Numerical Analysis of EBR-II Shutdown Heat Removal Test-17 using 1D Plant Dynamic Analysis Code coupled with 3D CFD Code

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Abstract. In sodium-cooled fast reactors, a natural circulation is expected to remove the core decay heat when the plant gets into a station blackout. From a perspective of reactor safety, the core hot spot temperature arising in the natural circulation should be evaluated accurately. To this end, Japan Atomic Energy Agency is trying to couple a 1-D plant dynamics analysis code Super-COPD and a 3-D CFD code AQUA to solve the thermal-hydraulic field in the whole plant under natural circulation condition. To validate the coupled analysis code, the code is applied to an analysis of EBR-II shutdown heat removal test in the cooperation with Argonne National Laboratory. The test simulated a protected loss of flow accident by introducing a simultaneous trip of the two primary pumps and the control rod scram. The obtained numerical results reasonably agree with the measured data, which demonstrates the validity of the coupled code.

Key Words: fast reactor, natural circulation, thermal-hydraulics, 1D-3D CFD analysis

1. Introduction

The importance of decay heat removal during a long-term loss of power supply has been recognized again after the Fukushima Daiichi nuclear power plant accident. When the sodium-cooled fast reactors (SFRs) get into a station blackout, a natural circulation is expected to remove the decay heat from the core and eventually to discharge the decay heat to the atmosphere by air cooling systems. During the past three decades, a lot of works have performed to estimate the capability of the natural circulation to remove the decay heat. Among them, to estimate the core hot spot temperature arising in the natural circulation has been an important matter in particular.

It may be difficult, however, to treat the core hot spot temperature as a criterion of reactor safety. The reason is that a quantitative estimation of the natural circulation requires an accurate balance of the thermal inertia and pressure loss in the reactor system where the temperature distributions and the sodium flow rate strongly connect each other. Thus, Japan Atomic Energy Agency (JAEA) has developed a plant dynamics analysis code Super-COPD, which can treat the flow re-distribution and the inter-subassembly heat transfer by modelling all subassemblies as independent channels, and the inter-subassembly gap flow paths in the core barrel as a flow network [1, 2].

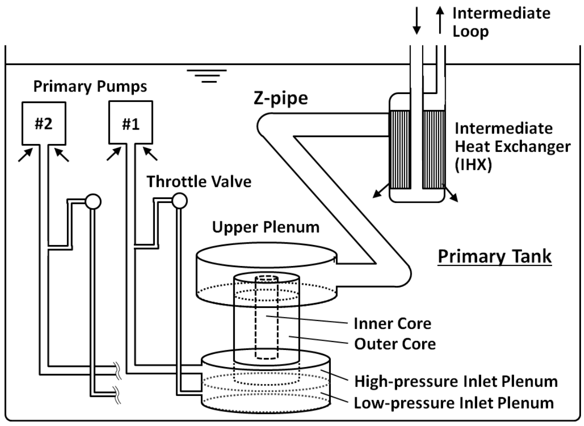
First, Super-COPD code has been applied to the analysis of Shutdown Heat Removal Test (SHRT) -17 performed by using the Experimental Breeder Reactor II (EBR-II) as a benchmark in IAEA's Coordinated Research Project (CRP) [3]. This test was conducted by Argonne National Laboratory. The SHRT-17 test simulated a protected loss of flow accident by introducing a simultaneous trip of the two primary pumps and the control rod scram. The obtained numerical results showed that the Super-COPD could provide a numerical result of the core hot spot temperature during the natural circulation decay heat removal in SFRs. A discrepancy was found, however, between the numerical result and the test result in the place where the sodium flow becomes multi-dimensional, e.g. in the plenum and the horizontal pipe [4].

Thus, JAEA is trying to couple a 1-D plant dynamics analysis code Super-COPD and a 3-D CFD code AQUA to solve the thermal-hydraulic field more precisely in the natural circulation arising in SFRs. This paper describes the analytical models and the coupling method of the codes, and then the numerical results obtained by applying the codes to EBR-II SHRT-17 test are compared with the test results of the natural circulation behavior.

