Nucl. Sci. and Tech., Vol.6, No. 2 (2016), pp. 21-30

Nuclear Science and Technology

Journal homepage: https://jnst.vn/index.php/nst

Representative neutronic characteristics calculations for the VVER-1000 reactors using SRAC and MCNP5

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Abstract: This paper presents the results of neutronic calculations using the deterministic and Monte-Carlo methods (the SRAC and MCNP5codes) for the VVER MOX Core Computational Benchmark Specification and the VVER-1000/V392 reactor core. The power distribution and k_{eff} value have been calculated for a benchmark problem of VVER core. The results show a good agreement between the SRAC and MCNP5 calculations. Then, neutronic characteristics of VVER-1000/V392 such as power distribution, infinite multiplication factor (k-inf) of the fuel assemblies, effective multiplication factor k_{eff}, peaking factor and Doppler coefficient were calculated using the two codes.

Keywords: *VVER-1000/V392, SRAC, MCNP5, power distribution, multiplication factor, Doppler coefficient*

I. INTRODUCTION

For the Ninh Thuan 1 nuclear power project, Russia was selected as the international partner. At present, we are considering three versions of the VVER reactor technology: AES-91, AES-92 and AES-2006; in which the AES-92, an abbreviation of the VVER-1000/V392, may satisfy most of our requirements about technology and safety criteria. Therefore, one of the important tasks for Nuclear Power Center, Institute for Nuclear Science and Technology (INST) is to investigate the neutronic characteristics of the AES-92 technology. This is also included in the strategy of human resource development and research capability enhancement at INST in the period of 2010-2015.

In framework of the OECD/NEA Expert Group on Reactor based Plutonium Disposition the VVER-1000 MOX Core Computational Benchmark [1] has been proposed to investigate the physics of a whole VVER-1000 reactor with 30% MOX fuel. The benchmark problem has been resolved with three difference codes (MCU, MCNP and RADAR) and different nuclear databases. A comparison of the results shows a good agreement among the various codes, with maximum deviation of the average fission rate in the central assembly obtained via MCU and MCNP by 4% for state S4. In 2009, Thilagam et al. [2] re-analyzed this benchmark problem using the Indian calculation codes including EXCEL, TRIHEX-FA and HEXPIN. It was reported that the difference in the pin-by-pin fission rate distributions calculated using the HEXPIN diffusion code and the “Benchmark Mean (BM)” [1] was about 18% for the S6 state (the state with all control rods inserted into the core). In addition, the inter-comparison of the evaluated nuclear data libraries (JEFF-3.1 and JEF-2.2) was also performed with the

https://doi.org/10.53747/jnst.v6i2.143
Received 04 February 2016, accepted 04 July 2016

benchmark problem [3]. The results revealed that the difference of fission rates calculated using JEFF-3.1 and JEF-2.2 were as high as 9.2% due mostly to the difference in the cross-sections of the reflector isotopic compositions in the two libraries. More recently, the CNUREAS and MCNP5 calculations [4] were carried out to compare with the benchmark results, showing a difference up to 20% for fission rate distributions in case of the MCNP5 calculations with generic cross section libraries.

In this paper, the neutronic calculations were performed to examine the above benchmark problem by using the two codes, SRAC and MCNP5, with different nuclear data. The results obtained using these two codes were compared to each other and also to the published benchmark results. Subsequently, the neutronic characteristics of the VVER-1000/V392 reactor were investigated using two different calculation methods: Deterministic with SRAC (Standard thermal Reactor Analysis Code) and Monte-Carlo with MCNP5 (Monte Carlo N-Particle version 5). The purpose is to reveal the typical neutronic characteristics of the VVER-1000/V392 reactor in relation to those presented in the reference [5] for this type of reactor, namely Belene, of Bulgaria.

This paper consists of two parts: The first part presents the results from the benchmark calculations for 30% MOX reactor in comparison with the OECD/NEA report [1]; and the second one shows the results of VVER-1000/V392 calculations.

