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Burnable Absorbers in CANDU Fuel Bundle Depletion with U_WB_1 Code

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Abstract

U_WB_1 nuclear fuel depletion code is being developed by Lovecký *et al* to conduct burnable neutron-absorber research for fast and thermal reactor designs. The use of neutron absorber in CANDU to gain operating margin was proposed by Chan *et al*. The development of U_WB_1 and the use of n-absorbers in CANDU were published in 2014. Research and development are still ongoing.

This paper describes the simulation of CANDU fuel bundle depletion. The accuracy and the speed of the code are compared to WIMS, Serpent and MCNP6 reference codes. The results show that U_WB_1 is suitable to be used as a depletion code to study the removal of the initial transient and suppression of the plutonium peaks in CANDU fuel reactivity. U_WB_1 code introduces an advantage in the depletion calculation time.

Keywords: U_WB_1, Nuclear Fuel, Depletion, Burnable Absorber.

1. Introduction

U_WB_1 nuclear fuel depletion code is being developed in [1] to conduct burnable neutron-absorber (BA) research for fast and thermal reactor designs. The use of BA in CANDU to gain operating margin was proposed in [2]. Research and development are still ongoing, both the code and the selection of BA for CANDU are currently being investigated.

U_WB_1 code has been in development since the end of 2012, and the current version (February 2015) presented in this paper is able to calculate CANDU fuel bundle depletion. The objective of the paper is to show that the accuracy of the code is sufficient to initialize the BA optimization study. Accuracy of the code is compared on two calculation cases (with and without BA) between U_WB_1 code and industry standard toolset WIMS, MCNP6 and Serpent. Their calculation times are compared. The speed of U_WB_1 code could be used in the BA optimization study that will comprise of two parts, pre-selection of BA materials or combination of materials with U_WB_1 code and recalculation of prospective designs with state-of-the-art codes. Optimal quantity of BA in the nuclear fuel could be determined to ensure a small loss in burnup.

The necessity to employ fast calculation tool for CANDU fuel bundle depletion is emphasized by large calculation time of CANDU depletion when compared to LWR. CANDU reactors operate

without BA, however, several designs were proposed, the most promising design for 37 element bundle is currently the combination of gadolinium and europium oxides [2]. Large degree of freedom is present in the CANDU optimization study since the fuel enrichment could also an important parameter.

The paper summarizes the main results presented in [3].

2. Calculation codes

Four different calculation codes are included in the comparison study. WIMS deterministic code is used as the reference code, because it is widely recognized by CANDU industry as a standard for safety analysis. The Monte Carlo code, MCNP6, is part of the study since it is theoretically one of the best probabilistic approaches in reactor physics calculations. Recent Finnish code, Serpent, was added to the study as the representative of new and fast calculation tool.

WIMS deterministic code [4] is able to model arbitrary 2-D geometry. The code version used for CANDU fuel depletion is known as WIMS-AECL, the methodology from original UK code is maintained by Chalk River Laboratories. The code benefits from analytical solution of transport equation along characteristic directions within a computational cell, the advantage of the method is distinctly lower calculation time when compared to discrete ordinates codes. The drawback is higher calculation time for 3-D spatial discretization. WIMS code was used with 89 energy group nuclear data library based on ENDF/B-VII.0 library.

U_WB_1 depletion code [1] is developed in Czech Republic as a part of the burnable absorber research. The code main features include burnup solver, Monte Carlo transport solver and 2sPC depletion scheme. Burnup solver handle Bateman equations with CRAM method for 3820 nuclides, Monte Carlo solver is used for neutron flux determination in 4308 energy group structure with neutron data library based on ENDF/B-VII.1 (423 nuclides). A two dimensional fuel cell is being modelled with arbitrary number of concentric cylinder regions, ray-tracing algorithm and uniform nuclear data library with 4308 energy groups are the main components of the solver. The two-step predictor-corrector method 2sPC couple burnup and transport solver in a scheme that use only three transport solver calling through the fuel depletion. The main assumption behind 2sPC scheme is that effective cross sections can be approximated from initial fuel state and final burnup fuel state with regression based on nuclide densities. The code development focused on the minimization of CPU usage at the expense of RAM demands.

