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A COMPARATIVE STUDY ON CANDU 6 REACTOR CRITICALITY THROUGH MCNPX AND SCALE 6.0 CODES

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ABSTRACT

The CANDU 6 is a nuclear reactor developed in Canada by Atomic Energy Canada Limited (AECL) which uses heavy water as both coolant and moderator, what results in a better neutron economy and allows natural uranium to be used as fuel. This nuclear system has been widely studied and evaluated making use of different computational codes that simulate nuclear systems estimating the neutronic parameters. In this sense, the present work aims to model the CANDU 6 reactor to ensure the reliability of the k_{eff} values obtained as well as to contribute to future studies regarding the CANDU 6 criticality. For this purpose, the MCNPX code was used to simulate the reactor for than to make comparisons and discussions about the criticality values found in multiple cases, based on previous results in the SCALE 6.0 code. For instance, different fuels compositions and reactivity devices configurations were analyzed.

1. INTRODUCTION

The CANDU (Canadian Deuterium Uranium) 6 reactor was developed in Canada by Atomic Energy Canada Limited (AECL) and has become one of the most successful nuclear reactors, widely used in several countries beyond Canada such as Argentina, China, Korea and Romania. Many factors have contributed to its acceptance worldwide, including one of the main features of CANDU 6 reactors which is the use of heavy water as both moderator and coolant. This use results in a better neutron economy and allows natural uranium to be used as fuel, therefore reducing fuel manufacturing costs. Another defining feature of the CANDU reactor is the horizontal calandria and pressure tubes, this allows the reactor to refuel online and may also reduce costs. With regard to safety, every nuclear system must include means of adjusting reactivity, and the CANDU 6 reactor has two emergency systems as well as four control devices which will be analyzed in the present work.

In order to study different types of reactors, geometries, and fuels, computational nuclear codes were developed and the CANDU 6 nuclear system [4] has been extensively studied and evaluated making use of different computational codes. Previous studies evaluated the criticality of the CANDU 6 reactor in the SCALE 6.0 code in multiple cases [1] and these results are now being compared to the k_{eff} values obtained using the MCNPX (Monte Carlo N-Particle Transport) code.

The objective of the present work is to evaluate the neutronic behavior of the CANDU 6 reactor [4] in these two different codes, in different core configurations. The first part of the present work aims to perform a reactivity device analysis, the second consists in examining the impact of the insertion of reprocessed uranium fuel channels, the third part aims to look into the consequences of the insertion of depleted uranium pins inside the reprocessed uranium fuel bundles. Finally, the last part is a study on a particular case considered optimal, with three different fuel bundle configurations.

2. METHODOLOGY

The standard CANDU 6 fuel bundle has 37 natural uranium fuel pins surrounded by the heavy water coolant, the coolant itself is surrounded by the pressure tube, then by the calandria tube and heavy water moderator as shown in Fig.1 and laterally in Fig.2. Each fuel channel has 12 horizontal fuel bundles and the core has 380 fuel channels as shown in a cross-sectional view in Fig.3.

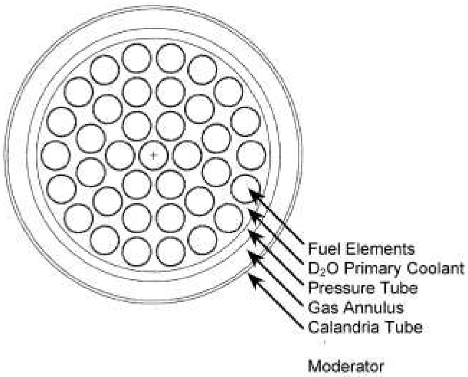

Fig.1. Cross-sectional view of the CANDU 6 Fuel Bundle [2].

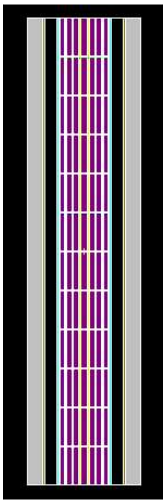

Fig.2. Lateral view [3].

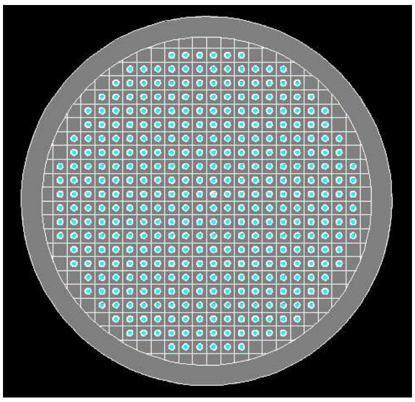

Fig. 3. Cross-sectional view of the CANDU 6 core [3].

