NEUTRON ANALYSIS OF REPLACEMENT POTENTIAL OF $^{NAT}UO_2$ ROUTINE FUEL OF A TYPICAL HEAVY WATER REACTOR WITH DIFFERENT (ThO_2-$^{ENRICHED}UO_2$) MATRIXES

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Abstract

Many interests exist with thorium-based fuels that make them outstanding for utilizing as economic alternatives. Mixture of ThO_2-UO_2 is widely investigated in both thermal-conductivity and neutron behavior point of views. Improved physical properties, elongated cycle length, breeder proficiency and proliferation resistance are the most attractive features of these kinds of the thorium-based fuel. In the present study, neutronic evaluation of three different uranium-enriched ThO_2-UO_2 fuel loads in a heavy water research reactor instead of $^{nat}UO_2$ routine loading has been proposed. The computational MCNPX 2.6.0 code was used to model the neutron parameters of a research core. The mixed fuel contents were fixed so that presents the same excess reactivity of $^{nat}UO_2$ routine loading. The UO_2 enrichment of the mixed thorium-based fuel was changed as 5, 10 and 15%. The obtained computational results showed ^{235}U weight fraction enhancement in the mixed fuel increases fission per absorption up ~71% in comparison with the routine loading. The fuel temperature reactivity coefficient does not significantly change. The UO_2 coolant and moderator temperature reactivity coefficient is considerably more negative than the thorium-based mix fuel. However, the (ThO_2-UO_2) mixed fuel resulted in higher negative fuel temperature reactivity coefficient, but its total reactivity coefficient is less negative because of the higher positive coolant void reactivity coefficient. All uranium- enriched mixed oxide fuels could obtain a total negative reactivity effects but are considerably less than the $^{nat}UO_2$ routine loading. The different mixed fuels did not result in far β and β_{eff} parameters than the routine loading. The fuel incineration rate calculation showed reduction of UO_2 quota in the mixed fuel to ~11%, (ThO_2-$^{15\%\text{-Enriched}}UO_2$) decreased ^{239}Pu production up <1.5 kg after 29.2 GWd burn-up, which is one tenth of the routine burnt fuel. Longer cycle length is another advantage of the used mixed fuel; all the investigated fuels resulted in approximately identical the cycle elongation. Approximately constant reactivity worth of the ^{135}Xe and ^{149}Sm neutron poisons during the burn-up of the mixed fuel can be considered as another advantage of the investigated thorium-based fuel in comparison with $^{nat}UO_2$ fuel.

Key words: neutron parameters, UO_2, (ThO_2-UO_2), MCNPX2.6.0, thorium-based fuel, burn up

1. Introduction

The thorium fuel cycle has several interesting advantages such as more chemically stable property, higher radiation resistance and favorable thermophysical properties over UO_2-based fuels. Also, thorium is three to four times more abundant in the earth's crust than uranium. It should be mentioned contrasting the ^{238}U/Pu-cycle, the Th/^{233}U-fuel cycle can therefore also achieve breeding for thermal neutron spectra [1].

Recently, more emphasis has been placed on decreasing the long-term toxicity of nuclear waste. In this respect, the fact that the toxicity of minor actinides (MA's) resulting from thorium reactors is at least one order of magnitude less compared to MA's resulting from uranium reactors; favors the thorium fuel cycle. Such a fuel cycle will generate a minimum quantity of actinide waste, the radiotoxicity of which would be lower than the existing reactors workingon ^{238}U -^{235}U /^{239}Pu fuel cycle for the first 50 000 years after disposal [2].
Thorium fuel can be used in all type of reactor such as the BWR [3], PWR [4], fast reactor [5], ADSR, fusion–fission hybrid [6] and PHWR [7]. Many thorium-based fuel assemblies were investigated and tested until now [8-14]. Hence, neutron investigation of different thorium-based fuel usage in a typical heavy water-cooled and moderated reactor has been proposed in this work. The most important research aim is investigation of a thorium-based fuel matrix to be used instead of the UO_2 fuel alternatives in the research reactor to achieve non-proliferation issues in weapon-grade ^{239}Pu production.

