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Analysis of Fuel Temperature Reactivity Coefficients According to Burn-up and Pu-239 Production in CANDU Reactor

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1. Introduction

The fuel temperature coefficient (FTC) of reactivity in CANDU reactors has recently been of concern, as it is evaluated as a positive value over equilibrium states from calculation with some reactor physics codes and methodology [1]. There seem to be two reasons largely effecting positive FTCs, i.e., spatial resonance processing in U-238 related to fuel temperature [2], and Pu-239 build-up with burnup. In particular, the resonances for some kinds of nuclides such as U-238 and Pu-239 are not easy to be accurately processed. In addition, the Pu-239 productions from burnup are also significant in CANDU, where the natural uranium is used as a fuel.

In this study, the FTCs were analyzed from the viewpoints of the resonance self-shielding methodology and Pu-239 build-up. The lattice burnup calculations were performed using the TRITON module in the SCALE6 code system, and the BONAMI module was executed to obtain self-shielded cross sections using the Bondarenko approach [3]. Two libraries, ENDF/B-VI.8 and ENDF/B-VII.0, were used to compare the Pu-239 effect on FTC, since the ENDF/B-VII has updated the Pu-239 cross section data.

2. Materials and Methods

The SCALE code system typically performs two stages to apply the resonance self-shielding methods. For the calculation of the effective multi-group (MG) cross sections, BONdarenko AMPX Interpolator (BONAMI) module is first applied with equivalence theory and narrow resonance (NR) approximation. Next, the Continuous ENergy TRansport Module (CENTRM) module is executed to the compute high-resolution flux spectra by solving the discrete ordinates transport equation using continuous nuclear data. In particular, the BONAMI module uses the Bondarenko method for resonance self-shielding. The value for the shielded cross section is interpolated from the MG library.

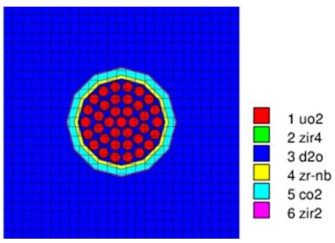

Fig. 1 The Unit Lattice Cell with 37 Fuel Pins of CANDU Reactor (Modeled using TRITON)

Figure 1 shows a unit lattice cell modeled with the TRITON sequence of SCALE6. The reactor lattice consists of 37 fuel pins (0.711% U-235) and D_2O coolant contained in a Zr-Nb pressure tube arranged in a D_2O moderator [1]. The edge planes were set to have a reflective boundary condition. For the 2-D lattice burnup calculation, the time-dependent depletion (T-DEPL) mode was used. NEWT was incorporated into the SCALE TRITON sequence for eigenvalue calculations.

3. Results and Discussions

The depletion calculations were performed up to 8000 MWd/MTU with an average power density of 33.4902 MW/MTU. The reactivity according to the fuel temperature was evaluated on fresh fuel under mid-burnup conditions (conservatively 4.0 GWD/MTU). Fuel temperatures were applied from 298.15 K to 1473.15 K to verify the overall reactivity changes on temperature.

As shown in Figure 2, the FTCs in two burnup conditions are evaluated to be obviously negative over the entire fuel temperature range. Compared with the ENDF/B-VI library, the reactivity change evaluated with ENDF/B-VII is slightly higher and becomes slightly lower based at the operating temperature of 960.15 K.

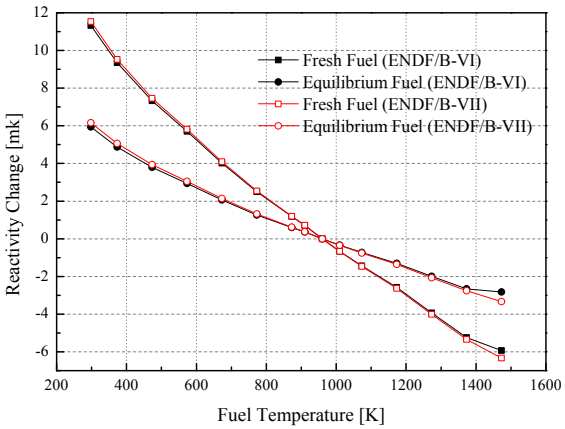

Fig. 2 The Reactivity Change According to Fuel Temperature for Fresh and Equilibrium Fuel

Figure 3 shows the FTCs varying with burnup increase. The FTCs with burnup were obtained at 960.15±50 K. Some reactor physics codes have resulted in slightly positive FTCs in a range above the mid-burnuup point (about 3600 MWd/MTU). However,

the FTCs evaluated in this study were turned out to be slightly negative over the entire burnup range.

