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Original Article

Core with unenriched MOX fuel for a Microreactor PWR

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Abstract: Microreactors, with their compact design, high efficiency, and robust operational characteristics, represent a promising solution for clean and reliable energy generation, particularly in remote regions or emergency scenarios. A key aspect of the development of these systems is the diversification of nuclear fuels beyond conventional uranium dioxide, aimed at enhancing fuel-cycle efficiency, reducing radioactive waste, and mitigating nuclear proliferation risks. This work presents the conceptual design of a microreactor core using MOX fuel without uranium enrichment. Several initial compositions of $PuO_2 + UO_2$ were evaluated, and the mixture that yielded the best performance, discussed throughout the article, was 15% PuO_2 and 85% UO_2. This composition met all design constraints and enabled a fuel cycle of approximately 15 years, operating at microreactor power levels. The SCALE computational system was employed to define the fuel cell and to perform the core burnup calculations. The results demonstrate the feasibility of using unenriched MOX in microreactors, highlighting its potential to expand fuel-cycle flexibility and contribute to a more sustainable and diversified energy supply.

Keywords: PWR Microreactor, MOX Microreactor, Core design, SCALE.

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Núcleo com combustível MOX sem enriquecido para um Microreator PWR

Resumo: Microreatores nucleares, devido ao seu design compacto, à elevada eficiência e robustez operacional, apresentam-se como uma alternativa promissora para a geração de energia limpa e segura, especialmente em regiões remotas ou em situações de emergência. Um aspecto fundamental no desenvolvimento desses sistemas é a diversificação do combustível, além do dióxido de urânio convencional, visando aumentar a eficiência do ciclo, reduzir a quantidade de resíduos radioativos e mitigar riscos associados à proliferação nuclear. Neste trabalho, foi desenvolvido o projeto conceitual do núcleo de um microrreator a combustível MOX, sem enriquecimento de urânio. Foram avaliadas diferentes composições iniciais de PuO_2 + UO_2, e a mistura que apresentou o melhor desempenho, conforme discutido ao longo do artigo, foi a de 15% de PuO_2 e 85% de UO_2. Essa composição atendeu às restrições do projeto e permitiu alcançar um ciclo de combustível de aproximadamente 15 anos, operando em potência de microrreator. As análises foram realizadas utilizando o sistema computacional SCALE, empregado tanto para a definição da célula combustível quanto para os cálculos de queima do núcleo. Os resultados obtidos demonstram a viabilidade do uso de MOX sem enriquecimento em microrreatores, destacando seu potencial para ampliar a diversidade de combustíveis e contribuir para uma matriz energética mais sustentável.

Palavras-chave: Microrreator PWR, Microrreator MOX, Núcleo Design, SCALE.

1. INTRODUCTION

The pursuit of cleaner, safer, and more adaptable energy solutions to meet the diverse demands across Brazil's territory has driven strategic initiatives in the nuclear field. Among them, a prominent example is the project led by the National Nuclear Energy Commission (CNEN) to develop a national nuclear microreactor, aimed at supplying power to remote regions, research centers, and strategic facilities such as military bases. As highlighted by CNEN in a recent report, the microreactor initiative is part of a broader energy diversification policy, reinforcing the importance of technological autonomy and mastery of the nuclear fuel cycle in Brazil [1].

In this context, the present work proposes the design of a core for a Pressurized Water Reactor (PWR)- type microreactor, using as fuel a mixture of metallic oxides (MOX) composed of natural uranium dioxide (UO_2) and plutonium dioxide (PuO_2). The choice of this type of fuel is justified by both the potential to reuse nuclear materials from irradiated fuel and strategic stockpiles, and the opportunity to reduce dependence on uranium enrichment, a complex and sensitive stage of the fuel cycle [2].

The use of natural uranium, however, poses technical challenges that require careful evaluation of the oxide ratios in the MOX. The proportion of UO_2 to PuO_2 directly influences the core's reactivity performance, affecting key parameters such as initial criticality, fuel behavior during burnup, operational safety, and especially the reactor's operational lifespan. To fulfill its goal of continuous, remote, and low-maintenance operation, the microreactor must ensure not only safety and efficiency but also a long service life [3].

Therefore, this study is justified by the need to investigate, through neutronic modeling and simulation, the most appropriate MOX composition to ensure safe, efficient, and long-lasting performance of the proposed microreactor. Given the strategic relevance of the national microreactor project and the innovation in the proposed fuel concept, this research advances the technical knowledge needed to implement compact, sustainable nuclear technologies in Brazil.

2. MATERIALS AND METHODS

The conceptual core design was developed through numerical simulations using the SCALE computational system (Standardized Computer Analyses for Licensing Evaluation), a suite of codes widely employed for criticality safety analysis, fuel burnup, shielding design, neutron and photon transport, decay heat calculations, and reactor physics modeling. All simulations were performed under steady-state conditions, unless otherwise specified.

