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IMPACT OF THE 37M FUEL DESIGN ON REACTOR PHYSICS CHARACTERISTICS

R. Perez[1,*] and P. Ta[2]
[1]AMEC NSS, 4th Floor, 700 University Avenue,
Toronto, Ontario, M5G 1X6, Canada
[2]Bruce Power NSAS, 123 Front St., Toronto, Ontario, M5J 2M3, Canada
[*]Corresponding Author: Email ricardo.perez@amec.com, Phone (416) 592-4959

ABSTRACT – For CANDU nuclear reactors, aging of the Heat Transport System (HTS) leads to, among other effects, a reduction on the Critical Heat Flux (CHF) and dryout margin. In an effort to mitigate the impact of aging of the HTS on safety margins, Bruce Power is introducing a design change to the standard 37-element fuel bundle known as the modified 37-element fuel bundle, or 37M for short. As part of the overall design change process it was necessary to assess the impact of the modified fuel bundle design on key reactor physics parameters. Quantification of this impact on lattice cell properties, core reactivity properties, etc., was reached through a series of calculations using state-of-the-art lattice and core physics models, and comparisons against results for the standard fuel bundle.

Introduction

Continuous operation of the nuclear reactors at Bruce Power site has resulted in the aging of their Heat Transport System (HTS) causing, among other effects, a reduction on the Critical Heat Flux (CHF) and dryout margin.

In an effort to mitigate the impact of aging of the HTS on safety margins Bruce Power is pursuing the implementation of a fuel design change which involves primarily reducing the central fuel element size (diameter) from that for the standard 37-element fuel bundle (37R). The new fuel bundle design is referred to as the modified 37-element fuel bundle, or 37M for short.

To support the Safety Analysis Case for implementation of the 37M fuel bundle in Bruce Nuclear Generating Station Units 3 and 4, design calculations of the impacts of the design change on predicted reactor physics characteristics of the fuel at the lattice cell level and the core as a whole are needed.

This paper summarizes development of the WIMS[1] lattice cell model, the device incremental cross-sections using DRAGON[2], and the RFSP[3] core model corresponding to Bruce Power's 37M fuel bundle design. Reactor physics characteristics for the new fuel bundle design, based on lattice cell and full core calculations, including radial power profiles, reactivity device worth and static core reactivity effects due to single parameter changes are provided. Conclusions of the study are then presented.

[1] Neutron transport code (WIMS - Winfrith Improved Multigroup Scheme).
[2] Neutron transport code.
[3] Neutron diffusion code (RFSP - Reactor Fuelling and Simulation Program).

WIMS models

The design of the 37M fuel bundle is primarily based on a reduction in the size of the central fuel element in the bundle to increase the cross sectional area of the surrounding coolant subchannels [1]. The specifications of the 37M fuel bundle design were prepared by Bruce Power.

In general, the specifications of design parameters of relevance to the modelling of the 37M fuel bundle design using the reactor physics codes are identified by a range. The range makes allowance for such things as manufacturing variability. For the purpose of the study presented here such variability was not taken into account and therefore nominal dimension values were selected for the construction of the WIMS models.

The WIMS models prepared for this study assumed that:

- When the bundles are placed inside the reactor core at normal operating conditions the sheath will collapse onto the fuel pellet, closing the gap that is left in the fabricated bundles. The value of the outer radius of the collapsed sheath is calculated by conserving the mass of Zircaloy-4 of the original sheath.
- Assuming the fuel density is the same for 37R and 37M, the reduction of the central element diameter leads to a proportional reduction of the UO_2 mass.
- The 37M fuel bundle design includes no new materials relative to the 37R fuel bundle, as the changes are all related to the central element which remains of a standard Zircaloy-clad UO_2 configuration.
- The fuel composition used in the WIMS models for both fuel designs is the one corresponding to natural uranium.

The WIMS models were used in two ways, the first one was to complete simulations describing lattice cell based fuel characteristics and the second one was to produce the WIMS tables containing the macroscopic cross section tables to be used in the full core simulations with RFSP.

Lattice cell simulations

The reduction in the size of the central fuel element implies a small reduction in the mass of UO_2 in the bundle. This reduction in mass of uranium has a direct, albeit small impact on some of the fuel characteristics as detailed here.

It is important to note that due to the design change, the rate at which the fuel burns in the 37R and 37M fuel bundles is slightly different. In the burnup simulations in WIMS, the same bundle power was used for both designs and since there is less fuel mass (smaller total cross sectional area of the fuel elements) in the 37M bundle, the number of fission events per unit of uranium mass within an arbitrary unit of time (burnup) needed to produce the required power will be higher. Therefore using the same series of time steps in the burnup calculation, WIMS produces fuel burnups that are slightly higher for the 37M fuel design when compared to those produced for the 37R fuel design. This difference increases from 0.0 MWh/kgU at 0.0 days to 3.05 MWh/kgU after 647.5 days of fuel burnup simulation. Since these differences in burnup are small, for comparison purposes, results obtained after the same burnup simulation time are used.

Fuel isotopic composition can change in the long term as a result of irradiation, and can change in the short term as a result of operational transients.

Figure 1 presents the variation of the lattice cell reactivity as a function of the fuel burnup (assuming irradiation at a constant power level) due to the long term changes in the fuel isotopic composition. Although the differences in reactivity between the two fuel designs at a given burnup simulation time are small (Figure 2 shows a maximum difference of 0.93%), the reduction in reactivity in the 37M lattice cell is slightly faster. These differences are in part due to the method of comparison, and the differences would be reduced if the calculations were performed for consistent bundle burnup steps instead of time steps. Irrespective of the method of comparison, the differences are small, particularly in relation to the estimated uncertainty in the WIMS-predicted isotopic concentrations of 2-4% [2].

Changes in the isotopic composition of the fuel can occur also in the short term due to operational transients, such as changes in power level arising from shutdown, setback and startup. During these transients the concentration of fission products can decay, build up or reach a new equilibrium according to the flux level change.

To assess the impact of the fuel design change on the saturated fission products effect, the reactivity worth (xenon load) due to the ^{135}Xe concentration was simulated with fuel at different burnup points: low burnup (fresh fuel, i.e., 0 MWh/kgU), mid-burnup and high burnup. The mid-burnup and high-burnup cases considered for this comparison were 101 MWh/kgU and 212 MWh/kgU for 37R fuel and 102 MWh/kgU and 213 MWd/TU for 37M fuel, respectively.

The results presented in Table 1 demonstrate that the impact of the fuel design change on reactivity worth is small (the maximum difference in reactivity worth is about 0.01 mk). This is as expected since the production of ^{135}Xe is driven by the power levels at which the simulations are performed and by the actual fissile concentration on the fuel, leading to the reactivity effects due to ^{135}Xe being effectively saturated in CANDU reactors. In the simulations the power level was the same for both fuel designs and the difference in the ^{135}Xe concentration due to the design change is small.

The changes in fuel bundle design from the perspective of the impact on fuel isotopic change reactivity are shown to be sufficiently small that there will be no impact on normal reactor operation arising from the fuel design change. The changes are sufficiently small that it is judged that there will be at most a very small impact on accident analysis. The changes in bundle uranium mass will result in small increases in fuelling rates and impacts on bundle-power distributions due to fuelling operations.

