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BRAZILIAN JOURNAL OF RADIATION SCIENCES

doi.org/10.15392/2319-0612.2024.2630
2024, 12(4B) | 01-25 | e2630
Submitted: 2024-24-08
Accepted: 2025-05-06

Closed nuclear fuel cycle strategy for NuScale-like reactor: safety parameters evaluation

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Abstract: The present study evaluates the potential implementation of two different closed fuel cycle strategies for a NuScale-like reactor core. After undergoing three burnup cycles of approximately 12 MWd/kgU in the NuScale-like initial core and five years of cooling in a spent fuel pool, the spent fuel was theoretically reprocessed using GANEX or UREX+ methods. These reprocessed fuel compositions were then mixed with thorium (Th) and inserted into specific batch positions of the core. As a result, the proposed NuScale-like core configurations contain fuel assemblies loaded with both conventional uranium-based fuel and reprocessed fuel, resulting in the following combinations: UO_2 and GANEX spiked with Th, and UO_2 and UREX+ spiked with Th. The primary goal is to assess the safety margins of the proposed cores and compare them with the reference case. The results indicate that all scenarios with reprocessed fuel improved the fuel temperature reactivity coefficient and maximum initial excess of reactivity, while varying the boron concentration in the coolant. Additionally, it was found that both proposed cores met the power peak factor (PPF) design requirements, despite the high PPF value observed at the C03 central position. The simulations were performed using the Serpent code version 2.1.32, developed by VTT.

Keywords: SMR, NuScale, GANEX, UREX+, closed nuclear fuel cycle.

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Estratégia de ciclo fechado de combustível nuclear para um reator do tipo NuScale: avaliação de parâmetros de segurança

Resumo: O presente estudo avalia a implementação potencial de duas estratégias diferentes de ciclo de combustível fechado para o reator NuScale. Após passar por três ciclos de queima de aproximadamente 12 MWd/kgU em um núcleo do tipo NuScale e cinco anos de resfriamento em uma piscina para refrigeração, o combustível irradiado foi reprocessado teoricamente usando métodos de reprocessamento GANEX ou UREX+. Estas composições de combustível reprocessado foram então misturadas com tório (Th) e inseridas em posições específicas do núcleo. Como resultado, as configurações de núcleo propostas para o reator do tipo NuScale contêm elementos combustíveis carregados com combustível convencional à base de urânio e combustível reprocessado, resultando nas seguintes combinações: UO_2 e GANEX enriquecidos com Th, e UO_2 e UREX+ enriquecidos com Th. O objetivo principal é avaliar as margens de segurança dos núcleos propostos e compará-las com o caso de referência. Os resultados indicam que todos os cenários com combustível reprocessado melhoraram o coeficiente de reatividade de temperatura do combustível e o excesso máximo inicial de reatividade, variando a concentração de boro no refrigerante. Adicionalmente, constatou-se que ambos os núcleos propostos atenderam aos requisitos de projeto com relação ao fator de pico de potência (PPF), apesar do alto valor do PPF observado na posição central C03. As simulações foram realizadas utilizando o código Serpent versão 2.1.32, desenvolvido pela VTT.

Palavras-chave: SMR, NuScale, GANEX, UREX+, ciclo do combustível nuclear fechado.

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1. INTRODUCTION

The so-called small modular reactors (SMRs) are nuclear reactors with a power capacity of up to 300 MWe per unit [1]. Compared to large nuclear power plants, they feature simpler designs, smaller structures, and flexible power generation capabilities. This makes them suitable not only for large-demand countries but also for regions with smaller and mixed-capacity electric grids [2,3,4]. Additionally, the modularization of systems allows components to be manufactured and assembled at a factory, then transported as a unit for on-site installation and operation.

In terms of safety, SMR projects mostly rely on inherent safety features, passive systems, and lower operating parameters such as thermal power and pressure drop. This approach diminishes the need for operator intervention or external power to manage reactivity insertion during various abnormal scenarios [5]. The referred characteristics represent an essential increase in the operational reliability of the facility since critical systems operate based on common physical phenomena such as natural circulation, gravity, and self-pressurization. That results in safety margins higher than those achievable by external or engineered controls, ensuring a safe and reliable self-shutdown of the reactor [6].

Currently, the most mature SMR design is the NuScale Power Modular and Scalable Reactor, developed by NuScale Power LLC, being the first project to receive approval and certification by the US Regulatory Commission (USNRC) [7]. Besides incorporating the aforementioned improvements from SMRs concept, the NuScale core also provides versatility regarding nuclear fuel composition. Such versatility includes allowing the utilization of uranium-based fuel at different enrichment levels in an OUT-IN loading pattern strategy or even a uranium-plutonium mixture of oxide fuel (MOX) with minimal

effect on the reactor's operation by design [8], attaining for a closed nuclear fuel cycle strategy implementation.

In order to explore new possibilities for fueling SMRs, the present paper proposes the implementation of two different closed nuclear fuel cycle strategies for the NuScale reactor core. To achieve that, we utilized a neutronic benchmark [9] to simulate the first NuScale-like burnup cycle, taking it as a reference case. Subsequently, we simulated the second and third burnup cycles to attain equilibrium in the NuScale core according to [7]. Following approximately 12 MWd/kgU over three burnup cycles and a five-year cooldown period, the spent fuel composition of the NuScale-like core was theoretically reprocessed using GANEX [10] or UREX+ [11] methods.

The two reprocessed fuel compositions were mixed with thorium fertile material to reduce the initial reactivity caused by the high levels of fissile material in the reprocessed fuel. Subsequently, these fuels compositions were charged into specific batch positions within the core limited to one third of the total fuel assemblies. As a result, the proposed NuScale-like core configurations include fuel assemblies loaded with both conventional uranium-based fuel and reprocessed fuel, resulting in the following combinations: enriched UO_2 and GANEX spiked with Th and enriched UO_2 and UREX+ spiked with Th.