To model the MOX fuel, isotopic densities in atoms/barn·cm were calculated and used as input to the SCALE code. Fuel density varies with composition, temperature, and porosity. In this case, the density increases with increasing PuO2 and decreases with increasing temperature [4].

To execute the simulation, it was necessary to define in detail the composition of all materials used in the project. The atomic density of each constituent isotope was included explicitly, expressed in atoms/barn·cm. The isotopes considered in the MOX fuel are listed in Table 1. For the remaining materials in the system, the nuclear data available in the SCALE V5-44 library were adopted.

Table 1: Abundance percentage of the isotopes present in the MOX fuel. [5]

Isotopes	Abundance (%)
U-234	0.01
U-235	0.71
U-238	99.28
Pu-238	0.10
Pu-239	77.32
Pu-240	19.35
Pu-241	2.68
Pu-242	0.55

A linear relationship defined by the Oak Ridge Laboratory [4] was used to calculate the densities. Considering a temperature of 300K, the density of UO_2 is 10.961 kg/m³, and the density of PuO_2 is 11.450 kg/m³, resulting in the following relationship:

For the fuel density calculation, the linear relation proposed by Oak Ridge National Laboratory [4] was used. Considering the reference temperature of 273K, the density of UO_2 is 10,970± 70 kg/m³, and the density of PuO_2 is 11,460±80 kg/m³. Thus, the apparent density of the MOX fuel is calculated by Equation 1.

$$\rho_{MOX} = \rho_{UO_2}(1 - y) + \rho_{PuO_2} . y \quad (1)$$

- ρ_{MOX}: apparent density of the MOX fuel (kg/m^3)
- ρ_{UO_2}: density of UO_2
- ρ_{PuO_2}: density of PuO_2
- y: mass fraction of PuO_2 in the MOX fuel

As the temperature increases, the fuel undergoes thermal expansion, which leads to a progressive reduction in its density. To represent this volumetric variation over the relevant temperature range, the empirical correlations provided in the Oak Ridge National Laboratory report were employed [4].

With this, the apparent densities of UO2 and PuO2 can be obtained from equations 2 and 3. Finally, to obtain the density of each isotope present in the MOX fuel, it is necessary to multiply the apparent density by the abundance percentage, as shown in Table 1.

$$\rho'_{UO_2} = (1 - y) . \rho_{MOX} \quad (2)$$

$$\rho'_{PuO_2} = y . \rho_{MOX} \quad (3)$$

Reactivity is a fundamental parameter in nuclear reactor physics, as it governs the system’s behavior and determines whether the neutron population will increase, decrease, or remain constant during operation. It reflects the balance between neutron production, absorption, and leakage within the core. A related quantity, widely used in design analyses, is the effective neutron multiplication factor (k_{eff}), which represents the system's ability to sustain a chain reaction. It is defined as the ratio of the number of neutrons in successive generations. For a reactor to be critical or supercritical, its value must be equal to or greater than unity [5].

Another key aspect influencing reactivity is the relationship between the infinite multiplication factor (k_∞) and the moderator-to-fuel volume ratio (V_m/V_f). This ratio directly affects the efficiency of moderation and, consequently, the neutron energy spectrum. Proper moderation increases the probability of fission events in fissile isotopes, thereby improving reactivity. For this reason, each fuel cell geometry has an optimal V_m/V_f value that maximizes k_∞. To determine this optimum, a $k_\infty \times (V_m/V_f)$ curve was generated using the T-NEW sequence of the SCALE code with the V5-44 nuclear data library. This parametric study enabled the identification of the half-pitch dimension that yields the highest k_∞ , which was then adopted as the reference geometry for subsequent analyses.

After the optimal cell geometry was established, additional neutronic parameters, such as microscopic fission and absorption rates, transport coefficients, and the average number of neutrons produced per fission, were extracted to support subsequent design stages. Assuming the fuel element (FE) can be approximated as an equilateral cylinder, a common simplification used to balance neutron leakage minimization and fuel utilization [10], the diffusion equation was applied to estimate the required radius and height for the FE to achieve $k_{eff} \geq 1.30$. This criterion ensures sufficient excess reactivity at the beginning of life, providing appropriate operational and safety margins in the presence of modeling uncertainties and the progressive depletion of fissile isotopes over the reactor cycle. Although the ideal level of excess reactivity depends on the reactor type and operational constraints, the chosen margin is consistent with conceptual studies aimed at enabling long fuel cycles without compromising criticality control.

Using the SCALE code, specifically the KENO module for three-dimensional calculations, together with the V5-44 nuclear data library, the full 3D model of the fuel element was constructed, enabling accurate determination of the physical dimensions required to achieve the specified keff.

