Evaluation Method for Core Thermohydraulics during Natural Circulation in Fast Reactors* (Numerical Predictions of Inter-Wrapper Flow)

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Decay heat removal using natural circulation is one of significant functions for a reactor. As the decay heat removal system, a direct reactor auxiliary cooling system has been selected in current designs of fast reactors. In this system, cold sodium is provided in an upper plenum of reactor vessel and it covers the reactor core outlet. The cold sodium can penetrate into the gap region between the subassemblies. This gap flow is referred as inter-wrapper flow (IWF). A numerical estimation method for such phenomena was developed, which modeled each subassembly as a rectangular duct with gap region and also the upper plenum. This numerical simulation method was verified by a sodium test and also a water test. We applied this method to the natural circulation in a 600 MWe class fast reactor. The temperature in the core strongly depended on IWF, flow redistribution in the core, and inter-subassembly heat transfer.

Key Words: Numerical Analysis, Natural Convection, Fast Breeder Reactor, Heat Transfer, Passive Safety

1. Introduction

Decay heat removal using natural circulation is one of the important functions for the safety of fast reactors. Flow rate through the reactor core depends on temperature distribution in the cooling systems under natural circulation condition. The flow rate and temperature distribution couples with each other and thermohydraulic phenomena have complex relations of cause and effect. As a decay heat removal system, DRACS (Direct reactor auxiliary cooling system) has been selected in current designs of fast reactors, for example, Japanese Demonstration Fast Breeder Reactor (DFBR)⁽¹⁾. DRACS has dumped heat exchangers (DHXs) in an upper plenum of reactor vessel. Figure 1 shows schematic of the DHX in the reactor vessel.

In this design, cold sodium provided by the DHXs covers the reactor core outlet and also makes thermal stratification in the upper plenum. The cold sodium in the upper plenum can penetrate into the gap region between the subassemblies and cause natural convection. This gap flow is referred as inter-wrapper flow (IWF).

As for IWF, water experiments^{(2),(3)} were carried out for the European Fast Reactor which also featured DRACS. Here the whole core was simulated by 253 circular tubes; 19-pin electric heater was installed in each tube to simulate a fuel subassembly. IWF was observed in the experiments and the cooling effects on the simulated core were evaluated. It was found that IWF influenced the highest temperature in the core during natural circulation. Evaluation methods using numerical simulations were studied for IWF under the natural circulation conditions of Japanese DFBR⁽⁴⁾. Decrease of the peak temperature in the core due to IWF was predicted during a scram transient in the total black out event.

Here, a numerical estimation method for such natural circulation phenomena in a reactor core is proposed, which uses the multi-dimensional thermohy-

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Fig. 1 Schematic of DRACS

draulic analysis code, AQUA⁽⁵⁾. The prediction method considers each subassembly as a rectangular duct and models gap region between the subassemblies and the upper plenum in a reactor vessel together. This modeling enables us to calculate IWF, thermal interaction between IWF and flow in the subassemblies, inter-subassembly heat transfer due to radial temperature distribution across the core, and flow redistribution in the core.

This numerical simulation method was verified based on experimental data of a sodium test using 7-subassembly core model⁽⁶⁾ and also a water test which simulated IWF using an 1/12 sector model of a reactor core. Thermal coupling between IWF and temperature distribution in the subassemblies was shown in the 7-subassembly sodium experiment. The calculation method was examined for the temperature decrease in the subassembly due to IWF. In the water experiment, velocity profile of IWF in the core was obtained by using flow visualization and particle image velocimetry. The calculated velocity profile was compared with the experimental data.

We applied the estimation method to the natural circulation in a 600 MWe class fast reactor. Parameters of the calculations are geometry of subassembly pad, which keeps the pitch of subassemblies, and a flow guide for IWF, which leads the cold fluid into the core. Thermohydraulic phenomena in the reactor core was predicted. Key functions required for the simulation methods are discussed on the points to predict the highest temperature in the core.

2. Simulation Method

Thermohydraulic phenomena considered in a reactor vessel during natural circulation are as follows;

1) thermal stratification in the upper plenum; cold sodium provided by the DHXs stratifies in lower part of the upper plenum,

- 2) IWF; gap flow between the subassemblies,
- 3) penetration flow into the subassemblies from the stratified upper plenum,
- 4) inter-subassembly heat transfer; radial heat transfer across the core,
- 5) flow redistribution among the subassemblies due to buoyancy force.

