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Initial Safety Parameter Evaluation of a PWR Loaded with Thorium and Reprocessed Fuel

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Abstract: The once-through cycle (OTC) of nuclear fuel results in storing large quantities of high-radioactive isotopes. Alternatively, the closed cycle (CC), which involves reprocessing and reusing spent nuclear fuel, improves fuel utilization and reduces high-level radioactive waste. This study evaluates the feasibility of incorporating reprocessed fuel into a Pressurized Water Reactor (PWR) core. The PWR core was simulated based on the component dimensions, material definitions, and fuel compositions described in the available Benchmark for Evaluation and Validation of Reactor Simulations (BEAVRS). This core configuration originally contained only uranium-based fuels enriched at different levels, namely 1.6, 2.4, and 3.1 wt-% of ^{235}U, and was used as a reference case. Subsequently, the spent fuel of a PWR that attained a burnup of approximately 33,000 MWd/tHM and a cooling period of five years was theoretically reprocessed using two techniques: GANEX or UREX+. Both reprocessed fuels compositions were spiked with thorium dioxide. Then, four other PWR core configurations were simulated: two with the insertion of fuels reprocessed by the GANEX or UREX+ technique replacing uranium-based fuel enriched at 2.4% and two with the insertion of fuels reprocessed by the GANEX or UREX+ techniques replacing uranium-based fuel enriched at 3.1%. The Serpent code was used to simulate the reactor and assess the neutron flux, temperature reactivity coefficients and the impact of the boron concentration in the coolant on the effective multiplication factor. The findings indicate that using reprocessed fuel in this PWR core is not only feasible but also advantageous.

Keywords: PWR, GANEX, UREX+, reprocessed fuel, closed nuclear fuel cycle.

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Análise dos Parâmetros de Segurança Iniciais de um PWR Abastecido com Tório e Combustível Reprocessado

Resumo: O ciclo aberto do combustível nuclear resulta no armazenamento de grandes quantidades de isótopos altamente radioativos. Alternativamente, o ciclo fechado, que envolve o reprocessamento e reutilização de combustível nuclear irradiado, leva a uma melhor utilização do combustível e à redução de resíduos radioativos de alto nível. Este estudo avalia a viabilidade de incorporar combustível reprocessado em um núcleo de Reator de Água Pressurizada (PWR). O núcleo PWR foi simulado com base nas dimensões dos componentes, definições de materiais e composições de combustível descritas no *Benchmark for Evaluation and Validation of Reactor Simulations* (BEAVRS). A configuração do núcleo contém originalmente apenas combustíveis à base de urânio enriquecidos em diferentes níveis, nomeadamente 1.6%, 2.4% e 3.1% em fração de peso de ^{235}U, e foi tomada como caso de referência. Em seguida, o combustível irradiado de um PWR que atingiu uma queima de aproximadamente 33,000 MWd/tHM e um período de resfriamento de cinco anos foi teoricamente reprocessado usando duas técnicas: GANEX ou UREX+. Após o reprocessamento foi adicionado dióxido de tório às composições. Sendo assim, foram simuladas outras quatro configurações de núcleos PWR: duas com inserção de combustíveis reprocessados pela técnica GANEX ou UREX+ em substituição ao combustível à base de urânio enriquecido a 2.4% e outras duas com inserção de combustíveis reprocessados pelas técnicas GANEX ou UREX+ em substituição ao combustível à base de urânio enriquecido em 3.1%. O código Serpent foi utilizado para simular o reator e avaliar o fluxo de nêutrons, os coeficientes de reatividade de temperatura e o impacto da concentração de boro no refrigerante no fator de multiplicação efetivo. Os resultados indicam que o uso de combustível reprocessado neste núcleo PWR não é apenas viável, mas também vantajoso.

Palavras-chave: PWR, GANEX, UREX+, combustível reprocessado, ciclo do combustível nuclear fechado.

1. INTRODUCTION

The once-through cycle (OTC) of nuclear fuel, also known as the open fuel cycle, produces a significant volume of long-lived radioactive waste and has a low efficiency in terms of utilizing nuclear fuel [1]. During burnup, plutonium isotopes and minor actinides (Np, Am, and Cm) are produced through the transmutation of ^{238}U [2], normally leading to an increase in radiotoxicity levels. Because these high-level wastes require safe storage for long periods [3,4], their production is a significant concern for nuclear power utilization [5].

A feasible path for building a future with sustainable nuclear energy production is the implementation of the closed fuel cycle (CC) [6]. This involves reprocessing the spent nuclear fuel and reusing it prior to permanent storage. Thus, several significant fissile isotopes, such as ^{239}Pu and ^{235}U, and some minor actinides are recovered during reprocessing, guaranteeing a higher efficiency of nuclear fuel utilization [7]. According to the World Nuclear Association [8], the CC enables an additional 25% to 30% energy yield compared to direct fuel disposal and decreases the production of long-lived and highly radioactive fission products generated during the first burnup.

