NUCLEAR SAFETY ASPECTS ASSOCIATED TO INCREASING OF U235 ENRICHMENT FOR 37-ELEMENTS CANDU FUEL BUNDLES WITH SEU AND RU FUELS

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Abstract

The paper presents several results of a comparative neutronic analysis on main parameters of interest at cell level calculations, for CANDU fuel bundles with 37 elements and U-based fuel, considering the increasing in U^{235} enrichment. Two modified fuel bundle projects (C37M1 and C37M2) have been analyzed in comparison with the standard CANDU fuel bundle design, C37. The compositions considered in this paper were natural uranium (UNAT), slightly enriched uranium (SEU) and recovered uranium (RU), respectively. Elementary cell calculations were performed by means of WIMS and DRAGON codes, using the IAEA's nuclear data library on 172 energy groups. The paper aimed to evaluate the neutronic parameters of interest for economic considerations (discharge maximum burnup, uniform power distribution in the bundle, U^{235} consumption to produce a unit of energy), and for nuclear safety ensuring (void reactivity coefficient, Doppler effect, resistance to proliferation based on estimation of minor actinides amount resulting from the fuel burnup).

Keywords: *Modified CANDU Fuel, Recovered Uranium, Cell Calculations, Neutronic Parameters*

Introduction

CANDU flexibility in using advanced fuel bundles was addressed by extensive studies performed in Canada, Korea, China and Romania during the last 30 years. The lower overall performance compared to the PWR reactors, required and launched a continuous development and implementation of new fuel bundles with advanced characteristics.

Since the early 1980s, the Fuel Performance and Physics Analysis Group of the Nuclear Research Institute (RATEN-ICN PITESTI), aims to develop and test new advanced fuels for CANDU reactors. Over time, several types of fuel bundles have been developed: fuel bundles with 37 modified elements [1, 2], with 43 elements [3] and with an increased number of elements [4, 5]. At the same time, many fuel compositions have been taken into account, [6 - 11].

This paper proposes an neutronic analysis for a modified CANDU type fuel bundle project, at lattice cell level, comparative to the CANDU standard fuel bundle. The considered fuel bundles contain the same fuel in all the 37 elements. Fourteen different fuel compositions were analyzed, namely: Natural Uranium fuel, nine fuel compositions of Recovered Uranium (RU), and four fuel compositions of Slightly Enriched Uranium (SEU).

For the lattice cell assessment, 2-D neutron transport calculations were performed to estimate the following neutronic parameters: the maximum burnup at discharge, the radial power distribution expressed by the Power Peaking Factors (PPFs), the consumption of U^{235} for the unit of energy production, the void reactivity expressed by the Coefficient of Void Reactivity (CVR), the Doppler effect expressed by the fuel temperature coefficient of reactivity, and the inventory of minor actinides in the spent fuel concerning for the proliferation resistance.

WIMS [12] and DRAGON [13] transport codes were used in simulation, and the necessary nuclear data based on ENDF/B-VII were taken from a 172-energy group library generated in an IAEA sponsored project [14].

Fuel bundle description

All considered fuel bundles are expected to contain the same fuel (natural uranium, recovered uranium or slightly enriched uranium) in sintered pellets. The C37 bundle has the standard pellet diameter of about 12 mm, while the modified fuel bundle variants, C37M1 and C37M2, have smaller central pellet diameter, namely ~ 11 and ~ 10 mm.

Nine fuel compositions based on Reprocessed Uranium were considered, the difference consisting in the percentage mass of the U^{235} isotope, the isotopic component of the oxide and the source of the spent fuel. As for the Slightly Enriched Uranium fuel, 4 compositions were considered based on the difference in the percentage mass of U^{235} isotope [15, 16]. All fourteen fuel configurations considered for analysis are presented in Table 1.

