



Assessment of the IEA-R1 nuclear reactor using a nonstandard fuel assembly with six fuel plates of the Brazilian Multipurpose Reactor

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ABSTRACT

In order to qualify the fuel plates of the Brazilian Multipurpose Reactor (RMB), a nonstandard Instrumented Fuel Assembly (IFA) was designed and is being constructed to be burned in the IEA-R1 nuclear research reactor. IFA has fuel plates of different uranium densities (10 fixed fuel plates of 3.0 gU/cm^3 – IEA-R1 standard; 6 removable fuel plates of 3.7 gU/cm^3 – RMB; and a central aluminum plate). This paper is the first step to demonstrate that IEA-R1 can safely operate with this IFA. To verify the IFA thermal behavior inside the IEA-R1 core during reactor operation and certify the no power peaks occurrence, the power distribution was calculated for each fuel plate. LEOPARD and HAMMER-TECHNION codes were utilized to calculate the core thermal neutron cross section and CITATION code to calculate the core power distribution. Calculations were performed for 5 MW reactor power considering the IFA placed in a core peripheral position. The RMB fuel plates average power was 4.73 % higher compared to IEA-R1 fuel plates. This was expected due to the higher density of uranium in these plates. The power of each IFA fuel plate was compared with a fresh IEA-R1 Fuel Assembly (FA) at the same core position. The power in the IFA hottest plate is only 6.79 % higher than the correspondent IEA-R1 fuel plate. The IFA power distribution was also compared to the hottest FA of the core. The power of each IFA fuel plate was below its correspondent hottest FA fuel plate. In addition, the total IFA power is 18.40 % less than the hottest FA in the core. No significant power peaks occur in the IFA during operation. As future works, thermal-hydraulic calculations will be performed considering this calculated power distribution and no hot spots are expected.

1. Introduction

Since 1988, IPEN (Nuclear and Energy Research Institute) located in São Paulo, Brazil, has been producing and qualifying LEU (Low Enriched Uranium – 19.9 % of ^{235}U) Fuel Assemblies (FAs), for the IEA-R1 research reactor. The first IPEN FAs were produced using $\text{U}_3\text{O}_8\text{-Al}$ dispersion with uranium densities of 1.9 gU/cm^3 in the fuel plates (from 1988 to 1996), and 2.3 gU/cm^3 (from 1996 to 1999). Since September 1999, IPEN FAs have been produced with $\text{U}_3\text{Si}_2\text{-Al}$ dispersion and uranium density of 3.0 gU/cm^3 (Maiorino, 1999). IPEN produces and consumes its own nuclear fuel and does not third party with its nuclear fuel (Negro et al., 2018). In addition to produce its own nuclear fuel, IPEN produce uranium metal annular targets. The targets needed to produce Mo-99 at the Brazilian Multipurpose Reactor (RMB) are planned to be produced in Brazil (Durazzo et al., 2021; 2022). Durazzo et al. (2017) describe the uranium metal fabrication at IPEN. The availability

of uranium metal in Brazil encouraged the development of the annular foil target. The uranium silicide dispersion fuel ($\text{U}_3\text{Si}_2\text{-Al}$) is currently recognized as the best performance fuel for research reactors with a fuel loading up to 4.8 gU/cm^3 (Gan et al., 2011).

Fuel performance evaluation and nuclear fuel qualification require post-irradiation assessment (Terrani et al., 2020; IAEA, 2009). Silva et al. (2008) described that “IPEN has no hot cells to provide destructive analysis of the irradiated nuclear fuel. As a consequence, non-destructive methods have been utilized to evaluate irradiation performance of the fuel elements. A complete fuel element is irradiated in the IEA-R1 core and the fuel element evaluation consists of two items: (i) monitoring the fuel performance during the IEA-R1 operation, concerning the following parameters: reactor power, time of operation, neutron flux at the position of each fuel assembly, burnup, inlet and outlet water temperatures in core, water pH, water conductivity, chloride content in water, and radiochemistry analysis of reactor water; and (ii) periodic underwater visual inspection of fuel assemblies and

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Nomenclature

AP	Absorber Plate
CFA	Control Fuel Assembly
CFD	Computational Fluid Dynamics
IEA-R1	IPEN Research Reactor Facility – 5 MW power
IFA	Instrumented Fuel Assembly
FA	Fuel Assembly
K_{eff}	Effective neutron multiplication factor
IPEN	Nuclear and Energy Research Institute
IPEN/MB-01	IPEN Research Reactor Facility – zero power
LEU	Low Enriched Uranium
MCNP	Monte Carlo N-Particle
MTR	Materials Testing Reactor
RMB	Brazilian Multipurpose Reactor
SPND	Self Powered Neutron Detector

eventual sipping tests for fuel element suspect of leakage. Irradiated fuel elements have been visually inspected periodically by an underwater radiation-resistant camera inside the IEA-R1 reactor pool, to verify its integrity and its general plate surface conditions.” No information can be obtained about the conditions of the internal fuel plates.

Regarding the fluid–structure interaction on the FA fuel plates, a test section was designed and built to verify the critical flow velocity (Castro, 2017). Experiments were performed and results compared to theoretical Miller correlation (Miller, 1958). It was achieved an average fluid velocity of 14.5 m/s, resulting in plates instability and consequent plastic deformation. Comparatively, the value obtained for critical velocity in the test section through the experiments was of the order of 85 % of the value obtained by calculation with Miller’s theoretical expression, (Castro, 2017; Andrade et al., 2022).

Some numerical analysis using the CFD code ANSYS CFX (ANSYS Inc., 2017) were conducted to analyze the fluid–structure interaction of a single fuel plate bounded by two coolant channels (Mantecón and

Mattar Neto, 2018; 2019a; b). These numerical results are qualitatively consistent with theoretical expectations and experimental investigations and have allowed gaining understandings of coupled problems in fuel elements (Mantecón and Mattar Neto, 2019c).

Additionally, an experimental program on the hydrodynamic behavior of fuel assemblies was proposed as part of qualification process of the RMB FA (Torres et al., 2020).

In order to verify and compare calculated and experimental neutronic parameters of the RMB core, a similar RMB core was assembled in the IPEN/MB-01 research reactor facility. The IPEN/MB-01, located in IPEN, is a zero-power research reactor (max. power 100 W) and its core has a 4×5 array, with 19 fuel assemblies of U_3Si_2-Al , with 2.8 gU/cm^3 and 19.75 % enriched in U-235, and one aluminum block (internal irradiation position) (Santos et al., 2019).

The development of Instrumented Fuel Assemblies (IFAs) showed up as an important issue in order to obtain experimental data (Torres et al., 2003; Hainoun et al., 2014; Umbehaun, 2016; Maprelian, 2018; Maprelian et al., 2020; IAEA, 2019). To qualify the RMB fuel plates under irradiation, an IFA was designed and is under construction in IPEN. RMB FAs have near dimensions and geometry of IEA-R1 FAs, but they are not identical and therefore an actual RMB FA does not fit in the IEA-R1 core. Hence, the IFA has the main characteristics and external dimensions of a standard IEA-R1 FA. It has 16 fuel plates total, 10 of which are fixed standard IEA-R1 fuel plates with uranium density of 3.0 gU/cm^3 , six are removable RMB fuel plates of 3.7 gU/cm^3 of uranium density and a central aluminum plate. The two central fuel plates were replaced by a solid thick aluminum plate, which has two holes that cross axially the IFA for passage of a thermocouple and a SPND (Self Powered Neutron Detector). After a specified burning-up period, the removable fuel plates will be individually tested by gamma spectrometry (Terremoto et al., 2000), thickness measurements and visual inspections (Silva et al., 2008). The objective of the main research is to qualify the RMB FAs, and the objective of this work is to calculate the power distribution in the IFA and to verify if no power peaks will occur during its operation inside the core.