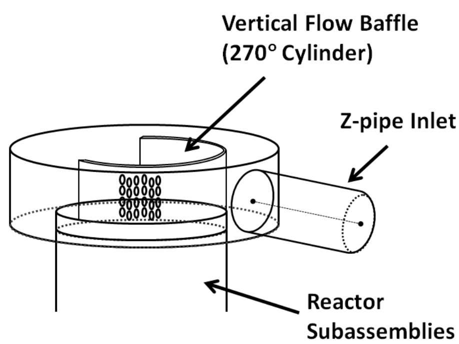
2. EBR-II SHRT-17 Test

Shutdown Heat Removal Test (SHRT) -17 was conducted to simulate the protected loss of flow event by using the Experimental Breeder Reactor II (EBR-II) in 1984. The plant was operated at full power and full flow as an initial condition, and then a simultaneous trip of two pumps was started with the control rod scram. The purpose of this test was to illustrate the natural circulation could remove the decay heat without any problems.

Figure 1(a) shows the primary tank layout including the reactor, the intermediate heat exchanger, Z-pipe and two primary pumps. All the major primary system components are submerged in the primary tank. Two primary pumps provides the sodium flow from the primary tank to the two inlet plena of the core. The sodium flow entering the high-pressure inlet plenum goes upward through the inner core subassemblies, while the sodium flow entering the low-pressure inlet plenum goes upward through the outer core subassemblies. Those sodium flows are mixed in the upper plenum, then the mixed sodium flow goes into the intermediate heat exchanger through the outlet piping, Z-pipe. As shown in FIG. 1(b), the upper plenum consists of two regions: a cylindrical region above the inner and outer core subassemblies and an annular region surrounding the cylindrical region. The vertical flow baffle plate plays a role of mixing the sodium flows.



(a) Primary tank layout



(b) Upper plenum configuration

FIG. 1. EBR-II Plant Overview

3. Analytical Models

JAEA is trying to couple a 1-D plant dynamics analysis code Super-COPD [1,2] and a 3-D CFD code AQUA [5] to solve the thermal-hydraulic field in SFR plants under the natural circulation condition. The former is used to solve the thermal-hydraulic field in a whole plant, while the latter is applied to the core upper plenum and in the half of Z-pipe.

3.1. 1-D Plant Dynamics Analysis Model

The one-dimensional flow network model for primary heat transport system (PHTS) of EBR-II is illustrated in FIG. 2. The upper plenum is divided into two mixing volumes; one is corresponding to the cylindrical region above the inner and outer core subassemblies, the other one is corresponding to the annular region surrounding the cylindrical region. The inlet plenum is divided into the high-pressure plenum volume and the low-pressure plenum volume. The core subassemblies are modelled with the independent flow channels, as shown in FIG. 3(a), to calculate the flow re-distribution caused by the buoyancy force under the natural circulation condition. The sodium flow rate of the inner core subassembly and of the outer core subassembly is calculated according to the pressure difference of ΔP_i and ΔP_o, respectively. The sodium temperature of the subassemblies is calculated in each of the center region and the six peripheral regions (see FIG. 3(b)). The radial heat transport paths and the sodium flow paths between the wrapper tubes are depicted in FIG. 3(b) and 3(c). The core thermal power, the rotational speeds of the primary pumps, the sodium inlet temperature and the sodium flow rate of the intermediate heat exchanger in the secondary loop side are given as the boundary conditions. TABLE I gives the correlations of the friction factor and the heat transfer coefficient used in Super-COPD.

