Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 23-24, 2019

Multi Physics Coupling of RAST-K for Multi-cycle Core Calculation

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1. Introduction

A nuclear reactor system is a system where various physical phenomena are interrelated. For this reason, there has been increasing interest in best estimate analysis of nuclear reactors by a multi-physics code system for licensing and 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. In the framework of the EU funded NURESAFE project, CTF [3] and neutronics code DYN3D has been integrated and coupled on the Salomé platform [4].

Ulsan National Institute of Science and Technology (UNIST) has developed a nodal diffusion computational code RAST-K [5] sponsored by Korea Hydro & Nuclear Power Central Research Institute (KHNP-CRI). To perform high-fidelity of nuclear reactor core analysis, the RAST-K has been integrated and coupled with the subchannel thermal hydraulics (TH) code, CTF and steady state fuel performance code, FRAPCON [6].

2. Code Description

2.1. Reactor core analysis code, RAST-K

RAST-K utilizes the non-linear scheme based on multi-group coarse mesh finite difference acceleration with three-dimensional multi-group unified nodal method to solve steady state and transient problems with assembly-wise nodes. The lattice physics code STREAM [7] generates nuclear data of fuel assembly and reflector models used in the RAST-K. Chebyshev Rational Approximation Method (CRAM) with micro depletion method are implemented for the depletion calculation and the predictor-corrector method improves the accuracy of the depletion calculation. The pin power reconstruction method is applied in RAST-K for pin-level resolution of the power distribution.

2.2. Subchannel thermal hydraulics code, CTF

CTF [6], originally developed by Northwest Laboratory in 1980, is a TH simulation code designed for LWR vessel analysis. It is available to solve sub-channel forms of 8 conservation equations by using a two-fluids, three-field (fluid film, fluid drops, and vapor) modeling approach. CTF has capability to calculate channel to channel flow (cross-flow) during the simulation, giving more accurate coolant properties than other TH codes which simulate single fuel rods. CTF provides coupling interface to be coupled with external code consisting of various functions from initialization, setting variables, running simulation, getting the results, and finalization. A module that converts channel-centered variables to rod-centered variables make it easy to be coupled with a neutronics code. MPI based parallelization is implemented in CTF for acceleration of the simulation.

2.3. Fuel behavior code, FRAPCON

FRAPCON [6] calculates the steady-state, thermo-mechanical response of a nuclear fuel rod in LWRs during long-term burnup conditions of 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. Accurate fuel temperature can be obtained because oxide fuel property changes are considered 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.

3. Multi-physics coupling feature

In the multi-physics coupled code system, CTF and FRAPCON are linked to RAST-K. At beginning of the coupled calculation, CTF input files should be generated. A fuel rod analysis interface (FRAPI) has been developed in UNIST. Restart capability with multi-rod analysis has been implemented the FRAPI. The FRAPI module provides interface for external a code to control the FRAPCON including initialization, finalization, setting and getting variables, saving and loading states, writing and reading restart files for a state, writing restart for transient calculation by FRAPTRAN. The FRAPI makes it easy to couple with neutronics and thermal-hydraulics code in an iterative coupling scheme. In the coupled system, burnup calculations of RAST-K and FRAPCON are performed separately to preserve accuracy of their own depletion modules. A predictor-corrector coupled depletion scheme is introduced in FRAPCON calculation to maintain the consistency of burnup models as shown in Table 1. The flowchart of the multi-physics coupled calculation is shown in Figure 1. For each burnup step, the 3-dimensional pin-wise power distribution is computed by solving the nodal diffusion equation and pin power reconstruction method. provides it to CTF and FRAPCON. CTF computes coolant properties by sub-channel TH calculation by using the pin-wise power distribution. Coolant temperature and density are passed to RAST-K, and bulk coolant temperature and coolant pressure is moved to FRAPCON. Because bulk temperature and coolant pressure are provided as fuel rod boundary condition, the axial heat conduction calculation of the FRAPCON calculation is skipped. FRAPCON supplies the fuel rod temperature to RAST-K as well. The pin-wise TH properties such as coolant temperature, density, and fuel temperature provided by the CTF and FRAPCON are volume-averaged to be used in RAST-K cross-section update which is based on a node. Such a coupled calculation is repeated until critical boron concentration (CBC), coolant temperature, coolant density, fuel temperature, and gap pressure are converged.