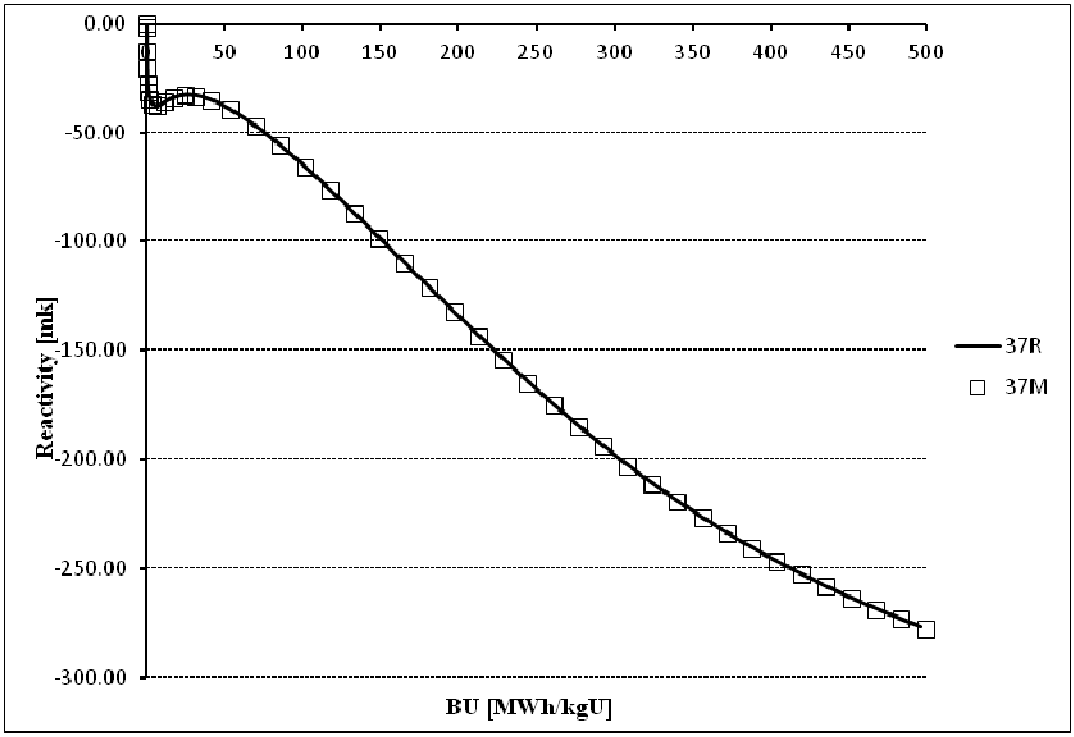

Figure 1: Lattice cell reactivity change as a function of the fuel burnup.

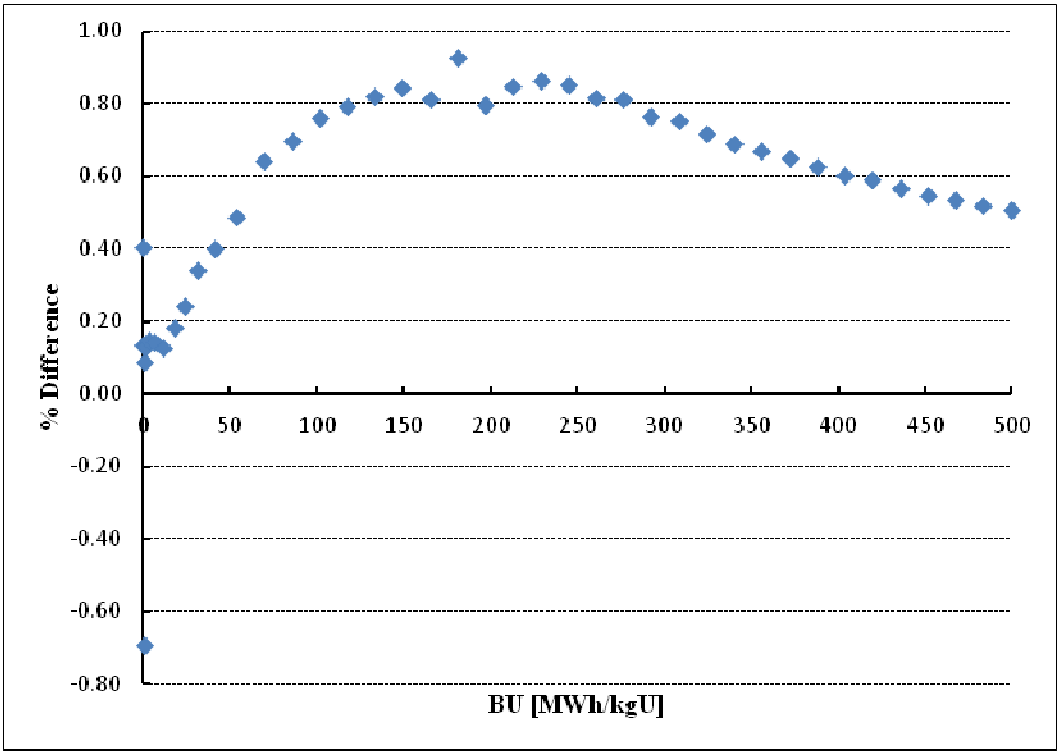

Figure 2: Percentage difference in the lattice cell reactivity of the 37M fuel with respect to the 37R fuel as a function of burnup.

BU	WIMS model	$k_{eff,ref}$	$k_{eff,actual}$	$\Delta\rho$ [mk]
	^{135}Xe Concentration	**Nominal**	**0.0**	
Low	37R	1.12422	1.12422	0.000
	37M	1.12418	1.12418	0.000
	Difference			0.000
Medium	37R	1.0471	1.07667	26.229
	37M	1.04652	1.07605	26.223
	Difference			-0.006
High	37R	0.96923	0.99376	25.468
	37M	0.96807	0.99254	25.467
	Difference			-0.001

Table 1: Reactivity changes associated with the changes in the ^{135}Xe concentration in the fuel. The ^{135}Xe concentration at the perturbation point is zero.

H-factors and radial power distribution

The fuel management operations for the 37M fuelled core are designed to achieve the same channel power distribution as in the core fuelled with standard 37-element fuel bundles and as such there is essentially no impact on the time-averaged channel power distribution for the converted core.

The implementation of the design change will have a small impact in the bundle power per unit cell flux, namely the fast and thermal H-factors. The H-factors are calculated by WIMS during the process of generating fuel tables and are used by RFSP and SORO[4].

The H-factors depend on the neutron spectrum and the actual concentration of fissile material via the macroscopic fission cross section. Therefore the reduction in the mass of UO_2 results in a reduction of the fast and thermal H-factors, H1 and H2. To a first order approximation, the change in the thermal-neutron H-factor (the bundle power to cell-average thermal flux ratio) is expected to be proportional to the change in the bundle mass. The change in H-factor is also influenced by the variation in the effectiveness of uranium at different locations in the bundle, and as the mass change associated with the modified 37-element bundle is at the center of the bundle where the thermal-neutron flux is at a minimum, the change in H-factor (particularly for H2) will be smaller than the change in mass.

Figure 3 and Figure 4 show a comparison between the dependence of the H-factors on fuel burnup for both fuel designs. The maximum reduction in the H-factors in the 37M fuel bundle with respect to the 37R fuel bundle is about 0.76 % and 0.36 % for H1 and H2, respectively[5].

[4] Neutron diffusion code (SORO - Simulation Of Reactor Operation).
[5] These differences are in part due to the method of comparison, as discussed earlier, and the differences would be reduced if the comparison were performed for consistent bundle burnup steps.

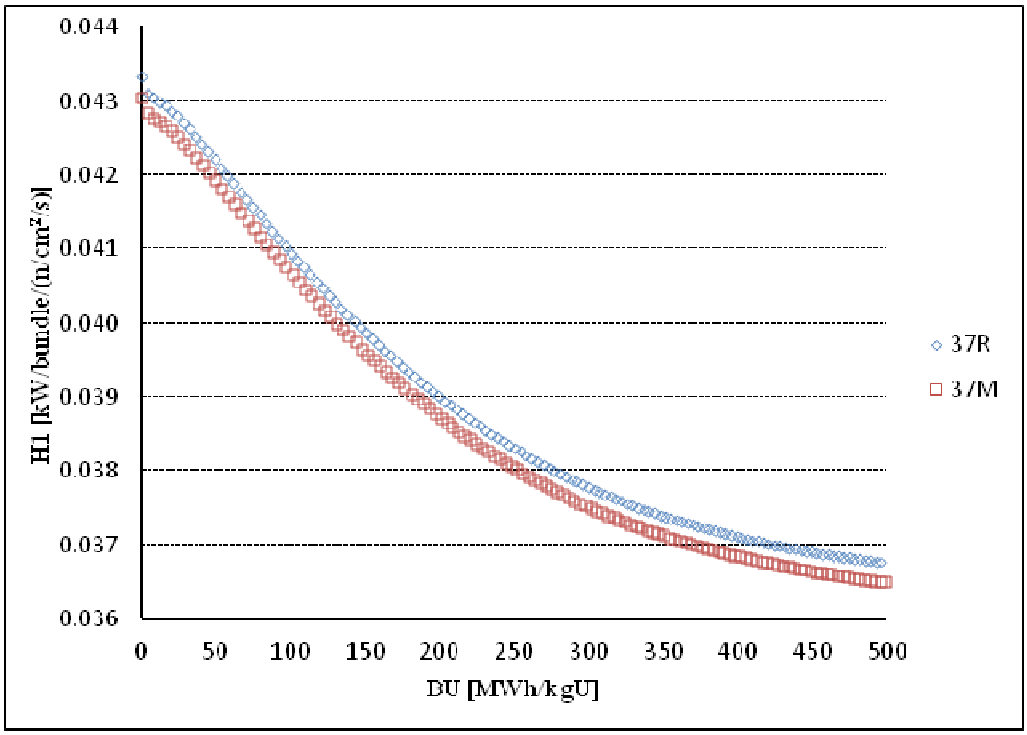

Figure 3: Comparison of Fuel Burnup dependence for Fast H Factor (H1) between 37R and 37M fuel bundles.

