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Experimental investigation on boiling heat transfer of high pressure water in a SCWR sub-channel

Han Wang [a], Weishu Wang [b,*], Qincheng Bi [c]

[a] School of Nuclear Science and Engineering, North China Electric Power University, Beijing 102206, PR China
[b] North China University of Water Resources and Electric Power, Zhengzhou 450011, PR China
[c] State Key Laboratory of Multiphase Flow in Power Engineering, Xi'an Jiaotong University, Xi'an 710049, PR China

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ABSTRACT

An experiment has recently been completed to obtain the wall temperature and heat transfer coefficient of water at subcritical pressures in a SCWR sub-channel. The test section was wire-electrode cut to simulate the central sub-channel of a 2×2 rod bundle. Experimental parameters covered the pressures of 11–19 MPa, mass fluxes of 700–1300 kg/m^{2}s and heat fluxes of 200–600 kW/m^2. Heat transfer characteristics in single-phase and two-phase regions were analyzed with respect to the variations of heat flux, system pressure and mass flux. For a given pressure, it was found that the wall temperature increases with increasing heat flux or decreasing mass flux in the steam-water two-phase region. Departure from Nucleate Boiling (DNB) was observed from the wall temperature profiles in the sub-channel. Experimental results showed that the soaring wall temperature at DNB becomes dramatic with the increase of pressure. Correlation assessments have also been conducted against the current set of experimental data. The comparisons indicated that the Fang correlation agrees well against the two-phase heat transfer coefficient. Heat transfer difference in the sub-channel at subcritical and supercritical pressures was compared. It was concluded that the wall temperature at sub-critical pressure may be lower or higher than that of supercritical pressure depending on q/G ratio and the occurrence of DNB.

1. Introduction

Two-phase flow boiling heat transfer of water has been widely-applied in industry, such as compact heat exchangers, reactor core of Boiling Water-cooled Reactor (BWR), steam generator, furnace of fossil-fired power plant etc. The critical pressure of water is 22.115 MPa, below which boiling occurs if the bulk temperature reaches the saturated temperature for a given pressure. It is recognized that two types of heat transfer deterioration, Departure from Nucleate Boiling (DNB) and Dryout, exist at subcritical pressure. DNB usually occurs at low steam-quality and high heat-flux conditions when the heat transfer transforms from nucleate boiling to film boiling, causing a rapid rise in wall temperature. Dryout occurs at high steam-quality conditions when the flow pattern transforms from an annular flow to a mist flow. The adherent liquid film is torn or evaporated due to acceleration of the core steam, which leads to heat transfer deterioration [1].

In recent years, the R&D of Supercritical Water-cooled Reactor (SCWR) becomes a hot point under the Generation-IV International Forum [2]. SCWR is a once-through water-cooled reactor operating at a pressure of 25 MPa, which reduces the capital and operational costs by eliminating the steam generator or steam-water separator in Light Water-cooled Reactors (LWRs) [3,4]. In addition, SCWR provides a thermal efficiency of about 45% which is higher than the 33% efficiency for the current LWRs [5]. Although SCWR operates at supercritical pressure, the reactor core is in subcritical pressure region during the load up and shutdown processes or in accidents [6]. At subcritical pressures, deteriorated heat transfer of DNB causes a sudden rise in cladding temperature. The issue is more complicated in SCWR where tight rod bundle (gap of 1.44 mm) was introduced in the conceptual design. These tight rod bundles may affect the thermal boundary development and the heat transfer to the coolant. Therefore, an improved understanding of heat transfer characteristics at relevant conditions is necessary for the design of SCWR core. It is the purpose of our ongoing research to experimentally investigate the flow boiling heat transfer of water in a SCWR sub-channel.

The experiments of flow boiling heat transfer in tubes have been performed by many researchers since the 1960s. Swenson et al. [7] investigated the heat transfer of water in smooth tube and internally-ribbed tube for vertical upward flow. It was found

* Corresponding author.
E-mail address: wangweishu@ncwu.edu.cn (W. Wang).

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Nomenclature

c_p	specific heat [J/kgK]
d	hydraulic diameter [mm]
G	mass flux [kg/m^{2}s]
h	heat transfer coefficient [kW/m^{2}K]
H	enthalpy [kJ/kg]
H_{fg}	latent heat of vaporization [kJ/kg]
K	temperature [K]
L	length of the test section [m]
Nu	Nusselt number [–]
P	pressure [MPa]
Pr	Prandtl number [–]
q	heat flux [kW/m^2]
Q	mass flow rate [kg/s]
Re	Reynolds number [–]
t	temperature [°C]
x	equilibrium steam quality [–]