II. RESULTS OF THE OECD/NEA BENCHMARK CALCULATION [1]

A. Benchmark brief specifications

The benchmark model consists of a full-size 2-D VVER-1000 core with heterogeneous 30% MOX-fuel loading. The core was mixed of uranium oxide (UOX) and MOX fuel. A 2-D model of the VVER-1000 core was considered. Pattern of the VVER core with 30% MOX-fuel loading is presented in Fig.1.

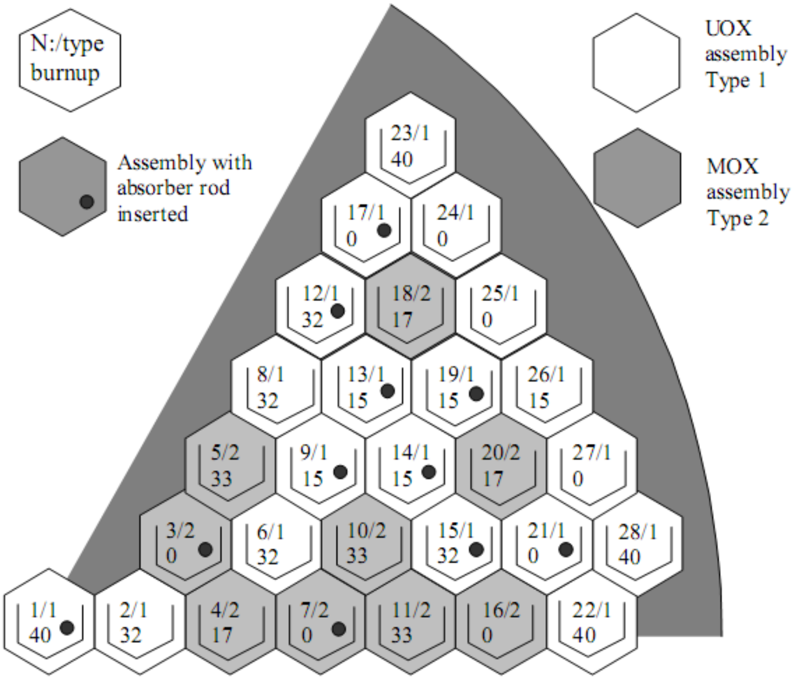


Fig. 1: Pattern of the VVER core with 30% MOX-fuel loading [1]

The fuel pins contain the fuel pellets with the radius of 0.386 cm and the pin pitch of 1.275 cm. The inside and outside diameters of the cladding are 0.772 cm and 0.910 cm, respectively. The fuel assembly cell types and the geometry data for the assembly (both UOX and MOX) are referred from the benchmark report [1].

The core consists of burnt and fresh fuel assemblies (FA):

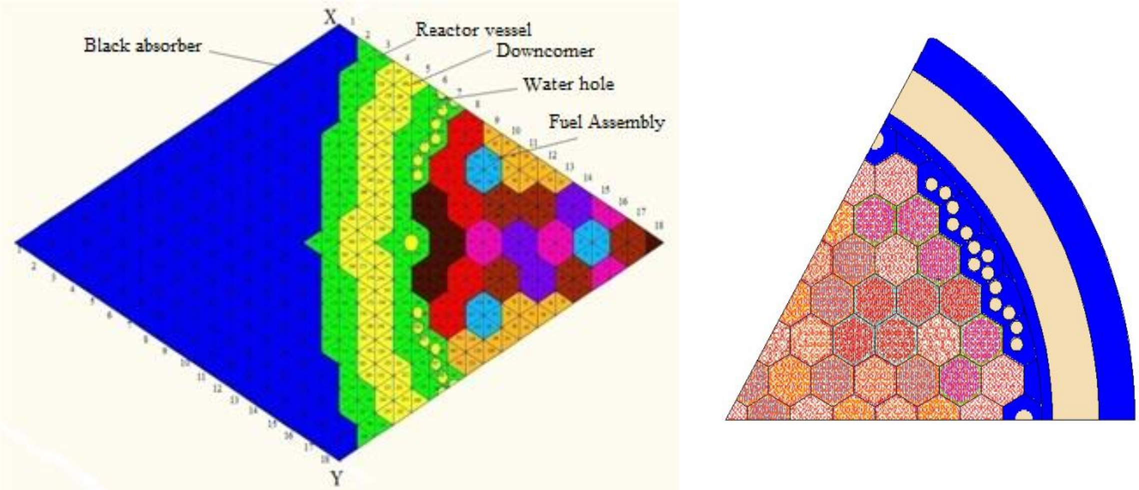
- 70% UOX type including 4 burn-up values (0, 15, 32 and 40 MWd/kg).
- 30% MOX type including 3 burn-up values (0, 17, and 33 MWd/kg).

