Nuclear Mathematical and Computational Sciences: A Century in Review, A Century Anew
Gatlinburg, Tennessee, April 6-11, 2003, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2003)

DEVELOPMENT AND VALIDATION OF A COUPLED CODE SYSTEM RETRAN-3D/MASTER

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ABSTRACT

A coupled code system of RETRAN-3D/MASTER has been developed for best-estimate simulations of interactions between reactor core neutron kinetics and plant thermal-hydraulics by incorporating the 3-D reactor core kinetics analysis code MASTER into system transient code RETRAN-3D. Two different coupling methods are developed, that is, standard analysis and refined analysis. In the standard analysis, T/H conditions are all handled by RETRAN-3D, whereas in refined analysis, the core inlet and outlet conditions are provided by RETRAN-3D, and then the COBRA III-C/P built in MASTER determines more detailed T/H conditions by performing subchannel analysis. The soundness of the consolidated code system is confirmed by simulating the MSLB benchmark problem developed by OECD/NEA. All three analysis methods of RETRAN-3D/MASTER standard and refined analyses as well as RETRAN-3D point kinetics are performed. By incorporating the multi-dimensional reactor kinetics code into the T/H code, the interaction between T/H model and reactor neutronics model is handled efficiently. Especially, the refined calculation of RETRAN-3D/MASTER is seen to lower the power peaking factor by as much as 33%, compared to their standard coupled analysis.

Key Words: PWR, Safety, Simulations

1. INTRODUCTION

It is well known that the reactor core neutron kinetics and plant thermal-hydraulics ought to be closely related to attain improved accuracy in nuclear power plant simulation during its transient behaviors. For the effective data transfer between the two different models, simultaneous analysis accounting for feedback effects of each other has been desirable, but rather hardly achievable due to the enormous computational efforts required, especially for the system showing highly asymmetric behaviors where multi-dimensional treatments are demanded. Thus, manually assigning the boundary conditions based on the counter part results has been widely adopted for such problems. Lately, however, digital computer technology is so much advanced that directly coupled neutron kinetics and thermal-hydraulic codes such as RELAP/PARCS, RELAP5/PANBOX, TRAC-PF1/NEM, CRONOS2-FLICA4, and MARS/MASTER have been successfully developed [1].

In this study, a realistic system analysis code RETRAN-3D has been coupled with the MASTER code. RETRAN-3D[2] developed by EPRI is a realistic system analysis code with two-fluid thermo-hydraulic model, heat structure model, point kinetics model, 3-D neutron kinetics model, and other subsidiary models, such as valve, pump, and turbine, needed for analyzing transient behavior of the light water reactor plants. Current RETRAN-3D is already equipped with 3-D neutron kinetics module ARROTTA, thereby allowing coupled analysis with thermal-hydraulic calculation. ARROTTA, however, has not been widely used in Korea, mainly due to the compatibility problem in neutron group constants with other existing reactor core neutron analysis codes. The point kinetics model, therefore, had been more frequently used, which motivated coupling RETRAN-3D and multi dimensional reactor kinetics code MASTER. MASTER is a two-group, three-dimensional neutron diffusion code developed in KAERI [3]. The code is capable of microscopic depletion, xenon dynamics, on-line DNB (departure from nuclear boiling) analysis, and kinetics calculation in both rectangular and hexagonal geometries. The transient thermal-hydraulic analysis in MASTER is achieved by COBRA III-C/P module employing homogeneous equilibrium model [4]. Also, MASTER has a broad input database covering each neutron fuel cycle for all domestic nuclear power plants. Both RETRAN-3D and MASTER have Generic License for Non-LOCA analysis, from USNRC and KINS, respectively.

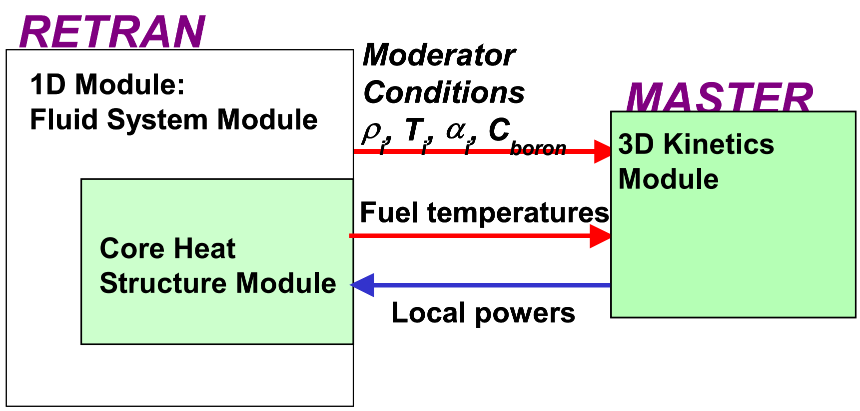
Through the coupled analysis with general purpose realistic system analysis code RETRAN-3D and MASTER, it would be possible to optimally simulate the multi-dimensional reactor kinetics effects on plant transient behavior, thereby substantially reducing inefficiency, inaccuracy, and unnecessary conservatism related to traditional iterative approach once needed for applying 3-D kinetics model.

