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Feasibility to Convert the NuScale SMR from UO_2 to a Mixed (U, Th)O_2 Core: A Parametric Study of Fuel Element—Seed-Blanket Concept

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Abstract

Delays in the construction of nuclear reactors due to licensing issues have been a problem across the world, affecting projects in Finland, France, and the United States. Small Modular Reactors (SMRs) emerge as a transition between Generations III+ and IV in order to make nuclear energy more competitive with other energy sources, including renewables. In this study, the SMR NuScale, one of the most promising projects today, is investigated for its conversion into a U-233-producing reactor through the Radkowsky seed-blanket fuel element concept, applied in the Shippingport reactor, in a parametric study. Initially, a validation of the reference reactor (NuScale) was carried out with data from technical documents and papers, thus demonstrating the agreement of the computational model carried out with the SERPENT code. Then, a parametric study is carried out to define the area of the seed and blanket region, proportions of enrichment and pitch length. Finally, a comparison is made between the production of U-233, TRU reduction, burn-up extension and neutronic and thermohydraulic safety parameters. This study demonstrates an improvement in the conversion factor and a considerable reduction in the production of TRU, in addition to the production of U-233 with a low proportion of other uranium isotopes that can lead to the beginning of the thorium cycle with already consolidated technologies.

Keywords

Thorium, SMR, Seed-Blanket, NuScale

1. Introduction

The nuclear industry has been in constant evolution since its emergence during World War II. In addition, nuclear energy plays an important role in the development of electric energy generation in Brazil and in the world, since the great world powers use this energy, not just to produce electricity, but also to move ships, submarines, applications in the medical field, etc. Generation I reactors were early power reactor designs, developed in the 1950s and built in the 1960s. Thus, Generation II reactors were developed with safety improvements and greater fuel economy. Generation II reactors represent 90% of current reactors in operation and were built before 2000. Generation III reactors were developed as advanced Light Water Reactors (LWRs) with several design enhancements. The Generation III+ reactors, in turn, incorporate evolutionary improvements to the Generation III reactors, such as passive cooling, in some projects, and some units have already been built around the world, as is the case with the reactors at the Lufeng site in China.

Generation IV reactors are the promise of revolutionary projects with greater use of natural resources, greater passive security and lower capital costs. The speculation of the first Generation IV reactors was for 2030 [1] as shown in **Figure 1**, but with the appearance of Small Modular Reactors (SMRs), it is estimated that this deadline will be postponed, as explained below.

Generation III+ reactors around the world have experienced construction delays due to licensing issues. We can cite the case of Olkiluoto 3, in Finland, whose initial cost estimate was €$3 billion, with construction starting in 2005 and forecast for completion in 2010, which ended up costing €$11 billion [2] and had its first maximum power reached in September 2022, with regular operation scheduled for December 2022.

Finland's case is no exception and has been repeated all over the world, as in the US where Westinghouse filed for bankruptcy after the cost of Vogtle 3 and Vogtle 4 increased to US$30 billion [3], due to regulatory problems with the NRC. In France, the Flamanville 3, whose commercial operation was expected for 2013

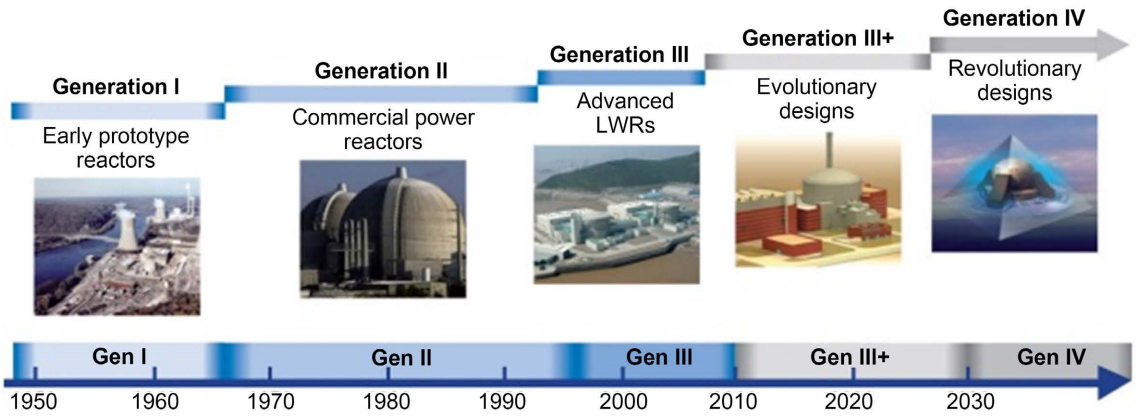

Figure 1. Nuclear energy generations: time intervals correspond to the design and first deployments of different generations of reactors [1].

and with an initial cost of €$3 billion, had its commercial operation postponed to mid-2023 and the cost increased to €$12.7 billion [4]. Licensing problems, schedule delays and rising costs led to the development of Small Modular Reactors (SMRs) as a response to the problems of economic competitiveness of nuclear energy with other sources, with respect to costs and construction time [5] [6] [7].

