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Experimental investigation of horizontal forced-vibration effect on air-water two-phase flow

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ABSTRACT

In order to investigate the potential seismic vibrations effect on two-phase flow in an annular channel, experimental tests with air-water two-phase flow under horizontal vibrations were carried out. A low-speed eccentric-cam vibration module capable of operating at motor speed of 45–1200 rpm ($f = 0.75$–20 Hz) was attached to an annular channel, which was scaled down from a prototypic BWR fuel sub-channel with inner and outer diameters of 19.1 mm and 38.1 mm, respectively. The two-phase flow was operated in the ranges of $\langle j_f \rangle = 0.25$–1.00 m/s and $\langle j_g \rangle = 0.03$–1.46 m/s with 27 flow conditions, and the vibration amplitudes controlled by cam eccentricity (E) were designed for the range of 0.8–22.2 mm. Ring-type impedance void meters were utilized to detect the area-averaged time-averaged void fraction under stationary and vibration conditions. A systematic experimental database was built and analyzed with effective maps in terms of flow conditions ($\langle j_g \rangle$-$\langle j_f \rangle$) and vibration conditions (E-f and f-a), and the potential effects were expressed by regions on the maps. In the $\langle j_g \rangle$-$\langle j_f \rangle$ maps, the void fraction was found to potentially decrease under vibrations in bubbly flow regime and relatively lower liquid flow conditions, which may be explained by the increase of distribution parameter. Whereas and the void fraction may increase at the region closed to bubbly-to-slug transition boundary under vibrations, which may be explained by the changes of drift velocity due to flow regime change from bubbly to slug flows. No significant change in void fraction was found in slug flow regime under the present test conditions.

1. Introduction

In the past decades, earthquakes have caused several unusual events for nuclear reactors. In 1993, a minor earthquake (M4.0) in Tohoku, Japan, caused a scram event at the Onagawa nuclear power plant (boiling water reactor, BWR) (Japan Business, 1993; Nariai, 1994). In March of 2011, a strong earthquake (M9.0) accompanying huge tsunami attached Tohoku, Japan, and resulted in an INES-7 nuclear accident at Fukushima Daiichi nuclear power plant (BWR). Further in August of 2011, an earthquake (M5.8) in Virginia, USA, induced in a scram event at North Anna nuclear power station (pressurized water reactors, PWR) (NRC, 2011). These earthquake events have brought the public's attention to the potential seismic effects on nuclear reactor integrity and thermal hydraulics. According to the official reports, the scram events in Onagawa and North Anna during earthquakes were due to the changes of water gaps among fuel structures caused by the seismic vibrations. However, several possible reasons may cause scram events from the view point of nuclear thermal hydraulics and void-reactivity feedbacks. For a BWR, the local void fraction inside the reactor core may change under seismic vibrations; for a PWR, the thermal boundary layer (TBL) may be disturbed under seismic vibrations. If the local void fraction or TBL were affected by vibrations, the reactor core power may change due to the changes of moderator density and thermal neutron populations (Hixson, 2011). Since in general the earthquakes may vibrate at relatively low frequency, usually $f \leq 20$ Hz, the effects of low frequency vibrations on reactor core thermal hydraulics, specifically for the two-phase flow inside the coolant sub-channel geometry should be reexamined in detail.

Several research groups have attempted to investigate the vibration effects on adiabatic and boiling two-phase flows by performing experiments. Nangia and Chon (1967) carried out an experimental investigation of interfacial vibration effect on saturated pool boiling. The vibrations were applied in the vertical direction

