Analytical and experimental research into boron dilution events

V. Teschendorff, K. Umminger *, F. P. Weiss **

Gesellschaft für Anlagen und Reaktorsicherheit (GRS) mbH – Forschungsgelände, 85748 Garching
* Framatome ANP GmbH - Freyeslebenstr. 1 - 91050 Erlangen
** Forschungszentrum Rossendorf, Institut für Sicherheitsforschung
Bautzner Landstraße 128, 01328 Rossendorf

Abstract: Research activities are being performed in Germany with the aim to improve and validate the methods for predicting boron dilution events. Integral experiments in the PKL test facility investigate the thermal-hydraulic system behaviour in a wide range of conditions. The latest test program comprises small break LOCA scenarios with boron dilution. For these tests, boric acid in the coolant is used together with an advanced instrumentation that can measure boron concentration during the transient. Mixing processes in the downcomer and lower plenum under the influence of various loop operating conditions are studied in the transparent 1:5 ROCOM four-loop test facility equipped with advanced wire mesh sensors to follow the transient concentration patterns. Analytical R&D activities include further model development and validation in the thermal-hydraulic system code ATHLET as well as assessment calculations for detailed three-dimensional mixing in the reactor pressure vessel with CFD-codes.

1. INTRODUCTION

Safety studies for low power and shut-down conditions of nuclear power plants performed in several countries during the last decade directed the interest to accident sequences for which a local boron dilution in the reactor core by a plug of pure water or water of reduced boron concentration might occur.

In Germany, GRS is performing probabilistic safety investigations for a Pressurised Water Reactor (PWR) of KONVOI type with the aim to test the available methodology for Levels 1 and 2. These investigations comprise events during shut-down conditions including event sequences with boron dilution. Also, for loss-of-coolant accidents with small leaks, sequences have been identified that involve boron dilution.

In the following, some research activities are described that are being performed in Germany with the aim to improve and validate the methods for predicting boron dilution events.

1.1. Boron dilution events

Boric acid is used as a soluble neutron absorber in the primary coolant of pressurised water reactors. The main functions of boric acid are to compensate for fuel burnup and xenon poisoning with the required reactivity margin during normal operation, and to provide the necessary subcriticality of the core during refuelling and maintenance. The smaller the fuel burnup is, the higher the boron concentration must be. During normal reactor operation, the specified boron concentration is maintained by the volume control system. In case of a loss-of-coolant accident, borated water is injected by the emergency core.
cooling system (high-pressure safety injection pumps, accumulators, and low-pressure pumps).

Causes for inadvertent boron dilution events could be of two kinds:

- Unintentional injection of unborated water from outside the primary cooling system by malfunctions of injecting systems, (e.g. pump sealing water, malfunction of the volume control system),

- Separation of primary coolant into highly borated water and almost boron-free water by evaporation and condensation.

The first kind of events would normally progress slowly. In case of running reactor coolant pumps or under natural convection conditions - it could be easily detected and corrected, and should not pose problems for analysis. Stagnant flow conditions (e.g. mid-loop operation) or transients of the second kind could result in event sequences that are more difficult to analyse and constitute a challenge to the analytical methods. This is why R&D work into boron dilution events has been intensified in Germany.

For the event sequences with boron dilution considered in safety analyses, multiple failures of systems are assumed. It is a common feature of such events that during periods of reduced coolant inventory in the primary system and with the reactor shut down, evaporation takes place in the core and condensation in the steam generators. The boron is mainly retained in the liquid phase so that the vapour phase and condensate is almost boron-free. If this situation prevails over a long time without injection of borated water from outside large amounts of unborated or low-borated water can accumulate in the primary system and form “plugs”, e.g. in the pump crossover legs of the main coolant loops. If at a later time circulation in the cooling system is reestablished either by natural circulation following refilling or by restarting of a main coolant pump, water of low boron concentration may be driven through downcomer and lower plenum into the core. The effect on reactivity depends strongly on the spatial and temporal concentration profile.

Current safety studies in Germany are mainly considering the following initiating events:

- Loss of residual heat removal under shut down conditions (mid-loop operation),

- Small leak with multiple failures in the emergency core cooling system.

Both events have been extensively analysed under a variety of assumptions. For the small leak, various combinations of leak size, leak location, and failures of the emergency core cooling system are being studied.

1.2. Phenomena that require further research

Several observations from these investigations are relevant for further R&D effort, namely for code development and additional experiments:

- Influence of system parameters such as injection location, injection rate, break location on plug formation and onset of natural circulation in the different loops

- Mixing of unborated water with borated water within the steam generators, the primary pipes, in the downcomer and lower plenum of the reactor pressure vessel is decisive for the boron concentration at the core inlet and thereby for the resulting reactivity insertion. System code analysis should be complemented by multi-dimensional component code analysis and findings from mixing experiments.

- Break-down and restart of natural circulation in one or several loops is sensitive to the behaviour of the main coolant pumps at low velocities and to the direction of injected water
in the hot leg (flowing co-current with the two-phase mixture to the steam generator or counter-current to the upper plenum).

- Models for counter-current flow limitation of vapour and liquid in the horizontal and inclined part of the hot leg during the period of “reflux condenser mode” should be validated on experimental data obtained for large pipe diameters,

- System codes with coarse first order space discretisation should be supplied with a shape-preserving algorithm to prevent “numerical diffusion” of concentration profiles,

- Strong interaction between core behaviour and cooling system behaviour requires a system of coupled codes.

In the following, recent analytical and experimental research work is presented that is performed in support of more accurate predictions for boron dilution events.

