ANALYSIS OF ACCIDENT-TOLERANT FUEL BEHAVIOUR IN CANDU REACTORS

GEORGETA ROXANA BEZDEDEANU and CRISTINA ALICE MĂRGEANU

Institute for Nuclear Research (RATEN ICN) Pitesti, Romania
e-mail: roxana.bezdedeanu@nuclear.ro

Abstract

The paper presents several results of a comparative neutronic analysis on main parameters of interest at cell level calculations of the accident-tolerant fuel bundles (ATF) compared to the CANDU standard fuel bundle currently exploited in CANDU reactors. Two accident-tolerant fuel approaches were investigated, one assuming the use of dopants (Al_2O_3, Cr_2O_3) in the fuel pellets' composition, and the other considering both the use of dopants in the fuel pellets and a coated chromium clad. The fuel compositions considered were based on natural UO_2, namely: UO_2 doped with chromium oxide, UO_2 doped with aluminium oxide and UO_2 doped with both types of oxides, 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 (maximum discharge burnup, power distribution in the bundle, U^{235} consumption), 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: *Accident Tolerant Fuel, Doped fuel, Coated clad, Cell Calculations, Neutronic Parameters*

Introduction

The main advantage of CANDU reactors is the versatility of using different fuel cycles (natural uranium, equivalent uranium, slightly enriched uranium, depleted uranium, reprocessed uranium, mixed oxide of uranium and thorium, mixed oxide of uranium and plutonium, etc.).

Currently, there are 31 CANDU reactors in operation, the countries with CANDU technology including Canada, South Korea, China, Argentina, Romania, India and Pakistan [1].

Since the early 1980s, the Fuel Performance group at the Institute for Nuclear Research (RATEN ICN) Pitesti, aims to develop and test new advanced fuels for CANDU type reactors. Over time, several types of fuel compositions have been taken into account, [2-6]. At the same time, many CANDU type fuel bundle designs have been developed: modified fuel bundles with 37 elements [7, 8], fuel bundles with increased number of elements (43, 52, and 61 elements, respectively) [9-12].

This paper aims to evaluate the economic and nuclear safety performances of the CANDU accident-tolerant fuel projects, by performing calculations at the elementary cell level, compared to the standard CANDU fuel bundle with 37 fuel elements and natural UO_2 fuel.

Eight different fuel compositions based on natural uranium were analysed, as follows: the UO_2 standard composition, a composition of UO_2 doped with Al_2O_3, 3 compositions of UO_2 doped with Cr_2O_3, and 3 compositions of UO_2 doped with both Al and Cr oxides, respectively.

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}, 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.

The neutron transport calculations were performed using the collision probability code DRAGON (vers. 5, [13]), as well as the discrete ordinates (SN) option in WIMS (vers. D5B, [14]), together with an adequate nuclear data library based on ENDF/B-VII evaluated nuclear data [15].

Fuel bundle description

The CANDU type fuel bundle has cylindrical shape and includes 37 identical elements, arranged on 3 concentrical rings (of 6, 12 and 18 elements, respectively) around a central element. All the fuel elements contain the same fuel composition, natural UO_2 sintered or doped sintered pellets. The fuel pellets (~ 12 mm in diameter), are obtained by pressing UO_2 powder in a cylindrical shape, which by heat-treatment at high temperature becomes very dense.

All the eight fuel configurations considered for analysis are presented in Table 1. The content of Cr oxide in the doped fuel pellets was between 0 and 0.16%, corresponding to the solubility limit of the dopant in UO_2 at the applicable sintering conditions.

In three of the considered fuel compositions, both Cr and Al oxides were added. The addition of Al_2O_3 contributes to the grains' growth and provide beneficial features for the fuel performance: dimensional stability, improved behaviour in case of water/steam ingress and a higher fission gas retention capability, which could lead to lower fission gas release during potential accidents, [16,17].

Table 1. Identification of the studied cases

Case ID	Cr/U mass ratio	Al/U mass ratio
C37	0	0
ATF1	0	0.02 %
ATF2	0.05 %	0
ATF3	0.05 %	0.02 %
ATF4	0.10 %	0
ATF5	0.10 %	0.02 %
ATF6	0.16 %	0
ATF7	0.16 %	0.02 %

Another issue addressed by the study was the effect on the main neutronic parameters due to the addition of a thin layer of metallic Cr (between 1 – 10 μm thick) on the zircalloy cladding. The chromium layer is used to improve the high temperature steam corrosion behaviour, [18]. Three thicknesses of the chromium layer were considered in the paper, namely: 1 μm, 5 μm and 10 μm, respectively.

