ENCLOSURE 3

UAP-HF-09567

M-RELAP5 Additional Code Assessment Using LOFT/L3-1 and Semiscale/S-LH-1 Test Data

December 2009
(Non-Proprietary)

M-RELAP5 Additional Code Assessment Using LOFT/L3-1 and Semiscale/S-LH-1 Test Data

PREFACE

M-RELAP5, which is currently used for the US-APWR small break loss-of-coolant accident (SBLOCA) safety analysis, has been validated by using the experimental data obtained in various Separate Effects Test (SET) and Integral Effects Test (IET) facilities. In particular, the code ability to predict the IET is important in assessing the code applicability to SBLOCAs where several thermal-hydraulic phenomena and processes interact in a complicated manner. In the framework of the M-RELAP5 development, the ROSA/LSTF SB-CL-18 test is selected so as to demonstrate the code applicability to the US-APWR SBLOCA analysis.

Independent from the code assessment described above, M-RELAP5 code is assessed based upon the requirement from the TMI action plan, which prescribes various requirements for the plant safety features, operator actions, and safety analyses. The action plan defines that the computer codes used for the safety analysis shall be validated using the simulated-SBLOCA IET data, specifically, obtained in the LOFT and Semiscale test facilities (Item II.K.3.30). In conformance to the requirement, the LOFT/L3-1 and Semiscale/S-LH-1 tests are selected as additional code assessment problems to demonstrate the M-RELAP5 ability for the PWR SBLOCA analyses. The present report describes the M-RELAP5 code validation results using these experimental data.

Table of Contents

List of Tables

List of Figures

List of Acronyms

ACC	Accumulator
APWR	Advanced Pressurized-Water Reactor
BST	Blowdown Suppression Tank
CCFL	Counter-Current Flow Limitation
CHF	Critical Heat Flux
DC	Downcomer
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
HPIS	High Pressure Injection System
IET	Integral Effects Test
INL	Idaho National Laboratory
LBLOCA	Large Break Loss-of-Coolant Accident
LOCA	Loss-of-Coolant Accident
LOFT	Loss-of-Fluid Test
LP	Lower plenum of reactor vessel
LPIS	Low Pressure Injection System
LSTF	Large Scale Test Facility
MHI	Mitsubishi Heavy Industry, Ltd.
PCT	Peak Cladding Temperature
PWR	Pressurized-water Reactor
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
ROSA	Rig of Safety Assessment
RV	Reactor Vessel
SBLOCA	Small Break Loss-of-Coolant Accident
SCS	Secondary Coolant System
SET	Separate Effects Test
SG	Steam Generator
SI	Safety Injection
T_{COLD}	Cold Leg Temperature
T_{HOT}	Hot Leg Temperature
UP	Upper Plenum
USNRC	United States Nuclear Regulatory Committee

1. INTRODUCTION

M-RELAP5[1], which is currently used for the US-APWR small break loss-of-coolant accident (SBLOCA) safety analysis, has been validated by using the experimental data obtained in various Separate Effects Test (SET) and Integral Effects Test (IET) facilities. In particular, the code ability to predict the IET is important in assessing the code applicability to SBLOCAs where several thermal-hydraulic phenomena and processes interact in a complicated manner. In the framework of the M-RELAP5 development, the ROSA/LSTF SB-CL-18 test is selected so as to demonstrate the code applicability to the US-APWR SBLOCA analysis. Although the ROSA/LSTF-SB-CL-18 test was originally conducted to obtain the simulated SBLOCA experimental data for Westinghouse-type 4-Loop PWR design, the data is sufficiently scalable also to the US-APWR SBLOCAs as examined in References 2, 3 and 4.

Independent from the code assessment described above, M-RELAP5 code is assessed based upon the requirement from the TMI action plan[5], which prescribes various requirements for the plant safety features, operator actions, and safety analyses. The action plan defines that the computer codes used for the safety analysis shall be validated using specific SBLOCA IET data obtained in the LOFT and Semiscale test facilities (Item II.K.3.30). In conformance to the requirement, the LOFT/L3-1 and Semiscale/S-LH-1 tests are selected as additional code assessment problems to demonstrate the M-RELAP5 ability to predict the complicated thermal-hydraulic phenomena and processes, which is reported in the present material.

LOFT (Loss-of-Fluid Test) L3-1[6] was the first nuclear powered SBLOCA experiment. The test was designed to simulate a 4-in diameter equivalent single-ended break in the cold leg of a large PWR. The primary purpose of code validation using the LOFT/L3-1 data is to assess the code ability to predict the plant response following the small break.

The Semiscale/S-LH-1 experiment[7] was conducted in the Mod-2C test facility. The Semiscale Mod-2C is a small-scale, nonnuclear, experimental system with an electrically heated core. The S-LH-1 simulated the 5% cold leg SBLOCA, where the upper head to downcomer bypass flow was calibrated to 0.9% of the recirculation flow to retard steam venting through the spray nozzle during the transient. Therefore, the core uncovery occurred prior to clearing of the loop seal in the crossover leg, as was also observed in the Semiscale/U-UT-8 experiment[8]. The primary purpose of code validation using the Semiscale/S-LH-1 data is to assess the code ability to predict not only the system response but also the core heat-up behavior occurring during the loop seal period.

2. LOFT/L3-1 EXPERIMENT ANALYSIS

2.1 Test Description

2.1.1 Test Facility

The Loss-of-Fluid Test (LOFT) reactor system, in particular the primary coolant system and reactor core, is a fully operational, scaled representation of a commercial pressurized water reactor (PWR). Details of the test facility scaling are given in Reference 9. As such, transients resulting from accident initiating events are representative in complexity and nature of those accidents which may occur in commercial PWRs. The experimental assembly comprises five major subsystems which have been instrument such that system variables can be measured and recorded during the test. The subsystems include a) the reactor vessel, b) the intact loop, c) the broken loop, d) blowdown suppression system, and e) the emergency core cooling system (ECCS). The LOFT major components are shown in **Figure 2.1-1**.

The LOFT reactor vessel, which simulates the reactor vessel of a commercial PWR, has an annular downcomer, a lower plenum, lower core support plates, a nuclear core, and an upper plenum. The downcomer is connected the cold legs of the intact and broken loops and contains two instrument stalks. The upper plenum is connected the hot legs of the intact and broken loops. The core contains 1300 unpressurized nuclear fuel rods arranged in five square (15x15 fuel assemblies) and four triangular fuel modules located at the corner, shown in **Figure 2.1-2**. The fuel rods have an active length of 1.67-m and an outside diameter of 10.72-mm. The fuel consists of UO_2 sintered pellets with an average enrichment of 4.0 st% fissile uranium (U^{235}) and with a density that is 93% of theoretical density. Fuel pellet diameter and length are 9.29 and 15.24-mm, respectively. Both ends of the pellets are dished with the total dish volume equal to 2% of the pellet volume. Cladding material is Zircaloy-4. Cladding inside and outside diameters are 9.48 and 10.72-mm, respectively. The details are given in Reference 10.

The intact loop simulates three loops of a commercial four-loop PWR and contains a steam generator (SG), two primary coolant pumps in parallel, a pressurizer, a venturi flow meter, and connecting piping. The broken loop consists of a hot leg and a cold leg that are connected to the reactor vessel and the blowdown suppression tank (BST) header. Each leg consists of a break plane orifice, a quick-opening blowdown valve (QOBV), a recirculation line, an isolation valve, and connecting piping. The break for Experiment L3-1 is located in the broken loop cold leg. The recirculation lines establish a small flow from the broken loop to the intact loop and are used to warm up the broken loop. The broken loop hot leg also contains a simulated steam generator and simulated pump. These simulators have hydraulic orifice plate assemblies which have similar resistances to flow as an active steam generator and a pump.

The blowdown suppression system is comprised of the BST header, the BST, the nitrogen pressurization system, and the BST spray system. The blowdown header is connected to the suppression tank downcomers which extend inside the tank below the water level. The header is also directly connected to the BST vapor space to allow pressure equilibration. The nitrogen pressurization system is supplied by the LOFT inert gas system and uses a remote controlled pressure regulator to establish and maintain the specified BST initial pressure. The spray system consists of a centrifugal pump that discharges through a heat-up exchanger and any of three spray headers or a pump recirculation line that contains a cool-down heat exchanger. The spray pump suction can be aligned to either the BST or the borated water storage tank. The three spray headers have flowrate capacities of 1.3, 3.8 and 13.9 ℓ/s, respectively, and are located in the BST along the upper centerline.

The LOFT ECCS simulates that of a commercial PWR, which consists of two accumulators, a high-pressure injection system (HPIS), and a low-pressure injection system (LPIS). Each system is arranged to inject scaled flowrates of emergency core coolant directly into the primary coolant system. The accumulator, HPIS, and LPIS were used during the L3-1 test. Each system was arranged to inject scaled flowrates of ECC directly into the primary coolant system (RCS) cold leg. To provide these scaled flowrates, accumulator ACC-A, HPIS Pump A, and LPIS Pump A were utilized. Accumulator ACC-A was preset to inject the ECC at a system pressure of 4.22 MPa. HPIS Pump A was set to initiate injection at a system pressure of 13.16 MPa. The pressure setpoint for automatic LPIS injection was 0.98 MPa.

Details of the LOFT system are described in Reference 11.

2.1.2 Experimental Results

Important results from the experiment are discussed in Reference 12 and summarized below.

LOFT/L3-1 was the first nuclear powered SBLOCA experiment. The test was designed to simulate a 4-in diameter equivalent (2.5%) single-ended break in the cold leg of a PWR. Coolant from the accumulator, HPIS and LPIS was injected into the intact loop cold leg. The reactor was scrammed manually at 2 seconds prior to the break initiation (defined to occur at time zero) when the cold leg blowdown valve was opened. The pumps were tripped at the break initiation and coasted down in about 19 seconds. The HPIS flow initiated automatically at about 5 seconds. The pressurizer was empty by 17 seconds and the upper plenum fluid was saturated by 25 seconds.

Natural circulation began as the pumps completed their coastdown and continued until 390 seconds when the primary system pressure dropped below the secondary pressure and the steam generator was no longer a heat sink. The break flow was sufficient, however, to remove the decay heat and to continue system depressurization. At about 630 seconds the accumulator started injecting the ECC. The accumulator emptied of water and nitrogen entered the system at about 1750 seconds. The LPIS setpoint was purposely lowered from a normal pressure of 2.12 MPa to 0.98 MPa to assure nitrogen injection from the accumulator to the RCS. No effects of the nitrogen on the RCS response were observed in the measurements.

