Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 23-24, 2019

Production of Axial Burnup Profiles of Spent Nuclear Fuels Discharged from OPR-1000 using STREAM/RAST-K Core Follow Calculations

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1. Introduction

The safe management of spent nuclear fuels of PWRs are a big issue in nuclear industry because the most spent fuel storage pools are expected to be saturated within 5~19 years. For example, the spent fuel storage pool of Kori units is expected to be saturated from 2024 in spite of the considerations of extended racks and inter-transportations between the different units. The criticality safety analysis of the spent fuel storage and transportation facilities are very important to show that they are kept under subcriticality for normal or accident situations. In particular, the application of burnup credit to the criticality safety analysis is required to reduce the excessive conservatism with fresh fuel in order to allow cost-effective and higher density storage of spent nuclear fuels. The axial burnup profiles are considered in the criticality safety analysis with burnup credit to accurately estimate the reactivity of spent fuel facilities. In particular, this effect of axial burnup profile on the criticality is known as the end effect. So, the detailed axial burnup profiles are needed to be evaluated in order to find the conservative (or bounding) one giving the largest reactivity.

The objective of this work is to evaluate the detailed axial burnup profiles of the spent nuclear fuels discharged from OPR1000 using the core follow calculations with STREAM/RAST-K code system which has been developed by UNIST and KHNP. Additionally, the results of the core follow calculations are compared with those of DeCART2D/MASTER.

2. Methods and Results

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 multi-group transport equation with MOC (Method of Characteristics) for two-dimensional assembly and reflector models, and generates homogenized fuel assembly cross sections and form functions as the function of many parameters such as burnup, boron concentration, and temperatures in 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 slowisng down) method and equivalence theory for resonance self-shielding effect and by the CRAM (Chlastebyshev Rational Approximation) for depletion. The RAST-K code is an advanced nodal diffusion code which uses the multi-group CMFD (Coarse Mesh Finite Difference) method coupled with 3D multi-group unified nodal method.

2.2 Characteristics of OPR1000 Cores

OPR1000 rates 2815 MWt (1000MWe). Its reactor core consists of 177 fuel assemblies and each fuel assembly has 16x16 fuel array lattice structure which is comprised of 236 fuel rods, 5 guide tubes for control rods, and 1 guide tube for in-core instrumentation. The cladding is Zircaloy-4, ZIRLO and the pellet of the fuels is UO_2. The fuel rods and guide tubes are supported by Inconel-718, ZIRLO and Zircaloy-4 grids. The followings summarizes the main design characteristics of the fuel assemblies used in the 1st to 13th cycles :

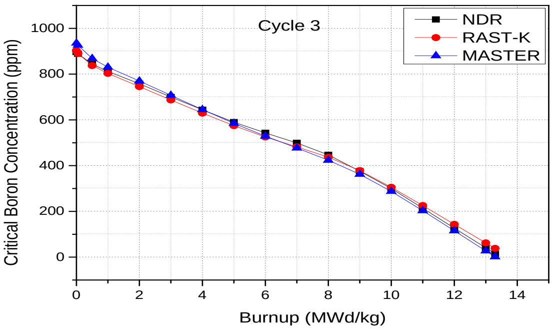
- 16x16 KSFA assemblies with Gd type BP of 4.0~8.0w/o Gd_2O_3 contents with natural uranium enrichment are used for 1th ~ 8th cycles.
- 16x16 GUARDIAN assemblies with Gd type BP of 4.0~8.0w/o Gd_2O_3 contents with natural uranium enrichment are used for 9th ~ 10th cycles.
- 16x16 PLUS7 assemblies with Gd type BP of 6.0~8.0w/o Gd_2O_3 contents with 2.0w/o uranium enrichment are used for 11th ~13th cycles.

However, there are some unclear points in NDRs (Nuclear Design Report) that for example there are no specifications of the PLUS7s pellet density, and so we assumed the pellet density of 10.313g/cm^3.