2. RETRAN-3D/MASTER COUPLING METHODOLOGY

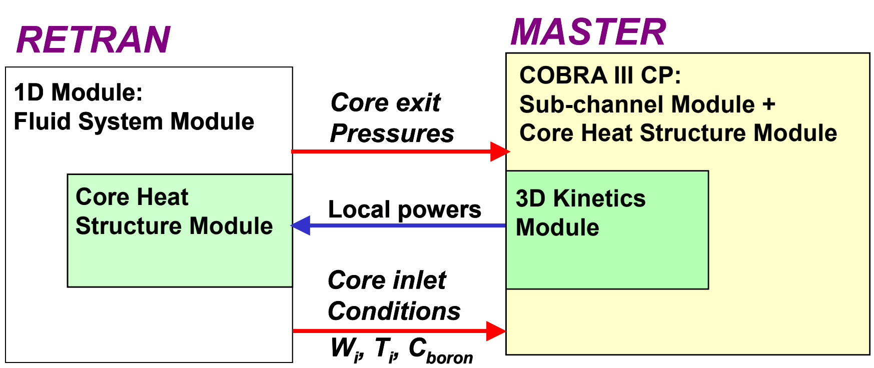
RETRAN-3D/MASTER coupling has been carried out by replacing the RETRAN-3D point kinetics model with the MASTER 3D neutron kinetics model. The neutron kinetics model in RETRAN-3D calculates heat generation rate based on thermo-hydraulic parameters such as moderator temperature, moderator density from the previous time step, then transfers to heat structure model. The heat structure model in connection with the thermo-hydraulic module, determines the thermo-hydraulic flow conditions and heat flux. Therefore, heat generation rate is obtained separately from the thermal-hydraulic and heat structure modules, and RETRAN-3D standalone calculation strategy could be maintained, even with the coupled MASTER module.

One way of coupling MASTER with RETRAN-3D would be converting MASTER into the subprogram of RETRAN-3D, especially since both two codes use same FOTRAN programming language. Since the coupled program, made by this method, would be basically a single process, all the global variables would reside in the same Share Memory location. This could cause a compilation problem if RETRAN-3D and MASTER share the same subroutine name, or same Named Common Block. Such an approach could therefore produce unexpected reliability problems as well as unnecessary maintenance efforts. Separate development independent with each other wouldn't be possible. MASTER, to avoid such a problem, has been converted into Dynamic Link Library (DLL) where shared memory is maintained separately from memory space occupied by other main programs, in our case RETRAN-3D. Generally, the coupled program through DLL is basically multi-program having separate memory space, excluding memory conflicts inherently, and only interfacing variables need to be arranged for data communication between program modules. The algorithm of RETRAN-3D and MASTER, therefore, could be maintained straightforward irrespective of other linked programs if the DLL method is applied. In addition, since DLL routine is invoked into the memory space only when called upon, standalone RETRAN-3D calculation is also possible.

Figure 1 depicts variable transfer concept between RETRAN-3D and MASTER. There are two different kinds for the coupled calculations, the standard calculation and refined calculation. In both cases, RETRAN-3D receives reactor core power distributions from MASTER and plant thermal-hydraulic calculation is performed in the same manner. They differ, however, in the way MASTER determines core power distribution. In standard calculation, RETRAN-3D provides thermal-hydraulic parameters needed for deciding core power distribution, whereas in refined calculation [5], MASTER determines those values by its own T/H module performing so-called subchannel analysis. Here, the inlet and outlet flow conditions are provided by RETRAN-3D. In the subchannel T/H analysis, fuel-assembly-wise nodding or even quadrant of an assembly nodding are possible with little computational overhead. The subchannel T/H analysis is carried out by COBRA III-C/P module built in the MASTER code.