This study aims to analyze the insertion of reprocessed fuel into a pressurized water reactor (PWR) core. The PWR core was simulated based on the data described in the available Benchmark for Evaluation and Validation of Reactor Simulations (BEAVRS) [9]. Subsequently, the spent fuel discharged from a typical PWR was theoretically reprocessed by two different techniques: Group Actinide Extraction (GANEX) [10] and Uranium Extraction Plus (UREX+) [11,12], both of which were spiked with thorium dioxide (ThO_2). Adding ThO_2 was necessary to reduce the enrichment of the reprocessed fuels, as the reprocessing techniques recover several fissile materials, leading to higher fuel reactivity levels. Subsequently, four other core simulations were performed: two regarding the insertion of fuels reprocessed by the GANEX technique and two regarding the insertion of fuels reprocessed by the UREX+ technique. The Serpent code [13] version 2.1.32 was used to

assess the fuel and moderator temperature reactivity coefficients at the beginning of life (BOL) of the five simulated cores and their initial effective multiplication factors for different boron concentrations in the coolant.

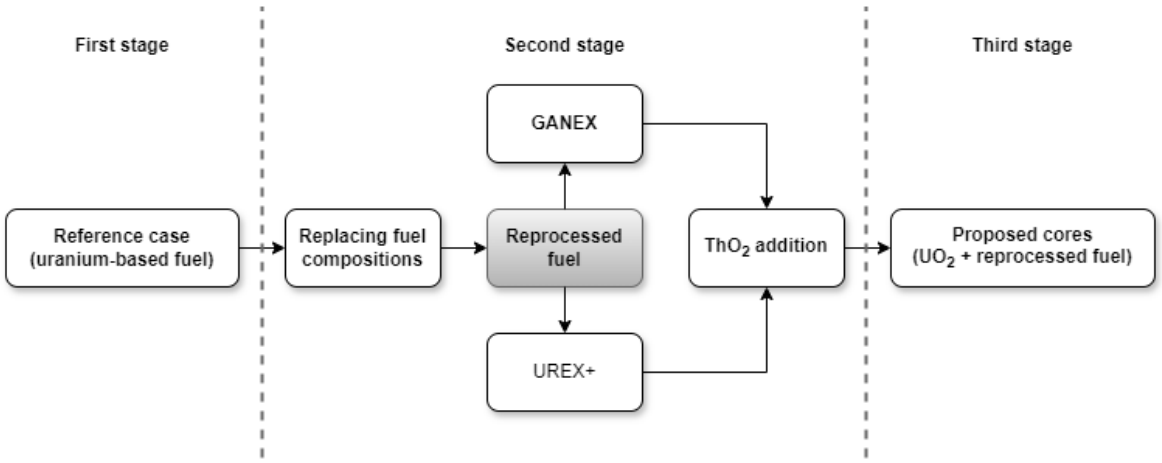
2. MATERIALS AND METHODS

Figure 1 illustrates the theoretical framework of this study. In summary, the methodology was divided into three stages, starting with the simulation of a PWR core as a reference case based on a neutronic benchmark [9]. The referred core configuration employs an OUT-IN strategy with uranium-based fuel enriched at varying levels, 1.6 and 2.4 wt-% of ^{235}U in the central positions and 3.1 wt-% of ^{235}U in the outermost positions.

Subsequently, the methodology progressed to the second stage, in which the batch positions with fuel enriched at 2.4 and 3.1 wt-% of ^{235}U were replaced with reprocessed fuel compositions obtained by GANEX or UREX+ methods. These techniques are well-known non-proliferation methods for reprocessing spent nuclear fuel because they do not recover plutonium isotopes separately and are therefore adopted in the present work. The spent fuel composition came from a PWR that attained a burnup of approximately 33,000 MWd/tHM and a cooling time of five years. Subsequently, once the reprocessing techniques were applied, the compositions were spiked with different ThO_2 percentages depending on the fuel loading pattern.

Finally, the third stage examined the options for inserting these reprocessed fuel compositions into the PWR core, totaling four proposed cases aside from the reference case conducted in this work. The primary objective was to compare their initial safety parameters to investigate the feasibility of inserting reprocessed fuel into the PWR core from a safety perspective.

Figure 1: Theoretical framework for designing the proposed cores.



