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RELAP5 and TRACE Simulation of Hot Leg Break LOCA Experiment on LSTF

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ABSTRACT

Confidence in the computational tools and establishment of their validity for a given application depends on the assessment. The purpose of this study is therefore to independently assess the TRACE computer code for hot leg break test. A pressurized water reactor (PWR) hot leg break loss-of-coolant accident experiment SB-HL-02 was performed on the Large Scale Test Facility (LSTF) in the Rig of Safety Assessment-IV (ROSA-IV) program with a break size equivalent to 10% cold leg cross sectional area. For calculations the RELAP5/MOD3.3 Patch 5 and TRACE V5.0 Patch 4 computer codes were used. The RELAP5/MOD2 input model was obtained within the framework of International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP) on Evaluation of Uncertainties in Best Estimate Accident Analysis (2006-2010). The obtained input model was first adapted to RELAP5/MOD3.3 and then converted to TRACE using Symbolic Nuclear Analysis Package (SNAP), requiring also manual corrections. 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 results suggest that TRACE calculation is comparable to RELAP5 calculations and that results obtained by both codes agree well with the experimental data. Finally, it was also demonstrated that advanced SNAP graphical user interface has the capabilities to graphically present complex phenomena like collapsed liquid level distribution in the loop, helping to understand natural circulation flow in different regimes.

TABLE OF CONTENTS

LIST OF FIGURES

Page

LIST OF TABLES

EXECUTIVE SUMMARY

Confidence in the computational tools, and establishment of their validity for a given application depends on the assessment. The purpose of this study is therefore to independently assess the TRACE computer code for hot leg break test. A pressurized water reactor (PWR) hot leg break loss-of-coolant accident (LOCA) simulation experiment SB-HL-02 was performed on the Large Scale Test Facility (LSTF) in the Rig of Safety Assessment-IV (ROSA-IV) program with a break size equivalent to 10% cold leg cross sectional area. For calculations the RELAP5/MOD3.3 Patch 5 and TRACE V5.0 Patch 4 computer codes were used.

The RELAP5/MOD2 input model was obtained within the framework of International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP) on Evaluation of Uncertainties in Best Estimate Accident Analysis (2006-2010). The obtained input model was first adapted to RELAP5/MOD3.3 and then converted to TRACE using Symbolic Nuclear Analysis Package (SNAP). Manual corrections were also needed (break model, tees for accumulator connection, steady-state calculation).

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 was located at the side of horizontal hot leg pipe in the loop without pressurizer. Total failure of high pressure injection system and auxiliary feedwater as well as loss of off-site power concurrent with the scram were assumed as the experimental conditions. The accident started with break valve opening. Scram signal and safety injection signal were generated. Only passive accumulators and low pressure safety injection were available for injection. Core heatup was experienced before first injection. The experimental data were available for the first 1000 s.

In the report the comparison between calculated and experimental data is shown. Base case and calculations by adjusting the break flow coefficients to fit the primary pressure have been performed. The results suggest that TRACE calculation is comparable to RELAP5 calculations and that results obtained by both codes agree well with the experimental data, especially for calculations in which the break flow coefficients were adjusted.

For results presentation the SNAP animation of the ROSA/LSTF facility for RELAP5 has been used. It was demonstrated that advanced SNAP graphical user interface has the capabilities to graphically present complex phenomena like collapsed liquid level distribution in the loop, helping to understand natural circulation flow in different regimes. Even more, comparison with measured values could be done if data available.

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The authors acknowledge the financial support from Krško Nuclear Power Plant and Slovenian Nuclear Safety Administration within CAMP program (project no. POG-U3-KE-R4/104/12 – also NEK no.: 3120118) and the financial support from the Slovenian Research Agency (research core funding no. P2-0026).

ABBREVIATIONS

ACC	accumulator
AFW	auxiliary feedwater
ASCII	American Standard Code for Information Interchange
BETHSY	Boucle d'Etudes Thermohydrouliques Systeme
CRP	Coordinated Research Project
ECCS	emergency core cooling system
HPI	high pressure injection
IAEA	International Atomic Energy Agency
LOCA	loss-of-coolant accident
LPI	low pressure injection
LSTF	Large Scale Test Facility
MFW	main feedwater
MSIV	main steam isolation valve
PORV	power operated relief valve
PRZ	pressurizer
PWR	Pressurized Water Reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RELAP	Reactor Excursion and Leak Analysis Program
ROSA	Rig of Safety Assessment
RPV	reactor pressure vessel
SBLOCA	small-break loss-of-coolant accident
SG	steam generator
SI	safety injection
SNAP	Symbolic Nuclear Analysis Package
TRACE	TRAC/RELAP Advanced Computational Engine

1 INTRODUCTION

Confidence in the computational tools and establishment of their validity for a given application depends on the assessment. For independent assessment of the RELAP5 and TRACE computer codes, the SB-HL-02 hot leg break test was selected, which has not been used by code developers for TRACE code assessment. During such test natural circulation phenomena in the primary loops, including those in the two-phase stratified and countercurrent flow regimes could be investigated. Because these phenomena are significantly dependent on facility scale and geometry, large-scale tests in a primary system geometry representative of operational nuclear power plants are required. The Rig of Safety Assessment (ROSA)/Large Scale Test Facility (LSTF) is the world largest integral test facility and therefore satisfies requirement regarding large-scale tests. For calculations the RELAP5/MOD3.3 Patch 5 and TRACE V5.0 Patch 5 computer codes were used. In this way, the TRACE computer code agreement with experimental data could be qualitatively compared to RELAP5 computer code agreement with experimental data. The post test simulations with RELAP5 have been already performed in the past. For example, the study (Ref. 1) present simulations of the test with the RELAP5/MOD3.2, while study (Ref. 2) present simulations with earlier version of RELAP5/MOD3.3 (released before 2010). However, the author is not aware of TRACE assessment against ROSA/LSTF SB-HL-02 hot leg break test, which is main purpose of this paper.

In Section 2 the methods used are described. First, the ROSA-IV/LSTF facility is described. The SB-HL-02 test, simulating hot leg break on the ROSA-IV/LSTF facility with the break size equivalent to 10% cold leg break is described. Then the RELAP5 and TRACE thermal-hydraulic system computer codes used are briefly described, followed by input model description for both computer codes and initial and boundary conditions. The calculations have been performed with TRACE computer code using default option for critical flow model, and RELAP5 computer code using both Henry-Fauske and Ransom-Trapp critical flow model. For each code, default and user-defined critical flow model coefficients have been used used, what gives in total six calculations of SB-HL-02 test. Then, results of the hot leg LOCA calculations are presented in Section 3, including graphical presentation using advanced SNAP graphical user interface and discussion of the result. Finally, main conclusions are drawn.

2 METHODS USED

2.1 ROSA-IV/LSTF Facility Description

The ROSA/LSTF (Ref. 3) was designed to simulate thermal-hydraulic phenomena peculiar to small-break loss-of-coolant accidents (SBLOCAs) and operational transients by having prototypical component elevation differences, large loop-piping diameters, prototypical primary-pressure levels, and simulated system controls. The ROSA/LSTF has volumes scaled at 1/48 of a typical 3423 MWt 4-loop pressurized water reactor (PWR) plant (see Figure 1). The four primary loops in the reference PWR are represented by two symmetric loops in the ROSA/LSTF, each one including an active steam generator and an active reactor coolant pump. The component elevations are preserved full scale to simulate natural circulation phenomena peculiar to SBLOCAs and transients. The ROSA/LSTF initial core power is 10 MW because of the limitation in the capacity of the power supply in the test facility. This initial power corresponds to 14% of the volumetrically scaled (1/48) nominal core power of the PWR. To obtain prototypical initial fluid temperatures, core flow rate in ROSA/LSTF is set to 14% of the scaled nominal flow rate of the PWR. Besides the major components, the reactor protection systems and equipment controls, the secondary and various auxiliary systems are included, too. These systems include emergency core-cooling systems, feedwater, condensate and steam systems together with component service systems such as the cooling water, instrument air, water purification, etc.

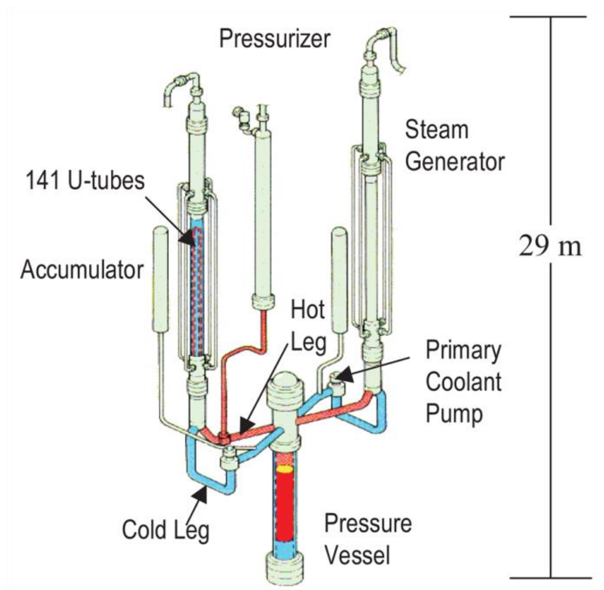

Figure 1 Schematic View of the Large Scale Test Facility (LSTF) (Ref. 4)

2.2 SB-HL-02 Test Description

The SB-HL-02 test was conducted on June 30, 1987 using the LSTF facility in ROSA-IV program. The break size was equivalent to 10% cold leg break using 31.9 mm ID sharp-edge orifice at downstream of horizontal pipe connected to hot leg break nozzle (see Figure 2) in loop without pressurizer. The break size of 10% is the largest among integral experiments on PWR break loss of coolant accidents (LOCAs) that are being performed at the ROSA/LSTF. Total failure of high pressure injection system and auxiliary feedwater as well as loss of off-site power concurrent with the scram were assumed as the experimental conditions.

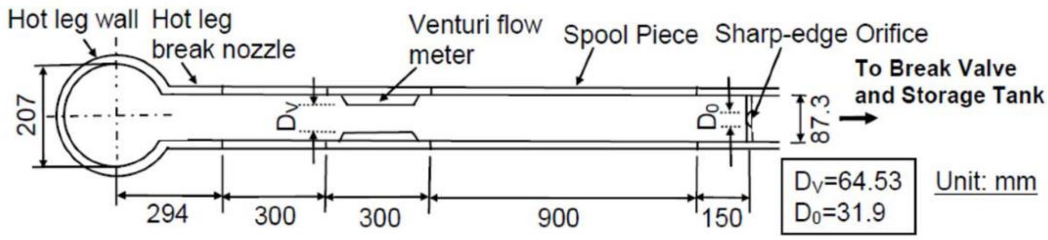

Figure 2 Configuration of Break Unit (Ref. 4)

Detailed thermal-hydraulic data on a PWR hot leg break LOCA were obtained through the ROSA/LSTF experiment. The hot leg break LOCA transient was characterized by vapor condensation on accumulator coolant in cold legs induced loop seal clearing and effectively enhanced core cooling thereafter. The experimental data were open to publics through publishing the data report (Ref. 4).

The main sequence of events is shown in Table 1. The transient started with break valve opening at 0 s. At pressuriser pressure 12.97 MPa and 12.27 MPa the scram and safety injection signal are simulated, respectively. At 42 s core power decay is started. At 160 s cold leg fluid started flashing and primary pressure is lower than steam generator (SG) secondary-side pressure. Then primary coolant pumps were stopped at 261 s. At 300 s to 350 s the core uncovered and superheating occurred, with loop seal clearing at 340 s. At 900 s low pressure injection (LPI) system in loop with pressurizer (PRZ) is initiated at pressurizer pressure 1.29 MPa. The measured data have been provided for 1000 s.

Table 1 Sequence of Major Events during SB-HL-02 Test

Time (s)	Event
0	Break valve open
6	Scram signal (primary pressure = 12.97 MPa)
9	Safety injection signal (primary pressure = 12.27 MPa)
10	Break flow from single-phase liquid to two-phase flow
42	Core power decay started
160	Cold leg fluid started flashing, primary pressure lower than steam generator (SG) secondary-side pressure
261	Primary coolant pumps stopped
280	Break flow to single-phase vapor
300 to 350	Core uncovery, superheating
330	Initiation of accumulator system (primary pressure = 4.51 MPa)
340	Loop seal clearing
900	Initiation of low pressure injection (LPI) system in loop with PZR (pressure vessel lower plenum pressure =1.29 MPa)

2.3 <u>Computer Codes Used</u>

At the time of calculations the latest RELAP5 and TRACE thermal hydraulic system codes were used: U.S. NRC RELAP5/MOD3.3 Patch 5 (Ref. 5) and TRACE Version 5.0 Patch 4 (Ref. 6), respectively. The RELAP5/MOD3.3 Patch 5 has built in two models for critical flow: Henry-

Fauske critical flow model which is default and Ransom-Trapp critical flow model (Option 50 need to be used). The TRACE has built in as default the critical flow model, which is extension of Ransom and Trapp critical flow model.

2.4 <u>RELAP5 Input Model</u>

Sample input deck for RELAP5/MOD3.2 code analysis of ROSA/LSTF experiment has been obtained in the frame of International Atomic Energy Agency (IAEA) Coordinated Research Programme (CRP) on Evaluation of Uncertainties in Best Estimate Accident analysis (Ref. 4). The RELAP5/MOD3.2 input model was first adapted to RELAP5/MOD3.3 computer code, for which also animation model has been created using Symbolic Nuclear Analysis Package (SNAP) (Ref. 7). The adapted RELAP5 input model of ROSA/LSTF consists of 159 Hydraulic Components and 44 Heat Structures in terms of SNAP (see Figure 3). In terms of RELAP5, the input model consists of 212 volumes, 221 junctions and 213 heat structures with 1305 mesh points.

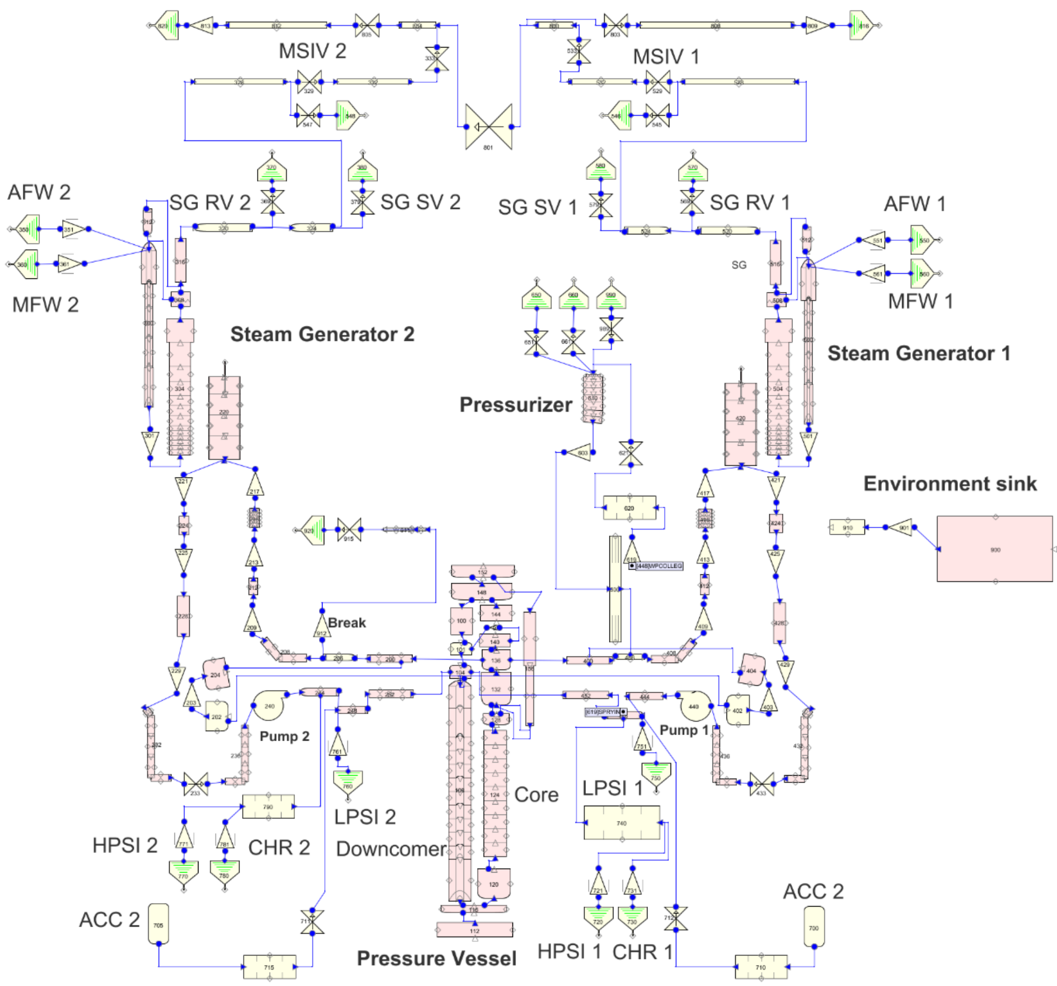

Figure 3 RELAP5 Input Model of ROSA/LSTF Represented by SNAP

Modeling of the primary side includes the reactor pressure vessel (RPV), both loops, the pressurizer, pressurizer spray lines and valves, pressurizer power operated relief valves (PORV) and pressurizer safety valve and reactor coolant pump (RCP). Emergency core cooling system (ECCS) piping includes high pressure safety injection (HPI) pumps, accumulators (ACCs), and low pressure safety injection (LPI) pumps. The secondary side consists of the SG secondary side, main steam line, main steam isolation valves (MSIVs), SG relief and safety valves, and main feedwater (MFW) piping. The turbine is represented by time dependent volume. The MFW and AFW (auxiliary feedwater) pumps are modeled as time dependent junctions.

2.5 TRACE Input Model

The TRACE input model was obtained from RELAP5 through conversion by SNAP. Manual corrections like break model, tees for accumulator connection, and corrections for steady-state calculation have been performed. The TRACE input model, shown in Figure 4, consists of 171 Hydraulic Components and 44 Heat Structure.

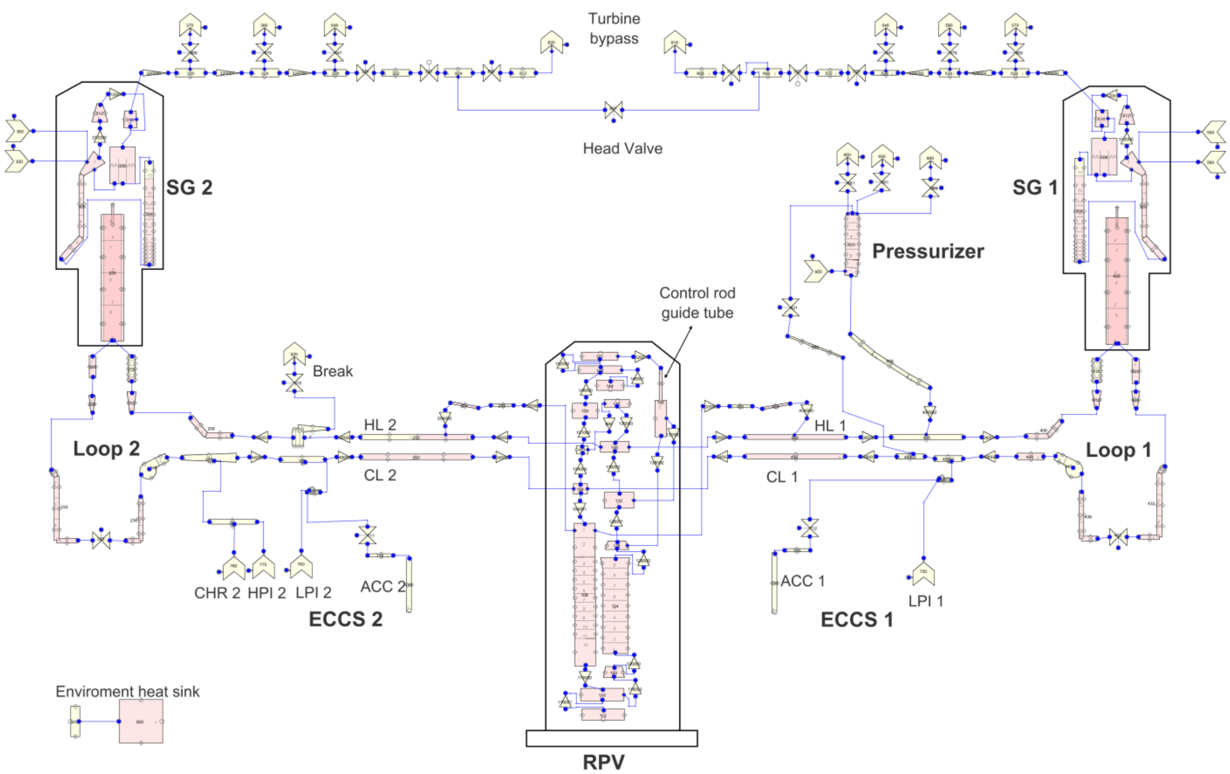

Figure 4 TRACE Input Model of ROSA/LSTF Represented by SNAP

2.6 Initial and Boundary Conditions

The initial and boundary conditions used are shown in Table 2. The agreement for both RELAP5 and TRACE is good. In case of RELAP5 there is some deviation for cold leg fluid temperature, while in case of TRACE there is some deviation in hot leg temperature. In case of TRACE there is also slightly lower secondary-side pressure comparing to measured data. This is an indication that the steam generator heat removal rate due to different heat transfer models is not same as in RELAP5, therefore secondary-side pressure was not matched. Namely, when initializing the cold leg temperature and secondary side pressure, only one variable can be set while the other is dependent. Similar behavior has been observed when converting from RELAP5 to TRACE the BETHSY input model (Ref. 8).

Table 2 Initial and Boundary Conditions for SB-HL-02 Test

Parameter	Measured (loop 1/2)	RELAP5	TRACE	Unit
Initial core power	10.06	10.06	10.06	MW
Initial hot leg fluid temperature	598.6 / 598.8	598.2 / 598.2	597.5 / 597.5	K
Initial cold leg fluid temperature	563.5 / 563.5	564.6 / 564.4	563.9 / 563.7	K
Initial mass flow rate	26.4 / 25.62	26.4 / 25.62	26.4 / 25.66	kg/s
Initial PRZ pressure	15.5	15.5	15.5	MPa
Initial PRZ liquid level	2.7	2.7	2.7	m
Initial secondary-side pressure	7.34 / 7.37	7.35 / 7.35	7.1 / 7.1	MPa
Initial secondary-side liquid level	9.75 / 9.65	9.74 / 9.65	9.73 / 9.66	m
Initial main steam flow rate	2.63 / 2.62	2.78 / 2.69	2.75 / 2.68	kg/s
Initial main feedwater flow rate	2.58 / 2.77	2.58 / 2.77	2.58 / 2.77	kg/s
Main feedwater temperature	495.4 / 495.4	495.4 / 495.4	495.4 / 495.4	K
Accumulator water temperature	318.8 / 323.7	318.8 / 323.7	318.8 / 323.7	K
LPI system fluid temperature	311.8	311.8	311.8	K

2.7 <u>Simulated SB-HL-02 Test Cases</u>

As has been already mentioned, the RELAP5/MOD3.3 computer code using Henry-Fauske critical flow model (R5_HF label) and Ransom-Trapp critical flow model (R5_RT label) has been used for calculations. TRACE has only default critical flow model, which is extension of Ransom and Trapp critical flow model (TRACE label) and this was used for calculations. Both RELAP5 and TRACE have option to select user-defined critical flow model coefficients. The values of break flow model coefficients used in the selected calculations (six in total) are shown in Table 3.

