THERMAL-HYDRAULICS AND SAFETY CONCEPTS OF SUPERCRITICAL WATER COOLED REACTORS

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Abstract

The paper summarizes the status of safety system development for supercritical water cooled reactors and of thermal-hydraulic codes needed to analyze them. While active safety systems are well understood today and expected to perform as required, the development of passive safety systems will still need further optimization. Depressurization transients have successfully been simulated with some codes by a pseudo-two-phase flow simulation of supercritical water. Open issues of thermal-hydraulic codes include modeling of deteriorated heat transfer in one-dimensional system codes and predictions of heat transfer during depressurization transients from supercritical to sub-critical conditions.

1. Introduction

The concept of a Supercritical Water Cooled Reactor (SCWR) is basically following the trend of coal fired power plants to increase the live steam temperature for increasing thermal efficiency and specific turbine power. This goes along with an increase of live steam pressure to maximize the turbine enthalpy difference at a given condenser pressure. Coal fired power plants built in the 1960ies were designed with drum boilers, in which a two-phase flow from the evaporator was separated in the drum, such that saturated steam was supplied to the superheater and liquid was recirculated either by natural convection or by a recirculation pump to the evaporator inlet. Since the 1990ies, the live steam pressure exceeded the critical pressure and steam and liquid could not be separated any more. Consequently, the recirculation of liquid was omitted which avoided also the costs of the drum and of the recirculation system. This simplified system is known as a once through boiler.

Boiling water reactors (BWR) are still comparable with drum boilers with their separators and dryers and with their flow recirculation through the downcomer inside the reactor. An innovative approach to improve this system with a superheater was tried in the HDR reactor in Grosswelzheim in Germany [1], which failed unfortunately already during the commissioning phase due to a design error. A better performance is expected from supercritical systems. Besides simplification of the system, supercritical water has the advantage of an excellent heat transfer in the entire heat up range from liquid to superheated steam because dryout is physically excluded. Being a once through system, the separators and dryers of the BWR are omitted in the SCWR, as well as fluid recirculation through the downcomer. This latter simplification, however, causes a basic difference to the control of the steam cycle.

The simplified control systems of a BWR and an SCWR are compared in Fig. 1. In a BWR, the feedwater pump is controlling the liquid level in the reactor pressure vessel, the steam pressure is controlled by the turbine governor valve, and the core power is either controlled by the control rods or by the speed of the recirculation pumps, indicated with blue circles in Fig. 1, left. The SCWR, on the right hand side, does not include any recirculation loop. The feedwater pump can either control the steam temperature at the core outlet if the core power is controlled by the control rods, or it can control the core power if the steam outlet temperature is controlled by the control rods. Again, the steam pressure is controlled by the turbine governor valve in both cases.

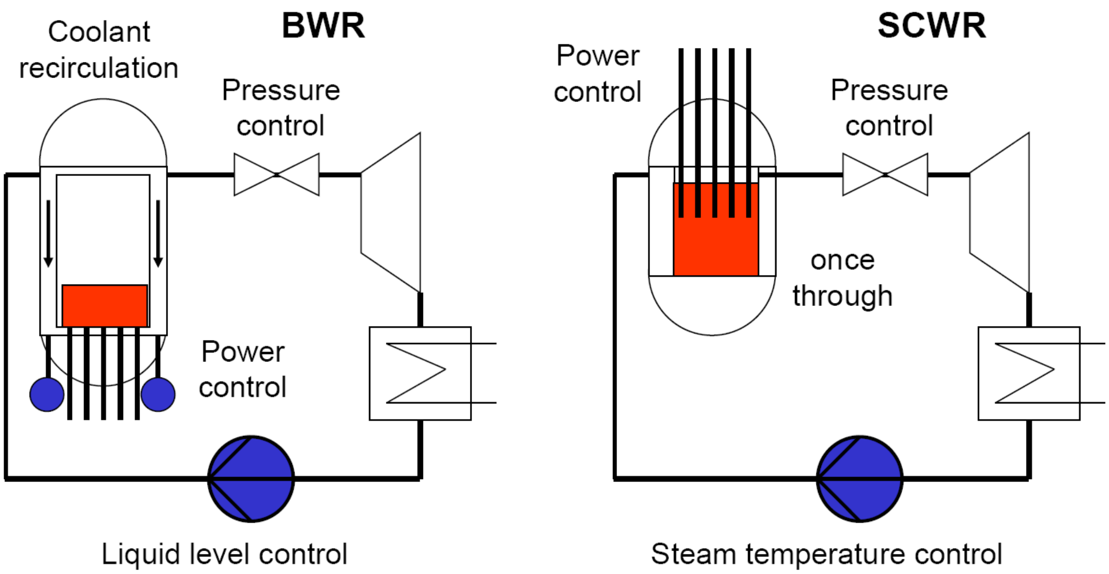

Fig. 1: Comparison of BWR and SCWR general features.

The once through system is causing also a basic difference to the general requirements of the safety system. Having a closed coolant loop inside the reactor, the BWR can remove residual heat by natural convection, driven by the rising steam in the core and above, and the safety system has to ensure sufficient coolant inventory in the pressure vessel to keep the core covered with water. The SCWR, on the other hand, can remove the residual heat only by forced convection inside the reactor, which may be driven by a natural convection loop outside, but the requirement for the safety system, in general, is to ensure sufficient coolant mass flow rate instead.