2.1. Reference case

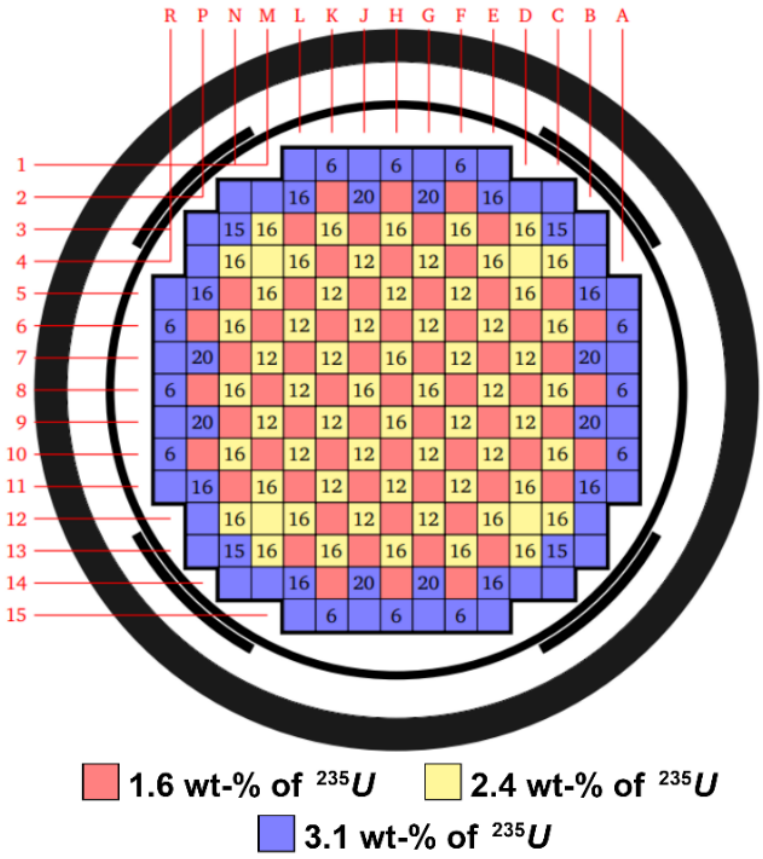
The component dimensions, materials definitions, and fuel composition from the Benchmark for Evaluation and Validation of Reactor Simulations (BEAVRS) were used in this study to simulate a PWR core. This is a multi-cycle full-core PWR depletion benchmark built from the description of fuel assemblies, burnable absorbers, core loading patterns, and other components of a real 4-loop PWR Westinghouse plant. The core power output corresponded to 3411 MWth with an operational pressure of 2250 psia.

Figure 2 shows the loading patterns of the initial core. The reference case contained 193 fuel assemblies distributed throughout the core in an OUT-IN strategy, with fuel enriched at different levels. The center positions were subsequently charged with lower fuel enrichments, namely 1.6 and 2.4 wt-% of ^{235}U, while the fuel with the highest enrichment occupied the outmost positions (3.1 wt-% of ^{235}U). Each fuel assembly had a 17×17 array configuration with a lattice assembly pitch of 21.50364 cm, an active fuel length of 365.76 cm, and composed of 264 fuel rods, 24 guide tubes, and one central instrumentation tube, which was modeled as an empty guide tube. The fuel rods and guide tubes are spaced 1.25984 cm apart using a pin lattice.

Burnable absorber (BA) rods containing borosilicate are strategically located in fuel assemblies to handle the initial excess reactivity. The concentration of BA rods depends on the position of the fuel assembly in the core, as shown in Figure 2. The core also featured other types of reactivity controls, including control rods that were structurally composed of two different absorber regions, B_4C and Ag-In-Cd, and boron dissolved in the water-coolant material.

The entire core is surrounded by a heavy steel reflector to help in neutron economy and is bounded by a cylindrical barrel made of stainless steel. All the materials and dimensions of the simulated core and their specifications for the structural components are described in detail in the benchmark taken as a reference case [9].

Figure 2: Loading pattern and BA concentration (6, 12, 15, 16, or 20 rods) in each assembly of the reference case [9].



2.2. Reprocessed fuel compositions for replacing specific uranium-based fuel assemblies

The reprocessed fuel composition was obtained as follows: first, the spent fuel of a PWR that attained a burnup of approximately 33,000 MWd/tHM and a cooling period of five years was reprocessed using GANEX or UREX+ techniques and then spiked in a fertile composition of ThO_2. These fuel compositions were loaded into the fuel assemblies containing the uranium-based fuels enriched at 2.4 or 3.1 wt-% of ^{235}U, thereby replacing their fuel content from the traditional UO_2 to reprocessed fuel compositions.

In terms of GANEX results, according to [10], Np, Pu, Am, and Cm were recovered together in one liquid flow, and the losses were estimated to be lower than 0.5% (neptunium essentially), corresponding to a recovery yield of actinides exceeding 99.5%. The decontamination factors for some lanthanides (especially Nd, Sm, and Eu) in the actinide product was approximately 5%.

The other reprocessing technique used was UREX+. 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 the plutonium in a pure mixture. UREX+ is an improvement of UREX because it extracts plutonium mixed with some minor actinides [11,12]. The percentages of recovered isotopes from the spent fuel matrix were: 99.95% of U, 99.50% of Pu, 71% of Np, 98% of Am, and 79% of Cm [11,12].