(a) Standard Calculation



(b) Refined Calculation

Figure 1. Two different modeling methods allowed in RETRAN-3D/MASTER analysis

The main characteristics of the two analysis options, the standard and the refined analyses, are as follow.

- - Standard Analysis: T/H parameters necessary for determining the core reactivity is provided by RETRAN-3D (See Figure 1.a). The necessary T/H parameters are moderator density, moderator temperature, void fraction, boron density distribution determined by T/H models. In addition, fuel temperature distribution, determined based on heat structure model is also prepared. Then, all these parameters are transferred to MASTER which, in turn, calculates reactor core power distribution. The newly calculated values are sent back to RETRAN-3D.

- - Refined Analysis: The core inlet and outlet conditions from RETRAN-3D are transferred to COBRA III-C/P module in MASTER (See Figure 1.b). The main parameters include (i) core inlet coolant mass flow rate, (ii) core inlet coolant temperature, (iii) boron density at core inlet, and (iv) core exit pressure. COBRA III-C/P uses these parameters, as boundary conditions, for calculating the reactor core T/H conditions in detail. The reactor core kinetics module in MASTER uses the refined core T/H conditions for the neutron cross-section data. The core power distribution obtained in this manner therefore reflects more realistic fuel and moderator temperature effects on neutronics data. Finally, the refined core power distribution is passed to RETRAN-3D for global T/H calculation.

3. VERIFICATION CALCULATIONS AND RESULTS

The verification calculations were performed for RETRAN-3D/MASTER coupled codes. As a test problem, OECD NEA MSLB Benchmark Problem was selected for comparison between the results from this study and those from the various participants of the problem [5,6].

3.1 OECD/NEA Main Steam Line Break (MSLB) Benchmark Problem Outline

The reference plant selected for the MSLB benchmark problem is the TMI-1, 2772 MWth B&W designed pressurized water reactor plant [7]. The coolant system consists of two hot legs, four cold legs, and two once-through type steam generators. Two steam nozzles are attached to each steam generator and the four steam lines are connected to Main Steam Isolation Valve (MSIV) then combined to common header linked to turbine via Turbine Stop Valve. The reactor core consists of 177 fuel assemblies. The specifications provide all the data needed for the neutronics and T/H modeling. The group constant data consisting of 438 unrodded and 192 rodded group constant tables represent an end-of-cycle core. As for the control rod worth, two rodded group constant sets were provided in the specification: for the best-estimate and the return-to-power. As far as the combination of neutronic and T/H modeling is concerned, there are three exercises defined in the benchmark: (i) Exercise I, point kinetics plant simulation, (ii) Exercise II, coupled 3D neutronics/core thermal-hydraulics response evaluation, and (iii) Exercise III, best-estimate coupled core/plant transient modeling. The safety system related with simulating the MSLB problem includes Main Steam Safety Valves connected to each steam line. The purpose of these valves is to limit the allowable pressure for the steam generator. The other safety feature of concern is Safety Injection at the primary system.

In the benchmark problem, the break is assumed at the main steam line located between the steam generator and MSIV. Such incident would cause excessive cooldown effect on the fault-side reactor coolant loop. Together with the single stuck rod assumption, the MSLB event is expected to provide adequate scenario for evaluating the code capability for modeling the asymmetric thermal-hydraulic and neutronics effects during the transient. The main interests during the MSLB test are returning to power and regaining the reactor core criticality. Breaks were assumed to occur at both two steam lines connected to the steam generator in the loop with pressurizer, one being the double-ended break and the other one 8" slot break. The overcooling effect has been augmented by assuming the full power operation where coolant inventory is in its highest level. Also, in order to worsen the accident consequences, single failure of feed water valve stuck open for the faulted-loop steam generator was assumed. The feed water was assumed to stop at 30 sec after the accident initiation, by closing the Feedwater block valve. Reactor coolant pumps were assumed to continue their operation during the accident, thereby maximizing the heat transfer from primary side to secondary system. To enhance the depressurizing effect, the pressurizer heater was not considered. Also, negative reactivity due to boron injection was not modeled to consider a more serious situation. The reactor scram, following the accident initiation, was preset for 114% reactor power with 0.4 sec delay, or pressurizer pressure of 13.41 MPa (1945 psia) with 0.5 sec delay. The setpoint for High Pressure Safety Injection was pressurizer pressure of 11.34 MPa (1645 psia) with 25 sec delay. The three cases of point kinetics, RETRAN-3D/MASER standard analysis, and RETRAN-3D/MASTER refined analysis were tested.

