A neutronic evaluation of reprocess fuel and depletion study of VHTR using WIMSD5 and MCNPX codes

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Among the projects of IV generation reactors available nowadays, the (High Temperature Reactors) HTR, are highlighted due to their desirable characteristics and they have been studied by the ***Instituto Nacional de Ciências e Tecnologia de Reatores Inovadores/CNPq****(Brazil). For this work, it evaluated the neutronic behavior and fuel composition during the burnup using the codes (Winfrith Improved Multi-Group Scheme) WIMSD5 and the MCNPX2.6, inserting different percentages of reprocessed fuel in the core. The fuel type "C" coming from Angra-I nuclear power plant, in Brazil, enriched with 3.1% was burnt by three typical cycles and then reprocessed. It recovered (Pu) and minor actinides (MA)being neptunium (Np), americium (Am), curium (Cm), and processed six different fuels varying percentage insertion of reprocessed fuel and enrichment uranium. It analyzed the multiplication factor, temperatures reactivity coefficients, and the composition during the burnup. The results showed, in the analyzed conditions, only one of these fuels is possible to be used. To compare, a reference fuel using 15% enrichment (235U) was too evaluated*

INTRODUCTION

Nowadays the strategies for the management of spent fuel in many countries consist in direct storage at nuclear power plants for future disposal in geological repositories [1,2]. However, if no decisions on management strategies are taken as soon as possible, large amounts of spent fuel will soon raise on storage. An alternative is the reprocessing and recycling it. Some reprocessing techniques have been proposed, and for Pu and Minor Actinides, techniques such as UREX+ or GANEX have been evaluated [3,4]. Assuming reprocessing, the recovered TRU holds potential source of energy in its fissile and fertile nuclides. In this work, it supposes that there are technologies for reprocessing and fabrication (TRU+Uranium) into usable fuel forms to utilization in nuclear power reactors. Based on this assumption, it is interesting to evaluate the neutronic behavior of a VHTR core with different percentage of insertion of reprocessed fuels [5]. It will evaluate seven cores: six of them are based on reprocessed fuels and one is used as reference (15% of 235U enrichment). Neutronic parameters and depletion behavior will be evaluated during the burnup. To this study a cluster-homogenized of the reactor was simulated using WIMSD5 code [6,7]. To compare the results, the same model was evaluated using MCNPX2.6 [8,9].

METODOLOGY

The VHTR reactor uses helium gas as a coolant and has a burnup of 90,000.0 MWd/THM [10]. System operation at a high temperature, however, puts additional constraints on the design of the VHTR reactor, with the greatest implications on the optimal material selection. For this reason, TRISO-coated fuel particle is the main nuclear fuel type considered to be used in the VHTR (Figure 1)[11,12]. In a TRISO-coated fuel particle, fissile fuel is enclosed within layers of temperature resistant materials including SiC and pyrolithic graphite. This structure is then formed either into pebbles ready to be burned or in fuel compacted to be meshed into graphite prismatic blocks. Each fuel pin has six layers containing in this order: fuel, porous carbon, pyrocarbon, silicon carbide, pyrocarbon and graphite [13].

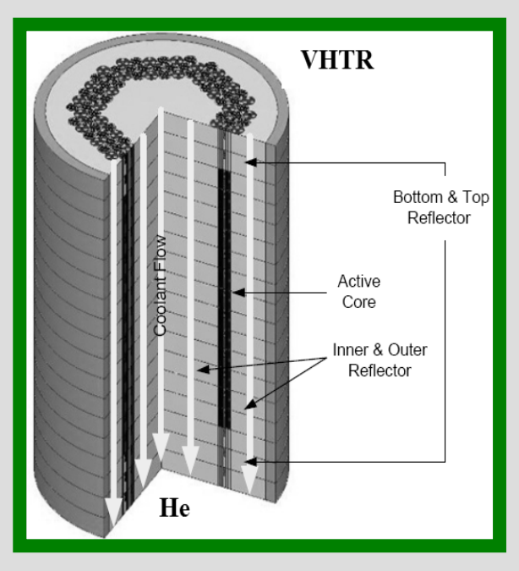

Figure 1- Reactor core geometry

The VHTR design used in this study was based on the *MHTGR nuclear physics benchmarks* [14,15,16]. A graphite cylinder containing 36 fuel pins distributed in three regions (or rings) models the core configurations: inner ring, central ring and outer ring. A central reflector made of graphite, an external reflector, also made of graphite and a medium region that receive the fuel and coolant channels, constitutes these models. Keeping the same ratio VM/VF (moderator volume/ fuel volume) of active core reactor, it was possible to model it in a single representative fuel block, representing the different coating fuel layers. Figure 2 shows the simplified model evaluated using the WIMSD-5B and MCNPX 2.6.0 codes.

