INTEGRAL EFFECT TESTS IN THE PKL FACILITY WITH INTERNATIONAL PARTICIPATION

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For over 30 years, investigations of the thermohydraulic behavior of pressurized-water reactors under accident conditions have been carried out in the PKL test facility at AREVA NP in Erlangen, Germany. The PKL facility models the entire primary side and significant parts of the secondary side of a of pressurized water reactor at a height scale of 1:1. Volumes, power ratings and mass flows are scaled with a ratio of 1:145. The experimental facility consists of four primary loops with circulation pumps and steam generators (SGs) arranged symmetrically around the reactor pressure vessel (RPV). The investigations carried out encompass a very broad spectrum from accident scenario simulations with large, medium, and small breaks, over the investigation of shutdown procedures after a wide variety of accidents, to the systematic investigation of complex thermohydraulic phenomena.

The PKL tests began in the mid 1970s with the support of the German Research Ministry. Since the mid 1980s, the project has also been significantly supported by the German PWR operators. Since 2001, 25 partner organizations from 15 countries have taken part in the PKL investigations with the support and mediation of the OECD/ NEA (Nuclear Energy Agency).

After an overview of PKL history and a short description of the facility, this paper focuses on the investigations carried out since the beginning of the international cooperation, and shows, by means of some examples, what insights can be derived from the tests.

KEYWORDS : PKL, Thermalhydraulics, PWR, Accident Scenarios, Experiments, OECD-PKL Project

1. INTRODUCTION

Since the commencement of experiments at the PKL Test Facility more than 30 years ago, the various phases of the experiments have always reflected on and given priority to current safety issues. Reactor safety research in the seventies centered above all on the theoretical and experimental analysis of large-break (LB) loss of coolant accidents (LOCAs), focusing on verifying the effectiveness of the ECCS required to control such events. Based on this original objective, the PKL Test Facility was constructed at Siemens/KWU (now AREVA NP) with the support of the German Ministry for Research and Technology and placed in service in 1977.

The LB-LOCA experiments were interrupted in 1979 in the wake of the accident at Three Mile Island Unit 2 (TMI-2) for the performance of experiments at the PKL Test Facility de-signed to contribute to gaining information as quickly as possible on issues raised by this event. Consequently, only a short time after the accident it was possible to simulate small-break (SB) LOCA event sequences, which also included the TMI-2 scenario, at the PKL Test Facility. The in-vestigations, focused on demonstrating the safety margins of the operating units through the ex-perimental verification of the effectiveness of the engineered safety features in the event of large- and small-break LOCAs, were covered within the test programs PKL I and II [1,2].

The subject of the subsequent PKL III program, started in 1986, has been the investigation of so-called accident transients with and without LOCAs. While the first test series within PKL III covered design-basis accidents and cool-down procedures detailed in the operating manual, the main interest was then focused on beyond-design-basis accidents and the experimental verification of Accident-Management (AM) procedures [3]. Typical topics of investigation studied within the test series PKL III A to PKL III D were:

- Cool-down procedures with and without reactor coolant pumps under symmetric and asymmetric boundary conditions (e.g. one or more steam generators isolated on the secondary)
- Cool-down procedures following small break LOCAs or steam generator (SG) tube ruptures, partly in

combination with additional system failures
- Accident management (AM) procedures (e.g. secondary or primary side bleed-and-feed) following total loss of feed water or multiple failure situations
- Failure of residual heat removal system under cold shut-down conditions
- Systematic investigations within parametric studies such as:
  - Single and two-phase flow, reflux condensation, counter current flow limitation
  - Influence of non-condensable gas on heat removal from the primary to the secondary

Since 2001, the PKL project has been continued in the course of an international project initiated by the OECD. The major topics covered by the experiments between 2001 and 2007 were:
- Boron dilution events following SB-LOCA [4]
- Loss of residual heat removal under shut-down conditions [5]

Additional topical safety issues, such as main steam line break or boron precipitation processes in the core following LB-LOCA, are subject to investigation in the current OECD-PKL 2 project, which is scheduled to run until September 2011.

The overriding objective of all PKL experiments has been and remains the experimental investigation of thermal-hydraulic processes in PWRs with respect to the response both of the overall system and of individual components and subsystems during the simulation of operational transients and accidents. The tests performed to date (in total more than 150 integral experiments) 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. In addition, tests have supplied valuable information regarding safety margins available in the plants. The test results have also found concrete application in the validation and further development of thermal hydraulic computer codes, so-called system codes. Tests have also provided numerous initial and boundary conditions for further detailed investigations in special experimental facilities, as, for example, in the ROCOM facility [6], which is operated at the Dresden-Rossendorf research center in Germany, or the JULIETTE facility at AREVA NP in Le Creusot, France. Mention should also be made at this point of the upper plenum test facility (UPTF [7]), which was also operated by Siemens/KWU (now AREVA NP) in Germany and has now been dismantled. The results of the tests performed at the UPTF have proven to be an excellent complement for the PKL experiments. Whereas the UPTF provided a full-scale model of large parts of the reactor coolant system (RCS) for investigating the response of plant subsystems and in particular for studying individual phenomena, which are highly dependent on the geometry, the PKL project concentrated on studying overall system response.