3.2 RETRAN-3D/MASTER Code Input Modeling

The nodding diagram of RETRAN-3D/MASTER code input prepared for the MSLB Benchmark Problem is as shown in Figure 2. The TMI-1 plant was simulated with a total of 121 volumes and 147 junctions. All four reactor coolant pumps were modeled. Ten fill junctions were used to model the eight valves, high pressure injection, and the breaks. The reactor vessel is composed of 22 volumes and reactor core itself was modeled by separately considering the faulted side and intact side, each having 6 volumes. The fuels rods were divided into 12 heat structures to match the nearby hydraulic volumes. Since the breaks were assumed in each steam line, both of the steam lines were modeled, whereas a single volume was assigned for the steam lines in the intact side. The 24" pipe where double-ended break assumed to occur was provided with 2 valves connected to time dependent volume defining the containment conditions. The other valve connecting the two broken pipe area is closed at the time of the break. This prevents flow jump through the double-ended break, and assures steam flow from the broken side steam generator is only to the break area of the corresponding connected steam line. The 8" slot break is simulated with separate valve connected to time dependent volume for containment. The steam flow through the main steam line safety valve was modeled using negative fill junction rather than the normal valve so that the flow rate could be governed by steam line pressure. The heat structure model was employed to model the heat transfer through the steam generator tubes, whereas other structures were not considered.

For the reactor core kinetics module of MASTER, each fuel assembly was assigned single radial mesh (total of 177). Axially, 28 meshes were allocated to each radial mesh. The reactor core meshes for RETRAN-3D and those for MASTER do not generally align each other. The linear interpolations therefore were used to find the corresponding values of parameters of interest.

