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Neutronic evaluation of initial NuScale-like core loaded with reprocessed fuel

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Abstract. The present study evaluates the potential insertion of reprocessed fuel in a NuScale-like core. The main idea consists of assessing the beginning of life (BOL) safety parameters of the core loaded with reprocessed fuel and comparing them to the reference case using conventional uranium-based fuel. Two different reprocessed fuels were studied: one of them reprocessed using the GANEX process, while the other one used the UREX+ process. Overall, the cases in which the core was simulated with reprocessed fuel achieved better results than conventional UO_2 for fuel temperature coefficient and normalized peak factor. Furthermore, the usage of reprocessed fuel could also reduce the dependence on chemical control for reactivity, which is a desired condition in the event of borated water unavailability. The Serpent code version 2.1.32 developed by VTT was used to perform the simulations.

1 Introduction

The nuclear power generation industry faces some notable issues regarding its potential beneficial exploitation in a sustainable future context. To compete and eventually overcome other technologies, the industry must concentrate efforts on adhering to three main aspects [1]: (1) economic attractiveness, which includes reducing construction time and costs of installations, (2) demonstrating safety features improvements, (3) implementing high-level waste management strategies to deal with conventional solid spent fuels. With these prime considerations, this kind of energy source may indeed play a crucial role in a road map for the global transition towards a CO_2 free power generation.

The so-called small modular reactors (SMRs) concept emerges as a response to this scenario, especially addressing the first two aspects. SMRs are advanced nuclear reactors with a power capacity of up to 300 MWe per unit [2]. Compared to large nuclear power plants, they are simpler in design, smaller, and have flexible power generation, which justifies their utilization not only in large-demand countries but also in countries with a small and mixed-capacity electric grid. Moreover, the modularization of systems makes it possible for components to be manufactured and assembled at the factory, and posteriorly shipped as a whole to a site for installation and operation.

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Transitioning to the safety viewpoint, SMRs often rely on inherent safety, passive systems, and lower operating parameters such as thermal power and pressure. This means that no human intervention or external power is required to shut down the reactor in a variety of cases of deviation from normal operation state, mainly because critical safety systems work through physical phenomena [3]. Therefore, natural circulation, gravity, and self-pressurization characteristics assure a safer operation of SMRs installations by design.

Several advantages can be outlined from the SMRs concept. Combining those advantages with the operational experience of Pressurized Water Reactors (PWRs), it is quite reasonable that most of the SMRs in development utilize PWR technologies. So far, the NuScale Power Modular and Scalable Reactor, developed by NuScale Power LLC, stands as the most mature SMR design and the first one to receive approval and certification by the US Regulatory Commission (USNRC) [4].

Despite its potential adherence to the mentioned aspects, the question regarding high-level waste management remains. The NuScale core utilizes conventional enriched uranium dioxide (UO_2) as fresh fissile material to maintain the fission reaction chain, also by-producing plutonium and other minor actinides remarkably known by their high radiotoxicity, as an example of large nuclear power plants. The approach to managing this nuclear-spent fuel involves its disposal to a final deep geological repository without reprocessing or the reprocessing of spent nuclear fuel to extract fissile materials for recycling, consequently reducing the volume of high-level waste.

The present study evaluates the potential insertion of reprocessed fuel in a NuScale-like core. The main idea consists of assessing the beginning of life (BOL) safety parameters of the core loaded with reprocessed fuel and comparing them to the reference case using conventional uranium-based fuel. Two different reprocessed fuels were studied: one of them reprocessed using the GANEX (Group ActiNide EXtraction) process [5], while the other one using UREX+ process [6], both after three burnup cycles of approximately 12 MWd/TU of NuScale initial core. The Serpent code version 2.1.32 [7] developed by VTT was used to perform the simulations.

2 Methodology

2.1 Reference case

This work simulated the NuScale-like core according to the existing benchmark and was taken as the reference case [8]. With an active length of 200 cm and outer radii of 99.06 cm, 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 MWth.

