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Original Article

A Lattice-Based Monte Carlo Evaluation of Canada Deuterium Uranium-6 Safety Parameters

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ABSTRACT

Important safety parameters such as the fuel temperature coefficient (FTC) and the power coefficient of reactivity (PCR) of the CANada Deuterium Uranium (CANDU-6) reactor have been evaluated using the Monte Carlo method. For accurate analysis of the parameters, the Doppler broadening rejection correction scheme was implemented in the MCNPX code to account for the thermal motion of the heavy uranium-238 nucleus in the neutron-U scattering reactions. In this work, a standard fuel lattice has been modeled and the fuel is depleted using MCNPX. The FTC value is evaluated for several burnup points including the mid-burnup representing a near-equilibrium core. The Doppler effect has been evaluated using several cross-section libraries such as ENDF/B-VI.8, ENDF/B-VII.0, JEFF-3.1.1, and JENDL-4.0. The PCR value is also evaluated at mid-burnup conditions to characterize the safety features of an equilibrium CANDU-6 reactor. To improve the reliability of the Monte Carlo calculations, we considered a huge number of neutron histories in this work and the standard deviation of the *k*-infinity values is only 0.5–1 pcm.

1. Introduction

In the CANada Deuterium Uranium-6 (CANDU-6) reactor, heavy water (D_2O) is used as coolant and moderator simultaneously inside a unique lattice arrangement, thus enabling utilization of natural uranium fuel without any enrichment and both on-power fuel loading and unloading. To obtain a highly thermalized neutron spectrum in CANDU-6, a pressurized cylindrical fuel channel is submerged in a bulky, thermally insulated, and near-static D_2O moderator, and the nuclear heat produced in the fuel channel is removed by a small volume of D_2O coolant, which plays a limited role in the neutron moderation. In a standard CANDU-6 fuel lattice, the coolant volume fraction is only about 4.3% and the coolant-to-moderator volume ratio is about 0.052. The energetic fission neutrons are first slightly moderated by the D_2O coolant and then they are fully thermalized in the large, cold moderator. When the thermalized neutrons reenter the fuel channel, they

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can even be upscattered by the hot coolant. Because of these relatively complicated neutron interactions, the CANDU-6 reactor shows unique behavior in terms of the major safety parameters such as coolant temperature coefficient (CTC), fuel temperature coefficient (FTC), and the resulting power coefficient of reactivity (PCR) [1].

It is well-known that the coolant void reactivity (CVR) of CANDU-6 is positive for operational conditions and the CTC is also positive. The positive CVR and CTC are mainly ascribed to two neutronic phenomena. First, when the coolant is removed from a fuel channel, the resonance escape probability is slightly increased because the slowing down of fission neutrons by coolant disappears and resonance neutron capture by ^{238}U is reduced. Second, more thermal neutrons reach the central fuel elements because the spatial self-shielding of the fuel bundle is slightly weakened due to the coolant voiding. The positive CVR is a long-standing safety concern of CANDU-6 [1,2].

One of the important safety requirements in nuclear reactors is that the prompt feedback from fuel temperature change (i.e., FTC) should be negative. In CANDU-6, due to the unique arrangement of the fuel channel and moderator and the resulting very soft neutron spectrum, the FTC is small because the neutron capture by ^{238}U resonance is relatively small [1,2]. A recent deterministic study [2] shows that the FTC of CANDU-6 is clearly negative for fresh fuel and it can be positive for highly burned fuel. In addition, FTC becomes less negative or more positive with the fuel temperature. Roh et al. [2] showed that the positive FTC in CANDU-6 is mainly ascribed to the neutron upscattering by oxygen in the fuel and the large thermal fission resonance of ^{239}Pu.

The PCR of a nuclear reactor is a combined effect of FTC and CTC because both fuel and coolant temperatures change with reactor power. A negative PCR is necessary to improve the inherent safety features and stability of nuclear reactors. The PCR of CANDU-6 is traditionally known to be slightly negative or close to zero for the full power condition. The small PCR of CANDU-6 is largely due to the small FTC and a clearly positive CTC. In CANDU-6, for a negative PCR, the FTC value should be clearly negative because CTC is already positive under operational conditions [1,2]. Recently, for an equilibrium CANDU-6 core, the PCR and FTC were reported to be slightly positive when the newly developed Industry Standard Toolset reactor physics codes were used. The PCR was calculated to be +1.7 pcm/% power at 100% power level [3]. This indicates that the PCR and FTC evaluation in CANDU-6 is subject to nontrivial calculational uncertainties and the evaluation method should be improved for a more accurate evaluation of the CANDU-6 safety characteristics.