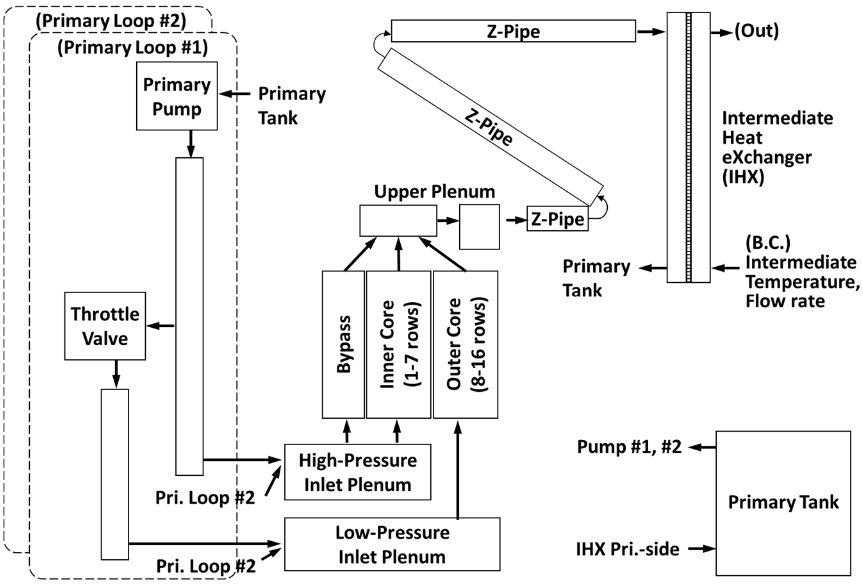
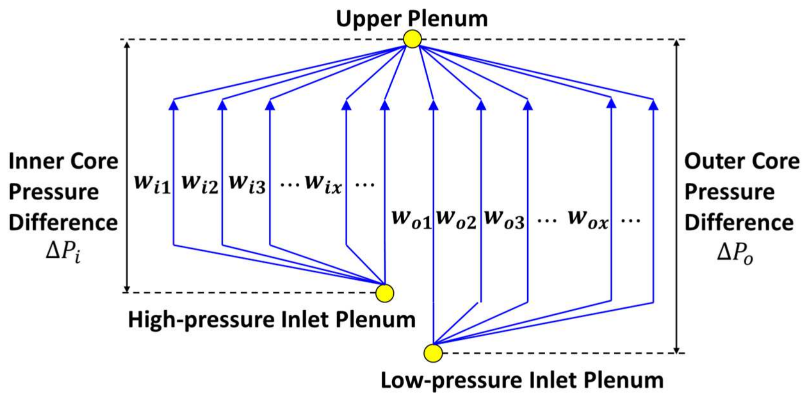


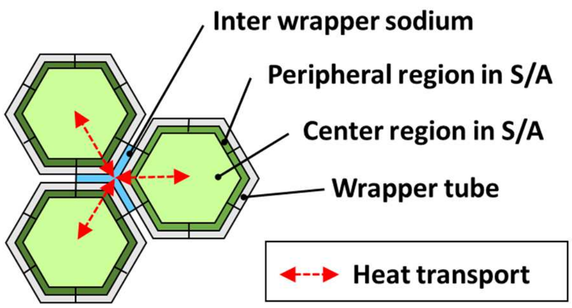
FIG. 2. Model for primary heat transport system (PHTS) of EBR-II

TABLE I. CORRELATIONS USED IN 1D PLANT DYNAMICS ANALYSIS MODEL

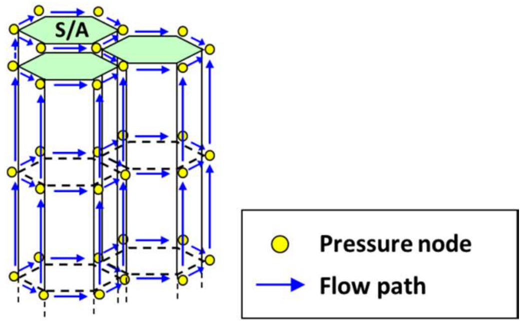
Locations	Correlations
Friction Factor	
Fuel Pin Bundle	Cheng, Todreas[6]
Inter-Wrapper Tube	Parallel Plate Correlations
IHX Tube Bundle	Pipe Flow Correlations
Heat Transfer Coefficient	
Fuel Pin Bundle	Lyon[7]
Wrapper Tube	Parallel Plate Correlations
IHX Tube Bundle	Mikityuk[8]
Z-Pipe	Seban, Shimazaki[9]



(a) Subassembly flow



(b) Radial heat transport



(c) Inter-wrapper gap flow

FIG. 3. Core thermal-hydraulic model

3.2. 3-D CFD Model

The 3-D CFD calculations are performed with AQUA code. The three-dimensional model of the core upper plenum and the lower half of Z-pipe is illustrated in FIG. 4. The vertical flow baffle is modeled as a wall with the permeability of 0.2 and with a gap to the top of the core upper plenum. The thermal baffle attached at the top of the core upper plenum cannot be modeled due to lack of data on the structure. The total cell number is 131,094. For calculation of the conservation equations of momentum and energy, the 2nd-order QUICK-FRAM scheme [10] is used to discretize convective terms and the 2nd-order central difference scheme is used to the remaining terms. A fully-implicit method of SIMPLEST-ANL [11] is employed. The RNG k-e model is used for turbulence modeling. Heat loss from the Z-pipe to the primary tank is considered by using the equivalent thermal conductivity through the Z-pipe, the outer tube and the stagnant sodium filled in a gap between the two tubes. As for boundary conditions, the sodium temperature and the flow velocity at the outlet of all subassemblies and the sodium temperature in the primary tank are given by Super-COPD code.

