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Comparison of Bias and Uncertainty of ENDF/B-VII with ENDF/B-VI for Spent Fuel Criticality Analysis

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1. Introduction

Accurately evaluating the criticality of spent fuel storage rack is of importance to the analysis for the safety of the fuel storage rack and the reliability of shielding design. Because of this request for assessment, the Monte Carlo simulation has been an appropriate approach in the criticality analysis to estimate due to its general geometry modeling capability, correct representation of transport effects and continuous energy cross sections. Validation and bias evaluation test for the criticality analysis of spent fuel storage rack with Montel Carlo Code MCNP5 with ENDF/B-VI library to use criticality evaluation were performed in 2010 [1, 2].

In this paper, based on the same methodology in the past test, bias and uncertainty are recalculated by using Monte Carlo Code MCNP6.1 with ENDF/B-VII library [3]. The benchmark problem cases were selected from the International Handbook of Evaluated Criticality Safety Benchmark Validation Experiments (IHECSBE) [4] by considering the typical PWR specifications such as the use of LEU and square-shape lattice structure of fuel assembly.

2. Method and Result

2.1 Methodology

As mentioned, Monte Carlo code has been widely employed in criticality analysis and shielding design field. For the validity of the code, nevertheless, it shall be proved in order to be applied in analysis. The reliability of codes can be confirmed by comparing the calculated effective multiplication factor (k_{eff}) with the measured value.

In general, k_{eff} in nominal condition is calculated with given geometric configuration and material compositions. Moreover, abnormal conditions are evaluated, which includes geometric malposition, change of designed variables, mechanical errors, and so on. For the criticality analysis considering burnup, additionally, the effective multiplication factor shall be calculated with all the fission products and heavy nuclides generated in depleted fuel materials. Because it is not straightforward to measure the fission products and heavy nuclides directly, however, burnup calculations shall be used to estimate the spent fuel fission product.

The validity of the method employed in this paper has been confirmed, and the final effective multiplication factor can be calculated as follows [5]:

$$k_{eff} = k_{cal} + \Delta k_{bias} + \Delta k_{uncertainty} + \Delta k_{burnup}$$

where,
k_{cal}: Calculated nominal value of the effective multiplication factor
Δk_{bias}: Bias in criticality analysis methodology
$\Delta k_{uncertainty}$: Uncertainty due to design variables and calculation
Δk_{burnup}: Calibration factor due to axial burnup

Bias for the criticality analysis methodology is a mean value of difference between 1 and the ratio of calculated effective multiplication factor to the value of the effective multiplication factor of the critical experiment. The bias value is calculated as following equation [2].

$$\Delta k_{bias,i} = \frac{k_{exp,i} - k_{cal,i}}{k_{exp,i}}$$

$$\Delta k_{bias,avg} = \frac{1}{n}\sum_{i=1}^{n} \Delta k_{bias,i}$$

Also, a standard deviation of the bias calculation value is obtained via the following equation.

$$\sigma(\Delta k_{bias,avg}) = \sqrt{\frac{1}{n-1}\sum_{i=1}^{n}(\Delta k_{bias,i} - \Delta k_{bias,avg})^2}$$

A statistical uncertainty of the calculated bias value must be considered in the criticality safety evaluation. To apply 95% probability of uncertainty at the 95% confidence interval, one sided tolerance limit factor, which corresponds to the number of benchmark cases used in this analysis, shall be considered. The one sided tolerance limit factor that corresponded to 272 criticality experiments evaluated according to the Coefficient Calculation Method [6] was calculated, which was 1.772. However, 2.0 was applied in this paper as the one sided tolerance limit factor for conservative calculation so that uncertainty was calculated via $2\sigma(\Delta k_{bias,avg})$. [2]

2.2 Case Selection

For the criticality validation calculation, criticality experiments using UO_2 fuel in the IHECSBE [4] had been selected due to appropriate consideration of characteristics for performing criticality analysis on the typical PWR fuel pool. These critical experimental data with several types of nuclear safety validation data from all over the world are used to validate the criticality calculation method. For the selection, the characteristics of light water reactor fuel storage facilities had been considered as follows [2]:

- Fuel (UO_2) is mounted inside the cladding as a pellet type.
- The alignment of fuel rod is composed of a square lattice structure.
- Water is present as a moderator.
- Neutron absorber or stainless steel plate is present between fuel assemblies.
- Fission occurs by thermal neutron mainly.

The selected criticality cases were 18 types out of 272 LEU criticality experiments due to characteristics shown above. The cases are listed in Table 1,youtu and one sample of the geometry shape is shown in Figure 1. The bias value by the criticality analysis methodology is significantly influenced by the cross-section data used in the calculation and computational code. Therefore, endf70c(.70C) for nuclear fuel materials and sab20022(.66t) for thermal neutron scattering reactions were used for this calculation [3]. For setting up the criticality calculation, input files that include the KCODE card which are 30000 particle histories, 50 inactive cycles, and 250 active cycles were set.