Table 3 Break Flow Model Coefficients Used for RELAP5 and TRACE Calculations

Calculation ID	Break flow model coefficients	Case
R5_HF(1.0_0.14)	CD=1.0, C=0.14	base
R5_HF(0.75_0.14)	CD=0.75, C=0.14	tuned
R5_RT(1.0_1.0)	CD1=1.0, CD2=1.0	base
R5_RT(0.9_0.9)	CD1=0.9, CD2=0.9	tuned
TRACE(1.0_1.0)	CHM12=1.0, CHM22=1.0	base
TRACE(1.0_0.7)	CHM12=1.0, CHM22=0.7	tuned

The meaning of abbreviations in Table 3 is: CD is discharge coefficient and C is thermal non-equilibrium constant, CD1 is subcooled discharge coefficient, CD2 is two-phase discharge coefficient, CHM12 and CHM22 are subcooled and two-phase multipliers, respectively. As shown in Table 3, with each code and critical flow model calculations two cases have been performed, by default coefficient of critical flow model (base case) and coefficients selected by user, which gave the best agreement for pressurizer pressure between calculation and experiment tuned case). Namely, the sequence of events strongly depended on the primary (pressurizer) pressure.

3 RESULTS

The results of calculations are shown in Figures 5 through 46. The animated results showing mass distribution in LSTF are shown in Figures 47 through 52. Comparison between calculations and experimental data is made separately for base cases using code default break flow coefficients and tuned cases using user-defined break flow coefficients (see Sections 3.1 and 3.2, respectively). Key plant parameters are shown like primary and secondary pressures, hot and cold leg temperatures, mass flowrate in loops, emergency coolant injection flow rates, break flow rate, fuel rod surface temperature etc.

3.1 Comparison Between Base Case Calculations and Experiment

Figure 5 shows the pressurizer pressure, which starts to drop simultaneously by the break at time zero. Due to large break size 10% the depressurization is fast. The scram signal is generated in few seconds after the break when the PZR pressure decreased to 12.97 MPa. The scram signal generation caused the closure of SG main steam isolation valves (MSIVs) and the coastdown of primary coolant pumps. The steam generator pressure increased rapidly due to MSIV closure, reaching the relief valve setpoint. Later the valve is cycling. From Figure 5 it can be seen that RELAP5 (‘R5_HF(1.0_1.0)’ case) and TRACE (‘TRACE(1.0_1.0)’ case) calculations behave similarly when default break flow models with default critical coefficients are used, while calculated pressurizer pressure obtained by optional RELAP5 Ransom-Trapp critical flow model is in better agreement (‘R5_RT(1.0_1.0)’ case). Secondary-side pressures are shown in Figures 6 and 7. TRACE calculation (‘TRACE(1.0_1.0)’) is in slightly better agreement with experimental data than RELAP5 calculations for secondary pressure, but after pressure reversal (at 160 s in experiment) steam generator no more served as heat sink.

Figure 8 shows the break mass flow rate. The break flow rate decreased when the break flow turned from single-phase liquid to two-phase flow at 10 s first, and then to single-phase vapor when hot legs became empty of liquid as shown in Figures 22 and 23. RELAP5 ‘R5_RT(1.0_1.0)’ case in initial 300 s the best predicts the break flow comparing to other two predictions (‘R5_HF(1.0_0.14)’ and ‘TRACE(1.0_1.0)’), what explains the best prediction of primary pressure for ‘R5_RT(1.0_1.0)’. When looking integral of break mass, ‘R5_HF(1.0_0.14)’ and ‘TRACE(1.0_1.0)’ calculations overpredicted the break flow in the first 160 s. Later, TRACE calculation is closer to experimental break flow than RELAP5 calculation as shown in Figure 9.

Figures 10 and 11 show the primary loop mass flow rate measured at the primary coolant pump suction in loop no. 1 and 2, respectively. In the experiment forced circulation ceased 250 s after the scram signal generation. The primary loop mass flow rate after the initiation of the accumulator coolant injection indicates the flow of steam induced by condensation on accumulator coolant. The indicated mass flow rate after about 330 s is thus incorrect. The calculations are in good agreement with experimental data in the first 100 s. Later, the calculated flow is lower and ceased earlier than in experiment. It should be noted that pump coastdown was simulated modelled using experimental data.

Figures 12 and 14 show hot leg temperatures in loop no. 1 and 2, respectively. Hot leg fluid became saturated immediately after the break. Therefore the calculated trends till 350 s are similar as for primary pressure. The time differences in temperature decrease are also similar as for primary pressure calculations. Figures 13 and 15 show cold leg temperatures in loop no. 1 and 2, respectively. In the experiment the cold leg fluid became saturated and started flashing soon after the break. Similar conclusion as for hot leg temperature calculations can be made for cold leg temperature calculations till 350 s. Later the qualitative difference is observed for TRACE calculation in which temperature drops significantly after 520 s.

Figure 16 shows the core collapsed liquid level. The best timing in core uncovery was obtained in calculation with the best break flow and primary pressure agreement (i.e. 'R5_RT(1.0_1.0)' case). Figure 17 shows fuel rod surface temperature. The time of peak clad temperature occurrence is again the best for calculation with best primary pressure prediction. However, in all calculations the peak is relatively small due to short core uncovery.

Figures 18 and 19 show that the coolant injection flow rate from the accumulator tank in the loop no. 1 was about three times larger than that in the loop no. 2. The accumulator coolant injection occurred twice in the experiment, while in the calculations only in the 'R5_RT(1.0_1.0)' case the injection was twice, while in the other two calculations was once with oscillatory behavior at the end of injection for 'R5_HF(1.0_0.14)' case. Also, in all calculations the accumulator injection was earlier due to faster primary pressure drop in calculations.

Figures 20 and 21 show low pressure injection flows in loop no. 1 and 2, respectively. The experimental flow in loop no. 2 was zero, as the flow was going only into loop no. 1. This was not observed in calculations. The modelled injection flow in loop no. 1 is three times smaller than in loop no. 2, what reflected also in the calculations. The reason for such modelling is not explained in the report (Ref. 4).

Figures 22 and 23 show hot leg fluid densities in loop no. 1 and 2, respectively. In the experiment, the liquid level behaviors in the hot legs were in asymmetrical during the time period from about 75 to 170 s due to relatively large size break at the hot leg. Hot legs became empty of liquid at about 280 s, causing the termination of two-phase flow discharge from the break. The hot leg liquid level started to recover after the initiation of the accumulator coolant injection. Also in calculations such trends were predicted qualitatively.

Finally, Figures 24 and 25 show cold leg fluid densities. Cold leg fluid became saturated and started flashing soon after the break. Cold legs became almost refilled but temporarily twice during the accumulator injected coolant in the experiment. In the calculations cold leg was refilled just half in loop no. 1. Only first refill was qualitatively predicted. This is closely related to the accumulator injection.