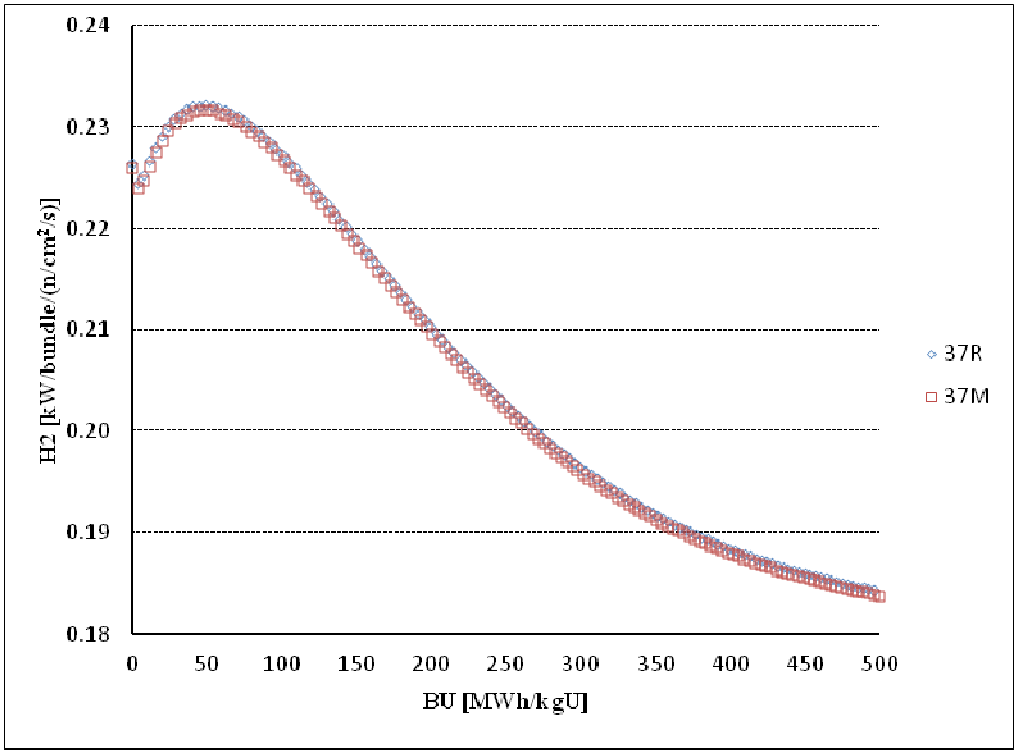

Figure 4: Comparison of Fuel Burnup dependence for Thermal H Factor (H2) between 37R and 37M fuel bundles.

The impact of this change in power-to-flux ratio will depend on the core configuration:

- The differences in fuelling ripple between the 37M and 37R fuelled cores will be minimal. Differences in lattice properties between the two bundle designs are small. As a result, both the cell averaged flux and the resultant bundle power differences will also be small. Thus, the impact on normal operation and safety analysis due to ripple related differences will similarly be small.
- The reduction in the thermal H-factor for the 37M relative to the 37R fuel bundles implies that a slightly higher flux is required to achieve the same power. In a transition core where both 37M and 37R fuel bundles are present in the reactor core, the potential exists for a 37R channel to be surrounded by 37M fuelled channels. In this scenario, higher flux and power may result in the 37R channel, which may affect the magnitude of the fuelling ripple of the channel with 37R bundles (i.e., ratio of the channel powers before and after a fuelling operation). Such impacts will be small based on the small differences in calculated thermal H-factors.

Within a fuel bundle, the flux and power distributions among the fuel elements in the radial direction are primarily determined by the fuel-pin and coolant sub-channel geometric configurations. The outer elements of the 37R fuel bundles have the highest power rating among the elements. In the axial direction, the flux and power profiles peak toward the interfaces between bundles. This axial peaking is dependent on the bundle-end-region dimension, material properties and separation between bundles.

The power distribution within the fuel bundle impact on two aspects of safety analysis: the power density distributions drive how the fuel temperature changes; and the linear power distribution of the fuel elements affects the calculations of dryout.

In terms of the radial power distribution inside the bundle, the design change has a small impact on the power density distribution with small increases in the relative power density distribution for the central and inner fuel elements (with maximum increases of about 0.95% and 0.32%, respectively) and reductions in the intermediate and outer fuel elements (with maximum reductions of about 0.14% and 0.30%, respectively) as shown in Figure 5. The actual power densities increase in all the fuel elements in the 37M fuel bundle design with respect to the values for the 37R fuel bundle design (for a given bundle power) with mean values of approximately 1.5%, 0.9%, 0.5% and 0.4% for the central, inner, intermediate and outer fuel elements.

The impact of the fuel design change on the relative linear power ratings is presented in Figure 6. The results show a large decrease in the relative linear power rating corresponding to the central element (close to 22%) and slight increases in the corresponding values for the inner, intermediate and outer elements of about 0.9%, 0.5% and 0.4%.

For most applications, the increase in the linear power rating of the outer elements is considered the most significant impact arising from the changes in intra-bundle power distribution since the outer elements are generally limiting from the point of view of fuel and fuel sheath temperatures under accident conditions.