The pump inlet loop seal did not clear during the transient as expected because of the large core bypass paths from the upper plenum to the cold leg which allowed pressure equalization between the hot and cold legs.

At about 3600 seconds, secondary bleed and feed was initiated by the operator action, which imposed a 38.8 to 50 K/hr cool-down rate on the secondary system. This procedure had no effect on the primary system pressure because the primary and secondary systems were thermally decoupled.

The mass inventory in the reactor vessel was sufficient at all times to keep the core completely covered, consequently the core remained cooled with the clad temperatures following the coolant saturation temperature.

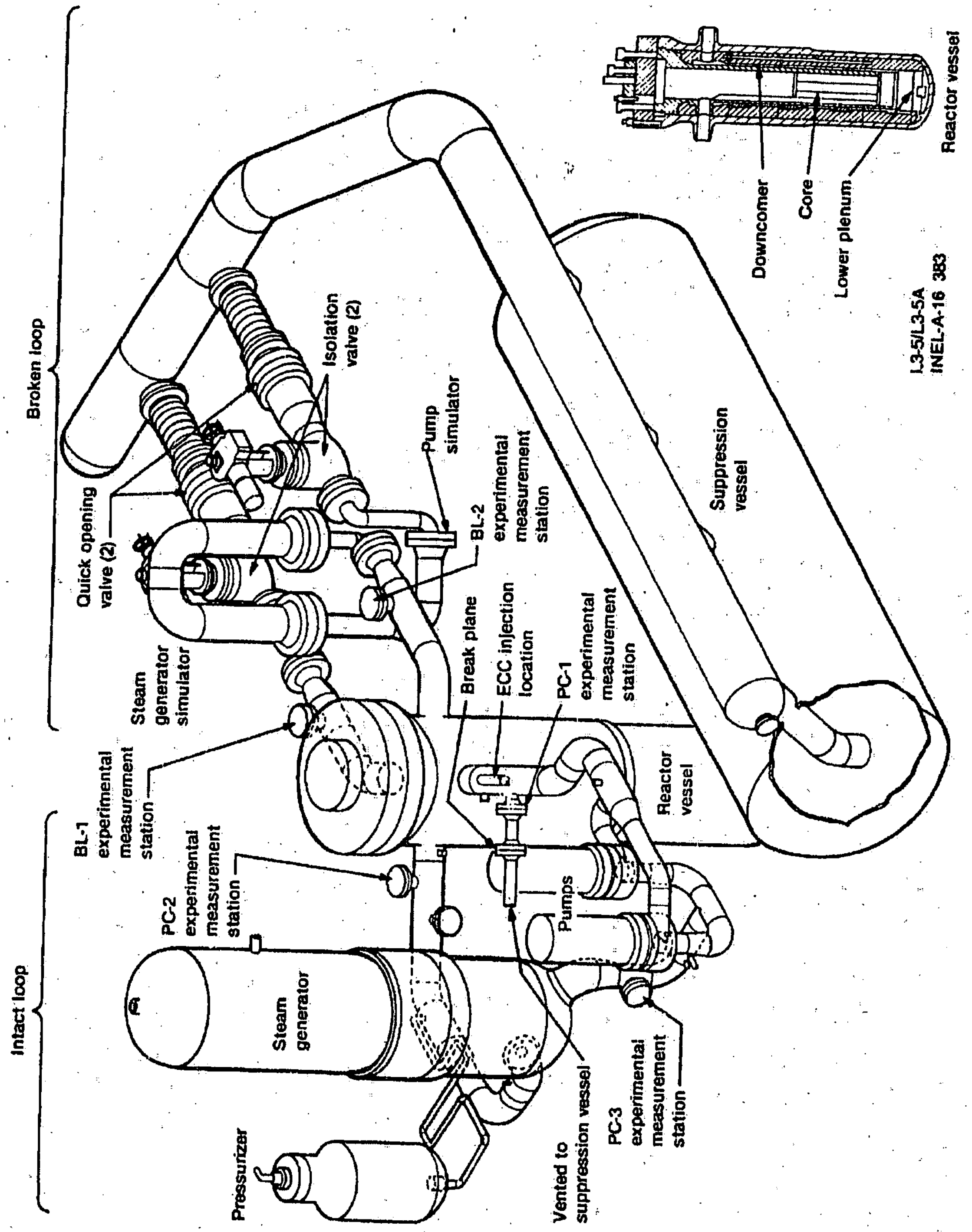

Figure 2.1-1 Schematic of LOFT Major Components[6]

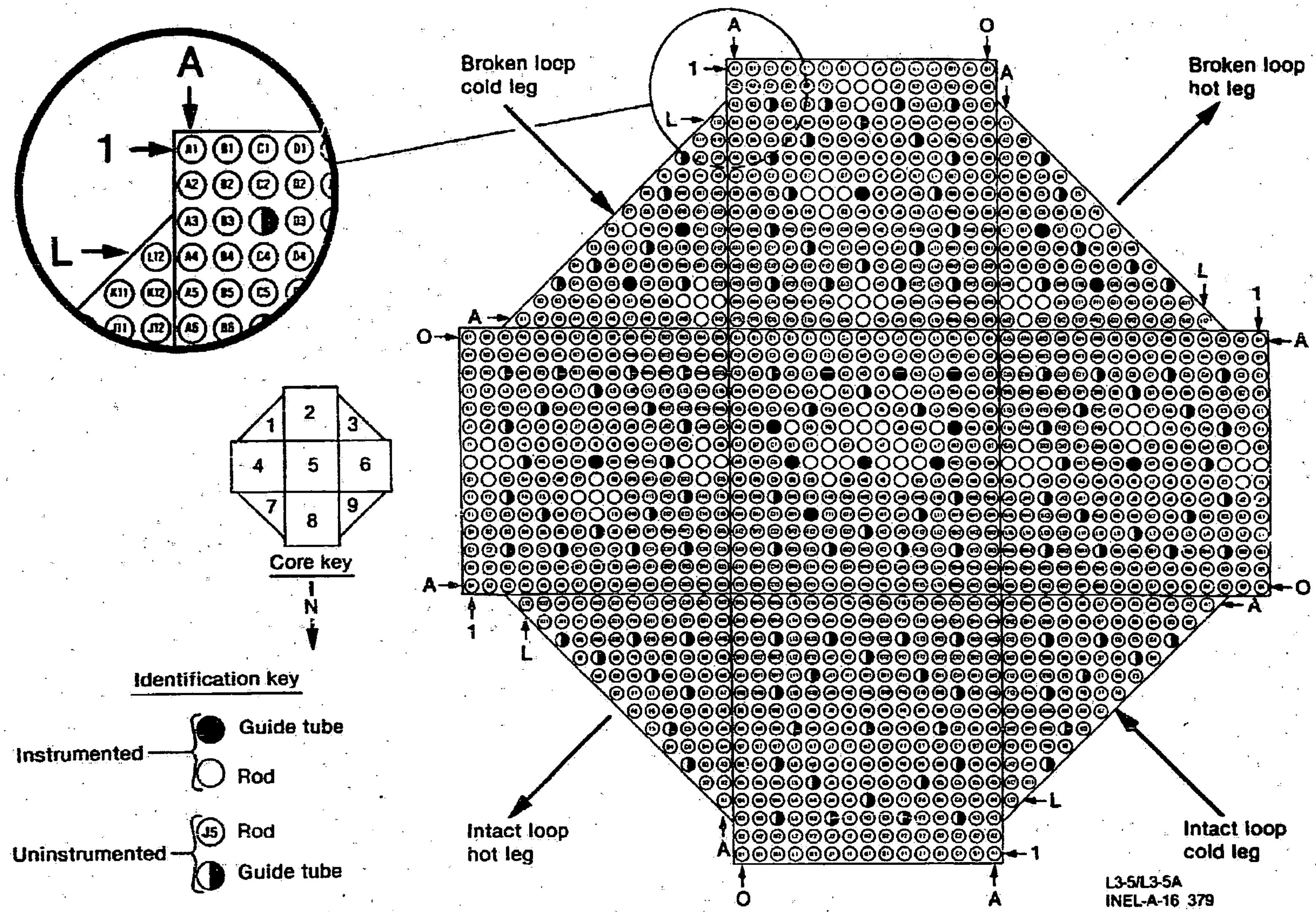

Figure 2.1-2 LOFT Core I Configuration[6]

2.2 M-RELAP5 Code Validation

2.2.1 Analysis Model

The LOFT/L3-1 test was simulated by M-RELAP5 such that the code ability to predict the SBLOCA test was examined as well as done for the other IET analyses, the ROSA/LSTF SB-CL-18 and the Semiscale/S-LH-1. The M-RELAP5 LOFT model used here is based on the input model developed by INL[12]. However, the noding scheme and the thermal-hydraulic model options have been modified so as to conform to the models applied to the US-APWR SBLOCA analysis. The noding diagram is shown in **Figure 2.2-1**.

The M-RELAP5 LOFT model primarily consists of the a) reactor vessel, b) pressurizer, c) steam generator, d) intact loop, e) broken loop, f) ECCS, and g) break assembly. **[**

]

Heat conduction in the nuclear core fuel rods and the reactor component structures are taken into account. **[**

]

The counter-current flow limitation (CCFL) occurring in the piping with a smaller diameter is taken into account for the calculation. The CCFL in the SG U-tubes is modeled using the Wallis correlation[13], where $\beta=0.0$, $c=0.88$, *and* $m=1.0$ are applied. This modeling is identical to that for the US-APWR plant calculation, because the geometric scaling of the SG U-tubes is almost identical between the LOFT and US-APWR. **[**

]

The post-test analysis report[12] states that the steam control valve of the SG secondary system did not seat 100% nor did it seat the same each closure although the valve began to close at 5%/s during the transient test. The actual steam leakage from the secondary was not measured directly. In the present calculation, therefore, the secondary system pressure is imposed as a boundary condition based on the measurement.

