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International Journal of Heat and Mass Transfer 47 (2004) 413–417
International Journal of HEAT and MASS TRANSFER
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Technical Note

On void fraction distribution during two-phase boiling flow instability

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Received 17 July 2002; received in revised form 23 June 2003

1. Introduction

Flow instability is an important consideration in the design of nuclear reactors because of the possible occurrence of flow excursion during a postulated accident. The present investigation addresses the static flow instability in a heated channel. Static flow instability may arise in narrow cooling channels due to steam formation in the case of loss of coolant accidents. The formation of steam in a heated channel can have a significant effect on the overall pressure drop along the channel. For fully developed single-phase flow, at high coolant velocities, the pressure drop is comprised primarily of effects due to wall friction, associated with surface roughness condition, which decreases the velocity. With a reduction of the flow velocity, two-phase effects initially appear in a single-phase liquid at the point of incipient boiling (IB) and become more pronounced with the onset of significant voiding (OSV) as demonstrated in Fig. 1. With the presence of vapour in the bulk liquid during boiling, it causes an increase of frictional drag, and also acts to affect the pressure drop through acceleration and buoyancy effects. At some intermediate velocity, the increase in pressure drop due to boiling completely offsets the decrease in pressure drop due to channel frictional components. Further velocity reductions cause the pressure drop to rise which results in a minimum point, the onset of flow instability (OFI) point, in the pressure drop versus velocity curve (see Fig. 1). If parallel flow paths exist, such as in fuel element designs, this increase in pressure drop in one channel may cause flow to be diverted to alternate channels, destabilising the system, and resulting in excursive or Ledinegg [1] stability.

Extensive validation against experimental data covering a wide range of conditions with significant improvements incorporated within the computational fluid dynamics (CFD) two-phase boiling flow model in a generic computer code CFX4.4 has been recently performed in Tu and Yeoh [2] and Yeoh et al. [3]. Good agreement was achieved between the predicted and measured void fraction distributions. Implementation of the modified boiling model to RELAP5 has also yielded good agreement in particular the axial void fraction distribution (Yeoh and Tu [4]). We further extend our investigation in this study to demonstrate the feasibility of employing this boiling model to predict the pressure drop along an upward boiling flow in a vertical heated channel and its relationship with the void fraction distribution. An important consideration is the identification of the velocity at which the OFI point occurs and the capability of the model to predict this point is assessed. The locations of IB and OSV points can also be obtained through the void fraction distribution along the heated channel. These results are also reported.

2. Two-phase boiling flow model

In a two-fluid mechanistic model of subcooled flow boiling, both the vapour and liquid phases are treated as continua. The model solves two sets of conservation equations governing mass, momentum and energy. Since the macroscopic fields of one-phase are dependent on those of the other phase, closure relationships for the interaction terms coupling the transport of momentum, energy and mass of each phase across the interface are

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0017-9310/$ - see front matter
doi:10.1016/S0017-9310(03)00366-1

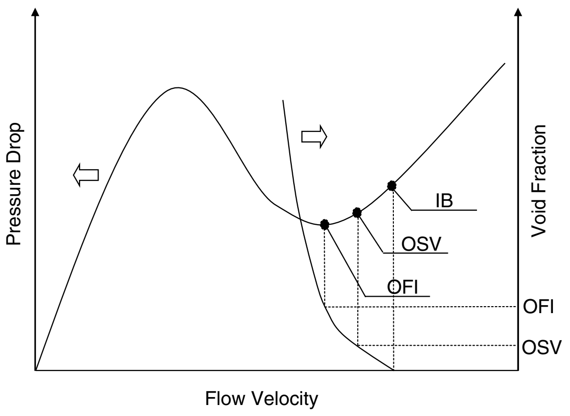

Fig. 1. Presence of IB, OSV and OFI along the pressure drop and void fraction versus flow velocity curves.

required. These terms such as inter-phase drag, heat and mass transfer terms in the field equations as well as the governing equations for the two-fluid model have been highlighted elsewhere (Tu and Yeoh [2]) and will not be repeated here.

The correlation developed by Zeitoun and Shoukri [5] is adopted for determining the mean bubble diameter because of its applicability for subcooled boiling flow at low-pressures. For the bubble departure diameter d_{bw} at the heated wall, the correlation of Kocamustafaogullari and Ishii [6] is employed: $d_{bw} = 0.0208\theta(\frac{\sigma}{g\Delta\rho})^{1/2}$ where $\Delta\rho = \rho_l - \rho_g$ and the equilibrium contact angle θ is taken to be at 80° as suggested in Rogers and Li [7].