3.3.Coupling Method of 1-D Plant Dynamics Analysis Code with 3-D CFD Code

The 1-D plant dynamics analysis code Super-COPD is coupled with the 3-D CFD code AQUA to calculate the natural circulation behavior in the upper plenum and in the Z-pipe. The outlet of a subassembly channel in Super-COPD code is connected to corresponding 4 meshes on the core top in AQUA code at the interface. All the meshes at the outlet of the lower half of Z-pipe in AQUA code are connected to the inlet of the upper half of Z-pipe in Super-COPD code.

The pressure loss and the gravity head in the upper plenum and in the lower half of Z-pipe are calculated by using AQUA; these results are given to Super-COPD. The flow velocities in all the subassemblies calculated by Super-COPD are given to AQUA as the boundary conditions.

In the calculation of the sodium temperature field, the sodium temperature at the upstream of the interface between Super-COPD and AQUA is used. During the rated power operation, the sodium temperature at the outlet of all the subassemblies calculated by using the Super-COPD is given to AQUA as the boundary condition. When the sodium flow rate is very low in the core, the sodium flow may go back to an unheated subassembly from the upper plenum. In this case, the sodium temperature near the outlet of subassembly in the upper plenum calculated by using AQUA is given to Super-COPD.

A weak coupling methodology (sequential two-way coupling methodology) is used to couple the codes (see FIG. 5). In this coupling methodology, the 3-D CFD is firstly conducted with the boundary conditions given by the 1-D plant dynamics analysis results in the previous time step, and then the 1-D plant dynamics analysis is conducted with the boundary conditions given by the 3-D CFD results in the same time step. The two codes are synchronized within 0.01 sec by using a code coupling platform developed by JAEA.

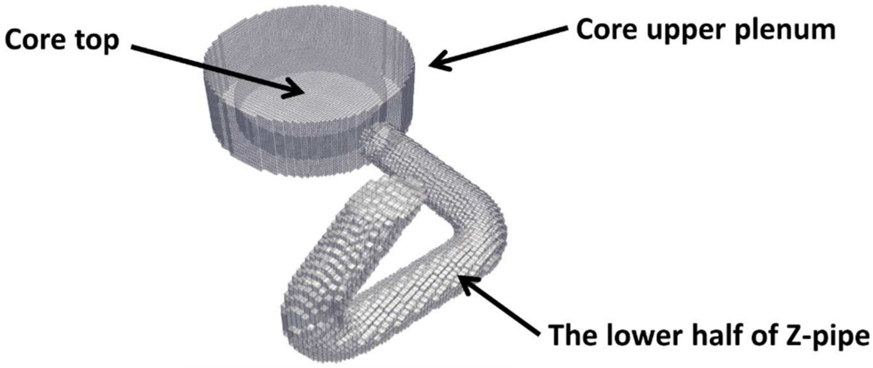


FIG. 4. Model of the upper plenum and the lower half of Z-pipe for EBR-II.

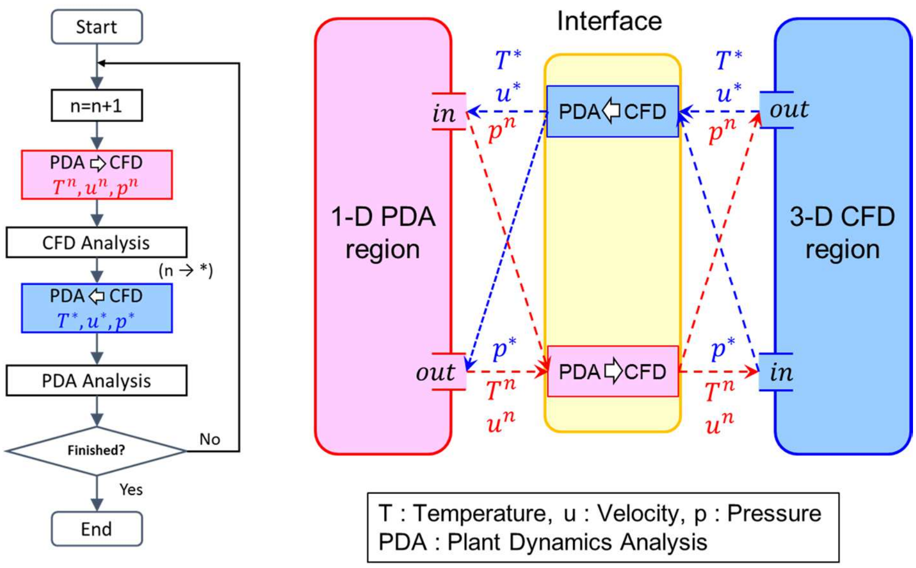


FIG. 5. Weak coupling methodology between 1-D code and 3-D CFD code.