Combining the three variables (fuel composition, thickness of Cr layer and computer code used for simulation), a number of 8 x 3 x 2=48 simulations have been performed. The study presents the neutronic parameters of interest evolution with fuel burnup: maximum discharge burnup, power distribution in the bundle, U^{235} consumption, void reactivity coefficient, Doppler effect, minor actinides amount in the spent fuel.

Methodology for neutronic analysis and key parameters

In order to be analysed in CANDU6 reactor conditions, fuel bundles must respect the other characteristics of the standard CANDU elementary cell (pressure tube, Calandria tube, coolant and moderator), as illustrated in Fig. 1.

These characteristics were taken from a benchmark problem proposed by AECL, [19]. For fuel burnup simulation, a 35 kW/kgU constant power rate was assumed, roughly corresponding to a bundle power of 700 kW.

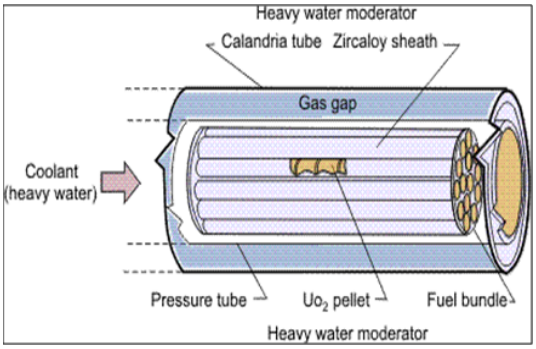


Fig.1. CANDU fuel bundle in a fuel channel, [20]

Based on considerations related to the 1-D modeling of the elementary cell and from the need to conserve the mass of uranium dioxide along the entire length of the fuel bundle, the "*corrected*" densities of the fuel pellets contained in the fuel bundle were determined. Similarly, to preserve the zircaloy mass per cell, the "*corrected*" densities of the claddings were calculated. The corrected value obtained for the fuel pellets was 10.289 g/cm^3, while the corrected value for the zircaloy claddings was 7.469 g/cm^3.

A lattice cell program solves the neutron balance equation in this simplified geometry, taking into account the nuclear properties of the materials that make up the cell. The plot images, corresponding to the studied fuel bundles, were obtained using the PSP:module of the DRAGON computer code and are shown in Fig. 2.

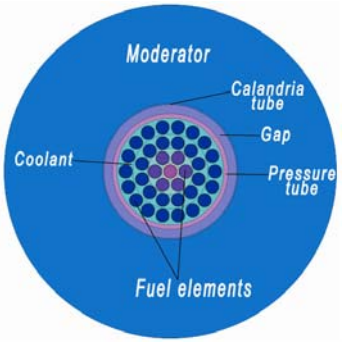


Fig. 2. CANDU cell with 37-element fuels bundles (DRAGON plots)

The reactivity of an initially supercritical cell decreases with fuel burnup due to consumption of fissile isotopes and accumulation of "poisons" (Xe, I), the cell becoming subcritical. The maximum burnup is defined as the fuel burnup degree for which the cell is still critical, k_{inf} =1.0 (multiplication constant equals 1, so fission still possible).

Uniform burnup was also considered a key parameter for this analysis, because it ensures a good fuel performance, both from the cost effectiveness point of view and from the safety one. Fuel elements located on the same ring are considered identical. As a result, only 4 PPFs (power peaking factors) need to be defined (see formula below), corresponding to the central element and the 3 rings with 6, 12 and 18 fuel elements, respectively:

$$PPF_i = \frac{w_i}{N_i} \tag{1}$$

where: w_i are the fractions of total bundle power produced by each radial fuel region, N_i is the number of fuel elements per region (i = 1 ÷ 4).

The void effect, defined as the variation of cell reactivity upon loss of coolant, is an important safety indicator in defining the intrinsic safety of a fuel design. The coefficient of void reactivity is defined in [21] as the decrease/increase in reactivity of the cell upon total or partial cooling agent emptying:

$$CVR_{(f)} = \left(\frac{1}{k_{ref}} - \frac{1}{k}\right) \times \frac{1000}{f_v} \tag{2}$$

where k_{ref} is the multiplication factor corresponding to the cooled cell, k is the multiplication factor corresponding to the partially emptied cell, and f_v is the void fraction (in %).

The homogeneous coolant approximation was used, so the loss of coolant is approximated by decreasing its density:

$$d_f = d_0 \times \left(1 - \frac{f_v}{100}\right) \quad (3)$$

The fuel temperature considered in the calculations was 1200°K. The temperature coefficients were evaluated by modifying the nominal (design) values with ±200 K for both computer codes, using the formula below:

$$c_T = \left(\frac{1}{k_{ref}} - \frac{1}{k_T}\right) \times \frac{1000}{T - T_{ref}} \quad (4)$$

where: k_{ref} is the multiplication factor corresponding to the reference cell, k_T is the multiplication factor corresponding to the modified fuel temperature, T is the modified fuel temperature and T_{ref} is the nominal fuel temperature.

