Nuclear Safety
NEA/CSNI/R(2022)6
June 2024
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NEA PKL-4 Project
Summary Report

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NEA
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Nuclear Energy Agency

NEA/CSNI/R(2022)6

Unclassified **English text only**

11 June 2024

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA PKL-4 Project

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JT03545832

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The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes;
- to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management and decommissioning, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS (CSNI)

The Committee on the Safety of Nuclear Installations (CSNI) addresses Nuclear Energy Agency (NEA) programmes and activities that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations.

The Committee constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development and engineering, to its activities. It has regard to the exchange of information between member countries and safety R&D programmes of various sizes in order to keep all member countries involved in and abreast of developments in technical safety matters.

The Committee reviews the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensures that operating experience is appropriately accounted for in its activities. It initiates and conducts programmes identified by these reviews and assessments in order to confirm safety, overcome discrepancies, develop improvements and reach consensus on technical issues of common interest. It promotes the co-ordination of work in different member countries that serve to maintain and enhance competence in nuclear safety matters, including the establishment of joint undertakings (e.g. joint research and data projects), and assists in the feedback of the results to participating organisations. The Committee ensures that valuable end-products of the technical reviews and analyses are provided to members in a timely manner, and made publicly available when appropriate, to support broader nuclear safety.

The Committee focuses primarily on the safety aspects of existing power reactors, other nuclear installations and new power reactors; it also considers the safety implications of scientific and technical developments of future reactor technologies and designs. Further, the scope for the Committee includes human and organisational research activities and technical developments that affect nuclear safety.

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List of abbreviations and acronyms

ACC	Accumulator
AFWS	Auxiliary feed water (provided by mobile or steam-driven [turbo-] pumps)
AM	Accident management
CCFL	Counter current flow limitation
CEA	Commissariat à l'énergie atomique et aux énergies alternatives (Alternative Energies and Atomic Energy Commission, France)
CET	Core exit temperature
CL	Cold leg
CNPRI	China Nuclear Power Technology Research Institute Co. Ltd
CRGA	Control rod guide assembly
CSN	Consejo de Seguridad Nuclear (Nuclear Security Council, Spain)
CSNI	Committee on the Safety of Nuclear Installations (NEA)
DBC	Design-basis condition
DC	Downcomer
DEC	Design-extension-condition also beyond-design basis
DIA	Data interpretation and application
ECC	Emergency core cooling
ECCI	Emergency core coolant injection
EDF	Électricité de France
ELAP	Extended loss of AC power
EPR	European pressurised water reactor
HP	High pressure
HPSI	High pressure safety injection
IB	Intermediate break
IB-LOCA	Intermediate-break loss-of-coolant accident
IRSN	Institut de Radioprotection et de Sûreté Nucléaire (Institute for Radiation Protection and Nuclear Safety, France)
KAERI	Korea Atomic Energy Research Institute
LB	Large break
LHSI	Low-head safety injection
LOCA	Loss-of-coolant accident
LOOP	Loss of offsite power
LPSI	Low pressure safety injection

LS	Loop seal
LSC	Loop seal clearing
LUT	Lappeenranta-Lahti University of Technology (Finland)
MB	Management Board
MCP	Main coolant pump
NC	Natural circulation
NEA	Nuclear Energy Agency
NPIC	Nuclear Power Institute of China
NRA	Nuclear Regulation Authority (Japan)
OECD	Organisation for Economic Co-operation and Development
PCT	Peak (fuel-rod) cladding temperature
PKL	Test facility (German acronym for “Primärkreislauf”, i.e. primary circuit)
PPPT	Passive pressure pulse transmitter
PRG	Programme Review Group
PRZ	Pressuriser
PSI	Paul Scherrer Institut (Switzerland)
PWR	Pressurised water reactor
RC	Reflux-condenser
RCCA	Rod cluster control assemblies
RCS	Reactor cooling system
RHRS	Residual heat removal system
RPV	Reactor pressure vessel
SACO	Safety condenser
SB	Small break
SB-LOCA	Small-break loss-of-coolant accident
SG	Steam generator
SNPSDC	State Nuclear Power Software Development Center (China)
SSD	Secondary-side depressurisation
SSM	Strålsäkerhetmyndigheten (Radiation Safety Authority, Sweden)
STUK	Säteilyturvakeskus (Radiation and Nuclear Safety Authority, Finland)
T/H	Thermal-hydraulics
UCP	Upper core plate
UH	Upper head
UP	Upper plenum
UPTF	Upper plenum test facility

USNRC	United States Nuclear Regulatory Commission
USP	Upper support plate
VTT	Teknologian tutkimuskeskus (VTT Technical Research Centre of Finland)
VVER	Water-cooled water moderator energy reactor (Russian design)

Executive summary

In continuation of the NEA-guided experimental programmes at the PKL test facility in Germany operated by Framatome GmbH, the fourth phase of the NEA PKL project was conducted between June 2016 and September 2020. The topics of the experimental investigation had been defined in collaboration with the project partners and were considered of high interest. On certain topics, the PKL experiments were complemented by experiments at the PWR PACTEL test facility (operated by Lappeenranta-Lahti University of Technology, LUT, Finland) and the PMK-2 test facility (operated by Centre for Energy Research, Hungarian Academy of Sciences, Hungary). A brief introduction to the test topics, the key safety objectives addressed and the conclusions drawn from the results are given in the following sections.

Test i1: Parametric studies on two-phase flow phenomena related to LB-LOCA

The objective of test i1 on LB-LOCA (double-ended guillotine break in cold leg) was to investigate two-phase phenomena in the RPV during the refilling and flooding phase that determine the re-establishment of core cooling (e.g. quench front propagation in the core) and during the long-term cooling phase by means of integral testing, i.e. taking into account the effects introduced by SGs (acting as heat sources, contributing to steam binding) and the main coolant pumps (MCP, acting as main contributors to the overall loop pressure losses). For the first run addressing phenomena in the refilling and flooding phase, repeated quenching procedures were conducted from an already heated core (≈ 400°C peak cladding temperature) to constitute a parameter study. The test matrix underlying this study was set up to evaluate the impact of changes in key parameters and boundary conditions on the course of events and to identify significant contributors. A main conclusion was that the operating status of the MCP (running or standstill) contributes significantly to the outcome of the test with respect to quenching of the heated core and the re-establishment of core cooling. The objective of the second test run (set in the long-term cooling phase) was to capture the swell level position in the RPV and the inventory mass for steady-state points at different pressure and power levels in order to support the validation of calculation models for properties of two-phase flow (interphase momentum exchange).

Tests i2: DEC SB-LOCA and IB-LOCA

Test i2.1 addressed a 1% hot-leg break (SB-LOCA) with the additional failure of the HPSI (DEC). Secondary-side depressurisation was implemented as a preventive AM-procedure; cold-side accumulators (ACCs) and a low pressure safety injection (LPSI) were assumed available. Phenomena of particular interest included heat transfer in the steam generator (SG) in presence of nitrogen (from ACCs not being cut off) and core quenching due to a swell level increase as a result of primary side pressure reduction and loop seal clearing initiated by ACC injection. The inflow of nitrogen from still open ACCs and its accumulation in the SGs led to the deterioration of heat transfer and diminished the pressure reduction on the primary side. This, in turn, indisposed triggering of the LPSI. These phenomena are relevant for the outcome of the transient and could be replicated as observed in the reference test for counterpart testing (ROSA/LSTF SB-HL-12).

For 13%, 17% and larger cold-leg IB-LOCA cases, dry-out as well as the earliest and most intense core heat-up is expected to occur around the point of flow stagnation in the core that develops from dynamic two-phase flow phenomena in the RPV and loop. Experiments confirmed the theory that the position of flow stagnation in the core is mostly predetermined by break size and composition and the relation of pressure differentials along the hot (UP, SG, cross-over leg, MCP) or cold path (RPV, downcomer, CL) to the break location. It was concluded that the absolute pressure level observed for these phenomena (flow stagnation and quenching of the core) in PKL and in the reference experiments at ROSA/LSTF (IB-CL-03 and IB-CL-05) has only a minor influence on the outcome compared with the composition of the loop pressure losses.

For both SB- and IB-LOCA cases, loop seal clearing (LSC) – albeit induced by different root causes – had a dominant influence on core quenching.

As regards the intrusion of nitrogen into the reactor coolant system (RCS) after the ACC depletion and its impact on cooldown, PKL i2-tests showed that the size and position of the break (hot or cold side) are decisive for the distribution and resultant effect of the N_2 on the outcome of the transients. In this regard, PWR PACTEL experiments on the impact of nitrogen on the primary side on cooldown during SB-LOCA course of events complemented the PKL experiments.

