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Criticality Analysis of Spent Fuel Storage System Using McCARD Code

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1. Introduction

Spent fuels generated during the operation of commercial nuclear reactors generate radiation and decay heat for a long period of time because they contain fission products even after withdrawal from the reactor. Therefore, it is very important to safely store and manage spent fuel.

The purpose of the criticality analysis is to ensure that the spent fuel storage system is in a subcritical state under sufficiently conservative conditions. Recently, the Monte Carlo(MC) code has been widely used to establish subcritical limits. This is because MC code is suitable for complex geometrical modeling in three dimensions and is easy to interpret for neutron transport [1]. However, the MC code can not be applied to criticality analysis without a series of validation procedures. Therefore, all MC codes must be verified before they can be used for criticality analysis. McCARD, one of the MC codes, was developed by Seoul National University. This code has been permitted by the Korea Institute of Nuclear Safety(KINS) in the design process of Jordan reactor. However, the McCARD has not been used for criticality analysis of spent fuel storage and transport systems.

In this study, the applicability of McCARD to the analysis of the spent fuel storage system was validated and the Upper Subcritical Limit(USL), which is the criterion for determining the subcriticality of the system, was determined. Finally, a criticality analysis of the spent fuel storage system in actual operation will be performed based on the established USL of McCARD.

2. Upper Subcritical Limits

The United State standard defines the USL as a criterion for determining that an unknown system is in a subcritical state even within a sufficient margin. The USL is determined based on the uncertainty and bias associated with the code and data used in the calculations for well-known systems.

To be sure that the multiplication factor(k_s) calculated in the unknown system, which is expected to be a subcritical state, is less than or equal to the maximum multiplication factor based on the multiplication factor of the sample, uncertainty, and additional margin terms [2].

$$k_s \le k_c - \Delta k_s - \Delta k_c - \Delta k_m, \tag{1}$$

where
k_c = mean value of resulting from the calculation of benchmark criticality experiments using a specific calculational method and data,
Δk_s = uncertainty in the value of k_s,
Δk_c = uncertainty in the value of k_c,
Δk_m = additional margin to ensure subcriticality.

If the calculational bias(β) is defined as $\beta = k_c - 1$, the uncertainty of the calculational bias is equal to the uncertainty of k_c (i.e., $\Delta\beta = \Delta k_c$). If β is a positive value, β is assumed to be 0 to conservatively determine USLs. Using the newly defined β and the $\Delta\beta$, Eq (1) can be rewritten as Eq (2) and consequentially the USL can be defined as Eq (3):

$$k_s + \Delta k_s \le 1 - \Delta k_m + \beta - \Delta\beta, \tag{2}$$

$$USL = 1 - \Delta k_m + \beta - \Delta\beta. \tag{3}$$

Based on a statistical analysis of the critical experiments, the USL is determined as a function of design parameters that affect the criticality of the fuel system, such as Average Energy of Fission(AEF), uranium enrichment, and fuel/moderator ratio. For the USL establishing, a linear regression line is derived from the regression analysis between the design parameters and the multiplication factors. The USL is finally established by applying the additional margin, the uncertainty presented by each methodology.

In this section, the USL will be established based on 167 critical experimental data of spent fuel storage and transport systems provided by reference 3. It is performed based on uranium enrichment, which utilized the five methodologies described in NUREG/CR-6361[2] and NUREG/CR-6698[4]. A detailed of the five methodologies are given in reference 2 and reference 4. Table I shows the USL set using five methodologies as a function of uranium enrichment based on the 167 critical experiments [5].

Table I: Upper Subcritical Limits as a function of uranium enrichment based on 167 experiments [5]

Case	Method 1	Method 2	Method 3	Method 4	Method 5
USL	0.9454	0.9909	0.9414	0.9712+4.000E-05*X	0.9403

3. Criticality Analysis of Spent Fuel System

In this section, a criticality analysis of the actual spent fuel storage pool(SFP) was performed using the McCARD. In general, the storage of spent fuel is preferred to a wet storage method that is efficient in shielding radiation and removing residual heat. The wet storage method stores spent fuel assemblies in a tank filled with light water. Fig. 1 is the configuration of the actual SFP.