In large commercial reactors, axial enrichment zoning is commonly applied to control the axial power shape and extend the fuel cycle [11]. However, in low-power microreactors with compact cores, such strategies become unnecessary. Because the neutron mean free path is comparable to or larger than the characteristic core dimensions, the neutron flux tends to be nearly flat, with only minor differences between central and peripheral regions [12]. Therefore, maintaining a uniform PuO_2 concentration along the fuel rods does not compromise thermal margins or burnup behavior, making compositional grading unnecessary.

Another essential parameter in core design is the control and safety rod system. Reactivity control in nuclear reactors requires precise and reliable mechanisms, and safety rods play a central role in this function. Designed to absorb both thermal and fast neutrons within the core, these rods provide a direct and immediate means of regulating the chain reaction. Their insertion must be capable of driving the system to a subcritical condition ($k_{eff}<1$), promptly halting fission and ensuring reactor safety under both normal operation and emergency conditions. Consequently, the positioning, number, and absorber material of the safety rods become critical parameters in the neutronic and mechanical design of microreactor cores.

2.1. The linear power density q'

The linear power density assessment was carried out to verify whether the proposed core design can safely remove the heat generated during operation. This calculation provides a first-order consistency check between the neutronic power distribution and the fuel rod's thermal–hydraulic limits. The theoretical linear heat rate q' (Equation 4) is estimated using a one-dimensional radial conduction model that includes the thermal conductivity of the fuel and cladding (k_f), the fuel radius (R_f), the gap thickness (t_c), and the effective heat-transfer coefficients across the gap (h_g) and from the cladding to the coolant (h_s). Equation 5 relates the operating thermal power to the number and length of the fuel rods[6].

$$q' = \frac{2\pi\Delta T}{\left[\frac{1}{2k_f}+\frac{1}{R_f h_g}+\frac{t_c}{k_c R_f}+\frac{1}{h_s(R_f+t_c)}\right]} \tag{4}$$

Table 2 summarizes the average thermal parameters adopted in the simplified thermo-hydraulic evaluation.

Table 2: Thermal parameters used in the conduction and convection model

Parameter	Value
$k_{f\ (mox)}$	0.030 W/cm.K(8)
$K_{c(zircaloy)}$	0.107 W/cm.K(6)
$h_{s(water)}$	2.000 W/cm^2.K(6)

The parameters listed in Table 2 represent average values selected to provide a first-order approximation of the thermal behavior of the fuel rod under typical microreactor operating conditions, with fuel temperatures potentially reaching up to approximately 1300K. In this regime, the thermal conductivities of MOX fuel and Zircaloy cladding are treated as effective mean values, representing the integrated behavior along the temperature gradient rather than temperature-dependent correlations. Accordingly, the resulting temperature estimates should be interpreted as global approximations, suitable for the conceptual nature of the present study. A more detailed future thermo-hydraulic model may incorporate temperature-dependent material properties and axial profiles to refine these results and improve local temperature predictions.

The experimental linear power density must remain consistent with the theoretical limit to ensure that the heat generated in the fuel is effectively removed through combined conduction and convection. According to Equation 5, the linear power density is directly determined by the total thermal power, the number of fuel rods, and their active length.

$$q'_0 = \frac{P}{n.L} \tag{5}$$

This parameter also governs the rate at which the fuel is consumed and, consequently, the progression of burnup throughout operation. Once the linear power is defined, depletion calculations can be performed using the T6-DEPL module of the SCALE code, allowing the isotopic evolution of the fuel and the associated reduction in reactivity to be monitored over time. This methodology enables the estimation of the core cycle, defined as the point at which the effective multiplication factor, k_{eff}, falls below unity, indicating that the system can no longer sustain criticality without design modifications or the insertion of additional reactivity.

However, evaluating keff alone is insufficient to fully characterize reactor behavior over the burnup cycle. It must be complemented by a detailed analysis of the isotopic evolution during irradiation, as the fuel composition undergoes significant changes throughout operation. Monitoring isotopic concentrations enables identification of the phenomena of conversion, accumulation, and depletion of key nuclides that directly influence the reactor's neutronic performance and overall safety [13].

3. RESULTS AND DISCUSSIONS

MOX fuel can be formulated with different proportions of UO_2 and PuO_2, and the compositions evaluated in this study are presented in Table 3. To identify the most suitable fuel mixture for the core of a PWR-type microreactor under predefined operating conditions, two main criteria were considered: the expected cycle length and the power the core could sustain. The initial analyses were carried out using natural UO_2; if the performance obtained had not met the requirements, the plutonium content would have been gradually adjusted. However, such an adjustment was not necessary, as composition 3 exhibited the best overall performance. Therefore, because it represents the most promising configuration among all alternatives investigated, this work focuses its analyses and discussions exclusively on the results obtained for this composition.

