Validation of UNIST MCS Monte Carlo Code System for OPR-1000

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1. Introduction

MCS is a Monte Carlo (MC) neutron transport simulation code which has been developed exclusively since 2013 by the COmputational Reactor physics and Experiment laboratory (CORE) of Ulsan National Institute of Science and Technology (UNIST) for the neutronics design of nuclear reactors. To validate the capabilities of UNIST MCS, the OPR-1000 is modeled and the 3D whole-core depletion calculation is coupled with various feedback options, one dimensional thermal hydraulics (TH1D), depletion, Critical Boron Concentration (CBC), Equilibrium Xenon (Eq-Xe), and OpenW. Then, the calculated results of the axial and radial power distributions at Middle Of Cycle (MOC) and End Of Cycle (EOC) are compared with measured data. In addition, the result of the CBC calculation in MCS is compared with the measurement data and Nuclear Design Report (NDR) data. Overall, this study shows good agreement between MCS and the measured data. Therefore, good performance of the MCS modelling capabilities for OPR-1000 is demonstrated.

2. OPR-1000 Core Design

The OPR-1000 reactor is a pressurized water reactor (PWR) designed to operate at 2815 MW thermal power. The fuel consists of uranium dioxide (UO_2) pellets with slightly enriched ^{235}U. The core consists of 177 fuel assemblies (FA) in a 16x16 array with 236 fuel rods and 5 guide tubes welded to spacer grids. The burnable absorber (BA) is sintered UO_2-Gd_2O_3, and the Gd_2O_3 contents is 6-8 w/o [1]. The Cycle 1 core features a 3 batch, and the loading pattern is shown in Fig.1.

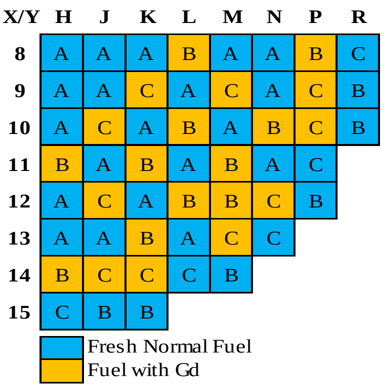

Fig. 1. Loading Pattern for OPR-1000 Cycle 1

3. MCS Monte Carlo Code

MCS is developed to solve complex whole-core problems for neutronics analyses of PWRs such as BEAVRS [2]. MCS features detail whole-core geometry modelling with universe and lattice functions, probability tables; for unresolved resonance region cross sections, free-gas treatment, S(a,b) treatment, and neutron up-scattering with Doppler broadening rejection correction. MCS uses three types of estimators, such as collision estimator, track length estimator, and absorption estimator. MCS uses a parallel fission bank algorithm for the parallel fission source iteration which is faster and can be used effectively with more processors than the parallel master-slave algorithm [3]. MCS uses continuous-energy cross section libraries and detailed geometrical data to estimate neutronics design parameters of a nuclear reactor such as effective multiplication factor, neutron flux, and fission power. MCS can perform 3D whole-core depletion calculation coupled with different modules, such as depletion, TH1D, Eq-Xe, CBC, OpenW, and quadradic depletion etc [4-5]. MCS depletion calculation can be performed either cell-wise or material-wise, based on user specification. The TH1D channel consists of a group of fuel pins and coolant paths surrounding them. It receives the power distribution in the fuel pin and updates the fuel temperature and coolant behavior based on the inlet temperature and flow rate. The TH1D module does not consider the cross flow through the pin. Eq-Xe is used to prevent non-physical oscillation in MC due to the fluctuation of the Xenon number density and flux during depletion for high dominance problems, by updating the number density of Xe-135 in each depletion step. The OpenW module is On-The-Fly Cross section generation based on the multipole method provided by MIT for 72 nuclides. MCS also has the capability to perform multi-cycle calculations. The shuffling function is added to the depletion capability which enables users to shuffle the fuel assemblies based on the shuffle map designed by the user.