MCNP6 Monte Carlo code [5] is a general purpose neutron transport code. The code is the result of merging MCNP5 and MCNPX codes. Large validation effort, general geometry definition, large number of physics treatment methods, tally determination, variance reduction methods and particle types are the features that make the code a recognized standard in Monte Carlo codes. General methods leads to the disadvantage of the code that is generally slower than other Monte Carlo codes. Latest ENDF/B-VII.1 nuclear data library in continuous energy format was used in the study.

Finnish code Serpent [6], [7] is a Monte Carlo code with development focus on reactor physics. The code runs significantly faster than MCNP6 code. The code was written in C language, other codes in this study are written in Fortran language. Standard surface-to-surface tracking is accelerated with Woodcock delta-tracking method, however, the efficiency of the method is expected to diminish when strong absorbers are present in the geometry. Serpent input files and nuclear data libraries are

similar to MCNP6. Specific uniform energy grid treatment for cross sections speeds up the calculation. The memory demands of current version 1.1.19 are significantly higher than for the rest of the codes in the study. In contrast to majority of fuel depletion codes, Serpent includes convenient definition of CANDU lattice type (concentric rings of fuel pins in a square lattice). ENDF/B-VII.0 nuclear data library was used in the study.

3. Calculation model

CANDU fuel bundle depletion was modelled by those codes identified in Section 2. Two dimensional model of a 37 element fuel bundle was investigated. Real geometry is shown in Figure 1, and a geometry used for modelling is depicted in Figure 2. Fuel pins in the rings were homogenized because of the limitations of U_WB_1 code that currently allows only concentric cylinders and square or triangular lattice to represent a basic fuel cell. Differences in neutron multiplication factor for heterogeneous and homogenized fuel bundle, less than 0.002, is negligible from fuel design point of view. The reason of U_WB_1 limitation is the focus on PWR fuel assembly that can be simplified into fuel cell that comprises of two annular regions (fuel, cladding) and surrounding moderator. Lattice of CANDU fuel bundle is very heterogeneous, therefore, homogenization was chosen as the best approach to simplify the geometry.

Two cases were compared in the study: natural uranium fuel bundle with and without the presence of burnable absorber. BA content of 150 mg Gd_2O_3 and 300 mg Eu_2O_3 per fuel bundle was chosen according to a preliminary study [2]. BA in placed in the CANLUB layer between fuel and its sheath. Because fuel sheath is almost transparent to neutrons, homogenization of CANLUB layer and fuel sheath was incorporated into fuel depletion model.

Very high absorption cross section of Gd-157 nuclide causes fast burning of gadolinium that is used for the compensation of initial excess reactivity, initial transient peak is removed. Europium with longer influence on fuel reactivity is used to suppress the plutonium peak. The total cross section for the thermal energy of 0.0253 eV for each of the burnable isotopes are: 60801 barns for Gd-155, 253929 barns for Gd-157, 9190 barns for Eu-151 and 367 barns for Eu-153.

The central fuel pin represents the first fuel ring, the 6 fuel pins in the inner ring are homogenized into a secondary fuel ring, the 12 fuel pins are homogenized into a tertiary ring, and the last 18 outer fuel pins are homogenized into a final, outer ring. Dimensions and materials are summarized in Table 1 and 2, respectively. Periodic boundary was preferred, when not applicable, mirror boundary condition was applied.

Fuel lattice depletion was performed with constant power 32.908 MW/MTU for 42 depletion time steps. Final calculation burnup slightly over 20000 MWd/MTU was achieved in 647 effective full power days.

All codes except the WIMS calculation were executed under RedHat OS with 2700 MHz frequency processor on the same computer in order to evaluate the speed of the codes.

