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Solving Thermal Hydraulic Safety Issues for Current and New Pressurised Water Reactor Design Concepts

Primary Coolant Loop Test Facility (PKL2) Project – Final Report

JT03425550

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ACKNOWLEDGEMENTS, CONTRIBUTORS AND CREDITS

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PKL III G – Project

The tests, inclusive of evaluation and documentation, within the PKL III G test series have been performed by AREVA NP.

The personnel involved are as follows: K. Umminger, project management; R. Mandl, W. Stelzer, B. Reinhold, process and systems engineering, and test specification; H. Kremin, R. Güneysu, E. Bechtgold, P. Reghenzani, W. Schatz, mechanical engineering and performance of tests; P. Burwitz, W. Heumann, K. Endres, instrumentation and control, data acquisition; L. Dennhardt, B. Schoen, S.P. Schollenberger evaluation and documentation; G.-J. Seeberger, analytical pre- and post-test calculations.

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Special thanks are hereby expressed to the authors that provided valuable contribution to the data interpretation and application (DIA) Chapter of the report (for contributions see Chapter 5).

PKL2 Project

Since April 2008, the Primärkreislauf (PKL) Project has been carried out within the collaborative programme PKL2 which was initiated by the NEA with the following international participants: BEL V jointly with Suez-Tractebel SA, Belgium; Nuclear Research Institute, Czech Republic; Valtion Teknillinen Tutkimuskeskus and Säteilyturvakeskus, Finland; Commissariat à l'énergie atomique et aux énergies alternatives, Institut de radioprotection et de sûreté nucléaire, Électricité de France, France; Technische Vereinigung der Grosskraftwerksbetreiber, AREVA NP GmbH, Gesellschaft für Anlagen- und Reaktorsicherheit, Germany; Kuratorium für Forschung im Küsteningenieurwesen Atomic Energy Research Institute jointly with the Paks nuclear power plant, Hungary; Japan Nuclear Energy Safety Organisation, Japan; Korea Atomic Energy Research Institute in association with the Korean Institute of Nuclear Safety, Korea; Netherlands Ministry of Housing, Spatial Planning and the Environment, Netherlands; Consejo de Seguridad Nuclear, Spain; Statens Kärnkraftinspektion, Sweden; Paul Scherrer Institut, Switzerland; Health and Safety Executive, United Kingdom; United States Nuclear Regulatory Commission, United States.

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LIST OF ABBREVIATIONS AND ACRONYMS

ACC	Accumulator, *also*: ACCU
AEKI	Atomic Energy Research Institute (Hungary)
BMBF	Federal Ministry for Education, Science, Research and Technology (Germany)
CATHARE	Code avancé de thermohydraulique pour les accidents sur les réacteurs à eau, T/H system code
CCFL	Counter-current flow limitation
CEA	Commissariat à l’énergie atomique et aux énergies alternatives (France)
CET	Core exit temperature
CFD	Computational fluid dynamics
CI	Coolant inventory
CL	Cold leg
CM	Coolant mass
COL	Cross-over leg, *also*: Loop seal
COMBO	Continuous measurement of boron concentration
CRDA	Control rod drop accident
CSNI	Committee on the Safety of Nuclear Installations (NEA)
CT	Cladding temperature
DC	Downcomer
ECC	Emergency core coolant
ECCI	Emergency core coolant injection
ECCS	Emergency core cooling system
EPR	European Pressurised Reactor
FFT	Fast Fourier transform
GRS	Global Research for Safety, Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH (Germany)
HL	Hot leg
HPSI	High-pressure safety injection
HZDR	Helmholtz-Zentrum Dresden Rossendorf, research centre (Germany)
IAEA	International Atomic Energy Agency
I&C	Instrumentation and control system
KONVOI	Pressurised water reactor of German design
LB-LOCA	Leg break loss-of-coolant accident
LOCA	Loss-of-coolant accident
LP	Lower plenum
LPIS	Low-pressure injection system
LPSI	Low-pressure safety injection
LSTF	Large-scale test facility
MB	Management board
ME	Elevation of measurement

MS	Main steam
MSL	Main steam line
MSLB	Main steam line break
MSRCV	Main steam relief control valve
MSRV	Main steam relief valve
NC	Natural circulation
NCI	Natural-circulation interruption
NEA	Nuclear Energy Agency
NPP	Nuclear power plant
OA	Operating agent
OECD	Organisation for Economic Co-operation and Development
PANDA	Passive Nachwärmeabfuhr- und Druckabbau – test facility
PCI	Primary coolant inventory
PCT	Peak (fuel rod) cladding temperature
PKL	Test facility, (German acronym for "Primärkreislauf", meaning: primary circuit, RCS)
PMK	Integral type test facility (Hungary)
PORV	Pressure-operated relief valve
PRG	Programme Review Group
PRZ	Pressuriser
PTS	Pressurised thermal shock
PWR	Pressurised water reactor
RC	Reflux condenser
RCL	Reactor coolant line
RCP	Reactor coolant pump
RCS	Reactor cooling system
RELAP	The Reactor Excursion and Leak Analysis Programme, T/H computer code
RHRS	Residual heat removal system
ROCOM	Rossendorf coolant Mixing test facility, operated by HZDR
ROSA	Rig of Safety Assessment Project
RPV	Reactor pressure vessel
RV	Regulatory valve
RVLIS	Reactor vessel level indicating system
SAMG	Severe accident management or mitigation
SB	Small break
SBO	Station blackout
SETH	SESAR Thermal Hydraulics project (see SESAR below)
SESAR	Senior Group of Experts on Nuclear Safety Research
SESAR/FAP	Senior Group of Experts on Nuclear Safety Research Facilities and Programmes
SLB	Steam line break
SV	Safety valve

SG	Steam generator
SI	Safety injection
SIS	Safety-injection systems
SL	Swell level
SOT	Start of test
TC	Thermocouple
T/H	Thermal hydraulics
TRACE	TH-system code
UPTF	Upper plenum test facility (dismantled large-scale test rig)
VVER	Water-Water Energetic Reactor

Latin and Greek symbols for physical values

1Φ	single-phase	
2Φ	two-phase	
[B]	ppm	Boron concentration
$\dot{m}$	kg/s	Mass flow
p	bar, Pa	Pressure
Δp	bar, Pa	Pressure difference
P	kW	Power
t	s	Time
T	°C, K	Temperature
ρ	kg/m^3	Density

Subscripts

el	Electrical
max	Maximum
min	Minimum
prim	Primary
sec	Secondary
sat	Physical value at saturation conditions

EXECUTIVE SUMMARY

Background

For many years, extensive experimental investigations into the system response of pressurised water reactors (PWRs) under accident conditions have been being conducted at the large-scale Primärkreislauf (PKL) test facility (operated at AREVA NP, Germany) which constitutes a full-height model of the entire reactor cooling system (RCS) and major parts of the secondary side of a PWR.

Since 2001, the PKL Project has been continued in the course of an international project initiated by the Nuclear Energy Agency (NEA) (as recommended in the SESAR report[1]) with the objective of preserving competence in the long term and sufficient infrastructure in the field of reactor safety. The major topics covered by the experiments between 2001 and 2007 within the framework of the SETH and PKL projects were:

- Boron dilution events following small-break loss-of-coolant accidents (SB-LOCA).
- Loss of residual heat removal under shutdown conditions.

The subsequent PKL2 (PKL III G) experimental project considered the needs of the different countries and organisations and thus also reflects topics with high safety relevance according to the SESAR/SFEAR report. Between April 2008 and September 2011 the PKL2 project has been accomplished with the contribution of the following international participants:

- BEL V jointly with Suez-Tractebel SA, Belgium.
- Nuclear Research Institute of the Czech Republic.
- Valtion Teknillinen Tutkimuskeskus and Säteilyturvakeskus (Radiation and Nuclear Safety Authority), Finland.
- Commissariat à l’énergie atomique et aux énergies alternatives, France.
- Institut de radioprotection et de sûreté nucléaire (Institute for Radiological Protection and Nuclear Safety), France.
- Électricité de France, France.
- Technische Vereinigung der Grosskraftwerksbetreiber, Germany.
- AREVA NP GmbH, Germany.
- Gesellschaft für Anlagen- und Reaktorsicherheit, Germany.
- Kuratorium für Forschung im Küsteningenieurwesen Atomic Energy Research Institute (AEKI) jointly with the Paks NPP, Hungary.

1. NEA (2001), *Nuclear Safety Research in OECD Countries- Summary Report of Major Facilities and Programmes at Risk*, OECD, Paris (www.oecd-nea.org/nsd/reports/nea3144-research.pdf).

- Japan Nuclear Energy Safety Organisation, Japan.
- Korea Atomic Energy Research Institute, in association with the Korea Institute of Nuclear Safety, Korea.
- Netherlands Ministry of Housing, Spatial Planning and the Environment, the Netherlands.
- Consejo de Seguridad Nuclear, Spain.
- Statens Kärnkraftinspektion, Sweden.
- Paul Scherrer Institut, Switzerland.
- Health and Safety Executive, United Kingdom.
- Nuclear Regulatory Commission, United States.

Objective of the work

The general objective of the PKL experiments is to contribute to a better understanding of the sometimes highly complex thermal-hydraulic processes involved in various accident scenarios and to allow a better assessment of the countermeasures implemented for accident control and the demonstration of safety margins available in the plants. In addition, the experimental results aim at the application in the validation and further development of thermal-hydraulic computer codes, also called system codes.

The PKL2 experimental project comprised eight integral experiments at the PKL test facility with a total of twelve test runs. Additional tests in the PMK and Rossendorf Coolant Mixing (ROCOM) test facilities complement the PKL experiments, either by consideration of scenario-relevant specifics of a VVER-type PWR (e.g. horizontal steam generators – SG) in the case of PMK (replication of a VVER 440/213 PWR) or by separate effect tests in the ROCOM test facility with respect to mixing phenomena in the reactor pressure vessel (RPV) downcomer (DC) and lower plenum (LP).

The PKL2 tests investigated safety issues relevant for current PWR plants as well as for new PWR design concepts and focused on complex heat transfer mechanisms in the SGs under postulated accident situations.

The first scenario G1 included tests addressing the heat transfer mechanisms in the SGs in the presence of nitrogen, steam and water, in both vertical (PKL tests) and horizontal (PMK tests) SGs. Asymmetric cool-down procedures in the case where SGs have partly dried out on the secondary sides have been covered by scenario G2. Further tests addressed heat transfer between the primary and secondary sides, either with respect to overcooling transients on the primary side resulting from a main steam line break (MSLB) and completed by tests on mixing of hot and cold water in the RPV DC and the LP in the ROCOM test facility (G3), or – in a more generic approach – with a focus on reflux-condenser (RC) condition in the form of parameter studies (G4).

Further topics in the PKL2 project addressed boron precipitation processes in the core following large-break loss-of-coolant accidents (G5), the upper head void behaviour during cool down under off-normal condition (G6) and the effectiveness of countermeasures for the mitigation of multiple- failure beyond-design-basis course of events following small-break loss-of-coolant accidents (SB-LOCA) (G7).

In detail:

- **G1 – Failure of residual heat removal system (RHRS) under cold shutdown conditions (investigation on heat transfer in the SG)**

  Loss of residual heat removal in ¾ loop operation with closed RCS can lead to RC-like conditions effectuating a continuous boron depletion of individual sections of the reactor coolant system as a result of coolant being transported from SG inlet-to-outlet-side. The previous test programmes SETH and PKL (specifically test PKL E3.1 and test series PKL F2, have already identified a range of heat transfer mechanisms and basic coherences involving the inlet-to outlet-side transport of coolant with different effects in transient tests. Test series G1 comprises systematic studies on heat transfer in SG U-tubes in the presence of nitrogen (N_2), steam and water, to be used for the support of conclusions on plant operation following loss of RHRS as regards stabilisation of primary pressure and the prevention/mitigation of boron dilution, but primarily to supply a detailed data basis for the validation of T/H Codes on the relevant T/H phenomena.

  Particularities of the scenario in a VVER PWR: Characterisation of VVER SG heat transfer during preparation for core unloading

  During shutdown, the core cooling in a VVER is not assured by primary side RHRS – as in PWRs –, but by heat removal through the SG with natural circulation (NC) on the primary side and the secondary side filled with water. Systematic investigation of the degradation of SG heat transfer and of the disturbance of NC during lowering of the primary system level in preparation of core unloading is of high safety importance.

- **G2 – Cool down under asymmetric boundary conditions (i.e. isolated SGs)**

  The background for the G2 test is a scenario in a PWR that requires a reactor cool down via an only partially available secondary side (two isolated SGs), that is, under asymmetric conditions. Such a process could be required, for example, after a feed water or MSLB as well as for a SG U-tube break, with simultaneous loss of off-site power, in either case. By means of the G2 test here, the influence of, in particular, isolated SGs with boiled-off secondary sides on the unit cool down and the natural-circulation behaviour is systematically investigated. The objective of test G2.1 was to clarify whether NC is maintained in loops with isolated and boiled-off SGs at a continuous cool down (gradient of 50 K/h), with and without additional primary-side pressure reduction by spraying with the volume control system and, if not, whether NC can be maintained by a stepwise cool down.

  The main purpose of the G2.1 test was also the supply of a detailed data basis for the validation of T/H Codes on the relevant T/H phenomena.

- **G3 – Cold-water transients following MSLB**

  The transient behaviour resulting from a non-isolable break in the main steam line (MSL) of a SG was investigated in this test. The influence of the MSLB on the primary-side system conditions during the boiling-off phase of the affected SG and the subsequent primary-side injection from two safety-injection pumps were of particular interest. Non-isolable MSLBs (up to a double-ended break) cause a rapid decrease in secondary-side pressure in the affected SG and an overcooling transient in the associated loop on the primary side. Safety issues arise from the possibility of a re-criticality within the core and the resulting power excursion in a part of the core due to the entry of cold water into the reactor core area, and from a possible pressurised thermal shock (PTS) at the RPV inlet nozzles in case of safety-injection system start after reaching the activation criteria due to a rapid volume contraction on the primary side.

The results of the PKL test have been used as boundary conditions for the ROCOM experiments on mixing in the RPV down comer to determine the temperature distribution in the core inlet plane for an assessment of the prospect of re-criticality in the core.

· **G4 – Influence of secondary-side parameters on heat transfer under RC condition**

RC conditions may occur in the course of SB-LOCA only if safety-injection systems (SIS) are operating under reduced availability. Accordingly, the level in the RCS can decrease temporarily to the point that the decay heat is transferred to the secondary side under RC condition (provided the secondary-side heat sink is available).

Test G4.1 was primarily designed against the background of a need for an extension of the data base on heat transfer to secondary side under RC conditions as a function of the secondary-side parameters (secondary-side water level and cool-down rate).

· **G5 – Boron precipitation in the RPV following leg break loss-of-coolant accidents (LB-LOCA)**

The test scenario is set in the long-term cooling phase following a large cold LB-LOCA with assumed continuing boiling in the reactor in the long-term. During continuous evaporation, the boron (B) injected with the emergency core coolant (ECC) water remains in the liquid phase in the core region and accumulates continuously. At B values which exceed the solubility limit of boron, locally initiated crystallisation then effectuate the formation of solid boron particles which may influence the core cooling capabilities. The objective of test G5.1 was to provide information and detailed data on the relevant processes and phenomena such as the size of the mixing volumes in the RPV, the speed of the boron enrichment process and the hot leg (HL) ECC flow required to “flush” the core (i.e. a reversion of the precipitation process).

· **G6 – Formation and behaviour of upper head void during cool down**

The main objectives of test G6.1 were to demonstrate the progress of the upper head void growth under cool down in emergency power mode (i.e. under NC). A void growth in the RPV that extends into the HL may form a possible complication of the cool-down process and switch over to RHRS operation. G6.1 was designed to record the physical principles involved in the upper head void growth in detail and to supply a data basis for code validation or model development on the observed phenomena.

· **G7 – Effectiveness of secondary-side depressurisation to prevent a core-melt scenario during a SB-LOCA transient with additional system failures**

The main test objectives comprise questions on the efficiency of the accident management (AM) procedures (SG depressurisation followed by accumulator injection) in a temporal restoration of the secondary-side heat sink to re-establish heat removal from primary side and to initiate a primary-side pressure decrease, as well as on the differences between core exit temperature (CET) and maximum measured cladding temperature (CT) as a function of the boundary conditions. The G7.1 test was designed as a counterpart test between the PKL and ROSA/LSTF test facilities.

Main conclusions on the significance for nuclear safety and recommendations

The Programme Review Group (PRG) has reviewed the test results with respect to their relevance for the operation of PWRs or the solution of safety issues. The following conclusions have been obtained:

G1: It was the common consensus that the experimental data corresponding to the heat transfer to the secondary side in the low-pressure range, condensation and transport of condensate under RC conditions, NC, swell levels (SLs) formation and relocation of the coolant masses (CM) inside the primary side are thermal-hydraulic principles which are expected to occur also in a PWR and are thus useful for the evaluation of real PWR T/H behaviour. G1 test series demonstrated the coherence between rising primary side coolant inventory and the increasing level of primary pressure required for the system to self-stabilise.

As regards the particularities of a VVER PWR plant, the PMK test results allow the conclusion that in case of a loss of primary inventory in a VVER-440 plant during lowering of the primary system level in preparation of core unloading, the operator has to close all venting lines before starting the refill process. This would help to resume NC, which might have been interrupted by coolant loss, and assure core outlet temperatures well below saturation ones. Only after having achieved these conditions should the operator vent air trapped in vessel head, SG collectors and pressuriser, in order to return to normal conditions.

G2: Test G2.1 proved that a stepwise cool down (or a reduced mean cooling rate) process maintains NC in all loops provided that the magnitudes of the cool-down steps and cool-down gradients are adequately controlled. However, this stepwise cool-down procedure cannot be directly transposed to most PWRs. Therefore, it could be interesting in the future to look for strategies that allow avoiding natural-circulation interruption (NCI) and recovery if NCI occurs for some operating procedures, as for instance in the event of a SG U-tube rupture (SGTR), that need to impose a large cool-down rate in order to eliminate the break flow rate. Therefore, further investigations in order to assess the impact of key parameters such as the (continuous) cool-down rate, the decay heat level and the mixing in the DC on the proposed cool-down strategy remain to be investigated.

G3: The fast cool-down tests performed in both the PKL and the ROCOM facilities complete the databases for the validation of thermal-hydraulics system codes (validation against heat transfer in the affected SG and influence on loop flow rate), and of computational fluid dynamics (CFD) codes (buoyancy-driven mixing patterns in the RPV). With respect to the re-criticality aspect of the scenario, the ROCOM experiments demonstrated the effectiveness of mixing in the RPV DC, a sector formation in the core inlet plane was not observed.

G3.1 was selected by the PRG and Management Board (MB) for performing a common analytical activity among participants to demonstrate the codes’ capacities to deal with the relevant phenomena involved in the scenario. The results of this benchmark exercise which included a quantitative assessment of code performances based on accuracy evaluation of their results have been issued in a separate report.

G4: The G4.1 tests provided details on the effectiveness of RC operation in the RCS in heat transport from primary to secondary sides under different secondary-side operating conditions (variation of secondary-side fill level and cool-down rate). The test demonstrated the preservation of heat transport from primary to secondary sides even for minimum secondary-side fill level, as long as the feed-water flow remains intact. For the variation of the cool-down rate it was demonstrated that even for cool-down rates of up to 500 K/h the primary-side pressure remained closely coupled to the secondary pressure, the core cooling was never compromised, even for initial primary coolant inventories of approx. 40%.

G5: One of the objectives in performing test G5.1 was to evaluate the likelihood of occurrence of this upwardly-oriented, large-split-break scenario; and after taking extraordinary measures to facilitate its manifestation, the test results showed that the scenario, as theorised, could not be replicated with the current PKL configuration. Even after directly injecting very low emergency core cooling system (ECCS) flow into the loop seals and blocking all vessel bypass flow paths, the flow resistance across the loops was not large enough to block the loop seals and cause a core depression. The liquid level in the vessel consistently exceeded the core region and progressed towards the SGs. As a result of the large mixing volume the boron precipitation progressed only slowly.

In further investigations on the topic a modification of the PKL test rig to impose considerably higher loop flow resistances may be employed to realise a parameter study on the dependence of swell-level depression on loop flow resistances and its consequences for the mixing volumes and progress of the boron enrichment in the core.

G6: Test G6.1 showed that during cool down under NC condition the temperature distribution in the upper plenum (UP) and RPV dome strongly affects the void growth in the RPV dome. In the PKL test sub-cooled NC present at the core outlet and in the upper plenum limited the void growth at about 0.5 m above the reactor coolant line (RCL) upper edge within the upper plenum. The upper head void persisted under NC-conditions and condensed only slowly, due to RPV dome heat losses.

G7: The test demonstrated the effectiveness of a secondary-side depressurisation in the restoration of the heat removal from the primary side and in the initiation of an accordingly fast primary-side pressure reduction. In addition to identifying and verifying relevant phenomenology of a PWR to a SB-LOCA (e.g. loop seal formation and clearing, eventual core uncovery, mass distribution in the RCS, core heat up, reflooding, accumulator performance), the G7.1 helped in the discussion on adequacy of CET measurements as an indication of core temperatures to trigger AM actuations.

By means of a suitable comparison of ROSA and PKL, as well as T/H Code results, not only a confirmation of the expected value of the counterpart test may be achieved, but the outcomes of this analysis will also be helpful to support the steps involved in designing integral plant model qualification procedures and uncertainty evaluation methodologies.

The collaboration on an international scale proved to be a very effective way to preserve test facilities and know-how on the one hand, and to solve PWR safety issues in close collaboration among operating agents of test facilities and T/H code users on the other hand. A further intensification of co-operation of individual test facilities by means of counterpart and complementary testing may be of use for the deduction of scaling effects or conclusions on safety relevant issues for different plant designs and geometries.

Applicability of experimental results to real plant conditions

The data application and interpretation (DIA) chapter featuring a review of the PKL experimental results with respect to safety relevance, scaling value and lessons learnt from the execution of experiments forms an essential part of the final report.

Experimental results obtained in the PKL2 project extended the previous reference database with a valuable collection of data useful for the assessment of current thermal-hydraulic codes and for code development. As in the previous PKL project, PKL2 participants have kept an active involvement and promotion of analytical activities during the execution of the project with respect to the understanding of important thermal-hydraulic phenomena/processes and the predictive capabilities as well as the strengths and limitations of existing tools.

Apart from the fact that most of the experiments addressed code validation or benchmark activities some test results are representative of PWR behaviour and may allow conclusions on plant performance and operation for the particular scenario.

The PKL integral test facility is well suited to experimentally investigate PWR behaviour under various conditions such as natural circulation (NC) (G2, G3, G6), reflux condensation (G4, G7) – also in presence of non-condensable gases (G1) – or scenarios related to boron enrichment processes following LB-LOCA (G5). In particular, the mechanisms of heat transfer in the SG U-tubes are considered representative of PWR thermal-hydraulic behaviour due to the PKL design features, its scaling and the possibility to employ boric acid and appropriate measuring techniques.

The interaction of experiments and system codes could be useful to obtain qualified results for PWRs. For instance, some experimental results discussed in the report and adopted for transfer to PWR scale can be compared with corresponding analyses using thermal-hydraulic system codes and, for some local phenomena, with additional separate effect tests. Of course, the transfer of PKL tests to real plant condition in general is facilitated in case of results being applied to a German KONVOI PWR, as the PKL test facility replicates the German PWR design, but with the support of adequate analytical studies the conclusions drawn from the PKL experiments will also lead to qualified results for PWR designs similar to KONVOI.

1. INTRODUCTION

1.1 Background and purpose of the PKL2 Project

In 1997, the Committee on the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency (NEA) set up an international working group whose aim was to identify how existing expertise in the field of reactor safety and an appropriate experimental infrastructure could be sustained in the future. One of the main tasks of this Senior Group of Experts on Nuclear Safety Research Facilities and Programmes (SESAR/FAP) was to identify those test facilities and research programmes which were threatened by closure in the next years and to select facilities and programmes which, if they were able to continue through involvement in NEA projects, would be of particular benefit to the member countries [1]. In selecting these experimental facilities, the group not only based their decisions on the technical capabilities provided by a specific facility but also assessed the competence of the team working there, including the way in which use was made of test results for analytical applications; e.g. for code validation.