SMR projects leverage five factors to be more economically competitive than large reactors: 1) Their reduced size implies a lower capital cost for construction, when compared to a large unit; 2) The Components of an SMR can be manufactured on an assembly line and shipped to site, thereby reducing the cost of construction. Since in some projects, they arrive on the site as ready-made packages (project simplification); 3) Its reduced size makes passive features easy to implement; 4) Some SMRs are designed to operate without refueling for their entire operational life. In this way, they do not need to plan for refueling or storing fuel. The seller undertakes to remove the entire plant, exempting the concessionaire from decommissioning plans; 5) Reduced construction schedule. The cost of an SMR is in the order of US$1.5 billion [8] [9]. In the US, the SMR with the most advanced project is the NuScale, already with licensing well advanced with the NRC and with its first unit will be operational in mid-2029, with the remaining modules coming into operation for full operation of the plant by 2030 [10].

Thorium has a high melting point, around 3300˚C, and is naturally found in the form of oxides, phosphates and silicates. The element in question still has a higher melting point than that of UO_2, which has a melting point of 2800˚C, in addition to releasing fewer gases from fission than UO_2. All these characteristics make thorium a great attraction to be used in the nuclear industry. These properties tend to improve the thermo-hydraulic characteristics of Mixed Oxide (MOX) fuels, with thorium and uranium, compared to current uranium oxide fuels. Thorium available in the Earth's crust probably has more available energy than all uranium and fossil fuel sources combined. There is no standard classification for thorium resources, as there is for uranium resources, as it is not a targeted exploration resource for the mineral industry. Its availability is normally estimated in relation to uranium and rare Earth resources and their application in the fuel cycle has implications for the reduction of long-lived actinides and Transuranic (TRU) elements [11] [12] [13].

The motivation for using thorium also has an economic character, due to the increase in the cost of uranium extraction in this century. Maintaining the current consumption rate, it is estimated that extractions costing less than US$80 per kilogram of uranium should last for the next 40 years, for costs of US$130 per kilogram of uranium with a durability of 60 years and for costs below US$260 per kilogram of uranium about 100 years [14]. The mentioned costs limit uranium as a primary source of energy to this century, if no recycling process is used and only with the use of thermal reactors, without taking advantage of the remaining plutonium in the material used or from other sources [15].

Given these concerns, there are already studies on the use of thorium in large PWR reactors, such as the AP1000 reactor [16] [17]. In these studies, the authors show that the use of the thorium cycle in LWR reactors using Radkowsky's seed-blanket concept [13] is advantageous, since its use as a homogeneous MOX with a high amount of uranium implies the accumulation of isotopes such as U-236, which end up making the process uneconomical and increasing the need for enrichment with each new cycle with reprocessing. However, the use of the seed-blanket concept implies a reduction of heat production areas and consequently higher heat fluxes in the seed region when compared to the original reactor. Thus, as SMR reactors have lower power, the use of the seed-blanket fuel element concept becomes viable, without the need for power reduction.

The objective of this work is, taking the NuScale as a reference reactor, to carry out a parametric study to design a Radkowsky's seed-blanket concept fuel element for this reactor, reducing the amount of plutonium produced and producing U-233 for later use, increasing the sustainability of the cycle of fuel.

Some recent studies have advanced in the investigation of Small Modular Reactors (SMRs) of Pressurized Water (PWR) with the use of thorium, but there are still few in the literature. From the manuscripts found, we can mention the use of TRISO particles applied to mixed oxide fuels [18], the conversion with minimal alterations of the SMART reactor with the reduction of the amount of burnable poisons necessary for reactivity control [19], the development of reactors for marine propulsion [20] [21] [22] [23], experiments to validate codes used to simulate reactors [24] and thermo-hydraulic analysis with advanced fuels [25]. These studies are the only ones found in the Web of Science base with the keywords thorium, SMR and PWR, demonstrating that there is still much to be explored in this field, especially in new designs in fuel elements, which is the case of this study and which in its the vast majority are carried out by the same research groups, being little disseminated.

The originality of this article lies in the application of the seed-blanket concept proposed by Radkowsky to the NuScale reactor with different fuel rod pitches, without changing the core heat removal system, thus being a way of starting the thorium cycle in PWR reactor technologies already well-consolidated in the nuclear industry, producing 1 kg of U-233 per cycle.

2. Materials and Methods

2.1. The NuScale Reference Model

The NuScale is a small modular reactor moderated by light water, currently in the licensing stage, with a capacity to generate 77 MWe. Each core module is approximately one-third the size of a large-scale reactor core. Its design includes modern Generation III+ security systems like passive cooling. The core module is composed of 37 assemblies. Each assembly has a grid of 17 × 17 rods, 264 for fuel, 24 guide tubes and a central tube for instrumentation. The used fuel is UO_2 with U-235 enriched between 2.60 and 4.55 w/o, depending on the assembly lo-

cation. In 12 assemblies, the fuel contains homogeneously mixed gadolinium oxide, to control its burning.

To have a reference model against which to compare the converted core, we created a computational model of a NuScale reactor, using the parameters found in the NuScale specifications [26]. The relevant geometric parameters of a NuScale assembly are presented in **Table 1** and **Figure 2**.

The reactor core has a total of 37 fuel assembly in which 12 of these has a mixture of uranium-gadolinium oxide and the remaining 25 fuel assembly are built with pure UO_2. In this way, there are two types of fuel rods as shown in **Figure 2**.

The two types of fuel assembly described are shown in **Figure 3** with the arrangement of the UO2 and (Gd, U)O_2 rods.

In the reactor core, there are 12 fuel assemblies with a concentration of 2 w/o of Gd_2O_3 and an average enrichment of 4.32 w/o (**Figure 4** in dark blue); 12 fuel assemblies with a concentration of Gd_2O_3 of 6w/o and an average enrichment of

Table 1. Design parameters for the NuScale assembly.