Besides this difference, there are several common safety system requirements, which can be taken directly from BWR concepts without significant modifications. These are:

- The reactor shut down system by control rods or by a boron injection system as a second, divers shut down system.
- Containment isolation by active and passive containment isolation valves in each line penetrating the containment to close the third barrier in case of an accident.
- Steam pressure limitation by pressure relief valves.
- Automatic depressurization of the steam lines into a pool inside the containment through spargers to close the coolant loop inside the containment in case of containment isolation.

- A coolant injection system to refill coolant into the pressure vessel after intended or accidental coolant release into the containment.
- A pressure suppression pool to limit the pressure inside the containment in case of steam release inside the containment.
- A residual heat removal system for long term cooling of the containment.

The following chapter will show some examples of safety concepts which had been worked out recently to meet these requirements.

2. Concepts of Safety Systems for Supercritical Water Cooled Reactors

A minimum set of safety systems which fulfill the above mentioned requirements is sketched in Fig. 2. The reactor shut down system is provided by shut down rods which can fall into the reactor from the top like in a pressurized water reactor (PWR), since separators and dryers are not complicating the design like in a BWR. An example is given by Koehly et al. [2] for the High Performance Light Water Reactor (HPLWR). Control rod drives outside the reactor as well as control rod guide tubes inside can be taken from PWR design without significant modifications. A vessel with boron acid must be provided inside or outside the containment for additional shut down under accidental conditions. Different from PWR control, however, this boron acid cannot be used to compensate excess reactivity during normal operation.

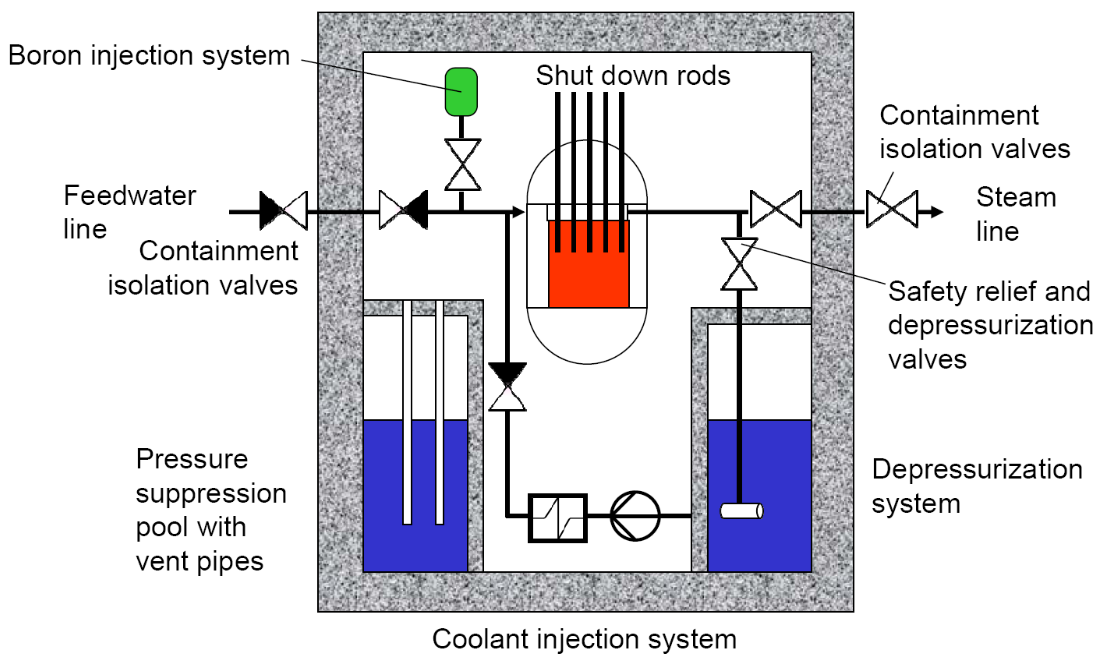

Fig. 2: Minimum set of safety systems for SCWR

Containment isolation valves can be check valves in feedwater lines, which need to be damped to avoid a water hammer, and steam isolation valves with hydraulic and medium controlled actuators as described by Sempell in [3]. A pressure suppression pool with vent pipes is keeping the containment pressure below the design limits, which can also serve as a heat sink for the automatic depressurization system in the simplest case. A low pressure coolant injection system with a heat exchanger to a secondary emergency coolant system is shown underneath the reactor in Fig. 2, but is meant to be placed somewhere inside or outside the containment in a sheltered position. These systems look quite similar as those of conventional BWR. The different response of these systems in SCWR, however, shall be discussed by studying a loss of feedwater accident.

Let us imagine the case of a simultaneous trip of all feedwater pumps caused e.g. by a station black out. These feedwater pumps are high pressure, multistage centrifugal pumps which must be equipped with a check valve each to avoid backflow in case of a trip of a single pump. These check valves, as well as those for containment isolation, will stop the feedwater flow within a few seconds and even a potential flywheel of the feedwater pumps could not extend the short coast down time. Different from a feedwater pump trip in a BWR, therefore, this case is equivalent with a loss of coolant flow to the core within a few seconds, requiring a scram of the reactor. As a consequence, the system must be depressurized immediately, being the only option to maintain a coolant mass flow rate through the core, either through the turbines and through the turbine by-pass valves as an immediate action, or through the automatic depressurization system inside the containment to avoid loss of coolant to the outside of the containment. It is not wise to close the turbine governor valve in this case to keep a high system pressure, like in a BWR, as such measure would stop the steam flow simultaneously, which would overheat the core.

