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Data Report of ROSA/LSTF Experiment SB-PV-03

-0.2% Pressure Vessel Bottom Break LOCA with
SG Depressurization and Gas Inflow-

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An experiment denoted as SB-PV-03 was conducted on November 19, 2002 using the Large Scale Test Facility (LSTF) in the Rig of Safety Assessment-V (ROSA-V) Program. The ROSA/LSTF experiment SB-PV-03 simulated a 0.2% pressure vessel bottom small-break loss-of-coolant accident in a pressurized water reactor (PWR). The test assumptions included total failure of high pressure injection system of emergency core cooling system (ECCS) and non-condensable gas (nitrogen gas) inflow to the primary system from accumulator (ACC) tanks of ECCS. Secondary-side depressurization of both steam generators (SGs) as an accident management (AM) action to achieve the depressurization rate of 55 K/h in the primary system was initiated 10 min after the generation of a safety injection signal, and continued afterwards. Auxiliary feedwater injection into the secondary-side of both SGs was started for 30 min with some delay after the onset of the AM action.

The AM action was effective on the primary depressurization until the ACC tanks began to discharge nitrogen gas into the primary system. The core liquid level recovered in oscillative manner because of intermittent coolant injection from the ACC system into both cold legs. Therefore, the core liquid level remained at a small drop. The pressure difference between the primary and SG secondary sides became larger after nitrogen gas ingress.

Core uncover occurred by core boil-off during reflux condensation in the SG U-tubes under nitrogen gas influx. When the maximum cladding surface temperature of simulated fuel rods exceeded the pre-determined value of 908 K, the core power was automatically reduced to protect the LSTF core. After the automatic core power reduction, coolant injection from low pressure injection (LPI) system of ECCS into both cold legs led to the whole core quench.

After the continuous core cooling was confirmed through the actuation of the LPI system, the experiment was terminated.

This report summarizes the test procedures, conditions, and major observations in the ROSA/LSTF experiment SB-PV-03.

Keywords: PWR, LSTF, Pressure Vessel Bottom Break, Small Break LOCA, Accident Management, Steam Generator Depressurization, Gas Inflow, Core Boil-off

ROSA/LSTF 実験 SB-PV-03 データレポート
—ガスが流入する条件での 0.2%圧力容器底部小破断冷却材喪失事故時蒸気発生器減圧—

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(2024 年 10 月 9 日受理)

ROSA-V 計画において、大型非定常実験装置(LSTF)を用いた実験(実験番号:SB-PV-03)が 2002 年 11 月 19 日に行われた。ROSA/LSTF SB-PV-03 実験では、加圧水型原子炉(PWR)の 0.2%圧力容器底部小破断冷却材喪失事故を模擬した。このとき、非常用炉心冷却系(ECCS)である高圧注入系の全故障とともに、蓄圧注入(ACC)タンクから一次系への非凝縮性ガス(窒素ガス)の流入を仮定した。また、アクシデントマネジメント(AM)策として両蒸気発生器(SG)二次側減圧を安全注入設備信号発信後 10 分に一次系減圧率 55K/h を目標として開始し、その後継続した。さらに、AM 策から少し遅れて両 SG 二次側への 30 分間の補助給水を開始した。

ACC タンクから一次系への窒素ガスの流入開始まで、AM 策は一次系減圧に対して有効であった。ACC 系から両低温側配管への間欠的な冷却材注入により、炉心水位は振動しながら回復した。このため、炉心水位は小さな低下にとどまった。窒素ガスの流入後、一次系と SG 二次側の圧力差が大きくなった。

窒素ガス流入下における SG 伝熱管でのリフラックス凝縮時に、ボイルオフによる炉心露出が生じた。模擬燃料棒の被覆管表面最高温度が LSTF の炉心保護のために予め決定した値(908K)を超えたとき、炉心出力は自動的に低下した。炉心出力の自動低下後、ECCS である低圧注入(LPI)系から両低温側配管への冷却材注入により、全炉心はクエンチした。

LPI 系の作動を通じた継続的な炉心冷却を確認後、実験を終了した。

本報告書は、ROSA/LSTF SB-PV-03 実験の手順、条件および実験で観察された主な結果をまとめたものである。

Contents

1.	Introduction	1
2.	Overview of LSTF	4
3.	Test Conditions and Procedures	6
3.1	Initial Steady State and Boundary Conditions	7
3.2	Test Procedures	8
3.3	Instrumentation	9
3.3.1	Measured Data	10
3.3.2	Data Conversion, Reduction, and Calibration	10
3.3.3	Data Qualification	11
4.	Experimental Results	18
4.1	Initial and Boundary Conditions	18
4.2	Thermal-hydraulic Responses Concerning Boundary Conditions	18
4.3	Transient Thermal-hydraulic Responses	20
4.3.1	Thermal-hydraulic Responses in Pressure Vessel	21
4.3.2	Thermal-hydraulic Responses in Primary Loops	23
4.3.3	Thermal-hydraulic Responses of Steam Generators	24
5.	Summary	55
	Acknowledgement	56
	References	56
	Appendix A Available Experimental Data List	58

目次

1. 緒言	1
2. LSTF の概要	4
3. 実験条件および手順	6
3.1 初期定常および境界条件	7
3.2 実験手順	8
3.3 計装	9
3.3.1 計測データ	10
3.3.2 データ変換、処理および校正	10
3.3.3 データ評価	11
4. 実験結果	18
4.1 初期および境界条件	18
4.2 境界条件に係る熱水力応答	18
4.3 過渡熱水力応答	20
4.3.1 圧力容器内の熱水力応答	21
4.3.2 一次系ループ内の熱水力応答	23
4.3.3 蒸気発生器の熱水力応答	24
5. 結言	55
謝辞	56
参考文献	56
付録 A 利用可能な実験データリスト	58

List of Tables

Table 1-1	Major features of LSTF and PKL.....	3
Table 1-2	Major information of LSTF tests for counterpart testing in PKL-related OECD/NEA Projects.....	3
Table 3-1	Specified initial steady state and boundary conditions.....	12
Table 3-2	Specified control logic, operation set points, and conditions.....	14
Table 3-3	Specified pump rotation speed ratio after scram signal.....	14
Table 3-4	Pre-determined core power decay curve after scram signal.....	15
Table 3-5	Pressure vessel lower plenum pressure versus LPI flow rate.....	15
Table 3-6	Pre-determined secondary pressure drop curve after AM action onset	15
Table 4-1	Initial steady state conditions.....	26
Table 4-2	Chronology of major events until break valve closure.....	27
Table A-1	List of available experimental data for LSTF SB-PV-03.....	59

List of Figures

Fig. 2-1	Schematic view of the Large Scale Test Facility (LSTF)	5
Fig. 2-2	Schematic diagram of the Large Scale Test Facility (LSTF)	5
Fig. 3-1	Configuration of break unit	16
Fig. 3-2	Core heater rod zone and heater power	16
Fig. 3-3	Schematic view of SG steam line	17
Fig. 4-1	Core power.....	28
Fig. 4-2	Pressurizer heater power.....	29
Fig. 4-3	Pressurizer liquid level	29
Fig. 4-4	Primary coolant pump rotation speed.....	30
Fig. 4-5	Primary loop mass flow rate.....	30
Fig. 4-6	SG main steam flow rate.....	31
Fig. 4-7	SG main feedwater flow rate.....	31
Fig. 4-8	SG secondary-side collapsed liquid level.....	32
Fig. 4-9	Auxiliary feedwater flow rate.....	32
Fig. 4-10	Time-integrated break flow and break flow rate.....	33
Fig. 4-11	Lower plenum fluid temperature.....	33
Fig. 4-12	Liquid level in accumulator tank.....	34
Fig. 4-13	Coolant injection flow rate from accumulator tank (0 to 10000 s).....	34
Fig. 4-14	Coolant injection flow rate from accumulator tank (3000 to 8000 s)	35
Fig. 4-15	Coolant injection flow rate from LPI system.....	35

Fig. 4-16	Primary and secondary pressures	36
Fig. 4-17	SG relief valve flow rate	36
Fig. 4-18	Upper plenum collapsed liquid level (0 to 10000 s)	37
Fig. 4-19	Upper plenum collapsed liquid level (3000 to 7000 s)	37
Fig. 4-20	Core collapsed liquid level (0 to 10000 s)	38
Fig. 4-21	Core collapsed liquid level (3000 to 7000 s)	38
Fig. 4-22	Typical core exit temperatures (0 to 10000 s)	39
Fig. 4-23	Typical core exit temperatures (8400 to 9800 s)	39
Fig. 4-24	Typical cladding surface temperatures at Positions 9-5 (0 to 10000 s)	40
Fig. 4-25	Typical cladding surface temperatures at Positions 9-5 (3000 to 7000 s)	40
Fig. 4-26	Typical cladding surface temperatures at Positions 4-1 (0 to 10000 s)	41
Fig. 4-27	Cladding surface temperatures at Position 9 (8400 to 9800 s)	41
Fig. 4-28	Cladding surface temperatures at Position 8 (8400 to 9800 s)	42
Fig. 4-29	Cladding surface temperatures at Position 7 (8400 to 9800 s)	42
Fig. 4-30	Cladding surface temperatures at Position 6 (8400 to 9800 s)	43
Fig. 4-31	Cladding surface temperatures at Position 5 (8400 to 9800 s)	43
Fig. 4-32	Cladding surface temperatures at Position 4 (8400 to 9800 s)	44
Fig. 4-33	Cladding surface temperatures at Position 3 (8400 to 9800 s)	44
Fig. 4-34	Downcomer collapsed liquid level	45
Fig. 4-35	Upper-head collapsed liquid level	45
Fig. 4-36	Hot leg liquid level	46
Fig. 4-37	Hot leg fluid temperature	46
Fig. 4-38	Cold leg liquid level	47
Fig. 4-39	Cold leg fluid temperature	47
Fig. 4-40	Crossover leg downflow-side collapsed liquid level	48
Fig. 4-41	Crossover leg upflow-side collapsed liquid level	48
Fig. 4-42	SG U-tube upflow-side collapsed liquid level in loop with pressurizer	49
Fig. 4-43	SG U-tube downflow-side collapsed liquid level in loop with pressurizer	49
Fig. 4-44	SG U-tube upflow-side collapsed liquid level in loop without pressurizer	50
Fig. 4-45	SG U-tube downflow-side collapsed liquid level in loop without pressurizer	50
Fig. 4-46	SG U-tube downflow-side fluid temperature in loop with pressurizer	51
Fig. 4-47	SG U-tube downflow-side fluid temperature in loop without pressurizer	51
Fig. 4-48	Nitrogen gas mole fraction of SG U-tube in loop with pressurizer	52
Fig. 4-49	Nitrogen gas mole fraction of SG U-tube in loop without pressurizer	52
Fig. 4-50	SG inlet plenum collapsed liquid level	53
Fig. 4-51	SG secondary-side fluid temperature in loop with pressurizer	54
Fig. 4-52	SG secondary-side fluid temperature in loop without pressurizer	54

Acronyms and Abbreviations

ACC	Accumulator
AFW	Auxiliary Feedwater
AM	Accident Management
CET	Core Exit Temperature
ECCS	Emergency Core Cooling System
HPI	High Pressure Injection
LPI	Low Pressure Injection
LSTF	Large Scale Test Facility
NEA	Nuclear Energy Agency
OECD	Organisation for Economic Co-operation and Development
PKL	Primärkreisläufe Versuchsanlage
PWR	Pressurized Water Reactor
PZR	Pressurizer
ROSA	Rig-of-Safety Assessment
SBLOCA	Small-Break Loss-of-Coolant Accident
SG	Steam Generator
SI	Safety Injection
UCP	Upper Core Plate

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1. Introduction

An event occurred at the South Texas Project Unit-1 pressurized water reactor (PWR) in the US in 2003 [1]. Namely, inspections from outside of the pressure vessel revealed that a small amount of residue containing boron was found around the circumference of two instrument-tube penetration nozzles of pressure vessel lower-head. This raised a safety issue concerning structural integrity of pressure vessel. Multiple instrument-tubes may be ejected on account of circumferential cracks at nozzles of pressure vessel lower-head. This would have caused a small-break loss-of-coolant accident (SBLOCA) at the pressure vessel bottom. Liquid-phase break is characterized by fast loss of coolant inventory with slow depressurization of the primary system [2][3]. As for an accident management (AM) measure against SBLOCA, steam generator (SG) secondary-side depressurization being coupled with auxiliary feedwater (AFW) injection into the SG secondary-side is important to depressurize the primary system [4]. This may activate low pressure injection (LPI) system of emergency core cooling system (ECCS), which enables long-term core cooling during SBLOCA especially when high pressure injection (HPI) system of ECCS is in failure. However, experimental data have been scarcely obtained for such a pressure vessel bottom SBLOCA with the AM measure under complete failure of HPI system [5].

An experiment denoted as SB-PV-03 was carried out on November 19, 2002, on a PWR pressure vessel bottom SBLOCA [6]. This test utilized the Large Scale Test Facility (LSTF) [7] in the Rig-of-Safety Assessment-V (ROSA-V) Program. The LSTF simulates a Westinghouse-type four-loop 3423 MW (thermal) PWR by a full-height and 1/48 volumetrically-scaled two-loop system. The break size was equivalent to 0.2% of the volumetrically-scaled (1/48) cross-sectional area of the reference PWR cold leg. Thus, in the LSTF test even for a break at non-cold leg, the break size is based on the scaled cold leg cross-sectional area. Water injection from HPI system of ECCS is expected as a function to mitigate the consequences in such 0.2% break LOCA case. On the contrary, the SB-PV-03 experiment assumed that HPI system of ECCS was unavailable.

In the new regulatory requirements for PWR in Japan, an event of loss of ECCS water injection functions was postulated as one of the accident sequences, which should be evaluated the effectiveness of measures against core damage [8]. A medium-break LOCA or SBLOCA with the assumption of a totally-failed HPI system is the event that represents the loss of ECCS water injection functions [9]. Appropriate measures for this event should be taken such as the primary depressurization through early depressurization of the SG secondary-side because of steam condensation in the SG U-tubes and early actuation of the AFW system to provide feedwater into the SG secondary-side.

In the SB-PV-03 experiment, secondary-side depressurization of both SGs as an AM action

to attain the depressurization rate of 55 K/h in the primary system was taken 10 min after the generation of a safety injection (SI) signal, and continued thereafter. AFW was injected for 30 min into the secondary-side of both SGs with some delay after the initiation of the AM action, which may produce less heat removal from the SG secondary-side system. The SB-PV-03 experiment also supposed non-condensable gas (nitrogen gas) ingress to the primary system from accumulator (ACC) tanks of ECCS due to failure of the ACC system isolation after the coolant injection initiation. The effectiveness of the AM action should thus be confirmed under the influence of nitrogen gas influx from the ACC tanks.

The SB-PV-03 test objectives were as follows; (i) to elucidate thermal-hydraulic phenomena involved, (ii) to clarify the AM action effectiveness for both core cooling and primary depressurization, and (iii) to provide experimental data for the assessment of thermal-hydraulic safety analysis computer codes. The LSTF test data would be useful to define the conditions of counterpart testing by means of other integral test facilities such as PKL (Primärkreisläufe Versuchsanlage) [10]. **Table 1-1** compares major features of the LSTF and PKL. Volumetric scaling is 1/48 in the LSTF, whereas it is 1/145 in the PKL. The LSTF runs at full pressure, whereas the PKL pressure is limited to 5 MPa. The number of loops is two in the LSTF, and four in the PKL. The axial core power profile is a 9-step chopped cosine in the LSTF, whereas it is flat in the PKL. The aim of the counterpart testing is to address scaling problems through thermal-hydraulic phenomena at similar or identical pressure ranges. As summed up in **Table 1-2**, in this regard, the LSTF test data have been provided on SBLOCAs under different conditions of break, AM action and its onset timing, and nitrogen gas inflow to the primary system from the ACC system for the counterpart testing in the PKL-related OECD/NEA Projects [5], [11]-[20].

This report summarizes the test procedures, conditions, and major observations in the ROSA/LSTF experiment SB-PV-03. All the experimental data were processed carefully and qualified to acquire the best possible accuracy.

Table 1-1 Major features of LSTF and PKL

Item	LSTF	PKL
Height	Full	Full
Volumetric scaling	1/48	1/145
Number of loops	2	4
Pressure vessel downcomer	Cylindrical	Double-pipe
Number of U-tubes per SG	141	30
Pressure	Full	Up to 5 MPa
Core power	14%	10%
Axial profile	Chopped cosine	Flat
Radial profile	3-region	3-region

Table 1-2 Major information of LSTF tests for counterpart testing
in PKL-related OECD/NEA Projects

Test ID [test year]	Break size, location [orientation]	AM action and onset timing	Nitrogen gas inflow	OECD/NEA Project [references]
SB-HL-18 [2011]	1.5%, hot leg [upward]	[Low pressure phase test] Full open of SG relief valves with AFW injection into SG secondary- side at core exit temperature of 623 K	No- consider	PKL-2 [11][12][13]
SB-CL-32 [1996]	1%, cold leg [horizontal]	SG secondary-side depressurization to achieve depressurization rate of 200 K/h in primary system with AFW injection into SG secondary-side 10 min after break	No- consider	PKL-3 [14][15][16]
SB-HL-12 [1998]	1%, hot leg [downward]	Full open of SG relief valves with AFW injection into SG secondary- side at cladding surface temperature of 600 K	Consider	PKL-3 [16][17][18]
SB-PV-09 [2005]	1.9%, pressure vessel upper head	Full open of SG relief valves with AFW injection into SG secondary- side at core exit temperature of 623 K	Consider	ETHARINUS [5][19][20]

Remarks: (1) The common test assumption was total failure of HPI system. (2) The SB-HL-18 test, broadly divided into the high pressure phase test and low pressure phase test, was conducted in the OECD/NEA ROSA-2 Project. The low pressure phase test corresponded to the counterpart testing. (3) The SB-PV-09 test was performed in the OECD/NEA ROSA Project.

2. Overview of LSTF

Figure 2-1 shows a schematic view of the LSTF that simulates a typical 3423 MW (thermal) four-loop Westinghouse-type PWR with a two-loop system model by full height and 1/48 in volume [7]. **Figure 2-2** gives the schematic diagram of the LSTF. The reference PWR of the LSTF is Tsuruga Unit-2 of Japan Atomic Power Company. The LSTF is composed of a pressure vessel, pressurizer (PZR), and primary loops. Each loop includes an active SG, primary coolant pump, crossover leg, hot leg, and cold leg. The crossover leg is a primary piping that connects the primary coolant pump to the SG. Loops with and without PZR are designated as loop-A and loop-B, respectively. The hot leg and cold leg, 207 mm in inner-diameter each, are sized to conserve the volumetric scale (2/48) and the ratio of length to square root of pipe diameter to better simulate flow regime transitions in the primary loops [21]. Each SG is furnished with 141 full-size U-tubes (inner-diameter of 19.6 mm each), inlet and outlet plena, boiler section, steam separator, steam dome, steam dryer, main steam line, four downcomer pipes, and other internals (see pp.260-273 in Ref. [7]). Each SG is equipped with six instrumentation tubes, which are available to measure differential pressures through differential pressure transducers and fluid temperatures with thermocouples (see pp.53-54 in Ref. [7]). Tubes 1 and 6 are short tubes (Type 1; see p.267 in Ref. [7]), Tubes 2 and 5 are medium tubes (Type 5), and Tubes 3 and 4 are long tubes (Type 9). The LSTF represents the reference PWR bypasses involving eight upper-head spray nozzles (inner-diameter of 3.4 mm each) (see p.202 in Ref. [7]) and the hot leg nozzle (see p.196 in Ref. [7]) leakage. The spray nozzles allow bypass flow that amounts to 0.3% of the total core flow rate during the initial steady state, while bypass area of the hot leg nozzle is set to allow 0.2% bypass flow for each loop. The LSTF core, 3.66 m in active height, mainly consists of 1008 electrically heater rods in 24 rod bundles to simulate the fuel rod assembly and 96 non-heating tie rods to simulate control rod guide thimble (see p.220 in Ref. [7]). The angle of the PZR surge line to the hot leg is 45°. All the types of ECCS fitted to the reference PWR are installed in the LSTF. Nitrogen gas is used for pressurization of ACC tanks of ECCS.

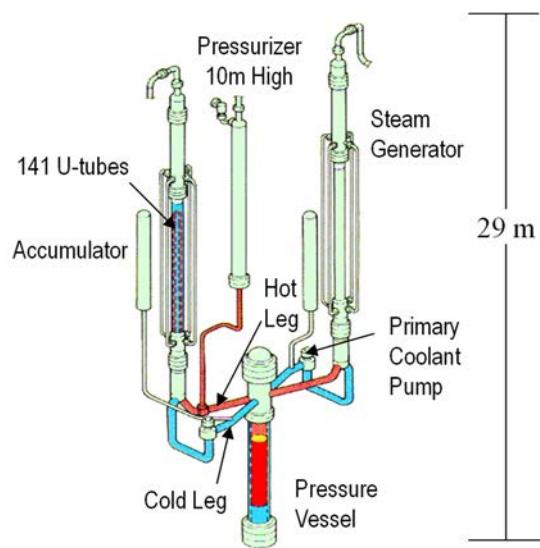


Fig. 2-1 Schematic view of the Large Scale Test Facility (LSTF)

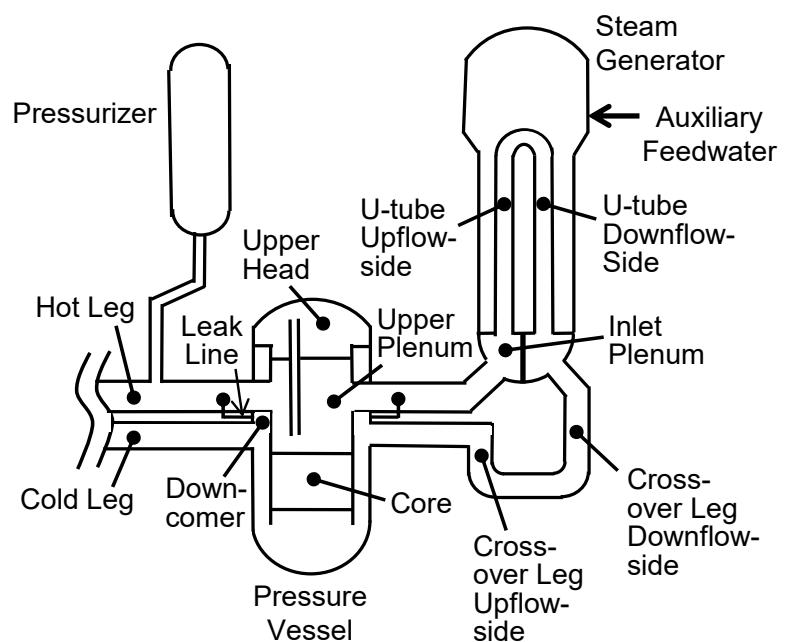


Fig. 2-2 Schematic diagram of the Large Scale Test Facility (LSTF)

3. Test Conditions and Procedures

With respect to the break, ECCS, and control logic, following assumptions were posed to the SB-PV-03 test conditions.

The assumption related to the break is the following;

- 1) Break size (flow area) corresponds to 0.2% of the volumetrically-scaled (1/48) cross-sectional area of the reference PWR cold leg. This is equal to the size of ten in-core instrumentation tubes (inner-diameter of 10.16 mm each) of pressure vessel lower-head. As presented in **Fig. 3-1**, the break is simulated by using a 4.6 mm inner-diameter sharp-edge orifice fixed at the downstream of a horizontal pipe that is connected to a lower plenum nozzle.

The assumptions concerning the ECCS are as follows;

- 2) Total failure of HPI system
- 3) When the primary pressure reduces to 4.51 MPa, ACC system is actuated in both loops. Coolant is provided from the ACC system into both cold legs. The ACC coolant injection flow rate ratio of loop with PZR to loop without PZR is designed to become 1:1. The ACC coolant injection temperature is 320 K.
- 4) For each loop, non-condensable gas (nitrogen gas) inflow to the primary system from the ACC tank takes place owing to failure of the ACC system isolation after the ACC coolant injection initiation.
- 5) When the pressure vessel lower plenum pressure declines to 1.24 MPa, LPI system is activated in both loops. Coolant is fed from the LPI system into both cold legs. The LPI coolant injection flow rate ratio of loop with PZR to loop without PZR is intended to become 1:1. The LPI coolant injection temperature is 310 K.

The assumptions regarding the control logic are as below;

- 6) When the primary pressure drops to 12.97 MPa, a scram signal is generated, causing closure of SG main steam stop valve, coastdown of primary coolant pumps, termination of SG main feedwater, and manual closure of SG main steam isolation valves.
- 7) Loss of off-site power concurrently with the scram signal generation
- 8) When the primary pressure lowers to 12.27 MPa, a SI signal is obtained.
- 9) Secondary-side depressurization of both SGs as an AM action to achieve the depressurization rate of 55 K/h in the primary system is made 10 min after the SI signal generation, and continues thereafter. The primary depressurization rate of 55 K/h is decided with reference to the depressurization rate during procedures of an operator action in PWR accidents where heat removal by SG(s) is possible [6]. The time delay of 10 min is taken account of a grace period to start an operator action [6].
- 10) AFW is supplied into the secondary-side of both SGs with some delay after the AM action

onset. The AFW injection duration is 30 min, supposing the use of turbine-driven AFW pump due to malfunction of electric AFW pump. The AFW flow rate into the SG secondary-side is planned to be about 0.59 kg/s in loop with PZR and about 0.72 kg/s in loop without PZR. This purpose is to look into an impact of the different AFW flow rate on the SG secondary-side collapsed liquid level. The AFW injection temperature is 310 K.

11) Thresholds of maximum cladding surface temperature for the LSTF core protection system are as follows;

908K=75%, 918K=50%, 919K=25%, 920K=10%, of pre-determined value [22], according to the conditions mentioned in **Table 3-1**.

