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Final Integration Report of the OECD/NEA ROSA Project

2005-2009

BETTER POLICIES FOR BETTER LIVES

NUCLEAR ENERGY AGENCY

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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The OECD is a unique forum where the governments of 34 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, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Republic of Korea, 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.

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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 31 countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Russian Federation, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes 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 friendly and economical use of nuclear energy for peaceful purposes, as well as
– 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 policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, 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. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

THE COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

"The Committee on the Safety of Nuclear Installations (CSNI) shall be responsible for the activities of the Agency that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations, with the aim of implementing the NEA Strategic Plan for 2011-2016 and the Joint CSNI/CNRA Strategic Plan and Mandates for 2011-2016 in its field of competence.

The Committee shall constitute 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 shall have 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 shall review the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensure that operating experience is appropriately accounted for in its activities. It shall initiate and conduct programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach consensus on technical issues of common interest. It shall promote 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, and shall assist in the feedback of the results to participating organisations. The Committee shall ensure that valuable end-products of the technical reviews and analyses are produced and available to members in a timely manner.

The Committee shall focus primarily on the safety aspects of existing power reactors, other nuclear installations and the construction of new power reactors; it shall also consider the safety implications of scientific and technical developments of future reactor designs.

The Committee shall organise its own activities. Furthermore, it shall examine any other matters referred to it by the Steering Committee. It may sponsor specialist meetings and technical working groups to further its objectives. In implementing its programme the Committee shall establish co-operative mechanisms with the Committee on Nuclear Regulatory Activities in order to work with that Committee on matters of common interest, avoiding unnecessary duplications.

The Committee shall also co-operate with the Committee on Radiation Protection and Public Health, the Radioactive Waste Management Committee, the Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle and the Nuclear Science Committee on matters of common interest."

FOREWORD

OECD/NEA ROSA Project was performed to resolve issues in thermal-hydraulics analyses relevant to light water reactor (LWR) safety by using the ROSA/LSTF of JAEA for 4 years from April 2005. In particular, it focused on the validation of simulation models and methods for various complex phenomena that may occur during design-basis event (DBE) and beyond-DBE and to increase the level of detail and accuracy in the analyses of the key phenomena during transients and accidents of interest.

To meet these objectives, the ROSA Project performed 12 LSTF experiments for the following 6 types of subject as either separate-effect tests or integral-effect tests.

- Test 1 Temperature stratification and coolant mixing during ECCS coolant injection
- Test 2 Unstable and destructive phenomena such as water hammer
- Test 3 Natural circulation under high core power conditions
- Test 4 Natural circulation with superheated steam
- Test 5 Primary cooling through SG secondary depressurization
- Test 6 Subjects open to participants

These subjects were defined among the Project participants from 14 countries who share the need to maintain or improve their technical competence in thermal-hydraulics for nuclear reactor safety evaluations. System modifications, addition of various types of new instrumentation and/or preparation of new test sections were performed for the ROSA/LSTF to obtain detailed and specific data necessary to achieve the objectives.

The LSTF experiments generated the data that fulfill current and future needs for the development, verification and validation of best estimate (BE) and computational fluid dynamics (CFD) codes. The obtained data as well as technical information and experiences through the post-test analysis by using the BE and CFD codes have been shared among the participants through the discussions mostly during program review group (PRG) meetings.

This report briefly summarizes major findings in all the 6 subjects of the ROSA Project with a description of background and objectives for each subject. Appendices describe new measurement instruments and new test sections furnished for the ROSA Project, and summarize the post-test analyses including answers to a questionnaire on the analysis effort.

EXECUTIVE SUMMARY

Background of ROSA Project

Computer codes are utilized in safety evaluation of light water reactors (LWRs) in order to simulate plant behavior during design basis event (DBE) and beyond-DBE (BDBE). Complex multi-dimensional single-phase and two-phase flow conditions occur during these events and non-condensable gas is present in many cases. For this purpose, thermal-hydraulic safety analysis codes, especially best estimate (BE) loss-of-coolant accident (LOCA) analyses codes, have been developed with significant efforts to achieve high predictive capability especially for one-dimensional phenomena such as flows in piping at rather high flow rate. Such BE codes include APROS, ATHLET, CATHARE, MARS, RELAP5, TRAC and TRACE. However, there remain needs for experimental work and code development and validation for the complex flow conditions during DBE and BDBE. Furthermore, the increased use of the BE analysis codes and methodologies in the safety evaluation of reactors, replacing traditional conservative approaches, requires the quantification of uncertainties in the simulation models and methods.

Many experimental facilities have contributed to the thermal-hydraulic databases available today including those for the CSNI code validation matrix (CCVM). However, a large portion of current data is insufficient for future codes that are to incorporate multi-dimensional simulation capabilities, mainly because the spatial resolution of measurement and the measurement parameters are not enough to assess the simulation models and methods.

The CSNI set up an international working group SESAR/FAP (Senior Group of Experts on Nuclear Safety Research Facilities and Programmes) in 1997 to identify how existing expertise in the field of reactor safety and an appropriate experimental infrastructure could be sustained in the future. One of the main tasks of the SESAR/FAP was to identify test facilities and research programs threatened by closure shortly and to select facilities and programmes that would be of particular benefit to the member countries, if they were able to continue in a form of OECD project. Several international co-operative research projects had been started in the field of thermal hydraulics safety research to implement the recommendation by the SESAR/FAP: these include the SETH (SESAR Thermal-Hydraulics) Project using two test facilities; PANDA of PSI, Switzerland and PKL of AREVA, Germany, and PKL Project. Later, the CSNI established the SESAR/SFEAR (Support Facilities for Existing and Advanced Reactors) in 2005 as a follow-up of the SESAR/FAP to assess the need for and strategy of maintaining key research facilities. In both working groups, the LSTF (Large Scale Test Facility) in JAEA (former JAERI) was discussed and evaluated as an excellent facility for the thermal-hydraulic safety research. In the early 2000s, JAEA decided to propose an OECD/NEA project to make the LSTF utilized more effectively and efficiently among the member countries.

The LSTF was constructed in 1985 for the fourth-phase of ROSA (Rig-of-Safety Assessment) Program in JAEA following the TMI-2 accident to experimentally investigate thermal-hydraulic phenomena that may appear during postulated abnormal transients and small-break LOCAs (SBLOCAs). The LSTF is a full-pressure full-height two-loop model of a 4-loop Westinghouse-type PWR with a volumetric scaling of 1/48. More than 180 experiments were performed prior to the OECD/NEA ROSA Project, indicating its excellent capabilities to simulate the thermal-hydraulic phenomena under conditions close to those in the reference reactor. One of the important benefits of utilizing the LSTF for the OECD/NEA Project would be the combination of the facility with the latest instrumentation to measure various types of measurement parameters with more detailed spatial resolution than ever. In the late 1980s and 1990s, the measurement instrumentations for the LSTF experiments were designed to be suitable for the verification of one-dimensional BE codes. The observed phenomena, however, were often characterized by multi-dimensional effects, including parallel channels, unstable, multi-phase flows under influences of non-condensable gas. The measured data from these early experiments sometimes lacked the spatial resolution and parameters to well describe the multidimensional phenomena.

This ROSA Project tries to generate and share the necessary detailed data among the OECD member countries to fulfill the current and future needs for the development, verification and validation of both the BE and computational fluid dynamics (CFD) codes, by taking a view of the combined utilization of BE code with uncertainty evaluation methods such as BEPU (best estimate plus uncertainty).

Objectives of ROSA Project

The ROSA Project is to resolve issues in thermal-hydraulics analyses relevant to LWR safety by using the LSTF. In particular, it intends to focus on the validation of simulation models and methods for complex phenomena that may occur during DBE and beyond-DBE of LWRs. The key objective of the ROSA Project is to provide integral and separate-effect experimental database to validate code predictive capability and accuracy models. Especially, the LSTF experiments were planned to include phenomena coupled with multi-dimensional mixing, stratification, parallel flows, oscillatory flows and non-condensable gas flows.

A group to pursue the key objectives was formed among the OECD Member countries who share the need to maintain or improve the technical competence in thermal-hydraulics for nuclear reactor safety (NRS) evaluations. The experimental program was defined among the participants of the ROSA Project to provide a valuable and broadly usable database to achieve the objectives.

Work performed in ROSA Project

To meet the objectives, the ROSA Project performed 12 LSTF experiments for the following 6 types of subject as either separate-effect test (SET) or integral-effect test (IET).

Test 1 Temperature stratification and coolant mixing during ECCS coolant injection

An IET and an SET were performed to clarify temperature stratification in cold legs and upper downcomer in pressure vessel (PV) under reactor-typical conditions, which is important for the evaluation of pressurized thermal shock (PTS).

Test 2 Unstable and destructive phenomena such as water hammer

A series of SETs was performed to clarify influences of system pressure on occurrence and intensity of condensation-induced water hammer (CIWH) during ECCS injection into cold legs.

Test 3 Natural circulation under high core power conditions

Two IETs were performed to clarify thermal-hydraulic response adverse to core cooling during a SBLOCA or an operational transient, both with high core power because of failure of reactor scram.

Test 4 Natural circulation with superheated steam

Two SETs were performed to demonstrate a natural circulation of superheated steam during a severe accident that may cause containment bypass in case of the rupture of steam generator (SG) U-tubes at high-pressure sequence.

Test 5 Primary cooling through SG secondary depressurization

Two IETs simulated SBLOCA with a break in a cold leg, focusing on the long-term core cooling at low system pressures by natural circulation induced by the SG secondary depressurization under influences of non-condensable gas in SG U-tubes from accumulators.

Test 6 Open subjects

The subjects for this test series were open to the project participants. Candidates were discussed as follows:

1. PV upper-head SBLOCA -- Wall thinning found at the Davis Besse reactor in the US in 2002 raised a safety issue concerning vessel structural integrity,
2. PV bottom SBLOCA -- A small amount of residue found at around the circumference of two instrument-tube penetration nozzles at the South Texas Project Unit-1 in the US in 2003 raised a safety issue concerning vessel structural integrity,
3. Large-break LOCA (LBLOCA) focusing on the direct-contact condensation of steam on ECCS water in cold legs – This phenomenon should influence the containment pressure. A good database was lacking too.
4. SBLOCA focusing on accident management (AM) measures with symptom-oriented operator actions.

Three LSTF experiments were dedicated for this test series. The 4th subject above was then decided being coupled to the 1st and 2nd subjects. Consequently, two IETs (**Tests 6-1** and **6-2**) simulated the 1st and 2nd subjects with SG depressurization as an AM measure. One SET (**Test 6-3**) was performed corresponding to the 3rd subject. A new test section was prepared to simulate prototypical steam flow conditions that may appear in cold legs; high flow rate with large superheating at near-atmospheric pressure.

Pre-test analyses were done by JAEA for all the experiments to survey optimum test conditions by using such computer codes as RELAP5/MOD3.2.1.2, SKETCH-INS/TRAC-PF1 and FLUENT version 6.2 and 6.3. The obtained results were distributed and discussed among the Project participants to define detailed conditions of the LSTF experiments.

System modifications such as addition of **new measurement instrumentation** and/or preparation of **new test sections** were performed to obtain detailed and specific data for each test. The new instrumentations and/or preparation of new test sections include followings:

(1) thermocouple arrays in cold and hot legs, PV downcomer and SG inlet plenum
(2) non-condensable gas detectors at SG outlet plenum and PV upper-head
(3) gas velocity meters for low-velocity gas flow in hot legs
(4) video probes employing bore scopes for visual observation of flows in cold legs
(5) nuclear data for the preparatory PWR analyses to simulate LOCA and anticipated transient, both without scram
(6) test section for CIWH with fast-response pressure sensors, a pitot tube and a gamma-ray densitometer
(7) test section for direct steam condensation on ECCS water downstream of the break unit horizontally connected to the LSTF cold leg in the loop without pressurizer

All above except (5) were newly prepared for the ROSA Project by JAEA experimental team.

The obtained LSTF data has been utilized for the **post-test analyses** for the development, verification and validation of the thermal-hydraulic BE safety analysis codes and models as well as CFD codes. The technical information and experiences obtained through the post-test analysis including methods to improve the analysis accuracy were shared among the Project participants through the discussions in PRG (Program Review Group) meetings and over e-mail communications.

A **questionnaire** was performed further with the help from the Project participants who performed **Test 6-1** post-test analysis to clarify their specific efforts in the analysis method to obtain better agreement with the LSTF data.

Major Results and Significances

The 12 LSTF experiments for 6 subjects provided the following technical findings suitable for the V&V (verification and validation) of safety analysis codes and new issues for the assurance and improvement of LWR safety.

Test 1 Temperature stratification and coolant mixing during ECCS coolant injection
An SBLOCA experiment with a 1% break at hot leg (**Test 1-1**, IET) and a steady natural circulation experiment (**Test 1-2**, SET) were performed focusing on the temperature stratification in cold legs and PV downcomer during ECCS injection. In **Test 1-1**, the primary inventory decreased gradually because of break flow discharge under influences of the ECCS injection following plant control sequence. The inventory was stepwise decreased in **Test 1-2** where the ECCS injection was done at pre-determined constant flow rates after steady condition was achieved in each step.
The two experiments revealed that the temperature stratification condition in cold leg depends on the injection manner of the ECCS and that the liquid level in horizontal leg and flow condition may control low-temperature coolant flowing down the PV downcomer along either PV inner wall side or core shroud outer wall side.
Namely, the temperature stratification in cold legs was seen widely in both tests under both single- and two-phase natural circulation conditions. The degree of temperature stratification was larger in the intact-loop cold-leg (loop with pressurizer), since the shape and the location of the ECCS injection nozzles are different between two loops. It was found that the cold water from the cold leg flowed down in the downcomer along the PV inner wall under the natural circulation condition when the downcomer liquid level was higher than the cold leg. The cold water flowed down along the outer wall of the core barrel cover when the loop flow was almost stagnated or the downcomer liquid level was lower than the cold leg.
Such temperature stratification important for the evaluation of pressurized thermal shock (PTS) was observed under reactor typical pressure, temperature and flow conditions, and utilized for the validation of both CFD and BE codes.

Test 2 Unstable and destructive phenomena such as water hammer
A series of SETs was performed to study onset conditions and consequences of water hammer in a new horizontal pipe test section attached to the LSTF under wide range of system pressure. It was confirmed that the Condensation Induced Water Hammer (CIWH) occurs at high ambient pressure up to about 7 MPa.
The LSTF experiments clarified that the water hammer generates greatest pressure pulse peak when the system pressure was at around 1 MPa and it may take place up to 7 MPa. The averaged amplitude of pressure pulses in each CIWH test increased with the system pressure when it was less than 1 MPa, and decreased as the system pressure increases. A pressure pulse whose amplitude was larger than 10 MPa formed even at the system pressure of 7 MPa, although the averaged amplitude was less than 1 MPa. The estimated superficial steam velocity at the inlet of test section was related to the averaged amplitude of pressure pulses, probably because the steam velocity is proportional to the steam condensation rate. This observation suggested that the large volumetric shrinkage induced by the significant condensation rate

produced the large pressure pulse. The water injection rate seemed to affect the dependency of the pressure pulse amplitude on the system pressure. The effect of the water level at the outlet of test section on the intensity was small. No thermal stratification formed in the liquid phase once the CIWH was induced. The effective mixing of the condensate with the subcooled water may have resulted in the large condensation rate.
The onset conditions of and intensity of pressure pulses due to the CIWH was experimentally clarified as above, which may threaten the integrity of reactor structures even under pressures typical of LWRs.

Test 3 Natural circulation under high core power conditions
Two IETs were performed under an assumption of failure of reactor scram to reveal various complex thermal-hydraulic phenomena adverse to core cooling under high core power conditions and difficult for code analyses. The core power for two tests was obtained from PWR analysis by using the SKETCH-INS/TRAC-PF1 code. Total failure of high-pressure injection (HPI) system was assumed further in both tests.
An SBLOCA with 1% cold leg break was simulated with **Test 3-1**. The natural circulation under high core power contributed to maintain core cooling effectively for relatively long time until core uncover. Several phenomena specific to the natural circulation under high core power were observed such as supercritical flow in hot legs and counter-current flow limiting (CCFL) in SG inlet plenum and U-tubes because of high-velocity steam flow. Coolant accumulation in the SG inlet plenum and U-tubes by the CCFL during reflux condensation mode caused a small drop in the core liquid level. Steam condensation on coolant injected from accumulator tank into cold leg induced loop seal clearing in the broken loop.
A loss-of-feedwater (LOFW) transient with delayed auxiliary feedwater actuation was simulated with **Test 3-2**. Core power was maintained at about 7.4% for more than 7 hours until the automatic power reduction due to high core temperature, which was caused by a gradual coolant inventory loss due to cycle opening of the PORV. Two-phase natural circulation started in very early stage of the transient. Significant level oscillation, in a form of slow fill and dump, occurred in all the instrumented SG U-tubes with gradual decrease in the primary loop flow rate. Very complex reflux cooling phenomena appeared in the SG U-tubes in non-uniform manner such that a water column developed rather randomly among the U-tubes with a lower frequency than that of the cycle opening of the RVs and PORV. Temperature excursion appeared in the core in the reflux condensation mode, even when the liquid level was still in the upper plenum, suggesting departure from nucleate boiling (DNB) due to the high core power.
Various and very complicated thermal-hydraulic responses were observed, including inadequate core cooling in the reflux condensation mode, associated with the natural circulation under high core power.

Test 4 Natural circulation with superheated steam
Two SETs were performed, with argon (**Test 4-1**) and super-heated steam (**Test 4-2**) as test fluid at the system pressure of 1 MPa and 4 MPa respectively, to clarify the onset conditions of superheated steam natural circulation that may cause bypass sequence during severe accident if SG U-tube(s) rupture under high temperature and pressure conditions. The experiment conditions were defined and confirmed through the CFD (FLUENT) code analyses and LSTF preparatory experiments. The FLUENT analysis indicated that the gas temperature at the hot leg inlet should be higher than that in the SG secondary side by at least 100 K.
The gas temperature at the hot leg inlet in the PV upper plenum was higher than that in the SG secondary side by about 300 K in **Test 4-1** and by 150 K in **Test 4-2**, which was sufficiently larger than the estimated temperature difference. Gas velocity measured in the hot legs was about 0.3 m/s at maximum for both tests, which was comparable to those estimated with FLUENT code.
Forward high-temperature gas flow from PV to SG and backward low-temperature gas flow from SG to PV developed in the upper and the lower volumes of hot legs, respectively. In the upper volume, a flat temperature profile formed at around the leg inlet and the gas temperature decreased toward SG. A flat temperature profile formed in the lower volume near the SG inlet plenum and the gas temperature increased toward PV with similar change of profile. This temperature profile could be related to the growth

of thermal boundary layer and the mixing of gas under the counter-current gas-gas flow conditions.
The natural circulation of superheated steam was confirmed to form in the LSTF primary loops including counter-current gas-gas flow in hot legs. Further it was found that forward and backward flow existed at the same time in SG U-tubes and hotter and colder gases mixed in SG inlet plenum. No large qualitative difference appeared between argon and steam cases.

Test 5 Primary cooling through SG secondary depressurization
Two IETs, both simulating SBLOCA with a 0.5% cold leg break, were performed without (**Test 5-1**) or with (**Test 5-2**) non-condensable gas from accumulator tanks of ECCS to clarify non-uniform parallel SG U-tube behavior during the SG depressurization as an AM action. In both experiments, SG relief valves (RVs) were fully opened at 10 minutes after the safety injection signal. Total failure of HPI system was assumed in both tests. To well observe the natural circulation phenomena at low pressures, an enhanced SG depressurization by fully opening the safety valves (SVs) was performed when the primary pressure decreased to 2 MPa as well as no actuation of low-pressure injection (LPI) system.
Two-phase natural circulation developed in the primary loops soon after the break valve opening and continued until the primary pressure decreased far below 1 MPa with non-uniform flow among the SG U-tubes in **Test 5-1** without non-condensable gas ingress.
In **Test 5-2** with the gas ingress, on the other hand, the primary depressurization rate decreased after non-condensable gas started to enter primary loops and SG U-tubes, suggesting the occurrence of degradation in the condensation heat transfer in the SG U-tubes. Collapsed liquid level decreased gradually in the U-tubes after the initiation of non-condensable gas inflow while two-phase natural circulation continued with non-condensable gas. However, some U-tubes indicated a rapid coolant drain, while single-phase liquid natural circulation continued in some other tubes, probably because of preferential accumulation of non-condensable gas into such drained U-tubes. Asymmetrical natural circulation between two loops also appeared in **Test 5-2** due probably to different number of forward flow U-tubes under the influences of non-condensable gas. The natural circulation continued and contributed to maintain core cooling for a long time even with the gas ingress into SG U-tubes.
In the two LSTF experiments various non-uniform flows in SG U-tubes under the influences of non-condensable gas onto heat transfer during the natural circulation was systematically observed especially at low system pressures which is important for long-term core cooling.

Test 6 Open subjects
In response to the requests from Project participants, following three experiments (2 IETs and 1 SET) were performed;

Test 6-1: An SBLOCA with a break at PV upper-head was simulated under assumption of total failure of HPI system. The break size was equivalent to 1.9% cold leg break to simulate the ejection of one whole CRDM penetration nozzle. SG secondary depressurization was performed as a symptom oriented AM measure based on core exit temperature (CET).
The primary pressure started to decrease after the break and became lower than the SG secondary-side pressure almost simultaneously with the core uncovery, resulting in no reflux flow from the SG. The relatively lower primary pressure at the onset of SG depressurization after the superheating detection by CETs made the initiation of coolant injection from accumulator tanks too late to adequately cool the core in time. A significant core temperature excursion then appeared and core power was terminated due to high core temperature. Slow and late response of CETs relative to the core temperature excursion during core boil-off was revealed, suggesting a necessity of further study on the CET responses because CETs are utilized worldwide as one of key parameters to start AM operator action.
The result of this test raised a new safety issue on the reliability of the CET to detect core uncovery. A new task group was then formed in 2008 under WGAMA (working group on analysis and management of accident) of OECD/NEA/CSNI and summarized the state-of-the-art knowledge with recommendations as a CSNI report "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor" NEA/CSNI/R(2010)9. A paper was presented further for OECD/NEA special session at the

NURETH-14 at Toronto in 2011.
This issue is currently under investigation, including a ROSA-PKL counterpart-test with collaboration between the OECD ROSA2 and PKL2 Projects, because the WGAMA task group identified that multi-dimensional behavior of superheated steam in the core should be a key and scaling-related when the obtained findings are applied to reactor accidents.

Test 6-2: An SBLOCA with a break at PV bottom was simulated with an asymmetrical SG secondary depressurization as AM measure under assumption of total failure of HPI system. The break size was equivalent to 0.1% cold leg break to simulate the ejection of one whole instrumentation nozzle. An asymmetrical SG depressurization in loop w/o pressurizer (active loop) only was employed, expecting unbalanced natural circulation between loops and unbalanced transport of non-condensable gas from accumulator tanks. The depressurization was initiated at 30 minutes after the safety injection signal, considering an international common understanding on grace period to start operator action. The SG depressurization rate was controlled such that the primary depressurization rate becomes to be 55 K/hr. The SG depressurization was started before the core uncovery, because of very small break and slow primary depressurization, resulting in continuous core cooling by two-phase natural circulation through the active loop. After non-condensable gas ingress started from accumulator tanks, however, the primary depressurization rate greatly decreased because of the degradation of condensation heat transfer in the SG U-tubes, resulting in a large pressure difference between the primary and SG secondary. The core uncovery finally started due to gradual boil-off because of the difficulty in the initiation of LPI system.
After the core uncovery started, CETs increased on the inactive loop side, while not on the active loop side due to condensate fall-back from the SG. The LPI was actuated as the steam generation rate decreased in the core, and most of the core was quenched. The core boil-off, however, started again because the LPI was stopped when the primary pressure raised due to the increase in the steam generation rate in the core. Core power was finally decreased automatically because of too-high core temperature during the second core boil-off.
A unique database was provided by **Test 6-2** including an asymmetric natural circulation core cooling under the influences of non-condensable gas, which indicated a possibility of insufficient core cooling in a long-term transient due to high primary pressure relative to LPI actuation pressure.

Test 6-3: Direct steam condensation on ECCS coolant in cold legs during reflood phase of LBLOCA was simulated in a test section newly installed downstream of the break unit that is horizontally attached to the LSTF cold leg. The test section consisted of a simulated cold leg pipe (4.5 m-long, 102.3 mm i.d.), a simulated ECCS injection nozzle, three viewers, a separator, separator downstream pipes for steam and water. Steam velocity at the inlet of the simulated cold leg pipe ranged from about 20 to 50 m/s being controlled by the LSTF core power, covering conditions estimated from the PWR LBLOCA analyses. The inlet steam superheating was as high as possible to best simulate the PWR conditions. Simulated ECCS water was at temperature of 310 K with a mass flux ranging from about 12 to 207 kg/(m^2s) so as to preserve the ratio of ECCS water injection flow rate to the inlet steam flow rate.
Significant steam condensation was observed between the subcooled ECCS coolant and highly superheated steam in a short distance from the ECCS injection point. Fluid temperature distribution was non-uniform at about 50 mm downstream from the ECCS injection point but became to be almost uniform in less than about 350 mm. Total steam condensation rate estimated from the difference in the steam flow rates between the inlet and outlet of test section was in proportion to the simulated ECCS water mass flux until the complete condensation of steam. Overall condensation rate was linearly proportional to the injected ECCS coolant.
Image data on liquid droplets under condensation in the high-velocity steam flow was obtained further by high-speed video camera for in-depth investigation of the flow conditions under non-equilibrium steam condensation, necessary for the verification and validation of the models and safety analysis codes.

Conclusions and Recommendations

The OECD/NEA ROSA Project to resolve issues in thermal-hydraulics analyses relevant to LWR safety was conducted during 4 years since April 2005 among 14 countries by using the LSTF of the ROSA Program in JAEA. Major objectives of the ROSA Project are as follows:

(1) Provide integral and separate-effect experimental database to validate code predictive capability and accuracy of models for complex phenomena that may occur during DBE and beyond-DBE.

(2) Clarify the predictive capability of codes currently used for thermal-hydraulic safety analyses as well as of advanced codes presently under development.

The LSTF is a full-pressure full-height two-loop model of an 1100 MWe 4-loop Westinghouse-type PWR with a volumetric scaling of 1/48, and has been utilized for the investigation of thermal-hydraulic response during accidents and abnormal transients since 1985, by fully utilizing the capability to demonstrate the key phenomena.

The ROSA Project investigated various types of thermal-hydraulic responses that may encounter during accidents and abnormal transients of LWR through 6 subjects with 12 LSTF experiments in either IET or SET. Care was taken to include such complex phenomena as multi-dimensional mixing, stratification, parallel flows, oscillatory flows and/or those with non-condensable gas. Efforts were taken further to increase data parameters necessary for better understanding of the phenomena by employing four types of JAEA-designed new instrumentation as well as two new test sections for specific subjects of CIWH and direct contact steam condensation during reflood phase of large break LOCA, both with much of the specially designed instrumentation.

For all the experiments, JAEA performed pre-test analyses to survey optimum test conditions to meet the objectives of each test and the ROSA Project. The obtained results were distributed and discussed among the Project participants to define detailed test conditions.

From the 12 LSTF experiments, a unique database was obtained, and was readily shared among the Project participants to utilize it for the post-test analyses for the verification and validation of the safety analysis codes and models as well as CFD codes, both under development and improvement. This database may also be added as appropriate to the CCVM for its next revision. The technical information obtained through such effort was discussed and shared through the PRG meetings, partly through a questionnaire for the post-test analysis of PV upper-head break LOCA test (Test 6-1), and over e-mail communications.

Inspired by the result of Test 6-1, a safety issue was newly raised on the effectiveness of CET that is utilized to start an operator action to cool uncovered core as an accident management measure. A task group was formed in WGAMA, issued a CSNI report of NEA/CSNI/R(2010)9 and prepared a co-authored paper for the NURETH-14 in 2011.

Through the ROSA Project, the LSTF has been recognized useful and suitable for the international research activities to resolve issues in thermal-hydraulics analyses relevant to LWR safety. Further utilization of the LSTF would be beneficial to reinforce LWR thermal-hydraulic safety by addressing such specific issues as V&V of BE and CFD codes and scaling of thermal-hydraulic phenomena necessary for the accurate safety evaluation including BEPU through in depth understanding of the complicated phenomena that may encounter during DBAs and beyond-DBAs.

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1. BACKGROUND, OBJECTIVES & METHODS

1.1 Background of ROSA Project

Computer codes are utilized in safety evaluation of light water reactors (LWRs) in order to simulate plant behavior during design basis event (DBE) and beyond-DBE (BDBE). This involves complex multi-dimensional single-phase and two-phase flow conditions, which may include non-condensable gas in many cases. For this purpose, thermal-hydraulic safety analysis codes, especially for best estimate (BE) loss-of-coolant accident (LOCA) analyses codes, have been developed with significant efforts to achieve high predictive capability especially for one-dimensional phenomena such as flows in piping at rather high flow rate. Such BE codes include APROS, ATHLET, CATHARE, MARS, RELAP5, TRAC and TRACE. However, there remain needs for experimental work and code development and validation for the complex flow conditions during DBE and BDBE. Further, the increased use of the BE analysis codes in the safety evaluation of reactors, which is replacing traditional conservative approaches, requires the validation and the quantification of uncertainties in the simulation models and methods.

Many experimental facilities have contributed to the thermal-hydraulic databases available today including those for the CSNI code validation matrix (CCVM). However, a large portion of current data is insufficient for future codes that are to incorporate multi-dimensional simulation capabilities, mainly because the spatial resolution of measurement and the measurement parameters are not enough to assess the simulation models and methods.

Meanwhile, the Committee on the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency (NEA) of OECD set up an international working group SESAR/FAP (Senior Group of Experts on Nuclear Safety Research Facilities and Programmes [1-1]) in 1997 to identify how existing expertise in the field of reactor safety and an appropriate experimental infrastructure could be sustained in the future. One of the main tasks of the SESAR/FAP was to identify test facilities and research programs threatened by closure shortly and to select facilities and programmes that would be of particular benefit to the member countries, if they were able to continue in a form of OECD project. Several international co-operative research projects had been started in the field of thermal hydraulics safety research to implement the recommendation by the SESAR/FAP, which included the SETH (SESAR Thermal-Hydraulics) Project using two test facilities; PANDA of PSI, Switzerland and PKL of AREVA, Germany, and PKL Project. Later, the CSNI established the SESAR/SFEAR (Support Facilities for Existing and Advanced Reactors [1-2]) in 2005 as a follow-up of the SESAR/FAP to assess the need for and strategy of maintaining key research facilities. In both working groups, ROSA/LSTF [1-3] in JAEA (former JAERI) was discussed and evaluated as an excellent facility for the thermal-hydraulic safety research. However, the ROSA/LSTF was not considered for the OECD/NEA Project because of heavy utilization for Japanese domestic safety research in the late 1990s, for example, to confirm the effectiveness of passive safety installations such as PCCS (passive containment cooling system) with a horizontal heat exchanger in collaboration with industry. In the early 2000s, JAEA decided to propose an OECD/NEA project to make the ROSA/LSTF utilized more effectively and efficiently among the member countries.

The ROSA/LSTF was constructed in 1985 at JAEA (formerly JAERI) following the TMI accident to experimentally investigate thermal-hydraulic phenomena that may appear during postulated abnormal transients and small-break LOCAs (SBLOCAs) as described in **Chapter 2**. More than 180 experiments were performed prior to the ROSA Project, indicating its excellent capabilities to simulate the thermal-hydraulic phenomena under conditions close to those in the reference reactor. One of the important benefits of utilizing the ROSA/LSTF for the OECD/NEA Project would be the combination of the facility with the latest instrumentation to measure various types of measurement parameters with more detailed spatial resolution than ever. In the late 1980s and 1990s, the data obtained from the ROSA/LSTF experiments were designed to be suitable for the verification of one-dimensional BE codes. The observed phenomena, however, were often characterized by multi-dimensional effects, including parallel channels, unstable, multi-phase flows under influences of non-condensable gas. The obtained data in such early days were sometimes lack in the spatial resolution and parameters to well describe the multidimensional phenomena.

The proposed ROSA Project tries to generate and share the necessary data among the OECD member countries to fulfill the current and future need for the development, verification and validation of both the BE and computational fluid dynamics (CFD) codes, by taking a view of the combined utilization of BE code with uncertainty evaluation methods.

1.2 Objectives of ROSA Project

The OECD/NEA ROSA Project is to resolve issues in thermal-hydraulics analyses relevant to LWR safety using the ROSA/LSTF facility of JAEA [1-3]. In particular, it intends to focus on the validation of simulation models and methods for complex phenomena that may occur during DBE and beyond-DBE. The key objective of the ROSA Project is to provide integral and separate-effect experimental database to validate code predictive capability and accuracy models. Especially, the ROSA/LSTF experiments were planned to include phenomena coupled with multi-dimensional mixing, stratification, parallel flows, oscillatory flows and non-condensable gas flows.

A group to pursue the key objectives formed among the OECD Member countries who share the need to maintain or improve the technical competence in thermal-hydraulics for nuclear reactor safety (NRS) evaluations.

The experimental program was defined to provide a valuable and broadly usable database to achieve the objectives. Twelve experiments were conducted in six types of subject, which are briefly explained in the following section.