We adopt the wall heat flux partition concept of Judd and Hwang [8], which are applicable for low-pressure subcooled boiling flow. The surface quenching heat flux is determined through the relationship: $Q_q = [\frac{2}{\sqrt{\pi}}\sqrt{k_l\rho_l C_{pl}\sqrt{f}}]A_q(T_w - T_l)$, where the bubble departure frequency of Kocamustafaogullari and Ishii [6] is given by: $f = \frac{1.18}{d_{bw}}[\frac{\sigma g\Delta\rho}{\rho_l^2}]^{0.25}$, and A_q is the fraction of wall area subjected to cooling by quenching calculated from: $A_q = n(\pi d_{bw}^2/4)$. The density of active nucleation sites n is obtained from Kocamustafaogullari and Ishii's [6] correlation of data: $n = \frac{1}{d_{bw}^2}[\frac{2\sigma T_{sat}}{(T_w - T_{sat})\rho_g h_{fg}}]^{-4.4} f(\rho^*)$, where $\rho^* = \Delta\rho/\rho_g$ and the function $f(\rho^*)$ is a known function of a density ratio described by: $f(\rho^*) = 2.157\times 10^{-7}\rho^{*-3.2}(1 + 0.0049\rho^*)^{4.13}$.

The heat transferred by evaporation or vapour generation is calculated from Bowring [9]: $Q_e = nf(\frac{\pi}{6}d_{bw}^3)\rho_g h_{fg}$ while the heat transferred by turbulent convection according to the definition of local Stanton number, St, is: $Q_c = St\rho_l C_{pl} u_l(T_w - T_l)(1 - A_q)$.

3. Results and discussion

Solution to the two sets of governing equations for the balance of mass, momentum and energy of each phase was sought. The conservation equations were discretised using the control volume technique. The velocity–pressure linkage was handled through the SIMPLE procedure. The discretised equations were solved using Stone's strongly implicit procedure. The two-fluid CFD boiling model predictions are compared against a series of comprehensive experiments performed at CEA-Grenoble in the vertical rectangular channel described in Fig. 2. The height, width and thickness of the channel flow are 600, 38 and 3.6 mm respectively. Since the wall heat flux was applied uniformly throughout the outer test section wall and taking advantage of the rectangular geometrical shape, only a quarter of the rectangular channel was considered as the domain for simulation. Uniform inlet velocity profiles were specified at the bottom wall while pressure boundary condition was employed at the top wall. Convergence was achieved within 4000 iterations when the mass residual dropped below a reasonably small value $\sim 1\times 10^{-7}$. More details regarding the experimental setup, instrumentation and measurement uncertainties are reported in the two CEA-Grenoble reports [10,11]. These experimental data were chosen because of the requirement to validate the model against the configuration of the eventual replacement of the present research reactor at ANSTO similar to those of the CEA test section geometry.

Fig. 3 shows the comparison of pressure drop versus the inlet velocity in the rectangular channel for a surface

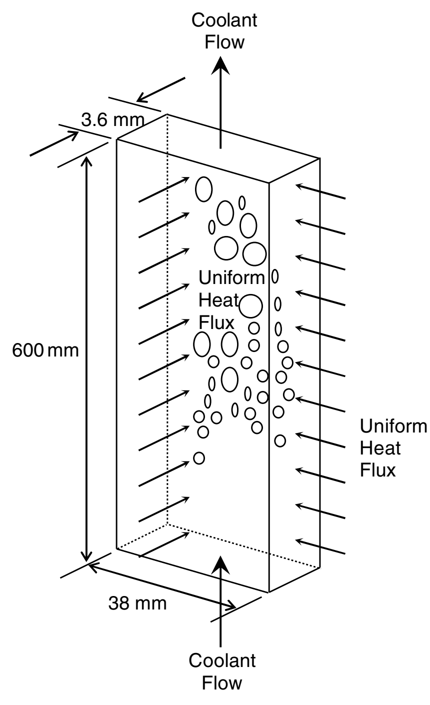

Fig. 2. Schematic illustration of an upward subcooled boling flow in a narrow rectangular channel.

