BJRS

BRAZILIAN JOURNAL
OF
RADIATION SCIENCES
08-03 (2020) 01-22

Neutronic evaluation of CANDU-6 core using reprocessed fuels

Silva[a] C. A. M., Velasquez[a] C. E., Almeida[a] M. C. B., Faria[a] R. B., Pereira[a] C.

[a]*Universidade Federal de Minas Gerais, Departamento de Engenharia Nuclear, Av. Antônio Carlos, 6627, Campus UFMG PCA 1 – Anexo Engenharia, Pampulha, 31270-901, Belo Horizonte, MG*

clarysson@nuclear.ufmg.br

ABSTRACT

The spent fuel from a PWR still contains some amount of fissile materials depending on their initial enrichment and the burnup. Thus, spent fuel from PWRs containing about 1.5% of fissile material could be used as fuel for CANDU reactors after some fission products are removed from it. Thus, an important proposal is the DUPIC cycle, where spent fuels from a PWR are packaged into a CANDU fuel bundle using mechanical reprocessing but without the need of chemical reprocessing. When it is refueled with reprocessed fuel, the reactivity of the system increases, and this behavior may affect the safety parameters of the reactor. Therefore, this work studies the neutronic parameters of two reprocessing fuel techniques: AIROX and OREOX, which are evaluated for two different cores configuration. The first one considers heavy water as a moderator and coolant. The second one considers heavy water and light water as moderator and coolant, respectively. These studies evaluate the core behavior based on the different number of reprocessed fuels channels and compare them with the reference core. To perform the simulation, MCNPX was used to calculate the effective multiplication factor, void reactivity coefficient, and neutron flux, which were evaluated at steady state condition for the different cases. The results show that the presence of parasitic absorbers in the reprocessed fuels hardens the neutron spectrum. This behavior provokes an increase in the core reactivity, and in the void reactivity coefficient. Among these parameters, the use of light water reduces the core reactivity but does not improve the void reactivity coefficient.

Keywords: CANDU reactor, DUPIC cycle, AIROX, OREOX, MCNPX 2.6.0

ISSN: 2319-0612
Accepted: 2020-07-24

1. INTRODUCTION

Nowadays, LWRs (Light Water Reactors) and HWRs (Heavy Water Reactors) represent a great share of the total nuclear power plants in the world, about 65% and 11%, respectively [1]. The CANDU (CANada Deuterium Uranium reactor) is an HWR type that uses natural uranium as fuel. On the other hand, LWRs use enriched uranium as fuel and after burnup have some percentage of fissile isotope which depends on the initial enrichment and the fuel burnup conditions. The development of the fuel cycle technology for recycling the SF (Spent Fuel) of nuclear reactors has particular importance to improve the uranium utilization, to reduce the high-level nuclear waste. Several studies verify the use of reprocessing LWR fuel into the CANDU fuel cycle [2-13].

Then, SF from LWRs could be used as fuel for CANDU, if some fission products are removed from it. Nonetheless, the most important process is the DUPIC (Direct Use of spent PWR fuel In CANDU), which reuses SF from a LWR into an HWR core, using a direct re-fabrication method without separating fissile materials. Among the reprocessing techniques there are two dry pyrochemical process proposed: AIROX (Atomics International Reduction Oxidation) and OREOX (Oxidation and REduction of OXide fuel). The AIROX is a precursor of the DUPIC concept, developed by Atomic International, by recycling LWR spent fuels retaining most of the fission-product inventory in reconstituted fuel assemblies. This method avoids the generation of high-level liquid waste streams, recycling the fertile ^{238}U and unburned fissile transuranics. The OREOX oxidation is performed at a temperature higher than the AIROX process, resulting in more effective removal of the volatile fission products. In fact, the OREOX is an improvement of AIROX process that present maximum flexibility in fuel design and satisfy the requirement of compatibility with existing CANDU system. Regardless of the two techniques, the DUPIC fuel cycle offers attractive possibilities for saving natural uranium fuel from CANDU and for a substantial reduction of spent fuel compared with the once-through cycle. The main purpose of the DUPIC fuel cycle is improving the utilization of uranium fuels while at the same time reducing the volume of radioactive waste.

This work evaluates the insertion of two reprocessed fuels, FRA (Fuel Reprocessed by AIROX technique) and FRO (Fuel Reprocessed by OREOX technique), into the CANDU-6 core. These fuels have a higher percent of fissile isotopes than natural uranium. Thus, their insertion into the core will increase the reactor reactivity, which may affect the safety core parameters. The CANDU uses heavy water as moderation/coolant, which has higher moderation ratio than light water. Thus, if H_2O is used as a coolant, the reactivity drops off to a level that could improve the safety core parameters using the FRA and FRO fuels.

Thereby, two core configurations were simulated: the first one considers the CANDU using D_2O as coolant and moderator. The second one evaluates the use of D_2O and H_2O as moderator and coolant, respectively. The goal is to study the neutronic behavior of core with different number of RFC (Reprocessed Fuel Channels) in the two configurations and compare them to the reference core loaded with FNU (Fresh Natural Uranium). This core was based on preceding works [13,14], which evaluates a hypothetical extreme condition considering all CANDU fuel channels with a fresh fuel bundle. Therefore, the FNU comprises a fuel bundle at zero burnup. The MCNPX 2.6.0 (Monte Carlo N-Particle eXtended - version 2.6.0) code is used in the simulations where the effective multiplication factor, void reactivity coefficient, and neutron flux were evaluated at steady state condition.

Since nuclear power plants can play an important role in future power generation and considering that spent fuel is a challenge for the nuclear industry, the current paper contributes to study the use of reprocessed fuels. The DUPIC cycle, considered in the simulations has highlights among the advanced fuel cycles due to its synergism and non-proliferation characteristics. Furthermore, the current work links the neutronic analysis of reprocessed fuels and light water as coolant in the CANDU core. These studies may contribute to future analyzes of advanced CANDU reactors which aims inherent safety and reduced capital cost construction.

2. METHODOLOGY

2.1. Configuration of CANDU-6 core

This work simulates the full-core reactor and considers the dimensions, geometric characteristics and composition of main structural components from previous studies [13-16]. Two models were simulated to evaluate the neutronic behavior due to different coolant types. The first one uses heavy water (D_2O) as a moderator and coolant. The second one uses heavy water as a moderator and light water (H_2O) as coolant. Table 1 shows the main features of the simulated cores and Table 2 presents the temperature of the principal components from the evaluated system.

Table 1: Main dimensions of simulated CANDU-6 core [14-17].

Reactor Zone	Description		Value (cm)
Core	Sub-*calandria* inner radius		3.378×10^2
	Main *calandria* inner radius		3.797×10^2
	Fuel channel length		5.944×10^2
	Lattice pitch		2.857×10^2
	Calandria tube inner radius		6.448
	Calandria tube outer radius		6.587
	Pressure tube inner radius		5.169
	Pressure tube outer radius		5.603
Fuel Bundle	End plate radius		5.100
	Fuel pin radius		6.103×10^{-1}
	Fuel pin height		4.820×10^2
	Clad radius		6.522×10^{-1}
	Clad height		4.953×10^1
	Fuel pins rings	Inner radius	1.448
		Middle radius	2.875
		Outer radius	4.330

Table 2: HFP temperature of the principal components.