1.3 Scope of Work of ROSA Project

The OECD/NEA ROSA Project consists of the following six types of subject; **Test 1** through **Test 6**, which are either separate-effect test (SET) or integral-effect test (IET). Twelve LSTF experiments were performed for the six types of subject as shown in **Table 1-1**. **Table 1-1** contains major subject, test classification (SET or IET), specific safety significance, suitability of data to use for the validation of BE and/or CFD code, and additional specific features such as new instrumentation, facility modification and/or preparatory (pre-test) analyses.

Major significances for each of 6 subjects are as follows:

Test 1 Temperature stratification and coolant mixing during ECCS coolant injection

This test is intended to clarify temperature stratification in cold legs and upper downcomer in pressure vessel (PV), which is important for the evaluation of pressurized thermal shock (PTS).

Two LSTF experiments were intended to simulate the temperature stratification under the reactor-typical conditions; pressure, temperature, temperature difference and flow transient in rather realistically-shaped horizontal legs with an emergency core cooling system (ECCS) injection port and a concentric downcomer.

Test 2 Unstable and destructive phenomena such as water hammer

This test is intended to clarify influence of system pressure on occurrence and intensity of condensation-induced water hammer (CIWH) during ECCS injection.

A series of LSTF experiments was intended to simulate the CIWH condition under reactor-typical pressure. Namely, the relevant conditions including temperature difference between steam at saturation temperature and ECCS water at room temperature were achieved under wide range of the system pressure up to about 7 MPa.

Test 3 Natural circulation under high core power conditions

This test is intended to clarify thermal-hydraulic response adverse to core cooling during a small- break LOCA and operational transient with high core power because of failure of reactor scram.

Two LSTF experiments were intended to simulate two different IETs with well-simulated core power transient after the initiation of accident and transient, being coupled with the reactor-typical thermal-hydraulic responses; pressure, core power and corresponding steam generation rate and resultant flow transient including natural circulation with high flow rates of steam and liquid.

Test 4 Natural circulation with superheated steam

This test is intended to experimentally indicate the natural circulation with superheated steam during a severe accident that may cause containment bypass in case of the rupture of steam generator (SG) U-tubes at high-pressure sequence.

Two LSTF experiments were intended to simulate the natural circulation including gas-gas counter-current flows under the reactor-typical conditions; pressure, temperature, temperature difference between PV upper plenum and SG secondary side, and flow transient in realistically shaped hot leg and SG.

Test 5 Primary cooling through SG secondary depressurization

This test is intended to clarify thermal-hydraulic response in LOCAs, especially for the long-term core cooling at low system pressures by natural circulation induced by the SG secondary depressurization under influences of non-condensable gas from accumulators.

Two LSTF experiments were intended to simulate IETs through the transient from the initial steady condition at the nominal operating condition for pressurized water reactor (PWR) to the long-term core cooling at near-atmospheric pressure, with and without the influences of non-condensable gas migration into SG U-tubes.

Test 6 Open subjects

This test is intended to fulfill the requests from the project participants by means of three LSTF experiments that respectively simulated two types of small-break LOCA with the break at the top of upper-head or at the bottom of the PV, and one local phenomena as direct steam condensation on ECCS coolant in cold legs during the reflood phase in a large-break LOCA of PWR.

Three LSTF experiments were intended to perform two IETs (**Tests 6-1 and 6-2**) that simulate LOCAs from the nominal operating condition of PWR with an accident management (AM) measure, and a series of SETs (**Test 6-3**) under prototype steam flow conditions; high flow rate and large superheating at near-atmospheric pressure.

The twelve LSTF experiments above were intended to provide data for the development, validation and verification of computer codes and models, aiming to increase the level of details and accuracy in the analyses of the key phenomena during transients and accidents of interest.

To meet this objective, system modifications, addition of new instrumentation and/or preparation of new test sections were performed for the ROSA/LSTF, which are mainly composed of the following;

(1) thermocouple arrays in cold legs and downcomer, hot legs and SG inlet plenum

(2) non-condensable gas detectors at SG outlet plenum and PV upper-head

(3) gas velocity meters in hot legs

(4) glass-fiber type video probes for visual observation of flows in cold legs

(5) nuclear data for the preparatory PWR analyses to simulate LOCA and anticipated transient, both without scram

(6) test section for condensation-induced water hammer (CIWH) with fast-response pressure sensors, a pitot tube and a gamma-ray densitometer

(7) test section for direct steam condensation on ECCS water downstream of break unit for the cold leg (horizontal orientation) in the loop without pressurizer

Measurement instrumentation (2), (3) and (4) were developed by JAEA for the LSTF utilization and applied to the corresponding ROSA Project experiments. (1), (2), (3), (4), (6) and (7) are presented in **Appendix 1**.

References

[1-1] OECD/NEA, 2001. Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk (SESAR/FAP). OECD/NEA, Nuclear Safety, ISBN 92-64-18463-5.

[1-2] OECD/NEA, 2007. Nuclear Safety Research in OECD Countries: Support Facilities for Existing and Advanced Reactors (SESAR/SFEAR). Nuclear Safety NEA/CSNI/R 6, ISBN 978-92-64-99005-0.

[1-3] The ROSA-V Group, 2003. ROSA-V Large Scale Test Facility (LSTF) System Description for The Third and Fourth Simulated Fuel Assemblies. JAERI-Tech 2003-037, Japan Atomic Energy Research Institute.

Table 1-1 ROSA/LSTF Experiments for OECD/NEA ROSA Project

Test	IET or SET[1)]	Subjects	Safety Significances[2)]	Data Utilization[3)]	Additional Specific Features (New Instrumentation, Facility Modifications and/or Preparatory Analysis)
1		Temperature stratification and coolant mixing in cold legs and downcomer during ECCS coolant injection	• Clarify temperature stratification conditions important for evaluation of pressurized thermal shock (PTS)	BE & CFD	• Array of thermocouples to cold legs for detailed measurement of temperature distribution • Video Probe to cold legs for visual observation of flows
1-1	SET	Temperature stratification under quasi-steady condition during natural circulation			
1-2	IET	Temperature stratification under transient during hot leg small break LOCA (1% cold leg break equivalent)			
2	SET	Unstable and destructive phenomena such as water hammer	• Clarify onset conditions of and intensity of pressure pulses due to condensation-induced water hammer (CIWH) that may threaten the integrity of reactor structures under pressures typical of LWRs	BE & CFD	• New test section with 67 mm i.d. and 2050 mm in length for well-defined boundary conditions • Fast response pressure sensors • Array of thermocouples • Pitot tube to measure steam velocity (failed) • Gamma-ray densitometer • Pre-test analyses by RELAP5 code to expect flow conditions
3		Natural circulation under high core power conditions	• Clarify thermal-hydraulic responses adverse to core cooling during both accident and abnormal transient under high core power condition due to failure of reactor scram	BE & CFD	• Pre-test analyses on PWR and LSTF to define core power curve respectively by SKETCH-INS/TRAC-PF1 and RELAP5/MOD3.2 codes • LSTF preparatory tests to define break size and core power curve
3-1	IET	1% cold leg break LOCA without scram			
3-2	IET	Loss-of-feedwater (LOFW) transient without scram with auxiliary feedwater injection			
4		Natural circulation with superheated steam during high pressure sequence of severe accident	• Clarify onset conditions of superheated steam natural circulation that may cause bypass sequence during severe accident due to rupture of SG U-tubes • Measure low-velocity gas flows during natural circulation, which were scarcely done due to difficulty in measurement	BE & CFD	• Pre-test analysis with FLUENT code to define test conditions • Array of thermocouples to hot legs and SG inlet plenum for detailed measurement of gas temperature distribution • JAEA-developed pulsed-wire velocity meter for measurement of slow gas flow
4-1	SET	Natural circulation of argon simulating superheated steam			
4-2	SET	Natural circulation of superheated steam			

Table 1-1 ROSA/LSTF Experiments for OECD/NEA ROSA Project (continued)

Test	IET or SET[1]	Subjects	Safety Significances[2]	Data Utilization[3]	Additional Specific Features (New Instrumentation, Facility Modifications and/or Preparatory Analysis)
5		Primary cooling through SG secondary depressurization as accident management (AM) measure	● Clarify thermal-hydraulic responses, especially for long-term core cooling by natural circulation at low system pressures under influences of non-condensable gas ● Confirm effectiveness of AM measure	BE	● JAEA-developed non-condensable gas sensor to measure gas concentration in SG outlet plenum and PV upper-head ● RELAP5/MOD3.2 code analyses on PWR and LSTF to define break size and onset timing of AM action
5-1	IET	0.5% cold leg break LOCA without non-condensable gas inflow from accumulator			
5-2	IET	0.5% cold leg break LOCA with non-condensable gas inflow from accumulator tanks			
6		Open subject (requests from participants)			
6-1	IET	Pressure vessel upper-head break LOCA with symptom oriented AM measure (1.9% cold leg break equivalent)	● Clarify consequences of SBLOCA and effectiveness of AM measure ● Investigate responses of core exit thermocouples (CETs) during core boil-off	BE & CFD	● RELAP5/MOD3.2 code analyses on PWR and LSTF to define break size and to study influences and onset timing of AM action
6-2	IET	Pressure vessel bottom break LOCA and AM effectiveness (0.1% cold leg break equivalent)	● Clarify consequences of SBLOCA and effectiveness of AM measure	BE	
6-3	SET	Steam condensation on ECCS water at reflood phase during large-break LOCA (LBLOCA)	● Clarify conditions of steam direct-contact condensation on ECCS coolant during PWR LBLOCA that may impact on containment pressure ● Obtain steam condensation data under typical LBLOCA condition without influences of non-condensable gas, which was scarce due to difficulty in measurement	BE & CFD	● New test section to simulate flow in cold leg under well-defined boundary conditions ● High-speed video for visual observation of flows ● Array of thermocouples ● Pitot tube to measure steam velocity ● Gamma-ray densitometer ● RELAP5/MOD3.2 code analyses on PWR and LSTF to define velocity and temperature of steam at test section inlet

1) IET: Integral-effect test, SET: Separate-effect test

2) Common to all the tests: Provide LSTF database for verification and development of BE (and/or CFD) codes

3) BE: best estimate codes, CFD: computational fluid dynamics codes

2. ROSA/LSTF

2.1 Facility Outline and Unique Design Features

The LSTF (Large Scale Test Facility) [2-1] shown in **Fig. 2-1** is a full-pressure and full-height integral test facility constructed in 1985 to investigate thermal-hydraulic response during PWR transients and accidents for the fourth phase of ROSA (Rig-of-Safety Assessment, ROSA-IV) Program at JAEA (Japan Atomic Energy Agency). The LSTF simulates a 1100 MWe four-loop Westinghouse (W)-type PWR: Tsuruga Unit-2 reactor of the JAPC (Japan Atomic Power Company).

The Three Mile Island Unit-2 (TMI-2) reactor accident in 1979 led to a reorientation of the LWR safety research to pursue SBLOCAs and operational/abnormal transients. Of primary importance for the LSTF was then a good simulation of the natural circulation in the primary loops involving two-phase flows and counter-current flows. The major design characteristics of the LSTF are summarized in **Table 2-1**.

Table 2-1 Major Design Characteristics of LSTF to Reference PWR

Items	LSTF	PWR	PWR/LSTF
Primary/Secondary Pressures (MPa)	16 / 7.4	16 / 6.13	1 / 0.83
Primary/Secondary Temperatures (K)	598 / 562	598 / 550	1 / 0.98
Core Height (m)	3.66	3.66	1
Number of Fuel Rods	1008	50952	50.55
Primary Fluid Volume V (m^3)	8.14	347	42.6
Total Core Power Q (MW)	10	3423(t)	342
Q/V (MW/m^3)	1.23	8.8	8.0
Core Inlet Flow (ton/s)	0.0488	16.7	342
Pressure Vessel Downcomer Gap (m)	0.053	0.26	4.91
Number of Primary Loops	2	4	2
Hot LegInner Diameter D (m)	0.207	0.737	3.56
Length L (m)	3.69	6.99	1.89
$L/\sqrt{D}$ ($m^{1/2}$)	8.11	8.14	1.0
Volume $\frac{\pi}{4}D^2L$ (m^3)	0.124	2.98	24.0
Number of Tubes in Steam Generator (SG)	141	3382	24.0
Average Length of SG Tubes (m)	20.2	20.2	1

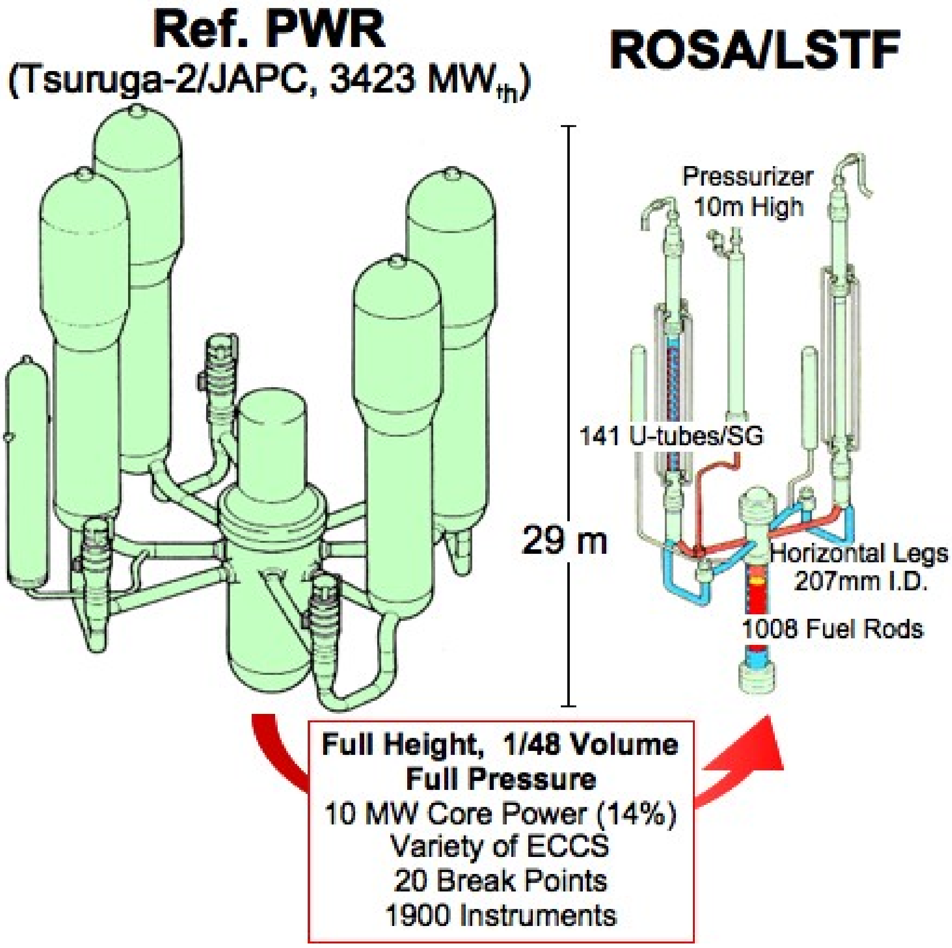

Fig. 2-1 Large Scale Test Facility (LSTF) of Rig-of-Safety Assessment (ROSA) Program in JAEA

The ROSA/LSTF is the world largest integral test facility for W-type PWR. Multi-dimensional thermal-hydraulic response during reactor transient is simulated by using a 10 MW electrically heated full size core rod bundle with more than 1000 simulated fuel rods, about 50 mm width annulus downcomer in the pressure vessel, 207 mm inner diameter hot and cold legs, 141 full size U-tubes in each of two steam generator.

The four primary loops of the reference PWR are represented by two equal-volume loops to best simulate two-phase flows during reactor accidents and transients mainly by achieving the large-diameter horizontal legs. The volumetric scaling ratio of the primary loops is 1/48 to those of the reference PWR. The LSTF can handle wide ranges of primary and secondary pressures from reactor nominal operating pressure to atmospheric pressure.

The LSTF is equipped with all the ECCS (emergency core cooling system) with additional new features such as ECCS coolant temperature control, temperature stratification in accumulator coolant tank to inject coolant by flashing of high temperature coolant portion without N_2 gas pressurization, though these features were not used in the OECD/NEA ROSA Project.

Instrumentation around 1900 channels provide detailed information on thermal-hydraulic conditions by measuring such parameters as temperature, liquid level, pressure loss, flow rate, density. Visual observation of flow is possible using “video-probe”; a periscope that withstands high-temperature steam/water conditions employing glass-fiber image guide or bore scopes. Non-condensable gas detectors and detailed temperature measurement was furnished further to the LSTF in the OECD/NEA ROSA Project.

Since the shakedown test in 1985, the LSTF has provided experimental data including the 5% cold-leg LOCA simulation for OECD/NEA ISP-26, the Mihama Unit-2 SGTR (steam generator tube rupture) simulation for the Nuclear Safety Commission of Japan and the AP600 reactor simulation for the USNRC. Throughout its operational history, the facility has demonstrated excellent experimental capabilities and has provided unique data for non-equilibrium, non-homogeneous and multi-dimensional phenomena.

2.2 Design Philosophy and Scaling of Major Components

Design philosophy and scaling of major components of the LSTF are summarized as follows:

Design Pressure covers that in the reference PWR. The rated pressure of LSTF pressure vessel is 17.26 MPa. The experiments can thus be performed under full-pressure conditions to simulate the thermal-hydraulic phenomena in the reference PWR via corresponding fluid properties.

Core Power is limited to 10 MWt, which is scaled by 1/48 at core power equal to and less than 14% of the scaled rated power of the reference PWR.

Elevations are preserved, i.e., one to one correspondence with the reference PWR, considering good representation of the driving force of natural circulation. As for the horizontal legs, pipe inner diameter (207 mm) is far smaller than that of the reference PWR (737 mm). The top elevation of hot leg inner diameter is thus set equal to that of the reference PWR. As for the cold legs, elevation of leg horizontal axis is set equal to that of the reference PWR.

Volumes of major components such as pressure vessel (PV) and pressurizer are scaled by 1/48 to the reference PWR. Since two loops are lumped into one, the volume of SG is twice as much as scaled one. The scaling factor is about 1/21 to the two-loop PWR such as Mihama Unit-2.

Flow Area is scaled by 1/48 in the pressure vessel and by 1/24 in the primary loops and SGs. The flow area in the horizontal legs is scaled to conserve the ratio of the length L to the square-root of pipe diameter D; Error! Objects cannot be created from editing field codes. of the reference PWR to better simulate the flow regime transitions in the primary loops (Froude number basis). The time scale of simulated thermal-hydraulic phenomena is thus one to one to those in the reference PWR.

Fuel Assembly has mostly the same dimensions for the following points as those of PWR 17x17 fuel assembly; diameter, length and pitch of fuel rod, diameter, length and pitch of control rod guide thimble, and the ratio of number of fuel rods to number of the guide thimbles. This preserves the heat transfer characteristics of the core. The total number of the simulated fuel rods is scaled by about 1/48. There are 1008 heated rods in the current fourth fuel assembly.

Steam Generators are designed to simulate primary-to-secondary thermal interactions during most of SBLOCAs or operational transients. The rated 10 MWt power sets the maximum steady-state steam and feedwater flow requirements to 14% of the scaled flow required for the reference PWR operating at full power. The secondary pressure is elevated at the initial steady state operation to suppress and control the heat transfer with the scaled heat transfer area of U-tubes and 14% primary loop flow rate. U-tube bundles have mostly the same dimensions with those of the reference PWR.

Flow Capacities are scaled by 1/48 where practical.

Pressure Loss due to Fluid Flow is designed to be equal to that in the reference PWR for scaled flow rates.

Break in the reference PWR is simulated by using a break unit that can be connected at 19 locations of the LSTF. Break size is controlled by orifice or nozzle. The tested maximum size of break has been 10% cold leg break equivalent for the primary loop.

ECCS is designed with adjustable operational envelopes that cover capacity at the normal envelope and beyond. Piping and nozzles are provided so that ECCS fluid can be injected into several possible locations around the primary coolant system other than those typical to the reference PWR for some parameter investigations. All types of ECCS furnished to the reference PWR are provided; high-pressure injection system (HPIS), accumulator injection system (AIS), low-pressure injection system (LPIS) and residual heat removal (RHR) system. The LSTF is equipped with a gravity-driven injection system (GDIS) as a kind of passive safety feature, though it was not used in the OECD/NEA ROSA Project.

Process Control System includes logic that considers the following four experimental needs;

a) A computer controlled sequence control program drives steady-state and transient events. Constants vary according to demands for each experiment.

b) The interlock program incorporates special control features of the reference PWR for facility equipment trip control.

c) The component control programs contains logic specific to key individual component control such as core power decay, pressurizer heater control and pump rotation increase/coastdown.

d) The facility protection control system governs the equipment safety interlocks such as core over-temperature protection.

Instrumentation are designed

a) to simulate the process instrumentation of the reference PWR to control the LSTF system during SBLOCA and operational transient simulation experiments,

b) to measure thermal-hydraulic phenomena as precisely as possible, and

c) to develop new instruments for advanced measurements in need.

As for the point b), three-dimensional fluid behaviors are to be detected by a combination of thermocouples, water level measurements and gamma-ray densitometers with aid of visual observation by the video probe at hot and cold legs. As for the point c), non-condensable gas detectors and low-velocity gas velocity meter were developed in JAEA for the OECD/NEA ROSA Project.

References

[2-1] The ROSA-V Group, 2003. ROSA-V Large Scale Test Facility (LSTF) System Description for The Third and Fourth Simulated Fuel Assemblies. JAERI-Tech 2003-037, Japan Atomic Energy Research Institute.

3. TEST DESCRIPTION AND MAJOR RESULTS

3.1 Test 1 Temperature stratification during ECCS coolant injection

3.1.1 Background

During a small-break loss-of-coolant accident (LOCA) of pressurized water reactor (PWR), the emergency core cooling system (ECCS) is actuated and cold water is injected into cold legs. Injected water mixes with hot primary coolant and flows into the pressure vessel through the downcomer. Some of injected water may flow at the bottom of the cold leg, forming a cold layer less mixed with the upper hot layer. Insufficient mixing of cold water and hot coolant results in temperature stratification. In case that liquid level forms in the cold leg, steam condensation on injected water occurs, and local pressure variations may cause unstable flow oscillations, which in turn promote coolant mixing. Such multi-dimensional and non-equilibrium flow phenomena shown in **Fig. 3.1-1** are of concern for pressurized thermal shock (PTS) in view of aging and plant life extension. Experiments for thermal hydraulic phenomena pertaining to PTS, are summarized in several project reports [3.1-1~3.1-3].

Computational fluid dynamics (CFD) is a powerful tool for predicting various flow phenomena especially in single-phase flows. Some mixing phenomena have been simulated using commercial and in-house CFD codes relating to boron dilution and PTS scenarios [3.1-4~3.1-7], and two-phase cases have also been reported [3.1-8]. Applications of CFD codes to analyze such complicated nuclear safety problems as the temperature stratification phenomena are of interest especially under two-phase flow conditions [3.1-9]. Although some experimental results have been used for validation of CFD codes [3.1-2, 3.1-3], experimental data obtained under high-pressure and high-temperature single- and two-phase flow conditions corresponding to PWR are limited.

3.1.2 Objectives

The objective of Test series 1 was to obtain the three-dimensional temperature distributions in cold legs and downcomer during ECCS coolant injection under PWR conditions for verification of computer codes and simulation models. Two types of experiments were conducted: steady-state and transient tests. In **Test 1-1**, the ECCS coolant was injected into the cold legs under steady-state single- and two-phase natural circulation conditions, and the steady-state mixing phenomena under controlled boundary conditions were observed. These data would be suitable for an assessment of turbulent mixing and condensation models. A hot-leg small-break LOCA was simulated in **Test 1-2** to obtain the temperature distribution under transient conditions with actuation of high-pressure and accumulator injection systems. These data would be suitable for an assessment of simulation codes for various types of mixing phenomena during LOCAs with system responses under influences of ECCS coolant injection.

3.1.3 Description of Experiments

Test 1-1 (ST-NC-34 in JAEA) was conducted as a separate-effect experiment. The primary and secondary pressures for the single-phase natural circulation condition were 15.5 and 6.7 MPa, respectively, with the core power of 2% scaled nominal value and the full primary mass inventory. The two-phase natural circulation was established at the primary and secondary pressures of about 6.7 MPa with the same core power but reduced inventories. The ECCS injection rates were 0.3 and 1.0 kg/s for both the cold-legs A and B. The injection duration was 80 s. The injections with the rate of 1.8 kg/s were additionally performed for the cold-leg B under the two-phase condition.

Test 1-2 (SB-HL-17 in JAEA) was conducted simulating a 1% hot-leg small-break LOCA. The high pressure and accumulator injection systems were actuated at the specified operational set points. The flow rate of ECCS was a half of the scaled value since a single failure of safety system was assumed.

For Test series 1, 144 thermocouples were newly installed in cold legs and downcomer to obtain the three-dimensional temperature distribution data. New video probes were also installed in the cold legs, and the flow phenomena during ECCS coolant injection were observed visually.

3.1.4 Experiment Results

Two types of ECCS injection tests were performed in Test series 1: steady-state natural circulation condition and transient LOCA condition. The temperature stratification in cold legs was seen widely in both tests under both single- and two-phase natural circulation conditions. The degree of temperature stratification was larger in the cold-leg A, since the shape and the location of the ECCS injection nozzles are different between two loops in the LSTF. It was found that the cold water from the cold leg flowed down in the downcomer along the inner wall of the pressure vessel under the natural circulation condition with the downcomer liquid level higher than the cold leg (**Fig. 3.1-2(a)(b)**), while it flowed down along the outer wall of the core barrel cover when the loop flow almost stopped (**Fig. 3.1-3**) or the downcomer liquid level was lower than the cold leg (**Fig. 3.1-2(b)**). Besides the temperature variation and distribution in the cold legs and downcomer, the injection of ECCS coolant with the variation of liquid level was observed visually. These experimental data would be useful for evaluation and validation of both the CFD and BE codes and models: Test 1-1 is for simulation models including turbulent mixing and condensation [3.1-10, 3.1-11], and Test 1-2 is for transient system analyses with PTS estimation under actual LOCA conditions.

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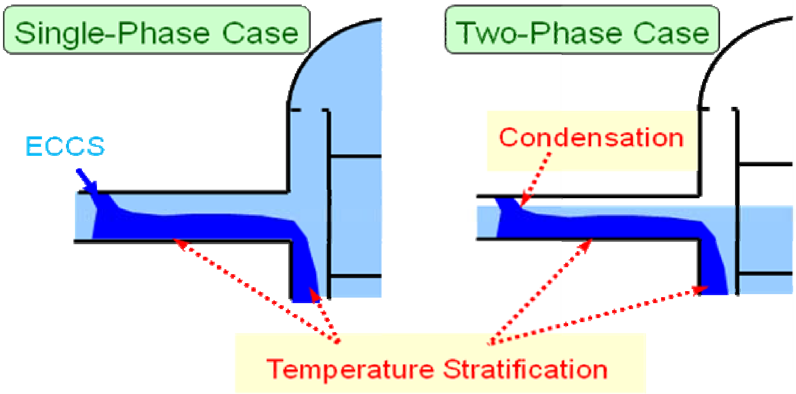

Fig. 3.1-1 Schematic diagram of temperature stratification phenomena in Test 1

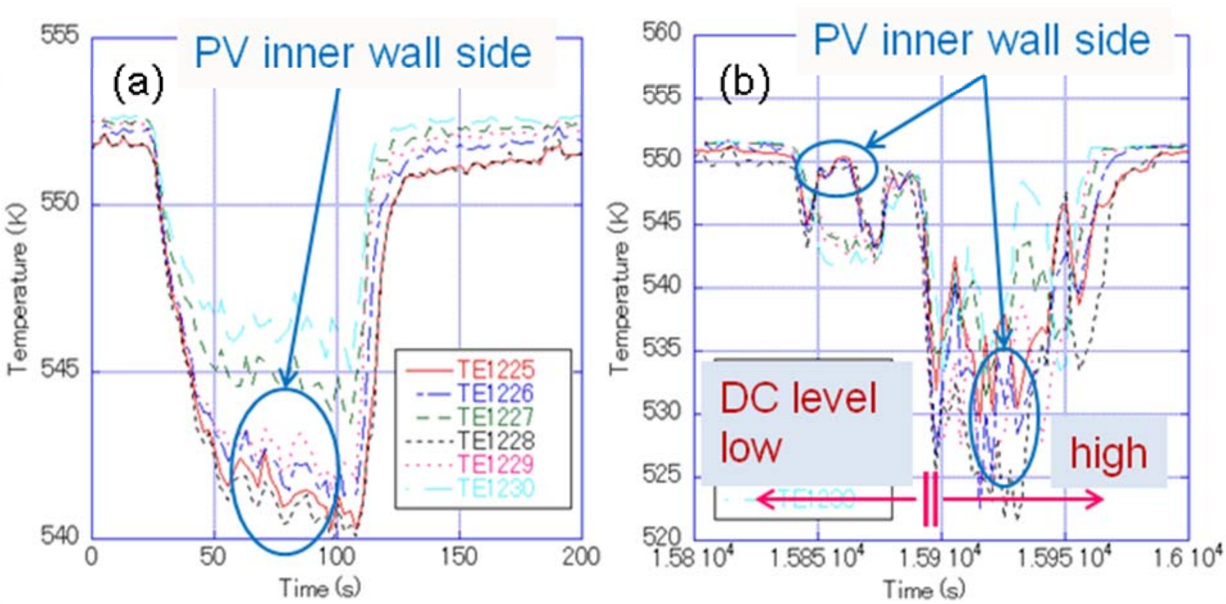

Fig. 3.1-2 Downcomer temperature in Test 1-1: (a) single phase case (b) two-phase case

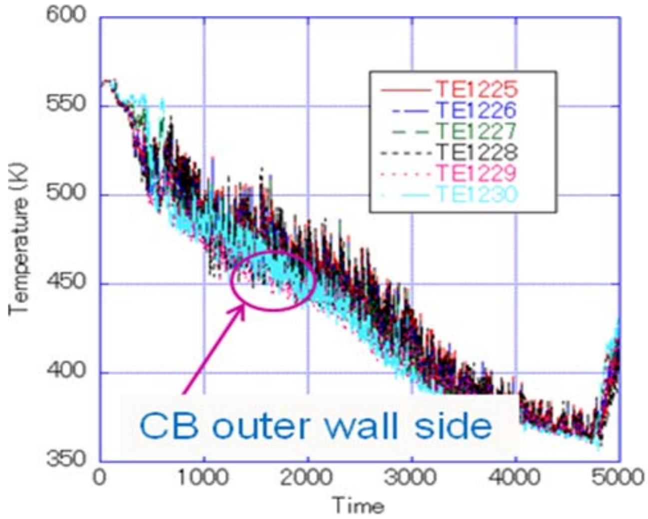

Fig. 3.1-3 Downcomer temperature in Test 1-2

3.2 Test 2 Unstable and destructive phenomena such as water hammer

3.2.1 Background

Condensation-induced water hammer (CIWH) may occur when highly subcooled water contacts directly with steam, resulting possibly in failures of the reactor components and structures such as branch piping due to the generation of large mechanical energy. Thus, data to describe characteristics of CIWH have been obtained through many studies. However, those studies focused mainly on CIWH under low pressure conditions because of large shrinkage of steam volume. Low ambient pressure appears in the course of an accident management for the depressurization of the primary system to activate ECCS such as accumulator and low-pressure injection (LPI) systems. CIWH may also occur in some passive safety reactors, in which significant depressurization of the primary system is performed to enable gravity water injection from the in-containment refuelling water storage tank (IRWST).

Though higher ambient pressure is supposed to tend to suppress the onset of CIWH due to higher steam density, it makes subcooling of the injected ECCS water larger, which is preferable for the onset of CIWH. That is, under high pressure conditions, there could be a competition between contradictory effects of pressure and water subcooling. In order to evaluate the influence of CIWH on the integrity of the surrounding components and structures, it is crucial to clarify the onset criteria and the intensity of CIWH under a wide range of pressure and subcooling conditions.

3.2.2 Objectives

The major objectives of Test 2 are to obtain detailed transient thermal-hydraulic data concerning CIWH in a horizontal pipe through a series of separate-effect tests, and to clarify the onset criteria and the intensity of CIWH. In particular, the effect of system pressure up to approximately 7 MPa in combination with water subcooling is studied. The obtained data could be suitable for the verification of computer codes and models for the prediction of CIWH.

3.2.3 Description of Experiments

Separate-effect tests (ST-WH-05, 06, 07, 08, 09, 10 and 11 in JAEA) were performed under well defined boundary conditions with a newly furnished horizontal pipe test section that was made of approximately 2.3 m-long stainless steel pipe with inner diameter of 66.9 mm [3.2-1]. The primary parameters were system pressure, water subcooling and liquid level (local void fraction) in the test section. One end of the test section was connected to the pressure vessel (PV) downcomer of LSTF. The other end was closed using a sealing plate. Room temperature water in the LSTF refueling water storage tank (RWST) was injected to the bottom of the test section near the closed end using the LSTF high pressure injection (HPI) system. The injected water is discharged to the LSTF downcomer.

