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EXPERIMENTS ON MAIN STEAM LINE BREAK IN THE TEST FACILITIES PKL AND ROCOM

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Abstract

Non-isolable main steam line breaks in PWRs cause a rapid depressurization of the affected steam generator. This leads to increased heat transfer from the primary to the secondary side and thereby to a fast cooldown transient on the primary side. Under certain boundary conditions the reactor pressure vessel integrity considering PTS (pressurized thermal shock) and potential recriticality following entrainment of colder water into the core area are important aspects to be assessed. Complementary tests in the PKL (system behavior) and ROCOM (mixing behavior in the reactor pressure vessel (RPV) downcomer and lower plenum) facilities have been performed on this subject. This paper summarizes the main outcome of these experiments and their use for safety analysis.

Introduction

For many years extensive experimental investigations into the system response of PWR under accident situations have been conducted in the integral test facility PKL. The experiments conducted in the PKL test facility cover a very broad spectrum of topics ranging from large- and small-break loss-of-coolant accidents when the facility was first built up to the simulation of transients including the effects of measures to prevent severe accident situations. The PKL tests performed to date have altogether contributed to a better understanding of the sometimes highly complex thermal-hydraulic processes involved in various accident scenarios and to a better assessment of the countermeasures implemented for accident control. Another important benefit of the PKL tests is that they provide an extensive database for use in the further development and validation of thermal-hydraulic computer codes.

Since 2001 the PKL project has been included in an international program set up by the OECD with 15 participating countries [1]. The current OECD-PKL 2 test program is focusing on the detailed investigation of heat transfer mechanisms in the steam generators under accident situations (e.g. reflux condensation in presence of nitrogen, cooldown procedures in presence of steam generators that are isolated on the secondary side, fast secondary side cool down transients). The experiments in PKL are complemented by experiments in the test facilities PMK (operated at KFKI, Hungary) on heat transfer in PWRs with horizontal steam generators and ROCOM (operated at HZDR, Germany) on mixing in the RPV downcomer and lower plenum. One major topic of the current test program which will last until the end of 2011 is dealing with a main steam line break (MSLB).

A non-isolable main steam line break (up to double ended guillotine break) in a PWR causes a rapid boil off and depressurization of the affected steam generator. As a consequence the temperature difference between the primary and secondary side increases and a large heat sink develops on the secondary side leading to a subcooling transient and a decrease in primary side pressure (due to volume contraction). In this context the assessment of the reactor pressure vessel integrity considering PTS (pressurized thermal shock) aspects is one important point for this accident scenario. The assessment of potential recriticality following entrainment of colder water into the core area is another important subject. Tests in the PKL and ROCOM test facilities were performed based on these investigation goals (PTS, recriticality). In PKL the overall system behavior was investigated. The results (that is, the measured mass flows and temperatures at the RPV inlet) provided the boundary conditions for complementary tests on mixing cold and hot water in the RPV downcomer and lower plenum in ROCOM.

1. PKL-Test

1.1 The Large Test Facility PKL III

The large-scale test facility PKL (Fig. 1) is a scaled-down model of a pressurized water reactor of KWU design of the 1300 MW class. It is employed to experimentally investigate the thermal-hydraulic behavior of a PWR in accident scenarios, as well as the effectiveness of emergency measures. The reference plant is Philippsburg 2 nuclear power plant, which has a thermal power of 3770 MW[1].

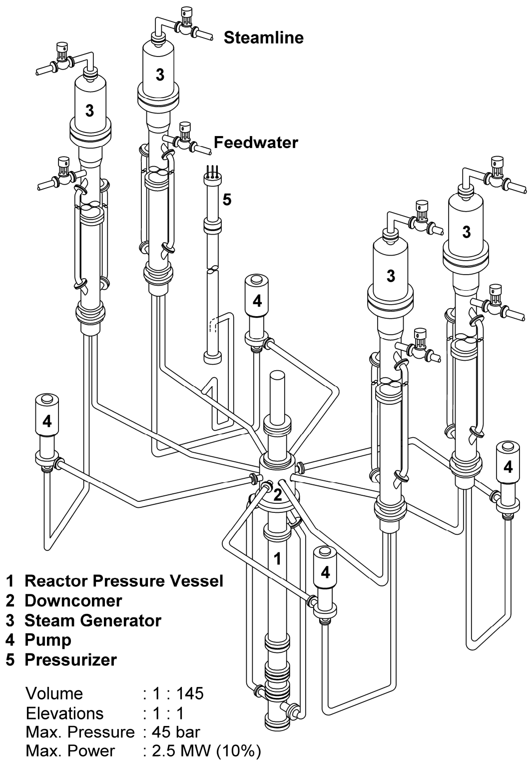


Figure 1: PKL Test Facility

[1] At the time of the design of the PKL test facility.

The PKL test facility models the entire primary side and essential parts of the secondary side (without turbine and condenser) of the reference plant. All elevations are scaled 1:1, volumes, power and mass flows are modeled at a scale of 1:145. The test rig is equipped, as is the reference PWR, with 4 loops on the primary side symmetrically arranged around the reactor pressure vessel (RPV). In the test facility, the reactor core is simulated by a bundle of 314 electrically heated rods with a total power of 2.5 MW corresponding to 10 % of the rated thermal power. The maximum pressure on the primary side is 45 bar.

