EXPERIMENTAL STUDY ON THE STABILITY OF A NATURAL CIRCULATION DRIVEN SUPER-CRITICAL WATER COOLED REACTOR

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Abstract
The supercritical water reactor (SCWR) is one of the advanced nuclear reactor designs currently under investigation. Considering the large density difference which the coolant experiences when travelling through the core, using natural circulation as the driving force seems feasible. This avoids the use of pumps and therefore adds an inherent safety feature. The same large density difference however, is also expected to induce flow instabilities, as is known from natural circulation boiling water reactors. These instabilities are due to inherent feedback mechanisms, either pure thermo-hydraulic or neutronic – thermo-hydraulic. In this study, the stability of a natural circulation SCWR is studied experimentally. The setup (DeLight) is a scaled version using Freon R23. The thermo-hydraulic feedback naturally occurs within a closed loop, but to mimic a nuclear reactor, neutronic feedback and the fuel rod behaviour is artificially implemented. This was done based on local density measurements and with a first order time delay model. It was found that there are no pure thermal-hydraulic instabilities, but applying the neutronic feedback does result in significant instabilities for certain operating conditions. A parameter study was performed by examining the impact of the power distribution and the fuel time constant on the nature of the instabilities.

Keywords: natural circulation, supercritical water reactor, stability

INTRODUCTION

The SuperCritical Water Reactor (SCWR) is one of the six concept designs that is being studied as part of the international generation IV effort to develop more efficient, safer and proliferation resistant nuclear reactors. By using supercritical water, the exit temperature can be raised significantly, most designs consider an exit temperature of 500°C and a pressure of 25 MPa. This results in a higher efficiency (up to 42-45%), which is well above that of current nuclear reactors. Using supercritical water also results in a reduced complexity of the auxiliary systems and plant components, cutting investment costs, as reported by Buongiorno and MacDonald (2003). Over the course of the past decades a number of core designs have finalized, including recently a European design (Fischer et al., 2009). These designs differ considerably with regards to the fuel assembly, flow layout or neutron spectra which are used. The European design (HPLWR, High Performance Light Water Reactor) is remarkable in having a three-pass core layout (Fig. 1A) combined with water rods for moderation. The system operates at 25 MPa, with an inlet and exit temperature of 280 °C and 500 °C. Between the passes mixing plena are used to reduce peak cladding temperatures by homogenizing the flow into the passes. More details on the HPLWR fuel assembly can be found in Hofmeister et al. (2007).

As is well known, supercritical fluids experience strong changes in fluid properties. This is illustrated in Fig 1B. The density varies between 780 kg/m³ and 90 kg/m³ with a sharp change near the pseudo critical point. This large density difference could be used as the driving force for natural circulation. This removes the need for large feed water pumps, and adds a layer of inherent safety to the system. Currently most reactor designs only consider natural circulation for emergency situations, despite its inherent safety aspect. So far only one design, the ESBWR (as studied by e.g. Marcel et al., 2008), has actually been constructed in a small size at Dodewaard, the Netherlands and was operated for decades. However, a number of reactors are currently under development with single phase natural circulation: e.g. the REX-10 (Regional Energy Reactor – 10 MW_{th}) which uses water at 2 MPa (Jang et al., 2011). GE also continues to research the ESBWR design for future application.