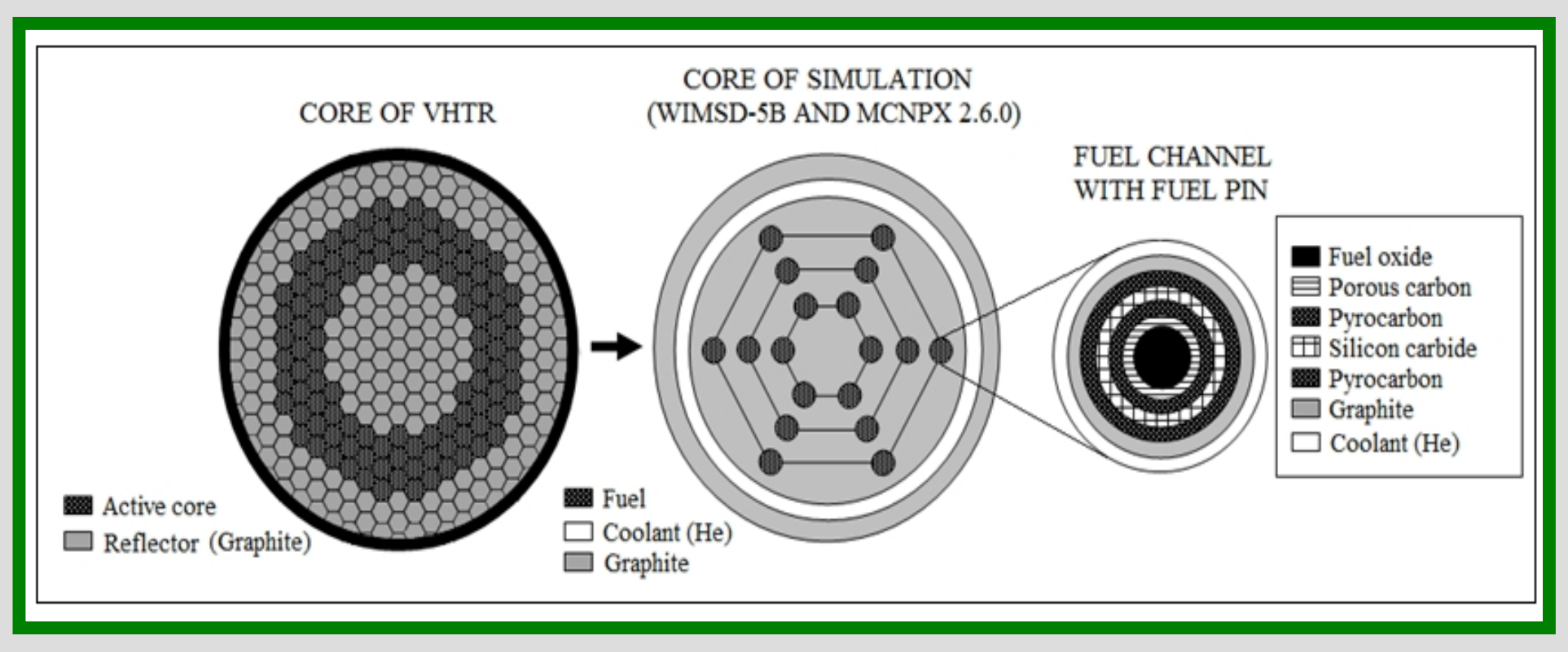

Figure 2- Reactor core geometry (type cluster)

The first VHTR core was loaded with a standard fuel with 15% of enrichment. The other six fuels are based on a mixture of uranium oxide together with reprocessed transuranic nuclides to achieve 15% of fissile material (235U, 239Pu and 241Pu). The Pu, Am, Np and Cm were recovered from the PWR spent fuel vectors obtained from the simulation of the fuel type "C" coming from Angra-I [17,18,19]. The Table 1 shows the isotopic compositions for studied fuels and Table 2 presents the percentage of 235U isotope and reprocessed fuel inserted on the matrix, to achieve 15% of fissile material. The fuels were evaluated during the burnup up to 100,000.0 MWd/THM, during 990 day with no reloads [20,21].

RESULTS

The first analysis is based on the fuel temperature coefficient (αTF) and the moderator temperature coefficient (αTM). The temperature coefficient is obtained by the following equation:

$$\alpha_T = \frac{\Delta\rho}{\Delta T} = \frac{\rho_2 - \rho_1}{\Delta T} \Rightarrow \alpha = \frac{k_2 - k_1}{k_2 \cdot k_1} \cdot \frac{1}{\Delta T}$$

where k2 is the effective multiplication factor due to the temperature variation (ΔT) and k1 is the effective multiplication factor when it operates at work temperature. ΔT for both coefficients, (αTF) and (αTM) was 50ºC. At Zero Power (300ºC) the αTF for all fuels was negative, however, only the fuels 1 and 2 presents αTM negative to burnup over 90,000.00 MWD/tU. At Full Power (900 oC) all fuels present negative values of αTF and αTM.

Table 1- Isotope composition on the fuel matrix for the major comparative samples

VHTR				
15% Fuel Enrichment				
Percentage of Minor Actinide Insertion				
Isotope	**Fuel Type**			
	1	**2**	**3**	**4**
^{238}Pu	0.00000	0.19774	0.22191	0.26773
^{237}Np	0.00000	0.37333	0.41897	0.50548
^{241}Am	0.00000	0.40018	0.44910	0.54183
^{243}Am	0.00000	0.15028	0.16865	0.20347
^{242}Cm	0.00000	0.00001	0.00001	0.00001
^{243}Cm	0.00000	0.00049	0.00055	0.00066
^{244}Cm	0.00000	0.03669	0.04118	0.04968
^{235}U	13.22005	6.14419	5.27920	3.63957
^{238}U	74.91361	70.65816	70.13796	69.15188
^{239}Pu	0.00000	5.77014	6.47551	7.81258
^{240}Pu	0.00000	2.41981	2.71562	3.27634
^{241}Pu	0.00000	1.30798	1.46787	1.77095
^{242}Pu	0.00000	0.68865	0.77284	0.93241
^{242}Am	0.00000	0.00107	0.00120	0.00145
^{16}O	11.86634	11.85128	11.84944	11.84595
Isotope	**Fuel Type**			
	5	**6**	**7**	
^{238}Pu	0.33077	0.35619	0.36568	
^{237}Np	0.62451	0.67250	0.69042	
^{241}Am	0.66942	0.72086	0.74007	
^{243}Am	0.25138	0.27070	0.27792	
^{242}Cm	0.00002	0.00002	0.00002	
^{243}Cm	0.00082	0.00088	0.00090	
^{244}Cm	0.06138	0.06610	0.06786	
^{235}U	1.38357	0.47406	0.13436	
^{238}U	67.79511	67.24812	67.04383	
^{239}Pu	9.65228	10.39396	10.67098	
^{240}Pu	4.04785	4.35889	4.47506	
^{241}Pu	2.18798	2.35610	2.41890	
^{242}Pu	1.15198	1.24050	1.27356	
^{242}Am	0.00179	0.00193	0.00198	
^{16}O	11.84114	11.83921	11.83848	

