VERA Solutions by UNIST MC Code MCS

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1. Introduction

In reactor physics, two widely known numerical methods are used to solve neutron transport equations: deterministic method [1] and the Monte Carlo (MC) method [2]. The MC method is useful for solving complicated three-dimension (3D) problems. This method is especially suitable for complicated geometries which cannot be simulated explicitly by using deterministic code systems. However, it requires much larger computation time than the deterministic method to achieve a small statistical uncertainty. Fortunately, the immense development of computer science and technology permits applying the MC method in reactor core simulations and analyses.

To improve the accuracy of reactor analysis tools for neutron-physics simulation, Ulsan National Institute of Science and Technology (UNIST) has been developing the MC code named MCS [3-6] to achieve a high accurate whole-core solution. Various types of verification and validation (V&V) work have been conducted on MCS to ensure its accuracy and reliability for nuclear power system design and safety evaluation. The primary goal of this work is to verify MCS using the VERA Core Physics Benchmark problems [5], which is directed towards the results of 3D multi-assembly and whole-core simulations.

2. Benchmark Specification

The Consortium for Advanced Simulation of LWRs (CASL) provided the detailed specifications for the VERA Core Physics Benchmark Progression in 2014 including 10 problems based on the actual fuel and plant data of the Watts Bar Nuclear 1 (WBN1) initial startup core [7]. This information is for nuclear industry software/method developers and analysts to model U.S. nuclear power reactors and their operations. Problems 4, 5 and 9 were chosen for analysis in this paper. Fig. 1 (left and right) illustrates the radial 3x3 assembly and quarter core layout of fuel assemblies and poison configurations employed in Problems 4 and 5, respectively.

3. Monte Carlo Codes

Two MC codes were used for this study - one used for calculation and the other used as reference.

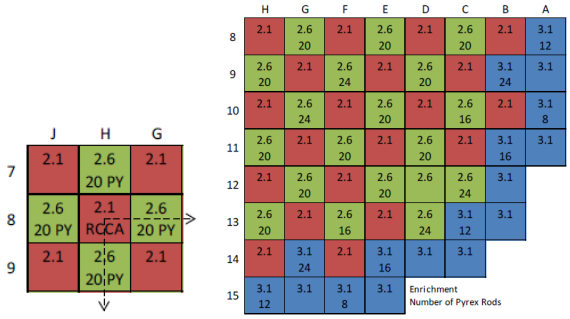

Fig. 1. Radial configurations: 3x3 assembly, poison and control layout (left) and VERA core fuel and poison loading pattern (quarter symmetry- right).

The first MC code is the UNIST inhouse code MCS. This code is a 3D continuous-energy neutron-physics code for particle transport based on the MC method, under development at UNIST [3, 4]. Two kinds of calculations are allowed by MCS: criticality runs for reactivity calculations and fixed-source runs for shielding problems. MCS neutron transport capability is validated and verified with the International Criticality Safety Benchmark Experimental Problem (ICBEP), and Jordan Research and Training Reactor (JRTR). MCS is capable of whole-core simulation with pin-wise depletion and thermal-hydraulics feedback, and it is validated against the solution of BEAVRS cycle 1 [5, 6].

The second MC code is the KENO-VI code from the SCALE 4.4 System for "Standardized Computer Analysis for Licensing Evaluation" [8] developed for the U.S. Nuclear Regulatory Commission (NRC) to enable standardized analyses and evaluation of nuclear facilities.

4. Numerical Results

MCS utilizes a continuous cross-section library ENDF/B-VII.0 at exactly 564.45K for all simulations. Those simulations were executed on a Linux cluster with 40 processes (Intel Xeon E5-2620 @ 3.00 GHz). A total of 400 million histories were run with 10 inactive cycles, 40 active cycles, 20,000 histories per sub-cycle and 400 sub-cycles.

Solutions computed by KENO-VI for VERA benchmark problems are taken from the benchmark document and used as a reference for verification.