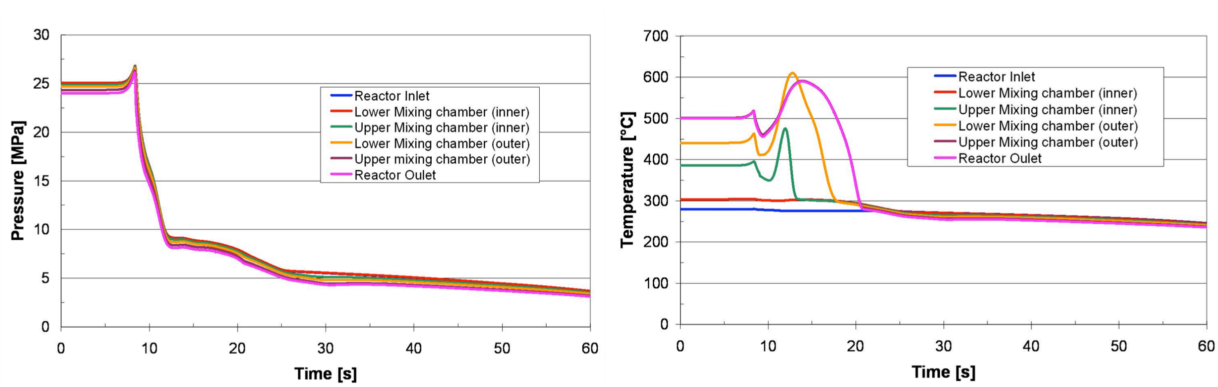

Fig. 3: Depressurization after containment isolation; left: pressure history, right: coolant temperatures at different locations of the HPLWR core concept [4].

The pressure and coolant temperature history in case of containment isolation of all feedwater and steam lines has been simulated by Schlagenhaufer et al. [4] for the HPLWR with its coolant flow path as described by Koehly et al. [2]. We see in Fig. 3 (left) that the containment isolation will first cause a short pressure peak, which actuates the automatic depressurization system (ADS) of the steam lines, followed by rapid pressure decrease. The temperature history shown in Fig. 3 (right) gives a short temperature peak, caused by a 0.6 sec delay time of the ADS and 3.5 sec shut down time of the control rods, but the coolant temperature falls rapidly afterwards to the feedwater temperature of 280°C because of the high coolant mass flow rate during depressurization. Within 20 sec after containment isolation, the pressure has reached the saturation pressure of the feedwater inside the reactor and the feedwater in the upper plenum will start boiling. This situation will keep a minimum pressure in the vessel of initially 6.4 MPa which is decreasing slowly such that the core will be well cooled for about 10 min.

If the low pressure coolant injection system will be designed with a pressure head of at least 6 MPa, and if the emergency power supply and the ramp up of the coolant injection pump can be provided within 20 sec in total, the core will be well cooled for long term as the cooling circuit is now closed within the containment before a significant amount of coolant was lost, and the residual heat will be removed to the secondary coolant system. This time is short but feasible in conventional boiling water reactors. It would be advantageous, however, to have a longer grace period. The feedwater volume stored in the reactor pressure vessel has sometimes been called an "in-vessel accumulator" [5], suggesting that this water volume may be spent for cooling before the low pressure coolant injection becomes available. Indeed, the core is well cooled by this water for about 10 min, as described above. As soon as cold water is injected, however, the steam pressure in the reactor breaks down and the core flow is stopped until the reactor has been filled up again. The in-vessel accumulator acts as a pressurizer, providing a driving pressure head for the coolant only as long as its temperature is sufficiently hot.

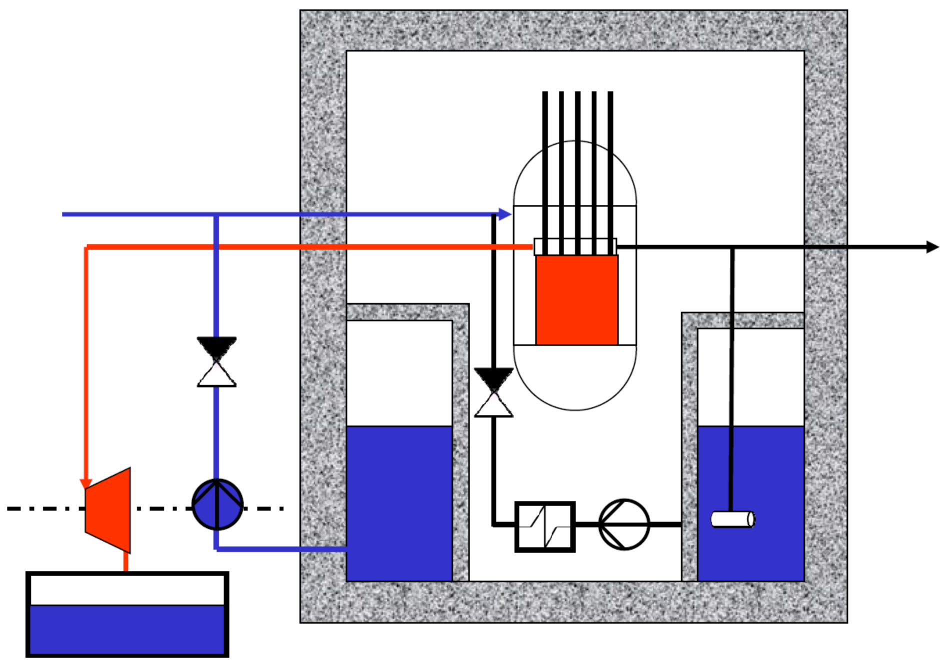

Fig. 4: Depressurization through a steam turbine driving a high pressure coolant injection pump

