Chapter 5

PRESSURE DROP RELATIONSHIPS

NOMENCLATURE

A	flow area
A_r	flow area ratio (< 1)
A_g	projected grid cross section
C_f	friction coefficient
C_d	drag coefficient
C_v	modified loss coefficient
C_0	distribution coefficient
D, d	diameter
e	absolute roughness
F	correction coefficient
Fr	Froude Number
f	friction factor
f_l	laminar friction factor
f_t	turbulent friction factor
G	mass flux
G_{SL}	superficial liquid mass flux ($\rho_L j_L$)
g	gravitational acceleration
H	wire pitch
h	heat transfer coefficient
h_{fg}	latent heat
I	specific enthalpy
j	volumetric flux
K	loss coefficient
L	length
p_t	rod pitch
P, p	pressure
q	heat flux
Re	Reynolds Number
S	slip ratio
T	temperature
t	thickness
u	velocity
V	velocity
v	specific volume
v_{fg}	$v_G v_L$
W	mass flow rate
x	mass quality
Z	length, elevation

GREEK SYMBOLS

α	void fraction
θ	angle of direction of flow with vertical

β	homogeneous void fraction
Δ	difference
δ	thickness of annular film
ϕ^2	two phase friction multiplier
μ	dynamic viscosity
ρ	density
σ	surface tension
χ	Martinelli parameter

SUBSCRIPTS

a	acceleration
av	average
B	bundle
b	bulk
cir	circular
crit	critical
e	elevation
f	film, frictional
G	vapour
GO	gas only
h	hydraulic
i	inlet
L	liquid
LO	liquid only
l	local
m	mean
o	outlet
R	relative
s	spacer
SPF	single-phase flow
TP, TPF	two-phase flow
tot	total
w	wall

5.1. INTRODUCTION

In the nuclear industry, pressure drop correlations find extensive application for design and analysis of many systems and components. For example, validated pressure drop correlations (PDCs) are required to determine the extent of orificing needed to match the channel flow to the power, pumping power required, the riser height required to achieve a certain circulation rate in natural circulation BWRs, recirculation ratio in natural circulation type steam generators, stability analysis, transient and accident analyses, etc. Some of the above applications require correlations for both single-phase and two-phase flows. Two-phase flows are encountered during normal operation of BWRs, transients and accidents in PWRs and PHWRs, and in certain components like the steam generators.

Two-phase flow pressure drop depends on a large number of independent parameters like geometric configuration of the duct, mass and volume fractions of the individual phases, pressure, fluid properties, mass flux, orientation of the duct (i.e. horizontal, vertical or inclined), flow direction (i.e. vertical upflow, downflow or counter-current flow) and flow patterns. Further, in many engineering applications, two-phase flow systems can be adiabatic, diabatic, one-component, two-component or multi-component. To cater to the needs of these diverse applications, a very large number of two-phase flow pressure drop correlations are reported in literature. Many of these correlations, being empirical in nature, are applicable only for limited parameter ranges. Even mechanistic models are based on certain assumptions and careful examination of the particular application is necessary to ensure that the assumptions made in deriving the model hold good. For many practical situations, designers and analysts often require some guidance to choose the appropriate correlation.

The parameter ranges of two-phase flow in some of the above applications can be quite different. For example, natural circulation reactors are characterised by relatively low mass flux and driving pressure differential compared to forced circulation systems. Therefore, correlations chosen for the analysis of natural circulation systems require improved accuracy at low mass fluxes. For the analysis of critical flow, following a break in high pressure systems, pressure drop correlations valid for very high mass fluxes (10–20 Mg/m^2s) are required. For investigations on the start-up procedure for natural circulation boiling water reactors, correlations valid over a wide range of pressures starting from atmospheric pressure are required.

In this document, some of the commonly used and often-cited pressure drop correlations are compiled along with their range of application. Later on assessments of these PDCs reported in literature are reviewed and their recommendations summarized. Limitations of the reported assessments are brought out and a rational assessment procedure for diabatic flow is proposed. As per this procedure assessment of pressure drop correlations cannot be carried out in isolation. For example, a rational assessment of diabatic flow pressure drop requires pre-assessment of models for the onset of nucleate boiling (ONB) and void fraction. Assessment of flow pattern specific pressure drop correlations also require pre-assessment of the criteria for flow pattern transitions.

5.2. SURVEY OF SITUATIONS WHERE PRESSURE DROP RELATIONSHIPS ARE IMPORTANT

In a nuclear reactor, the generated power, Q_G, is extracted from the core by means of a fluid coolant. The first purpose of the thermohydraulic design of the reactor is to ensure that, during the nominal steady state reactor operating conditions, the extracted power, Q_E, is equal to the generated one. Secondly, for accidental conditions, the evaluation of the difference between Q_G and Q_E is necessary for predicting the behaviour of the plant. The evaluation of the extracted power is performed by means of the well known relationship:

$$Q_E = W \Delta I \tag{5.1}$$

where

ΔI is the enthalpy difference between the core outlet and inlet and W is the mass flow rate.

For the evaluation of the extracted power, it is then necessary to know the flow rate. In some cases, it can be measured (total flow rate in the main loop) but generally at design level, it has to be computed and this calculation requires a knowledge of the pressure loss through the different parts of the plant.

It is necessary to take into account the fact that the total pressure loss is due to different components, namely distributed pressure loss due to friction, local pressure losses due to sudden variations of shape, flow area, direction, etc. and pressure losses (the reversible ones) due to acceleration (induced by flow area variation or by density change in the fluid) and elevation (gravity effect).

A general purpose relationship for the evaluation of the pressure loss in any possible case does not exist up to now and thus it becomes necessary to collect a set of relationships applicable to the different configurations, conditions, etc. A list of the factors on which the pressure loss depends is shown in Table 5.1.

An important factor affecting the pressure loss is the geometry. In a reactor plant, we have to deal with several basic geometrical shapes (circular pipes, annuli, etc.) and with a number of special devices, like rod bundles, heat exchangers, valves, headers, plenums, pumps, pools, etc. Other factors are then concerned with the fluid status (single or two phase/one component, two-component or multi-component), the flow nature (laminar or turbulent), the flow pattern (bubbly, slug, annular, etc.), the flow direction (vertical upflow, downflow, inclined flow, horizontal flow, counter-current flow, etc.) and the operating conditions (transient or steady state).

TABLE 5.1. FACTORS ON WHICH THE PRESSURE DROP DEPENDS

Geometry	basic shapes	circular pipe, rectangular channel, annulus, etc.
	other shapes & devices	rod bundle, spacer, valve, heat exchanger, orifice, plenum, header, pump etc.
fluid status	single phase	
	two phase	one-component, two-componnet & multi-component
flow nature	laminar	
	turbulent	
flow patterns	bubbly, slug, annular, etc.	
flow direction	vertical upflow, downflow, inclined flow & horizontal flow	co-current & counter-current flow
operating conditions	steady state transient	
driving force	forced convection	
	natural convection	

A final, very important issue, is concerned with the driving force depending on whether the flow is sustained by a density difference in the fluid (natural convection) or by a pump (forced convection), or whether there will be feedback between the pressure loss and the extracted power or not. Once more, in case of natural convection, some differentiation could arise from what is called microscopic natural convection: normally the pressure loss inside a device does not depend on the fact that the flow is sustained by a pump or by a density difference (macroscopic natural convection); however, in some circumstances, local effects could happen and, as a consequence, the pressure loss will be influenced by the driving force.

By looking at Table 5.1, it appears clearly that it generates a very big matrix of conditions and to fill all the matrix cells is a very hard job. At the same time, it becomes immediately clear that the filling of the whole matrix is not necessary. For example, with respect to the geometry, mainly the basic geometrical shapes have to be taken into account. Some of the geometric conditions of interest are identified in the next section. The pressure loss correlation for special devices is usually given by the manufacturer.

5.2.1. Distinction between core and system approach

The term *thermalhydraulic analysis* is often used to identify two widely different analytical approaches. The first one can be called *core approach* and is mainly concerned with the reactor core. In this case, a very detailed analysis is performed at subchannel level and, consequently, only the basic geometrical shapes are taken into account. For instance, the pressure drop in rod bundles is usually computed by subdividing them into subchannels of simple shape. The bundle pressure drop is then computed based on the pressure drop in single subchannel and, in principle, no special pressure drop correlation for bundles is needed. The special devices are limited to the spacers, a relatively limited class. Due to the fact that the analysis is a very detailed one, it is normally performed for steady state conditions or for slow transients, computed as subsequent steady states. This approach is the basic one for design purposes.

The second one can be called *system approach* and deals with the whole plant. In this case, each component is represented by a small number of mesh points. For instance, no detailed geometrical description of the core is considered. All the subassemblies are usually represented by means of one pin, from a thermal point, and the pressure drop is then computed by means of a bundle pressure drop correlation. Again, basic geometrical shapes are needed (circular pipe, annulus, etc.) but the several complex geometries of interest are represented by means of adhoc empirical relationships. This approach is mainly used in safety analysis and consequently deals with transient conditions.

5.2.2. Geometric conditions of interest

Geometric conditions of interest to nuclear power plants (NPPs) only are considered here. Emphasis is made on geometric conditions that are relevant to the primary loop of NPPs. The secondary loop of NPPs (the steam generator and the piping up to the main steam isolation valve (MSIV) and the feedwater valves in case of PWRs and PHWRs) is also important and is to be considered. In addition, the emergency core cooling (ECC) lines from the ECC pumps to the injection point along with the different types of valves may also be considered. Also, there are quite a few advanced designs to be dealt with (examples are SBWR, AP-600, CANDU-3, CANDU-9, EPR, AHWR, etc.). Again, it becomes a difficult task to cover typical geometries relevant to all these designs. For the purpose of this report, the various geometries relevant to NPPs can be classified into two categories:

5.2.2.1. Simple geometry for steady state and design calculations

In case of NPPs, the attention is generally limited to the nuclear fuel. The geometries of interest for local pressure drop are the spacer grids, tie plates, etc. Similarly for distributed pressure drop the geometries of interest are the channel and subchannels (various types, i.e. central, lateral, middle-lateral) for the square and the triangular array.

TABLE 5.2. LOCATIONS IN A PWR WHERE LOCAL AND DISTRIBUTED PRESSURE LOSSES ARE IMPORTANT

Local pressure drop in the RPV:	Cold leg to downcomer	
	Downcomer to lower plenum entry	
	Core inlet	
	Spacers	
	Core outlet	
	Upper plenum to hot leg	
	Bypasses:	Lower plenum to core bypass
		Core bypass to upper plenum
		Downcomer to hot leg
		Downcomer to upper head
		Upper head to upper plenum (direct)
		Upper head — Control Rod Guide (CRG)
		CRG-Upper Plenum (different positions)
Local pressure drops in the primary loop:		
	Hot leg bends	
		Hot leg to steam generator inlet water box entry
		U-tube bends
		U-tube exit
		Steam generator outlet water box to cold leg entry
	Loop seal bends	
		Pump inlet
		Pump (inside with various situations for the rotor)
		Pump outlet
		Pressurizer to surge line entry
		Hot leg to surge line connection
		Surge line bends if any
		Accumulator to pipe entry
		Accumulator pipe bends
		Accumulator line check valve

Similarly distributed (due to skin friction) pressure drops are important for the following locations in a PWR:

Distributed pressure drops in the RPV:	Downcomer
	Core and Core bypass
	Upper Plenum
	CRG
Distributed pressure drops in the primary loop:	Hot leg
	Surge line
	U-Tubes
	Cold leg — loop seal
	Cold leg — horizontal

In addition, the reactor system consists of pipes of various sizes, annulus, etc. The flow paths on the secondary side of the steam generators and the water boxes could be considered separately.

5.2.2.2. Complex geometry (or system) for safety — transient-analysis

During a transient, both direct (i.e. the nominal direction of the flow) and reverse flow directions are relevant. Both transient and steady state knowledge is relevant (as already mentioned). Both single phase (liquid or steam only) and two phase flows are relevant. Flows with phase opposition including counter current flow limit (CCFL) may happen in any discontinuity.

A knowledge of local and distributed pressure drops is necessary for transient analysis, (e.g. LOCA calculations). For example, in a typical PWR, local loss coefficients for direct and reverse flow must be supplied by the code user for each of the locations identified in Table 5.2. Table 5.2 also identifies the locations where distributed pressure drops are important.

Similar tables can be prepared for other reactors. For example, in a pressure tube type heavy water reactor, additional local loss coefficients required are listed in Table 5.3.

TABLE 5.3. LOCATIONS IN A PHWR WHERE ADDITIONAL LOCAL PRESSURE LOSSES ARE IMPORTANT

Entry loss from steam generator outlet pipes to header
Header to feeder entry loss
Inlet feeder bends
Inlet grayloc
Inlet grayloc to Liner tube entry
Liner tube to channel entry
Fuel locator
Junction between two bundles
Channel to liner tube entry
Liner tube to outlet grayloc
Outlet grayloc
Outlet feeder bends
Feeder to header entry loss
Header to steam generator inlet pipe entry

5.3. CORRELATIONS FOR DESIGN AND ANALYSIS

5.3.1. Components of pressure drop

The overall static pressure drop, Δp, experienced by a fluid while flowing through a duct comprises of the following components:

$$\Delta p = \Delta p_f + \Delta p_l + \Delta p_a + \Delta p_e \quad (5.2)$$

where

ΔP_f, ΔP_l, ΔP_a and ΔP_e are the components of pressure drop due to skin friction, form friction (also known as local friction), acceleration and elevation respectively. The skin friction pressure drop is also known simply as friction pressure drop.

5.3.1.1. Friction pressure drop

This is the irreversible component of pressure drop caused by shear stress at the wall and can be expressed as:

$$\Delta p_f = \frac{fL}{D_h}\frac{W^2}{2\rho A^2} \tag{5.3}$$

where

D_h is equal to 4 times flow area/wetted perimeter.

The pressure drop occurs all along the length and hence referred to as distributed pressure drop sometimes. This equation is applicable for single-phase and homogeneous two-phase flows, although, the method of calculation of the friction factor, f, and density, ρ, differ in the two cases. Pressure drop across tubes, rectangular channels, annuli, bare rod bundle (i.e. without spacers), etc. are examples of this component.

5.3.1.2. Local pressure drop

This is the localized irreversible pressure drop component caused by change in flow geometry and flow direction. Pressure drop across valves, elbows, tee, spacer, etc. are examples. The local pressure drop is given by

$$\Delta p_l = K\frac{W^2}{2\rho A^2} \tag{5.4}$$

where

K is the local loss coefficient, the correlations for which differ for different geometries and for single-phase and two-phase flows.

5.3.1.3. Acceleration pressure drop

This reversible component of pressure drop is caused by a change in flow area or density. Expansion, contraction and fluid flowing through a heated section are the examples. The acceleration pressure drop due to area change for single-phase and two-phase flow can be expressed as

$$\Delta p_a = \frac{(1 - A_r^2)W^2\phi}{2A_0^2\rho_L} \tag{5.5}$$

where A_0 = smaller flow area

$\phi = 1$ for single-phase flow and for two-phase flow ϕ is given by:

$$\phi = \left(\frac{x^3}{\rho_G^2\, \alpha^2} + \frac{(1-x)^3}{\rho_L^2 (1-\alpha)^2}\right)\left(\frac{\rho_G\, \rho_L}{x\, \rho_L + (1-x)\rho_G}\right) \tag{5.6}$$

The acceleration pressure drop due to density change for single-phase and two-phase flows can be expressed as:

$$\Delta p_a = G^2 \left\{\left(\frac{1}{(\rho_m)_o}\right) - \left(\frac{1}{(\rho_m)_i}\right)\right\} \tag{5.7}$$

For single-phase flows, this component is negligible, but can be significant in two-phase flows. For two-phase flow, the above equation can be used with ρ_m given by:

$$\frac{1}{\rho_m} = \left(\frac{x^2}{\rho_G\, \alpha} + \frac{(1-x)^2}{\rho_L (1-\alpha)}\right) \tag{5.8}$$

To evaluate the acceleration pressure drop due to density change, accurate prediction of the density of fluid is necessary. For single phase flow, density of fluid can be predicted reasonably well with established relationships for thermophysical properties of the fluid. For two phase flow, it is necessary to predict void fraction accurately to determine density and in turn acceleration pressure drop. Hence, correlation for void fraction needs to be chosen judiciously.

