Development of a CANDU Full-core Model with MCNP for a Characterization of Decommissioning Wastes

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1. Introduction

Recently, an attempt to develop a cost evaluation system for a reactor decommissioning was initiated in our country[1)]. Its eventual goal is to minimize the conservatism from operation data, an analysis tool, and cost data, etc.

In the past, one- or two-dimensional transport codes such as the ANISN or DORT have been used to calculate the neutron flux for a region of interest in a source term analysis for decommissioning. It was revealed that the nuclide inventory estimated by these codes in the structural components at the inner core agreed well within about 20 percent with the measured data. The nuclide inventory in the structural components at the outer core, however, deviated up to factors of 10. To enhance the accuracy in the estimation of the nuclide inventory, the Monte Carlo code was propsed[2)]. However, an implicit modeling describing the core with a few regions through geometry and composition averaging technique rather than an explicit modeling describing a heterogeneous geometry was carried out in these studies.

In this paper an MCNP full-core model for an arbitrary equilibrium CANDU core with an explicit modeling methodology was developed and verified.

2. Characteristics of Wolsong Unit 1

Initial core was designed to have 4,400 bundles of a 0.71 weight percent (wt.%) natural uranium fuel and 160 bundles of 0.5wt.% ^{235}U depleted fuel at the 8^{th} and 9^{th} bundle locations to suppress the neutron flux in the center region of the core. As the fissile materials were depleted, 8 fresh bundles at a time were loaded into the channel to provide excess reactivity. By repeating the 8-bundle shift fueling scheme, the core eventually reaches to an equilibrium core.

Although the equilibrium conditions achieved, it is continuously fluctuating because of the 8-bundle shift fueling scheme. It is revealed that, because locations 01 to 08 are occupied by fresh fuel at the time of a fueling, the burnup value at these locations varies from 0 to 3,500 MWhr/bundle as time passes. Because locations 09 to 12 are occupied by fuel that has been burnt in a previous cycle, these locations have a burnup value ranging from 1,000 to 4,000 MWhr/bundle. Since the neutron flux in regions with a low burned fuel is higher than that in the regions with high burnup fuel, the neutron flux at each bundle location is expected to fluctuate under normal operation.

A neutron flux variation from 6,579 to 7,655EFPD for a variety of bundle locations was investigated Although the burnup values at locations 01 to 08 were varied from 0 to 3,500 MWhr/bundle within this range of operation time, the neutron flux stays within a 10 percent variation from the time-averaged mean value, resulting in a 5 % variation at 1σ. This implies that the developed equilibrium core model is a reasonable reference core model useful in the characterization of decommissioning wastes.

3. Equilibrium Core Modeling

3.1 Irradiated Fuel Composition

Prior to pointing out what nuclides should be considered, importance of each nuclide treated in WIMS-AECL was analyzed in terms of neutron capture cross-section. All the actinides are considered to establish an equilibrium core model. Considered in the core modeling are 24 nuclides for the fission products including ^{103}Rh, ^{105}Rh, ^{131}Xe, ^{135}Xe, ^{149}Nd, ^{147}Pm, ^{149}Sm, ^{151}Sm, etc. Finally, 40 nuclides are treated in the equilibrium core modeling, which is over 0.99 in the accumulative capture cross sections.

The continuous cross-section libraries generated at 1,200K, based on ENDF/B-VI nuclear data were used for ^{235}U, ^{238}U, ^{239}Pu. For the other nuclides in the fuel the libraries generated at 300K, based on ENDF/B-VI nuclear data were used. For the coolant and moderator, the cross-section libraries generated at 900K and 600K were used, respectively.

3.2 Structural Components

Due to a small excess reactivity, if a reactivity device and its support, and structural components installed at both sides of the core were excluded, the flux distribution through this model would not be reliable. Therefore all the side structural components such as the calandria-side tube sheet, fuelling machine side tube sheet, steel ball shielding, lattice tubes, annulus gas, and end fitting are implicitly attached in the equilibrium core model.

