REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**167-8191**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	
Date of RAI Issue:	**08/20/2015**

Question No. 09.01.01-2

RAI 9.1.1-4: Criticality prevention under normal and accident conditions of fuel handling

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b) sets specific requirements for the demonstration of nuclear criticality prevention in fuel storage. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 9.1.1, guides the reviewer to verify that the new and spent fuel will remain subcritical during fuel handling, in accordance with GDC 62 and 10 CFR 50.68(b). Section 9.1.1 of NUREG-0800 also guides the reviewer to verify that the applicant has identified a comprehensive set of normal conditions and has modeled them conservatively. It also tells the reviewer that the accidental or erroneous placement of a fuel assembly outside of, but next to, the fuel storage racks should be considered as an abnormal condition. The NRC memorandum from Laurence Kopp to Timothy Collins, dated August 19, 1998, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” is also cited by the applicant and adds further clarification as follows: “However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis.” With regard to the dry storage and handling of new fuel, the Kopp memorandum also notes that the accident conditions of flooding with optimum-density hydrogenous moderator (i.e., per 10 CFR 50.68(b)(3)) and flooding with full-density water (i.e., per 10 CFR 50.68(b)(2)) are the principal conditions that require evaluation and that the simultaneous occurrence of other accident conditions need not be considered. As noted in the applicant-cited NRC Interim Staff Guidance DSS-ISG-2010-01, the normal conditions of fuel storage and handling include not only static storage but also anticipated fuel handling activities such as inspection, cleaning, reconfiguration, and movement of fuel in and around the fuel storage racks and associated

fuel handling systems. Conditions during normal fuel handling operations are among the initial conditions to be considered in the analysis of postulated accidents.

ISSUE

It appears that the applicant has not provided information that demonstrates compliance with GDC 62 and 10 CFR 50.68(b) under normal and accident conditions of fuel handling in and around the various dry and wet fuel storage and handling systems. The wet fuel handling systems described in the DCD include those that handle fuel in and around the fuel elevator, the fuel transfer system, the refueling pool, the refueling cavity, and the spent fuel cask loading area. The staff notes for example that the fuel carrier of the fuel transfer system can hold two side-by-side fuel assemblies in an up-ended configuration and appears to have ample adjacent space into which an additional fuel assembly could be dropped by accident.

INFORMATION NEEDED

a) For dry handling of new fuel: In its response and in the DCD or its incorporated references, the applicant should identify a comprehensive set of normal conditions and accident conditions for dry handling of new fuel and provide conservative analyses that show compliance with GDC 62 and 10 CFR 50.68(b)(1)-(3) under such conditions. Where appropriate, the applicant should clarify where administrative control procedures are relied upon to address this issue.

b) For wet handling of new and used fuel: In its response and in the DCD or its incorporated references, the applicant should identify a comprehensive set of normal conditions and accident conditions of wet fuel handling and provide information showing that all such handling conditions comply with GDC 62 and with 10 CFR 50.68(b)(4) where applicable. The information should conservatively evaluate the most reactive normal and accident conditions of fuel handling operations throughout the fuel storage and handling area of the auxiliary building and outside the reactor vessel in the reactor building. The wet fuel handling equipment and operations to be addressed include those that involve the following items as identified in the DCD: the pool racks, the fuel elevator, the fuel transfer system, the refueling canal, the refueling pool, the refueling cavity, the reactor cavity, and the cask loading area. Where appropriate, the applicant should clarify where administrative control procedures are relied upon to address this issue.

Response - (Rev.1)

1. For Dry Handling of New Fuel

1.1 New Fuel Inspection Area

The k_{eff} of the new fuel inspection area (NFIA) is expected to be very low since the fuel assemblies are inspected in a dry environment. Therefore, criticality safety evaluations for NFIA are not necessary.

1.2 New Fuel Storage Racks

Under normal condition, the k_{eff} of the new fuel storage rack (NFR) is very low since the fuel assemblies are stored in dry environment. Therefore, criticality safety evaluations for normal condition of NFR are not necessary.

