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Experimental investigation on transient heat transfer in 2 × 2 bundle during depressurization from supercritical pressure

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ABSTRACT

The experimental investigation on transient heat transfer of supercritical water during depressurization in 2 × 2 bundle with wire wraps has been performed on the SWAMUP-II co-constructed by CGNPC and SJTU. The test section consists of two channels separated by a square steel assembly box with round corners. Water flows downward in the first channel and then turns upward in the second channel to cool the 2 × 2 bundle with wire wraps installed inside the assembly box. The bundle consists of four heater rods of 10 mm in O.D. arranged with pitch-to-diameter ratio of 1.18. Experimental parameter ranges cover pressure from 16 to 26 MPa, mass flux from 850 to 1450 kg/m^2 s, heat flux from 250 to 650 kW/m^2, inlet fluid temperature from 345 to 365 °C, and depressurization rate of 1 and 2 MPa/min. The experimental data are obtained and heat transfer characteristics during depressurization from supercritical to sub-critical conditions are discussed. The fraction of heat transfer from inner channel to outer channel is lower than 15% under the test conditions. Four different transient heat transfer phenomena during depressurization are observed. The boiling crisis is likely to occur in SCWR during depressurization as the operating condition of SCWR is compared with the test condition. The boiling crisis is less likely to occur with the increase of mass flux, and more likely to occur with the increase of heat flux or inlet fluid temperature. The wall temperature increment during boiling crisis is smaller with higher mass flux, and larger with higher heat flux or inlet fluid temperature. The depressurization rate affects the transient heat transfer itself trivially.

1. Introduction

Several concepts of SCWRs have been studied in Japan, Europe, Canada and China in the last 10–20 years to explore the future potential of water cooled nuclear reactors, aiming at higher efficiencies and lower specific plant erection costs than current PWRs or BWRs. Examples are the Super Light Water Reactor and the Super Fast Reactor concepts studied by Oka et al. (2010), the High Performance Light Water Reactor documented by Schulenberg and Starflinger (2012), the Canadian SCWR outlined by Yetisir et al. (2013), or A Chinese SCWR concept presented by Li et al. (2013). For these SCWR concepts, the coolant is assumed to be operated at well above the critical pressure of around 25 MPa, with a core inlet temperature of 280 °C and a core outlet temperature of 500 °C or more to improve the thermal efficiency. The surface heat flux of a fuel rod is more than 1500 kW/m^2 according to these concepts, and an average mass flux of ~1600 kg/m^2 s turned out to be sufficient to keep the peak cladding temperature below 650 °C (Schulenberg and Starflinger, 2012), as supercritical water is an excellent coolant. Operation at sub-critical pressure, i.e. below 22.064 MPa, is not considered in these concepts except for start-up, shut down or accidental conditions. In such cases, however, the low critical heat flux at near the critical pressure might cause a temporary boiling crisis with significantly higher cladding surface temperatures.

Research activities are ongoing worldwide to develop advanced nuclear power plants of SCWR type (Oka and Koshizuka, 2001; Schelenberg et al., 2008). Studies of thermal–hydraulic behavior of supercritical fluids have been performed since 1950s. The existing experimental and theoretical studies on heat transfer at supercritical pressure conditions were reviewed and published by several authors. Exhaustive literature search, carried out by Cheng & Schulenberg (2001) and Pioro and Duffey (2005), showed that the majority of experimental data were obtained in vertical tubes, while experimental investigations devoted to heat transfer in bundles cooled with supercritical fluid are very limited in open

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Nomenclature

C_p	specific heat at constant pressure (J/kg °C)
D	diameter (m)
e	error (–)
G	mass flux (kg/m^2 s)
H	specific enthalpy (J/kg)
I	current (A)
L	length (m)
Nu	Nusselt number (–)
M	mass flow rate (kg/s)
P	pressure (Pa)
Pr	Prandtl number (–)
Q	heat transfer per unit in vertical length (W/m)
q	heat flux (W/m^2)
q_v	volumetric heat flux (W/m^3)
R	radius (m)
S	perimeter (m)
T	temperature (°C)
U	voltage (V)
W	power (W)
z	axial location (m)

