Computational Thermofluid Dynamics in Rod Bundles of Liquid Metal Reactor

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1. Introduction

The Battery Omnibus Reactor Integral System (BORIS) is being developed as a multipurpose integral optimized fast reactor with an ultra long life core [1]. BORIS is operated by natural circulation of the primary system lead (Pb) coolant without pump power. BREST is a liquid metal reactor with rod bundles in square arrangement. Its mockup test section contains one spacer grid located near the midplane of the rod bundle. These bundles have two zones. The rod diameters and heat production differ between the two zones [2]. The experiment served as a standard thermal and hydraulic problem. This study focuses on thermohydrodynamics in the primary system of liquid metal cooled reactor utilizing the computational fluid dynamics (CFD) code CFX.

2. Rod Bundle Design and Grouping

The BREST-type reactor rod bundle has two sections differing in the rod diameter and heat flux [2]. The rod bundle has different thermal conditions. This study has shown how the thermal conditions differ by group, and the temperature rise is always higher on the high power rod region. Fig. 1 displays the grouping by diameters. In this study, the bundle is three-dimensionally modeled using CATIA as shown in Fig. 2.

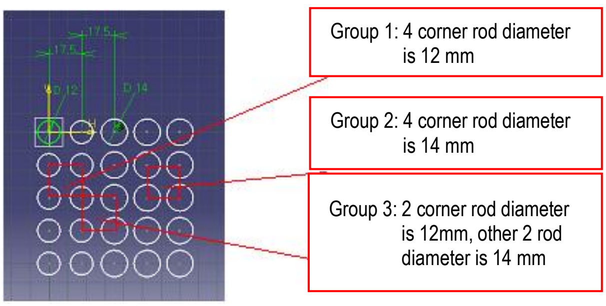

Fig. 1. Grouping of fluid region.

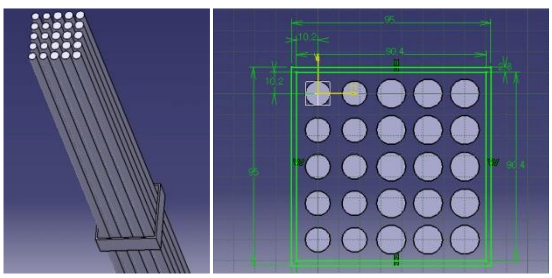

Fig. 2. 3-D Modeling of rod bundle with grid spacer.

3. Input Conditions

CFX allows testing systems in a virtual environment. CFX has been applied to simulation of coolant flowing in the BREST-Type reactor core.

Since it is difficult enough to measure temperature and velocity profiles in the reactor core, visualization of the liquid metal hydrodynamic behavior by CFX affords more detailed information.

CFX requires the boundary condition parameter and specific coolant data. This study used the boundary condition of BORIS in Table I and two coolants of Pb and water [3].

Table I: CFX Boundary Condition

Boundary Type	Inlet	Outlet	Wall (Fuel)
Mass and Momentum	Normal speed: 0.415 m/s	Average static pressure: 101325 Pa	
Heat Transfer	Total temperature: 713.15 K		Heat flux: W/m^2 G1: 52319.2 G2: 30270.2 G3: 41294.7

The pressure drop and heat transfer coefficient in the reactor core are important for system design of BORIS because the thermal energy from fission is removed by natural circulation coolant. Thus, this study focuses on the pressure and temperature profiles.

The Inlet temperature in the BORIS core is 713.15 K [3]. The Pb thermophysical properties in Table II [4] are applied to simulation of the Pb coolant in CFX-Pre.

4. CFX Analysis Results

Fig. 3 shows the temperature profiles in the rod bundle with different coolants while the heat flux was fixed at 52319.2 W/m^2. The temperature increased linearly. The Pb temperature is higher than that of water in the same group. Figs. 4 and 5 present the axial temperature and pressure profiles by the group. The temperature of the mixed region in group 3 linearly increases for Pb. Fig. 6 displays the higher coolant temperature rise on the side group. It is similar to results obtained by other analysis codes FLUENT and MATRA [5] at the height of 676 mm.

5. Conclusions

In this study, the higher coolant temperature rise was observed on the side where the higher rod power was supplied. The result of this study is almost the same as those calculated by FLUENT and MATRA. More exact simulation requires accurate modeling of the mixing vane geometry and computation of the flow in the grid spacer region.

