Progress in Nuclear Energy 83 (2015) 73–81

Contents lists available at ScienceDirect

Progress in Nuclear Energy

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

Experimental study of natural circulation instability with void reactivity feedback during startup transients for a BWR-type SMR

Shanbin Shi [a], Zeyun Wu [a], Zhuo Liu [a], Joshua P. Schlegel [b], Caleb S. Brooks [c], Jaehyuk Eoh [a], Yikuan Yan [a], Yang Liu [d], Won Sik Yang [a], Mamoru Ishii [a,*]

[a] School of Nuclear Engineering, Purdue University, 400 Central Drive, West Lafayette, IN 47907, USA
[b] Department of Mining and Nuclear Engineering, Missouri University of Science and Technology, 301 W 14th St, Rolla, MO 65409, USA
[c] Department of Nuclear, Plasma, and Radiological Engineering, University of Illinois, 104 South Wright St, Urbana, IL 61801, USA
[d] Department of Mechanical Engineering, Virginia Tech, 460 Old Turner St, Randolph Hall, Blacksburg, VA 24061, USA

ARTICLE INFO

Article history:
Received 20 November 2014
Received in revised form
2 March 2015
Accepted 10 March 2015
Available online

Keywords:
Flow instability
Void reactivity feedback
Fuel dynamics
Startup transients

ABSTRACT

The natural circulation boiling type SMR can experience flow instability during the startup transients due to the void reactivity feedback. A BWR-type natural circulation test loop has been built to perform the nuclear coupled startup transient tests for Purdue Novel Modular Reactor (NMR). This test loop is installed with different instruments to measure various thermal hydraulic parameters. The testing process can be monitored and controlled through PC with the assistance of LabVIEW procedure. The effects of power ramp rate on the flow instability during the nuclear coupled tests were investigated by controlling the power supply based on the point kinetics model with coolant void reactivity feedback. Two power ramp rates were investigated and the results were compared with those of the thermal hydraulic startup transients without void reactivity feedback. The time trace of power supply, system pressure, natural circulation rate, and void fraction profile are used to determine the flow stability during the transients. The results show that nuclear coupled startup transients also experience flashing instability and density wave oscillations. The power curves calculated from point kinetics model for startup transients show some fluctuations due to void reactivity feedback. However, the void reactivity feedback does not have significant effects on the flow instability during the startup procedure for the NMR.

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1. Introduction

Flow instability is an important subject to study for the initial startup, normal operation and accidents management for a natural circulation boiling water reactor (NCBWR). The flow instability in boiling water reactor (BWR) affects the design, control and safety of the reactor. The investigation of flow instability was initially of very interest to the thermal–hydraulic researchers. However, the nuclear-coupled flow instability obtained additional interest after flow oscillations reported in two commercial BWRs, i.e. Coarso in Italy (Gialdi et al., 1985) and LaSalle 2 in USA (US NRC, 1988). The first event was widely reported to experience neutron flux oscillations in the out-of-phase mode. The second event was reported to experience an excessive neutron flux oscillation while it was on the natural circulation after the pump trip. Muto et al. (1990) investigated the nuclear thermal hydraulic instability and thermal margin of LaSalle-2 using the time-domain code STANDY. Rao et al. (1995, 1996) studied the nuclear-coupled flow instabilities of two-phase flow in a boiling channel by taking into account of void reactivity coefficient and fuel time constant. Uehiro et al. (1996) applied the linear stability analysis in the frequency domain to reveal the instability mechanism for parallel boiling channels. Two modes of flow instability, i.e. in phase mode and out of phase mode, were found to exist in the stability map. Koh and Hagen (1997) used the scaled natural circulation loop DESIRE to study the effects of real-time void reactivity feedback on the power by taking into account the reactor fuel time constant. Van Bragt and Hagen (1998) performed a parametric study by coupling neutronics to thermal–hydraulics for the stability of natural circulation BWRs. It was found that the neutronic feedback effect stabilized the Type-I oscillation (caused by gravitational pressure drop in the riser) but destabilized Type-II oscillation (caused by frictional pressure drop). Furuya et al. (2005) developed the SIRUS-N facility to investigate regional and core-wide stability of natural circulation BWRs. The

* Corresponding author. Tel.: +1 765 494 4587; fax: +1 765 494 5951.
E-mail address: ishii@purdue.edu (M. Ishii).

http://dx.doi.org/10.1016/j.pnucene.2015.03.003
0149-1970/Published by Elsevier Ltd.

Nomenclature

Latin letters

A	area (m^2)
K_α	void reactivity coefficient (−)
K_D	Doppler coefficient (−)
l_{oR}	length ratio (−)
$n(t)$	neutron amplitude function (−)
$n_E(t)$	heater power input for the control system (kW)
t	time (s)
T	temperature (K)

Greek letters

α	void fraction (−)
β	effective fraction of delayed neutrons (−)
λ	precursor decay constant (s^{-1})
$\bar{\lambda}$	one group decay constant (s^{-1})
Λ	neutron generation time (s)
ξ	reduced precursor concentration (−)
ρ	total reactivity (−)
ρ_a	void reactivity (−)
ρ_D	Doppler reactivity (−)
ρ_{ext}	external reactivity (−)
τ_{AD}	artificial time delay constant (s)
τ_c	fuel element time constant (s)

Subscripts

f	liquid
F	fuel element
$F\infty$	fuel element to coolant
i	ith group of precursor
o	reference point/component
M	model
P	prototype
p	fuel pellet
R	ratio of model over prototype

experimental results showed that stability margin was sufficient for the loop under normal operating conditions with considering void reactivity feedback and fuel-rod heat conduction. Recently, Kuran (2006) derived the scaling criteria for fuel heat conduction with void reactivity feedback between the prototype and the model. Based on Kuran's work, Woo (2008) performed the quasi-steady tests to obtain the stability map for a natural circulation BWR. The tests showed that the power oscillation caused by void fraction oscillation through void reactivity feedback might be damped at low frequency oscillation at low pressure.

