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STUDY ON CRITICALITY AND NEUTRONIC SAFETY PARAMETERS OF NUSCALE FUEL ASSEMBLY

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ABSTRACT

STUDY ON CRITICALITY AND NEUTRONIC SAFETY PARAMETERS OF NUSCALE FUEL ASSEMBLY. NuScale, a typical Pressurized Water Reactor (PWR) Small Modular Reactors (SMRs), offers a new opportunity for the future of nuclear industry. With 160 MW thermal power, NuScale has several advantages such as flexibility due to its modularity in construction. This work is focused on the study of criticality and neutronic safety parameters of NuScale fuel assembly using MCNP6 code and ENDF/B-VII library. The calculation results shows that criticality of fuel assembly type D is the highest among other assembly types because it has a fuel pin with pure UO_2 without Gd_2O_3 concentration. The Doppler temperature coefficient (DTC) of fuel assembly type C is the most negative among other assemblies due to Doppler broadening effect on resonance region of capture cross section of ^{238}U which is the highest concentration. The moderator temperature coefficient (MTC) of fuel assembly type D is the most negative among the other assembly types. The effective delayed neutron fraction (β_{eff}) does not reflect a consistent trend among fuel assembly types. Fuel assembly type D shows the highest prompt neutron lifetime (ℓ) while the highest neutron generation time (Λ) is shown in assembly type C. It can be concluded that this study provides adequate results that can be used as a first step to carry out the neutronic computation and analysis of the NuScale full core.

Keywords: Criticality, safety parameters, NuScale fuel assembly, MCNP6, ENDF/B-VII

ABSTRAK

STUDI KRITIKALITAS DAN PARAMETER KESELAMATAN NEUTRONIK PERANGKAT BAHAN BAKAR NUSCALE. *NuScale, reaktor modular kecil PWR tipikal membuka peluang baru untuk masa depan industri nuklir. Dengan daya termal 160 MW, NuScale memiliki beberapa kelebihan seperti fleksibilitas karena modularitasnya dalam konstruksi. Riset ini difokuskan pada studi kritikalitas dan parameter keselamatan neutronik perangkat bahan bakar NuScale menggunakan program MCNP6 dan pustaka ENDF/B-VII. Hasil perhitungan menunjukkan bahwa kritikalitas perangkat bahan bakar tipe D adalah yang paling tinggi diantara jenis perangkat lainnya karena memiliki pin bahan bakar dengan* UO_2 *murni tanpa konsentrasi* Gd_2O_3*. Koefisien temperatur Doppler (DTC) perangkat bahan bakar tipe C paling negatif diantara perangkat lainnya karena efek pelebaran Doppler pada daerah resonansi dari tampang lintang tangkapan* ^{238}U *yang merupakan konsentrasi tertinggi. Koefisien temperatur moderator (MTC) perangkat bahan bakar tipe D paling negatif diantara tipe perangkat lainnya. Fraksi neutron tunda efektif (*β_{eff}*) tidak mencerminkan kecenderungan yang konsisten di antara jenis perangkat bahan bakar. Perangkat bahan bakar tipe D menunjukkan waktu hidup neutron serempak (*ℓ*) tertinggi sedangkan waktu generasi neutron (*Λ*) tertinggi ditunjukkan dalam perangkat tipe C. Dapat disimpulkan bahwa studi ini memberikan hasil perhitungan cukup memadai yang dapat digunakan sebagai langkah pertama. untuk melakukan komputasi dan analisis neutronik teras penuh NuScale.*

Kata kunci: *kritikalitas, parameter keselamatan, perangkat bahan bakar NuScale, MCNP6, ENDF/B-VII*

