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NEUTRONIC EVALUATION OF ANNULAR FUEL RODS TO ASSEMBLIES 13X13, 14X14 AND 15X15.

Raphael H. M. Silva[1], Mario C. Ramos[1], Carlos E. Velasquez[1], Clarysson A. M. da Silva[1], Cláubia Pereira[1] and Antonella L. Costa[1]

[1] Departamento de Engenharia Nuclear
Universidade Federal de Minas Gerais
Av. Antônio Carlos, 6627, Campus UFMG PAC 1 – Anexo Engenharia, Pampulha,
31270-90 Belo Horizonte, MG
rapha.galo@hotmail.com, marc5663@gmail.com, carlosvelcab@hotmail.com, clarysson@nuclear.ufmg.br, claubia@nuclear.ufmg.br, antonella@nuclear.ufmg.br

ABSTRACT

Research and development in nuclear reactor field has been proposed a new concept of fuel rod such as annular shape. The design of the annular fuel rods allows the coolant flow through the inner and outer side of it. Such project was proposed as an alternative to the traditional fuel rods used in LWR reactors. This new geometry allows an increase in power density in the reactor core with greater heat transfer from the fuel to the coolant which reduces the temperature in central region of the rod, in which a better configuration and dimension of fuel elements are aimed due to improvement of cooling in possible replacement of PWR traditional rods for annular rods. The aim of this work is to evaluate the neutronic parameters of fuel element with annular fuel rods where three configurations were studied: 13x13, 14x14 and 15x15. The goal is compare the neutronic between the advanced and the standard fuel assembly 16x16. In these studies, the external dimension and the moderator to fuel volume ratio (V_M/V_F) of standard 16x16 is the same in all annular fuels assemblies. The MCNPX 2.6.0 (Monte Carlo N-Particle eXtended – version 2.6.0) code was used in all simulations. After all procedures, the annular fuel assembies 13 have obtained greater neutronics parameters and were selected to more neutronics simulations.

1. INTRODUCTION

Reactors with LWR technology are responsible for generating the bulk of electricity generated by nuclear power plants in the world. Almost 90% of these plants with this technology, work exclusively using the uranium oxide (UO_2). The operation of nuclear fuel cycle has been subjected of studies during few decades, aiming to optimize the process energy generation, as well as, improve system performance [1].

Currently, there are approximately 439 nuclear power plants in the world related to different technology. Brazil, has a PWR nuclear power plant complex at Angra, where are localized two power plants in operation Angra I and II producing 640 MW / 1350 MW, respectively. The third nuclear plant, named Angra III, is in under construction. The electric production associated to the nuclear power plant complex represents about 2% of the electricity consumed in the national energy system [2].

Since the 1950's, researchers have been developing a new technology related to fuel rod design, which allowed coolant flow internal and external to the fuel used. Such project was called annular fuel and it has been proposed as an alternative to the traditional fuel rods used in LWR reactors. This new geometry allows an increase in power density in the reactor core demonstrated through neutronics and thermal-hydraulic parameters simulated [4].

In addition to the significant advantages mentioned above, annular fuel rods provide other positive factors in the context studied such as the thermo-hydraulic aspect, which heat transfer from the fuel to the coolant rods is considerably higher due to the incorporation of internal cooling channels in geometry. The use of this technology minimizes the problems related to cracks, ruptures or deformation of the cladding, due to lower temperatures in the fissile material region [3].

The study related to this fuel rod type, represents a significant performance improvement proposal for LWR (light water reactors) in operation around the world. In addition, experimental and theoretical studies aimed at improving the neutronic and thermo-hydraulic approach to maintain or even expand security parameters [4,5].

In this context, the present study discusses different neutronics parameter associated to assemblies' configurations with annular rods. The goal is to evaluate the replacement of traditional rods for typical PWR assemblies (16x16) by annular one. In this work, it was considered Angra II fuel assembly [6] for comparison reference, which composition and the external dimensions were used in simulation in the ECAs. To achieve the objective mentioned above, the standard assemblies (ECP) and the annular assemblies (ECAs) will be simulated using the MCNPX 2.6.0 (Monte Carlo N-Particle eXtended-Version 2.6.0) code where neutronics parameters were calculated. Such simulations intending to compare the neutronics behavior between the ECP and the ECAs by criticality analysis, neutron flux in fuel assemblies and fuel cells.

