REACTOR PHYSICS ASSESSMENT OF MODIFIED 37-ELEMENT CANDU FUEL BUNDLES

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ABSTRACT

Reducing the central element diameter in order to improve the total flow area of CANDU fuel bundle and redistribute the power density of all remaining elements was studied in Canada and Korea when considering the effect of aging pressure tube diametral creep.
The aim of this paper is to study the modified bundle behavior using the transport codes WIMS and DRAGON. In calculations, a WIMS nuclear data library on 172 energy groups was used.
2-D transport calculations were performed with WIMS and DRAGON, leading to similar results in estimated cell parameters. Additionally, 3-D DRAGON calculations were carried on in order to evaluate the local flux distribution shift, as well as the incremental cross sections for supercells containing modified CANDU bundles and reactivity devices.
The overall effect of using modified fuel bundles was meaningless for both cell and supercell parameters, thus ensuring this possibility of fuel improvement for thermal-hydraulic purposes only.

Key words: DRAGON, WIMS, Modified CANDU Fuel

Introduction

Important studies were carried out in Canada ([1], [2], [15]) and Korea ([3], [4]) concerning the improvement of thermal hydraulic properties of CANDU bundles in fuel channel. One of the selected solutions was reducing the central element diameter. The power density of the remaining elements is thus redistributed. This phenomenon is particularly important for pressure tube diametral creep caused by aging, when the flow area is to be kept as constant as possible. As stated in [15], the remaining 36 fuel elements, end plates, bearing pads and spacers are unchanged. Demonstrative irradiations were performed by OPG for 37M bundles and the conclusions of the preliminary assessment were that for a CANDU core with a limited number of channels that contain 37M bundles, the thermal hydraulics features are not affected. The studies at both lattice cell and reactor core levels were carried out using WIMS-IST and RFSP-IST codes.
This paper intends to study the reactor physics assessment of the modified CANDU bundle at both cell and supercell levels, in the aim to prove that reducing the central element diameter, although important from the thermal hydraulics point of view, is rather meaningless for overall bundle neutron features.

2-D neutron transport calculations were performed for lattice cell assessment, i.e. to estimate the maximum burnup, the radial power distribution, as well as the temperature and void reactivity coefficients. WIMS ([6]) and DRAGON ([7]) transport codes were used, and the necessary nuclear data based on ENDF/B-VII were taken from a 172-group library issued from an IAEA sponsored project ([8]).

Additionally, 3-D DRAGON calculations were done in order to study the effects of burning modified fuel bundles on the supercells containing reactivity devices such as adjuster rods, mechanical control absorbers, light water zone control units and guide tubes. Incremental cross sections were calculated and local flux perturbations were determined.

Lattice cell description

The "reference" bundle (referred to as 37Mref) has the usual 12.22 mm pellet diameter, while the "reduced" (referred to as 37Mred1, 2, 3 and 4) cells have smaller central pellet diameter: 11.84, 11.34, 10.84 and 10.7 mm respectively. These bundles are intended to be used in a CANDU6 reactor; therefore, the lattice cell's characteristics - apart from those of the fuel bundle itself - correspond to a standard CANDU lattice cell, as presented in a specific benchmark problem ([9]). The considered cell power rating was 42 kW/kgU, which corresponds to a bundle power of about 800 kW. The lattice arrangement is given in Fig. 1 (DRAGON plot obtained using the `PSP:` module). For WIMS, the DSN (`SEQUENCE 1` option) method was used, while for DRAGON the `EXCELT:` tracking module was selected.