INTRODUCTION

In the last decade, development of Small Modular Reactors (SMRs) have gain an interest on nuclear industry because it offers the various advantages, such as modularity, lower capital investment, flexibility, etc. Many nuclear reactor physicists carry out reviews and studies on neutronic aspects of several SMRs core. Many of SMRs design concepts are based on Light Water Reactor (LWR) technology. Numerous designs are being promoted by nuclear industry companies, such as NuScale, AREVA, Babcock & Wilcox (mPower), General Atomics, and Westinghouse (IRIS) [1]. Some other designs are being developed by national research institutes, in examples Argentina, China, Japan, Korea, and Russia[2],[3]. SMRs design concept makes it possible for several remote location or some locations that are not suitable for large units to utilize nuclear energy, and some designs can also be used for non-electric applications, like hydrogen production [4]. It is hoped that SMR can provide an overall cost per unit of electricity that can compete with large Nuclear Power Plants (NPP), and could be a key to meet the growing demand for nuclear energy in coming decades.

Among light water cooled SMR designs, NuScale has opens up a new opportunity for the future of nuclear industry. NuScale is based on PWR technology without using primary coolant pump, so natural circulation is used as primary heat transfers. With a thermal power of 160 MW per module, make NuScale has its modularity and flexibility to increase power, up to 12 modules in on facility, and advantages due to small footprint[5],[6]. Financial budget for SMR is also lower than large NPP because it eliminates uses of pumps and several pipelines [7],[8] as well as simpler manufacture and transport of reactor component. In addition, there are reduced probability for some accidents such as Loss of Coolant Accident (LOCA) or Loss of Flow Accident (LOFA) which based on primary coolant pump and pipeline failure [9]-[11].

This work is focused on the study of criticality and neutronic safety parameters of NuScale fuel assembly with a 17 × 17 size consisting of 264 fuel rods, 24 guide tubes and one instrumentation tube. Four different fuel assembly types in the UO_2 and UO_2+Gd_2O_3 fuel rod configurations were investigated. A series of calculations were performed using the Monte Carlo transport code MCNP6 [12] and the ENDF/B-VII continuous energy nuclear data library [13]. A number of criticality and neutronic safety parameters were calculated and analyzed including temperature coefficient of reactivity related to Dopper broadening (DTC), moderator temperature coefficient of reactivity (MTC), and kinetic parameters. The results of these calculations are expected to be use as initial study before performing overall calculation and analysis on neutronic behavior of NuScale reactor core.

METHODOLOGY

a. Description of NuScale

NuScale is a typical PWR Small Modular Reactor with a thermal power of 160 MW that contains a reactor core, pressurizer and steam generators integrated in a reactor pressure vessel (RPV) and placed inside a compact steel containment. NuScale reactor design is illustrated in Figure 1 and design parameters are presented in Table 1. NuScale core configuration consists of 37 fuel assemblies (FA) and 16 control rod assemblies (CRA). The core is surrounded by a stainless steel heavy neutron reflector which improves fuel utilization by preventing the escape of neutrons radially from the core. The reflector also provides the core envelope and directs the flow through the core. NuScale reactor core design parameter is presented in Table 2.

The NuFuel HTP2™ is fuel that used in NuScale core, and its design features are similar to those used in the PWR fuel assembly. The fuel assembly is arranged in a 17 × 17 square lattice fuel assembly (FA) with 21.4 cm width and 200 cm active height. This shorter height is the only significant difference between the NuScale and other PWR fuel designs. Fuel assembly is supported by five spacer grids, 24 guide tubes, top and bottom sides of a nozzles which together provide a structural framework for those 264 fuel rods. Each fuel assembly has a central instrumentation tube. With total 37 fuel assemblies in NuScale core, 25 FAs consist of homogeneous fuel mixture of UO_2 and Gd_2O_3 as burnable absorber while remaining 12 FAs uses UO_2 fuel only. Fuel design parameter is presented in Table 3 and its axial separation is illustrated in Figure 2.