### 2. CODE DEVELOPMENT

Safety analysis for boron dilution events requires a system of computer codes that comprises a thermal-hydraulic system code, a neutron kinetics code with three-dimensional core simulation coupled with the thermal-hydraulic code, and a tool for three-dimensional mixing calculations in certain regions of the core [1]. In the following, examples from code development and validation are presented.

#### 2.1. Improving the system code ATHLET

ATHLET is a thermal-hydraulic system code developed by GRS¹ for the analysis of transients, design basis accidents and beyond design basis accidents without core melting in light water reactors [2]. It has been systematically validated on a comprehensive set of integral experiments and separate effects tests [3]. The code is being applied for safety evaluation for nuclear power plants in Germany and abroad, with numerous user organisations in central and eastern Europe.

An R&D project² for the improvement of analytical methods for deboration events was initiated. It comprised improvement and validation of ATHLET models, evaluation of mixing experiments, analyses with CFD codes, and generic reactor calculations. The following two subchapters describe important model additions to ATHLET that are relevant for boron dilution events.

---

¹ Development and validation of ATHLET is sponsored by the German Federal Ministry of Economics and Technology (BMWi).

² Project RS 1125 sponsored by the German Federal Ministry of Economics and Technology (BMWi).
2.1.1. Boron transport modelling in ATHLET

An essential part of the boron simulation model of ATHLET is the boron transport model, i.e. the method to calculate the boron concentration transported in the junctions. Due to the 1st order local approximation in ATHLET, the application of the pure donor principle would cause high numerical diffusion which slopes down the spatial boron concentration profiles when they move through the control volumes. To mitigate this effect, a simple time lag method was used up to now. For that purpose, the time history of the boron concentration is stored for every control volume (CV) and the transit time of the flow through the CV \( i \) is calculated as

\[
\tau = \frac{V_i}{A_i w_{Li}}
\]

with

- \( V_i \) CV volume (m\(^3\))
- \( A_i \) CV flow area (m\(^2\))
- \( w_{Li} \) CV liquid velocity (m/s)

The transported boron concentration at current time \( t \) equals the donor CV boron concentration at \( t - 0.45\tau_i \).

This method reduces the numerical diffusion significantly but not sufficiently. In particular, it is not suitable for frequent flow reversals as it is observed for many boron dilution transients. Hence, a new method has been developed which better preserves the boron concentration profiles even under oscillating flow conditions. For that, the spatial boron profile within every CV is stored as a table, resulting in a one-dimensional, continuous profile for the complete coolant circuit. This profile is transported with the liquid flow in the main (longitudinal) direction; branching flows are not considered for the profile transport.

In parallel, the boron mass in the CVs is explicitly integrated balancing all in- and out-flowing boron masses, and the average CV boron concentration is calculated considering evaporation and condensation. The boron concentration profile is then adjusted to the average concentration.

The transported boron concentration in the main junctions is that of the right (left) end of the donor control volume for positive (negative) flow direction. Branching junctions are still calculated with the time lag method.

The boron profiles are updated once in a FEBE integration time step (loose coupling). Through the interpolation of the concentration profiles in upstream direction during the FEBE substeps, a quasi-close coupling is achieved for the boron transport.

A simple test illustrates the effect of the different methods on the mitigation of numerical diffusion. Fig. 2.1.1-1 shows a pipe with 0.75m diameter, subdivided into 9 CVs, each with the length of 1 m. This is a typical nodalisation for a PWR main coolant pipe. An initial spatial boron concentration profile of \( 0 \rightarrow 1000 \rightarrow 0 \) ppm is specified. Through a fill at the left boundary, a nearly rectangularly oscillating pure water mass flow is defined. The time dependent volume at the right boundary provides constant pressure, temperature and boron concentration (0 ppm). Three complete cycles are calculated, where the boron plug moves back and forth within the pipe. Without any numerical diffusion, the final boron concentration profile should be equal to the initial one.

In Fig. 2.1.1-2 the concentration profiles obtained by the different transport models at the end of the calculation are compared with the initial profile. The result of the donor method is also included although it is actually not a valid ATHLET option. The comparison demonstrates clearly the advantage of the profile method.
2.1.2. Hot leg injection

During an accident with a small leak in the primary coolant system, natural circulation of the coolant establishes after the primary coolant pumps’ coast down. In the hot legs, liquid or mixture flows with moderate velocity from the reactor vessel (RV) towards the steam generators (SG). In Siemens design PWRs, emergency core cooling (ECC) water is injected into the hot legs via a scoop shaped flow device which directs the injected liquid through its momentum towards the upper plenum (UP). In addition, gravity forces may cause the injected liquid to flow – counter current to the natural circulation flow in the hot leg – completely or partially towards the UP. If the hot leg is largely filled with saturated liquid coming from the UP, and if the velocity is adequately small a thermal stratification will establish with subcooled ECC liquid below saturated water. The cold liquid is retained at the hot leg bend and the inclination of the cold layer forces it towards the UP (Fig. 2.1.2-1).

In ATHLET only one velocity is calculated for each phase, hence liquid–liquid counter current flow cannot be considered. All injected water is transported with the liquid main flow in the hot leg. Therefore, a dedicated model has been developed which distributes the injection between the hot leg injection point and the UP.