The problem can be overcome if the system is depressurized through a steam turbine driving a high pressure coolant injection pump, as sketched in Fig. 4. Such system has often been used already in conventional BWR. As the condenser behind this turbine must be at lower elevation than the turbine outlet, but the pump intake must be lower than the water reservoir, this concept is usually designed with 2 coolant pools at different elevation. As sketched in Fig. 4, Ishiwatari et al. [6] propose to use a separate condensate storage tank at lower elevation, like in a BWR. Now the missing coolant will be refilled already during depressurization. The steam mass flow must be high enough to ensure that the maximum cladding surface temperature in the core does not exceed the envisaged limit, but small enough to maximize the grace period for the active, low pressure coolant injection system.

A passive system without rotating components could be a closed loop which condenses the steam in an additional upper pool inside the containment, as sketched in Fig. 5. This system decreases the system pressure slowly, but the flow rate could eventually be too small to cool the core. Therefore, de Marsac et al. [7] propose to drive the coolant loop additionally with a steam injector. After an initial short depressurization through the ADS, the subcritical steam is supplied to the steam injector which drives a closed coolant loop through the condenser in the upper pool. Coolant is lost to the containment pool only during the short initial depressurization phase, and the steam supplied to the injector afterwards is condensing in the closed system. This innovative system, however, has never been analyzed in detail.

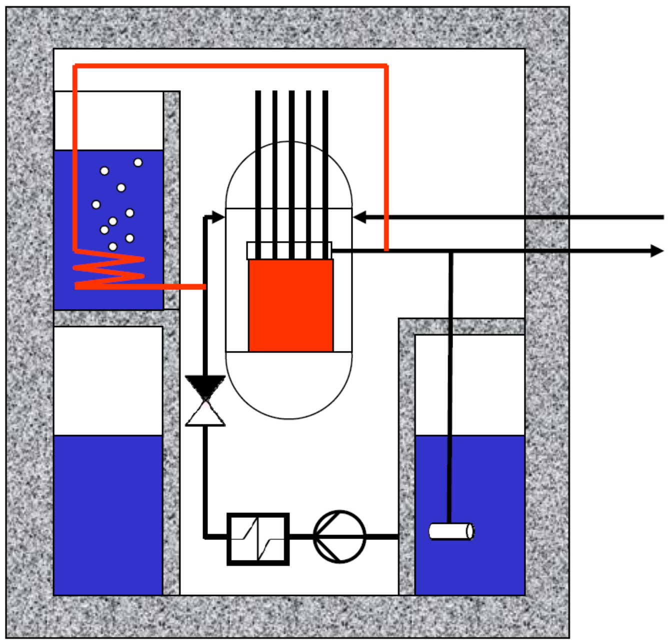

Fig. 5: Depressurization by condensation in a closed loop inside an upper containment pool

Instead, Schlagenhaufer et al [4] propose to use a motor driven recirculation pump to drive the closed coolant loop through the condenser in the upper pool, which is easier to control and thus easier to optimize for this purpose. Fig. 6 shows the response of the reactor coolant during a depressurization transient using this system. The transient was initiated at time step 5 sec by inadvertent containment isolation, which caused a pressure peak, scram and activation of the ADS as described above. Despite the peak mass flow rate of the steam leaving the reactor outlet (a), a short temperature peak of the coolant cannot be avoided again (b). Simultaneous with scram activation, a recirculation pump in the condensate line of the closed loop is started now, and the ADS is closed again as soon as the pressure is less than 10 MPa. As a consequence, the coolant temperature at reactor inlet drops suddenly to 20°C at time step 20 sec, as the condensate stored in the loop has been cold during normal operation. Around 1 min after scram, the closed loop has stabilized, and the condensate temperature increased to the saturation temperature at actual system pressure (c). The core outlet temperature is controlled by the recirculation pump such that is stays slightly superheated, which minimizes the coolant mass flow rate and thus the required power of the pump. Fig. 6 (d) shows more than 90% void in the core after 5 min and even the feedwater inside the reactor “in-vessel accumulator” is boiling, but the core remains to be cooled sufficiently. The total peak power of 4 recirculation pumps needed for this system was 1 MW. This exercise could serve as a starting point for a passive system, e.g. with a condenser at higher elevation.