As early as 1998, SESAR/FAP presented CSNI with a first set of results which not only comprised general and strategic recommendations but also proposed actions for immediate or near-term implementation. The safeguarding of integral test facilities for studying thermal-hydraulic issues was one of the actions assigned top priorities, provided test programmes with unquestionable scientific interest are proposed. At its annual meeting in December 1999, CSNI issued the recommendation that an international collaborative project be set up in the field of thermal hydraulics (T/H) to implement the recommendation made to this effect by SESAR/FAP. Based on programmes and time and cost schedules proposed by the various companies operating the test facilities, a proposal for a project called SETH (SESAR Thermal Hydraulics) was elaborated in consultation with the NEA Secretariat and was submitted to the member countries in mid-2000. The proposal, which was based on experiments to be conducted in the Passive Nachwärmeabfuhr- und Druckabbau (PANDA) and Primärkreislauf (PKL2) test facilities, was approved by the project partners within NEA.

The PKL tests conducted at AREVA NP's PKL test facility [2, 3, 4, 5, 6] in Erlangen, Germany, between April 2001 and March 2007 in the frame of the **SETH (PKL III E**) [7] and **PKL (PKL III F)** [11] projects addressed the pressurised water reactor (PWR) issues of inherent boron dilution [8, 9] after occurrence of either a small-break loss-of-coolant accident (SB-LOCA) or loss of the residual heat removal system (RHRS) under cold shutdown condition (¾-loop operation) [10]. At the end of the PKL project all questions of interest concerning the system behaviour and important phenomena related to post-SB-LOCA boron dilution had been understood. In contrast, for the course of events evolving from failure of RHRS under cold shutdown conditions (¾-loop operation), new phenomena concerning heat transfer in the steam generators (SG) in presence of nitrogen could be identified on the basis of the PKL experimental results. Some specific processes such as the transitions between the different flow patterns in the SG tubes and their consequences for safety relevant aspects such as boron dilution had not yet been understood in detail. In order to close the remaining questions on this subject and to solve other open issues for current PWR plants as well as for new PWR design concepts, a programme proposal for a new NEA project on PKL has been worked out in co-operation with the national and international partners.

The new **PKL2 (PKL III G)** experimental project presented with the report at hand considered the needs of the different countries and organisations and thus also reflected topics with high safety relevance according to the SESAR/SFEAR report [12]. In this report the PKL facility is mentioned among those to be maintained and that have unique capabilities and high relevance to the resolution of current safety issues (indicated by its high numerical ranking) as well as the potential to be highly relevant in support of the resolution of advanced PWR safety issues.

The PKL2 experimental project comprised eight integral experiments with a total of twelve test runs in the PKL test facility covering the following topics:

- **G1**: Heat transfer in the SG (failure of RHRS under ¾-loop operation).
- **G2**: Cool down under asymmetric boundary conditions (i.e. isolated SGs).
- **G3**: Cold-water transients following main steam line break (MSLB).
- **G4**: Influence of secondary-side parameters on heat transfer under reflux-condenser (RC) condition.
- **G5**: Boron precipitation in the reactor pressure vessel (RPV) following leg break loss-of-coolant accidents (LB-LOCA).
- **G6**: Formation and behaviour of upper head void during cool down.
- **G7**: Effectiveness of secondary-side depressurisation, performance of the core exit temperature (CET) measurement for the evaluation of the core cooling status.

In addition, the PKL test on heat transfer mechanisms and fast cool-down transients MSLB have been complemented by experiments in the PMK test facility with respect to VVER behaviour and at the **Ro**ssendorf **Co**olant **M**ixing (ROCOM) test facility on coolant mixing in RPV downcomer (DC) and lower plenum (LP), respectively.

1.2 Test facilities

1.2.1 PKL III test facility

The large-scale test facility PKL (Figures A1-A3 and [2-6]) is a model of a PWR of KWU-design of the 1 300 MW class. Reference plant is the Philippsburg 2 NPP. 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 modelled by the scaling factor 1:145.

As for other test facilities of this size, the scaling concept aims to simulate the thermal-hydraulic system behaviour of the full-scale power plant. The following features serve to meet this requirement:

- Full-scale hydrostatic head.
- Power, volume, and cross-sectional area scaling factor of 1:145.
- Full-scale frictional pressure loss for single-phase flow.
- Simulation of all four loops with identical piping lengths.
- Core and SGs are simulated as a “section” from the actual systems, in other words, full-scale rod and U-tube dimensions, spacers, heat storage capacity are used; the numbers of rods and tubes are scaled down.

- In cases of conflicting requirements, simulation of the phenomena was given preference over consistent simulation of the geometry, e.g. in order to account for important phenomena in the HLs such as flow separation and counter-current flow limitation, the geometry of the HLs is based on conservation of the Froude number and was finally designed on the basis of experiments at the full-scale Upper Plenum Test Facility (UPTF).
- The RPV DC is modelled as an annulus in the upper region and continues as two stand pipes connected to the LP. This configuration permits symmetrical connection of the 4 cold legs to the RPV, preserves the frictional pressure losses and does not unacceptably distort the volume/surface ratio.

PKL is worldwide the only test facility with four identical reactor coolant loops arranged symmetrically around the RPV. This configuration permits accidents to be investigated under realistic conditions, including those accidents characterised by non-symmetrical boundary conditions between the loops. Modelling of a 3-loop plant is possible by simply isolating one loop. Each loop is equipped with an active reactor coolant pump with speed controllers to enable any pump characteristics to be reproduced. Under natural-circulation conditions (i.e. reactor coolant pumps – RCPs – not in operation) the flow resistance of blocked pumps is simulated.

The reactor core is modelled by a bundle of 314 electrically heated rods with a maximum core power of 2.5 MW which is equivalent to 10% of nominal rating. Each of the 4 SGs is equipped with 30 U-tubes of original size and material. Allowance has been made for the differing elevations (1.5 m) between the tubes with the smallest and largest bending radius.

As the functions of all major primary and secondary operational and safety 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 1 300 measuring points, the PKL facility is extensively instrumented, something which permits detailed analysis and interpretation of the phenomena observed in the tests. Besides conventional measurements for temperature, pressure and mass flow rates, also special measurement techniques for the determination of the boron concentration (see Section 1.4) were used for the experiments described in this report.

The maximum operating pressure of the PKL test facility is 45 bar on the primary side and 56 bar on the secondary side. Due to this pressure limitation, it is not possible to simulate the high-pressure portion of accident sequences (such as small-break LOCAs) starting from a PWR's actual operating pressure (155-160 bar) under original conditions. Hence, the PKL tests "start" at a primary system pressure of 45 bar and with initial conditions corresponding to those that would prevail in a real plant at this time (i.e. when the primary system pressure is at this level). These initial conditions are obtained from analyses conducted using system codes (such as the Reactor Excursion and Leak Analysis Program (RELAP) 5) for a real PWR geometry and corresponding boundary conditions and are realised within a so-called conditioning phase. The remainder of the accident sequence, where the most relevant phenomena are expected to occur (e.g. for the small break (SB) LOCA tests described here: refilling, onset of natural circulation (NC) and transport of low-boron water in the direction of the RPV) is then simulated in the tests using real PWR pressures. Accidents scenarios which would occur in the PWR under shutdown conditions (e.g. loss of residual heat removal system during mid-loop operation) are simulated in the PKL test facility under original pressure conditions.

The PKL test facility was designed, built and commissioned by Siemens/KWU (now AREVA NP) in the seventies. At that time reactor safety research was centred above all on the theoretical and experimental analysis of LB-LOCA, focusing on verifying the effectiveness of the emergency core cooling system (ECCS) required for controlling these accidents. In line with this original objective and considering topical issues, Siemens/KWU carried out the first PKL tests in the years from 1977 to 1986 in the course of the projects PKL I and PKL II which were sponsored by the German Federal Ministry for Education, Science, Research and Technology (BMBF).

The PKL III project, which was started subsequently, had the main goal of investigating experimentally the thermal-hydraulic processes on the primary and the secondary side of a 1 300 MW PWR of KWU-design during various accident scenarios with and without loss-of-coolant. Within the scope of this project Siemens/KWU conducted tests concerning the investigations of transients from 1986 to 1999 with financial support from both the German Utilities operating PWRs[2] and the BMBF and the BMWi (German Federal Ministry for Economics and Technology). One focus of these activities was on the effectiveness of beyond-design-basis accident management-measures being initiated manually by the operators. These measures for accident mitigation were theoretically analysed within the German Risk Study, Phase B [13].

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. In addition, they have supplied valuable information regarding safety margins available in the plants. 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, also called system codes. These codes employed in designing and licensing NPPs have to be validated beforehand.

1.2.2 ROCOM test facility

The ROCOM test facility models the primary circuit of a German KONVOI-type PWR in a linear scale of 1:5 [37, 38]. The test facility was designed for the investigation of a wide spectrum of coolant mixing scenarios inside the primary circuit of a KONVOI-type PWR. The RPV was manufactured from acrylic glass and forms the main part of the test facility (Figure A4). The geometrical similarity between the model and the original reactor is fully respected within the region in between the bends in the cold legs, which are closest to the reactor inlet and to the core entrance. The geometry of the inlet nozzles with their diffuser segments and the curvature radius of the inner wall at the junction with the pressure vessel were modelled in detail. Similarity is also taken into account for the core support plate with the orifices for the coolant. The KONVOI reactor has a perforated sieve drum (flow skirt below the core barrel), which is also placed in the LP of the vessel in the ROCOM test facility (Figure A4).

ROCOM is a four-loop test facility with individually controllable pumps in each loop, which enables the possibility of performing tests over a wide range of flow conditions, from NC to nominal flow rates and this includes the use of ramped flow rate changes to mimic normal or natural operation conditions (Figure A4).

The design parameters of the test facility are presented in the Table C2 together with the data of the original reactor.

The facility is operated with de-mineralised water at room temperature. 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 a de-borated or an undercooled slug of coolant. The distribution of this tracer in the test facility is measured by special wire-mesh electrical conductivity sensors developed by Helmholtz-Zentrum Dresden

2. In 1995 the project was joined by the nuclear power plants Gösgen (Switzerland) and Trillo (Spain).

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 realised 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 1 000 Hz and are subsequently compared to reference values in order to estimate the position of the labelled volume and its transport. The result is a dimensionless mixing scalar that characterises the instantaneous share of the coolant originating from the labelled volume at a given position inside the flow field. Based on the boundary conditions the mixing scalar can be converted into a boron concentration or temperature value.

Wire-mesh sensors can be installed at different positions inside the test facility. One sensor is integrated into the core support plate just below the fuel element inlets in such a way that one measurement position for each fuel element inlet is available (Figure A4). In the DC two axial sensor planes are available each spawning a measuring grid of 64*29 positions around the whole DC. A third sensor type is available for measurements in the cross-section of the loops. In the current experiments between one and six of such sensors were installed in dependence on the investigated scenario (Figure A4).

1.2.3 PMK test facility

The PMK-2 facility (Figure A5) is a scaled down model of the Paks NPP and it was primarily designed for investigating small-break loss-of-coolant accidents (SB-LOCA) and transient processes of VVER-440/213 plants. The Paks NPP is equipped with VVER-440/213-type reactors. Such plants are slightly different from PWRs of usual design and have a number of special features, viz.: 6-loop primary circuit, horizontal SGs, loop seal in hot and cold legs, safety-injection tank (SIT) set-point pressure higher than secondary pressure, the coolant from SITs is directly injected to the upper plenum and DC, residual heat is removed via the SGs. The main aim of the facility was to investigate the consequences of these differences on the transient behaviour of the system.

The flow diagram of the PMK-2 facility is shown in Figure A5 in the Appendix.

The volume and power scaling of PMK-2 are 1:2070. Transients can be started from nominal operating conditions. The ratio of elevations is 1:1 except for the LP and pressuriser. The six loops of the plant are modelled by a single active loop. In the secondary side of the SG the steam/water volume ratio is maintained. The coolant is water under the same operating conditions as in the NPPs.

The main characteristics of the PMK-2 facility are given in Table D1 in the Appendix.

The core model consists of 19 electrically heated rods, with uniform power distribution. Core length, elevation and flow area are the same as in the Paks NPP.

In the modelling of the SG primary side, the tube diameter, length and number were determined by the requirement of keeping the 1:2070 ratio of the product of the overall heat transfer coefficient and the equivalent heat transfer area. The elevations of tube rows and the axial surface distribution of tubes are the same as in the reference system. On the secondary side the water level and the steam to water volume ratios are kept. The temperature and pressure are the same as in the NPP.

An automated data acquisition system collects signals of the over 100 measurement points, generally with a frequency of 1 Hz. The list of measurements includes the following parameters: coolant and structure temperatures (the most important ones being the cladding temperatures (CTs) of the core model), absolute and differential pressures, liquid levels, flow rates (including break flow).

A large number of void probes allow tracing of the swell level (SL) at different locations of the primary circuit. About half of them are coupled with micro-thermocouples, these permit to follow the propagation of non-condensable gases in the primary circuit.

1.3 Accident scenarios and crucial phenomena under investigation

1.3.1 Failure of RHRS during ¾-loop operation

In the recent years, an increasing attention has been paid to the accident scenarios in shutdown states; indeed, probabilistic safety assessments have shown that they account for an important contribution to the overall core damage frequencies, especially regarding the duration of these operating conditions. Among these accidents, the loss of RHRS has been investigated in numerous studies.

Analyses by the GRS have furthermore indicated that operating the secondary side[3] as an alternative heat sink following the loss of the residual heat removal system could lead to inherent boron dilution.

Loss of residual heat removal in ¾ loop operation with closed reactor coolant system (RCS) can lead to reflux condenser-like conditions (see Figure B1, Case 1). Individual sections of the RCS can become boron-depleted as a result of coolant being transported from SG inlet-to-outlet-side. The previous test programmes PKL III E and F (specifically test PKL III E3.1 [7, 15] and test series PKL III F2, [11, 16, 17] and corresponding preliminary work relating to these tests) have already identified a range of heat transfer mechanisms and basic coherences involving the inlet-to-outlet-side transport of coolant with different effects (see Figure B1).

- **Slugs of sub-cooled water in all SG tubes with nitrogen enclosed in the tube bends above (see Figure B1, Case 4)**. The tall columns of sub-cooled condensate reduce the heat transfer zone to the bottom part of all tubes. Degraded heat transfer capacity in the SGs leads to relatively high RCS pressure. Coolant or steam cannot overflow to the outlet side of tubes containing water columns with nitrogen above so there is no prospect of boron dilution here (see PKL III E3.1, [15]).

- **Intermittent spillover of boron-depleted coolant through individual short SG tubes (see Figure 4, Case 3)**. Following the onset of intermittent spillover, slugs of boron-depleted coolant are transported over the apex of individual tubes because the condensation rate exceeds the coolant mass (CM) flow spillover. Equilibrium pressure on the RCS side is moderate and constant at 4-5 bar. This process leads to continuous boron dilution starting below the SG outlet tubes (see PKL III E3.1, [15]; PKL III F 2.1 Run1, [16]; PKL III F2.2 Run1, [17]).

- **Continuous circulation in some short tubes (see Figure 4, Case 5)**. There is sufficient reactor coolant inventory (CI) to initiate continuous overflow in the short SG tubes. Depending on the RCS CI and pressure, 1Φ or 2Φ flow is initiated. The mass flow of transported coolant in this case is greater than the condensation rate, and borated coolant is transported over the apex of the tubes (see PKL III F2.1 Run 1, [16]). Boron dilution resulting from the heat transfer process can be ruled out under these boundary conditions.

Test series PKL III F2 already revealed some fundamental coherence between primary inventory and occurrence of the different heat transfer modes above.

3. The operating manuals for Siemens PWRs state that for mid-loop mode at least one SG must be filled on its secondary side and be ready for operation to provide an alternative heat sink in the theoretical event of total loss of the residual heat removal system.

In absence of forward coolant transport, heat transfer may deteriorate (i.e. reactor coolant system temperature and pressure are higher) as more inventory is displaced into the heat-removing SG, whereas the major influence parameters on the primary inventory are as follows:

- influence of higher initial level in the RCS, see Test F2.1, [16];
- hot pressuriser (PRZ) (less water displaced into PRZ during heat up compared to cold PRZ);
- only one operable SG (inventory is displaced towards this SG);
- additional injection of inventory (see below for specific effects).

The expulsion of N_2 and onset of flow in SG tubes generally improves heat transfer to the secondary side (i.e. stabilisation at a lower pressure and temperature), but can lead to boron dilution in the loop seal due to condensate spillover in the U-tubes (as seen for only one operable SG).

For two or more operable SGs the following differences (compared to only one operable SG) are evident:

- more homogeneous distribution of CI among all heat-removing SGs;
- less CI in the individual heat-removing SG;
- larger heat transfer area in the individual heat-removing SG (no sub-cooled slugs);
- lower primary equilibrium pressure level; and
- significantly lower prospect of boron dilution.

The overall objective of the G1 test series on this topic is to gain an understanding of the flow conditions and of the changing of the heat transfer mechanisms described, dependent on the primary inventory.

1.3.2 Cool down and natural circulation (NC) under asymmetric boundary conditions

Cool down under asymmetric boundary conditions may be required, for example, after a feed water or MSLB as well as for a SGTR, with simultaneous loss of off-site power, in either case. After isolating its main steam (MS) and feed water, the secondary side of a defective SG continues being heated from the primary side reaching a condition where the temperatures on the inlet and outlet sides of the SG become equal. In this condition the driving force for NC diminishes until eventually only the driving force from the RPV is present for the affected loop or loops. Through cooling the RCS via the intact SGs, an opposing driving force occurs in the loops with the isolated SG (heat up of the main coolant during the flow through the SG that is warmer on the secondary side, and the development of a negative temperature difference between the SG inlet and outlet sides), by which the entire flow through the loop can stagnate (Figure A8).

1.3.3 Fast cool-down transients

In a German PWR, for a MSLB (10% of area) inside of containment (Figure A8, Scenario G3), a rapid feed water and main-steam-side isolation of the affected SG would occur at the beginning of the accident because of the violation of protection limit values (for example, pressure gradient limitation set-points). Secondary-side coolant enters containment through the break and the affected SG boils off. As the event progresses, the reactor coolant pumps (RCPs) (coast down curve) and the pressuriser heater (pressuriser level < 2.28 m) are shut off.

A large heat sink develops on the secondary side of the affected SG because of the MSLB and a large amount of heat is transferred from the primary side to the secondary side, as the temperature difference between primary and secondary side increases. On the primary side, the increased energy removal due to boiling in the affected SG leads to a sub-cooling transient and to a decrease in primary-side pressure due to volume contraction. The question is then, how much sub-cooled water from the affected loop reaches the core through the RPV DC and can contribute to re-criticality there.

In some PWRs in other countries, the primary-side pressure decrease activates a primary-side injection through the safety-injection pumps (SIPs) into the cold legs and thus to a rapid pressure increase until the PRZ safety valve (SV) is actuated. Primary-side coolant then flows through the PRZ SV into the blowdown tank. The question in this case is to what extent pressurised thermal shock (PTS) aspects must be considered due to the entry of cold emergency coolant into the RPV DC.

Tests in the PKL and ROCOM test facilities were performed based on these investigation goals (re-criticality and PTS). In the PKL facility, the overall thermodynamic behaviour was investigated. The results of PKL Test G3.1 (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 DC and in the LP in the ROCOM test facility.

Furthermore, the test results are used for the validation and optimisation of analytical tools for analytical benchmark activities among the participating partners, that is, for post-test calculations with system programmes (in connection with PKL) or computational fluid dynamics (CFD) calculations (in connection with ROCOM). The final goal is to be able to make a better confidential level for PWRs with regard to re-criticality 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.

1.3.4 Influence of secondary-side parameters on heat transfer under RC conditions

In case of a LOCA[4] the RCS inventory varies throughout the course of events depending on the number of SIPs in operation and the discharge rate via the break, which is itself a function of break size, RCS pressure and coolant condition at the break location.

RC conditions in the RCS may temporarily occur in the RCS if the leakage rate at high pressure is greater than the injection rates of the safety-injection pumps as a result of a coolant reduction in RCS inventory, in particular, if the SIPs are postulated to operate at reduced availability. In this state of operation the decay heat from the core is then transported to the SGs in RC mode.

The effectiveness of RC conditions in heat removal lies within the condensation mechanism on the U-tube primary sides, a dropwise or film condensation represents the most effective way of heat transfer Figure A9). As a result of the effectiveness of heat transition under RC conditions, the emerging temperature differential between primary and secondary side is very low (among 3-5 K), enabling a direct coupling of the primary and secondary sides temperature and pressure levels and thus facilitating a cool down of the primary side.

During LOCA scenarios whose course of events are postulated to involve cool-down procedures either featuring reduced safety-injection systems (SIS) availability or being initiated at a late or very late point of time, the most significant parts of the transient (comprising the shift to low-pressure injection) are presumed to be conducted under RC or RC-like conditions. Therefore, the correct reproduction of the thermal-hydraulic interaction between primary and secondary side for this state of operation is a central

4. “Small”-break LOCAs are deemed large enough to resist make-up by the chemical and volume control system but still small enough that the SGs are needed for heat removal.

demand for thermal-hydraulic (TH-) computer codes nowadays employed in safety analyses by licensing authorities, expert organisations and manufacturers.

Although, heat transition to secondary side is supposed to be widely replicated correctly by today's TH-system codes within a broad spectrum of boundary conditions, the special data basis provided with the results of test G4.1 enable a verification or further improvement of the existing computer models on the reproduction of the complex physical phenomena involved in heat transition from primary to secondary side under steady-state or transient RC conditions in the RCS.

In this way, test G4.1 serves the further development of the basics required for the safety-related assessments of PWR TH-behaviour. The test results are dedicated to further extend the data basis for the validation of T/H Codes on the complex thermal-hydraulic phenomena associated with RC conditions.

1.3.5 Boron precipitation processes following LB-LOCA

The G5 test scenario is set in the long-term cooling phase following a large cold LB-LOCA (Figure A10). Continuous boiling in the reactor core is expected during the long-term phase of a large cold LB-LOCA in PWR plants with sole cold leg injection of emergency core coolant (ECC) water. In this state the reactor core is rewetted and covered by two-phase mixture of water and steam but due to mixture of cold ECC water with steam generated in the core the ECC fluid fed to the core is almost at boiling temperature. Thus, the boiling process in the core continues in the long term. During evaporation the B injected with the ECC water remains in the liquid phase in the core region and accumulates continuously. At B values which exceed that of the ECC water by a factor of 20 or more the limits of boron solubility may be reached and even exceeded; locally initiated crystallisation then effectuates the formation of solid boron particles which may influence the core cooling capabilities. In the long-term cooling phase the containment pressure (= RCS pressure) is assumed to vary around 2.5 bar (T_{sat} ~127 °C), under these conditions the solubility limit for boron in water is reached at approx. 66 000 ppm.

The most decisive parameters for the speed of the boron enrichment process in the core are the core power (residual heat) and the size of the mixing volume i.e. amount of liquid water available for mixing in the RPV. The direct impact of the residual heat released in the core manifests in an increased evaporation rate on one hand and an increased amount of steam (void volumes) in the core region on the other hand; the consequences of both mechanisms are relatively easy to determine. The RPV volumes involved in the mixing process determine the amount of liquid coolant available for mixing. It is crucial to determine if large sections like the reflector gap of the LP contribute to the mixing volume, and if yes, to what extent. The contribution of these sections – even if only partially – may significantly increase the mixing volume and thus prolong the grace time until precipitation occurs.

A mechanism whose impact is not so easy to quantify lies within the magnitude of the flow resistance of the cross-over legs (COLs): In a cold leg 1A break scenario, a high flow resistance between upper plenum (close to steam source) and cold side (break, heat sink) is postulated to impose a high pressure on the SL in the core. The higher the flow resistance, in particular for a partial or complete blockage of one or multiple COL by ECC or two-phase mixture, the steeper the gradient and the higher the pressure in the upper plenum. In conjunction with high steam loads (high core power) this may cause a swell-level depression in the core and a significant reduction of the mixing volumes in the RPV. Another consequence of this pressure balance between hot (core, upper plenum) and cold side (break) is the limitation of the ECC flow from the cold side via DC to the core as the upper plenum (UP) pressure acts against the flow. In sum, for the stationary pressure and fill/swell-level balances between hot and cold side (considering the temperatures of water columns of core and DC), and at the power levels in the G5.1 scenario, the ECC flow is expected to compensate only for the evaporation rate in the core, regardless of the ECC flow rates as any excess ECC is lost via the break. A distinct sub-cooling span in the core cannot be restored by high cold side ECC flow rates. Moreover, a high cold side ECC flow tends to increase the flow resistance of the COLs which again increases the pressure gradient between hot and cold side of the RPV.