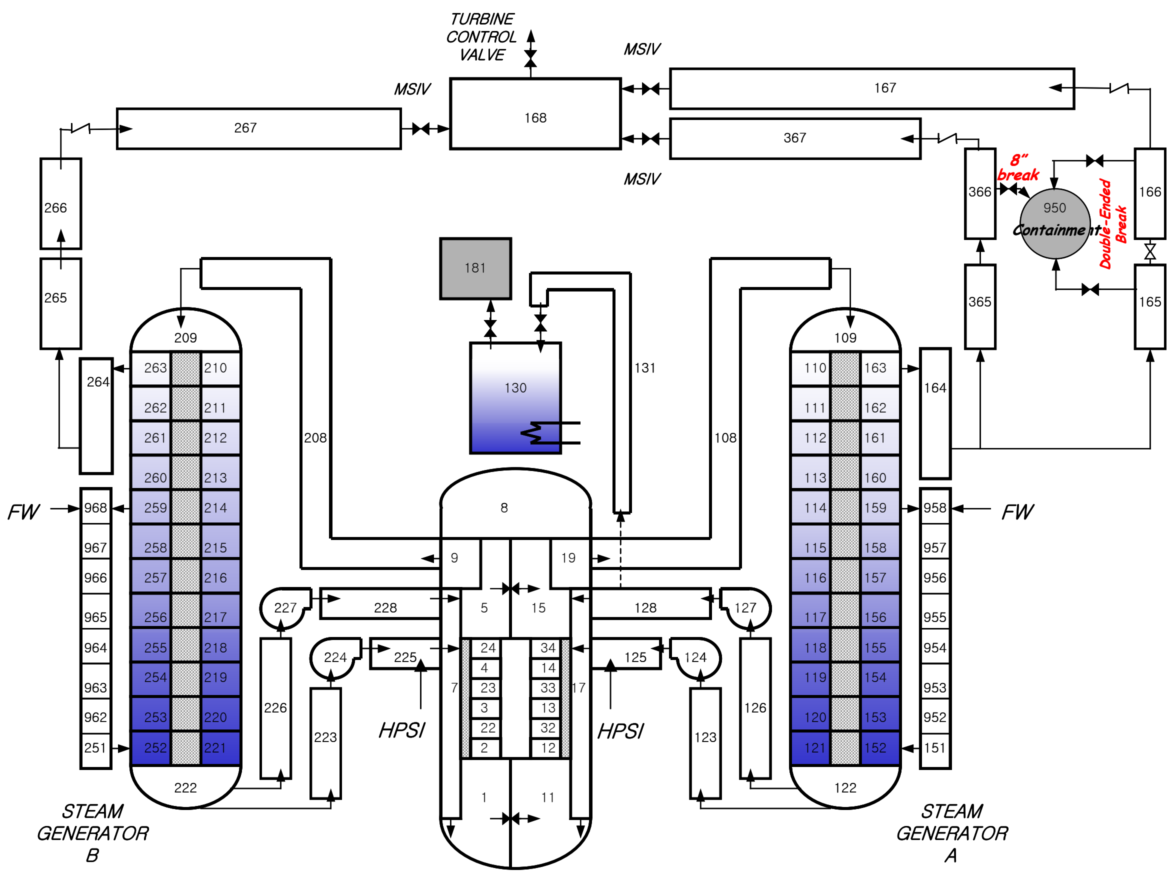


Figure 2. TMI-1 Plant NSSS System Nodalization for MSLB Benchmark Problem

3.3 Steady State Analysis Results

The initial state conditions obtained from RETRAN-3D/MASTER calculation and the Benchmark final specifications, together with their relative differences, are listed in Table I. Even though the RETRAN-3D/MASTER coupled analysis results showed a slight deviation from those by point kinetics calculation, the differences were practically negligible. The table values presented are, therefore, only for point kinetics analysis. As seen from the table, the initial state conditions correspond well with each other, with less than 1% deviations for most parameters. The steam generator steam superheat, however, is showing a noticeably higher difference. This could arise from the steam inventory fluctuation at the steam generator upper region, and the thermal energy release errors due to the steam super heating are not considered to cause appreciable differences during the plant transient behavior.

Table I. Initial State Conditions for OECD MSLB Benchmark Problem

Parameters	Specified Values	RETRAN-3D Results	%diff
Core Power, MW	2772.0	2772.0	0.0
RCS cold leg temperature, K	563.76	563.0	0.1
RCS hot leg temperature, K	591.43	590.7	0.1
Lower plenum pressure, MPa	15.36	15.40	0.3
Outlet plenum pressure, MPa	15.17	15.19	0.1
RCS pressure, MPa	14.96	15.02	0.4
Total RCS flow rate, kg/s	17602.2	17602.0	0.0
Core flow rate, kg/s	16052.4	16052.1	0.0
Bypass flow rate, kg/s	1549.8	1549.8	0.0
Pressurizer Level, m	5.59	5.599	0.2
Steam Flow per OTSG, kg/s	761.59	760.7	0.2
OTSG outlet pressure, MPa	6.41	6.41	0.0
OTSG outlet temperature, K	572.63	569.4	0.6
OTSG superheat, K	19.67	16.3	17.1
Initial SG inventory, kg	26000	26326	1.3
Feedwater temperature, K	510.93	510.1	0.2

3.4 Transient Analysis Results

The major sequence of events of interest during the MSLB and their occurrence times are listed in Table II, with the break incident set to time 0.001 sec. As soon as the main steam lines are broken, steam is released through the faulted-side steam generator. Meanwhile, due to the check valve between common header and broken steam line, steam from the intact steam generator does not reach the broken area, being directed to the turbine. This continues until the turbine stop valve is engaged.

The reactor scram occurs with 0.4 sec delay after high neutron flux trip setpoint is reached. Then, both turbine stop valve and main steam isolation valve are closed, ceasing the steam release from the intact side steam generator. The steam from the broken side steam generator continues to be released through the broken area since the flow path is not isolated by the valve operations. The pressure for the intact side steam generator is then elevated, due to its isolation, causing the steam line safety valve operation until the pressure stabilizes to the set point. High pressure safety injection is started 25 sec after the primary side pressure is reduced to the setpoint.

Figures from 3 to 9 depict system transient behaviors of main interests from the RETRAN-3D point kinetics analysis, RETRAN-3D/MASTER standard analysis, and RETRAN-3D/MASTER refined calculation.

