BJRS

BRAZILIAN JOURNAL
OF
RADIATION SCIENCES
10-03A (2022) 01-13

Criticality and depletion analysis of the European Lead-Cooled Training Reactor (ELECTRA)

Oliveira[a] A. L. P., Silva[a] C. A. M.

[a] UFMG/Departamento de Engenharia Nuclear, Av. Antônio Carlos, 6627, Campus UFMG, PCA01, Anexo Engenharia, Pampulha,31270-90, Belo Horizonte, MG, Brasil
pereiraoliveiraana@gmail.com, clarysson@nuclear.ufmg.br

ABSTRACT

In this article, a criticality and depletion analysis of the European Lead-Cooled Training Reactor (ELECTRA), a low-power, compact, fast lead-cooled reactor, was performed. The active core model and simulations were made by using the MCNPX code. First, the model is compared with the reference and the influence of the reflector radius on the system is analyzed. Secondly, it was simulated the fuel combustion after 30 years of continuous operation of the ELECTRA, managing to evaluate the transmutation of a reprocessed fuel in a compact lead-cooled reactor.

Keywords: **ELECTRA, Fast Nuclear Reactor, Lead-Cooled Reactor, MCNPX.**

ISSN: 2319-0612
DOI: https://doi.org/10.15392/2319-0612.2022.1940
Submitted: 2022-01-31
Accepted: 2022-11-27

1. INTRODUCTION

Fast nuclear reactors are part of Generation IV Nuclear Reactors, joined in 2000 by the Generation IV International Forum (GIF), completing cooperation in development and research to optimize existing reactors. The main objectives of this generation of reactors are (a) more efficient use of natural resources; (b) improve nuclear safety; (c) improve proliferation resistance; (d) minimize the production of burnt fuel; and (e) decrease the cost of building and operating nuclear power plants [1]. Among the proposed fourth-generation reactors is the lead-cooled fast reactor, such as the ELECTRA.

ELECTRA is a fast, low-power reactor (0.5 MWth) – because it is made initially for research proposals –, which, in addition to having inherent safety, enables the use of reprocessed fuels, such as $(Pu_{0.4}, Zr_{0.6})N$ (in molar percentage) [2]. It is remarkable that these characteristics are aligned with the objectives (a), (b) and (d) of the first paragraph, which makes ELECTRA part of the studies for the IV generation of nuclear reactors. Advantages not only economical, but also ecological, which make it a smart option to help meet the growing demand for energy through the use of alternative and diversified sources.

The fuel studied, $(Pu_{0.4}, Zr_{0.6})N$, was developed by the Royal Institute of Technology, with Uppsala University and Chalmers of Technology, in Sweden [2]. It comes from pressurized water reactor UOX fuel, burning 43 GWd/t, cooled for 4 years before reprocessing and for 2 years storage before charging to the core [2]. Nitride fuels have attractive advantages, such as high thermal conductivity, low expansion, low gas release and good thermal compatibility with the coolant in question [3].

As for the coolant/reflector (lead), the highlights are the low neutron absorption, the relatively low melting point (327ºC) and the high boiling point (1740ºC) [1] [2]. In addition, it does not react significantly with water or air – unlike sodium, which makes the reactor safer – and is a convenient choice for a fast reactor coolant such as ELECTRA.

This article presents a criticality study and a fuel depletion analysis of the ELECTRA using the MCNPX 2.6.0 (Monte Carlo N-Particle eXtended). The objective of the study is to evaluate the effective multiplication factor (k_{eff}) of the system and to study and analyze the transmutation of a reprocessed fuel in ELECTRA, a lead-cooled compact reactor.

2. MATERIALS AND METHODS

First, the modeling of the reactor core was made, considering the dimensions and composition of the references [2, 4]. The actinide composition of the fuel, (Pu 0,4, Zr 0,6) N, is shown in Table 1 below.

Table 1: Actinide composition of the fuel.

