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Axial and radial void fraction measurements in convective boiling flows

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HIGHLIGHTS

- Axial and radial void fraction measurement for vertical forced convective boiling flow.
- γ- ray densitometry has been used for measuring void fraction distribution profiles.
- Drift flux model description used to rationalize the experimental observations.

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ABSTRACT

This work is inspired by the need to have measurements and predictive capability on vertical saturated boiling flows, which are of importance in boiling water nuclear reactors. In these systems, sub-cooled water flows upwards in vertical tubes which contain a multiplicity of nuclear fuel rods, and the heat is taken away from these rods by natural convection boiling. A close analog of this situation exists when the liquid is in forced flow, and the heat flux from the vertical rods is quenched by forced convection boiling of the upward liquid water flux.

In this work, we present experimental data has for axial, and radial vapor void fraction distributions in an annular boiling channel for low mass-flux forced the flow of water at high inlet sub-cooling. A single centrally placed electrical rod, designed to mimic the nuclear fuel rod has been used. The void fraction measurement is made using gamma ray densitometry technique. Axial and radial vapor void fractions have been reported, as a function of inlet liquid flux and inlet liquid temperature. The experimental data has been rationalized using a simple one-dimensional drift flux model adapted to the conditions of the experiment.

1. Introduction

Convective boiling flows are of great importance to nuclear reactor systems and have been the subject of numerous theoretical and experimental investigations (Todreas and Kazimi, 2012). Boiling occurs at a planar interface in contact with liquid when the temperature of the liquid is raised sufficiently beyond the saturation temperature at that pressure. Boiling may occur under a quiescent fluid condition, which is referred to as *pool boiling*; or under forced flow conditions, which is referred to as *forced convective boiling* (Collier and Thome, 1994). When boiling flow occurs under forced flow condition, heat transfer includes a contribution from both convective as well as from nucleate boiling (Collier and Thome, 1994). The process of flow boiling is most commonly affected inside vertical tubes, in horizontal tubes, in annuli, and on the outside of horizontal tube bundles. The local flow boiling heat transfer coefficient is primarily a function of vapor void fraction, mass flux, heat flux, flow channel geometry and orientation, two-phase flow pattern, and fluid properties.

The particular relevance of convective boiling to the nuclear industry is for applications in thermal hydraulics in boiling water reactors. In these applications, liquid phase water is brought in typically under sub-cooled conditions and made to flow around nuclear fuel pins (vertical rods), held concentrically within a large vertical cylindrical containing vessel. The fuel pins serve as the principal source of heat, usually with very high energy fluxes. The liquid water undergoes phase change in the vicinity of the heated fuel rods, even when other parts of the column may continue to be under sub-cooled or saturated liquid conditions. This kind of configuration leads to a differential distribution of vapor and liquid phases, with the vapor tending to segregate both radially and axially, as it forms along the vertical height of the vessel. Indeed, in turn this segregated vapor drives the liquid circulation, and in natural circulation boiling water reactors, is the sole cause of the flow of the two-phase mixture to occur. On the other

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hand, in forced circulation boiling, in which there is a default flow profile even when the energy flux from the heater rods is zero, this local segregation of vapor and liquid is significantly modulated both by the energy flux from the rods as well as the flow patterns as a result of the inlet hydrodynamic conditions as well as the geometry available for the flow and heat transfer to occur.

One of the main challenges in operating this kind of a reactor system are in the complexities of two-phase flow around the rods driven by a vertically distributed heat flux in the rods (Todreas and Kazimi, 2012). This is because the void fraction (vapor fraction) distribution significantly affects the reactor power and is one of the important parameters that determine the heat transfer capability and the possible occurrence of critical heat flux (Collier and Thome, 1994). The complex phenomenon of convective flow boiling can be divided into several regimes on a qualitative basis, depending on the local flow conditions, namely bubbly, slug, churn, annular, wispy annular and mist flow (Todreas and Kazimi, 2012). Bubbly flow regime is characterized by the vapor bubbles that are dispersed in the form of discrete bubbles in the continuous liquid phase. The shapes and sizes of the bubbles may vary widely, but they are notably smaller than the pipe diameter. Moving up the vertical tube, the vapor fraction increases, and slug flow regime is observed. In this regime, vapor bubbles collide and coalesce to form larger bubbles similar in size to the pipe diameter. Further moving up the containing tube, churn flow is observed, the flow becomes unstable, and the liquid travels up and down in a chaotic manner, although the net flow is directed upwards. Annular flow regime follows churn flow regime, where the bulk of the liquid flows as a thin film on the wall with the vapor as the continuous phase flowing up the center of the tube, forming a liquid annulus with a vapor core whose interface is disturbed by both large-magnitude waves and chaotic ripples. Annular flow regime is followed by wispy annular and mist flow, where the entrained droplets congregate to form large lumps or wisps of liquid in the central vapor core with a very disturbed annular liquid film. In mist flow regime, the annular liquid film becomes very thin, such that the shear of the vapor core on the interface is able to entrain all the liquid as droplets in the continuous vapor phase (this regime is inverse of the bubbly flow regime) (Todreas and Kazimi, 2012).