5.3.1.4. Elevation pressure drop

This reversible component of pressure drop is caused by the difference in elevation and can be expressed as:

$$\Delta p_e = \rho g \Delta z \cos \theta \tag{5.9}$$

where

θ is the angle with the vertical in the direction of flow. For two phase flow,

$$\rho = \rho_L (1-\alpha) + \rho_G\, \alpha \tag{5.10}$$

In many instances with vertical test sections, the elevation pressure drop is the largest component. For such cases, accurate prediction of the void fraction is essential which again calls for a judicious choice of the correlation for void fraction.

5.3.2. Configurations

For the purpose of design of advanced reactors, the required correlations mainly cover the following configurations. For friction pressure loss, circular pipe, annulus, rectangular channels and rod cluster and for local pressure loss, spacer, bottom and top tie plates, flow area changes like contraction, expansion, bends, tees, valves etc. are most common. For CANDU type fuel bundles, the alignment of two adjacent fuel bundles also is important in estimating the pressure drop. In addition, in-core effects like radiation induced creep, blister formation, swelling, corrosion, etc. are also important factors affecting the pressure drop which are not dealt with here. Following is an account of the pressure drop correlations described configuration-wise and generally used for design.

5.3.3. Friction pressure drop correlations

The present compilation of pressure drop correlations is applicable to steady state fully developed flow. Fully developed flow conditions are expected to occur in long components like the steam generator U-tubes.

5.3.3.1. Circular pipe

5.3.3.1.1. Adiabatic single-phase flow

For fully developed laminar flow, the friction factor is given by:

$$f = 64/Re \tag{5.11}$$

which is valid for Reynolds number less than 2000. For turbulent flow in smooth pipes several friction factor correlations are proposed and in use. A few commonly used correlations for smooth pipe are given below.

Blasius (1913) proposed the following equation:

$$f = 0.316\, Re^{-0.25} \tag{5.12}$$

valid in the range $3000 \le Re \le 10^5$. The following equation valid in the range of $3000 \le Re \le 10^6$ is also often used for design.

$$f = 0.184\, Re^{-0.2} \tag{5.13}$$

Drew et al. (1932) proposed the following equation:

$$f = 0.0056 + 0.5\, Re^{-0.32} \tag{5.14}$$

valid in the range $3000 \le Re \le 3\text{x}10^6$. The following equation proposed by Nikuradse (1933)

$$\frac{1}{\sqrt{f}} = 0.86 \ln(Re\sqrt{f}) - 0.8 \tag{5.15}$$

is valid over the entire range of Reynolds number. Colebrook (1938) proposed the following equation

$$\frac{1}{\sqrt{f}} = 0.86 \ln\left(\frac{e/D}{3.7} + \frac{2.51}{Re\sqrt{f}}\right) \tag{5.16}$$

valid for smooth and rough pipes for the whole range of Reynolds number above 3000. The following explicit equation proposed by Filonenko (1948) is a good approximation of Colebrook equation for smooth tube in the range $4 \times 10^3 \leq Re \leq 10^{12}$.

$$f = [1.82 \log(Re) - 1.64]^{-2} \qquad (5.17)$$

An explicit form of the Colebrook equation valid for smooth and rough tubes has been obtained by Selander (1978) for use in computer codes.

$$f = 4\,[3.8 \log(10/Re + 0.2e/D)]^{-2} \qquad (5.18)$$

It may be noted from the above that well established correlations for friction factor do not exist in the transition region between $2000 \leq Re \leq 3000$. Further, in many transients, the flow may change from laminar to turbulent, or vice versa, necessitating a switch of correlations. Numerical calculations, often encounter convergence problems when such switching takes place due to the discontinuity in the friction factor values predicted by the laminar flow and turbulent flow equations. A simple way to overcome this problem is to use the following criterion for switch over from laminar to turbulent flow equation.

$$\text{if } f_t > f_l \text{ then } f = f_t \qquad (5.19)$$

where

f_t and f_l are friction factors calculated by turbulent and laminar flow equations respectively. This procedure, however, causes the switch over from laminar to turbulent flow equation at $Re \approx 1100$. Solbrig's (1986) suggestion to overcome the same is to use friction factor as equal to greater of $(f_t)_{4000}$ and f_l below Reynolds number of 4000. $(f_t)_{4000}$ is the friction factor calculated by the turbulent flow equation at Re = 4000. Effectively this leads to

$$f = (f_t)_{4000} \text{ for } 2000 \leq Re \leq 4000 \qquad (5.20)$$

In addition, a condition to avoid infinite friction factor is required to take care of flow stagnation (i.e. $Re \approx 0$).

5.3.3.1.2. Diabatic single-phase flow

Generally isothermal friction factor correlations are used with properties evaluated at the film temperature $T_f = 0.4\,(T_W - T_b) + T_b$, where T_W and T_b are the wall and bulk fluid temperatures [Knudsen & Katz (1958)]. Sometimes the friction factor for non-isothermal flow is obtained by multiplying the isothermal friction factor with a correction coefficient, F. The correction coefficient accounts for the temperature gradient in the laminar layer and the consequent variation in physical properties of the fluid. The correction coefficient can be expressed as a function of the temperature drop in the laminar layer, ΔT_f as given below:

$$F = 1 \pm C\Delta T_f \qquad (5.21)$$

The negative sign shall be used for heat transfer from wall to the fluid, and

$$\Delta T_f = q/h \tag{5.22}$$

Different values of the constant C are given by different investigators. El-Wakil (1971) gives a value of 0.0025, while Marinelli and Pastori (1973) give a value of 0.001.

An alternative approach is to express the correction factor in terms of the viscosity ratio. This approach is more widely used and the following empirical equation proposed by Leung and Groeneveld (1993) is recommended.

$$F = (\mu_b/\mu_w)^{-0.28} \tag{5.23}$$

where

the subscripts b and w refer to the bulk fluid and wall respectively.

5.3.3.1.3. Adiabatic two-phase flow

A large number of two-phase flow pressure drop correlations can be found in literature. These correlations can be classified into the following four general categories.

(1) Empirical correlations based on the homogeneous model,
(2) Empirical correlations based on the two-phase friction multiplier concept,
(3) Direct empirical models,
(4) Flow pattern specific models.

In addition, computer codes based on the two-fluid or three-fluid models requires correlations for the partitioning of wall friction between the fluids and interfacial friction correlations.

Some of the widely used and often cited correlations in each of the above category are given below.

Homogeneous flow model

In the homogeneous flow model, the two-phase frictional pressure gradient is calculated in terms of a friction factor, as in single-phase flow. The friction factor is calculated using one of the equations given in Section 5.3.3.1.1, with the use of the two-phase viscosity in calculating the Reynolds number. Several models for two-phase viscosity are available some of which are given in Appendix VII.

Many of the models for mixture viscosity do not yield significantly different results. Further, homogeneous models are expected to give good results for high mass flux flows with low and high void fractions where the bubble diameter is small compared to the duct diameter. Hussain et al. (1974) recommend a value of $G = 2700$ kg/m^2-s ($\approx 2 \times 10^6$ lb/h-ft^2) above which homogeneous models are applicable.

Correlations based on the multiplier concept

In this case, the two-phase pressure drop is calculated from the single-phase pressure drop by multiplying with a two-phase friction factor multiplier. The following definitions of two-phase friction multipliers are often used.

$$\phi^2_{LO} = \frac{(dp/dz)_{TPF}}{(dp/dz)_{LO}}; \quad \phi^2_{GO} = \frac{(dp/dz)_{TPF}}{(dp/dz)_{GO}};$$
$$\phi^2_{L} = \frac{(dp/dz)_{TPF}}{(dp/dz)_{L}} \quad \textit{and} \quad \phi^2_{G} = \frac{(dp/dz)_{TPF}}{(dp/dz)_{G}} \qquad (5.24)$$

where

the denominators refer to the single-phase pressure gradient for flow in the same duct with mass flow rates corresponding to the mixture flow rate in case of ϕ_{LO}^2 and ϕ_{GO}^2 and individual phases in case of ϕ_L^2 and ϕ_G^2. Among these, ϕ_{LO}^2 is the most popular friction multiplier. Some of the multiplier based correlations are briefly described in Appendix VIII.

There are many more empirical correlations (other than those in Appendix VIII) given under the multiplier concept, inclusion of all of which is outside the scope of the present report. Care has been taken to include all those correlations which are of interest to current and advanced reactor designs. In passing, it may be mentioned that all of the homogeneous models given in the previous section can also be expressed in terms of a two-phase friction multiplier.

Direct empirical models

In this category, the two-phase friction pressure drop is directly expressed as a function of mass flux, mixture density, length, equivalent diameter, etc. without reference to single-phase pressure drop. Examples in this category are the models proposed by Lombardi-Pedrocchi (1972), Lombardi-Ceresa (1978), Bonfanti et al. (1982) and Lombardi-Carsana (1992). These correlations also specify the use of the homogeneous model for the calculation of the gravitational and accelerational pressure drop. Such correlations are expected to provide accurate values of the calculated total pressure drop rather than the individual components of the pressure drop. Since Lombardi-Carsana is the latest in this series only this correlation is given in Appendix IX.

Flow pattern specific models

In general, two methods are being used to generate flow pattern specific correlations. In the first, empirical correlations are obtained by correlating the data for each flow pattern. In the second method mechanistic models which take into account the distribution of phases in each flow pattern have been developed. Examples of the first approach are those due to Baker [see Govier and Aziz (1972) and Hoogendoorn (1959)] for horizontal flows and Hughmark (1965) for horizontal slug flow. Examples of mechanistic models are those due to Taitel and Dukler (1976a) and Agrawal et al. (1973) for stratified flow; Wallis and Dobson (1973) and Dukler and Hubbard (1975) for slug flow and Hewitt and Hall-Taylor (1970) for annular flow. Some of the empirical and mechanistic models for calculating pressure gradient for horizontal and vertical flows are given in Appendices X and XI respectively.

To apply flow pattern specific correlations, we must also have a method to identify flow patterns. This can be done by the use of flow pattern maps proposed by different authors for horizontal, vertical and inclined flows.

Interfacial friction models

The two-fluid model used in many of the advanced system codes require correlations for interfacial friction in addition to wall friction. Complete description of the models used in computer codes like TRAC-PFI/MOD1 [Liles and Mahaffy (1984)] and RELAP5/MOD3.2 [the RELAP5/MOD3 development team (1995)] are readily available in the open literature. For specific flow patterns, models are proposed by Wallis (1970), Coutris (1989), Putney (1991) and Stevanovic and Studovic (1995). For use in computer codes, it is also essential that such correlations for the various flow patterns be consistent. For example, when the flow pattern changes from bubbly to slug, the interface force predicted at the transition point by correlations for the bubbly and slug flow should be same. A consistent set of interfacial and wall friction correlations for vertical upward flow has been proposed by Solbrig (1986) along with a flow pattern map for use in two-fluid models (Appendix XII).

5.3.3.1.4. Diabatic two-phase flow

The correlations discussed so far are applicable to adiabatic two-phase flow. The effect of heat flux on two phase pressure drop has been studied by Leung and Groeneveld (1991), Tarasova (1966) and Koehler and Kastner (1988). Tarasova (1966) observed that two phase friction pressure drop is higher in a heated channel compared to that in an unheated channel for same flow condition. However, Koehler and Kastner (1988) concluded that two phase pressure drops are same for heated and unheated channels. Studies conducted by Leung and Groeneveld indicate that the surface condition is significantly influenced by heat flux. Effective surface roughness increases due to the formation of bubbles at heated surface leading to larger pressure drop. They concluded that for the same flow conditions, the two phase multiplier is larger for low heat flux than high heat flux. They further observed that maximum value of two phase multiplier is obtained when heat flux approaches critical heat flux value. In the absence of established procedure to take the affect of heat flux into account the following procedure for calculation of two phase diabatic pressure drop is generally followed.

For diabatic two-phase flow, the quality, void fraction, flow pattern, etc. change along the heated section. To calculate the pressure drop in such cases, two approaches are usually followed. In the first approach, the average ϕ_{LO}^2 is calculated as:

$$\phi_{LO}^2 = \frac{1}{L}\int_0^L [\phi_{LO}^2(z)]dz \qquad (5.25)$$

The approach can be used in cases where the $\phi_{LO}^2(z)$ is an integrable function. Numerical integration is resorted to in other cases. An example of such an approach is proposed by Thom (1964). Thom has derived average values of $\phi_{LO}^2(z)$ which are reproduced in Table 5.4. Similar integrated multiplication factors for diabatic flow as a function of outlet quality are also available for the Martinelli-Nelson method. Thom has also obtained multiplication factors for calculating the acceleration and elevation pressure drops for diabatic flow in this way.

In the second approach the heated section is subdivided into a large number of small segments. Based on average conditions (i.e. x_i, α_i and flow pattern) in that segment, the pressure drop is calculated as in adiabatic two-phase flow using one of the models described previously.

TABLE 5.4. VALUES OF FRICTION MULTIPLIER FOR DIABATIC FLOW [THOM (1964)]

Outlet	Pressure (bar)				
Quality	17.24	41.38	86.21	144.83	206.9
0.000	1.00	1.00	1.00	-	-
0.010	1.49	1.11	1.03	-	-
0.015	1.76	1.25	1.05	-	-
0.020	2.05	1.38	1.08	1.020	-
0.030	2.63	1.62	1.15	1.050	-
0.040	3.19	1.86	1.23	1.070	-
0.050	3.71	2.09	1.31	1.100	-
0.060	4.21	2.30	1.40	1.120	-
0.070	4.72	2.50	1.48	1.140	-
0.080	5.25	2.70	1.64	1.190	1.050
0.100	6.30	3.11	1.71	1.210	1.060
0.150	9.00	4.11	2.10	1.330	1.090
0.200	11.40	5.08	2.47	1.460	1.120
0.300	16.20	7.00	3.20	1.720	1.180
0.400	21.00	8.80	3.89	2.010	1.260
0.500	25.90	10.60	4.55	2.320	1.330
0.600	30.50	12.40	5.25	2.620	1.410
0.700	35.20	14.20	6.00	2.930	1.500
0.800	40.10	16.00	6.75	3.230	1.580
0.900	45.00	17.80	7.50	3.530	1.660
1.000	49.93	19.65	8.165	3.832	1.740

In many cases, the pressure drop is to be calculated for a component with subcooled inlet flow (for example rod bundles in BWRs). For such cases a single-phase friction model is used in the non-boiling part of the test section and a two-phase model is used in the boiling zone. A model is also required to establish the onset of boiling in such cases. Usually, the thermal equilibrium model is used. But in many cases a model taking into account the effect of subcooled boiling is also used. The Saha and Zuber (1974) model is preferred by many investigators for this purpose [Marinelli & Pastori (1973), Vijayan et al. (1981), Snoek & Ahmad (1983)].