2. MATERIAL AND METHODS

The development of the paper occurred in two stages. In the first step, the ECP neutron reference validation. For this, were used the MCNPX codes 2.6.0 and SCALE 6.0. In the second step, based on the external dimensions and the isotopic composition of the ECP, the ECAs were simulated using the MCNPX codes 2.6.0 to compare the neutronics parameters between ECP and ECA.

2.1. Validation of ECP (16x16) Reference

In the first stage of this work, the ECP (16x16) was simulated using the MCNPX 2.6.0 and SCALE 6.0 codes. The library used by MCNPX 2.6.0 code was the ENDF/B-VII. 1 and SCALE 6.0 the case was run with ENDF/B-VII collapsed in 238 energy groups available on SCALE package. A criticality comparison between the two codes were made for different temperatures (Hot Full Power "and " Zero Cold Power ") and different enrichment of UO_2. The principal objective of this validation is to verify that the ECP (16x16) have simulated in MCNPX and the SCALE codes have the similar behavior.

Under "Hot Full Power" conditions the fuel / gap and the cladding / moderator were simulated at temperatures of 873 K / 873 K and 618 K / 587 K, respectively. Under "Zero Cold Power" conditions all regions were modeled at 293. The gaps thicknesses, the cladding properties and guides tubes were performed in accordance to reference document [6]. The Fig. 1 shows the fuel rods disposition and the guides tubes, Table 1 presents the main data of the ECP and this system was set up by a square-meshed panel, where each cell of this mesh represents rods related to fuel and guide tube. This fuel assembly was named ECP 16.

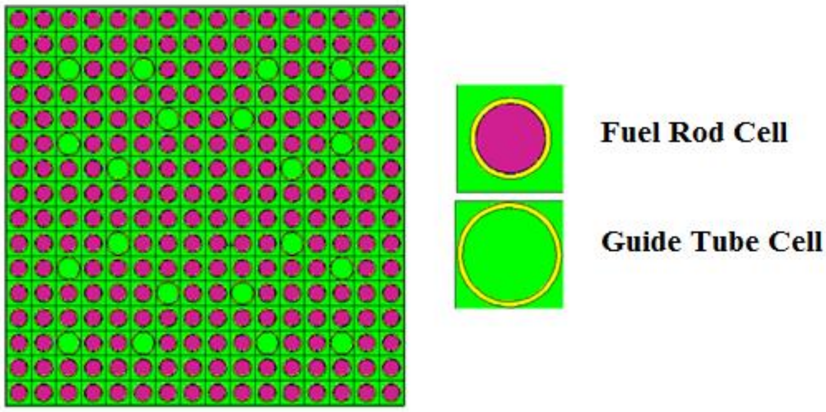

Figure 1: Reference (16x16) typical fuel assembly.

Table 1: Dates of simulation into MCNPX 2.6.0 e no SCALE 6.0 codes.

<table>
<tr><td colspan="3">External Dimension (cm)</td><td>$2.288\ 10^{1}$</td></tr>
<tr><td colspan="3">Active length (cm)</td><td>$3.916\ 10^{2}$</td></tr>
<tr><td colspan="3">Number of Fuel Rods</td><td>$2.360\ 10^{2}$</td></tr>
<tr><td colspan="3">Number of Guide Tubes</td><td>$2.000\ 10^{1}$</td></tr>
<tr><td colspan="3">Pitch Distance (cm)</td><td>1.430</td></tr>
<tr><td rowspan="2">Radius Guide Tubes (cm)</td><td colspan="2">Inner</td><td>$6.200\ 10^{-1}$</td></tr>
<tr><td colspan="2">Outer</td><td>$6.900\ 10^{-1}$</td></tr>
<tr><td rowspan="6">Radius Fuel Rod (cm)</td><td rowspan="3">Zero Cold Power</td><td>Outer Radius of Fuel (R_{FO})</td><td>$4.555\ 10^{-1}$</td></tr>
<tr><td>Inner Radius of Cladding (R_{COI})</td><td>$4.650\ 10^{-1}$</td></tr>
<tr><td>Outer Radius of Cladding (R_{COO})</td><td>$5.375\ 10^{-1}$</td></tr>
<tr><td rowspan="3">Hot Full Power</td><td>Outer Radius of Fuel (R_{FO})</td><td>$4.583\ 10^{-1}$</td></tr>
<tr><td>Inner Radius of Cladding (R_{COI})</td><td>$4.659\ 10^{-1}$</td></tr>
<tr><td>Outer Radius of Cladding (R_{COO})</td><td>$5.385\ 10^{-1}$</td></tr>
</table>

The isotopic composition used in the simulations is presented in Table 2. Based on the reference document [6], three enrichments were considered: 1.9%, 2.5% and 3.2%, the cladding is Zircaloy-4 material and all simulations do not considered boron diluted in moderator.