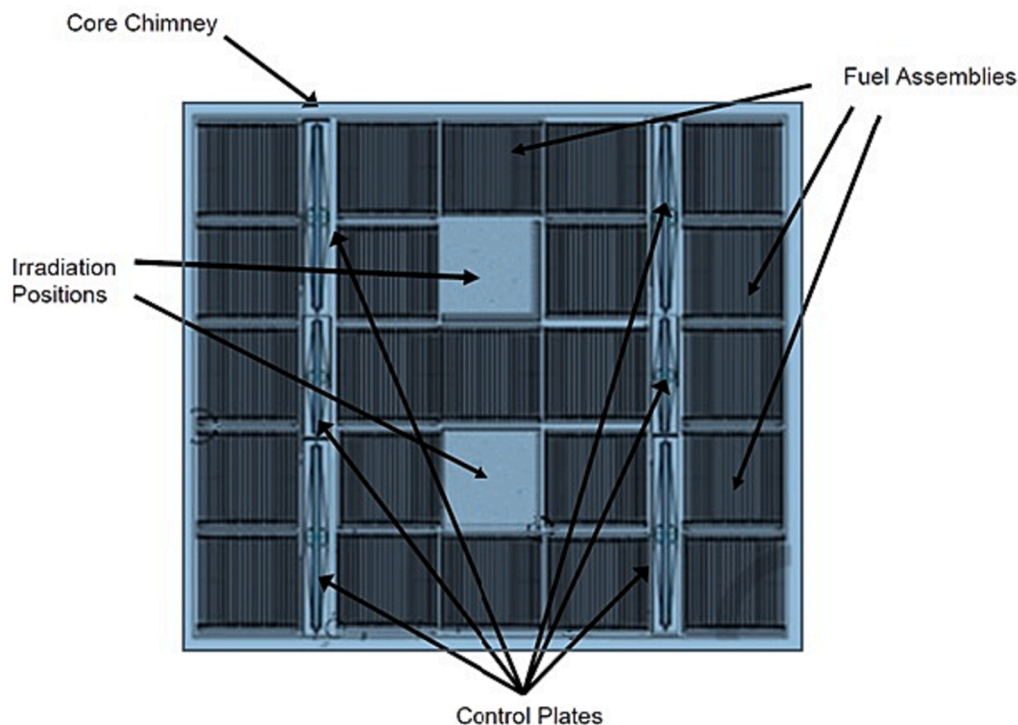


Fig. 1. RMB reactor core (CNEN, 2017).

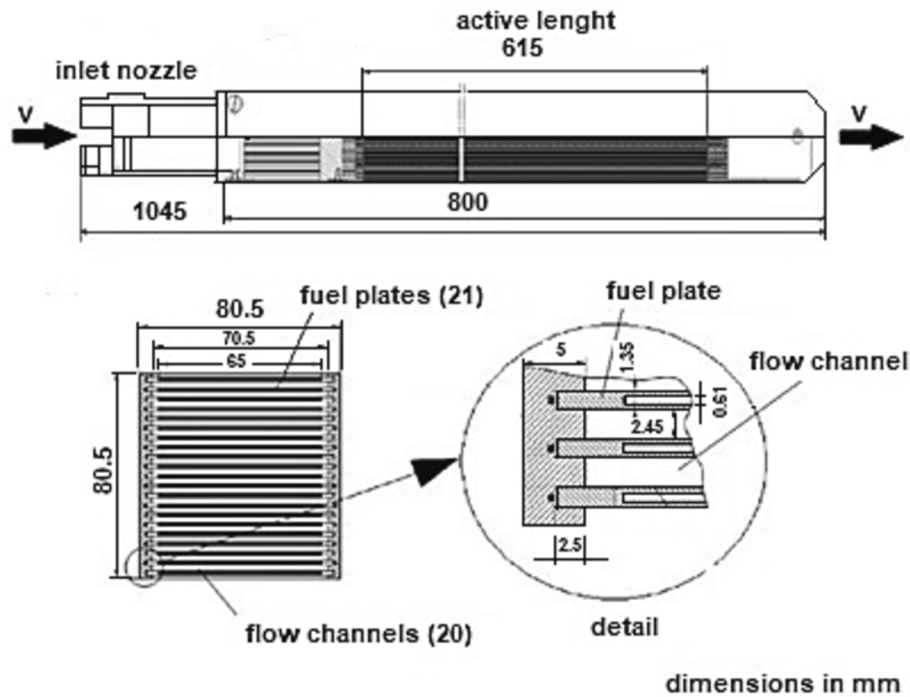


Fig. 2. RMB fuel assembly (CNEN, 2017).

2. RMB and IEA-R1 core description

2.1. Brazilian multipurpose reactor (RMB)

The Brazilian Multipurpose Reactor (RMB) represents a significant advancement in Brazilian nuclear research and technological capabilities. The facility will be part of a new nuclear research center to be built at Iperó city, about 100 km from São Paulo city in the southeast part of Brazil. Obadia and Perrotta (2010) provide information about the new research center, which will have several facilities. The RMB is a multipurpose research reactor designed for various functions, including the radioisotopes production for medical and industrial applications, materials testing, neutron activation analysis, and fundamental research in nuclear science and technology. It is a 30 MW open pool type reactor, which uses MTR (Materials Testing Reactor) fuel assemblies, light water as moderator and coolant fluid, and heavy water as reflector. The reactor core is assembled in a 5x5 array, with 23 FAs, 2 positions for material irradiation and tests, and 6 control plates. Overall external dimensions of the RMB FA are: 80.5 mm x 80.5 mm x 1045 mm. Each FA has 21 fuel plates, 1.35 mm thick, 75 mm width and 655 mm length, forming 20 internal channels of 2.45 mm thick and 70.5 width. The meat (0.61 mm x 65 mm x 615 mm) has 3.7 gU/cm³ of LEU (19.75 %) in the form of uranium silicide and aluminum dispersion (U₃Si₂-Al) clad with aluminum (Perrotta and Soares, 2014). Perrotta et al., (2019a) describe the RMB fuel cycle management and Perrotta et al., (2019b) present an update on the nowadays overall development status of the RMB project.

Figs. 1 and 2 show simplified drawings of the RMB core and its fuel assembly, respectively.

2.2. IEA-R1

The IEA-R1 is an open pool type research reactor. It is a light water cooled and moderated that uses MTR fuel elements. The reactor is located at IPEN in São Paulo and was designed and built by Babcock and Wilcox Co. in 1957. IEA-R1 research reactor is a multipurpose research reactor. It has been used for basic and applied researches, training and mainly for radioisotope production for applications in medicine.