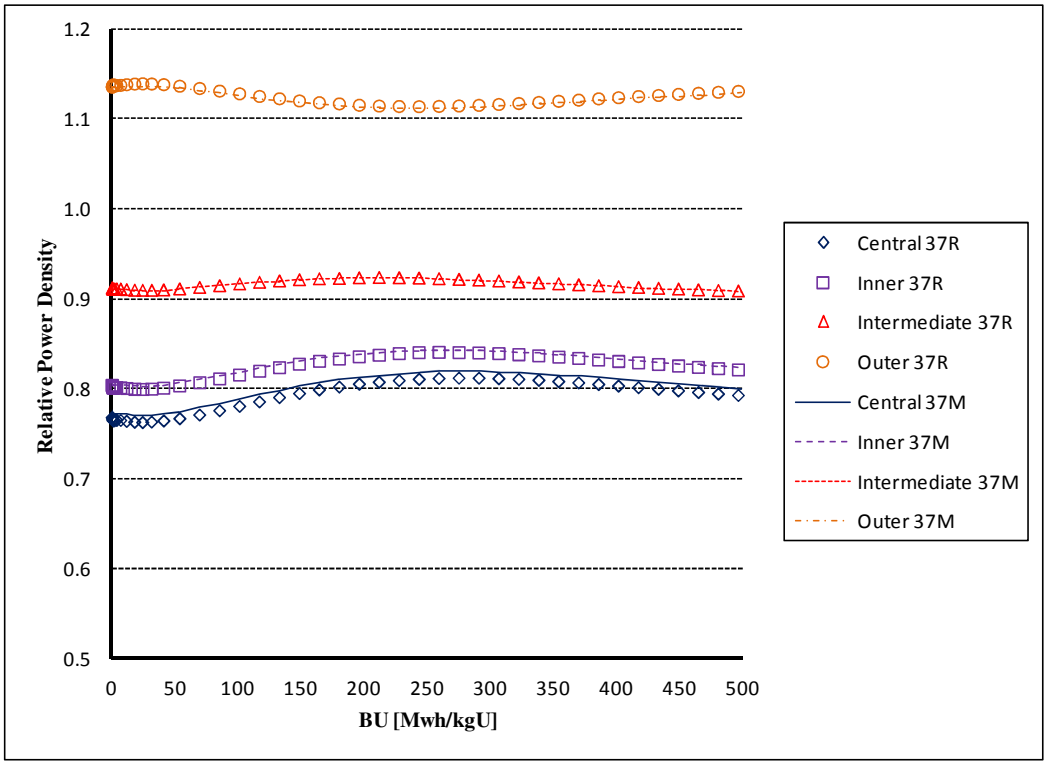

Figure 5: Relative Element Power Distribution as a function of Fuel Burnup

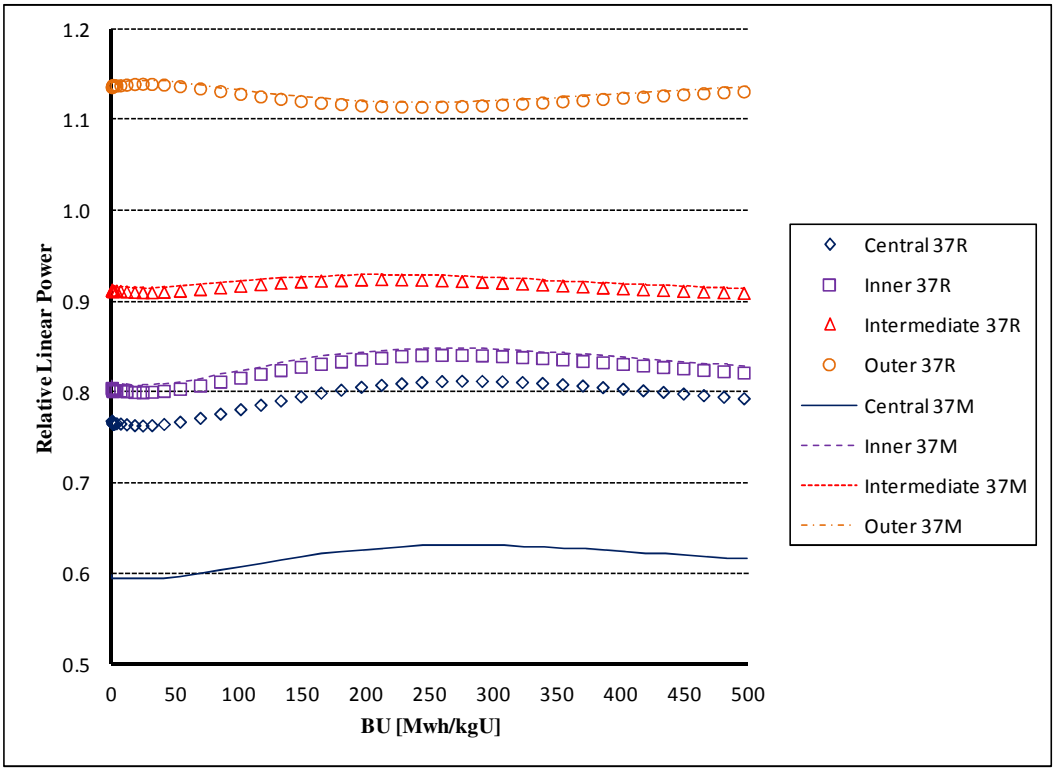

Figure 6: Relative Element Linear Power as a function of Fuel Burnup

Neutron flux in the pressure tube

The increase of the diameter of the pressure tube as a function of time in operating CANDU reactors is known as pressure tube diametral creep. One of the most important mechanisms influencing the diametral creep of the pressure tube is due to the collisions of fast neutrons with the pressure tube atoms producing changes on the underlying atomic lattice. The number of these collisions is related to the fast neutron fluence (flux integrated over time).

Figure 7 shows the fast fluxes in the pressure tube as a function of burnup, with the lines and the squares correspond to the 37R and 37M fuel bundle designs respectively. The change in the values of the fluxes is small and it cannot be readily distinguished in the figure. In general, the fluxes are higher for the 37M fuel design, with a maximum percentage increase of 0.2% when compared with the corresponding flux for the 37R fuel bundle design. Note that this percentage difference is well within the accuracy established in [2] regarding the flux-power spatial distributions in the radial direction through the fuel bundle.

Based on these results, the increase in the fast neutron flux at the location of the pressure tube arising from the fuel design change is sufficiently small that it is expected that the rate at which the pressure tube diametral creep develops will be insignificantly impacted.

Neutron spectrum

Negligible changes in the neutron spectrum are predicted due to the design change in the lattice cell. WIMS calculations were performed at low, medium and high burnup levels for both fuel designs.

Figure 8 presents the neutron spectra for both fuel designs at three burnup levels. The results for the 37M fuel show a decrease of about 0.2 % in the high energy end of the spectrum around 0.5-0.9 MeV and an increase in the range of 0.3 % - 0.4 % in the thermal energy region below 0.14 eV with respect to the results obtained for the 37R fuel design. The behaviour is consistent for all the calculated burnup levels.

The increase in the neutron flux in the thermal region can be attributed to the reduction in UO_2 mass in the 37M and assumption of constant power level. The decrease in neutron flux at high energies is a normalization effect.

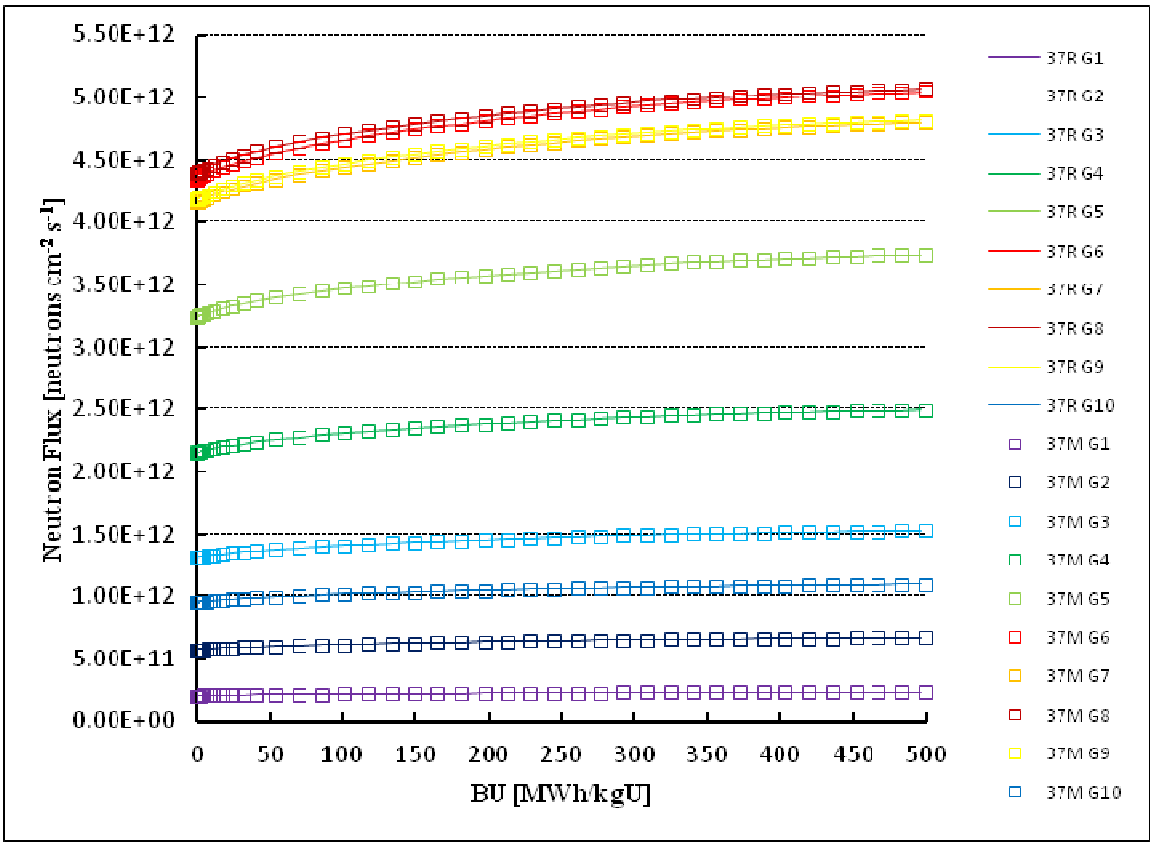

Figure 7: Average Fast Fluxes in the Pressure Tube[6].

