DEVELOPMENT OF A COUPLED CODE SYSTEM BASED ON SPACE SAFETY ANALYSIS CODE AND RAST-K THREE-DIMENSIONAL NEUTRONICS CODE

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ABSTRACT

Recently, a new system thermal hydraulic analysis code named SPACE (Safety and Performance Analysis Code for Nuclear Power Plants) has been developed by the Korean nuclear industry. Similar to other system thermal hydraulic analysis codes, the SPACE code uses point kinetics for core power calculation. To model asymmetric core power distribution, 3-dimensional neutron kinetics calculation is required. A coupled code system of SPACE/RAST-K has been developed for simulation of interactions between a three-dimensional reactor core kinetics code, RAST-K and the system safety and performance analysis code, SPACE. The linkage is direct and explicit coupling of the two codes. The coupling provides a method of executing the RAST-K three-dimensional neutronics using the nuclear steam supply system boundary conditions, such as the core inlet temperatures, core inlet flow distribution, core outlet pressure, trip information and control rod movements, calculated by the SPACE thermal hydraulics code. The RAST-K calculates suitable core conditions based on the boundary conditions provided by the SPACE code. The results of the RAST-K code, such as the local power and reactivity, are transferred to the core heat structures of SPACE code. The soundness of the coupled code system is confirmed by simulating the main steam line break (MSLB) benchmark problem developed to verify the performance of a coupled kinetics and system transient codes by OECD/NEA. The calculation results show that the developed coupled code system works well as predicted.

KEYWORDS

System code, coupling of neutronics code, SPACE, RAST-K, MSLB

1. INTRODUCTION

The Korean nuclear industry launched the SPACE code development project in 2006 to develop a new thermal-hydraulic system analysis code. The SPACE code adopts advanced physical modeling of two-phase flows, mainly two-fluid three-field models and has the capability to simulate 3D effects by the use of structured and/or non-structured meshes [1]. Also, the programming language for the SPACE code is C++ for new generation of engineers who are more comfortable with C/C++ than FORTRAN language. The SPACE code will replace outdated vendor supplied codes and will be used for the safety analysis of operating pressurized water reactor (PWR) and the design of an advanced reactor. Through various validation and verification programs using the separated or integral loop test data and the plant operating data, the code has been released and the topical reports on the code and related safety analysis methodologies have been prepared for license works. Additionally, various efforts have been adopted to expand the application area of the code, such as the coupling with multi-dimensional neutronics analysis code, the robustness improvement and the modification of three-dimensional system analysis features.

Postulated events with large positive moderator and thermal-hydraulic feedback, e.g. main steam line breaks (MSLB) are important neutronic and system transients that need to be evaluated to determine the overall safety and risk of operating nuclear power plants. These types of transients, as well as others, have been simulated using large thermal-hydraulic system and kinetics computer codes that have been developed by international regulatory bodies as well as the nuclear industry. Compared to the conventional system codes employing lower dimensional models, e.g. point kinetics models, the coupled system codes are expected to provide substantially more accurate prediction in analysis of the system transients involving strong interactions between neutronic and thermal-hydraulic phenomena. This is because that a realistic representation of the physical system in the three-dimensional space reduces the errors associated with the assumptions introduced in the lower dimensional models. One of the assumptions made in using a point kinetics model is that the reactivity change during the transient can be properly represented by a few sets of reactivity coefficients. However, the validity of the reactivity coefficients, which are to be generated from a set of steady-state neutronic calculation, is not guaranteed in the transient conditions in which the actual core condition is far from the conditions for which the reactivity coefficients are generated [2].

There are two primary concerns in the analysis of the steam line break accident: return-to-power and departure from nucleate boiling (DNB). The coupled codes are required to address these concerns in the best-estimate manner. Since return-to-power is a globally occurring phenomenon, sufficiently accurate prediction of the core power level would be possible if the transient inlet coolant condition and the core reactivity are properly determined by the coupled code [3]. A point kinetics model may predict a return-to-power while a three-dimensional kinetics model does not. It is expected that the coupled 3-dimensional kinetics codes can significantly reduce the conservatism of the lower dimensional models. The coupling scheme is direct and explicit in interface of system analysis code and neutron kinetics code. The soundness of the coupled code system is confirmed by simulating the main steam line break benchmark problem developed to verify the performance of a coupled kinetics ad system transient codes by OECD/NEA.

2. CODE DESCRIPTIONS

2.1. SPACE code descriptions

The SPACE code is composed of input and output package, hydrodynamic model package, heat structure model, control system model and reactor kinetic model, etc. The hydrodynamic model package is composed of hydraulic solver, constitutive models, special process models and component models. The input/output package performs read of input file and restart file, input error check, storage space setup, as well as variable initialization, preparation of main output file, plot files, and restart file. The hydraulic solver adopts two-fluid, three-field governing equations which are comprised of gas, continuous liquid and droplet fields. The three fields are allowed to be at non-homogeneous and non-equilibrium state, while the gas field is assumed to be a homogenous equilibrium mixture of vapor and non-condensable gas. The governing equations also involve porosity to take into account the structural material impact on the fluid flow [4]. Each field equation is discretized by applying the finite volume method (FVM) to the very unique SPACE mesh system which naturally encompasses various three dimensional structured and/or unstructured mesh systems, as well as one-dimensional pipe meshes [5]. Non-linear terms appeared in the temporal and source terms of the phasic mass and energy equations are linearized by using the Taylor expansion technique. Semi-implicit scheme is chosen as the basic numerical time advancement scheme in the SPACE.

The proper physical models can significantly improve the accuracy of the prediction of a nuclear reactor system behavior under many different transient conditions because those models are composed of the source terms for the governing equations. To develop the physical models and correlations for the SPACE code, various models currently used in major nuclear reactor system analysis codes have been reviewed. In addition, a literature survey of recent studies has been performed in order to incorporate the up-to-date models into the SPACE code. Unlike RELAP5, TRACE or CATHARE which are major best-estimate nuclear reactor system analysis codes that only consider liquid and vapor phases, the SPACE code incorporates a dispersed liquid field in addition to vapor and continuous liquid fields; interfacial interaction models between continuous, dispersed liquid phases and vapor phase have to be developed separately.