For the CANDU reactor, the temperature coefficient is negative [22] due to the U^{238} resonance.

The efficiency of U^{235} consumption is determined by comparing the inventories at the first burnup step (BoI, Beginning of Irradiation) and at the moment of discharge from the reactor core (EoI, End of Irradiation), as in the formula below:

$$q = \frac{\left(N_1 + 6N_2 + 12N_3 + 18N_4\right)_{B_{max}} - \left(N_1 + 6N_2 + 12N_3 + 18N_4\right)_{B_0}}{B_{max}} \quad (5)$$

where: N_i is number density of U^{235} on the ring i, B_{max} is the maximum burnup, and B_0 is the burnup for fresh fuel.

A current concern of fuel designers, from the point of view of resistance to nuclear proliferation, is the prevention of recovery the fissile materials from spent fuel elements. The amount of Pu resulting from fuel burnup was estimated, with particular interest in Pu^{239} amount. Minor Actinides (Am, Cm and Np, respectively) production was also of interest for this study, the formula given below being used to estimate number densities of the isotopes in the spent fuel.

$$q = \frac{\left(N_1 + 6N_2 + 12N_3 + 18N_4\right)}{37} \quad (6)$$

where: N_i is number density of isotope on ring i.

Main cell results and discussions

The maximum burnup for the standard CANDU fuel bundle, C37, was 6571 MWd/tU (DRAGON estimate) and 6897 MWd/tU (WIMS estimate), respectively.

In Fig. 3, the multiplication factor evolution with burnup is presented for all considered compositions of doped fuel in comparison with the standard fuel, UO_2, in C37 fuel bundle. According to both lattice cell codes used, a similar behaviour as for the standard CANDU fuel bundle was observed, but with slightly reduced values.

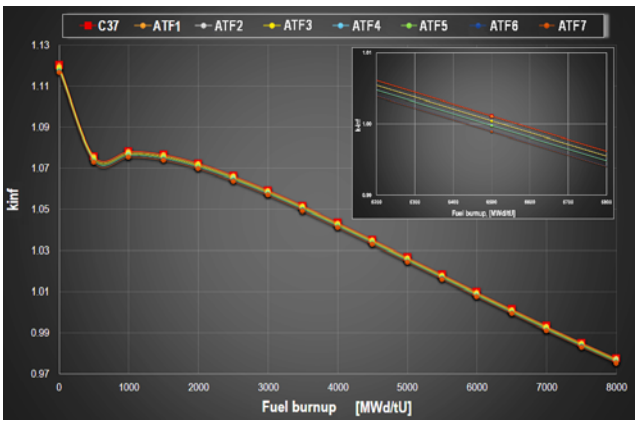


Fig. 3. k_{inf} evolution with burnup for dopped fuel pellets (DRAGON estimations)

Fig. 4 shows the k_{inf} multiplication factor evolution with burnup for the doped fuel bundles with additional Cr layer on the fuel element claddings. Due to very small differences between the Cr_2O_3 doped fuel compositions and those containing both dopants, only the compositions containing both Cr and Al oxides dopants were represented (ATF3, ATF5 and ATF7 cases, respectively). Only the minimum and maximum thickness of Cr layer on the fuel element cladding were selected, to increase the representation clearness.

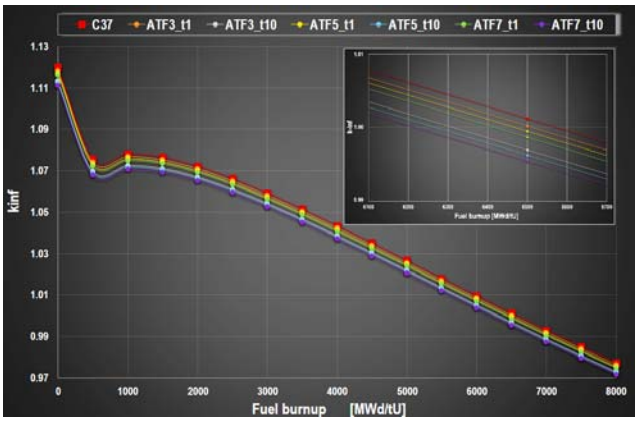


Fig. 4. k_{inf} evolution with burnup for dopped fuel pellets and additional Cr layer on fuel element cladding (DRAGON estimations)

The maximum burnup obtained for the doped fuel cases were between 6437 MWd/tU and 6567 MWd/tU, values lower by 2% compared to those characterizing C37 fuel bundle. For the coated chrome cladding, the maximum burnup was between 6215 MWd/tU and 6507 MWd/tU, values lower by up to 5.4% than the ones obtained for the C37 fuel bundle.