Fast response sensors were used for the measurement of CIWH pressure pulses generated and propagated below water level in the test section. Many thermocouples were installed in the test section to estimate the liquid level, temperature stratification in water layer and the passage of water slugs. A gamma-ray densitometer was also employed to measure the liquid level.

The test procedures are schematically illustrated in **Fig. 3.2-1**. The liquid level in the LSTF downcomer was kept higher than the top surface of the test section during the initial state. Water at room temperature was then supplied to the test section at a specified injection rate. Immediately after this operation, the water in the PV of LSTF was gradually discharged from the PV bottom. During the test phase, the liquid level was developed in the test section by the water discharge operation and saturated steam entered into the test section from the PV. This situation may resemble a two-phase flow encountered in the ECCS injection line after the injection rate decreased.

3.2.4 Experiment Results

Through a series of separate-effect tests, it was confirmed that CIWH occurred even at high ambient pressure of approximately 7 MPa. The following insights were obtained for the characteristics of CIWH under a various combination of system pressure and water subcooling.

- ✓ The averaged intensity of pressure pulses in each CIWH test increased with the system pressure when it was less than 1 MPa, and decreased in the case of higher system pressure as shown in **Fig. 3.2-2 (a)**.
  - ➢ The maximum intensity was much larger than the average intensity. A pressure pulse whose intensity was larger than 10 MPa formed even at the system pressure of 7 MPa as plotted in **Fig. 3.2-2 (b)**, although the averaged intensity was less than 1 MPa.
  - ➢ When the reflection wave of a pressure pulse and the original wave were superimposed, the intensity became almost twice as much as that of the original wave.
  - ➢ The water injection rate seemed to affect the dependency of the pressure pulse intensity on the system pressure. The effect of the water level at the outlet of the test section on the intensity was small.
  - ➢ It was confirmed that the formation of the intermittent flow such as a slug flow was related to the onset of CIWH.
  - ➢ No thermal stratification formed in the liquid phase during the period while the CIWH was induced. The effective mixing of the condensate with the subcooled water may have resulted in the large condensation rate.
  - ➢ The steam mass flow rate estimated using mass and energy balances in the test section monotonically increased with the system pressure up to 4 MPa. The mass flow rate became almost insensitive to the system pressure when it was larger than 4 MPa.
  - ➢ The estimated superficial steam velocity was related to the averaged intensity of pressure pulses, probably because the steam velocity is proportional to the steam condensation rate. This observation suggested that the large volumetric shrinkage induced by the significant condensation rate produced the large pressure pulse.

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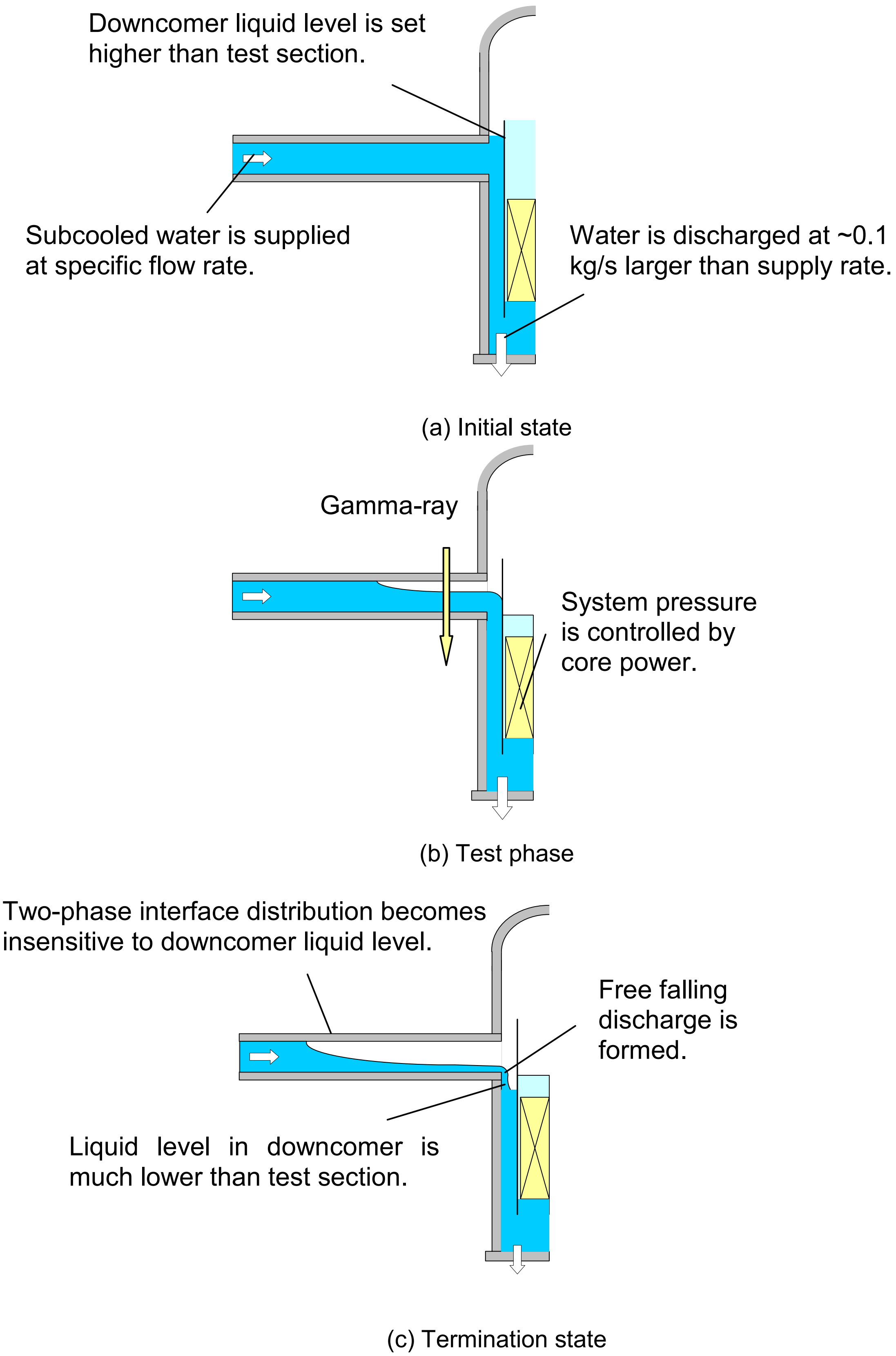
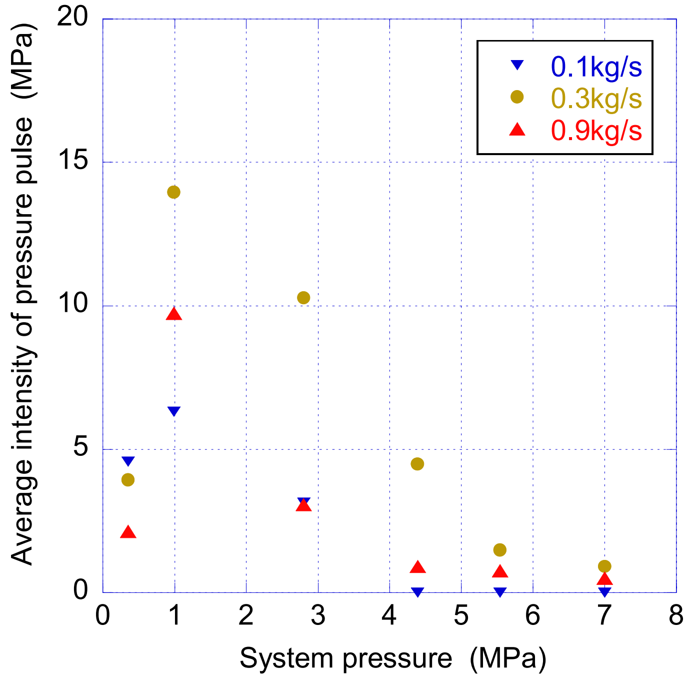
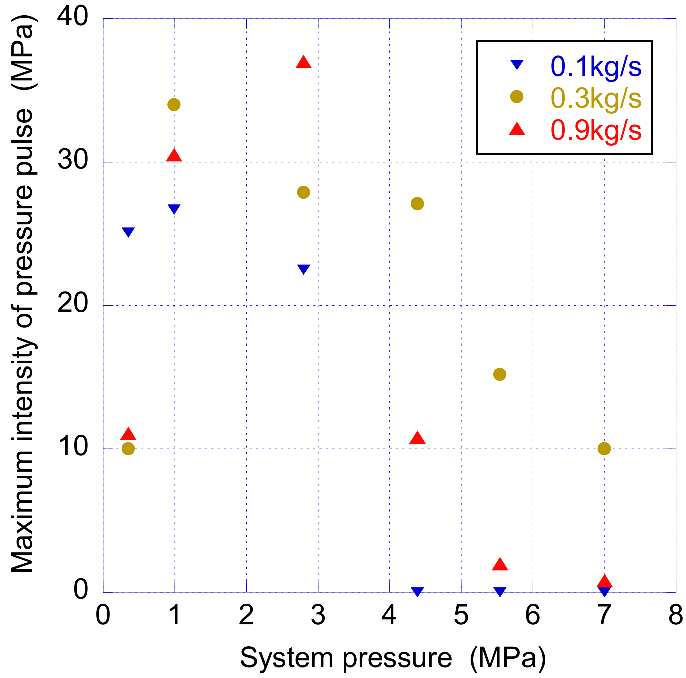


Fig. 3.2-1 Schematic diagram for test procedures



(a) Averaged intensity



(b) Maximum intensity

Fig. 3.2-2 Influence of system pressure on CIWH intensity

3.3 Test 3 Natural circulation under high core power conditions

The high reliability of control rods results in relatively low risk for anticipated transient without scram (ATWS) of PWR. Failure of scram, however, would lead to relatively high core power for a long time and degradation in core cooling especially when coolant mass inventory decreases. The thermal-hydraulic data on the natural circulation under high core power, however, is scarce, especially for the data taken from integral tests simulating accidents and anticipated transients where the primary coolant inventory decreases. The data obtained through the ROSA/LSTF experiments would then be valuable to understand the natural circulation phenomena during the high core power transient and for the validation of the safety analysis codes.

Based on the motivation above and discussions among the project participants, the following two tests were performed by using ROSA/LSTF. These tests may include natural circulation under high core power conditions and several thermal-hydraulic phenomena associated with the natural circulation. The primary inventory of the both tests decreases, though the mass decrease rate differs much depending on the size and location of the opening (break) on the primary pressure boundary.

Test 3-1 Small-break loss-of-coolant accident (LOCA) without scram

Test 3-2 Loss-of-feedwater (LOFW) transient without scram

Background, objectives, description and results are explained for each of the two tests in the following sub-sections.

3.3.1 Test 3-1 Small-break LOCA without scram

3.3.1.1 Background

Thermal-hydraulic phenomena during small-break LOCA (SBLOCA) without scram may include high-power natural circulation with large flow rate for both the steam and liquid phases in hot legs. The liquid flow is highly accelerated to be supercritical (Froude number (Fr) > 1.0) by high velocity steam at the entrance of hot legs where the liquid velocity becomes higher than the speed of a surface long-wave that depends on liquid depth [3.3-1, 3.3-2]. The hot leg liquid level then drops largely. The correct prediction of the transition into supercritical flow is important for reactor safety analysis, because the supercritical flow in the hot leg may affect liquid carryover and liquid accumulation condition at around the inlet plenum and U-tubes of steam generator (SG). Counter-current flow limiting (CCFL) may take place at the SG inlet plenum and at the bottom of SG U-tubes that may hold a large amount of coolant, as shown in **Fig. 3.3-1**. The core cooling condition would then be much degraded when the natural circulation turns into reflux cooling. Experimental data, however, have been scarcely obtained for such a SBLOCA transient without scram where the thermal-hydraulic phenomena change as the coolant mass inventory decreases with time.

3.3.1.2 Objectives

The objectives of Test 3-1 are to clarify thermal-hydraulic response adverse to core cooling under high core power condition due to failure of reactor scram during the SBLOCA transient. Obtained test data on the complicated coolant behaviors during natural circulation are suitable for the assessment of predictability of computer codes and models for system integral analyses.

3.3.1.3 Description of Experiment

Test 3-1 [3.3-3] was conducted under assumptions of total failure of high pressure injection (HPI) system and an actuation of auxiliary feedwater as well as loss of off-site power concurrent with the scram signal. Break size and core power curves were defined through discussions among the project participants based on the LSTF pre-test analyses with RELAP5 code [3.3-4] and PWR SBLOCA analyses with JAEA-developed coupled three-dimensional kinetics and thermal-hydraulics code SKETCH-INS/TRAC-PF1 [3.3-5] as well as LSTF preparatory experiments. Size of the break located at the side of cold leg in the loop without pressurizer (PZR) (loop-B) was 1% that corresponds to 2.8-inch break in the reference PWR. The core power curve was pre-determined based on the PWR SBLOCA analyses with a detailed core model, but the portion higher than 10 MW (14%) was cut off.

3.3.1.4 Experiment Results

Two-phase natural circulation took place under high core power, which caused SG relief valves (RVs) kept opened. Flow in hot legs became supercritical due to high velocity steam and liquid flows during the two-phase natural circulation. The hot leg liquid level then became quite low and affected liquid carryover into the SGs (**Fig. 3.3-1**). Liquid accumulation in the SG U-tube upflow-side and inlet plenum happened during reflux condensation mode probably because of CCFL at the inlet of the U-tubes and the bottom of the inlet plenum due to high steam velocity (**Fig. 3.3-1**). The liquid accumulation caused a small drop in the core liquid level. It, however, had no significant influences adverse to core cooling in this test. Steam condensation on accumulator coolant in cold legs induced loop seal clearing in the loop-B, but after the initiation of automatic core power decrease due to core temperature excursion during boil-off.

Consequently, the natural circulation under high core power contributed to maintain core cooling effectively for relatively long time until core uncovery, while several phenomena specific to the natural circulation under high core power such as supercritical flow and CCFL appeared in hot legs and SG inlet plenum and U-tubes, respectively.

3.3.2 Test 3-2 Loss-of-feedwater transient without scram

3.3.2.1 Background

Thermal-hydraulic phenomena during LOFW transient without scram include high-power natural circulation with liquid entrainment in hot leg at the surge line nozzle, and CCFL at the PZR bottom that may hold a large amount of coolant in the PZR, as shown in **Fig. 3.3-2**. In the transient following LOFW, PZR power-operated relief valve (PORV) may continue cycle opening, resulting in loss of primary coolant inventory. The core cooling condition would then be degraded especially after the natural circulation turns into reflux cooling. A LOFW-induced ATWS experiment performed in the LOFT (Loss of Fluid Test) program revealed that the primary pressure is kept at about 17.2 MPa by cycle opening of the PZR PORV and safety valve while the primary fluid temperature gradually increases [3.3-6]. Experimental data, however, have been scarcely obtained for such a LOFW transient without scram where the thermal-hydraulic phenomena change as the coolant mass inventory gradually decreases with time.

3.3.2.2 Objectives

The objectives of Test 3-2 are to clarify thermal-hydraulic phenomena adverse to core cooling under high core power condition due to failure of reactor scram during the LOFW transient. Obtained test data on the complicated coolant behaviors during natural circulation are suitable for the assessment of predictability of computer codes and models for system integral analyses.

3.3.2.3 Description of Experiment

Test 3-2 [3.3-7] was conducted under assumptions of total failure of HPI system and an actuation of auxiliary feedwater (AFW) as well as loss of off-site power concurrent with the scram signal. Onset timing of the AFW and the core power curve were defined through discussions among the project participants based on the LSTF pre-test analyses and PWR LOFW analyses with RELAP5 code. To avoid injection of cool AFW coolant onto high-temperature structures that may cause significant deformation of SG structures including U-tubes (bottom) due to thermal shock, AFW was initiated when the SG secondary-side collapsed liquid level decreased to about 0.5 m. The core power curve was pre-determined from volumetrically-scaled (1/48) power obtained from the PWR LOFW analyses with one-point neutron kinetics model, and the portion higher than 10 MW (14%) was cut off. The core power curve and AFW conditions are thus different from those in TMLB' scenario (station blackout with no recovery of AFW) [3.3-8].

3.3.2.4 Experiment Results

Core power was maintained at about 7.4% for more than 7 hours until the automatic power reduction due to high core temperature. AFW was actuated as planned, providing a continuous primary-to-secondary heat removal. The primary pressure was maintained at around 16 MPa by cycle opening of PZR PORV until the core uncovery started. The SG secondary-side pressure was kept at around 8 MPa by frequent cycle opening of the relief valves till the end of the test. The primary coolant inventory decreased gradually due to cycle opening of the PORV. Two-phase natural circulation started in very early stage of the transient. Liquid level decreased very slowly after the actuation of AFW but remained in the PZR because of CCFL at the PZR bottom due to high steam discharging rate through the PORV (**Fig. 3.3-2**). Significant level oscillation, in the form of a slow fill and dump, occurred in all the instrumented U-tubes of SGs with gradual decrease in the primary loop flow rate with some oscillation (**Fig. 3.3-2**). A water column developed in the SG U-tubes rather randomly and with a lower frequency than that of the cycle opening of the RVs and PORV. An intermittent increase in the fuel rod surface temperature preceded core boil-off and major temperature excursion in the reflux condensation mode, even when the liquid level was still in the upper plenum. The high core power may have caused departure from nucleate boiling (DNB). The experiment was terminated when the primary and SG secondary-side pressures reached nearly-equilibrium condition with well-cooled core after the automatic core power decrease started.

Consequently, various and complicated thermal-hydraulic responses appeared associated with the natural circulation under high core power, including inadequate core cooling in the reflux condensation mode.

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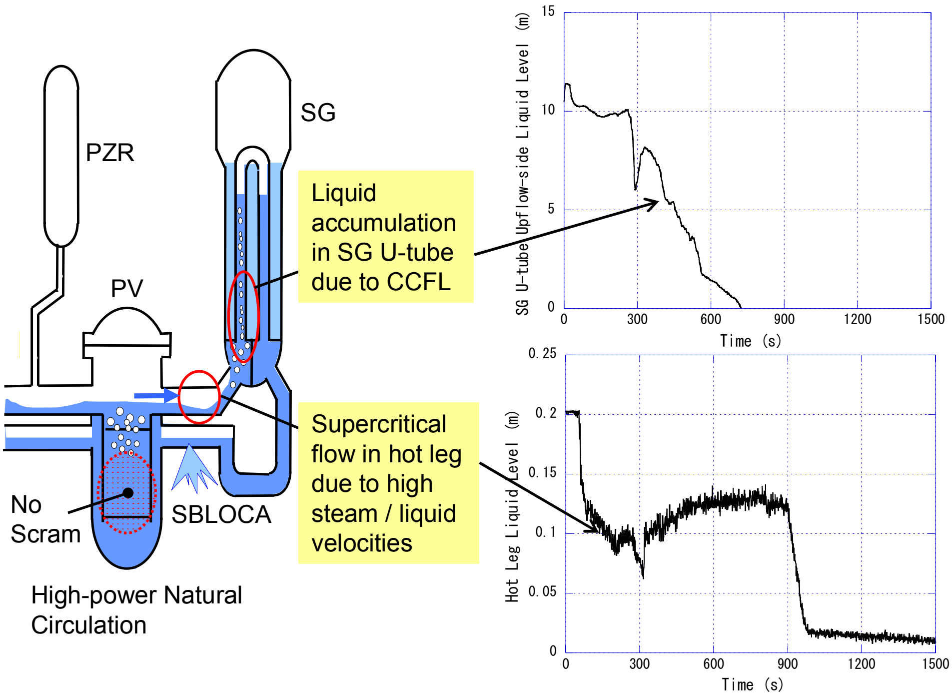

Fig. 3.3-1 Local phenomena during SBLOCA without scram in Test 3-1

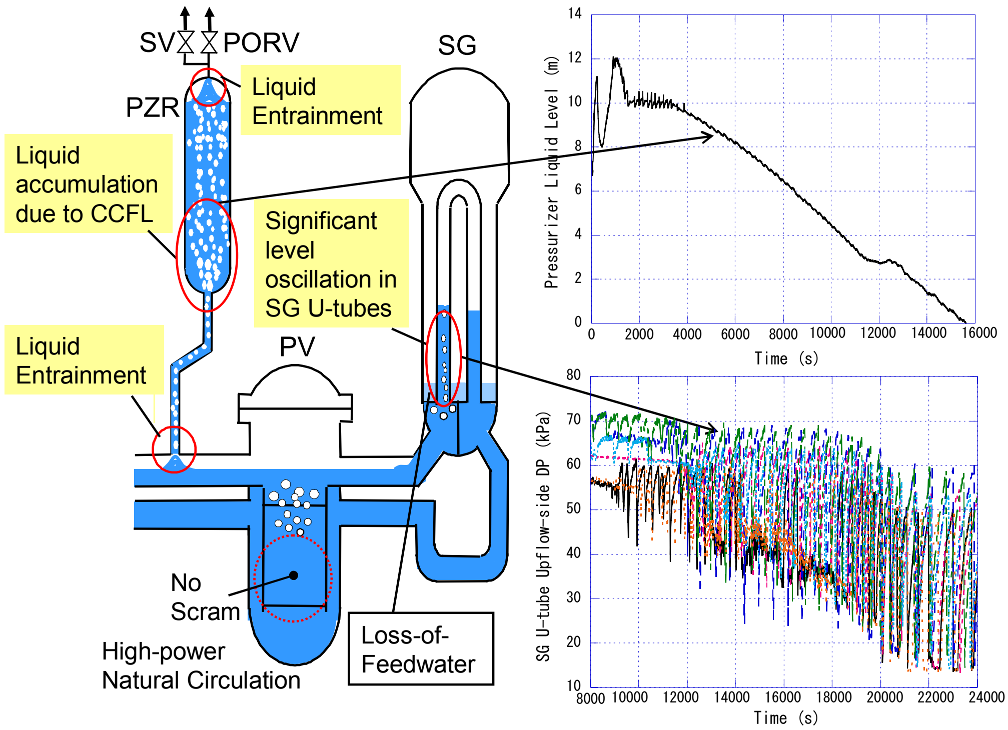

Fig. 3.3-2 Local phenomena during LOFW transient without scram in Test 3-2

3.4 Test 4 Natural circulation with superheated steam

3.4.1 Background

High pressure sequences of a PWR severe accident could involve natural circulation of highly superheated steam in the primary cooling system, which acts as a carrier of thermal energy generated in the degraded reactor core to the boundary structures. The natural circulation is characterized by the formation of a counter-current gas-gas flow in hot leg (HL), forward and backward flows in steam generator (SG) U-tubes and the mixing of hotter and colder gases in inlet plenum (IP) of SG.

The failure of reactor coolant piping in the primary system is concerned to occur due to high temperature degradation of piping materials in such a severe circumstance. Depending upon the flow characteristics, it is possible that the natural circulation allows high temperature superheated steam to enter SG U-tubes. In case that the natural circulation of superheated steam results in so-called severe accident-induced steam generator tube rupture (SGTR) so that containment bypass of radioactive materials occurs. Though the probability in occurrence of the severe accident-induced SGTR is low, the assessment of risk from this event should be made with a reasonably realistic manner since it may cause a serious consequence to the public and the environment [3.4-1, 3.4-2].

Thermo-fluid dynamics of such a natural circulation flow were investigated through past studies. A series of experiments was performed with a 1/7th scale SG model for Westinghouse PWRs [3.4-3]. An experiment was carried out with LSTF before the OECD-ROSA Project [3.4-4]. In those experiments, the spatial distribution of gas temperature was mainly examined. However, no data was obtained for gas flow velocity of the natural circulation. The quantification of the flow velocity is anticipated to be of importance to characterize the natural circulation and to aid the assessment of thermal loads onto the boundary structures.

3.4.2 Objectives

The present test series is composed of two quasi-steady separate-effect tests, Tests 4-1 and 4-2, to be performed by using LSTF with newly installed instrumentation for detailed measurement of gas temperature distribution and flow velocity especially at HLs and IP of SGs. The major objectives of the present test series are, considering the limitations of LSTF application to severe accident studies, to clarify important phenomena associated with the natural circulation of superheated steam under conditions as typical as possible and to provide a database for assessment of analytical tools including CFD codes.

3.4.3 Description of Experiments

Argon was applied as a simulated gas of the primary system in **Test 4-1** [3.4-5] at relatively low pressure (1.0 MPa) and superheated steam formed in the core of LSTF was used in **Test 4-2** [3.4-6] at higher pressure (4 MPa). The secondary side of SGs was filled with air and steam in Test 4-1 and Test 4-2, respectively. The procedures for both tests to satisfy specified test conditions including gas temperatures described below were deeply examined in advance. Those were confirmed to be appropriate by preliminary tests, resulting in successful performance of both tests as planned.

The dominant driving force for the natural circulation is density difference of gas, namely temperature difference of gas, between the inlet and outlet sides of SG U-tubes. Consequently, the most important condition is the difference in gas temperature between the primary and secondary systems since the variation of gas temperature in SG U-tubes is largely influenced by the primary to secondary heat transfer. This condition was defined through a series of pre-test analysis with FLUENT code (version 6.2) [3.4-7], which indicated that the gas temperature at the inlet of HL should be higher than that in the SG secondary side by at least 100 K. With this temperature difference as the representative for both tests, Rayleigh number (Ra) was estimated not to be far from that of a PWR based on MELCOR analysis ($\sim10^{15}$) when the gas temperature in the secondary system was assumed at 300 K for Test 4-1 and the saturation temperature at 4 MPa for Test 4-2.

The newly installed instrumentation includes rakes of thermocouples (TCs) and pulsed-wire velocity meters (see **Appendix 1** for concept and roles of measurement). The rakes of TC and the pulsed-wire velocity meters were positioned at pressure vessel (PV) and SG sides of each HL. Additional TC rakes were placed in IP of each SG. The pulsed-wire velocity meter is vertically traversable, and it has a structure of resistance thermometer wires arranged in series in the flow direction. The most upstream wire is electrically heated and others act as sensors to detect the arrival of the hot gas volume generated by the upstream heater wire. The gas velocity is evaluated from time difference of detection and distance between two wires. When natural thermal disturbances exist in a flow, all wires can be applied as sensors.

3.4.4 Experiment Results

The gas temperature in the vicinity of the HL inlet in the upper plenum of PV was higher than that in the SG secondary side by approximately 300 K in Test 4-1 and 150 K in Test 4-2, which was sufficiently larger than the specified target mentioned above.

The measurement using the pulsed-wire velocity meters without wire heating showed that the signals from two wires were correlated with a certain width of time difference, which was considered to correspond to the convection of thermal disturbances in a flow. Summarizing all signals, it was identified that forward flow from PV to SG and backward flow from SG to PV developed in the upper and the lower volumes of HL, respectively. There was also a tendency that the time difference was largest, that is, the gas velocity was lowest at around the center of HL. Roughly estimated gas velocity was approximately 0.3 m/s at maximum for both tests, which was comparable to the results of pre-test analysis with FLUENT code.

A clear temperature stratification: hotter gas in the upper volume and colder in the lower volume, developed in HL. In the upper volume of HL, a flat temperature profile formed at the vicinity of the inlet and the temperature decreased toward SG with changing profile to an inclined one. On the other hand, a flat temperature profile formed in the lower volume near the IP of SG and the temperature increased toward PV with similar change of profile. This feature in gas temperature profile could be related to the growth of thermal boundary layer and the mixing of gas under the counter-current gas-gas flow conditions.

No large spatial distribution of gas temperature was found in the IP of SG. This result implied that gas in the volume of the IP of SG was well mixed.

The gas temperature in SG U-tubes showed a decreasing profile from the inlet to the outlet in some of six instrumented U-tubes. The inversed profile or uniform profile in the remaining instrumented U-tubes. The temperature profile of gas indicated the co-existing of U-tubes with forward flow and those with backward flow.

As a summary, the findings obtained from the present test series are conceptually illustrated in **Fig. 3.4-1**. The natural circulation of superheated steam was confirmed to form in the primary loops of LSTF, including counter-current gas-gas flow in HL, co-existing of forward and backward flow U-tubes and the mixing of hotter and colder gases in IP of SG. There was no large qualitative difference between argon and steam cases except steam velocity at the top volume of HL. It is anticipated that detailed and quantitative data of the present test series is of sufficient value for the assessment of the predictive capability of analytical codes including CFD codes.

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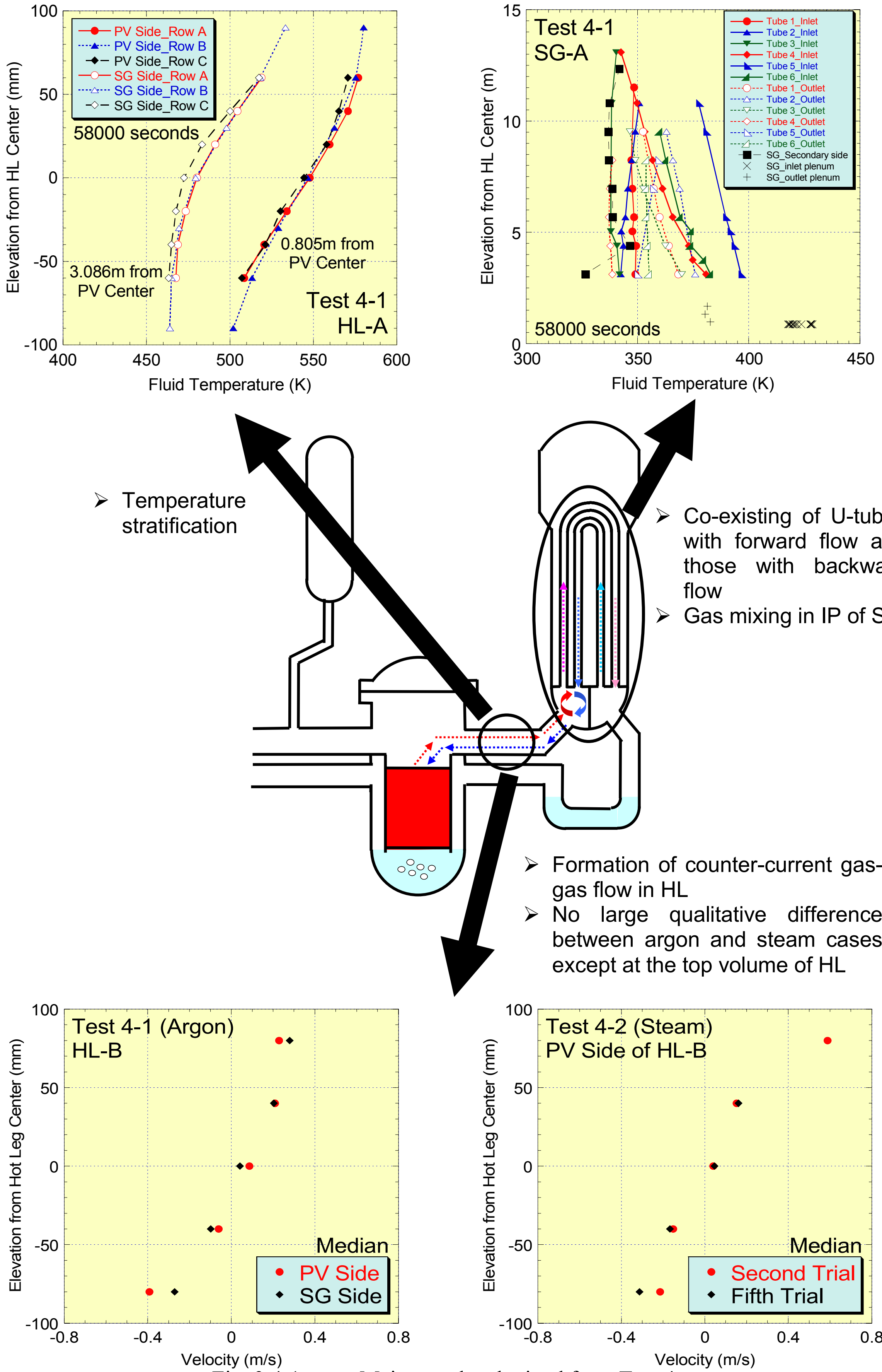

Fig. 3.4-1 Major results obtained from Test 4

3.5 Test 5 Primary cooling through SG secondary depressurization

3.5.1 Background

Steam generator (SG) secondary-side depressurization through the relief valves (RVs) is one of major accident management (AM) measures to cool and depressurize the primary loop via natural circulation during a small-break loss-of-coolant accident (SBLOCA) of current PWR. The establishment of prediction methods for natural circulation cooling at near-atmospheric pressure is essential to evaluate the effectiveness of such SG secondary-side depressurization for long-term core cooling in the current PWR and of gravity-driven injection in some future reactor design.

The primary cooling through the SG secondary-side depressurization would be influenced by such factors as coolant mass inventory, break flow, water injection from emergency core cooling system (ECCS) and presence of non-condensable gas. The LSTF experiments performed before OECD/NEA ROSA Project revealed that non-uniform flow occurs among SG U-tubes during single-phase liquid and two-phase natural circulation under influences of system pressure and coolant mass inventory [3.5-1~3.5-3].