The break flow history is imposed as a boundary condition which is specified by the input data based on the measurement. The Moody critical flow model[16] has been implemented into M-RELAP5 for the plant safety analyses[1], in conformance to the requirement prescribed in Appendix K to 10 CFR 50. The Moody critical flow model is known as a model which maximizes the break flowrate. In the framework for the M-RELAP5 code assessment using the IET data, therefore, MHI practically employed an approach to impose the measured break flowrate data as a boundary condition, excluding excessive conservatism and distortions caused by applying the Moody critical flow model. It is noted that the experimental test report[6] mentions that the uncertainty for the measured break flow rate was ±15%. [

]

The core fission power and decay power history are also given through the input data table for the present calculation. Although Reference 12 mentions that no significant effects of noncondensable gas from the accumulator were observed after the accumulator emptied, the noncondensable gas model simulating the nitrogen entering the RCS was applied in the present calculation.

The M-RELAP5 transient calculation simulated the experiment from the break initiation until shortly before the operators manually initiated the steam bleed of the secondary coolant system (SCS). The latter portion of the experiment was not simulated because the behavior of the LOFT facility after the onset of the steam bleed is not relevant to the behavior of the US-APWR.

2.2.2 Analysis Results

The steady-state calculation was performed by M-RELAP5. The converged plant parameters are listed in **Table 2.2-1**, in which the calculation results are compared with the measurements. The table shows that M-RELAP5 accurately reproduces the steady-state condition prior to the transient test for the LOFT/L3-1.

The chronology during the LOFT/L3-1 test is listed in **Table 2.2-2**, where the experimental and calculated results are compared. The transient calculation was initiated by the simulated break flow data shown in **Figure 2.2-2**. The measured secondary system pressure was also given by a boundary condition as shown in **Figure 2.2-3**. [

], a good agreement was obtained for the primary system pressure as shown in **Figure 2.2-4**. It is noted that the M-RELAP5 accuracy for the SG heat transfer has been validated using the ROSA-IV/LSTF SB-CL-18 test data[17]. Following the break initiation, the RCS rapidly decreases to the secondary system pressure during the blowdown phase. The temporal change of pressurizer liquid level was well reproduced by M-RELAP5 as shown in **Figure 2.2-5**. The natural circulation begins as the pumps complete their coastdown, and then the primary and secondary pressures equivalently decrease. Around 400 seconds after the break initiation, the primary system pressure falls below the secondary system pressure, which is the end of the natural circulation phase. After that, the SG no longer behaves as a heat sink.

Calculated differential pressures in terms of the crossover leg downhill-side and uphill-side are compared with the measurements in **Figure 2.2-6** and **Figure 2.2-7**, respectively. The differential pressure is essentially due to the liquid level after the natural circulation period ends. In the experiment, the loop seal formed in the intact loop crossover leg was not cleared because the steam generated in the core was able to be vented through the bypass paths. Reference 12 describes that the core bypass fractions were 3.6% of primary loop flow for the lower plenum to upper plenum path, 6.6% for the inlet annulus (downcomer) to upper plenum path, and 1.3% for the reflood assist bypass valve at the test initiation. It was also noted that the valve leakage area for the reflood assist bypass changed with the pressure difference across the valve. Similar to the measurement, the M-RELAP5 calculation predicts that the loop seal in the intact loop crossover leg does not clear throughout the transient as shown in **Figure 2.2-6** and **Figure 2.2-7**.

The accumulator started injecting the safety coolant as the RCS pressure fell below the initial accumulator pressure around 640 seconds. The nitrogen gas in the accumulator tank expands and ejects the safety coolant to the RCS. The accumulator emptied of the water and the nitrogen began to enter the RCS at about 1750 seconds. These behaviors are well simulated in the M-RELAP5 calculation as shown in **Figure 2.2-8** for the tank pressure, and in **Figure 2.2-9** for the tank level, respectively. This validates the accumulator model implemented in M-RELAP5.

No fuel cladding heat-up was observed in the LOFT/L3-1 test or calculated with M-RELAP5 as shown in **Figure 2.2-10**.

Table 2.2-1 Steady-State Parameters for LOFT/L3-1

Parameter	Experiment[6,12]	M-RELAP5
Primary system pressure [MPa]	14.81 ± 0.04	14.82
Primary system mass flowrate [kg/s]	484.0 ± 6.3	484.0
Cold leg temperature [K]	554.0 ± 3	554.0
Hot leg temperature [K]	574.0 ± 1	573.0
Steam generator pressure [MPa]	5.43 ± 0.11	5.38
Steam generator mass flowrate [kg/s]	25.0 ± 0.4	25.0
Pressurizer level [m]	1.16[4)] ± 0.01	1.16
Core bypass fraction (LP to UP)[1)] [%]	3.6	3.45
Core bypass fraction (DC to UP)[2)] [%]	6.6	6.62
Core bypass fraction (RABV)[3)] [%]	1.3	1.30
Core power [MW]	48.9 ± 1.0	48.9

1) Core bypass fraction from lower plenum to upper plenum.
2) Core bypass fraction from downcomer to upper plenum.
3) Core bypass fraction through the reflood assist bypass valve.
4) Including the instrumentation elevation difference.

Table 2.2-2 Primary Test Chronology for LOFT/L3-1

Event	Experiment[6] (sec)	M-RELAP5 (sec)
Reactor scram	-2.15	-2.15
LOCA initiated	0.0	0.0
Primary coolant pumps tripped	0.04 ± 0.01	0.04
Scaled HPIS initiated	4.6 ± 0.5	0.95[1)]
Pressurizer empty	17.0 ± 1	23
Pump coastdown complete	19.0 ± 1	31[2)]
Accumulator injection initiated	633.6 ± 0.5	655.85
Accumulator liquid level below standpipe	1570.0 ± 1	1558
Accumulator line empty of fluid	1741.0 ± 1	1690
SCS steam bleed initiated	3622.5 ± 1	-
LPIS injection initiated	4240.0 ± 1	-
Experiment completed	4368.0 ± 1	-

1) Determined when the RCS pressure is less than 13.07 MPa.
2) Determined when the RCP head is less than 0.0m.