The analysis of the UPTF-TRAM A3 experiments [4] point out that the split of the injection flow can be quantified by modified Wallis parameters. The Wallis parameter is the ratio of the hydrostatic force due to the density difference of the phases and the friction force of the concerning phase. The square root of this ratio can be considered as a dimensionless velocity \( J^* \):

\[
J^*_{v} = \sqrt{\frac{J^2_{v} \rho_v}{(\rho_l - \rho_v)gD}} \quad \text{Wallis parameter for steam}
\]

\[
J^*_{i} = \sqrt{\frac{J^2_{i} \rho_i}{(\rho_l - \rho_v)gD}} \quad \text{Wallis parameter for liquid}
\]

The modification considers the fact that, as a result of the steam flow, only a portion of the hot leg flow area is available for the counter current liquid flow. The correlation obtained from the UPTF-TRAM experiments is given by

\[
0.8 = \sqrt{J^*_{HL,J}} + 0.5\sqrt{J^*_{ECC\rightarrow UP}}
\]

with the Wallis parameter for the hot liquid flow from the UP to the hot leg entrance

\[
J^*_{HL,J} = \frac{\dot{M}_{HL,J}}{A_i \sqrt{gD \rho_{UP} (\rho_{ECC} - \rho_{UP})}}
\]

and the Wallis parameter for the counter current ECC flow from the hot leg to the UP

\[
J^*_{ECC\rightarrow UP} = \frac{\dot{M}_{ECC\rightarrow UP}}{A_i \sqrt{gD \rho_{ECC} (\rho_{ECC} - \rho_{UP})}}
\]

For a typical high pressure injection rate of about 40 kg/s, the experimental data represented in Fig. 2.1.2-2 show that with low circulation flow \( J^*_{HL,J} < 0.6 \) the injected ECC water \( J^*_{HL,J} \approx 0.25 \) is directed nearly completely to the UP. After the counter current flow limitation has been reached \( J^*_{HL,J} \geq 0.6 \) the reversal of the injection flow towards the steam generator starts.
Fig. 2.1.2 includes also results from the PKL test facility. The PKL results confirm the UPTF-TRAM findings although it is volumetrically scaled by 1:145.

From the formulae above, the Wallis parameter for the counter current ECC liquid flow from the hot leg to the UP is derived as

\[ J_{\text{ECC} \rightarrow \text{UP}, 1}^* = \left( 6 - 2 \sqrt{J_{\text{HL}, 1}^*} \right) \]

The ATHLET model calculates the fraction of the ECC water directed immediately to the UP as

\[ \dot{M}_{\text{ECC} \rightarrow \text{UP}, j} = J_{\text{ECC} \rightarrow \text{UP}, 1}^* A_j \sqrt{gD_{\text{ECC}, 1} \rho_{\text{ECC}, 1} - \rho_{\text{UP}, j}} \]

If the ECC water contains boron or dissolved gas it will be directed to the UP in an according manner. The remaining water, boron, and dissolved gas is added to the injection control volume.

2.2. Coupled codes

For transients and accidents which are characterised by strong interaction between local core behaviour and coolant system behaviour, thermal-hydraulic system codes and 3D neutronic codes should be applied with close coupling.

2.2.1. ATHLET - QUABOX/CUBBOX

GRS has developed a coupled version of the 3D transient neutronic code QUABOX/CUBBOX with ATHLET.

QUABOX/CUBBOX [5] solves the neutron diffusion equations with two prompt neutron groups and six groups of delayed neutron precursors. The coarse mesh method is based on a polynomial expansion of neutron flux in each energy group. The time integration is performed by a matrix-splitting method which decomposes the solution into implicit one-dimensional steps for each spatial direction. The reactivity feedback is taken into account by dependence of homogenized cross-section on feedback parameters, the functional dependence can be defined in a very general and flexible manner.

The coupling approach for 3D neutronics models implemented in ATHLET has been described [6]. It is based on a general interface, which separates data structures from neutronics and thermo-fluiddynamic code and performs the data exchange in both directions. The approach has been successfully applied to couple other 3D neutronics codes. The internal coupling method has following features: The fluiddynamic equations for the primary circuit and the flow channels in the reactor core region are completely modelled and numerically solved by ATHLET methods. The time integration in the neutronics code QUABOX/CUBBOX is performed separately. Therefore, both codes keep their capabilities. The time step size is synchronized during the transient, whereby the accuracy control is preferably done by the fluiddynamic code. The coupling allows a flexible mapping defined by input between fuel assemblies of the core loading and the thermo-fluiddynamic channels.

The potential of the coupled system was demonstrated in the framework of the OECD PWR MSLB-Benchmark [7]. Currently GRS participates in the BWR turbine trip benchmark using advanced versions of these codes.

The code system has already been applied for safety analyses of RBMK reactors [8]. Generic analysis of various boron dilution events in German PWRs is in progress.
2.2.2. RELAP5 / PANBOX

At Framatome ANP GmbH the coupled three-dimensional (3-D) thermal-hydraulic/kinetics code RELAP5/PANBOX [9] is used for PWR transient and accident analysis. The code system is capable of modelling reactivity transients, including boron dilution transients. RELAP5/PANBOX – in short R/P/C – is a part of the integrated code system package CASCADE-3D [10] used at Framatome ANP GmbH in the area of core design and safety analysis. R/P/C is comprised of the core simulation package PANBOX coupled directly to the best-estimate plant simulation code RELAP5. Coupling of RELAP5 has been achieved via the general RELAP5 interface package EUMOD (External User MODels). R/P/C has the capabilities of RELAP5 with added ability to calculate 3-D neutron kinetics and thermal margins with COBRA, the core thermal-hydraulic module of PANBOX. External I&C models may also be linked through EUMOD to R/P/C. Control of RELAP5 variables and PANBOX control assembly position is possible. The input is simple to adapt from the stand alone versions of the codes.

2.2.3. ATHLET / DYN3D

The 3D reactor core model DYN3D [11] has been developed at FZR to improve the simulation of reactivity initiated accidents, where space-dependent effects in the reactor core are relevant. The neutron flux distribution is calculated by solving the two group neutron diffusion equation by means of different nodal expansion methods for hexagonal and for square fuel assembly geometry. The thermo-hydraulic part of the code comprises a one- or two-phase coolant flow model, a fuel rod model describing the thermo-mechanical behaviour of the fuel and cladding, and a heat transfer regime map ranging from one-phase liquid flow to superheated steam.

