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.:	**90-7939**
SRP Section:	**09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling**
Application Section:	
Date of RAI Issue:	**07/20/2015**

Question No. 09.01.01-1

RAI 9.1.1-1: Detailed description of the spent fuel burnup credit analysis method used in APR1400 design

10 CFR 50.68 prescribes fresh and spent fuel storage and handling system criticality safety requirements. General Design Criterion (GDC) 62 requires that criticality in the fuel storage and handling system be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The APR1400 DCD subsection 9.1.1 and technical report APR1400-Z-A-NR-14011-P, Rev. 0, titled “Criticality Analysis of New and Spent Fuel Storage Racks,” state that the spent fuel pool rack design takes credit for burnup in spent fuel rack criticality safety analyses. Credit was taken for 28 isotopes, grouped into nine major actinides and 19 minor actinides and fission products.

In addition, on page 46 of APR1400-Z-A-NR-14011-P, Rev 0, the applicant refers to depletion uncertainty and the worth of minor actinides and fission products under specific ranges of applicability.

From these statements, it seems that the applicant used a mixture of the approaches provided in the Kopp Memo (Reference 11 of the technical report) and NUREG/CR-7109. However, it was not clear from the DCD or the technical report how these methods were used in the burnup credit analysis for the spent fuel rack. Because both the Kopp Memo and NUREG/CR-7109 treat modeling bias and bias uncertainties as a percentage of the base value of the corresponding reactivity reduction calculated from the depletion analysis code and criticality safety analysis code, the user of these methods must first determine the base values in the separate calculations. However, neither the DCD nor APR1400-Z-A-NR-14011-P, Rev. 0 provides a clear description on how the base values of reactivity reduction were

determined. The applicant is requested to provide a clear description of the method it used in the spent fuel rack criticality safety analysis and demonstrate that the assumptions used and the results produced are conservative. Specifically, the applicant is requested to:

1. Explain how the reactivity reduction was determined for the fuel depletion. This explanation should include all assumptions used.

2. Explain how the base value of the burnup credit is determined for the 28 isotopes, including a description of the method (e.g., correction factor method, direct difference method, or a method beyond those presented in NUREG/CR-6811). If a method other than those presented in NUREG/CR-6811 is used, the applicant should provide a clear technical basis for the validity and reliability of the method.

3. If a mixture of approaches is being used, staff needs an explanation of how the Haut Taux de Combustion (HTC) data were used to benchmark the computer code(s).

4. Update the technical report and DCD as appropriate to clarify the items above.

The staff needs this information to determine if the APR1400 spent fuel rack design meets the regulatory requirements of 10 CFR 50.68 and GDC 62.

Response - (Rev.1)

1. Explain how the reactivity reduction was determined for the fuel depletion. This explanation should include all assumptions used.

The depletion calculations were performed using the ORIGEN-ARP code of SCALE version 6.1.2. The ORIGEN-ARP code performs the depletion calculation with problem-dependent cross section libraries, which were generated by an interpolation algorithm with pre-generated cross section data. However, the pre-generated cross section data for PLUS7 fuel assembly was not included in the ORIGEN-ARP code. Therefore, additional analyses were performed to calculate the pre-generated cross section data for PLUS7 fuel assembly.

The pre-generated cross section data for PLUS7 fuel assembly were calculated by using the TRITON/NEWT sequence of SCALE version 6.1.2. To calculate the pre-generated cross section data a two dimensional reactor physics model for the APR1400 design was constructed and depletion calculations were performed up to the maximum burnup for each initial enrichment. Once the pre-generated cross sections were determined, they were inputted into the ORIGEN-ARP code to determine the reactivity reduction.