Table 3: Proportions between UO_2 and PuO_2

	MOX composition - %	
	PuO_2	UO_2
1.	5	95
2.	10	90
3.	15	85
4.	20	80
5.	25	75
6.	30	70

The analysis begins by correcting the density equation (Equation 1) as a function of temperature, yielding Equation 6 (300 K, corresponding to the initial core temperature) and Equation 7 (1300 K, representing the operating condition).

$$\rho_{MOX}(300) = 10961 + 489.y \qquad [Kg/m^3] \tag{6}$$

$$\rho_{MOX}(1300) = 10628 + 474.y \qquad [Kg/m^3] \tag{7}$$

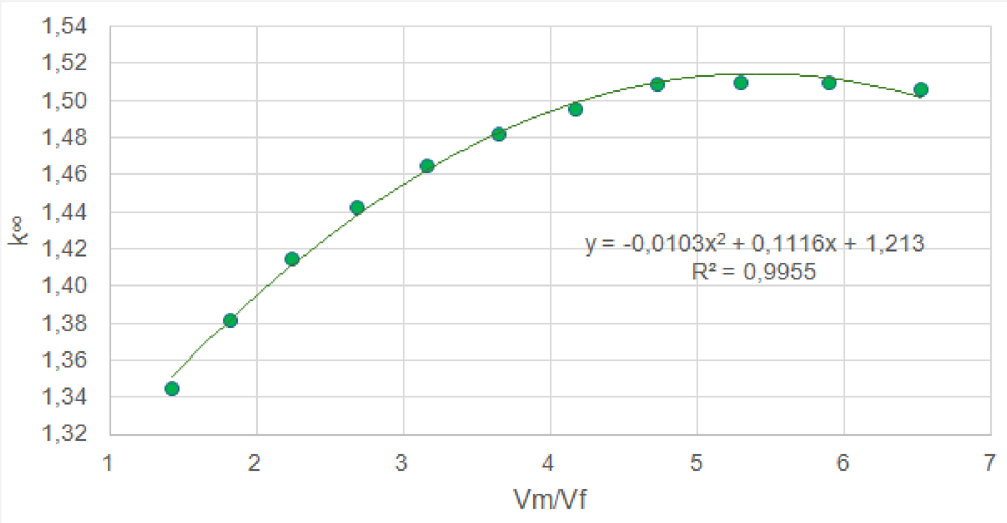
The values obtained from Equations 6 and 7 are presented in Table 4. All results are expressed in atoms/barn·cm and were calculated using Equations 2 and 3, taking into account the isotopic abundances as provided in Table 1.

Table 4: Isotopes density for 15% PuO_2 fuel.

Density$\left(\frac{atm}{b.cm}\right)$	Isotope	300k	1300k
U	U-234	0.000001	0.000001
	U-235	0.000149	0.000144
	U-238	0.020769	0.020138
Pu	Pu-238	0.000004	0.000004
	Pu-239	0.002792	0.002707
	Pu-240	0.000699	0.000678
	Pu-241	0.000097	0.000094
	Pu-242	0.000020	0.000019
O_2	O-16	0.049061	0.047571

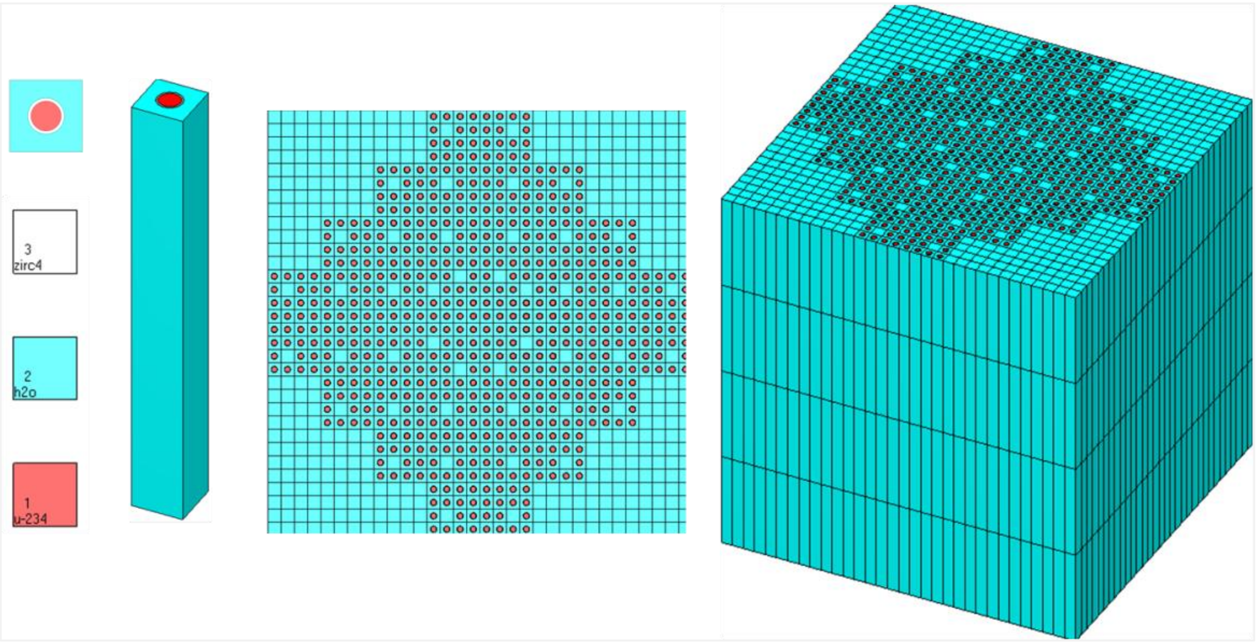
The $k_\infty \times V_m/V_f$ curve, used to determine the cell size, was obtained considering a fuel radius of 0.5 cm, a Zircaloy-4 cladding with a thickness of 0.1 cm, and light water acting as both moderator and reflector. The moderator region was varied by increasing its radius from 0.65 cm in increments of 0.05 cm until the maximum value of k_∞ was reached. The parameters adopted were based on PWR reactors, such as Angra II [9]. The resulting curve is shown in Figure 1, and the point at which the k_∞ (= 1.5098) was obtained corresponds to a *halfpitch* of 1.20 cm.

