The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14
Toronto, Ontario, Canada, September 25-30, 2011

NURETH14-170

DEVELOPMENT OF CORE HOT SPOT EVALUATION METHOD FOR NATURAL CIRCULATION DECAY HEAT REMOVAL IN SODIUM COOLED FAST REACTOR

N. Doda[1], H. Ohshima[1], H. Kamide[1], O. Watanabe[2] and Y. Ohkubo[2]
[1] Japan Atomic Energy Agency, Oarai, Ibaraki, Japan
[2] Mitsubishi FBR Systems, Inc., Shibuya, Tokyo, Japan

Abstract

Toward the commercialization of fast reactors, a design study of Japan Sodium-cooled Fast Reactor (JSFR) is being performed, in which fully natural circulation system is adopted as the decay heat removal system. A new evaluation method of core hot spot which can be applied to natural circulation decay heat removal has been developed. The new method consists of three-step thermal hydraulics analyses in order to consider the effects of physical phenomena particular to natural circulation, such as inter-fuel-assembly heat transfer and flow redistribution in the core due to buoyancy force. From the viewpoint of calculation cost reduction, we have also developed a simplified model substituting for the third step analysis (subchannel analysis). The new method was applied to the evaluations of a loss-of-external-power event and of a sodium leakage accident in a secondary loop of a large scale reactor.

1. Introduction

Toward the commercialization of fast reactors, a design study of Japan Sodium-cooled Fast Reactor (JSFR) is performed [1]. In this design study, fully natural circulation system is adopted as the decay heat removal system to satisfy the requirements of economical competitiveness and higher reliability. In order to adopt the system, we must evaluate the maximum cladding temperature, i.e. core hot spot, under the natural circulation conditions.

When the core flow rate is controlled by the primary pumps or pony motors, the flow rate distribution in the core is determined so as to balance all the fuel assembly flow resistances. Under such forced circulation conditions, the core hot spot can be evaluated by the conventional method using a plant dynamics code in which the core is modeled as a few representative flow channels, including a hot pin channel which has the worst thermal-hydraulic condition of all the fuel pins to account for various uncertainties. However, during the natural circulation decay heat removal operations, the core flow rate is fundamentally not controlled and is driven by the buoyancy force. Flow redistribution in the core occurs due to the buoyancy force balance and the radial heat transfer among the fuel assemblies becomes nonnegligible comparing with the axial heat transport because the core flow rate is small as 3 to 5 % of the rated power operation condition. These effects cause flattening of temperature distribution in both core and fuel assemblies. Appling the conventional method to the core hot spot evaluation under the natural circulation decay heat removal operations, the flattening of transverse temperature profiles resulting from buoyancy-induced flow redistribution and other inter- and intra-assembly phenomena are not considered, thus the evaluation results might be excessively conservative.

We have developed a new evaluation method in order to consider the effects of physical phenomena particular to natural circulation. In addition, from the viewpoint of calculation cost reduction we have also developed a simplified model substituting for the subchannel analysis which is a part of the new method to simulate the flow and temperature distributions in a fuel assembly and evaluates the highest fuel cladding temperature. In this paper, the theoretical basis of the new method and the simplified model are described. Description of the treatment of uncertainties is also included. The proposed method was applied to a loss-of-external-power event and a sodium leakage accident in a secondary loop of a large scale reactor. The applicability of the proposed method to the core hot spot evaluation for natural circulation decay heat removal is discussed.

2. Natural circulation decay heat removal in JSFR

In the JSFR design, there is a natural circulation decay heat removal system (NC-DHRS) to remove the decay heat as a backup system when the normal heat sink fails. This system consists of one direct reactor auxiliary cooling system (DRACS) and two primary reactor auxiliary cooling systems (PRACS) as shown in Figure 1. The system works under the passive natural circulation conditions, because it only requires DC-power supply from battery back-up systems to operate the air cooler dampers in DRACS and PRACS. For this system, fully natural circulation decay heat removal operation is performed in a wide range of events including manual trip, loss-of-external-power, and leakage of sodium coolant from secondary loop, while it is a backup option in the event of total blackout in the Monju design which has pony motors in the primary and secondary loops.