Table 1. Coupling feature of RAST-K and FRAPCON

BU step	PC	RAST-K	FRAPCON
i = n	Predictor	$N^{R,P}_n = f(N^R_{n-1}, P_{n-1})$	
	Nodal iteration	$P^P_n = f(N^P_n, T^P_n)$	$N^{F,P}_n = f(N^F_{n-1}, P_{n-1})$ $T^P_n = f(N^{F,P}_n, P^P_n)$
	Corrector	$N^R_n = f(N^R_{n-1}, avg(P_{n-1}, P^P_n))$	$N^F_n = f(N^F_{n-1}, avg(P_{n-1}, P^P_n))$
	Nodal iteration	$P^C_n = f(N^C_n, T^C_n)$	$T^C_n = f(N^F_n, P^C_n)$
	Converged	$P_n = P^C_n$	$T_n = T^C_n$

4. NUMERICAL TESTS

4.1.Modeling condition of OPR-1000

A numerical test of the coupled calculation was performed on Cycle 1-4 of a typical OPR1000 reactor. 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~F. Axially, 46 fuel meshes and 2 reflector meshes are modeled in the neutronics calculation, and 10 meshes are used in the CTF and FRAPCON model (active fuel region). Each length of 6 CTF and FRAPCON 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, 17~22 burnup steps are analyzed up to 13~18 GWD/MTU. The total core power is 2,815 MWth, the coolant total flow rate is 15,000~16,000 kg/sec, and the inlet temperature is 296.1 °C.

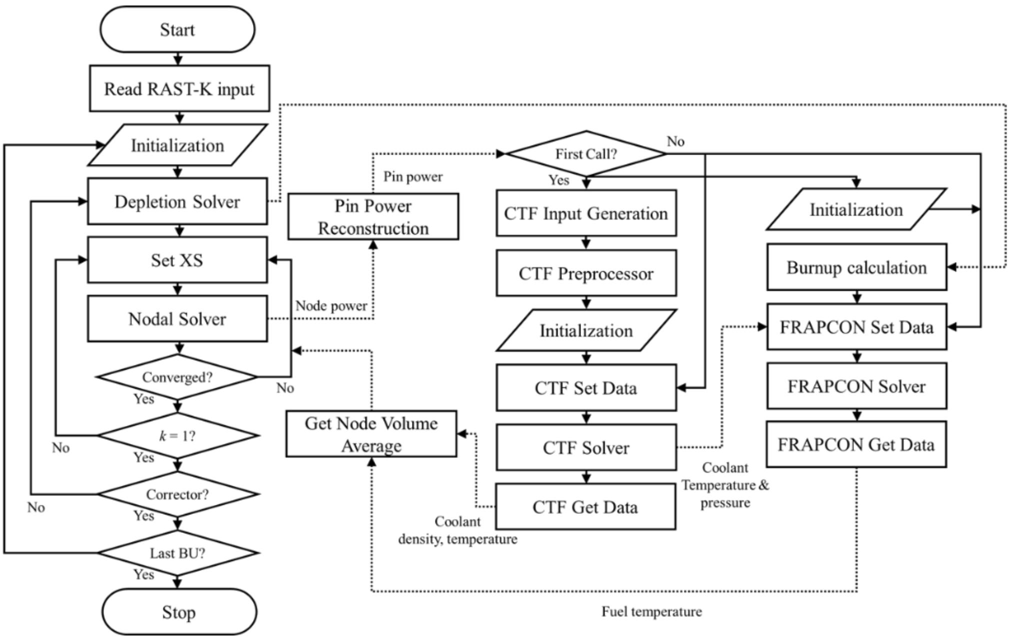


Figure 1. Flowchart of coupled calculation

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-K, CTF, and FRAPCON, respectively. Table 2 shows difference of nuclear design parameters for Cycle 4 calculated by the multi-physics (MP) coupled calculation and RAST-K standalone calculation (R) with internal simple TH solver. BOC at 0.5 GWd/MTU, MOC at 6.0GWd/MTU, and EOC at 18.0GWd/MTU. Difference of centerline temperature, peaking factor, FTC, MTC, and CR worth are in relative difference [%].

