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The complex nonlinear dynamics in the multiple boiling channels coupling with multi-point reactors with constant total flow rate

Jin Der Lee [a,*], Chin Pan [b]

[a] Nuclear Science and Technology Development Center, National Tsing Hua University, 101, Sect. 2, Kuang Fu Road, Hsinchu 30013, Taiwan, ROC

[b] Department of Engineering and System Science, National Tsing Hua University, 101, Sect. 2, Kuang Fu Road, Hsinchu 30013, Taiwan, ROC

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ABSTRACT

The present study explores the effect of nuclear-coupled feedback on the oscillation modes and nonlinear phenomena of a five nuclear-coupled boiling channel system by a nonlinear dynamic model previously developed by the authors. The results show that the combined effects of stable neutron interaction and unstable void-reactivity feedback generate distinct influence on the system stability, particularly a significant unstable effect as in the $4C_\alpha$ cases. The effect of channel-to-channel interaction will drive the 5-channel system more unstable than a 3-channel one. Such a nuclear-coupled effect may affect the oscillation modes and nonlinear phenomena among the channels substantially. For the present system with a constant total flow rate, the superimposition of the dominant single-phase frictional pressure drop and strengthening void-reactivity feedback may result in the departure from the out-of-phase mode oscillations at some system states. The results demonstrate the appearance of different bifurcation phenomena in the unstable region and complex nonlinear phenomena, i.e. various periodic oscillations and complex Rossler type of chaotic oscillations, in such a system subject to certain nuclear-coupled feedbacks. A special type of complex P-3 oscillations is identified in this system. It suggests that there may be immeasurable types of the periodic nonlinear oscillations in the limited unstable space of this five nuclear-coupled boiling channel system.

1. Introduction

Density wave oscillations (DWOs) are a typical type of dynamic instability occurring in the boiling system (Boure et al., 1973). The self-sustained DWOs are well-known to be triggered by the multiple thermohydraulic feedbacks among flow rate, pressure drop, flow enthalpy and density or void fraction. Most two-phase flow systems consist of multiple parallel boiling channels and studies concerning DWOs combined with channel-to-channel instability in a boiling system are of significant interest. Guido et al. (1991) indicated that in-phase and out-of-phase oscillations were the fundamental oscillation modes for an identical double channel system. Podowski et al. (1990) revealed complex interaction modes involving coupled parallel channels caused by the differences among the channels. Furthermore, Lee and Pan (1999) reported that complex channel-to-channel interactions might drive the system more unstable with increasing number of channels. The oscillations among channels were essentially out-of-phase in the multi-channel system having a constant total flow rate.

The issue of nuclear-coupled thermal–hydraulic stability is extremely important for the design, operation and safety of boiling water reactors (BWRs) and advanced boiling water reactors (ABWRs). Because of the strong coupling between the thermal–hydraulics and the neutronics, two types of instability, i.e. core-wide (in-phase) and regional (out-of-phase) oscillations, might exist in the nuclear boiling system (March-Leuba and Blakeman, 1991). Regional instability events were reported in Caorso, Ringhals 1 and Confrentes in the unstable region of low flow rates and middle high power conditions (OECD, 1997). During such an instability event, the flow rate and the heat generation rate in one half of the core might oscillate out-of-phase with the other half with a large magnitude while the average power oscillation was still quite small (March-Leuba and Blakeman, 1991). Out-of-phase instabilities were favored when the gain of the thermal–hydraulic feedback was stronger than that of the neutronic feedback. Thus, the high two-phase pressure drop would cause the dominance of the out-of-phase mode in the core unstable phenomenon (March-Leuba and Blakeman, 1991).

Out-of-phase instabilities are commonly dealt with the boundary conditions of the same channel pressure drop among parallel channels with a constant total inlet flow rate. Some reduced order

* Corresponding author. Tel.: +886 3 5742889; fax: +886 3 5713849.
E-mail address: ctlee2@mx.nthu.edu.tw (J.D. Lee).

