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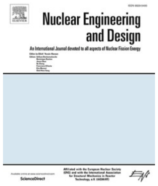

Core reduction for increasing neutron flux and radioisotope production in the IEA-R1 research reactor

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ABSTRACT

This work aims to present and analyze a new reactor core configuration for the IEA-R1 reactor core with the objective of enhancing its neutron irradiation capabilities. The reactor was modeled using the KENO-VI module from the SCALE Code System, provided by ORNL. Results have shown that implementing a near-cylindrical configuration in the IEA-R1 core could substantially increase neutron flux, particularlly in the center irradiation positions such as the EIBe (Beryllium Irradiated Element). After comparing various configurations, we conducted a brief analysis of the proposed core configuration's potential to produce ^{177}Lu, resulting in a 34% increase in the specific activity of ^{177}Lu considering a 14-day full power cycle.

1. Introduction

The increasing demand for radioisotopes for medical and industrial applications have induced reactor operators to study alternatives to increase their facilities capacity to produce radioisotopes and provide new services (Ipen, 2009; AHMED, 2007; Mokhtari et al., 2017; SAFAEI ARSHI, 2021). The alternatives for improving radioisotope production in research reactors include the utilization of high-density fuel (AHMED, 2007), placing special irradiator elements in the core to increase local epithermal and thermal neutron flux (Mokhtari et al., 2017; SAFAEI ARSHI, 2021; STEFANI, 2021; Sairanbayev et al., 2021), placing more efficient reflectors (AHMED, 2007), and changing the core geometry (SAFAEI ARSHI, 2021). In Brazil the demand for radioisotopes is growing and places significant challenges to radioisotope producers (BRASIL, 1988; SANTOS, 2023) including to the operators of the IEA-R1 research reactor, the oldest nuclear reactor in Brazil, built under the Atoms for Peace Program by Babcok & Wilcox Company and achieved its first criticality in 1957 (Ipen, 2009). It is currently the main supplier of radiochemical including all the ^{99m}Tc production (AHMED, 2007).

To address the increasing demand studies have been carried out to enhance the IEA-R1 reactor radioisotope production. Stefani et al. (STEFANI, 2021) investigated new core configurations with the Monte Carlo SERPENT code (Mokhtari et al., 2017) aiming to elevate the neutron flux at the reactor's irradiation positions. They concluded that a near cylindrical configuration could significantly increase the neutron flux in irradiation positions.

Recently, the IEA-R1 was modelled using the SCALE 6.2 (Wieselquist et al., 2020) and previous studies are being reviewed to verify the consistency of this new model. In this study, the work from Laranjo et. al. was conducted considering this new simulation model, in addition to the insertion of the burn up rates in the fuel elements.

In the Methods section we present the information employed to generate a 3D model as close as possible to the real-life reactor, and some of the techniques applied to obtain the needed information from the simulation's modules. In Validation section we have compared the output data from the standard configuration with data obtained from the actual core, such as xenon build-up, reactivity, control rod influence and thermal flux profile. In the Results section, the different configurations were compared with each other, first all of them, and further analysis were conducted only with the promising ones. New configurations, beyond the analyzed ones in the reference study, were modeled to verify the full potential of the core configuration changing, such as new

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irradiation positions and ^{177}Lu production.

2. Methods and data

The approach taken in this study involved proposing seven different configurations for the IEA-R1 reactor core, with the primary goal of enhancing the neutron flux levels in the irradiation positions. These positions were identified as: EIF, EIBRA 1 EIBRA 2, EIBe A and EIBe B. To achieve this enhancement, the approach taken was to change fuel elements configuration in the core, changing the reflector material from Graphite to Beryllium, and changing the position of irradiator positions in the core. The reactor core's thermal power level remained at 4.5 MW.

The fuel and moderator temperature coefficients of reactivity were calculated since significant changes in the core were implemented. Furthermore, the fuel cycle length was also verified to account for potential alterations resulting from the removal of fuel element in specific configurations.

To validate the accuracy of k_{eff} and neutron flux estimations at the irradiation positions we compared the calculated results for the E01 configuration with those presented in the IEA-R1 research reactor Safety Analysis Report. Below we present the reactor core descriptions, details of the seven core configurations and specifics from the KENO-VI calculations.

Finally, a steady state single heated channel thermal analysis was performed in order to verify the thermal behavior of the chosen configuration.

2.1. Neutron flux, k_{eff} and burnup calculations

The reactor core was modeled based on configuration number 263, which was in place during the year 2022, using the KENO-VI code, a Monte Carlo module from the SCALE Code System. This method was chosen due to its reduced computational requirements in comparison with deterministic approaches when dealing with high complexity modelling scenarios such as nuclear reactors. All the simulations were executed using the 252 multigroup ENDF VII.0 library, embedded in the SCALE code system. The multigroup library was chosen due to its good performance capabilities (BOSTELMANN, 2023). This work focused on the thermal flux, considering neutrons with energies below 0.625 eV.

To achieve accurate flux, simulations were conducted using the mesh feature of KENO-VI, which allows to compute the fluxes in a grid-based system. In this work the grid was defined with each mesh having dimensions 2.59 cm x 2.7 cm x 1.36 cm, returning in a flux map with 211,680 positions.

For depletion (burnup) and activation simulations the TRITON module of SCALE was used, providing a depletion sequence using the same model employed in KENO-VI (Wieselquist et al., 2020). The construct the depleted IEA-R1 configuration E01 (configuration 263) each fuel element burnup was updated using the Burnup Monitoring Report provided by IEA-R1 operator (RODRIGUES, 2022).

2.2. Reactor core description

The IEA-R1 reactor is a 5 MW Material Test reactor (MTR) located at the Institute for Nuclear and Energy Research (IPEN) at the University of São Paulo campus, São Paulo, Brazil. The IEA-R1 reactor core consists of a matrix plate with 80-positions (8x10) accommodating fuel elements, neutron reflectors and irradiation positions. The MTR type fuel elements have 19.75 % enriched uranium, the dispersed uranium silicide alloy (U_3Si_2-Al) cladded with an Aluminum plate.

Fig. 1 shows the core configuration 263 of the IEA-R1 research reactor, comprising 24 Standard Fuel Elements (SFE) and 4 Control Fuel Elements (CFE) surrounded by beryllium and graphite reflectors. For material irradiation, 11 positions are available throughout the core (Ipen, 2009).

Each SFE comprises 18 fuel plates measuring 7.17 cm x 0.152 cm x 62.5 cm. The fuel meat is positioned within a framing module and compressed between two aluminum plates, which act as cladding. The active region itself has dimensions 6 cm x 0.076 cm x 60 cm.

The CFE share similarities with the SFE, albeit with twelve fuel plates. The remaining spaces serve as a pathway for the insertion of control rods. These control rods are made of Ag-In-Cd alloy with proportion of 80 %, 15 % and 5 %, respectively. Fig. 2 shows the modeled SFE and CFE used in this work, being blue, green, and white the water (moderator), aluminum (cladding and structure), and control rods, respectively. The colored plates situated within the plates represent the active region of the fuel.

