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Technical Note

Criticality effect according to axial burnup profiles in PWR burnup credit analysis

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ABSTRACT

The purpose of the critical evaluation of the spent fuel pool (SFP) is to verify that the maximum effective multiplication factor (K_{eff}) is less than the critical safety limit at 100% stored condition of the spent fuel with the maximum reactivity. At nuclear power plants, the storage standard of spent fuel, ie, the loading curve, is established to prevent criticality from being generated in SFP. Here, the loading curve refers to a graph showing the minimum discharged burnup versus the initial enrichment of spent fuel. Recently, US NRC proposed the new critical safety assessment guideline (DSS-ISG-2010-01, Revision 0) of PWR SFPs and most of utilities in US is following it. Of course, the licensed criterion of the maximum effective multiplication factor of SFP remains unchanged and it should be less than 0.95 from the 95% probability and the 95% confidence level. However, the new guideline is including the new evaluation methodologies like the application of the axial burnup profile, the validation of depletion and criticality code, and trend analysis. Among the new evaluation methodologies, the most important factor that affects K_{eff} is the axial burnup profile of spent fuel. US NRC recommends to consider the axial burnup profiles presented in NUREG-6801 in criticality analysis. In this paper, criticality effect was evaluated considering three profiles, respectively: i) Axial burnup profiles presented in NUREG-6801. ii) Representative PWR axial burnup profile. iii) Uniform axial burnup profile. As the result, the case applying the axial burnup profiles presented in NUREG-6801 showed the highest K_{eff} among three cases. Therefore, we need to introduce a new methodology because it can be issued if the axial burnup profiles presented in NUREG/CR-6801 are applied to the domestic nuclear power plants without any other consideration.

1. Introduction

Korea Hydro & Nuclear Power Co., Ltd (KHNP) is the electric power company which generates approximately 26.8% of the total electric power generated in South Korea. KHNP operates 24 nuclear power plants and lots of spent fuel assemblies are stored in spent fuel pool (SFP). SFP is evaluated in accordance with various standards like criticality, seismic performance, thermal-hydraulic and radiation shielding evaluation. Especially, criticality analysis is very important and performed to show that the spent fuel is under the sub-criticality condition to verify SFP safety. However, US NRC proposed new critical safety assessment guideline (DSS-ISG-2010-01, Revision 0) of PWR SFPs recently [1]. It recommends to consider the axial burnup profiles of spent fuel assembly presented in NUREG-6801 in criticality analysis [2]. Fuel assembly loaded into the reactor will burn with a slightly cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the center regions than in the upper and lower ends. Consequently, at high burnup, the more reactive fuel around the ends of the fuel assembly (less than the average burnup) occurs in regions near the ends of the fuel where the neutron important occurs due to neutron leakage. This phenomena increases the reactivity in comparison that obtained with uniform axial burnup profile. In this paper, the criticality effect according to the axial burnup profiles in PWR burnup credit analysis was evaluated considering three profiles, respectively:

i) Various axial burnup profile presented in NUREG-6801
ii) Representative PWR axial burnup profile
iii) Uniform axial burnup profile a reactor core design code

For reference, the axial burnup profiles presented in NUREG-

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6801 are based on the 3-D deletion calculations reflecting the burnup history of spent fuel assemblies discharged from a lot of US PWR plants [2]. Representative PWR axial burnup profile is a typical shape derived by a reactor core design code and most of PWR fuel assemblies have similar axial burnup shapes like Fig. 5, even though several fuel assemblies deviate from the typical shape [10].

2. Methodology

2.1. Computer codes

Criticality analysis is performed to show that the spent fuel is under the sub-criticality condition. Namely, the licensed criterion of the maximum effective multiplication factor (k_{eff}) in the SFP should be lower than 0.95 from the 95% probability and the 95% confidence level without credit for soluble boron for the normal condition. The objective of this analysis is to evaluate criticality effect according to the axial burnup profiles of spent fuel assembly in SFP. SCALE (**S**tandardized **C**omputer **A**nalyses for **L**icensing **E**valuation) ver. 6.1.3 is used for the criticality safety analysis. It was developed at ORNL (Oak Ridge National Laboratory) in US and widely being used around the world. Especially, STARBUCS is an analysis sequence in SCALE for automating criticality safety and burnup loading curve analyses of spent fuel systems employing burnup credit. It automatically performs all necessary calculations to determine spent fuel compositions and the k_{eff} of the spent fuel configuration. In addition, for burnup loading curve analyses, STARBUCS performs iterative calculations to search for initial fuel enrichments that result in an upper subcritical limit. STARBUCS allows the user to simulate axial burnup profiles of the spent fuel assembly, select the actinides and fission products that are to be needed in the criticality analysis, and apply ICF (isotopic correction factors) to the calculated spent fuel nuclide inventory to account for calculation bias and uncertainties. Depletion calculations were performed for each of fuel assemblies and ORIGEN code was used. They assume the average soluble boron concentration of 1000 ppm, the coolant temperature of 627 °C (900 K), and the specific power of 37 MW/MTU during the core operation [3,4]. It is conservative to assume a high-concentration boron and high-temperature coolant during the depletion process because the lower the burnup of the spent fuel, the higher the reactivity.