Figure 1: k_∞ versus V_m/V_f.



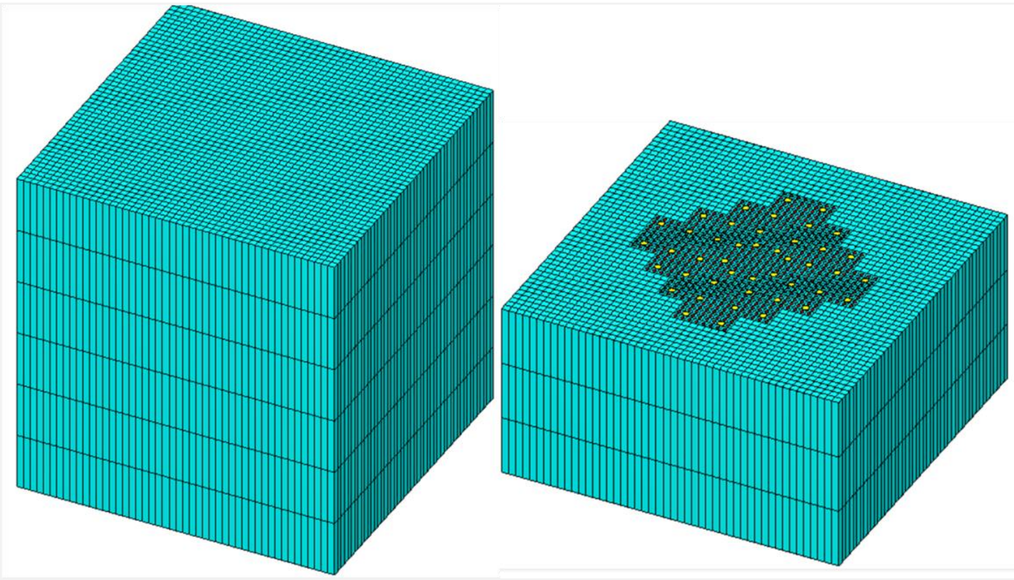
Using the cross-section values obtained from the SCALE output and considering a halfpitch of 1.20 cm, the diffusion equation was applied to estimate the required radius of the fuel element. The result indicated that a radius of approximately 41.7 cm would be sufficient to achieve a k_{eff} greater than 1.3. Based on this initial dimensioning, the fuel rods were arranged in 4×4 groups, positioned to approximate the fuel element geometry as closely as possible to a cylindrical shape. As a result of this configuration, the final fuel element consisted of an 8×8 lattice of rod groups. The radius and height corresponding to this new geometry were 84.4 cm and 78 cm, respectively. As shown in Figure 2, the proportions and overall geometry of the resulting core configuration are clearly evident.

Figure 2: Cell Proportions and Core Size Estimation



Subsequently, using the SCALE KENO application, the fuel assembly was modeled, accounting for factors that can reduce the Keff value, such as neutron leakage and limited size. To mitigate this leakage, a 19.2 cm radial and 25 cm axial layer of light water was added to act as a reflector, as illustrated in Figure 3.

