NEUTRONIC EVALUATION OF TRANSURANICS IN A GFR MODEL USING MCNPX AND SCALE 6.0

Anderson A. P. Macedo, Victor F. Castro, Clarysson A. M. da Silva,
Carlos E. Velasquez and Claubia Pereira

Departamento de Engenharia Nuclear – Escola de Engenharia. Universidade Federal de Minas Gerais,
Av. Antônio Carlos, 6627, Pampulha, Belo Horizonte MG, CEP 31270-901, Brasil, Phone/Fax: 55-31-34096662

macedo@nuclear.ufmg.br; victorfariacastro@gmail.com; clarysson@nuclear.ufmg.br;
carlosvelcab@ufmg.br, claubia@nuclear.ufmg.br

ABSTRACT

In this study, a GFR core model with 100 MWt was evaluated using three different fuel compositions: a conventional (U, Pu)C and two reprocessed fuels reprocessed by UREX+ technique one spiked with depleted uranium, (U,TRU)C, and the other one reprocessed spiked with thorium, (Th,TRU)C. The reprocessed fuel came from a PWR standard fuel (33,000 MWd/T burned) with 3.1% of initial enrichment and left in the pool by 5 years. Some important nuclides were followed during burnup and k_{inf} was evaluated for 1400 days. The results also include analysis of the B_4C insertion and the temperature coefficient. The simulations were performed comparing results between MCNPX and SCALE 6.0 codes. The main goal is to validate the model and evaluate the possibility to use TRU spiked with Th in a GFR.

1. INTRODUCTION

The Nuclear Engineering Department at Universidade Federal de Minas Gerais DEN/UFMG has been studied IV generation reactors proposed by the GEN-IV International Forum [1]. One of the models studied was the GFR [5, 6]. The first stage was to validate the GFR proposed by [2] using the (U,Pu)C with the SCALE 6.0 code. After that, an innovative nuclear fuel based on UREX+ reprocessed technique [11] was used to test the efficiency and to improve the nuclear fuel cycle on this reactor. Its focus was on criticality safety analysis, burnup extension and non-proliferation [5,6]. In this studies, the results show that the most suitable percent of the fissile material for each nuclear fuel is 11.33% for (U,Pu)C, 13.06% for (U,TRU)C and 15.78% for (Th,TRU)C.

In this work, the goal is to compare the results obtained using SCALE 6.0 with those obtained using MCNPX 2.6.0 code. Three fuels were simulated, (U,Pu)C, (U,TRU)C, and (Th,TRU)C, and the k_{inf} during 1400 days has been analyzed to verify the extended burnup and the trasnuranics buildup. In addition, using only SCALE 6.0, the fuel temperature coefficient of reactivity and the insertion of B_4C were also analyzed. In this test, seven control rods initially filled with helium were filled with boron carbide absorber (B_4C), in the following proportion: 90% ^{10}B and 10% ^{11}B. The goal is to verify the core criticality every 10 cm of rods insertion. The GFR model used by the DEN/UFMG is a heterogeneous reactor that works at 100 MWt [6].

2. PROPOSED MODEL

The core dimensions and properties can be found in a previous work [3]. The fissile material used for each nuclear fuel remains as mentioned before, which can be verified also in a reference [3]. The criticality and burnup calculations were performed with the MCNPX2.6.0 and SCALE6.0 both of them use the ENDF-B-VII.0 library. The MCNPX uses this library as a continuous one. On the other hand, the SCALE6.0 code uses 238 groups collapsed from the ENDF-B-VII.0 continuous library (V7-238). The MCNPX-2.6.0 performs the burnup using the CINDER90 using 63 energy groups and the SCALE6.0 uses the ORIGEN-S, which performs the burnup using 3 energy group. Figure 1 shows an assembly scheme details and the fuel rod of the reactor. The operating conditions in both codes are identical, the burnup was performed during 1400 days, divided into 12 equal steps, with 4000 particles used in 500 cycles dismissing the first 30 cycle. The specific power for both, the assembly and core evaluation is 48 W/g.