Fig. 1 shows the (a) core loading pattern and (b) control rod locations. As can be seen in Fig. 1 (a), the core comprises A, B, and C assemblies groups, 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, totaling 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 Fig. 1 (b)). Each regulating and shutdown CRAs includes two banks that move 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). Fig. 2 shows the radial layouts of the lattices of un-poisoned (left) and poisoned (right) fuel assemblies. 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 deeply described in [8].

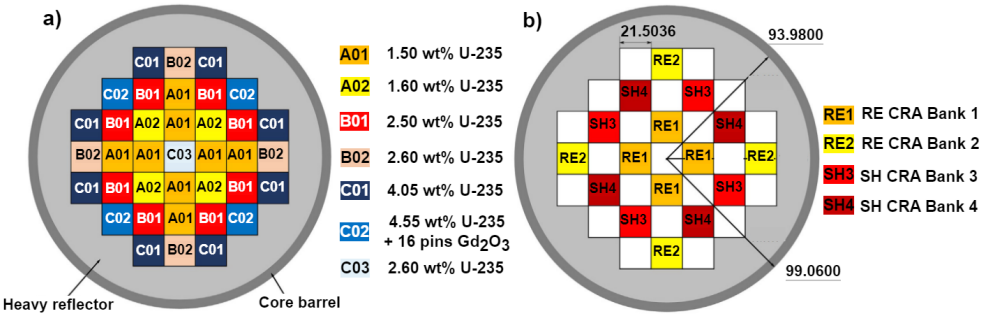


Figure 1. (a) Core loading pattern and (b) control rod assembly locations [8]. Dimensions are in cm.

Table 1. Isotopic composition of all fuel types in weight fraction [8].

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
sum	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00

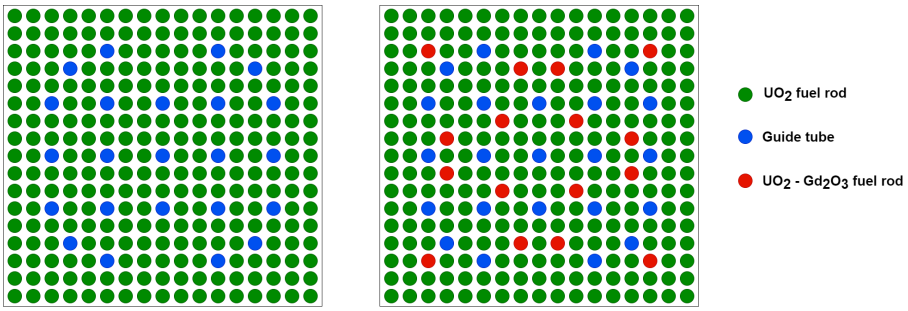


Figure 2. Radial layouts of fuel assemblies: un-poisoned (left) and poisoned (right) [8].

2.2 Reprocessed fuel compositions

The burnup of the initial NuScale-like core described in subsection 2.1 was simulated in SERPENT Monte Carlo [7] code for 24 months, corresponding to its fuel length cycle. After that, the core configuration was changed for a second burnup cycle, reallocating C01 fuel assemblies to B01 positions, as well as C02 fuel assemblies to B02 positions. B01 and B02 fuel assemblies underwent the same reallocating process to A01 and A02 positions, while A01 and A02 fuel assemblies were extracted from the core. Fresh fuel assemblies were used in C01, C02, and C03 positions according to their respective isotopic composition described in Table 1.

Subsequently, the core configuration for the third cycle in turn also had a distinct layout. The new configuration involved fuel assemblies burned twice in the A01 and A02 positions, fuel assemblies burned once in the B01 and B02 positions, and finally fresh fuel assemblies in the C01, C02, and C03 positions. Each burnup cycle achieved approximately 12 MWd/TU, following the NuScale power burnup of cycles 1, 2, and 3 and their corresponding fuel assemblies disposition. After three cycles of burnup and a cooling time of five years, which was also simulated using SERPENT Monte Carlo, the three-times-burned spent fuel located in the A01 and A02 positions was theoretically reprocessed by GANEX and UREX+ reprocessing techniques.