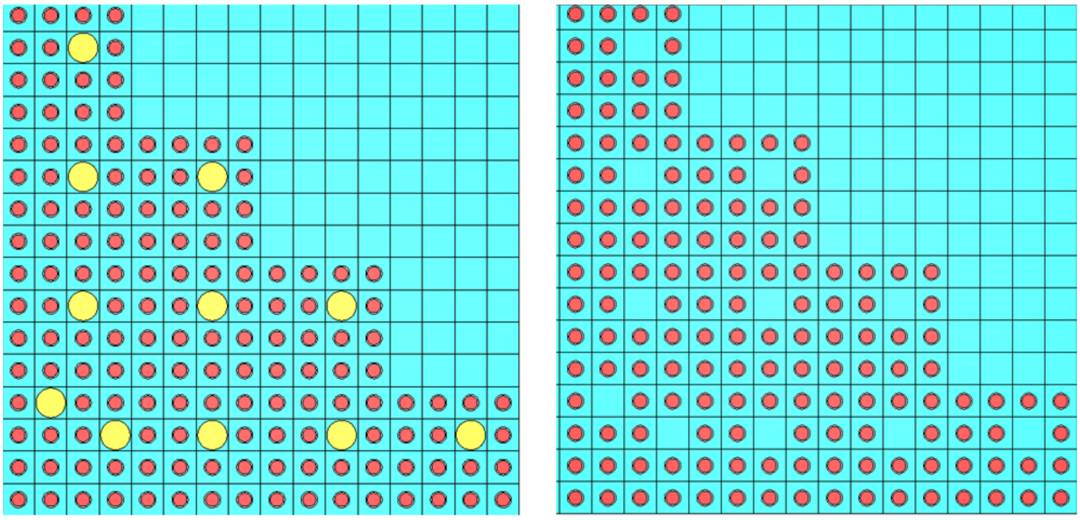
Figure 3: Size core and a horizontal cross-section at mid-height



Subsequently, boron carbide (B_4C) control rods were added, each with a radius of 1.1 cm and enriched to 90% in B-10. To render the core subcritical, a total of 44 control rods were required, distributed as 11 per quadrant, as illustrated in Figure 4a. In this configuration, the calculated keff was 0.9737 ± 0.0019.

During reactor operation, however, the control rods remain withdrawn; therefore, the space they would occupy is filled with moderator. Under this condition, the k_{eff} increases to 1.2942±0.0019, considering a total of 596 fuel rods, as shown in Figure 4b.

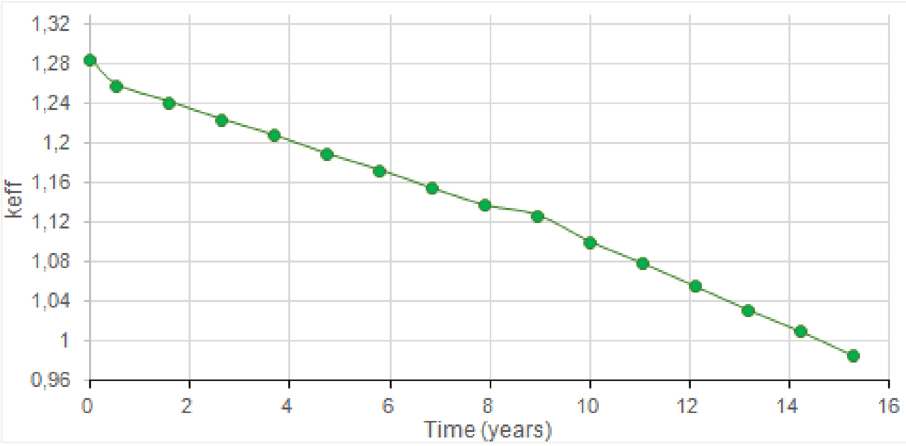
Figure 4: Top View of the Fuel Element Configuration



(a) (b)

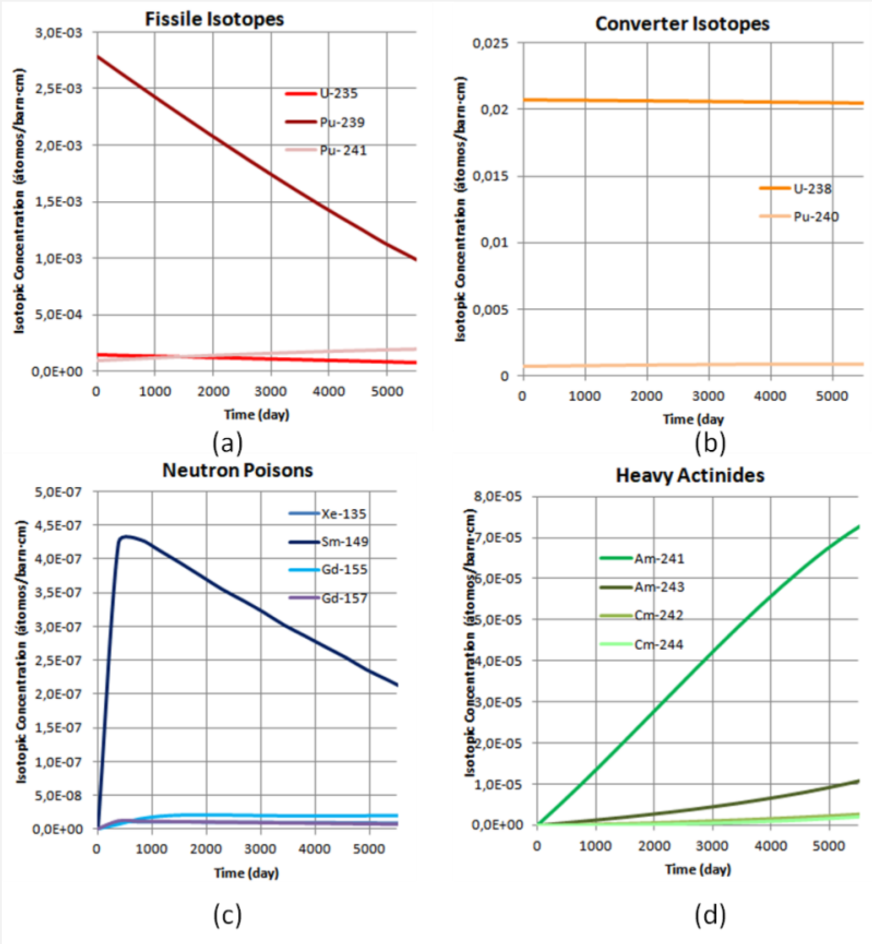
Equation 4 indicates that the maximum linear power is approximately 300 W/m, while Equation 5 shows that the burnup can be performed at a maximum thermal power of 13.3 MWt. Thus, when the depletion calculations are carried out assuming an operating power of 12 MWt, the cycle can be extended to about 15 years without refueling. The burnup behavior over time is presented in Figure 5.