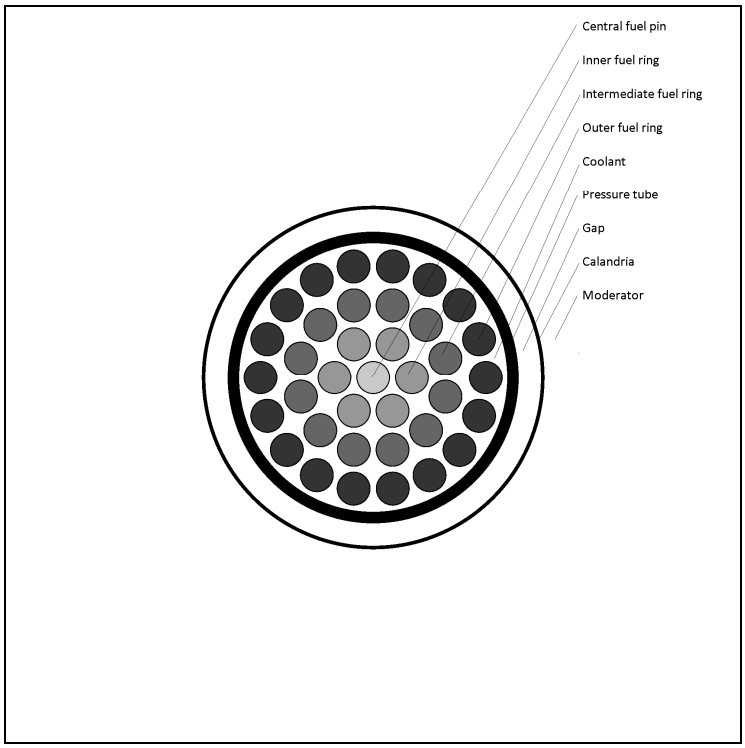

Figure 1 CANDU 37-element fuel bundle

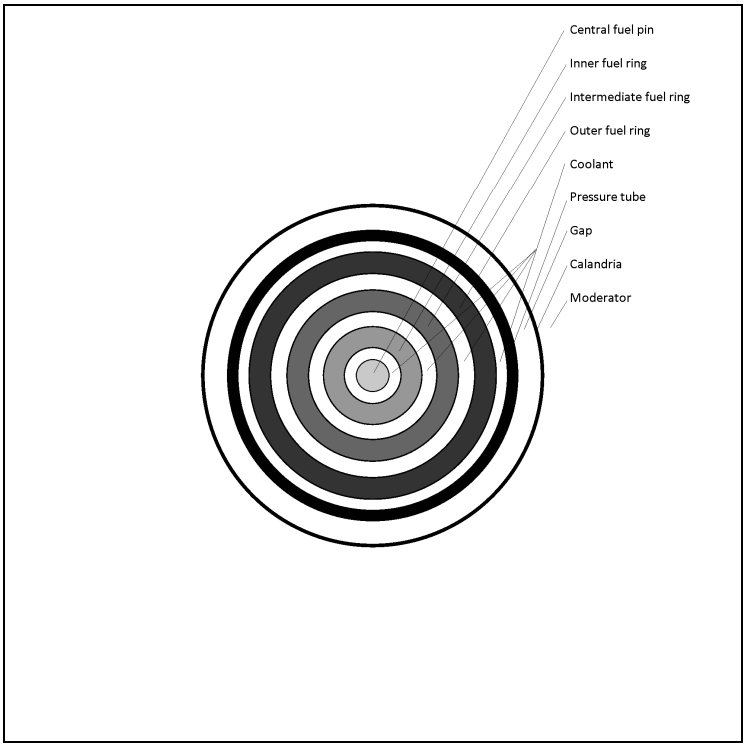

Figure 2 Homogenized CANDU fuel bundle

Region	Parameter	Value (cm)
Fuel	Outer radius	0.609
CANLUB layer	Outer radius	0.6096
Sheath	Outer radius	0.65046
Inner fuel ring	Pitch circle radius	1.4885
Intermediate fuel ring	Pitch circle radius	2.8755
Outer fuel ring	Pitch circle radius	4.3303
Coolant	Outer radius	5.15
Pressure tube	Outer radius	5.59308
Gap	Outer radius	6.4478
Calandria	Outer radius	6.5875
Moderator	Half pitch	14.2875

