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Recent Development of the Inter-Assembly Flow Analysis Tools for SFR Core Thermal Hydraulics

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1. Introduction

A typical SFR core is generally comprised of hundreds of hexagonal type ducted subassemblies. And these subassemblies have hundreds of fuel rods with a triangular channel arrangement forming a closed circuit by themselves without any flow path between them.

Subchannel analysis is considered to be the most suitable method for the LMR subassembly analysis when considering the geometrical complexities and computational resources needs. MATRA-LMR was developed as an analysis code to predict flow and temperature fields in SFR subassemblies.

In the SFR core, flow redistribution can be occurred in the inter-assembly region of the core. The hotter counter flow from the upper center region of the LMR core may have a significant effect on the thermo-mechanical integrity of the duct wall.

This paper describes the recent development of the inter-assembly flow analysis tools for SFR core thermal hydraulics and shows a few calculation results.

2. Inter-assembly Flow

In SFR, some parts of the subassembly outlet flow are affected by an overpressure profile because of the upper internal structure (UIS) at the core upper outlet region. A simplified description of the inter-assembly flow redistribution effect is shown in Figure 1.

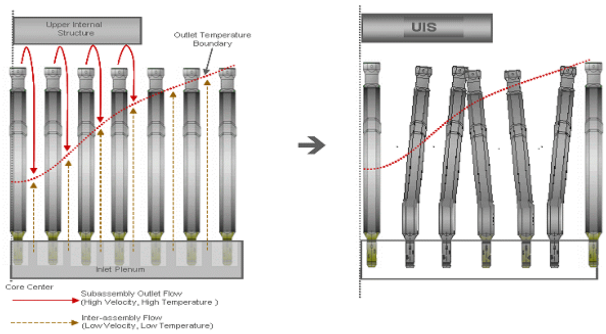

Figure 1. A sketch for the inter-assembly flow effect

The reversed flow rate generated by the overpressure can be higher than the up-and-coming inter-assembly flow rate, which is a leak flow at the lower part of the subassemblies. The hotter counter flow in the upper region runs down into the center region and it may raise the duct wall temperature. This characteristic of SFR core may have a significant effect on the thermo-mechanical integrity of the duct wall. Therefore, it is required to develop an appropriate method to analyze the inter-assembly flow in SFR core [1].

As a part of KALIMER core thermal hydraulic analysis codes, a combined calculation method is being developed at KAERI. The objective of the method is to obtain the temperatures of a subassembly duct walls, strongly influenced by the inter-assembly flow redistribution in the core. The method is based on subchannel analysis and porous media approach and will be integrated into a LMR whole core thermal hydraulic code.

3. Method and Results

The overall tools for the core thermal hydraulic design and analysis, which is used in the KALIMER core conceptual design, are summarized as shown in figure 2, i.e., flow grouping and peak pin temperature calculation, pressure drop calculation, steady state and detailed subchannel analyses, and inter-assembly flow analysis. These tools are expected to be extensively used for a basic data production during the conceptual design phase.

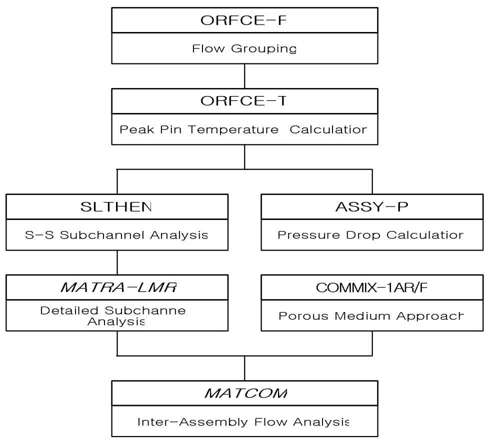

Figure 2. KALIMR core thermal hydraulic analysis tools

Some of the calculation results using these tools are given in this paper. Figure 3 shows duct temperatures in the center region assembly according to the pressure distribution at the assembly outlet. Duct temperatures in the outer region assembly with and without calculation iteration are shown in figure 4. The pressure distribution and iteration effect can be seen in the figures.