3.1 Initial Steady State and Boundary Conditions

The specified initial steady state and boundary conditions are listed in **Table 3-1**. Initial steady state conditions such as PZR pressure, fluid temperatures in the hot leg and cold leg were 15.5 MPa, 598 K and 563 K respectively (to be indicated in **Table 4-1**), in accordance with to the reference PWR conditions.

The LSTF initial core power is limited to 10 MW, on account of a limitation in the capacity of power supply. The 10 MW power corresponds to 14% of the volumetrically-scaled (1/48) PWR nominal core power (3423 MW). Radial peaking factors of high-, mean-, and low-power rod bundles are 1.51, 1.00, and 0.66, respectively, in “Case 3” (see p.228 in Ref. [7]) shown in **Table 3-1**. The radial power profile is divided into two high-power heater rod zones (H1 and H2), one mean-power heater rod zone (M), and three low-power heater rod zones (L1, L2, and L3) (see p.228 in Ref. [7]), as mentioned in **Fig. 3-2**. The total number of the heater rods is 360, 180, and 468 for the high-power, mean-power, low-power heater rod zones, respectively. The specified power per heater rod is 9.69 kW in “Case 3”. Axial core power profile is a 9-step chopped cosine with a peaking factor of 1.495 (see p.227 in Ref. [7]). To attain the prototypical initial fluid temperatures with this core power, core flow rate was set to 14% of the 1/48-scaled nominal flow rate. Initial SG secondary-side pressure was raised to 7.3 MPa to limit the primary-to-secondary heat transfer rate to 10 MW, while 6.1 MPa is nominal value in the reference PWR. Initial PZR liquid level was about 7.4 m that is equivalent to about 66% of the PZR vessel height (see p.396 in Ref. [7]). Initial SG secondary-side collapsed liquid level was about 10.2 m that corresponds to the SG medium tube height.

Proportional heaters in the PZR are employed to trim the pressure, while backup heaters mitigate system heat losses. Powers of the PZR proportional and backup heaters were 2.8 kW and 33.7 kW, respectively, as the initial condition. The powers of the PZR proportional and backup heaters turned off when the PZR liquid level became below 2.3 m. Many regions of the LSTF are provided with trace heaters to mitigate environmental heat losses.

The configuration of the break unit is drawn in **Fig. 3-1**. The break was simulated by utilizing a 4.6 mm inner-diameter sharp-edge orifice No.11 (see p.307 in Ref. [7]). The orifice was mounted at the downstream of a horizontal pipe (inner-diameter of 87.3 mm) that was attached to a lower plenum nozzle (N-6b; see p.96 and p.195 in Ref. [7]). The orifice flow area was consistent with 0.2% of the 1/48-scaled cross-sectional area of the reference PWR cold leg. Venturi flow meter (FE-570-BU; see p.303 in Ref. [7]) was installed in the break unit for the purpose of qualitatively monitoring the break flow during the experiment.

For controlling ACC water injection volume (see p.322 in Ref. [7]), the specified initial water level and volume above the standpipe were 1.58 m and 1.12 m³ respectively for both loops. The specified initial volume of non-condensable gas (nitrogen gas) in the ACC tank was 0.46 m³ for both loops. The angle of the coolant injection from the ACC tank into the cold leg via the ECCS nozzle (N-14a or N-14b; see p.83 and pp.239-240 in Ref. [7]) is 90° for loop with PZR (loop-A) and 45° for loop without PZR (loop-B) (see pp.397-398 in Ref. [7]).

3.2 Test Procedures

Table 3-2 shows the specified control logic, operation set points, and conditions. The experiment was launched at time zero by opening a break valve located downstream of the break orifice at the pressure vessel bottom. At the same time, rotation speed of primary coolant pumps was increased up to about 1550 rpm in 4 s for better simulation of pressure and temperature transients in the reference PWR.

When the primary pressure dropped to 12.97 MPa, a scram signal was generated. This produced the closure of the SG main steam stop valve, the coastdown of the primary coolant pumps, the termination of the SG main feedwater, and the manual closure of the SG main steam isolation valves. **Table 3-3** shows the specified rotation speed ratio of primary coolant pump after the scram signal. The specified pump rotation speed was reduced to zero 250 s after the scram signal.

Table 3-4 shows the pre-determined core power decay curve after the scram signal, based on calculations by making use of the RELAP5 code considering delayed neutron fission power and stored heat in PWR fuel rod [22]. The core power was held at the initial value of 10 MW for 18 s until the scaled PWR core decay power dropped to 10 MW. The LSTF core power began a decay afterwards according to the specified core power.

The pressure set points for opening and closure of the SG relief valves are 8.03 MPa and 7.82 MPa, respectively, referring to the corresponding values in the reference PWR. The SG relief valve was simulated by employing a 16.2 mm inner-diameter sharp-edge orifice to provide steam flow rate of 2 kg/s when the SG secondary-side pressure is 8 MPa.

The ACC system is actuated in both loops at the primary pressure of 4.51 MPa according to the reference PWR. The ACC coolant injection flow rate ratio of loop with PZR to loop without PZR was designed to become 1:1 (to be plotted in **Figs. 4-13 and 4-14**). The ACC coolant injection temperature is 320 K, which is the same as that in the reference PWR.

The LPI system is activated in both loops at the pressure vessel lower plenum pressure of 1.24 MPa. As seen in **Table 3-5**, the LPI coolant injection flow rate is decided on the basis of Q-H curve for the LPI pump. The LPI coolant injection flow rate ratio of loop with PZR to loop without PZR was intended to become 1:1 (to be indicated in **Fig. 4-15**). The LPI coolant injection temperature is 310 K, which is equal to that in the reference PWR.

When the primary pressure lowered to 12.27 MPa, a SI signal was obtained. **Figure 3-3** shows the schematic view of the SG steam line (see pp.257-258 in Ref. [7]). There are two types of valves (i.e., gate valve and flow control valve) in the turbine bypass line, being parallel to the SG relief valve line. The SG secondary-side depressurization exploiting the flow control valves [Valve No.; FCV (flow control valve) -441 and FCV-442] (see p.106 in Ref. [7]) in both SGs was controlled such that the primary depressurization rate becomes to be 55 K/h. At that time, the gate valves [Valve No.; AOV (air-operated valve) -150 and AOV-180] (see pp.111-112 in Ref. [7]) and the relief valves [Valve No.; AOV-151 and AOV-181] (see pp.111-112 in Ref. [7]) in both SGs were open and closed, respectively. The SG secondary-side pressure drop curve after the initiation of the AM action was pre-determined, as listed in **Table 3-6**. This was based on the primary saturation temperature under an assumption that the primary pressure may closely follow the SG secondary-side pressure.

The AFW was supplied at 310 K into the secondary-side of both SGs with some delay after the start of the AM action (to be presented in **Figs. 4-8 and 4-16**). This avoids a significant drop in the SG secondary-side collapsed liquid level. The operator manually injected the AFW into each SG secondary-side for 30 min by monitoring the narrow-range (0-1 kg/s) feedwater flow rate [Tag Name; FE520B-PAA (for secondary-side of SG in loop with PZR), FE530B-PAB (for secondary-side of SG in loop without PZR)] (see pp.257-258 in Ref. [7]). The AFW flow rate into the SG secondary-side was expected to be about 0.59 kg/s in loop with PZR and about 0.72 kg/s in loop without PZR (to be indicated in **Fig. 4-9**). The AFW flow rate of about 0.72 kg/s in loop without PZR was nearly equivalent to the volumetric-scaled flow rate of the reference PWR.

3.3 Instrumentation

Instruments are equipped in the LSTF to understand and evaluate thermal-hydraulic responses during simulated accidents and transients.

3.3.1 Measured Data

A list of available experimental data is shown in **Table A-1**, which is composed of Sequential No., Function ID., Tag Name, measurement location, range, unit, and uncertainty. The Tag Name is a fixed naming unique to each measurement. The alphabetical prefix in the Function ID. and Tag Name represents the kind of variable or the kind of measurement as follows;

- TE, fluid temperature,
- DT, differential temperature,
- TW, heater rod and structure temperature,
- FE, flow rate measured with conventional (differential pressure) flow meters,
- PE, pressure,
- MI, miscellaneous instrumented-signal (power, pump rotation speed, etc.),
- LE, liquid level,
- DP, differential pressure,
- DE, fluid density measured with gamma-ray densitometer.

After the experiment, data from these measurements are processed to obtain the “secondary” data, such as area-averaged fluid density derived from measurement with three-beam gamma-ray densitometer. These data are stored with Function ID. starting with a prefix of “RC”. The measurement uncertainty is assessed according to the accuracy of the relevant instrument.

3.3.2 Data Conversion, Reduction, and Calibration

The instrumented-signals are recorded in volts by the data logger of DARWIN system (Yokogawa Electric Corporation), and are converted into engineering units utilizing appropriate conversion equations and factors. Differential pressure (DP) cell is a device that measures the differential pressure between two inputs. Some parameters such as flow rate (FE) and liquid level (LE) that employ DP cell data require the calculation of the single-phase coolant density based on local pressure and fluid temperature data using steam table.

DP cell data for both the differential pressures and liquid levels are corrected on the basis of a similar calibration test for static pressure effect, which is performed separately. Three-valve manifold is operated for each of DP cells to obtain zero calibration data for 200 s twice at a little before the break valve opening and at a little after the closure of the break valve.

The applicability of flow rates measured with the conventional flow meters employing venturi, orifice or nozzle and DP cell is limited in principle to either single-phase liquid or vapor flow. In addition, the accuracy is poor when the readings are below about 20% of the measurement range. This is explained by the fact that the flow rate is proportional to the square root of the

measured DP. For example, a zero level drift of 1% in the DP cell output may result in the flow rate reading of 10% of the measurement range especially when the actual flow rate is nearly equal to zero. It is thus good to pay attention when the flow rate is below about 20% of the measurement range even though the data are corrected based on a calibration test for static pressure effect.

Two-phase flow instruments, such as gamma-ray densitometers, use certain conversion equations considering attenuation effects of gamma-ray that goes through coolant flow.

After the data acquisition, some experimental data are calibrated. The high-range pressure data in the PZR and the upper plenum, for example, are corrected on the basis of a zero level shift using the low-range pressure data first, and then all the density data are calibrated at two points with different fluid conditions.

3.3.3 Data Qualification

The experimental data are qualified manually. Thermocouple data are reviewed by employing pre-test ambient temperature data for anomalous readings, and are mutually compared with readings of instruments in the same vicinity. Pressure transducers are checked for zero level drift as well as any other suspicious behaviors. The outputs of conductance probe, power meters, pump speed and vibration meters, and valve position indicators are individually reviewed for inconsistent readings.

The flow meters, DP transducers, gamma-ray densitometers, and drag disk transducers require extensive manual qualification efforts. The validity of the flow meters and differential pressure data mostly depends on whether the reading is in the sensitive range of the measurement or not. The data from these instruments are presented with appropriate corrections based on calibration data for each transducer.

Available experimental data are “Good” defined as follows. “Good” means that the type of data has been reviewed manually, and is presumed to lie within the range and uncertainty values of the instruments based on the design specification which are published in the reference [7]. However, certain measurements may be affected by various extraneous factors such as flow velocity, flow regime, and wall effects. **Table A-1** shows the list of available experimental data qualified as “Good” for LSTF SB-PV-03 (Run ID named to be SP3).

Table 3-1 Specified initial steady state and boundary conditions (1/2)

Core	Initial core power	10 MW
	Radial core power profile	Case 3
	Axial core power profile	9-step chopped cosine, peaking factor = 1.495
Primary Loops	Initial hot leg fluid temperatures	598.1 K
	Initial cold leg fluid temperatures	562.4 K
	Initial mass flow rate	24.3 kg/s / loop
	Initial downcomer-to-hot leg bypass	0.049 kg/s / loop
Pressurizer (PZR)	Initial pressure	15.5 MPa
	Initial liquid level	7.2 m
	Inner-diameter of relief valve orifice	6.83 mm
	Relief valve open / closure	Primary pressure = 16.20 / 16.07 MPa
	Inner-diameter of safety valve orifice	14.4 mm
	Safety valve open / closure	Primary pressure = 17.26 / 17.06 MPa
Steam Generators (SGs)	Initial secondary-side pressure	7.3 MPa
	Initial secondary-side liquid level	10.3 m
	Initial main steam flow rate	2.74 kg/s
	Initial main feedwater flow rate	2.74 kg/s
	Main feedwater temperature	495.2 K
	Inner-diameter of relief valve orifice	16.2 mm
	Relief valve open / closure	SG secondary-side pressure = 8.03 / 7.82 MPa
	Inner-diameter of safety valve orifice	26.6 mm
	Safety valve open / closure	SG secondary-side pressure = 8.68 / 7.69 MPa

Table 3-1 Specified initial steady state and boundary conditions (2/2)

Break

Location	Pressure vessel lower plenum (see Fig. 3-1)
Type	Sharp-edge orifice
Inner-diameter of orifice	4.6 mm

ECCS

High pressure injection (HPI) system		Not actuated
Accumulator (ACC) system	Initiation of system	Primary pressure = 4.51 MPa
	Water temperature	320 K
	Initial water level above tank bottom in both loops	6.8 m
	Standpipe level above tank bottom in both loops	5.22 m
	Initial water volume above standpipe in both loops	1.12 m ³ *
	Initial gas volume in both loops	0.46 m ³
	Cross-sectional flow area above standpipe	0.7085 m ²
	Orifice diameter (d) in loops with / without PZR	38.2 mm / 35.0 mm
	Connecting pipe diameter (D) in both loops	97.1 mm
	Contraction ratio (d/D) in loops with / without PZR	0.393 / 0.360
Low pressure injection (LPI) system	Coolant injection flow rate ratio of loop with PZR to loop without PZR	1:1
	Injection location	Cold legs in both loops
	Initiation of system	Pressure vessel lower plenum pressure = 1.24 MPa
	Q-H pump characteristic	See Table 3-5
	Water temperature	310 K
	Coolant injection flow rate ratio of loop with PZR to loop without PZR	1:1
	Injection location	Same with ACC system

* $(6.8 - 5.22 \text{ [m]}) \times 0.7085 \text{ [m}^2\text{]} \approx 1.12 \text{ [m}^3\text{]}$

LSTF Core Protection System Logic

Control of core power to	Maximum cladding surface temperature reaches
75%	908 K
50%	918 K
25%	919 K
10%	920 K

Table 3-2 Specified control logic, operation set points, and conditions

Event	Condition
Break	Time zero
Generation of scram signal	Primary pressure = 12.97 MPa
PZR proportional heater off	Generation of scram signal or PZR liquid level < 2.3 m
PZR backup heater off	PZR liquid level < 2.3 m
Initiation of core power decay curve simulation	Generation of scram signal
Initiation of primary coolant pump coastdown	Generation of scram signal
Closure of SG main steam stop valve	Generation of scram signal
Manual closure of SG main steam isolation valves	Generation of scram signal
Termination of SG main feedwater	Generation of scram signal
Generation of safety injection (SI) signal	Primary pressure = 12.27 MPa
Initiation of secondary-side depressurization of both SGs as AM action to achieve depressurization rate of 55 K/h in primary system	10 min after SI signal generation
Initiation of auxiliary feedwater injection into secondary-side of both SGs	With some delay after initiation of AM action
Initiation of ACC system in both loops	Primary pressure = 4.51 MPa
Initiation of LPI system in both loops	Pressure vessel lower plenum pressure = 1.24 MPa

Table 3-3 Specified pump rotation speed ratio after scram signal

Time (s)	Rotation Speed Ratio	Time (s)	Rotation Speed Ratio	Time (s)	Rotation Speed Ratio
0	1.000	30	0.280	80	0.125
2	0.850	40	0.220	90	0.110
5	0.730	50	0.185	100	0.100
10	0.540	60	0.160	250	0.000
20	0.370	70	0.140		

Table 3-4 Pre-determined core power decay curve after scram signal

Time (s)	Power (MW)	Time (s)	Power (MW)	Time (s)	Power (MW)
0	10	150	2.589	2000	1.404
18	10	200	2.429	3000	1.262
20	8.316	300	2.246	4000	1.169
30	5.532	400	2.166	5000	1.103
40	4.670	500	2.079	6000	1.052
50	4.072	600	1.998	8000	0.980
60	3.704	800	1.743	10000	0.929
80	3.209	1000	1.653		
100	2.929	1500	1.508		

Table 3-5 Pressure vessel lower plenum pressure versus LPI flow rate

Pressure vessel lower plenum pressure (MPa)	LPI flow rate in loop with PZR (kg/s)	LPI flow rate in loop without PZR (kg/s)
1.24	0.76	0.76
1.20	1.07	1.07
1.15	1.41	1.41
1.10	1.70	1.70
1.05	1.94	1.94
1.0	2.15	2.15
0.9	2.49	2.49
0.8	2.76	2.76

Table 3-6 Pre-determined secondary pressure drop curve after AM action onset

Time (s)	Pressure (MPa)	Time (s)	Pressure (MPa)	Time (s)	Pressure (MPa)
0	7.82	3600	3.24	8100	0.77
60	7.7	4500	2.5	9000	0.54
900	6.35	5400	1.9	10800	0.25
1800	5.11	6030	1.46		
2700	4.06	7200	1.07		

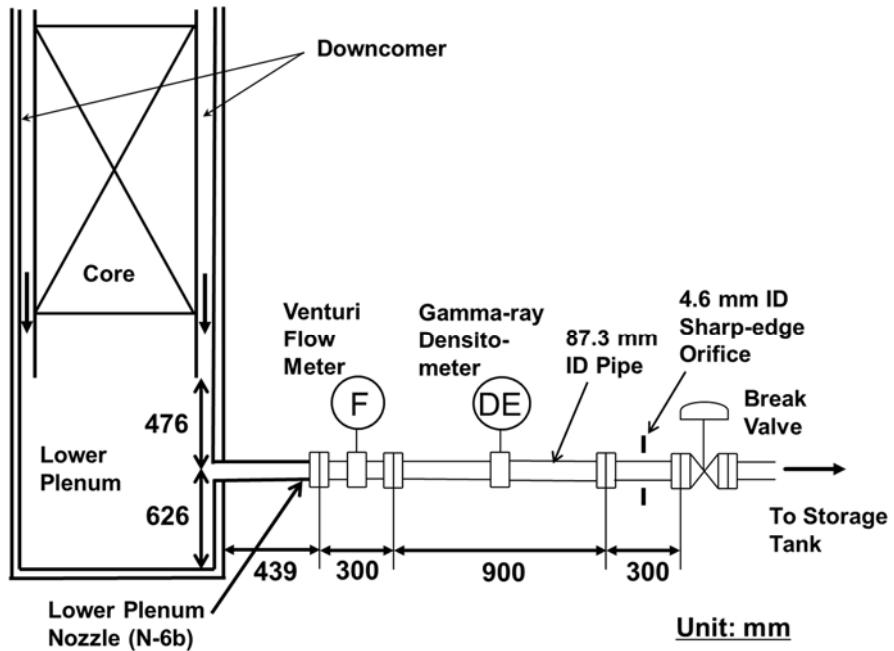
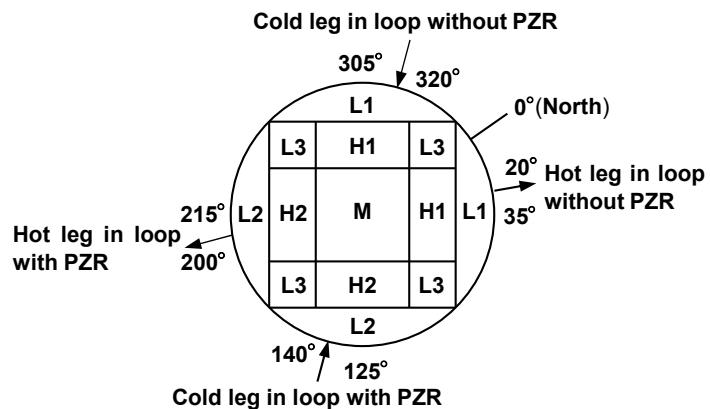


Fig. 3-1 Configuration of break unit



Heater Rod Zone	No. of Heater Rods	Radial Profile = Case 3	
		Peaking Factor	Heater Power
L1 (low-power)	144	0.66	$9.69\text{kW} \times 0.66 \times 144 = 921\text{kW}$
L2 (low-power)	144		$9.69\text{kW} \times 0.66 \times 144 = 921\text{kW}$
L3 (low-power)	180		$9.69\text{kW} \times 0.66 \times 180 = 1151\text{kW}$
H1 (high-power)	180	1.51	$9.69\text{kW} \times 1.51 \times 180 = 2634\text{kW}$
H2 (high-power)	180		$9.69\text{kW} \times 1.51 \times 180 = 2634\text{kW}$
M (mean-power)	180	1.00	$9.69\text{kW} \times 1.00 \times 180 = 1744\text{kW}$

Fig. 3-2 Core heater rod zone and heater power

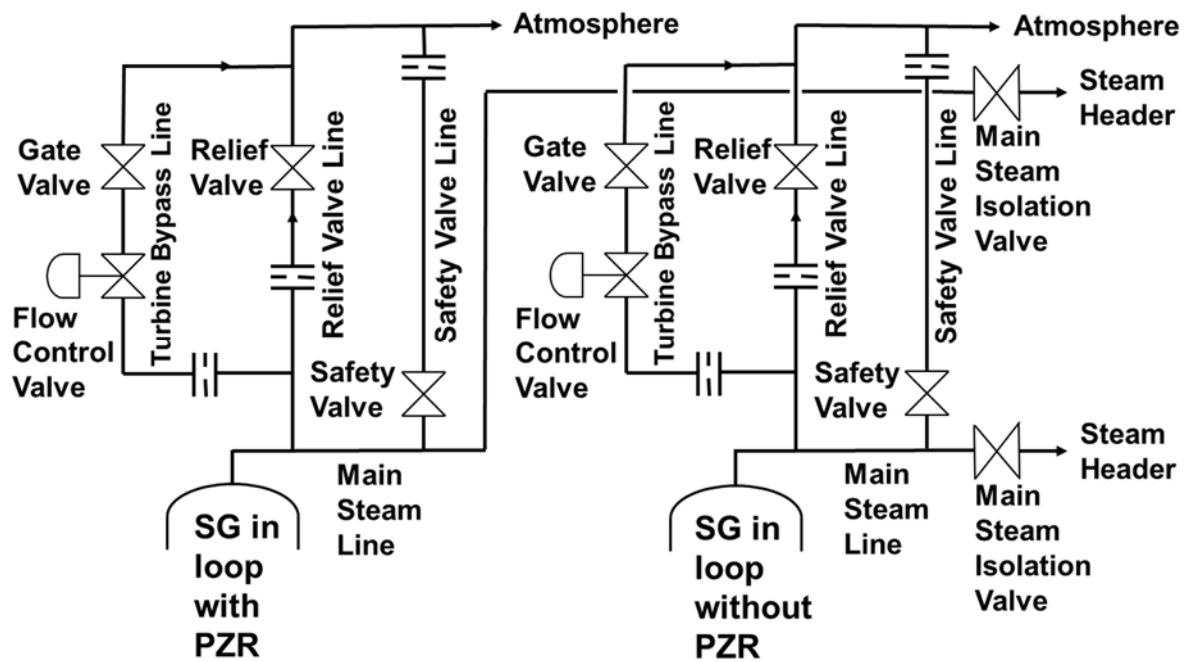


Fig. 3-3 Schematic view of SG steam line

4. Experimental Results

4.1 Initial and Boundary Conditions

Initial steady state conditions achieved in the experiment were in reasonable agreement with the specified values, as mentioned in **Table 4-1**. The measured values indicated are those averaged for the last 60 s prior to the onset of the break. Initial SG re-circulation ratio estimated from the measured flow rates in the downcomer and SG main steam line was 6.5 in loop with PZR (loop-A) and 6.3 in loop without PZR (loop-B). **Table 4-2** shows the chronology of major events until 9935 s when the break valve was closed.

4.2 Thermal-hydraulic Responses Concerning Boundary Conditions

As presented in **Fig. 4-1**, the core power began to decay at 270 s. The core power was automatically reduced by the LSTF core protection system at 9180 s when the maximum cladding surface temperature exceeded 908 K (to be depicted in **Fig. 4-24**). The core power was controlled to 10% of the pre-determined value [22] at 9198 s when the maximum cladding surface temperature reached 920 K. The core power turned off at 9870 s.

As noted in **Fig. 4-2**, the power values of the PZR proportional and backup heaters were initially kept constant at 2.8 kW and 33.7 kW, respectively. After the break, they began to increase up to 8.2 kW and 87.2 kW, respectively. The PZR proportional and backup heater powers turned off by 183 s and 184 s, respectively, because the PZR liquid level became below 2.3 m (as set out in **Fig. 4-3**). The PZR liquid level degraded immediately and monotonously after the break. The PZR became voided by 330 s.

Figures 4-4 and 4-5 show the primary coolant pump rotation speed and the primary loop mass flow rate measured by using a venturi flow meter at each primary coolant pump suction leg, respectively. The pump rotation speed started an increase simultaneously with the break, and attained about 1550 rpm in 4 s. The pump coastdown initiated at 252 s, following the scram signal. The pump rotation speed reduced thereafter, which led to a decrease in the primary loop mass flow rate. The primary loop was in two-phase natural circulation following the primary coolant pumps stop. The primary loop mass flow rate gradually declined after water column drained in instrumented SG U-tubes in non-uniform manner (to be plotted in **Figs. 4-42 through 4-45**). The primary loop mass flow rate oscillated during intermittent ACC coolant injection periods (to be presented in **Fig. 4-13**). After the termination of the ACC coolant injection, the primary loop mass flow rate became close to zero, suggesting that reflux condensation should occur in the SG U-tubes (to be noted in **Figs. 4-42 through 4-45**).