Table 1 Fuel bundle model dimensions

Region	Material	Density (g/cm^3)	Temperature (K)
Fuel	UO_2	10.5057	1011
Sheath	Zircaloy-4	6.3918	561
Coolant	Heavy water 99.11 wt%	0.8179	561
Pressure tube	Zr2.5Nb	6.5041	561
Gap	CO_2	0.001195	450
Calandria tube	Zircaloy-2	6.4003	339
Moderator	Heavy water 99.97 wt%	1.08699	339

Table 2 Fuel bundle model materials

4. Results

Comparison of neutron multiplication factor during fuel depletion is depicted in Figure 3 for standard CANDU fuel and in Figure 4 for CANDU fuel with burnable absorbers. It can be seen from the figures that the initial transient peak is removed as a consequence of adding burnable absorbers. The plutonium peak is not removed, however, it is reduced and appears at higher burnups. No residual suppression of reactivity is visible for the combination of gadolinium and europium oxides, and the neutron multiplication factor for fuel with BA converges to the case without BA. The first part of fuel depletion is characterized by plutonium accumulation and the creation of plutonium peak. U_WB_1 code gives the highest neutron multiplication factor values, while WIMS reports the lowest values. In the second part of fuel depletion, from plutonium peak to burnups around 10000 MWd/MTU, the lowest and highest estimates of neutron multiplication factor values are from U_WB_1 and WIMS code respectively. In the last part of fuel depletion, the two previously mentioned codes once again exchange the boundary behavior and the highest values are calculated by U_WB_1 code. MCNP6 and Serpent codes generally predict neutron multiplication factor between U_WB_1 and WIMS codes. Comparison of initial neutron multiplication factor is summarized in Table 3.