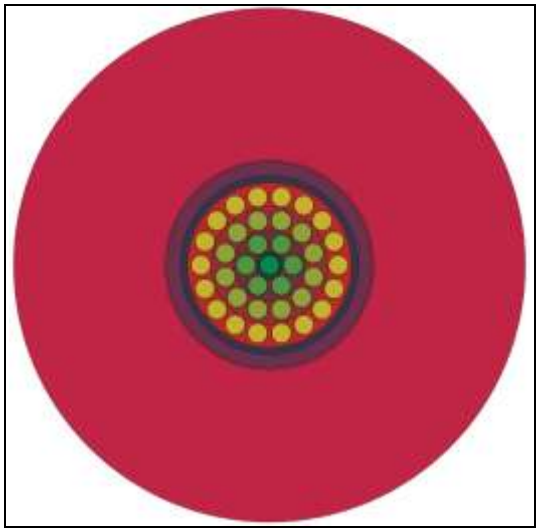


Fig. 1. *Lattice cell - DRAGON plot*

Main cell results

The main key parameter was the maximum burnup, i.e. the maximum burnup for which the cell is still critical.

The Radial Power Peaking Factors (RPPFs) are defined in [3] for each fuel ring k considering the Average Fuel Pin Power (AFPP) in ring k and the Bundle Average Fuel Power (BAFP) as:

$$RPPF_k = \frac{AFPP_k}{BAFP}, \text{ for } k = 1, 2, 3 \text{ and } 4. \quad (1)$$

The maximum burnup for the standard CANDU cell, as well as for the modified bundles, was about 6700 MWD/tU, for both codes used in calculations (Fig. 2). Therefore, this important parameter was not perturbed by reducing the central element diameter.

Another expected result was the RPPF behavior: the importance of the central element decreases with its diameter for the fresh fuel as well as for 3500 and 7000 MWd/tU burnup (Fig. 3). Both WIMS and DRAGON calculations offered the same values.