For coupling DYN3D with ATHLET two different ways were pursued [12]. The first one uses only the neutron kinetic part of DYN3D and integrates it into the heat transfer and heat conduction model of ATHLET. This is a very close coupling, the data have to be exchanged between all core nodes of the single models (internal coupling). For the second way of coupling, the whole core is cut out of the ATHLET plant model (external coupling). The core is completely modelled by DYN3D. The thermohydraulics are split into two parts: the thermohydraulic model of DYN3D describes the thermal and fluid dynamics of the core and ATHLET models the coolant system.

This code combination has been successfully applied in international benchmark exercises, e.g. [13] as well as to NPP transients and accidents.

2.3. CFD codes calculations of ISP 43

Computational Fluid Dynamics (CFD) methods are an effective tool to calculate three-dimensional mixing flows. Due to the rapid development of computer hardware and software it has now become feasible to simulate transient flow conditions and the transport of deborated water. However, the applied numerical methods and turbulence models require experimental validation with detailed local resolution of flow- and temperature fields. Therefore, the International Standard Problem ISP 43 was defined specifically for the validation of CFD programs at the University of Maryland College Park (UMCP) in cooperation with US NRC.
The UMCP 2x4 Loop Facility is a scaled model of the TMI-2-reactor of Babcock & Wilcox with a detailed reconstruction of the reactor pressure vessel (RPV). The instrumentation of the test rig is concentrated in the downcomer (DC) and lower plenum in order to allow a detailed comparison with CFD results. A description of the facility is given by Gavrilas et al. [14]. Fig. 2.3-1 shows the experimental configuration and the location of thermocouples.

The ISP 43 experiments are focused on rapid boron dilution transients. A deborated slug is set into motion by actuation of the main coolant pump of cold leg CLA1. On its way through the system the deborated water mixes with the borated water in the primary system. Due to 3D-mixing effects, variations of the boron concentration occur in time and space at the core entrance. In test case B the borated coolant in the primary system is simulated as warm water (T = 69 °C) and the deborated plug as cold water (T_{min} = 15 °C). The transient mass flow rate and temperature distribution at the inlet of CLA1 is shown in Fig. 2.3-2. At the beginning of the simulation the system is filled with stagnant, warm water. Heat losses at the outer walls are negligible.

The calculation of the transient, turbulent and incompressible flow was performed by GRS with CFX-TASCflow. It is based on the solution of three-dimensional ensemble averaged conservation equations for mass, momentum and energy. The discretisation methods are second order in time (implicit Euler method) and space (Linear-Profile Skewed Upwind combined with Physical Advection Correction-Method). The turbulent transport terms in the conservation equations of the momentum and energy equations are calculated with the standard k-ε turbulence model [15].

The numerical solution is based on a conservative, element-based Finite-Volume method with co-located variable storage. The numerical grid consists of non-orthogonal, hexahedral flux elements. The coupled non-linear, algebraic equation system is linearized by using a substitution or Picard iteration scheme. At every iterations step pressure and velocity fields are solved simultaneously as one block-coupled system. The coefficient matrices are solved with an algebraic multi-grid method. This solution technique provides robust and rapid convergence [16].

The computational geometry is based on the 3D surface model available in IGES Format on the web side of UMCP. The model was imported into the Powermesh software of ICEM CFD and transformed into a block-structured grid with 470 000 hexahedral flux elements. Fig. 2.3-3 shows the numerical grid with the cold legs, the lower plenum and the inner DC wall. The outlet is placed at hot leg HLA. The core is simulated as porous media with a pressure loss of 2000 Pa given by the experiment.

The temperature distribution in the DC, see Fig. 2.3-4, at 20 s after start of the transient, shows 3D effects of the non-symmetric injection of the deborated water plug. Cold water spreads at the entrance of the DC, downward and in circumferential direction. High temperatures appear at the cross section where the DC width increases. These are caused by local recirculation due to the geometric discontinuity. After 20 s cold water has already reached the lower plenum where strong recirculation zones develop.

The DC level 4 azimuthally averaged temperature is the principle figure of merit for ISP 43 [17]. In Fig. 2.3-5 the transient, azimuthally averaged temperatures of experimental and numerical results are compared, as well as maximal and minimal temperatures. The standard deviation of the experiment is delimited by the shaded area. The numerical injection starts after 9 s and ends after 60 s. At the beginning of the transient the calculated maxima are above and the minima are below the experimental data. This indicates that mixing is slightly underestimated. However, the azimuthal, local temperature distributions on level 4 at time = 20 s, 30 s, 40 s is in good agreement with the experiment, see Fig. 2.3-6. It shows that the local distribution of cold water in the DC is simulated correctly [18].
3. EXPERIMENTS

3.1. PKL test facility and program

3.1.1. Investigation of PWR system behaviour in the PKL test facility

For nearly 25 years now, Framatome ANP's large-scale test facility PKL in Erlangen, Germany (PKL is a German acronym for "primary system") has been used for extensive experimental investigations to study the integral behaviour of pressurised water reactor (PWR) plants under accident conditions [19]. The PKL facility (Fig. 3.1.1-1) replicates the entire primary system and most of the secondary system (except for the turbine and condenser) of a 1300-MW PWR plant, with elevations scaled 1:1 and diameters reduced by a factor of 12. The core is modelled by a bundle of 314 electrical heater rods. The core geometry – like the geometry of the steam generators – is represented as an "original segment"; i.e. the individual heater rods and U-tubes have the same geometry as in a real plant but the number of rods in the core and the number of tubes in the steam generators have been reduced by a factor of 1:145 (volume and power scaling). Modelling of the primary system with four identical reactor coolant loops arranged symmetrically around the reactor pressure vessel (Fig. 3.1.1-2) permits accidents to be investigated under realistic conditions, including those accidents characterised by non-symmetrical boundary conditions. As the functions of all major primary and secondary auxiliary systems and other connecting systems are also replicated in the test facility, integral system behaviour as well as the interaction between individual systems can be investigated under a wide variety of different accident conditions and the effectiveness of either automatically or manually initiated actions can be examined. With its total of around 1500 measuring points, the PKL facility is extensively instrumented, which permits detailed analysis and interpretation of the phenomena observed in the tests.

