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Flow Instability in Material Testing Reactors

Salah El-Din El-Morshedy
Reactors Department, Nuclear Research Center,
Atomic Energy Authority
Egypt

1. Introduction

Research reactors with power between 1 MW and 50 MW especially materials testing reactors (MTR), cooled and moderated by water at low pressures, are limited, from the thermal point of view, by the onset of flow instability phenomenon. The flow instability is characterized by a flow excursion, when the flow rate and the heat flux are relatively high; a small increase in heat flux in some cases causes a sudden large decrease in flow rate. The decrease in flow rate occurs in a non-recurrent manner leading to a burnout. The burnout heat flux occurring under unstable flow conditions is well below the burnout heat flux for the same channel under stable flow conditions. Therefore, for plate type fuel design purposes, the critical heat flux leads to the onset of the flow instability (OFI) may be more limiting than that of stable burnout. Besides, the phenomenon of two-phase flow instability is of interest in the design and operation of many industrial systems and equipments, such as steam generators, therefore, heat exchangers, thermo-siphons, boilers, refrigeration plants and some chemical processing systems. In particular, the investigation of flow instability is an important consideration in the design of nuclear reactors due to the possibility of flow excursion during postulated accident. OFI occurs when the slope of the channel demand pressure drop-flow rate curve becomes algebraically smaller than or equal to the slope of the loop supply pressure drop-flow rate curve. The typical demand pressure drop-flow rate curves for subcooled boiling of water are shown in Fig. 1 (IAEA-TECDOC-233, 1980). With channel power input S_2, operation at point d is stable, while operation at point b is unstable since a slight decrease in flow rate will cause a spontaneous shift to point a. For a given system, there is a channel power input S_c (Fig. 1) such that the demand curve is tangent to the supply curve. The conditions at the tangent point c correspond to the threshold conditions for the flow excursive instability. At this point any slight increase in power input or decrease in flow rate will cause the operating point to spontaneously shift from point c to point a, and the flow rate drops abruptly from M to M_c. For MTR reactors using plate-type fuel, each channel is surrounded by many channels in parallel. The supply characteristic with respect to flow perturbations in a channel (say, the peak power channel) is essentially horizontal, and independent of the pump characteristics. Thus, the criterion of zero slope of the channel demand pressure drop-flow curve is a good approximation for assessing OFI, i.e.
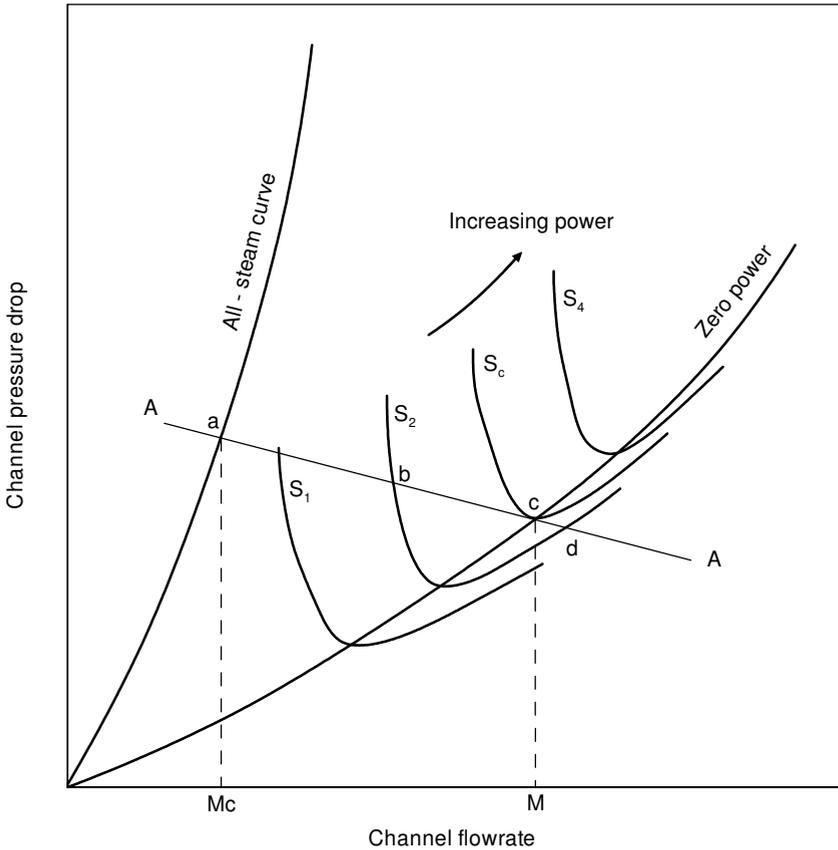
Fig. 1. Typical S-curves to illustrate OFI, (IAEA-TECDOC-233, 1980)

$$\frac{\partial (\Delta P)_{channel}}{\partial G} = 0 \qquad (1)$$

Functionally, the channel pressure drop-flow curve depends on the channel geometry, inlet and exit resistances, flow direction, subcooled vapor void fraction, and heat flux distribution along the channel.

2. Background

There is a lot of research work in the literature related to flow instability phenomenon in two-phase flow systems. (Ledinegg, M., 1938) was the first successfully described the thermal-hydraulic instability phenomenon later named Ledinegg instability. It is the most common type of static oscillations and is associated with a sudden change in flow rate. (Whittle & Forgan, 1967) and (Dougherty et al., 1991) were performed an experimental investigations to obtain OFI data in a systematic methodology for various combination of operating conditions and geometrical considerations under subcooled flow boiling. (Saha et