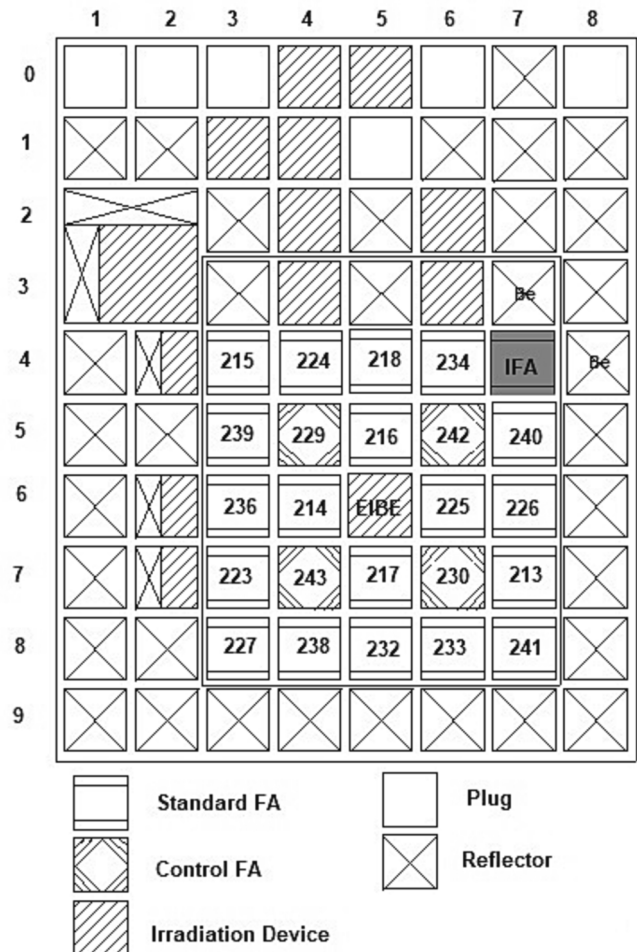


Fig. 3. IEA-R1 reactor core with emphasis on the IFA position (IPEN, 2017).

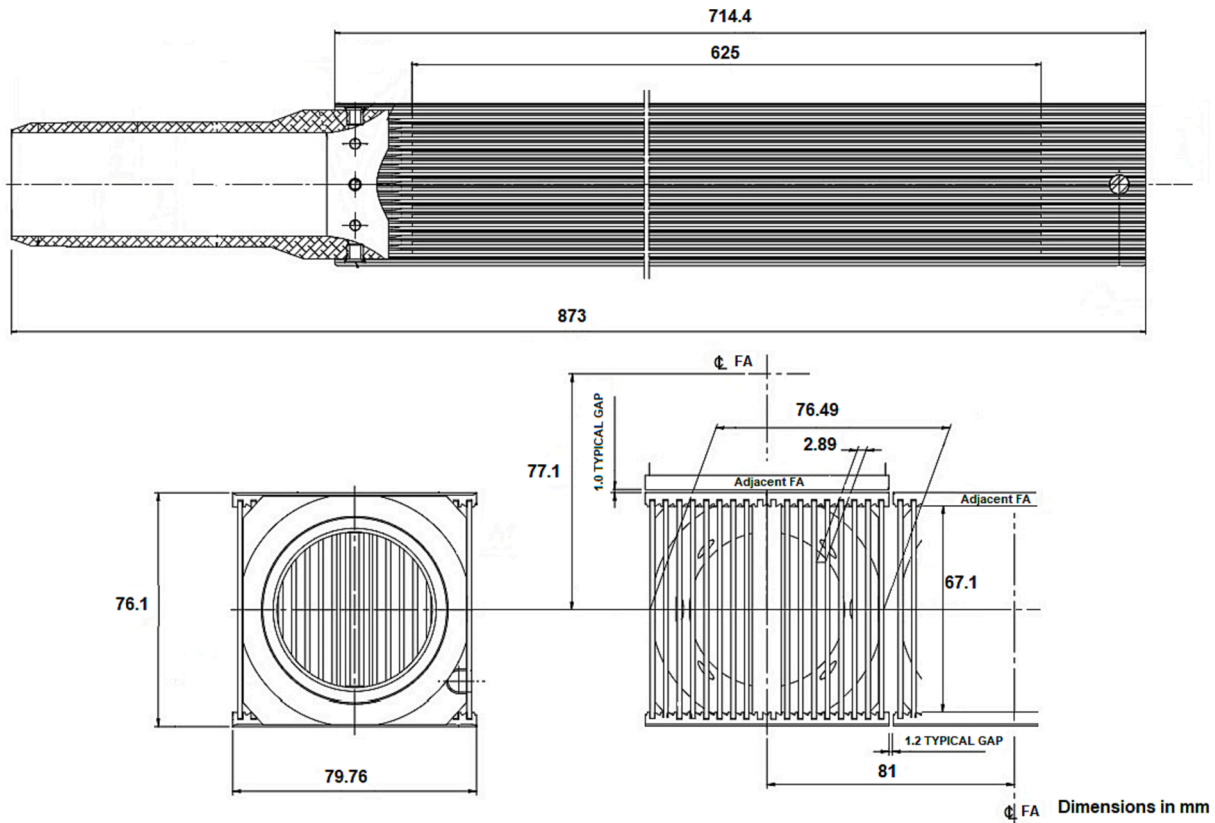


Fig. 4. IEA-R1 fuel assembly (IPEN, 2017).

Table 1
IEA-R1 and RMB main parameters.

Parameter	IEA-R1	RMB
Reactor power (MW)	5	30
Core array	5x5	5x5
Fuel assemblies	20	23
Control fuel assemblies	4	-
In-core irradiation position	1	2
Control plates	4 couples	6
FA fuel plates	18	21
Fuel plate enrichment (U^{235} %)	19.75	19.75
Uranium density (gU/cm^3)	3.0	3.7
Fuel plate dimensions (mm) (thickness x width x length)	1.52 x 70.75 x 625	1.35 x 75 x 655
Fuel plate meat (mm) (thickness x width x length)	0.76 x 60.75 x 600	0.61 x 65 x 615
FA internal channels	17	20
FA internal channels (mm) (thickness x width)	2.89 x 67.1	2.45 x 70.5

Genezini et al. (2019) details about IEA-R1 history in more than 60 years of operation.

Figs. 3 and 4 show simplified drawings of the IEA-R1 core and its fuel assembly, respectively. The IEA-R1 core is assembled on a 10x8 matrix plate. The core has Fuel Assemblies (FAs), Control Fuel Assemblies (CFAs), irradiation devices, graphite and beryllium reflectors and plugs. The active core is assembled on the matrix plate in a 5x5 array with 20 FAs, 4 CFAs and a central beryllium irradiation device (EIBE). The CFAs are numbered in the current configuration core as: 243, 230, 242 and 229, in which a couple of neutron Absorber Plate (AP: 1, 2, 3 and 4, respectively) are inserted from the top in order to control the core reactivity. These APs move into the CFAs and each CFAs have 12 fuel plates, 6 fuel plates less than a standard IEA-R1 FA. Overall external dimensions of the IEA-R1 FA are: 79.76 mm x 76.1 mm x 873 mm. Each FA has 18 fuel plates, 1.52 mm thick, 70.75 mm width and 625 mm

length, forming 17 internal channels of 2.89 mm thick and 67.1 mm width. The meat (0.76 mm x 60.75 mm x 600 mm) has 3.0 gU/cm^3 of low enriched uranium (19.75 %) in the form of U_3Si_2-Al dispersion clad with aluminum. Table 1 shows a comparison between IEA-R1 and RMB main parameters.