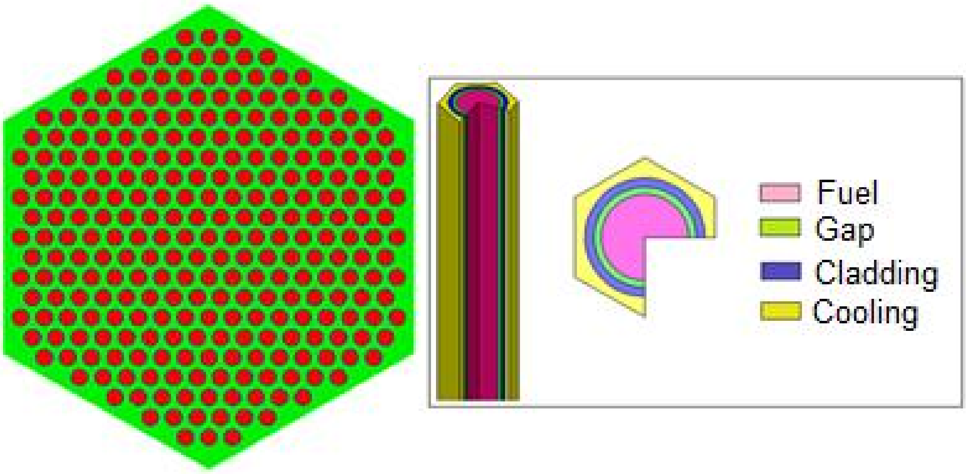

Figure 1: Heterogeneous assembly and details of a fuel cell.

2.1. The GFR core

Fig. 2 shows a scheme for the simplified core on SCALE 6.0. Table 1 presents the temperature for each component in the reactor used in the simulation. It is on the work from *WFG van Rooijen* [7], *Anthony M. Judd* [8] and *Peter Yarsky* [9].

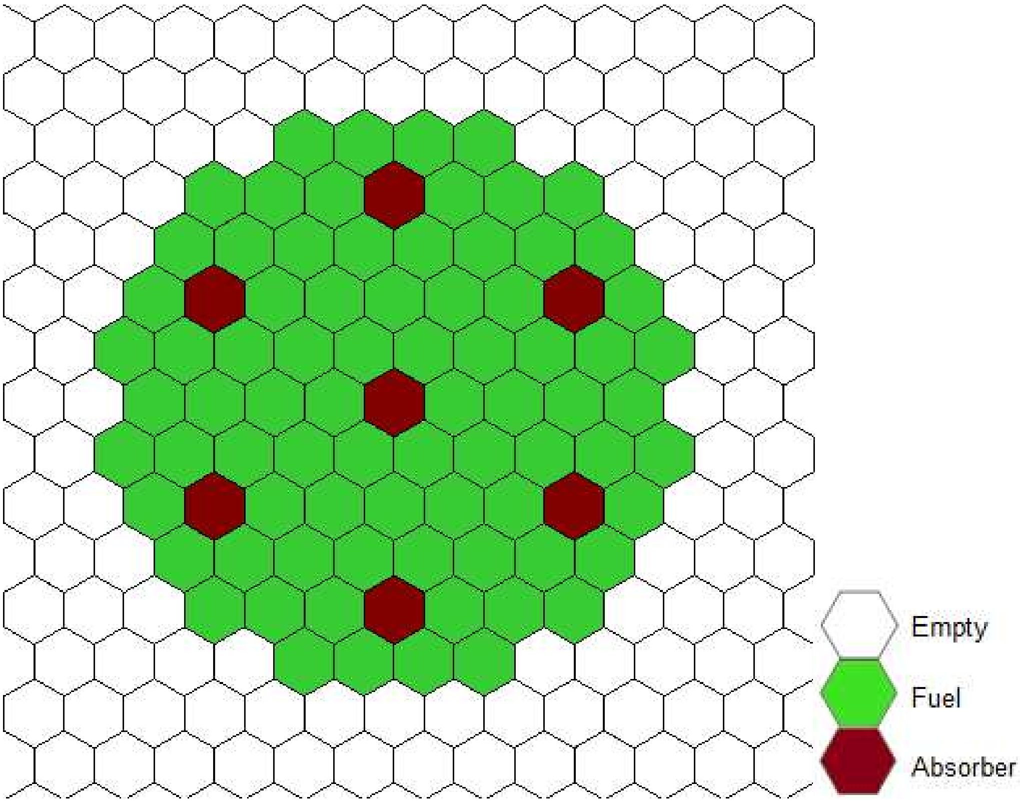

Figure 2. Simplified GFR core configuration.