Table 2: Percentage of reprocessed fuel on the fuel on different enrichment of the ^{235}U isotope

Samples	^{235}U isotope on the fuel (%)	Percentage of total uranium on the fuel (%)	Percentage of reprocessed fuel on the fuel (%)
1	15	100	0
2	8	87.12	12.87
2a	7	85.55	14.44
2b	5	82.57	17.42
2c	2	78.47	21.52
3- U natural	0.07	76.81	23.18
3a – U depleted	0.02	76.19	23.80

The following graphics present the obtained results. To simplify, only the standard Fuel 1 (15% of enrichment and no MA insertion), and the Fuels 2 (minimum MA insertion) and 7 (maximum MA insertion) were represented in the illustrations. The behavior of the other Fuels (3, 4, 5 and 6) is intermediary between the curves of the Fuels 2 and 7. The Figures 3 and 4 present the αTF and αTM behavior during the burnup at Zero Power obtained using WIMSD5. The Figures 5 and 6 present the coefficients to full power. Figures 7 and 8 present the multiplication factor to zero and full power respectively. Based on the results obtained, analyzing all possibilities of MA insertion among the reprocessed fuels, only the Fuel 2 presents condition to be used in the VHTR core. In a future work, a possibility to improve the behavior of the fuel with a major MA insertion from the reactivity coefficient viewpoint, is to change the ratio VM/VF.

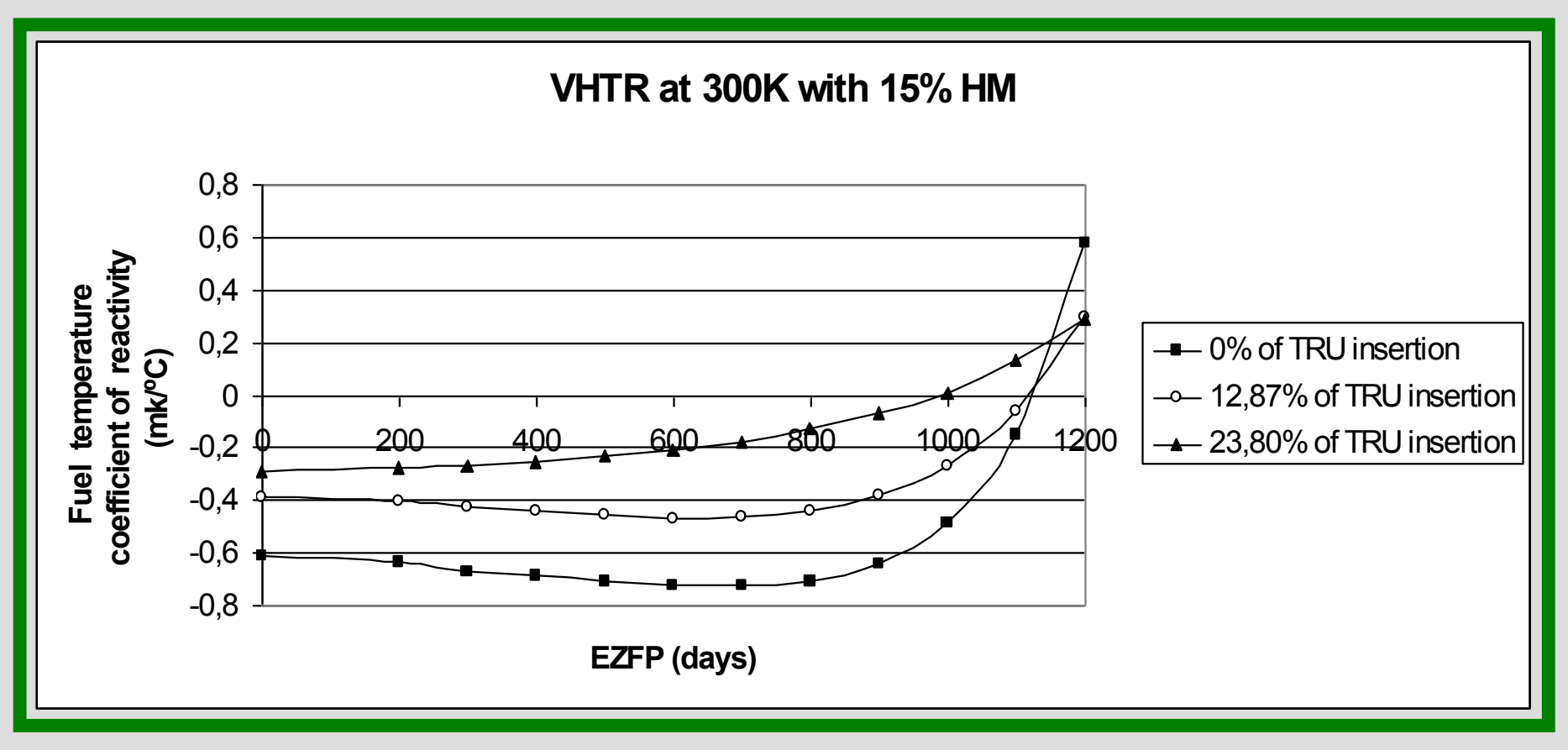

Figure 3– Fuel temperature reactivity coefficient on VHTR reactor at Zero Power (300K) for the three studied insertion of MA from the WIMS5D code