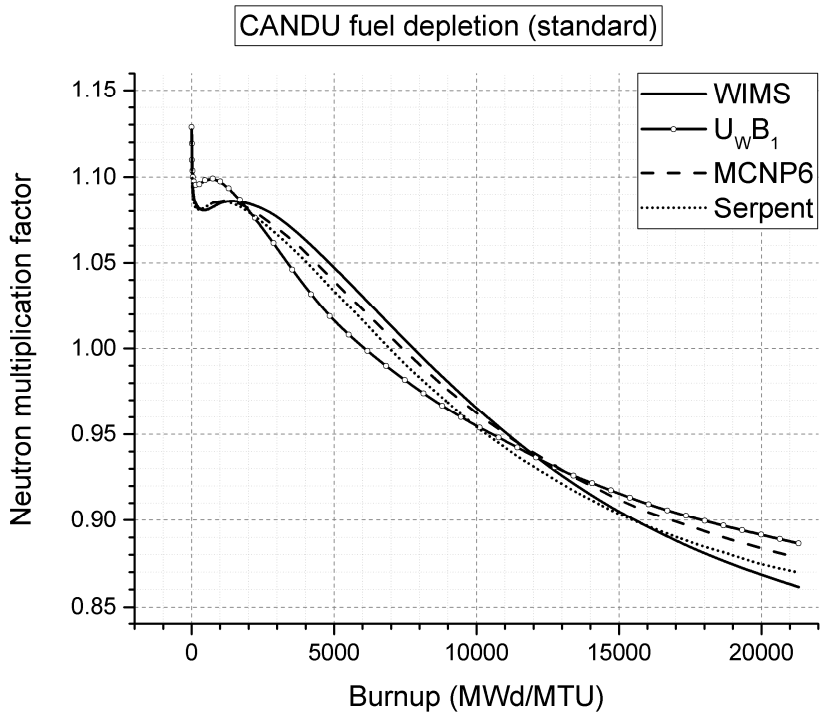

Figure 3 Standard CANDU - neutron multiplication factor during depletion

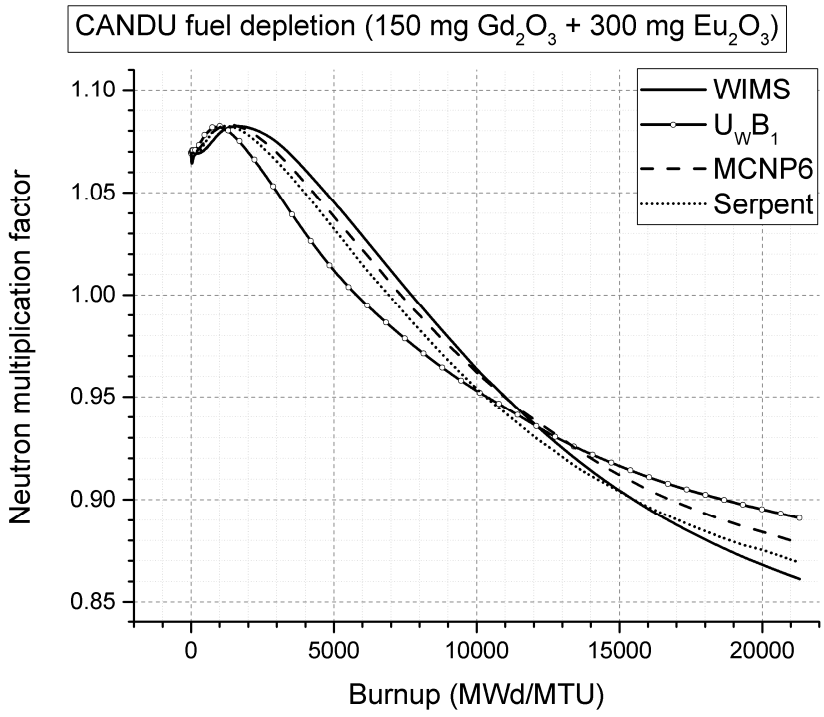

Figure 4 CANDU with BA - neutron multiplication factor during depletion

Code	Standard CANDU	CANDU with BA
WIMS	1.1263	1.0675
U_WB_1	1.1287 ± 0.0006	1.0695 ± 0.0006
MCNP6	1.1257 ± 0.0003	1.0693 ± 0.0003
Serpent	1.1271 ± 0.0002	1.0689 ± 0.0002

Table 3 Initial neutron multiplication factor values

Comparison of neutron multiplication factor near plutonium peak is shown in Figure 5 and Figure 6. Burnup on the horizontal axis was replaced by irradiation days and statistical uncertainty belt was added for Monte Carlo codes. The statistical uncertainty for two million active neutrons (the same number used for all statistical codes) per time depletion step is much lower than the differences between the codes. The difference between the highest and lowest neutron multiplication factors is 0.003. Difference between U_WB_1 and WIMS codes are around 0.002. Plutonium peak is predicted by WIMS at 42 irradiation days for standard CANDU fuel bundle and it is delayed by burnable absorbers by around 10 days. Other codes calculated plutonium peak at around 30 days, U_WB_1 code at only 22 days. However, the delay of plutonium peak around 10 days was calculated by all the codes, i.e. the impact of introducing burnable absorbers is predicted the same by all the codes.

