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EVALUATION OF ABSORBERS EFFECTS IN THE DUPIC FUEL CYCLE BUNDLES FOR CANDU 6 REACTORS TYPE

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ABSTRACT

The DUPIC (Direct Use of spent PWR fuel In CANDU) cycle concept arose in the needs to find out another fuel cycle options that have planned to have positive environmental and economical effects. The reprocessed fuel (DUPIC fuel) results in a fissile content greater than natural uranium fuel so the core reactivity is augmented. In previous works depleted uranium has shown a potential application to adjust the initial core k_{eff} to the standard natural uranium fueled CANDU core. The present work aims to verify neutron absorber materials in sense of reactivity control and analyze the neutronics four factors so the system k_{∞} response to these schemes. Thorium, Hafnium and Dysprosium was employed as absorbers in fuel bundle central pins (4 and 7 fuel pins replacements) and results were compared with depleted uranium. The deterministic module of SCALE 6 code system was used to model the fuel cell problem application. The best results were achieved with Thorium and Hafnium applications.

1. INTRODUCTION

The Direct Use of spent PWR In CANDU (DUPIC) concept is a proposal of advanced fuel cycles for CANDU-PWR reactors and enhances fuel utilization through synergism of both reactors technical aspects. Implementation ways to turn this application reliable through novel reactivity control systems for CANDU reactor has been studied [–]. This work presents an assessment of neutronic response in sense of application of different absorbers materials due fact the higher reactivity in DUPIC fuel (about 1.5% of fissile content []). Thorium is used as an application problem to make a prime evaluation of reactivity reduction and the fact of his breeding properties []. The NEWT module of SCALE 6 was used to solve neutronics of the nuclear system applications [**Erro! Fonte de referência não encontrada.**], where considered the standard CANDU 6 fuel bundle, the DUPIC fuel (reprocessed fuel – RF) and RF plus absorbers of solid cylinders pins of Depleted Uranium (DU), thorium oxide, dysprosium oxide and hafnium oxide in sequence of previous works [– **Erro! Fonte de referência não encontrada.**].

2. METHODOLOGY

Calculations was performed using NEWT code in a fuel cell 28,575cm diameter standard 37-element CANDU-6 fuel bundle by composing 50×50 discretization matrix. Geometry and material data used for the 2D model (Fig. 2.**Erro! Fonte de referência não encontrada.**) are described in [– **Erro! Fonte de referência não encontrada.**]. The rods were subdivided to be a length shorter than neutron free mean path on deuterium to obtain an accurate result on fuel region. The application cases for configurations 4 and 7 pins substitutions are shown in Fig. 2.**Erro! Fonte de referência não encontrada.**. Convergence error was set to 10^{-5} on system criticality solution.

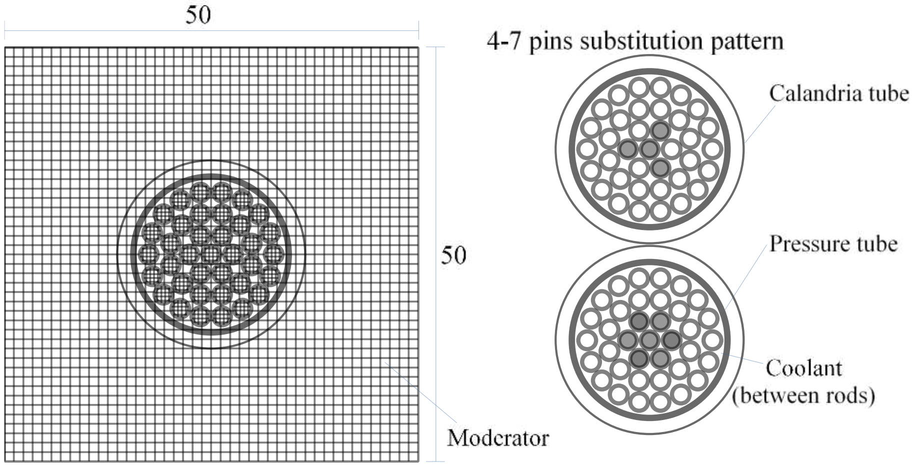

Fig. 2. .1. CANDU-6 fuel cell NEWT model scheme and 4-7 absorbers pins setup.

The application cases are labeled as NU (standard Natural Uranium bundle), RF (DUPIC reprocessed fuel), RF**xA**, where are **x** absorbers rods (4 or 7) of absorber **A** (DU, ThO_2, Dy_2O_3 and HfO_2) with physical densities respectively 10.6 [**Erro! Fonte de referência não encontrada.**,], 10.0, 7.8 and 9.7 g·cm^{-3}[].

