Estimation of neutron flux distribution in reactor vessel and bio-shield of Kori unit 1 with core follow calculations

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1. Introduction

Korea's commercial nuclear reactor, Kori Unit 1, started commercial operation in April 1978 and received a permanent shutdown on June 12, 2015. After operating until June 2017, it has permanently stopped. While the decommissioning plan for Kori Unit 1 initially aimed for approval in June 2022 with a projected completion by 2032, the dismantling approval has been delayed to June 2024 [5].

Throughout the operational life of Kori Unit 1, reactor main components such as the reactor vessel internals, reactor pressure vessel (RPV), and bio-shield had been activated due to the fission-induced neutrons. In nuclear power plants operating for more than 30 years, neutron-induced activation of these major structure accounts for the overall radioactivity of the nuclear power plant.

The assessment of radioactivity and radioactive nuclide inventories based on an accurate computational analysis on the neutron flux to reactor components is essential to establish a decommissioning plan.

Most of the previous works on the activation analysis for Kori Unit 1 were based on the neutron fluxes at a specific time point of a representative cycle (e.g., BOC at the first cycle). However, the flux distribution actually changes as time in a cycle, which can reduce the accuracy of the activation calculation. In this work, it is suggested that the fuel rod-wise burnup change distribution through a cycle is used as the source distribution in Monte Carlo neutron transport calculations because the number of the produced neutrons from fission will be approximately proportional to the burnup change through a cycle.

In this work, the fuel rod-wise burnup change distribution for each cycle was evaluated with the core follow calculation using DECART2D/MASTER4.0 code system for Kori unit cycle 1 to cycle 20. A representative cycle was selected for neutron flux calculation based on the burnup distributions and the source terms for Monte Carlo transport calculation was generated using the burnup difference between the beginning of cycle (BOC) and the end of cycle (EOC) for each fuel rod. The MCNP modeling of Kori unit 1 was made based on Kori unit 1 specifications and previous research [2]. Additionally, the Automated Variance reduction Parameter Generation (ADVANTG) code [6] was used to apply variance reduction method to the MCNP6 [6] neutron transport calculation in order to reduce the computational time for neutron flux calculations. The neutron flux values were calculated based on the tally for RPV and bio-shield.
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2. Methods and Results

2.1 Calculation Methodology

To generate source term for neutron flux calculation, the core following calculations using DECART2D/MASTER4.0 code system developed by Korea Atomic Energy Research Institute (KAERI) were used to generate the fuel rod-wise burnup distributions. Depletion calculations were conducted from the first cycle to 20th cycle for Kori Unit 1. The fuel pin-wise source distribution for neutron transport calculation at a cycle were assumed to be proportional to the fuel rod-wise burnup change distribution.

The Monte Carlo transport MCNP6 code was used to calculate the neutron flux on the RPV and bio-shield with the continuous energy ENDF/B-VIII.0 cross section library.

We modeled the full core with maintaining fuel rod-wise heterogeneities because the fuel assemblies are asymmetrically loaded in the core for the representative cycle are asymmetric. To reduce the computational time, the variance reduction method with FW (Forward-Weighted)-CADIS (Consistent Adjoint Importance Sampling) [6] can be applied in MCNP6 calculations. The weight window file for FW-CADIS application in MCNP6 was generated using the ADVANTG code [6] weight window, one of the variance reduction method, was applied.

Fig.1 shows the code and computational process used for neutron flux calculations in this paper.

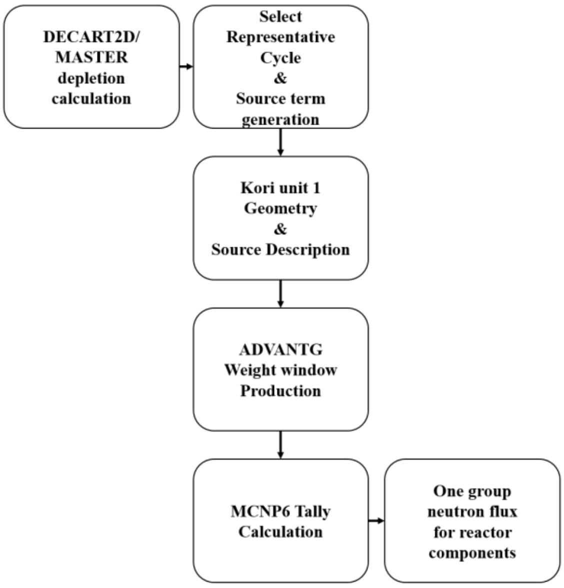

Fig.1 Flow diagram of calculation methodology

2.2 DECART2D/MASTER4.0 core follow calculation

Using DECART2D/MASTER4.0 code, the core follow calculations were conducted from cycle 1 to cycle 20 referring to the specification and operation history of Kori unit1 [5].