4.1. 3D 3x3 assembly control rod worth

This problem is based on the 3D assembly problem by the addition of multiple assemblies and rod cluster control assemblies (RCCA). Its solutions are to show the ability to predict the eigenvalue (k_{eff}) and power distribution without thermal-hydraulic feedback or depletion in the presence of black neutron absorbers. The control rod reactivity worth, related to the movement of RCCA, is commonly used for the verification of nuclear numerical methods. The differential rod worth (DRW) and integral rod worth (IRW) were also obtained by MCS and compared to the reference results. The MCS numerical results compared to KENO-VI results, are summarized in Table I. It can clearly be seen that MCS solutions are consistent with KENO-VI reference results since the largest difference of k_{eff} is 34 pcm. In addition, the maximum DRW and IRW discrepancies are only 11 and 14 pcm, respectively at the 20% RCCA withdrawal. The corresponding RCCA DRW and IRW are also compared between MCS and KENO-VI in Figs. 2 and 3. The normalized radial and axial power profiles are shown in Figs. 4 through 6. Due to the large number of particle histories, the radial and axial power uncertainties of the MCS code are less than 0.6%. The power profiles of MCS also show good agreement with KENO-VI solutions because the relative differences are less than 0.05% for radial distribution and 4% for axial distribution.

Table I: Solution Results for a 3D 3x3 Assembly Control Rod Worth

RCCA % Withdrawn	Diff. (MCS-KENO)		
	k_{eff} [± 3 pcm]	DRW [± 4 pcm]	IRW [± 4 pcm]
257.9 cm	-23	--	2
0	-34	-8	-11
10	-27	10	-2
20	-37	-11	-14
30	-28	5	-3
40	-33	-7	-8
50	-26	6	-1
60	-32	-6	-6
70	-25	2	0
80	-27	-2	-2
90	-26	0	0
100	-25	--	--

4.2. Neutronic performance at zero power condition

The work presented consists of a whole core of Westinghouse 17x17-type fuel assemblies in the WBN1 initial loading pattern. The solutions by MCS are the eigenvalue k_{eff} and core power profiles at zero power condition without thermal-hydraulic feedback or depletion.

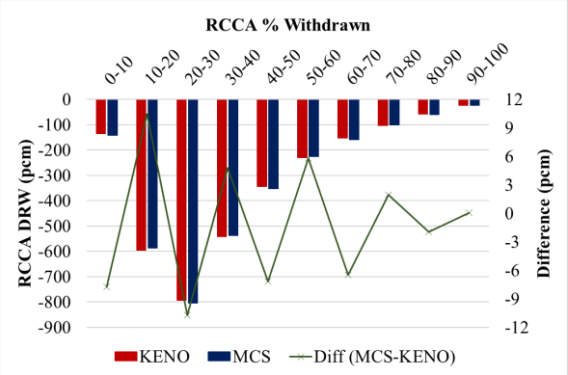

Fig. 2. RCCA differential rod worth.

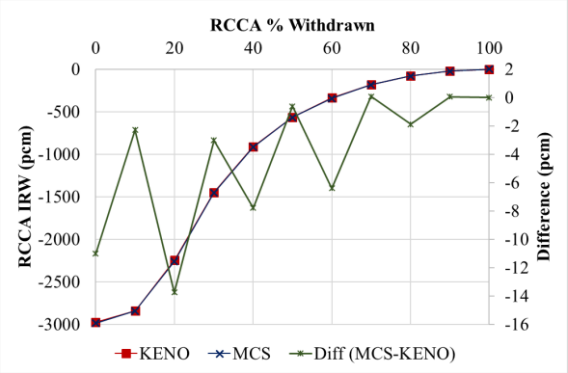

Fig. 3. RCCA integral rod worth curve.

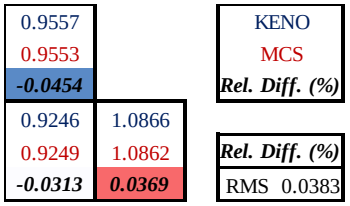

Fig. 4. Normalized assembly-wise radial power distribution for an octant 3x3 assembly lattice.