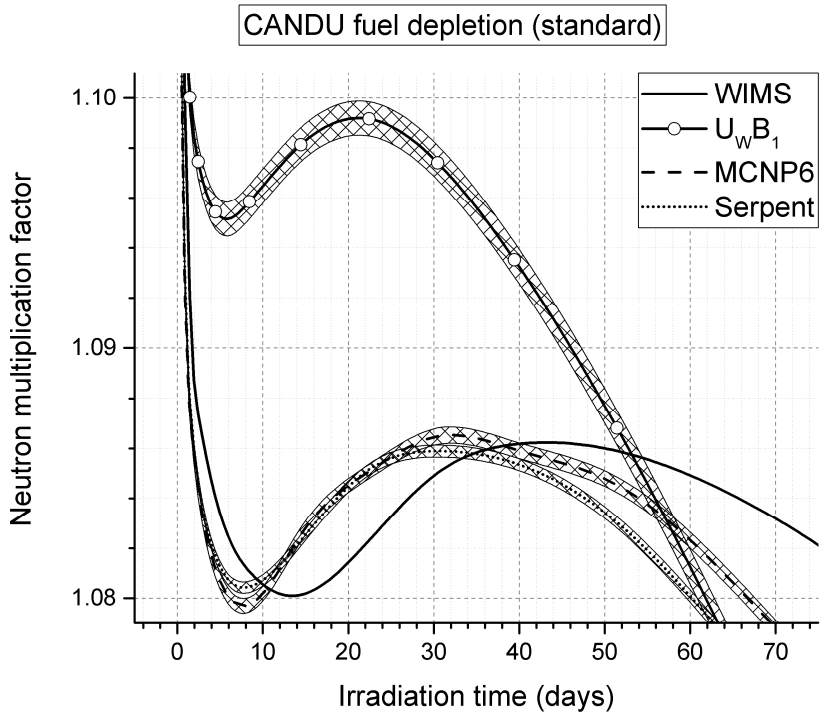

Figure 5 Standard CANDU - neutron multiplication factor near plutonium peak

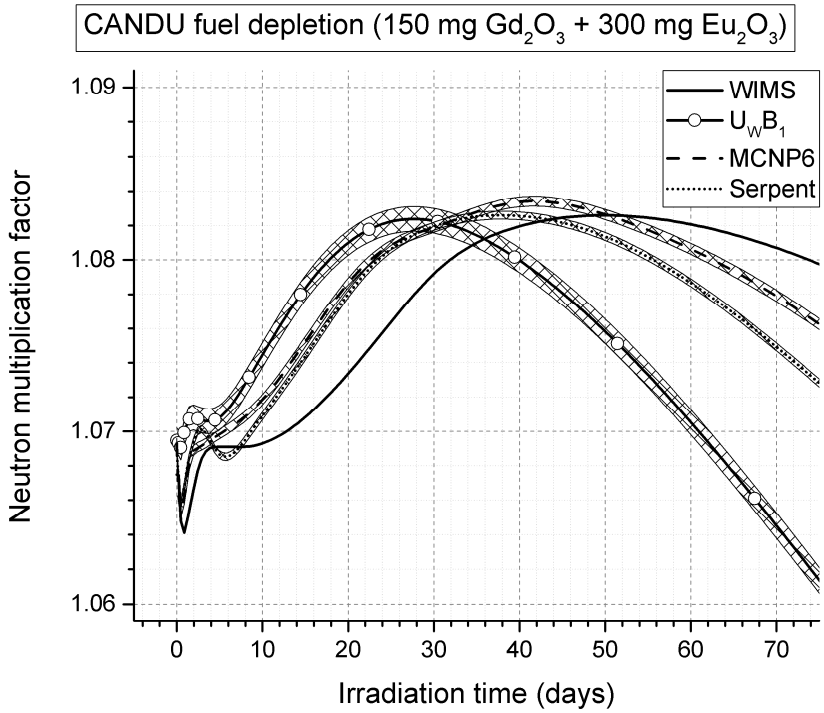

Figure 6 CANDU with BA - neutron multiplication factor near plutonium peak

Comparison of neutron multiplication factor between the codes in the means of ratio of neutron multiplication factors is depicted in Figure 7 for standard CANDU fuel and in Figure 8 for the case of CANDU with BA. Ratios were prepared with WIMS as a reference code. Statistical uncertainty error bars are included. Fuel depletion can be divided into three parts, consistently with the differences between the codes as described in neutron multiplication factor comparison. In the first and the last part, U_WB_1 values are the highest, in the second part of fuel depletion WIMS predicts the highest values. The typical difference between the codes around 0.01 reaches up to 0.03. The closest values to WIMS are generally achieved by MCNP6 code, the second closest is Serpent code, followed by U_WB_1 code. For each of the four codes in the comparison study, it is possible to locate a burnup interval where the code predicts the highest or the lowest neutron multiplication factor when compared to other codes. However, without experimental involvement, the true value is not known and probably lies between the predictions of the various codes.

