

Development of Numerical Analysis Methods for Natural Circulation Decay Heat Removal System Applied to a Large Scale JSFR

O. Watanabe¹, M. Suemori¹, J. Endoh¹, K. Oyama¹, T. Koga², H. Kamide³

¹ Mitsubishi FBR Systems, Inc. (MFBR), Tokyo, Japan

² Central Research Institute of Electric Power Industry (CRIEPI), Chiba, Japan

³ Japan Atomic Energy Agency (JAEA), Ibaraki, Japan

Abstract

A decay heat removal system utilizing passive natural circulation is applied to a large scale Japan Sodium-cooled Fast Reactor. As preparing for the future licensing, a one-dimensional flow network method and a three-dimensional numerical analysis method were developed to evaluate core cooling capability and thermal transient under decay heat removal modes after reactor trip. The one-dimensional method was applied to a water test simulating the primary system of the reactor, while the three-dimensional method was applied to the water test and a sodium test focusing on the decay heat removal system. The numerical results of both methods have turned out to agree well with the test results. And then the thermal-hydraulic behavior under a typical decay heat removal mode of the reactor has been predicted by the three-dimensional method.

Introduction

The Japan Atomic Energy Agency (JAEA) is conducting “Fast Reactor Cycle Technology Development (FaCT)” project [1], in which a decay heat removal system (DHRS) utilizing passive natural circulation was selected as one of the innovative technologies to be applied to the Japan Sodium-cooled Fast Reactor (JSFR) of 1500MWe output. In order to adopt such a passive DHRS for JSFR [2], a water test was performed using a 1/10-scale model [3] which physically simulates the reactor core and the two primary loops with heat exchangers for a Direct Reactor Auxiliary Cooling System (DRACS) and two sets of Primary Reactor Auxiliary Cooling System (PRACS). A sodium test [4] was also performed using Plant Dynamics Test Loop (PLANDTL) in which the heat transfer characteristics of the passive DHRS were examined especially for the heat exchanger of PRACS installed in the upper plenum of Intermediate Heat Exchanger (IHX). These tests have demonstrated that the sufficient natural circulation flow rate required for core cooling is established in the primary system under the reactor trip transient conditions including asymmetrical events of loop operation.

In this study, not only usual one-dimensional flow network method but also a three-dimensional numerical analysis method was developed for the future licensing of JSFR, since there was no experience of applying the passive DHRS to such a large scale Sodium Fast Reactor (SFR), also three-dimensional phenomena might occur in the large diameter pipes and the large plenum in primary components of JSFR. The one-dimensional method was applied to the water test analyses and the three-dimensional method was applied to the water test analyses and the sodium test analyses. Both the numerical results turned out to agree well with the flow rate and

temperature transients measured in the tests. It was also recognized that the three-dimensional method is quite useful for simulating the details of such phenomena as flow oscillations between two parallel cold-leg pipes provided in the primary loops, natural convections and thermal stratifications in whole of the primary components. The thermal-hydraulic behavior under a typical reactor trip condition was predicted by the three-dimensional method for JSFR and the feasibility of the passive DHRS of JSFR was discussed.

1. Decay Heat Removal System of JSFR

The primary system of JSFR consists of a Reactor Vessel (RV) including the core and two primary loops. Each primary loop consists of a hot-leg pipe, a pair of parallel cold-leg pipes and an integral type IHX into which a primary pump is installed. The DHRS of JSFR consists of a combination of one unit of DRACS and two units of PRACS attached to each loop as shown in Figure 1. PRACS has a heat exchanger (PHX) installed in the upper plenum of IHX, a couple of hot- and a cold-leg secondary pipes and an air cooler with an air stack. DRACS has a heat exchanger (DHX) installed in the upper plenum of RV and the secondary system similar to PRACS. These safety systems can be operated under passive natural circulation conditions with no active operation of pumps and blowers. The system requires only DC-power supplied by the back-up battery systems to operate the dampers of the air coolers. The dampers are closing under the usual power operation condition for minimizing the heat loss of the plant and they must open just after the reactor trip for removing the decay heat. The dampers are designed to have redundancy to keep the cooling function even if the single failure criterion is applied. Each air cooler has redundant dampers placed on the air flow path in parallel. Thus a failure to open one damper causes less than 50 percent reduction of the air flow rate. Furthermore, component diversity with respect to DRACS and PRACS is incorporated in the mechanical design of the air cooler dampers to prevent a common mode failure.

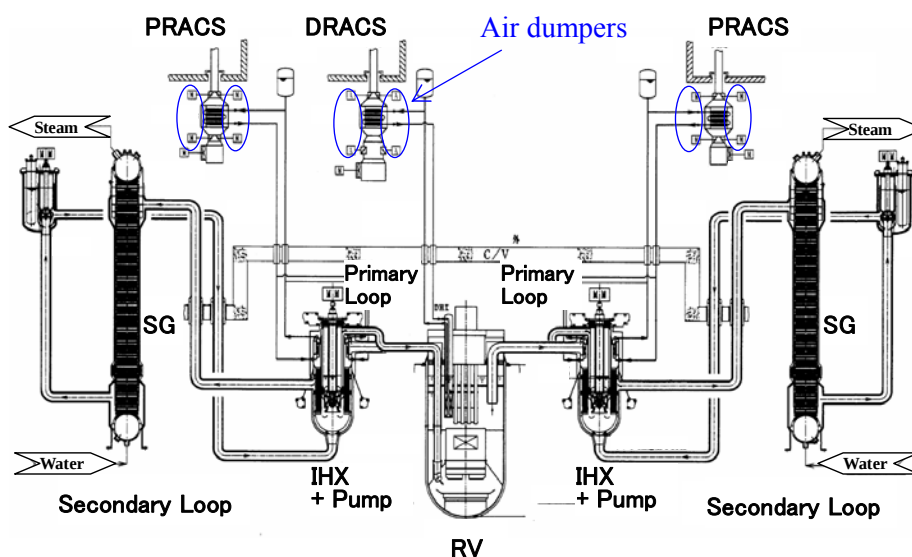


Figure 1 Schematic of JSFR Heat Transport Systems

2. Development of Computational Methods

2.1 One-dimensional flow network method

One-dimensional flow network methods are widely applied to safety analyses for SFR plants because of simplicity and easiness of handling [5]. And the one-dimensional methods have been applied to the safety analysis for the licensing of Japanese prototype SFR “MONJU”. Therefore the one-dimensional method is expected useful for the future licensing of JSFR. However, the one-dimensional method needs to be verified because the flow network model is set up according to unique geometry and heat transport system of each plant. In this study the one-dimensional method has been newly constructed for JSFR in which the development is made mainly for large plenum such as reactor upper plenum, large diameter primary pipes and so on where buoyancy induced three-dimensional complex flows are expected.