The conservative assumptions for parameters used in depletion calculation that are provided in Table 3.5-1 of the criticality analysis technical report (TeR) are as follows:

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The following considerations related to burnup credit (BUC) are provided below:

- Axial burnup profile: When modeling the fuel assembly in the criticality analysis, the reactivity is affected by the distribution of burnup along the axial length of the fuel assembly. Therefore, sensitivity analyses were performed with and without a bounding axial burnup profile to evaluate the magnitude of the end effect, which is applied as a bias. Biases due to axial burnup profile were provided in Table 3.5-17 of TeR.

- Burnable absorber: Burnable absorber effect on the reactivity of the fuel should be evaluated because it hardens the energy spectrum during operation. As a result of this evaluation, the k_{eff} for the fuel assembly without burnable absorbers is greater than that for the fuel assembly with burnable absorbers. The reason for this result is that gadolinium always has a net negative reactivity worth. This result is consistent with NUREG/CR-6760. Therefore, burnable absorbers were not considered in the criticality analysis.

- Axial blanket: The PLUS7 fuel assembly has blankets at the top and bottom of the fuel rod. Sensitivity analyses have been performed to assess the magnitude of the blanket effect. The result of comparison of the k_{eff} with and without blankets show that the non-blanketed fuel is more reactive up to about 3 % in k_{eff} as presented in Tables

3.5-15 and 3.5-18 of TeR. Therefore, axial blankets in the fuel rod are not considered for conservatism.

The following biases and uncertainties related to BUC are provided below:

- Depletion Uncertainty: To cover the uncertainty in the isotopic number densities generated during the depletion calculations, the depletion uncertainty provided in Kopp's memo is applied to the criticality analysis. Detailed information about depletion uncertainty is provided in the response to the 2nd question in this response.

- Validation of nuclides considered in the criticality analysis: To validate the nuclides considered in the criticality analysis, the recommendations provided in NUREG/CR-7109 were applied to the criticality analysis. Detailed information about validation of nuclides considered in the criticality analysis is provided in the response to the 2nd question in this response.

- Reactor Burnup Record Uncertainty: The reactor burnup record uncertainty is an uncertainty representing the deviation between the true burnup of a fuel assembly and the burnup based on the reactor record. The reactor burnup record uncertainty is typically less than 5 % based on NUREG/CR-6998. Therefore, the reactivity difference due to the 5 % change in burnup is applied to the reactor burnup record uncertainty. The uncertainties due to reactor burnup record are provided in Table 3.5-12 of TeR.

2. Explain how the base value of the burnup credit is determined for the 28 isotopes, including a description of the method (e.g., correction factor method, direct difference method, or a method beyond those presented in NUREG/CR-6811). If a method other than those presented in NUREG/CR-6811 is used, the applicant should provide a clear technical basis for the validity and reliability of the method.

To validate the methodology used in the depletion analysis, a mixture of the approaches provided in Kopp's memo (Reference 11 of TeR) and NUREG/CR-7109 were applied to the criticality analysis for spent fuel pool (SFP) of the APR1400.

Depletion Uncertainty

To cover the uncertainty in the isotopic number densities generated during the depletion calculations, the depletion uncertainty provided in Kopp's memo is applied to the criticality analysis.

The depletion uncertainty is taken to be 5 % of the reactivity difference between the reactivity at the fresh fuel condition and the reactivity at the burned fuel condition. The depletion uncertainty is calculated by the following equation:

$$\text{Depletion Uncertainty} = 0.05 \times \left(k_{fresh} - k_{burned} + 1.645 \sqrt{\sigma_{fresh}^2 + \sigma_{burned}^2} \right),$$

Where,

k_{fresh}: Calculated k_{eff} of fresh fuel condition,
k_{burned}: Calculated k_{eff} of burned fuel condition,
σ_{fresh}: Standard deviation of k_{fresh},
σ_{burned}: Standard deviation of k_{burned}, and
1.645: One-sided 95/95 confidence interval factor.