The experiments conducted at the PKL test facility cover an extremely broad spectrum of topics, ranging from the tests performed to study large- and small-break loss-of-coolant accidents (LBLOCAs and SBLOCAs) when the facility was first built, to the simulations of accident transients – including, especially, the effects of beyond-design-basis events – that have dominated experimental programs in recent years. The current PKL III E test series includes investigations into the topic of inadvertent boron dilution events. In April 2001, the PKL test program was continued as part of an international collaborative project set up by the OECD (Organisation for Economic Co-operation and Development). Integration of the PKL experiments into this joint project allows additional tests to be performed and will also ensure that the test facility remains available for use for a prolonged period of time.

3.1.2. PKL boron dilution tests

In all PKL tests carried out to date on the topic of boron dilution, the initiating event was assumed to be an SBLOCA. First tests were already performed within the PKL III D test program which was completed at the end of 1999. These tests which were carried out with unborated water as reactor coolant supplied important information regarding the establishment of natural circulation in the individual reactor coolant loops as a function of various boundary conditions.

In order to realistically simulate also the mixing of differently borated water flows in the loops and in the steam generators during the refill phase and after the onset of natural circulation – which is very important for assessing the effects of boron dilution accidents – the decision was made to try to use actual boric acid resembling that employed in a real plant during the tests of the current PKL III E series. In December 2000, the first PKL test (PKL III E1.1) was
carried out using boric acid and with suitable provisions made for boron measurement (e.g. an on-line system for continuous measurement of the boron concentrations).

Test E1.1 was based on symmetrical coolant injection into the cold legs of the loops and should therefore be regarded as a fundamental study based on generic assumptions regarding symmetry (representing an adverse boundary condition in terms of boron dilution) and not as a test intended to simulate a specific PWR accident sequence. The prime objective of this test was – apart from studying issues related to the onset of natural circulation – to determine the minimum boron concentration at the RPV inlet resulting from the establishment of natural circulation in the individual loops.

Since the maximum operating pressure of the PKL facility is 45 bar, it is not possible to simulate the high-pressure portion of such accident sequences starting from a PWR's actual operating pressure (160 bar). Hence, in the tests, the accident "starts" at a primary system pressure of less than 45 bar and with initial conditions corresponding to those that would prevail in a real plant at this time. The initial conditions used for Test E1.1 consisted of a partially emptied primary system at a pressure of 40 bar from which heat is being symmetrically removed via all four steam generators through operation in the reflux condenser mode (Fig. 3.1.2-1). As the facility was operated in the reflux condenser mode for a sufficiently long period of time prior to commencing the test, the water in the loop seals was largely free of boron ([B] < 50 ppm). At the same time, the boron concentration at the core exit had risen to approximately 3600 ppm (verified by water samples).

Starting with these initial conditions, borated emergency coolant ([B] = 2200 ppm) was then fed into the primary system. Parallel to water being injected into the primary system, the steam generators were cooled down on their secondary sides at a rate of 100 K/h. As soon as primary system pressure had dropped below 10 bar, coolant injection was taken over by the LP systems (with the water likewise being fed into the cold legs of all four loops). No use was made of the accumulators in order to avoid any possibility of accumulator injection resulting in asymmetrical conditions.

An overview of the main results of Test E1.1 is presented in Fig. 3.1.2-2 which depicts the process of refilling of the steam generator tubes, circulation in the individual loops (mass flow measured at steam generator outlet) as well as the boron concentration measured in the cold leg of Loop 1 at the RPV inlet. Despite the fact that the greatest possible degree of symmetry was obtained across the loops and refilling of the individual steam generators was also largely symmetrical, the onset and maintenance of natural circulation in the individual loops was found to differ. As the steam generators are not functioning as a heat sink during the period with SIPs in operation, steady-state natural circulation does not become established at this time. It is only when the SIPs have been shut down, primary system pressure has dropped below 10 bar and the LP pumps are started up (to inject coolant into each of the four loops) that steady-state circulation becomes established in all loops at roughly the same level of mass flow. This form of loop circulation results in only a minimal reduction in boron concentration at the RPV inlet – and this only when the SIPs have been shut down.

Obviously, during and after refilling, the (low-boron) condensate contained in the loop seals prior to commencement of the test mixes almost completely with the injected coolant. Besides the mixing in the loops during the refill phase and after initiation of natural circulation, the processes in the steam generators play an important role here. As illustrated in Fig. 3.1.2-3 for steam generator 1, circulation starts in the various tubes at different points in time. The overall effect of this was that the condensate contained in the various tubes was discharged at different points in time into the steam generator outlet plenum, something which promoted mixing considerably.
Reliable information regarding flow conditions in the individual steam generator tubes can be obtained from the temperature measurements taken (using thermocouples) in the steam generator tubes, especially in the bottom sections of the inlet side (i.e. immediately adjacent to the inlet plenum). For example, after completion of the refill phase, a primary-side temperature approaching that on the secondary side is a clear indication of flow stagnation in the respective tube (e.g. tube 29, Fig. 3.1.2-3), whereas if it approaches that prevailing in the steam generator inlet plenum then this is a sign that natural circulation is starting in the tube (e.g. tube 1, Fig. 3.1.2-3).

