THORIUM FUEL BUNDLE BEHAVIOUR UNDER IRRADIATION CONDITIONS. A COMPARATIVE STUDY

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Abstract

The aim of this paper was to perform thermo-mechanical calculations for Thorium-based fuel used in a CANDU 600 type reactor. Recent studies indicate that fuel bundles with different enrichment configuration have the potential to achieve a higher burn-up. In this regard, the study was conducted towards investigation of disparity of heterogeneous fuel bundle counter to the widely regarded homogenous type. The neutronic and thermo-mechanical fuel behaviour investigations were performed using the lattice code DIREN together with the thermo-mechanical codes TRANSURANUS, ELESIM-TORIU and ROFEM-TORIU, respectively. The following key parameters were taken into account for analysis: temperature, fission gas release, internal pressure and sheath plastic strain. The paper's conclusions underline the burnup and operation improvements brought by the utilization of Thorium mixed oxide placed in a heterogeneous configuration, with 30 % increase in fuel discharged burnup, being directly linked to the decrease of discharged fuel bundles.

Keywords: *Thorium, CANDU, TRANSURANUS, ROFEM, ELESIM*

Introduction

Actual trends in energy production emphasize the need for sustainability and implementation of low-carbon emission technologies [1]. In this regard, nuclear industry [2, 3] plays a key role by assuring a clean energy production and efficient utilization of mineral resources. In order to meet the future energy production challenges, an imperative need to develop advanced fuel cycles arises [4, 5].

This paper contributes to a broader study conducted towards finding of the most suitable composition for innovative fuels that can be irradiated in CANDU type reactors [6]. The outcome is closely related to the enhancement of nuclear fuel cycle efficiency by increasing the amount of energy extracted per fuel bundle.

Specifically, we were interested in displaying the enhance of nuclear performance under irradiation of Thorium-based mixed fuel inserted in a 37-element bundle with heterogeneous configuration in terms of fuel pellets enrichment, referred later on as HET bundle.

Thus, a preliminary study was conducted with the purpose of displaying the irradiation behaviour of singular elements placed on each bundle rings, designated as R1 (outer ring), R2 (intermediary ring) and R3 (inner ring) and the central pin (R4). Obviously, for a complete picture, the output had to be presented in a comparative manner. Therefore, a final study was performed, which assessed the in-pile behaviour of HET bundle versus a 37-element bundle with homogeneous configuration (HOM bundle). For this particular case we considered adequate to evaluate only the behaviour of outer elements, since they are most subjected to failure due to the higher thermal stress. In order to increase the accuracy of the results, the second study was conducted using three computer codes, namely: TRANSURANUS, ELESIM-TORIU, and ROFEM-TORIU.

Fuel bundle general description

For both considered fuels a single bundle geometry was used, i.e. 37-element CANDU standard fuel bundle [7], which is nowadays used in actual CANDU power reactors (37 fuel elements disposed in the CANDU widely-known configuration: one central pin, and 3 concentric rings (of 6, 12 and 18 elements). The 37- elements bundle geometry, fuel elements and (Th, U)O_2 fuel pellets are depicted in Fig 1.

Fuel pellets, for both considered bundles, were obtained from a mixture of SEU (Slightly Enriched Uranium) and (Th, U)O_2. In the case of HOM bundle, the fuel composition was characterized by: 9% content of U^{235} in the mixture, a wTh/U weight ratio of 0.3 corresponding to 77 - 23% heavy element (HE) share, and all fuel elements had the same pellet composition regardless of their position in the bundle's structure. Whereas, for the HET bundle the pellet composition was considered, as follows:

- **R4**. For the central pin only ThO_2 fuel was considered
- **R3 & R2**. For the elements placed on the inner and intermediary rings: 8% content of U^{235}, and wTh/U weight ratio of 0.667 (60 - 40% HE share)
- **R1**. For the elements placed on the outer ring: 8% content of U^{235} in mixture, wTh/U weight ratio considerably decreased to 0.257 (80 - 20% HE share).