al., 1976) and (Saha & Zuber, 1976) carried out an experimental and analytical analysis on the onset of thermally induced two-phase flow oscillations in uniformly heated boiling channels. (Mishima & Nishihara, 1985) performed an experiment with water flowing in round tube at atmospheric pressure to study the critical heat flux, CHF due to flow instability, they found that, unstable-flow CHF was remarkably lower than stable-flow CHF and the lower boundary of unstable-flow CHF corresponds to the annular-flow boundary or flooding CHF. (Chatoorgoon, 1986) developed a simple code, called SPORTS for two-phase stability studies in which a novel method of solution of the finite difference equations was devised and incorporated. (Duffey & Hughes, 1990) developed a theoretical model for predicting OFI in vertical up flow and down flow of a boiling fluid under constant pressure drop, their model was based on momentum and energy balance equations with an algebraic modeling of two-phase velocity-slip effects. (Lee & Bankoff, 1993) developed a mechanistic model to predict the OFI in transient sub-cooled flow boiling. The model is based upon the influence on vapor bubble departure of the single-phase temperature. The model was then employed in a transient analysis of OFI for vertical down-wards turbulent flow to predict whether onset of flow instability takes place. (Chang & Chapman, 1996) performed flow experiments and analysis to determine the flow instability condition in a single thin vertical rectangular flow channel which represents one of the Advanced Test Reactor's (ATR) inner coolant channels between fuel plates. (Nair et al., 1996) carried out a stability analysis of a flow boiling two-phase low pressure and down flow relative to the occurrence of CHF, their results of analysis were useful in determining the region of stable operation for down flow in the Westinghouse Savannah River Site reactor and in avoiding the OFI and density wave oscillations. (Chang et al., 1996) derived a mechanistic CHF model and correlation for water based on flow excursion criterion and the simplified two-phase homogenous model. (Stelling et al., 1996) developed and evaluated a simple analytical model to predict OFI in vertical channels under down flow conditions, they found a parameter, the ratio between the surface heat flux and the heat flux required to achieve saturation at the channel exit for a given flow rate, is to be very accurate indicator of the minimum point velocity. (Kennedy et al., 2000) investigated experimentally OFI in uniformly heated micro channels with subcooled water flow using 22 cm tubular test sections, they generated demand curves and utilized for the specification of OFI points. (Babelli & Ishii, 2001) presented a procedure for predicting the OFI in down ward flows at low-pressure and low-flow conditions. (Hainoun & Schaffrath, 2001) developed a model permitting a description of the steam formation in the subcooled boiling regime and implemented it in ATHLET code to extend the code's range of application to simulate the subcooled flow instability in research reactors. (Li et al., 2004) presented a three dimensional two-fluid model to investigate the static flow instability in subcooled boiling flow at low-pressure. (Dilla et al., 2006) incorporated a model for low-pressure subcooled boiling flow into the safety reactor code RELAP5/Mod 3.2 to enhance the performance of the reactor code to predict the occurrence of the Ledinegg instability in two-phase flows. (Khater et al., 2007a, 2007b) developed a predictive model for OFI in MTR reactors and applied the model on ETRR-2 for both steady and transient states. (Hamidouche et al., 2009) developed a simple model based on steady-state equations adjusted with drift-flux correlations to determine OFI in research reactor conditions; they used RELAP/Mod 3 to draw the pressure drop characteristic curves and to establish the conditions of Ledinegg instability in a uniformly heated channel subject to constant outlet

pressure. From the thermal-hydraulic point of view, the onset of significant void (OSV) leads to OFI phenomena and experimental evidence shows also that OSV is very close to OFI (Lee & Bankoff, 1993; Gehrke & Bankoff, 1993). Therefore, the prediction of OFI becomes the problem of predicting OSV. The first study that addressed the OSV issue was performed by (Griffith et al., 1958), they were the first to propose the idea that boiling in the channel could be divided into two distinct regions: a highly subcooled boiling region followed by a slightly subcooled region, they defined the OSV point as the location where the heat transfer coefficient was five times the single-phase heat transfer coefficient. A few years later, (Bowring, 1962) introduced the idea that OSV was related to the detachment of the bubbles from the heated surface and the beginning of the slightly subcooled region was fixed at the OSV point. (Saha & Zuber, 1974) developed an empirical model based on the argument that OSV occurs only when both thermal and hydrodynamic constraints are satisfied, where a general correlation is developed to determine OSV based on the Peclet and Stanton numbers. (Staub, 1968) postulated that OSV occurs when steam bubbles detach from the wall and assumed a simple force balance on a single bubble with buoyancy and wall shear stress acting on detach the bubble with surface tension force tending to hold it on the wall. He also postulated that the bubble could grow and detach only if the liquid temperature at the bubble tip was at least equal to the saturation temperature. (Unal, 1977) carried out a semi-empirical approach to determine and obtain a correlation of OSV point for subcooled water flow boiling. (Rogers et al., 1986; Chatoorgoon et al., 1992) developed a predictive model which relates the OSV to the location where the bubble first detaches assuming that bubble grow and collapse on the wall in the highly sub-cooled region. (Zeiton & Shoukri, 1996, 1997) used a high-speed video system to visualize the sub-cooled flow-boiling phenomenon to obtain a correlation for the mean bubble diameter as a function of the local subcooling, heat flux, and mass flux. (Qi Sun et al., 2003) performed a predictive model of the OSV for low flow sub-cooled boiling. The OSV established in their model meets both thermodynamic and hydrodynamic conditions. Several coefficients involved in the model were identified by Freon-12 experimental data.

It is clear that, there are several predictive models for OSV and OFI have been derived from theoretical and experimental analysis in the literature. However, their predictions in vertical thin rectangular channels still have relatively high deviation from the experimental data. Therefore, the objective of the present work is to develop a new empirical correlation with lower deviation from the experimental data in order to predict more accurately the OFI phenomenon as well as void fraction and pressure drop in MTR reactors under both steady and transient states.