Table II. Major Sequence of Events

Event	Time (s)		
	Refined Calc.	Standard Calc.	Point Kinetics
Break open	0.001	0.001	0.001
High neutron flux setpoint reached	5.81	5.91	5.41
Reactor trip	6.21	6.31	5.81
Turbine stop valve closes	6.71	6.81	6.31
Main steam line isolation valve closes	7.21	7.31	6.81
Steam line B small safety valve opens	6.95	7.05	6.54
Steam line B safety valve groups 1, 2 open	7.02	7.11	6.59
Steam line B safety valve groups 1, 2 close	7.23	7.33	6.85
Steam line B small safety valve closes	7.32	7.42	6.92
Steam line B small safety valve opens	8.02	8.12	7.63
Steam line B safety valve groups 1, 2 open	8.09	8.18	7.69
Steam line B safety valve groups 1, 2 close	8.45	8.53	8.10
Steam line B safety valve groups 1, 2 open	8.73	8.83	8.32
Steam line B safety valve group 3 opens	9.37	9.47	8.96
Steam line B safety valve group 3 closes	25.18	25.40	28.12
Steam line B safety valve groups 1, 2 close	32.82	33.44	N/A
High pressure safety injection starts	43.15	43.29	44.14
Steam line B small safety valve closes	51.18	52.05	N/A
Broken SG dry out	~80	~80	~80
Point of max. Power after trip	69.(34.7%)	69.(37.3%)	68.(58.0%)
Transient ends	100.	100.	100.

