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Integrated Neutronics and Thermal-Hydraulics Simulation of the APR 1400 with iDTMC/START Coupling

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1. Introduction

The Monte Carlo (MC) method is one of the most accurate approaches for reactor analysis, as it explicitly simulates neutron transport. For accurate reactor modeling, it is essential to account for feedback effects, since variations in density and temperature directly influence nuclear cross sections. This underscores the importance of multi-physics simulations that couple neutronics with thermal-hydraulics. To mitigate the high computational cost of the MC method, the Improved Deterministic Truncation of Monte Carlo (iDTMC) method was developed at KAIST in-house MC code, iMC. By applying deterministic acceleration during inactive cycles and truncation during active cycles, iDTMC enables pin-wise solutions to be obtained within only a few active cycles, significantly improving computational efficiency and reducing simulation cost. For the thermal-hydraulics aspect, the Steady and Transient Analysis of Reactor Thermal-hydraulics (START) code has been developed at KAIST. START is a subchannel analysis code based on a homogeneous two-phase flow model, which solves the mass, momentum, and energy conservation equations axially while accounting for lateral momentum exchange between subchannels. It provides subchannel fluid conditions, fuel and cladding temperatures through a heat conduction model, and evaluates the margin to critical heat flux (CHF). In this study, we present coupled neutronics and thermal-hydraulics simulations of the Advanced Power Reactor 1400MW (APR1400) core using the iDTMC/START framework.

2. Methodology

2.1 The iDTMC Method

The iDTMC method is a hybrid approach that combines stochastic and deterministic analyses of the reactor [1]. In the conventional MC method, the iDTMC framework enhances the convergence of the fission source distribution (FSD) during inactive cycles by coupling the partial-current-based Coarse Mesh Finite Difference (p-CMFD) method. During active cycles, pin-wise solutions are obtained using the partial-current-based Fine Mesh Finite Difference (p-FMFD) method. Both p-CMFD and p-FMFD solve the same one-group neutron balance equation, given in Equation (1).

$$\sum_s \frac{A_s}{V_i}(J_{s1} - J_{s0}) + \Sigma_a^i \varphi_i = \frac{1}{k_{eff}} \nu \Sigma_f^i \varphi_i \qquad (1)$$

Here, A_s is the surface area, V_i is the volume of node i, s is the surface index, φ is the flux, J is the current, Σ_a and $\nu\Sigma_f$ are the absorption and nu-fission cross sections, respectively, and k_{eff} is the effective multiplication factor. The difference between p-CMFD and p-FMFD lies in the node configuration: p-CMFD assigns sub-assemblies as nodes, whereas p-FMFD uses individual fuel pins as nodes, as illustrated in Figure 1.

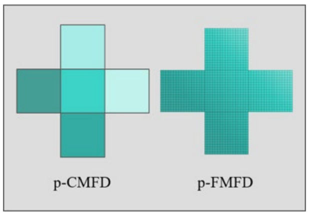

Figure 1. Node configurations
in p-CMFD (left) and p-FMFD (right)

The iDTMC method consists of two major steps. First, p-CMFD accelerates the convergence of the FSD by being directly coupled to the MC simulation during the inactive cycles. Second, p-FMFD obtains pin-wise solutions using the tallied parameters accumulated after a few skip cycles. To ensure the stability of the simulation, p-FMFD is applied in a decoupled manner during the active cycles. The overall flow of the iDTMC method is shown in Figure 2.

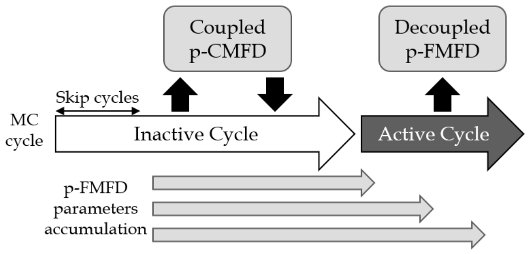

Figure 2. Schematic of the iDTMC method

2.2 The START Code

The START code is a subchannel analysis tool capable of evaluating thermal-hydraulic behavior under both steady-state and transient conditions [2]. It determines local thermal-hydraulic properties, calculates fuel and cladding temperatures using a fuel conduction model, and evaluates margins to CHF. The START code is based on a homogeneous two-phase flow model, which assumes that liquid and vapor phases share the same velocity. This assumption reduces the governing equations to a single momentum conservation equation per subchannel. An example of the discretization for a 2×2 fuel pin cluster is shown in Figure 3.

