International Agreement Report

Assessment of TRACE 5.0 Against ROSA-2 Test 5, Main Steam Line Break with Steam Generator Tube Rupture

Prepared by:
S. Gallardo, A. Querol, Lorduy, and G. Verdu

Universitat Politecnica de Valencia
Instituto Universitario de Seguridad Industrial, Radiofisica y Medioambiental
Cami de Vera s/n
46022 Valencia, SPAIN

Kirk Tien, NRC Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
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ABSTRACT

The purpose of this work is to overview the results obtained by the simulation, using the thermal-hydraulic code TRACE5, of Test 5 (SB-SG-14) in the frame of the OECD/NEA ROSA-2 Project. This test, conducted at the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA), simulates the thermal hydraulic responses after a PWR Steam Generator Tube Rupture (SGTR) induced by a Main Steam Line Break (MSLB). The result of these simultaneous breaks is a depressurization in both primary and secondary systems because they are connected through the SGTR. The actuation of the Accumulator Injection System was suppressed to keep primary coolant discharge to the Steam Generator secondary-side as low as possible.

The STGR is considered one of the main accidents in nuclear safety due to steam generator reliability and performance are serious concerns in the PWR operation. Through several studies, it has been reported that the severe accident management procedures such as foresee flooding and the primary system depressurization are used to minimize the release from the affected steam generator. These actions may significantly reduce the source term in SGTR accidents.

A comparison between experimental and simulation results is provided throughout several graphs, which represent the main thermal-hydraulic variables. In general, TRACE5 shows an acceptable behavior reproducing the experimental data in the entire transient.

FOREWORD

Thermalhydraulic studies play a key role in nuclear safety. Important areas where the significance and relevance of TH knowledge, data bases, methods and tools maintain an essential prominence are among others:

- assessment of plant modifications (e.g., Technical Specifications, power uprates, etc.);
- analysis of actual transients, incidents and/or start-up tests;
- development and verification of Emergency Operating Procedures;
- providing some elements for the Probabilistic Safety Assessments (e.g., success criteria and available time for manual actions, and sequence delineation) and its applications within the risk informed regulation framework;
- training personnel (e.g., full scope and engineering simulators); and/or
- assessment of new designs.

For that reason, the history of the involvement in Thermalhydraulics of CSN, nuclear Spanish Industry as well as Spanish universities, is long. It dates back to mid 80's when the first serious talks about Spain participation in LOFT-OCDE and ICAP Programs took place. Since then, CSN has paved a long way through several periods of CAMP programs, promoting coordinated joint efforts with Spanish organizations within different periods of associated national programs (i.e., CAMP-España).

From the CSN perspective, we have largely achieved the objectives. Models of our plants are in place, and an infrastructure of national TH experts, models, complementary tools, as well as an ample set of applications, have been created. The main task now is to maintain the expertise, to consolidate it and to update the experience. We at the CSN are aware on the need of maintaining key infrastructures and expertise, and see CAMP program as a good and well consolidated example of international collaborative action implementing recommendations on this issue.

Many experimental facilities have contributed to the today's availability of a large thermal-hydraulic database (both separated and integral effect tests). However there is a continuous need for additional experimental work and code development and verification, in areas where no emphasis have been made along the past. On the basis of the SESAR/FAP[1] reports "*Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk" (SESAR/FAP, 2001)* and its 2007 updated version "*Support Facilities for Existing and Advanced Reactors (SFEAR) NEA/CSNI/R(2007)6"*, CSNI is promoting since the beginning of this century several collaborative international actions in the area of experimental TH research. These reports presented some findings and recommendations to the CSNI, to sustain an adequate level of research, identifying a number of experimental facilities and programmes of potential interest for present or future international collaboration within the nuclear safety community during the coming decade. The different series of PKL, ROSA and ATLAS projects are under these premises.

CSN, as Spanish representative in CSNI, is involved in some of these research activities, helping in this international support of facilities and in the establishment of a large network of international collaborations. In the TH framework, most of these actions are either covering not enough investigated safety issues and phenomena (e.g., boron dilution, low power and shutdown conditions, beyond design accidents), or enlarging code validation and qualification data bases incorporating new information (e.g., multi-dimensional aspects, non-condensable gas effects, passive components).

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[1] SESAR/FAP is the *Senior Group of Experts on Nuclear Safety Research Facilities and Programmes* of NEA Committee on the Safety of Nuclear Installations (CSNI).

This NUREG/IA report is part of the Spanish contribution to CAMP focused on:

- Analysis, simulation and investigation of specific safety aspects of PKL2/OECD and ROSA2/OECD experiments.
- Analysis of applicability and/or extension of the results and knowledge acquired in these projects to the safety, operation or availability of the Spanish nuclear power plants.

Both objectives are carried out by simulating the experiments and conducting the plant application with the last available versions of NRC TH codes (RELAP5 and/or TRACE).
On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermalhydraulics analysis of accidents in the Spanish nuclear power plants. Nuclear safety needs have not decreased as the nuclear share of the nations grid is expected to be maintained if not increased during next years, with new plants in some countries, but also with older plants of higher power in most of the countries. This is the challenge that will require new ideas and a continued effort.