To evaluate the effects of the reprocessed fuel load in the NuScale-like core, the initial safety parameters were analyzed as a primary requirement and compared to the reference case containing only conventional enriched uranium. These safety parameters evaluation included fuel and moderator reactivity temperature coefficients, effective multiplication factor as function of boron concentration in the coolant and power peak factor. The aim is proving the proposed modification against chemical dependency and core resistance to reactivity-initiated accident. We also assessed the kinetic parameters since they are vital for the effective control of fission reactions within the reactor core, mainly maximizing the

reactor period with a greater delayed neutron emission rate. The Monte Carlo Serpent code version 2.1.32 [12] developed by VTT was used to perform the simulations.

2. MATERIALS AND METHODS

2.1. NuScale-like (reference case)

In this work, we simulated the NuScale-like core according to the existing benchmark [9] and established it as the reference case. The reactor has an active length of 200 cm and outer radii of 99.06 cm and the coolant flow through the core is driven by natural circulation, avoiding pumps in the primary system. The simulated power output is equal to 160 MW_{th}.

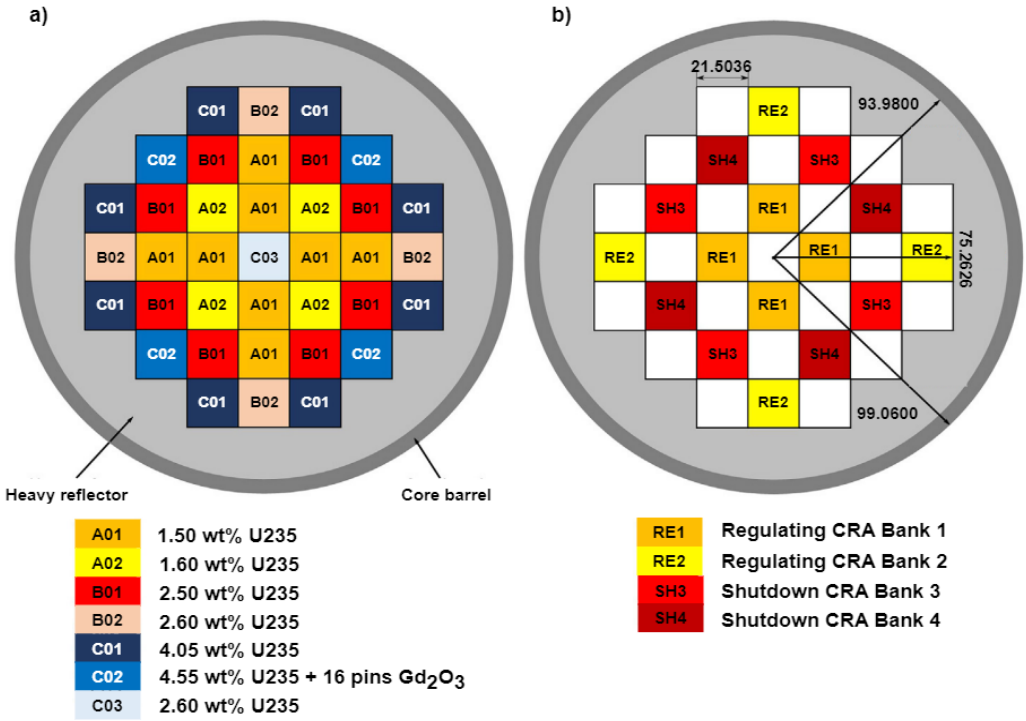
Figure 1 shows the (a) core loading pattern and (b) control rod locations. As can be seen in Figure 1 (a), the core comprises A, B, and C assemblies' batches, in which A and B split into 01 and 02 positions, with eight and four fuel assemblies respectively. The C assembly group is divided into 01, and 02 positions as well, but with a center position (C03) in addition, resulting in 37 fuel assemblies. All fuel assemblies are loaded with fresh uranium-based fuel at six different enrichment levels, which results in an average enrichment of 2.95 wt% ^{235}U. Table 1 presents the isotopic composition of each fuel type in weight fraction.

The control rod assemblies (CRAs) located at B01, B02, and A01 positions manage the core reactivity through the regulating (RE) and shutdowns (SH) banks (see Figure 1 (b)), being structurally composed of two different absorber regions, namely Ag-In-Cd (AIC) and B_4C. These materials were adopted from BEAVRS benchmark specifications [13]. Each regulating and shutdown CRAs include two banks capable of moving independently. The entire core is surrounded by a heavy steel reflector to help in neutron economy and bounded by a cylindrical barrel.

To cope with initial excess reactivity, the fuel assemblies located at C02 positions are loaded with integral burnable poison (BP) in the content of a homogeneous mixture of UO_2 and gadolinia (Gd_2O_3). Figure 2 shows the radial layouts of the lattices of un-poisoned (left) and poisoned (right) fuel assemblies, highlighting fuel rods with BP as the red ones. Both lattices have typical 17x17 fuel rod positions, with a pitch inside the assembly of 1.2598 cm and an assembly pich inside the core of 21.5036 cm.

Besides the 264 fuel rods, the fuel assembly also contains 24 guide tubes (GT) into which control rods can be inserted, and a central instrumentation tube, modeled as an empty guide tube. All materials and dimensions of the simulated core are described in details in [9]. To validate the NuScale model, we took the referred neutronic benchmark for the NuScale and modeled the reactor core at six different conditions for comparison, varying the insertion of control rod banks. The validation results are exhibited in subsection 3.1.

Figure 1: (a) Core loading pattern and (b) control rod assembly locations.

