Thermal-hydraulic stability of a natural circulation system with nuclear feedback*

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Abstract The stability of low temperature nuclear heating reactor with various subcoolings of reactor core inlet has been studied by means of simulating experiments. The thermal-hydraulic system and the data acquisition and processing system are presented. Especially, the process of realizing the simulating nuclear feedback is introduced in detail. Finally, the experimental results are discussed in the opinions of nuclear reactor physics and thermal-hydraulics. The conclusion is that the nuclear reactor can operate stably only when the subcooling of reactor core inlet is high enough.

Keywords Stability, Nuclear feedback, Density-wave oscillation, Natural circulation

1 Introduction

The problem of stability may arise in many two-phase flow systems, such as nuclear reactors, power station boilers, heat exchangers, liquid-propellant engines and some chemical facilities. An instable two-phase flow, of which thermal-hydraulic parameters often drift aperiodically or oscillate periodically, may bring about mechanical vibration or alternating thermal stress, even rapid degradation of performances. Thus, the instability of two-phase flow will make these systems run abnormally and unsafely.

Because water is not only coolant but also neutron moderator in light-water nuclear reactors, a change in void fraction may lead to a change in nuclear reactivity. Therefore, the thermal-hydraulic instability may result in the instability of nuclear reaction. For this reason, it is necessary to study the stability of two-phase flows of nuclear reactors.

Especially for low temperature nuclear heating reactor, of which primary loop is a natural circulation system under low pressure and low dryness, the problem of flow stability is more striking. Because, as shown in Fig.1, void fraction is susceptible to dryness under low pressure when dryness is relatively low, that is, a very little variation of dryness may result in a great change of void fraction. Therefore, in such a system, two kinds of feedback, thermal-hydraulic feedback and nuclear feedback, are very liable to be erected spontaneously. Referring to Fig.2, the processes of nuclear feedback and thermal-hydraulic feedback are manifested by the left and right circuits, respectively. Assuming that a little variation, for example, a little decrement of dryness happens because of

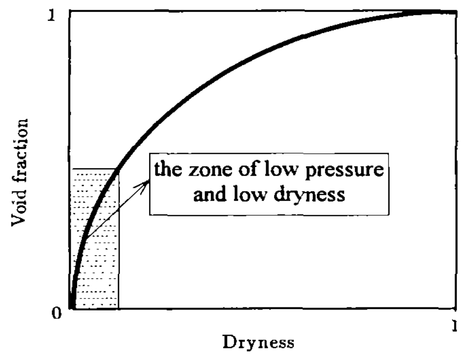

Fig.1 Relation between dryness and void fraction under low pressure

some random factors, then a relatively great decrement of void fraction is unavoidable. This decrement of void fraction, on one hand, will result in an increment of nuclear reactivity then an increment of nuclear power, finally lead to an increment of dryness, so it is obvious that the nuclear feedback is a negative one anyway. On the other hand, this decrement of void fraction will result in a decrement of driving pressure

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head and a decrement of two-phase friction factor simultaneously. The former, which is disadvantageous factor to circulation, will lead to a decrement of circulation flow rate then a decrement of dryness. In this sense, the thermal-hydraulic feedback is a positive one. Hence, the property of thermal-hydraulic feedback is contingent on specific system parameters, such as subcooling of reactor core inlet, altitude of reactor core and system pressure.[1] When the thermal-hydraulic feedback is positive, density-wave oscillations may occur because of the interdigitation between nuclear feedback links and thermal-hydraulic feedback links.

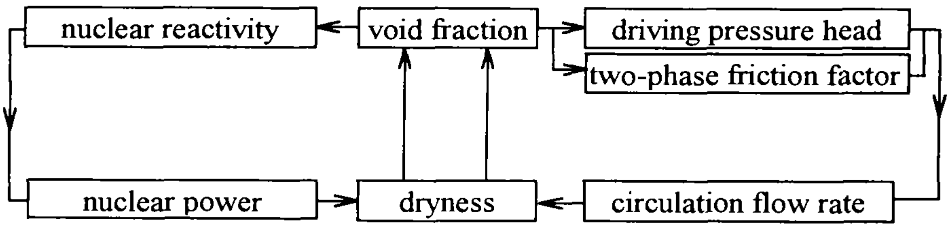


Fig.2 Nuclear feedback and thermal-hydraulic feedback

Previous studies on the stability of natural circulation system have been done without nuclear feedback[2~5], so it is necessary to research this topic with simulating nuclear feedback. Therefore, an experimental natural circulation circuit with simulating nuclear feedback, which simulates the primary loop of low temperature nuclear heating reactor, has been erected. Two experiments have been carried out on the system with different subcoolings of heating section inlet. One has been performed with a high inlet subcooling, the other with a low inlet subcooling. In the end, the stability of low temperature nuclear heating reactor has been discussed.

