Multi-physics Solution of VERA Core Physics Benchmark using CUPID and nTER

Youyeon Choi [a *], Hyoung Kyu Cho [a], and Jin Young Cho [b]
[a]*Department of Nuclear Engineering, Seoul National University, 1, Gwanak-ro, Gwanak-gu, Seoul 08826*
[b]*Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon 34057*
[*]*Corresponding author: choiyy14@snu.ac.kr*

1. Introduction

A high fidelity multi-physics analysis has attracted increasing attention with respect to economical reactor design and safety margin. The purpose of high fidelity simulation is to improve the economics of nuclear reactor design by reducing unnecessary safety margin and it can be achieved with a realistic high-resolution analysis tool [1]. With the availability of high-performance computing resources, there have been several studies to develop high fidelity multi-physics simulation platforms by coupling each physics code such as VERA [2] and MOOSE [3].

CUPID [4] is a three-dimensional two-phase thermal-hydraulics analysis code developed by KAERI, which adopts the two-fluid/three-field approach. The semi-implicit method is used for numerical scheme and CUPID is highly parallelized with a domain decomposition using MPI which enables to achieve economical computation time in subchannel scale analysis. In the previous studies, subchannel models such as turbulent mixing, void drift, and grid-directed cross-flow model were implemented in CUPID and subchannel analysis against APR1400 whole core was progressed [5, 6]. To perform an advanced high fidelity reactor core simulation, a whole core neutron transport code, nTER [7], was coupled with CUPID. The steady-state multi-physics solution for OPR1000 reactor core was introduced in the previous study [8].

In the present study, a coupled life-cycle calculation was performed using CUPID/nTER for VERA Core Physics Benchmark problem 9. During the simulation, CUPID provides thermal-hydraulics feedback while nTER performs a depletion calculation for a typical 18-month fuel cycle. The critical boron concentration was calculated as a result of coupling simulation and it was compared with the measured data provided in the report [9].

2. VERA Core Physics Benchmark

The VERA Core Physics Benchmark Problems were proposed by the Consortium for Advanced Simulation of Light Water Reactor (CASL) to provide a method for developing and demonstrating capabilities of reactor physics codes [9]. Each of the problems is based on the actual fuel and core geometries in Watts Bar Nuclear 1 (WBN1) initial core loading. In this study, problem 9 was selected for the validation of multi-physics simulation using CUPID and nTER. The specifications of the problem are illustrated in the following section.

2.1 Problem Specifications

Problem 9 represents the depletion of the fuel and burnable absorbers during a typical 18-month fuel cycle [9]. The type of fuel assembly in WBN1 is a 17x17 Westinhouse type and Fig. 1 shows the assembly and control rod layout used in the problem. Table 1 lists the given input conditions. In the present study, a quarter of the reactor core was modeled and simulated in the CUPID calculation. To validate the simulation results, the measured critical boron concentrations are provided [9] and compared with the CUPID/nTER coupling results.

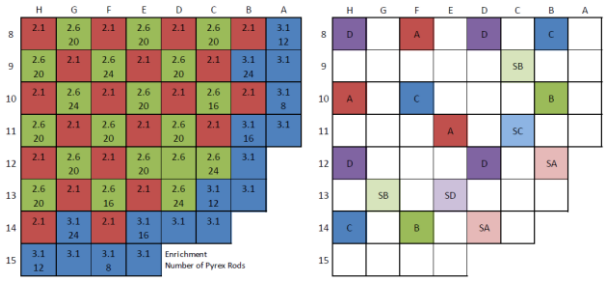

Fig. 1. Assembly and control rod layout for VERA Core Physics Benchmark problem 9 (quarter symmetry) [9]

Table 1: Problem 9 input specification [9]

Input	Value
Rated power (100%)	3411 MW
Rated coolant mass flow (100%)	16591.68 kg/s
Reactor pressure	15.51 MPa
Cycle length	441.0 EFPDs
EOC Exposure	16.939 GWd/MT

2.2 Coupling Scheme

In a power reactor simulation, thermal-hydraulics feedback is required for neutronics calculation. In this study, thermal-hydraulics code, CUPID and neutron transport code, nTER were externally coupled through the socket-based server program. Both codes run simultaneously and the coupling variables are exchanged via a socket communication. In the current status, Picard iteration, which is one of the fixed-point iterations, was adopted as a data exchange method between CUPID and nTER. Using this method, the coupling variables are exchanged between two codes until the converged result is obtained. For the simulation of VERA Benchmark problem 9, the Picard iteration was progressed at each burnup step. The schematic of the CUPID/nTER coupled

simulation during the depletion calculation is illustrated in Fig. 2.