2.2 Three-dimensional numerical analysis method

A three-dimensional numerical analysis method has been developed to deal with phenomena such as local natural convections and thermal stratifications in whole of the primary system under decay heat removal conditions. The computational fluid dynamic analysis code “STAR-CD” is employed as the main frame of the three-dimensional method. A whole core thermal-hydraulic code “TREFOIL” [6] and one-dimensional flow network models for simulating the secondary side of PRACS, DRACS and IHX were assembled into the three-dimensional method as shown in Figure 2. “TREFOIL” code can analyze the core thermal-hydraulics taking into account of inter-subassembly radial heat transfer, inter- and intra-subassembly flow redistribution that are quite important phenomena under the natural circulation conditions. RNG k- ϵ turbulent model [7], [8] and a second order advection scheme named MARS (Monotone Advection and Reconstruction Scheme) [9] are applied to the three-dimensional method as a set of options in “STAR-CD” code that tend to analyze flow velocity and temperature gradients more steeply than the other set of options such as standard k- ϵ turbulent model and first order advection scheme, the pressure losses in the core subassemblies are calculated based on the empirical formulas by Cheng and Todreas [10] and that in the tube bundles of the heat exchangers and in the pumps are calculated based on design results of each component. In the design of heat exchangers, the pressure loss in the tube bundle is estimated in total of the frictional loss in the bundle and the form losses at the bundle inlet and outlet. In the design of pumps, the pressure loss in the pump is estimated based on the homologous curve of the pump. These pressure losses are incorporated into “STAR-CD” code as a body force which is a function of the flow velocity in the flow conduit. The other pressure losses such as in pipes with elbows and in plenum are calculated based on the standard logarithmic law of wall friction built-in “STAR-CD” code.

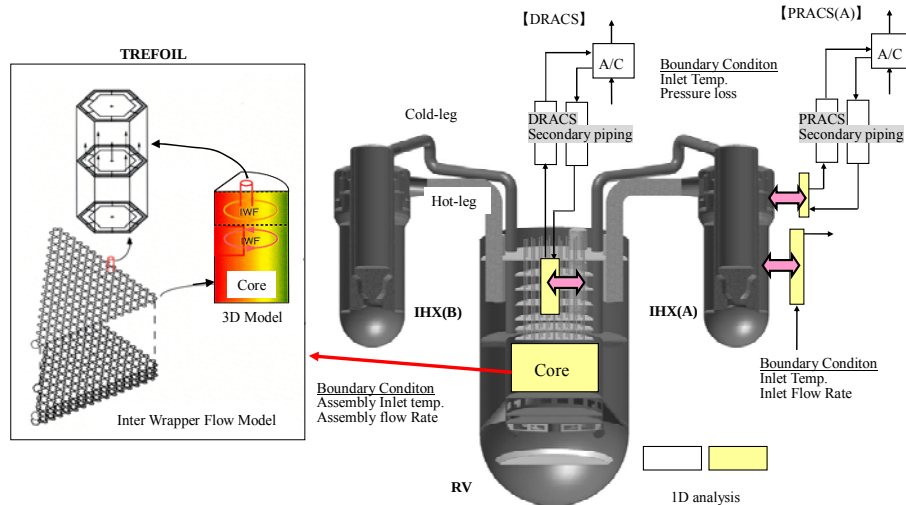


Figure 2 Structure of three-dimensional method

3. Application of the methods to tests

3.1 Outline of water and sodium tests

3.1.1 Water test apparatus

A water test was conducted using a 1/10-scale model at Civil Engineering Research Laboratory of Central Research Institute of Electric Power Industry (CRIEPI). The test apparatus physically simulates RV with DHX and the two primary loops with PHX as shown in Figure 3. The similarity rule of natural circulation tests had been discussed in a previous study for the top-entry type FBR [11], indicating that there were three major dimensionless parameters to be considered: the Euler number (Eu), the modified Boussinesq number (Bo) and the modified Grashof number (Gr). Eu stands for the ratio of pressure loss to inertial force, which is equivalent to half the value of pressure loss coefficient in a continuous flow conduit, and the summation of Eu along the primary circuit is equal to Richardson Number (Ri) that stands for the ratio of natural circulation force to inertial force. $Bo^{1/2}$ and $Gr^{1/2}$ are equivalent to Peclet number (Pe) and Reynolds number (Re) in case of forced convection, respectively. $Bo^{1/2}$ and $Gr^{1/2}$ are convenient for identifying the natural circulation test conditions because the representative flow velocity depends on the heating power. It is ideal to carry out the scale model test based on the similarity rule matching the above dimensionless numbers to those of JSFR. However, it is practically impossible to match the three dimensionless numbers at the same time. Therefore, the water test was performed with matching Eu while $Bo^{1/2}$ of the water test is set at the same order as that of JSFR (about three times larger than that of JSFR). Under this test condition, though $Gr^{1/2}$ equivalent to Re is reduced to about 1/350 of JSFR, its absolute value is as high as several thousands in the primary piping. Therefore, flows during the water test are likely to stay in a turbulent state in major parts of the apparatus. The simulation tests were conducted for representative events selected from the Design Basis Events (DBEs) in JSFR. Typical events were “loss of the external electric power supply”, “sodium leakage in a secondary loop”, and “sodium leakage in a secondary loop of PRACS”.