After the reprocessing, the fuel compositions contained massive amounts of fissile isotopes, making their initial reactivity higher than desired. To correct this, the reprocessed fuels were spiked with ThO_2 to reduce their criticality. For the replacement of uranium-based fuel enriched at 2.4%, the percentage of ThO_2 used was 96.43% and 96.50% when the fuel was reprocessed using the GANEX and UREX+ methods, respectively. Table 1 lists the final compositions used in the simulations. Subsequently, the percentage of ThO_2

used to replace the uranium-based fuel enriched at 3.1% was 93.35% and 93.62% when the fuel was reprocessed using the GANEX and UREX+ methods, respectively. Table 2 lists the final compositions used in the simulations. In this study, the percentages of ThO_2 were adjusted to achieve the same initial effective multiplication factor (k_{eff}) as that obtained from the PWR core used as the reference case. The primary purpose of such equivalence between their initial k_{eff} values is to facilitate comparisons between the simulated cases with regard to their safety parameters.

Table 1: Compositions of the fuels theoretically reprocessed and spiked with ThO_2 used for replacing uranium-based fuel enriched at 2.4 wt-% of ^{235}U.

Fuel reprocessed by the GANEX technique and spiked with 96.43 wt-% of ThO_2				**Fuel reprocessed by the UREX+ technique and spiked with 96.50 wt-% of ThO_2**			
Isotope	**Weight frac.**	**Isotope**	**Weight frac.**	**Isotope**	**Weight frac.**	**Isotope**	**Weight frac.**
Th-232	8.47546E-03	Am-243	3.58208E-06	Th-232	8.48194E-03	Am-243	3.43013E-06
U-234	4.91820E-10	Am-241	2.63788E-06	U-234	2.40285E-09	Am-241	2.52599E-06
U-235	2.55963E-08	Am-242	4.85457E-09	U-235	1.25054E-07	Am-242	4.64865E-09
U-236	1.30844E-08	Cm-242	8.21396E-07	U-236	6.39256E-08	Cm-242	6.34059E-07
U-238	3.11157E-06	Cm-244	9.41647E-07	U-238	1.52020E-05	Cm-244	7.26884E-07
U-233	6.58199E-15	Cm-245	3.27668E-08	U-233	3.21571E-14	Cm-245	2.52936E-08
U-237	1.86421E-11	Np-237	1.50638E-05	U-237	9.10781E-11	Np-237	1.05032E-05
Pu-238	5.88848E-06	Np-238	2.46801E-08	Pu-238	5.72501E-06	Np-238	1.72081E-08
Pu-239	1.54195E-04	Np-239	1.55829E-06	Pu-239	1.49914E-04	Np-239	1.08651E-06
Pu-240	5.27132E-05	Nd-144	3.91437E-06	Pu-240	5.12498E-05	O-16	1.21042E-03
Pu-241	4.95637E-05	Sm-150	7.84971E-07	Pu-241	4.81878E-05		
Pu-242	1.87311E-05	Eu-152	1.66920E-07	Pu-242	1.82111E-05		
		O-16	1.21077E-03				
sum = 1.000E+00				sum = 1.000E+00			

Table 2: Compositions of the fuels theoretically reprocessed and spiked with ThO_2 used for replacing uranium-based fuel enriched at 3.1 wt-% of ^{235}U.

Fuel reprocessed by the GANEX technique and spiked with 93.35 wt-% of ThO_2				Fuel reprocessed by the UREX+ technique and spiked with 93.62 wt-% of ThO_2			
Isotope	**Weight frac.**	**Isotope**	**Weight frac.**	**Isotope**	**Weight frac.**	**Isotope**	**Weight frac.**
Th-232	8.20543E-01	Am-243	6.67305E-04	Th-232	8.22971E-01	Am-243	6.25333E-04
U-234	9.16211E-08	Am-241	4.91411E-04	U-234	4.38053E-07	Am-241	4.60503E-04
U-235	4.76833E-06	Am-242	9.04358E-07	U-235	2.27980E-05	Am-242	8.47477E-07
U-236	2.43750E-06	Cm-242	1.53018E-04	U-236	1.16540E-05	Cm-242	1.15593E-04
U-238	5.79654E-04	Cm-244	1.75419E-04	U-238	2.77141E-03	Cm-244	1.32515E-04
U-233	1.22616E-12	Cm-245	6.10412E-06	U-233	5.86243E-12	Cm-245	4.61117E-06
U-237	3.47283E-09	Np-237	2.80624E-03	U-237	1.66041E-08	Np-237	1.91479E-03
Pu-238	1.09696E-03	Np-238	4.59766E-06	Pu-238	1.04370E-03	Np-238	3.13713E-06
Pu-239	2.87249E-02	Np-239	2.90294E-04	Pu-239	2.73302E-02	Np-239	1.98078E-04
Pu-240	9.81993E-03	Nd-144	7.29208E-04	Pu-240	9.34314E-03	O-16	1.20945E-01
Pu-241	9.23322E-03	Sm-150	1.46232E-04	Pu-241	8.78492E-03		
Pu-242	3.48942E-03	Eu-152	3.10955E-05	Pu-242	3.32000E-03		
		O-16	1.21004E-01				
sum = 1.000E+00				sum = 1.000E+00			