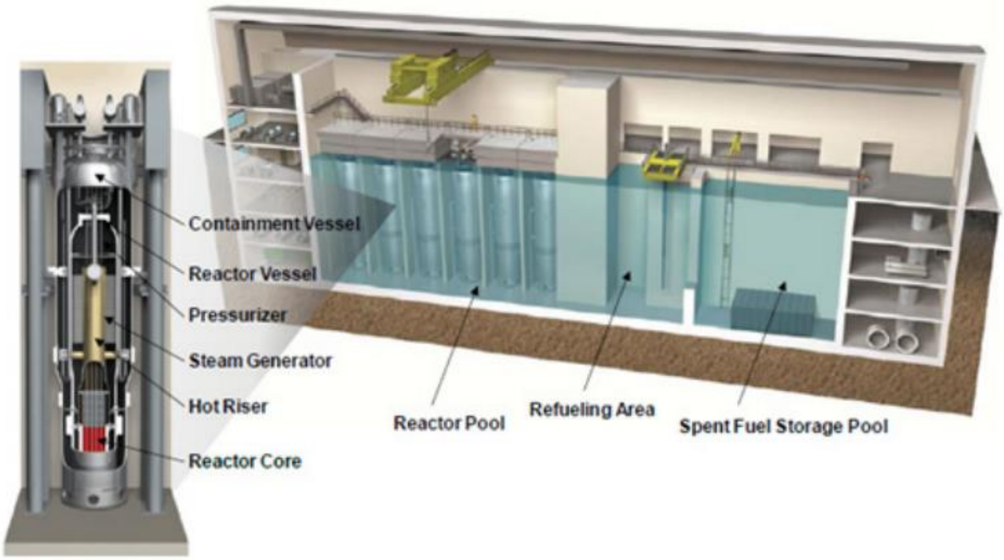

Figure 1. NuScale reactor design [6].

Table 1. NuScale reactor design parameter [6].

Key reactor parameter	Value
Core thermal output (MWt)	160
System pressure (psia)	1850
Inlet temperature (^oF)	497
Core average temperature (^oF)	543
Average temperature rise in core (^oF)	100
Best estimate flow (kg/hr)	2.11E+06
Core bypass flow (%) (best estimate)	7.3
Average linear power density (kW/m)	8.202
Peak linear power for normal operating conditions (kW/m)	16.404
Normal operation peak heat flux (kW/m^2)	536.545
Total heat flux hot channel factor, F_Q	2.0
Heat transfer area on fuel surface (m^2)	583.022
Normal operation core average heat flux (kW/m^2)	268.272
Core flow area (m^2)	0.9095
Core average coolant velocity (m/sec)	0.823

Tabel 2. NuScale core design parameter [6].

Parameter	Value
Core	
Diameter of active core (m)	1.506
Number of fuel assemblies	37
Height-to-diameter ratio of active core	1.33
Total cross section area of active core (m^2)	1.711
Core barrel ID/OD (m)	1.8796/1.9812
Reflector	
Height (m)	2.33045
Width (m)	0.0635 to 0.30988

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Table 3. NuScale fuel design parameter [6].

Fuel Assembly	
Fuel design	NuFuel HTP2™
Length (m)	2.436
Nominal UO_2 per assembly (kg)	249.24
Rods per fuel assembly	264
Fuel assembly pitch (cm)	21.504
Fuel rod pitch (cm)	1.259
Number of grids per assembly	5
Span of grids (cm)	51.054
Number of guide tubes per assembly	24
Number of instrument tubes per assembly	1
Guide tube dashpot region ID (cm)	1.00838
Guide tube dashpot region OD (cm)	1.22428
Guide tube above dashpot ID (cm)	1.143
Fuel Rod	
Peak rod exposure core design criteria for UO_2 rods (GWd/MTU)	62
Gd_2O_3 concentration	≤ 8%
Cladding outside diameter (cm)	0.94996
Cladding inside diameter (cm)	0.82804
Cladding thickness (cm)	0.06096
Fuel rod-cladding diametral gap (cm)	0.01651
Cladding material	M5®
Fuel column length (cm)	199.9996
Overall fuel rod length (cm)	215.9
Fuel rod material	UO_2
Fuel rod diameter (cm)	0.81153
Fuel rod density (g/cm^3) (96 % theoretical density)	10.53
Fuel rod length (cm)	1.016
Fissile enrichment	< 4.95%