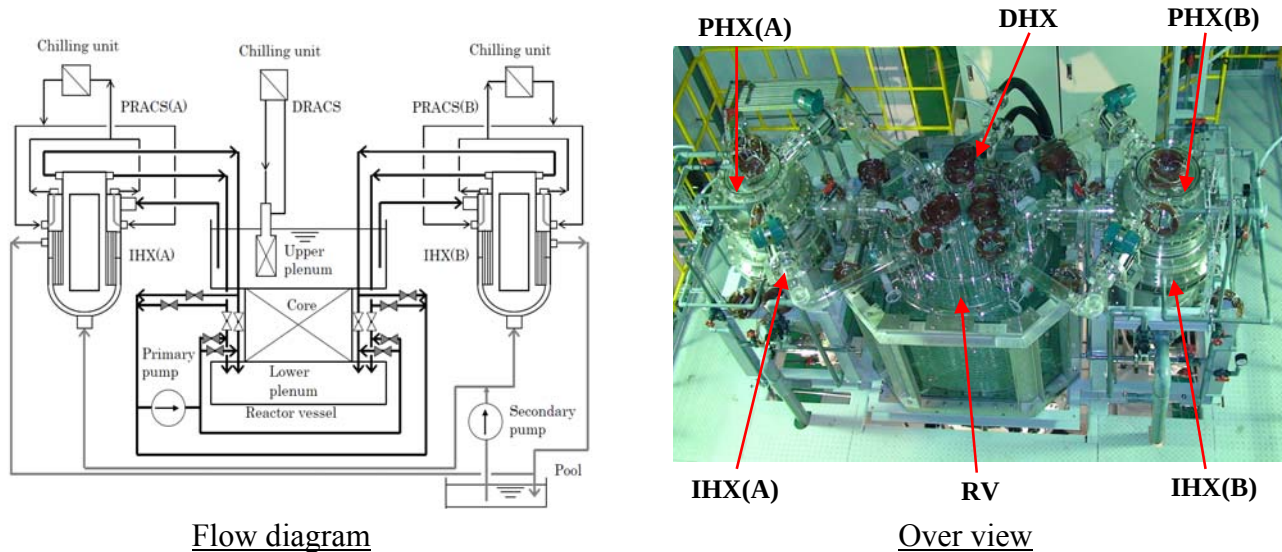


Figure 3 Water test apparatus

3.1.2 Sodium test apparatus

The sodium test was conducted using Plant Dynamics Test Loop (PLANDTL) [12] of JAEA O-arai research and development center. A new IHX partially simulating PHX was manufactured and installed in PLANDTL as shown in Figure 4. The primary system consists of the core with 7 mock-up fuel subassemblies, RV simulating the upper plenum, a couple of the hot- and cold-leg pipes and the new IHX. The secondary system consists of the main air cooler and a couple of the hot- and cold-leg pipes. The auxiliary cooling system for decay heat removal consists of PHX, a couple of the hot- and cold-leg pipes and the air cooler with the air stack and so on. The geometry of the primary system of PLANDTL is not an exact mock-up of the JSFR design as compared with that of the 1/10-scale water test apparatus, however the sodium test can be conducted under the equivalent Eu condition of the DHRS of JSFR. The heat removal capability of PRACS was determined as 100kW taking into account of the Ri similarity and the model scale of nearly 1/8 of JSFR. On the other hand, the PHX consists of straight heat transfer tubes arranged along the inner shroud installed in the upper plenum of IHX. Under high flow rate conditions of the rated power operation the primary coolant flows down uniformly through the annular region between the outer and the inner shrouds with a low pressure loss. Under the low flow rate conditions of the natural circulation regime the coolant flows around the PHX heat transfer tubes due to the buoyancy effect and flows down transferring the heat on the tube walls. It was required for the sodium test to employ heat transfer tubes of the same scale as the actual plant in order to verify the heat transfer characteristics of PHX under the equivalent Pe condition with JSFR.

The objectives of the sodium tests are to evaluate the heat transfer characteristics of PHX under the natural circulation condition and to confirm the transition behaviour from the stationary forced circulation conditions to the natural circulation conditions employing an integrated system consisting of the primary sodium system, the secondary sodium system of PRACS and the tertiary air cooling system.

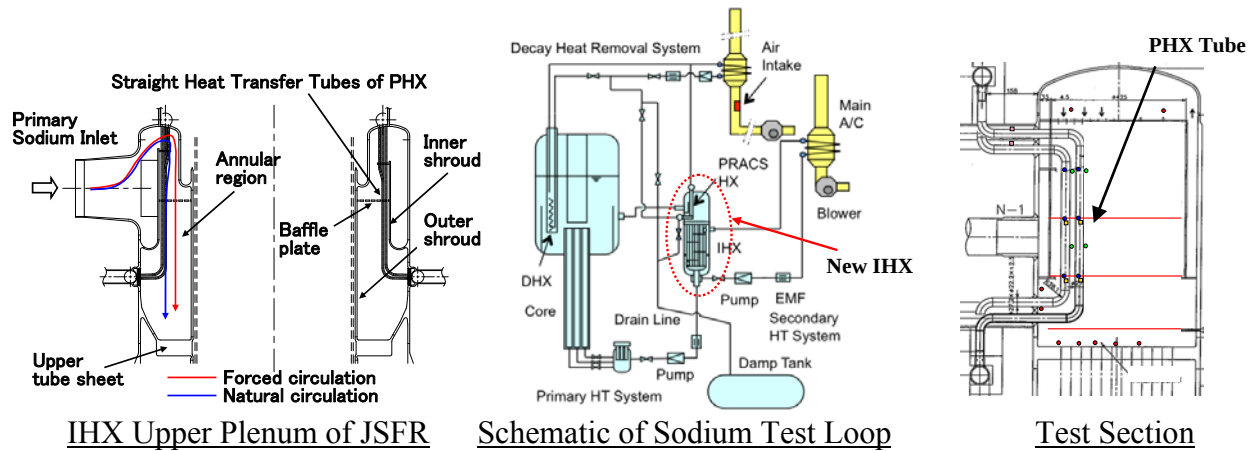


Figure 4 Sodium test apparatus

3.2 Application of one-dimensional method

The one-dimensional method was applied to the 1/10-scale water tests, in which two cold-leg pipes arranged in a primary loop were modeled as one flow conduit and the characteristics of the apparatus such as the pressure losses in the primary system, the heat transfer coefficients in IHX, PHX and DHX, and the thermal-hydraulic properties of water were incorporated into the one-dimensional method as input data. The boundary conditions of the analyses such as the core heating power decrease, the primary pump coast-down curves, the secondary side inlet flow rates and the temperatures of IHXs, PHXs and DHXs were also incorporated into the one-dimensional method as input data. The pump head proportional to the square of the measured pump rotation was added as a body force to the momentum change of the flow conduit. The basic plan of the flow network model applied to the water test is the same as that to JSFR.

The application has been made for 5 representative events which were simulated by the water test. Among them two events are picked up in this paper as typical cases. The analysis result for “loss of the external electric power supply” is shown in Figure 5 comparing with the test result. Since the two primary loops including PHX are operated symmetrically in this event, the primary flow rate, inlet and outlet temperatures of RV in both loops behave very similarly and the analysis result shows good agreement with the test result.

The analysis result for “sodium leakage in a secondary loop” is shown in Figure 6. In this case the secondary flow rate of IHX with the failure loop-A firstly decreases, then the core heating power rapidly decreases to a decay heat level, and the primary pumps are subsequently tripped. Therefore the inlet temperature of RV on the side of loop-A increases earlier than that of the normal loop-B, the natural circulation force in the loop-A becomes smaller than that in the loop-B, then the primary flow rate in the loop-A decreases to almost zero. Afterwards, the primary flow rate in the loop-A gradually increases to the level of that in the normal loop-B because the cooling due to PHX begins in the upper plenum of both IHXs, then a stable natural circulation is established. The analysis result of inlet and outlet temperatures of RV in the loop-A and the primary flow rate in both loops shows good agreement with the test result as well as in the “loss of the external electric power supply” case.

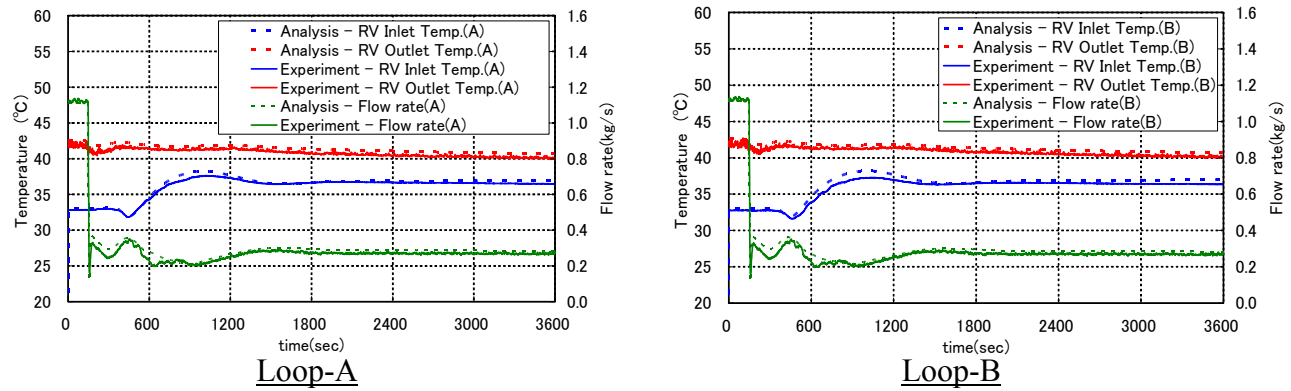


Figure 5 Analysis Results by 1-D method for “Loss of the external electric power supply”