2 Experimental systems and methodologies

2.1 Thermal-hydraulic system

The principle diagram of experimental system is shown in Fig.3. The system includes two circuits, one is a primary loop, the other is an auxiliary loop, where heat is transferred to environmental atmosphere, a terminal heat sink by an air-cooler.

The primary loop, a natural circulation system, which simulates that of nuclear heating reactor, mainly includes a heating section (simulating reactor core), an ascending section, a descending section, a steam-water separator, a condenser and a heat exchanger. The loop is designed to meet the primary similarity criteria of single/two-phase flow.[6] Moreover, by comparison, the experimental loop has the same size of heating rod, the same hydraulic diameter, the same altitude of heating section, the same altitude of ascending section and the same circuit resistance coefficient as the primitive model. The main design parameters of the loop are shown as follows: system pressure $\leq$2.5 MPa, heating section altitude=1.2 m, ascending section altitude$\approx$3 m, inlet subcooling of the heating section is 5 to 30°C, outlet dryness of the heating section <0.05, maximal heating power=180 kW, upper temperature limit of the heating rod surface is 500°C.

In the process of circulation, subcooled liquid flows upward along the heating section driven by the pressure resulting from the density difference of coolant between the up-section (including the heating section and the ascending section) and the down-section (including the heat exchanger and the descending section). Owing to being heated, the liquid begins to micro-boil near the outlet of the heating section. Then the mixture of steam-water passes through the ascending section and gets to the separator. Here, the vapor being separated enters the condenser and is condensed to liquid in it, then comes back to the separator. Simultaneously, the liquid being separated in the separator, after being cooled in the heat exchanger, flows down the descending section, and returns to the heating section inlet through the throttle valve and the flow meter equipped on the bottom of the loop.

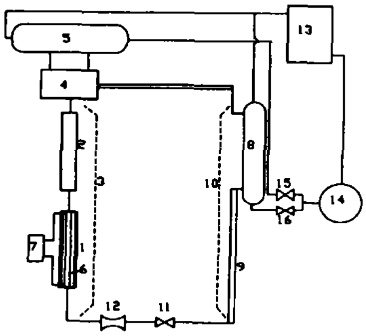


Fig.3 Principle diagram of thermal-hydraulic system
1 Heating section (simulating reactor core), 2 Ascending section, 3 Up-section, 4 Steam-water separator, 5 Condenser, 6 Heating rod, 7 Silicon controlled rectifier (SCR), 8 Heat exchanger, 9 Descending section, 10 Down-section, 11 Throttle valve, 12 Flow meter, 13 Air-cooler, 14 Pump, 15 Valve B, 16 Valve A

The auxiliary loop, a forced circulation loop, which is designed to adjust the operation conditions of the primary loop, consists of the condenser , the heat exchanger, the air-cooler, a pump and relating valves and pipes. The fluid out of the pump is separated into two paths, one passes through the heat exchanger, the other passes through the steam-water separator, both of the two flow rates can be regulated by adjusting the valves before the heat exchanger and the separator, respectively. The two paths of fluid converge at the air-cooler, and are cooled in it, then return to the pump.

The two loops are linked at the heat exchanger and the condenser.

2.2 Realization of simulating nuclear feedback

In one word, simulating nuclear feedback has been realized by software and hardware. The whole process consists of 5 steps. The first step is to calculate the density change of coolant in the heating section, $\Delta\rho$, which simulates that of nuclear reactor core, according to the differential pressure of this section, $\Delta P_{\text{pressure}}$, which is measured by a differential manometer. Secondly, $\Delta\rho$ is multiplied by density-reactivity coefficient which is determined by the nuclear reactor, the product is the change in nuclear reactivity, named as $\Delta\eta$. Next, through solving point-reactor kinetics equations, the change in neutron flux, ΔN, can be got. Then the variation of nuclear power, $\Delta P_{\text{nuclear}}$, can be got from the product of ΔN and the power coefficient determined by the nuclear reactor.