In traditional reactor analysis, the so-called asymptotic scattering kernel has been often used, assuming that the target nucleus is at rest during the scattering reaction with a neutron. However, it is now clearly accepted that, in a scattering reaction, thermal movement of the target can noticeably affect the scattering reaction in the vicinity of scattering resonance and it enhances neutron capture by the capture resonance. Related works indicate that the thermal motion of ^{238}U noticeably affects the scattering reaction and the resulting Doppler broadening of the scattering resonance enhances the FTC of thermal reactors including pressurized water reactors by 10–15% [4–7].

The basic impact of the Doppler broadening of the scattering resonances on the criticality and FTC was first investigated for a clean, fresh CANDU fuel lattice in a recent work [8], in which a modified MCNPX [9] code was used. In the study by Dagan et al. [8], to take into account the thermal motion of ^{238}U, the so-called Doppler broadening rejection correction (DBRC) method [5,6,8,10] was adopted. It was shown that consideration of the ^{238}U thermal motion results in a slightly enhanced FTC (more negative). With the same DBRC method, the authors also performed a preliminary evaluation of the FTC of CANDU-6 at near-equilibrium condition [11,12].

2. Materials and methods

In this study, the safety parameters (FTC and PCR) of CANDU-6 are re-evaluated using the continuous-energy Monte Carlo code MCNPX with the DBRC method to simulate the thermal vibration of ^{238}U. The analysis is performed for a standard CANDU-6 fuel lattice. Temperature-dependent FTC is calculated for several burnup conditions, and power-dependent PCR is evaluated for a mid-burnup condition. The FTC and PCR are quite small in CANDU-6 and the statistical uncertainty should be very small for an accurate evaluation of the parameters. For an accurate Monte Carlo calculation, an extremely large number of neutron histories (3–20 billion) is used in this work.

2.1. *Doppler-broadened rejection correction method*

The DBRC method [5,6,8,10] has been developed to deal with thermal movement of target nuclei in the Monte Carlo simulations of a neutron scattering reaction. It is a statistical approach based on the use of a complementary rejection technique in the Monte Carlo simulation. In MCNPX, a probability function is used to simulate the target velocity and the angle between a neutron and its target. In the DBRC method, a modification of the original probability density function is necessary. With the corrected probability density function, the modified MCNPX (MCNPX–DBRC) is able to include the effect of the energy dependence of the cross sections on the scattering kernel.

From the Doppler broadening theory, the rigorous target probability density function can be written as follows:

$$P(V, \mu_t) = \frac{\sigma_s(E_r, 0) v_r p(V)}{2\sigma_s(E, T) v} \quad (1)$$

where v is the neutron speed, V is the speed of the target, v_r is the neutron velocity relative to the target at rest, μ_t is the scattering cosine, and $p(V)$ is the target velocity distribution at temperature T. In MCNPX, the Maxwell–Boltzmann distribution is used for $p(V)$. $\sigma_s(E, T)$ is the Doppler-broadened scattering cross section at temperature T.

The corrected probability density function in the DBRC method can be written as Eq. (2), $\beta = [Am/(2k_B T)]^{1/2}$, in which A is the atomic number of the target, m is neutron mass, k_B is the Boltzmann constant, and C is a normalization constant as shown in Eq. (3). In Eq. (2), $\sigma_s^{max}(E_\xi, 0)$ is the largest scattering cross section in a specific interval in the vicinity of the actual scattering cross section $\sigma_s(E, T)$. $\xi = \sqrt{AE/k_BT}$ is the dimensionless speed of the neutron. In Eq. (2), the second and third brackets are the constraints on the chosen values of V and μ_t in the first bracket [6]. Initially, the target velocity V in the first bracket is chosen by sampling a specific velocity for the target nucleus out of a Maxwell–Boltzmann distribution. Then, a rejection technique is used in MCNPX for the second and third brackets. If a random number (between 0 and 1) is less than the value of the second bracket, the values chosen for V and μ_t are accepted. The same procedure is applied to the third bracket. If the random number is less than the value of the third bracket, the sampled values of V and μ_t are rejected. If the sampled value is rejected, the sampling of V is repeated for the first bracket. A more detailed description of this method can be found in the work by Becker et al. [6].

$$P(V, \mu_t) = C\left[\frac{2\beta^4V^3e^{-\beta^2V^3} + (\beta\upsilon\sqrt{\pi}/2)(4\beta^3/\sqrt{\pi})V^2e^{-\beta^2V^3}}{1+\beta\upsilon\sqrt{\pi}/2}\right]\left[\frac{\upsilon_r}{\upsilon+V}\right] \times \left[\frac{\sigma_s(E_r, 0)}{\sigma_s^{\max}(E_\xi, 0)}\right] \quad (2)$$

$$C = \frac{\sigma_s^{\max}(E_\xi, 0)(1+\beta\upsilon\sqrt{\pi}/2)}{2\upsilon\sigma_s(E, T)\beta\sqrt{\pi}/2} \quad (3)$$

