

WATER TEST OF NATURAL CIRCULATION FOR DECAY HEAT REMOVAL IN JSFR

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Abstract

Japan Atomic Energy Agency (JAEA) is conducting a design study of Japan Sodium-cooled Fast Reactor (JSFR), in which a decay heat removal system (DHRS) utilizing natural circulation (NC) is applied as an innovative technology. The Central Research Institute of Electric Power Industry (CRIEPI) has carried out a water test to verify the applicability of NC to decay heat removal. The test used a 1/10-scale model of JSFR. Key issues on thermal-hydraulics were identified through simulation tests on representative events. Measures were also proposed to resolve these issues. The present study has demonstrated that a sufficient and stable NC is established in each event.

1. Introduction

The coolant used in the primary and the secondary loops of a sodium-cooled fast reactor (SFR) has a low vapor pressure, a low viscosity and a high thermal conductivity. These properties give big advantages in passive decay heat removal of SFR. That is, a large temperature rise from core inlet to outlet utilizing the low vapor pressure brings large buoyancy forces, the low viscosity decreases pressure drops, and the high thermal conductivity improves efficiency of heat exchangers and avoids rise in temperature of fuel elements. All of these features contribute much to the implementation of decay heat removal utilizing natural circulation (NC) of cooling systems. However, in most of the previous and existing SFRs, such as the Japanese experimental SFR JOYO and the Japanese prototype SFR MONJU, decay heat removal after a scram has relied on forced circulation [1] and NC was utilized to decay heat removal only in hypothetical accidents as all primary pumps stopped. A number of studies have been conducted as safety assessments of NC [2], [3]. A confirmation test at JOYO has clearly demonstrated that decay heat removal is established with NC of the primary and the secondary cooling system [4]. A water test has also shown that NC works quite well in decay heat removal for the demonstration fast breeder reactor of the top-entry type [5]. These reactors appeared to have the capability to remove the decay heat by NC alone. In the future, NC will be utilized for decay heat removal in

various accidents so as to strengthen the safety of SFR. There may be problems on thermal-hydraulics, where increased efforts should be devoted to resolve them [6]. Calling for an enhancement of safety and a reduction in construction costs, a design study has been conducted for a large scale SFR named Japan Sodium-cooled Fast Reactor (JSFR) in “Fast Reactor Cycle Technology Development (FaCT)” project by Japan Atomic Energy Agency (JAEA) [7]. The NC of the cooling systems is utilized to remove the core decay heat in the design basis accidents as one of the innovative technologies introduced in the FaCT project. The JSFR is a loop type reactor and has two primary cooling systems. Each primary loop consists of a pair of parallel cold-leg pipes, one hot-leg pipe and an integral type of Intermediate Heat Exchanger (IHX) into which a primary pump is installed. The Decay Heat Removal System (DHRS) consists of two sets of Primary Reactor Auxiliary Cooling Systems (PRACS's) and a Direct Reactor Auxiliary Cooling System (DRACS). The heat exchanger of PRACS (PHX) is placed in the upper plenum of IHX. The heat exchanger of DRACS (DHX) is installed in the upper plenum of reactor vessel. In decay heat removal, PRACS plays a major role and DRACS is a diverse system to this. The whole DHRS starts to remove the decay heat immediately after the scram. Buoyancy forces are generated both in core and in PHX. These forces are expected to bring about stable flows. Transient behaviors in NC will have a major impact on the structural design of the cooling system. Rapid temperature changes in the primary system should be adequately moderated for securing the structural soundness and countermeasures should be taken in extreme cases.

The Central Research Institute of Electric Power Industry (CRIEPI) has conducted a water test to examine the feasibility of extended application of decay heat removal utilizing NC to various types of accidents. Serious issues in NC were identified and studied through the simulation tests for representative events of JSFR. Previous experiments dealt with the total black-out accident alone, which produced a symmetric behavior on loop operation. Many accidents have been addressed and tested in the present study, including those causing an asymmetric behavior on the loop operation. These extensive tests have been conducted for the first time. The test results were directly converted to actual plant state in order to use in the design study of JSFR and were also applied to validation of the analysis codes. This paper describes an outline of the water test, such as the experimental apparatus, procedure, results, estimation and the conclusion.

2. Experimental apparatus

2.1 Outline of water test equipment

The system configuration of test apparatus is depicted in Figure 1. The apparatus consists of the core, the reactor vessel, the primary cooling loop, the secondary cooling loop, DHRS and the auxiliary systems. Unlike in the case of original design, the primary pump is not installed in the IHX vessel and is located outside of the model due to a small space. The external pump drives forced circulations of the primary loops in the steady state, but NC is simulated inside the model. Accordingly, the flow path must be switched from the forced circulation line to the NC one at the moment of scram. Principal dimensions of the model are equal to a scale of 1/10 of JSFR. The small scale makes the test apparatus impossible to mimic the actual shape especially for the core, the heat exchanger and the pump. Despite the difference in the local geometry, the similarity rule described hereinbelow will be satisfied in the test. For this purpose, an orifice and a control valve

are specially equipped respectively in the fuel assembly and in the primary loop. Their pressure loss coefficients are adjusted to coincide with those of JSFR by isothermal tests representing NC flow. A heater power and a primary flow rate can be programmed to simulate the actual transient. Each of the primary and secondary loops is independently regulated to permit a faithful reproduction of asymmetric event. The primary flow rate is measured in each of four cold-leg paths. In the steady state, the core power is transferred through IHX to the secondary loop connecting to the heat sink of a large pool. Each of PRACS's and DRACS has own cooling unit. The core with a maximum power of 120 kW is divided into an active, a blanket and a shielding region. A pipe channel with built-in bar heater is used as a substitute for the fuel subassembly.