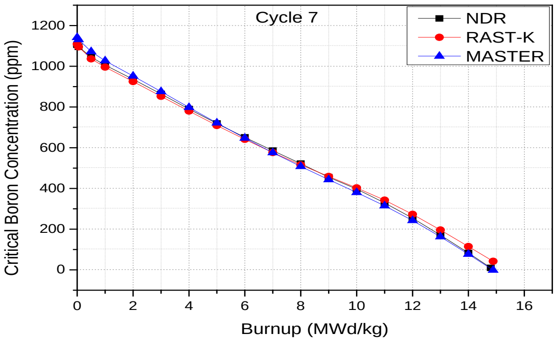
2.3 Results of the Core Follow Calculations

This section presents the results of the core follow calculations. Figs. 1. (a), (b), (c) and (d) compare the evolutions of the critical boron concentrations for the selected four cycles (i.e., 3rd, 7th, 12th, 13th), respectively. For every cycle, the core depletion calculation was performed with the cycle burnup specified in NDR. For 3rd cycle, the maximum difference in CBC between NDR and STREAM/RAST-K is 23 ppm while the one between NDR and DeCART2D/MASTER is 38 ppm. For 7th cycle, the maximum difference in CBC between

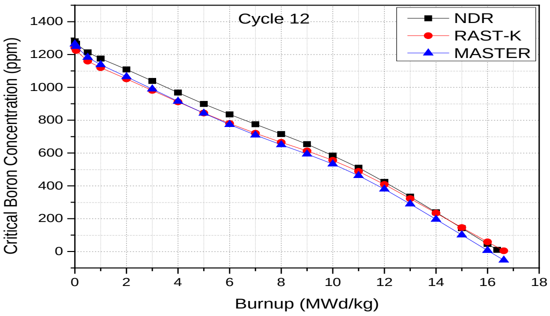
NDR and STREAM/RAST-K is 32 ppm while the one between NDR and DeCART2D/MASTER is 36 ppm. For 12th cycle, the maximum difference in CBC between NDR and STREAM/RAST-K is 57.7 ppm while the one between NDR and DeCART2D/MASTER is 30 ppm. For 13th cycle, the maximum difference in CBC between NDR and STREAM/RAST-K is 58 ppm while the one between NDR and DeCART2D/MASTER is 45 ppm. These results show that the both code systems give very good agreements in CBC over these cycles.



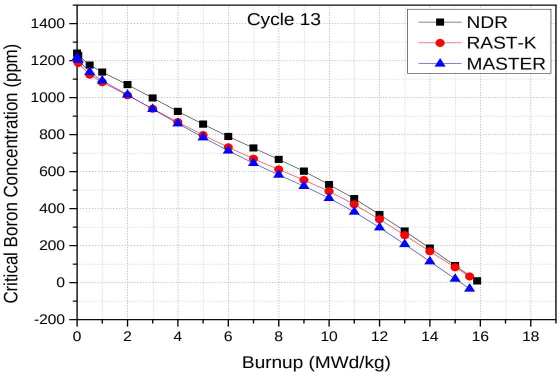
(a) 3rd cycle



(b) 7th cycle



(c) 12th cycle



(d) 13th cycle

Fig. 1 Comparison of the CBC evolutions

Fig. 2 compares CBC values estimated by DeCART2D/MASTER and STREAM/RAST-K at EOC for all the cycles. On the other hand, it should be noted that the CBC value at EOC for all the cycle except for 1th, 3th and 4th cycle is 10 ppm in NDR (15 ppm for 1th, 3th and 4th cycle). From this figure, it is noted that STERAM/RAST-K overestimates CBC values at EOC for most cycles in comparison with NDR and the maximum CBC of ~70 ppm occurs at 4th cycle while DeCART2D/MASTER gives the highest CBC of 35ppm at 4th cycle and the lowest CBC of -60ppm at 11th cycle.