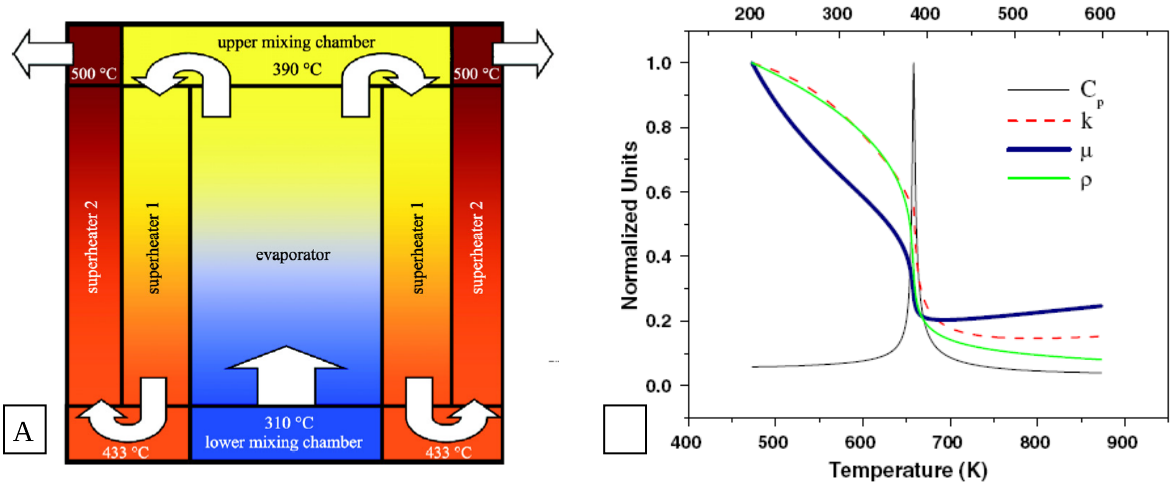


Figure 1: A: Three pass core arrangement proposed for the HPLWR (Fischer et al., 2009), B: normalized fluid properties for water at 25 MPa for a range of temperatures

Natural circulation loops however can become unstable under specific operational conditions (e.g. high power and low flow rate). Bouré et al. (1973) presented a classification of the different types of instabilities. A static instability (flow excursion) can be described using only the steady state equations. In this case, a small change in the flow conditions will result in a new steady state not in the vicinity of the original one. For dynamic instabilities, such as density wave oscillations or DWO, the steady state equations are not sufficient to predict the system behavior, not even the threshold of instability. In such a situation, multiple competing solutions exist for the governing equations. The system will move from one solution to the other, driven by feedback mechanisms. March-Leuba and Rey (1993) presented a detailed explanation of the DWO and the feedback mechanisms, which is driven by the interaction of inertia and friction for the thermo-hydraulic modes. In a nuclear reactor another feedback mechanism is present: the neutronic feedback. This couples the fluid density to the power production through the moderation done by the water molecules. This makes the fission rate and subsequently the heat generated in a reactor core directly correlated to the neutron density. This results in a much more complex behavior, as shown by Van Bragt and Van der Hagen (1998) for the ESBWR and recently by Yi et al. (2004) for the US design of a SCWR.

Experimental data on natural circulation supercritical loops is rare in open literature. Most published results on the stability of supercritical flows are numerical and they consider either a forced single pas system (e.g. Ambrosini and Sharabi, 2008) or an idealized loop geometry (e.g. Jain and Uddin, 2008 and Sharma et al., 2010). Lomperski et al. (2004) performed an experimental study on a rectangular supercritical CO_2 loop . They reported steady state data and were unable to find any instabilities within the considered range. These findings did not agree with the accompanying numerical work by Jain (2005), who did find a stability boundary at much lower power.

The goal of this study is to examine the stability boundary of a naturally circulating HPLWR experimentally. To this end a scaled setup has been designed and built. Neutronic feedback has been implemented artificially, and the time delay between the power production and the wall heat flux has been modeled using a single fuel time constant. In the subsequent paragraphs first the measurement setup and procedure will be described before presenting the results. Because of the scarcity of experimental data on the stability of a supercritical loop in open literature, this data could serve as an important benchmark tool for existing codes.