Figures 4-6 and 4-7 show the SG main steam and feedwater flow rates, respectively. The SG main steam was terminated by the closure of the SG main steam stop valve [Valve No.;

AOV-220] (see p.113 in Ref. [7]) at 250 s concurrently with the scram signal. The main feedwater injection into the secondary-side of both SGs was ended by the closure of the SG main feedwater line valve [Valve No.; AOV-260] (see p.113 in Ref. [7]) at 253 s, following the scram signal. The main steam isolation valves of both SGs [Valve No.; AOV-170 and AOV-200] (see pp.111-112 in Ref. [7]) were manually closed at 280 s, following the SG main steam stop valve closure.

Figures 4-8 and 4-9 show the SG secondary-side collapsed liquid level and AFW flow rate, respectively. The SG secondary-side liquid level began to rise after the main steam stop valve closure and the main feedwater termination (**Figs. 4-6 and 4-7**). This seems to be attributed to the decrease in the net upward steam flow through the SG boiling section. The SG secondary-side liquid level gradually dropped with some fluctuation owing to cyclic opening of the SG relief valves (to be indicated in **Fig. 4-17**). During the time period other than the AFW pump operation, flow meter (see pp.257-258 in Ref. [7]) for AFW counts flow rate in the return line from the AFW pump to the refueling water storage tank. The AFW injection into the secondary-side of both SGs was initiated at 950 s following the AM action onset, and terminated at 2750 s. The AFW flow rate into the SG secondary-side was about 0.59-0.60 kg/s in loop with PZR and about 0.72-0.73 kg/s in loop without PZR. The SG secondary-side liquid level steadily increased after the AFW injection start. The reason is that the AFW flow rate was greater than the steam discharge flow rate through the SG flow control valve employed for the AM action. The SG secondary-side liquid level turned to fall after the end of the AFW injection. The liquid level difference in the secondary-side of both SGs was the largest at the AFW injection termination, and then descended little by little. The SG secondary-side was not emptied of liquid throughout the experiment, which produced continuous heat removal from the SG secondary-side system.

The break unit as illustrated in **Fig. 3-1** is located at about 1.7 m below the core bottom (see p.194 in Ref. [7]). **Figure 4-10** shows the time-integrated break flow evaluated from the liquid level increase in the storage tank, and the break flow rate derived as the differential of the time-integrated break flow. **Figure 4-11** depicts the lower plenum fluid temperatures at 0.3 m, 1.5 m, 1.8 m, and 2.1 m below the core bottom. The break flow rate reduced roughly stepwise. The break flow turned from single-phase liquid to two-phase flow at around 1300 s because the lower plenum fluid became saturated. This brought about a decrease in the break flow rate. The break flow returned to single-phase liquid at around 9300 s after the actuation of the LPI system (to be mentioned in **Fig. 4-15**).

Figure 4-12 shows the liquid level in the ACC tank. **Figures 4-13 and 4-14** give the coolant injection flow rate from the ACC tank derived from the liquid level history in the ACC tank. Another name for the ACC tank in loop with PZR is ACC-cold tank, while that for the ACC tank

in loop without PZR is ACC-hot tank. The coolant injection temperature is 320 K in both the ACC-cold tank and ACC-hot tank, but the heater capacity of the ACC-hot tank is designed to be twice that of the ACC-cold tank. When the primary pressure lowered to 4.53 MPa, the ACC system initiated at 3230 s in loop with PZR and 3250 s in loop without PZR. The initial water level above the ACC tank bottom was 6.72 m in loop with PZR and 6.73 m in loop without PZR. The final water level above the ACC tank bottom was 5.28 m in loop with PZR and 5.24 m in loop without PZR. When the primary pressure dropped to about 1.6 MPa, the ACC system ended at 6740 s in loop with PZR and 7100 s in loop without PZR. The coolant injection periods from the ACC-cold tank included around 3230-5800 s, 5850-5910 s, 5960-6030 s, 6060-6150 s, 6200-6450 s, and 6590-6740 s. On the other hand, the coolant injection periods from the ACC-hot tank contained around 3250-5800 s, 5850-5910 s, 5960-6030 s, 6060-6150 s, 6200-6450 s, 6590-6830 s, and 7030-7100 s.

When the pressure vessel lower plenum pressure reduced to 1.23 MPa, coolant was fed from the LPI system into both cold legs at 9280 s in accordance with the discharge pressure of the LPI pump, as set out in **Fig. 4-15**.

4.3 Transient Thermal-hydraulic Responses

Figure 4-16 shows the primary and secondary pressures. The primary pressure initiated a decrease at time zero when the break valve was opened. Small size of break resulted in a slow primary depressurization. When the primary pressure lowered to 12.97 MPa, the scram signal was generated at 250 s. This gave rise to the closure of the SG main steam stop valve, the coastdown of the primary coolant pumps, the termination of the SG main feedwater, and the manual closure of the SG main steam isolation valves. The SG secondary-side pressure rapidly raised up to about 8 MPa after the closure of the SG main steam stop valve. The SG secondary-side pressure fluctuated between 8.03 MPa and 7.82 MPa. This is because the relief valves of both SGs cyclically opened during the time period of around 200-950 s, as presented in **Fig. 4-17**. When the primary pressure dropped to 12.27 MPa, the SI signal was obtained. Secondary-side depressurization of both SGs as the AM action to achieve the primary depressurization rate of 55 K/h was undertaken at 945 s: 10 min after the SI signal generation, and went on afterwards. The primary pressure fell following the SG secondary-side pressure after the AM action onset, which led to the activation of the ACC system (**Figs. 4-13 and 4-14**). When the primary pressure declined to about 1.6 MPa, the ACC system was completed. As plotted in **Fig. 4-16**, the primary depressurization rate decreased and thus the pressure difference increased between the primary and SG secondary sides after around 7200 s. This was induced by the accumulation of nitrogen gas in the SG U-tubes (to be described in **Section 4.3.3**). Nitrogen gas enters cold legs through the ECCS nozzles first, and then migrates to hot legs through downcomer and bypass lines simulating hot leg leakage that

connect the downcomer to the hot legs (as denoted in **Chapter 2**). Nitrogen gas finally flows into the SG U-tubes. The nitrogen gas accumulation implies degradation in the condensation heat transfer in the SG U-tubes. After the initiation of the automatic core power reduction at 9180 s, the LPI system was actuated in both loop at 9280 s (**Fig. 4-15**).

The major components of the LSTF, whose thermal-hydraulic responses are described below, can be seen in **Fig. 2-2**.

4.3.1 Thermal-hydraulic Responses in Pressure Vessel

Liquid level behaviors in upper plenum and core

The upper plenum collapsed liquid level is shown in **Figs. 4-18 and 4-19**, while the core collapsed liquid level is given in **Figs. 4-20 and 4-21**. A small drop began in the core liquid level at 4710 s after the upper plenum became empty of liquid at 4670 s. The cladding surface temperature increase at Position 9 (near the core top) remained at about 8 K at around 5450-5785 s (to be depicted in **Fig. 4-25**). The liquid levels recovered in the upper plenum and core in oscillative way because of the periodic coolant injection from the ACC system into both cold legs (**Figs. 4-13 and 4-14**). The upper plenum became voided again at 8300 s, following the primary coolant inventory depletion after the termination of the ACC system. A significant depression started in the core liquid level at 8340 s. Core uncovering thus took place owing to core boil-off at 8575 s (to be noted in **Figs. 4-24, 4-26, and 4-28**). The core liquid level abruptly increased after the actuation of the LPI system at 9280 s (**Fig. 4-15**). The entire core was quenched by 9682 s (to be presented in **Figs. 4-24 and 4-26**). The liquid level recovered in the upper plenum at around 9730 s.

Responses of core exit temperature (CET) and cladding surface temperature of simulated fuel rods

Figures 4-22 and 4-23 show the typical CETs at the center [Tag Name; TE-EX040-B21-UCP, -B23-UCP], peripheral region [TE-EX040-B11-UCP, -B14-UCP], and outer region [TE-EX040-B01-UCP, -B02-UCP] of upper core plate (UCP). The representative CETs account for the first and second highest CETs at each region of UCP. Thickness of UCP is 76.2 mm. The core exit thermocouples are installed 13 mm above the upper surface of UCP. The vertical and radial measurement positions of the CETs are presented in Ref. [7] (pp.377-378) and Ref. [7] (pp.385-386), respectively. The CETs were maintained saturated until 9180 s even after the core uncovering began partly at 8575 s (to be set out in **Fig. 4-28**). This was ascribed to condensate falling from hot leg nozzles (see p.196 in Ref. [7]). Subsequently, the CETs started to rise. The CET was elevated to 510 K, which was not above the initial value. Therefore, the CET increase remained at about 45 K. The CET, which recorded 510 K, was seen at the center [TE-EX040-B23-UCP] of UCP located above the B23 rod bundle at 9680 s. The CETs were kept saturated again after around 9725 s.

Figures 4-24 through 4-26 show the typical cladding surface temperatures of the simulated fuel rods at Positions 9 through 1. Positions 9, 8, 7, 6, 5, 4, 3, 2, and 1 are placed at 3.610 m, 3.048 m, 2.642 m, 2.236 m, 1.830 m, 1.242 m, 1.108 m, 0.612 m, and 0.050 m, respectively, above the core bottom (see p.222 in Ref. [7]). The representative cladding surface temperatures stand for the cladding surface temperatures of the rod bundles where the highest temperature emerges at each position. The arrangement of high-, mean-, and low-power rod bundles is presented in Ref. [7] (p.228). The cladding surface temperatures were affected by the axial core power profile (i.e., the 9-step chopped cosine) and the core liquid level behavior (**Figs. 4-20 and 4-21**). No cladding surface temperatures at Positions 2 and 1 rose. During the core uncover period the cladding surface temperatures at Positions 9, 8, 7, 6, 5, 4, and 3 began to increase at 8695 s, 8575 s, 8655 s, 8755 s, 8860 s, 8960 s, and 9110 s, respectively. The core power was controlled to 10% of the pre-determined value at the maximum cladding surface temperature of 920 K (**Fig. 4-1**). Just after that, the peak cladding temperature of 925 K was observed at Position 6. The maximum cladding surface temperatures at Positions 9, 8, 7, 5, 4, and 3 were 629 K, 779 K, 913 K, 862 K, 738 K, and 568 K, respectively. They reached at 9595 s, 9560 s, 9200 s, 9200 s, 9200 s, and 9205 s, respectively. The whole core was quenched by 9682 s due to the core liquid level recovery because of coolant injection from the LPI system into both cold legs.

Figures 4-27, 4-28, 4-29, 4-30, 4-31, 4-32, and 4-33 show the distributions of the cladding surface temperatures of the simulated fuel rods at Positions 9, 8, 7, 6, 5, 4, and 3, respectively. At Position 9 (**Fig. 4-27**), the cladding surface temperatures rose earlier in the order of the B17, B22, and B08 rod bundles. The maximum cladding surface temperature was higher in the order of the B17, B22, and B08 rod bundles. The quenching of the cladding surfaces of the B08, B17, and B22 rod bundles was simultaneous.

At Position 8 (**Fig. 4-28**), the cladding surface temperature increases were earlier in the order of the B13, B10, and B03 rod bundles. The maximum cladding surface temperature was higher in the order of the B13, B10, and B03 rod bundles. The cladding surfaces of B03, B10, and B13 rod bundles were quenched at the same time.

At Position 7 (**Fig. 4-29**), the cladding surface temperature rises of the B13 and B17 were simultaneous, and were earlier than those of other rod bundles. The cladding surface temperatures increased faster in the order of the B10, B08, and B03 rod bundles. The maximum cladding surface temperature was higher in the order of the B13, B17, B22, B10, B08, and B03 rod bundles. The quenching of the cladding surface of the B03 rod bundle was a little earlier than that of other rod bundles.

At Position 6 (**Fig. 4-30**), the cladding surface temperatures increased earlier in the order of the B13, B10, and B03 rod bundles. The maximum cladding surface temperature was higher

in the order of the B13, B10, and B03 rod bundles. The cladding surfaces of B03, B10, and B13 rod bundles were quenched at the same time.

At Position 5 (**Fig. 4-31**), the cladding surface temperatures of the B08, B17, and B22 rod bundles rose at the same time. The maximum cladding surface temperature was higher in the order of the B17, B22, and B08 rod bundles. The quenching of the cladding surfaces of the B08, B17, and B22 rod bundles was simultaneous.

At Position 4 (**Fig. 4-32**), the cladding surface temperatures of the B03, B10, and B13 rod bundles increased simultaneously. The maximum cladding surface temperature was higher in the order of the B13, B10, and B03 rod bundles. The cladding surfaces of the B03, B10, and B13 rod bundles were quenched at the same time.

At Position 3 (**Fig. 4-33**), the cladding surface temperatures of the B08, B17, and B22 rod bundles rose at the same time. The maximum cladding surface temperature was higher in the order of the B17, B22, and B08 rod bundles. The quenching of the cladding surfaces of the B08, B17, and B22 rod bundles was simultaneous.

Coolant behaviors in pressure vessel

Figure 4-34 shows the downcomer collapsed liquid level. The liquid level formed in the downcomer at 560 s. Similar to the core liquid level after the emptying of the upper plenum, a small reduction appeared in the downcomer liquid level (**Figs. 4-20 and 4-21**). The oscillating recovery was seen in the downcomer liquid level resulting from the periodic coolant injection from the ACC system into both cold legs (**Figs. 4-13 and 4-14**). A notable drop initiated in the downcomer liquid level at 7810 s. The downcomer liquid level recovered at 9320 s with the coolant injection from the LPI system into the cold legs (**Fig. 4-15**).

Figure 4-35 shows the upper-head collapsed liquid level. An abrupt depression started in the upper-head liquid level at 430 s. After 940 s the upper-head liquid level was maintained at around 0.07-0.12 m, except for the intermittent ACC coolant injection periods.

4.3.2 Thermal-hydraulic Responses in Primary Loops

Figures 4-36 and 4-37 show the liquid levels and fluid temperatures in the hot legs, respectively. The hot leg liquid level is estimated from the fluid densities measured by use of a three-beam gamma-ray densitometer. The hot leg fluid became saturated at 360 s almost concurrently with the liquid level formation in the hot leg. The liquid level in the hot leg with an inner-diameter of 0.207 m was kept at around 0.12-0.14 m in both loops at around 1390-3650 s. The hot leg liquid level became close to zero at 4000 s after an abrupt decrease initiated in the upper plenum liquid level at 3960 s (**Figs. 4-18 and 4-19**). The liquid level recovery was oscillative in the hot leg with the periodic ACC coolant injection into the cold leg (**Figs. 4-13**

and 4-14). The hot leg fluid temperature indicated superheating after around 8800 s because the hot leg liquid level became close to zero again.

Figure 4-38 and 4-39 show the liquid levels and fluid temperatures in the cold legs, respectively. The cold leg liquid level is evaluated from the fluid densities measured by means of a three-beam gamma-ray densitometer. The cold legs were filled with subcooled water until the primary pressure dropped to around 10 MPa (**Fig. 4-16**). The liquid levels formed in the cold legs at 680 s. When the liquid level in the cold leg with an inner-diameter of 0.207 m depressed to around 0.16 m, the cold leg fluid became saturated. The cold leg liquid level dropped to around 0.02 m thereafter. The oscillating recovery was observed in the cold leg liquid level with the intermittent ACC coolant injection into the cold leg (**Figs. 4-13 and 4-14**). The cold leg fluid temperature showed some superheating after around 7500 s especially in loop with PZR because the cold leg became empty of liquid. After the activation of the LPI system (**Fig. 4-15**), the recovery in the cold leg liquid level brought about the reduction in the cold leg fluid temperature.

Figures 4-40 and 4-41 show the collapsed liquid levels in the downflow-side and upflow-side of the crossover legs, respectively. The crossover leg downflow-side liquid level started to drop at 2380 s in loop with PZR and 2960 s in loop without PZR. A large decline laid in the crossover leg liquid level at 5730 s in the downflow-side and 5780 s in the upflow-side. During the intermittent ACC coolant injection periods (**Figs. 4-13 and 4-14**) the liquid levels in the downflow-side and upflow-side of the crossover legs decreased and increased, respectively.

4.3.3 Thermal-hydraulic Responses of Steam Generators

Figures 4-42 through 4-45 show the collapsed liquid levels in the instrumented SG U-tubes. The instrumented SG U-tubes designated as Tubes 1 and 6 are short tubes (Type 1; see p.267 in Ref. [7]), Tubes 2 and 5 are medium tubes (Type 5), and Tubes 3 and 4 are long tubes (Type 9). The liquid level in the upflow-side balanced that in the downflow-side for each SG U-tube. Non-uniform flow behaviors were seen in the SG U-tubes. The trend of the liquid level change was mostly similar for the two same-length SG U-tubes consisting of short, medium, and long tubes. The liquid level behaviors in the SG U-tubes were asymmetrical between two loops. The liquid level drop in the downflow-side and upflow-side of the SG U-tubes continued down to the crossover leg downflow-side (**Fig. 4-40**) and the SG inlet plenum (to be plotted in **Fig. 4-50**), respectively. The SG U-tubes were emptied of liquid the earliest in Tube 1 or Tube 6 and the latest in Tube 2.

To investigate the accumulating status of nitrogen gas in the SG U-tube, nitrogen gas mole fraction M in the steam-gas mixture is expressed by

$$M = \frac{P - P_s}{P} \quad (1)$$

where P is the total pressure of the steam-gas mixture that is represented by the PZR pressure (**Fig. 4-16**). P_s is the steam pressure that is equivalent to the steam saturation pressure based on the SG U-tube fluid temperature by assuming 100% relative humidity of saturated steam in the steam-gas mixture. The fluid temperatures in the downflow-side of the SG U-tubes at Position 1 (= at 0.811 m above the SG U-tube bottom) were chosen as the SG U-tube fluid temperatures for evaluating the nitrogen gas mole fraction. This selection considered that the SG U-tubes remained empty of liquid after the termination of the ACC system (**Figs. 4-42 through 4-45**). **Figures 4-46 and 4-47** show the fluid temperatures at Position 1 in the downflow-side of the SG U-tubes in loops with and without PZR, respectively, from 7200 s to 8800 s. Here, the fluid temperatures of Tube 4 (one of two long tubes) in loop with PZR and Tube 5 (one of two medium tubes) in loop without PZR were unavailable on account of the bad trends. The tendency of the fluid temperature variation among the SG U-tubes was similar, except for the long tube around 7550-7750 s. The SG U-tube fluid temperature was lower in loop without PZR than in loop with PZR. **Figures 4-48 and 4-49** show the estimated nitrogen gas mole fractions of the SG U-tubes in loops with and without PZR, respectively. The nitrogen gas mole fractions of the SG U-tubes exceeded zero at around 7200 s. The nitrogen gas mole fraction of the SG U-tube in loop without PZR was generally greater than that in loop with PZR. The reason is that the steam pressure P_s was lower in loop without PZR than in loop with PZR, relying on the SG U-tube fluid temperature. For example, the nitrogen gas mole fractions of the SG U-tubes in loops with and without PZR were evaluated to be approximately 0.35 and 0.38, respectively, at around 8600 s (i.e., at early stage of the cladding surface temperature rise).

Figure 4-50 shows the SG inlet plenum collapsed liquid level. A substantial depression initiated in the SG inlet plenum liquid level at 2400 s in loop with PZR and 2980 s in loop without PZR. The SG inlet plenum became empty of liquid at 3000 s in loop with PZR and 3640 s in loop without PZR. The SG inlet plenum liquid level recovered up to about 0.2 m at 7230-7360 s in loop with PZR and about 0.4 m at 7230-7560 s in loop without PZR. The liquid level behavior of the SG inlet plenum was in response to that of the hot leg (**Figs. 4-36 and 4-37**).

Figures 4-51 and 4-52 show the SG secondary-side fluid temperatures in loops with and without PZR, respectively. Positions 1, 3, 5, 7, and 9 are placed at 0.811 m, 2.101 m, 3.381 m, 5.941 m, and 8.501 m, respectively, above the SG U-tube bottom. The SG secondary-side fluid temperature is dependent on the SG secondary-side liquid level. The SG secondary-side fluid temperature at Position 1 was held saturated. The reason is that the SG secondary-side liquid level remained at around 0.8 m when the LPI system was actuated in both loops (**Figs. 4-8 and 4-15**). The SG secondary-side fluid temperatures at Positions 3, 5, 7, and 9 indicated

superheating after 9280 s, 8580 s, 7000 s, and 5400 s, respectively, in loop with PZR. By contrast, the SG secondary-side fluid temperatures at Positions 3, 5, 7, and 9 showed superheating after 6000 s, 8850 s, 9360 s, and 9340 s, respectively, in loop without PZR.

Table 4-1 Initial steady state conditions

Items	Tag Name (Loops with / without PZR)	Specified (Loops with / without PZR)	Measured *1 (Loops with / without PZR)
Pressure vessel			
Core power (MW)	WE270A-T	10.0±0.07	10.07
Downcomer-to-upper head bypass (%)	None	0.3	Not Measured
Primary loop			
Hot leg fluid temperature (K)	TE020C-HLA / TE160C-HLB	598.1±2.75	598.6 / 598.2
Cold leg fluid temperature (K)	TE070C-CLA / TE210C-CLB	562.4±2.75	563.8 / 563.4
Mass flow rate (kg/s / loop)	FE020A-LSA / FE160A-LSB	24.3±1.25	24.70 / 24.30
Downcomer-to-hot leg bypass (kg/s)	FE010-HLA / FE150-HLB	0.049±0.01	0.049 / 0.044
PZR			
Pressure (MPa)	PE300A-PR	15.5±0.108	15.53
Liquid level (m)	LE280-PR	7.2±0.25	7.27
Steam generator			
Secondary-side pressure (MPa)	PE430-SGA / PE450-SGB	7.3±0.054	7.37 / 7.40
Secondary-side liquid level (m)	LE430-SGA / LE450-SGB	10.3±0.38	10.23 / 10.21
Steam flow rate (kg/s)	FE440-SGA / FE480-SGB	2.74±0.10	2.67 / 2.59
Main feedwater flow rate (kg/s)	FE430-SGA / FE470-SGB	2.74±0.05	2.79 / 2.62
Main feedwater temperature (K)	TE430-SGA / TE470-SGB	495.2±2.63	495.4 / 495.2
Auxiliary feedwater temperature (K)	TE880-RWST	310±2.37	309.4
Accumulator system			
Pressure (MPa)	PE650-ACC / PE660-ACH	4.51±0.054	4.53 / 4.53
Temperature (K)	TE660-ACC / TE700-ACH	320±2.3 / 2.4	321.7 / 322.8
Water level above tank bottom (m) *2	LE650-ACC / LE660-ACH	6.8±0.12/0.15	6.72 / 6.73
Low pressure injection system			
Temperature (K)	TE840-PL	310±2.63	309.9

*1 Averaged for 60 s (-60 to 0 s)

*2 Distance from standpipe top to tank bottom is 5.22 m.