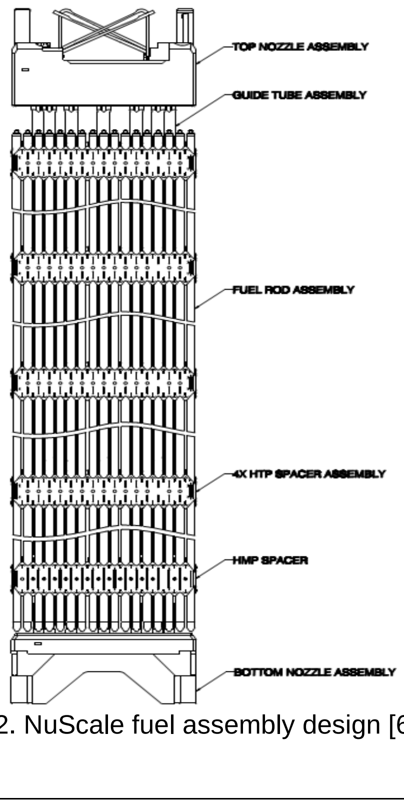


Figure 2. NuScale fuel assembly design [6].

b. Calculation model

In this experiment, a series of calculations using MCNP6 code with ENDF/B-VII library have been carried out to study the criticality and neutronic parameters of NuScale fuel assembly. MCNP6 is a general-purpose Monte Carlo transport code developed by Los Alamos National Laboratory (LANL) which has ability to track several types of particles over a wide energy range in a modeled geometry. The advantages of MCNP6 in simulating 3-D fuel assembly and reactor core configurations with geometrical complexity are well known. MCNP6 has successfully demonstrated its capability to analyze neutronic behavior and also fuel depletion for various types of reactor [14]-[25].

The first step of MCNP6 calculations is to model fuel pin, guide tube and instrumentation tube in a cubic or square lattice. Fuel pin cell consists of a 0.4060 cm radius fuel rod surrounded by helium and zircalloy-4 cladding which have thicknesses of 0.0082 cm and 0.0609 cm, respectively. Water with boron concentration of 1184 ppm occupies the region outside the fuel cell in the lattice. MCNP6 model for fuel pin cell is shown in Figure 3 and its material composition was given in Table 4.

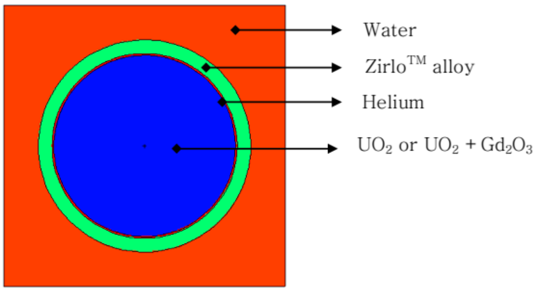


Figure 3. MCNP6 model for NuScale fuel pin cell.

Table 4.Composition of NuScale fuel pin cell.

No	Material	Radius (cm)
1	UO_2 or UO_2+Gd_2O_3	0.4060
2	Helium	0.4142
3	Zirlo^{TM} alloy	0.4751
4	Water	1.2590^a

^{a} pitch of cubic lattice

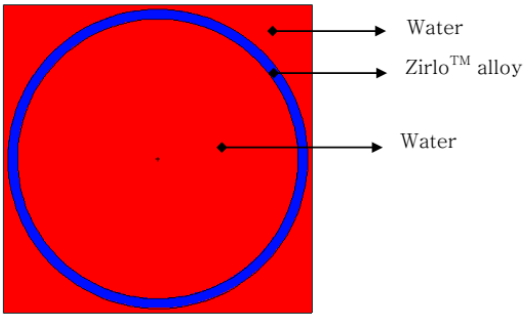


Figure 4. MCNP6 model for NuScale guide and instrumentation tubes.