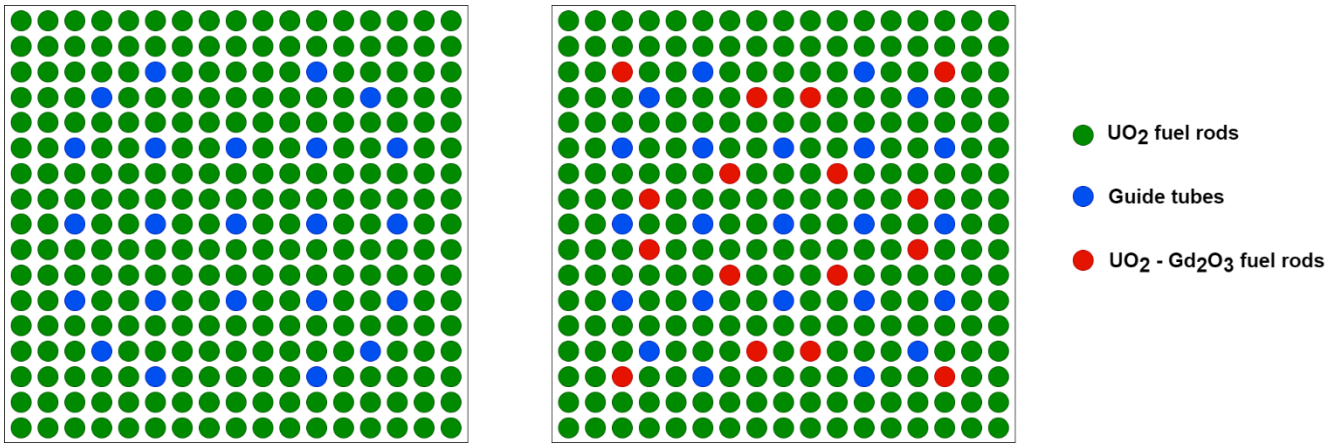


Source: (FRIDEMAN, 2023) [9].

Table 1: Initial isotopic composition of all fuel types in weight fraction [9].

Isotope	A01	A02	B01	B02/C03	C01	C02
U-235	1.32228E-02	1.41043E-02	2.20376E-02	2.29191E-02	3.57001E-02	3.68989E-02
U-238	8.68294E-01	8.67411E-01	8.59466E-01	8.58583E-01	8.45782E-01	7.74066E-01
O-16	1.18483E-01	1.18485E-01	1.18495E-01	1.18498E-01	1.18517E-01	1.19632E-01
Gd-152	-	-	-	-	-	1.34100E-04
Gd-154	-	-	-	-	-	1.48094E-03
Gd-155	-	-	-	-	-	1.01195E-02
Gd-156	-	-	-	-	-	1.40867E-02
Gd-157	-	-	-	-	-	1.08389E-02
Gd-158	-	-	-	-	-	1.73134E-02
Gd-160	-	-	-	-	-	1.54296E-02

Figure 2: Radial layouts of fuel assemblies: un-poisoned (left) and poisoned (right).

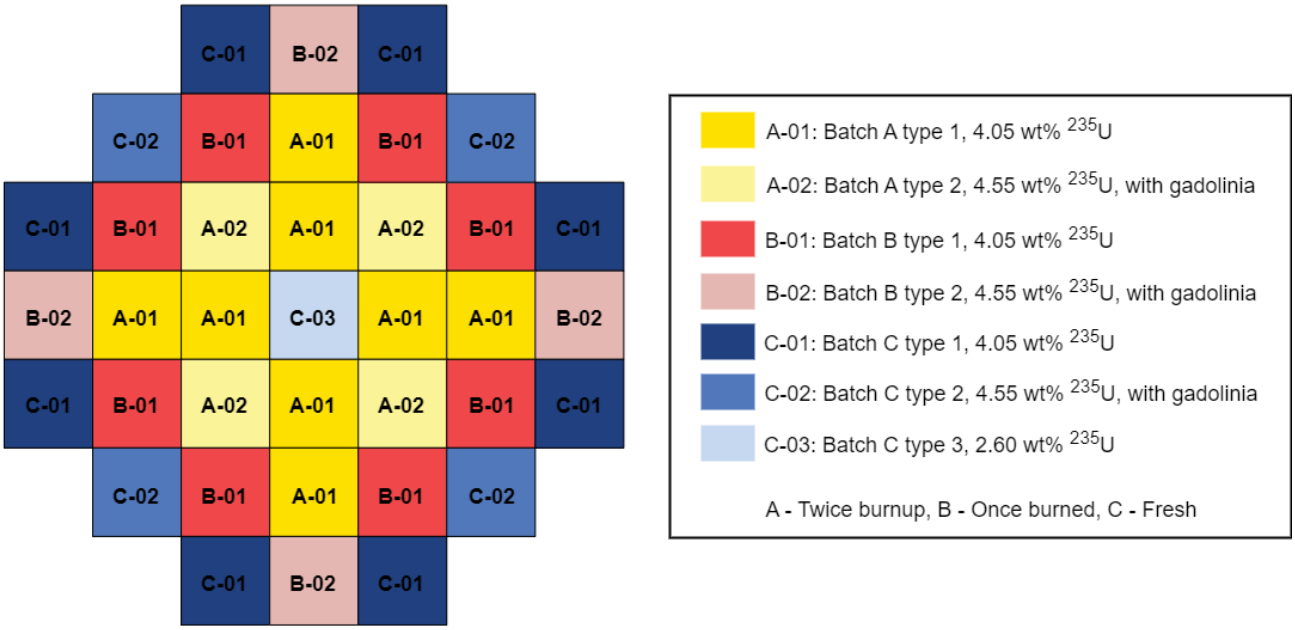


Source: (FRIDEMAN, 2023) [9].

The 24-month burnup cycle of the NuScale-like core was simulated using the SERPENT Monte Carlo code. Following that, the core configuration was modified for a second burnup cycle, shuffling C01 burned fuel assemblies to B01 positions and C02 fuel assemblies to B02 positions. B01 and B02 fuel assemblies were similarly reassigned to A01 and A02 positions, while the A01 and A02 fuel assemblies were discharged from the core. Fresh fuel assemblies were then placed in the C01, C02, and C03 positions according to their respective isotopic compositions described in Table 1.

Subsequently, the core configuration for the third cycle featured a distinct layout. This new configuration included fuel assemblies that had been burned twice in the A01 and A02 positions, assemblies burned once in the B01 and B02 positions, and fresh fuel assemblies in the C01, C02, and C03 positions. This setup corresponds to the NuScale equilibrium core described in the NuScale Final Safety Analysis Report (FSAR) [7], as illustrated in Figure 3. Each burnup cycle achieved approximately 12 MWd/kgU.