1.2 PKL Test for a Main Steam Line Break

Non-isolable main steam line breaks (up to a double-ended break) cause a rapid decrease in secondary-side pressure in the affected steam generator (SG). This leads to increased heat transfer from the primary to the secondary side, and therefore to pronounced cooling of the primary coolant in the affected loop (subcooling transient). An important question during this process is, whether a localized recriticality of the core and the resulting power excursion can occur due to the entry of cold water into the reactor core area.

The PKL test was started from hot-standby conditions; this is because low reactor power leads to a larger decrease in coolant temperature, which represents a disadvantageous boundary condition for subcooling and recriticality. With a completely filled primary circuit, the transient was started by completely opening a valve (representing a break) in the main steam line of SG 1 and the coastdown of the reactor coolant pumps (due to the MSLB). The cross-section of the opening was chosen to represent the transient conditions of a 10% break (corresponding to the most disadvantageous break size as determined by preparatory RELAP 5 analyses of the subcooling). It is assumed that the main steam line break is located inside containment, and therefore cannot be isolated (Fig. 2). The other SGs are isolated from the break (there is no connection through the turbine bypass, MS isolation valves are closed). Due to the limiting maximum pressure of the PKL test facility, the processes that normally occur at higher pressures were represented at a reduced pressure of 45 bar. For the extrapolation to real plant conditions additional calculations with thermal hydraulic system codes (after validation by the experiment) are necessary.

An additional, important aspect of this accident scenario concerns RPV integrity under consideration of pressurized thermal shock (PTS) due to the introduction of cold water in the RPV downcomer. This is important above all when the cooling of the primary coolant is intensified by injection of emergency cooling water into the cold leg at high primary-side pressure (up to the actuation pressure of the pressurizer safety valve). This case is not relevant for German PWRs but for a few PWR plants in other countries and was investigated in the second phase of the PKL test described here, that is, after the affected SG was completely emptied. In this process, the primary-side pressure was increased by injection from the safety-injection pumps (SIPs, cold-side injection in 2 of 4 loops, Loop 1 and 4) until the pressurizer safety valve (PRZ SV) responded. Steam flow out of the PRZ SV was followed later by water flow. Earlier computer calculations of this scenario indicate that, under certain boundary conditions, a reduction and partial stagnation, or even a reversal, of the natural circulation flow, can occur in the primary side loops with the intact steam generators.

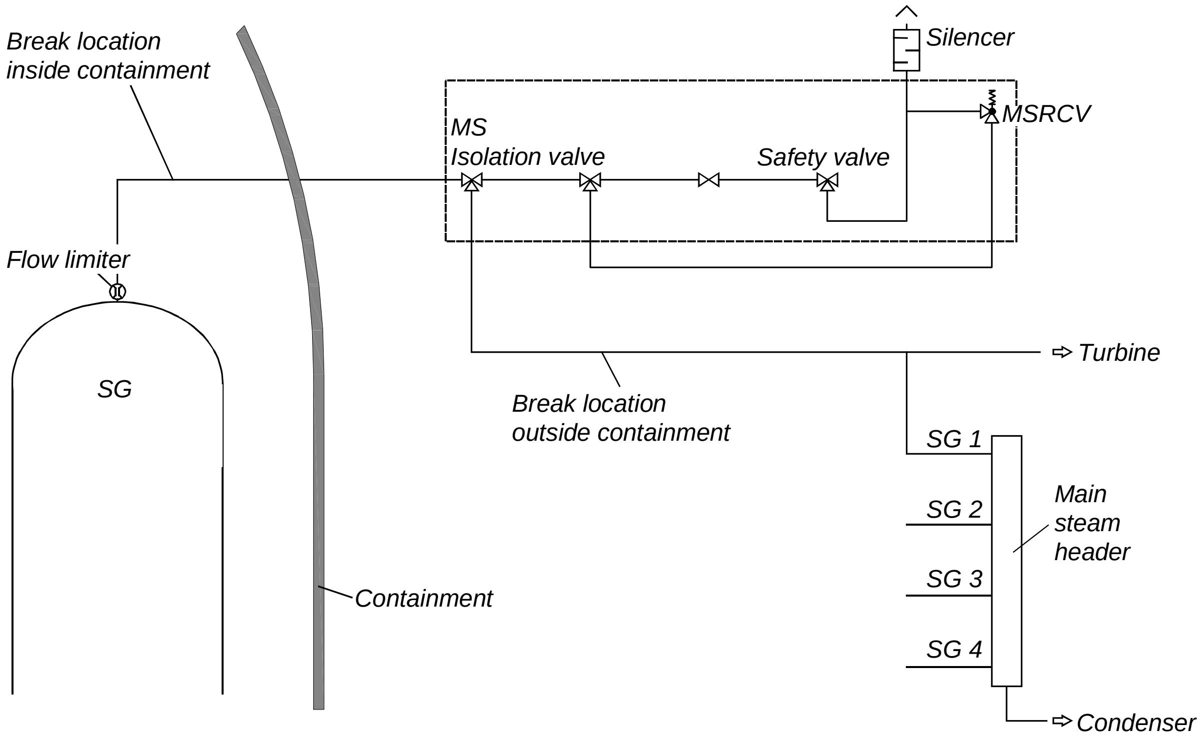


Figure 2: Possible break locations

A general goal of the test was to create a reliable database for validating computer programs. In view of the test goals (concerning PTS and recriticality) the following parameters are of decided importance:

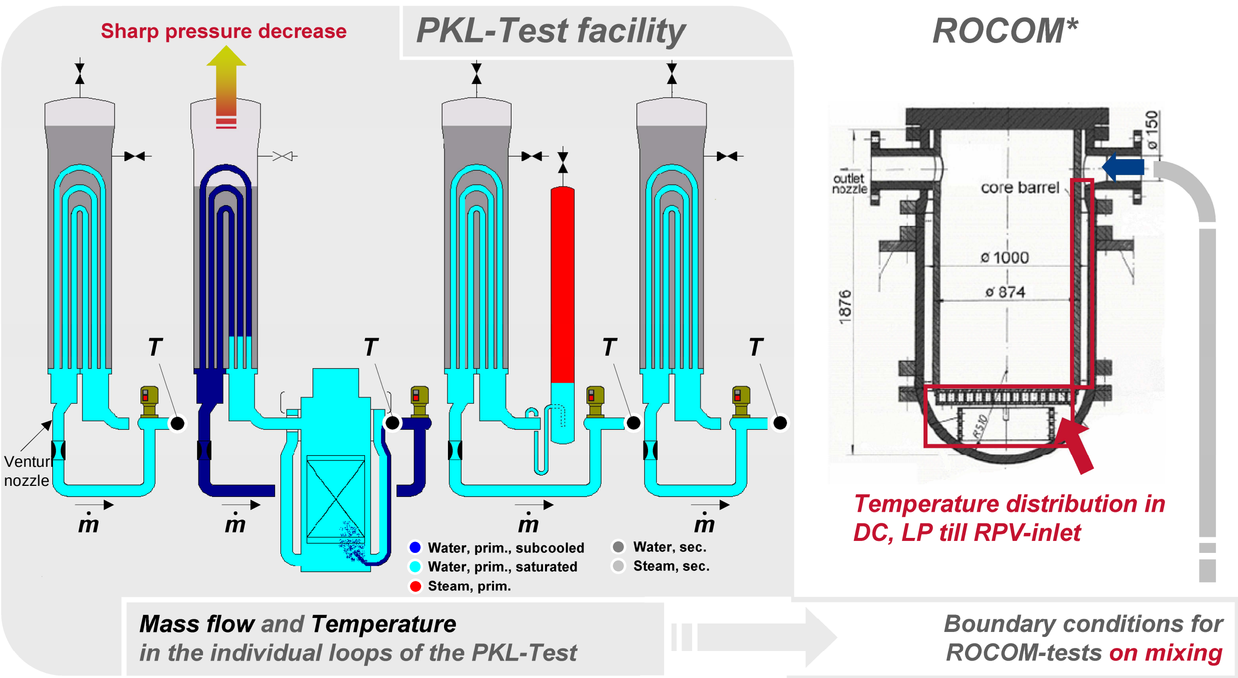
- The heat transfer in the affected SG and the determination of the minimum coolant temperature at the SG outlet and at the RPV inlet
- The circulation flow in the loops with the affected and the intact steam generators

Furthermore, the results of this PKL test, which is oriented to PWR system behavior, also provide the boundary conditions for complementary tests in the ROCOM facility[2] ([2], Fig. 3) on mixing cold and hot water in the RPV downcomer as well as in the lower plenum, and for determining the fluid state at the core inlet.

1.3 PKL Test Results

With the occurrence of the MSLB, the coolant on the secondary side of the affected SG started to boil off (the secondary side of the remaining SGs were isolated from the break and the main steam isolation valves were isolated). Steam flowed out of the secondary system through the break location. As a result of the loss of coolant, the pressure, level and coolant temperature in the affected SG sank. The SG inventory evaporated completely within about 1000 s; this lead to increased heat removal from the primary side to the secondary side in the affected loop until a fill-level of about 10% of the initial value was reached.

2 Rossendorf Coolant Mixing Test Facility, operated in the research center at Dresden-Rossendorf



* Rossendorf Coolant Mixing Test facility, operated in HZDR

Figure 3: Main Steam Line Break in the PKL Test Facility

Because of the increased heat transfer, a temperature decrease (subcooling transient) occurred on the primary side at the SG outlet and the RPV inlet of the affected loop. The minimum temperature at the RPV inlet was reached approximately 600 s after the beginning of the accident (Fig. 4).