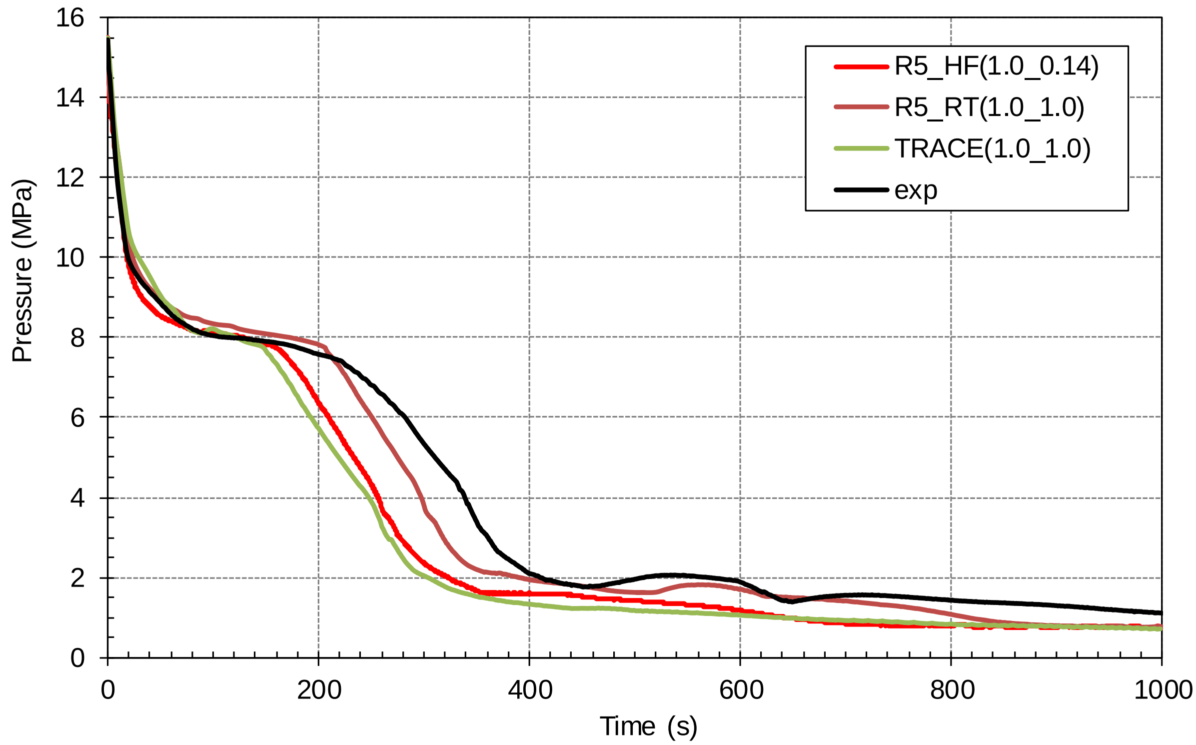

Figure 5 **Pressurizer Pressure – Base Case**

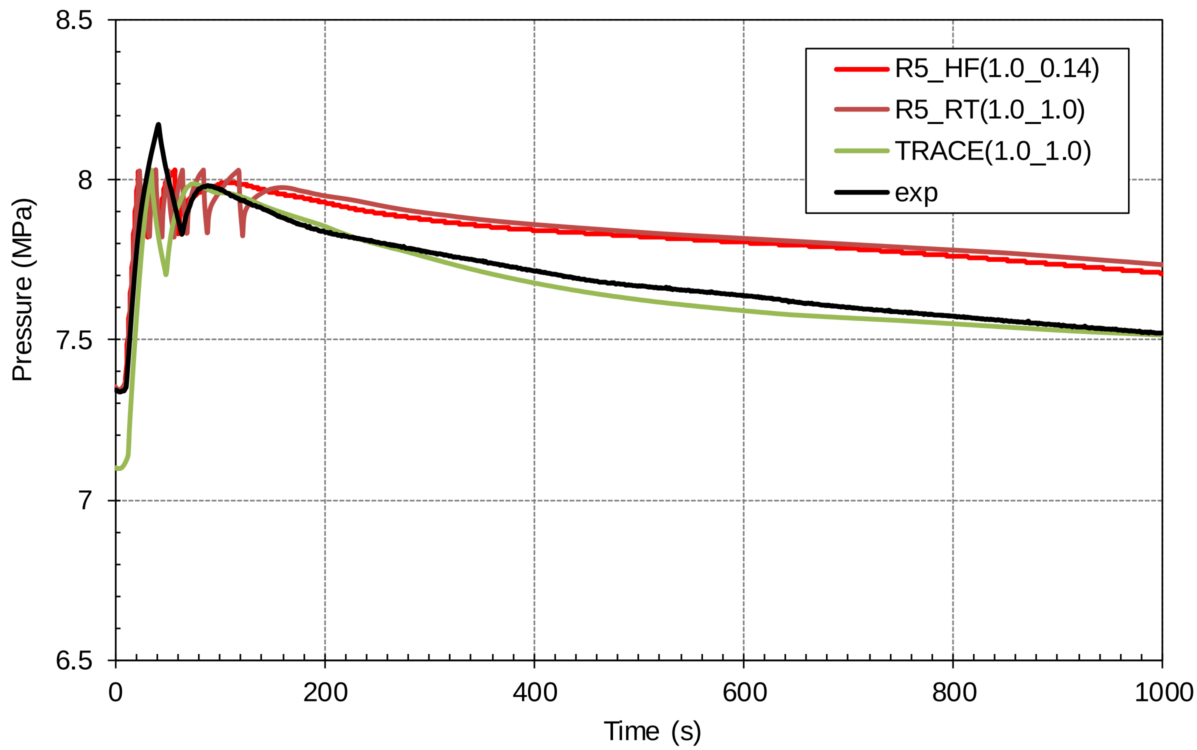

Figure 6 **Secondary-Side No. 1 Pressure – Base Case**

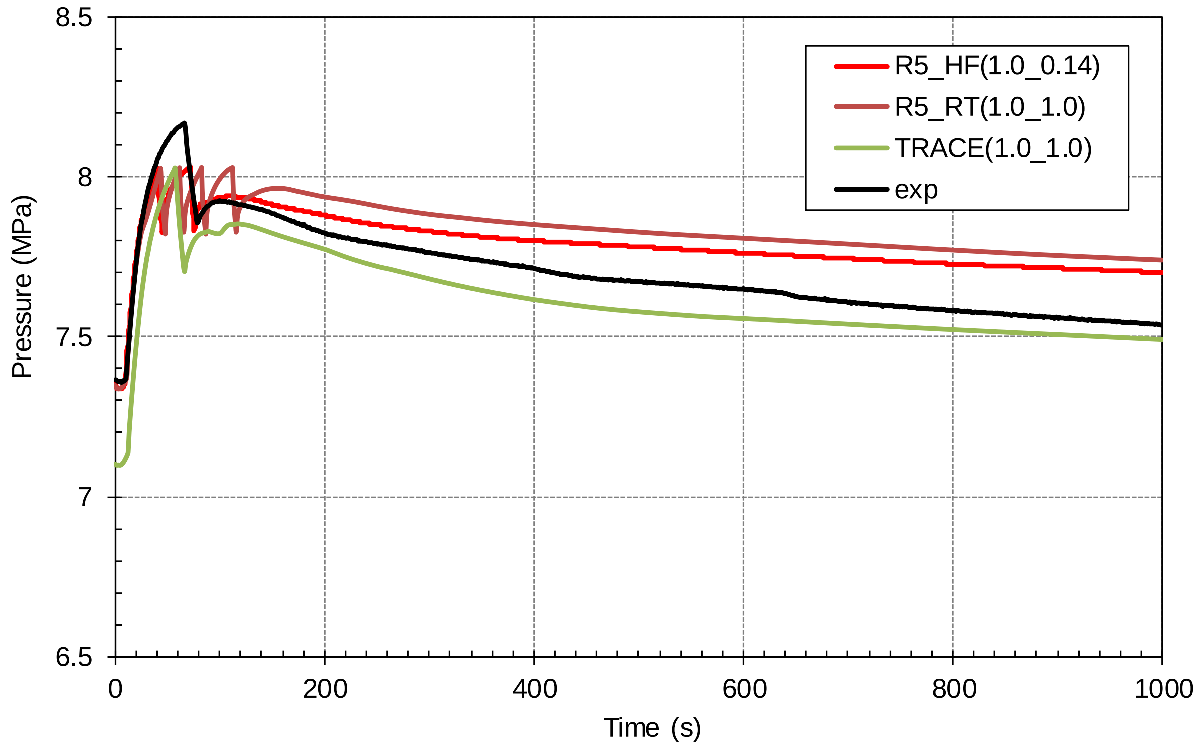

Figure 7 **Secondary-Side No. 2 Pressure – Base Case**

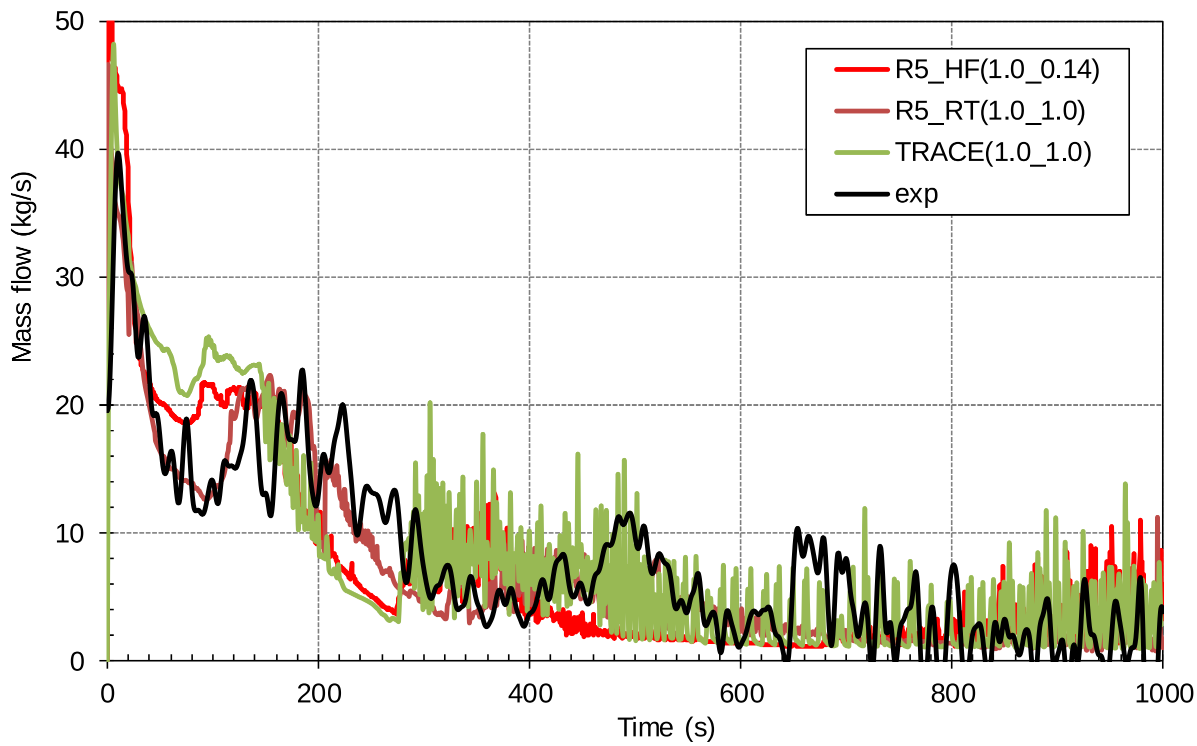

Figure 8 **Break Mass Flow Rate – Base Case**

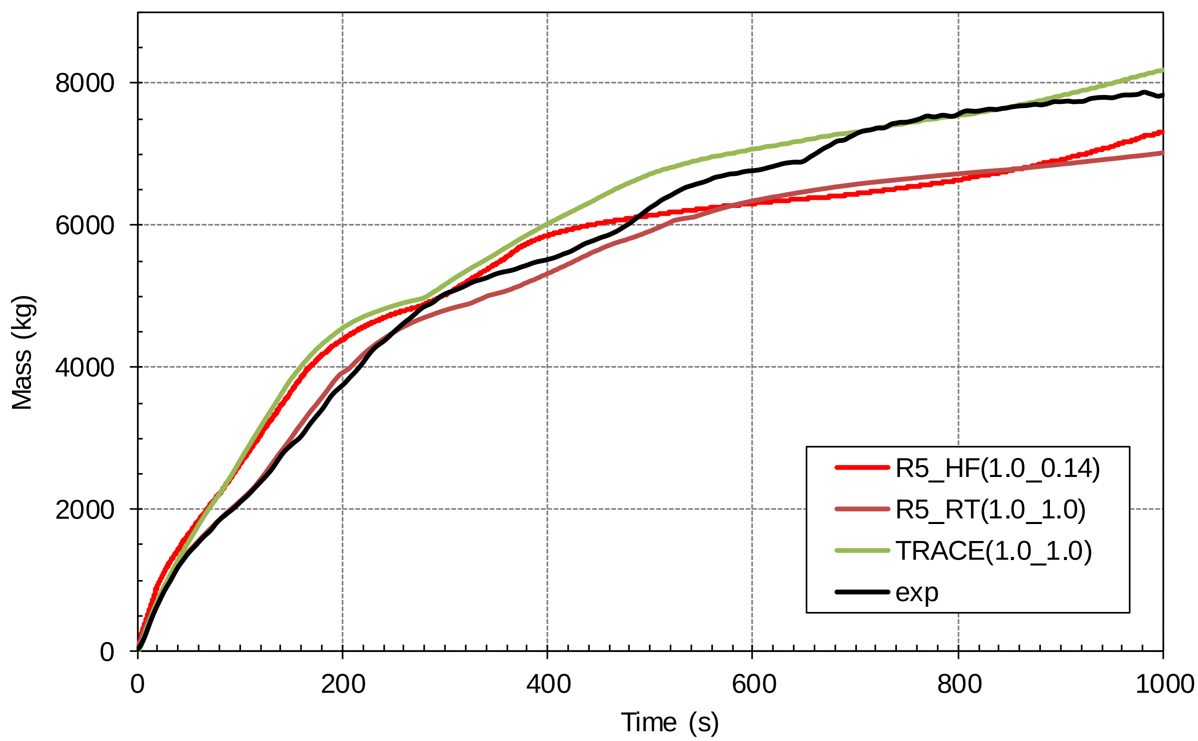

Figure 9 **Integral of Break Mass Flow Rate – Base Case**

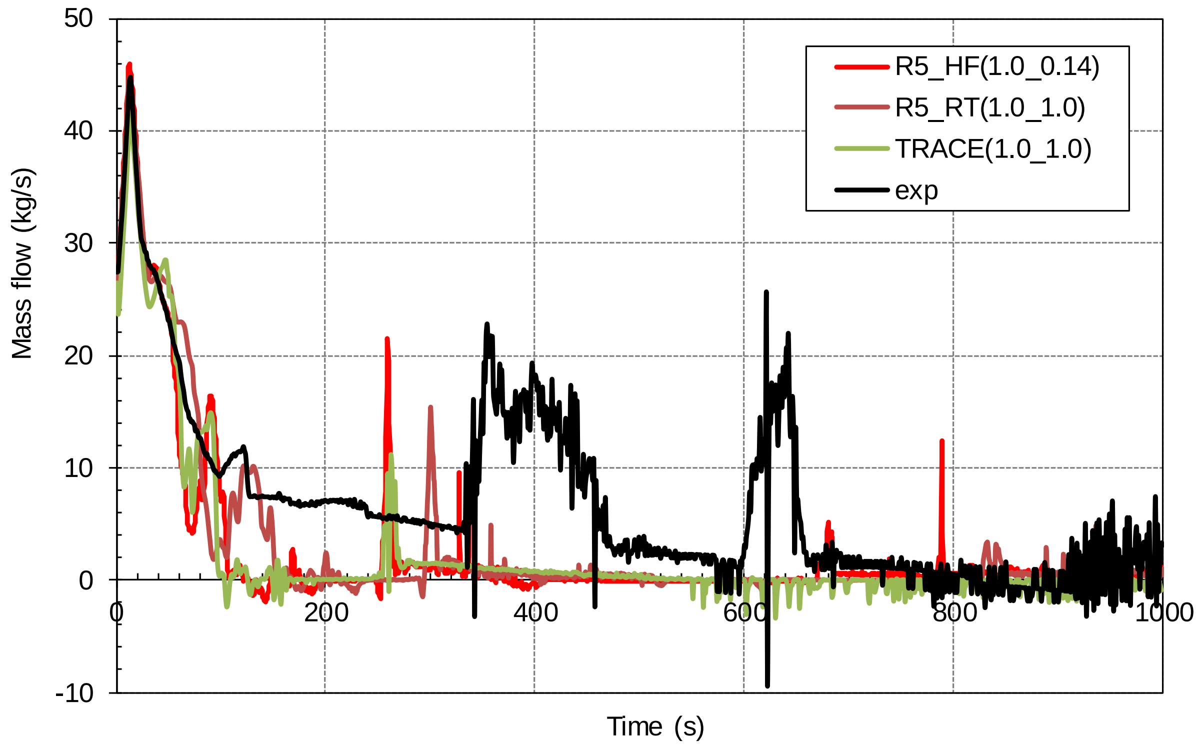

Figure 10 **Primary Loop No. 1 Mass Flow Rate – Base Case**

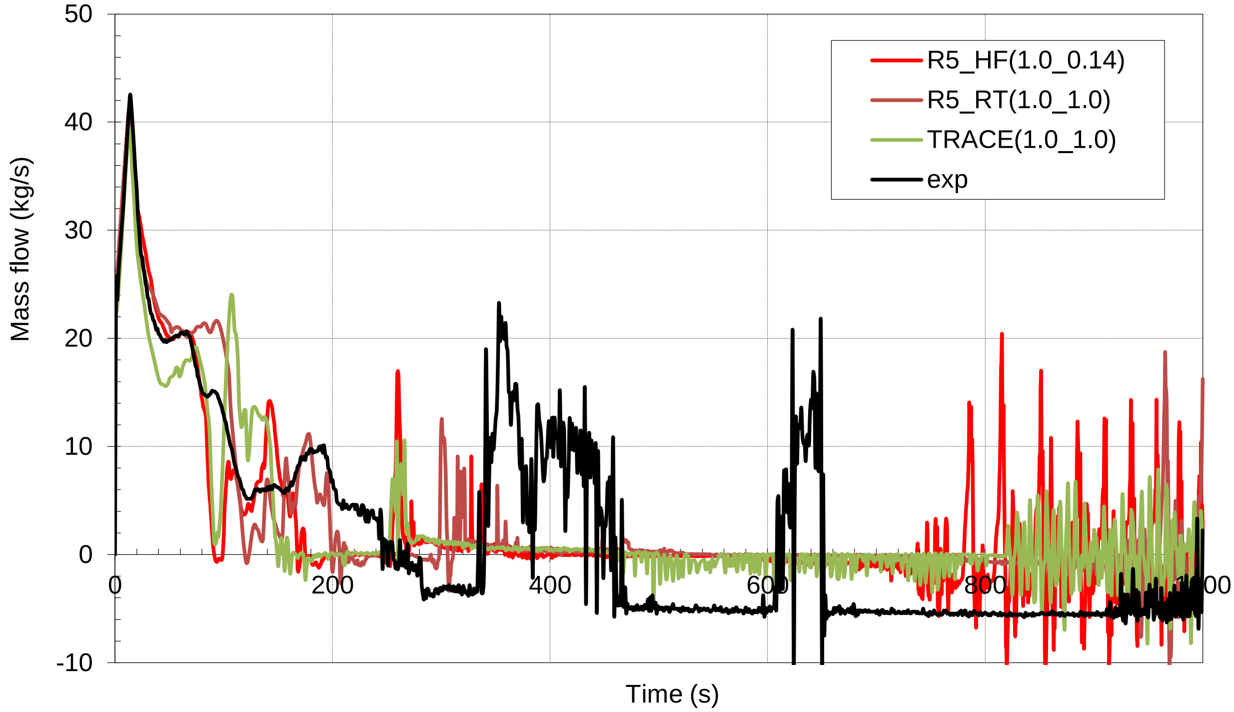

Figure 11 **Primary Loop No. 2 Mass Flow Rate – Base Case**

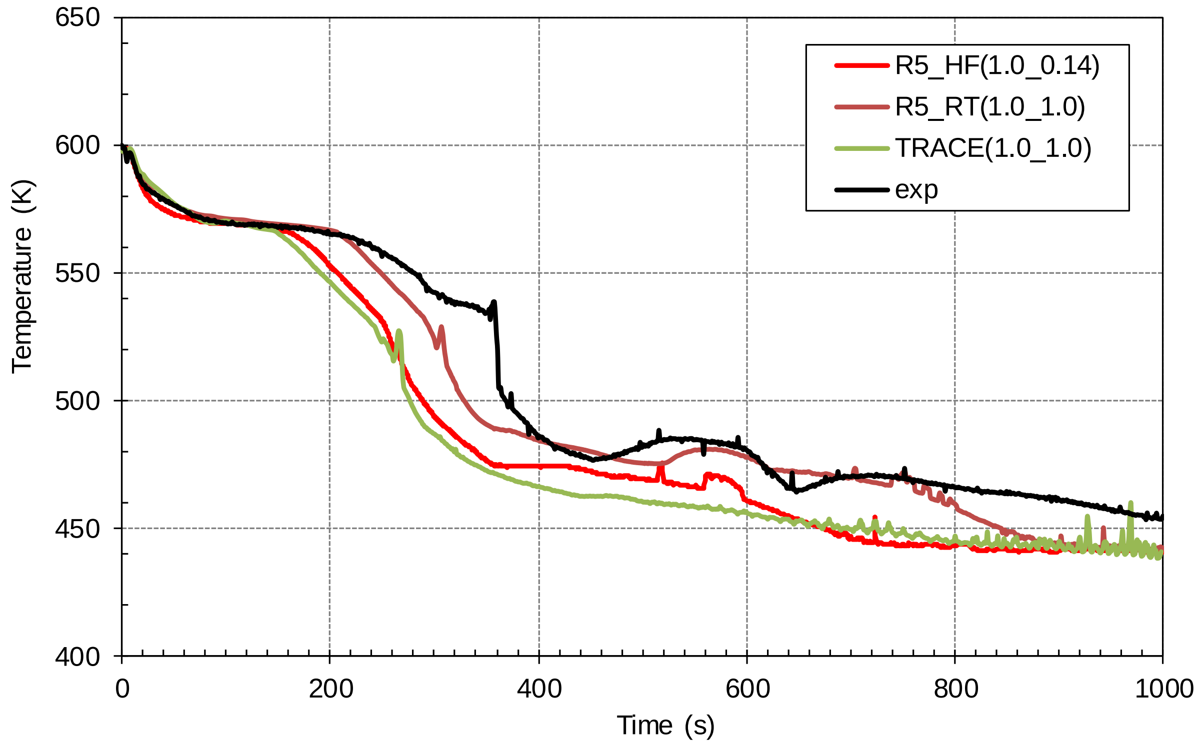

Figure 12 **Hot Leg No. 1 Fluid Temperature – Base Case**

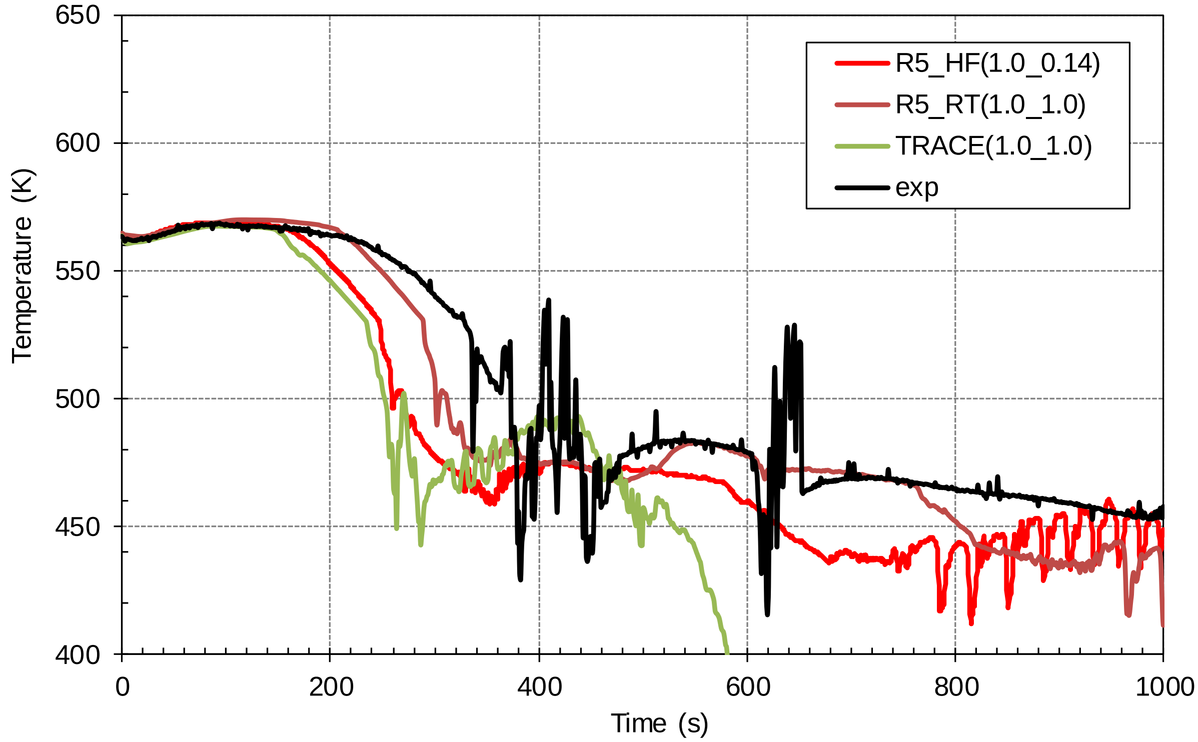

Figure 13 Cold Leg No. 1 Fluid Temperature – Base Case

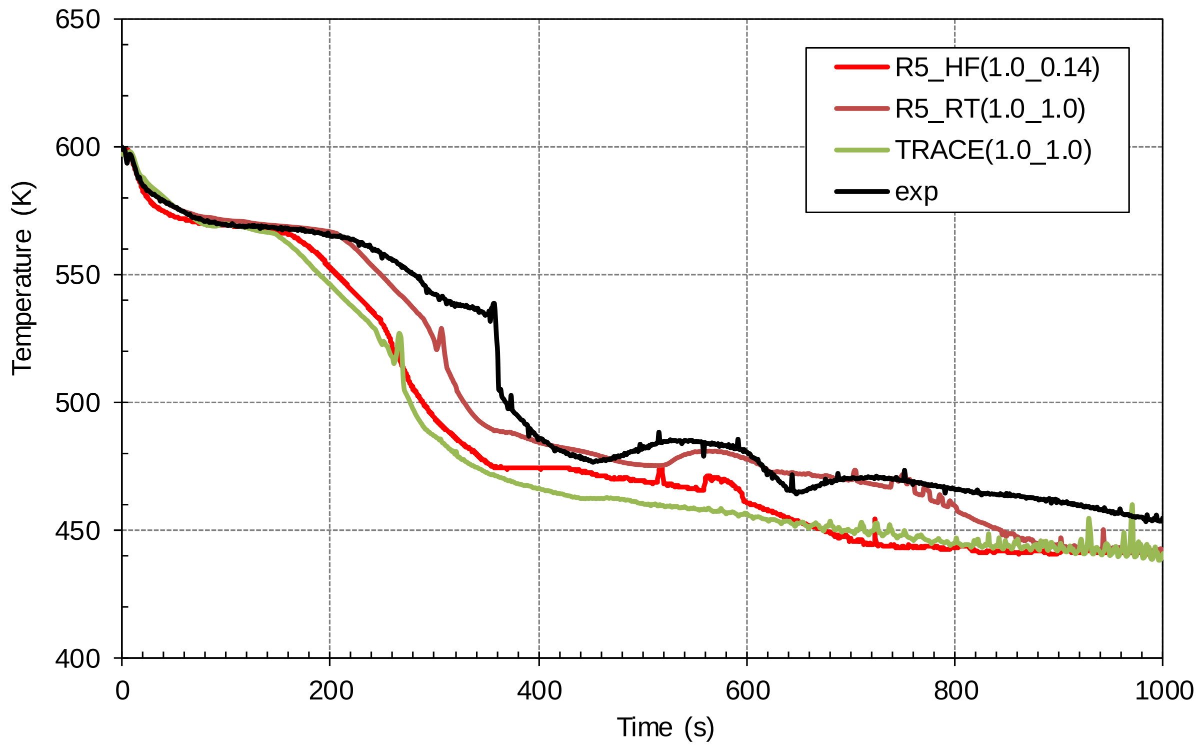

Figure 14 Hot Leg No. 2 Fluid Temperature – Base Case

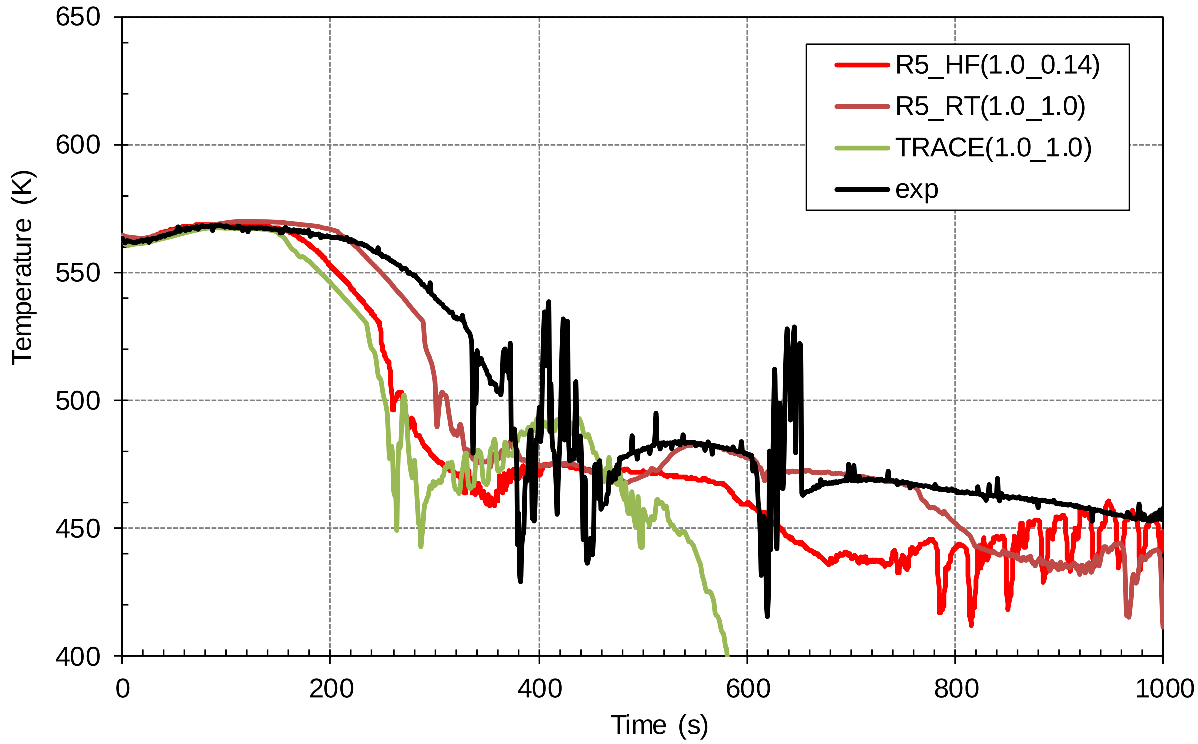

Figure 15 **Cold Leg No. 2 Fluid Temperature – Base Case**

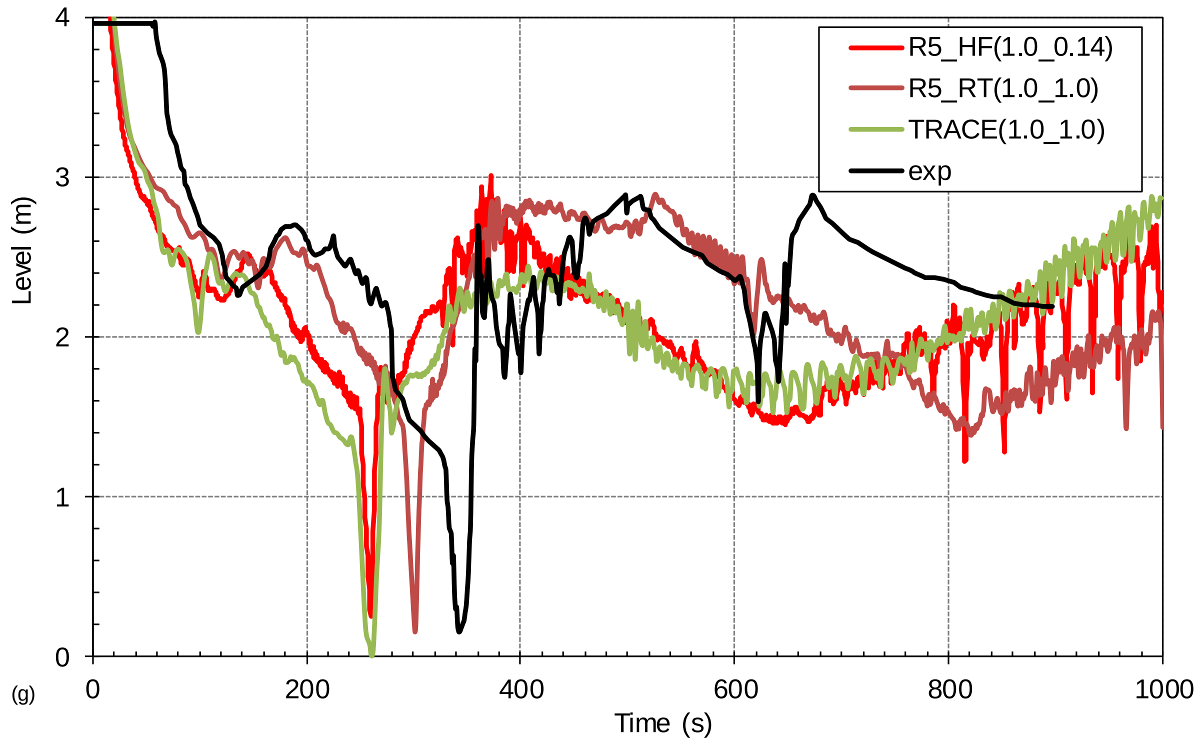

Figure 16 **Core Collapsed Liquid Level – Base Case**

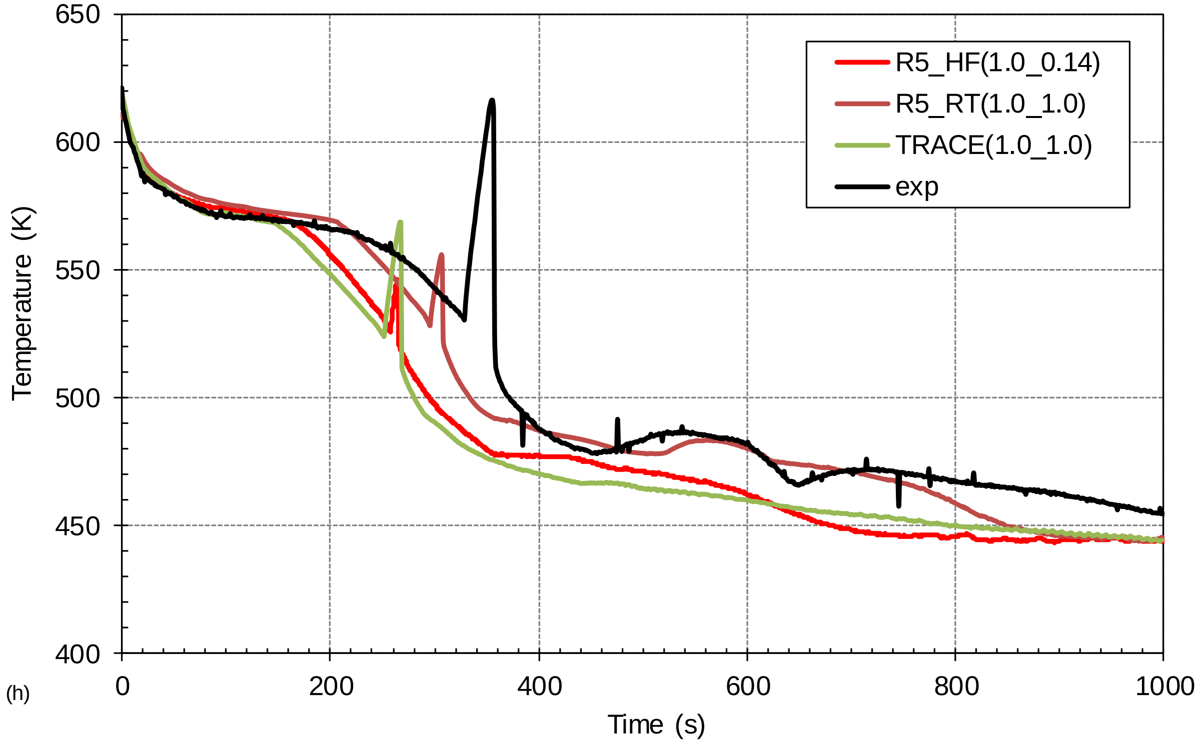

Figure 17 **Fuel Rod Surface No. 7 Temperature – Base Case**

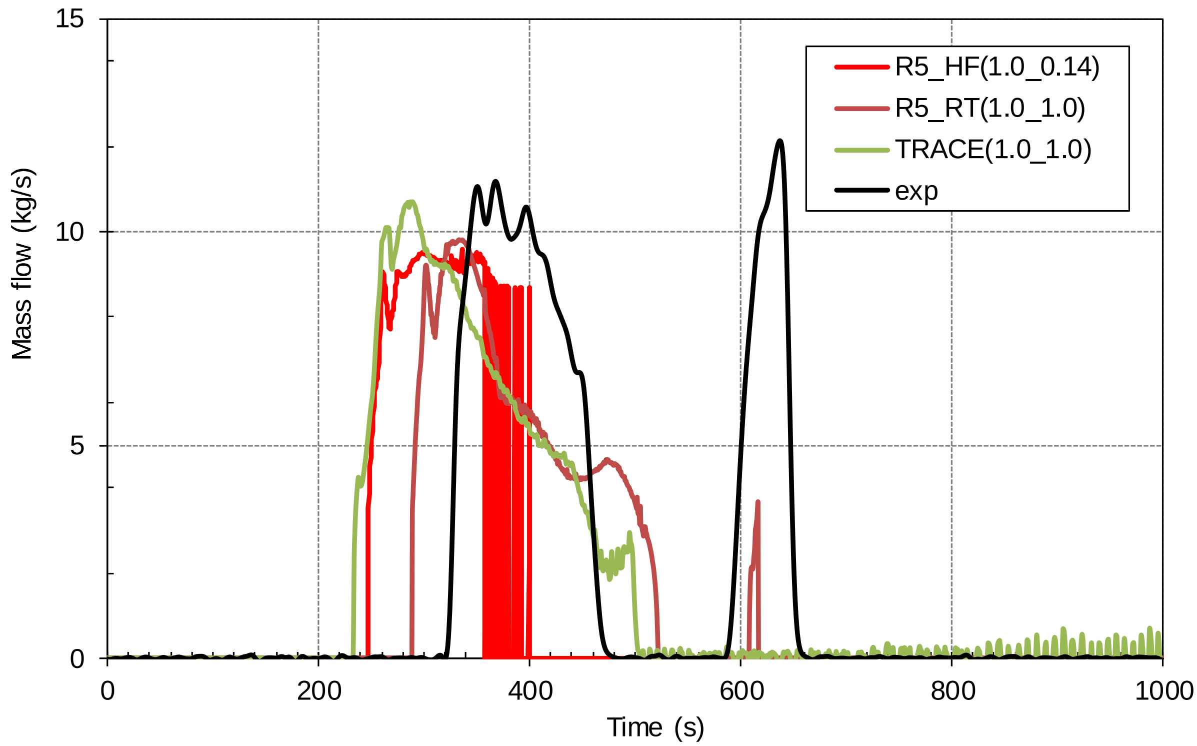

Figure 18 **Accumulator No. 1 Flow Rate – Base Case**

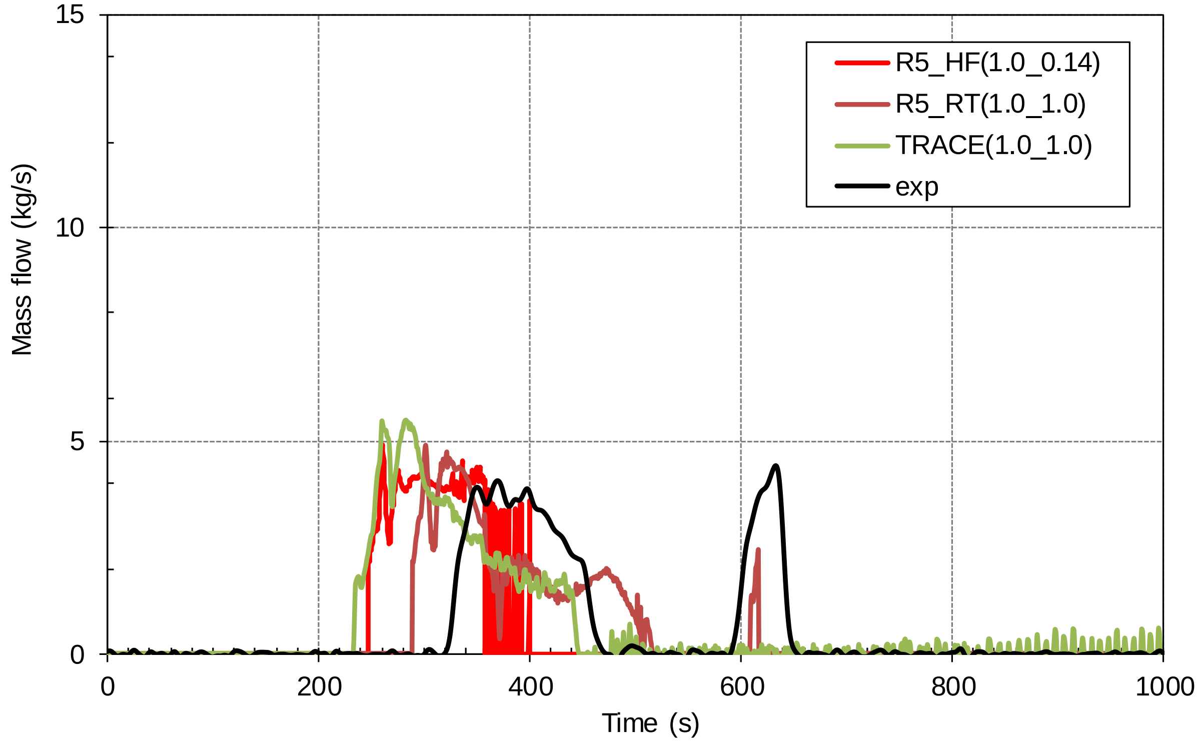

Figure 19 Accumulator No. 2 Flow Rate – Base Case

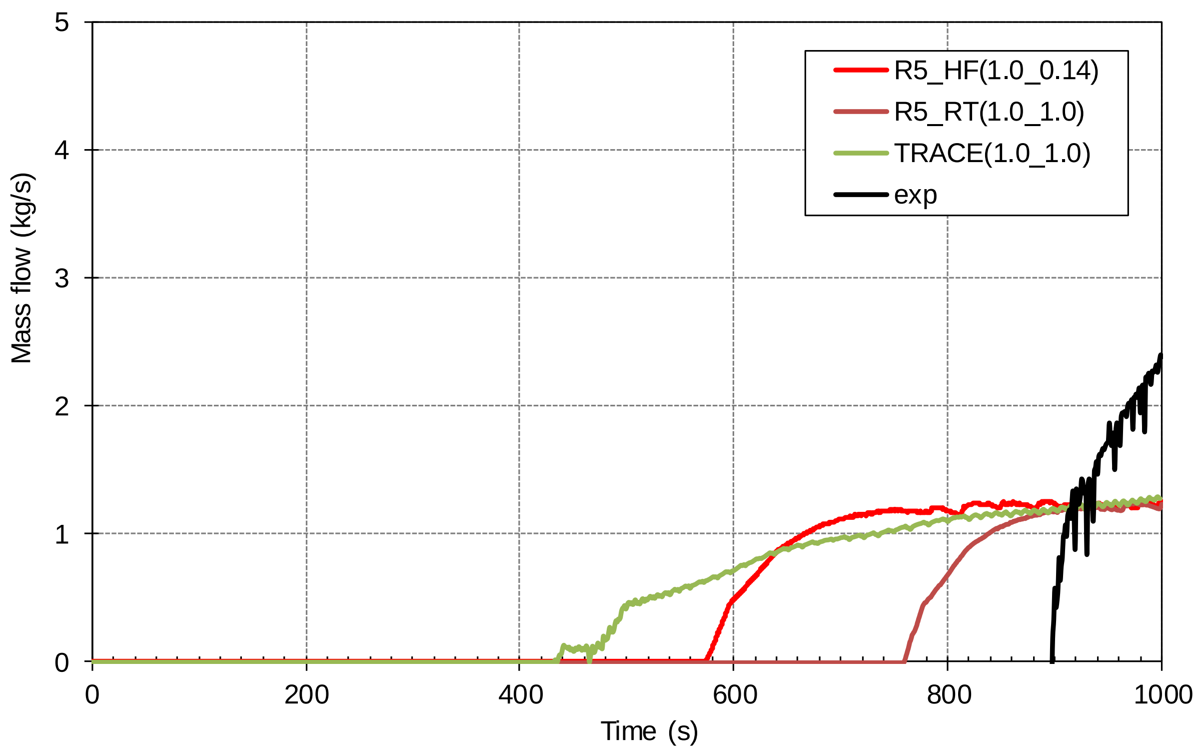

Figure 20 LPI No. 1 Flow Rate – Base Case

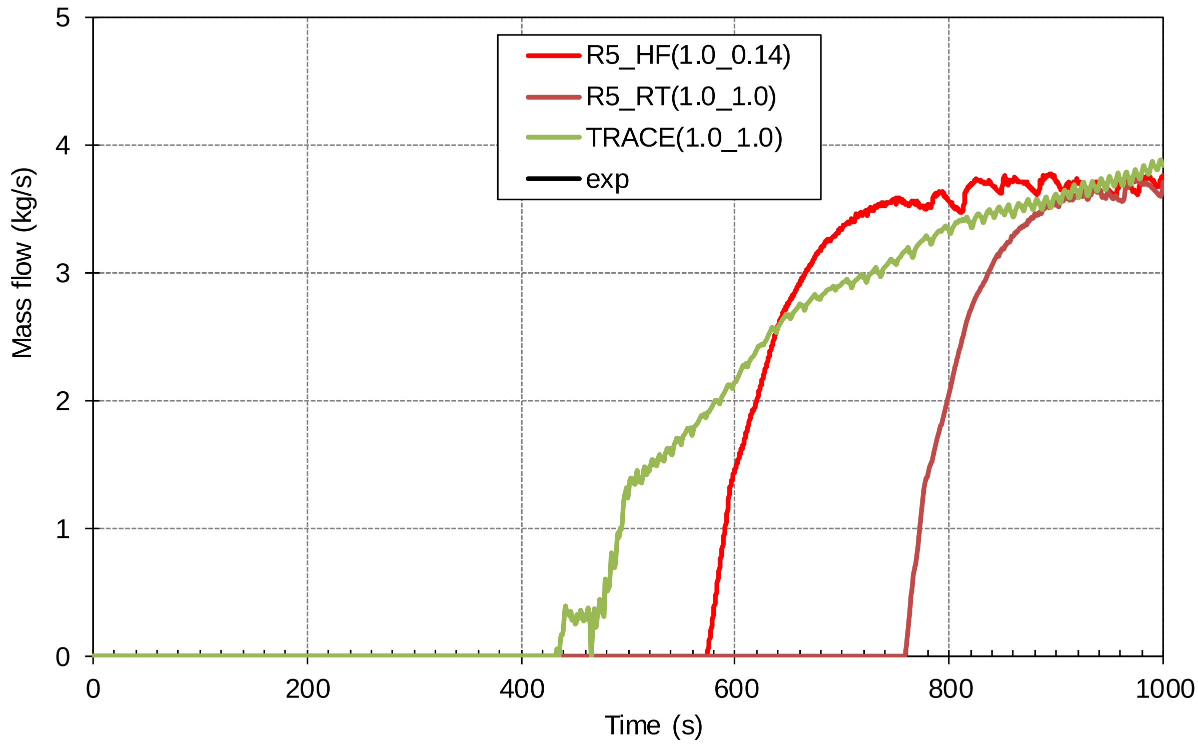

Figure 21 **LPI No. 2 Flow Rate – Base Case**

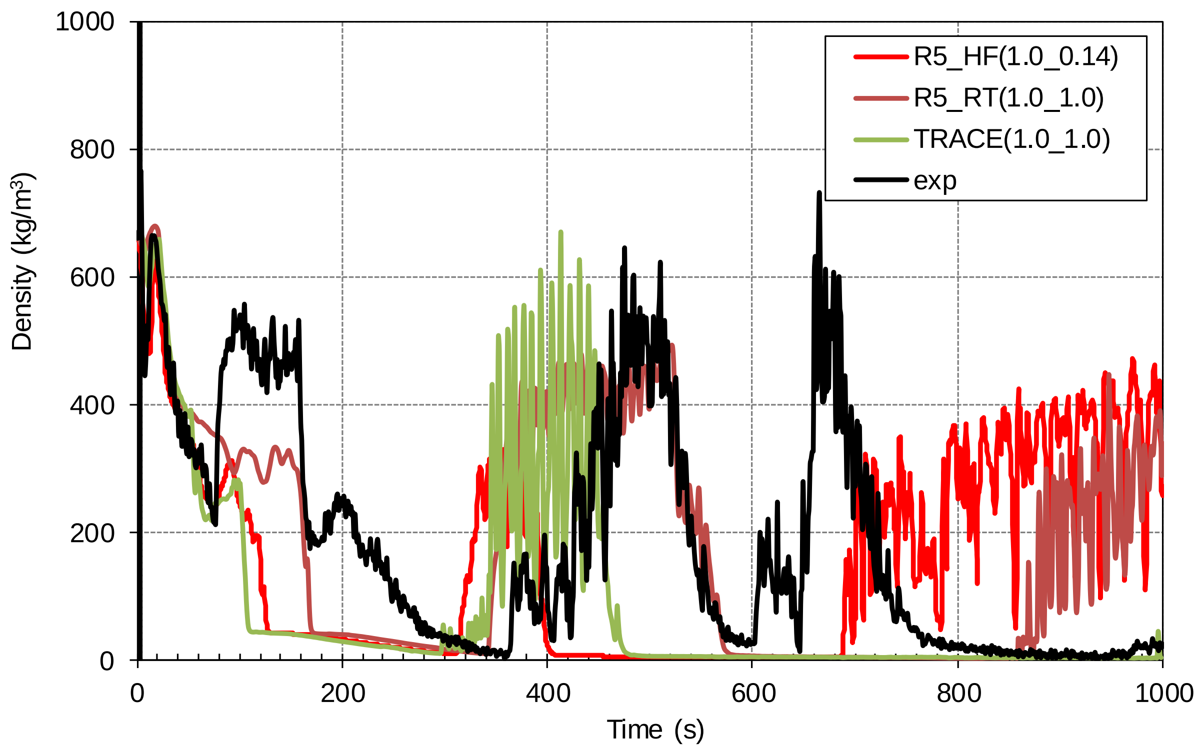

Figure 22 **Hot Leg No. 1 Flow Density – Base Case**

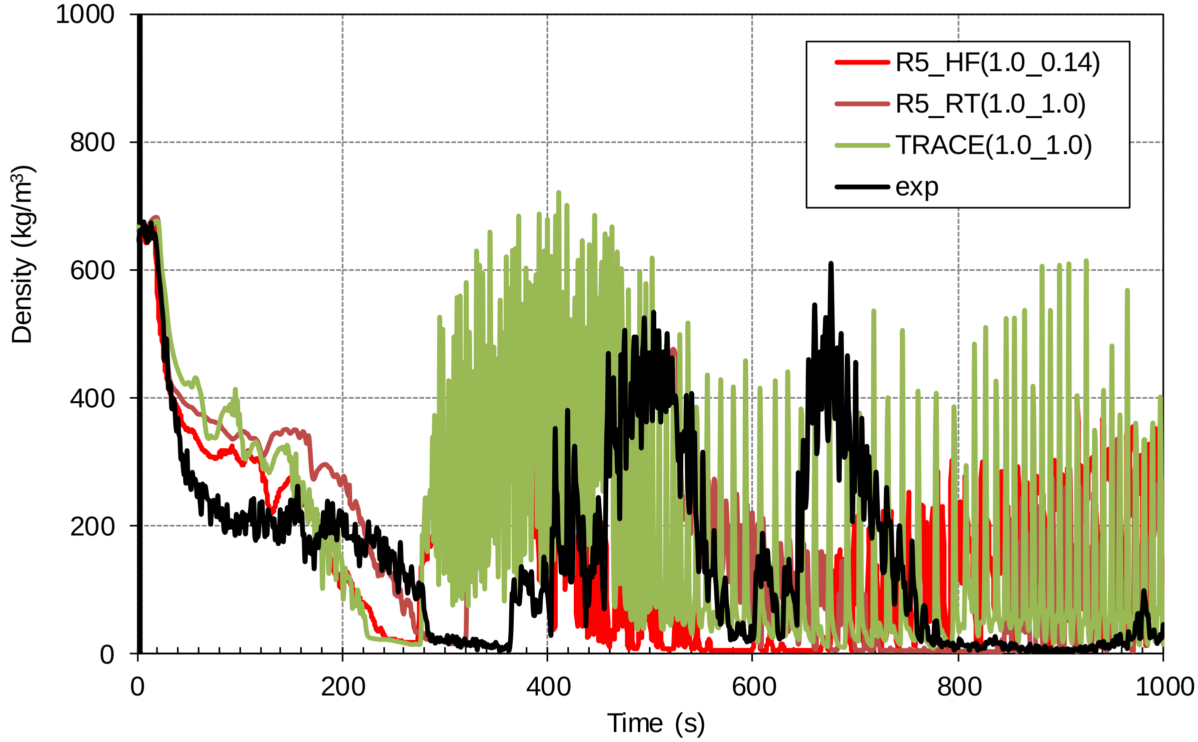

Figure 23 Hot Leg No. 2 Flow Density – Base Case

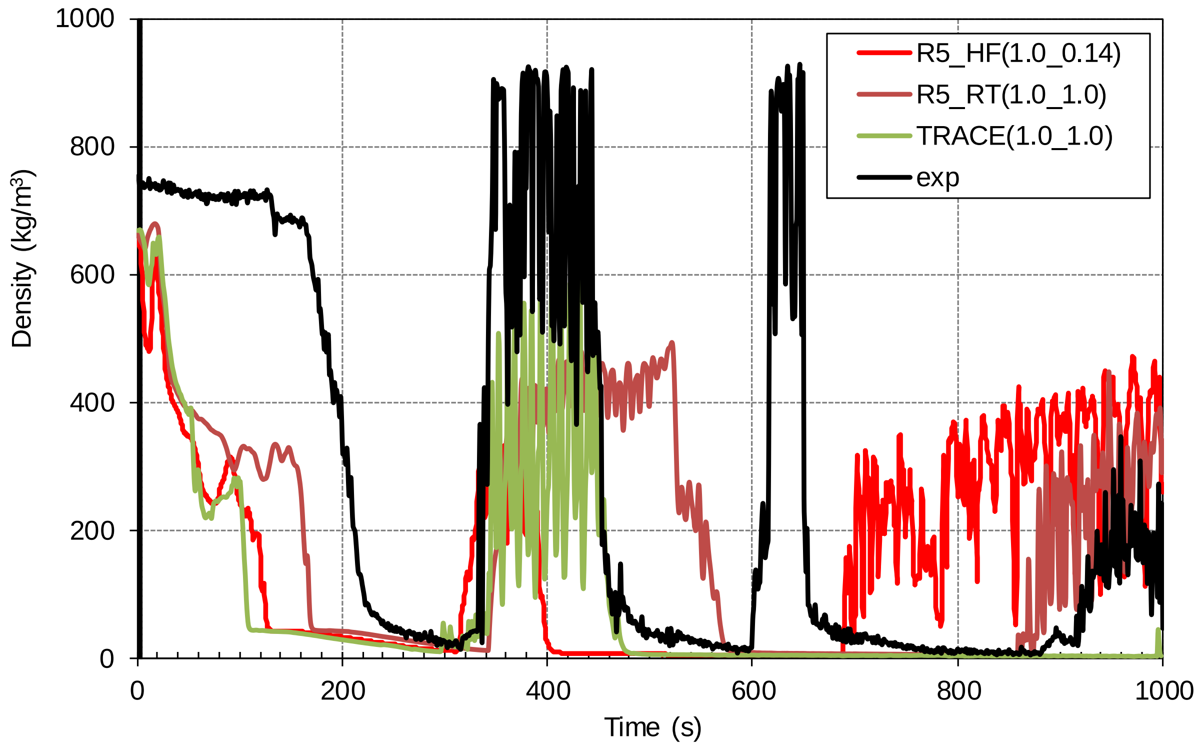

Figure 24 Cold Leg No. 1 Flow Density – Base Case

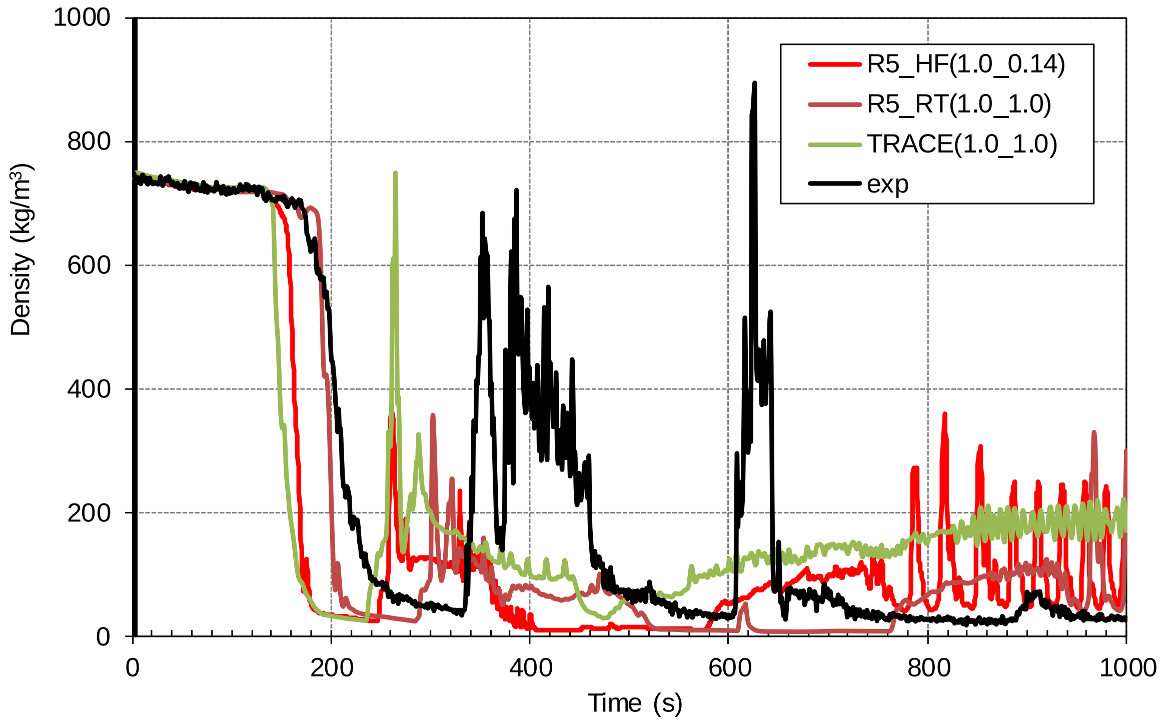