3. RESULTS

In Fig. 3.**Erro! Fonte de referência não encontrada.** it is shown, as expected, a hardening on flux spectrum due RF material properties and one can observe the relative flux values drop to about 0.3 in thermal region in all RF applications. Should be stressed that the standard NU bundle case maintain the highest flux values overall energy range (10^{-5}eV – 20MeV). Fig. 3.**Erro! Fonte de referência não encontrada.** was plotted the difference between RF bundle and RF plus absorbers and its effects on flux.

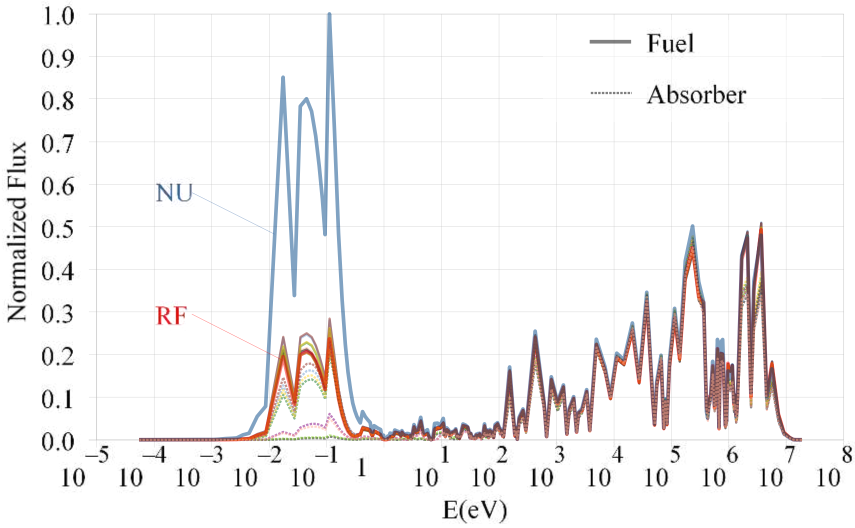

Fig. 3. .2. Normalized flux spectrum for all application problems.