3. Mathematical model

3.1 Correlation development

Experimental evidence shows that, the onset of significant voids, OSV is very close to the onset of flow instability, OFI (Lee & Bankoff, 1993; Gehrke & Bankoff, 1993). Therefore, the prediction of OFI in the present work becomes the problem of predicting OSV. Due to the complicated nature of the subcooled nucleate boiling phenomenon, it is often convenient to predict OSV by means of empirical correlations. In the present work, an empirical correlation to predict the onset of significant void is proposed takes into account almost all

the related affecting parameters. The proposed correlation is represented best in terms of the following dimensionless groupings form:

$$\frac{\Delta T_{OSV}}{\Delta T_{sub,in}} = k_1 Bo^{k2} \Pr^{k3} \left(L/d_h \right)^{k4} \tag{2}$$

Where ΔT_{OSV} is the subcooling at OSV $= T_{sat} - T_{OSV}$

$\Delta T_{sub,in}$ is the inlet subcooling $= T_{sat} - T_{in}$ and

Bo is the boiling number $= \dfrac{\phi}{\rho_g U_g I_{fg}}$ where U_g is the rise velocity of the bubbles in the

bubbly regime (Hari & Hassan, 2002)

$$U_g = 1.53 \left[\frac{\sigma g \left(\rho_f - \rho_g \right)}{\rho_f^2} \right]^{1/4} \tag{3}$$

By taking the logarithmic transformation of equation (2) and applying the least squares method, the constants k_1, k_2, k_3 and k_4 are evaluated as 1, 0.0094, 1.606 and -0.533 respectively. So the developed correlation takes the following form:

$$\frac{\Delta T_{OSV}}{\Delta T_{sub,in}} = Bo^{0.0094} \Pr^{1.606} \Big/ \left(L/d_h \right)^{0.533} \tag{4}$$

with all water physical properties calculated at the local bulk temperature. This correlation is valid for low pressures at heat flux ranges from 0.42 to 3.48 MW/m^2 and L/d_h ratios from 83 to 191.

3.2 Bubble detachment parameter

A parameter, η (the bubble detachment parameter) which indicates the flow stability is defined as follows (Bergisch Gladbach, 1992):

$$\eta = \frac{U \times \Delta T_{sub}}{\phi} \tag{5}$$

where U is the local velocity, ΔT_{sub} is the local subcooling and ϕ is the local heat flux. The physical meaning of η is that it controls the behavior of the steam bubbles formed at active sides of the heating surface. If η decreases below a certain value (η_{OFI}), the steam bubble will detach from the wall, otherwise it will stay there. In order to be sure of the maximum power channels are protected against the occurrence of excursive flow instability, the parameter η must be higher than η_{OFI} by a considerable safety margin. Based on the developed correlation, η_{OFI} can be determined by:

$$\eta_{OFI} = \frac{U \times \Delta T_{sub,in}}{\phi} \times Bo^{0.0094} \Pr^{1.606} \Big/ \left(L/d_h \right)^{0.533} \tag{6}$$

3.3 Void fraction modeling

The ability to predict accurately the void fraction in subcooled boiling is of considerable interest to nuclear reactor technology. Both the steady-state performance and the dynamic response of the reactor depend on the void fraction. Studies of the dynamic behavior of a two-phase flow have revealed that, the stability of the system depends to a great extent upon the power density and the void behavior in the subcooled boiling region. It is assumed that the void fraction in partially developed region between onset of nucleate boiling (ONB) and the OSV equal to 0 and in the fully devolved boiling region from the OSV up to saturation, the void fraction is estmated by the slip-ratio model as:

$$\alpha = 1 / \left[1 + \left\{ (1-x)/x \right\} S \rho_g / \rho_f \right] \tag{7}$$

Where the slip, S is given by Ahmad, 1970 empirical relationship as:

$$S = \left(\frac{\rho_l}{\rho_g} \right)^{0.205} \left(\frac{G d_e}{\mu_l} \right) \tag{8}$$

The true vapor quality is calculated in terms of the thermodynamic equilibrium quality using empirical relationship from the earlier work of (Zuber et al., 1966; Kroeger & Zuber,1968) as:

$$x = \frac{x_{eq} - x_{eq,OSV} \, \exp \left(\frac{x_{eq}}{x_{eq,OSV}} - 1 \right)}{1 - x_{eq,OSV} \, \exp \left(\frac{x_{eq}}{x_{eq,OSV}} - 1 \right)} \tag{9}$$

Where the thermodynamic equilibrium quality, x_{eq} is given by:

$$x_{eq} = \frac{I_l - I_f}{I_{fg}} \tag{10}$$

and the thermodynamic equilibrium quality at OSV, $x_{eq,OSV}$ is given by:

$$x_{eq,OSV} = \frac{I_{l,OSV} - I_f}{I_{fg}} \tag{11}$$

3.4 Pressure drop modeling

Pressure drop may be the most important consideration in designing heat removal systems utilizing high heat flux subcooled boiling such as nuclear reactors. The conditions in which the pressure drop begins to increase during the transient from forced convection heat transfer to subcooled flow boiling are related to the OSV. The pressure drop is a summation of three terms namely; friction, acceleration and gravity terms.

3.4.1 Pressure drop in single-phase liquid

The pressure drop terms for single-phase liquid regime are given by:

$$\Delta P\big|_{friction} = \frac{2 f G^2 \Delta z}{\rho_l d_e} \tag{12}$$

where f is the Darcy friction factor for single-phase liquid. It is calculated for rectangular channels as:

for laminar flow (White, 1991)

$$f = 12 \Big/ \frac{G d}{\mu_l} \tag{13}$$

for turbulent flow (White, 1991)

$$\frac{1}{f^{1/2}} = 2.0 \log\left(\mathrm{Re}\, f^{1/2}\right) - 1.19 \tag{14}$$

$$\Delta P\big|_{acceleration} = (1/\rho_l - 1/\rho_{li})G^2 \tag{15}$$

$$\Delta P\big|_{gravity} = \mp \rho_l\, g\, \Delta z \tag{16}$$

3.4.2 Pressure drop in subcooled boiling

The pressure drop terms for subcooled boiling regime are given by:

$$\Delta P\big|_{friction} = \frac{f G^2}{2 \rho_l d_e} \int_0^z \phi^2(z)\, dz \tag{17}$$

where $\phi^2(z)$ is the two-phase friction multiplier and is obtained from (Levy, 1960) correlation as:

$$\phi^2(z) = \left[\frac{1 - x(z)}{1 - \alpha(z)}\right]^{2-m} \tag{18}$$

where m is 0.25 as suggested by (Lahey & Moody, 1979)

$$\Delta P\big|_{acceleration} = G^2 \int_0^z \frac{d}{dz}\left[\frac{(1-x)^2}{(1-\alpha)\rho_l} + \frac{x^2}{\alpha_g \rho_g}\right] dz \tag{19}$$

$$\Delta P\big|_{gravity} = g \int_0^z \left[\alpha \rho_g + (1-\alpha)\rho_l\right] dz \tag{20}$$

