Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 18-19, 2023

Preliminary Multi-Physics Analysis of a 2x2 Rod Array Using CUPID/GIFT Coupled Code

Yeong Hun Lee, Kyuseok Sim, Youho Lee, Hyoung Kyu Cho*
Department of nuclear engineering, Seoul National Univ., 1 Gwanak-ro, Gwanak-gu, Seoul 08826
**Corresponding author: chohk@snu.ac.kr*

1. Introduction

With an increase in discharge burnup of pressurized water reactor (PWR) fuels, the fuel pellet and cladding undergo embrittlement, which can have detrimental effects on their integrity under both normal operating conditions and postulated accident conditions [1]. In this regard, the U.S. NRC has established new fuel safety criteria for design basis accidents such as loss-of-coolant accidents (LOCA) and reactivity-initiated accidents (RIA) [2,3]. KINS also has been making efforts to revise the emergency core cooling system (ECCS) acceptance criteria [4]. As high burnup fuels experience deformation such as creep, swelling, relocation, and oxidation formation during operation, the multi-physics simulation, including fuel behavior, is essential for safety analysis.

Recently, KINS developed MARS/FRAPTRAN, a system analysis code MARS coupled with NRC's transient fuel performance (F/P) code FRAPTRAN [5]. In addition, KAERI developed CUPID/FRAPTRAN, a subchannel thermal hydraulics (T/H) code with FRAPTRAN [6]. CUPID/FRAPTRAN can simulate more realistic phenomena occurring in the reactor vessel as it is based on a three-dimensional T/H model, which accounts for both axial and lateral flow in subchannels. For safety analysis using CUPID/FRAPTRAN, burnup-dependent parameters are initialized with steady-state F/P code results. In the previous study of CUPID/FRAPTRAN, however, fuel rod conditions obtained from FRAPCON analysis were applied to whole fuel rods as the initial conditions. For more realistic simulation, pin-wise initial conditions are required and it became the motivation of this study. In the present study, CUPID was coupled with a F/P code, GIFT, which is developed at Seoul National University, to generate pin-wise initial conditions for transient T/H-F/P coupled analysis.

2. Code descriptions

2.1 Subchannel analysis code CUPID

CUPID is a three-dimensional, transient T/H code developed by KAERI and it incorporates a subchannel module called CUPID-RV for a reactor vessel and core simulation in subchannel scale [7]. Several subchannel models, including crossflow, turbulent mixing model, and void drift model were implemented to enhance subchannel analysis capability [8]. In the previous study, the subchannel module of CUPID was employed to conduct multi-physics simulations including CUPID/nTER coupling with neutron transport code [9] and CUPID/FINIX coupling with F/P code [10]. The subchannel analysis capability during established multi-physics simulation has been demonstrated.

2.2 Fuel performance code GIFT

GIFT is an in-house F/P code developed at Seoul National University. It is designed to simulate steady-state fuel behavior in PWR. The phenomena that are modeled by GIFT include thermal conduction, fuel deformation, fission gas release, and cladding oxidation. In particular, GIFT calculates robustly cladding mechanical deformation considering radial and axial interaction. The calculation method including axial interaction has a negligible effect under general conditions, but it is crucial when rapidly changing temperatures and stress occurred such as reflood and burst. Additionally, GIFT can generate initial conditions including burnup history results for transient fuel rod analysis. These initial conditions generated by GIFT can be used for FRAPTRAN.

2.3 Coupling methodology

CUPID/GIFT is externally coupled using TCP/IP socket programming, allowing data transfer between two different codes. A simple schematic of the coupled code is shown in Figure 1, which includes an interface program called GIFT2CPD. This interface program controls the socket communication and transfers coupled variables such as cladding radius and temperature, coolant temperature, and cladding-to-coolant heat transfer coefficient to each code. GIFT was initially designed for a single fuel rod analysis but has been parallelized using MPI programming to simulate multiple fuel rod analysis.

