Id.: 967882

Neutronics Analyses of a NuScale Power Reactor

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Key words: Micro Nuclear Reactor, Neutronics,PWR.

Area: Nuclear Engineering
Sub-area: Neutronic Analysis of Nuclear Systems

Abstract

The study presents a comprehensive neutronic analysis of the NuScale Power Reactor, a small, modular pressurized water reactor (PWR) designed in the United States. With a compact, integrated, and modular architecture, NuScale reactors aim to deliver approximately 77 MW(e) and 250 MW(t), supporting energy needs in remote areas or industrial applications with an operational span of up to 18 months without refueling. The design's compact nature optimizes site usage while maximizing energy production efficiency. This research utilizes the Serpent Monte Carlo code for neutronic modeling to evaluate critical safety and operational parameters, including the effective multiplication factor (k_{eff}), reactivity coefficients, and control rod assemblies (CRA) configurations. The Serpent code is complemented with computational fluid dynamics (CFD) for thermal-hydraulic modeling, detailing the heat transfer and fluid dynamics behavior within the core. Results confirm alignment with expected benchmarks, supporting NuScale's design and safety objectives and setting a foundation for future adaptations, such as integrating thorium as an alternative fuel. This analysis contributes valuable insights into the deployment potential of SMRs for sustainable, scalable nuclear energy solutions.

1. Introduction

Delays in the construction of nuclear reactors in recent decades have been faced by many countries: Vogtle-3 and Vogtle-4 in the USA [1], Olkiluoto 3 in Finland[2] and Flamanville-3 in France[3]. This difficult reality required the nuclear industry to "reinvent" itself. These delays increased construction costs by up to 3 times the initial estimate. Exactly during the renaissance of nuclear energy in the 21st century, in the transition between the current installed base and generation III and IV reactors, in response to this tragedy from the perspective of investment and return, nuclear technology matured and proposed almost 100 new concepts of small modular reactors (SMRs). Their construction time of 3 to 4 years, in addition to installation and licensing requirements that are more lenient than large reactors[4]. These factors make SMRs competitive with other energy sources, allowing for more efficient development of the nuclear industry. Furthermore, their reduced size means they can be manufactured on an assembly line, which is not possible with large reactors. Among the reactors in the most advanced implementation phase is NuScale, whose first unit in the USA is scheduled to begin operating in 2030.

The VOYGR^{TM} from Nuscale is a small light-water-cooled pressurized-water reactor (PWR), the main design is to feature more than one module or as they call NuScale Power Module^{TM} each one of this module is capable of provide approximately 250 MW(t) and 77 MW(e) (gross), each module can operate is a self-contained and operates independently of the others in a multi-module configuration, as the same time they are all managed from one single control room. The application of the reactor is for electricity production, with a flexible operation to load follow and for non-electrical process using heat application, wich includes the cogeneration of heat and electricity.

In this work, some thermal and neutronic calculations of the reactor were proposed, to validate computational models to describe its behavior and compared with reference [3] so that it could be used in future analyses with other projects from our research group in converting SMR reactors for use in the thorium fuel cycle. The reference reactor is essential for quantifying the performance of new reactors that will be developed.

2. General description of VOYGR^{TM}

The radial core layout is loaded with 37 fuel assemblies at six different u-235 enrichments levels, four fuel assemblies contain 16 fuel pins with Gd_2O_3 burnable poison, in total there are 7 different fuel assembly types. The core has 16 control rod assemblies (CRA) divided into two regulating and others two shutdown, surrounded by heavy steel reflector and bounded by one cylindrical core barrel (Fig. 1) [5][6].