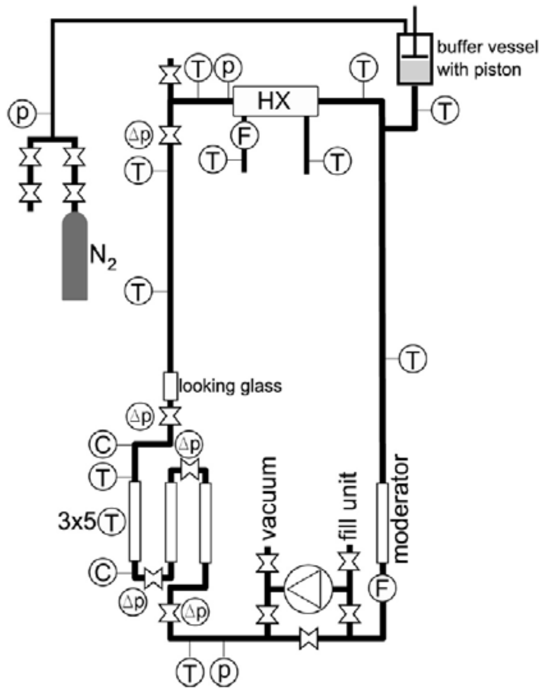
EXPERIMENTAL SETUP

In order to reduce the pressure and temperature level and the power requirements imposed by the supercritical water to more suitable lab values, a scaling fluid was used. To design a scaled version of the HPLWR, the governing 1-D equations (conservation of mass, momentum and energy and the equation of state) of the system should first be considered and made non-dimensional. This is done by selecting a reference state (the pseudo-critical point). Rohde et al. (2011) describe the scaling procedure based on the conservation of the Froude number and the friction distribution. It was shown that the friction distribution, rather than the actual value determines the linear stability behaviour. After comparison of a large number of different fluids, Freon R23 (CHF_3) was selected as the scaling fluid based on the power requirement, the temperature (the pseudo-critical temperature is only 33°C), the pressure (5.7 MPa) and safety (non flammable). The non-dimensional fluid properties agree well, with a maximum deviation of 8% for the density far away from the pseudo-critical point. Some relevant pseudo-critical fluid properties and scaling values are indicated in Table 1. Through linear stability analysis of a channel with supercritical water and of its scaled R23 counterpart, it was shown that the scaling rules result in the same stability behaviour, confirming the proposed scaling procedure and fluid selection (see Rohde et al., 2011).

Table 1: Comparison of selected pseudocritical properties of H_2O and R23 and the considered scaling rules, Rohde et al. (2011)

	R23	H_2O	Scaling factor	
Pressure (MPa)	5.7	25	Length	0.191
Temperature (°C)	33.2	384.9	Diameter	1.06
Density (kg/m³)	537	316.82	Power	0.0788
Enthalpy (kJ/kg)	288.3	2152.9	Mass flux	0.74
Core inlet temperature (°C)	-21	280	Time	0.438
Core exit temperature (°C)	105	500		

Based on the derived scaling rules, an experimental facility has been constructed, named 'DeLight' (Delft Light water reactor facility). A schematic drawing is shown in Fig. 2 and some of the dimensions are listed. The loop is constructed using stainless steel tubing (6mm ID for the core sections, 10 mm ID for the riser and downcomer). The total height of the loop is 10 m. Up to 18 kW of heating (twice the scaled power requirement) can be added in 4 heating sections (3 cores and the moderator channel which mimics the water rod presence). Heating is done electrically (providing a uniform heat flux boundary) by sending a current through the core tubes (up to 600A per core element using Delta SM15-200 power units). The power rating of each heating section can be controlled individually, as the power distribution in the HPLWR core is non uniform, with the first heating section accounting for 53% of the total power, the second for 30% and the last for 17%. Each heating section is electrically insulated from the rest of the setup using a PEEK ring mounted in between 2 flanges. To provide a stable pressure level, a buffer vessel is present at the top of the loop which has a moveable piston (Parker Series 5000 Piston Accumulator) connected to a nitrogen gas cylinder. By positioning this piston higher or lower the pressure level in the loop can be set at 5.7 MPa. Two heat exchangers (HX in Fig. 2) are mounted in series at the top section of the loop to extract the heating power and to set the inlet conditions. Due to the differential thermal expansion of the heating sections, the tubes are connected to the wall using moveable spacers which contain 2 prestressed springs.



Heating section length: 0.8 m
Moderator section length: 1.4 m
Riser length: 7.53 m
Downcomer length: 10.26 m

HX1: SWEP© B16DW U
Secondary side: water
secondary side flow rate: 0.49 l/s
secondary side Tin: ±18°C

HX2: Vahterus Oy© Q-Plate
Secondary side: R507a
secondary side flow rate: variable
secondary side Tin: variable (-40 to -5°C)

Figure 2: Schematic of the DeLight facility with dimensions and equipment specifications.

The loop contains a large number of sensors to closely monitor the thermohydraulics. At the top and bottom absolute pressure sensors are presents (p symbol in Fig. 2, ± 0.15%). The different heating sections each contain 5 type K thermocouples to measure the local fluid temperature (T symbol in Fig. 2, ± 0.1K). These thermocouples also have to be insulated electrically from the core to prevent the feed current passing through them. This was done using PEEK rings. The individual thermocouple channels were calibrated carefully using 3 reference thermocouples which were calibrated over the entire temperature range by a certified body. As shown in Fig. 2 additional thermocouples are placed in the riser and downcomer section, as well as on the secondary side of the heat exchangers to monitor the heat removal. The R23 mass flow rate is measured using a coriolis mass flow meter (F symbol in Fig. 2, ± 0.25%). Apart from the core sections the entire setup is insulated using Armacell© (25 mm thick) to reduce any heat loss to the exterior. A magnetic rotor pump is present in the loop, but a bypass can be set to allow for natural circulation, as shown in Fig. 2.

