Assessment of Material Homogenization for Fast Reactor Design Using MCS and MC²/TWODANT/REBUS

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1. Introduction

ALFRED program (Advanced Lead-cooled Fast Reactor European Demonstrator) [1] is in the framework of the LEADER project (EURATOM VII Framework Programme) [2]. The analysis of ALFRED is devoted to the demonstration of the technology of fast reactors cooled by lead.

The purpose of this work is to present how the ALFRED core has been designed and characterized by simulating reactor core and calculating criticality results. The effective multiplication factor k_{eff} computation with UNIST Monte Carlo code MCS and deterministic code MC²-3/TWODANT/REBUS-3 [3-5] are also presented and evaluated to assess accuracy of the MC²-3/TWODANT/REBUS-3 in modeling of Lead-cooled Fast Reactor design.

2. ALFRED Core Design

ALFRED (Advanced Lead-cooled Fast Reactor European Demonstrator) is a demonstrator of the lead fast reactor technology, which generates 300 MW of thermal power. The core has a hexagonal lattice composed of 171 fuel assemblies (FA) - which is subdivided into two radial zones with different plutonium enrichment, 12 control rods (CR), 4 safety rods (SR) and surrounded by two rows of 108 dummy elements serving as a reflector, as shown in Fig. 1.

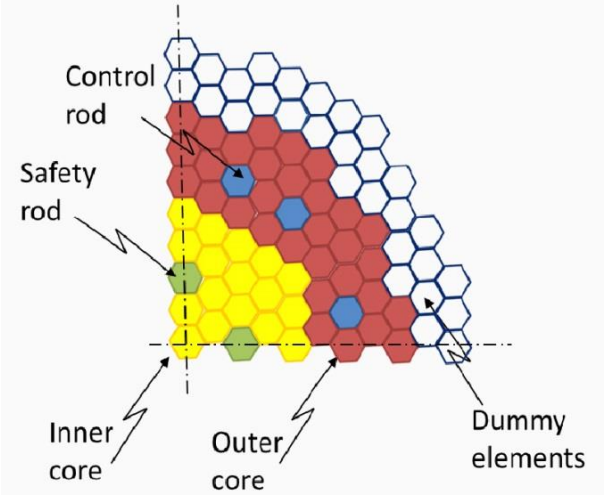

Fig. 1. Scheme of two enrichment zones of a quarter core.

In the following Table I, the main parameters of ALFRED are briefly summarized.

Table I: ALFRED - Main Parameters

Parameter	Unit	Values
Thermal Power	MW	300
Electrical Power	MW	125
Active height	mm	600
Pins per FA	-	127
Inner vessel radius	cm	165
Primary Coolant	-	Pure Lead
Core inlet temperature	°C	400
Core outlet temperature	°C	480
Fuel type maximum Pu content in the fuel	- %	MOX 30
Maximum fuel temperature	°C	~2000
Clad material	-	15-15 Ti
Maximum clad temperature at nominal condition	°C	550

3. Computational Modeling

3.1. Computer Codes

In this study, MCS Monte Carlo code and MC²-3/TWODANT/REBUS-3 deterministic code system were utilized to simulate ALFRED reactor core.

3.1.1. Monte Carlo code.

MCS is a 3D continuous-energy neutron-physics code for particle transport based on the Monte Carlo method, under development at UNIST [6,7]. ENDF/B-VII.0 and ENDF/B-VII.1 nuclear cross section libraries are utilized as continuous energy cross section data. MCS allows two types of calculations: criticality runs for reactivity calculations and fixed-source runs for shielding problems. The verification of criticality calculation ability is conducted by selected benchmark problem cases, namely the International Criticality Safety Benchmark Experimental Problem (ICBEP), BEAVRS and VERA. Another ability of MCS is multi-physics simulation, which is also verified with the solution for BEAVRS cycle 1.