Description	Value
Fuel pellet diameter	0.81153 cm
Gap outer diameter	0.82804 cm
Rod diameter	0.94996 cm
Rod pitch	1.25984 cm
Rod height	215.9 cm
Active height	200.0 cm

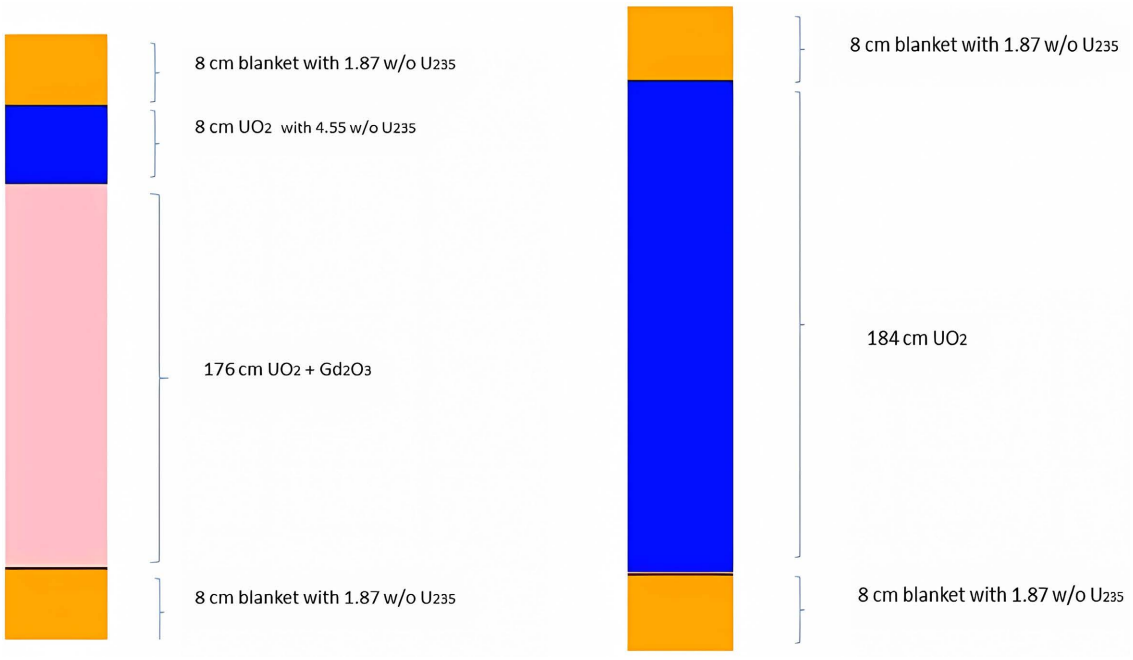

Figure 2. The axial schemes of two types of fuel rods are shown [26].

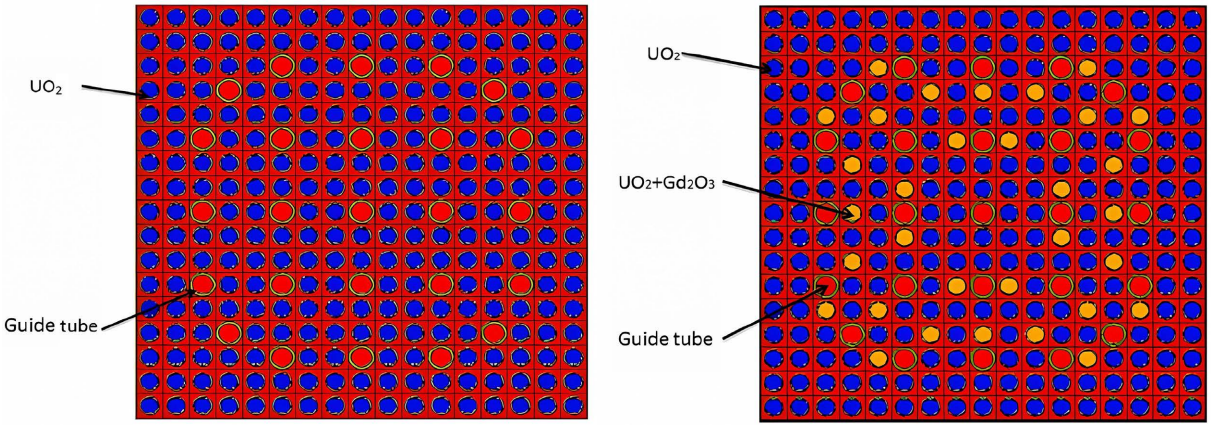

Figure 3. Cross view of two types of fuel assembly including Gd_2O_3 pins (right) and pure UO_2 pins (left) are shown [26].