The data acquisition system consists of a PC with one National Instruments PCI-6259 data acquisition card, connected to a National Instruments SCXI-1001 rack with two SCXI-1102B 32-channel amplifiers. This system is used for monitoring the experimental setup and for recording sensor signals. The measured and processed data are displayed on a PC screen which allows for continuous monitoring. Additionally, seven signals (three temperature values, two pressure values, and the R23 and cooling water flow rates) are connected to a separate stand-alone data acquisition system. This system is used for safety monitoring and will shut down the power supplies if one of the signals exceeds prescribed limits. A third PC is used to control the setup, setting the pressure level and the power input.

In order to mimic the behaviour of a nuclear reactor, neutronic feedback has to be included within the setup. This feedback mimics the moderating effect of the water, whereby an increase of the density (= lower temperature ~ drop in power) raises the moderating effect and thus increases the probability of fission, which then again raises the power level to the original level. This is described through the 'reactivity' $\Re$ of the system. The artificial feedback is implemented by measuring the average core density ρ with the help of the 15 installed thermocouples and the equation of state for the density (similar to that used by Marcel et al., 2008). To allow for a fast computation, the equation of state was implemented as a series of splines $\rho = f(T)$. The node points for the splines (NIST database v.8) were carefully selected and spaced more closely near to the pseudocritical point. Once the neutronic feedback is engaged, the measured density variations are used to calculate the reactivity via Eq. (1). The change in power due to the reactivity feedback is then calculated with the help of a linearized six-group, point-kinetic model (Duderstadt and Hamilton, 1976). This model for the neutronic feedback was also used by Yi et al. (2004). The precursor decay constants and fractions can be found in Table 2 and the delayed neutron faction β was set to 0.0056. These values were obtained from Ortega-Gomez (2009), but because of the time scaling in the facility, the decay constants need to be modified using the time scaling. The mean generation time was set to 22 µs, which is the scaled equivalent of a BWR, taken from Marcel et al. (2008). The density coefficient $r_\rho$of the reactivity $\Re$, is defined based on the work of Schlagenhaufer et al. (2007) who studied the density feedback in the HPLWR. This value also needed to be scaled.

$$\Re = r_\rho(\rho(t) - \bar{\rho}) \tag{1}$$

Table 2: Delayed neutron fractions and decay time constants used, Ortega-Gomez (2009)

fractions		Decay constants	
β_1	0.038	λ_1	0.0290
β_2	0.213	λ_2	0.0724
β_3	0.188	λ_3	0.263
β_4	0.407	λ_4	0.710
β_5	0.128	λ_5	3.20
β_6	0.026	λ_6	8.84

Using the point-kinetics equations, the instantaneous change of fission inside the fuel pellets is determined based on the density pertubation. In reality however, there is a time delay between the release of the fission energy and the moment when the coolant is exposed to a change in heat flux. This is because of the finite pellet size and its thermal conductivity. This time delay was modeled as a first order process with a time constant of 6 seconds. This value is based on earlier experimental work on the ESBWR fuel rods (Van der Hagen, 1988). Because this value is related to the actual dimensions of the fuel rod and its inner gap, it can vary significantly between the different reactor designs which are currently being studied. It can be as low as 2 seconds. Therefore this value will be varied in this study, to determine its effect on the stability.

EXPERIMENTAL PROCEDURE

To experimentally determine the stability behaviour, the following procedure was used. First, the pump was used to start the circulation in the loop and a small amount of heating was added (1.5 kW). The pump was then switched off and bypassed, resulting in a naturally circulation. The pressure was then raised above the critical pressure and the cooling setup was turned on. By simultaneously controlling the position of the piston and slowly incrementing the added heat, the system was brought to the required testing conditions (5.7 MPa, and a specified power input in the HPLWR distribution). To control the inlet temperature, the expansion pressure on the secondary side of HX2 was set to levels such as -38°C, -30°C, -20°C… To reach values in between or for finer control of the inlet temperature, the moderator section at the bottom of the downcomer (see Fig. 2) was used. By adding more or less heat to this section, the temperature could be controlled to within 0.2°C, and this was used to set intermediate inlet temperatures. To judge if the system was stable, a number of signals were monitored: three temperatures (core inlet and outlet and heat exchanger outlet): variation < 0.2 °C, and the absolute pressure variations in the loop (< 0.025 MPa).