All the work mentioned above are mainly focus on flow instability phenomena in conventional BWRs. Recently Small Modular Reactor (SMR) designs are receiving a surge of interest in nuclear community due to their inherent safety and other advanced features. To better satisfy the safety concerns in SMR design, it is necessarily needed to investigate the flow instability related to SMR designs which has some essential differences from those of conventional BWRs. In this paper, the authors design a natural circulation test facility to investigate the flow instability during the startup transients and quasi-steady states of the Purdue's 50-MWe Novel Modular Reactor (NMR-50) concept (Ishii et al., 2013). The thermal hydraulic flow instability occurring during the initial startup transients was investigated using different power ramp rates (Ishii et al., 2014). The initial startup transients can be generally divided into four phases, i.e. single-phase natural circulation in core, periodic flashing in chimney, core net vapor generation also called transition phase, and two-phase natural circulation. The flashing instability and density wave oscillations are the two main instability mechanisms observed. The flashing instability is mainly observed in the phase of core net vapor generation, and the density wave oscillation is observed during the two phase natural circulation. In this paper, the effects of void reactivity feedback on the core wide flow instability are investigated during the initial startup transients. The point kinetics model (PKM) is utilized to control the power supply to mimic the power behavior during the startup transients. The heat conduction difference between the reactor fuel and the loop heater rod is also addressed with void reactivity feedback.

This paper describes the point kinetics model for the void reactivity feedback in Section 2. Section 3 discusses the difference between the reactor fuel and the electric heater rods used in test facility. Section 4 shows the experimental test facility and control program for the startup transients. The experimental results and analyses are presented in Section 5. Finally, Section 6 summarizes the key conclusions of this research.

2. Neutron kinetics

2.1. Point kinetics model

For the BWR startup tests, the point kinetics model (PKM) gives a good estimate for the core-wide oscillations in the NCBWR where the whole core oscillation is in phase (Woo, 2008). Thus the PKM was used in this study to control the power supply in a way to represent the power behavior with coolant void feedback during the startup transients of NMR-50. The integration of the time-dependent neutron transport and precursor equations yields the exact point kinetics equations for the total flux and precursor levels integrated over the energy and the spatial domain (Ott, 1985). These equations contain no approximation as long as the rigorous definitions of the kinetics parameters are used, i.e., the actual time-dependent flux shape is calculated. The exact point equations are reduced to the point kinetics approximation under the assumption that the initial flux shape remains constant. Using the standard notation, the point kinetics equations for the neutron flux amplitude function $n(t)$ and the six-group reduced precursor concentrations ξ_i can be written as (Ott, 1985)

$$\frac{dn(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda} n(t) + \frac{1}{\Lambda} \sum_{i=1}^{6} \lambda_i \xi_i \tag{1}$$

$$\frac{d\xi_i(t)}{dt} = -\lambda_i \xi_i(t) + \beta_i n(t), \quad i = 1, 2, \ldots, 6 \tag{2}$$

where β_i is the delayed neutron yield of the ith group, λ_i is the decay constant of ith group delayed neutron precursor, ξ_i is the ith group reduced precursor concentration, $\rho(t)$ is the dynamic reactivity, and Λ is the neutron generation time.

During the startup transients, the variation of the core reactivity $\rho(t)$ has an inherent relationship with the change of the neutron flux amplitude, which can be thought of linearly proportional to the

power level. Therefore it is crucial in this research to develop a reactivity feedback model that can accurately predict the power behavior. For the analysis of NMR-50 startup transient behaviors, the dynamic reactivity $\rho(t)$ can be decomposed into the following three components:

$$\rho(t) = \rho_{\text{ext}}(t) + \rho_{\alpha}(t) + \rho_{\text{D}}(t) \tag{3}$$

where $\rho_{\text{ext}}(t)$: reactivity due to movements of control rods or other control elements, $\rho_{\alpha}(t)$: reactivity due to void fraction change or moderate density change, $\rho_{\text{D}}(t)$: reactivity due to fuel pellet temperature change or the Doppler effect.

The external reactivity ρ_{ext} represents the reactivity inserted by control rods movement to produce the scaled power ramp. The moderator density change due to void fraction variation and the nuclear fuel temperature change are the main reactivity feedback mechanisms in a BWR. The two reactivity coefficients are defined by

$$K_{\alpha} = \frac{\partial \rho}{\partial \langle \alpha \rangle} \tag{4}$$

$$K_{\text{D}} = \frac{\partial \rho}{\partial \langle T \rangle_{\text{p}}} \tag{5}$$

where $\langle \alpha \rangle$ is the volume-averaged void fraction in the core, and $\langle T \rangle_{\text{p}}$ is the averaged fuel pellet temperature. In the current startup experiment for the BWR test facility, the void reactivity feedback is the predominant feedback mechanism compared to the Doppler-reactivity. The Doppler-reactivity becomes important when there are large amplitude power oscillations causing significant fuel temperature changes.

2.2. Reactivity calculation

In the simulation of the nuclear-coupled startup transient, the external reactivity can be calculated to produce a given power ramp. The reactivity $\rho(t)$ to produce a power ramp $n(t)$ can be obtained by solving the inverse problem of the PKM. The precursor equation can be solved analytically with the one delayed neutron group approximation. The PKE with one delayed neutron group can be written as

$$\frac{\mathrm{d}n(t)}{\mathrm{d}t} = \frac{\rho(t) - \beta}{\Lambda} n(t) + \frac{1}{\Lambda} \bar{\lambda} \xi(t) \tag{6}$$

$$\frac{\mathrm{d}\xi(t)}{\mathrm{d}t} = -\bar{\lambda}\xi(t) + \beta n(t) \tag{7}$$

$$\rho(t) = \rho_{\text{ext}}(t) + \rho_{\alpha}(t) \tag{8}$$

with the steady state initial conditions:

$$n(0) = n_0 \tag{9}$$

$$\xi(0) = \xi_0 = \frac{\beta}{\bar{\lambda}} n_0 \tag{10}$$

$$\alpha(0) = \alpha_0 \tag{11}$$

$$\rho(0) = 0 \tag{12}$$

where $n(t)$ and $\xi(t)$ are neutron and reduced precursor concentration respectively. β is the total delayed neutron yield for six groups

$$\beta = \sum_{i=1}^{6} \beta_i \tag{13}$$

and $\bar{\lambda}$ is the effective one group decay constant defined as

$$\frac{1}{\bar{\lambda}} = \frac{1}{\beta} \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i} \tag{14}$$

The effective decay constant $\bar{\lambda}$ highlights the importance of the long-lived precursor groups. This gives accurate results in slow transients such as reactor startup.

