XA0201520

CIRCUS and DESIRE: experimental facilities for research on natural-circulation-cooled boiling water reactors

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Abstract. At the Delft University of Technology two thermohydraulic test facilities are being used to study the characteristics of Boiling Water Reactors (BWRs) with natural circulation core cooling. The focus of the research is on the stability characteristics of the system. DESIRE is a test facility with freon-12 as scaling fluid in which one fuel bundle of a natural-circulation BWR is simulated. The neutronic feedback can be simulated artificially. DESIRE is used to study the stability of the system at nominal and beyond nominal conditions. CIRCUS is a full-height facility with water, consisting of four parallel fuel channels and four parallel bypass channels with a common riser or with parallel riser sections. It is used to study the start-up characteristics of a natural-circulation BWR at low pressures and low power. In this paper a description of both facilities is given and the research items are presented.

1. INTRODUCTION

Natural circulation is a key item in the design of innovative natural-circulation-cooled Boiling Water Reactors (BWRs). Instead of using recirculation pumps to provide the cooling flow for the core, the core flow is driven by the density differences between the two-phase mixture in the core and the essentially single-phase flow in the downcomer. The natural-circulation core flow is enhanced by using a riser section on top of the core. Because the core flow cannot be controlled by means of a pump, the recirculation core flow is an internal variable of the system. Natural circulation has been used in the early stages of reactor development. Both the Experimental Boiling Water Reactor (EBWR) [1] and the Vallecitos Boiling Water Reactor [2] could be used with natural circulation. Later, the Humboldt Bay atomic unit and the Dodewaard plant have been operated as commercial BWR/1 plants with natural circulation. More recently the interest in natural circulation as a possibility for the core cooling has been renewed. This can be seen in the design of the Simplified Boiling Water Reactor, based on which new designs such as the ESBWR have been proposed [3]. The trend in these designs with respect to the reactor core is towards larger cores and higher power, combined with larger risers to enhance the natural-circulation core flow.

Because the core flow responds to changes in power the stability of a natural-circulation BWR is somewhat different from the stability of a forced-circulation BWR. Therefore, the stability of a natural-circulation BWR requires special attention. It has been shown that two different instability types exist for such a reactor, denoted by type-I and type-II [4]. Type-I oscillations are typical for natural-circulation BWRs and are driven by the gravitational pressure drop over the core and riser. Type-II oscillations are driven by the interplay between single-phase and two-phase friction in the core. This division in different types is not sharp. The transition from one type to the other occurs gradually. Although the character of both types of oscillations is different one could describe both of them as density-wave oscillations.

The type-I oscillations may occur during the start-up phase of the reactor, because this unstable region broadens as the pressure decreases and because it is associated with low-power operating conditions. The flashing of water in the riser at low pressures induces this type of oscillation. The neutronic feedback is not important for this type of instability. The core region is essentially single-phase and thus the power oscillations will be small. Moreover, at such a low level power oscillations cannot cause any damage to the fuel. However, large flow oscillations should be avoided in view of their possible effect on structural materials. Different types of oscillations could be possible in view of the different possible configurations of parallel fuel bundles with common or parallel riser sections.

Type-II oscillations may occur during high-power/low-flow conditions in both natural-circulation or forced-circulation BWRs. Because the density in the core region fluctuates, the nuclear feedback is essential in the analysis of these types of oscillations. A division is made between core-wide, regional, and local oscillations. In core-wide oscillations the total power in the core will vary and all fuel bundles oscillate in-phase. This mode is favoured neutronically because of the subcriticality of the regional modes. In regional oscillations the total power will remain nearly constant and the power distribution will vary periodically. This mode is favoured thermohydraulically because the total flow will be nearly constant. A third type of oscillation is a local thermohydraulically unstable fuel bundle which induces small changes in the power and the power distribution.

