Preliminary Test of Thermal-Hydraulic Experimental Facility for RIA Safety Study

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1. Introduction

Accidental insertion of reactivity to reactor core can cause a sudden increase in fission rate and reactor power leading to a reactivity initiated accident (RIA) [1]. In pressurized water reactors (PWRs), the control rod ejection accident due to mechanical failure of the control element assembly nozzle or control rod dive mechanism housing results in the coolant pressure to repel the control rod assembly completely out of the core. As a consequence, rapid increase of reactivity in the core causes a localized power excursion. 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].

Recently, test results from NSRR (Nuclear Safety Research Reactor, Japan) and Cabri research reactor (Cadarache, France) 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. However, most studies are focused on pre-DNB (Departure from Nucleate Boiling) failure mechanisms such as the PCMI (pellet-clad mechanical interaction) rather than the post-DNB phenomena. These result in a lack of knowledge related to the thermal hydraulic phenomena during the RIA [5].

In this paper, we introduce a recently constructed thermal hydraulic experimental facility for RIA safety study. The objective of this study is to characterize the fast-transient flow boiling phenomena for forced convection flow in a vertical tube which simulates the PWR RIA conditions for a wide range of pressure conditions. Current experimental designs and major specifications of the constructed test loop are introduced. Finally preliminary test results are presented and discussed.

2. Description of the Test Facility

2.1 Test Loop

The principal operating conditions of the transient flow boiling test facility for RIA are as followings:

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

Fig. 1 shows the schematic diagram of the RIA thermal-hydraulic test facility, THERA (Thermal-Hydraulic Experimental facility for RIA Applications). It consists of a circulation pump, preheater, RIA test section, steam/water separator, pressurizer, cooling lines, and feed water lines. Measuring variables are also indicated in the figure. The inlet flow rate, inlet/outlet fluid temperatures and pressures, tube surface temperatures, and voltage and current for analyzing the applied power are measured. Especially to measure the very fast transient surface temperature responses, high-speed infrared pyrometers having minimum response time of 0.5 ms are adopted. The applied voltage and current are also measured for every 1 ms using a high speed data acquisition system to analyze the fast variations in the applied power from the DC (Direct Current) pulsed power supply.

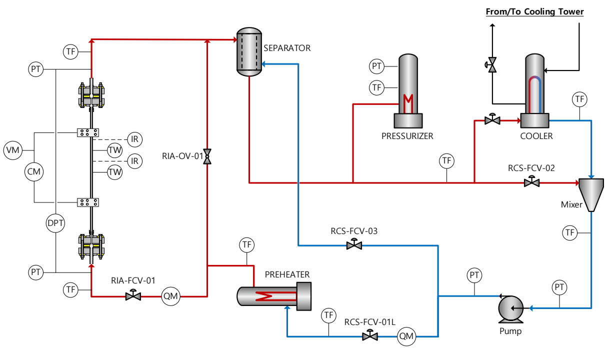

Fig. 1. Schematic diagram of the test facility

2.2 DC Power Supply

A conventional DC power supply is modified 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 capacity of 450 kW with maximum voltage and current of about 75 V and 6000 A, respectively. To control the output voltage and the pulse width, the power control circuits and logics of the power supply is modified. The resulting pulse shape is similar to a step function and the pulse width is adjustable within 20~330 ms considering the actual RIA conditions. The stepwise waveform is originated from the inherent power supply design and the pulse width is adjusted by cutting the number of passing cycles in the input AC voltage.

2.3 RIA Test Section

Water flows inside of vertical heated tube to reproduce the thermal hydraulic conditions of RIA. It can be regarded as a bundle flow in the reactor core since its heated and hydraulic perimeters are the same. Inconel-600 tube of 8 mm in inner and 10 mm in outer diameters is used in this study. The 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. Thus, the heated length of the tube becomes 0.5 m. The outer surface of the tube is thermally insulated to prevent the heat loss to air. The outer temperatures of the tube are measured at two positions of 0.43 m and 0.48 m from the beginning of the heated region through small holes in the insulator. During the power transient slight decrease of the tube conductivity due to the temperature increase results a small degradation in the heating power. However, the degradation is not significant since it only affects in the temperature rising behavior during a short duration and gives no impact on the overall heat transfer behavior.