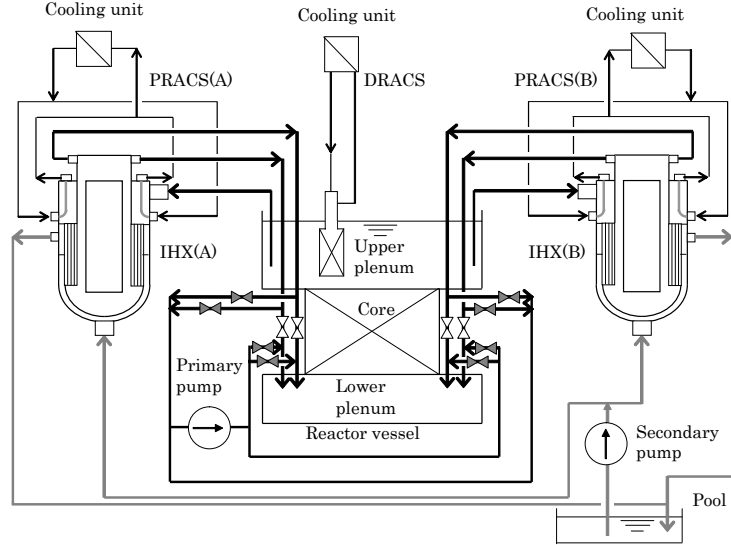


Fig. 1 Schematic of water test equipment.

2.2 Similarity rule

For NC in a closed loop, there are no readily available reference velocity and temperature, so that these variables need to be defined. Here, a single loop with a heater and a cooler is considered to discuss the similarity rule of NC. Two physical balances are established in a steady state of this NC. One is the balance between buoyancy force and pressure loss, and the other is between heat generation and cooling. Two correlation equations describing these balances are solved for the velocity and the temperature rise. These variables are formulated in terms of influential parameters that are the characteristic length, the transferred heat and the properties of water. Dimensionless variables can be defined by using these reference ones. Three-dimensional equations of momentum, energy and continuity for an incompressible fluid flow are rearranged to non-dimensional forms by writing them in terms of the dimensionless variables [8]. These non-dimensional equations contain three dimensionless parameters on the similarity rule which are the Euler number (Eu), the modified Grashof number (Gr) and the modified Boussinesq number (Bo). Gr and Bo are defined as:

$$Gr = \frac{(\beta g \rho^{-1} C^{-1})^{2/3} L^{4/3} Q^{2/3}}{\nu^2} \quad (1)$$

$$Bo = \frac{(\beta g \rho^{-1} C^{-1})^{2/3} L^{4/3} Q^{2/3}}{\kappa^2} \quad (2)$$

Here β , g , ρ , C , L , Q , ν and κ represent, respectively, expansion coefficient, gravity, density, specific heat, length, heat generation rate, kinematic viscosity and thermal diffusivity. Eu is a parameter relevant to the flow resistance and is equivalent to half of the pressure loss coefficient corresponding with the Richardson number (Ri) under NC. Gr is a parameter representing transport phenomena of fluid under NC and $Gr^{1/2}$ is equivalent to the Reynolds number (Re). Bo is a parameter representing transport phenomena of heat under NC and $Bo^{1/2}$ is equivalent to the Peclet number (Pe). All these dimensionless parameters in the water test using a reduced scale model could not match those in JSFR. A value of 1/10 is chosen as a reduced scale of the test model, where Eu is identical to that of JSFR and $Bo^{1/2}$ is set at the same order (about three times larger than that of JSFR). In addition, $Gr^{1/2}$ is reduced to 1/300, but the flow is kept in a turbulent state almost everywhere as in JSFR. On Bo similarity, both a short distance added by the small scale size and a low heating rate roughly offset the effect of low thermal conductivity given by water. The resultant temperature field becomes similar to the actual one and generates an equivalent buoyancy force distribution. In this way, the present test could reproduce an essential trend of NC flow, but it has a limitation in simulation of local flow and heat transfer especially in the core and the heat exchangers. Substituting the typical value of parameter into the reference variable, we obtain conversion equations respectively on representative velocity, temperature rise and time between the simulation model (m) and the actual plant (p):

$$\frac{\Delta u_m}{\Delta u_p} = 0.10 \quad (3)$$

$$\frac{\Delta T_m}{\Delta T_p} = 0.06 \quad (4)$$

$$\frac{\Delta t_m}{\Delta t_p} = \frac{L_m}{\Delta u_m} \cdot \frac{\Delta u_p}{L_p} = 1.04 \quad (5)$$

Here u , T , and t represent, respectively, velocity, temperature and time. The estimates are made for a typical case. These ratios had to be pre-calculated for each test and were used for conversion of the service condition and the transient change between test and actual phenomena.