Constitutive models of the SPACE code is composed of the surface area and surface heat transmission correlation, surface-wall friction correlation, droplet separation and adhesion correlations, wall-fluid heat transmission mode and correlations, all correlations required for governing equations, and the type of the correlation is determined by the flow-form map. The SPACE code contains special process and system component models to limit or modify the solution of the basic governing equations reflecting the physical phenomena and to provide the capability to simulate the systems of nuclear power plant [1].

Nuclear fission heat of a nuclear fuel rod is calculated by point kinetics approximation and is treated as a heat source in the heat conduction equation. Reactivity feedbacks of moderator density, moderator temperature, fuel temperature, boron concentration, reactor scram, power defect are considered and decay heat of ANS-73, -79 and -94 models are implemented. Conductivity of the gap between fuel pellet and cladding is dynamically calculated considering the conduction of the gap gas mixture, solid contact conduction, and the radiation across the gap. Five kinds of gap gases such as Helium, Argon, Krypton, Xenon, and Nitrogen are considered.

2.2. RAST-K code descriptions

The RAST-K (Reactor Analysis code for Steady and Transient – KHNP) has been developed for the evaluation of dynamic control ability of control rod and the application for the real-time core simulator in 21 Korean operating PWRs, and licensed as a core transient analysis code in zero power condition of PWRs from Korean regulatory body at 2006.

The RAST-K solves the three-dimensional two group neutron diffusion equations using the non-linear nodal expansion method (NEM) CMFD (Coarse Mesh Finite Difference) method which is performed through the coupling the high order NEM to get more accurate core power distribution. To reflect the reactivity feedback effect caused by fuel temperature and coolant temperature (or density) variation, the RAST-K code has own time dependent one-dimensional fuel heat transfer solver and time dependent one-dimensional thermal-hydraulic property model [6].

For a nuclear fuel assembly, applying different assembly discontinuity factor (ADF) for six directions, the transverse neutron leakage terms are calculated with second-order polynomials. The burnup rate is calculated with macroscopic cross section and the power could be obtained excluding the reflector using the ‘core-reflector boundary condition’. Because of these features, RAST-K has advantages for simulating real-time reactor core. The neutron cross section of RAST-K is described as a function of fuel temperature, temperature and density of coolant, boron concentration and number density of poison, and can be obtained from commercial codes and nuclear design code also. The fuel temperature is calculated by solving time-dependent one-dimensional heat transfer equation and the coolant temperature and density are calculated with solving homogeneous equilibrium model for two-phase flow adopting drift flux model.

3. SPACE / RAST-K COUPLING

To model accurate asymmetric core power distribution, three-dimensional neutron kinetics calculation is required. The coupled code system of SPACE/RAST-K has been developed for simulation of interactions between a three-dimensional reactor core kinetics code, RAST-K and the system safety and performance analysis code, SPACE. The linkage is direct and explicit coupling of the two codes. The coupling provides a method of executing the RAST-K three-dimensional neutronics using the nuclear steam supply system boundary conditions, such as the core inlet temperatures, core inlet flow distribution, core outlet pressure, trip information and control rod movements, calculated by the SPACE thermal hydraulics code. The RAST-K calculates suitable core conditions based on the boundary conditions provided by the SPACE code. The results of the RAST-K code, such as the local power and reactivity, are transferred to the core heat structures of SPACE code as depicted in Figure 1.

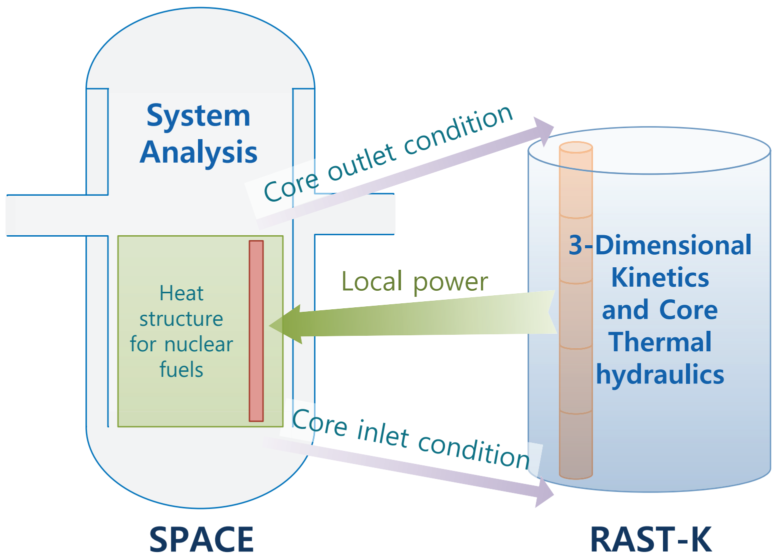

Figure 1. Schematics of SPACE/RAST-K coupling

The coupling of SPACE and RAST-K is implemented with dynamic link library (DLL) of Microsoft Windows operating system. The use of DLL has the advantage in maintaining the integrity of each code independently as well as keeping coupled code structures. The system analysis code SPACE is host code and the neutron kinetics code RAST-K supply library function. In the coupling scheme, major element is interface data array for transfer between two codes. There are parameters of the highest level: number of radial and axial nodes in neutronics code. The number is necessary for dynamic allocation of array in system code. The other parameters are flag for call modes, total power, total reactivity, axial shape index, trip signal, time step, control rod position and thermal-hydraulic condition for inlet and outlet.