Comparison of diabatic two-phase pressure drop predictions with experimental data by Snoek and Leung (1989) showed that the Saha & Zuber model is not adequate to predict the onset of nucleate boiling (ONB) in 37-rod bundles with non-uniform heat flux due to enthalpy maldistribution in the subchannels. They found that the Saha and Zuber correlation overpredicted the single-phase region length by as much as 50%. They modified the Saha & Zuber correlation for the case of Peclet number >70 000 as:

$$x_{ONB} = -568\frac{q}{Gh_{fg}} \tag{5.26}$$

Knowing the thermodynamic quality, x_e, the true quality, x_t, is obtained as:

$$x_t = x_e - x_{ONB} \exp\left(\frac{x_e}{x_{ONB}} - 1\right) \tag{5.27}$$

They also tested this correlation with the available data and found that better agreement is obtained in the prediction of single-phase length in case of nonuniform heat flux. With uniform heat flux, however, the single-phase length is underpredicted to some extent.

5.3.3.2. Annulus

Correlations for circular pipe are normally used for the calculation of single phase pressure drop in annulus using the hydraulic diameter concept. For two-phase pressure drop, the same concept is expected to be applicable. The accuracy of this method can be checked by comparison with experimental data. Examples of available experimental data are those due to Adorni (1961), CISE (1963), Moeck (1970), etc.

5.3.3.3. Rod bundle

The rod bundle geometries used in advanced designs differ in several ways. In PWRs and BWRs, the fuel bundles are long (»1.8 to 4.5 m) whereas in CANDU type heavy water reactors short fuel bundles of about 0.5 m are used. Generally grid spacers are used in PWRs and BWRs while split-wart spacers are used in CANDUs. In certain fast breeder reactors wire-wrapped bundles are still used. In PWRs and BWRs, the total pressure drop is obtained by summing up the pressure drop in bare rod bundle and the spacers. For wire-wrapped bundles empirical correlations for the pressure drop in the bundle considering the geometric details of the wire wraps are available. For prototype CANDU type bundles, the total pressure drop is sometimes expressed in terms of an overall loss coefficient due to the closeness of the spacers and the complex geometry of the end plates [Vijayan et al. (1984)] and alignment problem at the junction between two bundles [Pilkhwal et al. (1992)].

5.3.3.3.1. Pressure drop in wire wrapped rod bundles

In the case of wire wrapped rod bundles, the geometry and shape of the system is quite rigid and the development of a general correlation for predicting the pressure drop is a reasonable task. Such a correlation proposed by Rehme (1968 and 1969) is given below:

$$\Delta P = f_R \frac{L}{D_h} \frac{\rho u_R^2}{2} \frac{U_B}{U_G} \tag{5.28}$$

where

$U_B = U_S + U_D$ is the bundle perimeter
$U_G = U_S + U_D + U_K$ is the total perimeter

U_K, U_S and U_D are the shroud perimeter, pins perimeter and wire perimeter respectively. The reference velocity, u_R, is defined as:

$$u_R = u\sqrt{F} \tag{5.29}$$

where

u is the average velocity in the rod bundle

The geometrical factor F depends on the pitch to diameter ratio and on the ratio between the mean diameter and the wire pitch (H).

$$F = \left(\frac{p_t}{D}\right)^{0.5} + \left[7.6 \frac{d_m}{H} \left(\frac{p_t}{D}\right)^2\right]^{2.16} \qquad (5.30)$$

where

d_m is the mean diameter of wire wraps. The reference friction factor f_R is calculated by means of the following correlation based on Rehme's experimental data.

$$f_R = \frac{64}{Re_R} + \frac{0.0816}{Re_R^{0.133}} \qquad \text{for } 2 \times 10^3 \leq Re_R \leq 5 \times 10^5 \qquad (5.31)$$

where

$$Re_R = Re \sqrt{F} \text{ and } Re = (u_R D_h)/\nu \qquad (5.32)$$

These are valid in the range $1.12 < p_t/D < 1.42$ and $6 < H/dm < 45$. Later on Dalle Donne and Hame (1982) extended the validity of the correlation to lower p_t/D ratios by multiplying F with a correction factor C for $p_t/D < 1.03$.

$$C = 1.6 - e^{-\frac{P_t/D-1}{0.05873}} \qquad (5.33)$$

The measurements on wire wrapped bundles performed in ENEA when compared with the general correlation were found to be in very good agreement for a wire pitch of 140 mm. The discrepancy in the whole Reynolds number range was about 4-5 per cent. The agreement for the 160 mm pitch was a little worse, up to 13 per cent which is attributed to measurement uncertainty. Later on, pressure drop measured by ENEA in prototype fuel elements of the PEC reactor were found to be in good agreement with the predictions of Rehme's correlation thus confirming its general validity [Cevolani (1996)].

5.3.3.3.2. Pressure drop in CANDU type fuel bundles

Several short bundles are stacked end to end in CANDU type PHWRs compared to a long single fuel bundle used in PWRs and BWRs. Due to the basic change in design concept some of the problems and geometries are unique to the design.

Snoek & Ahmad (1983)

Snoek & Ahmad suggested the following empirical correlation for friction factor based on experiments on a 6 m long electrically heated horizontal 37 rod cluster.

$$f = 0.05052\, Re^{-0.05719} \qquad \text{for } 108{,}000 \leq Re \leq 418{,}000 \qquad (5.34)$$

Venkat Raj (1993)

Venkat Raj proposed the following equations based on a set of experiments with prototype horizontal 37 rod clusters for PHWRs with split-wart type spacer which includes the junction pressure drop.

$$f = 0.22\, Re^{-0.163} \qquad 10{,}000 \leq Re \leq 1{,}40{,}000 \qquad (5.35)$$

$$f = 0.108\, Re^{-0.108} \qquad 1{,}40{,}000 \leq Re \leq 5{,}00{,}000 \qquad (5.36)$$

5.3.3.3.3. Pressure drop in bare rod bundles

Single-phase

Correlations for circular pipes are commonly used to calculate pressure drop using hydraulic diameter of the rod bundle in the absence of experimental data. Some of the commonly used correlations are:

Kays (1979)

For rod clusters

$$f = f_{cir} K_1 \qquad (5.37)$$

where

K_1 — is provided as a function of p/D (pitch to diameter ratio) based on the work by Diessler and Taylor (1956).

f_{cir} — can be calculated using correlations given for circular pipe.

Rehme (1980)

For non-circular channels

Laminar flow;

$$f\,Re = K \qquad (5.38)$$

where K is a geometry parameter that only depends on the configuration of the channel.

Turbulent flow;

$$\sqrt{(8/f)} = A[2.5 \ln Re\sqrt{(f/8)} + 5.5] - G^* \qquad (5.39)$$

where

the empirical factors A & G* can be determined from the diagrams given in Rehme (1973a)

Grillo and Marinelli (1970)

Grillo and Marinelli proposed the following equation based on their measurements on a 4 × 4 square array rod bundle with rod diameter of 15.06 mm and p/D of 1.283

$$f = 0.1626\ Re^{-0.2} \tag{5.40}$$

Two-phase

In the absence of experimental data, the method used for diabatic two phase flow in Section 5.3.3.1.4 can be used with hydraulic diameter of the bundle in place of pipe diameter. Lombardi-Carsana (1992) (CESNEF-2) correlation discussed in Appendix IX is also applicable for rod bundles. In addition, there are some empirical equations proposed for rod bundles some of which are given below.

CNEN correlation (1973)

$$\Delta p_{TPF} = 1.7205\text{x}10^{-6}\ (L\ M^{0.852})/D_h^{1.242} \tag{5.41}$$

where

M is given by:

$$M = [x v_G + (1 - x) v_L] G^2 \tag{5.42}$$

where

M is in [N/m^2]
L & D_h are in metres,
Δp_{TPF} is obtained in metres of water at 25°C.

This equation is applicable for square array fuel bundles with pitch to diameter ratio = 1.28, D_h = 1.31 cm, peripheral rod-channel gap = 0.55 × pitch, $8 < P < 70$ kg/cm^2 and $680 < G < 2700$ kg/m^{2}s.

Grillo and Marinelli (1970)

$$\xi(G) = \frac{\phi^2_{LO}}{(\phi^2_{LO})_{M-N}} \tag{5.43}$$

$$\xi(G) = 0.56 + 0.315\left(\frac{10^6}{G}\right) \tag{5.44}$$

where $(\phi_{LO}^2)_{M-N}$ is calculated using the Martinelli-Nelson method (Appendix VIII).

Unal (1994)

For rod cluster

$$f = 0.1\ Re_{av}^{-0.3} \tag{5.45}$$

$$Re_{av} = GD/\mu_{av} \tag{5.46}$$

where μ_{av} corresponds to average of inlet and outlet under post CHF dispersed flow condition.

5.3.3.4. Steam generator secondary side

Two-phase pressure drop calculations are important for natural circulation type steam generators. The driving force for natural circulation flow is resisted by pressure losses which oppose the flow. The natural circulation driving force is provided by the difference between the density of the water in the downcomer and that of the steam-water mixture in the heating zone and riser. Calculation of pressure losses in a steam generator is therefore an integral part of evaluating the circulation and flow rate through the heating zone. Pressure drop correlations specific to steam generator tube banks are not readily available. For design and analysis purposes, however, the frictional pressure losses can be calculated by the procedure listed for diabatic two-phase flow discussed in Section 5.3.3.1.4 with the hydraulic diameter of the tube bank used in place of the pipe diameter [ORNL-TM-3578 (1975)].

5.3.4. Local pressure drop

5.3.4.1. Grid spacers

Because of variation and complexity of geometry, it is extremely difficult to establish a pressure loss coefficient correlation of general validity for grid spacers. But methods of calculation reasonably accurate for design purpose can be achieved. For more precise determination of pressure drop across spacers, experimental studies are required. Some correlations used to determine pressure drop across grid spacers are discussed below.

5.3.4.1.1. Single-phase flow

Single-phase pressure drop is calculated using a spacer loss coefficient, K, as:

$$\Delta p = K\ \rho\ V_B^2/2 \tag{5.47}$$

In some cases, it may be possible to obtain a reasonable value of the spacer loss coefficient if its geometry can be approximated to one of those considered in Idelchik (1986). For other cases, the different empirical models for K, described below may be used.

Rehme (1973b)

$$K = C_v\, \varepsilon^2 \tag{5.48}$$

where

$\varepsilon = A_g/A_B$.

For $Re_B > 5 \times 10^4$, $C_v = 6$ to 7 and for $Re_B \leq 5 \times 10^4$ C_v values are given in graphical form as a function of Re_B. Subsequently Rehme (1977) studied the effect of roughness of rod surface on the pressure drop across spacers. Cevolani (1995) proposed $C_v = 5 + 6133Re^{-0.789}$ for square bundles and $\ln(C_v) = 7.690-0.9421 \ln(Re) + 0.0379 \ln^2(Re)$ for triangular bundles with an upper limit of $K = 2$ if the calculated value is greater than 2.

Mochizuki & Shiba (1986)

$$K = 2.7-1.55(\log Re_B-4) \text{ for } Re_B \leq 8 \times 10^4 \tag{5.49}$$

$$K = 1.3 \quad \text{for } Re_B > 8 \times 10^4 \tag{5.50}$$

This correlation is valid only for the specific spacer used for the experimental studies with 37 rod cluster.

Kim et al. (1992)

$$K = (C_d + 2LC_f/t)\, \varepsilon/(1-\varepsilon)^2 \tag{5.51}$$

where

C_d the drag coefficient varies from 0.8 to 1.0 for a thin rectangular plate depending on the aspect ratio of the plate,

C_f the friction coefficient can be obtained from the flat plate flow solution. For turbulent boundary layer preceded by laminar region.

$$C_f = 0.074\,(Re_B\, L/D_h)^{-0.2} - 1740\,(Re_B\, L/D_h)^{-1} \tag{5.52}$$

For fully laminar flow

$$C_f = 1.328(D_h/L)^{0.5}(Re_B)^{-0.5} \tag{5.53}$$

Transition Reynolds number is assumed to be 5×10^5.

5.3.4.1.2. Two phase flow

In general, the homogeneous model or the slip model is used for the estimation of the two-phase pressure drop across grid spacers.

Homogeneous model

$$\Delta p = K(Re_{sat})\, v\, G^2/2 \tag{5.54}$$

where

$K(Re_{sat})$ is the form loss coefficient for single phase flow estimated at the Reynolds number corresponding to the total flow in the form of saturated liquid and v is the specific volume given by

$$v = x\, v_G + (1-x)\, v_L \qquad (5.55)$$

This model may be used when experimental data are not available. Beattie (1973) has provided the following equation to calculate the pressure drop in rod spacers, sudden expansion, etc. if the flow is churn-turbulent at the obstruction.

$$\phi^2_{LO} = [1 + (x\frac{\rho_L}{\rho_G} - 1)]^{0.8}\ [1 + x(\frac{3.5\rho_L}{\rho_G} - 1)]^{0.2} \qquad (5.56)$$

Slip model

According to this model, the form loss coefficient for two phase flow can be obtained from

$$\Delta p_{TPF} = \frac{K_{SPF} G^2}{2\rho} = K_{SPF}\frac{\rho_L}{\rho}\frac{G^2}{2\rho_L} = K_{TPF}\frac{G^2}{2\rho_L} \qquad (5.57)$$

where

ρ is given by

$$\rho = \alpha\rho_G + (1-\alpha)\rho_L\ ; \qquad \alpha = \frac{1}{1+\left(\frac{1-x}{x}\right) S\frac{\rho_G}{\rho_L}}$$

It may be noted that this equation reduces to the homogeneous model if S = 1. Grillo and Marinelli (1970) recommend a value of S = 2 for grid spacers.

Tie plate

Generally, tie plates are used at the ends of rod cluster fuel elements which structurally joins all the fuel pins. Unlike spacers, the flow areas at the downstream and upstream sides of the tie plates are different. Also, these are generally located in the unheated portion of the bundle. Reported studies on pressure drop for the tie plates are few in number. An approximate calculation for design purposes can be made using the contraction and expansion model for local pressure losses. In addition the friction losses in the thickness of the tie plates can be calculated using the hydraulic diameter concept. For two-phase pressure losses, the homogeneous or the slip model described above can be employed in the absence of experimental data.

5.3.4.2. Area changes

Single-phase

The pressure losses due to area changes are calculated by Equation 5.4 with loss coefficients calculated for the relevant geometry from Idelchik (1986).

Two-phase

In general, the irreversible pressure drop due to area changes is estimated from the knowledge of single-phase loss coefficient using the homogeneous model. When details of the slip ratio are available, then the slip model given above can be used.

Sudden expansion

Romey [see Lottes (1961)] expresses the two-phase pressure drop across sudden expansion by the following equation:

$$\Delta p = G^2 A_r^2 \frac{(1-A_r)}{\rho_L}\left\{1 + x\left(\frac{\rho_L}{\rho_G} - 1\right)\right\} \tag{5.58}$$

Beattie (1973) model given above can also be used (Eq. 5.56).

Fitzsimmons (1964) provides the following equation to calculate the pressure change across abrupt expansion

$$\Delta p = \frac{G^2 A_r^2}{\rho_L}\left\{\frac{\rho_L}{\rho_G} x^2 \left(\frac{1}{\alpha_1 A_r} - \frac{1}{\alpha_2}\right)\right\}\left\{(1-x)^2\left(\frac{1}{(1-\alpha_1)A_r} - \frac{1}{(1-\alpha_2)}\right)\right\} \tag{5.59}$$

where

subscripts 1 and 2 refer respectively to the upstream and downstream locations of the abrupt expansion. An assessment carried out by Husain et al. (1974) suggests that better agreement with data is obtained when α_1 and α_2 are calculated by assuming slip flow.

5.3.4.3. Bends and fittings

The single-phase pressure drop due to bends and fittings can be calculated using the appropriate loss coefficients from Idelchik (1986).