Rosario Velasco García, CSN Vice-president
Nuclear Safety Council (CSN) of Spain

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EXECUTIVE SUMMARY

This report presents the main results obtained in the simulation with the thermal hydraulic code TRACE5 of the Test 5 (SB-SG-14) in the frame of the OECD/NEA ROSA-2 Project. Test 5 simulates the thermal hydraulic responses after a PWR Steam Generator Tube Rupture (SGTR) induced by a Main Steam Line Break (MSLB) in the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA). These simultaneous breaks resulted in a depressurization of secondary and primary sides because both systems are connected through the SGTR. The actuation of the Emergency Core Cooling System (ECCS), which consists of the High Pressure Injection and the Low Pressure Injection systems, is assumed. Furthermore, the failure of the Accumulator Injection System is considered to keep primary coolant discharge to the Steam Generator secondary-side as low as possible.

A detailed TRACE5 model of the LSTF has been used to reproduce the transient. The simulation results have been compared with the experimental measurements in several graphs, including primary and secondary pressures, SGTR and MSLB mass flow rates, primary mass flow rates and collapsed liquid levels (in pressurizer, hot and cold legs, U-tubes, etc.).

ACKNOWLEDGMENTS

This paper contains findings that were produced within the OECD-NEA ROSA-2 Project. The authors are grateful to the Management Board of the ROSA Project for their consent to this publication, and thank the Spanish Nuclear Regulatory Body (CSN) for the technical and financial support under the agreement STN/1388/05/748.

ABBREVIATIONS AND ACRONYMS

AFW	Auxiliary Feedwater
AIS	Accumulator Injection System
AM	Accident Management
BE	Best Estimate
CAMP	Code Assessment and Management Program
CET	Core Exit Temperature
CPU	Central Processing Unit
CRGT	Control Rod Guide Tubes
CSN	Nuclear Safety Council, Spain
DBE	Design Basis Event
ECCS	Emergency Core Cooling System
HPI	High Pressure Injection
IBLOCA	Intermediate Break Loss-Of-Coolant Accident
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
JC	Jet Condenser
LOCA	Loss-Of-Coolant Accident
LPI	Low Pressure Injection
LSTF	Large Scale Test Facility
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
NV	Normalized to the Steady State Value
PA	Auxiliary Feedwater Pump
PCT	Peak Cladding Temperature
PF	Feedwater Pump
PGIT	Primary Gravity Injection Tank
PJ	High Pressure Charging Pump
PL	High Pressure Injection Pump
PORV	Power-Operated Relief Valve
PV	Pressure Vessel
PWR	Pressurized Water Reactor
PZR	Pressurizer
RHR	Residual Heat Removal
RV	Relief Valve
SBLOCA	Small Break Loss-Of-Coolant Accident
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SNAP	Symbolic Nuclear Analysis Package
SRV	Safety Relief Valve
ST	Storage Tank

1 INTRODUCTION

Evaluation of accident scenarios in actual plant conditions is very important in the operation of Nuclear Power Plants (NPP). One of the main accidents in the field of nuclear safety is the Steam Generator Tube Rupture (SGTR). Steam generator (SG) reliability and performance are serious concerns in the operation of PWR [1].

The flow rate through the broken U-tube depends on the primary-to-secondary side differential pressure in the affected SG, the primary coolant subcooling, and the break location along the U-tube [2].

Through several studies related to the SGTR [3, 4, 5], it has been reported that the severe accident management procedures such as foresee flooding of the secondary side through the emergency feedwater system and the primary system depressurization by opening the Power-Operated Relief Valve (PORV) can be used to minimize the release from the affected steam generator [6]. These actions may significantly reduce the source term in SGTR accidents.

In this frame, a simulation of the OECD/NEA ROSA-2 Project Test 5 (SB-SG-14) [7] has been performed using the thermal hydraulic code TRACE5 [8, 9]. Test 5 performed in the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA) [10] reproduced a Main Steam Line Break (MSLB) with a Steam Generator Tube Rupture (SGTR) in a PWR.

LSTF is a Full Height Full Pressure (FHFP) facility designed to simulate the Tsuruga unit II Nuclear Power Plant, a 4-loop Westinghouse PWR of 3423 MWt. The volumetric scaling factor is 1/48. The four primary loops of the reference PWR are represented by two equal-volume loops. The core power used to simulate the decay core power is 10 MW, corresponding to 14% of the 1/48 volumetrically scaled reference PWR rated power.

The result of simultaneous SGTR and MSLB is a depressurization in the secondary and primary systems of the affected loop because both systems are connected through the SGTR. In this experiment, the activation of the Emergency Core Cooling System (ECCS), which consists of High Pressure Injection (HPI) and Low Pressure Injection (LPI) systems, and the failure of the Accumulator Injection system (AIS) is assumed. The objective in this type of action is to minimize the primary coolant flow to the secondary system.

A detailed model of the LSTF with TRACE5 code has been used to reproduce the transient. The simulation results have been compared with experimental data, provided by the organization [10], in several graphs including primary and secondary pressures, SGTR and MSLB mass flow rates, primary mass flow rates, and collapsed liquid levels (in pressurizer, hot and cold legs, U-tubes, etc.).