The k_{fresh} values were calculated by criticality calculations with fresh fuel condition, and other bias and uncertainty are not included in these values. The k_{burned} values were calculated by criticality calculations with burned fuel condition. To model burned fuel, isotopic number densities obtained from the depletion calculations were applied to the criticality calculations, and other bias and uncertainty are not included in these values. The k_{fresh} and k_{burned} values are provided in Table 3.5-5 of TeR, and σ_{fresh} and σ_{burned} values are provided in Table 3.5-6 of TeR. The depletion uncertainties are provided in Table 3.5-13 of TeR.

Validation of isotopes applied to the criticality analysis

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3. If a mixture of approaches is being used, staff needs an explanation of how the Haut Taux de Combustion (HTC) data were used to benchmark the computer code(s).

As discussed in the response to the 2nd question, the validation of major actinides was performed using benchmark analysis including HTC experiment data. Detailed information about the benchmark analysis is provided in Reference 7 of TeR (WCAP-17889-P, “Validation of SCALE 6.1.2 with 238-Group ENDF/B.VII.0 Cross Section Library for APR1400 Design Certification”).

4. Update the technical report and DCD as appropriate to clarify the items above.

DCD 9.1.1 and TeR Section 3 will be revised to clarify the BUC methodology applied to the criticality analysis for spent fuel pool of APR1400.

Impact on DCD

DCD subsection 9.1.1.3.4 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

Subsections 3.4 and 3.5 of the criticality analysis TeR will be revised as indicated in Attachment 2.

g. No CEAs or burnable poison rods in the fuel assembly, and no neutron absorption effect in the fuel assembly support materials are assumed to be present.

h. The rack is filled with fuel assemblies up to the initial onsite storage capacity with the SFP filled with water.

i. The bias and bias uncertainty obtained from benchmark calculation are reflected in the calculated K_{eff}. Uncertainties from mechanical tolerances and variations in the design parameters are added to the total uncertainty. For region II analyses, the effects of axial burnup profile and burnable poison rods, and uncertainty from the depletion calculation methodology are considered in the calculation K_{eff}.

Calculated K_{eff} including all biases and uncertainties is compared with design criterion in 10 CFR 50.68 in order to show the subcriticality of the new and spent fuel storage racks for normal conditions and postulated accidents. For the spent fuel pool region II, the curve for the minimum-burnup and initial loading enrichment is generated based on the K_{eff}s calculated for each enrichment and burnup.

9.1.1.3.4 Criticality Analysis for New and Spent Fuel Storage Racks

~~Criticality analyses for new and spent fuel storage racks are performed. The results show that the design criterion in 10 CFR 50.68 is met and that the subcriticality is maintained (Reference 10).~~

This paragraph should be replaced with paragraphs on pages 2 and 3 of Attachment 1.

9.1.2 New and Spent Fuel Storage

9.1.2.1 Design Bases

The following design bases are imposed on the storage of new and spent fuel assemblies:

New Fuel Storage

a. The new fuel storage pit is protected from the effects of natural phenomena, including earthquakes, tornadoes, hurricanes, floods, and external missiles. The new fuel storage racks meet the guidance in NRC RG 1.13 (Reference 11), NRC RG 1.29 (Reference 12), NRC RG 1.115 (Reference 13), NRC RG 1.117

9.1.1.3.4 Criticality Analysis for New and Spent Fuel Storage Racks

Criticality analyses for the new and spent fuel storage racks have been performed to demonstrate compliance with 10 CFR 50.68. A high level overview of each analysis is given below. Full details of these analyses can be found in Technical Report APR1400-Z-A-NR-14011-P, Rev.0 (Reference 10).

9.1.1.3.4.1 Criticality Analysis for New Fuel Storage Racks

The new fuel storage racks are required to maintain a K_{eff} no greater than 0.95 under fully flooded conditions and no greater than 0.98 under optimum moderation conditions at a 95 percent probability with 95 percent confidence level (95/95 level).