Two-phase pressure drop

Chisholm (1969) provides the following general equation for the calculation of two-phase pressure drop in bends and fittings.

$$\frac{\Delta p_{TP}}{\Delta p_L} = 1 + \left(\frac{\Delta p_G}{\Delta p_L}\right) + C\left(\frac{\Delta p_G}{\Delta p_L}\right)^{0.5} \tag{5.60}$$

$$C = \left\{1 + (C_2 - 1)\left(\frac{v_{fg}}{v_G}\right)^{0.5}\right\}\left\{\left(\frac{v_G}{v_L}\right)^{0.5} \left(\frac{v_L}{v_G}\right)^{0.5}\right\} \tag{5.61}$$

where

$v_{fg} = v_G - v_L$, and
C_2 is a constant.

Bends

For bends C_2 is a function of R/D, where R is the radius of curvature of the bend and D is the pipe diameter.

R/D →	1	3	5	7
C_2 for normal bend	4.35	3.40	2.20	1.00
C_2 for bend with upstream disturbance within 50 L/D	3.10	2.50	1.75	1.00

Chisholm provided the above values of C_2 by fitting Fitzsimmons (1964) data.

Chisholm & Sutherland (1969)

For 90° bends: $C_2 = 1 + 35\,N$ (5.62)

For 180° bends: $C_2 = 1 + 20\,N$ (5.63)

N is the number of equivalent lengths used for calculating single-phase pressure drop.

Tees:

$C_2 = 1.75$

Valves:

C_2 = 1.5 for gate valves
= 2.3 for globe valves

Alternatively the homogeneous model may be used.

Orifices:

For separated flow (stratified) at obstruction, Beattie (1973) obtained the following expression for ϕ_{LO}^2.

$$\phi_{LO}^2 = \left\{1 + x\left(\frac{\rho_L}{\rho_G} - 1\right)\right\}^{0.8} \left\{1 + x\left(\frac{\rho_L \mu_G}{\rho_G \mu_L} - 1\right)\right\}^{0.2} \qquad (5.64)$$

5.3.5. Importance of void fraction correlations

Void fraction plays an important role, not only in pressure drop calculation, but also in flow pattern determination and neutron kinetics. All the four components of pressure drop directly or indirectly depend on the void fraction. For certain situations of practical interest, accurate prediction of all the components are required. For example, steady state flow prevails in a natural circulation loop when the driving pressure differential due to buoyancy (i.e. the elevation pressure drop) balances the opposing pressure differential due to friction and acceleration. For natural circulation loops, therefore, the largest contribution to pressure drop arises from the elevation pressure drop. Also, the acceleration pressure drop can be 10–15% of the total core pressure drop. For such cases, accurate estimation of each component of pressure drop is required. Therefore, it is very important to have a reliable relationship for the mean void fraction. Significant deviations are observed between the predicted flow rate using different models for friction and void fraction.

In many experiments with diabatic vertical test sections, the friction pressure loss is obtained as shown below:

$$\left(\frac{dp}{dz}\right)_{TPF} = \left(\frac{dp}{dz}\right)_{m} - g\left(\frac{\alpha}{v_G} + \frac{1-\alpha}{v_L}\right) - G^2\frac{d}{dz}\left(\frac{x^2}{\alpha}v_G + \frac{(1-x)^2}{1-\alpha}v_L\right) \qquad (5.65)$$

where

$(dp/dz)_m$ is the measured pressure drop.

It is observed from the above equation that the void fraction, α, and quality, x, play an important role in deducing the frictional term from the measured static pressure drops. Usually, the acceleration and elevation drops are calculated with the help of a void fraction value, which may not be measured but calculated by a correlation.

The stability predictions of natural circulation loops are also strongly influenced by the friction and mean void fraction model [see Furutera (1986)]. The use of certain friction models can completely mask the stability phenomenon. In coupled neutronic thermalhydraulic calculations, the void fraction plays an important role in the calculation of reactor power [Saphier and Grimm (1992)]. For such calculations, it is essential to use the best models for each component of pressure drop which indirectly also implies the use of the best void fraction model. Hence, it is necessary to make a judicious choice of the void fraction correlations. Some of the commonly used void fraction correlations are described briefly in the following section.

5.3.5.1. Void fraction correlations

In general, the void fraction correlations can be grouped into three; viz.,

(a) slip ratio models,
(b) Kβ models and
(c) correlations based on the drift flux model.

In addition, there are some empirical correlations, which do not fall in any of the three categories. Some of the commonly used correlations in all the above categories are described below.

5.3.5.1.1. Slip ratio models

These models essentially specify an empirical equation for the slip ratio, S ($=u_G/u_L$). The void fraction can, then be calculated by the following equation:

$$\alpha = \frac{1}{1+\left(\frac{1-x}{x}\right) S \frac{\rho_G}{\rho_L}} \tag{5.66}$$

For homogeneous flow, $u_G = u_L$ and S = 1. At high pressure and high mass flow rates the void fraction approaches that of homogeneous flow, and can be calculated by setting S = 1 in the above equation. But usually, the slip ratio is more than unity for both horizontal and vertical flows. For vertical upward flows, the buoyancy also assists in maintaining S > 1. The common slip ratio models are given in Appendix XIII.

5.3.5.1.2. Kβ models

These models calculate α by multiplying the homogeneous void fraction, β, by a constant K. Well-known models in this category are due to Armand (1947), Bankoff (1960) and Hughmark (1965) which are given in Appendix XIV.

5.3.5.1.3. Correlations based on the drift flux model

By far the largest number of correlations for void fraction reported in the literature are based on the drift flux model. The general drift flux formula for void fraction can be expressed as

$$\alpha = \frac{j_G}{C_0[j_G + j_L] + V_{Gj}} \tag{5.67}$$

where

V_{Gj} is the drift velocity (= $u_G - j$, where j is the mixture velocity) and for homogeneous flow C_0 = 1 and V_{Gj} = 0. The various models (see Appendix XV) in this category differ only in the expressions used for C_0 and V_{Gj} which are empirical in nature.

The Chexal and Lellouche (1996) correlation is applicable over a wide range of parameters and can tackle both co-current and counter-current steam-water, air-water and refrigerant two-phase flows. The correlation is used in RELAP5 [the RELAP5 Development Team (1995)] and RETRAN [Mcfadden et al. (1992)] and is given in Appendix XV for steam-water two-phase flow.

5.3.5.1.4. Miscellaneous correlations

There are a few empirical correlations which do not belong to the three categories discussed above. Some of the more common ones are given in Appendix XVI.

Significant differences exist between the void fraction values obtained using different correlations. This necessitates a thorough assessment of the void fraction correlations.

5.3.6. Review of previous assessments

Several assessments of pressure drop and void fraction correlations reported in literature are reviewed and their recommendations summarized in this section.

5.3.6.1. Pressure drop correlations

In general, two different approaches are followed while assessing the predictive capability of pressure drop correlations. In one of these, a particular correlation is chosen and compared with all available two-phase flow pressure drop data disregarding the flow pattern to which the data belong. This approach is adequate for adiabatic flows while assessing correlations valid for all flow patterns, and is followed by Idsinga et al. (1977), Friedel (1979 & 1980), Beattie and Whalley (1982), Snoek & Leung (1989) and Lombardi & Carsana (1992).

In the other approach correlations are chosen for a particular flow pattern and compared against data obtained for that flow pattern. Since flow pattern specific pressure drop data are limited, the flow pattern to which the data belong is identified with a flow pattern map to facilitate the selection of the correlation. This approach requires a pre-assessment of flow pattern maps. Examples of such assessments are those due to Mandhane et al. (1977), Hashizume & Ogawa (1987) and Behnia (1991). Some assessments like those of Dukler et al. (1964) and Weisman & Choe (1976) combine both these approaches.

Some limited assessments for investigating parametric effects are also reported. For example, Simpson et al. (1977) and Behnia (1991) assessed data from large diameter pipes while D'Auria and Vigni (1984) studied the effect of high mass velocity flows. Most assessments employed statistical methods, but the parameter and the correlations chosen for assessment are widely different. Some salient results of these assessments are presented here.

5.3.6.1.1. Homogeneous model

Beattie and Whalley (1982) compared 12 pressure drop correlations including 5 homogeneous models using the HTFS (Heat transfer and fluid flow services) databank containing about 13500 adiabatic data points for steam/water and non-steam water mixtures. This study used roughly about 8400 horizontal flow data points and 5100 vertical flow data points. They used the homogeneous void fraction model to calculate the elevation head for the homogeneous friction models whereas an unpublished void fraction correlation (HTFS-1981) was used for the other models. From this study Beattie and Whalley conclude that the homogeneous model is as good as the others in predicting the two-phase flow pressure drop over the range of parameters considered. The main results of Beattie and Whalley are summarized in Table 5.5.

Idsinga et al. (1977) compared 18 different correlations (4 homogeneous models) against 3500 steam-water pressure drop measurements under both adiabatic and diabatic flow conditions. Most of the data were from vertical pipes ranging in diameter from 0.23 to 3.3 cm. Also, the amount of low mass flux data (less than 300 kg/m^2s) was much less. They used the thermodynamic equilibrium model for the calculation of single-phase length in case of diabatic data. The void fraction model used is the homogeneous model for all homogeneous friction models and for other models, consistent void fraction correlations recommended by the original authors were used. Assessment by Idsinga et al. (1977) shows that best results are obtained from the homogeneous models proposed by Owens (1961) and Cicchitti (1960). Incidentally, these models were also considered for assessment by Beattie and Whalley (1982) and were found to give reasonable results for steam/water flow, although not as good as that of Beattie and Whalley model.

Assessment by Weisman and Choe (1976) showed that the homogeneous models of McAdams (1942) and Dukler et al. (1964) give better results in the homogeneous flow regime ($G > 2712.4$ kg/m^{2}s). Interestingly, the homogeneous model by Dukler (1964) gave consistently good results for all flow regimes except the separated (stratified) flow regime.

5.3.6.1.2. Correlations based on the multiplier concept

Several comparisons of these correlations have been reported previously. One of the earliest assessment was carried out by Dukler et al. (1964). They also compiled a databank consisting of about 9000 data points. They have selected 5 correlations [Baker (1954), Bankoff (1960), Chenoweth and Martin (1955), Lockhart and Martinelli (1949) and Yagi (1954)] for assessment. Their assessment showed that the Lockhart and Martinelli correlation is the best out of the five correlations for two-component two-phase flow.

Idsinga et al. (1977) assessed 14 multiplier based models against 3500 steam-water pressure drop data. The multiplier based models recommended by Idsinga et al. (1977) are the ones due to Baroczy (1966) and Thom (1964).

Friedel (1980) compared 14 pressure drop correlations against 12 868 data points obtained by 62 authors from circular and rectangular channels. Both horizontal and vertical flow adiabatic data in pipes ranging from 1 to 15 cm in diameter were studied. While applying the correlations no distinction is made as to whether they were derived for horizontal or vertical two-phase flow. Overall, the Chisholm (1973) and the Lombardi-Pedrocchi (DIF-1) correlations were found to be the most accurate. However, these two correlations are equivalent and are unexpectedly inadequate for prediction of the measured values in gas/water and gas/oil flows.

TABLE 5.5. MAIN RESULTS OF BEATTIE AND WHALLEY

Fluid used	NDP(a)	Orientation	Recommended Correlation
Non steam-water	7168	horizontal	HTFS, L-M(b) & B-W(c)
Non steam-water	2011	vertical	L-M, HTFS & B-W
Steam-water	1236	horizontal	Dukler et al, B-W & Isbin
Steam-water	3095	vertical	B-W, HTFS, & Friedel

(a) NDP: No. of data points,
(b) L-M: Lockhart-Martinelli (1948),
(c) B-W: Beattie and Whalley (1982).

Friedel (1979) derived two-phase friction pressure drop correlations for horizontal, vertical upflow and downflow based on his databank. He has also compared the predictions of these correlations with the Chisholm (1973) and DIF-2 correlations using an enhanced databank consisting of about 25 000 data points. The data pertain to one-component and two-component mixtures flowing in straight unheated sections with horizontal, vertical upflow and downflow in tubes, annular and rectangular ducts under widely varying conditions. The Friedel correlation was found to be better than the other two.

5.3.6.1.3. Flow pattern specific models

To assess flow pattern specific pressure drop correlations, the first step is to select a flow pattern map applicable to the geometry. Previous review of flow pattern specific pressure drop correlations have been carried out by Weisman and Choe (1976), Mandhane et al. (1977), Hashizume & Ogawa (1987) and Behnia (1991) for horizontal two-phase flow. In the reviews by Mandhane et al. and Behnia, the flow pattern to which the data belong has been obtained with the help of Mandhane's flow pattern map. Hashizume and Ogawa (1987) used a modified Baker map in their assessment. Weisman and Choe used their own flow pattern map.

Using the AGA-API databank (enhanced by the addition of Fitzsimmons (1964), Petrick (1961) and Miropolski (1965) data), Weisman and Choe made a flow pattern specific assessment for horizontal two-phase flow. Their assessment covers four basic flow patterns referred to as separated flow (Stratified flow), homogeneous flow, intermittent (slug) flow and annular flow. The transition criteria used by them are given in Table 5.6.

Based on their assessment the correlations recommended for different flow patterns are given in Table 5.7. Their assessment shows that the scatter obtained using the different correlations (11 in all) for separated flow is substantially large. Ten different correlations were assessed for the homogeneous flow pattern and in this regime, the homogeneous models give better predictions. Most of the correlations tested for the intermittent flow regime were found to give reasonably good values, although the best predictions are obtained with the Dukler et al. (1964) correlation followed by Lockhart-Martinelli correlation. These two correlations are also seen to give consistently good results for annular flow.

TABLE 5.6. TRANSITION CRITERIA FOR HORIZONTAL FLOW (WEISMAN & CHOE)

Flow pattern	Transition criteria
Separated flow	$J_G^* < 2.5 \exp[-12(1-\alpha)] + 0.03\alpha$ where $J^*_G = \rho_G^{0.5} J_G / [g\, D(\rho_L - \rho_G)^{0.5}]$
Annular flow	$G > 10(G_L)^{-0.285} (D/D_c)^{0.38}$ where G_L is in lb/ft^{2}hr and $D_c = 1.5''$ (0.0381 m)
Homogeneous flow	$G \geq 2712.4$ kg/m^{2}s (2×10^6 lb/h ft^2)

Mandhane et al. (1977) compared 16 pressure drop correlations against the University of Calgary Pipe Flow Data Bank containing about 10 500 data points. The data were grouped by predicted flow pattern using the Mandhane et al. (1974) flow pattern map. Each correlation was then tested against all the data points contained within each flow pattern grouping. The correlations recommended by Mandhane et al. are given in Table 5.8. Hashizume and Ogawa (1987) also carried out an assessment of 5 pressure drop correlations using selected data (only 2281 data) from the HTFS databank. This, however, contained some very low mass flux data. In this analysis they have used the modified Baker (1954) map for flow pattern identification. They concluded that their correlation gives the best prediction for refrigerant data.

