Experimental Thermal and Fluid Science 45 (2013) 169–179

Contents lists available at SciVerse ScienceDirect

Experimental Thermal and Fluid Science

journal homepage: www.elsevier.com/locate/etfs

Experimental investigation on heat transfer characteristics of high pressures water in a vertical upward annular channel

Han Wang, Qincheng Bi *, Zhendong Yang, Wu Gang, Richa Hu

State Key Laboratory of Multiphase Flow in Power Engineering, Xi'an Jiaotong University, Xi'an 710049, China

ARTICLE INFO

Article history:
Received 8 June 2012
Received in revised form 31 October 2012
Accepted 6 November 2012
Available online 16 November 2012

Keywords:
Heat transfer
High pressure water
Annular channel
DNB
Heat transfer correlation
Critical heat flux

ABSTRACT

Heat transfer experiments of high-temperature high-pressure water in a vertical upward annulus have been completed at Xi'an Jiaotong University. Particular attention was paid on heat transfer of water at sub-critical pressures with respect to the engineering application of nuclear reactor in this pressure region. Experimental parameters covered the pressures of 11–25 MPa, mass fluxes of 350–1000 kg/m^2 s and heat fluxes up to 600 kW/m^2. The gap of the annular flow channel was 6.0 mm with an effective heated length on the inner heated rod of 1400 mm. According to the experimental results, it was found that the increase of pressure and heat flux may lead to heat transfer deterioration in the steam-water two-phase region, whereas the increase of mass flux enhances heat transfer significantly. Two types of heat transfer deterioration, DNB and Dryout, were discussed and analyzed in detail. Based on the experimental data, heat transfer correlations were obtained to predict heat transfer in single-phase and two-phase regions. In addition, the present paper compared heat transfer difference of sub-critical pressure water in annulus and circular tube. At similar test parameters, annular channel has a better heat transfer performance in comparison with circular tube. Finally, heat transfer characteristics of water at sub-critical and super-critical pressures in the annulus were compared and discussed, which differed a lot at both normal and deteriorated heat transfer conditions.

1. Introduction

With the decreasing fossil fuel on the earth, nuclear energy power plants have gain an increasingly important part in the world electric power industry, which, on the other hand, puts forward a strict demand on the safety of the nuclear reactor core [1–3]. From the thermal hydraulic point of view, heat transfer in rod bundles are quite complicated and affected by various factors, such as the arrangement of the fuel rod, the physical property of the coolant and the turbulent mixing at a cross-section. In Boiling Water-cooled Reactor (BWR), two-phase boiling heat transfer is the main heat transfer mode in reactor core, while in Pressurized Water-cooled Reactor (PWR) and Heavy Water-cooled Reactor (HWR), local boiling heat transfer also exists in the primary circuit [4]. The prediction of heat transfer in the rod bundles of a nuclear reactor is of great importance to the reactor design and operation. At sub-critical pressures, heat transfer deterioration may occur with a sharp wall temperature rise, which poses a great threat to the safety of the reactor fuel bundles. Single-rod annular flow channel is representative of the rod bundles and the investigation on heat transfer of water in annulus contributes to understanding the heat transfer in bundles.

Heat transfer of two-phase flow has gained much attention since the 1960s and a lot of work has been done in this research field. Bringer and Smith [5] experimentally investigated heat transfer of carbon dioxide at near critical pressures in 4.57 mm I.D. smooth tube. They observed that heat transfer in near critical pressure region was very complicated and the conventional semi-theoretical correlation gave a maximum 30% error in predicting heat transfer coefficient. Swenson et al. [6] studied heat transfer of water in smooth tube and internally-ribbed tube. It was found that heat transfer deterioration in smooth tube occurred at a steam quality of 0.03, while the internally-ribbed tube did not see any deterioration before the steam quality reached 0.9 at the same test conditions. Nishikawa et al. [7] compared heat transfer of water in smooth tube and different internal-structure tubes. They concluded that B&W internally-ribbed tube has excellent heat transfer ability and the coefficients are four times higher than those of smooth tube. Hong et al. [8] experimentally studied heat transfer of R-134a with internally heated annulus at near critical pressures. The critical heat flux (CHF) and heat transfer for pressure deduction transients have been investigated. Uchida and Fujita [9] performed experiments of flow boiling in 1.6 mm-gap annulus with water at 0.05 MPa. They concluded that heat transfer was

* Corresponding author. Tel.: +86 29 8266 5287; fax: +86 29 8266 9033.
E-mail address: qcbi@mail.xjtu.edu.cn (Q. Bi).