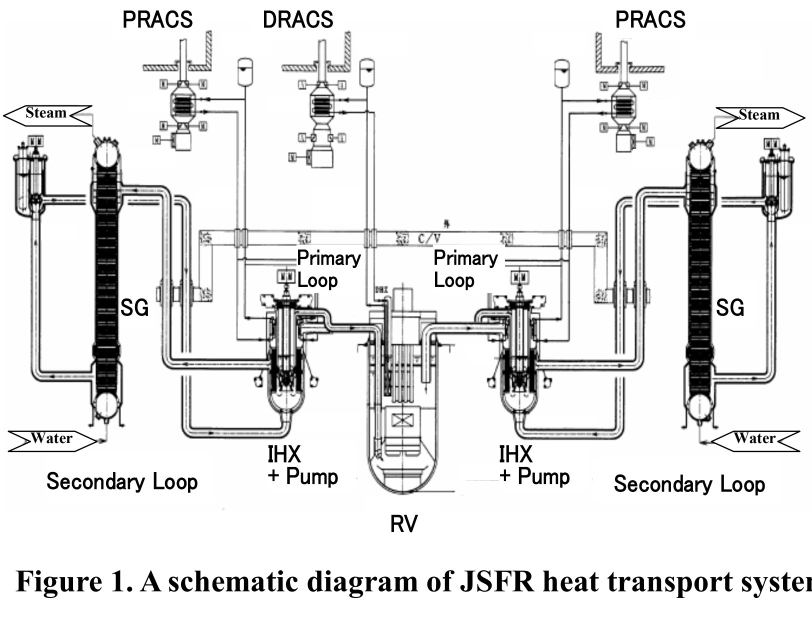


Figure 1. A schematic diagram of JSFR heat transport systems

3. Evaluation method of core hot spot under natural circulation conditions

Under the natural circulation decay heat removal operations, the core flow rate is fundamentally not controlled and is driven by the buoyancy force. When the flow rate is small, the flow resistance is not high enough to be a dominant force to distribute the flow rate in the core and the flow redistributions occur so as to give more flow rates to the hotter fuel assembly and the hotter subchannel in a fuel assembly due to the buoyancy force. The flow redistribution also affects the temperature distribution, thus the changes of temperature and flow rate distributions occur dynamically. The radial heat transfer among the fuel assemblies becomes relatively important. Since the adjacent fuel assemblies influence each other, the thermal-hydraulics in the fuel assemblies can not be treated individually. In a fuel assembly, the flow redistribution among subchannels also occurs due to buoyancy force, thus the subchannels are related to each other and should be treated in the entire region of a fuel assembly. Uncertainties should be dealt with in different way from under the forced convection conditions. Under the natural circulation conditions, uncertainty does not always affect in one way and we can not previously assume it conservatively. In addition, we should add the consideration of spatial distribution of uncertainty, because the fuel assemblies and subchannels are thermal-hydraulically connected to each other. For core hot spot evaluation, the hottest fuel assembly and the hottest fuel pin should be evaluated with all the others simultaneously accounting for uncertainties.

3.1 Three-step analysis for core hot spot evaluation under natural circulation conditions

Figure 2 shows the new evaluation method, which consists of a plant dynamics analysis and a subchannel analysis of fuel rod bundle to consider the flow redistributions in the core and in the fuel assembly due to buoyancy force and the radial heat transfer among the fuel assemblies. To account for uncertainties, the uncertainties were categorized into three groups, which affect the whole core, in-core local, and intra-assembly quantities, respectively. To find the hottest fuel assembly, plant dynamics analyses are curried out twice, imposing the whole core uncertainties and the in-core local uncertainties to the boundary conditions in each step analysis. A subchannel analysis is curried out once because the hottest fuel pin is located at a local position in the fuel assembly. The three-step thermal hydraulics analyses are carried out in the following order:

Step 1. A plant dynamics analysis, in which the core is modeled with all channels of fuel assembly to include the inter-fuel-assembly heat transfer and the flow redistribution in the core driven by buoyancy force, is carried out to identify a hottest channel candidate in the core. In this analysis, the uncertainty contributors in the whole core category shown in Table 1 are considered in the boundary conditions.

Step 2. A plant dynamics analysis is carried out again to obtain the boundary conditions of the subchannel analysis for the hottest assembly. The boundary condition of the second plant dynamics analysis includes the uncertainty contributors in the in-core local category as well as in the whole core category to make the hottest channel candidate the hottest channel in the core. The bulk maximum temperature in the hottest channel is obtained in this analysis with accounting for the inter-fuel-assembly heat transfer and the flow redistribution in the core driven by buoyancy force.