The pressure dynamics and the feedwater (inlet subcooling) dynamics of the system should also be taken into account; this might especially be important for type-I oscillations in natural-circulation BWRS for which water flashes into large volumes of steam giving rise to large flow oscillations.

At the Delft University of Technology two thermohydraulic test facilities are being used to study the instability types in natural-circulation BWRs: DESIRE for type-II oscillations and CIRCUS for type-I oscillations. A description of DESIRE is given in Section 2, a description of CIRCUS is given in Section 3. Both facilities can be used to produce valuable experimental data needed for further model development in the system codes applied for nuclear power plant analyses[5].

2. DESIRE

DESIRE is a simulated fuel bundle of a natural-circulation BWR with freon-12 as scaling fluid. Figure 1 shows an overview of the primary loop of the current facility, which is a scaled model of the Dodewaard natural circulation reactor[6]. The fuel bundle consists of 35 fuel rods in a 6 × 6 array. The fuel rods are 958 mm long with an heated length of 880 mm. The diameter is 6.35 mm. The fuel rods have either a chopped cosine or a flat uniform axial profile. Six independent power supplies can be connected arbitrarily with individual fuel rods, facilitating a wide range of radial power distributions. The nominal power is 22.3 kW, but the maximum power is about 50 kW. The pressure ranges from 8 to 13 bar (nominal pressure is 11.6 bar). In this manner, using freon-12 as a coolant, a full 1.8 m BWR bundle is simulated operating at 75 bar and 1116 kW nominal conditions. The inlet friction of the fuel bundle can be varied by means of a valve. The riser section can be varied in length (1.1 m–1.9 m) by means of a telescopic riser section. At the top of the riser a free surface exists at which the vapour is separated from the liquid which returns through the downcomer. Part of the vapour is also drawn into the downcomer, the so-called carry-under. The temperature distribution in the loop is measured with chromel-alumel thermocouples. The absolute pressure is measured at different positions in the loop, and the pressure difference is measured over the upper part of the downcomer tube and part of the steam dome, which gives an indication of the collapsed liquid level above the riser exit. The total recirculation flow as well as the steam flow and the feedwater flow is measured with vortex flow meters. Gamma transmission is used to determine the average void fraction at a given height in the core region. This void fraction is also being used to implement an artificial feedback on the power of the fuel rods. The fuel rod thermal time constant as well as the void reactivity coefficient can be varied artificially. Gamma transmission can also be used to reconstruct the void-fraction on a subchannel basis, using tomographic techniques. In order to study the stability characteristics of the system, noise-analysis techniques are being used.

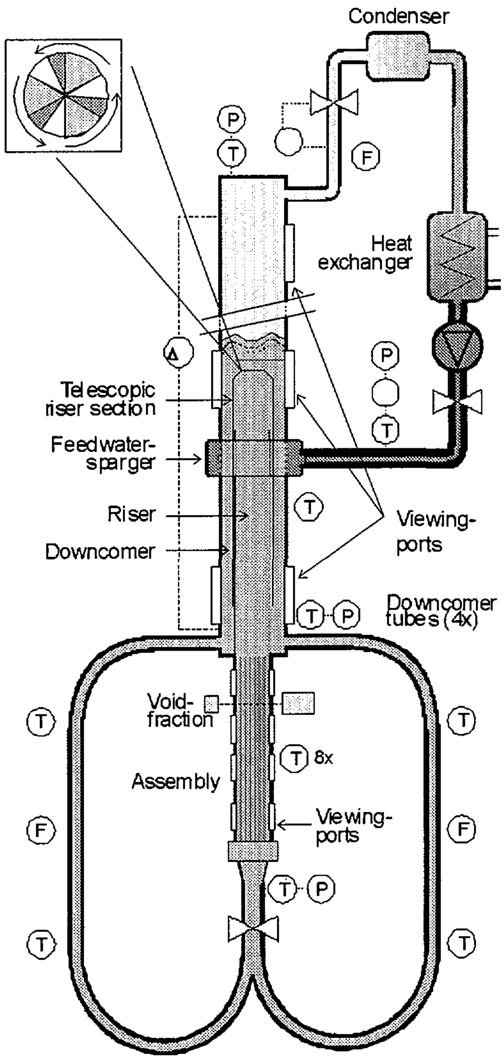

FIG. 1. Overview of the DESIRE facility. The position of the instrumentation is shown as well (T=Temperature, P=Pressure, ΔP=Pressure Difference and F=Flow).

