

Summary Report of the NEA ATLAS-2 Joint Project

**NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS**

Summary Report of the NEA ATLAS-2 Joint Project

JT03489522

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 38 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, Colombia, Costa Rica, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Korea, Latvia, Lithuania, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 34 countries: Argentina, Australia, Austria, Belgium, Bulgaria, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Romania, Russia, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission and the International Atomic Energy Agency also take part in the work of the Agency.

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.

This document, as well as any data and map included herein, are without prejudice to the status of or sovereignty over any territory, to the delimitation of international frontiers and boundaries and to the name of any territory, city or area.

Corrigenda to OECD publications may be found online at: www.oecd.org/about/publishing/corrigenda.htm.

© OECD 2022

You can copy, download or print OECD content for your own use, and you can include excerpts from OECD publications, databases and multimedia products in your own documents, presentations, blogs, websites and teaching materials, provided that suitable acknowledgement of the OECD as source and copyright owner is given. All requests for public or commercial use and translation rights should be submitted to neapub@oecd-nea.org. Requests for permission to photocopy portions of this material for public or commercial use shall be addressed directly to the Copyright Clearance Center (CCC) at info@copyright.com or the Centre français d'exploitation du droit de copie (CFC) contact@cfcopies.com.

Committee on the Safety of Nuclear Installations (CSNI)

The Committee on the Safety of Nuclear Installations (CSNI) is responsible for 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 to its activities from their respective backgrounds in research, development and engineering. 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 considers the safety implications of scientific and technical developments of future reactor technologies and designs as well as human and organisational research activities and technical developments that affect nuclear safety.

Table of contents

Executive summary	5
1. Introduction	8
2. Summary of test results.....	9
2.1 Test B1 series	9
2.2 Test B2 series	10
2.3 Test B3 series	12
2.4 Test B4 series	13
2.5 Test B5 series: Counterpart test for benchmark analysis.....	15
2.6 PKL-4-ATLAS-2 joint workshop.....	16
3. Conclusions and recommendations.....	18

Figures

Figure 2.1. System pressures and core temperatures in the B1.1 test	10
Figure 2.2. System pressures and segmented core levels in the B2.1 test	11
Figure 2.3. System pressures and segmented core levels in the B2.2 test	12
Figure 2.4. Comparison of maximum core temperatures in the B3.1 and B3.2 tests	13
Figure 2.5. Schematic diagrams for simulating an SLB and a SGTR in the B4.1 test	14
Figure 2.6. Schematic diagram for simulating a mid-loop operation and the maximum core heater rod surface temperatures in the B4.2 test	14
Figure 2.7. Comparison of the system pressures and the PCT in the B5.1 and the SB-PV-07 tests	16

List of abbreviations and acronyms

ADV	Atmospheric dump valve
ATLAS	Advanced Thermal-hydraulic Test Loop for Accident Simulation (KAERI)
BDBA	Beyond design-basis accident
CEA	French Alternative Energies and Atomic Energy Commission (Commissariat à l'énergie atomique et aux énergies alternatives, France)
CET	Core exit temperature
CNPRI	China Nuclear Power Technology Research Industry
CSN	Spanish Nuclear Safety Council (Consejo de Seguridad Nuclear)
CSNI	Committee on the Safety of Nuclear Installations (NEA)
CUBE	Containment Utility for Best-Estimate Evaluation
DBA	Design-basis accident
DEC	Design extension conditions
DVI	Direct vessel injection
FANR	Federal Authority for Nuclear Regulation (United Arab Emirates)
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (Germany)
IBLOCA	Intermediate break loss-of-coolant accident
IET	Integral effect test
IRWST	In-containment refuelling water storage tank
JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
KHNP-CRI	Korea Hydro and Nuclear Power-Central Research Institute
KINS	Korea Institute of Nuclear Safety
LBLOCA	Large break loss-of-coolant accident
LPP	Low pressuriser pressure
LSTF	Large scale test facility (JAEA)
LWR	Light water reactor
MSSV	Main steam safety valves
NEA	Nuclear Energy Agency
NPIC	Nuclear Power Institute of China
OA	Operating Agent

PAFS	Passive auxiliary feedwater system
PCT	Peak cladding temperature
PECCS	Passive emergency core cooling system
POSRV	Pilot operated safety relief valve
PRA	Probabilistic risk assessment
PRG	Project Review Group
PSI	Paul Scherrer Institute (Switzerland)
PWR	Pressurised water reactor
RCS	Reactor coolant system
RHRS	Residual heat removal system
RPV	Reactor pressure vessel
SBLOCA	Small break loss-of-coolant accident
SGTR	Steam generator tube rupture
SIP	Safety injection pump
SIT	Safety injection tank
SLB	Steam line break
SPICRI	State Power Investment Corporation Research Institute (China)
ÚJV Řež	Nuclear Research Institute Rez (Czech Republic)
US NRC	United States Nuclear Regulatory Commission

Executive summary

The Nuclear Energy Agency (NEA) Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS) phase 2 project (hereafter, NEA ATLAS-2) is an NEA-supported joint project started in October 2017 for a three-year period. In the framework of the NEA ATLAS-2 project, a total of eight integral effect tests (IETs) in five different topics were performed with the ATLAS facility.