Figure 5: K_{eff} Value Over the Years Extracting 15 MWt



In this study, the following isotopic groups of interest were monitored: (a) fissile isotopes, represented mainly by U-235, Pu-239, and Pu-241, which are directly responsible for energy generation through fission; (b) fertile and converter isotopes, such as U-238 and Pu-240, which, upon capturing neutrons, form new fissile nuclei and contribute to the sustainability of the fuel cycle; (c) neutron poisons, including Xe-135, Sm-149, Gd-155, and Gd-157, which are fission products or burnable absorbers with high neutron-capture cross sections that progressively reduce core reactivity; and (d) heavy actinides, such as Am-241, Am-243, Cm-242, and Cm-244, formed through successive neutron captures and decay processes, which influence reactivity and increase the radiotoxicity of the irradiated fuel. The results obtained for each isotopic group are presented in Figure 6.

Figure 6: Evolution of isotopic concentrations over the cycle for the fuel with 15% PuO_2: (a) fissile isotopes, (b) fertile and converter isotopes, (c) neutron poisons, and (d) heavy actinides.



Throughout the burnup cycle, the fissile isotopes U-235, Pu-239, and Pu-241 exhibit distinct behaviors: Pu-239 undergoes the greatest reduction due to its intense participation in fission reactions, U-235 is consumed more gradually because it is present in smaller proportions, and Pu-241 increases steadily as it is produced through neutron capture in Pu-240. Among the fertile isotopes, U-238 and Pu-240 show minimal variation, reflecting a balance between production and consumption that is characteristic of systems with partial breeding. The nuclear poisons display specific trends: Sm-149 rises rapidly at the beginning of the cycle and then decreases slowly due to the dynamic equilibrium between its production and neutron capture; Xe-135 remains nearly constant given the low power of the microreactor; and the gadolinium isotopes (Gd-155 and Gd-157) maintain low and nearly flat concentrations, consistent with their slow burnup as long-term absorbers. Finally, the heavy actinides (Am-241, Am-243, Cm-242, and Cm-244) accumulate progressively over the cycle, with Am-241 showing the most pronounced increase due to the decay of Pu-241. This buildup of non-fissile actinides contributes to the gradual reduction in core reactivity and increases the radioactivity and toxicity of the irradiated fuel.

Finally, we present a summary (Table 5) of the key information from the proposed design for the core of a PWR microreactor, using MOX fuel with natural uranium.

Table 5: Core characteristics

Fuel radius	0.5 cm
Clad thickness	0.1 cm
Fuel composition	MOX 85% natural UO_2 15% PuO_2
Clad material	Zircaloy
Cell geometry	Square
Pitch	2.4 cm
Equivalent active radius	38.4 cm
Number of fuel rods	596
Fuel assembly	8 X 8
Active height	78 cm
Core volume	361 liters
Initial reactivity (300 K)	33,770 pcm
Cycle time	13 years
Thermal power	12 MWt
Linear power density	300 W/m
Number of absorber rods	44
Absorber rod composition	B_4C
Absorber rod radius	1.1 cm
Reflector (light water)	Radial: 19.2 cm; Axial: 25 cm

4. CONCLUSIONS

In conclusion, the development of microreactors represents a promising response to the growing global demand for clean, reliable, and decentralized energy sources. Their compactness, inherent safety characteristics, and operational flexibility make them strong candidates for reducing carbon emissions and supporting long-term environmental goals. Consequently, continued research and technological investment in microreactor concepts remain essential for a more sustainable and resilient energy future.

