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Critical heat flux experiments and a post-CHF heat transfer analysis using 2D inverse heat transfer☆

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ABSTRACT

The critical heat flux (CHF) is one of the main thermal-hydraulics safety limits in water-cooled reactors. At CHF, a film of vapor is formed on the heated wall and the wall temperature can sharply increase in a short period of time, which may lead to damage of the heater surface. In spite of many studies, the CHF mechanisms are still not well understood due to the complexity of this two-phase phenomenon and its dependence on local thermal-hydraulic conditions. Two-phase boiling experiments have been conducted to observe CHF and post-CHF behavior in prototypical conditions for small modular reactors (mass flux: 560–1570 kg/m^2.s, pressure: 7.6–16.4 MPa, inlet subcooling: 160–440 kJ/kg) in a 2 × 2 square rod bundle geometry with a chopped-cosine power profile at the University of Wisconsin-Madison. The fuel rods temperature history during and following CHF were used in the solution of the two-dimension inverse heat transfer problem to estimate the wall temperatures and wall heat fluxes. This analysis showed different transient boiling curves for different flow regimes that differ qualitatively and quantitatively from the typical boiling curve considered in steady-state two-phase heat transfer analysis. The experiments suggest that the inverse heat transfer analysis approach can be used to estimate the post-CHF heat transfer and to better understand the CHF mechanisms at high pressure typical of water-cooled reactors.

1. Introduction

The ability to transfer heat with a minimal temperature difference is a limiting factor in heat transport systems that operate at high heat fluxes. In many industrial applications, such as power plant technologies, the heat transport systems operate with two-phase vapor-liquid flows to optimize heat transfer and minimize temperature differences. The knowledge of the heat transfer mechanisms in different two-phase flow regimes is important to define safety margins during normal and abnormal operational conditions. In these two-phase heat transport systems, such as exist in the reactor core of light water reactors (LWR), the maximum operating power is restricted to a maximum heat flux between the liquid and the heated surface. This limiting condition is known as the critical heat flux (CHF) phenomenon.

At CHF, a film of vapor is formed on the heated wall and the wall temperature can sharply increase in a short period of time, which may lead to damage of the nuclear fuel rods. This phenomenon is also characterized by more mechanistic terminology known as film dryout or departure from nucleate boiling (DNB). The former term refers mostly to conditions found in boiling water reactors (BWR), where the fuel rod surface is completed dried out downstream of the CHF point without an annular liquid film present. The latter term refers to conditions found in pressurized water reactors (PWR), where vapor bubbles accumulate near the heated wall and coalesce into a film of vapor, thereby leading to a localized increase in the heater wall temperature.

The difference between the DNB and dryout is discussed by Kitto (1980) as a limiting quality phenomenon. The DNB occurs at lower quality levels while dryout occurs at higher qualities. Kitto (1980) defines a limiting quality region in which there is a considerable change in the critical heat flux for a small change in quality. The critical heat flux is predicted in thermal-hydraulic computational codes, such as TRACE (USNRC, 2013), and CTF (Salko and Avramova, 2015), by correlations from data gathered in either BWR or PWR conditions. One exception is the look up table 2006 (Groeneveld et al., 2007) that covers critical heat flux data in a large range and presents a limiting quality region. However, the thermal-hydraulic codes fail to relate the flow regime to the critical heat flux mechanisms. As shown in this paper and stated in Groeneveld et al. (2007), void fractions in the annular flow regime exist below the limiting quality. In fact, the annular flow above the limiting quality is defined by a very thin liquid film (Groeneveld et al., 2007)

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and the CHF mechanisms are not well understood despite of the many studies of critical heat flux (Carey, 2008). To understand the different mechanisms of vapor formation at different local qualities, new critical heat flux data were analyzed using two-dimensional inverse heat transfer.

The CHF experiments were performed at the high pressure heat transfer facility built to study the heat transfer at light water small modular reactor conditions, *i.e.*, low mass fluxes and moderate to high pressures (Greenwood et al., 2017). The flow channel was modified and new grid spacers were designed for the current work. A set of pre-CHF experiments in single-phase and two-phase flow were performed and compared to the subchannel code CTF in order to estimate the heat losses of the test section and the contact resistance of the thermocouples embed into the heater's walls. A new set of CHF experiments are presented in this paper. The modifications have shown an improvement of the CHF prediction when compared to the look up table 2006 (Groeneveld et al., 2007).

2. Experimental facility

The High Pressure Heat Transfer Facility (HPHTF) was designed to study heat transfer under a variety of pressures in single-phase, two-phase, and supercritical fluid flows (Greenwood et al., 2017). In this work, the facility was configured to examine two-phase flow conditions and critical heat flux under small modular reactor conditions. The facility (Fig. 1) consists in a test section (4) with a 2 × 2 rod bundle electrically heated and with a heated length of 2 m, a high pressure pump (1), a heat exchanger (5), an accumulator used to pressurize the system with argon gas (7), and two bypass valves (3, 6). The orifice flow meter (2) was calibrated to measure the inlet mass flow rate. Details of the loop can be found in Greenwood (2015) and Greenwood et al. (2017). The new test section flow channel and the spacer grids, which were modified for these experiments, are described in this section.