Table 5. Composition of NuScale guide and instrumentation tubes.

No	Material	Radius (cm)
1	Water	0.5715
2	Zirlo^{TM} alloy	0.6120
3	Water	1.2590^a

^{a} pitch of cubic lattice

Guide tubes and instrumentation tubes have identical geometries, despite each has different functions. A guide tube is a part of the structure designed to provides a channel for neutron absorber rods (control rods), burnable absorbers rods, and neutron source rods. An instrumentation tube is designed to provides a place for neutron detectors or in-core instrumentation and other measuring installation. Those two tubes were modeled in a similar technique to the fuel pin cell model, but only replacing fuel rod with water. Water inside guide tubes and instrumentation tubes is used as an additional neutron moderation. Zircaloy-4 cladding radius is slightly bigger than those in fuel pin. All square lattices have the same pitch of 1.2590 cm. MCNP6 model for NuScale guide tubes and instrumentation tubes is shown in Figure 4 and its composition is given in Table 5.

Each NuScale fuel pin cell has 200 cm height but it has different number of axial zones as illustrated in Figure 5. Fuel assembly with burnable absorber (UO_2+Gd_2O_3) consist of 4 axial zone, besides, there are only 3 zones in the axial direction, see Table 6. UO_2 fueled pin has a 4.33% ^{235}U enrichment on the middle zone, while the middle zone of UO_2+Gd_2O_3 fueled pin cell has different ^{235}U enrichment, 4.32%, 4.30%, and 4.29% with several Gd_2O_3 concentration, 2%, 6%, and 8%, respectively.

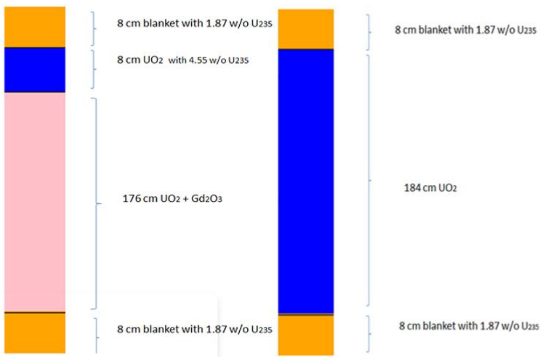


Figure 5. Axial zone of NuScale fuel pin cell [26].

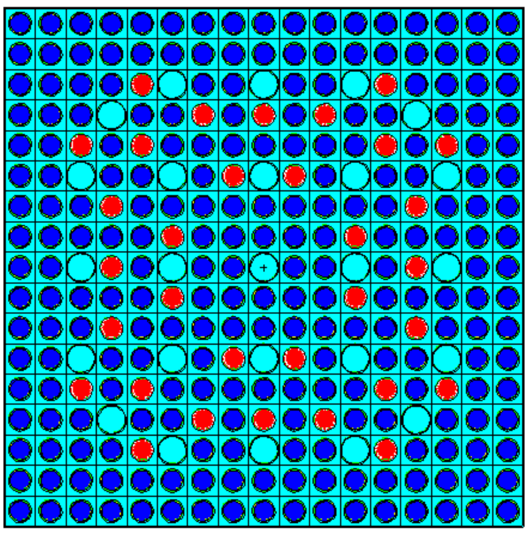
The next step is to model the NuScale fuel assembly into MCNP by constructing a 17×17 square lattice consisting of 264 fuel pin, 24 guide tube and 1 instrumentation tube cells. Four different fuel assembly types consist of UO_2 and UO_2+Gd_2O_3 fuel rod configurations are summarized in Table 7 and its MCNP6 model can be seen in Figure 6. In the calculation, each axial zone with its corresponding fuel composition, burnable poison, and zone height for each type of UO_2 and UO_2+Gd_2O_3 fuel assembly are modeled with all sides of the fuel assembly geometry were set as a reflective surface.