2.3. Core enrichment

Each of the fuel elements were modeled with its own enrichment according to the facility burn up report from February 2022, each of the

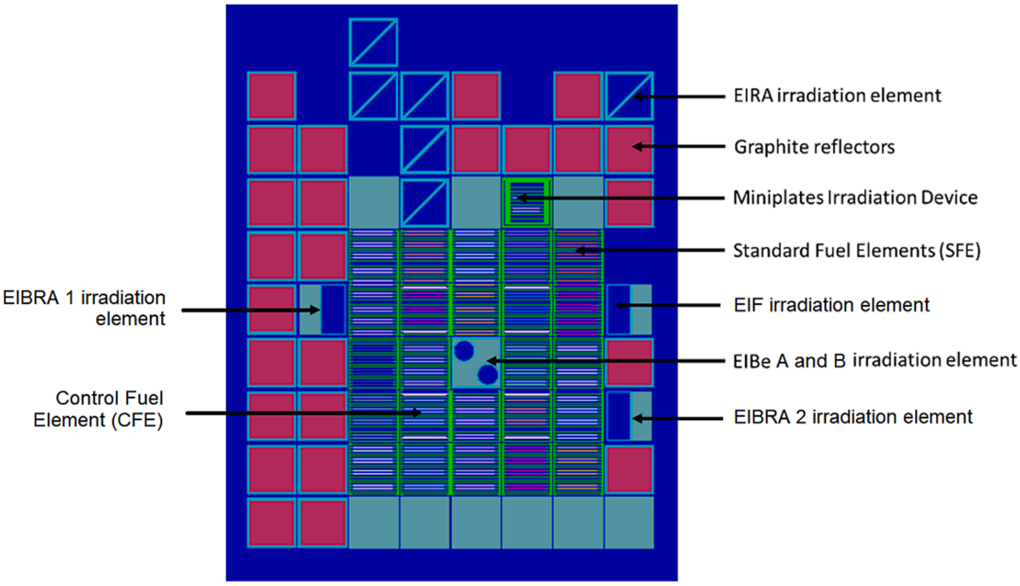


Fig. 1. Core configuration number 263 (Configuration E01) from the IEA-R1 research reactor. Top view identifying fuel elements, irradiation positions, neutron reflectors and other devices.

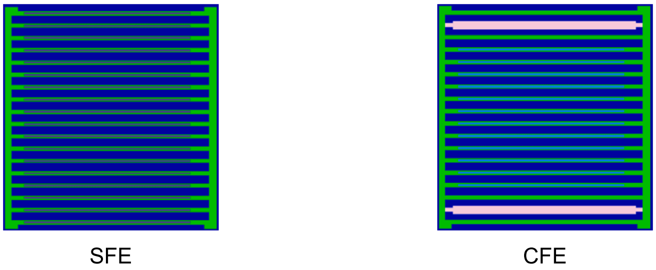


Fig. 2. Top view of Standard Fuel Element (SFE) and Control Fuel Element (CFE).

fuel elements were modeled with a different enrichment. Fig. 3 shows the values considered.

In the chosen configuration (E05) the fuel elements were arranged in a manner that the four most depleted ones were removed and the adjacent fuel elements were moved to theirs position ion the following disposition:

- 43 → 44 → 45 removed
- 47 → 46 → 55 removed
- 83 → 84 → 85 → 75 removed
- 87 → 77 → 67 → 66 removed

2.4. *Proposed core configurations to enhance neutron flux at irradiation positions*

In addition to the standard configuration (E01), six alternative arrangements were modeled and designated as configurations E02 to E07 as shown in Fig. 4 In the configuration E02, the Graphite reflectors adjacent to the fuel elements were replaced by Beryllium ones. In the E03, all the graphite reflectors present in the core were replaced by Beryllium reflectors. In the E04 configuration, four of the standard fuel elements were removed in a cross pattern, and in the E05 configuration, an "X" pattern was followed to remove four standard fuel elements.

The removal of 4 SFE elements from the core opens the opportunity to incorporate four additional irradiation positions. Two additional configurations (E06 and E07) were devised to represent these new conditions. In configuration E06, EIRA elements were inserted on the positions where the SFE were removed, and in configuration E07, EIBRA elements were inserted for this purpose.

A single-channel model was developed using the ANSYS CFX computational fluid dynamics (CFD) package to analyze the thermal characteristics of this new core proposal for the IEA-R1 reactor. We used the benchmark (Hainoun et al., 2014) as a reference for defining the geometric model, boundary, and operating conditions of the IEA-R1 standard core and the new proposal, respectively. We designed a 1/2 three-dimensional model of a fuel element (composed of the fuel plate, cladding, and coolant) with the ANSYS SpaceClaim tool, taking advantage of the symmetry conditions, as illustrated in Fig. 5.

The proposed new core holds the structural dimensions of the standard core with fewer fuel assemblies. Therefore, the geometry of the fuel element was modeled according to the dimensions presented in Table 1.

A spatial discretization of the computational domain was designed using the ANSYS Meshing tool, in which we built a mesh of hexahedrons for all regions under analysis, taking advantage of the geometry sweep condition. Fig. 6 illustrates the mesh used in the present study.

The fuel and cladding regions were discretized with six elements in thickness, while the coolant region has 20 elements to capture local phenomena accurately. As a result, the mesh used in this study has 5.04