2.2. *CANDU-6 lattice model problem*

A standard CANDU fuel lattice is modeled and analyzed in this work to characterize the generic safety parameters of CANDU-6. In the CANDU-6 reactor, each fuel channel is placed in a large D_2O moderator region. Therefore, fuel channels are rather loosely coupled with neighboring ones, and an appropriately modeled lattice can be utilized to investigate the general safety characteristics of the whole core.

As shown in Fig. 1, the standard fuel bundle consists of 37 fuel rods. The fuel bundle is loaded into a pressure tube and a calandria tube surrounds the pressure tube that physically separates the moderator from the coolant. Heavy water is used for both coolant and moderator. Table 1 shows the major design parameters used for the fuel lattice in this work. The fuel lattice pitch is 28.575 cm and the average linear power is about 12.94 kW/cm.

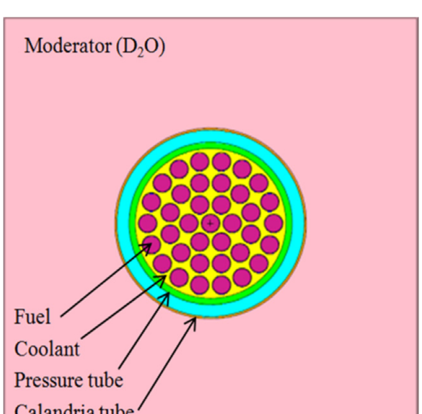

Fig. 1 – The standard CANada Deuterium Uranium fuel lattice.

Table 1 – Design parameters of the standard CANada Deuterium Uranium fuel lattice.

Parameter	Value
Fuel pin	
Number of pins	37
Fuel pellet radius	0.608 cm
Fuel temperature	960.16 K
Fuel density	10.492 g/cm^3
Clad thickness	0.04 cm
Clad temperature	561.16 K
Clad density	6.520 g/cm^3
Pressure tube	
Inner radius	5.179 cm
Outer radius	5.613 cm
Temperature	561.16 K
Density	6.515 g/cm^3
Calandria tube	
Inner radius	6.450 cm
Outer radius	6.590 cm
Temperature	342.16 K
Density	6.544 g/cm^3
Coolant	
Purity	99.89 wt%
Temperature	561.16 K
Density	0.808 g/cm^3
Moderator	
Purity	99.98 wt%
Temperature	342.16 K
Density	1.085 g/cm^3

3. Numerical results and discussion

The safety parameters of CANDU-6 strongly depend on the fuel burnup because the fuel composition changes a lot with burnup. In particular, ^{239}Pu quickly builds up with burnup and it affects a lot the safety features of the CANDU-6 core. With the modified MCNPX code [10] based on the ENDF/B-VII.0 cross-section library, the CANDU fuel lattice has been depleted up to 230 days or 7.5 GWD/tU. The usual average discharge burnup of CANDU-6 natural U fuel is about 7.2 GWD/tU. To see the impact of the DBRC scheme, the lattice was also depleted with the standard MCNPX. In each depletion calculation, 27 burnup steps are used. A total of 150 Monte

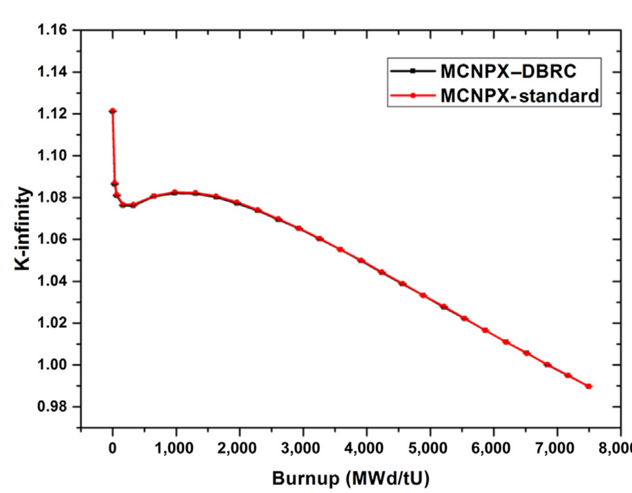

Fig. 2 – The multiplication factor versus burnup.