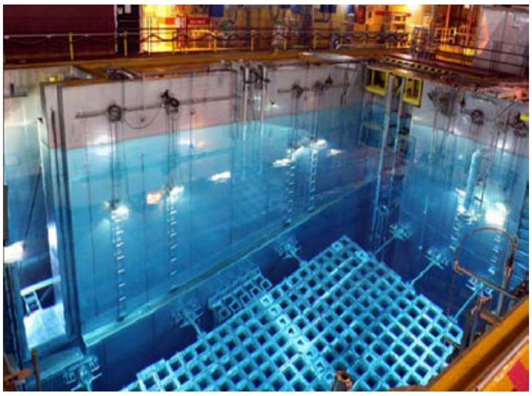

Fig. 1. Spent fuel storage pool and rack

A typical SFP consists of a region I storing fresh fuel assembly, a region II storing spent fuel, and region II is divided into regions A and B again. The model to be used for the criticality analysis was selected as the SFP region II-A in Yeonggwang unit 5 and 6.

3.1. Depletion Calculation

Criticality analysis for the SFP is performed in two steps. The first step is to simulate the actual burning of nuclear fuel assemblies in the core. This process is called the depletion calculation. This step using the MC code confirms the variation of the nuclides generated by the nuclear fission in the fuel assembly loaded in the core [6]. Depletion calculation using the McCARD were performed on the PLUS7 type fuel assembly of the APR-1400 reactor. The design parameters for the depletion calculation model are summarized in Table II, and the configuration is shown in Fig. 2 [7].

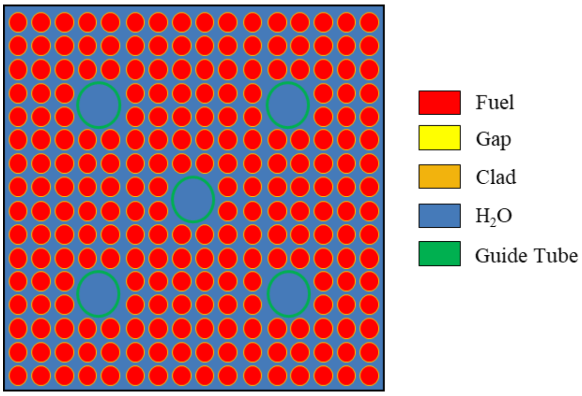

Fig. 2. Configuration of the depletion model

Table II: Specifications of depletion model [7]

Parameter	Value
Total power, MWt	2,815
Fuel density, g/cm^3	10.313
Moderator density, g/cm^3	0.66
Array	16 x 16
Pellet diameter, cm	0.8192
Clad inner diameter (ID), cm	0.8357
Clad outer diameter (OD), cm	0.9500
Pin pitch, cm	1.2852
Assembly width, cm	20.229
Guide tube inner diameter, cm	2.2860
Guide tube outer diameter, cm	2.4890
Fuel active height, cm	381.0
Fuel temperature, K	900
Moderator temperature, K	600
Boron concentration, ppm	680

The spent fuel stored in the actual SFP is targeted to the all fuel assemblies depleted in the reactor. However, the depletion model used for the criticality analysis used a single fuel assembly. This is to increase the conservativeness of the criticality analysis by assuming that the fuel assemblies are arranged infinitely in the X and Y directions and to exclude structures that do not affect the criticality. Depletion calculations using the McCARD code are generally performed based on the assumptions presented below [6]:

A. The fuel assemblies are infinitely arranged in the X and Y directions.
B. A water reflector with a thickness of 30 cm is placed on the top and bottom of active fuel.
C. Neutron absorber (burnable poison, control rod) are not considered.
D. The actual fuel assemblies are zoned to decline the power peaks of the fuel rods adjacent to the guide tubes and corners, but apply the same degree of enrichment to all fuel rods in the depletion calculation model.

When performing depletion calculations using the McCARD, 100 active cycles, 20 inactive cycles, and 100,000 histories per cycles were applied. In addition, a reflective boundary condition was applied in the radial direction and a vacuum boundary condition was applied in the axial direction in order to assume that a single fuel assembly was arranged radially infinitely. Depletion calculations of the fuel assembly were performed at 0.5 wt% intervals from 1.5 wt% to 5.0 wt%, and the nuclide data was calculated by dividing each enrichment by the 2,500MWd/MTU step at 0-60,000MWd/MTU burnup range. The ENDF/B-VII.0 library was used to calculate the fuel assembly depletion using the MCARD. Fig. 3 is a change in the behavior of the multiplication factor depending on the initial enrichment of the depletion model.