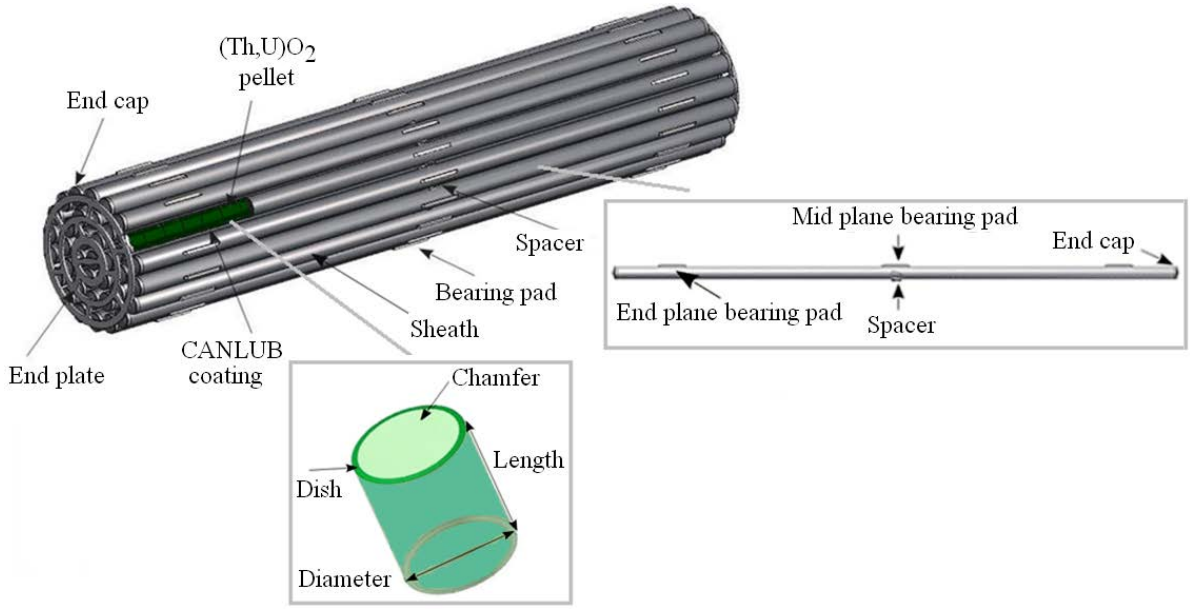


Fig. 1. 37-element CANDU fuel bundle geometry

The simulated configurations are described in Table 1 and illustrated in Fig. 2.

Table 1. Thorium and Uranium mixed oxide fuel configuration

Bundle type	**HOM**	**HET**		
	R1, R2, R3 & R4	**R1**	**R2 & R3**	**R4**
U^{235} content [%]	9	8	8	-
UO_2 content [%]	23	20	40	0
ThO_2 content [%]	77	80	60	100
Ratio wTh/U	0.3	0.257	0.667	-
Fuel Density [g/cm^3]	9	9	9	9

* R1 - Outer ring fuel element; R2 - Inner ring fuel element; R3 - Intermediary ring fuel element, R4 - Central fuel element

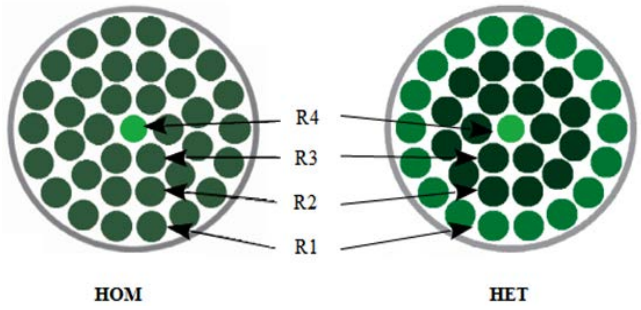


Fig. 2. Fuel elements configuration regarding the enrichment distribution: heterogeneous bundle (left) and homogenous bundle (right)

Fuel behaviour analysis

Lattice calculations were performed using the WIMS computer code [8] with the associated IAEA library [9], in order to obtain neutronic properties with respect to the burnup. The linear power envelope directly derives from burnup histories, it consisting in the maximum OELP values recorded at every burnup, from zero to exit (discharge) value.

In Table 2 are presented the Time-Average (TA) and Core Follow Simulations (CFS) results for the considered Thorium-based fuel bundles. The evidence indicates a reduction of discharged bundle number which is directly linked to the increase of average discharged burnup.