Table 2: Dates of all composition used in simulation.

Region	**Isotopes**	**Concentration** (Atomic Density)
Fuel (1.9 %)	^{235}U	$4.462\ 10^{-4}$
	^{238}U	$2.275\ 10^{-2}$
	^{16}O	$4.639\ 10^{-2}$
Fuel (2.5 %)	^{235}U	$5.872\ 10^{-4}$
	^{238}U	$2.261\ 10^{-2}$
	^{16}O	$4.639\ 10^{-2}$
Fuel (3.2 %)	^{235}U	$7.516\ 10^{-4}$
	^{238}U	$2.244\ 10^{-2}$
	^{16}O	$4.640\ 10^{-2}$
Gap	^{3}He	$1.504\ 10^{-7}$
	^{4}He	$1.504\ 10^{-1}$
Cladding	Zr (nat)	$4.254\ 10^{-2}$
	Sn (nat)	$4.825\ 10^{-4}$
	Fe (nat)	$1.485\ 10^{-4}$
	Cr (nat)	$7.597\ 10^{-5}$
	Hf (nat)	$2.213\ 10^{-6}$
Moderator (0 ppm of boron)	^{1}H	$4.719\ 10^{-2}$
	^{16}O	$2.359\ 10^{-2}$

2.2. Annular Assemblies Simulation.

Three annular fuel assemblies were simulated in this step. The rods number of such assembly was based on previously work [7], but the external dimensions, as well as, the V_M/V_F (Moderator Volume/Fuel Volume) ratio were based on the ECP 16. In the Fig. 2, the configuration details of the ECAs and the parameters evaluated are described below.

- ECA 13 - Annular Fuel Assembly - dimension 13 x 13;
- ECA 14 - Annular Fuel Assembly - dimension 14 x 14;
- ECA 15 - Annular Fuel Assembly - dimension 15 x 15.

In this part of the work the MCNPX 2.6.0 was used to simulate the ECAs to compare the results with the ECP 16. These simulations have occurred in two separate steps. At first, the ECAs were simulated according to the originally used in ECP enrichment, defined at 3.2%. In the second step, the ECAs were simulated by varying the enrichment of 3.3% to 4.0%. This methodology was used to obtain the same values of k_{inf} from assembly of reference (ECP 16). After these steps, the ECAs have set up with the new enrichment values were simulated to calculate criticality and neutron flux.

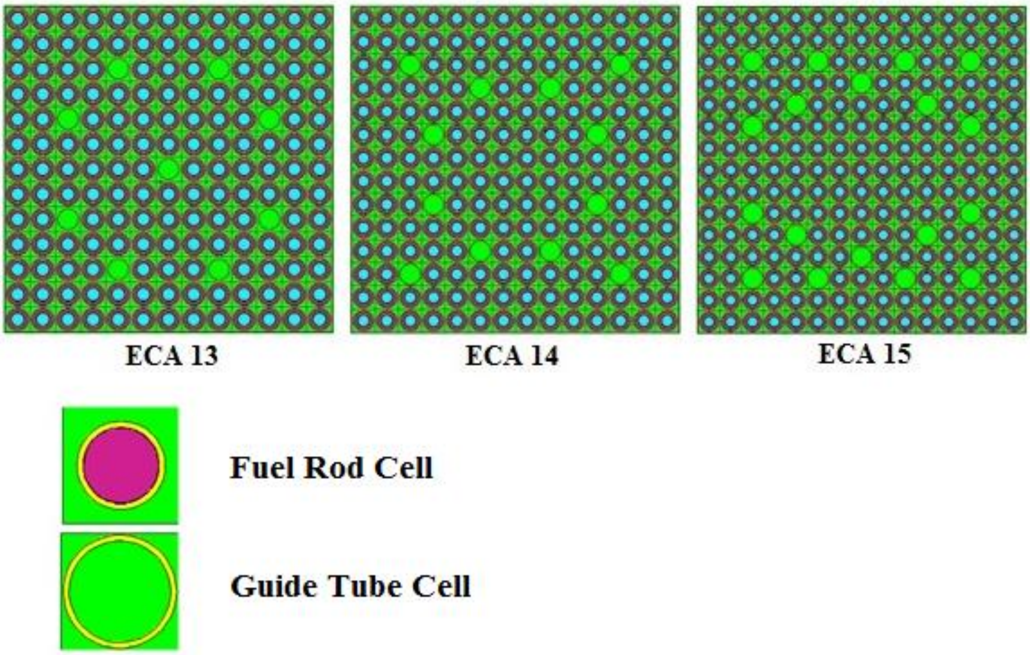

Figure 2: Annular fuel assemblies have simulated in research.