Greek symbols

α	heat transfer coefficient (W/m^2 °C)
δ	thickness (m)
η	efficiency of heat transfer (–)
λ	thermal conductivity (W/m °C)
μ	average deviation (–)
σ	standard deviation (–)

Subscripts and superscripts

b	bulk
bo	bottom
c	calculated value
e	experimental data
i, j	index
in	inlet
o	outer wall
out	outlet
pc	pseudo-critical
v	volume
w	wall

Acronyms

BWR	Boiling Water Reactor
CGNPC	China General Nuclear Power Corporation
ISO	International Standard Organization
LOCA	Loss of Coolant Accident
NI	National Instrument
NIST	National Institute of Standards and Technology
PWR	Pressurized Water Reactor
SCWR	Supercritical Water-Cooled Reactor
SJTU	Shanghai Jiao Tong University
SWAMUP-II	Supercritical Water Multipurpose Test Loop II

literature. Silin et al. (1993) reported on a large database for water flowing in large bundles at supercritical pressures, Razumovskiy et al. (2008, 2009) experimentally investigated the mean heat transfer of supercritical water in vertical 3-rod and 7-rod bundles. Li et al. (2015) carried out an experiment on the heat transfer of supercritical water in a 2 × 2 bundle of which the heat transfer enhancement caused by the grid spacers and a non-uniform circumferential wall temperature distribution were observed. Wang et al. (2014) performed heat transfer experiments with supercritical pressure water flowing upward in a 2 × 2 bundle without spacing device. They assessed eight selected correlations against the experimental data, and found that the correlations of Jackson and Hall (1979) and Jackson (2002) and Ornatsky et al. (1970) provided the best prediction accuracies. However, all these heat transfer researches mentioned in bundles are steady-state heat transfer at supercritical pressure.

A few researches have been published on the transient heat transfer during depressurization. A typical test with a boiler tube, showing these phenomena during depressurization, has been reported by Köhler and Hein (1986), using a vertical tube which was heated with a uniform heat flux of 619 kW/m^2 and cooled from inside with water with a mass flux of 2000 kg/m^2 s at an inlet temperature of 347 °C. The pressure, starting from supercritical pressure of 24.7 MPa, was reduced with ~30 kPa/s, and kept constant once a pressure of 19.1 MPa was reached. A boiling crisis with the cladding temperature suddenly increased from 375 to ~580 °C has been recorded. Aoki et al. (2006) reported a test to clarify the characteristics of the transient heat transfer during pressure reduction from supercritical to subcritical pressure using a 3-rod bundle test section, and measured heat transfer characteristics were compared with the data obtained using a single-tube test section. Kang and Chang (2009) has carried out the pressure transient heat transfer experiments in vertical tube using the Freon, HFC-134a as coolant. The pressures were varied from 3.8 to 4.5 MPa, the mass flux 600–2000 kg/m^2 s, the heat flux 10 and 140 kW/m^2, and the pressure transient rates 1.1–13.6 kPa/s. They found that the variations of heat transfer rates according to the pressure transient rates are trivial.

The SCWR core concept, reported by Li et al. (2013), employs a new type of closed assembly with double-row fuel rods in square geometry, showed in Fig. 1. The transient heat transfer test needs to be carried out in 2 × 2 bundle based on the core concept, as literature about the tests in this kind of channel are scarce. The effects of thermal-hydraulic parameters on the transient heat transfer during depressurization are to be analyzed. The experimental results are expected to provide supporting data for CFD and system codes validation.

2. Description of experiment

2.1. Experimental facility and measurements

Fig. 2 shows a schematic diagram of the facility SWAMUP-II coconstructed by CGNPC and SJTU, which is designed to perform heat transfer tests with supercritical water or steam-water two phase flows, and also with fast pressure variations. Distilled and deionized water from the water tank is driven through a filter by two high pressure plunger-type pumps with an operating pressure up to 35 MPa. The main flow goes through the re-heater to absorb the heat of the hot fluid coming from the test section, then passes the preheater where it is heated up to a pre-defined temperature and enters into the test section, and exits the test section with a high temperature up to 550 °C. The preheater is directly heated by AC power with a maximum heating capability of 600 kW. The test section is heated by DC power with a maximum heating capacity of 900 kW. Another flow is led through the bypass line to the