Using the initial conditions in Eqs. (9)–(12), Eq. (7) can be solved for the reduced precursor concentration as

$$\xi(t) = \frac{\beta}{\bar{\lambda}} n_0 e^{-\bar{\lambda}t} + \beta \int_0^t n(t') e^{-\bar{\lambda}(t-t')} \mathrm{d}t' \tag{15}$$

Therefore, by inserting Eq. (15) into Eq. (6), the reactivity for a known power transient $\tilde{n}(t)$ can be determined as

$$\begin{aligned} \tilde{\rho}(t) = & \Lambda \frac{\mathrm{d}}{\mathrm{d}t} \{\ln[\tilde{n}(t)]\} \\ & + \beta \left[1 - \frac{\tilde{n}_0}{\tilde{n}(t)} e^{-\bar{\lambda}t} - \frac{\bar{\lambda}}{\tilde{n}(t)} \int_0^t \tilde{n}(t') e^{-\bar{\lambda}(t-t')} \mathrm{d}t' \right] \end{aligned} \tag{16}$$

In a startup thermal–hydraulic transient without reactivity feedback, the power transient $\tilde{n}(t)$ can be expressed as a linear function of time as

$$\tilde{n}(t) = a_0 + a_1 t \tag{17}$$

In this case, the reactivity in Eq. (16) can be calculated as

$$\tilde{\rho}(t) = \beta + \Lambda \frac{a_1}{\tilde{n}(t)} - \frac{\beta}{\tilde{n}(t)} \left[a_0 + a_1 t - \frac{a_1}{\bar{\lambda}} \left(1 - e^{-\bar{\lambda}t} \right) \right] \tag{18}$$

If the void fraction transient corresponding to the known power transient $\tilde{n}(t)$ is given by $\tilde{\alpha}(t)$, the reactivity in Eq. (18) represents the sum of the external reactivity and the void reactivity feedback:

$$\tilde{\rho}(t) = \rho_{\text{ext}}(t) + \rho_{\tilde{\alpha}}(t) \tag{19}$$

Consequently, if the external reactivity insertion in Eq. (16) is to be simulated using a known power transient $\tilde{n}(t)$ with the associated void fraction transient $\tilde{\alpha}(t)$, the reactivity in Eq. (6) should be evaluated as

$$\rho(t) = \rho_{\text{ext}}(t) + \rho_{\alpha}(t) = \tilde{\rho}(t) - \rho_{\tilde{\alpha}}(t) + \rho_{\alpha}(t) \tag{20}$$

where $\tilde{\rho}(t)$ is given by Eq. (16) and the void coefficient $\rho_{\alpha}(t)$ can be obtained through a perturbation approach with the core simulation. The detailed procedure of generating power supply for the nuclear coupled transients can be seen in Fig. 1.

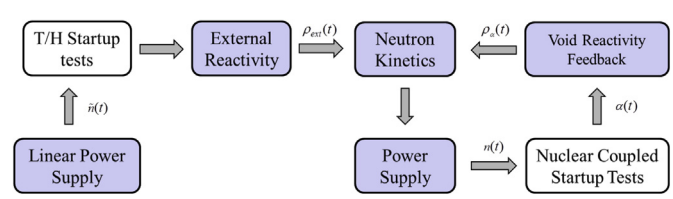

Fig. 1. Procedure to calculate the power supply with void reactivity feedback.

3. Fuel cycle design and analyses

In order to simulate the effect of void reactivity feedback in nuclear coupled tests, the differences between the electric resistance heaters and typical fuel element must also be considered. As can be seen in Fig. 2, there are similarities between an NMR-50 fuel element and commercial electric heater rods used in the test facility. Kuran (2006) utilized the two-region lumped model to describe the fuel dynamics for both fuel element and electric heater rods in his Ph.D thesis. In a typical electric heater rod, usually magnesium-oxide is used for electric insulation in the oxide central region. The heating coils are placed near the periphery of this region. The oxide-region is enclosed without a gap by cladding material, which is normally stainless steel or incoloy alloys.

The fuel time constant, which characterizes the time needed to transfer the heat to the coolant, is different between the reactor fuel element and the electric heater rod due to different geometry, structure and material property. In nuclear coupled test considering the void reactivity feedback, the power level for a new step is calculated from Eq. (21). The similarity criteria for the fuel dynamics between the fuel element and the electric heater rod are dominated by the Fourier number (Kuran, 2006). Thus, when calculating the heater power response in void reactivity simulation, a delay equation is needed in addition to the equation for the neutron kinetics as

$$\tau_{AD}\frac{dn_E(t)}{dt} = n(t) - n_E(t) \tag{21}$$

where $n(t)$ is the solution from the point kinetic equations and $n_E(t)$ is the signal which is sent to the heater power controller. The artificial time delay, τ_{AD}, can be determined as

$$\tau_{AD} = [\tau_c]_P\sqrt{l_{oR}} - [\tau_c]_M \tag{22}$$

where $\tau_c = (\langle\rho c\rangle_F A_F)/U_{F\infty}$ is the time constant of the fuel element or the heater rod.

The difference between the NMR-50 fuel elements and the electric heater rods should be identified. Table 1 gives the geometrical and thermo-physical information for the NMR-50 fuel elements and the facility electric heaters. The NMR-50 time constant ($[\tau_c]_P$) is about 6.7 s under 1000 W/m^2 of gap conductance (Woo, 2008). The heater rod time constant ($[\tau_c]_M$) calculated based on the design parameter in the experiment is about 2.13 s. The artificial time delay enforced between the heater power and nuclear reactor power will be about 4.0 s. That is, τ_{AD} in Eq. (22) can be approximately set as 4.0 s in the nuclear coupled startup transients.

Table 1
Geometrical data and thermo-physical properties of the NMR-50 fuel element and the facility heaters.