Once a steady state situation has been reached, the measurement was started. First over a period of 2 minutes the average core density is recorded. Then the neutronic feedback would be switched on. If the system is unstable, the neutronic feedback results in a fluctuating power input with a growing amplitude. These signals would then be recorded until saturation is reached. The saturation value was set to 10% of the power input to prevent large pressure fluctuations in the loop. If the system was stable, and no large oscillations were present two minutes after switching on the feedback, a step increase in the power (250 or 500W) was done for 5 seconds. The decaying signal was then recorded until it was no longer distinguishable. Once the measurement was completed the power was raised by 250W and the next point was recorded. This was continued until the signal variations were no longer measurable. The power was then increased to the next power level to measure the next point. This was continued until the core exit temperature exceeded 110°C (safety limit). The instabilities could be seen in all the recorded signals but they were most apparent in the temperature signals. So it was decided to use the inlet temperature signal of the riser as input for the signal analysis. Two examples of this measured temperature are shown in Fig. 3.

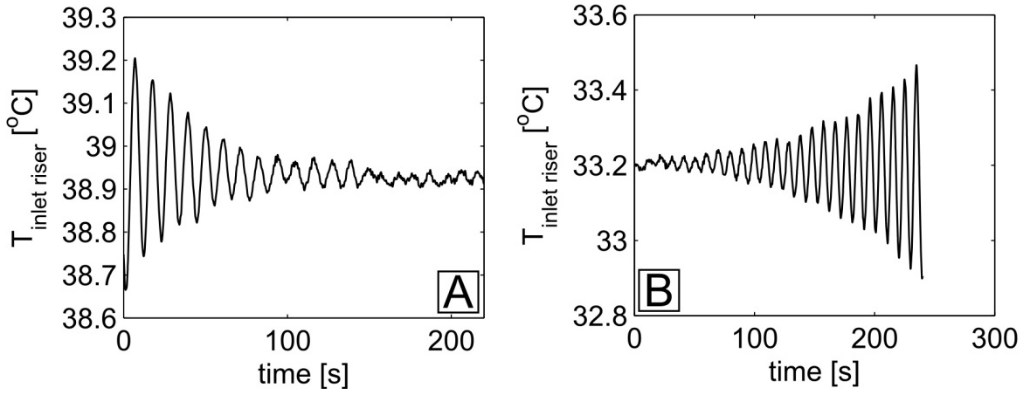


Figure 3: Examples of a measured stable (A) and unstable (B) temperature signal

These temperature signals were then processed using signal analysis tools. All the sensor signals are sampled with a frequency of 120 Hz and then resampled to 20 Hz. Before resampling, the signals are filtered with a cut-off frequency of 9Hz. This was done using a digital filter implemented in Matlab©. The resampling is done by taking a running average of 6 samples. These resampled data values were then used to determine the decay ratio 'DR'. This was done by fitting the equation $y = (1 - c_1 - a_1)e^{b_1 t} + c_1 + a_2 e^{b_2 t} \cos \omega t$ to the first two periods of the auto correlation function (ACF) of the signal. The DR is then defined by equation (2). These equations have been derived based on the work of Marcel (2007) for a natural circulation BWR. As an extra check for the resonance frequency, the auto power spectral density is also determined, verifying it contains a single well defined peak at the frequency f. In the following section the measured DR values will be presented.

$$DR = e^{\frac{2 \cdot \pi \cdot b_2}{|\omega|}} \tag{2}$$

RESULTS

To represent the results in a more general form, a set of non-dimensional numbers are required. Because previous studies have used different formulations of the 1D transport equations, different non-dimensional units were derived to show the results. These numbers are mostly inspired by the earlier work done on boiling systems, seeking to extend the concept of the subcooling number and the phase change number into the supercritical range, as can be read in Ortega Gómez (2009) and Ambrosini and Sharabi (2008). Rohde et al. (2011) proposed a scaling procedure to preserve the stability behavior of a supercritical loop and suggested the pseudo phase change number N_{PCH} and subcooling number N_{SUB} (Eqs. (3)-(4)), using the pseudo-critical enthalpy h_{pc} as reference (288.03 kJ/kg). N_{PCH} represents the ratio of the power input (heat flux q" multiplied by the tube length L and the wetted perimeter P_h) to the mass flow rate (mass flux G multiplied by the flow surface area A).

$$N_{PCH} = \frac{q'' \cdot P_h \cdot L}{G_{in} \cdot A \cdot h_{pc}} \tag{3}$$

$$N_{SUB} = \frac{\left(h_{pc} - h_{in}\right)}{h_{pc}} \tag{4}$$

Steady-state power to flow map

Figure 4 shows three measured power flow curves for various inlet temperatures. As can be seen, these series show the expected trend: as the power increases, first the flow rate increases as well. This is due to the increased density difference that drives the flow. Starting from a given power the flow rate begins to decrease again. This is due to the increasing friction within the loop, which increases sharply as the velocity goes up (decreasing density). Furthermore, increasing the inlet subcooling relative to the pseudo-critical point increases the flow rate, and shifts the maximum flow rate to a higher power. By increasing the subcooling at the inlet, the inlet density goes up, requiring thus a larger amount of heating in order to make it sufficiently low so that friction can balance the increased driving force. Note how the slope of the second half of the power flow curve becomes steeper at increased subcooling. These trends are consistent with the published numerical work on supercritical loops (Jain and Uddin, 2008 and Sharma et al., 2010).