During natural circulation, flow rate through the core is so small that the highest temperature in a subassembly is affected by the heat transportation through the wrapper tube wall due to IWF and inter-subassembly heat transfer. The penetration flow occurs when negative buoyancy force is larger than inertia force of upward flow in a subassembly⁽⁷⁾. The penetration flow also influences the temperature in the subassemblies. The flow redistribution flattens the radial temperature distribution across the core.

The three-dimensional thermohydraulic analysis code, AQUA, is used to predict above mentioned phenomena in the core and upper plenum. AQUA is an in-house developed, finite difference code. The thermohydraulics in the inter-wrapper gap regions is connected to the upper plenum, which is a source of cold sodium, and is additionally coupled with the flow in the subassemblies. Transverse temperature gradient in the subassembly becomes pronounced due to heat removal through the wrapper tube wall. These temperature gradients thus influence the heat transfer between the subassemblies and the gap regions. Therefore, three regions; 1) the inter-wrapper gap, 2) the upper plenum, and 3) the subassemblies, have been analyzed together.

The horizontal mesh scheme for the core part is shown in Fig. 2. Cartesian coordinate system is used to model the above mentioned three regions together. The calculation domain is a kind of sector model for 1/12 of the core. The hexagonal shape of the subassembly is converted to a rectangular shape while maintaining the perimeter of the wrapper tube facing the gap region, which corresponds to the heat transfer area between the subassembly and the gap region. The IWF in the gap region removes heat from the subassemblies. In a horizontal plane of the gap region, one part has hot upward flow and the other part has cold downward flow. Thus relative position between an element of the gap region and a side of hexagonal wrapper tube is maintained in the mesh scheme to simulate the heat flow between the IWF and the subassemblies.

In the subassembly region, the meshes facing the wrapper tube are modeled as a wall subchannel and the inner meshes are modeled as an inner subchannel. Such a manner of classifying subchannels is known to be useful for simulating flow and the temperature field

in a subassembly, using a staggered half-pin mesh arrangement^{(8),(9)}. The total flow area in each mesh type is set equal to that in the subassembly by using a porous body model. The fraction of heater power in each mesh is also set in the same manner. Flow resistance coefficients at the wall and inner subchannel regions are set based on the correlations developed for subchannel analyses by Cheng⁽¹⁰⁾. The width and flow area of the gap region are identical to the real geometry.

Figure 3 shows vertical cross section of the core and the upper plenum. At the height of the core outlet, the subassembly regions and the gap region are connected to the upper plenum region. All subassemblies including control rod channels are connected to a high pressure plenum at the bottom. A low pressure plenum, to which the radial reflector and control rod channels are connected in the ordinary way, is neglected to be a simple model. There is no flow connection between the gap region and the high pressure plenum. The DHX is also modeled in the upper plenum region. Boundary conditions are as follows: the flow rate, the temperature into the high pressure plenum, the decay heat in each subassembly, and the flow rate and inlet

Fig. 2 Mesh scheme of subassemblies and core matrix

Fig. 3 Mesh scheme of core and upper plenum

temperature in the secondary loop of the DHX. A first order upwind scheme is used for convection terms differencing and a laminar flow model is applied because of simple mesh modeling in the core for the complex geometry in reality (e.g., wire wrapped fuel pin bundle). The above mentioned calculation method is referred as a brick subassembly model.

3. Verification Based on a Sodium Experiment

Thermal interaction between IWF and flow in the subassemblies has been investigated in the sodium experiments using PLANDTL-DHX facility^{(6),(9)}. The brick subassembly model was applied to the sodium experiment⁽⁶⁾. Here, some comparisons between experiments and calculations are shown to see the validity of the modeling for the thermal interaction.

Axial temperature distributions along the center subchannel in the center subassembly are compared between experiments and calculations in Fig. 4. Flow velocity in the pin bundle was ~ 5 cm/s and it corresponds to $\sim 1\%$ of full power condition in a reactor. All of core power was removed by the DHX. Axial temperature distributions along the colder gap region beside the center subassembly are also shown in Fig. 4. The upper half of the gap region showed lower temperature than the middle height in the experiment. This is due to the cold downward flow (IWF) from the upper plenum into the gap region. The center subchannel in the pin bundle also showed temperature decrease in the upper non heated region. This is due to heat removal by IWF.

Figure 5 shows non-dimensional peak temperature in the center subassembly. The horizontal axis is flow velocity ratio in the center subassembly as

Fig. 4 Axial temperature distributions in PLANDTL-DHX core

compared with the full power condition in a reactor. The flow rate was kept constant by a pump. The non-dimensional peak temperature, T^* , was obtained by the following equations:

$$T^* = \frac{T_{\text{peak}} - T_{\text{in}}}{\Delta T}, \quad (1)$$

$$\Delta T = \frac{Q}{WC_p}. \quad (2)$$

Here, T_{peak} is the highest temperature in the core, T_{in} is the inlet temperature into the core, Q is heat input in a subassembly, W is a mass flow rate, and C_p is specific heat of sodium.