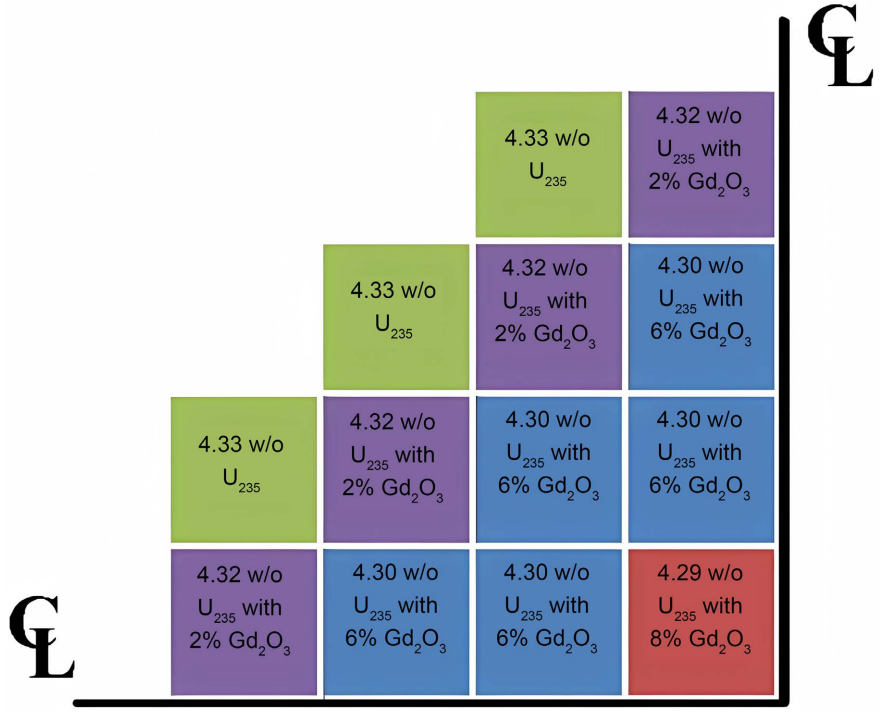

Figure 4. NuScale full core (1/4 simmetry) [26].

4.30 w/o (**Figure 4** in light blue); and one fuel assembly with a Gd_2O_3 concentration of 8 w/o and an average enrichment of 4.29 w/o (**Figure 4** in red). The rest are built with pure UO_2 (**Figure 4** in green). Fuel enrichment without Gd_2O_3 is 4.55 w/o, with gadolinium oxide is 4.11 w/o.

The operating states of the reactor used in the simulations are shown in **Table 2**.

The SERPENT code was used to simulate the full core model using the Monte Carlo method. To validate the model, we compared the results with those from Sadegh-Noedoost *et al.* [26], which were obtained using the MCNPX code. Subsequently, simulations of a single NuScale assembly were used as a reference to compare the results obtained with the MOX fuel element.

Table 2. Operation states.

Operation state	Temperature	
	Fuel	Moderator
Cold Zero Power (CZP)	300 K	300 K
Hot Zero Power (HZP)	492 K	492 K
Hot Full Power (HFP)	700 K	557 K

2.2. Homogeneous Model

To increase the nuclear fuel cycle, Radkowsky (Galperin *et al.* [13]) proposed the Seed-Blanket Unit (SBU) concept, where the nucleus is divided into two regions, one with fissile elements (the seed) surrounded by another with fertile elements (the blanket). Besides exploring the vast amount of thorium natural reserves, a seed-blanket reactor has other advantages, such as the reduction of transuranic waste, which prevents their reuse for the construction of nuclear weapons.

The first step in converting a reactor to mixed fuel is to look for the proportion of seed and blanket volumes that provides an infinite multiplication factor (k_∞) above one and maximizes the Conversion Rate (CR).

A first approach to this problem is to use a *homogeneous model*, which is a simplified reactor model with a single fuel assembly, where the geometry of the rods is removed and two parallelepipeds delimit the seed and blanket regions, each containing a homogeneous mixture of the materials that compose them (**Figure 5**). Then, the blanket volume is varied until the desired condition is found.

Initially, following the results of Galperin *et al.* [13], we used a moderator to fuel volume ratio of $z_s = \left(V_m/V_f\right)_s = 3.1$ in the seed region and $z_b = \left(V_m/V_f\right)_b = 2.0$ in the blanket region. From these values, one can calculate the molecular fractions for water in each mixture, x_s and x_b, then the material compositions can be written as

- $x_s H_2O + (1 - x_s)UO_2$ for the seed and
- $x_b H_2O + (1 - x_b)Th_{0.9}U_{0.1}O_2$ for the blanket,

both with 20 w/o U-235 enrichment.

The assembly has an active volume of $21.26 \times 21.26 \times 200$ cm^3. The analysis of the homogeneous reactor proceeded as follows, 13 simulations were carried out so that the ratio between the blanket and total volumes $\left(V_b/V_t\right)$ comprised the range between 11.50% and 94.73%. Then, the values of k_∞ and CR obtained from the simulations were analyzed to choose $\left(V_b/V_t\right)$ such as to obtain a slightly supercritical reactor $k_\infty > 1$ and maximize the conversion rate. The optimal value, $V_b/V_t = 0.72$, was used to build a detailed reactor model, as described in the next section.

2.3. Heterogenous Model

In the heterogeneous model, the geometry of the rods is included (pellet, gap, clad). The dynamics of a seed-blanket reactor is different from that of a reactor

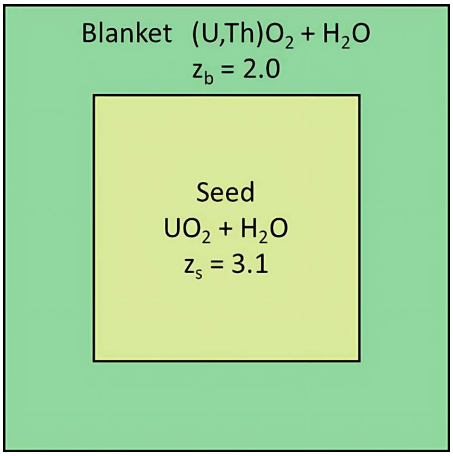

Figure 5. The homogeneous model cross-section.

