A Study on Reconstruction of Intra Fuel Pin Power and Flux Distributions with the iDTMC Method in the Monte Carlo Reactor Analyses

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1. Introduction

For a practical application of the Monte Carlo (MC) method in commercial reactor problems, a whole core depletion calculation should be affordable in terms of the memory and time requirement. Although the performance of the computer has been steadily advanced, the large-scale whole core burnup calculation on an intra-pin level is still demanding and unfeasible.

Many research groups have studied to decrease the numerical cost of the depletion calculation in MC method. The Chebyshev rational approximation method has been developed to enhance the computational accuracy and efficiency in solving the burnup matrix by examining characteristics of the matrix [1]. The predictor-corrector method has been widely used in the depletion calculation [2], which can improve the stability of the established MC burnup calculation. Both studies showed improvements over other methods, but challenges to be solved still remained.

In the meantime, the iDTMC method has been previously developed to decrease the computing time and stochastic uncertainties for the steady-state MC eigenvalue analysis [3][4]. This study showed that the iDTMC method can substantially improve the accuracy and reliability of the pin-wise power and flux profiles compared to the conventional MC method. Since the power and flux distributions are essential parameters in the depletion calculation, it is expected that the burnup calculation can be also significantly improved by the iDTMC method.

For a preliminary study on the application of the iDTMC method in the fuel pin depletion calculation, a reconstruction scheme of intra pin power and flux profiles is presented in this paper. The intra pin power and flux distributions are estimated in one of fuel assemblies of the APR1400 problem, and are compared to the standard MC method to verify the efficiency.

2. Methods

2.1 iDTMC method application in depletion analyses

The change of material compositions with the time in a reactor core system can be estimated by the following Bateman equation:

$$\frac{dN_A(t)}{dt} = -(\sigma_A^a\phi + \lambda_A)N_A(t) + \sigma_C\phi N_c(t) + \lambda_B N_B(t) \quad (1)$$

where N_x is the number density of a nuclide *X* (i.e. *A*, *B*, and *C*), σ is the cross section, λ is the decay constant. Fig. 1 describes the decay chain and reactions between the nuclides associated with Eq. (1). The nuclide *A* can be either produced or lost by a decay and nuclear reactions.

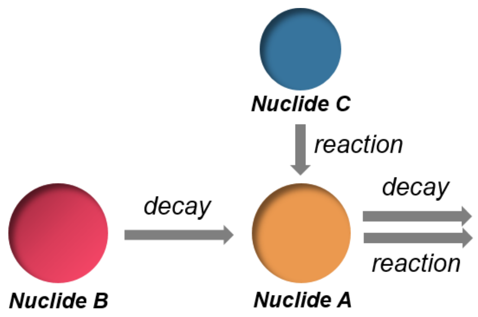

Fig. 1. Decay chain of a nuclide A

As presented in Eq. (1), the neutron flux plays an important role in the accurate evaluation of the material compositions. In other words, the accurate and reliable neutron flux would improve the neutronic results.

Meanwhile, the iDTMC method has validated its numerical performance on the evaluation of the pin-resolved power and flux distribution. The iDTMC method can decrease the computing time of the steady-state eigenvalue calculation at every depletion step, while providing the more reliable and accurate power and neutron flux solutions. Therefore, total numerical cost regarding the depletion calculation also can be decreased with the application of the iDTMC method.

2.2 Intra-pin reconstruction for flux distribution

The iDTMC method can provide the detailed pin-resolved power and flux distribution on the lattice-based fine mesh grid. The FMFD parameters like the cross sections are homogenized and condensed on each pin node. In short, the intra-pin level detailed information cannot be calculated with the iDTMC method.

However, for the practical depletion analysis in thermal reactors, the intra-pin information is necessary. This is because the flux level in the outer region is highly different from that in the central region due to the self-shielding effect. To take into account the flux gradient in fuel pins, the fuel pin should be divided into several sub-rings.