The VVER assemblies are hexagonal in shape consisting of 331 lattice locations in a hexagonal array. The pitch of the fuel assembly is 23.6 cm. Each assembly contains 312 fuel pins, 18 guide tubes, and 01 instrumentation tube. The pins have cylindrical shape with Zr–Nb cladding.

The six reactor states and the models of one-sixth reactor core used in SRAC and MCNP5 are shown in Table I and Fig. 2, respectively. It is noted that the CITATION module functions the full core calculations in SRAC, where the Finite Different Method (FDM) was used to solve the diffusion equations.

Table I. Reactor state descriptions [1]

State	State name	Fuel temperature, ^{0}K	Moderator in FA temperature, ^{0}K	Moderator in FA material	Reflector temperature, ^{0}K	Water gap, water hole, down-comer material	Absorber rod
S1	Working state	1027	575	M575B1.3	560	M560B1.3	-
S2	State with constant temperature	575	575	M575B1.3	560	M560B1.3	-
S3	Cold state with high boron content	300	300	M300B2.8	300	M300B2.8	-
S4	Working state without boron	1027	575	M575B0	560	M560B0	-
S5	State with constant temperature without boron	575	575	M575B0	560	M560B0	-
S6	State with control rods inserted	553	553	M553B0	553	M553B0	Inserted



a, One-sixth reactor core model in SRAC

b, One-sixth reactor core model in MCNP

Fig. 2: One-sixth reactor core modeling in SRAC and MCNP5

B. Cross-section data

The ENDF/B-VII.1 nuclear data library was used in the MCNP5 calculations, where the cross sections for fuel and non-fuel materials were created by NJOY99 at 300K, 553K, 560K, 575K and 1027K. In the SRAC code the nuclear data library ENDF/B-VI.8 and Collision Probabilistic Method (CPM) were used to calculate the neutronic parameters of the fuel rods and fuel assemblies.

C. Results and discussion

Effective multiplication factor (k_{eff})

The k_{eff} calculation results for six states using the two codes are shown in Table 2, where a good agreement is shown between the SRAC, MCNP5, and benchmark values. In MCNP5, total of 100x10^6 neutrons history was used. For all states, the k_{eff} with an estimated standard deviation did not exceed 0.005%.

Table II. The results k_{eff} for six states

States	MCNP5	SRAC	MCNP-4C* [1]	MCNP5-MCNP4C (pcm)	SRAC-MCNP4C (pcm)	MCNP5-SRAC (pcm)
S1	1.04159	1.038176	1.03770	373.47	45.85	327.77
S2	1.05536	1.051769	1.05132	382.81	42.69	340.26
S3	0.93815	0.93568	0.93416	425.31	162.45	263.28
S4	1.14112	1.139747	1.13871	211.20	90.99	120.32
S5	1.15854	1.155145	1.15400	391.87	99.12	293.04
S6	1.05125	N/A	1.04729	377.26	N/A	N/A

