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.:	**179-8190**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**DCD Tier 2, Section 9.1.1**
Date of RAI Issue:	**09/01/2015**

Question No. 09.01.01-15

RAI 9.1.1-9: Deformation of neutron absorber material

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. In addition, NRC Interim Staff Guidance DSS-ISG-2010-01 states that any degradation in rack materials should be modeled conservatively.

ISSUE

In DCD Section 9.1.1, the applicant states that “only 75 percent of B-10 density in the neutron absorbing materials is assumed in order to reflect the deformation of the neutron absorbing material,” but does not indicate the cause or nature of the material deformation.

INFORMATION NEEDED

The applicant should explain in its response and clarify in the DCD or its incorporated references the purpose of the stated spent fuel rack criticality analysis assumption with regard to the potential existence of material deformation, degradation, or other material characteristics or phenomena (e.g., non-uniform poison concentrations, poison granularity effects of neutron channeling or streaming) that could reduce or degrade the performance of the neutron absorbing material.

Response - (Rev.2)

The purpose of the assumption with regard to 75 percent of B-10 areal density is to ensure conservative calculation results. 90% of B-10 areal density is a conservative assumption to consider degradation of METAMIC™. However, 75 percent of B-10 areal density is used for more conservative analysis.

DCD Tier 2, Section 9.1.1 will be revised to clarify the analysis conditions for neutron absorber plates as indicated in the Attachment.

Impact on DCD

DCD Tier 2, Section 9.1.1 will be revised as indicated in the Attachment.

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 any Technical, Topical, or Environment Report.

<u>Spent Fuel Storage Rack</u>

The following analysis conditions are considered in the design of the spent fuel storage racks:

a. The fuel assembly is assumed to have a maximum enrichment of 5 wt% of U-235 in the criticality calculation for spent fuel storage rack region I. For the normal condition, an infinite array of fresh fuel assemblies is modeled in the criticality calculation. Criticality for damaged fuel assemblies is separately evaluated and the effects of gap between racks are also evaluated.

b. For the region II analyses, an infinite array of 2 × 2 fuel assemblies with various U-235 enrichments from 1.8 to 5.0 wt% is used for the criticality calculation. The mo
limits t
assume

Credit is taken for the neutron absorption in the neutron absorbing materials, and only 75 percent of B-10 areal density in the neutron absorbing materials is credited in the analysis. These assumptions provide additional margin in the event that deformation, degradation, or damage to the neutron absorber occur.

c. ~~Credit is taken for the neutron absorption in the rack structural materials and neutron absorbing materials. The steel plate thickness is conservatively set to a minimum, and only 75 percent of B-10 density in the neutron absorbing materials is assumed in order to reflect the deformation of the neutron absorbing material.~~

d. The neutron absorption effects associated with the soluble boron normally in the SFP water are neglected in the criticality analysis for normal operations and are considered for the postulated accidents. The SFP boron concentration is assumed to be about one-half of the minimum concentration level defined in the Technical Specifications.

e. No cooling time is assumed to avoid fission product accumulation and Xe-135 is not included in the criticality calculation to conservatively evaluate the K_{eff}.

f. The nuclear characteristics of the spent fuel are affected by the core operation parameters, such as coolant temperature, soluble boron concentration in the coolant, and axial burnup profile. Thus, the most severe operating conditions are conservatively assumed in the fuel burnup calculation.

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.:	**179-8190**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**Criticality Analysis TeR Subsection 3.5**
Date of RAI Issue:	**09/01/2015**

Question No. 09.01.01-16

RAI 9.1.1-19: Compliance results for spent fuel pool region II without soluble boron

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 to verify that the applicant has provided sufficient design and analysis information to support the evaluation findings of compliance.

ISSUE

The criticality analysis report does not clearly present the analysis results that show compliance with the regulatory requirement to demonstrate that the neutron multiplication factor k-eff without soluble boron remains below 1.0 at a 95 percent probability, 95 percent confidence level. The report’s Table 3.5-24 lists ranges of k-eff values that support development of the burnup credit loading curve but these include values above 1.0.

INFORMATION NEEDED

The applicant should provide in its response and in the DCD or its incorporated references the final analysis results that clearly show compliance without soluble boron. In particular, the analysis results should indicate the compliance margin by presenting a calculated maximum k-eff value that is a finite amount less than 1.0 at a 95 percent probability, 95 percent confidence level.

Non-Proprietary

Response - (Rev.1)

The loading curve provided in Figure 3.5-7 of the critical analysis technical report (TeR) was generated based on target k_{eff} of 1.0 with additional margin. However, it was found that k_{eff} of a fuel assembly with an initial enrichment of 4.5 wt% and burnup of 38.5 GWd/MTU slightly exceeds the regulatory limit (k_{eff} <1.0). To correct this error, the loading curve was regenerated based on target k_{eff} of 0.998 with additional margin.

The regenerated loading curve is provided in Figure 1. The summary of the loading conditions (initial enrichment and minimum burnup) is provided in Table 1, and it is shown that final k_{eff} values are a finite amount less than 1.0 at a 95 percent probability, 95 percent confidence level.

The criticality analysis TeR and Figure 3.7.16-1 of technical specifications will be revised to incorporate the regenerated loading curve.

TS

Table 1. k_{eff} with bias and uncertainty

TS

Figure 1. Loading Curve (Minimum Burnup versus Initial Enrichment)

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

Subsection 3.5 of the criticality analysis TeR will be revised as indicated in Attachment 1.

