STEADY STATE CALCULATIONS OF THE PWR MOX/UO_2 CORE TRANSIENT BENCHMARK WITH MCNP6

Nguyen Thi Dung[1]*, Tran Vinh Thanh[2], Tran Viet Phu[1], Nguyen Huu Tiep[1], Thanh Mai Vu[3], Donny Hartanto[4], Pham Nhu Viet Ha[1]**

[1]Institute for Nuclear Science and Technology (INST), Vietnam Atomic Energy Institute (VINATOM), 179 Hoang Quoc Viet Str., Cau Giay Dist., Hanoi 100000, Viet Nam

[2]Vietnam Agency for Radiation and Nuclear Safety (VARANS), 113 Tran Duy Hung Str., Cau Giay Dist., Ha Noi 100000, Viet Nam

[3]VNU University of Science, Vietnam National University, 334 Nguyen Trai Str., Thanh Xuan Dist., Hanoi 10000, Viet Nam

[4]Department of Mechanical and Nuclear Engineering, University of Sharjah, P.O. BOX 27272, Sharjah, United Arab Emirates

*E-mail: *dungvlk52@gmail.com, **hapnv0915@gmail.com*

Abstract: This paper presents the steady-state analysis results of the OECD/NEA and U.S. NRC PWR MOX/UO_2 (MOX: Mixed Oxide) Core Transient Benchmark with the modern MCNP6 Monte Carlo code based on the up-to-date ENDF/B-VII.1 nuclear data library. The purpose is to verify the MCNP6 code and the ENDF/B-VII.1 library for calculations of a heterogeneous MOX/UO_2 fuelled PWR core, which has different neutronic characteristics from the popular homogeneous ones loaded with the UO_2 fuel due to its partial loading of the MOX fuel. The effective neutron multiplication factor, assembly power distributions, and control rod worths calculated using MCNP6 show a generally good agreement with the available benchmark data. The MCNP6 model and results obtained in this study will be applied to verify our MOX/UO_2 fuelled PWR core model being developed with the reactor kinetics code PARCS, which in turn will be used for further analyses of rod ejection accidents in a MOX/UO_2 fuelled PWR core.

Keywords: PWR, MOX/UO_2, MCNP6

I. INTRODUCTION

It has been recognized that utilizing the recycled plutonium as mixed-oxide (MOX) nuclear fuel in light water reactor cores could save the natural uranium resources and reduce either the amount of weapon-grade plutonium or the plutonium amount which has to be disposed to the final storage. Nevertheless, special concern on the control rod ejection accident (REA), which is a consequence of mechanical failure of the control rod drive mechanism casing located on the reactor pressure vessel top and categorized as design-basis reactivity-initiated accident in pressurized water reactors (PWRs), has also been raised for MOX fueled cores since their delayed neutron fractions are significantly smaller than those of UO_2 cores [1]. It is noticed that the control rod ejection transient can result in significant, localized perturbations of the neutronic and thermal-hydraulic core parameters, which can be difficult for reactor core simulators to predict accurately, especially in a heterogeneous MOX/UO_2 fueled core. Therefore partial loading of MOX fuel in a PWR core might call for an improvement of the calculation methods applied in the reactor core simulators. In this regard, the OECD/NEA and U.S. NRC PWR MOX/UO_2 Core Transient Benchmark [1] has been well defined with a complete set of reactor core steady-state and transient benchmark input and unique solutions designed to provide the framework to assess the heterogeneous transport and nodal diffusion transient methods and codes like the reactor kinetic code PARCS [2] to predict the control rod ejection transient response of a four-loop Westinghouse-type PWR core partially loaded with weapons grade MOX fuel, which is similar to that chosen for plutonium disposition in the U.S.

In this study, the steady-state calculations of the OECD/NEA and U.S. NRC PWR MOX/UO_2 Core Transient Benchmark were performed using the modern MCNP6 Monte Carlo code [3] and the up-to-date ENDF/B-VII.1 evaluated nuclear data library [4]. The goal of the study is to verify the MCNP6 code and the ENDF/B-VII.1 library for calculations of a heterogeneous MOX/UO_2 fuelled PWR core, which has different neutronic characteristics from the popular homogeneous ones loaded with the UO_2 fuel due to its partial loading of the MOX fuel. The effective neutron multiplication factor, assembly power distributions, and control rod worths were calculated using MCNP6 and compared against the available benchmark data. These values obtained with MCNP6 and ENDF/B-VII.1 are expected to be a new full-core heterogeneous transport solution for the benchmark at the steady-state conditions.