WIMS code estimations were systematically 5% higher than the DRAGON code ones. According to both codes, for the considered configurations of doped fuel and doped fuel with Cr coated cladding, the maximum burnup at discharge is lower compared to the CANDU standard one, as can be seen in Fig. 5.

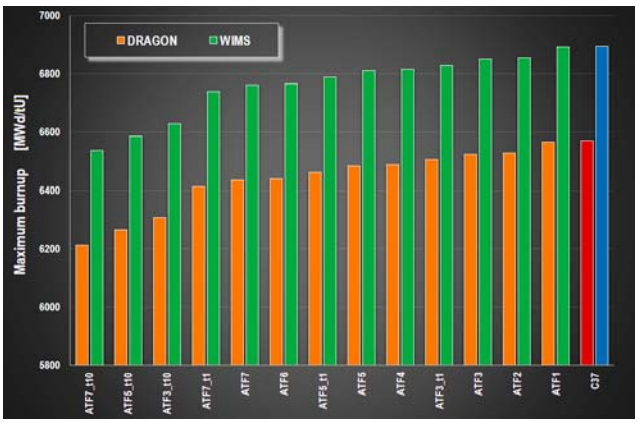


Fig. 5. Maximum burnup [MWd/tU] at the EoI (WIMS and DRAGON calculations)

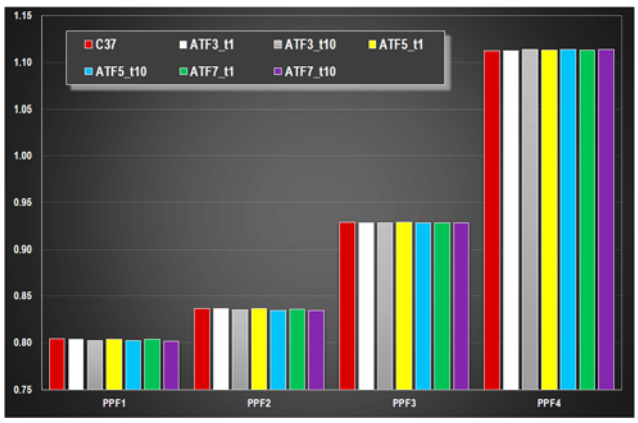


Fig. 6. Power peaking factor for analysed fuel compositions (doped fuels and Cr layer on cladding)

The power peaking factors were evaluated for each ring of fuel elements both at the beginning of burnup (B=0, fresh fuel) and at discharge moment ($B=B_{max}$). PPF=1 indicates the uniform power distribution. For the fuel bundle with UO_2 fuel (C37), PPF_1, PPF_2 and PPF_3 are less than 1, while PPF_4 is about 1.1.

Using doped fuel compositions instead of the UO_2 fuel does not improve the radial power distribution, as can be seen in Table 2. The power peaking factors (PPF) estimated for the doped fuel bundles have the same trend as for the reference fuel bundle: the inner and intermediate elements (1+6+12) are characterized by PPF values less than 1, while the external 18 elements exceed the average by up to 22% (fresh fuel, BoI) and 20% (spent fuel, EoI), respectively.

Table 2. Power peaking factors for analysed fuel compositions (DRAGON estimates at BoI and EoI)

Case ID	PPF_1	PPF_2	PPF_3	PPF_4	Rel. diff. from average	Rel. diff. PPF_1 vs PPF_4
Fresh fuel (BoI)						
C37	0.7750	0.8112	0.9152	1.1320	-22.50%	-31.54%
ATF1	0.7750	0.8112	0.9152	1.1320	-22.50%	-31.54%
ATF2	0.7749	0.8112	0.9152	1.1320	-22.51%	-31.55%
ATF3	0.7749	0.8112	0.9152	1.1320	-22.51%	-31.55%
ATF4	0.7748	0.8111	0.9151	1.1321	-22.52%	-31.56%
ATF5	0.7748	0.8111	0.9151	1.1321	-22.52%	-31.56%
ATF6	0.7747	0.8110	0.9151	1.1321	-22.53%	-31.57%
ATF7	0.7747	0.8110	0.9151	1.1321	-22.53%	-31.57%
Spent fuel (EoI)						
C37	0.8047	0.8365	0.9288	1.1128	-19.53%	-27.68%
ATF1	0.8047	0.8365	0.9288	1.1128	-19.53%	-27.68%
ATF2	0.8045	0.8363	0.9287	1.1129	-19.55%	-27.72%
ATF3	0.8045	0.8363	0.9287	1.1129	-19.55%	-27.71%
ATF4	0.8042	0.8361	0.9287	1.1131	-19.58%	-27.75%
ATF5	0.8042	0.8361	0.9287	1.1131	-19.58%	-27.75%
ATF6	0.8039	0.8358	0.9286	1.1132	-19.61%	-27.79%
ATF7	0.8039	0.8358	0.9286	1.1132	-19.61%	-27.79%

Addition of a thin layer of metallic chromium on the zirconium alloy cladding does not improve the radial power distribution, but it does not negatively influence this parameter either (see Fig. 6).