3.5 Prediction of OFI during transients

In order to apply the present correlation on transient analysis, both the momentum and energy equations are solved by finite difference scheme to obtain the velocity variation and temperature distribution during transient. The conservation of momentum for unsteady flow through a vertical rectangular channel of length L and gap thickness d and heated from both sides is:

$$\rho\frac{dU}{d\tau} = \frac{dP}{dz}(\tau) - \frac{\tau_w}{d} \tag{21}$$

with the initial condition $U = U_0$ at $\tau = 0$.

where the wall shear stress, is defined by:

$$\tau_w = \frac{f\rho U^2}{8} \tag{22}$$

and the friction factor, f is given by Blasius equation as:

$$f = 0.316\,\mathrm{Re}^{-0.25} \tag{23}$$

The conservation of energy for unsteady state one-dimensional flow is:

$$\rho C_P\left(\frac{\partial T}{\partial \tau} + U(\tau)\frac{\partial T}{\partial z}\right) = \frac{\phi(t)}{d} \tag{24}$$

with the boundary condition $T = T_i$ at $z = 0$ and Initial condition $T = T_0(z)$ at $\tau = 0$.

The initial steady-state coolant temperature distribution is calculated from a simple heat balance up to the distance z from the channel inlet taking into account that, the channel is heated from both sides.

-    for uniform heat flux distribution:

$$T_0(z) = T_{in} + \frac{\phi\,z}{G\,C_P\,d} \tag{25}$$

-    for chopped cosine heat flux distribution:

$$T_0(z) = T_{in} + \frac{2W_h L_p \phi_0}{\pi\,GCp\,W\,d} \times \left[\sin\frac{\pi(z - L/2)}{L_p} + \sin\frac{\pi L}{2L_p}\right] \tag{26}$$

where the axial heat flux distribution is given by:

$$\phi(z) = \phi_0 \cos\left(\frac{\pi(z - L/2)}{L_p}\right) \tag{27}$$

Where:

Lp: is the extrapolated length, $L_P = L + 2e$,

e : is the extrapolated distance and

ϕ_0 : is the maximum axial heat flux in the channel, $\phi_0 = \bar{\phi} \times PPF$

Where . $\bar{\phi}$ is the average surface heat flux and PPF is the power peaking factor.

The coolant temperature distribution during transient resulted from the solution of equation (24) by finite difference method is:

$$T_j^{p+1} = \frac{K_1 \times T_{j-1}^{p+1} + T_j^p + K_2}{1 + K_1} \tag{28}$$

where $K_1 = U^{p+1} \times \dfrac{\Delta \tau}{\Delta z}$ and

- for uniform heat flux distribution:

$$K_2 = \frac{2\phi^p \Delta \tau}{\rho C_p d} \tag{29}$$

- for chopped cosine heat flux distribution:

$$K_2 = \frac{2\phi_0^p \Delta \tau L_p}{\pi \rho C_p d \Delta z} \times \left[\sin\left(\frac{\pi\left(z_j - L/2\right)}{L_p} \right) - \sin\left(\frac{\pi\left(z_{j-1} - L/2\right)}{L_p} \right) \right] \tag{30}$$

4. Results and discussion

4.1 Assessment of the developed correlation

The subcooling at OSV is evaluated by the present correlation and the previous correlations described in table 1 for (Whittle & Forgan, 1967) experiments. All the results and experimental data are plotted in Fig. 2. The solid line is a reference with the slope of one is drawn on the plot to give the relation between the predicted and measured data. The present correlation shows a good agreement with the experimental data, it gives only 6.6 % relative standard deviation from the experimental data while the others gives 20.2 %, 26.4 %, 27.4 % and 35.0 % for Khater et al., Lee & Bankoff, Sun et al. and Saha & Zuber correlations respectively as shown table 1.

The experimental data of (Whittle & Forgan, 1967) on light water cover the following operating conditions:

- Rectangular channel with hydraulic diameter from 2.6 to 6.4 mm.
- Pressure from 1.10 to 1.7 bar.
- Heat flux from 0.66 to 3.4 MW/m².
- Inlet temperature from 35 to 75°C.
- Velocity from 0.6096 to 9.144 m/s.

Correlation	Description
Khater et al., 2007	$\Delta T_{OSV} = \dfrac{\sqrt{1 + 16\left(\dfrac{\rho_f}{\rho_g}\right)\left(\dfrac{C_P\phi}{0.172\,h_i I_{fg}}\right)d_h} - 1}{2\left(\dfrac{\rho_f}{\rho_g}\right)\left(\dfrac{C_P}{I_{fg}}\right)}$
Lee & Bankoff, 1993	Approximated by: $St = 0.076 Pe^{-0.2}$
Sun et al., 2003	$\Delta T_{OSV} = \dfrac{\sqrt{1 + 16\left(\dfrac{\rho_f}{\rho_g}\right)\left(\dfrac{C_P\phi}{A_i h_i d_h I_{fg}}\right)} - 1}{2\left(\dfrac{\rho_f}{\rho_g}\right)\left(\dfrac{C_P}{I_{fg}}\right)}$ with $h_i = C1\dfrac{k}{d_b}\mathrm{Re}_{fg}^{C2}\mathrm{Pr}^{C3}\left(\dfrac{\rho_f}{\rho_g}\right)^{C4}$
Saha & Zuber, 1976	$Nu = \dfrac{\phi\,d_h}{k\,\Delta T_{OSV}} = 455$ for $Pe \le 70000$ $St = \dfrac{\phi}{G C_P \Delta T_{OSV}} = 0.0065$ for $Pe > 70000$