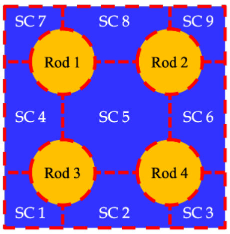

Figure 3. Discretization in the top view

In the axial direction, the START code solves the conservation equations for mass, momentum, and energy, while also accounting for lateral momentum exchange between neighboring subchannels. To address limitations of the homogeneous flow assumption, various empirical correlations and models are incorporated. The correlations and models applied in START for specific phenomena and parameters are summarized in Table 1.

Table 1. Correlations and models used

Phenomenon / Parameter	**Correlation or Model**
Two-phase friction multiplier	Armand correlation
Space grid pressure loss coefficient	Rehme correlation
Subcooled boiling	Lellouche model
Void fraction	Armand-Massena correlation
Heat transfer coefficient (single phase convection)	Dittus-Boelter correlation
Heat transfer coefficient (nucleate boiling)	Thom correlation
CHF prediction	2006 Groeneveld CHF Lookup Table

The START code has been validated through benchmark problems from the NUPEC PWR Subchannel and Bundle Tests. The validation results demonstrate that START can reliably predict thermal-hydraulic properties at both the single-subchannel level and the full fuel assembly level [2].

2.3 The iDTMC/START Coupled Simulation

In this study, neutronics and thermal-hydraulics were coupled using the iDTMC method and the START code. The overall flow of the coupled simulations is illustrated in Figure 4.

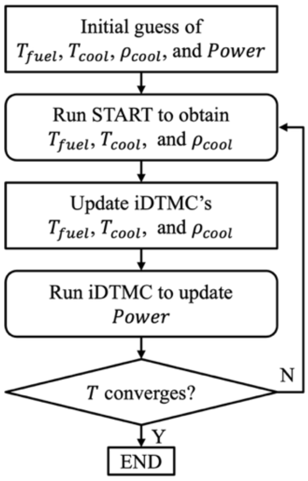

Figure 4. Flowchart of the iDTMC/START framework

The calculation begins with an initial guess of the power distribution. Based on this homogeneous power assumption, the START code computes the fuel temperature (T_{fuel}), cladding temperature (T_{clad}) coolant temperature (T_{cool}), and coolant density (ρ_{cool}). The resulting thermal-hydraulic data are then transferred directly to the iDTMC calculation without relaxation. In particular, the fuel temperature is obtained from the heat conduction module of the START code, which employs a finite volume discretization with 12 equi-volume radial nodes. These data are mapped onto 4 equi-volume nodes of the fuel in the the iDTMC calculation, ensuring high-fidelity geometrical coupling. This information is then used in iDTMC to update the cross sections and account for Doppler broadening. The new power distribution from the iDTMC calculation is subsequently passed back to the START code. This exchange between iDTMC and START proceeds in a Picard iterative manner until both fuel and coolant temperatures converge. The convergence criterion is defined using the l^2 norm, as shown in Equation (2).

$$\frac{\|T_x^n - T_x^{n-1}\|_{l^2}}{\|T_x^n\|_{l^2}} \leq \epsilon_x \; where \; x = fuel, cool \qquad (2)$$

Here, T_x^n represents the solution vector for either fuel or coolant temperature at the n^{th} Picard iteration, and ϵ_x is the user-defined convergence criterion for the corresponding temperature vector. Because the neutronics calculation introduces stochastic error, the iteration error does not vanish even with a large number of iterations but instead converges to a finite value.

3. Numerical Results

For the neutronics-thermal hydraulics coupled simulation with the iDTMC/START framework, the first operating cycle of the APR1400 core was analyzed at BOC, ARO, and zero soluble boron, as shown in Figure 5.