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

0.998

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

TS

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

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

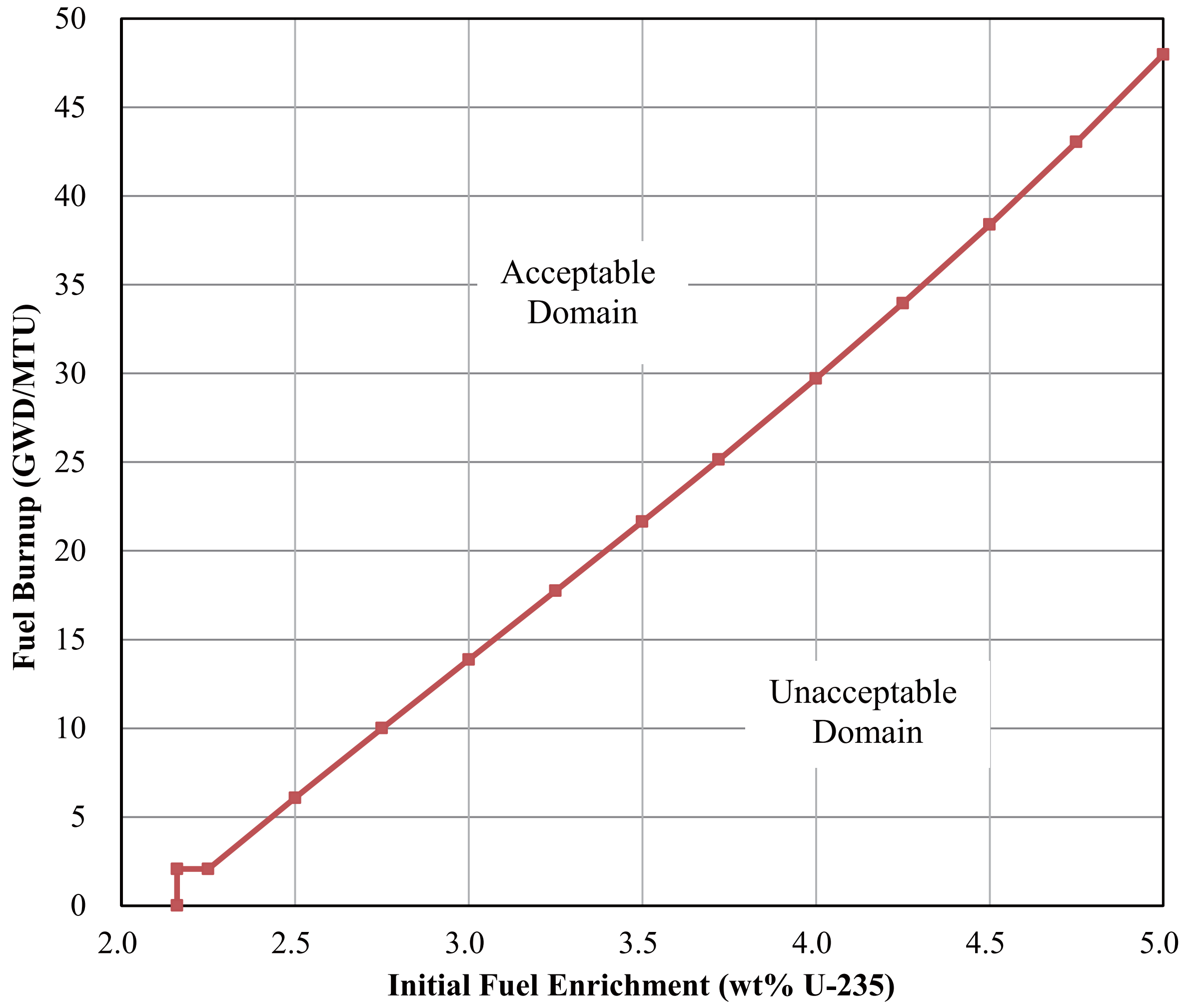
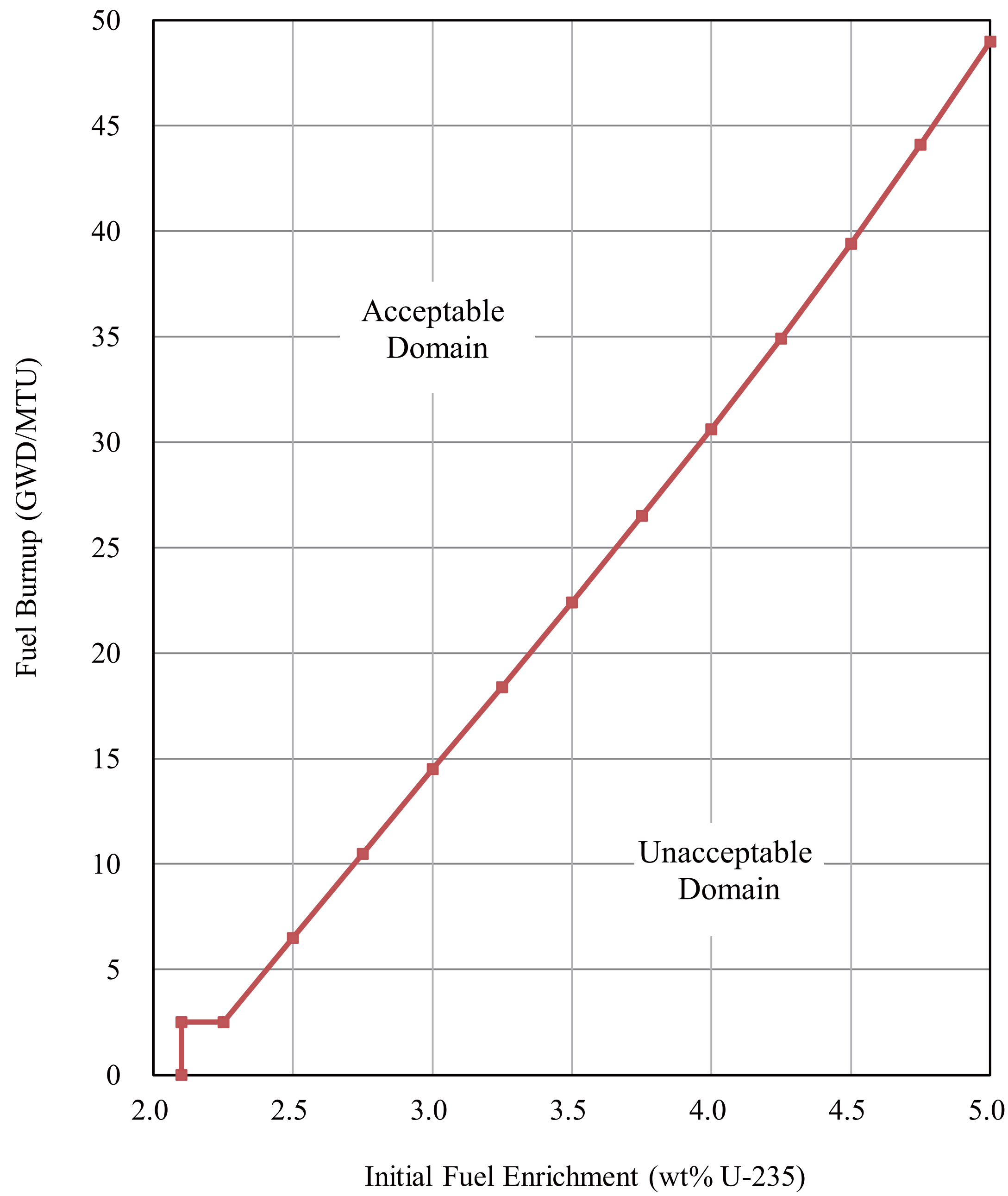


Figure 3.7.16-1 (page 1 of 1)
Discharge Burnup vs. Initial Enrichment for Region II Racks

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



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

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.:	**179-8190**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**DCD Tier 2, Section 9.1.1**
Date of RAI Issue:	**09/01/2015**

Question No. 09.01.01-19

RAI 9.1.1-22: Citation of applicant documents that support Tier 2 Section 9.1.1

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 applicant has provided sufficient design and analysis information to support the evaluation findings of compliance.

ISSUE

The NRC staff makes its safety finding based on information provided in the DCD and any documents incorporated by reference into the DCD. It is essential that all documents with essential information that is not contained in the DCD be docketed and clearly referenced in the DCD wherever needed to support or supplement the Tier 2 design and analysis information.

The staff notes that Section 9.1.1 of DCD Tier 2 fails to cite or reference in any manner both the criticality analysis report and the criticality validation report. Both reports clearly contain essential information that the staff must use in reaching its safety finding. Therefore, both reports should be clearly cited wherever used to support information in Tier 2 Section 9.1.1.

INFORMATION NEEDED

In its response, the applicant should add to Tier 2, Section 9.1.1, appropriately placed citation references to supporting applicant documents deemed essential by the NRC staff the applicant to support the claims of compliance.

Response - (Rev.1)

The criticality analysis technical report and the criticality code validation report will be cited to support the description of DCD Tier 2, Section 9.1.1. Also, misstatements in DCD Tier 2, Section 9.1.1 will be corrected to have consistency between the DCD Tier 2, Section 9.1.1 and the criticality analysis technical report.

Impact on DCD

DCD Tier 2, Section 9.1.1 will be revised as indicated in Attachment 1.

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 Report.

For criticality safety design, the following analyses are performed to evaluate the degree of subcriticality and to verify conformance with the following design criteria: (Reference 10)

a. New fuel storage rack: The K_{eff}s are calculated for new fuel storage pit filled with various density of unborated water. The K_{eff} must not exceed 0.95 for flood with unborated water and 0.98 for flood with optimum moderation conditions, assuming single failure of sources of moderation and potential firefighting activities. (Section 2 of Reference 10)

b. Spent fuel storage rack: The minimum required soluble boron concentrations are evaluated for normal conditions pursuant to the criteria of 10 CFR 50.68(b), item (4). Postulated accident conditions are considered for boron dilution accident in the SFP drop of a fuel assembly, abnormal location of a fuel assembly, and ~~rack movement in the event of seismic activity~~. SFP boron dilution analysis demonstrates that the dilution of the SFP below the boron concentration required for subcriticality is not credible considering the unborated water volume required. The undetected addition of 698,152 gallons of unborated water to the SFP is required to dilute the SFP such that the subcriticality limit (K_{eff} less than or equal to 0.95) is exceeded. This volume of water movement would be easily detected by pool level alarm and visible observation of spillover by operators. If a boron dilution event is detected, the borated water could be added to the SFP from boric acid storage tank (BAST) through boric acid makeup pump (BAMP) or by other means. (Section 4 of Reference 10)

Criticality analysis conditions are described below, including the design criteria, criticality analysis code with its validation for establishing code bias and bias uncertainty, and calculation model.

9.1.1.3.1 Design Criteria

The design criteria are pursuant to 10 CFR 50.68(b), items (2) and (3) for the new fuel storage rack, and item (4) for the spent fuel storage racks.

For new fuel storage racks, the maximum K_{eff} value, including all biases and uncertainties, must not exceed 0.95 for the flooded condition with unborated water and 0.98 for optimum moderation, at a 95 percent probability and 95 percent confidence level. Rack cells are assumed to be loaded with fuel of the maximum fuel assembly reactivity.

For spent fuel storage racks, the maximum K_{eff} value, including all biases and uncertainties, must remain below 1.0 with full density unborated water, at a 95 percent probability and 95 percent confidence level. Rack cells are assumed to be loaded with fuel of the maximum fuel assembly reactivity.

9.1.1.3.2 Computer Codes and Validation

SCALE 6.1 (Reference 5) is used for the depletion and criticality calculations. The TRITON module is used for the determination of the isotopic content with depletion, and the CSAS5 module is used for the calculation of the neutron multiplication factor, k, using the TRITON-produced atom densities. The SCALE analysis can be done with a number of different cross-section libraries. The ENDF/B-VII (Reference 6) 238-group library is used for the depletion and criticality calculations.

