OVERVIEW OF RECENT EFFORTS THROUGH ROSA/LSTF EXPERIMENTS

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JAEA started the LSTF experiments in 1985 for the fourth stage of the ROSA Program (ROSA-IV) for the LWR thermal-hydraulic safety research to identify and investigate the thermal-hydraulic phenomena and to confirm the effectiveness of ECCS during small-break LOCAs and operational transients. The LSTF experiments are underway for the ROSA-V Program and the OECD/NEA ROSA Project that intends to resolve issues in thermal-hydraulic analyses relevant to LWR safety. Six types of the LSTF experiments have been done for both the system integral and separate-effect experiments among international members from 14 countries. Results of four experiments for the ROSA Project are briefly presented with analysis by a best-estimate (BE) code and a computational fluid dynamics (CFD) code to illustrate the capability of the LSTF and codes to simulate the thermal-hydraulic phenomena that may appear during SBLOCAs and transients. The thermal-hydraulic phenomena dealt with are coolant mixing and temperature stratification, water hammer up to high system pressure, natural circulation under high core power condition, and non-condensable gas effect during asymmetric SG depressurization as an AM action.

KEYWORDS : LSTF, ROSA Program, Integral-effect and Separate-effect Tests, Thermal-hydraulics, OECD/NEA ROSA Project

1. INTRODUCTION

The ROSA (Rig-of-Safety Assessment) Program was started in 1970 at Japan Atomic Energy Research Institute (JAERI, the predecessor of Japan Atomic Energy Agency (JAEA)) to study thermal-hydraulic response of light water reactors (LWRs) during loss-of-coolant accidents (LOCAs) and operational transients. It was first dedicated to separate-effect tests mainly for fundamental blowdown phenomena and core heat transfer in the ROSA-I Program [1]. The fundamental experimental techniques to handle steam/water two-phase flows at reactor typical pressure were developed through experiments including mechanical reactions of two-phase discharge flows onto the facility. The ROSA Program then proceeded to confirm the effectiveness of an emergency core cooling system (ECCS) by system integral tests for pressurized water reactors (PWR) in the ROSA-II Program [2] and for boiling water reactors (BWR) in the ROSA-III Program [3]. These early phases of the ROSA Program have utilized rather small test facilities with a half-height core and pressure vessel, yet were designed to study major phenomena and responses under typical LWR conditions with nominal operating pressure and temperature during large-break LOCA as a design-basis accident (DBA).

The Three Mile Island Unit-2 (TMI-2) reactor accident in 1979 led to a reorientation of LWR safety research to consider small-break LOCAs (SBLOCAs) and operational transients. Emphasis was put onto such phenomena as natural circulation through the primary loops involving two-phase stratified flows and counter-current flows. Since such phenomena depend greatly on the component height and facility scaling, large-scale tests in the primary system geometry representative of operational PWRs were required. Development of best-estimate (BE) analysis computer codes was accelerated worldwide to prepare tools to evaluate the complicated thermal-hydraulic phenomena with accuracy by eliminating conservative assumptions put in evaluation model (EM) codes to accurately evaluate safety margins in safety analyses.

Based on this background, JAERI initiated the ROSA-IV Program in 1980 [4] that includes large-scale integral simulation of a Westinghouse (W)-type PWR plant using the Large Scale Test Facility (LSTF) [5] (Fig. 1). The LSTF experiments were designed to provide data for the following three fundamental areas:

(a) Plant behavior definition - to simulate and define plant behavior and thermal--hydraulic phenomena during SBLOCAs and operational transients.
(b) Plant recovery methods investigation - to investigate

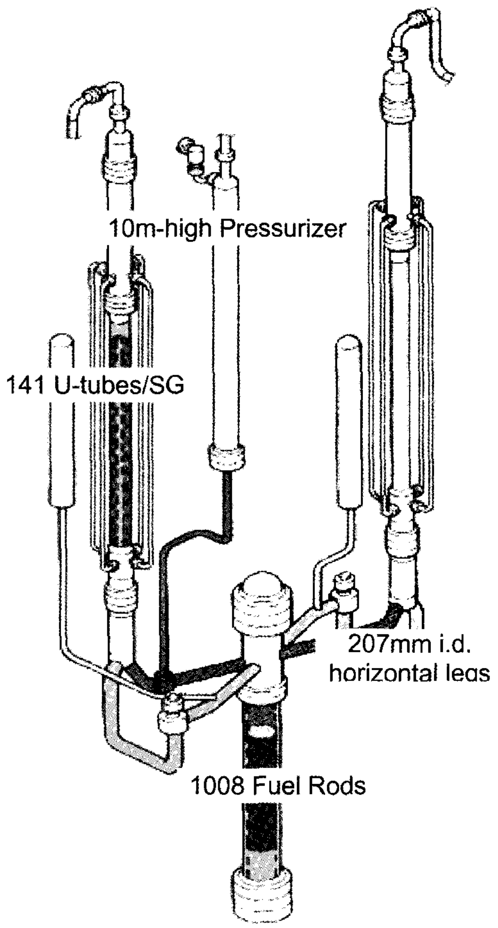

Fig. 1. Schematic of ROSA/LSTF

operator procedures and innovative recovery equipment for possible use during SBLOCAs and operational transients.

(c) Creation of a code assessment database -- to obtain a database for assessment and verification of advanced thermal-hydraulic codes.

There are many secondary objectives within these three areas. The LSTF was thus designed to simulate the thermal-hydraulic phenomena peculiar to SBLOCAs and operational transients by realizing prototypical component elevation differences, large loop piping diameters, prototypical pressure levels, simulated system controls, and core power levels sufficient to simulate the decay power through rather long-term transients during SBLOCAs. The LSTF has been used to investigate the PWR plant behavior including the effects of plant recovery methods for a wide spectrum of accident and transient conditions since May 1985, including simulation of equipment specific to other vendor designs.