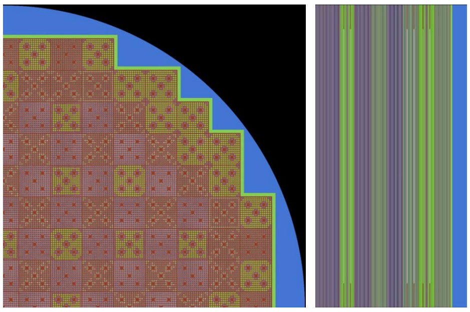

Figure 5. Radial (left) and axial (right) core configuration of APR1400

The detailed core configuration follows Ref. [1] and [4]. For the water temperature in guide and instrument tubes, the average temperature of the neighboring subchannels was applied. Based on previous studies [1], 2×10^7 histories per cycle were used. From the Shannon entropy, 40 inactive cycles with 20 skipped cycles were used.

The Picard iteration error of the iDTMC/START simulation is shown in Figure 7 for 1 and 10 active cycles.

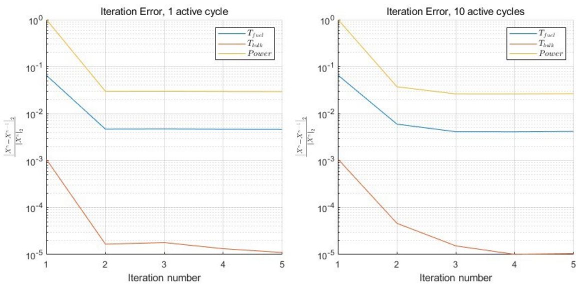

Figure 6. Iteration error of power(black), fuel (blue), and coolant (orange) 1 (left), 10 (right) active cycles

Fuel and coolant temperature errors reached convergence in five iterations for every configuration examined. This indicates that, within iDTMC, the coupling convergence is insensitive to the number of active cycles. The eigenvalues reported in Table 2 are consistent, and their variance were computed using Dirichlet distribution based sampling; which will be reported in a separate manuscript.

Table 2. Converged eigenvalues

Number of active cycles	k_{eff}
1	1.15436 ± 2.85 pcm
10	1.15429 ± 2.29 pcm

Figures 7–10 present the converged normalized power and temperature distributions of the APR1400 core.

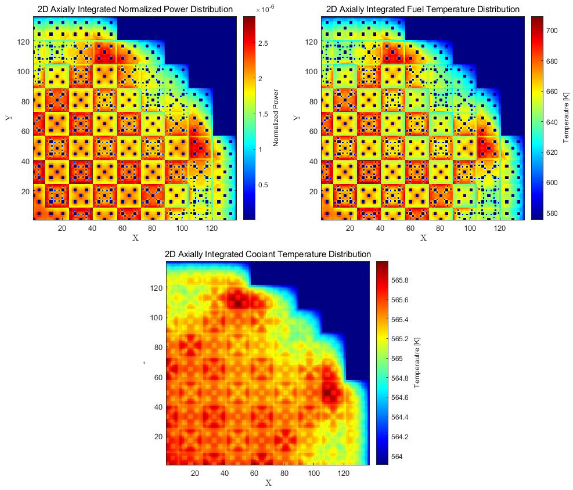

Figure 7. Axially integrated distributions of power (top left), fuel (top right), and coolant temperature (bottom)

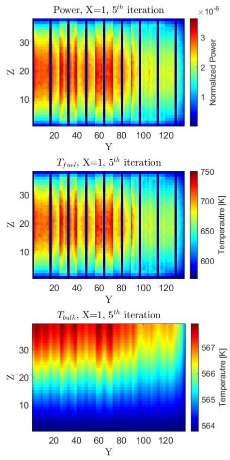

Figure 8. Converged power (top), fuel (middle) and coolant (bottom) temperature at the YZ plane, $\boldsymbol{X = 1}$