The ΔT corresponds to the average temperature increase under adiabatic condition at the wrapper tube wall. Thus, T^* means the heat removal ratio of IWF (heat removal through the wrapper tube) to the heater power in the center subassembly. When the flow velocity ratio was less than 1.5%, the non-dimensional temperature became smaller than unity. The non-dimensional temperature was 0.5 when the velocity ratio was 0.5%. It means that IWF removes 50% of heater power in the center subassembly. The calculated data was in good agreement with the experimental results within the measurement error, which was mainly resulted from flow rate measurement in the subassembly.

The experimental analyses revealed that the brick-subassembly model calculation could simulate temperature decrease in the gap region between the subassemblies and also the decrease of the peak temperature in the subassembly. The heat transportation via IWF was well calculated by this model.

4. Verification of Flow Field Based on a Water Experiment

4.1 Test apparatus

A water experiment, TRIF (Test rig of inter-

Fig. 5 Non-dimensional peak temperatures in PLANDTL-DHX core

wrapper flow), has been carried out to investigate flow field of IWF in the core. The PLANDTL-DHX sodium experiment revealed the cooling characteristics of IWF in the seven-subassembly partial core model. However, the reactor core has several regions, e.g., inner core and radial reflector. Flow field of IWF in the core will depend on such core matrix. In a simple assumption, it is predicted that cold sodium in the bottom of the upper plenum flows downward in the radial reflector region and flows up in the inner core region. The TRIF experiment was planned to see the flow field of IWF in the whole core.

Figure 6 shows a schematic of the reactor vessel model of TRIF. A 30° sector of reactor core and upper plenum is modeled. Flow field of IWF is of interest in this experiment. Flow visualization and particle image velocimetry, PIV⁽¹¹⁾ were used to obtain the flow field in a complex geometry. For the visualization, no heater is installed inside of the subassemblies and flow through the subassembly is not simulated. The wrapper tube of subassembly is made of transparent resin. Glass heater panels, those are coated by tin-oxide and electrically heated, are set at the wrapper tube wall to simulate heating by the core fuel subassemblies. This subassembly model enables to see the flow in the gap region between the subassemblies from outside of the reactor core model. Non-heated subassemblies are set in outer region of the core to simulate radial reflectors. Button type pads are set at the wrapper tube wall; the pad is used to keep the pitch of subassembly arrangement. This geometry of the pad permits vertical flow through the pad position, i.e., fluid exchange between the gap region in the core and the upper plenum.

Cooling panels are dumped in the upper plenum to

Fig. 6 Schematic of TRIF experimental apparatus

simulate the DHX. Cold water provided by the cooling panel flows into the gap region between the heated subassemblies and IWF occurs.

4.2 Flow field of IWF

Schematic of visualized flow pattern is shown in Fig. 6. Water in the upper plenum was cooled by the DHX and stratified in the bottom. The cold water penetrated into the inter-wrapper gap in outer non heated region in the core. At the bottom of core, radial flow toward the center of the core was observed. In the inner region of core, upward flow was observed.

The velocity field of IWF in the gap region, which was measured by PIV, is shown in Fig. 7. Detail of the measurement technique was reported in Ref.(11). The velocity vectors are time averaged value of 60 data during 2 seconds. The velocity vectors were measured in the gaps of P1 to P4 in the heated region of the core (see Fig. 6). Radial component of velocity was small in these gap regions. The upward flow velocity of IWF increased as it flowed up. The upward flow velocity in the inner gap was larger than that in the outer gap. Flow pattern and magnitude of flow velocity of IWF were grasped in the geometry of button type pad.

4.3 Calculation results of TRIF

The calculated temperature and velocity field in a vertical cross section of $J=3$ is shown in Fig. 8. The horizontal mesh arrangement of the core is also

shown in Fig. 8. This cross section ($J=3$) includes the gaps of P1 to P4 where flow field was measured. The calculated flow pattern was in good agreement with the visualized pattern, however the radial flow component in the heated subassembly region of the core was larger than measured velocity vectors (see Fig. 7). Figure 9 shows a comparison of vertical velocity components between experiment and calculation in the gap of P1. The measured data were scattered depending the transverse position in the gap. The calculated distribution of flow velocity traced the trend of experimental data.