Figure 2.2-1 M-RELAP5 Noding Diagram for LOFT/L3-1 Analysis

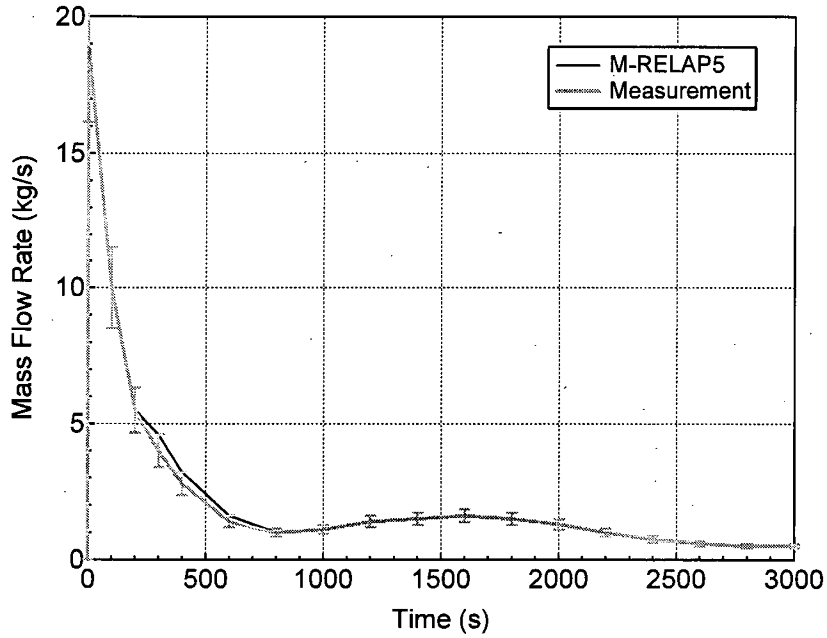

Figure 2.2-2 Break Mass Flowrate for LOFT/L3-1

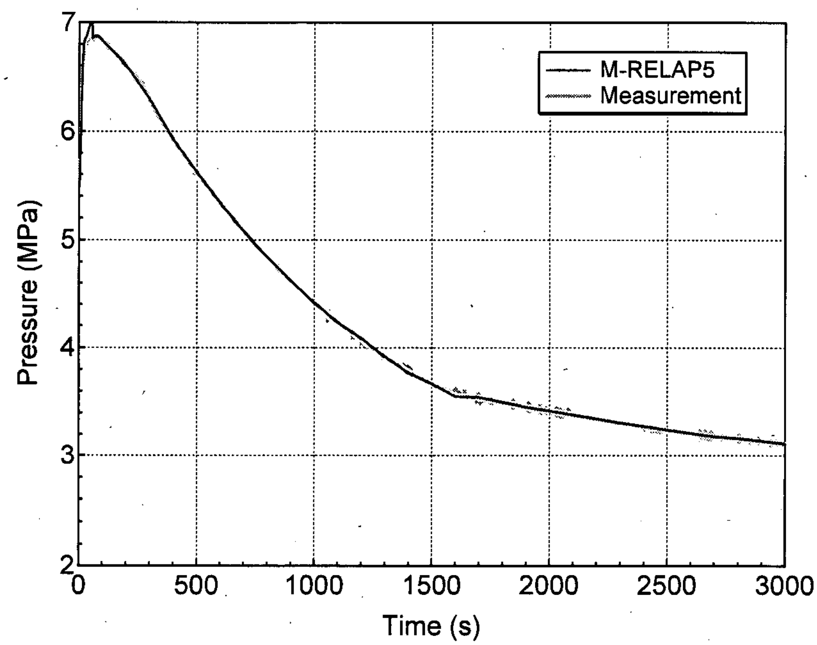

Figure 2.2-3 Secondary System Pressure for LOFT/L3-1

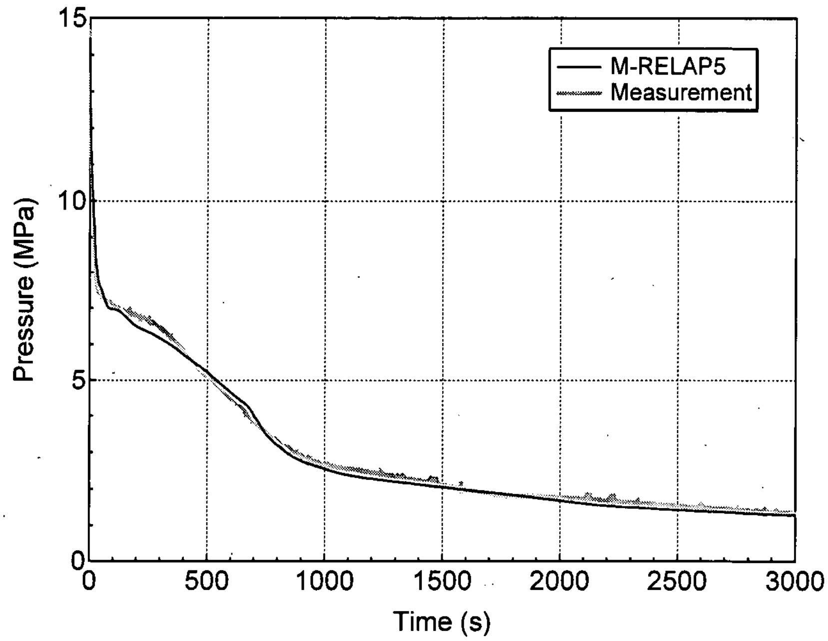

Figure 2.2-4 Primary System (Upper Plenum) Pressure for LOFT/L3-1

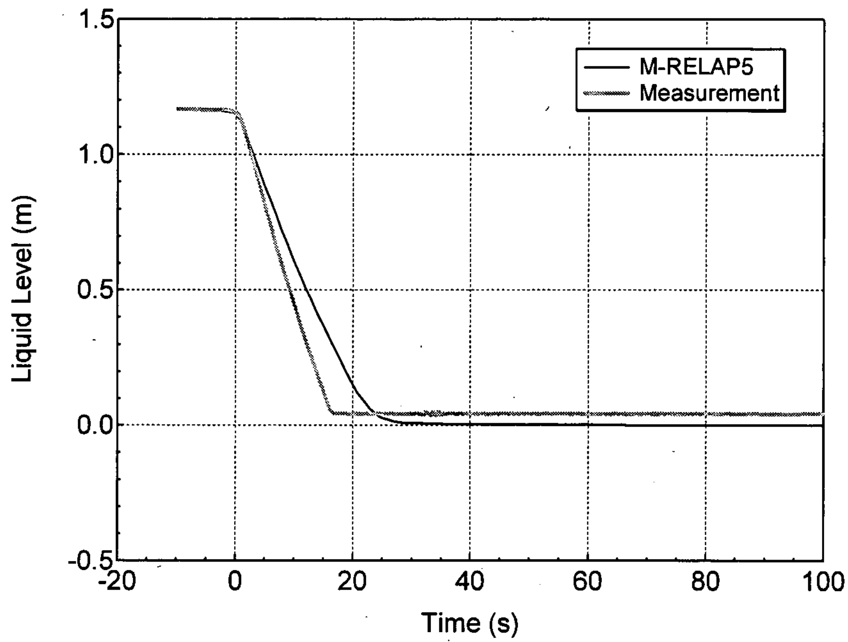

Figure 2.2-5 Pressurizer Liquid Level for LOFT/L3-1

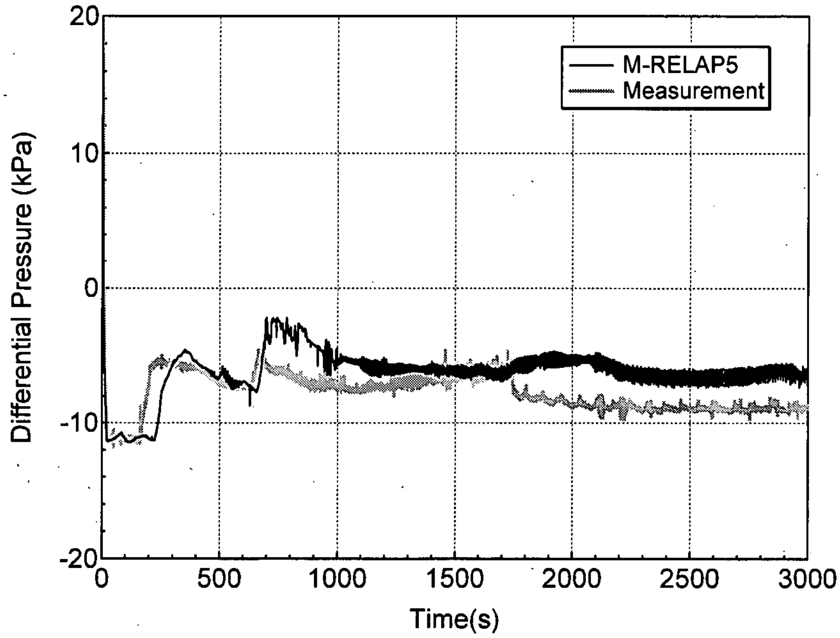

Figure 2.2-6 Differential Pressure in Intact Loop Crossover Leg for LOFT/L3-1 (SG-Side)

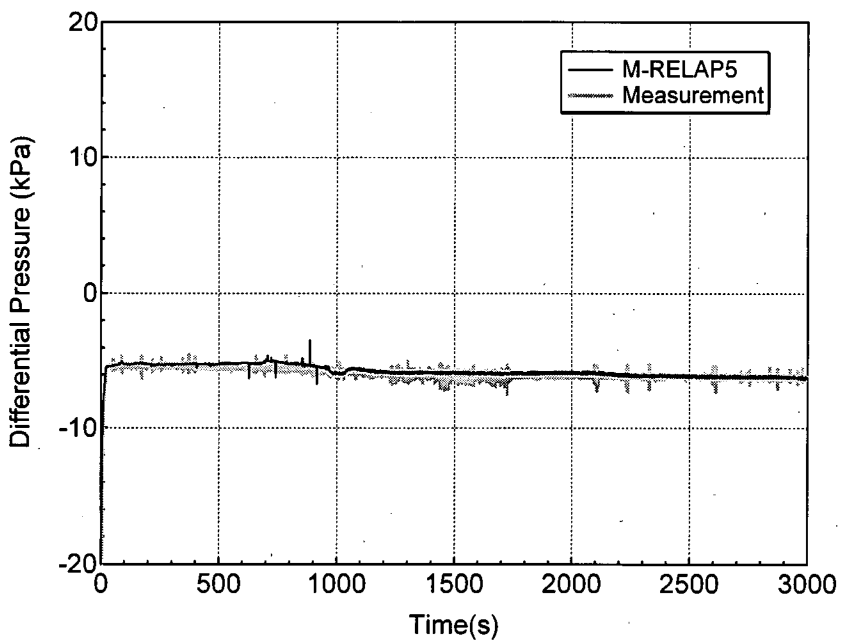

Figure 2.2-7 Differential Pressure in Intact Loop Crossover Leg for LOFT/L3-1 (RCP-Side)

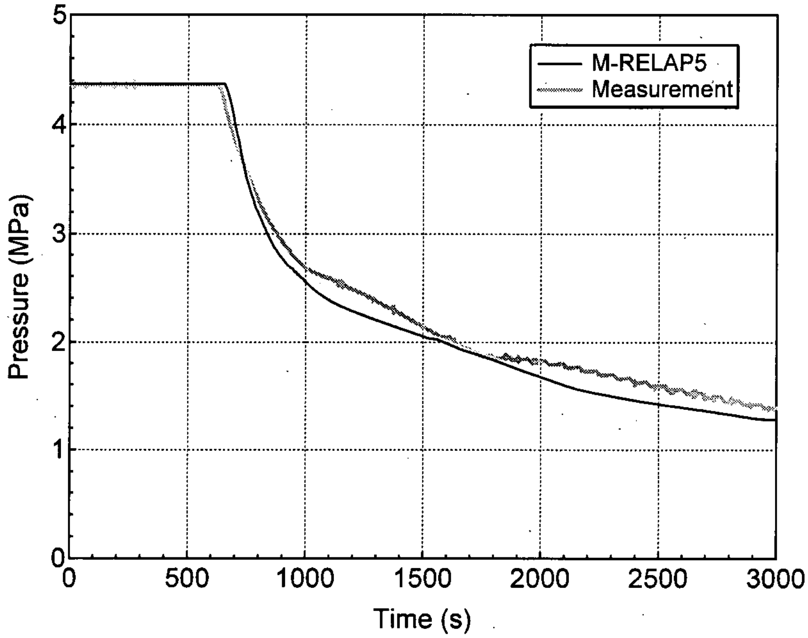

Figure 2.2-8 Accumulator Tank Pressure for LOFT/L3-1

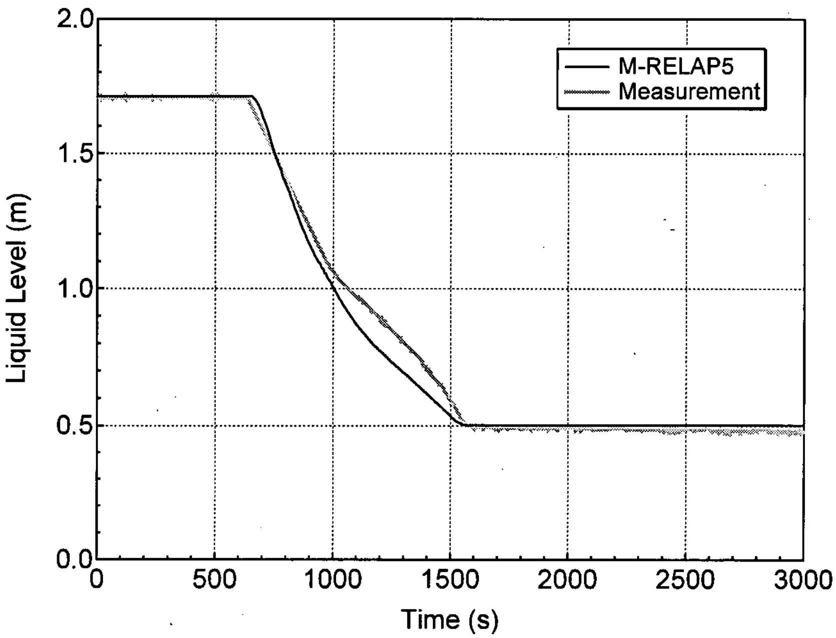

Figure 2.2-9 Accumulator Tank Water Level for LOFT/L3-1

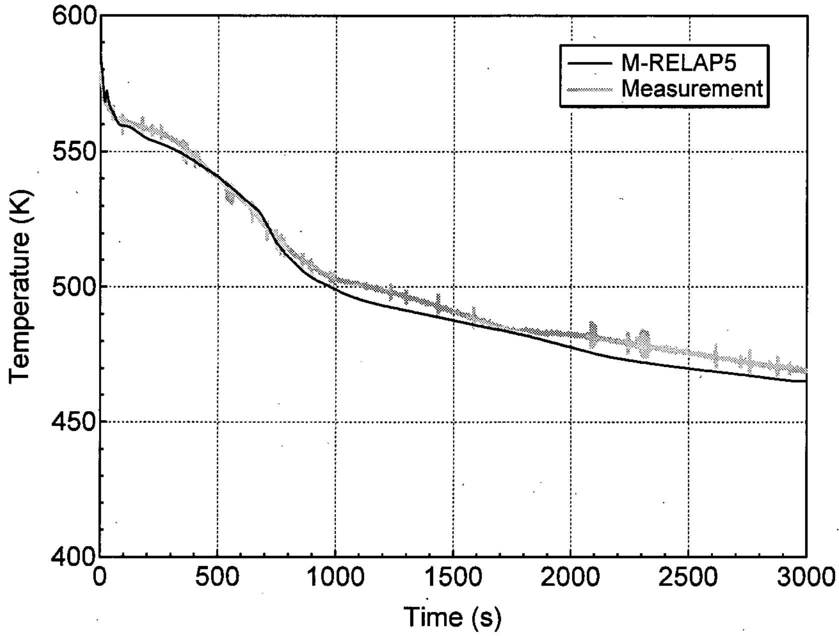

Figure 2.2-10 Fuel Cladding Temperature for LOFT/L3-1 (Z=62-in)

2.3 Summary

The LOFT/L3-1 experiment was simulated by using M-RELAP5 to validate the code's ability to predict the plant response occurring under SBLOCAs. The primary purpose is to assess the M-RELAP5 models and noding scheme, which are also applied to the plant analysis, using the experimental test data.