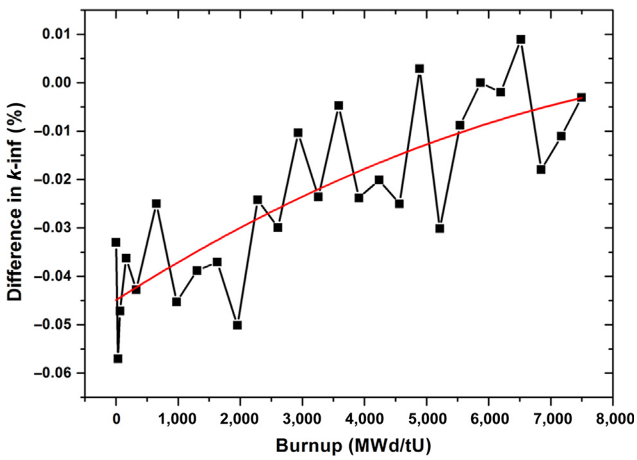

Fig. 3 – Difference in *k*-infinity (*k*-inf) between the modified and standard MCNPX codes.

Carlo cycles with 300,000 histories/cycle were used for each time step. The standard deviation of the *k*-infinity (*k*-inf) values is < 8 pcm. The results of the depletion calculations are shown in Fig. 2.

As shown in Fig. 3, MCNPX–DBRC provides a slightly smaller multiplication factor and the maximum absolute difference between the original and the modified MCNPX is around 0.05% when the burnup is very low. It is also observed that the discrepancy decreases gradually with the fuel burnup. The lower reactivity with the DBRC module is ascribed to enhanced neutron capture by ^{238}U, as will be discussed later. Figs. 4 and 5 show the impact of the DBRC module on the inventory of major fissile nuclides, that is, ^{235}U and ^{239}Pu, as a function of burnup. In Fig. 5, it is clearly noted that the modified MCNPX provides a little higher accumulation of ^{239}Pu by about 0.14%, which is again due to the higher neutron capture reaction of ^{238}U resulting from the Doppler broadening of the scattering resonances. As a result, the consumption of ^{235}U is reduced slightly, which is observed in Fig. 4. The average behaviors of the discrepancies are determined by fitting the Monte Carlo results to polynomials and they are indicated in Figs. 3–5 as well.

To find out the effect of the DBRC scheme on the modified code, the capture reaction rates of ^{238}U at 3.9 GWD/tU are evaluated for several energy intervals and the results are summarized in Table 2. It is worthwhile to note that all the energy bins except the first one contain at least a resonance of ^{238}U. The first resonance of ^{238}U occurs at about 6.6 eV, which is included in the second energy bin. One can clearly note that the neutron capture rate increases noticeably in the resonance-containing energy intervals. In particular, in the energy range of 27.7–48.1 eV, MCNPX–DBRC gives over 2% higher capture reaction rates compared with the MCNPX standard. Table 2 indicates that the total neutron capture rates by ^{238}U are enhanced by about 0.09% due to the consideration of the ^{238}U thermal motion.

The FTC is responsible for a prompt reactivity feedback effect in nuclear reactors and a negative fuel temperature feedback is one of the important components supporting the inherent safety. In a natural U-loaded CANDU-6 reactor, it is well-known that the FTC changes significantly with burnup because it is largely governed by accumulation of ^{239}Pu. The FTC is calculated as a function of burnup and fuel temperature based on the previous model problem. First, the FTC is evaluated at a burnup of 3.9 GWD/tU, which is a little higher than the typical mid-burnup of 3.6 GWD/tU.

Six temperature points (600 K, 798 K, 910 K, 960 K, 1,010 K, and 1,200 K) are considered to estimate the FTC at 3.9 GWD/tU. Fig. 6 provides the reactivity change as a function of the fuel temperature. In Fig. 6, the reactivity change is determined with respect to a reference fuel temperature of 960 K. To examine sensitivity to the nuclear data, several data libraries (ENDF-B/VI.8, JENDL-4.0, and JEFF-3.1.1) are also compared with ENDF-B/VII.0. For the library comparison, we replaced the actinide cross sections of the fuel with the new library data, that is, the original ENDF-B/VII.0 data are still used for the fission products. Although the current library comparison is not a complete one, we believe that the comparison should provide the essential characteristics of each library because the FTC is largely governed by the actinide nuclides in the fuel. For a high-fidelity Monte Carlo evaluation of the FTC values, the standard deviation of the *k*-inf value is reduced by 0.6–1 pcm for the FTC evaluation. For such a low statistical uncertainty, over 3 billion neutron histories are used to