3. Neutronic codes description

3.1. Summary

The neutronic calculation methodology used to follow up the operation of the IEA-R1 reactor core is based on the following codes: LEOPARD (Barry, 1963; Kerr et al., 1991), and HAMMER-TECHNION (Barhen et al., 1978) to calculate the cross sections. TWODB or 2DB (Little and Hardie, 1969) is a two-dimensional multigroup diffusion code used to calculate the effective neutron multiplication factor (K_{eff}), fissile material consumption, neutron flux, burnup and power density. 2DB cross sections results are used as input data for CITATION, which has a three-dimensional model, necessary to evaluate the insertion of the control rods/plates in the core. The control rods are used for power maneuvers; to control and shutdown the reactor. CITATION calculates the effective multiplication factor K_{eff} , maximum neutron flux, as well as maximum power density in the core.

3.2. Lattice physics codes

Lattice physics codes are used to generate cross section data for nodal codes, analyzing axial segments of fuel assemblies, to determine the detailed spatial and spectral distribution of neutrons and photons across the segment (Knott and Yamamoto, 2010).

The LEOPARD code, in the original version (Barry, 1963), was applied only for square and hexagonal cell geometry, suitable for LWR reactors with slightly enriched fuel. The code was modified at the

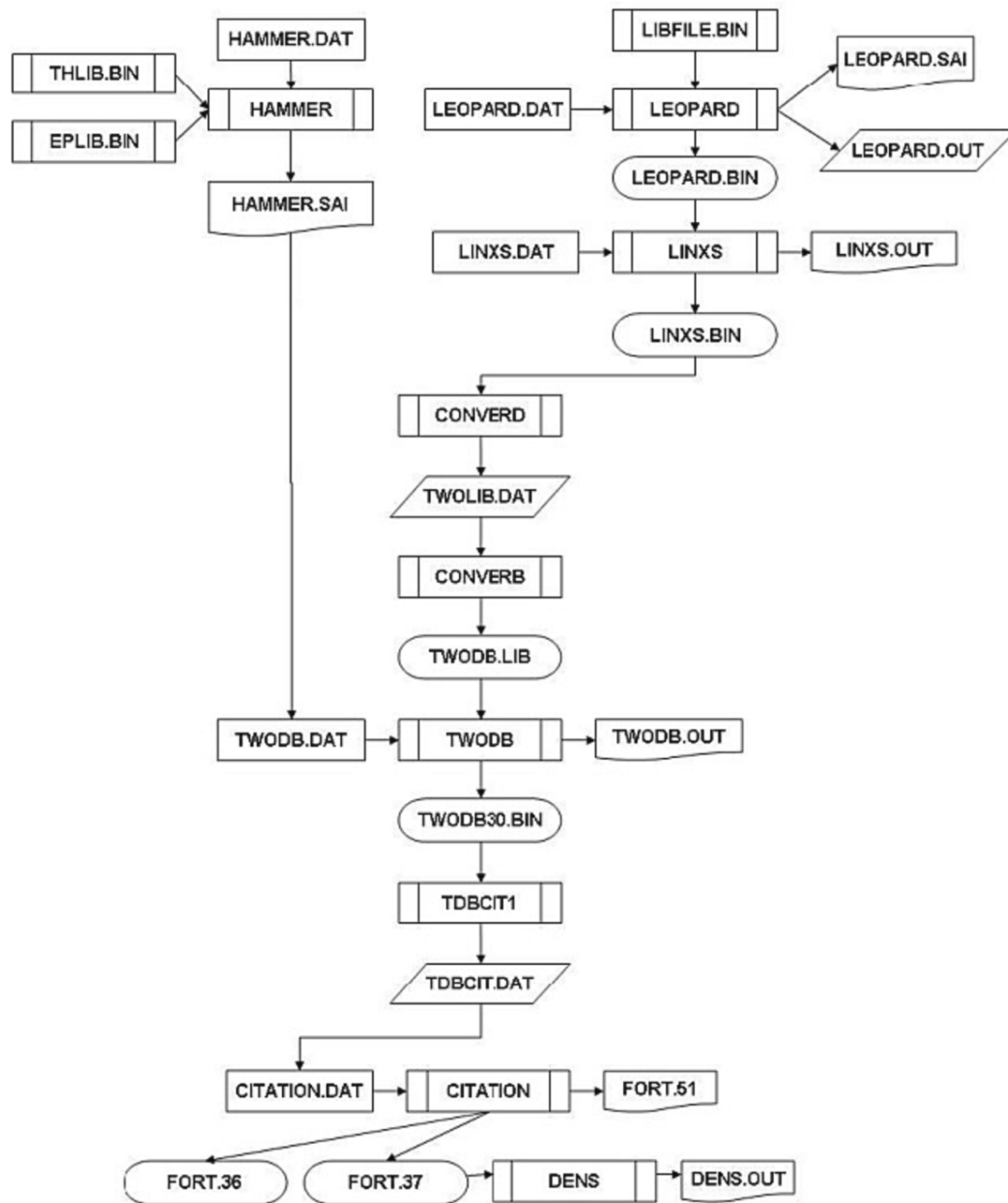


Fig. 5. Neutronic calculation flowchart (Yamaguchi, 1997).

University of Michigan, where the option of plate-type (slab) geometry was added. Plate-type geometry is common in MTR-type research reactors. The modifications are described in (Kerr et al., 1991). The LEOPARD calculated the cross sections of all combustible regions and the HAMMER-TECHNION code calculated the cross sections for non-combustible region. HAMMER-TECHNION was used to calculate the reflectors (graphite and beryllium elements) and the thick aluminum plate (cell composed by: water + aluminum) of the IFA. All calculated cross sections feed the 2DB code input. LEOPARD and HAMMER-TECHNION codes use ENDF/B-II and ENDF/B-IV data libraries, respectively.

3.3. CITATION code

Although the CITATION (Fowler et al., 1972) is quite old, it has still been used in nuclear power and research reactor core calculations, along

with other codes that calculate neutronic parameters. Santos et al. (2011) used experimental data from IPEN/MB-01 as a benchmark in order to qualify reactor codes such as CITATION. Amin et al. (2014) and Benaalilou et al. (2019) compared the CITATION with Monte Carlo N-Particle (MCNP). Both calculations showed good agreement and accuracy, proving that CITATION is a good and efficient calculation tool. Rahmani et al. (2013) had good agreement between the Safety Analysis Report and CITATION calculation for the effective multiplication factor and the reactor thermal power three-dimensional distribution. Shirazi (2021) used the CITATION results as a benchmark to validate a Monte Carlo method to calculate the neutron flux in a spherical reactor core.

The CITATION is designed to solve multigroup neutronics problems with application of the finite-difference diffusion theory approximation to neutron transport in up to three-dimensional geometry (Fowler et al., 1972).