M-RELAP5 attained reasonable agreement compared to the measured RCS pressure and the pressurizer, loop seal, and accumulator behaviors. A large core bypass fraction caused the loop seal in the crossover leg to not clear during the LOFT/L3-1 test or in the M-RELAP5 calculation. In addition, M-RELAP5 predicted no cladding heat-up during the test, which was consistent with measured results.

Hence, it can be concluded that M-RELAP5 is able to reproduce the transient behavior, phenomena and processes of interest during the LOFT/L3-1 SBLOCA test.

3. SEMISCALE/S-LH-1 ANALYSIS

3.1 Test Description

3.1.1 Test Facility

The Semiscale Program was a part of the Water Reactor Research Test Program Division of EG&G Idaho, Inc., which conducted research of the thermal-hydraulic phenomena associated with simulated accident conditions in a PWR. The Semiscale Mod-2C system as structured during the S-LH-1 and S-LH-2 experiments simulated centerline cold leg small break loss-of-coolant accidents (5% SBLOCAs)[18].

Semiscale Mod-2C is a scaled model representation of a PWR plant, with a fluid volume of about 1/1705 of a PWR (**Figure 3.1-1**). The modified-volume scaling philosophy followed in the design of the Mod-2C system preserves most of the first-order effects thought important for SBLOCA transients. Most notably, the 1:1 elevation scaling of the Semiscale system is an important criterion for preserving the factors influencing signature response to a SBLOCA. Details of the scaling principle and the scaling results are described in Reference 19.

The Mod-2C system[18] consists of a pressure vessel with external downcomer and simulated reactor internals: an "intact loop," with a shell and inverted U-tube active steam generator (SG), pressurizer, and pump; and a "broken loop," including an active pump, active SG, and associated piping to allow break simulations. The intact loop simulates three "unaffected loops" of a four-loop PWR, and the broken loop simulates an "affected loop" in which the small break is assumed to occur. The break simulates a 5% cold-leg, centerline, communicative break in the loop piping between the pump and vessel. The intact loop SG consists of six inverted U-tubes, and the broken loop SG consists of two inverted U-tubes. Vessel internals include a simulated core, consisting of a 5 x 5 array of internally heated electric rods, of which 23 were powered as shown in **Figure 3.1-2**. The rods are geometrically similar to nuclear rods, with a heated length of 3.66 m (12 ft) and an outside diameter of 1.072 cm (0.42 in.).

3.1.2 Experimental Results

The Semiscale S-LH-1 test exhibited phenomena not generally observed during previous SBLOCA transients. In particular, a severe core uncovery occurred prior to clearing of the loop seals at the suction of the reactor coolant pumps. Following the initial depressurization, significant CCFL (counter-current flow limit) occurred in the hot leg piping and uphill-side of the SG U-tubes, particularly in the broken loop which consisted of smaller piping than that of the intact loop. During the loop seal phase, the core liquid level was depressed by the liquid holdup in the hot legs and uphill-side of the U-tubes. The upper portion of the core uncovered and heater rod cladding started heating up. The heat-up was terminated by an increase in the core liquid level following the loop seal clearance.

Since a large flow resistance occurred at the spray nozzle between the upper head and downcomer, relatively little steam was vented from the core, resulting in the significant core liquid level depression prior to loop seal clearance. Early seal clearance was observed in the intact loop because a larger amount of liquid flooded at the hot leg and U-tubes in the broken loop compared to the intact loop.

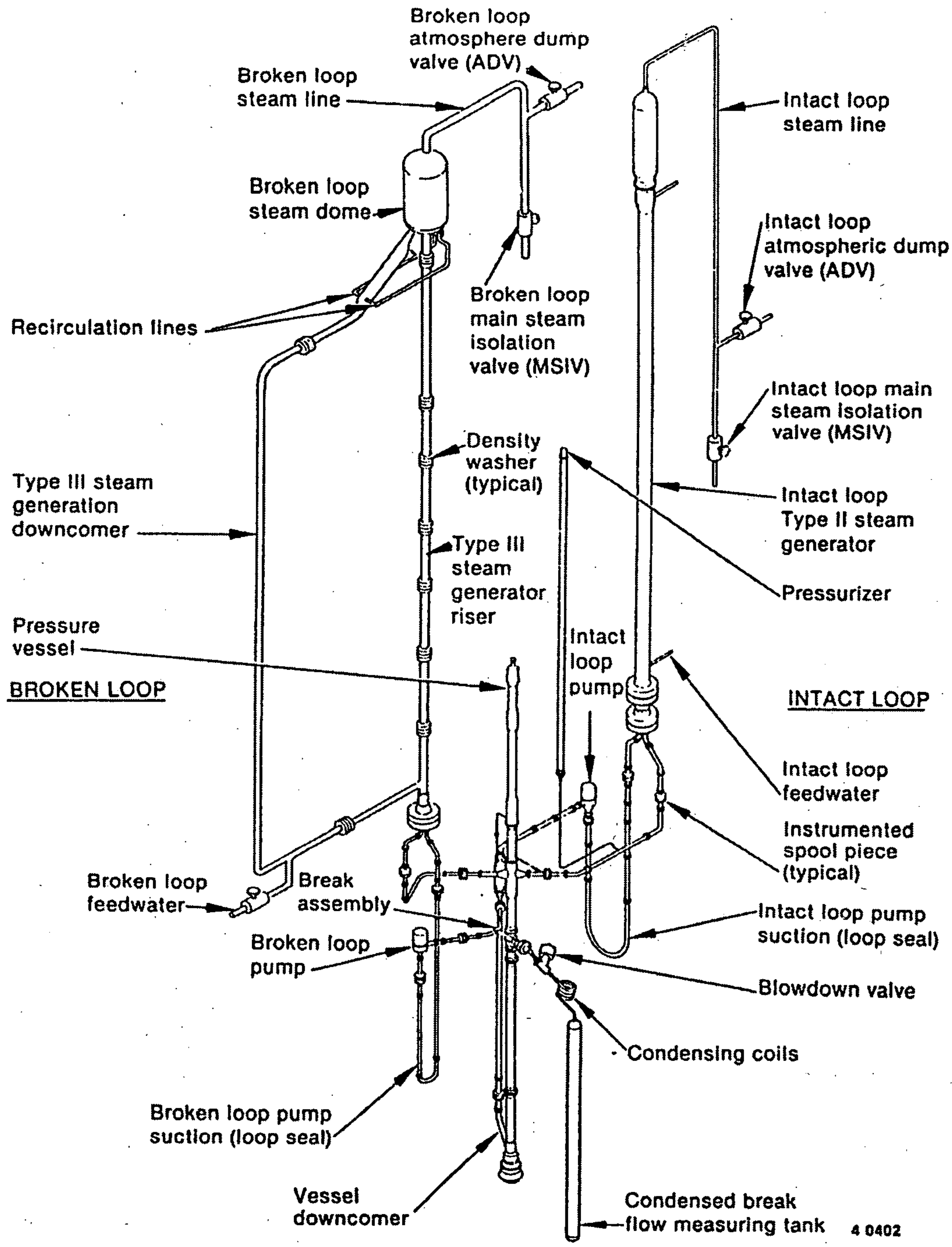

Figure 3.1-1 Semiscale Mod-2C System Configuration[7]