Using Eq. (5), the efficiency of U^{235} consumption in ATF fuel bundles was determined (see Fig. 7). For C37 fuel bundle, the reference fuel bundle, U^{235} consumption was 6.14 (DRAGON estimate) and 6.82 (WIMS estimate), respectively. U^{235} consumption for the doped fuels with or without Cr coated cladding had values between 6.14 and 6.23 (DRAGON estimations), and 6.82 and 6.96 (WIMS estimations), respectively.

Therefore, the efficiency of U^{235} consumption decreases by up to 0.6% in the case of doped fuels, and by up to 2% in the case of doped fuels with Cr layer on Zy cladding.

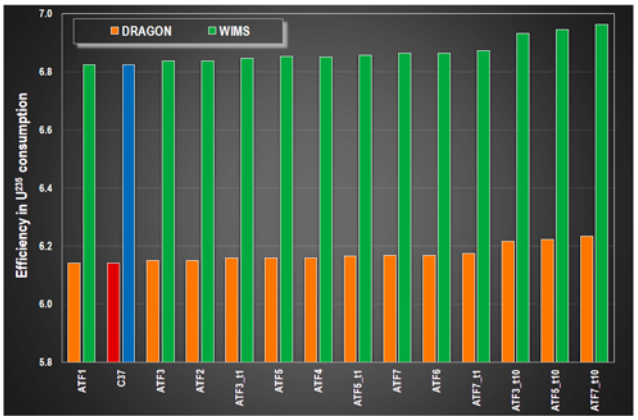


Fig. 7. U^{235} consumption (doped fuels with and without Cr layer on Zy cladding)

The void reactivity is an important parameter describing the evolution of cell reactivity during a LOCA (Loss of Coolant Accident). The void fraction (f_v) considered in this simulation was 99%, and the density of the coolant (considered homogeneous) decreases accordingly.

As it can be seen in Fig. 8, the cell reactivity increases when the cooling agent is lost for all analysed fuel bundles. For the doped fuels, the reactivity value decreases by up to 10% from the reference value for UO_2 fuel (C37 fuel bundle).

By adding a 10 μm thick Cr layer on the Zy cladding, the reactivity value decreases by up to 28% compared to the reference value corresponding to C37 fuel bundle without Cr coating.

The void coefficient (CVR) associated to CANDU reactor operation conditions is positive for fresh fuel: 15.39 mk/% (DRAGON estimate) and 16.77 mk/% (WIMS estimate). The void coefficient decreases to 11 - 12 mk/% for the spent fuel, at the discharge burnup, depending on the code used. Doped fuels were characterized by CVR values close to the reference value. The addition of a 10 μm Cr layer on Zy cladding still keeps a positive void effect, but with slightly lower values (by 0.9%).

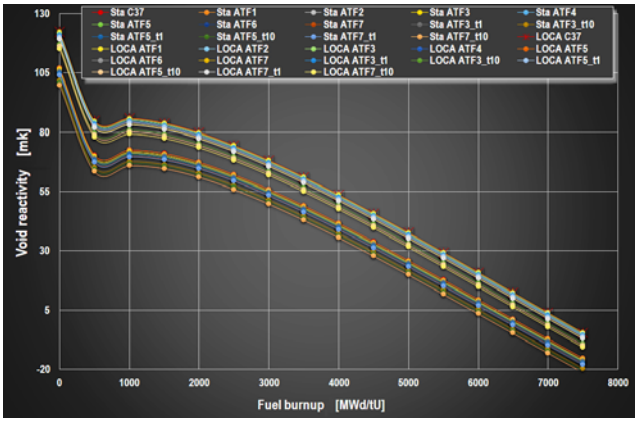


Fig. 8. Void reactivity in ATF fuel bundles

The temperature coefficients (cT) express the change in the cell reactivity as a result of the change in fuel temperature. For a CANDU reactor, cT is negative. Tables 3 and 4 presents the temperature coefficients calculated for fuel, the results exhibiting no influence coming from the use of dopants in the fuel composition or the addition of a thin layer of Cr on the Zy cladding.