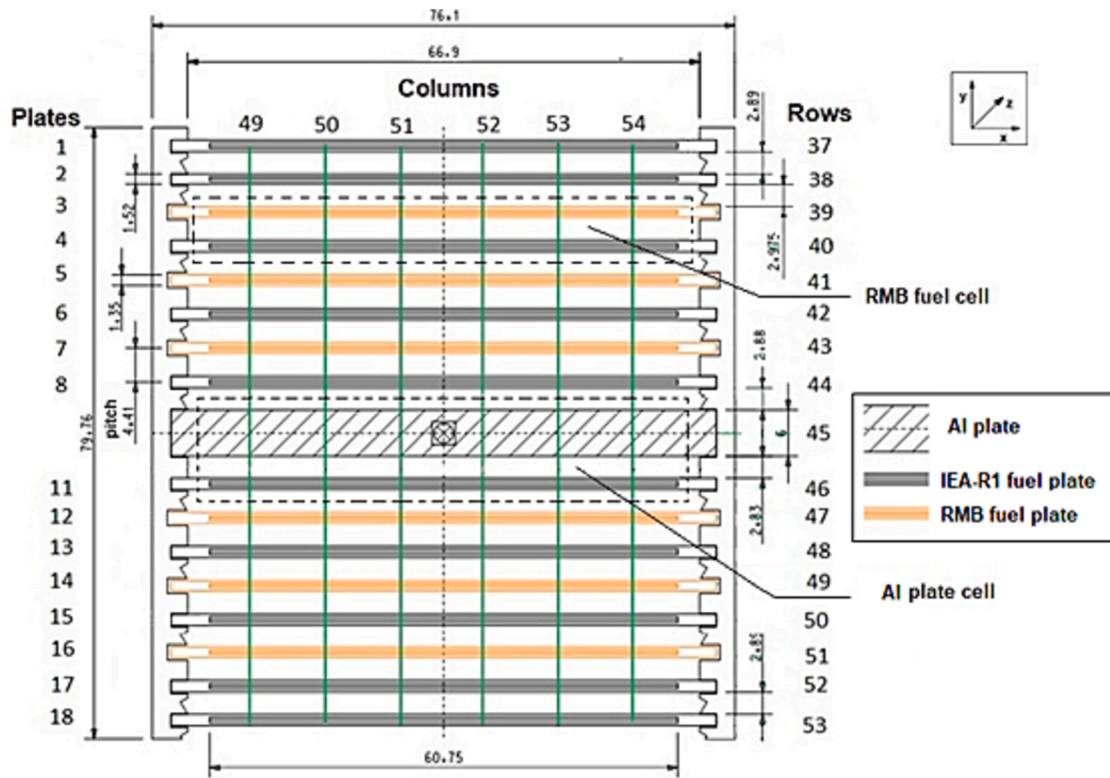
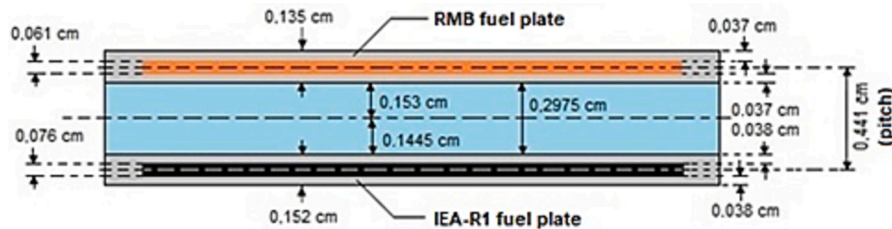


Fig. 6. Instrumented Fuel Assembly – IFA with xy Citation discretization model. (Soares, 2023).

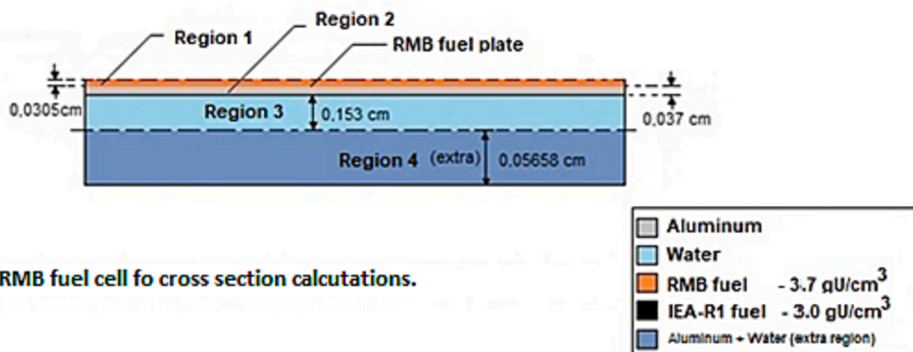
3.4. IPEN neutronic calculation methodology description

Fig. 5 shows the neutronic calculation flowchart. This methodology was developed by IPEN and has been used for a long time for analysis in the IEA-R1 reactor. Stefani and Conti (2017) developed a single management code that integrated the nuclear reactor codes in order to automatically apply this methodology mentioned above to IEA-R1.

The calculation of fuel cross sections is done with the LEOPARD program, using the standard cell model (fuel, cladding and moderator) with an extra region to homogenize the fuel cell. Data of cells are provided in the LEOPARD.DAT file and the output files LEOPARD.SAI and LEOPARD.OUT are generated. The neutron cross sections are generated with an interval approximate of 30 burning days until reaching 50 % burnup of U^{235} . They are recorded in the binary file LEOPARD.BIN. The



(a) IFA details dimensions: RMB and IEA-R1 fuel plates.



(b) RMB fuel cell fo cross section calculations.

Fig. 7. IFA details dimensions considered for LEOPARD code (Yamaguchi and Torres, 2023).

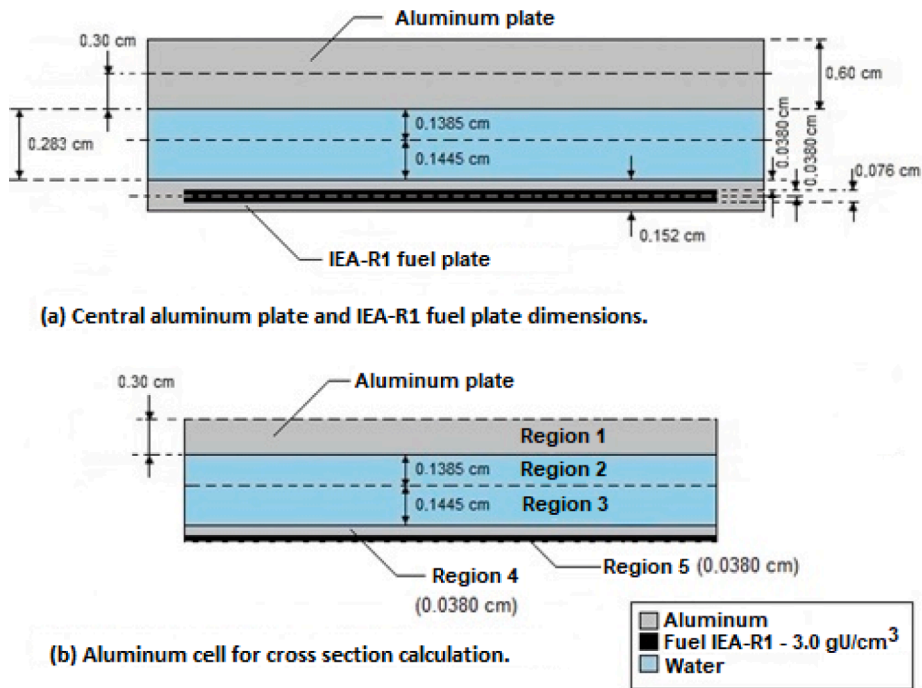


Fig. 8. Aluminum plate dimensions considered for HAMMER-TECHNION code (Yamaguchi and Torres, 2023).