2. DESCRIPTION OF THE TEST FACILITY

The layout of the PKL-III facility (Fig. 1) is based on

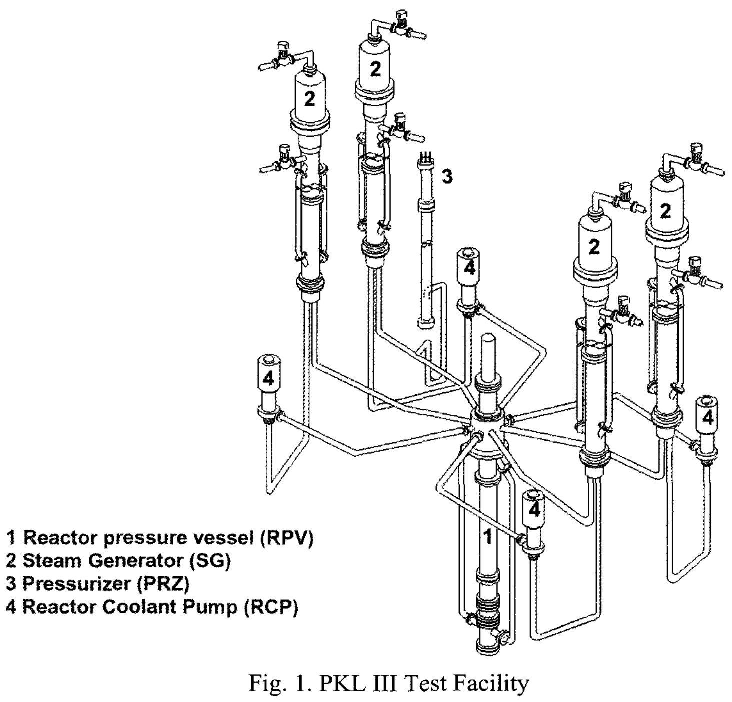

Fig. 1. PKL III Test Facility

the "Vorkonvoi" type (4-Loop, 1300 MWe) of KWU pressurized water reactor, with the Philippsburg 2 nuclear power plant serving as the reference plant. The entire primary side and the most significant components of the secondary side (excluding turbines and condenser), including the appropriate system technology, is represented. Because the essential construction principles of the western types of PWRs are similar, it is possible to make statements concerning the behavior of other companies' plants. In any case, the analysis of plant-specific reactor transients must then be made with the help of computer codes.

Following the scaling concept, all geodetic heights are represented in a 1:1 ratio. The entire volume of the primary side and, as far as possible, the partition of the individual volumes correspond to a scaling factor of 1:145, which corresponds to a hydraulic diameter reduction of 1/12. For some components, the exact volume scaling was not applied in order to simulate certain thermohydraulic phenomena, for example CCFL (Counter Current Flow Limitation) in the hot legs. This allowed dimensionless numbers (for example, the Froud number) to be maintained in the correct parameter range. The single-phase pressure losses correspond to a large extent to the values in a PWR. The core is modeled by a bundle of 314 electrically heated rods. The core geometry is, like the SG geometry, constructed as an "actual section;" that is, the individual heated rods and U-tubes have the actual geometry, but the number of heated rods in the core and the number of U-tubes in the SG are reduced by the scaling factor 1:145 (volume and power scaling), as compared to the original plant.

The representation of the primary side by four identical and symmetric loops arranged around the reactor pressure vessel (RPV) allows the realistic investigation of accidents, even with unequal boundary conditions in the individual loops. Through the representation of all significant interface and auxiliary system functions on the primary and secondary side, the system behavior as well as the interaction between the individual systems can be investigated under a wide range of accident conditions, and the effectiveness of the automatic or manually performed counter actions can be checked. For the realistic representation of the events in secondary-side bleed-and-feed operations, the complex geometry of the feed water system (all heights 1:1, volumes 1:145, pressure losses 1:1) was modeled with the possibility of setting the corresponding temperature distributions.

With approximately 1500 measurement points, the PKL facility is comprehensively instrumented. This allows detailed analysis and interpretation of the phenomena that develop in the course of the tests. Although certain constraints are placed on the application of insights gained from the test results to a PWR plant because of the primary pressure limit of 45 bar and because of the geometric scaling of the test facility, in most cases a qualitative application of the observed phenomena is possible because of the detailed scaling concept. To what extent the test results from other test facilities (for example, LSTF and BETHSY, pressure 1:1, or UPTF, geometry 1:1) [7,8] or results from computer programs (for example, ATHLET or RELAP) must be included, is considered for individual cases.

With regard to system technology and system design layout, the test facility is constantly being modified and expanded according to new project definitions and investigational emphases, and the applied measurement techniques are continually being updated to the state-of-the-art technology.

3. MAIN AREAS OF CONCENTRATION OF THE THREE INTERNATIONAL PROGRAMS OECD-SETH, OECD- PKL AND OECD- PKL2 SINCE 2001:

Since April 2001, the PKL project has proceeded as part of an international project initiated by the OECD. The involvement of the OECD, or more specifically of its suborganization the NEA, developed with the declared goal of ensuring the long term preservation of competence and sufficient infrastructure in reactor safety research in the international arena.

