Progress in Nuclear Energy 83 (2015) 229–247

Contents lists available at ScienceDirect

Progress in Nuclear Energy

journal homepage: www.elsevier.com/locate/pnucene

Drift-flux model for rod bundle geometry

Tetsuhiro Ozaki [a,*], Takashi Hibiki [b]

[a] Tokai Works, Nuclear Fuel Industries Ltd., 3135-41 Muramatsu, Tokai-mura, Naka-gun, Ibaraki 319-1196, Japan
[b] School of Nuclear Engineering, Purdue University, 400 Central Drive, West Lafayette, IN 47907-2017, USA

ARTICLE INFO

Article history:
Received 1 July 2014
Received in revised form 29 September 2014
Accepted 19 March 2015
Available online 21 April 2015

Keywords:
Drift-flux model
Rod bundle
Sub channel
Void fraction
Two-phase flow

ABSTRACT

In view of an important role of a one-dimensional drift-flux correlation in nuclear thermal-hydraulic system analysis codes, several drift-flux correlations such as Lellouche–Zolotar, Chexal–Lellouche, TRAC-BF1 and Ozaki correlations have been reevaluated by rod bundle test data taken in FRIGG and NUPEC test facilities. The mean absolute error of void fraction representing a correlation bias of the Lellouche–Zolotar, Chexal–Lellouche, TRAC-BF1 and Ozaki correlations are, respectively, −1.0, 0.5, −6.3 and −3.3% for the FRIGG test data and 2.0, 2.3, −0.4 and −0.7% for the NUPEC test data. The effects of unheated rods, axial and radial power distributions, large unheated center rod and geometry of a shroud or casing on void fraction are identified. The presence of unheated rods with similar size of other heated rods tends to increase a distribution parameter in a drift-flux correlation, whereas the presence of a large unheated center rod tends to decrease the distribution parameter. The axial and radial power distributions do not have significant effect on void fraction within the tested axial and radial power distribution range. The Ozaki correlation is recommended for predicting void fraction in a BWR core but it is suggested to reduce the distribution parameter in the Ozaki correlation if a large unheated center rod exists in the core. It is indicated that drift-flux correlations developed based on bounded rod bundle test facility data may overestimate the distribution parameter for a PWR core.

1. Introduction

Detailed understanding of thermal-hydraulic behaviors in nuclear power plants (NPPs) is of importance to secure safety operation of the NPPs. Several two-fluid model (Ishii and Hibiki, 2011) based nuclear thermal-hydraulics system analysis codes such as TRACE (U.S. NRC, 2008), RELAP5 (ISL, 2001) and TRAC-BF1 (Borkowski and Wade, 1992) have been utilized to simulate thermal-hydraulic responses in the NPPs. To elucidate the simulation accuracy and its uncertainty, best estimate plus uncertainty (BEPU) methodology and code scaling, applicability, and uncertainty (CSAU) methodology have been proposed in safety analysis procedures (AESJ, 2008; Boyack et al., 1989). In the BEPU and CSAU methodologies, component model uncertainty and model scalability should be discussed in detail. One of the important component models in codes is an interfacial shear term which has a large impact on void fraction prediction. The interfacial shear term can be formulated by rigorous drag laws but this approach requires additional interfacial area concentration constitutive equations which have not been developed well (Hibiki and Ishii, 2002a; Hibiki et al., 2006a; Ozar et al., 2012). Due to lack of accurate interfacial area concentration models, thermal-hydraulics codes have often utilized drift-flux correlations to estimate the interfacial shear term (Brooks et al., 2012). The use of a proper rod bundle drift-flux correlation to predict the interfacial shear term is thus indispensable to ensure successful nuclear safety analyses (Griffiths et al., 2014).

A one-dimensional drift-flux correlation includes two important parameters, namely distribution parameter and drift velocity. The distribution parameter represents the effect of void fraction and mixture volumetric flux distributions on void fraction, whereas the drift velocity represents the effect of a relative velocity between phases on void fraction. The distribution parameter is affected by various factors such as flow pattern (Ishii, 1977), flow channel geometry (Julia et al., 2009; Chen et al., 2012), flow channel size (Kataoka and Ishii, 1987; Hibiki and Ishii, 2003), flow orientation (Goda et al., 2003), pressure (Ishii, 1977), liquid velocity (Clark et al., 2014), bubble size (Hibiki and Ishii, 2002b), gravitational acceleration (Hibiki et al., 2006b) and phase change (Hibiki et al., 2003). Although some researchers have attempted to model the distribution parameter analytically using assumed void fraction and mixture volumetric

* Corresponding author.
E-mail address: te-ozaki@nfi.co.jp (T. Ozaki).

http://dx.doi.org/10.1016/j.pnucene.2015.03.015
0149-1970/

flux distributions (Zuber and Findlay, 1965; Hibiki et al., 2003; Julia et al., 2009), empirical distribution parameter correlations have been commonly adopted (Coddington and Macian, 2002). The drift velocity is affected by various factors such as flow pattern (Ishii, 1977), flow channel confinement (Julia et al., 2009), and flow channel size (Kataoka and Ishii, 1987; Hibiki and Ishii, 2003). Since the drift velocity arises due to a density difference between phases, the frame work of the drift velocity can be made analytically based on rigorous drag laws (Ishii, 1977).