Eighty system integral experiments were performed so far until March 1993 in the ROSA-IV Program. These include TMI-type experiments [4], SBLOCA experiments with different break conditions [6-8] and plant recovery actions [8], an experiment for the 26^{th} international standard problem (ISP-26) of OECD/NEA/CSNI [9], counter-part tests [10] with the BETHSY of CEA in France, the PKL of GRS in Germany and the Semiscale of INEL in USA, plant accident simulation for the Mihama Unit-2 steam generator tube rupture (SGTR) accident [11,12], and natural circulation tests under various system pressures with and without non-condensable gas [13,14].

Based on knowledge and experience gained by the ROSA-IV Program, the ROSA-V Program was inaugurated in April 1991. The LSTF experiments were continued with the following main objectives:

(i) To study the effectiveness of accident management (AM) measures in the case of Beyond Design Basis Accidents (BDBAs).
(ii) To investigate passive safety features for next generation reactors.
(iii) To develop and improve computer codes.

These objectives are based on the increased need to prevent severe accidents after the Chernobyl accident in 1986. Objective (i) includes as a secondary objective the confirmation of capability and reliability of accident detection measures to prevent severe core damage in current and next generation PWRs and to develop improved detection measures that can anticipate inadequate core cooling conditions. The improved accident detection measures include a primary coolant mass indication system that may provide a measure to clearly detect the core heat-up phenomena. The obtained database is utilized for the assessment and improvement of the thermal-hydraulic computer codes that are applied to both the DBAs and BDBAs.

Following the main objectives, more than 100 experiments have been performed so far in the ROSA-V Program. These include experiments to confirm the effectiveness of passive safety features for the AP600 reactor design in collaboration with USNRC [15] and the horizontal heat exchanger of a passive containment cooling system (PCCS) for a next-generation BWR [16], various SBLOCA experiments to confirm the effectiveness of AM measures such as steam generator (SG) secondary depressurization cooling with and without non-condensable gas ingress from accumulators [17], and a loss-of-RHR (residual heat removal) system during plant outage including mid-loop operation [18].

The LSTF had been intensively utilized to identify, observe, and clarify thermal-hydraulic phenomena that may occur during SBLOCAs and operational transients to confirm the effectiveness of ECCS. Various safety-relevant subjects and issues related to the prevention and evaluation of severe accidents, the passive safety features for both PWR and BWR, events during a plant outage, and simulation of real plant accident including SGTR have been investigated further.

The OECD/NEA ROSA Project was started in April 2005, within the framework of the ROSA-V Program in JAEA to foster effective international cooperative research for LWR thermal-hydraulics safety by fully utilizing the LSTF at its full capacities. Major results from the selected LSTF experiments for the OECD/NEA ROSA Project in the first 3 years are introduced with the major findings to illustrate the typical capabilities of LSTF experiments.

2. DESIGN PHILOSOPHY OF LSTF

The LSTF [5] is a full-pressure and full-height model of a typical Westinghouse (W)-type 4-loop PWR with 3423 MWt, namely the Tsuruga Unit-2 of Japan Atomic Power Company (JAPC) designed to accurately simulate the driving force of natural circulation. 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 large-diameter (207 mm) horizontal legs. The volumetric scaling ratio of the primary loops is 1/48 of the reference PWR, which is about 1/21 for a two-loop PWR such as the Mihama Unit-2. The flow area in the horizontal leg is scaled to conserve the ratio of the length L to the square-root of pipe diameter D; $L/\sqrt{D}$ of the reference PWR to better simulate the flow regime transitions in the primary loops. The time scale of simulated thermal-hydraulic phenomena is thus one to one to those in the reference PWR.

The Fuel Assembly has mostly the same dimensions as those of the PWR 17×17 fuel assembly. This preserves the heat transfer characteristics of the core. The total number of simulated fuel rods is scaled by 1/48; 1008 electrically-heated rods in the current fourth fuel assembly. Core power is equal to or less than 14% of the scaled rated power of the reference PWR. The maximum core power is 10 MWt.

The volume of each SG is twice as much as the scaled one. U-tube bundles have mostly the same dimensions with those of the reference PWR. The SG is designed and operated to accurately simulate primary-to-secondary thermal interactions. 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 initial secondary pressure is then elevated to suppress and control the heat transfer.

A break unit with an orifice or a nozzle to control break size can be connected at 19 locations of the LSTF. All types of ECCS furnished to the reference PWR are provided as well as a gravity-driven injection system (GDIS) as passive safety features.

More than 1600 instrumentations are installed to measure thermal-hydraulic phenomena as precisely as possible. Three-dimensional fluid behaviors are detected by a combination of thermocouples, water level measurements, and gamma-ray densitometers at hot and cold legs with the aid of visual observation by using a video probe: a periscope that withstands high-pressure and high-temperature steam/water two-phase flow.

3. OECD/NEA ROSA PROJECT

3.1 Background and Objectives

Great efforts have been taken worldwide, especially after the TMI-2 accident, to develop, verify, and utilize BE codes for safety evaluation of LWRs through the simulation of plant behavior during DBEs and BDBE transients. Such events involve complex multi-dimensional single-phase and two-phase flow conditions, which may include non-condensable gas in many cases. Although the developed BE codes have very high predictive capabilities especially for one-dimensional phenomena, there remains need for experimental work and code development/validation for such complex flows as multi-dimensional mixing, stratification, parallel flows, and oscillatory flows with or without the influence of non-condensable gas. The increased use of BE analysis methods in licensing, which is replacing traditional conservative approaches, further requires the validation and quantification of uncertainties in the simulation models and methods. Many experimental facilities have contributed to prepare the thermal-hydraulic databases available today. However, most of them are insufficient for future codes that are to incorporate multi-dimensional simulation capabilities more rigorously, mainly because the spatial resolution of the measurements is not sufficient to assess the simulation models and methods.

