NRTSC
NUCLEAR REACTOR TECHNOLOGY
AND SCIENTIFIC COMPUTATIONS

WSRC-TR-93-404

RETENTION:
PERMANENT

**A CRITICAL REVIEW OF PREDICTIVE MODELS FOR THE
ONSET OF SIGNIFICANT VOID IN FORCED-CONVECTION
SUBCOOLED BOILING (U)**

by
H. DORRA, S. C. LEE, S. G. BANKOFF

ISSUED: JUNE, 1993

D. Allan Cowles 8/9/93
Authorized Derivative Classifier/
Reviewing Official

SRTC

SAVANNAH RIVER TECHNOLOGY CENTER, AIKEN, SC 29808
Westinghouse Savannah River Corporation
Prepared for the U.S. Department of Energy under
Contract DE-AC09-89SR18035

MASTER

zp

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Document: WSRC-TR-93-404
Title: A CRITICAL REVIEW OF PREDICTIVE MODELS FOR
THE ONSET OF SIGNIFICANT VOID IN FORCED-
CONVECTION SUBCOOLED BOILING (U)
Contract: AX 811457
Task title:

Approvals

D. R. Muhlbauer, Responsible Manager	Date:
8/19/93
A. J. Garrett, Level 3 Manager	Date:

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

This report has been reproduced directly from the best available copy.

Available to DOE and DOE contractors from the Office of Scientific and Technical Information, P. O. Box 62, Oak Ridge, TN 37831; prices available from (615) 576-8401.

Available to the public from the National Technical Information Service, U. S. Department of Commerce, 5285 Port Royal Rd., Springfield, VA 22161

SRTC Perspective

D. A. Coutts

Senior Engineer

The coolant in the Savannah River Site (SRS) production nuclear reactor assemblies is circulated as a subcooled liquid under normal operating conditions. This coolant is evenly distributed throughout multiple annular flow channels with a uniform pressure profile across each coolant flow channel. During the postulated Loss of Coolant Accident (LOCA), which is initiated by a hypothetical guillotine pipe break, the coolant flow through the reactor assemblies is significantly reduced. The flow reduction and accompanying power reduction (after shutdown is initiated) occur in the first 1 to 2 seconds of the LOCA. This portion of the LOCA is referred to as the flow instability (FI) phase (1).

The reactor assemblies are subject to a type of FI known as a Ledinegg flow excursion. This type of instability is common in low pressure subcooled flow channels that exhibit a demand curve with a negative sloped region and a relative maximum and minimum as shown in Figure 1. The relative minimum is the result of the flow acceleration created by the change in fluid density, and an increase in the frictional pressure loss created by vapor nucleating on the heater surface and effectively increasing the flow channel roughness. The postulated LOCA can initiate such a flow excursion.

Assembly power limits are calculated for LOCA conditions using the FLOWTRAN-FI code. The code calculates a dimensionless ratio known as the Stanton number at each axial position in the channel through the course of the transient. This Stanton number was introduced by Saha and Zuber (2) to describe the onset of significant void (OSV) in channels where subcooled nucleate boiling is present.

$$(1) \quad St = \frac{\phi}{Gc_p(T_{sat} - T_b)}$$

where:

ϕ = local heat flux
 G = mass velocity
 c_p = liquid specific heat
 T_{sat} = saturation temperature
 T_b = bulk liquid temperature

The Stanton number will vary both with time and position in the assembly. The reactor operating power is limited to preclude Stanton numbers in excess of 0.00455 during the LOCA. This value is 70 percent of the 0.065 value proposed by Saha and Zuber (2). This difference is a safety margin that accounts for data and calculation uncertainties.

OSV and the Stanton number criterion are steady-state concepts. It is desirable to know the uncertainties associated with their application to a transient process. The Experimental Thermal-Hydraulics Group of the Nuclear Engineering Section contracted SGB Associates of Evanston, IL to study the transient FI process. This work has been completed. Table 1 presents a summary of the reports produced during this study.