Step 3. A subchannel analysis of fuel rod bundle is carried out to calculate the coolant flow and temperature fields and the cladding tube temperature distribution in the hottest assembly. The core inlet coolant temperature, the flow rate and the heat fluxes at the wrapper tube walls of the hottest channel, which are calculated by the second step of the plant dynamics analysis for the hottest channel in the core, are imposed into the boundary conditions. The boundary conditions also include the uncertainty contributors in the intra-assembly category as well as in the other categories. The core hot spot is consequently obtained.

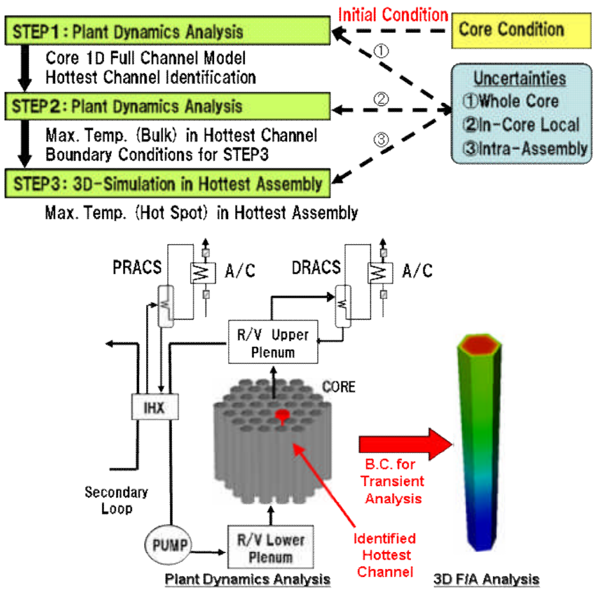


Figure 2. Three-step analysis method

3.2 Uncertainty contributors under natural circulation conditions

We identified the important uncertainty contributors under natural circulation conditions by referring the hot spot factors under the forced convection conditions and the previous studies on the natural circulation decay heat removal in a sodium-cooled fast reactor.

3.2.1 Categorization of uncertainty contributors

Table 1 shows the identified uncertainty contributors. These are classified by major physical quantities such as power, flow rate, temperature and the others, and then they are subdivided into three groups as follows:

Whole core. The deviations arising from the uncertainty contributors are the whole core averaged quantities, which are imposed to the boundary conditions in the first step analysis.

In-core local. The deviations arising from the uncertainty contributors have spatial distributions in a core but the averages of deviations over the whole core must be each zero. These are imposed to the boundary conditions in the second step analysis.

Intra-assembly. The deviations have spatial distributions in a fuel assembly but the averages of deviations over the whole fuel assembly must be each zero. These are imposed to the boundary conditions in the third step analysis.

The uncertainty contributors considered in the previous step analyses are imposed to the boundary conditions in the following step analyses.

Table 1. Categorization of uncertainty contributors

	Category	Applied Step
A. POWER		
Total Power		
Nuclear data (decay heat)	Whole Core	First
Power level measurement	Whole Core	First
Power Spatial Distribution		
Nuclear data (decay heat distribution)	In-Core Local	Second
Fuel maldistribution (plutonium enrichment)	In-Core Local	Second
Control rod position deviation	Intra-Assembly	Third
Fuel maldistribution (fabrication tolerance)	Intra-Assembly	Third
B. FLOW RATE		
Total Flow Rate		
Primary loop flow rate estimation	Whole Core	First
Flow Rate Distribution		
Inlet flow maldistribution	In-Core Local	Second
Inter-wrapper flow effect	In-Core Local	Second
Flow area (wrapper tube fabrication tolerance)	In-Core Local	Second
Flow area (duct swelling, etc)	In-Core Local	Second
Subchannel flow area (pin diameter, etc)	Intra-Assembly	Third
C. TEMPERATURE		
System Level		
R/V inlet/outlet temp. measurement	Whole Core	First
Temperature Distribution		
Wire spacer contact	Intra-Assembly	-
Pellet-cladding eccentricity	Intra-Assembly	-
D. OTHER		
Properties (specific heat, etc)	Whole Core	First

3.2.2 Consideration of spatial distribution of uncertainty contributor

We first added the consideration of spatial distribution of uncertainty contributor into the core hot spot evaluation because the flow rate distributions in a core and in a fuel assembly under the natural circulation conditions are strongly coupled with the coolant temperature distributions, while under the forced convection conditions the most conservative conditions can be assumed without accounting for buoyancy as the hottest fuel pin.