The following postulated accident conditions are considered and criticality analyses have been performed to evaluate the criticality safety.

- Optimum moderation condition (provided in TeR Subsections 2.4 and 2.5)
- Flooded by pure water condition (provided in TeR Subsections 2.4 and 2.5)
- Dropped fuel assembly (provided in the response to RAI 179-8190 09.01.01-21)

2. For Wet Handling of New and Used Fuel

2.1 New Fuel Elevator

2.1.1 Normal conditions

The new fuel elevator (NFE) is designed to accommodate one fuel assembly only. Therefore, the criticality safety evaluation for the normal condition of the NFE is not necessary.

2.1.2 Accident conditions

During the placement of the fuel assemblies in the cask loading pit where the NFE is located, it is possible to drop the fuel assembly from the fuel handling machine. This postulated accident is analyzed under the worst condition such that one fuel assembly is loaded in the NFE and the dropped fuel assembly is located next to the NFE as shown in Figure 1. The design input data is summarized in Table 1 and the bias and uncertainties are summarized in Table 2. The analysis results are presented in Table 4, and it is shown that the k_{eff} of this postulated accident condition is lower than the regulatory limit ($k_{eff} \leq 0.95$).

2.2 Spent Fuel Pool

Under normal condition, the most conservative condition is a static condition where all fuel assemblies are in the approved storage locations. Criticality safety evaluations for the static condition were performed in Subsections 3.4 and 3.5 of TeR.

The following postulated accident conditions are considered and criticality analyses have been performed to evaluate the criticality safety.

- Abnormal pool water temperature
  - As shown in the Subsections 3.4.3.5 and 3.5.3.5 of TeR, SFP has a negative moderation coefficient, i.e. k_{eff} at the lower temperature is higher than those at the higher temperature. Since the minimum water temperature (the maximum water density) is applied to the normal condition (static condition), additional evaluations are not necessary.
- Dropped fuel assembly (provided in Subsection 4.1 of TeR)

- Misloaded fuel assembly (provided in Subsection 4.2 of TeR)
- Boron dilution accident (provided in Subsection 4.3 of TeR)

2.3 Fuel Transfer System (Two Cavity Fuel Carrier)

2.3.1 Normal conditions

Under normal condition of the fuel transfer system, two fuel assemblies can be loaded in the fuel carrier assembly of the fuel transfer system (FTS). Figure 2 shows the analysis model for the normal condition of the FTS. The design input data is summarized in Table 1 and the bias and uncertainties are summarized in Table 3. The analysis results are presented in Table 4, and it is shown that the k_{eff} of the normal condition of the FTS is lower than the regulatory limit (k_{eff} < 1.0).

2.3.2 Accident conditions

During the placement of the fuel assemblies in the refueling canal, it is possible to drop the fuel assembly from the fuel handling machine. The worst condition for the dropped fuel assembly accident is that two fuel assemblies are loaded in the fuel carrier assembly and the dropped fuel assembly is located next to the fuel transfer system. The geometric configurations for these postulated accident cases are provided in the Figure 3, and the analysis results are provided in Table 4. As shown in the Table 4, the k_{eff} values of these postulated accident conditions are lower than the regulatory limit (k_{eff} ≤ 0.95).

TS

Non-Proprietary

TS

TS

TS

TS

TS

TS

TS

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on the Technical/Topical/Environmental Reports.

REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**167-8191**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**Criticality analysis TeR Sections 3.4 and 3.5**
Date of RAI Issue:	**08/20/2015**

Question No. 09.01.01-4

RAI 9.1.1-10: Modeled thickness of neutron absorber plates

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b)(4) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, guides the reviewer, in part, to verify that the criticality analysis conservatively incorporates fuel storage rack design data, including materials and dimensional data.

ISSUE

The applicant states that the absorber plates are assumed to have the maximum thickness allowed by tolerances. This staff considers this assumption to be potentially non-conservative in view of the lost neutron moderating effects of the water displaced by thicker plates.