Figure 3: Loading pattern reference equilibrium cycle.



Source: (USNRC, 2022) [7].

2.2. Fuel cooldown

The next step of the methodology involves transferring the eight discharged fuel assemblies from the A01 positions to a cooling pool for a period of five years. This step aims to observe the changes in the spent fuel composition due to nuclear reactions involving fission products and the decay of short-lived radioactive nuclides into more stable elements, which typically reduces the radiotoxicity of the spent fuel.

2.3. Fuel reprocessing

After three cycles of burnup and a cooling time of five years, the three-times-burned spent fuels located in the A01 positions were theoretically reprocessed by GANEX or UREX+ reprocessing techniques. These methods are well established as non-proliferation processes since they do not separate plutonium from the transuranic elements during reprocessing steps, avoiding any clandestine use of the mentioned nuclear material [10].

The GANEX process developed by *Commissariat à l´énergie atomique et aux energies alternatives* (CEA) for reprocessing Generation IV spent nuclear fuels is composed of two extraction cycles following the dissolution of the spent fuel. Once the uranium is selectively extracted from the dissolution solution by a monoamide solvent, the transuranic elements (Np, Pu, Am, and Cm) are separated from the fission products in a second cycle before the co-conversion step [11].

In terms of GANEX results, according to [15], neptunium, plutonium, americium, and curium were recovered altogether in one liquid flow and the losses were estimated at a value lower than 0.5% (neptunium essentially), corresponding to a recovery yield of actinides higher than 99.5%. The decontamination factors versus some lanthanides (especially Nd, Sm, and Eu) were much lower than expected and the mass of lanthanides in the actinide product was around 5% at the end. The amount of uranium after the reprocessing is 0.01% of the total amount of uranium in the spent fuel. The isotopic compositions after the GANEX reprocessing of A01 spent fuel assemblies were spiked with 74.50% of thorium.

Other reprocessing technique used was the UREX+. The UREX, a variant of the Plutonium Uranium Reduction Extraction (PUREX) process, is a series of five solvent-extraction flow sheets that separate uranium from spent fuel without recovering plutonium in a pure mixture. UREX+ is an improvement of UREX because it also extracts plutonium mixed with some minor actinides [12, 16]. The percentages of recovered isotopes are as follows: 99.95% of U, 99.50% of Pu, 71% of Np, 98% of Am, and 79% of Cm from the

matrix of spent fuel. After the UREX+ reprocessing of A01 spent fuel assemblies, the isotopic compositions were spiked with 74.67% of thorium.

The approach of incorporating reprocessed fuel compositions into different mixtures is driven by the need to reduce initial reactivity due to the high concentration of fissile material in the reprocessed fuel, particularly isotopes such as ^{235}U, ^{239}Pu, and ^{241}Pu. Fertile materials are commonly used for this purpose, not only to achieve the desired reduction in initial reactivity but also for further breeding nuclear fuel, capitalizing on their significant ability to produce fissile isotopes during reactor operation.

^{232}Th content is an attractive alternative compared to ^{238}U since thorium is about three times more abundant than uranium and is distributed in nature as an easily exploitable resource in many countries [17]. Moreover, as fertile material, the ^{232}Th isotope can transmute by neutron absorption and two beta decays in ^{233}U, which is one of the uranium fissile isotopes capable of undergoing fission reactions [18].

In this study, the thorium percentages were adjusted to match the initial effective multiplication factor (k_{eff}) of the NuScale-like core taken as the reference case. This adjustment needed slightly higher percentages of reprocessed fuel in cases utilizing the GANEX method. Such requirement is due to the higher recovery rates of GANEX for minor actinides with significant absorption cross-sections compared to the UREX+ method. Table 2 displays the final composition of the reprocessed fuel using GANEX reprocessing, and Table 3 shows the final composition using UREX+ reprocessing.

Table 2: Composition of reprocessed fuel spiked with thorium after GANEX and UREX+ process obtained from A01 three-times-burned spent fuel assemblies.

GANEX spiked with 74.50% of thorium				**UREX+ spiked with 74.67% of thorium**			
Isotope	**Weight frac.**	**Isotope**	**Weight frac.**	**Isotope**	**Weight frac.**	**Isotope**	**Weight frac.**
Th-232	6.55316E-01	Am-241	7.89474E-03	Th-232	6.56865E-01	Am-241	7.61478E-03
U-232	1.74467E-13	Am-242	1.99097E-05	U-232	8.58572E-13	Am-242	1.92036E-05
U-233	3.45182E-12	Cm-242	2.06620E-07	U-233	1.69868E-11	Cm-242	1.60655E-07

GANEX spiked with 74.50% of thorium				UREX+ spiked with 74.67% of thorium			
Isotope	**Weight frac.**	**Isotope**	**Weight frac.**	**Isotope**	**Weight frac.**	**Isotope**	**Weight frac.**
U-234	2.57294E-08	Cm-243	7.73398E-06	U-234	1.26617E-07	Cm-243	6.01344E-06
U-235	2.18822E-05	Cm-244	6.96071E-04	U-235	1.07685E-04	Cm-244	5.41220E-04
U-236	9.89423E-06	Cm-245	5.73651E-05	U-236	4.86906E-05	Cm-245	4.46034E-05
U-237	6.91938E-14	Cm-246	5.42278E-06	U-237	3.40510E-13	Cm-246	4.21640E-06
U-238	1.84346E-03	Cm-247	6.88836E-08	U-238	9.07189E-03	Cm-247	5.35594E-08
U-240	1.01195E-21	Cm-248	4.69677E-09	U-240	4.97990E-21	Cm-248	3.65191E-09
Pu-236	2.20006E-10	Cm-250	2.20479E-16	Pu-236	2.15452E-10	Cm-250	1.71430E-16
Pu-237	3.91182E-21	Np-235	1.74259E-12	Pu-237	3.83085E-21	Np-235	1.22384E-12
Pu-238	4.35568E-03	Np-236	4.42317E-08	Pu-238	4.26552E-03	Np-236	3.10643E-08
Pu-239	1.14943E-01	Np-237	1.15540E-02	Pu-239	1.12564E-01	Np-237	7.31095E-03
Pu-240	4.73297E-02	Np-238	3.60385E-12	Pu-240	4.63500E-02	Np-238	2.53102E-12
Pu-241	2.21785E-02	Np-239	2.13428E-09	Pu-241	2.17195E-02	Np-239	1.49893E-09
Pu-242	1.09806E-02	Nd-144	8.58666E-04	Pu-242	1.07533E-02	Nd-144	-
Pu-243	2.45597E-18	Sm-150	1.68929E-04	Pu-243	2.40514E-18	Sm-150	-
Pu-244	5.12367E-07	Eu-152	2.86118E-05	Pu-244	5.01762E-07	Eu-152	-
Am-243	2.49195E-03	O-16	1.20382E-01	Am-243	2.40358E-03	O-16	1.20310E-01
Sum = 1.0E+00				Sum = 1.0E+00			