Using the same simulation methodology the ECP 16 and ECAs were configured using a square mesh panels, where each cell of this mesh represents a fuel rod or guide tube. The main objective is to maintain the same ECP 16 ratio of moderator and fuel V_M/V_F previously validated. Thus, external radius (R_{FO}) of fissile material of ECAs, which it was calculated to obtain the same ECP 16 V_M/V_F ratio. This calculation was based on the following method:

$$\left[\frac{V_M}{V_F}\right]_{ECP16} = \left[\frac{V_M}{V_F}\right]_{ECAs} \equiv R \tag{1}$$

To calculate the news radius (R_{FO}) used in annular fuel assemblies and aiming to maintain the ratio calculated above. The new radius was obtained by Eq. (1), which has as main goal to sustain the same ratio inside the same external dimension. The temperatures have treated in simulation related to the standard fuel assembly were 873 K to the fissile material and gap, 618 K to the cladding and 587 K to the moderator. In the case of the annular fuel assemblies, the temperatures were 600 K and 587 K to the material fissile / gap and cladding / moderator, respectively.

$$R = \frac{S_{FA} - \pi \cdot n_f \cdot (R_{COO}^2 - R_{CII}^2) - \pi \cdot n_g \cdot (R_{COO}^2 - R_{COI}^2)}{\pi \cdot n_f \cdot (R_{FO}^2 - R_{FI}^2)} \tag{2}$$

where:

$\frac{V_M}{V_F} \equiv R$ = volume moderator / volume fuel ratio of ECP 16;

R_{CII} is the inner radius of inner cladding;
R_{CIO} is the outer radius of inner cladding;
R_{FI} is the inner radius of fuel;
R_{FO} is the outer radius of fuel;
R_{COI} is the inner radius of outer cladding;
R_{COO} is the outer radius of outer cladding;
S_{FA} is the area of fuel assembly;
n_f is the number of fuel rods; and
n_g is the number of guide tubes.

In the Fig. 3, we can see all radius commented above.

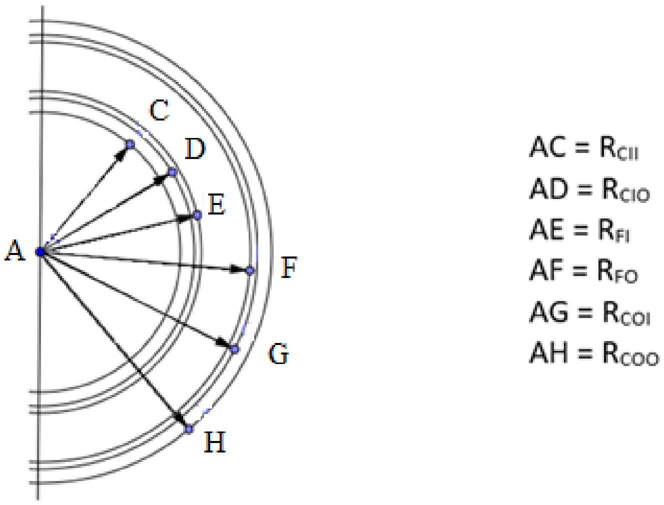

Figure 3: Radius of Annular Fuel Rods worked in the research.

Table 3, it can be seen the parameter used at ECP 16 and ECAs. These parameters were set up through works previously done [6,7]. This table show also the new radius calculated (in bold) used in simulation related to annular fuel assemblies ECA 13, ECA 14 and ECA15 in the comparison between the standard assembly ECP 16.

Table 3: Dates of simulation related to all assemblies simulated.

