The Effects of Fuel Burnup on Incremental Cross Sections in Actinide Fuels in CANDU Reactors

A. C. Morreale[1], M. R. Ball, D. R. Novog and J. C. Luxat
McMaster University, Hamilton, Ontario, Canada
([1]morreaac@mcmaster.ca)

Abstract

The burning of transuranic actinides from spent fuel in current thermal reactors is an active area of nuclear research. Mixed oxides can be used to combine actinides and natural uranium to produce a high burnup fuel suitable for use in CANDU®*. Proper safety assessment is needed to ensure standard operational behavior is maintained with advanced fuels. The interaction of control devices with the fuel and the incremental cross sections should be explored over the full burnup cycle of the fuel to determine the changes that may result from composition changes in the fuel over the burnup lifetime.

1. Introduction

Closing the nuclear fuel cycle and making efficient use of the stockpiles of spent fuel that have accumulated over more than 50 years of nuclear power generation is an important issue facing the nuclear industry and has led to the investigation of many fuel recycle techniques. One of the recycle approaches involves the reprocessing of transuranic actinides such as neptunium, plutonium, americium and curium from spent fuel and burning them in current reactors. The combination of actinides with uranium into a mixed oxide fuel (MOX) that can be used in current thermal reactor systems is an area of active research [1, 2, 3]. Numerous studies have been performed of MOX type fuel production and use in current thermal reactor designs, originating in response to the desire to reprocess excess nuclear weapons material, including studies on the feasibility of plutonium disposition in CANDU reactors [4, 5]. Currently pressurized water reactors (PWRs) in France, Belgium, Switzerland and Japan operate with a portion of MOX fuel (10-20% of core fuel load). In addition, feasibility studies of a full core fuelled with an actinide MOX have been performed for the CANDU 6 design [6].

The employment of these advanced fuels in a current system must meet the design requirements for safe operations and reactor control over the full burnup cycle of the fuel. The fuel designs are usually analysed for the full core with a neutron diffusion code (ex. RFSP) that derives its inputs from lattice and super-cell calculations (ex. WIMS-AECL and DRAGON). The homogenized cross sections for the fuel, coolant and moderator for a standard cell are provided by the lattice code while the local reactivity changes caused by interactions between the control devices and the lattice cell are assessed using super-cell calculations. The incremental cross section for a specific control device refers to the differences in the characteristics between the standard lattice cell and the super-cell calculation. In order to perform the full core calculations the diffusion code requires both the cross sections from the lattice cell calculations and the set of incremental cross sections for the system.

* CANadian Deuterium Uranium (CANDU) is a registered trademark of Atomic Energy of Canada Limited (AECL)

Incremental cross sections are computed by simulating groups of lattice cells with and without control device material present and determining the change in relevant two-group homogenized cross sections including $\Delta\Sigma_{Total}$, $\Delta\Sigma_{absorption}$, $\Delta\Sigma_{N\text{-}Gamma}$, and $\Delta\nu\Sigma_f$ and scattering resulting in $\Delta\Sigma$ values. For most light water reactor (LWR) designs the control devices operate parallel to the fuel elements and hence a rodded assembly can be well represented by a two-dimensional geometry. In the case of the CANDU design, the control devices operate perpendicular to the fuel channels and therefore require a three-dimensional super-cell simulation in order to determine the incremental cross sections.

The incremental cross sections used in previous feasibility evaluations were based on super-cell calculations with standard natural uranium (NU) fuel [6]. Further studies have shown that altered neutron spectra produced by these advanced fuels effects the incremental cross sections resulting in difference from those of standard NU fuel [7]. Furthermore, the fuel composition and hence the neutron spectrum will change over the burnup cycle which can also effect the incremental cross section results. The work presented in this paper studies the effects of fuel burnup on the CANDU incremental cross sections for both natural uranium (NU) and transuranic mixed oxide (TRUMOX) fuel over the full burnup cycle. The two designs explored are the standard 37 element CANDU bundle fuelled with NU and the 43 element TRUMOX bundle design fuelled with actinide mixed oxide fuel. The TRUMOX bundle contains a burnable poison (dysprosium) in the central fuel pin.

2. Evaluation of incremental cross sections

The incremental cross sections for CANDU systems are evaluated using the DRAGON 3D neutron transport code (v 3.06) with the ENDF-BVII library [8]. The CANDU control system places the control devices perpendicular to the fuel channels thus requiring a full 3D super-cell calculation with the device inserted and removed to produce an incremental cross section. The 3-D super-cell contains two fuel channels with a control device in between the channels, see Figure 1.

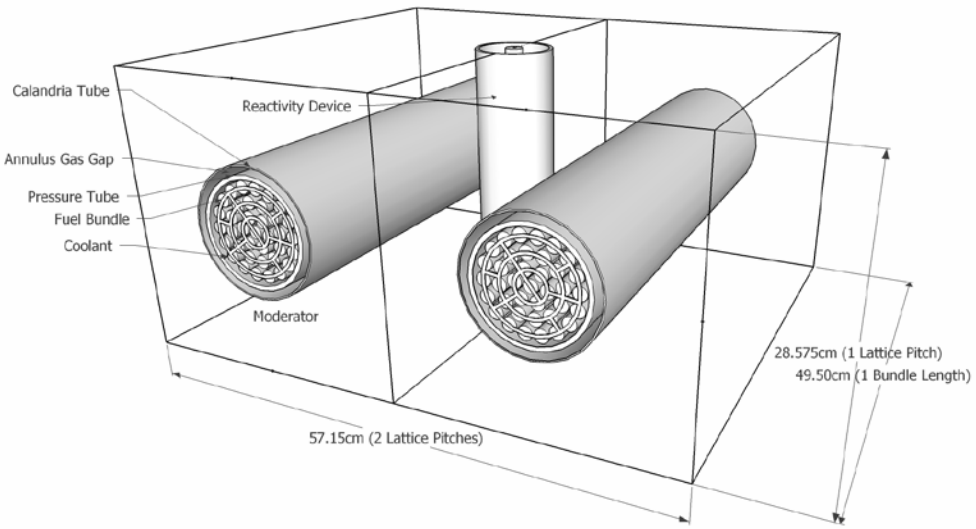

Figure 1 DRAGON 3D Super-cell Model (37 element bundle).

The 3D super-cell model run in DRAGON uses a fully detailed 2D model detailing the 37 or 43 elements of the fuel bundle for the self-shielding calculation, but the 3D transport solution is performed with annularized bundle properties. Specifically, the outer rings (rings 2, 3, 4) are depicted as homogenized annuli of fuel, coolant and cladding material rather than individually clad elements in coolant. The annular fuel rings maintain the volume of fuel, coolant and cladding to represent the fuel elements at the common radial distance from the centre of the bundle. Both the 37 and 43 element bundles are annularized in this manner with the thickness of the annuli equal to the diameter of the clad fuel element and the centre of the annuli located at the radial distance of the elements in the ring to the centre of the bundle. The solution of 3D transport with cluster geometries was impossible in earlier versions of DRAGON (3.04) and newer versions, such as 3.06 used herein, though capable of handling the geometry have high computational complexity [9]. Hence, this study utilizes the accepted technique of annular homogenization of fuel rings that has been used in a substantial amount of prior CANDU safety analysis.