2.2. Modeling approach and assumption

Region II storage racks in SFP are made of stainless steel boxes with a neutron absorber plate that is attached by stainless steel sheathing. There is only one neutron absorber plate between fuel assemblies. Since region II racks are not symmetric around a single rack, a 2 × 2 model is required. The calculation models consist of a group of four identical racks surrounded by periodic boundary conditions rather than reflective boundary conditions. So, the geometric models for spent fuel storage rack were constructed for the criticality analysis like Fig. 1 and it has an infinite array. It is also modeled for a SFP of KHNP plants. This paper considers V5H fuel

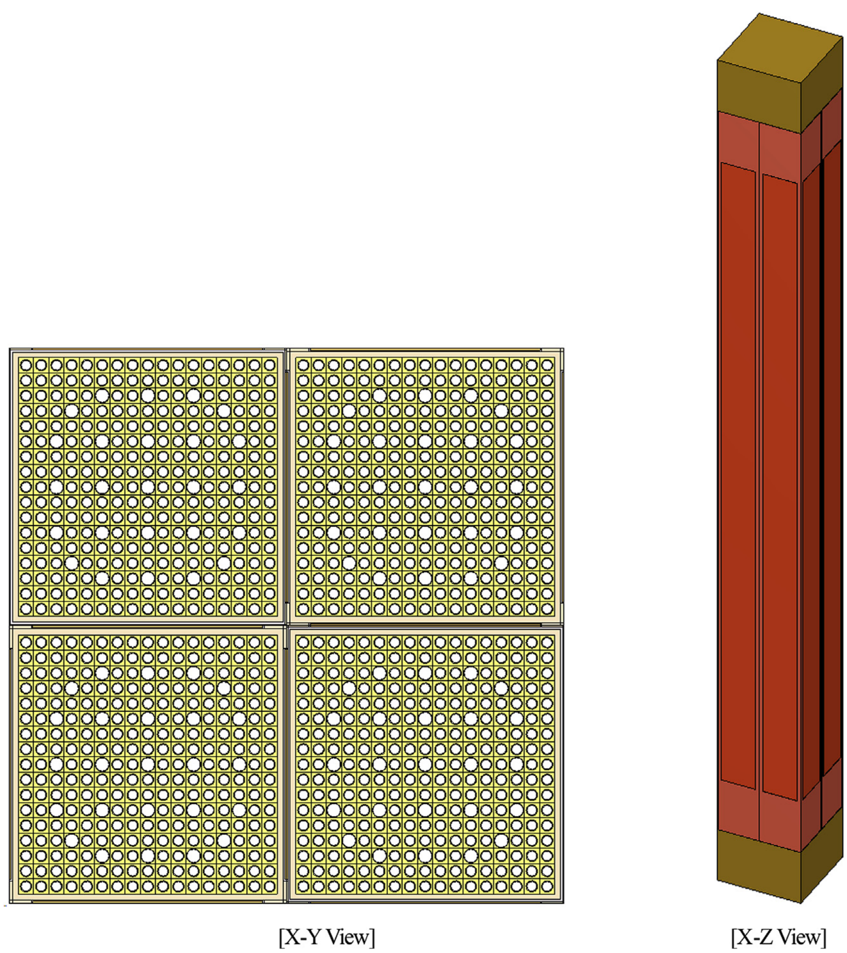

Fig. 1. Spent fuel storage rack with 4×4 array.

Table 1
Specifications of spent fuel storage rack.

Parameter	Value
Rack ID, mm	223.5
Box Wall Thickness, mm	1.9
Rack Pitch, mm	229.4
Rack Length, mm	3658
Moderator Temperature, Deg. C	4
Metamic Thickness, mm	2.7
Metamic Width, mm	190.5
Metamic Length, mm	3658
Metamic Pocket Thickness, mm	3.04
Metamic B_4C Weight Percent	30.5

Table 2
Specifications of V5H fuel assembly.