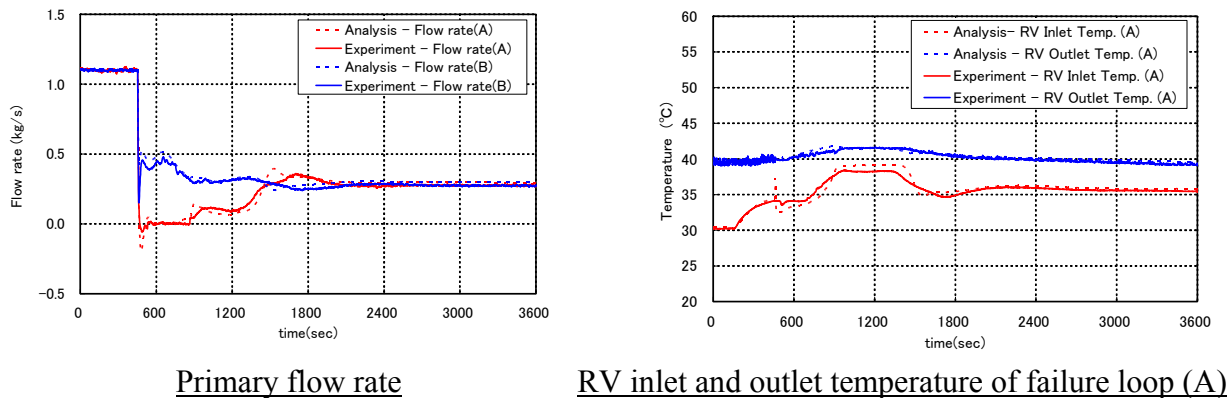


Figure 6 Analysis Results by 1-D method for “Sodium leakage in a secondary Loop “

3.3 Application of three-dimensional method

3.3.1 1/10-scale water test analysis

The three-dimensional method was applied to the 1/10-scale water test in which the three-dimensional mesh division simulating the geometry of the primary components was constructed for “STAR-CD” code as shown in Figure 7. The total number of meshes is about 3,900,000. The characteristic data and the boundary condition data were input into the three-dimensional method as in the one-dimensional analyses. Only the pressure drops due to wall friction in the primary components were internally calculated by “STAR-CD” code. The basic plan of the three-dimensional method applied to the water test analysis is same as that of JSFR.

The application has been made for two representative events that are “loss of the external electric power supply” and “sodium leakage in a secondary loop”. The analysis results for “loss of the external electric power supply” are shown in Figure 8 comparing with the test results. The analysis results of primary flow rates in the two cold-leg pipes arranged in a primary loop show good agreement with the test results including their oscillating phenomena around 600 seconds after starting of the transient. The core outlet temperatures are likely simulated in this analysis. The bumpy temperature transient at the radial shielding outlet around 1000 seconds is caused by a reverse flow in the radial shielding due to the unbalance of buoyancy force in the core during

the short period. The small discrepancies of the temperature transients between the analyses and the test results at the core outlet are expected to be caused by temperature mixing due to local convections. In the water test, each outlet of the heating channel is equipped with the orifice for simulating pressure losses in the core subassemblies including the radial blankets and the neutron shielding. The inner diameter of each orifice is less than several millimeters, so the discharge flow from the orifice makes too small convective flow for the three-dimensional method to analyze in the reasonable computational time. The inlet temperature of the tube bundle in IHX begins to decrease at 500 seconds when the PHX cooling starts. The temperature in lower plenum of IHX just below the tube bundle outlet once decreases due to the low temperature of the secondary side and gradually increases to the IHX inlet temperature level. This temperature increase causes Rayleigh-Taylor instability for an upward flow in the annular region of IHX as shown in Figure 8. The unstable temperature distribution in the annular region causes inlet temperature difference between two parallel pipes of the cold-leg, and it causes unbalance of buoyancy force in the parallel pipes and then the flow rate oscillation occurs.

The analysis result for “sodium leakage in a secondary loop” is shown in Figure 9 comparing with the test result. The analysis result of primary flow rates in the two cold-leg pipes shows good agreement with the test result including their oscillating phenomena. But, in the increase process of the flow rate in the failure loop-A after 800 seconds, the flow rates in the two cold-leg pipes simultaneously increase in the analysis while they increase separately in the test. It is expected to be caused by coincidence of the unstable phenomena. The core outlet temperatures are well simulated in this analysis including the bumpy temperature transient at the radial shielding outlet.

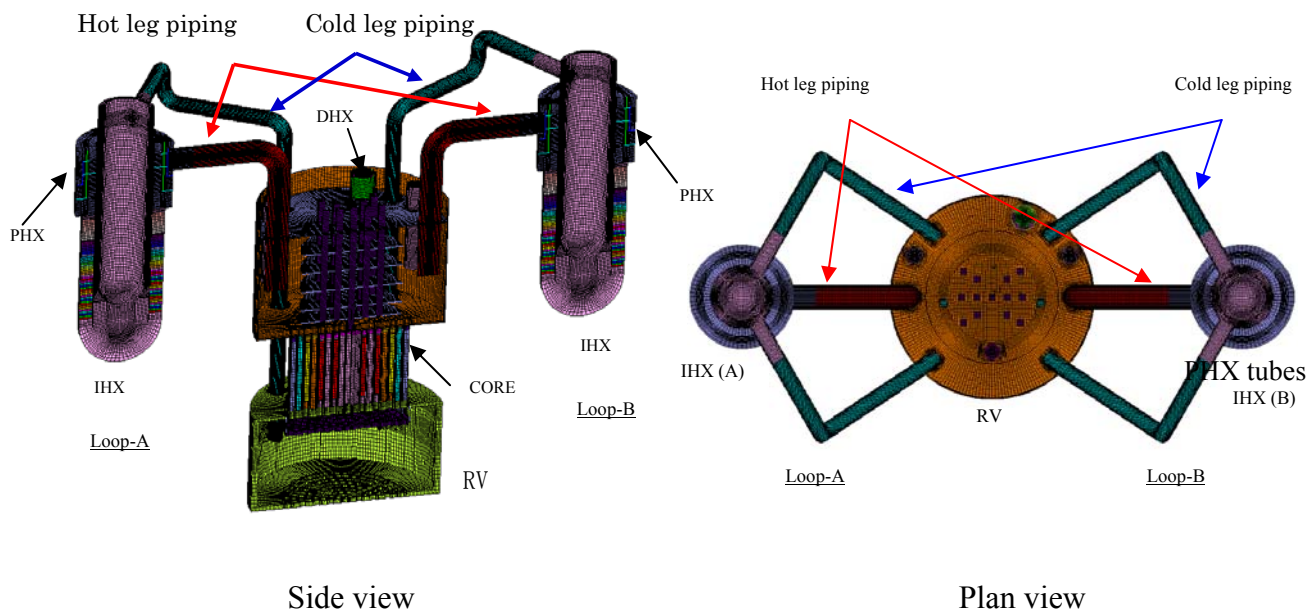
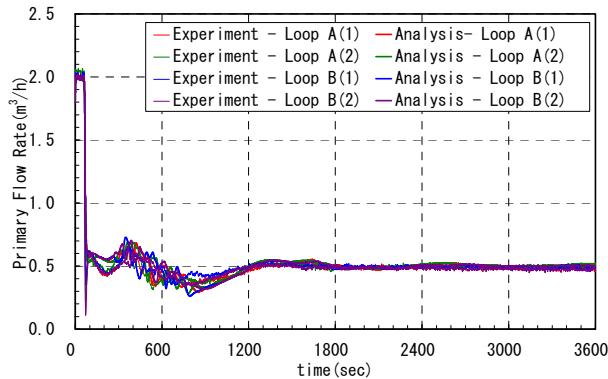
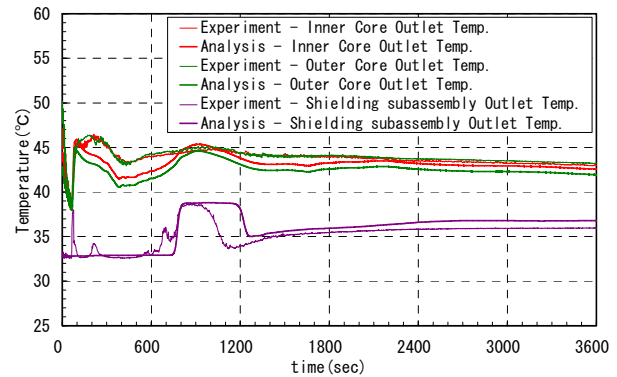