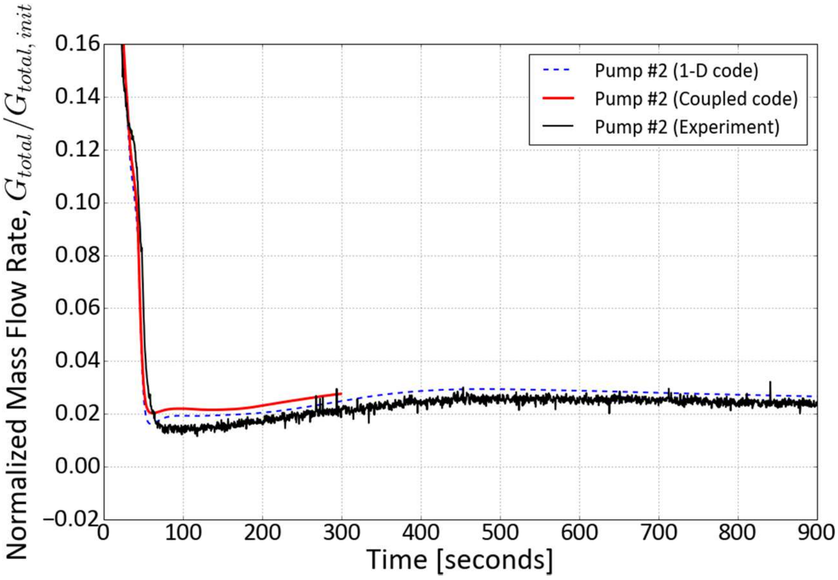
4. Numerical Results and Discussions

Numerical simulations of EBR-II SHRT-17 test were performed by using the coupled code. The core thermal power, the rotational speeds of the primary pumps, the sodium inlet temperature and the sodium flow rate of the intermediate heat exchanger in the secondary loop side are given as the boundary conditions. The sodium mass flow rates, the pressure differences between the core outlet and the discharge of pumps and the sodium outlet temperature of intermediate heat exchanger in the secondary loop side are given as the initial conditions shown as “Ref.” in TABLE II. The initial steady state condition was obtained by running null-transient calculations of the 1-D plant dynamics analysis code and the 3-D CFD code as a stand-alone code, and then running an additional null-transient calculation for stable coupling.

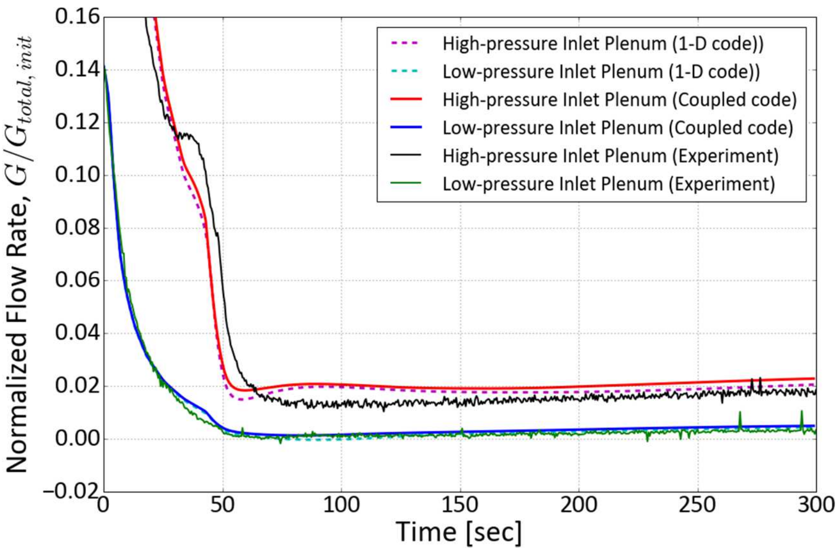
FIGURE 6(a) shows the sodium flow rate through the pump #2. In the experiment, the flow rate coasted down during the initial 60 sec after a simultaneous trip of the two pumps at 0 sec and then recovered to 2 - 3 percent of the initial flow rate due to the buoyancy force. The numerical results using the coupled code were almost the same as the 1-D code results in 300 sec. The flow rate after the initial 60 sec was slightly larger than the results of the 1-D code due to the difference in the estimation of the pressure loss and the gravity head in the Z-pipe. The flow rates from the pump #2 to the high- and the low-pressure inlet plenums are shown in FIG. 6(b). There were some discrepancy between the numerical results and the experiment in the high-pressure inlet plenum flow rate. At about 40 sec, since the pump #1 stopped earlier than the pump #2, the high-pressure inlet plenum flow through the pump #2 increased to balance the pressure between the primary loops connected to the two pumps. The numerical results of the coupled code and the 1-D code show that the flow rates increased only slightly at about 40 sec and the flow coasted down about 10 sec earlier than the experiment. In addition, the flow rates increased temporarily at about 90 sec when the measured flow rate was the lowest. It can be observed that there were some differences between the numerical results of the coupled code and the 1-D code. Thus, the multi-dimensional thermal-hydraulic behavior in the core upper plenum and in the Z-pipe may be influential to the primary flow rate when the flow rate is low under the natural circulation condition.

