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Calculation of 2-Dimensional PWR MOX/UO2 Core Benchmark OECD NEA 6048 with SRAC Code

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A B S T R A C T

The mixed uranium-plutonium oxide fuel (MOX/UO2) is an interesting fuel for future power reactors. This is due to the large amount of plutonium that can be processed from spent fuel of nuclear plants or from plutonium weapons. MOX/UO2 fuel is very flexible to be applied in thermal reactors such as PWR and it is more economical than UO2 fuel. However, due to the different nature of neutron interactions of MOX in PWR, it will change the reactor core design parameters and also its safety characteristic. The purpose of this study is to determine the accuracy of SRAC2006 code system in generation of cross-sections and calculation of reactor core design parameters such as criticality, reactivity of control rods and radial power distribution. In this study, PWR MOX/UO2 Core Transient Benchmark is used to verify the code that models a MOX/UO2 fueled core. SRAC-CITATION result is different from DeCART by 0.339% from. SRAC-CITATION result of single rod worth in All Rods Out (ARO) conditions are quite good with a maximum difference of 6.34% compared to BARS code and 4.74% compared to PARCS code. In All Rods In (ARI) condition, SRAC-CITATION results compared to the PARCS code is quite good where the maximum difference is 9.72%, but compared to BARS code, it spikes up to 33.24% at maximum difference. In the other case, overall radial power density results are quite good compared to the reference. Its maximum deviation from DeCART code is 5.325% in ARO condition and 6.234% in ARI condition. Based on the results of these calculations, SRAC code system can be used to generate cross-section and to calculate some neutronic parameters. Hence, it can be used to evaluate the neutronic parameters of the MOX/UO2 PWR core design.

1. INTRODUCTION

An important issue for the future of large Nuclear Power Plant (NPP) operations is the reliability of uranium fuel supply. If there is a limitation of uranium supply, an alternative fuel replacement must be found. One potential candidate for alternative fuel in the future is MOX fuel, which is based on a mixture of plutonium and uranium in its oxide form. With the use of MOX, the need for uranium fuel can be reduced. The main advantages of recycling plutonium to make a MOX fuels are reducing the amount of enriched uranium and reducing radioactive waste generated from spent nuclear fuel [1]. For this reason, research is still being carried out regarding the use of MOX fuel in PWR reactors [2][3][4]. Most light water-cooled (LWR) reactors have been licensed to use MOX fuels at fraction up to 30% or more on the reactor core [5][6]. Korean Utility Requirements (KUR) states that the design of nuclear reactors with MOX

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fuel can reach 30% of the core. At present, EUR (European Utility Requirements) has a nuclear design capability of up to 50% MOX in its core [5] and APR1400 has successfully demonstrated the ability to design a core with 50% MOX on it to obtain EUR certification [7].

The accuracy of the neutronic calculation code is not only influenced by its calculation method, but also by the nuclear data used, modeling of fuel assembly and the core. Because the modeling capabilities of neutronic calculation code are limited, some uncertainty will exist in the output of the code. Verification of neutronic calculation code becomes an important thing for user to do to ensure that they are using the code correctly in modelling the calculation cases, and also to compare the calculated result from the code with other code that has been done modelling the same cases. There has been a lot of research related to the validation of neutronic calculation code [8][9][10]. SRAC2006 is a code system applicable to neutronics analysis of a variety of reactor types [11]. The verification of the SRAC2006 code system carried out in this paper uses the Pressurized Water Reactor (PWR) MOX/UO2 Core Transient Benchmark reference from the OECD Nuclear Energy Agency (NEA). PWR MOX/UO2 Core Transient Benchmark issued by the Nuclear Science Committee of the OECD NEA was used as a reference by researchers around the world to verify calculations of UO2 / MOX fueled PWR core [12]. The most commonly code to use as PWR reactor core analysis is the NODAL3 program [13][14][15]. NODAL3 can use cross section data and group constant generated by PIJ module from SRAC2006.