The simulation of the CANDU 6 core [4] was performed using the geometry parameters described in Tab. 1. Both figures 2, 3 and 5 were obtained by the MCNPX geometry plot capability. All the simulations and data presented were obtained in stationary state by the MNCPX code using 50.000 neutrons during 200 active cycles and then compared to the SCALE 6.0 results presented in a previous work [1]. The cross-section library used in MCNPX was the ENDF/BVII continuous and the one used in SCALE 6.0 KENO-VI was the ENDF/BVII.0 collapsed in 238 energy groups [1].

Tab. 1. Geometric parameters of the CANDU 6 [3].

Parameter	**Value**
Fuel pins	37
Fuel bundles per channel	12
Fuel channels	380
Calandria diameter	759.4 cm
Subcalandria diameter	675.6 cm
Fuel channel length	594.05 cm
Fuel lattice pitch	28.575 cm
Calandria tube inner radius	6.4478 cm
Calandria tube outer radius	6.5875 cm
Pressure tube inner radius	5.1689 cm
Pressure tube outer radius	5.6032 cm
End Plate radius	5.1000 cm
End Plate thickness	0.17505 cm
Fuel Bundle length	49.5333 cm
Clad length	49.1832 cm
Clad radius	0.6522 cm
Pin radius	0.6013 cm
Ring 1 diameter – 6 pins	2.8960 cm
Ring 2 diameter – 12 pins	5.7510 cm
Ring 3 diameter – 18 pins	8.6610 cm

2.1 Reactivity Devices Analysis

In the CANDU 6 reactor [4], there are 4 ways of directly controlling the core reactivity, these devices are needed to maintain the reactor critical for normal operation, to allow power control and to permit fast reactor shutdown in an emergency condition. These devices are:

Liquid Zone Control Units: There are 14 ZCUs in the CANDU 6 reactor which are located in 6 vertically oriented tubes. In the two central tubes, there are three compartments and the four outer tubes are divided into two compartments each. The compartments can be filled with variable and controllable amounts of light water which contributes to absorption of neutrons.

Adjusters Rods: In the CANDU 6, there are 21 adjuster rods made by stainless steel and in these rods are fully inserted in the core during normal operation, contributing to optimize the neutron flux distribution.

Shutoff Rods: The CANDU 6 reactor has 28 shutoff rods made by cadmium clad with stainless steel. The function of these rods is to shut down the reactor in normal or accidental circumstances and its insertion is made through guide tubes made of a zirconium alloy.

Mechanical Control Absorbers: There are 4 MCAs in the CANDU 6 reactor, they are physically the same as the shutoff rods, made by cadmium clad with stainless steel and are normally parked outside the core.

There are also two emergency and independent shutoff systems that are not the focus of the present work and therefore were not modeled. The 4 types of control reactivity devices were simulated in the MCNPX code all with a cylindrical geometry with its respective poison/guides tubes materials and coordinates. Tab.2 shows the geometry parameters considered.

Tab.2. Geometric parameters of the CANDU 6 reactivity devices used in the simulation [3].

Reactivity Device	**ZCU**	**ADJ**	**SOR**	**MCA**
Poison Radius	5.00 cm	0.310 cm	3.500 cm	3.550 cm
Clad IR	-	3.607 cm	3.607 cm	3.607 cm
Clad OR	-	3.690 cm	3.725cm	3.690 cm
Guide tube	6.10 cm	4.519 cm	4.519 cm	4.519 cm
Guide tube	6.24 cm	4.572 cm	4.572 cm	4.572 cm

2.2 Insertion of Reprocessed Uranium (RF) Fuel Channels

The second part of the present work focused on the analysis of the reactivity behavior of the core as there were inserted fuel channels with reprocessed uranium by the OREOX [5] process as fuel instead of natural uranium, the core was without its reactivity devices and just the guide tubes were on place. The insertion was made with 4 new reprocessed uranium fuel channels each simulation (Fig. 4.) and the expected results was an increase in the effective multiplication factor.