Table 1: Components temperatures for the Core [7, 8, 9]

Components	Temperature (Kelvin)
Fuel	1000
Fuel cladding	800
Cladding to the fuel rods	600
Gap	1000
Cooled to the control rods	600
Cooled to the fuel assembly	600

3. RESULTS

Table 3 shows the values of k_0 at beginning of the cycle (BOC) and k_{12} represents the values at the end of the cycle (EOC).

Table 3. Initial and Final k_{inf} For Three Fuels in 1400 Days Burning

k_{inf}	FUEL / k_{inf}					
	MCNPX			SCALE 6.0		
	(U, PU)C	(U, TRU)C	(Th, TRU)C	(U, PU)C	(U, TRU)C	(Th, TRU)C
k_0	1.30270	1.30039	1.29860	1.30713	1.30499	1.30353
k_{12}	1.17824	1.18847	1.23279	1.17080	1.18142	1.22089

Figure 3 shows the k_{inf} evolution during burnup (1400 days). After 300 days of burnup, the nuclear fuel spiked with thorium undergoes a proportional decrease, indicating the ^{233}U build-up effects.

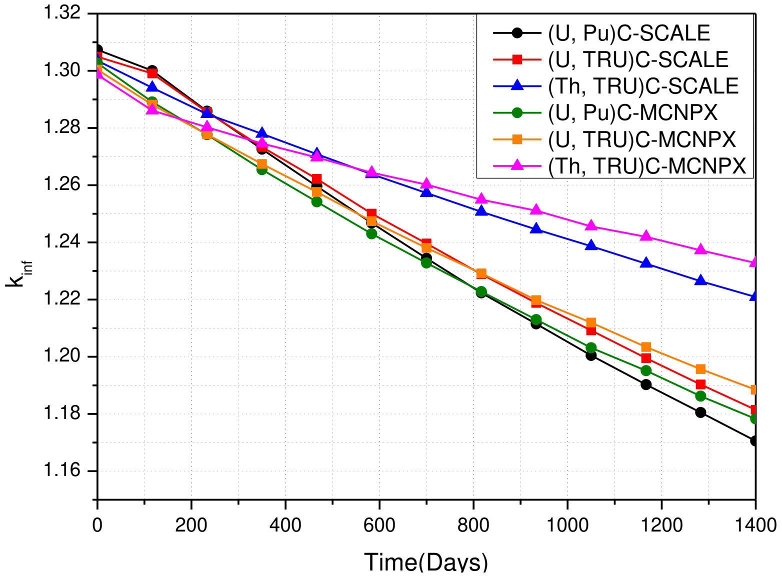

Figure 3: k_{inf} Comparison to the Three Fuels in 1400 Days Burning.

The small differences can be attributed to the data treatment between the codes, such as neutron transport and the energy groups used to calculate the burnup for each depletion code. The differences between the simulations in each code are calculated by the absolute difference between the k_{inf} values for each fuel presented in Equation (1). Figure 4 shows the differences in pcm for each nuclear fuel, where 100, 1300 and 1400 days presented a difference greater than 10^3 pcm. The differences at BOC are due to stabilization of the core after the initial reactivity insertion. On the other hand, the differences at EOC are due to the fissile material depletion and minor actinides production.

$$\Delta k_{inf}(pcm) = |k_{\mathrm{inf}\ |SCALE6.0} - k_{\mathrm{inf}\ |MCNPX}| x10^5 \qquad (1)$$

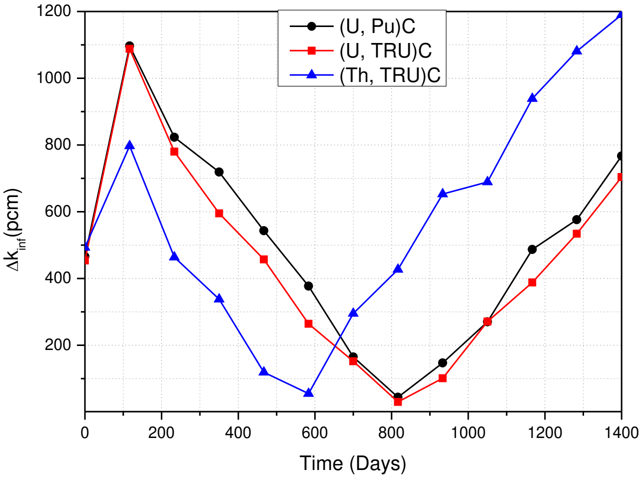

Figure 4: Absolute differences of the k_{inf} between SCALE 6.0 and MCNPX for each nuclear fuel