The non-uniform flow is characterized by the coexistence of concurrent flow, contributing to heat transfer from the primary loop to the SG secondary-side, and single-phase reverse flow or stagnant two-phase stratified flow, as shown in **Figs. 3.5-1 and 3.5-2**. The effective heat transfer area decreases when the number of stagnant U-tubes increases especially at low pressures. It is difficult to properly predict such complex phenomena using computer codes. The experimental data is limited on the natural circulation cooling at low pressures under well-defined boundary conditions. It is thus essential to conduct simulation experiments on SG secondary-side depressurization with detailed observation of local phenomena such that the number of forward or reverse flow U-tubes may change with the coolant mass inventory.

3.5.2 Objectives

The objectives of Test 5 are to clarify thermal-hydraulic responses, especially for the long-term core cooling at low pressures by natural circulation under influences of non-condensable gas, and effectiveness of the AM measure. Obtained test data on the non-uniform coolant behaviors under influences of non-condensable gas are suitable for the assessment of predictability of computer codes and models for system integral analyses.

3.5.3 Description of Experiments

Tests 5-1 [3.5-4] and **5-2** [3.5-5] were conducted with no gas inflow and with gas inflow respectively from accumulator tanks under the assumptions of an actuation of auxiliary feedwater and total failure of high pressure injection (HPI) system as well as loss of off-site power concurrent with the scram signal. The size of break at the side of cold leg in the loop without pressurizer (PZR) (loop-B) was 0.5% that corresponds to 2-inch diameter break in the reference PWR. SG RVs were fully opened 10 minutes after the safety injection signal. Such boundary conditions as break size, AM action and its onset timing were defined through discussions among the project participants based on the LSTF pre-test analyses with RELAP5 code [3.5-6], focusing on the observation of two-phase natural circulation and non-uniform flow among the SG U-tubes. Further assumptions were made to conduct enhanced SG depressurization by fully opening the safety valves when the primary pressure decreased to 2 MPa and no actuation of low pressure injection system, both to well observe the natural circulation phenomena at low pressures.

A JAEA-developed gas measurement device including Zirconia sensor shown in **Appendix 1** was newly furnished in the LSTF to directly measure oxygen concentration in steam by introducing air as non-condensable gas for accumulator pressurization. Non-condensable gas in accumulator tanks enters cold legs first and migrates to hot legs through pressure vessel (PV) downcomer and hot leg leak simulating lines that connected the downcomer to the hot legs. Non-condensable gas that entered in SG U-tubes should have gathered at the SG outlet plena after some U-tubes become empty of liquid. The gas measurement device was thus connected to the SG outlet plena and to the PV upper-head one by one.

3.5.4 Experiment Results

In **Test 5-1**, the AFW flow rate was almost a half of the planned flow rate in the loop with PZR, causing low liquid level in the SG secondary-side. Two-phase natural circulation continued in the loop-B until the primary pressure decreased to far below 1 MPa with non-uniform flow among the SG U-tubes (**Fig. 3.5-1**). The SG U-tube liquid level recovered greatly after the initiation of the enhanced depressurization especially in the loop with PZR where subcooled coolant column was formed in the U-tubes. Similar large level recovery happened in the two long U-tubes of the SG in loop-B. For these U-tubes, the circulation terminated similar to those of the SG in the loop with PZR (loop-A). The U-tube liquid level monotonically decreased with the primary pressure after the initiation of the enhanced depressurization but in different way in both SGs. In the loop-A, the natural circulation stopped before the initiation of accumulator coolant injection because of adverse heat transfer around the top portion of the SG U-tubes which emerged above the two-phase mixture level.

Test 5-2 was conducted with non-condensable gas inflow as a counterpart to **Test 5-1**. The SG secondary-side liquid level changed similarly in two loops with the AFW flow rate as planned. SG secondary-side depressurization was performed as planned but the depressurization rate decreased after non-condensable gas started to enter primary loops and SG U-tubes, suggesting degradation in the condensation heat transfer in the SG U-tubes. Collapsed liquid level decreased gradually in most of the instrumented U-tubes after the initiation of non-condensable gas inflow as two-phase natural circulation continued with non-condensable gas. However, such tubes with rapid coolant drain probably because of accumulation of non-condensable gas and with continuous single-phase liquid natural circulation co-existed in the SG in the loop-B (**Fig. 3.5-2**). Asymmetrical natural circulation between two loops also appeared in Test 5-2 due probably to different number of forward flow SG U-tubes under the influences of non-condensable gas. The natural circulation anyhow continued and contributed to maintain core cooling effectively for a long time. Air concentration in steam was about 40-50% at the SG outlet plena after the U-tube downflow-side became empty of liquid. A bit larger amount of gas was found to exist in the SG outlet plenum than the estimation based on the steam pressure and temperature. Oxygen concentration in steam at the PV upper-head, however, indicates almost zero through the whole measurement, implying that no non-condensable gas enters the upper-head.

In summary, various non-uniform flow behaviors were observed in the SG U-tubes in both **Test 5** experiments. In Test 5-2 with non-condensable gas ingress especially, three different types of U-tube behavior coexisted in a SG depending probably on gas accumulation rate as well as asymmetric natural circulation between two loops. The SG secondary depressurization as an AM action, however, was confirmed to be effective and no core heat-up took place through the whole transient in both experiments. Direct measurement of non-condensable gas in a form of oxygen concentration in steam-air mixture by the JAEA-developed gas measurement device was successful in Test 5-2 at the SG outlet plena where non-condensable gas accumulates.

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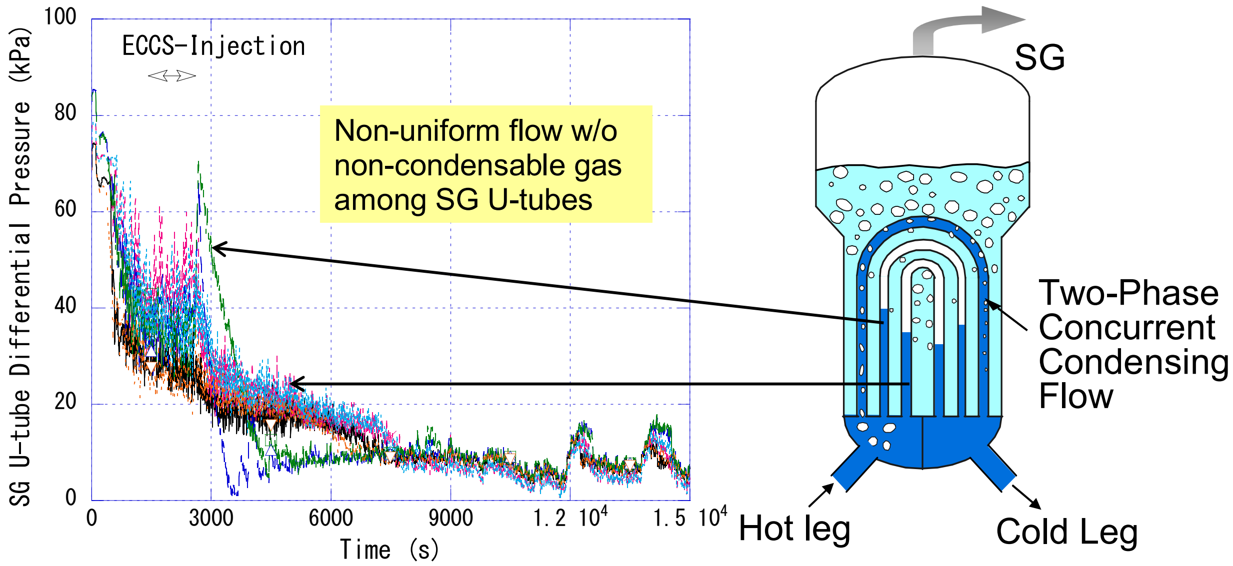

Fig. 3.5-1 SG U-tube liquid level behaviors without non-condensable gas inflow in Test 5-1

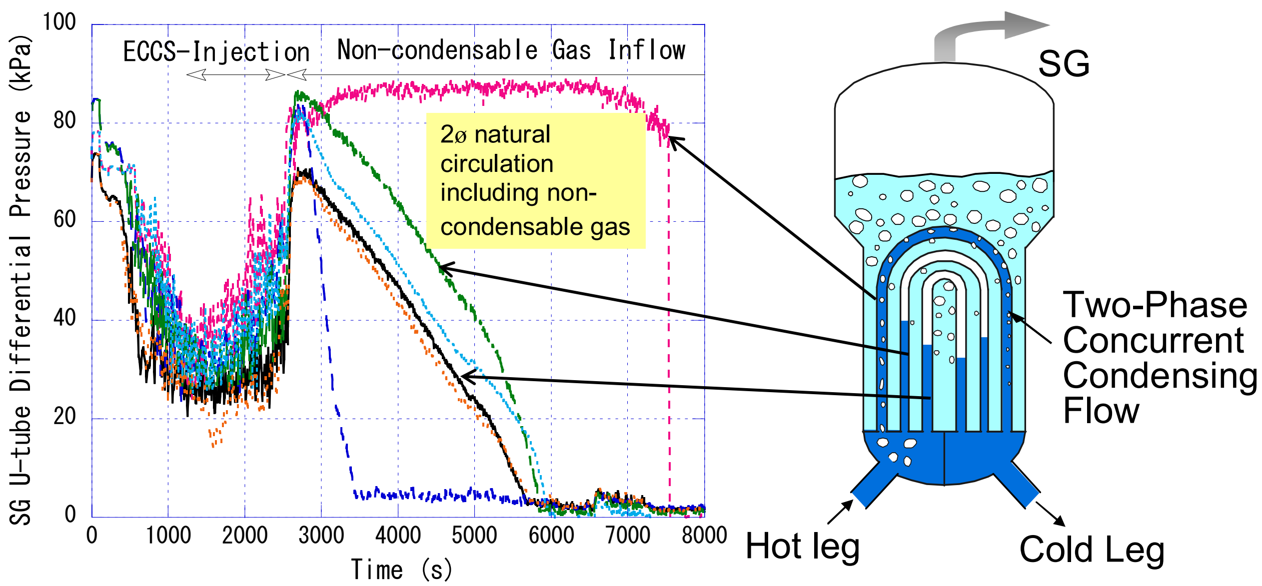

Fig. 3.5-2 SG U-tube liquid level behaviors with non-condensable gas inflow in Test 5-2

3.6 Test 6 Open subjects

The subjects for this test series are open to be defined through discussions among the project participants. The candidates based on the discussions were:

1. Pressure vessel (PV) upper-head small-break loss-of-coolant accident (LOCA)
2. PV bottom small-break LOCA
3. Large-break LOCA focusing on the direct-contact condensation in the cold leg and its effects on the containment pressure
4. Small-break LOCA focusing on accident management (AM) measures with symptom-oriented operator actions

The fourth was then decided being coupled to the first and second subjects. In summary, three LSTF experiments were defined for Test 6.

Background, objectives, description and results are explained for each of the three tests in the following sub-sections.

3.6.1 Test 6-1 PV upper-head break LOCA with AM measure

3.6.1.1 Background

Vessel head wall thinning was found at the Davis Besse reactor in the US in 2002 and raised a safety issue concerning vessel structural integrity [3.6-1]. Circumferential cracking of penetration nozzle for the control rod drive mechanism (CRDM) may cause a small-break LOCA (SBLOCA) with break at the PV upper-head, as shown in **Fig. 3.6-1**. Experimental data, however, had been scarcely obtained for such a PV upper-head SBLOCA especially with some recovery action such as steam generator (SG) secondary-side depressurization as a symptom-oriented AM action [3.6-2, 3.6-3]. During the core boil-off process, a core temperature excursion may be detected by core exit thermocouples (CETs), which can initiate the symptom-oriented AM operator action during the PV upper-head SBLOCA transient of interest, employing a certain criterion defined by industries and regulatory bodies. Detailed study on the thermal-hydraulic response associated with the PV upper-head break LOCA through the LSTF experiment was then requested from the Project participants in combination with the symptom-oriented AM action.

3.6.1.2 Objectives

The objectives of Test 6-1 are to clarify consequences of SBLOCA due to PV upper-head break with SG depressurization as a symptom-oriented AM measure based on core exit temperature measured by CETs during core boil-off, to confirm effectiveness of the symptom-oriented AM measure and to provide LSTF experimental data suitable for the assessment of predictability of computer codes and models for reactor safety analyses.

3.6.1.3 Description of Experiment

Test 6-1 [3.6-4] was conducted under assumptions of total failure of high pressure injection (HPI) system and an actuation of auxiliary feedwater as well as loss of off-site power concurrent with the scram signal. Break size, AM action and its onset timing were defined through discussions among the project participants based on the LSTF pre-test analyses and PWR SBLOCA analyses with RELAP5 code [3.6-5]. The size of break at PV upper-head was equivalent to 1.9% cold leg break to simulate the ejection of one whole CRDM penetration nozzle. The withdrawal of control rods was not simulated. SG secondary-side depressurization was started as a symptom-oriented AM operator action when the core exit temperature indicated a certain superheating. Namely, relief valves (RVs) of both SGs were fully opened immediately after the measured core exit temperature reached 623 K: a criterion for Japanese PWR.

3.6.1.4 Experiment Results

Liquid level in the upper-head was found to control break flow rate as coolant in the upper plenum entered the upper-head through control rod guide tubes (CRGTs) until the penetration holes at the CRGT bottom were exposed to steam in the upper plenum (**Fig. 3.6-2**). The cycle opening of the SG RVs induced oscillation in the upper-head mixture level. Relatively large size break resulted in a fast primary depressurization, especially after the break flow turned into single-phase vapor flow. The primary pressure became lower than the SG secondary-side pressure almost simultaneously with the core uncovery, resulting in no reflux flow from the SG.

It took, however, more than 230 s to initiate the AM action after the core temperature excursion started because of late and slow response of core exit temperature. The time delay in the detection of steam superheating by CETs was influenced by such factors as core power distribution, three-dimensional steam flow in the core due to steam flow towards break through CRGTs, low temperature steam flow from the core peripheral region and low-temperature metal structure around core exit [3.6-6]. Significant temperature distribution resulted in the core and at the core exit as typically shown in **Fig. 3.6-3**.

The AM action was not effective in the early stage onto the intended primary depressurization to start accumulator coolant injection, because the primary pressure was lower than the SG secondary-side pressure at the onset of AM action, as shown in **Fig. 3.6-4**. Automatic core power decrease procedure to protect LSTF core was then initiated due to high core cladding temperature during core boil-off. Loop seal clearing was induced when the accumulator injection started because of significant steam condensation in both cold legs. The experiment was terminated when long-term core cooling was confirmed by the actuation of low pressure injection (LPI) system.

In summary, the results of Test 6-1 indicated a possibility to have significantly high PCT in the PV upper-head break LOCA, though some assumptions adverse to core cooling such as total failure of HPI were made. Slow and late response of CETs during core boil-off was revealed, suggesting a necessity of further study on the CET responses because CETs are utilized worldwide as an important indicator to start AM operator action by detecting core temperature excursion during reactor accidents. A possibility was suggested further that the initiated AM measure becomes ineffective for a certain time duration once the initiation timing is so delayed, though the AM measure is expected to take effect shortly after the initiation. A task group on the CET effectiveness in AM was thus formed in the Working Group of Analysis and Management of Accident (WGAMA) of OECD/NEA/CSNI in 2008 to further discuss the CET effectiveness.

3.6.2 Test 6-2 PV bottom break LOCA with AM measure

3.6.2.1 Background

A small amount of residue including boron around the circumference of two instrument-tube penetration nozzles of PV lower-head was found at the South Texas Project Unit-1 in the US in 2003 and raised a safety issue concerning vessel structural integrity [3.6-7]. Ejection of two 1.5-inch outer-diameter instrument-tubes would have caused a SBLOCA at the PV bottom, as shown in **Fig. 3.6-1**. Liquid-phase break is characterized by fast loss of coolant mass with low primary depressurization. The SG secondary-side depressurization as an AM operator action is important to depressurize the primary system to introduce low-pressure ECCS especially when high-pressure ECCS is in failure. It is then important to confirm the effectiveness of the AM measure under the influences of non-condensable gas from accumulator tanks. Experimental data, however, have been scarcely obtained for such a PV bottom SBLOCA transient with the AM measure [3.6-8].

3.6.2.2 Objectives

The objectives of Test 6-2 are to clarify consequences of SBLOCA due to PV bottom break with an asymmetrical SG depressurization as an AM measure, to confirm the effectiveness of the AM measure, especially on the natural circulation core cooling and the primary depressurization under influences of non-condensable gas, and to provide a database suitable for the assessment of predictability of computer codes and models for system integral analyses.

3.6.2.3 Description of Experiment

Test 6-2 [3.6-9] simulated a SBLOCA with a break at PV bottom; equivalent to 0.1% cold leg break, to simulate the ejection of one whole instrumentation nozzle. The test was conducted under assumptions of total failure of HPI system, an actuation of auxiliary feedwater and loss of off-site power concurrent with the scram signal. Break size, AM action and its onset timing were defined through discussions among the project participants based on the LSTF pre-test analyses and PWR SBLOCA analyses by using RELAP5/MOD3.2.1.2 code. Discussion was made especially on the SG depressurization method. Consequently, asymmetrical depressurization only for the SG in the loop without pressurizer (PZR) was employed, considering it may cause unbalanced natural circulation between loops and unbalanced transport of non-condensable gas (nitrogen gas) from accumulator tanks (**Fig. 3.6-5**), which would be unique and suitable for the validation of computer codes and models. The AM action was taken 30 minutes after the safety injection signal, considering an internationally common understanding on a grace period to start operator action. The SG depressurization rate was controlled such that the primary depressurization rate becomes to be 55 K/hr.

3.6.2.4 Experiment Results

Very small size break resulted in slow primary depressurization with certain influences of SG secondary conditions. Large pressure difference appeared twice between primary and secondary-side of the SG in the loop without PZR after the AM action and after accumulator tanks started to discharge non-condensable gas. The large pressure difference in the early timing was caused by adverse heat transfer from non-depressurized SG to the primary loop where natural circulation stooped soon, while the natural circulation continued in the loop without PZR after the AM action. When non-condensable gas started to enter primary loops, primary depressurization rate decreased and the large pressure difference resulted, suggesting the degradation of condensation heat transfer in the SG U-tubes (**Fig. 3.6-5**).

The core exit temperatures on the side of the loop with PZR increased after the initiation of core uncovery in the late phase of transient. However, they were kept at the saturation temperature on the side of the loop without PZR due to condensate fall-back from the SG though non-condensable gas weakened steam condensation in the SG U-tubes. The LPI system was actuated when steam generation rate decreased in the core, and most of the core was quenched. The fuel rod surface temperature, however, started to increase again because of the termination of the LPI system when the primary pressure was increased due to the recovery of steam generation in the core following the core quench. The experiment was terminated when core cooling was confirmed by the second actuation of the LPI system after the initiation of automatic core power decrease procedure due to high core temperature in the second core boil-off.

In summary, Test 6-2 provided a unique database including asymmetrical natural circulation behavior because of significant effect of the asymmetric SG secondary-side depressurization under influences of non-condensable gas. The asymmetric SG depressurization emphasized heat transfer manner between the primary and SG secondary sides and the natural circulation under the influences of non-condensable gas in all the three modes: the single-phase liquid, two-phase and reflux condensation modes. The natural circulation, though asymmetrical, effectively cooled core as long as two-phase mixture covered the core. The degradation of condensation heat transfer in the SG U-tubes, however, caused a significant delay in the actuation of LPI as well as obstruction for the LPI actuation resuming.

3.6.3 Test 6-3 Direct-contact condensation in cold leg during large-break LOCA

3.6.3.1 Background

Steam direct-contact condensation may take place on emergency core cooling system (ECCS) water in cold legs during reflood phase of a large-break loss-of-coolant accident (LBLOCA) of PWR, as shown in **Fig. 3.6-6**. The steam condensation in the cold leg is ignored in the current safety analyses in Japan for the conservative evaluation of containment pressure response by the overestimation of steam discharge rate from the break. The steam condensation data are important for the realistic evaluation of the containment pressure, especially for the pressure peak that appears during reflood phase of LBLOCA.

Since the LBLOCA is design-basis accident of PWR, many of relevant databases have been obtained by such test facilities as LOFT [3.6-10], CCTF, SCTF and UPTF [3.6-11]. Experimental data, however, have been scarcely obtained for the steam condensation on ECCS water in cold legs because most of the LBLOCA experiments are under some influences of non-condensable gas. The boundary conditions were not well defined in several experiments either. The project participants, therefore, requested to prepare a new database through the LSTF experiments to validate the associated direct steam condensation models and the safety analysis codes.

3.6.3.2 Objectives

The objectives of Test 6-3 are to clarify the steam direct-contact condensation on ECCS coolant during PWR LBLOCA that may impact on containment pressure through the LSTF separate-effect experiment that properly simulates the flow conditions in the cold leg, and to obtain a database useful for the validation and development of safety analysis codes.

3.6.3.3 Description of Experiment

Test 6-3 [3.6-12] was conducted as a separate-effect experiment in a test section newly installed downstream of the break unit that is horizontally attached to the LSTF cold leg, as will be described in **Appendix 1**. The test section mainly consists of a simulated cold leg pipe (102.3 mm i.d.), a simulated ECCS injection nozzle, three viewers, a separator, separator downstream pipes for steam and water. The three viewers were installed in the downstream of the ECCS injection point to observe visually condensing two-phase flow behaviors by high-speed video camera with back-lighting through a light diffusion plate.

Experimental conditions on primary pressure, inlet steam velocity and superheating as well as mass flux of ECCS water in the simulated cold leg pipe were defined through discussions among the project participants based on PWR LBLOCA analyses by using RELAP5/MOD3.2.1.2 code. The primary pressure was near-atmospheric pressure, similar to the calculated condition in the PWR cold leg (698.5 mm i.d.) during reflood phase. Steam velocity at the inlet of the simulated cold leg pipe was controlled by the LSTF core power. The inlet steam velocity was about 20, 30, 40 and 50 m/s, which mostly covers conditions obtained by the PWR LBLOCA analysis. The inlet steam superheating was as high as possible in the LSTF to best simulate the PWR conditions. Simulated ECCS water was injected at temperature of 310 K into the simulated cold leg pipe through a nozzle at angle of 45 degree downward, similar to that in reference PWR. Simulated ECCS water mass flux was ranging from about 12 to 207 kg/(m^{2}s) so as to preserve the ratio of ECCS water injection flow rate to inlet steam flow rate, but less than that to cause complete condensation of steam as shown in **Fig. 3.6-6**.

3.6.3.4 Experiment Results

Significant steam condensation took place in a short distance from the simulated ECCS injection point, and the steam temperature in the test section decreased immediately after the initiation of the ECCS injection. Fluid temperature distribution was non-uniform at about 50 mm downstream from the ECCS injection point but became to be almost uniform in less than about 350 mm. Total steam condensation rate estimated from the difference between the test section inlet and outlet steam flow rates was in proportion to the simulated ECCS water mass flux until the complete condensation of steam as shown in **Fig. 3.6-6**. When the simulated ECCS water mass flux is about 148 kg/(m^{2}s); equivalent to full HPI with single-failure LPI conditions, for instance, it was found that Inlet steam was completely condensed if inlet steam mass flux is less than about 195 kg/(m^{2}s).

Clear image data were obtained on droplet behaviors through the viewer at about 0.2 m downstream from the ECCS injection point, especially for annular mist flow, as shown in **Fig. 3.6-7**. The number of droplets decreased with increasing distance from the ECCS injection point, and very few droplets appeared at the viewer at about 2.3 m downstream from the point.

In summary, a steam direct-contact condensation behavior was successfully observed in a new test section that simulated typical flow conditions that may appear in PWR cold legs during reflood phase of LBLOCA under well-defined boundary conditions. Steam condensation was so significant that it caused liquid and steam temperature to become equal in a short distance from the simulated ECCS injection point. The total steam condensation rate was in proportion to the simulated ECCS water mass flux until the complete condensation of steam. Clear image data were obtained on droplet behaviors, especially for annular mist flow, which is important for model validation and development.

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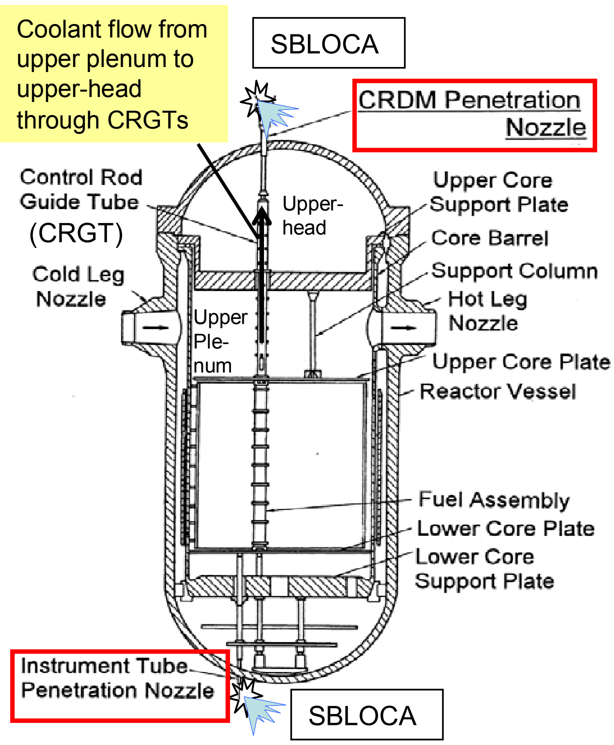


Fig. 3.6-1 Locations of penetrations at top and bottom of PWR pressure vessel

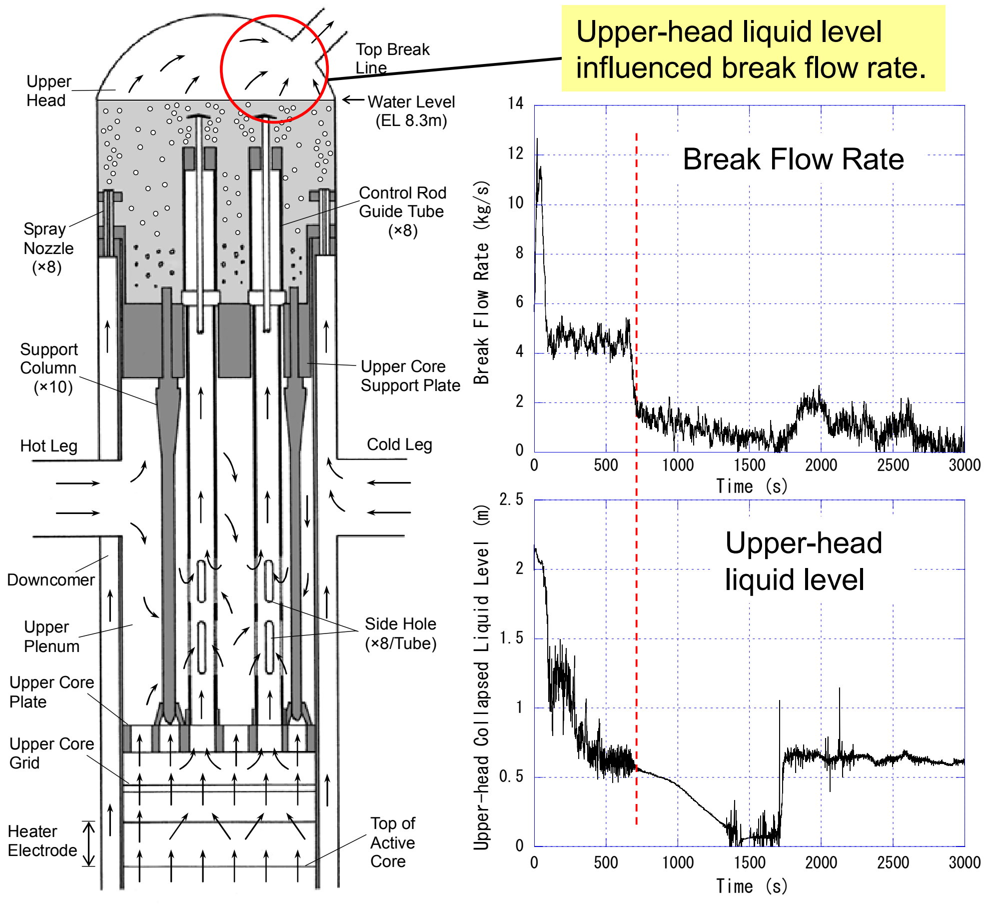


Fig. 3.6-2 Coolant flow in pressure vessel upper region at onset of AM action in Test 6-1

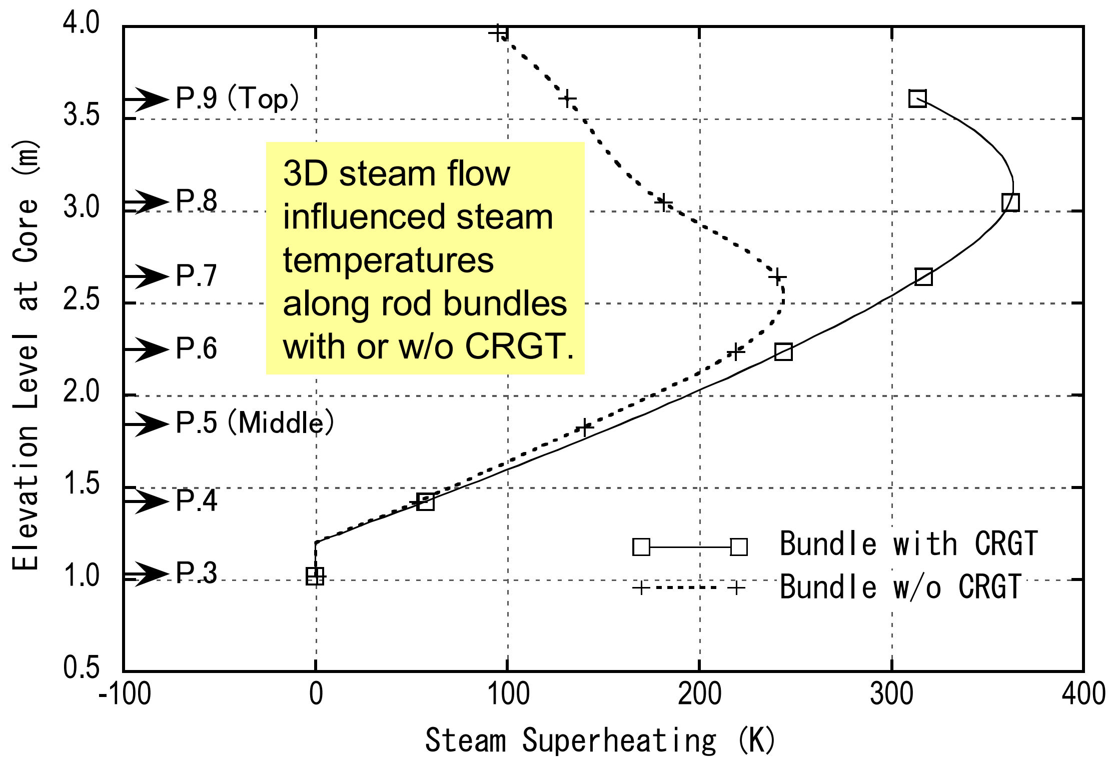


Fig. 3.6-3 Core steam temperatures 100 s after initiation of AM action in Test 6-1