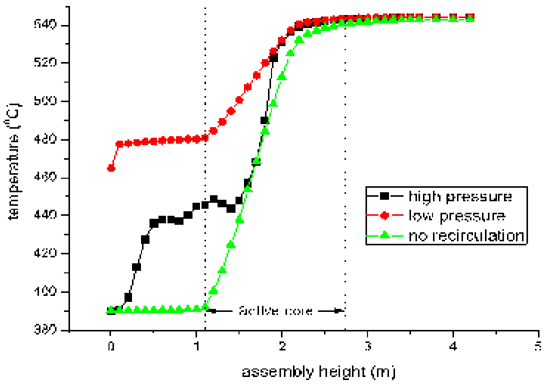

Figure 3. Duct temperatures in the center region assembly

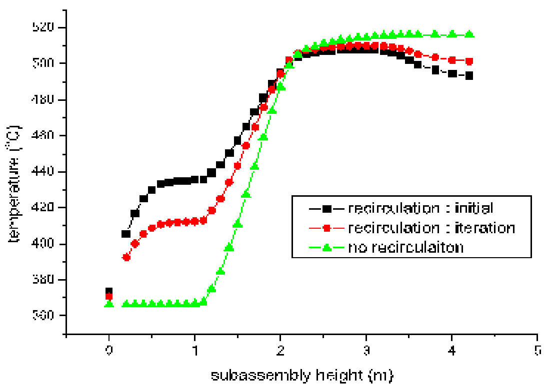

Figure 4. Duct temperatures in the outer region assembly

Figure 5 and 6 show temperature and velocity distributions in the inter-assembly region of the cores KALIMER and SPX, respectively. The temperature and pressure have almost the same distributions as seen in the figures.

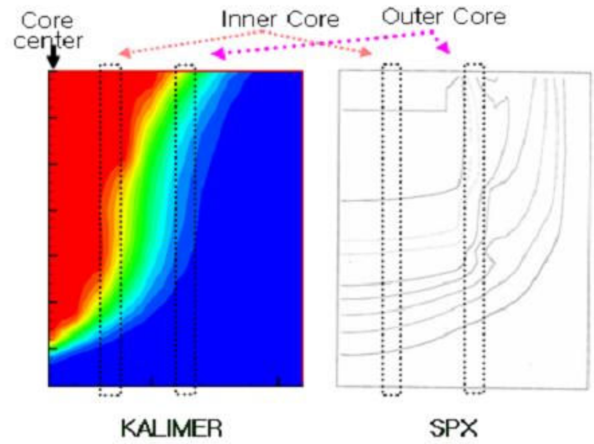

Figure 5. Temperature distribution of the inter-assembly flow

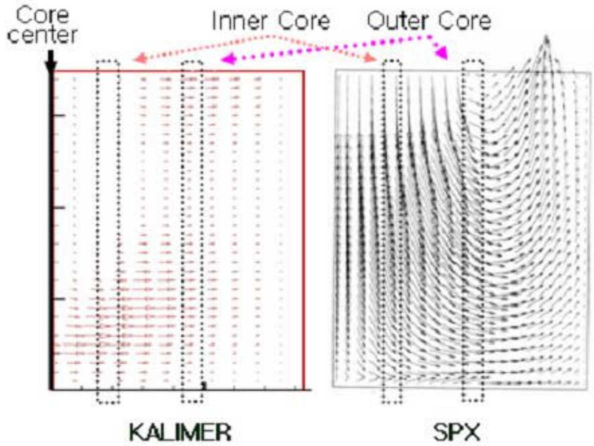

Figure 6. Velocity distribution of the inter-assembly flow

4. Summary and Further Studies

The recent development of the inter-assembly flow analysis tools for SFR core thermal hydraulics is explained and a few calculation results of KALIMER breakeven core are shown in this paper. The inter-assembly analysis tools will be improved and verified for the core conceptual design and evaluation

Acknowledgement

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REFERENCES

[1] Valentin B. et al., Evaluation of Wrapper Tube Temperatures of Fast Neutron Reactors using The TRANSCOEUR-2 Code, IAEA TCM, Obninsk, Russia (1998)