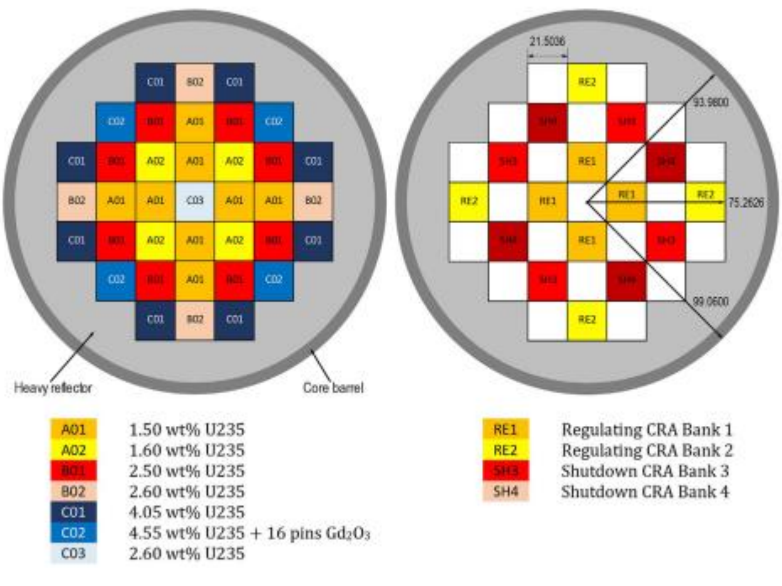

Fig 1. First core load pattern of NuScale core. **[6]**

The axial view of NuScale core is showed in Fig. 2.

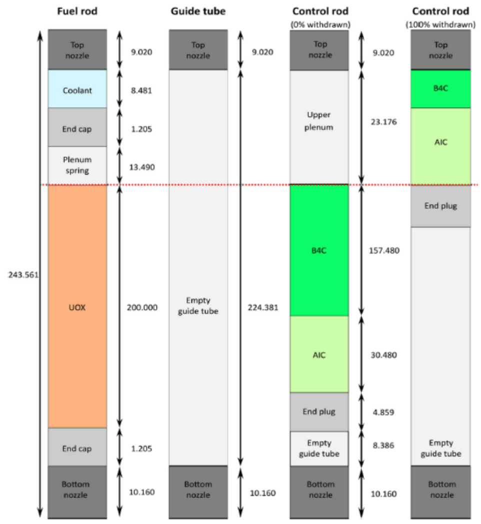

Fig. 2 Axial structure and axial alignment of the different components of the NuScale core. **[6]**

All fuel assemblies in $\mathbf{VOYGR}^{TM}$ has the layout of 17x17 lattice (Fig. 3), each fuel rod has a pitch in fuel assembly by $1.2598\ cm$, the core has $21.5036\ cm$, the actual BP configuration is not publicly available, but for this study is assume the configuration provide by [6].

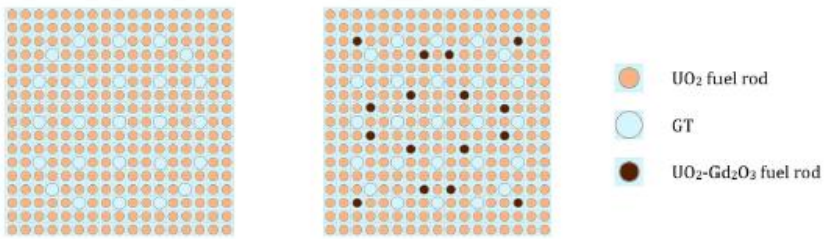

Fig. 3 Radial layouts of fuel assemblies [6]

3. METODOLOGY

3.2 Neutronic model

We use the Serpent code, a three-dimensional continuous-energy Monte Carlo particle transport code, developed for several purposes including reactor physics calculations encompassing fuel cell and assembly calculations, spatial homogenization, few energy-group cross-section generation, full core criticality calculations and fuel cycle studies [7][8]. The Serpent code version we utilized has a cross-section library based on the ENDF/B.VII.0 data files with data at the following temperatures: 300 K, 600 K, 900 K, 1500 K and 1800 K. Although one finds in the literature thermal-hydraulic interfaces for the Serpent code [9] we adopted flat temperature profiles for the fuel and moderator temperatures. In this study, no thermal-hydraulic feedback was used in the calculations to correct cross-sections due to variable fuel and moderator temperature distributions that occur at power conditions.