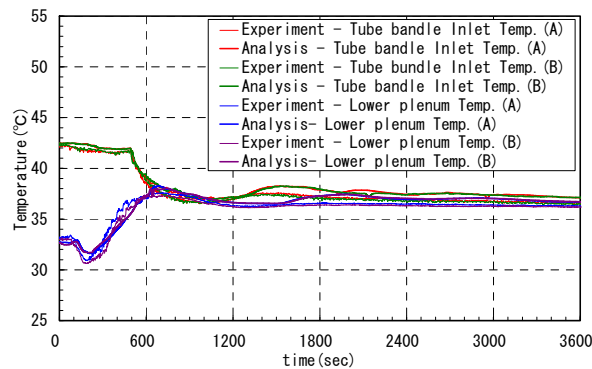
Figure 7 Three-dimensional mesh division for water test apparatus



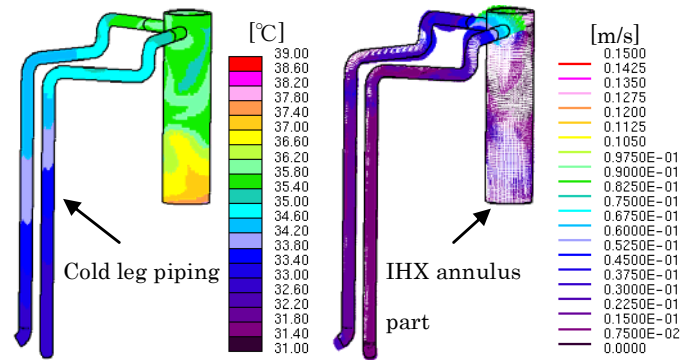
Primary flow rate in cold leg piping



Core outlet temperature

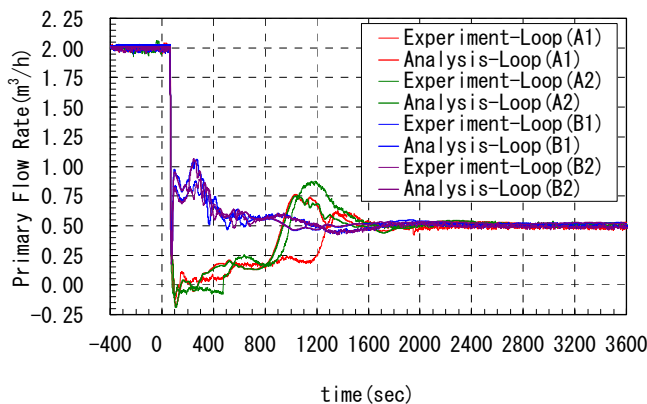


Inlet and outlet temperature of IHX tube bundle

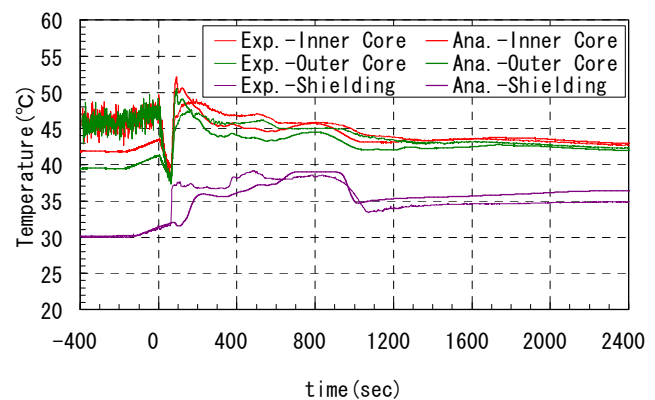


Temperature and flow distribution at 454 seconds

Figure 8 Analysis results by 3-D method for “Loss of the external electric power supply”



Primary flow rate in cold leg piping



Core outlet temperature

Figure 9 Analysis results by 3-D method for “Sodium leakage in a secondary Loop-A “

3.3.2 PLANDTL sodium test analysis

The three-dimensional method was applied to the PLANDTL sodium test in which the three-dimensional mesh division simulating the geometry of the primary system was constructed for “STAR-CD” code as shown in Figure 10. The total number of meshes is about 3,300,000. The one-dimensional flow network models for simulating the secondary side of PRACS and IHX

were assembled into the three-dimensional method. The PRACS model includes PHX, the secondary sodium piping and the air-cooler with the dumper and the air stack. The characteristics of the test apparatus such as the pressure losses in the core, the tube bundle of IHX, the baffle plate of PHX and the secondary side of PRACS including the air cooler with the air stack, and the heat transfer coefficients in the core subassemblies, tube bundles of IHX and PHX and the air cooler, and the thermal-hydraulic properties of sodium were incorporated into the three-dimensional method as input data. But, the pressure loss and heat transfer coefficient due to wall friction in the primary system are internally calculated by “STAR-CD” code. The basic plan of the three-dimensional method applied to the sodium test analysis is the same as that of JSFR.

The application has been made for a sodium test to confirm the transient phenomena from stationary forced circulation conditions to natural circulation conditions employing an integrated system consisting of the primary sodium system, the secondary sodium system of PRACS and the tertiary air cooling system. The analysis result of the sodium test is shown in Figure 11 comparing with the test result. The analysis results of the natural circulation flow rate and hot- and cold-leg temperatures in the primary system show good agreement with the test results, and those in the PRACS are also well simulated. Due to the buoyancy force, sodium cooled on the heat transfer tubes of PHX horizontally diffuses while it locally goes down around the tubes in the upper plenum of IHX. The vertical temperature profile in the upper plenum of IHX shows good agreement between the analysis and the test results. It is expected that the horizontal temperature gradient does not steeply occur on the upper tube sheet of IHX.