As shown in Figure 2(a), CUPID has subchannel-to-rod connectivity and the rod is surrounded by four subchannel cells. Thus, mapping is necessary to transfer accurate T/H data to the rod and reflect F/P feedback from the rod. Figure 2(b) shows that the averaged data of the four cells surrounding the rod is sent to GIFT. On the other hand, CUPID receives the sum of a quarter of the four adjacent rods as shown in Figure 2(c). During the simulation, GIFT provides power for each fuel rod as an input and calculates the fuel behavior including T/H feedback. CUPID performs thermal-hydraulic calculations that consider F/P feedback, which involves geometry and heat flux changes. The fuel rod's deformation influences the porosity of the cell and the hydraulic diameter, which are key parameters for the T/H calculation. Moreover, CUPID calculates the heat flux transferred to the subchannel cell using its own heat structure model, which uses the received cladding radius and temperature. In addition, GIFT is a steady-state analysis code that calculates values for a particular time step, whereas CUPID is a transient analysis code. In order to achieve convergence between the two codes, the coupled variables are iteratively transferred for a single time step in GIFT as shown in Figure 3.

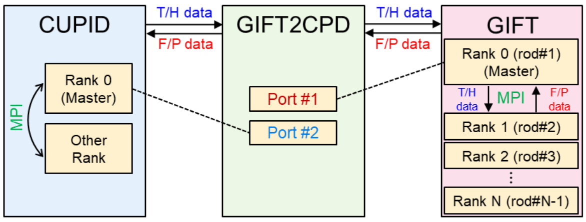

Figure 1. Schematic of CUPID/GIFT coupling

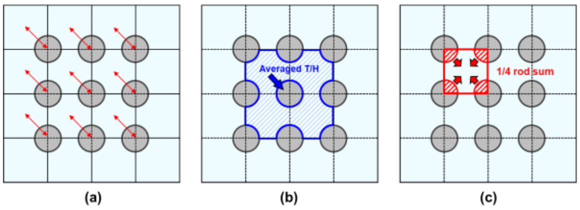

Figure 2. CUPID (a) subchannel-to-rod connectivity, (b) averaged T/H data transfer, (c) sum of fuel data transfer

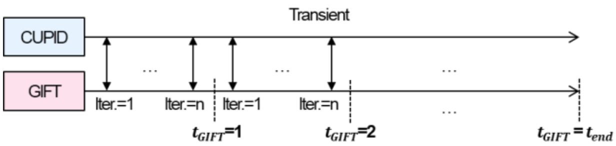

Figure 3. Calculation procedure of CUPID/GIFT

3. Calculation results

3.1 Problem description

For verification calculation, a 2×2 rod array was simulated. The rod array has a total height of 3.6576 m and employs 40 meshes in the axial direction for both CUPID and GIFT. Each rod is 9.5 mm in diameter with a rod pitch of 12.6 mm. The detailed geometry and size of the rod array is summarized in Figure 4. The boundary conditions of CUPID and input variables of GIFT are summarized in Table 1 and Table 2, respectively. The power conditions of the simulation were obtained by CUPID/nTER depletion calculation results for the first cycle of VERA benchmark problem #9. The average power profile for each rod is shown in Figure 5. Figure 6 shows the axial power distribution of rod #1, where the power has a cosine shape at the beginning of the cycle (BOC) and flattened at the end of the cycle (EOC).

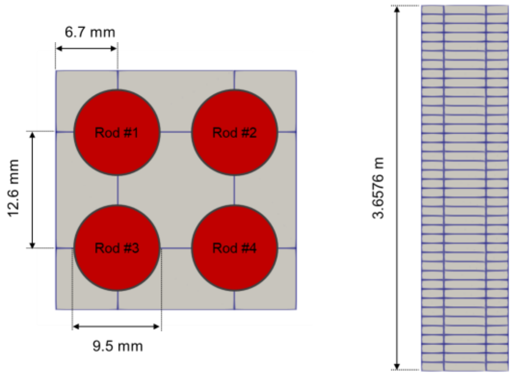

Figure 4. Geometry and size of a 2×2 rod array

Table 1. CUPID boundary condition

Parameter	Value
Pressure	15.513 MPa
Inlet temperature	565 K
Inlet velocity	4.76 m/s

