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CANDU-PHWR Core Analysis by the McCARD/SCAN Code System

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Abstract

Recently, a two-step CANDU-PHWR core analysis system, McCARD/SCAN, in which two-group homogenized and incremental cross sections are generated by the B_1 theory augmented Monte Carlo (MC) method has been successfully applied for hypothetical CANDU 6 core analysis problems. In this study, the developed McCARD/SCAN code system is used for the initial core analysis of Wolsong Nuclear Power Plant Unit 2. The effective multiplication factor k_{eff} and normalized channel power distribution estimated by McCARD/SCAN is compared with those from the MC whole core calculations. In addition, the reactivity worth of each adjuster rod bank calculated by McCARD/SCAN are compared with the physics measurement data and the McCARD whole core calculation results.

Keywords: Monte Carlo, CANDU-PHWR, few-group constant generation, McCARD/SCAN

1. Introduction

A B_1 theory augmented Monte Carlo (MC) homogenized few-group constant generation method [1,2] (B_1 MC method hereafter) has been proposed as an alternative way to generate homogenized few-group constants of nuclear fuel systems like fuel pins or fuel assemblies or bundles by deterministic fuel assembly spectrum codes like CASMO [3], HELIOS [4], WIMS-AECL [5], etc. The applicability of the B_1 MC method to PWR core analyses has been demonstrated by showing that few-group constants from the method implemented in a Seoul National University (SNU) MC code, McCARD [6] lead to nodal core neutronics calculations in a good agreement with whole PWR core reference MC calculations [2].

Recently a two-step procedure for the CANDU-PHWR core analysis coupling the McCARD few-group constant generation capability and a SNU diffusion theory code, SCAN [7], has been successfully applied for hypothetical CANDU 6 core analysis problems [8]. This study extends its validation to actual CANDU 6 initial core problems using the physics measurement data of Wolsong Nuclear Power Plant Unit 2 (Wolsong-2) [9]. Measurements of the reactivity worth of adjuster rods (ADJ) in seven banks are compared with those from the McCARD/SCAN code system and McCARD whole core transport calculations. The multiplication factor and normalized power distribution calculated by McCARD/SCAN are compared with those from the McCARD whole core calculations.

2. CANDU 6 Initial Core Analysis Problem

The CANDU 6 reactor is composed of the reactor core, the D_2O reflector, and the stainless steel Calandria reactor vessel as shown in Figure 1.

