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ASSESSMENT OF THE DELAYED NEUTRON FRACTION VARIATION DURING THE BURNUP OF A PWR LOADED WITH REPROCESSED FUEL SPIKED WITH THORIUM

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ABSTRACT

Delayed neutrons behaviour were evaluated using reprocessed fuel spiked with thorium in a PWR fuel assembly. Three standard UO_2 fuel assemblies with different enrichments based on the Angra-2 FSAR were modelled using the transport codes NEWT and validated with KENO-VI, both in SCALE6.0. The standard UO_2 fuel in each assembly was replaced by the reprocessed fuel with different fissile concentration. The multiplication factor desired should be the same as Angra 2 at Begin Of Life and Hot Zero Power. Therefore, the fissile material was set to reach the same multiplication factor. After that, the delayed fission neutron fraction of each assembly with UO_2 and $(Th\text{-}TRU)O_2$ were compared using the NEWT code. The evolution of the k_{inf} and the delayed neutron fraction (DNF) during burnup using the TRU fuel was evaluated and compared with the standard UO_2. The results show that the DNF of the assemblies using reprocessed fuel is smaller than the standard assemblies for all of them, which is due to the ^{239}Pu presence and the ^{233}U production, which contribute to the low values obtained for delayed fission neutron fraction. These lower values of DNF suggest that an assembly fuelled with TRU fuel is harder to control.

1. INTRODUCTION

Since the early days of nuclear development, there had been studies on the use of thorium as fuel in nuclear systems. The thorium fuel cycle is considered a safe and viable alternative of uranium, due to the reduction in the yield of transuranic actinides, plutonium in particular, and the production of the fissile isotope ^{233}U from the fertile isotope ^{232}Th, that composes 98.98% of natural thorium. In thermal reactors, it also has a negative temperature coefficient that improves the reactor safety. The production of fission products from ^{233}U is about 25% less than for the ^{235}U or ^{239}Pu, which lowers the reactor poisoning [1]. Natural thorium can replace depleted uranium that is added to reprocessed fuels and can be used during all the nuclear fuel cycle.

Combining the advantages of the thorium fuel cycle with the need to reduce nuclear waste, studies using thorium with reprocessed spent nuclear fuel have been carried out. The Department of Nuclear Engineering at UFMG, since the 90's, has been developing studies of reprocessing techniques with the addition of thorium [2, 3, 4, 5], and more recently included core criticality and burnup core calculations [6, 7]. Experimental studies were also initiated in critical reactors at Institute of Energy and Nuclear Research (IPEN) [8].

In this work, the multiplication factor and the behaviour of the delayed fission neutron fraction (DNF) during burnup were analysed and compared using UREX+ reprocessed nuclear fuel spiked with thorium (TRU) loaded in a PWR (Pressurized Water Reactor) with a standard UO_2 fuel. The DNF plays a vital role in the reactor control and safety.

Firstly, the CSAS6 sequence including the 3D Monte Carlo neutron transport KENO-VI, was used only as a reference case for the criticality calculations and model validation. The TRITON module was used for the depletion calculations. It prepares problem-dependent cross-sections linking codes for deterministic neutron transport (NEWT code), depletion and decay evaluation (ORIGEN-S) calculations. NEWT is a 2D deterministic neutron transport code and was used in the assessment of the reactor assemblies. It calculates the multiplication factor of the system and the delayed neutron fractions that were the parameters of interest in this work. It also has been used to generate cross-sections for future 3-D neutronics thermal-hydraulics coupling for nuclear reactor safety analysis. They are all part of the SCALE6.0 code package [9].

2. METHODOLOGY

The fuel assembly modelled is based on the data available from the FSAR (Final Safety Analysis Report) of Angra-2 reactor (Eletrobras Termonuclear S.A., 2013). **Tab. 20** presents the main parameters of the fuel assemblies modelled. Tab. 2 shows the six fuel assemblies types described in Angra-2 FSAR [10]. To minimize the uncertainties, only the assemblies without burnable poison are analysed ELE01, ELE02, and ELE04.

Tab. 20. Parameters of the fuel assemblies

Parameter	Value
Assembly type	16x16
Assembly pitch	23.11 cm
Number of rods per assembly	236
Number of guide tubes per assembly	20
Rod diameter	1.43 cm
Active length	391.6 cm
Cladding and guide tube material	Zircaloy 4
Fuel temperature	873 K
Cladding temperature	618 K
Moderator temperature	583 K

Tab. 21. Assembly types in the Angra-2 reactor core.