that uses only UO_2 as fuel. To accommodate the volumes found in the previous analysis, we need to make allowances for the pitch and diameter of the rods. Let X and L be the sides of the cross section of the seed and blanket regions, respectively. It is possible to calculate $X^2 = L^2\left(1-V_b/V_t\right)$. Remembering that the moderator to fuel volume ratio is $z = V_m/V_f = A_m/A_f$, where A_m and A_f are the moderator and fuel areas, respectively, we can write the following relationships,

$$A_f^{(s)} = \frac{X^2}{z_s+1} = \left(m^2 - M\right)\frac{\pi}{4}D_s^2, \tag{1}$$

$$A_f^{(b)} = \frac{L^2}{z_b+1}\times\frac{V_b}{V_t} = \left(17^2 - n^2 - N\right)\frac{\pi}{4}D_b^2, \tag{2}$$

where m and n are the number of grid steps in the seed and empty grid steps in the blanket, $M\left(N\right)$ is the number of guide tubes, $D_s\left(D_b\right)$ is the fuel diameter for the seed (blanket). Equations (1) and (2) can be used to make an initial estimate of the fuel diameters. We have, of course, the condition $D_{s/b} < P_{s/b}$, where $P_s = X/m$ and $P_b = L/17$ are the pitch of the rods. A choice of initial parameters, using $V_b/V_t = 0.72$, is presented in **Table 3**.

With these parameters, we have the same amount of guide and instrument tubes, but 304 fuel rods instead of 264 in the original NuScale design. This configuration is shown in **Figure 6**.

As in the previous study, using this detailed model, we performed a parametric study varying the diameters D_s and D_b of the rods, to find the best conversion factor without exceeding the power limit. Finally, the values of average power, effective multiplication factor and fraction of delayed neutrons in the BOC and EOC were calculated to compare against those of the original NuScale.

3. Results and Discussion

3.1. Reference Model Validation

The study consists of converting the NuScale reactor core from a UO_2 core to

Table 3. Initial parameters for the heterogenous model.

	Parameter				
	Steps	Guide tubes	Fuel rods	Pitch (cm)	Diameter (cm)
Seed	$m = 11$	$M = 9$	112	1.023	0.592
Blanket	$n = 9$	$N = 16$	192	1.251	0.848

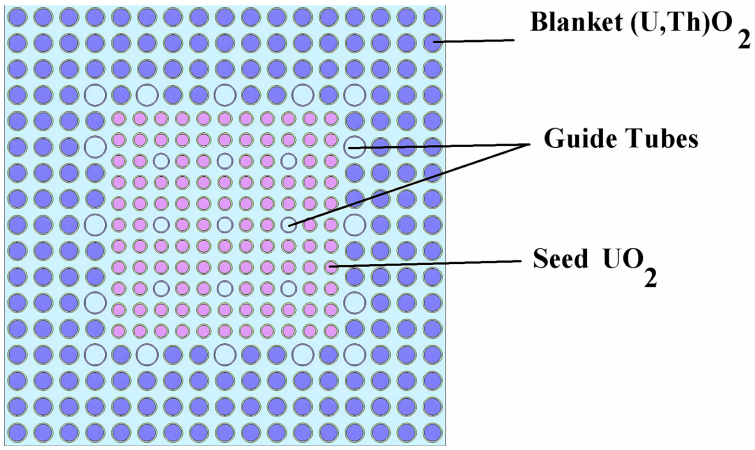

Figure 6. NuScale-Th fuel element configuration.

a (Th, U)O_2 MOX, so we need to ensure that we are modeling the desired reactor. To validate the model, we took the reference [1] and modeled the reactor core in 3 different states: Cold Zero Power (CZP), Hot Zero Power (HZP) and Hot Full Power (HFP). The description of these states can be found in Section 2, and the results obtained are shown in **Table 4**.

In **Table 4**, we note a good compatibility between the reference and modeling values, with a maximum difference of the order of 200 pcm for k_{eff}. This is an acceptable value given that the data were generated using different cross-section libraries, being ENDF/B-VII for our study and ENDF/B-VI for the reference [26]; and different Monte Carlo codes, SERPENT for this study and MCNP for the reference.

3.2. Parametric Study of Seed and Blanket Areas

In the second part of this study, we searched for the optimal ratio of blanket volume to total volume which maximizes the conversion factor and that provides an infinite multiplication factor so that the reactor maintains an excess of reactivity for operation. In this first study, the model was simplified to two homogeneous regions for the seed and blanket, constituted by a weighted mixture with the material proportions of fuel and moderator, so it would be possible to study the model in a simplified way. The proportions taken from the ratio volume of the moderator by the volume of fuel were 3.1 and 1.9 in the seed and in the blanket respectively, these values are the same used by Galperin *et al.* (as cited in Kasten [27]). The values of k_∞ and CR as a function of the ratio of blanket volume to total volume are shown in **Figure 7**.

Table 4. Parameters for validation of the reference reactor modeling.