References

1. *Action Plan to Upgrade Thermal-Hydraulic Limits for Savannah River Site Production Reactors (U)*. 30 June 1989. WSRC-RP-89-373.
2. Saha, F., and N. Zuber, 1974, Point of Net Vapor Generation and Vapor Void Fraction in Subcooled Boiling. In *Proceedings of the 5th International Heat Transfer Conference, Tokyo, Japan, September 3-7, 1974*, 175-79.

Figure 1. Demand curves for subcooled isothermal and diabatic systems

Table 1--Reports prepared as part of AX-1811457

Authors	Title	Document number
Bankoff, S. G., S. C. Lee, & A. Knaani	Literature Review on Forced-Convection Subcooled Boiling	WSRC-TR-91-98
Lee, S. C., & S. G. Bankoff	Stability of Forced Convection Subcooled Boiling	WSRC-TR-92-233
Dorra, H., S. C. Lee, and S. G. Bankoff	A Critical Review of Predictive Models for the Onset of Significant Void in Forced-Convection Subcooled Boiling	WSRC-TR-93-404
Lee, S. C., and S. G. Bankoff	Prediction of the Onset of Significant Void in Transient Subcooled Flow Boiling	WSRC-TR-93-405
Gehrke, Volker, & S. G. Bankoff	Stability of Forced-Convection Subcooled Boiling in Steady-State and Transient Annular Flow	WSRC-TR-93-406
Dorra, H., S. C. Lee, and S. G. Bankoff	Onset of Significant Void in Internal Flow, Forced Convection, Subcooled Boiling	WSRC-TR-93-407

A Critical Review of Predictive Models for the Onset of Significant Void in Forced-Convection Subcooled Boiling

H. Dorra, S. C. Lee¹ and S. G. Bankoff
Department of Chemical Engineering
Northwestern University
Evanston, IL 60208-3120

June 30, 1992

¹ Visiting Scholar

Permanent Address: Mechanical Engineering Department, Yeungnam University,
Gyongsan, Korea 712-749

ABSTRACT

The predictive models for the onset of significant void (OSV) in forced-convection subcooled boiling are reviewed and compared with extensive data. Three analytical models and seven empirical correlations are considered in this review. These models and correlations are put onto a common basis and are compared, again on a common basis, with a variety of data. The evaluation of their range of validity and applicability under various operating conditions are discussed. The results show that the correlations of Saha-Zuber (1974) seems to be the best model to predict OSV in vertical subcooled boiling flow.

KEYWORDS: Onset of Flow Instability, Onset of Significant Void, Forced-Convection Subcooled Boiling, Analytical Models, Empirical, Comparison

1. INTRODUCTION

A Ledinegg instability may occur in a heated channel when two-phase pressure loss exceeds the driving pressure difference between the channel inlet and outlet. This results in a sudden reduction of the flowrate, thus giving rise to significant vapor generation inside the channel. This, in turn, may cause channel wall overheating, leading to a dangerous burnout condition. This instability is commonly characterized by a minimum in the pressure drop versus flow rate curve, which is referred to as the onset of flow instability (OFI). Experimental evidence shows that the onset of significant void (OSV) generally signals OFI in a system with pressure-driven boundary condition. (Johnston, 1988; Dougherty, et al., 1990) This is because the sudden decrease in mean flow density results in an increased pressure gradient owing to the combined effects of friction and acceleration. With pressure boundary conditions, rather than specified inlet flowrates, this can lead rapidly to flow instability, and consequent dryout and overheating.

According to Collier (1981), OSV is a transition from partial boiling to fully-developed subcooled boiling, where the heating surface is covered with bubbles. Prior to OSV vapor bubbles cannot survive in the relatively cold core liquid. Prior to, and after, detachment of individual bubbles from the heating surface, a bubble boundary layer builds up at the wall. Thus, OSV may be regarded as the destabilization of this bubble boundary layer, principally by shear forces exerted by the core liquid.