Therefore, in the iDTMC method, the intra-pin information is obtained by the reconstruction process with the MC tallies as shown in Fig. 2. The magnitude of the pin neutron flux is determined by the iDTMC method, but the detailed profile is determined by the MC method. Then, the iDTMC-corrected intra-pin flux distribution can be calculated by multiplying two parameters:

$$\phi_{i,r} = ff_{i,r} \times \phi_i^{DTMC'} \quad (2)$$

where i is the node index, r is the ring index, ff is the form function determined by the whole geometry MC calculation, $\phi_i^{DTMC'}$ is the actual neutron flux level determined by the iDTMC method. The form function obtained from MC simulation is calculated for the average to be unity:

$$ff_{i,r} = \frac{\phi_{i,r}^{MC}}{\sum_{r=1}^{N_r} \phi_{i,r}^{MC} / N_r} \quad (3)$$

where ϕ^{MC} is the neutron flux calculated from the MC simulation. The iDTMC flux is also normalized by the nominal reactor power *P*:

$$\phi_i^{iDTMC'} = C \times \phi_i^{iDTMC} \quad (4)$$

where C is the normalization factor defined as:

$$C = \frac{P}{\sum_i \kappa\Sigma_f^i \phi_i^{iDTMC} V_i}$$

$\kappa\Sigma_f$ is the fission cross section multiplied by the fission energy, and V_i is the volume of the node i .

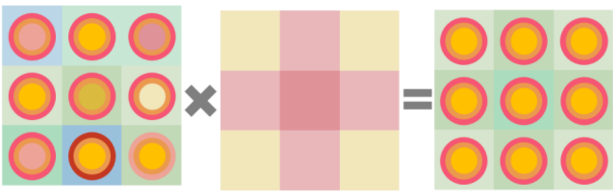

Fig. 2. Reconstruction of intra-pin flux distribution with iDTMC method

The conventional generation-based MC method only tracks the birth and death of the neutron particles, so the neutron balance in the reactor system cannot be strictly taken into account. On the other hand, the iDTMC method is supposed to preserve the nodal balance at every cycle at each node by the current-based correction and homogenized flux-weighted cross sections. Therefore, the iDTMC method can provide the more balanced and accurate reactor parameters than the standard MC method in the same calculation condition.

3. Numerical Results

A single fuel assembly of the APR1400 is considered for the numerical analysis. A B1 type fuel assembly in Fig. 3 consists of two different uranium enriched UO_2 fuel pins, Gadolinia burnable absorbers, and guide thimbles. For the fuel depletion, the fuel pin is divided into 3 sub rings to be equi-volume as shown in Fig. 4. A reflective boundary condition is assigned to the all surfaces surrounding the fuel assembly.

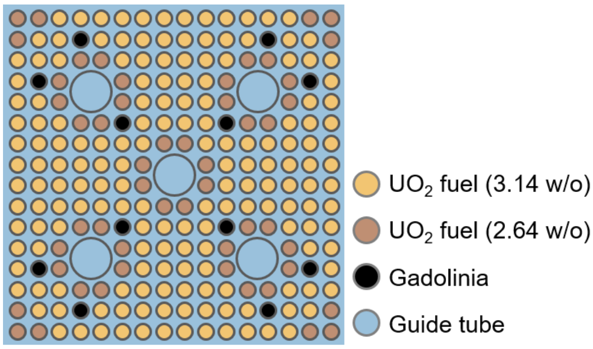

Fig. 3. Configuration of fuel assembly (type B1)

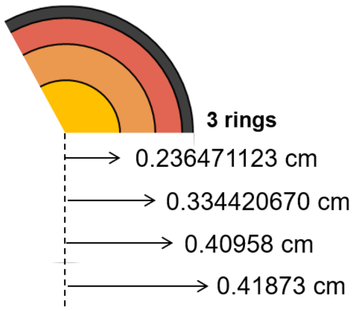

Fig. 4. Division of a fuel pin

Neutronic analysis is implemented with the in-house MC code named iMC. Total 15 inactive cycles, 10 active cycles, and 100,000 particles per cycle are used. The reactor power is set to be 10 MWth.

The intra pin power and flux distributions reconstructed by the iDTMC method were evaluated and compared with the results of the standard MC method. The intra pin power distribution was nicely regenerated with the iDTMC method showing a good agreement with the MC method. However, it was clearly shown that the iDTMC method provided more reliable power distribution with the lower standard deviations. Fig. 6 presents the distribution of the standard deviation (SD) for the intra pin power distribution.

Similarly to the power distribution, the intra pin flux distribution was also calculated and compared. Fig. 7 describes the intra pin flux distribution for the MC and the iDTMC method. They also showed a good agreement with each other though the reliability is different. The reliability of the intra pin flux distribution can be found in Fig. 8. The iDTMC method provided much lower standard deviations.