2. Methodology

This study evaluates neutron behavior of three different uranium-enriched ThO_2-UO_2 fuel loads in a typical heavy water research reactor instead of $^{nat}UO_2$ routine fuel. The research reactor is a heavy water-cooled and moderated. The core is fueled by 150 fuel assembly contained 18 fuel rods.
The 3D thermal reactor core was modeled by MCNPX 2.6.0 computational code in details [15]. The cross sectional view of the hexagonal-configurated core is shown in Fig.1.

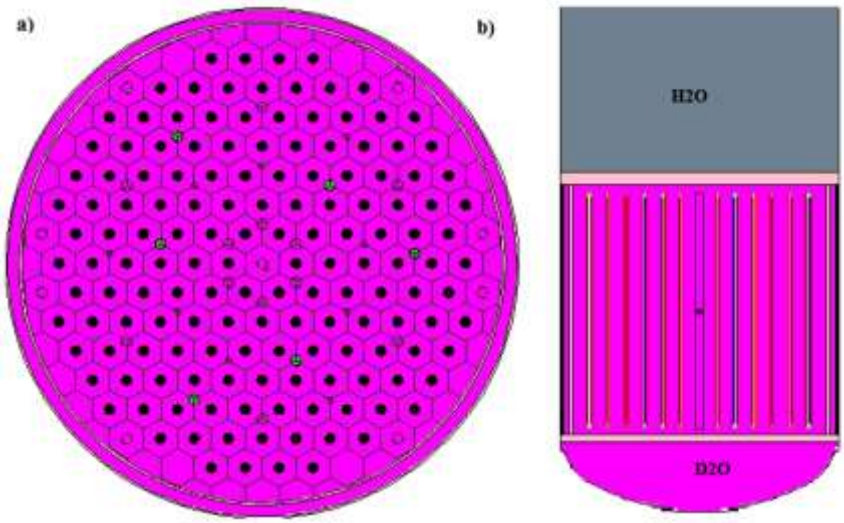

Figure 1 Cross sectional view of the modeled core a) Radial, b) Axial

Four types of fuel with different composition are studied in the modeled reactor as presented in Table 2. For the (ThO_2-UO_2) mixed fuels, weight fraction of fissile element have been fixed so that the excess reactivity of the loaded fresh cores are in good conformity with difference of <30 pcm.

Table 1 Elements weight fraction in 4 types of fuel with different composition

Fuel/ Element	Composition ThO_2%-UO_2%	Enrichment of uranium	O%	^{235}U%	^{232}Th%	^{238}U%	Density (g/cm^3)
UO_2 natural	0-100	0.7%	0.11853	0.006170	–	0.87531	10.14
(ThO_2-UO_2)	71.22-28.78	5%	0.12045	0.01268	0.62587	0.24099	9.6757
(ThO_2-UO_2)	84.38-15.62	10%	0.12081	0.01377	0.74152	0.1239	9.5945
(ThO_2-UO_2)	89.31-10.69	15%	0.12094	0.01413	0.78484	0.08008	9.5645

To compare $^{nat}UO_2$ routine fuel and three different uranium-enriched ThO_2-UO_2 fuels on neutron performances of the modeled core, the following neutron parameters have been investigated. Neutron spectra and average fission per absorption ratio have been calculated. Reactivity coefficients of fuel, coolant, moderator and void reactivity variations of the coolant have been calculated for the different fuel loads in the core. Delayed neutron fraction and effective delayed neutron fraction have been calculated. Burn up calculation has been performed at 40MW power for 2 years.

3. Results and discussion

According to the obtained computational results presented in Table 2, the (ThO_2-$^{Enriched}UO_2$) fuels did not result in far effective delayed neutron fraction (β_{eff}) and delayed neutron fraction (β) than the $^{nat}UO_2$ routine loading. The calculated results showed that the uranium weight fraction enhancement in the mixed fuel increases fission per absorption up and about 71% in comparison with the routine fuel loading.

