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Modified distribution parameter for churn-turbulent flows in large diameter channels

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HIGHLIGHTS

- Void fraction data collected in pipe sizes up to 0.304 m using impedance void meters.
- Flow conditions extend to transition between churn-turbulent and annular flow.
- Flow regime identification results agree with previous studies.
- A new model for the distribution parameter in churn-turbulent flow is proposed.

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ABSTRACT

Two phase flows in large diameter channels are important in a wide range of industrial applications, but especially in analysis of nuclear reactor safety for the prediction of BWR behavior and safety analysis in PWRs. To remedy an inability of current drift-flux models to accurately predict the void fraction in churn-turbulent flows in large diameter pipes, extensive experiments have been performed in pipes with diameters of 0.152 m, 0.203 m and 0.304 m to collect area-averaged void fraction data using electrical impedance void meters. The standard deviation and skewness of the impedance meter signal have been used to characterize the flow regime and confirm previous flow regime transition results. By treating churn-turbulent flow as a transition between cap-bubbly dispersed flow and annular separated flow and using a linear ramp, the distribution parameter has been modified for churn-turbulent flow. The modified distribution parameter has been evaluated through comparison of the void fraction predicted by the drift-flux model and the measured void fraction.

1. Introduction

Two-phase flows are important in a wide range of industrial applications, but especially in analysis of nuclear reactor safety. In Boiling Water Reactors (BWRs), two-phase flows exist during both normal operation and during transients or accident scenarios. Especially for natural circulation BWR designs, the ability to predict the void fraction in the region above the reactor core is essential for the prediction of the natural circulation flow rate and liquid inventory in the Reactor Pressure Vessel (RPV). During accident scenarios, the region above the reactor core in Pressurized Water Reactors (PWRs) may also be occupied by two-phase flow.

Typically two-phase flows in reactor systems are predicted using advanced predictive codes such as RELAP or TRACE based on the one-dimensional form of the two-fluid model. In addition, three-dimensional CFD codes such as CFX and FLUENT use the three-dimensional form of the two-fluid model as a basis for the prediction of two-phase flow behavior. The two-fluid model, when implemented correctly, is the most detailed model available for predicting large-scale flow behavior, but is also the most computationally intensive and, as reported in the literature (Ishii and Hibiki, 2010; Delhaye, 2001), a number of closure relations still require additional development. Further, as discussed by Wulff (2011), great care must be taken in the implementation of the two-fluid model in computational methodologies. It treats each phase independently, with interfacial transfer terms to describe the transfer of mass, momentum and energy between the two phases. In one-dimensional approaches such as that used in advanced reactor system analysis codes, interfacial momentum transfer is expressed by the generalized interfacial drag and is one of the most important factors in determining the void fraction. Typically calculation of the generalized interfacial drag requires the use of constitutive models based on the drift-flux model (Ishii and Mishima, 1984; Ishii and Hibiki, 2010) to compute the area-averaged relative velocity from

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Nomenclature

Latin characters

C_0	distribution parameter
D	diameter [m]
D_H	hydraulic diameter [m]
G	measured impedance [Ω]
g	gravitational acceleration [m/s^2]
j	volumetric flux [m/s]
N	total number
N_{mf}	viscosity number
s	skewness
$\bar{V}_{gj}$	mean drift velocity [m/s]
V_{gj}	void-weighted drift velocity [m/s]
v	velocity [m/s]
v_{gj}	void-weighted drift velocity [m/s]
x	data value

Greek characters

α	void fraction
σ	surface tension [N m]; standard deviation
$\Delta\rho$	density difference [kg/m^3]
ρ	density [kg/m^3]

Subscripts/superscripts

*	non-dimensional value
+	non-dimensional value
0	value at void fraction of 0
1	value at void fraction of 1
AN	value at transition to annular flow
CT	value at transition to churn-turbulent flow
exp	experimentally measured value
f	value for liquid
g	value for gas
Ishii	value from Ishii (1977)
n	*n*th value
pred	value predicted by model

Operators

$\langle\cdot\rangle$	area-averaged value
$\langle\langle\cdot\rangle\rangle$	void-weighted area-averaged value

the void-weighted average gas and liquid velocities calculated by the momentum equation.