The OSV phenomenon has attracted much attention since an early work of Bowring (1962). In recent years, interest in the OSV phenomenon has grown in conjunction with the safety analysis of nuclear reactors. In a particular reference loss-of-flow accident in a pressurized-water reactor, the flow through the channels decreases rapidly, owing to the decrease in the pressure difference across the channels resulting from the pipe break. The power also decreases, following the reactor trip. There is thus a race between the decreases in flowrate and power which determines the conditions within the channels, and ultimately the temperature of the fuel elements. Normally, the flow is all-liquid, but as the flow decreases, it is possible to have the onset of nucleate boiling (ONB). This corresponds to the appearance of the first bubbles on the walls. Since the bulk

of the liquid at the point of appearance is subcooled, the bubbles usually do not immediately detach from the wall. As the flow decreases further, the point of ONB moves backwards along the wall surface, resulting in a formation of a bubble boundary layer in the vicinity of the wall. The appearance of the bubble boundary layer causes the pressure gradient to increase, owing to the increased apparent roughness of the wall. At some point, owing to the detachment of the individual bubbles or the instability of the bubble boundary layer, there is significant vapor generation and mixing of bubbles with the core liquid. The onset of significant void precedes and is very close to OFI. Much work has been carried out in order to improve the understanding of the OSV phenomenon in connection with the nuclear safety analysis.

A number of investigators have attempted to derive the criteria for OSV in vertical, forced-convection subcooled boiling flow. However, a purely theoretical approach is very difficult, and most investigators have therefore set up simplified equations with extensive idealization and employ some empirical correlations for flow and heat transfer parameters. In general, they show that these analytical models agree well with their own experimental data or selected data in their range of validity. A number of small scale experiments have been performed in order to investigate OSV under various conditions and flow variables. These extensive experiments have produced a number of useful empirical correlations which are of practical importance.

Despite these efforts, the initiation mechanism of significant vapor generation is not well-understood as yet. Some discrepancies between models and the experimental data are found in the literature and significant scatter of data is also revealed from one investigator to another. It is the objective of this review to put these models onto a common basis and to compare them, again on a common basis, with a variety of data. The evaluation of their range of validity and applicability under various operating conditions will be also included in this paper.

2. A BRIEF REVIEW OF THE OSV MODELS

A number of predictive models for the onset of significant void (OSV) have been developed over the past thirty years. A few models have been derived from theoretical analyses, although some empirical adjustments or parameters were needed to fit the data. Examples of this category include the work of Levy (1967), Staub (1968), and Rogers, et al. (1987). Several empirical models have also emerged. In these models the OSV data have been correlated into simple equations, which include geometrical dimensions, fluid properties, and flow parameters, such as the mass velocity and the heat flux. The work of Bowring (1962), Saha and Zuber (1974), and Unal (1975) is notable among these models. These models will be introduced very briefly here.

2.1. Analytical models

Three analytical models can be found in the literature up to the present time. These models are mechanistic models for predicting the OSV point in vertical, upwards, subcooled flow boiling. Attention was focused in the models on the thermal and hydrodynamic conditions under which a bubble starts to depart from the heated surface. A force balance exerted on a single bubble was set up to establish a criterion for the onset of bubble departure. The balance has three components in vertical upflow, which are buoyancy and wall shear forces acting to detach the bubble, and the surface tension force tending to hold it on the wall. Note that the drag force was substituted for the wall shear force in the model of Rogers, et al. (1987), following Al-Hayes and Winterton (1981).

From the force balance, a dimensionless distance to the top of the bubble is calculated. It is argued in all three models that the liquid temperature at the top of the bubble should be at least equal to the saturation temperature when the bubble leaves. Incorporating the classical temperature profile for fully-developed, single-phase, turbulent flow obtained first by Martinelli (1947), the degree of liquid subcooling at OSV has been calculated.