Table 1. Previous correlations used in comparison
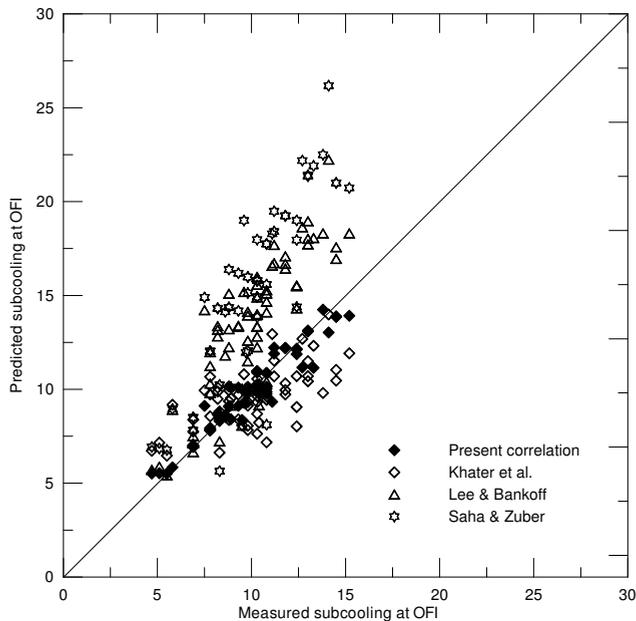
Fig. 2. Comparison of the present correlation with previous models

Correlation	Relative standard deviation
Present correlation	0.066
Khater et al.	0.202
Lee & Bankoff	0.264
Sun et al	0.274
Saha & Zuber	0.350

Table 2. Relative standard deviation from experimental data for subcooling at OSV

4.2 Prediction of S-curves

The pressure drop for Whittle & Forgan experimental conditions is determined and depicted in Figs 3 and 4 against the experimental data. The present model predicts the S-curves with a good agreement achieved with the experimental data. A well defined minimum occurred in all the S-curves. The change in slope from positive to negative was always abrupt and the pressure drop at the condition of the minimum was always approximately equal to that for zero-power condition. As subcooled liquid heat ups along the wall of a heated channel, its viscosity decreases. Increasing the wall heat flux causes further reduction in liquid viscosity. Therefore, pressure drop associated with pure liquid flow decreases with increasing wall heat flux. The trend changes significantly when bubbles begins to form. Here, increasing wall heat flux increases both the two-phase frictional and accelerational gradients of pressure drop. Pressure drop therefore begins to increase with increasing heat flux.
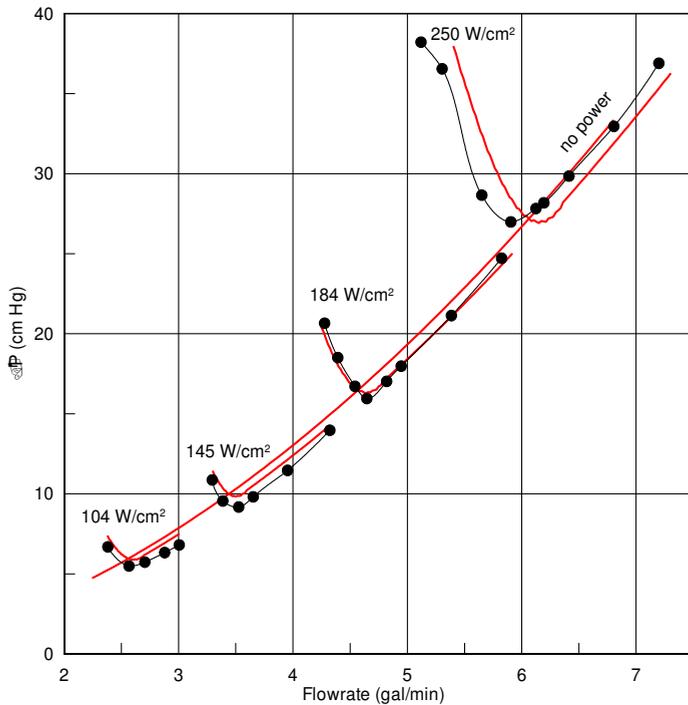

Fig. 3. S-curves prediction for (Whittle & Forgan, 1967) experiments (No. 1 test section)
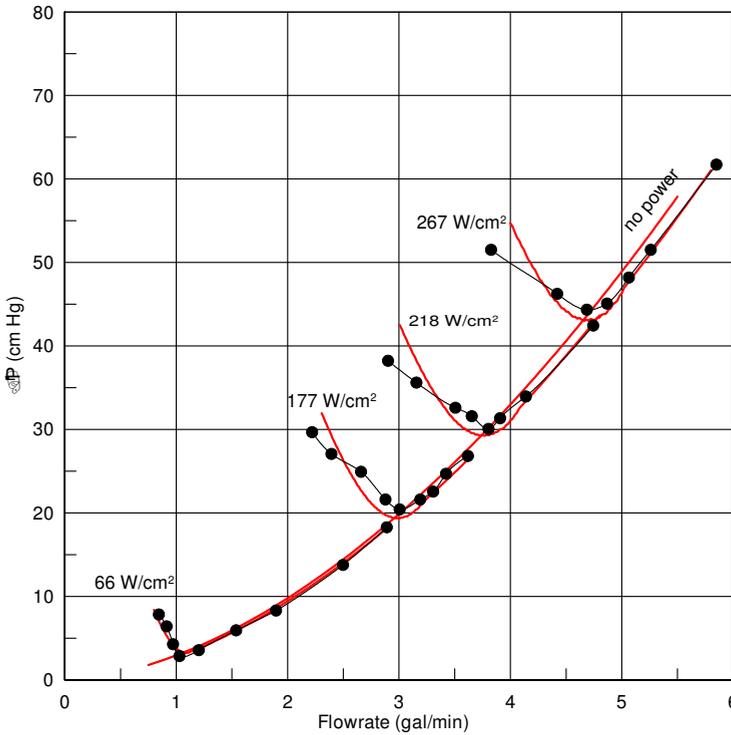
Fig. 4. S-curves prediction for (Whittle & Forgan, 1967) experiments (No. 3 test section)