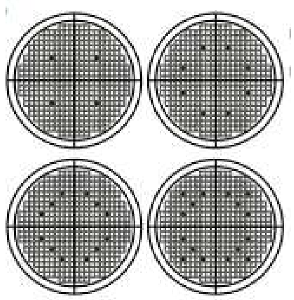

Fig. 4. Configurations of Reprocessed Uranium Channels simulated [1].

2.3 Insertion of Depleted Uranium pins inside the Reprocessed Uranium Fuel Bundles

Because of the fact that the reprocessed fuel has more fissile material than the natural uranium standard fuel, it is expected as increase in the effective in multiplication factor as seen in part 2. This increase in k_{eff}, however, can cause safety issues and therefore a poison material such as Depleted Uranium, which at times is a subproduct of fuel enrichment, can be used to balance this effect. Consequently, it is expected that adding 7 pins of depleted uranium will reduce the multiplication factor compared to the one obtained in the second part of this study. The third part of the present work aims to analyze the effect of introducing Depleted Uranium (Tab. 3.) in the 7 central pins of the Reprocessed Uranium (Fig. 5.) fuel bundles studied in the previous part.

Tab. 3. Depleted Uranium composition. [3]

Nuclide	Mass Fraction (%)
O-16	1.1847 E+01
U-234	8.6679 E-04
U-235	1.7410 E-01
U-238	8.7978 E+01

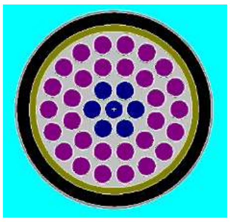

Fig. 5. Central pins with Depleted Uranium [3].

2.4 Optimal Case

In a previous work [1], there was a case analyzed which called attention for presenting the lowest reactivity coefficient as well as the highest content of reprocessed fuel. This case consists in 44 reprocessed fuel channels, 244 Natural Uranium with 4 pins of Depleted Uranium fuel channels and 92 standard Natural Uranium fuel channels, in a configuration showed in Fig. 6. This configuration was simulated and the results compared to the ones previously obtained.

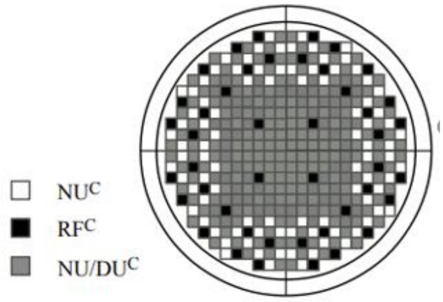

Fig. 6. Fuel Channels configuration of the “optimal case” [1].

3. RESULTS

3.1 Results of the Reactivity Devices Analysis

The simulations were made considering different configurations of reactivity devices: no devices, only ZCUs, just ADJs, only SOR, just MCAs and finally all the devices located into the core. The results of the effective multiplication factor of the core in both SCALE 6.0 KENO-VI and MNCPX in these configurations as well as the absolute difference between them were presented in Tab. 4.

Tab. 4. Results of the reactivity device analysis [3].

Configuration	SCALE 6.0 [1]	MCNPX	Difference
No device	1.01046	1.01547	0.00501
MCAs	1.00309	1.00674	0.00365
ADJs	0.99439	0.99997	0.00558
SORs	0.94340	0.94227	0.00113
ZCUs	1.00717	1.01261	0.00544
All devices	0.92488	0.91879	0.00609

As it can be seen in the third column of Tab. 3, the difference between the results is small, so it is possible to conclude that the results are similar as well as the neutronic behavior and the agreement was good enough. Beyond that, it can be noticed that the highest reactivity impact comes from the Shutoff Rod and most importantly that as more reactivity devices are added, the smaller is the effective multiplication factor.

3.2 Results of the Insertion of Reprocessed Uranium (RF) Channels

In the section 2.2 of the present work, the methodology behind the simulation of the core with Reprocessed Uranium was introduced as well as the awaited results. Firstly, it is expected that in each simulation the reactivity increases because of the addition of 4 channels filled with more fissile material than the Natural Uranium one. Furthermore, it is expected that the results of the SCALE 6.0 KENO-VI and the MCNPX codes are sufficiently close. For this, there are presented Tab. 5 with the simulation results and Fig. 7 which compares them graphically.

Tab. 5. Results of the RF Insertion Analysis [3].