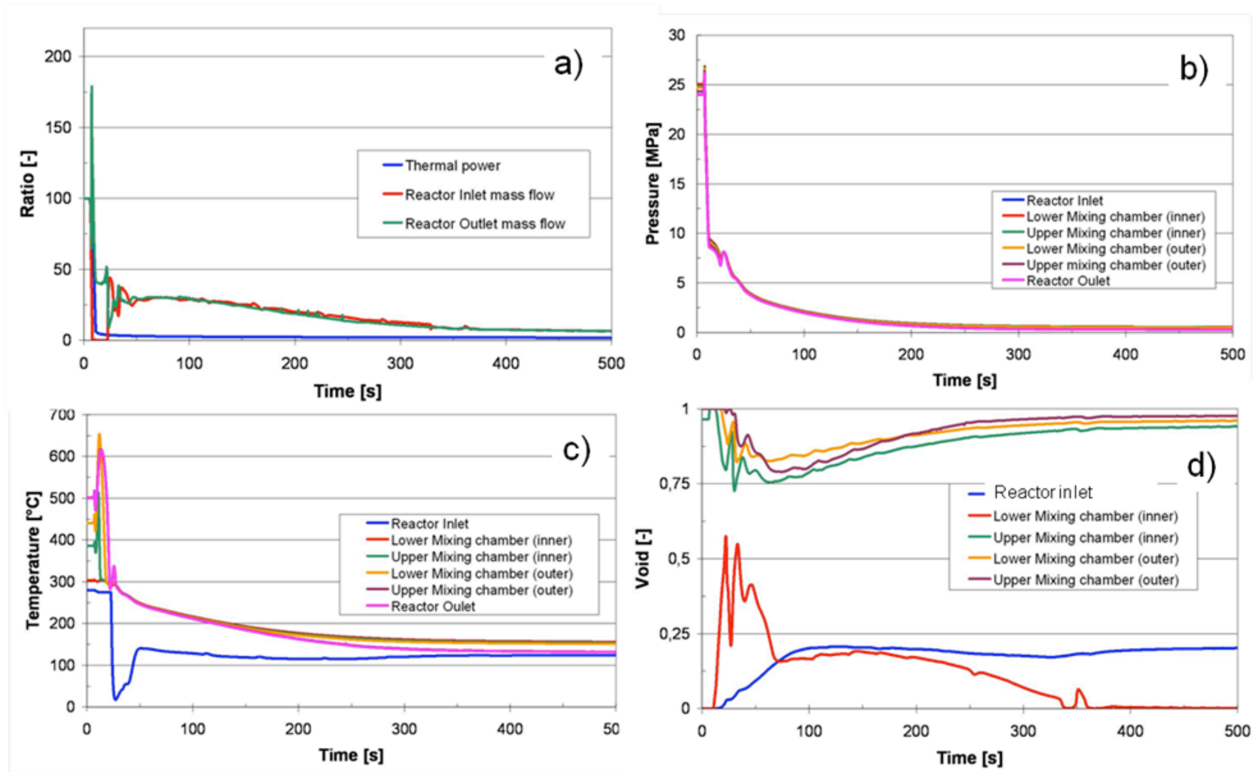

Fig. 6: History of core power and coolant mass flows (a), coolant pressure (b), temperature (c) and coolant void (d) during depressurization in a closed loop [4].

The potential arrangement of such safety systems inside the HPLWR containment is illustrated in Fig. 7. The active low pressure coolant injection system and its heat exchangers for residual heat removal are placed underneath the annular pressure suppression pool. A condensate recirculation pump of a closed loop, as described above, would need to be installed there as well to have enough pressure head at its intake avoiding cavitation. Four upper pools serve as a heat sink for the ADS and for the closed loop condenser sketched in Fig. 5. Containment condensers hanging from the ceiling of the containment transfer residual heat to pools above the containment when the temperature of the four upper pools and of the pressure suppression pool has reached the saturation temperature, limiting the containment pressure. With 26.7 m total height and 21.6 m outer diameter, the containment is surprisingly compact, indicating a significant potential for cost reduction, but it is storing more than 2000 m^3 of water.

Several system codes, i.e. RELAP5, CATHARE, APROS, KIKO3D-ATHLET and SMABRE/TRAB-3D, have been used to analyze this system under loss of flow, loss of off-site power, loss of coolant and control rod ejection accidents, as summarized by Andreani et al. [9]. Results confirm the appropriateness of the proposed active safety systems, so far, but passive systems will still need to be optimized. On the other hand, the thermal-hydraulics codes themselves are not yet considered to be optimal either, as will be discussed next.