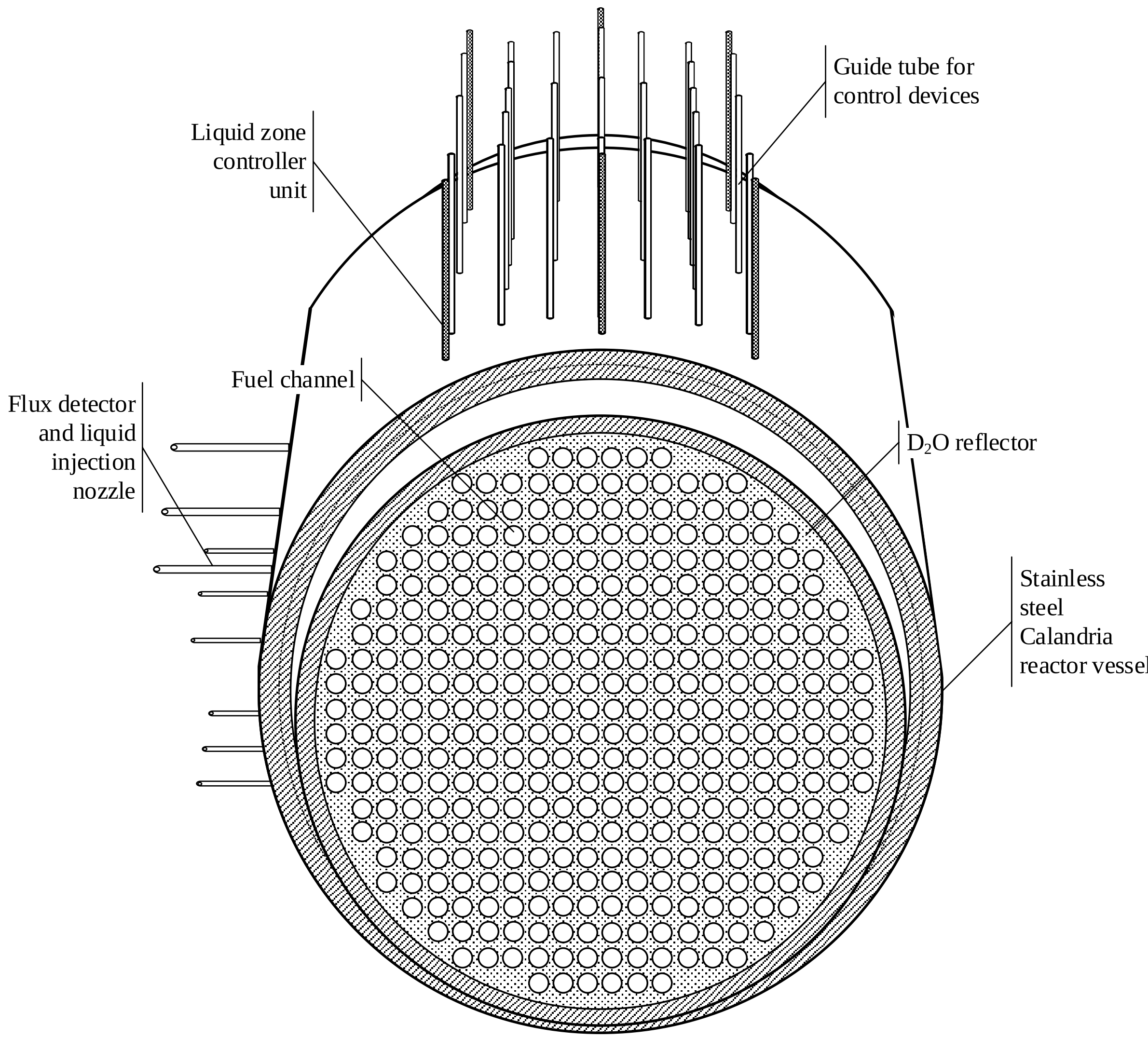

Figure 1 Over view of the CANDU 6 reactor

The Calandria is a horizontal cylindrical vessel which envelopes 380 fuel channels comprising the core and contains heavy water moderator and reflector. As shown in Figure 2, eighty out of the 380 fuel channels are depleted ones while the rest are fresh ones. Each fuel channel consists of 12 fuel bundles aligned horizontally inside the pressure tube. Depleted fuel channels contain 2 depleted fuel bundles each, which are positioned at the 4^{th} and the 5^{th} sites of the 12 fuel bundle sites in the order from the front or the end of the fuel channel while all the fresh fuel channels comprise the fresh fuel bundles. The depleted fuel channels are arranged bi-directionally in the centre region of the core so that none of two adjacent depleted fuel channels are aligned in the same direction.