Most correlations of the distribution parameter and drift velocity utilized in the thermal-hydraulics analysis codes were developed in the 1980s when the concepts of model uncertainty analysis and model scale-up capability were not well-developed. The performance of drift-flux correlations heavily depends on data accuracy and a range of flow and pressure conditions utilized for the correlation development as well as modeling strategy considering various factors affecting the correlation. Coddington and Macian (2002) evaluated the performance of 9 bundle drift-flux correlations using 9 bundle test data bases. The bias and random uncertainty (defined by standard deviation here) range from −4.1% to +5.7% and from 7.1 % to 12.6 %, respectively. Those comparisons are certainly useful for understanding correlation defects and their applicable ranges but the averaged uncertainty quantities such as bias and random uncertainty do not give detailed insights to identify dominant factors affecting the performance of drift-flux correlations.

This paper reevaluates the accuracy of several drift-flux correlations using two rod bundle test data taken in a wide range of mass flux and pressure conditions, and addresses the effects of unheated rods, axial and radial power distributions, and a large unheated center rod on void fraction.

2. Existing drift-flux correlations for rod bundle

2.1. Drift-flux model

A one-dimensional drift-flux model is formulated by Zuber and Findlay (1965)

$$\langle\langle v_g\rangle\rangle = \frac{\langle j_g\rangle}{\langle\alpha\rangle} = C_0\langle j\rangle + \langle\langle V_{gj}\rangle\rangle, \tag{1}$$

where v_g, j_g, α, C_0, j, and V_{gj} are the gas velocity, superficial gas velocity, void fraction, distribution parameter, mixture volumetric flux, and drift velocity, respectively. $\langle\rangle$ and $\langle\langle\rangle\rangle$ indicate the area-averaged and void fraction-weighted mean values, respectively. The void fraction-weighted mean drift velocity and distribution parameter are defined by

$$\langle\langle V_{gj}\rangle\rangle \equiv \frac{\langle\alpha V_{gj}\rangle}{\langle\alpha\rangle}, \tag{2}$$

and

$$C_0 \equiv \frac{\langle\alpha j\rangle}{\langle\alpha\rangle\langle j\rangle}. \tag{3}$$

Considering a limiting condition such that the distribution parameter should become unity as a density ratio approaches unity, a simple distribution parameter model is proposed as (Ishii, 1977)

$$C_0 = C_\infty - (C_\infty - 1)\sqrt{\frac{\rho_g}{\rho_f}}, \tag{4}$$

where ρ_f and ρ_g are, respectively, the liquid and gas phase densities and C_∞ is the asymptotic value of the distribution parameter. A correction factor as a function of void fraction is introduced in Eq. (4) for subcooled boiling flow (Ishii, 1977; Hibiki et al., 2003).

2.2. Existing drift-flux correlations for rod bundle geometry

Rod bundle drift-flux correlations utilized in NPP system analysis and steam generator analysis codes and recently developed correlations are summarized in Table 1. They are Lellouche–Zolotar correlation (1982), Chexal–Lellouche correlation (1991), TRAC-BF1 correlation (1992) and Ozaki correlation (2013).

2.2.1. Lellouche and Zolotar correlation (1982)

Lellouche–Zolotar correlation was developed based on 738 FRIGG data taken at Allmänna Svenska Elektriska Aktiebolaget (ASEA) and 46 CISE data taken at Centro Informazioni Studi Esperienze (CISE). 94% of total data utilized for developing the Lellouche–Zolotar correlation was taken in the FRIGG test program. The FRIGG tests were intended to simulate two types of fuel assembly; Marviken heavy-water type (6 or 36 rods mounted in a circular shroud, FRIGG-1, 2, 3 and 4 data) and the Oskarshamn light-water type (8 × 8 rod bundles in a square casing, FRIGG-5 data) reactors (Nylund, 1969). It can be said that the Lellouche–Zolotar correlation was practically developed by the single data source taken at ASEA. Although the Lellouche–Zolotar correlation considers the correlation performance at limiting cases such that $C_0 \to 1$ and $\langle\langle V_{gj}\rangle\rangle \to 0$ as $\langle\alpha\rangle \to 1$, the parameter dependence of the distribution parameter and drift velocity is purely empirical for co-current two-phase flows at a pressure higher than 1.38 MPa (200 psi) and the applicability of the Lellouche–Zolotar correlation beyond the data range has not been validated. A precursor of the Lellouche–Zolotar correlation is Lellouche correlation (1974) and an improved version of the Lellouche–Zolotar correlation is Chexal–Lellouche correlation (1991). Such the correlation development history clearly indicates the correlation performance heavily depending on databases utilized for developing the correlation. The Lellouche–Zolotar correlation is utilized in ATHOS3 code to simulate two-phase flow behavior in steam generators (AESJ THD, 1993).

2.2.2. Chexal and Lellouche correlation (1991)

Chexel–Lellouche correlation (1991) is an extended and improved version of the Lellouche–Zolotar correlation (1982). The basic model development concept of the Chexel–Lellouche correlation is similar to that of the Lellouche–Zolotar correlation. Several correlation packages in the Chexal–Lellouche correlation are available depending on thermal conditions (adiabatic and diabatic two-phase flows), channel orientations (vertical, inclined and horizontal two-phase flows), and flow directions (co-current upward and downward and counter-current two-phase flows). The Chexal–Lellouche correlation known as EPRI correlation has been utilized in RELAP5 code to simulate a two-phase flow behavior in NPPs (ISL, 2001).

2.2.3. Correlation utilized in TRAC-BF1 code (1992)

A drift-flux correlation utilized in TRAC-BF1 code (Borkowski and Wade, 1992) is the Bestion correlation for drift velocity (Bestion, 1990) and Rouhani correlation for distribution parameter (Rouhani, 1969). The Bestion correlation was developed based on a theoretical terminal velocity of a slug-shaped particle and experimental data taken in rod bundles with hydraulic diameters of 12 mm and 24 mm. The Bestion correlation is not applicable for low pressure conditions and large diameter channels (Clark et al., 2014). The Rouhani correlation was developed by considering the mass