4.3 Prediction of OFI during transients

The present model is used to predict the OFI phenomenon for the IAEA 10 MW MTR generic reactor (Matos et al., 1992) under loss of flow transient. The reactor active core geometry is 5×6 positions where both standard and control fuel elements are placed with a total of 551 fuel plates. A summary of the key features of the IAEA generic 10 MW reactor with LEU fuel are shown in Table 3 (IAEA-TECDOC-233, 1980). The pump coast-down is initiated at a power of 12 MW with nominal flow rate of 1000 m³/h and reduced as $e^{\tau/T}$, with T = 1 and 25 seconds for fast and slow loss-of-flow transients respectively. The reactor is shutting down with Scram at 85 % of the normal flow. The pressure gradient is proportional to mass flux to the power 2. Therefore, the pressure gradient during transient is considered exponential and reduced as $e^{2\tau/T}$, with T = 1 and 25 seconds for fast and slow loss-of-flow transients respectively with steady-state pressure gradient, $\left.\dfrac{dP}{dz}\right|_{\tau=0.0}$ = 40.0. The calculation is performed on the hot channel where the axial heat flux is considered chopped

cosine distribution of a total power peaking factor equal to 2.52 with the extrapolated length equal to 8.0 cm.

Coolant	Light Water
Coolant flow direction	Downward
Fuel thermal conductivity (W/cm K)	1.58
Cladding thermal conductivity (W/cm K)	1.80
Fuel specific heat (J/g K)	0.728
Cladding specific heat (J/g K)	0.892
Fuel density (g/cm^3)	0.68
Cladding density (g/cm^3)	2.7
Radial peaking factor	1.4
Axial peaking factor	1.5
Engineering peaking factor	1.2
Inlet coolant temperature	38.0
Operating pressure (bar)	1.7
Length (cm)	8.0
Width (cm)	7.6
Height (cm)	60.0
Number of fuel elements SFE/SCE	21/4
Number of plates SFE/SCE	23/17
Plate meat thickness (mm)	0.51
Width (cm) active/total	6.3/6.65
Height (cm)	60.0
Water channel thickness (mm)	2.23
Plate clad thickness (mm)	0.38

Table 3. IAEA 10 MW generic reactor specifications

Figures 5, 6, and 7 show the OFI locus on graphs of the flow velocity, the exit bulk temperature and the bubble detachment parameter as a function of time for fast loss-of-flow transient. The pressure gradient reduced exponentially from 40 kPa/m as $e^{-2\tau}$, while the average heat flux is maintained at a constant value. The transient time is 0.16 second which represents the period from steady-state to the time of 85% of the normal flow (just before Scram). The flow velocity decreases, the bulk temperature increases, and the bubble detachment parameter decreases. Figure 5 shows slight changes of the velocity variation depending on the magnitude of the heat added from both plates. In this figure OFI is reached at end of each initial heat flux curve. Figure 6 shows that, OFI is always predicted at exit bulk temperature greater than 104°C while, Fig. 7 shows that, OFI phenomenon is always predicted at bubble detachment parameter value lower than 22. In case of slow loss-of-flow transient, the pressure gradient reduced exponentially from 40 kPa/m as $e^{-0.08\tau}$, the transient time is 4.0 seconds which represents the period from steady-state to the time just before Scram at 85% of the normal flow.
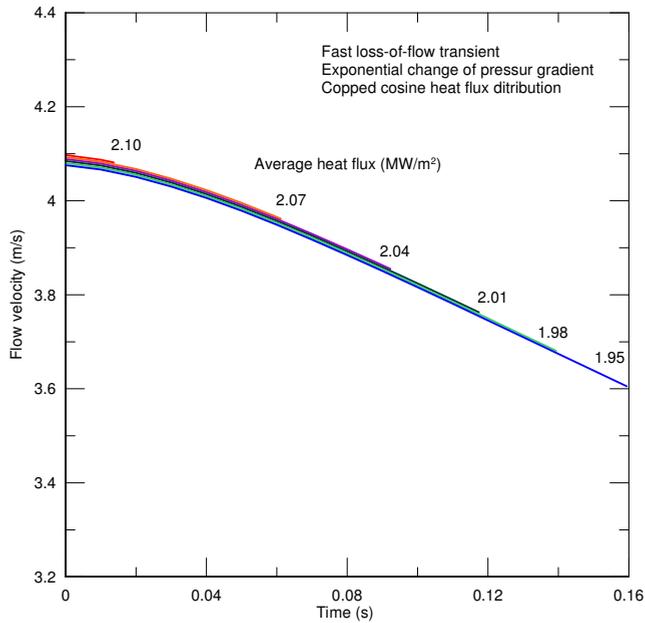
Fig. 5. Flow velocity variations for various heat fluxes under fast loss-of-flow transient, OFI reached at the end of each curve
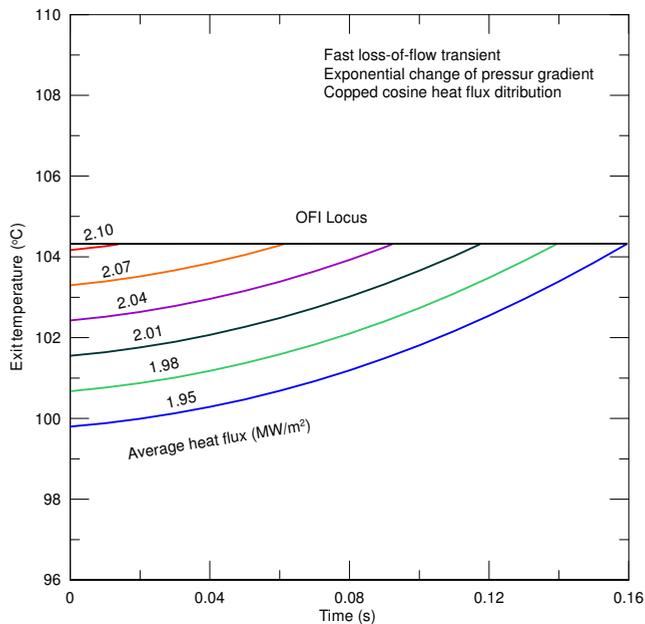