II. CALCULATION MODEL AND METHOD

The one-fourth symmetry of the PWR MOX/UO_2 core configuration and the main core design parameters are described in Fig. 1 and Table 1, respectively. The core has uniform fuel composition in axial direction and the axial reflector has the same width as the fuel assembly pitch. The axial reflector contains fixed moderator at the same condition with the core inlet and outlet for the bottom and top axial reflectors, respectively. The axial boundary condition (BC) is zero flux. The core is surrounded by a single row of reflector assemblies having the same width as the fuel assembly pitch. Each reflector assembly contains 2.52 cm thick baffle and has fixed moderator at the same condition with the core inlet. The outer radial BC is zero flux. The PWR UOX (UO_2) and MOX fuel assembly configurations are represented in Fig. 2. The UO_2 assembly configuration is a 17×17 lattice including 160 UOX rods, 104 UOX Integral Fuel Burnable Absorber (IFBA) pins and 24 control rods; whereas the MOX assembly configuration is a 17×17 lattice including 264 MOX rods and 24 Wet Annular Burnable Absorber (WABA) pins. The heavy metal composition in the fuel assemblies is shown in Table 2. Detailed information on the core geometry, material composition and benchmark problem can be found in Ref. [1]. In this calculation, the pure 2D problem of the above PWR MOX/UO_2 core (Part I of the benchmark), with no axial reflector and reflective boundary conditions in the axial direction, at fixed T/H conditions was examined using the Monte Carlo code MCNP6 based on the evaluated nuclear data library ENDF/B-VII.1. The reactor conditions that were analysed include HZP (hot zero power) ARO (all rods out) and HZP ARI (all rods in) with the boron concentration of 1000 ppm. The neutron multiplication factor (eigenvalue), assembly power distributions and single control rod worths were calculated using MCNP6 in relation to the available benchmark solutions.

It is worth noting that MCNP6 is a general-purpose Monte Carlo N-Particle transport code developed by the Los Alamos National Laboratory that can be used for neutron, photon, and electron or coupled neutron/photon/electron transport [3]. The code, as an advanced merger of MCNP5 and MCNPX, has many new features, capabilities, options in addition to those of MCNP5 and MCNPX, allowing its flexibility to be applied in various practical applications, e.g., criticality calculation, reactor design, safety analysis of nuclear facilities, reactor dosimetry, medical physics, etc. In addition, the recent released ENDF/B-VII.1 nuclear data library [4] has been extensively validated with the ICSBEP (International Criticality Safety Benchmark Evaluation Project) Benchmark, demonstrating its reliability in criticality calculations [5]. The ENDF/B-VII.1 library was further benchmarked with the MCNP6 Monte Carlo code using the ICSBEP Benchmark, showing its suitability in combination with MCNP6 especially for criticality calculations of the low-enriched uranium, compound fuel, thermal spectrum systems (LEU-COMP-THERM) like PWRs [6]. For that reason, the MCNP6 code was selected in the present study along with the ENDF/B-VII.1 library for calculations of the PWR MOX/UO_2 core.