Table 3. Temperature coefficients for doped fuels

Case ID	Fresh fuel		Spent fuel	
	DRAGON	WIMS	DRAGON	WIMS
C37	-0.0052	-0.0050	-0.0106	-0.0107
ATF1	-0.0052	-0.0050	-0.0106	-0.0107
ATF2	-0.0052	-0.0050	-0.0106	-0.0107
ATF3	-0.0052	-0.0050	-0.0106	-0.0107
ATF4	-0.0052	-0.0050	-0.0106	-0.0107
ATF5	-0.0052	-0.0050	-0.0106	-0.0107
ATF6	-0.0052	-0.0050	-0.0106	-0.0107
ATF7	-0.0052	-0.0050	-0.0106	-0.0107

Table 4. Temperature coefficients for doped fuels with Cr layer on Zy cladding

Case ID	Fresh fuel		Spent fuel	
	DRAGON	WIMS	DRAGON	WIMS
C37	-0.0052	-0.0050	-0.0106	-0.0107
ATF3_t1	-0.0052	-0.0052	-0.0106	-0.0105
ATF3_t10	-0.0050	-0.0050	-0.0104	-0.0103
ATF5_t1	-0.0051	-0.0051	-0.0106	-0.0105
ATF5_t10	-0.0049	-0.0049	-0.0104	-0.0103
ATF7_t1	-0.0052	-0.0052	-0.0106	-0.0105
ATF7_t10	-0.0049	-0.0049	-0.0104	-0.0103

To evaluate the fissile inventory in the spent fuel bundles (in order to assess the benefit of a potential reprocessing) it is necessary to estimate the amount of Pu^{239} at the EoI.

Therefore, using Eq. (6), the amount of Pu^{239} existing in the ATF spent fuel bundles was determined. A comparison with the Pu^{239} inventory existing in the C37 spent fuel at EoI was also performed, there results being presented in Fig. 9.

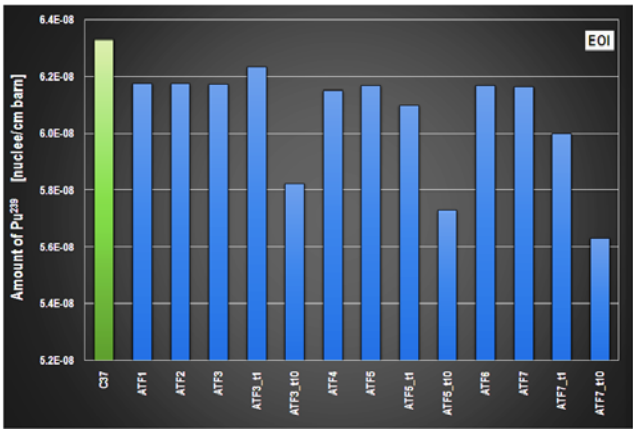


Fig. 9. Pu^{239} number densities [nuclei/cm·b] in the spent fuel for the considered fuel bundles (DRAGON estimation)

The amounts of the main minor actinides (Am, Cm and Np) contained in spent fuel were also estimated at EoI based on Eq. (6), and the results are presented in Figs. 10, 11 and 12, respectively. Similar behaviour can be observed in both plutonium and minor actinides cases.

The quantities of minor actinides found in the ATF spent fuels at EoI are lower compared to those corresponding to the standard fuel bundle, C37, thus reducing the attractiveness for reprocessing the spent fuel.

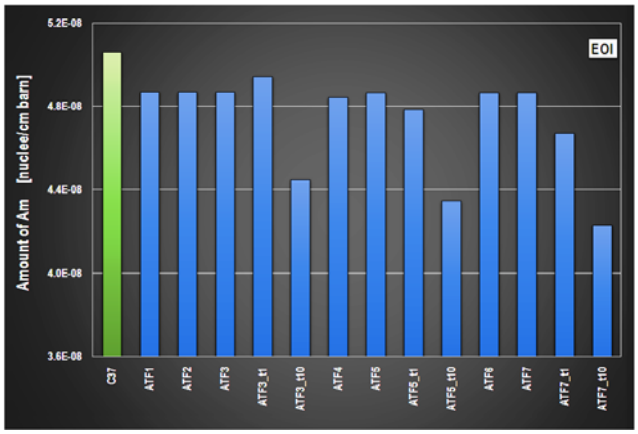


Fig. 10. Am number densities [nuclei/cm·b] in the spent fuel for the considered fuel bundles (DRAGON estimation)

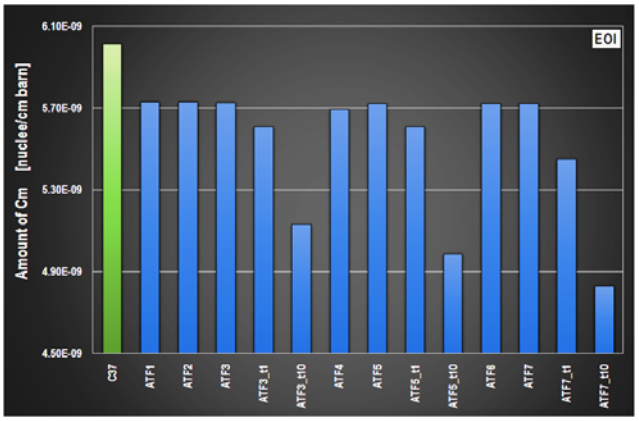


Fig. 11. Cm number densities [nuclei/cm·b] in the spent fuel for the considered fuel bundles (DRAGON estimation)