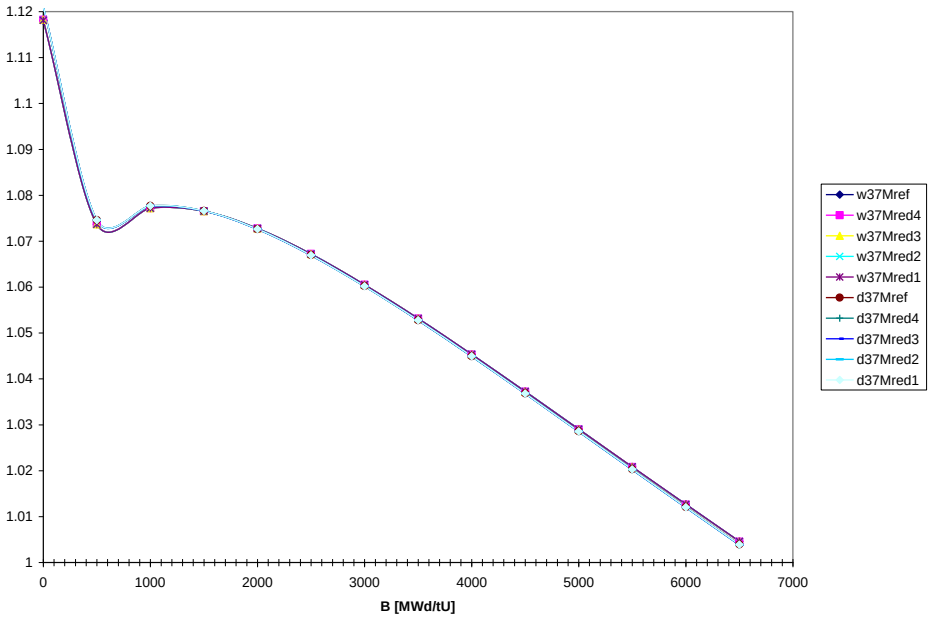


Fig. 2. *K-inf evolution with burnup (WIMS and DRAGON calculations)*

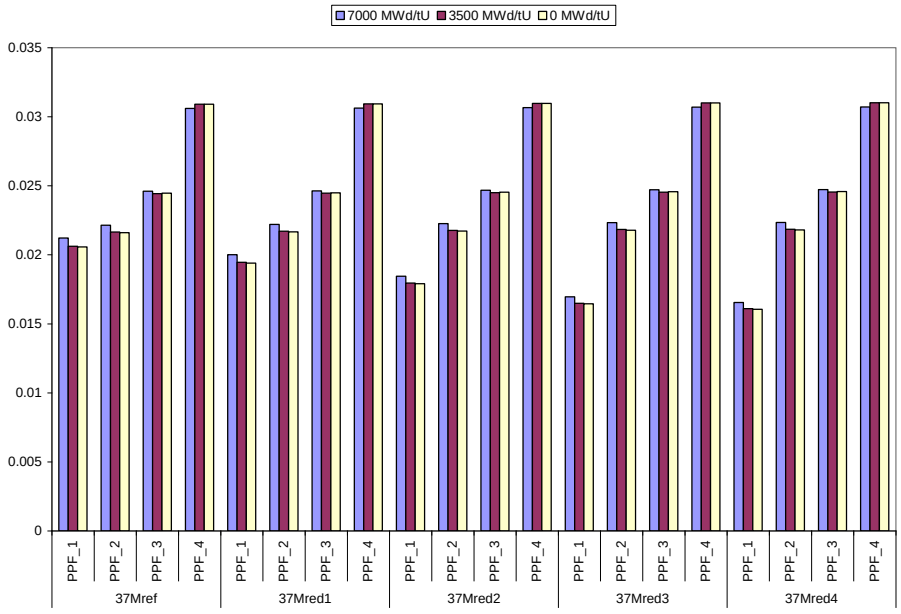


Fig. 3. *Power Peaking Factors for the considered fuel bundles at different burnup steps*

The temperature and void reactivity coefficients were estimated as "instantaneous perturbations" ([5]) for fresh fuel cells for which fuel, moderator and coolant temperatures, as well as coolant density were modified and the cell reactivity was estimated.

The temperature coefficients were evaluated by modifying the nominal (design) values with ± 5% and ± 10%, running the 2-D codes and calculating

$$Coeff\_Temp = \frac{\rho(T+\Delta T)-\rho(T)}{\Delta T} \tag{2}$$

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

$$dens_{fv} = dens_0 \cdot \left(1-\frac{fv}{100}\right) \tag{3}$$

The void reactivity is then calculated according to the directions in [3] as:

$$\Delta\rho_{fv} = \left(\frac{1}{keff_0}-\frac{1}{keff_{fv}}\right) \tag{4}$$

using the appropriate multiplication constants obtained for "cooled" and "voided" cells respectively.

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

$$Coeff\_Void = \frac{\Delta\rho(fv)-\Delta\rho(0)}{fv} \tag{5}$$

The reactivity ρ is expressed either in 10^{-3} (mk) or in 10^{-5} (pcm = per cent mille) fractions of $\frac{\Delta k}{k}$, therefore the temperature coefficients are expressed in mk/K or pcm/K, while the void coefficient is given in mk/% or pcm/% ([5], [10]).

The fuel temperatures considered in this study were *1209.15 K* and 1088.24, 1148.69, 1269.61 and 1330.07 K respectively. The moderator temperatures ranged around *339.16 K*: 305.24, 322.20, 356.12 and 373.08 K respectively. The coolant temperatures used in this analysis were *561.66 K*, 505.49, 533.58, 589.74 and 617.83 K respectively.

Table 1 presents the calculated temperature coefficients for fuel, moderator and coolant respectively, the results exhibiting a negligible influence of reducing the central element diameter.

Tab. 1. *Temperature coefficients estimated for different burnup values (0, 3000 and 6000 MWd/tU)*

Fuel type	CT_{fuel} (pcm/K)			CT_{mod} (pcm/K)			CT_{cool} (pcm/K)		
	0	3000	6000	0	3000	6000	0	3000	6000
37Mref	-0.506	-0.382	-0.320	-8.494	-1.272	2.941	2.709	4.035	5.135
37Mred1	-0.506	-0.382	-0.320	-8.494	-1.272	2.941	2.709	4.035	5.135
37Mred2	-0.506	-0.382	-0.320	-8.494	-1.272	2.941	2.709	4.035	5.135
37Mred3	-0.506	-0.382	-0.320	-8.494	-1.272	2.941	2.709	4.035	5.135
37Mred4	-0.506	-0.382	-0.320	-8.494	-1.272	2.941	2.709	4.035	5.135

The effect of fuel burnup on the estimated temperature coefficients is consistent with previous analyses ([11]). The fuel temperature coefficient decreases with burnup, while the moderator temperature coefficient increases with the burnup and becomes positive at high burnups. The coolant temperature coefficient is positive for any burnup, significantly increasing for end of life burnup.