	NMR-50	Facility
Fuel pellet outer radius (mm)	4.585	6.475
Cladding outer radius (mm)	5.276	9.475
Number of rods	23,296	4
Pellet thermal conductivity (W/m/K)	3.4	45
Cladding thermal conductivity (W/m/K)	14.3	13.85
Gap conductance (W/m^2/K)	Varied	n/a

4. Experimental test facility

A stainless steel natural circulation boiling test loop shown in Fig. 3 is scaled down from the Purdue's NMR-50 (Ishii et al., 2013) to study the nuclear coupled flow instability during the startup transients. The geometrical parameters of the test facility are determined based on a previous design of a simulated reactor core (Dixit et al., 2013). This test loop is about 7 m in elevation and composed of heated section, chimney (riser), separator, downcomer, and inlet plenum. The electric heater rods in the core section are arranged in a 2 × 2 layout with an active heated length of 1.13 m. The inner diameters of the core and riser section are close to a commercial three-inch pipe. Several Honeywell differential pressure transducers and absolute pressure transducers are installed to measure the inlet pressure differences and system pressures. T-type thermocouples are used to measure local temperatures at lower plenum, core inlet, instrumentation ports, steam dome, downcomer and so on. Three core impedance probes (IMP01–IMP03) and four probes (IMP04–IMP07) are installed to measure the void fractions at different locations. The home-made impedance probes can reach 0.5% in absolute value for low void fraction measurement. In addition, two Honeywell magnetic flow meters are installed to measure the loop natural circulation flow rate and condensation flow rates. A LabVIEW procedure is compiled to realize the operation and controlling through PC, which is also integrated with data acquisition model. Fig. 4 shows the display panel in LabVIEW for the nuclear coupled tests. All measured thermal–hydraulic parameters and void reactivity feedback coefficient for nuclear coupling can be displayed in the front panel.

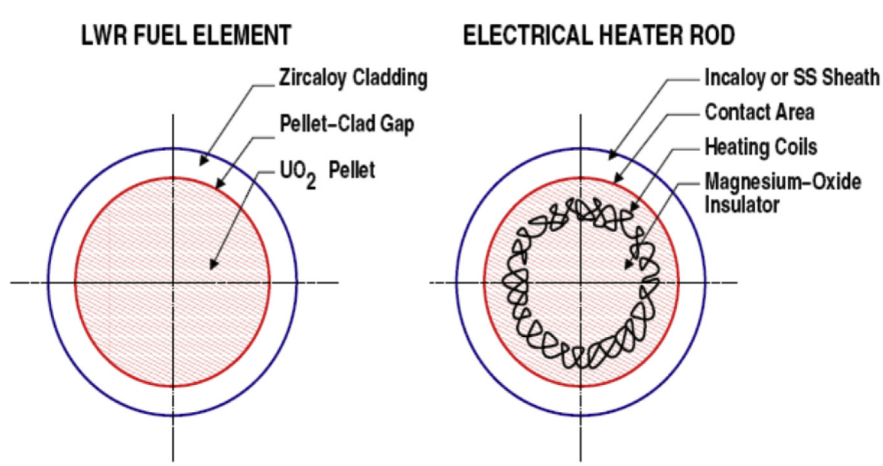

Fig. 2. Fuel element and electric heater rod (Kuran, 2006).

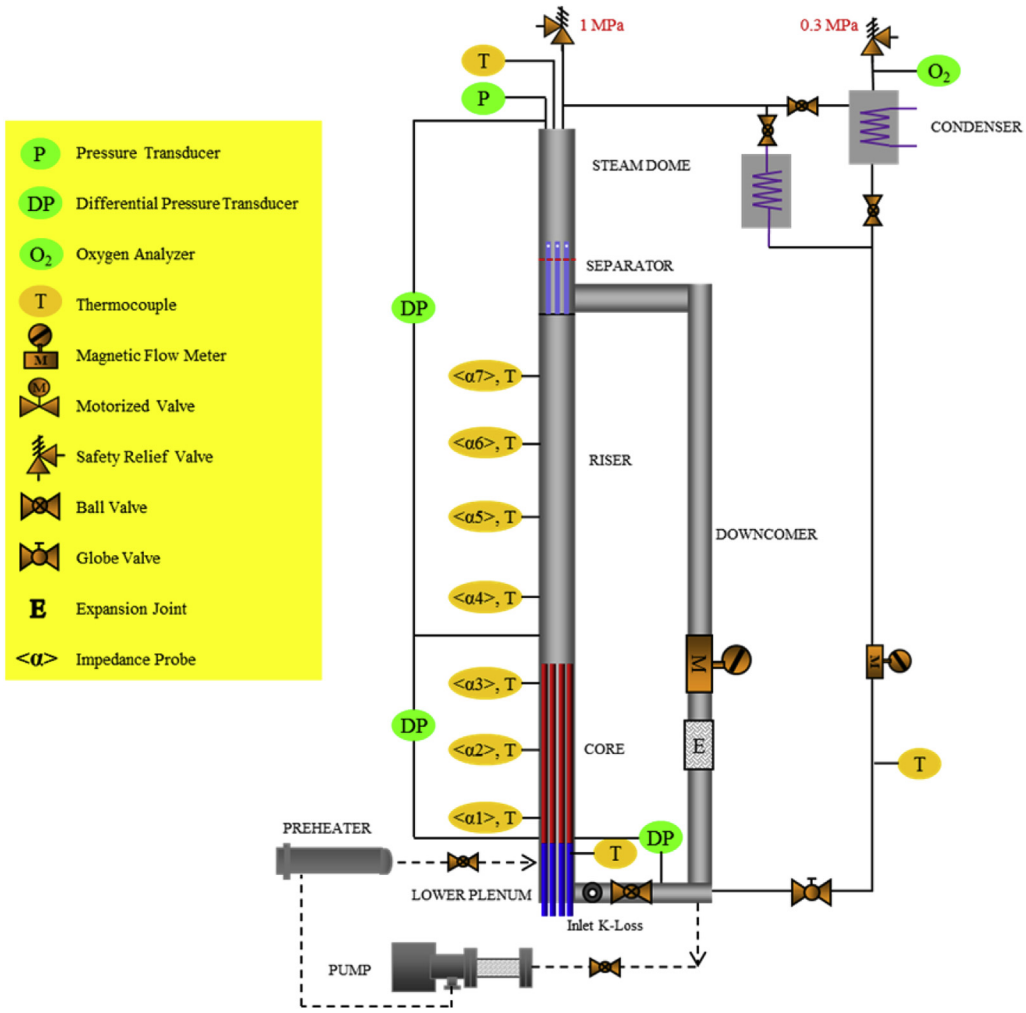

Fig. 3. Schematic diagram of experimental loop.