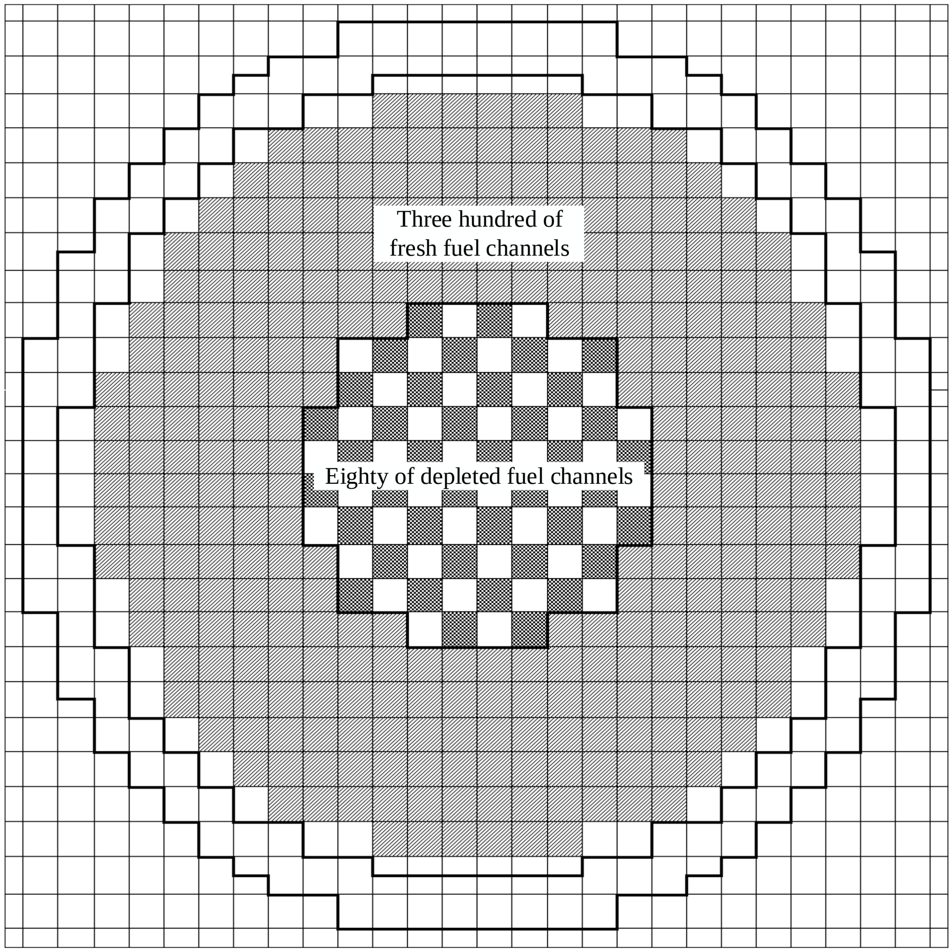

Figure 2 Positions of the depleted fuel channels

All the fresh fuel bundles and 160 depleted fuel bundles were loaded into 380 fuel channels. There are two major reactivity devices - the liquid zone controllers (LZCs) and ADJ - loaded inside. The light water level of each LZC is set to 35.6%, while all mechanical absorbers are fully withdrawn. Regional temperatures are 308.15 K at fuel and coolant, 304.15 K at moderator, and 293.15 K at structural material. The boron concentration in moderator is 8.945 ppm.

3. Numerical Results

3.1 Standard unit lattice cell & Supercell calculation

The two-step deterministic solutions to the CANDU 6 core analysis problems require specifying the two-group constants for every 3-dimensional node. The required two-group constants for the initial core condition are calculated through the standard unit cell and the supercell model. Figures 3 and 4 display geometry models of the two cells, respectively.