Parameter	Value
Stack Density, g/cm^3	10.28
Fuel Rod Pitch, mm	12.6
Number of Fuel Rods	264
Number of Guide Tubes	24
Number of Instrument Tubes	1
Fuel Temperature, Deg. C	680
Moderator Temperature, Deg C	323.2
Power Density, MW/MTU	37.074
Enrichment, w/o	2.0~5.0

assembly with 17 × 17 array that is the most conservative nuclear fuel type in criticality analysis. For reference, the reactivity of V5H fuel assembly is promoted by the increased moderator because the thickness of the guide rod of V5H fuel assembly is relatively thin [9]. Tables 1 and 2 are specifications of spent fuel storage rack and V5H fuel assembly. To assure that the actual k_{eff} will always be less than the calculated k_{eff}, the conservative design criteria and assumptions were employed like below;

1) Moderator is assumed to be un-borated water and 4°C that results in the highest k_{eff}.
2) No soluble poison or control rods are assumed to be present for normal operations.
3) Spacer grids of a fuel assembly are replaced by water.
4) No Integral Fuel Burnable Absorber (IFBA) rods are assumed to be present in the fuel rods.
5) An infinite radial array of fuel assemblies are assumed for spent fuel storage rack.
6) Depletion calculations assume the average soluble boron concentration of 1000 ppm and the coolant temperature of 627 °C (900 K) in-core operation.

2.3. Determination of upper safety limit (USL)

The ability of a calculation methodology to accurately predict the sub-criticality of a system must be well understood. The understanding of a calculation bias and uncertainty in predicting subcritical systems can be obtained through the validation process. Validation process is to verify the difference between the calculated and experimental results. This difference, called bias and uncertainty, is considered in combination with subcritical margin to establish an upper safety limit (USL). Sub-criticality is assured if calculated k_{eff} is below the USL and is within the area of applicability for the validation. Table 3 is summary of the criticality safety analyses for region II racks with 17 × 17 Assembly and criticality code validation is based on the NUREG/CR-6698 [5,6]. All possible

Table 3
Summary of the criticality safety analyses for Region II, 17×17 assembly.

Parameter	Value
Biases	
Criticality Calculational Bias	0.0029
Uncertainties	
Bias Uncertainty	0.00488
Criticality Calculaional Uncertainty(2σ)	0.00498
Depletion Uncertainty	0.00640
Discharged Burnup Uncertainty	0.00920
Manufacturing Tolerances	0.00688
Fuel Eccentricity	(negative)
Total Bias	0.0029
Statistical Combination of Uncertainties	0.0149
Upper Safety Limit (USL) = 0.95−0.0029 − 0.0149	0.9322

modifications of the reference configuration and all calculation biases are identified and individually evaluated, and their total contribution (ΔK_{Total}) is added to the USL. Normally, this evaluation is quantified by means of the multiplication factor difference between the modified configuration and the reference one. The calculation deviations and model approximations are treated as biases (ΔK_B), being directly added to USL. The calculation uncertainties, fabrication tolerances and off-nominal operating conditions are treated as uncertainties (ΔK_U) and they are statistically combined before being added to USL, according to the following equation;

$$\text{Maximum } K_{eff} = USL + \Delta K_{Total} = USL + \sum \Delta K_B + \sqrt{\Sigma(\Delta K_U)^2}$$

Where,

Maximum K_{eff}: Maximum Effective Multiplication Factor(0.95)
USL: Upper Safety Limit
ΔK_B: Criticality Calculation Biases
ΔK_U: Uncertainties including Manufacturing Tolerance, Burnup, and Depletion

The uncertainties could be treated assuming “worst case” conditions in the reference configuration. At least, the following uncertainties should be considered:

- Tolerance in boron loading into neutron absorber
- Tolerance in neutron absorber thickness and width
- Tolerance in storage rack inside diameter
- Tolerance in storage rack pitch
- Tolerance in storage rack thickness
- Tolerance in fuel enrichment
- Tolerance in fuel density
- Tolerance in fuel clad diameter
- Tolerance in fuel rod pitch
- Fuel assembly eccentric positioning in the rack

As the result of code validation, Total bias ($\Sigma\Delta K_B$) is 0.0529 and statistical combination of uncertainties including manufacturing tolerances is 0.0149. Therefore, USL is 0.9322 and it is in Table 3 [8].