Table 1. Identification of the studied fuel compositions

Nr. Crt.	Case	wU^{234} (%)	wU^{235} (%)	wU^{236} (%)	wU^{238} (%)	Spent Fuel source
1.	UNAT	0	0.71	0	99.29	-
2.	RU 1	0.02	0.9	0.3		LWR
3.	RU 2	0.04	1.3	0.7		LWR
4.	RU 3	0.02	0.9	0.2		CANDU
5.	RU 4	0.02	0.9	0.4		CANDU
6.	RU 5	0.02	1.0	0.2		CANDU
7.	RU 6	0.02	1.0	0.4		CANDU
8.	RU 7	0.02	1.1	0.2		CANDU
9.	RU 8	0.02	1.1	0.4		CANDU
10.	RU 9	0.024	0.99	0.45		PWR
11.	SEU 1	0	0.9	0	99.1	-
12.	SEU 2	0	1.0	0	99.0	-
13.	SEU 3	0	1.1	0	98.9	-
14.	SEU 4	0	1.2	0	98.8	-

Methodology for neutronic analysis and key parameters

A simplified model was used to simulate the fuel bundles behaviour in the CANDU core. Each bundle was assumed infinitely long, therefore a 2-D model was used, which allows to ignore the neutron axial leakage. Moreover, even the Zircaloy spacers, pads, end caps and end plates were ignored in the 2-D model, Zirconium mass per bundle was conserved, increasing the sheath density accordingly.

The standard CANDU cell model contains - apart from the fuel bundle itself - the pressure tube, the calandria tube, the corresponding coolant and moderator, characteristics to the CANDU 6 reactor. The data were taken from a "classical" IAEA benchmark problem dedicated to CANDU reactor [17].

Neutron transport calculations were performed using the lattice codes DRAGON and WIMS together with a data library based on ENDF/B-VII (172 energy groups). For WIMS calculations, the DSN (SEQUENCE 1 option) method was used, while for DRAGON calculations, the EXCELT: tracking module was selected.

For reasons related to the UO_2 mass conservation along the entire length of the fuel bundle, the "*corrected*" densities of the fuel pellets characterizing the considered geometry were determined. The standard fuel pellets "*corrected*" density value was calculated as 10.289 g/cm^3; for the modified C37 fuel bundles (C37M1 and C37M2), the central elements "*corrected*" density values were calculated as 0.27 g/cm^3 and 11.04 g/cm^3, respectively.

For fuel burnup simulation, a 35 kW/kgU constant power rate was assumed, roughly corresponding to a bundle power of 700 kW. The lattice cell arrangement is given in Fig. 1 (DRAGON plots given by PSP: module).

The combination of fuel composition, bundle geometry and computer code used resulted in 14x3x2=84 simulations to be prepared, run and interpreted for each of the investigated cell parameter.

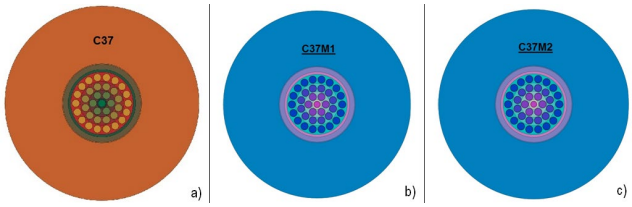


Fig. 1. Lattice cell - DRAGON plots for fuel bundles: a) C37; b) C37M1; c) C37M2

The main parameter to describe an innovative fuel performance is the maximum burnup i.e the maximum burnup for which the cell is still critical [18], ensuring enough radioactivity to maintain the chain reaction.

Uniform burnup was also considered a key parameter to ensure a good fuel performance, both from the cost effectiveness point of view and from the safety one, since an unbalanced radial power distribution may lead to uneven fuel sheath stress and thus to fuel.

A Power Peaking Factor (PPF) can be defined for each fuel element to express its contribution to the bundle power output. Since in the cell model used for DRAGON and WIMS the fuel elements on the same ring are equivalent, defining only 4 PPFs corresponding to the central element, respectively to the elements on the inner, middle and outer ring, should provide enough information:

$$PPF_i = w_i \cdot \frac{V_{total}}{V_i \cdot N_i}, \text{ i} = 1, 2, 3, 4, 5 \qquad (1)$$

where: w_i are the fractions of total bundle power produced by each radial fuel region, V_{total} is the total fuel volume, V_i is the volume of an element in ring i, and N_i are the number of fuel elements per region.

Innovative fuel projects are supposed to ensure economic improvements by more effective fissile utilization in energy production. Therefore, a quantitative parameter (q) was defined to express the amount of fissile needed to produce 1 Joule of energy:

$$q = \frac{\sum_{j=1}^{4} n_j \cdot v_j \cdot N_j(B_{\max}) - \sum_{j=1}^{4} n_j \cdot v_j \cdot N_j(B_0)}{B_{\max}} \cdot \frac{\sum_{j=1}^{4} n_j}{\sum_{j=1}^{4} n_j \cdot v_j} \qquad (2)$$

where: $N_j(B)$ is U^{235} isotopic number density for element ring j at burnup B (= 0 for fresh fuel and = B_{max} for maximum burnup), v_j is the fuel volume in each element on ring j, and n_j is the number of elements on ring j.