TABLE II. INITIAL STEADY STATE OF THE NUMERICAL RESULTS

	Ref.	Cal.
Power (MW)	57.3	57.3
Sodium Mass Flow Rate (kg/s)		
Inner Core	386.6	387.5
Outer Core	65.3	65.3
Pump #1	234.0	233.9
Pump #2	233.6	233.9
Pressure Difference from Core Outlet (kPa)		
Discharge of Pump #1	251.9	246.8
Discharge of Pump #2	245.0	246.5
Sodium Temperature (K)		
IHX Intermediate-side Outlet	714.2	715.6



(a) Flow rate through pump #2



(b) Flow rates from pump #2 to high- and low-pressure inlet plenums

FIG. 6. Sodium flow rate in EBR-II SHRT-17 test.

The subassembly outlet sodium temperature calculated in the 1-D code are given as the boundary conditions in the 3-D CFD calculation. Three subassembly outlet temperatures of numerical results are compared with the experimental data in FIG. 7. The overall response of temperature at each subassembly outlet was in good agreement with the experiment. However, after 50 sec the 2nd row driver slightly over-predicted, while the 6th row driver under-predicted the experimental results. And also, there was a phase lag at the peak point of the 2nd row driver. The over-/under-prediction of the temperatures may be solved by more accurately evaluating the radial heat transport between adjacent subassemblies in the core. The phase lag may be caused by the discrepancy in the flow rate before 60 sec in FIG.6, because the lower flow rate of subassembly leads to the delay of the hot sodium transport from the core region to the subassembly outlet.

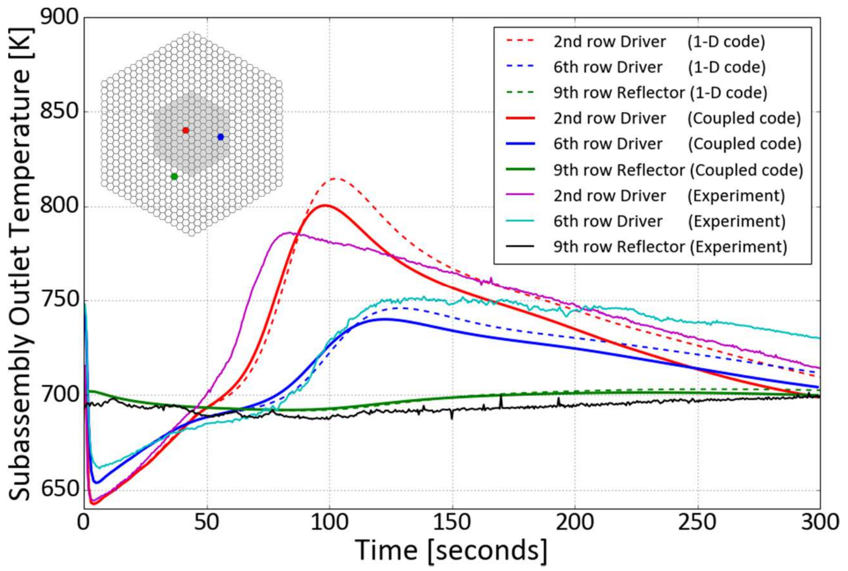


FIG. 7. Subassembly outlet sodium temperature in EBR-II SHRT-17 test.