Table 2 Evaluation of neutron parameters of the modeled reactor fed 4 types of fuel

Neutronic parameters	UO_2	ThO_2-$^{(5\%)}UO_2$	ThO_2-$^{(10\%)}UO_2$	ThO_2-$^{(15\%)}UO_2$
Effective multiplication factor	1.06102	1.06118	1.06110	1.06091
Excess reactivity(pcm)	5751	5765	5758	5741
Effective delayed neutron fraction (pcm)	662	655	642	647
Delayed neutron fraction (pcm)	687	656	658	636
Fission per absorption ratio	0.48	0.76	0.81	0.82
Fuel temperature reactivity (pcm/℃)	-1. 629	-1.919	-1.801	-1.890
Coolant temperature reactivity (pcm/℃)	-3.548	-1.887	-1.761	-1.706
Moderator temperature reactivity (pcm/℃)	-8.57	-4.609	-4.300	-4.221
Void reactivity (pcm /%void)	+3.461	+5.499	+5.987	+5.147
Radial power peaking factor	1.496	1.491	1.489	1.493
Axial power peaking factor	1.410	1.409	1.397	1.421

*****The variation was calculated during transit from 293 K to 599 K

The fuel temperature reactivity coefficients of the thorium-based mix fuels are nearly identical with the $^{nat}UO_2$ fuel. However, the (ThO_2-$^{5\%}UO_2$) mixed fuel resulted in higher negative fuel temperature reactivity coefficient than $^{nat}UO_2$ fuel with approximately relative discrepancy of 18%.The coolant and moderator temperature reactivity coefficients of the modeled core fueled $^{nat}UO_2$ are significantly more negative than the different mixed fuels. The coolant void reactivity coefficients of (ThO_2-UO_2) fuels are more positive than UO_2 fuel, after

10% volumetric void formation and the values are almost constant, which are close to each other (Fig.2).

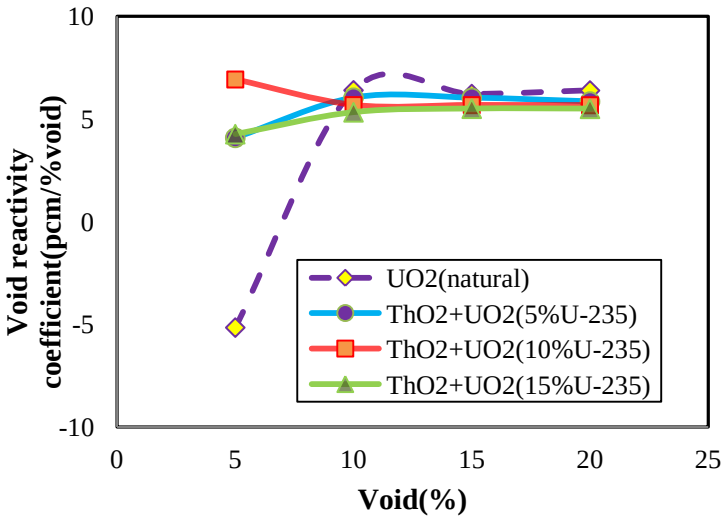

Figure 2 Dependence of void reactivity coefficient on volumetric void percent

Due to over-moderated design of the research reactor, moderator per fuel volume decreases during void formation in case of all the investigated fuel matrixes so that the modeled core bears a positive reactivity during void formation.

As shown in Table 3, assembly-wise power peaking factor differences between the different fuel loads in the modeled thermal reactor are insignificant.

Neutron spectra have been separately calculated for the core fueled with the four types of fuels. The nuetronic calculations demonstrated that the neutron spectra are nearly close in thermal section for (ThO_2-$^{enriched}UO_2$) in comparison with $^{nat}UO_2$ fuel; the 5%-enriched mixed oxide fuel presented fitter data. The thorium-based fuels resulted in noticeably underestimated data than the $^{nat}UO_2$ fuel in epithermal region as well as fast region up E_n<6 MeV (Fig. 4).