The GANEX process developed by *Commissariat à l´énergie atomique et aux energies alternatives* (CEA) for the reprocessing of Generation IV spent nuclear fuels is composed of two extraction cycles following the dissolution of the spent fuel [5]. In terms of GANEX results, 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 and A02 spent fuel assemblies were spiked with 71.7% and 83.9% of thorium, respectively, aiming for a decrease of initial reactivity caused by the massive presence of fissile isotopes in reprocessed fuels. Tab. 2 shows the final composition of the reprocessed fuel for both cases.

On the other hand, the UREX, a variant of the 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 [6]. In total, there are eight different UREX+ processes under commercial development.

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 and A02 spent fuel assemblies, the isotopic compositions were spiked with 72% and 84% of thorium, respectively. Tab. 2 shows the final composition of the reprocessed fuel for both cases.

Reprocessed fuel mixture with thorium content is an attractive alternative because it is about three times more abundant than uranium and is distributed in nature as an easily exploitable resource in many countries. Moreover, as fertile material, the ^{232}Th isotope can transmute by neutron absorption in ^{233}U, which is a fissile material capable of undergoing fission reactions. The percentages of thorium were chosen in order to achieve the same initial k_{eff} obtained from the NuScale-like core taken as a reference case.

Table 2. Composition of reprocessed fuel spiked with thorium after GANEX and UREX+ process obtained from A01 and A02 three-times-burned spent fuel assemblies.