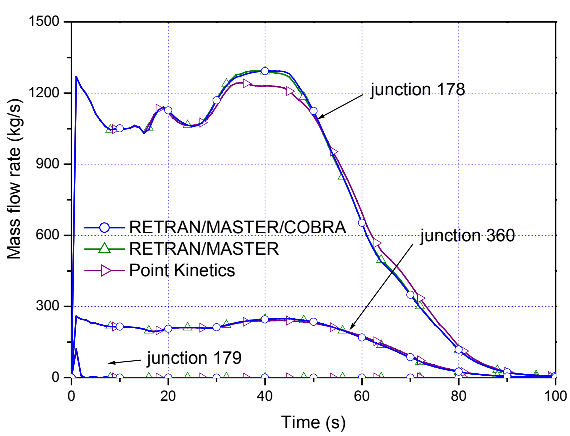


Figure 3. Break Flow Rates

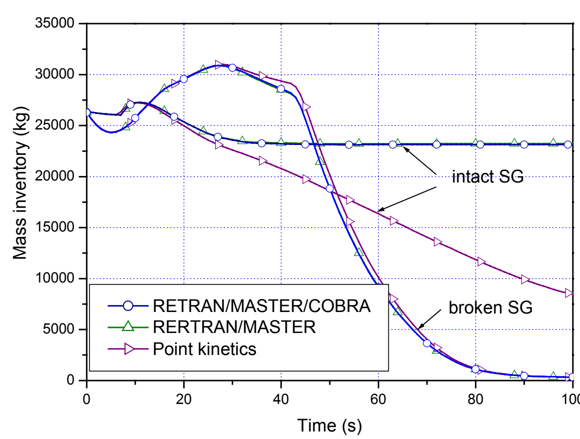


Figure 4. Steam Generator Feedwater Inventories

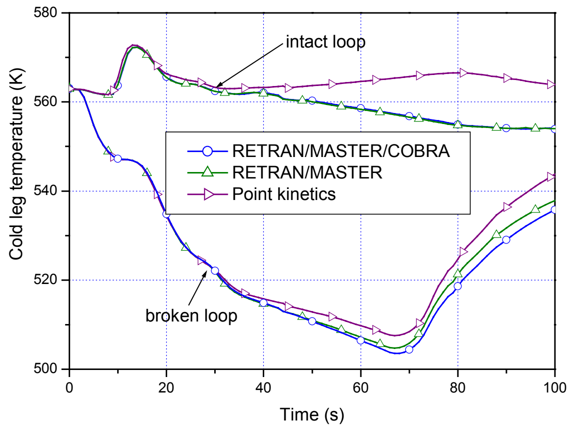


Figure 5. Coolant Temperatures for Cold Legs

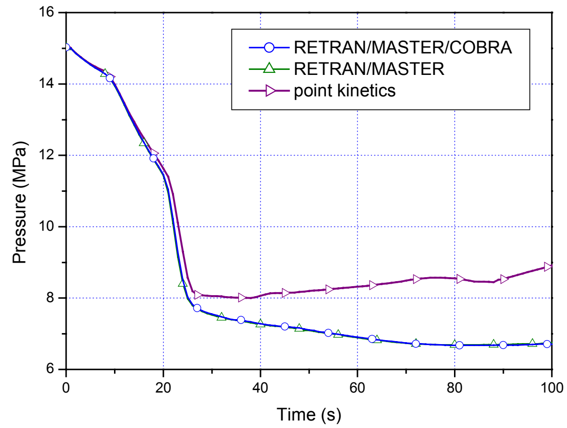


Figure 6. Pressurizer Pressures

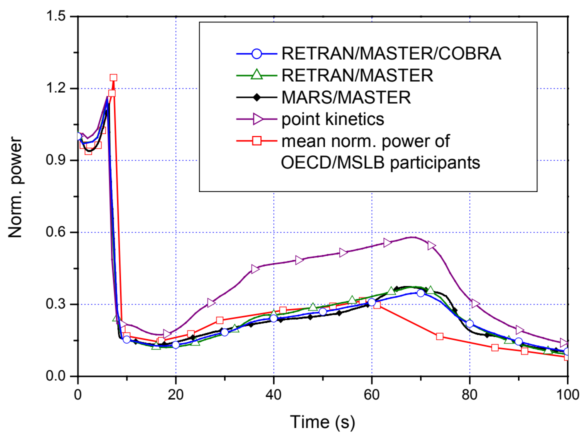


Figure 7. Reactor Core Powers

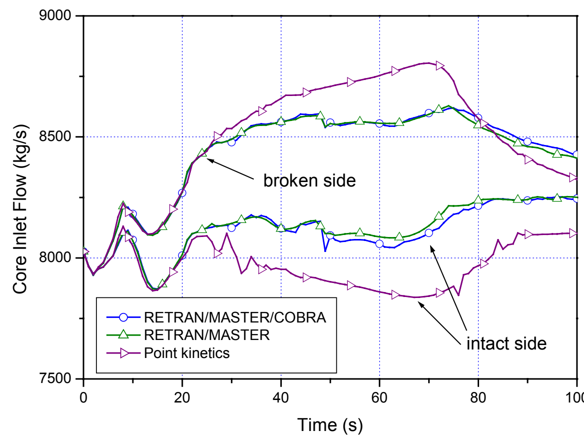


Figure 8. Core Inlet Flow Rates

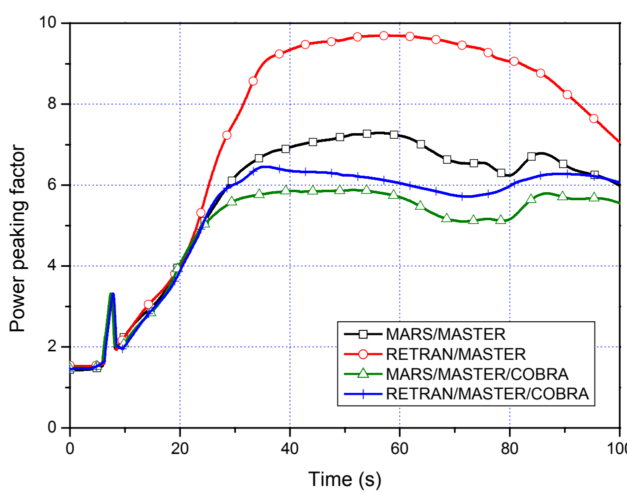


Figure 9. Power Peaking Factors

Figure 3 shows break flow rates from these analyses for both 24" and 8" breaks. Once the break is initiated, critical flow condition is immediately reached, reaching maximum flow rate. The next peak appears at around 40 sec after the break when feedwater is barely supplied. The flow rate continues to decline as feedwater inventory in the steam generator is reduced. Break flow from intact side steam line diminishes almost immediately due to the check valve between the common header and the main steam line break. Figure 4 shows feed water inventory changes for both intact and broken side steam generators. The feedwater inventory for the broken steam generator is reduced initially due to the large amount of steam release, but it recovers by the feedwater supply. The inventory start to reduce at a higher rate starting at around 45 sec when feedwater is no more supplied. As seen from this figure, the feedwater inventories for the faulted side steam generator are quite similar for both point kinetics analysis and RETRAN-3D/MASTER coupled analysis. This is basically due to the fact the break flow is critical flow for both analyses. In case for the intact side steam generator, however, the trends are quite different. The reactor power increase is relatively lower for the coupled analysis and the intact side valves remain closed during the initial 50 secs, allowing no inventory changes during this period. In case for point kinetics modeling, due to the higher core power increase, a few opening of the steam line safety valves leads the continuous decrease in inventory. Figure 5 shows the temperature variations of the cold legs. The temperatures for the break side cold leg are seen to show steeper drop due to the excessive cooling effect caused by steam release. The trend continues until around 70 sec when the cooling effect by steam release starts to be outweighed by the reactor power. The cooling is reduced as the steam generator feedwater inventory is lowered. It is noted that the feedwater for the steam generator in the break side almost drains out at around 80 sec (See Figure 4). The intact side cold leg temperature starts to increase soon after the turbine trip. It remains more or less uniform values as the open/close operations of the main steam line safety valves.