With the shutdown of the RCPs (the coastdown was simulated), a transition from forced flow to natural circulation took place on the primary side. Because of the larger energy removal (driving force) in Loop 1, a larger mass flow arose than in the other loops. In the unaffected loops, however, there was also a stable circulation flow at a very low value, it was lower than in the affected loop because the SGs in the unaffected loops were acting as energy sources (counter drive against natural circulation). In both phases of the test (first phase: subcooling transient during emptying of the secondary side; second phase: cold side injection of emergency cooling water), the driving impulse from the RPV (temperature difference between downcomer and core region resulting in a positive pressure difference between the RPV outlet and inlet) apparently is enough to maintain the circulation in all loops.

The second phase of this test run was performed directly following the first phase. As the first action of this phase, the safety injection pumps (SIPs) were turned on in Loops 1 and 4 (cold side) and cold emergency cooling water being injected into the RPV. The primary side pressure increased and the pressurizer filled with coolant (the PRZ steam cushion was compressed). The injection lead to a pronounced temperature stratification over the height of the cold legs in the injection loops with rather distinctive temperature oscillations. This was especially observed in Loop 4, because the circulation mass flow was smaller and thus less mixing could take place with the warmer water from the natural circulation.

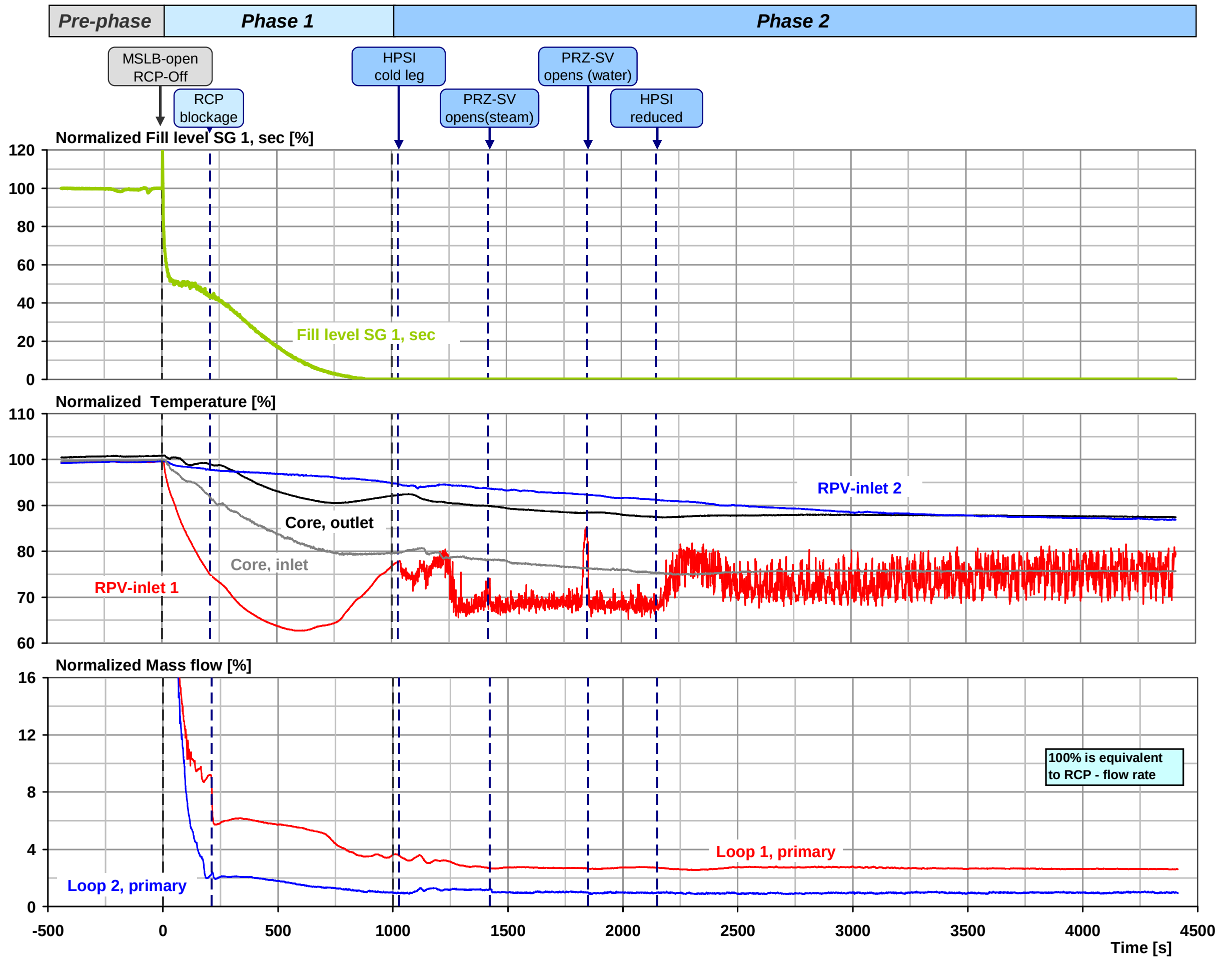


Figure 4: PKL Test results

In Loop 1 (the loop with the main steam line break), the temperature stratification was less. Furthermore, in both injection loops, no reverse flow of cold emergency cooling water flowed through the RCPs into the pump loop seals (Fig. 5). As already done for previous PKL experiments with stratification in the cold legs during ECC injection, such test results can serve as data base for CFD codes, [5]. If qualified against experiments (ideally including results from 1:1 scaled experiments like UPTF), CFD can then be used to transfer experimental results to real PWRs.

