Multi-physics Coupled Reactor Core Analysis System of RAST-K2.0 with CTF and FRAPCON

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1. Introduction

In the past decades, the importance of best estimate analysis of nuclear reactors by a multi-scale, multi-physics code system has increased for nuclear safety analysis in nuclear engineering. Consortium for Advanced Simulation of Light water reactors (CASL) [1] project has been conducted by the US DOE to develop the capability of advanced modeling and simulation tools for improved performance of currently operating Light Water Reactors (LWR), and the MPACT [2] code has been coupled with CTF as a part of the CASL project. Seoul National University (SNU) has established a multi-physics coupled code system based on nTRACER [3].

For nuclear reactor design and safety analysis of LWRs, Ulsan National Institute of Science and Technology (UNIST) CORE group has developed a two-step approach nuclear reactor analysis code system called STREAM/RAST-K2.0 [4], [5]. The nodal code (RAST-K2.0) was coupled with a thermal/hydraulics (TH) code and a fuel performance (FP) code to construct a multi-scale, multi-physics analysis code system. The sub-channel TH code CTF [6] and the FP code FRAPCON [7] were selected to be coupled. Chapter 2 will describe the computational codes representing neutronic (N), TH, and FP. Chapter 3 will describe the coupled calculation scheme. Chapter 4 will present numerical results for the first cycle of the OPR1000 reactor.

2. Computational Codes

2.1. Neutronic code

RAST-K 2.0 [4] code is a reactor core analysis code being developed at UNIST for in-core fuel management study, core design calculation, load follow simulation, and transient analysis in neutronics. It solves a nodal diffusion equation by using a 3-dimensional 2-group UNM (unified nodal method), and it adopts CRAM (Chebyshev Rational Approximation Method) with a micro depletion method for the depletion calculation. The 2-group cross section and group constant data are provided by STREAM, which is a lattice physics code also developed at UNIST. Due to the functionalized cross section model in RAST-K2.0, cross section feedback is utilizable.

STREAM [5] solves the transport equation to generate the nuclear data of fuel assembly and reflector models used in RAST-K2.0 by 2-dimensional MOC. By adopting a Pin based Slowing down Method (PSM) as the resonance treatment method, STREAM can obtain higher accuracy of numerical results.

STORA code links STREAM and RAST-K2.0 by gathering STN files containing cross section and group constant data calculated by STREAM and reformatting it for use in RAST-K2.0. Fig. 1 presents a flowchart of the STREAM/RAST-K2.0 code system.

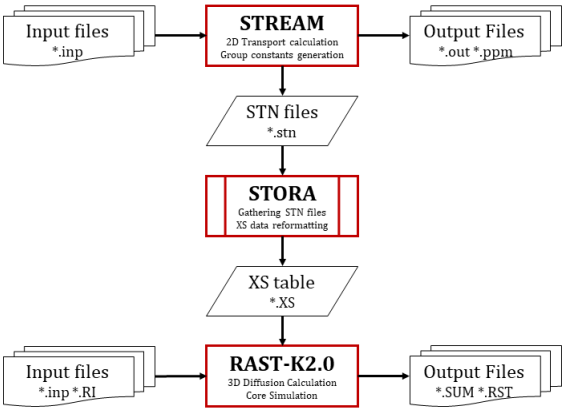

Fig. 1. Two-step flowchart of ST/R2 code system.

2.2. Thermal/Hydraulics code

CTF [6], originally developed by Northwest Laboratory in 1980, is a TH simulation code designed for LWR vessel analysis. It is available for solving sub-channel forms of 9 conservation equations by using a two-fluids, three-field (fluid film, fluid drops, and vapor) modeling approach. Because CTF models multi-rod arrays in the reactor core, channel to channel flow (cross-flow) can be considered in the simulation, giving more accurate coolant properties than other TH codes which simulate single fuel rods. Because CTF provides a module that converts channel-centered channel index to rod-centered channel index and a coupling interface module, it can be easily coupled with a neutronics code. MPI based parallelization accelerates the simulation.

2.3. Fuel performance code

FRAPCON [7] calculates the steady-state, thermal mechanical response of oxide fuel rods in LWRs during long-term burnup conditions which occur during normal power reactor operation. For each time step, 1) heat conduction in the axial direction is calculated by using

user-defined boundary conditions to determine the coolant bulk temperature. 2) The temperatures of fuel and cladding are determined by heat transfer calculation from the cladding surface to the fuel, using the coolant temperature as boundary condition. 3) Deformation of fuel and cladding, and 4) fission product generation and release are computed. Fig. 2 shows the FRAPCON calculation flowchart of a single time step. Accurate fuel temperature can be obtained because oxide fuel property changes are taken into account in the heat transfer calculation. Moreover, other parameters, such as fission gas release, cladding corrosion, cladding hoop strain, and gap thickness are available to be quantified. Steady-state fuel behavior calculated by FRAPCON can be used as the initial input condition for FRAPTRAN, which analyzes transient fuel behavior.