The incremental cross section simulations are performed with a standard stainless steel adjuster rod interacting with the system for the 37 element bundle fuelled with natural uranium fuel and the 43 element bundle fuelled with the TRUMOX actinide fuel. The simulations are performed with the control device at 0% and 100% inserted and the 2 group homogenized cross sections are computed. The fast and thermal incremental cross sections: $\Delta\Sigma_{Total}$, $\Delta\Sigma_{absorption}$, $\Delta\Sigma_{N\text{-}Gamma}$, and $\Delta\nu\Sigma_f$ along with the $\Delta\Sigma_{scatter1,\,2}$ and $\Delta\Sigma_{scatter2,\,1}$ are computed. The 3-D models use a tracking mesh of 12 angles and 20 lines/cm^2. The self-shielding is performed in 2-D with similar meshing of 12 angles and 20 lines/cm. Each calculation is performed for a specific burnup level.

To properly account for the fuel composition at the specific burnup, a separate 2D infinite lattice simulation of the cluster model is performed for the full burnup cycle of the fuel. The fuel composition in each ring of fuel is extracted for the desired burnup level and is used as an input into the fuel composition in the 3D super-cell calculation. The natural uranium fuel is modelled from fresh to the standard CANDU 9 exit burnup of 9,000 MWD/T at a power level of 32.0 W/g. The TRUMOX fuel is modelled from fresh to the exit burnup of ~30,000 MWD/T at a similar power level. Both models use a meshing of 20 angles and 20 lines/cm for self-shielding and 20 angles and 35 lines/cm for the transport solution.

By combining the fuel composition derived from the burnup calculation and the 3D super-cell calculation of the effect of the control device, the incremental cross sections at several specific burnup levels can be analysed for the two fuel types. In general, most incremental cross section calculations are performed at a mid-cycle burnup level as this most closely approximates the fuel composition that the devices will interact with during reactor operations.

2.1 Fuel bundle design and composition

The two fuel designs studied are the 37 element natural uranium (NU) bundle and the 43 element TRUMOX actinide fuel bundle. The 37 element NU bundle contains 37 elements with a common diameter of ~13.1mm all fuelled with natural uranium fuel that is 0.711 wt% U-235 and the remainder U-238 (fresh). The 37 elements are arranged in a cluster geometry of 5 concentric rings of 1, 6, 12, and 18 elements. The 43 element TRUMOX bundle has a large centre element with a

diameter of 17.4 mm which is surrounded by concentric rings of smaller elements (11.4 mm diameter) of 7, 14 and 21 elements. Both bundle designs are displayed in Figure 2.

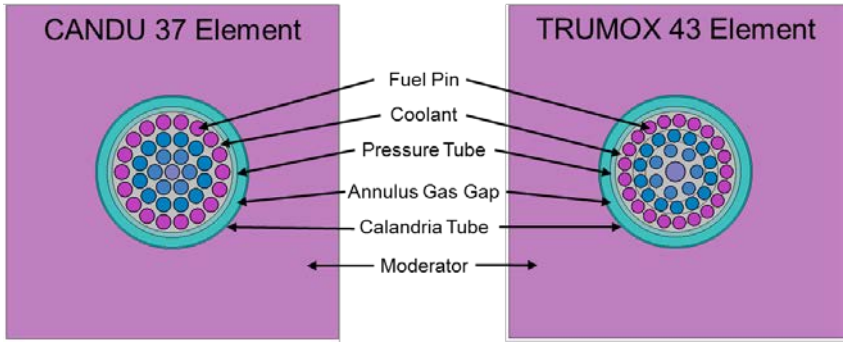

Figure 2 2D Lattice Cell Models for 37 element CANDU and 43 element TRUMOX.

The fuel composition for the TRUMOX fuel is a mixture of actinides extracted from 30 year cooled spent light water reactor fuel and natural uranium. The actinide composition is based on data from Oak Ridge National Laboratories that predicts the probable yields of actinides from spent fuel reprocessing [10]. The centre element contains a dysprosium-zirconium burnable neutron absorber within standard cladding. The composition of the actinide mixture and the uranium matrix along with the central burnable absorber material is provided in Table 1.

Table 1: Fuel Composition, Actinide Uranium and Dysprosium Zirconium Oxide

Actinides			**Uranium Mix**			**Dysprosium Zirconium Oxide**		
Isotope	**Type**	**Wt %**	**Isotope**	**Type**	**Wt %**	**Isotope**	**Type**	**Wt %**
Np-237	Actinide	4.698	U-234	U Mix	0.0054	Dy-160	Absorber	1.346
Pu-238	Actinide	1.301	U-235	U Mix	0.7110	Dy-161	Absorber	10.943
Pu-239	Actinide	56.243	U-238	U Mix	99.2836	Dy-162	Absorber	14.858
Pu-240	Actinide	20.099				Dy-163	Absorber	14.597
Pu-241	Actinide	3.040	**Actinide Oxide (AOX)**			Dy-164	Absorber	16.633
Pu-242	Actinide	3.800	Actinides	88.207	wt %	Zr-90	Zr Mix	12.388
Am-241	Actinide	9.907	Oxygen	11.793	wt %	Zr-91	Zr Mix	2.732
Am-243	Actinide	0.763				Zr-92	Zr Mix	4.221
Cm-243	Actinide	0.001	**Uranium Oxide (UO2)**			Zr-94	Zr Mix	4.371
Cm-244	Actinide	0.072	Uranium	88.150	wt %	Zr-96	Zr Mix	0.719
Cm-245	Actinide	0.012	Oxygen	11.850	wt %	O-16	Oxygen	17.192
Cm-246	Actinide	0.001						

The TRUMOX fuel is a blended mixture of 3.10 wt% actinide oxide and 96.90 wt% natural uranium and is designed for a larger burnup target of 30,000 MWD/T (compared to 7500-9000 MWD/T for NU fuel) and as such the fissile elements in the fuel (Pu^{239}, Pu^{241}, and U^{235}) make up a higher percentage of the fuel, 2.53 wt% vs 0.71 wt% U-235/U in natural uranium CANDU fuel.