PMK2-tests on DEC IB-LOCA for 17% cold- and hot-leg break cases exemplarily showed the course of events for VVER 440-type PWRs. Together with the PKL tests and PWR PACTEL experiments on SB-LOCA, they contributed to the collective database on SB/IB-LOCA transients provided by the NEA PKL-4 project.

Test i3: Test on the failure of the RHRS from cold shutdown condition

PKL III i3 revisited the scenario of a failure in the residual heat removal system for a 3-loop plant. The main objectives of the i3.1 test comprised operational aspects (e.g. progression of temperatures at the RHRS suction positions in the hot legs) as well as safety aspects (inherent boron dilution). The impact of the secondary-side pressure (2 bar vs. 5 bar) on relevant phenomena was investigated by conducting two separate runs. The i3.1 test confirmed conclusions applied from older tests, most notably that: a quasi-steady-state condition with assured heat removal to the secondary side always becomes established, even if only one SG is operable and in the absence of operator actions – regardless of secondary-side pressure level; the occurrence of boron dilution can be avoided by having more than one SG ready for operation.

Tests i4: Investigations on cooldown procedures (DBC and DEC)

Within the PKL III i-experimental programme, the PKL facility was modified to better represent the T/H design of RPVs of modern PWRs (e.g. EPRs); this modification encompassed a change of the RPV vessel lid, UP internals, embedding of the upper support plate (USP) and a new upper core plate (UCP), and installation of replications of CRGA and RCCA.

The background to test i4.1 was a procedure for operational cooldown in LOOP condition. The results of test i4.1 and their comparison against the NEA PKL-2 G6.1 test illustrated differences in the thermal-hydraulic behaviours of the different UH/UP configurations, in particular the process of void formation in the UH and further propagation of the steam bubble through the CRGAs, leading to formation of a second void – in the UP, below the USP.

The DEC scenario for test i4.2 was an extended loss of AC power (ELAP), in which an alternative preventive AM-procedure based on partial depressurisation of a SG with subsequent feed from auxiliary feed water system (AFWS) was tested. AFWS signalling was generated by a passive pressure pulse transmitter (PPPT). The results of test i4.2 demonstrated that an early partial cooldown of one SG in presence of continuous NC on the primary side was sufficient for a secured heat removal and cooldown; a sustained and sufficiently high SG secondary water level was assured by reliable activation and deactivation of the AFWS by the PPPT.

Test i5: Boron precipitation following LB-LOCA

Boron precipitation occurs in the long-term cooling phase following the 1-A cold-leg break (LB-LOCA), a break size large enough to avoid subcooling of the core by cold-leg safety injection (due to ECC loss through the break). Consequently, continuous boiling in the reactor core is expected during the long-term phase. The most decisive parameters for the speed of the boron enrichment process are the core power (residual heat) and the mixing volume i.e. amount of liquid water in the RPV riser. The conclusion was reached that the crystallisation of boric acid is not expected for the analysed scenario within several hours of operation even under extremely favourable conditions for boron content increase (e.g. closing of the reflector gap bypass, minimising the mixing volume). Furthermore, the difference between the density of the water/boric acid mixture and the density of the subcooled fluid in the lower plenum dictates the involvement of the lower plenum volume in the mixing process. The reflector gap re-circulation flow plays only a smaller role.

Test i6.1 - Investigations into multiple steam generator tube ruptures

The background scenario for test i6.1 involved multiple U-tube ruptures in three out of four steam generators as a result of high seismic loads. The objectives of test i6.1 include investigation of break flows between the primary and secondary sides and corresponding pressures courses as well as boron dilution on the primary side, but the key safety issue is the effectiveness of preventive AM-procedures to reduce primary side pressure and temperature. An experimental analysis of multiple SG tube ruptures and corresponding AM-procedures performed in test i6.1 indicated that the secondary-side depressurisation of intact SGs is an effective measure leading to a reversion of the break flow to defective SGs, and thus to an increase of inventory in the primary side and a reduction of activity release to the secondary side. If primary side bleed is required for further pressure reduction, feed water supply into a defective SG counteracts the loss of primary side inventory, successfully reduces the coolant depletion and assures continued core cooling – but with a loss of control of the boron concentration.

The next NEA-guided project at the PKL test facility (ETHARINUS) will address the open questions from the NEA PKL-4 project and will also encompass the assessment of the performance of innovative passive (decay heat removal) systems.

This report was approved by the NEA Committee on the Safety of Nuclear Installations on 12 August 2021 (see Summary Record of the Seventieth (70^{th}) Meeting of the Committee on the Safety of Nuclear Installations - NEA/SEN/SIN(2021)2/REV).

1. Background

For many years, extensive experimental investigations into the system response of pressurised water reactors (PWRs) under accident conditions have been conducted at the large-scale PKL integral test facility operated by Framatome GmbH. The facility constitutes a reduced-scale model of the entire reactor coolant system and major parts of the secondary side of a PWR.

Since 2001, the NEA SETH and subsequent PKL, PKL-2 and PKL-3 projects addressed open questions on current topics in the field of PWR safety for various scenarios with a broad variation of primary- and secondary-side boundary conditions.

The NEA PKL-4 project, which comprised the PKL III i-experimental programme, was conducted between June 2016 and September 2020.

The programme, presented in brief in this report, considered the needs of project partners from different countries and organisations and is thus considered to reflect topics of high safety relevance:

- Test i1: Two-phase flow phenomena related to LB-LOCA, investigated through parametric studies
- Tests i2: Design-extension conditions SB-LOCA and IB-LOCA (counterpart testing with PKL III H.1 and ROSA/LSTF)
- Test i3: Failure of RHRS from cold shutdown condition
- Tests i4: Cooldown procedures under cooldown in natural circulation operation mode for design-basis conditions (DBC) and design-extension conditions (DEC)
- Test i5: Boron precipitation following LB-LOCA
- Test i6: Multiple steam generator tube ruptures

In addition, the PKL experiments on the listed topics were complemented by experiments in the PWR PACTEL test facility (operated by Lappeenranta-Lahti University of Technology, LUT, Finland) and the PMK-2 test facility (operated by the Centre for Energy Research, Hungarian Academy of Sciences, Hungary).

The number of experiments conducted for each topic can be found in the NEA PKL-4 TEST MATRIX at the end of this report.

2. NEA PKL-4 project partners

The NEA PKL-4 project was carried out with the contribution of the following international participants:

- GdF Suez – Tractebel S.A., Div. Tractebel Engineering jointly with the Bel-V, Belgium
- State Nuclear Power Software Development Center (SNPSDC), the China Nuclear Power Technology Research Institute Co. Ltd. (CNPRI), and the Nuclear Power Institute of China (NPIC), China
- Ustav jademeho vzykumu (UJV) Řež a.s., Czechia
- Lappeenranta-Lahti University of Technology LUT jointly with the Teknologian tutkimuskeskus (VTT) and Säteilyturvakeskus (STUK), Finland
- Commissariat à l'énergie atomique et aux énergies alternatives (CEA), jointly with the Institut de Radioprotection et de Sûreté Nucléaire (IRSN), France
- Électricité de France (EDF), France
- Framatome GmbH and the Gesellschaft für Anlagen- und Reaktorsicherheit, Germany
- MTA Centre for Energy Research (MTA-EK) jointly with the Paks Nuclear Power Plant, Hungary
- Paks Nuclear Power Plant, Hungary
- Nuclear Regulation Authority (NRA), Japan
- Korea Atomic Energy Research Institute (KAERI), Korea
- State Atomic Energy Corporation "Rosatom", Russia jointly with OKB "GIDROPRESS", Russia
- Consejo de Seguridad Nuclear (CSN), Spain
- Ringhals AB (RAB), Sweden
- Strålsäkerhetmyndigheten (SSM), Sweden
- Paul Scherrer Institut (PSI), Switzerland
- United States Nuclear Regulatory Commission (USNRC), United States

3. Objectives

The general objective of integral testing 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 plants. In addition, the experimental results aim to validate and further develop thermal-hydraulic computer codes, so-called system codes.

Within the NEA PKL-4 project, the PKL facility was modified two times. The first modification included the installation of nozzle bypass lines at the very beginning of the project. These lines enable a direct flow between the RPV downcomer vessel and the hot legs, which exists also in a full-scale PWR. The second modification of the PKL facility included an exchange of the RPV upper head (UH) vessel and the upper plenum (UP) internals. Due to the installation of the upper support plate (USP) between the UP and UH, the PKL facility can now also replicate PWRs of different designs. In particular, phenomena in EPR-type UH/UP can be simulated in the PKL facility. Since the KONVOI-type configuration can be installed back in the PKL facility, the replication of both designs is possible nowadays.