Actinide	**Pu 238**	**Pu 239**	**Pu 240**	**Pu 241**	**Pu 242**	**Am 241**
N (atm/cm^3)	7.05E-03	1.04E-01	4.75E-02	2.33E-02	1.56E-02	2.39E-01
Actinide	**Zr 90**	**Zr 91**	**Zr 92**	**Zr 94**	**Zr 96**	**N 14**
N (atm/cm^3)	1.56E-01	3.38E-02	5.10E-02	5.06E-02	7.99E-3	5.00E-1

Source: the authors. [2]

For the fuel cladding, T91 steel coated with FeCrAlY [5] was used. Figure 1 illustrates the active core of the ELECTRA reactor, plotted for simulation, with a reflector radius equal to 30.0 cm, while Table 2 shows some system parameters.

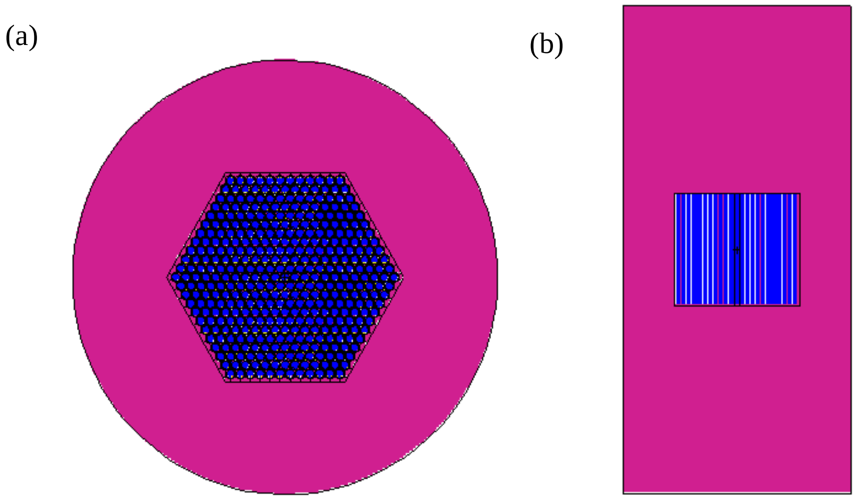

Figure 1: *Simple model of ELECTRA fuel assembly in a lead pool for the reflector impact study. (a) Plan xy; (b) Plan xz.*

Table 2: Characteristics of the Model of ELECTRA's active core.

Parameter	Value	Unit
Hexagon flat-to-flat distance	28.2	cm
Coolant inlet temperature	673	K
Coolant outlet temperature	773	K
Total number of pins	397	-
Pin pitch	1.4	cm
Gap thickness	0.05	mm
Total height	130	cm
Fuel height	30	cm
Cladding inner/outer diameter	1.26/1.16	cm
Fuel pellet diameter	1.11	cm
Fuel pellet density	9.44	g/cm^3

In the first stage of the work, the effective multiplication factor (k_{eff}) and the fraction of neutrons that return to the core (f) were compared between the reference and the model configured in MCNPX. So, the criticality analysis of ELECTRA at steady-state criticality analysis was performed for different reflector radius values. The code returned the value of the effective multiplication factor (k_{eff}) and the fraction of neutrons returning to the core (f), due to the variation of the reflector radius, calculated according to the definition of Spriggs et al. [6], according to Equation (1),

$$f = 1 - \frac{k_c}{k_{eff}} \tag{1}$$

where k_c is the k_{eff} value of the reactor without a reflector. The differences (D) between the values obtained for k_{eff} and f were calculated by Equations (2) and (3) shown below:

$$\mathrm{D(pcm)} = [\, k_{eff}(reference) - k_{eff}(MCNPX)\,] * 10^5 \tag{2}$$

$$\mathrm{D(\%)} = \frac{[\, f(MCNPX) - f(reference)\,]}{f\,(MCNPX)} * 100\% \tag{3}$$

Furthermore, in the second stage of the work, in order to determine the total anti-leakage probability of the core (P), the infinite multiplication factor (k_{inf}) of the system was also estimated by MCNPX, where P was calculated by Equation (4),

$$P = \frac{k_{eff}}{k_{inf}} \tag{4}$$

Beyond that, the neutron flux as a function of the radius and as a function of energy were calculated and plotted. Not only that, but the calculations of fuel burnup were done for the time of thirty years – given as the reactor lifetime.