There are potentially an infinite number of patterns of spatial distribution for the in-core local and the intra-assembly uncertainties. We assumed that the probability density function of the uncertainty contributors obeys the normal distribution, and then we assumed plausible conservative spatial distribution patterns. For example, the larger values are near the center of a core or a fuel assembly and the smaller values are at the outer circumference, or inversely. By calculation results, the most conservative spatial distributions should be chosen.

3.3 Simplified model for thermal hydraulics in a fuel assembly

From the viewpoint of calculation cost reduction, we developed a simplified model for thermal hydraulics in a fuel assembly. We focused on one subchannel in the fuel assembly where the maximum secondary peak of the coolant temperature appears just after the flow coastdown ends. We used the eddy diffusivity concept to derive the momentum exchange between the hottest subchannel and its adjacent subchannels. The ratio of the transverse mass flux to the axial mass flux is assumed to be a constant during the transition from normal operation to natural convection. In terms of energy conservation of coolant, we assumed that the axial convective heat transport and the radial heat transport from the cladding surface are dominant and the energy mixing among the adjacent subchannels is negligible. The basic equations are described as follows:

Coolant momentum:
$$\frac{L}{A_c}\frac{dG}{dt} = \Delta P - \Delta P_F + \Delta P_H + \Delta P_E\,, \tag{1}$$

Coolant energy:
$$A_c \rho_c Cp_c \frac{\partial T_c}{\partial t} + G Cp_c \frac{\partial T_c}{\partial z} = a_c K_c (T_e - T_c)\,, \tag{2}$$

Cladding energy:
$$A_e \rho_e Cp_e \frac{\partial T_e}{\partial t} = 2\pi s K_b (T_f - T_e) + a_c K_c (T_c - T_e)\,, \tag{3}$$

Fuel energy:
$$\rho_f Cp_f \frac{\partial T_f}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(\lambda_f r \frac{\partial T_f}{\partial r}\right) + q_f\,. \tag{4}$$

3.3.1 Pressure difference between the top and bottom of a fuel assembly

To simulate the flow redistribution in a fuel assembly due to buoyancy, the transient pressure difference between the top and bottom of a fuel assembly which is calculated in the second step analysis is applied to the boundary conditions.

3.3.2 Pressure loss through a fuel assembly

The pin bundle pressure loss in a fuel assembly can be described in some experimental correlation equations. In this study the Cheng and Todreas correlation [2] is used for wire-wrapped pin bundle and the form pressure loss is considered for the other part.

3.3.3 Pressure head

Pressure head is the summation of the coolant density from the inlet to outlet of a fuel assembly. In the fuel pin bundle part of a fuel assembly, the coolant density of the subchannel is summed up. Above the fuel pin bundle part there is a plenum where the flows coming from the subchannels of the fuel pin bundle gather and the coolant temperature becomes uniform in the horizontal plane. The coolant density in this plenum is calculated from the fuel-assembly-averaged temperature obtained by the second step analysis.

3.3.4 Momentum exchange between subchannels

In a wire-wrapped fuel assembly, the cross flow occurs along a wire spacer spirally wrapping up a fuel pin. Such flow pattern promotes the mixing of the coolant in a fuel assembly. A simplified way to model the wire sweeping mixing is to use the eddy diffusivity concept. The sweeping flow is modeled as a constant transverse flow across the interior gaps between fuel pins. A dimensionless effective eddy diffusivity ε^* is defined to represent the sweeping effects as follows:

$$\varepsilon^* = \varepsilon / (V \cdot \eta). \tag{5}$$

The momentum exchange between the hottest and its adjacent subchannels is given by

$$\Delta P_E = -\frac{\varepsilon}{\delta}(J_a - J_h). \tag{6}$$

Assuming that the adjacent subchannel axial mass flux J_a is equal to the fuel-assembly-averaged axial mass flux J_f, we then arrive at

$$\begin{aligned} \Delta P_E &= -\frac{\varepsilon^* \cdot V \cdot \eta}{\delta}(J_a - J_h) = -\frac{\varepsilon^* \cdot \eta}{\delta} \cdot \frac{J_f}{\rho}(J_a - J_h) \\ &\cong -\frac{\varepsilon^* \cdot \eta}{\delta} \cdot \frac{1}{\rho} \cdot J_f(J_f - J_h) = C \cdot J_f(J_f - J_h). \end{aligned} \tag{7}$$

3.3.5 Accounting for uncertainties

Uncertainties should be considered through the boundary conditions as the heat generation of the hottest fuel pin and the flow area of the hottest subchannel.