INFORMATION NEEDED

In its response and in the DCD or its incorporated references, the applicant should resolve the apparent inconsistency by correctly characterizing and the modeling assumption of absorber plate thickness and by revising the modeling assumption as appropriate to avoid non-conservatism in this regard. If the assumption is revised, the applicant should also provide an updated criticality analysis for the storage pool racks.

Response - (Rev.1)

The design values of the neutron absorber plate (NAP) are provided in the table below. It is found that the NAP dimensions applied to the criticality analysis TeR are not consistent with the design values of NAP. So, these inconsistencies will be corrected and the criticality analysis TeR will be revised to incorporate the design values of NAP with revised assumptions.

Description	Design values	Applied values (Tables 3.1-1, 3.1-2 of TeR)
Thickness (mm)	[ ]TS	[ ]TS
Width (mm)	[ ]TS	[ ]TS
Length (mm)	[ ]TS	[ ]TS

The revised NAP assumptions are as follows:

1. The nominal dimensions of NAP thickness and width are applied to the criticality analysis,
2. The effect of the tolerance of NAP thickness and width on k_{eff} will be evaluated and included in the total uncertainty, and
3. The uncertainty due to tolerance of NAP length will not be evaluated, because the NAP length applied to the criticality analysis is assumed to be identical to the active fuel length ([ ]TS) which is shorter than the minimum length of NAP ([ ]TS).

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

Figure 3.7.16-1 of technical specifications will be revised as indicated in Attachment 2.

Impact on Technical/Topical/Environmental Reports

Sections 3.1, 3.2, 3.4, 3.5, 4.1, 4.2 and 4.3 of the criticality analysis TeR will be revised as indicated in Attachment 1.

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

TS

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

TS

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, and

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.

The uncertainties due to mechanical tolerances about the thickness and width of neutron absorber plates (NAP) are also evaluated and provided in Table 3.4-4. The uncertainty due to tolerance of the NAP length will not be evaluated, because the NAP length applied to the criticality analysis is assumed to be identical to the active fuel length ([ ]TS) which is shorter than the minimum length of NAP ([ ]TS).

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:

TS

[ ]

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~~ 0.92649 without applying soluble boron and the k_{eff} for the damaged fuel storage cells is ~~0.93655~~ 0.93681 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~~ 0.92649	≤ 0.95	< 1.00
K_{eff} for the Damaged Fuel Storage Cell	~~0.93655~~ 0.93681		

Table 3.4-4 Uncertainty due to Mechanical Tolerances

TS

This table should be replaced with the table in the next page.

Table 3.4-4 Uncertainty due to Mechanical Tolerances

TS

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

TS

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

TS

This table should be replaced with the table in the next page.

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

TS

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:

[ ]TS

The uncertainties due to mechanical tolerances about the thickness and width of neutron absorber plates (NAP) are also evaluated and provided in Table 3.5-8. The uncertainty due to tolerance of the NAP length will not be evaluated, because the NAP length applied to the criticality analysis is assumed to be identical to the active fuel length ([ ]TS) which is shorter than the minimum length of NAP ([ ]TS).

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:

[ ]TS

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~~ 0.998) 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.

TS

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

Table 3.5-7 Mechanical Tolerances

TS

This table should be replaced with the table in the next page.

Table 3.5-7 Mechanical Tolerances

TS

Table 3.5-8 Uncertainty due to Mechanical Tolerance

TS

This table should be replaced with the table in the next page.

Table 3.5-8 Uncertainty due to Mechanical Tolerance

TS

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

TS

This table should be replaced with the table in the next page.

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

TS

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

TS

This table should be replaced with the table in the next page.

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

TS

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

TS

This table should be replaced with the table in the next page.

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

TS

Table 3.5-25 Minimum Burnup Calculated with Raw Fitting Equation

TS

This table should be replaced with the table in the next page.

Table 3.5-25 Minimum Burnup Calculated with Raw Fitting Equation

TS

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

TS

This table should be replaced with the table in the next page.