3. Experimental procedure and result

The simulation tests have been conducted for representative events selected from the Design Basis Events (DBE) in JSFR. Among them, three events are picked up in this paper as typical cases. These are 1) a loss of the external electric power supply, 2) sodium leakage in a secondary loop and 3) sodium leakage of a secondary PRACS. Event 1) causes flows symmetrically between two primary loops. The other two events, 2) and 3), cause flows asymmetrically. Test procedures, an evolution of accident and typical results are presented in this chapter.

3.1 Experiment of the loss of external electric power supply

The reactor is shut down by a signal of blackout in the event 1), and the core decay heat is removed with NC of the primary cooling system and DHRS. Figure 2 depicts handling of the cooling systems in the simulation. The test starts under conditions similar to those of JSFR. The initial conditions of temperature in the primary loop are established through the steady state operation and transient change at the reactor trip. The core power changes rapidly from a steady one to a decay level at the moment of scram. The primary flow is shut off in several tens of seconds after the power change. Since then, NC starts in the primary loop. The delay of primary pump shutdown is set to produce the same effect as that brought by pump flow coast down in JSFR. The experimental results are shown in Figure 3. As described previously, two primary loops behave symmetrically. Forced circulation of the primary loop is blocked off at the scram, but a certain flow is kept by the buoyancy force arising in the steady state. Shut down of the secondary loop causes a rise in temperature of the cold-leg fluid, which produces a decrease in buoyancy force of the primary flow. Thus the primary flow of NC is gradually reduced. At 600 seconds after the scram, DHRS comes into operation, which brings about a buoyancy force in PHX. The primary flow then stops decreasing and begins to increase. As a result, a stable primary flow (corresponding to about 2-3% rated flow of the actual plant) is maintained. The

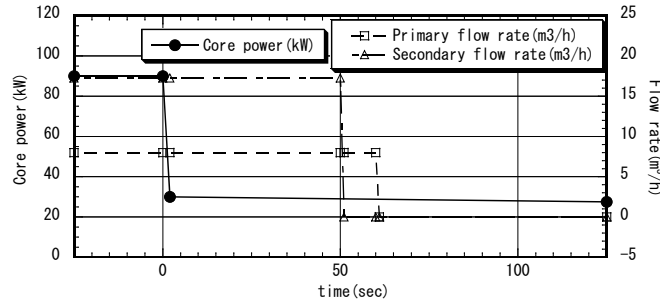


Fig. 2 Test procedures for the loss of the external electric power supply.

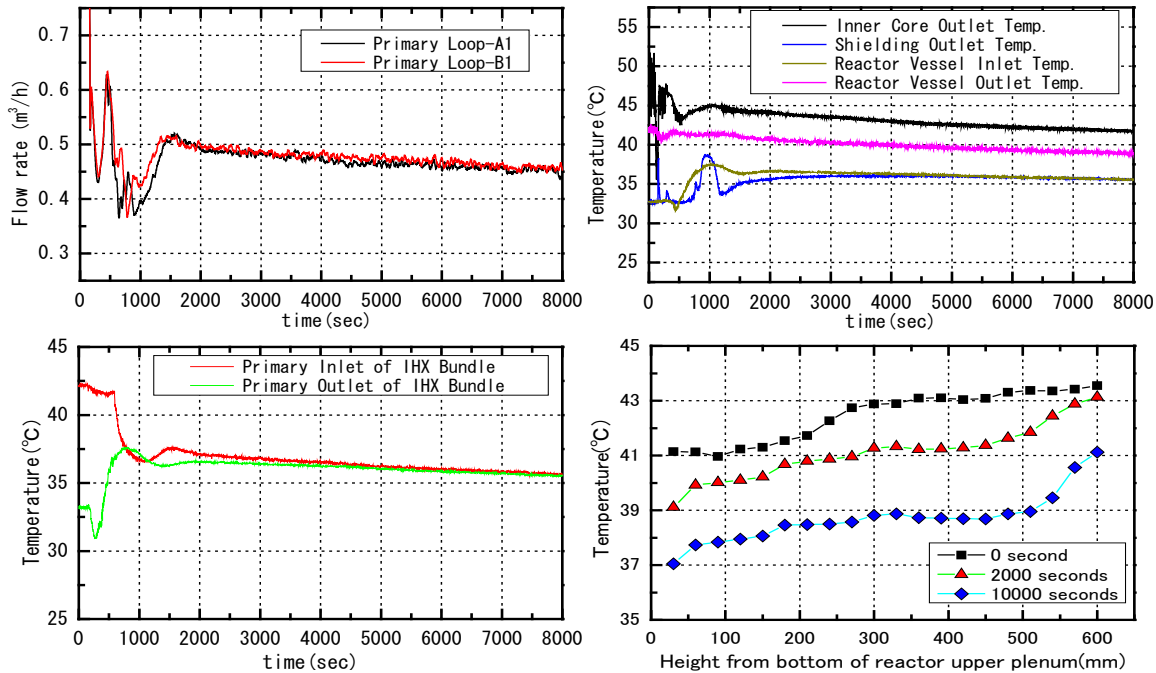


Fig. 3 Test results for the loss of the external electric power supply.