Fig. 6. Exit bulk temperature variations for various heat fluxes under fast loss-of-flow transient
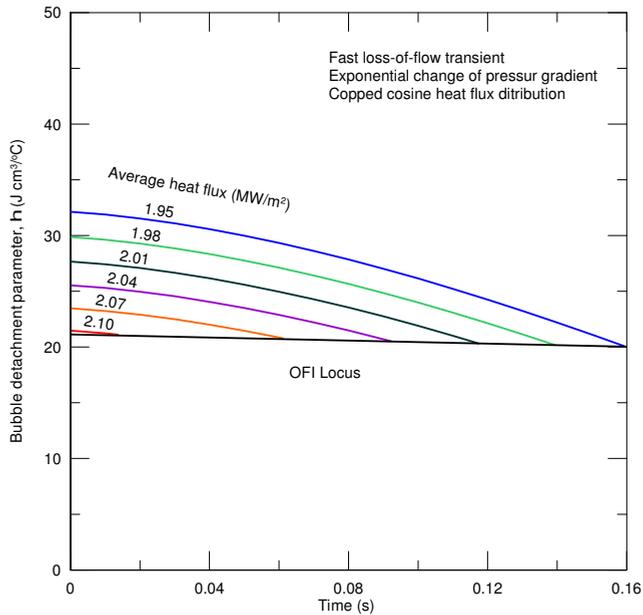
Fig. 7. Bubble detachment parameter variations for various heat fluxes under fast loss-of-flow transient
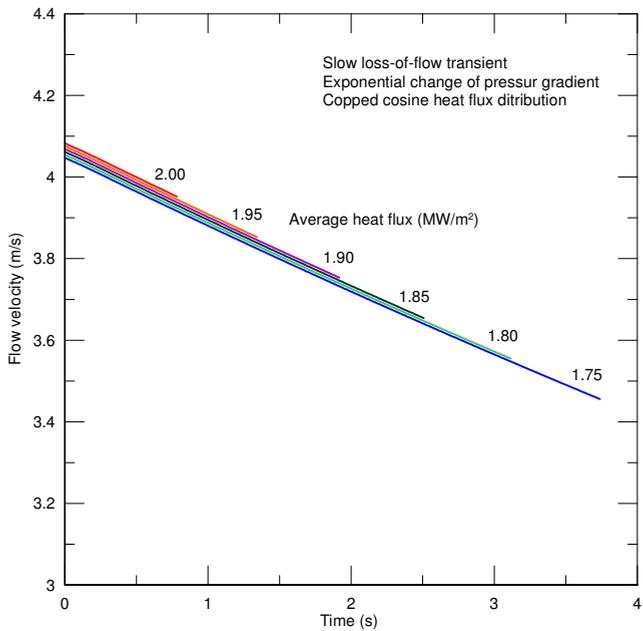

Fig. 8. Flow velocity variations for various heat fluxes under slow-of-flow transient, OFI reached at the end of each curve
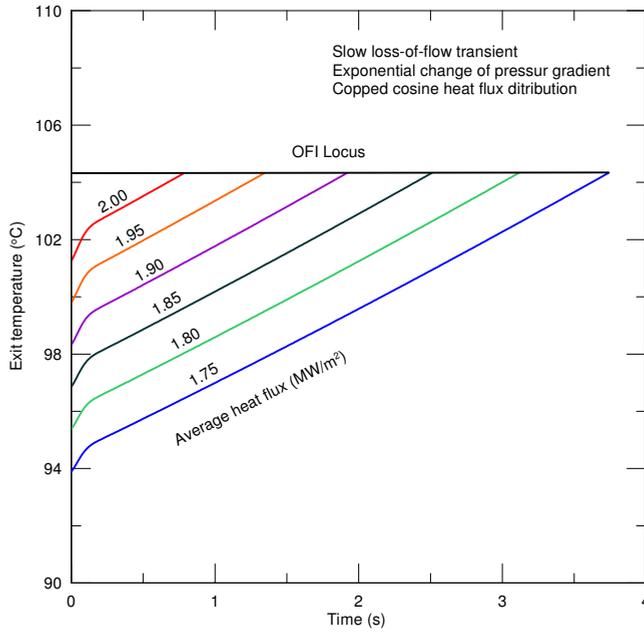
Fig. 9. Exit bulk temperature variations for various heat fluxes under slow-of-flow transient
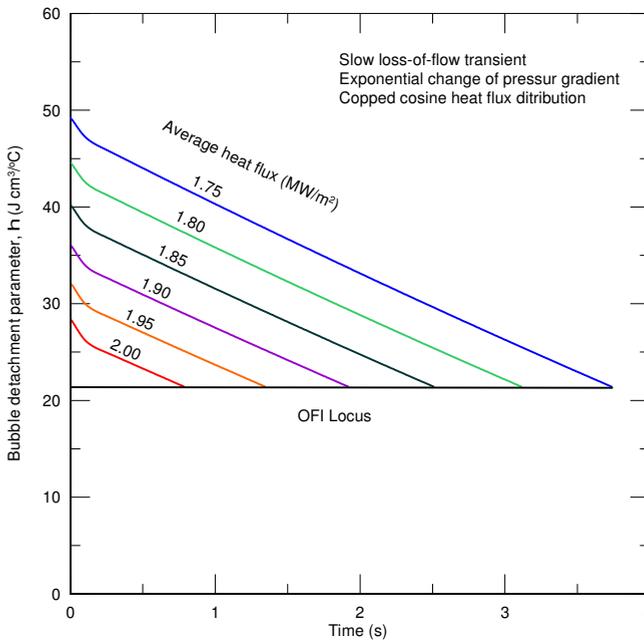