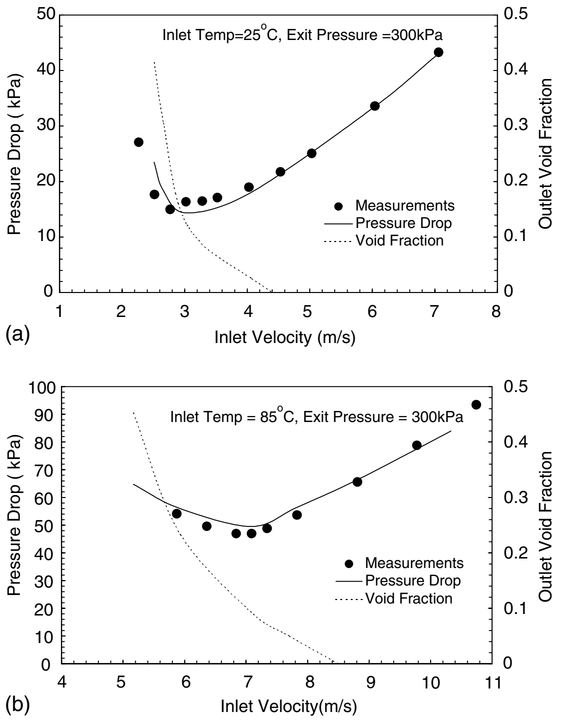

Fig. 3. Pressure drop and void fraction versus inlet flow velocity profiles for a heat flux of 3 MW/m^2 and exit pressure of 300 kPa with inlet subcooled temperatures: (a) 25 °C and (b) 85 °C.

heat flux of 3 MW/m^2 and exit pressure of 300 kPa at two inlet subcooling temperatures of 25 and 85 °C. Accompanying the pressure drop profiles in these results are also the outlet void fraction distributions for each of the subcooling temperatures. Excellent agreement between the experimental results and model prediction was achieved. At low inlet subcooling temperature, with reference to Fig. 1, IB was ascertained to occur around an inlet velocity of 4.4 m/s as indicated by the absence of any void fraction at the channel outlet. At high inlet subcooling temperature, as expected, IB occurred much sooner at an inlet velocity of 8.5 m/s. The predicted OFI points, corresponding to the minimum pressure drop versus inlet velocity curves, for inlet subcooling temperatures of 25 and 85 °C are 3.1 and 7.3 m/s respectively. These values are in very good agreement with experimental measured OFI points at 2.9 and 7.0 m/s for the same inlet subcooling temperatures.

Fig. 4 illustrates the sensitivity of the pressure drop and outlet void fraction distributions subjected to varying inlet temperatures and heat fluxes. The predicted distribution of the pressure drop at an inlet subcooling temperature of 45 °C and exit pressure of 300 kPa with three wall heat fluxes of 2, 3 and 4 MW/m^2 applied to the channel wall is shown in Fig. 4(a). The predicted OFI points corresponding to the minimum pressure drop versus inlet velocity curves, for the wall heat fluxes of 2, 3 and 4 MW/m^2 were found to be 2.1, 3.9 and 5.4 m/s, respectively. These values compared very well with the measured OFI points at 2.3, 3.6 and 5.2 m/s for the corresponding three wall heat fluxes in the experiments. As additional heat was applied to the test section, with reductions to the inlet velocity, the onset of static flow instability occurred more rapidly at higher inlet velocities. In Fig. 4(b), the pressure drop and outlet void fraction distributions at the OFI point against the inlet temperature are presented. The pressure drop increased with increasing inlet temperature as confirmed by Nair et al. [12]. Nevertheless, Fig. 4(a) and (b) results surprisingly showed that the void fraction distribution was insensitive to the varying wall heat fluxes as well as for the range of inlet subcooling temperatures between 25 and 85 °C. In both cases, the outlet void fraction at the point OFI yielded almost a constant value at around 10% void fraction at the channel exit. This interesting finding suggested that the onset flow instability occurred at this void fraction value could be treated as a guideline or criterion for nuclear safety analysis of static flow (Ledinegg) instability problems. Interestingly enough, we are also able to predict the criteria commonly used in many nuclear safety analyses for the OSV point at about 5% void fraction through our computational model of the pressure drop versus inlet velocity demand curve (see Fig. 3).

Wang et al. [13] has indicated four regions of dynamic instability in a forced-convection upflow boiling system. In this paper, we focused on the boiling phenomenon due to boiling onset oscillations within the OSV point in the rectangular channel. The two-phase boiling flow in the vicinity around the OFI point is rather unstable; the fluid flow and heat transfer from the heated wall fluctuates due to the bubble formation, growth and detachment from the wall. During our numerical simulations of the subcooled low-pressure boiling flow, the model revealed the possible presence of these oscillations as observed during the experiments. Fig. 4(c) and (d) present the axial distributions of the pressure drop and void fraction along the channel height. The results corresponded to an inlet subcooling temperature of 45 °C, exit pressure of 300 kPa and a wall heat flux of 3 MW/m^2. The profiles are recorded at three different situations during the reduction of inlet velocity—at OSV, OFI and the point after OFI. The increasing irregularity or waviness of the profiles for the axial pressure drop and axial void fraction at the OFI point as the boiling passes through the OSV regime suggested possible occurrence of the phenomenon due to boiling onset oscillations. The flow became stable again after it passed through beyond the OFI point as evidenced by the smooth axial profiles, which was also

