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Summary Report of the NEA-Advanced Thermal-Hydraulic Test Loop for Accident Simulation (ATLAS) Joint Project

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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

AM	Accident management
APR+	Advanced Power Reactor Plus
ATLAS	Advanced Thermal-Hydraulic Test Loop for Accident Simulation
ATWS	Anticipated transient without scram
BARC	Bhabha Atomic Research Centre (India)
CEA	Commissariat à l'énergie atomique et aux énergies alternatives (France)
CFD	Computational fluid dynamics
CNPRI	China Nuclear Power Technology Research Institute (China)
CSN	Spanish Nuclear Safety Council (Spain)
CSNI	Committee on the Safety of Nuclear Installations (NEA)
DBA	Design basis accident
DEC	Design extension conditions
DEGB	Double-ended guillotine break
DVI	Direct vessel injection
ECC	Emergency core cooling
EDF	Électricité de France (France)
EOP	Emergency operating procedure
FFTBM	Fast Fourier Transform Based Method
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit, a German non-profit organisation which deals with technical-scientific research and provides expertise (Germany)
IBLOCA	Intermediate break loss-of-coolant accident
IBRAE-RAS	Nuclear Safety Institute of the Russia Academy of Sciences (Russia)
IET	Institution of Engineering and Technology
KAERI	Korea Atomic Energy Research Institute (Korea)
KEPCO E&C	Korea Electric Power Corporation Engineering & Construction Company, Inc. (Korea)
KEPCO-NF	Korea Electric Power Corporation Nuclear Fuel (Korea)
KINS	Korea Institute of Nuclear Safety (Korea)

KHNP-CRI	Korea Hydro & Nuclear Power Central Research Institute (Korea)
LBLOCA	Large break loss-of-coolant accident
LOCA	Loss-of-coolant accident
LPI	Low pressure injection
LSC	Loop seal clearing
LSTF	Large scale test facility
LWR	Light water reactor
MPa	Mega Pascal
MSSV	Main steam safety valves
MTA-EK	Hungarian Academy of Sciences Centre for Energy Research (Hungary)
MWe	Megawatt electric
NCFM	Natural circulation flow map
NEA	Nuclear Energy Agency
NINE	Nuclear and Industrial Engineering
NITI	A.P. Aleksandrov Scientific Research Technological Institute (Russia)
NRA	Nuclear Regulation Authority (Japan)
OKBM	OAO I. I. Afrikantov OKB Mechanical Engineering (Russia)
PAFS	Passive auxiliary feed water system
PCCT	Passive condensation cooling tank
PCHX	Passive condensation heat exchanger
PCT	Peak cladding temperature
PKL3	Primary Coolant Loop Test Facility
POSRV	Pilot operated safety relief valve
PRG	Project Review Group
PSI	Paul Scherrer Institute (Switzerland)
PWR	Pressurised water reactor
RCP	Reactor coolant pump
RCS	Reactor coolant system
RIR	Risk-informed regulation
ROCOM	Rossendorf Coolant Mixing Model
ROSATOM	State Atomic Energy Corporation (Russia)
SBLOCA	Small break loss-of-coolant accident
SBO	Station blackout
SCS	Shutdown cooling system

SG	Steam generators
SGTR	Steam generator tube rupture
SIP	Safety injection pump
SNPSDC	State Nuclear Power Software Development Centre (China)
SSM	Swedish Radiation Safety Authority (Sweden)
STUK	Radiation and Nuclear Safety Authority (Finland)
TLOFW	Total loss of feed water
U.S. NRC	United States Nuclear Regulatory Commission (United States)
VTT	Technical Research Centre of Finland Ltd (Finland)

1. Introduction

The NEA Advanced Thermal-Hydraulic Test Loop for Accident Simulation (ATLAS) Project is one of the NEA supported joint projects under the auspices and with the support of the NEA. It started from April 2014, with a three-year project period. This joint project focused on key light water reactor (LWR) thermal-hydraulic safety issues related to multiple high risk failures highlighted from the Fukushima Daiichi accident utilising a large scale test facility of ATLAS.