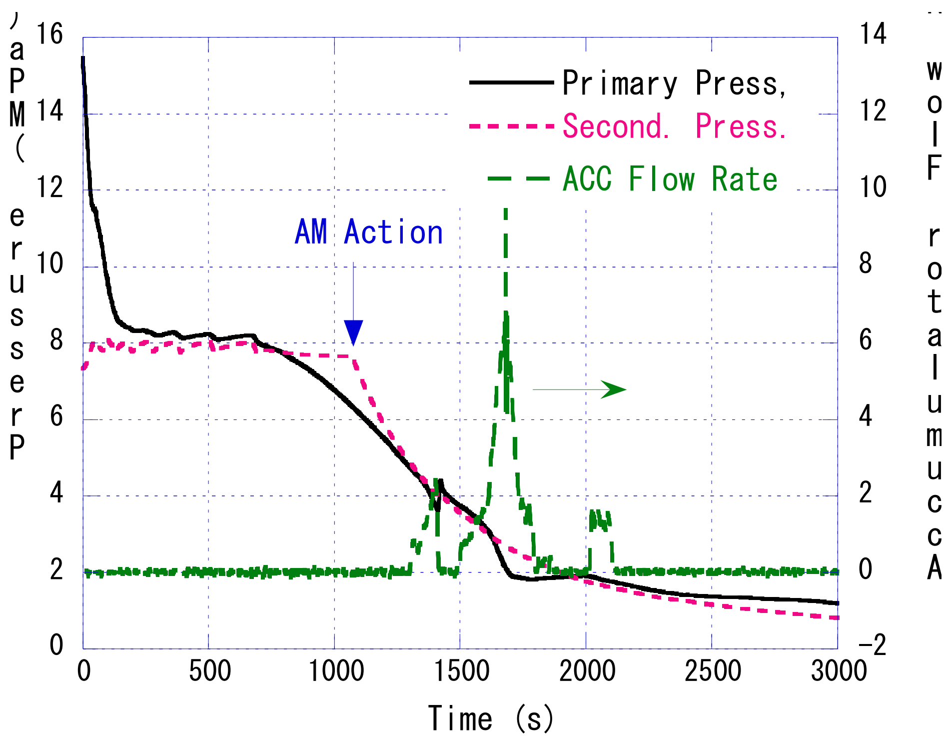


Fig. 3.6-4 Primary and secondary pressures and accumulator flow rate in Test 6-1

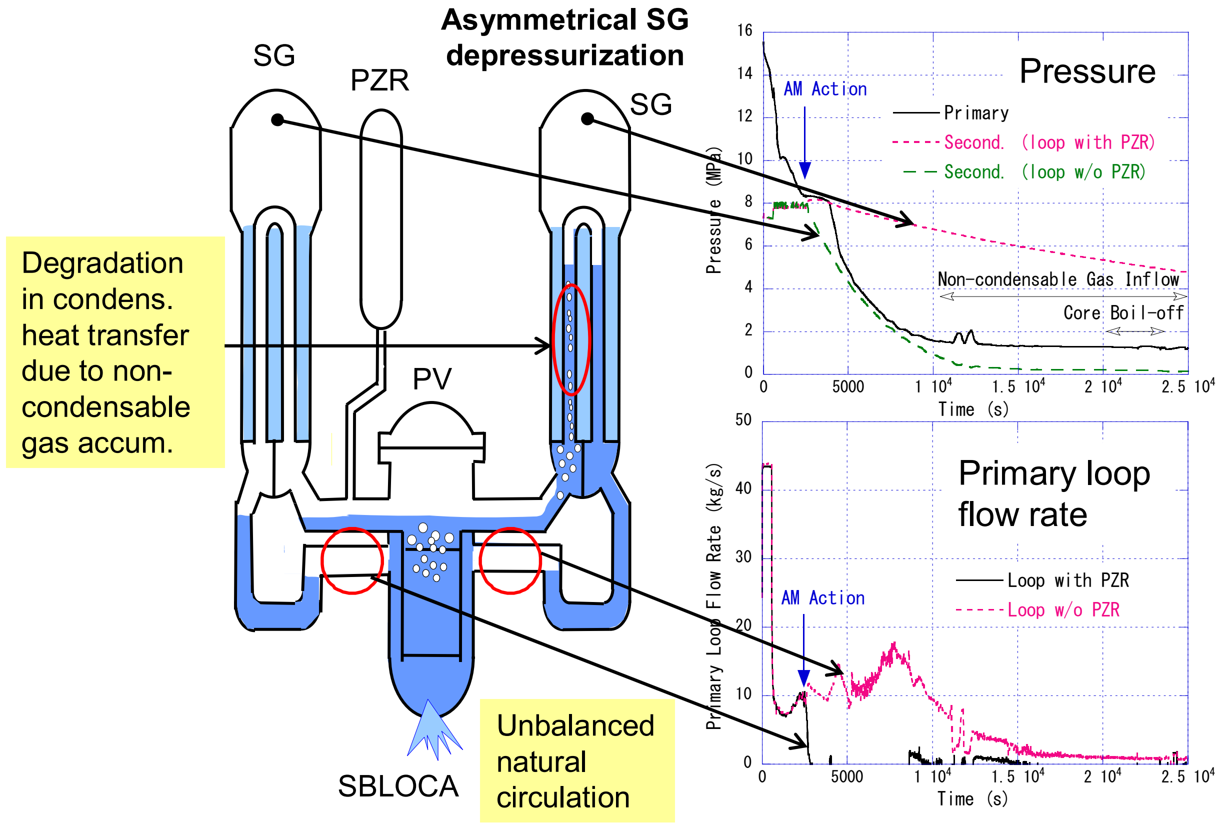


Fig. 3.6-5 Local phenomena during SBLOCA with asymmetrical depressurization in Test 6-2

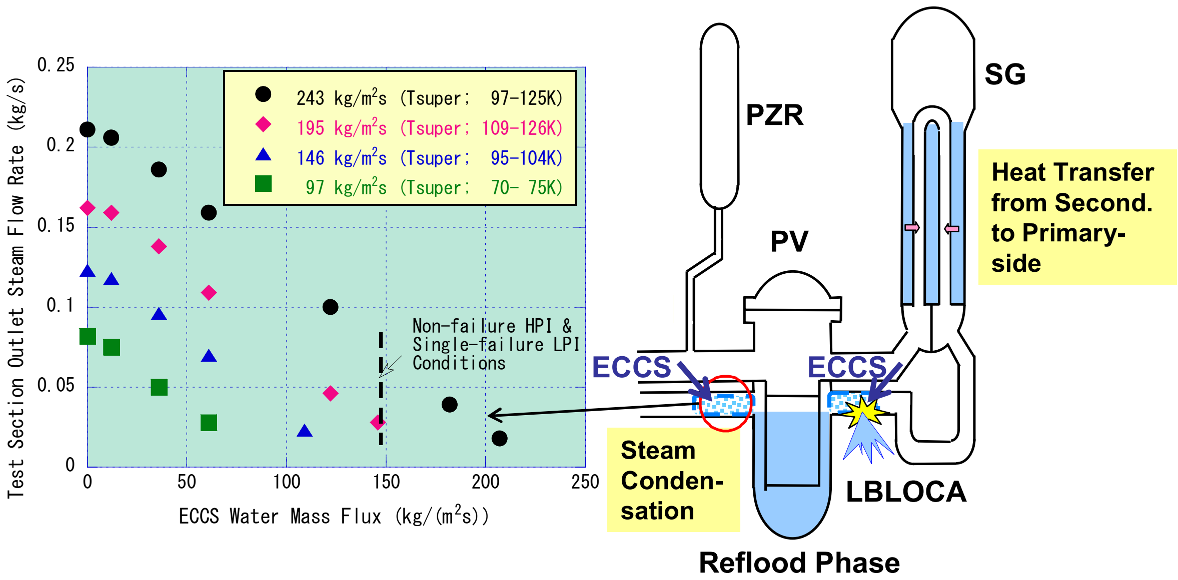
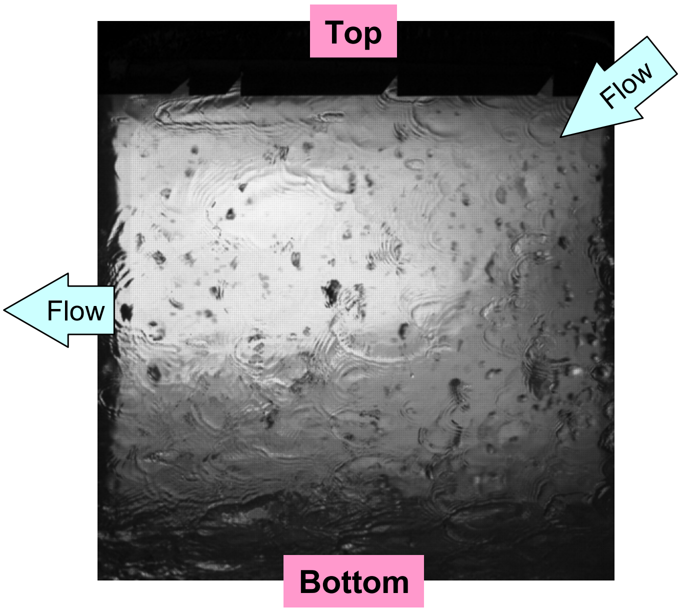
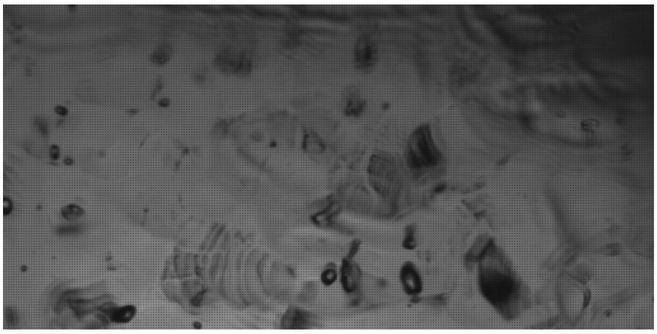


Fig. 3.6-6 ECCS water mass flux versus test section outlet steam flow rate (inlet steam mass fluxes of 97, 146, 195 and 243 kg/(m^{2}s)) in Test 6-3



(a) Overall view for flow observation



(b) Center view for droplet viewing

Fig. 3.6-7 Image taken through viewer at about 0.2 m downstream from ECCS injection point (inlet steam velocity of 20 m/s, ECCS water mass flux of 12 kg/(m^2s)) in Test 6-3

4. SUMMARY

OECD/NEA ROSA Project was carried out for 4 years from April 2005 among 14 countries to resolve issues in thermal-hydraulics analyses relevant to LWR safety using the ROSA/LSTF facility of JAEA. Major objectives of the ROSA Project are as follows:

(1) Provide integral and separate-effect experimental database to validate code predictive capability and accuracy of models for complex phenomena that may occur during DBE and beyond-DBE.

(2) Clarify the predictive capability of codes currently used for thermal-hydraulic safety analyses as well as of advanced codes presently under development.

The **LSTF** (Large Scale Test Facility) of ROSA (Rig of Safety Assessment) Program in JAEA is a full-pressure and full-height two-loop integral test facility constructed in 1985 to investigate thermal-hydraulic response during PWR accidents and abnormal transients. The LSTF simulates an 1100 MWe four-loop Westinghouse (W)-type PWR: Tsuruga Unit-2 reactor of the JAPC (Japan Atomic Power Company) with a volumetric scaling of 1/48.

Especially, **12 experiments** for **6 subjects** were planned and performed for the ROSA Project in either integral-effect test (IET) or separate-effect test (SET) to widely cover thermal-hydraulic responses that may encounter during reactor accidents and abnormal transients. Care was taken for each of the following ROSA/LSTF experiments to include phenomena related to multi-dimensional mixing, stratification, parallel flows, oscillatory flows and/or non-condensable gas flows,

Test 1 Temperature stratification and coolant mixing during ECCS coolant injection

Two experiments (IET and SET) relevant for pressurized thermal shock (PTS) under reactor typical pressure, temperature and flow conditions were performed.

The experiments revealed that thermal stratification condition in horizontal leg depends on ECCS injection manner and that liquid level in horizontal leg and flow condition may control low-temperature coolant flowing down along either pressure vessel inner wall side or core shroud outer wall side.

Test 2 Unstable and destructive phenomena such as water hammer

A series of SETs was performed on onset conditions and consequences of water hammer in a horizontal pipe under wide range of system pressure.

The experiments clarified that water hammer may take place up to 7 MPa and it generates greatest pressure pulse peak at around 1 MPa.

Test 3 Natural circulation under high core power conditions

Two IETs were performed under assumption of failure of reactor scram to reveal various complex thermal-hydraulic phenomena adverse to core cooling under high core power conditions and difficult for code analyses, which are as follows;

Test 3-1 (1% cold leg break LOCA): Supercritical flow in hot legs under high velocity steam flow, coolant accumulation in SG inlet plenum and U-tubes by CCFL.

Test 3-2 (LOFW transient): Complex reflux cooling phenomena in parallel SG U-tubes in non-uniform manner.

Test 4 Natural circulation with superheated steam

Two SETs were performed, respectively with argon or steam as test fluid, to clarify consequences that may cause containment bypass because of the rupture of SG U-tube(s) during a severe

accident. The experimental conditions were carefully defined and confirmed by using CFD code analyses and LSTF preparatory experiments.
The experiments provided detailed low-velocity steam flow distribution within hot legs, steam mixing in SG inlet plenum and parallel non-uniform flow in SG U-tubes.

Test 5 Primary cooling through SG secondary depressurization

Two IETs simulating 0.5% cold leg break LOCA with or without non-condensable gas ingress were performed to clarify non-uniform parallel SG U-tube behavior during the SG depressurization as an accident management (AM) action to mitigate reactor accidents by effective primary cooling.
The experiments systematically observed various non-uniform flows in SG U-tubes under influences of non-condensable gas onto heat transfer during the natural circulation especially at relatively low system pressures that is important for long-term core cooling.

Test 6 Open subjects

Three experiments (2 IETs and 1 SET) were performed to provide necessary information in response to the requests from Project participants, which are as follows;

Test 6-1 (IET): PV upper-head break small-break LOCA (SBLOCA) with SG secondary depressurization as a symptom oriented AM measure based on core exit temperature. This test raised a safety issue on the reliability of core exit temperature to detect core uncovery. A new task group was formed under WGAMA (working group on analysis and management of accident) of OECD/NEA/CSNI in 2008.

Test 6-2 (IET): PV bottom break SBLOCA with asymmetrical SG secondary depressurization as an AM measure. Asymmetric natural circulation core cooling was observed under influences of non-condensable gas, which indicated a possibility of insufficient core cooling in a long-term transient because of relatively high primary pressure.

Test 6-3 (SET): Direct steam condensation in ECCS coolant in cold legs during reflood phase of large-break LOCA (LBLOCA). Significant steam condensation was observed in a short distance from ECCS injection point between the subcooled ECCS coolant and highly superheated steam. Overall condensation rate was linearly proportional to the injected ECCS coolant. Image data on liquid droplets under condensation were obtained.

For all the experiments, **pre-test analyses** were done to survey optimum test conditions to meet the test objectives. The obtained results were distributed and discussed among the Project participants. The pre-test analyses employed such computer codes as RELAP5/MOD3.2.1.2, SKETCH-INS/TRAC-PF1 and FLUENT version 6.2 and 6.3.

To obtain necessary data during each of ROSA/LSTF experiments, JAEA-designed **new instrumentation** were furnished especially to measure temperature distribution at fine spatial resolution in cold and hot legs, downcomer, SG inlet plenum, non-condensable gas concentration in steam-gas mixture at PV upper-head and SG outlet plenum, low-velocity gas flow distribution in hot legs and direct visual observation of flows in all of the four horizontal legs.

New test sections were furnished further especially for each of Test 2 and Test 6-3 to observe local phenomena in detail with much of the instrumentation specially designed for the purpose.

The obtained data has been readily distributed to and shared among the Project participants and utilized for the post-test analyses for the validation, verification and development of the safety analysis codes and models as well as advanced codes such as CFD codes. **The obtained technical information and experiences**, partly through **a questionnaire** for Test 6-1 post-test analysis, were shared among the Project participants through the discussions during in PRG (Program Review Group) meetings and over e-mail communications.

In total, the ROSA Project, with the Project participants, successfully planned and performed ROSA/LSTF experiments and shared the obtained data to conduct the verification, validation and development of the computer codes including future CFD codes.

This Final Integral Report is published in a form of a DVD with the Data Report for all the 12 ROSA/LSTF experiments with the measured data except for some specific data such as fast response pressure sensor data from Test 2.

APPENDIX 1 FACILITY MODIFICATION AND NEW INSTRUMENTATION

Table A1-1 shows a list of new instrumentation and new test section prepared and utilized in the ROSA Project.

New instrumentation installed for Test 1 (temperature stratification) consists of thermocouples in the cold legs and downcomer to measure the detailed temperature distributions and one video probe in each of cold legs to visually observe flow behaviors below the ECCS nozzle, as shown in **Fig. A1-1**. The thermocouples at distance of 746.5 or 1633.5 mm from the PV center are arranged at 20 mm intervals spot-welded on each of three stainless steel support plates fixed to the inner wall of the cold leg, as shown in **Fig. A1-2**. Each video probe mainly consists of a sensitive CCD camera, a bore scope and light sources, as shown in **Fig. A1-3**. The windows for both bore scope and light are made of sapphire with high temperature and pressure resistant.

A new test section for Test 2 (water hammer) was connected to the LSTF downcomer to clarify the onset criteria and the intensity of condensation-induced water hammer (CIWH) under well-defined boundary conditions, as shown in **Fig. A1-4**. The test section is an about 2 m-long and 11 mm-thick stainless steel horizontal pipe with inner-diameter of 66.9 mm. The test section end is closed by a sealing plate. Water is injected into the test section through the water supply line at the test section bottom. The test section was equipped with instruments for fluid and wall temperatures, pressure, pressure fluctuation, fluid density and so on. Fast response pressure fluctuation sensors were used for the measurement of CIWH pressure pulses.

New instrumentations have been installed for Test 4 (super-heated natural circulation). The instrumentations consist of thermocouples in the hot legs and SG inlet plena to clarify the thermal stratification and mixing of hot and cold gases and JAEA-developed pulsed-wire velocity meters in the hot legs to measure the velocity distribution in the vertical direction, as shown in **Fig. A1-5**. The vertically-traversable pulsed-wire velocity meter has resistance thermometer wires made of platinum-rhodium arranged in series in the flow direction, as shown in **Fig. A1-6**. A wire at the most upstream is electrically heated while others act as sensors to detect the arrival of the hot gas volume generated by the upstream heater wire. The gas velocity is evaluated from time difference of detection and distance between the two wires.

A JAEA-developed gas measurement device was used for Test 5-2 (SG depressurization with non-condensable gas inflow) to directly measure oxygen concentration in steam at the SG outlet plena and PV upper-head by introducing air as non-condensable gas, as shown in **Fig. A1-7**. The gas measurement device includes a Zirconia sensor, a separator, a flow meter and piping. The measurement uncertainty of Zirconia sensor is ±1.5% of full-scale (25%) for oxygen concentration in steam. Conditions of fluid sampled at the SG outlet plena is confirmed by draining it from the separator bottom while the glove valve and valve in the drain line are opened. Small-volumetric flow rate of steam/gas mixture is measured by using the flow meter in the downstream of Zirconia sensor vessel.

A new test section for Test 6-3 (steam condensation) was furnished in the downstream of the break unit horizontally attached to the LSTF cold leg to simulate a steam direct-contact condensation on ECCS water in cold legs during reflood phase of PWR LBLOCA under well-defined boundary conditions, as shown in **Fig. A1-8**. The test section mainly consists of a simulated cold leg pipe, a simulated ECCS injection nozzle, three viewers, a separator, separator downstream pipes for steam and water. The simulated cold leg pipe is an about 4.5 m-long and 6 mm-thick stainless steel horizontal pipe with inner-diameter of 102.3 mm. The three viewers were installed in the downstream of the ECCS injection point to observe condensing two-phase flow behaviors by using high-speed video camera. The separator separates steam-water two-phase flow into single-phase steam and water flows at the downstream end of the test section. The test section was equipped with instruments for flow rate, fluid and wall temperatures, pressure, fluid density and so on. Two vortex-type flow meters were mounted at the inlet of the simulated cold leg pipe and at the separator downstream steam pipe to measure volumetric flow rates of steam, from which velocities and mass flow rates of steam were estimated.

Table A1-1 List of new instrumentation and new test section

Test No.	Subjects	Contents
Test 1	Temperature stratification	Thermocouples in cold legs and downcomer
		Video probes in cold legs
Test 2	Water hammer	New test section with new instrumentation that horizontally attached to LSTF downcomer
Test 4	Super-heated natural circulation	Thermocouples in hot legs and SG inlet plena
		JAEA-developed pulsed-wire velocity meters in hot legs
Test 5-2	SG depressurization with non-condensable gas inflow	JAEA-developed gas measurement device connected to SG outlet plena and to PV upper-head
Test 6-3	Steam condensation	New test section with new instrumentation furnished downstream of break unit horizontally attached to LSTF cold leg

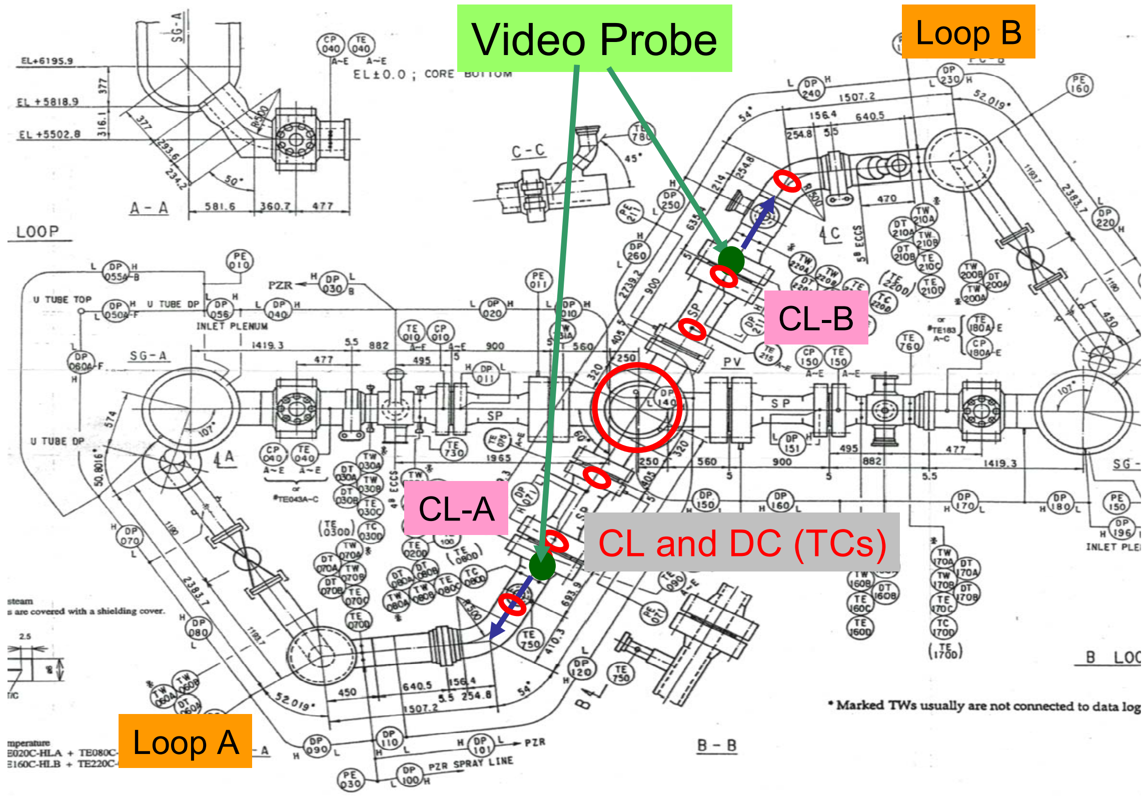


Fig. A1-1 Locations of new instrumentation for Test 1

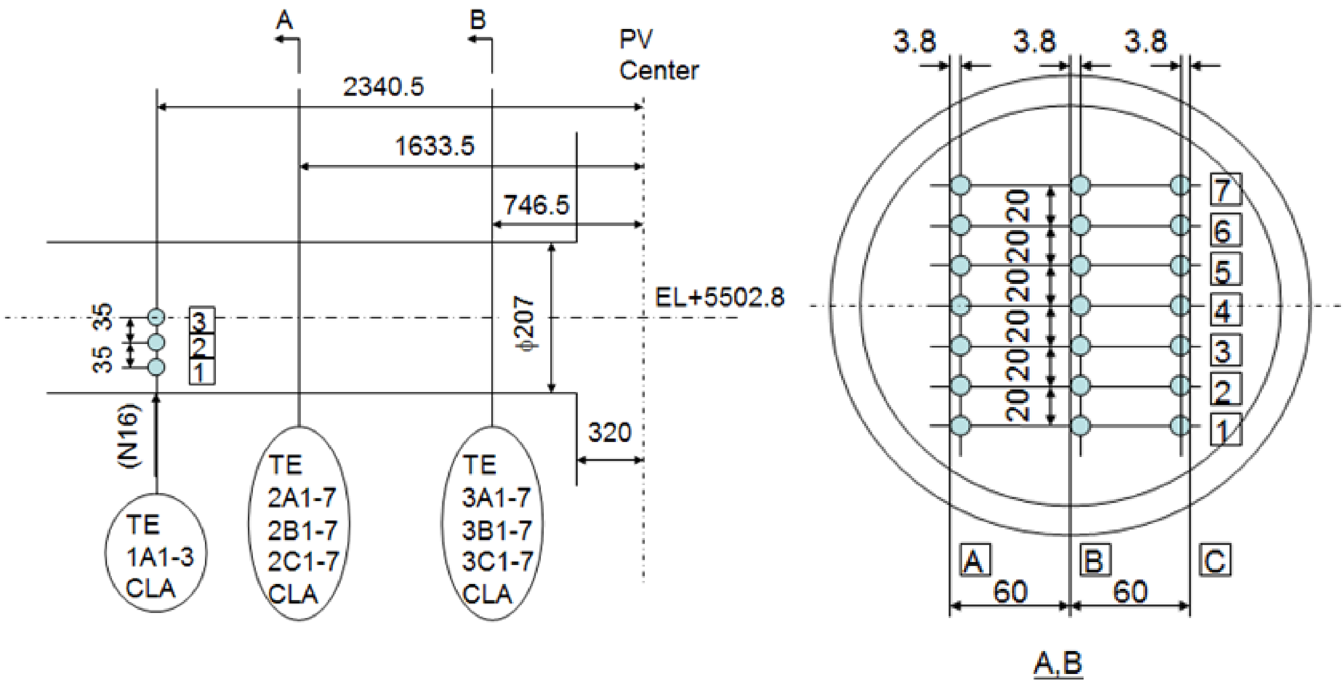


Fig. A1-2 Locations of new thermocouples in cold leg in loop A for Test 1

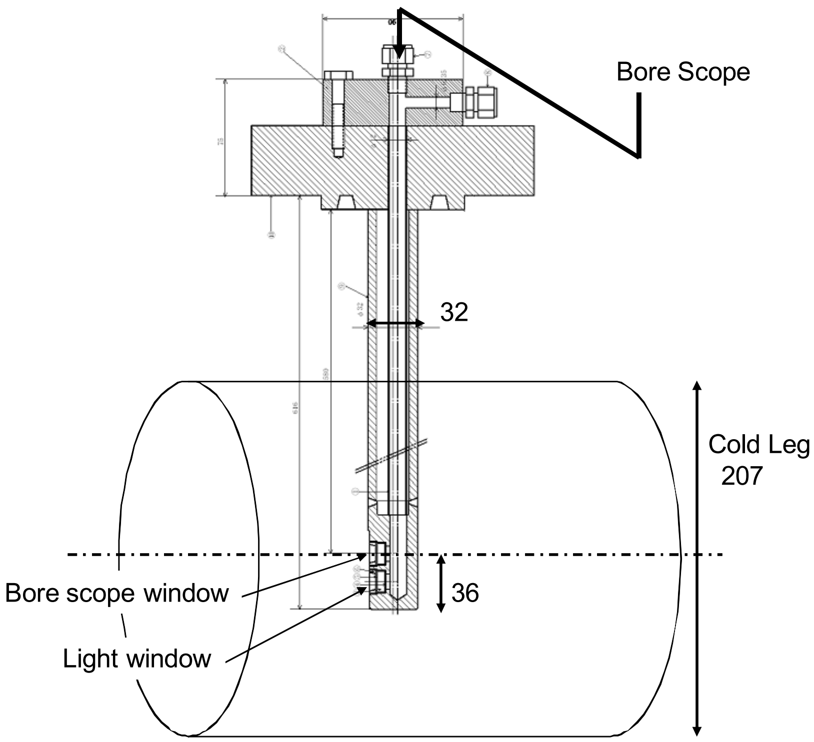


Fig. A1-3 Schematic view of new video probe in cold leg for Test 1

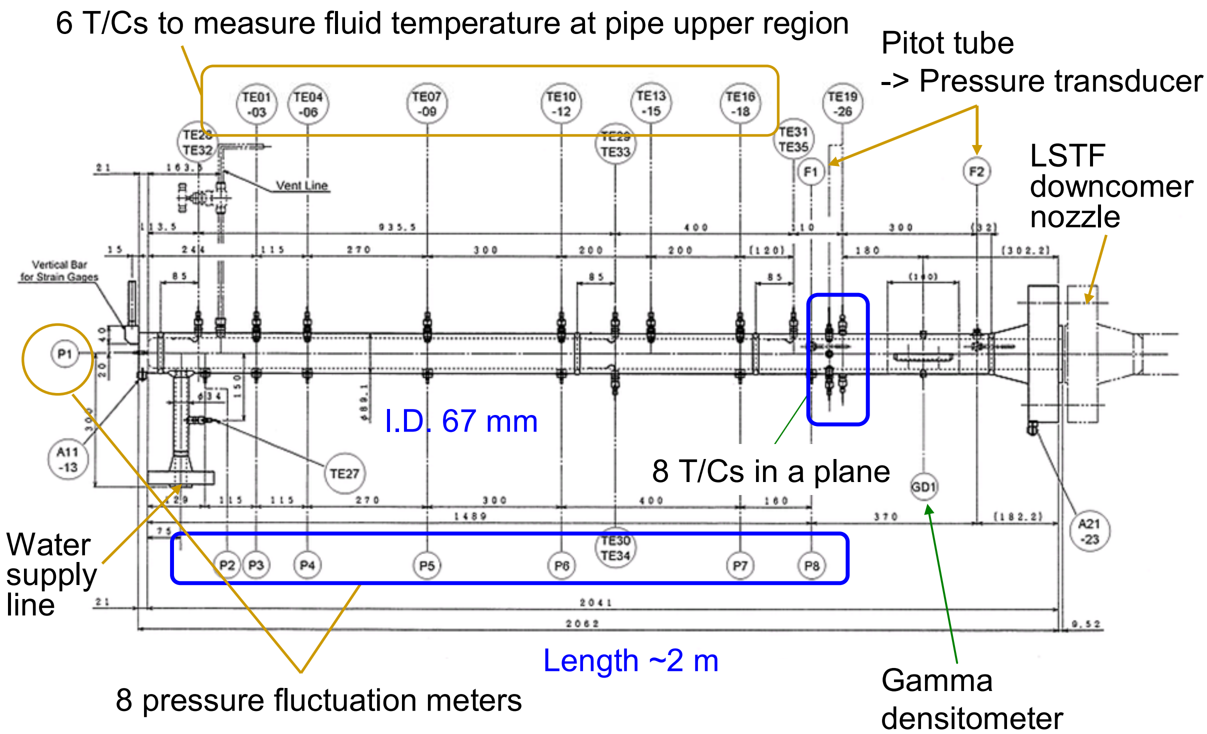


Fig. A1-4 Geometry of new test section for Test 2

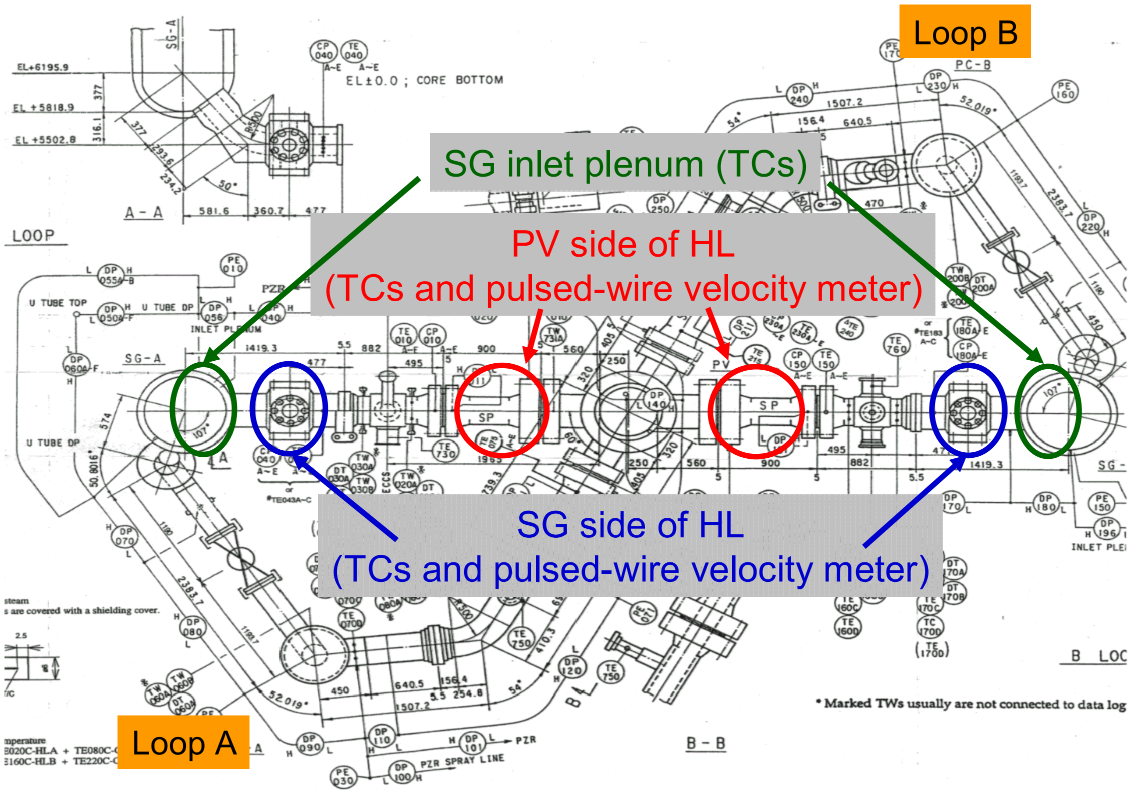


Fig. A1-5 Locations of new instrumentation for Test 4

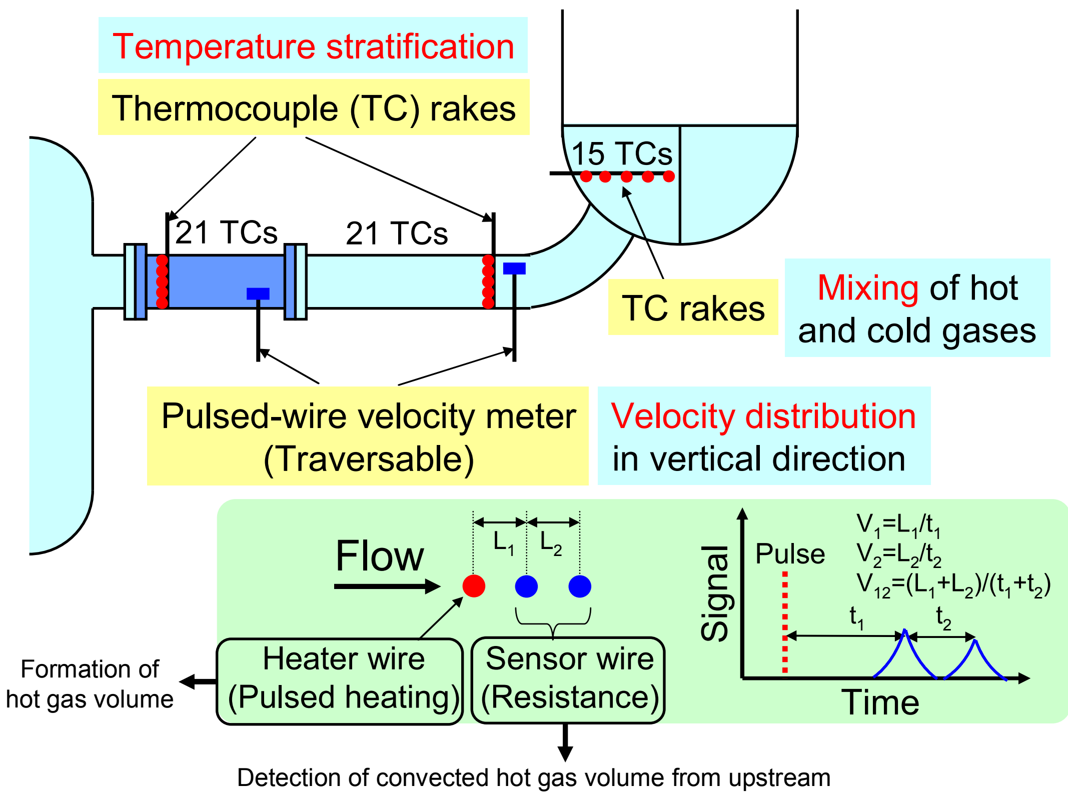


Fig. A-1-6 Conceptual scheme for newly installed instrumentation in Test 4

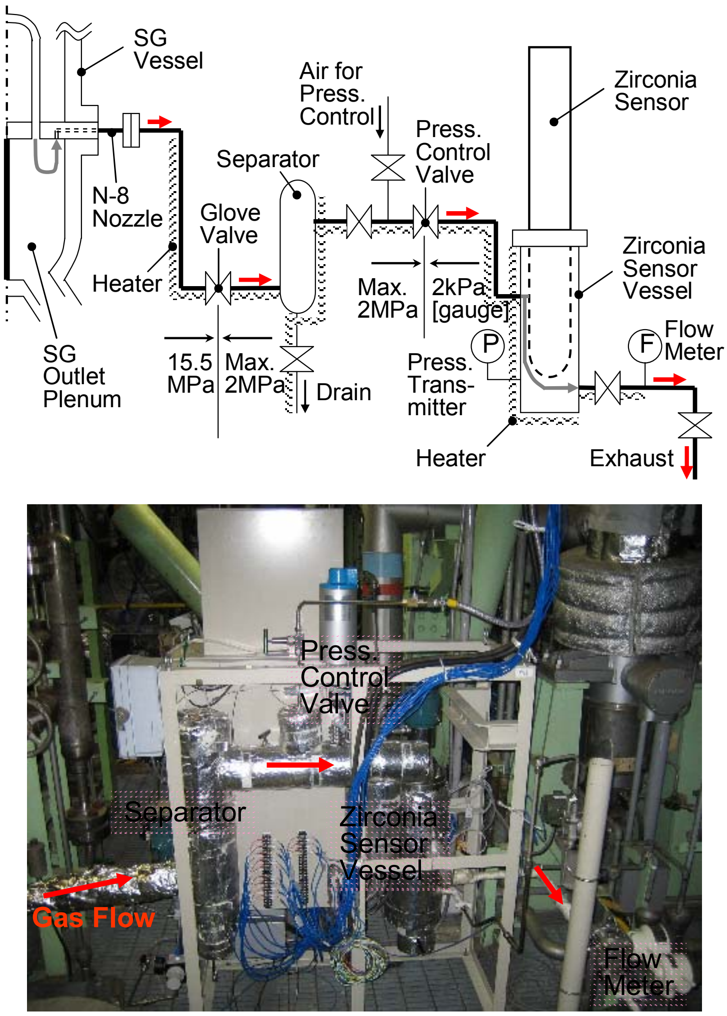


Fig. A1-7 Overview of gas measurement device connected to SG outlet plenum in Test 5-2