Major findings of the NEA ATLAS-2 project can be summarised as follows;

- During a two-inch cold leg small break loss-of-coolant accident (SBLOCA) transient with total failure of safety injection pump, the reactor core was quenched after an operation of the passive auxiliary feedwater system (PAFS).
- Passive core makeup systems are very effective in removing the core decay heat during a Station Blackout (SBO) and SBLOCA transient.
- The break location and number of available safety injection tanks can affect the core heat-up and asymmetric temperature distribution during an intermediate break loss-of-coolant accident (IBLOCA) transient.
- The reactor coolant system can be successfully cooled down with the proper operation of safety systems against a multiple failure accident.
- A counterpart test for an SBLOCA was performed and the scaling issue was addressed. The differences between the two tests can be attributed to the different design of prototype nuclear power plant for each facility.

Utilising the established integral effect test (IET) database, simulation models and methods for complex phenomena of high safety relevance to thermal-hydraulic transients in design-basis accident (DBA) and beyond-DBA were validated. The project participants carried out very active analyses with their analysis codes such as RELAP, TRACE, CATHARE, ATHLET, MARS and SPACE.

The present NEA ATLAS-2 project aims to enhance the safety of operating nuclear power plants by simulating the various accident transients in connection with the safety analysis technology. Considering there remain working areas where safety analysis technology can be improved and eventually prevent a severe accident in any case, the ATLAS follow-up project (NEA ATLAS-3) is planned to further address the safety relevant issues.

1. Introduction

Within the context of the Nuclear Energy Agency (NEA) Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS) project, from April 2014 to March 2017 a series of tests were performed to resolve key thermal-hydraulic safety issues related to multiple high risk failures highlighted from the Fukushima Daiichi accident, by utilising a thermal-hydraulic integral effect test (IET) facility of ATLAS. Notwithstanding the distinguished achievement of the NEA ATLAS project, a general consensus between the project partners was reached to continue the second phase of the project with the aim of enhancing the nuclear safety analysis technology and improving the best guidelines for accident management. In particular, the NEA ATLAS phase 2 project (hereafter, NEA ATLAS-2) focused on the validation of simulation models and methods for complex phenomena of high safety relevance to thermal-hydraulic transients in a design-basis accident (DBA) and beyond-DBA (BDBA).

ATLAS is an integral effect facility simulating an APR1400 (advanced power reactor 1 400 MWe) with a 1/2 reduced height. The scaling factor of the fluid volume is 1/288. Under the framework of the NEA ATLAS-2 project, a total of eight integral effect tests in five different topics were performed at the ATLAS facility.

The NEA ATLAS-2 project started in October 2017 for a three-year period. Due to the worldwide crisis resulting from the COVID-19 pandemic, the project was extended to the end of 2020. Eighteen organisations from eleven countries participated in the project as follows: Belgium (Bel V, ENGIE), China (State Power Investment Corporation Research Institute [SPICRI], China Nuclear Power Technology Research Industry [CNPRI], Nuclear Power Institute of China [NPIC]), the Czech Republic (Nuclear Research Institute Rez [ÚJV Řež]), France (French Alternative Energies and Atomic Energy Commission [CEA], EDF), Germany (Gesellschaft für Anlagen- und Reaktorsicherheit, [GRS]), Japan (Japan Atomic Energy Agency [JAEA]), Korea (Korea Atomic Energy Research Institute [KAERI], Korea Institute of Nuclear Safety [KINS], Korea Hydro and Nuclear Power-Central Research Institute [KHNP-CRI], KEPCO E&C), Spain (Spanish Nuclear Safety Council [CSN]), Switzerland (Paul Scherrer Institute [PSI]), the United Arab Emirates (Federal Authority for Nuclear Regulation [FANR]) and the United States (United States Nuclear Regulatory Commission [US NRC]). Japan joined the NEA ATLAS-2 project as an in-kind contributor providing the experimental data for the counterpart test against the large scale test facility (LSTF). The Operating Agent (OA) established a national consortium together with Korean nuclear players and contributed to this project by performing pre- and post-test calculations.

2. Summary of test results

The key outline of the tests performed in the Nuclear Energy Agency Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS) phase 2 project (NEA ATLAS-2) is as follows:

- B1: small break loss-of-coolant accident (SBLOCA) with total failure of safety injection pump (SIP) under passive auxiliary feedwater system (PAFS) operation (one test);
- B2: performance of passive core makeup during a station blackout (SBO) and an SBLOCA (two tests);
- B3: intermediate break loss-of-coolant accident (IBLOCA) of pressuriser surge line and direct vessel injection (DVI) line break (two tests);
- B4: design extension conditions of a steam line break (SLB) accompanied by a steam generator tube rupture (SGTR) and shutdown coolability without residual heat removal system (two tests);
- B5: counterpart test for SBLOCA of 1% reactor pressure vessel (RPV) top break (one test).