In sum, boiling in the core is assured for a longer period in the long-term cooling phase following a LB-LOCA.

An effective measure to avoid or reverse a present or developing boron enrichment process in the core is the switch to hot leg (HL) emergency core coolant injection (ECCI). A HL ECC flow high enough may create a net flow that displaces the high boron RPV CI via DC towards the break ("flushing" of the core).

However, an early switch to HL injection is inappropriate due to high steam velocities present in the loops at the early state of the long-term cooling phase which are assumed to divert the ECC injected in the HLs into the SGs and thus avoid a continued cooling of the core. Instead, the boron injected with the ECC may accumulate in the U-tubes as a result of evaporation in the U-tubes as the secondary side remains at elevated pressure and temperature. Consequently, for PWR preferably featuring cold leg ECCI, the switch to HL injection for a reversion of the enrichment process in the core has to be delayed until the steam velocities have reduced according to the decrease of the decay heat.

1.3.6 Formation and behaviour of upper head void during cool down under NC-conditions

In a PWR, following the scram at occurrence of the incident the plant is cooled down via MS relief from secondary side into turbine bypass tank or, - in case of unavailability – into atmosphere.

Following the trip of the RCPs, the flow conditions shift from forced to NC thereby changing the temperature distribution in the RPV significantly. The heat up span across the core for decay power of ~ 1.9% increases from approx. 2-3 K under forced circulation (RCPs operating) to approx. 25 K under NC-conditions. During depressurisation on the primary side, a void formation in the RPV dome may occur if the fluid in the RPV is still hot and enters flash evaporation as the RCS pressure drops.

It no longer seems possible that the void expansion in the RPV dome could progress up to the point where the steam volumes enter the reactor coolant line (RCL).

During depressurisation of the RCS the fluid in the RPV dome reaches the local saturation conditions and enters flash evaporation, thereby creating the upper head void volumes.

Following the activation of RCPs the void volumes are expected to partially collapse at least. If a complete and rapid collapsing of the upper head void occurs, the strain imposed on the RPV dome structures resulting from the hydraulic and thermal shocks may produce unfavourable loads.

1.3.7 SB-LOCA with beyond-design-basis additional system failures

In PWR, the defence-in-depth strategy assures prevention and control of events and accidents at several engineering and procedural levels in order to ensure the effectiveness of the protection of physical barriers against the release of radioactive materials. In addition to engineering and procedures which reduce the risk and severity of accidents, all plants have guidelines for severe accident management (AM) or mitigation (SAMG).

Background for the G7.1 Test is a HL SB-LOCA scenario superposed by additional system failures (no high-pressure safety injection, no automatic secondary-side cool down). The postulated additional system failures result in a course of events that necessitates AM measures to prevent a core-melt scenario. A secondary-side depressurisation was employed as AM measure for restoration of the secondary-side heat sink aiming for a fast reduction of the primary pressure. The reduction of the primary pressure down to accumulator (ACC) injection pressure then effectuates the transition to the low-pressure phase with the low-pressure safety injection (LPSI) active.

Three main topics of investigation have been pursued in the frame of the G7.1 test concept:

- **Effectiveness of the secondary-side depressurisation** being employed as AM measure for the reduction of the primary pressure.

- **CET performance**, plausibility of the measuring signals acquired under beyond-design-basis loads.

  > "…the CET shall be used to get reliable information about the thermal-hydraulic state in the reactor core. The most important information concerns the quality of the cooling of the core. Any inadequate core cooling should be detected as soon as possible in order to take adequate countermeasures to re-establish a proper core cooling in a timely manner. In case of an inadequate core cooling, the key information concerns the temperatures in the reactor core. The general overall temperature situation in the core is important, but the hottest fuel rod cladding temperature in the core is of paramount interest because it mainly determines the timing of AM actions.
  >
  > The technical challenge consists therefore in deducing the core temperature, which is not directly measured in the reactor case, from CET readings. That means, physically spoken, some heat from the superheated core region has to be transported to the CET measurement sensor, where it can be detected. This energy transport can only (in a timely manner and for limited temperatures) be realised through the convection of cooling fluid from the core region to the CET measurement location. In the case of a superheated core which is at least partially covered by liquid water the fluid consists mainly of steam and entrained water which is generated by the core decay heat. Hence, the overall energy transport from the fuel rod to the CET sensor has the following three main steps: heat transfer from the rod surface to the fluid, convective fluid transport to the CET sensor location, and heat transfer from the surrounding fluid to the CET sensor."
  >
  > NEA CSNI report NEA/CSNI/R(2010)9, p. 127.

- **Counterpart testing with ROSA/LSTF** for the determination of scaling effects between the PKL and ROSA/LSTF test facilities was encouraged as it was expected to be beneficial in order to draw relevant conclusions and recommendations for a possible further work on the CET performance issue.

1.4 Application of boric acid and measurement of boron concentrations

For some tests within the PKL2 project, boric acid was added to the primary coolant of the PKL test facility:

- **G1** test series: Inherent boron dilution following RC-like heat transfer in the SG evolving from failure of RHRS under cold shutdown condition.

  Inherent boron dilution was only addressed as the result of failure of RHRS under cold shutdown conditions in the test series G1 (test G1.1 and G1.2). COMBO (Continuous Measurement of Boron Concentration) devices have been employed in these tests for detection of the boron concentration. The COMBO system was originally developed by Siemens (now AREVA NP) for use in real PWRs. The measuring principle is based on the absorption of neutrons by the isotope boron-10, which varies according to the boron content of the coolant. The COMBO system, which basically consists of a neutron source (positioned on one side of the reactor coolant line) and two counter tubes (installed close together on the opposite side of the reactor coolant line), can be mounted on the outer wall of the RCL so that it has no effect on the fluid being measured. As a result of using this neutron transmission method, the measurements are averaging over the entire flow cross-section in the pipe.

Apart from the COMBO systems, the test facility is equipped with sampling points for taking grab samples at all the locations of the primary system (see appropriate G1 test reports [19, 20]).

The grab samples analysed by a titration apparatus are used to verify the COMBO measurement signals.

- **G5** test: Boron precipitation in the RPV following LB-LOCA. For the measurement of high boron concentrations of over 10 000 ppm in the RPV conductivity probes have been installed at different locations at the PKL RPV (see individual reports). Just as the COMBO devices, the conductivity signals have been verified by grab samples.

The titration apparatus is calibrated once a year. The pH measurement is checked prior to and after each test.

2. TEST OBJECTIVES

2.1 PKL2 tests

The Primärkreislauf (PKL2) test matrix with test subjects and safety relevant key phenomena can be found in Table B1 in the Appendix B.

2.1.1 Objectives of Test G1.1/G1.2

Test series G1 comprises systematic studies on heat transfer in steam generator (SG) U-tubes in presence of nitrogen (N_2) and with variable primary coolant inventory (PCI) to establish different swell-level heights (with and without coolant in SG U-tubes) that result in different heat transfer mechanisms:

- Blocking of U-tubes by sub-cooled coolant slugs with N_2 enclosed.
- Continuous and discontinuous 1Φ or 2Φ coolant transport in U-tubes (prospect of boron dilution).

The emerging test results are dedicated for the support of conclusions on plant operation following loss of residual heat removal system (RHRS) as regards:

- Stabilisation of primary pressure and – temperatures at moderate levels.
- Prevention/mitigation of boron dilution on one hand, but primarily represent a data basis for the validation of thermal hydraulics (T/H) Codes on the beforehand mentioned phenomena.

To support post-test analysis or reproduction with T/H system codes, special emphasis has been put on establishment of clear boundary conditions for each test phase.

The investigations in the frame of G1 therefore focus on the different heat transfer mechanisms and corresponding states of system stabilisation emerging from a specific primary inventory.

The individual test runs of G1 tests were both performed either with single (G1.1, G1.1a) or double loop operation (G1.2) and without pressuriser PRZ (Figure B2).

2.1.2 Objectives of the Test G2.1

In Test G2.1 the asymmetric unit cool down under a loss of off-site power is investigated with two intact SGs (SG 2 and 4) and two SGs that are isolated and boiled-off on the secondary side (SG 1 and 3).

The objective of Test G2.1 is to investigate a plant cool down without reactor coolant pumps (RCPs) under asymmetric conditions between the individual loops, i.e. two SGs are intact and two SGs isolated and boiled-off on the secondary sides. Of main interest is the behaviour of natural circulation (NC) in the loops with isolated and boiled-off SGs.

It must be clarified whether NC is maintained in loops with isolated and boiled-off SGs at a continuous cool down (gradient of 50 K/h), with and without additional primary-side pressure reduction by spraying with the volume control system (runs 1 and 2), and if not, whether NC can be maintained by a stepwise cool down (run 3).

The following aspects are of special interest in this context:

- Natural-circulation behaviour in isolated SGs 1 and 3 as dependent upon the decrease in the reactor pressure vessel (RPV) outlet temperature.
- Dependence of NC in the loops with isolated SGs on the temperature difference that arises between inlet and outlet sides of the SG.
- Determination of the negative temperature difference between inlet and outlet sides in the isolated SGs that leads to the interruption in primary-side NC.
- Heat transfer from the secondary side of the isolated SGs by NC on the primary side.
- Analysis of the opposing driving force (density effect) in the isolated SGs, and the hysteresis of the NC flow in the loops with the isolated SGs.

2.1.3 Objectives of Test G3.1

The transient behaviour resulting from a break in the main steam line of a SG was investigated in this test. Of particular interest were the effects of the main steam line break (MSLB) on the primary-side system conditions during the boiling-off phase of the affected SG and the subsequent primary-side injection from two safety-injection pumps (SIPs).

Non-isolable MSLBs (up to a double-ended break) cause a rapid decrease in secondary-side pressure in the affected 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 (sub-cooling transient). An important question during this process is, whether a local re-criticality 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 sub-cooling and re-criticality. With a completely filled primary circuit, the transient was started by completely opening a valve (representing a break) in the main steam line (MSL) of SG 1. The cross-section of the opening was chosen to represent the transient conditions of a 10% break. Due to the limiting maximum pressure of the PKL test facility, the processes that normally occur at higher pressures were represented departing from a reduced pressure of 45 bar (Figure B13).

An additional, important aspect of this accident scenario concerns RPV integrity under consideration of pressurised thermal shock (PTS) due to the introduction of cold water in the RPV downcomer (DC). This is intensified by injection of emergency core coolant (ECC) into the cold leg (CL) in a rather cold primary circuit (due to steam line break). Moreover, the pressure increase at the end of the transient tends to increase the mechanical load. This case is relevant for some pressurised water reactor (PWR) plants and was investigated after the affected SG was completely emptied. In this process, the primary-side pressure was increased by injection from the SIP, (cold side injection in 2 of 4 loops) until the pressuriser PRZ safety valve (SV) responded. Steam flow out of the PRZ SV was followed later by water flow.

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

- What is the heat transfer from the primary to the secondary side in the affected SG during the boil-off on the secondary side?
- What temperature is established at the RPV inlet in the affected and the unaffected loops because of the heat transfer?
- What is the NC flow in the affected and the intact loops?

2.1.4 Objectives of Test G4.1

Main test objective of G4.1 is to investigate the phenomena associated with a changing of the heat transfer conditions between primary and secondary under reflux-condenser (RC) conditions in the reactor cooling system (RCS) by application of various secondary-side parameter changes. The variation of secondary-side parameters were pursued in two ways:

- **Change of the secondary-side fill level**. Under specified normal operation the SG U-tube bundle is completely covered by secondary-side water, ensuring the complete availability of the heat transfer area between primary and secondary-side. The reduction of the secondary-side water level reduces the wetted U-tube surface available on the secondary-side. Consecutively, the local heat flow rate in the bundle between primary and secondary-side shifts to higher fluxes in the lower still wetted parts of the SG U-tube bundle resulting in higher thermal area loading in the lower parts of the bundle. By keeping the primary side operating conditions constant at RC conditions run 1 provides a data basis for effectiveness and characteristics (e.g. $\Delta T_{prim\text{-}sec}$) of heat transfer and heat transition only depending on the secondary-side fill level.

  During phases with lowered secondary-side water level, the feed-water flow delivered at the feed-water nozzle with distinct sub-cooling span is injected into saturated or overheated steam. Depending on the secondary-side level the heat up and admixture of feed water may influence temperature distribution on the secondary-side.

- **Increase of general heat flow rate from primary to secondary**. By imposing different cool-down procedures via secondary side, the general heat flow rate from primary to secondary is increased depending on the cool-down gradient. An intense heat flux to secondary side imposes a displacement of coolant towards the heat sinks on the primary side (SG U-tubes), the higher the heat flux, the larger the magnitudes of dislocations. Sustainment of core cooling for already reduced primary inventory (RC conditions in RCS with approx. 40% CI) with imposed additional dislocations of CI is the main objective in the second test run.

  Consequently, the secondary-side fill levels (i.e. the heat transfer area to secondary) and the RCS CI were kept at constant conditions throughout the cool-down procedures. In this way, run two provides a data basis for the change of characteristics of heat transfer in the U-tubes' primary sides for different heat flow rates to secondary side, in particular on the occurrence of counter-current flow limitation (CCFL), causing persistent coolant removal from RPV.

In sum, Test G4.1 serves the further development of the basics required for the safety-related assessments of PWR thermal-hydraulic behaviour. The test results are dedicated to further extend the data basis for the validation of T/H codes on the complex thermal-hydraulic phenomena associated with RC conditions.

2.1.5 Objectives of Test G5.1

The considerations on the theoretical background of the post leg break loss-of-coolant accidents (LB-LOCA) long-term core cooling phase return the following important points as subjects to be clarified by Test PKL III G5.1:

- **Size of the mixing volumes in the RPV**.

  The amount of RPV sections and volumes which contribute to the mixing volumes is analysed in G5.1 by boron concentration measurements in different RPV sections throughout the test run.

  The impact of core power (evaporation rate, void volumes in the core, pressure level in upper plenum) on the enrichment process is quantified in G5.1 by a variation of the core power level within the power range typical of the scenario: 1-2%.

  The impact of the flow resistance of the cross-over legs (COLs) on the size of the mixing volumes in the core is analysed in G5.1 by a variation of the cold side ECC flow rates and injection sequences with the (modified) chemical/volume control system directly into the COLs.

- **Speed of the boron enrichment process** depending on core power and the size of the mixing volumes.

  For the determination of a point in time to switch from cold to hot leg (HL) emergency core coolant injection (ECCI) (core flushing) the speed of the boron precipitation and the decrease of the decay power (reduction of steam velocities in the loops) are the basis for calculations for real PWRs.

  In test G5.1 different sizes of mixing volumes with different speeds of the enrichment process have been established as a result from the broad variation of parameters

- **Hot leg ECC flow required to “flush” the core**.

  At a later stage of the test (1% of core power) the ECCI was at first partially switched to combined hot and CL injection for a mitigation or reversion of the enrichment process. The flow required for a flushing of the core was roughly determined by a stepwise increase of the HL ECCI.

- **Supply of a data base for code verification and assessment.**

2.1.6 Objectives of Test G6.1

Test G6.1 (Figure B26, B27) was designed to create a data basis for code validation or development of new models to describe RPV dome void fraction behaviour. Therefore, the test procedure addresses different thermal-hydraulic phenomena that may occur in the frame of cool down under NC-conditions with upper head void fraction from formation of the upper head void to its collapsing under RCP re-activation. Main objective of test PKL III G6.1 was to investigate the magnitude of the “sweeping flow” and its impact on the fluid temperature evolutions in the RPV dome during cool down and the formation and evolution of the upper head void volumes during depressurisation, and to provide a data basis for code validation on the emerging T/H phenomena:

- Evolutions of fluid temperatures in RPV dome during transition from forced to natural circulation and under pure NC-conditions.

- Formation of upper head void during depressurisation of the RCS via auxiliary spraying and interaction of RPV and PRZ (dislocation of inventory).
- Slow condensation of upper head void due to heat losses across RPV dome structures and the presence of continuous NC.
- Temperature stratification in the RPV in steam and liquid coolant above/below the interphase during and after formation of the upper head void fraction in presence of continuous NC.
- Partial collapse of upper head void during re-activation of RCPs.

2.1.7 Objectives of Test G7.1

The main test objectives comprise questions on the efficiency of the accident management (AM) measures performed (SG depressurisation followed by accumulator injection) on re-establishment of core cooling, as well as on the differences between core exit temperature (CET) and maximum measured cladding temperature (CT) as function of the boundary conditions.

Significant phenomena and effects to be investigated by the G7.1 test are:

- Core uncovery due to loss of inventory (boil-off) with the formation of superheated steam as well as the primary-side pressure behaviour before and after occurrence of core uncover.
- Effectiveness of a secondary-side depressurisation and its influence on the primary pressure as well as the accumulator (ACC) injection after the SG depressurisation and its influence on the core cooling.
- Relation between the P[5]cladding temperature and CET during these processes.
- Scaling effects between the PKL and ROSA/LSTF test facilities.

During an accident scenario with the occurrence of core uncovery the measured CET is of special interest. For the assessment of the CET performance, the PKL test facility is equipped with multiple thermocouples (TCs) at the core exit in different radial positions and installation configurations, e.g. replication of PWR instrumentation (TCs with and without shield tube). The CET thermocouple reading is used as criterion for the initiation of the AM procedures involving emergency operating procedures and/or severe accident management in various countries. Although, the actual temperature set point may vary among different PWR types. The performance of the CET mainly depends on the installation point of the TCs and the scenario. The test was performed to assess the reliability of the CET measurement and its correlation to the maximum CTs and to provide information on physical phenomena responsible for the CET performance.

The test G7.1 was designed as counterpart test between the PKL and ROSA/LSTF test facilities.

2.2 OECD Rossendorf Coolant Mixing (ROCOM) Tests

The ROCOM test matrix with test subjects and safety relevant key phenomena can be found in Table C1 in the Appendix C.

5. The peak cladding temperature (PCT) – being a parameter and definition used in reactor safety research – cannot be measured by the PKL instrumentation as the upper part of the heater rod – above ME 7 up to the end of the active length - is not instrumented. The maximum measured rod cladding temperature is measured at heater rod m9 at elevation (ME) 7.

2.2.1 Objectives of ROCOM Tests 1.1 and 2.1

The objective of the **ROCOM Test 1.1** was the investigation of the 3D flow behaviour inside the RPV under flow conditions typical for a MSLB scenario. For that reason the concrete boundary conditions were derived from the corresponding PKL experiment (Test G3.1). Quasi-stationary flow conditions were foreseen to be established derived from the time point of minimum loop temperature in the PKL experiment. The temperature distribution inside the DC should help in the clarification of the main mixing phenomena taking place after entering of the flows from the different loops with different temperatures into the vessel. A sector formation between the coolants from different loops was expected to be formed inside the vessel. The test should provide information on the position of the transition region between the established sectors and a more or less homogeneous temperature distribution. Finally, the test should allow drawing conclusions on a possible re-criticality during such a postulated accident. The resulting temperature distribution at the core inlet should be assessed.

The objective of **ROCOM Test 2.1** was to assess the influence of the combination of loop flow rates and temperature differences on the position of the transition region between sectored and homogenised temperature fields. For that reason the flow and temperature conditions from an earlier time point of the PKL Test G3.1 were used as boundary conditions.

An additional objective of all ROCOM tests was to provide data for code validation (especially for computational fluid dynamics (CFD) codes).

2.2.2 Objectives of ROCOM Tests 1.2 and 1.3

A second safety issue during a postulated MSLB accident is the temperature load on the loop and RPV wall. The objective of the **ROCOM Tests 1.2 and 1.3** was the assessment of the mixing of the ECC water with the ambient water on the way from the injection position till the core inlet. Special focus was put on the stratification in the loops, on a possible flow reversal in the loops and on the temperature distribution in the downcomer. In this respect the position of the lowest coolant temperature and the possible presence of fluctuations of the ECC water stripe are of special interest.

2.2.3 Objectives of ROCOM Test 2.2

In addition to the quasi-stationary ROCOM Tests 1.1 and 2.1 the **ROCOM Test 2.2** was performed under transient conditions. The main objective of this test was the assessment of the influence of changing mass flow rate in the non-affected loops on the position of the transition region between the sectors and a more or less homogeneous temperature distribution.

2.3 OECD-PMK Tests

The PMK test matrix with test subjects and safety relevant key phenomena can be found in Table D2 in the Appendix D.

2.3.1 Objectives of PMK Test 1

PMK test 1: Disturbance of NC during lowering of primary system level

The objective of the test was the investigation of the degradation of SG heat transfer and of the disturbance of NC during lowering of the primary system level in preparation of core unloading. During shut down, the core cooling in a VVER is not assured by RHRS – as in PWRs –, but by heat removal through the SG with NC on the primary side and the secondary side filled with water. In preparation of reactor vessel lid removal valves are opened at the top of reactor vessel, pressuriser and SG collectors and the primary system inventory is drained allowing air to fill up the higher parts.

Test 1 addresses the situation, when – due to leakage or an operator error – the vessel level drops below the one specified for vessel head opening. The aim of the test was to systematically investigate the natural-circulation behaviour and the resulting primary/secondary heat transfer at different fill levels of the primary side, with special respect to the effect of the amount of steam produced in the core and of the status of the valve at the top of the pressuriser. For that purpose primary mass was drained in several steps (Test 1.1A and 1.2) or continuously (Test 1.1B) until the level decreased close to the top of the core and replenished in each test run in several steps.

2.3.2 Objectives of PMK Test 2

PMK Test 2: SB-LOCA during cool down of the plant to cold shutdown conditions

The objective of Test 2 is the investigation of SG heat transfer effectiveness in case of a SB-LOCA during the cool down of the plant to cold shutdown conditions. According to the original VVER-440 cool-down procedures below 2.5 MPa the pressuriser steam atmosphere is replaced by nitrogen, hydro accumulators and high-pressure emergency core cooling system (ECCS) are disconnected from the primary system and the automatic start-up of low-pressure ECCS is disabled. A LOCA in this situation results in N_2 injection to the primary system, most of it being collected in the SGs. Since emergency injection is not available, there is a competing process between CT rise and pressure reduction to the set point of low-pressure injection that strongly depends on the heat transfer effectiveness in the SGs, this latter being impacted by the presence of N_2. Obviously, secondary bleed is an important action for reaching LPIS injection in time.

3. TEST CONDITIONS

3.1 Test conditions – Primärkreislauf (PKL2) tests

3.1.1 Conditions of Test Series G1

Attached figures related to test series G1: Figures B1-B5.

Test series G1 comprises **parameter studies** for the investigation of heat transfer mechanisms (occurring in the course of a loss of residual heat removal system (RHRS) transient under cold shutdown conditions) and the dependence of their formation on the primary coolant inventory (PCI). The boundary conditions the three test runs G1.1, G1.1a and G1.2 therefore reflect cold shutdown conditions for the operating loop:

- single-(G1.1, G1.1a) or double loop (G1.2) operation , remaining loops blocked by blank flanges in cold and hot legs (HLs) close to reactor pressure vessel (RPV), associated steam generator (SG) initially filled with water on the secondary side;
- sequence of steady states (i.e. stable heat flow from primary to secondary at stable primary pressure) with intermediate change of the PCI;
- primary inventory at ¾-loop level (homogeneous boron concentration [B] = 2 000 ppm), N_2 above at start of test (SOT);
- core power initially set to 180 kW corresponding to ~ 0.6% of scaled full load core power after approx. 24 h after shut down of reactor, plus compensation for heat losses; [5]
- pressuriser (PRZ) isolated;
- prior to SOT: Removal of decay power via RHRS (loop 1);
- temperature at core outlet approx. 60 °C at SOT;
- shutdown of RHRS at SOT.

Both tests (G1.1 andG1.2) of test series G1 feature multiple changes of PCI (reduction/replenishment) with phases of steady-state operation (stable heat flow from primary to secondary) at (quasi-) constant primary pressure awaited in between. Inventory was drained and replenished via LP drain valve and a modified volume control system piping injecting into lower sections of downcomer tubes, respectively, in order to avoid a disturbance of the steam formation in the core as a consequence of exterior influence.