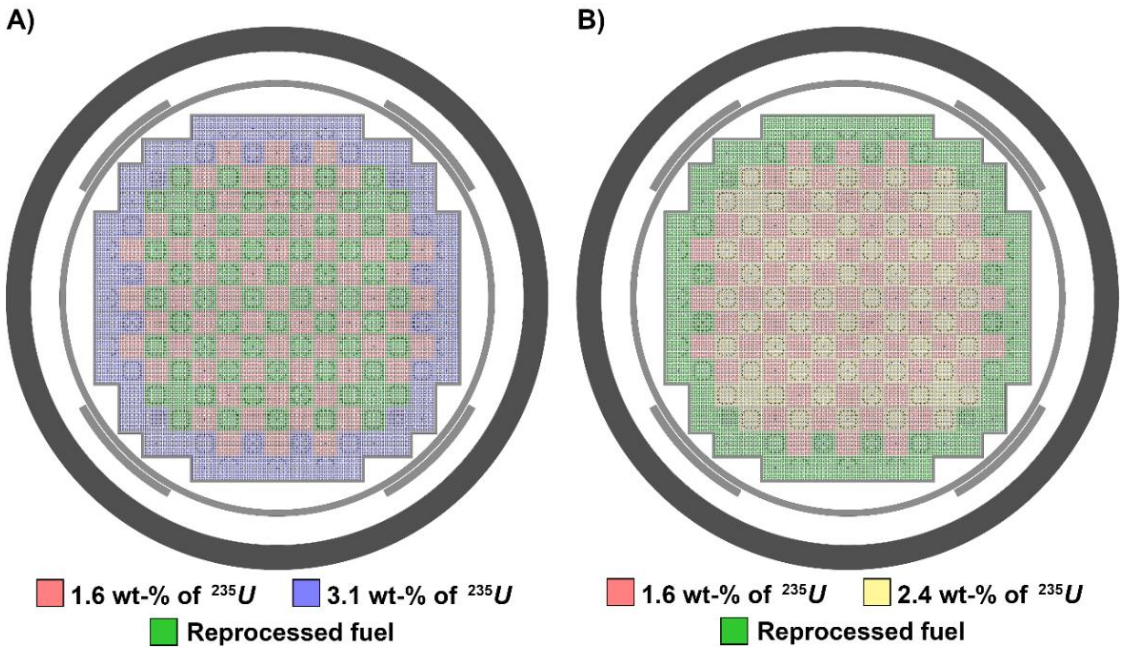
2.3. Proposed cores containing UO_2 and reprocessed fuel

Figure 3 shows the two possibilities proposed in this study for inserting reprocessed fuel into the PWR core, changing the fuel content from traditional UO_2 for reprocessed fuel compositions in different batch fuel assemblies in the core. The first possibility involves changing the fuel assemblies that originally contained uranium-based fuel enriched at 2.4 wt-% of ^{235}U for the fuel reprocessed using the GANEX or UREX+ methods, as presented in Table 1. The new core configuration is shown in Figure 3 (A). The other possibility examined in this study is the replacement of fuel assemblies that originally contained uranium-based fuel enriched at 3.1 wt-% of ^{235}U for the fuel reprocessed using the GANEX or UREX+, as presented in Table 2. The new core configuration is shown in Figure 3 (B). Therefore, in

addition to the uranium-based PWR taken as a reference case, four cases were performed, as itemized below to facilitate the understanding:

- **Case 1:** Core configuration containing uranium-based fuels enriched at 1.6 and 3.1 wt-% of ^{235}U and fuel reprocessed using the GANEX method spiked with 96.43% of ThO_2.
- **Case 2:** Core configuration containing uranium-based fuels enriched at 1.6 and 3.1 wt-% of ^{235}U and fuel reprocessed using the UREX+ method spiked with 96.50% of ThO_2.
- **Case 3:** Core configuration containing uranium-based fuels enriched at 1.6 and 2.4 wt-% of ^{235}U and fuel reprocessed using the GANEX method spiked with 93.35% of ThO_2.
- **Case 4:** Core configuration containing uranium-based fuels enriched at 1.6 and 2.4 wt-% of ^{235}U and fuel reprocessed using the UREX+ method spiked with 93.62% of ThO_2.

Figure 3 : (A) Loading pattern of cases 1 and 2. (B) Loading pattern of cases 3 and 4.



Each core configuration was simulated with fuel reprocessed using GANEX or UREX+ methods, totaling four different cases with reprocessed fuel insertion into the core. The primary objective was to compare their initial safety parameters to investigate the feasibility of inserting reprocessed fuel into the PWR core. The safety parameter evaluation included the corresponding neutron fluxes, fuel and moderator reactivity coefficients, delayed neutrons and decay constants, and reactivity response to boron concentration.