Table 2. Core results

Parameter	**HOM**	**HET**	**HOM vs. HET** (%)
Heavy Element (HE) mass per bundle [kg]	17.05	17.05	-
Number of discharged bundles	6216	3012	48.45
Average Burnup [MWd/kgHE]	12.86	17.89	- 28.11
Average Discharge Burnup [MWd/kgHE]	15.86	22.19	- 28.52
Maximum Bundle Power [kW]	893	983	- 9.15
Maximum Linear Power [kW/m]	58.3	58.6	- 0.51
Number of points above the OELP fuel failure curve	0	0	0
Number of points above the OELPB fuel failure curve	1458	2308	- 38.83
Number of points overriding both fuel failure curves	0	0	0

* OELP - Maximum Outer Element Linear Power,
OELPB - Maximum Outer Element Linear Power Boost

Heterogeneous bundle assessment

In order to perform this study, the most important thermo-mechanical parameters were assessed using the ELESIM-TORIU computer code.

The results represented in comparative manner for the fuel elements placed on each bundle ring were, as follow:
- linear power for outer fuel element,
- fuel pellet central and surface temperature,
- fuel sheath inner and outer surface temperature,
- radial gap,
- plastic strain deformation,
- fission gas release volume and
- internal pressure.

These parameters are depicted in Figs. 3 – 6, with respect to fuel burnup.