Number of RF Channels	SCALE 6.0 [1]	MCNPX	Absolute Difference
0	1.01046	1.01547	0.00501
4	1.01459	1.01840	0.00381
8	1.01709	1.02041	0.00332
12	1.02098	1.02300	0.00202
16	1.02251	1.02464	0.00213

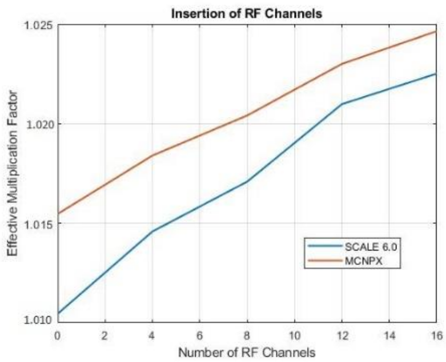

Fig. 7. Graphic of Number of RF Channels vs k_{eff} in both codes [3].

As shown in Tab.5 and in Fig. 7, the differences between the two codes were small and the k_{eff} increased as more Reprocessed Uranium fuel was added into the core, therefore, the results were considered reasonable and within what was expected.

3.3 Results of the Insertion of Depleted Uranium (DU) pins in the RF Bundles

The gradual addition of Reprocessed Fuel in the fuel channels is responsible for an increase in multiplication factor (Fig.7) and therefore it is desirable to have a counterpart to this growth. As a way to do it, the 7 central pins (Fig. 1) of each fuel bundle that was previously added to the core with only Reprocessed Fuel as fuel in sections 2.2 and 3.2, will now have Depleted Uranium. In conclusion, it is expected an increase in the reactivity as the fuel channels are being added, but the growth in this parameter will be smaller than the ones presented in the previous analysis. The results will again be presented in a table and in a graphic.

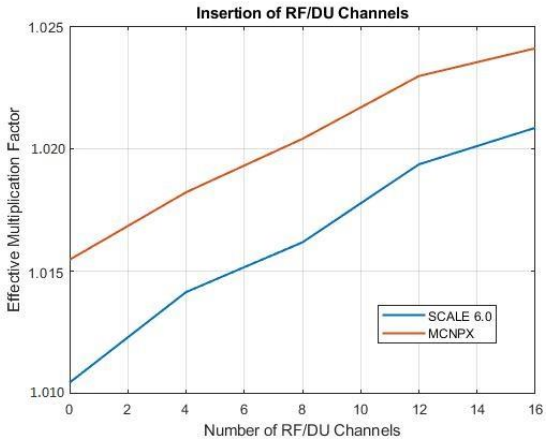

Fig. 8. Graphic of Number of RF/DU Channels vs k_{eff} in both codes [3].

Tab. 6. Results of the RF/DU Insertion Analysis [3].

Number of RF/DU Channels	SCALE 6.0 [1]	MCNPX	Absolute Difference
0	1.01046	1.01547	0.00501
4	1.01415	1.01822	0.00407
8	1.01618	1.02040	0.00422
12	1.01936	1.02296	0.00360
16	1.02085	1.02409	0.00324

As shown in Tab.6 and in Fig. 8, the results of the two codes were close and the k_{eff} increased as more RF/DU fuel was added into the core, therefore, the results were considered reasonable and within what was expected. Beyond that, there was a small decrease of the values when comparing Tab.5 and Tab.6, as expected.

3.4 Special Case Result

The case which results will be presented in this section is better described in section 2.4 and in a previous work [1]. The evaluation of multiple conditions led to the fact that the combination which provided more efficiency, safety and neutron economy involved 44 reprocessed fuel channels, 244 Natural Uranium with 4 pins of Depleted Uranium fuel channels and 92 standard Natural Uranium fuel channels, in a configuration showed in Fig. 6. The previous results in this case are expected to be similar to the MCNPX one.

Tab. 7. Optimal Case Results [3].

SCALE 6.0 [1]	MCNPX
1.01051	1.03030

Although the results presented in Tab. 7 were not as close as in previous parts, considering this first model, the high number of different materials, cross sections and computational complexity involved in it, the result difference of around 19mk was considered acceptable in this particular case. Future studies can be done in order to reduce this difference.

4. CONCLUSION

On balance, the comparison made between SCALE 6.0 and MCNPX showed a good agreement and the criticality behaved as it was initially expected in multiple conditions and configurations of the CANDU 6 core, therefore, the modelling of the reactor as well as the reliability of the k_{eff} values obtained previously could be well verified. Finally, the present work leaves space for further simulations and analysis on the criticality of the CANDU 6 reactor.

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