Parameters		ECP 16	ECA 13	ECA 14	ECA 15
Radius Fuel Rod (cm)	R_{cii}	-	$4.317\ 10^{-1}$	$3.767\ 10^{-1}$	$3.367\ 10^{-1}$
	R_{cio}	-	$4.888\ 10^{-1}$	$4.338\ 10^{-1}$	$3.938\ 10^{-1}$
	R_{FI}	-	$4.950\ 10^{-1}$	$4.400\ 10^{-1}$	$4.000\ 10^{-1}$
	$\boldsymbol{R_{Fo}}$	$4.583\ 10^{-1}$	**$7.314\ 10^{-1}$**	**$6.656\ 10^{-1}$**	**$6.155\ 10^{-1}$**
	R_{coi}	$4.659\ 10^{-1}$	$7.390\ 10^{-1}$	$6.732\ 10^{-1}$	$6.231\ 10^{-1}$
	R_{coo}	$5.385\ 10^{-1}$	$8.116\ 10^{-1}$	$7.458\ 10^{-1}$	$6.957\ 10^{-1}$
Radius Fuel Rod (cm)	Inner	0.620	0.700	0.700	0.700
	Outer	0.690	0.768	0.768	0.768
Number of Fuel Rods		$2.360\ 10^{2}$	$1.600\ 10^{2}$	$1.840\ 10^{2}$	$2.070\ 10^{2}$
Number of Guide Tubes		$2.000\ 10^{1}$	9.000	$1.200\ 10^{1}$	$1.800\ 10^{1}$
Pitch Distance		1.430	1.760	1.630	1.520
Volume (cm^3)	V_M	$1.180\ 10^{5}$	$1.104\ 10^{5}$	$1.092\ 10^{5}$	$1.079\ 10^{5}$
	V_F	$6.073\ 10^{4}$	$5.682\ 10^{4}$	$5.621\ 10^{4}$	$5.551\ 10^{4}$
V_M/V_F		1.944	1.944	1.944	1.944

2.2.1 Assessment of criticality.

In this stage, the fuel behavior as a function of temperature variation was analyzed, this process is important because verify if the evaluated systems are intrinsically safe in relation to temperature variation. In this case, the fuel temperatures have varied and the other components temperature remained constant at 273 K. The fuel's temperatures were associated to 300 K / 400 K / 500 K / 600 K / 800 K / 900 K / 1000 K and 1200 K. The enrichment used in this study was determined after the initial criticality analysis described earlier, which is 3.2% for the ECP 16 and 3.8% and 3.9% to ECA 13 and ECA 14 / ECA 15, respectively.

Table 4, shows the three enrichments used in the annular fuel assemblies, which were calculated previously, aiming to obtain the same k_{inf} of the standard assembly ECP 16.

Table 4: Composition of the new enrichment used in annular fuel assembly.

Enrichment	**Isotopes**	**Concentration** (Atomic Density)
Fuel (3.8 %)	^{235}U	$8.925\ 10^{-4}$
	^{238}U	$2.230\ 10^{-2}$
	^{16}O	$4.640\ 10^{-2}$
Fuel (3.9 %)	^{235}U	$9.159\ 10^{-4}$
	^{238}U	$2.228\ 10^{-2}$
	^{16}O	$4.640\ 10^{-2}$

2.2.2 Analysis of Neutronic Flux

In this way, to calculate the flux a 3D mesh is configured within pattern system. MCNPX estimates the flux in each cell of this mesh. This process has set up a 16 x 16 x 01 mesh and obtained the values analyzed, the mesh is configured coincident to the dimensions of the pitch distance of the fuel assembly. The ECA 13, was modeled after a 13 x 13 x 01. In this way, it is estimated the flux within each ECP and the ECAs cell. The flux also was analyzed related to the total dimensions of the assemblies, which it was established and demonstrated in the table 1 to the all cases simulated.

3. RESULTS AND DISCUSSION

3.1. Validation of ECP (16x16) Reference

In General, the values obtained between the two codes used in simulations presented minors in Zero Cold Power than Hot Full Power, but in all cases such differences are in the third decimal place.

In the simulations carried out on Zero Cold Power, the differences between the values were less than 4.000 10^2 pcm. In cases related to Hot Full Power, all simulations described have remained above the rate previously mentioned, but still it lower than 1.000 10^3 pcm. This behavior can be explained by the different libraries used in both programs.

The results observed in the validation confirm that the values have found in the simulations into MCNPX 2.6.0 and SCALE 6.0 codes. These differences are acceptable in criticality analysis, therefore, the models configured in such code can be used for calculating the k_{inf} of these systems. All these values can be seen below in the Table 5.

Table 5: Values related to the comparison between both codes.