2.2 Changes in composition over the burnup cycle

Both the NU and TRUMOX fuels undergo composition changes during their burnup cycle that have effects on the neutronics of the fuel and will affect the cross section behaviour. The U-238 in the NU fuel will undergo neutron capture and produce Pu-239 adding fissile content to the depleting amount of U-235 (and also undergoing further absorptions that produce transuranic species). The fissile content in NU is dominated by U-235 during the initial portion of the cycle but towards the end the Pu-239 content surpasses that of U-235 (~6800MWD/T). The initial fissile content is entirely U-235 which is burned up and only accounts for 35% of fissions by the end of the cycle. The Pu-239 and Pu-241 content climb from zero to 58% and 7% of the fissile content. For TRUMOX, the fissile content is 68% plutonium for fresh fuel and is dominated by Pu-239 for the full burnup. The U-235 content is lower beginning at 28% and gradually declining down to 18%. The Pu-241 content in TRUMOX builds linearly from 3.7% to 22.6% during the cycle. Figure 3 shows the concentrations for the three fissile elements over the burnup cycle for both fuels.

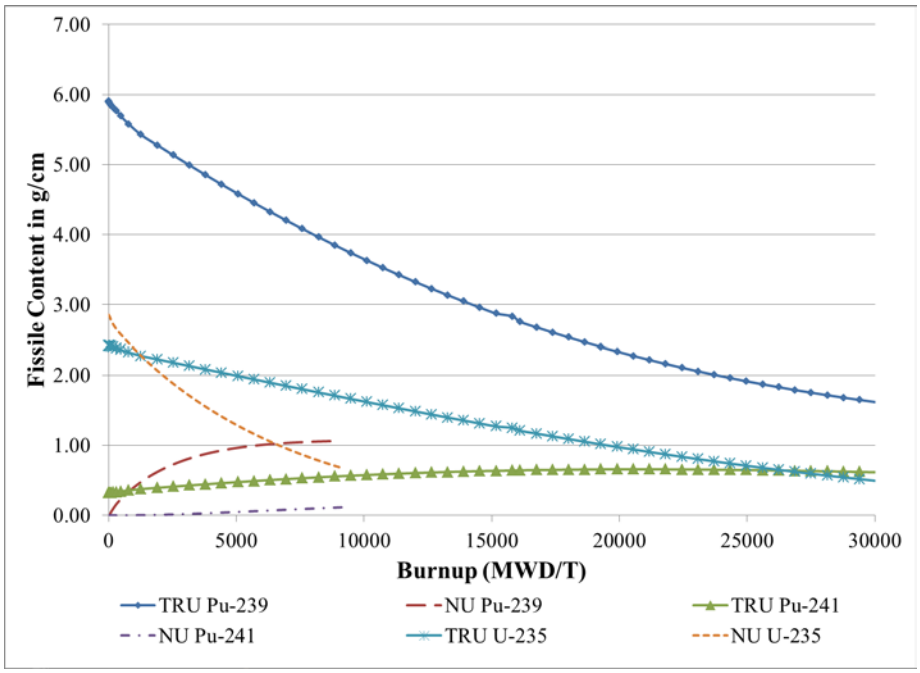

Figure 3 Concentrations of fissile isotopes over the burnup cycle

The changes in the fuel composition during burnup also alter the delayed neutron fraction, β, affecting the relative dollar worth of control devices (Δρ/β). The delayed neutron fraction for a single lattice cell is tracked over the relevant burnup cycle for each fuel based on the composition of fissile isotopes and is displayed in Figure 4. It should be noted that a full core with many channels all at different burnup levels will produce an average β for the full core based on the overall fuel composition in the core (for NU CANDU $\beta_{\text{Full Core}}$ = 5.82 mk) [11]. For the infinite lattice case, the NU beta value is initially high (6.82 mk) and decreases as more plutonium is produced in the fuel (down to 4.15 mk at 9000 MWD/T, a drop of 2.67 mk over the burnup cycle). The TRUMOX fuel has higher plutonium content and thus lower beta values (~3 mk less than NU). Beta starts at 3.66

mk and rises slightly to a peak of 3.88 mk and then decreases to 3.75 mk at 30000 MWD/T, a range of only 0.22 mk, less than 10% of the range for NU, making it more constant over the burnup providing a more stable delayed fraction for the TRUMOX fuel.

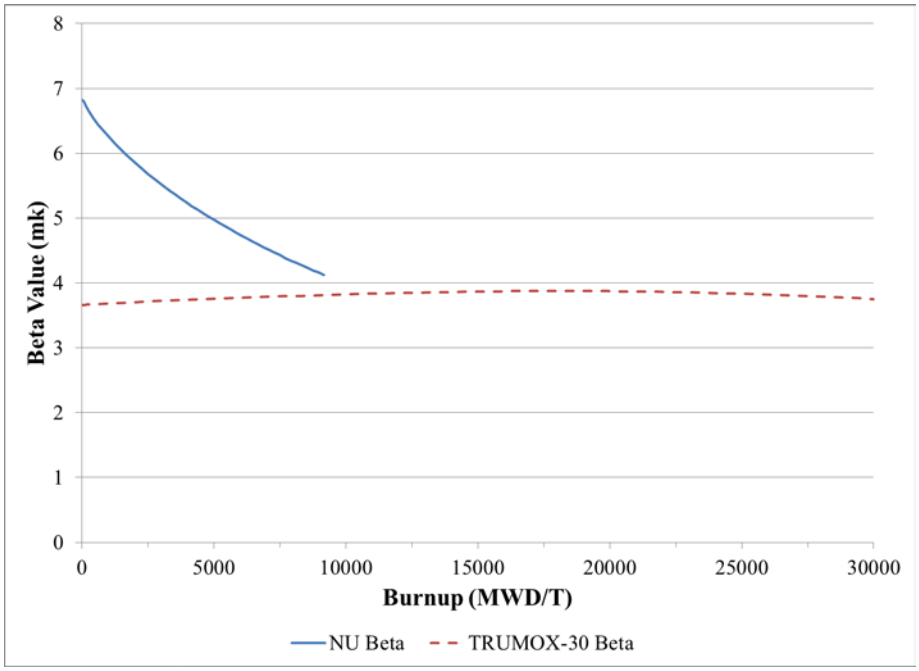

Figure 4 Effects of burnup on delayed neutron fraction, β (infinite lattice case)

3. Simulations and Results

The burnup simulations for the two fuel types were used to produce fuel compositions at various burnup levels which were then transferred into the 3D simulations to determine the reactivity changes and incremental cross sections resulting from the insertion of a standard CANDU type stainless steel adjuster rod. The adjuster rod is a pin in tube design with an inner pin of 0.8 cm and a 3.8 cm OR tube with a thickness of 0.124 cm. Both the pin and tube are stainless steel (304L alloy) with a composition defined in Table 2. The tube is open at both ends allowing moderator to fill the inner gap surrounding the solid pin. The adjuster assembly (pin and tube) is moved within a perforated zirconium guide tube that is fixed in the core.

Table 2: Stainless Steel Adjuster Composition

Material	Wt %	Material	Wt %	Material	Wt %
C	0.037	Si	0.460	Mn	1.282
Fe	72.671	Ni	8.430	Cr	17.120

The 3D incremental cross section simulations, as described earlier in Section 2, consisted of a simulation with the rod fully inserted and with the rod fully extracted. The perforated zirconium alloy guide tube is present in both models. The simulations were performed for natural uranium fuel and for the TRUMOX actinide fuel.