These models show that the degree of liquid subcooling at OSV, ΔT_{OSV} , is in general expressed as follows:

$$\Delta T_{OSV} = \Delta T_{OSV}(q'', h, \rho_f, Pr, C_{pf}, \tau_w, Y_b^+)$$
(1)

where Y_b^+ is the dimensionless distance from the wall to the top of the bubble. Levy (1967) and Rogers, et al. (1987) used the Dittus-Boelter equation in calculating the heat transfer coefficient, h . However, the definition of Y_b^+ is different from one author to another. Levy (1967) derived from the force balance:

$$Y_b^+ = 0.015 \sqrt{\sigma D_b \rho_f} \frac{1}{\mu_f} \quad (2)$$

while Staub (1968) suggested the following equation:

$$Y_b^+ = \frac{D_b}{2} \frac{\rho_f v_f \sqrt{f/8}}{\mu_f} \quad (3)$$

Meanwhile, the following equation was derived by Rogers, et al. (1987):

$$Y_b^+ = \frac{D_b}{2} \frac{\rho_f}{\mu_f} \sqrt{\frac{\tau_w}{\rho_f}} (1 + \cos \theta) \quad (4)$$

These three models are very similar in nature. All use the same method for establishing a criterion for the onset of bubble departure and adopt the same condition for the temperature at the bubble top. However, it is interesting to note that different bubble shapes were assumed in the analyses from one author to another. Levy (1967) considered a full spherical bubble, but Staub (1968) postulated that the bubble shape at OSV would be a hemisphere. In the work of Rogers, et al. (1987), the bubble was treated as a truncated sphere, with a contact angle at the surface equal to the equilibrium contact angle.

2.2. Empirical Correlations

Due to the complicated nature of the subcooled nucleate boiling phenomenon, it is often convenient to predict OSV by means of empirical correlations. Several correlations by various investigators have been produced over the past thirty years. Seven correlations can be found in the literature up to the present time: Bowring (1962), Thom, et al. (1965), Ahmad (1970), Dix (1971), Saha and Zuber (1974), Sekoguchi, et al. (1974), and Unal (1975). According to these correlations, the degree of liquid subcooling at OSV may be expressed in terms of the following quantities:

$$\Delta T_{OSV} = \Delta T_{OSV}(P, D_b, q'', G, T_{fi}, \text{fluid - properties}) \quad (5)$$

The earliest work, that of Bowring (1962), includes the pressure term explicitly in the correlation, whereas the other correlations contain the effect of pressure only through the fluid properties. The inlet temperature is included only in the Ahmad correlation (1970). The correlations of Bowring (1962) and Thom, et al. (1965) were presented in dimensional form. On the other hand, the correlations of Saha and Zuber (1974), Unal (1975), Ahmad (1970), Sekoguchi, et al. (1974) and Dix (1971) were given in dimensionless form. Saha and Zuber (1974) suggested two dimensionless parameters, the Stanton number and the Nusselt number, based upon the argument that at low mass fluxes OSV is controlled by local thermal conditions, whereas at high mass fluxes it is hydrodynamically controlled. The remaining correlations show the ratio of the heat flux transferred to the liquid phase to the total heat flux at OSV. Unal (1975) proposed that this ratio has two different values for water, depending upon the liquid velocity. In the correlation by Dix (1971), this ratio is a function of the Reynolds number. In contrast, the Ahmad correlation (1970) uses the local heat transfer coefficient at OSV, as in the Dittus-Boelter type equation.

The correlations are introduced briefly below.

. Bowring (1962)

$$\Delta T_{OSV} = \eta \frac{q'' \rho_f}{G} \quad (6)$$

and

$$\eta \times 10^6 = 14.0 + 0.1 P \quad (7)$$

where P is the system pressure in bars and G , q'' , and ρ_f are expressed in SI units.