Description	Temperature (K)
Uranium Dioxide Pellets	1.000×10^3
Zircaloy Fuel Sheath	1.000×10^3
Coolant	5.630×10^2
Moderator	3.430×10^2

The present work simulates the full-core reactor, which consists of 380 horizontal fuel channels, each one comprises of 12 bundles arranged in a square lattice. Figure 1 illustrates the radial and axial view of the reactor core, which is simulated at HFP (Hot Full Power).

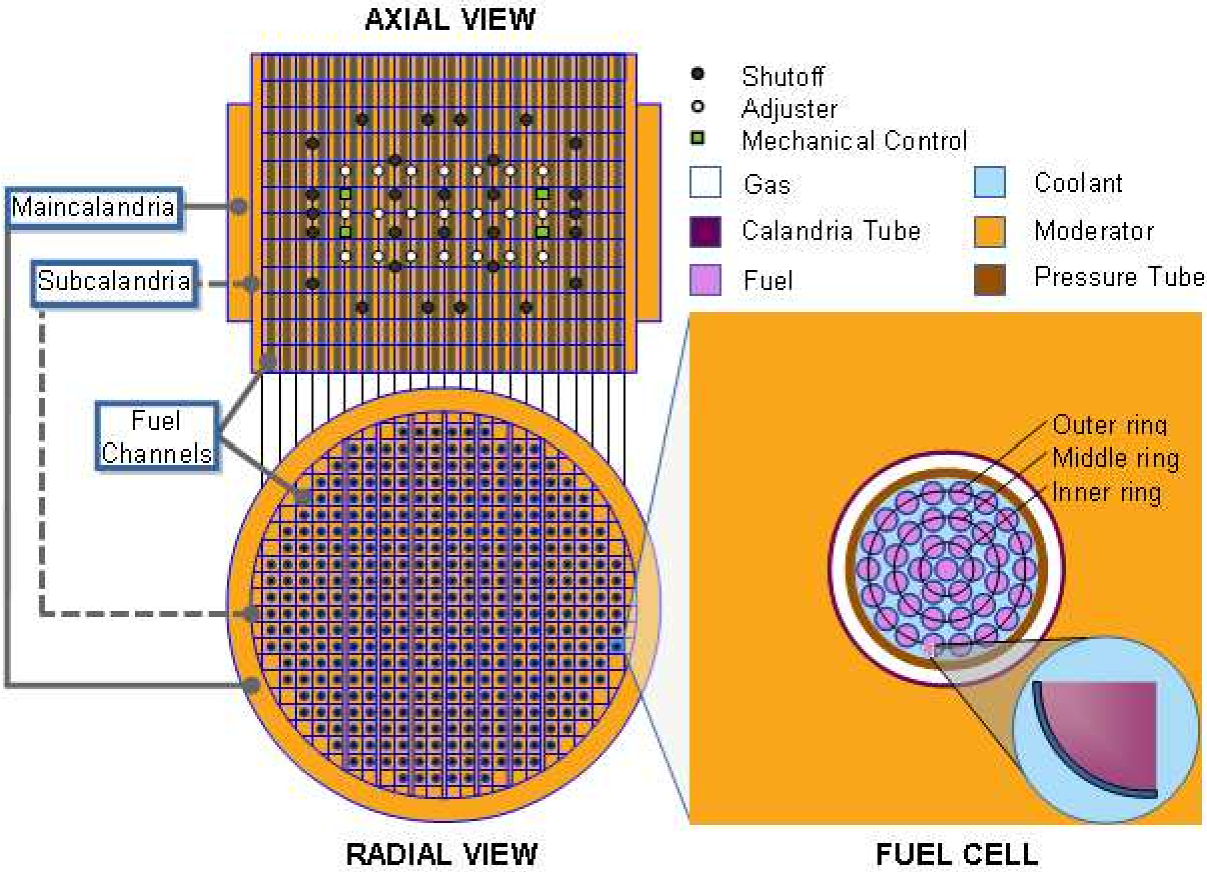

Figure 1: *Full-core CANDU-6 configuration.*

The fuel cell corresponds to a bundle column surrounded by a moderator (D_2O). Each fuel bundle has 37 fuel pins with zircaloy cladding, the bundle has a central pin surrounded by three rings of fuel pins and the coolant (D_2O or H_2O) flows between these fuel pins. Figure 2 illustrates the CANDU fuel bundle, which has several components. Due to the complexity of this bundle geometry, the simulations take into considerations the zircaloy fuel sheath, uranium dioxide pellets and the end plates. The end plate is represented by a homogenized disk consisting of 29% zircaloy and 71% heavy water by volume.

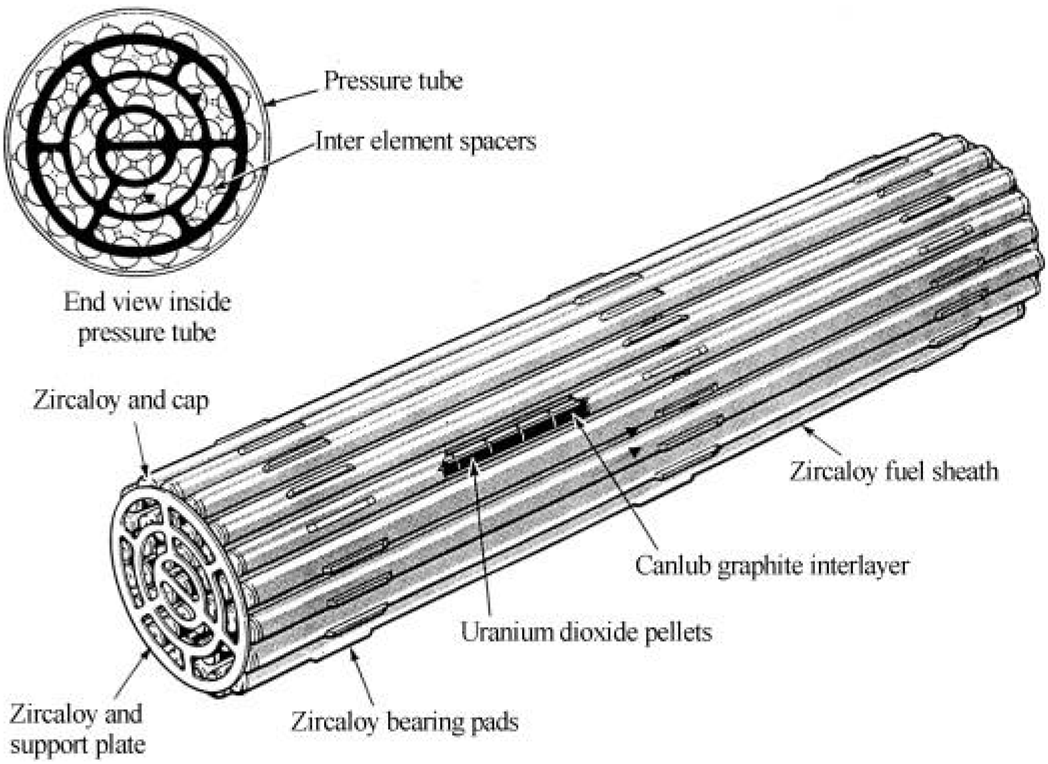

Figure 2: *Standard fuel bundle of CANDU-6.*
Source: Garland, W.J., 2014. The Essential CANDU.