The OECD/NEA ROSA Project intends to resolve such issues in thermal-hydraulics analyses relevant to LWR safety by using the LSTF with a focus on the validation of simulation models and methods for complex phenomena that may occur during DBEs and BDBE transients. Key objectives of the ROSA Project are as follows;

(1) To provide integral and separate-effect experimental databases to validate code predictive capability and the accuracy of models. Especially, phenomena coupled with multi-dimensional mixing, stratification, parallel flows, oscillatory flows and non-condensable gas flows are to be studied.

(2) To clarify the predictability of codes currently used for thermal-hydraulic safety analyses as well as of advanced codes presently under development, thus creating a group among the OECD member countries who share the need to maintain or improve technical competence in thermal-hydraulics for nuclear reactor safety (NRS) evaluations.

More than 17 organizations from 14 OECD/NEA member countries have joined the ROSA Project.

3.2 LSTF Experiments for ROSA Project

The experimental program is defined to conduct twelve experiments of the following six types to achieve the objectives above;

1) Temperature stratification and coolant mixing during ECCS coolant injection
2) Unstable and destructive phenomena such as water hammer
3) Natural circulation under high core power conditions
4) Natural circulation with superheated steam
5) Primary cooling through SG secondary depressurization
6) Request-based experiments such as small break (SB) LOCA with a break at the top or bottom of the reactor pressure vessel (PV) both being coupled with accident management (AM) measures with symptom-oriented operator actions, and steam condensation on ECCS water during large break (LB) LOCA

4. MAJOR RESULTS OF OECD/NEA ROSA PROJECT

Selected results of four types of ROSA Project tests are briefly introduced in this section to illustrate the capability of LSTF experiments that simulate major thermal-hydraulic phenomena.

4.1 Coolant Mixing and Temperature Stratification (Test 1)

Two experiments; a steady natural circulation as a separate-effect test and a hot leg SBLOCA system integral test, both with ECCS coolant injection into cold legs, were performed to observe temperature stratification and coolant mixing in the cold legs and downcomer. The separate-effect test results are briefly introduced and discussed here.

4.1.1 Background and Objectives

During a SBLOCA of PWR, cold ECCS water injected into the cold legs mixes with hot primary coolant and flows into the vessel downcomer. A portion of the injected water may flow at the cold leg bottom forming a cold layer less mixed with the upper hot layer, thus causing temperature stratification. When a liquid level forms in the cold leg, steam condensation may take place on injected water, causing local pressure variations and unstable flow oscillations, which in turn promote coolant mixing. Such multi-dimensional and non-equilibrium flow phenomena are of concern for pressurized thermal shock (PTS) in view of aging and plant life extension [19,20].

The LSTF experiment simulated a quasi-steady natural circulation to observe coolant mixing and temperature stratification in the cold leg and downcomer. The influences of liquid level formation in the cold leg were studied by step-wise decreasing coolant inventory. Three-dimensional temperature distributions were measured in the cold legs and downcomer by newly installing thermocouples (T/Cs) at high spatial density. Further three-dimensional flow analyses were performed using FLUENT code [21] to evaluate the predictive capability of such computational fluid dynamics (CFD) code using experimental data as boundary conditions.

4.1.2 Test Facility and Test Condition

Two or three temperature measurement stations were added into the two cold legs that have different ECCS nozzle geometries. Each measurement station is composed of three vertical stems on the same cross-section with 7 T/Cs on each stem. In the downcomer, three radially-arranged T/Cs were put at two different depths from the bottom of the cold leg nozzle at three chordal locations. For each cold leg, a video probe was installed in each of two cold legs to obtain visual images especially for the mixing below and downstream of the ECCS nozzle.

The experiment was started by single-phase liquid natural circulation (100% coolant inventory) with the SG secondary-side pressure of 6.5 MPa. Room-temperature (RT) coolant is then injected into the cold legs at a constant injection rate for a certain time duration. Two different ECCS injection flow rates were tested at four different coolant inventories: 100%, 80%, 70% and 50%. Coolant mixing was observed in two cold legs. Core power was kept constant at 2% of the nominal scaled power.

4.1.3 Test and Analysis Result

The temperature stratification occurred in the cold leg during the ECC coolant injection, extending beyond the cold leg exit under both the single-phase and two-phase natural circulation conditions. The degree of temperature stratification was somehow larger in cold-leg A where the ECCS nozzle is located nearer to the PV than in cold-leg B.

The cold layer of coolant was found to flow down along the PV inner wall under the single-phase liquid NC condition. This condition continued even under the two-phase NC condition if the downcomer liquid level was as high as or higher than the bottom of cold-leg. The cold water, however, flowed down along the core barrel outer wall when the downcomer liquid level became lower than the bottom of cold-leg.

Steam condensation took place on the injected ECC coolant and influenced the downcomer liquid level increase rate. The fluctuation of liquid level was observed by video probe in the cold legs as well as the coolant stream falling down from ECCS nozzle. The density fluctuation was seen in single-phase liquid flow and in the liquid flow below the liquid level when the window of the video probe was submerged in the liquid stream.

