Core Modeling with MCNP for a Quantification of Decommissioning Wastes from Korean Standard Nuclear Power Plants

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1. Introduction

The source terms of decommissioning wastes are used for the determination of decommissioning and decontamination technologies, prediction of an exposure for workers, delay time to a decommissioning initiation, and input parameters for safety analysis, prior to a decommissioning of nuclear reactors. Therefore, a reasonable source term evaluation is inevitable for a proper cost estimation of decommissioning of nuclear power plants.

In this paper, a quarter-core model to assess the decommissioning wastes was developed by the MCNP code, and verified by the FSAR data. First, analysis system to quantify the waste from a structural component surrounding a core was proposed, and then the MCNP core model was established.

2. Analysis System for Activated Material

2.1. Nuclide Buildup in Structural Component

The nuclide inventory change in an activated material can be simply expressed by Eq. (1).

$$\frac{dN_i}{dt} = \sum_j \delta_{ij}\lambda_j N_j + \sum_k f_{ik}\sigma_k\phi N_k - (\lambda_i + \sigma_i\phi)N_i \qquad (1)$$

where, δ_{ij} =fraction of radioactive disintegration by other nuclides, which leads to formation of species i, f_{ik} =fraction of neutron absorption by other nuclides, which leads to formation of species i. Other nomenclatures represent conventional meaning.

The nuclide inventory in a structural component can be quantified by ORIGEN2 with a built-in neutron cross section library. Because the neutron spectrum in structural components, however, is softer than that in a core, the radioactive nuclide inventory generated by the (n, r) reaction of parent nuclide is estimated to be lower than a realistic value, if the neutron capture cross section is applied without a correction. As listed in Table 1, it is revealed that the neutron capture cross section in structural components surrounding a core is three times higher than the value in the built-in library which is made by reflecting the neutron spectrum of a core. Table 1 compares the capture cross sections in the PWRU library in ORIGEN2 with the cross sections generated by the MCNP simulation for the isotopes mainly composing a reactor vessel.

Table 1. Comparison of neutron capture cross sections

Nuclide	ORIGEN2 (PWRUS Library)	MCP (core/vessel)
Fe-54	0.196	0.207/0.686
Co-59	4.919	4.653/12.14
No-61	0.224	0.213/0.741

2.2. Proposed System

The analysis system should evaluate and apply the neutron flux and the capture cross section properly in the interest region, as explained above. In this study, MCNP/ORIGEN2 was chosen as an analysis tool. In this system MCNP is used to simulate the neutron from the reactor core to the structural component in the outer region of the core, and to calculate the flux and the cross-section at the region. And then, the flux and cross-sections from the MCNP simulation are applied to ORIGEN2 to obtain the radionuclide inventory. The one-group capture cross section needed to solve Eq. (1) is retrieved by MCNP through Eq. (2).

$$\bar{\sigma}_\gamma = \frac{\iint \sigma_\gamma(E)\,\phi(E)\,dEdV}{\iint \phi(E)\,dEdV} \qquad (2)$$

By replacing the neutron capture cross section for a relevant nuclide with the cross section generated by MCNP, the inventory for a structural component can be reasonably estimated.

3. Core Modeling with MCNP

3.1. Model Description

An explicit model for an initial core of Yonggwang unit 5, which is a Korean Standard Nuclear Power Plant, was established by the MCNP code. This core consists of 177 fresh fuel assemblies. Four mainly different initial enrichments are 1.28, 2.34, 2.84, and 3.34 weight percents of ^{235}U. The fuel assemblies including burnable poison rods and guide thimbles, and a radial structural component surrounding the core, were exactly modeled without an approximation. However, structural components supporting the core such as the lower support structure and in-core instrumentation nozzle, and the upper guide structure assembly in the below and above regions of the core had not been modeled yet, because their influence on the reactivity and power distribution of the core is negligible. For the fuel, the continuous cross-section library generated at 900K, based on ENDF/B-V nuclear data was used. For the moderator, the cross-section library generated at 600K was used. The moderator density calculated at 15.5 MPa and 312°C was applied to the entire core model.