3. RESULTS AND DISCUSSION

The results of the simulations are presented in Tables 3 and 4. First, from Table 3, there is a comparison between the results of the MCNPX and the reference [2]; There is a tendency in the reduction of the differences as the reflector radius increases. The standard deviation for the effective multiplication factor simulations found was only 0.00067.

Other tendence that can be noted is that the effective multiplication factor, k_{eff}, is directly proportionally related to the reflector radius. Moreover, from 100.0 cm onwards the impact of the increase in radius becomes less significant. The phenomenon follows the same pattern for the fraction of neutrons returning to the core, f. Furthermore, the value of P increases with increasing radius of the reflector, indicating that the probability that neutrons do not escape the core is greater for cases where the radius of the reflector is larger. As for the differences (D) between the results calculated in this work and those presented in the reference article [2] and for the standard deviations, it is verified that the greatest difference related to k_{eff} occurs for the reactor without the coolant, while the smallest difference occurs for the case in which the radius of the reflector is equal to 200.0 cm. Regarding the fraction f, the largest and smallest differences are also related to the cases of the reactor with radium of 0.0 cm and the 200.0 cm radius, respectively. Facts that show a tendency to reduce differences with increasing radius of the reflector.

Table 3: Effective Multiplication Factor (k_{eff}) and Fraction of Neutrons (f) that return to the Core calculated for Different Radius of the Reflector.

Reflector Radius, cm	**k_{eff}**			**f**		
	Reference	**MCNPX**	**D, pcm**	**Reference**	**MCNPX**	**D, %**
0.0 ($k_{eff} = k_c$)	0.77316	0.75055	2261	0	0	0
30	0.99083	0.98176	907	0.22	0.24	8.33
50	1.06525	1.05985	540	0.27	0.29	6.9
70	1.09338	1.0878	558	0.29	0.31	6.45
100	1.10725	1.10269	456	0.3	0.32	6.25
130	1.11038	1.10493	545	0.3	0.32	6.25
200	1.11111	1.10678	433	0.3	0.32	6.25

Table 4 shows the values of the Infinite Multiplication Factor, calculated by MCNPX. This parameter does not consider neutron leakage in the system. The table also shows the values obtained for the Total Anti-Leakage Probability of the Core. As the radius increases, P increases, as it reduces leakage, contributing to the increase in the keff value.

Table 4: Infinite Multiplication Factor (k_{inf}) and Total Anti-Leakage Probability of the Core (P) calculated for different radius of the reflector.

Reflector Radius, cm	**k_{eff} (MCNPX)**	**k_{inf}**	**$P = k_{eff}/k_{inf}$**	**P - 1**
0.0 ($k_{eff} = k_c$)	0.75055	1.3831	0.54266	-0.4573
30	0.98176	1.81581	0.54067	-0.4593
50	1.05985	1.46446	0.72371	-0.2763
70	1.0878	1.28076	0.84934	-0.1507
100	1.10269	1.1858	0.92991	-0.0701
130	1.10493	1.1611	0.95162	-0.0484
200	1.10678	1.15441	0.95874	-0.0413
Standard Deviation	0.13001	0.23695	0.18496	0.18496

Figure 2 below represents the radial neutron flux profile of ELECTRA core. The flux values were calculated considering the average volume by radial zones. The error for MCNPX calculations shouldn't be bigger than 5% for coherent results. In this simulation, the error found was around

0.4%, showing that the analysis is very accurate. There is a flattening in the neutron flux with the increase of the reflector radius, because the largest reflector layer contributes to the lowest neutron leakage, which increases the neutron flux at the core periphery and decreases it at the core center, since the reflector has the function of increasing the fraction of neutrons that return to the core.