Table 6. Axial zone of NuScale fuel pin cell.

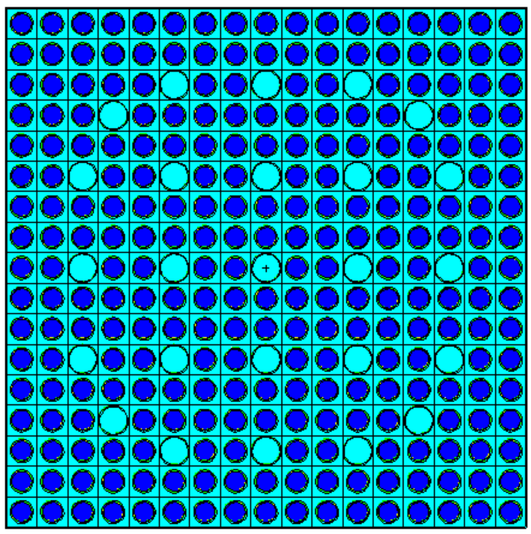
	UO_2 fuel pin cell		UO_2+Gd_2O_3 fuel pin cell	
Zone 1	Upper blanket	UO_2 fuel with ^{235}U enrichment of 1.87% and 8 cm height	Upper blanket	UO_2 fuel with ^{235}U enrichment of 1.87% and 8 cm height
Zone 2	Middle fuel	UO_2 fuel with 184 cm height	Upper fuel	UO_2 fuel with ^{235}U enrichment of 4.55% and 8 cm height
Zone 3	Lower blanket	UO_2 fuel with ^{235}U enrichment of 1.87% and 8 cm height	Middle fuel	UO_2+Gd_2O_3 fuel and 176 cm height
Zone 4	-	-	Lower blanket	UO_2 fuel with ^{235}U enrichment of 1.87% and 8 cm height

Table 7. Four NuScale fuel assembly types.

Type		No. of fuel pin cell per axial zone
A	Zone 1	264 UO_2 fuel with ^{235}U enrichment of 1.87%
	Zone 2	264 UO_2 fuel with ^{235}U enrichment of 4.55%
	Zone 3	232 UO_2 with ^{235}U enrichment of 4.55%, and 32 UO_2+Gd_2O_3 fuel with ^{235}U enrichment of 4.32% and Gd_2O_3 concentration of 2%
	Zone 4	264 UO_2 fuel with ^{235}U enrichment of 1.87%
	Zone 2	264 UO_2 fuel with ^{235}U enrichment of 4.55%
B	Zone 1	264 UO_2 fuel with ^{235}U enrichment of 1.87%
	Zone 2	264 UO_2 fuel with ^{235}U enrichment of 4.55%
	Zone 3	232 UO_2 fuel with ^{235}U enrichment of 4.55%, and 32 UO_2+Gd_2O_3 fuel with ^{235}U enrichment of 4.30% and Gd_2O_3 concentration of 6%
	Zone 4	264 UO_2 fuel with ^{235}U enrichment of 1.87%
C	Zone 1	264 UO_2 fuel with ^{235}U enrichment of 1.87%
	Zone 2	264 UO_2 fuel with ^{235}U enrichment of 4.55%
	Zone 3	232 UO_2 fuel with ^{235}U enrichment of 4.55%, and 32 UO_2+Gd_2O_3 fuel with ^{235}U enrichment of 4.29% and Gd_2O_3 concentration of 8%
	Zone 4	264 UO_2 fuel with ^{235}U enrichment of 1.87%
D	Zone 1	264 UO_2 fuel with ^{235}U enrichment of 1.87%
	Zone 2	264 UO_2 fuel with ^{235}U enrichment of 4.33%
	Zone 3	264 UO_2 fuel with ^{235}U enrichment of 1.87%



Type A/B/C (232 UO_2, 32 UO_2+Gd_2O_3) of Zone3



Type D (264 UO_2) of Zone2

Figure 6. MCNP6 model for NuScale fuel assembly.