Therefore, the overall conclusion to be drawn from this test is that, despite symmetrical injection into the loops, the onset of natural circulation is asynchronous and that only a small reduction in boron concentration is to be observed at the RPV inlet due to efficient mixing in the steam generators and in the loops.

Because of its design (all elevations are scaled 1:1 and the four reactor coolant loops are arranged symmetrically), the PKL integral test facility is highly suitable for resolving questions related to natural circulation. Due to its exact replication of the steam generators as a whole and especially of the steam generator tubes (including, for example, their differing heights) the processes observed in these regions can be applied very well to real power plant configurations. Hence, heterogeneous behaviour in the steam generator U-tubes can likewise be assumed for an actual PWR plant under similar boundary conditions. The results of the PKL boron dilution tests provide vital information regarding the onset of natural circulation, boron distribution, mixing processes in the loops and in the steam generators, and especially boron concentrations at the RPV inlet. They therefore represent an important database for validating and further developing system codes as well as for supplying boundary conditions for 3D computational fluid dynamics (CFD) codes (e.g. for analyzing downcomer mixing).

To perform a final evaluation of such analyses and provide them with experimental verification, further integral tests are needed in which the boundary conditions are varied (to simulate different pipe break and coolant injection configurations). These investigations will be performed as part of the current, internationally supported OECD program officially started in April 2001.

It has to be mentioned here, that for the realistic simulation of the mixing processes taking place in the RPV downcomer and lower plenum, test facilities in which the relevant components are modelled in full scale or at least in a larger scale are more suitable. Mixing experiments of this kind have been performed as part of the UPTF TRAM project using a full-scale mock-up of the RPV downcomer and the connected primary system piping [20]. In this connection, the tests results from PKL and UPTF also demonstrate the mutually supplementary character of the two test facilities.
3.2. UPTF test facility

The tests in the Upper Plenum Test Facility (UPTF) show that in terms of mixing, the cold legs supplied with emergency coolant exhibited an ideal mixing behaviour of condensate and emergency core coolant. It was also demonstrated that the mixing behaviour of the mixed coolant injected from the cold legs (supplied with emergency coolant) into the downcomer annulus with the inflowing condensate from the cold legs not supplied with emergency coolant is good to ideal.

As an example, Fig. 3.2-1 provides an impression of the mixing processes in the downcomer during run 13a. It shows the water temperature distribution in the unwrapped RPV downcomer at a time 1337 s after starting the test run 13a. The water mixture (comprising emergency core cooling water and hot water) flowing into the RPV downcomer from cold leg 2 forms a cold water plume. In the upper part of the downcomer the water is thermally layered outside the plume and there is a significant temperature difference between this water and the cold water plume. The intensive mixing of the cold water with the surrounding hot water results in an azimuthally uniform temperature profile without a plume present at and lower than 3300 mm below the lower edge of the cold leg, and a relatively uniform temperature throughout the entire height. Although Fig. 3.2-1 only illustrates the situation at a certain point in time during one selected test run, the following important overall test result can be established: During all test runs and at all times during the experimental transients, the mixing between the condensate and the emergency core cooling water in the downcomer was completed 4000 mm below the lower edge of the cold leg.

3.3. The mixing test facility ROCOM

ROCOM (Rossendorf Coolant Mixing Model) is a test facility for the investigation of coolant mixing operated with water at room temperature. The facility models a KONVOI type reactor in a linear scale of 1:5. ROCOM is a four-loop test facility with a Reactor Coolant Pump (RPV) mock up made of transparent plexiglas® (Figs. 3.3-1 and 3.3-2). The transparent material for the pressure vessel allows the measurement of velocity profiles in the downcomer by laser Doppler anemometry. Individually controllable pumps in each loop give the possibility to perform tests in a wide range of flow conditions, from natural circulation to nominal flow rate including flow ramps (pump start up).

Special attention was paid to components, which significantly influence the velocity field such as core barrel with lower core support plate and core simulator, perforated drum in the lower plenum, inlet and outlet nozzles. Thus, the geometry of the inlet nozzles with their diffuser segments and the curvature radius of the inner wall at the junction with the RPV was modelled in detail.

The water inventory of the loops is kept in the scale of 1:125 and the travelling time of the coolant is identical to the original reactor. Reynolds numbers in the range of some $10^5$ and at least $\text{Re} = 3.4 \times 10^4$ at the core inlet in case of one loop operation indicate a highly turbulent flow being a sufficient condition for the transferability of the measured flow field and the concentration distributions to the original reactor as long as density differences are negligible. This is fulfilled if the Froude number of the flow is much greater than 1, what has to be assessed for each experimental scenario. The transferability to original reactor conditions is substantiated by Dräger [21] comparing experimental results of a 1:5 scaled air-
operated model of the Russian VVER-440 with measured data from the original reactor. The same conclusion was drawn from own numerical work for KONVOI type reactors [22].

The test facility is operated with de-mineralised water. During cold water or boron dilution transients after RCP start up, the mixing is dominated by turbulent mechanisms. Therefore, both boron concentration and temperature fields can be modelled by the concentration field of a NaCl-tracer solution. The disturbance is created by computer controlled injection of salted water with a conductivity between 100 and 2000 μS/cm into the cold leg of one of the loops (disturbed loop) near the inlet nozzle of the reactor. A special mixing device is installed at that position, which equally distributes the tracer over the cold leg cross section.