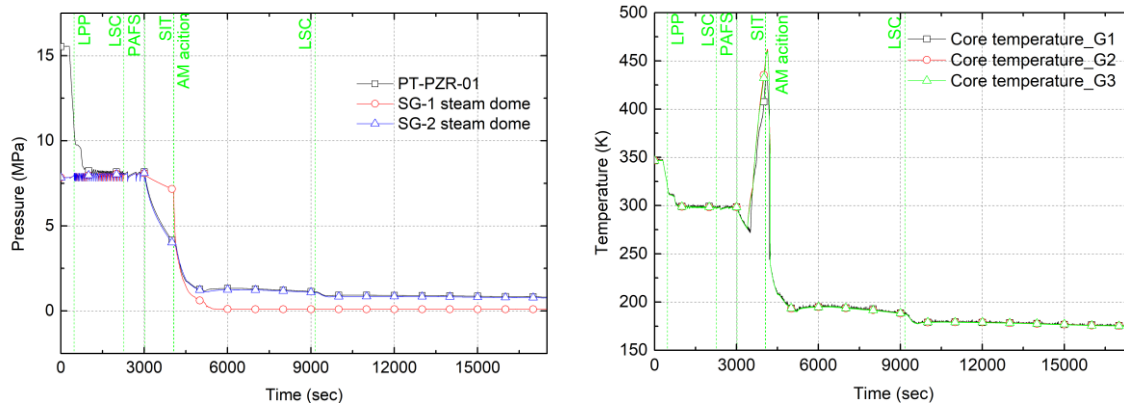
2.1 Test B1 series

An SBLOCA at a cold leg is one of the most important design-basis accidents (DBAs). In order to mitigate the consequences of an SBLOCA transient, pertinent safety systems should be utilised. Since the Fukushima Daiichi accident, various passive safety systems have been proposed to improve the safety and reliability of an ultimate heat removal system without any operator action during DBA and beyond-DBA (BDBA) transients. A PAFS is one of the advanced safety features that is intended to replace a conventional active auxiliary feedwater system. The driving force for a PAFS is a natural convection mechanism, i.e. condensing steam in nearly-horizontal U-tubes submerged inside a large water pool. The experimental data with a single train of PAFS can be used to validate a prediction capability of safety analysis codes for the natural circulation phenomenon and asymmetric cooling effect. One test was performed to simulate an SBLOCA at a cold leg by utilising a single train of PAFS.

In the B1.1 test, a two-inch cold leg SBLOCA was simulated with total failure of safety injection pump under an operation of PAFS. A single train of PAFS was connected to the steam generator number two (SG-2) of ATLAS. When the collapsed water level of the secondary side in SG-2 reached 25% in a wide range scale, PAFS started to operate. In the B1.1 test, an accident management action was simulated by way of the secondary side depressurisation of steam generator number 1 (SG-1). When the maximum heater rod surface temperature in the core reached 450°C, the accident management action was initiated by fully opening an atmospheric dump valve (ADV) of SG-1. After PAFS actuation, the secondary side water levels of steam generator maintained stable values. After accident management action, however, the secondary side water level and the

secondary system pressure of SG-1 sharply decreased because the steam was vented out. Flow rates of loop-2 were larger than those of loop-1, which could be attributed to the asymmetric cooling by a single train of PAFS operation. The B1.1 test result showed that during a two-inch cold leg break SBLOCA with total failure of safety injection pump, the reactor core was quenched after an operation of PAFS and accident management action. The asymmetric cooling condition induced asymmetric thermal-hydraulic behaviour in the present test.

Figure 2.1. System pressures and core temperatures in the B1.1 test



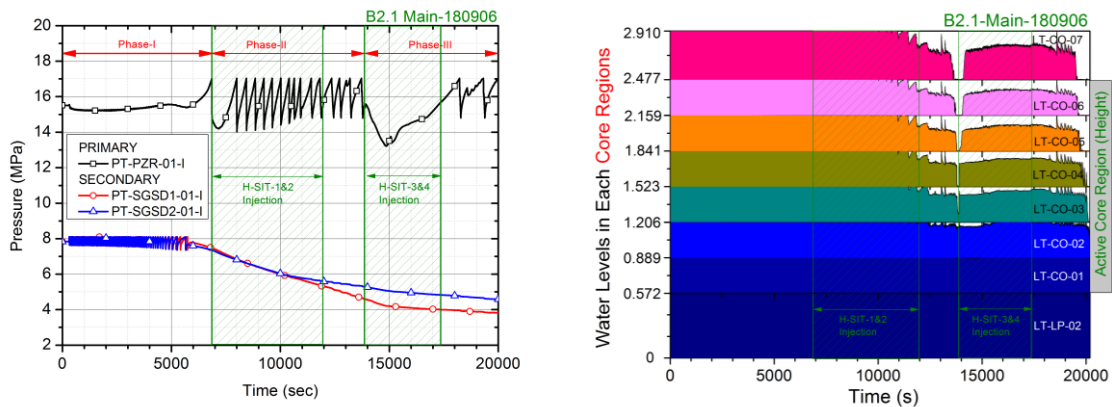
Source: NEA, 2021.

2.2 Test B2 series

Following the Fukushima Daiichi Nuclear Power Plant accident, demand for safety enhancement in nuclear power plants has increased. The accident showed that to prevent core meltdown, the core makeup at high pressure of the reactor coolant system (RCS) is crucial, and that even in an SBO situation, the core makeup water must be supplied efficiently. The concept of a hybrid safety injection tank (H-SIT) is a passive safety injection system that allows high-pressure core makeup over the operating pressure of a light water reactor (LWR). The H-SIT can be pressurised equivalently to the RCS through a pipe connection between the H-SIT and the pressuriser, along with nitrogen charging, in which case the coolant can be injected by gravitational head between the RCS and the H-SIT. As a similar concept to the H-SIT, the passive emergency core cooling system (PECCS) can be pressurised equivalently to the RCS through a pipe connection between the safety injection tank (SIT) and a cold leg. It is worth investigating the thermal-hydraulic phenomena anticipated in these passive core makeup systems to produce clear knowledge of the actual phenomena and to provide the best guideline for accident management. Two tests were performed on the topic of the passive core makeup.