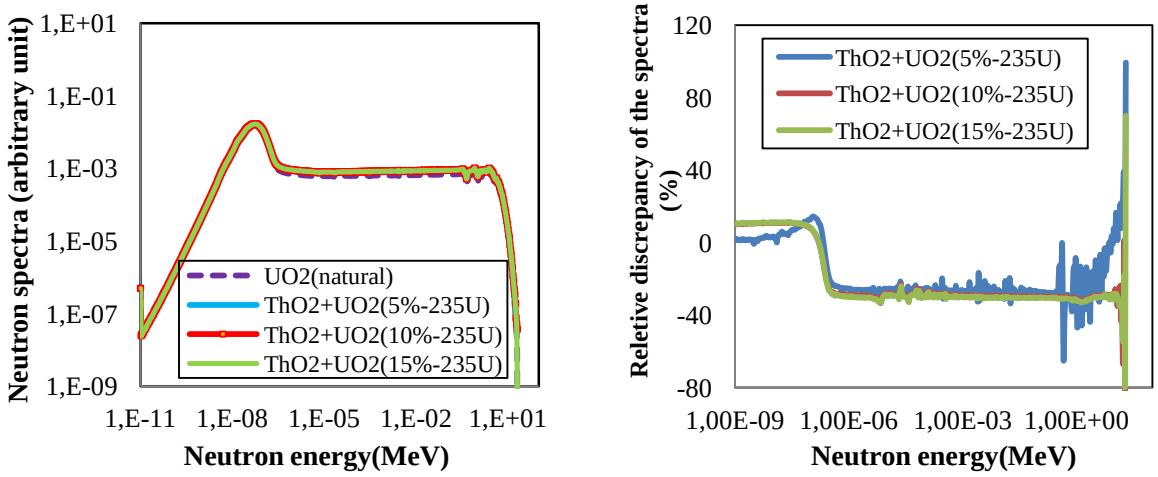

Figure 3 Neutron spectra and reletive discrepancy spectra of the mixed oxide fuel to the routine fuel

Fuel reactivity variation calculations during burn-up at 40 MW power displays the cycle

length is longer for the thorium-based mix fuels than $^{nat}UO_2$ fuel (Fig.4). The considerably-enhanced cycle length of (ThO_2-$^{\%Enriched}UO_2$) comparing the $natUO_2$ fuel is one of benefits of thermal reactor fueling with thorium–based matrixes; nevertheless ^{235}U initial load in thorium based fuel was double than the corresponded value of $^{nat}UO_2$ fuel.

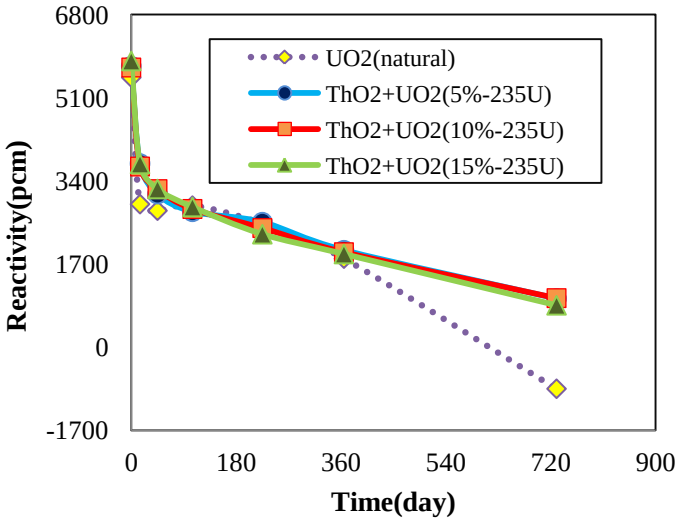

Figure 4 The modeled core reactivity variation on the burn-up at 40 MW power

The reactivity falling at the beginning of the cycle is directly related to ^{135}Xe and ^{149}Sm buildups. Whereas, thorium-based fuels have higher conversion ratio, ^{233}U fissile inventory compensates the sharp dropping which is seenin case of UO_2 fuel. As it is seen at Fig.5, a neutron flux in order of 10^{13} n/s·cm^2 is accessible in the thermal core; UO_2loading provides higher integrated neutron flux than the thorium-based fuel (~1.3 times).