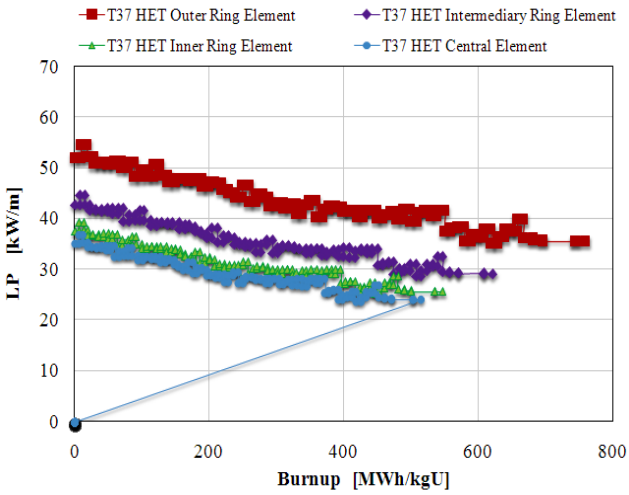


Fig. 3. Input data – irradiation histories - *LP*

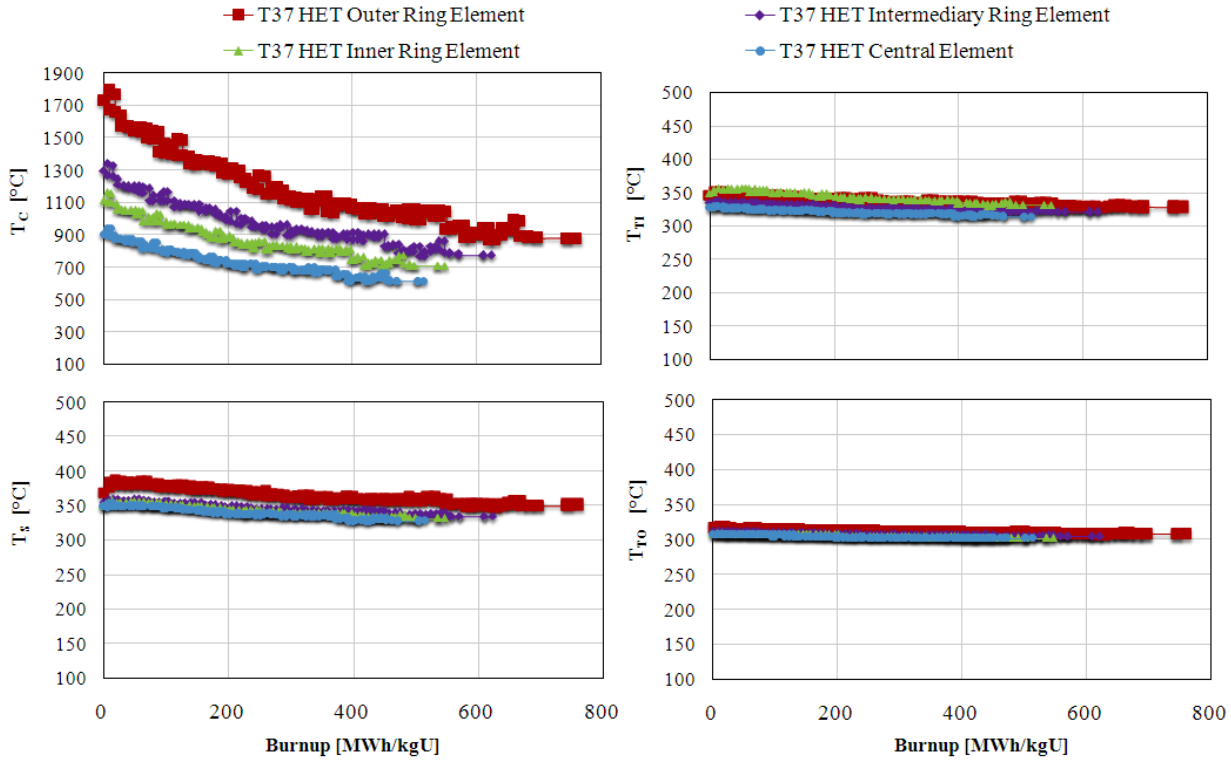


Fig. 4. Evolution of temperature in various regions: central pellet - T_C (up left), pellet surface - T_S (bottom left), inner sheath - T_{TI} (up right) and outer sheath - T_{TO} (bottom right)

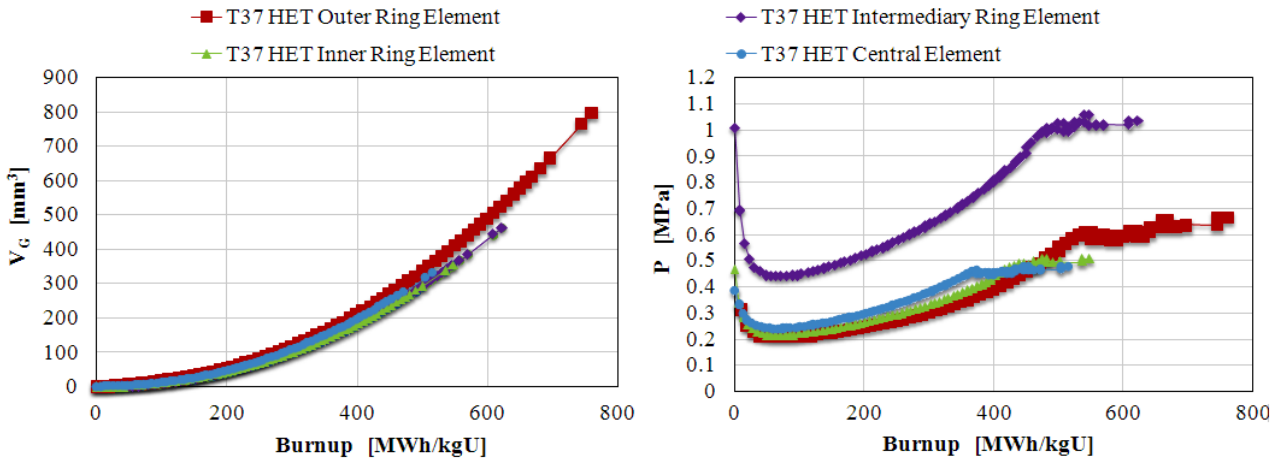


Fig. 5. Evolution of fission gas release - *V* (left) and internal pressure - *P* (right) with respect to burnup