Compliance with these regulatory requirements has been demonstrated by analyzing the storage of fuel of the maximum allowable reactivity in the new fuel storage racks at moderator densities between 0.01 and 1.00 g/cm^3. To ensure that the required K_{eff} s were met at a 95/95 level, the analysis accounted for the potential reactivity impact of the tolerances associated with both the rack and fuel design. The analysis additionally accounted for the bias and bias uncertainty associated with the Monte Carlo code's calculation of reactivity. The details of this analysis are provided in Section 2 of Reference 10.

9.1.1.3.4.2 Criticality Analysis for Spent Fuel Storage Racks

The spent fuel storage racks are required to maintain a K_{eff} of less than 1.0 under normal conditions at a 95/95 level assuming the pool is flooded with unborated water and a K_{eff} not greater than 0.95 at a 95/95 level under normal and accident conditions if flooded with borated water.

Compliance with these regulatory requirements has been ensured by developing storage requirements as a function of burnup and initial enrichment. To create the storage requirements, the following steps are taken:

a. Conservative isotopics are developed by performing depletion calculations modeling reactor operating conditions to be more limiting than anticipated operating conditions
b. A single limiting set of biases and uncertainties are calculated which bound all initial enrichments
c. Burnup requirements are set by calculating the burnup at which nominal system reactivity plus the biases and uncertainties, remains below 1.0 assuming an unborated pool at various initial enrichments

Once the burnup requirements have been determined, the amount of soluble boron necessary to maintain a K_{eff} no greater than 0.95 under normal conditions is calculated. The details of the analyses performed to develop the burnup requirements are provided in Section 3 of Reference 10.

Finally, accident conditions are analyzed. For the boron dilution event it is demonstrated that a boron dilution event capable of causing SFP reactivity to increase above a K_{eff} of 0.95 under otherwise normal conditions is not credible. Each credible non-dilution accident condition is analyzed and the requisite amount of soluble boron necessary to maintain a K_{eff} no greater than 0.95 is calculated. The details of the accident analyses performed are provided in Section 4 of Reference 10.

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

a. Bias and bias uncertainty of the criticality calculation method,
b. Statistical uncertainty of the Monte Carlo calculation,
c. Uncertainty due to tolerances or variations in the design parameters,
d. Uncertainty due to eccentric fuel assembly positioning, and
e. Bias due to pool cooling water temperature.

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

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

3.4.3.1 Bias and Bias Uncertainty due to Methodology

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

Table 3.4-2 provides the 95/95 bias and bias uncertainty developed in Reference 7. Table 3.4-7 provides the value used in the criticality analysis, which doubled the bias uncertainty provided in Table 3.4-2 for the purpose of conservatism.

3.4.3.2 Uncertainty due to Monte

Statistical uncertainties due to Mont

3.4.3.3 Uncertainties due to Mech

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

3.4.3.3.1 Fuel Assembly

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

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

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3.5 Criticality Analysis for Spent Fuel Pool Region II

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

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

3.5.1 Depletion Calculations

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

Zero cooling time

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

3.5.1.1 Bounding Reactor Parameters for the Depletion Calculation

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

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

3.5.1.2 ORIGEN-ARP Calculation

These paragraphs should be replaced with paragraphs on page 4 of Attachment 2.

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

3.5.1.3 TRITON-NEWT Calculation

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

The bounding reactor parameters are used for the depletion calculation for conservatism. The reactor parameters are a fuel temperature, a fuel density, a moderator temperature, a soluble boron concentration and a power level as presented in Table 3.5-1.

a. Fuel temperature: Higher fuel temperature causes Doppler broadening and it results in increased plutonium production. Therefore, the maximum fuel temperature of 1450 K (peak fuel temperature at the power of 116 % with additional margins) is applied to the depletion calculations.

b. Fuel density: To maximize fissile material, the maximum pellet density of 10.519 g/cm^3 was applied to the depletion calculations.

c. Moderator temperature: Higher moderator temperature causes less moderation and it results in energy spectrum hardening. Therefore, the maximum moderator temperature of 617.98 K (saturation temperature at 2250 psia) is applied to the depletion calculations.