. Thom, et al. (1965)

$$\Delta T_{OSV} = 0.02 h_{fs} \frac{q''}{G h_{fs}} \quad (8)$$

. Ahmad (1970)

$$\Delta T_{OSV} = \frac{q''}{h_i} \quad (9)$$

where

$$\frac{h_l D_h}{k_f} = 2.44 \sqrt{Re} \left(\frac{C_{pf} \mu_f}{k_f} \right)^{\frac{1}{4}} \left(\frac{h_{in}}{h_{sat}} \right)^{\frac{1}{4}} \left(\frac{h_{fg}}{h_{sat}} \right)^{\frac{1}{4}} \quad (10)$$

. Dix (1971)

$$\Delta T_{Osv} = 0.00135 q'' \frac{Re^{0.5}}{h} \quad (11)$$

. Saha and Zuber (1974)

$$Nu = \frac{q'' D_h}{k_f \Delta T_{Osv}} = 455 \quad \text{for } Pe \leq 70\,000 \quad (12)$$

$$St = \frac{q''}{G C_{pf} \Delta T_{Osv}} = 0.0065 \quad \text{for } Pe > 70\,000 \quad (13)$$

. Sekoguchi, et al. (1974)

$$\Delta T_{Osv} = 13.5 \frac{h_{fg}}{C_{pf}} \left(\frac{q''}{h_{fg} G} \right)^{0.65} \quad (14)$$

. Unal (1975)

$$\Delta T_{Osv} \frac{h}{q''} = a = \text{const} \quad (15)$$

where

$$a = 0.24 \quad \text{for } V_f \geq 0.45 \text{ m/s}$$

$$a = 0.11 \quad \text{for } V_f < 0.45 \text{ m/s}$$

3. EXPERIMENTAL DATA

A number of experiments have been performed to study OSV or OFI over the past decades. In early work, experiments for relatively high pressures were carried out in connection with evaluation of the performance of high-pressure boiler system. (Egen, et al., 1957; Bartolemei and Chanturiya, 1967; Rouhani, 1968) Recently, interest in low-pressure data ($P \leq 5 \text{ bars}$) has grown in conjunction with the safety analysis of nuclear systems. The low-pressure data are summarized in Table 1. All data are for water.

Among these, the data of Whittle and Forgan (1967) and Dougherty, et al. (1990) were obtained in a systematic way for various combinations of geometrical and operating conditions. The data set of Whittle and Forgan (1967) encompasses the OFI data in vertical flow with rectangular ducts of four different sizes. The aspect ratio of the channels ranges between 7.9 and 18.2, so that the flow could be considered to be reasonably two-dimensional. The OFI data were determined, under various operating conditions, as a minimum point in the pressure drop-flowrate curve for constant heat flux. At a given geometrical condition, the OFI heat flux is presented, using the inlet temperature and the exit pressure as variables. Thus, these data can be regarded as the OSV data. Dougherty, et al. (1990) gives OFI data for vertical downwards flow in round tubes of six different diameters. The OFI data were determined by a similar method as by Whittle and Forgan (1967).

The other sets in Table 1 represent the OSV data in vertical upwards flow. All data were obtained by means of void fraction measurement either with electrical probes or by an optical technique. These sets include the data for circular tubes (Sekoguchi, et al., 1974; Edelman and Elias, 1981; Ferrell, 1964), rectangular channels (Staub, et al., 1967) and annuli (Evangelisti and Lupoli, 1969; Rogers, et al., 1987).

4. COMPARISON AND DISCUSSION

Comparisons of analytical models of Levy (1967), Staub (1968) and Rogers et al. (1987) with the low-pressure data are shown in Figures 1 - 3, respectively. The legends are indicated in Figure 1 and are identical for Figures 1 - 9. It is shown that the model of Staub (1968) (Figure 2) appears to be in good agreement with the data of Dougherty et al. (1990), Whittle and Forgan (1967), Staub et al. (1967), and Ferrell (1964). These data reflect moderate and relatively high heat flux conditions. The model of Levy (1967) (Figure 1) underpredicts ΔT_{OSV} for most of the data. However, the data scatter does not appear to be significant, except for the data of Sekoguchi et al. (1974), as shown in Figure 1. The Levy model seems to be reasonable if some modifications could be made, because it has no preference for any particular data set. It should be mentioned that the data of Sekoguchi et al. (1974) do not seem to be reliable because they deviate considerably from the values predicted by all other models and correlations, as seen in Figures 1 - 9. In Figure 3, a wide spread of the experimental data is indicated and a substantial difference between the data and the prediction by Rogers et al (1987) can be found. The model of Rogers et al (1987) originally aims at prediction of OSV at low velocities in upflow subcooled nucleate boiling. The range of the liquid velocity of their concern was less than 1.0 (m/s). In this region, the buoyancy effect on the mechanism of bubble departure is thought to be important. (Lee and Bankoff, 1992) This may be the main reason of the large discrepancies between the model of Rogers et al. (1987) and the experimental data, most of which have velocities larger than 1.0 m/s.