Table 4-2 Chronology of major events until break valve closure

Time (s)	Event
0	Break valve open, start of primary coolant pumps rotation speed increase (to 1550 rpm in 4 s)
250	Scram signal generation, closure of SG main steam stop valve
252	Initiation of coastdown of primary coolant pumps
253	Termination of SG main feedwater
270	Initiation of core power decay
280	Manual closure of SG main steam isolation valves
300-950	Cyclic opening of relief valves in both SGs
340	Safety injection signal generation
500	Stop of primary coolant pumps
945	Initiation of secondary-side of both SGs as AM action to achieve depressurization rate of 55 K/h in primary system
950	Initiation of AFW injection into secondary-side of both SGs
2750	Termination of AFW injection into secondary-side of both SGs
3230	Initiation of ACC system in loop with PZR
3250	Initiation of ACC system in loop without PZR
4670	Upper plenum became empty of liquid.
5450-5785	Cladding surface temperature increase at Position 9 remained at about 8 K.
6740	Termination of ACC system in loop with PZR
7100	Termination of ACC system in loop without PZR
7200	Start of degradation in primary depressurization after nitrogen gas inflow
8300	Upper plenum became empty of liquid again.
8575	Start of increase in cladding surface temperature at Position 8
9180	Initiation of automatic core power reduction at maximum cladding surface temperature of 908 K
9198	Core power was controlled to 10% of pre-determined value at maximum cladding surface temperature of 920 K.
9200	Peak cladding temperature of 925 K was observed at Position 6.
9280	Initiation of LPI system in both loops
9680	Core exit temperature, which recorded 510 K, was seen at center of upper core plate.
9682	Whole core quench
9870	Core power off
9935	Break valve closure

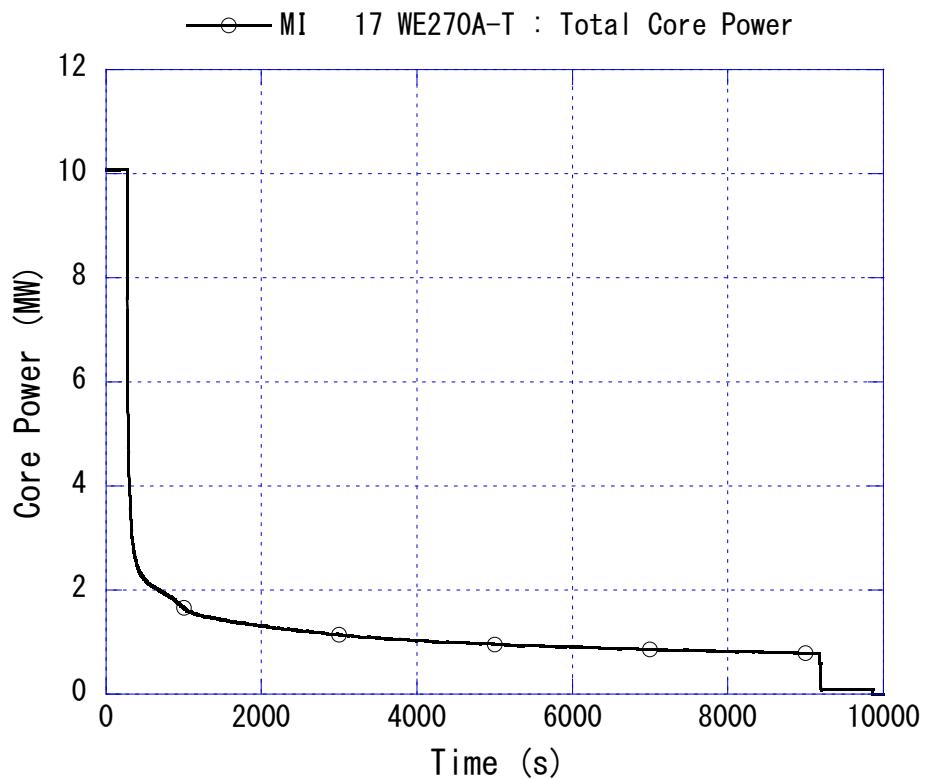


Fig. 4-1 Core power

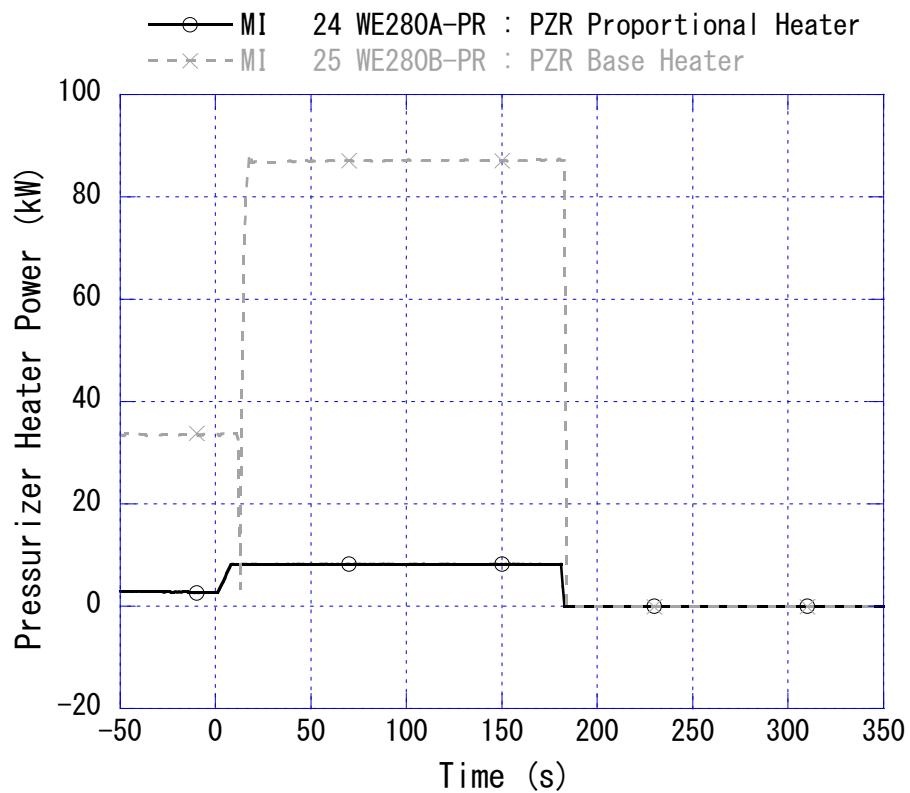


Fig. 4-2 Pressurizer heater power

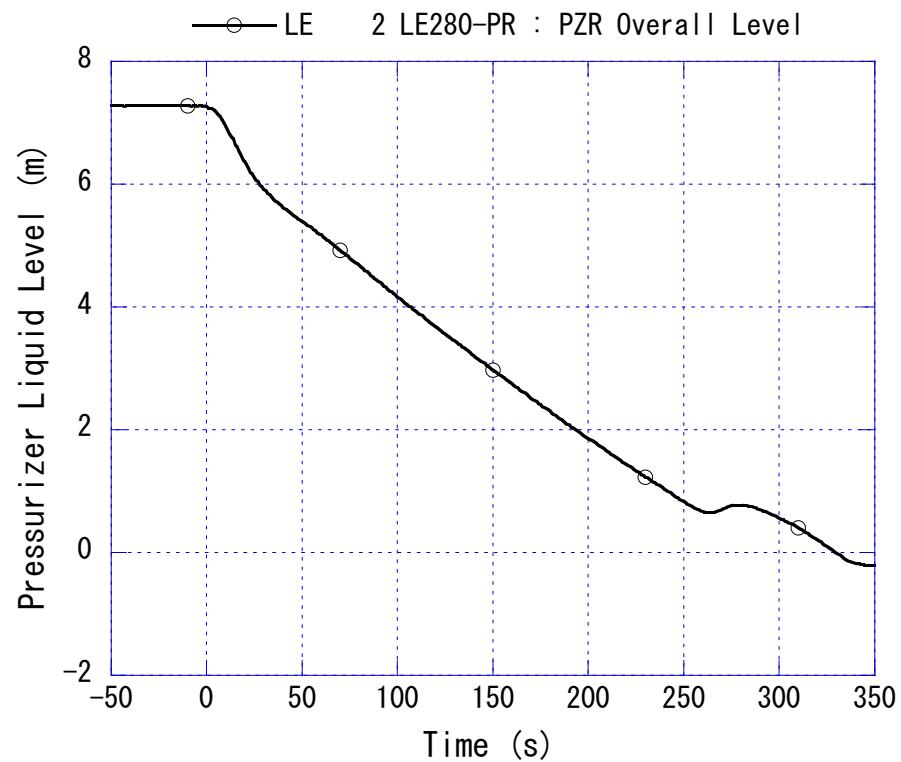


Fig. 4-3 Pressurizer liquid level

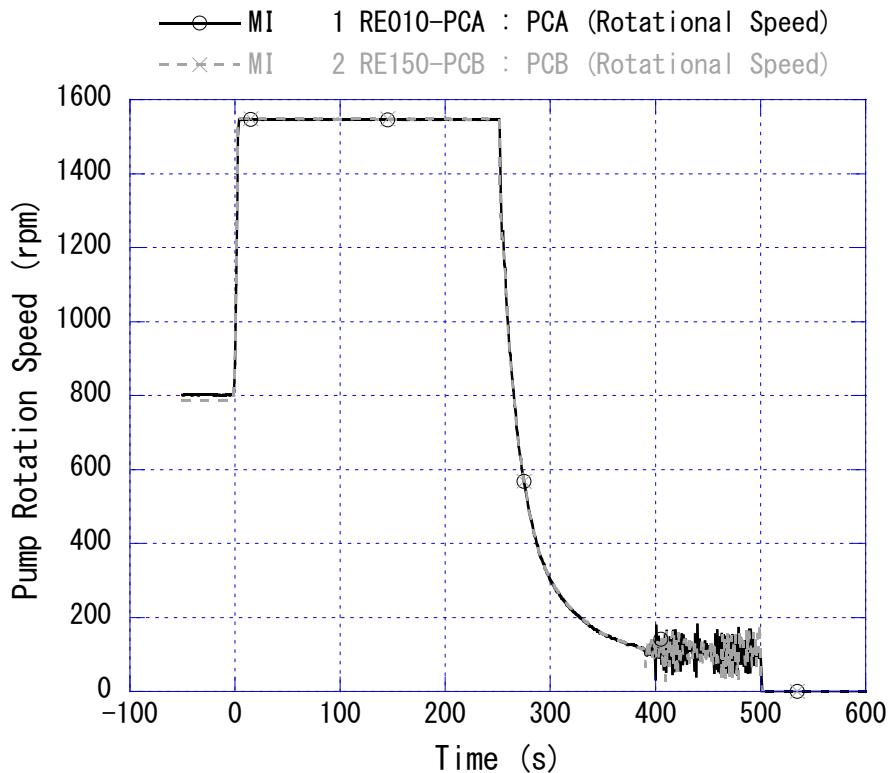


Fig. 4-4 Primary coolant pump rotation speed

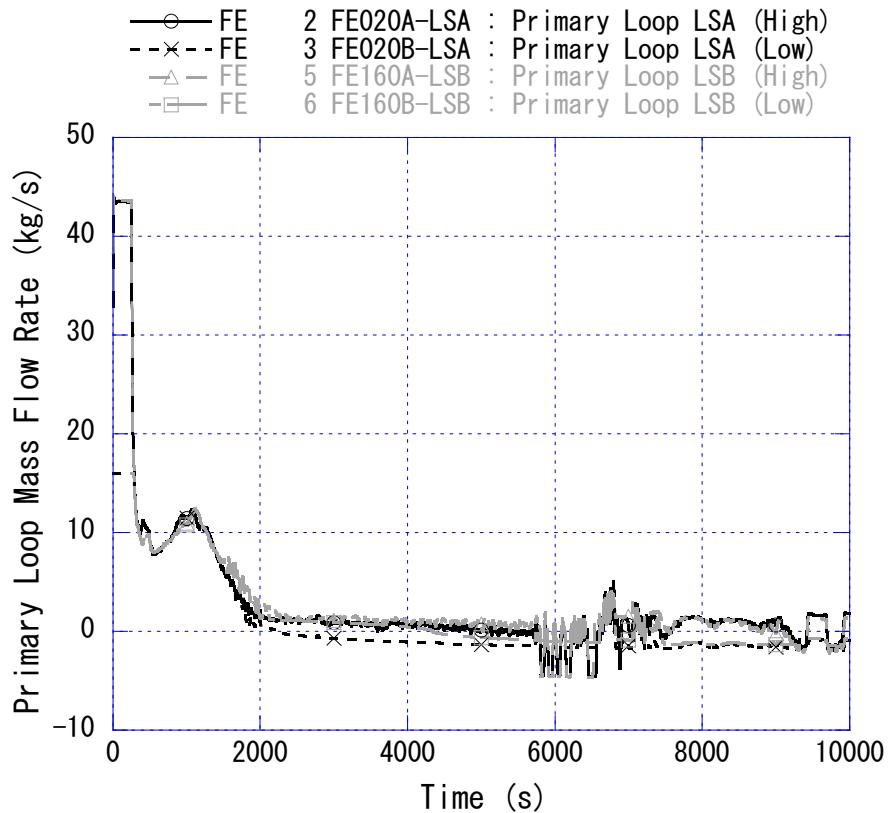


Fig. 4-5 Primary loop mass flow rate

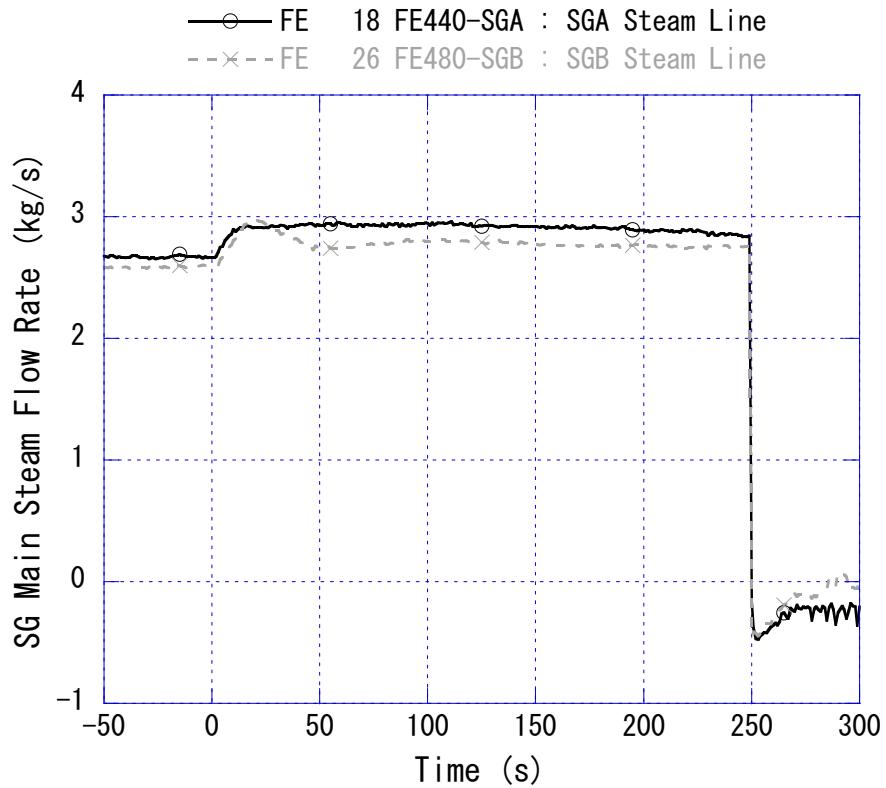


Fig. 4-6 SG main steam flow rate

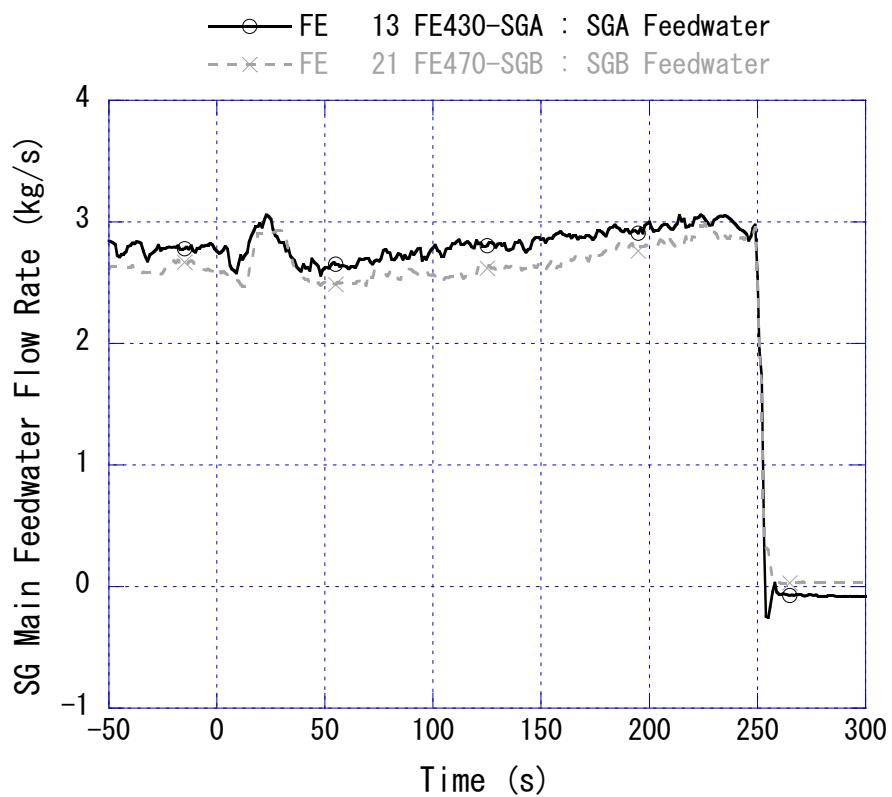


Fig. 4-7 SG main feedwater flow rate

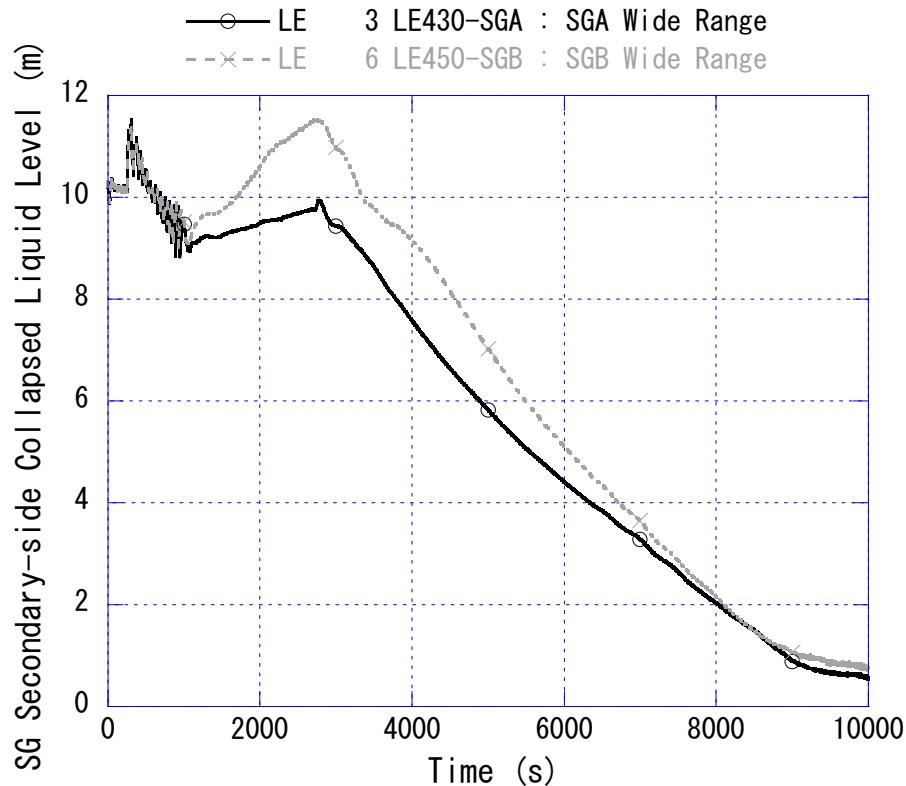


Fig. 4-8 SG secondary-side collapsed liquid level

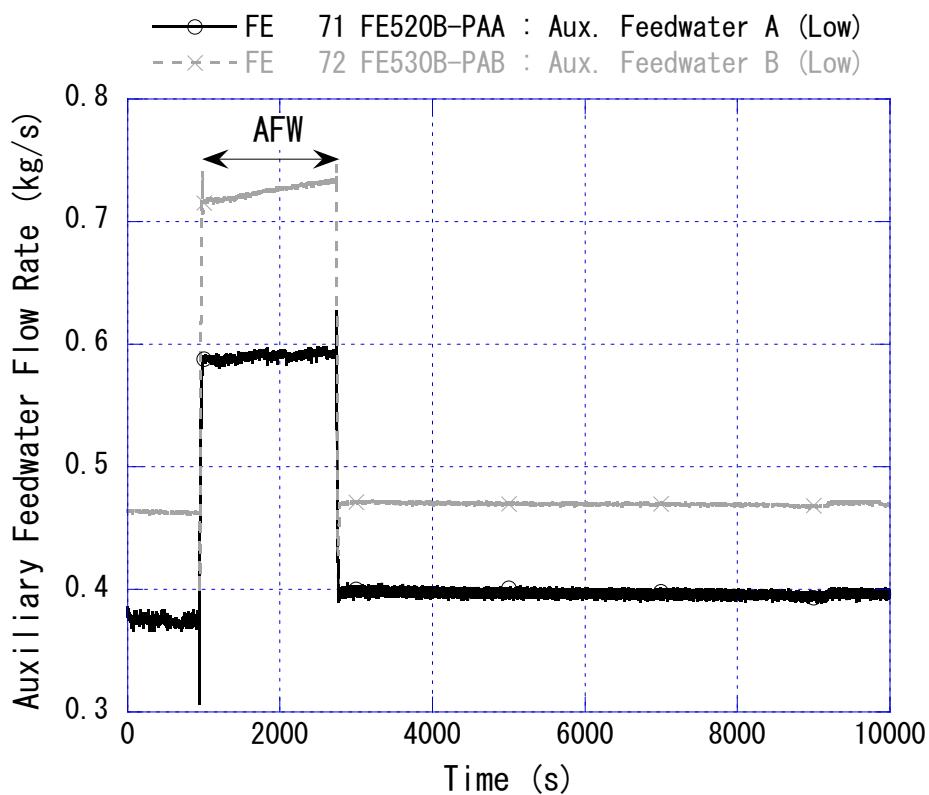


Fig. 4-9 Auxiliary feedwater flow rate

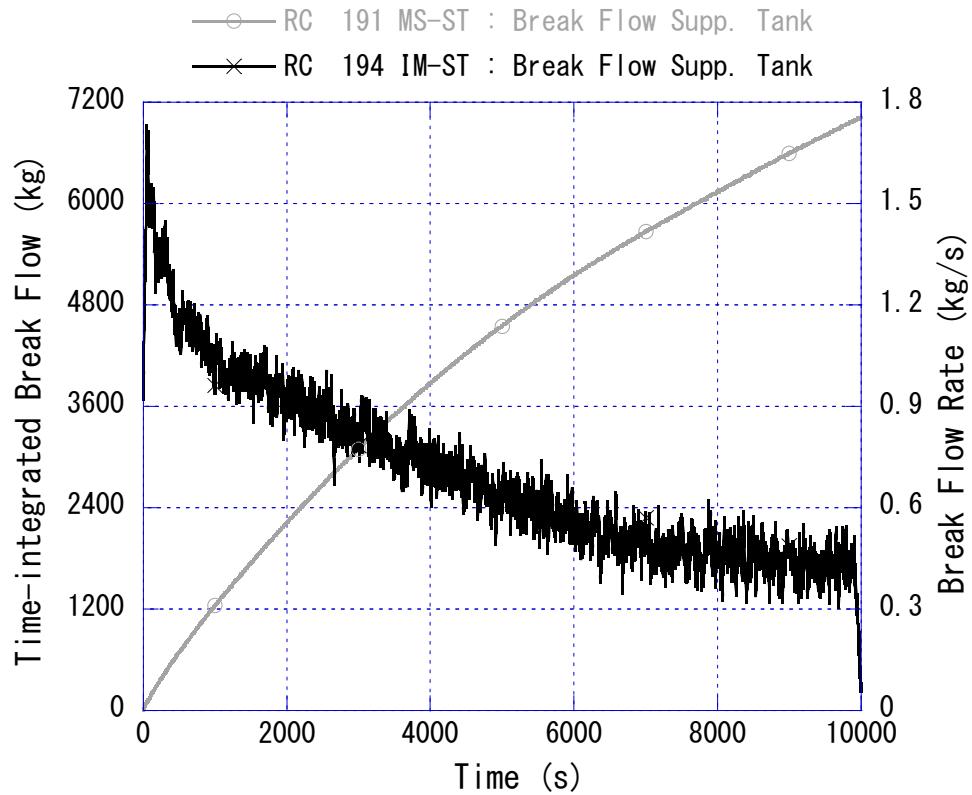


Fig. 4-10 Time-integrated break flow and break flow rate

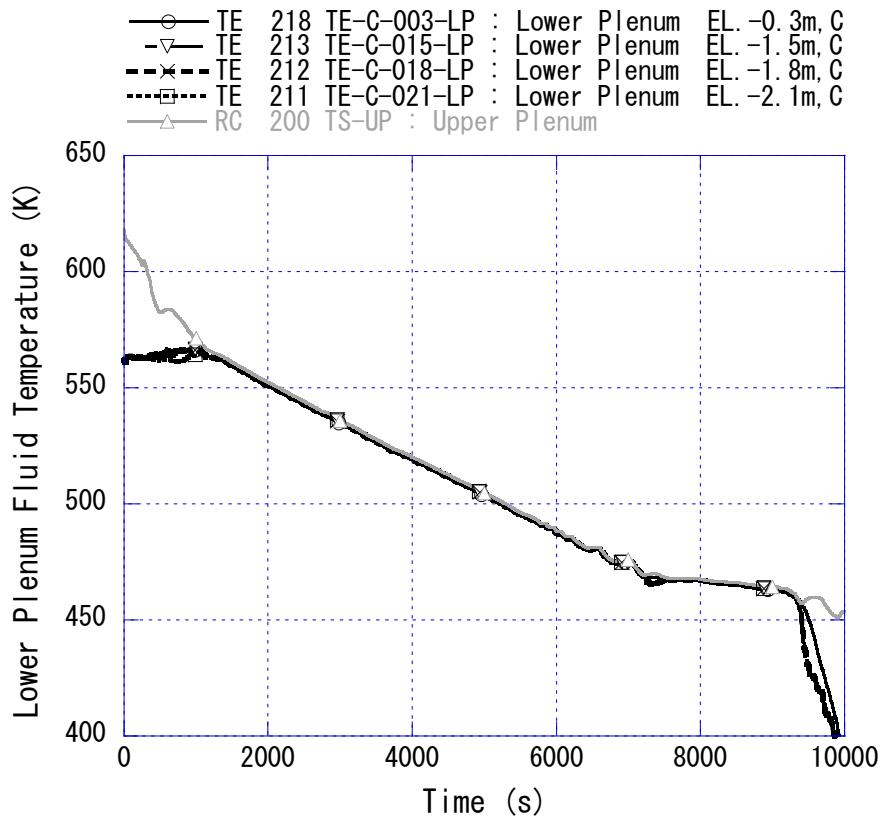


Fig. 4-11 Lower plenum fluid temperature

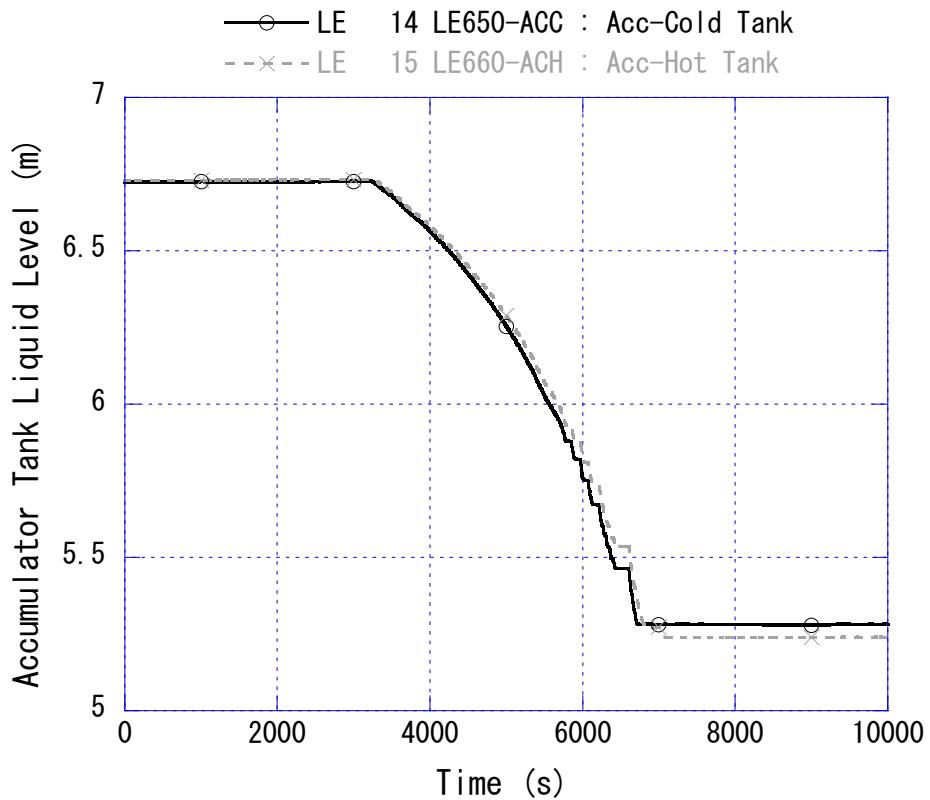


Fig. 4-12 Liquid level in accumulator tank

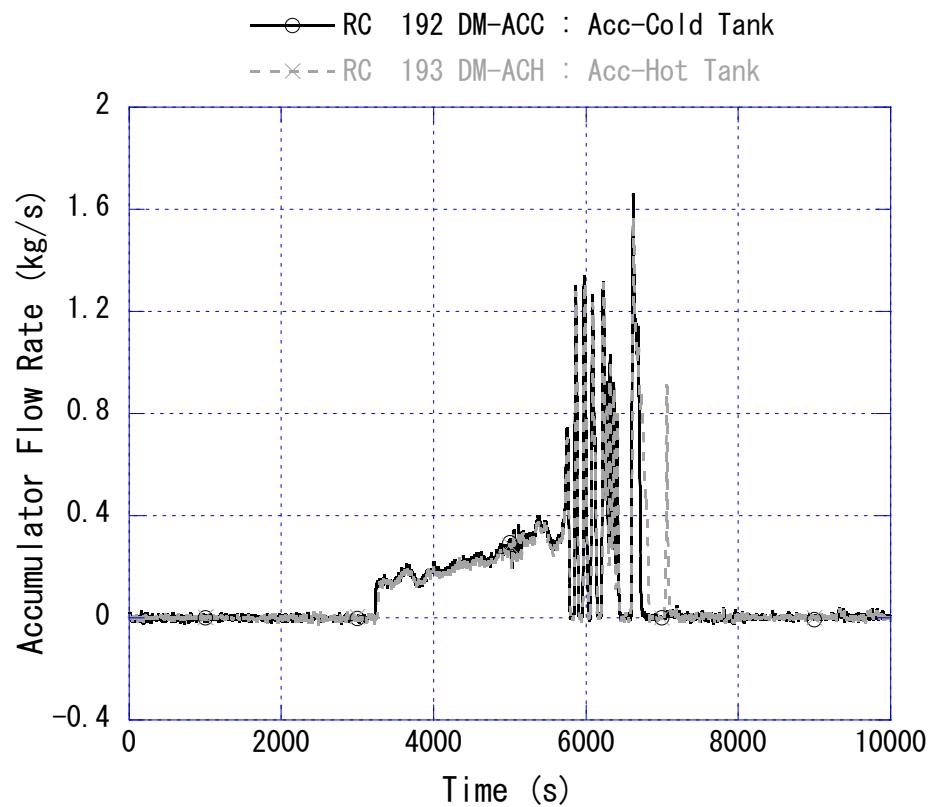


Fig. 4-13 Coolant injection flow rate from accumulator tank (0 to 10000 s)