**MCNP-4C: MCNP code used in the benchmark problem [1]*

Result for assembly average fission rate distribution for S1

Table III. Assembly average fission rate distribution for S1

No.	MCNP5	SRAC	MCNP-4C	Difference with MCNP-4C (%)	
				MCNP5	SRAC
1	0.764	0.759	0.764	0.06	0.65
2	0.938	0.935	0.928	1.11	0.75
3	1.276	1.221	1.226	4.06	0.41
4	1.158	1.102	1.100	5.25	0.18
5	0.951	0.932	0.940	1.15	0.85
6	0.988	1.004	0.994	0.57	1.01
7	1.199	1.177	1.180	1.58	0.25
8	0.988	0.996	0.999	1.08	0.30
9	1.261	1.295	1.296	2.73	0.08
10	0.914	0.908	0.922	0.90	1.52
11	0.836	0.857	0.864	3.24	0.81
12	1.018	1.006	1.009	0.87	0.30
13	1.369	1.385	1.389	1.43	0.29
14	1.326	1.360	1.361	2.59	0.07
15	0.949	0.979	0.977	2.84	0.20

16	1.149	1.143	1.160	0.92	1.47
17	1.218	1.247	1.201	1.44	3.83
18	1.209	1.127	1.150	5.13	2.00
19	1.313	1.326	1.319	0.48	0.53
20	1.120	1.074	1.096	2.23	2.01
21	1.122	1.209	1.165	3.67	3.78
22	0.545	0.572	0.566	3.75	1.06
23	0.377	0.374	0.366	3.05	2.19
24	0.883	0.919	0.868	1.68	5.88
25	0.984	1.038	0.983	0.07	5.60
26	0.817	0.842	0.821	0.50	2.56
27	0.787	0.852	0.807	2.42	5.58
28	0.345	0.361	0.353	2.33	2.27

In the state S1, the difference in the calculation results for the assembly average fission rates using MCNP4-C and RADAR [1] is about 1% to 4%. Table 3 shows the results obtained with SRAC and MCNP5 in comparison with those calculated using MCNP-4C. In general, both codes give acceptable deviation with the maximum of 5.25% for MCNP5 and 5.88% for SRAC. Such differences are reliable and acceptable in relation to those reported in the reference [2]. In detail, the maximum difference of average fission rate calculated using the CNUREAS deterministic code [2] and the “Benchmark Mean” [1] is about 14%; even such deviation from the “Benchmark Mean” might reach to 20% for the MCNP5 calculations.

III. NEUTRONIC CALCULATION RESULTS OF VVER-1000/V392

A. VVER-1000/V392 designs

In the fuel assembly (FA), there are 19 special channels. One of the channels is used to place neutron-measuring sensors in the in-core instrumentation system and the others are the guiding channels. Control Protection System (CPS) absorbing rods was inserted in guiding tube by mechanical drives.

At beginning of fuel cycle, the burnable absorber is used to decrease boric acid concentration and provide a negative coolant temperature coefficient of reactivity. As a result, this can make the radial power distribution flatter in the core. In the VVER-1000/V392 design, the burnable absorber is Gadolinium in form of oxide Gd_2O_3.

Control Protect System Control Rods (CPS CRs) are placed into the guiding channels of 121 non-periphery fuel assemblies. 103 CPS CRs are required for making the reactor to reach to sub-criticality even if there is no boric acid in the core.

The outer diameter of the fuel rod cladding is 9.1 mm; the inner diameter is 7.73 mm. The density and external diameter of Uranium dioxide (UO_2) pellets are 10.4-10.7 g/cm^3 and 7.6 mm respectively, and the diameter of central hole 1.2 mm . The U-Gd rods are enriched 5% Gadolinium oxide with structure similar to the UO_2 rods. The fuel rods were arranged in corners of the regular triangular lattice with a pitch of 12.75 mm.

The highest enrichments of ^{235}U in the fuel rod and the U-Gd rod are 4.95% and 3.6%, respectively. The effective length of the fuel rod is 3530 mm as shown in Table IV.