TABLE 5.7. CORRELATIONS RECOMMENDED BY WEISMAN & CHOE (1976)

Flow pattern	Recommended correlation	No. of correlations tested
Separated flow	Agrawal et al. (1973) and Hoogendoorn (1959)	11
Homogeneous flow	McAdams (1942), Dukler et al. (1964) & Chisholm (1968)	10
Intermittent flow	Dukler (1964), Lockhart-Martinelli (1949) & Hughmark (1965)	7
Annular flow	Dukler (1964) & Lockhart-Martinelli (1949)	6

TABLE 5.8. CORRELATIONS RECOMMENDED BY MANDHANE ET AL. (1977)

Flow pattern	Correlation
Bubble, elongated bubble	Chenoweth and Martin (1956)
Stratified	Agrawal et al. (1973)
Stratified Wavy	Dukler et al (1964)
Slug	Mandhane et al. (1974)
Annular, annular mist	Chenoweth and Martin (1956)
Dispersed bubble	Mandhane et al. (1974)

5.3.6.1.4. Assessment for diabatic flow

With modified Saha and Zuber correlation for the onset of nucleate boiling and the Armand correlation for void fraction, Snoek & Leung (1989) carried out an assessment of 9 different correlations using diabatic pressure drop data from horizontal 37 and 41 rod clusters relevant to CANDU type reactors. The databank consisted of 1217 measurements using either water or refrigerant-12. The correlations compared are the Beattie model (1973), Levy model (1974), Lombardi and Pedrocchi correlation, Martinelli-Nelson separated flow model (1948, 1949), Chisholm and Sutherland model (1969), Chisholm (1983), Reddy et al. (1982) and Beattie and Whalley model (1982). The acceleration pressure drop was calculated using Eqs. 5.6 and 5.7 given in Section 5.3.1 Friedel (1979) correlation was found to predict the experimental results best. Either of the Beattie models were found to yield small errors. Levy model was found to be good for water, but poor for refrigerant-12 data. Results of similar studies for vertical clusters are not available in open literature.

5.3.6.1.5. Parametric effects

Effect of diameter

Simpson et al. (1977) compared six pressure drop correlations with data from large diameter (12.7 and 21.6 cm) horizontal pipes. None of the pressure gradient correlations predicted the measured pressure drop accurately, suggesting the need for considering the effect of pipe diameter. Behnia (1991) has compared seven pressure drop correlations with data generated from large diameter pipe lines ranging in diameter from 7.6 cm to 48.4 cm. In order to identify the flow pattern to which the data belong he has used the Mandhane et al. (1974) flow pattern map. He concludes that the best predictions are obtained using the Beggs and Brill (1973) correlation followed by Aziz et al. (1972) correlation. However, it may be noted that the majority of the data is from large oil pipe lines of about 0.5 m in diameter.

Effect of high mass velocity two-phase flow

An assessment to identify a correlation suitable for predicting friction pressure losses in high velocity two-phase flows (characteristic of critical flow in long channels) has been carried out by D'Auria and Vigni (1984). The pressure drop measurements obtained in the exit nozzle of a pressure vessel was used to assess different pressure drop correlations. The investigations were in the range of pressures from 0.1 to 7.0 MPa and flow rate between 500 to 20 000 kg/m^2s. The assessments were carried out in two-phases; first 17 different correlations were compared with experimental data adopting a homogeneous equilibrium model. Later on a two-velocity model accounting for slip was considered and the correlations were compared with the same experimental data. Results from these studies indicate that practically none of the correlations is able to predict the measured $(\Delta p)_{tot}$ for high values of mass velocities ($G > 8000$ kg/m^2s) while for low values of the same quantity ($G < 2000 kg/m^2s$) nearly all correlations produce results which are within the experimental error band.

5.3.6.2. Assessment of void fraction correlations

Assessment of void fraction correlations are comparatively few in number. The reported assessments are due to Dukler et al. (1964), Friedel (1980), Chexal et al. (1991) Diener and Friedel (1994) and Maier and Coddington (1997). Dukler compared three holdup (i.e. $1-\alpha$) correlations, viz., Hoogendoorn (1959), Hughmark (1962) and Lockhart-Martinelli (1949). Hughmark correlation was found to give the best agreement with data.

Friedel (1980) compared 18 different correlations for mean void fraction using a databank having 9009 measurements of void fraction in circular and rectangular channels by 39 different authors. In his assessment no distinction was made as to whether the correlations were derived for horizontal or vertical two-phase flow. The mean void fraction correlation of Hughmark (1962) and Rouhani (I and II) (1969) were found to reproduce the experimental results considerably better than the other relationships, regardless of the fluid and flow directions. However, Rouhani equation II was found to reproduce the measured values more uniformly over the whole range of mean density. Hence, Friedel recommends Rouhani II relationship.

Chexal, Horowitz and Lellouche (1991) carried out an assessment of eight void fraction models using 1500 steam-water data points for vertical configurations representative of several areas of interest to nuclear reactors such as: (1) high pressure — high flows, (2) high pressure —

low flows, (3) low pressure — low flow, (4) counter current flooding limitation, (5) natural circulation flows and (6) co-current downflows. The data were representative of PWR and BWR fuel assemblies and pipes up to 18 inches in diameter. The correlations assessed and statistical comparision are given in Table 5.9.

Diener and Friedel (1994) made an assessment of mean void fraction correlations using about 24000 data points. The data consists of single-component (mostly water & refrigerant 12) and two-component systems (mostly air-water). In this assessment, they had compiled 26 most often used and cited correlations. These correlations were then checked for the limiting conditions [i.e. zero and unity value of void fraction for single-phase liquid ($x = 0$) and single-phase vapor ($x = 1$)]. Only 13 correlations were found to fulfill the limiting conditions and were selected for further assessment. In this assessment they have not differentiated the data on the basis of flow direction, although, in vertical upward flow the mean void fraction is expected to be lower than in case of horizontal flow under identical conditions (due to larger velocities caused by buoyancy effect). Most of the void fraction correlations reproduce the data with a rather acceptable accuracy. The three best correlations in the order of decreasing prediction accuracy are listed in Table 5.10 for various fluid conditions.

Maier and Coddington (1997) carried out an assessment of 13 wide range void correlations using rod bundle void fraction data. The database consisted of 362 steam-water data points. The data is from level swell and boil-off experiments performed within the last 10–15 years at 9 experimental facilities in France, Japan, Switzerland, the UK and the USA. The pressure and mass flux of the data range frm 0.1 to 15 MPa and from 1 to 2000 kg/m^2-s respectively. Of the 13 correlations considered, 5 were based on tube data. The remaining correlations either are specific to rod bundles or include rod bundle option.

TABLE 5.9. STATISTICAL COMPARISION OF THE EIGHT VOID FRACTION MODELS [CHEXAL, HOROWITZ AND LELLOUCHE (1991)]

Void fraction model	Mean error	Standard deviation
Chexal-Lellouche (1986)	–0.0041	0.049
Liao, Parlos and Griffith (1985)	0.002	0.094
Yeh and Hochreiter (1980)	0.050	0.142
Wilson et al. (1965)	0.013	0.099
Ohkawa and Lahey (1980)	0.025	0.057
Dix (1971)	0.023	0.094
GE ramp (1977)	0.012	0.062
Katoka and Ishii (1982)	0.031	0.101

All 13 correlations except Gardner (1980) are based on drift flux model. Some of the correlations e.g. Ishii (1977), Liao, Parlos and Griffith (1985), Sonnenburg (1989), Takeuchi et al. (1992), Chexal-Lellouche (1992) require iterations to calculate the void fraction. The important results of this assessment are:

(1) Two of the tube based correlations i.e.Liao, Parlos and Griffith (1985) and Takeuchi (1992), produce standard deviations which are as low as the best of the rod bundle correlations.

(2) Comlpex correlations like Chexal et al. (1992), or others requiring iterative solutions produce no significant improvement in mean error or standard deviation compared to more direct correlations of Bestion (1990), Inoue et al. (1993) and Maier and Coddington (1997).

TABLE 5.10. VOID FRACTION CORRELATIONS RECOMMENDED BY DIENER & FRIEDEL (1994)

Fluid	Total number of Data points	Recommended Correlation
Water/air mixture	10991	Rouhani I, Rouhani II, HTFS-Alpha@
1-component mixtures	9827	HTFS-Alpha, HTFS@, Rouhani II
2-component mixtures	14521	HTFS-Alpha, Rouhani I, Rouhani II
2-component mixtures with $G > 100$ kg/m^{2}s	11394	Rouhani II, Rouhani I, HTFS

@ proprietary correlations belonging to HTFS.

5.3.6.3. Limitations of the previous assessment procedure

Most of the well documented assessments of pressure drop correlations have been reviewed in the Section 5.3.6.1. Some limitations of these assessments are given below:

(1) To the best of our knowledge, none of the prior assessments of the two-phase friction correlations concentrate on low mass flux two-phase flows. Analysis using limited number of data (see Vijayan & Austregesilo) shows that there is considerable scatter in the predictions at the low mass fluxes typical of advanced designs. Hence it is desirable to assess the predictive capability of correlations reported in literature for use in the design of advanced reactors where better accuracy of prediction at low mass flux is the criterion of acceptability.

(2) Most assessment of PDCs are based on statistical approach. The correlations selected by a statistical method need not necessarily reproduce the parametric trends as shown by Leung & Groeneveld (1991). Reliable reproduction of parametric trends by PDCs is important to capture certain thermalhydraulic phenomena. An example in this regard is the flow pattern transition instability occurring near slug flow to annular flow transition [Boure et al. (1971)].

(3) Effect of pressure has not been studied separately. It is of interest to study this aspect for the advanced designs.

(4) Effect of pipe diameter needs to be assessed as the pipe diameters in advanced designs can be large. In this case, there is a need to generate additional data as most of the available data on steam water mixture are for small diameter pipes.

(5) Most assessments are for pipe flow data. The only assessment for rod bundles in the open literature is that reported by Snoek and Leung (1989) for CANDU type reactors.

(6) The database for vertical downflow is less extensive.

(7) In deriving certain empirical friction models, a specific void correlation is used to derive the experimental friction pressure drop data. Such empirical models, are to be used with the specified void correlation to predict the pressure drop. Such correlations may not be acceptable for natural circulation reactors where the flow rate is a dependent variable governed by the balance of the driving pressure differential due to elevation and the pressure losses. Therefore, applicability of such correlations needs to be assessed for natural circulation flow.

(8) To our knowledge, assessment of flow pattern specific pressure drop correlations for vertical flow are not reported so far. For the assessment of flow pattern specific correlations, the flow pattern to which the data belong is identified with the help of a flow pattern map which is different for different orientations of the duct. Therefore, separate assessments are required for identifying the best flow pattern map.

5.3.7. Proposed assessment procedure for diabatic vertical flow

For adiabatic vertical flows, the gravitational pressure drop is significant and therefore a void fraction correlation is necessary to derive the experimental friction pressure drop from the measured total pressure drop. For diabatic vertical two-phase flows with subcooled inlet conditions, which is relevant to nuclear reactors, a model for the onset of nucleate boiling is necessary in addition to void fraction correlation. This suggests that the frictional Pressure Drop Correlations (PDCs) cannot be assessed in isolation. In fact, a rational assessment of PDCs for diabatic flow requires a preassessment of models for onset of nucleate boiling (ONB), void fraction and flow pattern transitions. Therefore, a rational assessment procedure consists of the following steps:

(1) To review the literature and compile a set of correlations for ONB, void fraction, flow pattern and pressure drop,

(2) To compile a databank consisting of raw data for ONB, void fraction, flow patterns and pressure drop for forced and natural circulation conditions of one-component two-phase flow,

(3) Assessment of models for ONB, void fraction, flow pattern transitions and pressure drop.

This assessment also aims to investigate the parametric effects due to mass flux, pressure, quality, diameter, flow direction and geometry relevant to the advanced designs. An assessment is in progress in BARC. Some of the results available at this stage are given below.

5.3.8. Results of assessment

5.3.8.1. Compilation of databank

Several databanks exist for the pressure drop in two-phase flow. Examples are those due to Dukler et al. (1964), Friedel (1980), AGA-API, University of Calgary multiphase pipe flow databank, HTFS databank and MIDA [Brega et al. (1990)]. A databank has been compiled by Friedel (1994) for void fraction. Some databanks for flow patterns are also available. These databanks are not available to us at present and therefore a *t*wo-*p*hase *f*low *d*ata *b*ank (TPFDB) consisting of raw experimental data on the following phenomena is being compiled.

(a) Adiabatic and diabatic pressure drop in ducts of various geometry,
(b) Void fraction,
(c) Flow patterns,
(d) Flow pattern specific pressure drop.

In this compilation, special emphasis is given to steam-water flows although some data on air-water and refrigerant two-phase flows are included. The databank is being updated continuously. Currently, this databank consists of about 4000 data on pressure drop, 5000 data on void fraction, 3000 data on flow pattern and 500 data on flow pattern specific pressure drop. The sources from where the original data were compiled are shown in Appendix XVII.

5.3.8.2. Assessment of void fraction correlations

An assessment of the void fraction correlations given in Section 5.3.5.1 was carried out using a part of the void fraction data (about 3300 entries) contained in the TPFDB. The data used for assessment pertains to vertical upward flow of steam-water mixture in circular, annular and rectangular channels. Further details of the assessment are given in Appendix XVIII.

The present assessment showed that Chexal-Lellouche correlation performs better than other correlations. Clearly, all the statistical parameters considered above are minimum for this correlation, followed by Hughmark, Modified Smith and Rouhani correlations (Table 5.11). Previous assessments by Dukler et al. (1964) and Friedel (1980) have also shown that the Hughmark correlation to be the best. Assessment by Diener and Friedel (1994) have shown the Rouhani correlation to be among the best three correlations for predicting void fraction.

A generic problem of all good correlations mentioned above except Modified Smith correlation is that they overpredict the void fraction. This is clear from the mean error given in the table-11, which is positive for almost all the correlations (except Nabizadeh and Modified Smith correlations). Among the top four correlations only the Chexal-Lellouche and the modified Smith correlations satisfy the three limiting conditions (i.e. at $x = 0$, $\alpha = 0$; at $x = 1$, $\alpha = 1$ and at $P = P_{crit}$; $\alpha = x$) over a wide range of parameters (see also Appendix XVIII). Therefore, these correlations may be used in computer codes used for thermalhydraulic analysis.

5.3.8.3. Assessment of flow pattern maps for vertical upward two-phase flow

A large number of flow pattern maps are found in the literature. Many of these are based on experiments. Examples are those due to Griffith and Wallis (1961), Hosler (1967), Spedding and Nguyen (1980) and Weisman and Kang (1981). Since such flow pattern maps are based on limited data, these cannot be assumed to be of general validity. Therefore, theoretical flow pattern maps have been proposed by a few authors. In such maps, the transition criteria are physically based and can be considered to be of general validity. Examples of such maps are those proposed by Taitel et al. (1980), Mishima-Ishii (1984), Solbrig (1986), Bilicki and Kestin (1987) and McQuillan and Whalley (1985). In the present assessment, only three theoretical flow pattern maps for vertical upward flow, proposed by

Taitel et al. (1980), Mishima and Ishii (1984) and Solbrig (1986) are considered as they form the basis of the flow pattern maps used in computer codes for the thermal-hydraulic analysis of nuclear reactors.

A fairly large number of flow regimes are reported in literature. Examples are bubbly, dispersed bubbly, slug, churn, annular, wispy annular, wavy annular, annular mist, spray annular, droplet flow, etc. However, most investigators categorised the flow pattern data into mainly three regimes. These are the bubbly, slug and annular flow regimes. Even computer codes like RELAP5 consider only these as independent flow regimes. Therefore, in our assessment only these three flow patterns are considered. Corresponding to these three patterns the relevant transitions are bubbly-to-slug and the slug-to-annular.

Detailed results of this assessment are given in Appendix XVIII. Table 5.12 shows a summary of the comparison of the data with bubbly-slug together with slug — annular transition criteria. The characterization of bubbly flow data using the different transition criteria yield comparable results. Since it uses $\alpha = 0.52$, 95% of all bubbly flow data is characterized as bubbly by the Solbrig criterion. However, a large amount of slug flow data also fall in the bubbly flow regime.

The slug-annular transition criteria together with bubbly-slug transition criteria are required to assess the slug flow data. Table 5.13 shows the results of such an assessment. As seen all the criteria fare badly in characterizing slug flow data even though the Solbrig criterion I is somewhat better than others.