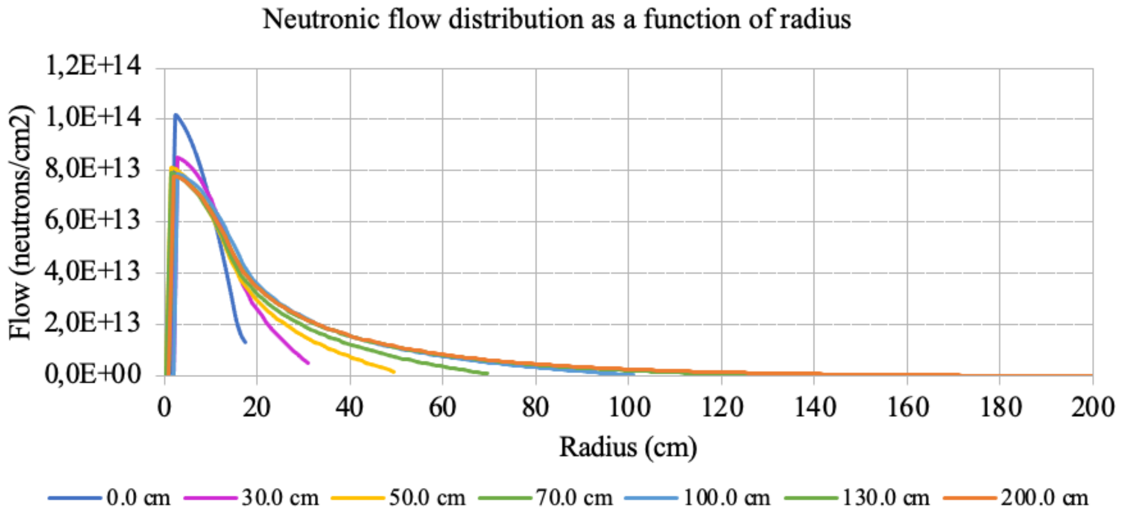

Figure 2: *Radial neutron flux profile of reactor core.*

Figure 3 illustrates the neutron flux spectrum to total volume of simulated systems. The code calculates the values considering the average neutron flux into the reactor core. By the results, it is plausible to say that the flux increases and is more consistent between the radius for the epithermal energy range, especially from 10^{-1} to 1 MeV. Another important conclusion is that the neutronic flux is directly related to the reflector radius.