3.3.6 Initial maximum coolant temperature

The initial maximum coolant temperature of the simplified model is set to be equal to the maximum coolant temperature calculated by using ASFRE code [5] by adjusting the initial subchannel flow rate G_0. Given the initial flow rate, the pressure loss and the pressure head can be calculated. The momentum exchange between subchannels is calculated by Eq.(8) which is derived by neglecting the unsteady term in Eq.(1).

$$\Delta P_E = -\Delta P + \Delta P_F - \Delta P_H. \tag{8}$$

Consequently, the constant value C in Eq.(7) should be determined.

4. Application to the evaluation of core hot spot under natural circulation decay heat removal operations in a large scale reactor

The three-step analysis method is applied to a loss-of-external-power event and a sodium leakage accident in a secondary loop of the 1500MWe reactor, which is designed in the feasibility study on commercialized fast reactor cycle systems [3].

4.1 Computer codes in each step analysis

In this study, the first step analysis of the new method is divided into two analyses due to the restriction of our analysis codes to use. At first, a plant dynamics code, in which the core, primary and secondary loops, and DRACS and PRACS systems are linked with flow network model, is used to estimate the core flow rate and the core inlet coolant temperature. Next, a whole core thermal-hydraulic code TREFOIL [4] is used to identify a hottest channel candidate in the core and to calculate the boundary conditions for the next step analysis. The TREFOIL code is used again for the second step analysis to calculate the flow rate and the coolant temperature in the hottest channel. For the third step analysis, a subchannel analysis code ASFRE [5] is used to simulate the coolant flow and temperature fields and the cladding tube temperature distribution in the hottest fuel assembly.

4.2 Analysis conditions

From the viewpoint of fuel pin structure integrity, it is necessary to evaluate the maximum cladding temperatures during several minutes or several hours after scram of the reactor. However, in this paper we focus on a relatively short period of time which is about 200s, because the physical phenomena particular to natural circulation, such as inter-fuel-assembly heat transfer and flow redistribution in the core and in each fuel assembly due to buoyancy force appear during the transition from normal operation to natural convection, and cause flattening of the transverse temperature distributions in the core and in the fuel assembly. PRACS and DRACS are not in operation yet in this early period.

Consider plant dynamics of JSFR in the loss-of-external-power event and in a sodium leakage accident in a secondary loop. These cases are assumed to be initiated from full power conditions, and then trips of primary and secondary loop pumps due to the loss of external power are followed by scram of the reactor. In both cases heat transfer of the steam generator (SG) connecting a secondary loop to a water loop was assumed to be zero, we imposed adiabatic condition on the heat transfer tube surface of the two SGs. In the case of a sodium leakage accident in one secondary loop, since heat transfer of the intermediate heat exchanger (IHX) connecting a primary loop to a secondary loop can be assumed to be zero, we imposed adiabatic condition on the tube surface of the IHX whose secondary loop is broken.

The deviations of uncertainties and the selected spatial distribution patterns are summarized in Table 2. Some uncertainty contributors, of which deviations could not be estimated directly because the target plant is being designed and the detailed data is not fixed, are tentatively set to be the values which are estimated in the installation permission of Monju reactor and in the previous study for the demonstration fast breeder reactor [6]. With regard to the spatial distribution pattern of uncertainty, it is basically assumed that the probability density function obeys the normal distribution and the value of deviation in the center of a core/fuel assembly is larger than that at the outer circumference. Some uncertainties are assumed to lead to more conservative estimation. The flow reduction due to fabrication tolerance of the entrance part of a fuel assembly is assumed to occur only in the most inner flow region of eight concentric regions in the core. Since the inter-wrapper flow enhances the heat transfer between fuel assemblies and reduces the maximum cladding temperature, it is assumed that there is no inter-wrapper flow in the core. The flow area deviation of the fuel assemblies due to creep and swelling deformation is assumed to be proportional to the power distribution in the core.