The boundary conditions have been arranged to suffice an in-depth investigation of the thermal-hydraulic processes occurring between primary and secondary sides. To serve this assumption, all potential disturbance variables (auxiliary heat sources for the compensation of heat losses) have been eliminated. According to this, the SGs secondary support heaters have been identified to possibly cause perturbations in the heat flow from primary and secondary and hence were not in operation in both runs, the same applies for RPV closure head trace heater and the reactor coolant pumps (RCP) cooling circuit.

3.1.2 Conditions of Test G2.1

See figures related to test G2.1: Figures B6-B12 in Index of tables and figures page 88.

The test is composed of three test runs in all. In these test runs, the asymmetric unit cool down under a loss of off-site power is investigated with two intact SGs (SG 2 and 4) and two SGs that are isolated and boiled-off on the secondary side (SG 1 and 3). The level in the pressuriser was regulated at 4 m by the volume control system in each run. The individual test runs differ in the following boundary conditions:

Run 1: Continuous cool down at 50 K/h by intact SGs 2 and 4 **with additional decrease in primary pressure** by spraying in the pressuriser with the volume control system (sub-cooling at core outlet approx. 10 K).

Run 2: Continuous cool down at 50 K/h by intact SGs 2 and 4 **without additional decrease in primary pressure** (higher sub-cooling at core outlet).

Run 3: Stepwise cool down without additional decrease in primary-side pressure.

The steps are chosen so that natural circulation (NC) is maintained in the loops with the isolated SGs: A total of three steps of 20 K, 30 K and 40 K are carried out, during which a cool-down gradient of 50 K/h is applied. After each cool-down step, a waiting period was planned so that temperature equilibrium between the inlet and outlet plenums in the isolated SGs could be established.

3.1.3 Conditions of Test G3.1

See figures related to test G3.1: Figures B13-B17 in Index of tables and figures page 88.

Because of the pressure limitation in the PKL test facility, Test G3.1 was performed at a lower pressure (pressure at start of test: p_{prim} = 42 bar; p_{sec} = 35 bar, Figure B13).The most important test boundary conditions were as follows:

- reactor cooling system (RCS) is completely filled with water and fully in operation;
- core power is 260 kW, equivalent to 0.8% of decay power (conservative assumption with respect to sub-cooling transient), including compensation for heat losses;
- pressuriser is hot, approx. 250 °C (PRZ heating is in operation until SOT);
- a break inside the containment in the SG 1 main steam line (MSL) (fill level = 9.2 m) is the initiating event;
- the unaffected SGs 2 – 4 are isolated from the break main steam (MS) isolation valves are closed) and filled with water (h = 12.2 m);
- all SGs are isolated from the feed-water system;
- RCPs are shut off at SOT, i.e. occurrence of break (coast down).

The replication of a main steam line break (MSLB) required additional piping and measuring devices being installed at the PKL test facility [22].

3.1.4 Conditions of Test G4.1

See figures related to test G4.1: Figures B18-B20 in Index of tables and figures page 88.

Both G4.1 test runs were specified according to the premise to keep reflux-condenser (RC) conditions and the primary inventory constant to observe the heat transfer from primary to secondary under RC conditions.

The most important test boundary conditions in both runs were:

- RCS inventory constant (RC conditions Figure B18).
- Core power constant in both test periods:
  - Run 1: 600 kW, 1.8% plus compensation for heat losses. The composition of the core power of 600 kW (corresponding to 1.8% in a real pressurised water reactor) is quite accurately close to the mean value for the expected real pressurised water reactor (PWR) power range for a SB-LOCA investigated scenario. The 1.8% consist of the typical 1.3% (for German PWR) plus a compensation for the heat flux emerging from structures and fluid volumes during the continuous cool-down procedure normally present in real accident scenarios.
  - Run 2: 540 kW, 1.6% (conservatively high value, averaged across typical cool-down transient in the course of a SB-LOCA).

Specifics of run 1:

- Parameter study for the investigation of heat transfer decreasing between primary and secondary sides as a function of the secondary-side fill level (Figure B19).
- Multiple stepwise changes of secondary-side fill level with intermediate phases (A à H) of steady-state operation (i.e. stable heat flux from primary to secondary) under RC conditions in the RCS at constant primary pressure and core power.

The system was allowed to recover (return to steady-state conditions) following every variation of secondary water inventory and pressure.

Specifics of run 2:

- Sequence of cool-down procedures via secondary side from 40 bar / 250 °C (core exit temperature – CET) with different cool-down gradients (100, 250, 470 K/h) under RC conditions in the RCS with intermediate phases of reset to initial conditions (Figure B20).

The system was reset to initial (steady-state) conditions after each cool-down procedure.

3.1.5 Conditions of Test G5.1

See figures related to test G5.1: Figures B21-B25 in Index of tables and figures page 88.

Figure B25 provides an overview on the procedures and measures employed in test PKL III G5.1. The test (as well the pre-test) was specified as a parameter study according to the premise to arrange different boundary conditions for the propagation of a boron precipitation in the core.

The most important test boundary conditions were:

- Start of test (Figure B21) with **most of the RCS emptied,** cross-over legs (COLs) cleared and low swell and fill levels in the RPV core region and downcomer, respectively, to minimise mixing volumes in the RPV in the first phases of the test.

- **Upper head/nozzle bypasses closed**, to maximise pressure differential between hot (UP) and cold (break) sides. The postulated higher pressure in the upper plenum is assumed to effect a swell-level depression in the core region to minimise the mixing volume in the RPV.
- **Core power constant at different levels** within the power range relevant for the scenario: 1-2% heater rod bundle power supply at 1.5% (450 kW), 2% (520 kW) and 1% (320 kW) of simulated decay heat.
- **Broad variation of emergency core coolant (ECC) flow rates** between half and multiples of the evaporation rate in the core starting with sole cold side injection, later switch to combined hot and cold side injection and further increase of HL injection (Figure B25C).
- Thereby: compensation for evaporation rates, refill of the RCS and eventual core flushing. ECC flow rates in the range of 0.06 kg/s (phase 6 of test G5.1) to 4 x 0.82 kg/s (pre-test, [24]).
- **Different fill levels in the primary system** (influenced by slight changes of ECC flow rate and/or injecting water into the pump seals).

The test conditions at SOT were chosen so that boron precipitation should be limited to the core region as much as possible. This required low emergency core coolant injection (ECCI) rates and a DC level lower than cold leg elevation (deduced from pre-test).

The Test G5.1 comprises the Phases 1 to 8 (Figures B24, B25) with quasi steady-state test parameter conditions, investigating the changes of boron concentrations in the core region and adjacent volumes.

3.1.6 Conditions of Test G6.1

See figures related to Test G6.1: Figures B26-B29 in Index of tables and figures page 88.

The initial facility conditions simulate a PWR under hot-standby conditions at a primary pressure of 40 bar (see Figure B26). The test procedure is based on PWR procedures as was composed of separate phases featuring sole cool-down processes or cool down with depressurisation in parallel (Figure B27). The following general boundary conditions have been applied:

- **Power level in the core: 1.9% decay power simulated, constant throughout test** (Figure B29). The power level of 1.9% represents an average across 3 000 s after scram for a 3 850 MW_{th} U/MOX core. In addition to the simulation of the decay heat the core power in the PKL test must also compensate for the heat losses of the primary side: another 100 kW are added to the core power (representing primary-side heat losses for 250 °C [5]). For the test phase with void growth this level of core power is conservatively high as the higher the core power level, the higher the CET and RPV dome temperatures. In the later test phases B and C featuring cool down under NC the core power level may be no longer conservative. Indeed, higher NC flow rates (due to higher core power) may provide increased sweeping flow magnitudes which cool down upper plenum (UP) structures by trend on one hand; but the dome temperature is strongly influenced by stratification which acts against the sweeping flow on the other hand.
- **Heat losses across the upper head RPV walls**. In real PWRs, the condensation of the fully formed upper head void fraction is mostly dominated by the cool-down rate due to heat losses is approx. 3-5 K/h and only to a lesser extent by the present NC. In the PKL test, this cool-down rate of the RPV dome fluid will be realised by the upper head support heaters.
- RCS completely filled, forced circulation by RCP operation in all four loops.
- Steady heat removal via secondary side; main steam (MS) and feed-water systems active.

- **Reduction of the upper head bypass by 50% to 0.25% of the core flow** in the PKL bypass lines for a reduction of the "sweeping flow". A constriction of the flow paths between UP, RPV dome and DC was assumed to create a less pronounced cool down of the UP and dome structures during the first cool-down phase, which then provide conservative boundary conditions for the formation RPV upper head void volume.

For the investigation of the fluid and wall temperatures in/on the relevant RPV volumes and structures, the PKL test facility was refitted with additional thermocouples (TC), distributed among TC-lances in the fluid volumes of the upper RPV and individual TCs attached to the RPV wall structures [25].

3.1.7 Conditions of Test G7.1

See figures related to Test G7.1: Figures B30-B33 in Index of tables and figures page 88.

The final PKL test boundary and initial conditions have been derived from its ROSA/LSTF counterpart test OECD Nuclear Energy Agency (NEA) ROSA-2 Test 3. The G7.1 test procedure and set-points (e.g. SI activation pressures) have also been adapted to fit the ROSA/LSTF experiment.

Background of test PKL G7.1 is a small break (SB) LOCA (break in HL) combined with postulated additional (safety) system failures: A total failure of the high-pressure safety injection (HPSI) and a total failure of the automatic secondary-side cool down. As accident management (AM) action a secondary-side depressurisation (for the re-establishment of the secondary side heat sink) at a CET of 350 °C was performed followed by a passive accumulator (ACC) injection. After the ACC injection the RCS was filled-up by the low-pressure safety injection (LPSI).

The initial conditions of the test were realised within a conditioning phase, prior to SOT. The pressure levels at SOT were similar to the situation in an EPR where a secondary-side depressurisation would be performed after a partial cool down. At SOT the inventory in the RCS was reduced to a mixture level in the SG inlet chambers and stationary RC conditions were present. At SOT the break in HL 1 was opened (upwards oriented with a size of 1.5%) and then the core power was removed by the break flow (SGs isolated on the secondary sides). The core power was held constant at 1.8% during the conditioning phase and the entire test.

All 4 SGs were depressurised (connected via main steam header) by fully opening of two main steam relief control valves (MSRCV) at a CET ≥ 350 °C. In the PKL test facility - according to the PWR - the maximum flow rate is limited by nozzles installed in the MSL with a diameter of 19.2 mm each.

Concerning the emergency core cooling system (ECCS) a full failure of the HPSI was assumed. The ACC injection into all four cold legs began at a primary pressure of 26 bar and was finished at a primary pressure of 10 bar. The LPSI into cold legs 1-4 was activated at a primary pressure of approx. 8 bar.

3.2 Test conditions –Rossendorf Coolant Mixing (ROCOM) Tests

3.2.1 Conditions of ROCOM Tests 1.1 and 2.1

The boundary conditions for **ROCOM Test 1.1** were derived from the PKL Test G3.1 at the time point of the minimum temperature in loop 1 during the overcooling phase (t = 609 s). The measured mass flow rates and temperatures in all loops were used to create the boundary conditions for the ROCOM test. These conditions are summarised in Table C3. They are derived on the basis of the fulfilment of the scaling criteria between original reactor and ROCOM test facility (see Section 5.2.3).

The wire-mesh sensor at the inlet nozzle of loop 1 into the vessel and the DC sensors together with the sensor in the core inlet plane were installed during this test.

Before the test a sugar solution with the given density difference was prepared. The test was conducted under quasi-stationary conditions with constant values for the flow rate in all loops. For that purpose, in these loops the corresponding flow rates were established about 30 s before the start of the test. With time t = 0 s the water with higher density was injected into loop 1 of the ROCOM test facility with the given flow rate. Injection took place over 90 s. During the whole time the wire-mesh sensors recorded the time-dependent data.

The test was repeated five times in order to damp statistical fluctuations. The averaged over all realisations as well as the data of the single realisations are available.

In order to assess the influence of the boundary conditions on the position of the transition region between sectored and nearly homogeneously mixed coolant, the boundary conditions for **ROCOM Test 2.1** were taken from an earlier time point of the PKL Test G3.1 (T = 130 s). The corresponding boundary conditions are summarised in Table C4. It should be noted that the scaling approach is different from the above described test. The similarity of the Froude number was achieved by reducing the velocity determined for reactor conditions and the density difference measured in the PKL Test G3.1 by a factor of 5 (for details see [27]).

The test procedure and the number and positions of the installed wire-mesh sensors were identical to those of ROCOM Test 1.1 with the only difference that the injection took place over a time span of 150 s. This test was also repeated five times with subsequent averaging.

3.2.2 Conditions of ROCOM Tests 1.2 and 1.3

The boundary conditions for **ROCOM Test 1.2** were also derived from the PKL Test G3.1. During the ECCI phase in this PKL test two different mass flow rate regimes of the ECCI system were used. For the ROCOM Test 1.2 the temperature and mass flow rate data from the time interval with higher ECC mass flow rate were used; the boundary conditions are based on data from the time t = 1 500 s and can be found in Table C5.

As the ROCOM Tests 1.1 and 2.1, the Test 1.2 was conducted under quasi-stationary conditions. In all loops the corresponding flow rates were established about 30 s before the start of the test. With time t = 0 s the water with higher density was injected through the ECCI system into loops 3 and 4 of the ROCOM test facility with the given flow rate. Injection took place over 100 s. During the whole time the wire-mesh sensors recorded the time-dependent data.

In order to answer specific questions of the plume behaviour in the DC the **ROCOM Test 1.3** was conducted with ECCI into one loop, only. Further, there was no base flow in all loops. In this test the injection took place over 120 s. The density and the flow rate of the injected water were identical to the data of Test 1.2 (Table C5).

Both tests were conducted five times; the data were averaged and are available together with the results of the single realisations.

In addition to the sensors in the vessel the cold legs of loop 3 and 4 were equipped with three additional sensors each (see Figure A4).

3.2.3 Conditions of ROCOM Test 2.2

In order to meet the objectives the **ROCOM Test 2.2** was conducted with changing time-dependent boundary conditions for the flow rate in the non-affected loops.

The initial conditions were selected in such a way that the results of this test can be compared with those of ROCOM Test 1.1. These conditions are summarised in Table C6. Equal flow rates in the initial state in all loops were chosen to meet the objective of creating experimental data for the validation of computational fluid dynamics (CFD) codes. Figure C1 shows the measured loop flow rates in all four loops during the test.

The injection of water with higher density took place over a time span of 150 s. Again, the test was repeated five times with subsequent averaging.

The wire-mesh sensor at the inlet nozzle of loop 1 into the vessel and the DC sensors together with the sensor in the core inlet plane were installed during this test.

3.3 Test conditions – OECD-PMK Tests

3.3.1 Conditions of PMK Test 1

In order to meet the test objectives**, three test runs** were performed. In Test 1.1A and 1.1B the valve at the top of the pressuriser was open throughout the tests, while in Test 1.2 it was closed as soon as the replenishment of the primary system started.

All test runs were started from about the same initial conditions:

- Natural-circulation heat transfer, with the primary system completely filled.
- Core power: ~12 kW.
- Primary/secondary-side pressure 0.1 and 0.4 MPa, respectively.
- Increased feed-water flow to maintain constant secondary-side conditions at ~ 303 K, with the secondary side completely filled.

Starting from the above conditions the following test procedure was defined:

- Opening of the top valves at the top of reactor vessel, pressuriser and SG collectors and stepwise (or continuous) decrease of the primary inventory.
- After each step a waiting period was scheduled to allow stabilisation of NC and heat transfer conditions.
- When NC was interrupted and boiling in the core developed, valves at top of the primary system were closed.
- Increase of primary inventory was performed in several steps.
- As during the drainage phase, after each step a waiting period was scheduled to allow stabilisation of NC and heat transfer conditions.
- The test was terminated, when nominal level in the pressuriser was reached.

3.3.2 Conditions of PMK Test 2

In order to meet the objectives described above three test runs were defined with the same break size of 1%:

- Test 2.1 was conducted with steam atmosphere in the pressuriser. As an AM action secondary bleed by one relief valve (RV) was started as soon as the cladding temperature (CT) exceeded 350 °C. This test constitutes the reference case for SG heat transfer, without non-condensables in the primary circuit.
- Test 2.2 repeated the same sequence as Test 2.1, but with air atmosphere in the pressuriser. In this way the test addressed the effectiveness of SG heat transfer during the blowdown and, especially, following the bleed action in the presence of non-condensables.
- Test 2.3 was a repetition of Test 2.2, the only difference being that instead of one, two RV were opened for secondary bleed. The test investigated the effect of bleed rate in presence of non-condensables.

The initial conditions of all the three test runs were practically identical:

- Constant core power at decay heat level: 11 kW.
- Primary pressure: 2.5 MPa.
- Forced circulation on the primary side, with the nominal flowrate of 4.5 kg/s.
- Core inlet temperature: 180 °C.
- Pressuriser liquid level: 8.8 m.
- Secondary-side pressure: 1.0 MPa.
- No feed-water injection.

4. TEST RESULTS

In the following subsections the test results are given in brief, for further details see the appropriate test report.

4.1 Primärkreislauf (PKL2) Test results

4.1.1 Results of Test G1.1/G1.2

For details, see test reports: [19, 20].

The separate effects and phenomena observed in the frame of the G1 series represent basic thermal-hydraulic principles and are as such expected to occur also in pressurised water reactor (PWR) scale.

Test series G1 (see Figures B1-B5) confirmed basic principles already observed in earlier PKL test programmes E and F (OECD-SETH and PKL) but also revealed additional coherences relating to heat transfer mechanisms and are useful for the evaluation of real PWR thermal hydraulics (T/H) behaviour.

- For a closed reactor cooling system (RCS) and at least one steam generator (SG) operational, heat removal from the core to secondary is maintained, i.e. core cooling is assured in either case.
- Pressure stabilisation occurs without the employment of countermeasures.

As regards the formation of heat transfer modes in the SG-U-Tubes, the following findings could be considered applicable to PWR geometry with additional justifications:

- Displacement of water into heat-removing SG(s) tends to deteriorate heat transfer (i.e. RCS temperature and pressure are higher) as long as no forward coolant transport in the U-tubes occurs.
- **Two active SGs** provide a larger heat transfer area and reduce the pressure level required for stabilisation.
  - Generally, the presence of two active SGs causes the coolant to distribute homogeneously among two active heat sinks. The impact on the pressure gradient resulting from injections of comparable (between G1.1 and G1.2) coolant masses (CM) is significantly lower.
  - With two active SGs boron dilution as a result of overspilling in the SG-U-Tubes as shown in Figure B1, case 3 is to be excluded.
- The expulsion of N_2 from SG tubes is required for the onset of coolant transport (which generally improves heat transfer to the secondary side.
- Onset of coolant transport (overspilling/continuous natural circulation) significantly improves heat transfer and effectuates a decrease of the primary pressure.

4.1.2 Results of Test G2.1

For details, see test report: [21]

The individual runs of the test produced the following significant results:

- After the secondary sides of two SGs have boiled-off and have been isolated, a new steady-state condition of stable natural circulation (NC) arises in all loops. This occurs through the intact SGs without cool down on the secondary side. The circulation in the loops with the isolated SGs is, in this case, approximately half as large as in the loops with the intact SGs (Figures B10-B12).
- For a **continuous cool down** at 50 K/h with a loss of off-site power, complete stagnation of the NC in the loops with isolated SGs must be expected, even without additional pressure decrease (Figures B10 and B11).
- If - during the continuous cool down - the primary pressure is lowered with the pressuriser spray from the volume control system (run 1), evaporation in the SG U-tubes must be expected (Figures B7 and B10), even though sub-cooling exists at the core exit (about 10 K in the test).
- A significantly higher sub-cooling at the core outlet (no or less pressuriser (PRZ) spray), prevents evaporation, as it leads to sustained sub-cooling in the bends of the SG U-tubes. However, flow stagnation in some or all U-tubes of an isolated and boiled-off SG can also occur due to the driving force counter to NC inside the U-tubes (negative temperature difference between SG inlet and outlet side) when the intact SGs are continuously cooled down at 50 K/h (run 2, see Figures B8 and B11).
- With a **stepwise cool down** (in the test the average cool-down gradient was 22 K/h, measured over the entire duration of the cool-down process), the NC in the loops with isolated SGs could be maintained, which provided for heat transfer from secondary side of the isolated SGs (run 3, see Figures B9 and B12). Here, the following individual phenomena are observed:
  - The (negative) temperature difference between the SG inlet and outlet plenums measured in the isolated loops is a gauge for the opposing driving force and thus a good indication of the preservation of circulation in the loop.

    However, the (negative) temperature difference between the SG inlet and outlet plenums, and thus the opposing driving force in the isolated SGs reaches a maximum only after the end of a cool-down step. Therefore, the minimum NC in the loops with isolated SGs is also reached only after the end of a cool-down step.

    In the test, the maximum temperature difference between the SG inlet and outlet plenums was - 8 K (approx. 7 min after the end of a cool-down step). The NC remained at this temperature difference.
  - As observed in the test, flow stagnation in individual U-tubes of the isolated SGs can also occur with the existing loop flow rate during the cool-down steps.

4.1.3 Results of Test G3.1

For details, see test report: [22]

From the Test PKL III G3.1, the following important conclusions for the system behaviour under a main steam line break (MSLB) inside containment can be drawn:

- Large temperature and pressure decreases occur in the affected SG after the initiation of the accident (Figures B15, B17).

- The water in the affected SG evaporates completely within about 1 000 seconds.
- Increased heat transfer from the primary to the secondary side occurs in the affected SG, which in turn leads to a sub-cooling transient in the primary circuit (Figure B14).
- The temperature at the reactor pressure vessel (RPV) inlet in the affected loop sinks from 240 °C to 150 °C (Figure B15).
- A larger natural-circulation flow is created in the affected loops than in the remaining loops, even after the secondary side boils off (Figure B16).
- Because of the injection with the SIP, the primary pressure and the pressuriser level increase (Figure B15).
- The pressuriser (PRZ) safety valve (SV) opens as a result of the primary-side pressure increase, which limits the primary pressure to 42 bar.
- Strongly sub-cooled water reaches the RPV through both loops with injection, which results in a pronounced temperature stratification over the height of the coolant pipe (Figures B15, B17).
- No reverse flow of cold emergency coolant takes place through the reactor coolant pumps (RCPs) into the pump loop seals.

Furthermore, the results of this PKL test, which is oriented to PWR system behaviour, also deliver the boundary conditions for complementary tests in the **Ro**ssendorf **Co**olant **M**ixing (ROCOM) facility on mixing cold and hot water in the RPV downcomer (DC) as well as in the lower plenum (LP), and for determining the fluid state at the core inlet.

4.1.4 Results of Test G4.1

For details, see test report: [23]

Run 1: Change of the secondary-side fill level, see Figure B19

- The heat removal from primary to secondary side (i.e. core cooling) remains preserved as long as the secondary-side feed is intact, even at very low secondary-side fill levels. The heat up and evaporation of secondary-side feed water absorbs the most part of the heat transferred to secondary-side.
- Down to a level of 0.25 m secondary-side water level, stable nucleate boiling on the secondary-side accounts for a relatively small required temperature differential between primary and secondary of less than 20 K (18.5 K at 0.25 m of sec. fill level).
- At nominal SG fill level (i.e. 12.2 m in PKL) with the U-tube bundle fully covered by water a maximum condensation ratio of < 50% was measured on the SG outlet sides. Furthermore, a repetition of the measurement at 4 m of SG secondary fill level revealed a tendency to equal shares of condensation for SG in- and outlet sides.

Run 2: Variation of secondary-side cool-down gradient, see Figure B20

- Generally, reflux-condenser (RC) conditions account for a very efficient heat removal from primary to secondary. Requiring only a small ΔT between primary and secondary, the primary pressure remains closely coupled with the secondary pressure, even at relative high secondary-side cool-down gradients of 500 k/h.

- Displacements of coolant on hot (core à SG) and cold (DC à core) sides occur as a result of enforced steam condensation on in- and outlet sides of the SG U-tubes and increase in magnitude with faster cool-down procedures (i.e. higher cool-down gradients). For coolant inventories of approx. 40% and cool-down gradients of up to ~ 500 K/h the displacements do not compromise core cooling.
- The frothing of the RPV inventory during depressurisation assures a 2Φ cooling of the entire heated length in the core for cool-down procedures of up to 500 K/h, even for coolant inventories of only 40%.
- Particularly of interest for German PWR[6]: No counter-current flow limitation (CCFL) in SG U-tubes was observed for primary pressures above 10 bar for cool-down rates of 100 K/h under RC operating conditions with coolant inventories of ~ 40% (constant inventory, no break).