The isotopic compositions of the fuel cladding, the central and guide tubes, the absorber cladding, the absorber rod, the steel buffer, the steel barrel and the reactor pressure vessel are referred from the reference [5].

Table IV. Details of the fuel assembly for VVER-1000/V392 [5]

STT	Type FA - A	Average enrichment of^{235}U, % (mass)	Quantity of fuel rods (enrichment ^{235}U, % in mass)		Characteristics of fuel rods with gadolinium		
			Fuel rod type 1	Fuel rod type 2	Quantity of U-Gd	Fuel enrichment in^{235}Uin U-Gd, % mass	Content of Gd$_2$O$_3$, % mass
1	13A	1,30	312(1,3)	-	-	-	-
2	22A	2,20	312(2,2)	-	-	-	-
3	30A9P	2,98	303(3,0)	-	9	2,4	5
4	39A9P	3,90	243(4,0)	60(3,6)	9	3,3	5
5	39A6P	3,91	246(4,0)	60(3,6)	6	3,3	5
6	40A9Q	3,98	303(4,0)	-	9	3,3	5
7	44A9Q	4,38	303(4,4)	-	9	3,6	5
8	44A9P	4,38	303(4,4)	-	9	3,6	5
9	47A6Q	4,68	306(4,7)	-	6	3,6	5
10	44A6Q	4,39	306(4,4)	-	6	3,6	5
11	47A9P	4,33	306(4,7)	-	9	3,6	5
12	44B4W	4,92	288/4,4	-	24	3,6	8
13	50B6W	4,91	306/4,95	-	6	3,6	8
14	50B9W	4,84	303/4,95	-	9	3,6	8
15	50B4W	4,33	288/4,95	-	24	3,6	8

The VVER-1000/V392 core loading pattern for the first fuel cycle has five types of fuel assemblies which are 13A, 22A 30A9P, 39A6P, 39A9P. Their arrangement in the core is shown in Fig. 3.

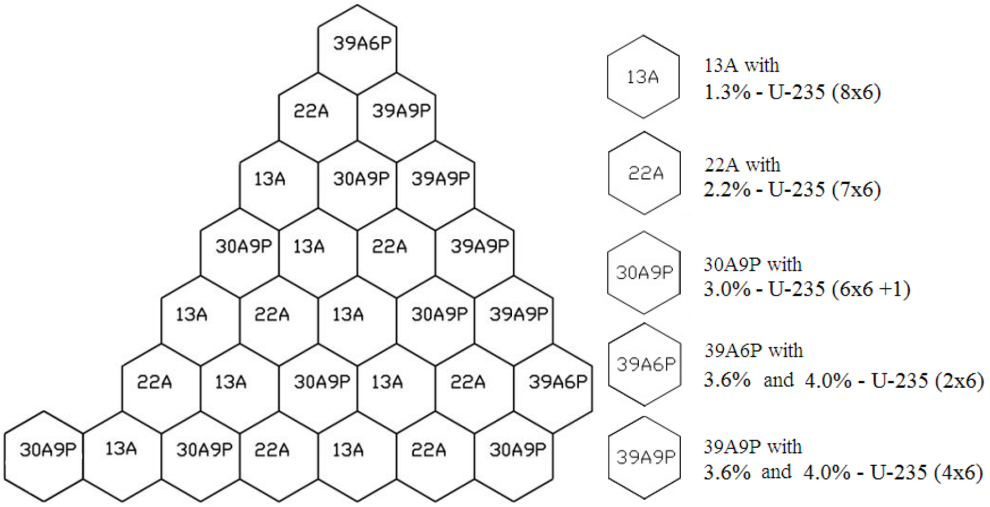


Fig.3: VVER-1000/V392 core in 60^{0}symmetry [5]

B. Results

Infinite multiplication factor of fuel assemblies

The k_{inf} values for fifteen fuel assemblies mentioned above were calculated using both SRAC and MCNP5. The boundary conditions consist of the six reflected planes covering each FA. The materials and geometry of FAs are taken from the reference [5]. The calculation results are shown in Table V, where a good agreement is obtained for the two codes.