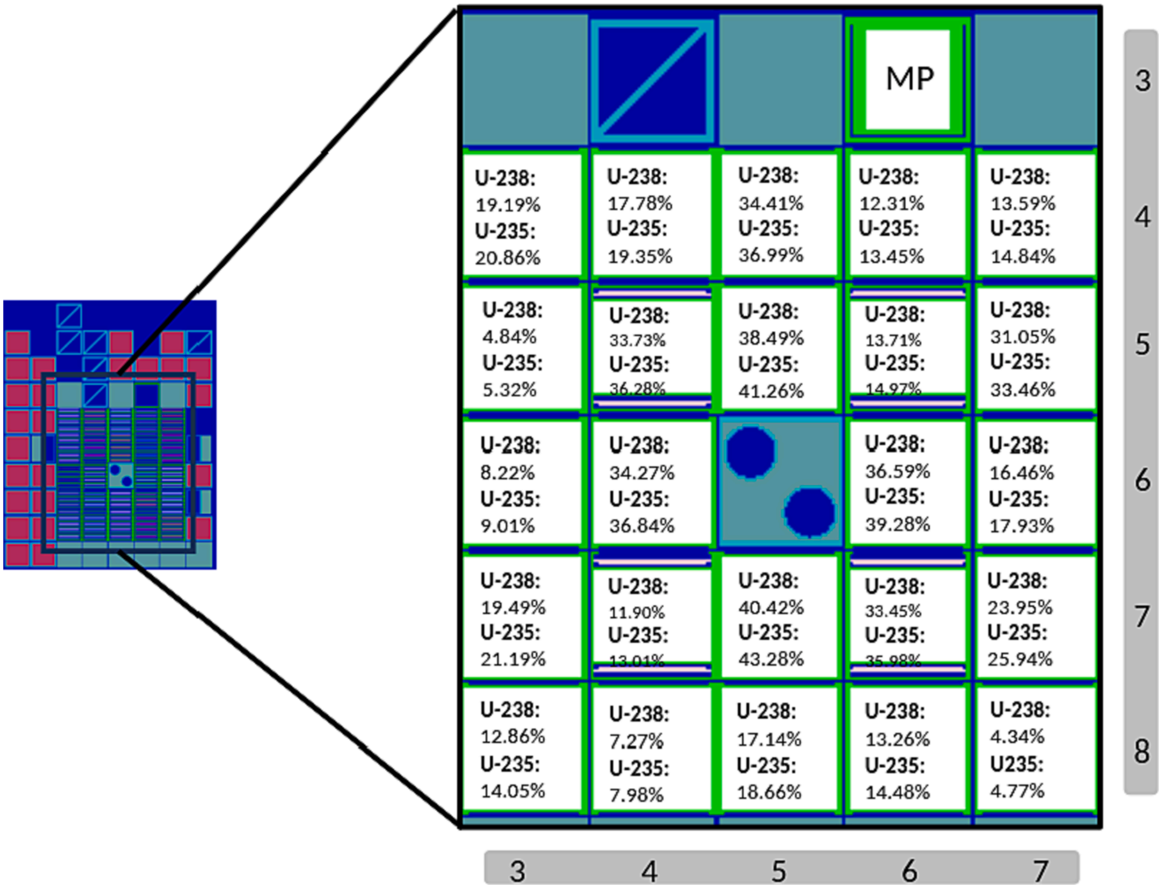
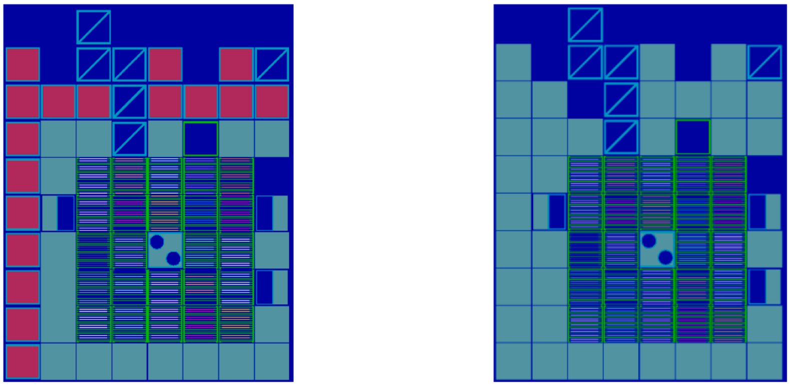
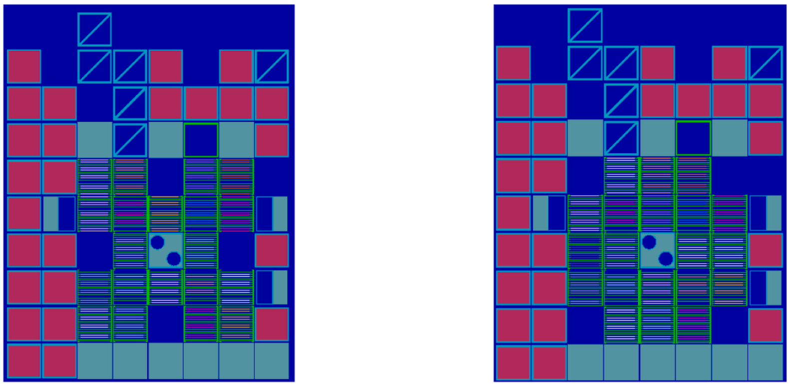


Fig. 3. Burn up for the E01 configuration.



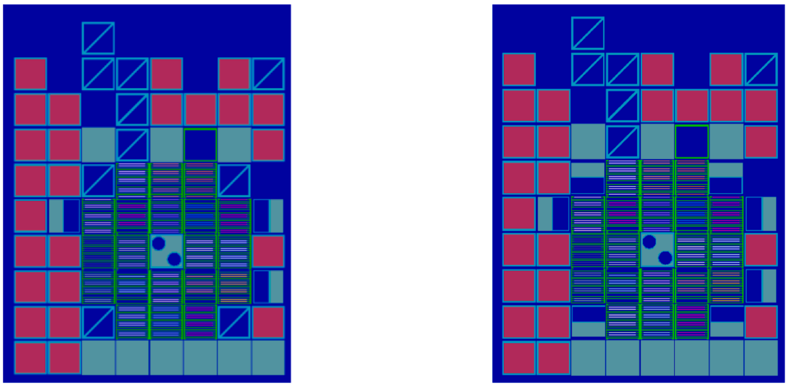
Configuration E02 with 24 fuel elements

Configuration E03 with 24 fuel elements



Configuration E04 with 20 fuel elements

Configuration E05 with 20 fuel elements



Configuration E06 with 20 fuel elements

Configuration E07 with 20 fuel elements

Fig. 4. Configurations E02 to E07.

million nodes and 4.85 million hexahedron elements. Table 2 presents the mesh quality parameters and their respective results.

By taking advantage of the geometry's sweep behavior, a mesh of excellent quality was build, which reflects the response accurately and with an optimized calculation time. To achieve this scenario, in addition to the refined work with the mesh stage, we performed physical modeling of the problem based on the benchmark (Hainoun et al., 2014) and the paper (Gong et al., 2015) to define the boundary/operation conditions and the thermophysical properties of materials, respectively. Table 3 presents the boundary conditions for steady-state analysis of the standard IEA-R1 reactor core.

In the current study, the materials used, except water, are not defined in ANSYS CFX. As a result, we set the thermophysical properties of the fuel (U3Si2 – Al) and the cladding (AL – 6061). Table 4 lists the constant values for thermal conductivity, density, and specific heat capacity suggested by (Gong et al., 2015).

The IAPWS-IF97 library was used to get water properties, and a the standard k-epsilon turbulence model with a scalable wall function was