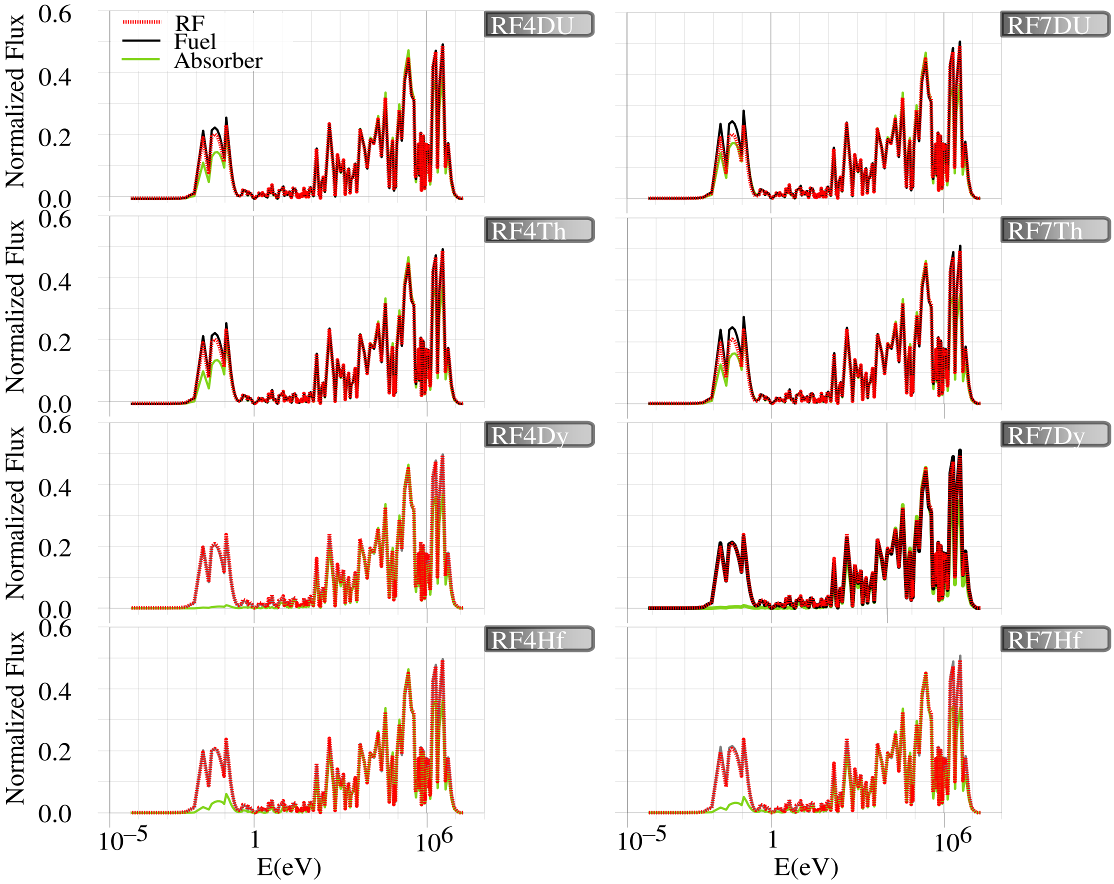

Fig. 3. .3. Normalized flux spectrum for RF-Absorber applications.

In Fig. 3.**Erro! Fonte de referência não encontrada.** the dotted line stands for the DUPIC fuel application only, and the black solid line refers to RF fuel problems combined with absorbers materials (green solid line).

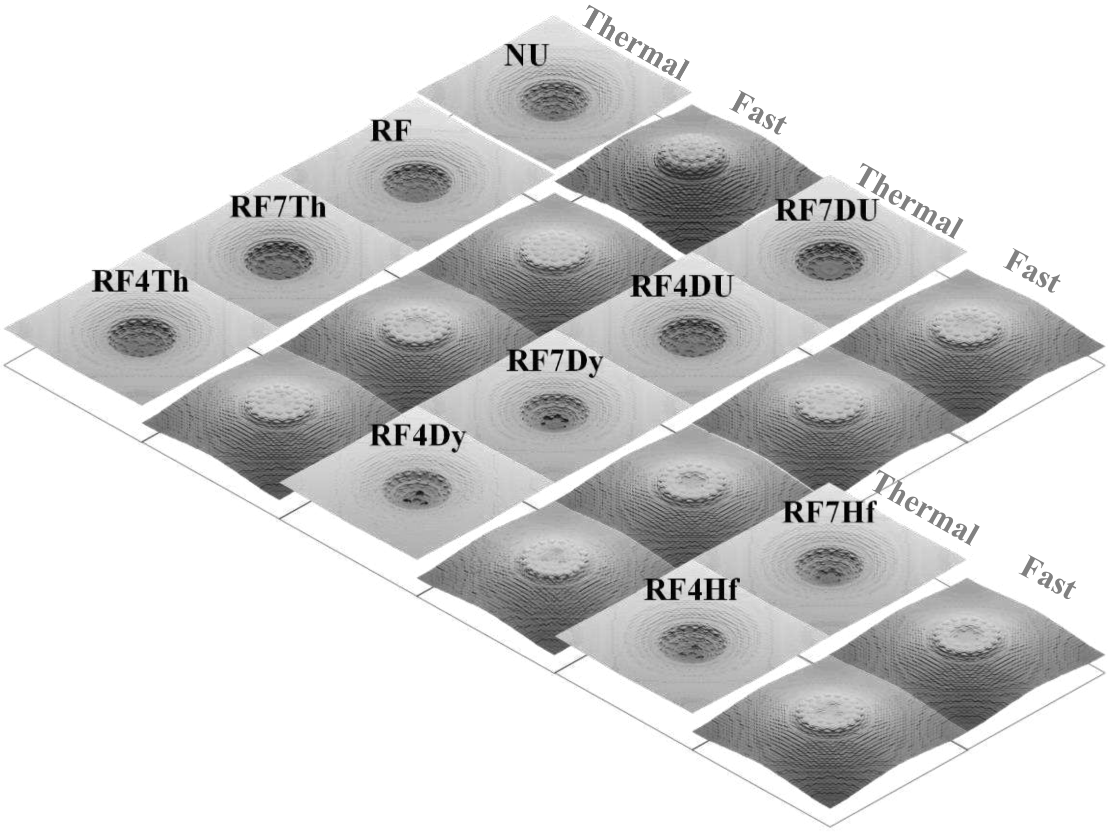

Fig. 3. .4. Spatial flux distribution along fuel bundle cases for 2-group problem.

In sense of the non-fission absorption reactions the Dy oxide cases reveal to be the stronger absorber due the expressive reduction of flux intensity on thermal region (<5%) resulting in a noticeable spatial self-shielding on fuel channel[1][]. The Fig. 3.**Erro! Fonte de referência não encontrada.** depicts an interpreted NEWT spatial flux distribution raw data to generate a flux surface distribution for 2 neutron energy groups (thermal from 10^{-5} – 4eV and fast 4eV – 20Mev[]) normalized to NU case where is shown the flux depression on absorbers positions. Again RF**x**Dy have the lower flux values at cell center showed by the darken region at the thermal range.