The calculations in the Serpent for determination of integral parameters such as k_{eff} were done using 200,000 histories and 2,000 cycles and for determination of differential quantities such as neutron flux and power density distribution, 4,000,000 histories and 2,000 cycles, in both cases were done using 100 inactive cycles. The computational cost is much higher for 4,000,000 stories and so it was only used in strictly necessary cases. The burnup calculations consider depletion zones for the assemblies of each fuel enrichment region with 50 axial divisions of 8.534 cm. The depletion steps were 1 day for the first 7 days to account for xenon effects and 30 days for the remaining effective days.

3.1 Thermal-hydraulic Model

We used the single heated channel analysis methodology in computational fluid dynamics (CFD) developed by [10] for the AP1000 reactor core and the AP-Th1000 concept. We adapted the standard configurations outlined in the submitted final safety analysis report to the NRC by NuScale [11] to meet the NuScale US600 Small Modular Reactor (SMR) operating conditions.

The adaptation process involved reorienting the methodology to analyze an internal subchannel with the coolant cell-centered to enhance the meshing stage. We utilized the Ansys SpaceClaim tool to create a three-dimensional model of the NuScale US600 subchannel based on the dimensions and characteristics outlined in the report [11]. Additionally, we leveraged the symmetric characteristics of the region to optimize the modeling, as depicted in Fig. 4.

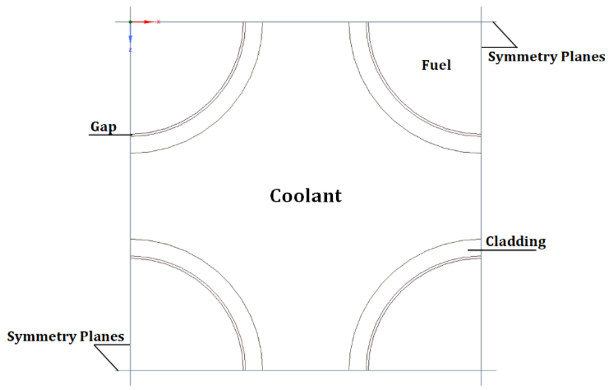

Fig. 4. Subchannel radial layout.

This approach enabled a more precise and efficient representation of the thermal and hydraulic behavior within the subchannel, aligning with the operational and safety requirements of the NuScale US600 SMR. We applied the Ansys Meshing tool to the subchannel region discretization, generating a mesh with 9211630 nodes and 8452000 elements, primarily hexahedral. We adopted the material definitions used by [10], with adjustments made to the enrichment of uranium dioxide and the substitution of Zircaloy-4 with M5 in the fuel rod cladding, in line with the AP1000 and NuScale reactor designs, respectively [11-12]. We use the thermophysical properties of M5 based on the work of [13].

To accurately capture the heat transfer process between the fuel rod and the coolant, we employed the k-omega SST turbulence model. Inlet and outlet boundary conditions were set as the mass flow and outlet pressure, respectively, along with symmetry constraints at the corner edges. According to the NuScale US600 project documentation, the core operates at a mass flow.

4. Results

4.1. Neutronic analysis

After modeling the reactor using the SERPENT Code we conclude the following results for Kef, CRA (Separate in RE1, RE2, SH3 and SH4) and the Reactivity in total CRA banks, all the results are presented in Tab. 4.

Tab. 4. Neutronic calculations in differente states

	Reference [6]		This study	
Core state (Fig. 1)	$keff$	σ_{keff}	$keff - Study$	$\sigma_{keff} - Study$
Rods Out	1.02768	0.00001	1.02760	0.0010
RE1 in	1.00723	0.00001	1.00693	0.00012
RE2 in	1.00313	0.00001	1.00299	0.00013
SH3 in	0.98978	0.00001	0.99015	0.00009
SH4 in	0.98971	0.00001	0.98920	0.00013
All Rods in	0.85791	0.00002	0.85691	0.00013

