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Validation of the MC^2-3/DIF3D Code System for Control Rod Worth via the BFS-75-1 Reactor Physics Experiment

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1. Introduction

To validate core neutronics design and related safety parameters for innovative reactor, uncertainty quantification for nuclear data, i.e., cross-sections is an essential work. Many studies had been performed to quantify uncertainty induced by cross-section based on the sensitivity and uncertainty methodology [1] based on both of deterministic [2, 3] and Monte Carlo method [4, 5]. However, the expected uncertainty for the innovative reactor such as the KALIMER-600 reactor [6] might be overestimated comparing to other measured data in physics experiments [7-9]. Hence, because of the limitation in up-to-date evaluated cross-section covariance data, an integral experiment is more reliable to validate core neutronics design and related safety parameters [10].

The BFS-75-1 critical experiment was carried out in the BFS-1 facility of IPPE in Russia within the framework of validating an early phase of KALIMER-150 design [11]. The Monte-Carlo model of the BFS-75-1 critical experiment had been developed [12]. However, due to incomplete information for the BFS-75-1 experiments, Monte-Carlo models had been generated for the reference criticality and sodium void reactivity measurements with disk-wise homogeneous model.

Recently, KAERI performed another physics experiment, BFS-109-2A, by collaborating with Russian IPPE. During the review process of the experimental report of the BFS-109-2A critical experiments, valuable information for the BFS-1 facility which can also be used for the BFS-75-1 experiments was discovered. Hence the previous MCNP models [13] were updated as as-built models and additional loading models were built for control rod. In addition, deterministic models were also built for the purpose of validating neutronics design code, the MC^2-3/DIF3D code [14, 15]. The established models were validated based on the ENDF/B-VII.0 cross-section library [16].

2. Description of the BFS-75-1 Critical Assembly

The BFS-75-1 critical assembly is the uranium metal fueled core with two enrichment zones. The inner core of the BFS-75-1 critical assembly is configured of 15.11 wt.% LEZ(Low Enriched Zone) and the outer core is configured of 19.96 wt.% HEZ(High Enriched Zone) as shown in Fig. 1. The cylindrical fuel rods of the critical assemblies are arranged into a hexagonal lattice with a pitch of 5.1 cm. The unit fuel cell of the fuel rod consists of several types of cylindrical disks surrounded by a cylindrical tube with an outer diameter of 5.0 cm. RB1 represents radial blanket region 1 which is composed of metal uranium and RB2 represents radial blanket region 2 which is composed of depleted UO_2. In each region, 0.4 cm radius steel stick rods are inserted to satisfy steel volume fraction. A fuel experimental rod is composed of eight fuel unit cells which are surrounded by lower axial blanket and upper axial blanket. Table I shows types of control rods tested in the BFS-75-1 reactor physics experiment.

Table II lists loading number for the configuration of control rod worth measurement in the BFS-75-1 reactor physics experiment. The control rod positions are described in the Fig. 1.

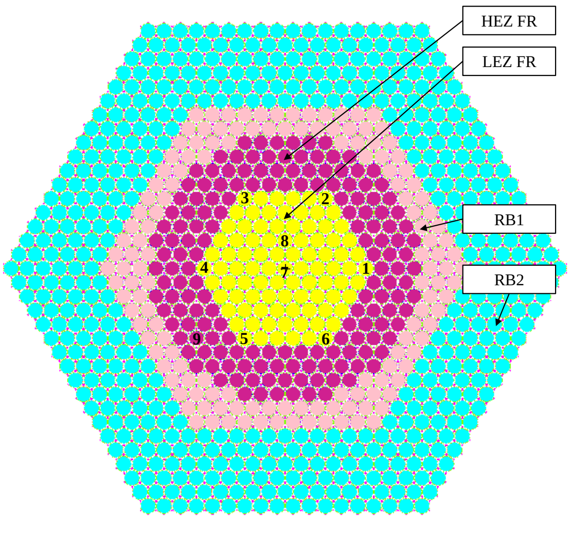

Fig. 1 Configuration of BFS-75-1 critical assembly

Table I: Types of control rods used in the BFS-75-1 critical experiments

Type	Description
1	Na disks (33 %), Steel disks (33 %), and natural B_4C disks (34 %)
2	80 wt.% enriched B_4C disks
3	Natural B_4C disks (~50 %) and Na disks (~50 %)
4	natural B_4C disks
5	Na disks (67 %) and Steel disks (33 %)

Table II: Loadings for the BFS-75-1 control rod worth

Loading number	Control rod type	Position
L000	Reference critical	
L101	Type 1	position 1
L102	Type 1	position 2
L103	Type 1	position 3
L104	Type 1	position 4
L105	Type 1	position 5
L106	Type 1	position 6
L107	Type 1	position 1, 4
L108	Type 1	position 1, 3, 5
L109	Type 1	position 1, 2, 3, 4, 5, 6
L110	Type 3	position 7
L111	Type 3	position 8
L112	Type 3	position 7, 8
L113	Type 4	position 7
L114	Type 4	position 8
L115	Type 4	position 9
L116	Type 1	position 7
L117	Type 2	position 7
L118	Type 5	position 7

3. Calculation Procedure for the MC^2-3/DIF3D Code

The calculation procedure of the MC^2-3/DIF3D codes for the fast reactor analysis is shown in Fig. 2. First, 1-D CPM (Collision Probability Method) calculation is performed with geometrical buckling for fuel unit cells to generate 1041 group homogenized cross-section using the MC^2-3 code. For non-fuel unit cells, 0-D slowing down calculation is performed to generate 1041 group homogenized cross-section using the MC^2-3 code.