ATLAS is an integral effect facility simulating an APR1400 (Advanced Power Reactor 1 400 megawatt electric [MWe]) with a 1/2 reduced height. The scaling factor of the fluid volume is 1/288. The NEA-ATLAS Project focused on the validation of simulation models and methods for following complex phenomena of high safety relevance to thermal-hydraulic transients in design basis accident (DBA) and beyond DBA. An international expert meeting was held at the Lappeenranta University of Technology (Lappeenranta, Finland) on 13-14 June 2013, to identify areas of interests and issues to be addressed in the joint project. Also there was discussion for the types of testing to investigate the defined issues. More than 30 experts from 13 countries participated in the expert meeting and they unanimously agreed to the proposal and recommended that the NEA Secretariat prepare with the Korea Atomic Energy Research Institute (KAERI) a project agreement with the technical content revised according to the comments received at the meeting. As a result, a total of eight integral effect tests from five different topics were specified and carried out with the ATLAS facility under the framework of the NEA-ATLAS Project. Repeatability of each ATLAS test was confirmed by performing necessary pre-tests before each official test.

Before the main test campaign started, natural circulation characterisation tests were performed to identify the natural circulation characteristics of the ATLAS facility from a viewpoint of scaling. Two tests were performed with a power level of 5.4% and 2.3% and it turned out that the natural circulation characteristics is similar to those of other similar integral effect test facilities when it is compared to the well-known NCFM (Natural Circulation Flow Map). It was confirmed that the ATLAS data fall into the envelope of the measured curves in PWR in spite of the half height scale.

2. Summary of Experimental Tests

The key outline of the tests performed in the NEA-ATLAS Project is as follows:

- **A1**: Prolonged SBO with active or passive secondary cooling (2 tests)
- **A2**: SBLOCA during SBO such as RCP seal failure or SGTR (2 tests)
- **A3**: Total loss of feed water assuming stuck half open of POSRV and half failure of safety injection pump (1 test)
- **A4**: Intermediate break LOCA of 17% cold leg break (1 test)
- **A5**. Counterpart tests for SBLOCA (1% cold leg break) and IBLOCA (13% cold leg break) (2 tests)

Design extension conditions (DECs) such as a station blackout (SBO) and a total loss of feed water (TLOFW), not seriously considered from a viewpoint of DBA, were incorporated in the test matrix. In particular, the prolonged SBO was highlighted in this project reflecting the great international interest after the Fukushima Daiichi accident. Possible emergency operating procedure (EOP) measures to be employed would be secondary-side depressurisation followed by dumping the main steam, accumulator injection, and an active feed of the secondary sides from a mobile diesel generator in the long run. Meanwhile, various passive safety systems have been recently proposed to improve safety and reliability of an ultimate heat removal system without any operator action during an SBO transient. The passive auxiliary feed water system (PAFS) attached to steam generators (SGs) of LWRs is one of the design options intended to completely replace the conventional active auxiliary feed water system to cope with an SBO transient. Two integral effect tests were carried out in the topic of prolonged SBO; active or passive secondary cooling was employed.

2.1. Test A1 series

In the A1.1 test, a prolonged SBO transient was simulated with two temporal phases: phase (I) for a conservative SBO transient without supply of auxiliary feed water, and phase (II) for asymmetric cooling through the delayed supply of auxiliary feed water only to one steam generator. The secondary side of SGs became empty, resulting from the inventory discharge through the cyclic opening and closing of the main steam safety valves (MSSVs) during the initial period of the transient. After the secondary side of the SGs became dried out, the primary system pressure started to increase due to a degradation of the heat removal capacity of the SGs. Periodic discharge of the primary inventory through the pilot operated safety relief valve (POSRV) resulted in core uncovery, and an excursion of the heater rod surface temperature eventually occurred. In order to simulate delayed supply of auxiliary feed water as an accident management (AM) measure, the auxiliary feed water was supplied when the maximum heater rod surface temperature in the core reached 450^{o}C. After the primary inventory loss started through the POSRV, the natural circulation flow became degraded. Depending on the heat removal capacity of the SGs, the natural circulation flow characteristics showed different trends in the primary loops. Similar to the behaviour of the system pressure, the coolant temperatures in the primary loops increased during the temporal phase (I) and then started to decrease after the supply of auxiliary feed water during the temporal phase (II). An asymmetric cooling pattern was clearly observed during the period when the auxiliary feed water was supplied.