The temperature and void reactivity coefficients were estimated as "*instantaneous perturbations*", [19] for fresh fuel cells for which fuel temperature, as well as coolant density were modified and cell reactivity was estimated.

The fuel temperature considered in the calculations was 1200 K. The temperature coefficients were evaluated by modifying the nominal (design) values with ±100 ˚K and ±200 ˚K, respectively, for both computer codes.

$$cT = \left(\frac{1}{k_{ref}} - \frac{1}{k_T}\right) \times \frac{1000}{T - T_{ref}} \quad (3)$$

To estimate the void reactivity, the homogenous approach was used, i.e. the overall coolant density was reduced by the void fraction (given in %):

$$dens_{fv} = dens_0 \cdot \left(1 - \frac{fv}{100}\right) \quad (4)$$

The void coefficient for a given void fraction is estimated as:

$$CVR(fv) = \left(\frac{1}{k_{ref}} - \frac{1}{k_{fv}}\right) \times \frac{1000}{fv} \text{ mk/\%} \quad (5)$$

In order to evaluate the attractiveness of a burned fuel for military purposes, it is necessary to estimate the amount of plutonium and Minor Actinides at the end of burnup. For this purpose, we used formula given below to estimate the number densities of the aim isotopes: Plutonium isotopes (Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241} and Pu^{242}), Americium isotopes (Am^{241}, Am^{242} and Am^{243}) and Curium isotopes (Cm^{242}, Cm^{243} and Cm^{244}) in the spent fuel.

$$ND_k = \frac{1 \cdot ND_k(1) + 6 \cdot ND_k(2) + 12 \cdot ND_k(3) + 18 \cdot ND_k(4)}{37} \quad (6)$$

Main cell results and discussions

The maximum burnup for the standard CANDU fuel bundle C37, as well as for the modified fuel bundles containing UNAT fuel, was about 6400 MWD/tU, for both codes considered (Fig.2). This important neutronic parameter was not perturbed by reducing the central element diameter.