fluid temperature of core outlet rises and reaches a peak just after the scram but does not exceed that in the steady state operation. The cold water coming from the primary side of DHX counterflows through flow paths of the shielding subassemblies at around 1000 seconds. Figure 3 also shows changes in the inlet and outlet temperatures of the primary side of IHX bundle. Shut down of the secondary cooling system produces an increase in temperature of the primary side outlet of IHX bundle. At 600 seconds, a fall in temperature at the primary side inlet of IHX bundle is caused by activation of PRACS. Transitions of the vertical temperature distribution are depicted for the upper plenum of reactor vessel in Figure 3. Thermal stratification occurs in every plenum but not in the primary piping. The temperature gradient across the density interface of thermal stratification is lower than that imposed in the decay heat removal operation using forced circulation. Start up of DRACS triggers a natural convection in the reactor upper plenum, by which the level of density interface goes up to the primary side inlet of DHX.

3.2 Experiment of sodium leakage in a secondary loop

The reactor is shut down by a signal of sodium leak in the secondary loop, and the decay heat is removed with NC of the primary cooling system and DHRS. Typical results are shown in Figure 4. Sodium leakage from the secondary loop-A disables heat removal through IHX (A), which causes a rise in temperature of the cold-leg fluid of the primary loop-A. Transient change of the simulation begins with shutdown of the secondary loop-A which has an equivalent effect of sodium leakage, and a scram follows after several tens of seconds. The primary loop-A loses its buoyancy force at the onset of the transient and the flow is interrupted. Decay heat removal in the initial transient is accomplished through NC of the intact loop-B without causing an excessively high temperature at the core outlet. The flow distribution of core hardly deviates from its normal one even in the early change of accident. After that, activation of DHRS induces NC even for the accident loop-A. The flow of the primary loop-A rapidly increases by gaining a buoyancy force at PHX. As a result, equivalent flows are brought to both primary loops by the buoyancy forces, and NC becomes stable in the later stages of transient.

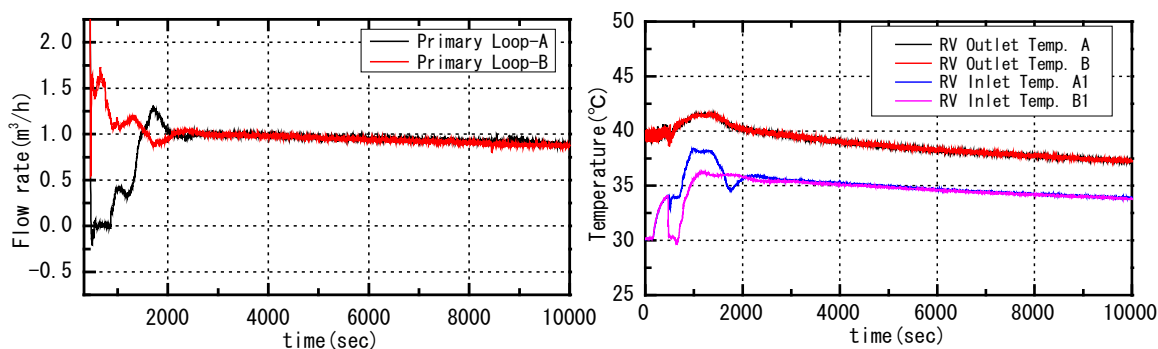


Fig. 4 Test results for sodium leakage in a secondary loop.

3.3 Experiment of sodium leakage in a secondary PRACS

Breakdown of a PRACS caused by sodium leakage makes it difficult for JSFR to continue its rated operation, since the accident condition does not meet the safety requirement on redundancy for DHRS. The reactor is to be shut down in the event 3). Sodium leakage is assumed in the secondary PRACS (A). The simulated event in the present study corresponds to a reactor trip without activation of PRACS (A). Hence, the test procedure is similar to that for event 1). A heat removal performance can be revealed in the test for the case of partial outage in DHRS. Typical results are shown in Figure 5. While NC has occurred once in both primary loops just after the scram, the flow rate of the primary loop-A is gradually reduced to almost zero since PRACS (A) does not work. The decay heat is then removed with NC induced by operable PRACS (B) and DRACS. The primary loop-B solely establishes a stable and sufficient flow of NC. The flow distribution in the core region results in no significant distortion. The core outlet temperature in the transient does not exceed that in the steady state.