The target scenario for the A1.2 test was a prolonged SBO with asymmetric secondary cooling through the supply of passive auxiliary feed water only to steam generator 2 (SG-2). The test objective was to investigate the primary cool-down performance by asymmetric passive secondary cooling as an accident mitigation measure. In this case, the PAFS was utilised instead of the turbine-driven auxiliary feed water. When the collapsed water level of the secondary side in SG-2 reached a 25% in a wide range scale, PAFS started to operate. PAFS played a key role in cooling down the primary system by the heat transfer and the natural circulation at the passive condensation heat exchanger (PCHX) and the passive condensation cooling tank (PCCT) water pool. During the whole test period, no flow instability in the PAFS loop was observed. With the actuation of PAFS, the fluid temperatures at the core inlet and outlet started to decrease without any excursion of the maximum heater surface temperature in the core. Asymmetric heat removal through the SGs resulted in different natural circulation characteristics in the primary loops. Contrary to cold leg-2A and -2B, degradation of natural circulation flow was clearly observed in cold leg-1A and -1B. The pressure and temperature of the reactor coolant system (RCS) continuously decreased during the heat removal by the PAFS operation, which indicates that PAFS can supply the auxiliary feed water to the steam generator and efficiently remove the core decay heat without any active system. This integral effect test data of A1.2 test can be used to evaluate the prediction capability of existing safety analysis codes and identify any code deficiency for an SBO simulation with an operation of a passive system such as PAFS.

In summary, the typical events of scenario expected to take place in the prolonged SBO accident were well reproduced in the tests and an asymmetric cooling pattern was clearly observed during the period when the auxiliary feed water was supplied. In the case of passive auxiliary feed water supply, it turned out that the PAFS is very effective to remove the decay core power. There was neither excursion of the heater rod surface temperature nor flow instability in the PAFS loop. In particular, the observed thermal-hydraulic behaviour during the PAFS operation attracted great interests from the project participants. The ATLAS Project provided an opportunity to investigate the efficacy of different modelling approaches concerning mixing phenomena in large pools of water. It was found that simplified one-dimensional modelling of these large open pools can indirectly approximate the mixing behaviour in such pools in an adequate way for the transients investigated. Across a variety of thermal-hydraulic (TH) system codes, thermal stratification was a parameter which proved difficult to predict with a high degree of accuracy. More thermal stratification was predicted in code calculations than the data. Thermal stratification, however, proved secondary to other factors for influencing prediction of key figures of merit.

Heat loss in an integral effect test facility is inevitable due to a larger ratio of the surface to the fluid volume in the reduced-scale components. The heat loss to the atmosphere reduces the conservatism of the experimental result for an accident simulation. So that, modelling the test facility with a system analysis code needs to realistically consider the effect of the heat loss in the RCS. In particular, the code calculation result for a long transient such as the station blackout scenario can be highly affected by modelling of the heat loss in the test facility. It was concluded that the heat loss should be carefully measured in the tests and the information is essential in modelling transient scenarios.

2.2. Test A2 series

A small break loss of coolant accident (SBLOCA) during SBO such as a reactor coolant pump (RCP) seal failure and a steam generator tube rupture (SGTR) has also been recognised as one of the risk-contributing sequences. Two integral effect tests were thus carried out in the subject of an SBLOCA during SBO.