GANEX							
A01: spiked with 71.7% of thorium				A02: spiked with 83.9% of thorium			
Isotope	weight frac.	Isotope	weight frac.	Isotope	weight frac.	Isotope	weight frac.
Th-232	6.30748E-01	Am-241	8.76247E-03	Th-232	7.37752E-01	Am-241	4.97555E-03
U-232	1.93643E-13	Am-242	2.20980E-05	U-232	1.16670E-13	Am-242	1.22597E-05
U-233	3.83121E-12	Cm-242	2.29330E-07	U-233	2.31595E-12	Cm-242	1.33191E-07
U-234	2.85574E-08	Cm-243	8.58403E-06	U-234	1.64504E-08	Cm-243	4.91668E-06
U-235	2.42873E-05	Cm-244	7.72578E-04	U-235	1.63387E-05	Cm-244	4.00462E-04
U-236	1.09817E-05	Cm-245	6.36702E-05	U-236	6.93316E-06	Cm-245	3.14992E-05
U-237	7.67990E-14	Cm-246	6.01881E-06	U-237	4.39450E-14	Cm-246	2.91200E-06
U-238	2.04608E-03	Cm-247	7.64547E-08	U-238	1.15728E-03	Cm-247	3.50789E-08
U-240	1.12317E-21	Cm-248	5.21300E-09	U-240	5.85504E-22	Cm-248	2.29600E-09
Pu-236	2.44187E-10	Cm-250	2.44713E-16	Pu-236	1.30800E-10	Cm-250	1.05428E-16
Pu-237	4.34178E-21	Np-235	1.93412E-12	Pu-237	2.50556E-21	Np-235	1.09559E-12
Pu-238	4.83442E-03	Np-236	4.90933E-08	Pu-238	2.76779E-03	Np-236	2.79253E-08
Pu-239	1.27576E-01	Np-237	1.15540E-02	Pu-239	7.26495E-02	Np-237	6.88107E-03
Pu-240	5.25318E-02	Np-238	3.99996E-12	Pu-240	2.94495E-02	Np-238	2.21916E-12
Pu-241	2.46162E-02	Np-239	2.36887E-09	Pu-241	1.40853E-02	Np-239	1.28349E-09
Pu-242	1.21875E-02	Nd-144	9.53044E-04	Pu-242	6.86507E-03	Nd-144	6.23636E-04
Pu-243	2.72592E-18	Sm-150	2.97787E-04	Pu-243	1.25067E-18	Sm-150	1.21767E-04
Pu-244	5.68683E-07	Eu-152	3.17566E-05	Pu-244	2.96441E-07	Eu-152	2.02168E-05
Am-243	2.76585E-03	O-16	1.20296E-01	Am-243	1.49851E-03	O-16	1.20677E-01
	sum =	1.0E+00			sum =	1.0E+00	
UREX+							
A01: spiked with 72% of thorium				A02: spiked with 84% of thorium			
Isotope	weight frac.	Isotope	weight frac.	Isotope	weight frac.	Isotope	weight frac.
Th-232	6.33442E-01	Am-243	2.65721E-03	Th-232	7.38676E-01	Am-243	1.44791E-03
U-232	9.49170E-13	Am-241	8.41830E-03	U-232	5.75151E-13	Am-241	4.80752E-03
U-233	1.87792E-11	Am-242	2.12300E-05	U-233	1.14170E-11	Am-242	1.18457E-05
U-234	1.39978E-07	Cm-242	1.77607E-07	U-234	8.10964E-08	Cm-242	1.03742E-07
U-235	1.19048E-04	Cm-243	6.64799E-06	U-235	8.05457E-05	Cm-243	3.82960E-06
U-236	5.38285E-05	Cm-244	5.98330E-04	U-236	3.41787E-05	Cm-244	3.11920E-04
U-237	3.76441E-13	Cm-245	4.93100E-05	U-237	2.16638E-13	Cm-245	2.45347E-05
U-238	1.00292E-02	Cm-246	4.66132E-06	U-238	5.70511E-03	Cm-246	2.26815E-06
U-240	5.50538E-21	Cm-247	5.92111E-08	U-240	2.88638E-21	Cm-247	2.73229E-08
Pu-236	2.38187E-10	Cm-248	4.03726E-09	Pu-236	1.28317E-10	Cm-248	1.78835E-09
Pu-237	4.23509E-21	Cm-250	1.89520E-16	Pu-237	2.45800E-21	Cm-250	8.21179E-17
Pu-238	4.71563E-03	Np-235	1.35298E-12	Pu-238	2.71526E-03	Np-235	7.70795E-13
Pu-239	1.24441E-01	Np-236	3.43423E-08	Pu-239	7.12704E-02	Np-236	1.96466E-08
Pu-240	5.12409E-02	Np-237	8.08241E-03	Pu-240	2.88905E-02	Np-237	4.84111E-03
Pu-241	2.40113E-02	Np-238	2.79809E-12	Pu-241	1.38179E-02	Np-238	1.56127E-12
Pu-242	1.18880E-02	Np-239	1.65709E-09	Pu-242	6.73476E-03	Np-239	9.02989E-10
Pu-243	2.65893E-18	O-16	1.20220E-01	Pu-243	1.22693E-18	O-16	1.20624E-01
Pu-244	5.54708E-07			Pu-244	2.90814E-07		
	sum =	1.0E+00			sum =	1.0E+00	

2.3 In-core distribution of fuel assemblies containing reprocessed fuel

Fig. 3 indicates the two possibilities for reprocessed fuel insertion in the presented NuScale-like core, changing the compositions of referred fuel assemblies containing UO_2 to the reprocessed fuel compositions proposed in subsection 2.2. The first possibility involves the fuel reprocessed from A01 spent fuel assemblies inserted in C01 positions (Fig. 3 (a)), while the other one includes C01 and C02 positions for insertion of fuel reprocessed from A01 and A02 spent fuel assemblies (Fig. 3 (b)), respectively. Therefore, aside from the uranium-based NuScale reference core, four cases were performed, as itemized above to facilitate the understanding:

- **case 1:** fuel reprocessed from A01 spent fuel assemblies by GANEX process and spiked with 71.7% of thorium. Inserted at C01 positions;
- **case 2:** fuel reprocessed from A01 and A02 spent fuel assemblies by GANEX process and spiked with 71.7% and 83.9% of thorium, respectively. Inserted at C01 and C02 positions.
- **case 3:** fuel reprocessed from A01 spent fuel assemblies by UREX+ process and spiked with 72% of thorium. Inserted at C01 positions.
- **case 4:** fuel reprocessed from A01 and A02 spent fuel assemblies by UREX+ process and spiked with 72% and 84% of thorium, respectively. Inserted at C01 and C02 positions.