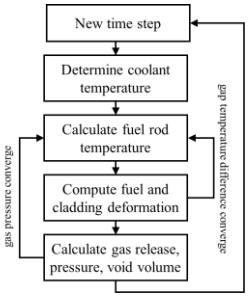

Fig. 2. FRAPCON calculation in a time step

3. Coupled Calculation Scheme

For the coupling of the multi-physics analysis system, CTF and FRAPCON are linked to RAST-K2.0 with a static library feature. During the coupled calculation, CTF input files should be created. Otherwise, the main program of FRAPCON is converted to a subroutine and input parameters are entered when calling FRAPCON without making an input file. Since FRAPCON performs analysis on a single pin, the subroutine is called for every fuel rod in the model. In the current coupling, FRAPCON simulates each of the burnup steps from fresh fuel. i.e., when RAST-K2.0 calculates at the n^{th} burnup step, FRAPCON simulates from the 1^{st} burnup step to figure out the fuel behavior in the n^{th} burnup step.

The multi-physics coupled calculation scheme is shown in Fig. 3. For each burnup step, the 3-dimensional node power distribution is computed by solving the nodal diffusion equation. Using the form function generated by STREAM, RAST-K2.0 reconstructs the pin-by-pin power distribution from the node power distribution and provides it to CTF and FRAPCON. At the first call of CTF, preprocessor inputs are generated and the CTF preprocessor tool creates an input deck for the CTF calculation model. Using the pin power distribution, the sub-channel TH calculation is performed. Coolant temperature and density are transferred into RAST-K2.0, and cladding surface temperature is moved to FRAPCON. The axial heat conduction calculation at the beginning of the FRAPCON calculation is skipped. Instead, the cladding surface temperature from CTF is used as the boundary condition. FRAPCON supplies the fuel rod temperature to RAST-K2.0 as well. While the TH properties of the coolant and fuel rods from CTF and FRAPCON are pin-wise, the RAST-K2.0 nodal calculation is node-wise, with mesh size greater than at the fuel pin level. Therefore, node average TH properties are computed and are used to update the cross section data. Such a coupled calculation is repeated until critical boron concentration (CBC) is converged.

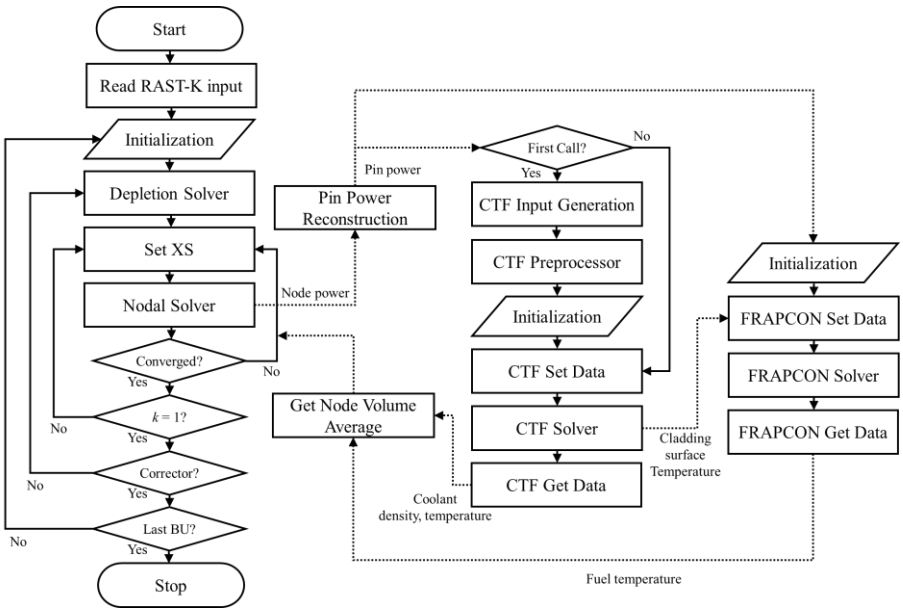

Fig. 3. Flowchart of coupled calculation.