At 1420 s after start of test, the primary pressure had increased so much that the PRZ SV was actuated. Steam could escape from the primary system through the PRZ SV, the primary pressure decreased again, and the PRZ SV closed again (approx. 50 s after opening). However, a small amount of steam could still escape through a small leak in the PRZ SV, so that, in spite of the injection with the SIPs and the associated filling of the PRZ, the primary pressure decreased further. This small leak leading to a discharge flow of about 0.01 kg/s in the PKL test facility (or 1.45 kg/s for a PWR) was not intended. However it did not remarkably influence the further course of the test. In a PWR a small leak after actuation and closing of the PRZ SV can also not be excluded. It is, however not possible to quantify its amount.

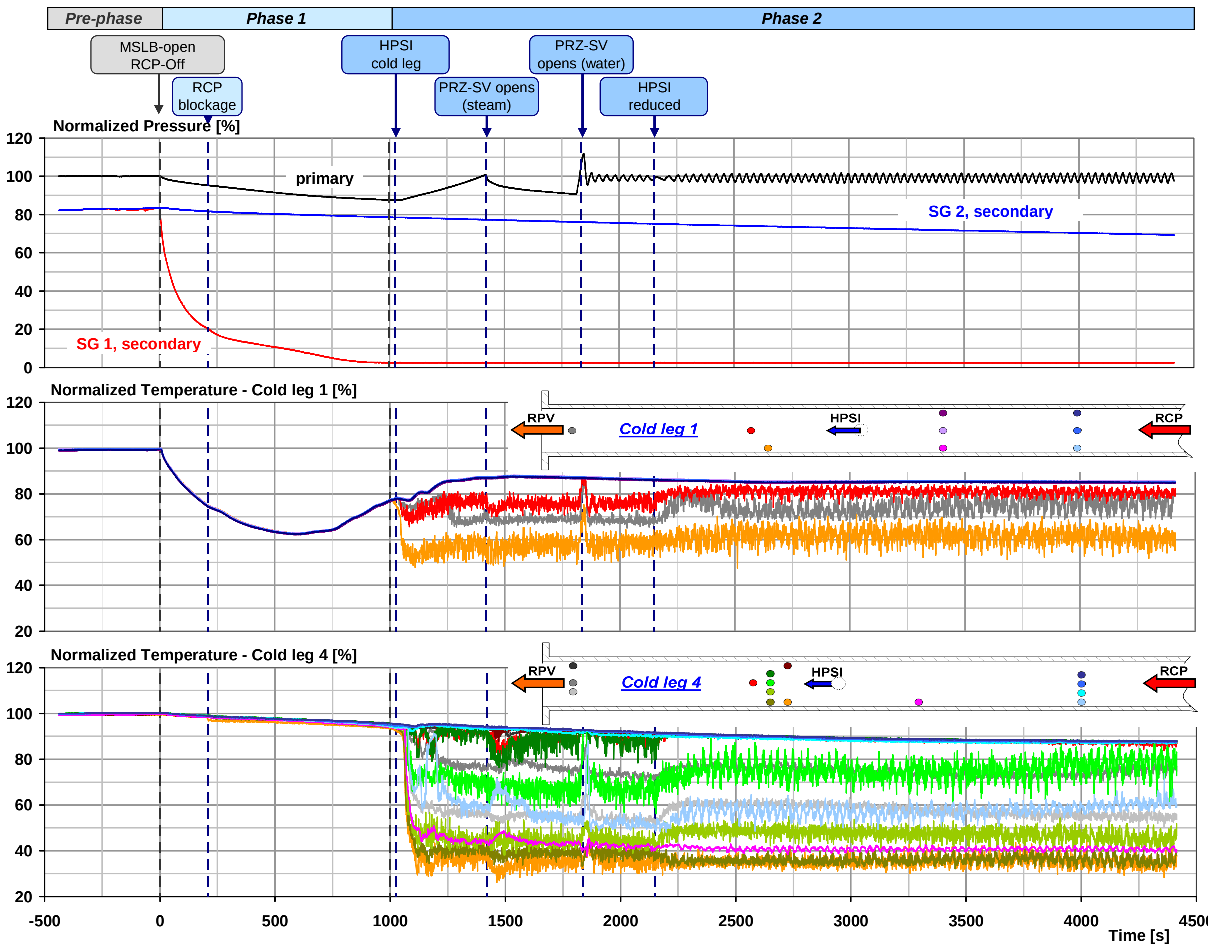


Figure 5: Temperature stratification at HPSI injection

After the pressurizer had completely filled with water and no more steam existed, a rapid primary-side pressure increase occurred, and therefore the PRZ SV opened again. Coolant flowed out of the primary system through the valve and the primary side pressure increase was stopped. The PRZ SV remained open until the end of the test and regulated the primary pressure. In a PWR, the primary system pressure increase associated with injection would lead to a reduction in the injection rate. To represent this process, the injection in each loop was reduced at 2150 s after start of test. In this phase the core power was removed under almost steady state conditions by injecting cold ECC-water into the cold legs and discharge of the same amount of hot water through the PRZ SV. The main flow path is from the cold legs via the RPV downcomer, the reactor core, the hot leg and the surgeline to the PRZ. By this way about 80 % of the power input in the primary system (decay power and heat transfer from the SG) is removed via the PRZ, the remaining part was discharged due to heat losses.