State	NuScale—Full Core (BOC)	
	Reference [26]	Calculation
CZP (k_{eff})	1.21151 ± 0.00007	1.21045 ± 0.00010
HZP (k_{eff})	1.16658 ± 0.00008	1.16988 ± 0.00011
HFP (k_{eff})	1.13926 ± 0.00009	1.13701 ± 0.00010
β_{eff}	0.00700 ± 0.00020	0.00695 ± 0.00050

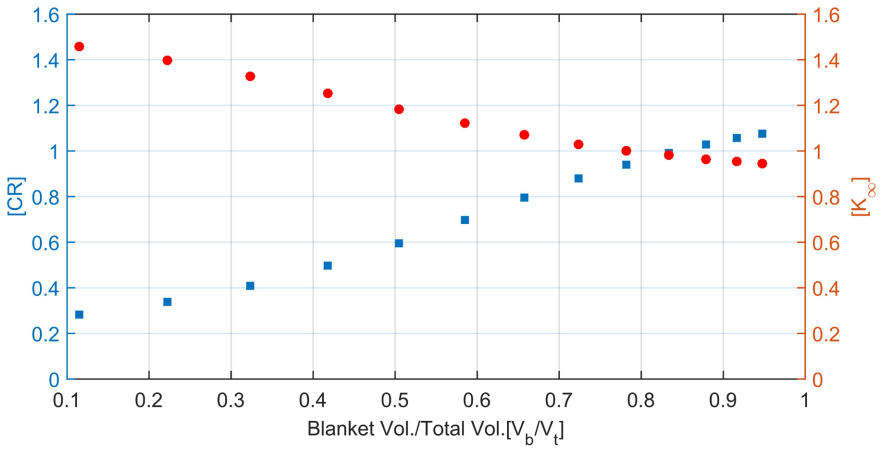

Figure 7. Conversion ratio and infinite multiplication factor as a function of the ratio between blanket volume and total volume.

In **Figure 7**, we note that values (V_b/V_T) greater than 0.8 can be discarded from the study as they result in a subcritical reactor ($k_\infty < 1$). Therefore, the construction of the heterogeneous model should have a ratio slightly less than 0.8, as it is also necessary to maximize the conversion rate at the BOC stage. The chosen value, 0.72, provides a good margin for parameter variation in the heterogeneous model. Results for the chosen model are presented in **Table 5**.

3.3. Modeling of the Seed-Blanket Fuel Element

From the choice of the best ratio between blanket and total volumes, with a k_∞ that remains greater than zero until the end of the cycle, in the third part of this study, we set up the seed-blanket fuel assembly as shown in **Figure 6**.

We note, in **Figure 6**, that the configuration for the blanket is of the type 17 × 17, while for the seed it is 11 × 11. For a better evaluation of the performance of this fuel element, we took the average power per fuel assembly in the NuScale core and modeled a reflected fuel element obtaining the parameters in **Table 6**, along with the equivalent for the reference reactor for comparison.

If we compare the value of k_∞ obtained by the homogeneous reactor in **Table 5** with the value obtained in **Table 6**, we notice that the heterogeneous reactor model has a value greater than what is expected. This can be explained by the formula of the 4 factors [28], given by

Table 5. Results for the homogeneous model (BOC).

Parameter	Homogeneous fuel assembly calculations
	Value
V_b/V_T	0.7235
k_∞	1.030720 ± 0.00008
β_{eff}	0.00686 ± 0.00050
CR	0.88 ± 0.00

Table 6. Reflected fuel element performance in 24 months of burnup.

Table head	Heterogeneous fuel assembly calculations	
	NuScale	SBU
V_m/V_f seed	-	1.70
V_m/V_f blanket	-	2.71
Power (W)	4.32 x 10^6	4.32 x 10^6
CR (BOC)	0.35 ± 0.00	0.48 ± 0.00
CR (EOC)	0.44 ± 0.00	0.52 ± 0.00
U-235 cycle consumption (kg)	3.13	3.22
k_∞ (BOC)	1.42334 ± 0.00009	1.26341 ± 0.00010
k_∞ (EOC)	1.21322 ± 0.00011	1.16182 ± 0.00011
β_{eff} (BOC)	0.00682 ± 0.00017	0.00675 ± 0.00017
β_{eff} (EOC)	0.00583 ± 0.00019	0.00573 ± 0.00020
U-233 production (kg)	**	1.00

**Negligible.

$$k_\infty = f\eta p\varepsilon, \tag{3}$$

where f is the thermal utilization, p is the resonance escape probability, ε is the rapid fission factor, and η is the average of neutrons emitted in fission per neutrons absorbed in the fuel. For the heterogeneous reactor, the thermal utilization will drop due to the depression of the neutron flux within the fuel. However, the resonance escape probability and the fast fission factor will increase, making the value of k_∞ considerably higher, this fact is well known within reactor physics [28].

Analyzing the conversion ratio, we noticed that there was a reduction in the value of the heterogeneous model when compared to the homogeneous model (**Table 5**) due to the self-shielding effect [28] that reduces the magnitude of the neutron flux in the fuel rods.

Analyzing **Table 6**, we note that there was a considerable increase in the conversion factor when the seed-blanket concept is used, despite the k_∞ at the beginning and end of cycle being lower in relation to the reference reactor, we note that the drop in reactivity is considerable (about 50% smaller). Thus, it is expected

that the core of this reactor will achieve an extension of the cycle lifetime or even that the fuel element will be able to remain in the core for more cycles, before its removal is necessary. We noticed a slightly lower consumption of U-235 (~3%) and that the value of the effective fraction of delayed neutrons remains approximately the same. Although, numerically, the value of U-233 seems small, the blanket can be maintained for an irradiation time of several cycles, not needing to be removed together with the seed, in this way it may imply a greater accumulation of U-233 before its removal of the core until its quantity becomes adequate for use. The higher EOC CR value demonstrates that the reactor becomes a better converter as the neutron spectrum becomes harder.

Figure 8 and Figure 9 below demonstrate the production of Pu-239 and Pu-241 in the fuel element during the operating cycle.

