Analysis on the End Effects for Spent Fuels having Axial Blankets for Spent Fuel Storage Pool

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1. Introduction

It is very important that the criticality safety analyses for spent fuel storage pool is conducted for checking subcriticality under normal and accident conditions. Many studies have been conducted to apply burnup credit to reduce conservatism in criticality safety analysis for dense storage racks of PWR spent fuel storage pool. Due to characteristics of long fuel assemblies, there are large burnup variations in the axial direction. Because it is difficult to take into account the axial burnup profile for all spent fuel assemblies, the US NRC suggested the bounding axial burnup profiles giving the conservative results depending on the divided burnup ranges [1]. However, it is necessary to select a bounding axial burnup profile in accordance with the domestic situation. In our previous study, the bounding axial burnup profiles were selected through core follow calculation of KORI Unit 1, 2, 3, and HANBIT Unit 3 [2]. Some fuel assemblies loaded into these cores have axial blankets, which were assumed to have the same enrichment as the normal fuel in the axial direction in our previous criticality calculations for determining the bounding axial burnup profiles. However, this treatment can lead to too conservative estimation of the end effect. The objective of this work is to quantify the end effects for the fuel assemblies having axial blankets.

2. Generation of Axial Burnup Profiles

2.1. Computer Code System

The two-step core design and analysis code system STREAM/RAST-K which has been developed by UNIST was used for core follow calculations. The STREAM code is an advanced lattice code, which solves the multi-group transport equation with MOC (Method of Characteristics) for two-dimensional assembly and reflector models. This code generates homogenized fuel assembly cross sections and form functions as the function of many parameters such as burnup, boron concentration, and temperatures in the STN file. The STORA program processes the STN file to generate the group constants which are used in the core nodal diffusion calculation with RAST-K. The STREAM code is characterized by its PSM (pin-based point-wise slowing down) method and equivalence theory for resonance self-shielding effect and by CRAM (Chebyshev Rational Approximation) for depletion [3]. The RAST-K code is an advanced nodal diffusion code that uses the multi-group CMFD (Coarse Mesh Finite Difference) method coupled with 3D multi-group unified nodal method [4].

2.2. Characteristics of Evaluation Reactor Core

The core selected in this study is KORI Unit 3, which is loaded with many fuel assemblies with axial blankets. KORI Unit 3 rates 2,775MWt until the 19th cycle and 2,900MWt from the 20th cycle to the 25th cycle. The core follow calculations of KORI Unit 3 were performed from 1st to 25th cycles. The core consist of 157 fuel assemblies and each fuel assembly has 17 x 17 fuel array lattice structure comprised of 264 fuel rods, 24 guide tubes for control rods, and 1 guide tube for in-core instrumentation. From 1st to 8th cycles, all the fuel assemblies have no axial blanket. However, from 9th to 25th cycles, fuel assemblies have 15.24cm (6 inches) thick axial blankets at the top and bottom of the fuel rod, in which uranium enrichment is 0.711wt% (natural uranium). The height of all fuel assemblies is 365.76cm. Table I shows the detailed characteristics of the fuel assembly types for each cycle.

Table I: Characteristics of fuel assemblies

Type	Cycle	BP
SFA	1	PYREX
OFA	2 - 5	WABA
KOFA	6 - 8	Gd_2O_3
V5H	9 - 15	WABA
RFA	16 - 19	Gd_2O_3
ACE7	20 - 25	Gd_2O_3

In Table I, it is noted that the core has different fuel assembly types and different types of burnable poison (BP) rods depending on different cycles. The SFA of which PYREX burnable poison has 12.5wt% B_2O_3 content borosilicate is used only for the 1st cycle. The OFA and V5H of which WABA burnable poison has 13.5wt% B_4C content in Al_2O_3, are used for 2nd ~ 5th cycles, and 9th ~ 15th cycles, respectively. The KOFA, RFA, and ACE7 of which enriched gadolinium burnable poison have different Gd_2O_3 contents with several uranium enrichments are used for 6th ~ 8th

cycles, 16th ~ 19th cycles, and 20th ~ 25th cycles, respectively.