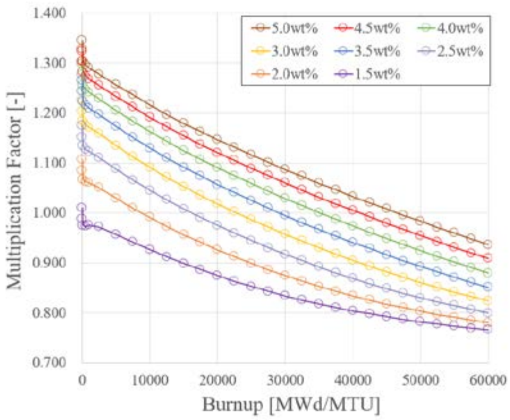

Fig. 3. Multiplication factor for PLUS7 fuel assembly as a function of burnup

3.2. Criticality Calculation

The criticality calculation using the McCARD was performed on a spent fuel storage rack II, which temporarily stores spent fuel, and the actual spent fuel storage rack of Yeonggwang unit 5 and 6 [8]. The specifications of spent fuel assembly are the same as the PLUS7 type described in Table II, which is located in the center of a spent fuel storage rack II with a width of 22.0 cm. The rack consists of a grid of 0.25 cm thick stainless steel, which serves to separate each spent fuel. Between these grids, a 0.25 cm thick neutron absorber consisting of BORAL is placed to lower the criticality of spent fuel assemblies. The specifications for the criticality calculation model are described in Table III and Fig. 4 shows the configuration of the spent fuel storage rack model. Fig. 4 is modeled around four corners of the storage rack adjacent to the spent fuel, and it is assumed that the storage racks are arranged infinitely if this model is repeated in the radial direction.

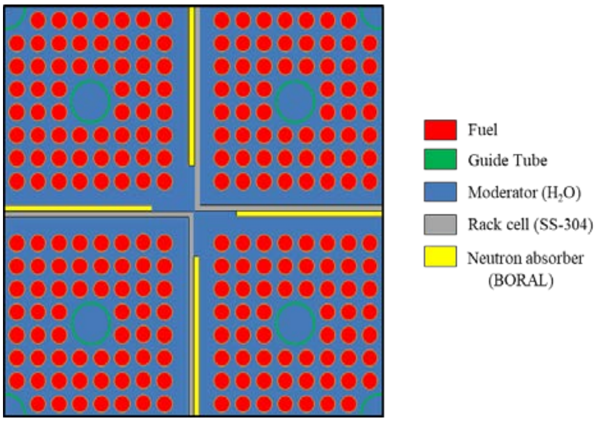

Fig. 4. Configuration of spent fuel storage rack region II using criticality calculation

Table III: Specifications of spent fuel rack II [8]

Parameter	Value
Cell height, cm	459.0
Rack cell pitch, cm	22.5
Cell inner width, cm	22.0
Cell wall thickness, cm	0.25
Cell material	SS-304
Neutron absorber thickness, cm	0.25
Neutron absorber material	BORAL
Neutron absorber width, cm	18.40

There are various uncertainties in the criticality analysis using MC codes. There sometimes has a non-conservative effect on the safety of the system. Therefore, some assumptions apply from a conservative condition [9].

A. The enrichment of U^{235} is uniform in the axial direction, and the axial blanket is not considered.
B. The structure supporting the assembly ignores the effect of neutron absorption assuming water.
C. The soluble boron contained in the moderator is not considered and the density is assumed to be 1.0 g/cc.
D. The spent fuel storage rack and fuel assemblies are arranged infinitely in all directions.
E. The cooling period after reactor shutdown of spent fuel is no considered.
F. It is assumed that the axial burnup distribution is uniformed.
G. The boron concentration of the neutron absorber plate adhered to the rack assumes 90% of the normal concentration.