The four steps above can be realized by software. The last step, from $\Delta P_{\text{nuclear}}$ to $\Delta P_{\text{thermal}}$, can be performed by a silicon controlled rectifier (SCR). Considering $\Delta P_{\text{nuclear}}$ obtained in step 4, then the control voltage of SCR, V_{ctrl}, can be got depending on its power/voltage performance curve. Then, the power of the heating assembly is controlled by the signal of V_{ctrl}.

SCR is a critical hardware for the realization of simulating nuclear feedback. Moreover, in order to simulate nuclear feedback, the SCR and the heating assembly must have excellent controllability, transient stability and time resolution characteristic.

2.3 Data acquisition and processing system

As shown in Fig.4, the system consists of

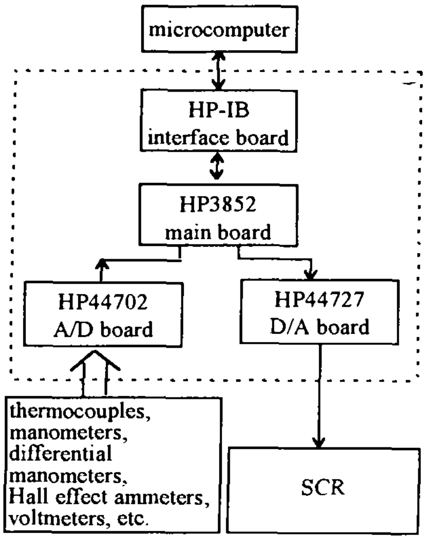


Fig.4 Data acquisition system

a microcomputer, a data acquisition equipment and different kinds of sensors. All collected analog signals are transferred to HP44702 high-speed digital voltmeter, and are converted to digital signals, which are inputted into the microcomputer through HP3852 main board and

HP-IB interface board. Thermal-hydraulic calculation is completed in the microcomputer, and the control voltage of SCR, V_{ctrl}, is figured out according to the principle of nuclear feedback described in section 2.2. Finally, the digital signal of V_{ctrl} is converted to analog signal in HP44727 D/A board and transmitted to SCR to control the heating power.

The main measuring points are shown in Fig.5. The corresponding codes and sensor types are listed in Table 1. Temperatures of fluid are measured by copper-constantan thermocouples. In order to augment signal-to-noise ratios, "NTT-T" temperature transformers are adopted. The temperature of the heating rod surface, which is relatively high, is measured by an alumel-chromel thermocouple. Absolute errors of temperatures are 1°C. Pressures and differential pressures are measured by "1151" manometers and "1151" differential manometers, respectively. Their relative errors are 0.5%, and their time-constants are 0.2 s. The heating current and voltage are measured by LT-2000-S Hall effect ammeter and LV-25 P voltmeter, respectively, of which relative errors are 0.5%.

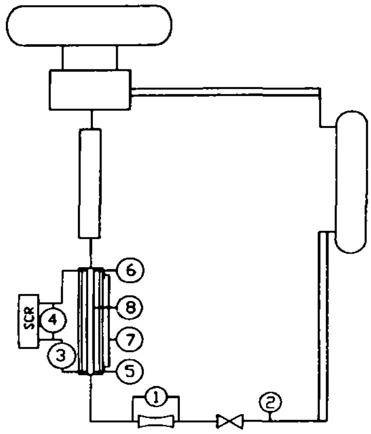


Fig.5 Location of main measuring points
1 Circulation flow rate, 2 Primary loop pressure, 3 Heating current, 4 Heating voltage, 5 Temperature of heating section inlet, 6 Temperature of heating section outlet, 7 Differential pressure of heating section, 8 Temperature of heating rod surface