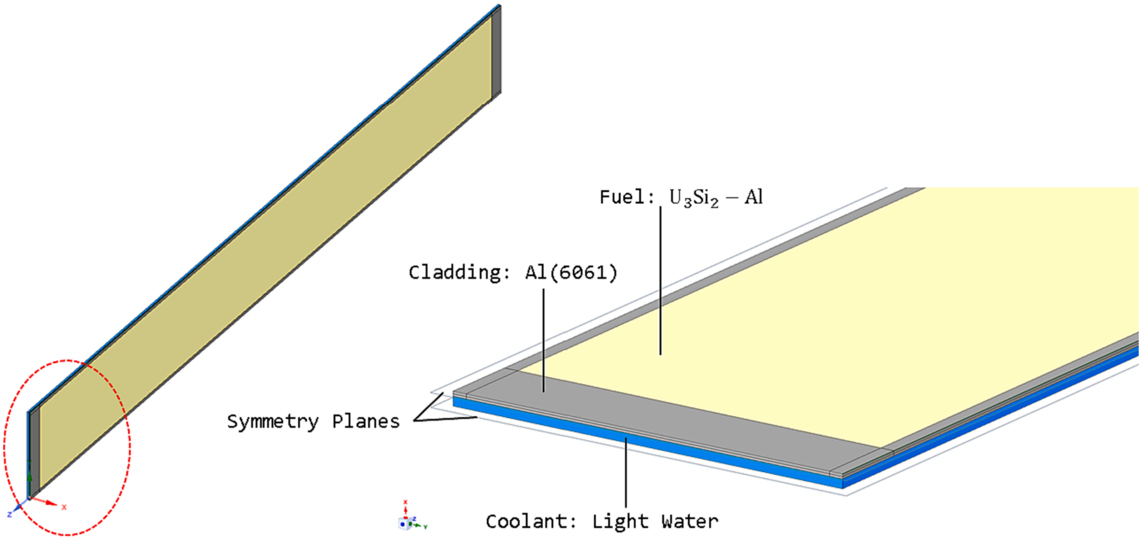


Fig. 5. Fuel element isometric view.

Table 1
Fuel element dimensions (Hainoun et al., 2014).

Domain	Values (mm) Thickness	Width	Length
Fuel	0.76	62.6	600
Cladding	1.52	67.1	625
Coolant	2.89	67.1	625

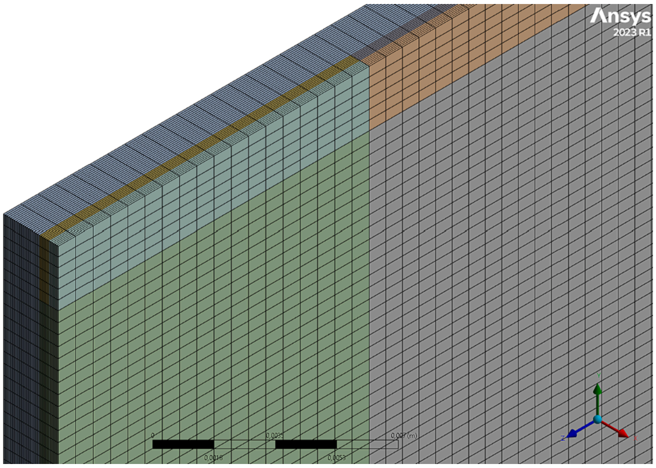


Fig. 6. Isometric view of the fuel element mesh.

Table 2
Mesh metrics.

Parameter	Minimum	Maximum	Average	Standard deviation
Skewness	1.3057x10^{-10}	5.9914x10^{-004}	8.3443x10^{-006}	3.3438x10^{-005}
Orthogonal Quality	1	1	1	2.136x10^{-007}
Aspect Ratio	9.2266	10.965	9.7299	0.63459

set. A high-resolution option was assumed for the advective scheme, with a convergence criterion of 1E-6 and a maximum number of interactions of 1E3.

Table 3
IEA-R1 steady state boundary conditions (Hainoun et al., 2014).

Parameter	Value
Nominal System Pressure	1.7 bar
Nominal Reactor Power	3.5 – 5.0 MW
Mass Flow Rate (per Instrumented Fuel Assembly)	6.27 kg/s
Reactor Core Inlet temperature	31.61 – 33.43 °C

Table 4
Materials thermophysical properties (Gong et al., 2015).

Domain	Parameter	Value
Fuel	**Thermal Conductivity**	50.03 W/mK
(U_3Si_2 – Al)	**Density**	6030 kg/m^3
	Specific Heat Capacity	338.4 J/kgK
Cladding	**Thermal Conductivity**	176.01 W/mK
(AL – 6061)	**Density**	2700 kg/m^3
	Specific Heat Capacity	998.56 J/kgK

2.5. Production of ^{177}Lu

To assess potential of producing ^{177}Lu, a sample was placed in the position with the highest flux (1.21x10^{14} at ~ 13 cm below the center of the active region). This work has considered only the direct route of production ^{176}Lu(n,γ)^{177}Lu due to the lower logistics costs (PILLAI, 2023).

The aluminum containment filled with air, as used in experimental situations, was modeled to hold the sample. The irradiated sample consists of a pure ^{176}Lu cuboid (100 % enriched) with a volume of 0.5 cm^3 (dimensions 1 cm x 1 cm x 0.5 cm). Special care was taken to account for the self-shielding effect on highly enriched lutetium samples (Sairanbayev et al., 2021). Fig. 7 shows the modelled containment with the ^{176}Lu sample inside.

3. Results

3.1. Validation

Once modelled some of the key parameters from the core configuration 263 were compared benchmark results. The parameters were the effective multiplication factor, xenon buildup and the core thermal flux at mapped positions inside different irradiator elements. The benchmark values were taken from the IEA-R1 reactor Safety Analysis Report (SAR)

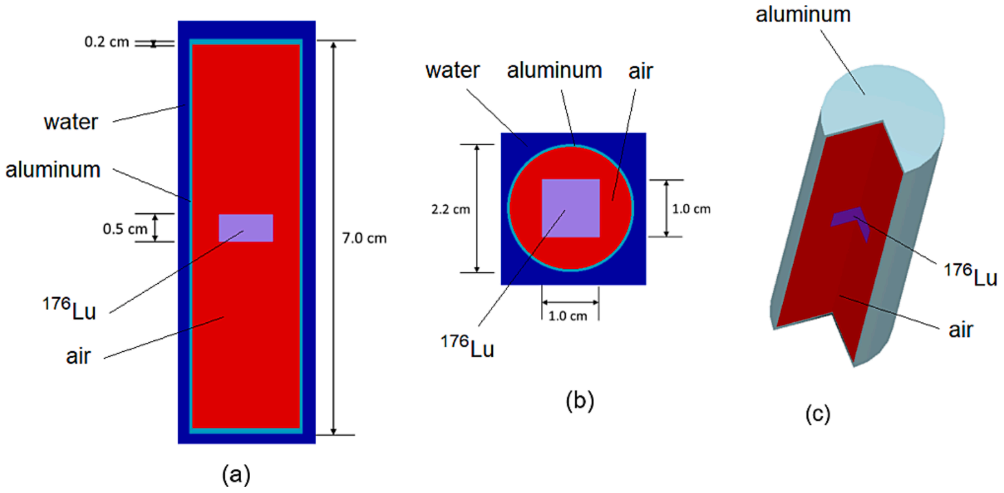


Fig. 7. Aluminum containment. Front view (a), Top view (b) and 3D view (excluded the water surrounding).

or furnished by the IEA-R1 operator.

The xenon buildup in the reactor core was obtained from a 60-hour simulation (48 h at 100 % power followed by 12 h at 0 % power). was conducted. The model yielded an average ^{135}Xe equilibrium concentration of 2.29 μg/cm^3 resulting in a total insertion of negative reactivity of 3135 ± 123 pcm. The SAR stated a negative reactivity insertion of 3200 pcm in equilibrium, resulting in a discrepancy of 2.07 %.