TABLE 5.11. COMPARISON OF VARIOUS VOID FRACTION CORRELATIONS

Correlation name	mean error (%)	absolute mean error (%)	r. m. s. error (%)	standard deviation (%)
Chexal-Lellouche	5.10	15.25	22.74	22.16
Hughmark	6.85	16.72	23.81	22.60
Modified Smith	–5.44	18.13	24.19	23.58
Rouhani	10.76	18.42	25.97	23.64
Zuber-Findlay	11.20	19.32	26.15	23.64
Bankoff	9.08	19.21	26.58	24.98
Osmachkin	1.32	18.91	26.59	26.56
Bankoff-Jones	12.50	20.78	27.95	25.00
Thom	6.72	21.11	28.88	28.08
Nabizadeh	-21.17	24.40	30.00	21.35
Armand	21.54	27.75	34.75	27.27
GE-Ramp	27.30	32.60	39.10	28.08
Bankoff-Malnes	30.98	36.57	44.15	31.45
Dix	17.81	39.92	48.52	45.14
Homogeneous model	44.90	49.03	55.51	32.65

TABLE 5.12. CHARACTERIZATION OF BUBBLY FLOW DATA USING THE VARIOUS TRANSITION CRITERIA

Item	Taitel et al.	Mishima-Ishii	Solbrig
PBB*	72.3	77.7	95.1
PBS+	21.1	17.8_	4.9
PBA**	6.7	4.6	0.0
PSB@	13.0	17.7	39.6
PAB#	0.7	1.6	4.0

*PBB:	Percentage of bubbly data characterized as bubbly;
+PBS:	Percentage of bubbly data characterized as slug;
** PBA:	Percentage of bubbly data characterized as annular;
@PSB:	Percentage of slug data characterized as bubbly;
# PAB:	Percentage of annular data characterized as bubbly.

TABLE 5.13. CHARACTERIZATION OF SLUG FLOW DATA USING VARIOUS TRANSITION CRITERIA

Item	Taitel et al.	Mishima-Ishii	Solbrig I	Solbrig II
PSS*	40.2	43.2	46.6	34.4
PSB	13.0	17.7	39.6	24.6
PSA**	47.0	39.2	13.5	40.5
PBS	21.1	17.8	4.9	11.8
PAS#	9.4	16.0	47.8	12.1

* PSS:	Per cent of slug data characterized as slug;
# PAS:	% of annular data characterized as slug;
** PSA:	Per cent of slug data characterized as annular.

TABLE 5.14. CHARACTERIZATION OF ANNULAR FLOW DATA WITH VARIOUS TRANSITION CRITERIA

Item	Taitel et al. (1980)	Mishima-Ishii (1984)	Solbrig I (1986)	Solbrig II (1986)
PAA*	90.1	82.7	48.1	85.5
PAS	9.4	16.0	47.8	12.1
PAB	0.7	1.6	4.0	2.4
PBA	6.7	4.6	0.0	0.7
PSA	47.0	39.2	13.5	40.5

· PAA: Percentage of annular data characterized as annular.

Limiting our attention to only the characterization of annular flow data shown in Table 5.14, Taitel et al. Mishima-Ishii and the Solbrig II criterion are found to perform well. However, an acceptable criterion shall not characterize slug flow data as annular and that is where all the three criteria fail.

5.3.8.4. Assessment of pressure drop correlations

A part of the pressure drop data from TPFDB for vertical upward two-phase flow in different geometries has been assessed against some of the correlations described earlier in this report. In the present assessment 2156 data points collected from literature for diabatic steam-water flow were assessed against the correlations listed in Table 5.15. Excepting Chisholm and Turner-Wallis the other correlations belong to the homogeneous model. The assessment is based on Colebrook equation for single-phase friction factor, Zuber-Findlay (1965) correlation for void fraction and Saha and Zuber (1974) model for the onset of nucleate boiling. The results are also given in Table 5.15. The table shows that the Chisholm correlation is the one with least R.M.S. error (37%) and least standard deviation (28%) followed by the homogeneous model given by Dukler et al. (1964) with 48% R.M.S. error and 46% standard deviation which suggests that the simple homogeneous models can give reasonable predictions for design purposes. Earlier assessment by Friedel (1980) had shown that the Chisholm (1973) correlation to be most accurate for adiabatic steam-water flow. Prior assessment by Weisman and Choe (1976) showed that the Dukler et al. (1964) gave consistently good results for all flow regimes.

5.4. COMPARISON OF CORRELATIONS AS THEY STAND IN CODES

Reference is made hereafter to system codes used in the safety and design analysis of nuclear power plants. The attention is focused toward RELAP5 and CATHARE owing to the direct experience gained in the use of these codes. The physical phenomenon addressed is the wall-to-fluid (steam and/or liquid) pressure drop excluding other phenomena that may contribute to the overall (steady state or transient) pressure drop.

TABLE 5.15. COMPARISON OF PRESSURE DROP CORRELATIONS

Correlation	Mean error (%)	R.M.S. error%	Standard Deviation %
Dukler et al. (1964)	12	48	46
McAdams (1942)	20	54	50
Beattie & Whalley (1982)	21	55	51
Cicchittie (1960)	31	65	57
Chisholm (1973)	24	37	28
Turner-Wallis (1965)	21	61	57

The comparison among correlations as they stand in the codes, implies two different steps:

(a) description of the physical models or constitutive equations or closure equations implemented in the codes;

(b) comparison among results produced by the code in terms of pressure drops, eventually including experimental data.

The item a) constitutes the objective of the Section 5.4.1, while item b) is addressed in the following discussion.

The calculation (better, the results of calculations) of pressure drop by system codes is a function of different types of parameters including :

- nodalization details,
- user assumptions,
- physical models for wall-to-fluid pressure drops (Section 5.4.1),
- general code hydraulic model and coupling with physical models other than pressure drops (e.g. heat transfer coefficient),
- numerical structure of the code.

The role of each set of parameters may be extremely different in the various code applications; i.e. user assumptions may be very important in one situation and (almost) not important in the another case; clearly, physical models are always important.

A huge amount of comparison among calculation results by system codes (including comparison with experimental data), is provided in the open literature (e.g. International Standard Problems organized by OECD/CSNI or Standard Problem Exercise organized by the IAEA). In the case of natural circulation, a detailed comparison among system codes, including evaluation of the effects of nodalization details, of boundary and initial conditions and of user choices can be found in D'Auria and Galassi (1992). In the framework of the present CRP some presentations focused on this item too [D'Auria and Frogheri (1996)].

Considering all of the above, it was preferred not to include results of time trends predicted by the code.

5.4.1. Physical models in system codes

The attention is focused hereafter to the two-phase wall-to-fluid friction in RELAP5/MOD3.2 [the RELAP5 Development Team (1995)] and CATHARE 2 v1.3 [Houdayer et al. (1982)] codes.

5.4.1.1. RELAP5

The wall friction model is based on the Heat Transfer and Fluid Flow Service (HTFS) modified Baroczy correlation, [see Chaxton et al (1972)]. The basic equation is

$$\left(\frac{dP}{dz}\right)_{2\phi} = \phi_L^2\left(\frac{dP}{dz}\right)_L = \phi_G^2\left(\frac{dP}{dz}\right)_G = \frac{1}{2D}\left\{\lambda_L \rho_L(\alpha_L u_L)^2 + C\left[\lambda_L\rho_L(\alpha_L u_L)^2 \lambda_G\rho_G(\alpha_G u_G)^2\right]^{1/2} \right.$$
$$\left. + \lambda_G \rho_G(\alpha_G u_G)^2 \right\} \tag{5.68}$$

where
$2 \leq C = -2 + f_l(G)\, T_l(\Lambda,G)$

where

$f_l(G) = 28 - 0.3\sqrt{G}$;
$T_l(\Lambda,G) = \exp\left[\{\log_{10}\Lambda + 2.5\}^2/\{2.4 - G(10^{-4})\}\right]$, and
$\Lambda = (\rho_G/\rho_L)(\mu_L/\mu_G)^{0.2}$

The same derivation implies the use of the Lockhart-Martinelli parameter, i.e. Eqs 1 to 3 in the Appendix VIII.

The partition between contributions to the total pressure drop due to liquid and steam is obtained following the theoretical basis proposed by Chisholm using the Z parameter defined as:

$$Z^2 = \frac{\lambda_L(Re_L)\rho_L u_L^2 \dfrac{\alpha_{LW}}{\alpha_L}}{\lambda_G(Re_G)\rho_G u_G^2 \dfrac{\alpha_{GW}}{\alpha_G}} \tag{5.69}$$

such that

$$\tau_L p_L = \alpha_L \left.\frac{dP}{dz}\right|_{2\phi} \left\{\frac{Z^2}{\alpha_G + \alpha_L Z^2}\right\} \tag{5.70}$$

$$\tau_G p_G = \alpha_G \left.\frac{dP}{dz}\right|_{2\phi} \left\{\frac{Z^2}{\alpha_G + \alpha_L Z^2}\right\} \tag{5.71}$$

In the last formulae (other than the already defined quantities) p_L and p_G are the section perimeters contacting with liquid and steam, respectively; in addition, α_{LW} and α_{GW} are the liquid and the vapor volume fraction respectively, in the wall film:

$$p_L/p = \alpha_{LW} \tag{5.72}$$

$$p_G/p = \alpha_{GW} \tag{5.73}$$

These are defined from the flow regime maps, on the basis of what can be referred as RELAP5 approach [the RELAP5 Development Team (1995)].

The single phase coefficient (Darcy-Weisbach friction factor) is computed from correlations for laminar and turbulent flows with interpolation in the transition regime. The laminar zone coefficient is obtained from the well known "64/Re" formula. The turbulent friction factor is obtained from the Zigrang-Sylvester approximation, [Zigrang and Sylvester (1985)], that is introduced into the already discussed Colebrook correlation. The transition region is computed by a linear interpolation that, again, can be reported as RELAP5 approach. Finally the heated wall effect is accounted for, by introducing the correlation adopted in the VIPRE code [Stewart (1985)].

5.4.1.2. CATHARE

In relation to two-phase wall-to-fluid friction, a simpler approach is included in the CATHARE Code [Bestion (1990)]. The complex interaction of this model with terms included in other code models (e.g. dealing with momentum transfer : interfacial friction, stratification criterion, drift velocity, droplet diameter) should be recalled: the overall result of the code predicted pressure drop comes from the combination of the effects of all the above mentioned models.

The wall friction is computed from the following formula (the index "K" may indicate either the liquid phase, K = L, or the vapor phase, K = G):

$$\tau_{WK} = -C_K C_{FK} \rho_K \frac{u_K |u_K|}{2} \tag{5.74}$$

where

C_{FK} is the single-phase friction coefficient

$$C_{FK} = C_{FK}(Re_k) \text{ with } Re_K = \alpha_K \rho_K u_K D_H / \mu_k \tag{5.75}$$

and C_K is the two phase flow multiplier deduced from the experiments.

In the case of stratified flow, this is the relative fraction of the wettable perimeter occupied by the phase K; C_K is only a function of the void fraction. In the other flow patterns, the vapour friction is assumed as negligible and only the liquid-to-wall friction is computed. This is assumed true in all cases except the case of very high void fraction. Specifically, the Lockhart-Martinelli (Appendix VIII) correlation for liquid was adopted for pressure below 2 MPa; for pressure larger than this value a slightly different correlation was adopted which corrects the pressure effect.

This approach was demonstrated to be acceptable with the exception of the situation of high quality in the annular-mist flow regime. A special correlation for $C_{K=L}$ is developed in such a case. It should be mentioned that an extensive experimental database was utilized to demonstrate the validity of the approach.

5.5. FINAL REMARKS

The performed activity gave an idea of the difficulty in synthesizing the current understanding of a fundamental phenomenon in thermohydraulics: the occurrence and the modelling of various components of pressure drop. Making only reference to the modelling, different approaches can be pursued for calculating friction pressure drops. In addition, a number of correlations, different from each other, have been developed and are currently in use. The areas and the modalities of application of the correlations are also different; in this context, system geometry (e.g. tubes, bundles), fluid status (single-phase, two-phase with or without interaction of phases), flow type (transient, steady state, fully developed or not), flow regime (e.g. in two-phase flow, bubbly or annular flow), can be distinguished. This makes it difficult to identify an 'agreeable' (or widely accepted) approach or to recommend a particular one.

The recommendations should also suit the objectives and the framework of the use of the correlations. Requirements of subchannel analysis codes and system codes should be distinguished. Detailed plans for future development are outside the purpose of the CRP, specifically the need to distinguish between the various applications. However, a few generic requirements that should be the basis of any future development are listed below.

(a) To identify the conditions for a suitable experiment (i.e. quality of facility design, of test design, of instrumentation and of recorded data)
(b) To identify "reference data sets"
(c) To define acceptable errors (as a function of application)
(d) To compare code and/or correlation results with selected "reference data sets".

In addition, a few specific requirements which need to be considered for future work are listed below.

The correlations selected based on assessment by statistical method need not necessarily reproduce the parametric trends. Therefore, future assessment should also examine the parametric trends for mass flux, pressure, quality and diameter.

Most of the reported assessments are for adiabatic pipe flow data. Assessment of pressure drop correlations for diabatic flow requires pre-assessment of the models for the on-set of boiling and void fraction. For flow pattern specific pressure drop correlations, a pre-assessment of flow pattern transition criteria is also required.

Only limited data are available for complex geometries like rod bundles, grid spacers, tie plates, etc. in the open literature. More data are required in this area.

The available database in the open literature is limited and further work is required to generate more pressure drop data for the following range of parameters:

Low (<500 kg/m^2 s) and high (>8000 kg/m^{2}s) mass flux two-phase flow
Large diameter pipe (>70 mm)
Low pressure (<10 bar)
Vertical down flow.

Simultaneous void fraction measurement is required along with pressure drop measurement to calculate individual components of pressure drop. The availability of flow pattern specific pressure drop data is very limited. More data are required to be generated in this area.

As final remarks, from a methodological point of view, we can limit ourselves to list the following various approaches for modelling pressure drops that can be considered when developing advanced thermohydraulic models (capabilities intrinsic to CFD — Computational

Fluid Dynamics or DNS — Direct Numerical Simulation are excluded from the present review) suitable for system codes.

Two-phase flow multiplier (developed having a boiling channel as reference): An average value of the two-phase pressure drop can be calculated. Users must be aware of the conditions under which the correlations are developed or tested (e.g. length of the channel, consideration of acceleration pressure drops, etc.).

Interfacial drag: The lack of knowledge of the interfacial area may noticeably lower the quality of such an approach.

Use of drift flux: The calculation of void fraction, based on correlations not tuned to the calculation of pressure drops may limit the validity of the approach.

Use of 6-equation model: The same observations as above applies here.

Calculation of pressure drop considering subchannels: Lack of appropriate knowledge of two- or three-dimensional flows, may limit the validity of the approach.

REFERENCES TO CHAPTER 5

ADORNI, et al., 1961, Results of Wet Steam Cooling Experiments: Pressure Drop, Heat Transfer And Burnout Measurements in Annular Tubes with Internal and Bilateral Heating, CISE-R-31.

ADORNI, N., GASPARI, G.P., 1966, Heat Transfer Crisis and Pressure Drop with Steam-Water Mixtures, CISE-R-170.

AGRAWAL, S.S., GREGORY, G.A., GOVIER, G.W., 1973, An analysis of stratified two-phase flow in pipes, Can. J. Chem. Eng. **51**, 280-286.

ARMAND, A.A., TRESCHEV, G.G., 1947, Investigation of the resistance during the movement of steam-water mixtures in heated boiler pipes at high pressure, Izv. Ves. Teplotekh. Inst. **4**, 1–5.

AZIZ, K., GOVIER, G.W. FOGARASI, M., 1972, Pressure drop in wells producing oil and Gas, J. Can. Petroleum Technol, 38–48.