The measurement errors of the temperature (thermo couples, position see Fig. 5), the pressure and the fill level are about 1 % of the measured values. The mass flows in the individual loops were measured by venturi nozzles (position see Fig. 3). The error depends on the measurement range with increasing uncertainties at low flow rates; values above 2% of the RCP mass flows (as illustrated in Fig. 4 for loop 1) can be measured with an error of 0.5 %. Lower mass flow rates lead to an higher error. For example the measured value for loop 2 (about 1% of RCP mass flow) is close to the lower measurement limit and can only be interpreted in qualitative terms. It indicates that the mass flow still exists, however at a low flow rate. The existence of a flow could be confirmed by other measurements (temperatures and flow direction indicators).

Due to the design features (4-Loop configuration, original elevations and friction pressure losses), the PKL test facility is well suitable to simulate the thermal hydraulic phenomena of a MSLB, thus the PKL test results can be qualitatively extrapolated to the PWR. Because of pressure scaling and other scaling affects (e.g. diameter, heat structures), the quantitative extrapolation is however only possible with the help of validated computer codes, i.e. by code calculations of the real plant scenario (using real plant geometry) with a computer code validated on the basis of post test calculations of the presented experiment.

2. ROCOM-Test

As already mentioned the PKL test provide the boundary conditions for complementary tests in the ROCOM facility on mixing cold and hot water in the RPV downcomer as well as in the lower plenum, and for determining the fluid state at the core inlet.

2.1 ROCOM Test Facility

ROCOM is a four-loop test facility (Fig. 6) for the investigation of coolant mixing operated with water at room temperature. The facility models a German KONVOI-type reactor with all details important for the coolant mixing along the flow path, from the cold-leg nozzles up to the core inlet, at a linear scale of 1:5. Special attention was given to components, which significantly influence the velocity field, such as the core barrel with the lower core support plate and core simulator, perforated drum in the lower plenum, and inlet and outlet nozzles. Individually controllable pumps in each loop give the possibility to perform tests over a wide range of flow conditions, from natural circulation to nominal flow rate, including flow ramps (pump and natural circulation start-up). Salt water or brine is used to alter the local electrical conductivity of the fluid in order to label a specific volume of water and thus simulate an under-borated or overcooled slug of coolant. The distribution of this tracer in the test facility is measured by special wire-mesh electrical conductivity sensors developed by the Helmholtz-Zentrum Dresden-Rossendorf (HZDR), which allow a high-resolution measurement of the transient tracer concentration with regard to space and time. These wire mesh sensors consist of two planes of electrodes, where the mesh spans the flow cross-section. The measurement of the instantaneous local conductivity of the medium is realized in the vicinity of each crossing point of two perpendicular wires. These measured local conductivities, which can be recorded with a frequency of up to 1000 Hz and are subsequently compared to reference values in order to estimate the position of the tracered water and its transport. The result is a dimensionless mixing scalar $\Theta x,y,z(t)$ that characterizes the instantaneous share of the coolant originating from the labeled volume (deborated or overcooled water) at a given position inside the flow field. It is calculated by relating the local instantaneous conductivity $\Theta x,y,z(t)$ to the amplitude of the conductivity change at the reference position σ_1 (usually the labeled slug in the cold leg) according to the following formula:

$$\Theta_{x,y,z}(t) = \frac{\sigma_{x,y,z}(t) - \sigma_0}{\sigma_1 - \sigma_0} \quad (1)$$

The lower reference value σ_0 is the initial conductivity of the water in the test facility before the experiment is started. Two sensors were used to observe the mixing in the test facility in the below described experiment. One sensor is integrated into the lower core support plate providing one measurement position at the entry into each fuel assembly (Fig. 7). The second sensor spans a measuring grid of 64 azimuthal, 29 axial positions over the height and two over the width of the downcomer. The measurement error of the wire mesh sensor was assessed with up to 3.5 % [2].