DESIRE has been extensively used to study the natural circulation and stability characteristics of the Dodewaard natural circulation reactor[7,8]. Recently a riser exit restriction has been implemented to be able to destabilize the system in order to study limit cycles and non-linear effects with the facility[9]. With this riser exit restriction the effect of steam separator friction can be studied. In the near future the intention is to use DESIRE in the framework of the European NACUSP project. A large set of thermohydraulic experiments without artificial feedback will be performed covering a wide operational range of the facility (power, feedwater temperature, pressure, riser liquid level, power distribution) for a large range of geometrical settings (inlet friction, outlet friction, riser length). For this purpose the instrumentation of the loop is currently being updated and extended. A dedicated set of experiments will be performed to study the capability of 3D-codes to calculate the void-distribution in the fuel bundle. The focus of these experiments is at low void fractions. Finally, part of the experiments will be devoted to optimize the artificial nuclear feedback currently being used in the facility. This optimization is needed to be able to extend the facility to multiple "parallel bundles" with which coupled neutronic/thermohydraulic regional oscillations can be simulated.

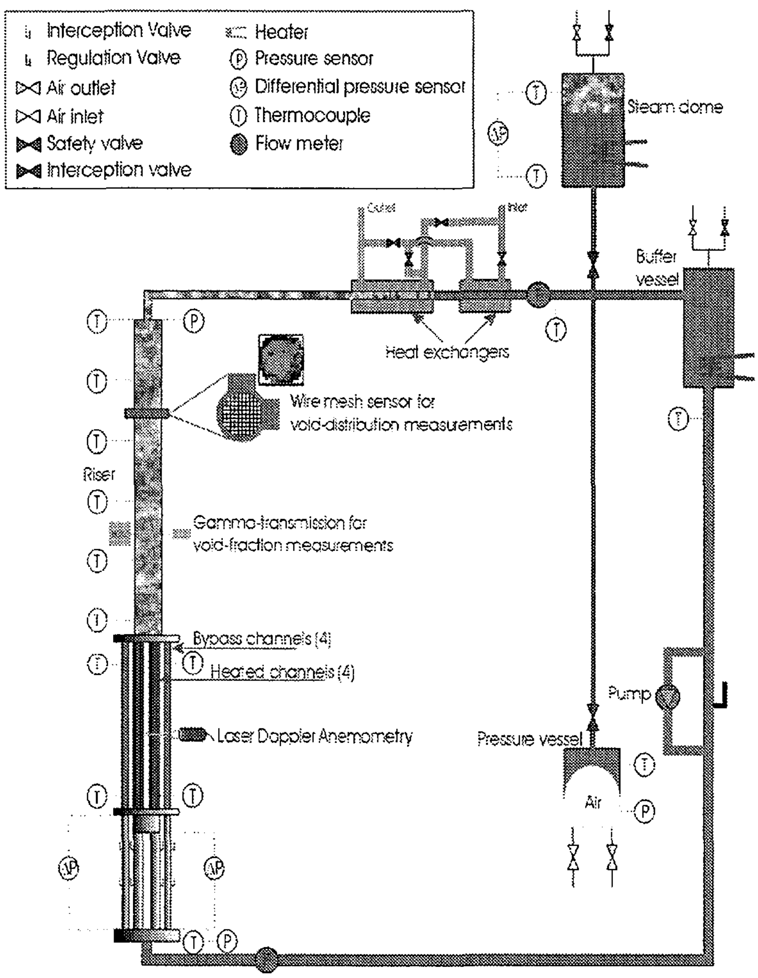

FIG. 2. Scheme of the CIRCUS facility (out of scale). The main instrumentation is also shown. (see legend).