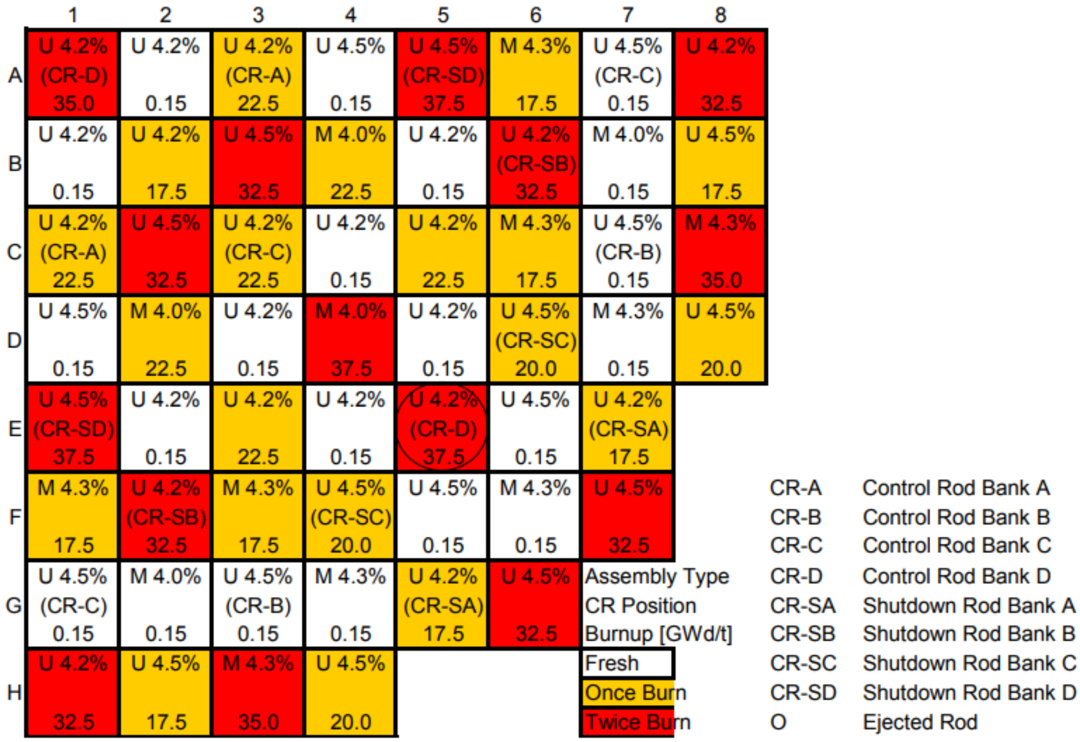

Fig. 1. PWR MOX/UO2 quarter-core configuration.

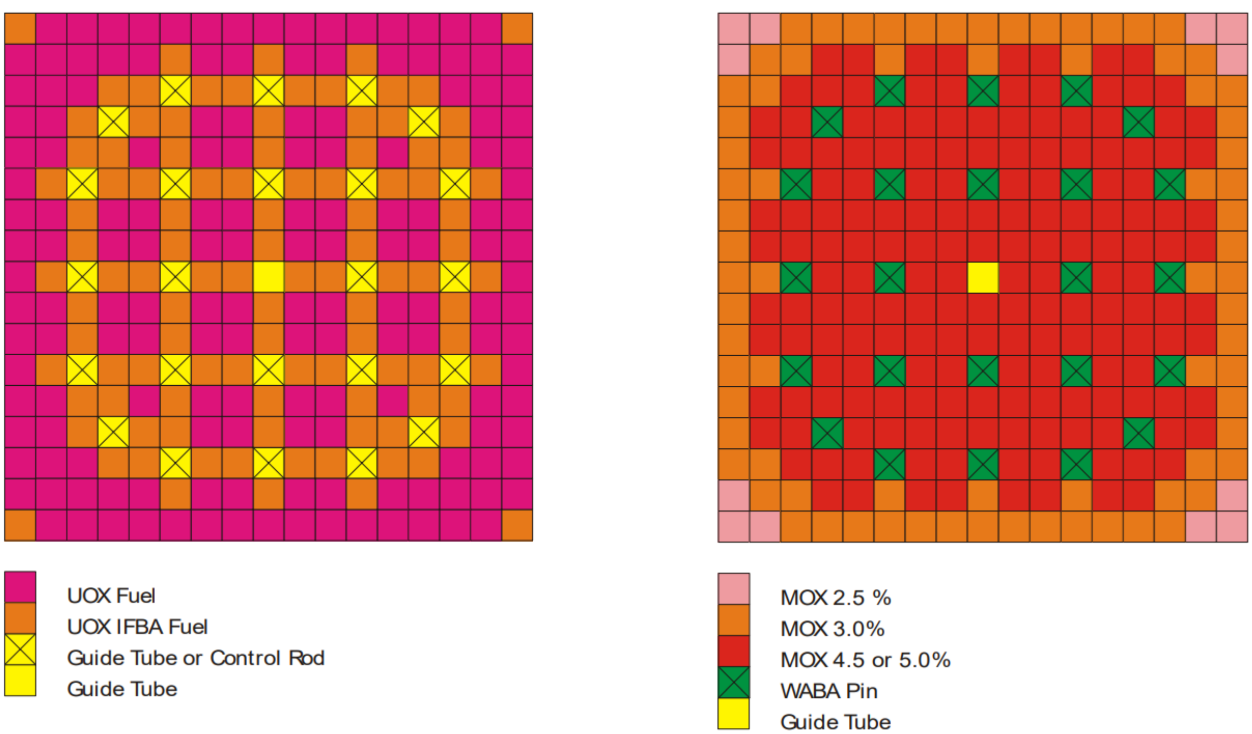

Fig. 2. UO_2 (right) and MOX (left) fuel assembly configuration.