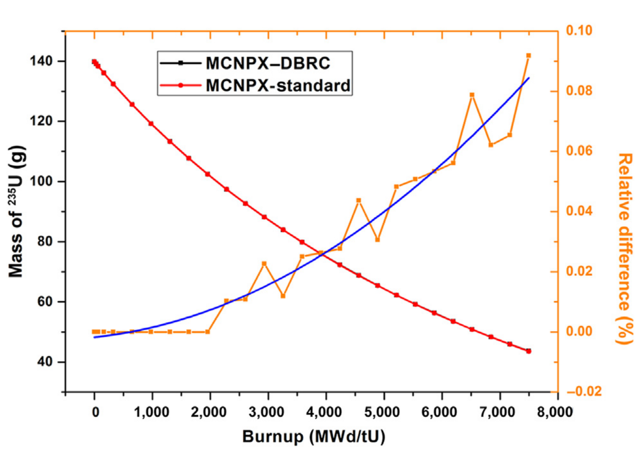

Fig. 4 – The ^{235}U inventory change versus burnup.

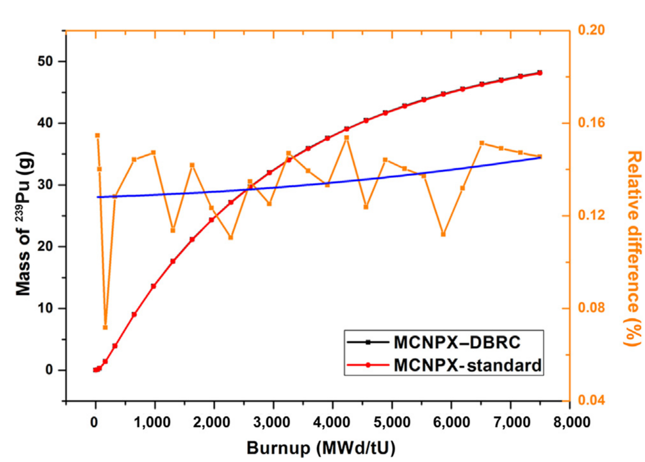

Fig. 5 – The ^{239}Pu inventory change versus burnup.

Table 2 – Comparison of the ^{238}U capture reaction rates between the MCNPX standard and MCNPX–DBRC.

Energy bin (eV)	MCNPX standard capture reaction rate	MCNPX–DBRC capture reaction rate	Differences (%)
1×10^{-3} to 4.0	$2.129 \times 10^{14} \pm 5.493 \times 10^{9}$	$2.128 \times 10^{14} \pm 5.490 \times 10^{9}$	-0.053 ± 0.0002
4.0–9.88	$1.702 \times 10^{13} \pm 3.405 \times 10^{9}$	$1.704 \times 10^{13} \pm 3.409 \times 10^{9}$	0.119 ± 0.0034
9.88–27.7	$1.072 \times 10^{13} \pm 2.910 \times 10^{9}$	$1.085 \times 10^{13} \pm 2.948 \times 10^{9}$	1.206 ± 0.0449
27.7–48.1	$8.161 \times 10^{12} \pm 2.448 \times 10^{9}$	$8.332 \times 10^{12} \pm 2.500 \times 10^{9}$	2.097 ± 0.0890
48.1–75.5	$3.697 \times 10^{12} \pm 1.479 \times 10^{9}$	$3.737 \times 10^{12} \pm 1.495 \times 10^{9}$	1.068 ± 0.0604
75.5–149	$7.302 \times 10^{12} \pm 2.191 \times 10^{9}$	$7.307 \times 10^{12} \pm 2.192 \times 10^{9}$	0.068 ± 0.0029
149–10×10^{7}	$4.846 \times 10^{12} \pm 4.847 \times 10^{9}$	$4.847 \times 10^{13} \pm 4.847 \times 10^{9}$	0.021 ± 0.0003
Total	$3.083 \times 10^{14} \pm 9.311 \times 10^{9}$	$3.085 \times 10^{14} \pm 9.340 \times 10^{9}$	0.084 ± 0.0004

DBRC, Doppler broadening rejection correction.

determine the k-inf at each temperature: 2.01 million histories in each Monte Carlo cycle are used and a total of 1,500 Monte Carlo cycles with 300 inactive cycles are used. The resulting FTC has an uncertainty of ±0.028–±0.030 pcm/K.

Fig. 6 clearly indicates that the reactivity obtained with the DBRC scheme decreases with fuel temperature up to 1,200 K, regardless of the cross-section library, whereas the reactivity tends to increase above ~1,000 K with the original MCNPX. When the fuel temperature is low, the reactivity decreases rather quickly with fuel temperature and the reactivity decreasing rate becomes smaller as the fuel temperature increases. It is also noted that both JEFF-3.1.1 and JENDL-4.0 provide very similar trends in the fuel temperature reactivity.