3. Simulation Results

The OPR-1000 core contains 41,772 fuel pins. In this work, the OPR-1000 quarter core is modeled using the MCS code. The inlet temperature is 569.26K with the core flow rate of 16315 kg/s and all rods out (ARO). The fuel temperature is 900K while the coolant temperature is 600K. The simulation was performed using the ENDF/B-VII.1 library with the OpenW having 72 nuclides on a Linux cluster with 84 processors of type “2.8 GHz, Intel (R) Xeon (R) CPU”. In this simulation, the important feedback modules needed for the power reaction simulation, such as TH1D, depletion, Eq-Xe, CBC, and OpenW were used. As in the TH1D calculation, the TH node is always larger than, or equal to, the corresponding neutronic nodes within the same boundaries. The OPR-1000 core is considered to have two types of fuel pins and therefore needs to be divided for the TH calculation. The UO_2 fuel pin is divided into ten axial meshes with one ring. The fuel pin that contains gadolinium is divided into ten axial meshes and ten rings.

In this simulation, 5 inactive cycles and 20 active cycles are simulated with 10,000 histories per sub-cycle and 100 sub-cycles.

The whole core depletion and TH feedback calculation for 34 burnup steps took around 3925 CPU hours and used 6500 MB memory. The MCS results of the radial power assembly distribution at MOC and EOC is compared with measured data collected during the process of reactor commissioning. Fig. 2 shows the comparison between MCS, the measurement data, and NDR data. The maximum CBC difference of the MCS code, compared to the measured data and NDR, is 19 ppm and 32 ppm respectively. The good agreement of axial and radial power distribution at MOC (6 GWD/MT) can be visualized in Figs. 3-4. The maximum relative radial power distribution at MOC is 3.32 %. The axial and radial power distributions at EOC (13.8GWD/MT) are shown in Figs. 5-6. The maximum relative radial power distribution at EOC is less than 5%.

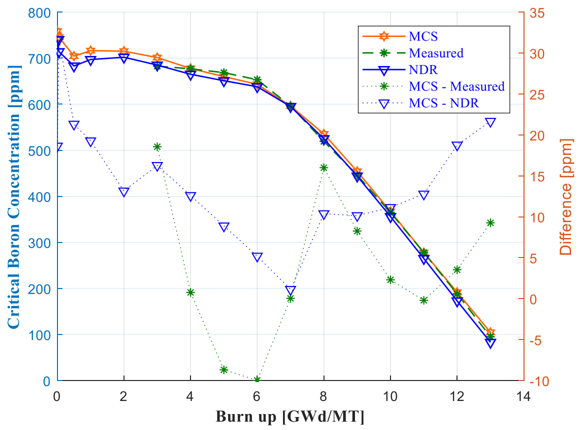

Fig. 2. The Comparison of Critical Boron Concentration of MCS versus Measured and NDR.

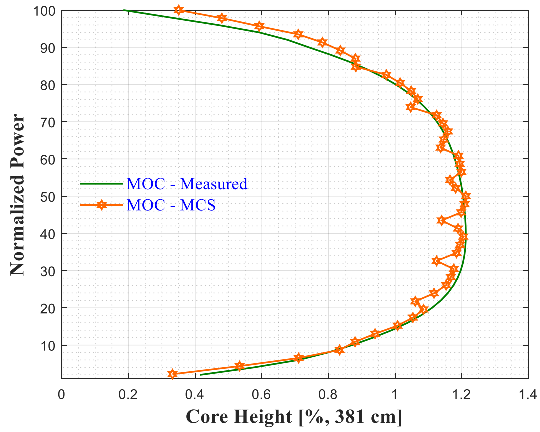

Fig. 3. Normalized Axial Power Distribution at MOC (6 GWD/MT).

	H	J	K	L	M	N	P	R
8	0.741	0.796	0.919	1.272	0.966	0.935	1.159	0.886
9	0.794	0.851	1.259	0.990	1.308	0.951	1.171	0.777
10	0.917	1.263	0.990	1.295	0.992	1.200	1.061	0.592
11	1.263	0.996	1.295	1.015	1.256	0.896	0.907	
12	0.965	1.309	0.998	1.260	1.228	1.024	0.579	
13	0.936	0.957	1.210	0.900	1.030	0.718		
14	1.170	1.186	1.064	0.900	0.577			
15	0.892	0.785	0.598					

RMS 1.10
Min -2.47
Max 3.32

-5 5

MCS
Rel. Error (%)

Fig. 4. The Normalized Radial Assembly Power Distribution compared with Measured at MOC (6 GWD/MT).