Fig. 10. Bubble detachment parameter variations for various heat fluxes under slow-of-flow transient

Figures 8, 9, and 10 show the OFI locus on graphs of the flow velocity, the exit bulk temperature and the bubble detachment parameter as a function of time for slow-of-flow transient. The graphs trends are same as for fast loss-of-flow-transient except that, OFI phenomenon could predicted at lower heat fluxes. Figures 9 and 10 show that, OFI phenomenon is always predicted at exit bulk temperature greater than 104°C and bubble detachment parameter value lower than 22 (the same values obtained for fast loss-of-flow-transient).

4.3 Safety margins evaluation

The safety margin for OFI phenomenon is defined as the ratio between the power to attain the OFI phenomenon within the core channel, and the hot channel power, this means that, OFI margin is equal to the ratio of the minimum average heat flux leads to OFI in the core channels and the average heat flux in the hot channel. It is found that, the OFI phenomenon occurs at an average heat flux of 2.1048 MW/m² for steady-state operation ($\tau = 0.0s$), and 1.7294 MW/m² just before Scram ($\tau = 4.0s$). Thus, these values can be regarded as the maximum possible heat fluxes to avoid OFI under steady-state operation and just before Scram respectively. The maximum hot channel heat flux is determined using the data of table 3 as 0.72595 MW/m² with an average value of 0.5648 MW/m². This means that, the reactor has vast safety margins for OFI phenomenon of 3.73 for steady-state operation, 3.45 and 3.06 just before Scram for both fast and low loss-of-flow transient respectively. Table 4 gives the estimated heat flux leading to OFI and the safety margin values for both the steady and transient states.

Description	Steady-state τ = 0.0 s	Transient τ = 0.16 s	Transient τ = 4.0 s
OFI heat flux (MW/m²)	2.1048	1.9491	1.7294
Safety margin for OFI	3.73	3.45	3.06

Table 4. Reactor safety margins for OFI phenomenon.

5. Conclusion

Flow instability is an important consideration in the design of nuclear reactors due to the possibility of flow excursion during postulated accident. In MTR, the safety criteria will be determined for the maximum allowable power and the subsequent analysis will therefore restrict to the calculations of the flow instability margin. In the present work, a new empirical correlation to predict the subcooling at the onset of flow instability in vertical narrow rectangular channels simulating coolant channels of MTR was developed. The developed correlation involves almost all parameters affecting the phenomenon in a dimensionless form and the coefficients involved in the correlation are identified by the experimental data of Whittle and Forgan that covers the wide range of MTR operating conditions. The correlation predictions for subcooling at OSV were compared with predictions of some previous correlations where the present correlation gives much better agreement with the experimental data of Whittle and Forgan with relative standard

deviation of only 6.6%. The bubble detachment parameter was also estimated based on the present correlation. The present correlation was then utilized in a model predicting the void fraction and pressure drop in subcooled boiling under low pressure. The pressure drop model predicted the S-curves representing the two-phase instability of Whittle and Forgan with good accuracy. The present correlation was also incorporated in the safety analysis of the IAEA 10 MW MTR generic reactor in order to predict the OFI phenomenon under both fast and slow loss-of-flow transient. The OFI locus for the reactor coolant channels was predicted and plotted against flow velocity, exit temperature and bubble detachment parameter for various heat flux values. It was found that the reactor has vast safety margins for OFI phenomenon under both steady and transient states.

6. Nomenclature

Cp   : specific heat, J/kg°C
d   : gap thickness, m
db   : bubble diameter, m
dh   : heated diameter, m
de   : hydraulic diameter, m
g   : acceleration of gravity, m/s^2
G   : mass flux, kg/ms^2
I   : enthalpy, J/kg
Ifg   : latent heat of vaporization, J/kg
k   : thermal conductivity, W/m°C
L   : active length, m

Nu   : Nusselt number, = $h\,d_e/k$

P   : pressure, Pa

Pe   : Peclet number, = $Re\,Pr$

Pr   : Prantdel number, = $\mu Cp/k$

Re   : Reynolds number = $G\,d_e/\mu$

St   : Stanton number, = Nu/Pe

T   : temperature, °C
U   : coolant velocity, m/s
W   : channel width, m
x   : steam quality
z   : distance in axial direction, m

Greek Letters

α   : void fraction, dimensionless

ΔP	: pressure drop, Pa
ϕ	: heat flux W/m^2
μ	: dynamic viscosity, kg/m s
ρ	: density, kg/m^3
σ	: surface tension, N/m
τ	: time, s
τ_w	: wall shear stress, N/m^2

Subscripts

f	: liquid phase,
fg	: difference of liquid and vapor,
g	: vapor phase,
h	: heated
in	: inlet
OFI	: onset of flow instability,
OSV	: onset of significant void,
s	: saturation,
w	: wall.

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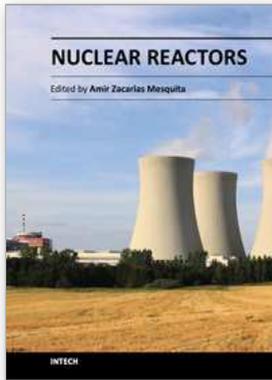
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