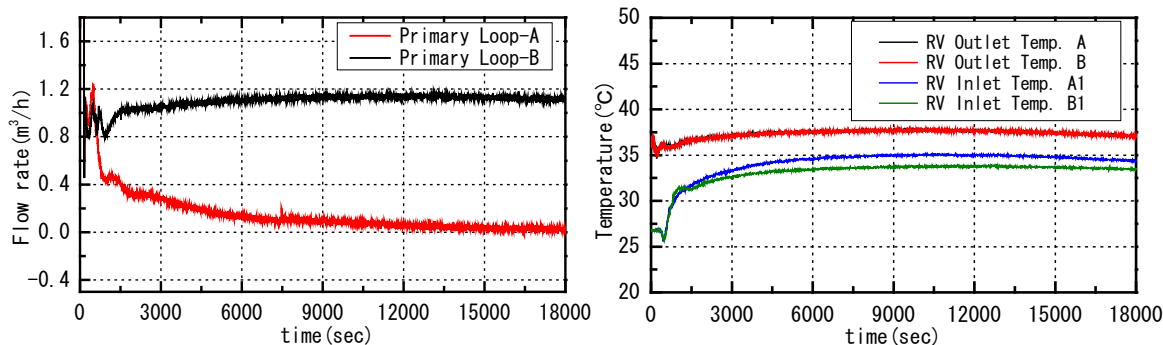


Fig. 5 Test results for sodium leakage in a secondary PRACS.

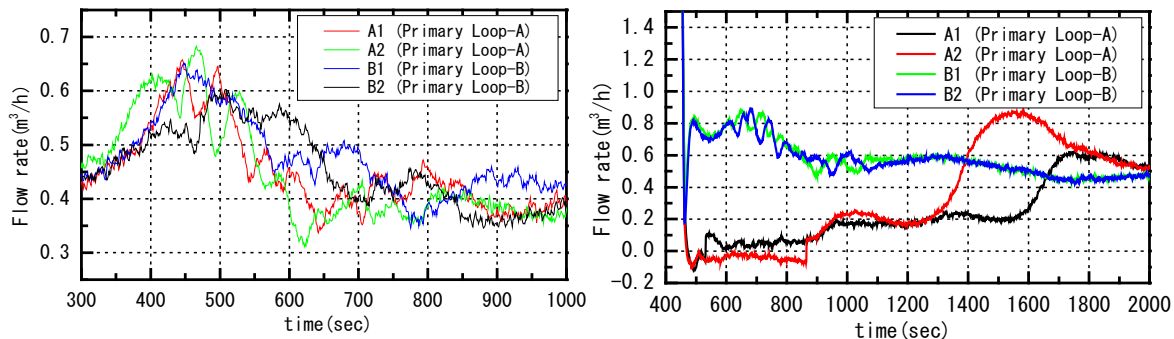
4. Issues and resolutions

Several thermal-hydraulic issues on NC have been identified in the simulation tests on the representative events in JSFR. Among them, three issues are addressed here. These are “flow interactions between parallel cold-leg pipes,” “temperature oscillations in the lower plenum of reactor vessel” and “optimum operation of DHRS.” Measures against problems have been proposed. Additional tests were conducted to confirm the effectiveness of measures. Concise summaries of thermal-hydraulic issues and resolutions are presented in this chapter.

4.1 Flow interactions between parallel cold-leg pipes

Each cold-leg line of the primary loop consists of two flow paths of a parallel pipe. The primary flow goes down the parallel pipe. Out-of-phase oscillations of flow usually occur between the parallel flows in the early stage of NC. Figure 6-(1) shows a typical result in the simulation test of the loss of the external electric power supply. From 450 to 700 seconds the cold-leg temperature is steeply rising and the primary flow rate is rapidly decreasing as shown in Figure 3, where interaction between buoyancy and inertial forces of fluid induces oscillations in the parallel flows. The event corresponds to a hot shock with flow decreasing for the cold-leg piping. In addition, mal-distribution of flows in the parallel pipes occurs in the event of sodium leakage

in a secondary loop. A typical result is shown in Figure 6-(2). Activation of PRACS restores NC eventually for the accident loop-A. When the cold-leg temperature of loop-(A) is rapidly dropping due to start-up of PRACS and the flow rate is steeply increasing, the current in the cold-leg is occasionally distributed to one side of the parallel paths for a short time as shown in Figure 6-(2). The event corresponds to a cold shock with flow increasing for the cold-leg piping. In order to prevent the occurrence of oscillations and mal-distribution in the cold-leg flows, the best plan is to change the piping system from parallel to single. Increase in the pressure drop of the parallel pipes may be also effective to remedy unequal distribution of flows.