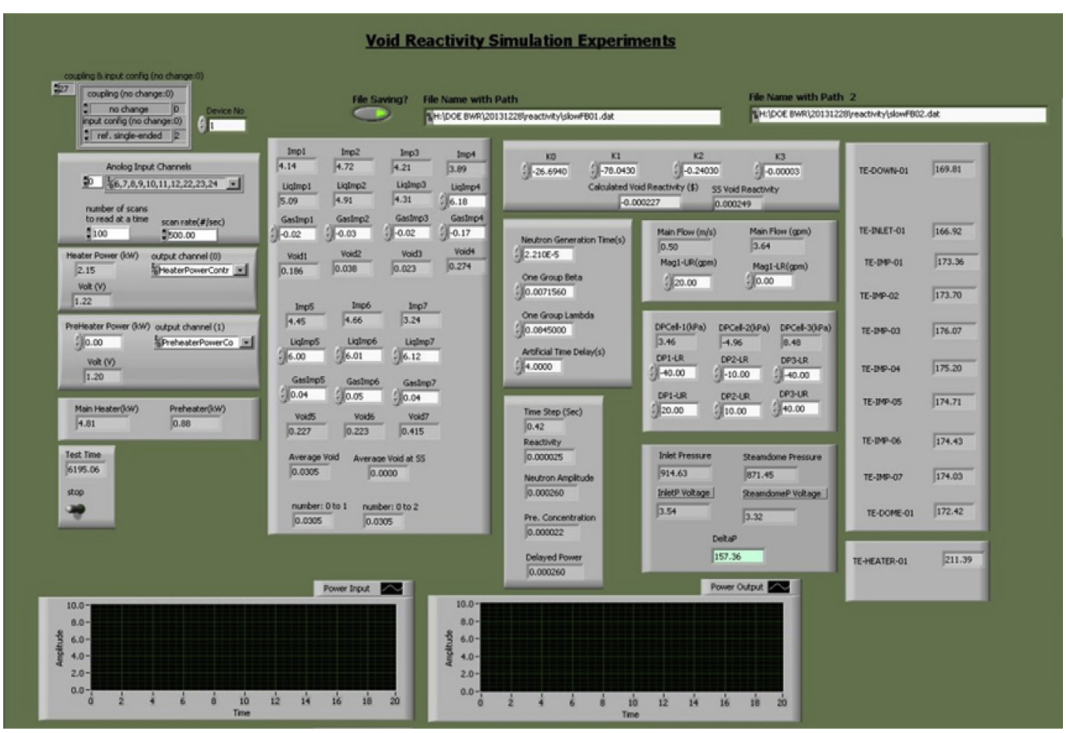

Fig. 4. Display panel for the nuclear coupled test.

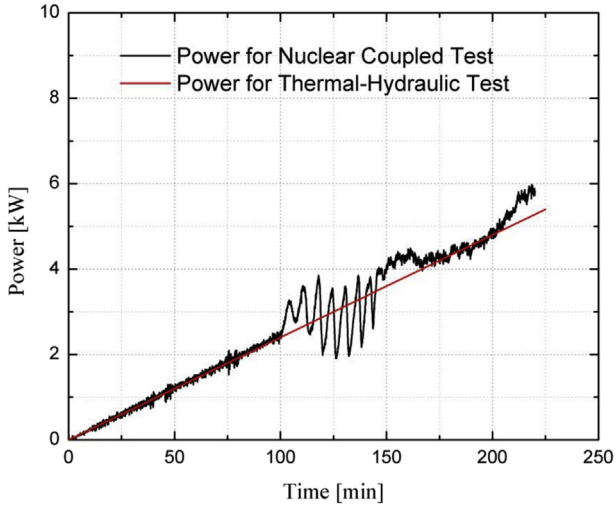

Fig. 5. Main heater power for the slow startup transient with void reactivity feedback.

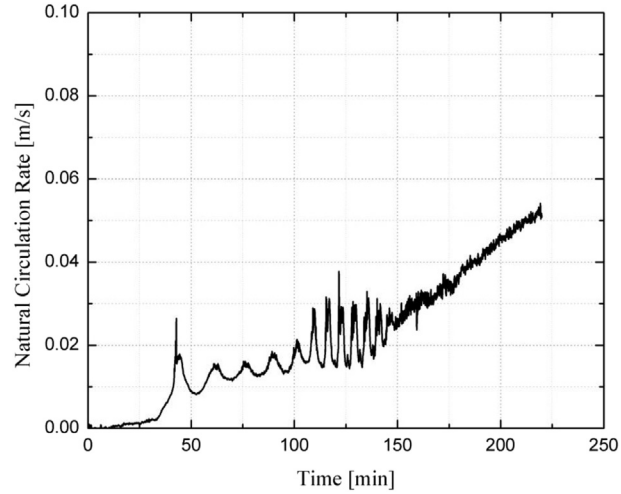

Fig. 7. Natural circulation rate for the slow startup transient with void reactivity feedback.

5. Test results

The startup transient tests with void reactivity feedback were carried out at two different power ramp rates, which are simulated using the PKM discussed in Section 2. The initial water level for the startup transient is set at 5.85 m. The previous thermal–hydraulic tests (Ishii et al., 2014) show that four phases exist during the low pressure startup transients, i.e. single-phase natural circulation in core, periodic flashing in chimney, core net vapor generation phase (transition phase), and two-phase natural circulation. The process of nuclear coupled tests can also be generally divided into above four phases. However, emphasis is put on the core net vapor generation phase and two-phase natural circulation with regard to void reactivity feedback mechanism.

5.1. Slow heat-up startup transients with void reactivity feedback

The results for the slow heat-up startup transient tests with void reactivity feedback are shown in Figs. 5–10. The main heater power is determined from the PKM and required to follow the known power ramp for the thermal–hydraulic tests under similar conditions. Fig. 5 displays the power curves for the nuclear coupled test with void reactivity feedback and the thermal hydraulic test without reactivity feedback.