The reactivity devices in CANDU reactors are the Reactor Regulating System (RRS) and two independent Shutdown Systems (SDS1 and SDS2). These devices comprise:

- RRS: 14 liquid zone controller compartments (ZCU) which can be filled with H_2O;
- RRS: 04 mechanical control absorber (MCA) rods;
- RRS: 21 adjuster (ADJ) rods;
- SDS1: 28 shutoff rods (SOR) which fall into the core from above; and
- SDS2: 06 poison-injection nozzles, horizontally oriented, that inject gadolinium nitrate ($Gd(NO_3)_3$) into the moderator in the event of SDS1 failure.

The simulated core comprises three types of reactivity control: SOR, ADJ and MCA rods, which are placed symmetrically on the axial and radial planes, as shown in Figure 1. The SORs and MCAs are tubes of cadmium enclosed by two concentric steel cylinders and inserted into guide tubes. The ADJs rods consist of a solid stainless-steel cylinder that is centered in a stainless-steel tube. All reactivity control rods are introduced into zirconium guide tubes, which are permanently located in the low-pressure moderator region. These guide tubes are situated interstitially between the calandria tubes where heavy water fills the gaps between the absorber material and the inner steel tube and between the inner steel tube and outer guide tube. Table 3 present the material of the control rods used in the simulations. The safety analysis of this work was focus on evaluating the behavior of the CANDU core under the insertions of different control rods without using the H_2O in the zone controller compartment or $Gd(NO_3)_3$ addition into moderator.

Table 3: Material of the reactivity control.

Component	SOR	ADJ	MCA
Absorber	Natural Cadmium	Stainless Steel 316	Natural Cadmium
Cladding	Stainless Steel 316	Stainless Steel 316	Stainless Steel 316
Guide tube	Natural Zirconium	Natural Zirconium	Natural Zirconium

2.2. Evaluated Fuels

As mentioned before, this works comprises three fuel types, FNU, FRA and FRO. The FNU comprises only fresh fuel bundles (zero burnup) while FRA and FRO contain reprocessed fuels which take into consideration the DUPIC cycle concept. In this cycle, after five cooling years of the Spent Fuel (SF) into the pool, it is shipped to a special plant for fuel bundles fabrication using dry processing, such as, AIROX or OREOX [12,18,19]. In this work, the composition of the spent fuel was obtained by previous studies [20], considering a typical PWR fuel with initial enrichment of 4.5% and a 50 GWd/MTU of burnup. A calculation was made to obtain the recovering fuel composition taking into consideration the percentage of fission product removal by AIROX [21] and OREOX [22]. The results comprise two reprocessed fuel that was used in the simulations namely: FRA (Fuel Reprocessed by AIROX technique) and FRO (Fuel Reprocessed by OREOX technique). Table 4, presents the isotopic composition of the simulated fuels, FRA, FRO and FNU (Fresh Natural Uranium).

Table 4: Weight fraction of simulated fuel types.

Isotope	FRA	FRO	Isotope	FRA	FRO	FNU
He^{4}	1.1995E-06	1.2007E-06	Sm^{150}	4.4670E-04	4.4718E-04	—
C^{14}	0.0000E+00	4.7355E-09	Sm^{151}	2.3929E-05	2.3954E-05	—
Cl^{36}	6.5925E-06	6.5995E-06	Eu^{151}	2.9740E-06	2.9772E-06	—
Ca^{41}	7.5074E-06	7.5154E-06	Sm^{152}	1.7605E-04	1.7624E-04	—
Ni^{59}	1.0798E-05	1.0810E-05	Eu^{153}	1.8570E-04	1.8590E-04	—
Y^{90}	1.4377E-07	1.4393E-07	Gd^{155}	1.5513E-07	1.5530E-07	—
Zr^{90}	2.4713E-04	2.4739E-04	Th^{230}	1.1431E-08	1.1444E-08	—
Zr^{93}	1.0838E-03	1.0850E-03	U^{233}	8.0252E-09	8.0338E-09	—
Nb^{93m}	6.0214E-09	6.0279E-09	U^{234}	2.4100E-04	2.4126E-04	5.4082E-05
Mo^{95}	1.0577E-03	2.1177E-04	U^{235}	1.0758E-02	1.0769E-02	7.1079E-03
Tc^{99}	1.2095E-03	1.2108E-03	U^{236}	6.5140E-03	6.5210E-03	—
Ru^{101}	1.2135E-04	2.4297E-04	U^{238}	9.5765E-01	9.5868E-01	9.9284E-01
Rh^{103}	6.5462E-04	6.5532E-04	Np^{237}	8.8084E-04	8.8178E-04	—
Pd^{107}	3.6255E-04	3.6294E-04	Pu^{238}	3.6255E-04	3.6294E-04	—
Ag^{109}	1.2125E-04	1.2138E-04	Pu^{239}	8.0352E-03	8.0439E-03	—
Sn^{126}	2.9660E-05	2.9691E-05	Pu^{240}	3.2012E-03	3.2047E-03	—
I^{129}	0.0000E+00	2.4246E-06	Pu^{241}	1.0275E-03	1.0286E-03	—
Cs^{133}	1.7022E-04	1.7040E-05	Pu^{242}	8.4324E-04	8.4414E-04	—
Cs^{135}	8.5621E-05	8.5713E-06	Am^{241}	1.1753E-03	1.1766E-03	—
Cs^{137}	1.3332E-04	1.3346E-05	Am^{242m}	2.1687E-06	2.1710E-06	—
Ba^{137}	5.5217E-04	5.5276E-04	Am^{243}	2.4954E-04	2.4981E-04	—
Nd^{143}	1.2196E-03	1.2209E-03	Cm^{242}	5.6625E-09	5.6685E-09	—
Nd^{145}	1.0104E-03	1.0115E-03	Cm^{245}	6.2597E-06	6.2664E-06	—
Sm^{147}	1.2678E-04	1.2692E-04	Cm^{246}	6.2536E-07	6.2604E-07	—
Sm^{149}	3.7683E-06	3.7723E-06	FM (%)	2.1877	2.1900	0.7108

In the present study, the reference CANDU core considers all channels loaded with fresh FNU bundle [13,14]. This methodology aims calculate the maximum criticality configuration using the traditional uranium fuel and to compare it with the insertion of reprocessed fuels. Therefore, the number of Reprocessed Fuel Channels (RFC) was increased gradually in the core, where the 12 fuel bundles into these channels have the same isotopic composition present in Table 4. Ten cases were simulated to each reprocessed fuel type, FRA and FRO. Table 5 presents the evaluated cases with the respective RFC number. Five of them use D_2O as the coolant and the others use H_2O.

Table 5: Number of RFC in the simulated cases.