2.4. Analysis method

Burnup credit is defined as the methodology of taking credit for the burnup of nuclear fuel in the performance of criticality safety analysis. The reactivity is generally reduced along with fuel burnup because of the change of concentration of actinide and the production of fission products absorbing neutrons. The change in the concentration of these isotopes that result in the reactivity reduction, is dependent upon the axial burnup profile of spent fuel assembly. When assuming an axially uniform profile of isotopes, the most reactive region of a fuel assembly is at the middle area during the depletion in a reactor because the further away from the center, the greater the neutron leakage. But the most reactive region of spent fuel is the end area of fuel assembly due to the low burnup at both ends of the fuel assembly. End effect is defined as the difference between K_{eff} considering the axial burnup profiles and K_{eff} considering the uniform axial burnup profile of fuel assembly [2];

$$End\ Effect(\Delta K) = K_{eff}^{with\ axial\ burnup} - K_{eff}^{uniform\ axial\ burnup}$$

This paper evaluates the criticality effect according to the axial burnup profiles of spent fuel assembly. Fig. 4 and Table 4 show the axial burnup profile presented in NUREG-6801 that US NRC recommends and Fig. 5 shows the representative PWR axial burnup profile [2,7,10]. The axial burnup profiles presented in NUREG-6801 were selected considering the most conservative case from 3169 PWR axial profiles of 1700 different assemblies for criticality analysis [2]. Representative PWR axial burnup profile is the typical shape derived by a reactor core design code and most of PWR fuel assemblies have similar axial burnup shapes even though several fuel assemblies deviate from the typical shape [10]. This paper evaluates the criticality effect according to three profiles including uniform axial burnup profile, respectively. The analysis calculations consider the storage racks to be fully stored with the most conservative fuel assemblies at the moderator temperature corresponding to the highest reactivity. The reactivity calculations must ensure a 95% confidence level with a 95% probability. The actinides and fission products listed in Table 5 are candidates for inclusion in criticality safety analyses of spent fuel pool. For reference, Xe-135 concentration in the spent fuel is conservatively set to zero [6].

3. Result

The objective of this analysis is to evaluate criticality effect according to the axial burnup profiles of spent fuel assembly in SFP. Spent WH17 × 17 was considered and modeled for a SFP of KHNP plants. First, K_{eff} of fuel assembly with the same enrichment was calculated to verify the criticality effect according to the axial burnup profiles. Secondly, loading curves of SFP for the storage of spent WH17 × 17 were produced according to the axial burnup profiles. For reference, the cases applying the representative PWR axial burnup profile and the uniform axial burnup profile is calculated considering a single profile regardless of burnup, and the case applying the axial burnup profiles presented in NUREG/CR-6801 is calculated considering profiles according to various burnups. Code inputs other than axial burnup profiles and burnup of the spent fuel are all the same.

Table 5
Nuclides for actinide and fission product burnup credit.

Ag-109	Cm-243	Gd-160	Nd-145	Rb-85	Sm-153	Te-130
Ag-110m	Cm-244	Ge-73	Nd-146	Rb-86	Sm-154	Te-132
Ag-111	Cm-245	Ge-76	Nd-147	Rb-87	Sn-115	U-234
Am-241	Cm-246	Ho-165	Nd-148	Rh-103	Sn-116	U-235
Am-242m	Cs-133	I-127	Nd-150	Rh-105	Sn-117	U-236
Am-243	Cs-134	I-129	Np-237	Ru-100	Sn-118	U-237
As-75	Cs-135	I-131	Np-238	Ru-101	Sn-119	U-238
Ba-134	Cs-136	I-135	Np-239	Ru-102	Sn-120	Xe-128
Ba-135	Cs-137	In-115	O-16	Ru-103	Sn-122	Xe-129
Ba-136	Dy-160	Kr-82	Pd-104	Ru-104	Sn-123	Xe-130
Ba-137	Dy-161	Kr-83	Pd-105	Ru-105	Sn-124	Xe-131
Ba-138	Dy-162	Kr-84	Pd-106	Ru-106	Sn-125	Xe-132
Ba-140	Dy-163	Kr-85	Pd-107	Ru-99	Sn-126	Xe-133
Br-81	Dy-164	Kr-86	Pd-108	Sb-121	Sr-86	Xe-134
Cd-110	Er-166	La-138	Pd-110	Sb-123	Sr-88	Xe-136
Cd-111	Eu-151	La-139	Pm-147	Sb-124	Sr-89	Y-89
Cd-112	Eu-152	La-140	Pm-148	Sb-125	Sr-90	Y-90
Cd-113	Eu-153	Mo-100	Pm-148m	Se-76	Tb-159	Y-91
Cd-114	Eu-154	Mo-95	Pm-149	Se-77	Tb-160	Zr-91
Cd-115 m	Eu-155	Mo-96	Pm-151	Se-80	Tc-99	Zr-93
Cd-116	Eu-156	Mo-97	Pr-141	Se-82	Te-122	Zr-95
Ce-140	Gd-152	Mo-98	Pr-143	Sm-147	Te-124	Zr-96
Ce-141	Gd-154	Mo-99	Pu-238	Sm-148	Te-125	
Ce-142	Gd-155	Nb-95	Pu-239	Sm-149	Te-126	
Ce-143	Gd-156	Nd-142	Pu-240	Sm-150	Te-127m	
Ce-144	Gd-157	Nd-143	Pu-241	Sm-151	Te-128	
Cm-242	Gd-158	Nd-144	Pu-242	Sm-152	Te-129m	