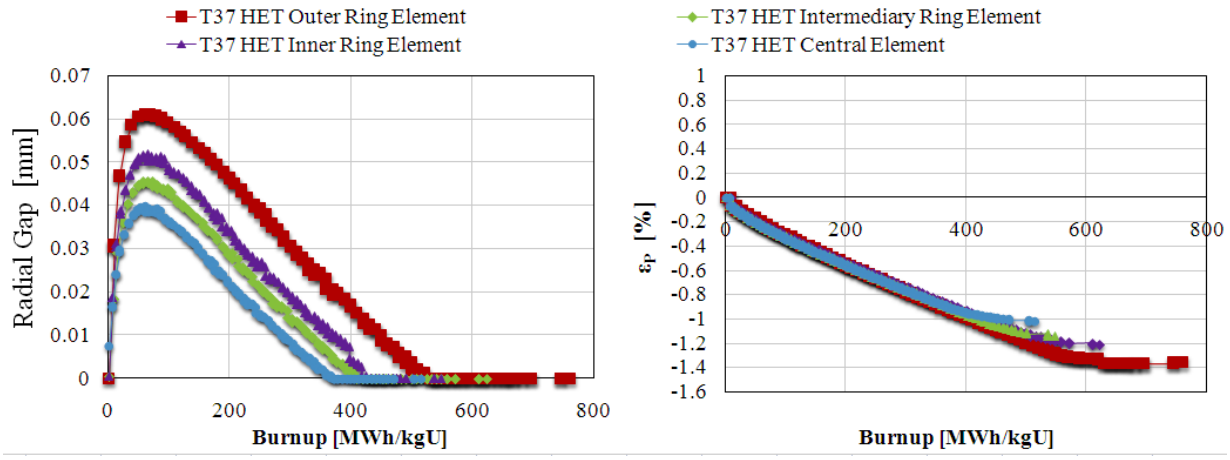


Fig. 6. Evolution of radial gap (left) and plastic sheath deformation - ε_P (right)

The central pellet temperature evolution, for all considered elements, follows the linear power shape, very closely. As a direct consequence of having the largest linear power values, the outer element has the highest operation temperatures in the centre and on the surface of pellets, an increase with 454 °C (25 %) and 25 °C (7 %) being observed.

At the beginning of the irradiation, the heat transfer between pellet and sheath marks an abrupt decrease, corroborated with the sheath collapsing on pellets stack, suggested in Fig. 3. As irradiation proceeds, a small amount of fission gasses are released therefore the difference between the pressure inside the fuel element (1 MPa) and the outside environment (coolant, 10.4 MPa) remains high, as shown in Fig. 5.

This behaviour is reflected in the evolution of pellet surface, inner and outer sheath temperature which remain almost constant throughout the simulated period. The maximum values obtained for the assessment of HET bundle are presented in the table below.

Table 3. Maximum values obtained for the parameters of interest using ELESIM-TORIU code

Computer Codes	Fuel type	Element location	k	T_C	T_S	T_{TO}	T_{TI}	V_g	P	gap	ε_P
ELESIM-TORIU	HET	R1	60.70	1798.51	385.51	349.30	317.70	797.48	0.66	0.061	0.0006
		R2	48.58	1343.81	360.36	338.46	312.48	464.02	1.06	0.045	-1.215
		R3	35.86	1164.38	357.34	357.1	309.62	353.73	0.51	0.052	-1.146
		R4	48.34	940.93	352.86	329.92	308.38	331.77	0.47	0.0399	-1.021

k- Heath Transfer coefficient, [kW/m2-K]; T_C - Pellet Central Temperature, [°C]; T_S - Pellet Surface, [°C]; T_{TO} - Fuel Sheath Outer Surface Temperature, [°C]; T_{TI} - Fuel Sheath Inner Surface Temperature, [°C]; V_g - Gas volume, [mm^3]; P - Gas pressure, [MPa]; gap - radial gap, [µm]; ε_P - Sheath elastic deformation, [%]

Heterogeneous versus Homogeneous bundle assessment

In this section, the thermo-mechanical behaviour of thorium based CANDU fuel elements is shown with a comparison between ELESIM-TORIU, ROFEM -TORIU and TRANSURANUS predictions.

The measurements have been carried out on the five thermal aged samples and on the non-aged cable, six cable samples in all. The IM values for the electric cable material made of PVC are presented in Table 3, [13].