4. Numerical Tests

4.1. Modeling condition

A numerical test of the coupled calculation was performed on the first cycle of a typical OPR1000 reactor. Fig. 4 shows the loading pattern of the core. It consists of 177 fuel assemblies (FA) with a 16 by 16 array of 236 fuel pins and 5 guide tubes. The fuel enrichment varies according to assembly types A, B, and C. Axially, 46 fuel meshes and 2 reflector meshes were modeled in the neutronics calculation, and 10 meshes are used in the TH and FP model (only for the fuel region). Each length of 6 TH and FP meshes at the middle corresponds to 5 neutronics meshes, and that of the other meshes at the top and bottom correspond to 4 neutronics meshes. From BOC (beginning of cycle) condition, 17 burnup steps are analyzed up to 13.8 GWd/MTU. The total core power is 2,815 MWth, the coolant total flow rate is 16,315 kg/sec, and the inlet temperature is 296.1 °C. Because the calculation model is Cycle 1 of the OPR1000, fresh fuel rods are loaded at BOC.

`	H	J	K	L	M	N	P	R
8	A	A	A	B	A	A	B	C
9	A	A	C	A	C	A	C	B
10	A	C	A	B	A	B	C	B
11	B	A	B	A	B	A	C	
12	A	C	A	B	B	C	B	
13	A	A	B	A	C	C		
14	B	C	C	C	B			
15	C	B	B					

Fig. 4. Loading pattern of OPR1000 Cycle 1

4.2. Numerical results

The numerical parameters which are exchanged during the coupled calculation are: pin power distribution, coolant temperature and density, and fuel temperature by RAST-K2.0, CTF, and FRAPCON, respectively. Fig. 5 to Fig. 8 show the exchanged parameters for BOC at 0.0 GWd/MTU, middle of cycle (MOC) at 6.0 GWd/MTU, and end of cycle (EOC) at 12.0 GWd/MTU.
During the simulation, the maximum obtained fuel temperature is 1,025 °C at BOC, and as the burnup increases, the spatial distribution of pin properties is mitigated.

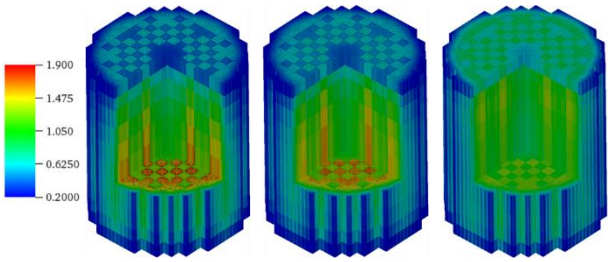

Fig. 5. Relative power distribution (-) from RAST-K2.0

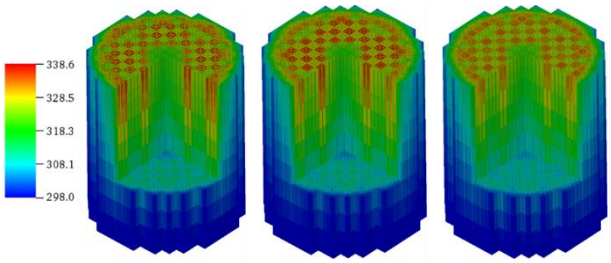

Fig. 6. Coolant temperature (°C) from CTF

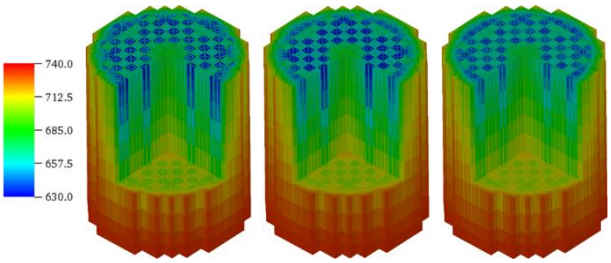

Fig. 7. Coolant density (g/cc) from CTF

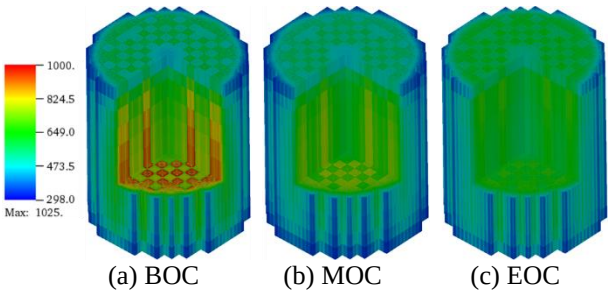

Fig. 8. Fuel temperature (°C) from FRAPCON

Other fuel properties computed by FRAPCON, shown in Fig. 9 to Fig. 11, are: gap thickness, ZrO_2 oxide thickness, and gap conductance.