The test facility is equipped with measurement devices, which allow a high resolution measurement of the transient tracer concentration in space and time. For that purpose, special wire-mesh sensors, based on the measurement of the electrical conductivity have been developed [23]. They consist of two orthogonal electrode grids put into the measuring cross section. The electrodes are electrically insulated wires. The wires of the first grid (transmitter electrodes) are supplied with short voltages pulses in a successive order. When a pulse is given to a certain wire, the individual currents, arriving at each of the wires of the second plane (receiver electrodes) are proportional to the local conductivity of the water at the crossing points between corresponding transmitter and receiver electrodes.

Four such sensors are installed in the facility with altogether about 1000 single measurement positions. One sensor is located in the cold leg inlet nozzle of the disturbed loop (Fig. 3.3-3, pos. 1), two in the downcomer just below the inlet nozzles (Fig. 3.3-3, pos. 2) and before the entrance into the lower plenum (Fig. 3.3-3, pos. 3), respectively. The fourth sensor (Fig. 3.3-3, pos. 4) is integrated into the lower core support plate and has one measurement position at each fuel element position. All sensors provide 200 measurements per second. In general, a measuring frequency of 20 Hz is sufficient, therefore usually ten successive images are averaged into one instantaneous conductivity distribution. Some of the measurements were carried out at 200 Hz to study turbulent phenomena.

The measured conductivity values are transformed into a dimensionless mixing scalar which is defined by relating the local instantaneous conductivity change to the amplitude of the conductivity change in the inlet nozzle of the disturbed loop.

3.3.1. Experiment with Start-up of the first RCP at the ROCOM test facility

Several hypothetical accident scenarios are discussed, where due to the start-up of the first RCP after flow stagnation, a slug of lower borated coolant could be driven into the RPV. A numerical analysis of such a hypothetical boron dilution event in a German PWR was performed by Reinders [24]. From that work, the following boundary conditions were derived for generic experiments at the ROCOM test facility:

- Flow stagnation in the initial state
- Start-up of the first RCP by linear increase of the pump frequency from 0 to the nominal value within 14 s
- Injection of water with high conductivity into the pump ramp

The experiments were carried out without density differences, because the criterion for neglecting the density differences between borated and unborated coolant outlined in [25] is fulfilled a few seconds after pump start-up. The normalised coolant velocity in the loop with the starting pump is shown on Fig. 3.3-4. In each of the remaining three loops with switched-off coolant pumps, a reverse flow establishes, which reaches about 10% of the flow rate in the starting loop (see Fig. 3.3-4).
The tracer simulating the coolant with lower boron content passes the wire-mesh sensor in the inlet nozzle (Fig. 3.3-3, pos. 1) from 7 to 14 s. This corresponds to an initial slug position behind the RCP. The dimensionless mixing scalar averaged over the cross section at the sensor position is given in Fig. 3.3-4, too.

3.3.2. Comparison of the experimental results with CFD calculations

For CFD calculations of the experiments, the code CFX-4 [26] is used. An incompressible fluid is assumed for the coolant flow. The turbulence is modelled using the standard (k, ε) approximation. The inlet boundary conditions (velocity, temperature, boron concentration etc.) are set at the inlet nozzles.

The boundary conditions at the core outlet are pressure controlled. For the description of boron respectively salt tracer concentration, scalar fields are used which are transported with the fluid and are subject of turbulent diffusion, but do not feedback on the fluid velocity field.

In the created grid models, special attention was paid to the exact representation of the nozzle region with the curvature radii at the four inlet nozzles, the passes of the hot legs through the downcomer and the diffuser-type extension of the downcomer below the nozzles. Due to the great importance of these parts for the coolant mixing, the grid is very dense in these regions. The generated grid consists of about 450000 nodes.

The core support plate, the core and the perforated drum are modelled as a porous region. The porosity is determined by relating the area of orifices to the total area of the sieve. The body forces taking into account distributed friction losses in the sieve plate are determined to obtain equal pressure drop over the structures as they are derived from the detailed construction.

The boundary conditions specified in section 3.3.1 were used as input for the calculations with CFX-4. In Fig. 3.3-6, the distribution of the mixing scalar (normalised boron dilution) at the core inlet obtained in the experiment and CFX calculation are shown for three time points. About 14 s after switching-on the RCP, the deboration front reaches the core inlet at two positions in the periphery of the core about 120 ° shifted from the azimuthal position of the inlet nozzle of the loop with the starting pump. In the CFX-calculation the deboration front reaches the core at the same positions but with a small time delay of less than 0.5 s. Then the maximum of the deboration front moves over the centre of the core to the side of the injecting loop. The described development of the core inlet distribution in time is caused by the flow pattern in the downcomer, which was obtained in the experiment and the CFX calculation in good agreement. After pump start-up, the main stream of the coolant entering the inlet nozzle is divided into two parts flowing right and left around the core barrel. A part of the flow is moving downwards from the inlet nozzle with lower velocity entering the core at later time (see also the downcomer snapshots of the CFX calculation in Fig. 3.3-6). The CFX calculation shows qualitatively the same behaviour, all main effects are to be seen, too. However, the deboration front is somewhat smoothed over the core inlet.

