Feasibility Study on MCNPX-based Burnup Calculation for CANDU Reactor Analysis

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1. INTRODUCTION

In recent years, particle transport codes based on Monte Carlo method have been widely used for analysis of various reactors such as PWR, FBR, and LMR, etc. Especially, the MCNP code, which is developed in Los Alamos National Laboratory (LANL), is a very powerful in solving gamma, neutron, and electron transport equation, and provides exact solutions (i.e., particle flux and dose rate) for a wide range of energy level and for various geometries. Besides, one of the latest extension versions of the MCNPX code (i.e., MCNPX 2.6.0) was integrated with CINDER90 depletion code [1]. Hence, burnup can be calculated for the various combinations of materials defined on the user input, as well as for each individual material containing a fissile actinide.

However, the neutronic analysis of the domestic CANDU reactors has been performed only by using the deterministic codes, such as WIMS, Dragon and RFSP. In this study, therefore, the performances of the MCNPX 2.6.0 code were evaluated to investigate the suitability and possibility of MCNPX-based burnup calculation in CANDU reactor. For this reason, the multiplication factor and physical parameters (η, f, p, and ε) as a function of burnup were calculated for an infinite lattice of natural uranium fuel. The results were then compared with those from the WIMS-IST code which is a multi-group transport code for lattice calculations.

2. METHODS AND MATERIALS

The unit cell model of a standard CANDU6 reactor was employed to investigate the performance of MCNPX-based burnup calculation. The unit cell consists of 37 fuel pins, coolant (D_2O), pressure tube (Zr-Nb), Calandria tube (Zircaoloy-2), and moderator (D_2O), as shown in **Fig. 1**. And, the fuel pin contains pellets of uranium dioxide with natural uranium (0.711 % U-235) and cladding (Zircaloy-4). In order to derive the infinite multiplication factor (k_∞), the edge planes of the unit cell model were set to have the reflective boundary condition. And, the ENDF/B-VI libraries (at room temperature), which are different with the cross-section data applied in WIM-IST code were employed in this problem because it was widely used in reactor physics analysis [2-3]. The *kcode* card was also used for defining the neutron source term, and initial spatial fission distributions were selected by the *sdef* card on behalf of *ksrc* card. The 1,000 neutrons per cycle and initial k_∞ of 1.0 were applied, and fifty cycles among total 250 cycles were skipped before data accumulation of multiplication factor.

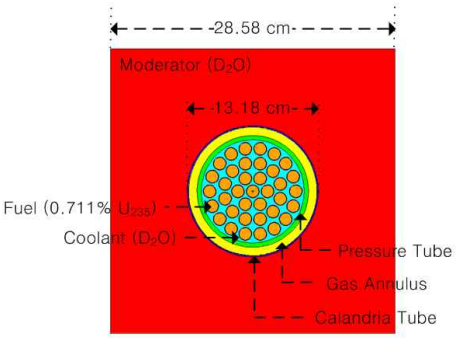

Fig. 1. Configuration of the Unit Cell Model for 37-Element Fuel (XY Cross-section Plane)

In reactor physics analysis, the four physical parameters (η, f, p, and ε) are also investigated to determine the multiplication of a nuclear chain reaction in an infinite medium. In this study, these parameters as a function of burnup were derived by following the Wolsong NPP design manual [4], as follows:

$$\eta = \left\{ \nu\sum_{f2}\phi_2 / \sum_{a2}\phi_2 \right\}_{fuel}$$

$$f = \left\{ \sum_{a2}\phi_2 V \right\}_{fuel} / \left\{ \sum_{a2}\phi_2 V \right\}_{cell}$$

$$p = \left\{ \sum_{a2}\phi_2 / \left[\sum_{a1}\phi_1 + \sum_{a2}\phi_2 \right] \right\}_{cell}$$

$$\varepsilon = \left\{ \left[\nu\sum_{f1}\phi_1 + \nu\sum_{f2}\phi_2 \right] / \nu\sum_{f2}\phi_2 \right\}_{fuel}$$