4.1.5 Results of Test G5.1

For details, see test report: [24]

The results of grab samples (Figure B24) as well of the online sensors indicate that during the complete test the mixing volume with increase of B consists **at least of the volumes of the core, the reflector gap and a significant part of the LP, down to the interconnection of the DC pipes (see Figure B23A)**. Theoretical increases of B assuming different sizes for the mixing volumes in the different parts of the mixing volume have been compared to the measured values during the test [24]. These theoretical considerations support the conclusions for the composition of the mixing volumes. The mechanisms of mixing that interconnect the different coolant volumes in the RPV sections are depicted in Figure B23.

It is obvious that theoretical or analytical approaches that consider the mixing volumes to be limited, for example to core region only, come up with much faster gradients for the precipitation process than obtained from the test.

In the KONVOI RCS geometry (simulated in PKL) the flow resistance across the loop towards the break is too low to maintain the swell level (SL) in the core low while DC level in parallel is high (postulated scenario in beforehand, Figure B22A). A depression of the SL in the core resulting from high cold side emergency core coolant (ECC)-rates or high flow resistances in the cross-over legs (COLs) is not possible. For RCS geometries similar to the KONVOI and realistic ECC flow rates it must be assumed that the mixing volumes always extend into upper plenum (UP), HLs and SG inlet plena (Figure B22C).

Apart from enlarged mixing volumes a high hot side SL provides a further mitigation of the precipitation process in the core by a partial boron deposit into SG-U-tubes. In the pre-test, which realised high ECC flow rates right from start of test (SOT), the precipitation process in the RPV was stopped below 10 000 ppm by boron deposit into SG U-tubes as a result of a high hot side SL (induced by a completely filled RPV DC, [24]).

In sum, for PWRs with RCS geometries (e.g. head losses, height of cross-over legs) similar to KONVOI configurations and at first sole cold leg emergency core coolant injection (ECCI), the boron precipitation issue does not seem to be a serious problem in the course of events following a leg break loss-of-coolant accidents (LB-LOCA) as it is not possible to reach the solubility limit in the core (assumed the availability of safety-injection systems (SIS) according to design-basis).

6. In German PWRs the LPSI has a maximum delivery head of approx. 11 bar. Moreover, the secondary cool down is automatically initiated at a rate of 100K/h under accidental conditions in German PWRs.

According to this, the grace period until a “core flushing” (by switch to HL injection) may be required to sustain/re-establish core cooling is long enough to allow the switchover to HL injection at significantly decreased decay power level (distinctly below 1% for t > 6 h after scram). For the reversion of the boron precipitation process in the core (“core flushing”) for a decay power of 1% it is assumed that a HL injection rate that exceeds the evaporation rate by 10-20% is sufficient to produce a net coolant flow from HL via core and DC towards the break.

4.1.6 Results of Test G6.1

For details, see test report: [25]

- The start of void growth under the top of the RPV dome dislocates coolant underneath the flashing layers (which are also close to saturation) from the dome downwards (Figure B28-B). In this way the void volumes may expand to below the top plate into the upper plenum.
- All coolant volumes in the RPV that cannot follow the cool-down process (i.e. remain sub-cooled) may enter flash evaporation during depressurisation of the RCS.
- Coolant volumes in the RPV **up to ~ 0.5 m above the reactor coolant line (RCL)** participate in the cool-down process which represents the limiting factor for the void growth. The coolant volumes up to approx. 0.5 m above RCL are subjects to fluid interchanges with the sub-cooled NC that result from the inertia of the coolant flow through the core overshooting the height of the RCL line (Figure B28-A).
  - The sub-cooling span present a core outlet is also present at ~ 0.5 m above the RCL.
    - The maximum volume of the RPV upper head void comprises the RPV dome volume down to ~ 0.5 m above RCL.
      - As long as a sub-cooling span is present at the core outlet, the void volumes in the dome cannot expand into the RCL.
- After activation of 2 RCPs the upper head void does not collapse completely. Instead, it collapses up to the point where the coolant flow into RPV dome via the upper head bypass joints is no longer injected into steam (no more free jet) but into the now elevated coolant level.

The condensation of the residual void volumes in the RPV is a slow process mostly depending on heat losses across the RPV dome – more generally, it depends on magnitude of bypass flow rate, geometry of upper core structures and cross-sectional area of the interphase.

4.1.7 Results of Test G7.1

For details, see test report: [26]

Main findings for PKL counterpart test G7.1 (complementing test OECD Nuclear Energy Agency (NEA) ROSA-2 test 3) are as follows.

As long as the core is completely covered with 2φ mixture the primary pressure is close to the secondary pressure. After the occurrence of a partial core uncovery the primary pressure drops below secondary pressure (Figure B33) as not all the heat from the core contributes to formation of steam during core uncovery but also to heat up of the uncovered structures (increase of cladding temperature – CT).

When the core exit temperature (CET) has increased to approx. 350 °C (criterion for the secondary-side depressurisation) almost half of the core is uncovered (Figure B30; superheated steam above the SL).

Important phenomena observed during the secondary-side depressurisation and the subsequent primary-side depressurisation are:

- **Flashing in the core region** due to depressurisation of the RCS (secondary-side bleed) which leads to partial rewetting of tube cladding and the limitation of CT and CET [26].
- **Clearing of COL** as a consequence of cold side accumulator (ACC) injection.

  Condensation effects on the cold side induced by ACC injections effectuated a temporary displacement of water from the core to the downcomer. Following the ACC injection in the loop seals water is displaced from the SG to the RCP sides and COL clearing takes place.
- **Rewetting of core following COL clearing.**

  Only after loop seal clearing the entire core region is rewetted by both the reactor coolant from the loop seals and the ACC) water injected and causes a reduction of the CT and CET to saturation temperature.

In sum, the high effectiveness of the secondary-side depressurisation (followed by the ACC injection) in the restoration of the secondary-side heat sink and thus in the depressurisation of the RCS could be demonstrated.

Differences between CT and CET as regards responses in time and temperature differences had been expected and have been observed.

- A time delay in the increase in temperature between maximum CT and CET of 270 s emerged (related to CET of 350 °C).
- The maximum temperature difference measured between the maximal CT and CET during the core uncovery was 150 K (Figures B31, B32).

The final test conditions are in good agreement with the corresponding part of the ROSA/LSTF test.

4.2 OECD-ROCOM test results

4.2.1 Results of ROCOM Test 1.1 and 2.1

For details, see the test report [28].

Figures C2 and C3 show the time evolution of the averaged over the whole sensor plane and the minimum temperature inside the vessel of the ROCOM test facility during the **ROCOM Test 1.1**. The test starts from isochoric conditions. With growing time a quasi-stationary density field established inside the vessel. During these quasi-stationary conditions in the final injection phase time-averaged temperature profiles were obtained for the sensor planes in the DC and the core inlet.

Figure C4 shows the temperature distribution in the outer sensor plane of the DC. As can be seen the overcooled coolant from loop 1 enters the DC 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 and the density difference. Below this transition region in the lower part of the DC a nearly uniform temperature distribution is observed.

Figure C5 shows the temperature distribution in the core inlet plane. The measured temperature difference over the core inlet plane is about 5 K, only. The density difference between the coolants from different loops in combination with the differences in the loop flow rates could partly be responsible for the nearly homogeneous distribution. The initial conditions (temperature and flow field in the RPV) could contribute to mixing phenomena.

The variation of the boundary conditions in the **ROCOM Test 2.1** (decrease of density difference and decrease of difference between the loop flow rates) leads to changes in the observed mixing pattern in the vessel of the test facility. Figure C6 shows the time-averaged temperature distribution in the outer plane of the DC sensor. Two main differences can be observed in comparison to the temperature pattern of Test 1.1. First of all it can be seen that the transition region has been shifted considerably downwards. The boundary between the sector-shape distribution and the nearly homogeneous distribution is now located at the outlet of the downcomer. Due to the increased fraction of the flow rate form the perturbed loop the stripe itself is also wider than in Test 1.1.

In the core inlet plane (Figure C7) again no sector formation can be observed. But contrary to Test 1.1 the temperature distribution is more heterogeneous; the measured temperature difference is about 23 K. The minimum is located in the middle of the core inlet plane.

4.2.2 Results of ROCOM Test 1.2 and 1.3

For details, see the test reports [27] and [28].

Figure C8 shows the time-averaged temperature distribution in the sensors just downstream of the injection position in **ROCOM Test 1.2**, Figure C9 the distribution at the outlet of the loops. The time-averaging was done at the quasi-stationary level which establishes after the beginning of the ECC water injection. In all four sensors a clear stratification of the flow can be seen with the coolant with lower temperature in the lower part of the pipe. The difference in the base flow rates in loop 3 and 4 influences the position of the boundary between cold and warm water; in the loop with lower base flow this boundary is located at a lower level (see the distributions in Figure C9). In the sensor plane directly downstream of the injection position (Figure C8) clearly the perturbation of the temperature field by the ECCI from the side is to be seen; more pronounced again in the distribution at the sensor in the loop with lower base flow rate (loop 4).

This lower base flow rate in this loop (see Table C5) is also responsible that a part of the injected ECC water is transported into the opposite direction. In the loop with higher base flow rate (loop 3) no reverse flow was detected.

Figure C10 show the time-averaged temperature distribution in the DC (outer plane of the sensor). Unification of the two stripes in the upper part can be seen. The resulting combined stripe is not located in the middle between the two loops receiving ECCI. This can be attributed to the differences in the loop flow rates. In this test a temperature gradient in radial direction is observed between inner and outer plane of the sensor. The smaller temperature values are measured at the RPV wall. The maximum difference is about 40 K. Further a transition region in axial direction is formed. Below this region a nearly uniform temperature distribution is measured.

In **ROCOM Test 1.3** the behaviour of a single ECCI stripe in the DC was investigated. As can be seen from Figure C11, the ECC water flows directly below the inlet nozzle (position is marked by the arrow) downwards. By help of the results of this single effect test it can be now concluded that the bend in the cold leg before the entry into the DC does not create a momentum which leads to an asymmetrical stripe behaviour.

Based on the results of both tests (1.2 and 1.3) it is clear that the differences in the base flow in the loops contribute to the shift of the position of the combined stripe from both loops in the DC (observed in test 1.2). Whether this difference in the base flow is only contributor to the observed asymmetry can finally be clarified only with an experiment with ECCI into two neighbouring loops with symmetrical base flow in these loops. That would be the case if under such conditions the two ECC stripes would also behave symmetrically (i.e. joining after entering the DC and moving straight downwards in the middle between the cold leg nozzles).

Small azimuthal fluctuations of the stripe in the DC have been observed in this test.

4.2.3 Results of ROCOM Test 2.2

For details, see the test report [28].

Figures C12 and C13 show the time evolution of the average and the minimum temperature in the different sensor planes inside the vessel of the ROCOM test facility during the **ROCOM Test 2.2**. Like the Test 1.1 this test starts from isochoric conditions. The main difference in the first part of the test is the difference in the loop flow rate of the affected loop. Contrary to Test 1.1, where the filling of the DC with overcooled water starts immediately after the stabilisation of the flow rate, in Test 2.2 the boundary between the sector-shape distribution and the nearly homogeneous distribution is located at a lower level. This state persists more or less until the flow rate in the unaffected loops has decreased by half of its initial value (see Figure C14; at t = 50 s). Also, the average DC temperature starts to decrease after t = 50 s when the flow rate in the unaffected loops has been decreased by half (Figure C13).

In the core inlet plane again no sector formation can be observed. The measured temperature difference over the core inlet plane at this time point (t = 50 s) is about 10 K (Figure C15), what is higher than in Test 1.1. This difference can be contributed to the reduced difference in the loop flow rates at this time point in Test 2.2 in comparison to Test 1.1. A reduced difference in the loop flow rate enhances the influence of the unaffected loops which results in a higher heterogeneity of the results (increased measured temperature difference over the core inlet plane).

After this time the average temperature and also the temperature difference decrease continuously till the end of the test.

4.3 OECD-PMK Test results

4.3.1 Results of PMK Test 1

For details, see test report: [29]

The two tests performed with the pressuriser valve open during replenishment of the primary system (Test1.1A and Test.1.1B) indicated the importance of boiling in maintaining NC: while in T1.1A (with substantial boiling) the lowest SG tubes are supplied by water practically throughout the test, in T1.1B (with only limited boiling) the temperature measurement at SG tube inlet shows an interruption of heat transfer to the secondary side (Figures D1 and D2).

After refilling of the system there is little difference between natural-circulation flow rates in these two tests, as shown by Figures D3 and D4. This can be explained by the fact that NC is affected by the number of SG tubes covered by water. In both test runs the air trapped in the SG collectors and tubes could not be sufficiently compressed during the refill process and, as a result, NC was re-established only through a limited number of heat transfer tubes. The resulting core flow rates were just enough to assure sub-cooled conditions at core outlet as indicated by the core outlet temperatures in Figure D5.

The test results of T1.2 (with the pressuriser valve closed during replenishment of the primary system) demonstrate that closure of all valves is beneficial with respect to re-establishment of NC, because the pressure increase resulting from primary system refill assures sufficient level both in the vessel and in the SG. Figure D6 shows the evolution of cold leg flow rate in Test 1.2: the final values are considerably higher than the ones in Figures D3 and D4. Thanks to the higher flow rates the core outlet temperature is decreasing, as shown in Figure D5, which has a beneficial effect on heater rod temperatures as well as demonstrated by a comparison of Test 1.1A and 1.2 results, Figures D7 and D8, respectively.

The test results allow the conclusion that in case of a loss of primary inventory in a VVER-440 plant during lowering of the primary system level in preparation of core unloading, the operator has to close all venting lines before starting the refill process. This will help to resume NC, which might have been interrupted by coolant loss, and to assure core outlet temperatures well below saturation ones. Only after having achieved these conditions he should vent air trapped in vessel head, SG collectors and pressuriser, in order to return to normal conditions.

4.3.2 Results of PMK Test 2

For details, see test report: [30]

The overall test behaviour is illustrated by the results of the reference case, i.e. those of Test 2.1. The primary depressurisation after opening of the break is shown in Figure D9. After the pressuriser is emptied, steam is formed in the hot parts of the system that leads to pressure stabilisation at a level of 1.1 MPa, slightly above the secondary pressure, since the break size is too small to evacuate the heat produced in the core. As a result of the clearing process of the hot leg (HL) loop seal a pressure increase can be observed after 1000 s, finally leading to steam passing the loop seal and condensing in the SG that provokes a sudden drop in the primary pressure.

The level decrease in different parts of the reactor vessel model is shown in Figure D10, where the moment of HL loop seal clearing is indicated by a local minimum in the levels LE11 and LE22. By about 2 500 s a further decrease of the primary system levels can be observed, with a minimum level of 2.1 m in the vessel. This is so low that heater rod temperatures at the three upper-most instrumented positions start to escalate, as shown by Figure D11 for the highest elevation. As the set point of 350 °C of heater rod temperature is reached, secondary bleed by opening one relief valve (RV) is started. The bleed action induces condensation of steam on the primary side of the SG and, as a consequence, the core level quickly recovers and the heater rods are quenched. The depressurisation rate is fairly low and it takes almost an hour to reach the set-point pressure of the LPIS pumps. Before this happens, the primary inventory decreases to a level that provokes overheating of the heater rods again. However, the maximum temperature is limited to 365 °C by LPIS injection and, with the vessel level rising, stable core cooling is assured.

In Test 2.2 (and Test 2.3) the primary depressurisation is much slower than in test T2.1 (Figure D12) due to the presence of non-condensables: the pressure stabilises only at 1 250 s. The difference to the secondary pressure is almost the double of that in test T2.1 that is a consequence of the air escaping from the pressuriser and filling up the primary side of the SG, thus reducing heat transfer to the secondary side. As a result of the clearing process of the HL loop seal a pressure peak can be observed just before 2 000 s, finally leading to steam passing the loop seal. It is interesting to note that the primary pressure decrease rate in this period is much slower than in test Test 2.1, indicating the deteriorated heat transfer effectiveness in the SG tubes.

Due to the continued loss of the primary coolant via the break the primary system inventory decreases to a minimum level of 2 m in the vessel by about 3250 s (Figure D13). Even if the mixture level is significantly higher, heater rod temperatures start to escalate at the three upper-most instrumented positions, as shown by Figure D14 for the clad temperatures at the highest elevation. As the set point of 350 °C of heater rod temperature is reached, secondary bleed by opening one RV is started at 3 349 s. Although the bleed action leads to steam condensation on the SG primary side, this is much less effective in presence of the air than it was in Test 2.1: this can be seen on the core level evolution in Figure D13 as well that recovers only slowly. Quenching of the heater rods takes much longer, as a result, the maximum CT is much higher than in Test 2.1, it reaches 570 °C.

Here again – the depressurisation rate being fairly low – the primary inventory decreases to a level that provokes overheating of the heater rods, before the set-point pressure of the LPIS pumps would be reached. The maximum temperature is limited to 435 °C by LPIS injection and the latter assures stable core cooling.

In Test 2.3 the behaviour of the system up to secondary bleed initiation is practically identical to Test 2.2, as it is witnessed by the primary depressurisation (Figure D12) and the level decrease in different parts of the primary circuit (Figure D15). (This proves a good repeatability of the test as well.) The secondary bleed with two RVs leads to faster pressure decrease also on the primary side than in Test 2.2, but the effect on core level is limited, it recovers only slowly. Quenching of the heater rods is accomplished a bit sooner than in Test 2.2 (Figure D16) and, as a result, the maximum CT is just above 500 °C. The increased depressurisation rate allows reaching the set-point pressure of the LPIS pumps, before the primary inventory would decrease to a level that would provoke overheating of the heater rods again.

Comparison of results of the first two tests indicates that – in the case of air-filled pressuriser – the primary pressure (Figure D12) stays about 1.5 bar higher than the secondary one after depressurisation due to degraded heat transfer in presence of air in the SG. However, this difference is reduced to about 0.3 bar by the time, when core heat up occurs, as a result of more and more steam passing the HL loop seal. In the tests with air, the secondary bleed is much less effective to recover the core level due to degraded heat transfer in the SG and quenching of the heater rods takes longer after initiation of the secondary bleed. As a consequence, CTs reach higher values. As indicated by the third test, faster depressurisation of the SG secondary side reduces the maximum temperature.

5. DATA INTERPRETATION AND APPLICATION (DIA)

The purpose of this chapter is, on one hand, to make a first approach on interpretation of the test results obtained in the experiments with respect to significant phenomena and processes relevant for the individual pressurised water reactor (PWR) accident scenario; and on the other hand to assess possibilities and give hints for further interpretation and application.

In addition and completion of the data from Primärkreislauf (PKL2), the considerations in this chapter are in parts extensions to the results of the PKL III F test series performed in the PKL project.

While the test series PKL III F and E were basically focused on the system behaviour in relation to inherent boron dilution, the PKL III G tests rather shifted the focus to the analyses of basic principles and separate effects either to provide better understanding of individual phenomena or to provide a data basis for code validation.

A detailed characterisation of the heat transport from primary to secondary side for the relevant scenario is fundamental to all PWR safety analyses since it is directly linked with core cooling. The PKL integral test facility is well suited to experimentally investigate heat transport under various conditions such as natural circulation (NC), in presence of non-condensable gases or related to boron enrichment processes following leg break loss-of-coolant accidents (LB-LOCA) due to its design features, its scaling and the possibility to employ boric acid and appropriate measuring techniques. Apart from the fact that most of the experiments addressed code validation or benchmark activities some test results on plant behaviour discussed hereafter have to be compared and assessed with corresponding analyses using thermal-hydraulic system codes and, for some local phenomena, with additional separate effect tests. The interaction between experiments and codes will finally lead to qualified results for PWRs.

Essential parts of Chapter 5 have been provided by an international experts group, featuring contributions from:

Ivan Tóth, *KFKI Atomic Energy Research Institute, Hungary,* **Dr Sören Kliem**, *Helmholtz-Zentrum Dresden Rossendorf,* **AREVA NP GmbH**, *Germany,* operating agents of PMK, ROCOM and PKL, Sections 5.1: Phenomena by experiments, 5.2.1: The quality of the experimental data.

Claire Agnoux, *Électricité de France, France,* Section 5.2.2: The safety relevance of the addressed issues.

Eugenio Coscarelli/Andriy Kovtonyuk, *University of Pisa, Italy,* Section 5.2.3: The scaling value of the data including the connection with expected nuclear power plant (NPP) behaviour.

Miguel Sanchez-Perea, (editor-in-chief for DIA Chapter), *Consejo de Seguridad Nuclear, Spain,* Section 5.2.4: The value for code assessment.

Shawn Marshall, *Nuclear Regulatory Commission, United States,* Section 5.2.5: Specific additional lessons learnt from the execution of experiments.

5.1 Phenomena by experiments

The following sections give a short summary on the most relevant phenomena with respect to reactor safety associated with each scenario. Phenomena are discussed hereafter along with the experiments they were observed in:

- ¾-Loop operation scenarios (Figure A6).
- Cool down under asymmetric boundary conditions (Figure A7).
- Fast cool-down transients (Figure A8).
- Heat transfer under reflux-condenser (RC) condition, influence of secondary-side parameters (Figure A9).
- Boron precipitation processes following LB-LOCA (Figure A10).
- Formation and behaviour of upper head void during cool down (Figure A11).
- Effectiveness of secondary-side depressurisation (Figure A12).
- Core exit temperature (CET) performance (Figure A12).
- Characterisation of VVER steam generator (SG) heat transfer.
- LOCA during cool down in VVER configuration.
- Coolant mixing inside the reactor pressure vessel (RPV).

5.1.1 The ¾-loop operation scenarios

Test series PKL III F2 in the former PKL project already revealed some fundamental coherence between primary inventory, occurrence of the different heat transfer modes with prospect of boron dilution in some cases and level of stabilisation of primary pressure.

Heat transfer modes in the U-tubes, stabilisation of primary pressure

Loss of the residual heat removal system (RHRS) during mid (or ¾) loop operation at closed reactor coolant system (RCS) causes the temperature in the core region to rise with resultant steam production once saturation temperature has been reached; steam production is in turn associated with an increase in RCS pressure up to the equilibrium pressure level required for the transfer of the entire decay power to the secondary side. This causes the coolant inventory (CI) in the core to froth up and the swell levels (SLs) rise variously in the SG depending on how many SGs are operable at the time. Secondary sides filled with water act as main heat sinks; hence the frothed primary inventory is predominantly displaced to heat-removing SGs, manifested by rising SLs in their U-tubes. This represents RC conditions superposed by a SL present on the U-tube inlet sides. In absence of forward coolant transport (blocked by N_2), heat transfer may deteriorate (i.e. RCS temperature and pressure are higher) as more inventory is displaced into the heat-removing SG. In either case, pressure and temperature (i.e. ΔT to secondary) in the RCS rise up to the level, where a steady-state heat flux to secondary becomes established, sufficient to transfer the whole core power to secondary side.

Even in the absence of active operator interventions, a quasi-steady-state condition with assured heat removal to secondary side always becomes established even if only one SG is operable. The primary equilibrium pressure required for the removal of the entire decay power depends on the heat transfer area in the U-tubes and is thus directly connected to the SLs in the U-tubes (i.e. the equilibrium pressure depends on the distribution of the primary inventory among the number of water filled SGs).

Major parameters influencing the primary inventory have also been identified already in the transient tests within the PKL programme:

- influence of higher initial level in the RCS, see Test PKL F2.1; [16]
- hot pressuriser (PRZ) (less water displaced into PRZ during heat up compared to cold PRZ);
- only one operable SG (inventory is displaced towards this SG); and
- additional injection of inventory by active or passive systems.

The previous test programmes PKL III E and F (specifically test PKL III E3.1 [7, 15] and test series PKL III F2 [16, 17] and corresponding preliminary work relating to these tests) have already identified a range of heat transfer mechanisms and basic coherences involving the inlet-to-outlet-side transport of coolant with different effects (see Figure B1).