Fig. 8 presents the thermal neutron flux at several core locations used for sample irradiation. The results were compared with the thermal flux map provided by the reactor's management to the users.

The comparison indicates a good adherence of the model with the flux map provided by the reactor operator since the mapped neutron flux could present errors up to 50 % depending on the irradiation time (SILVA, 2008). The model presented combined discrepancies of 7.5 % and 16.8 % for average and peak thermal neutron fluxes, respectively.

3.2. *Thermal neutron flux*

To compare the different configurations (E01, E02, E03, E04 and E05), the thermal neutron flux was obtained at five irradiation positions: EIF, EIBRA 1, EIBRA 2, EIBe A and EIBe B, identified in Fig. 1. Thermal neutrons were defined as those with energies below 0.625 eV. All neutron flux simulations were conducted with 100 million histories (2000 generations with 50.000 neutrons each) and the control rods adjusted to the critical position. Figs. 9 to 13 show the resulted thermal flux in each position for the configurations under review. It is possible to

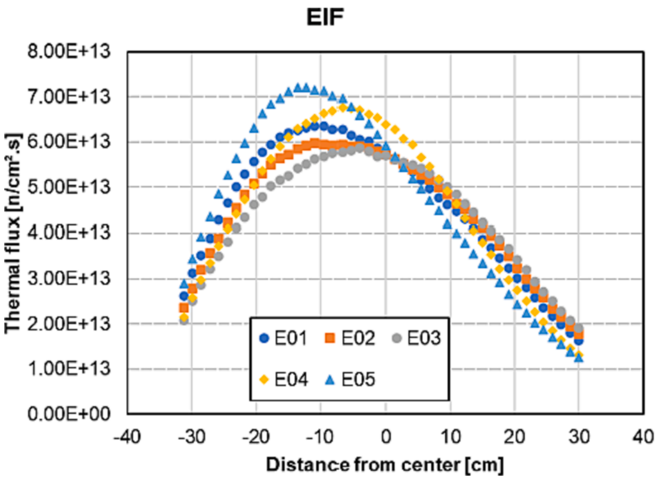


Fig. 9. Thermal neutron flux in the irradiation position EIF.

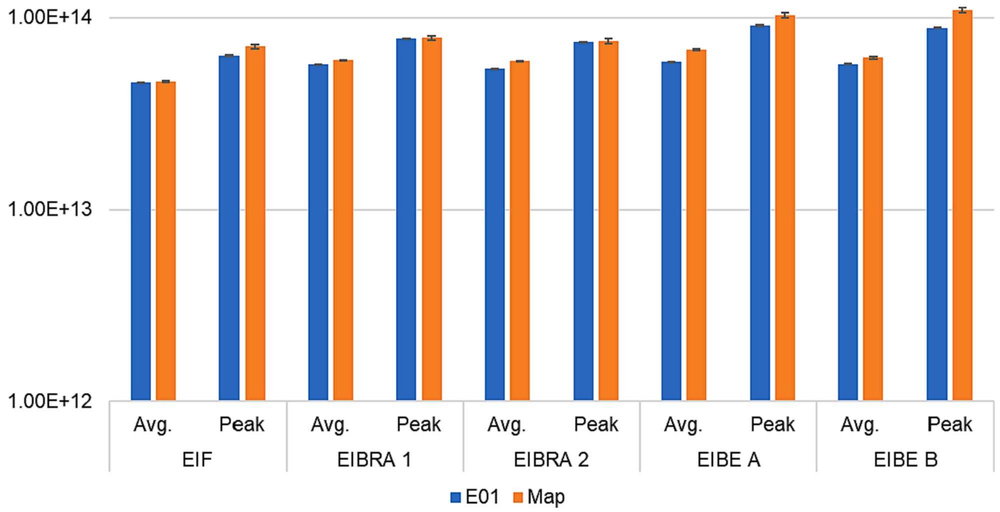


Fig. 8. Thermal neutron flux (n/cm^{2}s) comparison at different irradiation locations. Map means results provided by the IEA-R1 operator and E01 means results from this model. The results include average and peak thermal flux at the irradiation locations.

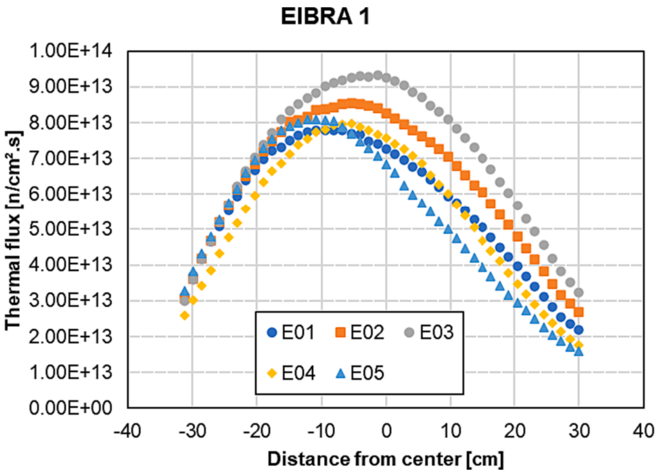


Fig. 10. Thermal neutron flux in the irradiation position EIBRA 1.

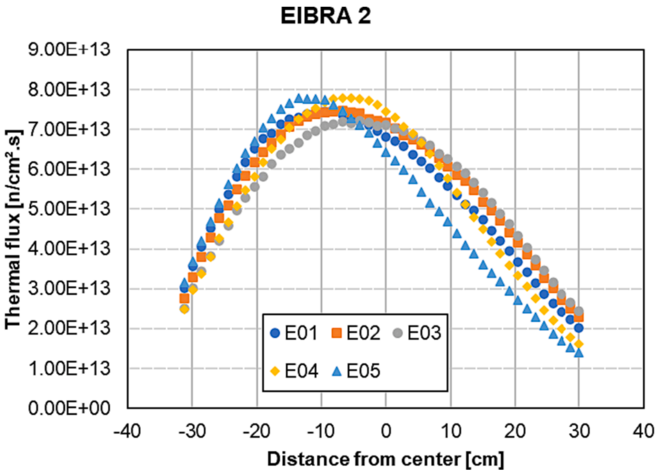


Fig. 11. Thermal neutron flux in the irradiation position EIBRA 2.

see that a great impact is made in the EIBe positions due to its central position in the core.

Table 5 summarize the fluxes obtained with respect to the standard configuration E01. Configurations E04 and E05 results present greater increases in the thermal neutron fluxes. At irradiation positions at the EIBE irradiation element it reaches over 30 % increase.

The configurations E04 and E05 with 20 fuel elements provide the greater average and peak neutron flux increases. Configuration E02 presents thermal neutron flux reduction in the irradiation positions and E03 presents a marginal increase of less than 3 % at peak locations.

3.3. *Moderator and fuel temperature coefficients of reactivity*

The E05 configuration was chosen, since it provided the most significant increase in neutron flux at peak locations to the fuel and moderator temperature coefficients of reactivity. The water temperature was raised from 294.5 K to 354.5 K in three steps of 20 K each, while the fuel temperature remained constant at 356.5 K. Similarly, the fuel coefficient was examined by simulating three steps of 30 K, 50 K and 100 K, ranging from 294.5 K to 474.5 K. Fig. 14 shows the evolution of the multiplication (k_{eff}) factor with the changes in the moderator and fuel temperatures.