Figure 25 Cold Leg No. 2 Flow Density – Base Case

3.2 Comparison Between Tuned Case Calculations and Experiment

When comparing tuned calculations shown in Figures 26 through 46 to base calculations shown in Figures 5 to 25, it can be seen that correct break flow modelling is the most important for loss of coolant accident progression.

Figure 26 shows the pressurizer pressure, which is well predicted by all calculations. Secondary-side pressures are shown in Figures 27 and 28, respectively. The calculated trends are comparable, being the TRACE calculated pressure slightly faster decreasing and closer to experimental data.

Figure 29 show the break mass flow rate. It may be seen that all tuned cases simulated break flows are in good agreement with the experimental data, what can be confirmed from Figure 30, showing integral of break mass flowrate. The agreement is perfect in the first 200 s due to tuning the break flow model coefficients. Later, the calculated break flows underpredicted the experimental data.

Figures 31 and 32 show the primary loop mass flow rate measured at the primary coolant pump suction in loop no. 1 and 2, respectively. As already mentioned, the indicated mass flow rate after about 330 s in experiment is incorrect. The calculations are in good agreement in the first 100 s. Later, the calculated flow is lower and ceased earlier than in experiment.

Figures 33 and 35 show hot leg temperatures in loop no. 1 and 2, respectively. The agreement is very good till 500 s. Later when the hot leg no. 1 is empty (see hot leg density in Figure 43), the temperature increases due to steam presence only. The timing is qualitatively different from experiment.

Figures 34 and 36 show cold leg temperatures in loop no. 1 and 2, respectively. Similar conclusion as for hot leg temperature calculations can be made. Comparing to base calculations, this time the TRACE results are closer to experimental data.