The neutronic parameters calculated in this study relate to the safety of reactor operations, namely the effective multiplication factor (K_{eff}) to calculate control rod reactivity, radial power peaking factor. The calculation results then will be compared with references data from DeCART or BARS to represent a heterogeneous solutions and PARCS as a nodal solutions programs [12]. Both of DeCART and PARCS generate cross-sections using HELIOS 1.7, which has often been used in PWR core calculations. The aim of this study is to verify SRAC2006 system code on generating cross-sections data and static parameters of the PWR MOX/UO2 benchmark core.

2. PWR MOX/UO2 CORE BENCHMARK

The reactor core from PWR MOX/UO2 benchmark is based on four-loop Westinghouse PWR that has a similarity to the reactor proposed for plutonium disposition program in the USA. Quarter core configuration is shown in Fig. 1 and core design parameter is shown in Table 1. Both UO_2 fuel assembly with 104 IFBA (Integral Fuel Burnable Absorber) and MOX fuel assembly with 24 WABA (Wet Annular Burnable Absorber) configuration is shown in Fig. 2 and the material composition for each type of fuel pin is shown in Table 2.

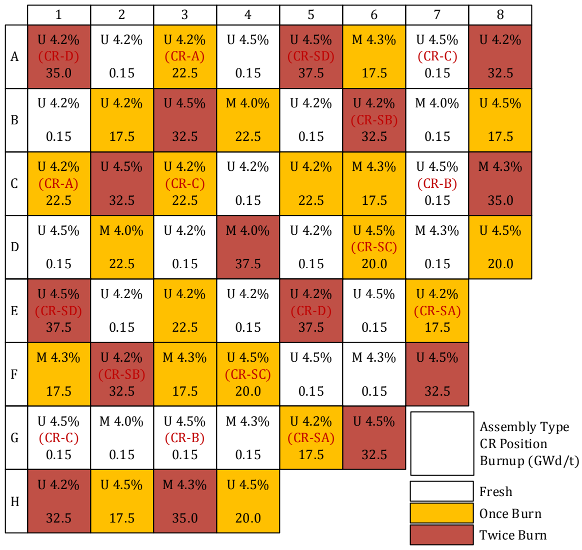

Fig. 1. Quarter-core geometry [12]

Table 1. Core design parameters [12]

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
Hot zero power (HZP) core average moderator temperature (K)	560
Hot full power (HFP) core average fuel temperature (K)	900
Fuel lattice, fuel rods per assembly	17 × 17, 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 (FH)	1.528
Design point-wise peaking (FQ)	2.5
Core loading (tHM)	81.6
Target cycle length (GWd/tHM) (months)	21.564 (18)
Capacity factor (%)	90
Target effective full power days	493
Target discharge burn-up (GWd/tHM)	40.0-50.0
Maximum pin burn-up (GWd/tHM)	62
Shutdown margin (SDM) (%Δρ)	1.3