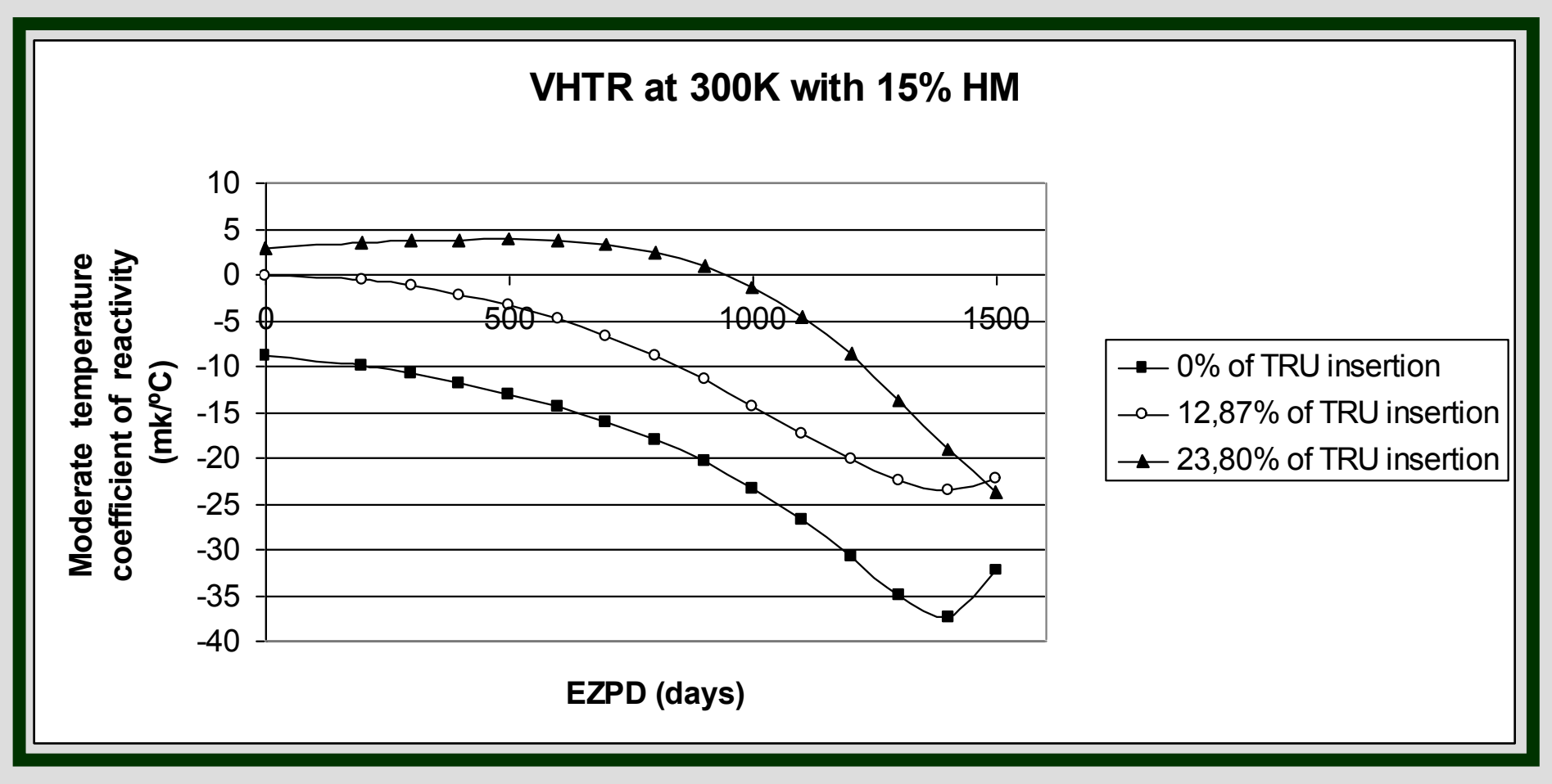

Figure 4– Moderate temperature reactivity coefficient on VHTR reactor at Zero Power (300K) for the three studied insertion of MA from the WIMS5D code