Table 2. Design parameter of a fuel rod in GIFT

Parameter	Value
Cladding diameter	9.5 mm
Cladding thickness	0.57 mm
Gap thickness	0. 084 mm
Pellet roughness	2.0E-3 mm
Cladding roughness	5.0E-4 mm
Fill gas property	Helium (100%)
Fill gas pressure	1.27 MPa
Grain size	1.0E-2 mm
Cold work	0.5
Fast neutron flux	2.46E+16 (#/m^{2}s)/(W/g)

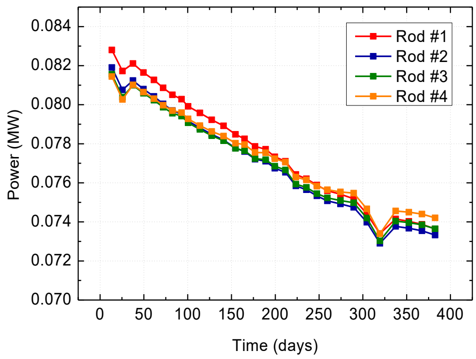

Figure 5. Average power profile for each rod

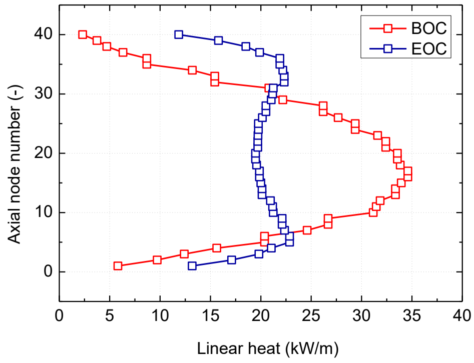

Figure 6. Power distribution at the BOC and EOC

3.2 Results and discussion

The total power of the CUPID/GIFT is compared with the CUPID result to evaluate the overall accuracy of the calculation. As shown in Figure 7, the calculation results match well, although there are some peaks in the coupled code result. These peaks are due to the iteration required for convergence between the two codes and it does not affect the calculation results as the converged results were merely used for the analysis.

The fuel radius profiles of rod #1 obtained using the coupled code at the BOC and EOC are shown in Figure 8. The fuel pellet undergoes expansion mainly due to thermal expansion and swelling, while the cladding experienced contraction due to creep. The pellet-to-cladding gap closure is observed in the middle of the fuel. Figure 9 shows the radial temperature distribution of rod #1 at the 20th axial node at the middle of the cycle (MOC) and EOC. The section where the temperature rapidly changes between the pellet and cladding disappears because the gap is closed in both cases during coupling calculation. Therefore, the fuel centerline temperature of coupled code decreases compared to that of CUPID.