Figure 37 shows the core collapsed liquid level. All calculations well predicted the time of minimum level occurrence and minimum level value. Figure 38 shows fuel rod surface temperature. The time of calculated peak clad temperature occurrence is again very good, when compared to experimental data. The agreement of calculated temperatures with experiment is good except for the peak clad temperature.

Figures 39 and 40 show that the coolant injection flow rate from the accumulator tank in the loop no. 1 is larger than that in the loop no. 2 in experiment and also in the calculations. In all calculations the time of first accumulator injection is well predicted.

Figures 41 and 42 show low pressure injection flows in loop no. 1 and 2, respectively. The timing for injection is a bit earlier than in the experiment for loop no. 1. The experimental flow in loop no. 2 was zero, as the flow was going only into loop no. 1. This was not the case for calculations. The modelled injection flow in loop no. 1 is three times smaller than in loop no. 2. The reason for this is not explained in the report (Ref. 4).

Figures 43 and 44 show hot leg fluid densities in loop no. 1 and 2, respectively. It can be seen that calculations are in qualitative agreement with the experiment.

Finally, Figures 45 and 46 show cold leg fluid densities. Cold legs refilling by accumulators and low pressure injection is qualitatively predicted. In TRACE calculation refilling is smaller than in RELAP5 calculation for cold leg no. 1. After 200 s all calculations underpredicted the experimental data.