The choice of fuel plays a central role in the feasibility of these systems. While uranium oxide is conventionally used in PWR-type reactors, MOX fuel, produced through the reprocessing of irradiated fuel, emerges as a compelling alternative, offering strategic advantages such as reducing the need for uranium enrichment and minimizing long-term plutonium stockpiles. In this context, the main objective of this work was to demonstrate that a PWR-type microreactor core can be effectively designed using MOX fuel without any uranium enrichment. This represents a significant benefit, given that enrichment is costly and technologically restricted to only a few countries. The use of plutonium therefore provides a dual advantage: decreasing existing inventories and preserving natural uranium resources, especially the fissile isotope U-235.

The calculations performed confirmed the feasibility of achieving a significantly extended burnup cycle, exceeding ten years of continuous operation at a thermal power of 12 MWt (approximately 4 MWe). This performance is attained while maintaining a compact core with reduced dimensions and simplified geometry, preserving the compactness and operational efficiency required for the proposed microreactor concept.

The main characteristics of the selected configuration include fuel pellets with a radius of 0.5 cm, Zircaloy cladding with a thickness of 0.6 cm, and light water serving as both moderator and coolant. The geometry of the arrangement, based on a reduced active volume, proved effective for both reactivity control and heat removal.

Finally, the results obtained reinforce the potential of MOX fuel as an innovative option for microreactors, promoting greater autonomy, efficiency, and sustainability for civil, remote, or strategic applications. The following research steps include coupled thermo-hydraulic analyses, a detailed investigation of reactivity coefficients, and the evaluation of control and shielding systems, to advance toward a complete and applicable conceptual design. The ongoing development of these technologies may consolidate microreactors' role as a viable alternative in the transition to a low-carbon energy matrix.

CONTRIBUTORSHIP

Methodology: A. C. N. R.

Conceptualization: S. O. V.

Validation: A. C. N. R.

Formal analysis: A. C. N. R.

Investigation: A. C. N. R.

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Writing – original draft: A. C. N. R.

Writing – review & editing: S. O. V.

CONFLICT OF INTEREST

All authors declare that they have no conflicts of interest.

DATA AVAILABILITY STATEMENT

The authors declare that the data supporting the results of this study are available in the article. Derived data supporting the conclusions of this study are available upon request from the corresponding author.

REFERENCES

[1] BRASIL. *Comissão Nacional de Energia Nuclear*. Com apoio da CNEN, Brasil avança no desenvolvimento de microrreatores nucleares. Portal Gov.br, 18 mar. 2025. Available at: https://www.gov.br/cnen/pt-br/assunto/ultimas-noticias/com-apoio-da-cnen-brasil-avanca-no-desenvolvimento-de-microrreatores-nucleares. Accessed on: 21 abr. 2025.

[2] *Associação brasileira de energia nuclear*. Combustível nuclear irradiado: um recurso valioso – não um resíduo. Available at: https://aben.org.br/combustivel-nuclear-irradiado-um-recurso-valioso-nao-um-residuo/. Accessed on: 21 abr. 2025.

[3] *International atomic energy agency*. Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management. Nuclear Energy Series No. NF-T-3.8. Viena: IAEA, 2011.

[4] S. G. Popov et al., *Thermophysical Properties of MOX and UO2 Fuels Including the Effects of Irradiation*. Oak Ridge National Laboratory, 2000.

[5] *National institute of standards and technology*. Atomic weights and isotopic compositions. Available at: https://physics.nist.gov/cgi-bin/Compositions/stand_alone.pl. Accessed on: 21 abr. 2025.

[6] Duderstadt, James J., e Louis J. Hamilton. *Nuclear Reactor Analysis. Department of Nuclear Enginnering*. The University of Michigan, 1975

[7] Kaplan, Irving. *Nuclear Physics*. *Aguilar*, 1961.

[8] International Atomic Energy Agency. *Advances in Thermal Hydraulics: Proceedings of an International Conference*. Viena: IAEA, 2022. Available on: https://www-pub.iaea.org/MTCD/Publications/PDF/IAEA-THPH_web.pdf.

[9] ELETRONUCLEAR. *Informações de Angra 2*. Available on: https://www.eletronuclear.gov.br/Nossas-Atividades/Paginas/Informacoes-de-Angra-2.aspx. Accessed on: 21 abr. 2025.

[10] LAMARSH, J. R.; BARATTA, A. J. *Introduction to Nuclear Engineering*. 3. ed. Upper Saddle River: Prentice Hall, 2001. Cap. 8.

[11] TODREAS, N. E.; KAZIMI, M. S. Nuclear Systems Volume I: Thermal Hydraulic Fundamentals**. 2.** ed. Boca Raton: CRC Press, 2011.

[12] VUJIC, J. L. et al. Small Modular Reactors: Simplified Designs, Safety Benefits, and Economic Advantages**.** *Energy*, v. 45, p. 288–295, 2012.

[13] FRANCIS, M. W. *Reactor Fuel Isotopics and Code Validation for Nuclear Applications*. Oak Ridge: Oak Ridge National Laboratory, 2014. ORNL/TM-2014.

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