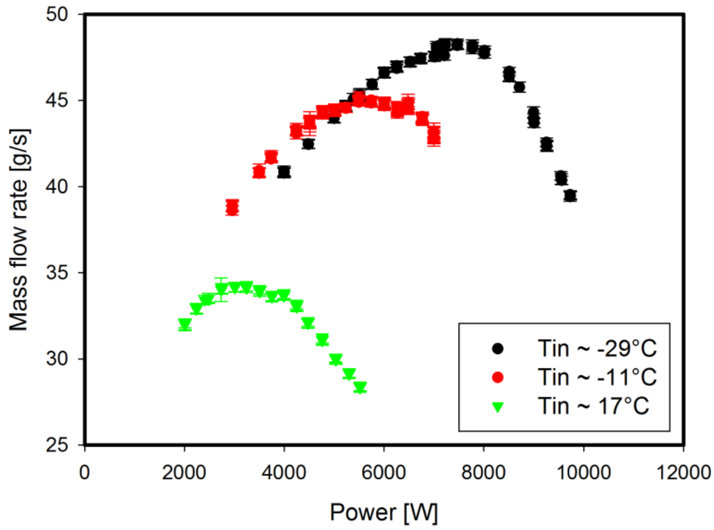


Figure 4: Examples of a measured power-flow curves for different inlet temperatures

Linear stability analysis: τ = 6s

DWO instabilities in reactors can be either purely thermo-hydraulic or coupled thermo-hydraulic – neutronic. During these experiments, no thermo-hydraulic instabilities (i.e. without neutronic feedback) were found in the considered power range. Switching on the neutronic feedback did result in the occurrence of instabilities. As such, all the shown data shown below is for the coupled neutronic – thermo-hydraulic instability. This is consistent with earlier work: Van Bragt and van der Hagen (1998) showed that for a natural circulation BWR the coupled neutronic – thermo-hydraulic mode is less stable than the pure thermo-hydraulic mode, shifting the stability 'peak' to the left in the stability plane. They also showed that for the coupled neutronic – thermo-hydraulic mode, the 'peak' in the stability region becomes much more narrow and extends up to higher N_{SUB}. Similar findings were reported by Yi et al. (2004) for a forced SCWR. A possible cause for the suppression of the thermo-hydraulic instabilities is the large interior volume of the heat exchangers (7.5 10^{-3} m^3) compared to that of the loop (roughly 2 10^{-3} m^3). They can thus act as strong dampers to perturbations. It is also known from literature that

About 1000 data points were measured with an inlet temperature varying between -29.7°C and 19.3 °C. The power ranged between 2 and 9.3 kW. The resulting contour plot of the DR values can be seen in Fig. 5 for τ = 6s. The black line indicates DR = 1, the neutral stability line. As can be seen, for a given inlet temperature (N_{SUB} = constant) the system undergoes two transitions. At low power, the system is stable (DR < 1), and raising the power results in increasing DR values until the system eventually crosses the stability boundary and becomes unstable. Continuing to increase the power makes the DR decrease again until the system is stable again. As such the DR shows a maximum. Jain and Uddin (2008) did not report this behavior, which could be because they only searched for the first transition by incrementing the power from a low value. This maximum behaviour was also found by Sharma et al. (2010) who studied a rectangular supercritical loop with water at 25 MPa. They also noted that above a certain inlet temperature no instabilities occur, which was also found here. For an N_{SUB} < 0.18 there is no instability found in the experiments. This could therefore be considered as an inherent safe zone for operation. As such, there exists a trajectory to move from zero power/low inlet temperatures towards high power/high inlet temperatures, something that is impossible for natural circulation BWR's as the boiling boundary and the neutral stability boundary cross each other at the origin of the stability plane. Such a trajectory might be exploited during the start-up phase of the HPLWR as long as the reactor vessel can be pressurized beforehand.

