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Nonuniform heat transfer of supercritical water in a tight rod bundle – Assessment of correlations

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ABSTRACT

The heat transfer coefficient of supercritical water in a rod bundle is essential for the fuel design of the Supercritical Water-Cooled Reactor (SCWR). Although numerous correlations have been proposed over the past few decades to predict the heat transfer coefficient, the conclusions are inconsistent due to the limited experimental data obtained in fuel bundle. In the present paper, 20 correlations were assessed against the experimental data obtained in a tight 2 × 2 rod bundle. Circumferential maximum wall temperature and minimum heat transfer coefficient were selected as the benchmark data to get a conservative conclusion in support of the fuel design and safety analysis. The assessments showed that the performances of these correlations vary greatly depending on the mass flux and heat flux. Most correlations give reasonable Nusselt numbers at normal and enhanced heat transfer regimes, but become worse when the heat transfer is impaired. Comparison of these correlations against the total of 714 data indicated that the correlation proposed by Chen-Fang is the best with an average error of −0.44% and a standard deviation of 6.4%. All of the experimental Nusselt numbers were successfully predicted within ±20% error band.

1. Introduction

The concept of using supercritical water as the coolant in a nuclear reactor gained much attention in the last two decades. Several conceptual designs of the named Supercritical Water-cooled Reactor (SCWR) have been proposed for R&D worldwide by GIF-IV (US DOE, 2002). According to Oka and Koshizuka (2000), a SCWR power plant has the competitive advantages of high heat efficiency, low capital investment, simplified loop design and nuclear nonproliferation. Owing to the postcritical operating pressure and temperature, the heat efficiency of a SCWR is expected to be 45% compared to 33% of the current Pressurized Water Reactor (Su et al., 2014). Therefore, SCWR is recognized as the most promising water-cooled nuclear systems in the future.

At supercritical pressures, there is no steam-water transition and the coolant could be treated as single-phase fluid. Therefore, critical heat flux is eliminated fundamentally in the design criterion of SCWR concepts. Cladding temperature becomes one of the main concerns which requires an accurate prediction of the heat transfer coefficient in fuel assembly. However, heat-transfer data with supercritical water flowing in rod bundle are extremely scare, and consequently, tube-data-based heat transfer correlations are applied in system codes and subchannel codes in support of the SCWR fuel design and safety analysis. This treatment is based on the hypothesis that flow geometry has negligible effect on heat transfer of supercritical water. However, recent studies (Wang et al., 2014, 2016; Gu et al., 2015, 2016) showed that the cladding temperature along the fuel rod is nonuniform, which is quite different from the case of in-tube flow. Heat transfer coefficients predicted by tube-data-based correlations are not conservative to the fuel design of SCWR. Therefore, it is essential to assess these heat transfer correlations using experimental database accumulated with fuel-assembly flow geometry.

Predicting the heat transfer coefficient of supercritical water is challenging due to the strong and non-linear variation in its thermophysical properties. Near the pseudo-critical point, the density, thermal conductivity and viscosity fall dramatically whereas the specific heat experience a sharp peak. The heat transfer coefficient against temperature varies similarly to specific heat at normal heat transfer region, however, an inverse profile is observed at deteriorated heat transfer region, as shown in Fig. 1 (Xu, 2004). The enhanced or deteriorated heat transfer depends strongly on thermal parameters, such as mass flow rate or heat flux (Wang et al.,

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Nomenclature

c_p	specific heat [kJ/kg K]
$\bar{c}_p$	mean specific heat, $\frac{H_w - H_b}{T_w - T_b}$ [kJ/kg K]
d	outer diameter of the heated tube [mm]
D_{hy}	hydraulic equivalent diameter [mm]
D_k	local hydraulic diameter [mm]
G	cross-sectional average mass flux [kg/m^2 s]
Gr	Grashof number, $\frac{(\rho_b - \rho_w) g D_{hy}^3}{\rho \nu^2}$ [–]
$\overline{Gr}$	mean Grashof number, $\frac{(\rho_b - \bar{\rho}) g D_{hy}^3}{\rho \nu^2}$ [–]
Gr^*	Grashof number based on heat flux, $\frac{g \beta D_{hy}^4 q}{\lambda \nu^2}$ [–]
h	heat transfer coefficient [kW/m^2 K]
H	enthalpy [kJ/kg]
K	temperature [K]
L	effective heated length [m]
Nu	Nusselt number, $\frac{h \cdot D_{hy}}{\lambda}$ [–]
P	pressure [MPa]
Pr	Prandtl number, $\frac{\mu \cdot c_p}{\lambda}$ [–]
$\overline{Pr}$	mean Prandtl number, $\frac{\mu \cdot \bar{c}_p}{\lambda}$ [–]
q	heat flux [kW/m^2]
Re	Reynolds number, $\frac{G \cdot D_{hy}}{\mu}$ [–]
T	temperature [°C]
x	Distance from the test-section inlet [mm]