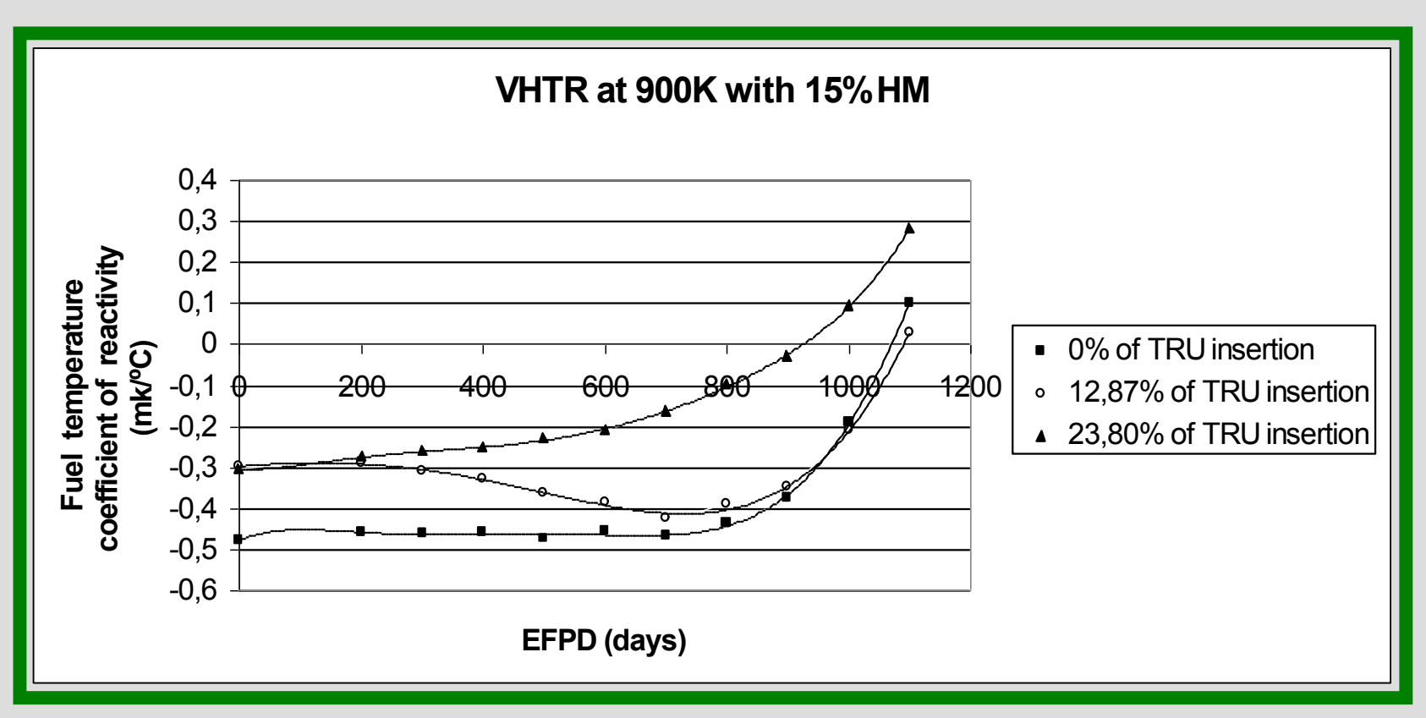

Figure 5– Fuel temperature reactivity coefficient on VHTR reactor at Full Power (900K) for the three studied insertion of MA from the WIMS5D code

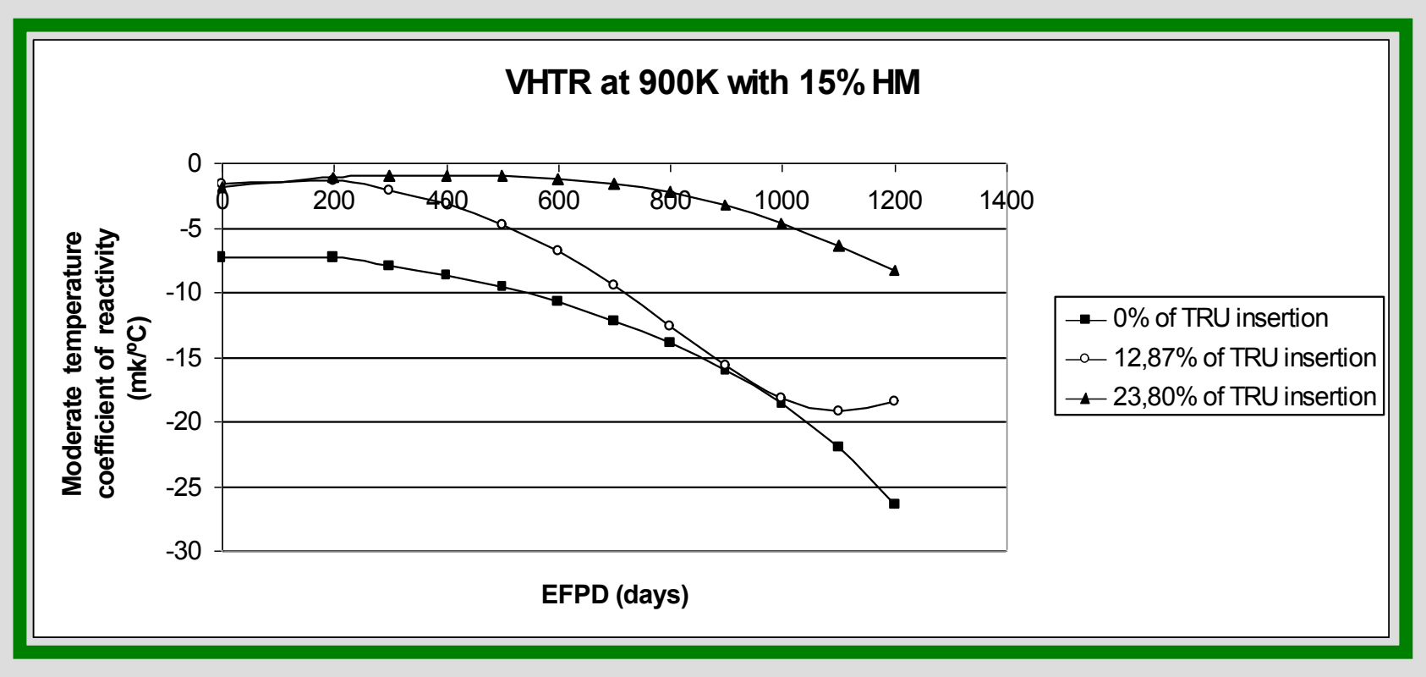

Figure 6– Moderate temperature reactivity coefficient on VHTR reactor at Full Power (900K) for the three studied insertion of MA from the WIMS5D code

