Proceedings of the Reactor Physics Asia 2015 (RPHA15) Conference
Jeju, Korea, Sept. 16-18, 2015

Hybrid Depletion Method for Monte Carlo Analysis of PWRs

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1. Introduction

Nowadays, high fidelity reactor analysis methods, especially Monte Carlo [1-2] and Method of Characteristics [3-4], are attracting people's attention. So far, the two step method [5] has been considered the only way to design a nuclear reactor core because it takes several thousand whole core simulations to design a new reactor core but the high fidelity methods need a lot of time and computer memory. A single step steady calculation also takes lots of time. A more severe problem is the depletion calculation. In the Monte Carlo depletion calculation, about 200~300 isotopes are considered and about 6 1-group reaction rates and 3-group fission reaction rate are tallied for simulating a Pressurized Water Reactor (PWR) which contains about 50,000 fuel pins [6]. In Kord Smith's challenge [7], he suggested dividing the fuel pin axially into 400 meshes and radially into 10 rings. To tally a 1-group reaction rate, three double precision floating point variables (8 bytes) is needed in a Monte Carlo transport simulation. 9 kinds of tally for every 300 isotopes in 200 million cells (50,000 fuel pins x 400 axial meshes x 10 radial rings) yield about 13 terabytes (Tb) of computer memory just for the tally. The tremendous computer memory consumption is a big problem and also the extremely slow computing time is a big obstacle in Monte Carlo depletion calculations. In this paper, a hybrid depletion method is suggested. This method adopts resonance treated multi-group cross sections with tallied multi-group neutron flux. The method does not need the reaction rate tally; it just needs the cell-wise multi-group flux tally. The hybrid Monte Carlo depletion method can reduce a considerable amount of computer memory and this memory reduction yields a computing speed improvement. The in-house Monte Carlo code MCS [8] and Lattice code STREAM [9] are used for relevant analyses with ENDF/B-VII.0 [10] nuclear data.

2. Model Problem Description

A 2-dimensional infinite pin cell model is analyzed in order to do a feasibility study of the hybrid Monte Carlo depletion using resonance treated multi-group cross sections. Fig. 1 is the geometry of a pin cell model which is one of the fuel pin in BEAVRS [6]. Table I contains the specific fresh fuel composition of the model problem.

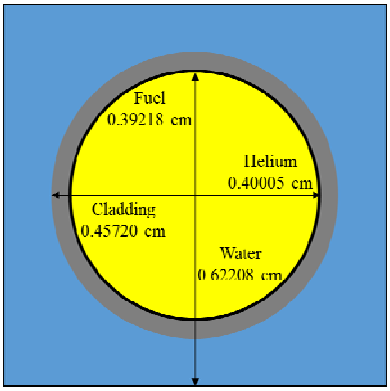

Fig. 1. 2.4 % enriched UO_2 fuel pin geometry [6]

Table I. Fresh Fuel Pin Composition [6]

Density (g/cc)	10.29748
Isotope	Number Density (atom/b-cm)
U-234	4.4842E-06
U-235	5.5814E-04
U-238	2.2407E-02
O-16	4.5828E-02
O-17	1.7457E-05
O-18	9.4176E-05

3. Conventional Monte Carlo Depletion

In Monte Carlo depletion simulation, the MCS tallies 1-group reaction rates for every isotope in the fuel pin. In this paper, (n,γ), (n,f), $(n,2n)$, $(n,3n)$, (n,α), (n,p) reactions are considered for every isotope and 3-group fission reaction rates are considered for fissile isotopes. Fig. 2 shows a flow chart of the conventional Monte Carlo depletion calculation.