UO_2 enrichment	Number of IFBA rods	Identifier
1.9%	**0**	**ELE01**
2.5%	**0**	**ELE02**
	12	ELE03
3.2%	**0**	**ELE04**
	8	ELE05
	12	ELE06

Fig. 19 shows a fuel assembly and the mesh used by KENO-VI and NEWT. This configuration was used by KENO-VI for the model validation. All of the analysed assemblies have the same geometry.

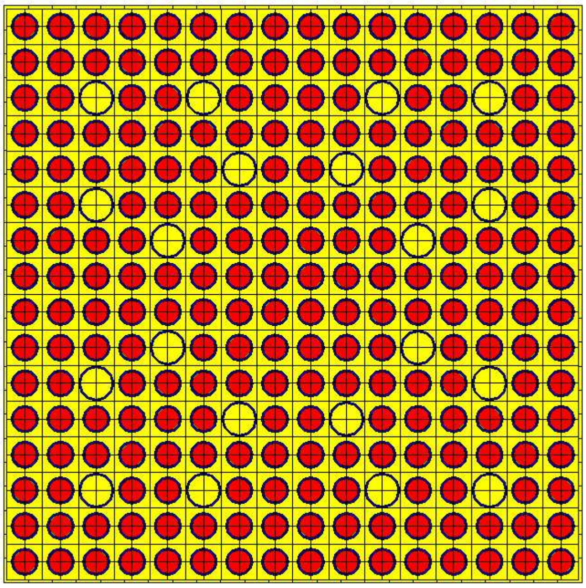

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Fig. 19. Fuel assembly design generated by KENO-VI and NEWT.

2.1. Criticality calculations

To validate the results from NEWT code, the assemblies loaded with standard UO_2 were compared with the results obtained with KENO-VI. The latter has been used in previous criticality analysis of reactors and assemblies fuelled with the studied TRU fuel. After validating the NEWT model, this code will be used for all subsequent neutron transport calculations using the TRU fuel.

The next step was to evaluate the insertion of $(Th\text{-}TRU)O_2$ in the assemblies. The fuel was obtained reprocessing spent fuel from a typical PWR fuel using the UREX+ technique and then spiked with thorium [11]. The fissile concentration was tested until reach the same k_{inf} value as the standard fuel in all three assemblies. Then, the DNF was also analysed for both fuels, $(Th\text{-}TRU)O_2$ and standard UO_2, at BOL.

2.2. Burnup calculations

The calculations from the fuel assemblies depletion were performed by the module TRITON using the ORIGEN-S code for the burnup and NEWT for the neutron transport. The k_{inf} values and the delayed neutrons fraction were analysed along the irradiation time. These calculations were performed at a constant power of 38 MW/MTHM during 868.5 days and without refuelling activities.

3. RESULTS

3.1. Results of criticality calculations

Tab. 3 shows the k_{inf} calculated by KENO-VI (stochastic method) and NEWT (deterministic method) using the standard UO_2 fuel. The values obtained show close similarity between both codes. The small difference can be explained by the different methods used for each one. As the fissile material build up increases, the difference becomes larger, suggesting a correlation that can be further studied.

Tab. 22. k_{inf} calculated by KENO-VI and NEWT for UO_2

	KENO-VI	NEWT	Difference
ELE01	1.22773 ± 0.00057	1.22512	-0.21%
ELE02	1.30480 ± 0.00058	1.30060	-0.32%
ELE04	1.36360 ± 0.00060	1.35912	-0.33%

These values were close enough to be considered satisfactory for the purpose of this work, which analyses the behaviour of the assembly when the UO_2 is replaced by the TRU fuel.

In the next step, various calculations varying the content of the TRU fuel were executed. The first step is to obtain a k_{inf} value similar to the standard fuel for all the fuel assemblies analysed. Tab. 23 shows the fissile concentration and composition of the TRU fuel used in each assembly type. Tab. 24 presents the k_{inf} values for the different assemblies.