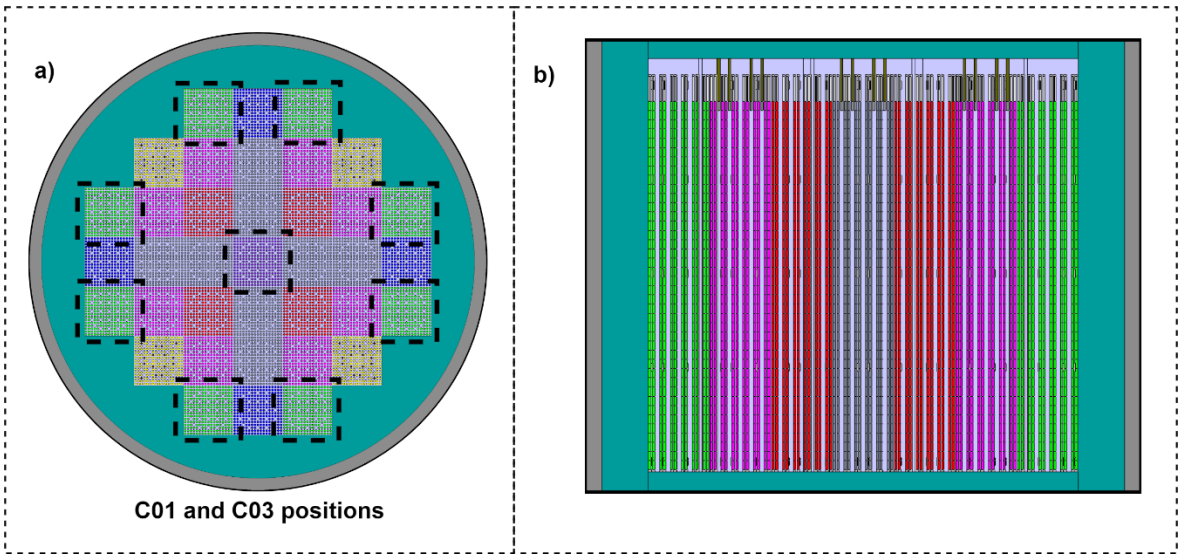
2.4. In-core distribution of fuel assemblies containing reprocessed fuel

After obtaining the reprocessed fuel composition, two main aspects were taken into consideration to decide upon optimum fuel loading patterns containing the new compositions proposed in subsection 2.3. It includes replacing fuel assemblies to a maximum of one-third of the core in order to have minimal impact on reactor`s safety parameters and maximizing the amount of reprocessed fuel usage.

Figure 4 displays the NuScale-like geometry generated in the SERPENT Monte Carlo code. Figure 4 (a) shows the horizontal cross-section of the core and indicate the proposed core configuration aligned with the constraints studied in the present work. It involves the fuel reprocessed from A01 spent fuel assemblies charged in C01 batch positions along with

C03 central position. Figure 4 (b) depicts a vertical cross-section of the core, where it is possible to verify the entire depletion zone composed by the fuel assemblies.

Figure 4: NuScale-like geometry: (a) horizontal and (b) vertical cross-sections.



Therefore, aside from the uranium-based NuScale-like reference core, two cases were simulated, as itemized bellow to facilitate the understanding:

- **Case 1:** fuel reprocessed from A01 spent fuel assemblies by the GANEX process and spiked with 74.50% of thorium, inserted at C01 and C03 positions. (Mass of reprocessed fuel = 2.45E+03 kg).
- **Case 2:** fuel reprocessed from A01 spent fuel assemblies by the UREX+ process and spiked with 74.67% of thorium, inserted at C01 and C03 positions. (Mass of reprocessed fuel = 2.45E+03 kg).

In other words, the proposed core configurations were simulated with reprocessed fuel whether using GANEX or UREX+ and spiked with thorium. Their corresponding safety margins were compared to the reference case in order to investigate the feasibility of implementation of closed nuclear fuel cycle strategies for the NuScale-like core. This safety analysis included reactivity feedback coefficients, chemical dependency of borated

water for core resistance to reactivity-initiated accident, power peak factor (PPF) and kinetics parameters.

The SERPENT Monte Carlo code version 2.1.31 [19] and ENDF/B-VII nuclear data library [20] were used to perform all simulations. SERPENT is a probabilistic Monte Carlo code used for a variety of applications in reactor physics, which includes steady-state simulations of reactor cores.

For fuel materials, cross-section libraries available in SERPENT were specified at hot full-power working temperatures, which were 900 K for containing fissile/fissionable materials and 600 K for the remaining materials. The thermal power generation was settled as 160 MW_{th}, as indicated by the FSAR of NuScale approval and certification [7], and the core were simulated with all control rods fully withdrawn. In the developed model for burnup calculations, we adopted an upper limit for the standard deviation at about 30 pcm, which required strict simulation parameters, such as 100 inactive cycles and 200 active cycles, a total of 200,000 neutron histories per cycle, and partitioning fuel material zones into 40 regions to accurately obtain volume and total mass of the materials.