3.1. Criticality effect according to axial burnup profiles

K_{eff} of fuel assembly with the same enrichment was calculated to verify the criticality effect according to the axial burnup profiles. As shown in Fig. 2, the case of the axial burnup profiles presented in NUREG/CR-6801 shows the highest K_{eff}. The case of the uniform

Table 4
Bounding axial burnup profiles [NURGE/CR-6801].

Burnup group	1	2	3	4	5	6	7	8	9	10	11	12
Axial height (%)	Burnup ranges (GWd/MTU)											
	>46	42–46	38–42	34–38	30–34	26–30	22–26	18–22	14–18	10–14	6–10	<6
2.78	0.582	0.666	0.660	0.648	0.652	0.619	0.630	0.668	0.649	0.633	0.658	0.631
8.33	0.920	0.944	0.936	0.955	0.967	0.924	0.936	1.034	1.044	0.989	1.007	1.007
13.89	1.065	1.048	1.045	1.070	1.074	1.056	1.066	1.150	1.208	1.019	1.091	1.135
19.44	1.105	1.081	1.080	1.104	1.103	1.097	1.103	1.094	1.215	0.857	1.070	1.133
25.00	1.113	1.089	1.091	1.112	1.108	1.103	1.108	1.053	1.214	0.776	1.022	1.098
30.56	1.110	1.090	1.093	1.112	1.106	1.101	1.109	1.048	1.208	0.754	0.989	1.069
36.11	1.105	1.086	1.092	1.108	1.102	1.103	1.112	1.064	1.197	0.785	0.978	1.053
41.69	1.100	1.085	1.090	1.105	1.097	1.112	1.119	1.095	1.189	1.013	0.989	1.047
47.22	1.095	1.084	1.089	1.102	1.094	1.125	1.126	1.121	1.188	1.185	1.031	1.050
57.80	1.091	1.084	1.088	1.099	1.094	1.136	1.132	1.135	1.192	1.253	1.082	1.060
58.33	1.088	1.085	1.088	1.097	1.095	1.143	1.135	1.140	1.195	1.278	1.110	1.070
63.89	1.084	1.086	1.086	1.095	1.096	1.143	1.135	1.138	1.190	1.283	1.121	1.077
69.44	1.080	1.086	1.084	1.091	1.095	1.136	1.129	1.130	1.156	1.276	1.124	1.079
75.00	1.072	1.083	1.077	1.081	1.086	1.115	1.109	1.106	1.022	1.251	1.120	1.073
80.56	1.050	1.069	1.057	1.056	1.059	1.047	1.041	1.049	0.756	1.193	1.101	1.052
86.11	0.992	1.010	0.996	0.974	0.971	0.882	0.871	0.933	0.614	1.075	1.045	0.996
91.67	0.833	0.811	0.823	0.743	0.738	0.701	0.689	0.669	0.481	0.863	0.894	0.845
97.22	0.515	0.512	0.525	0.447	0.462	0.456	0.448	0.373	0.284	0.515	0.569	0.525