	(nz-1)nr+1	(nz-1)nr+2	(nz-1)nr+3	……	……	nz·nr-2	nz·nr-1	nz·nr
	(nz-2)nr+1	……	……			……	(nz-1)nr-1	(nz-1)nr
	……					……	……	(nz-2)nr
	(z-1)nr+1	……		……	(z-1)nr+r			……
Axial	……		**Used**	……	……		**Unused**	
	3nr+1	……	……					……
	2nr+1	2nr+2	……				……	3nr
	nr+1	nr+2	nr+3		……	……	2nr-1	2nr
	1	2	3	……	r	……	nr-1	nr

Radial

Figure 2. Local power array for data transfer

Another interface parameter is local temporal power of each node. The active size of local power array is determined by number of radial and axial nodes as shown in Figure 2. The maximum size of 1-dimensional interface array is 46,000, which is limited by nr=40 and nz=1150. There are three kinds of call modes between SPACE and RAST-K. The first call mode is data initialization for calculation of RAST-K. The second mode is steady calculation and the third mode is transient calculation. In the parameter of caller function the interface array is treated as pointer. In explicit coupling the system thermal-hydraulic and neutronics calculations are performed at each time step. After the determination of thermal-hydraulic field the power distribution is updated subsequently.

4. EVALUATION OF COUPLED CODES

4.1. OECD MSLB problem

A postulated MSLB transient based on the Three Mile Island Unit 1(TMI-1) plant had been developed as a benchmark problem by OECD/NEA. The TMI-1 plant is a two-loop PWR of 2772MWt power with two vertical once-through steam generators. The plant has two hot-legs, four cold-legs and corresponding four reactor coolant pumps. Two steam nozzles are installed to each steam generator and the four steam lines are connected to a common header. The reactor core consists of 177 fuel assemblies which are 357cm long [7]. In the MSLB the overcooling in the broken-side cold legs occurs due to rapid depressurization accompanied with the loss of secondary inventory and energy in steam generator of the broken loop. An asymmetric cooling of core originated from the unequal cooling of steam generator cause an asymmetric radial power distribution through the reactivity feedback effect. The reactor coolant pumps continuously operate during the transient for maximizing reactivity feedback effect. The transient is analyzed for 100 seconds.

The benchmark problem consists of three phases. In the phase I point kinetics model is used [8]. In the phase II, coupled 3-dimensional neutronics and core thermal-hydraulics response is evaluated with inlet and outlet of core transient boundary conditions provided by the benchmark problem. For the phase III, best-estimate coupled core and plant system transient codes are used [9]. In this paper, calculation results of coupled SPACE/RAST-K codes are compared with the results of other participants of the phase III.

4.2. SPACE and RAST-K input models

The system analysis code SPACE inputs consist of cells (control volume), faces (flow path), heat structures, trips and control variables. For OECD MSLB benchmark problem, the SPACE input model is prepared with 156 cells, 184 faces and 36 heat structures. As shown in Figure 3, two steam lines of intact loop are modeled with single steam line and break in steam line are modeled as valves. Because a double-ended break of the 24-inch steam line and a slot break of the 8-inch pressure balance line should be specified in this analysis, each steam line, connected to the broken-side steam generator, is modeled individually. The feed lines modeled with the downcomer channel and feeding TFBC (Temporal Face Boundary Condition) components in the SPACE code [10]. The 6 main steam safety valves were modeled in the intact steam line using TFBC components which is activated by the pressure of steam line. The core region is modeled with 2 parallel flow channels and each flow channel has 8 axial nodes (include 6 active core nodes). The corresponding heat structures of active core region consist of 6 axial nodes. The tube or shell sides of a steam generator were modeled with 12 vertically stacked heat structures and cells, respectively.