3. RESULTS AND DISCUSSIONS

3.1. NuScale-like core (reference case)

The present study consists of converting the NuScale-like reactor core from a conventional UO_2 core to a core containing also reprocessed fuel to achieve a closed nuclear fuel cycle for this type of reactor. To validate the NuScale model, we took the neutronic benchmark for the NuScale reactor [9] and modeled the reactor core at six different conditions, varying the insertion of control rod banks positioned as depicted in Figure 3.

The obtained results are compared to the reference case in Table 3. It is notable the acceptable compatibility between the reference results and the obtained values from the present model, with a maximum relative difference of 262 pcm for effective multiplication factor. This is an acceptable value given that the output from the benchmark reference case averages several independent simulations performed in the framework of the *Euraton McSAFER* project [21].

Table 3: Parameters for validation of the reference reactor modeling [22].

	benchmark [9]		**calculation**		
Core state	k_{eff}	σ_{eff}	k_{eff}	σ_{eff}	**Relative diff (pcm)**
All rods out	1.02763	0.00010	1.02636	0.00019	123
RE1 in	1.00726	0.00010	1.00580	0.00020	144
RE2 in	1.00308	0.00010	1.00133	0.00020	174
SH3 in	0.98973	0.00010	0.98828	0.00018	146
SH4 in	0.98976	0.00010	0.98892	0.00019	84
All rods in	0.85794	0.00010	0.85569	0.00023	262

3.2. Safety parameters evaluation

Two of the most important parameters regarding the safe operation of a nuclear reactor involve the reactivity insertion in response to variations in the average temperature of the fuel and moderator material. These parameters are commonly called feedback effects and play a crucial role in reactor operation. Stated mathematically, such feedback effects are expressed as shown in Equation (1) and Equation (2), respectively [23]:

$$\alpha_f = \left(\frac{\partial \rho}{\partial T_f}\right) \approx \left(\frac{\Delta \rho}{\Delta T_f}\right) \quad \textbf{(1)}$$

$$\alpha_m = \left(\frac{\partial \rho}{\partial T_m}\right) \approx \left(\frac{\Delta \rho}{\Delta T_m}\right) \quad \textbf{(2)}$$

where T_f is the average fuel temperature, T_m is the coolant average temperature of the core and $d\rho$ is the incremental change in reactivity due to the above-mentioned feedback effects. To investigate the feasibility of the proposed cores, Tables 4 and 5 present a comparison of the fuel and moderator temperature reactivity coefficients for the reference case and both homogeneous and heterogeneous cores. We defined a step of $\Delta T_f = 100$ K to vary the fuel temperature while maintaining the coolant temperature constant to calculate fuel temperature reactivity. Additionally, we utilized a step of $\Delta T_m = 10$ K to vary the coolant temperature while keeping the fuel temperature constant to determine the moderator temperature reactivity coefficient.

Table 4 reveals that both proposed cases in which the core was simulated by combining conventional enriched uranium and reprocessed fuel assemblies exhibited lower values than the reference case regarding temperature reactivity coefficients. This behavior implies that in the event of an increase in fuel temperature, the homogeneous and heterogeneous cores demonstrate a higher negative reactivity insertion, leading to a rapid reduction in neutron population. This inverse behavior between increased fuel temperature and negative reactivity insertion is required from a safety perspective. In terms of averaged values, it represents a safety gain in this parameter of approximately 15% for case 1 core and 13% for case 2 compared to the reference case.

It is also notable that the case 1, which has slightly greater amounts of reprocessed fuel content, demonstrated lower values than the case 2. It could be explained by the Doppler-broadening absorption peak of Pu isotopes and some minor actinides, such as Am, Cm, and Np, for higher temperatures, thereby contributing to more absorption reactions in cases in which the core was loaded with more reprocessed fuel content.

Table 5 shows that the proposed cores also achieved lower performance than the reference case in terms of the moderator temperature reactivity coefficient. These cases presented a lower negative reactivity insertion in response to an eventual increase in the

temperature of moderator material, although closely approaching to the reference case value of $\alpha_m = -21.10439$ pcm/K. In summary, these results demonstrate that no problems related to a recursive increase in the core temperature and corresponding positive reactivity insertion would be caused in both studied cores.

Table 4: Comparison of the fuel temperature reactivity coefficients (pcm/K).

α_f			
$\Delta T_f\ (K)$	**reference case**	**case 1**	**case 2**
300 - 400	-3.29791	-4.35683	-4.53144
400 - 500	-3.41530	-3.45707	-3.74860
500 - 600	-2.75719	-3.36241	-2.81211
600 - 700	-2.54112	-3.15007	-2.93667
700 - 800	-2.15781	-3.86429	-2.63344
800 - 900	-2.58241	-2.77628	-2.29860
average	-2.79196	-3.22607	-3.16014

Table 5: Comparison of the moderator temperature reactivity coefficients (pcm/K).

α_m				
$\Delta T_m\ (K)$	$\Delta\rho\ (g/cm^3)$	**reference case**	**case 1**	**case 2**
550 - 560	0.76586 - 0.74766	-5.06334	-12.27187	-11.48977
560 - 570	0.74766 - 0.72778	-7.89335	-8.87095	-14.52935
570 - 580	0.72778 - 0.70573	-23.05092	-20.21978	-20.12799
580 - 590	0.70573 - 0.68067	-31.84494	-24.18533	-23.11255
590 - 600	0.68067 - 0.65106	-37.66941	-35.37457	-27.39860
average		-21.10439	-20.18452	-19.33615

We also evaluated the kinetic parameters related to each simulation performed in this work since they are vital for the effective control of fission reactions within the reactor core, mainly maximizing the reactor period with a greater delayed neutron emission rate. These kinetic parameters include the total fraction of fission neutrons that are delayed (β) and the

decay constant (λ). Mathematically, these parameters are computed through six delayed groups using Equation (3) and (4) [24]:

$$\beta = \sum_i \beta_i \tag{3}$$

$$\lambda = \left(\frac{1}{\beta} \sum_i \frac{\beta_i}{\lambda_i} \right)^{-1} \tag{4}$$

The obtained values for these parameters are summarized in Table 6. Note that the proposed cases exhibited a small reduction in these safety parameters compared to the reference case, being especially lower in case 1 because of the greater utilization of the reprocessed fuel in the entire core. Despite the decrease in these kinetic parameters, both proposed cores are still far enough from the prompt critical state, when the core is critical on prompt neutrons solely ($\rho = \beta$) and the reactor period becomes inappropriately short.