Tab. 23. Isotopic composition of the TRU fuel used for each assembly

Isotope	Weight percentage (%)			Isotope	Weight percentage (%)		
	ELE01 TRU 14%	ELE02 TRU 20.5%	ELE04 TRU 25%		ELE01 TRU 14%	ELE02 TRU 20.5%	ELE04 TRU 25%
Th-232	6.82E-01	5.91E-01	5.28E-01	Pu-246	7.58E-24	1.11E-23	1.35E-23
Np-235	3.20E-12	4.68E-12	5.71E-12	Am-241	6.95E-03	1.02E-02	1.24E-02
Np-236	7.44E-09	1.09E-08	1.33E-08	Am-242m	1.86E-05	2.73E-05	3.33E-05
Np-237	6.49E-03	9.51E-03	1.16E-02	Am-242	2.23E-10	3.26E-10	3.98E-10
Np-238	2.53E-12	3.71E-12	4.53E-12	Am-243	2.62E-03	3.83E-03	4.68E-03
Np-239	1.63E-09	2.38E-09	2.91E-09	Cm-241	1.15E-26	1.68E-26	2.05E-26
Pu-236	8.78E-09	1.29E-08	1.57E-08	Cm-242	1.51E-07	2.21E-07	2.70E-07

Pu-237	4.03E-21	5.90E-21	7.19E-21	Cm-243	8.01E-06	1.17E-05	1.43E-05
Pu-238	3.43E-03	5.03E-03	6.13E-03	Cm-244	6.03E-04	8.83E-04	1.08E-03
Pu-239	1.00E-01	1.47E-01	1.79E-01	Cm-245	3.18E-05	4.66E-05	5.68E-05
Pu-240	4.20E-02	6.16E-02	7.51E-02	Cm-246	4.09E-06	6.00E-06	7.31E-06
Pu-241	2.28E-02	3.34E-02	4.08E-02	Cm-247	4.50E-08	6.60E-08	8.05E-08
Pu-242	1.20E-02	1.75E-02	2.14E-02	Cm-248	2.63E-09	3.85E-09	4.70E-09
Pu-243	2.02E-18	2.96E-18	3.61E-18	Cm-250	1.43E-17	2.10E-17	2.56E-17
Pu-244	7.99E-07	1.17E-06	1.43E-06	O-16	1.20E-01	1.20E-01	1.20E-01

Tab. 24. k_{inf} values for each assembly type and fuel

	NEWT		
Fuel type	UO_2	TRU	Difference
ELE01	1.22512	1.22603	0.07%
ELE02	1.30060	1.30450	0.30%
ELE04	1.35912	1.35619	-0.22%

The delayed neutron fractions at BOL for both fuel materials are presented in Tab. 25. **Fig. 20** shows a comparison of both fuel for the ELE01 assembly and the decay constant for each group is shown in Tab. 26.

Tab. 25. β calculated defining six effective groups of delayed neutrons for the different fuel compositions

Group	ELE01		ELE02		ELE04	
	UO2	TRU	UO2	TRU	UO2	TRU
1	2.22132E-04	7.02032E-05	2.22474E-04	6.67111E-05	2.22848E-04	6.48902E-05
2	1.50321E-03	5.93028E-04	1.49904E-03	5.52468E-04	1.49540E-03	5.32692E-04
3	1.37970E-03	4.50765E-04	1.37350E-03	4.20678E-04	1.36835E-03	4.05686E-04
4	2.84788E-03	8.10498E-04	2.83447E-03	7.42117E-04	2.82442E-03	7.06880E-04
5	9.34094E-04	2.77607E-04	9.24720E-04	2.52163E-04	9.18126E-04	2.38992E-04
6	3.25877E-04	8.10102E-05	3.20964E-04	7.45471E-05	3.16728E-04	7.11956E-05
TOTAL	7.21289E-03	2.28311E-03	7.17517E-03	2.10868E-03	7.14587E-03	2.02034E-03

Tab. 26. Decay constant λ of each group

Group	ELE01		ELE02		ELE04	
	UO2	TRU	UO2	TRU	UO2	TRU
1	1.24638E-02	1.05744E-02	1.02055E-02	1.24653E-02	1.00429E-02	1.24681E-02
2	3.06674E-02	2.51678E-02	2.43516E-02	3.06791E-02	2.39944E-02	3.06942E-02
3	1.12803E-01	1.05645E-01	1.02712E-01	1.12764E-01	1.01468E-01	1.12757E-01
4	3.04794E-01	2.69834E-01	2.60711E-01	3.04740E-01	2.56706E-01	3.04756E-01
5	1.17208E+00	9.75122E-01	9.52471E-01	1.17486E+00	9.43279E-01	1.17833E+00
6	3.12252E+00	2.42353E+00	2.38531E+00	3.13131E+00	2.37200E+00	3.14252E+00

Fig. 20 shows the DNF of each group for the assembly ELE01.