The target scenario for the B2.1 test was a prolonged SBO with operation of the H-SIT as a passive core makeup system. Pressure balance lines connecting the pressuriser to the H-SITs were in a keep-open status throughout the test period. In the B2.1 test, typical events of an SBO scenario were well reproduced.

Figure 2.2. System pressures and segmented core levels in the B2.1 test



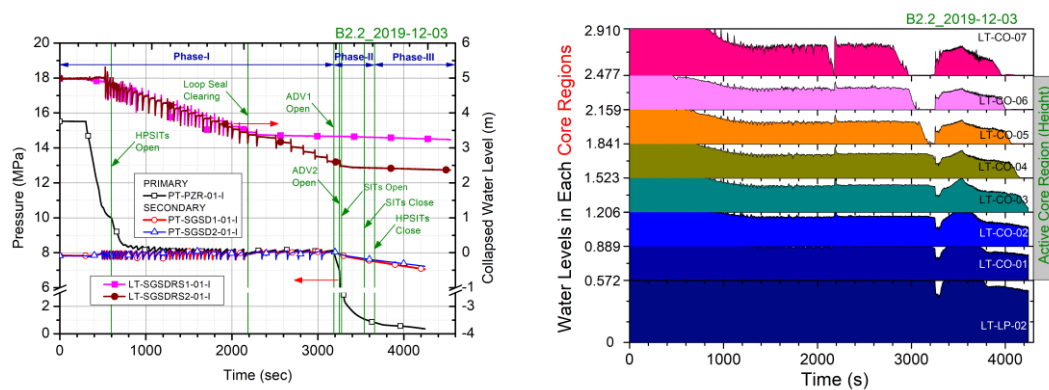
Source: NEA, 2021.

The secondary side of steam generators 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 steam generators became dried out, the primary system pressure started to increase due to a degradation of the heat removal capacity of the steam generators. Periodic discharge of the primary inventory through a pilot-operated safety relief valve (POSRV) of the pressuriser was observed. The hybrid safety injection tank number 1 (H-SIT-1) and hybrid safety injection tank number 2 (H-SIT-2) were activated to inject the coolant through the DVI nozzles with the first opening of a POSRV. The hybrid safety injection tank number 3 (H-SIT-3) and hybrid safety injection tank number 4 (H-SIT-4) were set to open when the maximum heater rod surface temperature in the core increased above 450°C. The core was effectively cooled by the safety injection flow from the H-SITs. Excursion of the heater rod surface temperature in the core was not observed during the safety injection from the H-SITs. The B2.1 test result shows that the H-SITs had an effective core cooling performance as a passive safety feature.

In the B2.2 test, a cold leg SBLOCA with an operation of a PECCS was simulated. The PECCS has two major functions to mitigate a DBA situation. The first one is automatic depressurisation of the primary system pressure through an automatic depressurisation valve on pressuriser. The second function is passive safety injection using two high-pressure safety injection tanks (HP-SITs) and two safety injection tanks (SITs). With the start of a break, the primary system pressure decreased to the low pressuriser pressure (LPP) set point, 10.7 MPa. The low pressuriser pressure signal isolated the secondary system. After the isolation, the secondary side water inventory continuously decreased to depletion with the open-close hysteresis of the MSSVs. High-pressure safety injection tank number 1 (HPSIT-1) and high-pressure safety injection tank number 3 (HPSIT-3) were activated when the pressuriser pressure decreased below 10.0 MPa. Continuous release of the primary system inventory induced an increase of cladding temperature in the core. Automatic depressurisation valves number one and two were actuated when the maximum heater rod surface temperature in the core increased above 380°C and 410°C, respectively. With safety injection from the HP-SITs and depressurisation through automatic depressurisation valves along with inventory depletion of the secondary side of the steam generators, the primary system pressure abruptly decreased below the activation set point of the SIT, 4.2 MPa. The safety injections from HP-SITs were not effectively injected during the early phase of the transient. Only after the opening of the automatic

depressurisation valves number one and number two was the safety injection water injected to the RCS, which resulted in an effective decrease of the maximum heater rod surface temperature in the core. After the termination of the safety injection from HP-SITs and SITs, the heater rod surface temperature in the core increased again above 450°C, which was the termination criterion of the present test. This could be attributed to exclusion of the long-term cooling system in prototypic PECCS. In fact, for the simplicity of test operation, the long-term cooling system from an in-containment refuelling water storage tank (IRWST) was not simulated in the present test.

Figure 2.3. System pressures and segmented core levels in the B2.2 test



Source: NEA, 2021.

From the test results of B2 series, it can be concluded that passive core makeup systems are very effective in removing the core decay heat during an SBO and an SBLOCA transient.