Fig. 3.3-5 shows the transient course of the maximum value of the mixing scalar at the core entry. The experiment was repeated several times to average over turbulent fluctuations of the velocity field observed in earlier experiments [27]. The results of the single realisations were used to perform a statistical analysis of the experimental data. The confidence-interval of two standard deviations (95.4 %) is shown, too. That maximum value or the minimum boron concentration is an indicator for possible reactivity insertion during a transient. In the experiment as well as in the calculations, the maximum value is determined at each time step over all fuel element positions. Therefore the position can vary, which has also an influence on the width of the confidence-interval of the experimental data. The maximum is reached in the experiment and the calculation at the same time. In the later part of the transient, the calculated concentration deviates more from the measured one, but is always inside the confidence interval.
4. CONCLUSION

Safety evaluation for boron dilution events like small leaks with multiple system failures or loss of the residual heat removal system during refueling and maintenance should be based on advanced analytical methods and findings from latest experiments.

Thorough analysis of such events requires a system of computer codes that comprises an advanced thermal-hydraulic system code, a neutron kinetics code with three-dimensional core simulation coupled with the thermal-hydraulic code, and a CFD-code for three-dimensional mixing calculations.

The code systems available in Germany are capable of performing the necessary analyses for safety evaluation. Uncertainties in prediction of boron dilution scenarios are further reduced by dedicated improvements of code models and validation on suitable experiments.

The present series of integral system tests in the PKL facility using boric acid in the coolant are of high value for a full understanding of the thermal hydraulic system behaviour, especially the reflux-condenser mode and the restart of natural circulation. Test results should now be analysed by system codes.

Experiments in the transparent 1:5 ROCOM test facility greatly enhanced our understanding of mixing processes in the reactor pressure vessel for various loop flow combinations. They should be supplemented by additional tests in the low flow region typical for natural circulation. The test results constitute a valuable data base for CFD-code validation.

The boron dilution issue is a convincing example how R&D effort can respond timely to the resolution of a safety topic.

REFERENCES


[22] T. Höhne: Comparison of Coolant Flow and Mixing in a Scaled Model of the PWR KONVOI with the Processes in the Original Reactor (in German); Report FZR-210, Rossendorf, Germany, 1998
Figures

**Fig. 2.1.1-1:** Nodalization of test sample

**Fig. 2.1.1-2:** Spatial boron concentration with different transport models
Fig. 2.1.2-1: Deflection of hot leg ECC injection during a small leak

Fig. 2.1.2-2: Limitation of ECC water flowing to the UP as function of the counter current warm water flow

\[ J_{ECC>UP}^* = \frac{\dot{m}_{ECC>UP}}{A_I \sqrt{g D \rho_{ECC} \Delta \rho}} \]

\[ J_{HL,I}^* = \frac{\dot{m}_{HL,I}}{A_I \sqrt{g D \rho_{UP} \Delta \rho}} \]

\[ \Delta \rho = \rho_{ECC} - \rho_{UP} \]

\[ A_I = \alpha A \]
Fig. 2.3-1: UMCP-Thermocouple positions

Fig.2.3-2: Inlet mass flow and temperature transient
Fig. 2.3-3: Numerical grid

Fig. 2.3-4: Temperature distribution after 20 s
Fig. 2.3-5: Transient averaged temperature on level 4

Fig. 2.3-6: Azimuthal, local temperature on level 4
Volume: 1 : 145
Elevations: 1 : 1
Max. Pressure: 45 bar
Max. Power: 2,5 MW (10%)

1 Reactor Pressure Vessel
2 Downcomer
3 Steam Generator
4 Pump
5 Pressurizer

Fig. 3.1.1-1: PKL III test facility primary side
Fig. 3.1.1-2: PKL III plan view
Fig. 3.1.2-1: Initial conditions for test PKL III E1.1

- Primary water: $p_{\text{prim}} = 40.0$ bar
- Secondary steam: $p_{\text{sec}} = 38.0$ bar
- Break: (40 cm$^2$)
- Water/steam mixture: 3500 ppm
- Low borated water: ~ 50 kg/loop
- Other concentrations: < 50 ppm
Fig. 3.1.2-2: Main results of PKL test E1.1
Fig. 3.1.2-3: Mixing by startup of circulation in different U-tubes at different times

Fig. 3.2-1: UPTF-TRAM C3 Run13a: Mixing of different borated (different temperature) water flows in the RPV-downcomer
Fig. 3.3-1: RPV plexiglas® model

Fig. 3.3-2: Test facility ROCOM

Fig. 3.3-3: Positions of the wire-mesh sensors in the test facility
Fig. 3.3-4: Normalized coolant velocity and created slug during the experiment

Fig. 3.3-5: Maximum of the mixing scalar at the core entry in comparison between experiment, and CFX-calculation (with confidence-interval of 95.4 %)
<table>
<thead>
<tr>
<th>Time [s]</th>
<th>15.0</th>
<th>17.0</th>
<th>19.0</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Inlet Experiment</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core Inlet CFX-calculation</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Time / s</th>
<th>Downcomer</th>
<th>Time / s</th>
<th>Downcomer</th>
<th>Time / s</th>
<th>Downcomer</th>
</tr>
</thead>
<tbody>
<tr>
<td>8.0</td>
<td><img src="image1" alt="Image" /></td>
<td>9.0</td>
<td><img src="image2" alt="Image" /></td>
<td>10.0</td>
<td><img src="image3" alt="Image" /></td>
</tr>
<tr>
<td>11.0</td>
<td><img src="image4" alt="Image" /></td>
<td>12.0</td>
<td><img src="image5" alt="Image" /></td>
<td>13.0</td>
<td><img src="image6" alt="Image" /></td>
</tr>
<tr>
<td>14.0</td>
<td><img src="image7" alt="Image" /></td>
<td>15.0</td>
<td><img src="image8" alt="Image" /></td>
<td>16.0</td>
<td><img src="image9" alt="Image" /></td>
</tr>
</tbody>
</table>

Fig. 3.3-6: Transport of the deboration front through the downcomer (CFX) and the core inlet (Experiment and CFX)