Reactivity devices such as the SORs, MCAs, ADJs, and LZCs are also modeled implicitly. Three types of LZCs were included in the MCNP model with an implicit geometry. When the implicit geometry was constructed, the mass and volume of a component were preserved.

3.3 Explicit Core Modeling

Two equilibrium core models, a snapshot assembly-averaged model and a snapshot ring-averaged model at a 6,579EFPD operation condition were developed by using the BUNDL program developed in this study. While the assembly-averaged model considers a number density homogenized over a fuel bundle, the ring-averaged model considers a number density homogenized over each fuel ring. All the models have an explicit skeleton of the fuel bundle, pressure tube, and calandria tube. Figures 1 shows a top-view of the equilibrium core model developed by MCNP, respectively.

4. Results

The multiplication factor and the power distribution for the final equilibrium core model were obtained by a KCODE simulation. Source convergence was passed through a fission source entropy check built in MCNP. The multiplication factor and the power distribution from the MCNP simulation were compared with the data calculated by POWDERPUFS/RFSP for the snapshot assembly-averaged model. As a result, the multiplication factor from the MCNP simulation was found to be 0.99721±0.00009. Figure 2 shows a comparison of the channel power distribution between the MCNP and POWDERPUFS/RFSP results. Minus value in the comparison result reveals that the MCNP underestimates the thermal power more than the RFSP. It was found that the results from MCNP and RFSP agreed well to within 4.1% of a root mean square error. This means that the developed core model was established properly.

5. Conclusions

The CANDU full-core MCNP model was developed for a source term characterization of decommissioning wastes through an explicit modeling strategy. Operation history was investigated to check whether the equilibrium core model is appropriate as a reference model or not. It was revealed that the equilibrium core model is appropriate because the flux fluctuation was maintained within 5% of the time-averaged mean value under a 1σ confidence level. Two equilibrium core models, a snapshot assembly-averaged and a bundle-averaged model, were finally proposed by using a BUNDL core modeling subprogram. The multiplication factor from the MCNP simulation was found to be 0.99721±0.00009 for the bundle-averaged model. And, the channel power distribution between the MCNP and RFSP results agreed well to within 4.1% of a root mean square error. Therefore it can be concluded that the CANDU equilibrium full-core equilibrium can be used for a source term characterization for decommissioning wastes and other related areas.

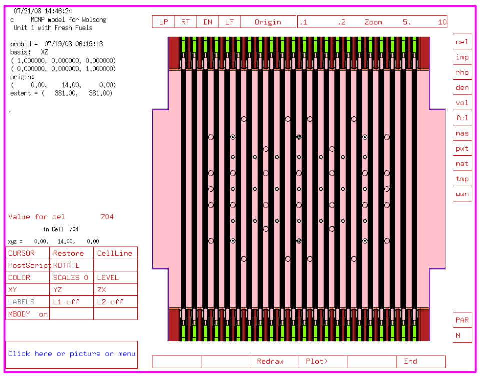

Fig. 1 Top-view of the Equilibrium Core Model

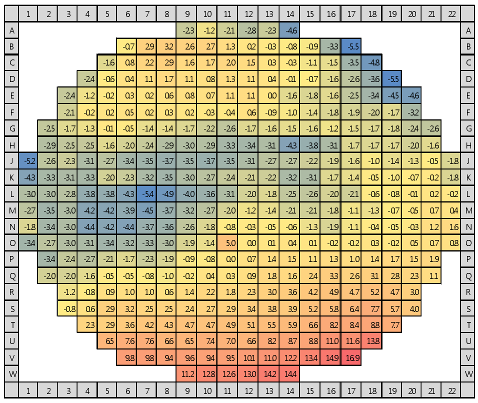

Fig. 2 Comparison of the Channel Power Distribution between the RFSP and MCNP Results

ACKNOWLEDGEMENTS

We would like to acknowledge that this work was funded by the Ministry of Knowledge Economy

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