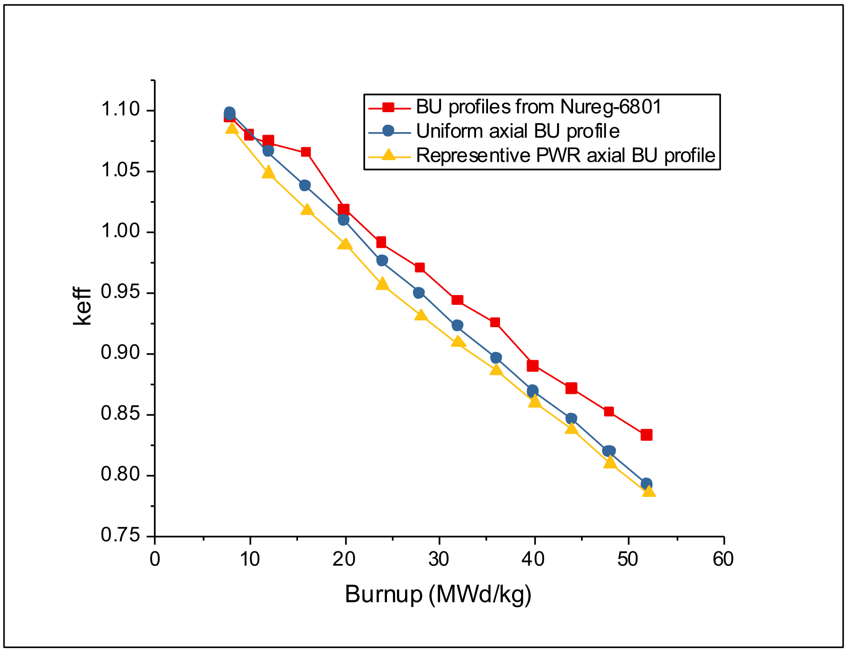

Fig. 2. K_{eff} according to axial burnup profiles (Enrichment = 4.5 w/o, Un-borated Water).

axial burnup profile shows higher K_{eff} than the case of representative PWR axial burnup profile, but shows lower K_{eff} than the case of the profile presented in NUREG/CR-6801. It means that the representative PWR axial burnup profile is not conservative comparing the uniform axial burnup profile and the profile presented in NUREG/CR-6801 in criticality analysis. So, it is necessary to identify which profile is reasonable, and to consider the additional bias or uncertainty for the end effect if the representative PWR axial burnup profile is applied in criticality analysis. This paper produced the SFP loading curves according to the axial burnup profiles. It is in Section 3.2.

3.2. SFP loading curve according to axial burnup profiles

The criticality analysis for the storage of spent WH17 × 17 (with an initial U-235 enrichment range of 2.0–5.0 wt%) has been performed according to the axial burnup profiles of spent fuel assemblies. The calculations were performed for a region II rack of KHNP's SFP. The region II rack calculations demonstrate that the maximum K_{eff} is less than 0.95 for the normal condition with no credit for soluble boron. Fig. 3 shows the initial enrichment vs. burnup combinations within the acceptable domain according to axial burnup profiles. As the result, the case applying the axial

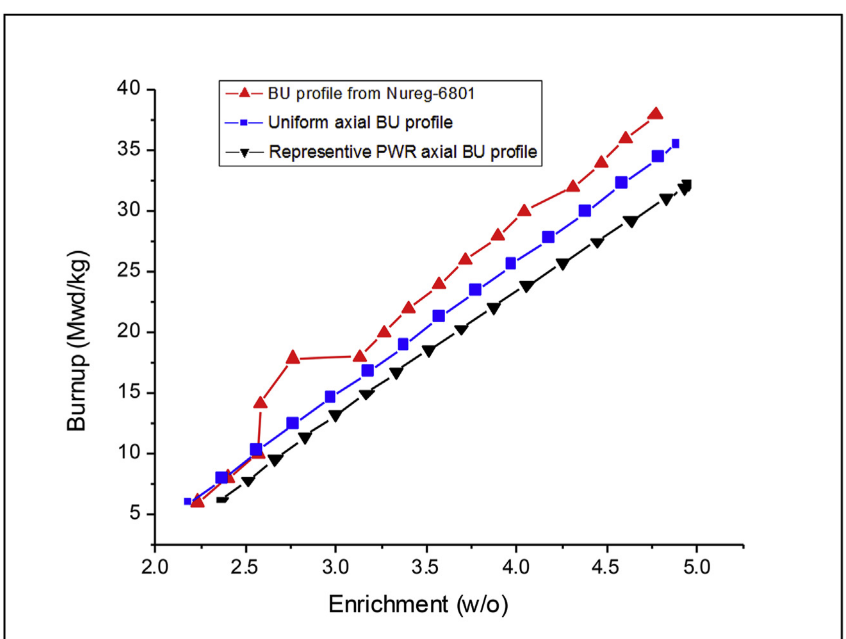

Fig. 3. Minimum required fuel assembly burnup as a function of initial enrichment in Region II.