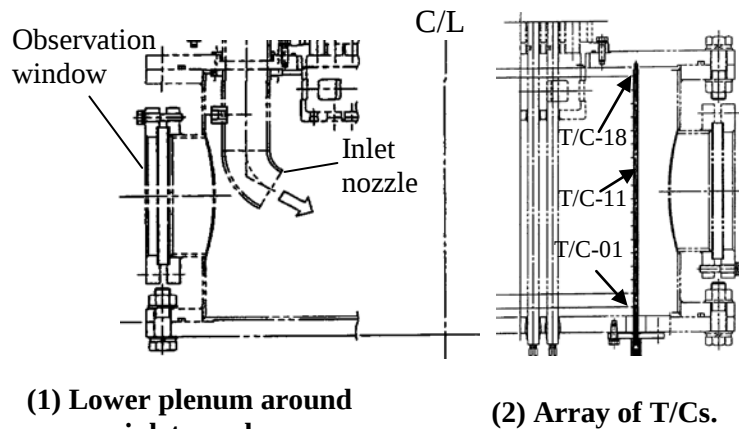


(1) The loss of the external electric power supply. (2) Sodium leakage in a secondary loop.

Fig. 6 Examples of flow interaction between parallel pipes.

4.2 Temperature oscillations in the lower plenum of reactor vessel

Thermal stratification is usually produced in each plenum of the reactor and IHX vessels of JSFR in the decay heat removal operation utilizing NC. The density interface of thermal stratification has a low temperature gradient, but significant temperature oscillations of fluid occur notably in the lower plenum of reactor vessel. Figure 7 depicts a vertical view around an inlet nozzle and an array of 18 thermocouples (T/Cs) in the reactor lower plenum. Figure 8 shows typical temperature changes obtained in the simulation test of the loss of the external electric power supply. Shut down of the secondary loop associated with a reactor trip produces a rise in



(1) Lower plenum around an inlet nozzle.

(2) Array of T/Cs.

Fig. 7 Schematic of lower plenum and measuring tool.

temperature at the primary outlet of IHX. The hot flow coming from IHX is injected obliquely downward in the lower plenum of reactor vessel through four inlet nozzles, by which thermal stratification is created. This density interface moves downward in the plenum, because the hot fluid is accumulated in the upper part during the early stage and the flow washes away the cold zone in the later phase. While the density interface lies above the level of inlet nozzle, a temperature rise with small-amplitude oscillations is observed in measurement of the thermocouples from T/C-11 to T/C-18, as shown in Figure 8. On the other hand, a temperature rise with large-amplitude oscillations occurs when the density interface is passing through the thermocouples placed beneath the inlet nozzle from T/C-1 to T/C-10. The result of temperature observation suggests that large-amplitude oscillations are produced from the inlet flow acting on the density interface. After the density interface falls below the nozzle level, the inflow strikes it. This impingement causes fluctuations of the density interface which pose large-amplitude oscillations of temperature. The most effective means for preventing severe oscillations is to locate the inlet nozzle at the lower end of plenum. Deceleration of the inlet velocity or alteration in the direction of jet flow is also available to suppress fluctuations of the density interface.

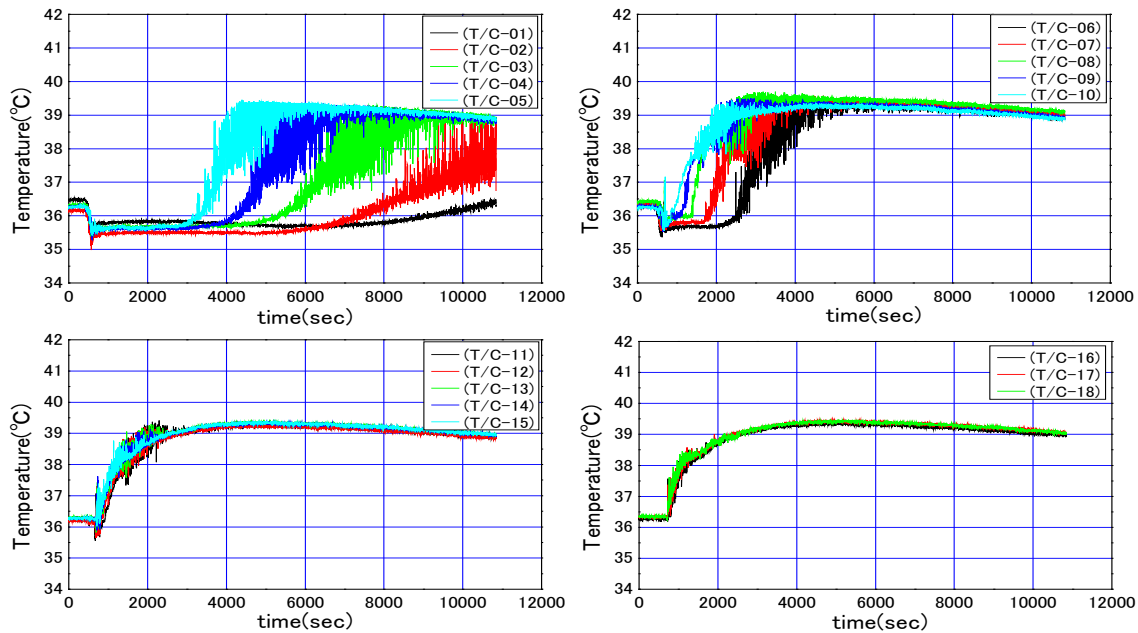


Fig. 8 Comparison of temperature changes in the vertical array of T/Cs.