Table II: Pb Thermophysical Properties

Density (kg/m^3)	10515.21
Specific Heat Capacity (J/kgK)	119
Dynamic Viscosity (Pa s)	0.002037
Thermal Conductivity (W/mK)	87.6465
Thermal Expansivity (1/K)	0.0001136

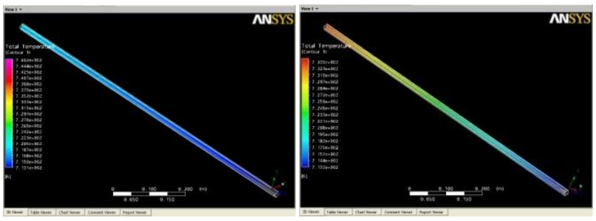

Fig. 3. Temperature contour with water and Pb coolant.

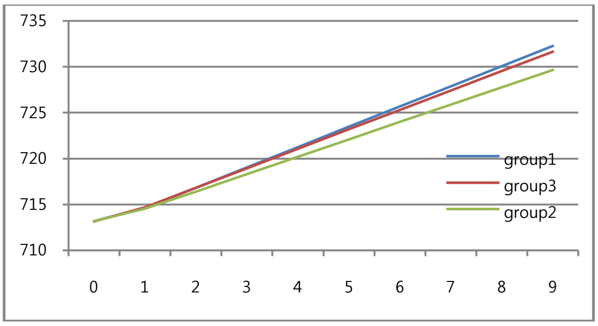

Fig. 4. Axial Temperature Distribution in Pb Coolant

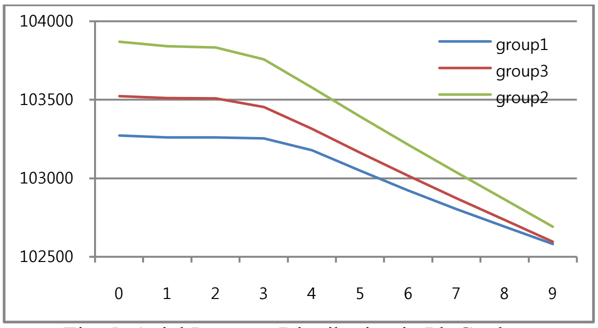

Fig. 5. Axial Pressure Distribution in Pb Coolant

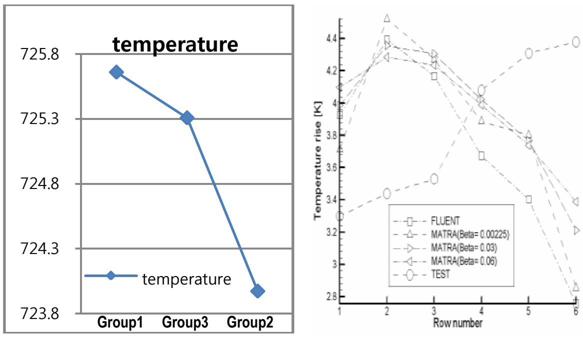

Fig. 6. Computed and measured temperatures.

REFERENCES

[1] H.M. Son, Y.H. Yu, and K.Y. Suh, "Primary System Design for Lead Cooled Battery Fast Reactor BORIS," Transactions of the American Nuclear Society Winter Meeting, Tracking ID: 163551, Albuquerque, NM, USA, 2006.
[2] H.M. Son and K.Y. Suh, "CFD Analysis of Liquid Metal Cooled Rod Assembly," Paper 7332, Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP), Nice, France, May 13-18, 2007.
[3] H.M. Son and K.Y. Suh, "Thermohydrodynamic Analysis of Reactor Vessel Auxiliary Cooling System for Lead-Cooled Battery Omnibus Reactor Integral System," Paper 8170, Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP), Anaheim, CA, USA, June 8-12, 2008.
[4] OECD/NEA, "Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies," Paris, France, 2007.
[5] H.M. Son and K.Y. Suh, "Solution to a Standard Problem Hydraulics and Heat Transfer in a Subassembly Model Cooled by Liquid Metal," Paper NTHAS5-F007, Proceedings of the 5th Japan-Korea Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS), Jeju, Korea, November 26-29, 2006.