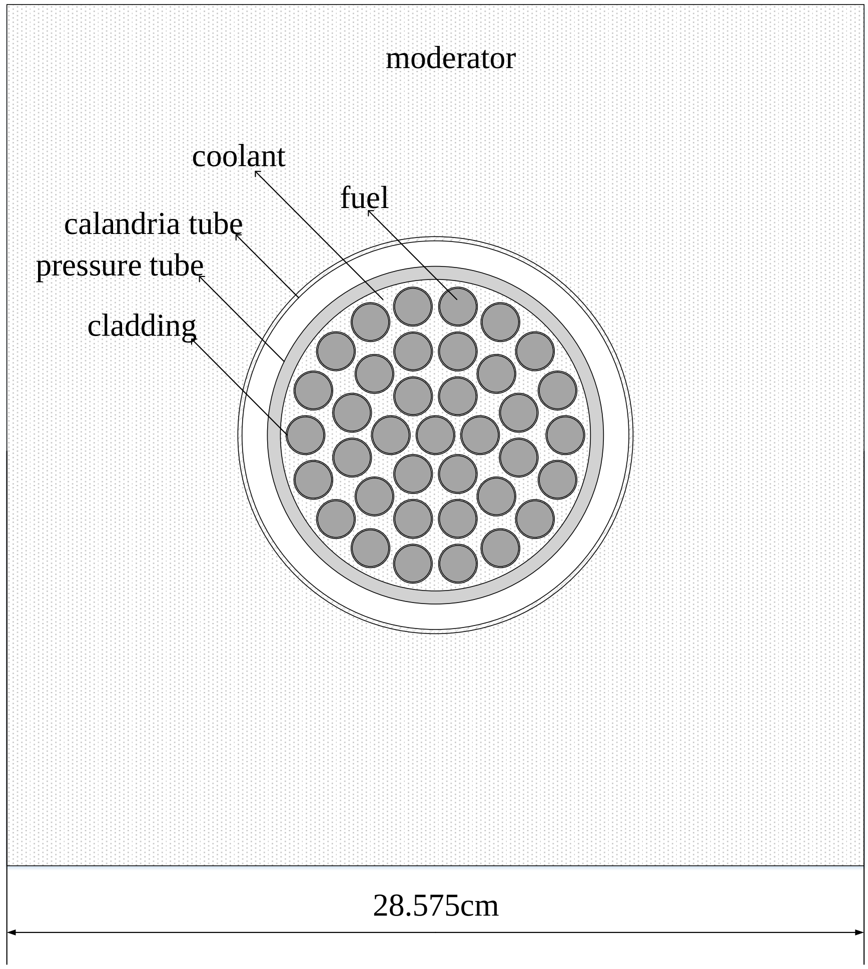

Figure 3 Standard lattice cell model

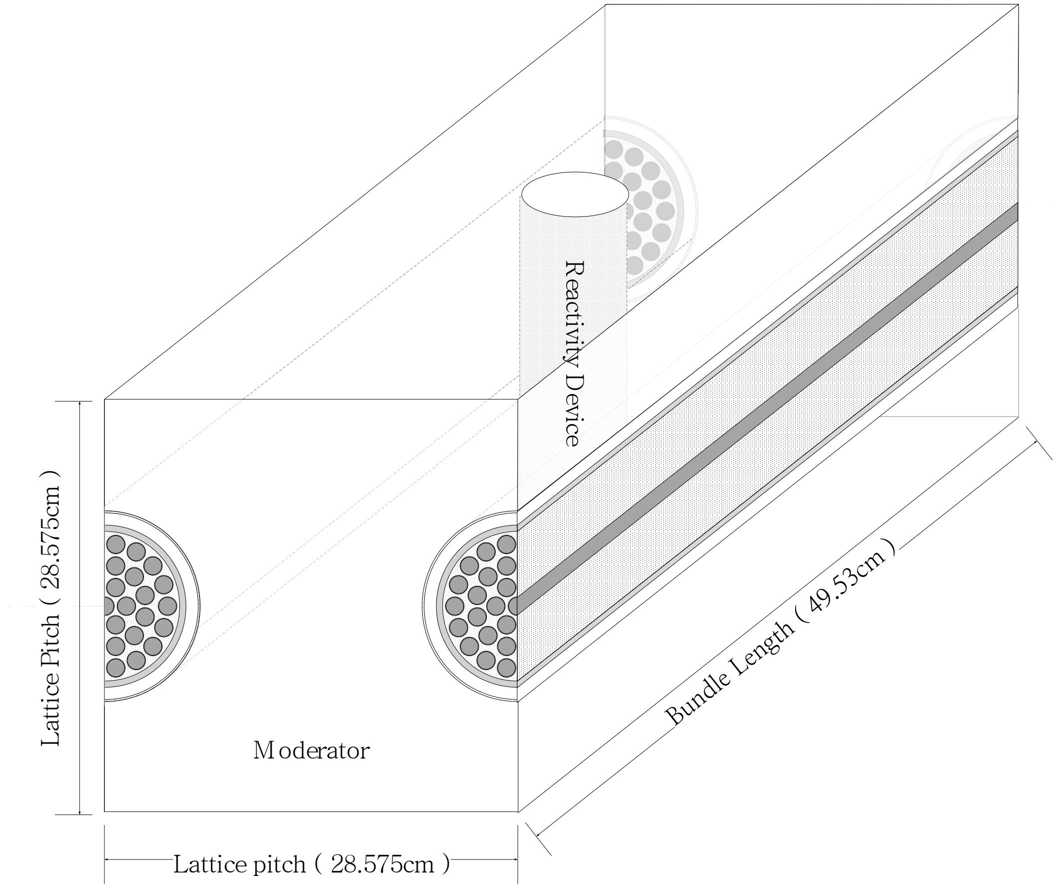

Figure 4 McCARD supercell model

The two sets of two-group constants for natural and depleted uranium fuel are generated through the standard lattice cell model by McCARD B_1 MC method based calculation. The incremental cross section for LZCs, ADJ, and all reactivity devices' guide tubes are generated in the same method through the supercell model with natural uranium fuel. Both calculations are done on the initial core conditions and their results are shown in Table 1, 2, 3, and 4.