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with frequencies of 20–115 Hz. They found that the heat transfer coefficient was increased and bubble diameter was decreased in their experiment. Shioyama and Ohtomi (1990) performed subcooled boiling tests with Freon-113 to investigate pressure fluctuations and bubble behaviors in a vertical tube with longitudinal vibrations at the frequency range of 5–50 Hz, and the bubble size, thermal boundary layer and void fraction were found affected under vibrations. Skoczylas and Ubranski (1992) examined the heat transfer in a thin-layer evaporator, and the vertical vibration with frequencies of 0–12 Hz was introduced. They found that heat transfer coefficient can be increased by 23.7–104.8%. Nariai and Tanaka (1994) carried out an experiment with vibration heater rod to study void fraction changes in a subcooled boiling flow. They tested with different amplitudes and frequencies and found that the void fraction may decrease drastically as the vibration frequency reached 10 Hz or higher. Kawamura et al., (1996a, 1996b) performed an experiment to investigate the changes of thermal neutron flux in the fuel assembly under vibrations. The neutron flux was found to increase up to 20% under vibrations (Kawamura et al., 1996a). Further, they tested the horizontal vibrations for subcooled boiling flow. The results showed only limited vibration effect on bubble collapse/growth was observed (Kawamura et al., 1996b). Hibiki and Ishii (1998) studied the local parameters of upward air-water two-phase flow in a circular pipe with flow-induced vibration. It was found that the bubble coalescence may be enhanced, and the distributions of local interfacial area concentration, bubble diameter and void fraction may be changed from wall-peaked to center-peaked profiles at relatively low liquid flow conditions. As a result, the distribution parameter and the bubble diameter may increase under vibrations. Umekawa et al. (1999) performed a flow oscillation experiment using an oscillator connected to the boiling test loop, and the critical heat flux (CHF) was found to decrease at the lower flow periods, which may limit the system heat removal performance. Osakabe et al. (2000) examined the self-excited oscillation effect on a thermosiphon system. The pressure fluctuation frequency can be up to about 10 Hz and the cooling rate can be increased by 10%. Abou-Ziyan et al., (2001) tested the performance of stationary and vibrated thermosiphon with water and R-134. The vertical vibrations were introduced with frequencies up to 4.33 Hz, and it was found that the boiling limits can be increased by 250%. Chou et al., (2002, 2003) investigated the boiling heat transfer in a heating container with different surface structures, different-sized balls and different vibration conditions. The tested frequencies were ranged from 0.5 to 60 Hz, and the boiling heat transfer was enhanced by 20–65%. Lee et al. (2004) investigated the enhancement of CHF in a vertical round tube by horizontal vibrations. The vibrations were operated with frequencies of 0–70 Hz, and the CHF can be enhanced up to 12.6%. Chen et al. (2010) carried out a preliminary subcooled flow boiling experiment with an annular channel and found that the area-averaged void fraction may be changed under low-frequency vibrations (0.5—6.5 Hz). Furthermore, Chen et al., (2012b and 2014) performed air-water two-phase flow experiment with similar vibration conditions (0.5—6.5 Hz) and analyzed the transient void fraction signals with fast Fourier transform (FFT) and Kohonen neural network, and suspected flow regime change around the transition boundary was pointed out by the neural network. Table 1 briefly summarizes the existing experimental tests and observed effects in the literature. Based on the present literature review, only limited systematic results have been proposed for the horizontal vibration effect on two-phase flows, and the vibration-fluid related phenomena should be further investigated. In order to understand the potential effects of horizontal seismic vibrations (commonly $f \leq 20$ Hz) on reactor core thermal hydraulics, specifically for variations of void fraction and flow regimes, common earthquake ranges (such as intensity and frequency) along with a wide range of two-phase flow conditions should be tested. With the systematic database, the effective flow/vibration conditions may be pointed out and analyzed in terms of flow regime maps or vibration conditions.

Nomenclatures

A	Amplitude [mm]
a	Acceleration [m/s^2]
C_0	Distribution parameter [-]
D	Diameter [m]
D_b	Bubble diameter [m]
D_H	Hydraulic diameter [m]
E	Eccentricity and vibration amplitude [m]
f	Frequency [Hz]
G^*	Non-dimensional voltage [-]
g	Gravitational acceleration [m/s^2], 1 g = 9.8 m/s^2
h	Heat transfer coefficient [W/m^{2}K]
j	Superficial velocity [m/s]
j_f	Superficial liquid velocity [m/s]
j_g	Superficial gas velocity [m/s]
ms	Motor speed [RPM]
V	Voltage [V]
v_g	Velocity [m/s]
v_{gj}	Drift velocity [m/s]
z	Axial length [m]
$< >$	Area averaged properties
$\ll \gg$	Void-weighted area-averaged properties

Greek Symbols

α	Void fraction [-]
α^*	Non-dimensional void change [-]
ρ	Density [kg/m^3]
σ	Surface tension [N/m^1]

Subscripts

f	Liquid phase
g	Gas phase
gj	Drift velocity term
H	Hydraulic diameter
in	Inlet properties
m	Measurement value
NV	No vibration condition
V	Vibration condition

Abbreviations

1-D	One dimensional
AL	Aluminum
BWR	Boiling water reactors
Cal.	Calculated
Non-D	Non-dimensional or dimensionless
NV	No vibration
PWR	Pressurized water reactors
RMS	Root mean square
SS	Stainless steel
Sup.	Superficial
TBL	Thermal boundary layer
V	Vibration
Vib.	Vibration

2. Experimental facility

The experimental facility was designed for a two-phase flow test system excited with simulated horizontal vibrations. The schematic of experimental system is shown in Fig. 1, and the whole system contains a two-phase flow loop and a vibration