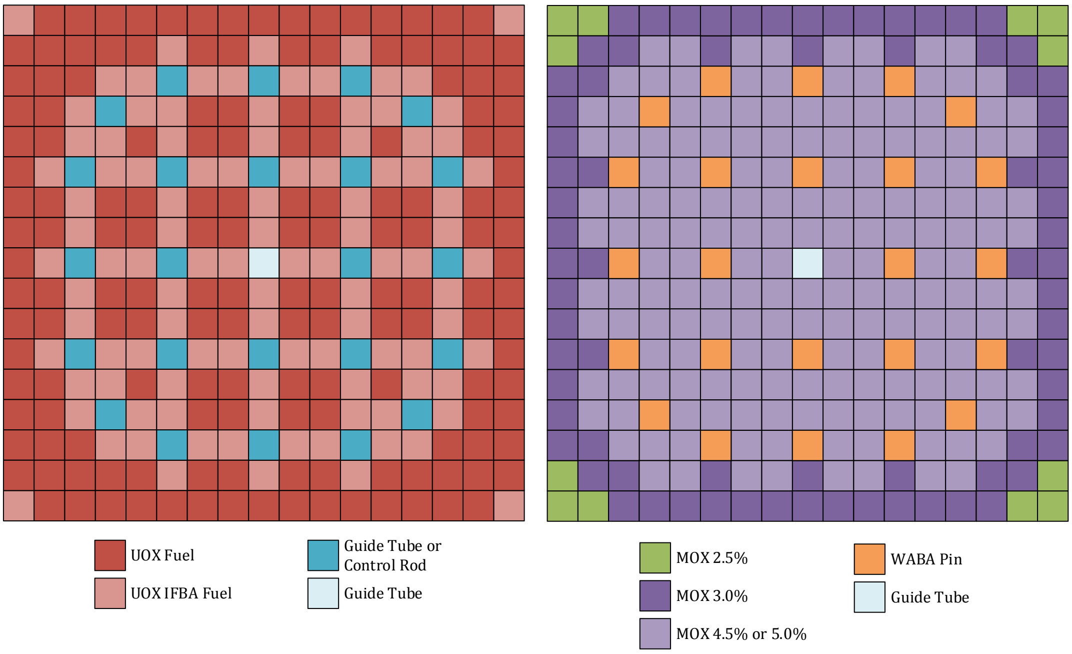

Fig. 2. UO2 fuel assembly with 104 IFBA and MOX fuel assembly with 24 WABA[12]

Table 2. Heavy Metal Composition in Fuel[12]

Assembly Type	Density [g/cm3]	HM Material	
UO2 4.2%	10.24	U-235: 4.2 wt%, U-238: 95.8 wt%	
UO2 4.5%	10.24	U-235: 4.5 wt%, U-238: 95.5 wt%	
		Pu-fissile (wt%)	
MOX 4.0%	10.41	Corner zone: 2.5 wt% Peripheral zone: 3.0 wt% Central zone: 4.5 wt%	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%
MOX 4.3%	10.41	Corner zone: 2.5 wt% Peripheral zone: 3.0 wt% Central zone: 5.0 wt%	

3. METHODOLOGY

The flowchart of this study can be seen in Fig. 3. Macroscopic cross-sections and group constants for fuel assembly are generated using the PIJ module from SRAC 2006 using material composition provided by Purdue University[16]. The PIJ module is based on the neutron transport theory by using collision probability method developed at JAEA[11]. In this study, neutron energy is condensed from 107 to 2 energy groups (59 fast, 48 thermal) using ENDF/B-VII cross-section data. Then the cross-section data and group constants for calculations on the reactor core are used to model the reactor core using SRAC-CITATION (2D).

Core modeling in SRAC-CITATION was carried out using baffles (2.52 cm) as in Figure 4. In 2D modeling in SRAC-CITATION, 10 mesh was used in X and Y directions of each assembly zone (21.42 cm) at full core and ¼ core models. Full core and ¼ core models is used for calculation of all control rods out (ARO) and all control rods in (ARI) using SRAC-CITATION to calculate total rod worth, while the calculation of single rod worth at ARO and ARI conditions is done using only full core model. To achieve relative flux change for the last iteration that lower than 10^{-8}, all calculations were performed with a maximum number of iterations of CITATION, which is 999 iterations. with hot zero power conditions, at a fuel temperature of 560 K, a moderator density of 752.06 kg/m3 (560 K), and a 1000 ppm boron concentration.