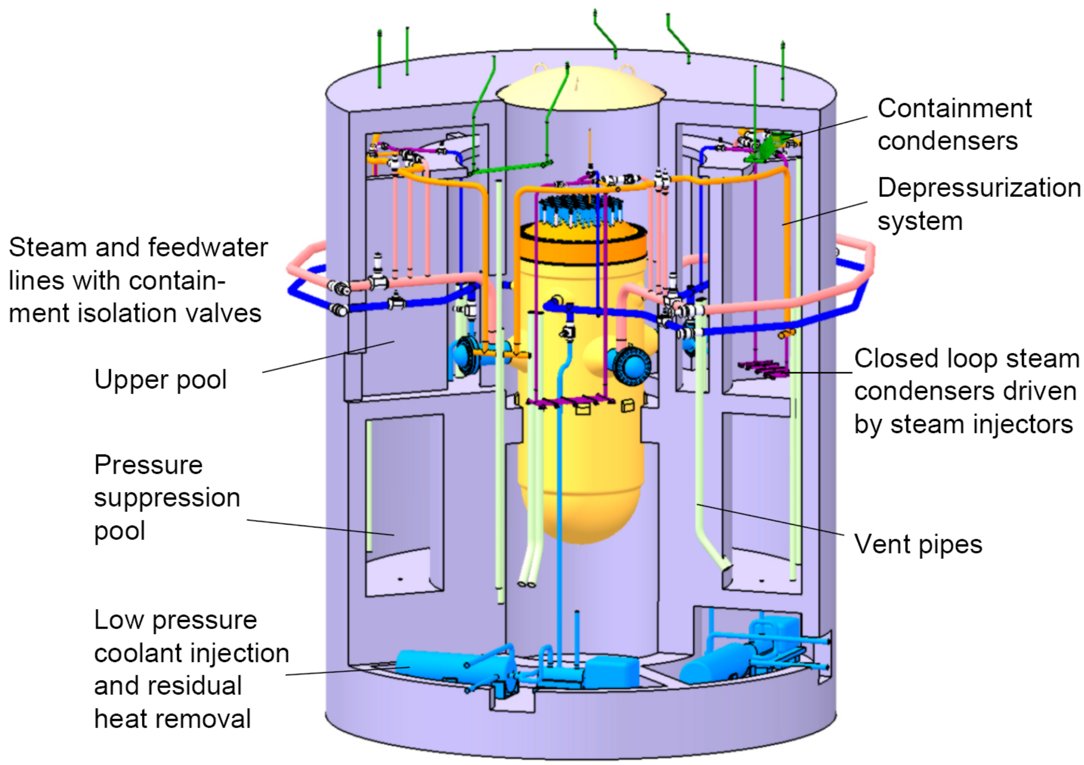

Fig. 7: Arrangement of safety systems in the HPLWR containment [8].

3. Challenges of thermal-hydraulic modeling of supercritical water

Extending a system code to supercritical pressure seems to be an easy task. As supercritical water is a single phase fluid, we just add another fluid using precise steam tables, heat transfer correlations and pressure drop correlations and the system code can handle such fluids as well. A problem arises, however, if we try to depressurize the reactor. As the coolant is a sub-cooled liquid at the reactor inlet but superheated steam at its outlet, the fluid passes the pseudo-critical line somewhere in the core, i.e. the line where liquid like properties are changing to steam like properties and the specific heat has a maximum. Once the critical pressure is reached in the core, this line will end in the critical point and a local two-phase flow region will appear instead. Several attempts to model this transition with RELAP5 and with ATHLET failed up to now: the iteration procedure diverged.

The problem was solved for the CATHARE code by Antoni and Dumaz [10] and later for the APROS code by Hänninen and Kurki [11] by modeling supercritical water as a pseudo two-phase flow. If the fluid enthalpy is less than the pseudo-critical enthalpy, the fluid is treated as liquid with zero void α, and as vapor with α=1 if the enthalpy is greater than the pseudo-critical enthalpy, as sketched in Fig. 8. A small artificial evaporation enthalpy is introduced to model the continuous transition as a small step when passing the pseudo-critical line. The flow is modeled with 6 equations for conservation of mass, momentum and energy, assuming that both pseudo-phases have the same velocity, that stratification is zero, and that surface tension is zero, which avoids any liquid entrainment. In the narrow range of pseudo-vaporization, the void is changing from zero to one like with sub-critical fluids. This approach allows modeling the supercritical range with the same set of equations as sub-critical fluids with a change of the parameters only. The additional uncertainty of pseudo-evaporation can be minimized by reducing the artificial heat of vaporization. Both codes worked successfully with a smooth transition to sub-critical pressures in all transient analyses.

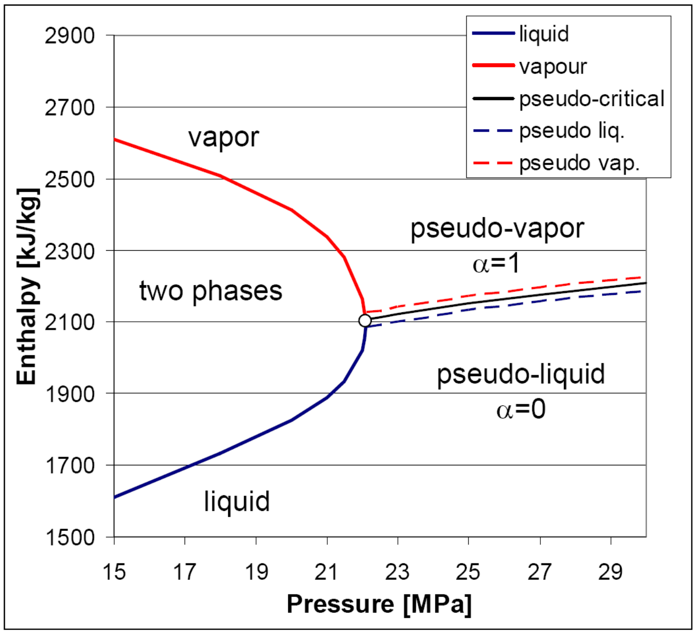

Fig. 8: Modeling of supercritical water with APROS and CATHARE

Another challenge of thermo-hydraulic modeling of supercritical water is its strange heat transfer characteristic. If the cladding surface temperature of fuel rods or of a heated tube is greater than the pseudo-critical temperature, but the bulk temperature is less than this, we risk a deterioration of heat transfer with high peak temperatures, as shown exemplarily in Fig 9 using the experimental data of Shitsman [12] for a heated tube of 8 mm inner diameter. Such peak temperatures are certainly a safety concern. Unfortunately, none of the heat transfer correlations published up to now can predict these phenomena properly.