Table 1 The two-group constants set generated by McCARD for the natural uranium fuel bundle and depleted fuel bundle

Two Group[1)] Constants	Natural uranium fuel	Depleted uranium fuel
Σ_{a1}	1.64×10^{-3}	1.61×10^{-3}
Σ_{a2}	4.06×10^{-3}	3.76×10^{-3}
$\nu\Sigma_{f1}$	8.90×10^{-4}	8.06×10^{-4}
$\nu\Sigma_{f2}$	4.66×10^{-3}	3.73×10^{-3}

Σ_{s12}	8.94×10^{-3}	9.18×10^{-3}
Σ_{s21}	4.68×10^{-5}	3.96×10^{-5}
D_1	1.32	1.32
D_2	0.83	0.83

Table 2 Incremental cross sections of liquid zone controllers

	Device free		Device type					
			AIR-1	H_2O-1	AIR-2	H_2O-2	AIR-3	H_2O-3
Σ_{tr1}	$2.5\text{x}10^{-01}$	$\Delta\Sigma_{tr1}$	$-1.2\text{x}10^{-02}$	$1.5\text{x}10^{-02}$	$-1.5\text{x}10^{-02}$	$1.7\text{x}10^{-02}$	$-1.0\text{x}10^{-02}$	$1.3\text{x}10^{-02}$
Σ_{tr2}	$4.0\text{x}10^{-01}$	$\Delta\Sigma_{tr2}$	$-1.7\text{x}10^{-02}$	$1.3\text{x}10^{-01}$	$-2.7\text{x}10^{-02}$	$1.5\text{x}10^{-01}$	$-7.3\text{x}10^{-03}$	$1.2\text{x}10^{-01}$
Σ_{a1}	$1.6\text{x}10^{-03}$	$\Delta\Sigma_{a1}$	$-7.5\text{x}10^{-06}$	$7.9\text{x}10^{-05}$	$-1.7\text{x}10^{-05}$	$9.0\text{x}10^{-05}$	$2.4\text{x}10^{-06}$	$6.8\text{x}10^{-05}$
Σ_{a2}	$4.1\text{x}10^{-03}$	$\Delta\Sigma_{a2}$	$1.6\text{x}10^{-04}$	$1.1\text{x}10^{-03}$	$7.7\text{x}10^{-05}$	$1.2\text{x}10^{-03}$	$2.5\text{x}10^{-04}$	$9.9\text{x}10^{-04}$
$\nu\Sigma_{f1}$	$8.9\text{x}10^{-04}$	$\Delta\nu\Sigma_{f1}$	$-2.1\text{x}10^{-05}$	$7.2\text{x}10^{-05}$	$-3.3\text{x}10^{-05}$	$8.5\text{x}10^{-05}$	$-1.0\text{x}10^{-05}$	$6.1\text{x}10^{-05}$
$\nu\Sigma_{f2}$	$4.7\text{x}10^{-03}$	$\Delta\nu\Sigma_{f2}$	$3.5\text{x}10^{-05}$	$-9.7\text{x}10^{-06}$	$3.7\text{x}10^{-05}$	$-7.3\text{x}10^{-06}$	$3.3\text{x}10^{-05}$	$-1.1\text{x}10^{-05}$
Σ_{s12}	$8.9\text{x}10^{-03}$	$\Delta\Sigma_{s12}$	$-3.0\text{x}10^{-04}$	$2.0\text{x}10^{-03}$	$-5.8\text{x}10^{-04}$	$2.3\text{x}10^{-03}$	$-5.5\text{x}10^{-05}$	$1.7\text{x}10^{-03}$
Σ_{s21}	$4.7\text{x}10^{-05}$	$\Delta\Sigma_{s21}$	$3.9\text{x}10^{-08}$	$-1.5\text{x}10^{-06}$	$5.7\text{x}10^{-07}$	$-2.1\text{x}10^{-06}$	$-5.5\text{x}10^{-07}$	$-1.2\text{x}10^{-06}$