The comparisons between empirical correlations and the identical data sets are shown in Figures 4 - 9. In these comparisons, the correlation of Thom et al. (1965) was excluded because of its significant underestimates of ΔT_{OSV} . The correlations of Saha and Zuber(1974) and Unal (1975) generally seem to be in good agreement with the data. The Saha-Zuber correlation is very simple and convenient, and has been widely used in computer codes for nuclear safety analysis. The Saha-Zuber correlation is based on Peclet number data up to 400,000. However, it was pointed out recently by Lee and Bankoff (1992) that the use of the correlation beyond this limit

may be questionable. The Bowring correlation (1962) (Figure 4) tends to underpredict ΔT_{OSV} for most of data whereas the Dix correlation (1971) (Figure 6) has an opposite trend. The Bowring correlation (1962) was originally developed based on the high-pressure data ranging between 11 and 138 (bars). It includes an empirical constant to consider the effect of pressure as seen in Equation (6). Thus, it may be said that the Bowring correlation (1962) can not be applied to low-pressure regions below the lower limit indicated by the author. On the other hand, the Dix correlation (1971) was suggested based on the R-114 data. From the comparison with the water data here, one may note that it is not preferable to apply the correlation to fluid mediums other than R-114, although the correlation is given in a dimensionless form. The Ahmad correlation (1970) seems to agree well with the data of low ΔT_{OSV} . The Ahmad correlation (1970) is the only one to include the effect of the liquid inlet enthalpy explicitly. The correlation of Sekoguchi et al. (1974) (Figure 8) fits well with their data only.

To evaluate each model on a common basis, the standard deviation, defined by the following equation, was calculated.

$$\sigma_s = \sqrt{\frac{1}{N} \sum_{i=1}^N [\Delta T_{OSV}(\text{measured})_i - \Delta T_{OSV}(\text{predicted})_i]^2} \quad (16)$$

The results show that the Saha-Zuber correlation (1974) has the lowest value of σ_s , 6.55 among the empirical correlations. For analytical models, the Levy model (1967) indicates that $\sigma_s = 9.83$, whereas the Staub model has 10.39.

Parametric effects on OSV will be discussed from the results of comparisons between selected models and the data of Dougherty et al. (1990) and Whittle and Forgan (1967). The models selected here for comparisons are those of Levy (1967), Staub (1968), Rogers et al. (1987), Bowring (1962), Saha and Zuber (1974), and Unal (1975). Figures 10 and 11 show comparisons of these models with the data of Dougherty et al. (1990). The tube diameters of the data are 0.0255 and 0.0284 (m) but the diameter was taken to be 0.027 (m), an arithmetic mean of two diameters, in evaluating the models. The operating pressures are 2.4 and 4.4 (bar), respectively. All models and the data predict that $\Delta T_{OSV}/q''$ simply decreases as the mass velocity is increased.

Among these models, the correlations of Saha and Zuber (1974) and Unal (1975) are the closest ones to the data. Similar trends are indicated in Figures 12 and 13, which show the data of Dougherty et al. (1990) for $D = 0.0091(m)$. The effect of pressure can be inferred from Figures 10 -13. It has a tendency to increase $\Delta T_{OSV}/q''$ at fixed mass velocities. This result coincides with the Bowring correlation which is the one to consider the effect of the pressure explicitly. Figures 14 and 15 show comparisons of the selected models with the data of Whittle and Forgan (1967). The prediction by the correlations of Unal (1975) and Saha and Zuber (1974) fits well with the data.