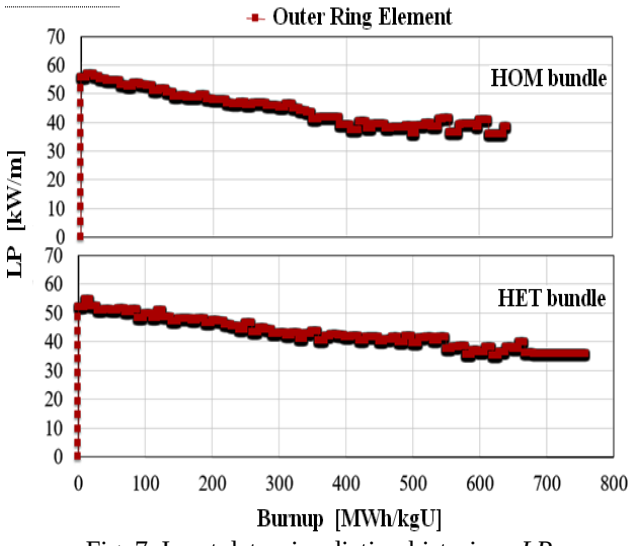


Fig. 7. Input data - irradiation histories - *LP*

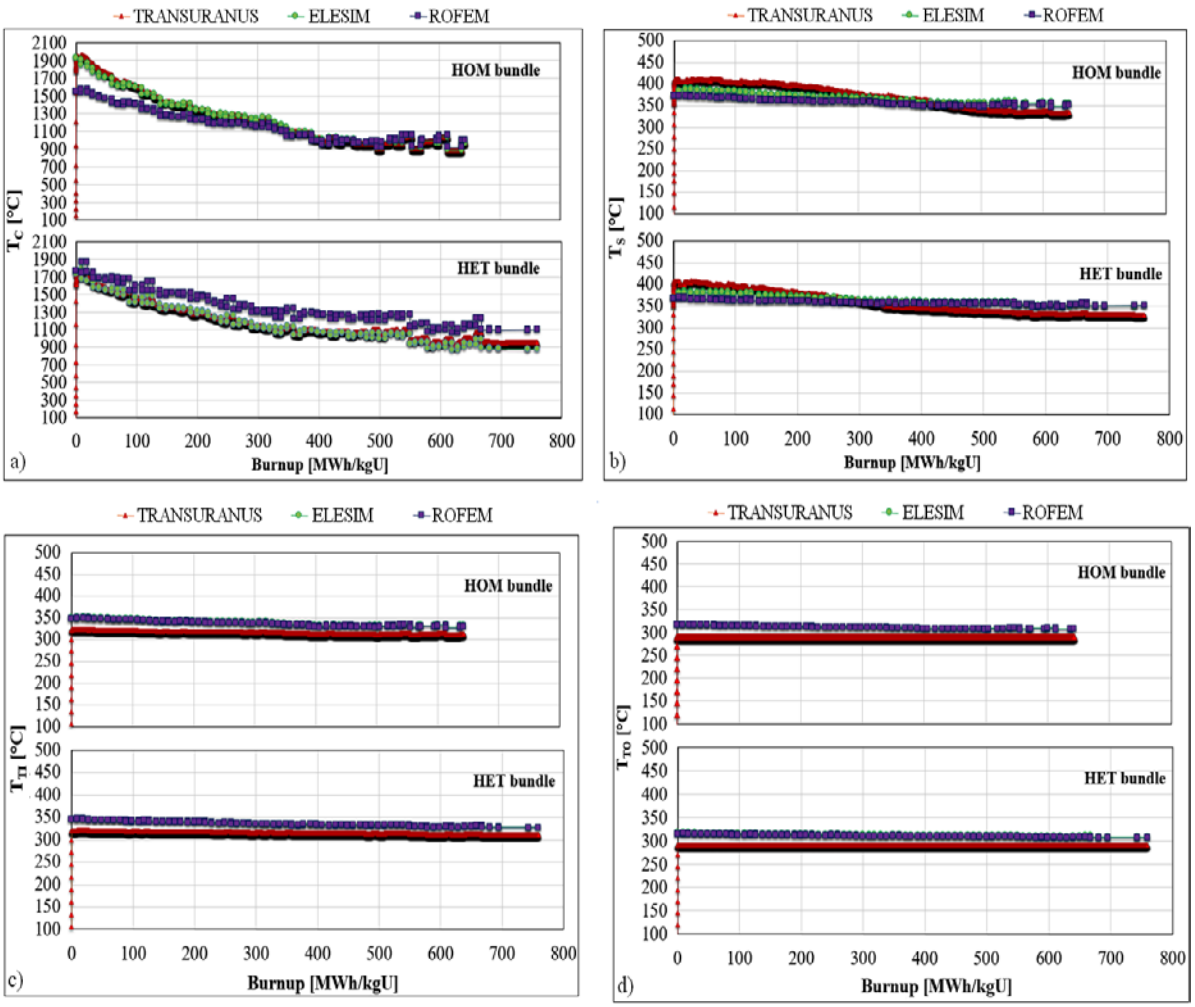


Fig. 8. Evolution of temperature in various regions: central pellet - T_C (a), pellet surface - T_S (b), inner sheath - T_{TI} (c) and outer sheath - T_{TO} (d)

Fig. 8 illustrates ELESIM-TORIU, ROFEM-TORIU and TRANSURANUS codes results for the fuel temperature evaluated on pellet region (central and surface) and sheath region (inner and outer). Good agreement seems to occur between the results generated by ROFEM-TORIU and ELESIM-TORIU, except the temperature calculated for central region. For this particular case, computations carried out by TRANSURANUS and ELESIM-TORIU codes follows the central temperature curve, in a very precisely way.