d. Soluble boron concentration: Higher soluble boron concentration causes energy spectrum hardening. Therefore, the maximum cycle average soluble boron concentration of 855 ppm (maximum cycle average soluble boron concentration with additional margin) was applied to the depletion calculations.

e. Power level: The sensitivity analysis is performed to identify the trend with respect to a power level, and it is found that there is no trend with respect to power level as shown in Table 3.5-2. So, the maximum power level corresponding to the maximum fuel temperature is used as a bounding reactor parameter.

f. Partial nuclide credit: The partial nuclide credit was taken for the purpose of conservatism. So only 28 nuclides presented in Table 3.2-1 are considered in criticality analysis.

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

3.5.2 Criticality Calculations

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

3.5.3 Bias and Uncertainty Calculations

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

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

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

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

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

3.5.3.1 Bias and Bias Uncertainty due to Methodology

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

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

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

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

Add superscript "*"

Add footnote

* Doubled bias uncertainty provided in Table 3.4-2 is applied to the criticality analysis for the purpose of conservatism.

3.5.3.4 Bias for Minor Actinide and Fission Product

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

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

a. Low enriched fuel (< 5.0 wt% U-235) with ENDF/B-VII cross section library,
b. Maximum burnup is 70 GWd/MTU, and
c. Total minor actinide and fission product nuclide worth does not exceed 0.1 in k_{eff}.

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

for all burnup and enrichment range.

3.5.3.5 Bias due to Pool Cooling Water Temperature

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

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

3.5.3.6 Uncertainty due to Eccentric Fuel Assembly Positioning

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

Reactor Burnup Record

3.5.3.7 Uncertainty due to ~~Burnup Measurement~~

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

This paragraph should be replaced with paragraphs on page 7 of Attachment 2.

3.5.3.8 Uncertainty due to Depletion

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

This paragraph should be replaced with paragraphs on page 8 of Attachment 2.

3.5.3.7 Uncertainty due to Reactor Burnup Record

The reactor burnup record uncertainty is an uncertainty representing the deviation between the true burnup of a fuel assembly and the burnup based on the reactor record.

The fuel assembly burnup is generated by using the CECOR computer code. The CECOR computer code synthesizes a three-dimensional assembly and the peak pin power distributions using signals from fixed in-core detectors. When performing a CECOR calculation, snapshots are selected to be representative of a period of reactor operation. The information contained in the burnup record includes the number of assemblies, number of axial nodes, core average exposure and axial exposures for all assemblies.

The reactor burnup record uncertainty is typically less than 5 % based on NUREG/CR-6998 (Reference 10). Therefore, the reactivity difference due to the 5 % change in burnup is applied to the reactor burnup record uncertainty. The uncertainties due to the reactor burnup record are provided in Table 3.5-12.

3.5.3.8 Uncertainty due to Depletion

To cover the uncertainty in the isotopic number densities generated during the depletion calculations, the depletion uncertainty provided in Kopp's memo (Reference 11) is applied to the criticality analysis.

The depletion uncertainty is taken to be 5 % of the reactivity difference between the reactivity at the fresh fuel condition and the reactivity at the burned fuel condition. The depletion uncertainty is calculated by the following equation:

$$\text{Depletion Uncertainty} = 0.05 \times \left(k_{fresh} - k_{burned} + 1.645 \sqrt{\sigma_{fresh}^2 + \sigma_{burned}^2} \right),$$

Where,

k_{fresh}: Calculated k_{eff} of fresh fuel condition,
k_{burned}: Calculated k_{eff} of burned fuel condition,
σ_{fresh}: Standard deviation of k_{fresh},
σ_{burned}: Standard deviation of k_{burned}, and
1.645: One-sided 95/95 confidence interval factor.

The k_{fresh} and k_{burned} values are provided in Table 3.5-5, and σ_{fresh} and σ_{burned} values are provided in Table 3.5-6. The depletion uncertainties are provided in Table 3.5-13.