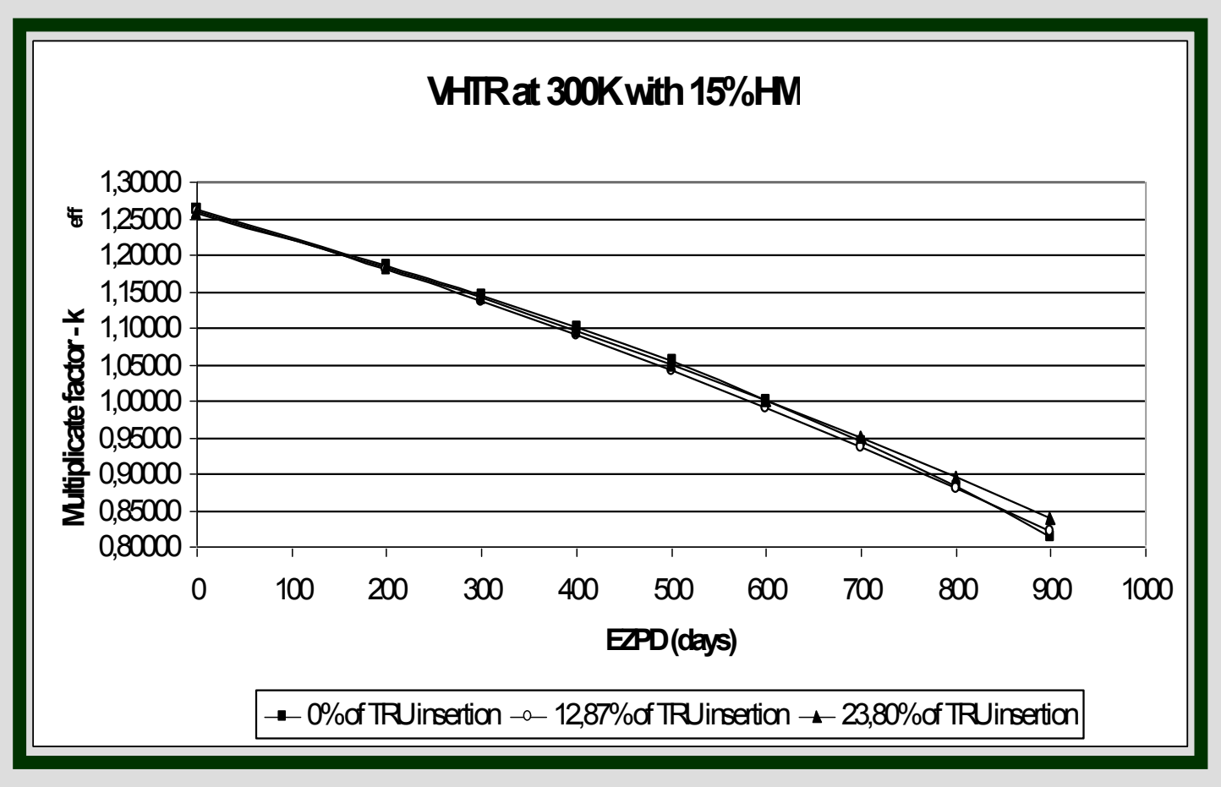

Figure 7 - Multiplication factor (keff) on the VHTR reactor at Zero Power (300K) for the three studied insertion of MA from the WIMS5D code