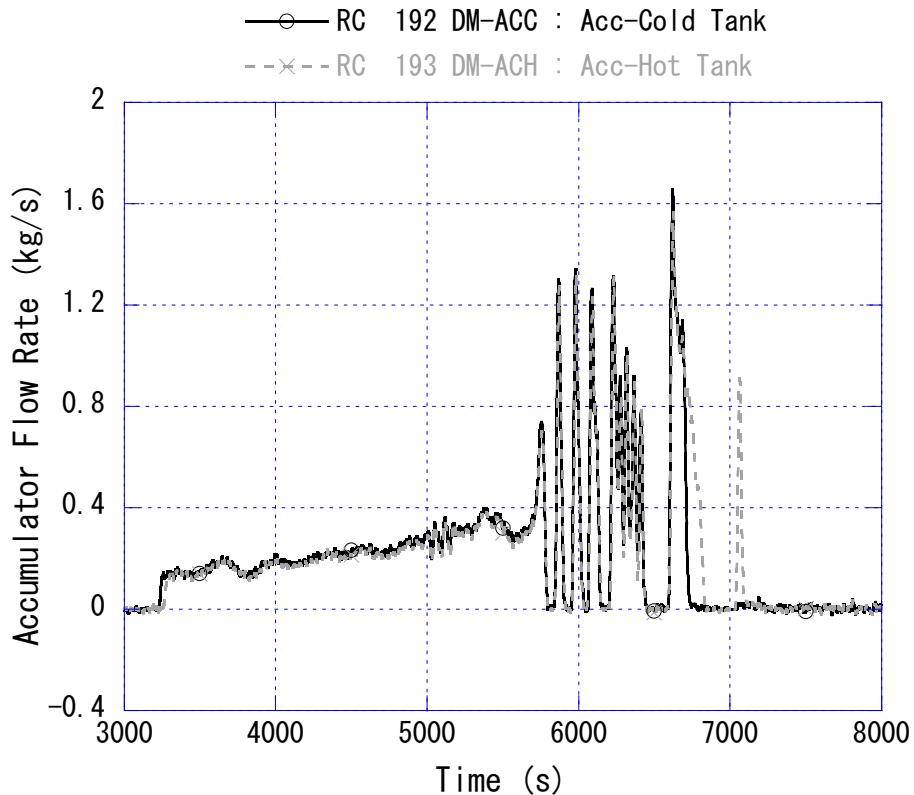


Fig. 4-14 Coolant injection flow rate from accumulator tank (3000 to 8000 s)

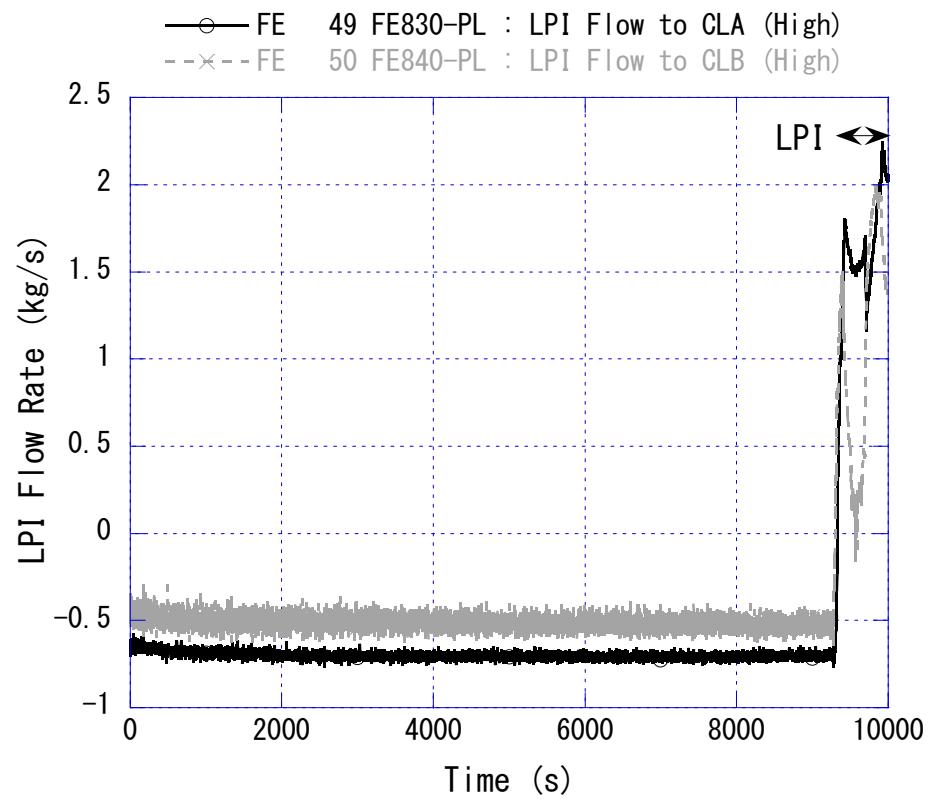


Fig. 4-15 Coolant injection flow rate from LPI system

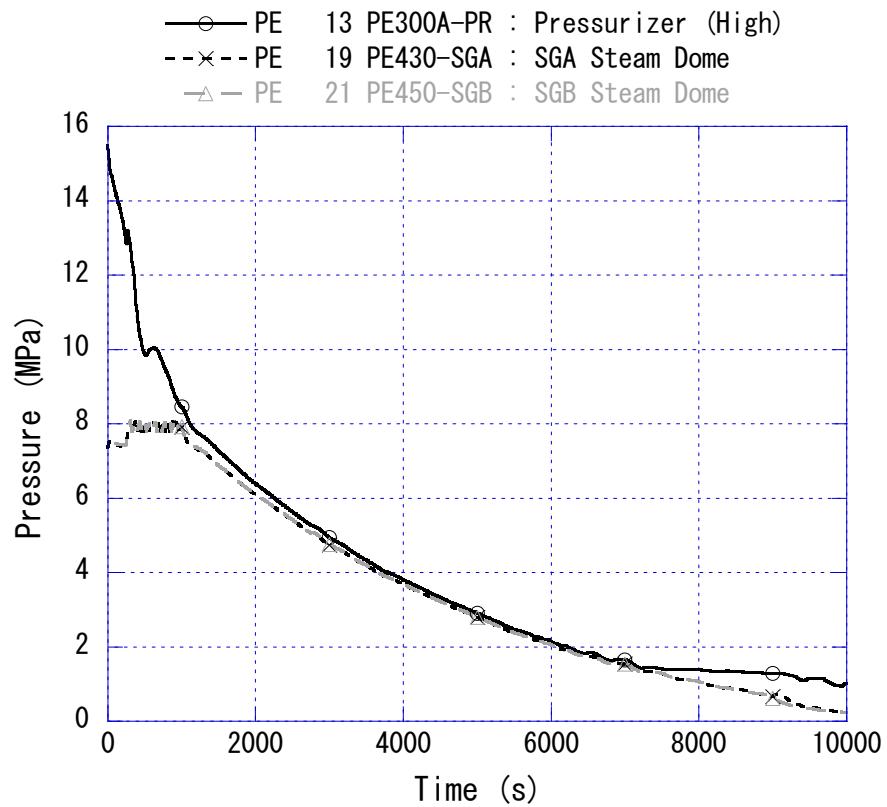


Fig. 4-16 Primary and secondary pressures

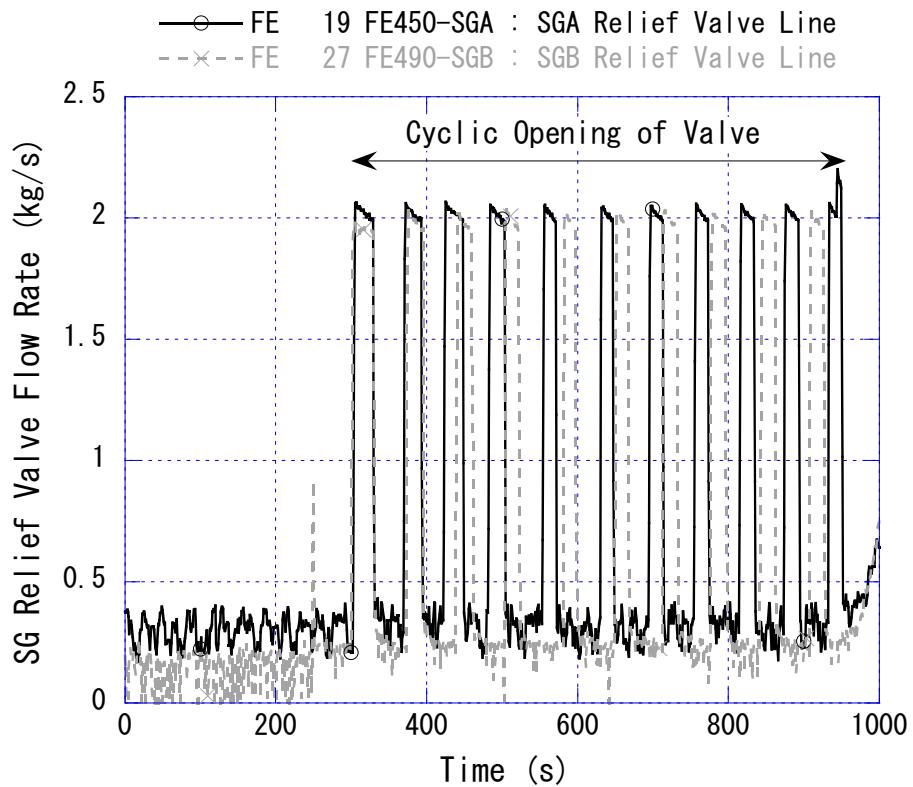


Fig. 4-17 SG relief valve flow rate

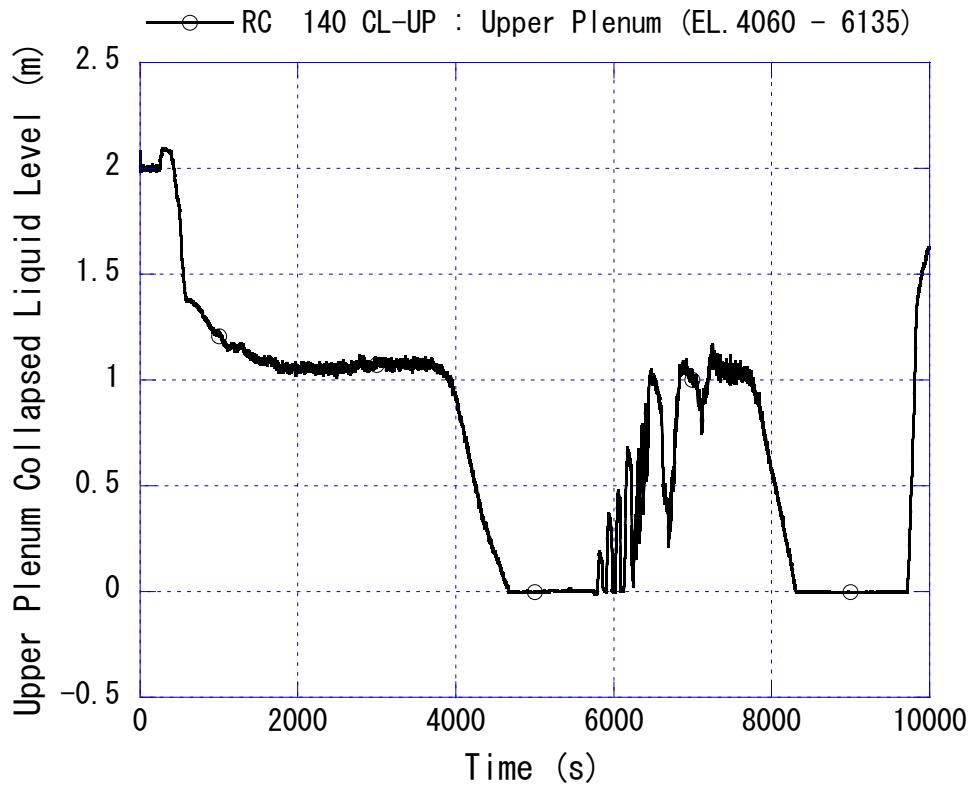


Fig. 4-18 Upper plenum collapsed liquid level (0 to 10000 s)

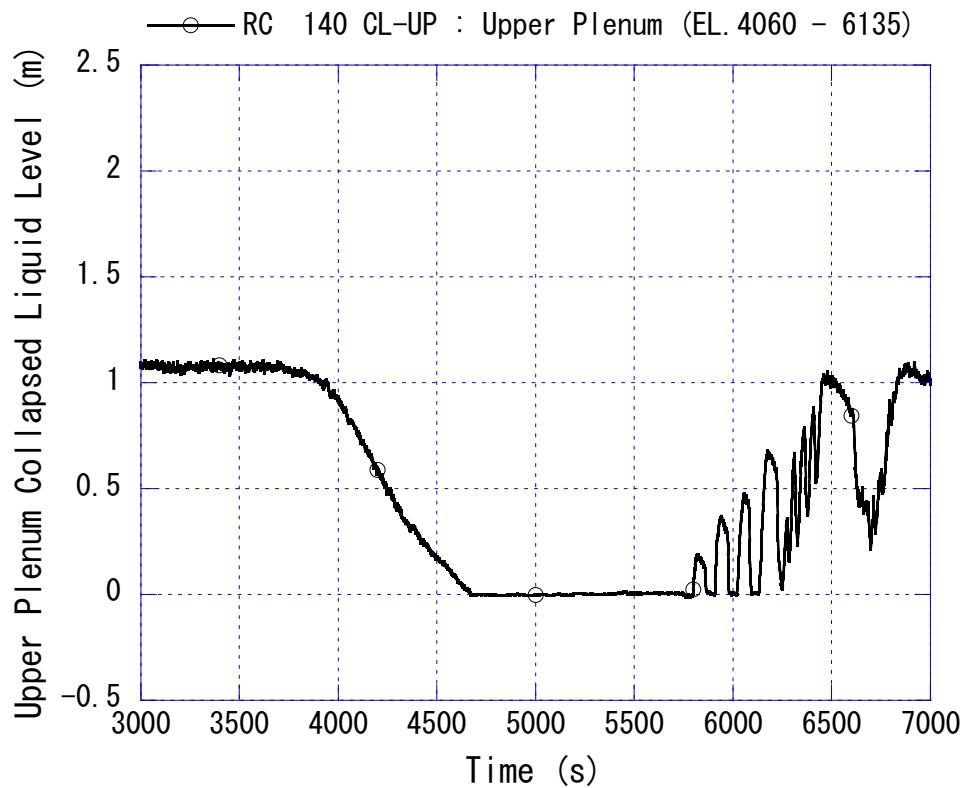


Fig. 4-19 Upper plenum collapsed liquid level (3000 to 7000 s)

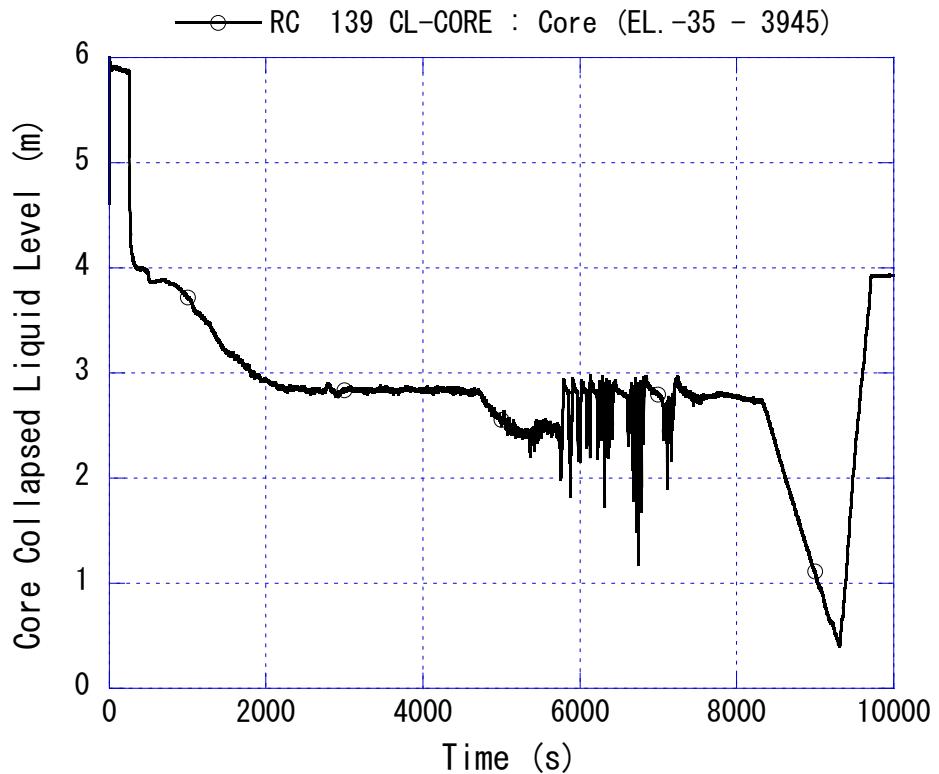


Fig. 4-20 Core collapsed liquid level (0 to 10000 s)

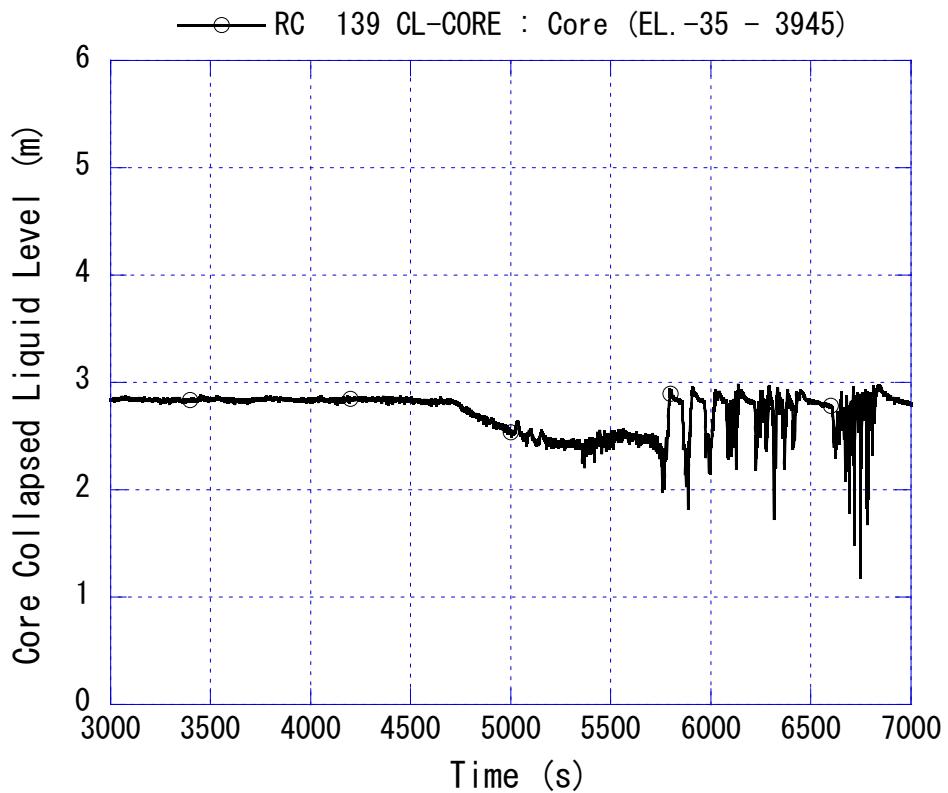


Fig. 4-21 Core collapsed liquid level (3000 to 7000 s)

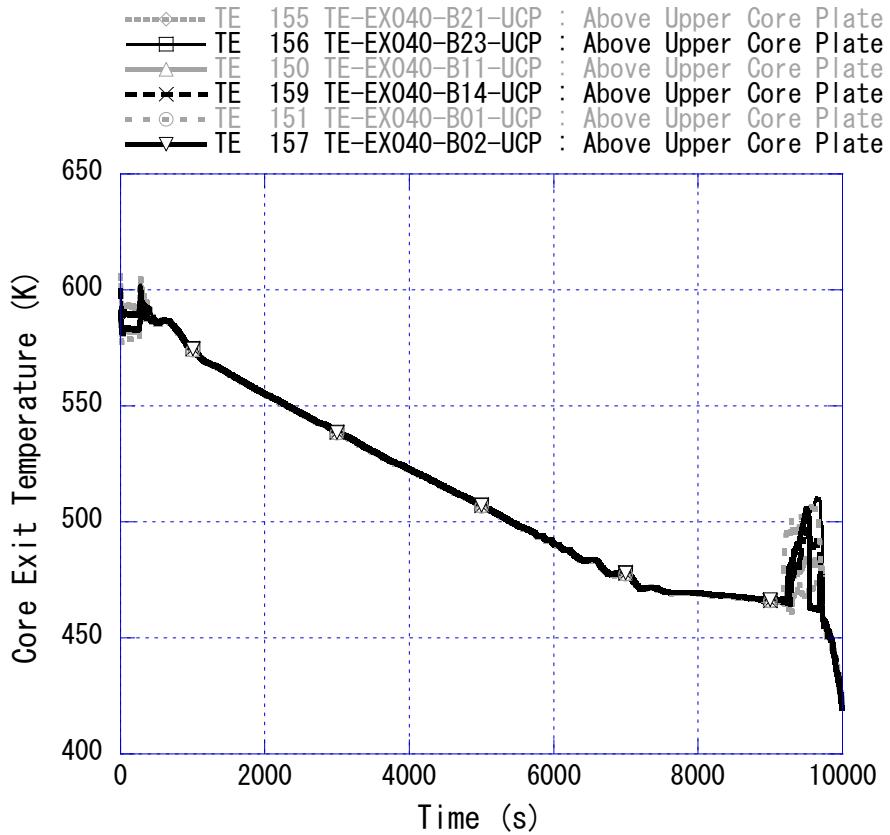


Fig. 4-22 Typical core exit temperatures (0 to 10000 s)