The NEA PKL-4 experimental programme comprised 8 integral experiments at the PKL test facility with a total of 14 test runs. Additional tests in the PWR PACTEL (LUT) and the PMK-2 (MTA-EK, Hungary) test facilities complemented the PKL experiments. The contribution of the other test facilities comprised a consideration of scenario-relevant specifics of different prototype PWRs (EPR and VVER) and an extension of the scope of investigation to different aspects of a certain scenario.

The NEA PKL-4 project was dedicated to investigating safety issues relevant to current PWR plants as well as to new PWR design concepts by means of systematic parameter studies on thermal-hydraulic phenomena and transient tests under postulated accident scenarios. The investigated subjects (i1 to i6) had been arranged according to the two focus areas implemented in the NEA PKL-4 project:

- parametric studies on thermal-hydraulic procedures for model development and the validation of system codes;
- experimental verification of cooldown procedures and operation modes for different incidents and accidents.

The first category addressed test subjects related to current safety issues that had either suffered from the lack of a dedicated database for analysis and validation of computer codes or from uncertainties in the safety evaluation stemming from open issues or questions. The extension to already existing databases related to these subjects was the foremost goal of this first category of experiments.

The second category of tests mostly contained transient tests either on test subjects already investigated in the former NEA PKL projects as answers to questions that could not yet be completed or on subjects which represented current topics addressed internationally on PWR safety.

The following sections summarise the objectives of the experiments conducted within the NEA PKL-4 project, with a focus on the subjects listed above.

4. Test i1: Parametric studies on two-phase flow phenomena related to LB-LOCA

For the PKL III i1 test series, one test with two separate runs was performed. The test aimed at investigating two-phase phenomena related to a large-break loss-of-coolant accident (LB-LOCA) in the cold leg. As break size, a double-ended guillotine break (2A) was used.

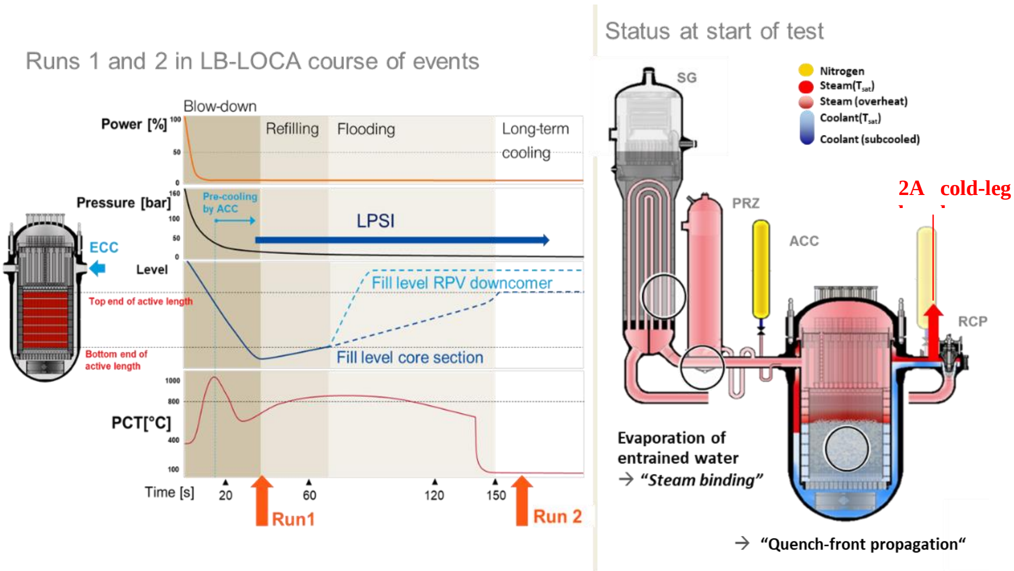
An LB-LOCA transient on the cold side can be divided into different phases. With the occurrence of the break, the blow-down phase starts, which is characterised by a huge inventory loss and thus also a large pressure drop on the primary side. Within a few seconds (30-40 s), the primary side becomes almost completely empty (water column below heated length in the RPV, the rest of the primary side is filled with steam) despite the HP and ACC-injections that have started in the meantime, and the pressure drops below the threshold for the low pressure safety injection (LPSI).

With the LPSI, a turnaround for the inventory is reached and the RPV gets refilled. The period until the active length of the meanwhile overheated fuel rods is reached is called the refilling phase. After the core quenching, the long-term cooling phase starts. In this phase, the water level in the RPV is almost constant and the LPSI ensures the cooling.

Both runs of test i1.1 are parametric studies, and each run consists of several tests (or sequences) with only one boundary condition changed between the sequences.

All sequences of run 1 started with the refilling phase, in which the primary side was almost completely empty and the LPSI started to fill the RPV and ended with the complete quenching of the core. Compared to a reference sequence (p_{prim} = 1 bar, p_{sec} = 50 bar, constant power profile, cold-leg LPSI into all 4 loops, same pressure losses as for MCPs in operation), one boundary condition, such as primary pressure, secondary pressure, core power profile, injection rate and MCP resistance, was changed in each sequence. By varying these individual parameters, the influence on the quench front propagation, entrainment of water into the still hot SGs and the pressure build-up due to the hot SGs (steam binding) was observed.

Figure 4.1. Test i1.1 - background scenario and key objectives



Source: Framatome GmbH, 2020.

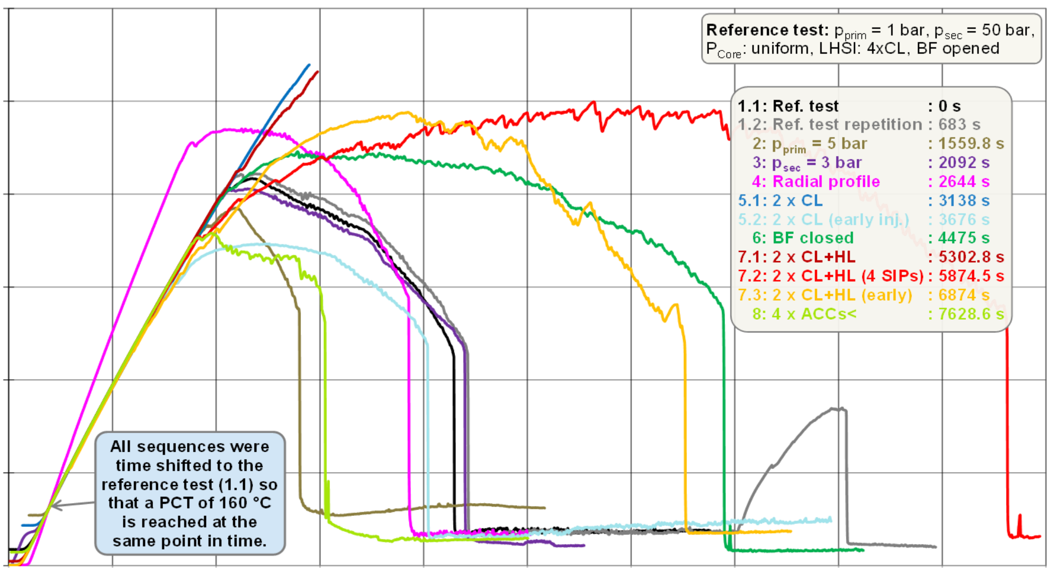
Run 2 also consisted of several sequences, in which the primary pressure and the core power were changed. The sequences of run 2 started after the end of the sequences of run 1, i.e. at the transition to the long-term cooling phase. The starting conditions for run 2 were a completely quenched core with still ongoing LPSI. The injection rate was adjusted between each sequence to compensate the break flow. The objective of the second test run (set in the long-term cooling phase) was to capture the swell level position in the RPV and the inventory mass for steady-state points at different pressure and power levels to support the validation of calculation models for properties of two-phase flow (interphase momentum exchange).

Figure 4.2. Test i1.1, run 1 test matrix

Sequence	1.1 (ref. test)	1.2 (ref. test rep-etition)	2	3	4	5.1 (1st test)	5.2	6	7.1	7.2	7.3	8
p_{prim}	1 bar	1 bar	5 bar	1 bar	1 bar	3 bar	1 bar	1 bar	1 bar	1 bar	1 bar	1 bar
p_{sec}	50 bar	50 bar	50 bar	3 bar	50 bar	50 bar	50 bar	50 bar	50 bar	50 bar	50 bar	50 bar
Core profile	no profile	no profile	no profile	no profile	radial profile	no profile	no profile	no profile	no profile	no profile	no profile	no profile
PCT at start ECC-Injection	400 °C	400 °C	400 °C	400 °C	400 °C	400 °C	200 °C	400 °C	440 °C	400 °C	220 °C	400 °C
LHS-injection	4 x CL	4 x CL	4 x CL	4 x CL	4 x CL	2 x CL	2 x CL	4 x CL	2 x CL+HL (2 SIP)	2 x CL+HL (4 SIP)	2 x CL+HL (2 SIP)	4 x ACC (10 bar)
Butterfly valve (RCP)	opened	opened	opened	opened	opened	opened	opened	closed	opened	opened	opened	opened
Quenching successful	YES	YES	YES	YES	YES	NO	YES	YES	NO	YES	YES	YES

Source: Framatome GmbH, 2020.