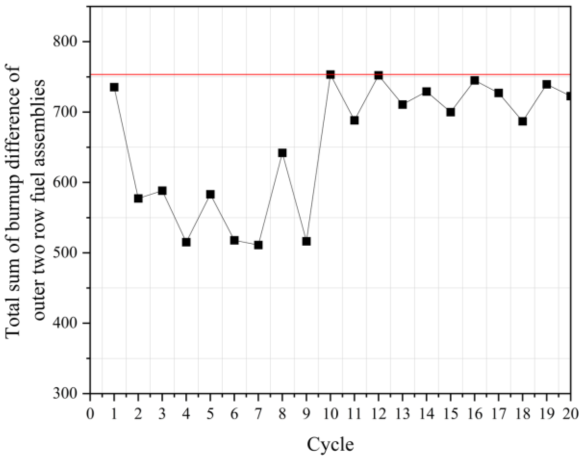

Fig.2 Burnup difference of outer two rows fuel assemblies, cycle 1 to cycle 20

From the PPI files, the output files of the MASTER4.0, the fuel rod-wise burnup distribution for each cycle was obtained. A representative cycle was selected based on the changes in the fuel rod-wise burnup through the cycle for the outer two rows of assemblies. Fig. 2 shows the burnup differences for the assemblies located at the outer two rows from the cycles 1 to 20. The representative cycle was selected as the cycle 10 having the highest burnup differences in the outer two ring rows of assemblies. The representative cycle was selected to simplify the activation calculation without considering all the cycles but with conservative estimation in activities. So, the neutron flux estimation was conducted only for this representative cycle. The fuel rod-wise source distribution was assigned to be proportional to the fuel rod-wise burnup change distribution through the representative cycle. Fig. 3 shows the fuel rod-wise source distribution.

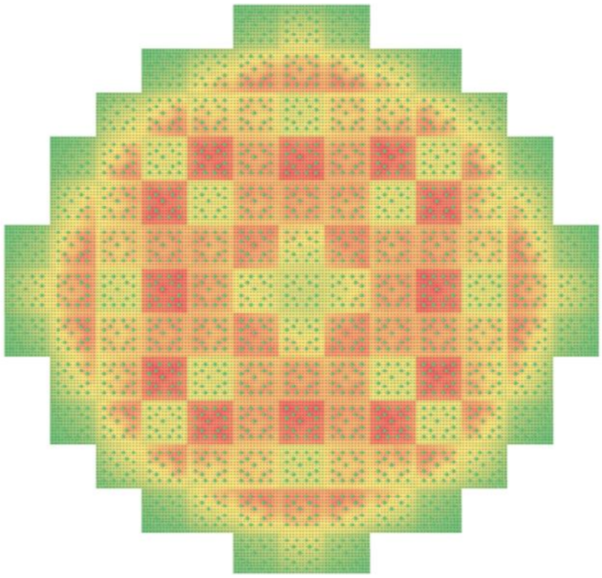

Fig.3 Source term based on cycle 10 pin-by-pin burnup difference between BOC and EOC

2.3 MCNP modeling of Kori Unit1 and source term definition

As seen in Fig.4, full core is modeled because fuel assemblies in the core of Kori unit1 cycle 10 are asymmetrically loaded. In this work, we used the fresh fuel compositions in MCNP6 model because it is known that there is little difference in neutron leakage outside the core between fuel composition of cycle average burnup and fuel composition of BOC [4].

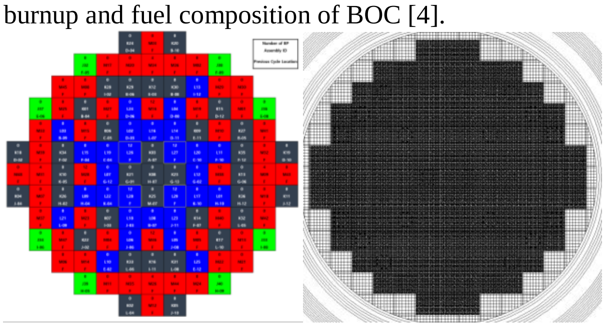

Fig.4 Kori unit1 cycle 10 fuel assembly loading pattern (left) and core modeling for MCNP6 (right)

Fig.5 shows the geometrical shape of reactor component outside the core such as upper support columns, upper support plate, lower support columns, lower support plate, and tie plate belong to the reactor vessel internals (RVI) and they were modeled in a homogenized form.

This figure also shows the geometrical shape of the reactor pressure vessel (RPV) which was modeled as a cylindrical shape with a hemispherical head. The bio-shield was also modeled as a cylindrical shape. The RPV and bio-shield regions were modeled by dividing into 10 segments in radial direction to calculate the radial flux distribution. And the Table I shows the dimensions of each structure in the modeling specifications of Kori unit 1. The bypass and downcomer, not specified in height, are passages through which coolant flows.