Table V. k_{inf} of fuel assembly

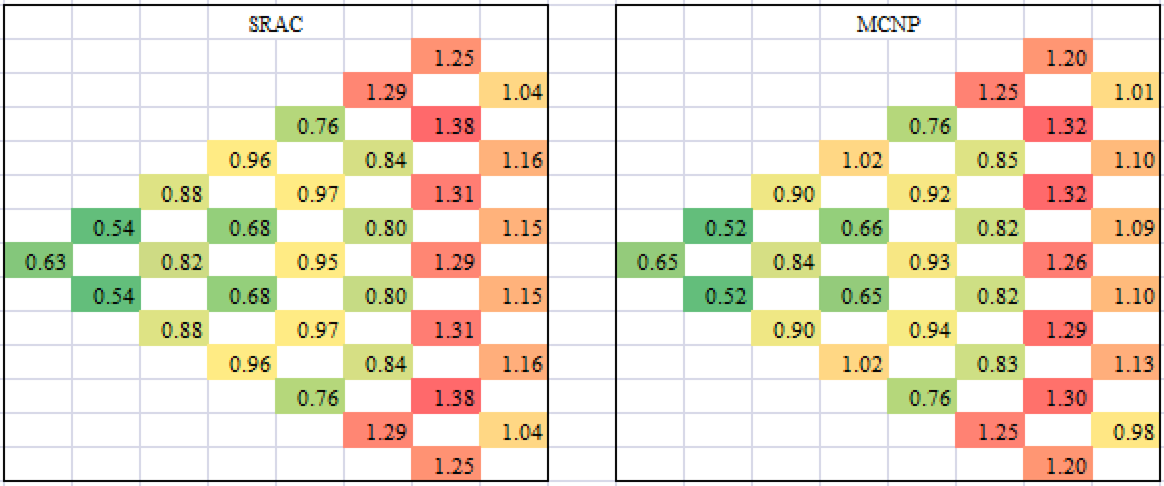
Name. of FAs	k_{inf}		Difference (pcm)
	SRAC	MCNP5	
13A	1.13020	1.12621	354.29
22A	1.30822	1.30251	438.38
30A9P	1.28843	1.28753	69.90
39A9P	1.36050	1.35916	98.59
39A6P	1.39289	1.39051	171.16
40A9Q	1.36351	1.36059	214.61
44A9Q	1.38592	1.38280	225.63
44A9P	1.38894	1.38633	188.27
47A6Q	1.42794	1.42393	281.61
44A6Q	1.41382	1.41021	255.99
47A9P	1.40482	1.40067	296.29
44B4W	1.23342	1.23422	-64.82
50B6W	1.44116	1.43766	243.45
50B9W	1.40783	1.40941	-112.10
50B4W	1.26348	1.26733	-303.79

Results of k_{eff} and power distribution

The full core calculations for the fresh fuel of the VVER-1000/V392 in the first fuel cycle were performed for two cases: (1) at the uniform temperature of 300 K assumed for the fuel and moderator, and (2) at the temperature of 1027 K assumed for the fuel and 576 K for the moderator. It is also assumed that in both cases, there is no boric acid in the moderator and all the control rods (included CR group 10) are withdrawn out of the core. The obtained results for k_{eff} values using SRAC and MCNP5 are shown in Table VI and the average fission rates in the one-sixth reactor core in Figs. 4 and 5 below.