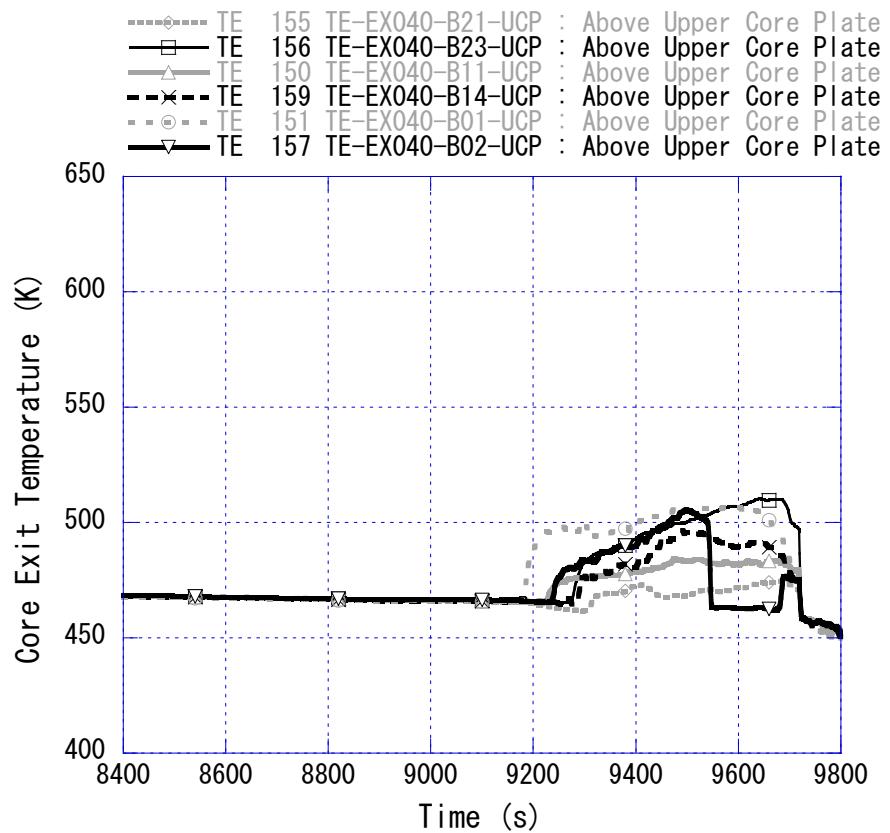


Fig. 4-23 Typical core exit temperatures (8400 to 9800 s)

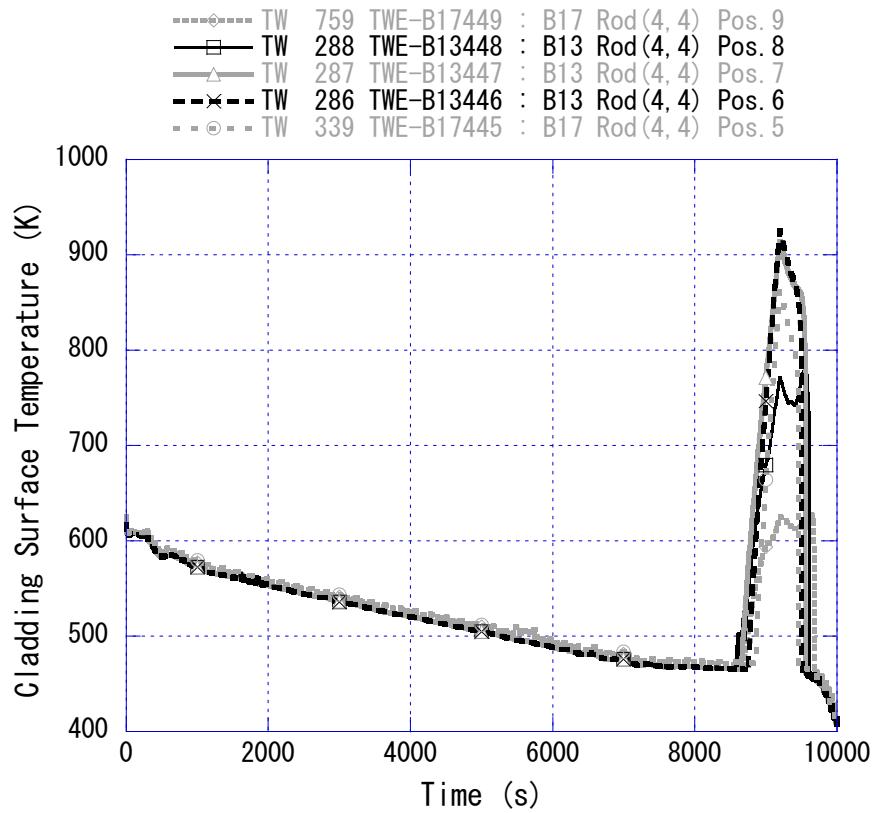


Fig. 4-24 Typical cladding surface temperatures at Positions 9-5 (0 to 10000 s)

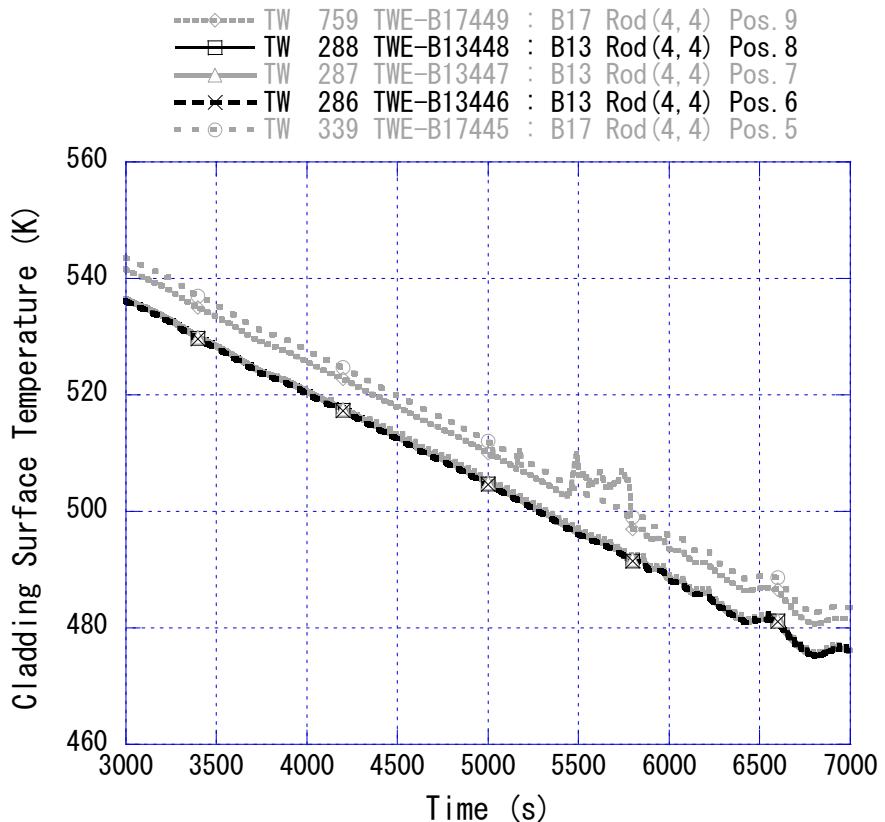


Fig. 4-25 Typical cladding surface temperatures at Positions 9-5 (3000 to 7000 s)

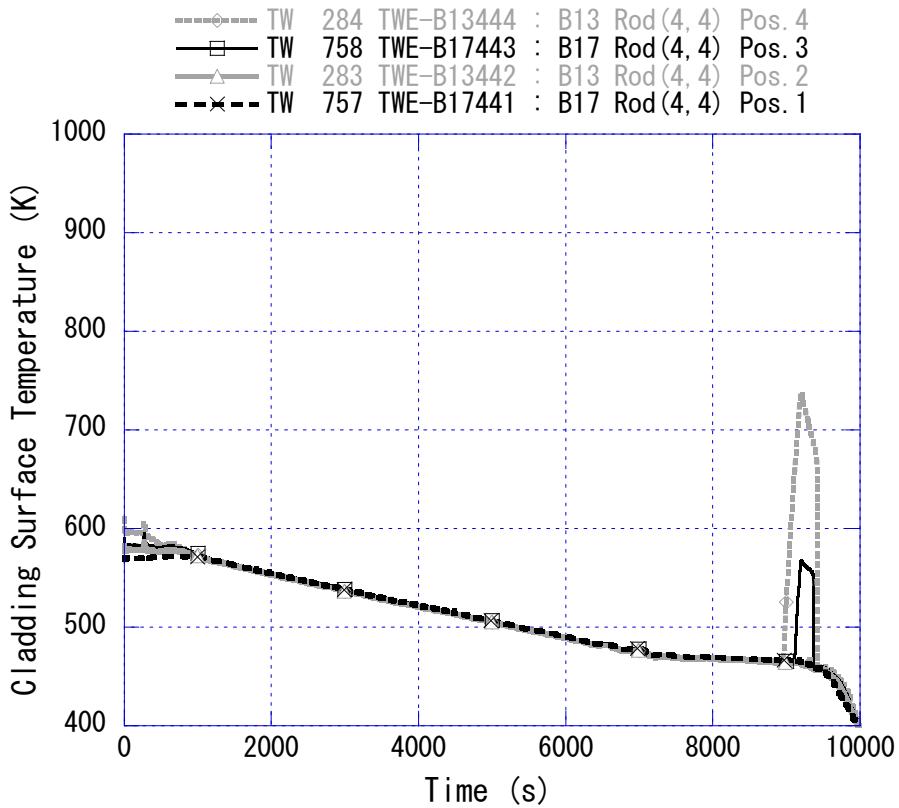


Fig. 4-26 Typical cladding surface temperatures at Positions 4-1 (0 to 10000 s)

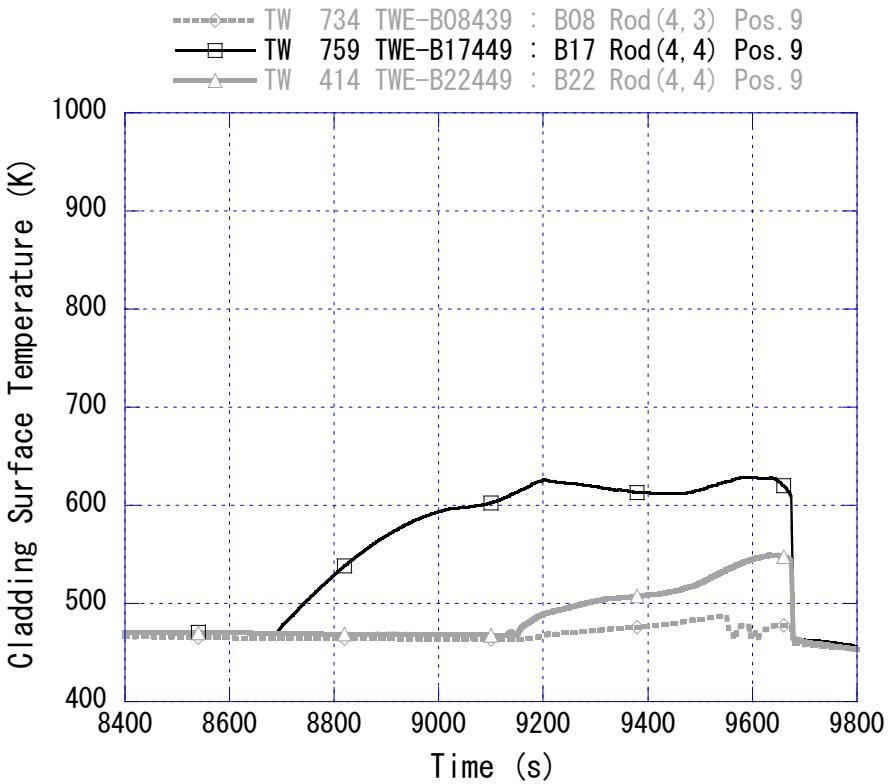


Fig. 4-27 Cladding surface temperatures at Position 9 (8400 to 9800 s)

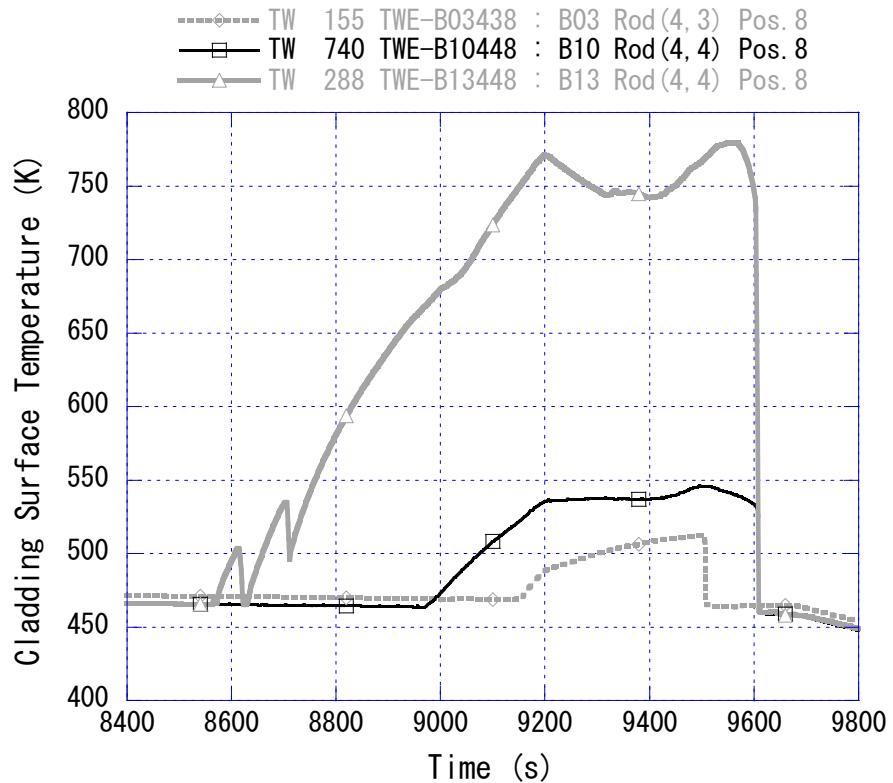


Fig. 4-28 Cladding surface temperatures at Position 8 (8400 to 9800 s)

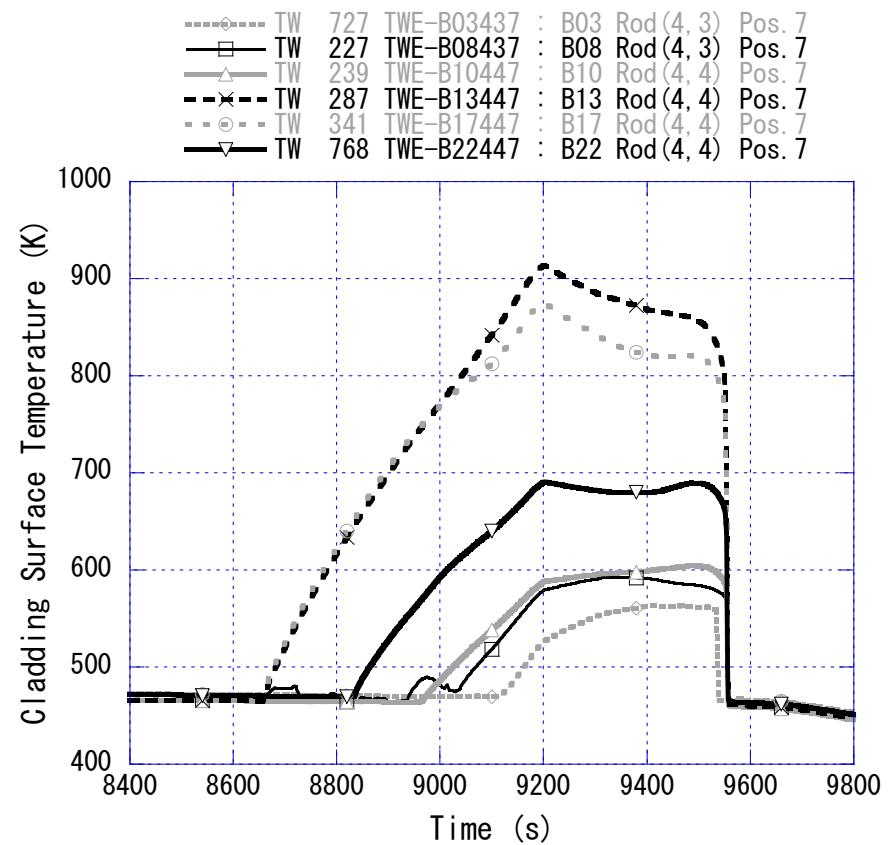


Fig. 4-29 Cladding surface temperatures at Position 7 (8400 to 9800 s)

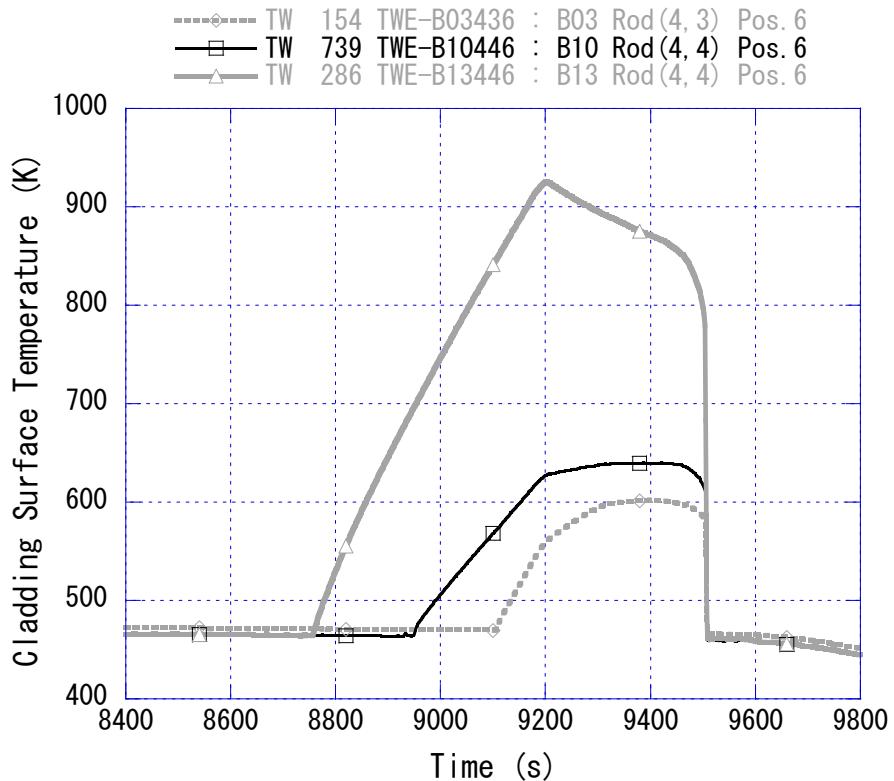


Fig. 4-30 Cladding surface temperatures at Position 6 (8400 to 9800 s)

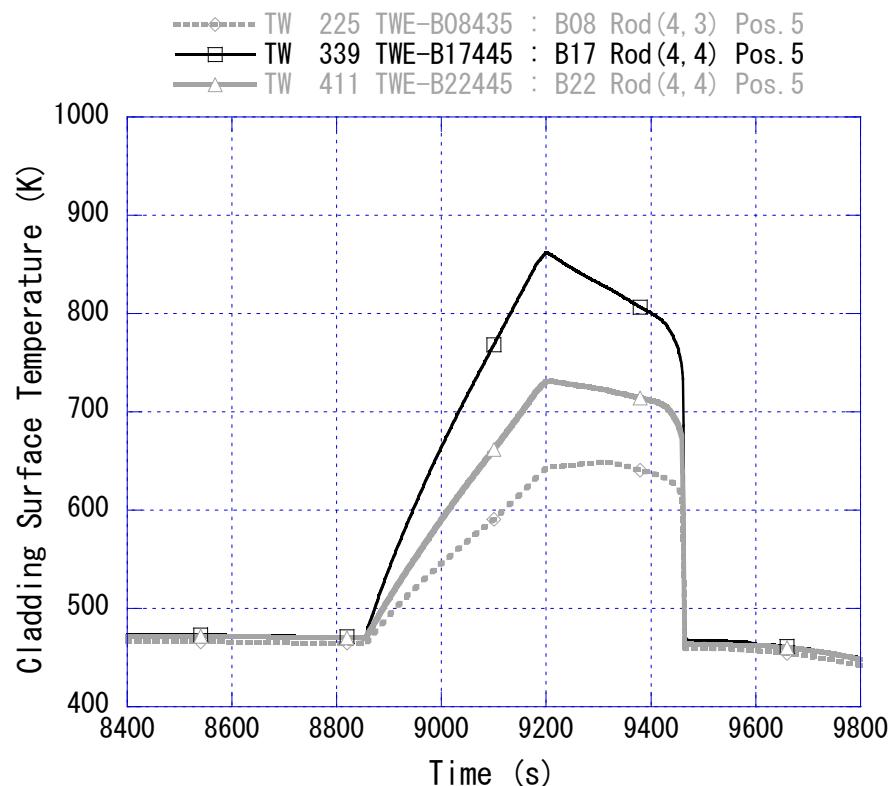


Fig. 4-31 Cladding surface temperatures at Position 5 (8400 to 9800 s)

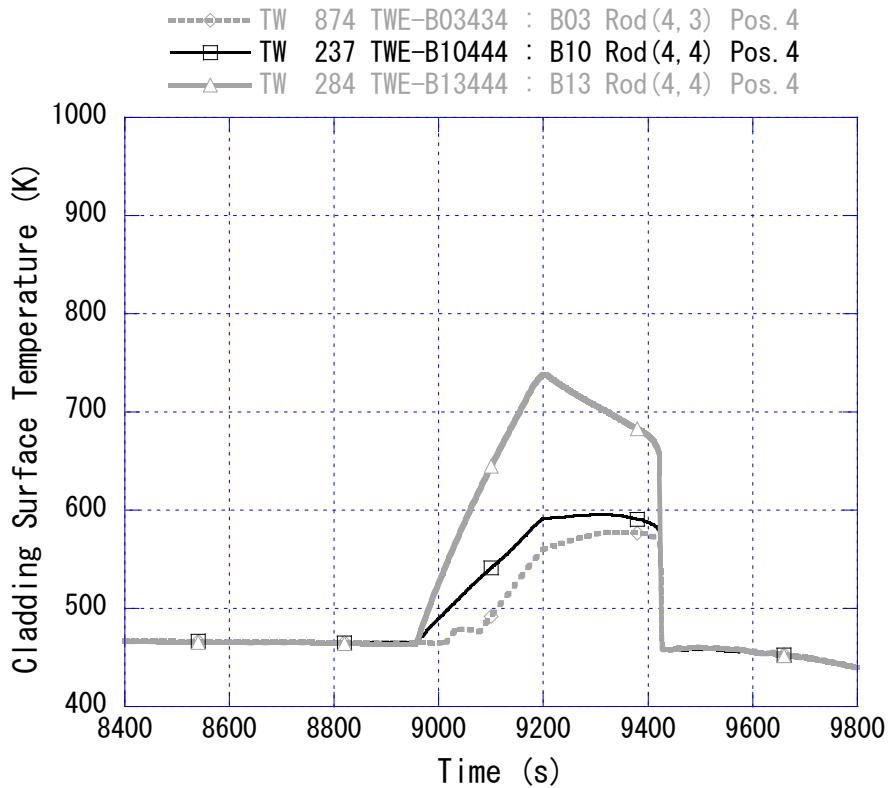


Fig. 4-32 Cladding surface temperatures at Position 4 (8400 to 9800 s)

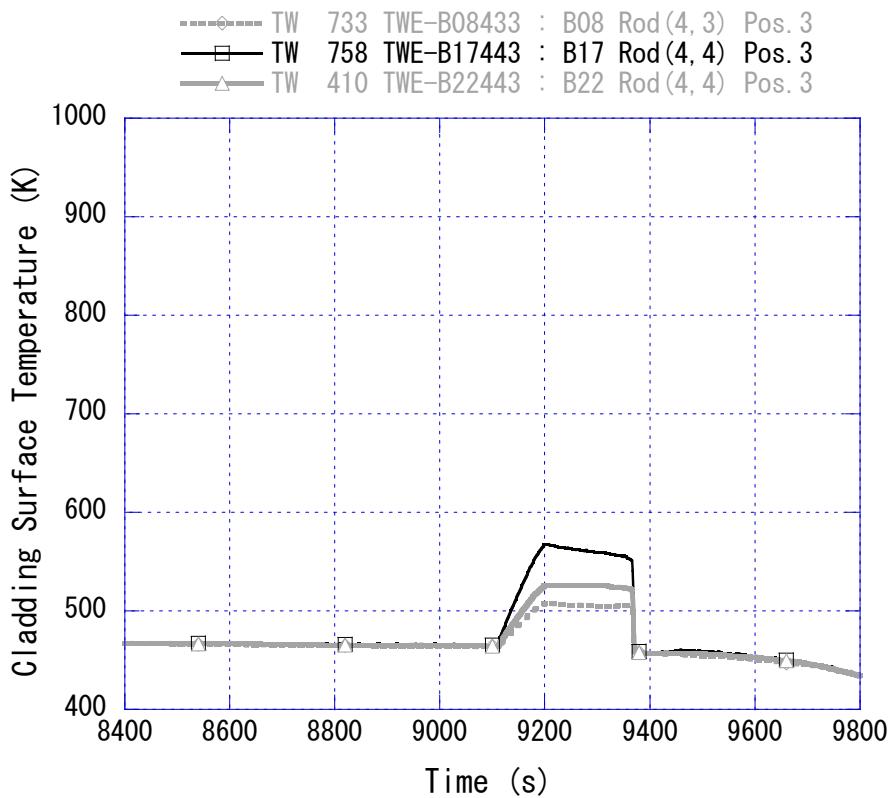


Fig. 4-33 Cladding surface temperatures at Position 3 (8400 to 9800 s)