Figure 4.3. Test i1.1 run 1: Different trends for the PCT depending on different sets of initial/boundary conditions



Source: Framatome GmbH, 2020.

4.1. Conclusions

Conducting **test i1.1** as an integral test (i.e. taking into account the interaction of many different sources of influence parameters on the primary and secondary sides) allowed for a detailed view of important phenomena such as steam binding. With its two runs, test i1.1 provided a broad spectrum of parameter settings for both phases of the LB-LOCA addressed: the flooding phase (quench procedure in the core) and the long-term cooling phase. MCPs at standstill contribute significantly to the overall head loss in the loop and the resulting pressure build-up in the core, with an impact on the propagation of the quench front.

5. Tests i2: Design-extension conditions SB-LOCA and IB-LOCA (counterpart testing with PKL III H.1 and ROSA/LSTF)

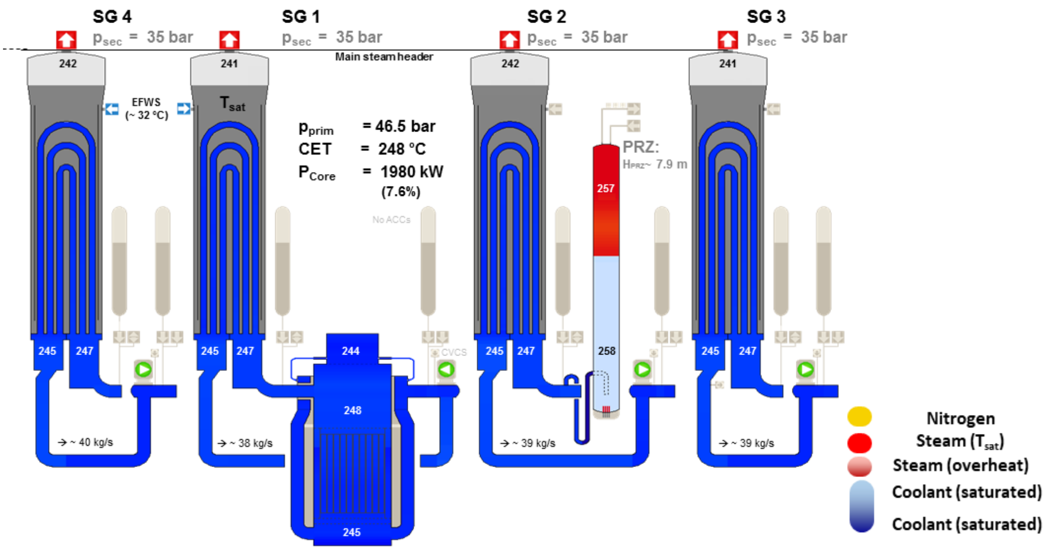
The PKL III i2 tests series aimed at investigating small break (SB) and intermediate break (IB) loss-of-coolant accidents. The first case was performed within the i2.1 test, which was divided into two runs, while the second scenario was simulated in the frames of the i2.2 test with different setups of the break, i.e. 13% and 17% upward breaks. Both tests constituted counterpart investigations with respect to experiments performed at the ROSA/LSTF test facility.

Test i2.1 addressed a 1% hot-leg break with additional failure of the HPSI. Secondary-side depressurisation was implemented as an accident management measure that was triggered once the core uncovering took place and the peak cladding temperature reached 400°C. ACCs were available at the cold-leg side and set to inject water into the primary circuit as soon as the pressure decreased to 40 bar. LPSI was assumed to be available at the cold-leg side, too. Apart from investigating the general course of events, the test served to examine phenomena of particular interest including heat transfer in the presence of nitrogen and core quenching due to the swell level increase as a result of primary side pressure reduction and loop seal clearing initiated by the ACC injection.

For test i2.2, a cold-leg IB-LOCA case was simulated. Similarly to test i2.1, i2.2 was designed to provide design-extension conditions with reduced availability of safety injections. Namely, run 1 and 2 assumed HPSI, LPSI and ACC available only in 2 out of 4 loops, while in run 3 the availability of HPSI and LPSI was reduced to only 1 out of 4 loops. Further variations between the i2.2 test runs included various break sizes: 13%, 17% and 17% in runs 1, 2 and 3, respectively.

Hence, test i2.2 was designed as a counterpart analysis including three different configurations of the same accident scenario. The motivation for structuring the investigation this way was to contribute to addressing questions on discrepancies with respect to PCT excursion between experiments (13% vs. 17% break) and post-test calculations.

Figure 5.1. Initial test facility condition before i2.2 tests on IB-LOCA

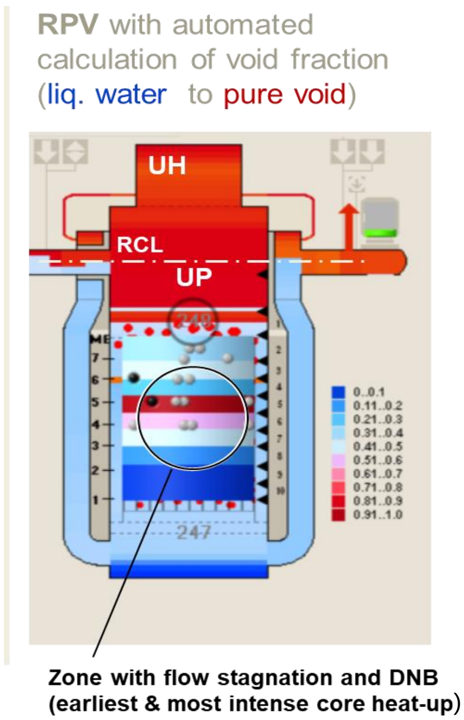


Source: Framatome GmbH, 2020.

The i2 test series provided a database for a detailed analysis on design-extension SB- and IB-LOCA with various configurations of emergency core cooling system availabilities. The results of test i2.1 (SB-LOCA) showed that the secondary-side depressurisation (SSD) is a well-suited measure to reduce the pressure at the primary side down to the set pressure of ACCs. Furthermore, bringing the pressure at the secondary side to below its value at the primary side led to an establishing of the reflux condenser operation mode, and thereby to a coupling of both pressures. Ongoing reduction of the primary side pressure triggered the ACC injection, which led to the loop seal clearing and efficient restoration of core cooling shortly after the beginning of the injection. In test i2.1, it was assumed that nitrogen can flow from ACCs to the primary circuit. The inflow of gas after the emptying of the ACC vessels and its accumulation in the SGs led to a hindering of heat transfer and diminished the pressure reduction at the primary side. This, in turn, indisposed triggering of the LPSI.

A different size and location of the break in test i2.2 (IB-LOCA) resulted in a distinctive course of events in this experiment. In case of a cold-leg (CL) break, the dry-out and core heat-up is expected to occur around the stagnation point in the core, which develops due to the establishment of a large pressure sink at the cold side, which dislocates the water from the downcomer into the MCP direction. The stagnation point position is mostly predetermined by the break size and pressure differentials along the hot (UP, SG, cross-over leg, MCP) and the cold side (RPV, downcomer, CL) to the break location. After the start of the quenching process, which was triggered by the loop seal clearing stemming from an accumulation of steam volumes on the hot side, the water hold-up drops, leading to a covering of the core at the stagnation point height and an uncovering of its top region. Similar to what was found in test i2.1 (hot-leg break), in case of a CL break, the LSC – albeit induced by different root causes – had a dominant influence on core quenching for both the SB-LOCA and the IB-LOCA sequences.

Figure 5.2. IB-LOCA with point of flow stagnation in RPV (picture from PKL data visualisation)



Source: Framatome GmbH, 2020.