2.3 Test B3 series

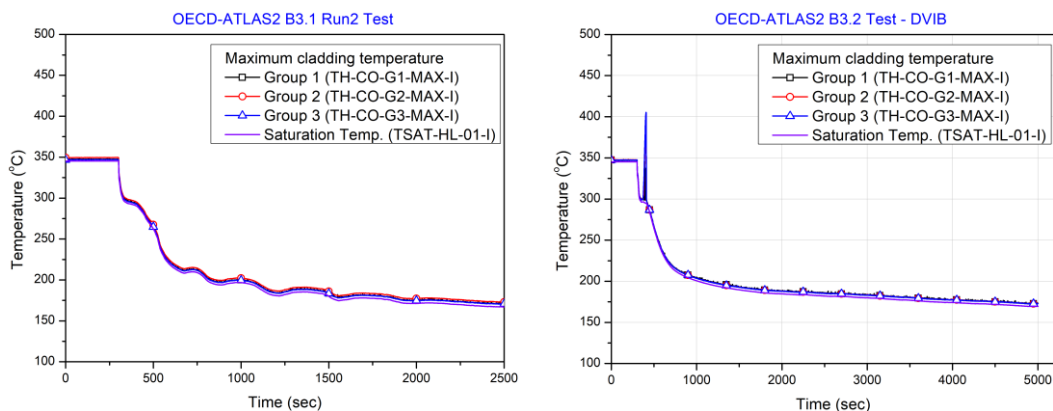
An IBLOCA has been recognised as one of the important topics in terms of risk-informed regulation. There is a widespread opinion that the frequency of a double-ended guillotine break of primary coolant circuit piping, such as the hot and cold legs of a pressurised water reactor (PWR), is quite low. Therefore, a rupture of an intermediate-size pipe is becoming relatively more important in risk-informed regulation. Although there is available experimental data for an IBLOCA, it is relatively limited. Thus, two tests for IBLOCA transients with a pressuriser surgeline break and a DVI line break were performed.

The target scenario for the B3.1 test is an IBLOCA with a pressuriser surgeline break, which corresponds to a 10-inch break in an APR1400. The test was composed of Run 1 and Run 2, according to the available number of SITs, with the aim of investigating asymmetric and multi-dimensional thermal-hydraulic phenomena. The pressuriser surgeline break induced a rapid depressurisation of the primary system and a blowdown of the coolant in the reactor pressure vessel (RPV). The safety injection from the SIPs and the SITs supplied sufficient safety injection water to the RCS and no excursion behaviour of the cladding temperature was observed in the core during the whole transient of Run 1 and Run 2 tests. A reverse heat transfer at the U-tube in the steam generators made a superheated condition of the cold leg flow. The intermittent safety injection from the SITs influenced a multi-dimensional temperature distribution in the downcomer of the RPV. The coolant in the lower downcomer was sufficiently mixed and there was no significant asymmetric behaviour of the coolant temperature at the lower plenum in both tests. In the

Run 2 test excluding one SIT, a different behaviour of the injection flow rate from SIT and the asymmetric temperature distribution were observed in the upper downcomer compared to the Run 1 test.

The B3.2 test was performed to simulate a DVI line break IBLOCA, which corresponds to an 8.5-inch break in APR1400. The DVI line break induced a steam pressure build-up in the core and an excursion behaviour of the cladding temperature. The minimum water level in the core was observed at the moment of the loop seal clearance. Since the break nozzle was located at the DVI line, the clearance of an upper downcomer could make an effective flow path of the steam towards the break. The B3.2 test result showed that the reactor core was quenched after the flow path of the steam towards the break was provided by the clearance of a loop seal and an upper downcomer. Compared to the B3.1 test result for a pressuriser surge line break, the B3.2 test indicated that the break location at the DVI line could significantly affect the behaviour of the core heat-up. While an excursion of the cladding temperature did not occur in the B3.1 test even with a larger break area than the DVI line, the simulation of the DVI line break scenario showed a core heat-up until the clearance of a loop seal and an upper downcomer.

Figure 2.4. Comparison of maximum core temperatures in the B3.1 and B3.2 tests

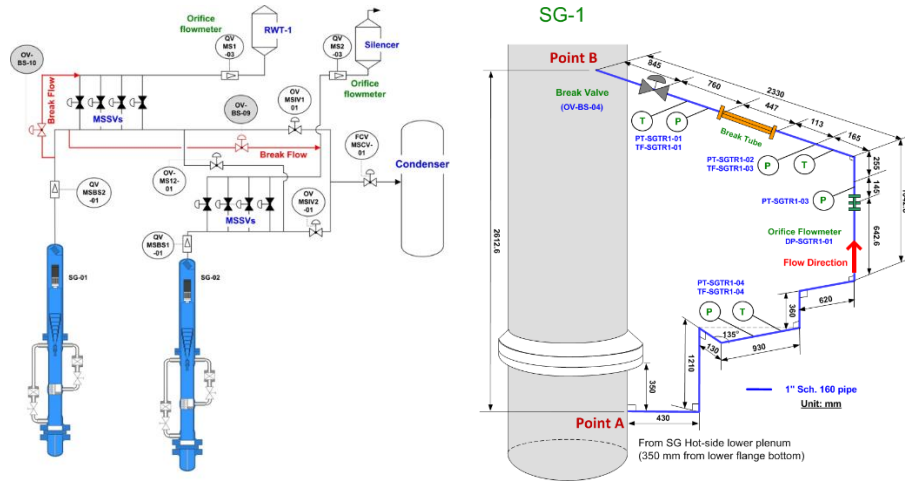


Source: NEA, 2021.