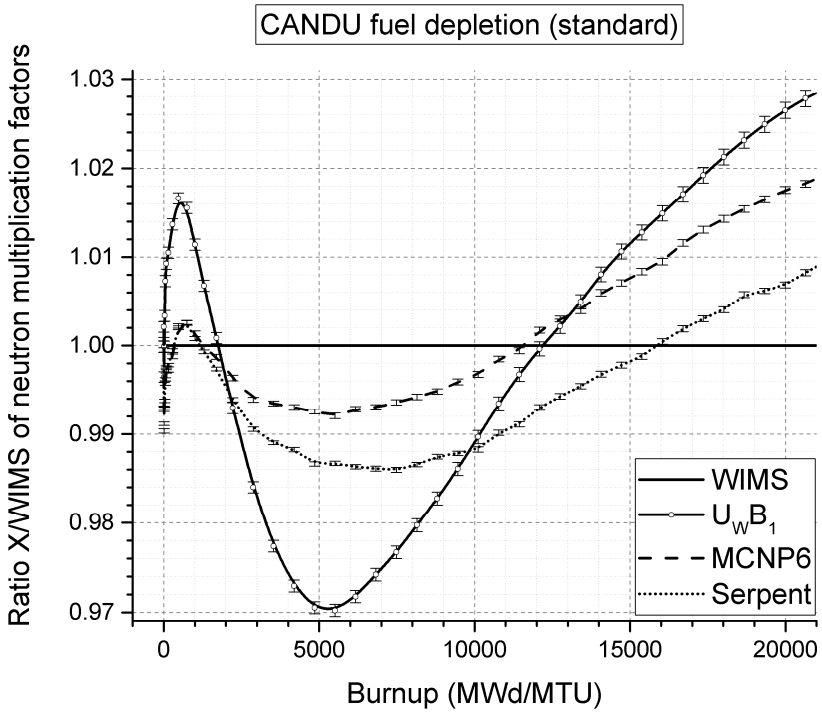

Figure 7 Standard CANDU - neutron multiplication factor comparison

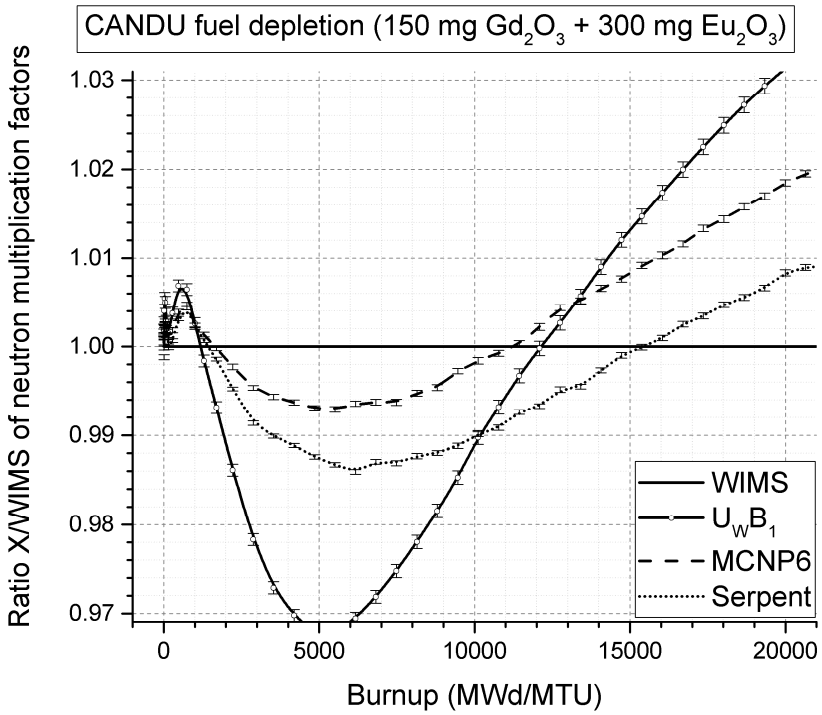

Figure 8 CANDU with BA - neutron multiplication factor comparison

The performance of the U_WB_1 code from the neutron multiplication point of view is comparable to other codes when fuel lattice is modeled. All the codes are marked by a degree of difference to the WIMS code, U_WB_1 code yield higher deviation that MCNP6 and Serpent codes.