Depletion calculations are performed using the TRITON t-DEPL sequence. The TRITON t-DEPL sequence uses the NEWT computer code to obtain the detailed 2D flux solutions. The neutron fluxes are used in multiple ORIGEN-S depletion calculations to get the isotope inventories compositions which can be used directly in a criticality calculation. The NEWT calculations are performed using the SCALE 238-energy-group ENDF/B-VII library.

Criticality or K_{eff} calculations are performed using the CSAS5 sequence and the ENDF/B-VII 238-energy-group library. CSAS5 uses the KENO V.a, a 3D Monte Carlo transport code.

The validation of KENO V.a code is performed using the benchmark experiments in the international handbook (Reference 7) and NUREG/CR-6361 (Reference 8), and HTC critical experiments (Reference 9). Through the validation of criticality code, the bias and bias uncertainty are evaluated for new and spent fuel storage racks. The guidance and procedure recommended in the NUREG/CR-6698 is followed for the validation of KENO V.a computer code.

(Reference 30)

9.1.1.3.3 Analysis Condition

New Fuel Storage Rack (Section 2 of Reference 10)

The following analysis conditions are considered in the design of the new fuel storage racks:

Thickness of concrete wall surrounding the new fuel storage racks is assumed to be 30 cm.

a. The new fuel stora … e structure. ~~All five sides of the new fuel storage racks are surrounded by a 0.61 m (2 ft) thick concrete reflector~~. The concrete walls surrounding the new fuel storage racks are adequately separated from the racks.

b. The dry new fuel storage pit for new fuel assemblies is designed to store two 8 × 7 racks. The criticality analysis is not performed for the normal condition because there is no moderator in a dry condition.

c. For postulated accidents, a pure water of density range from 0 to 100 percent of full density is assumed to fill the new fuel storage racks. Full density of unborated water is assumed to be ~~1,000 kg/cm^3~~ (62.4 lb/ft^3).

1.0 g/cm^3

d. ~~Actual geometrical model is used in the criticality calculation.~~ The fuel assembly is assumed to have a maximum enrichment of 5 wt% of U-235 and the rack is assumed to be filled with fuel assemblies to design capacity.

e. No burnable poison rods or other supplemental neutron poisons (e.g., control element assemblies [CEAs]) are assumed to be present in the fuel assemblies.

f. ~~The materials above the active fuel constitute a substantially poorer reflector than a thick concrete reflector.~~ The structural materials above the active fuel are assumed to be water.

g. Under postulated conditions of envelopment by aqueous foam or mist, a range of foam or mist densities is examined to provide reasonable assurance that the maximum reactivity of the array is established. The foam or mist is assumed to be pure water.

h. Uncertainties from fuel assembly fabrication tolerances, computer code and Monte Carlo calculation are considered in the calculated K_{eff}.

i. The K_{eff} must not exceed 0.95 in the event the fuel area becomes flooded with pure, unborated water. The K_{eff} must not exceed 0.98 in the event of envelopment of the entire array in a uniform aqueous foam or mist of optimum density that maximizes the reactivity of the finite array.

<u>Spent Fuel Storage Rack</u> (Section 3 of Reference 10)

The following analysis conditions are considered in the design of the spent fuel storage racks:

a. The fuel assembly is assumed to have a maximum enrichment of 5 wt% of U-235 in the criticality calculation for spent fuel storage rack region I. For the normal condition, an infinite array of fresh fuel assemblies is modeled in the criticality calculation. Criticality for damaged fuel assemblies is separately evaluated and the effects of gap between racks are also evaluated.

b. For the region II analyses, an infinite array of 2 × 2 fuel assemblies with various U-235 enrichments, from ~~1.8~~ 2.0 to 5.0 w The moderator of pure water is at the limits that yields the largest reactivity. assumed to be ~~1,000 kg/cm^3~~ 1.0 g/cm^3 (62.4 lbm/f

Credit is taken for the neutron absorption in the neutron absorbing materials, and only 75 percent of B-10 areal density in the neutron absorbing materials is credited in the analysis. These assumptions provide additional margin in the event that deformation, degradation, or damage to the neutron absorber occur.

c. ~~Credit is taken for the neutron absorption in the rack structural materials and neutron absorbing materials. The steel plate thickness is conservatively set to a minimum, and only 75 percent of B-10 density in the neutron absorbing materials is assumed in order to reflect the deformation of the neutron absorbing material.~~

d. ~~The neutron absorption effects associated with the soluble boron normally in the SFP water are neglected in the criticality analysis for normal operations and are considered for the postulated accidents. The SFP boron concentration is assumed to be about one-half of the minimum concentration level defined in the Technical Specifications.~~

Credit is taken for the soluble boron in the SFP. The SFP soluble boron concentration is assumed to be the same or less than the minimum concentration provided in the LCO 3.7.15.

e. No cooling time is as not included in the cri calculation to conservatively evaluate the k_{eff}.

f. The nuclear characteristics of the spent fuel are affected by the core operation parameters, such as coolant temperature, soluble boron concentration in the coolant, and axial burnup profile. Thus, the most severe operating conditions are conservatively assumed in the fuel burnup calculation.

22. NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity,” U.S. Nuclear Regulatory Commission, July 1980.

23. NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, 1979.

24. ANSI/ASME B30.9-2003, “Slings,” The American Society of Mechanical Engineers, 2010.

25. CMAA 70-00, “Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes,” Crane Manufacturers Association of America, 2000.

26. ASME B30.2-2005, “Overhead and Gantry Cranes – Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist,” The American Society of Mechanical Engineers, July 2011.

27. ANSI N14.6, “Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 Kg) or More,” American National Standards Institute, 1993.

28. ASME NUM-1, “Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type),” The American Society of Mechanical Engineers, 2004.

29. ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

30. WCAP-17889-P, "Validation of SCALE 6.1.2 with 238-Group ENDF/B-VII.0 Cross Section Library for APR1400 Design Certification," WEC, June 2014.

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.:	**179-8190**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**DCD Tier 2, Section 9.1.1**
Date of RAI Issue:	**09/01/2015**

Question No. 09.01.01-20

RAI 9.1.1-23: Incorporation of design and analysis details into Tier 2 Section 9.1

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 applicant has provided sufficient design and analysis information to support the evaluation findings of compliance.

ISSUE

The staff notes that neither the criticality analysis report nor its referenced criticality validation report are listed in DCD Tier 2, Chapter 1, Table 1.6-2, as documents to be incorporated by reference into Tier 2. Furthermore, the staff notes that the number of documents containing details that will ultimately have to be captured in the DCD Tier 2 section or else incorporated by reference by listing in Tier 2, Table 1.6-2, may have to grow in response to further RAI questions, audit items, and related discussions between the applicant and NRC staff at publicly noticed meetings.

INFORMATION NEEDED

In its response, the applicant should describe how it will either capture essential design and analysis details in the DCD sections or incorporate by reference the documents that contain such details. If the subject documents are to be incorporated by reference, then Tier 2, Table 1.6-2 should be updated accordingly.

Response - (Rev.1)

The criticality analysis technical report and the criticality code validation report will be cited to support the description of DCD Tier 2, Section 9.1.1; misstatements in DCD Tier 2, Section 9.1.1 will be also corrected to have consistency between the DCD Tier 2, Section 9.1.1 and the criticality analysis technical report.

Also, Table 1.6-2 of DCD Tier 2 will be revised to incorporate the criticality analysis technical report.

Impact on DCD

DCD Tier 2, Section 9.1.1 will be revised as indicated in Attachment 1.
DCD Tier 2, Table 1.6-2 will be revised as indicated in Attachment 2.

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

For criticality safety design, the following analyses are performed to evaluate the degree of subcriticality and to verify conformance with the following design criteria: (Reference 10)

a. New fuel storage rack: The K_{eff}s are calculated for new fuel storage pit filled with various density of unborated water. The K_{eff} must not exceed 0.95 for flood with unborated water and 0.98 for flood with optimum moderation conditions, assuming single failure of sources of moderation and potential firefighting activities. (Section 2 of Reference 10)

b. Spent fuel storage rack: The minimum required soluble boron concentrations are evaluated for normal conditions pursuant to the criteria of 10 CFR 50.68(b) item (4). Postulated accident conditions are considered for boron dilution accident in the SFP drop assembly, abnormal location of a fuel assembly, and ~~rack movement in the event of seismic activity~~. SFP boron dilution analysis demonstrates that the dilution of the SFP below the boron concentration required for subcriticality is not credible considering the unborated water volume required. The undetected addition of 698,152 gallons of unborated water to the SFP is required to dilute the SFP such that the subcriticality limit (K_{eff} less than or equal to 0.95) is exceeded. This volume of water movement would be easily detected by pool level alarm and visible observation of spillover by operators. If a boron dilution event is detected, the borated water could be added to the SFP from boric acid storage tank (BAST) through boric acid makeup pump (BAMP) or by other means. (Section 4 of Reference 10)

Criticality analysis conditions are described below, including the design criteria, criticality analysis code with its validation for establishing code bias and bias uncertainty, and calculation model.