2.4 Test B4 series

Design extension conditions (DECs) such as an SBO, multiple failure accidents involving an SLB accompanied by an SGTR, and a shutdown coolability without a residual heat removal system (RHRS), which have not been seriously considered from a viewpoint of DBA, were incorporated in the test matrix of the NEA ATLAS-2 project. Specifically, various kinds of potential multiple failure accidents have attracted worldwide attention post-Fukushima. Two tests were performed in the field of DECs.

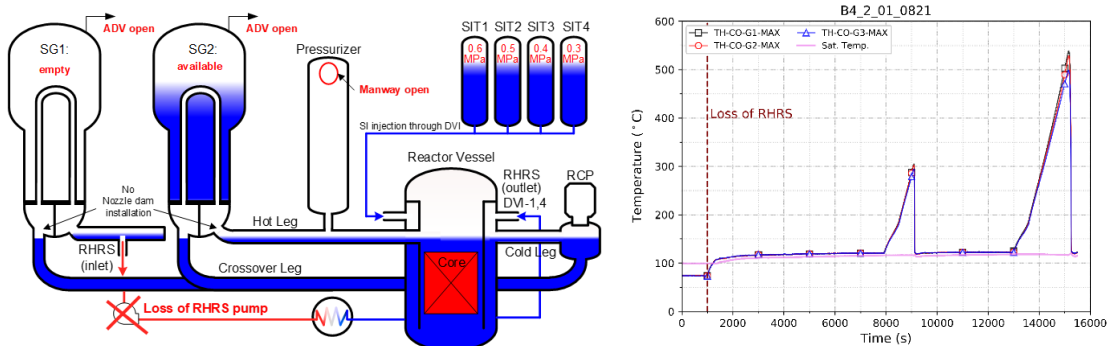
Figure 2.5. Schematic diagrams for simulating an SLB and a SGTR in the B4.1 test



Source: NEA, 2021.

The target scenario for the B4.1 test was a multiple failure accident of an SLB accompanied by an SGTR. The B4.1 test was started by opening two break valves at the main steam line. The secondary system pressure of the SG-1 decreased rapidly and the reactor scram signal was actuated when the secondary system pressure of the SG-1 reached 6.11 MPa. When the wide range level of SG-1 decreased to 0.1 m, an SGTR was initiated. Due to the inventory loss by the SGTR, the primary system pressure decreased. When the primary system pressure reached 10.72 MPa, the SIP actuation signal was activated. After that, the RCS cooled down with recovery of the collapsed water level of secondary side in SG-1. Finally, the test was terminated upon the operator’s decision when the recovered collapsed water level of SG-1 was over 7.0 m. In the B4.1 test, an SLB made the primary system pressure decrease due to the removal of excess heat through the break at the steam line. With the SGTR, the primary system inventory moved to the steam generator secondary side. After the injection of the auxiliary feedwater and the safety injection water from SIPs, however, the primary system pressure stabilised and the collapsed water level in the secondary side of the affected steam generator was recovered. From the present test result it can be concluded that the whole system can be successfully cooled down with the proper operation of safety systems against this kind of multiple failure accident.

Figure 2.6. Schematic diagram for simulating a mid-loop operation and the maximum core heater rod surface temperatures in the B4.2 test



Source: NEA, 2021.

The B4.2 test was performed to simulate a loss of the RHRS during a mid-loop operation. In that scenario, it has been shown that the safety of the reactor core and coolant system may be severely threatened by the boiling of coolant when the core decay heat is not properly removed. Therefore, an accident involving the loss of the RHRS is one of great concern since the accident has reoccurred and a number of probabilistic risk assessment (PRA) studies have identified this accident as the highest-risk scenario in low-power operation. The main purpose of this test was not only to investigate thermal-hydraulic transient in the RCS during a loss of the RHRS but also to evaluate the effectiveness of reflux condensation and safety injection from a SIT on shutdown coolability. In the B4.2 test, the pressuriser manway was opened and the initial water level in the primary system was at the centerline of the hot leg to simulate a mid-loop operation. The secondary system inventories were emptied at 5.0 m for SG-1 and SG-2, respectively. The SITs with variable initial pressure conditions were utilised in the present test. In the B4.2 test, the top part of the core was uncovered and the excursion of the heater rod surface temperature in the core occurred. The safety injection water from the safety injection tank number 1 (SIT-1) was supplied until the internal pressure of the SIT-1 was kept higher than the pressure of RPV downcomer. At the end, the SIPs were actuated when the heater rod surface temperature in the core exceeded 500°C since the safety injection water from SITs was not supplied due to a very small pressure difference between the primary system and the SIT-1. The B4.2 test result showed that the existence of secondary system inventory and the location of the pressuriser cause the asymmetric thermal-hydraulic behaviour in the RCS, and the safety injection from SIT and SIP can make up the uncovered core with the coolant and cool down the RCS during a mid-loop operation with a loss of RHRS.