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Nomenclature

A_{x-s}	cross sectional area of the channel (m^2)
$A^+_{x-s,j}$	non-dimensional cross sectional area of the j-th heated channel, $=A_{x-s,j}/A_{x-s,1}$
C_j	dynamic precursor concentration in j-th subcore ($\#m^{-3}$)
C_{j0}	steady state precursor concentration in j-th subcore ($\#m^{-3}$)
C^+_j	non-dimensional precursor concentration in j-th subcore, $=(C_j - C_{j0})/C_{j0}$
C_D	Doppler-reactivity coefficient ($\$/\Delta T_F$)
C_{pf}	liquid constant pressure specific heat ($J\,kg^{-1}\,K^{-1}$)
C_α	void-reactivity coefficient ($/%)
D_H	diameter of the channel (m)
$f_{1\phi}$	single-phase friction factor
$f_{2\phi}$	two-phase friction factor
Fr	Froude number, $= u_s^2/gL_H$
H_{jm}	interaction coefficient between subcores
h	heat transfer coefficient ($W\,m^{-2}\,K^{-1}$)
h_C	clad-to-coolant heat transfer coefficient ($W\,m^{-2}\,K^{-1}$)
h_{gap}	Pellet-to-clad gap conductance ($W\,m^{-2}\,K^{-1}$)
i_f	saturated liquid enthalpy ($J\,kg^{-1}$)
i_{fg}	latent heat of evaporation ($J\,kg^{-1}$)
i_g	saturated vapor enthalpy ($J\,kg^{-1}$)
i_i	inlet liquid enthalpy ($J\,kg^{-1}$)
k	thermal conductivity ($W\,m^{-1}\,K^{-1}$) or loss coefficient
L	length (m)
L_H	channel length (m)
L^+	non-dimensional length, $= L/L_H$
M	mass (kg)
M^+	non-dimensional mass, $=M/\rho_f L_H A_{x-s}$
N_j	dynamic neutron density in j-th subcore ($\#m^{-3}$)
N_{j0}	steady state neutron density in j-th subcore ($\#m^{-3}$)
N_s	number of nodes in the single-phase region
$N_{pch,j}$	phase change number for j-th channel, $= \frac{Q_j}{\rho_f A_{x-s} u_s}\frac{v_{fg}}{i_{fg} v_f}$
N_{sub}	subcooling number, $= \frac{i_f - i_i}{i_{fg}}\frac{v_{fg}}{v_f}$
N^+_j	non-dimensional neutron density in j-th subcore, $=(N_j - N_{j0})/N_{j0}$
P	system pressure (bar)
Q_j	heating power in j-th channel (W)
Q_0	steady-state heating power (W)
q''	heat flux ($W\,m^{-2}$)
q_0	steady state heat flux ($W\,m^{-2}$)
q''^+	non-dimensional heat flux, $= q''/q''_0$
q'''	volumetric heat generation rate ($W\,m^{-3}$)
r	radius (m)
T	temperature (K)
T_0	steady-state heated wall temperature
T_{sat}	saturation temperature (K)
T^+	non-dimensional temperature, $= (T - T_0)/T_{sat}$
t	time (s)
t^+	non-dimensional time, $= tu_s/L_H$
u	velocity (ms^{-1})
u_{i0}	steady state inlet velocity (ms^{-1})
u_s	velocity scale, $= u_{i0}$
u^+	non-dimensional velocity, $= u/u_s$
v_f	specific volume of saturated liquid ($m^3\,kg^{-1}$)
v_{fg}	difference in specific volume of saturated liquid and vapor ($m^3\,kg^{-1}$)
W	mass flow rate ($kg\,s^{-1}$)
x_e	exit quality
z	axial coordinate (m)
z^+	non-dimensional axial coordinate, $=z/L_H$

Greek symbols

α	void fraction or thermal diffusivity
β	delayed neutron fraction
ΔP	pressure drop (Pa)
ΔP^+	non-dimensional pressure drop, $= \Delta P/\rho_f u_s^2$
δx	$(x - x_0)$ for variable x, x_0 represents the steady-state value
ρ	density ($kg\,m^{-3}$)or reactivity ($\Delta K/K$, where K is multiplication factor)
ρ^+	non-dimensional density, $= \rho/\rho_f$
ρ_f	density of saturated liquid ($kg\,m^{-3}$)
Λ	neutron generation time (s)
λ	boiling boundary (m)
λ^+	non-dimensional boiling boundary, $= \lambda/L_H$
λ_C	decay constant of delayed neutron precursor (s^{-1})

Subscripts

ch	channel
e	exit of the channel
i	inlet of the channel
j	j-th channel or subcore
n	n-th node in the single-phase region
tot	total
0	steady state
F	fuel pellet
C	cladding

models were employed to analyze the out-of-phase mode characteristics of BWRs regarding the accuracy of the model as well as the simplicity of computation. The two-channel methodologies coupled with multimodal neutron kinetics, each representing one half of reactor core, were developed by Muñoz-Cobo et al. (2002, 2004) and Dokhane et al. (2007) individually to explore the nature of out-of-phase instability excited by the feedback gain of the first azimuthal neutronic mode. Dutta and Doshi (2009) investigated the out-of-phase oscillations between two halves of reactor core, each containing ten representative channels. They found that the out-of-phase instability could be suppressed by the void-Doppler reactivity feedback. On the other hand, some researchers employed best estimate whole system codes to accurately predict the quantitative behaviors of out-of-phase instability and to assure the operational safety of BWRs. Costa et al. (2008) analyzed a hypothetical out-of-phase instability in a BWR using RELAP5/PARCS code. They successfully simulated the out-of-phase transients with a typical frequency of about 0.5 Hz adopting the data from Peach Bottom BWR. Recently, both Wysocki et al. (2014) by TRACE/PARCS and Dokhane et al. (2014) with SIMULATE-3K demonstrated out-of-phase oscillations in BWRs with a rotating behavior between higher azimuthal neutronic modes through nonlinear coupling interaction between them.

Complex nonlinear phenomena might appear in a two-phase flow system, especially in a multi-channel nuclear-coupled boiling system, for which the large system codes were not suited to analyze it because of their complexity and time-consuming. A unique type of chaotic attractor, the so-called complex strange attractor, can be induced by the interactions among multiple loops or multiple boiling channels under some specific operating conditions. A complex strange attractor evolved from the Lorenz attractor as a result of loop interactions was reported by Satoh et al. (1998) in their single-phase natural circulation double loop system. Lee and Pan (2005) reported another type of complex strange attractor