3.1 Effects of burnup on k-infinity and reactivity change from adjuster insertion

The reactivity of the lattice cell will change as the fuel burnup increases and will also have an effect on the reactivity change caused by the insertion of the adjuster rod. In general, the reactivity of the lattice cell will decrease over the cycle as the fission content is steadily depleted. The k-infinity for both fuels was tracked over the burnup and is provided in Figure 5.

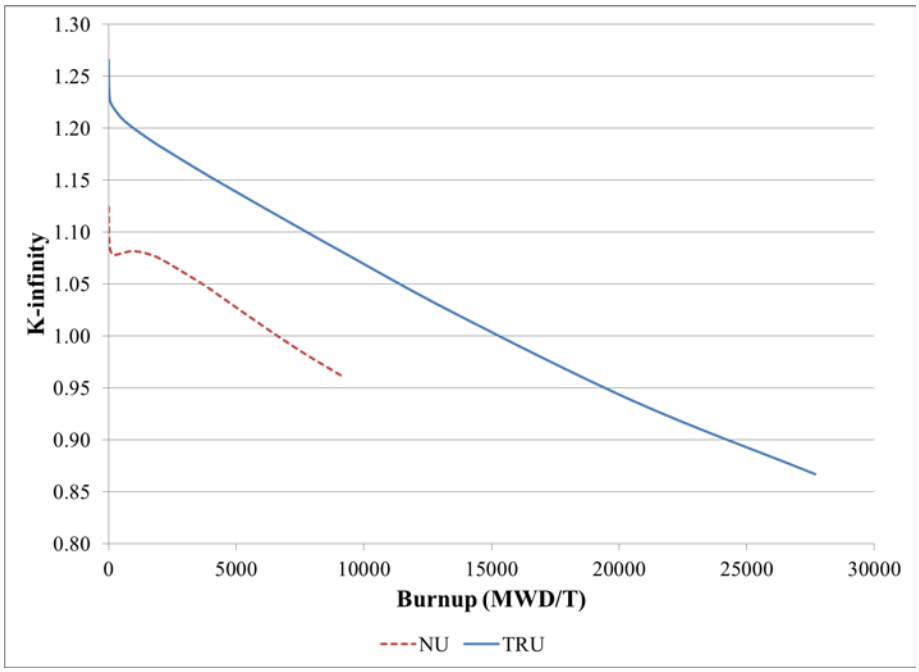

Figure 5 Effects of fuel burnup on infinite lattice k-infinity

The general reactivity response over the burnup cycle is similar for both fuel types, an initial strong reduction followed by a gradual linear decline. The initial decrease results from the production of neutron absorbing fission products, such as Xe-135, which build-up to a steady state value in the first week of operations. The NU case experiences a reactivity peak at about 1,100MWD/T due to the build-up of Pu-239 which transmutes from U-238 adding fissile content to the fuel. Early in the cycle these gains outpace the losses due to burnup producing a reactivity peak. The TRUMOX fuel has a much higher fissile content, much of it plutonium and a reduced U-238 content which makes the transmutation to Pu-239 a less dominating effect on the fuel reactivity and hence there is no reactivity peak.

Compared to NU fuel, TRUMOX fuel is more active and hence has a higher k-infinity for most of the burnup. It should be noted that the initial k-infinity is 12.6% higher in TRUMOX and at 1120MWD/T is 10.7% higher while at the end of burnup it is 9.9% lower than that of NU. For a critical operating reactor with fuels at all burnup levels, the active new fuel will balance out the older fuel. Since the TRUMOX fuel is more active to begin with it is able to stay in the core longer and reach a lower k-infinity value while overall full core reactivity is maintained.

The k-infinity values for the rod in and rod out models and the associated reactivity change for both the NU and TRUMOX fuels is provided in Table 3 and the reactivity change is graphed in Figure 6.

Table 3: Effects of Burnup on Super-cell Reactivity Change from Adjuster Rod Insertion

Natural Uranium Fuel				
Burnup(MWD/T)	*Rod out k-infinity*	*Rod in k-infinity*	*Reactivity change (mk)*	*Δρ/β ($)*
0	1.110372	1.04895	52.7	7.23
1760	1.068021	1.013743	50.1	8.41
3680	1.040618	0.989219	49.9	9.36
4480	1.027027	0.97665	50.2	9.81
7520	0.975944	0.928873	51.9	11.71
9120	0.952263	0.906633	52.9	12.77
TRUMOX Fuel				
Burnup(MWD/T)	*Rod out k-infinity*	*Rod in k-infinity*	*Reactivity change (mk)*	*Δρ/β ($)*
0	1.258476	1.217424	26.8	7.33
4480	1.142152	1.104555	29.8	7.96
9120	1.077236	1.040713	32.6	8.54
13600	1.017082	0.98161	35.5	9.21
20640	0.932344	0.898479	40.4	10.44
27680	0.863062	0.830615	45.3	11.93

There are distinct differences in the reactivity changes for adjuster insertion between the NU and TRUMOX fuels over their burnup cycles. The reactivity change for NU decreases initially and then after a flat period begins to linearly increase as the middle of the burnup cycle is reached. The range in the NU reactivity change is only 3 mk with the largest change coming at the end of burnup with 52.9 mk. The initial reactivity change is nearly as high at 52.7 mk with the decline caused by the build-up of plutonium in the fuel during the first half of the burnup cycle. The TRUMOX fuel undergoes a linear increase in reactivity change throughout the full burnup cycle that increases from 26.8 mk to 45.3 mk, a range of 18.5 mk. The TRUMOX values are lower by 50% initially and the gap decreases to 15% when the comparison is made at the end of the two burnup cycles. For a comparison of the reactivity change at the same burnup the TRUMOX fuel consistently lower: 50% less for fresh fuel, 41% less for 4480MWD/T and 38% less at 9120 MWD/T.