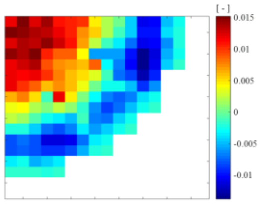
Table 2. Difference of nuclear design parameters for OPR1000 Cycle 4

Design parameter		Difference	Design parameter		Difference
Cycle length [day]		-2	Peaking factor, Fxy [%]	BOC	-0.63
				MOC	0.066
				EOC	-0.27
CBC [ppm]	BOC	3	FTC [%]	BOC	-3.7
	MOC	10		MOC	0.0
	EOC	7		EOC	-3.3
ASI [-]	BOC	-0.008	MTC [%]	BOC	7.3
	MOC	-0.003		MOC	4.7
	EOC	0.001		EOC	4.4
Centerline temperature [%]	BOC	-0.63	CR worth [%]	BOC	0.50
	MOC	0.066		MOC	0.35
	EOC	-0.27		EOC	0.28

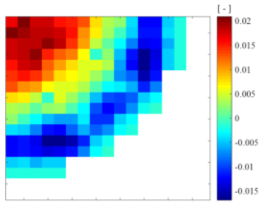
Figure 2~4 show difference of axially integrated relative node power distribution.
Coupled modes are defined as follow:

A. RAST-K, CTF, and FRAPCON
B. RAST-K and CTF
C. RAST-K and FRAPCON
D. RAST-K and internal TH

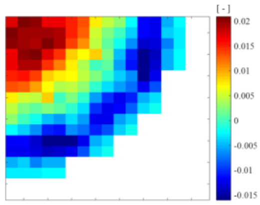
Comparing to power distribution calculated by internal TH solver (coupled mode D), power at central region is higher at BOC in coupled mode of A, B, and C which are based on pin-level with high fidelity codes. This is because power in periphery region is higher at BOC. On the other hand, the power in periphery region is higher at EOC.



(a) Diff. [A-D]

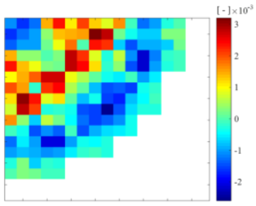


(b) Diff. [B-D]

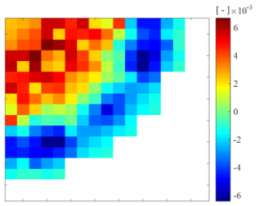


(c) Diff. [C-D]

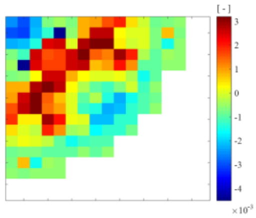
Figure 2. Difference of radial power distribution at BOC (0.0 GWd/MTU)



(a) Diff. [A-D]

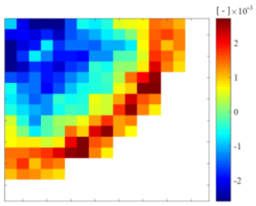


(b) Diff. [B-D]

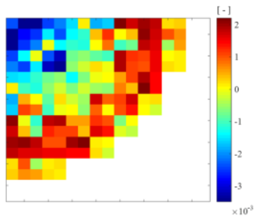


(c) Diff. [C-D]

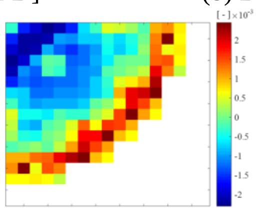
Figure 2. Difference of radial power distribution at MOC (6.0 GWd/MTU)



(a) Diff. [A-D]



(b) Diff. [B-D]



(c) Diff. [C-D]

Figure 3. Difference of radial power distribution at EOC (13.8 GWd/MTU)

5. Conclusion

The UNIST nodal diffusion code, RAST-K, the sub-channel TH code, CTF, and the steady-state fuel performance code, FRAPCON are successfully integrated. At every burnup step, RAST-K provides pin-level power distribution to CTF and FRAPCON. CTF computes the coolant temperature/density for RAST-K, and the coolant bulk temperature/pressure to be used as boundary condition in the fuel rod thermo-mechanical calculation of FRAPCON. FRAPCON returns the fuel temperature to RAST-K. Using coolant temperature, density, and fuel temperature, RAST-K 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 cycles 1-4 of a typical OPR-1000 reactor. The pin power distribution, coolant temperature and density, and fuel temperature are computed for parameter exchange among the codes. Comparison of design parameters for cycle 4 calculated by the multi-physics system and RAST-K standalone with simple internal TH solver shows decreased cycle length of 2 days and reduced peaking factor by 0.6%. This is because of the limitation in cross section feedback of the nodal diffusion code, which is based on node-wise even the pin-wise TH properties by CTF and FRAPCON are provided. Nevertheless, the multi-physics coupling is important because the coupled calculation gives the parameters of the behavior related to reactor safety during normal operation and provides initial states of accident analysis.

Acknowledgement

This research was supported by the project (L17S018000) by Korea Hydro & Nuclear Power Co. Ltd.

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