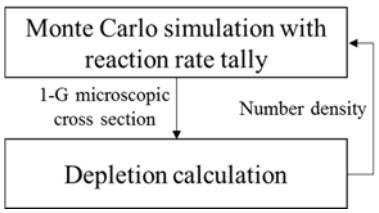

Fig. 2. Flow chart of conventional Monte Carlo depletion calculation

In the deterministic methods (e.g., method of characteristics), there are more assumptions than the Monte Carlo method. For instance, the deterministic methods adopt the multi-group cross sections. So far, there is no way to generate a multi-group cross section which is exactly equivalent to a continuous energy cross section. It is possible to consider that the accuracies of the deterministic transport solver and the Monte Carlo transport solver are almost the same, but the multi-group cross section involves more error than the continuous energy cross section. For that reason, Monte Carlo simulation results have been used as a reference solution of the deterministic simulation results, but the consumption of tremendous computer memory for the reaction rate tally is the most severe obstacle in the conventional Monte Carlo depletion calculation. According to Kord Smith's challenge [7], he suggested considering about 300 isotopes in a single burnup region and dividing the fuel pin axially into 400 meshes and radially into 10 rings. This would result in about 200 million burnup regions (50,000 fuel pins x 400 axial meshes x 10 radial rings) in a PWR core. The 6 types of reaction rates and 3-group fission reaction rate are tallied for depletion calculation in this paper. The 6 reaction rates are used to build a depletion matrix and the 3-group fission reaction rate is used to utilize multi-group fission yield data. To tally a single variable, three double precision floating point variables are needed in order to obtain an average value and its statistical error. Table II summarizes how much computer memory is required to simulate the pin-wise depletion of a commercial PWR core in the conventional Monte Carlo depletion method. By multiplying every value from the number of fuel pins to the number of the number of information in Table II, and by considering the computer memory requirement of double precision floating point variable, it is possible to obtain about 13 Tb of the computer memory consumption for the reaction rates tally.

Table II. Memory Consumption for Full Core Depletion Calculation in Conventional Monte Carlo Simulation

Fuel pin	50,000
Axial mesh	400
Radial ring	10
Isotope	300
Reaction tally	9
Information	3
Double precision	8 byte
Total	~13 Tb

4. Hybrid Monte Carlo Depletion

The same kinds of reactions are considered in the hybrid Monte Carlo depletion. The difference is the hybrid Monte Carlo depletion does not need a reaction-wise tally. It just tallies the cell-wise multi-group flux and utilizes resonance treated multi-group cross sections. The multi-group cross-sections are generated by using the lattice physics code STREAM. In the generation of multi-group cross section library of STREAM, 72 energy group structures are used. The 72-group structure is similar with other production codes [11-13]. The 12 fast energy groups (2.478E+5 eV ~ 1E+7eV) and 43 thermal energy groups (1E-5 eV ~ 4 eV) make accurate fast and thermal cross sections for LWRs applications. STREAM performs the resonance self-shielding calculation and generates shielded multi-group cross sections for the 15 resonance energy group (4 eV ~ 2.478E+5 eV) and extended resonance energy range(0.3 eV ~ 4 eV). An advanced resonance self-shielding method has been adopted in the STREAM code [9]. Fig. 3 presents a flow chart of the hybrid Monte Carlo depletion calculation.

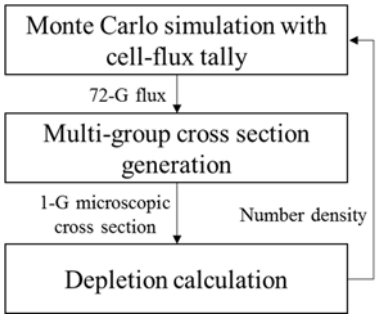

Fig. 3. Flow chart of hybrid Monte Carlo depletion calculation

Table III summarizes how much computer memory is required to simulate the pin-wise depletion of a commercial PWR core in the hybrid Monte Carlo depletion method. By the same way in Table II, it is possible to obtain about 345 gigabytes (Gb) of the computer memory consumption for the 72-group flux tally. This is just 1/40 the computer memory of the conventional Monte Carlo depletion method.

Table III. Memory Consumption for Full Core Depletion Calculation in Hybrid Monte Carlo Simulation

Fuel pin	50,000
Axial mesh	400
Radial ring	10
Group flux	72
Information	3
Double precision	8 byte
Total	~345 Gb

5. Results

Using the UO_2 fuel pin, the conventional and hybrid depletion calculation is conducted. The unit burnup step of the calculation is set as 600 MWd/MTU and the temperature of fuel, helium, cladding, and water is fixed as 300 K. Fig. 4 show the comparison between the conventional and the hybrid Monte Carlo depletion calculations. The overall standard deviation of the

multiplication factor is 15 pcm. The largest difference is about 150 pcm. The difference is several times larger than the overall standard deviation but it is still possible to have trust on the hybrid Monte Carlo depletion method in terms of a practical implementation.