[6] Only contributions of groups with energies larger than 1 MeV are accounted for. The lower cut-off energies are defined as (data taken from J. Griffiths, "WIMS-AECL User's Manual", COG-94-52, March 1994):

G1 ≥	7.7880 MeV
G2 ≥	6.0653 MeV
G3 ≥	4.7237 MeV
G4 ≥	3.6788 MeV
G5 ≥	2.8650 MeV
G6 ≥	2.2313 MeV
G7 ≥	1.7377 MeV
G8 ≥	1.3534 MeV
G9 ≥	1.0540 MeV
G10 ≥	0.8209 MeV

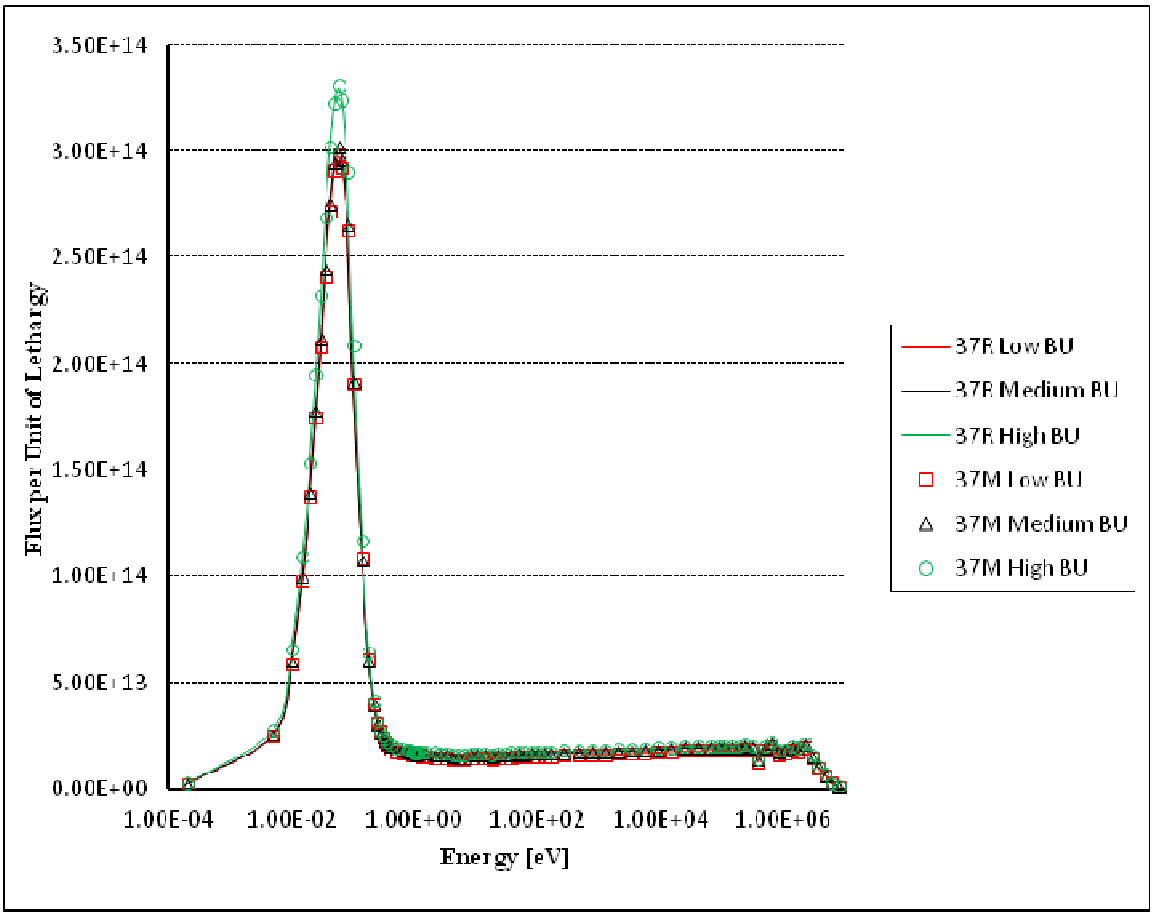

Figure 8: Neutron spectrum in the lattice cell for three different burnup levels

Incremental cross sections using DRAGON

The two energy group incremental cross sections for the Liquid Zone Controllers (LZC), Shutoff Rods (SOR) and Mechanical Control Absorbers (MCA) were re-calculated using DRAGON Version 3.04T and the "all-DRAGON" method [3] to account for differences between the 37R and the 37M fuel designs.

The percentage differences between the results for the two fuel types (37M relative to 37R fuel bundle designs) are in the range of -3.67% to +2.51%. No significant change in results for thermal absorption and thermal fission are observed. The largest differences are observed for the thermal fission for the top full and bottom full Liquid Zone Control compartments.

Time average reactor core

Time averaged (equilibrium) models of the reactor core are created in RFSP assuming fuelling using only one type of bundles, either 37R or 37M. The models share the same reactor geometry and structural material characteristics. Furthermore, the incremental cross sections used in the models correspond to the type of fuel that is represented in the core model. The same nominal values are used for all the parameters that define the state of the reactor (i.e. coolant density, moderator temperature, moderator isotopic purity, etc.).

The bundle irradiations are tuned to match a given target channel power distribution, for specific average core exit burnup. They are both independent of the type of fuel bundle that is present in the core. This reactor core model defines the reference state for the calculations described in the following sections.

Reactivity devices worth

The calculations of the reactivity worth for devices are performed under the assumptions that any perturbation introduced to the reference state of the reactor (insertion or withdrawal of the shutoff rods (SORs) or mechanical control absorber rods (MCAs), change in the average liquid zone controller (LZCs) fill level) is considered as an instantaneous perturbation and therefore the fuel irradiations are kept frozen. The response capability of the Reactor Regulating System (RRS) included in RFSP was not used. All calculations were performed using the *SIMULATE module from RFSP with the Simple Cell Methodology (SCM) and ensuring that no changes to the ^{135}Xe distribution occurs during the simulation of the perturbations.

The changes in core reactivity (from the reference state of the reactor) were calculated as the fill level of the LZCs was varied from 0% filled to 100% filled in intervals of 20% changes. For both fuel designs the reference state was assumed to have the LZC at 40% filled.

The estimated change in the LZC total reactivity worth (between the drained and filled states) for the 37M and 37R fuelled cores is negligible (0.01 mk). The impact of the design change (about 0.2 %) is smaller than the uncertainty in the control device reactivity worth calculation reported in Reference [2] (4.0%).

The MCAs are grouped by banks: Bank1 contain CAR03 and CAR04, and Bank2 has CAR01 and CAR02. Four cases were analyzed:

- Case 1: Bank1 half[7] inserted and Bank2 parked outside the core,
- Case 2: Bank1 fully[8] inserted and Bank2 parked outside the core,
- Case 3: Bank1 fully inserted and Bank2 half inserted,
- Case 4: and Bank1 and Bank2 fully inserted.

The impact of the fuel design change on the worth of the MCAs was small, with device worth slightly higher (i.e., 0.02 mk or approximately 0.3%) for the 37M fuelled core. Again, this change is smaller than the uncertainty in the code prediction.

The worth of the SORs was calculated for a nominal operation flux shape based on the assumption that 28 out of 30 SORs are fully inserted in the core. It is assumed that the most effective rod pair remains the same for the core fuelled with 37M bundles since the changes in neutron flux are small with regards to the configuration obtained for the nominal core fuelled with 37R fuel. This assumption of impairing the most effective pair of SORs is commonly used in the analysis of Large Break Loss of Coolant Accidents (LBLOCA) in order to conservatively simulate accident consequences.

[7] MCA Half inserted means that the bottom tip of the rod is located at the calandria horizontal mid plane.
[8] MCA Fully inserted means that the rod center is located at the calandria horizontal mid plane.

The results obtained for the worth of these 28 SORs show that the impact of the design change amounts to 0.05 mk (about 0.2 %).

The small changes in device reactivity worths resulting from the fuel design change are considered sufficiently small that there will not be any impact on normal reactor and fuelling operation, nor in accident analysis.

Static reactivity coefficients

Starting from the reference state of the reactor, a series of static full core calculations were performed for reactors fuelled with 37R and 37M fuel bundles using the *CERBERUS module from RFSP combined with SCM. Each series corresponds to the variation of a single parameter associated with a specific phenomenon from the set of Reactor Physics Validation Phenomena [2]. The impact of the fuel design change is assessed based on the difference in the reactivity response of each core to the introduced perturbations and on the difference in the estimated static reactivity coefficients calculated near the reference state. Comparison of the differences in the calculated reactivity coefficients with the uncertainties from validation recommended for each one of the phenomena under study provides support to the assessment and context against which to assess the impacts of the fuel design change.