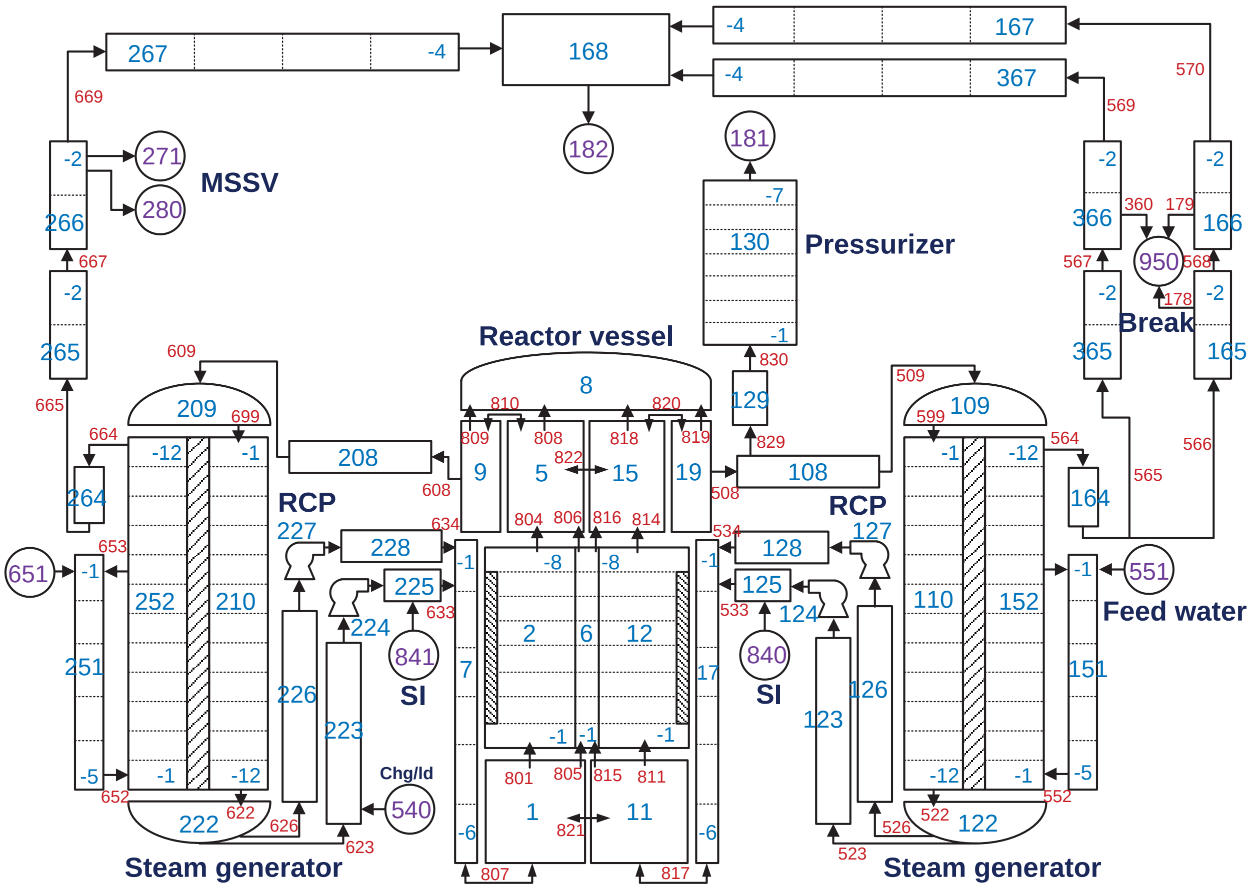

Figure 3. SPACE input nodalization of TMI-1MSLB

For coupled calculation of SPACE/RAST-K, neutronics nodes are mapped into thermal-hydraulic nodes radially and axially. In radial direction the RAST-K input model has 241 channels, but 64 channels are reflector channels and only 177 channels are used in actual coupling. In radial direction SPACE has 2 active core channels and one bypass channel as shown in Figure 3. The bypass channel is not used for code coupling and only 2 active core channels are used. The mapping information between 177 RAST-K channels and 2 SPACE channels in radial direction are presented in Figure 4. If two parallel channels were used in the interface, the 15 assemblies in central region are connected to both hydraulic channels in SPACE code. However, each assembly is limited to one inlet and one outlet thermal hydraulic boundary condition in SPACE/RAST-K coupling. To solve this problem, a logical third SPACE boundary condition channel is added using SPACE control system. The logical SPACE channel has average thermal hydraulic properties of 2 SPACE channels. In axial direction, the 24 axial nodes in RAST-K correspond to 6 axial nodes in SPACE.

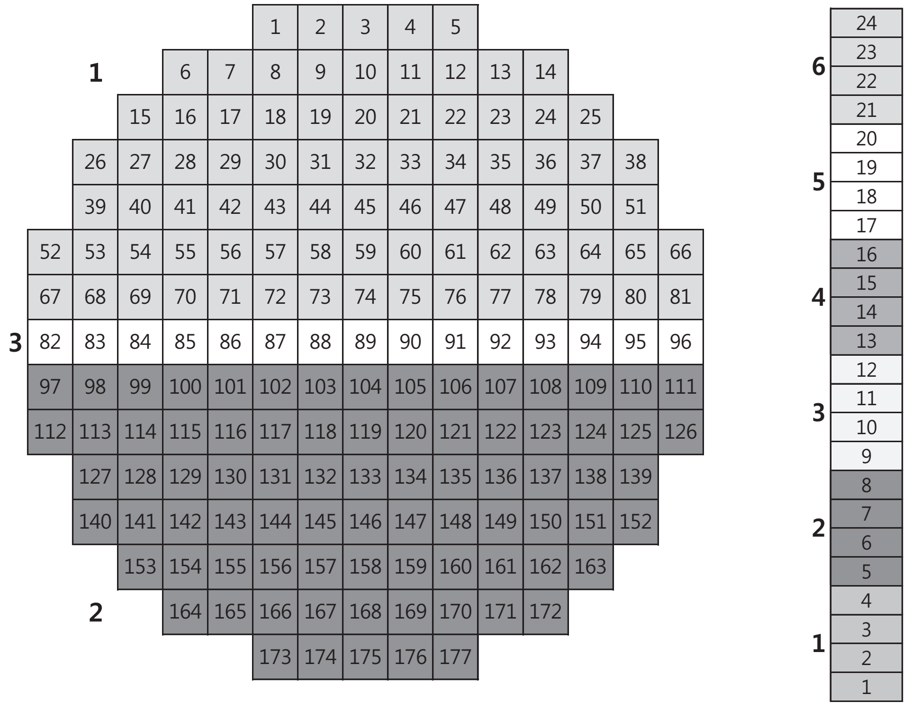

Figure 4. Models for core channels of TMI-1MSLB

4.3. Steady state results

The coupled calculation for steady-state condition is performed for 2000 seconds. The results of calculation in full-power condition of the plant are tabulated in Table I. As the initial condition of transient calculation, a coupled steady-state condition for TMI-1 is calculated. Most of the designed parameters of TMI-1 plant are calculated appropriately with the coupled SPACE/RAST-K code. The differences in the feed water flow rate, the SG secondary outlet pressure and SG inventory are noted. The outlet pressure of SG should be deliberately adjusted to obtain the core inlet temperature of target value. Since the overcooling of the primary coolant continues until the SG empties, it would be significant to get the initial SG inventory accurate. However, there is an uncertainty in total mass of water in secondary SG due to the feed line, the difference of -3.31% are considered as acceptable in this calculation. The steady-state-conditions are identical in coupled SPACE/RAST-K codes with 3-D kinetics and SPACE standalone calculation with point kinetics.