The spatial mapping was progressed to conserve the physical quantities due to the different geometrical modeling in CUPID and nTER. Both codes perform pin-by-pin scale analysis in a reactor core problem. However, CUPID uses channel-centered geometry while nTER uses rod-centered scheme. Therefore, as shown in Fig. 3, a single fuel rod in nTER was set to be faced with the adjacent four subchannels in CUPID.

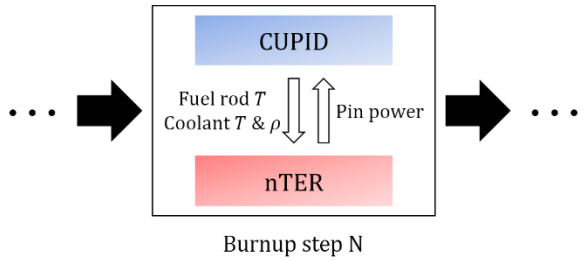

Fig. 2. CUPID/nTER coupling scheme in VERA Benchmark problem 9

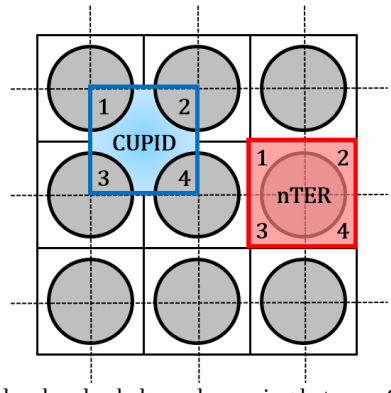

Fig. 3. Fuel rod and subchannel mapping between CUPID and nTER

3. Simulation Results

In this section, computational conditions for the coupling simulation are introduced. In addition, using CUPID/nTER, pin-by-pin coupling analysis was progressed against a quarter core of VERA Benchmark problem 9. The coolant and fuel rod temperature distributions were obtained in CUPID, while pin-wise power and burnup distributions were computed in nTER. Furthermore, critical boron concentration was calculated at each burnup step and the results were compared with the given measured data.

3.1 Simulation conditions

The computational mesh for CUPID was generated using the open-source tool, SALOME [10]. By adopting a quarter symmetry, 56 fuel assemblies of total 192 assemblies were modeled except for reflector region. The total number of fluid cells was 640,953 using 41 axially uniform meshes. Using the mass flow rate given in Table 1, the inlet liquid velocity was calculated and the initial and boundary conditions used in CUPID calculation are shown in Table 2.

Since the thermal-hydraulics properties such as fuel thermal conductivity and gap conductance were not specified in [9], MATPRO [11] thermal properties and dynamic gap conductance model [12], which were implemented in CUPID, were used in the thermal-hydraulics calculation. In addition, the simulation was also performed using thermal properties specified in NEACRP 3-D LWR Core Transient Benchmark [13], which is implemented in the default fuel rod model in nTER. In this paper, the result with NEACRP properties would be presented to compare with the result from nTER stand-alone (SA) calculation.

To guarantee a global convergence during the simulation, a user-defined damping factor was used. The damping factor was applied to the fuel rod temperature, which showed the most oscillating behavior among coupling variables, as follows:

$$T_{fuel}^{n+1} = \omega T_{fuel}^{n+1} + (1-\omega) T_{fuel}^{n} \qquad (1)$$

where n is a global iteration number and ω is a damping factor.

The power condition was fixed at hot full power (HFP) during the whole depletion calculation and the restart calculation was performed from 300 EFPD. The simulation was conducted in Linux cluster and the number of processors used for CUPID calculation was around 60, while for nTER was 73.