The flow behaviors in the first 100 min are almost single-phase natural circulation in the core section, and the void reactivity feedback can be neglected. The void reactivity feedback affects the power supply if there is a void fraction fluctuation in heated section. The output power shows certain oscillations in the transition phase from 100 min to 150 min. Due to the unstable flow conditions and heat conduction at the time of adding void reactivity, the system can have either positive feedback or negative feedback at the beginning such as the power ascension around 100 min in Fig. 5. It can be seen that the nuclear coupled power curve is able to stabilize around the reference power curve in the transition phase. After the transition phase, two-phase natural circulation is generated with much smoother power curve. Because of the limitation of the experimental conditions, the experiments stop at 220 min. However, it can be expected that the power curve follows the linear power curve by using the reactivity feedback model.

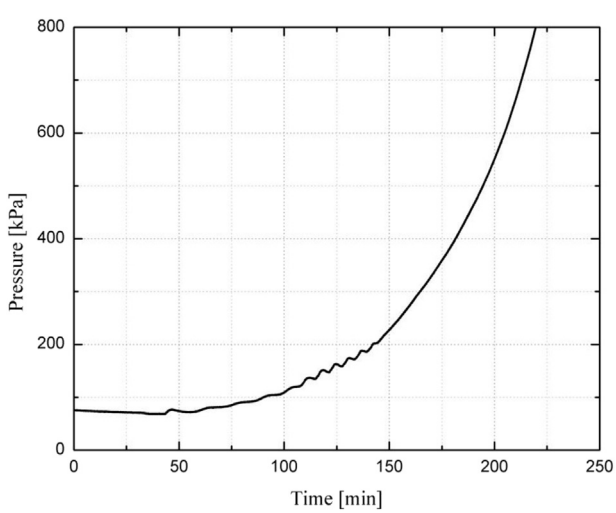

Fig. 6. Steam dome pressure for the slow startup transient with void reactivity feedback.

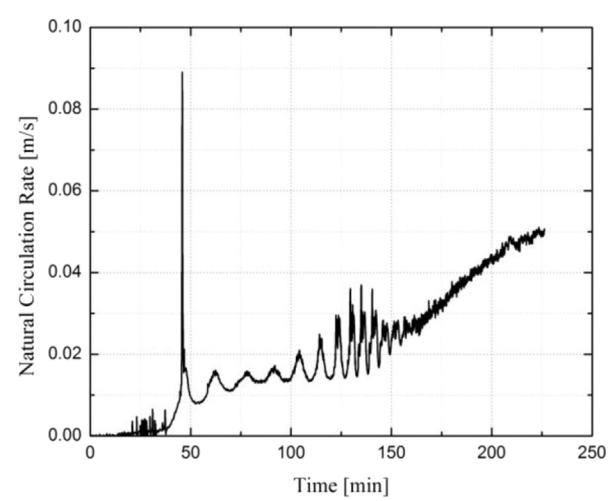

Fig. 8. Natural circulation rate for the slow startup transient without void reactivity feedback.

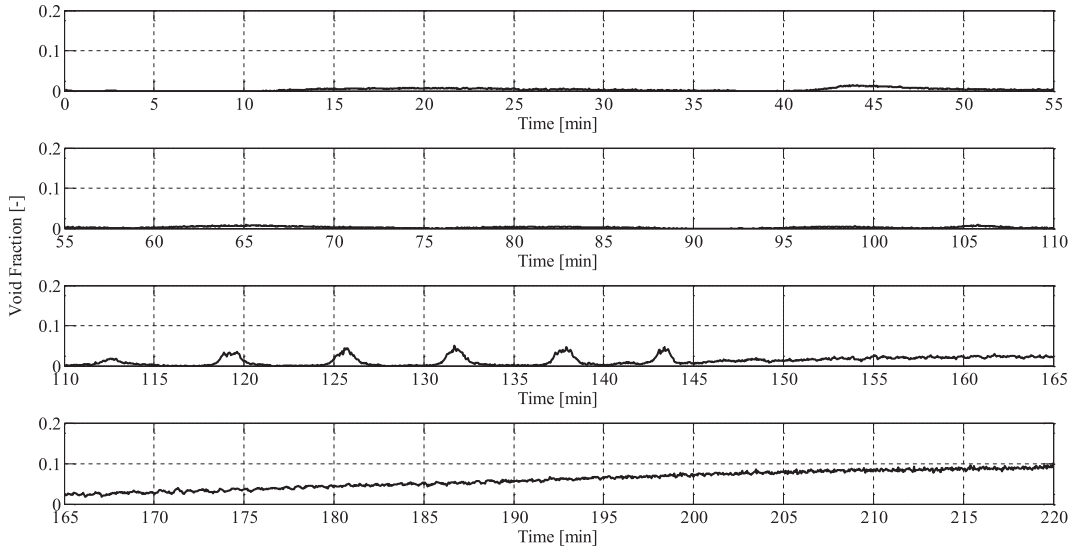

Fig. 9. Void fraction at the core exit (IMP03) for the slow startup transient with void reactivity feedback.

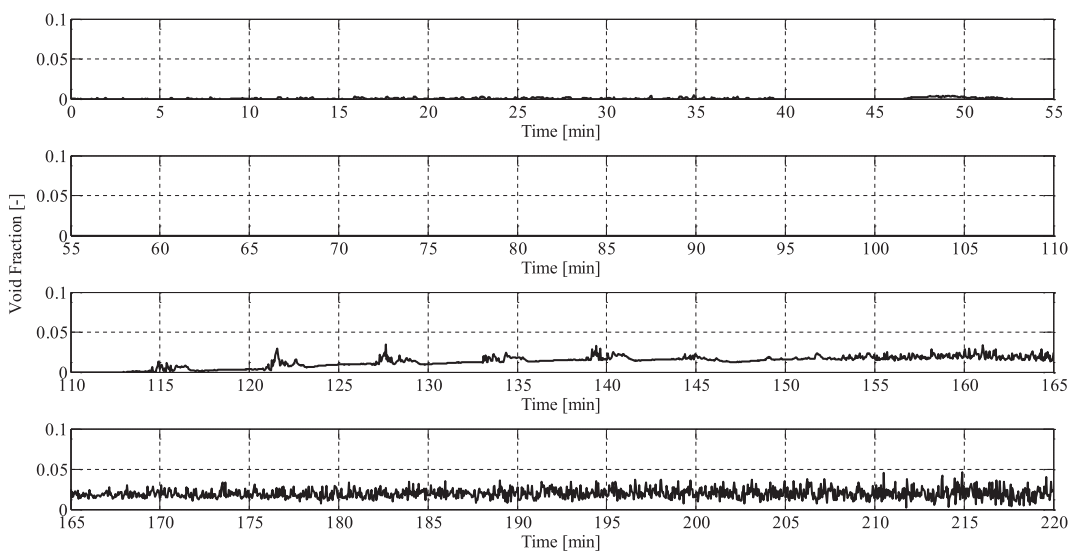

Fig. 10. Void fraction at the chimney outlet (IMP07) for the slow startup transient with void reactivity feedback.