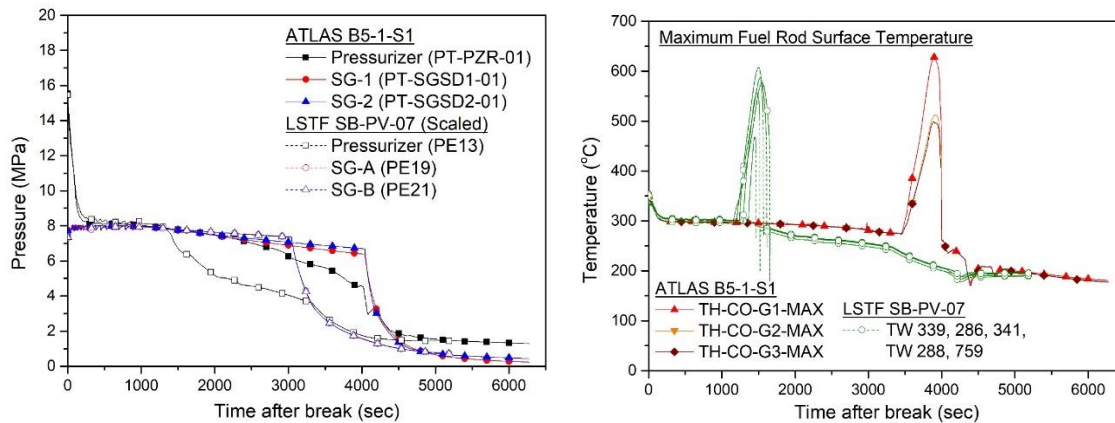
From the test results of the B4 series, it can be concluded that the reactor coolant system can be successfully cooled down with the proper operation of safety systems against a multiple failure accident.

2.5 Test B5 series: Counterpart test for benchmark analysis

Although many integral effect tests have been performed in the past two decades by utilising various large scale facilities, the scaling issue 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 verified before its data can be used for a safety analysis. It was agreed during the project that ATLAS could be utilised to reproduce one of the scenarios of LSTF in order to address the scaling issue.

The B5.1 test was defined as a counterpart test with respect to the LSTF SB-PV-07 test that simulated a 1% SBLOCA at RPV upper head under assumptions of total failure of the high-pressure injection system and non-condensable gas inflow to the primary system from accumulator tanks. The B5.1 test was also selected as a benchmark exercise during the project period of NEA ATLAS-2. A total of 11 participants adopting eight different codes participated in the benchmark exercise. This benchmark exercise consisted of blind and open analyses and it was co-ordinated by Tractebel. The participants adopted different approaches to modelling the RPV: 1D, quasi-3D or 3D. Even though the overall qualitative behaviour of code calculations in the blind phase is generally in agreement with the B5.1 test, there is quite a large spread in terms of timing and a notable difference in predicted peak cladding temperature (PCT). For the open phase of the benchmark, the main improvement made by almost all participants was break flow, for which adjustments to models were made in order to match the test data better. It mostly consisted of predicting a faster switch to single phase vapour at the break as in the B5.1 test. This resulted in a much better prediction of primary mass for most participants and consequently better matching of PCT occurrence timing.

Figure 2.7. Comparison of the system pressures and the PCT in the B5.1 and the SB-PV-07 tests



Source: NEA, 2021.

The initial steady-state conditions were achieved at a scaled power based on the core power that was supplied in the SB-PV-07 test. An SBLOCA at the RPV upper head was successfully simulated using the ATLAS facility as a counterpart test. When the maximum core exit temperature (CET) reached 623 K, the coolant was manually injected from the high-pressure injection system into cold legs in both loops as the first accident management action. The whole core was then quenched, and the accumulator system was actuated in both loops when the primary system pressure reduced to 4.51 MPa. After the scaled inventory was injected into the RCS from accumulator tanks, the accumulator tanks were not isolated from the RCS so the nitrogen gas could flow into the RCS. When the primary system pressure decreased to 4 MPa, the secondary system depressurisation was initiated by opening the atmospheric dump valves (ADV) in both steam generators as the second accident management action. At the same time as the second accident management action, the auxiliary feedwater injection was actuated. The overall sequence of transient scenario progressed later in the ATLAS B5.1 test than that of the LSTF SB-PV-07 test. This is mainly due to the different break flow rates between the two tests. ATLAS and LSTF have different inner geometries of the RPV upper head and it can have a significant effect on the RCS inventory, especially during the early transient period. The loop seal clearing phenomenon, which did not occur in the SB-PV-07 test, was clearly observed in the B5.1 test. This can be attributed to the different design of the intermediate leg, inner structure of the upper head, and location of the active core between two facilities that resulted from the different design of prototype nuclear power plants for each facility. These design differences can affect the pressure difference between the upper head and the downcomer region of RPV.

2.6 PKL-4-ATLAS-2 joint workshop

As the fourth phase of the NEA Primary Coolant Loop Test Facility (PKL) PKL-4 project had been under way since 2016, due to the links between the two programmes, the management boards of both projects decided in 2018 to organise a joint workshop of related analytical activities. The first joint workshop of the PKL-4 and ATLAS-2 projects took place in Barcelona, Spain, at the Technical University of Catalonia, from 7 to 9 November 2018. The workshop attracted 55 participants from 11 countries. It included 24 presentations covering the 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 application to reactor case. The joint workshop

provided an efficient way to evaluate the current code capabilities for the scenarios conducted in both projects. The conclusion of the first joint workshop prepared by the session chair together with the final summary integration report of these two projects will be issued as a public Committee on the Safety of Nuclear Installations (CSNI) report¹. The second joint workshop of both projects was originally scheduled to be held in Brussels, Belgium, at Tractebel Headquarters, from 3 to 5 November 2020. Unfortunately, due to the worldwide crisis resulting from COVID-19, the second joint workshop was postponed and a new date will be arranged in the near future during the next phases of both projects, i.e. the NEA ATLAS-3 and ETHARINUS projects.

