Whole-core Monte Carlo Analysis of MOX-3600 Core in NEA-SFR Benchmark Using MCS Code

Mai Nguyen Trong Nhan, Yunki Jo, Hyunsuk Lee, Alexey Cherezov, Deokjung Lee *
Department of Nuclear Engineering, Ulsan National Institute of Science and Technology, 50 UNIST-gil, Ulsan, 44919, Republic of Korea
**Corresponding deokjung@unist.ac.kr*

1. Introduction

In order to evaluate core performance characteristics of several Gen-IV Sodium-cooled Fast Reactor (SFR) concepts and enable different participants to compare their calculation results, a set of benchmark cases were introduced in The Benchmark for Neutronic Analysis of Sodium-cooled Fast Reactor Cores with Various Fuel Types and Core Sizes provided by Nuclear Energy Agency (NEA) [1]. Among several core concepts in this benchmark, the large size oxide core MOX-3600 is the most concerned due to inconsistencies between results from participating institutes. Hence, in this study, neutronic characterization of global parameters of the MOX-3600 including multiplication factor, sodium void worth, control rod worth and Doppler constant, are calculated together with the uncertainty analysis, using the Monte Carlo code named MCS developed in Ulsan National Institute of Science and Technology (UNIST).

2. Methods and Results

2.1 MOX-3600 benchmark specification

The axial fuel pin design in MOX-3600 is based on 1-meter active zone surrounded by two gas plenums and two axial reflectors. By using the fat $(U,Pu)O_2$ fuel pellets design, the MOX-3600 core enables self-breeding without fertile blanket [1]. The core is divided into the inner core and the outer core with different enrichment levels and axial distributions. The core layout is presented in Figure 1.

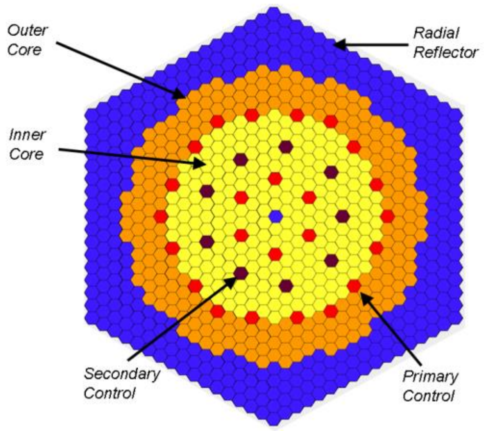

Figure 1. Core layout of the MOX-3600.

The main characteristics of the MOX-3600 core are summarized in Table I.

Table I: The main core characteristics the MOX-3600

Thermal power	3600 MW_{th}
Fuel	$(U,Pu)O_2$
Cladding/Duct material	ODS/EM10 steel
Number of assemblies: +Inner core +Outer core	 225 228
Coolant	Sodium
Number of control rod +Primary system +Secondary system	 24 (B_4C w. natural B) 9 (B_4C w. enriched ^{10}B)
Operating temperature +Fuel +Average structure temperature	 1500K 743K
Lattice pitch	21.2205 cm

2.2 MCS

MCS is a 3D continuous-energy neutron-physics code for particle transport based on the Monte Carlo method, under development at UNIST since 2013 [2]. Two kinds of calculations are allowed by MCS: criticality runs for reactivity calculations and fixed-source runs for shielding problems. MCS neutron transport capability is verified and validated with many benchmark problems including BEAVERS benchmarks, the International Criticality Safety Benchmark Experimental Problem (ICSBEP) and Jordan Research and Training Reactor (JRTR). [3,4]

2.3 Simulation and results

The MOX-3600 is modeled in MCS, using ENDF/B-VII.1 library. Each criticality simulation is run with 100 inactive cycles, 400 active cycles and 50,000 histories per cycle. The MCS simulations were executed on a Linux cluster (Intel Xeon E5-2620 @ 3.00 GHz) and cost 43.8 core-hours and 800MB of memory for criticality. The uncertainty analysis required the same amount of time but higher memory, 7.8GB.

The Doppler constant is calculated using Equation 1 [5]:

$$K_D = \frac{\frac{1}{k_{T_1}} - \frac{1}{k_{T_2}}}{\ln\left(\frac{T_2}{T_1}\right)}, \quad (1)$$

where K_D is the Doppler constant, k_{T_1} and k_{T_2} are the multiplication factors while T_1 and T_2 are fuel temperatures and are chosen at 1,500K and 1,200K, respectively.