To determine the FTC of the CANDU-6 lattice, the calculated k-inf values for the six temperatures are fitted into a continuous function and FTC is calculated as the derivative of the resulting fitting function. Taking into account the well-known relationship between the fuel temperature and reactivity in thermal reactors, the infinite multiplication factor (k_∞) is fitted into Eq. (4) [6], where T_f is the fuel temperature. The FTC is then calculated using Eq. (5) [6].

$$k_\infty = a + bT_f^{1/2} + cT_f \tag{4}$$

$$\frac{\partial \rho_\infty}{\partial T_f} = \frac{1}{k_\infty^2}\frac{\partial k_\infty}{\partial T_f} \tag{5}$$

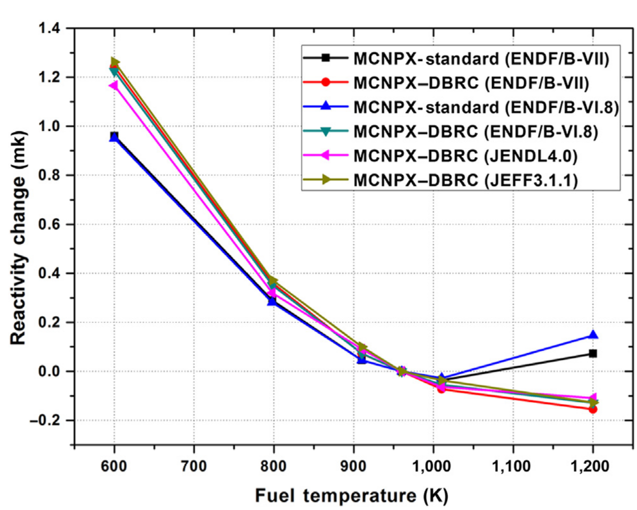

Fig. 6 – The reactivity change as a function of fuel temperature.

Fig. 7 shows the fitting results for the ENDF/B-VI.8 and ENDF/B-VII.0 libraries. From the results, one can see that the new FTC values calculated with MCNPX–DBRC are clearly more negative than those from the original MCNPX code for the two ENDF/B libraries. It is also observed that ENDF/B-VII.0 provides a slightly more negative (or less positive) FTC over a wide range of fuel temperatures for the CANDU-6 lattice problem. It is also noteworthy that the impact of the DBRC method on FTC is rather comparable for the two libraries.

In Fig. 8, the continuous FTC obtained with the DBRC method is plotted for several libraries. It is noted that all the libraries show similar FTC behavior over the fuel temperature change. Among the four libraries, JENDL-4.0 is observed to provide a slightly less negative (or more positive) FTC than the others. Table 3 shows actual predicted FTC values at 3.9 GWD/tU in the vicinity of the average fuel temperature at 100% power. Table 3 shows FTC values at several temperatures for the four cross-section libraries. It is clear that the more accurate analysis with the DBRC scheme provides a much more negative FTC than the standard approximate method; for example, the FTC is enhanced by over 100% at $T = 960$ K for the ENDF/B libraries. Table 3 indicates that the FTC from the JEFF-3.1.1 library is very comparable to that of ENDF/B-VII.0, whereas JENDL-4.0 provides a slightly less negative FTC.

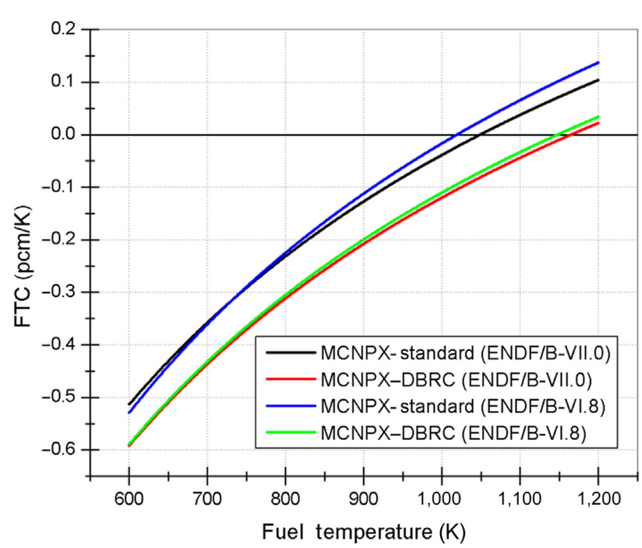

Fig. 7 – Fuel temperature coefficient (FTC) calculated using ENDF/B-VI.8 and VII.0 libraries (3.9 GWD/tU).