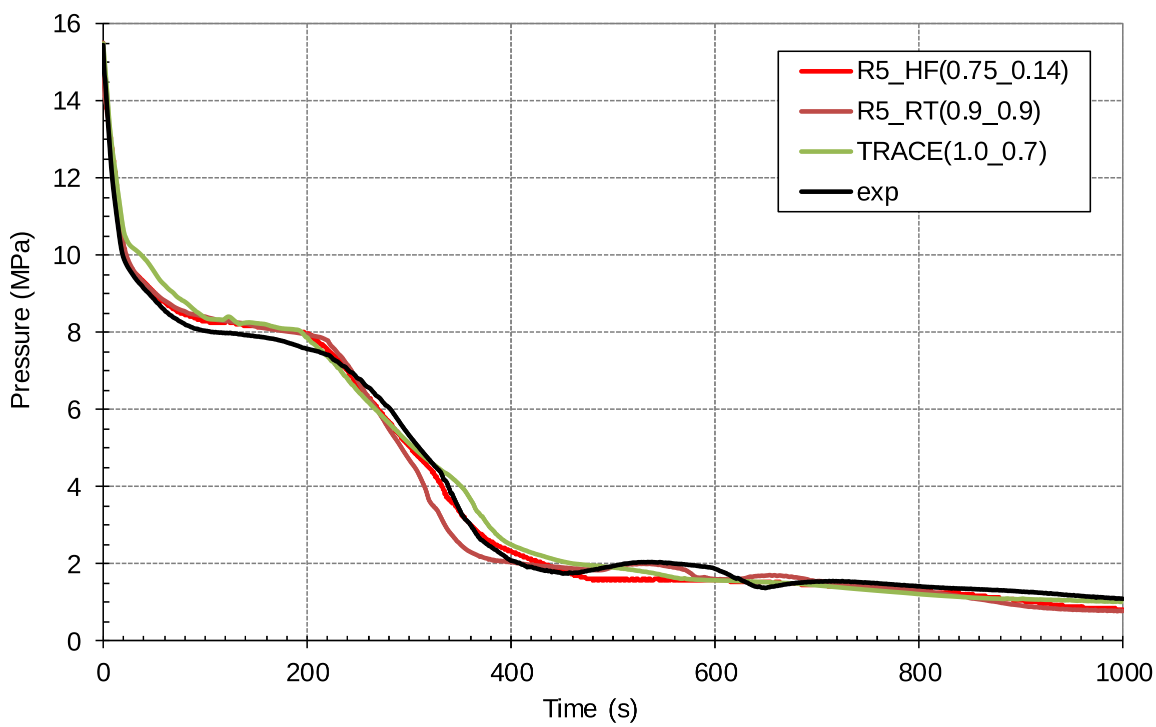

Figure 26 Pressurizer Pressure – Tuned Case

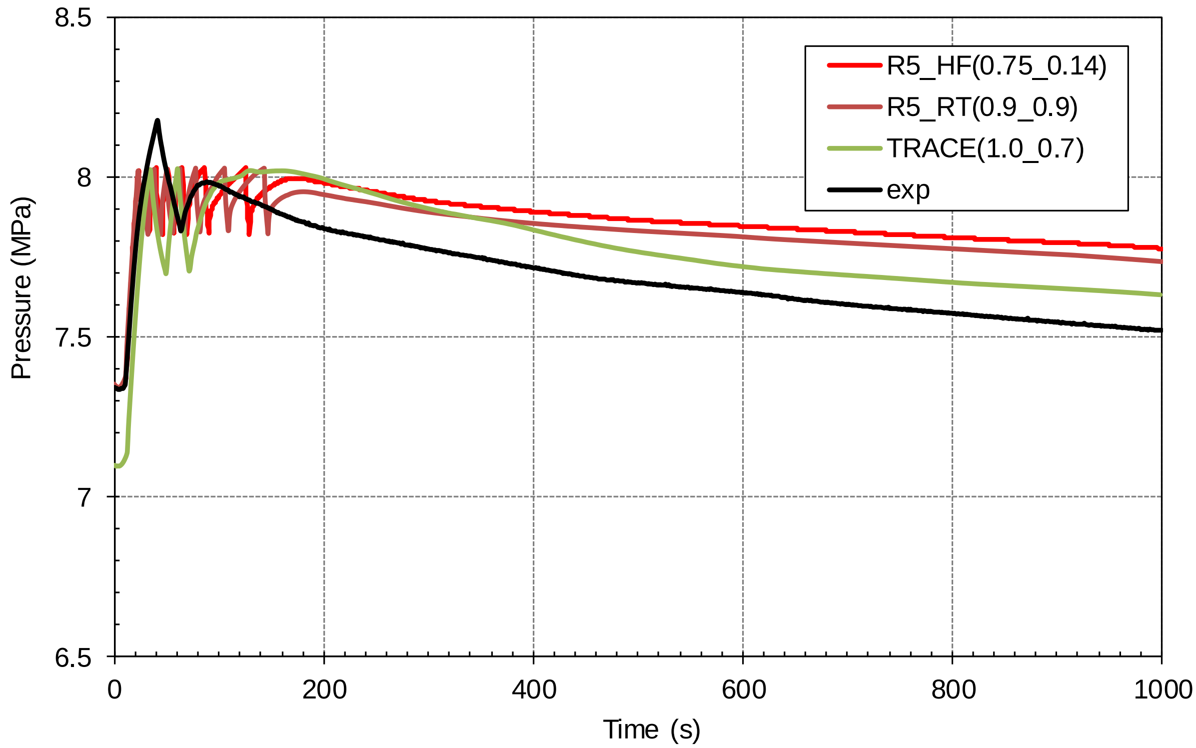

Figure 27 **Secondary-Side No. 1 Pressure – Tuned Case**

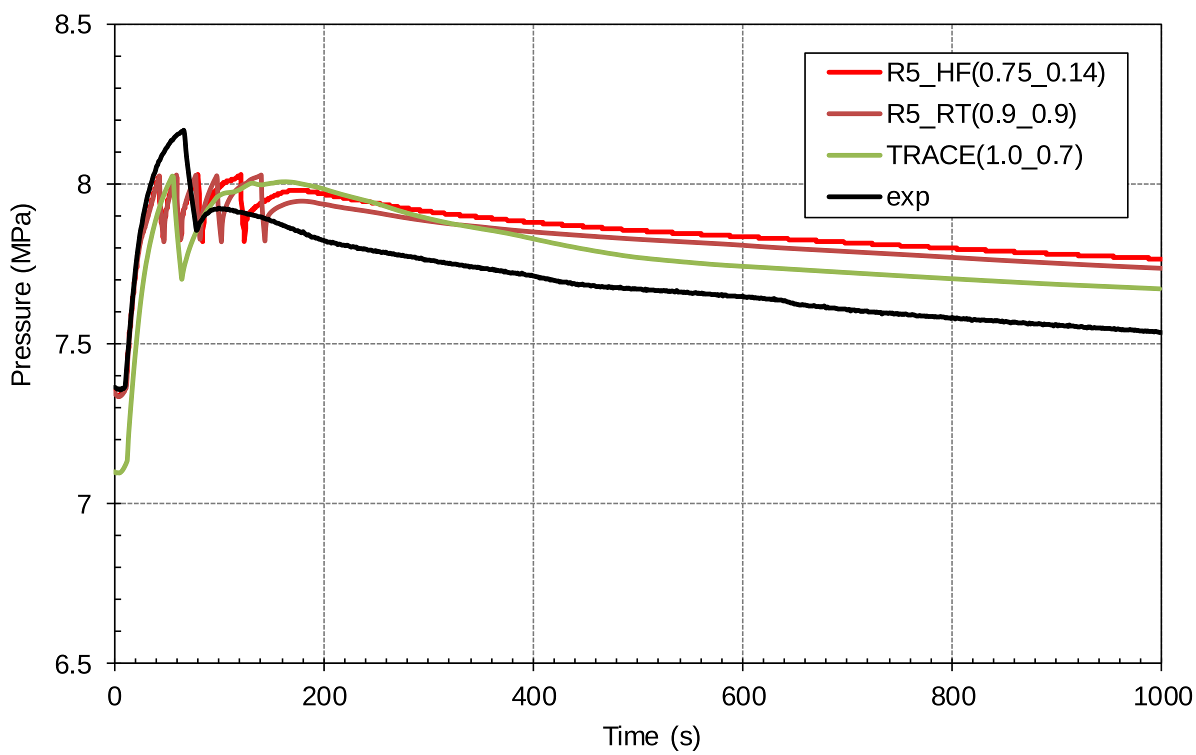

Figure 28 **Secondary-Side No. 2 Pressure – Tuned Case**

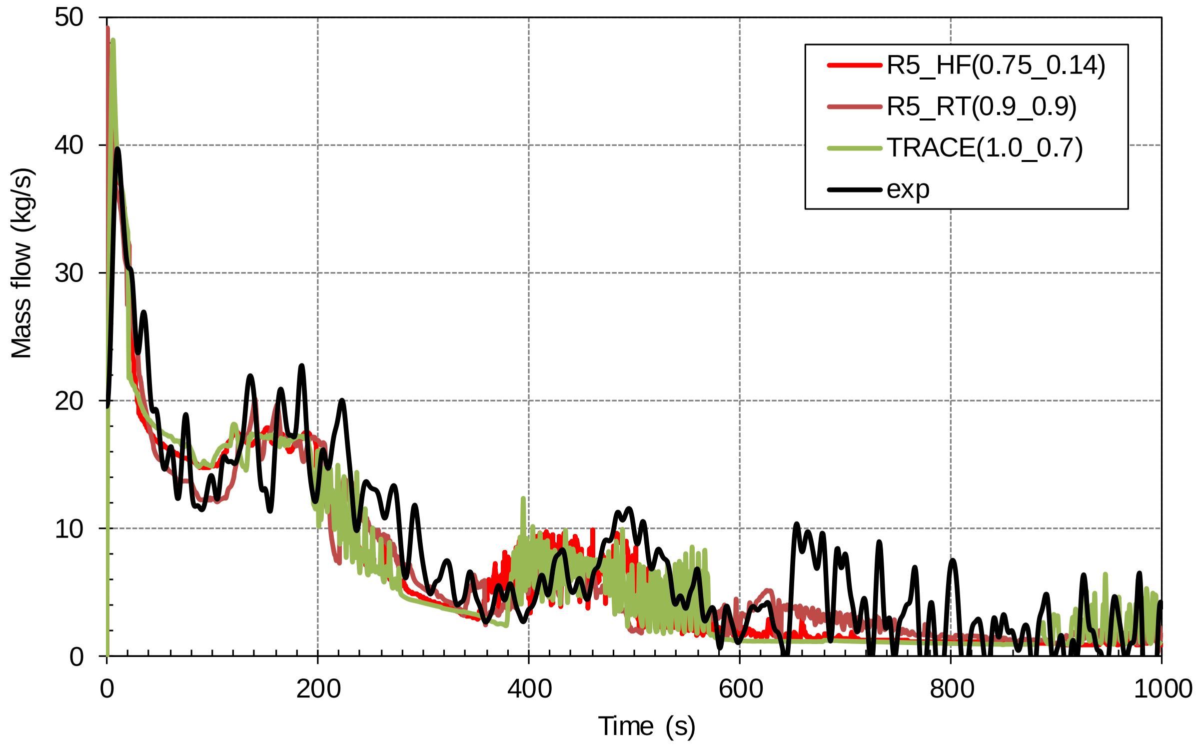

Figure 29 Break Mass Flow Rate – Tuned Case

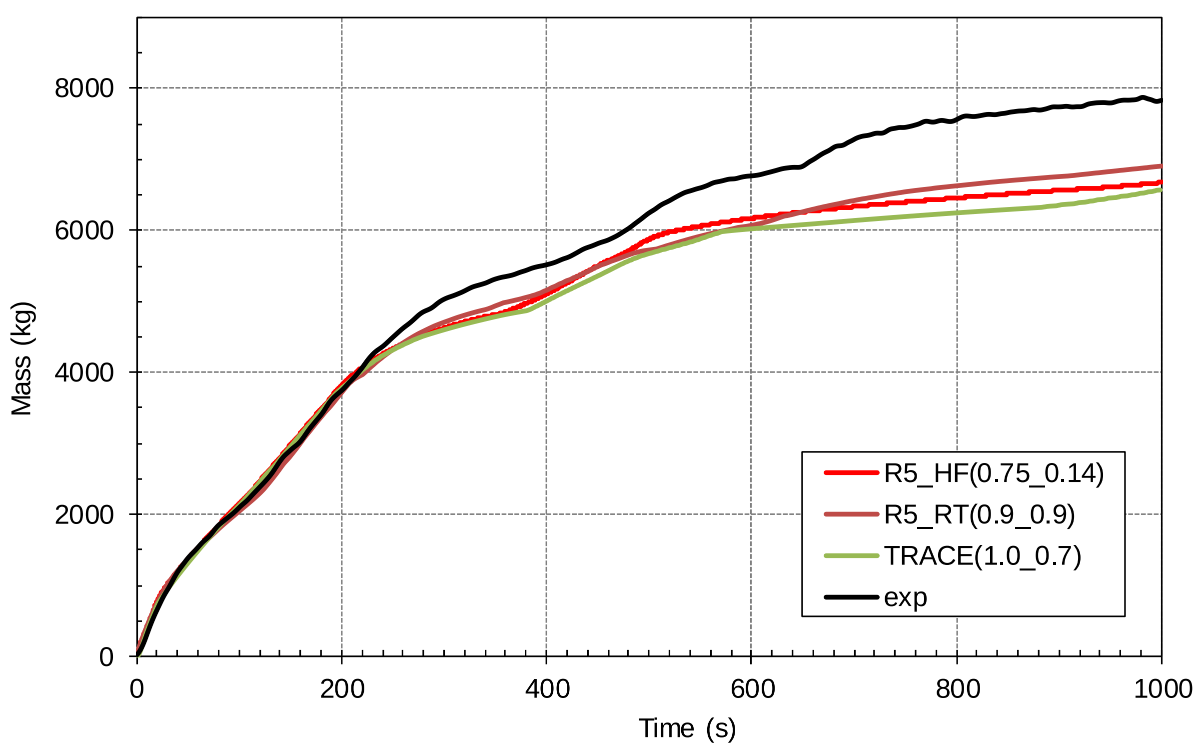

Figure 30 Integral of Break Mass Flow Rate – Tuned Case

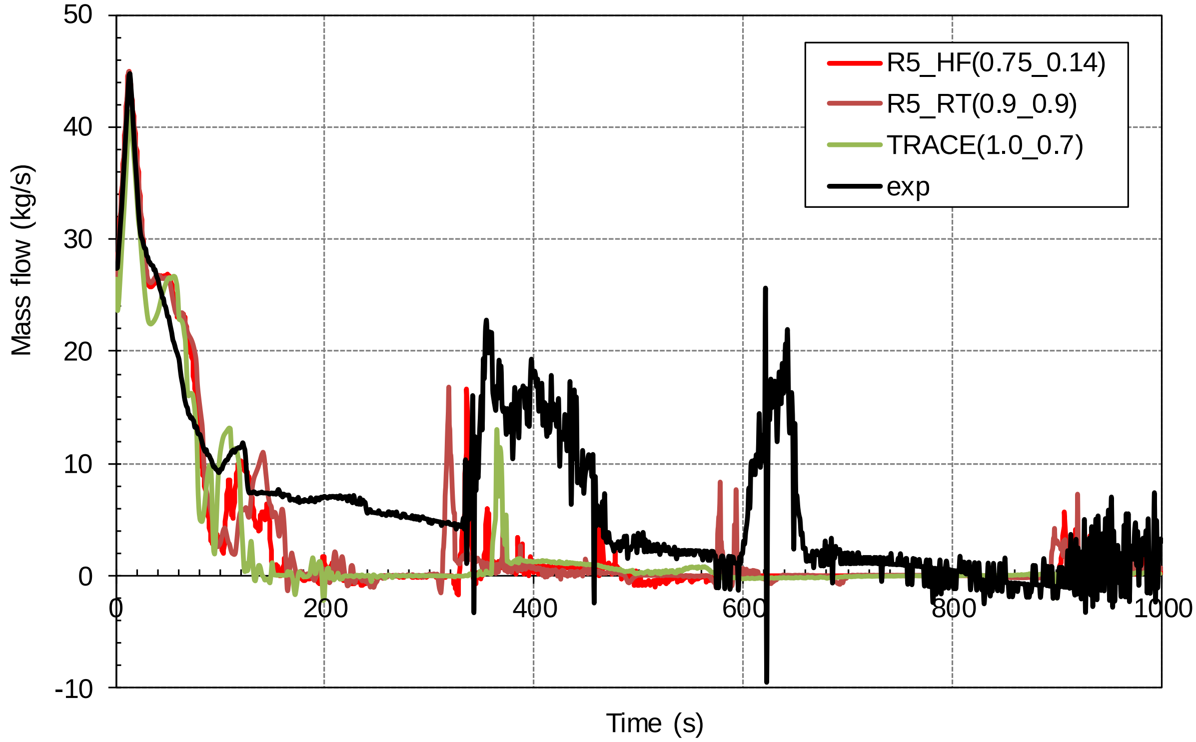

Figure 31 Primary Loop No. 1 Mass Flow Rate – Tuned Case

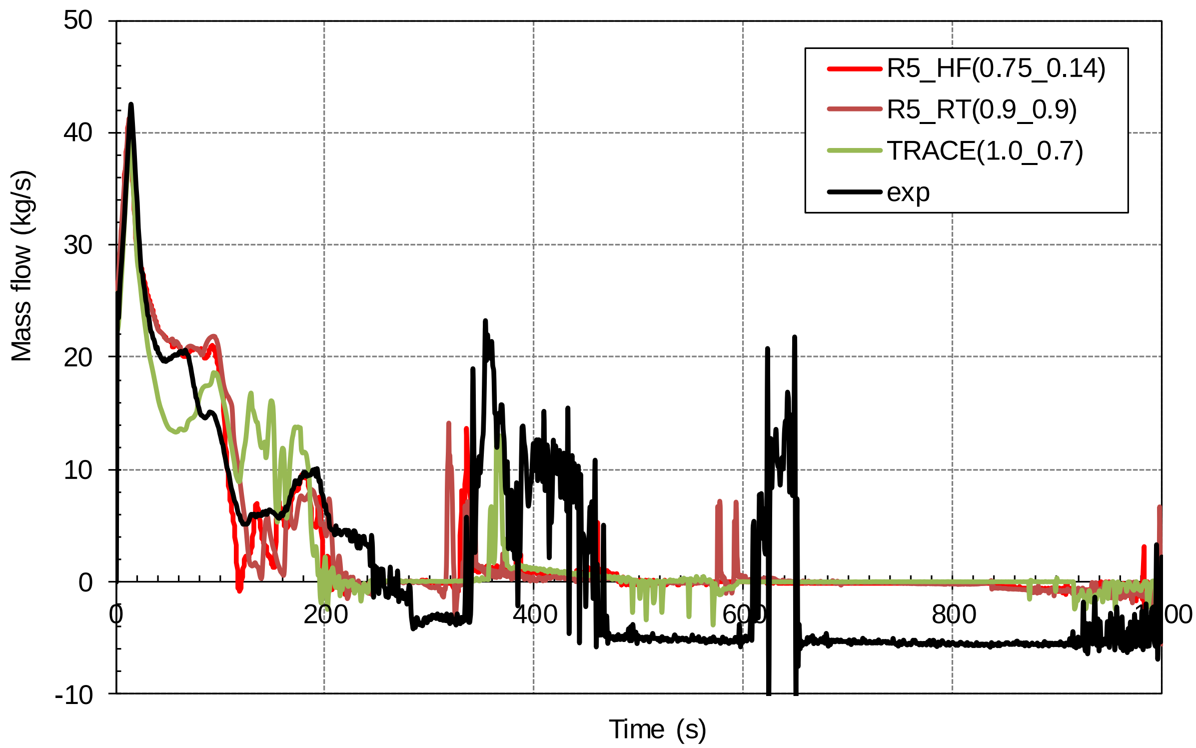

Figure 32 Primary Loop No. 2 Mass Flow Rate – Tuned Case

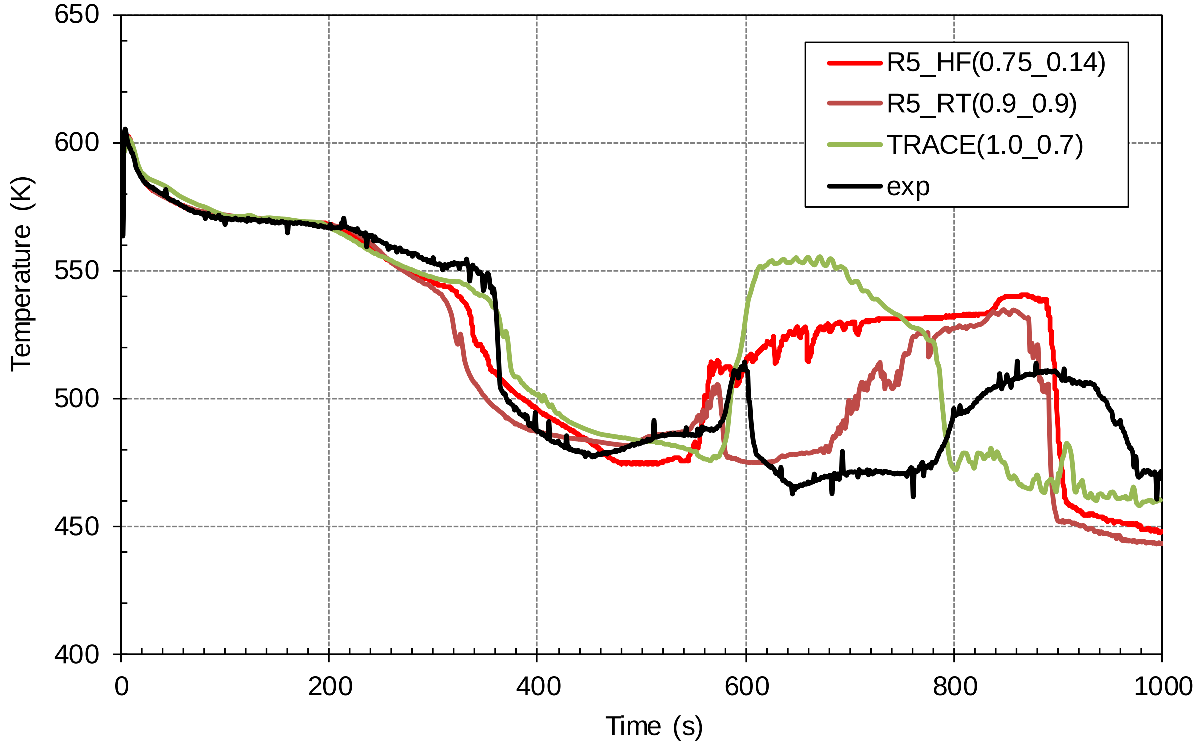

Figure 33 Hot Leg No. 1 Fluid Temperature – Tuned Case

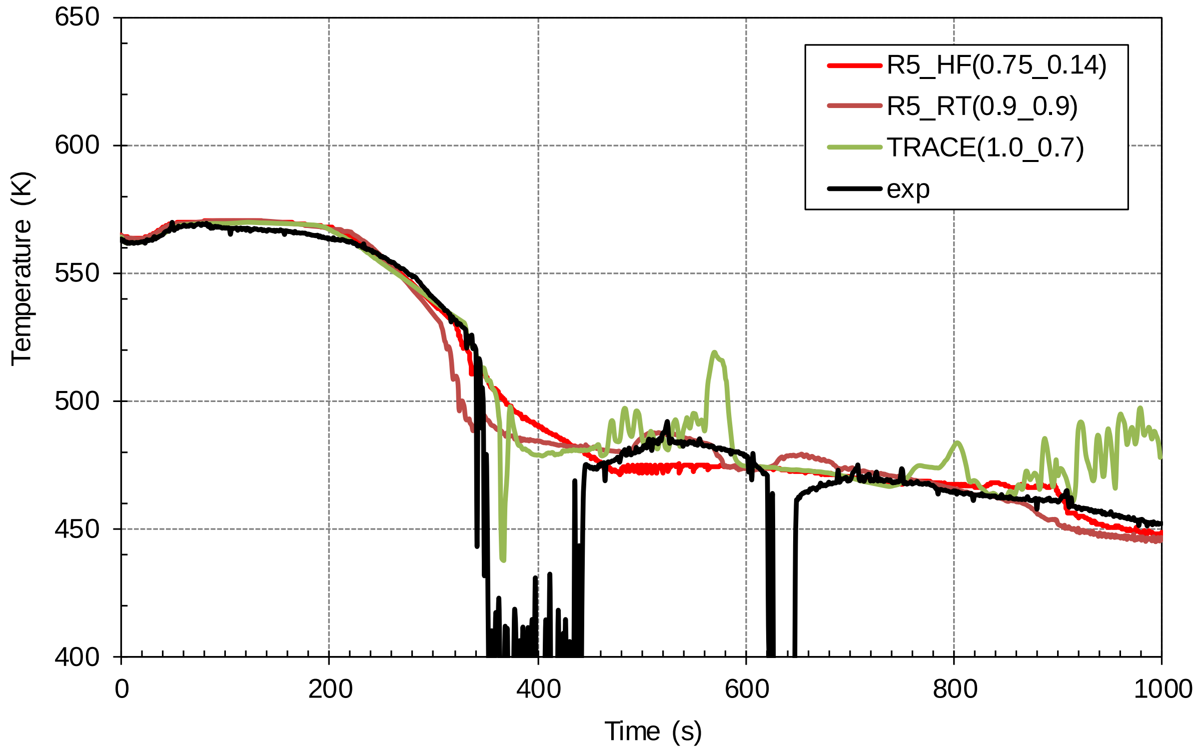

Figure 34 Cold Leg No. 1 Fluid Temperature – Tuned Case

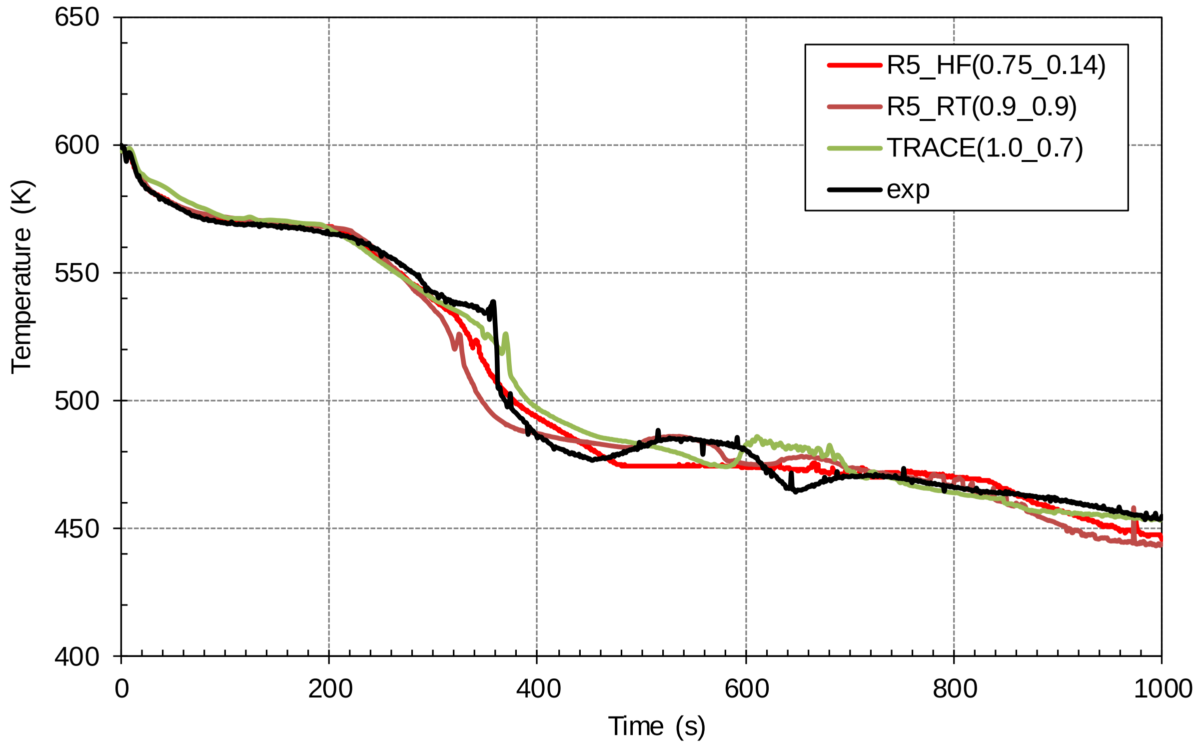

Figure 35 Hot Leg No. 2 Fluid Temperature – Tuned Case

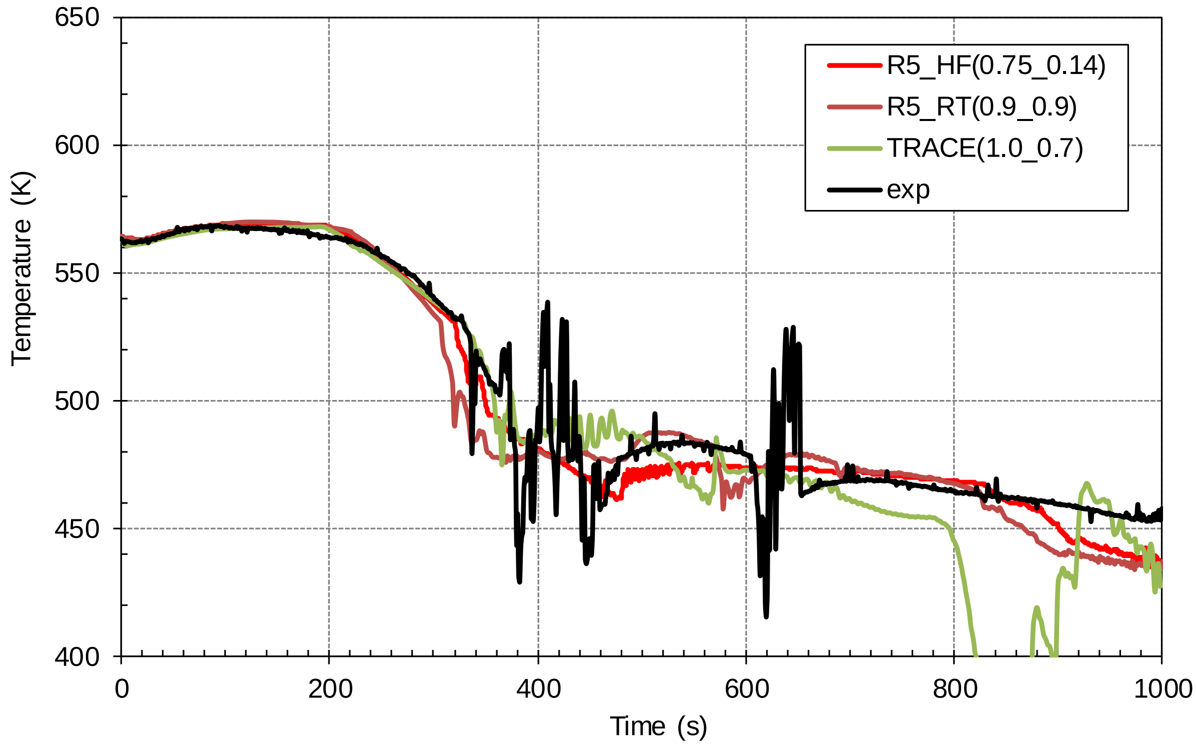

Figure 36 Cold Leg No. 2 Fluid Temperature – Tuned Case

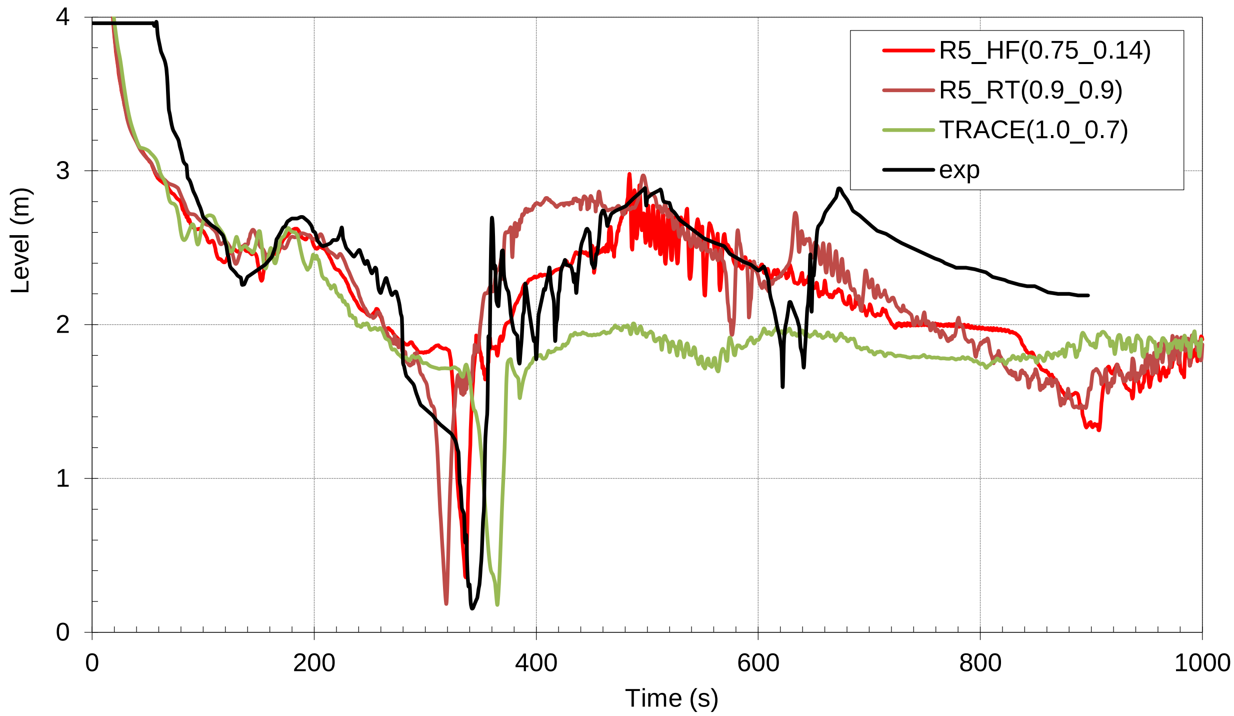

Figure 37 Core Collapsed Liquid Level – Tuned Case

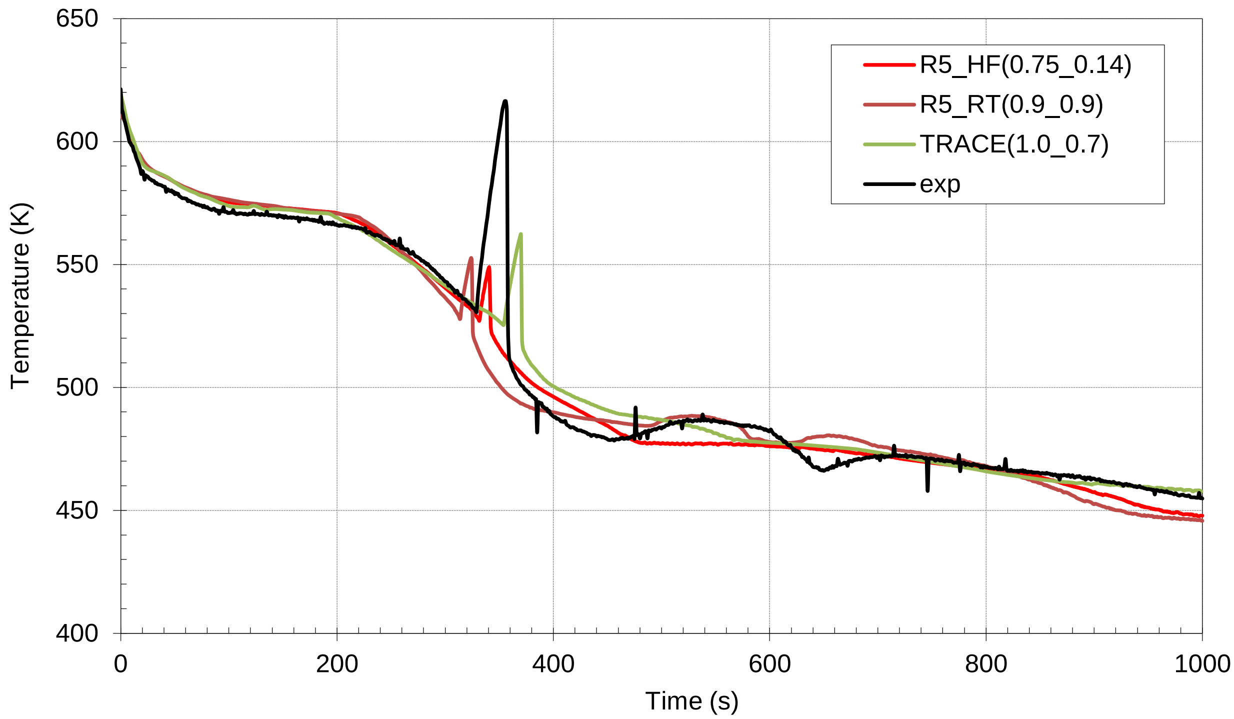

Figure 38 Fuel Rod Surface No. 7 Temperature – Tuned Case

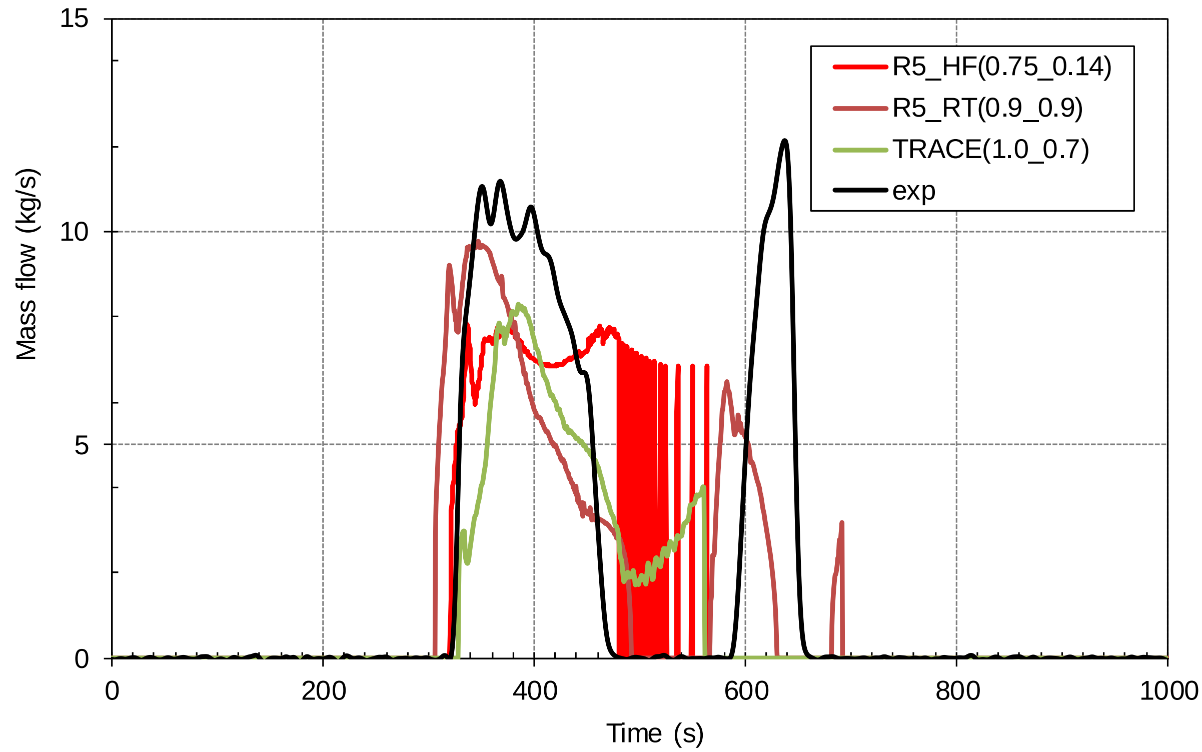

Figure 39 Accumulator No. 1 Flow Rate – Tuned Case

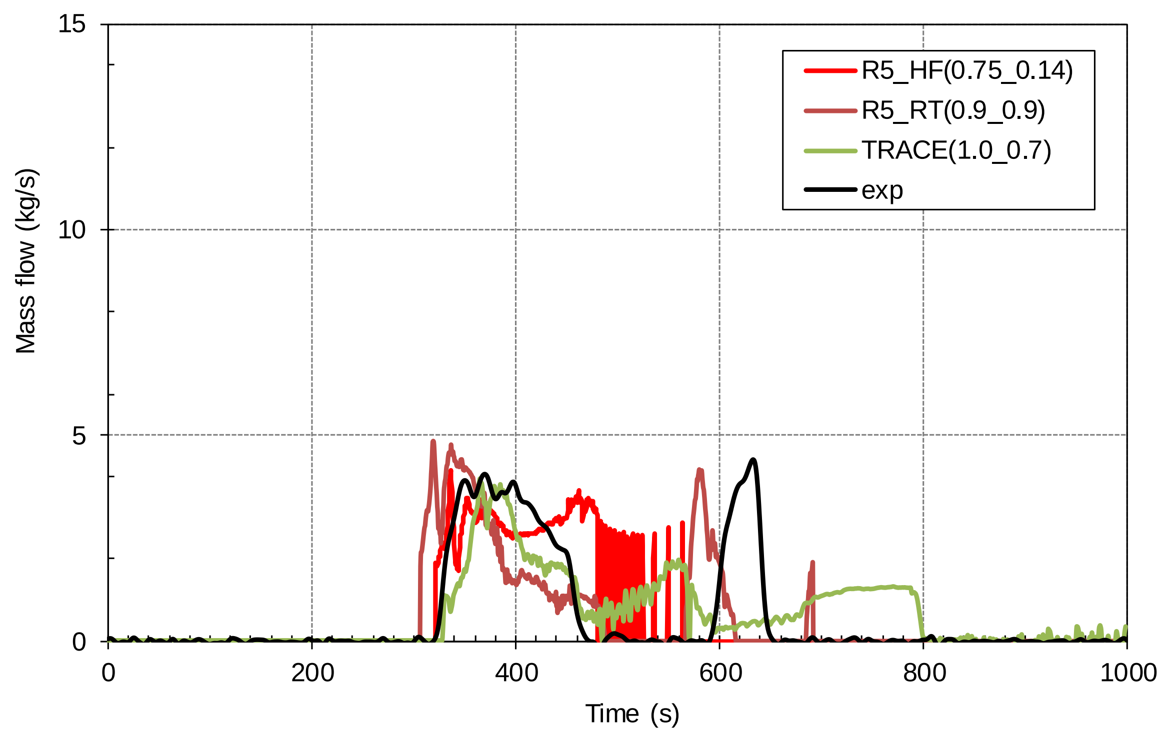

Figure 40 Accumulator No. 2 Flow Rate – Tuned Case

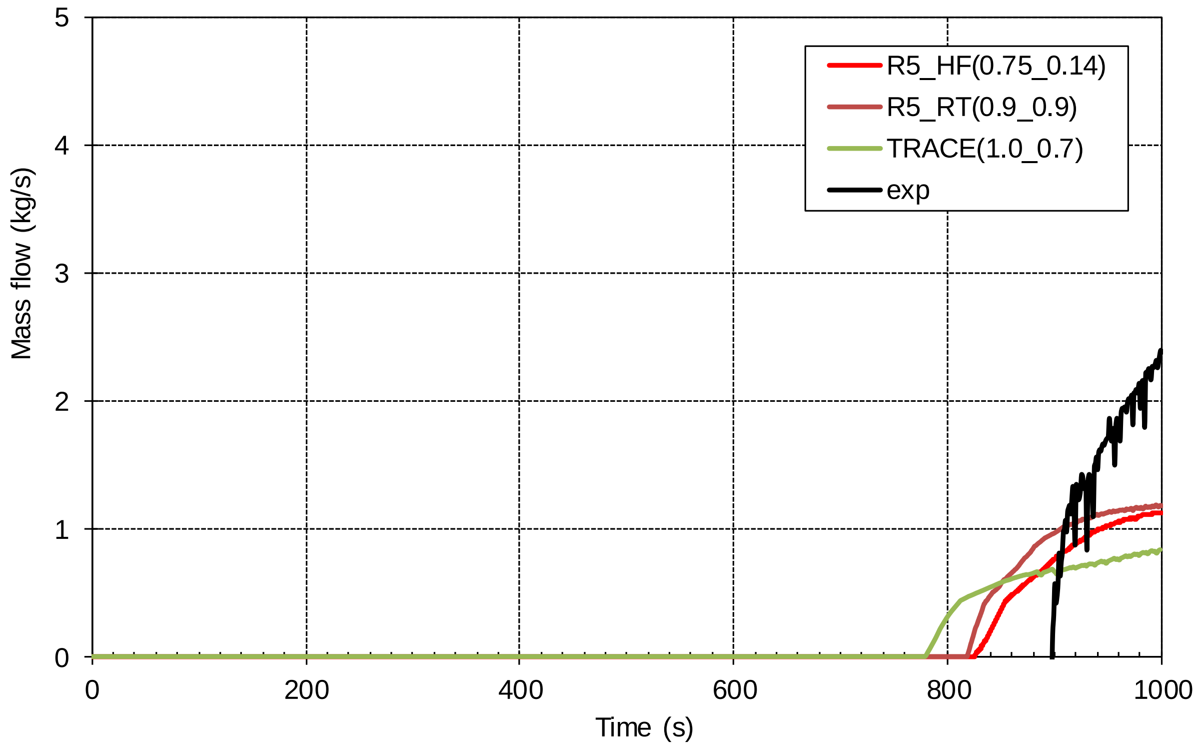

Figure 41 LPI No. 1 Flow Rate – Tuned Case

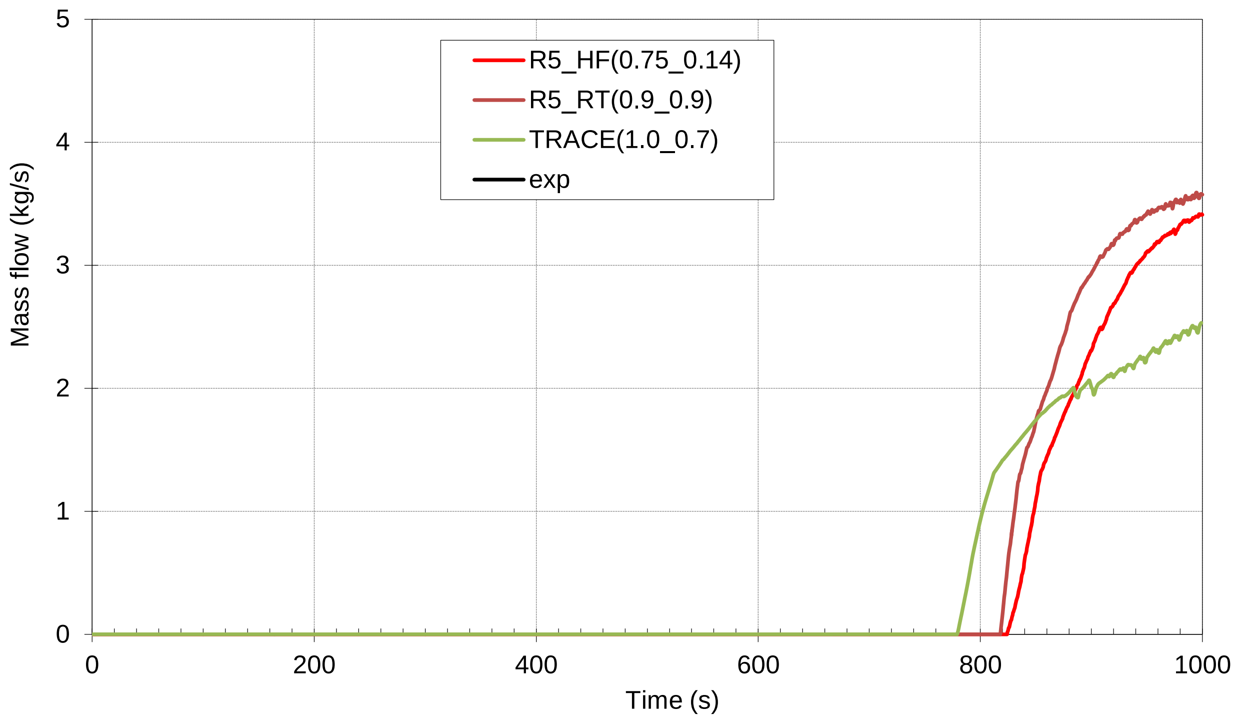

Figure 42 LPI No. 2 Flow Rate – Tuned Case

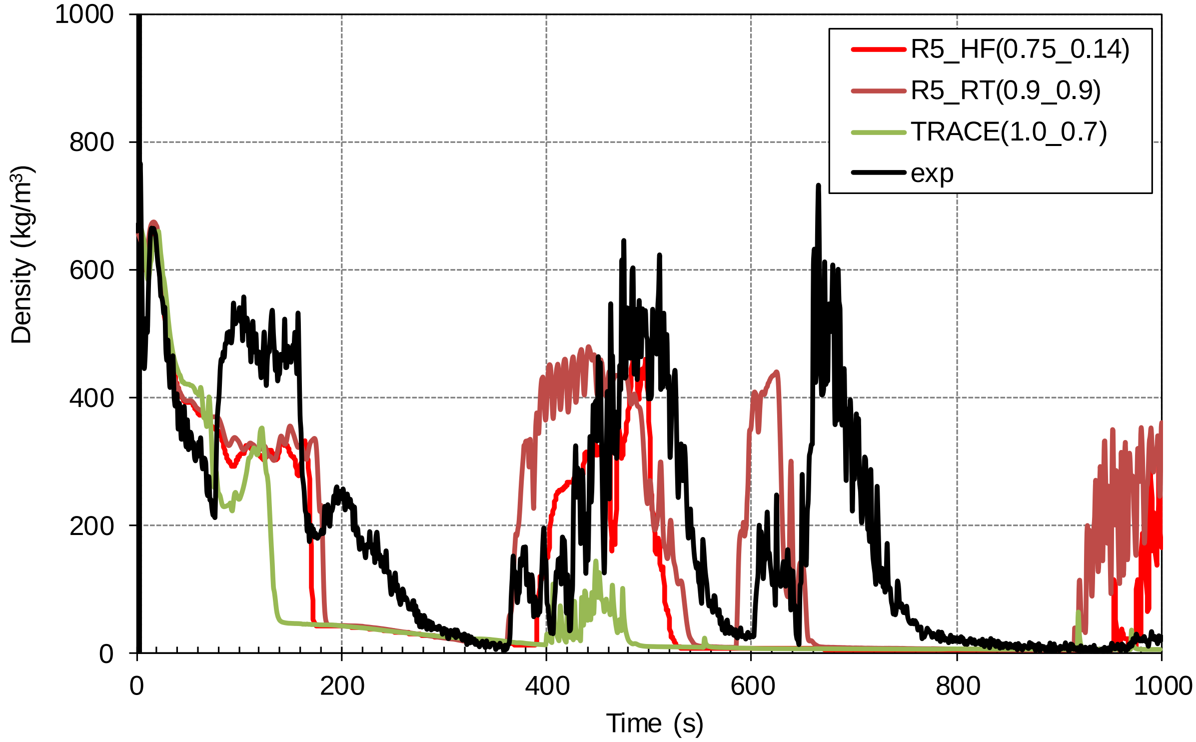

Figure 43 Hot Leg No. 1 Flow Density – Tuned Case

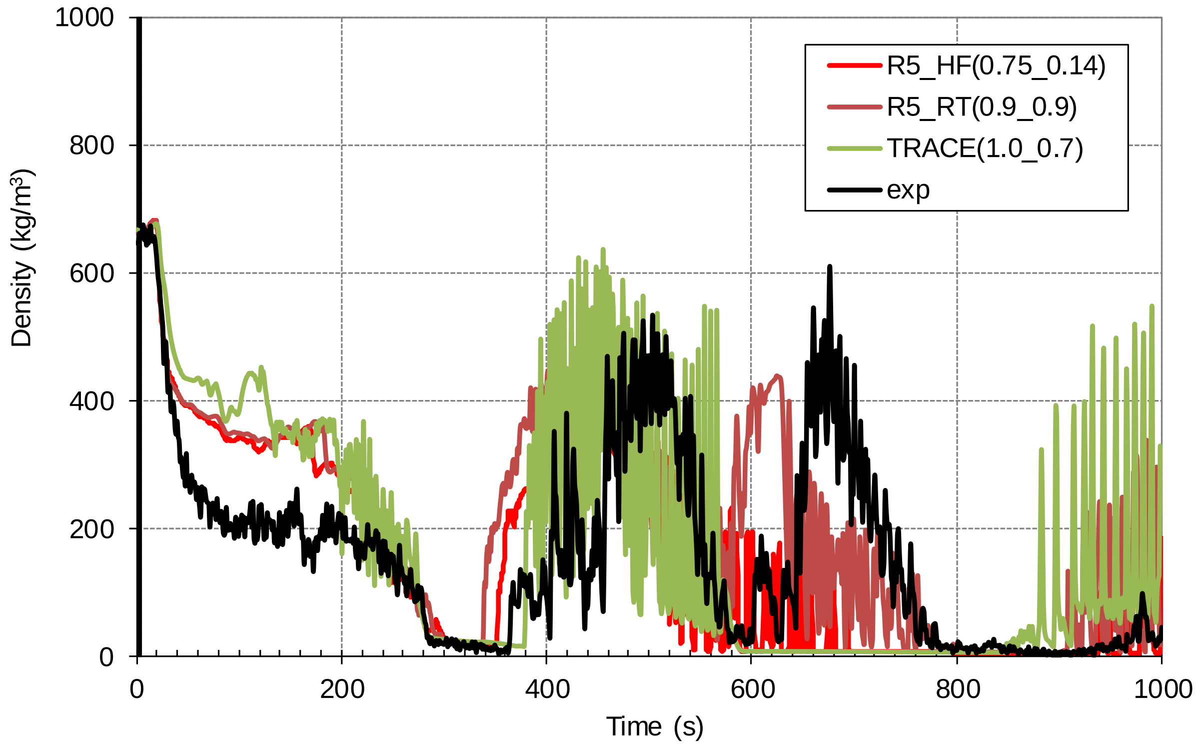

Figure 44 Hot Leg No. 2 Flow Density – Tuned Case

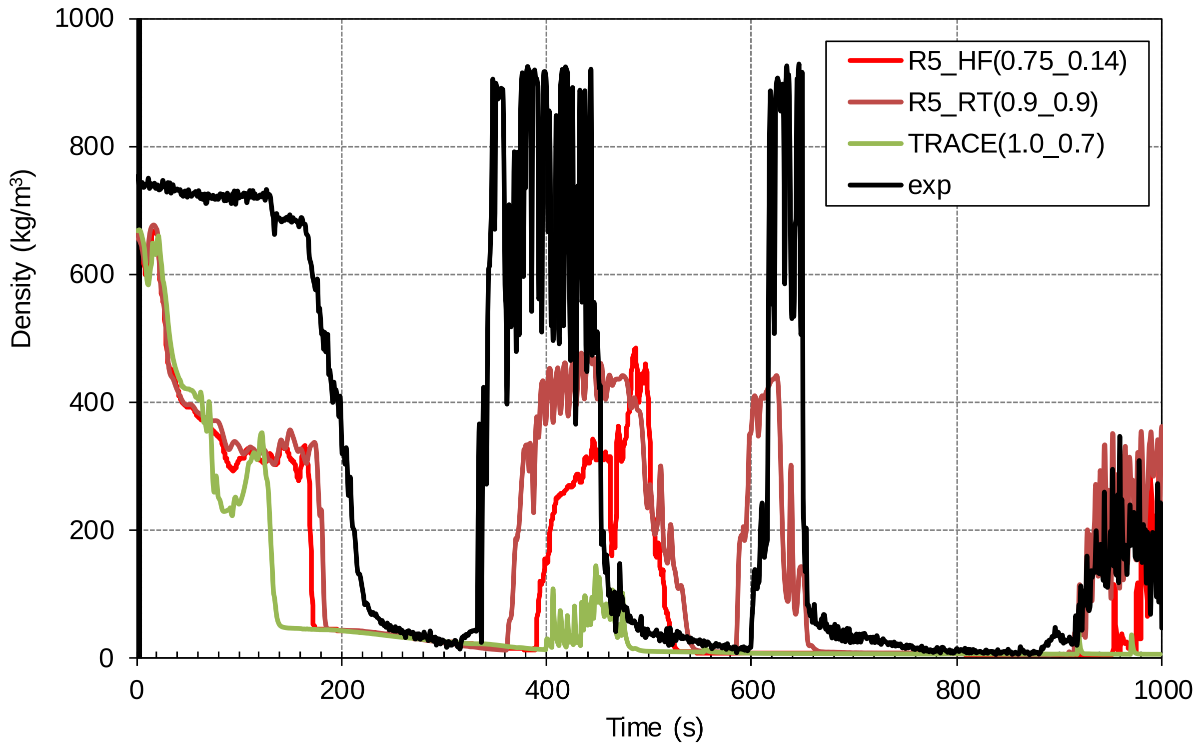

Figure 45 Cold Leg No. 1 Flow Density – Tuned Case

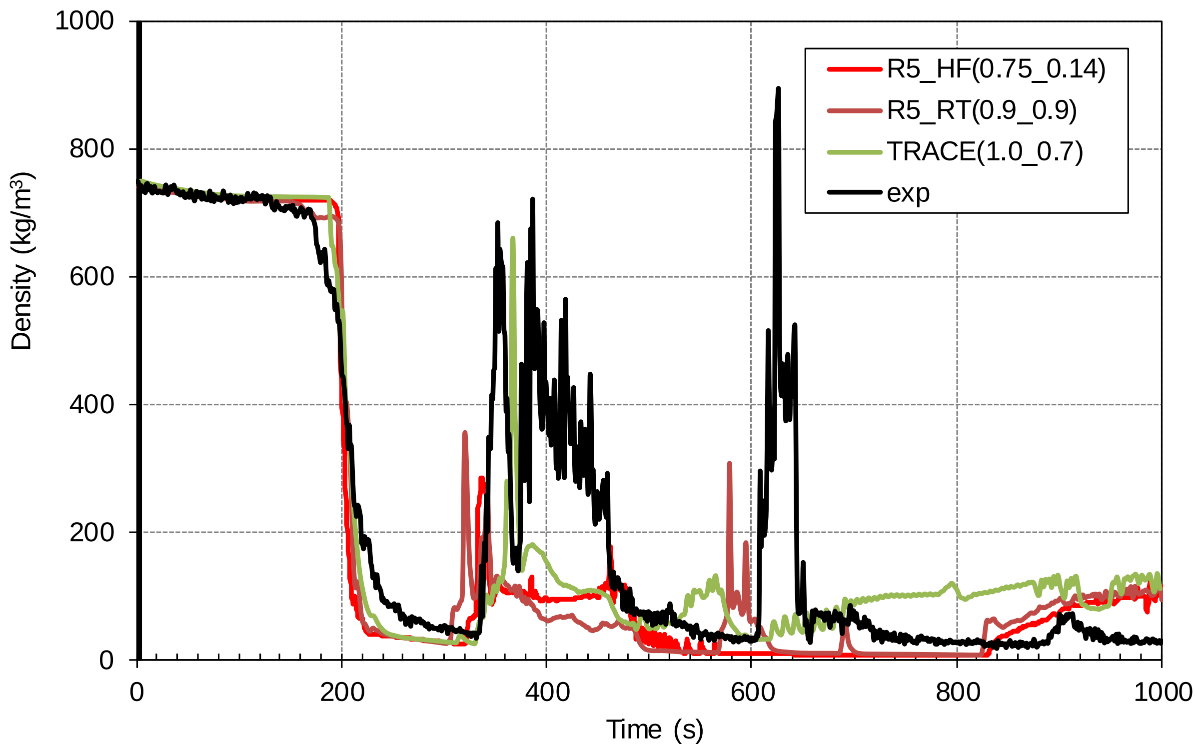

Figure 46 Cold Leg No. 2 Flow Density – Tuned Case

3.3 Animated Results

Figures 47 through 52 show mass distribution in LSTF. The SNAP animated results for void fraction obtained by RELAP5 calculation 'R5_RT(1.0_1.0)' are compared to the experimental collapsed liquid level data (Fig. 5.41 of Ref. 9). In RELAP5 animation liquid is represented by blue color (black shaded in case of measured data) and steam by white color. The comparison with experiment is not exact, when in the calculation the mixture is present, but it is true when only liquid or gas is present. Mass distribution in LSTF is shown at times 115 s, 195 s, 230 s, 300 s, 340 s and 400 s. The reader may refer to sequence of events shown in Table 1. As not all events are included in the sequence of events, in the following paragraph the events occurring at above times are briefly described.