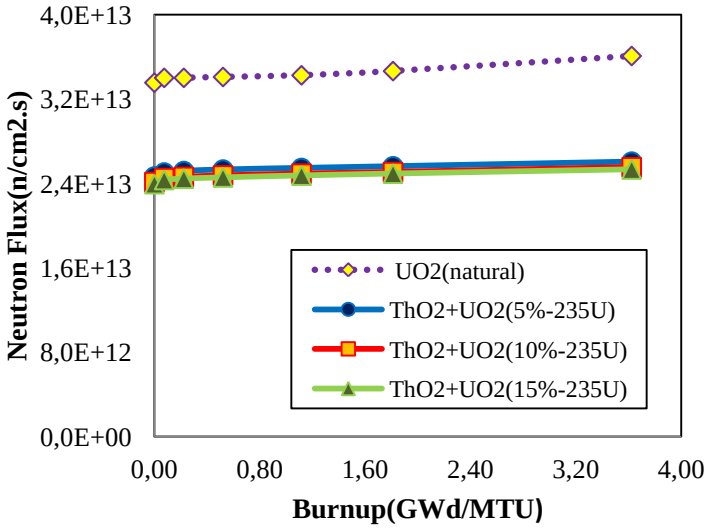

Figure 5 Dependence of neutron flux on the burn-up for different fuel loads

Dependence of the ^{135}Xe and ^{149}Sm mass production on the burn up time is presented in the Fig. 6 for four types of fuel. As the figure shows, the ^{135}Xe and ^{149}Sm production during the

(ThO_2-UO_2) burn-up is the most while the isotope concentration values are the least in case of $^{nat}UO_2$.

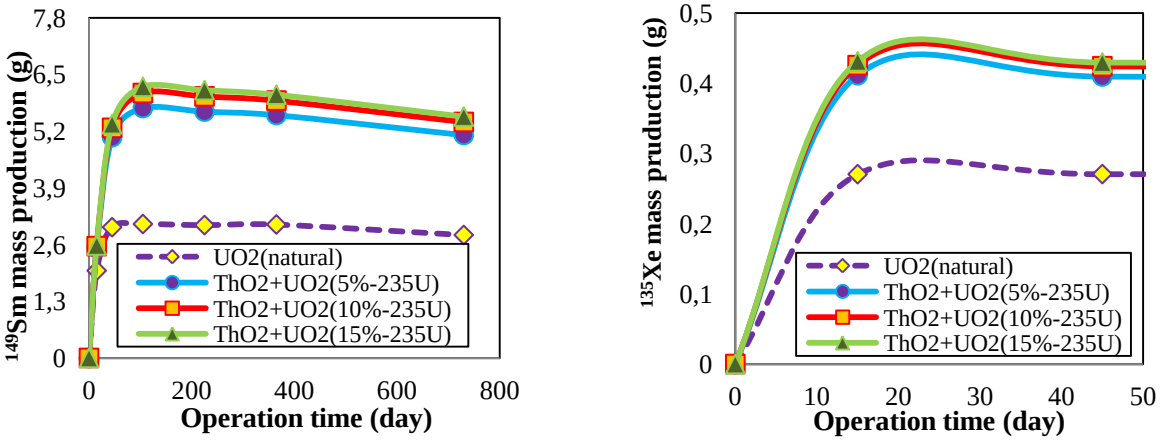

Figure 6 Comparison of ^{135}Xe and ^{149}Sm buildup during burn-up of the different fuels at 40 MW power

The dependence of ^{135}Xe and ^{149}Sm reactivity worth during the burn-up time is shown in Fig.7 for the different investigated fuels. As the figure displays, the modeled core fueled with the thorium-based mixed oxide fuel holds more independence to ^{135}Xe and ^{149}Sm fluctuations.