The control rod worth is the reactivity difference between the two states: when all control rods are withdrawed from the core during normal operation and when all control rods are fully inserted. The sodium void worth is calculated by replacing all sodium in the active core by void and is equal to the reactivity difference between this state and normal operation state.
The delayed neutron fraction and uncertainty values are provided in the output of MCS.

Results for the MOX-3600 from Helmholtz Zentrum Dresden Rossendorf (HZDR), Commissariat à l'énergie atomique et aux énergies alternatives (CEA), Centre for Energy Research (CER), Energy and Sustainable Economic Development (ENEA) are used for comparison purposes [6]. The results are presented in Table II.

Table II: Results for the MOX-3600

	UNIST	Diff. vs. HZDR	Diff. vs. CEA	Diff. vs. CER	Diff. vs. ENEA
Code	MCS	SERPENT	TRIPOLI4	SERPENT	MCNPX
Library	ENDF/B-VII.1	Same as MCS	Same as MCS	Same as MCS	ENDF/B-VII.0
k_{eff}	1.02804±0.00010	-1,464±10 pcm	-954±10 pcm	86±10 pcm	-1,724±10 pcm
β_{eff} (pcm)	375±19 pcm	-3.7%±4.9%	-3.7%±4.9%	-7.2%±4.7%	-6.1%±4.8%
Doppler constant	-938±4 pcm	-15.3%±5.3%	-4.7%±5.9%	-17.4%±5.1%	-5.7%±5.9%
Control rod worth	5,706±13 pcm	10.7%±0.3%	7.5%±0.2%	-2.6%±0.2%	-3.1%±0.2%
Sodium void worth	1,806±14 pcm	4.6%±0.8%	1.6%±0.8%	-0.9%±0.7%	11.4%±0.8%

Results for the multiplication factor exhibit apparent large discrepancies, while results for the delayed neutron fraction, sodium void worth and control rod worth appear to be relatively consistent. Among the four results used for comparison, HZDR's results show the most difference while relatively good agreement is observed between UNIST and CER except for the Doppler constant (relative difference is higher than 10%).

The core is re-simulated using SERPENT, a multi-purpose three-dimensional continuous-energy Monte Carlo particle transport code, developed at VTT Technical Research Centre of Finland, Ltd [7]. The library version, geometry and material specification together with the temperature are identical to those in MCS model. The results from MCS and SERPENT are compared in Table III.

Table III: Comparison of SERPENT and MCS for MOX-3600

	SERPENT	Diff. vs. MCS (pcm)
k_{eff}	1.02800±0.00031	-12±32
β_{eff} (pcm)	360±8	-15±21
Doppler constant (pcm)	-982±9	-43±10
Control rod worth (pcm)	5551±44	-155±46
Sodium void worth (pcm)	1728±32	-132±35

The results calculated from SEREPENT are closed to those from MCS even for the Doppler constant. As shown in table III, difference in Doppler constant is 43 pcm or 4.65% in terms of relative error.

The reason why Doppler constant from UNIST differ from other institutes probably lays in the difference of cross-section interpolation algorithms. While cross-sections of the fuel at 1500 K and 1200 K are available in the ENDF/B-VII.1 library, the cross-section of structural materials at 743 K require interpolation. Furthermore, other institutes might choose other fuel temperature values to compute the Doppler constant.

In order to evaluate the impact of the interpolation procedures, we repeated the above calculation at different temperatures of fuel and structural materials. The results in pcm unit are shown in Table IV.

Table IV: Doppler constant of MOX-3600 with different fuel/structure temperature

	T_1=1500K T_2=1200K T_{struc}=600K	T_1=1500K T_2=750K T_{struc}=600K	T_1=1500K T_2=750K T_{struc}=743K
MCS	-961±9	-979±3	-938±3
SERPENT	-871±9	-847±9	-708±9
Diff. vs. MCS	90±12	132±9	230±9

The MCS results differ insignificantly with different fuel and structure temperature while SERPENT results shown relatively large change in Doppler constant when the structure and fuel's temperature are varied. The interpolation of cross sections by different method in MCS and other codes could be a reason for different Doppler constant. It is seen that the Doppler constant is very sensitive to fuel temperature chosen for calculation and code used.