Table I. Steady state conditions

Parameters	Target	Calculated	Difference
Core power, MW	2772.0	2772.0	0.00%
RCS cold leg temperature, K	563.76	563.36	-0.07%
RCS hot leg temperature, K	591.43	591.57	0.02%
Lower plenum pressure, MPa	15.36	15.468	0.70%
Outlet plenum pressure, MPa	15.17	15.201	0.21%
RCS pressure, MPa	14.96	14.962	0.01%

Total RCS flow rate, kg/s	17602.2	17602.2	0.00%
Core flow rate, kg/s	16052.4	16030.8	-0.13%
Bypass flow rate, kg/s	1549.8	1571.4	1.40%
Pressurizer level, m	5.588	5.580	-0.15%
Feedwater flow per SG, kg/s	761.59	778.73	2.25%
SG outlet pressure, MPa	6.41	6.6664	4.00%
Initial SG inventory, kg	26000	25139.1	-3.31%
Feedwater temperature, K	510.93	510.93	0.00%

4.4. Transient results

For 100 seconds, a transient calculation is carried out. A transient begins with simultaneous opening of the break valves at 0.01 seconds. As a critical flow model at the break, Ransom-Trapp critical flow model is used with the discharge coefficient of 1.0. The calculation results with Henry-Fauske model and homogeneous equilibrium model show similar critical flow rate for the superheated single-phase steam. The Phase III of OECD MSLB benchmark problem has two scenarios. In this paper, only the first scenario is analyzed. The first scenario uses same assumptions as those used in the current licensing practice. With the first scenario, 3-D kinetics model usually do not predict return-to-power is used in the transient calculation [9]. The transient results of coupled SPACE/RAST-K codes with 3-D kinetics and SPACE standalone with point kinetics are compared with those of participants in OECD benchmark problem [7, 9]. The variable time steps of 0.01~0.001 seconds are used in analyses.

Transient results show similar behavior with results of participants in OECD benchmark problem. The return-to-power does not occur in both 3-D and point kinetics analyses. The core power starts to decrease right after the break due to the rapid depressurization resulting in the negative reactivity insertion as presented in Figure 5. The overcooled coolant brings positive reactivity into the core to cause the core power to increase. The scram is activated so that the core power decreases rapidly from the maximum core power of 116.1% (3218.03MW). The peak of power occurs 1.3 seconds later with 3-D kinetics than that with point kinetics due to larger negative reactivity during depressurization. The difference in the time of reactor scram affects the system pressure and the mass flow rate directly. The maximum negative reactivity of -0.0488 $\Delta k/k$ occurs at 9.6 seconds as shown in Figure 6. The break flow decreases continuously as the broken-loop steam line pressure decreases as identified in Figure 7 and 12.

The temperature of coolant continues to decreases due to the continuous overcooling of the coolant as presented in Figure 8~11. And the associated positive reactivity insertion results in the slight increase of power at 81.3 seconds. The decrease in the core reactivity and power is caused by the dryout of the broken SG and subsequent negative thermal feedback shown in Figure 13. After the dryout of the broken SG, the increase of coolant temperature of broken loop is observed in Figure 8 and 10.

Figure 14 shows the normalized radial power at the time of maximum reactivity after scram that is 81.3 seconds. The asymmetric distribution of radial power is clearly shown due to asymmetric coolant temperature and the presence of stuck rod. Because the hydraulic conditions of point and 3-D kinetics for transient are similar, most of system parameters are also similar. The capability of the coupled code system is confirmed by simulating the MSLB benchmark. The execution time for the coupled code is ten times of execution time of the standalone system analysis code. The coupled scheme should be optimized and improved to be more effective in the future.