The rod diameter, pitch, and power shape are comparable to typical small modular reactor designs, such as NuScale Power (2016). Each rod has ten thermocouples (TC) located in different angular and axial positions, and a cosine axial power profile. The thermocouples are located between a clad and a sheath, both made from Monel K500. The clad and sheath thickness are 0.38 mm and 0.89 mm, respectively, and the rod diameter is 9.5 mm. The inlet and outlet temperatures are measured with thermocouples Type K with 1/8 in OD located between rods #1 and #4, and the bulk temperatures are measured at six axial positions with nine Type K TC 1/16 in OD. Fig. 2 shows the TC locations with respect to the power shape and spacer grids locations, and Table 1 shows the rod and square flow channel dimensions.

The new grid spacers (Fig. 3) were designed to provide support to the rods and minimally affect the flow, with no mixing vanes. This design is not typical of nuclear reactors but similar grid spacer designs have been used in related experimental studies (Xiong et al., 2015; Moon et al., 2005). The lateral walls are part of the flow channel and the inner walls are 0.79 mm thick. There are four grid spacers in the heated length with 12.70 mm height and two grids outside the heated length with 6.35 mm height.

The power is shaped by a helical resistance filament made from Inconel 718 with a varying pitch to be close conformance with the axial profile (Eq. (1)). The filament is filled with BN which is also used as insulating sleeves between the filament and the sheath.

$$q(x) = \theta_0 + \theta_1 \cos\left(2\cdot\theta_2\left(\frac{x}{HL} - \frac{1}{2}\right)\right) \tag{1}$$

where

$\theta_0 = 0.82,\ \ \theta_1 = 0.68,\ \ \theta_2 = 2.44$

3. Pre-CHF heat transfer analysis

Steady-state experiments at different mass fluxes, pressures, inlet temperatures, and powers were performed in order to characterize the facility. The experiments were simulated using CTF (Salko and Avramova, 2015). CTF is a two-fluid computational code with three fluids fields equations (liquid, vapor, and liquid drops). It solves the energy, mass, and transversal and axial momentum equations in 1D (with capability to develop 2D and 3D models). It simulates each subchannel and each rod separately, as shown in Fig. 4, and connects each subchannel using crossflow correlations for turbulent flow. A total of seventy-eight axial nodes were used for the CTF channel. Table 2 summarizes the main model choices used in CTF simulations.

The experimental range covers single-phase and two-phase flows under subcooled and saturation boiling regimes:

Pressure: 7.8–16.0 MPa
Mass flow: 520–2200 kg/m^2.s
Inlet temperature: 63–309 °C
Power per rod: 1–91 kW.

The calibration of the thermocouples was verified at zero power and high mass flux. At these conditions, it is expected that all thermocouples would indicate the same temperature. The thermocouples are ungrounded Type K (Ni-Cr, Ni-Al) and were purchased in different batches (inlet, outlet, bulk, and wall thermocouples). The manufacture temperature range of Type K TC is from 0 to 1250 °C with a tolerance of 1.1 °C or 0.4%, whichever is greater. The maximum deviation found was 0.9 °C for the temperature range 18–105 °C, which is within the tolerance limit. The maximum temperature of these experiments was

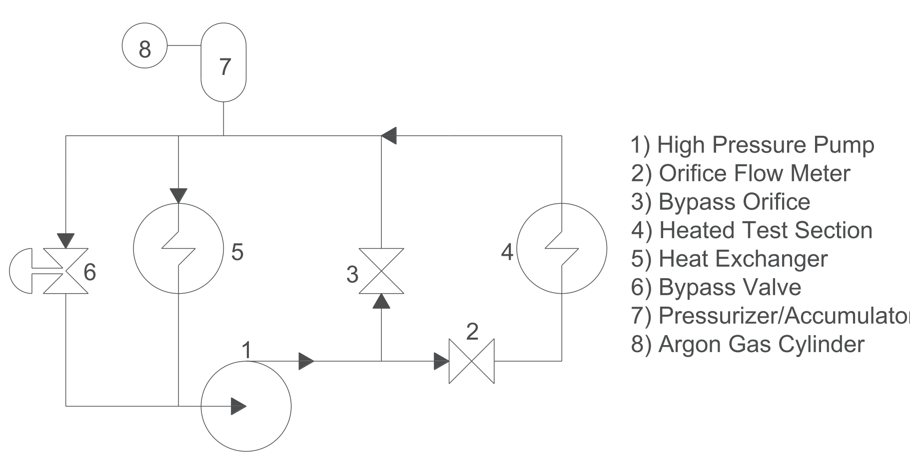

Fig. 1. Basic process flow diagram of the high pressure heat transfer facility (HPHTF) (Greenwood et al., 2017).