This decreasing behavior is actually expected since plutonium isotopes present in the reprocessed fuel matrices yield minor amounts of delayed neutrons. In the process of subsequent decay of radioactive fission, ^{239}Pu, ^{240}Pu, and ^{241}Pu produce less delayed neutrons than ^{235}U and ^{238}U, which are the primary isotopes capable of undergoing fission reactions for the reference case.

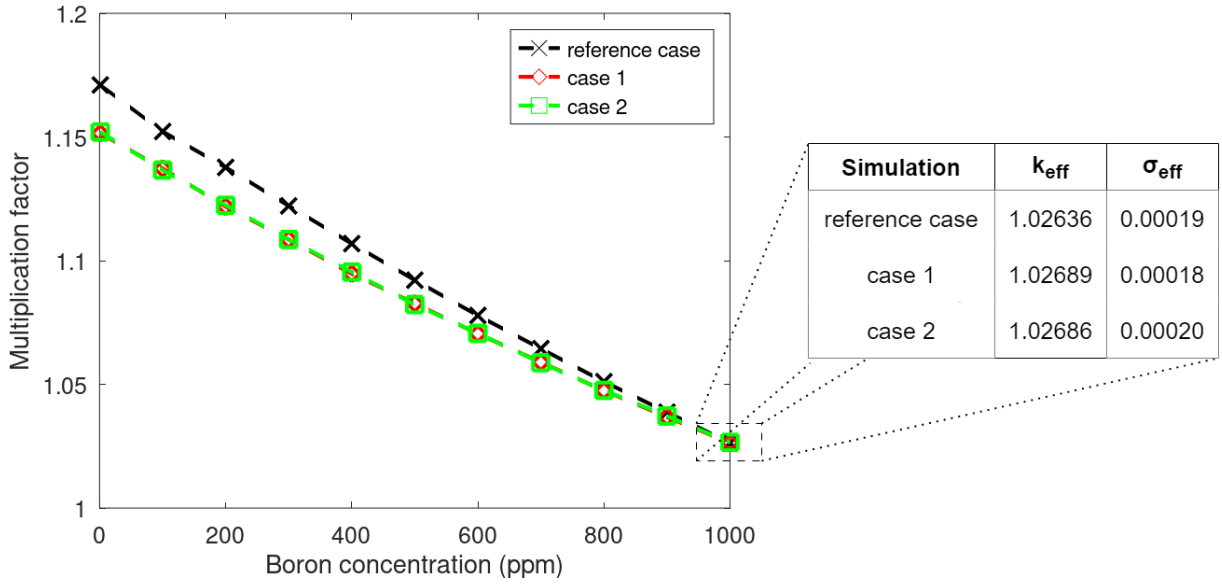
Table 6: Delayed neutrons yield and decay constant for each simulated case.

	β (n/f)			λ (s^{-1})		
Group	**reference case**	**case 1**	**case 2**	**reference case**	**case 1**	**case 2**
1	2.02222E-04	1.63618E-03	1.64691E-04	1.24908E-02	1.25343E-02	1.25342E-02
2	1.11388E-03	9.93404E-04	1.00153E-03	3.16496E-02	3.12908E-02	3.12918E-02
3	1.08898E-03	9.23333E-04	9.22597E-04	1.10197E-01	1.10450E-01	1.10428E-01
4	3.19217E-03	2.60972E-03	2.60429E-03	3.20679E-01	3.19999E-01	3.20087E-01
5	1.04188E-03	8.74281E-04	8.75947E-04	1.34547E+00	1.30272E-00	1.30352E+00
6	3.44220E-04	2.87397E-04	2.85242E-04	8.89685E+00	8.33934E-00	8.34528E+00
Total	6.98335E-03	5.85175E-03	5.85430E-03	8.08415E-01	7.69904E-01	7.67163E-01

In the present NuScale-like design, natural boron has been added to the primary coolant system water as a chemical control for reactivity management. Boron readily absorbs neutrons and therefore affects the neutron activity. In order to evaluate its significance for operational purposes, Figure 5 depicts a parametric study varying the boron concentration in the coolant materials, ranging from 0 ppm to operation state, namely 1000 ppm. All control rods (including regulating and shutdown) are assumed to be fully withdrawn, while the thermal power was settled to 160 MW_{th}. As can be verified, in each proposed core, the maximum excess of reactivity was considerably lower in comparison to the reference case, especially in the absence of boron dissolution in the coolant.

Furthermore, the multiplication factor decreases smoothly in case 1 and case 2 given an increase in boron concentration, both exhibiting a very similar behavior. According to these findings, adapting the NuScale-like core to the proposed cores could reduce the dependence on chemical control and improve core resistance to reactivity-initiated accidents, which is a desired condition in the event of borated water unavailability. For values higher than 600 ppm, both simulated cases approach the reference case, culminating in the very same values for k_{eff} at 1000 ppm used for simulations with the core at BOL conditions.

Figure 5: Multiplication factor as a function of boron concentration in the coolant.