This very effective international cooperation began with the Test Program PKL III E (in co-operation with the Swiss PANDA project, internationally designated as the OECD - SETH test program), and was followed by Test Program PKL III F (specifically OECD - PKL) in January 2004; this test regimen has been successfully continued by the current test program PKL III G (OECD - PKL 2) from April 2008 on. The topics emphasized in Test Programs PKL III F and E were investigations into the following:

- Boron dilution after primary side small break accidents, and
- Loss of residual heat removal under cold shut down conditions (Mid-Loop Operation)

The current test program PKL III G, which will last until September 2011, additionally addresses the following main topics:

- Subcooling transients after a break in the main steam line, (with supplementary tests in the ROCOM test facility for mixing in the RPV downcomer and in the lower plenum)
- Boron enrichment and precipitation in the core after primary side large break accidents
- Further systematic investigations of the heat transfer in the SGs in the presence of a non-condensable gas (with complementary tests in the Hungarian PMK test facility for horizontal SGs)
- Accident transients with reflux condenser operation for new PWR concepts

The investigation fields listed above are oriented to current issues and are being addressed in various national and international committees. The exact establishment of test parameters takes place prior to the tests, in close agreement with the project participants.

4. INVESTIGATION RESULTS

As already explained, the investigations serve to address various aspects such as the resolution of current safety issues, the optimization of reactor-related plant procedures, and the validation and improvement of thermohydraulic system codes. This will be further discussed by means of some examples:

4.1 Review of Current Issues for the Example of Boron Mixing for Small-break Accidents

The so-called inherent boron dilution can happen in a PWR for a small break on the primary side and when only some of the safety injection pumps (SIP) function as designed [9]. Under these conditions, the flow through the break at high pressure is greater than the injection rate of the SIP, and the primary coolant inventory is reduced. Energy from the core is temporarily transferred to the SG operating in reflux condenser mode. Boron reaches the steam phase only in small amounts. Thus, the condensate produced in the steam generators has considerably lower amounts of boron. This condensate can accumulate in certain locations in the primary loop, especially in the pump loop seals (Fig. 2, left hand). At lower pressure, the injection rate of the safety injection system exceeds the rate of flow through the break and the primary loop is filled up again. If natural circulation establishes itself, the less-borated masses of water (illustrated in light grey in the Fig. 2) can be transported toward the core. If insufficient mixing with borated water occurs, and the natural circulation in all loops happens simultaneously, a localized criticality in the reactor core cannot be ruled out.

Such a criticality depends strongly on the following boundary conditions:

- Size of the "condensate slugs" that develop
- Mixing in the SGs and during slug transport through loops with more highly borated water
- Flow rate of the natural circulation onset transient
- Difference in time between the start of natural circulation in the different loops
- Mixing with more highly borated water in the RPV annulus and the lower plenum

The first four points are the subjects of investigation in the PKL facility [4,6]. All the PKL boron dilution tests were performed with actual boric acid and with adequate measurement technology (continuous online measurement and probe samples) for determining boron concentration. Comprehensive investigations of mixing in the RPV downcomer and in the lower plenum were undertaken in the ROCOM test facility under boundary conditions tailored specially for the relevant accident scenarios [6]. For these tests, the results for natural circulation in the individual loops and for the conditions in the weakly-borated slugs (size, boron concentration distribution) at the entrance to the RPV obtained from PKL tests serve as boundary conditions.

In the PKL Test E 2.2, a break and cold-side injection with asymmetrical injection from only 2 SIPs simulates a possible failure configuration at some German PWR plants. The test's boundary conditions were purposely chosen to be conservative, with a very long operation in reflux condenser mode to obtain the maximum possible volume of accumulated condensate masses. However, although many times more condensate was produced than was predicted to develop in a PWR up to a primary pressure of 40 bar, the accumulated condensate slugs were markedly

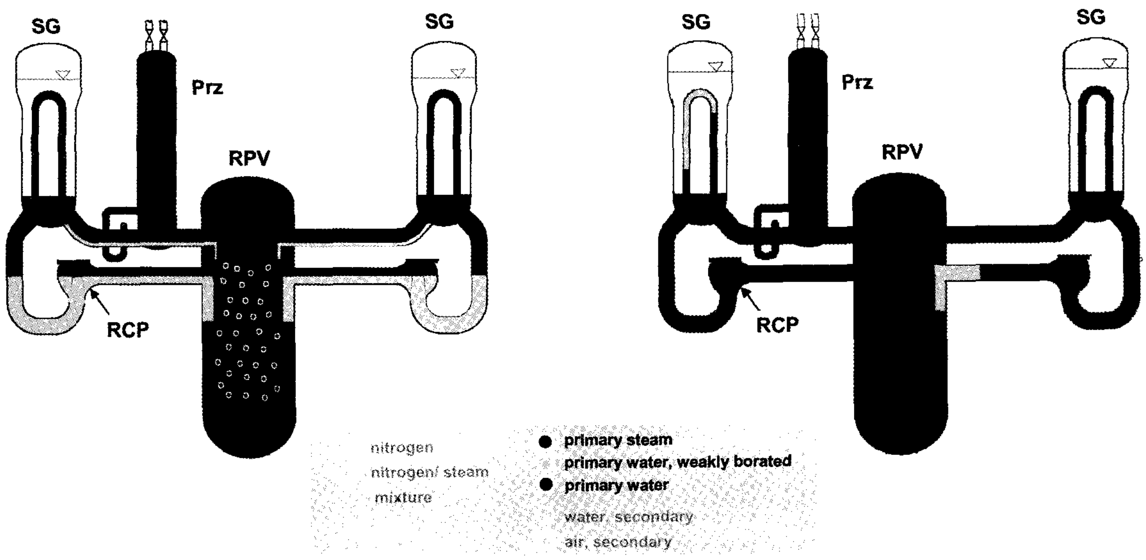

Fig. 2. Accumulation of Weakly Borated Slugs after Small-break Accidents (Schematic)

smaller than had been previously assumed. There are two reasons for this:

Through mixing processes resulting from the steam flow from the core to the steam generators, no weakly-borated water masses can remain unmixed during the filling processes in the SG inlet chambers and in the hot legs. In contrast to the results from thermohydraulic analyses, in the test, flow through individual U-tubes occurs as soon as the SG outlet chambers were filled to the lower edge of the tube sheet with coolant. Through this phenomenon, borated water is shifted from the SG inlet side to the SG outlet side, which in turn disturbs or hinders the collection of less-borated water in the outlet chambers and pump loop seals. Thus, on the cold side, condensate only accumulates when the SG outlet side is emptied; that is, when the liquid level lies clearly below the tube sheet. A further important result is that natural circulation arises in the different loops at different points in time (Fig. 3). Circulation arises first in the two loops that have no injection, where, furthermore, a noticeable time offset may also be observed between these two loops. As a consequence of this, the weakly-borated slugs from the two non-injected loops reach the RPV noticeably offset in time from each other. Through mixing with more highly-borated water during the refill phase and after the onset of circulation, a "dispersion" of the condensate slugs originally accumulated in the pump loop seals occurred. This means that the minimal boron concentration in the slugs at the entrance to the RPV after the onset of circulation in the non-injected loops was significantly higher than in the slugs that originally accumulated in the pump loop seals. In the loops with emergency cooling water injection - as has been observed in all other tests - no significant decrease in boron concentration appeared at the RPV inlet.

The PKL Test F1.1 was performed with symmetrical injection in all four loops (on the cold side), which corresponds to a typical injection configuration for a Framatome or a Westinghouse plant (Fig. 4). This means that in spite of the availability of only one (out of two) injection pumps, through injection of emergency cooling water (ECW) through a common header, all loops can be charged; however, the loops receive decidedly smaller injection rates. For this symmetrical injection configuration, the results obtained above were also confirmed: namely, no simultaneous circulation onset and a relatively low decrease in the boron concentration at the RPV inlet.

Systematic investigations under quasi-steady-state boundary conditions as, for example, in Test F4.1, confirm the findings of the assessment of small-break accident scenarios: the buildup of weakly-borated condensate in the pump loop seals can only occur when the primary side liquid level in the outlet side of the SGs lies below

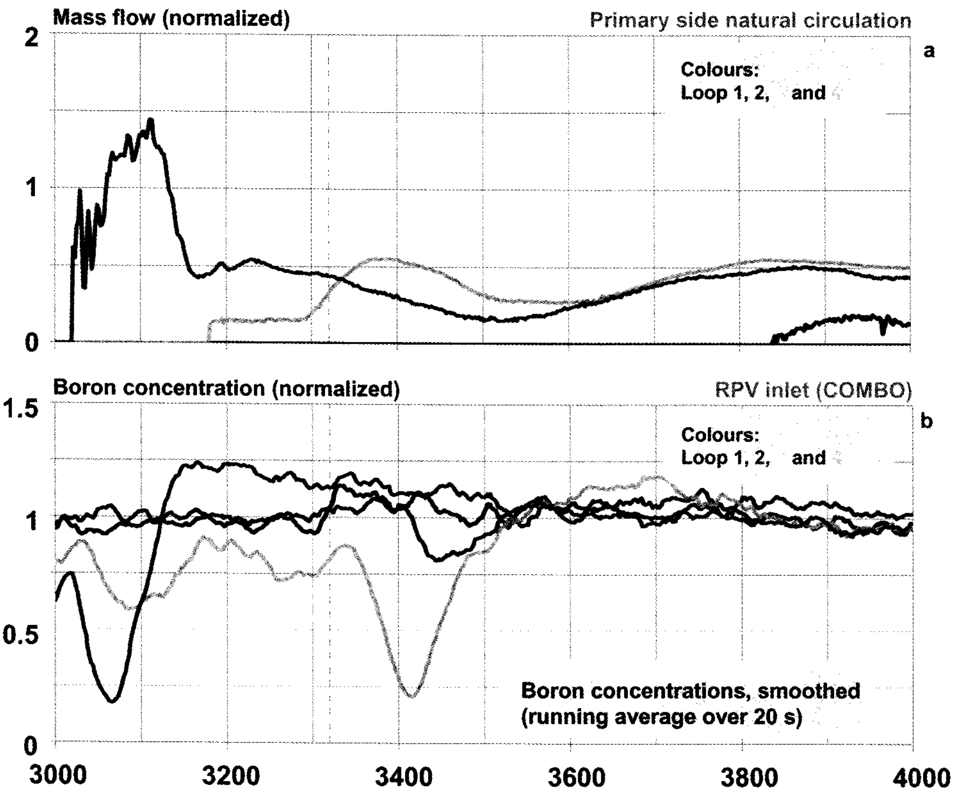

Fig. 3. Transport of Weakly-borated Slugs at the Onset of Circulation in the Refill Phase