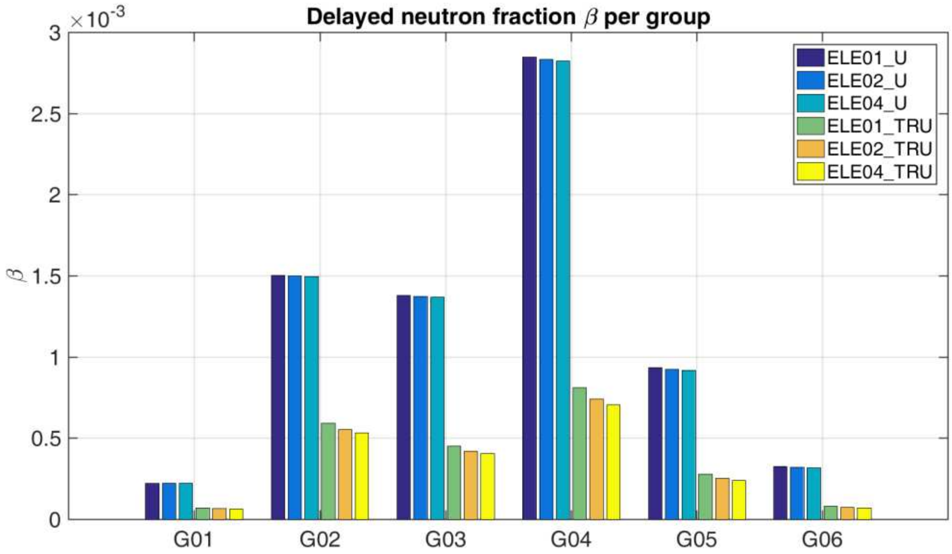

Fig. 20. β calculated in groups using different fuel.

For every group and each fuel composition, the DNF is considerably smaller using the TRU fuel, suggesting that a reactor fuelled with this type of fuel will be more difficult to control than the standard UO_2 fuel. It can be explained by the strong influence of ^{239}Pu in the TRU fuel. The total β of this isotope is β = 0.0022 in the thermal range, which is close to the obtained in all assemblies.

The delayed neutrons are responsible for the reactor response time and during a transient, it is desired that this parameter is as high as possible, especially when positive reactivity is inserted. For small reactivity insertions, the reactor period is mainly determined by the average neutron lifetime, including delayed neutrons, while these neutrons can be neglected for large positive insertions. Along with these effects, each precursor decay constant is also significantly smaller when using the TRU fuel, which suggests a strong shift of the reactor period and response time.

3.2. Results of burnup calculations

On the one hand, Fig. 21, Fig. 22 **and** Fig. 23 **show the evolution of the k_{inf} for all assemblies using both fuels during burnup. The figures display a k_{inf} decrease more slowly when using the TRU fuel. This is expected because the production of ^{233}U compensates the burnup of fissile and fissionable isotopes in the reactor using a thorium based fuel. On the other hand,**

Fig. 24, **Fig. 25** and **Fig. 26**, show the DNF variation during burnup for the TRU fuel value. For the TRU fuel, there is a small increase on DNF value, while the DNF from the UO_2 fuel assemblies decreases more steeply but maintaining its value above the TRU fuel. This behaviour can be explained by the production of ^{233}U, which also has a low value of $\beta = 0.0026$, maintaining the DNF value stable.