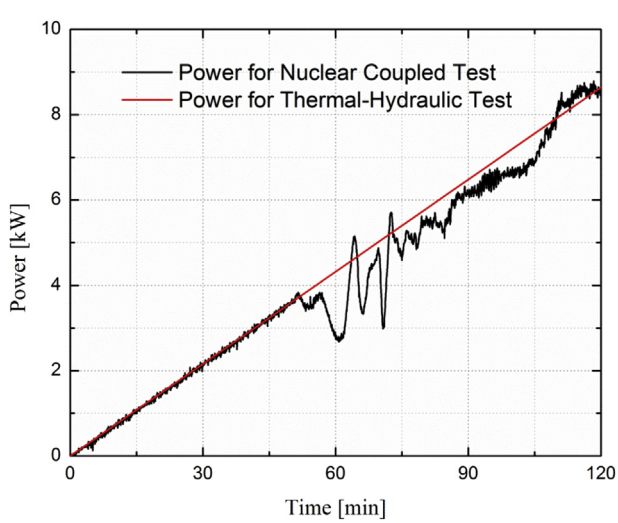

Fig. 11. Main heater power for the fast startup transient with void reactivity feedback.

The steam dome pressure profile during the slow heat-up startup transient with void reactivity feedback is demonstrated in Fig. 6. The overall trend of the pressure response is very similar to the thermal–hydraulic startup simulation. At the beginning of the test, the steam dome pressure is almost constant and close to the saturated steam pressure in the steam dome due to single-phase natural circulation. The steam dome pressure does not show big oscillations during the transition phase due to the flashing effect in the chimney. After then the steam dome pressure rises exponentially with much more vapor generation inside the test facility.

Fig. 7 shows the natural circulation rate for the slow heat-up case. The average flow velocity increases from 1.5 cm/s of single-phase natural circulation to 5 cm/s of two-phase natural circulation. The average flow velocity experiences periodic oscillations caused by flashing during the first 150 min. Due to the large amount of subcooling in the test section during the transition phase, the base power with fluctuations is not large enough to alter the flow regime in a short period. So the power oscillation does not show significant effects on the stability of natural circulation rate. The oscillation pattern is pretty much similar to that of the slow startup

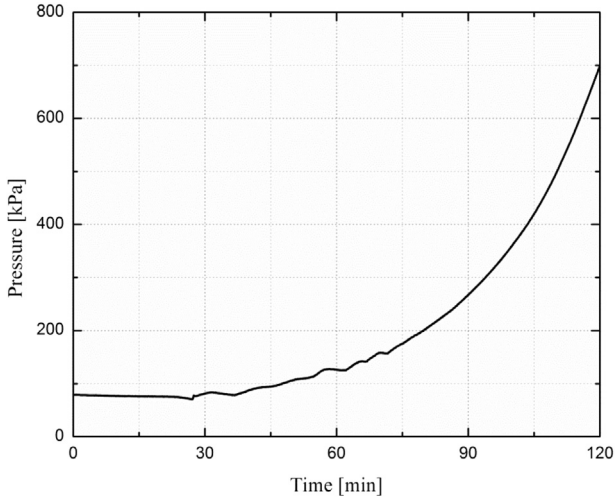

Fig. 12. Steam dome pressure for the fast startup transient with void reactivity feedback.

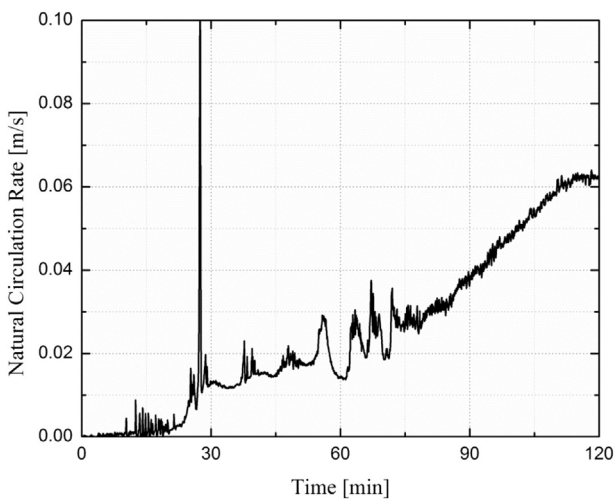

Fig. 13. Natural circulation rate for the fast startup transient with void reactivity feedback.

transients shown in Fig. 8 without considering the void reactivity feedback. The first peak in Fig. 8 occurs at an earlier stage when loop natural circulation rate is low. So the coolant heated with long residence time in the core section flashes extensively at the top of chimney.

In the two-phase natural circulation, the time delay of void reactivity feedback is much shorter than that in the transition phase. When the system becomes saturated, the void fraction will change in a shorter time with varied power level. As far as the oscillation frequency is concerned, density wave oscillation is a high frequency oscillation while flashing instability with large subcooling is a low frequency oscillation. Combined with the void fraction profile shown in Figs. 9 and 10, it can be seen that the void reactivity feedback has trivial effects on the flashing induced flow instability observed during the nuclear coupled slow startup transients.

5.2. Fast heat-up startup transients with void reactivity feedback

In order to investigate the effects of heat flux on the flow instability in startup transients with void reactivity feedback, the fast heat-up startup transient tests were performed based on the power curve for the startup thermal–hydraulic transients without void reactivity feedback. Fig. 11 shows the comparison between the reference power curve and the one used in the nuclear coupled test. Although the nuclear coupled power curve can follow the reference power curve, the average power deviation from the linear power is about 0.5 kW, which is larger than 0.3 kW in the slow heat-up case. Fig. 12 shows the steam dome pressure profile for the fast startup transient test with void reactivity feedback. The total test time for this condition is about 120 min.