3.2 Core calculation

The two-group constants and incremental cross sections are used for the core neutronics analysis of the CANDU 6 core to validate their qualification for the CANDU-PHWR neutronics design calculations by the SCAN code. The finite-difference method option in SCAN and the 84x68x40 fine mesh model are used for the whole core two-group diffusion theory calculation. Reference solutions are obtained by the continuous energy McCARD whole core transport calculations with 1200 cycles including 200 inactive cycles with 1,000,000 histories per cycle. The calculated effective multiplication factors are 1.00085 and 0.99999 ± 0.00002 for McCARD/SCAN and McCARD, respectively. It is noted that the two calculations have 86 pcm difference in k_{eff}.

Figure 5 shows a comparison of the channel power distributions from McCARD/SCAN and McCARD calculations. For the comparison, the full core results from McCARD code are folded into the 1/4 core model. From this figure, one can see that the channel power distribution from McCARD/SCAN two-step calculations agree very well with those from the reference McCARD calculations. The root mean square (RMS) difference of the SCAN predictions to the reference McCARD calculation is 0.61%. The maximum channel power difference is 1.37%.

0.989	1.010	1.050	1.090	1.150	1.250	1.260	1.230	1.090	0.870	0.600
0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.05
-0.11	-0.01	0.11	0.18	0.11	0.09	0.13	0.01	-0.19	-0.52	-0.58
1.010	1.030	1.060	1.100	1.160	1.250	1.250	1.210	1.060	0.837	0.568
0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.05
0.07	0.09	0.18	0.18	0.09	0.02	0.00	-0.01	-0.26	-0.66	-0.64
1.050	1.060	1.090	1.140	1.230	1.260	1.240	1.160	1.000	0.769	0.511
0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.05
0.25	0.24	0.28	0.27	0.04	-0.09	-0.03	-0.04	-0.23	-0.45	-0.12
1.100	1.100	1.130	1.180	1.260	1.260	1.200	1.090	0.910	0.679	
0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.04	
0.35	0.34	0.35	0.14	0.05	-0.10	-0.05	-0.02	-0.27	0.00	
1.140	1.150	1.220	1.260	1.260	1.220	1.130	0.992	0.791	0.562	
0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.04	0.05	
0.45	0.49	0.43	0.11	0.03	-0.03	-0.16	-0.22	-0.15	0.54	
1.230	1.240	1.250	1.260	1.220	1.140	1.020	0.862	0.657		
0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.04		
0.58	0.46	0.50	0.19	0.10	-0.09	-0.35	-0.42	0.07		
1.240	1.260	1.260	1.220	1.150	1.030	0.883	0.706	0.504		
0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.04	0.05		
0.36	0.22	0.17	-0.04	-0.24	-0.41	-0.80	-0.69	0.35		
1.180	1.190	1.170	1.110	1.020	0.879	0.713	0.539			
0.03	0.03	0.03	0.03	0.03	0.04	0.04	0.05			
0.41	0.27	0.20	0.05	-0.30	-0.61	-0.72	-0.13			
1.040	1.040	0.999	0.924	0.818	0.677	0.525				
0.03	0.03	0.03	0.04	0.04	0.04	0.05				
0.07	-0.12	-0.23	-0.28	-0.67	-0.90	-0.33				
0.829	0.813	0.761	0.687	0.579	0.455					
0.04	0.04	0.04	0.04	0.05	0.05					
-0.49	-0.49	-0.11	0.05	-0.31	-0.33					
0.572	0.551	0.503								
0.05	0.05	0.05								
0.18	0.06	1.37								

P_{Ref}
Rel. S.D. (%)
Diff. (%)

P_{Ref} = McCARD power
Rel. S.D. (%) = Relative Standard Deviation of McCARD power
P_{SCAN}= McCARD/SCAN power   Diff. (%) = $(P_{SCAN} - P_{Ref}) / P_{Ref} \times 100$

Figure 5 Comparison of normalized channel power distribution

Table 4 compares McCARD/SCAN and McCARD predictions with measurements of reactivity worth of the ADJ Bank. It is shown that the two methods predict similarly but that under-predict the reactivity worth of all the ADJ banks in comparison with the measured worth for them. The maximum relative differences of the McCARD/SCAN and McCARD calculations to the measured reactivity worth are observed 20 % and about 19 %, respectively, in the ADJ bank numbered 7. In this conjunction, it must be noted that the measurement data uncertainty of the reactivity device worth is reportedly $\pm 15.0\%$ on average.