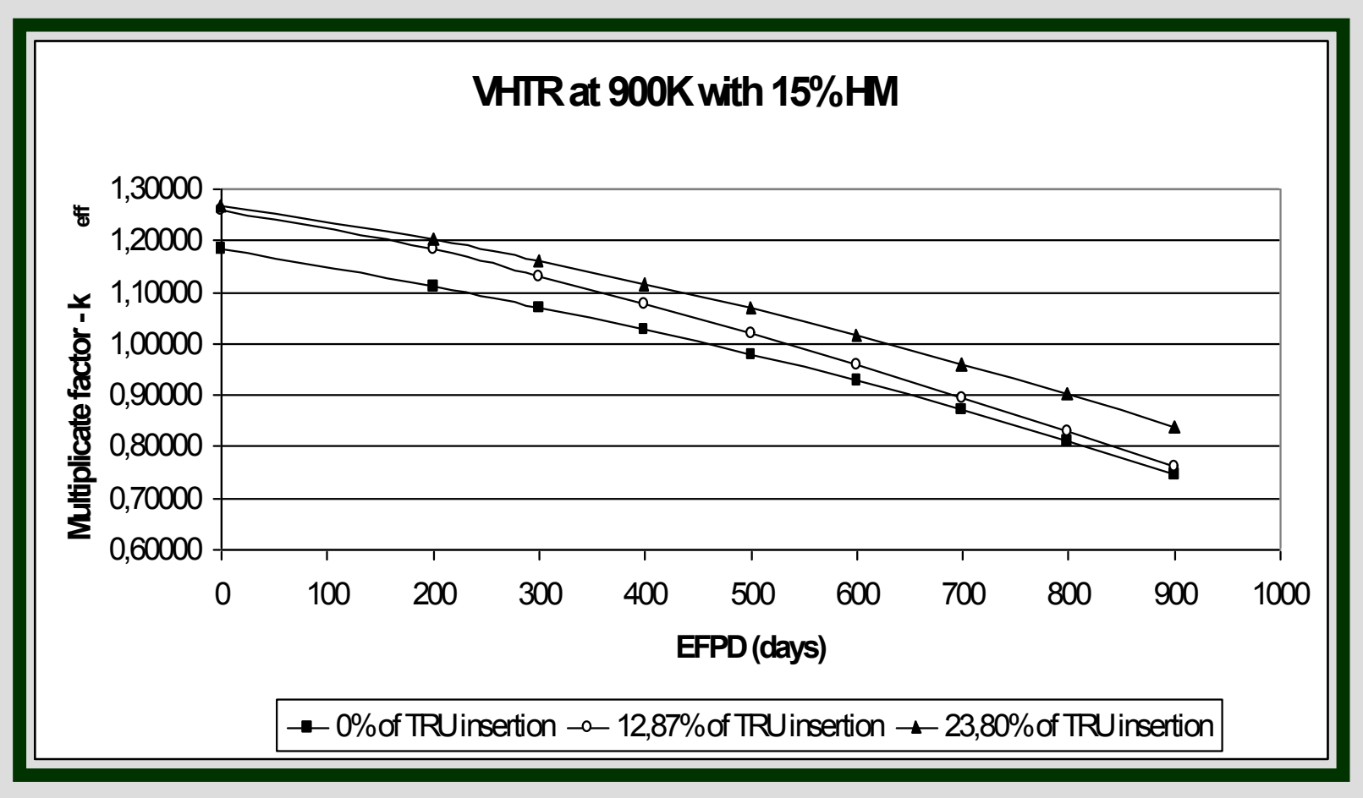

Figure 8- Multiplication factor (keff) on the VHTR reactor at Full Power (900K) for the three studied insertion of MA from the WIMS5D code

The multiplication factor was obtained using also the MCNPX2.6 code considering the same modeling (Figure 9). Only the Fuel 2 has been analyzed because it presented better behavior during the burnup as it was verified before. The result shows good agreement between the codes in the beginning of the burnup, but MCNPX2.6 presents a behavior overestimated relation to the WIMSD5 code. The difference is connected mainly to the different calculus methods employed in both codes and the treatment applied in the library during the burnup MCNPX/CINDER90 process as it has been verified in some works [22]. In addition, Figure 10 shows the 235U composition during the burnup to the fuel 1 (standard) and fuel 2 obtained from WIMDS5 and MCNPX2.6. The result presents very good agreement between the codes. To confirm such agreement, the composition of Pu, Am and Np during the burnup will be analyzed in a future work.[21]