9.1.1.3.1 <u>Design Criteria</u>

The design criteria are pursuant to 10 CFR 50.68(b), items (2) and (3) for the new fuel storage rack, and item (4) for the spent fuel storage racks.

For new fuel storage racks, the maximum K_{eff} value, including all biases and uncertainties, must not exceed 0.95 for the flooded condition with unborated water and 0.98 for optimum moderation, at a 95 percent probability and 95 percent confidence level. Rack cells are assumed to be loaded with fuel of the maximum fuel assembly reactivity.

For spent fuel storage racks, the maximum K_{eff} value, including all biases and uncertainties, must remain below 1.0 with full density unborated water, at a 95 percent probability and 95 percent confidence level. Rack cells are assumed to be loaded with fuel of the maximum fuel assembly reactivity.

9.1.1.3.2 Computer Codes and Validation

SCALE 6.1 (Reference 5) is used for the depletion and criticality calculations. The TRITON module is used for the determination of the isotopic content with depletion, and the CSAS5 module is used for the calculation of the neutron multiplication factor, k, using the TRITON-produced atom densities. The SCALE analysis can be done with a number of different cross-section libraries. The ENDF/B-VII (Reference 6) 238-group library is used for the depletion and criticality calculations.

Depletion calculations are performed using the TRITON t-DEPL sequence. The TRITON t-DEPL sequence uses the NEWT computer code to obtain the detailed 2D flux solutions. The neutron fluxes are used in multiple ORIGEN-S depletion calculations to get the isotope inventories compositions which can be used directly in a criticality calculation. The NEWT calculations are performed using the SCALE 238-energy-group ENDF/B-VII library.

Criticality or K_{eff} calculations are performed using the CSAS5 sequence and the ENDF/B-VII 238-energy-group library. CSAS5 uses the KENO V.a, a 3D Monte Carlo transport code.

The validation of KENO V.a code is performed using the benchmark experiments in the international handbook (Reference 7) and NUREG/CR-6361 (Reference 8), and HTC critical experiments (Reference 9). Through the validation of criticality code, the bias and bias uncertainty are evaluated for new and spent fuel storage racks. The guidance and procedure recommended in the NUREG/CR-6698 is followed for the validation of KENO V.a computer code. (Reference 30)

9.1.1.3.3 Analysis Condition

New Fuel Storage Rack (Section 2 of Reference 10)

The following analysis conditions are considered in the design of the new fuel storage racks:

Thickness of concrete wall surrounding the new fuel storage racks is assumed to be 30 cm.

a. The new fuel stora[illegible]e structure. ~~All five sides of the new fuel storage racks are surrounded by a 0.61 m (2 ft) thick concrete reflector.~~ The concrete walls surrounding the new fuel storage racks are adequately separated from the racks.

b. The dry new fuel storage pit for new fuel assemblies is designed to store two 8 × 7 racks. The criticality analysis is not performed for the normal condition because there is no moderator in a dry condition.

c. For postulated accidents, a pure water of density range from 0 to 100 percent of full density is assumed to fill the new fuel storage racks. Full density of unborated water is assumed to be ~~1,000 kg/cm³~~ 1.0 g/cm³ (62.4 lb/ft³).

d. ~~Actual geometrical model is used in the criticality calculation.~~ The fuel assembly is assumed to have a maximum enrichment of 5 wt% of U-235 and the rack is assumed to be filled with fuel assemblies to design capacity.

e. No burnable poison rods or other supplemental neutron poisons (e.g., control element assemblies [CEAs]) are assumed to be present in the fuel assemblies.

f. ~~The materials above the active fuel constitute a substantially poorer reflector than a thick concrete reflector.~~ The structural materials above the active fuel are assumed to be water.

g. Under postulated conditions of envelopment by aqueous foam or mist, a range of foam or mist densities is examined to provide reasonable assurance that the maximum reactivity of the array is established. The foam or mist is assumed to be pure water.

h. Uncertainties from fuel assembly fabrication tolerances, computer code and Monte Carlo calculation are considered in the calculated K_{eff}.

i. The K_{eff} must not exceed 0.95 in the event the fuel area becomes flooded with pure, unborated water. The K_{eff} must not exceed 0.98 in the event of envelopment of the entire array in a uniform aqueous foam or mist of optimum density that maximizes the reactivity of the finite array.

Spent Fuel Storage Rack (Section 3 of Reference 10)

The following analysis conditions are considered in the design of the spent fuel storage racks:

a. The fuel assembly is assumed to have a maximum enrichment of 5 wt% of U-235 in the criticality calculation for spent fuel storage rack region I. For the normal condition, an infinite array of fresh fuel assemblies is modeled in the criticality calculation. Criticality for damaged fuel assemblies is separately evaluated and the effects of gap between racks are also evaluated.

2.0

b. For the region II analyses, an infinite array of 2 × 2 fuel assemblies with various U-235 enrichments, from ~~1.8~~ to 5.0 wt%, The moderator of p 1.0 g/cm³ is at the ten limits that yields the largest reactivity. T assumed to be ~~1,000 kg/cm³~~ (62.4 lbm/ft³).

Credit is taken for the neutron absorption in the neutron absorbing materials, and only 75 percent of B-10 areal density in the neutron absorbing materials is credited in the analysis. These assumptions provide additional margin in the event that deformation, degradation, or damage to the neutron absorber occur.

c. ~~Credit is taken for the neutron absorption in the rack structural materials and neutron absorbing materials. The steel plate thickness is conservatively set to a minimum, and only 75 percent of B-10 density in the neutron absorbing materials is assumed in order to reflect the deformation of the neutron absorbing material.~~

d. ~~The neutron absorption effects associated with the soluble boron normally in the SFP water are neglected in the criticality analysis for normal operations and are considered for the postulated accidents. The SFP boron concentration is assumed to be about one-half of the minimum concentration level defined in the Technical Specifications.~~

Credit is taken for the soluble boron in the SFP. The SFP soluble boron concentration is assumed to be the same or less than the minimum concentration provided in the LCO 3.7.15.

e. No cooling time is as not included in the cri

f. The nuclear characteristics of the spent fuel are affected by the core operation parameters, such as coolant temperature, soluble boron concentration in the coolant, and axial burnup profile. Thus, the most severe operating conditions are conservatively assumed in the fuel burnup calculation.

22. NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity,” U.S. Nuclear Regulatory Commission, July 1980.

23. NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, 1979.

24. ANSI/ASME B30.9-2003, “Slings,” The American Society of Mechanical Engineers, 2010.

25. CMAA 70-00, “Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes,” Crane Manufacturers Association of America, 2000.

26. ASME B30.2-2005, “Overhead and Gantry Cranes – Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist,” The American Society of Mechanical Engineers, July 2011.

27. ANSI N14.6, “Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 Kg) or More,” American National Standards Institute, 1993.

28. ASME NUM-1, “Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type),” The American Society of Mechanical Engineers, 2004.

29. ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

30. WCAP-17889-P, "Validation of SCALE 6.1.2 with 238-Group ENDF/B-VII.0 Cross Section Library for APR1400 Design Certification," WEC, June 2014.

Table 1.6-2 (2 of 2)

Report Number (1)	Title	DCD Tier 2 Section
APR1400-F-A-NR-14002-P APR1400-F-A-NR-14002-NP	The Effect of Thermal Conductivity Degradation on APR1400 Design and Safety Analyses	15.4 15.6
APR1400-F-A-NR-14003-P APR1400-F-A-NR-14003-NP	Post-LOCA Long Term Cooling Evaluation Model	15.6
APR1400-H-N-NR-14012-P APR1400-H-N-NR-14012-NP	Mechanical Analysis for New and Spent Fuel Storage Racks	9.1.2
APR1400-K-I-NR-14005-P APR1400-K-I-NR-14005-NP	Staffing and Qualifications Implementation Plan	18.5
APR1400-K-I-NR-14009-P APR1400-K-I-NR-14009-NP	Design Implementation Plan	18.11
APR1400-Z-A-NR-14006-P APR1400-Z-A-NR-14006-NP	Non-LOCA Safety Analysis Methodology	15.0.2
APR1400-Z-A-NR-14007-P APR1400-Z-A-NR-14007-NP	LOCA Mass and Energy Release Methodology	6.2.1.3
APR1400-Z-J-NR-14001-P APR1400-Z-J-NR-14001-NP	Safety I&C System	7.1, 7.2, 7.3, 7.4, 7.5, 7.8, 7.9
APR1400-Z-J-NR-14003-P APR1400-Z-J-NR-14003-NP	Software Program Manual	7.1.4, 7.2.2.2, 7.3.1
APR1400-Z-J-NR-14004-P APR1400-Z-J-NR-14004-NP	Uncertainty Methodology and Application for Instrumentation	7.2.2.7, 7.3.2.7
APR1400-Z-J-NR-14005-P APR1400-Z-J-NR-14005-NP	Setpoint Methodology for Plant Protection System	7.2.2.7, 7.3.2.7
APR1400-Z-M-NR-14008-P APR1400-Z-M-NR-14008-NP	Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown	5.2, 5.3
APR1400-Z-A-NR-14011-P APR1400-Z-A-NR-14011-NP	Criticality Analysis of New and Spent Fuel Storage Racks	9.1.1