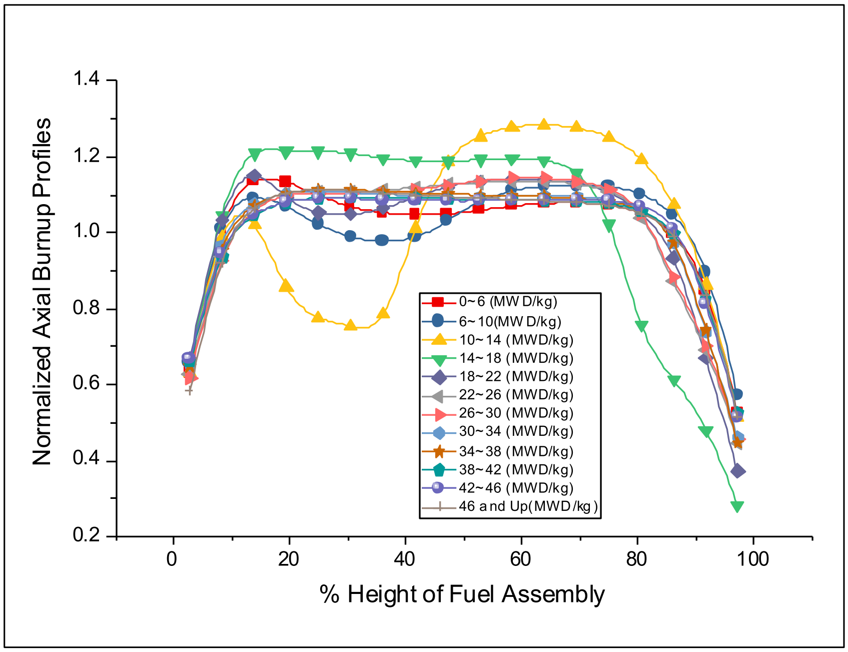

Fig. 4. Bounding axial burnup profiles [NURGE/CR-6801].

burnup profiles presented in NUREG/CR-6801 shows the highest loading curve. Especially, the loading curve obtained from the axial burnup profiles in NUREG-6801 isn't continuous and is relatively higher at the enrichment between 2.5 w/o and 3.2 w/o. That is why the loading curve is based on the values selected conservatively by the bounding concept, and the burnup at the top of the fuel assembly corresponding to the axial burnup profile between 14 and 18 MWD/MTU is the lowest among the axial burnup profiles presented in NUREG-6801. The case applying the representative PWR axial burnup profile shows a lower loading curve than the cases considering the uniform axial burnup profile and the axial burnup profile presented in NUREG/CR-6801. In other words, the case applying the representative PWR axial burnup profile can lead to non-conservative results comparing the others. US utilities are applying the axial burnup profile presented in NUREG/CR-6801 now, but it is not any issue because they are applying the boron credit at normal condition in criticality analysis. Therefore, we also need to introduce the boron credit or reproduce the axial burnup profiles based on the spent fuel assemblies discharged from the domestic nuclear power plants to mitigate the conservatism of the axial burnup profile presented in NUREG/CR-6801. Fig. 6 shows K_{eff} according to axial burnup profiles presented in NUREG-6801 considering boron concentrations. As shown in Fig. 6, K_{eff} decreases as the boron concentration increases. Actually, the boron