The uncertainty analysis results are shown in Figure 2. Uncertainties lower than 10^{-2} % are not shown.

taken at the nuclear data library temperature point with and without interpolation confirms this conclusion.

Uncertainty in cross sections data has considerable impact on the k_{eff} 's value of the MOX-3600 core, especially from capture ross section of ^{23}Na and inelastic cross section of ^{238}U. Further analysis on uncertainty and sensitivity will be carried out to assess the effect of uncertainty from nuclear library to a greater extend.

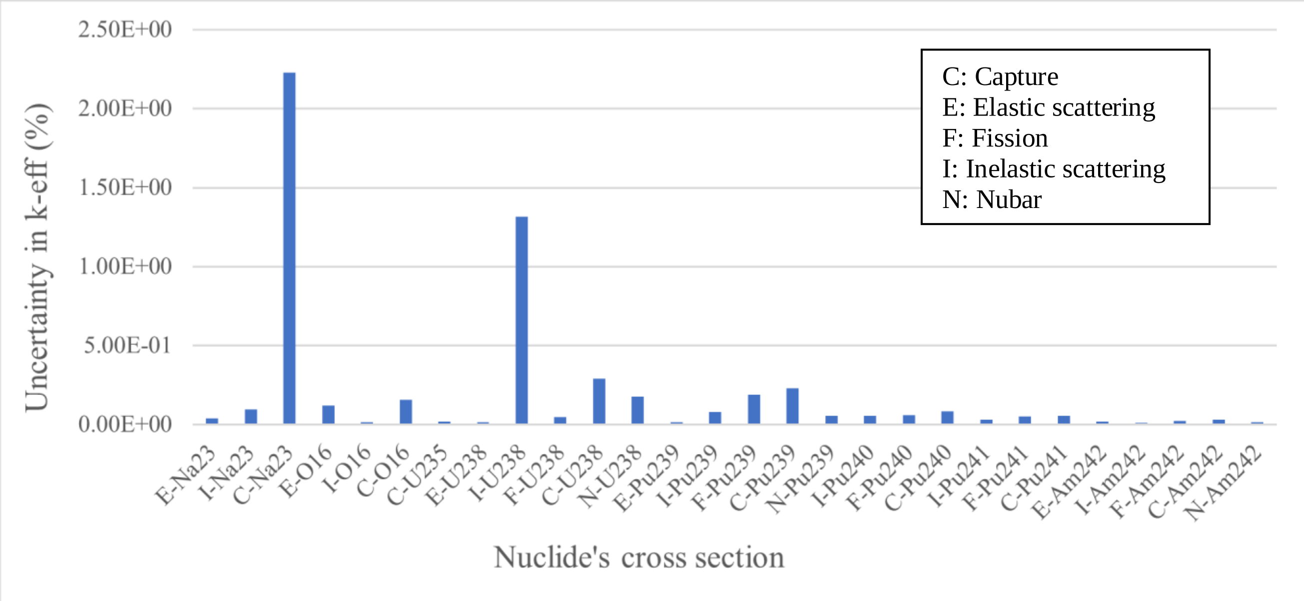

Figure 2. Uncertainty result in MOX-3600.

In total, the uncertainty due to cross sections of elements used in the MOX-3600 benchmark is around 5% with the capture cross section of ^{23}Na playing the most significant contributor to the uncertainty in k-eff (2.23%). Following is the inelastic and elastic scattering cross section of ^{238}U (1.32% and 2.91×10^{-1} %, respectively). Cross section of ^{239}Pu, ^{240}Pu and ^{242}Am also introduce around 10^{-1}% each to the k_{eff} uncertainty.

3. Conclusions

This study is devoted to the deterministic and uncertainty analysis of the MOX-3600 benchmark. For calculation of the benchmark, we use the Monte-Carlo code MCS, recently developed in UNIST. The calculation by MCS shows an agreement with SERPENT, TRIPOLI4, MCNPX with respect to fraction of delayed neutron, worth of control rods and sodium void effect. However, the large discrepancy in the Doppler effect is observed. We assumed that is due to a difference of the algorithms, used in the codes for interpolation of neutron cross-sections over a temperature. The calculation with the cross-sections

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