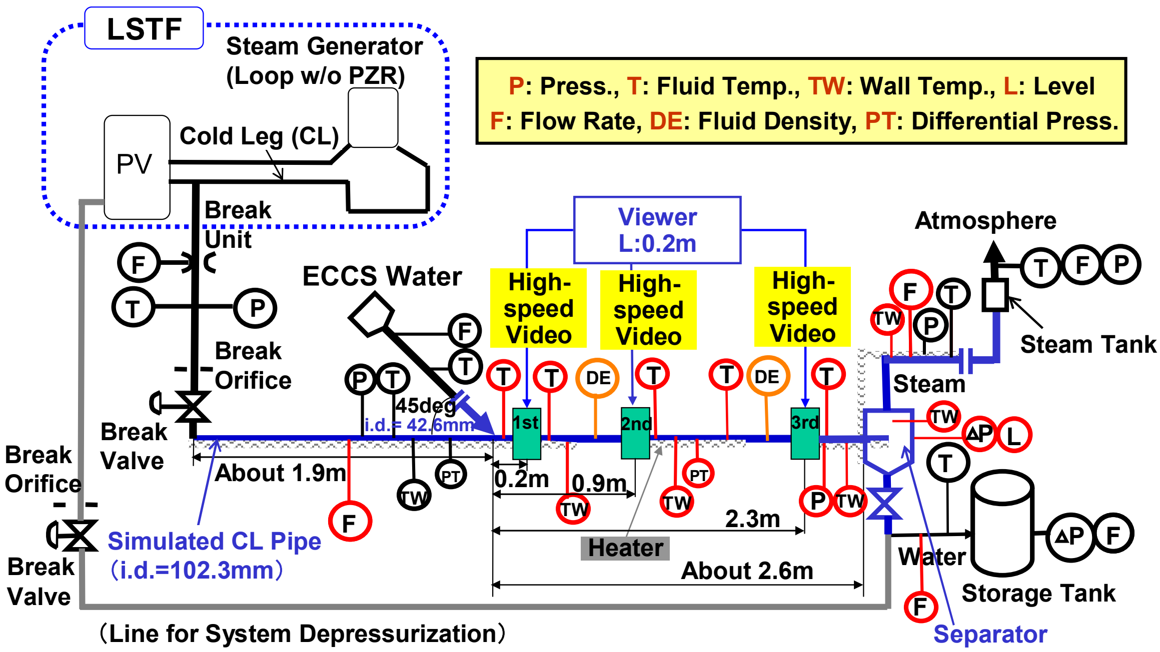


Fig. A1-8 Schematic view of new test section in downstream of cold leg break unit in Test 6-3

APPENDIX 2 POST-TEST ANALYSES

One of the key objectives of the ROSA Project is to clarify the predictability of computer codes currently used for thermal-hydraulic safety analyses as well as of advanced codes under development. Many pre-test and post-test analyses were performed for the ROSA Project by both the project participants and JAEA as operating agent to define the boundary conditions of the LSTF experiments and to understand the test results. **Appendix 2** summarizes the major findings obtained mainly through the post-test analyses.

A2-1 Pre-test Analysis and Post-test Analysis

(1) Pre-test analysis

The pre-test analyses are performed to provide information on the expected test results by changing values of several major parameters to define the test boundary conditions to achieve the objectives of all of the twelve LSTF experiments on the six subjects for the ROSA Project. The obtained results were discussed at the program review group (PRG) meetings and via e-mails among the project participants.

The pre-test analysis is a kind of blind calculation and provides a good chance to verify the code predictability. In the ROSA Project, only JAEA performed this analysis as the operating agent partly because it was considered that the analysis should be completed in relatively short time for the test preparation. JAEA then mainly used rather old version of the RELAP5 code (MOD 3.2.1.2) for the LOCA test cases because of long-term experience to use it including several code modifications. Other than the RELAP5 code, the SKETCH-INS/TRAC-PF1 code developed in JAEA for the three-dimensional thermal-hydraulics neutronics coupling in the core was used to define the core power transient in Test 3-1 (high-power natural circulation) with an assumption of failure of core scram during a LOCA. FLUENT code; computational fluid dynamics (CFD) code to analyze three-dimensional fluid flows, was used further to define conditions that cause natural circulation of superheated steam in Test 4.

In the course of the ROSA Project, however, many disagreements were observed between the pre-test calculation and LSTF test results, though the options and models were selected based on the previous experiences. One of such disagreements appeared for Test 6-1 (PV upper-head break LOCA) where the RELAP5 code expected that the single-phase steam discharge from the break, thus, fast primary depressurization, starts much later than observed in the LSTF test. This happened because of some errors in the break flow prediction, particularly in the definition of discharge coefficient in conjunction with the break upstream flow analysis as well as some trial and error effort in the guide tube flow representation. The faster core uncovery and temperature excursion appeared in the LSTF experiment than expected. Then, the late and slow response of core exit thermocouples (CETs) to detect steam superheating resulted late initiation of SG secondary depressurization as an accident management measure. This experience associated with the pre-test analyses has suggested us the necessity of thorough verification and validation (V&V) of computer codes and careful selection and definition of input parameters, options and models.

(2) Post-test analysis

The post-test analyses are performed to validate computer codes and models against the obtained (known) test data, which were performed by many of the project participants. The best estimate (BE) computer codes such as APROS, ATHLET, CATHARE, MARS, TRACE were used for integral effect tests (IETs) such as LOCA and operational transient simulations. CFD code FLUENT was mainly used for the separate-effect tests (SETs) such as thermal stratification. Some of BE codes, however, were used for some SETs such as water hammer. The calculated results were presented at the PRG meetings to share the results among the participants through the discussions on the results. One of the major discussions was done for Test 6-1 where a certain discrepancy appeared in the pre-test analysis as above. An effort was done in the ROSA Project further to make a mutual comparison on the analysis methods and encountered problems to extract lessons through the questionnaire on the analysis methods for Test 6-1.

A2-2 Post-test Analysis Results

This section summarizes general points found in the post-test analyses results presented at PRGs by the participants. Major outcomes of each result are summarized in **Table A2-2-1**. The presentation materials for the PRGs are attached to this report.

The code input for TRAC-PF1 and RELAP5/MOD3.2 code for the ROSA/LSTF with their noding schematics was given to the Project Participants from JAEA to assist to initiate their code calculations. Since the Project participants mostly utilize their own computer code, these inputs were utilized as reference to prepare their own code input. Some Participants utilized their own input that was originally used for the LSTF at ISP-26 in 1992 with many modifications according to the improvements in the computer code afterwards.

A2-2-1 Subjects Found for Models in BE Codes and Their Utilization

(1) Break Flow

For all the ROSA Project experiments, orifice was used to simulate the break. Most of the code calculations were then performed by adjusting the break flow rate by comparing the calculated results with the measured data. The adjustment was done for each of single-phase liquid, two-phase and single-phase steam (high quality) discharge flows, depending on the optional input structures for each of codes and, sometimes, by modifying the codes in the source program. The major parameters were discharge coefficients and break area in a form of break size and upstream flow conditions such as liquid level in the upstream pipe (e.g., horizontal leg) and liquid entrainment from a volume to branch pipe that has the break at its end. The primary pressure was calculated well once the break flow rate was well adjusted.

This process was necessary to validate the predictive capabilities of each computer code on the points other than the break flow rate. Further discussions may be necessary on the validation of break flow model(s) and their proper utilization with a certain discharge coefficient and relevant parameters to ascertain the appropriateness of the break flow model utilization in the safety analysis as far as these parameters are adjusted in each analysis.

(2) Steam Generator U-tubes

Steam generator (SG) of the LSTF has 141 U-tubes with 9 different lengths. Six tubes of them are instrumented. It is important to simulate well the SG U-tube behavior to simulate well both the mass inventory and primary-to-secondary heat transfer. Therefore, multiple U-tubes with different length were used in the code calculations to simulate thermal-hydraulic response in parallel channels well, considering that the measured flows indicated wide variation among the U-tubes in many LSTF experiments. The calculated results well simulated the general trend. However, they did not clearly show the variety among U-tubes.

Forward-flow tubes, reverse-flow tubes and even stalled tubes co-existed in many cases for the LSTF experiment. It was difficult to simulate such responses even when 9 U-tubes were used to simultaneously simulate different size of U-tubes. The natural circulation flow rate was overestimated. The origin(s) of the difficulties have not been well identified. A single volume representation of SG inlet plenum with branching to U-tubes may be one of the origins. However, the difference in the tube length may have small impact. Since the mixing and/or stratification and phase separation and distribution in the LSTF SG inlet plenum may differ from those in the reference reactor, care should be taken well when the reactor cases are dealt with.

(3) Interphase drag

Proper prediction of interphase drag between gas and liquid phases is indispensable to correctly simulate, for example, mixture level in the core, upper plenum and upper head, and liquid level in horizontal pipes and liquid accumulation around SG inlet plenum. Mixture level in the core was mostly well predicted once the mass inventory and distribution of coolant is well predicted. In the hot leg, the loop mass flow rate and interfacial drag controls the liquid level, thus the condition of criticality for horizontal flow based on Froude (Fr) number. In **Test 3-1** (high-power natural circulation), a few codes failed to properly simulate the liquid level and flow criticality in the hot legs, probably because of some error in the interphase drag evaluation, considering that the steam generation rate in the core is well represented by giving the measured core power history.

(4) Steam condensation

When the ECCS, namely accumulators, injects coolant into cold legs, significant steam condensation takes place on the subcooled water, temporarily causing fluctuation in the primary pressure, liquid level drop in the core, loop seal clearing as well as the enhancement of coolant injection from accumulators. Some difficulties were found to simulate well such responses because the phenomena depend on local steam-liquid interfacial area and temperature difference between steam and subcooled liquid in multi-dimensional mixing process. The computer codes simulated the overall phenomena rather well, but a large uncertainty sometimes appeared in the pressure response, etc.

(5) Influences of non-condensable gas

Steam condensation is significantly deteriorated when some non-condensable gas is mixed in the steam. The non-condensable gas (Nitrogen gas) enters the primary system after the completion of coolant injection from accumulators. The influences of non-condensable gas are then important for condensation heat transfer in the SG U-tubes especially after the SG secondary-side depressurization as one of accident management measures. Nitrogen gas entered cold legs moves to and accumulates in SG U-tubes and SG outlet plenum via steam flow towards SGs. Computer codes were found to well calculate the transfer of gas as a mixture with steam and the accumulation into SG U-tubes. Most of computer codes have some difficulties in the rigorous prediction of the heat removal through the condensation heat transfer under influences of non-condensable gas. Influences of gas ingress into the decreased-number of forward flow U-tubes are significant because flow reversal and stagnation take place in a large portion of U-tubes.

As a special case, non-condensable gas entered the primary loop under single-phase liquid natural circulation in **Test 6-2** (PV bottom break LOCA) causing temporal and fluctuational increase in the primary pressure. However, computer codes failed to correctly simulate the phenomena.

(6) Bypass Flow in Pressure Vessel

This subject is not related to physical models, but to the proper prediction of the pressure balance across the core shroud in the LSTF. It was pointed-out in several post-test analyses that some adjustment is necessary for the bypass flow between the upper plenum and downcomer and between hot leg and downcomer. Since the pressure balance between these two components directly affects the core uncovery behavior, care should be taken.

A2-2-2 Findings for CFD Codes

Test 1-1 was especially dedicated to the thermal-stratification under ECCS injection into cold legs into single-phase and two-phase natural circulation and almost stagnant conditions. The thermal-stratification appeared in cold legs and cylindrical downcomer, but in some different way in two cold legs with two different types of ECCS injection nozzle. FLUENT code results were reported from three countries with various types of turbulent models, basically with k-e model, mostly for steady flow results.

For the single-phase liquid flows, flow near the downcomer, thus downstream side in the cold leg, was rather well simulated, however, a certain discrepancy persisted for flow near the ECCS nozzle. The LSTF cold legs have a bend at the downstream or upstream of the ECCS nozzle in both loops, which generates secondary flow with some mixing effect. Three support columns (o.d. 6 mm) to hold thermocouples at two locations in the straight pipe portion of each cold leg should have caused further mixing effects. Anyhow, CFD codes failed to well simulate flows near pipe side-wall (both sides) under influences of such secondary flows and flow mixing.

For the two-phase flows, operating agent reported that a new model on steam condensation at steam-water interface may improve the vertical temperature distribution in liquid flow, thus the temperature stratification and radial temperature distribution in the downcomer too.

Table A2-2-1 OECD/NEA ROSA Project Code Evaluation by Post-test Analyses

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
PRG2 2005 November at JAEA, Tokai					
1-1	Preparation for Temperature Stratification Experiment (pre-test analysis)	JAEA Japan	T. Watanabe	FLUENT	✓ Single-phase temperature distribution in cold leg was calculated well using k-e model. ✓ Noding effect was about 4 K in cold leg for 0.13-0.21 million-node simulations. ✓ Secondary flow due to the curvature was predicted in the cold leg. ✓ Temperature stratification was seen in most part of the cold leg. ✓ Temperature stratification was not significant in the downcomer during single-phase natural circulation.
PRG3 2006 May at UPV/CSN, Valencia					
1-1	Preparation for Temperature Stratification Experiment (pre-test analysis)	JAEA Japan	T. Watanabe	FLUENT	✓ Noding effects of pump outlet and downcomer on the cold-leg temperature distribution discussed. ✓ Flow velocity increased at the downcomer inlet. ✓ New condensation model was prepared and shown to be necessary for correct thermal stratification prediction in two-phase case.
6-1	First Results of a Post-Test Calculation of LSTF Test 6-1 (SB-PV-09) with the Code ATHLET (post-test analysis)	GRS Germany	H. Glaeser, H. Austregesilo	ATHLET	✓ Good accuracy in the break flow rate and the influence of the upper head (UH) liquid level on break flow, but with a little delay in core boil-off start ✓ Calculated start of core uncovery is strongly dependent on simulation of coolant flow bypass between downcomer and UH/UP ✓ Mixture level tracking model applied to UH ✓ Break flow rates calculated with ATHLET specific 1D finite difference discharge model: discharge coefficient (Cd) = 1 for subcooled and saturated flow, Cd = 0.9 for steam flow ✓ Adequate simulation of flow resistances between UP and UH, not only within CRGTs but also for the downflow between HL nozzles and core upper plate, is essential for a correct prediction of the break flow rates!

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
6-1	Rosa LSTF upper-head SBLOCA: Conversion from TRAC-P to TRACE model. Steady state comparison (post-test analysis)	UPV Spain	V. G. Llorence, S. G. Bermell, G. V. Martín	TRACE	✓ TRACE input preparation converting from that for TRAC-P, which lead to good agreement with reference results. ✓ Old *stgen* TRAC-P component seems more flexible than *htstr* component in order to model heat transfer. ✓ This steady state point has been accepted as an initial point of the transient.
6-1 & 6-2	Status Report on the activities being performed by AVN in the context of the OECD ROSA project (post-test analysis)	AVN Belgium	A.Bucalossi	CATHARE2	✓ Input deck preparation based on literature. ✓ Vessel Coarse Meshing : ➢ 1 mesh (VOLUME) for Upper Head, Upper Plenum, and Lower Plenum. ➢ Fine meshing for Core, Downcomer and Guide tubes ✓ Calculated general trends seem reasonable for "Experimental Time-dependent interventions"-type transients except incorrect vessel level behavior ✓ More detailed input deck is necessary to correctly qualify the code on "Signal-dependent interventions"-type transients
6-1	PV Upper-head Break LOCA (post-test analysis)	JAEA Japan	T. Takeda	RELAP5/ MOD3.2	✓ Discharge coefficient 0.61 for two-phase break flow ✓ Nodalization corrected considering coolant paths to CRGT (control rod guide tube) bottom in post-test analysis ✓ Overpredict break flow rate due to underprediction of break-upstream void fraction especially during two-phase flow discharge period, suggesting that code has remaining problems in proper prediction of liquid level behaviors in vessel upper-head
6-2	PV Bottom Break LOCA (post-test analysis)	JAEA Japan	T. Takeda	RELAP5/ MOD3.2	✓ Primary pressure improved by adjusting loop-B SG secondary-side temperature to meet test data after initiation of AM action ✓ Primary loop flow rate improved by increasing interfacial drag by 10 times at hot leg bend, inlet plenum and upflow-side of SG in loop-B
PRG4 2006 November at JAEA Tokai					
6-1	Preliminary post test calculation of OECD/ROSA test 6.1 with APROS (post-test analysis)	VTT Finland	P. Inkinen, I. Karppinen, J. Poikolainen	APROS	✓ Very good accuracy for overall trend including core boil-off ✓ New input deck based on RELAP5 model for ISP-26 and of JAEA-provided ✓ Heat loss and upper head initial temperature conditions adjusted in detail ✓ Break model assumptions: break-line modeling, discharge coefficient of 0.86 ✓ CRGTs have to be modeled with 2 channels. Center tubes and peripheral tubes separately.

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
6-1	Results of Post-Test Calculation of LSTF Test 6-1 (SB-PV-09) with the Code ATHLET (post-test analysis)	GRS Germany	H. Austregesilo, H. Glaeser	ATHLET	✓ More detailed consideration on the previous calculated results ✓ Influence of LSTF downcomer to upper plenum is discussed through parameter calculation. ✓ Possibility of heat loss from pressure vessel suggested as a reason for temporary large mass in downcomer ✓ Video presentation on mass inventory distribution transient
6-1	PV Upper-head Break LOCA (post-test analysis)	KFKI Hungary	V. Kerekes G. ÉZSÖL	RELAP5/ MOD3.3	✓ Noding for break unit piping with Henry-Fauske model discharge coefficient =0.85 based on parameter analyses against measured break flow rate. However, 100 s sooner in the calculation for the break flow turned from two-phase flow to single-phase vapor ✓ The processes during single phase steam break flow are mainly influenced by correct prediction of the core level, which is strongly correlated with the modeling of the UP-CRGT connection. ✓ A little delay in core boil-off start and slower system depressurization
6-1	Case 6.1. Preliminary transient results (post-test analysis)	UPV Spain	V. G. Llorence, S. G. Bermell, G. V. Martín	TRACE	✓ Detailed discussions on obtained results for steady and transient up to 500 s after the break ✓ Break flow overestimated, but the reason and consequences not well discussed
6-2	Post-calculation of ROSA/LSTF Test 6.2 using TRACE v4.274 - Preliminary results (post-test analysis)	PSI Switzerland	A. Jasiulevicius, O. Zerkak, R. Macian-Juan	TRACE	✓ First presentation on TRACE (4.274) post-test calculation result -- preliminary result ✓ TRACE input preparation with 3D vessel component to study asymmetrical behavior between two primary loops of LSTF ✓ Good overall prediction but with over prediction in SG pressure and some fluctuation in primary pressure
			PRG5 2007 May at NEA Paris		
1-2	Temperature Stratification Experiment (pre-test analysis)	JAEA Japan	T. Watanabe	RELAP5/ MOD3.2	✓ Three cases were compared to define Test 1-2: (1) 5% cold-leg break without HPI, (2) 1% cold-leg break with HPI, and (3) 5% hot-leg break without HPI. ✓ Cold-leg flow conditions at the ECCS injection timing discussed.
3-1	High Power Natural Circulation (post-test analysis)	JAEA Japan	T. Takeda	RELAP5/ MOD3.2	✓ Break flow rate overpredicted due to failure in correct simulation of cold leg liquid level especially during two-phase flow discharge period, resulting in underprediction of primary pressure, early significant drop in hot leg liquid level

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
					and temporal drop in core liquid level with loop seal clearing ✓ Liquid accumulation in SG U-tube upflow-side improved by simulating SG U-tube with fine-mesh model and by applying CCFL model with Wallis-type correlation to all the SG U-tube junctions
6-2	First Results of a Post-Test Calculation of LSTF Test 6-2 (SB-PV-10) with the Code ATHLET (post-test analysis)	GRS Germany	H. Austregesilo, H. Glaeser	ATHLET	✓ Input modified from that for Test 6-1 ✓ Simulation of break line at bottom of pressure vessel with discharge coefficient Cd = 0.92 for subcooled and saturated flow & Cd = 0.8 for steam flow ✓ SG U-tubes divided in 2 groups with different length ✓ Good simulation of overall trend for the early phase because of good prediction of sub-critical flow at break for most part of the transient by adequate simulation of flow resistances at break line, while some discrepancy in cold leg liquid level under influences of accumulator coolant injection during natural circulation ✓ Some problems to simulate influences of non-condensable gas, causing oscillation in primary pressure with considerably increasing CPU time ✓ Necessary of multiple SG U-tubes modeling pointed out
6-2	Preliminary Post Test Calculation of OECD/ROSA Test 6.2 with APROS (post-test analysis)	VTT Finland	P. Inkinen, I. Karppinen	APROS	✓ SG U-tubes simulated by 3 groups with different length ✓ Good overall simulation of transient via good simulation of break flow (not the all the transient discussed) ✓ Problems in correct simulation of natural circulation via multiple SG U-tubes under influences of non-condensable gas at rather low-pressure condition
6-1 & 6-2	ROSA-V tests 6.1 and 6.2 simulations with TRACE (post-test analysis)	PSI Switzerland	A. Jasiulevicius, O. Zerkak, R. Macian-Juan	TRACE	✓ Test 6-1 (TRACE v.5.00): Good overall transient simulation, though with a problem in convergence in break upstream pipe stratified flow and in upper head liquid behavior (Discharge coefficient (Cd) not shown) ✓ Test 6-2 (TRACE v.4.274): Good overall transient simulation, though early SG U-tube drain initiation. TRACE v.5.00 over-predicted break flow rate, causing much early U-tube drain, early termination of natural circulation and higher primary pressure ✓ Necessary of multiple SG U-tubes modeling pointed out vs. one lumped U-tube for the reported calculation.
6-2	Case 6.2. Preliminary transient results (post-test analysis)	UPV Spain	V. G. Llorence, S. G. Bermell, G. V. Martín	TRACE	✓ Comparison of calculated results up to 5,000 s in about 24,000 s transient ✓ SG RV flow rate adjusted during the cycle opening after system isolation (MSIV closure)

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
					✓ Good agreement in break flow (single-phase liquid flow) ✓ SG U-tubes drain earlier, but horizontal leg emptying greatly delayed – reason(s) are not well understood ✓ Downcomer level decrease after ECCS injection due to steam condensation was milder than in LSTF test.
PRG6 2007 November at JAEA Tokai					
1-1	FLUENT Analysis of Test 1.1 (post-test analysis)	KFKI Hungary	T. Farkas, I. Tóth	FLUENT	✓ Best Practice Guidelines for the Use of CFD in Nuclear Reactor Safety Applications by WGAMA followed; cell skewness, size change between adjacent cells, aspect ratio, and special care for mesh around pipe connection ✓ Gambit mesh generation and steady-state calculations ✓ Results from 3 types of turbulent models compared; RSM seems to predict stratification most closely, while Realizable k-ε strongly underpredicts it. RSM (and to some extent Standard k-ε) tend to overpredict jet dynamics, especially in the measurement plane „2". Change from 1st to 2nd order discretization scheme in Realizable k-ε has little effect. ✓ Effect of the CL bend upstream of the injection appeared in all calculated results as secondary flows. ✓ Transient nature of experiment pointed-out; ECCS pipe fluid temperature
3-1	Results from the simulation of LSTF Test 3.1 (SB-CL-38) using TRACE v5.0 (post-test analysis)	USNRC USA	S.O.Marshall	TRACE	✓ Good calculation of overall response: pressure, natural circulation flow rate, supercritical flow in hot leg ✓ Unexplained temporary increase in primary-side pressure at ~500 s during high-power natural circulation period caused overestimation of break flow rate and earlier core uncovery and natural circulation flow rate, and failure of proper prediction of super-critical flow in hot legs ✓ Problems with accumulator modeling resulted in inaccurate fuel rod and core-exit fluid temperatures and hot-leg level recovery
3-1	Post test calculations of ROSA LSTF Test 3.1 using RELAP5/MOD3.3 (post-test analysis)	UPC Spain	V. Martínez, F. Reventós, C. Pretel, Ll. Batet	RELAP5/ MOD3.3	✓ Very good calculations in pressure, natural circulation flow rate, SG U-tube coolant accumulation until the initiation of accumulator injection ✓ Detailed adjustment (validation) of pressure difference distribution along the primary loop and in the pressure vessel, and of downcomer-hot leg (upper plenum) bypass ✓ Significant difference in loop seal behavior around 1000 s and an important delay in the loop seal clearing as a result of a wrong accumulator injection. ✓ Differences in the core refill and significant fluctuations in the core level

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
					appeared after accumulator started coolant injection, suggesting that vessel bypasses and upper head could be a source of uncertainty ✓ Low-pressure behavior under influences of non-condensable gas did not agree well with the data
5-1	Primary Cooling through SG Secondary-side Depressurization (post-test analysis)	JAEA Japan	T. Takeda	RELAP5/ MOD3.2	✓ Simulate 141 SG tubes by 9 parallel flow channels with different lengths, namely 24 nodes for short to medium tubes and 26 nodes for medium to long tubes ✓ Code has remaining problems in proper prediction of primary loop flow rate and SG U-tube liquid level behaviors especially after enhanced depressurization through fully opening of SG safety valves in both loops.
6-2	Preliminary post-test calculation of OECD/ROSA Test 6.2 with TRACE5 (post-test analysis)	UPV Spain	S. Gallardo, G. Verdú, V. Garcia	TRACE	✓ Good agreement in general for pressures even after the non-condensable gas ingress, and primary loop natural circulation flow rate - Note: There was a problem in N2 gas discharge from accumulator tanks ✓ Temporary underprediction of primary pressure between 1000-2000 s as found in APROS/VTT calculation. There were further discrepancies in primary mass flow rates, fluid temperatures and PV downcomer liquid level. Necessity of detailed study of upper head behavior is suggested. ✓ SG U-tubes divided in 3 groups (short, medium, long) ✓ Sub-critical flow at break -- Discharge coefficient Cd adjusted to 0.78 ✓ Accumulator valve flow area adjusted
	Development of ROSA model for TRACE 5.0 (post-test analysis)	UPM Spain	G. Jimenez, C. Queral, J. Barrera, A. Exposito, P. Niesutta, L. Valle	TRACE	✓ Translation of TRAC-PF1/MOD1 input model into a TRACE input model ✓ Preparation of Animation Mask with SNAP application ✓ Several new points:2-D PZR to avoid thermal stratification, 9 SG U-tubes, heat loss, break upstream tube ✓ Future revisions (adjustments): comparison with RELAP5 input model and ROSA facility descriptions, Check impact of level tracking option inside 3D vessel, check & adjust pressure losses in 3D vessel such as bypass flow to upper plenum
6-1 & 6-2	Simulation of OECD/ROSA Test 6.1 and 6.2 (post-test analysis)	UPM Spain	C. Queral, J. Barrera, G. Jimenez, P. Niesutta, L. Valle,	TRACE	✓ Utilization of the above-prepared input model for LSTF ✓ Similar nodalization rules to be applied to Almaraz NPP analyses ✓ Plant model is validated with a real plant transient Test 6-1 ✓ Break flow model with different nodalizations before and after flow restriction (critical flow point)

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
			A. Expósito		✓ Adjust bypass flow to upper head and temperature in upper head ✓ Late start of core uncovery due probably to bad simulation of coolant distribution and/or two-phase break discharge flow -- influences of SG U-tube modeling studied (1 tube & 9 tubes) ✓ Simulation with/without core protection system and possibility of PCT higher than 1200 °C pointed-out Test 6-2 ✓ Break flow model with different nodalizations before and after flow restriction (critical flow point) ✓ Primary pressure behavior adjustment by new heat structure & heat loss ✓ SG U-tubes simulation with coarse 1-tube model ✓ Overestimation of two-phase break flow rate and early core uncovery
6-2	Revised Post Test Calculation of OECD/ROSA Test 6.2 with APROS (post-test analysis)	VTT Finland	P. Inkinen, I. Karppinen	APROS	✓ Improvement in system nodalization; core region including upper plenum divided into 3 channels, and heat conduction from upper tie plate to core barrel modeled ✓ Initial 4000 s transient was presented with good overall agreement ✓ Temporary underprediction of primary pressure between 1000-2000 s as found in TRACE/UPC calculation ✓ Problems in modeling the Upper head and reproduction of liquid level in DC was noted and probably improper heat loss from the pressure vessel were pointed-out
			PRG7 2008 May at GRS Garching		
1-1	Stratification test 1.1 Preliminary results of CFD simulations with Ansys CFX (post-test analysis)	Tractebel engineering suez Belgium	C. Mandy, S. Keijers	Ansys CFX	✓ 45° injection case selected as more representative for Belgian plants ✓ Wide range simulation including pressure vessel downcomer and cold leg in the other side ✓ Shear Stress Turbulence model (Combined 2 equations model) ✓ Time step 0.05s for transient calculation with 7 internal iterations at maximum and polynomial function input for ECCS mass flow rate, temperatures and pressure boundary at downcomer outlet ✓ First-look analysis results indicated with future planning
3-1	OECD/ROSA Test 3.1 TRACE5 (post-test analysis)	UPV Spain	V. Abella, S. Gallardo, G. Verdú	TRACE	✓ Overall response was well simulated, including hot leg supercritical flows ✓ Primary pressure decrease rate over-estimated with early accumulator injection, late core uncovery, probably because of disagreement of break flow