Table 1. Main core design parameters.

Number of fuel assemblies	193
Power level (MWth)	3565
Core inlet pressure (MPa)	15.5
Hot full power (HFP) core average moderator temperature (K)	580.0
Hot zero power (HZP) core average moderator temperature (K)	560.0
Hot full power (HFP) core average fuel temperature (K)	900.0
Fuel lattice, fuel rods per assembly	17x17, 264
Number of control rod guide tubes	24
Number of instrumentation guide tubes	1
Total active core flow (kg/sec)	15849.4
Active fuel length (cm)	365.76
Assembly pitch (cm)	21.42
Pin pitch (cm)	1.26
Baffle thickness (cm)	2.52
Design radial pin-peaking (F_H)	1.528
Design point-wise peaking (F_Q)	2.5
Core loading (tHM)	81.6
Target cycle length (GWd/tHM) (months)	21.564 (18)
Capacity factor (%)	90.0
Target effective full power days	493
Target discharge burnup (GWd/tHM)	40.0-50.0
Maximum pin burnup (GWd/tHM)	62.0
Shutdown margin (SDM) (% $\Delta\rho$)	1.3

Table 2. Heavy metal (HM) composition in fuel.

<table>
<tr><th>Assembly type</th><th>Density [g/cm^3]</th><th colspan="2">HM material</th></tr>
<tr><td>UO_2 4.2%</td><td>10.24</td><td colspan="2">U-235: 4.2 wt%, U-238: 95.8 wt%</td></tr>
<tr><td>UO_2 4.5%</td><td>10.24</td><td colspan="2">U-235: 4.5 wt%, U-238: 95.5 wt%</td></tr>
<tr><td rowspan="3">MOX 4.0%</td><td rowspan="3">10.41</td><td>Corner zone:
2.5 wt% Pu-fissile</td><td rowspan="6">Uranium vector:
234/235/236/238 =
0.002/0.2/0.001/99.797 wt%

Plutonium vector:
239/240/241/242 =
93.6/5.9/0.4/0.1 wt%</td></tr>
<tr><td>Peripheral zone:
3.0 wt% Pu-fissile</td></tr>
<tr><td>Central zone:
4.5 wt% Pu-fissile</td></tr>
<tr><td rowspan="3">MOX 4.3%</td><td rowspan="3">10.41</td><td>Corner zone:
2.5 wt% Pu-fissile</td></tr>
<tr><td>Peripheral zone:
3.0 wt% Pu-fissile</td></tr>
<tr><td>Central zone:
5.0 wt% Pu-fissile</td></tr>
</table>

III. RESULTS AND DISCUSSION

As given in the benchmark, eight nodal diffusion method based solutions were obtained with the codes CORETRAN, EPISODE, NUREC, PARCS and SKETCH-INS, six of which were two-group (2G) and two were multi-group (MG). For the heterogeneous transport solutions, two cell homogeneous method based solutions were obtained with the codes BARS and DORT and two full-core heterogeneous method based solutions were obtained with the codes DeCART and MCNP4C2. It is noted that the transport codes without feedback and transient capability, e.g., DORT and MCNP, were able to perform only Part I of the benchmark. Accordingly, the full-core heterogeneous method with the MCNP6 code was applied in this work to solve Part I of the benchmark in comparison with the available benchmark solutions.

The comparison of the assembly power density obtained using DeCART and MCNP6 is shown in Fig. 3 and Fig. 4 at ARO and ARI conditions, respectively. These figures show a good agreement within ~6% between DeCART and MCNP6 in predicting the assembly power. Table 3 displays the comparison of eigenvalues and assembly power calculated using MCNP6 relative to the benchmark solutions. It can be seen that the MCNP6 results are in good agreement with the benchmark values obtained with different codes at both ARO and ARI conditions. Additionally, the total rod worth and the PWE (power-weighted error) and EWE (error-weighted error) values (see the definitions of PWE and EWE in Ref. [1]) obtained with MCNP6 were slightly higher than those obtained with the other codes most likely because of the fact that the recent released nuclear data library ENDF/B-VII.1 was used in the MCNP6 calculations as compared to the older ones that were used to obtain the benchmark solutions.