Tab. 3.**Erro! Fonte de referência não encontrada.** and Fig. 3.**Erro! Fonte de referência não encontrada.** are a summary of the system fission reaction rates over energy spectrum where details the contribution of each fissile isotope computed on SCALE code system on materials and absorber materials, when applicable. One found that the main isotopes are ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu, but no system enhance the fast-fissions probabilities keeping ^{235}U and ^{239}Pu the main fissile isotopes especially due the fact both are present in higher mass percentage. In a prime observation, it should be stressed that the RF+Th problem enhances the ^{239}Pu depletion and reduces the ^{235}U if compared with other

1 Composed of fuel and coolant inside the pressure tube on calandria vessel tubes.

bundle configurations (Tab. 3.**Erro! Fonte de referência não encontrada.**) so that could lead a boost on nuclear weapon proliferation resistance concept on DUPIC cycle.

Tab. 3. .1. System fission rates weights per fissile isotopes on applications.

	Isotope	Percentage values (%)									
		NU	RF	RF+DU		RF+Th		RF+Dy		RF+Hf	
				4 pins	7 pins	4 pins	7 pins	4 pins	7 pins	4 pins	7 pins
Fuel	^{230}Th		$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$
	^{233}U		$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$
	^{234}U	0.001	0.006	0.005	0.005	0.006	0.005	0.006	0.006	0.006	0.006
	^{235}U	**95.306**	**38.518**	**38.399**	**38.299**	**38.541**	**38.547**	**38.567**	**38.605**	**38.547**	**38.597**
	^{236}U		0.082	0.075	0.071	0.076	0.072	0.083	0.082	0.082	0.081
	^{238}U	**4.693**	**4.800**	**4.376**	**4.099**	**4.427**	**4.193**	**4.947**	**4.893**	**4.868**	**4.812**
	^{237}Np		0.024	0.022	0.020	0.022	0.021	0.024	0.024	0.024	0.024
	^{238}Pu		0.060	0.058	0.056	0.058	0.056	0.060	0.060	0.060	0.059
	^{239}Pu		**48.493**	**48.570**	**48.607**	**48.751**	**48.920**	**48.319**	**48.341**	**48.411**	**48.422**
	^{240}Pu		0.092	0.083	0.078	0.084	0.079	0.093	0.092	0.092	0.091
	^{241}Pu		**7.677**	**7.668**	**7.659**	**7.698**	**7.711**	**7.650**	**7.649**	**7.662**	**7.661**
	^{242}Pu		0.025	0.023	0.021	0.023	0.022	0.026	0.025	0.025	0.025
	^{241}Am		0.045	0.043	0.041	0.043	0.042	0.045	0.045	0.045	0.044
	^{242m}Am		0.104	0.104	0.104	0.104	0.105	0.103	0.104	0.104	0.104
	^{243}Am		0.005	0.005	0.005	0.005	0.005	0.005	0.005	0.005	0.005
	^{242}Cm		$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$
	^{245}Cm		0.069	0.069	0.069	0.069	0.069	0.069	0.070	0.069	0.070
	^{246}Cm		$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$	$<10^{-4}$
Absorber	^{232}Th					0.092	0.153				
	^{234}U			0.001	0.001						
	^{235}U			0.073	0.154						
	^{238}U			0.426	0.711						

Tab. 3.**Erro! Fonte de referência não encontrada.** and Fig. 3.**Erro! Fonte de referência não encontrada.** are a quantitative and qualitative frame of the four factor parameters of

system criticality, so due the hardening on neutron energy spectrum the fission density tend to be lower than NU but in other hand the higher fissile content results in a bigger system η for the application, naturally leading to higher values of the system multiplication factor where can be noticed on Tab. 3.**Erro! Fonte de referência não encontrada.** values keeping the RF only, the case with maximum value of k.