In the A2.1 test, a prolonged SBO together with the two RCP seal failures at loop-2 assuming asymmetric secondary cooling by PAFS only to SG-2 was simulated. The test objective was to investigate the primary cool-down performance by asymmetric passive secondary cooling as an accident mitigation measure when an SBO is combined with a loss-of-coolant accident (LOCA). In particular, conservative assumptions were applied in the test specification to see the effectiveness of the heat removal capability of PAFS; PAFS was activated with a delay at lower SG secondary water level (10%) than the normal set value (25%). During the transient, collapsed water levels of the reactor pressure vessel as well as the U-tubes and natural circulation characteristic in the primary and secondary loops were investigated. It was found that the coolant inventory of the primary side started to decrease from the pressuriser at first. After the depletion of the coolant inventory in the pressuriser, the upper head inventory of the reactor pressure vessel was reduced, and then the water levels in the U-tubes of the SGs decreased. The U-tube water levels of SG-1 were depleted faster than those of SG-2 resulting from the asymmetric secondary cooling by PAFS. With the depletion of the water level in the U-tube of SG-1, natural circulation of the loop-1 was terminated. After that, the whole primary system was cooled by natural circulation through loop-2. As the water levels in the U-tubes of SG-2 decreased, however, the cooling performance of natural circulation became degraded. With a continuous discharge of steam through the turbine bypass system, the coolant inventory of the secondary side of SGs was reduced until the actuation set point of PAFS. After the depletion of the secondary side water level of SG-1, the temperatures in the secondary side of SG-1 showed a slow decrease due to the structural heat loss. However, for SG-2, the secondary side water level was maintained at approximately 1.7 m, enough to induce natural circulation cooling through PAFS. It turned out that PAFS played a very effective role in cooling down the primary system by the heat transfer and natural circulation through the U-tubes. In spite of the reduced water level of the SG U-tubes, the heat removal capability through PAFS was maintained until the water level of U-tubes of SG-2 dropped too much.

In the A2.2 test, a single tube rupture at SG-1 during an SBO was simulated when the core water level fell below the top of the active core, and the auxiliary feed water was supplied to SG-2 when the maximum heater rod surface temperature reached 450°C. Following the SBO initiation, an isolation of the SGs resulted in the cyclic opening and closing of the MSSVs. The coolant discharge from the secondary side of the SGs through the MSSVs played a key role in removing the decay heat by natural circulation. After the secondary side of the SGs became empty, the primary system pressure started to increase because of the degradation of the heat removal capacity of the SGs. Periodic discharge of the primary system coolant through the POSRV resulted in core uncovery and a subsequent SGTR in SG-1. Due to the small size of the tube rupture, the occurrence of SGTR did not reduce the primary system pressure before the injection of the auxiliary feed water. Delayed supply of the auxiliary feed water after an excursion of the heater surface temperature successfully cooled the primary system until the end of the transient. The natural circulation flow characteristics in the primary system showed an asymmetric behaviour depending on the heat removal rate of the SGs. There was no remarkable effect of the change of the pressuriser connection to the hot leg in loop-1. It was found that the injection of the auxiliary feed water to the SG secondary side by either active or passive means is the most essential function to remove the decay core power through the primary to secondary heat transfer.

2.3. Test A3 series

A TLOFW accident has been considered to be important since it was identified as one of the major contributors to the severe core damage frequency in WASH-1400. An integral effect test, A3.1, was performed on this topic by assuming the additional failure of safety injection. The target scenario for the A3.1 test was a TLOFW with additional failures in order to simulate a combined accident that is typical of a beyond DBA with multiple failures. The main purpose of the A3.1 test was to see the effect of reduced safety injection pump (SIP) and POSRV flow on RCS cooling capability during feed & bleed operation. It was known from safety analysis that both core water level and peak cladding temperature (PCT) depend on the combination of the number of available SIPs and POSRVs as well as the operator's grace time before bleeding. In the present test, bleeding was simulated with a partially stuck-open (50%) POSRV taking into account an operator's waiting time of 15 minutes after the first opening of the POSRV. The bleeding increases depressurisation of the primary system. Also half capacity of SIP, only two SIPs out of four SIPs, was assumed available as a feed operation.