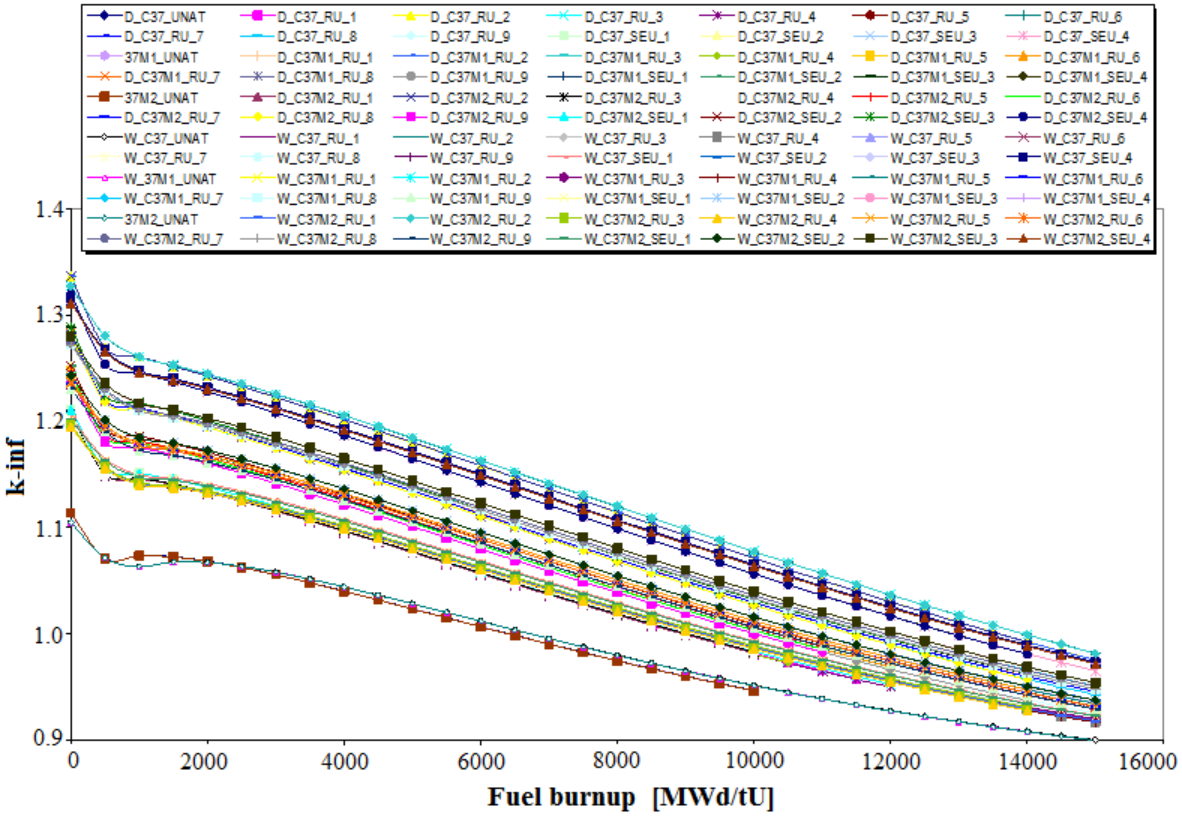


Fig. 2. k-inf evolution with burnup (WIMS and DRAGON calculations)

WIMS code estimations were systematically 5% higher than the DRAGON code ones. According to both codes, for all considered configurations of RU and SEU the maximum burnup at discharge exceeds the reference CANDU standard one.

Two fuel configurations (RU_2 and SEU_4) conducted to maximum burnup values twice higher than the reference one (as can be seen in Fig. 3). Maximum burnup values obtained for C37M1 fuel bundle project with UNAT, SEU or RU fuels were 0.04% higher compared with the ones characterizing the C37M2fuel bundle project.
Fig. 3 illustrates comparatively the maximum fuel burnup values obtained by means of DRAGON code for the selected fuel compositions considering the C37 and C37M2 bundle projects. The differences (in percent) from the value characterizing the reference fuel bundle C37 with natural uranium fuel are also mentioned.

PPF = 1 indicates an uniform power distribution. For the fuel bundles with UO_2 fuel (C37_UNAT), PPF1, PPF2 and PPF3 are less than 1, while PPF4 is about 1.1.

Using SEU or RU fuel instead of natural uranium does not improve the radial power distribution, as seen in Fig.4. In order to improve this parameter, fuel compositions with differentiated U^{235} enrichment on the fuel bundle rings should be used.