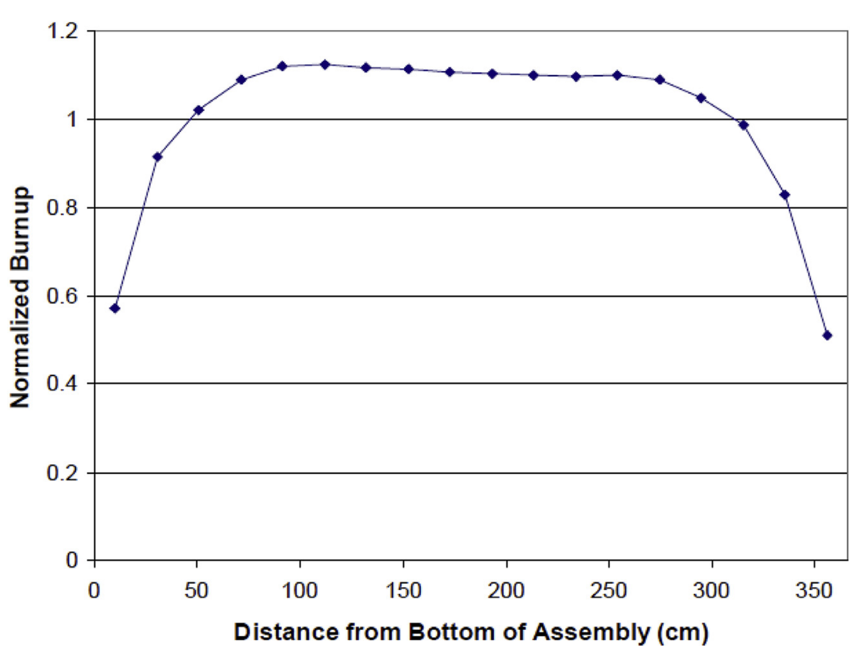

Fig. 5. Representative normalized PWR axial burnup profile.

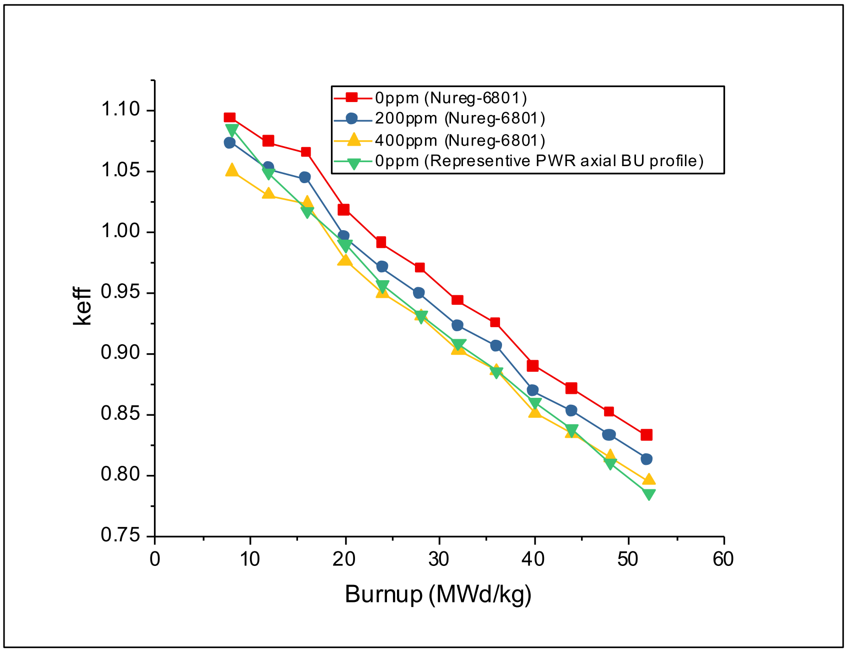

Fig. 6. K_{eff} according to axial burnup profiles of NUREG-6801 considering boron concentrations.

concentration of KHNP's spent fuel pool is over 2,000 ppm.

4. Conclusion

Criticality analysis is very important and performed to show that the spent fuel is under the sub-criticality condition to verify SFP safety. However, US NRC issued new critical safety assessment guideline of PWR SFPs recently. It recommends to consider the axial burnup profiles presented in NUREG-6801 in criticality analysis. Basically, K_{eff} is dependent upon the axial burnup profile of spent fuel because of the change in concentration of actinide and the production of fission products absorbing neutrons with fuel burnup. Therefore, criticality effect was evaluated considering the various axial burnup profiles. In this paper, three profiles were considered like below;

i) Various axial burnup profiles presented in NUREG-6801
ii) Representative PWR axial burnup profile
iii) Uniform axial burnup profile

As the result, the case applying the representative PWR axial burnup profile shows a lower K_{eff} than the cases of the uniform axial burnup profile and the axial burnup profiles presented in NUREG/CR-6801. In other words, the case applying the representative PWR axial burnup profile can lead to non-conservative results comparing the others. Therefore, it is necessary to identify which profile is reasonable, and to consider the additional bias or uncertainty to apply the representative PWR axial burnup profile in criticality analysis. And it can be issued if the axial burnup profiles presented in NUREG/CR-6801 are applied to the domestic nuclear power plants without any other consideration. So, we need to introduce the boron credit methodology or reproduce the axial burnup profiles based on the spent fuel assemblies discharged from the domestic nuclear power plants to mitigate the conservatism of the axial burnup profile presented in NUREG/CR-6801. Basically, K_{eff} decreases as the boron concentration increases and the boron concentration of KHNP's spent fuel pool is actually over 2,000 ppm.

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