* 1000 pcm of boron in moderator in Begin of Cycle

Tab. 5. Reactivity temperature coefficient

Fuel reactivity temperature coefficient			Moderator reactivity temperature coefficient		
Fuel Temp. (K)	Mod. Temp. (K)	α_F (pcm/K)	Fuel Temp. (K)	Mod. Temp. (K)	α_M (pcm/K)
600-700	600	-2,71	900	494,15-499,15	-19,21
700-800	600	-2,11	900	499,15-505,15	-17,55
800-900	600	-2,29	900	505,15-510,15	-20,42
900-1000	600	-2,26	900	510,15-516,15	-18,80
1000-1100	600	-1,68	900	516,15-521,15	-21,08
1100-1200	600	-2,00	900	521,15-527,15	-21,94
1200-1300	600	-1,85	900	527,15-533,15	-22,71
1300-1400	600	-1,87	900	533,15-538,15	-24,24
1400-1500	600	-1,63	900	538,15-544,15	-26,15
1500-1600	600	-1,95	900	544,15-549,15	-28,23
1600-1700	600	-1,38	900	549,15-555,15	-29,19
1700-1800	600	-1,76	900	555,15-560,15	-32,60

In Tab. 4, it is possible to observe a high compatibility with the values obtained for keff in different reactor states that may be linked to several factors such as the code, different nuclear library used or even a smaller neutron population for the simulations, not reaching the same convergence as the reference. Additionally, we also calculated the temperature reactivity coefficients, where although the reference did not make these calculations for comparison, they demonstrate safety and values compatible with other traditional PWRs and SMRs according to the reference[14][15][16].

4.2 Thermal-Hydraulic analysis

In this paper, we present a simplified thermal-hydraulic analysis of the standard core of the NuScale project. This analysis aims to determine the thermal and hydraulic limits of the project with a focus on the fuel rods and the coolant. The power generation term in Ansys CFX was assumed based on the sinusoidal approach defined in Todreas and Kazimi [17], with a maximum linear power of 13.87 kW/m calculated by SERPENT.

With the CFD model, we found 1104.25K and 648.37 K for the maximum temperature reached in the regions of the fuel pellet and its cladding. These temperature parameters are important engineering features to ensure that the fuel material and the core do not melt. Despite the lower heat transfer capacity of the M5 material used in the fuel rod cladding compared to Zircaloy-4, the standard core of the NuScale design demonstrated temperature levels within the safety limits of the materials. Fig. 5 displays the axial temperature distribution of the heated channel.