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.:	**179-8190**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**DCD Tier 2, Section 9.1.1, Criticality Analysis TeR Sections 2.4 and 2.5**
Date of RAI Issue:	**09/01/2015**

Question No. 09.01.01-23

Question 9.1.1-6: Calculation of sensitivities to new fuel rack tolerances and variations

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 new 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 the conservatism of normal-conditions models and the appropriateness of assumptions and approximations made therein. Sensitivity studies are one way to support normal-conditions model development.

ISSUE

For the new fuel rack criticality analysis, the applicant calculates sensitivities to design tolerances and variations on a model of an infinite-array of rack cells moderated by full-density water. The staff notes that this calculation model neglects the neutronic effects of storage pit concrete and does not address sensitivities under the potentially more limiting accident conditions of optimum moderation by low-density water.

INFORMATION NEEDED

The applicant should describe in its response and in the DCD or its incorporated references additional sensitivity calculations on an analysis model that addresses the neutronic effects of storage pit concrete as well as the respective conditions of optimum moderation by low-density water and moderation by full- density water. The results of the supplemental

sensitivity studies should be applied as necessary to the neutron multiplication factors computed for the respective moderation accident conditions.

Response - (Rev.1)

Sensitivity calculations on a finite array model describing the new fuel racks with storage pit concrete in optimum moderation and full-density water conditions have been performed to obtain uncertainties due to design tolerances. The criticality analysis TeR will be revised to include the sensitivity calculation results.

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

Sections 2.4.2 and 2.5, Tables 2.4-3, 2.4-4, and 2.5-1, Figures 2.4-2, 2.4-3, and 2.5-1 in the criticality analysis TeR will be revised as indicated in the attachment.

RAI 179-8190 - Question 09.01.01-23_Rev.1 **Non-Proprietary** Attachment (1/16)

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

TS

[ ]TS

[ ]TS

2.4 Criticality Analysis for New Fuel Storage Rack

2.4.1 Criticality Calculation

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

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

2.4.2 Bias and Uncertainty

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

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

b. Statistical uncertainty of the Monte Carlo calculation, and

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

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

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

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

a. Bias: [ ]TS

b. Bias uncertainty: [ ]TS

(2) Statistical uncertainty of the Monte Carlo calculation

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

a. Optimum moderation (2σ) : [ ]TS

b. Moderation by full density water (2σ) : [ ]TS

Statistical uncertainties of the criticality calculation for the optimum moderation condition and fully flooded condition are provided in Tables 2.4-3 and 2.4-4, respectively.

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

These items should be replaced with page 3 of Attachment.

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

a. UO_2 pellet stack density: [ ]TS,

b. UO_2 pellet diameter: [ ]TS,

c. Fuel rod pitch: [ ]TS,

d. Fuel clad outer diameter: [ ]TS,

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

f. NFR cell pitch: [ ]TS, and

a. U-235 enrichment: [ ]TS
b. UO_2 pellet stack density: [ ]TS,
c. UO_2 pellet diameter: [ ]TS,
d. Fuel rod pitch: [ ]TS,
e. Fuel clad outer diameter: [ ]TS,
f. Guide tube outer diameter: [ ]TS
g. Fuel assembly position in fuel rack cell: [ ]TS,
h. NFR cell pitch: [ ]TS, and

i. NFR cell thickness: [ ]TS.

~~g. NFR cell thickness: [ ]TS.~~

~~The uncertainty analyses are performed for the unit cell model of the NFR as shown in Figure 2.4-2.~~ The uncertainty analyses are performed for the finite array model including NFSP as shown in Figure 2.4-1. The calculation results for the uncertainty analysis due to the tolerances or variations in the design parameters are summarized in ~~Table 2.4-2~~ Tables 2.4-3 and 2.4-4. To determine the reactivity difference (Δk_i) associated with a specific manufacturing tolerance, the k_{eff} calculated for the reference model is compared to that for the model with an individual tolerance. The Δk_i due to a tolerance is then calculated as follows:

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

where:

k_i = k_{eff} with the tolerance,

k_R = k_{eff} for the reference model,

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

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

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

The resultant uncertainty due to tolerances or variations in the design parameters which is calculated as square root of the sum of the squares of individual Δk_i is presented in the last row of ~~Table 2.4-2.~~ Tables 2.4-3 and 2.4-4.

The evaluated total bias and uncertainty (Δk_{NFR}), ~~which includes~~ for optimum moderation and fully flooded conditions, which include bias for the criticality analysis of the NFR is determined as follows:

[ ]TS

This equation should be replaced with page 5 of Attachment.

For conservatism, a total uncertainty of 0.01721 will be applied to obtain final criticality analysis results for new fuel rack storage system.

[ ]TS

[ ]TS

Table 2.4-3 Tolerance Uncertainty for Optimum Moderation Condition

TS

This table should be added.

Table 2.4-4 Tolerance Uncertainty for Fully Flooded Condition

TS

This table should be added.

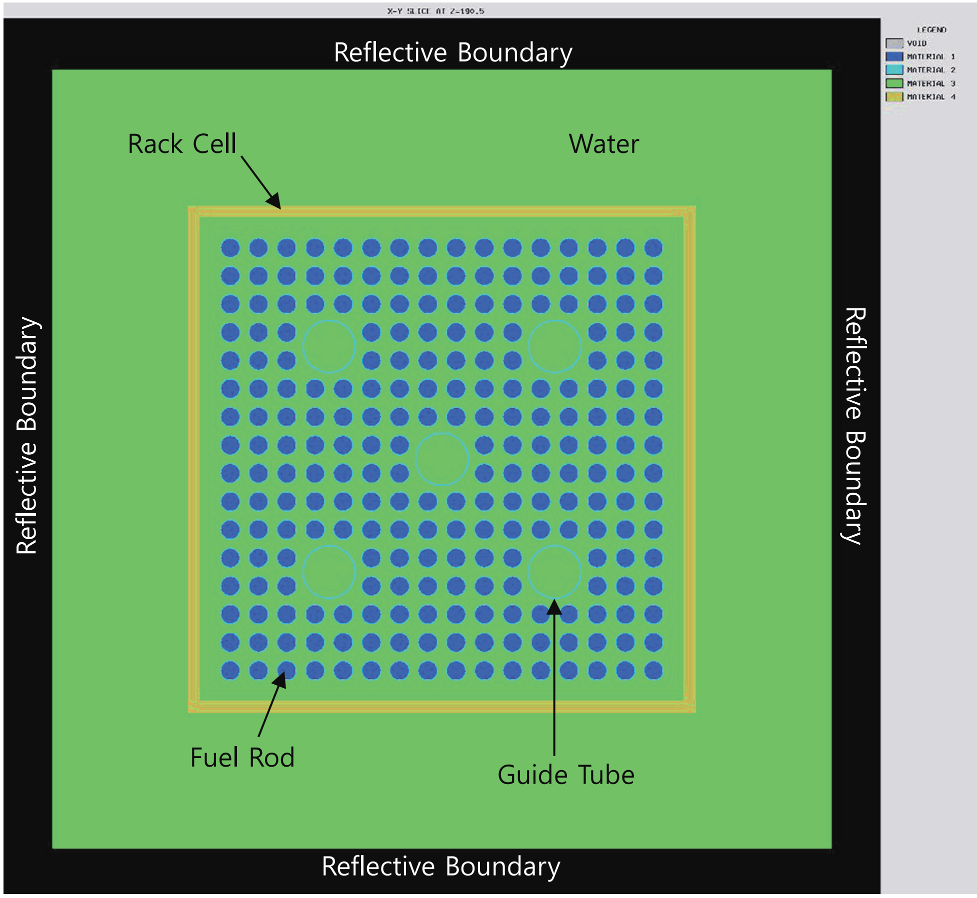


Figure 2.4-2 Unit Cell Model for Uncertainty Analysis

This figure should be replaced with page 9 of Attachment.