Temperature distributions are also compared between the experiment and calculations. Figure 10 shows radial temperature distributions in vertical cross sections of $J=7$. Parametric calculations were carried out for the vertical flow resistance coefficient at the pad. In the nominal case, the resistance coefficient was obtained according to the change of flow area through the pad position. The resistance coefficient was varied in a range, factor of 0.5 to 2.0 as compared with the nominal case. The radial temperature distribution at the top of heated length was influenced by the flow resistance through the pad position. The higher resistance case showed rather better agreement with the experimental results than the nominal case. The vertical velocity distributions in three calculational cases are also compared in Fig. 10. The region of upward flow expanded in outer

Fig. 7 Velocity field of IWF in TRIF

Fig. 8 Calculated temperature and velocity fields in TRIF

space in the heated subassembly region when the flow resistance increased. The higher flow resistance will reduce convective flow rate in the gap region. Thus requirement for the flow area of cold downward flow reduces. This change of the boundary between the hot upward flow and cold downward flow results in good agreement with the measured temperature distribution in the higher flow resistance case.

The calculations showed fairly good agreement of temperature distribution with the measured data. And it is also shown that the flow and temperature fields of IWF is sensitive to the flow resistance through the pad position.

5. Prediction Analysis for the Reactor Core

5.1 Boundary conditions

We carried out prediction analyses of thermal hydraulic phenomena in a reactor core during natural

circulation condition. The core matrix and mesh arrangement are shown in Figs. 2 and 3, respectively. The calculational conditions are listed in Table 1. The reactor core has inner and outer cores, radial blanket, and radial reflector. The flow rate into the lower plenum was a boundary condition and set at 1% of full power condition, which corresponds to a level of natural circulation flow rate.

Geometry of the subassembly pad is a calculational parameter. The reference case has a band type pad, which does not allow vertical flow through the pad position. The other is a slit type, which allows fluid exchange between the inter-wrapper gap region and the upper plenum⁽¹²⁾. The slit type pad is illustrated in Fig. 11. Geometry of core former, which is a plate and covers the region between the outermost subassemblies and the core barrel, is also changed as a calculational parameter. In a reference case, the core former closes the flow path from the upper plenum together with the band type pad. Flow guide through the core former is added to allow fluid exchange between the gap region in the core and the upper plenum as shown in Fig. 11. Several ducts penetrate through the core former, which allow the downward flow from the upper plenum and also prevent from mixing with hot fluid in higher part of the gap region. This duct will help to use the cold fluid

Fig. 9 Comparisons of vertical velocity component of IWF in TRIF experiment

Table 1 Calculational conditions

Item	Data
Time control	Steady state calculation
Decay heat level	1.7% of full power condition
Loop flow rate	1.0 % of full power condition
Core inlet temperature	450 °C

Fig. 10 Radial distributions of temperature and vertical velocity component along $J=7$

directly to remove the decay heat. Several flow holes are also set in the core former to release the hot fluid from the gap region to the upper plenum. Total of three cases, closed pad, slit pad, and flow guide, were calculated. The calculated cases are summarized in Table 2.

5.2 Calculational results for reactor core

Figure 12 shows the calculated temperature and velocity fields in a vertical cross section through the gap region ($J=3$, see Fig. 3) in the closed pad case and the slit pad case. Flow velocity of IWF in the slit pad case was larger than that in the closed pad case and the temperature along the inner core region in the slit pad case was lower than that in the closed pad case. Cold downward flow from the upper plenum was seen in the radial reflector region in the slit pad case. It decreased the temperature in the gap region.

Radial temperature distributions across the subassemblies ($J=1$, see Fig. 3) at the top of active core fuel are compared between three cases in Fig. 13. The slit pad case showed the lowest temperature in the core region among the three cases. The temperature gradient in each subassembly was larger in the slit pad case than the closed pad case, especially in the

inner core region. It means that the decay heat is removed by IWF through the wrapper tube in the slit pad case. The outer core subassembly had steep distribution in all cases. This is due to the inter-subassembly heat transfer to the adjacent blanket subassembly, which is much colder than the outer core subassembly. This outer core subassembly showed the highest temperature in the core in each case. One of the reasons is flow redistribution in the core, which is shown in Fig. 14.