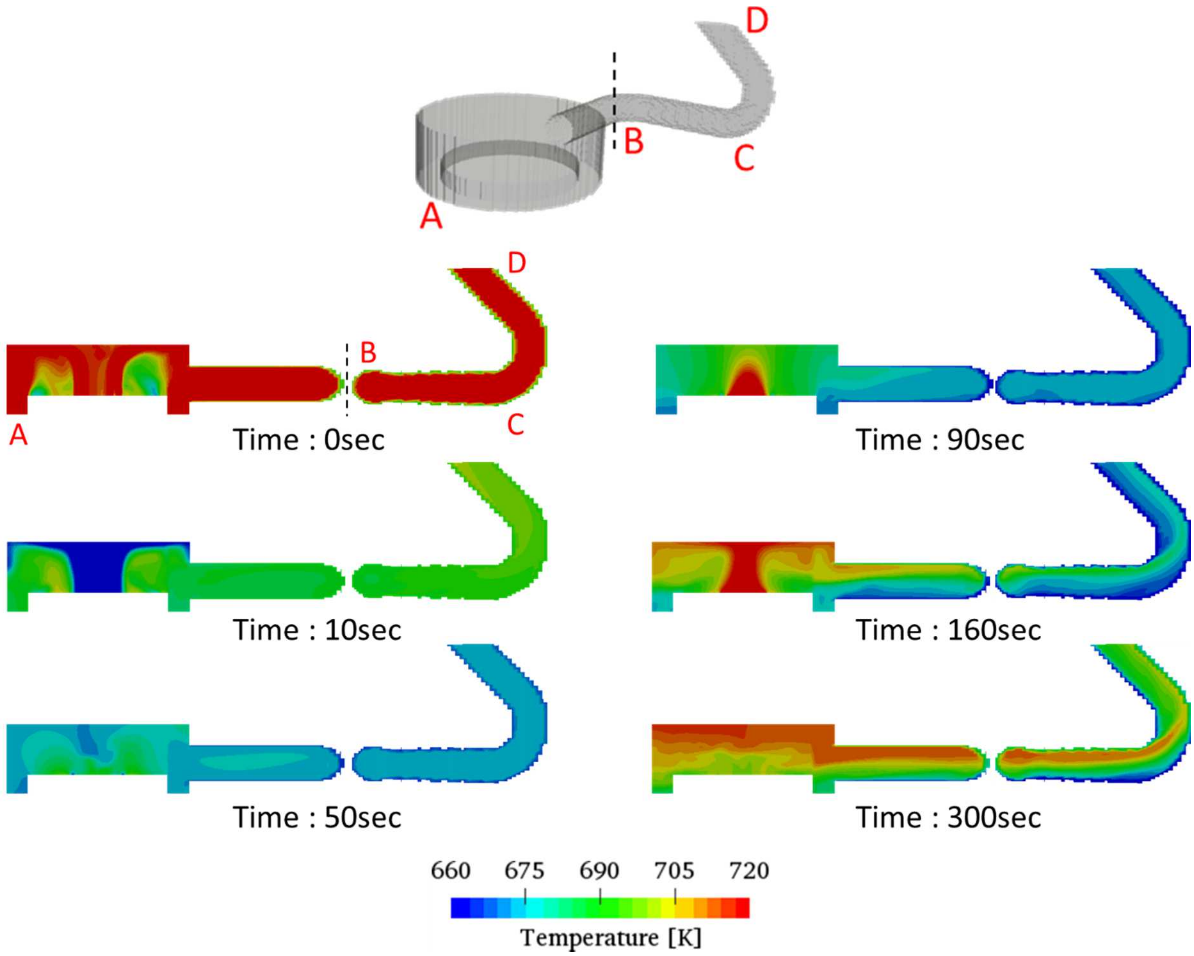


FIG. 8. Sodium temperature distributions in the upper plenum and in the lower half of Z-pipe calculated by using the coupled analysis code.