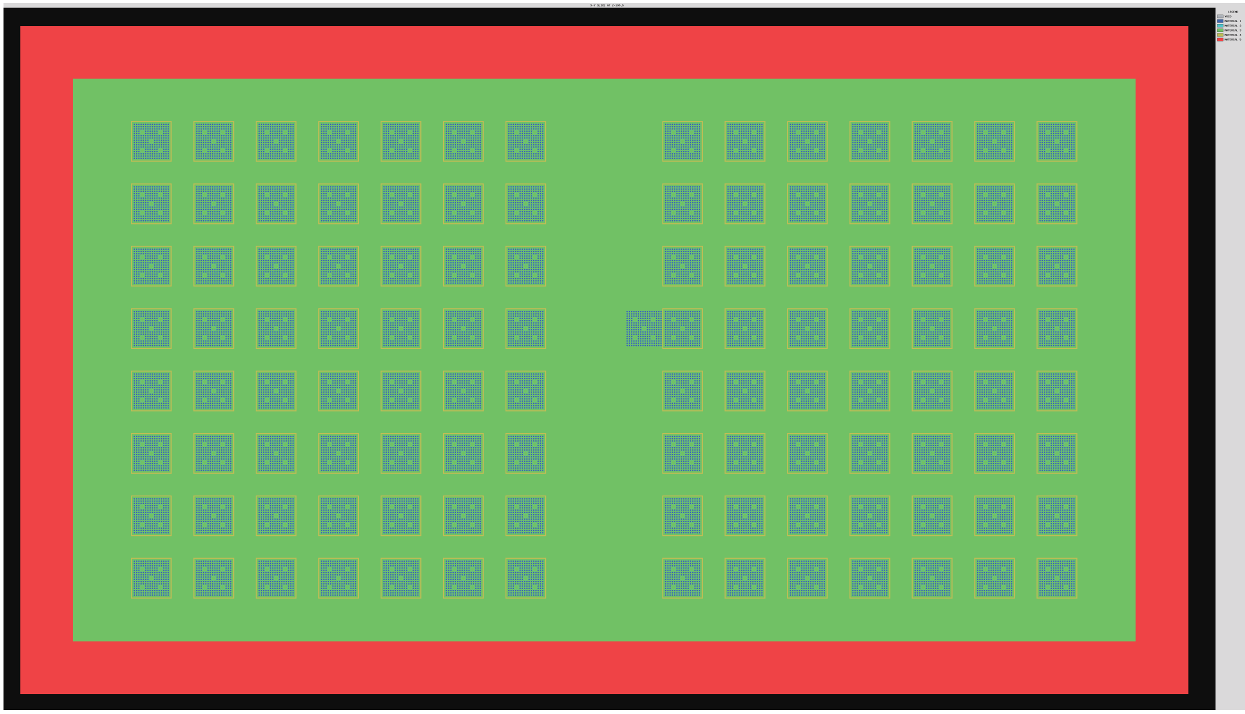


Figure 2.4-2 Criticality Calculation Model for a Dropped Fuel Assembly Accident (Contact)

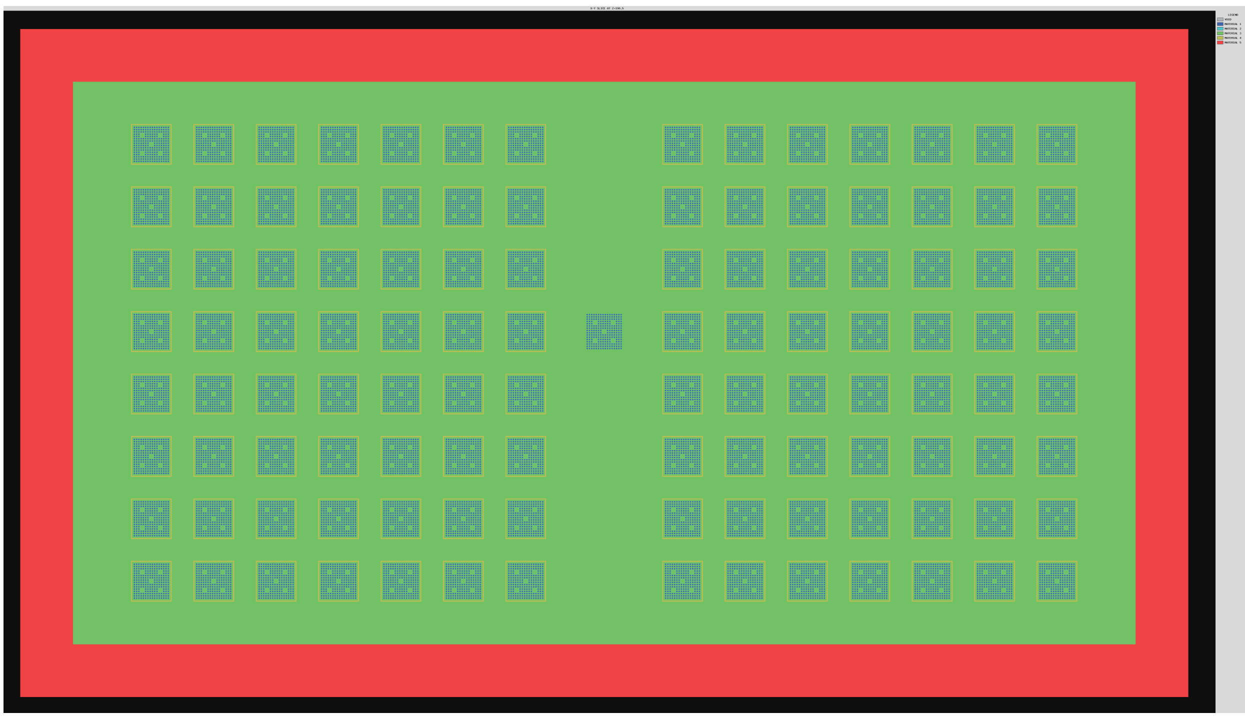


Figure 2.4-3 Criticality Calculation Model for a Dropped Fuel Assembly Accident (Centered)

This figure should be added.

Table 2.5-2 shows the evaluated k_{eff} for postulated accident cases of a dropped fuel assembly in the space between fuel racks.

2.5 Results

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

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

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

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

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

This table should be replaced with table on page 12 of Attachment.

Description	k_{eff}	Acceptance criteria
k_{eff} for flooded by pure water	0.91449	≤ 0.95
k_{eff} for optimum moderation	0.93495	≤ 0.98
k_{eff} for a dropped fuel assembly	0.55226	≤ 0.95

Table 2.5-1 Effective Multiplication Factors for the NFR

TS

This table should be replaced with page 14 of Attachment.

Table 2.5-1 Evaluation Results of k_{eff} for Various Water Densities

TS

TS

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

This figure should be replaced with page 16 of Attachment.

TS

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

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.:	**179-8190**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**DCD Tier 2, Section 9.1.1, Criticality Analysis TeR Sections 2.4 and 2.5**
Date of RAI Issue:	**09/01/2015**

Question No. 09.01.01-25

RAI 9.1.1-24: Dimensional tolerances for the new fuel storage racks

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 and handling. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 9.1.1 guides the reviewer, in part, to verify that normal and abnormal conditions are modeled correctly and that all modeling approximations and assumptions are appropriate. This includes appropriate handling of dimensional and material tolerances and uncertainties. The SRP Acceptance Criteria for Section 9.1.1 refer to ANSI/ANS-57.3, which provides in Subsection 6.2.4.1.5 a list of parameters that should be evaluated in the determination of the most reactive fuel assembly, including maximum fissile fuel loading, fuel rod pitch, and fuel rod cladding thickness.

ISSUE

Table 2.4-2 of the criticality technical report provides a list of the mechanical tolerances or variations for the input parameters to the new fuel storage rack model used in a sensitivity analysis for the new fuel storage rack criticality uncertainty evaluation. It is not clear to the staff why the tolerances in this table are not consistent with those for the region I spent fuel storage rack criticality uncertainty evaluations in that the new fuel tolerances do not consider uncertainty in uranium enrichment and both positive and negative tolerances in dimensions such as fuel rod pitch, fuel clad diameter, and rack cell thickness.

INFORMATION NEEDED

In its response and in the DCD or its incorporated reference, the applicant should either (1) provide justification for not including the uncertainty in uranium enrichment and both positive and negative dimensional variations or (2) update the sensitivity analysis for the new fuel storage racks to include these tolerances and apply the revised results to the new fuel storage rack criticality analysis accordingly.

Response - (Rev.1)

Sensitivity analysis for the new fuel storage rack criticality uncertainty evaluation has been performed to include additional items such as uranium enrichment and both positive and negative tolerances in dimensions, such as fuel rod pitch, fuel clad diameter, and rack cell thickness. Criticality analysis TeR will be revised to include the additional sensitivity calculation results.