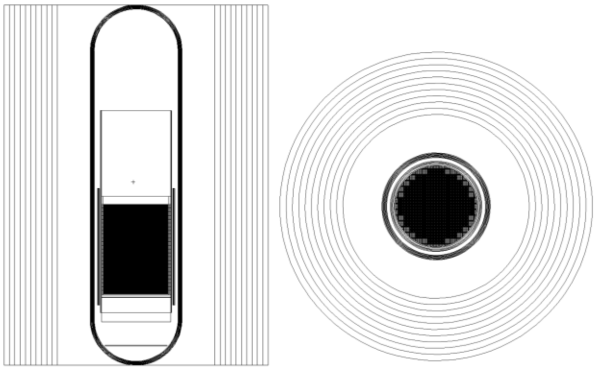

Fig.5 Kori unit1 MCNP modeling of reactor vessel internal, RPV and bio-shield vertical (left) and horizontal cross-sectional view of model (right)

Table I: Kori unit 1 Modeling Specification

	Radius (cm)	Height (cm)
Barrel	142.875	822
Bypass	146.685	-
Thermal shield	155.575	475
Downcomer	167.64	-
Pressure vessel	184.15	1467
Bio-shield	530	1467

The 'nonu' option in the MCNP6 transport calculation was employed to eliminate the neutron production from fission reactions originated from nuclear fuel material during simulation.

2.4 Variance reduction using ADVANTG

To reduce computational time using variance reduction method to MCNP transport calculation, the ADVANTG code was used. ADVANTG code calculates the adjoint flux based on the MCNP input file and produces the weight window file. The weight window applied in the MCNP input file assigns weights to source particles in the tally regions, such as RPV and bio-shield, effectively reducing the computational time for neutron flux calculation. To apply the weight window, the region of Kori unit 1 model in the MCNP input file was divided into 5cm in the x, y, and z directions. the Forward Consistent Adjoint Driven Importance Sampling (FW-CADIS) was applied for variance reduction.

2.5 Flux on RPV and bio-shield

The tally results for RPV and bio-shield obtained from the MCNP were multiplied by a normalization factor to get the actual neutron flux as follows:

$$\phi\left[\frac{neutrons}{cm^2 * s}\right] = \frac{tally\ result\left[\frac{1}{cm^2}\right] * P\left[\frac{J}{s}\right] * \nu\left[\frac{neutrons}{fission}\right]}{1.602E-13\left[\frac{J}{MeV}\right] * 200\left[\frac{MeV}{fission}\right]}. \quad (1)$$

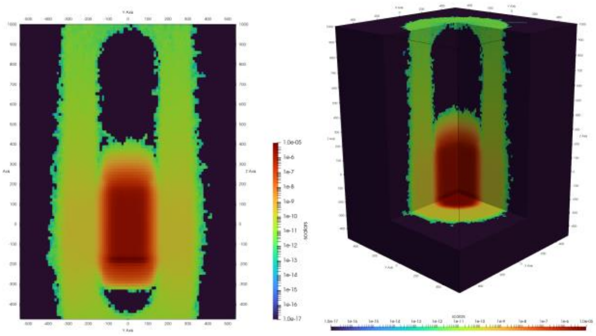

Fig.6 Flux distribution on Kori unit 1 MCNP modeling

As shown in the Fig.6, the neutron flux tends to decrease as moves outward form the reactor core. The average total neutron fluxes over the entire RPV and bio-shield are 2.52×10^{10} neutrons/cm^{2}s and 4.87×10^{8} neutrons/cm^{2}s, respectively. Fig. 7 and 8 show the axially averaged total neutron flux distributions over the RPV and bio-shield regions, respectively.

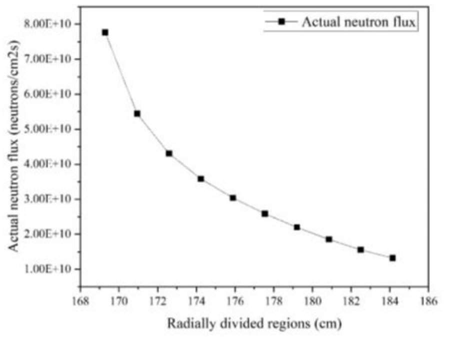

Fig.7 Axially averaged total neutron flux distribution over RPV

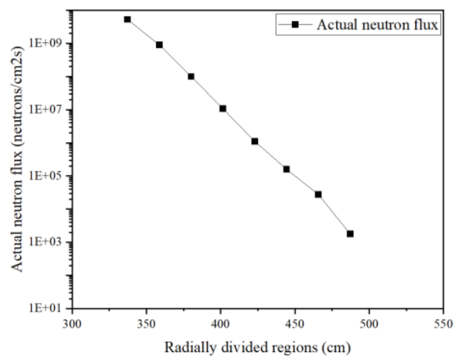

Fig.8 Axially averaged total neutron flux distribution over bio-shield

3. Conclusions

In this study, the neutron fluxes on the RPV and bio-shield of Kori 1 were calculated by the source terms obtained from the core follow calculations with DECART2D/MASTER code. In particular, the fuel rod-wise source distribution was assigned to be proportional to the fuel rod-wise burnup change distribution. The MCNP6 transport calculation was applied to a representative cycle which was also selected as the cycle having the highest burnup change in the outer two row assemblies. The neutron transport calculation using MCNP6 was done with the FW-CADIS variance reduction method using ADVANTG code. In the future, we have a plan to perform the detailed activation calculation coupled with the activation codes such as ORIGEN and FISPACT-II.

Acknowledgement

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