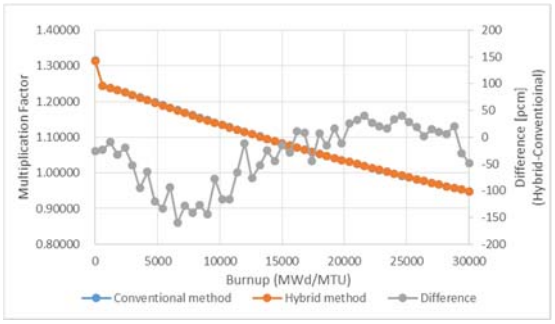

Fig. 4. Multiplication factor comparison between the conventional and hybrid Monte Carlo depletion calculation.

6. Conclusions

This is a preliminary study of the hybrid Monte Carlo depletion method. Using the infinite 2-dimensional fuel pin model, a feasibility test is performed. The hybrid Monte Carlo depletion method can reduce the computer memory for the tally by about 1/40 and the accuracy is also credible. The suggested method can be a potential solution to the computer memory problem in Monte Carlo depletion calculations. In the future, this method is going to be applied to an assembly and a whole core.

Acknowledgements

This work was partially supported by National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIP).

This work was partially supported by the Nuclear Power Core Technology Development Program of the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted financial resource from the Ministry of Trade, Industry & Energy, Republic of Korea (No. 20141520400090).

References

1. Pelowitz D, editor, “MCNP6 user’s manual, version 1.0,” Los Alamos National Security, LLC, Los Alamos, NM (2013)
2. Paul K. Romano and Benoit Forget, “The OpenMC Monte Carlo Particle Transport Code,” *Ann. Nucl. Energy*, 51, 274–281 (2013). http://dx.doi.org/10.1016/j.anucene.2012.06.040
3. William Boyd, Samuel Shaner, Lulu Li, Benoit Forget, and Kord Smith, “The OpenMOC Method of Characteristics Neutral Particle Transport Code.” Annals of Nuclear Energy, **68**, 43–52 (2014).
4. H. G. Joo, J. Y. Cho, K. S. Kim, C. C. Lee, and S. Q. Zee, “Methods and Performance of a Three-Dimensional Whole Core Transport Code DeCART,” *Proc.* PHYSOR 2004, Chicago, April 25-29, 2004, American Nuclear Society, CD-ROM (2004).
5. Tamer Bahadir, Sten-Orjan Lindahl, “Studsvik’s Next Generation Nodal Code SIMULATE-5,” ANFM IV, Hilton Head Island, South Carolina, April 12-15 (2009)
6. Benoit Forget, Kord Smith, “Benchmark for Evaluation and Validation of Reactor Simulations (BEAVRS), v1.1.1,” MIT Computational Reactor Physics Group, Octorber 30 (2013).
7. K. Smith and B, Forget, “Challenges in the Development of High-Fidelity LWR Core Neutronics Tools,” M&C 2013, Sun Vally, ID, May 5-9 (2013).
8. Hyunsuk Lee, Chidong Kong, and Deokjung Lee, “A New Monte Carlo Neutron Transport Code at UNIST,” KNS Spring Meeting, Jeju Convention Center, May 28-30 (2014).
9. Sooyoung Choi, Hyunsuk Lee, SerGi Hong, and Deokjung Lee, “Resonance Self-Shielding Methodolgy of New Neutron Transport Code STREAM,” *J. Nucl. Sci. Technol*., http://dx.doi.org/10.1080/00223131.2014.993738 (2015).
10. M. B. Chadwick, et al., “ENDF/B-VII.0: next generation evaluated nuclear data library for nuclear science and technology,” Nucl Data Sheets, Vol. 107, pp. 2931–3059 (2006).
11. J. Rhodes, K. Smith, M. Edenius, “CASMO-4E Extended Capability CASMO-4 User’s Manual,” Studsvik Scandpower Report SSP-09/442–U Rev 0 (2009).
12. D. Knott, A. Yamamoto, “Handbook of Nuclear Engineering: Lattice physics computations,” US: Springer, pp. 913-1239 (2010).
13. R. J. J. Stamm'ler, M. Abbate, “Methods of Steady-State Reactor Physics in Nuclear Design,” London, UK: Academic Press, Group constants in the resonance region, pp. 286-336 (1983).