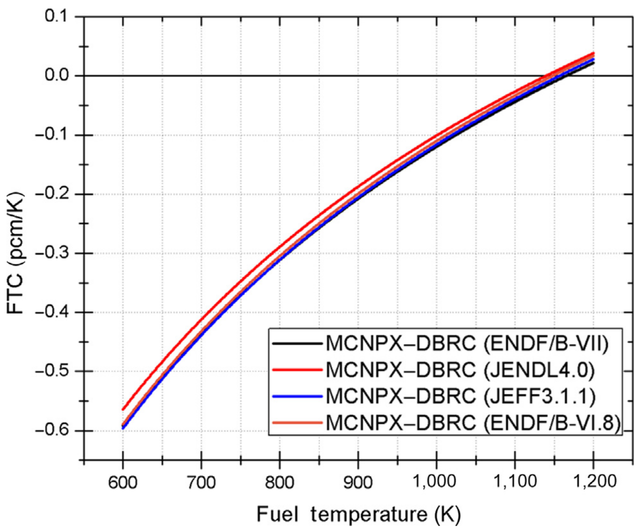

Fig. 8 – Fuel temperature coefficient (FTC) comparison between several nuclear data (3.9 GWD/tU).

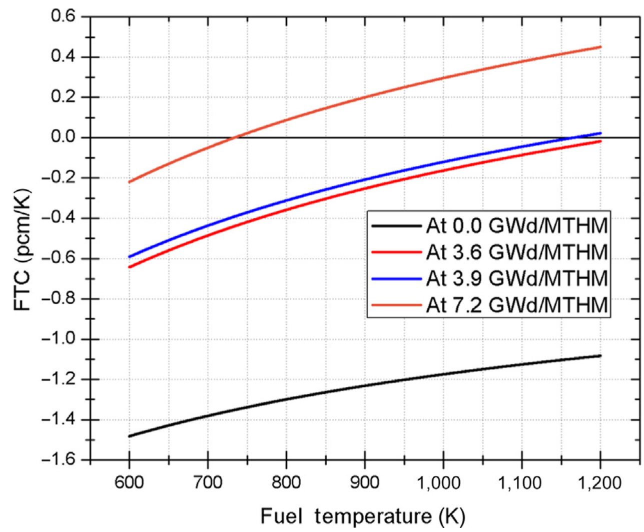

Fig. 9 – Fuel temperature coefficient (FTC) comparison at several burnup points (ENDF/B-VII.0).

Using the MCNPX–DBRC code, the FTC at several burnup points has been evaluated using the ENDF/B-VII library and the results are shown in Fig. 9 and Table 4. As is well-known, the FTC is strongly negative for the fresh CANDU fuel condition and it becomes less negative with burnup. At the near-equilibrium burnups (3.6 GWD/tU and 3.9 GWD/tU), the FTC is only slightly negative and it becomes clearly positive at a discharge burnup of 7.2 GWD/tU. Table 4 shows the FTC values at three fuel temperatures for several burnup points.

The inherent safety of nuclear reactors is largely dominated by the PCR, which is defined as the reactivity change in response to a unit change of the power. For self-regulation of a nuclear reactor with respect to a small perturbation, the PCR needs to be negative.

It has been shown that the PCR of the CANDU-6 reactor can be well-estimated using a lattice model if the coolant and fuel properties are appropriately determined [2]. The same method is used in this work to investigate the impact of the DBRC scheme on the PCR of CANDU-6. The power-dependent lattice parameters, such as fuel and coolant temperatures and coolant density, have been determined from a thermal-hydraulic-coupled three-dimensional neutronic analysis of the CANDU-6 reactor [13] and they are given in Table 5.

The PCR was evaluated at 3.6 GWd/tU and 3.9 GWd/tU because the mid-burnup (3.6 GWD/tU) well represents an equilibrium CANDU-6 core. For the PCR evaluation, seven power levels are considered (65%, 75%, 85%, 95%, 100%, 105%, and 110%), and the ENDF/B-VII.0 library is used in the Monte Carlo calculations. In this work, the PCR is directly approximated using a linear interpolation of the calculated discrete reactivity at the seven power levels. Because the power coefficient of CANDU-6 is very small, the calculated PCR is sensitive to the statistical uncertainty of the Monte Carlo results. Therefore, the standard deviation of the k-inf values for the PCR evaluation has been reduced to ~0.6 pcm for the PCR evaluation using many more neutron histories (~6 billion) than in the previous FTC prediction. The calculational results are provided in Fig. 10.