The single rod worths at ARO and ARI conditions were also calculated using MCNP6 and shown in Tables 4 and 5 in relation to the benchmark solutions, respectively. At ARO condition, the single rod worths predicted using MCNP6 agree with the benchmark values within 31 pcm. At ARI condition, the agreement between the MCNP6 and benchmark solutions was found to be within 56 pcm. Moreover, it can be seen that the MCNP results generally overestimated the benchmark solutions at both ARO and ARI conditions. Such discrepancy between the MCNP6 and benchmark solutions might be mainly contributed by (1) the use of the newer nuclear data library ENDF/B-VII.1 in the MCNP6 calculations as compared to the older ones that were used to obtain the benchmark solutions and (2) the difference in the neutron transport solution method that was employed in each code.

1.327	1.683	1.395	1.513	1.037	1.051	1.016	0.413
1.667	1.514	1.227	1.281	1.334	0.913	0.995	0.492
1.370	1.208	1.287	1.413	1.237	1.131	1.002	0.398
1.480	1.248	1.414	1.073	1.305	1.143	0.903	0.334
1.026	1.330	1.237	1.306	0.911	1.062	0.577	
1.054	0.927	1.134	1.150	1.073	0.766	0.280	
0.999	1.004	1.005	0.913	0.586	0.281		
0.418	0.495	0.407	0.347				

1.374	1.735	1.418	1.525	1.035	1.032	0.997	0.413
1.735	1.563	1.245	1.277	1.349	0.918	0.978	0.491
1.418	1.245	1.325	1.446	1.247	1.114	0.991	0.393
1.525	1.277	1.446	1.076	1.308	1.143	0.892	0.341
1.035	1.348	1.247	1.308	0.904	1.067	0.585	
1.032	0.917	1.114	1.142	1.067	0.754	0.281	
0.997	0.978	0.991	0.892	0.585	0.281		
0.413	0.491	0.393	0.34				

-3.440	-2.979	-1.652	-0.786	0.206	1.808	1.896	0.105
-3.944	-3.137	-1.455	0.291	-1.147	-0.571	1.766	0.186
-3.400	-2.993	-2.867	-2.296	-0.795	1.530	1.073	1.344
-2.982	-2.258	-2.221	-0.242	-0.199	0.022	1.235	-2.165
-0.872	-1.312	-0.782	-0.118	0.827	-0.510	-1.420	
2.087	1.067	1.811	0.721	0.602	1.601	-0.426	
0.242	2.695	1.382	2.302	0.250	0.082		
1.302	0.809	3.642	2.021				

Fig. 3. Assembly power distributions obtained by MCNP6 (this work) (top) and DeCART (middle) and relative difference (%) between MCNP6 and DeCART (bottom) at ARO condition.

1.182	2.484	1.213	2.219	0.764	0.681	0.301	0.207
2.487	2.442	1.830	2.187	1.897	0.466	0.499	0.267
1.224	1.837	1.235	2.556	2.034	1.045	0.342	0.200
2.249	2.193	2.541	1.925	1.758	0.554	0.462	0.179
0.760	1.889	2.022	1.735	0.533	0.708	0.197	
0.666	0.456	1.020	0.554	0.714	0.587	0.193	
0.288	0.478	0.330	0.452	0.195	0.191		
0.193	0.260	0.199	0.181				

1.209	2.533	1.202	2.196	0.742	0.669	0.300	0.205
2.533	2.459	1.812	2.103	1.832	0.449	0.489	0.268
1.202	1.812	1.198	2.452	1.944	0.985	0.329	0.198
2.196	2.103	2.452	1.823	1.675	0.531	0.450	0.186
0.742	1.832	1.944	1.675	0.508	0.696	0.190	
0.669	0.449	0.985	0.531	0.696	0.562	0.186	
0.300	0.489	0.329	0.450	0.190	0.186		
0.205	0.268	0.198	0.186				

-2.242	-1.918	0.903	1.052	2.931	1.803	0.449	1.144
-1.813	-0.682	0.999	4.013	3.542	3.720	2.115	-0.196
1.850	1.379	3.114	4.232	4.605	6.115	3.850	1.058
2.424	4.302	3.618	5.618	4.979	4.421	2.724	-3.595
2.489	3.135	4.024	3.594	4.831	1.750	3.638	
-0.447	1.561	3.557	4.395	2.538	4.391	3.771	
-3.857	-2.258	0.233	0.486	2.397	2.450		
-5.619	-3.014	0.310	-2.434				

Fig. 4. Assembly power distributions obtained by MCNP6 (this work) (top) and DeCART (middle) and relative difference (%) between MCNP6 and DeCART (bottom) at ARI condition.