Table 6 shows the resulting moderator and temperature coefficients for E05 configuration in comparison with the standard one (E01).

3.4. *Fuel cycle length*

The nominal thermal power operation level for the IEA-R1 reactor is 4.5 MW. The reduction in core size from 25 to 20 fuel elements elevates the average thermal power generation per fuel element from 187.5 kW to 225 kW or 20 % and hence, it reduces the fuel cycle length of the reactor. Fig. 15 compares the k_{eff} as a function of irradiation time for configurations E01 and E05.

Configuration E01 has a fuel cycle length of about 380 days while configuration E05, about 220 days. The fuel cycle length reduction is of 160 days or 42 %.

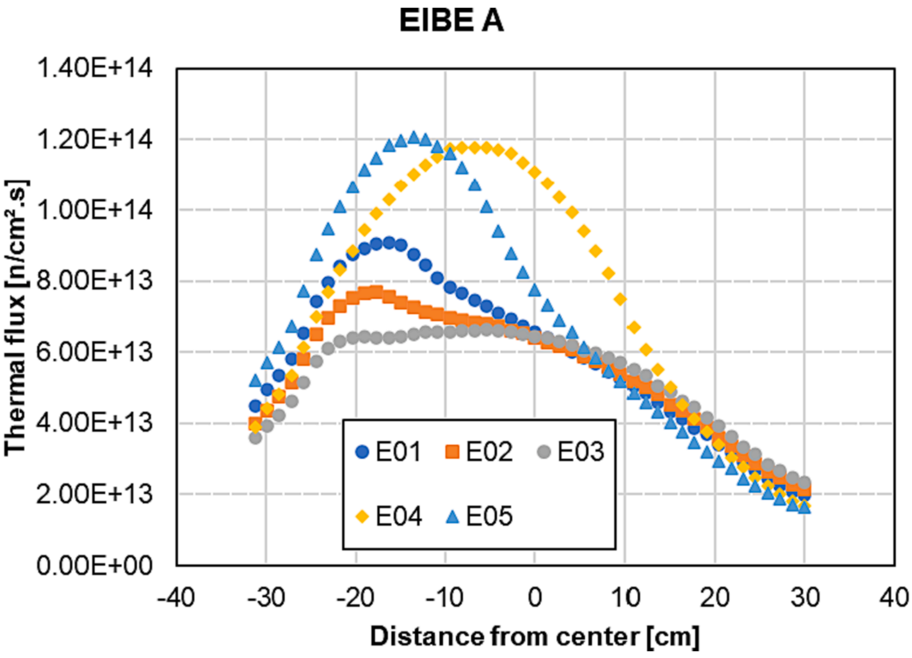


Fig. 12. Thermal neutron flux in the irradiation position EIBE A.

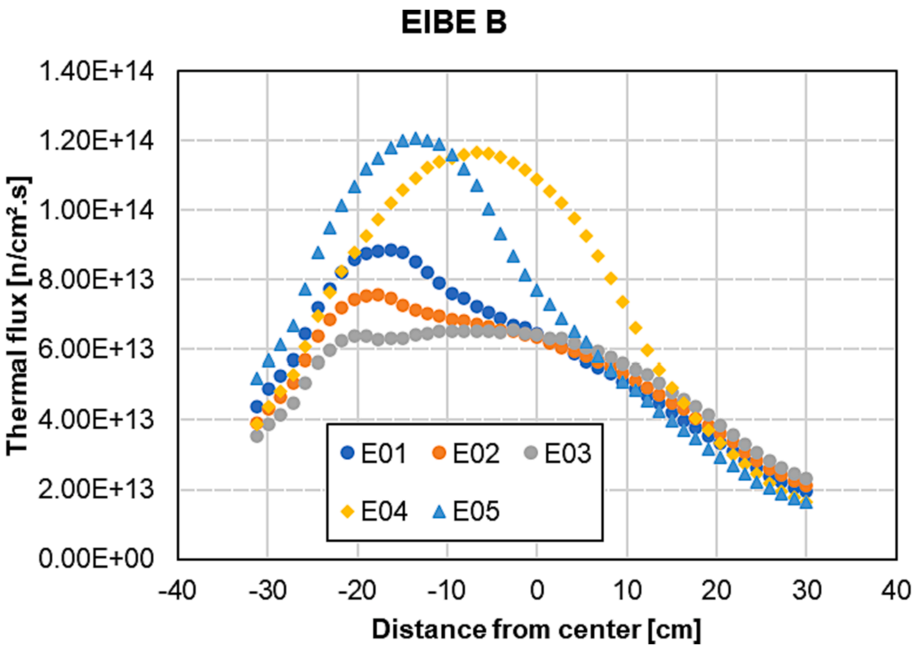


Fig. 13. Thermal neutron flux in the irradiation position EIBE B.Fig. 15. Lifecycle comparison between E01 and E05 configurations. The operation thermal power level is 4.5 MW.

Table 5
Thermal neutron flux variation with respect to the E01 configuration.

		E01	E02	E03	E04	E05
EIF	**Avg.**	$4.59x10^{+13}$	−2.09 %	−4.18 %	−0.42 %	2.96 %
	Peak	$6.35\ x10^{+13}$	−5.93 %	−7.54 %	6.37 %	13.49 %
EIBRA 1	**Avg.**	$5.73\ x10^{+13}$	11.20 %	22.21 %	−4.47 %	−6.01 %
	Peak	$7.79\ x10^{+13}$	9.52 %	19.67 %	2.45 %	3.84 %
EIBRA 2	**Avg.**	$5.44\ x10^{+13}$	2.73 %	0.34 %	−2.05 %	−5.88 %
	Peak	$7.46\ x10^{+13}$	0.19 %	−3.16 %	4.54 %	4.38 %
EIBE A	**Avg.**	$5.89\ x10^{+13}$	−6.50 %	−9.64 %	26.83 %	18.16 %
	Peak	$9.09\ x10^{+13}$	−15.41 %	−27.12 %	29.75 %	32.69 %
EIBE B	**Avg.**	$5.74\ x10^{+13}$	−5.50 %	−8.38 %	28.27 %	20.92 %
	Peak	$8.86\ x10^{+13}$	−14.54 %	−26.13 %	31.65 %	36.46 %
TOTAL	**Avg.**	$5.48\ x10^{+13}$	−0.01 %	0.18 %	10.29 %	6.36 %
	Peak	$9.09\ x10^{+13}$	−6.14 %	2.56 %	29.75 %	32.93 %