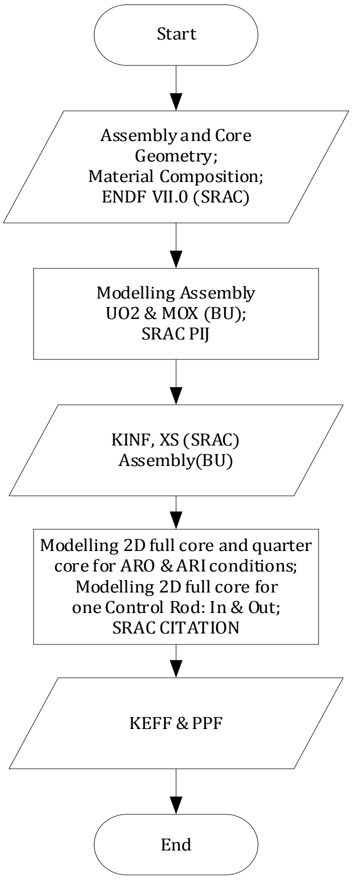

Fig 3. Calculation flowchart

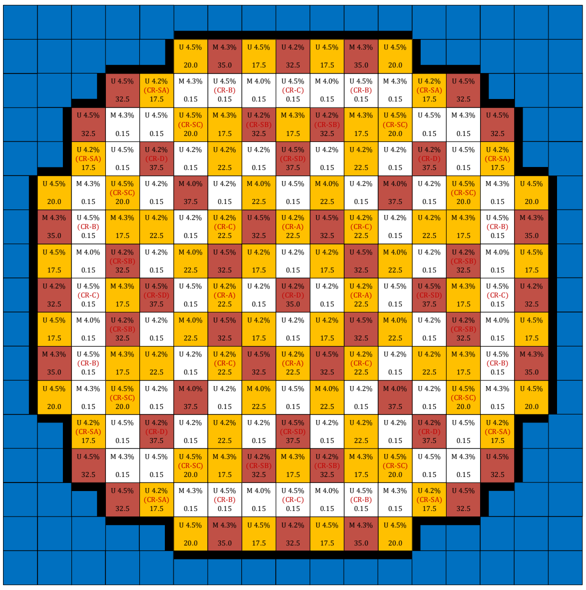

Fig. 4. SRAC CITATION reactor core model using 2.52 cm baffle

4. RESULTS AND DISCUSSION

The effective multiplication factor (K_{eff}) calculated by SRAC-CITATION at all rods out (ARO) and all rods in (ARI) conditions are presented in Table 3. Total rod worth are also presented in Table 3 and this results are then compared with the high-order heterogeneous multi-group transport DeCART as a reference. It also displays the calculation results of PARCS code that using nodal solutions. Total rod worth calculated by SRAC-CITATION show a difference of 0.339% in full core model and 0.337% in ¼ core model from DeCART total rod worth. The calculation results of the PARCS code itself compared to the reference have a difference of 0.720%. Based on these results, calculation of total rod worth using the SRAC code system is close to the reference data used.

Table 3. K_{eff} and total control rod worth

Code	K_{eff} ARO	K_{eff} ARI	Total Rod Worth (pcm)	error (%)
SRAC-CITATION 2D				
Full Core	1.060337	0.988785	6825	0.339
¼ Core	1.060331	0.988782	6824	0.337
PARCS 2G	1.063786	0.991536	6850	0.710
DeCART	1.058520	0.987430	6801	-

$$total\ rod\ worth = \frac{Keff_{ARO} - 1}{Keff_{ARO}} - \frac{Keff_{ARI} - 1}{Keff_{ARI}}$$

$$total\ rod\ worth\ error(X) = \left|\frac{(X - DeCART)}{DeCART}\right|$$

Calculation results of SRAC-CITATION 2D full core in single rod worth at ARO and ARI conditions are shown in Tables 4 and 5, respectively. The SRAC calculation results in this case are compared with the BARS code (heterogeneous) and PARCS code because the DeCART code does not gives results for single rod calculations. Rod worth value at ARO from SRAC is very close to the results of nodal solutions PARCS and heterogeneous solutions BARS. The difference is still below 7% with the highest difference to BARS occurring at (C,3) position of 6.34%, while that of PARCS is 4.74% at (A,1).