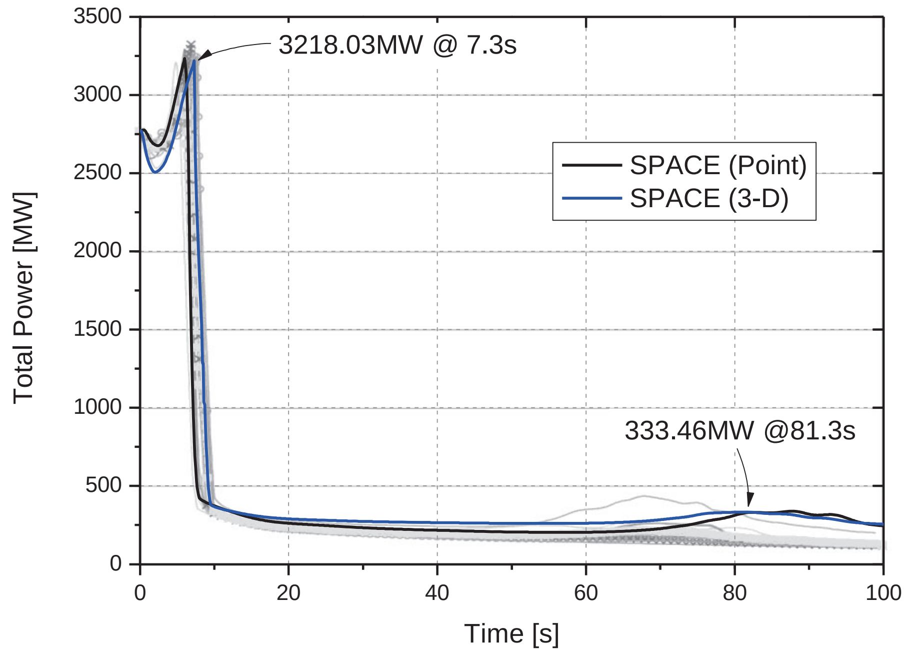

Figure 5. Transient total power

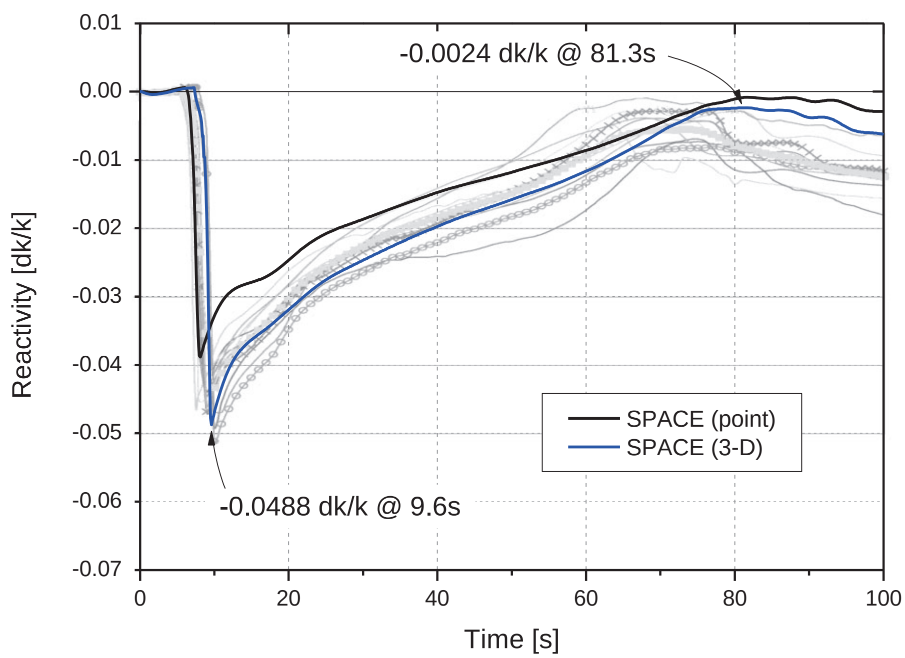

Figure 6. Transient total reactivity

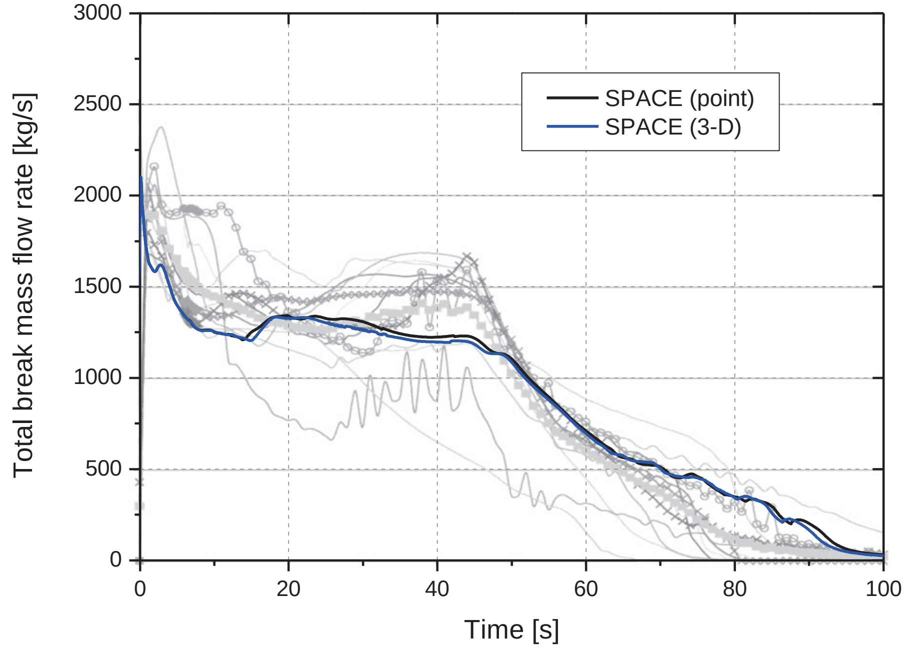

Figure 7. Transient total break mass flow rate

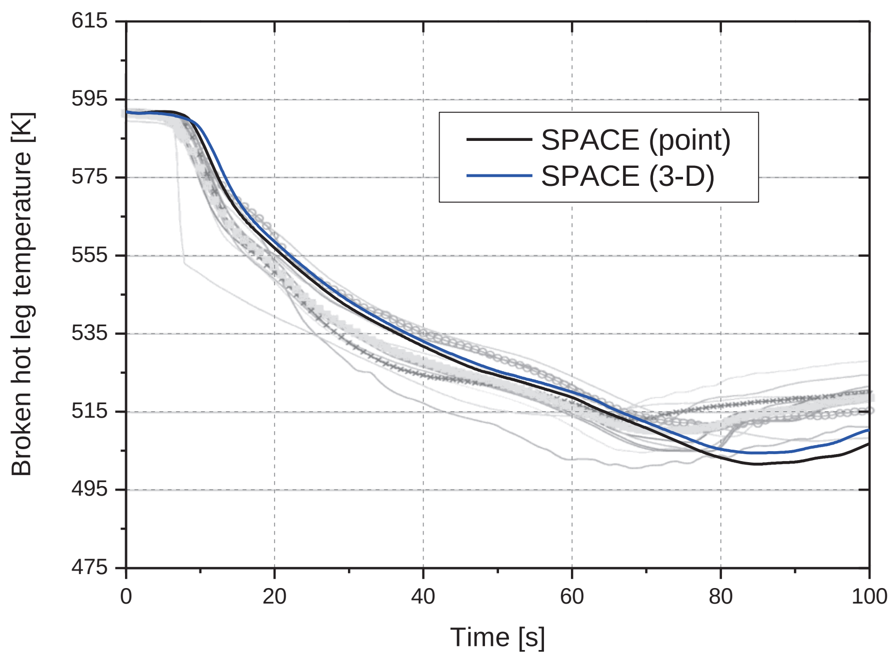

Figure 8. Broken hot leg temperature

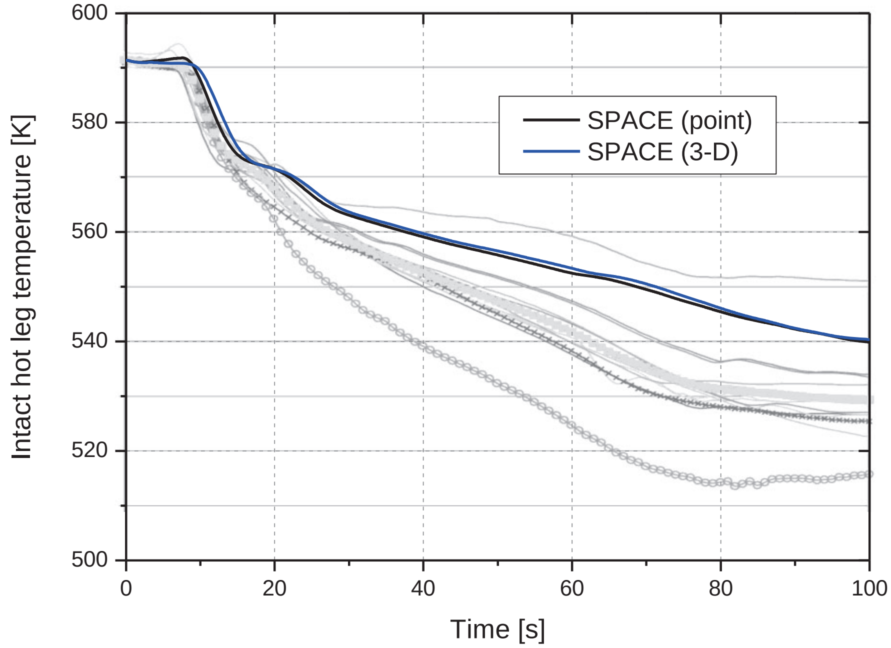

Figure 9. Intact hot leg temperature

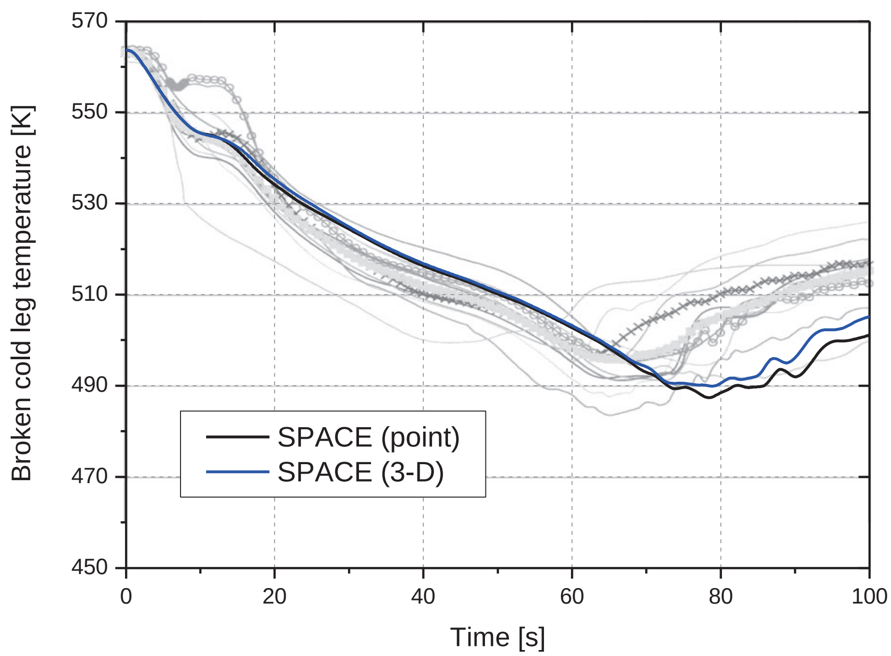

Figure 10. Broken cold leg temperature

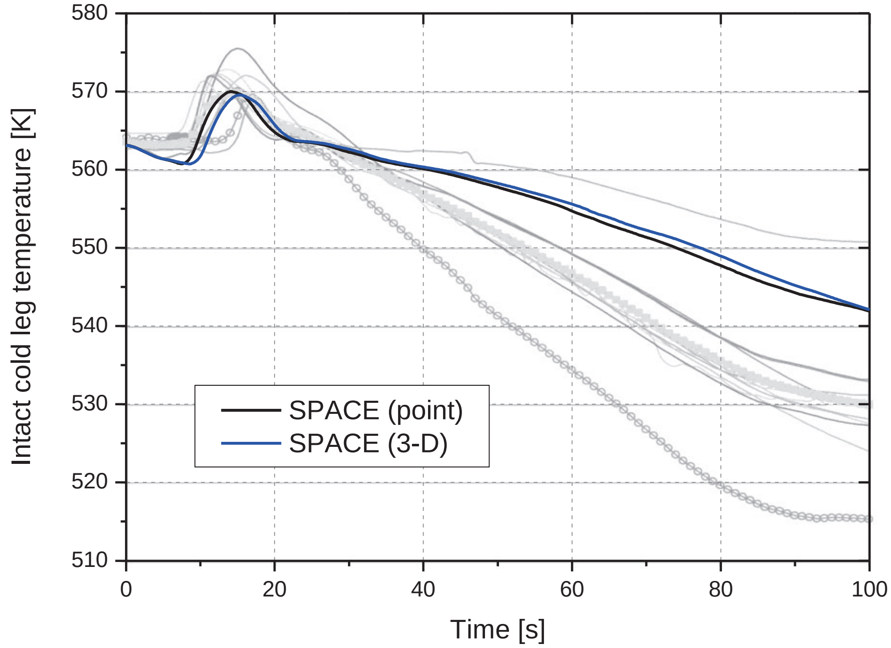

Figure 11. Intact cold leg temperature

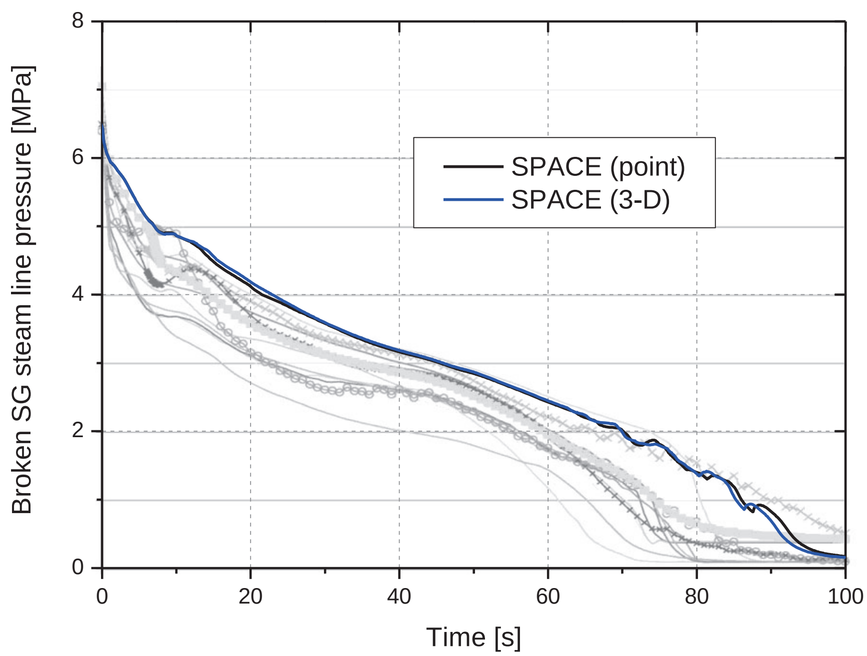

Figure 12. Broken SG steam line pressure

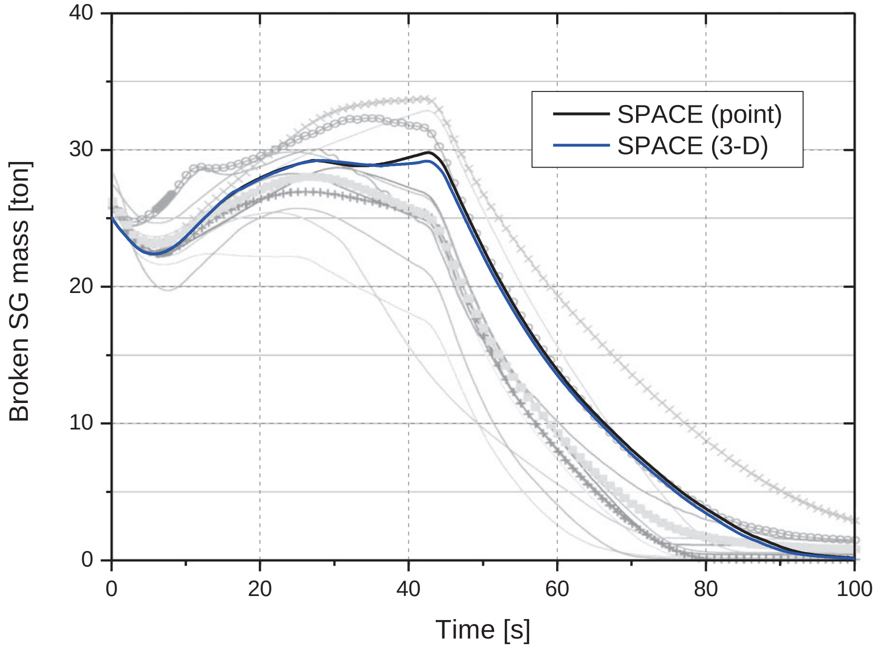