LINXS program converts this file into the binary file LINXS.BIN. The number of neutron energy groups is provided in the LINXS.DAT input file. The readable formatted file TWOLIB.DAT is generated by the CONVERD program. This last step is necessary when there are several types of fuel cells. The CONVERB program is used to convert the TWOLIB.DAT into the binary file TWODB.LIB. This binary file is a 2DB code neutron cross section input file (Yamaguchi and Torres, 2023). The LEOPARD and 2DB codes are part of the same system pack, so, the LEOPARD output file for combustible region is written in the adequate format for the 2DB.

For non-combustible region, HAMMER-TECHNION code performed cross section calculations, which are also used as input data for TWODB code, as showed in Fig. 5.

The 2DB code uses the TWODB.LIB and the TWODB.DAT as input files and, the calculated results are stored in the TWODB.OUT and TWODB30.BIN output files. The TDBCIT1 program is used to convert the binary file TWODB30.BIN in readable file TDBCIT.DAT. This file supplies the neutron cross section used in the CITATION input data (CITATION.DAT). After the running CITATION, Fort.36, Fort.37 and Fort.51 files are generate as output. Fort.51 is the main readable output file. The Fort.36 supply the neutron flux in the reactor core and the Fort.37 supply the core power density distribution. The DENS program is used to calculate the power density in some specific places defined by user converting the Fort.37 binary file in a readable file DENS.OUT.

4. IFA model description

The IFA is being built considering the main characteristics, external dimensions and structures of a standard IEA-R1 FA, similar as shown in Fig. 4. The amount of uranium in the removable fuel plates was recalculated since the width and the length are different from RMB standard fuel plates. The meat and cladding thicknesses were maintained for removable RMB fuel plates. Due to the difference of the adjacent fuel plate thicknesses, the flow channel increased about 3 %, which does not alter significantly the IFA thermo-hydraulic. Fig. 6 shows a simplified design of IFA, RMB fuel cell and aluminum plate cell used for neutronic cross section calculation. The columns and rows in the Fig. 6 represent the Citation model discretization.

Table 2

IFA neutronic model (Soares, 2023).

IFA dimensions	Aluminum plate	No. meshes	Mesh
Direction (cm)	7.709	-	6
Y Direction (cm)	8.100	1.044	17
Z Direction (cm)	60.000	-	30

Table 3

Main characteristics of the IEA-R1 and RMB fuel plates.

Plate	Dens. U (g/cm ³)	Mass U (g)	Enrich. U ²³⁵ (%)	Mass U ²³⁵ (g)	Dimensions of the meat (cm)		
					Thickness	Width	Length
IEA-R1	3.0	84.02	19.65*	16.51	0.076	6.035	60.0
RMB	3.7	83.17	19.65*	16.26	0.061	6.035	60.0

*19.65 % - Production batch enrichment (Silva, 2022).

Figs. 7 and 8 show the IFA geometric details and its lattice physics codes cells for cross sections calculations.

Table 2 shows the IFA modeling in both codes, 2DB (x and y directions) and CITATION (x, y and z directions). All cells are homogenized (water + aluminum + fuel) with the same pitch, except for the central aluminum plate.

Each fuel plate was divided into 6 meshes (x direction) of the same length, represented by columns numbered 49 to 54. Also, each fuel plate was divided into 17 lines (y direction) represented by rows 37 to 53 (y direction), as shown in Fig. 6. The IFA modeling carried out in both the 2DB and CITATIONs consider each fuel plate individually as a cell. The introduction of RMB fuel plates into an IEA-R1 FA causes small geometric changes in the hydraulic channels, as the RMB plates are slightly smaller (1.35 mm) than the IEA-R1 plates (1.52 mm). Therefore, the adjacent channels have 3 % larger flow areas.

Meshes size less than 2 cm have been adequate for CITATION modeling for this type of the IEA-R1 fuel assemblies. The mesh size is correlated with mean free path (λ), which is the average distance that a neutron moves between collisions. According to (Lamarsh and Baratta,

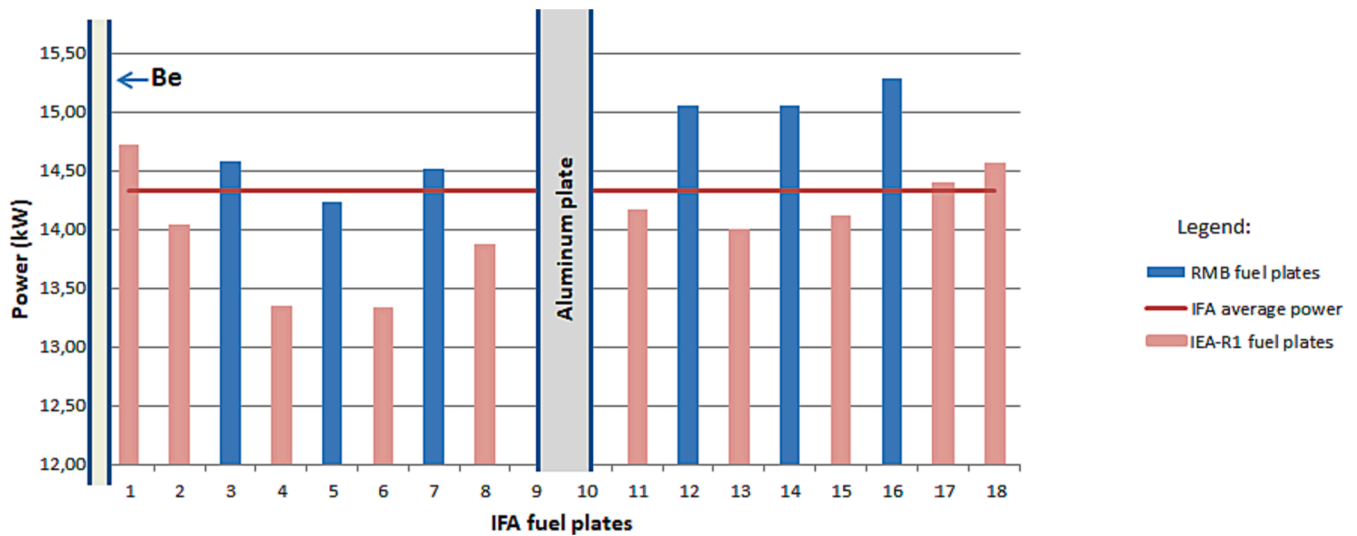


Fig. 9. Fuel plates power of IFA (Soares, 2023).

Table 4
Power of IFA fuel plates (Soares, 2023).

Plate	1	2	3	4	5	6	7	8	9
Type	IEA-R1	IEA-R1	RMB	IEA-R1	RMB	IEA-R1	RMB	IEA-R1	Al
(kW)	14.73	14.04	14.58	13.35	14.24	13.33	14.52	13.87	0

Plate	10	11	12	13	14	15	16	17	18
Type	Al	IEA-R1	RMB	IEA-R1	RMB	IEA-R1	RMB	IEA-R1	IEA-R1
(kW)	0	14.16	15.06	14.00	15.06	14.12	15.29	14.40	14.57

2001), the $\lambda = 1.35$ cm for UO_2 .

Table 3 shows the main characteristics of the IEA-R1 and RMB fuel plates. The uranium enrichment of this production batch for the IFA was 19.65 %, slightly less than the nominal value of 19.75 %. Although the uranium density of the RMB fuel plate is greater than that of IEA-R1, the thickness of the meat is smaller, so the quantity of U^{235} in the plates is very close (Table 3). The IEA-R1 plates have 1.54 % more U^{235} compared to the RMB plates (Silva, 2022).