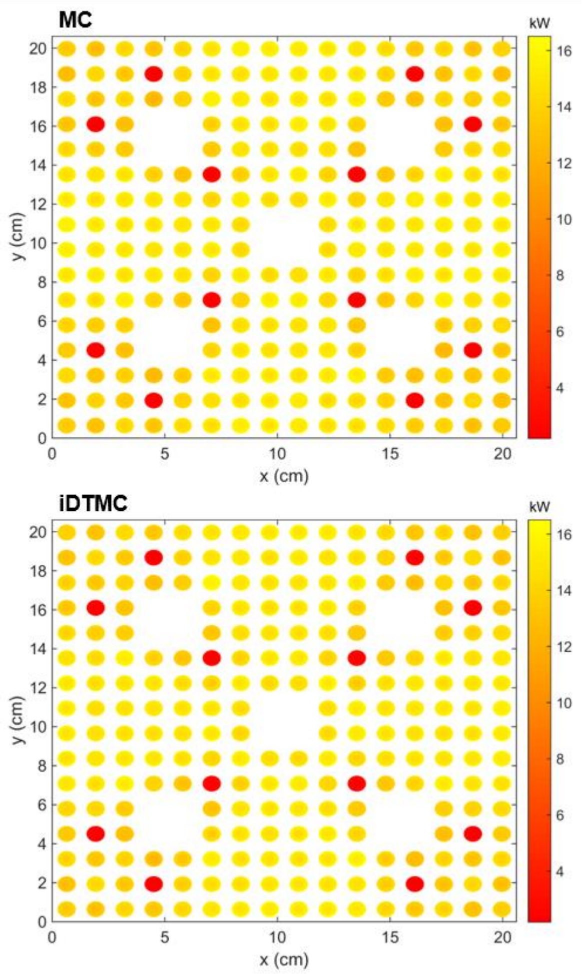

Fig. 5. Intra pin power distribution at active cycle 10

The reliability of the intra pin power and flux distributions is also demonstrated with the average value as given in Table 1. The table presents the average of standard deviations for power and flux distribution at the several specific cycles.

The iDTMC method clearly showed much smaller average standard deviation than the MC method. In particular, the iDTMC method had 13 times and 9 times lower stochastic errors at the active cycle 3 for the power and flux, respectively. Although the standard deviation in the iDTMC method more smoothly decreased along with the cycles than the MC method, the iDTMC method still provided much lower stochastic errors by the end of the simulation.

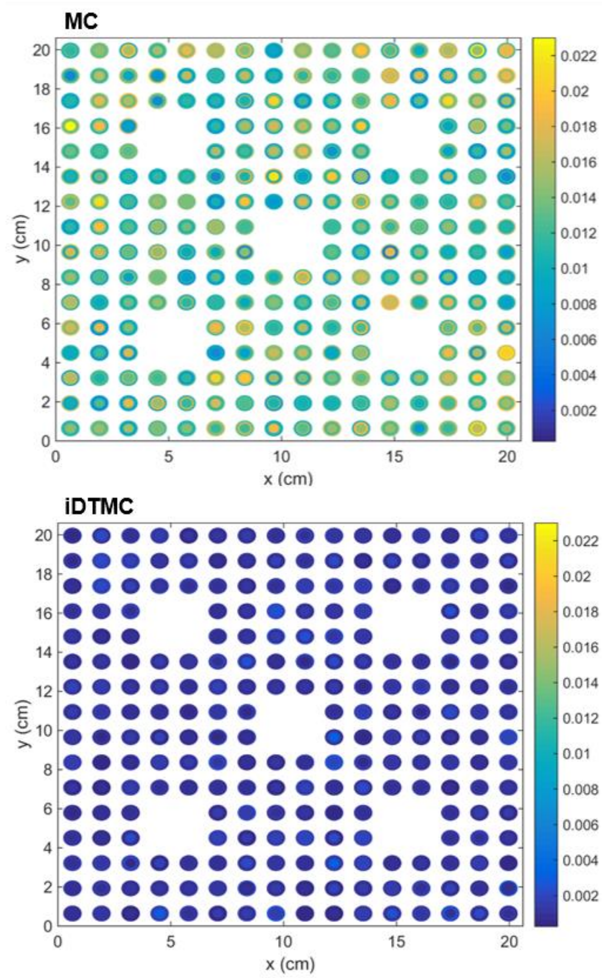

Fig. 6. SD of power distribution at active cycle 10

Table 1. Average standard deviation of intra pin power and flux distribution

Cycle	Power		Flux	
	MC	iDTMC	MC	iDTMC
3	0.0122	0.0009	0.0072	0.0008
5	0.0077	0.0007	0.0046	0.0008
10	0.0046	0.0006	0.0027	0.0005