Table 2: Initial and boundary conditions [9]

Conditions	Value
Boundary pressure	15.51 MPa
Inlet liquid velocity	4.76 m/s
Initial liquid/solid temperature	565 K

3.2 Simulation Result

As a result of coupling simulation, three-dimensional distributions of coolant and fuel rod temperature at the operating condition were obtained in CUPID at each burnup step. Fig. 4 depicts the coolant temperature distributions at the beginning and end of the cycle. The pin-wise power distributions were calculated in nTER and Fig. 5 shows the 2-dimensional relative power distributions at the beginning and end of the cycle. Compared with the distribution at the beginning of the cycle, relative power distribution became flatten due to increase in fuel burnup. This result was well reflected in the coolant temperature distribution as shown in Fig. 4.

Fig. 6 shows the variation of axial power distribution along the fuel cycle. At the beginning of the cycle, the highest power was located in the middle of the core. Therefore, the fuel burnup in the middle region became higher than the other regions, which led to the result of relatively lower power in that region. For this reason, the axial power shape became flatten in the middle of the core as the fuel cycle progressed.

At each of the burnup steps, critical boron concentration was calculated and the results were compared with the given measured data and the nTER SA calculation results. Fig. 7 shows the comparison of a critical boron concentration between measured data, nTER SA and CUPID/nTER coupling results. Among the given measured data, only the data with the condition over 90% of full power were illustrated in the figure. For the whole fuel cycle, the CUPID/nTER results showed good agreement with the measured data.

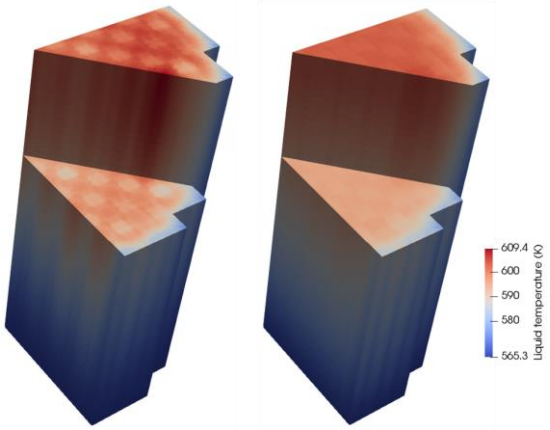

Fig. 4. The coolant temperature distributions at the beginning (left) and the end (right) of the cycle

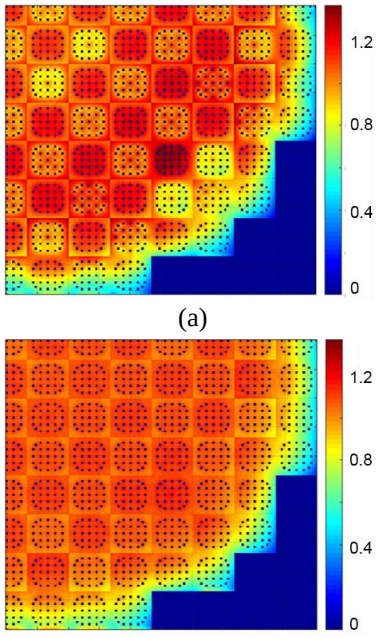

Fig. 5. 2-D relative power distribution at (a) the beginning and (b) end of the cycle

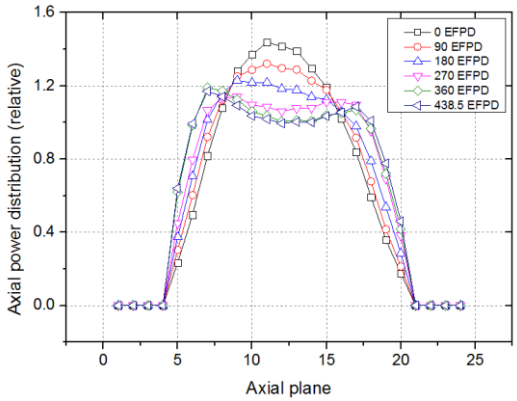

Fig. 6. Variation in axial relative power distribution along the cycle

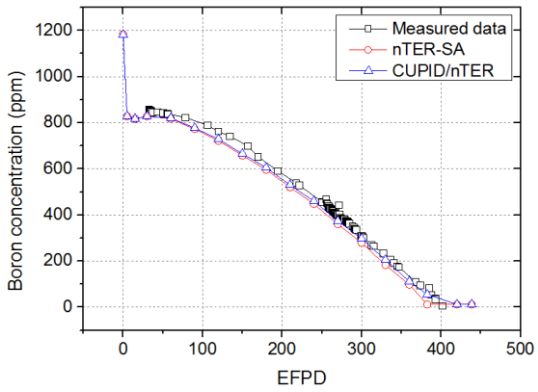

Fig. 7. Comparison of critical boron concentration between the measured data, CUPID/nTER, and nTER SA results