1. NEA (forthcoming), “Summary and Conclusions of the Joint PKL-ATLAS Workshop on Analytical Activities Related to the NEA PKL4 and ATLAS-2 Projects”, NEA/CSNI/R(2020)13, OECD Publishing, Paris.

3. Conclusions and recommendations

The second phase of the Nuclear Energy Agency Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS) phase 2 project (NEA ATLAS-2) ran successfully from October 2017 to December 2020. A total of eight integral effect tests in five different topics were carried out and 18 organisations from 11 countries participated in the project. Utilising the established IET database, simulation models and methods for complex phenomena of high safety relevance to thermal-hydraulic transients in design-basis accident (DBA) and beyond-DBA (BDBA) were validated. The present NEA ATLAS-2 project aims to enhance the safety of operating nuclear power plants by simulating various accident transients in connection with safety analysis technology. The thermal-hydraulic behaviours related to a passive core makeup, an intermediate break loss-of-coolant accident (IBLOCA), and a multiple failure accident such as a steam line break (SLB) combined with a steam generator tube rupture (SGTR), were investigated in a systematic manner. However, there are still working areas where safety analysis technology can be improved and eventually severe accidents could be prevented in any case.

One of the most interesting topics is investigating the coupling between reactor coolant system (RCS) and containment in integral effect test (IET). In 2019, a containment simulating vessel named Containment Utility for Best-Estimate Evaluation (CUBE) was constructed and connected with the RCS of ATLAS. By utilising the ATLAS-CUBE facility, thermal-hydraulic interaction between the RCS and the containment building can be experimentally investigated. In particular, the evaluation of multi-dimensional phenomena inside the containment and cooling capability of the passive heat sink and spray system can be highlighted. Unique experimental data on pressure build-up, thermal stratification and mixing, and condensation heat transfer in the containment combined with the RCS of ATLAS can be utilised not only to validate the mass/energy (M/E) and pressure/temperature (P/T) evaluation methodology but also to investigate complex thermal-hydraulic phenomena inside the containment during accident transients.

Since the Fukushima Daiichi accident, various passive safety systems have been proposed to improve the safety and reliability of an ultimate heat removal system without any operator action during DBA and BDBA transients. In the framework of the NEA ATLAS-2 project, various passive safety systems were utilised. However, performance and reliability of passive systems still needs to be tested further due to the inherently low driving force, multi-dimensional flow or mixing, asymmetric behaviour, flow oscillation and instability of these systems. Since the complex thermal-hydraulic phenomena are also greatly affected by the detailed design of the system, it is considered very challenging for the code developers and users. The one-dimensional simulation code needs to be validated with various experimental databases before it is applied to the passive safety system. Therefore, it is expected that more experimental and validation work is necessary to improve understanding of the physics and to improve simulation codes. It is also highly recommended that three-dimensional analysis be used in predicting the complex phenomena related to the passive safety system.

Natural circulation is a basic thermal-hydraulic phenomenon that determines the cooling of the RCS during a high-pressure accident sequence such as an SBO and also a very low

pressure accident sequence such as a mid-loop operation. Due to its weak driving force compared to the forced circulation, precise evaluation of the natural circulation by utilising system-scale safety analysis code is challenging. In particular, it is worth investigating the thermal-hydraulic characteristics anticipated in the natural circulation under asymmetric cooling conditions.

Even though the exact definition of design extension conditions (DEC) varies depending on countries, the objective of the DEC studies is to improve safety by enhancing the capability of nuclear power plants to withstand conditions generated by accidents that are more severe than DBAs or that involve additional failures. Some DEC, such as an SBO and a total loss of feedwater (TLOFW), were already taken into account in the previous NEA ATLAS project. A multiple failure sequence was experimentally investigated in the NEA ATLAS-2 project. In principle, the fuel degradation should be prevented in any case for sustainable nuclear energy. Thus, continuous utilisation of ATLAS is highly recommended for simulation of the various multiple failure accident and more severe DEC, such as total loss of heat sink.

In 2016, France eliminated a large break loss-of-coolant accident (LBLOCA) from the list of DBAs, which means that more licensing focus will be on the IBLOCA rather than the LBLOCA. These changes in regulatory position for LOCA will significantly affect the worldwide regulatory environment. In particular, a redefinition of a design-basis LOCA accident is a safety issue from the viewpoint of safety enhancement. Although two IBLOCA tests were performed in the present NEA ATLAS-2 project, there remain more areas to be investigated. More experimental programmes need to be designed in the IBLOCA area by utilising new design features of RCS-containment integrated systems and passive safety systems.

The scaling issue means research is needed on how to apply the experimental information obtained from small facilities to nuclear power plants. It is an unresolved issue and is a wish-and-see problem in the field of nuclear safety analysis. In order to address the scaling issue, analytical and experimental investigation should be done together. In particular, a systematic counterpart test programme is essential. Therefore, it is recommended that ATLAS be utilised together with other facilities to effectively address in the future the scaling issue as well as new nuclear safety interest