Greek letters

λ	thermal conductivity [W/m·K]
μ	dynamic viscosity [Pa·s]
ρ	density [kg/m^3]
κ	electrical resistance [$\Omega \cdot$ m]
σ	surface tension [N/m]

subscripts

b	bulk
g	vapor phase
in	inlet
l	liquid phase
w	wall
wi	inner wall
wo	outer wall
sat	saturated
tp	two phase

that heat transfer deterioration occurred at a small steam quality of 0.03 in the smooth tube, while no remarkable wall temperature increase was observed in the internally-ribbed tube until the steam-quality reaches 0.9. The internally-ribbed tube postpones the occurrence of DNB to high steam quality compared to the smooth tube. Marek et al. [8] conducted an early experimental investigation on heat transfer and pressure drop in 3×3 and 4×4 rod bundles. Circumferential temperature distribution along the heater was obtained. In analogy with the pressure drop, a new correlation was deprived to predict the Nusselt number in rod bundles. Li and Hahne [9] experimentally research the boiling heat transfer on finned tub bundle with the working media of R11. They found that boiling heat transfer is strongly enhanced by the two-phase flow induced by the tubes, especially in the intermediate region between natural convection and fully developed boiling. Qu and Mudawar [10] investigated the flow boiling heat transfer of water in two-phase micro-channels. Experimental results showed that the boiling heat transfer coefficient is a strong function of mass flux, but only a weak function of heat flux. Eleven heat transfer correlations were assessed against the test data. Comparisons signified that none of the correlations gives satisfying predictions on two-phase heat transfer coefficient. Kumamaru et al. [11] and Koizumi et al. [12] from Japan Atomic Energy Research Institute performed a set of post-dryout heat transfer experiments with steam-water two-phase flow in a 5×5 rod bundle. Several heat transfer correlations were compared with the experimental data, which showed that the performance of classical empirical correlations in predicting heat transfer coefficient relates to the flow conditions. A new correlation was proposed by improving the Groeneveld correlation [13]. Anghel and Anglart [14] also experimentally studied the post-dryout heat transfer of high-pressure water in vertical tube and annuli. It was found that the flow obstacles affect the onset of dryout and post-dryout significantly. Predicted heat transfer coefficients by eight correlations are highly deviated from the test data due to the introduction of the flow obstacles. Over the past twenty years, researchers of Chen et al. [15], Pan et al. [16], Wang et al. [17] and Shen et al. [18] from the State Key Laboratory of Multiphase Flow in Power Engineering at Xi'an Jiaotong University performed a series of steam-water two-phase heat-transfer experiments at subcritical pressures and accumulated a huge number of experimental data. Remarkable achievements have been obtained in heat transfer characteristics, heat transfer mechanisms analyses and correlation assessments.

From the literatures mentioned above, it is seen that a lot of works have been done on boiling heat transfer of water at subcritical pressures. However, the majority of past experiments were performed with smooth tube or large-scale rod bundle, few publications could be seen on boiling heat transfer in sub-channel using water as the test fluid. In nuclear reactor core, heat is generated from the fuel rods and transferred to the outside coolant, which is different from the in-tube flow. Therefore, it is necessary to perform specific experiments in the flow geometry relevant to tight fuel bundle.

A Fuel Qualification Test (FQT) is being planned to irradiate a fuel bundle at supercritical pressure conditions inside the LVR-15 research reactor in Rez, Czech Republic [19]. The proposed fuel bundle consists of four fuel rods of 8 mm in outer diameter and a length of 600 mm. It is inserted into a square assembly box with rounded corners (see Fig. 1). Recent heat-transfer experimental data in the 2×2 rod bundle at supercritical pressure have been reported by Wang et al. [20,21] and Gu et al. [22,23]. However, research on the flow boiling heat transfer in relevant flow channel has not been seen in publication. The main purpose of this paper is to present the heat transfer characteristics of water in a SCWR-FQT sub-channel and improve the understanding on flow boiling in tight rod bundle.

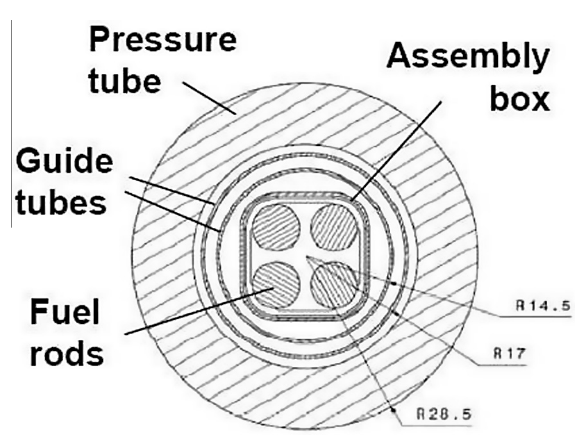

Fig. 1. Cross-sectional geometry of the fuel channel in the Fuel Qualification Test [19].