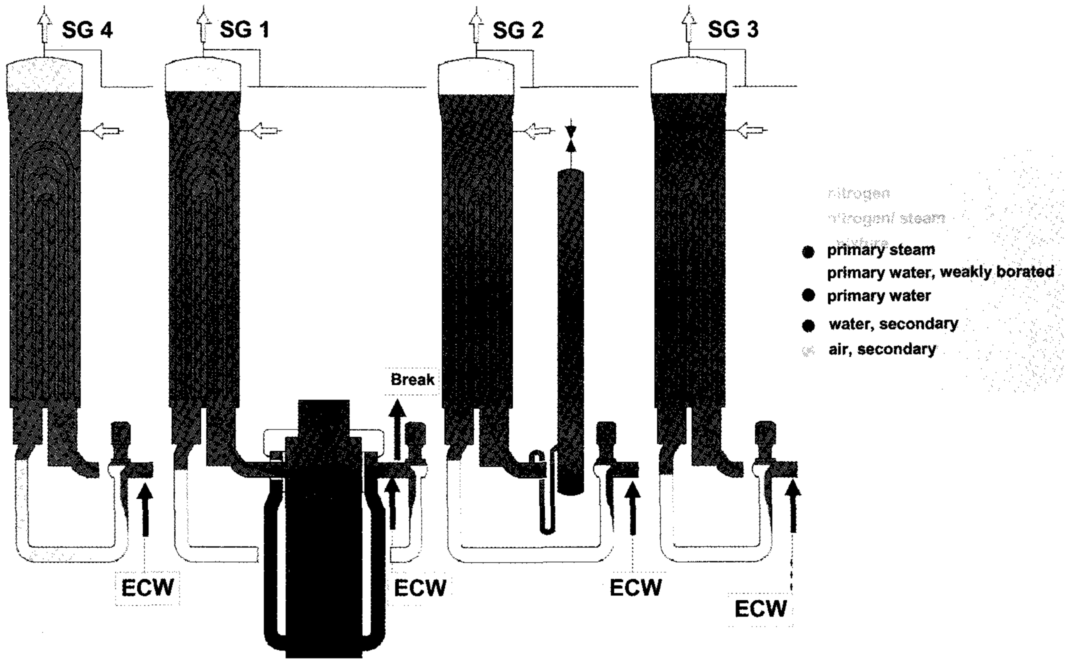

Fig. 4. PKL III F1.1 Condensate Distribution/Boron Concentrations Shortly after Start of Test

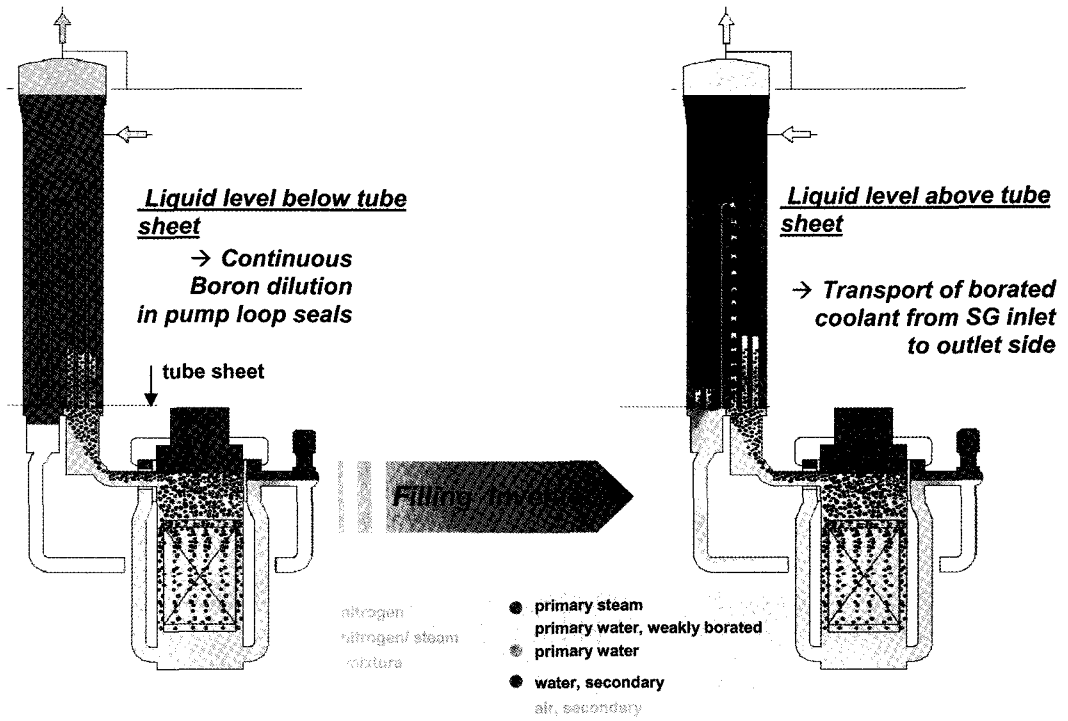

Fig. 5. Initiation of RC Operation und Coolant Transport Dependent Upon Liquid Level in SG Outlet Chamber

the tube sheet (Fig. 5). Even small amounts of natural circulation (for example, in a not yet or no longer fully developed reflux condenser situation) that can hardly be detected by conventional measurement techniques produce transport processes of more highly borated water from the SG inlet side to the outlet side and prevent low boron concentrations in the pump loop seals.