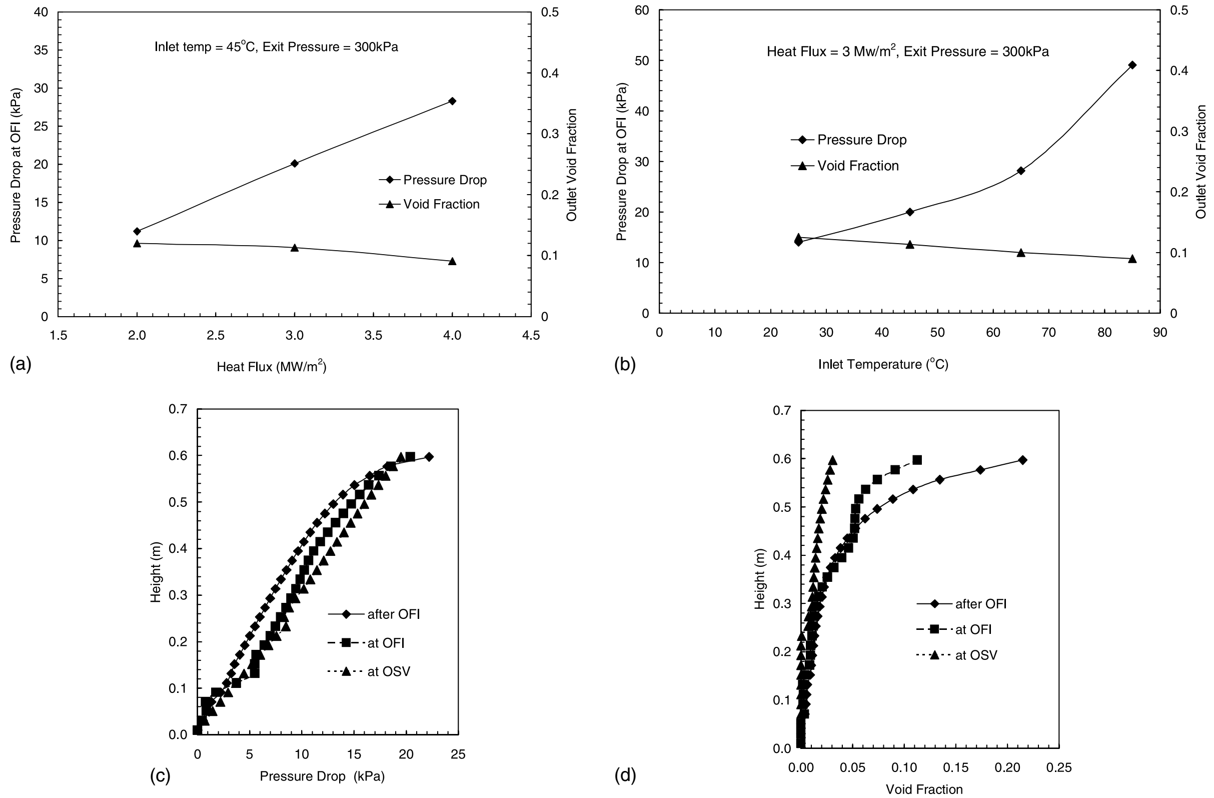

Fig. 4. Profiles of: (a) Both pressure drop and void fraction at OFI point against heat flux for inlet temperature of 45 °C and exit pressure of 300 kPa. (b) Both pressure drop and void fraction at OFI point against inlet temperature for a heat flux of 3 MW/m^2 and exit pressure of 300 kPa. (c) Local axial pressure drop distribution for an inlet subcooling temperature of 45 °C, exit pressure of 300 kPa and a wall heat flux of 3 MW/m^2 and (d) local axial void fraction distribution for an inlet subcooling temperature of 45 °C, exit pressure of 300 kPa and a wall heat flux of 3 MW/m^2.

similarly reported in Wang et al. [13]. Further work is in progress to further investigate the phenomenon due to boiling onset oscillations through the use of the current two-fluid CFD boiling model.

4. Conclusion

The presence of bubbles as a consequence of heating flow through a vertical channel has a significant effect on the overall pressure drop along the channel. A three-dimensional two-fluid CFD model is adopted to investigate the two-phase flow and heat transfer characteristics in the heated channel. Comparisons with experimental data for a variety of operating conditions show good agreement with the predicted results. The OFI velocity is accurately determined through the two-fluid CFD model and the predicted results of void fraction supplement a more fundamental understanding especially the occurrence of various points along the pressure drop versus the velocity curve regarding IB, OSV and OFI. At the OFI point, the current investigation suggests that the occurrence of onset flow instability at around 10% void fraction at the channel exit can be treated as a guideline or criterion for nuclear safety analysis of flow instability problems. The two-fluid CFD model also demonstrates the capability of possibly predicting the phenomenon of boiling onset oscillations within the OSV point as confirmed through Wang et al. [13] experiments.

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