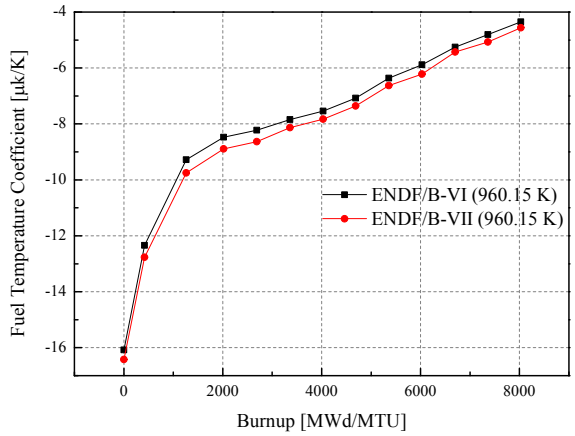

Fig. 3 The Fuel Temperature Coefficient with the Change of Burnup at the fuel temperature of 960.15 K

Pu-239 is known to contribute to positive reactivity feedback owing to the presence of its high fission resonances at around 0.293 eV of thermal energy region. This makes the FTC become less negative. This is especially noticeable in well-moderated systems in addition to the use of non-enriched uranium fuel. Furthermore, ENDF/B-VII.0 improved Pu-239's cross-section library for fission product yield from new experimental data (by LANSCE GEANIE) combined with older data and GNASH theoretical calculations [4]. Therefore, it is necessary to verify the reactivity differences by Pu-239 build-up of the CANDU reactor calculated from these two libraries. As tabulated in Table 1,the amount of Pu-239 with burnup is evaluated in using ENDF/B-VI library. If it is assumed that the Pu-239 cross-section is only considered to be advanced for library, it is analyzed that the Pu-239 effects have an average positive reactivity of ~0.0003 mk/K over the entire burnup region.

Table 1. The Pu-239 Concentration in the Fuel Cell Calculated with Two Kinds of Library

Burnup [MWd/MTU]	Number Density of Pu-239 [atoms/barn-cm]	
	ENDF/B-VI.8	ENDF/B-VII.0
840	1.48E-05	1.48E-05
1680	2.64E-05	2.63E-05
2351	3.31E-05	3.30E-05
3022	3.83E-05	3.81E-05
3692	4.24E-05	4.21E-05
4361	4.56E-05	4.53E-05
5029	4.82E-05	4.79E-05
5697	5.02E-05	4.99E-05
6365	5.19E-05	5.16E-05
7032	5.33E-05	5.29E-05
7700	5.44E-05	5.40E-05
8368	5.53E-05	5.49E-05

3. Conclusions

The FTCs of the CANDU reactor were newly analyzed using the TRITON module in the SCALE6 code system, and the BONAMI module was executed to apply the Bondarenko approach for self-shielded cross sections. When compared with some reactor physics codes resulting in slightly positive FTC in the specific region, the FTCs evaluated in this study showed a clear negativity over the entire fuel temperature range on fresh/equilibrium fuel. In addition, the FTCs at 960.15 K were slightly negative during the entire burnup. The effects on FTCs from the library difference between ENDF/B-VI.8 and ENDF/B-VII.0 are recognized to not be large; however, they appear more positive when more Pu-239 productions with burnup are considered. This feasibility study needs an additional benchmark evaluation for FTC calculations, but it can be used as a reference for a new FTC analysis in CANDU reactors.

Acknowledgements

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REFERENCES

[1] KOrea Power Engineering Company, INC, Design Manual: CANDU 6 Generating Station Physics Design Manual, Wolsong NPP 1, 59RF-03310-DM-00, Revision 0, 2009.
[2] Y. Kim and D. Hartanto, "Re-evaluation of the Fuel Temperature Coefficient of CANDU 6," Transactions of the Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, 2011.
[3] M. D. DeTart, TRITON: A Two-dimensional Transport and Depletion Module for Characterization of Spent Nuclear Fuel, ORNL-TM 0 2005/39, Version 6, 2009.
[4] M. B. Chadwick, *et al.,* "ENDF/B-VII.0: Next Generation Evaluated Nuclear Data Library for Nuclear Science and Technology," Nuclear Data Sheets 107, 2006.