5. SUMMARY AND CONCLUSIONS

The predictive models for the onset of significant void (OSV) in forced-convection subcooled boiling have been reviewed and compared with extensive data. Analytical models of Levy (1974), Staub (1968) and Rogers, et al. (1987) and empirical correlations of Bowring (1962), Thom, et al. (1965), Ahmad (1970), Dix (1971), Saha and Zuber (1974), Sekoguchi, et al. (1974) and Unal (1975) are included in this review. A brief introduction of each model was made. These models and correlations were compared to the extensive data at low pressures. The results show that the Saha-Zuber correlation (1974) is the best one to fit with whole sets of data among the correlations and the model of Levy (1967) among the analytical models. Selected models were compared with the data of Dougherty et al (1990) and Whittle and Forgan (1967) to investigate parametric effects on OSV. Both the models and the data predicted that $\Delta T_{OSV}/q''$ simply decreases as the mass velocity is increased. Among these models, the correlations of Saha and Zuber (1974) and Unal (1975) were the closest ones to the data. The pressure has a tendency to increase $\Delta T_{OSV}/q''$ at fixed mass velocities. This result coincides with the Bowring correlation which is the one to consider the effect of the pressure explicitly.

ACKNOWLEDGMENT

This work was supported by Westinghouse Savannah River Company. We thank the project monitor, B. S. Johnston for helpful comments. S. C. Lee received partial support from the Ministry of Education, Republic of Korea.

NOMENCLATURE

$C_{p,f}$	heat capacity ($\frac{J}{kg K}$)
D	tube diameter (m)
D_b	bubble departure diameter (m)
D_h	tube hydraulic diameter (m)
D_i	inner diameter (m)
D_o	outer diameter (m)
f	friction factor
G	mass velocity ($\frac{kg}{m^2 s}$)
g	acceleration due to gravity (m/s^2)
h	heat transfer coefficient ($\frac{W}{m^2 K}$)
h_{fg}	latent heat of vaporization ($\frac{J}{kg}$)
h_{in}	inlet fluid enthalpy ($\frac{J}{kg}$)
h_{sat}	liquid saturation enthalpy ($\frac{J}{kg}$)
k_f	thermal conductivity ($\frac{W}{m K}$)
N	the number of data
Nu	Nusselt number
P	pressure (bar)
Pe	Peclet number
Pr	Prandtl number
q''	heat flux ($\frac{W}{m^2}$)
Re	Reynolds number
St	Stanton number
T_f	fluid temperature ($^{\circ}C$)
$T_{f,in}$	inlet fluid temperature ($^{\circ}C$)

$T_{f,osv}$ fluid temperature at OSV ($^{\circ}C$)
 T_{sat} saturation temperature ($^{\circ}C$)
 ΔT_{osv} degree of subcooling at OSV, $T_{sat} - T_{f,osv}$ (K)
 V_f fluid velocity ($\frac{m}{s}$)
 Y_b^+ dimensionless distance from the wall to the bubble tip

Greek Symbols

θ contact angle
 μ_f fluid viscosity ($\frac{N}{m^2}$)
 ν kinematic viscosity ($\frac{m^2}{s}$)
 ρ_f fluid density ($\frac{kg}{m^3}$)
 ρ_g gas density ($\frac{kg}{m^3}$)
 σ surface tension ($\frac{N}{m}$)
 τ_w wall shear stress ($\frac{N}{m^2}$)