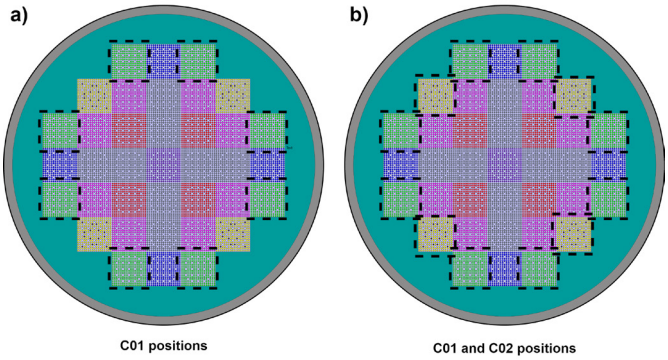


Figure 3. Core loading pattern with reprocessed fuel at (a) C01 and (b) C01 and C02 positions.

In other words, each one of the two core configurations was simulated with reprocessed fuel from both reprocessing techniques, and their corresponding safety parameters results were compared to investigate the feasibility of reprocessed fuel insertion in the NuScale-like core. The SERPENT Monte Carlo code version 2.1.31 [7] and ENDF/B-VII nuclear data library [9] were used to perform all simulations. SERPENT code is used for various applications in reactor physics, including steady-state simulations of reactor-modeled cores. The code utilizes the Monte Carlo method to simulate neutron interaction physics based on classical collision kinematics and ENDF reaction laws.

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, as indicated by the FSAR of NuScale approval and certification [4], and the core was simulated with all control rods fully withdrawn. In the developed model for steady-state 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

3.1 Reference case validation

The present study consists of converting the NuScale-like reactor core from a conventional UO_2 core to a core containing reprocessed fuel. To validate the model, we took a reference neutronic benchmark for the NuScale reactor [8] and modeled the reactor core at six different conditions, varying the insertion of control rod banks positioned as shown in Fig. 1. The obtained results are compared to the reference case in Table 3.

It is notable the good compatibility between the reference results and the obtained values from the present model, with a maximum relative difference of 262 pcm for k_{eff}. This is an acceptable value given that the output from the reference case averages several independent simulations performed in the framework of the Euraton McSAFER project.

Table 3. Parameters for validation of the reference reactor modeling.

	Benchmark [8]		Calculation by author		
Core state	k_{eff}	σ_{eff}	k_{eff}	σ_{eff}	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

One of the most important parameters regarding the safe operation of a nuclear reactor involves the reactivity insertion in response to variations in the average temperature of the fuel and moderator material. To investigate the feasibility of reprocessed fuel in a NuScale-like core, Table 4 and Table 5 present a comparison of the fuel and moderator temperature reactivity coefficients for the reference case and the four cases utilizing reprocessed fuel.

Table 4 reveals that all instances in which the core was simulated with reprocessed fuel exhibited lower values than the reference case. This behavior implies that in the event of an increase in fuel temperature, the cores loaded with reprocessed fuel demonstrate a more negative reactivity insertion, leading to a rapid reduction in neutron population, as desired from a safety perspective.

It is also notable that case 2 and case 4, which have greater amounts of reprocessed fuel in the core (C01 and C02 positions), demonstrated lower values than the simulated core with reprocessed fuel located only in C01 positions, which are case 1 and case 3. 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 reference core has achieved better performance overall than the cases with reprocessed fuel in terms of moderator temperature reactivity coefficient. Aside

from case 4, the remaining cases exhibited a small safety reduction for this parameter, although closely approaching the average reference value of α_M =-21.10439. On the other hand, case 4 exceeded the average reference value, which means that more negative reactivity would be inserted in the event of increasing temperature of moderator material. 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 all cores simulated with reprocessed fuel, as an example of the NuScale-like core taken as reference.