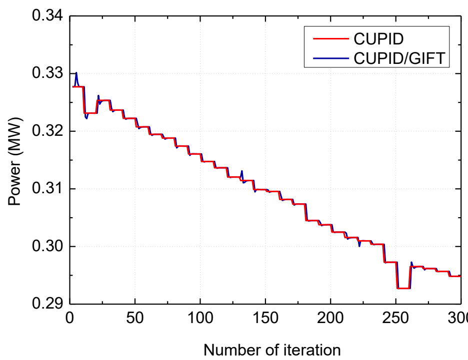

Figure 7. Total calculated power during the simulation

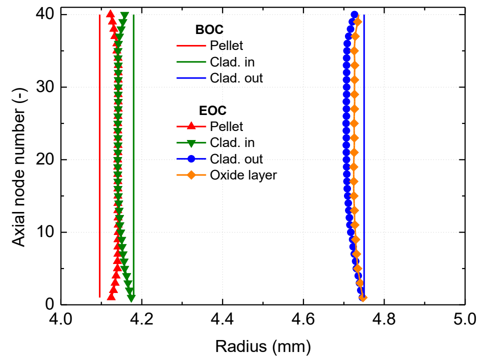

Figure 8. Results of fuel radius at the BOC and EOC

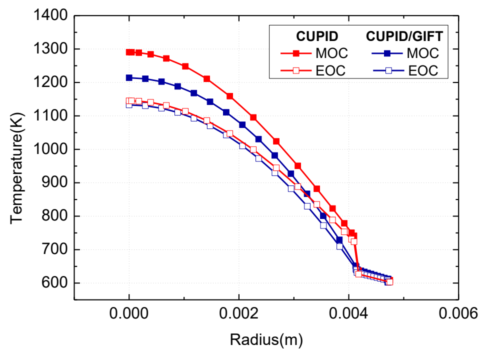

Figure 9. Results of fuel temperature at the MOC and EOC

The fuel deformation also influences the flow area. Figure 10 shows the coupled code result for hydraulic diameter and coolant velocity at the center subchannel of a 2x2 rod array at the BOC and EOC. The hydraulic diameter increases due to outer cladding contraction, resulting in decreased coolant velocity. The fuel deformation is well considered during subchannel T/H calculation. Figure 11 shows the difference in average coolant temperature surrounding each rod between coupled code and GIFT. The coolant temperature decreases at relatively higher power and increases at relatively lower power. The mixing among subchannels flattens the coolant temperature.

GIFT calculates mechanical gap and gap interface pressure to analyze the pellet-cladding mechanical interaction (PCMI). Figure 12 shows the mechanical gap and gap interface pressure of rod #1 during the cycle at the 21st axial node. The mechanical gap closure difference between GIFT and coupled code result is approximately 50 days, and the gap interface pressure also has different behavior. The cladding plastic strain results of rod #1 are shown in Figure 13, and coupled code has smaller strains compared to GIFT. These differences would be considered as an initial condition for transient analysis.

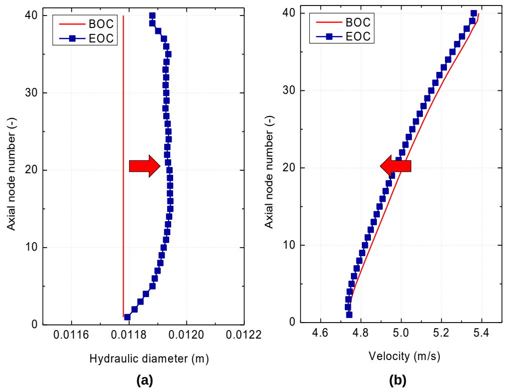

Figure 10. Results of (a) hydraulic diameter, (b) coolant velocity at the BOC and EOC

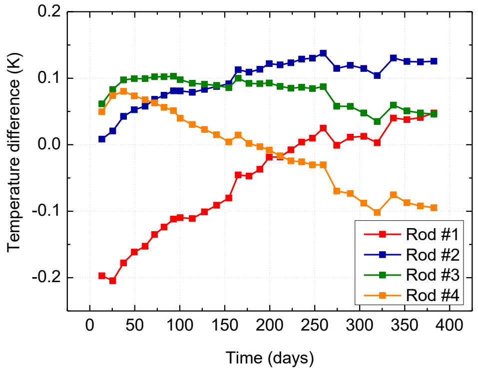

Figure 11. Difference in average coolant temperature surrounding each rod

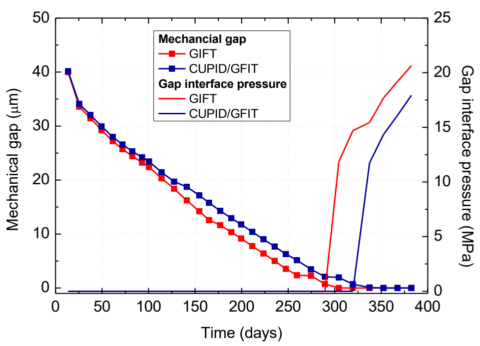

Figure 12. Results of mechanical gap width and gap interface pressure during simulation