The findings for Small-Break Accidents may be summarized as follows:

- For a large and long-lasting loss of coolant, weakly borated condensate slugs can build up in the pump loop seals.
- The size of these condensate accumulations are, however, significantly smaller than were determined in earlier analyses
- In the refill process, effective mixing processes are present in the SGs and the loops before onset of natural circulation.
- Simultaneous onset of natural circulation in the loops was not observed in any test.

Based on the PKL results, the mixing in the downcomer and in the lower plenum were investigated in the ROCOM facility. In spite of some assumed conservatism (for example, two loops initiating at the same time), these results yield a minimal boron concentration value noticeably greater than the critical concentration values at the core inlet.

4.2 Findings for Plant Operation for the Sample Case of Investigating Counter Measures for Loss of Residual Heat Removal in Mid-Loop Operation

In the test series up to now, several tests with loss of residual heat removal (RHR) accident scenarios have been carried out (Fig. 6), [5]. If a long-term failure of the residual heat removal system occurs while the primary side is still closed, the reactor water heats up to boiling temperature due to the core decay heat, creates steam, and expands correspondingly. For such a case, in a German PWR, the secondary side is maintained with at least one SG full of water and ready to operate. This SG takes over the primary side cooling and should thus avoid a continuous pressure increase. In this presentation, the findings related to the effectiveness of cooling through one or more SGs will be discussed (Fig. 7). These tests were also performed using actual boric acid and adequate measurement techniques for boron concentration.

With the availability of one operation-ready SG, a stable situation for heat transfer develops after several hours, when the primary side pressure has been stabilized. In this situation, the pressure and temperature difference to the secondary side, which is regulated at the minimum possible pressure, is sufficient with even a relatively small heat transfer surface. However, it was observed for the first time at PKL that in such a situation, a transfer of weakly borated water to the outlet side of the active SG can occur (Fig. 8). If this condition exists for a longer period of time, a not insignificant accumulation of weakly borated water in the stagnant coolant medium can develop in the pump loop seals and in the downcomer region. Because additional actions (such as accumulator injection or initiating low-pressure injection) are required to reestablish the RHR system start conditions (subcooling in the hot leg), fluid flow can occur, resulting in the undesirable presence of weakly-borated quantities of water. This risk should therefore be avoided by appropriate actions. As can be seen in Fig. 8, through the availability of 2 SGs and the smaller increase of primary-side inventory

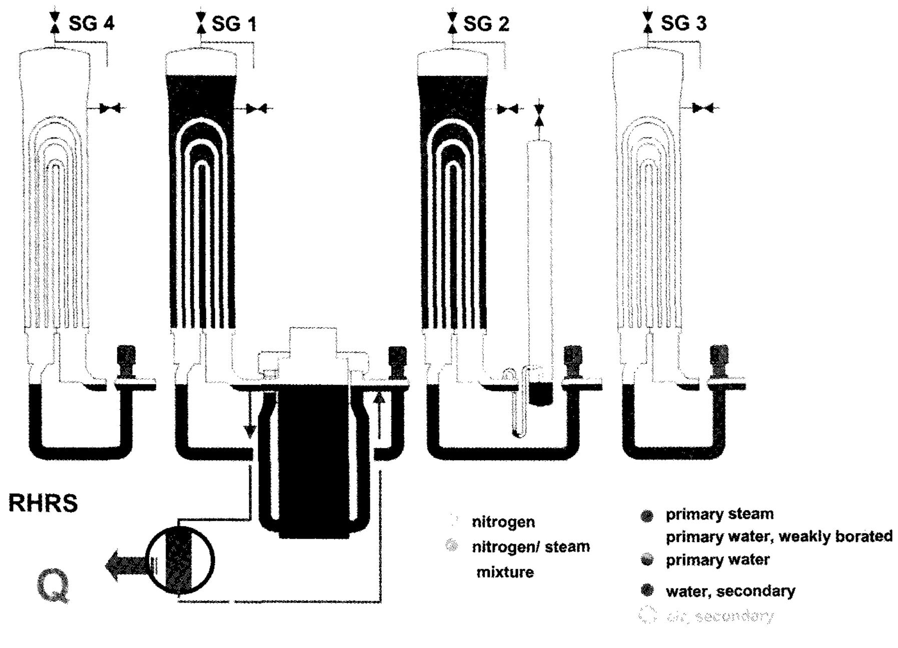

Fig. 6. Tests of Loss of Residual Heat Removal (Initial State)