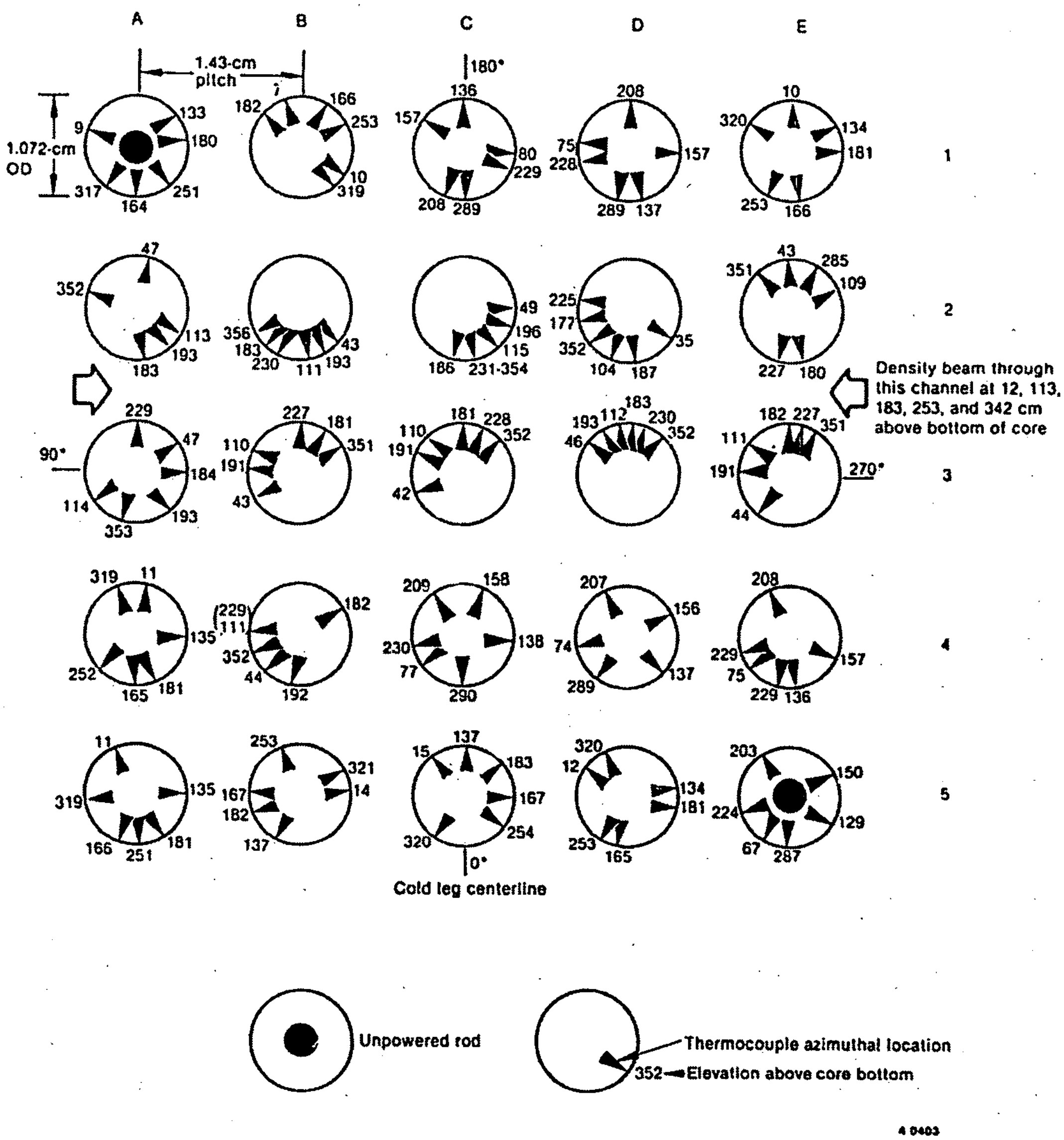

Figure 3.1-2 Semiscale Mod-2C Core Heater Rod Configuration[7]

3.2 M-RELAP5 Code Validation

3.2.1 Analysis Model

The Semiscale Mod-2C system is numerically represented by the noding diagram illustrated in Figure 3.2-1. The primary feature is that the system is nodalized with the same manner as for US-APWR SBLOCA calculations. The M-RELAP5 Semiscale Mod-2C model primarily consists of the a) reactor vessel, b) downcomer pipe, c) pressurizer, d) steam generator, e) intact loop, f) broken loop, and g) ECCS.

[

] The CCFL in the SG U-tubes is modeled by using the Wallis correlation[13], where *β=0.0, c=0.88, and m=1.0* are applied. This modeling is identical to that for the US-APWR plant calculation, because the geometric scaling of the SG U-tubes is almost identical between the Semiscale and US-APWR. [

]

The break flow history is imposed as a boundary condition which is specified by the input data based on the measurement so as to exclude uncertainties and distortions which are caused by applying the conservative Moody critical flow model as was done previously for the other IET calculations. The core power and secondary pressure are also given through input data tables for the present calculation.

3.2.2 Analysis Results

The steady-state calculation was performed by M-RELAP5. The converged plant parameters are listed in **Table 3.2-1** in which the calculation results are compared with the measurements. The table shows that M-RELAP5 accurately reproduces the steady-state condition prior to the transient test for the Semiscale/S-LH-1.

The chronology during the Semiscale/S-LH-1 test is listed in **Table 3.2-2**, where the experimental and calculated results are compared. The transient calculation was initiated by the simulated break flow data shown in **Figure 3.2-2**. The measured secondary system pressure was also given by a boundary condition as shown in **Figure 3.2-3**. M-RELAP5 well simulates the primary system pressure response as shown in **Figure 3.2-4**, indicating that the system mass and energy balances during the test are well reproduced by the M-RELAP5 calculation.

As described in Section 3.1.2, a complicated loop seal behavior was observed in the Semiscale/S-LH-1 test, where the coolant seal in the intact loop cleared first and the broken loop seal cleared about 90 s later. This loop seal behavior can be simulated by M-RELAP5 as shown in **Figure 3.2-5** and **Figure 3.2-6**, which demonstrate the code's ability to predict the loop seal behavior during SBLOCAs. It is noted that M-RELAP5 predicts transient decrease in the collapsed liquid level for the broken loop crossover leg as core liquid level depression during the loop seal period, while not observed in the measurement. However, the resultant core liquid level depression predicted by M-RELAP5 is deeper than the measurement, indicating that conservative prediction with respect to the loop seal PCT.

In addition, a severe reflux flooding occurred in the hot leg piping and SG-U-tubes in the S-LH-1 test and the core liquid level was significantly depressed during the loop seal phase. This was primary caused by the small core bypass flow fraction between the upper head and downcomer, which prevented the steam from venting from the core. This cause was experimentally validated by comparing the two tests, S-LH-1 (0.9% bypass) and S-LH-2 (3.0% bypass) of the Semiscale Program[18]. M-RELAP5 results are compared with the measurements from **Figure 3.2-7** to **Figure 3.2-9** in terms of the hot leg for the intact and broken loops, and the core liquid level, respectively. The severe flooding and core liquid depression can be well simulated by M-RELAP5.

As a result of the core liquid depression, the heater rod experienced the dryout and heat-up during the loop seal phase. This temperature excursion was terminated by increase in the core liquid level after the loop seal cleared as shown in Figure 3.2-9. Histories of the measured and calculated heater rod surface temperature are compared in **Figure 3.2-10** and the peak values are listed in **Table 3.2-3**. M-RELAP5 is capable of predicting the heater rod temperature behavior accurately. **[**

]

Figure 3.2-5 through Figure 3.2-9 show collapsed liquid levels. The approach used here is consistent with that used in Reference 18, where the calculated values were obtained by integrating liquid volume fraction distributions and the measured values were obtained from differential pressure measurements. Differences between the two methods affect the level comparisons before 90 s, when the pumps finish coasting down, because the measured differential pressures are affected by flow. Also, the difference between the calculated and measured cladding temperatures shown in Figure 3.2-10 prior to scram is caused by the comparison of a calculated surface temperature with a measured temperature inside the heater rod.

Table 3.2-1 Steady-State Parameters for Semiscale/S-LH-1

Parameter	Experiment[18]	M-RELAP5
Pressurizer pressure [MPa]	15.47 ± 0.14	15.47
Core ΔT	37.65 +1.5/-0.6	37.52
Intact loop flow rate	7/13	7.11
Broken loop flow rate	2.35	2.34
Intact loop cold leg temperature [K]	562.12 ± 2	562.06
Broken loop cold temperature [K]	564.05 ± 2	564.07
Intact loop Steam generator pressure [MPa]	5.72 ± 0.07	5.72
Broken loop Steam generator pressure [MPa]	6.08 ± 0.07	6.08
Pressurizer level [cm]	395 ± 14	394.95
Core bypass fraction [%]	0.9	0.9
Core power [MW]	2014.75 ± 0.15	2014.75

Table 3.2-2 Primary Test Chronology for Semiscale/S-LH-1

Event	Experiment[18] (sec)	M-RELAP5 (sec)
Pressurizer at 12.6MPa (trip level)	14.67	22.10
Core scram	19.57	26.70
Pump coastdown initiated		
Intact loop	21.35	26.80
Broken loop	20.76	26.45
HPIS initiated		
Intact loop	41.60	48.40
Broken loop	40.98	48.40
Minimum core liquid level reached	172.6	179
Intact loop pump suction cleared	171.4	183
Broken loop pump suction cleared	262.3	262