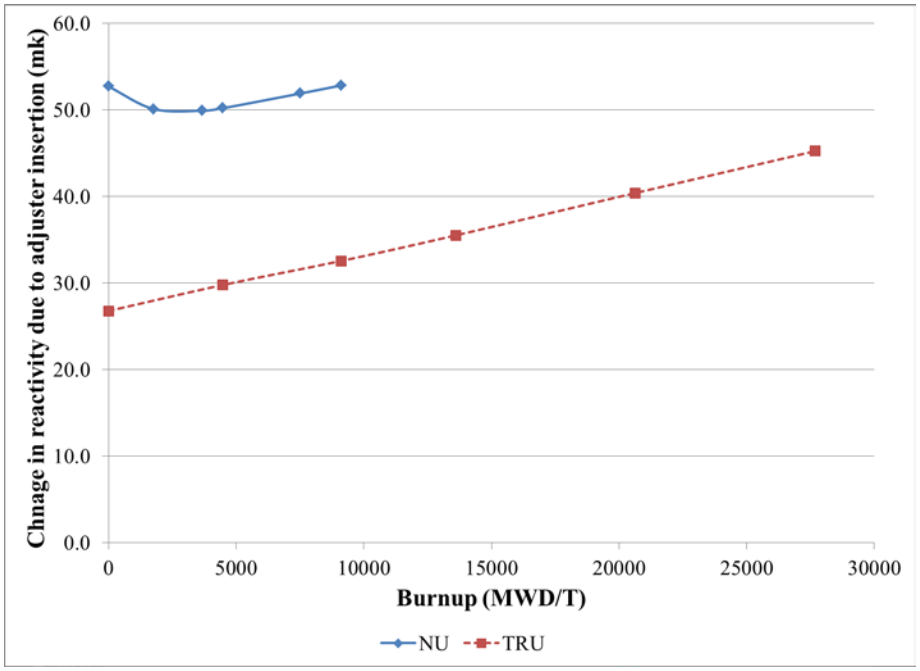

Figure 6 Effects of fuel burnup on reactivity change due to rod insertion

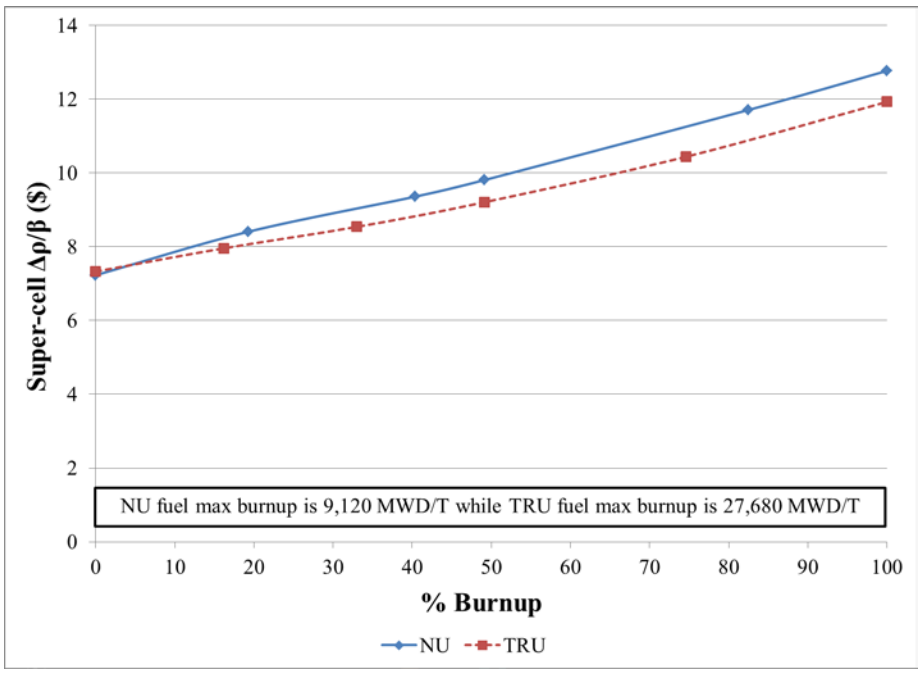

Figure 7 Effects of fuel burnup on relative reactivity change (Δρ/β) due to rod insertion

Using the β values computed in Figure 4, the relative reactivity change, (Δρ/β), was computed for each of the burnup steps. Comparing (Δρ/β), the two fuels show different results for a specific burnup with the TRUMOX 1.38% higher for fresh fuel, 18.9% lower at 4480MWD/T and 33.1% lower at 9120 MWD/T. However, the two fuels have similar relative reactivity when compared over their relative burnup cycles as seen in Figure 7. Both fuels experience a generally linear increase from about $7 to $12-$13 at the end of burnup. The dollar values for TRUMOX are 1.38% higher than NU for fresh fuel, 6.12% lower at the mid burnup stage and 6.58% lower at the end of burnup. Despite the different burnup behaviour of the two fuel types, their relative reactivity performance is quite comparable over their burnup cycle which would suggest similar controllability limits. This is an important issue when considering the use of TRUMOX fuel in an NU CANDU system.

In TRUMOX, the larger fission content and lower amount of U-238 means that transmutation to Pu-239 has less effect on the fission content and the Pu-239 and U-235 content decreases continuously during burnup (see Figure 3). The ratio of Pu-239 to U-235, the two primary fissile elements in the fuels, is a driver of the differences in reactivity change during rod insertion for the two fuels. In the NU case the ratio increases exponentially as the initial concentration of Pu-239 starts at zero and builds as the U-238 in the fuel is transmuted to Pu-239 causing the ratio to increase from zero to a ratio of about 1.53 at the end of the NU burnup cycle. Alternatively, the TRUMOX fuel begins at 2.44 and dips downward to 2.24 at 11400MWD/T then rises back up to 2.44 at 21000MWD/T and continues to rise during the next 6500MWD/T to 2.98 and reaches 3.28 at 30000MWD/T, the approximate end of the TRUMOX burnup cycle. This difference in Pu-239 behaviour explains the more constant nature of the reactivity change for the TRUMOX fuel and the initial dip seen in the NU fuel case. The TRUMOX case has a harder neutron spectrum due to the larger plutonium concentration and the presence of the actinides with high thermal capture cross sections (ex. Am-241 and Pu-240). The rod effectiveness is reduced since the absorption cross section of stainless steel follows a 1/v relationship and hence as the average velocity (energy) of the neutrons increases the absorption of the control device will decrease. The plutonium content is highest in the fresh fuel and the largest difference between the NU and TRUMOX cases in both plutonium content and spectrum is seen here explaining the gap in reactivity change.

3.2 Effects of burnup on incremental cross sections

The effects of burnup on fuel behaviour are also seen in the changes in the incremental cross sections of the adjuster rod insertion for each fuel type. The adjuster incremental cross sections are calculated by running a simulation with the adjuster rod fully out and then fully in and then subtracting the 2 group condensed cross sections for the rod out case from the rod in case. The adjuster incremental cross sections examined in these trials include fast and thermal $\Delta\Sigma_{Total}$, $\Delta\Sigma_{absorption}$, $\Delta\Sigma_{N\text{-}Gamma}$, and $\Delta\nu\Sigma_f$ along with the $\Delta\Sigma_{scatter1,\,2}$ and $\Delta\Sigma_{scatter2,\,1}$. These were computed at several burnup steps for each of the fuel types and are provided in Table 4 and 5 along with the percentage change of the cross section from the beginning to the end of the burnup cycle.