Fig. 13 displays the time trace of the natural circulation rate during this test. Flashing occurring in the phase of single-phase natural circulation increases the loop flow velocity. In the phase of net vapor generation, condensation at the chimney inlet and flashing near the top of the chimney can cause the intermittent oscillations from 50 min to 70 min. Figs. 14 and 15 show the time trace of void fraction during the nuclear-coupled fast startup transients. Density wave oscillations can be observed in Fig. 14 in the earlier period of two-phase natural circulation from 90 to 95 min and diminish with the power density continuously rising, which is similar as those of the thermal–hydraulic fast startup transients.

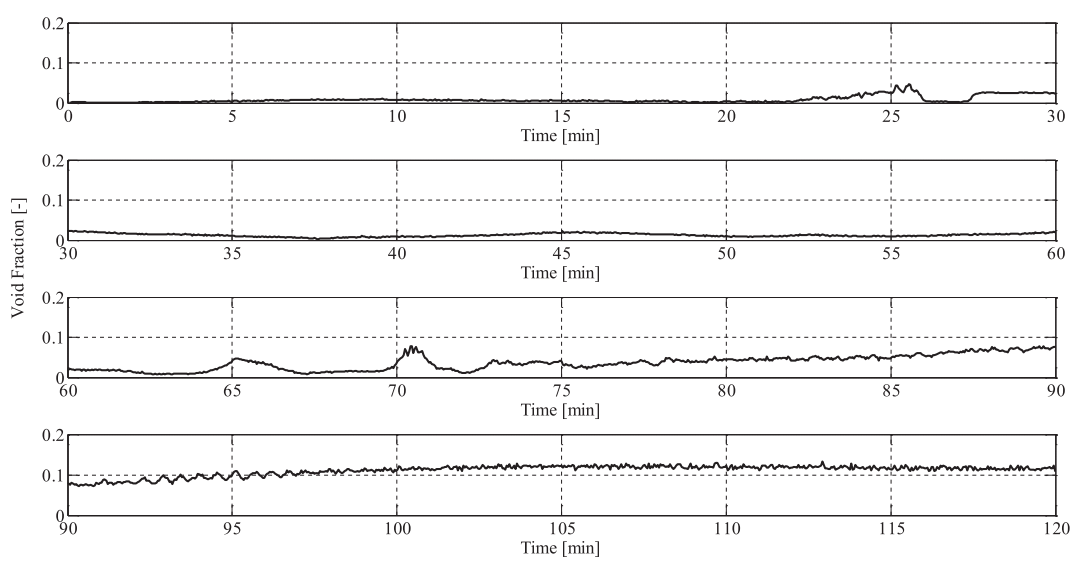

Fig. 14. Void fraction at the core exit (IMP03) for the fast startup transient with void reactivity feedback.

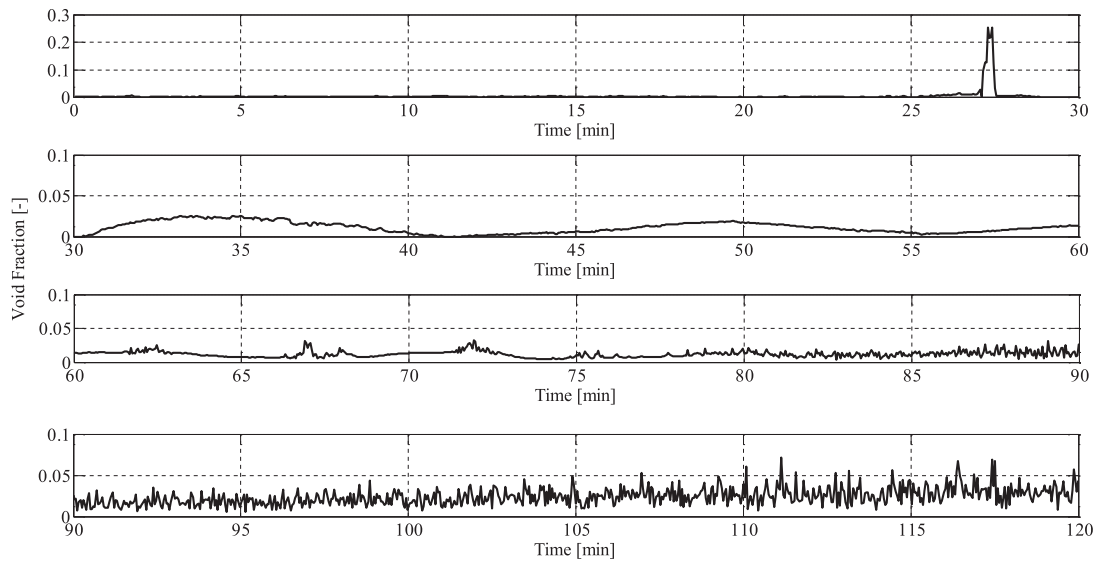

Fig. 15. Void fraction at the chimney outlet (IMP07) for the fast startup transient with void reactivity feedback.

6. Conclusions

The startup transients with void reactivity feedback are a series of research following the thermal–hydraulic startup transients to study the flow instability for the BWR-type SMR. The point-kinetic model (PKM) is used in this paper to calculate the power supply for the nuclear coupled startup transients. The difference of fuel dynamics between the electric heaters and the reactor fuels is considered. The artificial time delay is described by a delay equation during the startup transients.

The startup transients are performed in a natural circulation boiling test loop, which is designed to study the flow instability for the BWR-type NMR. Two nuclear coupled startup transient tests with different power ramp rates were performed. In general, the experimental nuclear-coupled transients do not show significant differences from the thermal hydraulic startup transients for the NMR. The power curves show a few oscillations due to the void reactivity feedback owning to flashing instability in the net vapor generation phase. However, the power curves always follow the reference thermal–hydraulic power curves for both slow and fast startup transients. In the two-phase natural circulation, the void reactivity feedback has small effect on the reactor power due to low coolant inlet subcooling and high natural circulation rate.

Acknowledgments

This material is based upon work supported under a Department of Energy Nuclear Energy University Program.

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