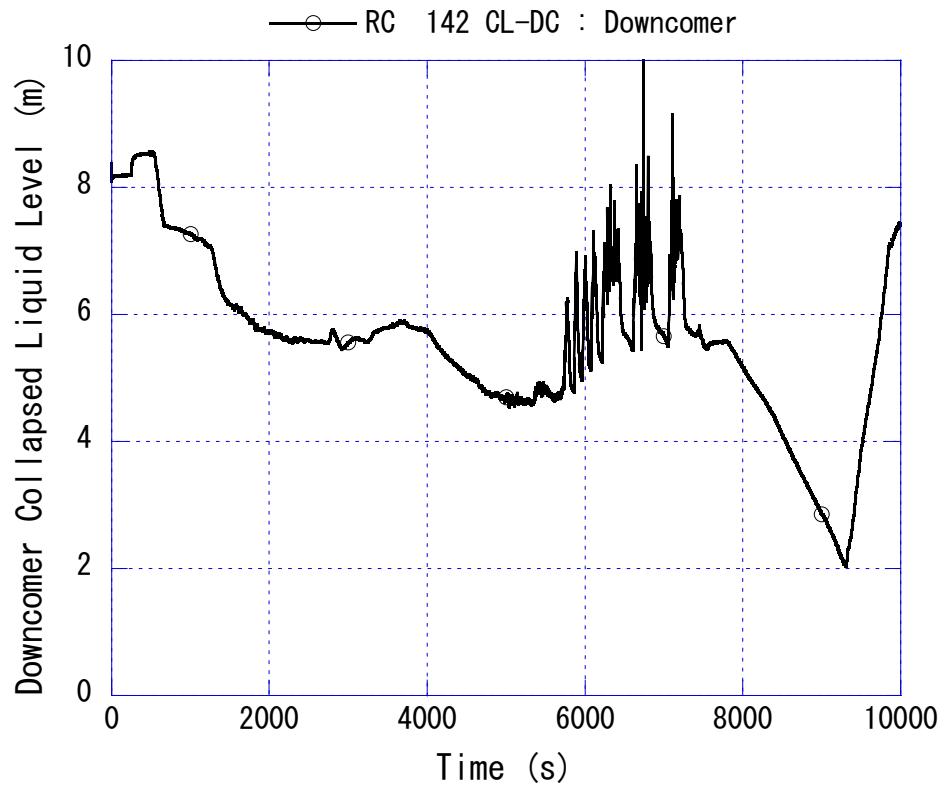


Fig. 4-34 Downcomer collapsed liquid level

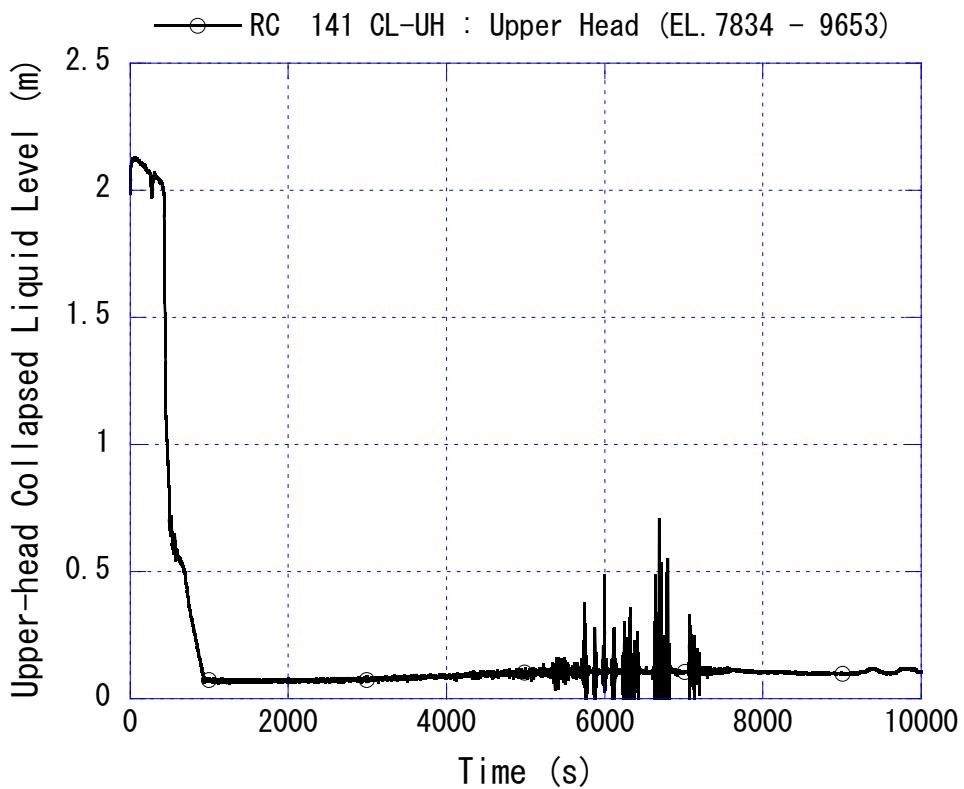


Fig. 4-35 Upper-head collapsed liquid level

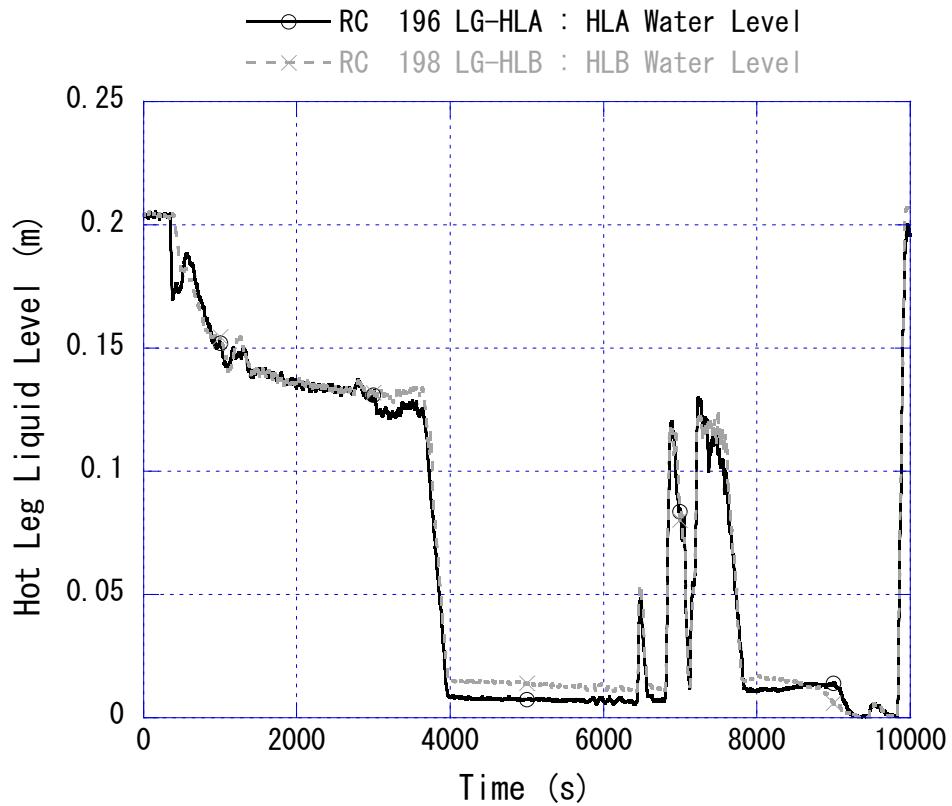


Fig. 4-36 Hot leg liquid level

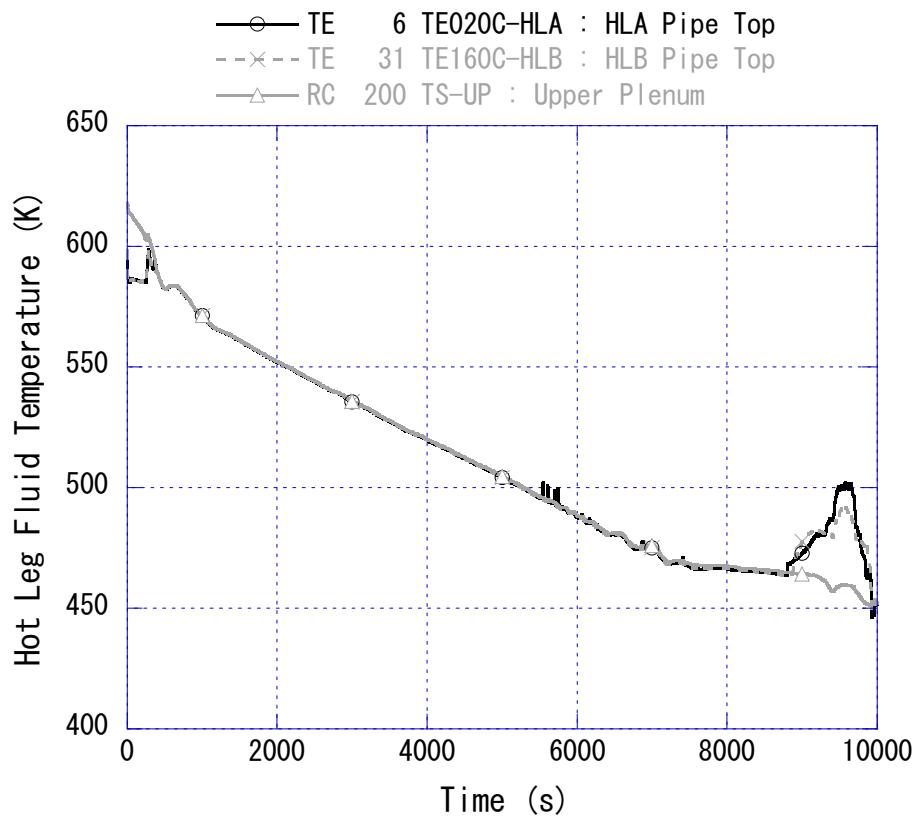


Fig. 4-37 Hot leg fluid temperature

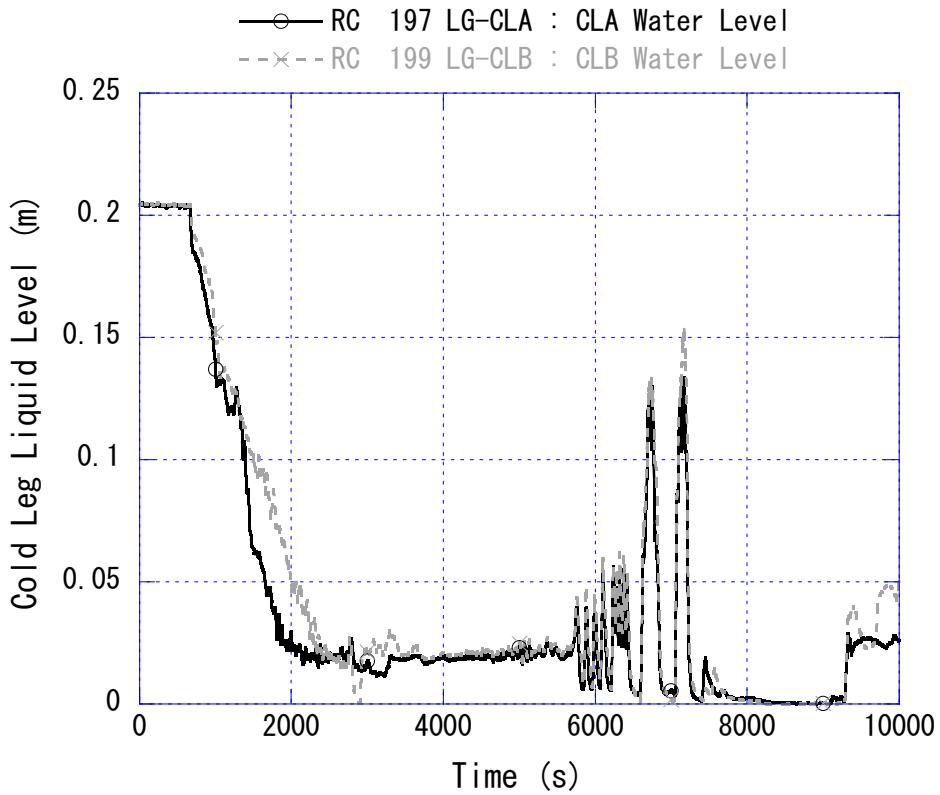


Fig. 4-38 Cold leg liquid level

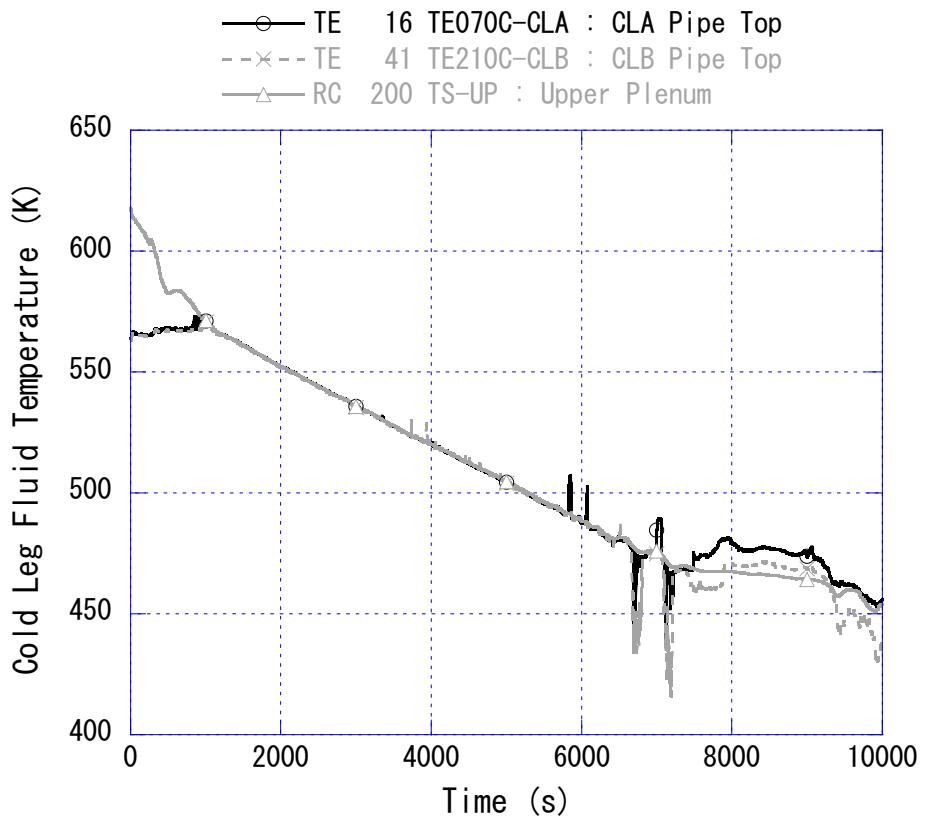


Fig. 4-39 Cold leg fluid temperature

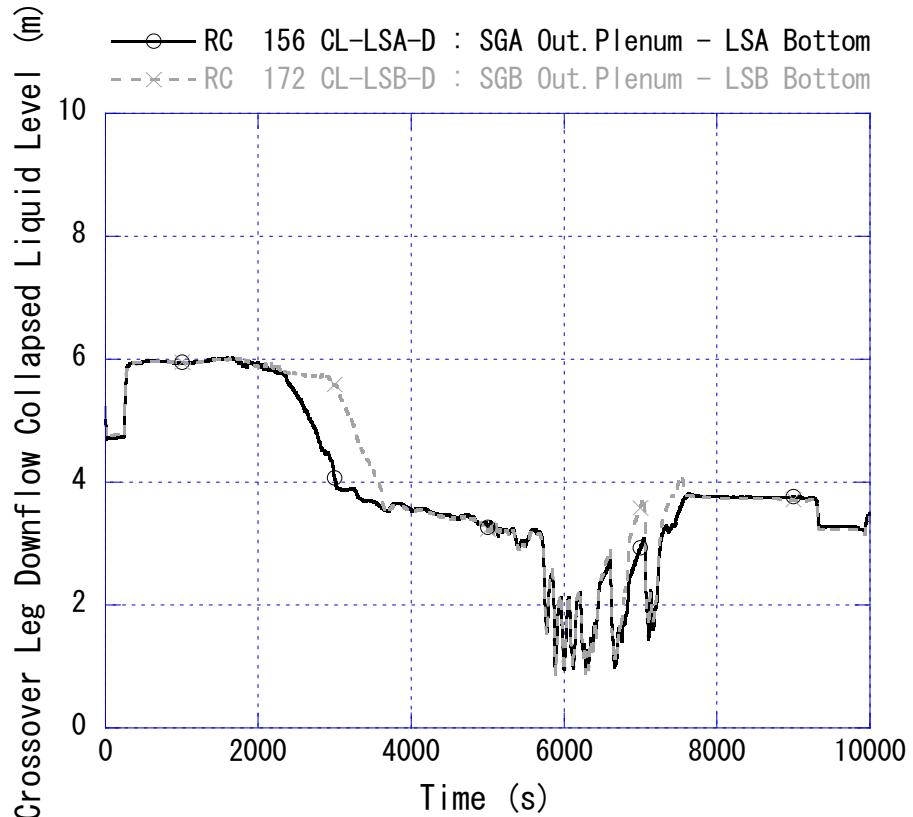


Fig. 4-40 Crossover leg downflow-side collapsed liquid level

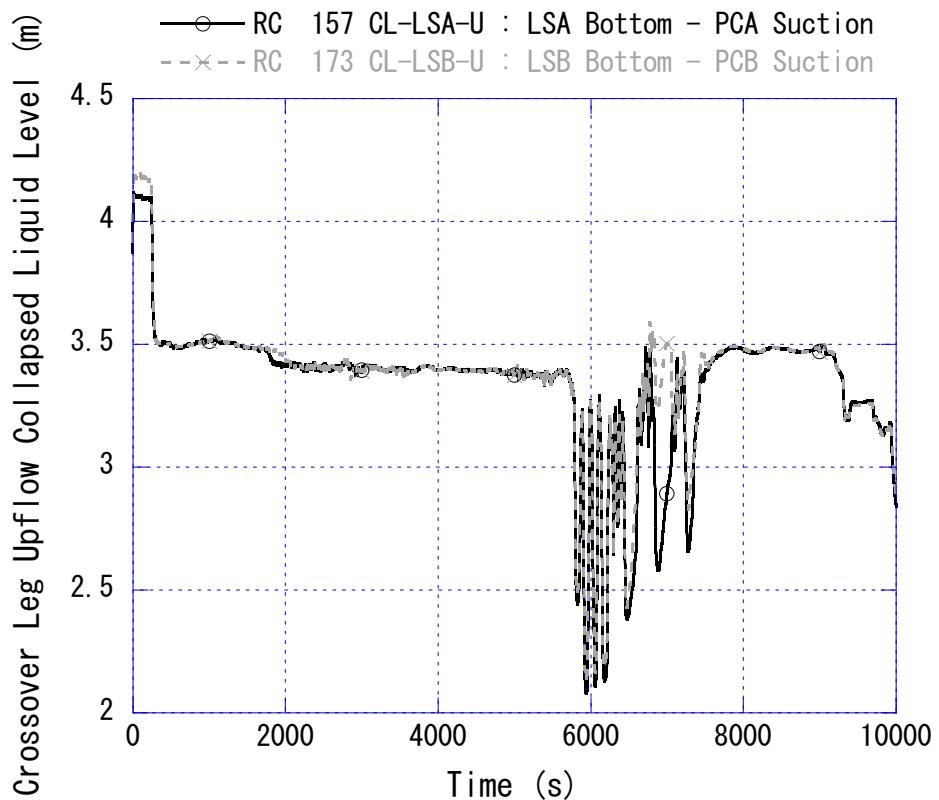


Fig. 4-41 Crossover leg upflow-side collapsed liquid level

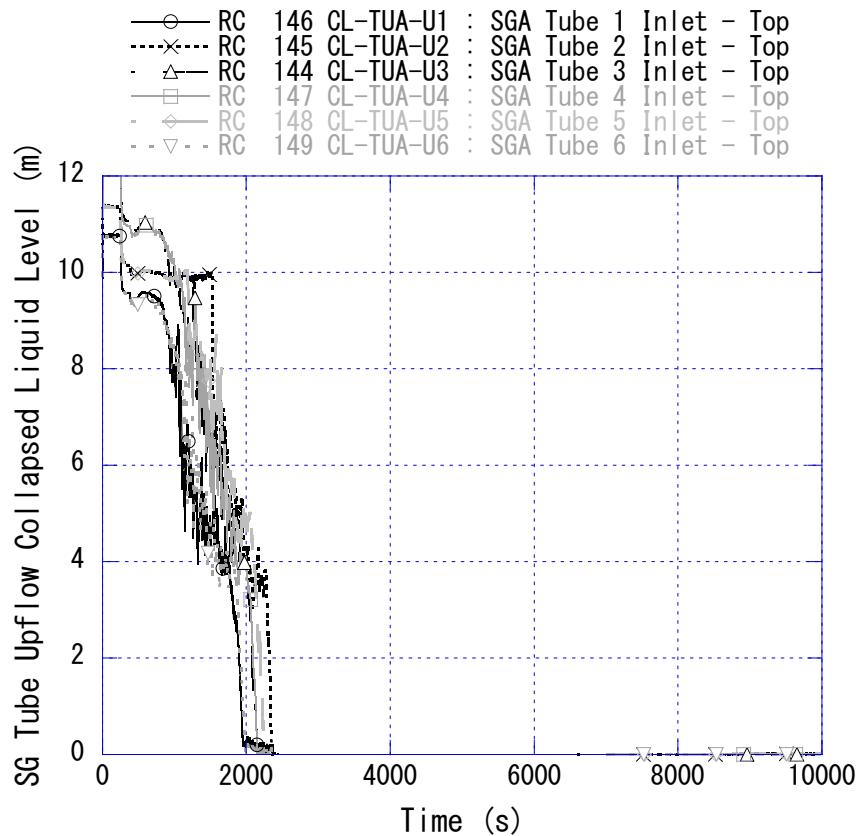


Fig. 4-42 SG U-tube upflow-side collapsed liquid level in loop with pressurizer

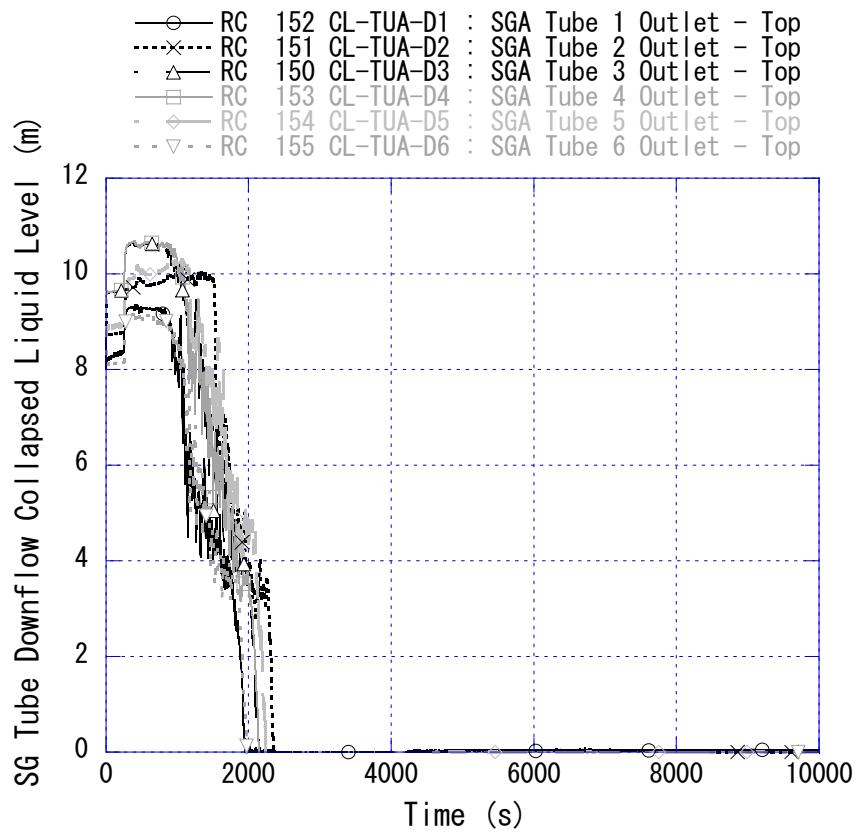


Fig. 4-43 SG U-tube downflow-side collapsed liquid level in loop with pressurizer

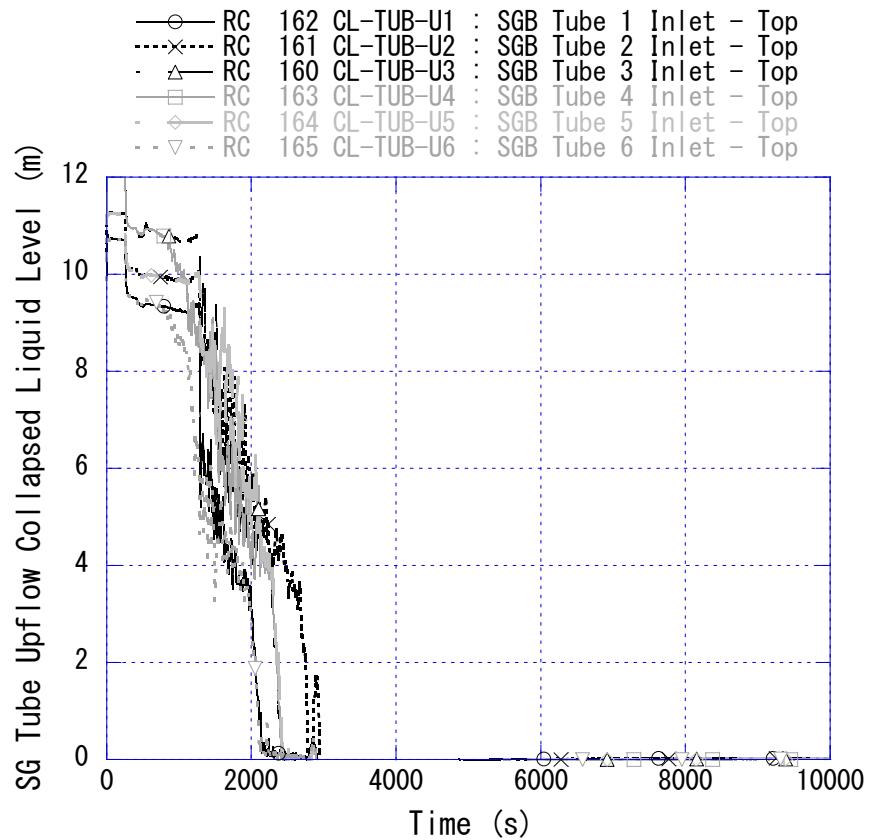


Fig. 4-44 SG U-tube upflow-side collapsed liquid level in loop without pressurizer