3.1.1 Nuclides Evolution

Figure 5 shows the depletion of the Pu isotopes, such as, ^{238}Pu, ^{239}Pu, ^{240}Pu and ^{241}Pu for both codes, which show good agreement between them. On the other hand, figure 6 shows a little difference between the calculation of some minor actinides such as ^{244}Cm and ^{241}Am. Figure 7 shows the buildup of ^{233}Th and ^{233}Pa, which are responsible for the ^{233}U production. The buildup of these nuclides show good agreement in the results between codes. The (Th,TRU)C was the fuel that most created fissile material, due to the build-up of ^{233}U. Its concentration reaches almost the maximum value exactly when the fuel has the highest reactivity by increasing its concentration after 150 days of burnup.

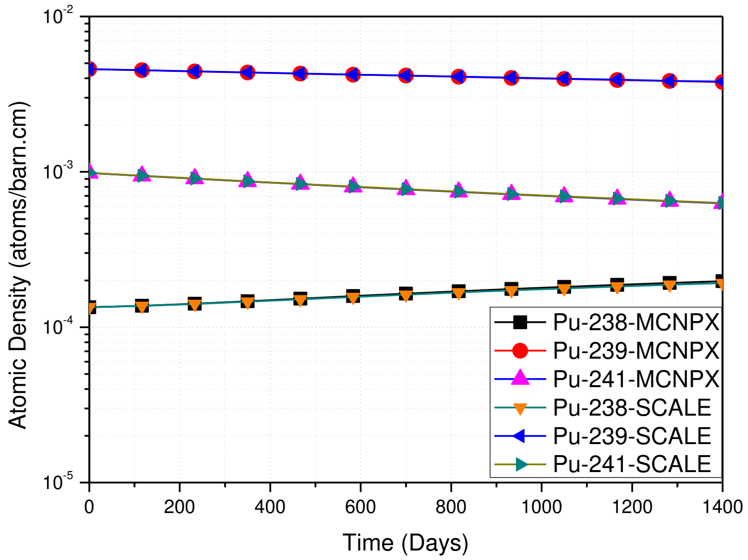

Figure 5: Depletion of Plutonium Isotopes during burnup.