The key assumptions for the calculations are as follows,

- The thermal hydraulics parameters were uniform throughout the core and the fuel design change would have an insignificant impact on the reference core state,
- Any perturbation introduced to the core parameters from their nominal values is considered instantaneous, allowing for no changes in the irradiation distributions through the core,
- A bias of -1.3 mk in the Coolant Void Reactivity (CVR) was used in the calculation of the DENSCALE parameter for both fuel types and,
- Estimates of the reactivity coefficients are calculated in the vicinity of the reference point by taking either the forward, backward or the mean value of the forward and backward finite difference. The estimates obtained are valid only within the range determined by the points used in its calculation. The estimates of the reactivity coefficients that are presented in this document are consistent with the precision of the effective neutron multiplication constant computed by RFSP and the assumed precision associated with the corresponding parameter under study.

The parameters and their corresponding range of variation used in the reactivity effects calculations are shown in graphical form. The reactivity change associated with each perturbation is calculated using the equation:

$$\Delta\rho = \frac{1}{k_{eff}^{0}} - \frac{1}{k_{eff}}$$

where k_{eff}^{0} is the effective multiplication constant obtained at the end of the last iteration in the *CERBERUS module for the nominal steady state, and k_{eff} is the effective multiplication constant obtained in the same way after the perturbation is introduced.

For those phenomena where the change in reactivity as a function of the corresponding parameter is close to a linear dependence, results of linear regressions are presented. The results of the linear regressions provide global estimate of the reactivity coefficients.

Coolant density

The results presented graphically in Figure 9 show an increase in the full core Coolant Void Reactivity (CVR) of 0.11 mk in the case of the 37M fuel with respect to the 37R fuel. This change in full core CVR can be understood from the reactor physics point of view based on the fact the reduction in the dimensions of the central fuel element results in an increase in the channel coolant volume. The change is considered small when compared with the uncertainty of ±1.1 mk attributed in Reference [2] to the random component in CVR calculations. The estimated value for the reactivity coefficient is the same for both fuel designs -15.3 mk/(g/cm^3) to the precision used in this study.

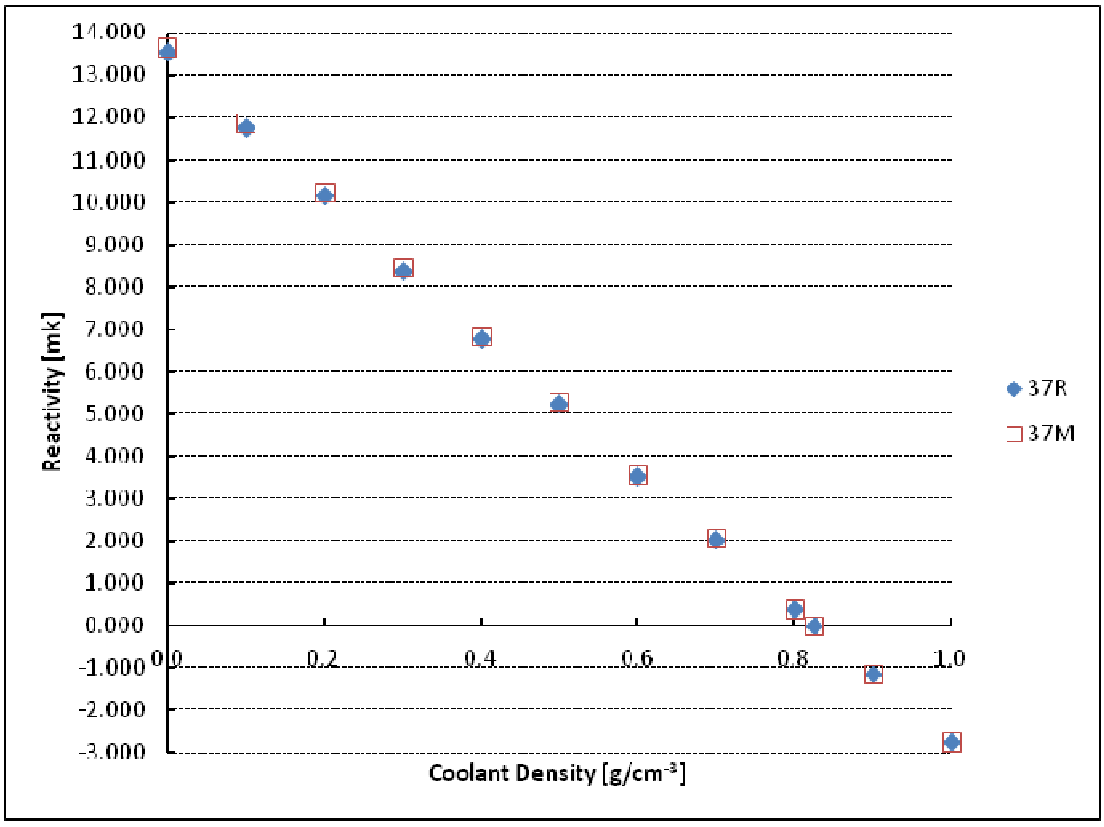

Figure 9: Reactivity change from the nominal conditions as a function of Coolant Density

Coolant temperature

The reactivity changes due to variations in the coolant temperature are generally small when compared with other reactivity effects. Figure 10 presents the impact of the fuel design change on coolant temperature reactivity as calculated by RFSP. The estimated reactivity coefficient for the 37M fuel design is marginally higher than the one calculated for the 37R fuel, but effectively (to the precision used in this study) they are the same (0.025 mk/^{o}C). This difference in coolant temperature reactivity coefficient is also small when compared with the recommended random uncertainty of 8% from Reference [2].

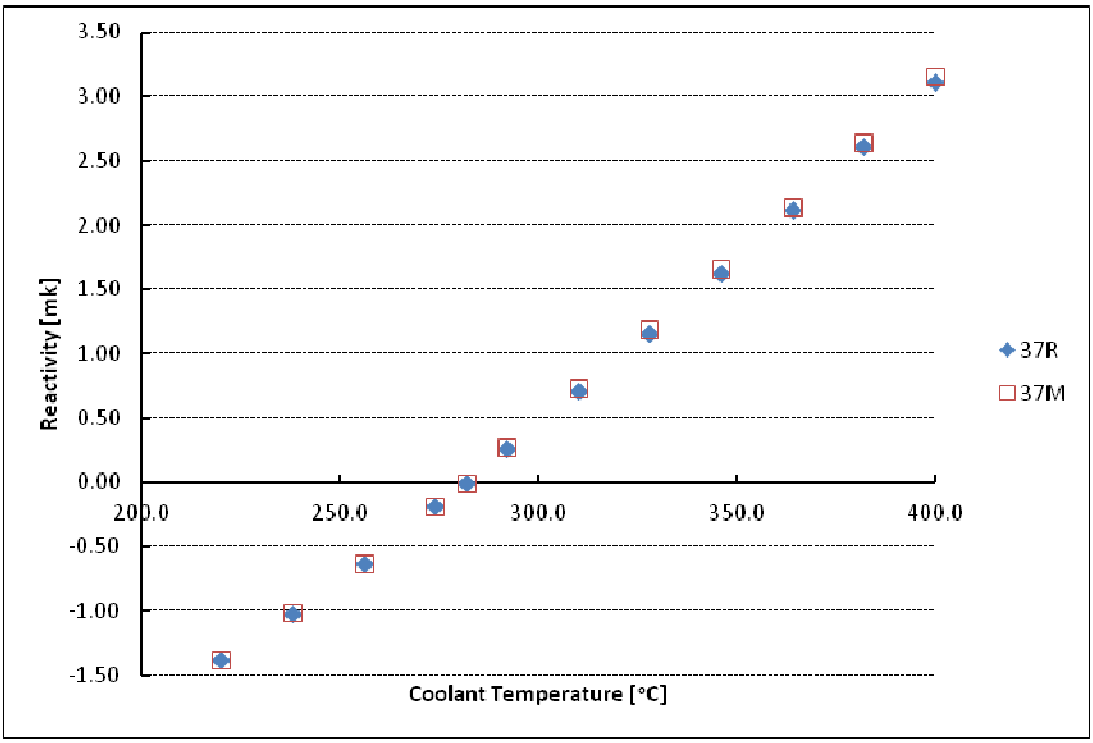

Figure 10: Reactivity change from the nominal conditions as a function of Coolant Temperature

Moderator poison

The design change in the 37M fuel bundle changes the fuel mass of the central element and the dimensions of the coolant subchannels surrounding it. These changes are located at the center of the bundle away from the moderator region. Therefore it is expected that the impact of the fuel design change on the reactivity effect related with the presence of neutron poison in the moderator will be small.