The assessment of fission gas release with burnup and implicitly the build-up of internal pressure inside the clad are presented in Fig. 9.

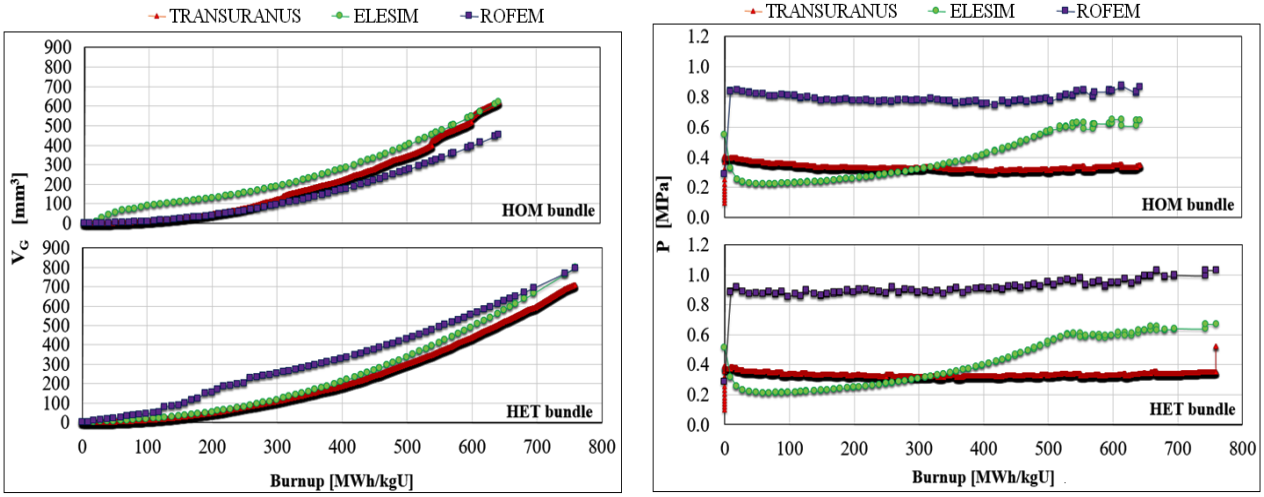


Fig. 9. Evolution of fission gas release - V (left) and internal pressure - P (right) with respect to burnup

Regarding the assessment of fission gas release acceptable agreement between the utilized computer codes are obtained, while significant dissimilarities occur for the evolution of internal pressure. The difference between codes predictions might be assigned to different models used for the evaluation of fission gas volumes. As far, the volume released by the HET bundle is faintly increased and the internal pressure does not exceed 0.9 MPa.

Fig. 10 shows the radial gap evolution and plastic sheath deformation, with respect to the burnup. Here are signalled some dissimilarities for HET and HOM behaviour. In the case of HOM bundle, the radial gap attains its maximum value at the beginning of irradiation period; it remains constant in the prediction of TRANSURANUS code and decreases evenly for ELESIM-TORIU and ROFEM-TORIU computations. Meantime, for the HET bundle a good agreement was obtained.

The comparative analysis indicates an overall good agreement between the results obtained using the three codes. However, small differences regarding the evaluation of radial gap and internal pressure are reported in Table 4.