Zero Cold Power					
Enrichment of Fuel	k_{inf}			σ	
	MCNPX	SCALE	Difference (pcm)	MCNPX	SCALE
1.9%	1.254	1.251	2.300 10^2	3.200 10^{-4}	5.800 10^{-4}
2.5%	1.328	1.324	3.780 10^2	3.800 10^{-4}	5.400 10^{-4}
3.2%	1.384	1.380	3.650 10^2	3.900 10^{-4}	5.500 10^{-4}
Hot Full Power					
Enrichment of Fuel	k_{inf}			Σ	
	MCNPX	SCALE	Difference (pcm)	MCNPX	SCALE
1.9%	1.233	1.223	9.880 10^2	3.800 10^{-4}	5.800 10^{-4}
2.5%	1.305	1.297	7.470 10^2	3.800 10^{-4}	5.700 10^{-4}
3.2%	1.360	1.353	7.170 10^2	4.100 10^{-4}	6.200 10^{-4}

3.2. Annular Assemblies Simulation.

3.2.1 Assessment of criticality.

Fig. 4 shows the k_{inf} values according on the previously analysis with 3.2%, 3.8% and 3.9% enrichment in comparison with the initial. Between the annular fuel assemblies, the ECA 14 has the k_{inf} closest to the ECP 16, this difference is estimated in (4.600 10^3 pcm). Considering criticality analysis, this difference is acceptable and, therefore, it can be considered that such enrichments generate k_{inf} values like ECP 16 reference.

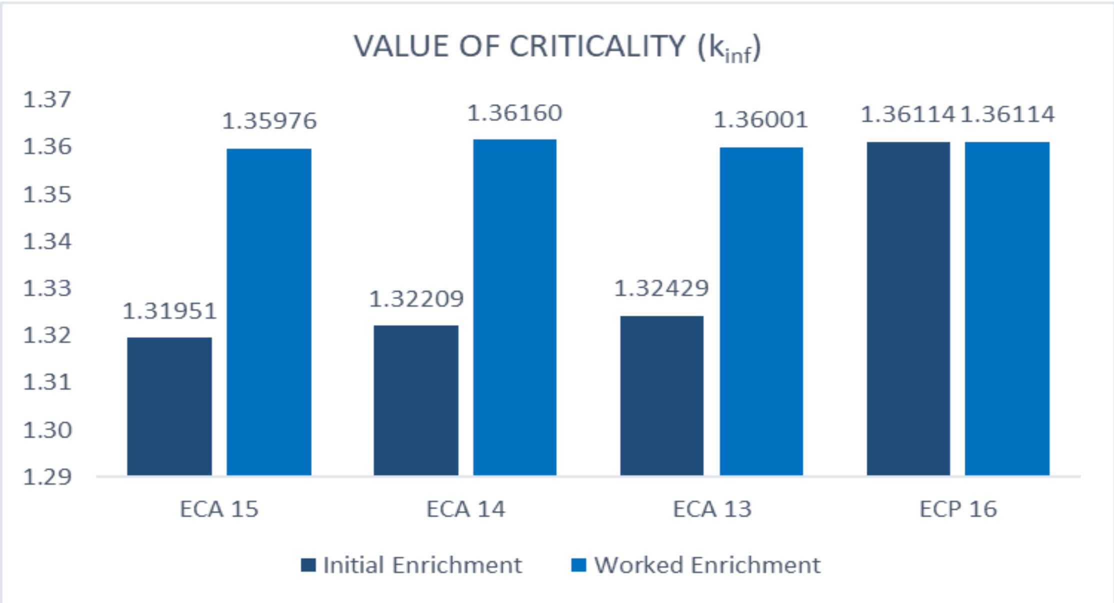

Figure 4: Values of criticality the all assemblies worked.

Fig. 5, shows the k_{inf} values related to the variation of the fuel's temperature. This figure shows the k_{inf} values decrease when the temperature is increased, this behavior can be explained through the Doppler effect [9]. In a general appearance, the ECP 16 presents higher k_{inf} values than other assemblies cases simulated at different temperatures. The ECA 14 has criticality value closest to the ECP 16 reference.