4.3 Optimum operation of DHRS

Decay heat removal in JSFR is implemented by using DHRS that consists of two sets of PRACS's and a DRACS. Each of PRACS and DRACS has distinctive characteristics on decay heat removal with NC. Two simulation tests have been conducted under an identical condition of cooling capacity for the event of the loss of the external electric power supply in order to reveal the characteristics of PRACS and DRACS. One is for the event cooled by all DHRS and the other by two PRACS's alone. The former case is called as "Two PRACS's and a DRACS" and the latter "Two PRACS's." Figure 9 compares experimental results between two cases for the

transient flow and temperature. The transients in two cases are similar to each other during 2,000 seconds in the early stage, because the initial flow rates of NC are commonly induced by the buoyancy forces arising in the same steady state and from identical activation of two PRACS's at 600 seconds. In this stage, PRACS plays an important role, but DRACS is less effective. In the later stages of transient, DRACS becomes fully effective, where the core outlet temperature gradually falls and the primary flow rate slightly decreases as shown in Figure 9-(1). In particular, DRACS contributes to direct cooling of hot fluid from the core, but the buoyancy force of the primary loop declines as a result of reduction in the hot-leg temperature and the flow rate decreases. Moreover, it is assumed that complex and unpredictable flows could occur between the primary side of DHX, the reactor upper plenum and the core. On the other hand, in the case of "Two PRACS's," both the core outlet temperature and the primary flow rate are very stable and nearly unchanged since 2,000 seconds as shown in Figure 9-(2). Considering optimum operation of DHRS, we have derived conclusions that the decay heat removal utilizing NC is assigned to PRACS and the usage of DRACS is restricted to the event where heat removal by PRACS fails. Activation of DRACS should be left to the plant operator's judgment. These proposals obviously satisfy the safety requirement on decay heat removal and contribute to the soundness of the core internals.

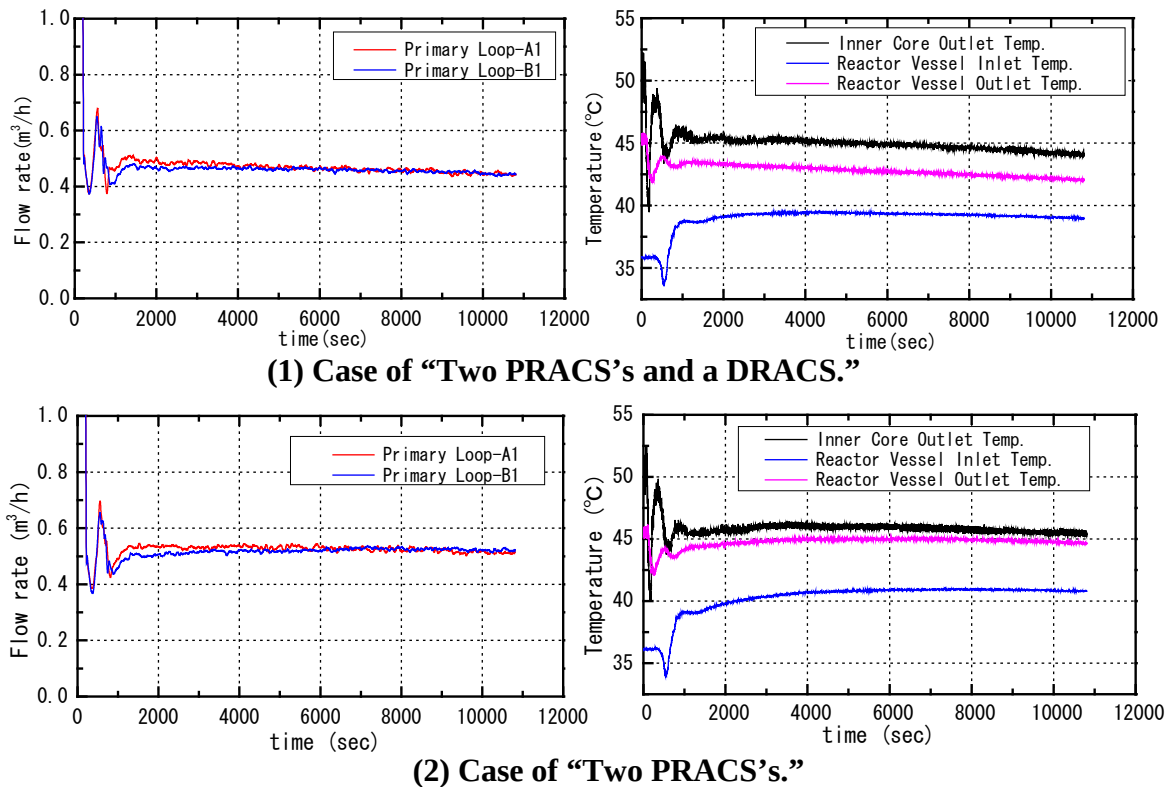


Fig. 9 Comparison of transient changes.