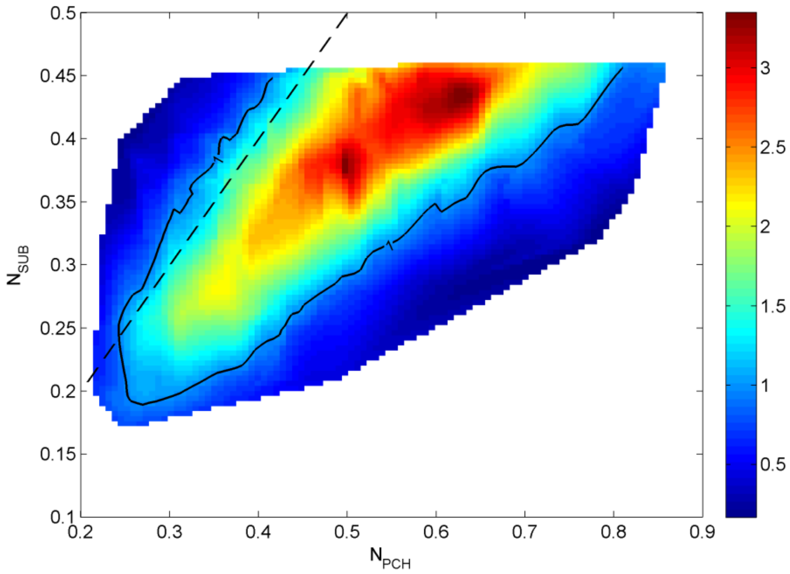


Figure 5: Contourplot of the measured DR values for τ = 6s, HPLWR power distribution.

The DR plot is similar to that of a natural circulation driven BWR, see Van Bragt and Van der Hagen (1998). In such a system, for a given subcooling number, the flow will undergo three transitions: (i) from stable to unstable, crossing the boiling line $h_{out} = h_{sat}$ and moving towards the low frequency type-I instability, (ii) from unstable to stable and (iii) then, at higher power, again from stable to unstable, moving towards the high frequency type-II instability. The measured frequency for these instabilities is about 0.1 Hz. In Fig. 5 the supercritical equivalent of the boiling line, the 'reference line', $h_{out} = h_{pc}$ is shown as a black dashed line. As can be seen, the instabilities occur very close to this line. As the density of a supercritical fluid is a continuous function (compared to the step change for a boiling medium), it should be clear that even if the temperature at the outlet is lower than T_{pc}, a strong density gradient occurs, which can trigger instabilities. So this 'reference line' can't be a fixed boundary for a supercritical system, as it is for a BWR. As such, the location of the instabilities in the stability plane and the recorded frequencies both clearly suggest that these instabilities are in fact the supercritical analogue of the type I instability in a BWR.

The instability threshold moves almost parallel to the reference line at $N_{SUB} > 0.27$. This suggests that an exit temperature close to T_{pc} could be used as a criterion for instability. The graphs reported by Sharma et al. (2010) show a similar trend for the thermo-hydraulic mode. They also showed that this first transition is insensitive to the loop height and local friction distribution (though only small variations was examined). This indicates that this boundary is a more general property of a supercritical loop, which is consistent with the aforementioned idea of extending the reference line from type I instabilities of BWR.

Impact of the fuel time constant

The fuel time constant has a strong effect on the stability of the system. This is highlighted in Fig. 6. Increasing the fuel time constant results in a more unstable system. It increases the size of the instable zone, shifting it to lower N_{PCH} and N_{SUB}. The peak value of the DR also increases considerably. For τ = 2s the peak value is about 1.3, whereas for τ = 6s this is almost 2.5. The frequency remains unaffected, and is about 0.1-0.12 Hz for all measured instabilities. This shows that the fuel time constant is an important system parameter for the coupled neutronic – thermo-hydraulic instability in a supercritical loop. As this is related to the size of the fuel rods, it is an important design parameter to consider for the safe operation of the system in natural circulation mode.