It is important to note that the above description is largely qualitative and based on photographic visualization of the flow fields. In general, the underlying flow physics, involving the nucleation of bubbles, their growth and departure, and possible coalescence as they meet other bubbles in the neighborhood, as well as re-condensation as they move to regions of relatively cooler liquid, is not totally understood and even more difficult to model from first principles. One of the important impediments to developing reliable models also has to do with the inability to have reliable *quantitative* experimental observations of flow variables such as local void fraction, local liquid velocity, and local temperature.

The present contribution relates to the measurement of local void fraction in such systems. The vapor void fraction inside a vertically heated tube varies axially as well as radially. The void fraction distribution in turn affects the liquid velocity distribution and hence is a characteristic feature of the prevalent flow regime. Void fraction distribution is dependent on the mass flux of liquid, inlet sub-cooling and heat flux of the heater. Even if the heat flux is independent of the elevation, as the liquid progressively vaporizes, the flow develops along the height of the boiling tube. Thus, it is pertinent to measure void fraction both along the height and along radial location in such a column. This information is crucial for providing validation data for thermal–hydraulic CFD codes, as well as for the design of nuclear safety systems based on more conventional methods.

Extensive research has been done in the area of sub-cooled boiling flow and comprehensive review of these works has been reported by Lee and Bankoff (1998) and Bartel et al. (2001). These reviews suggest that several researchers have attempted to measure void fraction in sub-cooled boiling flows; however all of them fall short on some aspect or the other. Roy et al. (1994) measured void fraction, gas velocity and bubble diameter in R-113 (refrigerant) boiling flows using dual sensor fiber optic probe, while Lee et al. (2002) measured void fraction distribution for water boiling flows using an intrusive conductivity probe. However, all these measurements were performed at certain axial positions only and hence no data on the axial development of local flow parameters have been reported. Situ et al. (2004) also measured void fraction distribution, interfacial area concentration, and interfacial liquid velocities using double sensor conductivity probe. Their experiments were conducted on sub-cooled flow boiling water and reported data on the axial development of void fraction, gas velocity, interfacial area concentration and interfacial velocity. They also validated constitutive equations of distribution parameters, drift velocity, and bubble Sauter mean diameter using their experimental data.

From the literature review, it become clear that most of the past work is limited to sub-cooled boiling and limited experimental conditions (there is almost no data on saturated boiling conditions, which are of greater relevance to the more modern nuclear reactor technologies). Further, it is limited to void fraction measurement at only a few axial locations, and that too with intrusive probes (which could potentially alter the flow significantly in these highly unstable boiling flows, hence making the measurements inaccurate). Additionally, the experimental results available in literature are mostly in the sub-cooled flow regime using invasive void fraction measurement techniques. Whereas, in present case the void fraction measurements are done at axial locations which are in subcooled as well as in saturated regime using densitometry, which is clearly non-invasive in nature. Additionally, the experimental conditions presented are for low flow rate forced convective boiling flows, whereas the results reported in literature are predominantly for high flow rate conditions.

In this contribution, measurements for radial and axial void fraction distribution are reported for vertical up-flowing boiling flows in a single cylindrical flow channel with a centrally located heating rod. All cases presented are for forced convection boiling, i.e., the liquid water is pumped into the system and flows independent of the boiling heat flux. For making the void fraction measurements, the gamma ray densitometry (GDT) technique has been used. GDT is a well-suited technique for measurement of void fraction non-invasively and has been extensively used for measuring void fractions in many non-boiling gas–liquid flows (Yadav et al., 2016), and in some flow boiling systems (Kok et al., 2001). The experimental data has been rationalized using the one-dimensional drift flux model.

2. Experimental

2.1. Boiling flow setup

The schematic diagram of the experimental setup is shown in Fig. 1. It consists of a tank for holding the water, preheater having a 1 kW heater for heating the liquid water to desired temperature (in the sub-boiling range), and a pump to deliver the water from tank to the inlet of glass column. The experiments were carried out in the vertical, concentric annular test section shown in Fig. 1. The outer tube of this section is a 75 mm inner diameter glass tube that allows for visual observation. The inner tube is a heating rod, which has an outside diameter of 40 mm with a heating length of 690 mm. The entire inner heater tube was connected to a 3 kW power supply. The inlet water temperature could be varied by changing the power of the heater fixed in the pre-heater tank. The applied heat flux was assumed to be constant and was calculated based on the electric power consumed by the heater rod and was verified by performing a heat balance across the test section for the case in which heat flux and inlet temperature were such that no bubbles were being formed at the