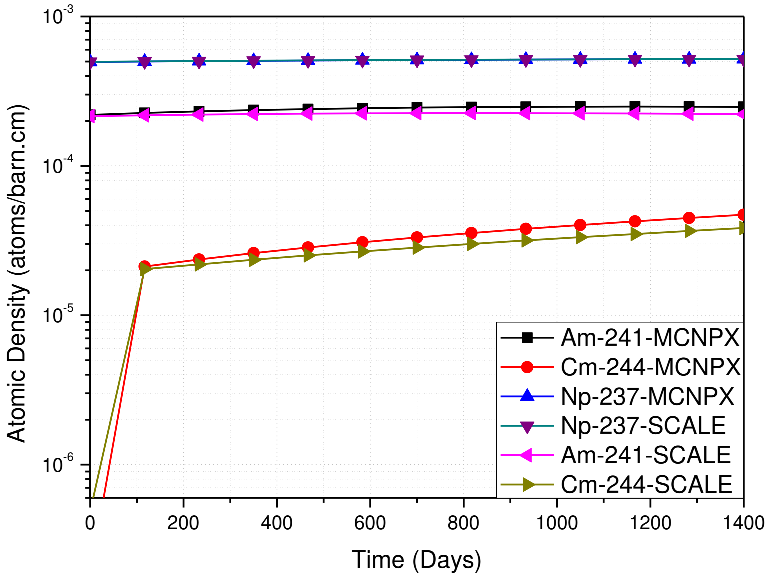

Figure 6: Depletion of Fissile Actinides during burnup

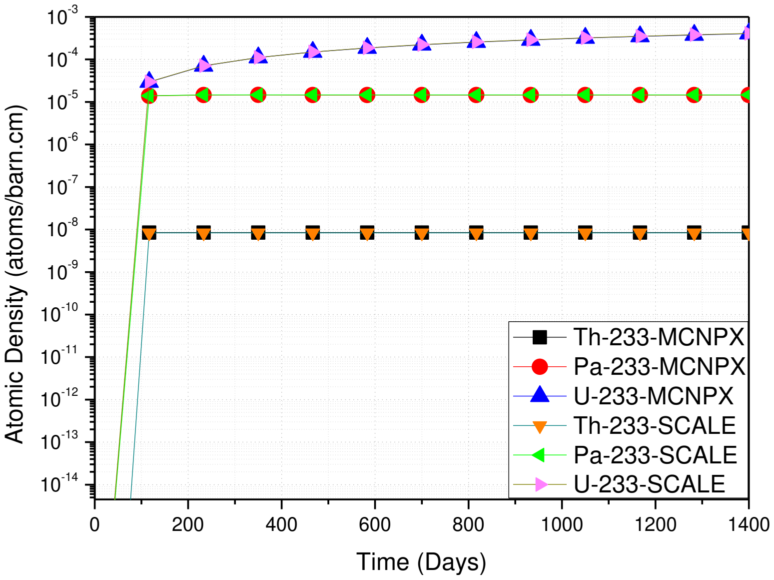

Figure 7: Buildup of ^{233}U, ^{233}Th and ^{233}Pa.

3.1.2 Fuel Temperature Coefficient of Reactivity

The fuel temperature coefficient of reactivity on SCALE 6.0 was calculated increasing 100 K the fuel temperature and calculating the k_{inf} for both temperatures during burnup. The equation 2 used to calculate the temperature coefficient of reactivity. Figure 8, shows the behavior of the temperature coefficient of reactivity for each of the three nuclear fuel evaluated, which shows that the temperature coefficient of reactivity remains negative during burnup.