In case of row worth at ARI condition, SRAC result compared to the BARS is not as good as in the ARO condition where there is a difference of up to 33.24% at position (E,7), 26.55% at position (A,7), and 14.99% at position (C,7), while it is still below 10% at other positions. The same thing happened as SRAC result compared to PARCS, however it is relatively better with a maximum difference of 9.72% at position (E,7) followed by 5.84% at position (A,1), whereas at other positions, the difference was much smaller.

Table 4. Rod worth at ARO (pcm)

Code	Control rod position									
	(A,1)	(A,3)	(A,5)	(A,7)	(B,6)	(C,3)	(C,7)	(D,6)	(E,5)	(E,7)
SRAC	174	146	91	51	68	124	50	68	66	28
PARCS 2G	166	143	91	53	70	123	51	68	64	27
BARS	166	139	87	49	66	117	49	66	63	27

Table 5. Rod worth at ARI (pcm)

Code	Control rod position									
	(A,1)	(A,3)	(A,5)	(A,7)	(B,6)	(C,3)	(C,7)	(D,6)	(E,5)	(E,7)
SRAC	-889	-887	-408	-56	-149	-1122	-78	-289	-260	-23
PARCS 2G	-840	-880	-405	-56	-152	-1127	-78	-290	-249	-21
BARS	-914	-921	-417	-44	-145	-1193	-68	-313	-268	-17

The radial power peaking factor is a core parameter that shows the heat generation density and distribution, which shows the ratio of the highest heat generation in fuel assembly with average heat generation of the entire core in radial direction. Normalized radial power distribution for ARO and ARI conditions are shown in Figures 5 and 6. The first row is the result of core calculation with SRAC-CITATION 2D, the second row is from DeCART and the third row is the absolute deviation of SRAC from DeCART.

Figure 5 shows that the calculation of SRAC-CITATION under ARO conditions is quite close to DeCART. The biggest difference is in the (H,4) of 5.325%, this is due to the interaction with MOX fuel assembly that surrounds (H,4) and its position near the baffle and radial reflector. From the safety aspect this has no effect because in that assembly, peak radial power factor is very low. The thing to note is the assembly with a large radial power factor such as at position (A,2) and (B,1) that reached 1.783 but the difference is only reaching 2.749% from DeCART.

Calculation of radial power distribution in ARI condition shows good results when compared to reference. The biggest difference between the calculation of SRAC and DeCART is 6,234% at position (G,5). This position is surrounded by a UO2 fuel assembly with low burnup fraction at (G,4) and (F,5), but burnup fraction at (G,6) is high enough, so SRAC-CITATION with it’s maximum iteration (999) could only achieve such a relatively high power fraction at (G,5).

	1	2	3	4	5	6	7	8
A	1.338 1.374 2.611	1.783 1.735 2.741	1.390 1.418 1.968	1.563 1.525 2.485	1.003 1.035 3.049	0.996 1.032 3.501	1.014 0.997 1.664	0.406 0.413 1.650
B	1.783 1.735 2.749	1.543 1.563 1.295	1.210 1.245 2.850	1.233 1.277 3.468	1.383 1.349 2.524	0.887 0.918 3.369	0.953 0.978 2.568	0.490 0.491 0.243
C	1.391 1.418 1.930	1.210 1.245 2.821	1.298 1.325 2.056	1.484 1.446 2.631	1.228 1.247 1.538	1.084 1.114 2.688	1.014 0.991 2.284	0.383 0.393 2.571
D	1.564 1.525 2.572	1.234 1.277 3.395	1.485 1.446 2.682	1.036 1.076 3.699	1.356 1.308 3.671	1.137 1.143 0.546	0.886 0.892 0.709	0.356 0.341 4.364
E	1.005 1.035 2.903	1.385 1.348 2.746	1.229 1.247 1.423	1.357 1.308 3.747	0.892 0.904 1.372	1.116 1.067 4.571	0.604 0.585 3.226	
F	0.999 1.032 3.243	0.889 0.917 3.020	1.086 1.114 2.484	1.139 1.142 0.304	1.117 1.067 4.669	0.759 0.754 0.680	0.294 0.281 4.749	
G	1.018 0.997 2.096	0.957 0.978 2.173	1.017 0.991 2.637	0.888 0.892 0.441	0.605 0.585 3.453	0.295 0.281 5.022		
H	0.410 0.413 0.808	0.494 0.491 0.598	0.386 0.393 1.850	0.358 0.340 5.325			SRAC-CIT DeCART %abs deviation	

