COE-INES Business Trip Report

Meetings attended: 11th Meeting of the Working Group on Advanced Nuclear Reactors Thermal Hydraulics "hydrodynamics and heat transfer in single and two-phase flow of liquid metals."
Location: Obninsk, Russia
Person attending: 原子核工学専攻博士後期課程 2 年 Georgy Sorokin
Dates: 4-10 July, 2004

Known as the First Russian Science City, Obninsk offers many diverse cultural, educational, industrial, and recreational opportunities. Obninsk has exceptional parks and recreational facilities, numerous service organizations and clubs. But more than that, Obninsk has 108,000 people who care about city and its guests. The location of Obninsk is comfortable. The proximity of major highways, railroads and Moscow International airports gives opportunity for any type of access to Obninsk.

The 11-th meeting of the IAHR Working Group on Advanced Nuclear Reactor Thermo hydraulics was held at the State Scientific Center of Russian Federation - Institute for Physics and Power Engineering named after A.I. Leypunsky (SSC RF-IPPE) in Obninsk, Russian Federation, from July 5 to 9, 2004. The general topic of the meeting was "Hydrodynamics and Heat Transfer in Single and Two-Phase Flow of Liquid Metals."

The objects of the meeting were to discuss new results of the researches performed in the field of liquid metal coolant thermohydraulics and to recommend lines of common physical and applied thermohydraulic researches for justification of existing and developing liquid metal cooled advanced nuclear reactor of new generation.

The meeting was attended by 72 participants representing 6 different countries (Japan, Nederland, Spain, Korea, India, Russia, as well as IAEA). During four days the plenary session and two technical sessions were held, one day more was devoted to discussion of benchmark calculations.

Meeting was opened by Head of Working Group on Advanced Nuclear Reactor Thermohydraulics Prof. A.P. Sorokin. He presented an overview of activity of International Association of Hydraulic Researches, history of creation and activity of the Working Group. Then Director of the Institute for Heat and Mass Transfer in Nuclear Power Engineering Prof. A.D. Efanov greeted participants.

The most participants were involved in the presentations and/or in the benchmark. They all noted that size of the meeting is considered to be nearly optimal for active and constructive discussions and fruitful information exchanges. Participants agreed that the spirit and philosophy of our meeting was successfully maintained: active participation by the attendees and informal and fruitful exchange of technical information with emphasis on discussions.

The papers were grouped into two sections:

- Thermohydraulics of fast reactor core cooled by liquid metals;
- Hydrodynamics and heat and mass transfer in liquid metals.

In the section "Thermohydraulics of fast reactor core cooled by liquid metals," approaches to experimental and numerical modeling, as well as results of researches of fast reactor

Fig. 1 During the presentation

thermohydraulics with sodium cooling were discussed. Also analysis of an influence of geometrical and flow parameters was presented. In the frame of the section a thermohydraulics of heavy metal cooled reactor was discussed. Results of investigations into specific features of heat and mass transfer in heavy metal cooled pin bundles induced by the availability of spacer grids were presented, too. It was noted that new methods of numerical modeling of thermohydraulics have received further development.

The work reported in the section “Thermohydraulics of fast reactor core cooled by liquid metals” has been carried out in the framework of innovative small liquid-metal-cooled reactor development program of COE-INES.

The investigations of thermalhydraulics on the experimental models of pin subassemblies of core modeling showed the possibility of cooling the core with the emergence of boiling in sodium. The nucleate, slugging and annular-dispersed two-phase flows of liquid metal and appearance of the crisis of heat exchange with natural convection conditions with low void fraction are observed. Therefore the studying the characteristics of heat exchange and nice numerical calculation of these regimes for the real parameters of active core is the very important task. The design feature of reactor core with the sodium cooling is the presence of the system hydrodynamically connected parallel channels (through the inlet and outlet headers). The mutual thermohydraulic influence of channels, the initiation of flow instability, changes of cooling conditions in the assemblies have been observed.

Fig. 2 During the presentation

The data which obtained in the experiments with the single fuel assembly model were shown that the different factors like construction design, hydrodynamic resistances of its different components, the geometry of pin bundle and length of the heating area, the parameters of coolant, the surface condition of heaters influence to initiation and evolution of the boiling in liquid metal at the assemblies.

In particular, the features of heat exchange are connected with the large difference between the vapor and liquid densities at coolant under the low pressure conditions in the reactor core. The data of liquid metal two-phase flow in the system with parallel channels are absent.

The experimental model for investigating the liquid metal boiling in the system of parallel channels, established on the faculty AR-1 in IPPE, simulates the typical geometry of the fast reactor core (BN-600, BN-800).

The experimental data which obtained with circulation of coolant in the right and left loops and also in the system of parallel loops have allowed analyzing the phenomena of assemblies' interaction. Two-fluid model at present is powerful tool for the simulation of hydrodynamics and heat exchange of two-phase flow in the channels. Its use has allowed simulating effects in the coolant with low velocity, under low pressure conditions, with mixing of phases, at acceleration of flow, with typical for the solving tasks. Using for the most critical part (for the power flux section) the SABENA code allows correctly considering these effects in solving the current task. An accounting of the upper plenum component influence, mixing of flows from different loops and so on is the trait of the closed loop task simulation. The verification of the developed code with the use of obtained experimental data is of fundamental importance. Both the results of experimental studies and the preliminary results of calculation were presented.

Fig. 3 Prof. F.A. Kozlov

Therefore the report arrested the special attention. Prof. E.F. Avdeev (Head of Thermophysical laboratory, Obninsk State Technical University for Nuclear Power Engineering (INPE)) gave an interesting note about importance and timeliness of ongoing activity at this field. Following the comment of Prof. F.A. Kozlov (Deputy Director of Thermophysical Division of SSC RF IPPE) a discussion was also held on the correct numerical approach adopted at the calculations. A comprehensive discussion of the SABENA code was also given, following the request from Prof. V.A. Rykov (Scientific Secretary of SSC RF IPPE) exploring the development history and applications of this code. After the presentation a discussion was with Prof. Kune Y. Suh (Seoul National University, Korea) about using the correlations for Na-K eutectic alloy in current calculations.

A visit to the First in the World Nuclear Power Plant and liquid metal test facilities of SSC RF IPPE was organized on 7th of July. Participants of the meeting visited: facility 6-B (Na, Na-K), AR-1 facility (Na-K), LIC facility (Li). On the whole the atmosphere of the meeting was very friendly and productive.