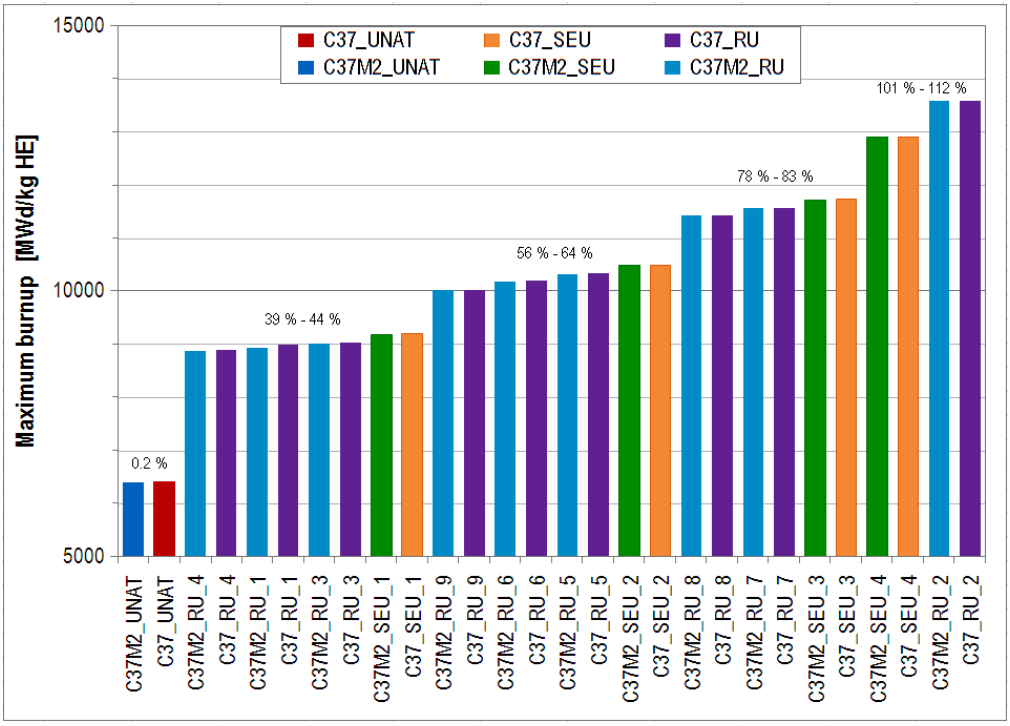


Fig. 3. Maximum burnup [MWd/kgHE] (WIMS and DRAGON calculations)

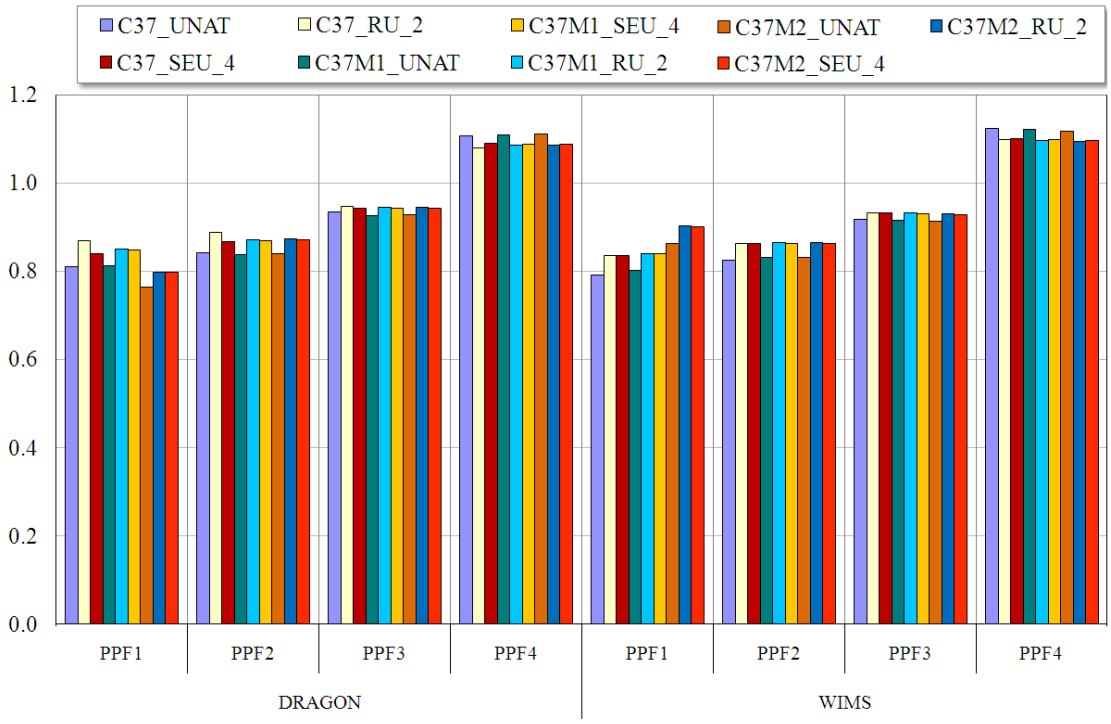


Fig. 4. Power peaking factor values for the most important fuel compositions

For CANDU standard fuel bundle, C37, U^{235} consumption parameter is 6.14 g/J. In the case of advanced fuel bundles, for this parameter a lower value is desirable, but the presence of U^{233} and U^{236} isotopes in SEU and RU fresh fuel may lead to higher values of this parameter.

Table 2 shows the values of the U^{235} consumption for all the analyzed fuel bundles and compositions.