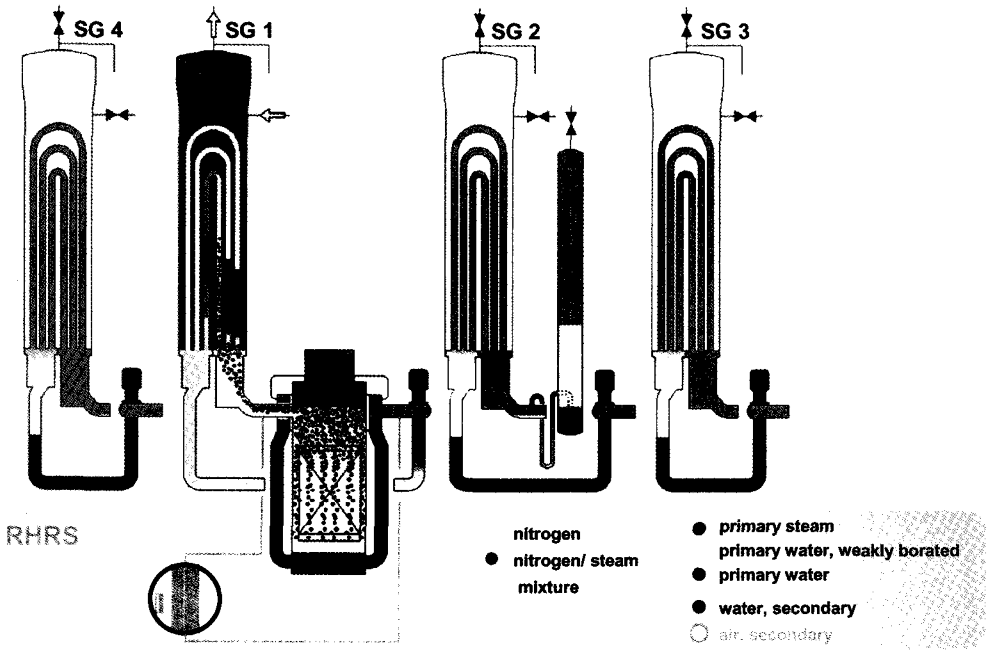

Fig. 7. State of Plant Many Hours after Loss of Residual Heat Removal

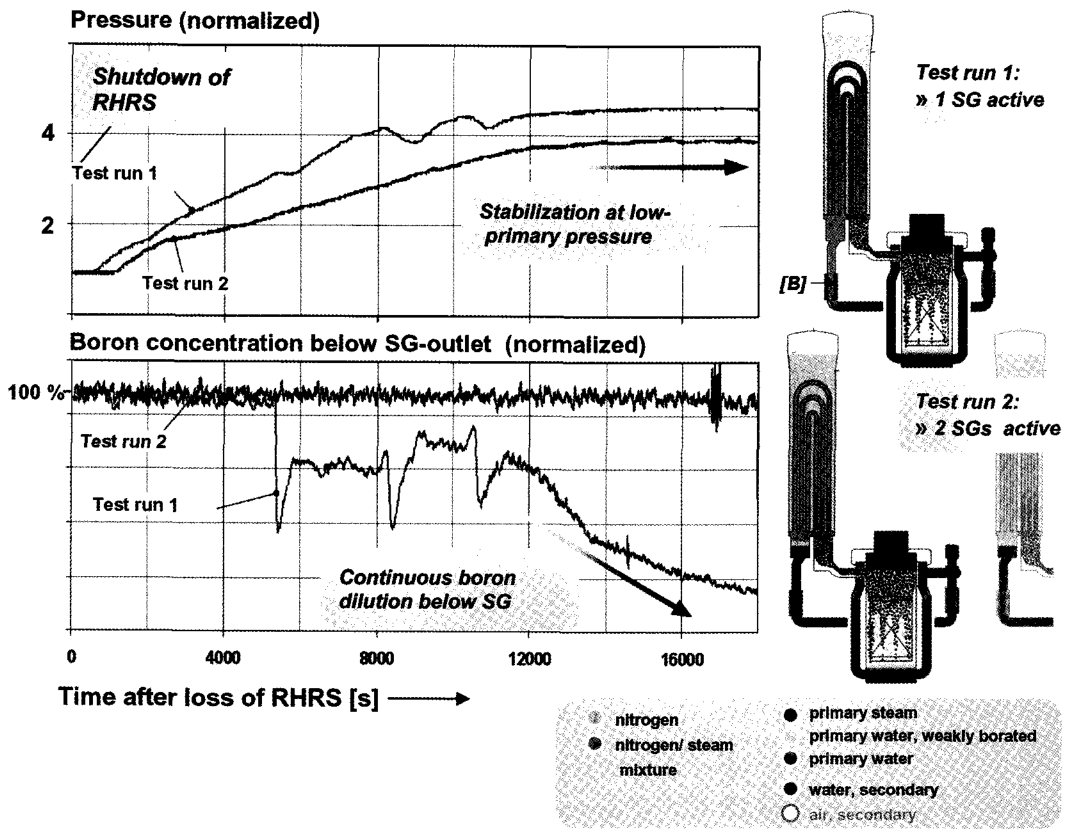

Fig. 8. Primary Pressure and Boron Concentrations for one and Two Standby SGs

in the SG that results, a transfer of weakly-borated water to the outlet side is avoided, and the system stabilizes at an even lower pressure level. Thus, the availability of 2 SGs should be considered.

4.3 Improvement and Optimization of Computer Programs

For the case: Detailed findings for heat transfer and flow behaviors in SGs with failure of residual heat removal (RHR).

A detailed review of heat transfer and flow behaviors in the individual heat exchange tubes of the SG in the tests on loss of RHR reveals extremely different processes among individual tubes (Fig. 9). Mainly in the longer tubes there is only a very short heat transfer zone at the inlet. Above this region - as the temperature measurements show - condensate that has cooled to secondary side temperature stagnates up to a height that can be maintained by the velocity head of the inflowing steam or steam-water mixture. This is, however, an unstable process subject to larger oscillations. In some other tubes, principally the shorter ones, a water-steam mixture can expand into the top of the U-tube. There, very good heat transport exists