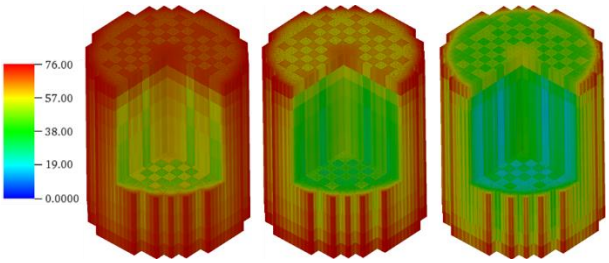

Fig. 9. Gap thickness (micron) during Cycle 1

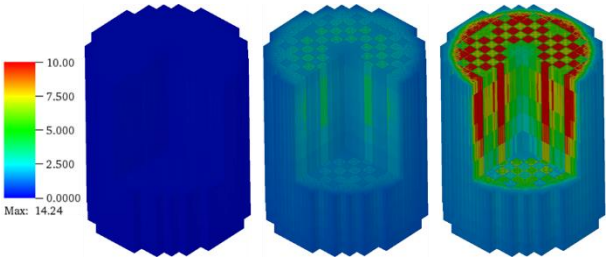

Fig. 10. ZrO_2 oxide thickness (micron) during Cycle 1

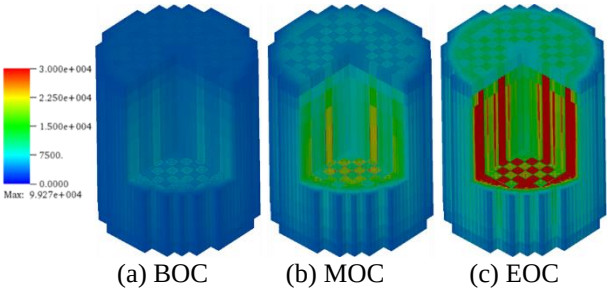

Fig. 11. Gap conductance (W/m^2-K) during Cycle 1

5. Conclusions

A steady-state multi-physics coupled reactor core analysis code system has been successfully established. UNIST nodal diffusion code (RAST-K2.0) was fully coupled with a sub-channel TH code (CTF) and a steady-state FP code (FRAPCON). For every burnup step, RAST-K2.0 provides pin power distribution to CTF and FRAPCON. CTF computes the coolant temperature and density for RAST-K2.0, and the cladding surface temperature to use as boundary condition in the FRAPCON heat transfer calculation in the fuel rod. FRAPCON provides fuel temperature to RAST-K2.0. Using coolant temperature, density, and fuel temperature, RAST-K2.0 updates the neutron cross section. The coupled calculation is repeated until certain parameters, k-value with boron concentration, are converged. A numerical test was performed on cycle 1 of a typical OPR1000 reactor, and pin power distribution, coolant temperature and density, and fuel temperature were computed for parameter exchange among the codes. In addition, gap thickness, oxide thickness, gap conductance, and hoop strain were compared at BOC, MOC, and EOC.

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REFERENCES

[1] The Consortium for Advanced Simulation of Light Water Reactors, www.casl.gov, accessed: 3/15/2018.
[2] B. Kochunas, et al., Demonstration of Neutronics Coupled to Thermal-Hydraulics for a Full-Core Problem using COBRA-TF/MPACT, CASL-U-2014-0051-000, April 1, 2014.
[3] Y.S. Jung, C.B. Shim, C.H. Lim, and H.G. Joo, Practical numerical reactor employing direct whole core neutron transport and subchannel thermal/hydraulics solvers, Annals of Nuclear Energy, Vol. 62, pp. 357-374, 2013.
[4] S. Choi, C. Lee, and D. Lee, Resonance treatment using pin-based pointwise energy slowing-down method, Journal of Computational Physics, Vol. 330, No.1, pp. 134-155, 2017
[5] J. Park, M. Park, J. Choe, P. Zhang, J. Jang, and D. Lee, Development Status of Dynamic Reactor Nodal Computational Code RAST-K v2.0, RPHA17, Chengdu, Sichuan, China August 24-25, 2017.
[6] M.N. Avramova, R.K. Salko, CTF–A Thermal Hydraulics Subchannel Code for LWRs Transient Analyses, User's manual, CASL-U-2015-0055-000, March 10, 2015.
[7] K. J. Geelhood, et al., FRAPCON-4.0: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup, PNNL-19418, Vol. 1, Rev. 2, September 2015.