5. Prediction of actual transients

By applying the similarity rule to the test results, we have predicted the actual behavior in JSFR. An actual transient could be estimated by dividing the test result by respective conversion ratios

from (3) to (5). Typical predictions are shown in Figure 10 and 11 for “the loss of the electric power supply” and “sodium leakage in a secondary PRACS,” respectively. These figures show the outlines of the actual transient, but the test result could not provide full information for the thermal-hydraulic phenomena because of geometrical differences posed by use of simplified models and relaxation of the similarity rule. The predicted curves for major transient were found to give close agreement with the results of 1-D numerical analysis obtained in the preliminary study of JSFR and were provided to the discussion of thermal-hydraulic design.

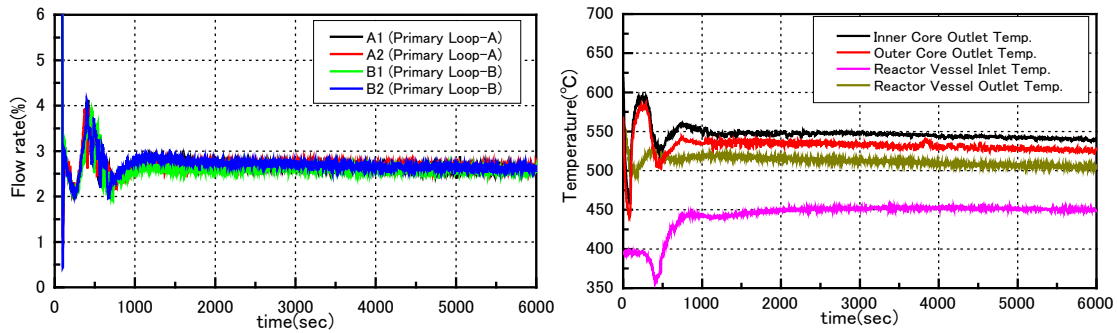


Fig. 10 Predicted actual transient for the loss of the electric power supply.

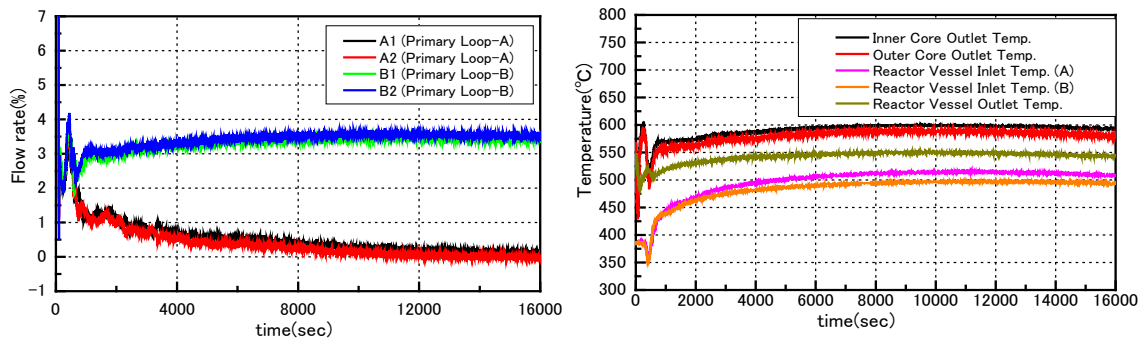


Fig. 11 Predicted actual transient for sodium leakage in a secondary PRACS.

6. Conclusion

A physical simulation has been implemented for the decay heat removal operation utilizing NC of the cooling systems by the water test using a 1/10-scale model of JSFR. In the experiment, the flow of the primary cooling loop was quasi one-dimensional and was kept under a similarity rule. Simulation tests have been conducted for representative events selected from DBE of JSFR. The result was converted to the actual behavior. These predictions well agreed with the 1-D computational results obtained in a previous research and were directly provided to the design study of JSFR. The test results have demonstrated that decay heat removal with NC gives slower changes in temperature of the primary system than that with forced circulation. On the other hand, some issues were identified on thermal-hydraulics. Performance tests also suggested that proper measures could resolve the issues. The present study affords convincing evidence that passive decay heat removal is fully established in various types of accidents for an appropriately designed SFR. Meanwhile, the water test using a reduced scale model was very effective in the study of NC. Improvements on the performance will be required to utilize NC of the cooling

systems in decay heat removal for the conventional design of SFR. The decay heat removal operation after the scram must be highly reliable and its safety feature needs to be assured by prior evaluations. This will firstly require the flow behavior fully predictable even in NC. It is advisable to simplify and stabilize the primary flow pattern so as to allow precise predictions. In general, pursuit of further increase in the primary flow rate under NC is valuable not only to enhance cooling performance of DHRS, but also to improve the adaptability of water test under similarity rule. When viewed from this standpoint, NC of the cooling systems suits well to decay heat removal of JSFR.

7. Acknowledgement

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