Table 1 List of criticality cases used in MCNP6.1 for bias calculations

Evaluation ID	Cases
LEU-COMP-THERM-001	8
LEU-COMP-THERM-002	5
LEU-COMP-THERM-003	22
LEU-COMP-THERM-004	20
LEU-COMP-THERM-006	18
LEU-COMP-THERM-008	17
LEU-COMP-THERM-009	27
LEU-COMP-THERM-010	30
LEU-COMP-THERM-012	10
LEU-COMP-THERM-013	7
LEU-COMP-THERM-014	7
LEU-COMP-THERM-016	32
LEU-COMP-THERM-017	3
LEU-COMP-THERM-042	7
LEU-COMP-THERM-051	19
LEU-COMP-THERM-062	15
LEU-COMP-THERM-065	17
LEU-COMP-THERM-079	10
Total	272

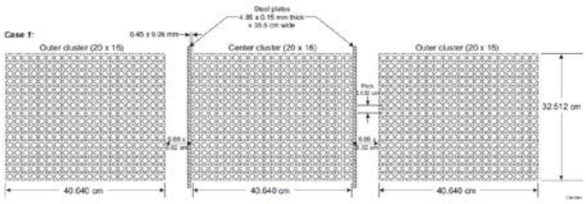

Figure 1 Critical Configuration of Case 1 of LEU-COMP-THERM-016 [7]

2.3 Result

The results of the effective multiplication factors with standard deviation by each case are presented in Figure 2 and distribution is shown in Figure 3. The criticality validation calculations show that the effective multiplication factors are in a range of 0.98094 to 1.00903. The mean effective multiplication factor of the validation calculation is 0.99735. The bias that shall be considered in the k_{eff} calculated by the MCNP6.1 was calculated as +0.00265. For a standard deviation of the bias calculation value, σ was calculated as 0.005167. The 2σ at 95% probability with 95% confidence interval was calculated as 0.010334 in which the conservative tolerance limit factor was applied. Compared to MCNP5 results [2], the mean of effective multiplication factor increases slightly which means bias decrease, and the value of uncertainty is about the same as shown in Table 2.

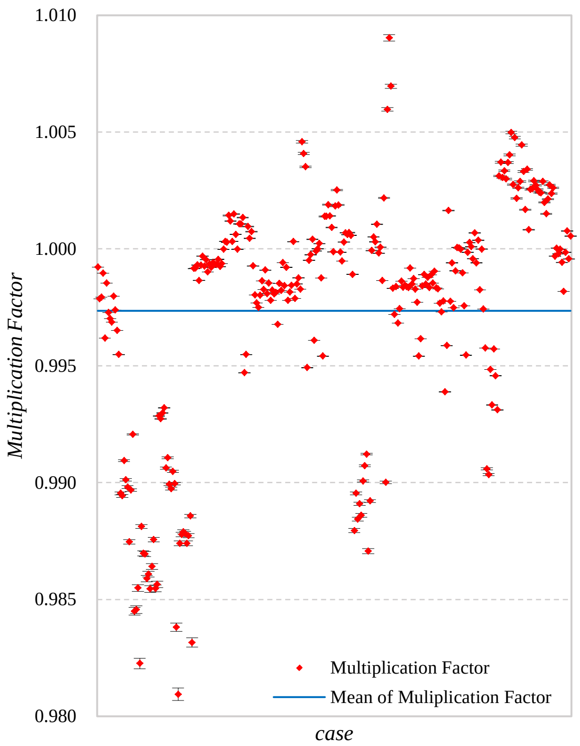

Figure 2 Effective multiplication factors with standard deviation used in MCNP6.1

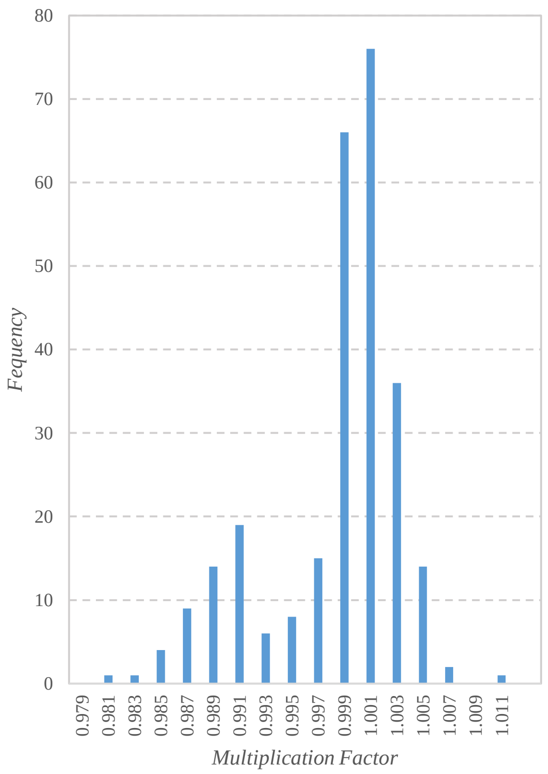

Figure 3 Distribution of effective multiplication Factors

Table 2 Calculation results of bias and uncertainty

	Mean of k_{eff}	Bias ($1-k_{cal}$)	Uncertainty (2σ)
ENDF/B-VI [2]	0.99448	+0.00552	0.01051
ENDF/B-VII	0.99735	+0.00265	0.01033

3. Conclusions

Bias and uncertainty evaluation of MCNP6.1 with ENDF/B-VII using criticality safety benchmark experiments had been performed. The criticality validation calculation shows that the bias is 0.00265 with having uncertainty 0.01033. Also, the effective multiplication factors are in a range of 0.98094 to 1.00903 and all calculation results are within 2% difference with those of experiment values. Therefore, the MCNP6.1 with ENDF/B-VII are appropriate to evaluate the typical PWR criticality safety analysis. This result will be applied in criticality analysis of spent fuel storage rack.

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