The SG voiding behavior was asymmetric between the upflow and downflow side for each SG, and also was asymmetric between the two SGs. The SG no. 1 U tubes became empty of liquid at 115 s for the downflow side and at 150 s for the upflow side; SG no. 2 became empty at 195 s for the downflow side and 230 s for the upflow side. At 300 s core uncovering started. The core level recovered quickly after loop seals were cleared at 340 s. At 400 s no specific event occurred, but the core cooling is stable. The accumulators are injecting into the system and the pressure drops to about 2 MPa.

Figure 47 (top) shows that SG no. 1 U tubes became empty of liquid at 115 s for the downflow side and that uplow side is also much emptied in the test. Also in the RELAP5 calculation the SG no. 1 U tubes for the downflow side are empty, while this is not the case for the SG no. 2 U tubes for the downflow side (see Figure 47 (bottom)). The cold legs and downcomer are full of liquid, while in the core there is some voiding. The qualitative agreement is very good.

Figure 48 shows that at 195 s the upflow side SG no. 1 U tubes and hot leg is empty of liquid both in the test and calculation. Also the level in the downcomer decreased comparing to facility state at 115 s. Also, SG no. 2 became empty at 195 s for the downflow side, while for the upflow side there is still some liquid, both in experiment and calculation.

Figure 49 shows that at 230 s the loop seal in loop no. 1 and loop no. 2 is present, but start slowly to clear. SG no. 2 become empty for the upwlow side at 230 s, while there is some small amount of liquid obtained in calculation. The level in the downcomer further decreased.

Figure 50 shows collapsed liquid level distribution at 300 s, when core uncovering started. Due to mixture the comparison on mass could not be made directly, but the reader can get qualitative picture. The agreement between calculation and experiment is very good, except for cold leg and downcomer.

Figure 51 shows mass distribution at 340 s, when in the experiment the loop seal clearing started. In the calculation, the loop seals are cleared and the core level is therefore already recovered.

Finally, Figure 52 shows mass distribution at 400 s. The core is covered and the liquid is present in cold leg and downcomer, as accumulators are injecting. There is also some liquid in hot legs. The qualitative agreement between experiment and calculation is good.

Above comparisons give deep insight into liquid mass distribution, helping to understand natural circulation flow in different regimes. By this the advancement of graphical presentation in the last 30 years is effectively demonstrated. Especially, as movies can also be generated.

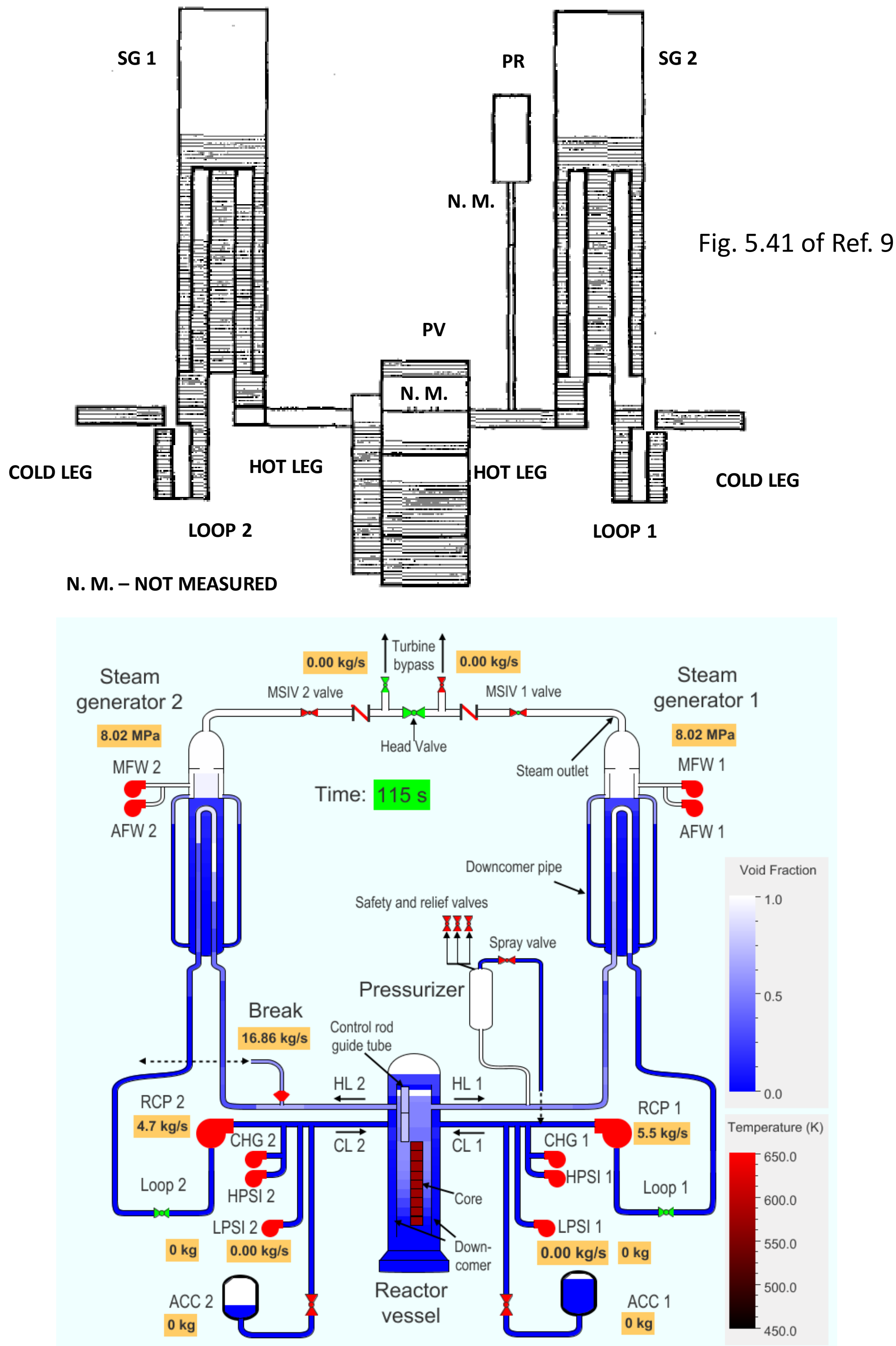

Figure 47 **Comparison Between Experiment (top) and RELAP5 (bottom) – Mass Distribution in LSTF at 115 s (SG no. 1 U tubes Empty in Downflow Side)**