Moderator	D_2O									
Coolant	D_2O						H_2O			
RFC Number	00	04	08	12	16	20	20	80	84	88	96
Fuel Type	FNU	FRA FRO	FRA FRO	FRA FRO	FRA FRO	FRA FRO	FRA FRO	FRA FRO	FRA FRO	FRA FRO	FRA FRO

Due to the reference CANDU model has all fuel channels with FNU, the neutron flux tends to be maximum in the central core region and decreases toward the boundaries. Radial flux (and power) flattening can be achieved by differential fueling, i.e. taking the fuel with a higher content of fissile nuclides in outer core regions than in inner core regions. Considering that reprocessed fuels has higher percentage of fissile isotopes than fresh FNU (see Table 4), the FRA and FRO were placed in the outer region of the core. This methodology aims to increase the fission number in this region to flat the core neutron flux. Figure 3 illustrates a radial view of the simulated cores where the white and the black cells represent the channels with FNU and reprocessed fuel respectively.

2.3. Evaluated Parameters

The MCNPX estimates the effective multiplication factor (k_{eff}) with the respective standard deviation (σ_{ST}) of the simulated model. This work evaluates the criticality and the neutron flux of the simulated system at steady state condition to FRA, FRO and FNU fuel types.

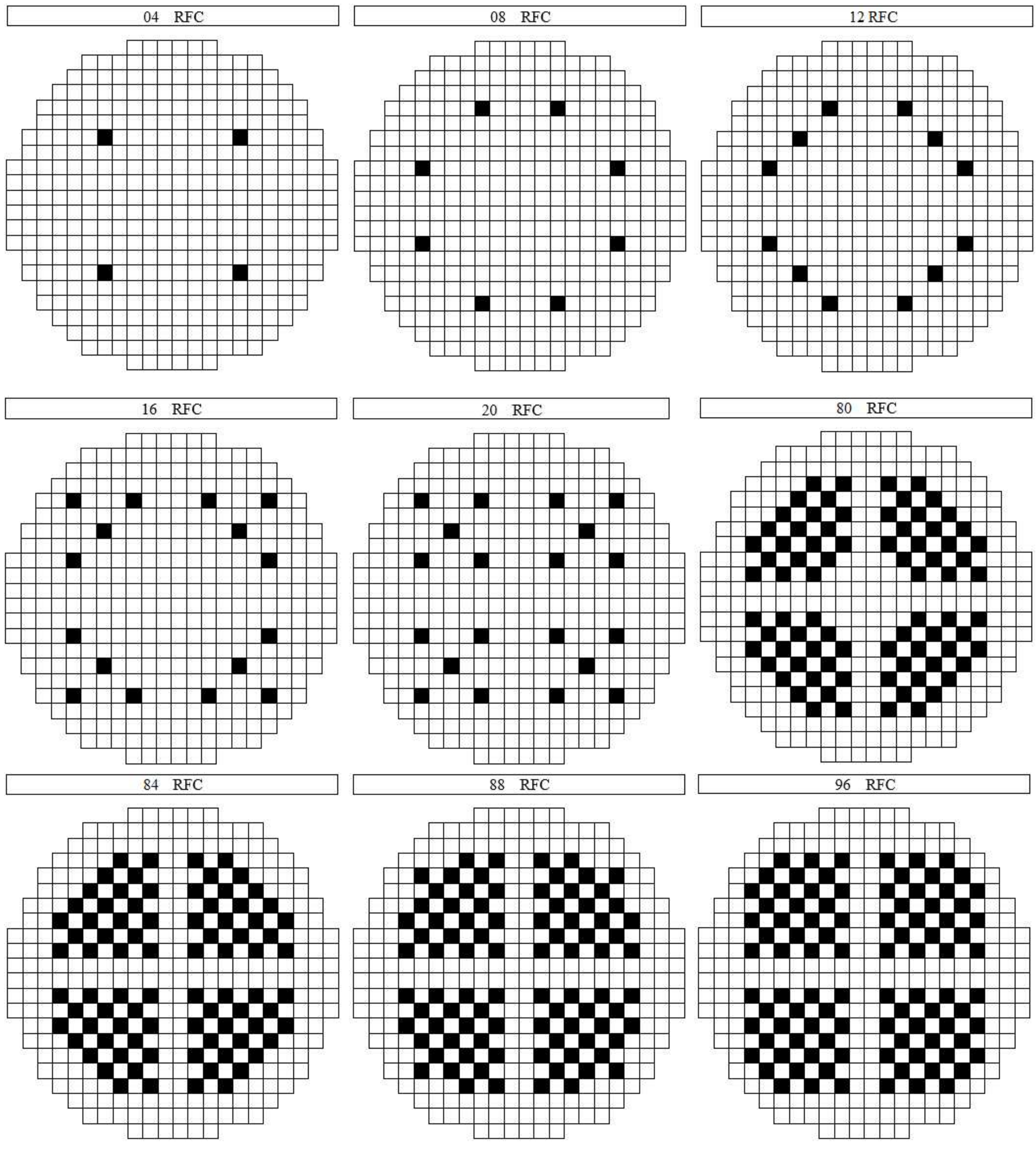

Figure 3: *Radial position of RFC (black cell) in simulated CANDU-6 core.*

The use of reprocessed fuel increases the percentage of fissile isotopes increasing the k_{eff} of the reactor. The device of reactivity control must assure a critical system. Thus, the effectiveness of the main regulating and shutdown system was evaluated. Although, the ADJs rods are fully inserted into the core, during normal operations [17], the simulations comprise the insertion and withdrawn of all rods aiming evaluates severe core conditions. First, the SORs, ADJs and MCAs were individually inserted in the core. After, all rods were inserted to evaluate the total negative reactivity contribution. In these studies, the k_{eff} variation (Δk) was calculated considering the rods withdraw and the rods fully-inserted into CANDU-6 core, through the equation:

$$\Delta k(mk) = (k_W - k_F) \cdot 10^3 \qquad (1)$$

The k_W and k_F are the respective k_{eff} when the control rods are withdrawn (*W*) and fully-inserted (*F*) in the reactor core.

The criticality analysis was considered at work temperatures, as shown in Table 2. However, the pre-defined library set of MCNPX 2.6.0 does not have the nuclear cross section for all the nuclides at different temperatures desired for representing the reprocessed fuels. Thus, nuclear data was downloaded from ENDF/BVII-1 (Evaluated Nuclear Data File) website [23], then processed with NJOY99 (Nuclear Data Processing System) [24] at the desired temperatures. After that, the data was added to the MCNPX 2.6.0 library.

In addition, in the criticality analysis, to study the influence of coolant density variation in the reactivity, the *CVR* (Coolant Void Reactivity) was calculated at the operating temperature condition (Table 2). It is important to know this effect in the reactor safety analysis that was calculated by:

$$CVR(mk) = \left[\frac{1}{k_S} - \frac{1}{k_P}\right] \cdot 10^3 \qquad (2)$$

where k_S and k_P is the k_{eff} calculated at standard (*S*) and perturbed (*P*) coolant condition respectively. For simulate an extreme core condition, the CVR was calculated with all control rods withdrawn of the core. In the simulations, the k_P was calculated by changing the coolant composition and density from the nominal value to the perturbed condition. Therefore, different values of *VF* (Void Fraction) were considered. The density variation was calculated as:

$$d_P = d_S \cdot (1 - VF) \qquad (3)$$

where d_S and d_P are the standard and perturbed coolant densities; *VF* is the void fraction.