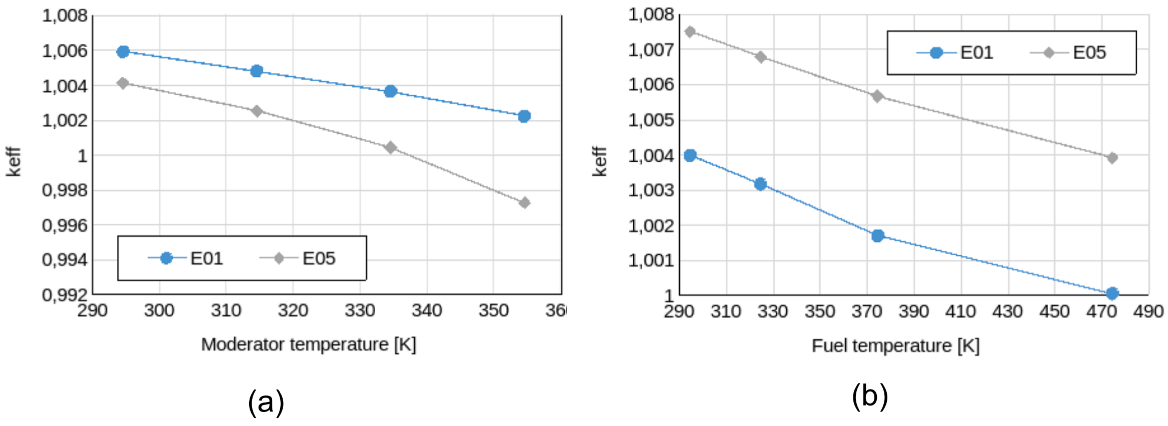


Fig. 14. K_{eff} evolution for temperatures changes in moderator (a) and fuel (b).

3.5. ^{135}Xe poisoning

In the E05 configuration the ^{135}Xe build up in the core remains at the same levels as in the standard configuration as illustrated in Fig. 16. The reactivity inserted in the core at this time would be of about 3600 pcm.

Table 6
Fuel and moderator coefficients.

Configuration	Fuel temperature (K)	Moderator temperature (K)	Value (pcm/K)
Moderator temperature coefficient of reactivity			
E01	356.5	314.5	−12.00
		334.5	−16.15
		354.5	−18.98
E05	356.5	314.5	−7.84
		334.5	−10.39
		354.5	−15.99
Fuel temperature coefficient of reactivity			
E01	309.5	354.5	−2.7
	349.5		−2.2
	424.5		−1.7
E05	309.5	354.5	−2.4
	349.5		−2.2
	424.5		−1.7

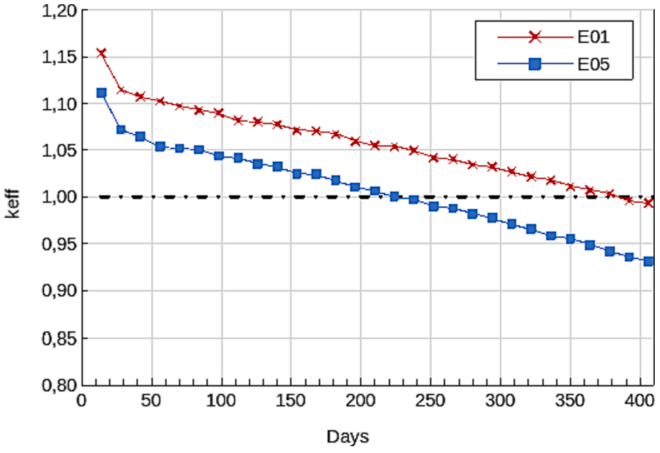


Fig. 15. Lifecycle comparison between E01 and E05 configurations. The operation thermal power level is 4.5 MW.

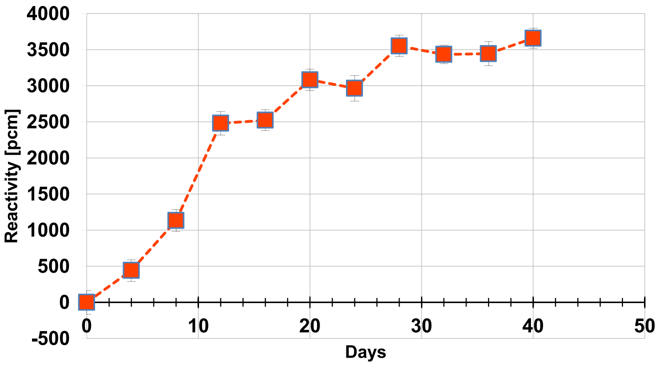


Fig. 16. ^{135}Xe buildup as a function of time for the E05 configuration.

3.6. *Lutetium-177 production*

Fig. 17 presents the production of 177Lu as a function of irradiation time for configurations E05 and E01. Considering 28 days of full-power operation, the E01 configuration produced a sample with 850.00 TBq/g, while the E05 configuration, 1025.45 TBq/g, i.e., a 20.64 % increase. The extended 28 days cycles was used in order to verify the production capacity in a theorical situation, since the IPEN reactor currently does not operate for more than 3 days a week.

Fig. 17 results are consistent with the expected production at this flux level (~10^{14}n.cm^{-2}.s^{-1}) (ZAHN, 2021).

3.7. *Thermal analysis*

As part of the evaluation process for the new core, studies in Computational Fluid Dynamics (CFD) were conducted using the operating conditions of the standard core. The goal was to confirm the applicability of the CFD model for the standard core configuration and then utilize it for the new core proposal. Table 7 presents the results of the CFD model compared with the experimental results available in the benchmark (Hainoun et al., 2014) for the standard IEA-R1 reactor core.

After comparing the CFD model with the experimental results of the standard configuration of the IEA-R1 reactor, the same model was applied to the new core proposal, which reduces the number of fuel assemblies to 20 and maintains the same power of 4.5 MW. In this case, the cosine approach was assumed to obtain the power profile and consider the same input conditions for the temperature (32.53 °C) and the mass flow rate (6.27 kg/s) in the instrumented fuel assembly.

The reactor's new core underwent an evaluation process in different power density scenarios, with an average core power density of 503.28 W/cm3 and a maximum power density of 710.10 W/cm3. As the benchmark only reports average operating values, we compared the temperature values of the new proposal with the default core configuration, as highlighted in Table 8.

It was noted that in terms of average values, the new core does not present excessively higher values, even in its hottest region. As the standard configuration of the reference reactor has an operating range between 2 and 5 MW, we compared the results (Table 9) of the new proposal (4.5 MW) with the values of the standard core for the 5 MW operating scenario.

Although several approximations, the new core proposal exhibits significant potential. Consequently, a comprehensive evaluation employing specialized codes such as RELAP, PARET-ANL, MTRCR, CATHARE, and others is imperative. The assessment will provide a more detailed understanding of the proposal's feasibility and efficacy, enabling to take more informed decisions regarding its implementation.

The fuel temperatures in tables 8 and 9 are well below 95 °C in the hot channel, thus ensuring the integrity of the fuel element, as the biggest limitation of the IEA-R1 reactor configurations is the aluminum alloy from which the reactor is made. fuel element, and cannot assume a value greater than this(Hainoun et al., 2014)(SANTOS, 2023).