3. CIRCUS

The CIRCUS-facility is a full-height scaled steam/water loop of the Dodewaard reactor. Figure 2 shows a schematic view of the current facility.The reactor core is simulated by 4 electrically heated fuel rods in coolant channels and by 4 separate bypass channels. The channels are made of glass in order to be able to visualize the flow. This is one of the important features of the CIRCUS facility, which enables the researcher to combine the measurements with observations of the flow. The friction of each individual coolant channel or bypass channel can be varied independently. On the top of the core section a long adiabatic glass tube is used to simulate the riser section. The two-phase mixture is condensed and cooled by means of a heat exchanger. The length and the secondary flow rate of the heat exchanger can be varied. This controls in combination with the heater power in the buffer vessel the subcooling of the liquid at the core inlet. The system pressure is regulated with a steam vessel, representing the steam dome in a BWR. A pressure vessel is used to pressurize the system. When the level in the steam dome drops below a certain level, the pressure vessel can be disconnected. The pressure vessel can also be used to perform alternative measurements with pressure feedback of the pressure vessel itself instead of pressure feedback of the steam volume in the steam dome. The position of the steam dome will be changed to a place before the heat exchanger to avoid problems with subcooled liquid entering the steam dome. Table 1 shows the main characteristics of the facility.

The temperature distribution in the loop is measured with chromel-alumel thermocouples and two Pt-100 temperature sensors for reference measurements. Absolute pressure is measured at the top of the riser and at the inlet of the core. The liquid level in the steam dome is measured with a differential pressure sensor. The differential pressure over the friction settings of the individual channels is a measure for the flow distribution over the coolant channels and bypass channels. The total flow in the loop is measured at 2 different positions with electromagnetic flow meters. The void fraction at a given height can be measured with gamma transmission techniques. At a fixed height at the top of the riser the radial void distribution is measured with a wire-mesh sensor, which measures the conduction of the two-phase mixture on a two-dimensional grid. Furthermore, laser doppler anemometry is used to study the local liquid velocity in the core or in the riser.

A limited set of experiments has been performed in which large flashing-induced oscillations have been observed[10]. As well as DESIRE, CIRCUS will also be used in the near future in the framework of the NACUSP project. A large set of thermohydraulic experiments will be performed covering a wide operational range of the facility (power, inlet subcooling, pressure, steam dome level, power distribution) for a range of geometrical settings (inlet friction, friction distribution). A dedicated set of experiments will be performed in order to study the flashing effect in the riser in detail. The gamma transmission technique and the wire-mesh sensor are of course important tools for this study. A third extensive set of experiments will be performed with two parallel riser sections instead of one common riser as in the current configuration.

TABLE. I. MAIN CHARACTERISTICS OF THE CIRCUS FACILITY

Power range per rod	0-3 kW
Pressure range	1-5 bar
Fuel channel diameter	20.4 mm
Fuel rod diameter	12.5 mm
Bypass channel diameter	10 mm
Fuel channel length	1.95 m
Riser diameter	47 mm
Riser length	up to 3 m

4. CONCLUSIONS

An overview of the facilities at the Delft University of Technology to study the natural circulation and stability characteristics of natural-circulation cooled BWRs has been given. The results of these studies are not only useful for these types of reactors but also for forced-circulation BWRs, because of the study on type-II instabilities. The results are also useful for LWR-reactors in general, because of the experimental data that will be generated at low power and low pressure. The experiments in the near future will be performed in the framework of the European NACUSP project in which the natural circulation characteristics and stability performance of natural-circulation cooled BWRs are studied.

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