In order to estimate the neutron flux inside the fuel cells that provides the neutron flux profile, a squared mesh was configured into MCNPX model of CANDU core. This mesh tally particles on each squared cell independent on the geometry problem [25]. In the model, there are 12 axial planes with 22 x 22 squared cells which have the same dimensions of the fuel cells. The absolute value of flux does not match the actual neutron source of the reactor. Thus, it is necessary to normalize the flux values initially calculated by MCNPX. In the simulations, this normalization was performed using the following equation [25]:

$$\phi_N = \phi_{MCNPX} \cdot \left[\frac{P \cdot \nu}{Q \cdot k_{eff}} \right] \tag{4}$$

where ϕ_N is the normalized flux; ϕ_{MCNPX} is the flux estimated by MCNPX; P is the reactor power level; ν is the average number of fission neutrons and Q is the recoverable energy per fission event. The values of ν, Q and k_{eff} are calculated by the MCNPX. The power level is defined at P = 2158.5 MW which is the fission power of the CANDU-6 [16,17].

3. RESULTS AND DISCUSSION

The simulations were performed in MCNPX code using 50,000 particles with 200 active cycles which results in low standard deviation of the k_{eff} around 2×10^{-4}. For the neutron flux calculations, the maximum relative error estimated by the code is around 4% and according to the MCNPX manual, results with errors less than 10% are reliable [25].

Both reprocessed fuel types (FRA and FRO) with the same RFC number, present similar k_{eff} values because these fuels present similar isotopic composition. Table 6 shows their k_{eff} values for the evaluated cases. The k_{eff} increase as the augment of RFC owing to reprocessed fuels have a higher percentage of fissile material than the natural uranium (Table 4).

Similarly, considering the same RFC number (i.e. 20) for two different coolant types (D_2O and H_2O), the D_2O presents higher k_{eff} than H_2O, because the heavy water has greater moderation ratio than light water. Under steady state conditions, the use of H_2O provides a negative reactivity in the core enabling the increase on the number of RFC to reach the same reactivity than D_2O. If compared with the reference core (FNU), the increase of k_{eff} due to the insertion of reprocessed fuels, leads to increase in the number of control devices on the core.

Table 6: k_{eff} to reprocessed fuel channels.

Moderator	Coolant	RFC Number	Fuel Type	Rods Withdraw	Type of inserted rod			
					MCA	ADJ	SOR	ALL
D_2O	D_2O	00	FNU	1.01139	1.00263	0.99624	0.93721	0.91760
		04	FRA	1.01421	1.00555	0.99926	0.94118	0.92095
			FRO	1.01452	1.00571	0.99865	0.94170	0.92069
		08	FRA	1.01682	1.00747	1.00166	0.94438	0.92369
			FRO	1.01679	1.00754	1.00160	0.94409	0.92340
		12	FRA	1.01968	1.01025	1.00469	0.94808	0.92674
			FRO	1.01977	1.01030	1.00456	0.94799	0.92713
		16	FRA	1.02094	1.01085	1.00572	0.94965	0.92820
			FRO	1.02079	1.01124	1.00611	0.94981	0.92864
		20	FRA	1.02564	1.01644	1.01022	0.95333	0.93245
			FRO	1.02530	1.01654	1.01015	0.95380	0.93315
D_2O	H_2O	20	FRA	0.94453	0.93719	0.93141	0.88625	0.86869
			FRO	0.94499	0.93733	0.93169	0.88618	0.86904
		80	FRA	1.00775	0.99898	0.99550	0.95379	0.93535
			FRO	1.00860	1.00000	0.99608	0.95480	0.93648
		84	FRA	1.01168	1.00404	0.99918	0.95667	0.93959
			FRO	1.01270	1.00518	0.99988	0.95723	0.93979
		88	FRA	1.01285	1.00470	1.00041	0.95825	0.94036
			FRO	1.01377	1.00569	1.00131	0.95915	0.94113
		96	FRA	1.01510	1.00647	1.00301	0.96121	0.94297
			FRO	1.01598	1.00765	1.00426	0.96207	0.94391

Table 7 presents the variation of k_{eff} (Δk) for each type of device control inserted in the core: shutoff (SOR), adjuster (ADJ), mechanical (MCA) and all rods (ALL).

Table 7: Variation of the effective multiplication factor (Δk).

Moderator	Coolant	Number RFC	Fuel Type	Type of inserted rod			
				MCA	ADJ	SOR	ALL
D_2O	D_2O	00	FNU	8.76	15.15	74.18	93.79
		04	FRA	8.66	14.95	73.03	93.26
			FRO	8.81	15.87	72.82	93.83
		08	FRA	9.35	15.16	72.44	93.13
			FRO	9.25	15.19	72.70	93.39
		12	FRA	9.43	14.99	71.60	92.94
			FRO	9.47	15.21	71.78	92.64
		16	FRA	10.09	15.22	71.29	92.74
			FRO	9.55	14.68	70.98	92.15
		20	FRA	9.20	15.42	72.31	93.19
			FRO	8.76	15.15	71.50	92.15
D_2O	H_2O	20	FRA	7.34	13.12	58.28	75.84
			FRO	7.66	13.30	58.81	75.95
		80	FRA	8.77	12.25	53.96	72.40
			FRO	8.60	12.52	53.80	72.12
		84	FRA	7.64	12.50	55.01	72.09
			FRO	7.52	12.82	55.47	72.91
		88	FRA	8.15	12.44	54.60	72.49
			FRO	8.08	12.46	54.62	72.64
		96	FRA	8.63	12.09	53.89	72.13
			FRO	8.33	11.72	53.91	72.07

In Table 7, the Δk was calculated by Equation (1) and using Table 6 values, where the largest Δk represents the high effectiveness of the control rods. The function of SOR rods in the reactor core is to provide a large amount of negative reactivity for reactor shutdown. Then, among the cases, the insertion of SOR generates more negative reactivity than MCA and ADJ because the different isotopic composition of the control rods (Table 3) and owing to the different number of these rods. The absorber material of SOR and MCA is cadmium, while for the ADJ is stainless-steel. As the neutron absorption cross section from cadmium is higher than the one for stainless-steel, the insertion of SORs causes a greater reduction of k_{eff} than MACs rods. Moreover, in the CANDU core, the number of SOR is higher than the number of MCA and ADJ (Figure 1).

In the simulations, the cases with 04, 08, 80 and 84 RFC number, present similar k_{eff} to case FNU (Table 6). In fact, the insertion of reprocessed fuels increases the quantity of transuranic elements in the core hardening the neutron spectrum due to the presence of absorbers and Pu on the reprocessed fuels [26, 27]. Therefore, as the RFC increases, the concentration of parasitic absorbers and Pu augment inducing an increase of the α_{TF}.

Figures 4 and 5 present the flux profiles for thermal neutrons on the reactor core for the different cases. As expected, these flux profiles are similar between the FRA and FRO owing to the isotopic composition comes from the same reprocessed fuels. Considering the same RFC number, the largest difference in the thermal neutron flux between FRA and FRO is about 6.6 %.