Table 3.2-3 Summary of PCTs during Loop Seal for Semiscale/S-LH-1

	Time (s)	PCT (K)
Measured PCT	182.4	624.4
M-RELAP5	191.5	634.1

Figure 3.2-1 Nodalization Diagram for M-RELAP5 Calculations

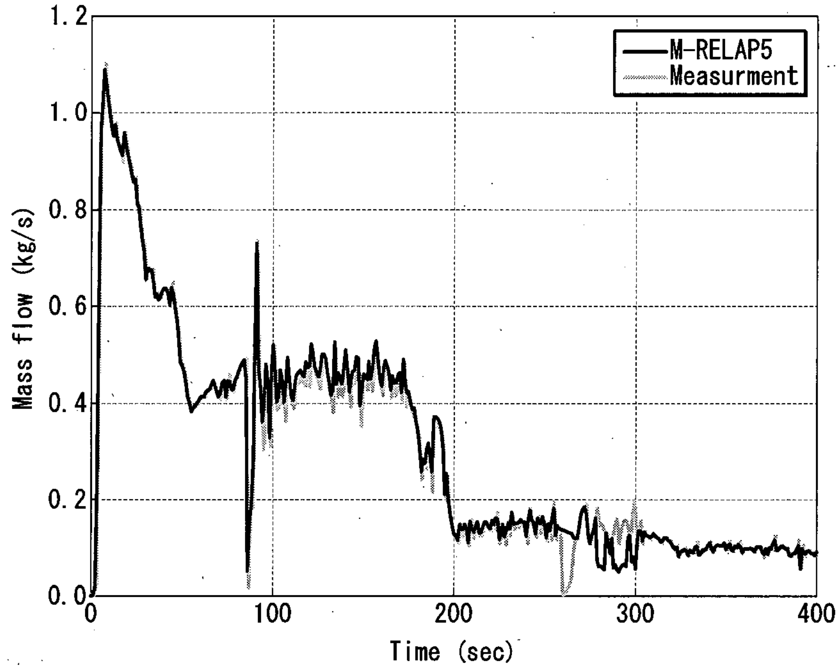

Figure 3.2-2 Break Mass Flowrate for Semiscale/S-LH-1

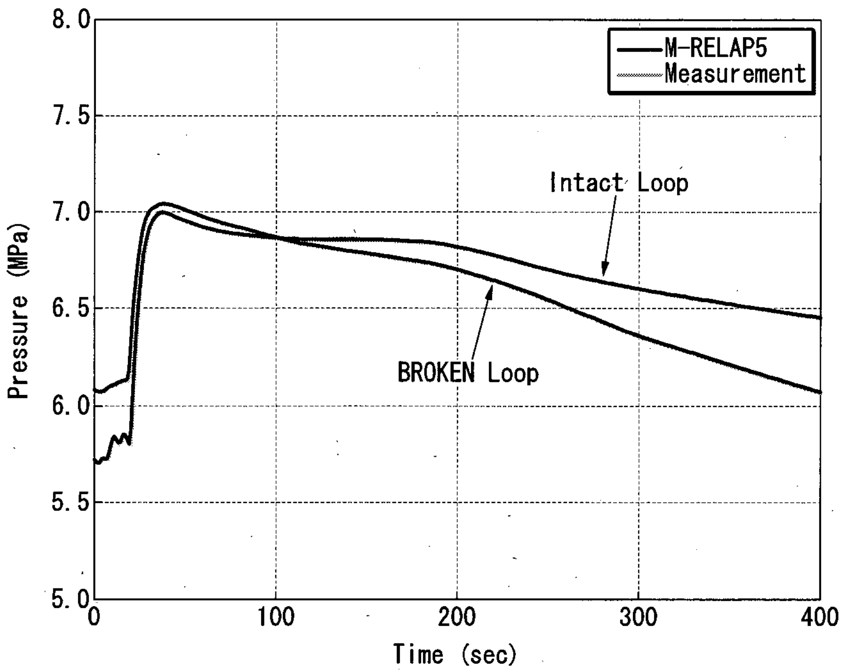

Figure 3.2-3 Secondary System Pressure for Semiscale/S-LH-1

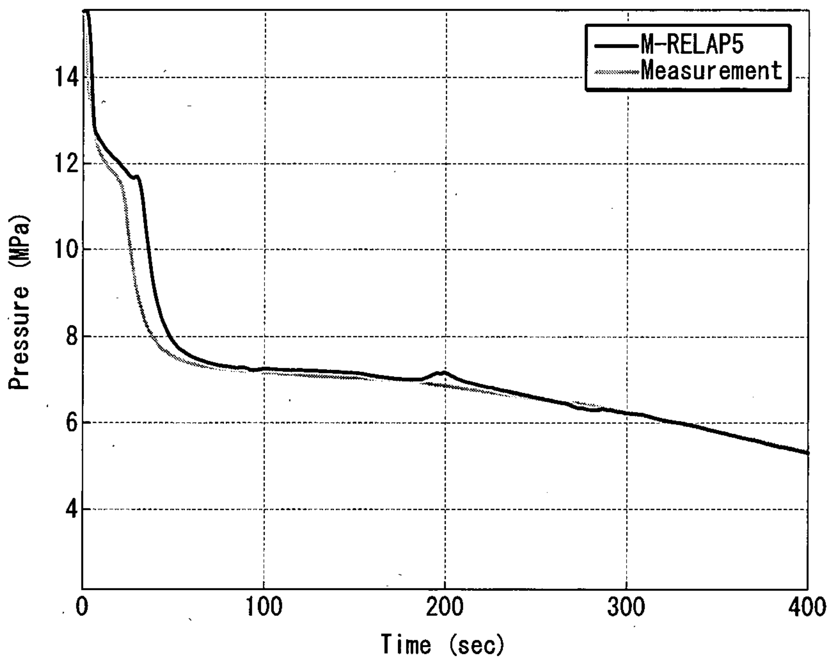

Figure 3.2-4 Primary System Pressure for Semiscale/S-LH-1

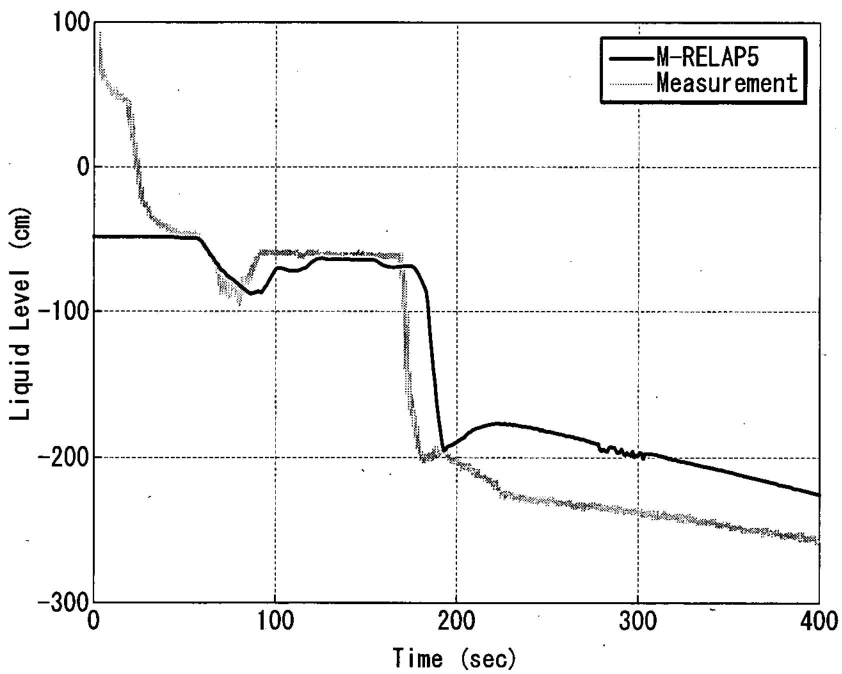

Figure 3.2-5 Collapsed Level in Uphill-Side of Intact Loop Crossover Leg for Semiscale/S-LH-1

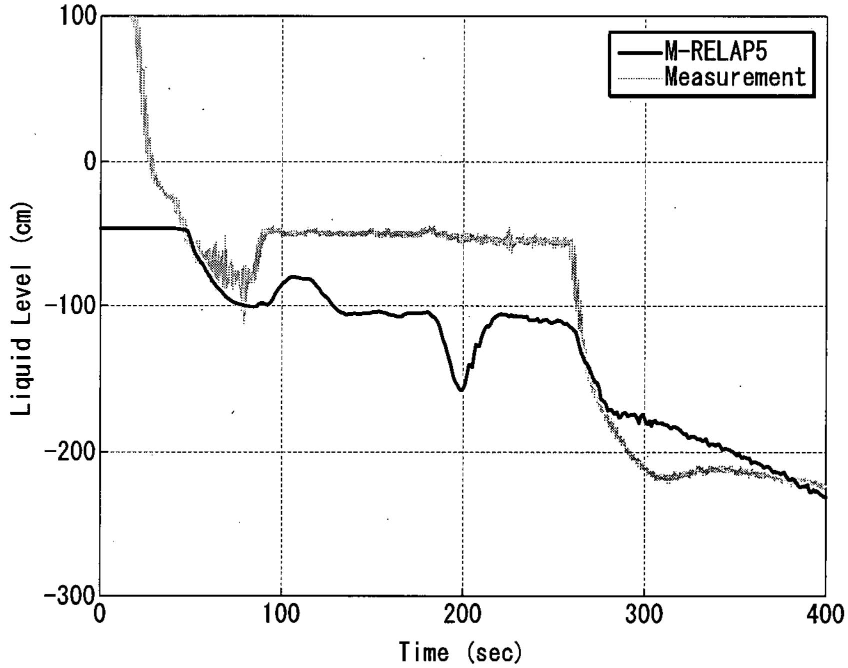

Figure 3.2-6 Collapsed Level in Uphill-Side of Broken Loop Crossover Leg for Semiscale/S-LH-1