The FLUENT6.2 code [21] was used for three-dimensional CFD analyses. The Navier-Stokes equations, together with turbulent flow models, are solved by the finite volume method. A entire cold leg, from the pump

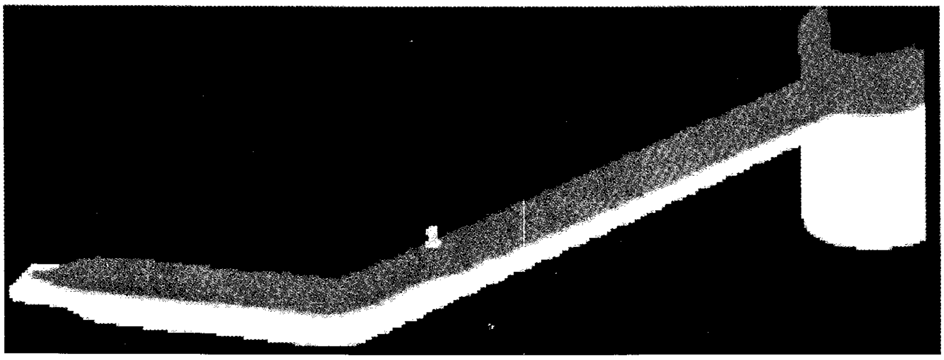

Fig. 2. Example of Density Distribution in Cold Leg and Downcomer by FLUENT Code

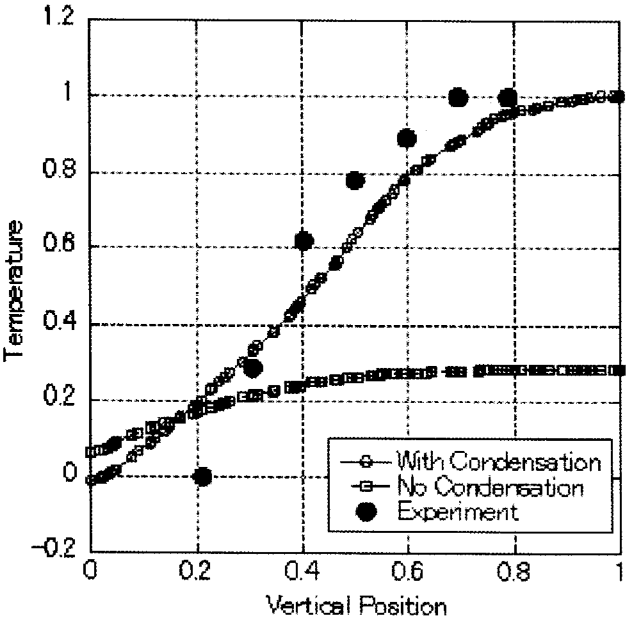

Fig. 3. Comparison of Vertical Temperature Distribution in Cold Leg (Liquid Level Case)

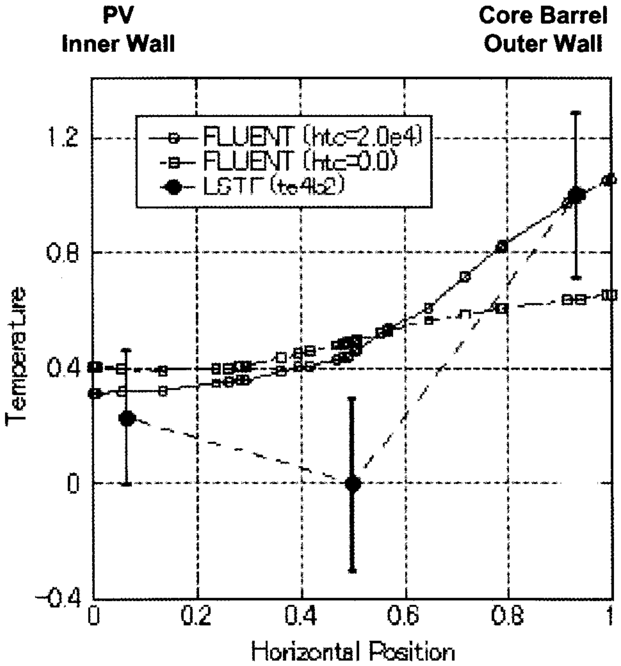

Fig. 4. Comparison of Horizontal Temperature Distribution in Downcomer

exit diffuser to a portion of vessel downcomer, was modeled with about 1.7 million volume elements as shown in Fig. 2. As for the numerical schemes, a second order implicit method in time and third order Monotone Upstream-centered Schemes for Conservation Laws (MUSCL) scheme in space and Pressure-Implicit with Splitting of Operators (PISO) method to calculate pressure and velocity were employed with realizable k-ε turbulent model as this combination would provide most accurate results. The coolant injection into a two-phase stratified flow was simulated using the Volume of Fluid (VOF) model. Since a phase change model is not provided by the code, a condensation model was prepared and applied, considering fluid temperatures, interfacial area, and the heat transfer rate in each unit cell.

The comparison with the experimental results revealed that the temperature stratification as a vertical temperature distribution near the ECCS nozzle was predicted well, though the mixing in the cold-leg was overestimated in both the single- and two-phase cases. For the two-phase flow cases with liquid level in the cold leg, the temperature distribution was well predicted when the effect of steam condensation was taken into account in the analysis as shown in Fig. 3. The trend of temperature distribution in the downcomer was predicted well as shown in Fig. 4.

4.1.4 Summary

Temperature stratification in the cold legs and vessel downcomer during ECCS coolant injection were studied using LSTF separate-effect natural circulation experiments under full-pressure and temperature conditions and by CFD code analyses. The detailed temperature distribution measurement revealed that the temperature distribution in the downcomer may depend on the liquid level in the cold leg. Visual observation of flow in the cold legs by using new video probes gave some complementary information. The three-dimensional CFD analysis using the FLUENT6.2 code indicated that the code can predict the trend of temperature stratification in a cold leg and vessel downcomer for both the single-phase liquid and two-phase flow conditions once steam condensation is taken into account, though the coolant mixing is overestimated.

4.2 Water Hammer at High Pressures (Test 2)

This test is composed of many small separate-effect tests where the test conditions were step-wise changed to cover wide ranges of system pressure. Major results are briefly described in this section.

4.2.1 Introduction

Significant steam condensation on cool ECCS water injected into a steam phase may cause unstable phenomena such as condensation-induced water hammer (CIWH), especially for passive safety features where cool water may be injected at low velocity yet high flow rate with small driving force thus at a rather unstable manner compared to the pump-driven ECCS injection in the conventional LWRs. The unstable phenomena may happen in the ECCS injection line etc. when the flow becomes rather stagnant and/or oscillative. While many experiments on the CIWH have been performed so far, the knowledge, especially at high-pressure condition typical to LOCAs and transients, is still insufficient [22]. A CIWH test was thus performed to observe the influences of system pressure.