Since the neutron flux is not homogeneous in the reactor core, its distribution or shape is very important because it will determine the distribution of power generated in the core. When the FNU is used the flux has a maximum at the centre of the core and reduces toward the boundaries. The plant's operating license typically puts limits not only on the total reactor power, but also on the maximum channel and bundle powers. The restrictions may not be possible with a peaked flux shape to generate the reactor power desired without exceeding the licensed maximum channel and bundle powers. However, the thermal neutron flux reduces when reprocessed fuels are inserted into the core tending to flatten it as the RFC number increase (Figure 4 and 5). This flattening provides better power distribution through the core, because the peak flux in the central region is reduced and more channels can generate more power. For maximum power output from a given reactor, it is desirable that each fuel bundle contribute equally to the total power output. This behaviour may reduce the reactor capital cost, by reducing the number of channels required to produce the same power with safe operation.

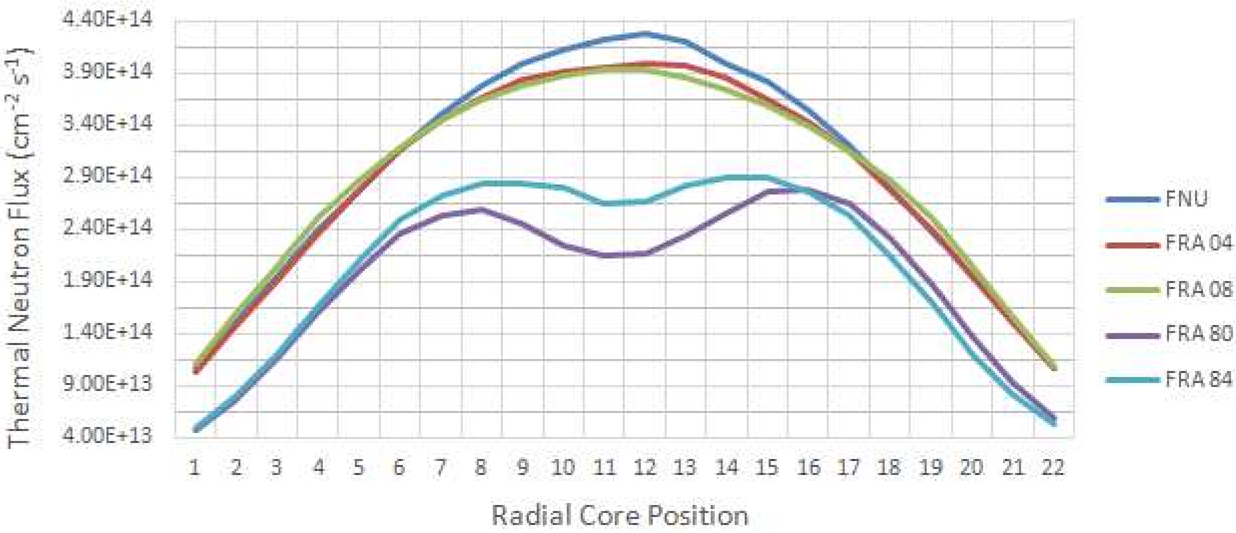

Figure 4: *Thermal neutron flux in central plane of reactor to FRA cases.*

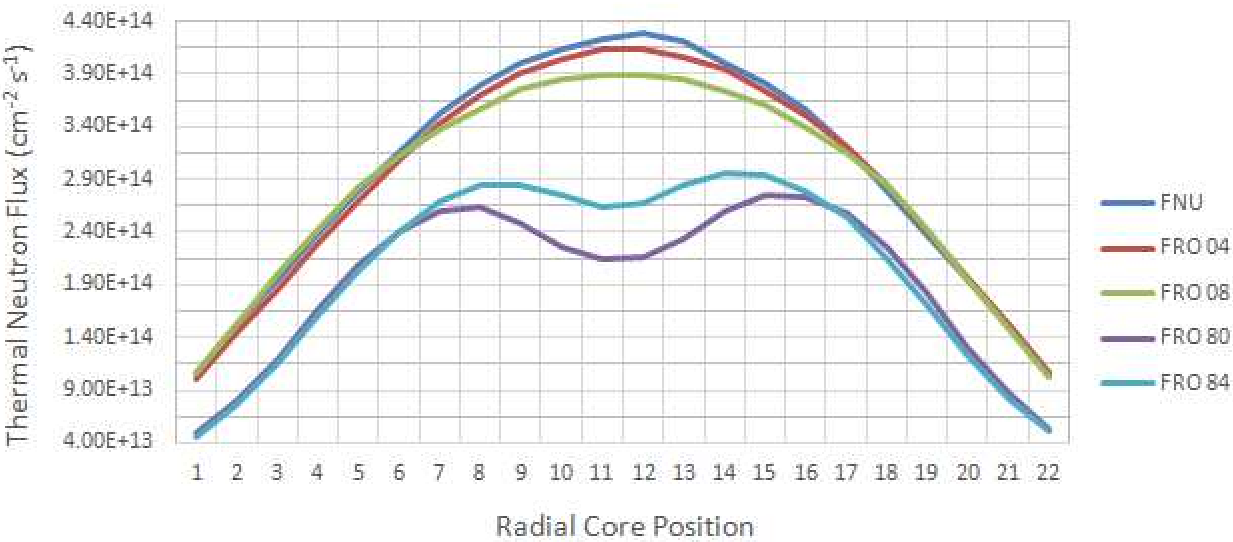

Figure 5: *Thermal neutron flux in central plane of reactor to FRO cases.*

To evaluate the *CVR* parameter, the reactor core was simulated with different coolant densities to calculate the k_{eff} due to this perturbed physical condition. Thus, MCNPX calculated the k_{eff} and the *CVR* values were determined by Equation (2). Table 8, presents the k_{eff} for different coolant *VF* and Figure 6 shows the behaviour of *CVR* as a function of *VF*. All cases present positive *CVR* that increase as *VF* augment, this is a typical behaviour of CANDU core. As known, the coolant is inside the pressure tubes while the moderator surrounds these tubes inside the *calandria*. The coolant and the moderator are placed in different regions in the core and a reduction in the coolant density does not affect the moderator density. Then, as the coolant density decrease, the neutron flux inside the pressure tube tends to harden and the fast fission probability increases generating a positive *CVR*.

Table 8: k_{eff} as a function of void fraction (*VF*) values.

Moderator		D_2O				D_2O			
Coolant		D_2O				H_2O			
RFC	0	4		8		80		84	
Fuel Type	FNU	FRA	FRO	FRA	FRO	FRA	FRO	FRA	FRO
0.0	1.01139	1.01421	1.01452	1.01682	1.01679	1.00775	1.00860	1.01168	1.01270
0.1	1.01317	1.01648	1.01638	1.02529	1.02551	1.01265	1.01335	1.01689	1.01774
0.2	1.01517	1.01804	1.01816	1.02771	1.02729	1.01786	1.01861	1.02179	1.02273
0.3	1.01642	1.02037	1.01951	1.02953	1.02979	1.02372	1.02463	1.02718	1.02861
0.4	1.01875	1.02162	1.02163	1.03188	1.03154	1.02938	1.03030	1.03329	1.03387
VF 0.5	1.02049	1.02384	1.02346	1.03390	1.03385	1.03590	1.03700	1.04024	1.04089
0.6	1.02252	1.02517	1.02554	1.03594	1.03611	1.04342	1.04456	1.04691	1.04792
0.7	1.02435	1.02715	1.02741	1.03809	1.03872	1.05136	1.05237	1.05504	1.05575
0.8	1.02666	1.02959	1.02991	1.04095	1.04076	1.06052	1.06150	1.06442	1.06541
0.9	1.02879	1.02899	1.02912	1.04354	1.04338	1.07143	1.07271	1.07524	1.07641
1.0	1.03100	1.03382	1.03409	1.04558	1.04550	1.08505	1.08555	1.08842	1.08923

The FNU case present higher *CVR* than values cited in some references (e.g. [17, 27-31]), because the FNU case considers all fuel channels loaded with fresh natural uranium. These core conditions contribute to *CVR* increases. Moreover, this parameter was evaluated with all control rods withdrawn from the core, as explained in methodology. In normal CANDU operation the Adjuster Rods (ADJ) are fully inserted in the core [17], which contributes to *CVR* reduction.