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

TS

TS

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

This figure should be replaced with the figure in the next page.

TS

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

TS

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

This figure should be replaced with the figure in the next page.

TS

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

TS

Figure 3.5-7 Minimum Burnup versus Enrichment Curve

This figure should be replaced with the figure in the next page.

TS

Figure 3.5-7 Minimum Burnup versus Enrichment Curve

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).

0.79742

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.

700.31

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

TS

This table should be replaced with the table in the next page.

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

TS

TS

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

This figure should be replaced with the figure in the next page.

TS

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~~ 0.86995, 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~~ 1230.34 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 4.2-2 Criticality Analysis Results for Misloaded Fuel Assembly Accident

TS

This table should be replaced with the table in the next page.

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

TS

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~~ 427.22 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~~ 430 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~~ 430 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~~ 430 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~~ 430 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~~ 430 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~~ 430 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.

430

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

TS

This table should be replaced with the table in the next page.

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

TS

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

TS

This table should be replaced with the table in the next page.

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

TS

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

This figure should be replaced with the figure in the next page.

TS

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

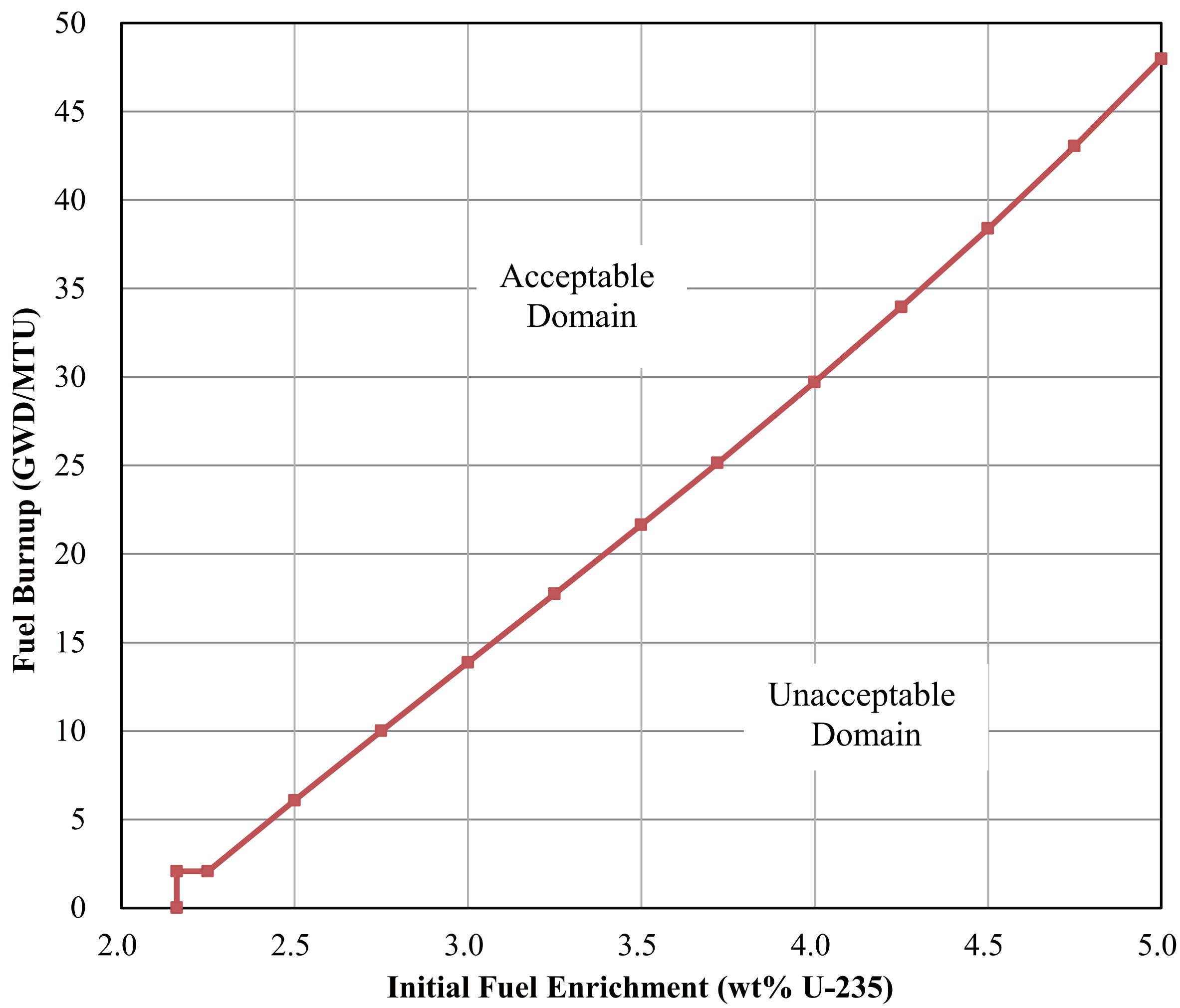
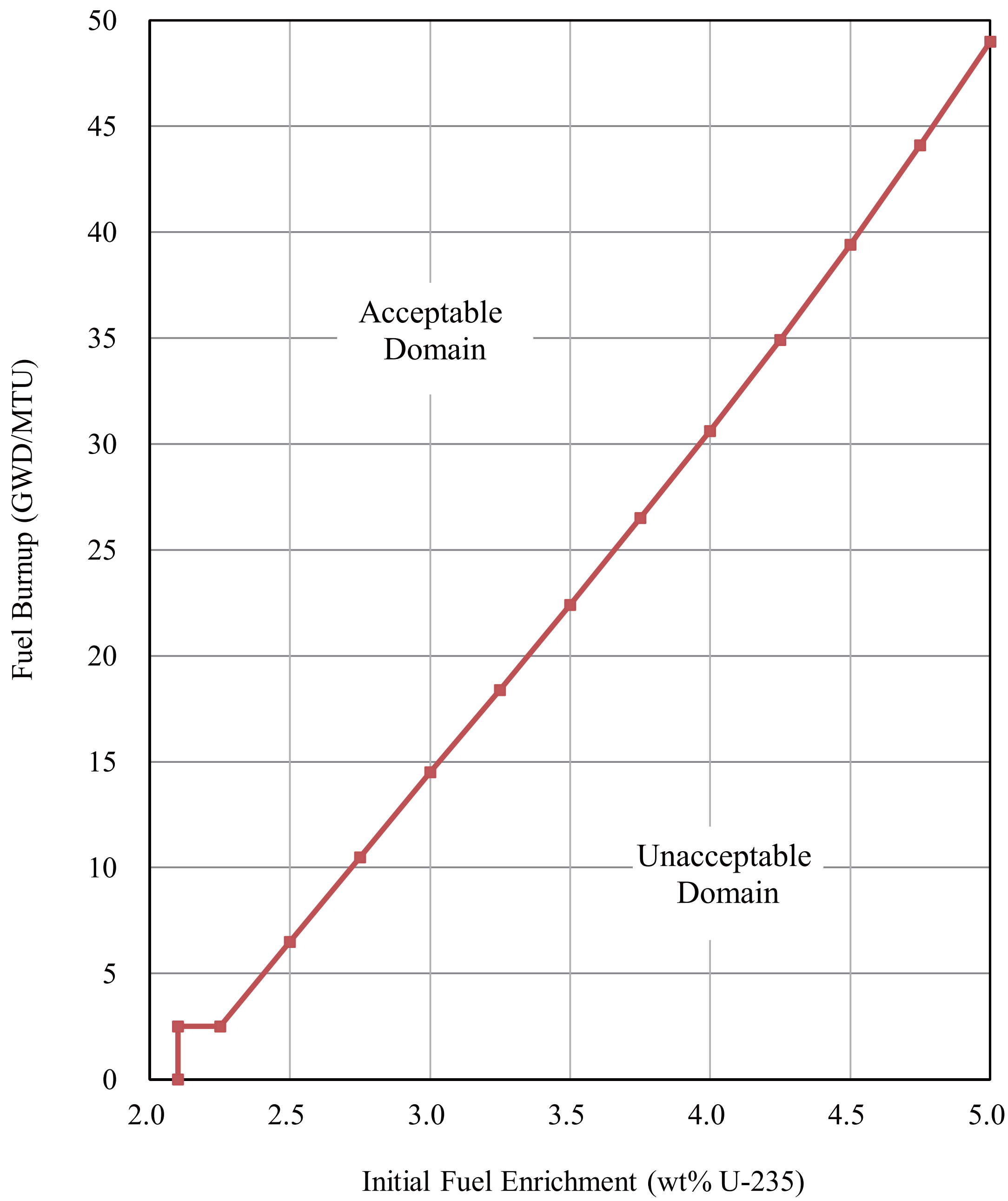