Table VI. Calculation result for k_{eff}

k_{eff}	SRAC	MCNP	Difference (pcm)
Case1	1.23288	1.23661	-302.54
Case2	1.16002	1.16415	-356.03



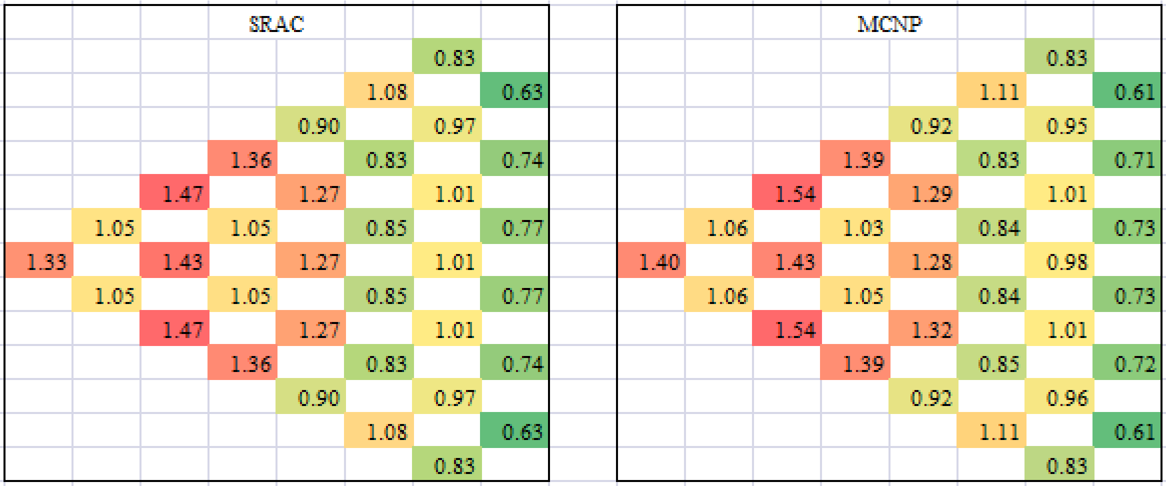
SRAC

						1.25	
					1.29		1.04
				0.76		1.38	
			0.96		0.84		1.16
		0.88		0.97		1.31	
	0.54		0.68		0.80		1.15
0.63		0.82		0.95		1.29	
	0.54		0.68		0.80		1.15
		0.88		0.97		1.31	
			0.96		0.84		1.16
				0.76		1.38	
					1.29		1.04
						1.25	

MCNP

						1.20	
					1.25		1.01
				0.76		1.32	
			1.02		0.85		1.10
		0.90		0.92		1.32	
	0.52		0.66		0.82		1.09
0.65		0.84		0.93		1.26	
	0.52		0.65		0.82		1.10
		0.90		0.94		1.29	
			1.02		0.83		1.13
				0.76		1.30	
					1.25		0.98
						1.20	

Fig.4: The average fission rate of one-sixth reactor core in Case 1



SRAC

						0.83	
					1.08		0.63
				0.90		0.97	
			1.36		0.83		0.74
		1.47		1.27		1.01	
	1.05		1.05		0.85		0.77
1.33		1.43		1.27		1.01	
	1.05		1.05		0.85		0.77
		1.47		1.27		1.01	
			1.36		0.83		0.74
				0.90		0.97	
					1.08		0.63
						0.83	

MCNP

						0.83	
					1.11		0.61
				0.92		0.95	
			1.39		0.83		0.71
		1.54		1.29		1.01	
	1.06		1.03		0.84		0.73
1.40		1.43		1.28		0.98	
	1.06		1.05		0.84		0.73
		1.54		1.32		1.01	
			1.39		0.85		0.72
				0.92		0.96	
					1.11		0.61
						0.83	

Fig.5: The average fission rate of one-sixth reactor core in Case 2

It can be seen from Figs. 4 and 5 that raising the temperature of moderator causes density of moderator decreases, which in turn causes the decrease of the moderator reflective efficiency in the area near the steel buffer. That is a reason why the peaking power tends to move to the center of the core. As shown in Table VI, the maximum difference for the k_{eff} is 356 pcm, and the discrepancies of the average fission rates for Case 1 and Case 2 are 6.23% and 5.61%, respectively.