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
					due to disagreement in the cold leg liquid level prediction ✓ Level drop in downcomer larger than in upper plenum and core ✓ Significant accumulator coolant injection with significant steam condensation and abrupt level recovery in the core, upper plenum and hot legs not well predicted ✓ Increase in the break valve friction by factor 2 caused decrease in the break flow rate and better primary pressure transient, but more delay in core uncovery initiation ✓ Increase in hot leg bypath friction caused better primary pressure, but not so significant improvement in the range tested ✓ More parameter survey are being planned (ex.) for friction factors in cold legs and pressure vessel, and CCFL coefficients and modes (location unclear)
3-2	High Power Natural Circulation (LOFW w/o Scram Test 3-2) (post-test analysis)	JAEA Japan	T. Takeda	RELAP5/ MOD3.2	✓ PZR liquid level behaviors improved by applying liquid entrainment model to surge-line inlet and CCFL model with Wallis-type correlation to PZR bottom ✓ Code has remaining problems in proper prediction of primary loop flow rate and SG U-tube liquid level behaviors with odd oscillation. ➢ Influences of multi-channel model for SG U-tubes are to be investigated by simultaneously applying CCFL model with Wallis-type correlation to inlet plenum and all the U-tube junctions of SG
5-1	Preliminary Post-test Calculation of ROSA/LSTF Test 5-1 with APROS (post-test analysis)	VTT Finland	P. Inkinen	APROS	✓ Noding completely 1-dimensional (including the core that is divided into 3 regions) with 3 SG U-tube groups ✓ Good overall prediction, except low-pressure transient especially after SG depressurization ✓ Flow rate in the long U-tube group behaved unexpectedly ➢ Flow stopped in the other U-tubes after the RV open as in the experiment ➢ After the opening of SVs flow rate decreased in the long U-tube group and increased in the other U-tube groups ✓ Cold leg liquid levels remain at higher level due to lower pressure vessel liquid level, causing an increase in break flow rate through the end of the transient
5-2	Primary Cooling through SG Secondary-side Depressurization (Non-condensable Gas Inflow	JAEA Japan	T. Takeda	RELAP5/ MOD3.2	✓ Simulate 141 SG tubes by 9 parallel flow channels with different lengths, namely 24 nodes for short to medium tubes and 26 nodes for medium to long tubes ✓ Code has remaining problems in proper prediction of primary loop flow rate

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
	Test 5-2) (post-test analysis)				and SG U-tube liquid level behaviors that were observed with different tendency among instrumented tubes especially after non-condensable gas inflow, suggesting that uncertainty remains in prediction of steam condensation in SG U-tubes which typically appears under influences of non-condensable gas.
6-1 & 6-2	OECD/ROSA Tests 6.1 and 6.2 with TRACE5 (post-test analysis)	UPV Spain	V. Abella, S. Gallardo, G. Verdú, V. Garcia	TRACE	✓ Similar models which differ mainly in the break situation ✓ Transform the model from TRAC-P to TRACE, including ➢ Re-design of components (i.e., SG, PV) ➢ Sensitivity analyses of physical parameters such as friction coefficients, discharge coefficients for (break) critical flow model, and for the heat transfer coefficients. Test 6-1 ✓ Good overall behavior prediction by using fitting discharge coefficients for single-phase liquid and two-phase break critical flows. Steam discharge rate was over-predicted because of lack in the coefficient fitting. ✓ Faster depressurization of SGs than in experiment, which made the SG pressure reached to the primary pressure far earlier. Heater rod surface temperature excursion rate was influenced by the fast decreasing primary pressure (saturation temperature). Test 6-2 ✓ Good overall behavior prediction by using a fitting discharge coefficient of 0.78 for break critical flow. Good primary pressure response after non-condensable gas ingress. ✓ Core uncovery started a little earlier than in experiment ✓ Qualitatively good calculation of unbalanced cold leg level. ✓ It was suggested that reverse friction coefficient may have a decisive influence onto the cold leg behavior calculation.
6-1	Post-test Thermal-Hydraulic analysis of ROSA Test 6.1 using TRACE (post-test analysis)	PSI Switzerland	J. Freixa, A. Manera, O. Zerkak, J. Dreier	TRACE5	✓ Single SG U-tube and all bypasses nodalized according to the geometry and the friction k-factors being adjusted to have the bypasses mass flows during steady-state match the experimental values ✓ Good overall response calculated including heater rod temperature that recorded PCT, by using default break discharge coefficients. A little delay in the transition from two-phase flow to single-phase vapor discharge, and overprediction of steam discharge rate. ➢ Additional (new) discharge coefficient of 0.8 for steam discharge improved

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
					the result. ✓ Phenomenological study performed to clarify reason(s) of discrepancy in the DC and cold leg level behavior during blowdown phase. Additional leakage at DC-UP by 1% of the core flow at steady state conditions dramatically found to improve the result.
	A TRACE model of the ROSA/LSTF	VTT Finland	S. Hillberg	TRACE5	✓ The model is built with SNAP, input has not been directly edited ✓ As for pressure vessel, for example, 16 axial levels, 4 radial rings, 4 azimuthal sectors ✓ A work in progress
PRG8 2008 November at OECD Paris					
1-1	FLUENT Analysis of a ROSA Cold Leg Stratification Test (post-test analysis)	KFKI Hungary	T. Farkas, I. Tóth	FLUENT	✓ Paper for XCFD4NRS in September, 2008 ✓ Follow-on of PRG6 results, utilizing almost the same method such as consideration to the OECD Best Practice Guidelines (BPG) ✓ 11 cases with variation in 3 turbulence models, standard wall functions, full buoyancy effects, SIMPLE pressure-velocity coupling etc. ✓ Standard k-ε & RSM overestimate buoyancy effect and indicate less intensive mixing of cold plume, while Realizable k-ε model underestimates buoyancy effect but with more effective mixing ✓ Necessity of more detailed information on the flow field such as velocity distributions with some turbulence properties pointed-out
1-1	Analysis of Temperature Stratification Phenomena Examination in Cold Leg with FLUENT code (post-test analysis)	JNES Japan	Y. Masuhara, F. Kasahara	FLUENT	✓ Analysis of temperature stratification phenomena during single-phase natural circulation mode with 2 different mesh numbers for each of 2 horizontal legs and a part of vessel downcomer ✓ Standard k-εmodel showed a good prediction among compared two types of k-εmodels ✓ In loop-B with inclined ECCS nozzle, thermal stratification was not properly simulated
3-1 & 3-2	OECD/ROSA Test 3.1 and 3.2 Simulation with TRACE5 (post-test analysis)	UPV Spain	V. Abella, S. Gallardo, G. Verdú	TRACE	Test 3-1 ✓ Revision of the effort for the previous PRG7 ✓ Results including liquid levels in core and cold leg improved, but with no explanation on the change(s) in the analysis method ✓ Coolant accumulation in SG U-tubes overestimated at early stage of transient, causing longer time to complete coolant drain

ROSA Test	Title	Organization	Author(s)	Code	Major results and suggestions to improve and develop code and/or model
					✓ Overestimation of break flow at low-pressure transient after completion of core reflooding, due probably to temporary overestimation of liquid level in cold leg-B Test 3-2 ✓ Overall transient was well simulated, but with overestimation of natural circulation flow rate in both loops suggesting remaining problems in the prediction of flows in multiple parallel SG U-tubes ✓ Overestimation of void fraction in the core during two-phase natural circulation phase, and rather abrupt drop in the liquid levels in pressure vessel and hot legs at the termination of the two-phase natural circulation
6-1	Plant Applications of ROSA 6.1 Test. Accident management actions in an Upper Head SBLOCA (post-test analysis)	UPM Spain	C. Queral, A. Expósito, L. Valle, G. Jiménez, E. Villalba, S. Beneyto	TRACE	Test 6-1 ✓ Basically, the same input was used for PRG6 in 2007 Nov. ✓ Influences of reflood option and reverse friction factors tested -- combined utilization may improve the core rod temperature (PCT). ✓ Mutual comparison among 5 Participant results and LSTF data -- AEKI is the best of all, but UPM still overestimated break flow rate at steam discharge phase, and late core uncovery happened. ✓ Revision of input (nodalization etc.) is to be performed. ✓ CET response and effectiveness of RVLIS (In-vessel liquid level monitor) discussed through parametric study on influences of AM depressurization timing onto PCT PWR analysis ✓ Break area adjusted to obtain similar pressure transient than in ROSA 6.1 test and power scaled with respect to primary volume ✓ Influences of available number of accumulators onto PCT studied parametrically in conjunction with RVLIS condition changes
6-2	Preliminary post-test calculation of OECD/ROSA Test 6.2 with TRACE5 (post-test analysis)	UPV Spain	S. Gallardo, G. Verdú, V. Garcia	TRACE	✓ During really transient of 1000-2000 s, underprediction of primary pressure, overprediction of primary loop mass flow rate and overprediction of downcomer liquid level appeared. ✓ Influences of nitrogen gas inflow were not well predicted in such parameters as primary pressure, collapsed liquid level in SG U-tube in loop without PZR and hot leg liquid level.

A2-3 Questionnaire for Test 6-1 Analysis by Project Participants

A2-3-1 Background and Objective

Many post-test analyses were performed for Test 6-1 (PV upper-head LOCA with AM measure) by using such best estimate (BE) LOCA analyses codes as ATHLET, APROS, CATHARE, RELAP5 and TRACE, and were presented at several PRG meetings of the ROSA Project. The calculated results agreed well with the experimental results. Several participants even extrapolated their results into a postulated condition, as a trial basis, to expect the true peak cladding temperature (PCT) by assuming that the core power did not trip-off even after the limiting core temperature in the LSTF experiment.

Based on the good agreement of results with the LSTF experiment data, and partly because the five major different computer codes were utilized so far by 4th PRG in 2006, a questionnaire was proposed to draw practical information mutually of interest among participants. The questionnaire asked some details of the models and methods used for their post-test analysis, if any. Efforts associated with performing the analyses such as manpower and time spent to complete the analyses were also asked as well as their computing circumstances. Replies were compared each other to clarify their practical efforts to obtain the good results by means of some practical analysis methods including selection and tuning of model(s) and calculation method(s) that would be applied for the reactor safety analysis (extrapolation from the scaled-down problem).

The questionnaire, however, was not meant to compare the calculated results themselves, considering that it was possible to incorporate some efforts into the post-test calculations by optimizing the code with the best accuracy. This attempt is, therefore, different from the ISP (international standard problem) in this respect.

The obtained results are summarized in the following.

A2-3-2 Questionnaire

The questionnaire and the mutual comparison of five replies are respectively shown in **Attachment A2-3-1 and A2-3-2**. The questionnaire is composed of the following four items;

1. Computer Code

   Specific features of the code and motivations to perform post-test analyses

2. Efforts

   Computer and computing time, and manpower characterization

3. Selection Method(s) of Model(s) and Method(s) specific to the Code

   Strategy to put priority on specific parameter(s), selection of specific model(s)/method(s), simulation of multi-dimensional phenomena

4. Models and Methods Specific To Test 6-1

   Method(s) to use five key models, and strategy to use the models in reactor analyses

A2-3-3 Major Results

For each of 4 items, major results are summarized as follows.

(1) Computer code

The following five computer codes were used; ATHLET Mod 2.1 Cycle A (GRS*), APROS 5.06 (VTT*), CATHARE2v15b mod5.2 (AVN*), TRACE 5.0RC3 (DSE-UPM*) and RELAP5 MOD3.2.1.2

* GRS: Gesellschaft für Anlagen- und Reaktorsicherheit mbH, Germany

(JAEA). Three TSOs (technical safety/support organizations) and two universities who support regulatory body provided answers to this questionnaire by June 28, 2007. The common major motivation to perform the post-test analysis was the validation and verification of their code in preparation for the safety analysis.

(2) Efforts

The calculation was performed mostly by using a so-called personal computer or its cluster with an operating system such as Linux and Microsoft Windows XP. The CPU time ratio to the physical time was ranging from about 0.3 to 13, depending on the code and number of meshes.

The total work hours by the personnel who made the calculation (analyst) significantly varied from three weeks to 1 year depending on their experiences to conduct code calculation. The analyst had at least one co-worker (and more at several organizations) to discuss with on their analyses.

(3) Selection Method(s) of Model(s) and Method(s) specific to the Code

Three organizations put their priority onto the correct calculation of the break flow rate and liquid levels in the core, upper head and upper plenum in the pressure vessel. The liquid level in the vessel upper head is important for the correct prediction of break upstream condition. One organization utilized an input used for ISP-26 (5% cold leg break LOCA test by ROSA/LSTF, 1992) with some modifications according to the facility modifications since the ISP-26.

The computer codes were used without any modifications except RELAP5 code, to which the break flow models were incorporated for better calculation of two-phase discharge flow through an orifice. The current computer codes are considered well improved through many of analysis experiences. However, care was taken to select appropriate models such as VOLUME model to calculate mixture level in the case of CATHARE code. As for the ATHLET code, 6-equation model (fully separated momentum equations) or 5-equation model (mixture of momentum equation and drift-flux approach) were applied to appropriate volumes. The break discharge coefficients and bypass flow were adjusted further to have good agreement with the measured data.

As for the simulation of three-dimensional phenomena, no special measures were taken except the 3-D noding in the pressure vessel employed by the ATHLET and TRACE codes. The other components were simulated basically by using zero-dimensional volume components (ATHLET and CATHARE codes) and one-dimensional components such as pipe.

(4) Models and Methods Specific To Test 6-1

Break Flow

The break flow is the most important parameter for the correct simulation of LOCA because it controls the mass and energy inventory in the system. Each computer code utilizes it's own set of critical (choking) flow models to calculate the break flow rate for single-phase (subcooled) liquid flow, (saturated) two-phase flow and single-phase steam flow with a discharge coefficient for each model. Non-equilibrium flow condition may be handled too. In some cases, analysts changed the model(s) for the proper simulation of the flow through orifice break used in the **ROSA Test 6-1**. When they use default models, they followed the guideline in the user manual. For all the cases, however, the discharge coefficients for three flow conditions were respectively adjusted against the measured data. The same set of models and methods including the upstream piping are to be used for the safety analysis. The discharge

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VTT: Valtion Teknillinen Tutkimuskeskus, Finland

AVN: Association Vinçotte Nuclear, Belgium

DSE-UPM: Departamento de Sistemas Energeticos - Universidad Politechnica de Madrid, Spain coefficient would then be one of the primary parameters to discuss the influences of uncertainty.

Liquid level in vessel upper head

In **ROSA Test 6-1**, this and the following two (in total three) parameters are important for the proper simulation of the break upstream pipe flow conditions. Two methods were used depending on the computer codes. The ATHLET and CATHARE codes used a single volume to simulate whole upper head, being composed of two sub-volumes that respectively represent upper steam phase with liquid entrainments and lower two-phase mixture phase. The other three codes used vertical stack of single volumes (pipe). The mixture level is defined from the void fraction. A level tracking model is employed by the TRACE code. Interphase drag was decreased by 1/10 for the proper simulation for the RELAP5/MOD3.2 code. The calculated results, however, seem to be influenced by the nodalization (number of slices of the vertically stacked volumes) for the latter three codes. For APROS code, flow resistances were adjusted in upper head as well as in control rod guide tubes, upper plenum and downcomer. The same model and methods (strategy) are to be used for the reactor safety analyses. This particular case, thus the simulation of mixture level in a large volume such as upper head, would not be so much influential in the scaling of phenomena.

Vapor/Liquid entrainment in vessel upper head

Every computer codes have an entrainment model for the branch pipe flow. This model was applied to the nozzle that connects to the break upstream pipe, except for the RELAP5 code case. The ATHLET and CATHARE codes employed further an optional vapor pull-through model considering the high-speed steam flow. A horizontal pipe was directly connected to the upper head from the break upstream pipe in the ATHLET code model, while a 45-degree inclined pipe was used to connect the upper head and the break upstream horizontal pipe in the LSTF. As for the RELAP5 code, fine mesh representation was performed to the volume above the top of CRGT. The interphase drag for these fine-mesh volumes was decreased to 1/10. The same model(s) and methods are to be used for the reactor safety analyses.

Flows through multiple Control Rod Guide Tubes (CRGTs)

There are eight simulated CRGTs in the LSTF, which connect the upper plenum and upper head. Good simulation of the flow through the CRGTs is indispensable for the proper prediction of the break upstream conditions. The parallel channels of CRGTs were simulated in the code calculations by either (two or eight) parallel multi-channels or one lumped flow channel, depending on the core and upper plenum noding. Eight PIPEs were employed for the TRACE analysis to achieve rigorous simulation by fully utilizing the 3D VESSEL component. Two parallel pipes were employed for the ATHLET and APROS analyses by considering the change in the flow conditions depending on the location of CRGTs; above the high-power or low-power fuel bundles. One lumped flow channel was selected for the CATHARE and RELAP5 analyses for simplicity. Consequently, It seems that the parallel multi-channel simulation gave better results in the flow through the CRGTs. Care was taken further in the ATHLET analyses such that the model option for the drift flux-based interfacial friction in the upper plenum volume was selected to have one for the pipe geometries rather than for the usual bundle geometries. The higher interfacial friction was resulted, causing higher resistance for the liquid downflow. In the RELAP5 analysis, the coolant conditions in the upper plenum were controlled by the number of slices of vertically stacked volumes. The same model(s) and methods are to be used for the reactor safety analyses.

Liquid level in the core and rod surface temperature

The interphase drag is defined based on the flow regimes in the core rod bundle for appropriate representation of vertical void fraction distribution under mixture level in the core. In the ATHLET code, the code-specific flooding-based drift flux model was used to define the interphase drag. For the RELAP5/MOD3.2 code, the interphase drag was reduced to 1/10 based on the previous experiences. The heat transfer in the core was calculated by using the default set of heat transfer correlations for Test 6-1 by all the codes, though the correlations are different from code to code. The same model(s) and methods are to be used for the reactor safety analyses.

A2-3-4 Summary

Five detailed replies to the questionnaire were compared and summarized to reveal the efforts done by the participants to attain the good simulation results for ROSA Project Test 6-1 by using the BE LOCA codes. Efforts associated with the analyses work such as manpower and time spent to complete the analyses were included further in the questionnaire as well as the computing circumstances. All the analysts were found to work for the regulatory body as TSO and university. The motivation to conduct the post-test analyses was the validation and verification of the BE codes to finally apply it to the reactor safety analyses.

It appeared that all the BE codes request the analysts to make many selections and decisions against many thermal hydraulic phenomena simulation, on the noding, models and combination of models and coefficients, especially when the multi-dimensional phenomena and the multiple parallel flows are involved. The analyst then may need very long time to build and adjust their input model, depending on their experiences. They referred user manual and had a cooperation of their colleague(s) to obtain the best results. This style of work is surely good to minimize the user effect.

The computer codes still have difficulties in the simulation of multi-dimensional phenomena except those in the pressure vessel for some codes that have a capability of three-dimensional node representation.

The break flow as the most important parameter for the rigorous simulation of LOCA was obtained well by mainly adjusting the discharge coefficients for break models that are different from code to code. Efforts were taken to achieve the proper simulation in the break upstream conditions and the mixture levels in the pressure vessel; core and upper head. Basically, the models and methods used for this post-test analysis are to be used for the reactor safety analyses without changes. It was difficult to fully clarify the influences of the phenomena scaling within the present questionnaire effort.

Good information was obtained in total to look into the motivation, circumstances and efforts associated with the post-test analysis of ROSA Project Test 6-1 by using the BE LOCA codes. The efforts taken for and experiences and information gained through the post-test analyses and the questionnaire would be beneficial in the preparation of PIRT (Phenomena Identification and Ranking Table) and in the BEPU (Best Estimate plus Uncertainty) analyses for the safety evaluation of current and future LWRs.

Attachment A2-3-1

OECD/NEA ROSA Project
Data/Information Transfer

January 10, 2007
Thermohydraulic Safety Research Group
Nuclear Safety Research Center
JAEA

General questionnaire on the analysis methods for Test 6-1

Various post-test analyses have been presented in PRG3 and PRG4 by the project participants to validate predictive capabilities of best-estimate (BE) computer codes such as ATHLET, CATHARE, TRACE, APROS and RELAP5/MOD3 against the ROSA/LSTF Test 6-1 data. Each presentation has shown nice calculated results with accuracy. However, the indicated results seem to rely on their specific models and nodings to simulate thermal-hydraulic phenomena in the test facility based on the long-term experiences against various subjects. In the presentation, details of such aspects are sometimes not included as a kind of common sense to the relevant researchers/engineers, while not for the other participants. Such a trend may be implicitly emphasized in post-test analysis. It is thus good to summarize the major aspects of computer codes and associated strategy to select primary models and calculation methods used to analyze the ROSA/LSTF Test 6-1 as well as motivations to perform the code verification and development.

Following is a list of tentative questionnaires to collect and summarize necessary information to clarify such details to thoroughly understand the calculated results for mutual comparison and discussions. Every participating organization of the OECD/NEA ROSA Project is requested to reply to the questionnaires, even when the code analysis is not performed, by assuming that some analyses are to be performed.

No.	Item	Questionnaire
		1. Computer Code
1-1	Name	Name of the code (used or to be used)
1-2	Specific Feature(s)	Specific feature(s) (as BE code)
1-3	Motivations	(1) Specific objective(s) to develop/improve the code (as BE code) (2) Reason(s) to perform analyses by using the code against ROSA Test 6-1 (3) Expected practical subjects for the code to be used for
		2. Efforts
2-1	Noding	Number of volumes and junctions
2-2	CPU Time	(1) Name of CPU and OS (2) Typical CPU time for physical time (3) CPU time ratio = (Typical CPU Time) / (Physical Time)

No.	Item	Questionnaire
2-3	Manpower	(1) Number of personnel (calculator) who performed the code calculation (2) Number of personnel who contributed the calculation work through discussions with calculator(s) (3) Calculator -- a/Regulator, b/Supporting Organization, c/University (i/Professor or ii/Student), d/Vendor, e/Utility, f/External employee of a/, b/, d/ or e/, g/others (specify) (4) Career (years) of calculator(s) to perform the code analysis of this kind (5) Total work hours for the main calculator to provide the results
	3. Selection Method(s) of Model(s) and Method(s) specific to the Code	
3-1	Strategy	Key information(s) and/or consideration(s) to put priority on major calculation parameters such as liquid level in upper plenum
3-2	Selection of Model(s) and Method(s)	(1) Method(s) to select the model(s) in the code and method(s) to handle the code with validation method(s) for the selection of the model(s) and/or the method(s). List all of the relevant model(s) and/or method(s). (2) Try & error process for the selection and/or elimination of the model(s), the specific constant(s) and/or assumption(s) for the model(s) and the method(s), if any.
3-3	Simulation of multi-dimensional phenomena	(1) Practical method(s) to simulate 3-dimensional two-phase flow phenomena by using the code ex.) 3-dimensional temperature distribution around the core during the core temperature excursion to have an exact prediction of the core exit temperature to start accident management action.
3-4	Statistical uncertainty definition method	(1) Experiences applicable, for example, for the optimized setup of noding in combination with liquid level tracking model, if any.
	4. Models and Methods Specific To Test 6-1	
4-1	Break flow	(1) Break flow model and specific coefficient with method(s) to define them (2) Strategy to apply them for reactor cases, namely in safety analysis
4-2	Liquid level in vessel upper head	(1) Method(s) and/or phase separation (inter-phase drag) model(s) to identify the location of liquid level (mixture level) in the vessel upper head, in conjunction with noding modeling (2) Strategy to apply them for reactor cases, namely in safety analysis
4-3	Vapor/liquid entrainment in vessel upper head	(1) Method(s) and/or model(s) to estimate the (two-phase) flow through rather large volume vessel upper head towards break pipe (2) Strategy to apply them for reactor cases, namely in safety analysis
4-4	Flows through multiple Control Rod Guide Tubes (CRGTs)	(1) Method(s) and/or two-phase flow (flow regime definition) model(s) to simulate multi-channel parallel pipe flows from upper plenum to upper head (2) Strategy to apply them for reactor cases, namely in safety analysis
4-5	Liquid level in the core and rod surface temperature	(1) Method(s) and/or phase separation (inter-phase drag) model(s) to simulate void fraction (distribution) in the core that defines two-phase mixture level in the upper plenum and core multi-channel (2) Boiling heat transfer model(s), definition of dryout, post-dryout heat transfer model(s) to simulate temperature excursion in the core (3) Strategy to apply them for reactor cases, namely in safety analysis

Attachment A2-3-2

Answers of General questionnaire on the analysis methods for Test 6-1 -- as of June 28, 2007 **--**

No.	Answers by GRS (ATHLET)	Answers by VTT (APROS)	Answers by AVN (CATHARE)	Answers by DSE-UPM (TRACE)	Answers by JAEA (RELAP5)
1-1	ATHLET Mod 2.1 Cycle A (version released in July 2006)	APROS 5.06	CATHARE2v15b mod 5.1	TRACE 5.0 RC3	RELAP5/MOD3.2.1.2
1-2	The thermal-hydraulic computer code ATHLET is being developed by the GRS for best-estimate analyses of plant transients, design basis and beyond design basis accidents in light water reactors. The code is composed of several basic modules: a thermo-fluiddynamic module, a heat conduction and heat transfer module, a neutron kinetics module, a general control simulation module, and a general solver of differential equation systems. The thermo-fluid-dynamic module includes a two-fluid formulation, with fully separated balance equations for liquid and vapour. It is complemented by mass conservation equations for up to 5 different non-condensable gases and by a boron tracking model. Specific models for pumps, valves, steam separators, mixture level tracking, critical flow, among others, are also available within ATHLET.	- 6 equation thermal hydraulic model - 1 and 3-D neutron kinetics - integrated containment model - comprehensive modeling of control and automation systems - graphical user interface with process component library	CATHARE2 (Code for Analysis of Thermal-Hydraulics during an Accident of Reactor and safety Evaluation) has been developed in Grenoble by the French Atomic Energy Commission (CEA), Electricité de France (EDF) and AREVA to perform Best-estimate calculations of PWR accidents. Based on a one-dimensional, two fluids, six equations model with a unique set of constitutive laws. <u>1-D AXIAL module</u> This module solves for each mesh the following equations: • 2 energy balance equations (liquid and vapour phase) • 2 momentum balance equations (liquid and vapour phase) • 2 mass balance equations (liquid and vapour phase) • 1 or 2 mass balance equations for 1 or 2 noncondensable gasses for a total of 6+2 equations per mesh. It is generally used for piping (i.e. hot legs, cold legs, core channels, SG U-tubes, SG downcomer, SG riser, feedwater lines, vapour lines, etc). <u>1-D VOLUME module</u> This module consists of 2 subvolumes (INF and SUP). In each one are solved: • 2 energy balance equations (liquid and vapour phase) • 2 mass balance equations	1) 1D/3D, six/eight equation models, non-equilibrium, non-homogeneous model. Also includes non condensable gas and boron transport. 2) It includes a 3D VESSEL model.	1) 1D, 2-fluid, non-equilibrium, non-homogeneous model for 2ø steam-water mixture that can contain non-condensable gas in steam phase and/or soluble gas in water phase 2) System component models such as pressurizer, accumulator, valve, pump, separator and control system 3) Core power is calculated by point-reactor kinetics models with reactivity feedback or is given by pre-programmed time-dependent power table.

No.	Answers by GRS (ATHLET)	Answers by VTT (APROS)	Answers by AVN (CATHARE)	Answers by DSE-UPM (TRACE)	Answers by JAEA (RELAP5)
			(liquid and vapour phase) • 1 or 2 mass balance equations for 1 or 2 noncondensable gasses For the total Volume is assigned one pressure equation for a total of 9+2 equations per mesh. For each junction connected to the VOLUME are associated two momentum balance equations (liquid and vapour phase).		
1-3	(1) The calculations have been performed in the frame of the code validation. (2) Post-test calculation of separate effect tests and integral tests are main contributors to the enhancement of the validation status of ATHLET. The selection of experiments for code validation takes into account the current status of code development, requirements of code users, current safety issues as well as the availability and adequacy of new experimental data. The OECD/ROSA Test 6-1 has been recently added to the ATHLET code validation matrix (Accident management for a non-degraded core in PWRs).	(1) To enhance code capabilities for safety analysis and licensing. (2) Code validation (3) Safety analysis of EPR	(1) Framework of code validation for safety analysis (2) Code Validation for Vessel SBLOCAs (3) Safety Analysis for Belgian Power plants	(1) Validate the code capabilities in these conditions. (2) We are interested in small break LOCA analysis at top and bottom vessel. (3) We want to simulate top/bottom vessel SBLOCA sequences with our TRACE model of Almaraz Nuclear Power Plant (Westinghouse 3 loop) and, we need to learn which are the nodalization, models and options adequate in this kind of sequences.	(1) Assess the predictability of RELAP5/MOD3.2 code and models for system integral analyses and modify the models (2) Find out problem(s) in RELAP5/MOD3.2 code to properly predict thermal-hydraulic phenomena during vessel upper head small break LOCA (3) Best-estimate plus uncertainty analyses with RELAP5/MOD3.2 code
2-1	80 thermo-fluid-dynamic objects and 56 heat conduction objects, with a total of 344 control volumes, 287 junctions and 261 heat slabs.	340 volumes, 342 junctions	456 scalar meshes	284 1D cells, 734 3D cells	About 190 volumes, about 200 junctions, about 180 heat slabs
2-2	(1) CPU: A Linux Cluster with 2 AMD single core processors (2.4 GHz). (2) The total CPU time up to the start of LPI injection was 2100 sec (3) An average CPU time ratio of 0.8.	(1) CPU: Intel Pentium 4, 3.00 GHz OS: Windows XP Professional (2) 3.75 (3) 12000/3200	(1) Intel Pentium 4 3.20 GHz, 4GB RAM, Linux Fedora Core 1.0 (2) 1000s for 13500s (3) 1000s/ 13500s	(1) vendor_id: AuthenticAMD, cpu family: 15, model: 37, model name : AMD, Opteron(tm) Processor 252, stepping : 1 cpu MHz : 996.164 cache size : 1024 KB (2) 3.2	(1) CPU; Intel Itanium 2, OS; SGI Linux (2) Typical CPU-time; 950 s (3) Typical CPU / physical time ratio; 0.32 (= 950 s / 3000 s)

No.	Answers by GRS (ATHLET)	Answers by VTT (APROS)	Answers by AVN (CATHARE)	Answers by DSE-UPM (TRACE)	Answers by JAEA (RELAP5)
2-3	(1) One senior analyst (2) One coworker has contributed to the analysis through discussions of the results. (3) (4) Several years of experience in code development and application for reactor safety analysis (5) The total manpower is estimated to 2 persons over 1 month.	(1) 1 person (2) 1 person (3) c ii (4) 0 (5) 600 hours = 4 months (ex. 5 hours a day for analysis work)	(1) 1 person (2) 1 person (3) b-TSO (4) 10 years (5) 3 weeks	(1) 1 person (2) three persons of DSE-UPM, one full time analyst and two part time analyst (3) c ii (4) one (5) one year, because we are performing a new model of ROSA facility.	(1) One analyst (2) Four persons of JAEA who contribute to analysis through discussions and one analyst (3) JAEA -- b (4) Analyst who engages in RELAP5 code analyses for several years (5) Over 1 month
3-1	No specific strategy was adopted for this calculation. The ATHLET input data was strongly based on the data set used for the calculation of the ROSA-IV LSTF small break test SB-CL-18 in the frame of code validation, taking into account the modifications performed in the facility for the ROSA-V Program.	Primary pressure and liquid level in the core	Priority on proper break flowrate. The major investigated parameters were levels within the vessel (especially Upper head, Upper plenum, and core)	No specific strategy was adopted for this calculation.	Proper prediction of break flow rate and liquid levels in upper head, upper plenum and core as essential boundaries of vessel upper head small break LOCA
3-2	(1) The selection of methods and modeling options was largely based on the recommended values presented in the code User's Manual /1/, which in turn reflect the experience acquired by the extensive code validation. For the primary circuit of the LSTF facility the 6-equation model (fully separated momentum equations) was applied, except for the pressurizer, control rod guide tubes, upper head and break line, where the 5-equation model (mixture momentum equation and drift-flux approach) was used. The secondary side was simulated with the 5-equation model. (2)	(1) Standard models used (2) Break flow discharge coefficient and flow resistances in different parts of the pressure vessel were adjusted to fit experiment.	(1) Standard CATHARE model of Upper Head and upper plenum (VOLUMEs model) (2) Break size, Finer Upper Plenum and Upper Head Modeling	(1) Standard models. Two dimensional pressurizer and several heights in steam generator U-tubes (2) Adjust head bypass flowrate	(1) A two-phase critical flow model is incorporated by JAEA [1], which employs the Bernoulli equation (incompressible flow) with a discharge coefficient (Cd) of 0.61 for single-phase liquid and the maximum bounding flow theory with a Cd of 0.61 for two-phase flow. A Cd of 0.84 is used for single-phase vapor. (2) 1) Modify noding for upper head above top of CRGT and interfacial drag in each node to properly predict liquid level in upper head 2) Adjust flow resistances of flow path between downcomer and upper head to fit downcomer-to-upper head bypass flow rate to 0.3% of total core flow rate during initial-steady state.
3-3	Multi-dimensional phenomena can be simulated approximately with the parallel channel technique (one-	No 3-dimensional modelling	No 3-D model was used	(1) 3D Vessel	Multi-dimensional phenomena cannot be analyzed by RELAP5/MOD3.2 code with one-dimensional model. As for simulation