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

Sections 2.4.2 and 2.5, Tables 2.4-3, 2.4-4, and 2.5-1, Figures 2.4-2, 2.4-3, and 2.5-1 in the criticality TeR will be revised as indicated in the attachment.

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

TS

[ ]TS

[ ]TS

2.4 Criticality Analysis for New Fuel Storage Rack

2.4.1 Criticality Calculation

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

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

2.4.2 Bias and Uncertainty

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

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

b. Statistical uncertainty of the Monte Carlo calculation, and

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

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

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

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

a. Bias: [ ]TS

b. Bias uncertainty: [ ]TS

(2) Statistical uncertainty of the Monte Carlo calculation

a. Optimum moderation (2σ) : [ ]TS

b. Moderation by full density water (2σ) : [ ]TS

Statistical uncertainties of the criticality calculation for the optimum moderation condition and fully flooded condition are provided in Tables 2.4-3 and 2.4-4, respectively.

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

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

These items should be replaced with page 3 of Attachment.

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

a. UO_2 pellet stack density: [ ]TS,

b. UO_2 pellet diameter: [ ]TS,

c. Fuel rod pitch: [ ]TS,

d. Fuel clad outer diameter: [ ]TS,

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

f. NFR cell pitch: [ ]TS, and

a. U-235 enrichment: [ ]TS
b. UO_2 pellet stack density: [ ]TS,
c. UO_2 pellet diameter: [ ]TS,
d. Fuel rod pitch: [ ]TS,
e. Fuel clad outer diameter: [ ]TS,
f. Guide tube outer diameter: [ ]TS
g. Fuel assembly position in fuel rack cell: [ ]TS,
h. NFR cell pitch: [ ]TS, and

i. NFR cell thickness: [ ]TS.

~~g. NFR cell thickness: [ ]TS.~~

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

The uncertainty analyses are performed for the finite array model including NFSP as shown in Figure 2.4-1.

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

Tables 2.4-3 and 2.4-4.

where:

for optimum moderation and fully flooded conditions, which include

k_i = k_{eff} with the tolerance,

k_R = k_{eff} for the reference model,

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

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

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

Tables 2.4-3 and 2.4-4.

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

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

[ ]TS

This equation should be replaced with page 5 of Attachment.

For conservatism, a total uncertainty of 0.01721 will be applied to obtain final criticality analysis results for new fuel rack storage system.

[ ]TS

[ ]TS

Table 2.4-3 Tolerance Uncertainty for Optimum Moderation Condition

TS

This table should be added.

Table 2.4-4 Tolerance Uncertainty for Fully Flooded Condition

TS

This table should be added.

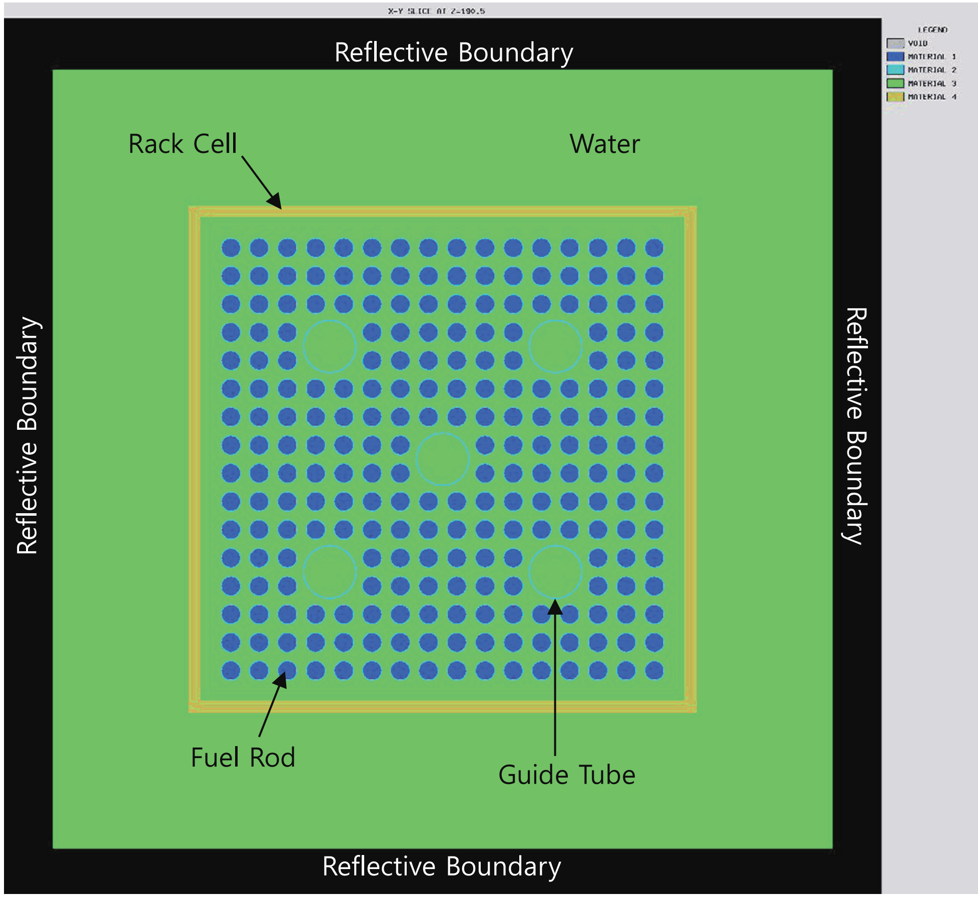


Figure 2.4-2 Unit Cell Model for Uncertainty Analysis

This figure should be replaced with page 9 of Attachment.

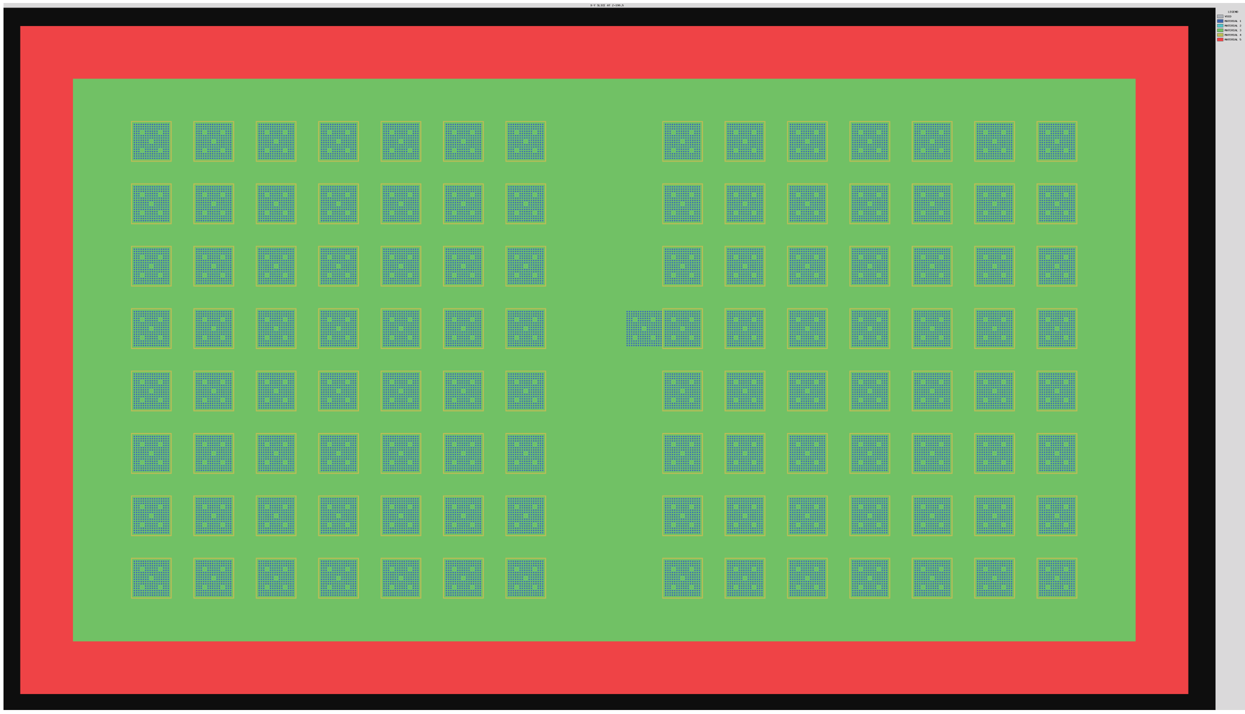


Figure 2.4-2 Criticality Calculation Model for a Dropped Fuel Assembly Accident (Contact)

