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Introduction of Transient Heat Transfer Experiment on Tube Flow for RIA Applications

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1. Introduction

A reactivity initiated accident (RIA) is a nuclear reactor accident due to a sudden increase in fission rate and reactor power[1]. RIA can damage the reactor core and even leads to disruption of the reactor in very severe cases. The cause of RIA is inadvertent insertion of reactivity to reactor core. Reactivity insertion events in power reactors can be divided broadly into control system failures, control rod ejections, events caused by coolant/moderator temperature and void effects, and events caused by dilution or removal of coolant/ moderator poison. Except the control rod ejections, most reactivity insertion events lead to relatively slow reactivity additions and they are considered as transients rather than accidents. However, the control rod ejection event is categorized to design basis accidents since it causes very rapid reactivity insertion[1].

In pressurized water reactors (PWRs), the control rod ejection events can occur by mechanical failure of the control element assembly nozzle or control rod dive mechanism housing. In that cases, the coolant pressure ejects a control rod assembly completely out of the core. As a consequence, the reactivity of the core is rapidly increased and the power excursion is resulted in the reactor core with large localized relative power increase. In the worst possible scenario[1], the reactivity addition can occur within about 0.1 s which is shorter than the actuation time of safety systems such as the emergency core cooling system. Therefore present RIA safety criteria are focused on maintaining the fuel integrity and core coolability during and after the RIA[2-4].

In 1990’s, test results from NSRR and CABRI reveal that high burn-up nuclear fuels could fail even at lower enthalpy than the previously accepted safety limit in case of a RIA, due to the effect of burnup-related and cladding corrosion-related phenomena on fuel rod performance. After that, interim acceptance criteria and guidance against the RIA are issued by NRC in 2007 and 2015[3, 4]. Unfortunately, those newly raised RIA safety criteria encroach previous safety margin due to the adoption of conservative evaluation method and high uncertainty in the analysis. Therefore a best estimate method is strongly required to secure the decreased safety margin by reducing the uncertainty in the RIA evaluations. However, there is a huge lack of knowledge related to the RIA up to now due to the difficulty and high costs for the RIA experiments. Especially, experimental studies on the thermal hydraulic phenomena are insufficient yet since it is difficult to realize the power excursion which simulates the RIA condition[5].

In this paper, we introduce current status of the ongoing thermal hydraulic RIA experiments by KAERI. In these experiments, the transient heat transfer characteristics including boiling curves will be measured by adopting simple internal tube flow under simulated RIA conditions. Current experimental designs and major specifications of the test loop under construction are presented and the experimental plan is exhibited.

2. Description of the Test Facility

2.1 Test Loop

The RIA simulation tests will be conducted in the reactor coolant system thermal hydraulics loop (RCS loop) of the KAERI. The principal operating conditions of the RCS loop are as followings:

- Operating pressure: 0.5~16.0 MPa
- Test section flow rate: 0~0.3 kg/s
- Maximum water temperature: 340 ℃
- Available DC pulse power: 450 kW

Fig. 1 shows a schematic diagram of the modified RCS loop facility for RIA simulation tests. It consists of a circulation pump, preheater, RIA test section, steam/water separator, pressurizer, cooling lines, and feed water lines.

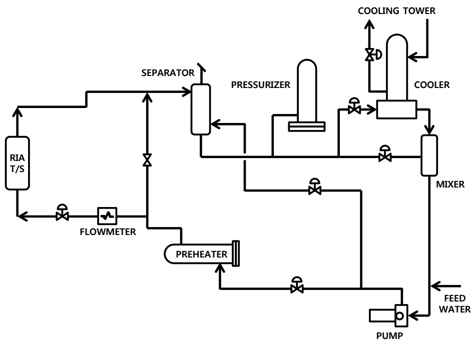

Fig. 1. Modified RCS thermal hydraulic loop for RIA tests.

2.2 DC Power Supply

A conventional direct current (DC) power supply will be used to generate a DC pulse which simulates sudden power excursion on the RIA test section. The pulse makes abrupt heat flux on the test section by direct

Joule heating. The power supply has maximum voltage of 75 V and current of 6000 A. To control the output voltage and the pulse width, the printed circuit board in the power supply is redesigned. The resulting pulse shape is similar to a step function due to an inherent power supply characteristic and the pulse width is designed to be adjustable in 16~330 ms considering the actual RIA conditions.

2.3 RIA Test Section

Fig. 2 shows a schematic diagram of the RIA test section. Only single tube is applicable to reproduce RIA conditions due to the limit of the power supply capacity. In addition, tube internal flow are adopted to simplify the bundle flows in the reactor core since its heated and hydraulic perimeters are the same. Inconel-600 tube is used for the test section to eliminate oxidation effect during the experiments.

The temperature, pressure, and flow velocity of the hot zero power (HZP) state of the PWR (APR1400), which can be the most severe RIA case, are about 291℃, 15.5 Mpa, and 5 m/s, respectively[6]. These conditions constitutes the principal experimental conditions of this study. Hence the outer diameter and the wall thickness of the test tube are selected considering allowable tube internal pressure along with the temperature. On the other hand, the tube heating length is limited by the available voltage and current ranges of the power supply. The bus-bars are spaced out 0.5 m apart considering the electrical resistance of the tube. Regarding the flow velocity, the heating length is enough for the pulse width which are less than 100 ms for the RIA heat transfer evaluation. The outer surface of the tube is thermally insulated to prevent the heat loss to air. Finally, the resulting temperature rise rate of the tube is estimated to be about 5000 K/s at the maximum.

2.4 Experimental Plan

This study started from 2017 and planned to 2021. Aims of the study are to evaluate previous heat transfer models in RIA conditions and to make improvement in the models which are applicable in generalized RIA applications. In 2017, a simplified pool boiling experimental facility is constructed and transient heat transfer tests are on the way under atmospheric pressure to analyze the effect of material oxidation, wettability, and properties in simulated RIA conditions[7]. From 2018, the high pressure flow boiling experimental facility for RIA conditions is under construction by modifying the previous RCS loop and power supply.

The flow boiling tests under RIA conditions are expected to start from the latter half of this year. The tests will be conducted under low pressure first and then the test pressure will be gradually increased to that of the HZP conditions. RIA flow boiling experimental data base for the low pressure conditions will be established at the end of 2019. The data base for the high pressure conditions are planned until 2021.

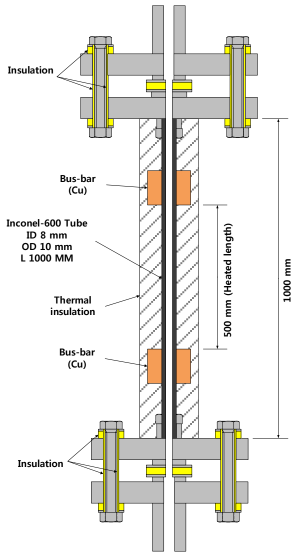

Fig. 2. Schematic diagram of the RIA test section.

3. Conclusions

RIA can result in disruption of the reactor due to a sudden increase in fission rate and reactor power. The current status of the KAERI's thermal hydraulic experiments on RIA are presented. Major facility design considerations, specifications, and test plan for this study are also described.

ACKNOWLEDGEMENT

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