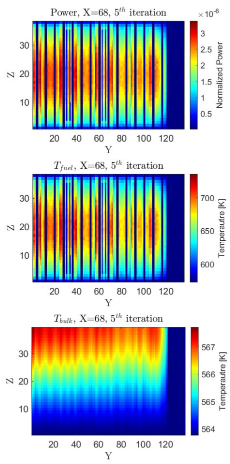

Figure 9 Converged power (top), fuel (middle) and coolant (bottom) temperature at the YZ plane, $\boldsymbol{X = 68}$

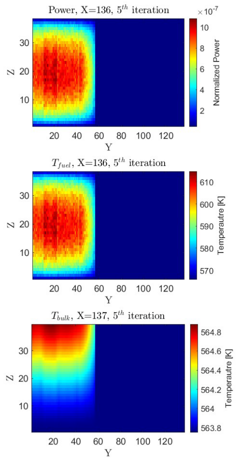

Figure 10. Converged power (top), fuel (middle) and coolant (bottom) temperature at the YZ plane, $\boldsymbol{X = 136}$

Figure 7 shows the axially integrated distributions of normalized power, fuel temperature, and coolant temperature. Figures 8–10 present the converged distributions at selected YZ planes. The planes at $X = 1$, $X = 68$, and $X = 136$ correspond to the mid-plane, quarter-point, and the edge of the core, respectively. All results in Figures 7–10 were obtained from the iDTMC method with a single active cycle. Overall, the fuel and coolant temperature distributions are consistent with the power distribution.

To examine whether the solution depends on the number of active cycles, we quantify the similarity between temperature fields as Equation (3).

$$\Delta T_x^{p,q} = \frac{\left|T_x^p - T_x^q\right|}{\bar{T}_x} \; where \; x = fuel, bulk \qquad (3)$$

Here, T_x^p denotes the temperature distribution of fuel or coolant computed with p active cycles in the neutronics calculations, and $\bar{T}_x$ is the average temperature distribution over all considered numbers of active cycles. $\Delta T_x^{p,q}$ is evaluated over the computational mesh, and we report its mean and standard deviation as summary at Table 3.

Table 3. $\boldsymbol{\Delta T_{fuel}^{p,q}}$ and $\boldsymbol{\Delta T_{bulk}^{p,q}}$

$(\boldsymbol{p,q})$	$(\boldsymbol{1,10})$
$\overline{\Delta T_{fuel}^{p,q}}$ [‰]	3.331 ± 2.812
$\overline{\Delta T_{bulk}^{p,q}}$ [‰]	0.009 ± 0.009

Comparing results for 1 and 10 active cycles, the fuel and coolant temperature distributions are effectively indistinguishable. The relative differences are negligible compared with the variance of the power distribution. To quantify this more rigorously, the variance of the thermal-hydraulic fields should also be calculated.

4. Conclusions

In this study, neutronics–thermal hydraulics coupled simulations of the APR1400 core were carried out using the iDTMC/START framework. Leveraging the efficiency of the iDTMC method, the framework enabled high-fidelity coupling at reduced cost.

In the full-core APR1400 calculations, the Picard iteration reduced the errors of the fuel and coolant temperatures to about 4×10^{-3} and 10^{-5}, respectively, within only a few iterations; because iDTMC is stochastic, these errors approach a finite value rather than vanishing with further iterations. The converged eigenvalue and thermal-hydraulic solutions were mutually consistent, indicating that the iDTMC/START framework reliably transfers information between neutronics and thermal-hydraulics. Importantly, varying the number of active cycles in the iDTMC method did not affect the converged solution. As summarized in Tables 2 and 3, the eigenvalues remain statistically consistent across cases, and the relative differences in the converged temperature fields are approximately 3‰ for fuel and 0.01‰ for coolant. This demonstrates that, in the coupled iDTMC/START simulation, using a single active cycle can deliver a consistent solution while significantly improving computational efficiency.

In conclusion, the iDTMC/START framework provides an efficient approach for high-fidelity multi-physics analysis of large reactor. Future research will focus on variance estimation of thermal-hydraulic quantities and their application, and compare efficiency with pCMFD-accelerated MC.

ACKNOWLEDGMENTS

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