Because the initial core has a symmetric geometry, a quarter-core model was established to decrease the computation burden for the simulation for a verification of the model. However, it can be easily extended to a full core model, when the transport calculation to obtain the flux and cross sections at a structural material in the top and bottom region of the core is needed. Fig. 1 represents a quarter-core model for Yonggwang unit 5 developed by MCNP. The 20cm-thick water layers were added to the top and bottom region of the core model, respectively.

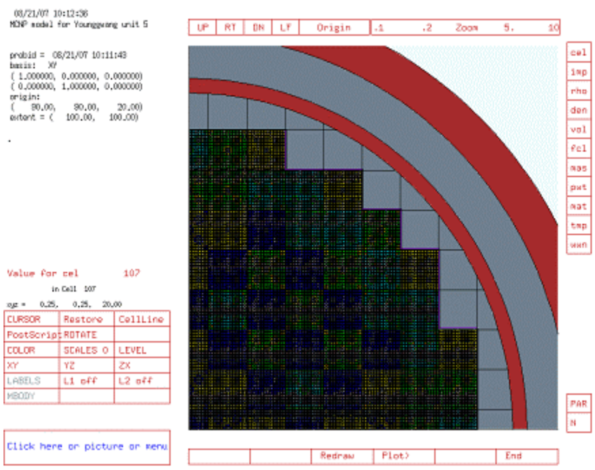

Fig. 1. MCNP Model for Yonggwang unit 5

The multiplication factor and relative power distribution from the MCNP simulation were compared with the data in FSAR for Yonggwang unit 5 to verify the model's adequacy. In this calculation, reflective boundary condition at the left and lower surface of the model in Fig.1 was applied. As a result, multiplication factor from the MCNP simulation was revealed to be 0.99749±0.00045. The root mean square error for the relative power distribution between the MCNP and FSAR results was revealed to be ~4.5%. This means that the developed core model was established properly.

1.12 1.02 8.93	0.95 0.86 9.47	0.62 0.59 4.84					FSAR MCNP %Error
1.31 1.22 6.87	1.27 1.17 7.87	1.22 1.17 4.10	1.12 1.12 0.00	0.63 0.62 1.59			
1.07 0.98 8.41	1.24 1.17 5.65	0.87 0.83 4.60	1.33 1.33 0.00	1.20 1.20 0.00	0.75 0.77 -2.67		
1.14 1.07 6.14	0.80 0.75 6.25	1.13 1.10 2.65	0.82 0.80 2.44	1.21 1.21 0.00	1.20 1.22 -1.67	0.63 0.64 -1.59	
0.76 0.74 2.63	1.15 1.17 -1.74	0.73 0.71 2.74	1.08 1.09 -0.93	0.82 0.82 0.00	1.33 1.39 -4.51	1.12 1.15 -2.68	
1.04 1.06 -1.92	0.74 0.75 -1.35	1.04 1.10 -5.77	0.73 0.75 -2.74	1.13 1.20 -6.19	0.87 0.88 -1.15	1.22 1.23 -0.82	0.62 0.63 -1.61
0.88 0.91 -3.41	1.06 1.12 -5.66	0.74 0.77 -4.05	1.15 1.24 -7.83	0.80 0.82 -2.50	1.24 1.27 -2.42	1.27 1.33 -4.72	0.95 0.97 -2.11
0.68 0.67 1.47	0.88 0.91 -3.41	1.04 1.10 -5.77	0.76 0.76 0.00	1.14 1.17 -2.63	1.07 1.13 -5.61	1.31 1.36 -3.82	1.12 1.00 10.71

Root Mean Square Error = 4.6%

Fig. 2. Comparison of Relative Power Distribution

4. Conclusions

A quarter-core model to assess the decommissioning wastes from Yonggwang unit 5 was developed and verified. An analysis system, MCNP/ORIGEN2 to evaluate the decommissioning waste was proposed. The multiplication factor and relative power distribution from the MCNP simulation were compared with the data in FSAR. As a result, it was concluded that the core model developed in this study was reasonable, because the multiplication factor and relative power distribution from the MCNP simulation agreed well within a justifiable value.

ACKNOWLEDGEMENTS

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