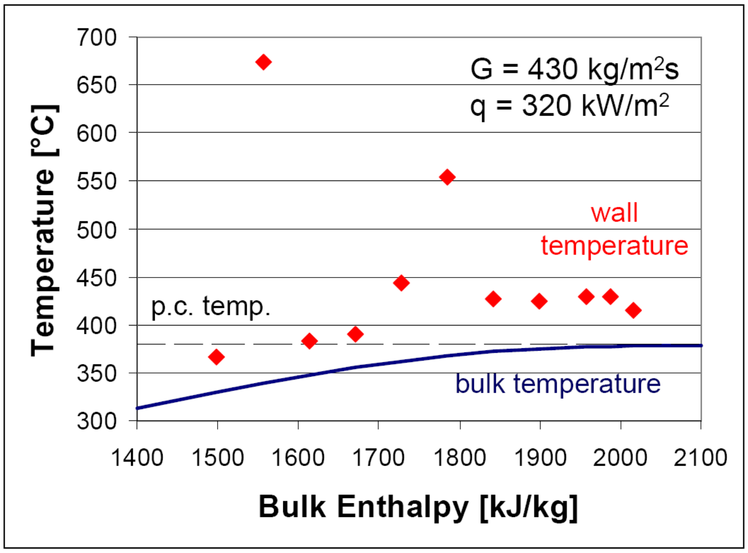

Fig. 9: Measured inner surface temperature of a heated tube with mass flux G=430 kg/m^{2}s, heat flux q=320 kW/m^2 at a pressure of 23.3 MPa, Shitsman [12].

We can explain the strange heat transfer in principle with the change of fluid properties in a fully developed laminar flow. Fig. 10 shows the fluid properties as a function of distance to the wall at 23.3 MPa assuming a wall temperature of 700°C, a heat flux of 1000 kW/m^2, and neglecting any convective heat transfer for the discussion. Such heat flux is not unusual in SCWR. It corresponds with a linear heat rate of 25.1 kW/m of a fuel rod with 8 mm outer diameter. For comparison, a boiler tube of a coal fired power plant is designed for a maximum heat flux of 400 kW/m^2 only. We see in Fig. 10 that the thermal conductivity in a near wall layer of around 30µm is significantly smaller than outside, which causes a thermal insulation. The small fluid density there will cause buoyancy effects, at least if the heated tube is vertical, and the small viscosity will act as a lubrication film, which reduces shear forces and thus the transition to turbulence. The transition zone from steam like to liquid like properties appears in a tiny layer of a few µm thickness, while the specific heat has a pronounced peak there.

We are convinced that a direct numerical simulation (DNS) of a turbulent flow can model the observed effects properly, if the mesh size is chosen to be small compared with the transition zone shown in Fig. 10. In other words, the Navier-Stokes-equations should be valid even in these extreme cases. The DNS approach, however, is not considered to be appropriate for safety analyses. The CPU time can be reduced significantly with a large-eddy-simulation which assumes, however, that fluid properties are constant at least within small scale turbulence structures. Conventional turbulence models like the k-ε or k-ω model are assuming even that fluid properties are constant within all turbulence structures, which does not seem to be justified looking at Fig. 10. It was surprising to see, therefore, that Palko and Anglart [13] succeeded to simulate the experiment of Shitsman [12], shown in Fig. 9, with the SST model which combines the k-ε model with the k-ω model. They used a very fine grid near the wall with a distance of the dimensional variable $y^+ < 1$ of the wall nearest grid point. Application of this method to heat transfer test of Ornatskii et al. [14], however, had not always been so successful.

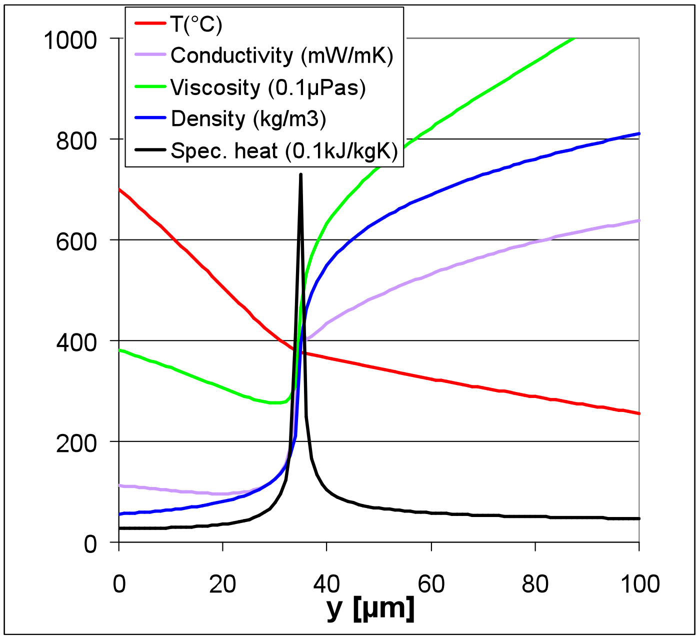

Fig. 10: Change of fluid properties normal to a hot wall in a fully developed laminar flow at 23.3 MPa pressure and with a heat flux of 1000 kW/m^2.

More effective for engineering applications should be the use of wall functions to enable a coarser grid and thus to allow modeling of a larger fluid domain. Conventional wall functions are assuming constant fluid properties in the entire boundary layer, which is certainly not valid regarding Fig. 10. Laurien [15] tried to improve these wall functions by integrating the energy equation algebraically over the viscous, heat conducting sub-layer with its laminar flow, and by improving Prandtl's mixing length theory for the logarithmic layer with a probability density function for the specific heat. This approach could show indeed an increase of wall temperatures of a turbulent boundary layer when the pseudo-critical temperature was exceeded at the wall. For validation, Laurien selected a case with a mass flux of 1530 kg/m^{2}s and a heat flux of 1610 kW/m^2 at a pressure of 25.5 MPa in a tube with 3 mm inner diameter. A comparison with experimental data of Ornatskii et al. [14] showed, however, that this method is still underestimating the observed temperature peaks by far.

