Development of multi-COBRA: A Transient Thermal-hydraulic Analysis Code for Fast Reactor Cores

Background

In liquid-metal-cooled fast reactors (LMFRs), the coolant flow coastdown events due to tripping the primary pumps lead to redistribution of coolant flow towards relatively high temperature subassembly channels, causing variation of core temperature profiles because the effects of buoyancy-induced flow and lateral heat transfer become conspicuous in low-flow transients. These phenomena should be taken into account in the core safety evaluations such as analyses of fuel integrity or core-bowing behavior under transient conditions. In order to obtain the detailed coolant flow rate and temperature distributions in each channel of the whole core, subchannel analysis for multi-assembly system is indispensable. However, there is not such a public calculation code.

Objectives

In this study, a multi-assembly transient thermal-hydroulic analysis code for fast reactor cores based on the subchannel-model is developed for analyzing low-flow transients from normal operation condition to natural circulation cooling.

Principal Results

1. Development of a subchannel analysis code for multi-assembly core

A transient thermal-hydraulic analysis code for fast reactor cores, multi-COBRA, has been developed as follows,

(1) On the basis of COBRA-IV-I, which is a subchannel analysis code for single-assembly, an advanced explicit solution method for solving fluid momentum equations as the linear functions of each calculative node pressure with pressure drop boundary condition was developed in order to analyze stably localized recirculating flow phenomena, and implemented in the code.

(2) Further, a WALL-model (Fig.1) to treat inter-assembly heat transfer and a NETWORK-model (Fig.2) to evaluate a whole core pressure drop by linking each subassembly have been incorporated in the code.

In multi-COBRA code, the transient thermal-hydraulic behavior of fast reactor cores from normal operating conditions to natural circulation cooling can be analyzed in detail using pressure drop or inlet flow boundary condition of the whole core.

2. Validation of multi-COBRA code

A validation of this code has been performed by a comparison with experimental data obtained by natural circulation transient test-7A of the Experimental Breeder Reactor-II (EBR-II) in the U.S. In this experiment, initial reactor power and flow levels were set at 28.5% and 32.1% of their respective full values, and the low-flow transient initiated by the primary pump trip and reactor scrum resulted in natural circulation cooling (Fig.4). The experimental analysis was performed on a partial core consisting of 7 assemblies described in Fig.3.

(1) The analytical results of top-of-core averaged temperatures in XX08 were in fair agreement with the experimental values about a rapid temperature drop by reactor scram and smooth transition to natural circulation cooling which follows an occurrence of second peak temperature at 55 seconds after the pump trip (Fig.5).

(2) It was predicted well that the peaking factor of coolant temperature decreases during loss-of-flow event (Table 1). Therefore, the flattening phenomena of transverse temperature profiles in the subassembly during low-flow transient could be simulated by considering buoyancy-induced flow and lateral heat transfer.

(3) It was analyzed that the allocation of coolant flow rate to XX08 in total flow of 7 assemblies decreased by approximately 10% after about 40 seconds from the beginning of the transient at which the pump rotation stopped completely and forced circulation power was lost (Fig.6). This result indicates that multi-COBRA can calculate the redistribution of coolant-flow towards higher temperature assemblies from XX08 of which the steady-state temperature is relatively low.

As the results, one may conclude that multi-COBRA is valid for quantitative analysis of thermal-hydraulic phenomena of fast reactor core and is particularly useful for the evaluation of low-flow transient behavior in which the effects of buoyancy-induced flow and transverse heat transfer become conspicuous.

Future Developments

The design studies of metal fuel fast reactor cores in which a passive safety feature is ensured even under the unprotected loss-of-flow events will be conducted through detailed analyses using multi-COBRA.

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Reference

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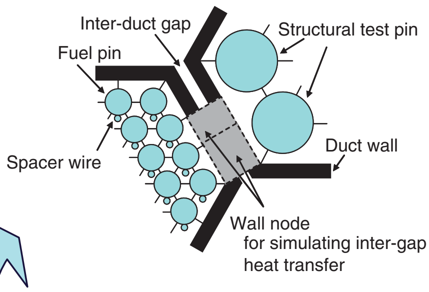