Doppler coefficient of reactivity

The reactivity change, $\Delta\rho$ due to the temperature change is calculated by [6]:

$$\Delta\rho = \frac{k_{eff}^{T_2} - k_{eff}^{T_1}}{k_{eff}^{T_2} \times k_{eff}^{T_1}} \qquad (1)$$

Where, $k_{eff}^{T_2}$ and $k_{eff}^{T_1}$ are the effective multiplication factors corresponding to T_2 and T_1 temperature conditions. The Doppler coefficient (Dc) is then estimated as the change in reactivity per degree change in fuel temperature using equation (2) and is expressed in pcm/K.

$$D_C = \frac{\Delta\rho}{\Delta T} \qquad (2)$$

Where ΔT is the change in fuel temperature ($\Delta T = 600K$ in this case)

The Doppler coefficient was calculated under the following conditions:

- The moderator temperature is equal to 600K, with no boric acid and with no control rod insertion for power control (without CR group 10);

- The fuel temperature is changed gradually every 100K from 600 – 1200K.

It is well known that the Doppler feedback plays a crucial role in reactor controls [7]. For that reason, the Doppler coefficient was calculated using Eq. (2) and shown in Table VII.

Table VII. Doppler coefficient and k_{eff} at different fuel temperatures

Temperature (K)	k_{eff}		Difference (pcm)
	MCNP5	SRAC	
600	1.17257	1.16712	467.0
700	1.16721	1.16360	310.2
800	1.16414	1.16039	323.2
900	1.16129	1.15734	341.3
1000	1.15908	1.15437	408.0
1100	1.15602	1.15158	385.6
1200	1.15369	1.14890	416.9
Dc (pcm/K)	-2.32607	-2.26464	

As can be seen in Table VII, the k_{eff} decreases when the fuel temperature increases. This is an expected behavior thanks to the fuel Doppler effect. The decrease of the Doppler coefficient corresponding to the increase of the temperature from 600K to 1200K is found as -2.32607 pcm/K with MCNP5 and -2.2646 pcm/K with SRAC. These values meet the requirement reported in the reference [5], that the change of the reactivity should not exceed from -3.3 to -1.7 pcm/K. In addition, we can see that the difference in the k_{eff} values calculated by MCNP5 (Monte Carlo code) with ENDF/B-VII.1 and SRAC (deterministic code) with ENDF/B-VI.8 is within about 310 - 467 pcm.

IV. CONCLUSIONS

In this paper, we have carried out the full-core calculations for the multiplication factors and average fission reaction rates for the benchmark state S1 using SRAC and MCNP5, where the cross section data used in MCNP5 were processed by NJOY code. The comparison results show a good agreement between those calculated using SRAC and MCNP5.

The full-core calculations for the k_{eff} values and power distribution were performed in the following two cases: The first case is uniform temperature of 300K assumed for the fuel and moderator. The second one is considering that the temperatures of the fuel (1027K) and the moderator (576K) for the first fuel cycle of the VVER-1000/V392 reactor. The difference in the k_{eff} between the deterministic and Monte Carlo methods is within about 350 pcm. Furthermore, the dependence of the k_{eff} on the fuel temperature was also calculated to clarify the Doppler effect. The results obtained by both codes showed the Doppler feedback added a negative reactivity when increasing the fuel temperature and met the requirement on the Doppler coefficient given in the reference [5].

In general, it was shown that our results compared well with the benchmark values [1]. Moreover, the SRAC and MCNP5 calculations for the VVER-1000/V392 showed reasonable agreement with the recommended parameters in the reference [5], demonstrating that the SRAC and MCNP5 codes are reliable for neutronic calculations of the VVER reactors. It is being planned that these calculation codes will be used for analysis of the neutronic characteristics of the LWRs.

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