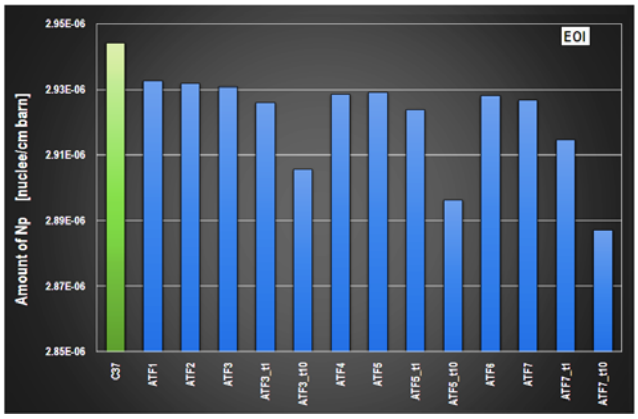


Fig. 12. Np number densities [nuclei/cm·b] in the spent fuel for the considered fuel bundles (DRAGON estimation)

Conclusions

A comparative neutronic analysis on the main parameters of interest at lattice cell level was presented for CANDU fuel bundles with 37 elements which contain natural uranium standard fuel or accident tolerant fuel, ATF.

The ATF was obtained considering two approaches:
• UO_2 fuel doped with chromium oxide or aluminium oxide, as well as a mixture of the two oxides, using different percentage of dopant,
• metallic chromium layer coating the Zy cladding, with different layer thicknesses.

Using the ATF fuel bundles (doped UO_2 fuel with and without Cr coated Zy cladding) in CANDU reactor does not significantly alter the neutronic features of the interest lattice parameters: maximum burnup, radial power peaking factors, void reactivity.

As expected, the use of dopants in the fuel composition led to penalties up to 2% in the maximum burnup. The addition of the Cr layer on Zy cladding decreased by up to 5.4% the maximum burnup comparatively with the reference value of C37 fuel bundle.

Doping the UO_2 fuel composition with Cr_2O_3 and/or Al_2O_3 and coating the Zy cladding with a Cr layer does not bring remarkable improvements in terms of uniform power distribution. The most likely cause is the homogeneity of the distribution of the amount of fissile in the fuel bundles.

The addition of dopants to fuel composition has led to a decrease in the efficiency of U^{235} consumption by up to 0.6%. The same parameter expressed for the Cr coated Zy cladding fuel bundles shown decreases of up to 2% in comparison with the reference, C37 fuel bundles.

The values of the void effect, for the considered doped fuel compositions and thicknesses of Cr layer on Zy cladding were positive, but slightly reduced in comparison with the values characterizing the reference, C37 standard fuel bundle.

The Doppler effect related to ATF fuel bundles (doped UO_2 fuel and Cr coated zirconium cladding) was negative, as the effect associated to the standard CANDU fuel, thus ensuring a very important safety feature.

The amount of Pu^{239} found in the doped spent fuel is up to 1.3% lower than in the case of C37 UO_2 fuel. The doped fuels with Cr coated Zy cladding led to a decrease of 2.5% more for the Pu^{239} inventory in the spent fuel.

A similar conclusion applies to the main minor actinides' inventories in the spent fuel at EoI. The ATF fuel contain up to 16% less americium, 20% less curium and 2% less neptunium, respectively.

From the point of view of neutronic cell calculations, the accident tolerant fuels analyzed in this paper work can be safely used in CANDU reactors

References

[1] Canadian Nuclear Association (2023) "Technology leadership: CANDU Reactors", https://cna.ca/research-and-advocacy/compete-globally/ .

[2] Bezdedeanu, G.R. (2020) "Neutronic characterization of CN43 CANDU type fuel bundles with natural uranium or reprocessed uranium", INR internal report, RI-12331, Institute for Nuclear Research (RATEN ICN) Pitesti, Romania (in Romanian).

[3] Bezdedeanu, G.R. (2021) "Comparative analyses on the neutronic characteristics of the CANDU type fuel bundles containing U-based fuel composition", INR internal report, RI-12821, Institute for Nuclear Research (RATEN ICN) Pitesti, Romania (in Romanian).