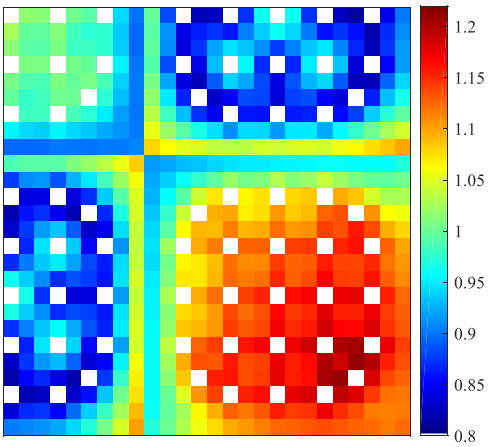

Fig. 5. Normalized pin-wise radial power distribution for a quarter 3x3 assembly lattice.

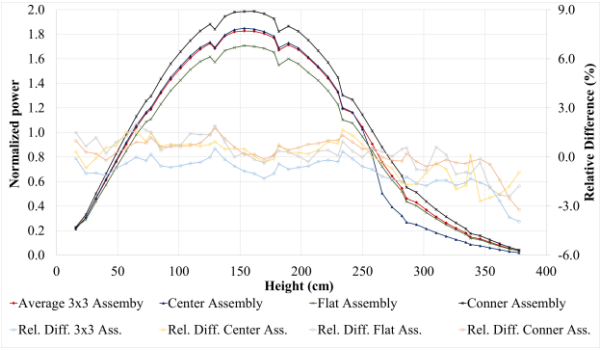

Fig. 6. Normalized average axial power distribution for 3x3 assembly lattice.

Table II summarizes MCS numerical results for the initial criticality configuration in which control bank D is partially inserted into the core. Other RCCA banks are also inserted. It is noted that the maximum difference of k_{eff} is 50 pcm. The comparison of initial criticality solutions between MCS and KENO-VI is shown in Fig. 7. Due to the huge number of particle histories, the deviation of power profiles is less than 1.6%. The normalized radial and axial power distribution (as shown in Figs. 8 and 9) all have relative differences of less than 0.8% and 1.6%, respectively. In addition, Figs. 10 and 11 show the average normalized pin-wise and 3D quarter core power distributions obtained by MCS. The CASL report does not provide the pin-wise power profile data, therefore, the MCS comparison to the reference was not made. Overall, it can be seen that the MCS solutions are in good agreement with KENO-VI reference solutions.

Table II: Summary of k_{eff} Results for Criticality Problems.

Case	RCCA Positions		k_{eff} (± 4pcm)	Diff. vs. KENO-VI (± 4pcm)
	Bank D Withdrawn Steps	RCCA Insertion		
1	167	-	0.99959	-31
2	230	-	0.99999	-33
3	97	Bank A	0.99844	-36
4	113	Bank B	0.99894	-42
5	119	Bank C	0.99852	-52
6	18	-	0.99884	-24
7	69	Bank SA	0.99856	-46
8	134	Bank SB	0.99882	-50
9	71	Bank SC	0.99911	13
10	71	Bank SD	0.99908	11

4.3. WBN1 Cycle 1 depletion with thermal-hydraulic feedback

This problem represents the depletion of the fuel and burnable absorbers and calculation of critical boron concentrations (CBCs) for WBN1 throughout the entire fuel cycle. WBN1 Cycle 1 simulation by MCS was performed with a quarter core geometry, and the fuel pin was divided into 1 radial ring and 10 axial meshes. The simulation was performed with 100% power, coolant inlet temperature at 565K, and Bank D at a position of 219 steps withdrawn. The number of MC cycles are as follows: 4 inactive cycles, 40 active cycles, 300 sub-cycle, and 15,000 histories per sub-cycle. The total simulation time was approximately 99 hours with 216 processes. In this work, the MCS solution was validated with the measured CBC data provided by CASL. The obtained CBC results by MCS compared to measured data are shown in Fig. 12. It is noted that the standard deviation of the computed CBC is less than 0.5 ppm. Good agreement with measurement is observed since most of the simulated points appear to be within 60 ppm. However, there is still a notable difference such as the maximum difference of 87 ppm at 160 EFPD. The observed CBC differences of the order of 60 ppm can be attributed to the difference in RCCA withdrawn steps, which cannot be changed in MCS simulation and the fuel was divided into only 10 axial meshes.