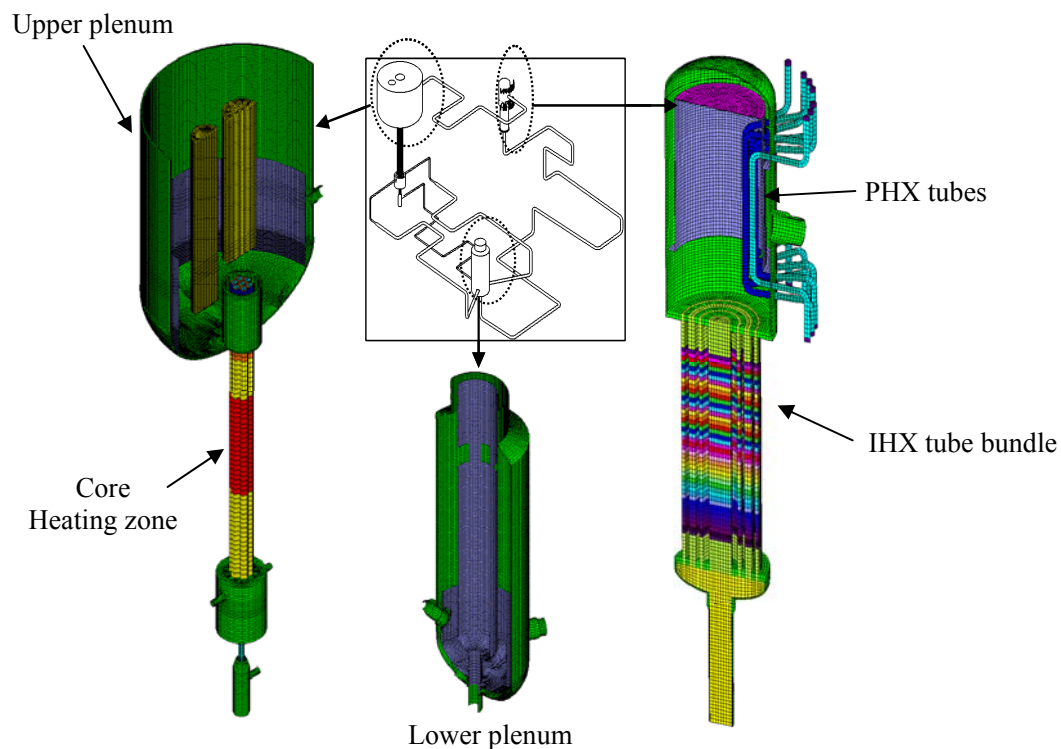


Figure 10 Three-dimensional mesh division for Sodium test apparatus

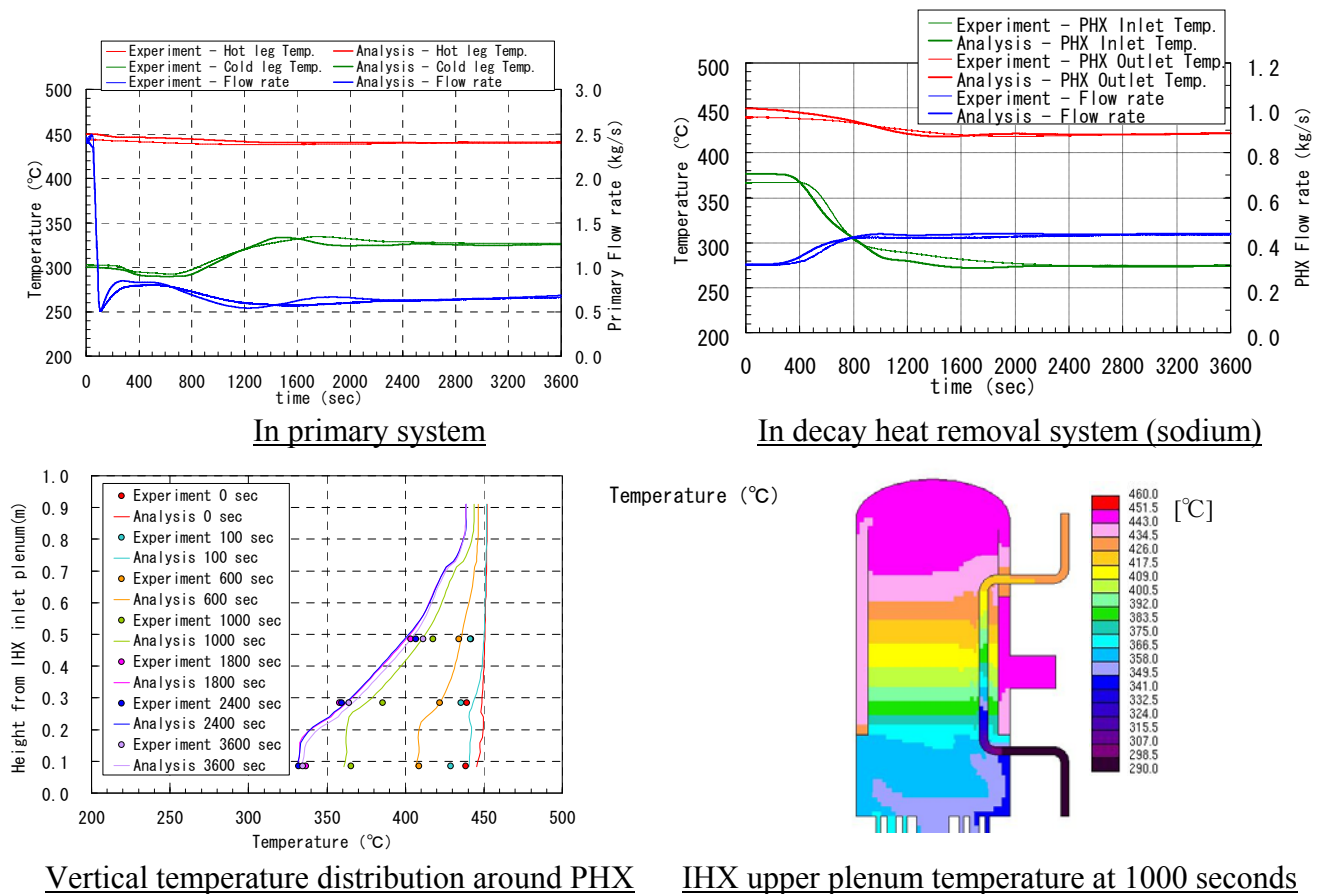


Figure 11 Analysis results by 3-D method

4. Prediction of passive decay heat removal in JSFR

The three-dimensional method was applied to the prediction of a representative transient event, “loss of the external electric power supply”, for JSFR. The three-dimensional mesh division simulating the geometry of the primary system was constructed for “STAR-CD” code as shown in Figure 12. The total number of meshes is about 6,500,000. The analysis model and conditions of the three-dimensional method have been already described in section 2.2. The progress of the event is as follows: the primary and secondary pumps are firstly tripped with their coast-down curves, simultaneously the core heat generation rapidly decreases due to the reactor scram following the decay heat curve, and then the dumpers installed in the air coolers of DRACS and PRACS are opened with the delay time for about 1 minute. The natural circulation in the secondary cooling system is ignored as a safety margin in the prediction analysis.