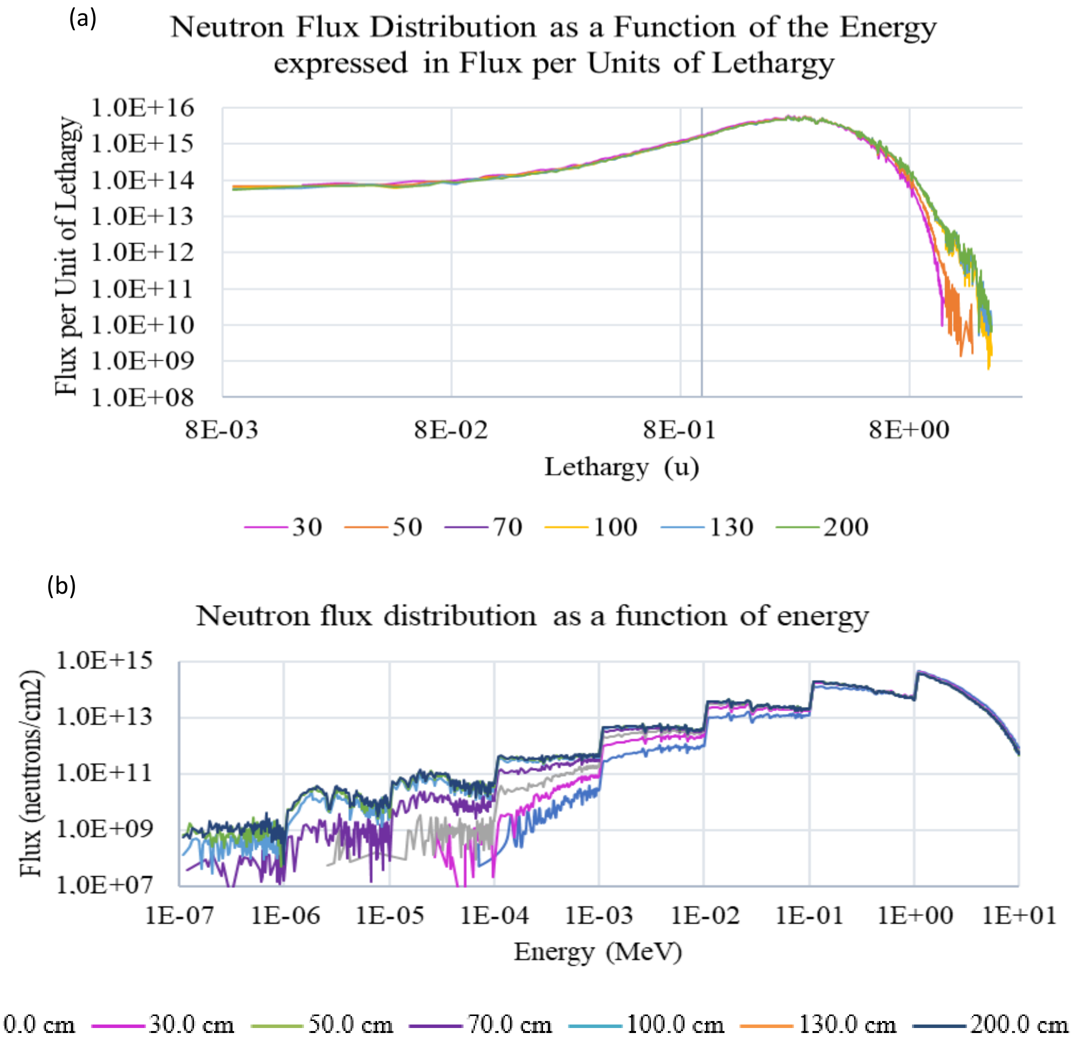

Figure 3: *Neutron flux spectrum to total core volume (a) Flux per Unit of Lethargy; (b) Flux per Energy (Mev).*

The Criticality of ELECTRA over 30 years of continuous fuel burning is shown in Figure 4. The graph allows to conclude that, in an ideal situation of continuous fuel burning for 30 years, the proposed core would be able to operate at 0.5MWth for about 25 years, not becoming subcritical until this period has passed.

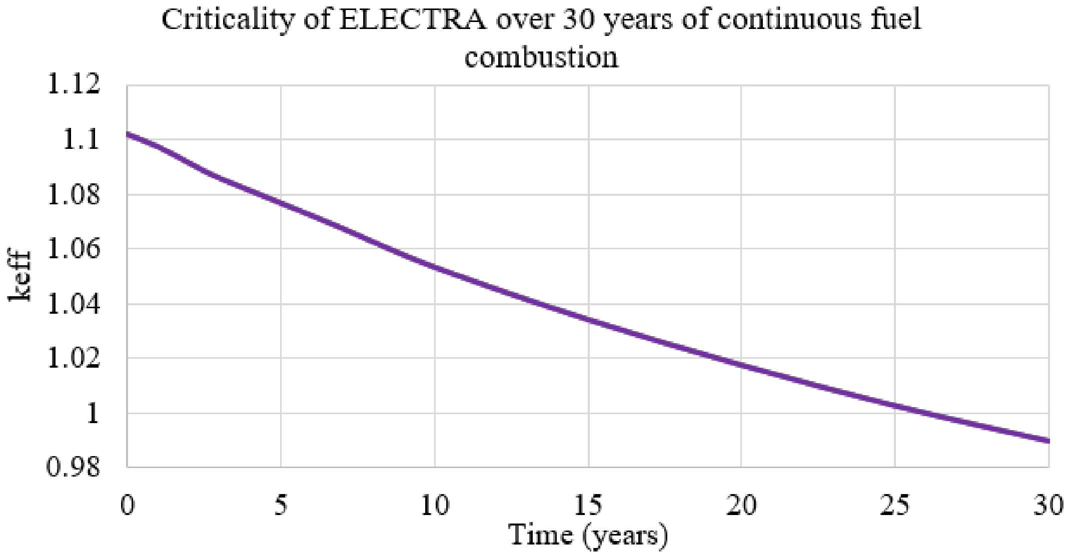

Figure 4: *Effective multiplication factor as function of time (years) for ELECTRA.*