The pressurizer upper region pressures are predicted as seen in Figure 6. In point kinetics analysis, the increased reactor power raised the primary side pressure. The high pressure injection signal, due to low RCS pressure, appears at ~18 sec. The real injection initiates at ~43 sec.

The reactor power variations are depicted in Figure 7, for RETRAN-3D point kinetics analysis, RETRAN-3D/MASTER standard/refined calculations, and MARS/MASTER analysis along with the mean normal values by the OECD/MSLB participants [8]. In all cases, the reactor powers are initially seen to decrease slightly. The negative reactivity by increased fuel temperatures caused the initial power recess in this case. Here, the fuel temperature increase is due to reduced coolant flow by the coolant contraction. It is noted that the moderator temperature coefficient and trip rod worth for the point kinetics calculation were given as specified in the OECD MSLB Benchmark specifications [7]. This explains the seemingly inconsistent steeper power recovery rate by point kinetics calculation for the similar moderator temperature behavior, especially during 20~40 secs (see Fig. 5 & 7). The core inlet flow rates, reflecting the coolant Cooldown effects, are as seen in Figure 8. After the initial power decrease, the coolant of lowered temperature rushes into the core, raising reactor power steeply as seen from the figure. At ~5 sec after the break, the reactor reaches the setpoint (114%) of its nominal power, and then trips. Once a reactor loses its nominal power substantially, the positive reactivity by low moderator temperature partly recovers the power. The reactor power continues to increase to the local maximum value at ~70 sec when primary side cooling effect ceases due to the feedwater drainage. At this point, moderator temperature reverts and starts to increase, causing negative moderator reactivity. The reactor power decreases thereafter.

As seen from Figures 3 to 8, the core-integrated characteristics, such as reactor core power, are not showing substantially different trends between RETRAN-3D/MASTER standard and refined calculations. From the local point of view, however, the refined calculation provides noteworthy improvement. This is demonstrated in Figure 9 where 3-D power peaking factors are compared. When the maximum return to power is observed, at ~60 sec, RETRAN-3D standalone T/H analysis shows Fq value of 9.7, whereas refined subchannel calculation using COBRA III-C/P predicts Fq of 6.5, lowering by 33%. This is due to the power flattening effects not only for the radial direction, but also along the core axial direction, coming from the detailed T/H and Doppler feedback effect. Here, Fq is lower for MAS/MASTER analysis than the value by RETRAN-3D/MASTER simulation. The 18 radial nodes for MARS/MASTER, rather than 2 for RETRAN-3D/MASTER, could have led to such differences with even higher power flattening effects.

4. SUMMARY AND CONCLUSIONS

A coupled code system based on the best estimate T/H code RETRAN-3D and 3-D reactor kinetics analysis code MASTER is developed, thereby establishing unified code system where interaction between T/H model and reactor kinetics are efficiently treated. Two different coupling methods are applied, standard analysis and refined analysis. In the standard analysis, T/H conditions are all handled by RETRAN-3D, whereas in refined analysis, COBRA III-C/P built in MASTER determines more detailed T/H conditions by incorporating subchannel analysis. The core inlet and outlet conditions for the refined analysis are provided by RETRAN-3D.

As a verification test, RETRAN-3D/MASTER coupled analyses were performed against OECD NEA MSLB Benchmark Problem prepared for evaluating coupled analysis of system T/H code and multi-dimensional reactor core kinetics codes. All three analysis methods of RETRAN-3D point kinetics, RETRAN-3D/MASTER standard and refined analyses were performed. The calculation results confirm successful integration of RETRAN-3D and MASTER, by showing similarities and consistencies among their predictions. As for computational time, approximately 1560 seconds were required to simulate 100 seconds of refined RETRAN-3D/MASTER calculation, which is considered reasonable for practical applications. By incorporating the multi-dimensional reactor kinetics code into the T/H code, the interaction between T/H model and reactor neutronic model could be handled efficiently. Especially, the refined calculation of RETRAN-3D/MASTER is seen to lower the power peaking factor by as much as 33%, compared to their standard coupled analysis.

ACKNOWLEDGMENTS

This work has been performed as a part of the Long-Term Nuclear R&D Program supported by the Ministry of the Science and Technology (MOST).

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