In the simulations, the insertion of the reprocessed fuels increases the *CVR*, i.e., as the *RFC* number increase, the *CVR* augment (Figure 6). The reprocessed fuel has several fissionable isotopes that are not present in the uranium fuel (Table 4). The hardening neutron spectrum due to the coolant density reduction may produce more fast fission reaction in reprocessed fuel than uranium fuel. Then, in void fraction conditions, the reprocessed fuels present higher core reactivity than traditional fuel (Table 8).

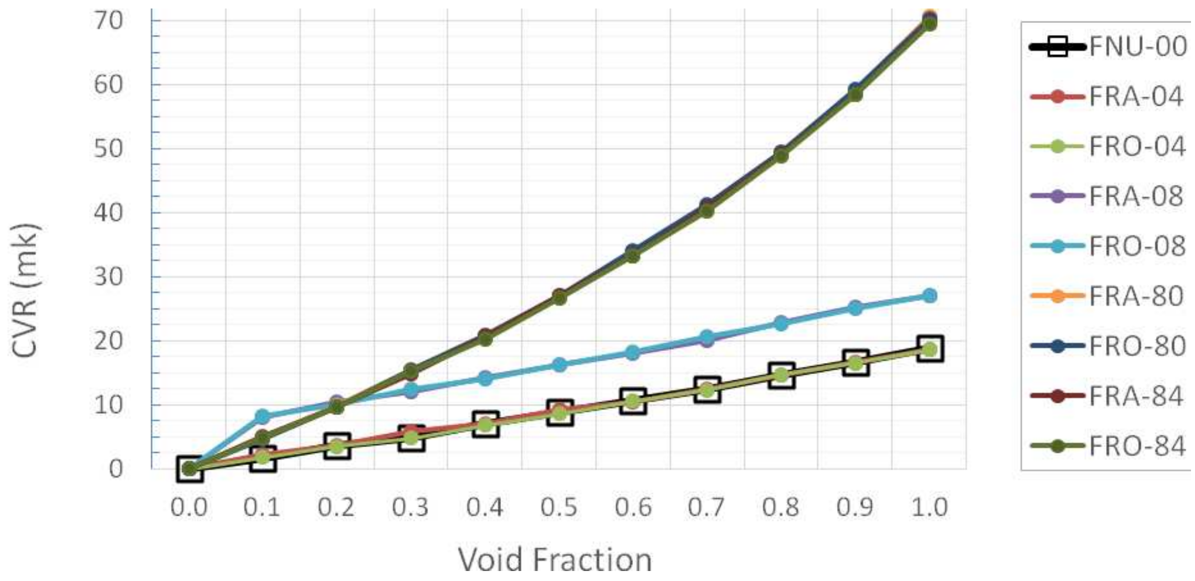

Figure 6: *Coolant Void Reactivity to void fraction values.*

In the simulations, the insertion of the reprocessed fuels increases the *CVR* values, i.e., as the RFC number increase, the *CVR* augment (Figure 6). The reprocessed fuel has several fissionable isotopes that are not present in the uranium fuel (Table 4). The hardening neutron spectrum due to the coolant density reduction may produce more fast fission reaction in reprocessed fuel than uranium fuel. Then, in void fraction condition, the reprocessed fuels present higher core reactivity than traditional fuel (Table 9).

It is important to note that the use of reprocessed fuels linked with light water reduces the initial core reactivity when *VF*=0 (Table 9). However, it does not reduce the *CVR* (Figure 6), because as *VF* augment, the coolant density decreases and the absorption effectiveness of H_2O also decreases.

4. CONCLUSIONS

The simulations show that FRA and FRO present similar behavior due to the small difference between their isotopic compositions. Comparing with natural uranium, these fuels have a higher percentage of fissile isotopes and absorber isotopes. The insertion of these reprocessed fuels in CANDU-6 core increases the k_{eff} of CANDU-6 and reduce the thermal neutron flux in the reactor core. The use of H_2O allows increasing in the number of RFC than the ones used with D_2O, because light water has lower moderation ratio than heavy water. When is used H_2O as a coolant on the CANDU, about 1/4 of core (80 or 84 RFC) should be loaded with FRA and FRO, this produces similar criticality results than standard fuel core. However, the insertion of the control devices provides a negative core reactivity. Nevertheless, as the RFC increase, the effectiveness of control rods decreases because the reprocessed fuels increase the core reactivity. The use of reprocessed fuel associated/coupled with the coolant H_2O could present the advantage in the fuel cycle of CANDU and the economic advantage of the light water. However, the increase of RFC provokes augment in the Coolant Void Reactivity which tends to be positive. Then, although the FRA and FRO have absorptive isotopes, the use of these reprocessed fuels requires the insertion of burnable poison in the fuel bundle. Dysprosium, Erbium and Boron present as promising burnable poison to be used in FRA and FRO fuel bundles. The present work evaluates the CANDU-6 at steady-state condition but future studies will analyses the influence of burnable poisons in the core parameter at steady-state and during the burnup. Furthermore, the simulations were based on a reference core in which all fuel channels are loaded with fresh natural uranium. Considering that each fuel channel of CANDU is refueled online, there is always a burnup distribution across the core. Thus, subsequent works will be developed to simulate the insertion of FRA and FRO in an equilibrium core.

ACKNOWLEDGMENTS

The authors are grateful to the following Brazilian research funding agencies: *Fundação de Amparo à Pesquisa do Estado de Minas Gerais* (FAPEMIG), *Coordenação de Aperfeiçoamento de Pessoal de Nível Superior* (CAPES), *Conselho Nacional de Desenvolvimento Científico e Tecnológico* (CNPq) and *Comissão Nacional de Energia Nuclear* (CNEN).

REFERENCES

[1] IAEA – International Atomic Energy Angency. **Nuclear power reactor in the world**. **Reference Data Series No. 2**, IAEA, Vienna, 2018.

[2] KO, W.I., GAO, F., 2012. Economic analysis of different nuclear fuel cycle options. **Science and Technology of Nuclear Installations**, v. 2012, p. 1-10, 2012.

[3] PARK, B. H.; GAO, F.; KWON, E.; KO, W. I. Comparative study of different nuclear fuel cycle options: quantitative analysis on material flow. **Energy Policy**, v. 39, p. 6916-6924, 2011.