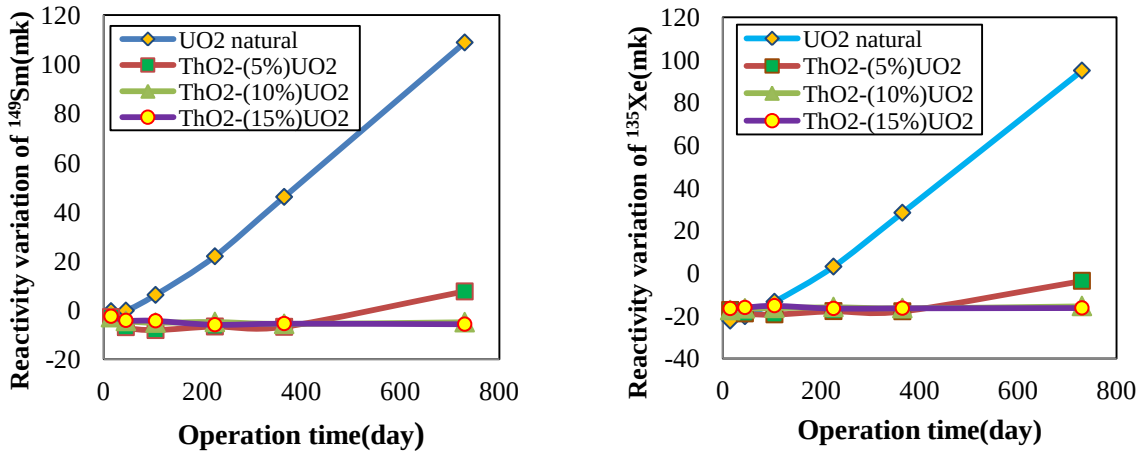

Figure 7 Comparison of ^{135}Xe and ^{149}Sm reactivity variation at 40 MW power

Dependence of the ^{233}Pa and ^{233}U mass production on the burn up time is presented in the Fig.8 for the thorium-based fuels. As the figure displays, the ^{233}Pa and ^{233}U production during the (ThO_2-$^{\%15}UO_2$) burn-up is more than the others.

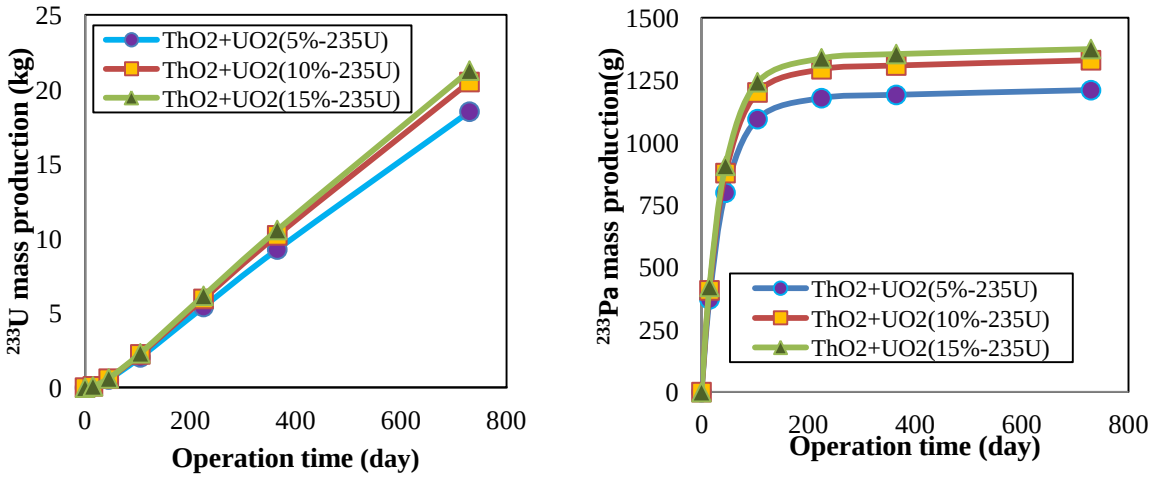

Figure 8 Comparison of ^{233}Pa and ^{233}U buildup duringburn-up of the different fuels at 40 MW power