Tab. 2. *Void coefficient estimated at different burnup values (0, 3000 and 6000 MWd/tU)*

Fuel type	CVR (pcm/K)		
	0	3000	6000
37Mref	15.158	12.511	12.352
37Mred1	15.158	12.511	12.352
37Mred2	15.158	12.511	12.352
37Mred3	15.158	12.511	12.352
37Mred4	15.158	12.511	12.352

The void coefficient estimated for all considered fuel bundles was not influenced by the central element reduced diameter. The effect of fuel burnup also confirms previous results, including the strong decreasing of the void coefficient with burnup ([11]).

Additionally, three flux-weighting factors, named THERM, RES and FAST ([12]) were estimated for all fuel cells (and also for all void fractions), to be used in further ORIGEN-S depletion calculations that usually follow fuel performance studies. Fig. 4 shows that the results based on DRAGON calculations are not influenced by the design perturbation, but RES and FAST values exhibit 10-20% differences given by codes behavior at high void fractions.

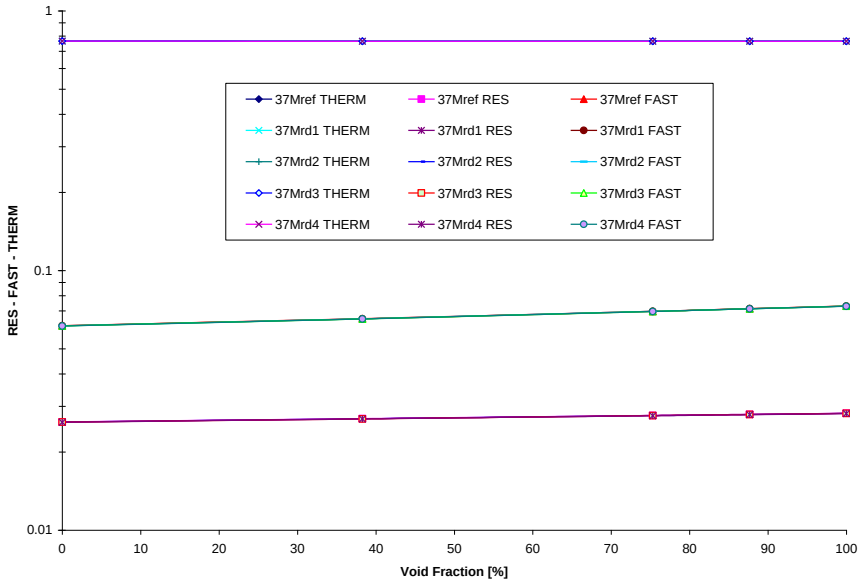


Fig. 4. *Estimated void flux-weighting factors THERM (up), RES (middle) and FAST (down)*

Supercell description

The "rod-like" reactivity devices in CANDU reactors (adjuster rods, control rods, shut-off rods and light water zone control units) are all inserted as in Fig. 5, therefore the DRAGON supercell model looks very much the same for all of them, still taking into account for the specific geometry and materials.