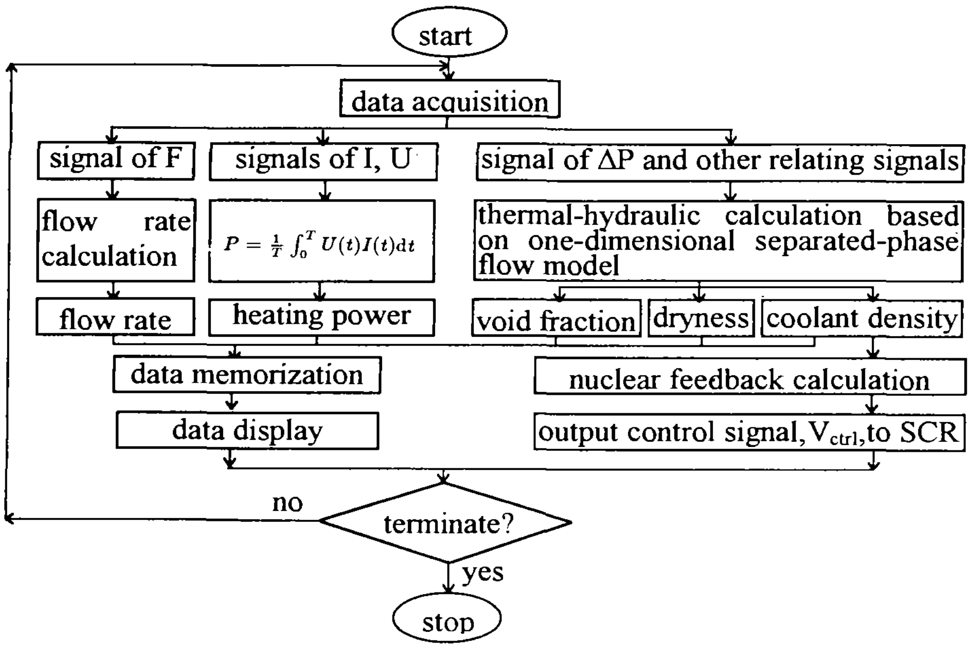


Fig.6 The schematic structure of the program

The software system, which is developed with the computer languages of Borland C++ and Assemble, is applied to completing data acquisition, calculating, memorization, display and control. Its schematic process is shown in Fig.6.

2.4 Operation conditions of the primary loop

In the beginning of experiments, the power of the heating assembly, which simulates nuclear power, is augmented to the rated level of 60 kW gradually, and natural circulation is erected in the primary loop. Then the required operation conditions of the primary loop can be obtained by adjusting the valves A and B (shown in Fig.3). Specifically, the required system pressure, 1.5 MPa, can be obtained by regulating the condenser feed-water through adjusting the valve B, the required subcooling of the heating section inlet, 5°C or 30°C, can be obtained by regulating the heat exchanger feed-water through adjusting the valve A. While the system is running stably under the required operation condition, simulating nuclear feedback can be thrown into the system. At the same time, the system responses are recorded by the data acquisition system.

Table 1 Measuring points, codes and sensor types

Measuring points	Codes	Sensor type
Circulation flow rate	F	Differential manometer
Primary loop pressure	P	Manometer
Heating current	I	Hall effect ammeter
Heating voltage	U	Voltmeter
Temperature of heating section inlet	T_{in}	Copper-constantan thermocouple
Temperature of heating section outlet	T_{out}	Copper-constantan thermocouple
Temperature of heating rod surface	T_w	Alumel-chromel thermocouple
Differential pressure of heating section	ΔP	Differential manometer

3 Experimental results and discussions

3.1 High subcooling of the heating section inlet

When the subcooling of the heating section inlet is about 30°C, the circulation flow rate, which is a typical parameter being indicative of whether the system is stable or not, declines slightly after nuclear feedback being thrown into the system, as shown in Fig.7. Although there are some slight fluctuations on the curve, the circulation flow rate drifts to another stable flow rate which is somewhat less than that before nuclear feedback. The variation of nuclear reactivity and the mean void fraction of heating section are shown in Figs.8 and 9, respectively. Although both the two curves have some vibrations, the relative amplitudes are within 10 percent. Therefore, the natural circulation is stable on the whole.