0894-1777/$ - see front matter
http://dx.doi.org/10.1016/j.expthermflusci.2012.11.001

Nomenclature

A	area (m^2)
Bo	boiling number (–)
Co	convection number (–)
c_p	specific heat capacity (kJ/kg K)
d	outside diameter of the inner tube (m)
D	inside diameter of the outer tube (m)
G	mass flux (kg/m^2 s)
h	heat transfer coefficient (kW/m^2K)
H	bulk enthalpy (kJ/kg)
H_{fg}	latent heat of vaporization (kJ/kg)
L	heated length (m)
P	pressure (MPa)
Pr	Prandtl number (–)
q	heat flux (kW/m^2)
Re	Reynolds number (–)
t	temperature (°C)
T	temperature (K)
W	mass flow rate (kg/s)
x	equilibrium steam quality
Greek letters	
λ	thermal conductivity (W/m K)
μ	dynamic viscosity (Pa s)
ρ	density (kg/m^3)
σ	surface tension
subscripts	
b	bulk
in	inlet
pc	pseudo-critical
w	wall

enhanced by the bubbles in low mass velocity and heat flux range, whereas deteriorated heat transfer occurred at high heat flux. Wu et al. [10] researched CHF of water in bilaterally heated annuli at pressures of 0.6–4.2 MPa. They found that the CHF increases with mass flow velocity and the annular gap size, but decreases with the rising critical steam quality. Heat transfer characteristics of high-pressure working fluids have also been investigated by many other researchers [11–18]. The influencing factors of heat transfer, such as experimental parameter, flow geometry structure, flow direction and heating method have been investigated systematically.

From the literatures mentioned above, it is seen that although much work has been done on heat transfer at high pressures, the majority of past experiments were performed with circular tubes and internally-ribbed tubes. Among the available published literatures focused on heat transfer in non-circular flow channels, little research can be found on heat transfer in annuli using water as the working fluid at high pressures. In the nuclear reactor core, heat is generated by the fuel rods and transferred to the outer coolant, which is different from the heat transfer of tube flow. To get a fundamental understanding of heat transfer in fuel bundles, Xi'an Jiaotong University performed heat transfer experiments of water in a single-rod vertical flow channel over a wide range of flow conditions. The objective of this paper is to present the heat transfer results of high pressure water in annulus and contribute to the safe heat-transfer in nuclear reactors.

2. Experimental facility and procedure

2.1. Experimental system

The experiments were carried out in a high-temperature high-pressure steam-water test loop shown schematically in Fig. 1. Distilled and de-ionized feed water from the water tank was driven through a filter by a high pressure plunger-type pump with a maximum operating pressure of 40 MPa. Part of the water was returned to the water tank through a bypass and the rest part of the water flowed through measuring orifices and adjusting valves into a heat exchanger to absorb the heat of the hot fluid coming from the test section. Then this part of water was heated to the test state by the pre-heater and test section transformer, using electrical AC power supplies with maximum capacities of 760 kW and 250 kW, respectively. Heat transferred from the test section to the coolant was extracted with a regenerative heat exchanger and a condenser. The feed water was directed back to the water tank and recirculated. The pressure and mass flux in the test section were controlled by adjusting the main valve and bypass valve.

2.2. Test section

The geometry of the test section is shown in Fig. 2. The annular channel was made up of a heated $\varnothing 8 \times 1.5$ mm stainless steel (0Cr18Ni9) circular tube within a $\varnothing 25 \times 2.5$ mm circular tube, forming an annular gap of 6.0 mm and a hydraulic diameter of 12.0 mm. The length of the test section was 1400 mm and was electrically-heated by low-voltage AC current so that variable heat flux can be obtained. The outer pipe was unheated and was thermally insulated to minimize heat loss. As illustrated in Fig. 2, a sealing structure was used on the ends of the test section to guarantee the hermetic demand and ensure the electric insulation between the inner and outer pipes. A double sealing structure was performed for the test section with a flat seal structure adopted between Flange 1 and Flange 2 with a gasket seal. There was a stuffing box between Flange 1 and the inner pipe so that a packing seal can be obtained by squeezing the hold-down bolt to the stuffing graphite.

2.3. Parameter measurements

The thermocouple arrangement is shown in Fig. 3. The current design permits the use of six thermocouples to measure the wall temperature of the test section. The spiral spacer arranged on the inner heated pipe was a string of 6 mm long ceramic tubes with an outside diameter of 3 mm. One pitch of the spiral spacer was 50 mm and the full length along the heated rod was 100 mm. The fluid pressure at the inlet of the test section was measured with a Rosemount-3051 capacitance-type pressure transmitter. A Rosemount-3051 differential pressure transducer was used to measure the pressure drop over the test section. Fluid temperatures at the test section inlet and outlet were measured using $\varnothing 3$ mm K-type sheathed thermocouples inserted into the fluid channel. Wall temperatures along the test section were measured by 0.2 mm diameter standard K-type thermocouples attached on the inside wall of the inner heated pipe. All instrumentation signals were monitored and collected with a computer-based IMP3595 distributed data acquisition system.