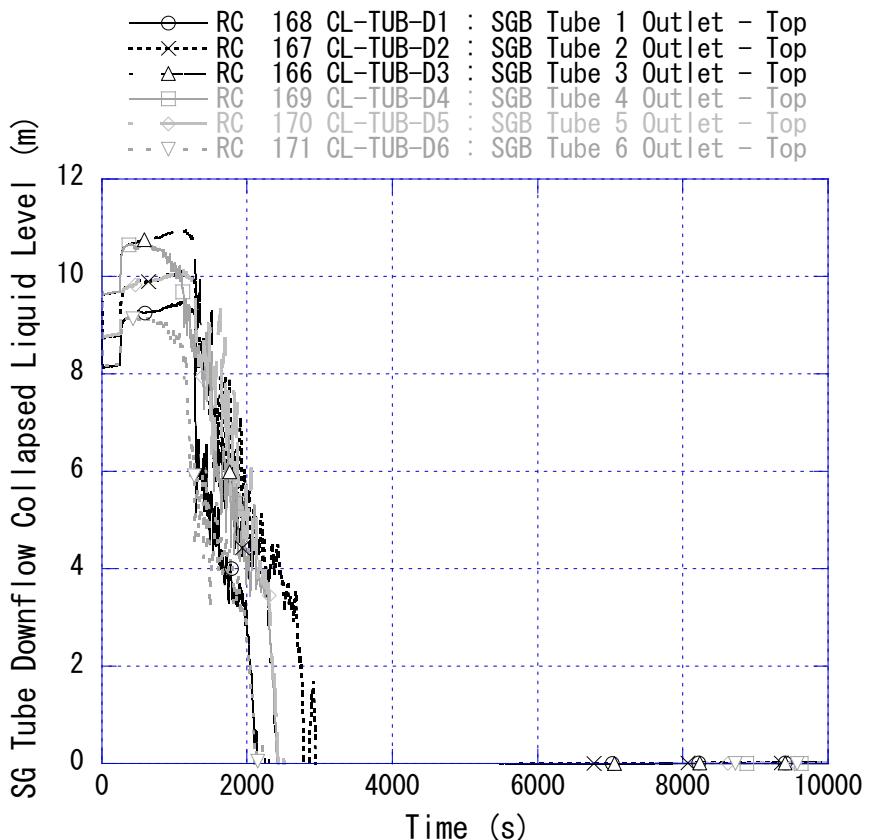


Fig. 4-45 SG U-tube downflow-side collapsed liquid level in loop without pressurizer

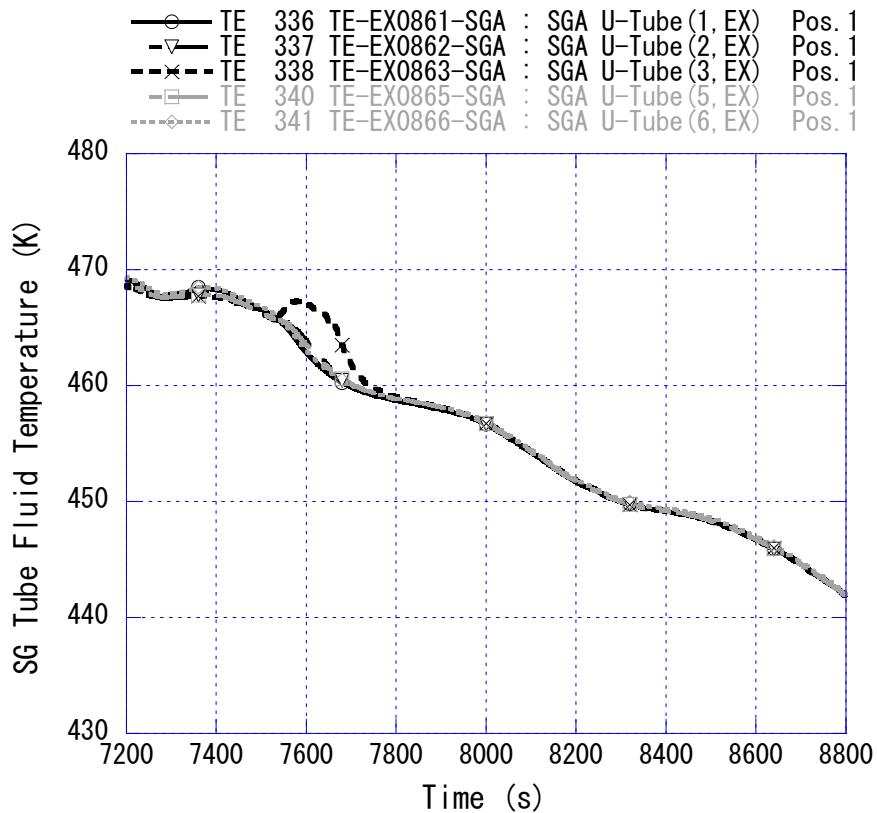


Fig. 4-46 SG U-tube downflow-side fluid temperature in loop with pressurizer

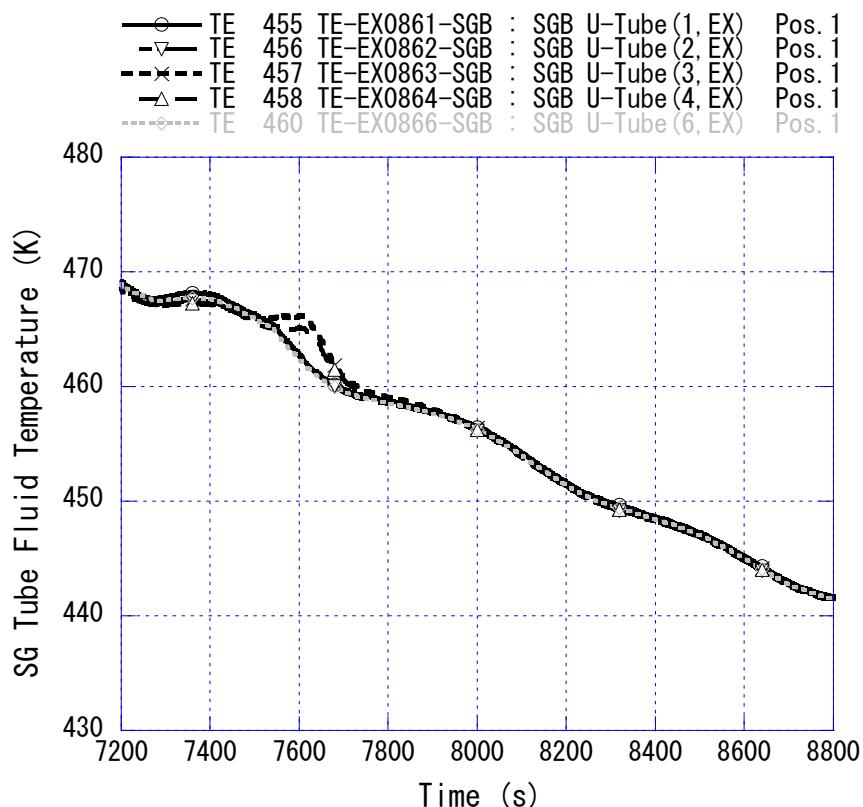


Fig. 4-47 SG U-tube downflow-side fluid temperature in loop without pressurizer

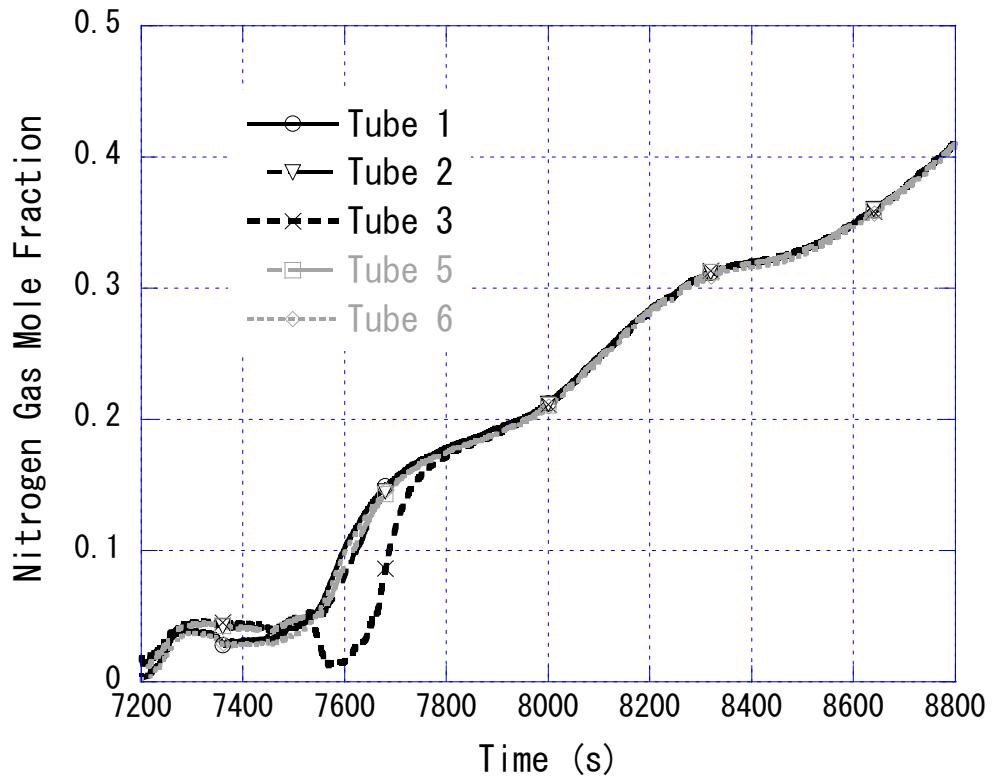


Fig. 4-48 Nitrogen gas mole fraction of SG U-tube in loop with pressurizer

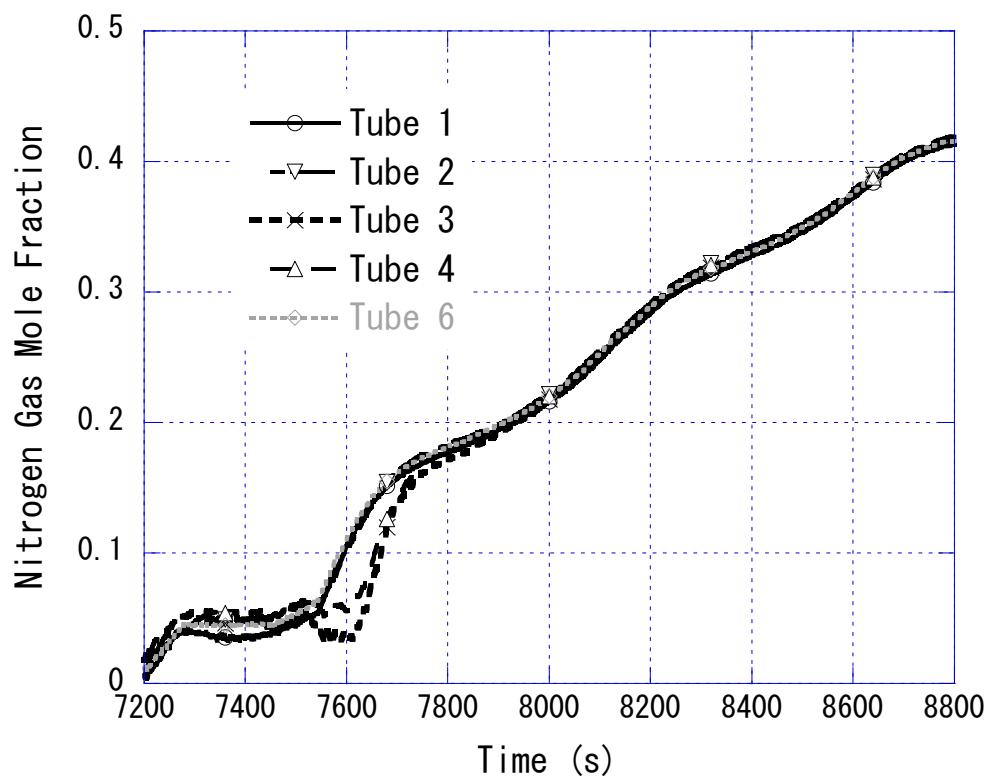


Fig. 4-49 Nitrogen gas mole fraction of SG U-tube in loop without pressurizer

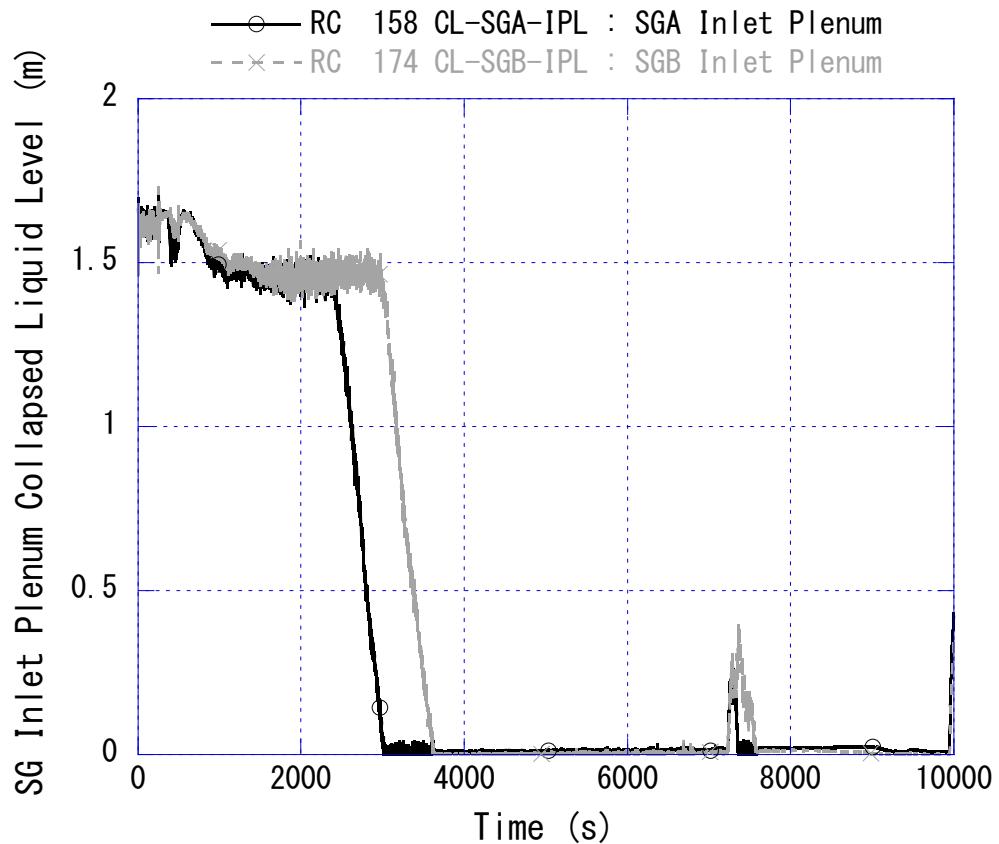


Fig. 4-50 SG inlet plenum collapsed liquid level

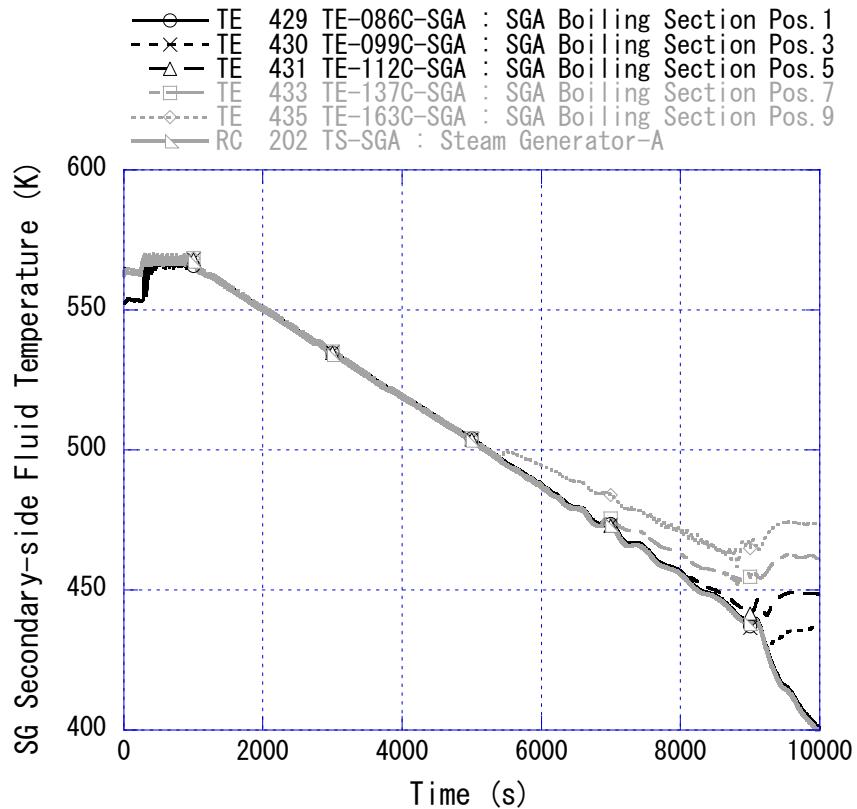


Fig. 4-51 SG secondary-side fluid temperature in loop with pressurizer

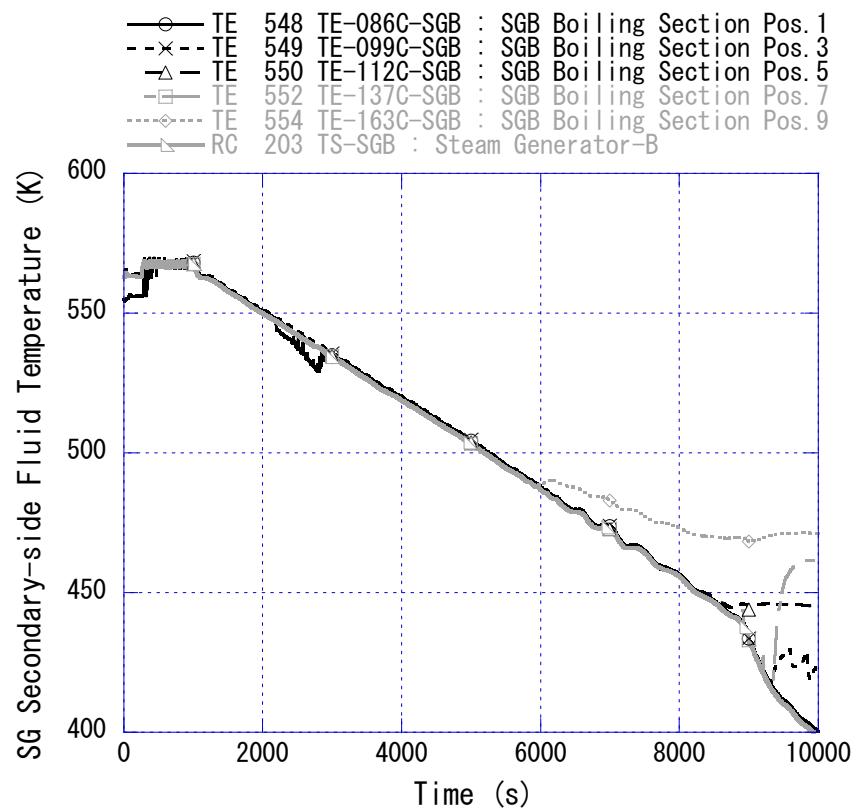


Fig. 4-52 SG secondary-side fluid temperature in loop without pressurizer

5. Summary

A ROSA/LSTF experiment named SB-PV-03 was carried out on November 19, 2002, simulating a PWR 0.2% pressure vessel bottom SBLOCA. The test assumptions were made such as unavailability of HPI system of ECCS and non-condensable gas (nitrogen gas) inflow to the primary system from ACC tanks of ECCS. Secondary-side depressurization of both SGs as an AM action to attain the depressurization rate of 55 K/h in the primary system was undertaken 10 min after a SI signal generation, and went on thereafter. AFW was supplied for 30 min into the secondary-side of both SGs with some delay after the initiation of the AM action. The AM action effectiveness was examined under the impact of nitrogen gas ingress from the ACC tanks. Further, the influence of a difference in the AFW flow rate on the SG secondary-side collapsed liquid level was surveyed. Major consequences are summarized as follows;

- (1) The AM action took effect on the intended primary depressurization, which caused the activation of the ACC system. The liquid level recovery was oscillative in the core because coolant was intermittently fed from the ACC system into both cold legs. Accordingly, there was a small drop in the core liquid level.
- (2) After the ACC tanks began to discharge nitrogen gas into the primary system, the pressure difference between the primary and SG secondary sides became larger due to degradation in the condensation heat transfer in the SG U-tubes.
- (3) The liquid level difference in the secondary-side of both SGs was the largest at the end of the AFW injection, and declined gradually afterwards. The SG secondary-side was not voided throughout the experiment, which gave rise to successive heat removal from the SG secondary-side system.
- (4) A large increase occurred in cladding surface temperature of simulated fuel rods during reflux condensation in the SG U-tubes under nitrogen gas influx. Automatic core power reduction procedure to protect the LSTF core was initiated when the maximum cladding surface temperature exceeded the pre-determined value of 908 K. The peak cladding temperature was 925 K. During the core uncover, by contrast, the CET was increased to 510 K, which did not exceed the initial value.
- (5) After the automatic core power reduction, LPI system of ECCS started the coolant injection into both cold legs. Resultantly, the core was entirely quenched owing to the liquid level recovery in the core.
- (6) After the continuous core cooling was confirmed because of the coolant injection from the LPI system, the experiment came to an end.

Acknowledgement

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Appendix A Available Experimental Data List

Table A-1 shows the list of available experimental data qualified as “Good” for LSTF SB-PV-03 (Run ID designated to be SP3). This table contains Sequential No., Function ID., Tag Name, measurement location, range, unit, and uncertainty. The alphabetical prefix of the Function ID. and Tag Name is explained as follows;

- (1) TE, fluid temperature,
- (2) DT, differential temperature,
- (3) TW, heater rod and structure temperature,
- (4) FE, flow rate measured with conventional (differential pressure) flow meters,
- (5) PE, pressure,
- (6) MI, miscellaneous instrumented-signal (power, pump rotation speed, etc.),
- (7) LE, liquid level,
- (8) DP, differential pressure,
- (9) DE, fluid density measured with gamma-ray densitometer,
- (10) RC, two-phase flow data calculated with DE and others.

Table A-1 (Cont'd)

SEQ No.	Function ID.	Tagname	Location	Range		Unit	Uncertainty	
				LO	HI		± ABS.	± %FR
1385	RC 174	CL-SGB-IPL	SGB Inlet Plenum	-	-	m	0.094	-
1386	RC 175	MC-UH	Upper Head (EL.6634 – 8860)	-	-	kg	79.08	-
1387	RC 176	MC-LSA-DW	SGA Out.Plenum+LSA Downflow	-	-	kg	42.68	-
1388	RC 177	MC-LSB-DW	SGB Out.Plenum+LSB Downflow	-	-	kg	41.33	-
1389	RC 178	MS-CORE	Core (EL.-35 – 3945)	-	-	kg	15.4	-
1390	RC 179	MS-UP	Upper Plenum (EL.4060 – 6135)	-	-	kg	24.37	-
1391	RC 180	MS-DC	Downcomer	-	-	kg	46.31	-
1392	RC 181	MS-TUA-UP-AV	SGA Tubes Upflow side	-	-	kg	8.3	-
1393	RC 182	MS-TUA-DW-AV	SGA Tubes Downflow side	-	-	kg	8.3	-
1394	RC 183	MS-SGA-IPL	SGA Inlet Plenum	-	-	kg	27.81	-
1395	RC 184	MS-LSA-UP	LSA Upflow side	-	-	kg	3.57	-
1396	RC 185	MS-TUB-UP-AV	SGB Tubes Upflow side	-	-	kg	9.65	-
1397	RC 186	MS-TUB-DW-AV	SGB Tubes Downflow side	-	-	kg	10.25	-
1398	RC 187	MS-SGB-IPL	SGB Inlet Plenum	-	-	kg	14.61	-
1399	RC 188	MS-LSB-UP	LSB Upflow side	-	-	kg	3.57	-
1400	RC 189	MS-ACC	Acc-Cold Tank	-	-	kg	39.26	-
1401	RC 190	MS-ACH	Acc-Hot Tank	-	-	kg	74.26	-
1402	RC 191	MS-ST	Break Flow Supp. Tank	-	-	kg	323.61	-
1403	RC 192	DM-ACC	Acc-Cold Tank	-	-	kg/s	13.55	-
1404	RC 193	DM-ACH	Acc-Hot Tank	-	-	kg/s	26.08	-
1405	RC 194	IM-ST	Break Flow Supp. Tank	-	-	kg/s	3.35	-
1406	RC 195	DM-RWST	RWST	-	-	kg/s	51.9	-
1407	RC 196	LG-HLA	HLA Water Level	-	-	m	0.012	-
1408	RC 197	LG-CLA	CLA Water Level	-	-	m	0.028	-
1409	RC 198	LG-HLB	HLB Water Level	-	-	m	0.012	-
1410	RC 199	LG-CLB	CLB Water Level	-	-	m	0.028	-
1411	RC 200	TS-UP	Upper Plenum	-	-	K	17.64	-
1412	RC 201	TS-PR	Pressurizer	-	-	K	17.64	-
1413	RC 202	TS-SGA	Steam Generator-A	-	-	K	7.82	-
1414	RC 203	TS-SGB	Steam Generator-B	-	-	K	7.82	-
1415	RC 279	DE291-SGB-EU	SGB Feedwater Line	-	-	kg/m ³	-	-