In addition, it is most reasonable to calculate the criticality calculation including all the nuclides generated, but the actual criticality analysis has been conservatively determined to take into account various uncertainties and safety. These nuclides are listed in Table IV [10].

Table IV: Nuclides used in criticality analysis [10]

Set for nuclides for actinides and fission products (28)			
^{234}U	^{235}U	^{236}U	^{238}U
^{238}Np	^{238}Pu	^{239}Pu	^{240}Pu
^{241}Pu	^{242}Pu	^{241}Am	^{243}Am
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh
^{109}Ag	^{133}Cs	^{143}Nd	^{145}Nd
^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd

The number density of 28 nuclides calculated from the depletion calculation using McCARD was used as the input data for the criticality calculation of the SFP.

3.3. Loading Curve

The multiplication factor of the SFP should not exceed the regulatory limit, including uncertainties and additional margins. Therefore, it is very important to establish regulatory limits on the SFP and to ensure that all fuel assemblies to be stored meet specified limits. This section evaluates the minimum burnup that meets the USLs established by the five methodologies described in section 2 is determined. And, a loading curve is generated as a function of the initial enrichment and minimum burnup that can be stored in the storage system. When performing a criticality calculation, 20 inactive cycles, 100 active cycles, and 100,000 histories per cycles were applied, and a reflective boundary condition was applied in the radial and axial directions. For the calculation, the ENDF/B-VII.0 continuous energy group library was used.

Fig. 5 show the results of multiplication factor and USL3 and USL5 of criticality calculation model depending on initial enrichments. Depending on initial enrichment and burnup of the spent fuel assemblies, the minimum burnup that satisfies USLs is determined. It is considered that the fuel assemblies burned more than the minimum burnup (area under USLs) can be stored in the SFP. On the other hand, assemblies in burnup areas that do not satisfy the USLs can not be stored.

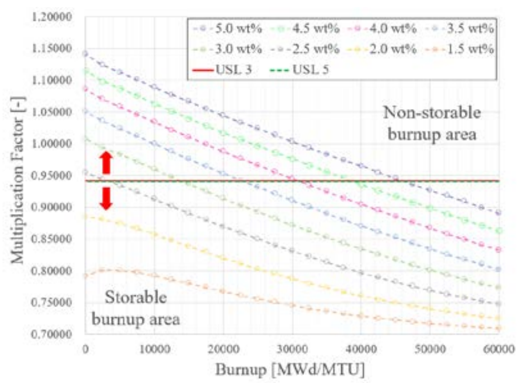

Fig. 5. Multiplication factor for SFP and USLs

Fig. 6 is the loading curve of the E1 rack of the SFP region II-A, which is determined based on the minimum burnup for each case with initial enrichment.

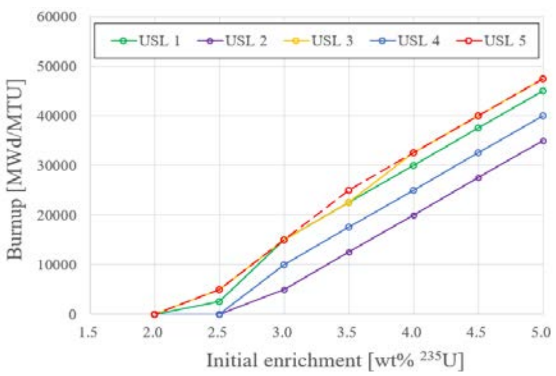

Fig. 6. Loading curve for SFP region II applying USL

4. Conclusion

In this study, criticality analysis was performed by applying the USL set by statistical method to actual SFP. The USLs of McCARD was set on the basis of the multiplication factor of 167 critical experimental problems presented in NUREG/CR-6361, and this criterion was applied to actual SFP of Yeonggwang unit 5 and 6. Through the criticality analysis, the minimum burnup satisfying the USL of McCARD was determined, a loading curve depending on each methodology for setting USL was produced. As a result, when the initial enrichment was low (>2.0wt%), all multiplication factors were satisfied in all burnup area. However, in the range where the enrichment exceeded 2.0 wt%, the fuel assemblies will be stored depending on USLs and burnup. This process confirms that the McCARD is applicable no only to nuclear design but also to criticality analysis of spent fuel storage system.

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