Figure 13. Transient broken SG mass

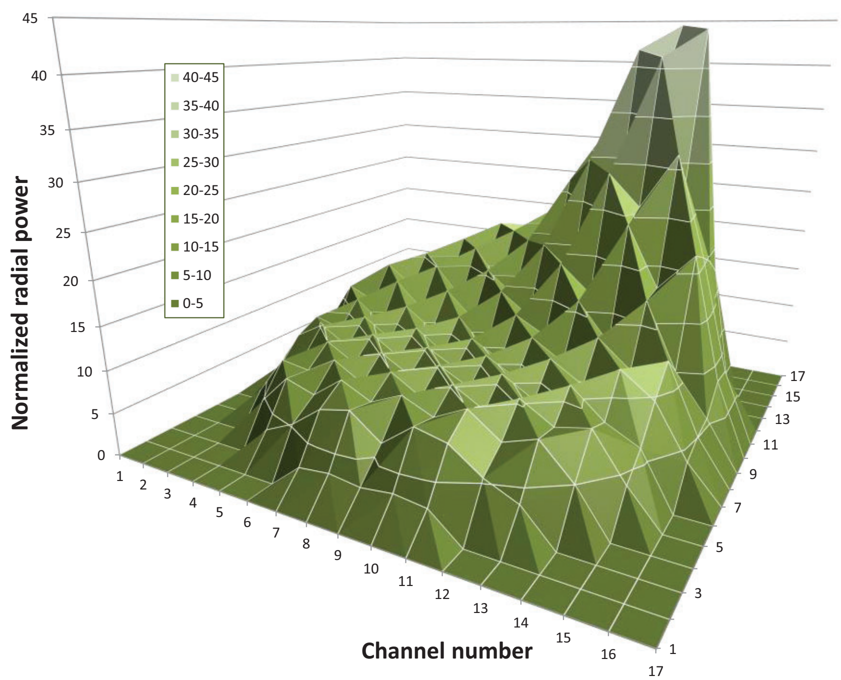

Figure 14. Normalized radial power distribution

5. CONCLUSIONS

The coupled code SPACE/RAST-K has been developed for simulation of interactions between a three-dimensional reactor core kinetics and the system safety and performance analysis. The explicit coupling of SPACE and RAST-K is implemented with DLL. The coupling provides a method of executing the RAST-K three-dimensional neutronics using the nuclear steam supply system boundary conditions, such as the core inlet temperatures, core inlet flow distribution, core outlet pressure, trip information and control rod movements, calculated by the SPACE thermal hydraulics code. The RAST-K calculates suitable core conditions based on the boundary conditions provided by the SPACE code. The results of the RAST-K code, such as the local power and reactivity, are transferred to the core heat structures of SPACE code. The capability of the coupled code system is confirmed by simulating the main steam line break benchmark problem. The calculation results show that the developed coupled code system works well as predicted. The performance of the coupled code should be optimized and improved to be more effective in the near future.

ACKNOWLEDGMENTS

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