Table 4: Effects of Burnup on Adjuster Incremental Cross Sections (ΔΣ), NU Fuel

Burnup	**GROUP 1 (Fast) – NU FUEL (cm^{-1})**				**Down Scatter**
(MWD/T)	**Total**	**Absorption**	**N-Gamma**	**NU-SIGMA-F**	**Gr1 to Gr2**
0	5.55E-04	1.10E-05	1.10E-05	-2.12E-07	-1.89E-05
1760	5.55E-04	1.09E-05	1.10E-05	-2.26E-07	-1.89E-05
3680	5.56E-04	1.09E-05	1.10E-05	-2.39E-07	-1.88E-05
4480	5.56E-04	1.09E-05	1.10E-05	-2.43E-07	-1.88E-05
7520	5.56E-04	1.09E-05	1.10E-05	-2.52E-07	-1.88E-05
9120	5.56E-04	1.09E-05	1.09E-05	-2.54E-07	-1.88E-05
% Change	0.24%	-0.87%	-0.73%	19.77%	-0.54%
Burnup	**GROUP 2 (Thermal) – NU FUEL (cm^{-1})**				**Up Scatter**
(MWD/T)	**Total**	**Absorption**	**N-Gamma**	**NU-SIGMA-F**	**Gr2 to Gr1**
0	2.31E-04	2.45E-04	2.42E-04	7.70E-06	4.64E-06
1760	2.47E-04	2.51E-04	2.45E-04	1.49E-05	4.72E-06
3680	2.54E-04	2.53E-04	2.47E-04	1.80E-05	4.72E-06
4480	2.56E-04	2.54E-04	2.47E-04	1.87E-05	4.72E-06
7520	2.60E-04	2.55E-04	2.48E-04	1.99E-05	4.70E-06
9120	2.62E-04	2.55E-04	2.48E-04	2.01E-05	4.69E-06
% Change	13.12%	4.15%	2.56%	161.18%	1.08%

Table 5: Effects of Burnup on Adjuster Incremental Cross Sections (ΔΣ), TRU Fuel

Burnup	**GROUP 1 (Fast) – TRU FUEL (cm^{-1})**				**Down Scatter**
(MWD/T)	**Total**	**Absorption**	**N-Gamma**	**NU-SIGMA-F**	**Gr1 to Gr2**
0	5.61E-04	1.08E-05	1.09E-05	-2.24E-07	-1.88E-05
4480	5.61E-04	1.08E-05	1.09E-05	-2.21E-07	-1.88E-05
9120	5.61E-04	1.08E-05	1.09E-05	-2.29E-07	-1.88E-05
13600	5.61E-04	1.08E-05	1.09E-05	-2.37E-07	-1.88E-05
20640	5.60E-04	1.08E-05	1.09E-05	-2.47E-07	-1.88E-05
27680	5.60E-04	1.08E-05	1.09E-05	-2.53E-07	-1.88E-05
% Change	-0.23%	-0.21%	-0.10%	13.04%	-0.28%
Burnup	**GROUP 2 (Thermal) – TRU FUEL (cm^{-1})**				**Up Scatter**
(MWD/T)	**Total**	**Absorption**	**N-Gamma**	**NU-SIGMA-F**	**Gr2 to Gr1**
0	3.36E-04	2.78E-04	2.66E-04	3.43E-05	4.67E-06
4480	3.31E-04	2.76E-04	2.65E-04	3.21E-05	4.66E-06
9120	3.24E-04	2.74E-04	2.63E-04	2.99E-05	4.64E-06
13600	3.18E-04	2.72E-04	2.62E-04	2.79E-05	4.62E-06
20640	3.09E-04	2.69E-04	2.60E-04	2.52E-05	4.59E-06
27680	3.01E-04	2.66E-04	2.58E-04	2.32E-05	4.57E-06
% Change	-10.20%	-4.16%	-2.87%	-32.32%	-2.26%

For both fuel cases, there is very little change in the Group 1 (fast) $\Delta\Sigma_{Total}$, $\Delta\Sigma_{absorption}$, and $\Delta\Sigma_{N\text{-}Gamma}$ values with the differences being less than ±1.0% and the directions are similar for both with the exception of the $\Delta\Sigma_{Total}$ value which increases by 0.24% in NU but decreases by 0.23% in the TRUMOX case. The $\Delta\Sigma_{scatter1,\ 2}$ values for NU and TRU both decrease by a small amount over their burnup cycle with the NU fuel decreasing by 0.54% while the TRU only decreases by 0.28%. The group 1 (fast) $\Delta\nu\Sigma_f$, values both increase with the NU case increasing by 6.8% more than the TRUMOX case. In group 2 (thermal) the two fuel types have opposite responses, the percentage changes in the NU fuel cases are all positive while the changes in the TRUMOX case are all negative. The behavior over the burnup for the two fuels is very different for the total absorption, N-Gamma and $\Delta\nu\Sigma_f$ values (Figure 8). In the NU case the adjuster incremental values generally increase sharply at the beginning and plateau during the last half of the burnup while the TRUMOX values decrease linearly with burnup. In general, these differences are caused by the plutonium build up in the NU fuel described earlier as the adjuster incremental cross sections tend to follow the Pu-239 concentration during burnup (Figure 3). The TRUMOX fuel already contains a sizeable amount of plutonium so the transmutation of U-238 to Pu-239 is not dominant in this case and the plutonium content steadily decreases during the burnup cycle which is reflected in the adjuster incremental cross sections.

In the specific case of adjuster incremental cross section for thermal absorption with TRUMOX fuel, the decrease over burnup is driven by actinides with high thermal absorptions that have been transmuted to other isotopes with lower absorption cross sections. Over the full burnup cycle, about 35% of the higher actinides, which have high thermal absorption cross sections, will be transmuted.

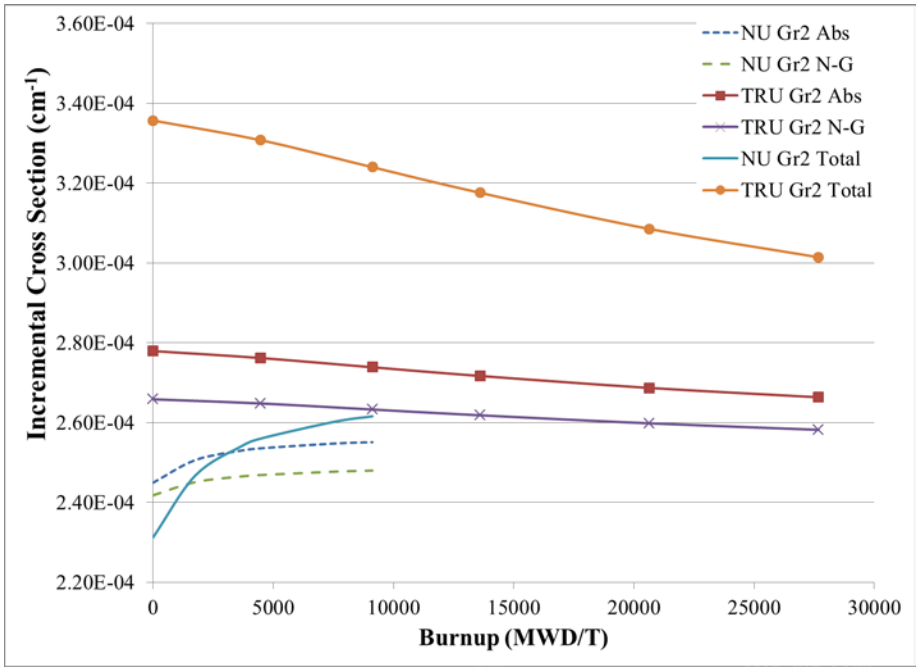

Figure 8 Change in Group 2 (Thermal) incremental cross sections over the burnup cycle