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

			α_f		
Δ_T (K)	reference case	case 1	case 2	case 3	case 4
300 - 400	-3.29791	-3.53289	-4.03864	-4.12881	-3.86429
400 - 500	-3.41530	-3.38107	-3.90997	-3.71044	-4.09192
500 - 600	-2.75719	-3.15984	-3.31420	-2.84148	-3.20676
600 - 700	-2.54112	-2.79075	-2.67395	-2.69408	-2.98383
700 - 800	-2.15781	-2.66215	-3.07088	-2.48705	-2.37914
800 - 900	-2.58241	-2.37631	-2.31824	-2.70809	-3.08783
average	-2.79196	-2.98384	-3.22098	-3.09499	-3.26896

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

			α_M			
ΔT (K)	$\Delta\rho$ (g/cm^3)	reference case	case 1	case 2	case 3	case 4
550 - 560	0.76586 - 0.74766	-5.06334	-11.88087	-8.66703	-9.06114	-13.63299
560 - 570	0.74766 - 0.72778	-7.89335	-19.22157	-18.63469	-9.26458	-19.61147
570 - 580	0.72778 - 0.70573	-23.05092	-10.84217	-14.26463	-18.95512	-15.53766
580 - 590	0.70573 - 0.68067	-31.84494	-28.30863	-27.13717	-27.04000	-24.58831
590 - 600	0.68067 - 0.65106	-37.66941	-27.79796	-34.40682	-33.02698	-32.62931
average		-21.10439	-19.61024	-20.62207	-19.46956	-21.19995

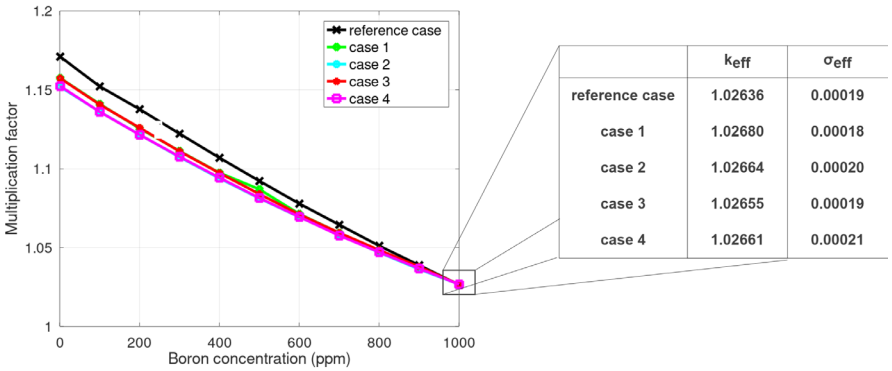
Fig. 4 depicts a parametric study varying the boron concentration in the coolant materials, ranging from 0 ppm to operation state, namely 1000 ppm. All control rods (including regulating and shut-down) are assumed to be fully withdrawn, while the thermal power was settled to 160 MW. In each simulated case with reprocessed fuel insertion, the maximum excess of reactivity was considerably lower if compared to the reference case, especially in the absence of boron dissolution in the coolant.

Furthermore, the multiplication factor decreases smoother in cases using reprocessed fuel in the core given an increase in boron concentration. According to these findings, the utilization of reprocessed fuel in the NuScale-like core could reduce the dependence on chemical control, which is a desired condition in the event of borated water unavailability. For values higher than 600 ppm, all cases simulated with reprocessed fuel approaches to the reference case, culminating in the very same values for k_{eff} at 1000 ppm used for simulations with the core at BOL conditions.

Another constraint and criterion on the safe margin of operations is the power peak factor (PPF). This parameter is calculated through the ratio between the normalized radial power distribution by the average power generated in all fuel assemblies. Fig. 5 depicts the normalized radial power peak factor for the reference case and all cases containing reprocessed fuel in a 1/4 symmetry.

For the reference case, the power distribution strongly concentrates in the C03 and B01 positions, which results in higher values of power peak factor for these fuel assemblies, as highlighted in red. The changing of C01 uranium-based fuel assemblies for the reprocessed fuel assemblies enables more power to be produced in the mentioned position, lowering the peak factor of B01 positions. This behavior can be observed in Fig. 5 (b) and (d), respectively for the fuel reprocessed by GANEX and UREX+ techniques.