3. Methods

3.1 Estimation of Heat Flux to Coolant

The heat flux from the tube inner wall to the coolant is estimated by inverse heat conduction calculation. The calculation is achieved in three steps: (1) frequency filtering of the measured signals to remove unavoidable noises, (2) inverse heat conduction calculation to get the temperature profile in the tube wall, and (3) extraction of heat flux to the water using the temperature gradient at the inner wall.

3.2 Test Conditions

Preliminary tests was conducted at low pressure conditions of around 10 bar. Mass flux of the flow was matched to that of normal operation value of PWR (~3600 kg/m^{2}s). Subcooling temperature of the inlet flow was adjusted within ~65 K. The applied power from the power supply was about 320 kW and the pulse widths were adjusted from 20 ms to 170 ms.

4. Results and Discussions

Fig. 2 shows the tube outer wall temperature measured by IR-pyrometer at the 0.49 m position from the beginning of the heated length. The temperature rise rate is about 3700 K/s. Occurrence of DNB is identified by the existence of film boiling after the peak temperature. In this experiment, DNBs are observed when the pulse widths are 87 ms and longer.

Typical inner wall temperature and the heat flux calculated from the inverse heat conduction calculation are shown in Fig. 3 for the pulse width of 170. The transition process of DNB, film boiling, and rewetting is remarkable in the temperature and heat flux profiles. The heat flux reaches to 11.5 MW/m^2 which is more than three times to the CHF (Critical Heat Flux) typical in steady flow boiling condition. At the end of power pulse, the temperature reaches to peak and then film boiling takes places. During the film boiling period, the temperature linearly decreases showing constant heat flux of about 1 MW/m^2. As the temperature decreases to around 500 K, rewetting heat transfer starts and the temperature drops rapidly. The peak rewetting heat flux reaches about 3 MW/m^2. Finally, the tube gradually cooled down to inlet temperature of the coolant by forced convection heat transfer.

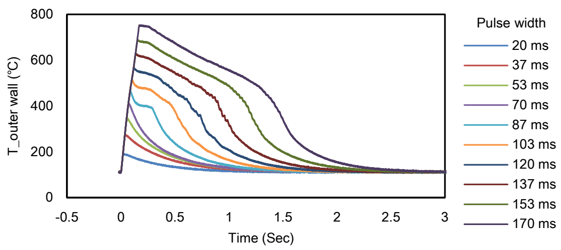

Fig. 2. Tube outer wall temperature response.

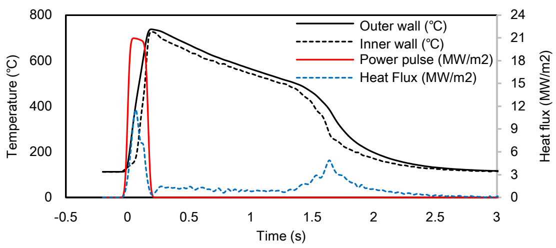

Fig. 3. Calculation of inner wall temperature and heat flux.

Table 1 shows the maximum outer wall temperatures and the corresponding heat fluxes as a function of the pulse width. First, as the maximum outer temperature increases, the heat flux also increases. However, when the pulse width exceeds 120 ms, the maximum heat flux is nearly saturated.

The CHF values reported by Bessiron et al. from the PATRICIA-PWR tests were about 4.2 to 5.9 MW/m^2 for system pressure of around 15 MPa [5]. Major difference of the present study and the PATRICIA-PWR test is the system pressure. In the PATRICIA-Pool tests which represents RIA in pool boiling conditions, it was reported that the CHF value significantly increases compared with the PWR pressure conditions [6]. The enhancement of heat flux in case of the fast wall temperature rise is explained by spontaneous nucleation at the surface cavities and the resulting latent heat absorption. Therefore, the CHF is higher in low pressure conditions since the vapor pressure of the water is low and the nucleation occurs favorably.

Table 1. Maximum wall temperature and heat flux

Pulse width (ms)	*87*	*103*	*120*	*137*	*153*	*170*
Max. T_{wall}(K)	424	486	542	607	671	734
Max. q'' (MW/m^2)	7.4	9.7	11.1	11.4	11.6	11.5

5. 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 preliminary test results are presented. The main tests are ongoing to investigate the effects of system pressure, flow rate, subcooling, and temperature rise rate under fast wall temperature rise conditions. The test results are expected to give important insights to the post-DNB phenomena for RIA safety study.

ACKNOWLEDGEMENT

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