Second, the TWODANT R-Z S_N transport calculation is performed to take into account global spectrum change based on the generated 1041 group cross-section as shown in Fig. 3. The purpose of the TWODANT R-Z S_N transport calculation is providing an inter-assembly spectrum difference in 1041 group structure, a rough calculation is sufficient for fast reactor analysis: S_8 angle quadrature and 5 cm axial mesh cell. Since different loading model results different global spectrum distributions, case-dependent TWODANT R-Z models were developed for the BFS-75-1 control rod worth calculation.

Third, 33 group homogenized cross-section is generated using both of TWODANT global flux distribution and 1-D or 0-D MC^2-3 calculations. Finally, 3-D hexagonal whole core calculation is performed using VARIANT option of the DIF3D code.

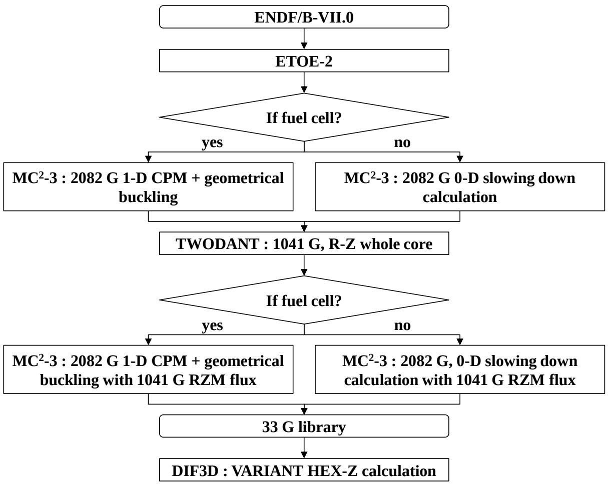

Fig. 2 MC^2-3/DIF3D calculation procedure

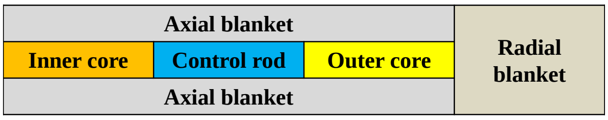

Fig. 3 Example of TWODANT R-Z model

Because real configuration of the BFS-75-1 unit cells is 3-D while capability of the MC^2-3 code is limited to 1-D, we adopted the 1-D homogenization method shown in Fig. 4 [17]. Hence to investigate 1-D homogenization effect, 1-D MCNP models were also built to verify 1-D MC^2-3 models.

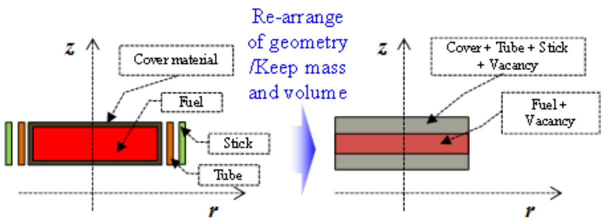

Fig. 4 Description of the 1-D homogenization method

4. Results

Table III shows C/E results of control rod worths in the BFS-75-1 reactor physics experiments for as-built MCNP, 1-D MCNP, and MC^2-3/DFI3D models. As-built MCNP model shows excellent agreement within 5.2 % maximum error for all types and all positions of control rods. 1-D MCNP models overestimate control rod worths by 1.1 % in average comparing to as-built MCNP models, while MC^2-3/DIF3D models overestimate control rod worths by 3.6 % in average comparing to as-built MCNP models. For L117 in which 80 wt.% enriched boron was used as control rod, MC^2-3/DIF3D model shows considerable overestimation comparing to 1-D MCNP model because highly enriched control rod may induce significant gradient in geometrical neutron flux distribution.

Table III. C/E results for the control rod worth of the BFS-75-1 reactor physics experiment, %

Loading	As-built MCNP	1-D MCNP	MC^2-3 /DIF3D
L101	1.4±1.0	1.7±1.0	4.2±0.6
L102	3.3±1.0	2.8±1.0	5.0±0.6
L103	1.8±1.2	3.2±1.2	5.5±0.9
L104	0.3±1.1	2.3±1.1	4.0±0.8
L105	1.4±1.1	2.0±1.1	4.2±0.8
L106	2.7±1.2	2.5±1.2	5.2±0.9
L107	0.2±0.6	0.8±0.6	3.5±0.5
L108	2.0±0.5	3.2±0.5	5.2±0.5
L109	1.4±0.4	2.7±0.4	5.5±0.4
L110	3.0±0.6	3.7±0.6	7.7±0.4
L111	-2.2±0.7	0.2±0.7	3.6±0.4
L112	-0.9±0.4	0.6±0.5	4.3±0.4
L113	4.8±2.0	5.9±2.1	8.6±2.1
L114	4.6±2.2	6.3±2.2	8.2±2.2
L115	4.2±3.5	5.2±3.6	4.3±3.5
L116	2.1±2.8	2.7±2.7	6.8±2.8
L117	1.0±0.8	5.4±0.9	7.7±0.9
L118	-5.2±4.9	-5.6±4.9	-2.9±4.9

5. Conclusions

In this paper, control rod worths of the BFS-75-1 reactor physics experiments were examined using continuous energy MCNP models and deterministic MC2-3/DIF3D models based on the ENDF/B-VII.0 library. We can conclude that the ENDF/B-VII.0 library shows very good agreement in small-size metal uranium fuel loaded core which is surrounded by the depleted uranium blanket.

However, the control rod heterogeneity effect reported by the reference [18] is not significant in this problem because the tested control rod models were configured by single rod. Hence comparison with other control rod worth measurements data such as the BFS-109-2A reactor physics experiment is planned as a future study.

Acknowledgements

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