Table 4 Reactivity worth of ADJ bank

			McCARD/SCAN		McCARD	
ADJ BANK Number	ADJ Rod Number	Measured reactivity (mk)	Calculated reactivity (mk)	Error (%)	Calculated reactivity (mk)	Error (%)
1	1,7,11,15,21	1.36	1.091	-19.8	1.147 ± 0.023	-15.6
2	2,6,18	1.53	1.470	-3.9	1.495 ± 0.025	-2.3
3	4,16,20	1.51	1.451	-3.9	1.430 ± 0.028	-5.3
4	8,9,13,14	2.33	2.091	-10.3	2.148 ± 0.028	-7.8
5	3,19	1.77	1.483	-16.2	1.469 ± 0.028	-17.0
6	5,17	1.79	1.511	-15.6	1.494 ± 0.028	-16.5
7	10,12	3.37	2.696	-20.0	2.722 ± 0.028	-19.2
Total		13.66	11.793	-13.7	11.906 ± 0.019	-12.8

4. Conclusion

The neutronics analyses for the Wolsong-2 initial core are conducted by the McCARD/SCAN code system. The application results show that the B_1 MC method generated two-group constants lead to the core analysis in a good agreement with the whole core reference McCARD calculations. It is worthy to note that the reactivity worth values of ADJ banks calculated by McCARD/SCAN agree with the physics measurements within their uncertainty range. These results merit further validation studies on the B_1 MC method in terms of the Wolsong-2 physics measurement data available. They will be reported in near future.

5. References

[1]. H. J. Shim, J. Y. Cho, J. S. Song, C. H. Kim, "Generation of Few Group Diffusion Theory Constants by Monte Carlo Code," *Trans. Am. Nucl*. Soc., **99**, 343 (2008).

[2]. H. J. Park, H. J. Shim, H. G. Joo, C. H. Kim, "Generation of Few-Group Diffusion Theory Constants by Monte Carlo Code McCARD," *Nucl. Sci. Eng*., **172**, 66 (2012).

[3]. M. Edenius, B. H. Forrsen, "CASMO-3 A Fuel Assembly Burnup Program User's Manual," Studsvik/NFA-89-3, Rev. 2, Studsvik AB (1992).

[4]. R. J. J. Stamm'ler, "HELIOS Methods," Studsvik Scanpower (2002).

[5]. J.D. Irish and S.R. Douglas, "Validation of WIMS-IST", *Proc. Of 23rd Annual Conference of Canadian Nuclear Society*, Toronto, Canada, June 2-5 (2002).

[6]. H. J. Shim, B. S. Han, J. S. Jung, H. J. Park, C. H. Kim, "McCARD: Monte Carlo Code for Advanced Reactor Design and Analysis," *Nucl. Eng. Technol*., 44[2], 161 (2012).

[7]. I. S. Hong, C. H. Kim, B. J. Min, and H. C. Suk, "Development of CANDU-PHWR neutronics code SCAN," in Proceedings of the 7th International CANDU Fuel Conference, vol. 1, pp. 77–86, Canadian Nuclear Society, Kingston, Canada, September 2001.

[8]. Seung Yeol Yoo, Hyung Jin Shim, and Chang Hyo Kim, "Monte Carlo Few-Group Constant Generation for CANDU 6 Core Analysis," *Science and Technology of Nuclear Installations*, vol. 2015, Article ID 284642, 11 pages, 2015. doi:10.1155/2015/284642.

[9]. Hangbok Choi, Gyuhong Roh, and Donghwan Park. "Benchmarking MCNP and WIMS/RFSP Against Measurement Data-II: Wolsong Nuclear Power Plant 2," *Nuclear Science and Engineering*, 150, 37 (2005).