REFERENCES

- Ahmad, S. Y., Axial distribution of bulk temperature and void fraction in a heated channel with inlet subcooling, *J. Heat Transfer*, Vol.92, 1970, pp.595-609.
- Al-Hayes, R. A. M. and Winterton, R. H. S., Bubble diameter on detachment in flowing liquids, *Int. J. Heat Mass Transfer*, Vol.24, 1981, pp.223-230
- Bartolomei, C. C. and Chanturiya, V. M., Experimental study of true void fraction when boiling subcooled water in vertical tubes, *Thermal Engineering*, Vol.14, 1963, pp.123-128.
- Bowring, R. W., Physical model based on bubble detachment and calculation of steam voidage in the subcooled region of a heated channel, Institutt for Atomenergi, Halden, Norway, 1962, Report No. HPR-10.
- Collier, J. G., *Convective boiling and condensation*, McGraw-Hill, New York, Chapter 6, 1981, pp.178-205.
- Dix, G. E., Vapor void fraction for forced convection with subcooled boiling at low flow rates, Ph. D. Thesis, University of California, Berkeley, CA, 1971.
- Dougherty, T., Fighetti, C., Reddt, G., Yang, B., Jafri, T., McAssey, E. and Qureshi, Z., Flow boiling in vertical down-flow, *Proceedings of 10th Int. Heat Transfer Conf.*, Vol. 6, 1990, pp.9-14.
- Edelman, Z. and Elias, E., Void fraction distribution in low flow rate subcooled boiling, *Nuclear Engineering and Design*, Vol.66, 1981, pp.375-382.
- Egen, R. A., Dingee, D. A. and Chastain, J. W., Vapor formation and behavior in boiling heat transfer, 1957, Report No. BMI-1163.
- Evangelisti, R. and Lupoli, P., The void fraction in an annulus channel at atmospheric pressure, *Int. J. Heat Mass Transfer*, Vol.12, 1969, pp.699-711.
- Ferrell, J. K., A study of convection boiling inside channels, Ph. D. Thesis, Department of Chemical Engineering, North Carolina State University, Raleigh, NC, 1964.
- Johnston, B. S., Subcooled boiling of downward flow in a vertical annulus, Savannah River Laboratory, Aiken, SC, 1988, Report No. DPST-88-891.
- Lee, S. C. and Bankoff, S. G., Prediction of the onset of significant void in downflow subcooled nucleate boiling, to be presented at the Symposium on Basic Aspects of Two-Phase Flow and Heat Transfer, National Heat Transfer Conference, San Diego, CA , August 9 -12, 1992.
- Levy, S., Forced convection subcooled boiling: prediction of vapor volumetric fraction, *Int. J. Heat Mass Transfer*, Vol. 10, 1967, pp.951-965.
- Martinelli, R. C., Heat transfer to molten metals, *Trans. ASME*, Vol.69, 1947, pp.941-947.

Rogers, J. T., Salcudean, M., Abdullah, Z., Mcleod, D. and Poirier, D., The onset of significant void in up-flow boiling of water at low pressure and velocities, *Int. J. Heat Mass Transfer*, Vol.30, 1987, pp.2247-2260.

Rouhani, S. Z., Calculation of steam volume fraction in subcooled boiling, *J. Heat Transfer*, Vol.90, 1968, pp.158-164.

Saha, P. and Zuber, N., Point of net vapor generation and vapor void fraction in subcooled boiling, *Proceedings of Fifth International Heat Transfer Conference*, Vol.4, 1974, pp.175-179.

Sekoguchi et al., Flow boiling in subcooled and low quality regions heat transfer and local void fraction, *Proceedings of Fifth International Heat Transfer Conference*, Vol.4, 1974, pp.180-184.

Staub, F. W., The void fraction in subcooled boiling: prediction of the initial point of net vapor generation. *J. Heat Transfer*, Vol.90, 1968, pp.151-157.

Staub, F. W., Walmet, G. E. and Niemi, R. O., Heat transfer and hydraulics-the effect of subcooled voids, General Electric Co., 1967, Report No. NYO-3679-8.

Unal, H. C., Determination of the initial point of net vapor generation in flow boiling system, *Int. J. Heat Mass Transfer*, Vol.18, 1975, pp.1095-1099.

Whittle, R. H. and Forgan, R., Correlation for the minima in the pressure drop versus flow-rate curves for subcooled water flowing in narrow heated channels, *Nuclear Engineering and Design*, Vol.6, 1967, pp.89-99.

**DATE
FILMED**

12/23/93

END