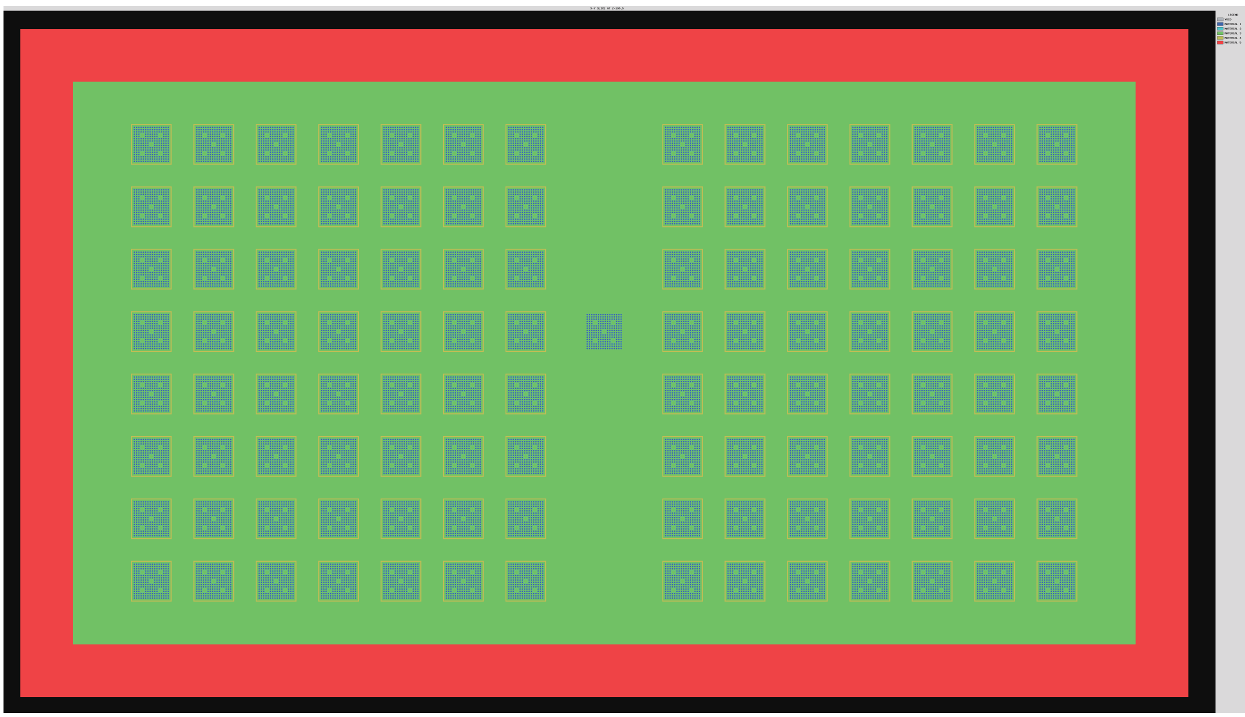


Figure 2.4-3 Criticality Calculation Model for a Dropped Fuel Assembly Accident (Centered)

This figure should be added.

2.5 Results

Table 2.5-2 shows the evaluated k_{eff} for postulated accident cases of a dropped fuel assembly in the space between fuel racks.

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

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

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

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

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

This table should be replaced with table on page 12 of Attachment.

Description	k_{eff}	Acceptance criteria
k_{eff} for flooded by pure water	0.91449	≤ 0.95
k_{eff} for optimum moderation	0.93495	≤ 0.98
k_{eff} for a dropped fuel assembly	0.55226	≤ 0.95

Table 2.5-1 Effective Multiplication Factors for the NFR

TS

This table should be replaced with page 14 of Attachment.

Table 2.5-1 Evaluation Results of k_{eff} for Various Water Densities

TS

TS

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

This figure should be replaced with page 16 of Attachment.

TS

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

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.:	**179-8190**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	**DCD Tier 2, Section 9.1.1, Criticality analysis TeR Sections 2.1, 3.1 and 3.5**
Date of RAI Issue:	**09/01/2015**

Question No. 09.01.01-26

RAI 9.1.1-25: Clarification items for APR1400-Z-A-NR-14011-P, Rev. 0

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 and handling. 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 the completeness and appropriateness of fuel and fuel storage rack design data and their use in the analyses.

ISSUES AND INFORMATION NEEDED

a. Technical report APR1400-Z-A-NR-14011-P, Rev. 0, “Criticality Analysis of New and Spent Fuel Storage Racks,” provides guide tube dimensions in Tables 2.1-1, 3.1-3, and 3.5-3. Table 2.1-1 identifies the dimensions as diameters, whereas Tables 3.1-3 and 3.5-3 identify the same measurements as radii. The staff notes that, based on the actual fuel assembly dimensions, “diameter” is correct for Tables 3.1-3 and 3.5-3. Therefore, the applicant should revise the affected tables in APR1400-Z-A-NR-14011-P, Rev. 0 to reflect that the guide tube dimensions listed correspond to diameters, not radii.

b. A dimension between the new fuel storage racks and the new fuel storage pit wall is shown above the right section of the new fuel storage racks in APR1400-Z-A-NR-14011-P, Rev. 0, Figure 2.1-1 but not labeled with a distance. The applicant should indicate the distance shown by the unlabeled arrow in Figure 2.1-1 of APR1400-Z-A-NR-14011-P, Rev. 0.

c. DCD Tier 2, Section 9.1.1 states that various U-235 enrichments from 1.8 to 5.0 wt% are used for the region II spent fuel storage rack criticality calculation. However, the tables in Section 3.5 of APR1400-Z-A-NR-14011-P, Rev. 0 seem to indicate that the enrichments range from 2.0 to 5.0 wt%. The applicant should revise DCD Section 9.1.1 and/or the criticality analysis technical report to accurately indicate the enrichment levels used for the criticality calculation.

Response - (Rev.1)

a. Tables 3.1-3 and 3.5-3 of the criticality analysis technical report will be revised to correct dimensions.

b. The unlabeled dimension indicated by the arrow in Figure 2.1-1 of the criticality analysis technical report is 178.5 mm, and it will be included in Figure 2.1-1. Also, Figure 2.1-1 will be revised to clarify dimensions of new fuel storage racks.

c. DCD Tier 2, Section 9.1.1 will be revised to correct the initial enrichment from 1.8 to 2.0.

Impact on DCD

DCD Tier 2, Section 9.1.1 will be revised as indicated in Attachment 3.

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

Tables 3.1-3 and 3.5-3 of the criticality analysis will be revised as indicated in Attachment 1. Figure 2.1-1 of the criticality analysis will be revised as indicated in Attachment 2.

Table 3.1-3 Fuel Assembly Design and Operating Data

TS

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

TS

TS

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

TS

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

<u>Spent Fuel Storage Rack</u> (Section 3 of Reference 10)

The following analysis conditions are considered in the design of the spent fuel storage racks:

a. The fuel assembly is assumed to have a maximum enrichment of 5 wt% of U-235 in the criticality calculation for spent fuel storage rack region I. For the normal condition, an infinite array of fresh fuel assemblies is modeled in the criticality calculation. Criticality for damaged fuel assemblies is separately evaluated and the effects of gap between racks are also evaluated.

b. For the region II analyses, an infinite array of 2 × 2 fuel assemblies with various U-235 enrichments, from ~~1.8~~ 2.0 to 5.0 wt%, The moderator of pure water is at the te limits that yields the large y. T assumed to be ~~1,000 kg/cm^3~~ 1.0 g/cm^3 (62.4 lbm/ft^3).

c. Credit is taken for the neutron absorption in the neutron absorbing materials, and only 75 percent of B-10 areal density in the neutron absorbing materials is credited in the analysis. These assumptions provide additional margin in the event that deformation, degradation, or damage to the neutron absorber occur.

~~Credit is taken for the neutron absorption in the rack structural materials and neutron absorbing materials. The steel plate thickness is conservatively set to a minimum, and only 75 percent of B-10 density in the neutron absorbing materials is assumed in order to reflect the deformation of the neutron absorbing material.~~

d. ~~The neutron absorption effects associated with the soluble boron normally in the SFP water are neglected in the criticality analysis for normal operations and are considered for the postulated accidents. The SFP boron concentration is assumed to be about one-half of the minimum concentration level defined in the Technical Specifications.~~

Credit is taken for the soluble boron in the SFP. The SFP soluble boron concentration is assumed to be the same or less than the minimum concentration provided in the LCO 3.7.15.

e. No cooling time is assu not included in the criticality calculation to conservatively evaluate the K_{eff}.

f. The nuclear characteristics of the spent fuel are affected by the core operation parameters, such as coolant temperature, soluble boron concentration in the coolant, and axial burnup profile. Thus, the most severe operating conditions are conservatively assumed in the fuel burnup calculation.