3.1.2. Deterministic code system.

A fast reactor analysis code system is used to analyze the fast spectrum reactor. MC²-3 code [3] is an initial procedure which generates a multi-group cross-section for a fast reactor. The solution for the time-independent, multi-group discrete ordinate form of the Boltzmann

transport equation is obtained by TWODANT in the DANTSYS [4] code package. REBUS-3 [5] is a system of programs designed for the analysis of fast reactor fuel cycles and it can accommodate different geometries including triangular and hexagonal mesh. REBUS-3 has various neutronics solution algorithms such as finite difference, spatial flux synthesis, and nodal diffusion theory methods to provide the flux solution.

3.2. ALFRED Models

In this study, two different MCS models were considered: (i) a detailed core model considering all geometrical and compositional details of the core design – heterogeneous model (Fig. 2) and (ii) a coarser model homogenizing each drawer with all constituents present initially in full core analysis – homogeneous model (Fig. 3). The MCS simulation used the ENDF/B-VII.0 continuous energy cross-section and was performed with 500,000 neutrons per cycle with 400 active cycles and 100 inactive cycles, which was 250 million histories in total.

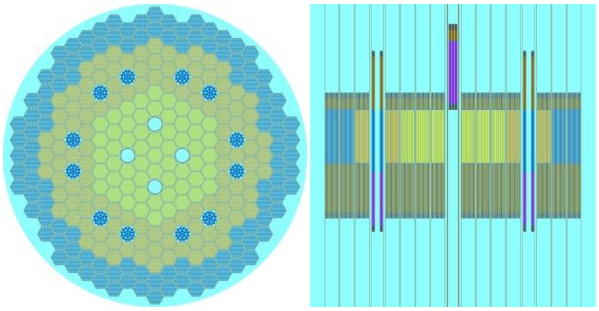

Fig. 2. Cross sections of MCS heterogeneous model.

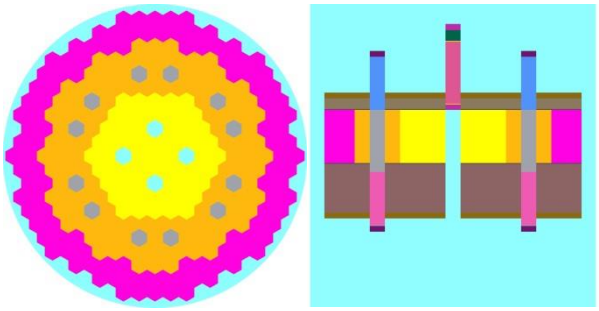

Fig. 3. Cross sections of MCS homogeneous model.

The deterministic model employed the MC^2-3 code in two different approaches to calculate the homogenized drawer's cross-sections in 1041 group energy structure. The first approach considered homogenized material of each plate in 2D R-Z geometry (Fig. 4), which was conducted by TWODANT.

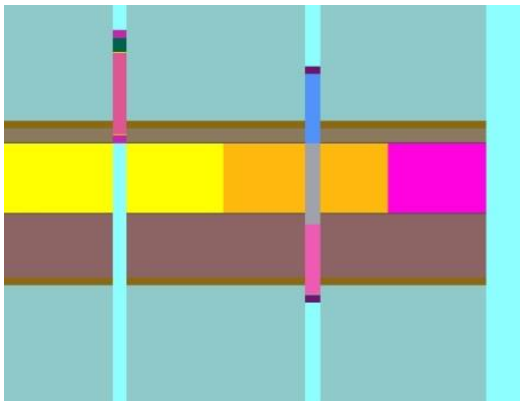

Fig. 4. A simplified axial scheme of ALFRED reactor defined in TWODANT.

The second approach homogenized the constituents of each drawer in a single cell, which was carried out by REBUS-3. Fig. 5 illustrates the fuel-contained layer in the quarter symmetry core that was defined in REBUS-3.