Fig. 5. Normalized radial power distribution at ARO condition

	1	2	3	4	5	6	7	8
A	1.176 1.209 2.768	2.521 2.533 0.456	1.196 1.202 0.461	2.229 2.196 1.492	0.743 0.742 0.113	0.638 0.669 4.668	0.313 0.300 4.223	0.200 0.205 2.264
B	2.521 2.533 0.455	2.369 2.459 3.641	1.726 1.812 4.763	2.034 2.103 3.274	1.886 1.832 2.937	0.454 0.449 1.167	0.472 0.489 3.522	0.267 0.268 0.385
C	1.197 1.202 0.456	1.726 1.812 4.758	1.212 1.198 1.161	2.549 2.452 3.969	1.938 1.944 0.319	0.950 0.985 3.582	0.347 0.329 5.501	0.194 0.198 2.091
D	2.229 2.196 1.505	2.034 2.103 3.263	2.550 2.452 3.980	1.776 1.823 2.577	1.735 1.675 3.556	0.547 0.531 3.017	0.442 0.450 1.866	0.194 0.186 4.099
E	0.743 0.742 0.134	1.886 1.832 2.958	1.938 1.944 0.301	1.735 1.675 3.569	0.510 0.508 0.339	0.712 0.696 2.239	0.202 0.190 6.095	
F	0.638 0.669 4.594	0.455 0.449 1.233	0.950 0.985 3.536	0.547 0.531 3.057	0.712 0.696 2.266	0.559 0.562 0.588	0.192 0.186 3.439	
G	0.313 0.300 4.498	0.473 0.489 3.271	0.348 0.329 5.706	0.442 0.450 1.712	0.202 0.190 6.234	0.193 0.186 3.628		
H	0.202 0.205 1.421	0.269 0.268 0.426	0.195 0.198 1.379	0.195 0.186 4.697			SRAC-CIT DeCART %abs deviation	

Fig. 6. Normalized radial power distribution at ARI condition

In general, the reason behind the differences in results of assembly power density of SRAC and DeCART is the same as differences of total rod worth and single rod worth at ARO ARI conditions before. It is caused by combination of different nuclide cross-section library and different methods (ie, diffusion or transport, number of energy groups, etc.) with its convergence criterion. However, the differences in calculation results obtained by SRAC (PIJ & CITATION) are not too large compared with references.

5. CONCLUSION

Verification of SRAC2006 code system for cross-section generation by SRAC-PIJ and calculation of core neutronic parameters by SRAC-CITATION have been carried out. Calculation results show very good agreement with the DeCART and BARS as a reference heterogeneous solutions code, and PARCS as a nodal solutions code. For this reason, the SRAC2006 system can be used to thoroughly evaluate the safety aspects of a PWR MOX/UO2 type NPP such as the AP1000. In the next study, SRAC2006 will be used to determine the derivative constant to evaluate the transient cases of PWR MOX/UO2 Core Transient Benchmark in collaboration with NODAL3 program.

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AUTHOR CONTRIBUTION

Wahid Luthfi carried out assembly and core modelling in SRAC2006 and Surian Pinem participated as reviewer and data analysis. Wahid Luthfi is the lead author of this paper and Surian Pinem as co-author. All authors read and approved the final version of the manuscript.

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