The results presented in Figure 11 for Boron and Figure 12 for Gadolinium confirm the expectation mentioned in the previous paragraph. The fuel design change results in small numerical differences in the moderator poison reactivity coefficients for Boron and Gadolinium. These differences are small when compared to the random uncertainty (2%) recommended in Reference [2] for reactivity effects related with this phenomenon. Effectively, to the precision used in this study, the reactivity coefficients are the same for both fuel designs (-8 mk/ppm for Boron and -27 mk/ppm for Gadolinium).

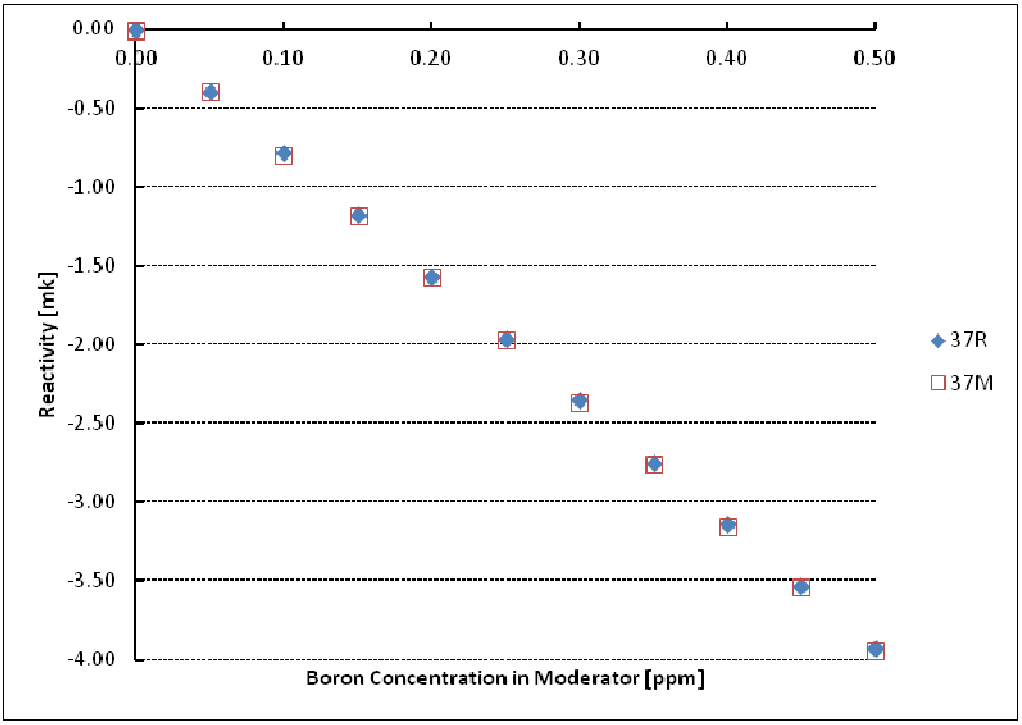

Figure 11: Reactivity change from the nominal conditions as a function of Boron Concentration in the Moderator

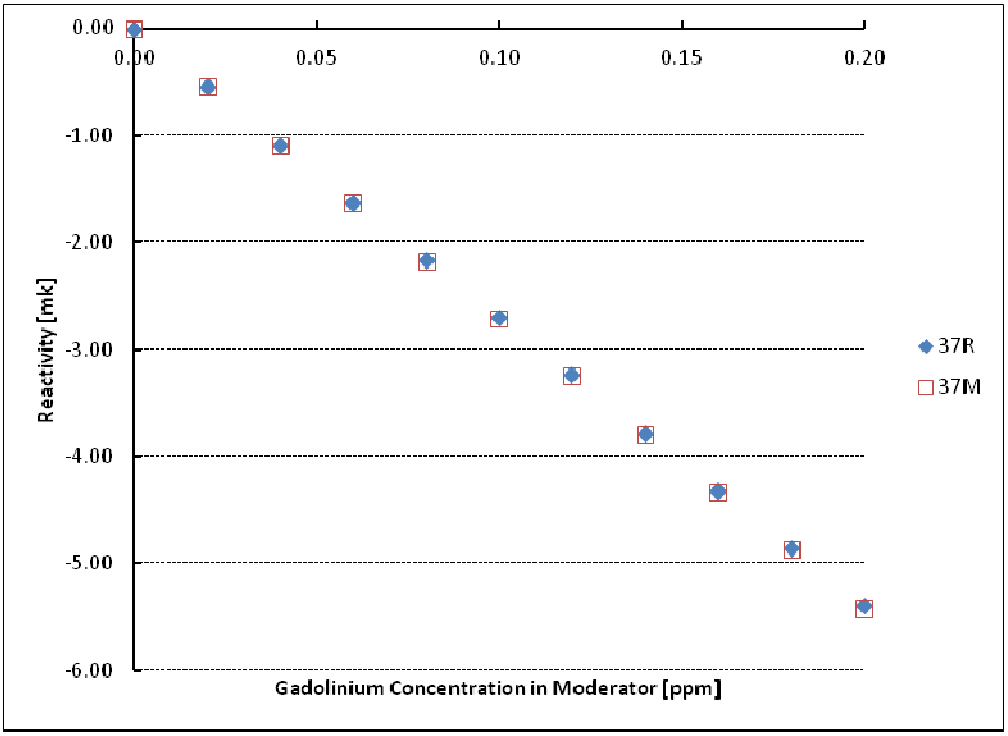

Figure 12: Reactivity change from the nominal conditions as a function of Gadolinium Concentration in the Moderator

Moderator purity

Due to the differences in neutron absorption cross sections between hydrogen and deuterium, changes in the isotopic purity of the moderator have a strong and direct impact on core reactivity.
In the case of the design change under study, the modification from the 37R fuel bundle to the 37M fuel bundle will affect the coolant and the fuel at the central element of the bundle. These changes are expected to have a weak impact on the reactivity effects related to moderator isotopic purity. Results presented in Figure 13 confirm the expectations; the change in the estimated reactivity coefficient, if any, is small compared with the 6 % uncertainty recommended in Reference [2]. Effectively, to the precision used in this study, the estimated reactivity coefficient is the same for both fuel designs (33 mk/at%).

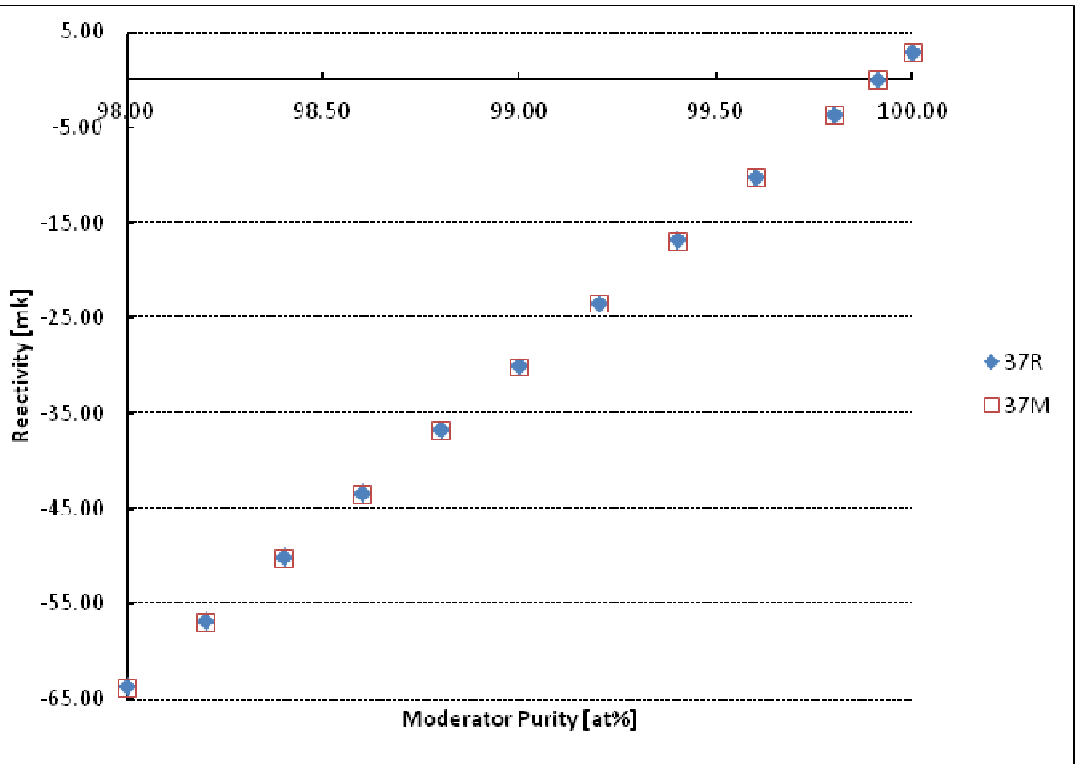

Figure 13: Reactivity change from the nominal conditions as a function of the Moderator Isotopic Purity

Fuel temperature

As discussed earlier in this, the design change has an impact on the radial distribution of the power densities with the larger effect on the central element. Nevertheless it is expected that the resulting changes in the average fuel bundle temperature will have a small impact from the fuel temperature reactivity perspective. The results presented in Figure 14 confirm this expectation. The change in the estimated fuel temperature coefficient is 6.45 % (0.0031 mk/^{o}C for 37R and 0.0028 mk/^{o}C for 37M) which is smaller than the random uncertainty (10 %) related to fuel temperature reactivity as recommended in Reference [2].