According to the obtained computational data, which is presented in Fig. 9, 15 kg of ^{239}Pu was produced at end of cycle in the $^{nat}UO_2$ spent fuel. The burn up calculations showed increase of weigh fraction of ^{235}U in the mixed fuel up15% decreased ^{239}Pu mass concentration up <1.5 kg after 2 years burn-up at 40 MW power (29.2 GWd), which the produced value is one tenth of the routine burnt fuel.

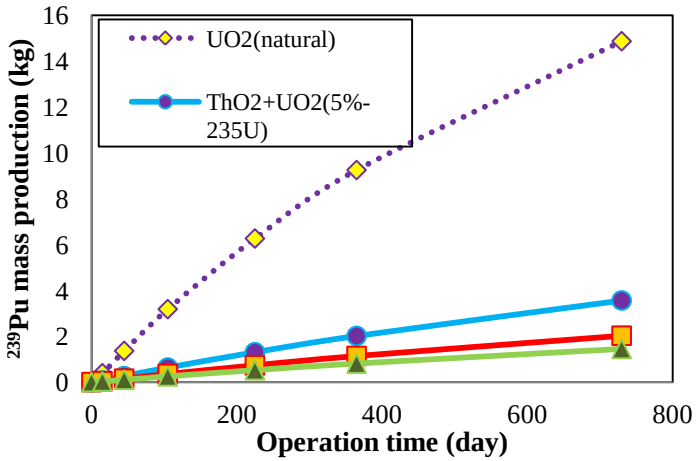

Figure 9 Comparison of ^{239}Pu buildup duringburn-up of the different fuels at 40 MW power

According to the achieved results presented in Table 3, maximum ^{233}U mass production after 2 years burn up is related to (ThO_2-$^{(15\%)}UO_2$) with 21.28 kg value.

Table 3 Comparison of inventory and consumption of fissile and fertile isotopes in different fuel (burn up time=2 year)

Fuel type	Inventory (kg)			Consumption(kg)		
	^{235}U	^{233}U	^{239}Pu	^{235}U	^{232}Th	^{238}U
UO_2	30.87	–	14.88	26.36	–	26.00
ThO_2-$^{(5\%)}UO_2$	86.01	18.52	3.56	31.52	23.00	5.00
ThO_2-$^{(10\%)}UO_2$	94.12	20.51	2.026	32.08	25.00	2.026
ThO_2-$^{(15\%)}UO_2$	97.28	21.28	1.449	32.32	26.00	1.458

4. Conclusion

Scientific reports show that Th-based fuels offer advantages over U-based fuels due to this fact that their proliferation resistance characteristic makes them as suitable alternatives for nuclear reactors. The obtained results in this work state that the β and β_{eff} for the modeled core fed (ThO_2-$^{Enriched}UO_2$) fuel and $^{nat}UO_2$ fuel are not considerably changed. Only the fuel temperature reactivity coefficients of the thorium-based mix fuels are more negative than $^{nat}UO_2$ while the other reactivity coefficients of the mixed fuels became more positive than $^{nat}UO_2$ routine fuel. However, the safety parameters of the modeled research core disturbed somehow by (ThO_2-$^{Enriched}UO_2$) loading, but it keeps the safe operation of the core if the coolant fluid two-phasing occurrence be avoided. Nevertheless its longer cycle length and lower ^{239}Pu mass production in the spent fuel are the advantages of the used mixed fuel. Approximately constant reactivity worth of the ^{135}Xe and ^{149}Sm neutron poisons during the burn-up of the mixed fuel can be considered as another advantage of the investigated thorium-based fuel in comparison with $^{nat}UO_2$ fuel.

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