Table 3. Comparison of eigenvalues and assembly power.

	Eigenvalue		Total rod worth (dk/k)	Assembly power error			
				ARO		ARI	
	ARO	ARI		%PWE	%EWE	%PWE	%EWE
nodal							
CORETRAN 1/FA	1.06387	0.99202	6808	1.06	1.69	2.01	2.52
CORETRAN 4/FA	1.06379	0.99154	6850	0.96	1.64	1.67	2.18
EPISODE	1.06364	0.99142	6849	0.96	1.64	1.66	2.16
NUREC	1.06378	0.99153	6850	0.96	1.63	1.64	2.16
PARCS 2G	1.06379	0.99154	6850	0.96	1.63	1.67	2.18
PARCS 4G	1.06376	0.99136	6865	0.90	1.42	1.61	2.26
PARCS 8G	1.06354	0.99114	6868	0.86	1.25	1.65	2.49
SKETCH-INS	1.06379	0.99153	6850	0.97	1.67	1.67	2.16
heterogeneous							
BARS	1.05826	0.98775	6745	1.29	1.92	3.92	10.30
DeCART	1.05852	0.98743	6801	reference	reference	reference	reference
DORT	1.06036	-	-	0.86	1.12	-	-
MCNP4C2	1.05699	0.98540	6873	0.67	1.26	1.33	3.67
MCNP6 (this work)	1.06064	0.98812	6920	1.59	2.21	2.9	3.59

Table 4. Comparison of single rod worths at ARO condition (dk/k).

	Rod position									
	(A,1)	(A,3)	(A,5)	(A,7)	(B,6)	(C,3)	(C,7)	(D,6)	(E,5)	(E,7)
nodal										
CORETRAN 1/FA	164	143	91	53	70	122	51	68	64	28
CORETRAN 4/FA	166	144	92	53	70	123	51	69	65	28
EPISODE	165	134	-	53	70	123	51	69	64	27
NUREC	166	143	91	53	70	122	51	68	64	27
PARCS 2G	166	143	91	53	70	123	51	68	64	27
PARCS 4G	167	144	91	53	70	122	51	68	64	27
PARCS 8G	168	144	91	52	69	123	50	68	64	27
SKETCH-INS	166	143	91	53	70	123	51	68	64	27
heterogeneous										
BARS	166	139	87	49	66	117	49	66	63	27
DeCART	-	-	-	-	-	-	-	-	-	-
DORT	-	-	-	-	-	-	-	-	-	-
MCNP4C2	-	-	-	-	-	-	-	-	-	-
MCNP6 (this work)	182	154	118	43	86	122	61	94	72	30

Table 5. Comparison of single rod worths at ARI condition (dk/k).

	Rod position									
	(A,1)	(A,3)	(A,5)	(A,7)	(B,6)	(C,3)	(C,7)	(D,6)	(E,5)	(E,7)
nodal										
CORETRAN 1/FA	-826	-875	-397	-57	-151	-1115	-78	-291	-246	-22
CORETRAN 4/FA	-840	-880	-405	-55	-152	-1127	-78	-290	-249	-20
EPISODE	-843	-884	-	-59	-155	-1130	-81	-293	-253	-24
NUREC	-840	-880	-405	-56	-152	-1127	-78	-290	-249	-21
PARCS 2G	-840	-880	-405	-56	-152	-1127	-78	-290	-249	-21
PARCS 4G	-849	-886	-407	-55	-153	-1134	-77	-290	-250	-21
PARCS 8G	-857	-889	-409	-54	-153	-1139	-76	-290	-253	-20
SKETCH-INS	-840	-880	-405	-56	-152	-1127	-78	-290	-249	-21
heterogeneous										
BARS	-914	-921	-417	-44	-145	-1193	-68	-313	-268	-17
DeCART	-	-	-	-	-	-	-	-	-	-
DORT	-	-	-	-	-	-	-	-	-	-
MCNP4C2	-	-	-	-	-	-	-	-	-	-
MCNP6 (this work)	-882	-895	-433	-73	-154	-1142	-97	-312	-285	-48