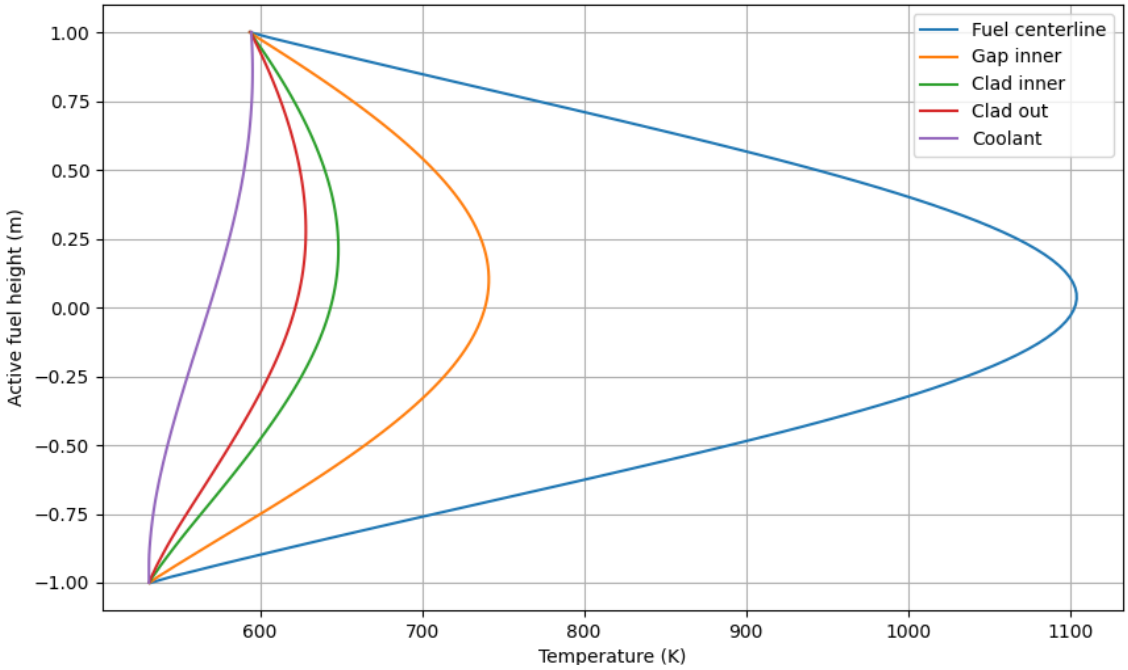

Fig. 5. NuScale temperature profiles for normal operating conditions.

The coolant flow channel also provides critical parameters for the safety analyses of nuclear reactor operation, as it evaluates the coolant heat removal capacity of the fuel rods. Tab. 6 shows the main results of the thermal-hydraulic model developed in CFD for the heated channel.

Tab. 6. CFD results for the NuScale standard core.

Design Parameters	**Value**
Fuel rod	
Peak linear power for normal operating conditions	13.87 kW/m
Maximum local heat flux for normal operations	544.17 kW/m²
Peak fuel centerline pellet temperature	1104.25 K
Maximum cladding temperature	648.37 K
Coolant	
Nominal system pressure	127.55 bar
Pressure drop	0.15492 bar
Core effective mass flow rate for heat transfer	587.15 kg/s
Core average coolant velocity	0.8741 m/s

Core inlet temperature	531.48 K
Average outlet coolant temperature	589.64 K

We compute 0.15492 bar for the pressure drop along the fluid flow channel, assuming only friction and gravity losses. We compared this result with the one-dimensional formulation available in Todreas and Kazimi [17], which showed a value of 0.14781 bar for the same operating conditions. The CFD model presented a 0.8741 m/s average coolant flow velocity along the active reactor core height, compared to 0.8230 m/s reported by the NuScale final safety report [11]. The same document gives 587.04 K for the average outlet core temperature against 589.64 K from the CFD model. Tab. 7 presents the highest fuel pellet temperature values for the proposed CFD model and those reported by the NuScale.

Tab. 7. Peak fuel pellet temperature for the proposed CFD model and the NuScale project approved by the NRC [11].

Normal Operating Condition	**Linear Power (kW/m)**	**Peak Fuel Pellet Temperature (K)**
NRC - Average linear power density	8.202	1019.26
CFD model	13.8737	1104.2
NRC - Peak linear power density	16.4042	-
NRC - Hot rod linear power density	21.3255	1408.15

Although NuScale's final safety analysis report lacks some thermal-hydraulic parameters, the current CFD model, even with many simplifications, presents consistent results for the reactor's normal operation condition

Conclusion

After extensive research about the data to analyze the reactor and using some information presented in the references, we conclude that the results presented in the benchmark work from [5] became valid with diference less than 100 pcm.

We have developed a three-dimensional CFD model of a typical PWR subchannel for the NuScale project to provide a simplified nuclear reactor thermal hydraulics analysis. We employed the methodology developed by Da Cunha et al. (2024) to set up the single heated channel analysis.

We found 1104.2 K and 648.37 K as the maximum temperature reached in the fuel pellet and cladding, respectively. Those values are within the reactor's normal operating limits and materials. As with the fuel rod, we estimated the fluid flow parameters. We obtained an average coolant outlet temperature of 589.64 K and an average coolant velocity of 0.8741 m/s in the reactor core. These values are slightly higher than those reported by NuScale and justified by the assumptions made in the CFD model proposed in this paper.