2.3. Generation of Axial Burnup Profiles

The core follow calculations were performed with 24 axial nodes that are not uniform. Because the active fuel lengths of the fuel assemblies were divided by considering the locations of the in-core detectors, the grids, and the length of an axial cutback or axial blanket. The axial burnup profiles for the 24 axial nodes are generated through the core follow calculations for 1,459 fuel assemblies discharged from 1st ~ 25th cycles. Of the 1,459 fuel assemblies, 903 fuel assemblies have axial blankets. To describe the effect of the axial burnup profiles on the criticality calculation, the axial burnup profiles were renormalized based on the average burnup of each spent fuel assembly by uniformly dividing the active fuel length except for the top and bottom ends with 24 axial nodes. The original axial burnup profiles for the 24 axial nodes used in core follow calculations are renormalized for the new axial node division in which top and bottom end nodes occupy 2.8% of the total active length and each of the other 22 nodes occupies 4.29%. The fine axial node division of the last two nodes is to more accurately represent the end effect near the top and bottom ends. For example, the axial burnup profiles for all the assemblies discharged from 8th cycle and 13th cycle are shown in Fig. 1, and Fig. 2, respectively. As shown in these figures, spent fuel assemblies discharged from 13th cycle having axial blankets have much more steep gradient in axial burnup profiles near the top and bottom ends than the assemblies from 8th cycle having no axial blankets.

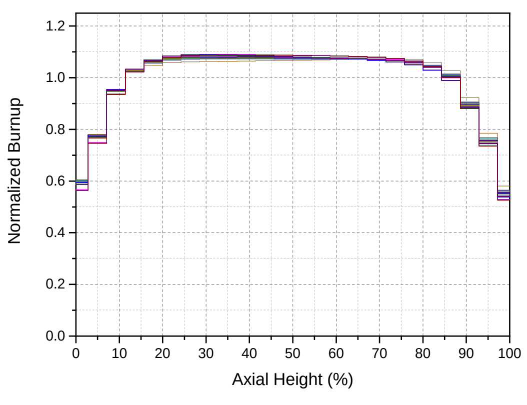

Fig. 1. Comparison of the axial burnup profiles for the fuel assemblies (having no axial blankets) discharged from 8th cycle

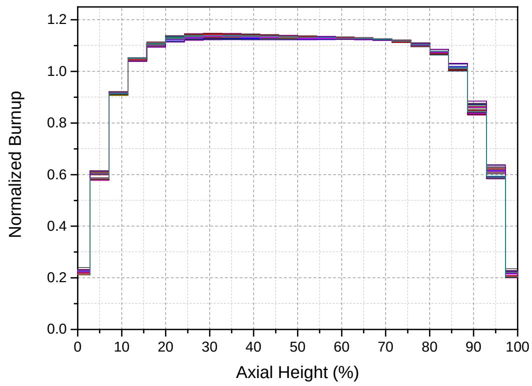

Fig. 2. Comparison of the axial burnup profiles for the fuel assemblies (having axial blankets) discharged from 13th cycle

3. Criticality Calculation

3.1. Computer Code System

The STARBUCS (Standardized Analysis of Reactivity for Burnup Credit using SCALE) sequence of SCALE6.1 was used to perform the criticality analysis with burnup credit for the spent fuel storage pool. The STARBUCS sequence automates the depletion calculations using the ORIGEN-ARP methodology to perform a series of cross-section preparation and depletion calculations to generate a comprehensive set of spent fuel isotopic inventories for each spatially-varying burnup region of an assembly. The spent fuel nuclide concentrations are subsequently input to CSAS5 and perform a criticality calculation of the system using the KENO V.a 3-D multi-group Monte Carlo criticality calculation [5].

3.2. Analysis of End Effect with Axial Blanket

In our previous study, the axial burnup profiles were divided into 12 burnup groups [2]. The first burnup group represents burnups higher than 50MWd/kg. The last burnup group represents burnups lower than 10MWd/kg. The intermediate burnup groups from the 2nd group to the 11th group were uniformly divided with 4MWd/kg intervals of burnup. The burnup representing each group is just the average of the discharge burnups in the group. For criticality analysis, the specific power for burnup calculation in STARBUCS sequence for all the fuel assemblies was set to 37MW/MTU, which is the value of the typical PWRs. At present, we did not consider cooling times, which leads to higher k_{eff} but lower end effects. All the other conditions are the same as in the our previous study [2]. The initial uranium enrichment for all the fuel assemblies was set to 4.5 wt%, and natural uranium is used in the axial blankets. We considered the following actinide and fission product nuclides for burnup credit [6] : U-234, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Nd-143, Nd-145, Eu-151, Eu-153, Gd-155, U-236, Am-243, and Np-237.