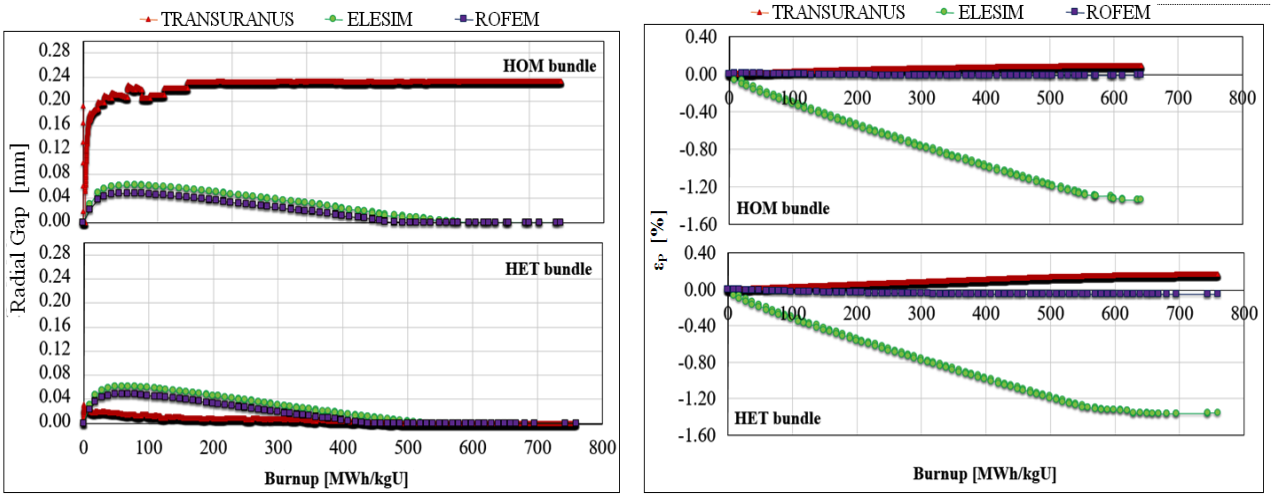


Fig. 10. Evolution of radial gap (left) and plastic sheath deformation - ε_P (right)

Table 4. Maximum values obtained for the parameters of interest using the three codes

Computer Codes	**Fuel type**	k	T_C	T_S	T_{TO}	T_{TI}	V_g	P	gap	ε_P
ELESIM-TORIU	HOM	62.54	1929.62	389.61	351.69	318.85	618.91	0.64	0.062	0.001
	HET	60.70	1798.51	385.51	349.30	317.70	797.48	0.66	0.061	0.0006
TRANSURANUS	HOM	42.10	1957.07	412.54	323.58	289.84	621.89	0.41	0.233	0.088
	HET	42.15	1789.94	408.71	322.33	289.85	710.04	0.52	0.03	0.16
ROFEM-TORIU	HOM	46.70	1577	346.68	349.96	316.45	452.04	0.74	0.048	0.008
	HET	62.44	1871.26	370.44	347.65	315.39	793.94	0.85	0.05	0.00

Table 5. Code-by-code comparison for the parameters of interest

Computer Codes	**Fuel type**	k	T_C	T_S	T_{TO}	T_{TI}	V_g	P	gap	ε_P
ELESIM-TORIU vs TRANSURANUS	HOM	48.55%	-1.40%	-5.56%	8.69%	10.01%	-0.46%	56.09%	-73.39%	-98.9%
	HET	44.01%	0.48%	-5.68%	8.37%	9.61%	12.31%	26.92%	103.33%	-99.6%
ELESIM-TORIU vs ROFEM-TORIU	HOM	33.92%	22.36%	12.38%	0.49%	0.76%	36.91%	-13.5%	29.16%	-87.5%
	HET	-2.77	-3.89%	4.07%	0.47%	0.73%	0.45%	-21.4%	22%	-

Conclusions

A thermo-mechanical analysis has been performed for thorium-based fuels inserted in a 37-element fuel bundle geometry using ELESIM-TORIU, ROFEM-TORIU and TRANSURANUS computer codes. The main goal of the present paper was to emphasize the increase of nuclear performance under irradiation of a CANDU standard fuel bundle with heterogeneous configuration, in terms of pellets enrichment.

For the present study, two cases were considered, namely:
- a heterogeneous bundle assessment using ELEIM-TORIU computer code where four elements from each bundle ring were evaluated
- a comparative evaluation of heterogeneous and homogenous fuel bundle using all three above mentioned computer codes, where a single element from the outer ring for each fuel bundle was evaluated.

The results for the first case reveal the improvement of thermo-mechanical performance of heterogeneous bundle, statement validated by temperature reduction, around 8%. Likewise, the output of the second study is overall in a good agreement.

Slight differences occurred for the assessment of some of the analysed parameters, such as radial gap and internal pressure; they are correlated with different approximations models implemented in codes structure.

Moreover, by supplying a higher Average Discharge Burnup (30% increase) and splitting the consumption of discharged bundle number, it can be concluded that the heterogeneous fuel bundle with considered configuration can improve the irradiation performance of Thorium-based mixed oxide fuel.

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