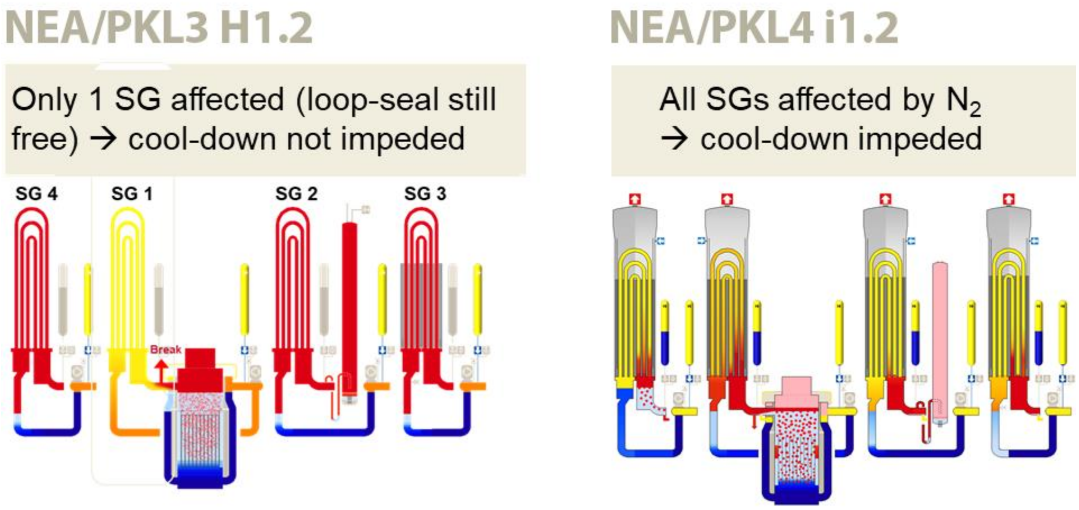
For test i2.2 (IB-LOCA) the main conclusions are:

- The stagnation point position (Figure 4) is mostly predetermined by the break size and pressure differentials along the hot (UP, SG, cross-over leg, MCP) and the cold side (RPV, downcomer, CL) to the break location; the absolute pressure level in the RCS plays a tangential role.
- For a correct replication of the main core heat-up sequence during the blow-down phase by T/H system codes, the flow stagnation phenomenon in the core during the blow-down phase needs to be replicated.

Regarding nitrogen intrusion into the reactor coolant system (RCS) after the ACC depletion, the size and position of the break (hot or cold side) are decisive for the impact of N_2 on the outcome of the transient. If the SGs are not needed as a heat sink (due to a large break size in case of IB-LOCA), the position of the break (hot or cold side) is irrelevant; if – in the case of an SB-LOCA where SGs are needed as heat sinks – the ACC injection points and the break are both on the same side (i2.2 IB-LOCA, break and injection on cold side), nitrogen mostly escapes through the break, again rendering its impact negligible. Accordingly, non-condensable gases may only develop adverse effects for transients in which SGs are needed for heat removal, if the non-condensable gas can accumulate in the SGs and, as demonstrated by the NEA PKL-3 H1.2 and its counterpart NEA PKL-4 i2.1 (Figure 5), if the non-condensable accumulations develop isochronously in all active SGs. In sum, test i2.1 (SB-LOCA) showed:

- that nitrogen presence and in particular its distribution in the RCS can have an impact on the outcome of an SB-LOCA transient;
- the advantages of having a 4-loop integral test facility representing a 4-loop PWR with respect to possible diversity in between individual loops.

Figure 5.3. Different distributions of nitrogen on primary side in two tests (H1.2 vs. i2.1) with different impacts on outcome of transient



Source: Framatome GmbH, 2020.

5.1. PWR PACTEL Tests SB-LOCA (NCG-21, NCG-22 and NCG-23)

Within the NEA PKL-4 project, additional experiments investigating the influence of nitrogen in the primary circuit were conducted at the PWR PACTEL test facility. The background for the PWR PACTEL tests is the possible impact of the presence of non-condensable gases in the RCS on the ability to perform core cooling. In some plants, the release of gaseous nitrogen to the primary side is prevented by automatically closing the accumulator injection line at the end of the discharge. If the automatic closure system fails, nitrogen can flow to the reactor cooling system when the accumulators are empty of water. The accumulator water is saturated with dissolved nitrogen, so a quantity of nitrogen will always enter the primary system with the accumulator discharge.

Nitrogen released into the primary cooling system can have a direct effect on the reactor cooling conditions, such as water distribution and condensation. Under suitable flow conditions, nitrogen released and flowing to a cold leg can cause a pressure rise in the cold leg end and the downcomer top parts with respect to the upper plenum pressure. Hence, nitrogen can temporarily increase the water level in the core due to a piston effect, i.e. shortly redistributing water masses in the vessel.

This effect has been studied in the UPTF, BETHSY, and PWR PACTEL test facilities. Similar experiments have also been carried out in the ROSA/LSTF, and PKL facilities. In the earlier PWR PACTEL experiments, the piston effect was not observed. The break location in the cold leg was quite close to the downcomer and allowed the accumulator injection water and nitrogen to easily escape from the system. In the PWR PACTEL experiments of the NEA PKL Phase 4 project, the break location was in a cold leg near the steam generator on the vertical side of the pipe. The effect of the line between the downcomer top part and the upper plenum was tested as well. In total, three tests (NCG-21, NCG-22, NCG-23) were performed; the first one served as a reference experiment that assumed no nitrogen inflow into the primary system, while the other two postulated the possibility of nitrogen intrusion into the reactor coolant system.

The PWR PACTEL tests in the NEA PKL Phase 4 project aimed at independently verifying whether the claimed positive effect of nitrogen on the core cooling could be reproduced and at generating data for the development and validation of thermal-hydraulic system codes.

5.1.1. Conclusions

During the experiments, the injected nitrogen did not cause any significant pressure rise in the cold leg and the downcomer with respect to the upper plenum, so no piston effect was observed in any of the tests with nitrogen inflow into the primary circuit. A temporal delay of the core heat-up occurred only when enough water was replaced with nitrogen in the break flow.

5.2. PMK-2 experiment on IB-LOCA (PMK-2 test 1, PMK-2 test 2)

The design-extension IB-LOCA topic was also adopted for the two PMK-2 experiments which were conducted as part of the NEA PKL-4 project. Above all, the PMK-2 experiments aimed at demonstrating the importance of the break location (hot or cold side); depending on plant geometry, there may be significant differences in the evolutions of hot and cold-leg break transients.

Two IB-LOCA tests (17% cold and hot-leg break, respectively; design-extension conditions: no HPSI 1 LPSI pump, hydro-accumulators [HA] available) have been performed at the PMK-2 facility in order to illustrate the different evolutions of the hot and cold-leg break case and gain experimental data to strengthen the validation database for the IB-LOCA case in general.

Validation calculations for a broad spectrum of break sizes (13-21%) have been completed before selecting the break size and formulating the experiments' final boundary conditions and procedures. Both experiments started from initial nominal conditions simulating a VVER 440 full-load mode of operation.

5.2.1. Conclusions

Under identical initial conditions, the hot-leg break (with the same diameter) produced significantly different results compared to the cold-leg break. The LPIS initiation at an elevated cladding temperature successfully mitigated the core heat-up for the cold-leg break. In the case of the hot-leg break, the same LPIS initiation strategy could not successfully restore core cooling (automatic power shutdown by core protection system).

In sum, the data analysis concluded that the IB-LOCA cases need further investigation.

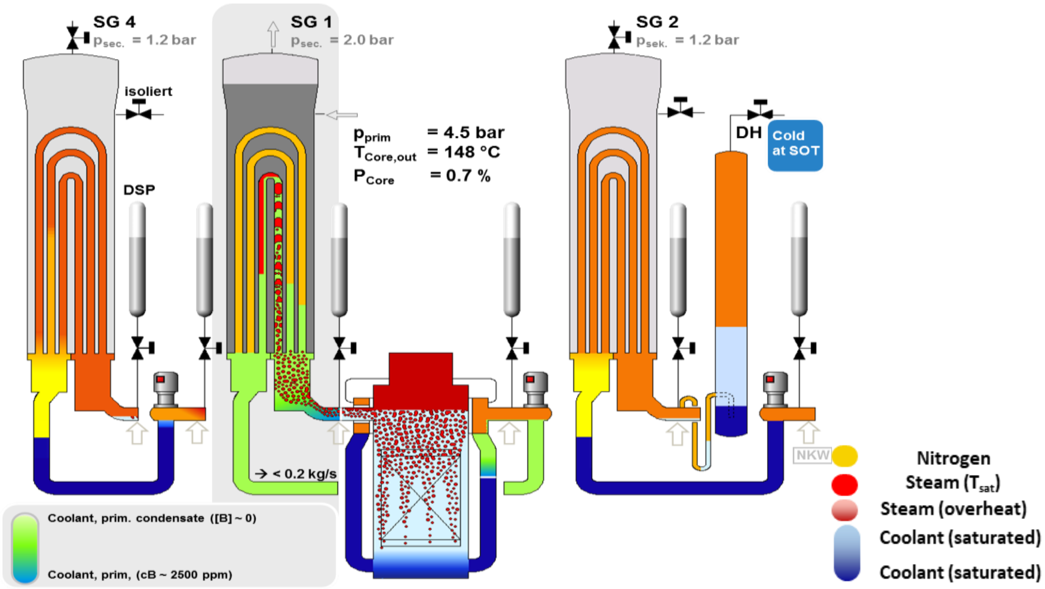
Together with the PWR PACTEL experiments on SB-LOCA, the PMK2-tests contributed to the collective database on SB/IB-LOCA transients provided by the NEA PKL-4 project.