We performed the thermal analysis according to the benchmark (Hainoun et al., 2014) for the reactor operation under steady-state conditions, and then we assumed constant values for the mass flow rates. It's worth noting that the maximum limit of the mass flow rate doesn't hold much significance for the thermal analysis conducted in this paper since we verified some specific reactor operating conditions of the steady-state scenario.

3.8. *New irradiation positions*

Removing 4 SFE elements from the core opens the possibility of adding four more irradiation positions in the IEA-R1 core. Two additional configurations (E06 and E07) were designed to account for the irradiation potential in these new positions.

In configuration E06, irradiation elements type EIRA were inserted in the positions where the SFE were removed. In configuration E07, EIBRA type irradiation elements were inserted in the liberated positions. The positions in the core are 43, 47, 83 and 87 and refer to core matrix being 43 the upper-left position and the 87 the lower-right position. The average thermal neutron flux in the positions is around 6x10^{13} n/cm^{2}s. The EIBRA irradiation element furnishes a thermal flux about 11 % greater than the EIRA irradiation element. The peak thermal flux in the EIBRA irradiation is greater than 8x10^{13} n/cm^{2}s in all positions and at

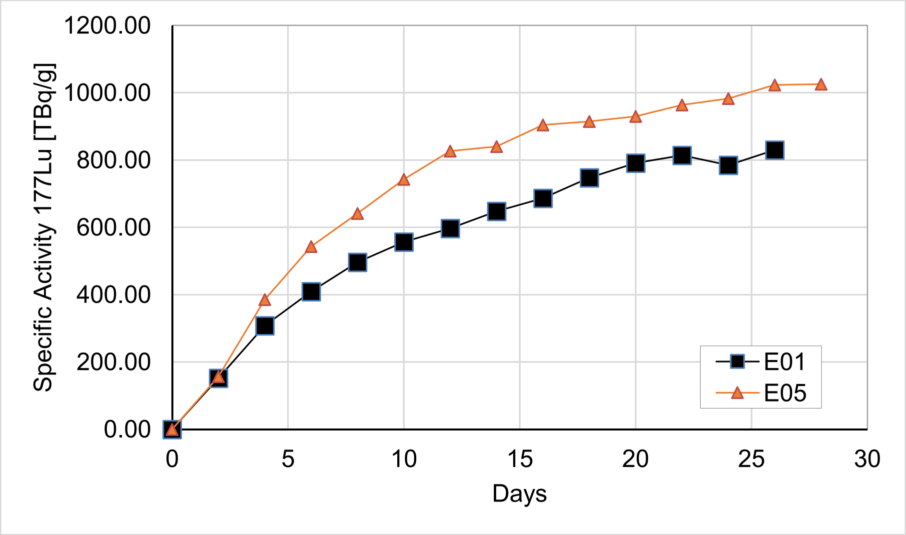


Fig. 17. Specific activity ^{177}Lu as a function of time.

Table 7
Experimental and CFD results for the IEA-R1 standard configuration.

Steady State Conditions		Coolant					Fuel
		Experimental data			CFD		CFD
Reactor Power (MW)	Nominal Operating Pressure (bar)	Mass Flow Rate (kg/s)	Inlet Temp. (°C)	Average Outlet Temp. (°C)	Average Outlet Temp. (°C)	Relative Error (%)	Maximum Temp. (°C)
3.5	1.7	6.27	32.69	37.57	37.58	0.01	47.107
4.0	1.7	6.27	31.61	37.23	37.23	0.01	48.218
4.5	1.7	6.27	32.53	38.69	38.70	0.02	50.559
5.0	1.7	6.27	33.43	40.21	40.21	0.01	53.038

Table 8
CFD model results for the new IEA-R1 reactor core arrangement.

Scenario	Reactor Power (MW)	Inlet Temp. (°C)	Mass Flow Rate (kg/s)	Outlet Temp. (°C)	Fuel Maximum Temp. (°C)
Benchmark	4.5	32.53	6.27	38.69	–
CFD of benchmark (Hainoun et al., 2014**)**	4.5	32.53	6.27	38.697	50.559
CFD of E05 configuration	4.5	32.53	6.27	41.833	59.325
CFD of E05 configuration hottest region	4.5	32.53	6.27	45.655	69.323

Table 9
Comparison between the new reactor core arrangement results with the 5.0 MW standard operating scenario.

Scenario	Reactor Power (MW)	Inlet Temp. (°C)	Mass Flow Rate (kg/s)	Outlet Temp. (°C)	Fuel Maximum Temp. (°C)
Benchmark	4.5	32.53	6.27	38.69	–
CFD of benchmark (Hainoun et al., 2014)	4.5	32.53	6.27	38.697	50.559
CFD of E05 configuration	4.5	32.53	6.27	41.833	59.325
Benchmark	5.0	33.43	6.27	40.21	–
CFD of benchmark	5.0	33.43	6.27	40.207	53.038

83 it is greater than 1x10^{14} n/cm^{2}s.

4. Conclusions

Changing the core configuration to a near cylindrical arrangement demonstrated significant potential of enhancing the IEA-R1 production capacity. This lead to an increasing in peak and average neutron flux, in the centered irradiation positions by approximately 20 % and 30 % respectively.

Another advantage of the E05 configuration is the ability to use four more irradiation positions, with an average neutron flux in 10^{13}n/cm^2/s range and a peak location of 10^{4}n/cm^2/s when using an EIBRA irradiation element. It is important to note that adding these new irradiation elements could provide 64 new irradiation locations (using EIRA elements with 2x8 locations each) or even 88 new locations (using an EIBRA with 22 locations each).

Taking into account the current fuel burnup, the faster consumption in the E05 configuration corresponds to a 60.71 % smaller lifecycle, implying the need for increased fuel element production and its management after use.

In the context of using the IEA-R1 reactor to keep up with the increasing demand for radioisotopes in Brazil, changing the IEA-R1 reactor to a nearly cylindrical geometry could serve as a solution for expand the production in the facility.

CRediT authorship contribution statement

Felipe Viggiano: Writing – review & editing, Writing – original draft, Investigation, Formal analysis, Data curation, Conceptualization. **Giovanni L. Stefani:** . **Frederico A. Genezini:** Writing – review & editing, Supervision, Methodology, Funding acquisition. **João M.L. Moreira:** Writing – review & editing, Supervision, Methodology, Formal analysis, Data curation. **Caio J.C.M. Cunha:** Software, Formal analysis.

Declaration of competing interest

The authors declare the following financial interests/personal relationships which may be considered as potential competing interests: Giovanni L Stefani reports financial support, administrative support, article publishing charges, equipment, drugs, or supplies, statistical analysis, travel, and writing assistance were provided by Universidade Federal do Rio de Janeiro. Giovanni Laranjo de Stefani reports financial support was provided by Federal University of Rio de Janeiro COPPETEC Foundation. Giovanni Laranjo de Stefani reports a relationship with Federal University of Rio de Janeiro that includes:. Giovanni Laranjo de Stefani has patent pending to Giovanni Laranjo de Stefani. There are no relationships that could generate conflicts of interest. If there are other authors, they 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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