IV. CONCLUSIONS

In this work, the steady-state calculations of the OECD/NEA and U.S. NRC PWR MOX/UO_2 Core Transient Benchmark were performed using the modern MCNP6 Monte Carlo code and the recent released ENDF/B-VII.1 evaluated nuclear data library. The eigenvalues, assembly power distributions, and control rod worths calculated using MCNP6 exhibited a generally good agreement with the available benchmark solutions. The discrepancy between the MCNP6 and benchmark solutions might be largely contributed by (1) the use of the newer nuclear data library ENDF/B-VII.1 in the MCNP6 calculations as compared to the older ones that were used to obtain the benchmark solutions and (2) the difference in the neutron transport solution method that was employed in each code. Consequently, these results obtained with MCNP6 and ENDF/B-VII.1 can be considered as a new full-core heterogeneous transport solution to supplement for the available benchmark solutions at the steady-state conditions. Moreover, the MCNP6 model and the results obtained will be applied to verify our MOX/UO_2 fuelled PWR core model being developed with the reactor kinetics code PARCS, which in turn will be used for further analyses of REAs in a MOX/UO_2 fuelled PWR core.

ACKNOWLEDGMENTS

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TÍNH TOÁN BÀI TOÁN BENCHMARK CHUYỂN TIẾP VÙNG HOẠT LÒ PHẢN ỨNG PWR NẠP TẢI NHIÊN LIỆU MOX/UO_2 TẠI CÁC TRẠNG THÁI ỔN ĐỊNH DÙNG MCNP6

Nguyễn Thị Dung[1]*, Trần Vĩnh Thành[2], Trần Việt Phú[1], Nguyễn Hữu Tiệp[1], Vũ Thanh Mai[3], Donny Hartanto[4], Phạm Như Việt Hà[1]**

[1]Viện Khoa học và Kỹ thuật Hạt nhân, Viện Năng lượng Nguyên tử Việt Nam, 179 Hoàng Quốc Việt, Cầu Giấy, Hà Nội, Việt Nam

[2]Cục An toàn Bức xạ và Hạt nhân, 113 Trần Duy Hưng, Cầu Giấy, Hà Nội, Việt Nam

[3]Đại học Khoa học Tự nhiên, Đại học Quốc gia, 334 Nguyễn Trãi, Thanh Xuân, Hà Nội, Việt Nam

[4]Khoa Kỹ thuật Cơ khí và Hạt nhân, Đại học Sharjah, Sharjah, Các Tiểu vương quốc Ả Rập Thống Nhất

*E-mail: *dungvlk52@gmail.com, **hapnv0915@gmail.com*

Tóm tắt: Bài báo trình bày kết quả phân tích bài toán benchmark chuyển tiếp vùng hoạt lò phản ứng nước áp lực (PWR) nạp tải nhiên liệu MOX/UO_2 (MOX: oxit hỗn hợp) của OECD/NEA và U.S. NRC tại các trạng thái ổn định sử dụng chương trình MCNP phiên bản mới nhất là MCNP6 và bộ thư viện hạt nhân mới ENDF/B-VII.1. Mục đích là để kiểm chứng MCNP6 và ENDF/B-VII.1 đối với các tính toán vùng hoạt PWR nạp tải MOX/UO_2, mà có các đặc tính vật lý neutron khác với các vùng hoạt PWR thông dụng chỉ nạp tải UO_2. Hệ số nhân neutron hiệu dụng, phân bố công suất của các bó nhiên liệu, và giá trị các thanh điều khiển được tính toán bởi MCNP6 nói chung cho thấy sự phù hợp tốt với các dữ liệu trong bài toán benchmark. Các kết quả đạt được ở đây sẽ được áp dụng để kiểm chứng mô hình PWR nạp tải MOX/UO_2 đang được phát triển bằng chương trình tính toán động học lò phản ứng PARCS, mà sẽ được dùng để phân tính các sự cố bật thanh điều khiển ra khỏi vùng hoạt PWR nạp tải MOX/UO_2.

Từ khóa: Lò phản ứng nước áp lực, nhiên liệu MOX/UO2, MCNP6