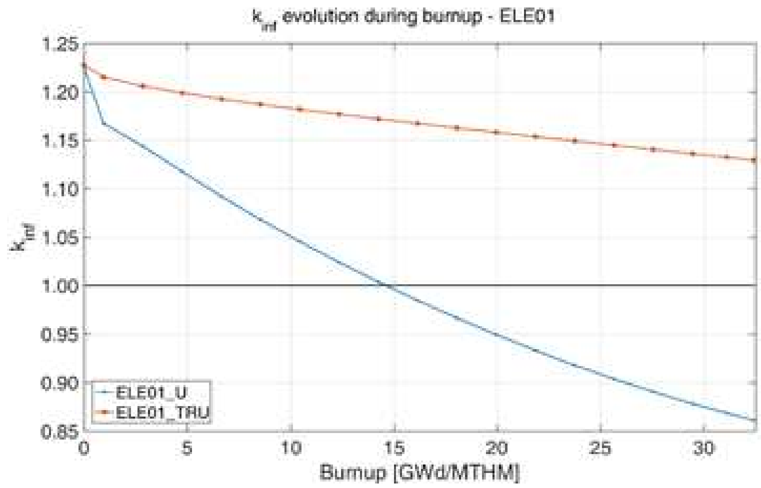

Fig. 21. k_{inf} during burnup – ELE01

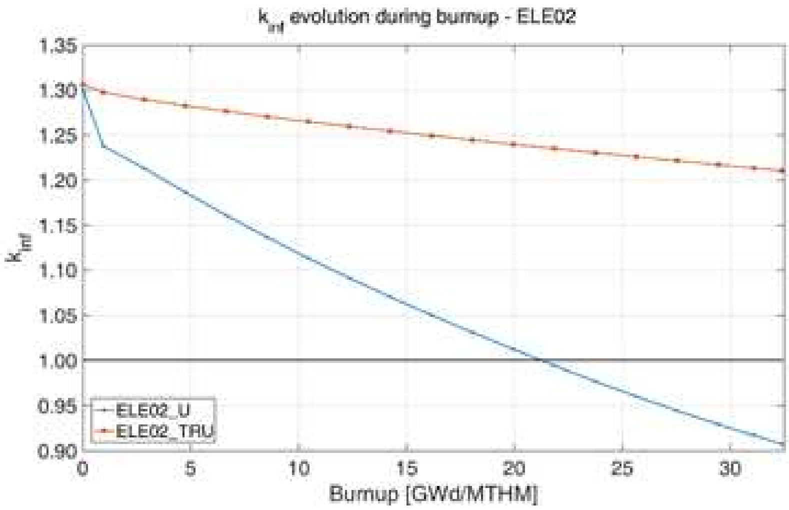

Fig. 22. k_{inf} during burnup – ELE02

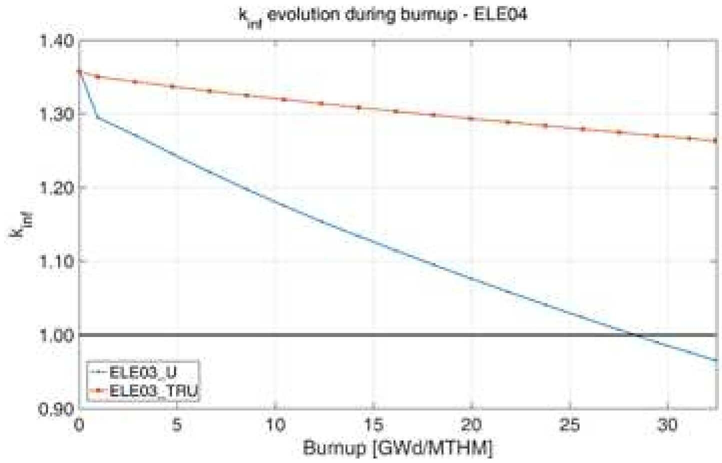

Fig. 23. k_{inf} during burnup – ELE04

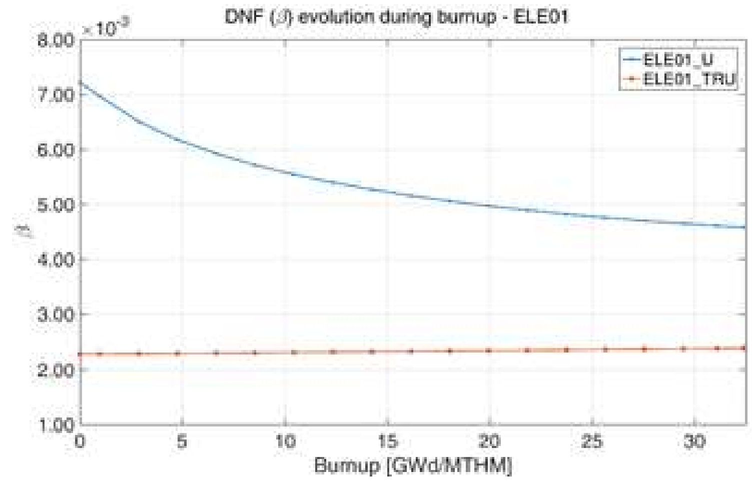

Fig. 24. DNF during burnup – ELE01

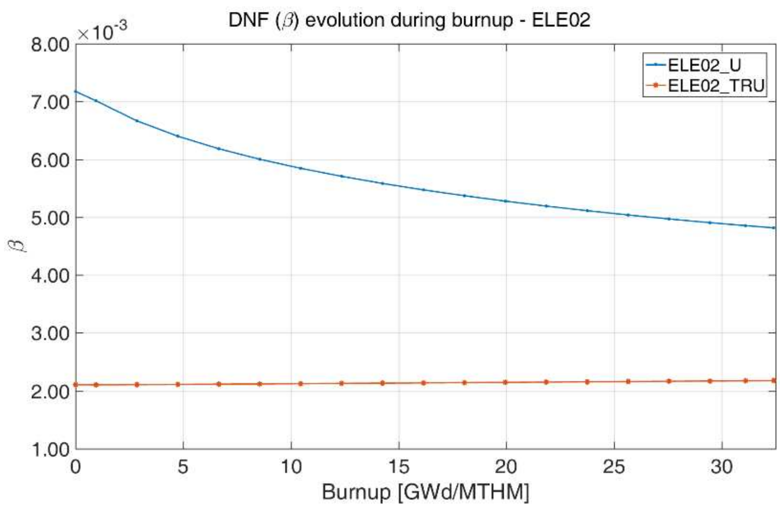

Fig. 25. DNF during burnup – ELE02

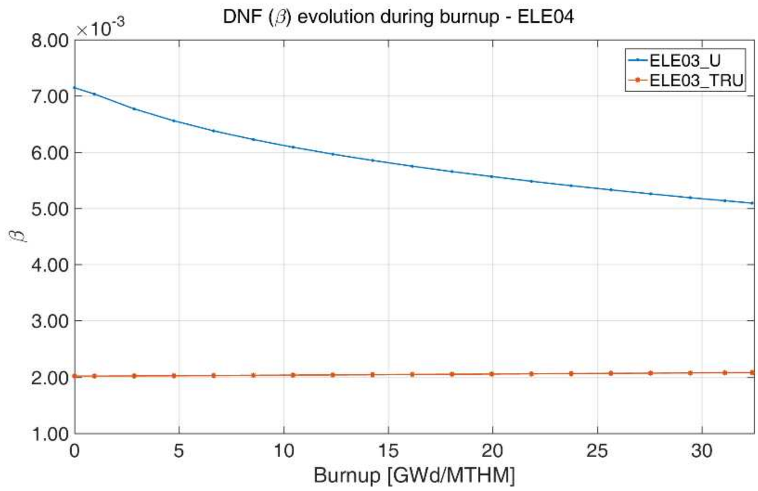

Fig. 26. DNF during burnup – ELE04

4. CONCLUSION

The NEWT model was validated by comparing the standard fuel with the KENO-VI code at BOL. The k_{inf} values have a maximum difference less than 0.35% for all cases. This difference is in the error margin enabling a suitable analysis from the behaviour of the assemblies when the UO_2 is replaced by the TRU fuel.

The composition of the TRU fuel was obtained at BOL by analysing the k_{inf} and the DNF. The assemblies fuelled with TRU and UO_2 were evaluated and compared. The results show that the DNF of the reprocessed fuel is considerably smaller than the UO_2. This is mostly due to a large amount of ^{239}Pu in the TRU fuel, that has a low value of total delayed neutrons, $\beta_{239pu} = 0.0022$. In contrast, the presence of ^{235}U in the UO_2 fuel increases the total DNF of the system.

During burnup, the k_{inf} decreases more slowly when using TRU fuel in comparison to the UO_2. According to the reference, it suggests that the TRU fuel allows the burnup extension. The slow decrease of the fuel assembly's multiplication factor using the TRU fuel can be explained by the production of the ^{233}U fissile isotope presents in the ^{232}Th reaction chain. So, the burnup of the ^{239}Pu is compensated by the production of ^{233}U.

The DNF calculated using both fuels shows a strong disagreement between them. When using the TRU fuel, it is believed to be more difficult to have a reactivity control of the assembly. However, while the DNF of the TRU fuel is significantly smaller than the UO_2 fuel, the decay constant is also smaller. Therefore, further studies are necessary to precise the influence of both parameters when using different fuels. A full core reactor assessment using a combination of both fuels might provide additional information about these safety parameters.

For future works, NEWT will be used to generate and expand a homogenized cross-section library for neutronic/thermal-hydraulics codes like PARCS/RELAP that will allow the transients safety analysis. An analysis using both fuels, UO_2 and TRU, is encouraged to simulate the behaviour of the reactor when using a combination of these fuels.

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