- (Figure B1, state 4) Stationary slugs of sub-cooled water present in all SG tubes with nitrogen enclosed in the tube bends above. Tall columns of sub-cooled condensate reduce the heat transfer zone to the bottom part of all tubes. Degraded heat transfer capacity in the SGs leads to relatively high RCS pressure.
- (Figure B1, state 3) Intermittent spillover of boron-depleted coolant through individual short SG tubes. Following the onset of intermittent spillover, slugs of boron-depleted coolant are transported over the apex of individual tubes because the condensation rate exceeds the coolant mass (CM) flow spillover. Equilibrium pressure on the RCS side is moderate and constant at 4-5 bar. This status provides potential for a continuous boron dilution process below the SG outlet tubes.
- (Figure B1, state 5) Continuous circulation in some short tubes. If CI is sufficient to initiate continuous overflow in the short SG tubes, regardless of 1Φ or 2Φ flow, boron dilution resulting from the heat transfer process can be ruled out.

The expulsion of N_2 (and onset of forward coolant transport) in SG tubes generally improves heat transfer to the secondary side (i.e. stabilisation at a lower pressure and temperature), but can lead to boron dilution in the loop seal due to condensate spillover in the U-tubes (as seen for only one operable SG). If an expulsion of nitrogen is impossible due to rising fill level on the SG outlet side prior to onset of coolant transport (e.g. as a result of additional emergency core coolant injections – ECCI) no overspilling or onset of 2Φ-NC will occur for primary pressures < 10 bar (for 1 SG activated).

In the systematic parameters studies on heat transfer (as function of the water inventory, see Figure B1) in the U-tubes of tests G1.1 and G1.2 almost all of the abovementioned heat transfer mechanisms and flow phenomena observed in the previous tests on the topic could be reproduced.

RC and occurrence of boron dilution

Generally, as RC conditions as such (regardless whether with or without SL) are associated with formation of condensate in the U-tubes, sections of the RCS volumes may thus become boron-depleted if low-boron coolant from the SG is being transported from SG inlet-to-outlet-side across the U-tubes. However, the frame for the occurrence of boron dilution (discontinuous flow phenomenon with high steam loads) as regards the combination of boundary conditions for the occurrence of the specific flow phenomena is very small. In fact, in the PKL tests of series F2 and G1, boron dilution (continuous accumulation of condensate below SG outlet(s)) as a result of coolant transport in the SGs was only observed for the following combination of parameters:

- RCS inventories corresponding to initial ¾-loop inventory. Additional ECCI at first create higher CMs that result in a rise of the pressure level and continuous circulation afterwards by trend.
- Only one active SG. The distribution of steam among at least two active SG would render the steam volumes and velocities too small to effectuate the flow phenomenon required for boron dilution. In addition, only half of the condensate is produced per SG.
- Steady-state (equilibrium) RCS pressure around four bar or less. Pressures at higher values result in a compression of steam volumes and therefore reduce the steam flow velocity in the U-tubes.
- Cold PRZ. The slow dislocation of coolant from the primary circuit after start of steam formation (condensation potential at cold PRZ wall structures) allows a distinct coolant reduction in the U-tubes of the heat sink and thus facilitates discontinuous flow phenomena in the SGs with higher steam loads (which provide a prospect of intermittent spillover as detailed for State 3).

5.1.2 Cool down under asymmetric boundary conditions

Stagnation of natural circulation

Stagnation of NC may arise from different effects. In the first case, the opposing driving force for NC in an isolated SG at elevated temperature level can become so large that it is larger than the driving force from the RPV. In the second case, saturation conditions in the U-tubes could be reached, causing evaporation and accumulation of steam in the U-tubes' apices as a result of the decrease in primary-side pressure and leading to interruption of the flow.

Under continuous cool down at 50 K/h with a loss of off-site power, the complete stagnation of NC in the loops with isolated SGs is expected, with or without additional pressure decrease via spraying in the PRZ. If auxiliary spraying in the PRZ is conducted to reduce the primary pressure, then, during a continuous cool down, evaporation must be expected in the SG U-tubes of the inactive loops even in the presence of sub-cooling at the core outlet.

In either case, the flow in the affected SG stagnates. Safety relevant consequences of interruption of flow in one or more loops may result from different aspects:

- The boron concentration in the affected loops cannot be increased.
- The activation of the RHRS cannot be assured if the temperature in the suction lines and primary pressure are not sufficiently low [31], and the RHRS remaining unavailable the cool down of the primary system becomes decreasingly inefficient due to the low SG cool-down capacity at low steam pressures.
- Stratification could take place as a consequence of stagnant flow and the operation of safety-injection systems (SIS), and cold plumes may cause thermal stresses on the RPV.

Test G2.1 proved a stepwise cool-down process to be able to maintain NC in all loops, regardless whether they have isolated (and emptied) or intact and operating SGs providing the magnitudes of the cool-down steps and cool-down gradients remain limited.

A cross comparison with older PKL test results allows comparisons on the magnitude of opposing pressure head for NC provided by the isolated but filled and emptied SG secondary sides (as conducted in G2.1). The test results provided by isolated but filled SGs [32] are distinct from the current ones. Indeed the water masses of hot isolated SG secondary sides are able to discharge more heat to the corresponding primary loops and therefore interruption of natural circulation (NCI) occurs earlier.

5.1.3 Fast cool-down transients

In the course of the main steam line break (MSLB) scenario postulating a 10% non-isolable break in the containment the affected SG is quickly isolated on feed water and main-steam-side due to the violation of protection limits. As a result of the discharge of heat from the RCS via the uncontrolled heat sink the coolant volume in the affected loop at first experiences an intense cool-down transient, a contraction of volume and consequently a pressure decrease in the RCS. In this context, the assessment of the RPV integrity considering pressurised thermal shock (PTS) aspects is one important point for this accident scenario. The assessment of potential re-criticality due to colder water entering the core area is another important subject.

Fast cool down in affected loop, impact on NC and RPV inlet temperature

The occurrence of the break initiates a rapid boil-off of the affected SG resulting in an intense heat extraction from the affected loop that leads to a distinct sub-cooling margin at the RPV inlet nozzle in the associated loop. The safety relevant aspect of re-criticality is assessed on the basis of the velocity and temperature field at core inlet. Crucial for the conditions of flow (velocity, temperature) at core inlet are: the evolutions of the NC loop flow rates from the loop featuring the affected SG as well as from loops with intact SGs and the coolant temperature in the outlet plenum of the affected SG. In the PKL G3.1 test, continuous circulation in all four loops is maintained, however, in the loop with affected SG the flow ratio increases in contrast to the flow in loops with intact SGs. Both parameters, NC and sub-cooling significantly determine the conditions of flow at the RPV inlet and therefore the mixing process in the DC and lower plenum (LP) as well. Complementary experiments in the **Ro**ssendorf **Co**olant **M**ixing (ROCOM) test facility have been conducted to analyse mixing in the DC and in the LP.

Activation of injection systems, stratification in cold leg, PTS

An additional, important aspect of this accident scenario concerns RPV integrity under consideration of PTS due to the introduction of cold water in the RPV DC. This case, relevant for some PWR designs, is important above all when the cooling of the primary coolant is intensified by injection of cold emergency core coolant (ECC) into the cold leg in a rather cold primary circuit due to MSLB. Moreover, the pressure increase at the end of the transient tends to increase the mechanical load.

The injection of cold ECC into already present NC, either from loops with affected or intact SG creates a more or less distinct stratification downstream of the ECCI nozzle. The effectiveness of mixing at and downstream of the ECCI nozzle is mostly dominated by the magnitude of the ECCI flow. Conservative boundary conditions as regards the magnitudes of stratification and PTS at the RPV inlet nozzle are provided by the loops that combine safety injection (SI) with low NC flow rates as it is the case for the loops with intact SGs. In such loops the tendency for mixing directly at or downstream from the injection nozzle is decreased and the cold plumes may cause thermal stresses on the reactor pressure vessel as they enter the RPV almost completely undisturbed.

A possible backflow of injected and stratified ECC towards the reactor coolant pump (RCP), in counter-current flow to the present NC would mitigate PTS. Mixing temperature and magnitude of a possible backflow also depend on the NC/ECC flow ratio as well as on the geometry of the RCP design (in particular on the height of the guide vane diffuser). In the PKL test no backflow was observed.

5.1.4 Influence of secondary-side parameters on heat transfer under RC conditions

RC conditions are characterised by low primary inventory and very effective heat removal based on the condensation mechanism within the U-tube primary sides. A dropwise or film condensation represents the most effective way of heat transfer. Due to the very small temperature difference between primary and secondary sides, both circuits remain closely linked in pressure and temperature even at higher cool-down speeds.

Counter-current flow limitation (CCFL) and dislocation of inventory at high specific SG loads

At high specific SG loads the condensate reflux in SG U-tubes and inlet plena may be hindered due to high steam velocities in counter-current direction (CCFL). Intense heat flux to secondary side imposes a displacement of coolant towards the heat sinks on the primary side (SG U-tubes). CCFL and dislocation of inventory may become safety relevant, for example, if the activation of the emergency feed-water system suddenly occurs at already reduced primary inventory (RC conditions) as a result of its re-availability in the course of a loss of feed-water transient or at the end of station blackout.

5.1.5 Boron precipitation processes following LB-LOCA

The speed of boron enrichment in the core depends on several influence parameters:

- The size of the mixing volumes in the RPV (amount of liquid mass participating in the enrichment process). The more liquid coolant is available in the mixing volume, the slower the boron enrichment process will be.

  Possible high flow resistances between upper plenum and cold leg break (e.g. cross-over legs filled with coolant of 2Φ mixture, high core power) induce a higher pressure in the upper plenum (UP) which results in a depression of the SL (and mixing volumes) in the RPV.

- The core power directly (evaporation rate, void fraction in core) and indirectly (pressure in the UP, swell-level depression) determines the precipitation progress ECCI rate and injection location as well as primary side filling status directly define the increase or decrease of boron concentration in the core region and adjacent volumes.

Mechanisms of mixing

Important for the size of the total mixing volume (i.e. the amount of liquid coolant available for mixing) are the mechanisms that interconnect the different coolant volumes in the RPV sections. The mechanisms of mixing in the RPV (see Figure 3) involve the core region, the reflector gap – even for SLs that just cover the core (by overflow across the grid plate) – and most parts of the lower plenum. For higher SLs the volumes of the UP, the hot legs (HL) and – by fluid exchange across the HLs – even the SG inlet chambers (resulting in boron depositing in the SG U-tubes) also participate in the enrichment process.

Swell-level depression

No less important for the size of the mixing volume and the sustainment of mixing mechanisms is the hot side SL above the core. Apart from the core power, the hot side SL is significantly determined by the pressure in the UP as the build-up of a higher pressure reduces the steam volumes and suppresses the SL (phenomena of "swell-level depression"). In such scenarios (cold leg LB-LOCA) the pressure balance between hot (UP) and cold sides (break) of the RPV determines the heights of the swell and fill levels in core and DC, respectively, and a high pressure in the UP reduces the height of the hot side coolant column above the LP. Of course the parameters for the UP pressure are mostly given by the PWR design concept:

- Upper head bypass flow, head losses along upper head bypass.
- Head losses along the loop, determined by SG U-tubing and RCPs at standstill.

With respect to the head losses in the upper head bypass, test G5.1 arranged the most unfavourable conditions by eliminating the bypass flow completely.

As regards the loop flow resistances towards the break the KONVOI RCS geometry (simulated in PKL) features head losses much too low to create a significant swell-level depression. It can be expected that for PWRs of different design, the head losses along the loop and the RCPs may be higher. In particular the height of the cross-over legs (COLs) relative to the core is of great influence, as a “lower”, more distinctive COL may hold a much larger coolant volume that requires a much higher pressure difference between break and the UP to be cleared.

Operator intervention – Core flushing

For the reversion of the precipitation process the ECCI must be at least partially switched to HL injection. The boundary limits for this point in time for the switch over are given by the projected time until formation of boron crystals on one hand, and by the time interval needed for the core power to decrease on the other hand. Due to the high core power at the beginning of the course of events, the early switch to HL injection is inappropriate due to high steam velocities associated with the core power of above 2%. In the early state of the long-term cooling phase the ECC injected at the hot side is assumed to be diverted towards the SGs and thus hinder a continued cooling of the core. Consequently, for PWRs preferably featuring cold leg ECCI, the switch to HL injection for a reversion of the enrichment process in the core has to be delayed until the steam velocities have reduced sufficiently as a result of decreased decay heat. According to the PKL test results with respect to the size of the mixing volumes, the grace period until a “core flushing” (by switch to HL injection) may be required to sustain/re-establish core cooling is long enough to allow the switchover to HL injection at significantly decreased decay power level (distinctly below 1% for t > 6h after scram).

5.1.6 Formation and behaviour of upper head void during cool down

During the shift from forced circulation to natural-circulation condition the temperature distribution in the RPV changes significantly. The heat up span across the core for decay power of ~ 1.9% increases from approx. 2-3 K under forced circulation (RCPs operating) to approx. 25 K under NC-conditions. During depressurisation on the primary side, a void formation in the RPV dome may occur if the fluid in the RPV dome is still hot and enters flash evaporation as the RCS pressure drops.

Fluid temperature in RPV dome – sweeping flow

Crucial for the maximum volume of the upper head void are the temperatures of fluid and structures in the upper RPV. Depending on the RPV geometry, – in particular on the flow paths and head losses between UP, RPV dome and upper downcomer (DC) – the coolant and structures in the RPV dome either slowly follow the increase of the core outlet temperature during shift to NC-conditions, or – in case of almost complete separation of RPV dome volumes from UP – remain at almost constant level (closer to core inlet temperature). Determining for the temperatures of fluid and structures in upper RPV are the magnitudes of the coolant flows possible between UP and dome that may establish after the reversion of pressure conditions in the RPV between hot (UP) and cold sides (DC) following the coast down of the RCPs. The pressure balance under NC-conditions produces a more or less significant flow of coolant (“sweeping flow”) from the UP across the RPV dome into the upper DC depending on the head losses of the flow paths given by the RPV design. In case of a more “open” configuration providing a stable flow from UP to DC, the dome fluid temperatures remain – with specific delay – connected to the core outlet temperature.

The geometry of the upper core structures of a KONVOI RPV which is replicated by the PKL RPV simulator allows a fluid exchange between UP, RPV dome and DC. Under stable NC-conditions the dome fluid temperatures follow the CET with a small time delay.

Another critical aspect for the formation and sustainment of the upper head void are the heat losses across the RPV dome structures. In the G6.1 test, the evolution of the dome temperatures during shift from forced to NC (in KONVOI closely linked to CET) and the cool down of the dome fluid temperatures due to heat losses (in the order of 3-5 K/h) were replicated very well.

Limitation of upper head void volumes

During depressurisation of the RCS the fluid in the RPV dome reaches saturation conditions and enters flash evaporation. The void volumes, in theory, grow up to the point where its lower boundary reaches sub-cooled water that cannot enter saturation conditions, either due to heat exchange with structures or as a result of fluid exchange with colder (distinctly sub-cooled) layers of coolant below (e. g. as a result of sub-cooled NC). In the PKL test the coolant volumes in the RPV **up to ~ 0.5 m above the** reactor coolant line (RCL) participate in the cool-down process which represents the limiting factor for the void growth. The coolant volumes up to approx. 0.5 m above RCL are subjects to fluid interchanges with the sub-cooled NC that result from the inertia of the coolant flow through the core overshooting the height of the RCL line.

For the steady void fraction, the dome structures act as heat sources for the void volumes. The heat released from the structures is used to further heat the steam volume which would result in an expansion of the void volumes. In the PKL test, featuring a rapid void formation up to its maximum expansion (limited by NC), the heat released from structures is removed either by absorption in the sub-cooled flow (as the void volumes cannot expand any further) or removed via heat losses across the RPV walls. Consequently, in the PKL test, superheating conditions at the dome and UP (indicated by the drop of the saturation temperature) was preserved by the heat released from walls and structures.

Collapsing of the upper head void following activation of RCPs

No complete collapsing of the upper head void volume was observed for the re-activation of two out of four RCPs at once. Instead, the quick rise of the RPV fill level stops as soon as the water injected via the bypass lines into the dome is injected into water. The condensation of the steam is highly effective, as long as the cold water (coming from the bypass lines) is injected as a free jet into the steam volumes in the dome. The high effectiveness of heat removal from the dome through absorption of heat by a cold free jet of water is lost as soon as the injection nozzles are covered by the rising RPV fill level and the water is then injected into water. Thenceforth, heat from the void volumes in the upper RPV dome (still in superheat) is transferred either across an interphase with adjacent saturated layers similarly to the conditions at a fully formed upper head void volume at lower fill levels; or – rather the dominating effect – by the heat losses across the RPV closure head, which are in the order of 3-5 K/h.

5.1.7 Effectiveness of secondary-side depressurisation

In the case of a small HL break, without high-pressure safety injection (HPSI) and no automatic secondary side cool down, the employment of accident management (AM)-measures is a necessity for the reduction of the primary pressure to reach the accumulator (ACC) injection pressure in time to prevent core damage due to loss-of-coolant inventory via the break. The point of imminent core heat up due to coolant loss defines the ultimate point in time for initiation of AM measures. One option to reduce the primary pressure is the employment of a secondary-side depressurisation. Main purpose of the secondary-side depressurisation is to induce a fast drop of primary pressure so that accumulator ECC becomes available before the peak cladding temperatures (PCT) rises too high. Regarding the effectiveness of these AM measures the most interesting feature is the primary pressure gradient that is required in order to reach very low pressures in a short period of time.

Phenomenology of secondary-side depressurisation

From initiation of the SG depressurisation until onset of ACC feed, the energy removed by the secondary side as a direct consequence of the AM measure can clearly be identified to be the dominant energy removal mechanism: RC conditions on the primary side U-tubes account for a very efficient heat removal from primary to secondary. As soon as the secondary-side pressure falls short of the primary pressure, the primary pressure is dragged down simultaneously; the availability of the full heat transfer area and film condensation on the primary side cause the primary pressure to remain closely coupled to the secondary pressure during the depressurisation process. Even at very high secondary-side cool-down gradients (~ 8 bar/min, 800 K/h between start of secondary-side bleed-and-feed and ACC injection) during depressurisation the required temperature differential between primary and secondary side was not greater that ~2 K, proving that secondary-side depressurisation is a highly effective way to reduce the primary-side pressure.

During depressurisation the flashing of the core inventory and the onset of rapid steam flows to the condensation areas in the U-tubes provide a limited core cooling effect; relevant parts of the primary (core region) and secondary sides are characterised by saturation conditions. In essence, this is even true for the period with cold-water injection from the cold side ACCs. The behaviours of the primary pressure and the ACC injection flow is then mainly determined by the balance between steam formation on one hand and steam removal by condensation and leakage on the other hand. With respect to steam formation this balance is mainly given by the residual heat modulated first by storage (during core heat up) and later by release of a part of this heat in and from the core structures, respectively. Further contributions result from heat releases from the vessel structures with onset of the sudden primary cool down due to secondary-side depressurisation. In Test G7.1 the ACC injection resulted in strong condensation on the cold side that temporarily dislocated coolant from the RPV core to the DC. The temperature quench of the superheated claddings occurred after COL clearing and temporarily caused additional steam formation in the core resulting in a temporary pressure support effect. Despite this temporary additional steam formation, ongoing energy removal to secondary side (steam condensation) kept on driving and supporting the primary cool down by maintaining a constant and complete ACC feed and a consecutive transition to the low-pressure phase with the low pressure safety injection (LPSI) active.

Depending on SG load (i.e. intensity of heat removal) and primary pressure, CCFL may occur in the SG U-tubes or HLs. This is relevant because a redistribution of coolant due to condensate holdup in the SGs and increasing lack of coolant in the core may result in a serious delay of temperature quench in the core region and may thus at first aggravate the situation.

The occurrence of CCFL is determined mainly by two conditions:

- **The SG load**, i.e. secondary-side cool down gradient, which is determined by the number of opened main steam relief valves (MSRV, i.e. the size of the main steam relief cross-section).
- **RCS pressure** determining the steam flow velocities in the HLs and U-tubes.

Provided the occurrence of distinct CCFL the impact on the core cooling then still depends on the **residual RCS CI** (influenced by ECCI).

Under the given boundary conditions in test G7.1, CCFL was not observed to cause displacements of CI sufficient to endanger the core cooling.

5.1.8 CET Performance

PKL Test G7.1 is representative of typical accident transients necessitating AM procedures. One of the main objectives is investigation of the performance of the CET during periods of core uncovery until recovery of core cooling and to provide answers to the question whether it reflects the situation in the core within acceptable margins. In principle, core heat up starts as the SL in the core drops below the upper edge of the heated length and the upper ends of the heater rods are no longer sufficiently cooled. A deterioration of heat removal from the rod surfaces results in the rod claddings heating up. The resulting superheat of the steam that passes the rod surfaces is then recorded by the CET, but with a considerable temperature difference and time delay between CET and the maximum measured cladding temperature (CT), – not necessarily identical to the PCT, see Section 2.1.7 and [26], which are influenced by various parameters.

In Test G7.1 a considerable gap of 150 K between the maximum CT and CET was measured. A significant contribution emerges from the heat transport mechanism between cladding surface and steam that mostly depends on the properties of the steam flow (density, moisture, flow velocity and pattern) and the thermal output distribution along the fuel rod (i.e. axial power profile). Poor heat transport from rod cladding to superheated steam that passes the surface accounts for a temperature difference of approx. 75 K between the rod surface and fluid temperatures at the corresponding elevation.

Between the rod surface and the CET thermo couple, additional possible sources of influence of the CET measuring signal with respect to time and temperature differentials result from heat exchanges with colder surrounding structures in the flow path or from shielding/protection tubes housing the CET measurements. A possible distortion of the CET signal by coolant reflux resulting from RC conditions was avoided in Test G7.1 by the break size being chosen to have the primary pressure falling short of the secondary-side pressure shortly after start of core heating up.

During depressurisation, being either slow due to continuous break flow or fast in case of secondary or primary-side depressurisation, the entrainment of water with the steam flow from the 2Φ mixture in the core (more pronounced at LPs) may also disturb the CET measurement. In summary, the measured temperature difference between CET and maximal CT of approx. 150 K in Test G7.1 was dominated by the heat transport between cladding and steam flow with a contribution of roughly 75 K. Further delays were caused by heat transfer to colder structures like the core plate and the shielding tube.

Generally, time delay and temperature difference between PCT and CET following the deterioration of the core cooling are characteristic for each type and position of the installations as well as for the accident transient. This dependence between magnitude of the temperature and time difference and the transient's history and boundary conditions have to be considered carefully for each PWR design, to always assure a reliable indication of the core cooling status.

5.1.9 Characterisation of VVER SG heat transfer during preparation for core unloading

Systematic investigation of the degradation of SG heat transfer and of the disturbance of NC during lowering of the primary system level in preparation of core unloading is of high safety importance. During shut down, the core cooling in a VVER is not assured by primary side RHRS – as in PWRs –, but by heat removal through the SG with NC on the primary side and the secondary side filled with water. In preparation of reactor vessel lid removal valves are opened at the top of reactor vessel, PRZ and SG collectors and the primary system inventory is drained allowing air to fill up the higher parts. It is important to know, how heat transfer to the secondary side is affected, when the primary level is decreased, especially, if it incidentally drops below the one specified for vessel head opening.

Changes in SG heat transfer effectiveness can be inferred from changes in the NC flow rate on the primary side as the primary mass inventory decreases. There is practically no change in NC for a while, as the primary level is decreased, even though the upper-most heat transfer tubes of the SG are depleted from water. In this period only a slight increase of the core outlet temperature can be observed. If the vessel level is further decreased, although still remaining above the HL, the decreasing number of active SG heat transfer tubes leads to substantially lower NC flows and, consequently, to higher core outlet and heater rod surface temperatures.

When the vessel level drops below the HL, single-phase NC is interrupted leading to strong oscillations both in the levels and in the flow rate, which is caused by boiling in the core. Depending on the amount of steam produced in the core and escaping via the valve at the vessel top, a fraction of the steam passes the HL loop seal. This may assure intermittent water supply to the lowest heat transfer tubes, assuring NC with strong oscillations.

An obvious mitigating measure to cope with the deteriorating NC is to refill the system. However, air trapped in the SG collectors and tubes may not be sufficiently compressed to re-establish efficient NC. This also depends on whether all valves at the top of the primary system are closed before refilling starts or e.g. the one in the PRZ remains open.