The analysis results are shown in Figure 13. The primary flow rates in two parallel cold-leg pipes rapidly decrease to about 2.5% of their rated condition due to the primary pump coast down and the minimum flow rate is determined by the natural circulation force in the primary circuit when the primary pumps completely stop. The difference among the minimum flow rates is expected to be caused by a small unbalance of buoyancy forces growing up mainly in the two cold-leg pipes of each loop starting with an infinitesimal numerical error brought in the steady solution of the initial (rated) condition, however the total flow rate in each loop (i.e. A1+A2 or

B1+B2) changes similarly and has almost the same minimum flow rate. The total primary flow rate for the core cooling behaves more stably than the flow rates in two parallel cold-leg pipes or in the primary loops. On the other hand, there are asymmetrical and small differences of the temperature and the flow rate between or among the core cooling loops under the usual power operating condition of actual SFR plants. Uncertain phenomena such as the minimum flow rate in the parallel pipes should be well considered in future design works and safety evaluations for the natural circulation decay heat removal system.

The oscillating phenomena in the parallel cold-leg pipes occur similarly to the water test and its analysis result after the primary pump trip, and then the amplitude of the oscillations increases violently at 1200 seconds because relatively cold fluid enters the vertical part in one of the cold-leg pipes (A1,B2) earlier than the other of the cold-leg pipes (A2,B1) and accelerates the downward flow in the cold-leg pipes (A1,B2) due to its buoyancy force when the fluid cooled by PHX reaches the cold-leg pipes. Afterward the primary flow rates in two parallel cold-leg pipes approach each other and a stable natural circulation is established in the primary system because the two sets of cold-leg pipes including the vertical part are filled with cold fluid cooled by PHX. The fuel bundle outlet temperature in the hottest core subassembly has a peak called “the secondary peak temperature” at around 150 seconds and the next peak called “the third peak temperature” at around 1000 seconds corresponding to the primary flow rate transient mentioned above. The second and the third peak temperatures are about 600°C and 630°C, respectively, and they are low enough compared with the maximum allowable fuel cladding temperature of 830°C in the safety criterion called “anticipated operational occurrences” in JSFR.

On the other hand, the flow rate oscillation between two parallel cold-leg pipes might be picked up from the point of view of the thermal transient for the structure of the primary cooling system, so an additional prediction has been performed in which a certain pressure loss is added to two sets of the parallel cold-leg pipes as a countermeasure. The results are shown in Figure 14. The flow rate oscillation such as the violent increase of the amplitude can be stabilized although slight increases of the second and the third peak temperatures have been observed.

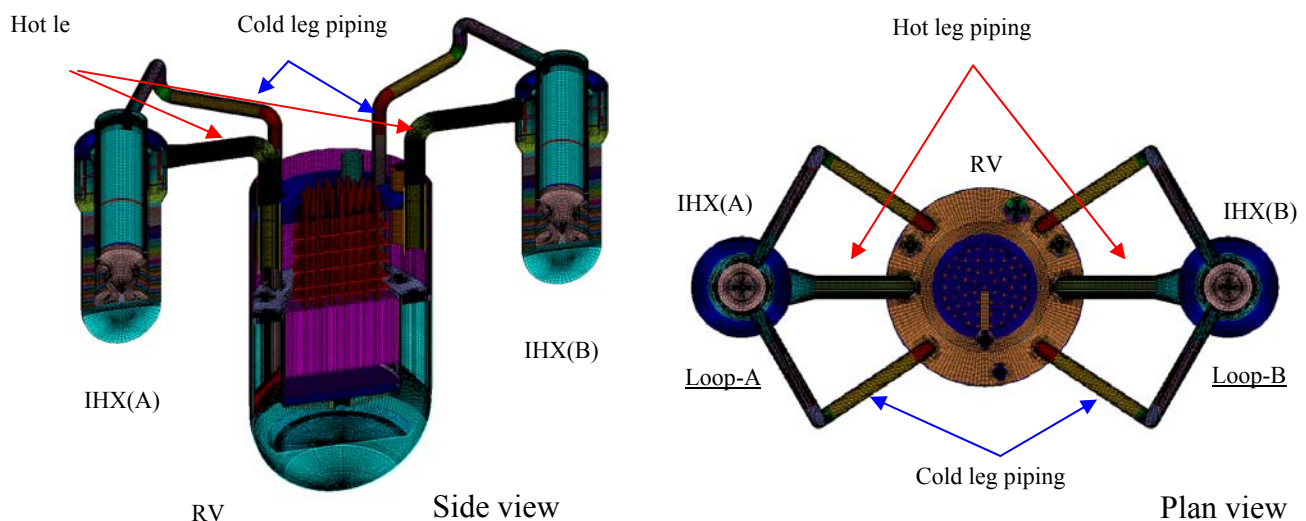
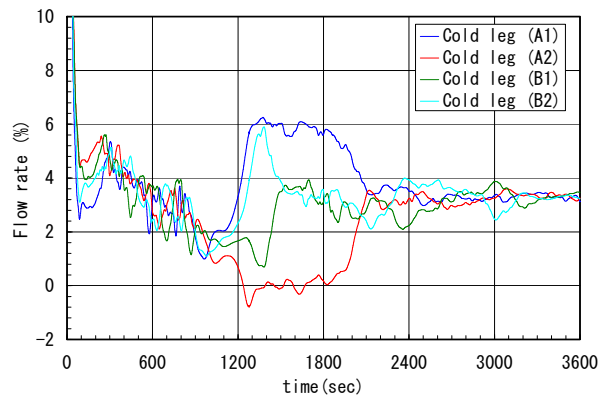
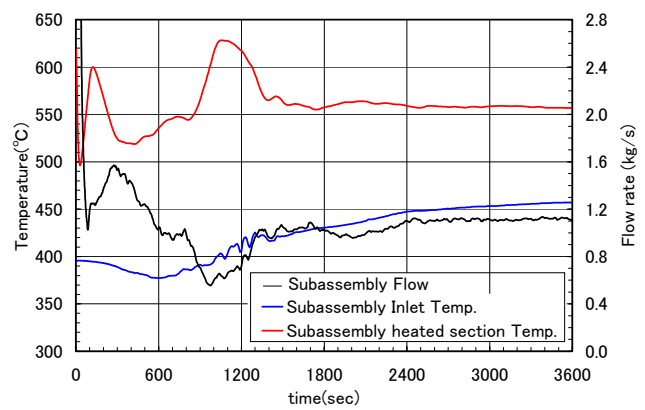


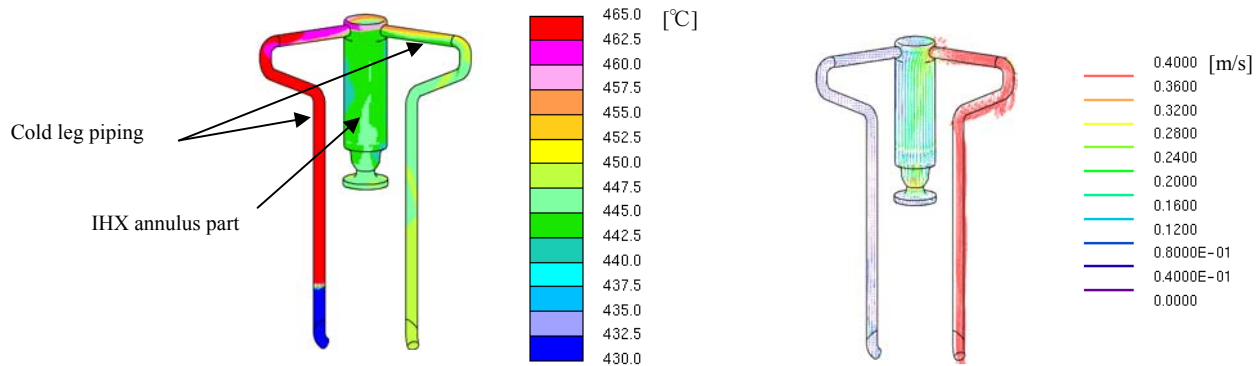
Figure 12 Three-dimensional mesh division for the primary system of JSFR