In Figure 8 and Figure 9, it is possible to notice that the seed-blanket concept reduces TRU production by more than 50% compared to the standard NuScale fuel element. This is also valid for long-lived actinides, thus representing a considerable reduction in tailings destined for geological deposits and reduction of decommissioning costs.

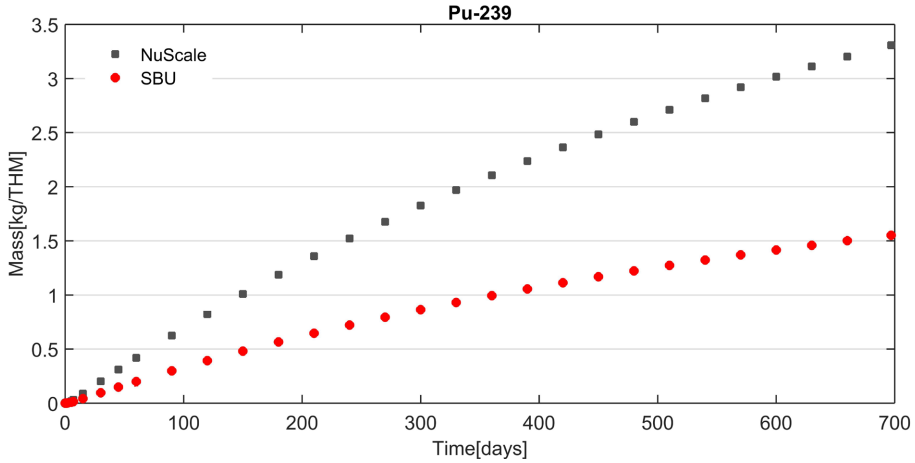

Figure 8. Production of Pu-239 in kg/THM.

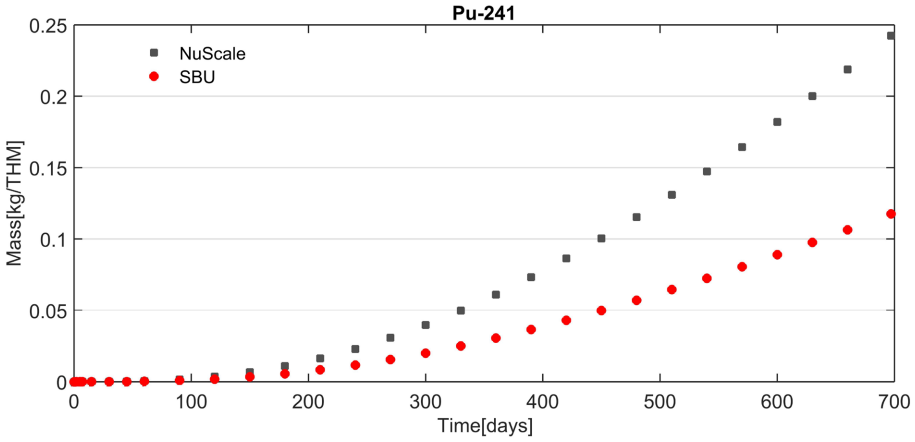

Figure 9. Production of Pu-241 in kg/THM.

Figure 8 demonstrates that Pu-239 is approaching a state of equilibrium, where the amount of material produced is equal to that consumed, while in Figure 9, we can see that Pu-241 is far from reaching this state. However, as the UO_2 fuel (seed) will stay in the core for less time than the blanket composed of (Th, U)O_2, where the amount of plutonium produced is much lower due to the low amount of UO_2 present, this will not affect the reactor at long term. This analysis is important because the blanket can remain in the core for a time many times longer than the seed, for a better use of thorium in the production of U-233.

3.4. Safety Parameters

In previous sections, the viability of the concept was verified and in the last part of this study, the operational safety for the new seed-blanket fuel element should be verified. Table 7 and Table 8 present the data for the temperature reactivity coefficients in BOC for the NuScale reference and the seed-blanket unit fuel elements.

We note that, in Table 7, there was a safety reduction for the temperature reactivity coefficient. This is due to the reduction in the fraction of water in the moderator volume in the seed and consequently in the hydrogen available for moderation, so its reduction ends up having a greater impact on core reactivity.

In the case of Table 8, we notice an increase in the safety of the temperature reactivity coefficient. This increase is due to the resonance integral of thorium being greater than that of uranium, thus increasing neutron capture with temperature [29].

A proprietary Matlab code was developed to verify the thermohydraulic operating limits of the proposed core. We validated the maximum temperature reached in the fuel and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) as reported by the NRC and described in Ref. [30].

Table 7. Moderator temperature reactivity coefficient.

T_{fuel} (K)	T_{mod} (K)	Heterogeneous fuel assembly calculations	
		NuScale	SBU
		α_M (pcm/˚C)	α_M (pcm/˚C)
700	410 - 494	−7.54 ± 0.31	−5.69 ± 0.18
700	494 - 505	−10.79 ± 0.31	−7.34 ± 0.18
700	505 - 516	−12.28 ± 0.31	−10.13 ± 0.18
700	516 - 527	−13.69 ± 0.30	−9.11 ± 0.18
700	527 - 538	−15.47 ± 0.30	−10.13 ± 0.17
700	538 - 549	−15.90 ± 0.30	−13.84 ± 0.18
700	549 - 560	−20.36 ± 0.30	−14.76 ± 0.18
700	560 - 571	−23.06 ± 0.30	−18.41 ± 0.17
700	571 - 582	−28.81 ± 0.30	−21.92 ± 0.18
700	582 - 593	−34.25 ± 0.30	−26.06 ± 0.18

Table 8. Fuel temperature reactivity coefficient.