Figure 3.7.16-1 (page 1 of 1)
Discharge Burnup vs. Initial Enrichment for Region II Racks

This figure should be replaced with the figure in the next page.


Fuel Burnup (GWD/MTU)
50
45
40
35
30
25
20
15
10
5
0
Acceptable Domain
Unacceptable Domain
2.0
2.5
3.0
3.5
4.0
4.5
5.0
Initial Fuel Enrichment (wt% U-235)

REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**167-8191**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**Criticality Analysis TeR Subsection 3.4**
Date of RAI Issue:	**08/20/2015**

Question No. 09.01.01-5

RAI 9.1.1-11: Modeling of damaged fuel

REQUIREMENTS AND GUIDANCE

In 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 62 requires the prevention of criticality in fuel storage and handling. 10 CFR 50.68(b)(4) sets specific requirements for the demonstration of nuclear criticality prevention in wet fuel storage. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 9.1.1, advises the reviewer, in part, to verify the conservatism of normal- and abnormal-conditions models and the appropriateness of assumptions and approximations made therein.

ISSUE

The design of the pool region I racks includes special rack cells for storing damaged fuel. The applicant has not characterized the allowed contents of those storage cells in terms of the assumed kinds and extent of fuel assembly damage or assembly reconfiguration analyzed for storage. As described in Section 3.4.2 and Figure 3.4-3 of the criticality analysis report, the applicant’s analysis model for the damaged fuel storage cells appears to contain an intact new fuel assembly. The NRC staff is concerned that this analysis model may not be bounding for the anticipated contents of damaged fuel and may be clearly bounding only for storing intact new fuel.

INFORMATION NEEDED

The applicant should provide in its response and in the DCD or its incorporated references a description of the allowed contents and conditions of fuel in the damaged fuel storage cells

and, as necessary, an updated analysis with stated assumptions with regard to the modeling of allowed damaged fuel contents. If the analysis assumes immediate insertion of dummy or replacement fuel rods in place of rods damaged in the reactor or elsewhere, this should be stated and justified in the DCD or its incorporated references along with the identification of any supporting information or criteria to be provided in a COL.

Response - (Rev.1)

The damaged fuel assemblies discussed in Subsection 3.4 of the criticality analysis technical report (TeR) were modeled under the following assumptions, and k_{eff} without biases and uncertainties is [ ]TS (Reference case).

1. Unburned new fuel with initial enrichment of 5.0 wt%, and
2. All gaps between the pellet and the cladding are assumed to be flooded by water.

Irradiation of fuel assemblies to high burnup may increase the possibility of fuel assembly failure and higher burnup results in decreased reactivity of fuel. Thus, the assumption of new fuel is a conservative assumption. Also all gaps between pellets and the cladding are assumed to be flooded by water. This assumption is conservative because flooded water in the gap causes a local area of higher moderation, which can increase k_{eff}.

Case studies are performed to clarify the allowed conditions of fuel in damaged fuel storage cell (DFC), and the results are as follows.