The fuel, coolant, and structure temperatures were 900 K, 566 K, and 600 K for the normal operation state, respectively. Moreover, the coolant contained a boron concentration of 975 ppm, and a thermal power generation of 3411 MWth (full rated power) was set. Detailed core design and fuel data are available in the BEAVRS specifications [9].

The Serpent code version 2.1.32 [13], developed by VTT, and the ENDF/B-VII nuclear data library [14] were used in all simulations of this study. Serpent is a probabilistic Monte Carlo code used for various applications in reactor physics, including steady-state simulations of reactor-modeled cores and burnup calculations. The Monte Carlo-based volume calculation routine of Serpent determined the material volumes, employing 10^{11} random sampling points in the geometry. Because the simulations varied only in material compositions and temperatures, we standardized the material volumes across all cases. Each simulation was performed using 200 active generations, 100 inactive generations, and 100,000 neutrons per generation, totaling 20,000,000 neutronic histories per cycle.

3. RESULTS AND DISCUSSIONS

Table 3 lists the k_{eff} at the BOL for the five simulated cores. The compositions of the reprocessed fuels used in cases 1, 2, 3, and 4 were selected by adjusting the ThO_2 percentages to obtain the same initial criticality as in the reference case. Thus, considering the confidence

interval, there was no evidence of a difference between the k_{eff} values of the simulated cases. This process ensured a meaningful comparison between the cases.

Table 3: Effective multiplication factor at BOL.

Simulation	k_{eff}
ref. Case	0.992022 ± 0.00015
case 1	0.992092 ± 0.00015
case 2	0.992037 ± 0.00014
case 3	0.991792 ± 0.00013
case 4	0.991820 ± 0.00014

Figure 4 shows the normalized neutron flux in the 2.4% enriched fuel positions of the reference case and in the reprocessed fuel positions of cases 1 and 2. Most of the neutrons in those fuel zones tend to have energies between 0.01 and 1 MeV in all simulated cases. Furthermore, the neutron spectrum of the reference case exhibits a prominent peak at approximately 10^{-7} MeV, which is characteristic of the ^{235}U isotope. This thermalization peak drastically reduced in the reprocessed fuels because of the predominant presence of plutonium isotopes as fissile materials, which resulted in fission neutron emissions with a harder energy spectrum than that of the uranium-based fuel.

Figure 4: Normalized neutron flux in the 2.4 wt-% of ^{235}U positions (ref. case) and in the reprocessed fuel positions (cases 1 and 2) at BOL.

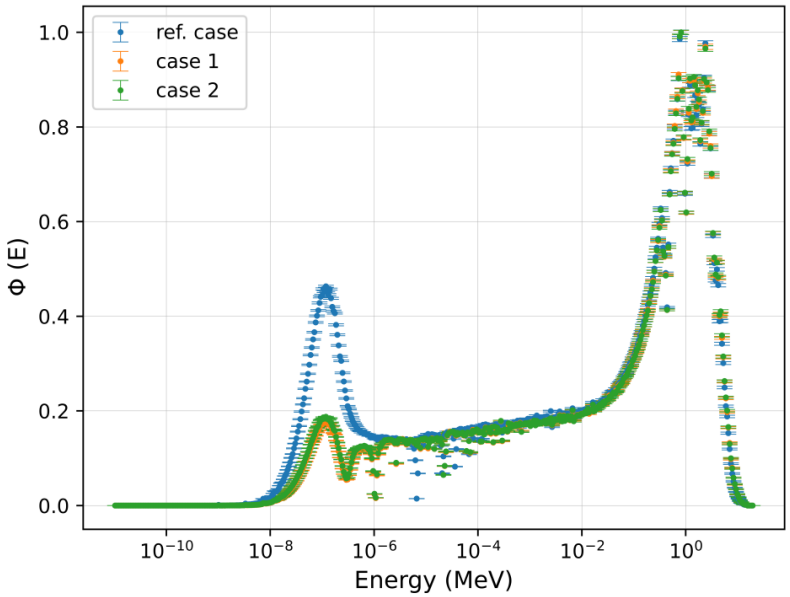


Figure 5 depicts the normalized neutron flux in the 3.1% enriched fuel positions of the reference case and in the reprocessed fuel positions of cases 3 and 4. Similarly, most of the neutrons in those fuel zones tend to have energies between 0.01 and 1 MeV in all simulated cases. Moreover, the neutron thermalization peak (at approximately 10^{-7} MeV) was still present in this fuel zone of the reference case. However, the thermalization peak was nearly absent in the reprocessed fuel zones because of their lower thorium content, which resulted in a higher concentration of Pu isotopes in their composition, hardening their neutron spectra.

Figure 5: Normalized neutron flux in the 3.1 wt-% of ^{235}U positions (ref. case) and in the reprocessed fuel positions (cases 3 and 4) at BOL.