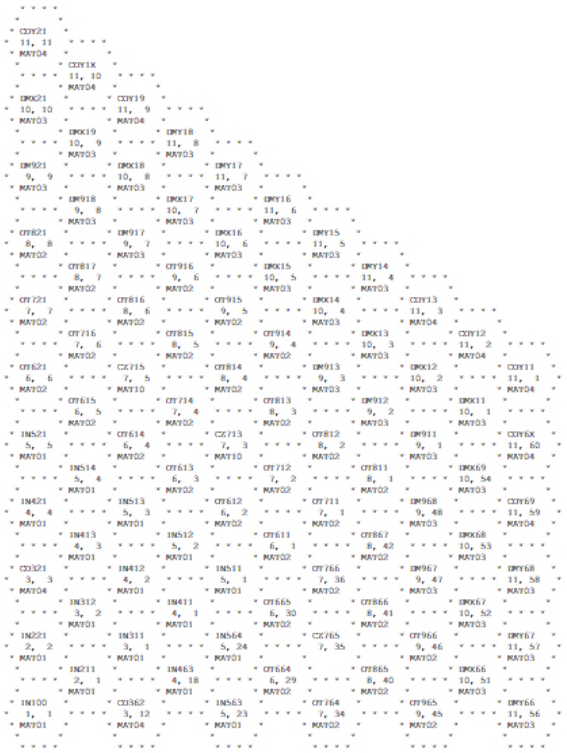

Fig. 5. A quarter symmetry core defined in REBUS-3.

Both deterministic method approaches were based on the ENDF/B-VII.0 data.

In addition, the temperatures chosen in this work were different for the core analysis and for other components:

- The core analysis, both for inner and outer fuel zone, was carried out at 900 K, the temperatures

available in the nuclear data of the basic library ENDF/B-VII.0;

- The other components: lead, dummy elements, control rods, safety rods, reflector, insulators, upper and bottom part of the system have been evaluated at 600 K.

4. Results and Discussion

In order to analyze the difference between heterogeneous and homogeneous models, the effective multiplication factor k_{eff} was computed and summarized in Table II. The gained k_{eff} from MCS heterogeneous model was selected as a reference (ref) to determine difference with k_{eff} results of remaining models.

Table II. Comparison of k_{eff} for ALFRED Core Calculated by MCS and MC^2-3/TWODANT/REBUS-3

Code	Modeling	k_{eff}	STD	Diff [pcm]
MCS	Heterogeneous	1.08337	0.00004	[ref]
MCS	Homogeneous	1.07432	0.00004	-905
MC^2-3/ TWODANT	Homogeneous	1.08112	-	-232
MC^2-3/ TWODANT/ REBUS-3	Homogeneous	1.07919	-	-418

The gained k_{eff} by MCS homogeneous model was 905 pcm lower compared to MCS heterogeneous model. The main reason explaining for this significant difference is the core models. Indeed, the FA consists of a ferritic steel wrapper enclosing 127 fuel pins arranged in a triangular lattice to form the bundle in the heterogeneous model; but as all materials were homogenized, this wrapper spread out the FA creating the decreasing in k_{eff} in the homogeneous model.

For the MC^2-3/TWODANT/REBUS-3 homogeneous simulation, the k_{eff} value obtained by this deterministic code was lower than one from the MCS heterogeneous calculations. Reason for this large difference (418 pcm) is possibly because of the different computation codes used and the treatment of the cross-section employed. Another factor is the impact of the core models (homogeneous vs. heterogeneous), which MCS employed a Monte Carlo approach with a detailed heterogeneous description.

As mentioned in Section 3.2, with similar homogenized materials, two models using deterministic code systems employed different approaches to simulate ALFRED reactor core: 2D R-Z geometry and 3D hexagonal geometry. In consequence, it leads to a 193 pcm in k_{eff} difference between these models.

Despite the fact that two homogeneous models by MCS and MC^2-3/TWODANT/REBUS-3 were all based on the same hexagonal geometry structure and materials, a considerable discrepancy in k_{eff} was still observed (483 pcm). It is mostly caused by the different neutronic methods, Monte Carlo and deterministic code.

5. Conclusion

In this work, four ALFRED models were simulated by both Monte Carlo and deterministic methods, including: MCS heterogeneous model, MCS homogeneous model, MC^2-3/TWODANT 2D homogeneous model and MC^2-3/TWODANT/REBUS-3 homogeneous model. By using ENDF/B-VII.0 continuous cross section library, criticality calculation was obtained by computing the effective multiplication factor of each model. The compared results indicate that the impact of homogenized materials to the fast reactor analysis is considerable due to a big difference between heterogeneous and homogeneous models. More effective and comprehensive system for analyzing fast reactors would be needed in further researches.

Acknowledgements

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