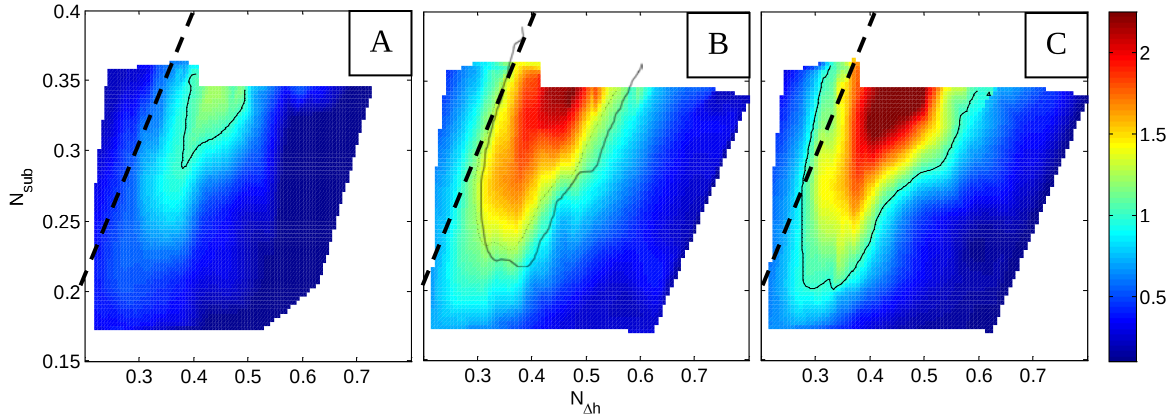


Figure 6: Contourplots of the measured DR values for τ = 2s (A), τ = 4s (B) and τ = 6s (C), HPLWR power distribution. The black line indicates the stability threshold, and the dashed line the 'reference' line

Impact of the power distribution

Because of the three pass core layout, whereby the evaporator is located at the center of the HPLWR, and superheater I and II concentrically around it, the power distribution per 'pass' is very different. The power distribution is 53% for the evaporator section, 30 for superheater I and 17% for superheater II (Fisher et al., 2007). This power distribution will have an effect on the density profile in the core, and as such on the stability of the system. To determine how strong the power distribution affects the stability, a uniform power distribution was considered here. Each pass then provides one third of the total power. The fuel time constant was set to 6 seconds. The results are shown in Fig. 7. As can be seen, the uniform power distribution is considerably more stable, as the unstable zone is shifted to higher N_{PCH}. The left boundary still runs parallel to the reference line, but is shifted to the right. The location of the tip is shifted to slightly higher N_{SUB}. The shift to higher power is related to the reduced density gradients occurring in the core sections, as the power is spread out more evenly. This also reduces the mean friction at a given power level, as the average core density is higher, lowering the mean core velocity. This shows that the power distribution is another important parameter to consider in the reactor design, not only from a neutronic point of view for the final power output, but also considering the stability of the natural circulation.

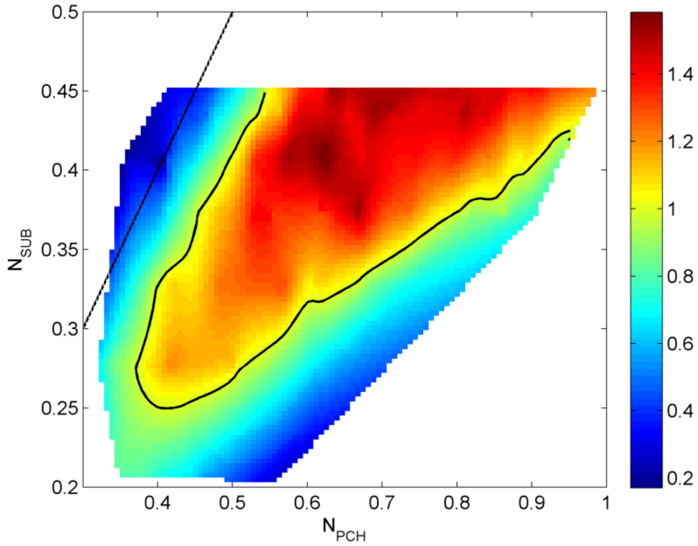


Figure 7: Contourplot of the measured DR values for τ = 6s, uniform power distribution.

CONCLUSIONS

An experimental study was performed on the thermo-hydraulic and coupled neutronic – thermo-hydraulic stability of a natural circulation HPLWR. This was done with the scaled DeLight facility using Freon R23 at 5.7 MPa. The result show there are no pure thermal-hydraulic instabilities, but applying the neutronic feedback does result an unstable zone for certain operating conditions. The measured frequency and the location in the stability plane indicate that the found instabilities are in fact the supercritical equivalent of the type I instabilities of a natural circulation BWR. A parameter study was performed by examining the impact of the power distribution and the fuel time constant on the nature of the instabilities. Lowering the fuel time constant and making the power distribution more uniform both make the system more stable. These results show how that the design of the fuel rods (size and enrichment) and the layout of the core are not only important from the point of view of the power generation or the heat transfer, but are also very important when considering a natural circulating system.

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