1. Loss of fuel rods from the fuel assembly

Loss of fuel rods from the fuel assembly results in a local area of higher moderation, which can increase k_{eff}. To quantify the effect of loss of fuel rods on the reactivity, case studies are performed.

First, the loss of single fuel rod case is evaluated. Figure 1 provides a base model for the loss of fuel rod study. The damaged fuel assembly (loss of single fuel rod) is stored in the 3rd DFC, and other DFCs are occupied by new intact fuel assemblies. To quantify the single fuel rod loss effect on reactivity, each fuel rod in the fuel assembly was removed one at a time and criticality calculations were performed. Also, k_{eff} without damaged fuel (intact fuel stored in DFC) was evaluated to calculate the k_{eff} deviation due to fuel rod loss. Since a base model has symmetry about the y-direction, only upper part (from 1st row to 8th row) of fuel assembly is considered in these calculations. The calculation results are provided in Table 1 and summarized as follows.

{ }TS

The maximum k_{eff} without biases and uncertainties is [ ]TS, which is much smaller than k_{eff} of the reference case discussed in TeR. Therefore, additional evaluations for single fuel rod loss are not necessary.

Loss of multiple fuel rods also can happen during the operation. Due to the extremely large number of possible cases for the multiple fuel rods loss, it is not practical to evaluate all possible cases. Hence, analysis results of the references 1 and 2 were applied to this evaluation. According to the references 1 and 2, k_{eff} increment due to the loss of multiple fuel rods is about 2%~2.42% Δk. So, k_{eff} of multiple fuel rods loss is estimated as follows.

[ ]TS

Since estimated k_{eff} satisfies the acceptance criteria (k_{eff} without soluble boron < 1.0), additional evaluations for the loss of multiple fuel rods are not necessary.

2. Cladding Failure

If the cladding of fuel rods is damaged, the gap between the pellet and the cladding will be flooded by water. The most severe case is that every fuel rod's cladding is damaged so that every gap is flooded by water. This case is the same as an assumption of the reference case. Therefore, additional evaluations for damaged fuel cladding are not necessary.

Also, the loss of cladding can cause loss of pellets. This is the same condition as loss of fuel rods. Therefore, additional evaluations for loss of fuel cladding are not necessary.

3. Fuel Rod Reconfiguration (Increase rod pitch)

Fuel rod pitch can be changed by removal of fuel hardware such as fuel assembly grid and the change of fuel rod pitch can lead to increase moderation and k_{eff}. Figure 2 provides a base model for the fuel rod reconfiguration. All DFCs are occupied by damaged fuel assemblies, and it is assumed that the damaged fuel assembly has the maximum uniform rod pitch within an inner dimension of the canister in the DFC. The calculation result for fuel rod reconfiguration is [ ]TS, which includes biases and uncertainties. Since k_{eff} satisfies the acceptance criteria (k_{eff} without soluble boron < 1.0), additional evaluations are not necessary.

As a conclusion, modeling of damaged fuel discussed in Subsection 3.4 of the criticality analysis TeR is performed under conservative and appropriate assumptions. Also it is concluded that damaged fuel assemblies under the conditions discussed above can be stored in the DFC. The criticality analysis TeR will be revised to clarify assumptions applied to modeling of damaged fuel.

References

1. Marshall W. J. and Wagner J. C., “Impact of Fuel Failure on Criticality Safety of Used Nuclear Fuel,” Proc. Conf. PSAM11, June 2012.

2. NUREG/CR-6835, “Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks,” U.S. Nuclear Regulatory Commission, September 2003.

TS

Table 1. Results for Loss of Single Fuel Rod

TS

Figure 1. Base Model for Loss of Single Fuel Rod

TS

Figure 2. Base Model for Fuel Rod Reconfiguration

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

Subsection 3.4 of the criticality analysis TeR will be revised as indicated in Attachment.

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^{TM} 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

For the purpose of conservatism, the damaged fuel assemblies are modeled under the following assumptions.

a. Unburned new fuel with initial enrichment of 5.0 wt%, and
b. All gaps between the pellet and the cladding are assumed to be flooded by water.