The adjuster incremental cross section for up-scatter in the NU case increases sharply to a maximum in the early burnup, plateaus shortly, and then decreases linearly for the last half of the burnup. In the TRUMOX case the adjuster incremental cross section maintains a steady decline that is less linear than for other values but is still fairly consistent (Figure 9). The change in up-scatter is proper for the TRUMOX case, a loss of 2.26%, but is inaccurate in the NU case which would result in a change of -0.68% if measured from the peak rather than the 1.08% change seen measuring from the fresh fuel initial value. The linear decrease in the NU case during second half of the burnup is driven by the fact that plutonium is becoming the dominant fissile isotope in the fuel.

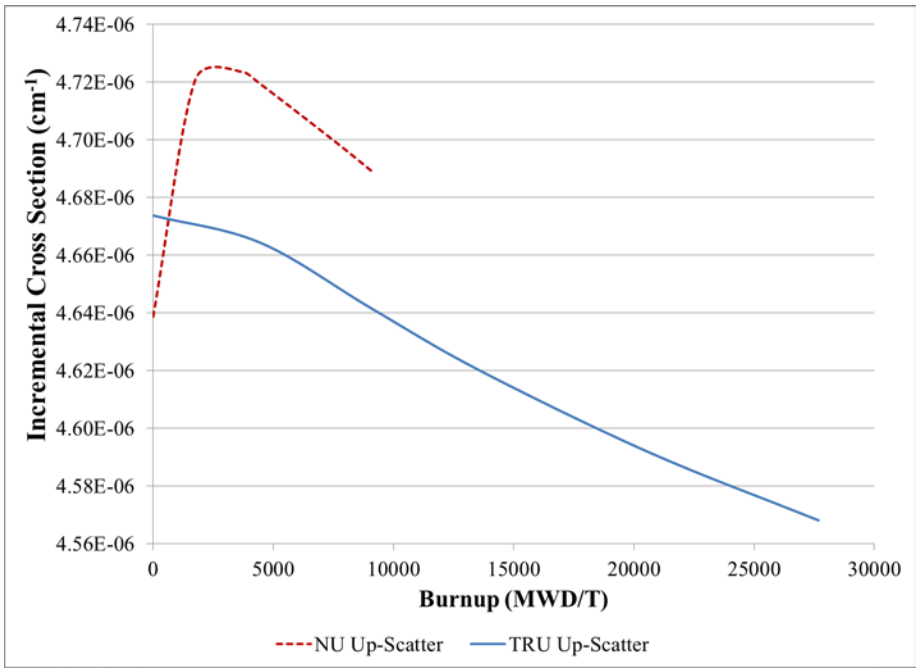

Figure 9 Change in the up-scatter incremental cross sections over the burnup cycle

Comparison between the adjuster incremental cross sections for NU and TRUMOX fuel can also be made either at specific burnup levels or at the relative places in the burnup cycle (ex. Fresh, mid, end). The percent difference of the adjuster incremental cross section values for the TRUMOX case are computed with respect to those of NU case for both types of comparisons in Tables 6 and 7.

The results of the comparisons of the adjuster incremental cross sections for the two fuel types are consistent over the two methods with the mid and end burnup values being larger than the direct burnup numbers due to the larger burnup cycle of TRUMOX. The group 1 (fast) values for Σ_{Total}, $\Sigma_{absorption}$, and $\Sigma_{N\text{-}Gamma}$ are relatively small (all within ±1.2%) as are the values for down scattering.

Table 6: Comparison of NU and TRUMOX Adjuster Incremental Cross Sections at Specific Burnup

Burnup	Gr 1 (Fast) – % diff from NU (TRU-NU)/NU				Down Scatter
(MWD/T)	Total	Absorption	N-Gamma	NU-SIGMA-F	Gr1 to Gr2
0	1.17%	-1.50%	-1.47%	5.57%	-0.52%
4480	0.97%	-0.94%	-1.01%	-9.06%	-0.19%
9120	0.81%	-0.70%	-0.79%	-9.75%	-0.09%
Burnup	**Gr 2 (Thermal) – % diff from NU (TRU-NU)/NU**				**Up Scatter**
(MWD/T)	**Total**	**Absorption**	**N-Gamma**	**NU-SIGMA-F**	**Gr2 to Gr1**
0	45.15%	13.46%	9.94%	345.20%	0.76%
4480	29.19%	8.92%	7.26%	72.23%	-1.16%
9120	23.86%	7.36%	6.17%	48.71%	-1.01%

Table 7: Comparison of NU and TRUMOX Adjuster Incremental Cross Sections at Burnup Stage

Burnup	Gr 1 (Fast) – % diff from NU (TRU-NU)/NU				Down Scatter
Stage	Total	Absorption	N-Gamma	NU-SIGMA-F	Gr1 to Gr2
Fresh Fuel	1.17%	-1.50%	-1.47%	5.57%	-0.52%
Mid Burnup	0.86%	-1.07%	-1.09%	-2.44%	-0.33%
End of Burnup	0.69%	-0.84%	-0.84%	-0.37%	-0.26%
Burnup	**Gr 2 (Thermal) – % diff from NU (TRU-NU)/NU**				**Up Scatter**
Stage	**Total**	**Absorption**	**N-Gamma**	**NU-SIGMA-F**	**Gr2 to Gr1**
Fresh Fuel	45.15%	13.46%	9.94%	345.20%	0.76%
Mid Burnup	24.05%	7.15%	6.06%	49.31%	-2.10%
End of Burnup	15.23%	4.41%	4.12%	15.37%	-2.58%
* NU Mid Burnup = 4480 MWD/T, NU End of Burnup = 9520 MWD/T					
** TRU Mid Burnup = 13600 MWD/T, TRU End of Burnup = 27680 MWD/T					

In the burnup stage comparison, the Δ-up-scatter differences change to -2.1% and -2.6% due to the higher burnup of TRUMOX. The burnup comparison also changes the perspective on the group 1 $\Delta\nu\Sigma_f$ value which comes closer to the NU value in the burnup stage comparison by 7-9%. The biggest differences from the NU case are in the group 2 (thermal) values for total and $\Delta\nu\Sigma_f$. The total values have differences as high as 45% for fresh fuel that decrease down to 15% over the burnup cycle.