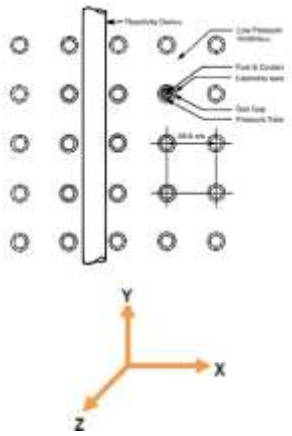
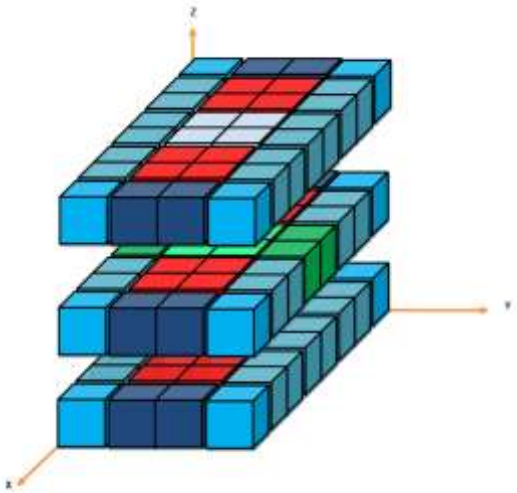


Fig. 5. *Generic CANDU "rod-like" reactivity device layout (from [13])*

The DRAGON supercell consists of two fuel bundles located in adjacent channels and a reactivity device inserted along the OY axis. The model consists of 5x3x3 = 45 cuboid blocks containing moderator (blocks A, B, C and E), "homogenized fuel" (fuel, sheaths and coolant) surrounded by "homogenized tubes" (pressure tube, CO2 gap and Calandria tube) in D blocks and reactivity devices surrounded by guide tubes and moderator in blocks F and G. In D blocks, the coaxial cylinders representing the fuel and tubes are placed along the OZ axis, in the middle of a moderator cuboid. The same, in the blocks F and G, the coaxial cylinders containing the reactivity devices and their guide tubes are placed along the OY axis, also in the middle of moderator cuboids. The blocks side lengths are whole multiples of LP/4 or BL/3 in order to strictly obtain a periodic reactor lattice. (LP = Lattice Pitch = 28.575 cm, BL = Bundle Length = 49.53 cm).



Block type = Number of blocks	Side length [cm]		
	X	Y	Z
A = 12	7.14375	7.14375	16.51
B = 16	14.2875	7.14375	16.51
C = 6	7.14375	14.2875	16.51
D = 6	14.2875	14.2875	16.51
E = 2	14.2875	14.2875	16.51
F = 2	14.2875	7.14375	16.51
G = 1	14.2875	14.2875	16.51

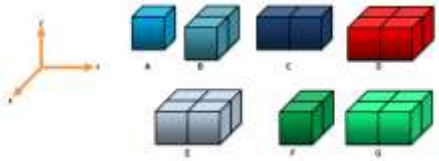


Fig. 6. *DRAGON supercell model for "rod-like" reactivity devices (from [14])*

In CANDU reactor physics, the neutron flux behavior in the vicinity of a reactivity device is expressed in terms of "perturbed" neighboring cells i.e. of cells with averaged macroscopic cross sections affected by (usually) small perturbations called incremental cross sections:

$$\Sigma^{pert} = \Sigma^{ref} + \Delta\Sigma \tag{6}$$

The main goal of a DRAGON supercell calculation is to estimate the multigroup $\Delta\Sigma$'s corresponding to the transport, absorption, yield and removal cross sections, as well as the modified "power-to-flux ratios". The neutron flux distribution is also supposed to be perturbed by the reactivity devices, yet the amplitude of the perturbation (and consequently the reactivity worth of the device) depends on the "reference" cell cross sections and consequently on the fuel type.

Incremental cross sections were estimated using DRAGON for the most important reactivity devices in CANDU i.e. adjuster rods segments A-inner, A-outer, B, C-inner, C-outer and D, mechanical control absorbers (identical to long shut-off rods), guide tubes, detector tubes and zone control units. In this respect, homogenous supercell calculations were performed, to evaluate the "reference" and "perturbed" sections in equation (6). The computed increments were subject to "permanent perturbations" ([5]), as they were estimated at B=0, 3200 and 6400 MWd/tU, thus covering the whole burnup range. Fig. 7 shows the low influence of the fuel bundle modification on the calculated incremental cross sections (fast transport, fast absorption, fast yield, removal, thermal transport, thermal absorption and thermal yield respectively) for the most effective reactivity device i.e. the mechanical control absorber (MCA).