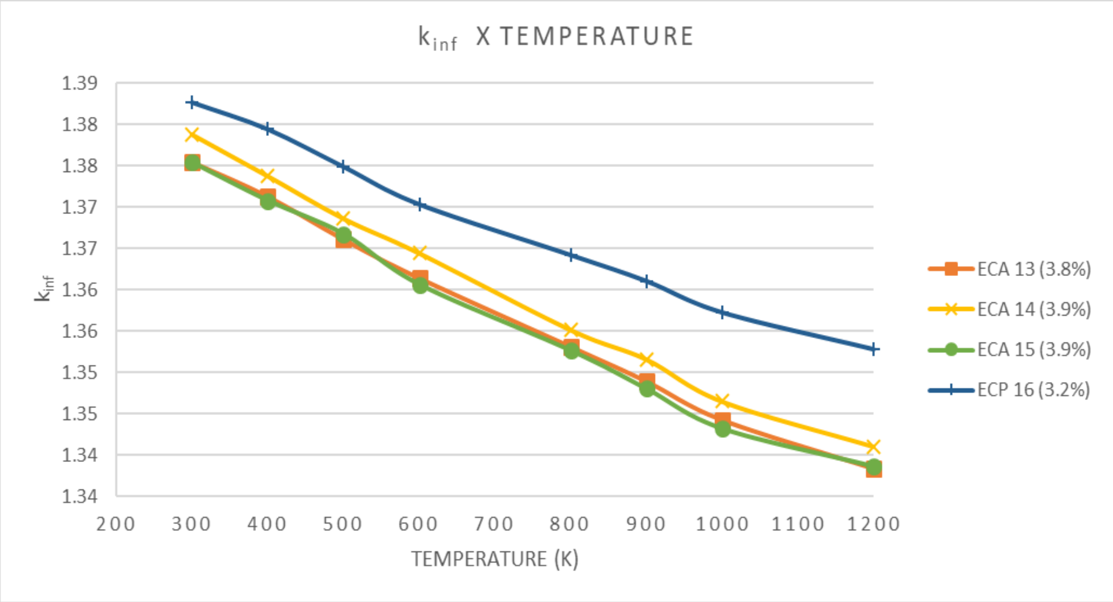

Figure 5: Relation between k_{inf} and temperature in simulation.

3.2.2 Analysis of Neutronic Flux

To compare the radial neutronic flux between the simulated assemblies worked, Fig. 6 presents the flux in central axis of all assemblies. It is observed that ECP 16 and ECA 15 have the smallest and largest values of flux analyzed respectively. Considering the annular fuel assemblies, the ECA 15 has the major flux while ECA 13 and 14 have similar shape. There is a relation between the number of rods and the flux, the highest the number of rods, the greatest level of flux obtained. This behavior is observed only among ECAs simulated, because the ECP 16 has the largest number of rods, but it has the lowest flux.

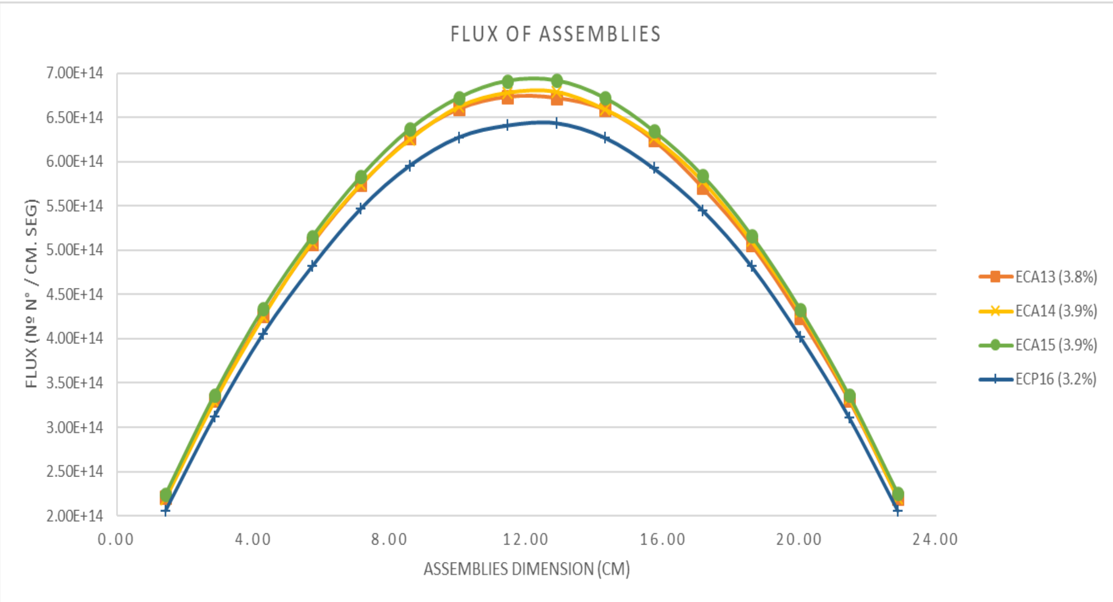

Figure 6: Relation of flux between all assemblies simulated.