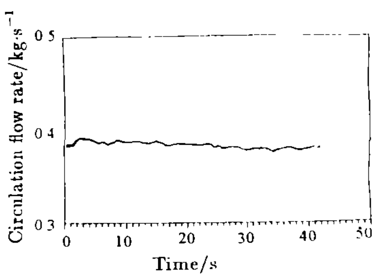


Fig.7 Circulation flow rate after nuclear feedback when subcooling of heating section inlet is 30°C

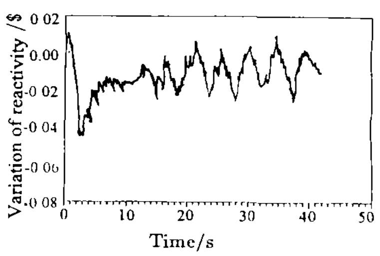


Fig.8 Variation of reactivity after nuclear feedback when subcooling of heating section inlet is 30°

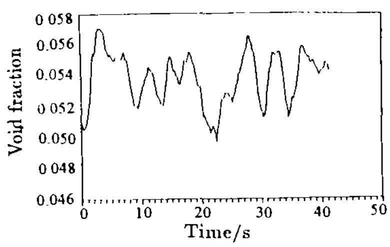
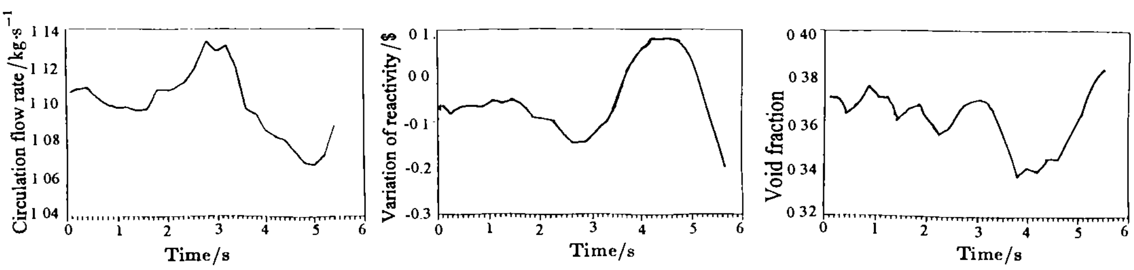


Fig.9 Void fraction of heating section after nuclear feedback when subcooling of heating section inlet is 30°C

Based on nuclear reactor physics, voids can degrade the slowing-down power of neutron moderator which also is coolant in nuclear heating reactor. Thus, the variation of reactivity, which is led into the system in the beginning of nuclear feedback, is negative, as shown in Fig.8. The negative change of reactivity means a decrement of the nuclear power which can result in decrements of void fraction and driving pressure head then a decrement of circulation flow rate. This is the reason why the circulation flow rate declines. Then the decrement of flow rate leads to an increment of void fraction. Therefore, the thermal-hydraulic feedback is negative, which is the basic reason why the natural circulation is stable. Certainly, the decrement of void fraction in the beginning of nuclear feedback can also diminish two-phase friction factor. However, this effect is negligible because the length of two-phase flow zone in the up-section is relatively short owing to the relatively high subcooling of the heating section inlet.

3.2 Low subcooling of the heating section inlet

When the subcooling of the heating section inlet is about 5°C, the circulation flow rate, which is shown in Fig.10, oscillates violently. Referring to Figs.11 and 12, both the variation of nuclear reactivity and the mean void fraction of the heating section also fluctuate violently. It is obvious that natural circulation is instable.



Figs.10-12 After nuclear feedback circulation flow rate(10), reactivity (11), and void fraction of heating section (12) as a function of time when subcooling of heating section inlet is 5°C

In the beginning of nuclear feedback, being similar to the situation stated in section 3.1, a negative reactivity is led into the system. Then decrements of nuclear power and void fraction occur. The decrement of void fraction can bring about two effects simultaneously. One is a decrement of driving pressure head, the other is a decrement of two-phase friction factor. However, the primary effect is not the former but the latter because the subcooling of the heating section inlet is relatively low and the length of two-phase flow zone in the up-section is relatively long. So the total effect is an increment of circulation flow rate then another decrement of void fraction. Therefore, the thermal-hydraulic feedback is positive, which is the fundamental reason why the system is instable.

4 Conclusions

On the basis of the simulating experiments, the subcooling of core inlet should be kept high enough in order to ensure the stable operation of low temperature nuclear heating reactor.

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