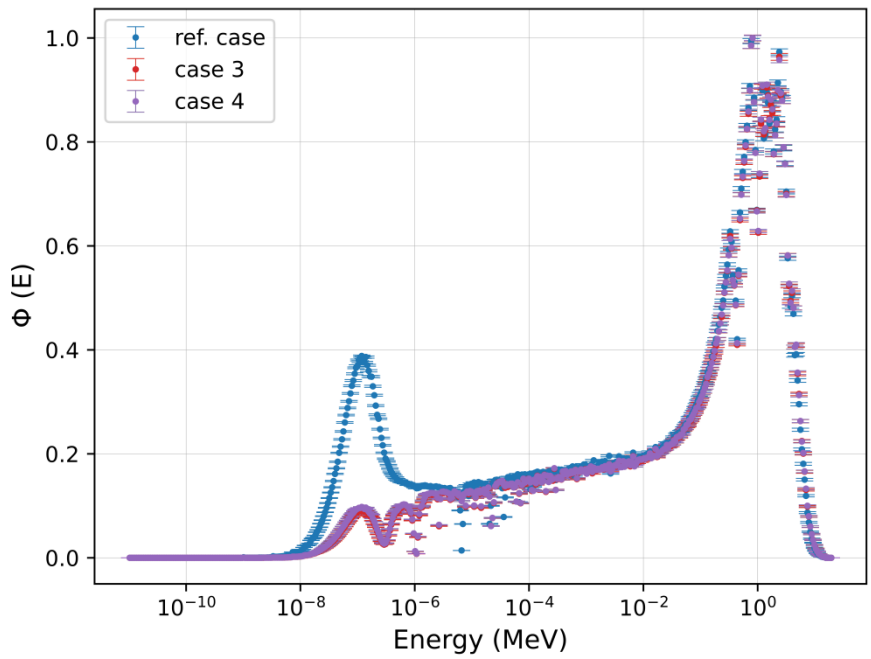


Table 4 compares the fuel temperature reactivity coefficients obtained for the five simulated cases. Because this parameter evaluates the resulting reactivity variation in response to an increase in the fuel temperature, negative values are desired from a safety perspective. To compute this parameter, the fuel temperatures varied from 300 K to 900 K, whereas the temperatures of the other materials remained fixed at the values defined for normal operation.

The reference case exhibited a less negative reactivity response to fuel temperature increases than the cases with reprocessed fuel insertion. This indicates that the use of reprocessed fuel can improve this safety parameter of the reactor. Also, a comparison between the cases with reprocessed fuel insertion shows that the use of the UREX+ technique exhibited a better response to increasing fuel temperatures than the use of the GANEX technique. Thus, case 2 had a more negative coefficient than case 1, and case 4 had a more negative coefficient than case 3. Furthermore, case 4 exhibited the most negative fuel temperature reactivity coefficient among all the simulated cases.

Table 4: Fuel temperature reactivity coefficients (pcm/K).

ΔT_f	ref. case	case 1	case 2	case 3	case 4
300-400	-3.38393	-3.87284	-4.32480	-4.13201	-3.99255
400-500	-3.32738	-3.98268	-3.64790	-3.63030	-3.51879
500-600	-2.47662	-3.12536	-3.41500	-3.10719	-3.96359
600-700	-2.45170	-3.05261	-3.30976	-3.46451	-2.86473
700-800	-2.76422	-3.12893	-2.27797	-2.55453	-2.97525
800-900	-2.06667	-2.55890	-2.91896	-2.75663	-2.48748
average	-2.74509	-3.28689	-3.31573	-3.27420	-3.30040

Table 5 compares the moderator temperature reactivity coefficients obtained for the five simulated cases. This parameter was evaluated by varying the coolant temperature from 510 K to 566 K, and the respective density variation was calculated using the normal operational pressure (2250 psia) and the data from the National Institute of Standards and Technology (NIST) [15]. The temperatures and densities of the other materials were maintained at the same values as those defined for normal operation.

The cases with reprocessed fuel insertion demonstrated significantly more negative reactivity insertion due to an increase in the moderator temperature than the reference case. Additionally, cases 3 and 4 exhibited very similar coefficients, with case 3 exhibiting the best response among all simulated cases to increases in moderator temperature.