The difference in the reactivity obtained with a non-uniform axial burnup profile and a uniform axial burnup is known as the "End Effect". If the sign of the end effect is positive, it means that the criticality calculation using the axial burnup profile is conservative. Fig. 3 shows the end effects estimated without considering the axial blanket in which 4.5 wt% enriched uranium is used in the axial blankets. As the burnup of the fuel assembly increases, the end effect overall increases as generally known in the literature [1]. In particular, it is noted in Fig. 3 that the end effect has a positive value at 38 MWd/kg burnup or more for the fuel assemblies having axial blankets, but the fuel assemblies with axial blanket show much larger end effects at 30 MWd/kg burnup or more than those of the fuel assemblies having no axial blankets. These larger end effects for the fuel assemblies having axial blankets are resulted from two aspects: 1) larger axial burnup gradients near the top and bottom ends and 2) consideration of enriched uranium (i.e., 4.5 wt%) instead of actual natural uranium in axial blankets during the criticality calculations. Fig. 4 shows the end effects estimated with considering natural uranium in the axial blankets. In particular, it is noted in Fig. 4 that many assemblies giving positive end effects without considering natural uranium in axial blanket shows negative end effects with considering natural uranium in the axial blankets. Table II summarizes the maximum and average end effects considering with and without axial blanket for each burnup group. Without considering the axial blanket, in the burnup groups higher than 38 MWd/kg, both the maximum and average end effects increase as burnup. The maximum positive end effects were ranged from 0.01% ~ 3.81% Δk depending on the burnup groups. Below 26 MWd/kg burnup, there are no axial burnup profiles giving positive end effects. On the other hand, the positive end effects estimated with considering axial blankets occurs from the fourth burnup group (i.e., 38 ~ 42 MWd/kg). The maximum positive end effects were ranged from 1.02% ~ 1.34% Δk depending on the burnup groups. The consideration of nature uranium axial blankets significantly reduces the maximum and average end effects.

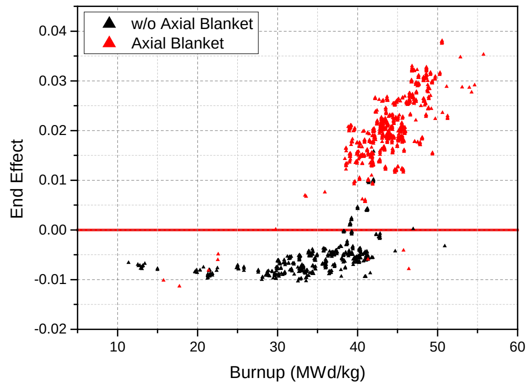

Fig. 3. End effects without considering the axial blanket

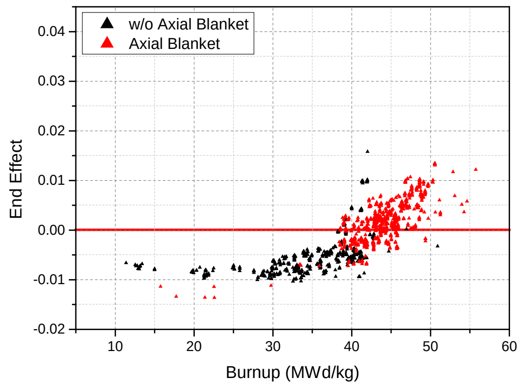

Fig. 4. End effects considering the axial blanket

Table II. Maximum and average end effect (Δk) for burnup groups

Burnup (MWd/kg) [Group]	Without axial blanket		With axial blanket	
	Maximum End Effect	Average End Effect	Maximum End Effect	Average End Effect
50~[1]	0.03808	0.02942	0.01345	0.00801
46~50[2]	0.03297	0.02659	0.01071	0.00583
42~46[3]	0.02688	0.01922	0.01579	0.00129
38~42[4]	0.02110	0.00709	0.01020	-0.00189
34~38[5]	0.00761	-0.00588	-0.00323	-0.00602
30~34[6]	0.00694	-0.00699	-0.00425	-0.00721
26~30[7]	0.00012	-0.00850	-0.00742	-0.00878
22~26[8]	-0.00488	-0.00746	-0.00706	-0.00820
18~22[9]	-0.00750	-0.00884	-0.00750	-0.00905
14~18[10]	-0.00775	-0.00808	-0.00775	-0.00832
10~14[11]	-0.00660	-0.00721	-0.00660	-0.00721

4. Conclusions

The axial burnup profiles of the spent fuels discharged from KORI Unit 3 were evaluated through the core follow calculations based on their NDRs and these axial burnup profiles were used to evaluate the end effect through the criticality calculations for spent fuel pool storage. 1,459 axial burnup profiles were classified depending on the presence or absence of the axial blanket in fuel assemblies. The analysis of the end effects with the evaluated axial burnup profiles showed that the maximum and average end effects increase as burnup. Also, it was found that the consideration of natural uranium in the axial blankets significantly reduces the end effect estimated with a single enriched uranium in the fuel rods having axial blankets. Therefore, based on the results of this study, it is strongly recommended to model the natural uranium the axial blankets, which significantly reduces the conservatism in the criticality safety analysis with a single enriched uranium in the fuel rods having axial blankets.

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