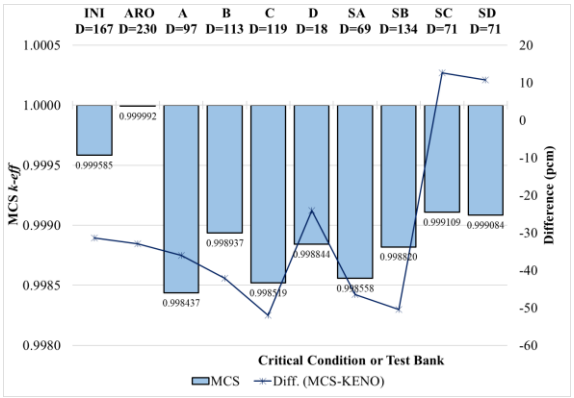

Fig. 7. Solution for whole core criticality problems.

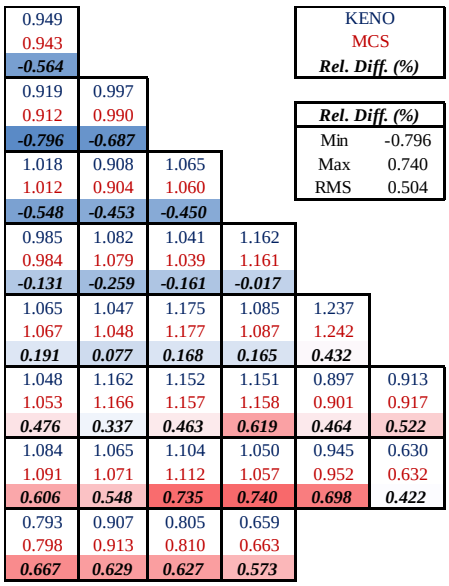

Fig. 8. Normalized assembly-wise radial power distribution for an octant symmetric core.

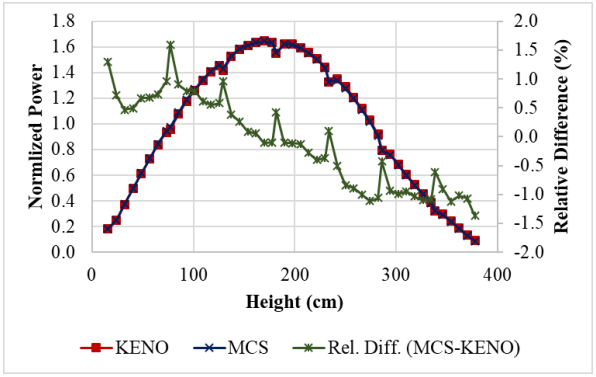

Fig. 9. Normalized average axial power distribution for a whole core.

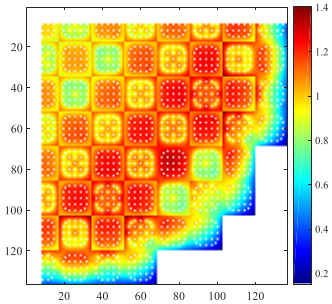

Fig. 10. Normalized pin-wise radial power distribution for a quarter symmetric core.