Fig. 1 Concept of inter-duct heat transfer WALL-model

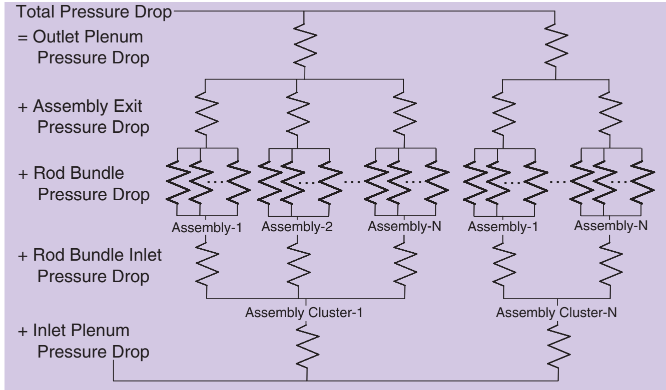

Fig. 2 Schematic description of pressure drop NETWORK-model

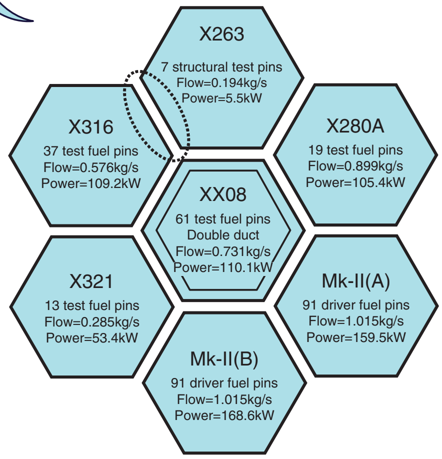

Fig. 3 Configuration of EBR-II test-7A analysis

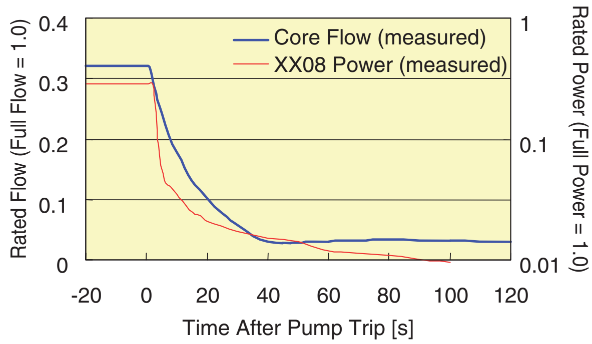

Fig.4 Coolant flow rate and XX08 power profiles during EBR-II test-7A

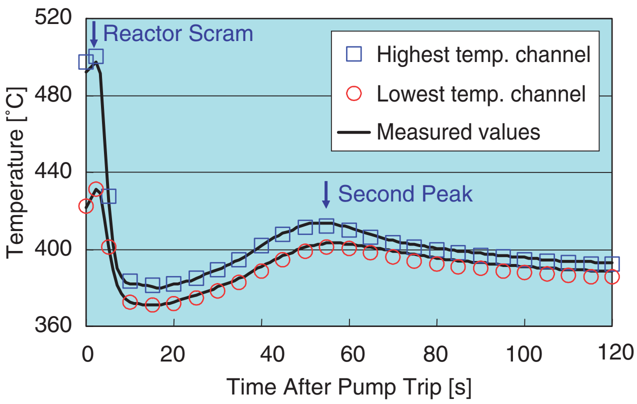

Fig.5 Top-of-core averaged temperature profile in XX08 during EBR-II test-7A

Table 1 Summery of Peaking Factors on Top-of-Core Coolant Temperature of XX08 in EBR-II test-7A

	Steady-State	*Second Peak*
Measured Value	1.419	1.114
Analyzed Value	1.426	1.156

Peaking Factor = $(\Delta T + 2\sigma)/\Delta T$
ΔT = Averaged core temperature rise
σ = Standard deviation

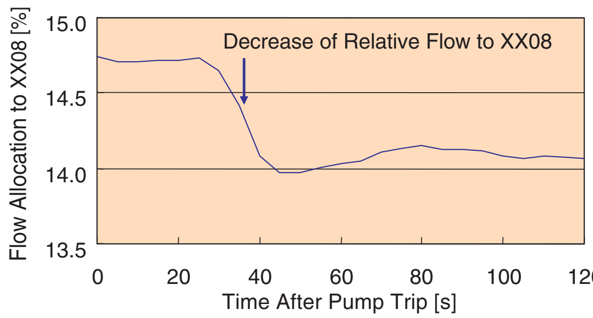

Fig.6 Analyzed coolant flow profile allocated to XX08 during EBR-II test-7A