Finally, but not least important, Tables 5 – 9 below shows the composition over 30 years of fuel burning of (a) Plutonium; (b) Neptunium; (c) Curium; (d) Americium; and (e) Uranium. It shows an overview of the transmutations results, where the total amount of Plutonium decreased to 85.96% of the amount of the first year, while the Uranium appeared going from 0 to 4.09E-03 (weight fraction) at the final of the 30th year. The Neptunium 237 went from 0 to 1.72E-03 (weight fraction) and the Curium amount increased until the 25th year, from 0 to 3.85E-03 (weight fraction). Finally, the Americium amount increased significantly over the years, going to 779% of the initial amount.

Table 5: Plutonium composition over 30 years of fuel burning.

Plutonium (94) composition (weight fraction) over 30 years of fuel burning						
Time (years)	**238**	**239**	**240**	**241**	**242**	**Total**
0	2.04E-02	3.02E-01	1.39E-01	6.82E-02	4.60E-02	5.76E-01
1	2.02E-02	3.02E-01	1.39E-01	6.48E-02	4.60E-02	5.72E-01
2	2.00E-02	3.01E-01	1.39E-01	6.17E-02	4.60E-02	5.68E-01
3	1.98E-02	3.00E-01	1.38E-01	5.87E-02	4.60E-02	5.63E-01
5	1.95E-02	2.99E-01	1.38E-01	5.31E-02	4.59E-02	5.56E-01
7	1.91E-02	2.98E-01	1.38E-01	4.80E-02	4.59E-02	5.49E-01
10	1.86E-02	2.96E-01	1.38E-01	4.13E-02	4.58E-02	5.40E-01
15	1.79E-02	2.92E-01	1.37E-01	3.22E-02	4.57E-02	5.25E-01
20	1.72E-02	2.89E-01	1.37E-01	2.52E-02	4.55E-02	5.14E-01
25	1.67E-02	2.85E-01	1.36E-01	1.97E-02	4.54E-02	5.03E-01
30	1.61E-02	2.82E-01	1.36E-01	1.55E-02	4.52E-02	4.95E-01

Table 6: Uranium composition over 30 years of fuel burning.

Uranium (92) composition (weight fraction) over 30 years of fuel burning				
Time (years)	**234**	**235**	**236**	**Total**
0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1	1.58E-04	8.55E-06	1.44E-05	1.81E-04
2	3.13E-04	1.71E-05	2.88E-05	3.59E-04
3	4.68E-04	2.57E-05	4.31E-05	5.37E-04
5	7.71E-04	4.30E-05	7.18E-05	8.86E-04
7	1.07E-03	6.04E-05	1.00E-04	1.23E-03
10	1.50E-03	8.67E-05	1.43E-04	1.73E-03
15	2.20E-03	1.31E-04	2.14E-04	2.55E-03
20	2.87E-03	1.76E-04	2.85E-04	3.33E-03
25	3.51E-03	2.22E-04	3.55E-04	4.09E-03
30	4.12E-03	2.68E-04	4.25E-04	4.81E-03

Table 7: Neptunium composition over 30 years of fuel burning.

Neptunium (93) composition (weight fraction) over 30 years of fuel burning	
Time (years)	**237**
0	0.00E+00
1	1.36E-05
2	3.20E-05
3	5.51E-05
5	1.14E-04
7	1.88E-04
10	3.25E-04
15	6.07E-04
20	9.43E-04
25	1.32E-03
30	1.72E-03

Table 8: Americium composition over 30 years of fuel burning.