Calculations were based on IFA position as showed in Fig. 3 for 5 MW reactor power. The IFA occupies a position at the periphery of the core, where it is close to two fuel assemblies (FA-234 and FA-240) and two beryllium reflectors. The introduction of a fresh FA inside the core usually occurs in a peripheral position, to decrease local power peaks and avoid high reactivity insertion. The IFA was modeled in the 2DB code as a brand new FA and the other fuel assemblies followed the IEA-R1 current burnup.

Table 5
IFA fuel plates power per groups (Soares, 2023).

Plates	N° of plates/group	Power density/group (W/cm ³)	Average power/group (kW)	(%) Power /group	Average power/plate (kW)	(%) Power/plate	(%) Difference power/plate
IEA-R1	10	62.0	140.6	61.30	14.1	6.13	4.73
RMB	6	65.3	88.7	38.70	14.8	6.45	-
Total Power - IFA			229.3	100.0	-	-	-

5. Results and discussion

5.1. IFA power distribution

Fig. 9 shows the power for each IFA fuel plate. The CITATION calculations were carried out considering control plates near the critical position. In this work, two neutron energy groups (thermal and fast) were used in CITATION code. The convergence criteria used in this work are the default values of the CITATION code: a) neutron flux = 0.0001 and b) k-effective = 0.00001. Each calculated mesh power densities, stored in a CITATION output file, were multiplied by correspondent mesh volume to generate the mesh power distribution. This mesh power distribution was summed along the length of IFA plates. Fig. 9 compares the power in (kW) among the IEA-R1 and RMB IFA fuel plates. Furthermore, the power of the RMB plates is higher than those IEA-R1 adjacent plates, as expected. In general, the RMB fuel plates presented higher power than the IFA average power, except for fuel plate 5. The difference about the power performed by fuel plate 5 and the IFA average power is very small, 14.24 kW and 14.30 kW, respectively. The IFA power of RMB plates is higher than the IFA IEA-R1 ones due to their higher uranium density (3.7 gU/cm³).

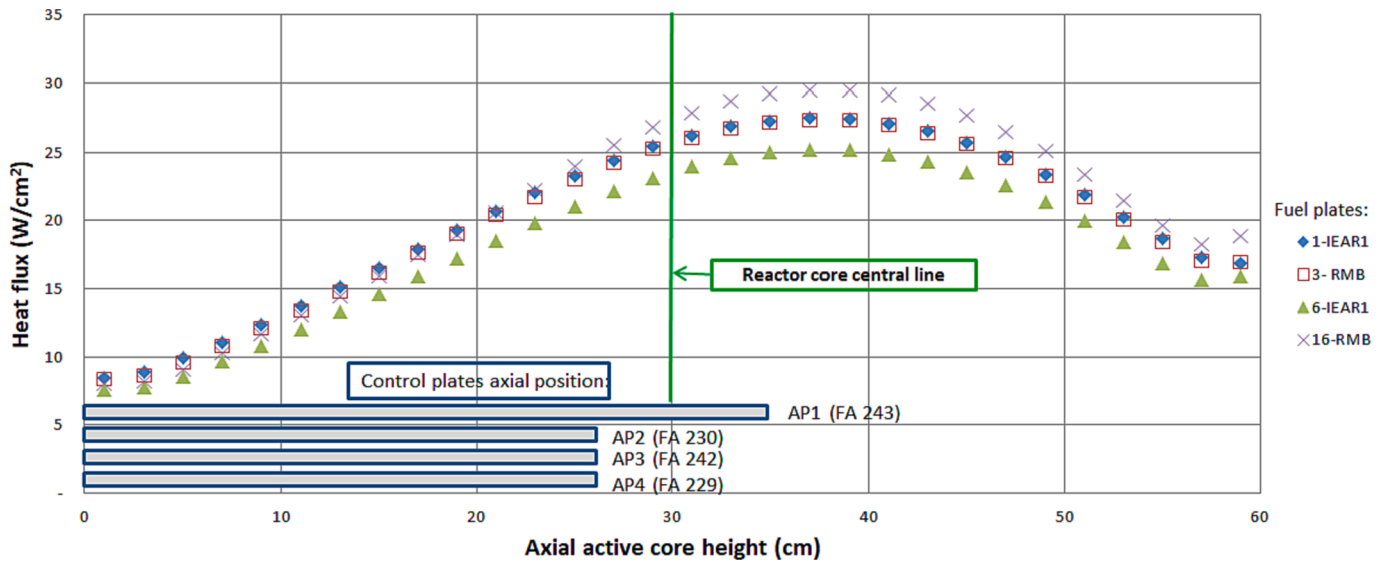


Fig. 10. Axial heat flux profile for 1, 3, 6 and 16 IFA fuel plates.

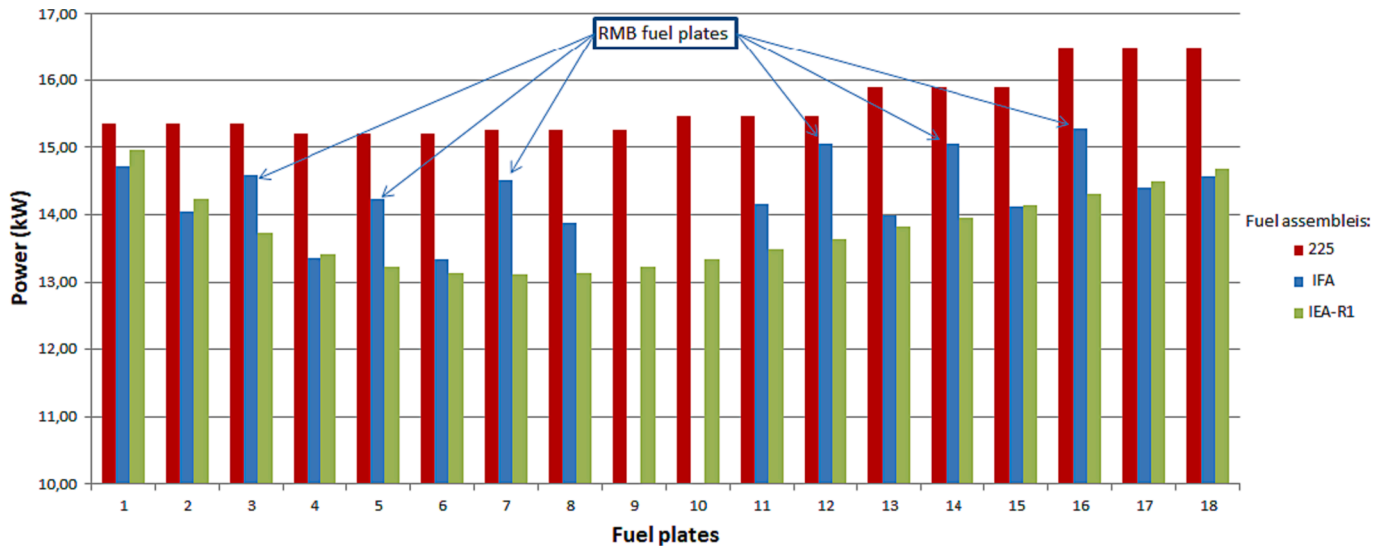


Fig. 11. Comparative power distribution among the FA: IFA, IEA-R1 and 225.