Simulation	k_{eff}	σ_{eff}
reference case	1.02636	0.00019
case 1	1.02689	0.00018
case 2	1.02686	0.00020

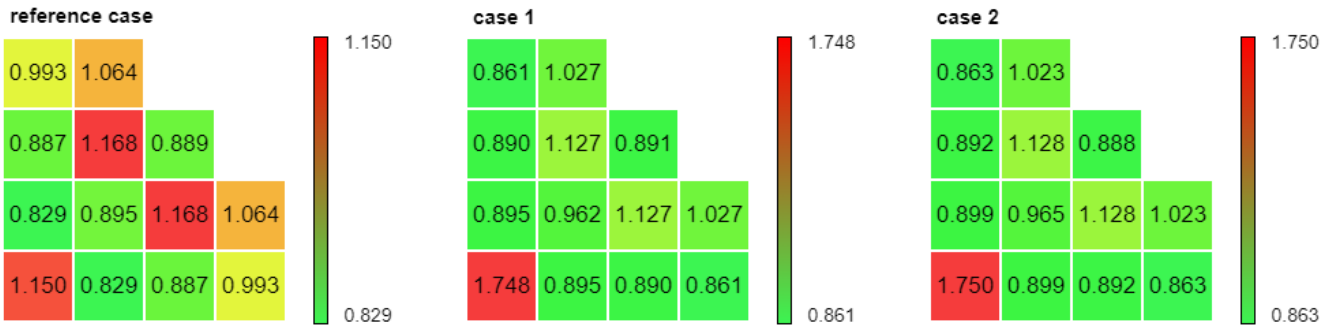
Another constraint and criterion on the safe margin related to both neutronics and thermal hydraulics is the power peak factor (PPF). This parameter is calculated through the ratio between the normalized linear power density ($\dot{Q}_l$) generated in a certain fuel assembly and the average linear power density generated in all fuel assemblies ($\dot{Q}_{average}$), as mathematically shown by Equation (5):

$$PPF = \frac{\dot{Q}_l}{\dot{Q}_{average}} \tag{5}$$

In the Serpent code, one can obtain the linear power density using a mesh-detector to compute the power distribution for each fuel assembly within the core. In the present work, we adjusted such radial mesh-detector dimensions to those exhibited by the fuel assemblies, considering the assembly pitch inside the core of 21.5036 cm. However, even though the core geometry is symmetric, the linear power distribution obtained from the neutronic code is not exactly symmetric due to the statistical nature of Monte Carlo transport method [25].

To overcome this issue, we averaged the results from symmetrical fuel-assembly positions as a part of the post processing to obtain a symmetric distribution. Therefore, the power peak factor (PPF) can be successfully assessed using Equation (5). Figure 6 depicts the resulted normalized power peak factor for the reference case and both cases containing reprocessed fuel compositions in a 1/4 symmetry.

Figure 6: Normalized power peak factor for all cases in 1/4 symmetry.



For the reference case, the power distribution concentrates in the C03 and B01 positions, which results in higher values of power peak factor for these fuel assemblies. The changing of a conventional NuScale-like to the proposed cores moderately reduces the power generation in B01, in addition to B02 and C01 batch positions. On the other hand, it drastically concentrates generation at C03 positions, thereby resulting in a power peak in this position around of $PPF = 1.750$ for both cases.

It is stated in NuScale FSAR that the *PPF* in the beginning of life for NuScale reactor lies in a range of $PPF = 1.9$ and 2.0 [7]. This value considers conditions such as the reactor operating at hot full power (100% of rated power) and all control rods out, conditions attained for computing this parameter. It means that the proposed modifications regarding the fuel composition and charging positions match the *PPF* requirements by design, despite presenting the linear power density profile very straightened in the center of the core.

4. CONCLUSIONS

The present study evaluates the potential adaptation of the NuScale-like core to a core combining enriched uranium and fuel reprocessed by GANEX and UREX+ processes to investigate the feasibility for closed nuclear fuel cycle implementation for this type of reactor. As a primary requirement, we assessed their safety parameters at beginning-of-life (BOL) and compared them to the reference case.

It was shown that no problems related to a recursive increase in the core temperature and corresponding positive reactivity insertion would be caused in both proposed cores. This finding reflects the fact that the fuel and moderator temperature reactivity coefficients exhibited negative values as an example of the NuScale-like core taken as a reference. Notably, both proposed cases achieved a significant increase of around 15% in fuel temperature reactivity coefficient and a small reduction in safety

margins considering moderator temperature reactivity coefficient, although closely approaching to the reference case value.

From the viewpoint of kinetic parameters, case 1 and case 2 presented lower values for delayed neutrons rate and decay constant than the reference case. It could be explained by the fact that plutonium isotopes present in reprocessed fuel compositions yield minor amounts of delayed neutrons in the process of subsequent decay of radioactive fission products. These values result in a shortened period of reactor. However, the proposed cores can still be considered suitable for operation because they are far enough from the prompt critical state.

Furthermore, both proposed cores presented considerably lower initial reactivity in the absence of boron dissolution in the coolant, which could allow a reduction of dependence on chemical control and improve core resistance to reactivity-initiated accidents. Finally, the changing of a conventional NuScale-like to the proposed cores moderately reduces the power generation in B01-batch positions and drastically concentrates generation at C03 positions. Although the high *PPF* value obtained in the referred position, case 1 and case 2 match the NuScale's *PPF* requirements by design.

The overall results reported in this paper show a feasible path for adopting the NuScale core to a core containing conventional enriched uranium and reprocessed fuel. Both options demonstrated valuable outcomes for implementing closed fuel cycle for this type of reactor considering the safety parameters analysis and might be considered for proposing the nuclear fuel recycling instead of locating spent fuel directly to the final deep geological repository. However, further efforts still need to be conducted to assess other options as positions to charging reprocessed fuel composition, especially for managing power generation across the core more equally and obtaining a flattened radial power distribution.

ACKNOWLEDGMENT

The authors are grateful to Brazilian research funding agencies CNEN – Comissão Nacional de Energia Nuclear, CNPq – Conselho Nacional de Desenvolvimento Científico e Tecnológico, CAPES – Coordenação de Aperfeiçoamento de Pessoal de Nível Superior, FAPEMIG – Fundação de Amparo à Pesquisa do Estado de Minas Gerais and FINEP - Financiadora de Estudos e Projetos for the support.

CONFLICT OF INTEREST

The authors declare that they have no competing financial interests or personal relationships that may have influenced the work reported in this study.

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