5.1.10 LOCA during cool down in VVER configuration

The original VVER-440 cool-down procedures below 2.5 MPa foresee the following actions:

- steam atmosphere of the PRZ is replaced by nitrogen;
- accumulators and high-pressure safety injection (HPIS) are disconnected from the primary system;
- automatic start-up of LPIS is disabled.

A LOCA in this situation results in nitrogen injection to the primary system, most of it being collected in the SGs, thus affecting heat transfer to the secondary side. In case of a small break (SB) – since accumulators and HPIS are not available – there is a competing process between cladding temperature rise and pressure reduction to LPIS set point that strongly depends on the effectiveness of heat transfer in the SG, the latter being impacted by the presence of nitrogen. Obviously, secondary bleed is an important action for reaching LPIS injection in time.

In case of a cold leg break, once the PRZ is emptied, nitrogen from the PRZ is transported towards the SG. Due to steam formation in the hot parts of the system the pressure stabilises slightly above the secondary pressure, since the break size is too small to evacuate the heat produced in the core. When the vessel level drops to the elevation of the HLs, the clearing process of the HL loop seal starts triggering a system pressure increase. More and more steam is produced in the core until the steam finally passes the loop seal and condenses in the SG. This provokes a sudden drop in the primary pressure.

Since the cold leg loop seal does not allow steam to escape to the break, the primary level soon drops down to the heated core and heater rod temperatures start to rise. If the cold leg loop seal is cleared, the core might be cooled temporarily, but with the continued mass inventory loss the only way to stop the heater rod temperature escalation is to actuate secondary bleed. The bleed action leads to condensation of steam on the primary side of the SG and, as a consequence, the core level quickly recovers and the heater rods are quenched. Since the secondary pressure is around 1 MPa, the depressurisation rate is fairly low and it takes a long time to reach the set-point pressure of the LPIS pumps (0.7 MPa). Before this happens, the primary inventory may decrease to a level that provokes overheating of the heater rods again. It is important to demonstrate that – in spite of the detrimental effect of nitrogen – the LPIS injection point can be reached in time to limit the maximum core temperature and assure stable core cooling.

5.1.11 Coolant mixing inside the RPV

Mixing of overcooled water; core inlet temperature distribution

After opening of the leak in the steam line the heat transfer from primary to secondary side in the corresponding loop increases causing a continuous overcooling of the coolant in this loop. This overcooled water mixes in a certain way with the coolant from the unaffected loops when entering the DC. This mixing is an inherent safety feature because the perturbation created in the loop is mitigated on the way to the reactor core. The established core inlet temperature distribution is sensitive to the velocity and temperature ratios of the individual loops. The better the mixing is (i.e. the more the core inlet distribution deviates from a sector formation where the overcooled water is concentrated in a quadrant of the core inlet plane) the smaller is the possibility of reaching re-criticality during such an accident. Different combinations of velocity and temperature ratios based on the PKL G3.1 test have been used in a three experiment test series at the ROCOM test facility to determine the core inlet temperature distribution. In all cases buoyancy-driven coolant mixing on the way to the core inlet leads to a smoothing of the perturbation and a nearly homogeneous temperature distribution. For all the test configurations with RCPs off, the existing density differences between affected and non-affected loops induced a good mixing in the DC and/or LP, even for small density differences.

Mixing of ECC water

The injection of cold ECC water into the loops causes stratification of the coolant in the cold leg. The height of the stratification plane between the ECC water and the coolant of the loop depends on the value of the NC in the loop. A low natural-circulation flow can lead to the situation that a part of the ECC water flows in the backward direction towards the main circulation pump. On the other hand the lower the natural-circulation flow in the loop also results in poorer mixing of the ECC water with the base flow. This increases the load onto the RPV wall when the ECC water stripe enters the downcomer. In the ROCOM experiment with activation of two ECC lines a coalescence of both ECC water stripes was observed after entering the downcomer. Further a gradient in radial direction was observed with the lowest temperature at the RPV wall. Due to the differences in the NC in both loops the minimum temperature was observed away from the inlet nozzle. Further fluctuations of the ECC water stripe were observed in the downcomer.

5.2 Evaluation of the experimental database

This section comprises the results of a discussion among the partners of the PKL2 project. Several individual items relative to the evaluation of the PKL experimental database have been discussed under different aspects.

5.2.1 The quality of the experimental data

The general perception of the project partners is that the quality of the experimental data is excellent. Moreover, a large effort has been made in performing data analysis (extraction of information from the test results). The documentation provided by the operating agents on the experimental data, including all the post-processed curves, was very detailed and well organised.

Detailed criteria to assure appropriate quality of experimental facilities and of test data for code assessment have been defined in [33]. The facility and test qualification matrix were developed in a parallel, because of the close interconnection between the qualities of facilities and the tests performed. According to the qualification matrix the following requirements should be fulfilled:

- representativeness of the test with regard to the reactor conditions including the range of parameters;
- quality of the data measured with adequate instrumentation and with acceptable uncertainties;
- quality and completeness of test documentation;
- scaling considerations and boundary conditions.

The first and the last aspect are discussed in Sections 5.2.2 and 5.2.3, respectively; in this section the quality of data and of the documentation are covered. The high quality of these latter requires adequate description of data, data acquisition, processing and evaluation as well as documentation, including quantification of measurement errors and uncertainties.

The evaluation and analysis of the safety relevant phenomena and processes of the scenarios addressed within the cited PKL2 tests relies above all on the correct characterisation of heat transport capabilities of both the primary and secondary sides. Therefore, the quality of the experimental data can be assessed against the relevant thermal-hydraulic phenomena and key parameters of the modelled transients.

PKL test facility

As regards the design concept (original heights, four loops, all relevant primary and secondary safety and operational systems) and instrumentation and control system (I&C), the PKL tests facility is well suited for the performance of integral large-scale experiments on PWR safety. Detailed characterisation tests (e.g. on heat & pressure losses) show the good accordance with reactor conditions on one hand and facilitate test interpretation and replication with thermal hydraulics (T/H) system codes on the other hand. The scaling concept of the test facility itself has proven to be suitable to yield consolidated findings that may mostly be applied to PWR scale at least in qualitative terms.

With respect to number and diversity of measuring installations needed for a reliable and detailed recording of the phenomena addressed by the individual experiments, the PKL facility provides sufficient numbers of measurements with enough redundancies and accuracy. To support the quality of the data with respect to depiction of relevant phenomena the available measuring installations (e.g. metering capacities) are adjusted during test preparation to fit the special thermal-hydraulic phenomena and system responses addressed by the test. If required, additional measuring installations are employed where appropriate (e.g. new TC-lances in the UP for test G6.1). All aspects of the measurement signals including actual installation point, the data acquisition system accuracy, and operational availability, are assured by regular comprehensive adjustment and checking (usually as part of the test preparation phase). In addition, the metering ranges of all voltage amplifiers are checked, their linear smoothing functions (comprising standard deviation, offset and amplification factors) are re-calculated prior to every test run, facilitating the identification of defective or non-linear amplifiers.

From a qualitative point of view the measured and recorded data undergo a quality and plausibility check by means of energy and mass balances and cross comparisons with complementary measurements. Possible aberrations and unforeseen events are recorded and documented during the test run.

The personnel operating the facility benefit from over 160 experiments performed at the PKL test facility up to date. The PKL test facility and the operating personnel are part of an accredited thermal-hydraulic laboratory platform[7] at AREVA NP.

7. Accredited test and inspection body under the terms of ISO 17025:2005 and ISO 17020:2004 Internationally accepted according to International Laboratory Accreditation Cooperation.

PMK test facility

The PMK facility and the tests performed were assessed against the facility and test qualification matrix in [34, 35] and it was demonstrated that the requirements were fully or, at least, partially satisfied.

In terms of the matrix the quality of the facility can be evaluated by the quality of the design, construction, operation, use in international framework and personnel qualification. It was shown that the design of the PMK facility ensures preservation of the time scale of the phenomena because of the full height of the main components, the full pressure and temperature, the use of water as coolant and correct consideration of power to mass and power to volume ratios. Design solutions were strictly followed in the construction and the capabilities of the facility were demonstrated by shake-down tests. The operation of PMK is ensured by qualified personnel, their skill has been enhanced by the long years of experience in running the facility. A considerable number of the PMK tests have been performed in international framework, e.g. in EU-PHARE and EU-Framework projects, or under the auspices of the IAEA and the US NRC CAMP programme.

The PMK facility is equipped with adequate instrumentation of sufficient accuracy to grasp the relevant phenomena expected in the test. Depending on the specific needs of the test, the instrumentation has been adapted, as it was also the case in the present PKL project. In the test addressing natural-circulation behaviour and SG heat transfer additional thermocouples (TCs) were added to the HL and SG primary and secondary sides, in order to better characterise SG heat transfer degradation. In addition, in order to measure the coolant velocity in different SG tubes a cross correlation technique was applied based on the signals of two TCs located near the inlet of the tubes. In the test addressing a LOCA during plant cool down, the main emphasis was on the effect of nitrogen escaping from the PRZ during lowdown. In order to trace the distribution of nitrogen in the primary circuit –besides the traditional void probes indicating the passage of coolant level at numerous locations in the facility– advanced local void probes have also been installed.

The electrode of these probes is replaced by a micro-thermocouple and in this way they are able to detect the phase (liquid/gas) and to measure the temperature exactly at the same location [36]. This makes it possible to distinguish portions of sub-cooled gas from sub-cooled liquid that allows detection of non-condensable in the given area.

ROCOM test facility

The ROCOM test facility was erected for the investigation of coolant mixing in the RPV of PWR. The experiments at ROCOM are carried out with the goal to measure the time-dependent distribution of the coolant temperature and/or boron concentration inside the complex geometry of the pressure vessel with its internals. All the internals of the vessel of the corresponding KONVOI PWR are replicated in full detail in the facility. Special emphasis was given to the scaling considerations. All relevant dimensionless numbers have been checked with the selected scale of 1:5. This scale is also in agreement with international facilities with a comparable designation. The ROCOM test facility has been used in different international projects (e.g.: [37, 38]). For several years the experimental data obtained at the ROCOM test facility are being directly used in the oversight process for German PWRs (e.g.: [39]).

The test facility is equipped with sensors provided with a special measurement technique that allows data to be obtained which help to clarify the coolant mixing mechanisms. These data form a data basis for the comprehensive validation of computational fluid dynamics (CFD) codes. In order to fulfill both objectives the sensors are distributed in the whole test facility forming a very dense grid of measuring points [40]. The main mixing phenomena take place in the DC of the facility. For that reason the number of single measuring points in the DC has been doubled for the experiments conducted within the PKL2 project. With ca. 3 800 measuring points only in the DC the facility is well suited for the creation of CFD-grade experimental data.

The personnel operating the facility rely on the experience of conducting mixing experiments at the ROCOM test facility over more than 10 years.

5.2.2 The safety relevance of the addressed issues

5.2.2.1 Systematic investigation of the heat transfer mechanisms in the steam generators in the presence of nitrogen, steam and water

The behaviour of a SG in the presence of non-condensable gases was thoroughly investigated in this experimental program by a PKL test and two PMK tests.

PKL¾-loop operation scenarios

Loss of RHRS transients have been the subject of an increased attention, as probabilistic safety assessments have shown that accidents under shutdown conditions have a larger contribution to core damage frequencies than originally anticipated.

Several tests on loss of RHRS have been carried out in the previous PKL test programme, either with an open or partly closed RCS. Tests performed with the RCS closed showed that residual heat removal is assured if at least one SG is still operable. However, these test results also show that under certain boundary conditions, the cold leg boron concentration can be significantly reduced in the long term. Within the PKL2 Project, the new tests provided a systematic analysis of the highly complex thermal-hydraulic processes that occur in the SGs in the presence of nitrogen, steam and water, and analysed the phenomena related to boron dilution and heat transfer to the secondary side.

The goal of the G1 test has been to contribute to the understanding of the overspill phenomena (in presence of N_2), which also plays a role in the boron dilution process.

The results can be used to assess real PWR T/H behaviour, since the pressure level and elevation height are at full scale, and the SG tubes are similar to PWR ones. The tests also provide for additional results to create a database for validating computer codes and their models under these conditions, as they give information on heat transfer mechanisms in the low-pressure range and in the presence of nitrogen.

PMK disturbance of NC during the lowering of the primary system level

The test has investigated the degradation of SG heat transfer and the disturbance of NC during the lowering of the primary system level in preparation for core unloading when the upper parts of primary (upper head and PRZ) are filled with air. If the level drops below that specified for vessel head opening, this leads to the NCI and boiling in the core develops.

Refilling of the system was performed in order to check the effectiveness of mitigating measures.

The different runs of the test allowed a study of how the amount of steam produced in the core before replenishment and the status of the valve in the PRZ (open or closed) during refill affected NC in the system and heat transfer to the secondary side.

The test results are not directly transposable to VVER reactors, mainly due to the single-loop design of the PMK facility: in a real plant some asymmetry can be expected among the coolant loops. In spite of that, the effect of the HL loop seal and that of the amount of steam produced in the core on NC will be very similar in the plant. Based on the test results it can also be stated that – should a loss of primary inventory occur in a VVER-440 plant during lowering of the primary system level in preparation for core unloading – the operator has to close all venting lines before starting the refill process. This will help to resume NC, (which might have been interrupted by coolant loss) and to assure core outlet temperatures remain well below saturation ones. Only after having achieved these conditions should the operator vent air trapped in vessel head, SG collectors and PRZ, in order to return to normal conditions.

PMK Small break LOCA during cool down in a VVER

In a VVER, the PRZ is filled with nitrogen in the final phase of normal operation cool down to cold shutdown. As a consequence, in the event of a primary break during cool down, the loss of primary inventory leads to the injection of nitrogen into the primary circuit. Nitrogen can compromise SG effectiveness. During cool down, when the system pressure is lowered to 2.5 MPa the accumulators and the HPIS are shut off. This situation is similar to a massive nitrogen injection in a PWR due to uncontrolled discharging of accumulators.

In these cases, only the availability of LPSI can limit core heat up. Nevertheless, – since it takes a long time until the primary pressure drops to the shut-off head of the LPSI pumps – a large amount of the primary inventory is lost before any safety injection (SI) can be started and core heat up occurs.

Three tests were carried out to study the influence of:

- Steam versus nitrogen: the goal is to check whether nitrogen can affect primary to secondary heat transfer and, with that, primary depressurisation by secondary bleed to an extent, which compromise LPSI in time.
- The number of secondary relief valve (RV) used for depressurisation in the presence of nitrogen.

Direct transposition of the test results to the power plant is not possible, the main reason being the single-loop design of the PMK facility. However, it can be expected that the test results showing the effect of nitrogen are pessimistic with respect to plant behaviour. In case of a small cold leg break in the plant the nitrogen of the PRZ would escape only to the SG of the loop connected to the PRZ and, then, most of it would be evacuated by the break. This means that deterioration of heat transfer can be expected in only one of the six SGs that would lead to a response closer to the test with steam atmosphere. The test results are, however, useful in validating the computer codes to be applied for analysis of this type of transient in VVER power plants.

5.2.2.2 Cool down under asymmetric boundary conditions

The background of this test is accident scenarios in a PWR with reactor cool down accomplished with an isolated SG, i.e. under asymmetric conditions in single-phase NC. Such a process could be required, for example, after a feed water or steam line break, as well as for a SGTR, with simultaneous loss of off-site power, in either case.

If the circulation in the loop corresponding to the isolated SG stagnates, then:

- In the case of a SGTR, there is a risk of heterogeneous dilution, since the clear water from secondary side would not mix with the borated primary coolant. Hence, a clear water plug can accumulate in the U-tubes.
- For all other cases, if the primary side is depressurised there is a risk of flashing in the U-tubes of that isolated SG. Depending on the total volume of U-tubes (which is about 25 m^3 per SG for most PWR), this phenomenon can suddenly fill the PRZ.

If the cool-down rate is fast, the NC in the inactive loops can stagnate. Hence, it is important to know whether there is a cool-down gradient permitting to maintain the circulation.

The aim of the first run of the test was to determine whether a cool-down rate of 50 K/h (maintaining an outlet core sub-cooling margin of 10 K) can lead to stagnation in the inactive loop. This stagnation leads to evaporation in the U-tubes of the SG when the primary side is depressurised. A sensitivity study has been also performed, to study the influence of primary pressure (controlled by auxiliary spray): this enables the determination of whether flow stagnation can also occur for a higher core sub-cooling margin.

The goal of the last run has been to determine the maximum cool-down strategy that prevents stagnation in the inactive loops. This is performed by a stepwise increase of the cool down.

As for the use of the test results, the temperature difference between the two SG plena is an indication of the counter-driving force that causes the loop stagnation. Nevertheless, such a measure is generally unavailable on most plants.

In German PWRs the stepwise cool-down procedure under asymmetric conditions has been applied as part of the standard cool-down procedure following SGTR. Nonetheless, the cool down in most PWR operating procedures is realised continuously, and not stepwise; so, it could be interesting in the future to study the occurrence of loop stagnation with a continuous cool down rate.

Finally, one must notice that for some operating procedures (as for instance in the event of a SGTR) a large cool-down rate is imposed by the need to depressurise rapidly in order to eliminate the break flow rate. In such conditions the loop stagnation is unavoidable. Therefore further investigations in order to assess the impact of key parameters as the decay heat, the recovery after the occurrence of NCI have to be envisaged.

5.2.2.3 Fast cool-down transients

Fast cool-down transients can be triggered off by a MSLB: it causes a rapid depressurisation of the affected SG, which intensifies the heat transfer between primary and secondary side, hence leading to a fast cool down of the primary side.

Two safety issues are related to this type of transient:

- Re-criticality issue: the fast cool down causes a pronounced overcooling in the primary side of the affected loop. Depending on the mixing in the DC and the LP with the unaffected loops, cold water can enter the core area: the question of re-criticality therefore has to be addressed.
- PTS: in a second phase of the transient, the introduction of cold water in the RPV DC raises the question of the RPV integrity due to PTS.

These two issues are related to the overall system behaviour, but also to strongly 3D mixing phenomena occurring in the DC and the LP. For this reason, complementary ROCOM tests were performed to investigate mixing in the downcomer, based on conditions taken from the PKL G3.1 transient.

The PKL test is composed of two parts:

- During the first phase of the transient, the affected SG boil-off intensifies the primary/secondary heat transfer, causing a sharp temperature decrease in the affected loop. Hence cold water enters the RPV inlet. After RCPs coast down, a higher NC flow rate establishes in the affected loop; meanwhile the flow rate in the unaffected loops decreases.

Stationary ROCOM tests have been performed for two points in time of the PKL transient (during the affected loop temperature decrease and at the minimum temperature) to study the mixing in the LP and in the downcomer. In addition, one transient test was performed to investigate the influence of changing loop flow rates on the coolant mixing. For both, stationary and transient ROCOM tests, the mixing phenomena are buoyancy-driven and are caused by density differences between the reactor loops. They are studied to address the re-criticality issue and need to be assessed to determine the heterogeneity of the temperature distribution at the core inlet plane including the possibility of the formation of a sector of overcooled water.

- During the second phase of the transient, ECC is injected in two loops (affected and unaffected). The flow rates in the unaffected loops should decrease due to counter-drive forces for NC. Strongly sub-cooled water reaches the RPV through both loops with SI.

  Stationary ROCOM tests have been performed for the injection phase to study temperature stratification in the RCS cold legs. Mixing phenomena in the DC have also been studied with a focus on the behaviour of the cold plumes. These plumes have a strong impact on the RPV DC wall temperature that needs to be studied in detail with respect to the PTS aspect.

The fast cool-down tests performed both on the PKL and ROCOM facilities help in completion of databases for the validation of thermal-hydraulics system codes (validation against heat transfer in the affected SG and influence on loop flow rate), and the validation of CFD codes (buoyancy-driven mixing patterns in the RPV).

5.2.2.4 Influence of secondary-side parameters on heat transfer under RC condition

The aim of the test has been to perform a parametric study of heat transfer between the primary and secondary side during RC operation. The parameters of interest focus on the secondary side: fill level in the SG and cool-down rate.

RC operation can occur for instance in case of a SB-LOCA: when the primary inventory decreases up to the point where NC cannot be maintained, the decay heat is removed by RC (provided the secondary side heat sink is available). It is therefore important to evaluate:

- the SG fill level required for heat removal; and
- the displacement of coolant: during the cool down required by AM procedures, coolant can be displaced towards the SGs and in the RPV from the DC to the core, which could compromise the core cooling, depending on the cool down gradient.

The tests were designed to check the efficiency of RC operation for heat removal, which implies only a small temperature difference between the primary and secondary sides. It also provides data for the CCFL occurrence, and for code and model validation.

5.2.2.5 Boron precipitation processes following LB-LOCA

This test is set in a long-term cooling phase of a large-break LOCA. During this phase (in the case of a cold leg break with cold leg water injection), water boils continuously into the core and, during this evaporation, boron injected with emergency core cooling system (ECCS) can concentrate and reach the solubility limit. Boron crystallisation inside the core would degrade the core cooling. Furthermore, the steam produced in the reactor core only transports a minute quantity of boron and can thus dilute the containment sump. Therefore, it is important to prevent boron crystallisation by switching the ECCS from sole cold leg injection mode to the mixed HL/CL mode (in order to obtain a liquid flow into the reactor core). The switching time depends on the core mixing volume: the more primary coolant is mixed, the more the switch over can be delayed. If the residual heat power is elevated (i.e. the switching time is short), the HL injection may not be sufficient to evacuate power: the switching criterion can hence limit the power upgrading in certain plants.

The first phases of the test G5.1 were a parametric study of the influence of decay heat power, ECCS flow rate and loop pressure drop (which influences the core SL) to generally investigate the mixing volume in the RPV, which may partly include the HLs and the LP.

Another point investigated has been the possibility to uncover the core when DC level is held at the bottom of cold legs, because of a strong pressure drop towards the break (across the broken loop).

Finally, the test aimed at studying the conditions under which the HL injection provides a good flushing of the core and then a reversion of boron concentration. This outcome is important because for some plants the HL injection must evacuate the decay heat power without evaporation, so the switching criterion can limit the power upgrading in certain plants.

5.2.2.6 Formation and behaviour of upper head void during cool down

During a NC cool down, the upper head cool-down rate is slower than the primary cool-down rate and the upper head is independent from the rest of the circuit. Hence, if the primary side is depressurised, a bubble can appear under the vessel head. This phenomenon can play a role for two kinds of transients:

- Transients with fast cool down and which require depressurisation of primary side (for instance, to annul a primary leak, during a SGTR with a secondary RV stuck open). Many procedures use the reactor vessel level indictor system (RVLIS) as an ECCS stop criterion: if the RVLIS measurement is distorted by the presence of a steam void, the transient duration could increase.
- Transients for instance associated with the total loss of the heat sink or the total loss of power supplies, for which the steam bubble must be avoided because it can completely fill the PRZ in case of a depressurisation.

Test G6.1 was proposed to characterise these phenomena. The main goals of this test were:

- To observe the dome cool-down gradient during NC cool down and to determine under which conditions the dome cool down is separated from the primary.
- To analyse the formation of a steam void under the vessel head during a NC primary cool down and more specifically the maximum void growth.
- The possibility to collapse a formed steam void under the vessel head with RCPs.

5.2.2.7 Effectiveness of secondary-side depressurisation 7.1/CET performance 7.1

This test consisted of a SB-LOCA with the total loss of medium head ECC. The main aim of this test was to perform a “counterpart test” to compare its results to an equivalent test performed on the ROSA facility, this topic being vital for code validation and assessment.

In the event of the unavailability of the medium head SI, the operating procedures prescribe:

- To bleed the secondary side, in order to depressurise and reach the pressure of the low head SI. In the test the secondary bleed was performed when the CET reaches 350 °C.
- Depending on the plant, to bleed directly the primary side by opening the pressure-operated relief valves (PORVs) to quickly depressurise the core if core cooling is compromised (core uncovered or superheated).

Effectiveness of secondary-side depressurisation

It is very important to assess the effectiveness of the secondary-side depressurisation on the efficiency of the SI.

CET performance

In previous experimental programmes it was shown that the CET may in some cases not detect superheating, because of the deposition of a water layer or water drops over the TCs. So, it is important to characterise the CET performances.

Yet, in most PWR both superheating and RVLIS are used to initiate secondary-side bleeding.