The accuracy of the U_WB_1 code was expected to be lower, however, it is still acceptable for fuel bundle design calculations. The decreased accuracy is a result of the code development focus on the speed with balanced accuracy. The speed-up factor of one or two orders of magnitude is experienced with U_WB_1 code, see Fig. 9. A fast Monte Carlo solver and the 2sPC depletion scheme both share a comparable part of the speed-up factor. Fuel bundle depletion takes around 5 minutes with WIMS code, around 1.5 hour with U_WB_1, around 17 hours with Serpent and around 670 hours with MCNP6 code. Between the Monte Carlo codes, U_WB_1 is significantly faster than the other codes. WIMS deterministic calculation proved to be faster than statistical approach. It is expected that in the future U_WB_1 could take advantage from the reserves in both the method and the computer evolution. Monte Carlo solver can be parallelized and 3-D approach can be implemented. On the other hand, third dimension extension for WIMS code will slow down the calculations distinctly and parallelization of deterministic methods can prove to be not as efficient as Monte Carlo parallelization for fuel bundle geometry.

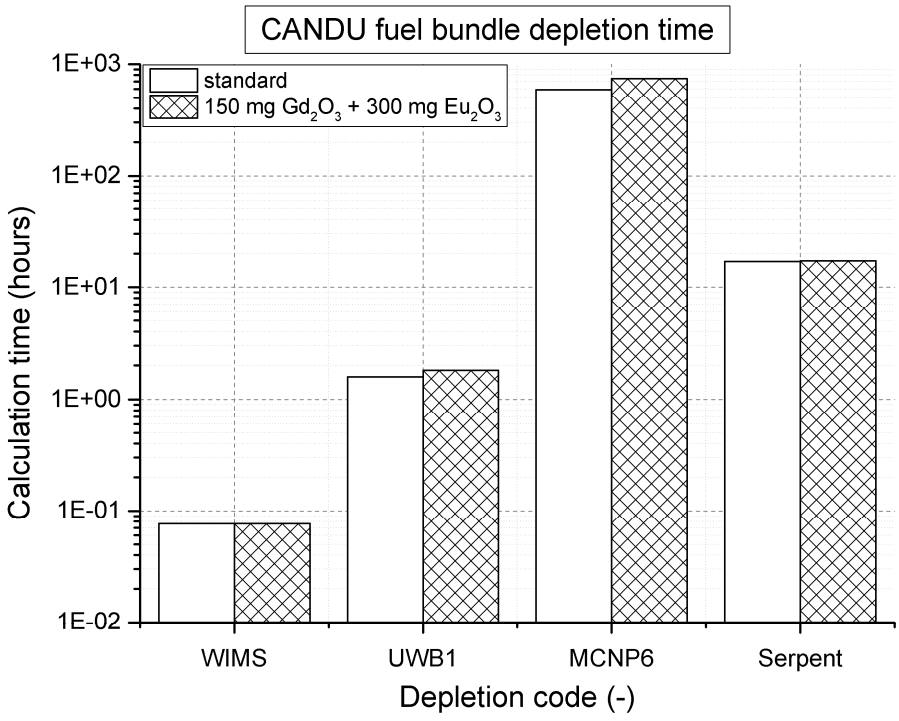

Figure 9 Calculation time comparison

5. Conclusions

The U_WB_1 depletion code was used in the comparison study with WIMS, MCNP6 and Serpent codes to model CANDU fuel bundle depletion. Two cases were compared in the study, natural uranium fuel with and without burnable absorber. Burnable absorber content of 150 mg Gd_2O_3 and 300 mg Eu_2O_3 per fuel bundle was chosen.

Fuel lattice model of CANDU nuclear fuel showed that the performance of the U_WB_1 code from the neutron multiplication point of view during fuel depletion is comparable to other codes. The comparison study concluded that all the codes are marked by a degree of difference to the WIMS code. Generally, U_WB_1 code yield higher deviation than that of MCNP6 and Serpent codes. The typical difference between the codes around 0.01 reaches up to 0.03 in neutron multiplication factor comparison. The best agreement with WIMS code is generally observed by MCNP6 code, the second closest code is Serpent code, followed by U_WB_1 code.

Among the compared codes, the lowest accuracy was both expected and observed for the U_WB_1 code, however, for fuel design calculations, it is still acceptable. The code development focused on the speed with balanced accuracy and it resulted in the decreased accuracy. A one or two orders of magnitude speed-up factor is achieved with U_WB_1 code. A comparable part of the speed-up factor is shared by both the fast Monte Carlo solver and the 2sPC depletion scheme.

6. Acknowledgements

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7. References

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