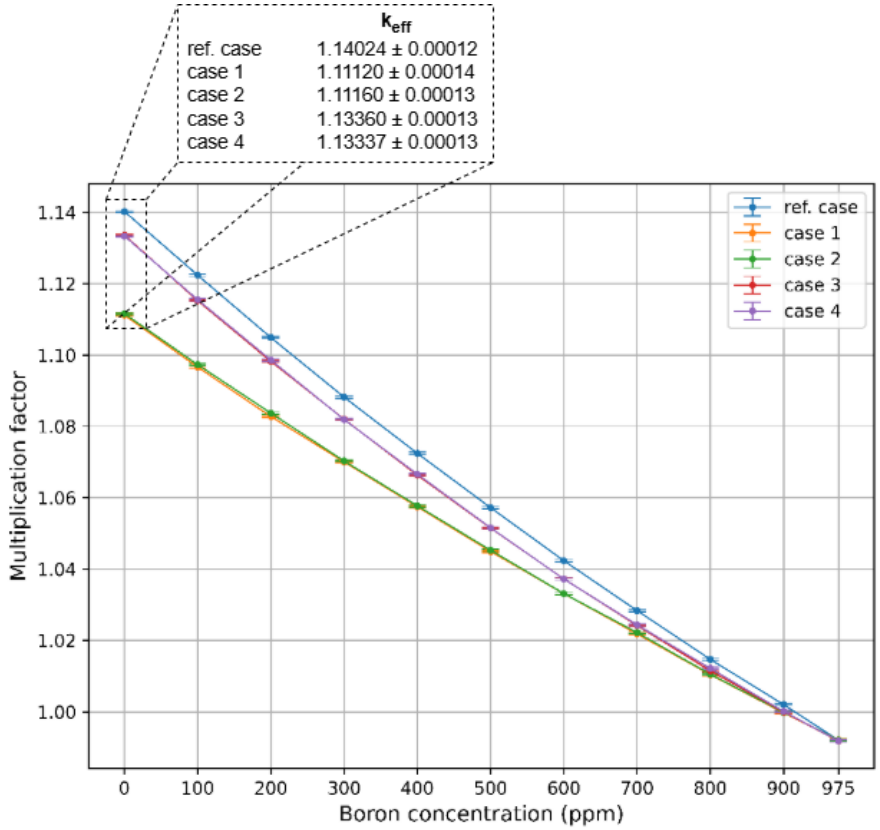
Table 5: Moderator temperature reactivity coefficients (pcm/K).

ΔT_m	ref. case	case 1	case 2	case 3	case 4
510-520	-1.65244	-4.20730	-2.54845	-6.73034	-7.84118
520-530	-3.61090	-2.95530	-3.72386	-12.30517	-9.00524
530-540	-0.36529	-0.87073	-4.11159	-5.00538	-8.42420
540-550	-2.31413	-5.15659	-2.80713	-10.67455	-8.20515
550-560	-1.92925	-9.48610	-7.63885	-11.26622	-9.19348
560-566	-4.62221	-4.07996	-7.38055	-4.21792	-7.46842
average	-2.41570	-4.45933	-4.70174	-8.36659	-8.35628

Figure 6 depicts the impact of the boron concentration in the moderator, ranging from 0 to 975 ppm (normal operation), on the multiplication factor. Cases 1 and 2 showed similar k_{eff} values for the simulated boron concentrations, as did cases 3 and 4. In addition, they exhibited lower k_{eff} values than the reference case at lower boron concentrations, particularly in the absence of boron. This result implies that the cores with reprocessed fuel insertion require less chemical control to reduce their excess reactivity compared with the reference case. Consequently, the resistance of the core to initial excess reactivity scenarios improves. Furthermore, cases 1 and 2 showed the lowest multiplication factor without boron.

Besides to safety enhancement, the potential for operating with reduced boron usage in the cores with reprocessed fuels insertion could lead to lower volumes of liquid radioactive waste, reduced corrosion damage to the reactor structure, and a lower operational radiation dose [16,17].

Figure 6: Effective multiplication factor as a function of the boron concentration in the moderator.



4. CONCLUSIONS

This study evaluated the feasibility of using reprocessed nuclear fuel in a PWR core by assessing initial safety parameters. A reference case was simulated using only uranium-based fuels, and four cases were simulated with reprocessed fuel insertion. Through detailed simulations, the study assessed the neutron flux, fuel and moderator temperature reactivity coefficients at the BOL, and the dependence on chemical control (boron usage) when reprocessed materials were introduced into the reactor core.

First, the neutron flux analysis revealed in the reference case, whether in the 2.4% or 3.1% enriched fuel, a prominent thermalization peak at approximately 10^{-7} MeV, characteristic of the neutrons from ^{235}U. This peak significantly diminished in the reprocessed fuels in all cases because of the higher concentration of plutonium isotopes, which emit

fission neutrons with a harder energy spectrum. Additionally, most of the neutrons in the analyzed fuel zones tend to have energies between 0.01 and 1 MeV in all simulated cases.

The cores with reprocessed fuel insertion exhibited safe operation characteristics, with more negative fuel and moderator temperature reactivity coefficients compared to the reference case, indicating an improvement in safety margins. Notably, the use of reprocessed fuel strongly improved the moderator temperature reactivity coefficient. Cases 2 and 4 exhibited the most negative fuel temperature reactivity coefficients. Meanwhile, cases 3 and 4 exhibited the most negative moderator temperature reactivity coefficients.

Furthermore, the cases with the use of reprocessed fuel, particularly cases 1 and 2, showed a lower need for boron usage to control their initial criticality than the reference case. This indicates an improvement in the resistance of the core to initial excess reactivity scenarios and suggests operational benefits.

These results support the continued investigation and potential implementation of reprocessed fuels in PWRs as a step toward a more sustainable nuclear cycle. Future studies should focus on evaluating other closed-cycle strategies for this PWR.

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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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