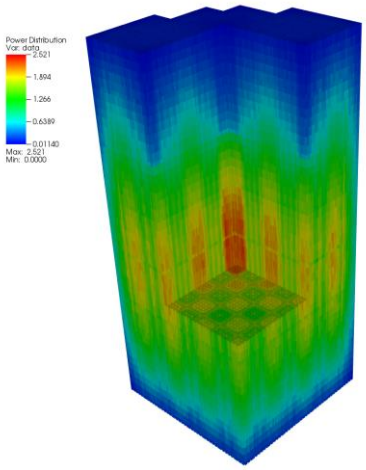

Fig. 11. Normalized 3D average power distribution for a quarter core.

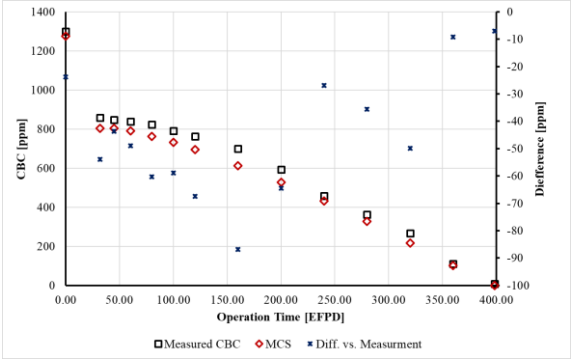

Fig. 12. Critical boron concentration for WBN1 Cycle 1.

5. Conclusion

In this study, the benchmark solutions of VERA problems obtained using UNIST MC code MCS has been presented. For both the 3D 3x3 mini-core and whole-core simulations, the MCS code shows good agreements with the reference results including cycle 1 depletion. It was successfully demonstrated that MCS could be used as a simulation tool for whole-core analysis in predicting the eigenvalue, pin power distribution, control rod worth, and depletion calculation. Future work will focus on the whole-core depletion calculation of WBN1 cycle 2 with thermal-hydraulic feedback coupled with the geometry thermal expansion.

Acknowledgements

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REFERENCES

[1] C. Kong, S. Choi, and D. Lee, Deterministic Lattice Code Development at UNIST, In: Proceedings of the PHYSOR Conference; Sep 28 – Oct 3; Kyoto, Japan Atomic Energy Agency, 2014. [USB]
[2] J. Leppänen, Development of a New Monte Carlo Reactor Physics Code, VTT Technical Research Centre of Finland, 2007.
[3] H. Lee, W. Kim, P. Zhang, A. Khassenov, Y. Jo, and D. Lee, Development Status of Monte Carlo Code at UNIST. In: Proceedings of the Korean Nuclear Society Spring Meeting, 2016 May 11-13, Jeju, Korean Nuclear Society, 2016. [USB]
[4] H. Lee, C. Kong and D. Lee, Status of Monte Carlo Code Development at UNIST. In: Proceedings of the PHYSOR Conference; Sep 28 – Oct 3; Kyoto, Japan Atomic Energy Agency, 2014. [USB]
[5] J. Jang, W. Kim, S. Jeong, E. Jeong, J. Park, M. Lemaire, H. Lee, Y. Jo, P. Zhang and D. Lee, Validation of UNIST Monte Carlo code MCS for criticality safety analysis of PWR spent fuel pool and storage cask, Annals of Nuclear Energy, Vol.114, p.495-509, 2018.
[6] H. Lee, W. Kim, P. Zhang, A. Khassenov, J. Park, J. Yu, S. Choi, H. Lee and D. Lee, Preliminary Simulation Results of BEAVRS Three-Dimensional Cycle 1 Whole Core Depletion by UNIST Monte Carlo Code MCS, In: Proceedings of the M&C conference, Apr 16-20, Jeju, Korean Nuclear Society, 2017 [USB].
[7] A.T. Godfrey, VERA Core Physics Benchmark Progression Problem Specifications, Rev. 4, Oak Ridge National Laboratory, Oak Ridge (TN), 2014. CASL-U-2012-0131-004.
[8] M.E. Dunn, C.L. Bentley, S. Goluoglu, L.S. Paschal, L.M. Petrie and H.L. Dodds, Development of a Continuous Energy Version of KENO Va. Nuclear Technology, Vol.119(3), p.306-313, 1997.