where,

$\nu\sum_{f1}$=Neutron Production Rate per Fast Fission

$\nu\sum_{f2}$=Neutron Production Rate per Thermal Fission

$\sum_{a1}$=Macro. Absorp. Cross-section for Fast Neutron

$\sum_{a2}$=Macro. Absorp. Cross-section for Thermal Neutron

ϕ_1 = Fast Neutron Flux

ϕ_2 = Thermal Neutron Flux

3. RESULTS AND DISCUSSIONS

In order to validate the performances of the MCNPX-based burnup calculation, the infinite multiplication factor and four physical parameters were compared with those from WIMS-IST code. **Table 1** shows the infinite multiplication factors as a function of burnup and the difference between calculated results for the two codes. As shown in **Table 1**, the multiplication factor

consistently decreased with increasing the burnup, and the k_∞ values from MCNPX 2.6.0 code were found to be slightly lower than those derived by the other code. However, differences between these results were gradually reduced from 7% to 4%, as burnup was changed from 0 to 8,367 MWD/MTU.

Table 1. Infinite Multiplication Factor derived from WIMS-IST and MCNPX Codes

Burnup [MWD/MTU]	k_∞		
	WIMS-IST	MCNPX	MCNPX / WIMS-IST
0	1.11892	1.03968	0.929
840	1.0788	1.00495	0.932
1,680	1.07714	1.0064	0.934
2,351	1.07042	1.00495	0.939
3,021	1.06136	0.9956	0.938
3,691	1.05009	0.99245	0.945
4,360	1.03988	0.98538	0.948
5,028	1.02839	0.97585	0.949
5,696	1.01679	0.9663	0.950
6,364	1.00519	0.95837	0.953
7,032	0.99382	0.95018	0.956
7,700	0.98268	0.94119	0.958
8,367	0.97186	0.933	0.960

The basic parameters (η, f, p, and ε) for reactor physics analysis were investigated by above-mentioned codes, as shown in **Fig. 2**. Particularly, reproduction factor, η, was significantly decreased with proceeding burnup, and it is shown that this factor from MCNPX code is more sensitive to the fuel burnup than the other one. The resonance escape probability (p) and thermal utilization factor (f) from WIMS-IST remained almost constant while those from MCNPX code steadily increased with respect to burnup. And, the last factor, ε, has the nearly same value, regardless of the used codes.

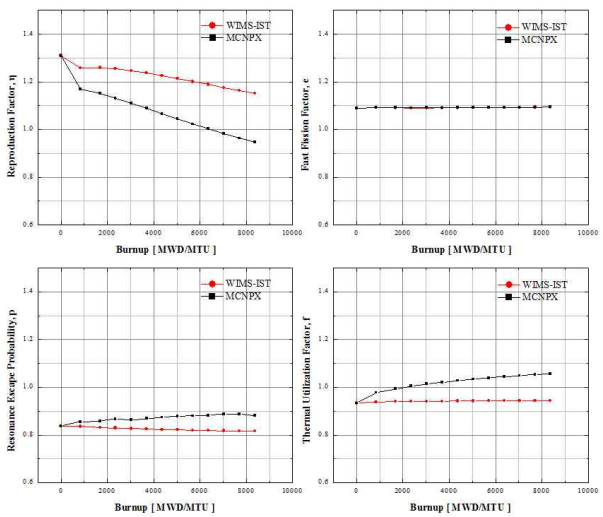

Fig. 2. Change of Four Physical Parameters as a Function of Burnup (0–8,367 MWD/MTU)

Based on these results, we can deduce that the ENDF/B-VI libraries (at room temperature) are not suitable for CANDU reactor analysis as a function of fuel burnup.

4. CONCLUSIONS

The performances of the MCNPX 2.6.0 code were evaluated to investigate the applicability of MCNPX-based burnup calculation in CANDU reactor. For this reason, the multiplication factor and physical parameters (η, f, p, and ε) as a function of burnup were calculated for an infinite lattice of natural uranium fuel. It was found that the infinite multiplication factor (k_∞) continually decrease with increasing the burnup, and these values from MCNPX code were slightly lower than those derived by WIMS code, which is the reference code for CANDU calculations. Other parameters, regardless of fast fission factor (ε), were also shown to have some differences between two codes. Therefore, it can be deduced that cross-section libraries at room temperature does not be suitable for CANDU reactor analysis, although these data was widely used in the analysis of reactor physics.

ACKNOWLEDGEMENT

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