[4] LEE J.; RYU H.; PARK G.; SONG K. Recent progress on the DUPIC fuel fabrication technology at KAERI, In: **ATALANTE** 2008 – **NUCLEAR FUEL CYCLES FOR A SUSTAINABLE, 2008**, Montpellier. Proceedings of Atalante 2008, 2008, p. 1-4.

[5] YANG, M. S.; CHOI, H.; JEONG, C. J.; SONG, K. C.; LEE, J. W.; PARK, G. I.; KIM, H. D.; KO, W. I.; PARK, J. J.; KIM, K. H.; LEE, H. H.; JOO HWAN PARK, J. H. The status and prospect of DUPIC fuel technology. **Nuclear Engineering and Technology**, v. 38, p. 359-374, 2006.

[6] KO, W. I., KIM, H. D. Analysis of nuclear proliferation resistance of DUPIC fuel cycle. **Journal of Nuclear Science and Technology**, v. 38, p. 757-765, 2001.

[7] HONG, J.S.; KIM, H. D.; YANG, M. S.; PARK, H. S.; MENLOVE, H.; ABOU-SAHARA, A.; ALSTON, W. **Safeguards experience on the DUPIC fuel cycle process. LANL Report LA-UR-01-0936.** New Mexico: LANL, 2001. 9p.

[8] KANG, K. H.; SONG, K. C.; PARK, H. S.; MOON, J. S.; YANG, M. S. 2000. Fabrication of simulated DUPIC fuel. **Metals and Materials**, v. 6, p. 583-588, 2012.

[9] SULLIVAN, J.D.; RYZ, M.A.; LEE, J.W. Aecl's Progress in DUPIC Fuel Development. **Atomic Energy of Canada Limited (AECL)**, v. 31, p. 300-306, 1997.

[10] KANG, J.; SUZUKI, A. Analysis on DUPIC Fuel Cycle in Aspect of Overall Radioative Waste Management. **Nuclear Fuel Cycle and Materials**, v. 4, p. 19-27, 1997.

[11] TUMINI, L. L. P.; FLORIDO, P.C.; SBAFFONI, M. M.; ABBATE, M. J.; MAI, L. A.; MAIORINO, J. R. Study of a TANDEM fuel cycle between a Brazilian PWR (Angra-I) and Argentinean CANDU (Embalse). **Annals of Nuclear Energy**, v. 22, p. 1-10, 1995.

[12] SULLIVAN J.D.; COX D.S. Aecl's progress in developing the DUPIC fuel fabrication process, In: **4TH INTERNATIONAL CONFERENCE ON CANDU FUEL**, 1995, Pembroke. Proceedings of 4th International Conference on CANDU Fuel, Canadian Nuclear Society, 1995, p. 300-310.

[13] SILVA, C. A. M.; PEREIRA, C.; VELOSO, M. A. F.; GALLARDO, S.; VERDÚ, G. Analysis of DUPIC fuel cycle using the MCNPX code, In: TOP FUEL 2015 - **REACTOR FUEL PERFORMANCE**, 2015, Zurich. Proceedings of Top Fuel 2015, European Nuclear Society, 2015. p. 85-94.

[14] POUNDERS, J. M.; RAHNEMA, F.; SERGHIUTA, D.; THOLAMMAKKIL, J. 2011. A 3d stylized half-core CANDU benchmark problem. **Annals of Nuclear Energy**, v. 38, p. 876-896, 2011.

[15] PARK C.J., CHOI H. Benchmarking WIMS/RFSP against Measurement Data of Wolsong Nuclear Power Plants. In: **JOINT INTERNATIONAL TOPICAL MEETING ON MATHEMATICAL AND SUPERCOMPUTING IN NUCLEAR APPLICATIONS**, 2007, Monterey. Proceedings of a Meeting Held. American Nuclear Society, 2007. p. 2-11.

[16] CHOI, H.; ROH, G.; PARK, D. Benchmarking WIMS/RFSP against measurement data II: Wolsong Nuclear Power Plants 2, **Nuclear Science and Engineering**, v. 150, p. 37-55, 2005.

[17] GARLAND, W. J. **The Essential CANDU**, 1st ed. Canada: UNENE. Hamilton, 2014.

[18] LEE, J. W.; KIM, W. K.; LEE, J. W.; PARK, G. I.; YANG, M. S.; SONG, K. C. Remote fabrication of DUPIC fuel pellets in a hot cell under quality assurance program. **Journal of Nuclear Science and Technology**, v. 44, p. 597-606, 2007.

[19] SULLIVAN, J. D.; COX, D.S., 1997. **Aecl's progress in developing the DUPIC fuel fabrication process**. **AECL Report CA9800574**. Toronto: AECL, 1997. 10p.

[20] RADULESCU, G.; WAGNER, J. C. **Burn-up Credit Criticality Benchmark. Phase VII - UO_2 Fuel: Study of Spent Fuel Compositions for Long-Term Disposal. NEA Report 6998**, Issy-les-Moulineaux: NEA, 2012. 182p.

[21] MAJUMDAR, D.; JAHSHAN, S. N.; ALLISON, C. M.; KUAN, P.; THOMAS, T. R. **Recycling of nuclear spent fuel with AIROX processing. DOE Report 10423,** Idaho: DOE, 1992. 68p.

[22] PARENT, E. **Nuclear Fuel Cycles for Mid-Century Deployment - Master Thesis**, Massachusetts: Institute of Technology, Department of Nuclear Engineering, 2003.

[23] ENDF Data - LANL. **ENDF/B-VII.1 Incident-Neutron Data**. Los Alamos, New Mexico, USA. Available at: <https://t2.lanl.gov/nis/data/endf/endfvii.1-n.html>. Last accessed: 15 November 2019.

[24] MACFARLANE, R. E.; MUIR, D. W.; BOICOURT, R. M.; KAHLER, A. C. **NJOY - The NJOY nuclear data processing system, version 2012**. Los Alamos National Laboratory: Los Alamos, 2012. 810p.

[25] HENDRICKS, J. S.; MCKINNEY, G. W.; FENSIN, M. L.; JAMES, M. R.; JOHNS, R. C.; DURKEE, J. W.; FINCH, J. P.; PELOWITZ, D. B.; WATERS, L. S.; JOHNSON, M. W. **MCNPX user's manual, version 2.6.0**. Los Alamos National Laboratory: Los Alamos, 2008. 636p.

[26] DUDERSTADT, J. J.; HAMILTON, L. J. **Nuclear Reactor Analysis**, 1st ed. New York: John Wiley & Sons, 1976.

[27] ROUBEN, B. **Reactivity Coefficients - Nuclear Reactor Analysis**. Hamilton: McMaster University, 2015

[28] TALEBI, F.; MARLEAU, G.; KOCLAS, J. A model for coolant void reactivity evaluation in assemblies of CANDU cells. **Annals of Nuclear Energy**, v. 33, p. 975–983, 2006.

[29] CANDU Owners Group Inc. **Reactor Physics - Science and Reactor Fundamentals. CNSC training course**, Toronto: CANTEACH Project, 2003.

[30] GROH, J. L. **Nuclear Theory II (Kinetics).** Toronto: CANTEACH Document, Chulalongkorn University, 1996.

[31] MOHAMED, N. M. A. Direct reuse of spent nuclear fuel. **Nuclear Engineering and Design**, v. 278, p. 182–189, 2017.