BAKER, O., 1954, Simultaneous flow of oil and gas, Oil Gas J. **53**, 185.

BANKOFF, S.G., 1960, A variable density single-fluid model for two-phase flow with particular reference to steam-water flow, J. Heat Transfer **82**, 265–272.

BAROCZY, C.J., 1966, A systematic correlation for two-phase pressure drop, Chem. Eng. Progr. Symp. Ser. **62**, 232–249.

BEATTIE, D.R.H., 1973, "A note on calculation of two-phase pressure losses", Nucl. Eng. Design, 25, 395–402.

BEATTIE, D.R.H. WHALLEY, P.B., 1982, A simple two-phase frictional pressure drop calculation method, Int. J. Multiphase Flow **8**, 83–87.

BECKER, K.M., HERNBORG, G., BODE, M., 1962, "An experimental study of pressure gradients for flow of boiling water in vertical round ducts", (part 4) AE-86.

BEGGS, H.D., BRILL, J.P., 1973, A study of two-phase flow in inclined pipes, J. Petroleum Technol **25**, 607–617.

BEHNIA, M., 1991, Most accurate two-phase pressure drop correlation identified, Oil & Gas J., 90–95.

BENNETT, A.W., HEWITT, G.F., KEARSEY, H.A., KAY, R.K.F., LACEY, P.M.C., 1965, Flow Visualisation Studies of Boiling at High Pressure, UKAEA Rep. AERE-4874.

BERGLES, A.E., CLAWSON, L.G., GOLDBERG, P., SUO, M., BOURNE, J.G., 1965b, Investigation of Boiling Flow Regimes and Critical Heat Flux, NYO-3304-5.

BERGLES, A.E., DOYLE, E.F., CLAWSON; SUO, M., 1965a, Investigation of Boiling Flow Regimes and Critical Heat Flux, NYO-3304-4.

BERGLES, A.E., GOLDBERG, P., CLAWSON, L.G., ROOS, J.P., BOURNE, J.G., 1965c, Investigation of Boiling Flow Regimes and Critical Heat Flux, NYO-3304-6.

BERGLES, A.E., ROOS, J.P., ABRAHAM, S.C., GOUDA, S.C., MAULBETSCH, J.S., 1968a, Investigation of Boiling Flow Regimes and Critical Heat Flux, NYO-3304-11.

BERGLES, A.E., ROOS J.P., 1968b, Investigation of Boiling Flow Regimes and Critical Heat Flux, NYO-3304-12.

BERKOWITZ, L. et al., 1960, Results of Wet Steam Cooling Experiments: Pressure Drop, Heat Transfer and Burnout Measurements with Round Tubes, CISE-R-27.

BESTION D., 1990, The physical closure laws in the CATHARE code, J. Nucl. Eng. Design **124**.

BILICKI, Z., KESTIN, J., 1987, Transition criteria for two-phase flow patterns in vertical upward flow, Int. J. Multiphase Flow **13**, 283–294.

BLASIUS, H., 1913, Mitt. Forsch. Geb. Ing.-Wesen **131**.

BONFANTI, F., CERESA, I., LOMBARDI, C., 1982, "Two-phase densities and pressure drops in the low flow rate region for different duct inclinations", Proc. 7th Int. Heat Transfer Conf. Munich.

BOURE J.A., BERGLES, A.E., TONG L.S., 1971, Review of two-phase flow instability, ASME Preprint 71-HT-42.

BREGA, E., BRIGOLI, B., CARSANA, C.G., LOMBARDI, C., MARAN, L., 1990, "MIDA: Data bank of pressure and densities data of two-phase mixtures flowing in rectangular ducts", 8th Congresso UIT, Ancona.

CARLSON, K.E., et al., 1990, RELAP5/MOD3 Code manual, NUREG/CR-5535, EGG-2596.

CEVOLANI, S., Sept. 5–8, 1995, "ENEA Thermohydraulic data base for the advanced water cooled reactor analysis", 1st Research Co-ordination Meeting of IAEA CRP on Thermohydraulic Relationships for Advanced Water Cooled Reactors, Vienna.

CHAWLA, J.M., 1967, VDI-Forsc-Heft, No. 523.

CHAXTON K.T., COLLIER J.G., WARS J.A., 1972, "HTFS Correlation For Two-Phase Pressure Drop and Void Fraction in Tubes", AERE Rep. R7162.

CHENOWETH, J.M., MARTIN, M.W., 1956, Pressure drop of gas-liquid mixtures in horizontal pipes, Petroleum Eng. **28**, C-42-45.

CHEXAL, B., LELLOUCHE, G., 1986, A Full Range Drift Flux Correlation for Vertical Flows (Revision 1), EPRI-NP-3989-SR.

CHEXAL, B., et al., 1996, Understanding Void Fraction in Steady and Dynamic Environments, TR-106326/RP-8034-14, Electric Power Research Institute.

CHEXAL, B., LELLOUCHE, G., HOROWITZ, J., 1992, A void fraction correlation for generalized applications, Progress in Nuclear Energy **27** 4, 255–295.

CHEXAL, B., HOROWITZ, J., LELLOUCHE, G., 1991, An assessment of eight void fraction models, Nucl. Eng. Design **126**, 71–88.

CHISHOLM, D., 1973, Pressure gradients due to friction during the flow of evaporating two-phase mixtures in smooth tubes and channels, Int. J. Heat Mass Transfer **16**, 347–358.

CHISHOLM, D., 1968, "The influence of mass velocity on friction pressure gradient during steam-water flow", Proc. Thermodynamics and Fluid Mechanics Conf., Institute of Mechanical Engineers, 182, 336–341.

CHISHOLM, D., SUTHERLAND, L.A., 1969, "Prediction of pressure gradient in systems during two-phase flow", Proc. Inst. of Mechanical Engineers Symp. on Two-phase Flow Systems, Univ. of Leeds.

CICCHITTI, A., LOMBARDI, C., SILVESTRI, M., SOLDAINI, G., ZAVATTARELLI, R., 1960, Two-phase cooling experiments: pressure drop, heat transfer and burnout experiments, Energia Nucleare **7**, 407–425.

CISE; 1963, A research programme in two-phase flow.

COLEBROOK, C.F., 1938, Turbulent flow in pipes with particular reference to the transition region between the smooth and rough pipe laws, J. Inst. Civ. Eng. Lond., 133–156.

COOK, W.W., 1956, Boiling Density in Vertical Rectangular Multichannel Sections with Natural Circulation, ANL-5621.

COUTRIS, N., DELHAYE, J.M., NAKACH, R., 1989, Two-phase flow modelling: The closure issue for a two-layer flow, Int. J. Multiphase Flow **15**, 977–983

DALLE DONNE, M., HAME, W., 1982, A Parametric Thermalhydraulic Study of an Advanced Pressurized Light Water Reactor with a Tight Fuel Rod Lattice, KfK 3453-WUR 7059e.

D'AURIA, F., VIGNI, P., 1984, "The evaluation of friction pressure losses in two-phase high velocity flow using non-homogeneous models", 1984 European Two-Phase Flow Group Meeting, Rome.

D'AURIA, F., GALASSI, G.M., 1992, Relevant Results obtained in the analysis of LOBI/mod2 Natural Circulation Experiment A2-77A, US NRC, NUREG/IA-0084.

DIESSLER, G., TAYLOR, M.F., 1956, Analysis of Axial Turbulent Flow Heat Transfer Through Banks of Rods or Tube, Reactor Heat Transfer Conference, TID-7529, Vol. 2.

DIENER, R., FRIEDEL, L., 1994, Proc. German-Japanese Symp. on Multiphase Flow, Karlsruhe, Germany.

DIX, G.F., 1971, Vapour Void Fraction for Forced Convection with Subcooled Boiling at Low Flow Rates, NEDO-10491.

DREW, T.B., KOO, E.C., MCADAMS, W.H., 1932, Trans. AIChE **28**, 56.

DUKLER, A.E., HUBBARD, M.G., 1975, A model for gas-liquid slug flow in horizontal and near horizontal tubes, Ind. Engrg. Chemistry Fundamentals **14**, 337–347.

DUKLER, A.E., WICKS, M., CLEVELAND, R.G., 1964, Frictional pressure drop and holdup in two-phase flow, Part A — A Comparison of existing correlations for pressure drop and holdup, B An approach through similarity analysis, AIChE J. **10**, 38–43 and 44–51.

EL-WAKIL, M.M., 1971, Nuclear Heat Transport, International text book company, 233–34.

EPRI, 1986, Advanced Recycle Methodology Program-02 Documentation, EPRI-NP-4574.

FILONENKO, G.K., 1948, On Friction Factor for a Smooth Tube, All Union Thermotechnical Institute (Izvestija VTI, No 10), Russia.

FITZSIMMONS, D.E., 1964, Two-phase Pressure Drop in Piping Components, Hanford Laboratory Rep., HW-80970.

FRIEDEL, L., 1979, "Improved friction pressure drop correlations for horizontal and vertical two-phase flow", European two-phase flow group mtg, Ispra.

FRIEDEL, L., 1980, Pressure drop during gas/vapor-liquid flow in pipes, Int. Chemical Engineering **20**, 352–367.

GARDNER, G.C., 1980, Fractional vapour content of a liquid pool through which vapour is bubbled, Int. J. of Multiphase Flow. **6**, 399–410.

GENERAL ELECTRIC COMPANY, 1978, Qualification of the one-dimensional core transient model for boiling water reactors, NEDO-24154, 78 Nucl. Eng. Design **290**.

GOVIER, G.W., AZIZ, K., 1972, The Flow of Complex Mixtures in Pipes, Van Nostrand Reinhold Company, New York, 538.

GRIFFITH, P., 1963, The slug-annular flow regime transition at elevated pressure", ANL-6796.

GRIFFITH, P., WALLIS, G.B., 1961, Two-phase slug flow, J. Heat Transfer **83** C (3), 307.

GRILLO, P., MARINELLI, V., 1970, Single and two-phase pressure drops on a 16-ROD bundle, Nuclear Applications & Technol. **9**, 682–693.

HASHIZUME, K., 1983, Flow pattern, void fraction and pressure drop of refrigerant two-phase flow in a horizontal pipe, Int. J. Multiphase Flow **9**, 399–410.

HASHIZUME, K., OGAWA, N., 1987, Flow pattern, void fraction and pressure drop of refrigerant two-phase flow in horizontal pipe — III Comparison of the analysis with existing pressure drop data on air/water and steam/water systems, Int. J. Multiphase Flow **13**, 261–267.

HASHIZUME, K., OGIWARA, H., TANIGUCHI, H., 1985, Flow pattern, void fraction and pressure drop of refrigerant two-phase flow in horizontal pipe — II: Analysis of frictional pressure drop, Int. J. Multiphase Flow **13**, 261–267.

HEWITT, G.F., HALL-TAYLOR, N.S., 1970, Annular Two-Phase Flow, Pergamon Press, New York.

HEWITT, G.F., OWEN, D.G., 1992, "Pressure and entrained fraction in fully developed flow", Multiphase Science and Technology, Vol. 3 (G.F. Hewitt, J.M. Delhaye and N. Zuber, Eds), Hemisphere Publishing, New York.

HOGLUND, B., et al., 1958, Two-phase Pressure Drop in a Natural Circulation Boiling Channel, ANL-5760.

HOOGENDOORN, C.J., 1959, Gas-liquid flow in horizontal pipes, Chem. Eng. Sci **9**, 205–217.

HOSLER, E.R., 1967, Flow Patterns in High Pressure Two-Phase (Steam-Water) Flow with Heat Addition, WAPD-TM-658.

HOUDAYER G., et al., 1982, "The CATHARE code and its qualification on analytical experiments", 10th Water Reactor Safety Information Mtg, Washington.

HUGHMARK, G.A., 1965, Holdup and heat transfer in horizontal slug gas-liquid flow, Chemical Eng. Sci. **20**, 1007–1010.

HUSSAIN, A., CHOE, W.G., WEISMAN, J., 1974, The applicability of the homogeneous flow model to pressure drop in straight pipe and across area changes, COO-2152-16.

IDELCHIK, I.E., 1979, Hand Book of Hydraulic Resistances, Hemisphere Publishing Company, New York.

IDSINGA, W., TODREAS, N., BOWRING, R., 1977, An assessment of two-phase pressure drop correlations for steam-water mixtures, Int. J. Multiphase Flow **3**, 401–413.

ISHII, M., 1977, One-dimensional Drift-flux Model and Constitutive Equations for Relative Motion Between Phases in Various Two Phase Flow Regimes, ANL-77-47.

INOUE, A., et al., 1993, "In bundle void measurement of a BWR fuel assembly by an X-ray CT scanner: Assessment of BWR design void correlation and development of new void correlation", Proc. ASME/JSME Nuclear Engineering Conf., Vol. 1, 39–45.

ITO, A., MOWATARI, K., 1983, Ring Grid Spacer Pressure Loss Experimental Study and Evaluation Method, American Nuclear Society, 972–978.

JANSSEN, E., KERVINEN, J.A., 1971, Developing Two-phase Flow in Tubes and Annuli, Part-I: Experimental Results, Circular Tubes, GEAP-10341.

JANSSEN, E., KERVINEN, J.A., 1964, Two-phase Pressure Drop in Straight Pipes and Channels: Water-Steam Mixtures at 600 to 1400 psia, GEAP-4616.

JONES, A.B., DIGHT, D.G., 1962, Hydrodynamic Stability of a Boiling Channel, KAPL-2208, Knolls Atomic Power Laboratory.

KATAOKA, I., ISHII, M., 1982, Mechanism and Correlation of Droplet Entrainment and Deposition in Annular Two-Phase Flow, NUREG/CR-2885 and ANL-82-44.

KAYS, W.M., 1975, Convective Heat and Mass Transfer, Tata-McGraw Hill Publishing Company Ltd., New Delhi.

KHARE, R., VIJAYAN, P.K., SAHA, D., VENKAT RAJ, V., 1997, Assessment of Theoretical Flow Pattern Maps for Vertical Upward Two-Phase Flow, BARC/1997/E010.

KIM, NAE-HYUN; LEE, S.K., MOON K.S., 1992, Elementary model to predict the pressure loss across a spacer grid without a mixing vane, Nuclear Technol. **98**, 349–353

KING, C.H., OUYANG, M.S., PEI, B.S., WANG, Y.W., 1988, Identification of two-phase flow regimes by an optimum modeling method, Nuclear Technol. **82**, 211–226.

KOEHLER, W., KASTNER, W., 1988, Two Phase Pressure Drop in Boiler Tubes, Two-Phase Flow heat Exchangers: Thermalhydraulic Fundamentals and Design (S. Kakac, A.E. Bergles E.O. Fernandes, Eds), Kluwer Academic Publishers.

KNUDSEN, J.G., KATZ, D.L., 1958, Fluid Dynamics and Heat Transfer, McGraw-Hill, New York, 178.

LAHEY, R.T., Jr., 1984, Two-phase Flow In Boiling Water Reactors, NEDO-13388, 59–70.

LAHEY, R.T., Jr., LEE, S.J., 1992, "Phase distribution and two-phase turbulence for bubbly flows in pipes", Multiphase Science and Technology, Vol. 3 (G.F. Hewitt, J.M. Delhaye, N. Zuber, Eds), Hemisphere Publishing, New York.

LAHEY, R.T., Jr., SHIRALKAR, B.S., RADCLIFFE, D.W., 1970, Two-phase Flow and Heat Transfer in Multirod Geometries, Subchannel Flow and Pressure Drop Measurements in a 9-Rod Bundle for Diabatic and Adiabatic Conditions, GEAP-13040.

LEUNG, L.K.H., GROENEVELD, D.C., 1991, "Frictional pressure gradient in the pre- and post-CHF heat-transfer regions", Multiphase flows' 91, Tsukuba, 1991, Tsukba, Japan.