Table 2. Deviation and spatial distribution pattern of uncertainty

Uncertainty contributors	Deviation	Distribution pattern
Whole Core		
Nuclear data (decay heat)	+10 %	-
Power level measurement	+2 %	-
Primary loop flow rate estimation	(conservative estimation)	-
R/V inlet/outlet temp. measurement	+3 ^{o}C	-
In-Core Local		
Nuclear data (decay heat distribution)	±5 %	larger in the center
Fuel maldistribution (plutonium enrichment)	±3 %	larger in the center
Inlet flow maldistribution	- 7 %	only in the most inner flow region in a core
Inter-wrapper flow effect	(no flow assumption)	
Flow area (wrapper tube fabrication tolerance)	±1%	smaller in the center
Flow area (duct swelling, etc)	-2% ~ -10%	proportional to power distribution
Intra-Assembly		
Fuel maldistribution (fabrication tolerance)	±1%	larger in the center
Subchannel flow area (pin diameter, etc)	±3%	smaller in the center
Wire spacer contact	+4 ^{o}C	-
Pellet-cladding eccentricity	+1 ^{o}C	-

4.3 Calculation results

4.3.1 Maximum coolant temperature transient

Figure 3 contains the calculation results in a loss-of-external-power event and in a sodium leakage accident in a secondary loop. Shown in these graphs are reactor power, reactor flow and the maximum coolant temperature which is almost the same as the maximum cladding temperature under such the low flow rate conditions. As can be seen from the nominal evaluation results which are calculated by using a whole core thermal-hydraulic analysis code

ACT [7], the coolant temperature in the core rises to first peak in a few seconds due to the short time-lag of control rod insertion after the pump trip, and then the coolant temperature drops immediately following the reduction of core heat generation because the primary loop pump coastdown characteristics retain the flow rate above the power curve. When the balance between heat generation and heat removal is reversed during the flow coast-down, the coolant temperature starts rising again. This temperature rise generates buoyancy force which causes natural circulation and limits the temperature rise to second peak. The proposed new method evaluates the hot spot temperature at the secondary peak 40°C lower than the conventional method and 50°C higher than the nominal simulation in a loss-of-external-power event. In another case, the proposed method also evaluates the temperature 80°C lower and 50°C higher, respectively. The evaluated the temperatures are summarized in Table 3. As has been demonstrated, the proposed method can estimate the hot spot with a reasonable degree of conservativeness.