Table 2. U^{235} consumption per unit energy yield

Case	C37		C37M1		C37M2	
	DRAGON	WIMS	DRAGON	WIMS	DRAGON	WIMS
UNAT	6.14	6.12	6.14	6.12	6.16	6.13
RU_1	6.28	6.43	6.54	6.44	6.55	6.55
RU_2	6.89	6.89	7.14	7.18	7.16	6.88
RU_3	6.36	6.41	6.29	6.22	6.30	6.42
RU_4	6.40	6.46	6.56	6.54	6.57	6.48
RU_5	6.48	6.50	6.58	6.51	6.60	6.52
RU_6	6.52	6.55	6.62	6.56	6.63	6.57
RU_7	6.60	6.61	6.65	6.62	6.66	6.63
RU_8	6.64	6.66	6.86	7.00	6.88	6.67
RU_9	6.51	6.55	6.56	6.56	6.57	6.68
SEU_1	6.34	6.27	6.24	6.22	6.26	6.39
SEU_2	6.47	6.48	6.54	6.48	6.56	6.49
SEU_3	6.59	6.59	6.61	6.59	6.63	6.60
SEU_4	6.71	6.68	6.84	6.78	6.86	6.79

Table 3. Temperature coefficients for considered fuels

Case	C37		C37M1		C37M2	
	DRAGON	WIMS	DRAGON	WIMS	DRAGON	WIMS
UNAT	-0.005	-0.005	-0.005	-0.005	-0.005	-0.005
RU_1	-0.005	-0.005	-0.005	-0.005	-0.005	-0.005
RU_2	-0.004	-0.005	-0.004	-0.004	-0.004	-0.004
RU_3	-0.005	-0.005	-0.005	-0.005	-0.005	-0.005
RU_4	-0.005	-0.005	-0.005	-0.005	-0.005	-0.005
RU_5	-0.005	-0.005	-0.005	-0.005	-0.005	-0.005
RU_6	-0.005	-0.005	-0.005	-0.005	-0.005	-0.005
RU_7	-0.005	-0.004	-0.004	-0.004	-0.004	-0.004
RU_8	-0.005	-0.005	-0.005	-0.005	-0.004	-0.005
RU_9	-0.005	-0.005	-0.005	-0.005	-0.005	-0.005
SEU_1	-0.005	-0.005	-0.005	-0.005	-0.005	-0.005
SEU_2	-0.005	-0.004	-0.005	-0.004	-0.005	-0.004
SEU_3	-0.004	-0.004	-0.005	-0.004	-0.004	-0.004
SEU_4	-0.004	-0.004	-0.004	-0.004	-0.004	-0.004

As we anticipated, the values of the uranium consumption parameter for one unit of power production, characteristic of advanced fuels based on SEU and RU are superior to the standard value.

Table 3 presents the calculated temperature coefficients for fuel, the results exhibiting no influence coming from reducing the central element diameter or the use of different fuels composition.

The void coefficient (CVR) associated with the CANDU reactor conditions of operation is positive. The void reactivity estimated values for the considered fuel bundles and compositions using DRAGON and WIMS are still positive, but by up to 12% lower than the standard value.

Figure 5 shows the Plutonium number densities (total and separate for each of its isotopes Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241} and Pu^{242}) in the spent fuel bundle, for the considered geometry and fuel composition in this paper work.

Other significant isotopes (Americium and Curium) from the point of proliferation resistance view were also investigated. Figures 6 and 7 illustrate the quantities of Americium (total and separate for each of its isotopes Am^{241}, Am^{242} and Am^{243}) and Curium (total and separate for each of its isotopes Cm^{242}, Cm^{243} and Cm^{244}).

The quantities of minor actinides found in the innovative burned fuels, exceed the quantities obtained for the standard fuel bundle, thus reducing the attractiveness for reprocessing the spent fuel.