Primary flow rates in cold leg piping

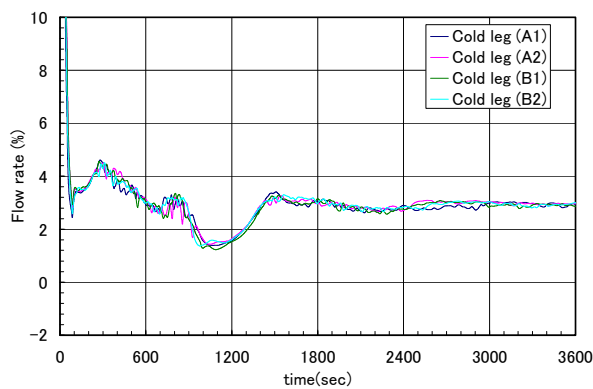


Core thermal transient

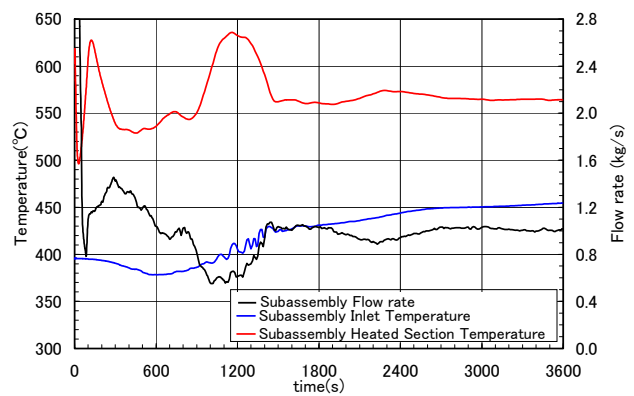


Temperature and flow distribution at 1320 seconds

Figure 13 Analysis results by 3-D method for “Loss of the external electric power supply”



Primary flow rates in cold leg piping



Core thermal transient

Figure 14 Primary flow rate and temperature transients of an additional case

5. Conclusion

In this study, not only usual one-dimensional flow network method but also a three-dimensional numerical analysis method was developed to evaluate core cooling capability and thermal transient behaviors under decay heat removal conditions after a reactor trip aimed for future licensing of JSFR. The one-dimensional method was applied to the water test simulating the primary system of the reactor with a scale of 1:10. The three-dimensional method was applied to the water test and the sodium test employing PLANDTL facility newly equipped with a PHX test section. The analysis results of both the methods have turned out to agree well with the test results. Especially the three-dimensional method made it possible to predict the details of the phenomena such as flow oscillations between two parallel cold-leg pipes arranged in the primary loop, the natural convections and the thermal stratification in whole of the primary components which the actual natural circulation analysis methods could not evaluate. And the thermal-hydraulic behavior in typical reactor trip events was predicted by the three-dimensional method for JSFR. The result has predicted that both the second and the third peak temperatures are low enough compared with the allowable fuel cladding temperature of the safety criterion of JSFR and the flow rate oscillation between two parallel cold-leg pipes occurs more violently than that in the water test. Therefore an additional analysis has been performed in which a certain pressure loss is added to each cold-leg pipe as a countermeasure. The results have shown the flow rate oscillation can be stabilized although slight increase of the second and the third peak temperatures have been observed. As a result, it is expected that the passive decay heat removal system employed in JSFR is feasible and can achieve high reliability. For the future licensing of JSFR, further development and verification studies for the evaluation methods and further optimization of the DHRS design would be necessary.

6. Acknowledgement

This paper describes the outlines of results obtained from one- and three-dimensional computational models development, which is conducted in a study of “Development of evaluation methods for decay heat removal by natural circulation under transient conditions” entrusted to “MITSUBISHI FBR SYSTEMS, INC. (MFBR)” by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

The authors are grateful to Mr. T. Murakami and Dr. Y. Eguchi of CRIEPI and Dr. H. Ohshima of JAEA for useful discussions of natural circulation phenomena and analysis methods.

7. References

- [1] Ichimiya, M., Mizuno, T., and Kotake, S., "A next generation sodium-cooled fast reactor concept and its R&D program", Nuclear Engineering and Technology, Vol.39, No.3, (2007), pp.171-186.
- [2] Kotake, S., Sakamoto, Y., Mihara, T., et al.(2010). "Development of advanced loop-type fast reactor in Japan", Nuclear Technology, 170, pp.133-147.
- [3] Ohyama, K., Watanabe, O., Eguchi, Y., Koga, T., et al.(2009). "Decay heat removal system by natural circulation for JSFR," Proc. of Int. Conf. on Fast Reactors and Related Fuel Cycles (FR09), 08-21P, Kyoto, Japan, Dec.7-11.
- [4] Kamide, H., Kobayashi, J., Ono, A., et al.(2010). "Sodium experiments on decay heat removal system of Japan sodium cooled fast reactor", The seventh Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety(NTHAS7), Chuncheon, Korea, Nov.14-17.
- [5] Yamaguchi, A., et al., "Plant wide thermal hydraulic analysis of natural circulation test at Joyo with MK-II irradiation core", Proc. 4th Int. Top. Mtg. on Nuclear Reactor Thermal-Hydraulics, Karlsruhe, F.R. Germany, Vol.1, (1989), pp.398.
- [6] Watanabe, O., Kotake, S., Kubo, S., Kajiwar, H., Fujimata, K., "Study on Natural Circulation Evaluation Method for a Large FBR", NURETH-8, Kyoto Japan, Proceeding Vol.2, 829-838.
- [7] Yakhot, V., and Orszag, "Renormalization group analysis of turbulence – I: Basic theory", J.Scientific Computing, 1, (1986), pp.1-51.
- [8] Yakhot, V., Orszag, Thangam, S., Gatski, T.B., and Speziale, "Development of turbulence models for shear flows by a double expansion technique", Phys. Fluids, A4(7), (1992), pp. 1510-1520.
- [9] STAR-CD, Methodology, Version 4.02, 2007. Computational Dynamics Limited.
- [10] Shih-Kuei Cheng and Neil E. Todreas , "Hydrodynamic Models and Correlations for Bare and Wire-Wrapped Hexagonal Rod Bundles - Bundle Friction Factors, Subchannel Friction Factors and Mixing Parameters", Nuclear Engineering and Design, Vol. 92, pp.227-251,(1986)
- [11] Koga, T., et al.(1993). " Natural circulation water test for top-entry loop type FBR", NURETH-6, Vol.2, Oct. 1993, pp.1302-1308.
- [12] Kamide, H., Hayashi, K., Isozaki, T., and Nishimura, M. (2001). "Investigation of core thermohydraulics in fast reactors, Interwrapper flow during natural circulation," Nuclear Technology, 133, p.77-91.