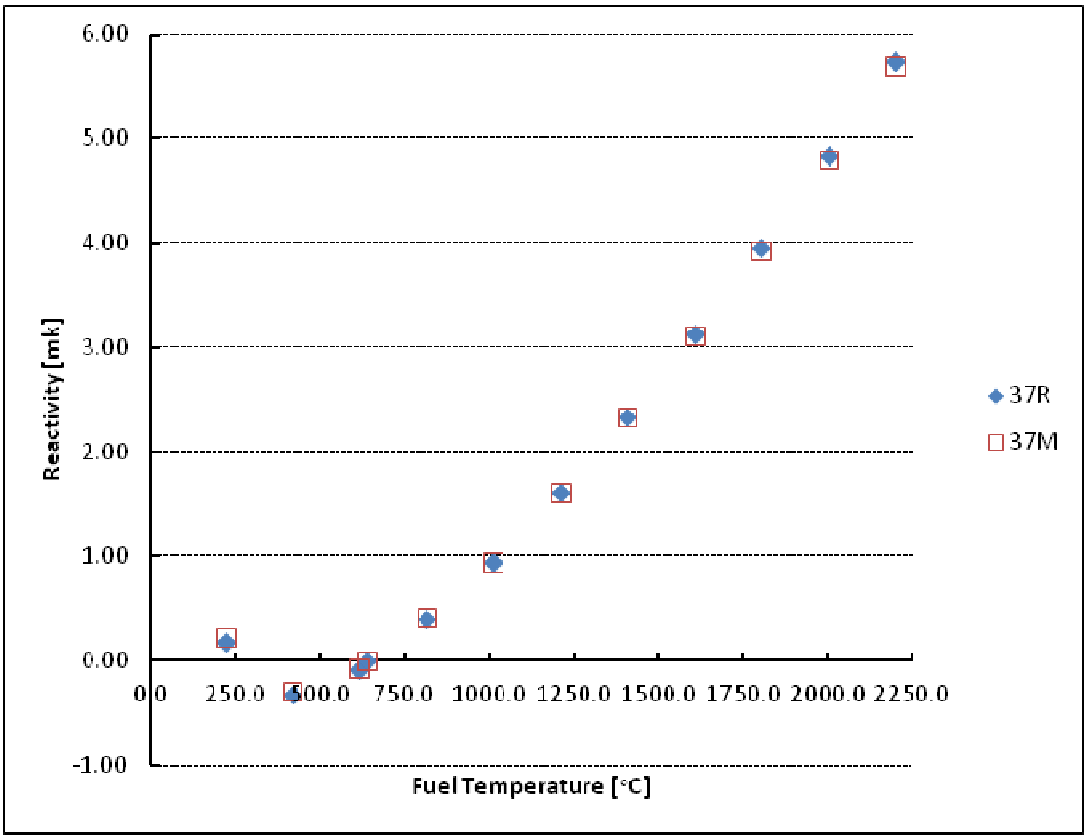

Figure 14: Reactivity change from the nominal conditions as a function of Fuel Temperature

Coolant Purity

Results presented in Figure 15 show that the impact of the design change on the coolant isotopic purity related reactivity effects is small. Effectively, to the precision used in this study, the estimated reactivity coefficients are the same for both fuel designs (0.8 mk/at%). The difference is well within the random uncertainty (12 %) for reactivity effects related with this type of phenomenon as recommended in Reference [2].

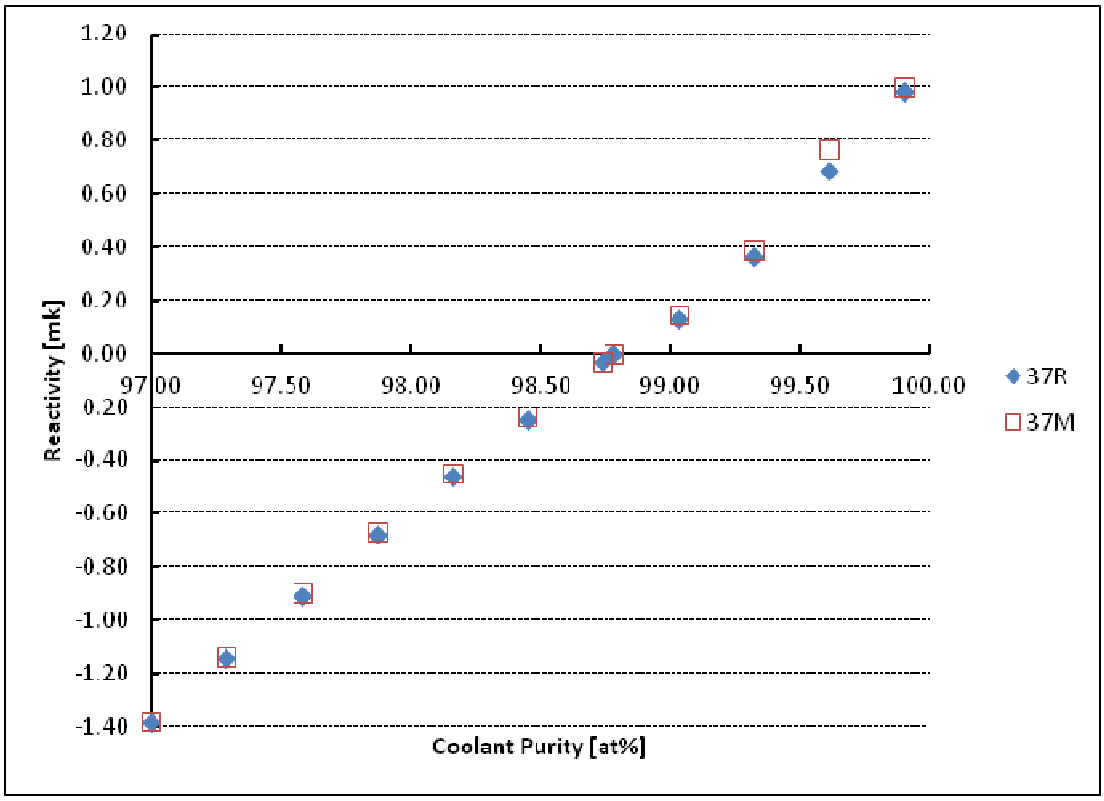

Figure 15: Reactivity change from the nominal conditions as a function of the Coolant Isotopic Purity

Conclusions

The impact on physics parameters due to the design change and the implementation of 37M fuel bundles in the Bruce NGS Units 3 and 4 cores has been assessed. This study has necessitated modifications to the WIMS, DRAGON and RFSP models developed previously for 37R fuel to allow for the assessment and comparison.

Using the updated models, lattice cell and full core simulations have being carried out in order to calculate the impact of the design change on reactor physics characteristics of the fuel and the core as a whole. The results presented in this paper support the conclusion that the impact of the design change in the reactor physics characteristics of the fuel and the core is small in general. The most significant impacts of the fuel design change from a reactor physics perspective are a slight increase in the full core CVR for the 37M fuel design (about 0.11 mk) compared to the predicted results for the 37R fuel design, as well as a redistribution in power between the fuel element rings.

In summary, the implementation of the 37M design has minimal impact on operation and safety of CANDU reactors from the reactor physics point of view while it has been demonstrated in other experiments and studies that the 37M design is effective as a design change to mitigate the effects of HTS aging on dryout margin.

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