6. Test i3: Test on failure of RHRS from cold shutdown condition

The PKL III i3 test series continued the investigation into the failure of residual heat removal systems. The postulated scenario was assumed to occur in a 3-loop plant in cold shutdown conditions with a ¾ loop primary side inventory level. The test was conducted in two runs with different set points for the activation of the feed water and main steam systems on the secondary side: 2 bar in run 1 and 5 bar in run 2 on the secondary side of the SG.

One of the main objectives of the i3.1 test was investigating the progression of temperatures at the RHRS suction positions in the hot legs immediately after the failure of residual heat removal and the waiting periods until saturation conditions are reached at the suction positions of the RHRS in the hot legs. Furthermore, the development of boron dilution mechanisms in the active SG (Figure 6), the impact of the secondary-side pressure on the primary side pressure level and boron dilution, as well as the effectiveness of procedures to restore the operation of the RHRS were of interest.

Figure 6.1. Situation after approx. 3h after failure of RHRS – coolant transport with boron dilution (green)



Source: Framatome GmbH, 2020.

Another objective of the i3.1 test was to draw a comparison against other tests investigating the failure of RHRS by verifying and applying conclusions on boron dilution from earlier tests for 4-loop plants and 3-loop plants.

6.1. Conclusions

Test i3.1 confirmed that in case of loss of residual heat removal in cold shutdown conditions, in the absence of active operator interventions, a quasi-steady-state condition with assured heat removal to the secondary side is always established even if only one SG is operable. The primary side pressure level is determined by the secondary-side pressure-control level. Heat transfer can be significantly improved (i.e. primary pressure is lower) if an additional SG is initially operable or activated promptly after failure of the RHRS in order to gain additional buffer time and space at low primary equilibrium pressure (p_{prim} < 5 bar) for actions/procedures (ECCI) to restore the fluid parameters required for reactivation of the RHRS in the long term.

The results on the development of the boron concentrations in both i3.1 test runs confirmed also the outcomes of an earlier PKL test series for a 4-loop PWR. In particular, for test i3.1 the main conclusions are:

- For a single active SG, continuous boron dilution (as a result of intermittent transport of condensate in the U-tubes) below the active SG cannot be completely excluded regardless of the secondary-side pressure level.
- As expected, at a pressure level of 5 bar on the secondary side, the tendency for continuous boron dilution is reduced; no continuous boron dilution process was measured for this case. Nevertheless, the results imply that without additional coolant injections, a boron dilution process cannot be completely excluded for only one active SG, regardless of whether the main steam pressure level is 2 or 5 bar. In contrast, if there are at least two active SGs, boron dilution below the active SGs can be excluded for the PKL facility.

7. Tests i4: Investigations on procedures under cooldown in natural circulation operation mode for design-basis and design-extension accidents

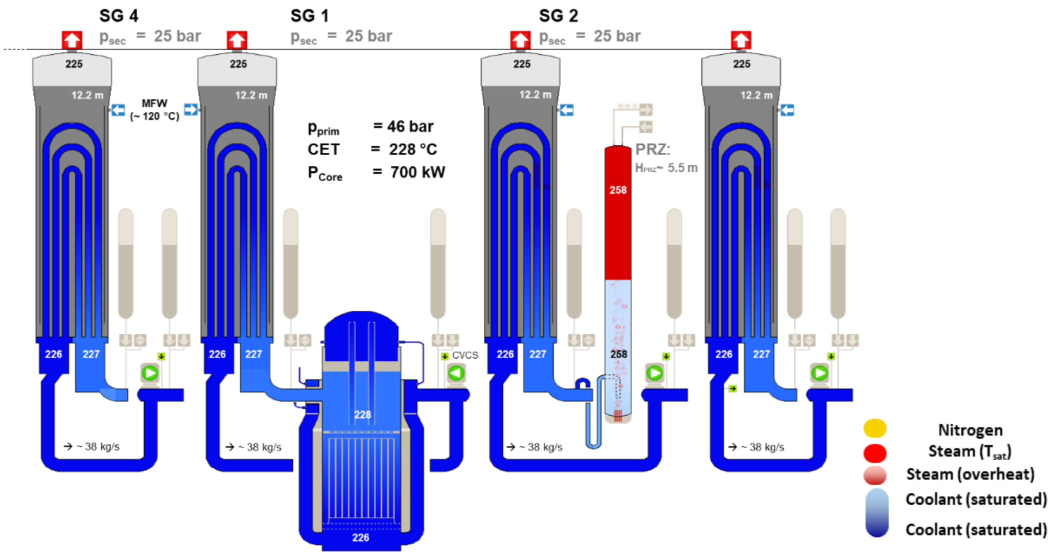
Within the PKL III i-experimental programme, the facility was modified to better correspond to modern reactors currently being built around the world. The modification process encompassed a change of the UH vessel and UP internals, embedding of the upper support plate and a new upper core plate (UCP), and installation of replications of control rod guide assemblies (CRGA) and rod cluster control assemblies (RCCA). These modifications have a significant influence on the plant operation both during normal operation and in case of an accident.

Both experiments in the i4 test series aimed at investigating the upper head void behaviour in NC cooldown operation mode. The background scenario for the i4.1 and i4.2 tests was loss of offsite power; each of the tests had, however, different assumptions with respect to the course of events.

The test i4.1 scenario employed cooldown in emergency power mode (Figure 7), i.e. assuming the reactor scram, the facility was cooled down via main steam relief from the secondary side. In parallel to the cooldown under NC conditions, pressure reduction of the primary side was conducted by the chemical volume control system which injected water into the pressuriser (PRZ) in the auxiliary spraying mode. Due to the RCS pressure reduction and the corresponding saturation temperature decrease, the fluid in the upper region of the RPV can undergo flash evaporation, causing void formation in the dome. By virtue of the recent facility modifications, the objective of this test was to investigate the void behaviour.

In contrast, the i4.2 test investigated a design-extension accident by assuming additional extended loss of AC power and corresponding failure of emergency diesel engines. As a consequence, the safety injection, the pressuriser spraying, and emergency feed water were not available. According to the operating manual, the primary side pressure under these conditions is decreased by opening of the PRZ relief valves. This solution has its disadvantages, however, so the i4.2 test aimed at investigating an alternative procedure. This alternative was based on partial depressurisation of the single steam generator which would then be fed with a mobile pump triggered passively by a passive pressure pulse transmitter (PPPT). Apart from investigating NC under beyond-design-basis accident, another objective of the test was thus verifying the PPPT’s system reliability as a passive signal for triggering other safety-related systems or devices.

Figure 7.1. Initial test facility condition for test i4.1 – replication of hot standby condition in PKL at reduced pressure



Source: Framatome GmbH, 2020.

7.1. Conclusions

The i4 test series encompassed an analysis of cooldown in a NC mode of operation with the focus either on the behaviour of the upper head void following design-basis events (e.g. emergency power mode; i4.1) or on the effectiveness of cooldown procedures in an NC mode of operation for beyond-design-basis conditions (ELAP) in the RPV configuration with installed USP.

Test i4.1 and its comparison with the G6.1 test conducted within the NEA PKL-2 project indicated discrepancies between different UH/UP configurations. In particular, it illustrated the process of void formation in the UH and further propagation of the steam bubble through the CRGAs leading to formation of a second void – in the UP, below the USP. Test i4.1 also showed heat transfer mechanisms and flow patterns, which could not be observed before in KONVOI-type RPV configuration. Despite these significant differences between the courses of events in the upper region of the RPV, test i4.1 confirmed that 15 K subcooling at the core exit supports preventing the steam bubble from expansion into the hot leg in the facility. The main conclusions of test i4.1 are:

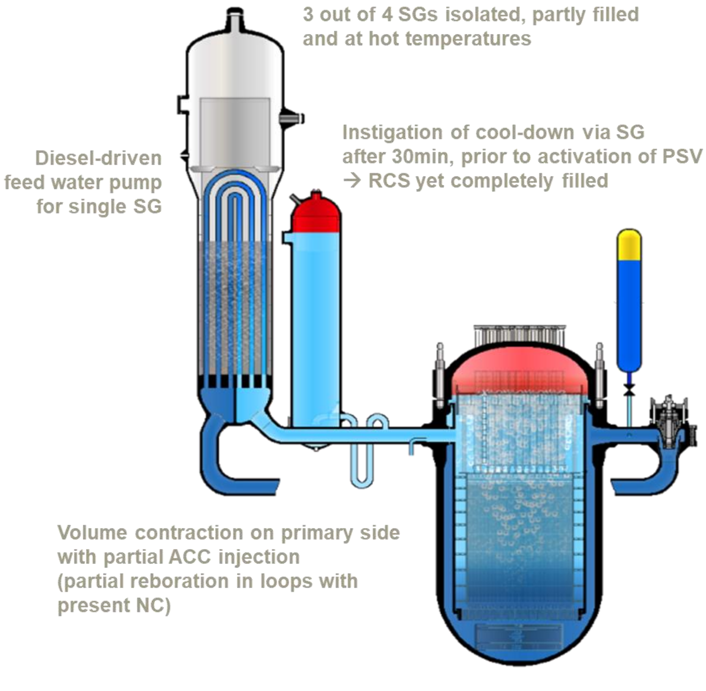
- The closed dome configuration (e.g. in the EPR) may produce two independent void bubbles during cooldown in NC mode of operation: one above the USP, one below.
- The activation of 2 out of 4 reactor coolant pumps results in a fast condensation of the steam bubble in the UP; the collapsing of void in the UH (second steam bubble) proceeds more slowly.
- T/H system codes employed for safety analyses of the given PWR and scenario need adequate models to represent these unique behaviours in order to correctly predict the void volume generated in the RPV during cooldown in an NC mode of operation (and the resultant PRZ level development).

Test i4.2 examined an alternative procedure for RCS cooldown after extended loss of AC power (ELAP, Figure 8). The test results demonstrated that partial cooldown of one SG during the postulated scenario was sufficient for secured heat removal. As long as a periodical water injection provided by a mobile pump, for instance, assured a sustained and sufficiently high SG water level, the RCS pressure and temperature could be effectively decreased. Apart from the potential influence of the auxiliary feed water injection, a passive mechanism aiming at triggering this injection was examined within the experiment. Test i4.2 also showed that under the boundary conditions established in the test, NC did not stop as long as saturation conditions were not reached in the SGs U-tube apexes. In other words, the flow-counteracting force, which stemmed from SGs acting as heat sources, did not hinder the flow sufficiently to stop circulation, and only reaching the saturation point led to flow stagnation in the affected loops.

Test i4.2 provided the following main conclusions:

- For safety analyses of the given ELAP scenario with T/H codes, capturing the flow-counteracting force (i.e. the formation of a negative temperature differential where SGs slowly start to act as heat sources) is crucial for the successful replication of the course of events.
- Under given test conditions, the PPPT is a reliable solution.

Figure 7.2. Scheme of PWR status before instigation of preventive AM-procedure – continuous NC on primary side, slight inventory loss from discharge at PRZ



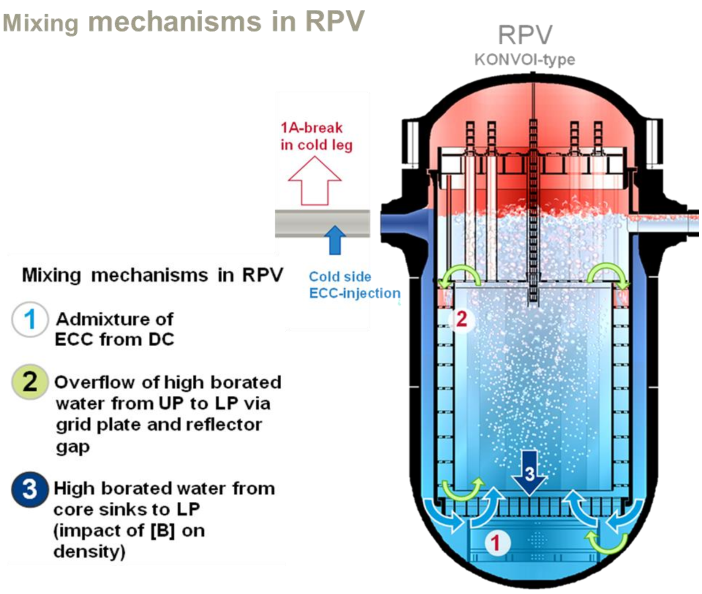
Source: Framatome GmbH, 2020.

8. Test i5: Boron precipitation following LB-LOCA

Boron precipitation occurs in the long-term cooling phase following a LOCA with large (here: 1A) cold-leg break, a break size large enough to avoid subcooling of the core by cold-leg safety injection (due to ECC loss through the break). In this case, continuous boiling in the reactor core is expected during the long-term phase in PWR plants with only cold-leg injection of ECC water. In this state, the reactor core is rewetted and covered by two-phase mixture. During evaporation, the boron injected with the ECC water remains in the liquid phase on the hot side and accumulates continuously. At boron values that exceed those 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.

The most decisive parameters for the speed of the boron enrichment process are the core power (residual heat) and the size of the mixing volume, i.e. the amount of liquid water on the hot side. The direct impact of the residual heat manifests in an increased evaporation rate on the one hand and an increased amount of steam (void volumes) on the hot side on the other hand. Buoyancy-driven mixing mechanisms in the RPV interconnect the RPV section volumes (riser/core section, lower plenum, reflector gap, Figure 8.1. An important factor for the size of the mixing volume in the RPV is the extent to which the water volumes of the reflector gap and the lower plenum volume both contribute to the mixing volume. The sizes of these additional mixing volumes are important because once mixing starts in the lower plenum, the precipitation process is slowed down in the core and in the upper plenum regions.

Figure 8.1. Scheme of PWR in long-term cooling phase following LB-LOCA



Source: Framatome GmbH, 2020.

Test NEA PKL-3 H5.1 confirmed the effectiveness of mixing mechanisms that interconnect the different coolant volumes in the RPV sections. They interconnect the volumes of the core region, but also the reflector gap (even for swell levels that just cover the core by overflow across the grid plate) and the entire lower plenum. For higher swell levels, the volumes of the upper plenum, hot legs and SG inlet chambers also contribute to the enrichment process. The moment in which the lower plenum starts to contribute to the mixing seems to be partially linked to the magnitude of the reflector water volumes and the size of the reflector bypass flow (through which the saturated water from the core exit may circulate back towards the lower plenum under continuous boiling condition with a swell level in the UP). For PWR designs which do not have a large water-filled reflector gap or which allow for only small reflector bypasses, the lower plenum may only contribute to the mixing volume at a later stage – when the impact of the boron concentration [B] on the coolant density outweighs the effect of buoyancy due to temperature – and/or the volume of the reflector gap may be smaller. In both cases, the volume of water available for mixing may be significantly smaller, and consequently the precipitation process may evolve faster.

For test i5.1, the reflector gap was closed by blocking the holes in the top plate, which led to disabling of the flow between the bypass gap and the UP. In doing so, the impact of the reflector bypass on the overall size of the mixing volume, i.e. on the combined effects of the reflector gap water volume (direct contribution) and the extent of the lower plenum participating in the mixing (indirect contribution), was to be assessed. Furthermore, a video inspection was installed in the upper and lower plenum in order to enable a visual illustration of the potential precipitation process in the respective components.

8.1. Conclusions

Test i5.1 further investigated the increase of boric acid concentration and the potential boron precipitation in the long-term cooling phase after an LB-LOCA. The main conclusions of test i5.1 are:

- Crystallisation of boric acid is not expected in the analysed scenario within several hours of operation, even under extremely favourable conditions for boron content increase (closing of reflector gap).
- Furthermore, the relationship between the density of water/boric acid and the subcooled fluid density in the lower plenum dictates the involvement of the lower plenum volume in the mixing process; the reflector gap re-circulation flow plays an inferior role.