T_{fuel} (K)	T_{mod} (K)	Heterogeneous fuel assembly calculations			
		Nuscale		SBU	
		(BOC)	(EOC)	(BOC)	(EOC)
		α_F (pcm/˚C)	α_F (pcm/˚C)	α_F (pcm/˚C)	α_F (pcm/˚C)
700 - 800	531	−1.48 ± 0.01	−1.59 ± 0.01	−2.92 ± 0.01	−3.23 ± 0.02
800 - 900	531	−1.25 ± 0.01	−1.78 ± 0.01	−2.55 ± 0.01	−2.65 ± 0.02
900 - 1000	531	−1.51 ± 0.02	−1.60 ± 0.02	−2.77 ± 0.01	−3.16 ± 0.02
1000 - 1100	531	−1.11 ± 0.02	−1.20 ± 0.02	−2.14 ± 0.02	−2.28 ± 0.04
1100 - 1200	531	−1.25 ± 0.01	−1.46 ± 0.01	−2.29 ± 0.02	−2.11 ± 0.03
1200 - 1300	531	−1.35 ± 0.02	−1.60 ± 0.02	−2.17 ± 0.01	−2.70 ± 0.02
1300 - 1400	531	−0.92 ± 0.02	−1.24 ± 0.02	−2.15 ± 0.00	−2.01 ± 0.03
1400 - 1500	531	−1.09 ± 0.01	−1.14 ± 0.01	−1.75 ± 0.02	−2.00 ± 0.03
1500 - 1600	531	−1.23 ± 0.02	−1.48 ± 0.02	−2.13 ± 0.02	−1.95 ± 0.00
1600 - 1700	531	−1.03 ± 0.01	−1.12 ± 0.01	−1.63 ± 0.03	−1.97 ± 0.00
1700 - 1800	531	−0.92 ± 0.01	−1.25 ± 0.01	−1.61 ± 0.02	−1.71 ± 0.02

Table 9. NuScale and SBU thermohydraulic parameters.

Table head	Reactor			
	NuScale (BOC)	NuScale (EOC)	SBU (BOC)	SBU (EOC)
Fuel center line temperature	1110 K	997 K	1579 K	1340 K
Clad maximum temperature	817 K	762 K	1138 K	999 K
MDNBR	2.72	3.37	1.25	1.59

We found a maximum fuel temperature slightly higher than that reported by the NRC for the standard NuScale. This is due to the use of a conservative sub-channel analysis model. For the SBU model, we also obtained an extrapolated maximum temperature value, however, under the thermal limit stipulated for UO_2, which is 2800˚C.

One of the main parameters in evaluating the safety of the reactor is the Minimum Departure from Nucleate Boiling Ratio (MDNBR), which describes the relationship between the critical heat flux and the local heat flux. This is a critical analysis in reactor engineering, as it is essential to ensure that nucleated boiling does not occur in PWR-type reactors. We used the Westinghouse non-uniform W-3 correlation [31] to calculate the critical heat flux. After validating the code used with the standard NuScale core, as shown in **Table 9**, we obtained 1.26 and 1.59 as minimum values for DNBR in BOC and EOC, respectively. According to Todreas and Kazimi [32], the DNBR values are considered safe.

4. Conclusions

The study demonstrated the viability of the seed-blanket concept applied to a

PWR-type SMR reactor, without the need for power reduction, since the lower power of a small reactor ensures that the thermal safety requirements are guaranteed.

In the validation of the reference reactor, a very close value of the effective fraction of delayed neutrons and effective multiplication factor was found, with a maximum difference of the order of 250 pcm. The value is acceptable considering that different cross-section libraries and codes were used.

In the parametric study carried out for the homogeneous seed-blanket reactor, a model with a conversion ratio of 0.87 and $k_{\infty} = 1.030720 \pm 0.00008$ was chosen, which, when transformed into a heterogeneous reactor, had its k_{∞} value increased to 1.26186 ± 0.00009 and the conversion factor reduced to 0.48 in BOC. These effects are expected due to changes in the 4-factor formula with increased resonance escape and fast fission factor, as well as the self-shielding effect.

The produced value of the U-233 isotope (1 kg) represents 1/3 of the mass burned of U-235, as each cycle is expected to remove approximately 1/3 of the reactor mass expected for the reactor model complete to be developed in the next phase of the study represents an almost sustainable production of fissile material. The use of devices such as neutron reflectors or even redistribution of flux by increasing the fissile material in the blanket can make the concept even more efficient. A possible way to increase the amount of fissile material in the blanket would be to change the MOX from (U, Th)O_2 to (Pu, Th)O_2. This would also contribute to the improvement of DNBR, which turned out to be the main safety factor affected in this analysis. The temperature reactivity coefficients and the fraction of delayed neutrons had little change from those for the model used as a reference.

Finally, the amount of TRU produced by the reactor had a reduction of ~50% of its mass value, when compared to the reference model. In the next stage of this study, this suggested fuel element concept will be examined and the complete reactor core will be modeled and studied throughout the first cycle. A more detailed analysis of the safety parameters will be carried out at the beginning, middle and end of the cycle.

Thus, we can list the main results of this study as the increase in the conversion value, reduction of TRU produced, production of fissile uranium and extension of the fuel cycle.

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Conflicts of Interest

The authors declare no conflicts of interest regarding the publication of this paper.

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