The scenario of A3.1 test consists of two temporal phases. The first one is a SG dry-out phase similar to an SBO accident and the other is a feed and bleed operation phase for cooling down the RCS. The first phase was initiated by terminating the main and auxiliary feed water supply at hot full power condition and both SGs dried out due to an inventory loss through actuation of the MSSVs. After the SG dry-out, the pressuriser water level and the primary system pressure continuously increased due to a loss of heat sink and finally the POSRV was opened at 17.03 mega Pascal (MPa). A large coolant inventory loss of the primary system through the POSRV during a feed and bleed operation resulted in a reduction of the core collapsed level but the minimum core water level was still above the top of the active core. As a result, the excursion of the heater rod surface temperature in the core was not observed. The second phase was initiated by maintaining the POSRV open (50% capacity) to simulate a bleed operation with 637 seconds delay (in ATLAS time) after the first POSRV opening. As the primary system pressure decreased to the SIP actuation point (12.47 MPa), the emergency core cooling (ECC) water started to be injected to the primary system via the direct vessel injection (DVI) nozzles by the SIP. Afterward, the feed (SIP) and bleed (POSRV) operation was maintained during the remaining test period. Delayed feed and bleed operation for the primary system was found to be still effective to cool down the RCS under the present test condition. Asymmetric loop behaviour caused by feed and bleed operation was identified. Thus, it was experimentally observed that the core water level was recovered without any excursion of heater rod surface temperature and the primary system conditions successfully reached the Shutdown Cooling System (SCS) operating condition.

2.4. Test A4 and A5 series

An intermediate break LOCA (IBLOCA) has been recognised very important topic in term of risk-informed regulation (RIR). There is a widespread opinion that the frequency of double-ended guillotine break (DEGB) of primary coolant circuit piping such as hot and cold legs of pressurised water reactor (PWR) is quite low. Therefore consideration of rupture of intermediate-size pipe is becoming relatively more important than ever in RIR-relevant safety analyses. Although there are available experimental data for the intermediate break LOCA, it is relatively quite limited. Thus, two integral effects tests for 13% and 17% IBLOCAs were defined and carried out.

In the A4.1 test, as a counterpart test for the Large Scale Test Facility (LSTF) 17% cold leg break IBLOCA, a single failure of the ECC injection (high pressure injection and low pressure injection) and a total failure of the auxiliary feed water to the secondary system were assumed. Following the break initiation, a rapid depressurisation and a blow down in the reactor pressure vessel led to an excursion of the heater rod surface temperature in the core. The maximum heater rod surface temperature was observed as 641^{o}C. The ECC water injection from the accumulator contributed to the effective recovery of the coolant inventory in the reactor pressure vessel and the subsequent quench of the core. After termination of the accumulator injection, no more excursion of the heater rod surface temperature was observed during the remaining transient. The LSTF test data were scaled down according to the scaling methodology and directly compared to the ATLAS test result. It showed that overall sequence of major events and thermal-hydraulic phenomena including transient behaviour of the system pressure, the temperature, and the break flow were reasonably reproduced in the ATLAS test. The maximum heater rod surface temperature in the ATLAS test was lower than that in the LSTF test. The natural circulation flow characteristics in the primary system showed an asymmetric behaviour and it affected the amount of the reverse heat transfer from the steam generator. From the multi-dimensional measurement of the fluid temperature distribution, thermal mixing behaviour in the RCS could be quantitatively investigated especially for the steam generator outlet plenum and the down comer.

In the A5.2 test, as a counterpart test for the LSTF 13% cold leg break IBLOCA, a full injection of the ECC and a total failure of the auxiliary feed water to the secondary system were assumed. The test result showed that the ECC water injection from the accumulator contributed to the effective recovery of the coolant inventory in the reactor pressure vessel and the subsequent quench of the core. Comparison with the LSTF test data showed that overall sequence of major events and thermal-hydraulic phenomena were reasonably reproduced in the ATLAS counterpart test. After initiation of the break, a rapid depressurisation and a blow down in the reactor pressure vessel led to an excursion of the heater rod surface temperature in the core by 589^{o}C. The overall sequence was similar to that in the case of A4.1. The LSTF test resulted in an earlier occurrence of loop seal clearing compared with the ATLAS counterpart test, so that a lower level of the reactor core and a higher maximum heater rod temperature were observed in the A5.2 test compared to the LSTF test.