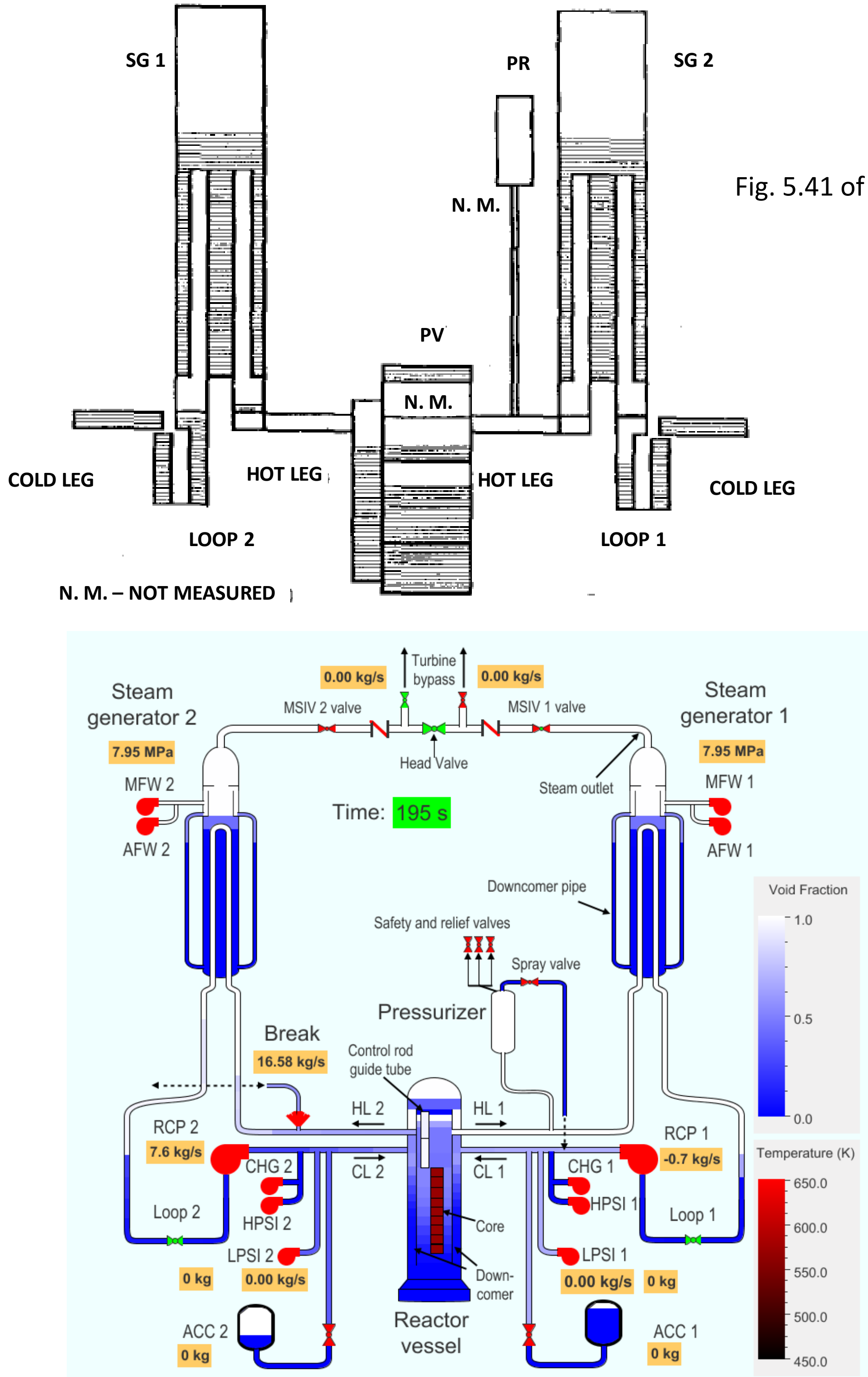

Figure 48 **Comparison Between Experiment (top) and RELAP5 (bottom) – Mass Distribution in LSTF at 195 s (SG No. 2 U Tubes Empty in Downflow Side)**

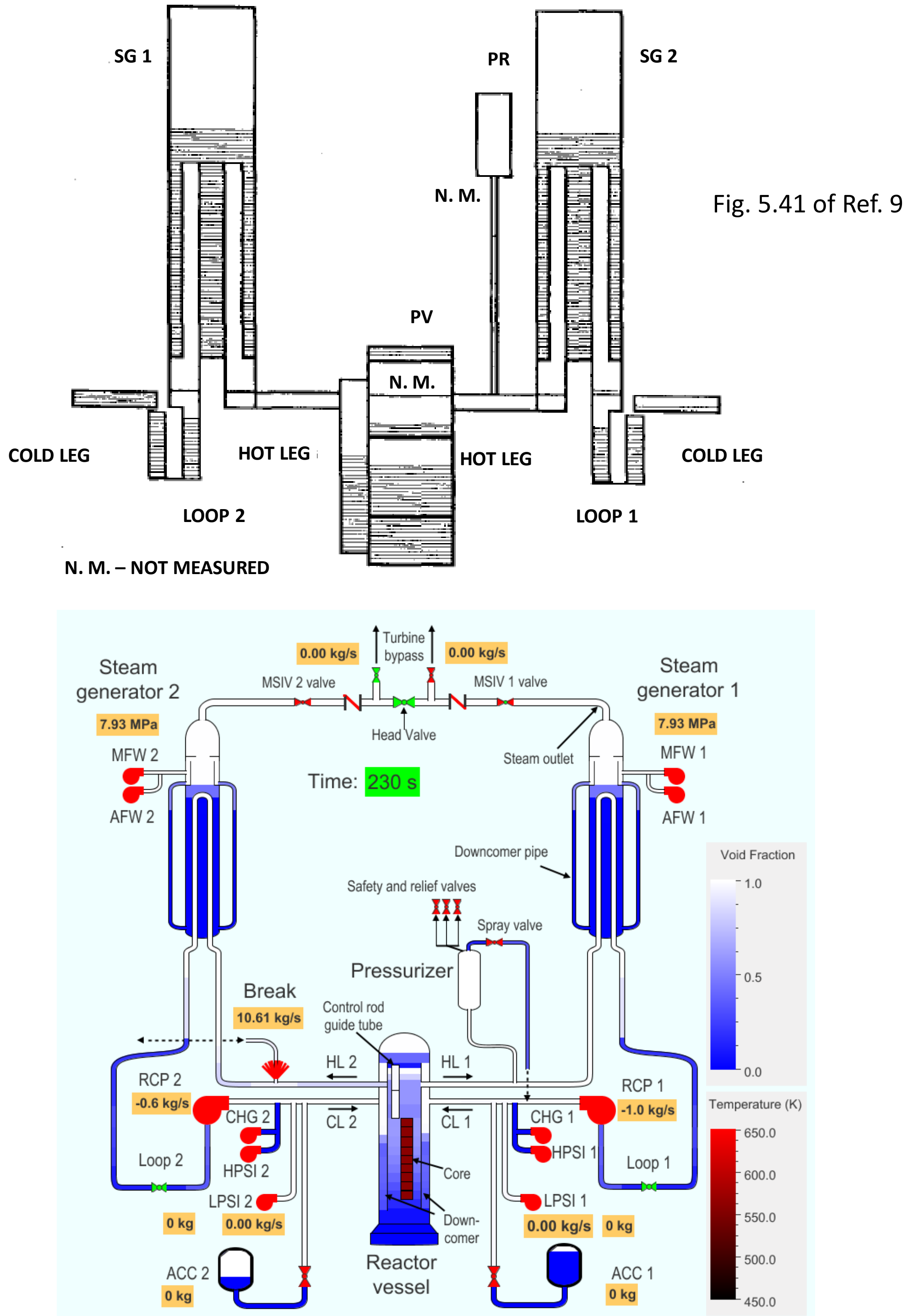

Figure 49 **Comparison Between Experiment (top) and RELAP5 (bottom) – Mass Distribution in LSTF at 230 s (Presence of Loop Seal in Test)**

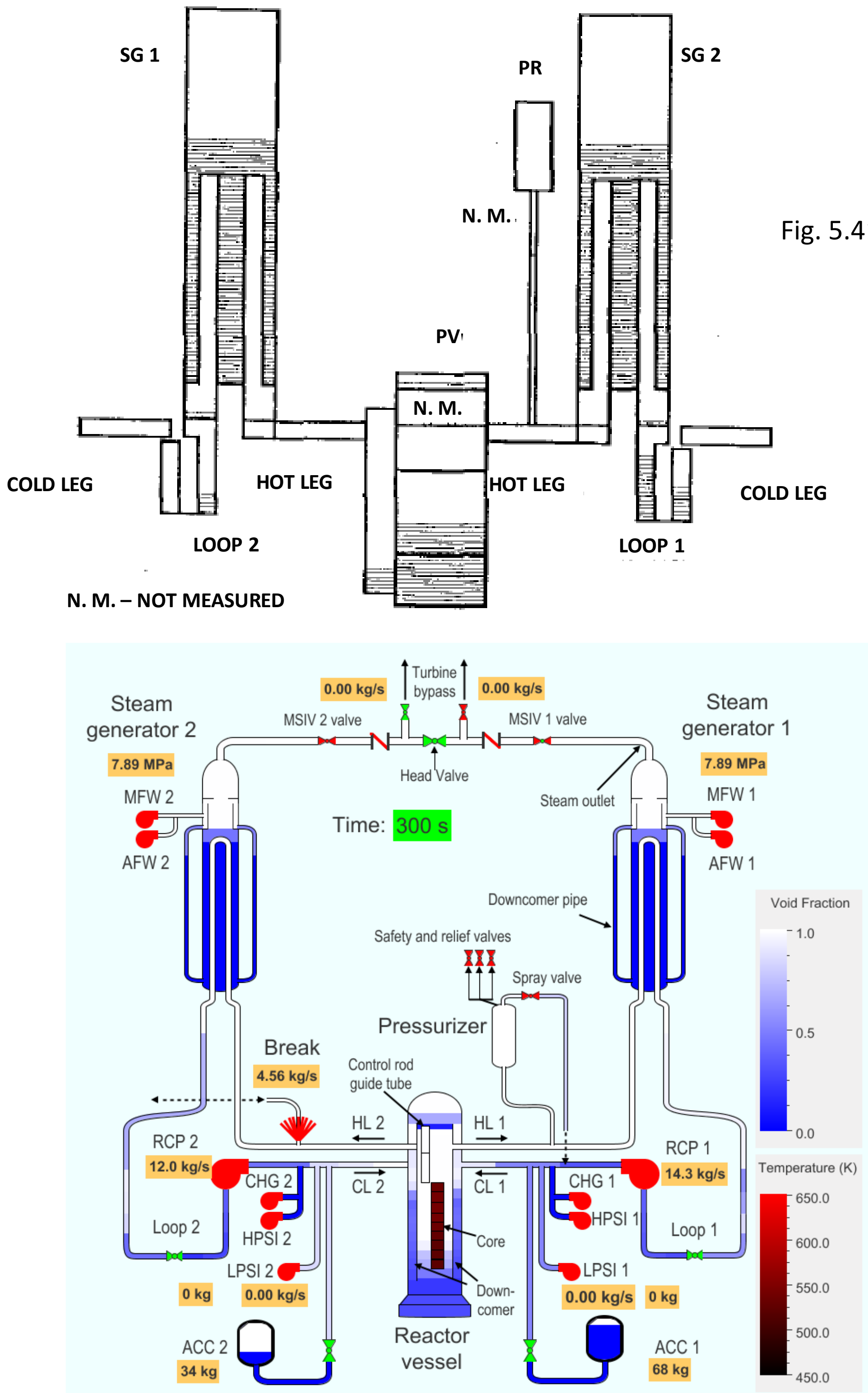

Figure 50 Comparison Between Experiment (top) and RELAP5 (bottom) – Mass Distribution in LSTF at 300 s (Core Uncovery Started in Test)

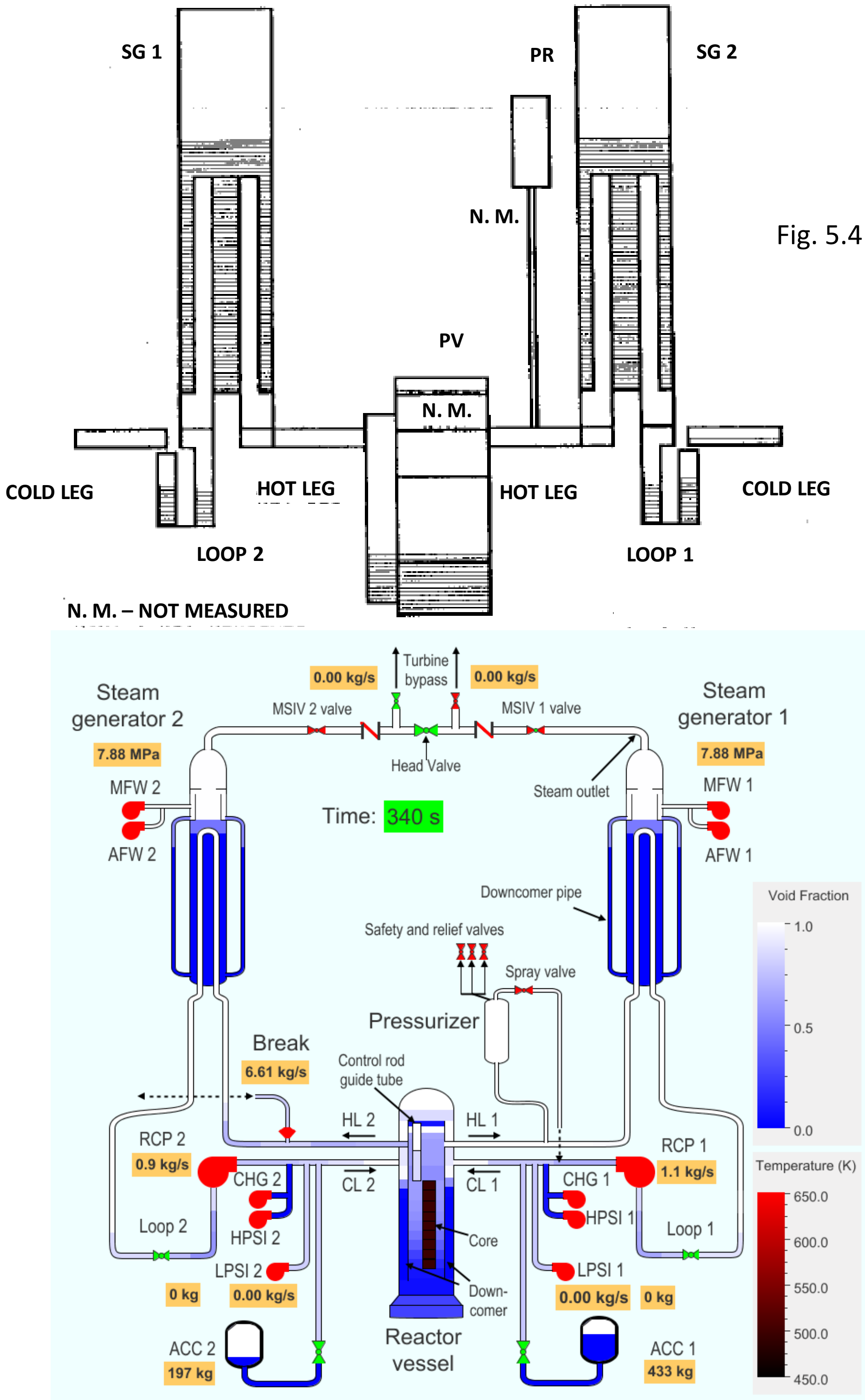

Fig. 5.41 of Ref. 9

Figure 51 Comparison Between Experiment (top) and RELAP5 (bottom) – Mass Distribution in LSTF at 340 s (Time of Loop Seal Clearing in Test)

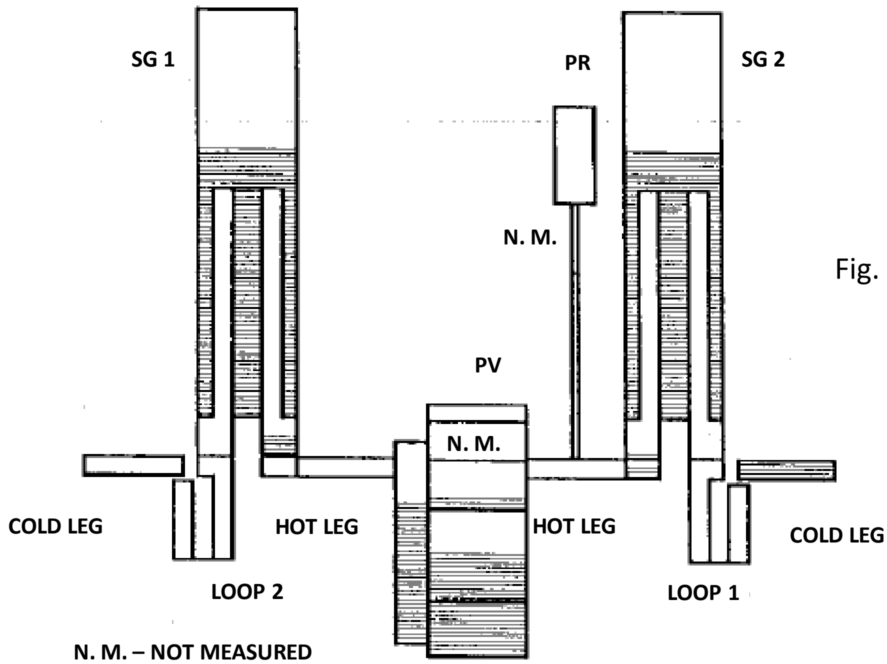

Fig. 5.41 of Ref. 9

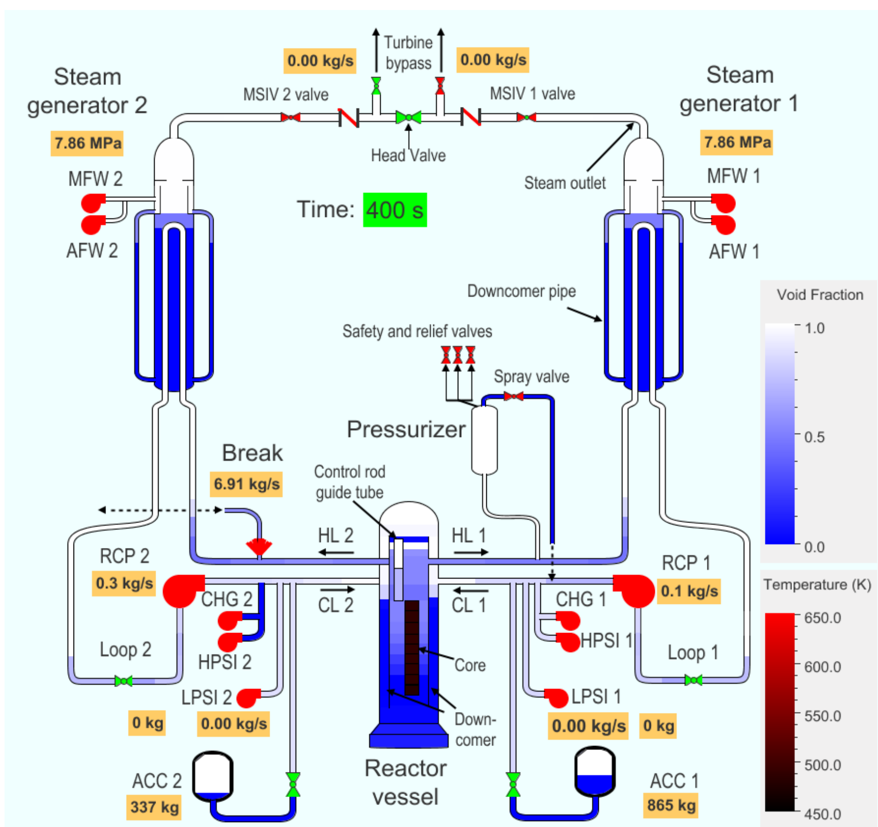

Figure 52 Comparison Between Experiment (top) and RELAP5 (bottom) – Mass Distribution in LSTF at 400 s (Core Covered and Accumulators Injecting in Test)

3.4 <u>Discussion of Results</u>

The results of calculations in which default break flow coefficients were used, do not agree perfectly with the experimental break flow. Primary pressure the most depend on break flow. In the base calculations the break flow in 'R5_HF(1.0_0.14)' and 'TRACE(1.0_1.0)' was initially too high, resulting in earlier pressure drop than in the test. Therefore also other parameters like fluid temperatures, core uncover and heatup, accumulators injection differ in timing. RELAP5 calculation 'R5_RT(1.0_1.0)' is better in this respect, but after 200 s the pressure drop is faster comparing to experimental data.

In a loss of coolant accident (LOCA) analysis of pressurized water reactor, the accurate prediction of break flow through the break during blowdown phase is very important in evaluating the remaining coolant inventory (Ref. 9). The coolant inventory has a first-order influence on the peak cladding temperature. During the blowdown phase of large break LOCA, the system pressure and liquid inventory are strongly affected by the critical flow model. The high pressure safety injection flow rate, pressurizer pressure, containment back pressure, and total energy release rate through the break are also affected by the critical flow model. If the code over-predicts (under-predicts) the break flow rate it will increase (decrease) the energy release rate from reactor coolant system (RCS). To show that the differences between the test and calculations are in great deal due to the break flow, the tuned calculations have been performed. Namely, with proper break flow modelling overall good agreement with experimental data both for RELAP5 and TRACE calculation can be obtained. The break flow coefficients were selected in such a way to best match the experimental pressure. The performed calculations confirm this hypothesis. The results suggest that TRACE calculation is comparable to RELAP5 calculations and that results obtained by both codes agree well with the experimental data.

4 CONCLUSIONS

A pressurized water reactor (PWR) hot leg break loss-of-coolant accident (LOCA) experiment SB-HL-02 was performed on the Large Scale Test Facility (LSTF) in the Rig of Safety Assessment-IV (ROSA-IV) program with a break size equivalent to 10% cold leg cross sectional area. For calculations, the RELAP5/MOD3.3 Patch 5 and TRACE V5.0 Patch 4 computer codes were used for base and break flow tuned calculations. The results suggest that TRACE calculation is comparable to RELAP5 calculations and that results obtained by both codes agree well with the experimental data for the break flow tuned valculations. Finally, it was also demonstrated that advanced Symbolic Nuclear Analysis Package (SNAP) graphical user interface has the capabilities to graphically present complex phenomena like collapsed liquid level distribution in the loop, helping to understand natural circulation flow in different regimes. Even more, comparison of calculated results with measured values could be done when data available.

5 REFERENCES

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11. ABSTRACT (200 words or less)

The accident at the Fukushima Dai-ichi nuclear power plant in 2011 demonstrated that external events could cause loss of all safety systems. In the Europe stress tests were performed and the need was identified to further improve the safety of the existing operating reactors. Therefore the safety upgrade programs were started. The objective of this study was to demonstrate that developed input model of two-loop pressurized water reactor (PWR) for TRACE thermal-hydraulic systems code can be used for independent calculations to be compared with RELAP5 computer code calculations. For demonstration the response of PWR to loss-of-coolant accident (LOCA) break spectrum from 10.16 cm (4 inch) to 30.48 cm (12 inch) was simulated. Only passive accumulators were assumed available. For calculations the latest TRACE Version 5.0 Patch 4 and RELAP5/MOD3.3 Patch 4 using both break flow models were used. The results showed that RELAP5 calculations using different break flow models are rather similar, therefore also other parameters are similar. The accumulators discharge was faster in TRACE calculation than in RELAP5 calculations. It can be concluded that different accumulator discharge influencing the break flow seems to be the largest contributor to the differences in the results between RELAP5 and TRACE.

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