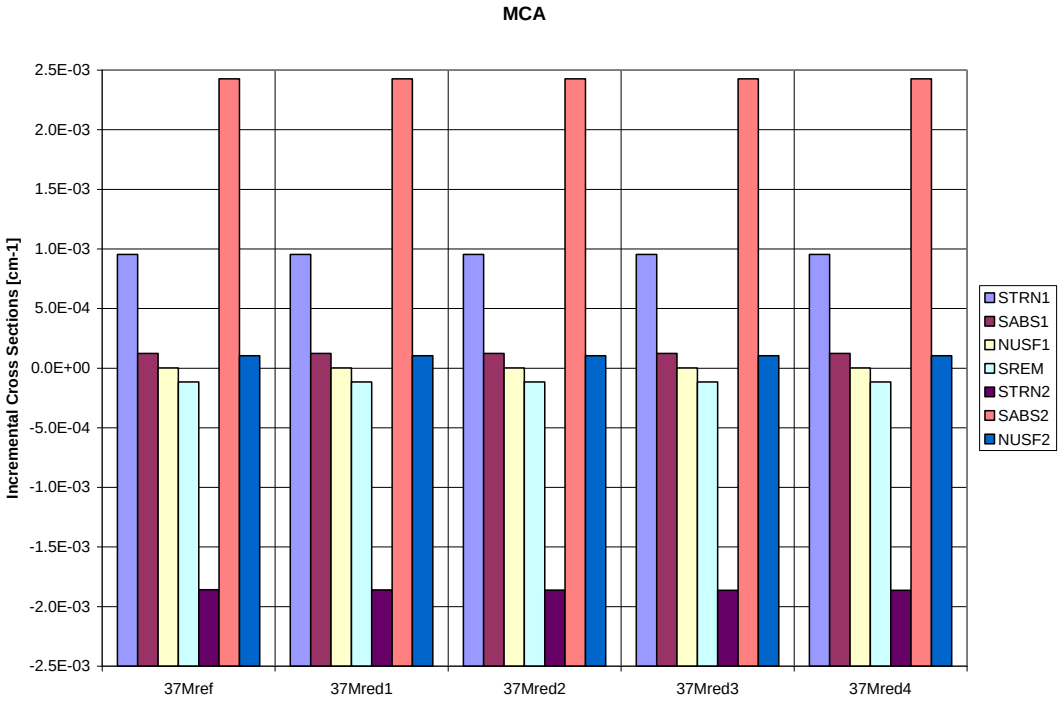


Fig. 7. *Incremental cross sections for MCA*

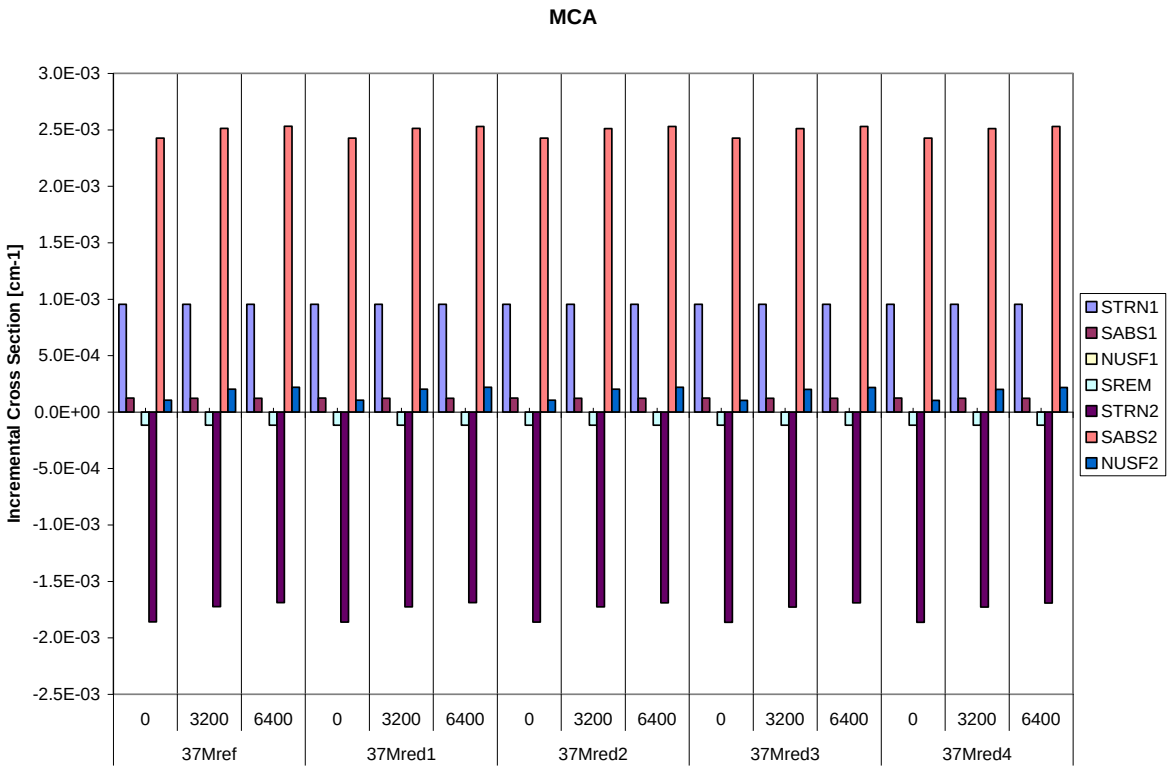


Fig. 8. *Incremental cross sections for MCA for different burnup values*

As seen in Fig. 8, the influence of fuel burnup on incremental cross sections is identical for all considered bundles.

Separate supercell calculations (for flux details purposes) were also performed for all considered fuel bundles and reactivity devices, in order to evaluate the thermal flux decrease in the presence of hard absorbers (Fig. 9).

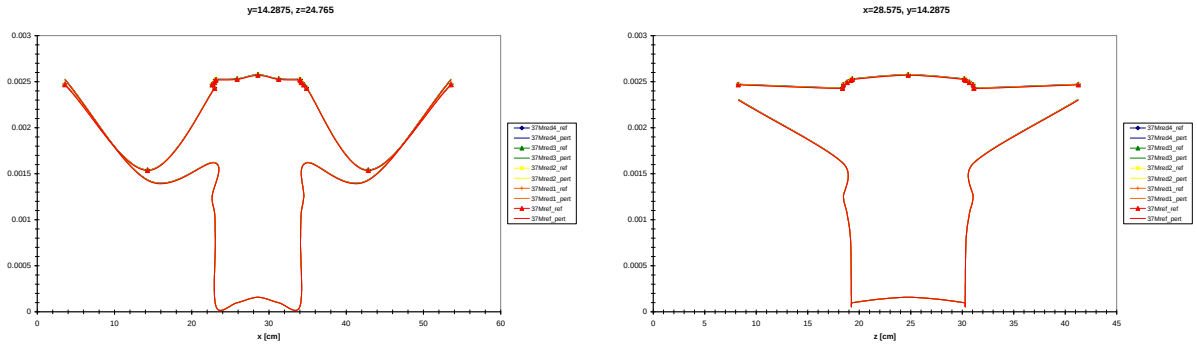


Fig. 9. *Thermal flux behavior near MCA along the OX (left) and OZ (right) axes; arbitrary flux units*

Conclusions

The most important result of this work was the qualification of the public domain computer codes WIMS and DRAGON for cell and supercell calculations regarding modified 37-element fuel bundles.

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 and flux-weighting factors.

Consequently, the supercell results (incremental cross sections and neutron flux shape perturbation) for the most important reactivity devices are not influenced by the central fuel element diameter reduction.

Therefore, the positive sub-channel results and the dry-out power enhancement presented in [3] can be applied to aging power plants without caring about the neutron balance alteration by this slightly bundle design changing.

However, for the full core transition (using 37M in all fuel channels), re-assessments are needed to demonstrate the improvement of the safety margins (SBLOCA, LBLOCA and LOR events).

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