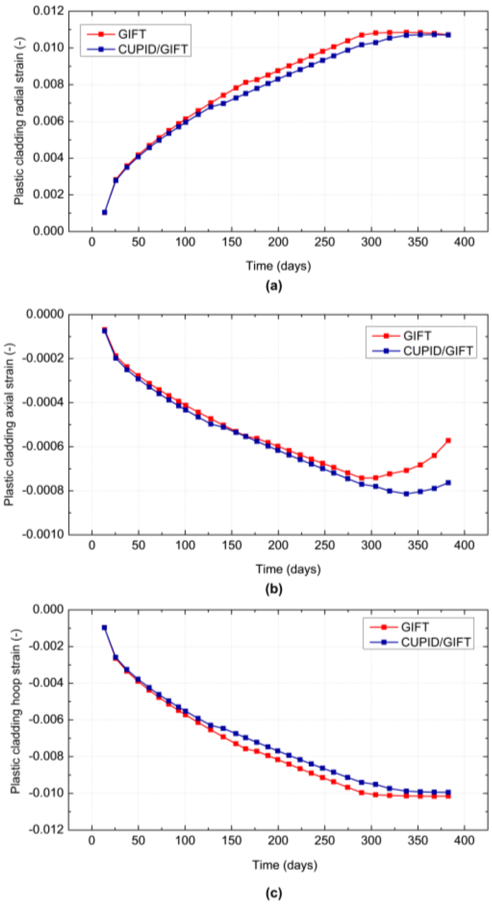

Figure 13. Results of plastic cladding strain (a) radial direction, (b) axial direction, (c) hoop direction

4. Conclusions

In this study, CUPID/GIFT coupling was established and the verification calculation was performed for a 2x2 rod array. The calculation results of the coupled code showed reasonable results physically and the fuel deformation was appropriately considered in the T/H calculation. The coupled simulation provided different results of the mechanical gap and several strains by considering mixing among subchannels and flattened fluid temperature.

The coupled code is expected to provide more accurate pin-wise initial conditions for a transient analysis using CUPID/FRAPTRAN.

ACKNOWLEDGEMENT

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety, granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea (No. 2106021).

REFERENCES

[1] J.S. Lee, I.S. Lee, and A.J. Cheong, "Development of Regulatory Audit Technology for Design Basis Accidents with Fuel Burnup Effects", Transactions of Korean Nuclear Society Spring Meeting, Jeju, Korea, May 17-18, 2018.

[2] US NRC, "Performance-Based Emergency Core Cooling System Requirements and Related Fuel Cladding Acceptance Criteria", March, 2016.

[3] NRC, DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents", Nov. 2016.

[4] J.S. Lee et. al., "Technical Basis for the Revision of ECCS Acceptance Criteria of Comestic PWR Plants", KINS/RR-1686, KINS, 2017.

[5] H.C. Kim et al., "Development of Fully Coupled MARS-KS/FRAPTRAN Code System for Simulation of Fuel Behavior during LOCA," Transactions of the Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, October 26-27, 2017.

[6] J.R. Lee and H.Y. Yoon, "Pin-wise Fuel Behavior in Non-LOCA by Multi-Scale and Multi-Physics Approach", Transactions of the Korean Nuclear Society Autumn Meeting, Changwon, Korea, October 20-21, 2022.

[7] H.Y. Yoon et.al., "A Multiscale and Multiphysics PWR Safety Analysis at a Subchannel Scale", Nuclear Science and Engineering, Vol.194(8-9), pp.633-649, 2020.

[8] S.J. Yoon et al., "Application of CUPID for Subchannel-Scale Thermal-Hydraulic Analysis of Pressurized Water Reactor Core Under Single-Phase Conditions," Nuclear Engineering Technology, Vol.50(1), pp.54-67, 2018.

[9] Y.Y. Choi, "Establishment and Validation of Reactor Core Simulator for Subchannel Thermal-hydraulic and Neutron Transport Coupled Analysis", Master Thesis, Seoul National University, 2020.

[10] J.K. Park, "Coupling between Subchannel Module of CUPID and Fuel Code FINIX for Multi-physics Transient Analysis of PWR Core", Master Thesis, Seoul National University, 2021.