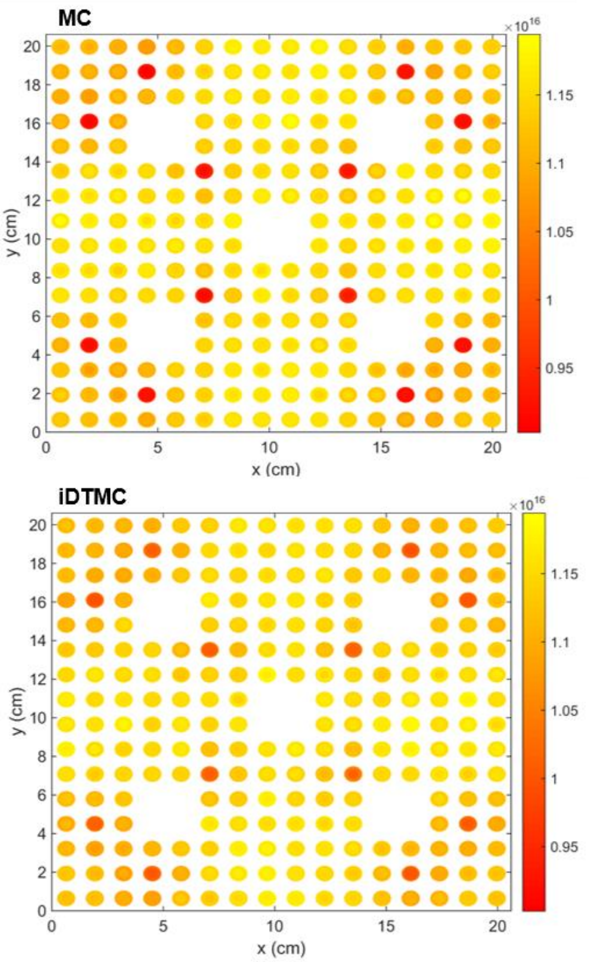

Fig. 7. Intra pin flux distribution at active cycle 10

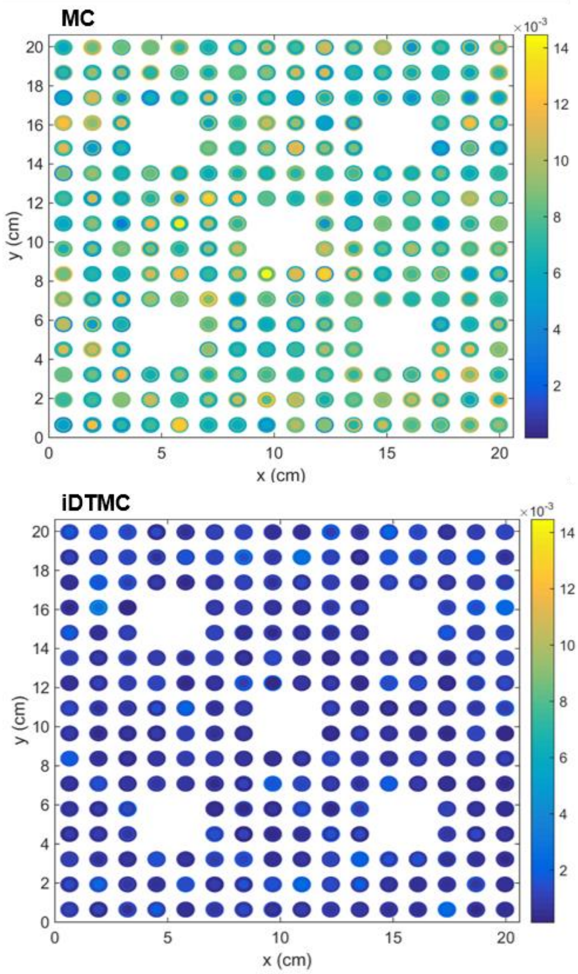

Fig. 8. SD of flux distribution at active cycle 10

3. Conclusions

For the Monte Carlo fuel depletion analysis, the intra-pin power and flux distributions are calculated by the iDTMC method with a simple reconstruction process, and it is confirmed that both detailed power and flux profiles are accurately reproduced. Meanwhile, the reliability of the reactor parameters are much improved with the iDTMC method. The standard deviation for the intra pin power and flux distribution is 6-10 times lower in the iDTMC method compared to the standard MC method on the average. Furthermore, the numerical cost is comparable between the MC and iDTMC methods because the deterministic calculation for the p-FMFD calculation and the intra-pin reconstruction process is almost negligible compared to the MC calculation. Therefore, it is expected that the iDTMC method also can provide more reliable and accurate depletion calculation. The actual depletion calculation will be implemented in the future.

ACKNOWLEDEGEMENT

This work was supported by the National Research Foundation of Korea grant NRF-2016R1A5A1013919 funded by the Korean government.

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