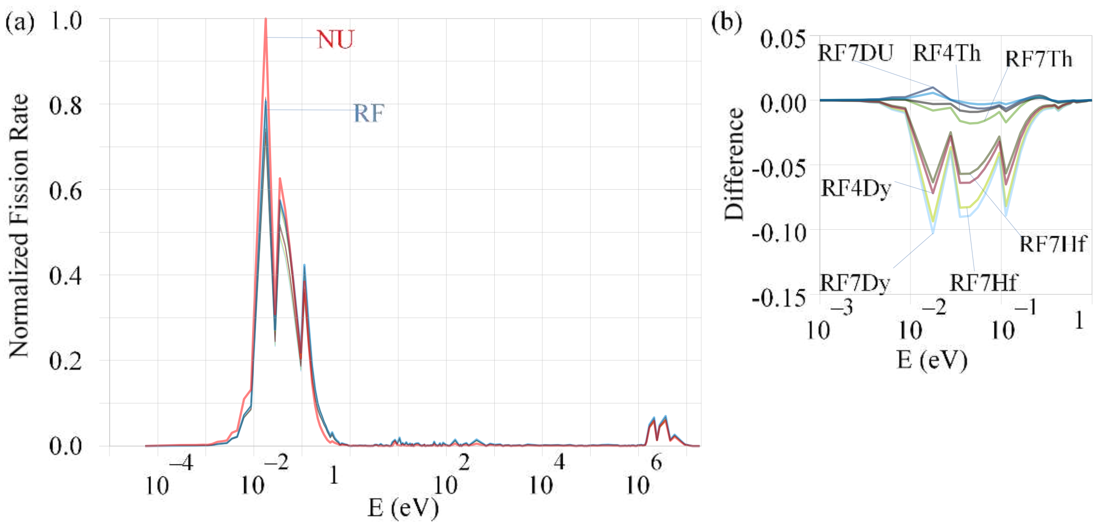

Fig. 3. .5. (a) System Reaction Rates. (b) Difference between RF-Absorber and RF only. Δ=0 means they are equal.

Tab. 3. .2. Criticality and four factor summary for all cases.

	NU	RF	RF4DU	RF7DU	RF4Th
k_{eff}	1.04214252	1.21192096	1.20355382	1.19495599	1.18748132
k_∞	1.04230800	1.21202200	1.20365600	1.19505400	1.18757700
η	1.31363700	1.49199700	1.48357600	1.47398900	1.46613600
f	0.86656200	0.92664000	0.92585100	0.92493100	0.92599300
p	0.84196300	0.77314700	0.77731700	0.78102200	0.77794100
ε	1.08749700	1.13388500	1.12733700	1.12232900	1.12443200
	RF7Th	**RF4Dy**	**RF7Dy**	**RF4Hf**	**RF7Hf**
k_{eff}	1.16437771	1.05877124	0.99572316	1.07623799	1.01228140
k_∞	1.16447500	1.05885800	0.99579900	1.07494700	1.01021000
η	1.43878300	1.49332400	1.49374200	1.49326500	1.49373800
f	0.92528400	0.83179600	0.79367500	0.84170100	0.80112700
p	0.78242600	0.75274700	0.74439200	0.75614900	0.74888400
ε	1.11793500	1.13244800	1.12837000	1.13105800	1.12725300

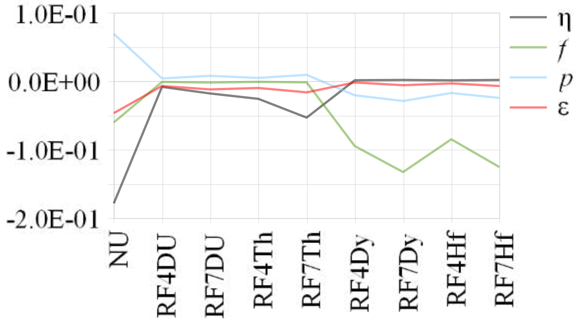

Figure 3. .6. Four factors differences to RF only case.

If we look to Fig. 3.**Erro! Fonte de referência não encontrada.**. we can notice that *f* values (RF**x**Dy and RF**x**Hf) drops to lower values than RF because these absorbers reduces the rate of neutron absorption in the fuel region considering isotopes of Dy and Hf that have microscopic cross-section around 10^3b at thermal energy of 0.025eV.
In relation of *p* and *ε* variations, the response of fast-fissions doesn't play a relevant change in the reaction rates and the absorber has the resonance region near of the energy range of the fuel causing small differences in all applications as observed for this particular system.

4. CONCLUSION

Beside the fact we can reach criticality values very near of the standard NU, the neutronic parameters are significantly different, but it is possible to reach a reasonable setup considering the reduction in the amount of fuel material (in RF) by replacing it with non multiplication media. In other hand this study showed that the use of DU or Th, can boost the depletion of plutonium in the fuel matrix in favor of fissile uranium isotope. For a more accurate assessment, thermal-hydraulics and kinetics analysis are needed to settle the operational limits for the proposed configurations.

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