Although the deterioration of heat transfer to supercritical water is observed in simplified smooth geometries, this phenomenon might well be absent in the core of a real reactor where the spacers and the fuel rod arrangement in a bundle enhances turbulent mixing significantly. The HPLWR fuel assembly, for instance, consists of 40 fuel rods in a square arrangement where a helical wire has been selected as spacer and mixing device. The helical wire-wraps guide the flow around the fuel rods, generating a strong swirl flow and effective mixing between the different sub channels in the fuel assembly. In addition the wire-wraps are flow obstacles that increase turbulence and improve heat transfer. The effect of a wire-wrap spacer on the heat transfer to supercritical water is studied with CFD for a four rod bundle by Chandra et al. [16]. Fig. **11:**. 11 shows a comparison of the calculated surface temperature on the fuel rods of an unwired and wired four rod bundle assembly for a mass flux of 1332 kg/m2s, a heat flux of 1375 kW/m2 and a pressure of 25 MPa. This clearly demonstrates that with wire present, the temperature on the fuel rod surface is more uniform. It was found that the wire-wrap reduces the average surface temperatures of the fuel rods and results in an improved heat transfer. Lycklama a Nijeholt et al [17] analyzed the effect of wire-wraps on the heat transfer to supercritical water for different representative geometries and conditions. They found that the average heat transfer rate is improved by about 10% by the presence of a wire-wrap. In this study it was also observed that heat transfer deterioration was mitigated by the presence of the wire-wrap for one of the considered geometries, i.e. the sub-channel. A similar mitigating influence of the wire-wrap is observed for a square annulus as well in calculations by Zhu and Laurien [18]. However, for a conclusive answer if wire wraps can prevent deterioration of heat transfer, the prediction of this phenomenon should be improved and further investigations are necessary.

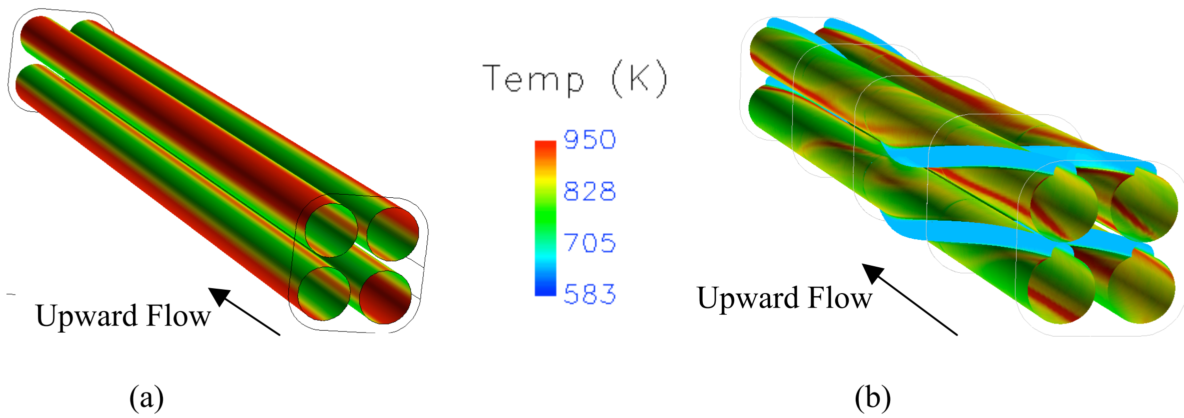

Fig. 11: Calculated surface temperature contours on the fuel rods for an unwired (a) and wired (b) four rod-bundle assembly [16].

Predictions of heat transfer are getting even more complicated when we look at depressurization transients. As the critical heat flux at near critical pressure is approaching a minimum towards the critical pressure, film boiling can easily occur after slow depressurization from supercritical conditions if the local wall temperature had been hotter than the critical temperature before. The consequences can be hot, local temperature peaks when the critical pressure is passed. An experiment of Kang and Chang [19] with Freon at near critical pressure gives a first indication of phenomena to be expected. They concluded from their tests that steady state heat transfer correlations overestimated the local heat transfer coefficient by around 30%, which is another cause for concern in safety studies.

4. Conclusion

Thermal-hydraulics and safety concepts for SCWR are the objectives of a joint international project of the Generation IV International Forum, in which the authors are managing the European contributions. The discussion of status and open issues in this paper illustrates that significant progress has been achieved in the last 5 years, but there is still a lot to be done.

- The concepts for active safety systems which had been studied for the SCWR in Japan and in Europe are promising, even though the SCWR is certainly more challenging than conventional light water reactors. The development of passive safety systems, however, will need further optimization.

- Several system codes needed to optimize SCWR safety systems had been extended to supercritical conditions in the past, but depressurization transients can still only be handled by a few codes. At least, we can recommend a pragmatic method to solve this issue.
- The deterioration of heat transfer at supercritical pressure in case of high heat flux and low mass flux is a safety issue which is still hard to be predicted properly, as well as the occurrence of film boiling phenomena after slow depressurization to sub-critical conditions. For application to safety system analyses, a simple heat transfer correlation would be favored, but even precise CFD predictions with fine resolution of the boundary layer turned out to be a challenge.

Prediction methods are intended to be improved in the upcoming years by test with prototypical components, such as the in-pile test of a small scale fuel assembly with 4 rods or out-of pile tests with electrically heated fuel bundles. The International Forum is sharing results and invites for worldwide collaboration.

5. Acknowledgement

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