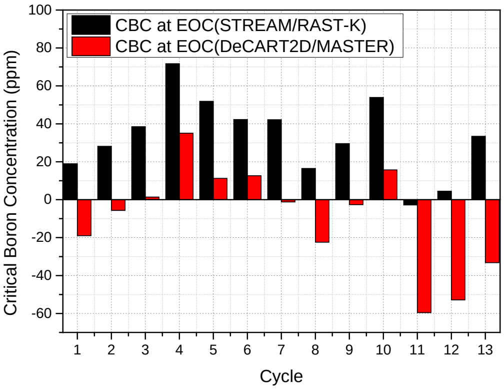


Fig. 2 Comparison of the differences in CBC between NDR and the present calculations

Fig. 3 represents the RMS (Root Mean Square) and the maximum errors of the assembly-wise powers between STREAM/RAST-K and NDR at BOC of all the cycles. The RMS errors are less than 2.5% for all the cycles and the maximum errors are less than 6.5% which occurs at 12th cycle.

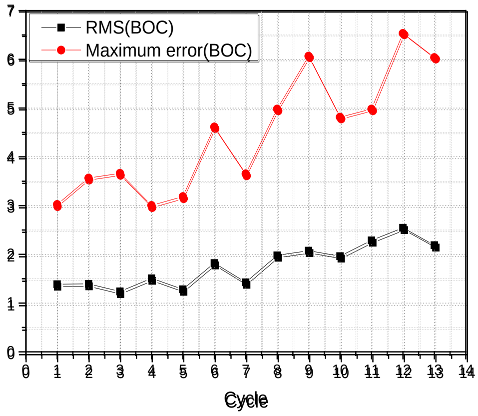


Fig. 3 The RMS and maximum errors between STREAM/RAST-K and NDR at BOC

2.4 Generation of Axial Burnup Profiles

The core follow calculations were performed with 24 axial nodes that are not uniform. The axial burnup profiles for the 24 axial nodes are generated from the core follow calculations for KSFA, GUARDIAN and PLUS7 fuel assemblies discharged from 1^{st} ~ 13^{th} cycles. The axial burnup profiles for the 24 axial nodes are renormalized for the other axial node division in which two end nodes occupy 2.8% of the total 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. The axial burnup profiles are finally renormalized with respect to the average discharge burnup of each fuel assembly. For example, the axial burnup profiles for some assemblies discharged from the first cycle are shown in Fig. 4. As shown in this figure, the axial burnups of the low regions are higher than those of the top regions due to the higher moderator densities of the lower regions and the axial burnups of the end regions are lower than the central regions due to the large neutron leakages in the end regions. These axial burnup profiles are considered to be typical profile for the fuel assemblies which do not use axial blanket fuels.

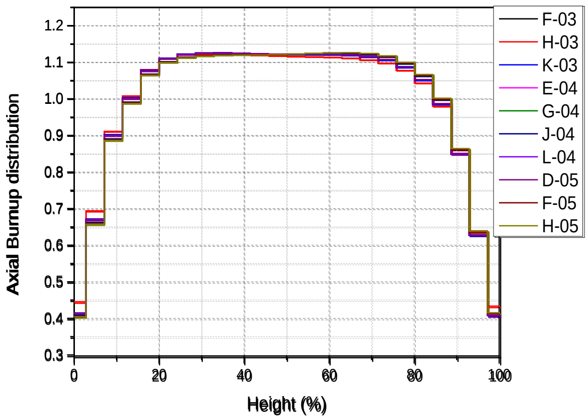


Fig. 4. Comparison of the axial burnup profiles for the fuel assemblies discharged from the 1st cycle

3. Conclusions

In this work, we performed the core follow calculations for 1st ~ 13th cycles of OPR1000 to generate the axial burnup profiles of the discharged fuel assemblies. The STREAM/RAST-K code system was used for the core follow calculations and the results of the calculations are compared with those given in NDRs and with those estimated with DeCART2D/MASTER. The results showed that the present core follow calculations with STREAM/RAST-K gives maximum 59 ppm difference in CBC in comparison with NDR and the RMS and maximum errors of assembly-wise power are less than 2.5% and 6.5% at BOC, respectively.

As the results of the core follow calculations, we produced the normalized axial burnup profiles for 24 axial nodes which are to be used in the criticality safety analysis with burnup credit.

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