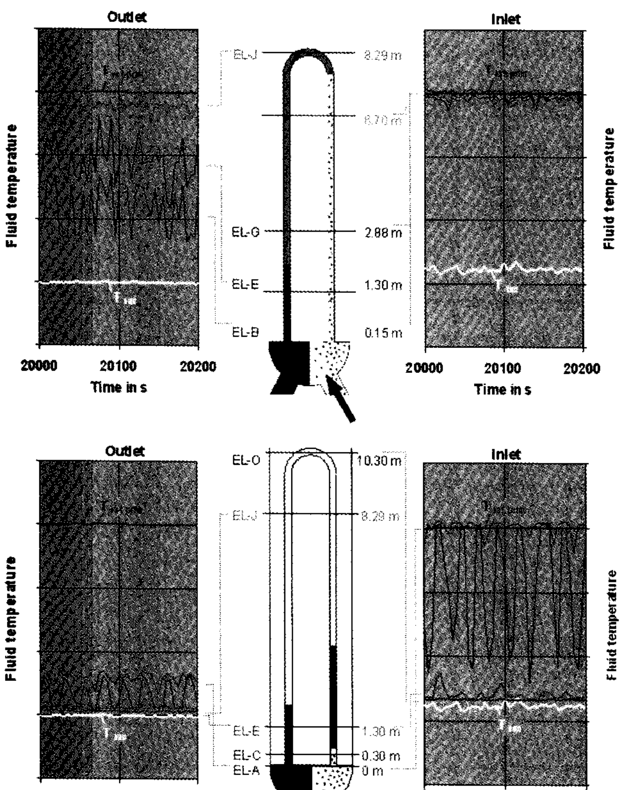

Fig. 9. Various Heat Transfer Processes in the SG U-tubes

over the full length of the inlet side, but there is also the risk of transporting lower-borated condensate to the outlet side of the SG. If the flow velocity is large enough, borated water reaches the top of the U-tube so that the deborating phenomena will not occur.

In the international group of OECD participants in the PKL Project, a so-called benchmark study with various computer programs (RELAP, CATHARE, ATHLET, etc.) was performed for a similar scenario [10]. The results display a large variation and make clear that to describe such complex processes, on one hand, a very detailed nodalization is required (for example, representing the SG tubes by three lengths appears insufficient), and on the other hand, it appears that the models of boron dilution are in need of improvement. Therefore, the PKL tests deliver important, detailed information for the further improvement of system codes and for the development of CFD programs that describe localized processes.

5. SUMMARY AND OUTLOOK

For more than 30 years, extensive experimental investigations into the system response of PWRs under accident conditions have been conducted at the PKL large-scale test facility, which constitutes a full-height model of the entire RCS and major parts of the secondary side of a pressurized water reactor in the 1300 MW class (with diameters a factor of 12 smaller). With the original studies on large- and small-break LOCAs and the modeling of transients in the following years, in particular considering beyond-design-basis conditions, the PKL experiments have covered a very broad spectrum of topics. These investigations have played a key role in German reactor safety research and certainly have also contributed to the safety assessment of PWR units worldwide.

Since April 2001 the PKL project has continued in the course of an international project initiated by the OECD. The investigations carried out in the first years of this international co-operation have delivered the only experimental data base worldwide on PWR system behavior for boron dilution scenarios, because actual boric acid has been used and detailed measurements of the local boron concentrations have been made. The results contribute to the answers to current safety questions, for example, with regard to the flow velocities, size and remaining boron concentration of condensate slugs, which must be considered under specific conditions at the RPV inlet, where the evaluation of the minimal boron concentration entering the core of a PWR is important.

The test results have also found concrete application in validation and further development of thermal-hydraulic computer codes as well as in verification and optimization of cooldown and emergency procedures for the PWR plants.

With the answers to these safety-relevant questions, the PKL investigations of system behavior for small breaks with boron dilution and of the plant shutdown behavior for loss of residual heat removal are, to a large extent, complete.

In the currently running OECD project, the following topics are being investigated, concluding at the end of 2011:

- System behavior for rapid cooldown transients in the SG (for example, a main steam line break) with focus on recriticality and/or pressurized thermal shock (PTS), more specifically subcooled flows in the RPV.
- Excessive boration in the core at very high boron concentrations, which, under certain conditions, can arise in the long-term phase of a large break accident with exclusively cold-side injection.
- Further systematic investigations of heat transfer in the SGs in the presence of a non-condensable gas (nitrogen)

Increasingly, the project partners bring up accident scenarios for new PWR design concepts such as the EPR system; these scenarios require geometric or system alterations in some details. The project is consciously structured with flexibility in the investigation goals, so that on short notice current issues can be addressed with the agreement of the partners.

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NOMENCLATURE

AM	Accident Management
BETHSY	Boucle d'Etudes Thermohydraulique Systeme (Integral Test Facility, France)
CCFL	Counter Current Flow Limitation
ECC	Emergency core coolant
HP	High pressure
LB	Large Break
LOCA	Loss of coolant accident
LP	Low pressure
LSTF	Large Scale Test Facility (Japan)
NEA	Nuclear Energy Agency
OECD	Organization for Economic Co-operation and Development
PKL	Primärkreisläufe-Versuchsanlage (integral test facility, Germany)
PWR	Pressurized water reactor
RCS	Reactor coolant system
RHR	Residual heat removal
RPV	Reactor pressure vessel
SB	Small Break
SETH	SESAR Thermal Hydraulics
SG	Steam generator
SIP	Safety injection pump
TMI	Three Mile Island
UPTF	Upper Plenum Test Facility

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