References

[1]https://www.eia.gov/todayinenergy/detail.php?id=61963#:~:text=Originally%20expected%20to%20cost%20%2414,be%20more%20than%20%2430%20billion. Acess in 10/21/2024.
[2]https://www.dw.com/en/finlands-much-delayed-nuclear-plant-launches/a-61108015#:~:text=The%20cost%20ballooned%20from%20an,2019%20World%20Nuclear%20Industry%20Report Acess in 10/21/2024.
[3]https://www.nucnet.org/news/fuel-loading-begins-at-flamanville-3-epr-nuclear-plant-in-normandy-5-4-2024. Acess in 10/21/2024.
[4] Tsolfanidis, N. The Nuclear Fuel Cycle. 3th edition, American Nuclear Society. 2013.

[5] https://rodare.hzdr.de/record/2025. Acess in 10/21/2024.

[6] E. Fridman, Definition of the neutronics benchmark of the NuScale-like core. *Nuclear Engineering and Technology*, Vol. 55, pp. 3639-3647 (2022).

[7] J. Leppanen, M. Pusa, T. Viitanen, V. Valtavirta, T. Kaltiaisenaho The Serpent Monte Carlo code: status, development and applications in 2013. *Annals of Nuclear Energy*, Vol. 82, pp. 142-150 (2015).

[8] http://montecarlo.vtt.fi. Acess in 10/21/2024.

[9] L. Kerby, A. Tumulak, J. Leppänen, V. Valtavirtaz. Preliminary serpent–MOOSE coupling and implementation of functional expansion tallies in serpente. International Conference on Mathematics & Computational Methods Applied to Nuclear Science & Engineering, Jeju, Korea, April 16-20 (2017).

[10] Da Cunha et al., Single heated channel analysis of the AP-Th 1000 concept. *Nuclear Engineering and Design*, Vol. 420, pp. 113033 (2024).

[11] LLC NuScale Power. Part 02 - Final Safety Analysis Report (Rev. 5) - Part 02 -Tier 02 - Chapter 04 - Reactor - Sections 04.01 - 04.06 (2020).

[12] Westinghouse, E.C, Westinghouse AP1000 Design Control Document Rev. 19. U.S. Nuclear Regulatory Commission, ML11171A500 (2011).

[13] KECEK, A.; TUCEK, K.; HOLMSTROM, S.; VAN UFFELEN, P. Development of M5 Cladding Material Correlation in the TRANSURANUS Code: Revision 1, Publication Office of the European Union, Luxembourg, (2016).

[14] AKBARI-JEYHOUNI, REZA ; REZAEI OCHBELAGH, DARIUSH ; MAIORINO, JOSE R. ; D'AURIA, FRANCESCO ; STEFANI, GIOVANNI LARANJO DE . The utilization of thorium in Small Modular Reactors - Part I: Neutronic assessment. *ANNALS OF NUCLEAR ENERGY*, Vol. 120, pp. 422-430, 2018.

[15] LARANJO DE STEFANI, GIOVANNI; LOSADA MOREIRA, JOÃO MANOEL ; MAIORINO, JOSÉ RUBENS ; RUSSO ROSSI, PEDRO CARLOS . Detailed neutronic calculations of the AP1000 reactor core with the Serpent code. *PROGRESS IN NUCLEAR ENERGY*, Vol. 116, pp. 95-107(2019).

[16] SILVA, M. V. ; SCHIRRU, R. ; STEFANI, GIOVANNI LARANJO ; NICOLAU, A. S. ; LIMA, A. M. M. ; PEREIRA, C. M. ; GUIMARAES, J. V. S. A. ; GONCALVES, D. M. E. . Optimized Modular Nuclear Reactor Project Utilizing Artificial Intelligence: Seed-Blanket Concept. *NUCLEAR ENGINEERING AND DESIGN*, Vol. 423, pp. 113187(2024).

[17] TODREAS N. E.; KAZIMI M. S. Nuclear systems I – Thermal hydraulic fundamentals. 1ª ed. New York: Taylor & Francis Group, (1990).