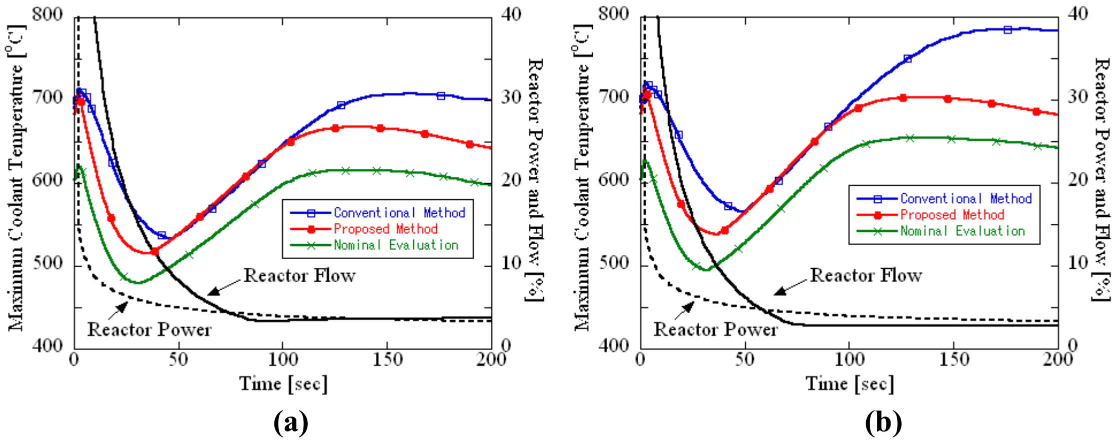


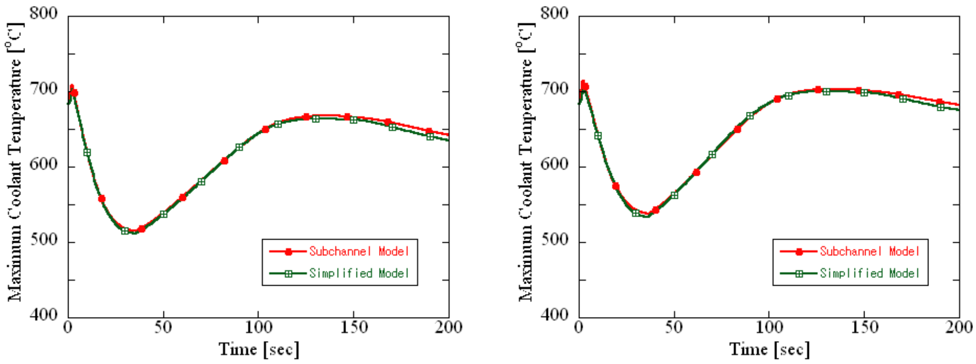
Figure 3. Maximum coolant temperatures: (a) in a loss-of-external-power event (b) in a sodium leakage accident in a secondary loop

Table 3. Maximum coolant temperature at secondary peak

Method	Loss of external power event	Sodium leakage accident in a 2nd loop
Conventional	708 °C	786 °C
Proposed	668 °C	703 °C
Nominal	616 °C	654 °C

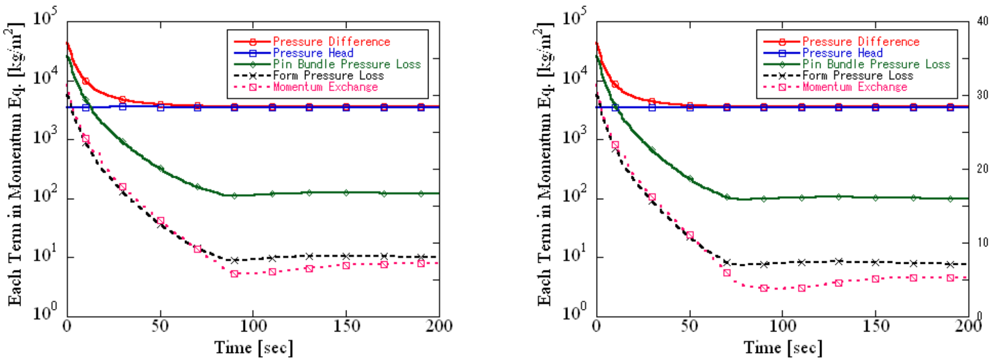
4.3.2 Evaluation using simplified model

Using the simplified model for thermal hydraulics in a fuel assembly, the maximum coolant temperatures in a loss-of-external-power event and in a sodium leakage accident in a secondary loop are calculated as shown in Figure 4. The simplified model results are in good agreement with the subchannel model results. The discrepancy in temperature between the two models is less than 4 °C during 200 seconds in each case. Figure 5 shows the values of each term in the momentum equation (Eq.(1)). Until the flow coastdown ends at about 70 seconds, the value of the momentum exchange between subchannels term shows the same trend as the value of the form pressure loss in a fuel assembly term, which is about 10 times smaller than the value of the pin bundle pressure loss term. After about 70 seconds, the pin bundle pressure loss and the pressure head terms are dominant and the sum of them consists of over 99% of the pressure difference between the top and bottom of a fuel assembly. This may explain that the flow redistribution between the hottest and the other subchannels ends during the flow coastdown. With regard to the calculation cost, in this study the simplified model calculation cost is about 600 times smaller than that of the subchannel model. As expected, the calculation cost is simply proportional to the number of subchannels in a fuel assembly.



(a) (b)

Figure 4. Maximum coolant temperatures by the simplified model evaluation: (a) in a loss-of-external-power event (b) in a sodium leakage accident in a secondary loop



(a) (b)

Figure 5. Values of each term in momentum equation: (a) in a loss-of-external-power event (b) in a sodium leakage accident in a secondary loop

5. Conclusion

A new evaluation method of core hot spot which can be applied to natural circulation decay heat removal has been developed. The new method consists of three-step thermal hydraulics analyses in order to consider the effects of physical phenomena particular to natural circulation, such as inter-fuel-assembly heat transfer and flow redistribution in the core and in each fuel assembly due to buoyancy force. The three-step analysis method was applied to the evaluation of a loss-of-external-power event and of a sodium leakage accident in a secondary loop of a large scale reactor. The calculated results were compared with those obtained by a conventional method and were also compared with a detailed three-dimensional simulation in the nominal condition. It was confirmed that the three-step analysis method can estimate the hot spot with a reasonable degree of conservativeness. In addition, from the viewpoint of calculation cost reduction, we also developed a simplified model substituting for the third step analysis (subchannel analysis) which simulates the flow and temperature fields and cladding tube temperature distributions in a fuel assembly. The calculation result of the simplified model is in good agreement with the subchannel analysis result and the calculation cost can be significantly reduced.