It was found from the two counterpart tests for IBLOCA that the overall sequence of major events and thermal-hydraulic phenomena including transient behaviour of the system pressure, the temperature, and the break flow were reasonably reproduced in the ATLAS test. However, the maximum heater rod surface temperature in the ATLAS test was lower than that in the LSTF test in the case of A4.1 (17% IBLOCA). It was concluded from analysis that the discrepancy may be attributed to the differences of the loop seal clearing characteristics, the core water level, and the uncovered position in the active core region. In the case of A5.2 (13% IBLOCA), the ATLAS counterpart test resulted in a delayed occurrence of loop seal clearing compared with the LSTF test. And a lower level of the reactor core and a higher maximum heater rod temperature were observed in the A5.2 test compared to the LSTF test. The present counterpart data set together with the LSTF data set would be a valuable database to address the ‘cliff-edge effect’ during an IBLOCA scenario. Further detailed post-test analysis is highly recommended.

2.5. Benchmark test, A5.1

Finally, properly accounting for scaling is one of the remaining major safety issues under debate between regulatory authorities and utilities. The scaling inherent in a certain facility needs to be justified before its data is used for a safety analysis. It was agreed at the Project Review Group (PRG) meeting that ATLAS can be utilised to reproduce one of the scenarios of LSTF in order to address scaling issues. The A5.1 test was defined as a counterpart test with respect to the LSTF SB-CL-32 test which simulated a 1% horizontal SBLOCA at the cold leg with cold leg injection of the ECC water and AM action. The test objective was to investigate the thermal-hydraulic phenomena during a cold leg SBLOCA such as core heat-up, loop seal clearing (LSC), and the effect of AM actions.

This test was selected as a benchmark exercise during the project period. A total of 15 participants adopting eight different codes participated in the benchmark. This benchmark exercise consisted of pre-test and post-test analysis and it was coordinated by NINE (Nuclear and Industrial Engineering) as an in-kind contribution to this project. The global average accuracy of each post-test calculation has been evaluated by FFTBM (Fast Fourier Transform Based Method) application. Almost all of the participants’ predictions were categorised in the “good prediction” range, with an overall reduction of a few percent of the discriminant parameter. In addition to this, all predictions of the average accuracy of the primary side pressure fell within the “good prediction” range. Detailed information is available from the separate benchmark report.

The scaling methodology used to determine the counterpart test conditions against the SB-CL-32 scenario was developed by comparing the geometrical differences between two facilities. This scaling methodology was also applied to set-up the test conditions of the A4.1 and A5.2 tests which were performed as counterpart tests of the LSTF IBLOCA tests. The initial steady state conditions were achieved at a scaled power based on the core power that was supplied in the SB-CL-32 test. Reactor scram occurred when the primary pressure decreased to 12.97 MPa after break initiation. The initiation of the AM action started at 456.7 seconds after the break valve opening and it was taken by depressurising the secondary side (SG-1 and SG-2, simultaneously). Two flow control valves (FCVs) were controlled by manual operation, and the depressurisation rates were maintained at 328 K/hr throughout the whole test period. The auxiliary feed water injection was actuated with some delay after the initiation of the AM action and had a time difference between the loops. The injection from the accumulators (ACCs) was actuated when the primary pressure decreased to 4.51 MPa. The ACC injection was set to be terminated when the total inventory, which is the same amount of scaled mass that was injected in the SB-CL-32 test, was supplied. The low pressure injection (LPI) signal was actuated when the primary pressure decreased to 1.21 MPa. The test ended 20 minutes after the LPI injection.

The targeted scenario was successfully simulated using the ATLAS facility, and the major thermal-hydraulic phenomena that can occur in a SBLOCA were observed. With the accident management action, the primary system pressure decreased following the depressurisation of steam generator secondary side and the overall system showed stable cool-down behaviour after the initiation of the AM action. LSC phenomenon occurred faster in the A5.1 test than observed in the LSTF SB-CL-32 test. This difference comes from the different intermediate leg design between LSTF and ATLAS. This difference affected the pressure difference between the upper head and down comer region of the reactor pressure vessel. An excursion of the heater rod surface temperature was not observed. Most of the active core of the ATLAS was submerged in the water during the test period because the height from the active core top to the bottom of reactor pressure vessel of the ATLAS was lower than that of the LSTF. The main differences in the LSC and the excursion of the heater rod surface temperature came from the different design of the intermediated-leg and the location of the active core. These differences are mainly attributed to a different design of the prototype nuclear power plants for each facility.