Americium (95) composition (weight fraction) over 30 years of fuel burning				
Time (years)	**241**	**242**	**243**	**Total**
0	6.99E-03	0.00E+00	0.00E+00	6.99E-03
1	1.02E-02	1.07E-06	1.11E-05	1.02E-02
2	1.32E-02	2.52E-06	2.22E-05	1.32E-02
3	1.60E-02	4.33E-06	3.35E-05	1.60E-02
5	2.13E-02	8.97E-06	5.62E-05	2.14E-02
7	2.60E-02	1.49E-05	7.90E-05	2.61E-02
10	3.21E-02	2.57E-05	1.14E-04	3.22E-02
15	4.03E-02	4.83E-05	1.73E-04	4.05E-02
20	4.64E-02	7.49E-05	2.34E-04	4.67E-02
25	5.08E-02	1.04E-04	2.95E-04	5.12E-02
30	5.40E-02	1.35E-04	3.57E-04	5.45E-02

Table 9: Curium composition over 30 years of fuel burning.

Curium (96) composition (weight fraction) over 30 years of fuel burning	
Time (years)	**242**
0	0.00E+00
1	4.22E-06
2	6.61E-06
3	8.55E-06
5	1.20E-05
7	1.53E-05
10	1.97E-05
15	2.59E-05
20	3.10E-05
25	3.85E-05
30	3.85E-05

4. CONCLUSION

The model developed in the MCNPX code allowed evaluating the criticality of the ELECTRA reactor, by calculating the effective multiplication factor, k_{eff}, the infinite multiplication factor, k_{inf}, and the neutron fraction that returns to the core, f, and verifying the results by comparison with the reference article [2]. The probability of non-leakage of the system as a function of the radius of the reflector was also calculated. Finally, the transmutation of the reactor over thirty years of fuel burning was analyzed. The results shows that the proposed core can operate at 0.5MWth for about 25 years continuously. Overall, these results corroborate with the understanding of the performance of fast reactors regarding the reactor criticality related to the reflector dimensions and its performance over the years.

ACKNOWLEDGMENT

The authors express their gratitude to Professor Claubia Pereira Bezerra Lima for running the simulations in the MCNPX 2.6.0 code. They also express its appreciations to the research funding agencies: FAPEMIG (Research Support Foundation of the State of Minas Gerais), CAPES (Coordination of Improvement of Personnel in Higher Education), CNPq (National Council for Scientific and Technological Development) and CNEN (National Nuclear Energy Commission).

REFERENCES

[1] BASTOS W. S. **Uma introdução a reatores rápidos de IV geração**. Instituto de Engenharia Nuclear – CNEN. Rio de Janeiro, Brazil (2002).

[2] SUVDANTSETSEG E. **Neutronics and Transient Analysis of a Small Fast Reactor Cooled with Natural Circulation of Lead.** Royal Institute of Techonology, Stockholm, Sweden (2014).

[3] RENKE C. A. C., BATISTA J. L., WAINTRAUB M., BASTOS W. S., AGHIMA L. O. B. **Análise Neutrônica Concernente à Utilização de Combustível Nitreto de Urânio/Plutônio em Reatores a Nêutrons Rápidos**. Instituto de Engenharia Nuclear – CNEN. Rio de Janeiro, Brazil (1991).

[4] KRÓLIKOWSKI I. P., CETNAR J. **Neutronic and Thermal-Hydraulic Coupling for 3D reactor core modeling combining MCB and fluent.** Faculty of Energy and Fuels, AGH University of Science and Technology, Krakow, Poland (2015).

[5] WEISENBURGER A., HEINZEL A., MÜLLER G., MUSCHER H., ROUSANOV A. **T91 cladding tubes with and without modified FeCrAlY coatings exposed in LBE at different flow, stress and temperature conditions**. Journal of Nuclear Materials, v. 3, p. 376 (2008).

[6] SPRIGGS G. D. ET AL. **Two-Region Kinetic Model for Reflected Reactors**. Ann. Nucl. Energ., 24 (1997).

[7] Nuclear Center at KAERI. **Table of nuclides**. Accessible in: https://atom.kaeri.re.kr/nuchart/.