RESULTS AND DISCUSSION

The calculation of criticality and neutronic safety parameters of NuScale fuel assembly was done using MCNP6 code and ENDF/B-VII library. KCODE as one option in MCNP6 was used to simulate 2.5 million neutrons histories obtained from 10,000 neutron neutrons per cycle, 50 skipped cycles, and 250 active cycles. KSRC as source definition option in MCNP6 was used to place the initial fission source at fuel cells. Standard deviation of criticality calculation was below 0.00070 with this configuration. S(α, β) library is also utilized to model the thermal scattering for hydrogen in light water.

The calculation results of criticality and temperature coefficient of reactivity are summarized in Table 8. Criticality is the condition in a nuclear reactor when the fissionable material can sustain a chain reaction by itself. It depends on the composition, size of assembly and also the arrangement of fuel materials within the assembly. In this calculation, the neutron leakage effect from the systems was assumed to be ignored, that's called k-infinity. The fuel temperature was modeled at 900 K, helium and Zirlo-4 cladding at 622 K and 565 K for water moderator on criticality calculation. From Table 8, it can be observed that the k-infinity of the fuel assembly type D is the highest among other types. It's because type D fuel assembly has a pin cell consist of pure UO_2 without any concentration of Gd_2O_3. The k-infinity of type A fuel assembly is greater than type B, and the smallest is type C fuel assembly. This is because in axial zone number 3, the assembly type A has a greater enrichment of ^{235}U (4.32%) and a smaller concentration of Gd_2O_3 (2%) than that of type B (4.30% ^{235}U, 6% Gd_2O_3) and type C (4.29% ^{235}U, 8% Gd_2O_3).

Temperature coefficient of reactivity is the amount of change reactivity when there are some changes on temperature. Two most dominant temperature coefficients are fuel temperature coefficient, better known as Doppler Temperature Coefficient (DTC), and Moderator Temperature Coefficient (MTC). In thermal reactors, Doppler broadening effect is primarily due to neutron capture in resonances region close to epithermal neutron spectrum for non-fissionable fuel isotopes, in this case ^{238}U. DTC is a very strong contributor for safety and stability of nuclear reactors during operation. MTC is primarily a function of moderator to fuel ratio that changes fuel assembly reactivity during moderator temperature changes.

The DTC reactivity was calculated by changing fuel temperature from 565 K to 900 K, preserving helium and cladding temperatures constant at 622 K, and water temperature constant at 565 K. The temperatures of 565 K and 622 K were modeled with provided nuclear data at a temperature of 600 K on material data card due to a limited number of the MCNP6 cross-section data library. However, this approach was corrected by adding a TMP card for interpolation at the actual temperature on each corresponding cell of cladding, helium, and coolant. Similarly, the MTC reactivity was simulated by varying moderator temperature from 565 K to 622 K and keeping helium and cladding temperatures constant at 622 K, and fuel temperature constant at 900 K.

Table 8 confirms that the DTC of the fuel assembly type C is the most negative among the other assembly types due to Doppler broadening effect on capture cross section of ^{238}U isotope which is the highest composition among all fuel assemblies. Table 8 also confirms that all types of fuel assembly had a negative value on MTC which means the reactor is under moderated, with fuel assembly type D is the most negative among other assembly. Negative value in MTC is desirable criteria because of its self-regulating effect on reactor operation.

Table 8. Criticality (k-infinity) and temperature coefficient of reactivity (Δk/k/K) of NuScale fuel assembly.