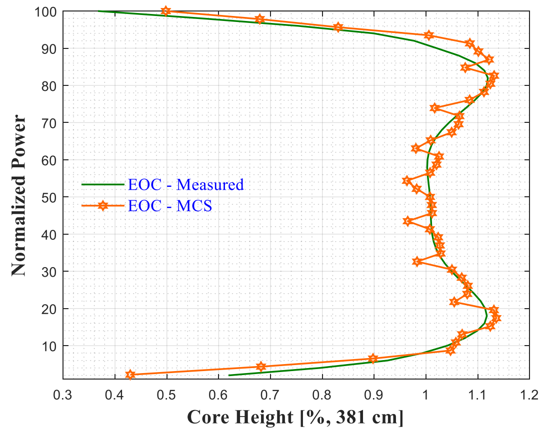

Fig. 5. Normalized Axial Power Distribution at EOC (13 GWD/MT).

	H	J	K	L	M	N	P	R
8	0.864	0.900	0.959	1.188	0.955	0.946	1.149	0.885
9	0.898	0.926	1.238	0.976	1.227	0.961	1.184	0.797
10	0.960	1.251	0.980	1.183	0.968	1.163	1.112	0.640
11	1.205	0.984	1.195	0.983	1.177	0.918	0.923	
12	0.968	1.248	0.974	1.176	1.193	1.085	0.632	
13	0.940	0.962	1.157	0.918	1.072	0.776	RMS Min	1.67 -3.27
14	1.126	1.169	1.092	0.914	0.628		Max	4.47
15	0.864	0.779	0.633		-5	5	MCS Rel. Error (%)	

Fig. 6. The Normalized Radial Assembly Power Distribution compared with Measured at EOC (13.8 GWD/MT).

3. Conclusions

The capabilities of the UNIST MCS Monte Carlo code system for OPR-1000 has been successfully validated. MCS is used to model the OPR-1000 and perform the 3D whole-core depletion calculation. The power distribution results of MCS at MOC and EOC shows good agreement with the measured data, with a maximum relative difference less than 5%. The critical boron concentration results are closer to the measured data than the NDR.

For future work, other important parameters such as fuel temperature, moderator temperature and density, and burn up distribution will be studied in MCS to compare with measured data for OPR-1000 cycle 1, 2, and so on.

ACKNOWLEDGEMENTS

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REFERENCES

[1] J. Choe, S. Choi, M. Park, P. Zhang, H. Lee, and D. Lee, Validation of the UNIST STREAM/RAST-K Code System with OPR -1000 Multi-cycle Operation, RPHA17 Conference, Chengdu, China, Aug. 24-25, 2017, Proceedings of the Reactor Physics Asia 2017.

[2] H. Lee, C. Kong, and D. Lee, Status of Monte Carlo Code development at UNIST, Proc. PHYSOR 2014, Kyoto, Japan, Sep. 28- Oct.3, 2014, American Nuclear Society, 2014.

[3] J. Jang, W. Kim, S. Jeong, E. Jeong, J. Park, M. Lemarire, H. Lee, Y. Jo, P. Zhang, D. Lee, Validation of UNIST Monte Carlo code MCS for criticality safety analysis of PWR spent fuel pool and storage cask, Ann. Nucl. Energy., 114 (2018), 495-509

[4] H. Lee, P. Zhang, A. Khassenov, and D. Lee, Preliminary Solution of BEAVRS Hot Full Power at BOC by Monte Carlo Code, KNS fall meeting, Jeju, Korea, October, 27-28, 2016.

[5] 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 Wholecore Depletion by UNIST Monte Carlo Code MCS, M&C 2017, Jeju, Korea, April 16-20, 2017.