The region above the reactor core in both reactor types is, in its most simple form, a large diameter channel. Large diameter channels are flow channels in which stable slug bubbles cannot be sustained. Stable slug bubbles are defined as Taylor cap bubbles which occupy the entire cross-section of the flow channel. When Taylor cap bubbles reach a certain size instability in the upper surface causes the bubble to collapse and break up into smaller bubbles. This size has been defined by Kataoka and Ishii (1987) as

$$D^* = \frac{D}{\sqrt{\sigma/g\Delta\rho}} = 40 \tag{1}$$

Therefore in any channel larger than the size predicted by Eq. (1), large Taylor cap bubbles cannot occupy the entire cross-section of the channel without becoming unstable and breaking up. This is the defining feature of large diameter channels. For air–water flows at atmospheric conditions this is 10.1 cm, while for BWR operating conditions, this value decreases to 6.3 cm. Many of the pipes in nuclear reactor systems are therefore considered 'large diameter'.

Because of the instability of larger cap bubbles, flows in large diameter channels behave very differently than flows in channels with smaller diameters. Without the stabilizing effect of the channel wall on the interface of the Taylor bubbles, the flow regime considered to be 'slug flow' does not exist. Instead, this region is occupied by many smaller Taylor cap bubbles. This causes significant differences in the local behavior of the two-phase flow in large and small channels under these flow conditions, affecting the models applicable to void fraction prediction as well as flow regime transition behavior. This means that the models applicable to void fraction prediction in small diameter channels, which are well-developed and benchmarked with a large experimental database, may not apply to large diameter channels.

The drift-flux model was developed by Zuber and Findlay (1965) and represents a more simplified tool for predicting the area-averaged void fraction in various two-phase flows. Many constitutive models exist for drift-flux models in various flow channel geometries and flow regimes. For small diameter pipes, the mechanistically developed model by Ishii (1977) is often used as a basis due to its simplicity and accuracy across a wide range of conditions. In addition, many drift-flux type models have been developed for application to large diameter channels. Many of these correlations, such as those of Hills (1976), Shipley (1982), Clark and Flemmer (1985, 1986), and Hirao (1986) are entirely empirical in nature and thus have not seen widespread use due to the inability of most empirical correlations to predict conditions outside those used in the initial benchmarking data set. Ishii and Kocamustafaogullari (1985) developed a mechanistic prediction of the drift velocity for 'slug' flow in large diameter channels by considering the rise velocity of a maximum-sized cap bubble, and this work was expanded on by Kataoka and Ishii (1987) who developed a semi-empirical correlation for the drift velocity for two-phase flows in pipes with various diameters, density ratios, continuous phase viscosities, etc. This model was thoroughly benchmarked against data in a wide range of pressure conditions, fluid combinations, and pipe sizes and was shown to scale well to nuclear power plant conditions. Hibiki and Ishii (2003) also developed a semi-empirical correlation, in this instance for bubbly flow conditions, based on numerous experiments (Hibiki and Ishii, 2001a,b). For large diameter pipes, the current state-of-the-art models are highlighted in Table 1 and include that of Hibiki and Ishii (2003) in the bubbly flow regime and Kataoka and Ishii (1987) in the cap-bubbly and churn-turbulent flow regimes. For annular flow Ishii's (1977) model is widely used when drift-flux type models are called for.

Table 2 shows many of the available research efforts of the past decades in the measurement of void fractions in large diameter channels. The highest gas flow rate achieved in any of these experiments was 8 m/s and the maximum void fraction is 0.85, but only a few studies included conditions with void fractions higher than 0.5–0.6. Additional data is needed for void fractions from 0.7 to 0.95 to confirm the data that is available and extend the existing database throughout churn-turbulent flow and, if possible, into annular flow. This will allow the evaluation of drift-flux correlations throughout this void fraction range.

Fig. 1 shows the void fraction measurements from several of these previous studies at void fractions up to 0.85 compared to the prediction of the existing models discussed above and illustrates the inability of the current drift-flux type correlations to predict high void fraction flows in large diameter channels. In this case the figure is for pool conditions, or liquid velocity of 0 m/s. This is chosen because the majority of high-pressure steam–water data was collected under pool conditions or for very low liquid velocities, as shown in Table 1. The data from Wilson (1961), Carrier (1963) and Styrikovich and Kutateladze (1976) was collected in high-pressure steam-water flows, while the data of Bailey et al. (1956), Hills (1976) and Schlegel et al. (2010) was collected for atmospheric pressure air–water flows. The figure indicates that