The calculated temperature field in the upper plenum and in the Z-pipe is shown in FIG. 8. In the initial steady state (0 sec), the hot sodium outflow from the inner core went up to the ceiling of the upper plenum and mixed with the cold sodium outflow from the outer core. The sodium in the Z-pipe had a uniform temperature. When the pump trip was started with the reactor scram, the sodium temperature in the cylindrical region in the upper plenum rapidly decreased, because cold sodium came out from the inner core (10 sec). After the end of flow coastdown (50 sec -), the sodium temperature in the upper plenum started to increase, and hot sodium went to the Z-pipe in which the heat loss to the primary tank became significant. After 160 sec, thermal stratification occurred in the Z-pipe and the hot sodium passed through the cold sodium without mixing.

The vertical distribution of sodium temperature at the outlet of cylindrical region in the upper plenum is shown in FIG. 9. As can be seen, the thermal stratification in the plenum was simulated and the temperatures at the top, 3/4 and 1/2 elevations in the plenum were in good agreement with the experiment from about 100 to 150 sec. After that, however, the temperatures at the top and at 3/4 elevations in the plenum came closer to the temperature at 1/2 elevation. It should be noted that the thermal baffle attached at the top of the upper plenum may affect the vertical distribution of temperature for a long term, because it reduces the effective mixing volume above the core and holds the sodium above the baffle without mixing.

Future work should therefore consider the effect of the thermal baffle attached at the top of the plenum on the primary flow as well as the effect of the vertical baffle. Further, it is necessary to refine the 3-D CFD modeling for more accurate estimation of the pressure loss and the gravity head in the plenum and the Z-pipe.

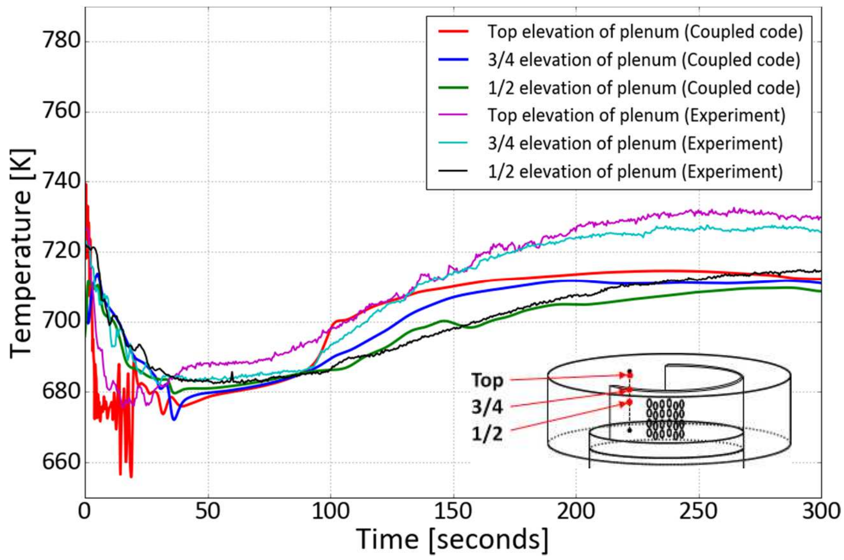


FIG. 9. Vertical sodium temperature distribution at the outlet of cylindrical region in the upper plenum in EBR-II SHRT-17 test.

5. Conclusions

Japan Atomic Energy Agency (JAEA) is trying to couple a 1-D plant dynamics analysis code Super-COPD and a 3-D CFD code AQUA to solve the thermal-hydraulic field more precisely in the natural circulation arising in SFRs. To validate the coupled code, the code was applied to an analysis of EBR-II shutdown heat removal test in the cooperation with Argonne National Laboratory. Super-COPD was used to solve the thermal-hydraulic field in a whole plant, while

AQUA was applied to the core upper plenum and in the lower half of Z-pipe. The two codes were coupled with a weak coupling methodology (sequential two-way coupling methodology) and synchronized within 0.01 sec by using a code coupling platform developed by JAEA. Thermal stratification in the plenum and in the Z-pipe was observed in the results of the 3-D CFD simulation, and it was influential to the transient behavior of the primary flow rate. In the future work, the effect of the thermal baffle attached at the top of the core upper plenum on the primary flow should be considered as well as the effect of the vertical flow baffle. Further, it is necessary to refine the 3-D CFD modelling for more accurate estimation of the pressure loss and the gravity head in the core upper plenum and the Z-pipe.

Acknowledgements

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