2.6. PKL3-ATLAS Joint Workshop

As the NEA Primary Coolant Loop Test Facility (PKL3) project was under way from 2012, due to the links between the two programs, the Management Boards of both projects decided in 2015 to organise a joint workshop of related analytical activities. The joint workshop of the PKL3 and ATLAS Projects took place in Lucca (Italy) at the Chamber of Commerce premise, from 13 to 15 April 2016. The workshop attracted 60 participants from 16 countries. It included 34 presentations covering the general overview of both programs, the analyses of the benchmarks exercises organised within the projects (on PKL3, Rossendorf Coolant Mixing Model [ROCOM] and ATLAS particular tests), and some analyses related to other PKL3 and ATLAS tests including application to reactor cases. The conclusion of this workshop prepared by the Session Chair together with the final summary integration report of these two projects are being issued as public CSNI report.

3. Conclusions and Recommendations

An NEA joint project utilising the Advanced Thermal-Hydraulic Test Loop for Accident Simulation (ATLAS) integral effect test facility has successfully progressed from April 2014 to March 2017. In this project, a total of eight integral effect tests covering five different topics were carried out and 27 organisations from 16 countries participated: Belgium (BelV, Tractabel), China (State Nuclear Power Software Development Centre [SNPSDC], China Nuclear Power Technology Research Institute [CNPRI]), France (Commissariat à l'énergie atomique et aux énergies alternatives [CEA], Électricité de France [EDF]), Finland (Radiation and Nuclear Safety Authority [STUK], Technical Research Centre of Finland Ltd [VTT]), Germany (GRS), Hungary (Hungarian Academy of Sciences Centre for Energy Research [MTA-EK]), India (Bhabha Atomic Research Centre [BARC]), Italy (Nuclear and Industrial Engineering [NINE]), Japan (The Nuclear Regulation Authority [NRA]), the Russian Federation (State Atomic Energy Corporation [Rosatom], Gidropress, Nuclear Safety Institute of the Russia Academy of Sciences [IBRAE-RAS], A.P. Aleksandrov Scientific Research Technological Institute [NITI], OAO I. I. Afrikantov OKB Mechanical Engineering [OKBM]), Spain (Spanish Nuclear Safety Council [CSN]), Sweden (The Swedish Radiation Safety Authority [SSM]), Switzerland (The Paul Scherrer Institute [PSI]), United Arab Emirates (FANR), United States (U.S. Nuclear Regulatory Commission [U.S.NRC]), and Korea (Korea Atomic Energy Research Institute [KAERI], Korea Institute of Nuclear Safety [KINS], Korea Hydro & Nuclear Power Central Research Institute [KHNP-CRI], Korea Electric Power Corporation Engineering & Construction Company, Inc. [KEPCO E&C], Korea Electric Power Corporation Nuclear Fuel [KEPCO-NF]). Italy joined the NEA-ATLAS Project as an in-kind contributor and co-ordinated a benchmark problem for the counterpart test against the Large Scale Test Facility (LSTF). Utilising the database established by the Institution of Engineering and Technology [IET, simulation models and methods for following complex phenomena of high safety relevance to thermal-hydraulic transients in design-basis accident (DBAs) and beyond-DBA was validated.

The present NEA-ATLAS joint project is aimed at safety of operating nuclear power plants in connection to a station blackout (SBO) accident. An SBO accident itself and multiple SBO-induced accidents, for instance, an SBO combined with a loss-of-coolant accident (LOCA), were investigated in a systematic manner. However, there are still remaining working areas where safety can be improved in order to further minimise the risk for severe accidents. One of the more interesting concepts is passive systems which can either replace or complement an active safety system. The Korean industry has developed the passive auxiliary feed water system (PAFS), completely replacing the existing active auxiliary feed water system and is going to apply it in the next generation light water reactors (LWRs) and Advanced Power Reactor Plus (APR+). However, performance and reliability of passive systems still needs to be validated further due to the inherent very low driving force and the very complex related phenomena such as multi-dimensional flow or mixing, asymmetric behaviour, flow oscillation and instability. In particular, there is much less overall experience in operation of a passive system