Table 4 shows the individual power of each IFA fuel plate. Plate 1 presented the highest power among the IFA IEA-R1 plates. This is due to the position of this plate close to a beryllium element, see Fig. 3, which acts as a neutron reflector, contributing to more fissions and consequently more power. According to Table 4, the greatest difference in power among IFA adjacent fuel plates occurs in plates 3 and 4. The difference is 1.23 kW, corresponding to 9.25 % higher.

Table 5 shows the total power calculated by CITATION in the IFA and the powers by groups of RMB and IEA-R1 plates. The differences in average power per plate between IEA-R1 and RMB plates are very small and it is only 4.73 %. This difference is not an error measure, it is just a value to verify if no exceptional power peaks occurred in the RMB fuel

plates. The power limit is defined mainly by the fuel cladding temperature. The maximum cladding temperature is about 95 °C due to corrosion effects on the fuel plates (Umbehaun, 2016). Previous thermo-hydraulic analysis showed that this temperature limit is not reached.

Fig. 10 shows the axial heat flux profile for 1, 3, 6 and 16 fuel plates. It is possible to observe that no exceptional and undesired power peaks occur. In the future works, these axial heat flux profiles will be used in thermal-hydraulic analysis and no hot spots are expected. As can be seen in the Fig. 10, the power profile in the fuel plates, 1-IEAR1 and 3-RMB, showed practically the same values.

5.2. Comparison among IFA, IEA-R1 FA and the hottest FA in the reactor core

For the core configuration presented in Fig. 3, a power calculation for the FA 225 (the hottest FA in the core) and for a hypothetical fresh IEA-R1 standard FA, placed in the same position as IFA, were performed to compare with the power of IFA. Fig. 11 compares the power distribution among IFA, IEA-R1 FA and the FA-225. As can be seen in Fig. 11, all the six RMB fuel plates presented less power compared to the respective FA

Table 6

Comparative total power: IFA, IEA-R1 FA and FA 225.

FA	Average FA	IFA	IEA-R1 FA	FA 225
Pot (kW)	229.88	229.33	248.03	281.04
FA (%) / (Average FA)	100.00	99.76	107.89	122.25

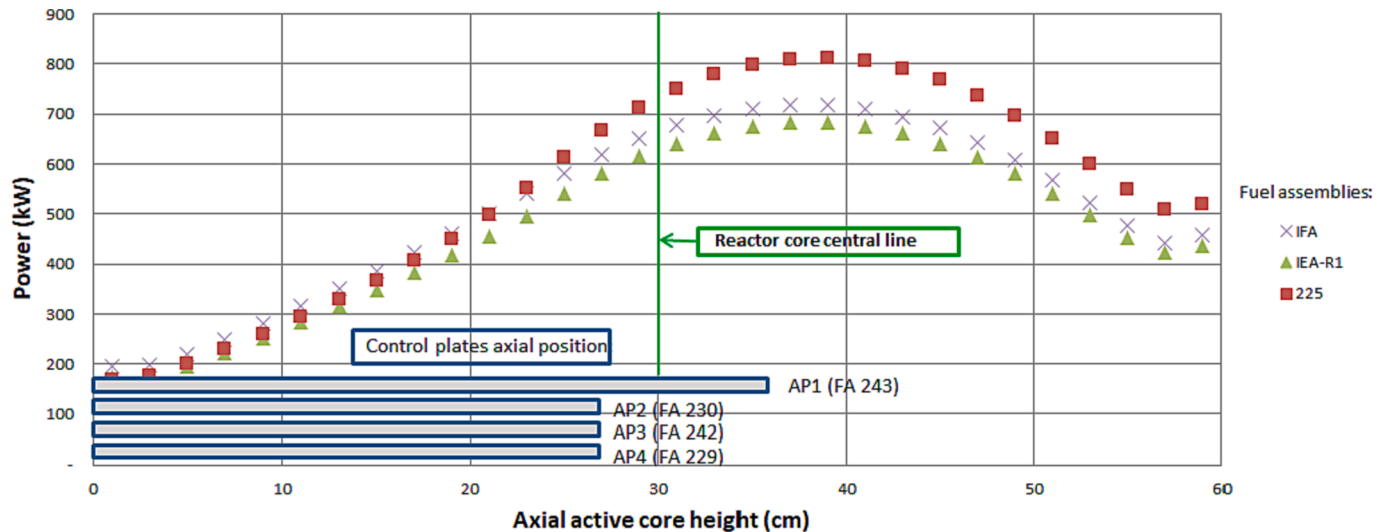


Fig. 12. Comparative axial power profile among plate 16 of IFA, IEA-R1 FA and FA-225.

225 plates. The IFA hottest plate power (16 – RMB fuel plate) is only 6.79 % higher than the IEA-R1 FA correspondent plate.

The power distribution in fuel assembly presents a standard shape, “U” form. The first fuel plates and the last ones presented more power than the fuel plates at the middle of the FA. Furthermore, the power presented by each fuel plate depends on the balance power distribution in the core and the burn up of neighbor FAs.

In terms of total power, the Table 6 shows the comparison among an IEA-R1 average power FA (100 %), IFA, IEA-R1 fresh FA and FA 225. Although the powers in RMB plates are higher than IEA-R1 plates of IFA, its total power is lower than IEA-R1 average FA, because it has 2 central fuel plates less.

Fig. 12 compares axial power profile among the IFA hottest fuel plate (16 plate) and the correspondent IEA-R1 FA and FA-225 plates. The IFA axial power profile is slightly higher than IEA-R1 FA, less than 5 % of power. On the other hand, the power peak in the FA 225 is 13.29 % higher than IFA. The axial IFA power profile presented the expected behavior.

6. Conclusions

This work presented the first steps to qualify a new MTR fuel plate for the Brazilian Multipurpose Reactor. The main conclusion of this paper is that there is no significant local power peak increase at the IFA due to the use of RMB fuel plates, and the IEA-R1 nuclear research reactor can safely operate with IFA placed at peripheral position. The power difference is relatively small; in average just 4.73 % more power compared to standard IEA-R1 plates. Additionally, the axial power peak in the FA 225 (hottest FA) is 13.29 % higher than IFA power.

The power generated by the IFA was compared with standard IEA-R1 FA and the hottest FA in the IEA-R1 core. The results showed that the total power generated by IFA was nearly average IEA-R1 FA, it was less than the fresh IEA-R1 FA and hottest FA 225 in the core. Radial and axially no power peak occurred in the IFA.

From neutronic point of view, the introduction of the IFA does not significantly affect the core power distribution, and probably will not affect the core thermal-hydraulic conditions either. In future works, thermal-hydraulic analyses using this power distribution will be carried out to confirm these conclusions.

CRedit authorship contribution statement

Humberto Vitor Soares: Data curation, Formal analysis, Investigation, Methodology, Resources, Software, Writing – original draft.

Walmir Maximo Torres: Conceptualization, Formal analysis, Supervision, Writing – original draft, Writing – review & editing, Project administration. **Pedro Ernesto Umbehaun:** Methodology, Supervision, Writing – review & editing. **Antonio Belchior Junior:** Conceptualization, Formal analysis, Methodology, Writing – review & editing. **Delvonei Alves de Andrade:** Formal analysis, Methodology, Supervision, Writing – review & editing.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

Data will be made available on request.

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