The adjuster incremental cross section for group 2 $\Delta\nu\Sigma_f$ is very different between the two cases for fresh fuel (345%) due to the much higher fissile content of the TRUMOX fuel which is about 3.5 times that of NU (2.53% vs 0.71%). As the fuel burns, the fissile content decreases explaining the differences between the specific burnup and burnup stage comparisons. By the time both fuels reach the end of their burnup cycle the difference in the values is about 15% since the TRUMOX fuel has a burnup cycle length that is 3 times as long as NU which utilizes the higher fissile content.

In summary, the incremental cross sections of the adjusters for the two fuels do behave differently over the burnup cycle of the fuels. The NU case shows an increase and then plateau while the adjuster incremental cross sections in the TRUMOX case undergo a continual decline. The main cause for the difference in behaviour is the different plutonium concentrations and build-up in the fuel types during the burnup cycle that alters the neutron spectrum. The NU fuel begins with no plutonium which builds from the transmutation of U-238 to become more than half the fissile content by the end of burnup. In the first part of the NU burnup cycle the Pu-239 concentration builds, adding fissile content to the steadily decreasing U-235 concentration faster than it is burned off producing a reactivity peak at approximately 1,100 MWD/T. From this point on, the reactivity constantly declines until the end of the cycle. The TRUMOX fuel has a much higher fissile content and a lower U-238 concentration and thus this transmutation effect has a lower impact resulting in a more linear reactivity and fissile content that decreases throughout the full burnup cycle.

4. Conclusion

The analysis of the effects of fuel burnup on the adjuster incremental cross sections in a CANDU reactor was performed using the DRAGON lattice physics code for both natural uranium and transuranic mixed oxide fuel. The 2D burnup and 3D incremental simulations yielded additional results for the reactivity change due to adjuster rod insertion. The use of higher enriched fuels such as TRUMOX with higher plutonium content and harder neutron spectra reduce the reactivity change during adjuster insertion due to the lower capture cross section of stainless steel. The NU fuel experiences a reactivity peak around 1,100MWD/T due to the transmutation of U-238 to Pu-239 which has an effect on the reactivity change and adjuster incremental cross section response over time. The TRUMOX fuel does not experience a transmutation driven peak because of its already high plutonium content and lower U-238 concentration. Despite the different composition and delayed neutron factor behaviour during the burnup cycles the relative reactivity changes due to rod insertion are quite similar between the two fuels indicating similar controllability.

The adjuster incremental cross sections tend to rise to a plateau for the NU case over the burnup cycle with the rise consistent with the period before the plutonium peak while the TRUMOX case shows a constant linear decline during the burnup cycle. The adjuster incremental cross section changes are minor for the fast group. In the thermal group, the TRUMOX case had significant differences in the $\Delta\Sigma_{absorption}$, $\Delta\Sigma_{N\text{-}Gamma}$ and $\Delta\Sigma_{Total}$ values over the burnup which initially start at 13.5%, 10% and 45% respectively and decrease down to 4.4%, 4.1% and 15% towards the end of burnup. The adjuster incremental cross section for thermal fission is much higher in TRUMOX case over the burnup cycle starting from 345% and trending down to 15% when both fuel cases are compared at their respective end of burnup. The differences are expected due to the much higher fissile content present in the TRUMOX fuel and the longer burnup it experiences. This analysis showed that the use of mixed oxide fuels and others that have higher fissile contents and fissile elements that harden the neutron spectrum have an effect on the incremental cross sections calculated using super-cell models and do not experience the plutonium transmutation driven reactivity peak seen in NU fuels. The biggest effects are in the adjuster incremental cross sections for thermal fission and are a related to the amount of fissile material in the fuel. This information can be utilized as input to full core diffusion calculations of a TRUMOX fuelled reactor to provide more realistic simulations suitable for detailed analysis and the modelling of accident scenarios.

5. Acknowledgements

The authors would like to thank S. Goodchild and Z. Catovic from Ontario Power Generation for their assistance in 3D super-cell modelling in CANDU. Funding for this research was provided by UNENE, Nuclear Ontario and NSERC.

6. References

[1] L.H. Baetsle and Ch. De Raedt, "Limitations of actinide recycle and fuel cycle consequences: a global analysis, Part 2: Recycle of Actinides in thermal reactors: impact of high burnup LWR-UO2 fuel irradiation and multiple recycle of LWR-MOX fuel on the radiotoxic inventory", *Nuclear Engineering and Design*, **Iss. 168**, p. 203-210, (1997).

[2] R. A. Wigeland, et al, "Impact on Geologic Repository Usage from Limited Actinide Recycle in Pressurized Light Water Reactors", *Nuclear Science and Technology*, **Vol. 44, No. 3**, p. 415-422, (2007).

[3] T. A. Taiwo, et al, "Assessment of a Heterogeneous PWR Assembly for Plutonium and Minor Actinide Recycle", *Nuclear Technology*, **Vol. 155**, p. 34-54, (2006)

[4] P. Boczar, et al, "Global Warming and Sustainable Energy Supply With CANDU Nuclear Power Systems", *Progress in Nuclear* Energy**, Vol. 32, No. ¾,** p. 297-304, (1998).

[5] P. G. Boczar, J. R. Hopkins, H. Feinroth and J.C. Luxat, "Plutonium Disposition in CANDU", *Proc. IAEA Technical Meeting on Recycling of Plutonium and Uranium in Water Reactor Fuels*, Windermere, U. K., July 1995.

[6] A. C. Morreale et al, "The reactor physics characteristics of a transuranic mixed oxide fuel in a heavy water moderated reactor" *Nucl. Eng. and Des.*, **Vol. 241**, pp. 3768-3776 (2011).

[7] A.C. Morreale, M.R. Ball, D.R. Novog, J.C. Luxat, "The Effects of Actinide Based Fuels on Incremental Cross Sections in a CANDU Reactor", *International Conference on Mathematics and Computational Methods Applied to Nuclear Science and Engineering (M&C 2011),* May 8-12, 2011, Rio de Janeiro, RJ, Brazil.

[8] G. Marleau, A. Hébert and R. Roy, "A User Guide for DRAGON," Report IGE-174 Rev.5, École Polytechnique de Montréal, April 2000.

[9] M. Dahmani, G. Marleau, R. Le Tellier, "Modeling reactivity devices for advanced CANDU reactors using the code DRAGON", *Annals of Nuclear Energy*, **Vol. 35**, pp. 804-812 (2008).

[10] E.D. Collins et al, "Can Thermal Reactor Recycle Eliminate the Need for Multiple Repositories", Rpt.# 37831-6170, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

[11] P. G. Boczar et al, "Plutonium Dispositioning in CANDU" *IAEA Technical Committee Meeting on Recycling of Plutonium and Uranium in Water Reactor Fuels*, Newby Bridge, U.K., July 1995.