Vertical axis of Fig. 14 is relative flow rate in the subassembly normalized by the loop flow rate. The flow rate in the inner core region increased as compared with that under full power condition. The excess flow rate was provided by reverse flow in radial blanket region and also radial reflector region. However, the flow rate in the outer core region was comparable with that under full power condition. This relatively low flow rate resulted in the highest temperature in the outer core subassembly in Fig. 13. Partial reverse flow (recirculation flow) was seen in that

Table 2 Calculational cases

Case	Geometry
Closed pad (Reference)	Pad: band type Core former: closed
Slit pad	Pad: slit type Core former: closed
Flow guide	Pad: band type Core former: open

Fig. 11 Geometry of pad and flow guide

Fig. 12 Velocity and temperature field in gap region ($J=3$) in a reactor core

Fig. 13 Radial temperature distributions across the subassemblies at top of active core

Fig. 14 Flow redistribution under natural circulation condition

subassembly due to the low flow rate. Further, cold radial flow in the gap region, above the pad position and coming from the upper plenum, could cool the upper part of the outer core subassembly, which is located in the outermost layer in the core fuel subassemblies. These cooling effects via the cold fluid from the DHX reduced the temperature in the upper non heated part of the outer core subassembly. Such temperature decrease also reduced the natural circulation head and the flow rate in that subassembly. Thus the outer core subassemblies could not gain the additional flow rate.

6. Discussion

6.1 Heat removal by IWF

Figure 15 shows non-dimensional peak temperature, T^* , in each subassembly along $J=1$ array. Definition of the non-dimensional temperature is the same as Eqs.(1) and (2). T^* means relative peak temperature as compared with the average temperature increase based on the power and flow rate in each subassembly.

In the closed pad case, T^* in the outer core subassembly (OC7) and the outermost subassembly (IC5) in the inner core showed 0.9 to 0.8. These subassemblies had higher temperature in the core (see Fig.13) and were cooled by the inter-subassembly heat transfer and IWF. In the inner core subassemblies, T^* showed around 1.1. This is resulted from heat transportation from the outer core subassemblies to the inner core subassemblies via IWF (see Fig. 12). In the slit pad case, the T^* in the inner core subassemblies showed 0.8. It means that IWF cools the inner core subassemblies by 20% of the decay heat. In the outer core subassembly, it reached nearly 40%. The non-dimensional temperature in the flow guide case

Fig. 15 Radial distributions of non-dimensional peak temperature in the subassemblies along $J=1$

was higher than that in the slit pad case by 5 to 10%. However, the flow guide functioned to utilize the IWF to reduce the core temperatures.

6.2 Key phenomena during natural circulation

It was revealed by the prediction analyses that IWF had significant influences on the core thermohydraulics. The highest temperature in the core was decreased by IWF in case of the slit type pad. The radial temperature profile in the core was affected by the flow redistribution, caused by buoyancy force, in the core. In the outer subassemblies, the buoyancy force was reduced by the penetration flow (partial reverse flow from the upper plenum) in the upper axial reflector and IWF via the temperature decrease in the upper non heated part in the subassemblies. The inter-subassembly heat transfer also reduced the temperature in the outer core subassembly due to the large temperature difference between the outer core subassemblies and the radial blanket subassemblies.

These phenomena were coupled with each other. Thus, a precise prediction method should have capabilities to consider all phenomena together. Especially following points are to be modeled ;

- 1) IWF,
- 2) inter-subassembly heat transfer,
- 3) flow redistribution in the core.

Transverse temperature distribution in the subassembly is also significant to simulate the heat removal through the wrapper tube by IWF and inter-subassembly heat transfer.

7. Conclusions

A multi-dimensional calculation using brick-subassembly model based on the finite difference method was proposed to simulate core thermohydraulics in a fast reactor during natural circulation.

The calculational method models the subassemblies, the gap region between the subassemblies, and the upper plenum together to simulate the gap flow between the subassemblies (inter-wrapper flow, IWF). The subassembly is modeled in a rectangular mesh geometry which considers the structure of inner and wall subchannels. It can simulate temperature decrease in wall subchannel region when IWF cools the subassembly through the wrapper tube.

The calculational method was verified through the water and sodium experiments on the points of the flow field of IWF and the temperature decrease in the subassemblies, respectively. The parametric calculations showed that the axial flow resistance coefficient in the gap region through the pad position should be set carefully.

Prediction analyses using this method were performed for a 600 MWe class fast reactor core which features the direct reactor auxiliary cooling system (DRACS). Pad geometry and flow guide through the core former had significant influence on IWF and consequently on the highest temperature in the core. Slit in the pad enhanced IWF and reduced the highest temperature in the core significantly as compared with a closed pad geometry.

It is pointed out that the prediction methods of core thermohydraulics during natural circulation should model IWF, inter-subassembly heat transfer and flow redistribution in the core.

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