5.2.3 The scaling value of the data including the connection with expected NPP phenomena

Among the main goals of the parametric studies performed within a framework of the PKL2 project is the aim to provide experimentally measured data for the purpose of:

- validation of thermal-hydraulic computer codes for correct prediction of phenomena occurring in test rigs;
- validation of CFD codes in order to be able to predict the corresponding phenomena for NPP scale and other geometries; and
- development of operational and emergency procedures and guidelines for NPP by utilities and regulatory bodies on the basis of experimental evidences.

Therefore, it is necessary to assess the scaling value of the measured experimental data and the possibility to transfer the main outcomes to the scale of a PWR.

This step has been taken by AREVA NP and the University of Pisa in as far as the following general phenomena occurring in the PKL III G series of experimental tests at PKL facility are concerned: single-phase and two-phase NC, heat transfer to the secondary side under asymmetric variable conditions and in the presence of non-condensable gases (including RC mode), fast cool-down transients and mixing processes and boron transport.

The experimental data corresponding to the heat transfer to the secondary side in the low-pressure range, condensation and transport of condensate under RC conditions, NC, SLs formation and relocation of the CMs inside the primary side and should be very useful for PWRs at least in qualitative terms, because:

- The development of the SL is largely a function of specific power (decay heat per flow cross-section), elevation heads, and the pressure level, and all these parameters are modelled on a scale of 1:1 of a KONVOI PWR.
- The SG tubes are identical to those used in a KONVOI PWR in terms of diameter, height and wall thickness so that the heat transfer mechanisms are transferable. Second-order influences such as heat losses and storage capacity of the primary-side structures delay the attainment of steady-state conditions but have negligible effect on the phenomena observed or the steady-state condition that become established.
- The driving force and counter-driving force for NC depend most importantly on the geodetic elevation and pressure losses. These dimensions are accurately modelled (geodetic elevations in 1:1 KONVOI scale, pressure losses in the primary circuit overall and in the individual components correspond to those in a KONVOI PWR).

Nevertheless some distortions between PKL and the reactor can be caused by the difficulties to perfectly scale-down the SG plena and the SG inlet bends. The two-phase flow distribution in this region can have an influence on the flow pattern in the different U-tubes, during the transition from two-phase NC to RC mode.

Moreover, at the University of Pisa, a comparison has also been made between the PKL NC behaviour and the so-called Natural-Circulation Flow Map created on the basis of experimental data from various PWR simulators [41]. A consistent performance by PKL is confirmed by this study.

Since the mixing volumes in the PKL facility (core, reflector gap, lower and upper plena etc.) are modelled to scale, the data obtained in test G5.1 constitute a scalable basis for expected boron mixing and precipitation processes in a PWR. It should be noted that the results and conclusions from the test are transferrable without major distortions only to the KONVOI design, since the flow resistance across the loops, which is the important parameter determining the mass distribution in the primary side, may notably differ in other PWR designs.

The transferability of phenomenon of void formation in upper head is strongly dependent on correct scaling of RPV bypass flow paths and head losses between upper plenum, RPV dome and downcomer. Additionally, other important parameters contributing to the fluid evaporation/condensation under depressurisation/pressurisation and heat input are the ratios between the available hydraulic volume, the volume of metal structures and the heat exchange surface (heat losses). Since the geometry of the RPV dome and bypasses were reproduced to scale in the PKL facility and the heat losses were replicated very well, the main outcomes from test G6.1 may provide useful information. However, it should be noted that the dynamics of void propagation, especially in the phase of collapsing after the restart of the pumps, may be distorted at the scale of a PWR due to the different geometrical configuration and volume versus height properties. Given this information, further analysis should be made to understand this phenomenology.

It should be also noted that the depressurisation rate of the affected SG during MSLB test may differ at the scale of NPP. This may lead to different minimum coolant temperature and the time of its occurrence in the affected loop of the primary side, which in turn can affect the reactivity effects and re-criticality phenomena at the scale of a PWR. In addition, the geometrical configuration of the DC in the PKL facility notably limits the transferability of measured data for further analysis of PTS. Therefore the issue is addressed by complementary experimental tests in ROCOM facility with boundary conditions provided from test G3.1. Yet, the scaling issue for each of the facilities has to be taken into account in this one-way coupling (i.e. PKL to ROCOM data transfer, without feedback) process. As a consequence, the outcomes from these tests are qualitatively applicable, but tests results should be transferred to PWR scale with caution, even though both PKL and ROCOM test facilities are well-scaled with respect to physical phenomena (as explained hereafter).

The density differences between the different coolants in the facility (reactor) play an important role for the phenomena in these tests. The Froude number is the main dimensionless similarity number, which can be used to characterise buoyant single-phase flows. The Froude number represents the ratio of inertia and gravitational forces. The influence of the inertia in flowing media is characterised by the density, the influence of the gravity by the density difference. That means that the density difference between injected and ambient water 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 experiments 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 $\sqrt{5}$.

There could be a small influence of the Reynolds number on the intensity of the coolant mixing but the main effect on the mixing is the presence of a turbulent flow itself which is ensured for all presented ROCOM experiments. In the past experiments were performed at the ROCOM test facility with assessment of the influence of the Reynolds number on the coolant mixing [42, 43]. It could be shown that with an increase of the Reynolds number the measured maximum mixing scalar (minimum boron concentration) stays within the uncertainty range of the measurement results induced by the statistical (turbulent) fluctuations of the flow field inside the vessel. It should be noted that due to the geometrical scaling it was not possible to increase the Reynolds till the corresponding value present in the original reactor.

5.2.4 The value for code assessment

Experimental results obtained at OECD Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) PKL2 programme extends the previous reference database with a valuable collection of data useful for assessment of current thermal-hydraulic codes and for code development. With a PIRT (Phenomena Identification Ranking Table) analysis in mind, Tables B1, C1 and D1 collect the key phenomena for all the tests performed (namely at PKL, PMK and ROCOM) with relevance for T/H code (system or CFD) verification and validation purposes.

As in previous PKL programmes, PKL2 groups have kept an active involvement and promotion of analytical activities during the execution of the project. These activities have aimed:

- to understand important phenomena/processes;
- to assess and compare the predictive capabilities of existing tools;
- to identify strengths and limitations of the existing tools, and draw conclusions on the possible use of the codes; and
- to develop a common understanding and to promote an exchange of knowledge.

Achievement of these objectives has been conducted through three principal actions:

- presentations from participants at the regular half-yearly Programme Review Group (PRG) meetings (pre- and post-test analyses);
- activities of benchmark exercise; and
- presentations and discussions within specific workshops on analytical activities.

Furthermore, several project partners have presented some of these analyses using their own TH simulation tools, for technical magazines publication and/or for congress presentations.

Benchmark exercise on test G3.1 (MSLB test)

G3.1 was selected by the PRG and Management Board (MB) for performing a common analytical activity among participants. The exercise was divided into two phases, namely “blind” pre-test [43] and post-test phases [45]). The main objective was to collect, analyse and document simulation results obtained by the participants, describing the performances of the code simulations and their capability to reproduce the relevant thermal-hydraulic phenomena observed in the experiment. Additionally, the exercise included a quantitative assessment of code performances based on accuracy evaluation of their results by using an FFT based method (see Appendix I of [44] for further details).

Five TH-system codes (CATHARE2, RELAP5, ATHLET, MARS and TRACE) were used in the exercise, demonstrating the codes’ capacities to deal with the relevant phenomena involved in both phases I (affected SG blowdown) and II (ECCS injection) of the transient. All of them were able to predict the trend of the main parameters during the phase I of the transient, with satisfactory results. In particular, the break flow, the affected SG depressurisation, the heat exchange primary to secondary, the RCS coolant temperatures at U-tubes outlet, and also the primary pressure are in agreement with the experimental results. Some difficulties appeared in capturing PZR variables behaviour during phase II; an imperfect knowledge of PZR PORV operation and/or PZR modelling have been identified as main causes of the discrepancies.

Quantitative indices of the results have been derived by the FFT based method during the benchmark to assess accuracy of the code and model performances. Results showed that all participants have an average accuracy during the phase I (outstanding for some variables), whereas there was a slight reduction in the overall accuracy during the later stages of the transient.

Pisa 2010 Workshop on PKL2 analytical activities

In the frame of the project, one specific workshop on analytical activities, held in University of Pisa in April 2010 [46], was organised. Although the main topic was the PKL G3.1 benchmark exercise, the workshop objectives also covered presentations and discussions on ROCOM and PMK analyses, on analyses of other PKL tests and on plant applications.

As in previous PKL-related projects, the workshop, which was attended by a significant number of participants, and provided an excellent opportunity for an effective and necessary interaction between code users and experimenters for the discussion of topical safety issues. It was a common understanding that the effort on analytical activities should be maintained within the projects and that this type of workshop was a successful step in the right direction. Different areas of interest to be considered as alternatives for future tests were also suggested.

Analysis of ROSA-PKL counterpart test

This co-ordinated joint ROSA and PKL action was a real new challenge for OA experimenters and also for code practitioners, as there has not been many similar exercises within the TH community in the past. Apart of investigating relevant phenomena occurring during AM, the action had also the objective to enrich the experimental database with a test having high value for code assessment and scalability purposes. These test results will be useful to illustrate and enlighten the scaling effects and techniques needed to scale-up to full NPP plants.

Several code practitioners have performed pre-test analyses and closely interacted with experimenters of both OAs in order to help in the design of a scenario to ensure compatibility with construction requirements of the two facilities. At the end a HL SB-LOCA with total failure of the HPSI with manual SG depressurisation (AM action) followed by ACC injection in cold leg has been selected.

In addition to identify and verify relevant phenomenology of a PWR to a SB-LOCA (e.g. loop seal formation and clearing, eventual core uncovery, mass distribution in the RCS, core heat up, reflood, accumulator performance), this test aims to observe the relationship among temperatures in the core and CET, and so help in the discussion on adequacy of CET measurements as indication of core temperatures to trigger AM actuations.

It is also foreseen to perform a suitable comparison of ROSA and PKL, as well as TH code results that will allow confirmation of the expected value of the counterpart tests. Outcomes of this analysis will be helpful to support the involved steps of integral plant model qualification procedures and uncertainty evaluation methodologies.

5.2.5 Specific additional lessons learnt from the execution of experiments

Boron precipitation issue

As has been previously indicated, the conclusions drawn from the PKL2 experiments involving the three forms of NC, cool-down methods and rates, and CET performance reinforce lessons learnt from previous experiments, reveal new insight into plant behaviour, and extend the database for code assessment. In addition, Test G5.1, the boron precipitation test, also provides evidence towards disproving a theory thought to challenge the assumption of a constant mixing volume when calculating boric acid concentrations during the long-term cooling phase following a LB-LOCA.

To ensure core cooling long after a LB-LOCA has occurred, operators are required to take actions to prevent the precipitation of boric acid; the accumulated precipitate can obstruct the flow of water to the core. The time at which such actions are taken, however, are crucial, particularly in US plants, and therefore must be calculated with reasonable conservatism. These calculations require an assumed mixing volume in which the concentration of boric acid can build, with the rate at which the concentration builds dependent on the size of the assumed volume. Generally, the volume is assumed to be of constant size and to consist of the core region, portions of the LP, and the upper plenum region below the top elevation of the HLs. For many years, these assumptions have been considered acceptable for LB-LOCA scenarios, but in recent years, analytical studies have shown that the assumption of a constant mixing volume may not be appropriate when the break is located on the top of the discharge leg, i.e. a large split-break. In theory, when a break is in this location, the loop seals will refill with liquid after the core has been reflooded and the steaming rate in the core has dropped during the long term. The filled loop seals represent a large resistance (in addition to the loop friction and geometric losses) to steam flowing through the external loop piping to the break. The blocked loop seals will cause the pressure to increase in the upper plenum. This pressure increase will ultimately lead to the depression of the two-phase level in the vessel, pushing the two-phase level downward towards the top elevation of the core, and thus, reducing the size of the mixing volume. With a smaller mixing volume, the rate of boric acid build-up in the core will be accelerated relative to cold leg breaks on the bottom of the pipe. The boric acid concentration in the core will increase more rapidly, decreasing the time to precipitation and the time available for the operators to take preventive measures.

One of the objectives in performing Test G5.1 was to evaluate the likelihood of occurrence of this upwardly-oriented, large-split-break scenario; and after taking extraordinary measures to facilitate its manifestation, the test results showed that the scenario, as theorised, could not be replicated with the current PKL configuration. Even after directly injecting very low ECCS flow into the loop seals and blocking all vessel bypass flow paths, the flow resistance across the loops was not large enough to block the loop seals and cause a core depression. The liquid level in the vessel consistently exceeded the core region and progressed towards the steam generators. These outcomes stand as evidence against the possibility of a decreasing mixing volume in long-term cooling calculations for an upwardly-oriented LB-LOCA scenario.

Impact on procedures related with asymmetric cool-down conditions:

In most PWR accident procedures, the cool-down phase is realised continuously through a fixed gradient. Under asymmetric cool-down conditions (following a feed water or MSLB as well as for a SGTR, with simultaneous loss of off-site power) the application of the cool-down procedure may lead to flow stagnation in the affected loop. Test G2.1 proved that a stepwise cool down (or a reduced mean cooling rate) process maintains NC in all loops provided that the magnitudes of the cool-down steps and cool-down gradients are adequately controlled.

However this stepwise cool-down procedure cannot be directly transposed to most PWRs. So, it could be interesting in the future to look for strategies that allow avoiding NCI and recovery if NCI occurs. Moreover, the impact of some key parameters (such as the decay heat level, the mixing in the downcomer) on the proposed cool-down strategy remain to be investigated.

Upper head void formation

During a natural-circulation cool down, the magnitude of the “sweeping flow” is strongly related to the flow paths geometry (through the control rod drop accident [CRDA] then the dome spray nozzles): hence, it depends on the reactor geometry. Nevertheless, the sweeping flow can be ineffective to cool down the vessel head. The vessel head cool down ratio then depends on the vessel head thermal losses.

The bubble formation, once the saturation conditions are attained, is fast (bulk boiling of the whole dome mass). Once the bubble is formed, the steam void can be superheated by the dome structures, which are hotter because of their thermal inertia.

During the natural-circulation cool down, the sub-cooling span at the core outlet is also present, in the PKL test, 0.5 meters above the top of the HL (by convection/conduction through structure phenomena). Therefore the steam void volume is limited at 0.5 meters above the top of the leg. This elevation may depend on the sub-cooling span and the upper plenum geometry; however it is expected, that the steam void does not enter the HLs.

The bubble collapsing is slow and mainly driven by the dome thermal losses: the formation of a saturated layer between the superheated steam and the sub-cooled water avoids any interface condensation. This saturated layer is relatively thin (10 cm thick in the PKL test).

At the very end of the test, two RCP were restarted. The re-establishment of a spray nozzles flow accelerates the bubble condensation: as long as the spray nozzles jet is in direct contact with steam, the condensation is fast. Once the spray nozzles jet is flooded (the spray nozzles in PKL are horizontal), the condensation is slower.

Re-criticality issue during the short term phase of SLB transient

The SLB initiates a rapid boil-off of the affected SG, which causes an increase in the sub-cooling margin at the RPV inlet nozzle in the associated loop. The inflow of unmixed (hence cold) water from the affected loop in a section of the core could lead to re-criticality. Mixing in the RPV was studied through several ROCOM tests, considering different boundary conditions representative of the PKL G3.1 SLB transient with RCPs off. Tests results show that different mixing mechanisms occur in the RPV according to flow conditions (velocity and density differences): sectorisation in the DC with strong mixing in the LP, or complete mixing in the DC below a certain limit depending on flow conditions.

The results of the experiments in the ROCOM test facility show that even small density differences induce mixing inside the RPV resulting in a more homogeneous temperature distribution at the core inlet under the investigated boundary conditions.

6. CONCLUSIONS

Within the Primärkreislauf (PKL2) test series, eight pressurised water reactors (PWR) integral tests (comprising a total of twelve test runs) were carried out from April 2008 to December 2011.

Due to the broad spectrum of topics, the Conclusions Chapter presents a short review of the results with respect to the fulfilment of the test objectives.

G1 – Loss of residual heat removal during ¾-loop operation

Previous tests on loss of residual heat removal (F2 series) revealed the essential thermal-hydraulic phenomena to be very strongly influenced by the primary-side water inventory in the region of the steam generator (SG) tubing; the G1 parameter studies now proved the perceptions of former tests on one hand and provided a systematic analysis of the complex thermal-hydraulic processes that occur in the SGs in the presence of nitrogen, steam and water on the other hand. They analysed the transition between the different flow patterns and the boundary conditions leading to different flow patterns and helped in finding conclusions on maintenance of core cooling and occurrence of boron dilution. The recalculation of test data represents a challenging task for the thermal hydraulics (T/H) system codes and may therefore be employed for code validation or model development.

G2.1 – Cool down under natural-circulation condition with isolated SGs

Basis for the conceptual design of Test G2.1 was the explicit demand for a data basis for the validation of T/H system codes against T/H phenomena associated with cool down under natural circulation (NC) condition in presence of isolated SGs. G2.1 provided important information on the behaviour of NC in presence of isolated SGs and pointed out options for cool-down procedures with preservation of natural circulation (stepwise cool-down procedure)

The general goal of the following analyses with qualified computer codes relates to the operational aspect of design of cool-down strategies under abovementioned scenarios with maintenance of NC.

The scope of work was fulfilled and possible continuation of investigations on the topic may include isolated but filled SG secondary side (in opposite to emptied SGs employed in Test G2.1, see Chapter 7 Outlook).

G3.1 – Fast cool-down transients following MSLB

The PKL G3.1 test results on the integral reactor cooling system (RCS) behaviour – extended by the perceptions on separate effects derived from the complementary Rossendorf Coolant Mixing (ROCOM) tests – in total provided a data basis that covers all relevant aspects of the scenario, suitable for the validation of both, T/H system codes (heat transport to secondary side, influence on loop flow rates) and computational fluid dynamics (CFD) codes and calculation models (buoyancy-driven mixing patterns in the reactor pressure vessel – RPV).

G4.1 – Parameter studies on reflux-condenser (RC) condition (influence of secondary side)

Test PKL III G4.1 was designed as a parameter study focused solely on separate effects to support the understanding of physical phenomena related to heat transfer between primary and secondary side under RC conditions and as such provided accurate boundary conditions for heat transport mechanisms as a basis for code validation and model improvement.

Cool down via secondary side was demonstrated to be a very efficient and safe way for primary-side pressure reduction, even (and especially) for cool down under RC condition on the primary side at coolant inventories around 40%, e.g. as a result of reduced availability of safety systems or cool down being initiated at a late point in time, and under considerably increased cool-down gradients of up to 450 K/h. Moreover, the effectiveness of heat transport to secondary side was verified to be preserved even for very low secondary side fill levels (<< 1 m).

Coolant dislocation from RPV to SG due to counter-current flow limitation (CCFL) was demonstrated to be of no issue for the preservation of the core cooling under the given boundary conditions.

G5.1 – Parameter study on Boron precipitation in the core following loss-of-coolant accidents (LB-LOCA)

Test G5.1 revealed the transient behaviour of a boron precipitation in the course of a LB-LOCA for different scenario-relevant parameter settings. Due to the presence of mixing mechanisms in the RPV, the grace period until switching criteria for a core flushing (by HL injection) are met was demonstrated to be of an order to allow the switchover to HL injection at a considerably later point in time at a significantly decreased decay power level.

In advance to G5.1 test specification swell-level depression in the RPV as a result of a pressure built-up in the UP was expected to considerably reduce the mixing volumes in the core and to cause adverse conditions for the chronological sequence of the boron concentration in the core. The current PKL loop configuration (representing German KONVOI configuration) only features small loop flow resistances insufficient to generate considerable swell-level depression. By means of test PKL III G5.1 the effects of a pronounced swell-level depression could not be verified.

A possible continuation of the investigations around boron precipitation may focus exclusively on swell-level depression, but requires modifications at the PKL test rig (see Chapter 7 Outlook).

G6.1 – Behaviour of the upper head void during cool down under NC condition

The PKL Test G6.1 allowed to observe the principles of limitation of void growth in the upper head and UP and supplied a detailed recording of the relevant physical background: Temperature distribution in the UP and RPV dome, intensity and effects of the sweeping flow via the closure head bypass persisting under NC-conditions and condensation of the upper head void under NC and forced full load circulation.

The test provided an important data source for the validation of existing computer codes or improvement of calculation models.

G7.1 – Beyond-design-basis SB-LOCA with additional safety system failures (counterpart testing with ROSA/LSTF)

By means of counterpart testing in PKL and ROSA/LSTF in the frame of Test G7.1 the high efficiency of a secondary-side bleed procedure employed for the reduction of the primary-side pressure depressurisation was demonstrated.

The consistent results for the relevant main phenomena in both test facilities account for a profound and valuable data basis for the evaluation of core exit temperature (CET) performance in accident management (AM) and especially for analysing scaling effects.

PMK Test 1 – Disturbance of NC during the lowering of the primary system level

The test provided information about the degradation of SG heat transfer and the disturbance of NC during the lowering of the primary system level in preparation for core unloading when the upper parts of primary (upper head and pressuriser – PRZ) are filled with air. The different runs of the test allowed a study of how the amount of steam produced in the core before replenishment and the status of the valve in the PRZ (open or closed) during refill affected NC in the system and heat transfer to the secondary side.

Although the test results are not directly transposable to VVER plants, it can be stated that in case of such an incident the operator has to close all venting lines before starting the refill process, because this will help to resume NC. The test produced challenging results for code validation.

PMK Test 2 – Small break LOCA during cool down in a VVER

The tests addressed the effectiveness of secondary side bleed with steam and nitrogen atmosphere in the SG and they demonstrated that core heat up cannot be avoided, but it is limited by low-pressure safety injection (LPSI).

The test results are useful in validating the computer codes to be applied for analysis of this type of transient in VVER power plants.

Taken together, the PKL2 project not only provided comprehensive and exclusive data bases for thermal-hydraulic system or CFD codes and review of operational procedures but vitalised relations within the international community of reactor safety related T/H and made significant contributions to maintain and develop its international expertise.

7. OUTLOOK

Following the Fukushima Daiichi accident in March 2011 it became obvious that beyond-design-basis or severe accident course of events needed further attention; analyses should reveal the capabilities of existing pressurised water reactor (PWR) to control a severe accident. The ensemble of former Primärkreislauf (PKL) tests on beyond-design-basis events already pointed out significant safety margins for the relevant events; so it may be of interest to impose further aggravated (worst-case) boundary conditions and perform the investigations on beyond-design-basis scenarios (e.g. station blackout (SBO) scenarios featuring lately employed accident management-measures) under perpetuation of the G7.1 main test goals: effectiveness of employed measures and plausibility of relevant measuring signals.

A logical continuation of the investigations on events and phenomena associated with events under cold shutdown state (i.e. failure of residual heat removal system – RHRS) – also with respect to current safety issues – may be the application of relevant principles, phenomena and procedures known to this day (F2 and G1 test series) to scenarios with open RCS.

Several other topics of the PKL2 (PKL III G) test programme may be subjects of continued investigation:

- **G2.1: Cool down under natural circulation (NC)-conditions with isolated steam generator (SGs).** A complementary test for a further **extension** of the already existing data base may be arranged under the premise to have **isolated but filled SG secondaries.** Based on the perceptions of Test G2.1 (preservation of NC through stepwise cool-down procedure) the deduction of a **maximum continuous** cool-down gradient may be the outcome of the possible new test.
- **G5.1: Boron precipitation in the core following leg break loss-of-coolant accidents (LB-LOCA).** A modification of the PKL test rig to impose considerably higher loop flow resistances may be employed to realise a parameter study on the dependence of swell-level depression on loop flow resistances and its consequences for the mixing volumes and progress of the boron enrichment in the core.

The workshop in Pisa furthermore suggested several other field of interest relating to identification of specific experimental needs connected with current plants (power uprates, core exit temperature performance, new SG replacements), or the obtainment of qualified data for the occurrence of counter-current flow limitation (CCFL) in the hot legs.

It was commonly accepted that the collaboration on the international scale proved to be a very effective way to preserve test facilities and know-how on one hand, and to solve PWR safety issues in close collaboration between OA of test facilities and thermal hydraulics (T/H) code users on the other hand.

A further intensification of co-operation of individual test facilities by means of counterpart and complementary testing may be of use for the deduction of scaling effects or conclusions on safety relevant issues for different plant designs and geometries.

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