4. Conclusion

The three-dimensional thermal-hydraulics analysis code, CUPID, and whole core neutron transport code, nTER, were coupled and performed a life-cycle depletion calculation of VERA Core Physics Benchmark problem 9. A quarter of the reactor core was used under quarter symmetry assumption and both codes were externally coupled using Picard iteration with damping factor.

Throughout the simulation, CUPID provided subchannel scale thermal-hydraulics feedback to nTER, while nTER performed neutronics calculation such as pin-wise power, burnup and critical boron concentration at each burnup step. From the coupled simulation, the neutronics and thermal-hydraulics results at the operating condition were obtained along the fuel cycle.

For future work, additional validation against BEAVRS would be conducted and further coupling simulation for APR1400 reactor would be progressed.

ACKNOWLEDGEMENT

This work was supported by Korea Atomic Energy Research Institute (KAERI) and funded by Korea Hydro and Nuclear Power Central Research Institute (KHNP CRI).

REFERENCES

[1] Y. Ma, S. Liu, Z. Luo, S. Huang, K. Li, K. Wang, G. Yu, and H. Yu, RMC/CTF multiphysics solutions to VERA core physics benchmark problem 9, Annals of Nuclear Energy, Vol. 133, pp. 837-852, 2019.
[2] S. Palmtag, Demonstration of neutronics coupled to thermal-hydraulics for a full-core problem using VERA, CASL Technical Report: CASL-U-2013-0196-000, 2013.
[3] D. Gaston, C. Newman, G. Hansen, and D. Lebrun-Grandié, MOOSE: A parallel computational framework for coupled systems of nonlinear equations, Nuclear Engineering and Design, Vol. 239 (10), pp. 1768-1778, 2009.
[4] Korea Atomic Energy Research Institute, "CUPID Code Manuals Vol. 1: Mathematical Models and Solution Methods Version 2.2", 2018.
[5] S. J. Yoon, S. B. Kim, G. C. Park, and H. K. Cho, Application of subchannel-scale thermal-hydraulic analysis of pressurized water reactor core under single-phase conditions, Nuclear Engineering and Technology, Vol. 50, pp. 54-67, 2018.
[6] S. B. Kim, J. H. Lee, G. C. Park, and H. K. Cho, The im8plementation of Mixing Vane Directed Cross Flow Model in CUPID for Subchannel Scale T/H Analysis, Transactions of the Korean Nuclear Society Spring Metting, Jeju, Korea, May 17-18, 2018.
[7] J. Y. Cho et al., Performance of a Whole Core Transport Code, nTER, The 6th International Conference on Nuclear and Renewable Energy Resources (NURER 2018), Jeju, Korea, Sep. 30-Oct. 3, 2018.
[8] Y. Choi, H. K. Cho, and J. Y. Cho, Preliminary Multi-Physics Analysis fo OPR1000 Reactor Core using Coupled CUPID and nTER, Transactions of the Korean Nuclear Society Spring Metting, Jeju, Korea, May 23-24, 2019.
[9] A. T. Godfrey, VERA Core Physics Benchmark Progression Problem Specifications, Rev. 4, CASL Technical Report: CASL-U-2012-0131-004, 2014.
[10] A. Ribes, and C. Caremoli, Salome platform component model for numerical simulation, 31th Annual International Computer Software and Applications Conference (COMPSAC 2007), Beijing, China, pp.553-564, 2007.
[11] W. G. Luscher, K. J. Geelhood, and I. E. Porter, Material Property Correlations: Comparisons between FRAPCON-4.0, FRAPTRAN-2.0, and MATPRO, PNNL-19417 Rev. 2, 2015.
[12] Korea Atomic Energy Research Institute, "MARS Code Manual Volume 1: Code Structure, System Models, and Solution Methods", KAERI/TR-2812/2004, 2004.
[13] H. Finnemann, and A. Galati, NEACPR 3-D LWR Core Transient Benchmark Final Specification, NEACRP-L-335 (Revision 1), 1992.