Greek letters

β	thermal expansion coefficient [1/K]
λ	thermal conductivity [W/mK]
μ	dynamic viscosity [Pa·s]
ρ	density [kg/m^3]
ν	kinematic viscosity [m^2/s]
ζ	frictional factor [–]

Subscripts

b	bulk
in	inlet
pc	pseudo-critical
w	wall

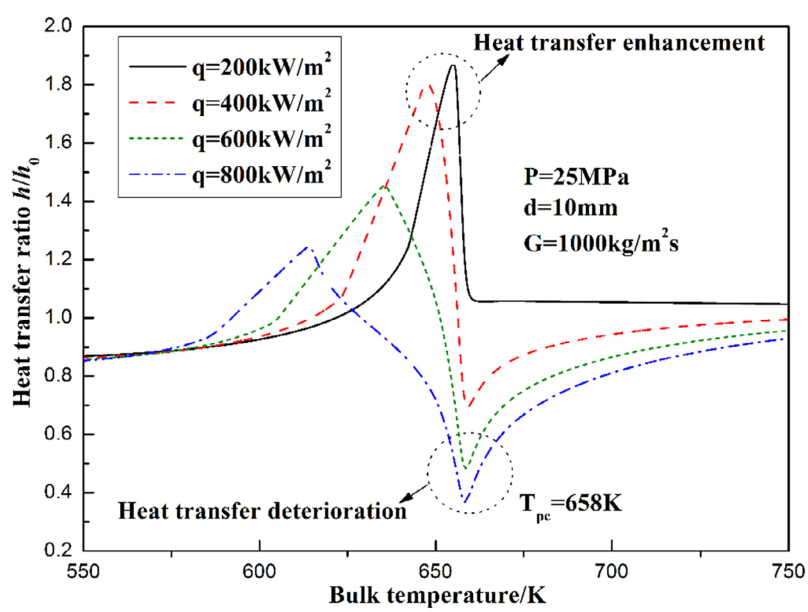

Fig. 1. Enhanced and deteriorated heat transfer at supercritical pressures (Xu, 2004).

2015). Most empirical correlations reasonably predict the heat transfer coefficient in low and high temperature regions, but fail to apply in the pseudo-critical region owing to the complicated variations in fluid properties.

A comprehensive review on the heat transfer of supercritical fluids was proposed by Pioro et al. (2004), which showed that the majority of correlations were developed using the experimental data obtained with tubes. Predicted heat transfer coefficients using these correlations show some distinctions for a given experimental parameter. Several research (Yu et al., 2009; Jager et al., 2011; Huang et al., 2012) reported that the correlation of Bishop et al. (1964) gives the most reasonable predictions against various tube-databases. Assessments of heat transfer correlation for supercritical fluids in subchannel, such as annuli or rod bundle, are seldom seen owing to the scarce of the experimental data. Yang et al. (2013) performed an experiment with supercritical water flowing in a narrow annulus. They found that the predicted Nusselt number by Swenson et al. (1965) agrees well with the experimental data. However, similar experiments conducted by Li et al. (2009) and Wu et al. (2011) indicated that the correlation of Jackson (2002) gives satisfying predictions in heat transfer coefficient. The reason for this discrepancy maybe lies in their enlarged annular gap-size and the introduction of wire-wrapped spacers compared with Yang et al. (2013). Bae (2011) believed that buoyancy plays an essential role to the heat transfer process of supercritical fluids. Assessment of correlations showed that the correlation of Watts and Chou (1982), which takes buoyancy into consideration, predicts the heat transfer coefficient best for supercritical CO_2 flowing in upward annulus.

As discussed above, consistent conclusion has not been drawn on which correlation provides reliable predictions in subchannel due to the limited data accumulated so far. An experiment with