LEUNG, L.K.H., GROENEVELD, D.C., AUBE, F., TAPUCU, A., 1993 New studies of the effect of surface heating on frictional pressure drop in single-phase and two-phase flow, NURETH-3, Grenoble, France.

LIAO, L.H., PARLOS, A., GRIFFITH, P., Heat Transfer Carryover and Fall Back in PWR Steam Generators During Transients, NUREG/CR-4376, EPRI NP-4298.

LILES, D.R., MAHAFFY, J.H., 1984, TRAC PF1-MOD1 An advanced best-estimate computer program for pressurised reactor thermal-hydraulic analysis, Los Alamos National Laboratory.

LOCKHART, R.W., MARTINELLI, R.C., 1949, Proposed correlation of data for isothermal two-phase, two-component flow in pipes, Chem. Eng. Prog. **45**, 39–48.

LOMBARDI, C., CARSANA, C.G., 1992, A dimensionless pressure drop correlation for two-phase mixtures flowing upflow in vertical ducts covering wide parameter ranges, Heat and Technol. **10**, 125–141.

LOMBARDI, C., CERESA, I., 1978, A generalized pressure drop correlation in two-phase flow, Energia Nucleare **25** 4, 181–198.

LOMBARDI, C., PEDROCCHI, E., 1972, A pressure drop correlation in two-phase flow, Energia Nucleare **19** 2, 91–99.

LORENZINI, E., SPIGA, M., TARTARINI, P., 1989, "Some notes on the two-phase flow friction multiplier", 7th Seminar on Thermal Non-equilibrium in Two-Phase Flow, Roma.

LOTTES, P.A., FLINN, W.S., 1956, A method of analysis of natural circulation boiling systems, Nuclear Science Eng. **19** 2, 91–99.

LU ZHONGQI ZHANG XI, 1994, Identification of flow patterns of two-phase flow by mathematical modelling, Nuclear Eng. and Design **149**, 111–116.

MAIER, D., CODDINGTON, P., 1997, "Review of wide range void correlations against an extensive database of rod bundle void measurements", Proc. ICONE, 5th Int. Conf. on Nuclear Engineering, Nice.

MALNES, D., 1979 “Slip ratios and friction factors in the bubble flow regime in vertical tubes, KR-110, Kjeller Norway (1966) and A new general void correlation”, European two-phase flow group Mtg., Ispra (1979).

MANDHANE, J.M., GREGORY, G.A., AZIZ, K., 1977, Critical evaluation of friction pressure-drop prediction methods for gas-liquid flow in horizontal pipes, J. Petroleum Technol. **29**, 1348–1358.

MANDHANE, J.M., GREGORY, G.A., AZIZ, K., 1974, A flow pattern map for gas-liquid flow in horizontal pipes, Int. J. Multiphase Flow **1**, 537.

MARCHATERRE, J.F., 1956, The effect of pressure on boiling density in multiple rectangular channels, ANL-5522.

MARCHATERRE, J.F., PETRICK, M., LOTTES, P.A., WEATHELAND, R.J., FLINN, W.S., 1960, Natural and Forced Circulation Boiling Studies, ANL-5735.

MARTINELLI, R.C., NELSON, D.B., 1948, Prediction of pressure drop during forced-circulation boiling of water, Trans. ASME **70**, 695–702.

MARINELLI, V., PASTORI, L., 1973, AMLETO — A Pressure drop computer code for LWR fuel bundles, RT/ING(73)11, Comitato Nazionale Energia Nucleare (CNEN).

MCADAMS, W.H., WOODS, W.K., HEROMAN, R.H., Jr., 1942, Vaporization inside horizontal tubes. II. Benzene-oil mixtures, Trans ASME **64**, 193.

MCFADDEN, J.H., et al., 1992, RETRAN-03: A program for Transient Analysis of Complex Fluid Flow Systems, Volume 1: Theory and Numerics, Electric Power Research Institute, EPRI NP-7450.

MCQUILLAN, K.W., WHALLEY, P.B., 1985, Flow patterns in vertical two-phase flow, Int. J. Multiphase flow **11**, 161–175.

MIROPOLSKI, E.L., SHITSMAN, M.E., SHNEENOVA, R.I., 1965, Thermal Eng. **12** 5, 80.

MISHIMA, K., ISHII, M., 1984, Flow regime transition criteria for upward two-phase flow in vertical pipes, Int. J. Heat Mass Transfer, 723–739.

MOCHIZUKI, Y., ISHII, Y., 1992, “Study of thermalhydraulics relevant to natural circulation in ATR”, Proc. 5th Int. Top. Mtg on Reactor Thermal Hydraulics, Vol. I, Salt Lake City, 127-134.

MOCHIZUKI, H., SHIBA, K., 1986, “Characteristics of natural circulation in the ATR plant”, 2nd Int. Top. Mtg. Nuclear Power Plant Thermal Hydraulics and Operations, Tokyo, 132–139.

MOECK, E.O., 1970, Annular Dispersed Flow and Critical Heat Flux, AECL-3656.

NABIZADEH, H., 1977, Modellgesetze and Parameteruntersuch fur den volumetrichen Dampfgehalt in einer zweiphasen Stroemung, EIR-Bericht, Nr. 323.

NIKURADSE, J., 1932, Mitt. Forsch Geb. Ing.-Wesen, **356**, 1.

OHKAWA, K., LAHEY, R.T., Jr., 1980, The analysis of CCFL using drift-flux models, Nucl. Eng. Design **61**.

ORNL-TM-3578, 1975, Design Guide for Heat Transfer Equipment in Water Cooled Nuclear Reactor Systems, Oak Ridge National Laboratory and Burns and Roe Incorporated, Tenn.

OSMACHKIN, V.S., BORISOV, V., 1970, Paper B4.9, IVth Int. Heat Transfer Conf., Versailes.

OWENS, W.S., 1961, "Two-phase pressure gradient", Int. Developments in Heat Transfer, Part II, ASME.

PETERSON, A.C., Jr., WILLIAMS, C.L., 1975, Flow patterns in high pressure two-phase flow: A visual study of water in a uniformly heated 4 — rod bundle, WAPD-TM-1199.

PETRICK, M., 1961, Two-phase Water Flow Phenomena, ANL-5787.

PETRICK, M., 1962, A Study of Vapour Carryunder and Associated Problems, ANL-6581.

PILKHWAL, D.S., VIJAYAN, P.K., VENKAT RAJ, V., 1992, "Measurement of pressure drop across the junction between two 37 rod fuel bundles for various alignments of the bundles", Proc. 19th National Conf. on Fluid Mechanics and Fluid Power, IIT, Powai, Bombay.

RANSOM, V.H., et al., 1987, RELAP5/MOD2 Code Manual, Volume 1: Code structure, System Models And Solution Methods, NUREG/CR-4312, EGG-2396, Rev. 1.

REHME, K., 1968, Systematische experimenetelle Untersuhung der Abhngigkeit des druckverlustes von der geometrischen Anordnung fur langs durchstromte Stabbundel mit Speraldraht-Abstandshaltern, KfK, Rep. 4/68-16.

REHME, K., 1969, Druckverlust in Stabbundeln mit Spiraldraht-Abstandshaltern, Forsh. Ing.-Wes.35 Nr.4.

REHME, K., 1973a, Simple method of predicting friction factors of turbulent flow in non-circular channels, Int. J. Heat Mass Transfer **16**, 933–950.

REHME, K., 1973b, Pressure drop correlations for fuel element spacers, Nuclear Technol. **17**, 15–23.

REHME, K., 1977, Pressure drop of spacer grids in smooth and roughened rod bundles, Nuclear Technol. **33**, 313–317.

REHME, K., TRIPPE, G., 1980, Pressure drop and velocity distribution in rod bundles with spacer grids, Nuclear Eng. Design **62**, 349–359.

ROUHANI, S.Z., 1966, Void Measurements in the Regions of Sub-cooled and Low Quality Boiling, AE-238.

ROUHANI, S.Z., 1966, Void Measurements in the Regions of Sub-cooled and Low Quality Boiling, AE-239.

ROUHANI, S.Z., 1969, Subcooled Void Fraction, AB Atomenergi, Rep. AWE-RTV-841.

ROUHANI, S.Z., BECKER, K., 1963, Measurement of Void Fractions for Flow of Boiling Heavy Water in a Vertical Round Duct, AE-106.

SAHA, P., ZUBER, N., 1974, "Point of net vapour generation and void fraction in subcooled boiling", Paper B4.7, Proc. 5th Int. Heat Transfer Conf., 175–179.

SAPHIER, D., GRIMM, P., 1992, Bypass Channel Modelling and New Void Correlations for the BWR Option of the SILWER Code, PSI-Bericht Nr.-119.

SEKOGUCHI, K., SAITO, Y., HONDA, T., 1970, JSME Preprint No. 700-7, 83.

SELANDER, W.N., 1978, Explicit Formulas for the Computation of Friction Factors in Turbulent Pipe Flow, AECL-6354.

SIMPSON, H.C., et al., 1977, "Two-phase flow studies in large diameter horizontal lines", European two-phase flow meeting, Grenoble.

SNOEK, C.W., AHMAD, S.Y., 1983, A Method of Predicting Pressure Profiles in Horizontal 37-Element Clusters, AECL-8065, Chalk River, Ontario (1983).

SNOEK, C.W., LEUNG, L.K.H., 1989, A model for predicting diabatic pressure drops in multi-element fuel channels, Nucl. Eng. Design **110**, 299–312.

SOLBRIG, C.W., 1986, "Consistent flow regime map and friction factors for two-phase flow", AIChE Annual meeting, Miami Beach, Florida.

SONNENBURG, H.G., 1989, "Full-range drift-flux model based on the combination of drift flux theory with envelope theory", NURETH-4 Proceedings, Vol. 2 (1003–1009).

SPEDDING, P.L., NGUYEN, V.T., 1980, Regime maps for air-water two-phase flow, Chem. Eng. Sci. **35**, 779.

STEINER, D., 1987, Pressure drop in horizontal flows, Multiphase Science and Technology, Vol. 3 (G.F. Hewitt, J.M. Delhaye, N. Zuber, Eds), Hemisphere Publishing, New York.

STEVANOVIC, V., STUDOVIC, M., 1995, A simple model for vertical annular and horizontal stratified two-phase flows with liquid entrainment and phase transitions: one-dimensional steady state conditions, Nucl. Eng. Design **154**, 357–379.

STEWART C.W., 1985, VIPRE-01: A Thermalhydraulic Code For Reactor Cores, EPRI NP-2511-CCM

SUO, M., BERGLES, A.E., DOYLE, E.F., CLAWSON, L., GOLDBERG, P., 1965, Investigation of Boiling Flow Regimes and Critical Heat Flux", NYO-3304-3.

TAITEL, Y., BORNEA, D., DUKLER, A.E., 1980, Modelling flow pattern transitions for upward gas-liquid flow in vertical tubes, AIChE J. **26**, 345–354.

TAITEL, Y., DUKLER, A.E., 1976, A model for predicting flow regime transitions in horizontal and near horizontal gas-liquid flow, AIChE J. **22**, 47–55.

TAITEL, Y., DUKLER, A.E., 1976a, A theoretical approach to the Lockhart-Martinelli correlation for stratified flow, Int. J. Multiphase Flow **2**, 591–595.

TAKEUCHI, K., YOUNG, M.Y., HOCHREITER, L.E., 1992, Nucl. Sci. Eng. **112**, 170–180.

TARASOVA, N.V., et al., 1966, "Pressure drop of boiling subcooled water and steam water mixture flowing in heated channels", Proc. of 3rd Int. Heat Transfer Conf. 178, ASME.

TAKEUCHI, K., YOUNG, M.Y., HOCHREITER, L.E., 1992, Generalized drift flux correlation for vertical flow, Nucl. Sci. and Eng. **112**, 170–180.

RELAP5 DEVELOPMENT TEAM; 1995, RELAP5/MOD3 Code Manual, Vol. 1 Code Structure, System Models and Solution Methods, NUREG/CR-5535 or INEL-95/0174, Idaho National Engineering Laboratory, Idaho Falls.

THOM, J.R.S., 1964, Prediction of pressure drop during forced circulation boiling of water, Int. J. Heat Mass Transfer **7**, 709–724.

TIPPETS, F.E., 1962, Critical heat flux and flow pattern characteristics of high pressure boiling water in forced convection, GEAP-3766.

TURNER, J.M., WALLIS, G.B., 1965, The Separate Cylinders Model of Two-Phase Flow", NYO-3114-6, Thayer School of Engg. Darmouth College, Hanover, New Hampshire.

TUTU, N.K., 1982, Pressure fluctuations and flow pattern recognition in vertical two-phase gas-liquid flows, Int. J. Multiphase Flow **8**, 443–447.

UNAL, C., BADR, O., TUZLA, K., CHEN, J.C., NETI, S., 1994, Pressure Drop at Rod Bundle Spacers in the Post-CHF Dispersed Flow Regime, Int. J. of Multiphase Flow **20**, 515–522.

VENKAT RAJ, V., 1993, "Experimental thermal hydraulics studies for pressurised heavy water reactors — An Update and Review, 3rd World Conf. on Experimental Heat Transfer, Fluid Mechanics and Thermodynamics, Hawaii.

VIJAYAN, P.K., PRABHAKAR, B., VENKAT RAJ, V., 1981, "Experimental measurement of pressure drop in diabatic two-phase flow and comparison of predictions of existing correlations with the experimental data", H-13, 6th National heat and mass transfer conf. Madras.

VIJAYAN, P.K., SAHA, D., VENKAT RAJ, V., 1984, Measurement of Pressure Drop in PHWR Fuel Channel with 19 Rod Bundles (Wire-Wrap Type) at Low Reynolds Numbers', BARC/I-811.

VIJAYAN P.K., AUSTREGESILO, H., 1993, Predictive Capability of the Models used in the ATHLET Code for the Estimation of Frictional Pressure Loss, Technical Note No. TN-VIJAYAN-93-1, Gesellschaft fur Reaktorsicherheit, Garching.

WALLIS, G.B., 1970, J. Basic Eng. Trans ASME Ser. D. 92, 59.

WALLIS, G.B., 1969, One-Dimensional Two-Phase Flow, McGraw-Hill, New York.

WALLIS, G.B., DOBSON, J.E., 1973, The onset of slugging in horizontal stratified air-water flow, Int. J. Multiphase Flow **1**, 173–193.

WEISMAN, J., KANG, S.Y., 1981, Flow pattern transitions in vertical upwardly inclined lines, Int. J. Multiphase Flow **7**, 271.

WEISMAN, J., CHOE, W.G., 1976, "Methods for calculation of pressure drop in cocurrent gas-liquid flow", Proc. Two-phase Flow and Heat Transfer Symp. Workshop, Fort Lauderdale, Two-Phase Transport and Reactor Safety. Vol. 1.

WILSON, S.F., LITTLETON, W.E., YANT, H.W., MAYER, W.C., 1965, Preliminary Separation of Steam from Water by Natural Separation, Part 1, Allis-Chalmers Atomic Energy Report No. ACNP-65002.

YEH, H.C., HOCHREITER, L., 1980, Mass effluence during FLECHT forced reflood experiments, Nucl. Eng. Design **60**, 413–429.

ZHAO, L., REZKALLAH, K.S., 1995, Pressure drop in gas-liquid flow at microgravity conditions, Int. J. Multiphase Flow **21** (1995) 837–849.

ZIGRANG, D.J., SYLVESTER N.D., 1985, A review of explicit friction factor equations, Trans. ASME, J. Energy Res. Technol. **107**.

ZUBER, N., FINDLAY, J.A., 1965, Average volumetric concentration in two-phase flow systems, J. Heat Transfer, Trans. ASME **87**, 453–468.