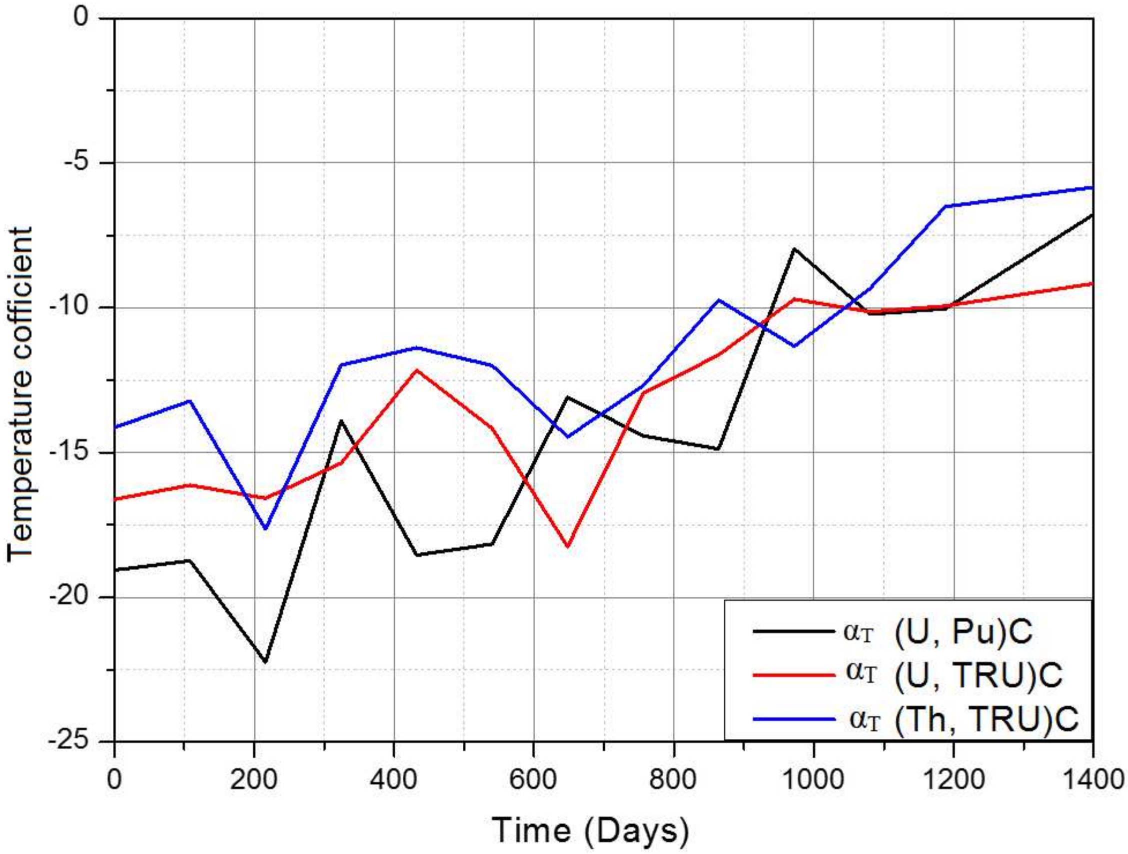

Figure 8: Evolution of temperature coefficient of reactivity

3.1.3 Criticality Evaluation with B_4C Absorber

Figure 9 shows the k_{inf} with B_4C rods insertion every 10 cm inside the core. Some tests increasing the radius of the rods were performed before obtaining a subcritical core with full B_4C rods insertion at BOC. After that, it was found that the absorbers rod radius must be increased 1.0 mm to reached values of k_{inf} at BOC below 1 with fully absorber rod insertion (100 cm). Therefore, the amount of boron carbide added was 143 kg, totaling an absorber mass of 415 kg. After analyzing the results from figure 9, it is shown that for both codes and with the fuel (U, Pu)C, the core becomes subcritical about 90 cm of rods insertion. For fuel (U, TRU)C, the subcriticality is only achieved at 90 cm for the SCALE 6.0, but not for the MCNPX. For this last one, a fully rods insertion is needed to reach subcriticality. However, (Th, TRU) C fuel also presents this difference in the results, being the k_{inf} calculations by the MCNPX (1.0006) are higher than the ones calculated by SCALE 6.0 (0.99543).

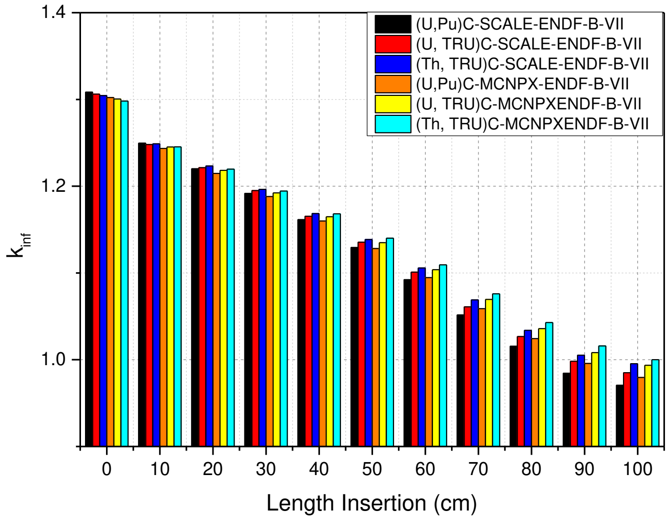

Figure 9: k_{inf} Behavior for the Three Fuels With the Height Variation of the Absorber.

Figure 10 shows the reactivity insertion effects with the B4C rods at every 10 cm. The reactivity was calculated using the equation (3), k_0 is the k_{inf} at BOC and k_1 is the k_{inf} for each absorber insertion length.