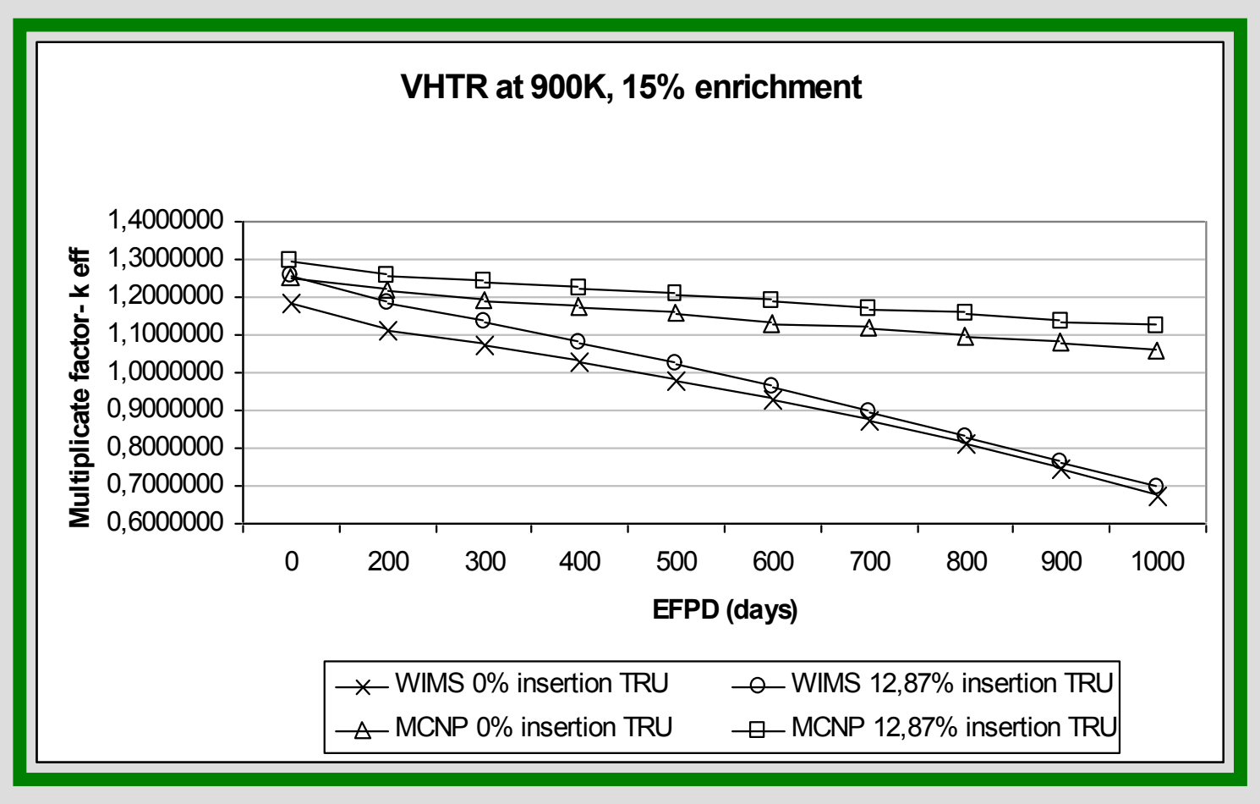

Figure 9 - Multiplication factor (keff) on the VHTR reactor at full Power (900K) for the two codes WIMS5D and MCNPX

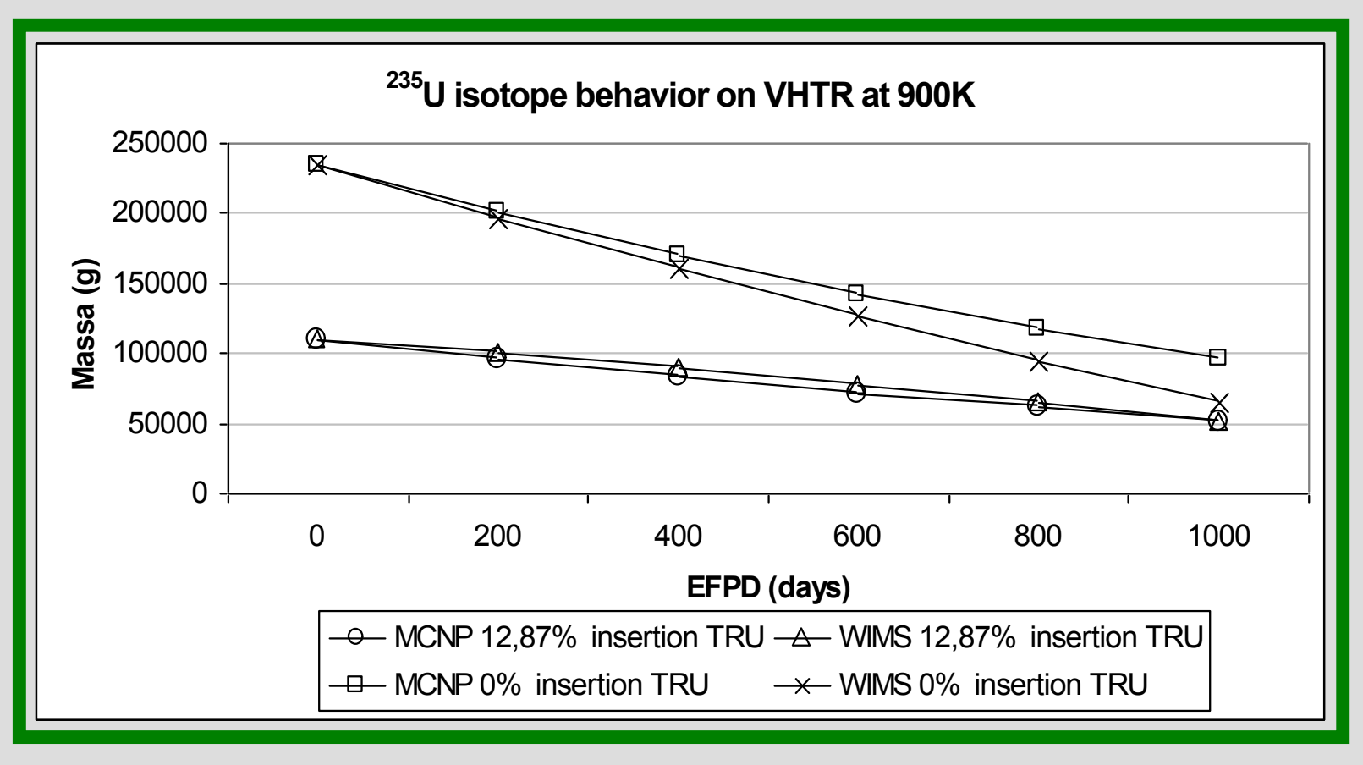

Figure 10- Evolution of the composition of the ^{235}U on the VHTR reactor for the WIMSD and MCNPX codes

CONCLUSIONS

In this work, it evaluated the neutronic behavior and fuel composition during its burnup in a VHTR reactor simulated with different percentages of reprocessed fuel in the core. The WIMSD5 code was used mainly to perform the analyses of multiplication factor and the temperature reactivity coefficients. Seven types of fuels were considered being a standard fuel (fuel 1) and fuels with TRU insertion varying from 12.87 up to 23.80 % (fuels 2, 3, 4, 5, 6 and 7). The analysis of the neutronic parameters demonstrates that the maximum TRU insertion allowed in the fuel is 12.87%. Such percentage assures a safe reactor behavior along the burnup. However, it is possible to increase the TRU percentage in the fuel changing carefully the Vm/Vf ratio as it will be performed in a future work. The decrease of 235U composition was evaluated during the burnup using the WIMSD5 code. To demonstrate the simulation capabilities of the WIMSD5, the MCNPX2.6 code was used to comparison. The results demonstrate very good agreement between the codes. A more detailed composition analysis including mainly Pu, Am and Np will be performed as a next step of this work using both codes.

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