4.2.2 Test Facility and Test Conditions

A horizontal pipe test section of about 2 m-long, 67 mm i.d. was prepared to clearly define boundary conditions and connected to the pressure vessel downcomer of the LSTF well below the cold leg. The instrumentations used were: 35 T/Cs to measure detailed temperature distribution, 8 high-response pressure fluctuation meters, and 1 gamma-ray densitometer to measure liquid level near the exit of test section.

Room-temperature water was supplied from the seal end of the test section by a HPIS pump, forming a horizontal stratified flow to prepare a counter-current flow condition against condensing steam flow from the downcomer as shown in Fig. 5. Steam was continuously generated in the core at low core power. Primary coolant was also continuously drained from the PV bottom to control the flow conditions in the test section via liquid level in the downcomer. The room-temperature water was supplied to the test section at 3 different flow rates while keeping the system pressure constant at 0.3 to 7 MPa.

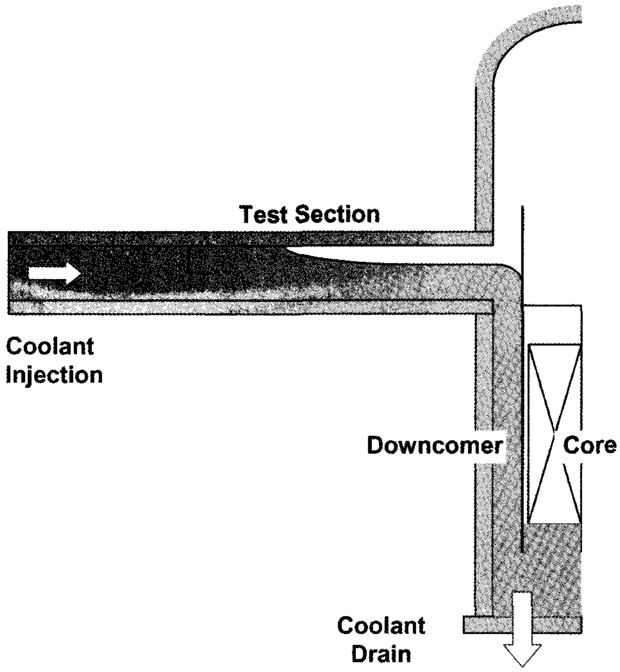

Fig. 5. Test Procedure for CIWH Experiment

4.2.3 Test Result

The averaged intensity of pressure pulses induced by CIWH shown in Fig. 6 was maximized at the system pressure of 1 MPa irrespective of the water supply condition range, and gradually decreased as the system pressure was increased, thereafter. In particular, a specific acoustic noise typical to the CIWH became weak beyond 4 MPa, while periodical pressure pulses kept appearing with high intensity. Some of the pressure pulses gave a loud noise even at 7 MPa.

The relationship between the averaged intensity of the pressure pulses and the averaged superficial steam velocity in the horizontal pipe was experimentally identified as shown in Fig. 7. The averaged intensity increased with the superficial steam velocity. Since steam from the downcomer was perfectly condensed in the test section, the steam velocity is in proportion to the volumetric condensation rate. The steam condensation rate should

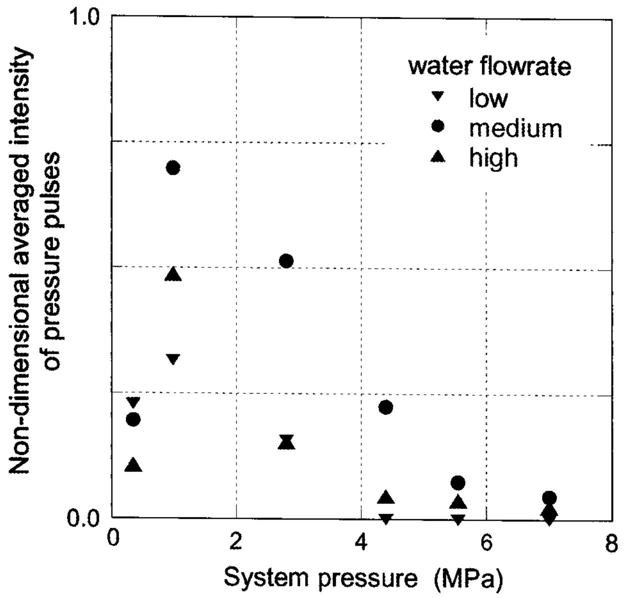

Fig. 6. Average Intensity of Pressure Pulses induced by CIWH

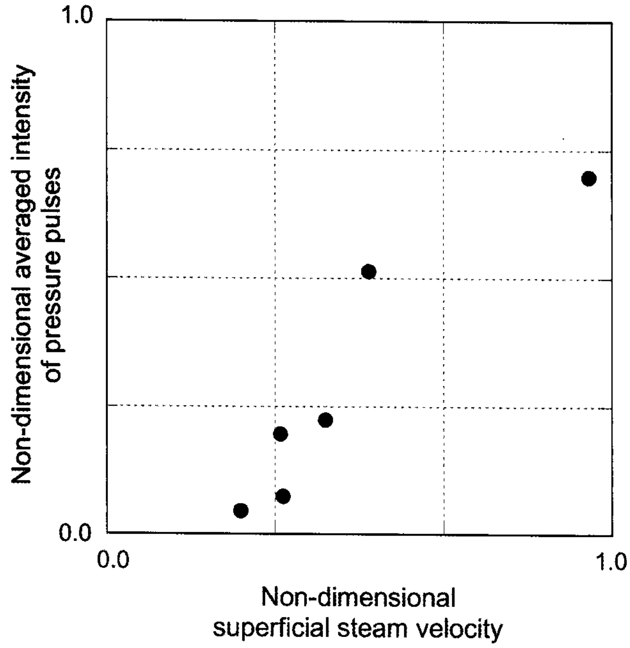

Fig. 7. Relation of Intensity of Pressure Pulses and Estimated Superficial Steam Velocity

then be highly related to the local depressurization rate of a steam bubble enclosed by a water slug that is formed just before the CIWH. The steam bubble disappears rapidly, inducing a large pressure pulse because of the high acceleration of the water slug when the steam condensation rate, thus the inlet steam velocity, is large. It is expected that the steam condensation rate is high enough to cause CIWH under high pressure condition up to 7 MPa.