No.	Answers by GRS (ATHLET)	Answers by VTT (APROS)	Answers by AVN (CATHARE)	Answers by DSE-UPM (TRACE)	Answers by JAEA (RELAP5)
	dimensional, parallel pipes connected by cross-connection junctions) in ATHLET. For Test 6-1, the core region has been modeled by two concentric rings, connected by a cross-connection object. The inner channel ring includes the high and middle powered bundles (13 to 20 and 21 to 24). The low powered bundles are combined in the outer channel ring. The upper plenum region up to the hot leg nozzles as well as the control rod guide tubes are divided accordingly.				of multi-dimensional core phenomena, for example, the use of computer code with three-dimensional core model such as TRAC-PF1 is needed.
3-4	In near future it is planned to apply the GRS methodology for uncertainty and sensitivity analyses for an uncertainty analysis of the Test 6-1 calculation, taking into account not only the code uncertainties (input parameters and modeling options) but also experimental uncertainties (e.g. core bypass flows).		None	No experience in evaluation of uncertainties in code analysis by using statistical procedure	No experience in evaluation of uncertainties in code analysis by using statistical procedure
4-1	(1) Critical discharge flow rates are calculated in ATHLET by a one-dimensional thermo-dynamic non-equilibrium model, taking into account the actual geometry of the discharge flow path /1/. This model generates tables of critical mass fluxes for a set of fluid conditions in the upstream discharge control volume. For pure steam flow, a discharge coefficient of 0.9 has been applied, for subcooled and saturated flow 1.0. (2) The same strategy is applied for reactor safety analyses.	(1) Critical flow model, discharge coefficient 0.86 (2)	(1) The Break is modeled with a TEE and AXIAL pipe and a Boundary condition as specified by the user guidelines. This is the recommended modeling procedure, particularly for breaks on horizontal pipes; since it takes into account the phase separation effects which are modeled in the TEE (see qualification reports). No specific phase separation law was developed in the TEE for a vertical pipe. Even without a phase separation phenomenon, the critical flow prediction is more accurate when calculated with 1-D modeling of the flow in the nozzle (see Qualification reports) than by using a critical flow correlation function of upstream parameters (Break sub-module).	(1) Default model. Discharge coefficients still not adjusted.	(1) A two-phase critical flow model employs the Bernoulli equation (the incompressible orifice flow equation) with a discharge coefficient of 0.61 for single-phase discharge liquid and the maximum bounding flow theory with a discharge coefficient of 0.61 for two-phase discharge flow. A discharge coefficient of 0.84 is used for single-phase discharge vapor. (2) The predictability of the break model has been assessed through analyses of LSTF small break LOCA experiments with break sizes of less than 2.5% [1]. However, it is not sufficient to assess the predictability of the break model for cases with larger break sizes. Reactor safety analysis is conducted based on knowledge obtained from the

No.	Answers by GRS (ATHLET)	Answers by VTT (APROS)	Answers by AVN (CATHARE)	Answers by DSE-UPM (TRACE)	Answers by JAEA (RELAP5)
			(2) The Same technique is applied to the reactor case		future uncertainty evaluation considering effects of break model.
4-2	(1) The liquid level in vessel upper head is determined by the ATHLET specific mixture level model /2/. This model is based on the definition of a non-homogeneous control volume, composed by two homogeneous sub-volumes, one for liquid with bubbles below the mixture level and the other for vapour with droplets above the mixture level. Mass and energy balances are performed separately for both sub-volumes, taking into account mass and energy exchanges at the interface. The upper head is defined as a branch object. (2) Similar strategy is applied for reactor safety analyses.	(1) Adjusting flow resistances in upper head, control rod guide tubes, upper plenum and downcomer, no phase separation applied (code standard model used) (2)	(1) The Upper head is modeled using a VOLUME component rather than an AXIAL one. The volume module represents a capacity with several connections. It is adapted for capacities with large dimensions compared to the diameters of the junctions. All thermal-hydraulic quantities are assumed uniform in horizontal planes. • The velocities inside the volume are small compared to the velocities at the junctions. • Inertial forces are assumed to be negligible compared to gravity forces. Consequently, the momentum equations are simplified and the pressure field is hydrostatic. • Phase stratification occurs (void fraction stratification). The stratification is represented by a two-node model with two sub-volumes (Ω-, Ω+). The interface between the sub-volumes has a variable level. In each sub-volume, enthalpies and void fractions are assumed to be uniform but not the pressure that has a hydrostatic gradient. It is assumed that there is liquid in the lower sub-volume possibly with gas rising towards the interface. In the upper sub-volume there is mainly gas, possibly with liquid drops or falling jets. Scalar variables are calculated inside the volume in front of each junction (at a distance from the junction equal to 2% of the volume height for vertical junctions). Flow distribution	(1) Level tracking and sensitivity to nodalization	(1) The RELAP5/MOD3.2 code overpredicted break flow rate during two-phase flow discharge. So, upper head above top of CRGT was represented by ten vertical nodes, and interfacial drag in each node was reduced to 1/10. Compare the calculated results before and after these modifications. (2) The same noding model and interfacial drag are applied to reactor safety analysis.

No.	Answers by GRS (ATHLET)	Answers by VTT (APROS)	Answers by AVN (CATHARE)	Answers by DSE-UPM (TRACE)	Answers by JAEA (RELAP5)
			between sub-volumes and the phase sorting phenomena are modeled at each junction, taking into account the two-phase jet effect pull-through process. Mass and energy transfer between the two sub-volumes (bubble rise, fall of droplets, condensation, evaporation) are modeled. For the one-phase liquid case, the upper sub-volume is residual (height =1 cm). Respectively the lower sub-volume is residual for one-phase gas fluid (height = 1 mm). (2) The same model is applied to the Reactor Case		
4-3	(1) The ATHLET specific T-junction model /2/ is applied at the junction connecting the vessel upper head and the break discharge line. This model simulates multidimensional flow phenomena of a T-branching as the superposition of two one-dimensional flows: one in the axial direction of the main flow (upper head) and one in the axial direction of the branch (break discharge line). One additional momentum equation is solved at the branching point. The phase separation takes into account liquid entrainment, vapour pull-through, and it is a function of the flow regime at the main flow direction. For Test 6-1 the junction connecting the upper head to the break discharge line was defined as horizontal. (2) The T-junction model is also applied for reactor safety analyses.	(1) code standard inter-phase drag and entrainment model used (2)	(1) Scalar variables are calculated inside the volume in front of each junction (at a distance from the junction equal to 2% of the volume height for vertical junctions). Flow distribution between sub-volumes and the phase sorting phenomena are modeled at each junction, taking into account the two-phase jet effect pull-through process. A junction is the connection between two adjacent modules. A junction is either upstream or downstream of the element. This choice is of no importance in terms of flow and is a topological control for the element and the circuit. In an element (except boundary condition) at least one upstream and one downstream junction should be defined; in a circuit, each junction should appear in two, and only two, different element definitions, each with a different direction key word. For each element, a junction consists of an assembly vector node and a scalar	(1) Standard TRACE entrainment model	(1) Upper head was modeled by vertical pipe to simulate the corresponding configuration, while entrainment model for vertical pipe is not prepared in RELAP5/MOD3.2 code. The code overpredicted break flow rate during two-phase flow discharge. So, upper head above top of CRGT was represented by ten vertical nodes, and interfacial drag in each node was reduced to 1/10. (2) The same interfacial drag is applied to reactor safety analysis.

No.	Answers by GRS (ATHLET)	Answers by VTT (APROS)	Answers by AVN (CATHARE)	Answers by DSE-UPM (TRACE)	Answers by JAEA (RELAP5)
			node that is a copy of the adjacent module node. The two-phase velocities Vl & Vg are defined at the vector node and the two momentum equations are centred on this node. Due to the solution method, the momentum equations must be written by one of the adjacent modules. (2) The same model is applied to the Reactor Case		
4-4	The 8 control rod guide tubes in the test facility are represented by two parallel pipes in the ATHLET nodalization. The calculations have shown that an adequate simulation of the flow resistance between upper plenum and upper head, not only within the CRGTs but also for the downflow between hot leg nozzles and core upper plate, is essential for a correct prediction of the two-phase break flow. One main factor contributing to the good agreement between calculation and measurement was the change of the model option for the calculation of the drift-based interfacial friction within the upper plenum volumes: instead of the usual correlation for bundle geometries, which results in a higher interfacial friction and thus in a higher resistance for liquid downflow, the correlation for pipe geometries was applied.	(1) CRGTs modeled with two separate channels, one for center tubes and one for peripheral tubes (optional CCFL models for parallel tubes available, but not used in this case) (2)	(1) One AXIAL pipe element was used to simulate the all Guide Tubes of the facility. The UH/UP bypass orifice were also modeled. (2) The same model is applied to the Reactor Case	(1) Every pipe from upper plenum to upper head is simulated with a PIPE connected to a 3D VESSEL.	(1) CRGTs were modeled with one lumped flow channel. Flow paths between the upper plenum and the CRGT were modeled to simulate coolant flow at the penetration holes and at the bottom. (2) The same noding model for CRGT is applied to reactor safety analysis.
4-5	(1) Interfacial friction coefficients are calculated in ATHLET as a function of the drift velocities, which in turn are determined by the ATHLET specific flooding based drift flux model taking into account the channel geometry and the flow conditions /2/. The driftflux model has been validated	(1) Flow regime dependent interfacial friction correlations (2) Heat transfer model is chosen according flow conditions. The user can select from several critical heat flux correlations,	(1) The flow regime is calculated automatically by CATHARE in an AXIAL pipe. (2) The heat-transfer regime is calculated automatically by CATHARE between a wall and a T-H element. (3) The same model is applied to the Reactor Case		(1) Upper plenum was represented by five vertical nodes considering coolant flow at penetration holes at CRGT bottom. Core was represented by nine vertical nodes according to 9-step chopped cosine in axial profile of core power, and interfacial drag in each node was reduced to 1/10 [2]. (2) Default heat transfer model used in

No.	Answers by GRS (ATHLET)	Answers by VTT (APROS)	Answers by AVN (CATHARE)	Answers by DSE-UPM (TRACE)	Answers by JAEA (RELAP5)
	against numerous separate effect tests, including several tests in the full-scale UPTF facility. For the core region the drift correlation for bundle geometries have been applied. (2) Heat transfer correlations used in the core region are the default ones, recommended in the code User's Manual. (3) The same correlations are applied for PWR safety analyses.	however, the default model (combination of Biasi and Zuber-Griffith correlations) was used for test 6-1. (3) Default models used, unless any special reason to use optional models (sensitivity analyses).			RELAP5/MOD3.2 code is chosen. The default model consists of Chen correlation for nucleate boiling, Chen-Sundaram-Ozkaynak correlation for transition boiling, and Bromley and Sun-Gonzales-Tien correlation for film boiling [3]. (3) The same noding model for upper plenum and core and default heat transfer model are applied to reactor safety analysis.
	References /1/ G. Lerchl, H. Austregesilo ATHLET Mod 2.1 Cycle A - User's Manual GRS-P-1, Vol. 1, July 2006. /2/ H. Austregesilo, C. Bals, A. Hora, G. Lerchl, P. Romstedt ATHLET Mod 2.1 Cycle A - Models and Methods GRS-P-1, Vol. 4, July 2006.				**References** [1] Asaka, H., *et al.*, 1990. Results of 0.5% Cold-Leg Small-break LOCA Experiments at ROSA-IV/LSTF: Effect of Break Orientation. Exper. Therm. Fluid Sci. Vol. 3, pp.588-596. [2] Kumamaru, H., *et al.*, 1999. RELAP5/MOD3 Code Analyses of LSTF Experiments on Intentional Primary-Side Depressurization Following SBLOCAs with Totally Failed HPI. Nucl. Technol., Vol. 126, pp.331-339. [3] Carlson, K. E., *et al.*, 1990. RELAP5/MOD3 code manual (draft). NUREG/CR-5535, EGG-2596.

APPENDIX 3 LIST OF EXPERIMENTAL DATA

TABLE A3-1 **IS A LIST OF LSTF EXPERIMENTAL DATA FOR THE ROSA PROJECT DISTRIBUTED BY MEANS OF CD-RS AND DVDS.**

Table A3-1 List of experimental data using CD-Rs and DVDs

No.	Test No.	Media	Contents
1	Test 1-1 (ST-NC-34)	CD-R	Measured test data of all instruments
2		DVD	Image data from video probe in hot leg in loop with PZR
3			Image data from video probe in hot leg in loop without PZR
4			Image data from video probe in cold leg in loop with PZR
5			Image data from video probe in cold leg in loop without PZR
6	Test 1-2 (SB-HL-17)	CD-R	Measured test data of all instruments
7		DVD	Image data from video probe in hot leg in loop with PZR
8			Image data from video probe in hot leg in loop without PZR
9			Image data from video probe in cold leg in loop with PZR
10			Image data from video probe in cold leg in loop without PZR
11	Test 2 (ST-WH-05, 06,07,08,09,10 and 11)	DVD	Test data of ST-WH-05
12			Test data of ST-WH-06
13			Test data of ST-WH-07
14			Test data of ST-WH-08 and 09
15			Test data of ST-WH-10 and 11
16	Test 3-1 (SB-CL-38)	CD-R	Measured test data of all instruments
17		DVD	Image data from video probe in hot leg in loop with PZR
18			Image data from video probe in hot leg in loop without PZR
19			Image data from video probe in cold leg in loop without PZR
20	Test 3-2 (TR-LF-13)	CD-R	Measured test data of all instruments
21		DVD	Image data from video probe in hot leg in loop with PZR
22			Image data from video probe in hot leg in loop without PZR
23			Image data from video probe in cold leg in loop with PZR
24			Image data from video probe in cold leg in loop without PZR

No.	Test No.	Media	Contents
25	Test 4-1 (ST-NC-39)	DVD	Measured test data of all instruments
26	Test 4-2 (ST-NC-40)	DVD	Measured test data of all instruments
27	Test 5-1 (SB-CL-39)	CD-R	Measured test data of all instruments
28		DVD	Image data from video probe in hot leg in loop with PZR
29			Image data from video probe in hot leg in loop without PZR
30			Image data from video probe in cold leg in loop with PZR
31			Image data from video probe in cold leg in loop without PZR
32	Test 5-2 (SB-CL-40)	CD-R	Measured test data of all instruments
33		DVD	Image data from video probe in hot leg in loop with PZR
34			Image data from video probe in hot leg in loop without PZR
35			Image data from video probe in cold leg in loop with PZR
36			Image data from video probe in cold leg in loop without PZR
37	Test 6-1 (SB-PV-09)	CD-R	Measured test data of all instruments
38		DVD	Image data from video probe in hot leg in loop with PZR
39			Image data from video probe in hot leg in loop without PZR
40	Test 6-2 (SB-PV-10)	CD-R	Measured test data of all instruments
41		DVD	Image data from video probe in hot leg in loop with PZR
42			Image data from video probe in hot leg in loop without PZR
43	Test 6-3 (LB-CL-05 and 06)	CD-R	Measured test data of all instruments in LB-CL-05
44			Measured test data of all instruments in LB-CL-06
45		DVD	Image data of high-speed video camera in LB-CL-05
46			Image data of high-speed video camera in LB-CL-06

Each data CD-R contains followings files other than LSTF data file(s).

(1) A folder BIN that contains lstfread.exe file

This file read data in CD-R, extract data and create data table in MS Excel format.

(2) CDread.bat file

This file activates lstfread. exe in BIN folder and specify CD-R drive name.

For example, current file specifies f drive. Please edit this bat file to change this CD-R drive to be specified by CDread.bat file.

Please note that the default drive for lstfread.exe is d. You do not need to use CDread.bat file when the CD-R drive of your computer is d.

(3) TAGLIST.xls file

This is a table of all the LSTF data. Please consult this table to seek necessary Function ID and LSTF System Description Report that indicates the location of sensors.

(4) Readme(HowToUse).doc file

The data reader program “lstfread. exe” converts the data file(s) and creates a csv file of LSTF data by the following procedures.

(1) Prepare a folder named ‘output’.

(2) Copy the folder BIN and CDread.bat file into a work directory the same as the folder ‘output’.

(3) Insert LSTF Data CD-R and confirm RUN ID and Function IDs you want to obtain data. ‘Function IDs’ are listed in ‘TAGLIST.xls’ file that is included in each of LSTF data CD-R. The contents of ‘TAGLIST.xls’ file may change according to the change in the measurement instrumentation

(4) Double click CDread.bat to activate it.

(5) Input several parameters requested as follows;

(Example)

RUN ID : SP9 <== FIX

Number of Data (MAX 10) : 4 <== Arbitrary

Function ID :

TE514

PE13

LE3

RC2

Start Time : 0.0 <== Arbitrary

End Time : 100.0 <== Arbitrary

This Example creates a data table file of Test 6-1 (abbreviated as SP9) in csv format that contains four data (TE514, PE13, LE3, RC2) for the same measurement time span.

(6) Program starts the preparation of a csv file into the folder ‘output’.

APPENDIX 4 LIST OF PUBLISHING

Tables A4-1 and A4-2 are lists of publishing papers and reports respectively based on ROSA Project.

Table A4-1 List of publishing papers

No.	Paper	Distribution Date F: file
1	A. Jasiulevicius et al. of PSI, "**SIMULATION OF OECD/NEA TEST 6.2 USING TRACE,**", ICONE15	F: March 1, 2007
2	M. Suzuki et al., "**Effects of three-dimensional steam flow on exit temperatures during core boil-off in PWR vessel top break LOCA simulation experiments by using ROSA/LSTF**", 2007 Fall Meeting of Atomic Energy Society of Japan	F: June 14, 2007
3	T. Watanabe,"**Simulation of Temperature Stratification during ECCS Water Injection using FLUENT**", NURETH-12 T. Takeda et al.,"**RELAP5 Analysis of ROSA/LSTF Vessel Upper Head Break LOCA Experiment**", NURETH-12	F: June 14, 2007
4	H. Nakamura et al., "**OECD/NEA ROSA Project (1) Current Status of The ROSA Project**", M. Kondo et al., "**OECD/NEA ROSA Project (2) Condensation-induced water hammer test employing system pressure as parameter**", H. Asaka et al., "**OECD/NEA ROSA Project (3) High-power Natural Circulation Experiment and RELAP5 Post-test Analysis**", Y. Maruyama et al., "**OECD/NEA ROSA Project (4) Pre-Test Analysis for Superheated Steam Natural Circulation Experiments and Verification of Pulsed-Wire Velocity Meter**", T. Takeda et al., "**OECD/NEA ROSA Project (5) Experiment on PWR PV Bottom Break LOCA and Post-test Analysis**", All for 2008 Annual Meeting of Atomic Energy Society of Japan	F: Feb. 8, 2008
5	M. Suzuki et al., "**Performance of Core Exit Thermocouple for PWR Accident Management Action in Vessel Top Break LOCA Simulation Experiment at OECD/NEA ROSA Project**", ICONE16 H. Nakamura et al., "**RELAP5/MOD3 Code Verification through PWR Pressure Vessel Small Break LOCA Tests in OECD/NEA ROSA Project,**", ICONE16	F: Feb. 8, 2008
6	S. Gallardo et al. of UPV, "**Post test calculation of OECD/ROSA Test 6.2,**", PHYSOR2008	F: Apr. 15, 2008
7	S. Gallardo et al. of UPV, "**OECD/ROSA Tests 6.1 and 6.2 with TRACE5,**", CAMP meeting	F: Apr. 28, 2008
8	T. Takeda et al., "**RELAP5 analysis of OECD/NEA ROSA Project experiment simulating a PWR small break LOCA with high-power natural circulation,**", PHYSOR2008	F: May 8, 2008

No.	Paper	Distribution Date F: file
9	T. Watanabe et al., "**OECD ROSA Project temperature stratification experiment and CFD analysis**", T. Takeda et al., "**ROSA Project Experiment on Loss-of-feedwater without Scram and RELAP5 Post-test Analysis**", Both for 2008 Fall Meeting of Atomic Energy Society of Japan	F: June 24, 2008
10	V. M. Martínez et al. of Technical University of Catalonia, "**Code Validation and Scaling of the ROSA/SLTF Test 3-1 Experiment,**", TOPSAFE2008 meeting	F: July 4, 2008
11	H. Nakamura et al., "**Overview of Recent Efforts in ROSA/LSTF Experiments**", NUTHOS-7	F: Sept. 8, 2008
12	T. Takeda et al., "**RELAP5 POST-TEST ANALYSES OF OECD/NEA ROSA PROJECT EXPERIMENTS ON STEAM GENERATOR DEPRESSURIZATION WITH OR WITHOUT NON-CONDENSABLE GAS INFLOW**", NTHAS-6	F: Sept. 8, 2008
13	T. Takeda et al., "**Plan of ROSA Project Experiment on Steam Condensation during Large-break LOCA**", 2009 Annual Meeting of Atomic Energy Society of Japan	F: Jan. 23, 2009
14	T. Farkas and I. Tóth of KFKI, "**FLUENT ANALYSIS OF A ROSA TWO-PHASE STRATIFICATION TEST,**", ICONE17	F: Feb. 12, 2009
15	T. Watanabe and H. Nakamura, "**CFD analysis of temperature stratification experiment in OECD/NEA ROSA project,**" NURETH-13	--
16	S. Gallardo, V. Abella and G. Verdú of UPV, "**FLUENT OECD/NEA ROSA Project Test 3-2. Simulation with TRACE5,**", CAMP Meeting	F: June 8, 2009
17	Y. Maruyama et al., "**Experiment on Superheated Steam Natural Circulation in ROSA Project -Characterization of Natural Circulation-**, T. Takeda et al., "**LSTF ROSA Project Experiment on Steam Condensation on ECCS Water during PWR Large-break LOCA**", Both for 2009 Fall Meeting of Atomic Energy Society of Japan	F: July 22, 2009
18	J. Freixa and A. Manera, "**Analysis of an RPV Upper Head SBLOCA at the ROSA Facility using TRACE,**" Nucl. Engng. and Des. 240(7) (2010) 1779-1788	--

Table A4-2 List of publishing reports

No.	Report	Distribution Date F: file, P: printed
1	The ROSA-V Group, "**ROSA-V Large Scale Test Facility (LSTF) System Description for The Third and Fourth Simulated Fuel Assemblies**," JAERI-Tech 2003-037 (March 2003)	P: before PRG1
2	Thermohydraulic Safety Research Group, "**A Report on RELAP5 Code Pre-test Analyses to Define LSTF PV-break SBLOCA Experiment Conditions**" (Sept. 12, 2005)	F: Sept. 12, 2005
3	Thermohydraulic Safety Research Group, "**Brief Description of Computer Code Inputs for ROSA/LSTF Analyses using RELAP5/MOD3 and TRAC-PF1/MOD1 Codes**" (June 17, 2005)	F: Oct. 14, 2005
4	Thermohydraulic Safety Research Group, "**A Report on RELAP5 Code Pre-test Analyses to Define ROSA/LSTF Experimental Conditions of PV Bottom LOCA Experiment**" (November 25, 2005)	F: Nov. 25, 2005
5	Thermohydraulic Safety Research Group, "**Quick-look Data Report of ROSA/LSTF Test 6-1 (1.9% Pressure Vessel Upper-head Small Break LOCA Experiment SB-PV-09 in JAEA),**" (February 16, 2006)	F: Feb. 17, 2006 P: Oct. 26, 2006
6	Thermohydraulic Safety Research Group, "**Quick-look Data Report of ROSA/LSTF Test 6-2 (0.1% Pressure Vessel Bottom Small Break LOCA Experiment SB-PV-10 in JAEA),**" (March 15, 2006)	F: March 16, 2006 P: Oct. 26, 2006
7	Thermohydraulic Safety Research Group, "**Heat Loss Characteristics of the ROSA/LSTF System**,“ (Aug. 7, 2006)	F: Aug. 16, 2006
8	Thermohydraulic Safety Research Group, "**Proposed Test Procedures of Temperature Stratification Experiment**," (Aug. 7, 2006)	F: Aug. 16, 2006
9	Thermohydraulic Safety Research Group, "**A Report on Pre-test Analyses and Preparatory Experiments to Define Experimental Conditions of ROSA/LSTF Test 3-1 on High Power Natural Circulation**," (Sept. 22, 2006)	F: Sept. 22, 2006
10	Thermohydraulic Safety Research Group, "**Final Data Report of ROSA/LSTF Test 6-1 (1.9% Pressure Vessel Upper-head Small Break LOCA Experiment SB-PV-09 in JAEA),**" (Dec. 1, 2006)	F: Dec. 5, 2006 P: ---
11	Thermohydraulic Safety Research Group, "**General questionnaire on the analysis methods for Test 6-1 -- tentative for discussions --**," (Jan. 10, 2007)	F: Jan. 10, 2007
12	Thermohydraulic Safety Research Group, "**Proposed Test Conditions and Procedures of Condensation-Induced Water Hammer Experiment**," (Jan. 10, 2007)	F: Jan. 10, 2007
13	Thermohydraulic Safety Research Group, "**Final Data Report of ROSA/LSTF Test 6-2 (0.1% Pressure Vessel Bottom Small Break LOCA Experiment SB-PV-10 in JAEA),**" (Jan. 17, 2007)	F: Feb.13, 2007 P: May 30, 2008
14	Thermohydraulic Safety Research Group, "**Quick-Look Data Report of OECD/NEA ROSA Project Test 1-1 (ECCS water injection under natural circulation condition: ST-NC-34 in JAEA),**" (Jan. 30, 2007)	F: Jan. 31. 2007 P: ---
15	Thermohydraulic Safety Research Group, "**Quick-look Data Report of ROSA/LSTF Test 3-1 (High Power Natural Circulation Experiment SB-CL-38 in JAEA),**" (March 20, 2007)	F: March 20, 2007 P: ---
16	Thermohydraulic Safety Research Group, "**Proposed Test Procedures of Temperature Stratification Experiment**," (March 23, 2007)	F: March 23, 2007

No.	Report	Distribution Date F: file, P: printed
17	Thermohydraulic Safety Research Group, "**Performance of Core Exit Temperatures for Accident Management Action in LSTF 1.9% Top Break LOCA Test** (Supplemental Report for Test 6-1)," (May 25, 2007)	F: May 25, 2007 P: May 30, 2008
18	Thermohydraulic Safety Research Group, "**A Report on Pre-test Analyses to Define Experimental Conditions of ROSA/LSTF Test 5-1 on Primary Cooling through Steam Generator Secondary-side Depressurization**," (May 24, 2007)	F: May 25, 2007
19	Thermohydraulic Safety Research Group, "**Quick-Look Data Report of OECD/NEA ROSA Project Test 1-2 (1% hot-leg break LOCA experiment with HPI: SB-HL-17 in JAEA)**" (Sept. 3, 2007)	F: Sept. 4, 2007
20	Thermohydraulic Safety Research Group, "**A Report on Pre-test Analyses to Define Experimental Conditions of ROSA/LSTF Test 5-2 on Primary Cooling through Steam Generator Secondary-side Depressurization**" (Sept. 5, 2007)	F: Sept. 5, 2007
21	Thermohydraulic Safety Research Group, "**Performance of Core Exit Temperatures for Accident Management Action in LSTF 1.9% Top Break LOCA Test (Supplemental Report for Test 6-1)**" (Sept. 19, 2007)	F: Sept. 19, 2007
22	Thermohydraulic Safety Research Group, "**A Report on Pre-test Analyses to Define Experimental Conditions of ROSA/LSTF Test 3-2 on High Power Natural Circulation**" (Oct. 9, 2007)	F: Oct. 9, 2007
23	Thermohydraulic Safety Research Group, "**Quick-look Data Report of ROSA/LSTF Test 5-1 (Primary Cooling through Steam Generator Secondary-side Depressurization SB-CL-39 in JAEA),**" (Nov. 30, 2007)	F: Nov. 30, 2007
24	Thermohydraulic Safety Research Group, "**Final Data Report of ROSA/LSTF Test 3-1 (High Power Natural Circulation Experiment SB-CL-38 in JAEA),**" (Jan. 9, 2008)	F: Jan. 15, 2008
25	Thermohydraulic Safety Research Group, "**Final Data Report of OECD/NEA ROSA Project Test 1-1 (ECCS water injection under natural circulation condition : ST-NC-34 in JAEA),**" (Jan., 2008)	F: Jan. 20, 2008
26	Thermohydraulic Safety Research Group, "**Results of Preliminary Test on Superheated Steam Natural Circulation in Framework of OECD/NEA ROSA Project,**" (Feb. 7, 2008)	F: Feb. 7, 2008
27	Thermohydraulic Safety Research Group, "**Quick-look Data Report of ROSA/LSTF Test 3-2 (High Power Natural Circulation Experiment TR-LF-13 in JAEA),**" (Apr. 11, 2008)	F: Apr. 11, 2008
28	Thermohydraulic Safety Research Group, "**Quick-look Data Report of ROSA/LSTF Test 5-2 (Primary Cooling through Steam Generator Secondary-side Depressurization Experiment SB-CL-40 in JAEA),**" (Apr. 11, 2008)	F: Apr. 11, 2008
29	Thermohydraulic Safety Research Group, "**Quick-look Data Report of OECD/NEA ROSA Project Test 2 (Condensation-induced Water Hammer Tests : ST-WH-05,06,07,08,09,10 and 11 in JAEA),**" (Mar. 14, 2008)	F: Apr. 14, 2008
30	Thermohydraulic Safety Research Group, "**Final Data Report of OECD/NEA ROSA Project Test 1-2 (1% hot-leg break LOCA experiment with HPI: SB-HL-17 in JAEA),**" (July 3, 2008)	F: July 3, 2008
31	Thermohydraulic Safety Research Group, "**Final Data Report of ROSA/LSTF Test 5-1 (Primary Cooling through Steam Generator Secondary-side Depressurization Experiment SB-CL-39 in JAEA)**," (July 3, 2008)	F: July 3, 2008
32	Thermohydraulic Safety Research Group, "**Final Data Report of OECD/NEA ROSA Project Test 3-2 (High Power Natural Circulation Experiment TR-LF-13 in JAEA),**" (Jan. 21, 2009)	F: Jan. 22, 2009

No.	Report	Distribution Date F: file, P: printed
33	Thermohydraulic Safety Research Group, "**Final Data Report of OECD/NEA ROSA Project Test 5-2 (Primary Cooling through Steam Generator Secondary-side Depressurization Experiment SB-CL-40 in JAEA),**" (Jan. 21, 2009)	F: Jan. 22, 2009
34	Thermohydraulic Safety Research Group, "**Quick-Look Data Report of Test 4-1 (Designated JAEA Internal Run ID of ST-NC-39) Natural Circulation of Superheated Steam in Primary Loops during High Pressure Sequences of LWR Severe Accidents with Argon Gas as Simulated Gas,**" (May 28, 2009)	F: June 2, 2009
35	Thermohydraulic Safety Research Group, "**Quick-Look Data Report of Test 4-2 (Designated JAEA Internal Run ID of ST-NC-40) Natural Circulation of Superheated Steam in Primary Loops during High Pressure Sequences of LWR Severe Accidents,**" (June 30, 2009)	F: July 6, 2009
36	Thermohydraulic Safety Research Group, "**Quick-Look Data Report of Test 6-3 (Steam Condensation Experiment during Large-break LOCA of PWR LB-CL-05 and LB-CL-06 in JAEA),**" (Sept. 1, 2009)	F: Sept. 2, 2009
37	Thermohydraulic Safety Research Group, "**Final Data Report of OECD/NEA ROSA Project Test 2 (Condensation-induced Water Hammer Tests : ST-WH-05,06,07,08,09,10 and 11 in JAEA),**" (Aug. 11, 2009)	F: Oct. 10, 2009
38	Thermohydraulic Safety Research Group, "**Final Data Report of Test 4-1 (Designated JAEA Internal Run ID of ST-NC-39) Natural Circulation of Superheated Steam in Primary Loops during High Pressure Sequences of LWR Severe Accidents with Argon Gas as Simulated Gas,**" (May 6, 2010)	F: May 19, 2010
39	Thermohydraulic Safety Research Group, "**Final Data Report of Test 4-2 (Designated JAEA Internal Run ID of ST-NC-40) Natural Circulation of Superheated Steam in Primary Loops during High Pressure Sequences of LWR Severe Accidents,**" (May 6, 2010)	F: May 19, 2010
40	Thermohydraulic Safety Research Group, "**Final Data Report of Test 6-3 (Steam Condensation during Large-break LOCA of PWR LB-CL-05 and LB-CL-06 in JAEA),**" (June 30, 2010)	F: July 1, 2010