Acknowledgments

The evaluation method of the hot spot in the core was conducted as part of 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 evaluation method of the hot spot in the core was performed by Japan Atomic Energy Agency (JAEA) under the contract with MFBR. Plant dynamics code and thermal-hydraulic analysis code runs were done by Mr. T., Iwasaki of NESI Corporation and Mr. A., Hashimoto of NDD corporation.

Nomenclature

a	linear density of contacting area	[m^2/m]
A	flow cross section	[m^2]
C	constant value	[-]
Cp	specific heat capacity at constant pressure	[J/kg K]
G	mass flow rate in a subchannel	[kg /s]
J_h	axial mass flux of hottest subchannel	[kg/m^2 s]
J_a	axial mass flux of adjacent subchannel	[kg/m^2 s]
J_f	fuel-assembly-averaged axial mass flux	[kg/m^2 s]
K	overall heat transmission coefficient	[W/m^2 K]
L	fuel assembly axial length	[m]
ΔP	pressure difference between the top and bottom of a fuel assembly	[Pa]
ΔP_F	pressure loss through a fuel assembly	[Pa]
ΔP_H	pressure head	[Pa]
ΔP_E	momentum exchange between subchannels	[Pa]
q	power density	[W/m^3]
s	gap distance between fuel and cladding	[m]
V	axial velocity	[m/s]

subscripts

b	gap between fuel and cladding
c	coolant
e	cladding
f	fuel

Greek Letters

δ	mixing scale	[m]
ε^*	dimensionless effective eddy diffusivity	[-]
ε	effective eddy diffusivity	[m^2/s]
η	subchannel centroid to centroid distance	[m]
λ	thermal conduction coefficient	[W/m K]
ρ	fuel-assembly-averaged coolant density	[kg/m^3]

References

[1] M. Ichimiya, T. Mizuno and S. Kotake, "A next generation sodium-cooled fast reactor concept and its R&D program", Nuclear Engineering and Technology, 39, 2007, No.3, pp.171-186.

[2] S. Cheng and N. E. Todreas, "Hydrodynamics Models and Correlations for Bare and Wire-Wrapped Hexagonal Rod Bundles - Bundle Friction Factors, Subchannel Friction Factors and Mixing Parameters", Nuclear Engineering and Design, 92, 1986, pp.227-251.

[3] Japan Atomic Energy Agency and The Japan Atomic Power Company, Feasibility Study on Commercialized Fast Reactor Cycle Systems—Phase II Final Report—, Japan Atomic Energy Agency Report, JAEA-Evaluation 2006-02 [in Japanese].

[4] O. Watanabe, S. Kotake, S. kubo, H. Kajiwara and K. Fujimata, "Study on Natural Circulation Evaluation Method for a Large FBR", Proceedings of the 8th International Topical Meeting on Nuclear Thermal-Hydraulics (NURETH-8), Kyoto, Japan, 1997, Vol.2, pp.828-838.

[5] H. Ohshima, H. Narita and H. Ninokata, "Thermal-Hydraulic Analysis of Fast Reactor Fuel Subassembly with Porous Blockages", Proceedings of the 4th International seminar on subchannel Analysis (ISSCA-4), Tokyo, 1997, pp.323-333.

[6] M. Ueta, T. Inagaki, Y. Shibata, K. Tarutani and K. Okada, "The development of demonstration fast breeder reactor (DFBR)", Transactions of the 13th International Conference on Structural Mechanics in Reactor Technology (SMiRT 13), Porto Alegre, Brazil, 1995, pp.359-369.

[7] H. Ohshima and M. Ohtaka, "Development of Computer Program for Whole Core Thermal-Hydraulic Analysis of Fast Reactors", Proceedings of the 10th International Conference on Nuclear Engineering (ICONE-10), Arlington, USA, 2002, ICONE10-22034.