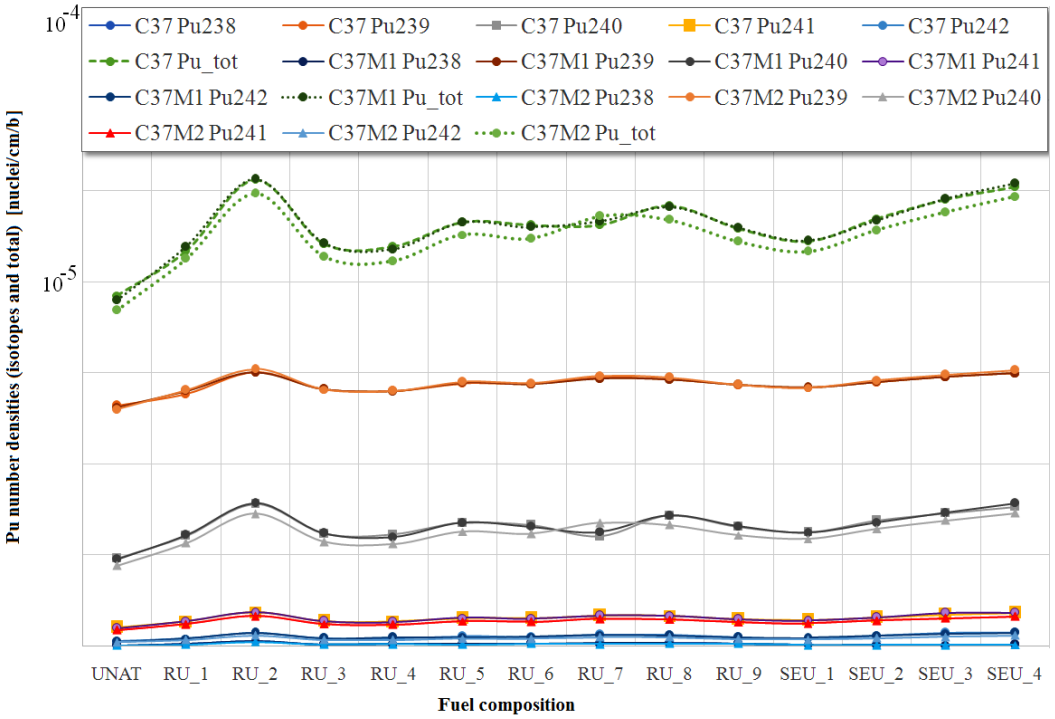


Fig. 5. Pu number densities [nuclei/cm/b] for the considered bundles and fuel compositions (DRAGON estimation)

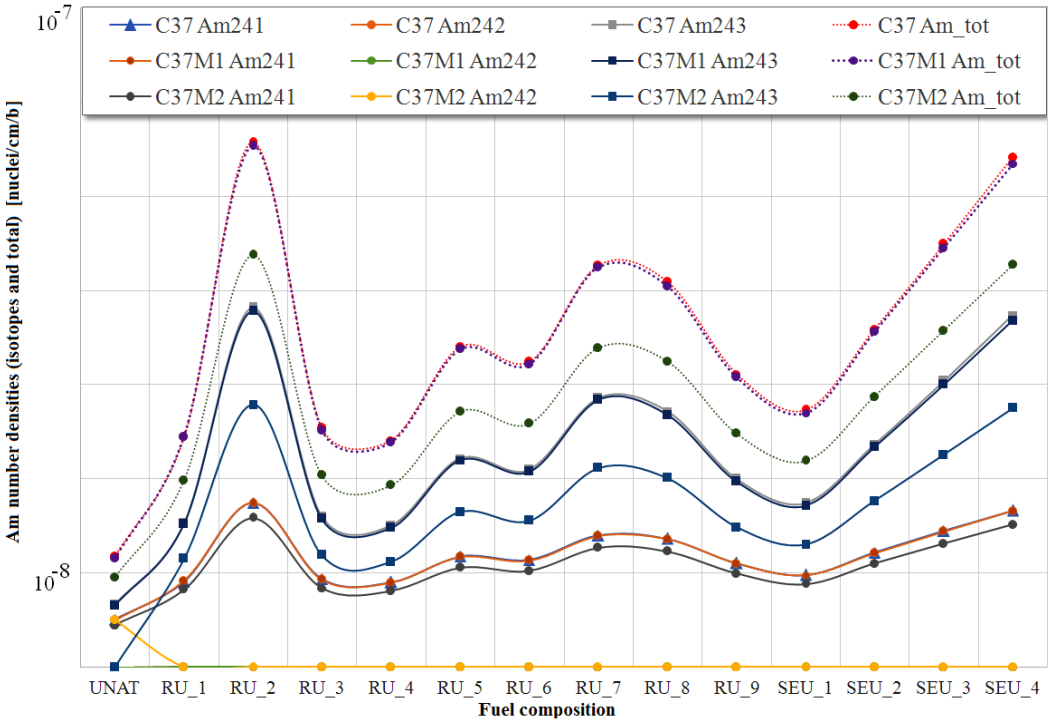


Fig. 6. Am number densities [nuclei/cm/b] for the considered bundles and fuel compositions (DRAGON estimation)