4.2.4 Summary

The influence of system pressure on the intensity of CIWH pressure pulses was observed in a horizontal pipe test section newly connected to the LSTF pressure vessel for the ROSA Project test. It was found that the averaged intensity of the CIWH pressure pulses has a peak at system pressure of around 1 MPa and decreases as the system pressure increases. The maximum intensity of the CIWH pressure pulses is still large even when the system pressure is 7MPa. This test indicates the possibility of CIWH occurrence even at high pressures.

4.3 High Core Power Natural Circulation (Test 3)

Two system integral tests were performed under the assumption of failure of the reactor scram to investigate natural circulation at high core power. One was a SBLOCA test with 1% break at the cold leg and the other was a loss-of-feedwater transient similar to TMLB' scenario but with auxiliary feedwater. The former SBLOCA test is introduced in this section.

4.3.1 Background and Objectives

High reliability of control rods results in relatively low risk for anticipated transient without scram (ATWS). Failure of scram during a SBLOCA, however, should lead to relatively high core power for a long time and significant thermal-hydraulic responses adverse to core cooling, under a condition of gradual loss of primary mass inventory from the break. The high-power natural circulation test assuming the failure of scram during cold leg SBLOCA was thus conducted with LSTF to investigate local phenomena peculiar to high-power natural circulation and the influences on core cooling. The experiment was analyzed by RELAP5/MOD3.2 code [23] to validate the code predictability.

4.3.2 Test Conditions

The break size of 1.0% was selected to keep natural circulation for relatively long time at high core power. The core power curve is defined based on the PWR SBLOCA analysis with three-dimensional thermal-hydraulics and neutron kinetics coupling code SKETCH-INS/TRAC-PF1 [24]. The LSTF core power curve was obtained as a 1/48 volumetrically-scaled PWR core power curve with a cut off for the portion higher than 10 MW (14%) due to the limitation in the electric core power supply. Loss of off-site power and actuation of auxiliary feedwater concurrent with the scram signal were assumed as well as total failure of the high-pressure injection system (HPIS).

4.3.3 Test Result and Comparison with Code Prediction

Two-phase natural circulation (NC) at high core power appeared soon after the break and continued until about 300 s, causing SG relief valves to be kept opened as shown in Fig. 8. During the two-phase NC period, horizontal separated flow appeared in the hot legs at high vapor and liquid flow rates and, therefore, became supercritical flow with a low liquid level as shown in Fig. 9. After the two-phase NC, a large liquid column formed in SG U-tubes in both the upflow and downflow sides until about 500 s because of counter-current flow limiting (CCFL) at the inlet of the U-tubes due to high vapor velocity. The SG inlet plenum was filled with a two-phase mixture. No CHF happened in the core in the early stage of the transient as the core power was around 8% at the transition from two-phase NC to the water-column stage.

The primary to secondary pressure difference was large, but gradually decreased by about 500 s. The length of the liquid column in the SG U-tubes was longer in the upflow side than in the downflow side, causing a liquid level drop in the upper plenum, though the influence was negligible to core cooling as the core was well covered by the mixture level. The SG secondary pressure was almost the same between the two SGs and gradually decreased after about 500 s because of continuous injection of cool water by auxiliary feedwater. The primary pressure followed the SG secondary pressure as shown in Fig. 8.

Unlike other cold leg SBLOCA experiments, no loop-seal clearing occurred. This was probably due to the

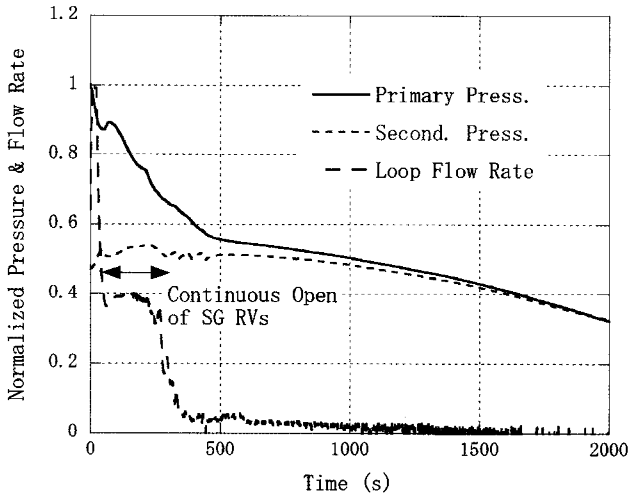

Fig. 8. Primary & Secondary Pressures and Loop Flow Rate

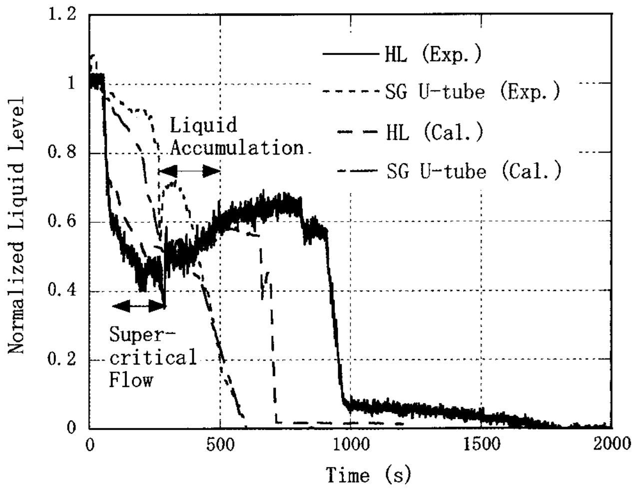

Fig. 9. Liquid Levels in Hot leg and SG U-tube Upflow-side (Comparison of Calculated and Measured Results)