$$\rho(pcm) = \frac{k_1 - k_0}{k_1 k_0} x 10^5 \tag{3}$$

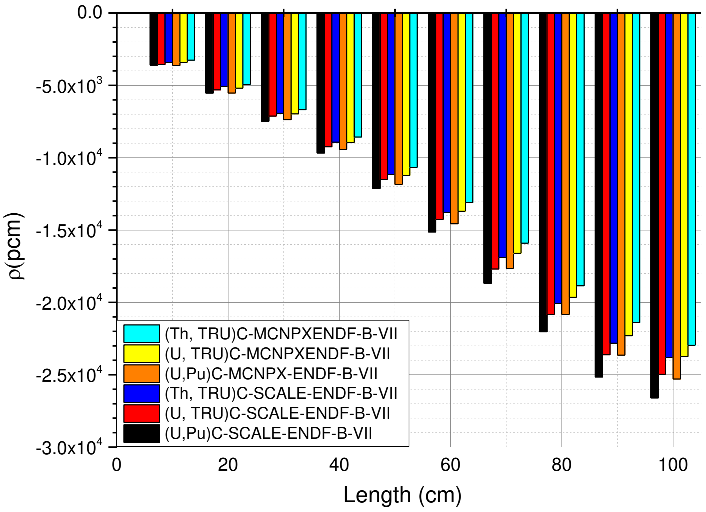

Figure 10: Reactivity Insertion for different lengths

4. CONCLUSIONS

The k_{inf} calculation shows little discrepancies between the MCNPX and SCALE 6.0 using the ENDF/B-VII.0. Nevertheless, the depletion rates for the main actinides are similar comparing the data provided by MCNPX and SCALE. The simulations showed that the fuel temperature coefficient of reactivity is consistent with the temperature increment. The substitution (U, Pu)C by (Th,TRU)C shows that the k_{inf} is higher than the (U,Pu)C and (U,TRU)C, which allows a burnup extension longer. Even though, the negative reactivity insertion (B_4C) should be much higher than the fuel using uranium. Therefore, the absorbers rods should be greater with higher amounts of B_4C to withstand the criticality excess. Finally, the nuclides build-up pointed out a similarity between the two codes used for the main actinides. Likewise, the model proposed here and the proposed using reprocessed fuels present expected behaviors and consistent with each other. DEN-UFMG is continuing its research with the aim of soon proposing its own GFR model based on the latest publications of the GEN-IV International Forum.

ACKNOWLEDGMENTS

The authors are grateful to the Brazilian research funding agencies, CNEN, CNPq, CAPES and FAPEMIG for the support.

REFERENCES

1. Annual Report – GEN-IV International Forum, 2009, 2012 and 2015
2. R. Reyes-Ramírez, et al. *"Comparison of MCNPX-C90 and TRIPOLI-4-D for fuel depletion calculations of a Gas-cooled Fast Reactor"*. Annals of Nuclear Energy **37**, 1101-1106, (2010).
3. C. Martín-Del-Campo et al. *"Contributions to the neutronic analysis of a gas-cooled fast reactor"*. Annals of Nuclear Energy **38**, 1406-141,(2011).
4. C. Martín-Del-Campo et al. *"Validation of simplified methods for fuel depletion calculations in gas-cooled fast reactors"*.
5. Macedo A. A. P. et al. *"Neutronic Analysis of Reprocessed Fuel in a Gas-Cooled Fast Reactor"*, Departamento de Engenharia Nuclear – Escola de Engenharia da Universidade Federal de Minas Gerais (UFMG), Brazil.
6. Macedo A. A. P., *"Proposta de um Reator Rápido Refrigerado a Gás Utilizando Transurânicos"*. Master degree dissertation, Escola de Engenharia da Universidade Federal de Minas Gerais (UFMG), Brazil.
7. W.F.G. Van Roijen. *"Improving Fuel Cycle and Safety of a Gas-cooled Fast Reactor"*. IOS Press NieuweHemweg-Amstherdam – Netherlands
8. Anthony M. Judd – *"An Introduction to The Engineering of Nuclear reactors"* — Cambridge Univerity Press – First edition 2014
9. Yasky, Peter – *"Core design and reactor physics of a breed and burn gas-cooled fast reactor"* - Massachusetts Institute of Technology. Dept. of Nuclear Engineering.
10. Thomas B. Cochan, et al. *"Fast Breeder Reactor Programs: History and Status"* A research report of the International Panel on Fissile Materials, February 2010
11. Argonne National Laboratory. Lab-Scale Demonstration of the UREX+ Process. WM'04 Conference, February 29 – March 4, Tucson, AZ. USA (2004).

12. Cota S., Pereira C. Neutronic Evaluation of the Non-Proliferanting Reprocessed Nuclear Fuels in Pressurized Water Reactors. Annual Nuclear Energy **24**, p. 829–834, 1996

13. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations, ORNL/TM-2005/39, Version 6, Vols. I–III, January 2009. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-750.