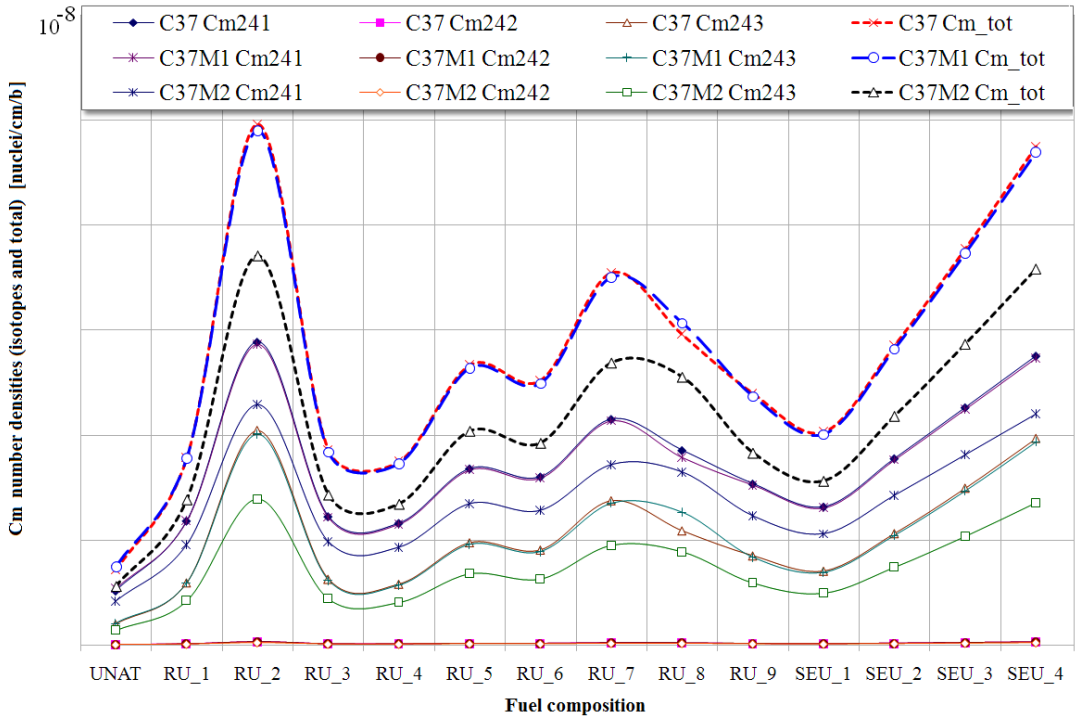


Fig. 7. Cm number densities [nuclei/cm/b] for the considered bundles and fuel compositions (DRAGON estimation)

Conclusions

A comparative neutronic analysis on the main parameters of interest at lattice cell level was presented in present work for CANDU fuel bundles with 37 elements which contain U-based fuel, taking into consideration the U^{235} enrichment increasing. The analysis was conducted for two fuel bundle projects, namely the CANDU standard fuel bundle (as reference) and a modified 37-element fuel bundle type CANDU for which the central element diameter was reduced compared to the reference element one. Natural Uranium, SEU or RU fuel compositions were analysed, the calculations being performed considering that all fuel bundle elements contain the same fuel composition.

Using the fuel bundles in CANDU with reduced central element diameter does not significantly alter the neutronic features of the lattice parameters: maximum burnup, radial power peaking factors, void reactivity.

For all considered configurations, the discharge burnup exceeds the reference CANDU standard one, for RU_2 and SEU_4 configurations being more than twice higher than the reference value.

Modified geometry or replacement of CANDU standard UO2 fuel with RU or SEU does not bring remarkable improvements in terms of uniform power distribution.

U^{236} presence in SEU/RU fresh fuel lead to higher values for U consumption compared with the reference value.

The void reactivity coefficient estimated for the considered fuel projects are still positive, but by up to 12% lower than the standard CANDU fuel value.

The Doppler effect is negative, the same as for C37 – these are an important feature from nuclear safety point of view.

Pu amount in the spent fuel is higher comparatively with the reference case; the presence of U^{232} and the amount of minor actinides, as Am and Cm, ensure for the resistance to proliferation concern.

Higher activities of minor actinides (produced by fuel burnup) and reduction of the void coefficient, support the possibility of using the studied fuels on the development of accident tolerant fuel projects.

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