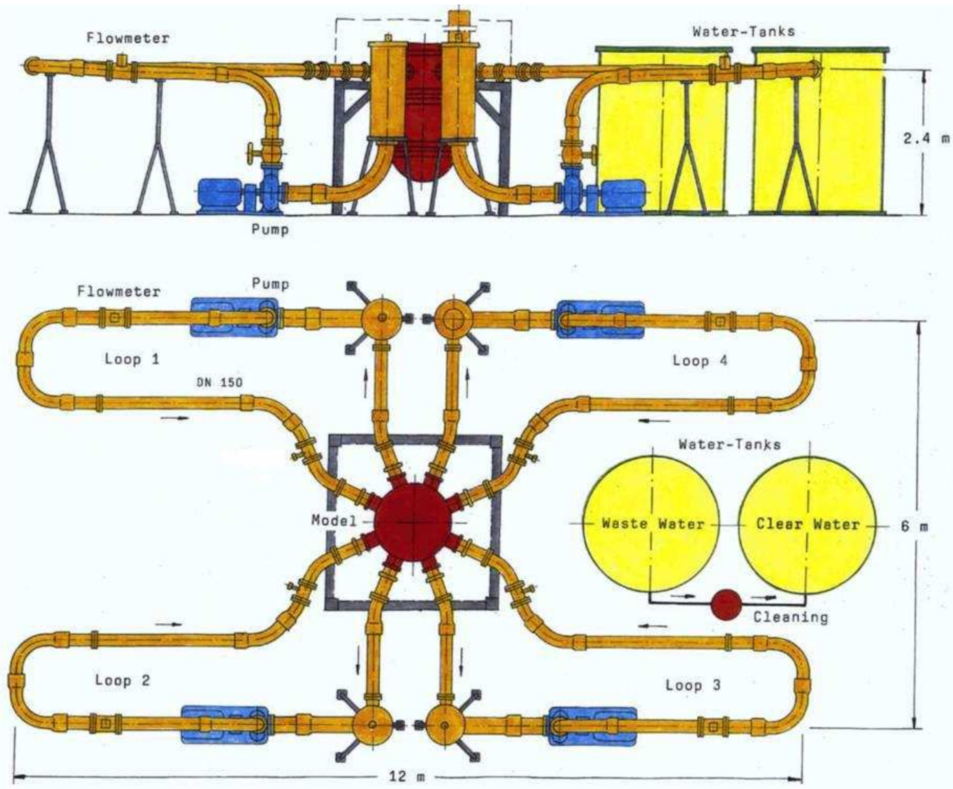
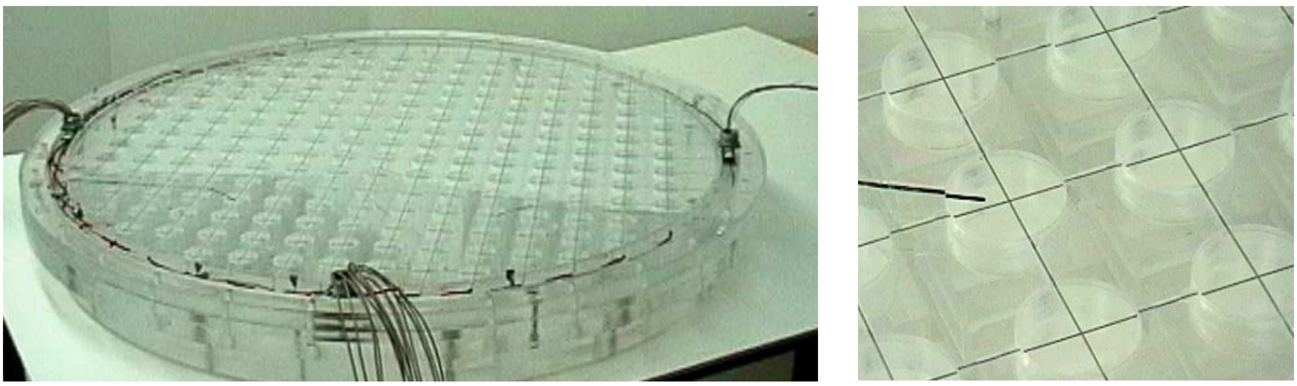


Figure 6: Scheme of the ROCOM facility



a) total view

b) electrodes

Figure 7: Lower core support plate with integrated wire mesh sensor

2.2 Boundary conditions

The above mentioned scaling factor of 1:5 must also be adopted in the application of the boundary conditions in order to replicate phenomena on the reactor scale. The density differences between the different coolants in the facility (reactor) play an important role for the phenomena in this test. The Froude number Fr is the dimensionless similarity number, which can be used to characterize buoyant single-phase flows:

$$Fr = \sqrt{\frac{\rho \cdot w^2}{\Delta\rho \cdot g \cdot L}} \qquad (2)$$

w is the velocity; g is the gravitational acceleration; ρ is the density; Δρ is the density difference; L is the characteristic length.

The Froude number represents the ratio of the inertia forces (numerator) and the influence of the gravitational force (denominator). The influence of the inertia in flowing media is characterized by the density, the influence of the gravity by the density difference. That means that the ratio Δρ/ρ is the key parameter in determining characteristics of the Froude number. The boundary conditions should be selected in such a way that the Froude number in the ROCOM experiment is identical to the Froude number under reactor conditions. In the current experiment the similarity of the Froude number was achieved by using the same density difference as under reactor (PKL) conditions and scaling the velocity determined for reactor conditions down by a factor of $\sqrt{5}$.