Similarly, the cases in which the core was simulated with reprocessed fuel located in C01 and C02 positions flatter the radial power distribution, which is evidenced by the lower peak factors in C03 and B01 positions than the reference case. Consequently, more power is generated in C01 and C02 positions. Overall, the most flattered radial power was achieved using fuel reprocessed by the GANEX technique in the C01 and C02 positions (Fig. 5 (c)) because it also reduced the peak factor in the center position (C03) and distributed power more equally throughout all positions inside the NuScale-like core.



	k_{eff}	σ_{eff}
reference case	1.02636	0.00019
case 1	1.02680	0.00018
case 2	1.02664	0.00020
case 3	1.02655	0.00019
case 4	1.02661	0.00021

Figure 4. Multiplication factor as a function of boron concentration in the coolant.

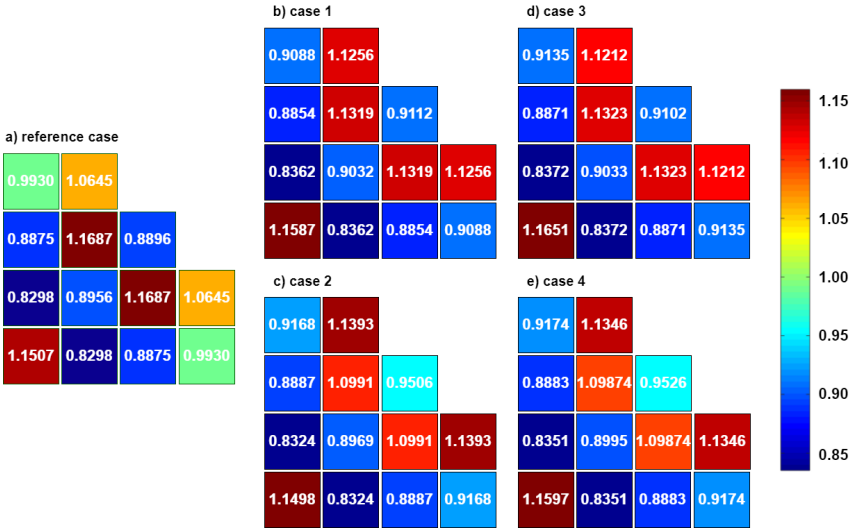


Figure 5. Normalized power peak factor for all cases in 1/4 symmetry.

4 Conclusions

Following the certification of NuScale for mixture oxide fuel (MOX) utilization, the present study evaluates the potential insertion of reprocessed fuel in a NuScale-like core by assessing the beginning of life (BOL) safety parameters of the core loaded with reprocessed fuel and comparing them to the reference case using conventional uranium-based fuel. Aside from the reference case, four cases using fuel theoretically reprocessed by GANEX and UREX+ reprocessing techniques were performed. Two of them were reprocessed using the GANEX process and spiked with 71.7% and 83.9% of thorium, posteriorly inserted in C01 and C02 positions, respectively. The other ones were reprocessed by the UREX+ technique and spiked with 72% and 84% of thorium, also inserted in the same positions.

It was shown that no problems related to a recursive increase in the core temperature and corresponding positive reactivity insertion would be caused in all cores simulated with reprocessed fuel. 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, Case 4, which represents the core loaded with fuel reprocessed by the UREX+ technique and inserted in C01 and C02 positions, achieved a more negative value for these parameters than all other cases (including reference case), thereby improving safety aspects regarding NuScale reactor operation.

Furthermore, the usage of reprocessed fuel evaluated in the four cases could also allow a reduction of dependence on chemical control, which is a desired condition in the event of borated water unavailability. Finally, the most flattered radial power was achieved using fuel reprocessed by the GANEX technique in the C01 and C02 positions once it reduced the peak factor in the center position (C03) and distributed power more equally through all positions inside the NuScale-like core. The overall results presented in this paper show a feasible path for the usage of reprocessed fuel in the referred core, even though burnup calculations still need to be performed, where we will further concentrate efforts.

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