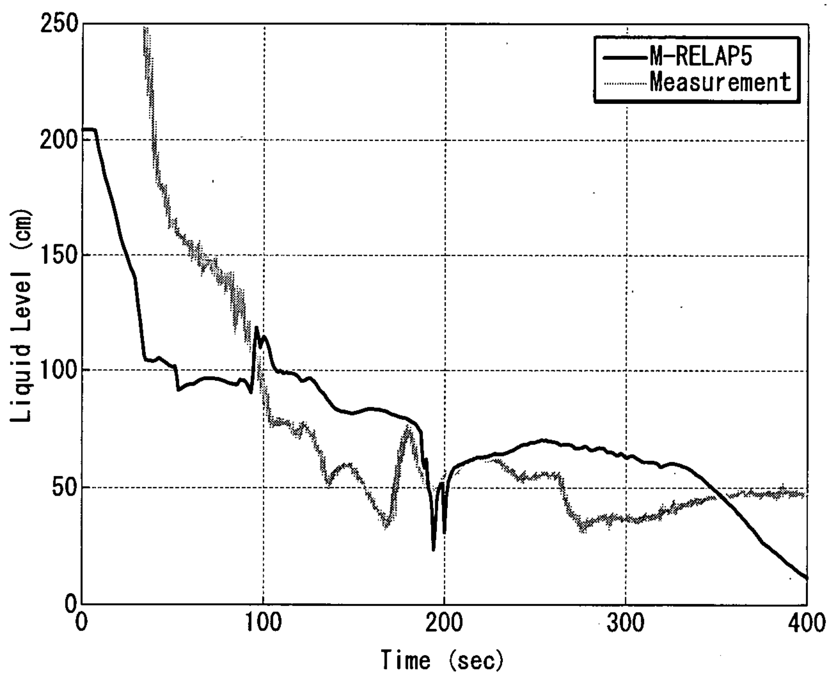

Figure 3.2-7 Collapsed Level in Intact Loop Hot Leg for Semiscale/S-LH-1

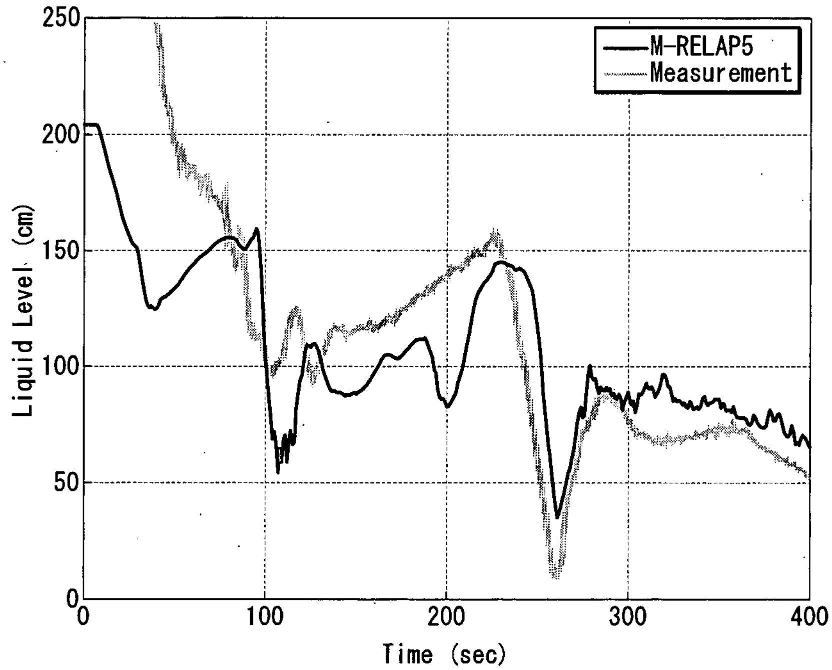

Figure 3.2-8 Collapsed Level in Broken Loop Hot Leg for Semiscale/S-LH-1

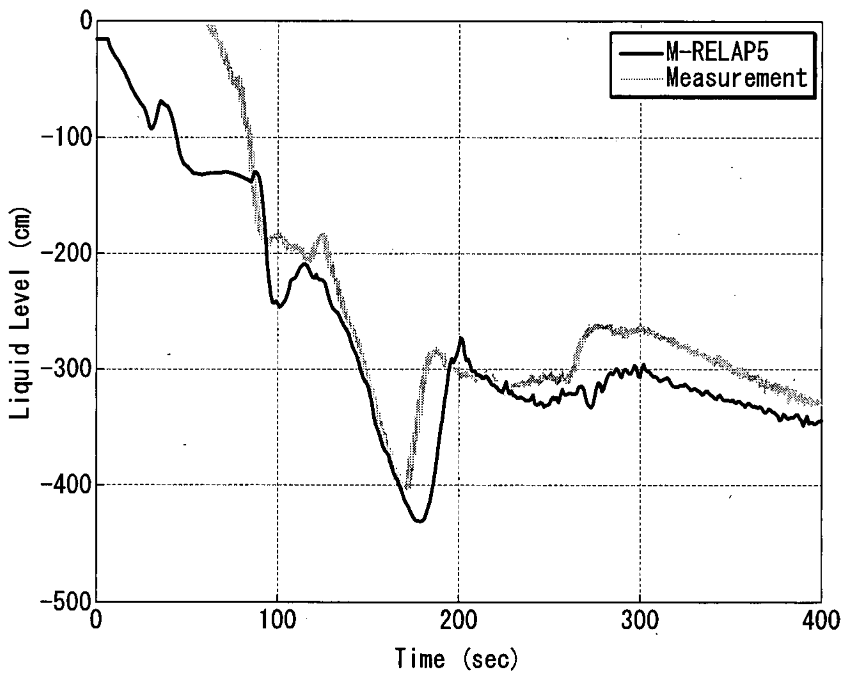

Figure 3.2-9 Core Collapsed Level for Semiscale/S-LH-1

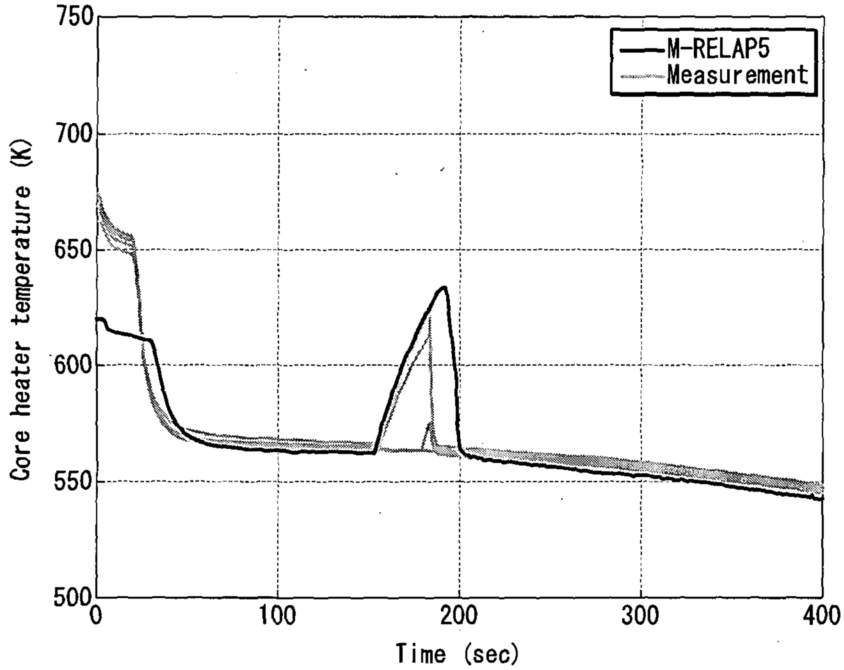

Figure 3.2-10 Core Cladding Temperature at 8.3-ft (253cm) Elevation

3.3 Summary

The simulation of Semiscale S-LH-1 was performed by M-RELAP5. The results demonstrate that M-RELAP5 well predicted the complicated plant responses, including the loop seal behavior. In particular, the severe core depression and heater rod temperature excursion during the loop seal phase were well reproduced by M-RELAP5, showing its high applicability to the PWR SBLOCA safety analysis.

4. CONCLUSIONS

In the process of M-RELAP5 code development, the code and its plant modeling scheme have been assessed via the code validation analyses using the IET data from ROSA-IV/LSTF SB-CL-18[1]. In the present report, additional code validations using the IET data from the LOFT/L3-1 and Semiscale/S-LH-1 are described.

For the LOFT/L3-1 test analysis, M-RELAP5 accurately predicted the plant responses, including the blowdown depressurization, the time-period of natural circulation, the accumulator behavior and so on. Since the large core bypass fraction efficiently facilitated the steam venting from the core, the core liquid level was not significantly depressed in the test, which was also reproduced by M-RELAP5 appropriately.

For the Semiscale/S-LH-1 test analysis, several phenomena and processes unique to PWR SBLOCAs appeared, including reflux flooding in the SG inlet and uphill-side of SG U-tubes, and the loop seal formation and clearance. In addition, the severe core level depression and the heater rod temperature excursion that occurred during the loop seal phase in the test were well reproduced by M-RELAP5.

The detailed scaling analyses that were done for the ROSA-IV/LSTF test were not performed for the LOFT/L3-1 and Semiscale/S-LH-1 tests in the present report. However, the LOFT and Semiscale Mod-2C test facilities were designed to be scaled to the typical Westinghouse 4-loop PWR plant, to which the US-APWR is scalable as mentioned in Reference 2. Hence, these code validation results, including the ROSA-IV/LSTF SB-CL-18 test analysis, demonstrate that M-RELAP5 and its plant modeling scheme adequately simulate the SBLOCA tests from various IET facilities with different scaling ratios, indicating that M-RELAP5 is reliably applicable to the US-APWR SBLOCA analysis.

5. REFERENCES

1 Mitsubishi Heavy Industries, Ltd., 'Small Break LOCA Methodology for US-APWR,' MUAP-07013-P (R0), July 2007.

2 Mitsubishi Heavy Industries, Ltd., 'Scaling Analysis for US-APWR Small Break LOCAs - Part 1 -,' UAP-HF-09472, September 2009.

3 Mitsubishi Heavy Industries, Ltd., 'Scaling Analysis for US-APWR Small Break LOCAs - Part 2 -,' UAP-HF-09541, November 2009.

4 Mitsubishi Heavy Industries, Ltd., 'Scaling Analysis for US-APWR Small Break LOCAs - Part 3 -,' UAP-HF-09xxx, December 2009.

5 US Nuclear Regulatory Commission, 'Clarification of TMI Action Plan Requirements, ' NUREG-0737, November 1980.

6 P. D. Bayless et al., 'Experimental Data Report for LOFT Nuclear Small Break Experiment L3-1,' NUREG/CR-1145, EGG-2007, January 1980.

7 G. G. Loomis, 'Experimental Operating Specifications for Semiscale Mod-2C 5% Small Break Loss-of-Coolant Experiment S-LH-1,' EGG-SEMI-6813, February 1985.

8 W. W. Tingle, 'Experiment Operating Specifications for Semiscale Mod-2A 5% Break Experiment S-UT-8,' EGG-SEMI-5685, December 1981.

9 L. J. Ybarrando et al., 'Examination of LOFT Scaling,' ASME Annual Meeting, 74-WA/HT-53, New York, NY, November 17-22, 1974.

10 M. L. Russell, 'LOFT Fuel Modules Design, Characterization, and Fabrication Program,' TREE-NUREG-1131, June 1977.

11 D. L. Reeder, 'LOFT System and Test Description (5.5-ft Nuclear Core 1 LOCEs),' NUREG/CR-0247, TREE-1208, July 1978.

12 K. G. Condie et al., 'Four-Inch Break Loss-of-Coolant Experiments: Posttest Analysis of LOFT Experiment L3-1, L3-5 (Pumps Off), and L3-6 (Pumps On),' EGG-LOFT-5480, October 1981.

13 G. B. Wallis, 'One Dimensional Two Phase Flow,' McGraw-Hill, 1969.

14 C. L. Tien, et al., 'Flooding in Two-Phase Countercurrent Flows,' EPRI NP-1283, December 1979.

15 P. S. Damerell et al., 'Use of Full-Scale UPTF Data to Evaluate Scaling of Downcomer (ECC Bypass) and Hot Leg Two-Phase Flow Phenomena,' NUREG-CP-0091 vol. 4.

16 F. J. Moody, 'Maximum Flow Rate of a Single Component, Two-Phase Mixture,' J. of Heat Transfer, Trans. ASME, Series C, Vol. 87, No. 1, pp134-142, February 1965.

17 Mitsubishi Heavy Industries, Ltd. 'MHI's 2nd Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 09/08/2009', Response to REQUEST 8.2.1-12, UAP-HF-09512, November 2009.

18 G, G. Loomis and J. E. Streit, 'Results of Semiscale Mod-2C Small-Break (5%) Loss-of-Coolant Accident Experiments S-LH-1 and S-LH-2,' NUREG/CR-4438, EGG-2424, R2, November 1985.

19 R. A. Shaw et al., 'A Description of the Semiscale Mod-2C Facility, Including Scaling Principle and Current Measurement Capabilities,' EGG-M-11485, January 1985.

20 H. T. Kim and J. C. No, 'Assessment of RELAP5/MOD3.2.2γ against flooding database in horizontal-to-inclined pipes,' Annals of Nuclear Energy, Vol. 29, Issue 7, May 2002.