A quasi-stationary mixing experiment was carried out using the data from the PKL-test at the time point of minimum temperature in loop 1 during the overcooling phase (t = 609 s, phase 1). As shown in Fig. 5, homogenous temperature distribution was observed over the hight of the cold legs in all loops throughout phase 2 of the PKL experiment. The goal of the ROCO experiment was the determination of the coolant temperature distribution inside the reactor pressure vessel. The data can be used to assess the possibility of re-criticality of the core and the thermal loading onto the pressure vessel wall.

Tab. 1 contains the normalized values for temperature, mass flow rate and density derived from the PKL experiment and used for boundary conditions for the ROCOM-test.

Loop	**1**	**2-4 (Average)**
Normalized Temperature [%]	63	97
Normalized mass flow rate [%]	5.46	1.41
Relative density [-]	1.12	1.00

Table 1: Conditions in PKL-test at t = 609 s

2.3 Test results

Fig. 8 shows the normalized temperature distribution in the downcomer (near the RPV wall) and in the core inlet plane. The value of 100 % corresponds to the temperature of the unperturbed loops, 0 to the temperature of the overcooled loop. As can be seen the overcooled coolant from loop 1 enters the downcomer as a stripe. In axial direction a transition region is formed. The position of this region is determined by the ratio of the loop flow rates. In the lower part of the downcomer as well as in the core inlet plane a nearly uniform temperature distribution is observed. The temperature difference over the core inlet is less than 5 K. The density difference between the coolant from different loops is responsible for the nearly homogeneous distribution. It differs from experiments with similar flow rates but without density differences. In those experiments [2] a sector formation at the core inlet can be observed.

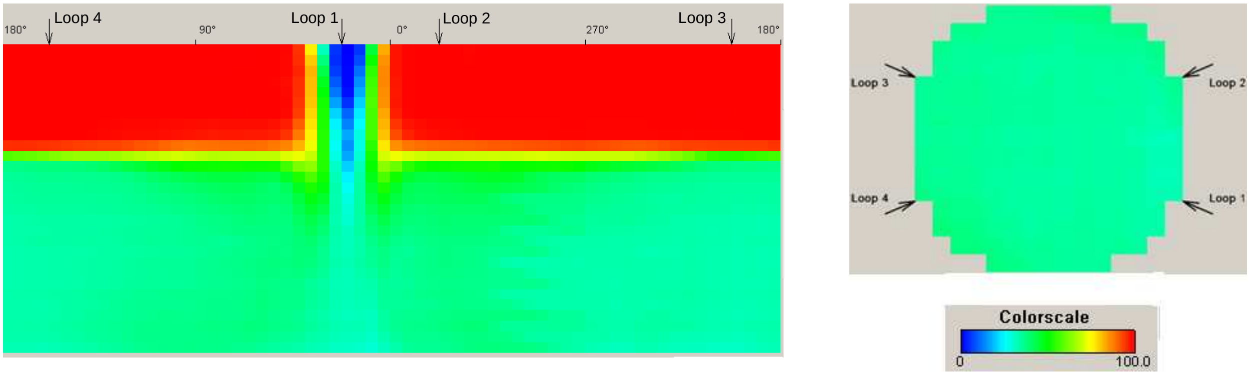


Figure 8: Temperature distribution in the downcomer (unwrapped (left)) and core inlet plane (right)

3. Conclusions and Outlook

Within the current OECD-PKL 2 test program, complementary experiments on main steam line break have been carried out in the PKL and ROCOM test facilities. Background scenario of these experiments was a 10% (non-isolable) main steam line break occurring under hot stand-by conditions.

In the PKL facility, the overall thermal hydraulic system behavior was investigated. The results of the PKL test (that is, the measured mass flows and temperatures at the RPV inlet in the individual loops) provided the boundary conditions for complementary tests on mixing cold and hot water in the RPV downcomer and in the lower plenum in the ROCOM test facility.

The results provide an important database for validation of computer programs in regard to recriticality and PTS, important results are:

PKL-test:

- Evaporation of affected SG-inventory within approx. 1000 s
- Temperature decrease at RPV-inlet in the affected loop down to approx. 60 % (in relation to the initial conditions)

- Intense natural circulation in the affected loop, also stable natural circulation (at lower intensity) in the unaffected loops

- Pronounced temperature stratification in the cold leg as a consequence of HPSI injection, thereby no backflow of cold water to RCPs

ROCOM-test:

- Complete mixing of hot and cold water already in the downcomer

- Temperature difference over the core inlet is less than 5 K

The combination of the PKL and of the ROCOM experiment covers all thermal hydraulic phenomena relevant for the MSLB scenario. The test results have been intensively used for validation and optimization of analytical tools, that is for system codes in connection with PKL [4] and for CFD in connection with ROCOM. The discussion of pre- and post-test calculations on PKL and ROCOM performed by the project partners was one major part of an analytical workshop hosted by University of Pisa. In addition a benchmark activity among the participating project partners on the PKL experiment including pre- and post test calculations with all relevant system codes currently used have been also co-ordinated by University of Pisa, [3].

The final goal now is to be able to make accurate predictions for PWRs with regard to recriticality and PTS for relevant scenarios by using the computer programs that have been validated in this way for plant calculations with PWR geometry and PWR boundary conditions.

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