[4] Mărgeanu, C.A., Moise, A. (2021) "Monte Carlo simulations for innovative fuel bundles with increased number of elements and natural uranium fuel", INR internal report, RI-12700, Institute for Nuclear Research (RATEN ICN) Pitesti, Romania (in Romanian).

[5] Rizoiu, A.C., Aioanei, C.M., Olteanu, G. (2013) "Reactor Physics View on using Thorium-based Fuel Bundles in CANDU Reactors", Proceed. of Nuclear 2013 Int. Conf., Institute for Nuclear Research Pitesti, Romania.

[6] Neacsa, G.R. (2017) "The use of T37 Fuel Bundles with mixed oxide of thorium and uranium to increase the economic performance of the CANDU Reactor", INR internal report, RI-11329, Institute for Nuclear Research Pitesti, Romania (in Romanian).

[7] Olteanu, G. (2014) "Preliminary 37M fuel bundle project intended for CANDU reactors", INR tech. note, S1-NT327, Institute for Nuclear Research Pitesti, Romania (in Romanian).

[8] Olteanu, G., Ionescu, D.V., Roman, M.R., Răduț, A.C. (2019) "Preliminary C37M2 fuel bundle project intended for CANDU reactors", INR technical note, S1-NT348, Institute for Nuclear Research Pitesti, Romania (in Romanian).

[9] Olteanu, G., Horhoianu, G., Zvanciuc, F. (2006) "Updating the SEU43 fuel bundle project", INR internal report, RI-7427, Institute for Nuclear Research Pitesti, Romania (in Romanian).

[10] Olteanu, G., Roman, M.R., Ionescu, D.V., Răduț, A.C. (2020) "Preliminary CN43fuel bundle project intended for use in CANDU type reactors", INR technical note, S1-NT358, Institute for Nuclear Research (RATEN ICN) Pitesti, Romania (in Romanian).

[11] Rizoiu, A.C. (2018) "Neutronic characterization of advanced CANDU type fuel bundle project with 52 elements", INR internal report, RI-11697, Institute for Nuclear Research Pitesti, Romania (in Romanian).

[12] Rizoiu, A.C. (2018) "Neutronic characterization of advanced CANDU type fuel bundle project with 61 elements", INR internal report, RI-11678, Institute for Nuclear Research Pitesti, Romania (in Romanian).

[13] OECD/Nuclear Energy Agency (1998) "WIMS-D5, OECD/NEA Data Bank Documentation", Package ID No. 1507/02, Paris, France, http://www.nea.fr/html/dbprog/.

[14] Marleau, G., Hébert, A., Roy, R. (2013) ″A User Guide for DRAGON Release 3.06L″, IGE-174 rev. 12, Institute de Génie Nucléaire, École Polytechnique de Montréal, Canada.

[15] Leszczynski, F., López Aldama, D., Trkov, A. (2007) ″WIMS-D library update: final report of a coordinated research project″, STI/PUB/1264, International Atomic Energy Agency, Vienna, Austria.

[16] Khatib-Rahbar, M., Krall, A., Yuan, Z., Zavisca, M. (2021) "Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases", NUREG/CR-7282, ERI/NRC 21-203, US NRC.

[17] Arborelius, J., Backman, K., Hallstadius, L., Limbäck, M., Nilsson, J., Rebensdorff, B., Zhou, G., Kitano, K., Löfström, R., Rönnberg, G. (2006) "Advanced Doped UO_2 Pellets in LWR Applications", *Journal of Nuclear Science and Technology*, Vol. 43(9), pp. 967-976.

[18] Bischoff, J., Blanpain, P., Brachet, J-C., Lorrette, C., Ambard, A., Strumpell, J., McKoy, C. (2014) "Development of fuels with enhanced accident tolerance", in IAEA-TECDOC-1797 "*Accident Tolerant Fuel Concepts for Light Water Reactors*", IAEA, Vienna, Austria.

[19] Roy, R., Marleau G., Tajmouati, J., Rozon, D. (1994) "Modelling of CANDU reactivity control devices with the lattice code DRAGON", *Annals of Nuclear Energy*, Vol. 21(2), pp. 115-132.

[20] Spinks, N.J. (2011) "CANDU Nuclear Power Reactors", Thermopedia, https://www.thermopedia.com/ content/611/.

[21] Rizoiu, A.C., Horhoianu, G. (2011) "Preliminary Reactor Physics Studies on Using Advanced Fuel Bundles in CANDU", Procced. of Nuclear 2011 Int. Conf., Institute for Nuclear Research Pitesti, Romania.

[22] Garland, W.J. (editor) (2014) "*The Essential CANDU: A Textbook on the CANDU Nuclear Power Plant Technology*", UNENE, Dept. of Engineering Physics, McMaster Univ., Hamilton, Ontario, Canada, https://www.unene.ca/education/candu-textbook .