	Type A	Type B	Type C	Type D
K-infinity	1.05759±0.00054	1.03212±0.00056	1.02478±0.00061	1.23776±0.00044
Doppler temperature coefficient (DTC)	-2.21101×10^{-5}	-2.10304×10^{-5}	-2.48007×10^{-5}	-1.88870×10^{-5}
Moderator temperature coefficient (MTC)	-1.32228×10^{-5}	-3.25345×10^{-5}	-8.19256×10^{-6}	-6.13324×10^{-5}

The calculation results of kinetic parameters are summarized in Table 9. The principal kinetics parameters of nuclear reactor are the effective delayed neutron fraction (β_{eff}), the prompt neutron lifetime (ℓ), the mean neutron generation time (Λ). Even though delayed neutrons constitute only a small fraction (<1%) of the total number of neutrons produced by fission, they play a dominant role in the control of fission chain reactions. If only the prompt neutrons existed, reactor operation becomes impossible due to the rapid changes in reactor power. Analysis of nuclear reactor control and accidents and conversion of reactor period to reactivity requires knowledge of the effective delayed neutron parameters and their decay constants. In a nuclear reactor chain, many fission products can be considered as potentially delayed neutron emitters.

Table 9. Kinetic parameters of NuScale fuel assembly.

	Type A	Type B	Type C	Type D
Effective delayed neutron fraction (β_{eff})	0.00601±0.00058	0.00626±0.00066	0.00571±0.00062	0.00615±0.00063
Prompt neutron lifetime (ℓ, sec)	1.3480×10^{-5} $\pm2.1409\times10^{-8}$	1.3746×10^{-5} $\pm2.3700\times10^{-8}$	1.3776×10^{-5} $\pm2.3502\times10^{-8}$	1.5099×10^{-5} $\pm1.3486\times10^{-8}$
Neutron generation time (Λ, sec)	1.5506×10^{-5} $\pm1.4874\times10^{-7}$	1.6892×10^{-5} $\pm1.6983\times10^{-7}$	1.7164×10^{-5} $\pm1.7320\times10^{-7}$	1.3448×10^{-5} $\pm1.2044\times10^{-7}$

The second important kinetic parameter that characterizes the timing behavior of the neutron population is the neutron generation time (Λ), which is defined as the average generation time between neutron birth and subsequent absorption inducing fission. If k~1, then Λ is essentially just the prompt neutron lifetime (ℓ). Neutron generation time depends on several parameters such as fuel enrichment, neutron fission cross-section, prompt neutron distribution function, the average number of neutrons released per fission, neutron flux, and adjoint flux. Prompt neutron lifetime (ℓ) is defined as the average time from a prompt neutron emission to either its absorption (fission or radiative capture) or its escape from the system. It depends on material composition, geometric configuration, and size of the system.

In this calculation, the KOPTS option in MCNP6 was activated. As criticality calculation, the fuel temperature was modeled at 900 K, helium and Zirlo-4 cladding temperatures at 622 K and temperature of water moderator at 565 K. From Table 9, it can be observed that the effective delayed neutron fraction (β_{eff}) does not reflect a consistent trend between assembly types A, B, C or D. On the other hand, assembly type D shows the highest prompt neutron lifetime (ℓ) and it is related to the highest criticality among all fuel assembly types. The highest neutron generation time (Λ) is shown in the assembly type C. The low kinetic parameters make it difficult to control reactor safety.

CONCLUSION

Study on criticality and neutronic safety parameters of NuScale fuel assembly has been done using MCNP6 code and ENDF/B-VII library. The calculation results show that the criticality of type D fuel assembly is the highest among other types of assembly. The Doppler temperature coefficient (DTC) of the fuel assembly type C is the most negative among other types of assembly Moderator temperature coefficient (MTC) of fuel assembly type D is the most negative among other types of assembly. The effective delayed neutron fraction (β_{eff}) does not show any significant difference. The assembly type D shows the highest prompt neutron lifetime (ℓ) while the highest neutron generation time (Λ) is shown on the assembly type C. It can be concluded that the study provides adequate calculation results that can be used as a preliminary study to carry out neutronic computation and analysis of the NuScale full core.

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