In relation to the flux in fuel cells, the Fig. 7 shows how is the behavior in cells simulated. In addition to the analyses of the neutron flux in the assemblies, were also made analyses of flux in a cell of the EC on each setting. This cell corresponds to the fuel region, moderator and cladding. Figures 7, presented the behaviors pattern within a cell of ECP where the term CEL 16 refers to the cell 16, CEL 15 is relative to the cell 15 and so on. Also, this figure shows the flux profile in the central cell axis. As expected, the CEL 16 has a different profile from other settings.

While CEL 16 has a peak in the central region, the other cells have this peak in its respective fissile material zone. Such behavior is generated by the different fuel zones, which causes maximum flux values in such positions. Among the evaluated cases, the CEL16 flux has increased in the region of fissile material. In relation to the annular cells, the fuel CEL 15 and 13, present the largest and the lowest flux in the fuel region, respectively. There is an obvious decrease in flux associated with the internal cooling channel. Between the annular rods, the CEL 13 has greater internal diameter and this feature changes the fission and moderation parameters.

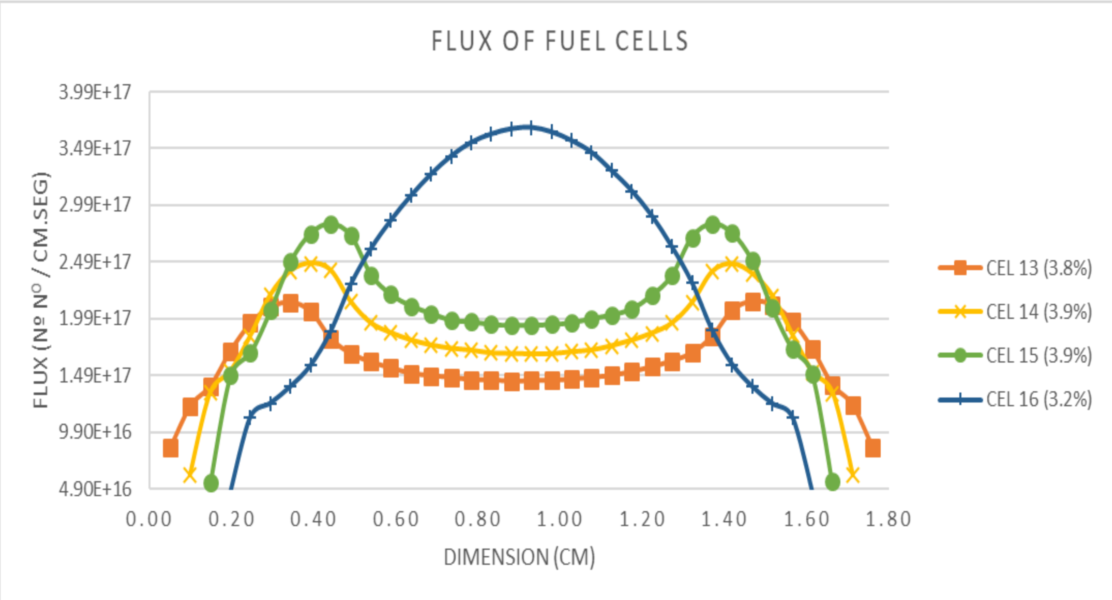

Figure 7: Relation of flux between all fuel cells simulated.

3. CONCLUSIONS

Through all procedure carried out in this research it is concluded, considering the parameters evaluated advantages from the cases studied, the ECA 13 simulated was selected to continue future researchers. Among the most representative parameters to achieve this propose, it can be cited the following feature for the annular fuel assembly simulated. The flattening in the curve of flux related to the assembly and the lower flux in the region related to fissile material.

However, the ECA(s) 14 and 15 have simulated in this research presented similar neutronic behavior to the ECA 13. Such characteristics related to the it, determine these annular fuel assemblies could be studied to more neutronic simulations.

Despite the ECA(s) are the most suitable for the an alternative PWR assemblies, other parameters such as temperature coefficient and neutronics calculations should be used aiming to complement the analysis. Other codes also will contribute to the comparison of the parameters studied, with the objective of greater reliability. Finally, the studies must be done considering not only fuel assemblies but the PWR reactor core.

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