It is clearly observed that the PCR value becomes less positive or more negative when the DBRC module is applied. This is mainly because of the enhanced Doppler effect of ^{238}U, that is, enhanced FTC due to the DBRC method. It is worthwhile to note that the PCR increases slowly with power level and it quickly increases when the power is above ~105%. The sudden increase of PCR is due to local coolant boiling in the exit of the coolant channel of CANDU-6. At 3.6 GWD/tU, the PCR predicted with the DBRC scheme is very close to zero at the full power condition, whereas it is likely to be more positive in the standard calculation. As expected from the FTC evaluation, PCRs at 3.9 GWD/tU are a little less negative or more positive than those at 3.6 GWD/tU, which is mainly ascribed to the unfavorable FTC change with burnup.

Table 3 – Fuel temperature coefficient at the operating temperature range for different nuclear libraries at 3.9 GWD/tU (Doppler broadening rejection correction method).

Fuel Temperature (K)	ENDF/B-VI.8	ENDF/B-VII.0	JENDL-4.0	JEFF-3.1.1
910.16	−0.189 (−0.101)[a]	−0.198 (−0.117)[a]	−0.178	−0.196
960.16	−0.144 (−0.053)[a]	−0.153 (−0.072)[a]	−0.133	−0.150
1,010.16	−0.101 (−0.007)[a]	−0.111 (−0.030)[a]	−0.093	−0.108

[a] Standard MCNPX method (without the Doppler broadening rejection correction method).

Table 4 – Fuel temperature coefficient at the operating temperature range for several burn points (ENDF/B-VII.0, Doppler broadening rejection correction method).

Fuel temperature (K)	0 GWD/tU	3.6 GWD/tU	3.9 GWD/tU	7.2 GWD/tU
910.16	−1.226	−0.242	−0.198	0.213
960.16	−1.197	−0.196	−0.153	0.261
1,010.16	−1.170	−0.154	−0.111	0.307

Table 5 – Fuel and coolant temperatures used in the power coefficient of reactivity calculations.

Power (%)	Fuel temperature (K)	Coolant temperature (K)
65	798	551
75	842	553
85	888	555
95	936	558
100	960	559
105	985	560
110	1,010	562

4. Conclusion

Based on a representative fuel lattice model of the CANDU-6 reactor, both the FTC and the PCR have been re-evaluated using the continuous-energy Monte Carlo MCNPX code, which was modified to take into account the Doppler-broadened elastic scattering resonance. From this study, the following conclusions are derived. At the fresh condition, the FTC of a CANDU-6 reactor is clearly negative and it becomes less negative with burnup, becoming positive at the discharge burnup of ~7,200 MWD/tU. The FTC can be noticeably enhanced by accounting for the thermal motion of ^{238}U in the elastic scattering reactions. At the mid-burnup condition (~3.6 GWD/tU) representing the equilibrium CANDU-6 core, the FTC turns out to be negative. Consequently, the resulting PCR is also noticeably improved when the scattering resonance Doppler broadening is correctly considered.

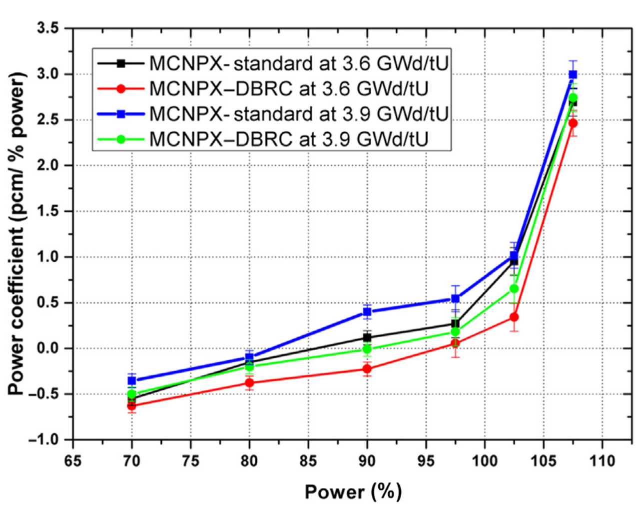

Fig. 10 – Power coefficient of reactivity as a function of power level.

The relatively simple lattice model analysis predicts that PCR will be negative below ~90% power and very close to zero at 100% power, but not always positive as reported. Thus, this work can be considered as a small contribution to a deeper and better understanding of the CANDU reactor core and its generic safety features. The new findings can be utilized to improve the future safety of the CANDU system. For a more concrete conclusion, a full three-dimensional core thermo-hydraulic-coupled neutronic analysis should be performed.

Conflicts of interest

All authors have no conflicts of interest to declare.

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