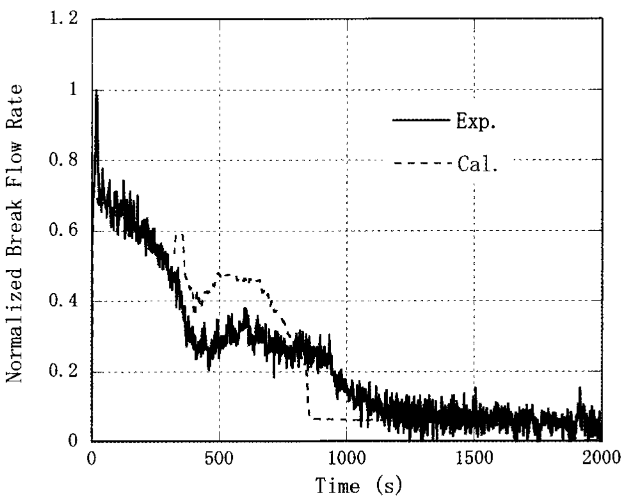

Fig. 10. Break Flow Rate (Comparison of Calculated and Measured Results)

formation of a large water column in the upflow side SG U-tubes. Core boil-off took place instead, rather rapidly. This caused significant temperature excursion. The LSTF protection system automatically started to decrease the core power before the actuation of accumulators (ACCs). Significant steam condensation occurred in the cold leg on the injected ACC coolant, which induced the loop-seal clearing in the loop without pressurizer. The whole core was quenched by about 2200 s by the ACC coolant (not shown).

The RELAP5/MOD3.2 code predicted the observed thermal hydraulic phenomena rather well, though it indicated several odd responses. The two-phase NC flow rate was overestimated, probably due to incorrect estimation of reverse flow in some of SG U-tubes, resulting in the fast primary depressurization (not shown). The break flow rate was overestimated, especially during the two-phase discharge period after the termination of the two-phase NC as shown in Fig. 10 because the cold leg liquid level was estimated to be much higher than measured. This caused an earlier decrease in the calculated primary mass inventory, thus earlier initiation of core uncovery (not shown). The liquid level overprediction in the cold leg was caused by an underprediction of vapor flow rate from the PV upper head to the downcomer through spray nozzles.

4.3.4 Summary

A 1.0% cold leg SBLOCA of PWR with a failure of scram was simulated by LSTF to investigate the thermal-hydraulic phenomena that may arise under a high core power transient. Several thermal-hydraulic phenomena peculiar to high-power conditions were observed. These included supercritical flow in hot legs, liquid accumulation in SG U-tubes in both upflow-side and downflow sides, and no loop-seal clearing because of a large water column in SG U-tubes. The RELAP5 code predicted the observed overall transient rather well, but it indicated several points for further improvement, such as the correct prediction of two-phase NC with a good representation of reverse flow SG U-tubes and liquid level in the horizontal leg.

4.4 Pressure Vessel Small-break LOCA (Test 6)

Two SBLOCA system integral tests were performed simulating a break at the pressure vessel (PV) in combination with simulated accident management (AM) measures. They were a PV top break test and a PV bottom break test. SG depressurization was performed as the AM measure. The former test has been introduced in [25,26] and the latter test is briefly introduced in this section.

4.4.1 Background and Objectives

A small amount of residue including boron was found around the circumference of two instrument-tube penetration nozzles of PV lower-head at the US South Texas Project Unit-1 in 2003. This event raised a safety issue concerning vessel structural integrity, as a SBLOCA may occur by the ejection of the PV lower-head instrument-tube. A PV bottom break SBLOCA test under an assumption of total failure of HPIS was performed using the LSTF to observe the consequences of such a LOCA and to provide a database for code assessment. Asymmetrical SG secondary-side depressurization was performed as an AM action. The effects of unbalanced natural circulation (NC) between loops and unbalanced transport of non-condensable gas during the depressurization were investigated. Further post-test analysis was conducted to validate the predictability of the RELAP5/MOD3.2 code.

4.4.2 Test Conditions

A 3.2 mm inner-diameter sharp-edge orifice was used to simulate the break; the size of the break is equivalent to 0.1% cold leg break to simulate the ejection of one whole instrument-tube (one-inch in outer diameter) of PWR PV bottom. Loss of off-site power concurrent with the scram signal when the primary pressure decreased to 12.97 MPa was assumed as well as the total failure of the HPIS. Then SG secondary-side depressurization as a symptom-oriented AM operator action was initiated 30 minutes after the safety injection signal when the primary pressure decreased to 12.27 MPa. The depressurization rate was controlled to achieve the primary depressurization rate of 55 K/h. The depressurization was performed asymmetrically only at the SG in the loop without a pressurizer (loop B). Nitrogen gas inflow into the primary loops from the accumulator (ACC) system was assumed to observe the influences of non-condensable gas.