9. Test i6: Investigations on multiple steam generator tube ruptures

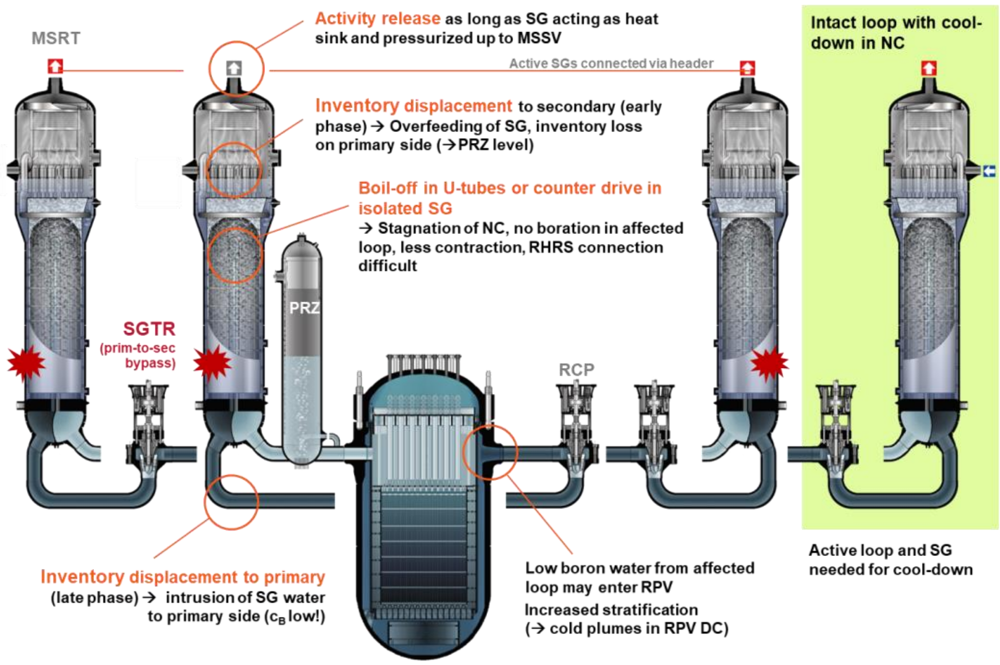
In the i6 test series, one experiment with 2 runs was performed. The background scenario for test i6.1 was multiple tube ruptures in 3 out of 4 steam generators (SG). Such an accident could possibly occur in a power plant as a result of high seismic loads, i.e. earthquakes. In case of such an event, coolant is enabled to flow from the primary to the secondary side, leading to a radioactivity release to the secondary-side circuit. As a result of the postulated event and corresponding inventory displacement from the primary to the secondary side, the pressure difference between the two circuits would decrease. Due to the pressure limitation of the PKL test facility, test i6.1 was started after the initial primary side pressure drop resulting from the break occurrence and before the start of the secondary-side cooldown implemented as an accident management procedure. The test was performed with the additional assumption of failure of the HPSI system and the chemical volume control system.

Test i6.1 was composed of two runs. In both runs, the first measure was the depressurisation of the one remaining intact SG by opening of the main steam relief valve followed by the feed water supply into one defective SG. Thereafter, run 1 assumed a primary side depressurisation via the pressuriser relief valve with subsequent accumulator and low pressure safety injection. The test was ended by activating the residual heat removal system after the attainment of subcooled condition in the hot legs. In run 2 – for the purpose of exploring alternative actions to reduce the primary side parameters – the MCPs were started at reduced rotation speeds. The conditions for ACC injection or LPSI were not reached in run 2. The test was ended with forced circulation in all loops and heat removal from the core by heat transfer to the remaining intact SG.

The main objectives of the i6.1 test include investigation of break flow between the primary and secondary side and corresponding pressure courses, boron dilution in the primary side, cooldown of intact SG and related reversion of the break flow, and initiation of NC in defective loop due to auxiliary feed water.

Furthermore, specific phenomena related to different procedures were of interest within the individual test runs. In particular, in run 1 the evaporation in the SG U-tubes apexes as a result of primary side depressurisation and influence of ACC injection was of interest. In run 2, besides a comparison with run 1, the objective was to investigate boron concentration before and after restart of the reactor coolant pumps (MCP).

Figure 9.1. MSGTR scenario with safety goals/objectives



Source: Framatome GmbH, 2020.

9.1. Conclusions

Experimental analysis of multiple SG tube ruptures and corresponding AM-procedures performed in test i6.1 indicated that the secondary-side depressurisation is an effective measure leading to a reversion of the break flow and therewith to an increase of inventory in the primary side and a reduction in the radioactivity release to the secondary side. Furthermore, feed water supply into a defective SG, implemented as a measure that counteracts loss of inventory due to primary side depressurisation via the PRZ relief valve, successfully reduced the coolant depletion. Although the SG auxiliary feed water injection and loss of inventory via the PRZ relief valve led to a boron dilution in the core, a rapid reduction of its concentration is not expected. Test i6.1 indicated that (auxiliary or passive) feed water supply of at least one SG on the secondary side is crucial for heat removal and, for this scenario in particular, the preservation of primary side water inventory.

10. Applicability of experimental results to real plant conditions

The data application and interpretation (DIA) section of the final report comprises a review of the PKL experimental results from the view of the project partners with respect to safety relevance, scaling value and lessons learnt from the execution of experiments. This section constitutes an essential part of the final report.

Experimental results obtained through the NEA PKL-4 programme added to 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 previous PKL-programmes, NEA PKL-4 participants were actively involved in promoting analytical activities during the execution of the project aiming at understanding important thermal-hydraulic phenomena/processes and assessing 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 may be used for improvement of plant performance and operation for a particular scenario. 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 can be compared with corresponding analyses using thermal-hydraulic system codes and, for some local phenomena, with additional separate effect tests.

11. Other activities

11.1. "Analytical benchmark" activity

In the NEA PKL-4 project, a so-called analytical benchmark exercise was performed. This activity included a blind benchmark on the PKL i2.2 experiment on IB-LOCA aiming at evaluating the current capabilities of the thermal-hydraulic codes on the domain of interest and developing a common understanding regarding the observed phenomena and their interactions.

Conclusions and recommendations were formulated with respect to:

- conducting the benchmark (i.e. how to improve further benchmarks and the significance of results);
- findings derived from the joint PKL-ATLAS workshop;
- phenomenological assessment (e.g. the identification of sources of uncertainty).

A separate project internal report was issued for this activity.

11.2. Joint PKL-ATLAS analytical workshop

An analytical workshop was arranged in co-operation with the ATLAS-2 PRG and took place in Barcelona following the 5th NEA PKL-4 PRG/MB meetings at the premises of the Technical University of Catalonia, from 7 to 9 November 2018 (NEA, 2018). This workshop aimed to bring together code users that performed calculations reproducing the PKL-4 (PKL and PACTEL) and ATLAS-2 experiments to present their simulation code results and to discuss modelling issues and problems. The workshop attracted 55 participants from 11 countries. It included 24 presentations covering a general overview of both programmes, the analyses of the benchmark exercise organised within the PKL-4 project, and some analyses related to other PKL-4 and ATLAS-2 tests, including applications to reactor cases. The activity provided an efficient way to evaluate the current code capabilities for the scenarios conducted in the projects. Emphasis was placed on the execution of blind calculations as a way of testing the predictive capability of codes. A general recommendation was the need to perform combined sensitivity and uncertainty analyses to qualify the code performance. As highlighted in previous similar analytical workshops (Barcelona 2003, Pisa 2005, Budapest 2006, Pisa 2010, Paris 2012, Lucca 2016), bringing experimental and analytical specialists together in a common conference has been very fruitful, particularly for the effective and necessary interaction between codes and experiments for the solution of topical safety issues.

12. Outlook and future perspective

The next NEA-guided project at the PKL test facility (ETHARINUS) will provide a logical continuation of the experimental activities of the NEA PKL-4 project with respect to addressing open questions and providing a more immersive look at chosen scenarios. Furthermore, it will add a topic related to the future of LWR design: assessing the performance of innovative passive systems.

13. NEA PKL-4 test matrix

Table 13.1. NEA PKL-4 test matrix

	Topic	No. of tests/runs
Design-extension-condition scenarios: effectiveness of AM measures		
PKL III i2.1	Design-extension SB-LOCA (hot leg) with additional safety system failures, counterpart test to PKL III H1.2 and ROSA/LSTF SB-CL-12	1/2
PKL III i2.2	Design-extension IB-LOCA (13% or 17% cold-leg break) with (varying) additional safety system failures, counterpart test to ROSA/LSTF IB-CL-05 and IB-CL-03	1/3
PKL III i4.2	ELAP – cooldown under NC conditions	1/1
PKL III i6.1	Multiple SG Tube Ruptures	1/2
PWR PACTEL tests	Influence of non-condensable gases – investigation of piston effect occurrence possibility	3/3
PMK-2 test 1	Design-extension IB-LOCA (17% cold-leg break) with additional safety system failures	1/1
PMK-2 test 2	Design-extension IB-LOCA (17% hot-leg break) with additional safety system failures	1/1
Events from cold shutdown states (failure of RHRS)		
PKL III i3.1	Failure of RHRS during ¾-loop operation in cold shutdown condition, RCS closed, 3-loop plant	1/2
Cooldown in NC condition		
PKL III i4.1	Investigation of UH/UP void behaviour during cooldown in NC mode of operation for EPR-type RPV	1/1
Investigation on boron precipitation following LB-LOCA		
PKL III i5.1	Boron precipitation following LB-LOCA (reflector bypass closed)	1/1
Parametric studies		
PKL III i1.1	Two-phase flow phenomena during quenching of the core (refilling and flooding phase) for LB-LOCA	1/2

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