4.4.3 Test Result and Comparison with Code Prediction

The primary pressure decreased after the break rather gradually and almost reached the SG secondary-side pressure in about 2500 s. Around this time the SG depressurization as an AM action was started in loop B. However, primary pressure did not start to decrease because the primary-to-secondary heat transfer was limited

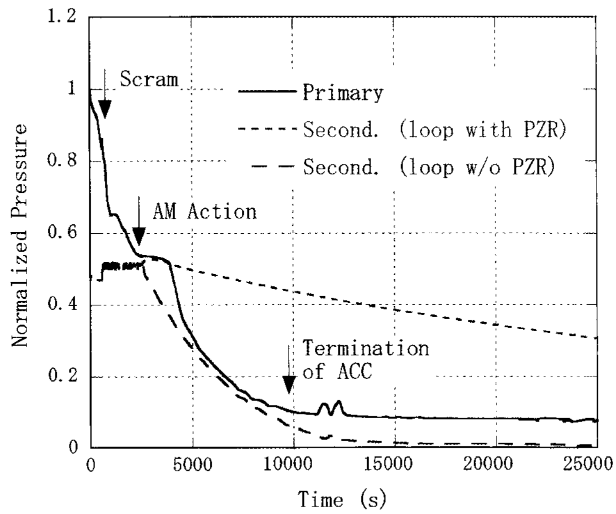

Fig. 11. Primary and Secondary Pressures

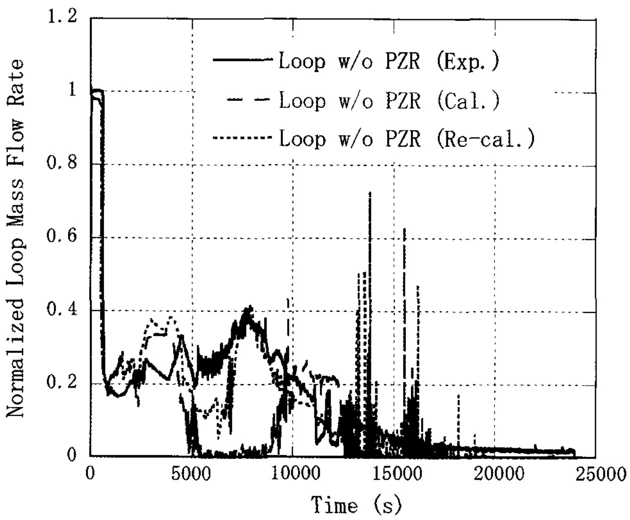

Fig. 12. Primary Loop Mass Flow Rate (Comparison of Calculated and Measured Results)

as a single-phase liquid NC was induced in the loop B while almost stagnant (reflux cooling) in the loop A (loop with pressurizer) where SG U-tubes were filled with steam. The heat transfer through NC in loop B controlled the primary pressure, and thus total system response thereafter.

The primary depressurization was greatly limited when nitrogen gas entered the SG U-tubes through cold legs, downcomer, and hot legs from ACC tanks after the completion of coolant injection. The primary pressure even increased temporarily as shown in Fig. 11. The primary pressure almost ceased decreasing. This caused a very late actuation of the low-pressure injection system (LPIS) well after the core uncovery due to core boil-off.

In the RELAP5/MOD3.2 code analysis, NC in the loop B was terminated after the initiation of the AM action due to the overprediction of void fraction and phase separation at the SG U-tube top. The code predicted accurately the loop mass flow rate in loop B by the continuous NC when the interfacial drag coefficient was 10 times at the hot leg bend to SG inlet, SG inlet plenum and U-tube upflow-side in the loop B as shown in Fig. 12.

4.4.4 Summary

A PWR PV bottom SBLOCA test was conducted with an asymmetrical SG depressurization as an AM measure. The observed thermal-hydraulic response was rather mild. The asymmetric SG depressurization, however, caused unique responses that include strongly asymmetric NC between two loops under influences of non-condensable gas from ACCs. It was clarified that the core uncovery may occur once the primary depressurization is obstructed when non-condensable gas is accumulated in the SG U-tubes under such conditions because of difficulty in the LPIS actuation. The post-test analyses revealed that RELAP5 code has a problem in the calculation of interfacial drag from the hot leg to SG inlet that is important for the correct prediction of two-phase NC.

5. SUMMARY AND FUTURE UTILIZATION

Since the TMI-2 Accident in 1979, significant effort has been done worldwide to provide experimental data for thermal-hydraulic safety of LWRs, especially concerning SBLOCAs and operational transients. The ROSA/LSTF experiments have been performed to identify, observe, and clarify thermal-hydraulic phenomena during the SBLOCAs and operational transients and to confirm the effectiveness of ECCSs by system integral and separate-effect tests since the shakedown in 1985. Various safety-relevant subjects and issues have been investigated. These included the passive safety features for both PWR and BWR, the events during plant outage and the simulation of real plant accident including SGTR. The obtained data has been utilized intensively for the verification and development of BE safety analysis codes.

The LSTF is currently utilized for the OECD/NEA ROSA Project to provide an experimental database on complex flows necessary to validate code predictive capability and accuracy of models relevant for safety analyses. The obtained database has new findings in every experiment and thus is good for the investigation of phenomena and the verification and development of computer codes including CFD codes. Detailed measurement with newly installed instrumentation including T/C arrays contributes to the preparation of database. The results of four experiments are briefly introduced here to illustrate the capabilities of the ROSA/LSTF to simulate, represent, and measure the phenomena of interest. The observed phenomena in the four experiments include coolant mixing and temperature stratification, water hammer up to high system pressure, natural circulation under high core power condition and influences of non-condensable gas during asymmetric SG depressurization as an AM action. The obtained data was utilized for the validation of one-dimensional LOCA code of RELAP5/MOD3.2.1.2 and CFD code of FLUENT to identify several problems for further improvement.

The ROSA/LSTF will be continuously utilized for the 2nd Phase of the OECD/NEA ROSA Project that may start in 2009 for safety improvement and regulation support mainly through simulations of intermediate-break LOCA and SGTR for estimation and minimization of radioactive release. The safety verification of a next-generation LWR under design among utilities, vendors and the government of Japan would then be performed through integral tests covering a wide range of accident and transient conditions.

The LSTF experiments are carefully planned among participants for cooperative research projects to continuously provide system integral and separate-effect test data for the evaluation and confirmation of reactor thermal-hydraulic safety during accidents and transients.

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