2 LSTF FACILITY DESCRIPTION

In this section, a brief description of the LSTF facility (JAEA) [10] is presented. LSTF simulates a PWR reactor, Westinghouse type, of four loops and 3423 MW of thermal power. The facility is electrically heated, scaled 1:1 in height and 1:48 in flow areas and volumes, with exception of the loops, which are defined by a scaling factor of 1:24 in flow areas and volumes. Figure 1 shows the LSTF facility. As it can be seen, the primary coolant system consists of the Pressure Vessel (PV) and two symmetrical primary loops: loop A with the pressurizer (PZR) and loop B.

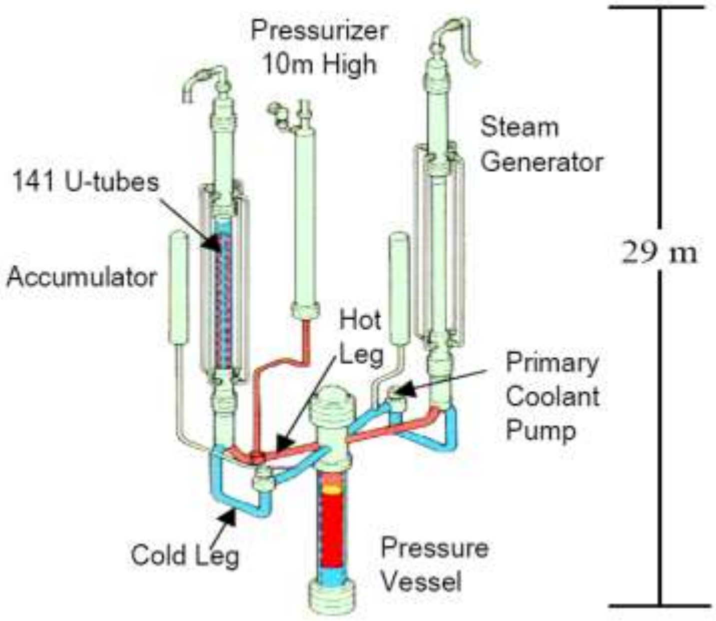

Figure 1 Schematic View of the LSTF Facility

Each loop contains a primary Coolant Pump (PC), a Steam Generator (SG) and an Accumulator tank. The secondary-coolant system consists of the jet condenser (JC), the Feedwater Pump (PF), the Auxiliary Feedwater Pumps (PA) and the necessary pipes to simulate the secondary system.

The pressure vessel is composed of an upper head above the upper core support plate, the upper plenum between the upper core support plate and the upper core plate, the core, the lower plenum and the downcomer annulus region surrounding the core and upper plenum. LSTF vessel has 8 upper head spray nozzles (of 3.4 mm inner-diameter). 8 Control Rod Guide Tubes (CRGTs) form the flow path between the upper head and upper plenum. The maximum core power of the LSTF is limited to 10 MW which corresponds to 14% of the volumetrically scaled PWR core power, being capable to simulate the PWR decay heat power after the reactor scram.

Regarding the steam generators, each of them contains 141 U-tubes, which can be classified in different groups depending on their length. The U-tubes have an inner diameter of 19.6 mm and an outer diameter of 25.4 mm (with 2.9 mm wall thickness). On the other hand, vessel, plenum and riser of steam generators have a height of 19.840, 1.183 and 17.827 m, respectively. The downcomer is 14.101 m in height.

3 TRANSIENT DESCRIPTION

In this section, the main actions and events characterizing this transient (SGTR and MSLB) are described. The complete control logic of this transient is listed in Table 1.

The experiment is initiated by opening simultaneously the two valves, which simulate the MSLB and the SGTR. The scram signal is produced when the Safety Injection (SI) signal is generated or the primary pressure drops to 12.97 MPa. This signal produces the initiation of the core power decay curve. In addition, the scram signal produces the initiation of the primary coolant pumps coast down and the turbine trip. The SI signal is generated when the broken steam generator secondary pressure decreases to 4.24 MPa or when the primary pressure decreases below 12.27 MPa. After the SI signal, the Auxiliary Feedwater (AFW) is started in the broken loop and it is finished when the steam generator secondary collapsed liquid level recovers the initial liquid level.

The High Pressure Injection (HPI) system is activated when the primary pressure drops to 12.27 MPa. About 30 minutes after the SI signal, the intact SG depressurization is started as Accident Management (AM) action by full opening the Relief Valve (RV). AFW is started in the intact loop simultaneously with the AM action.

Due to the quasi-equilibrium reached between the primary coolant loss through the SGTR and the coolant injected by the HPI, the primary pressure stagnates. In this moment, the Power-Operated Relief Valve (PORV) is opened to recover the liquid level of the pressurizer and it is closed when the pressurizer liquid level recovers 1 m. After the first PORV closure, the HPI mass flow rate is reduced to a half.

Some minutes later, a new stagnation of the primary pressure is reached. The PORV is opened again to recover liquid level in the pressurizer and it is closed at the same liquid level than the first closure. The HPI system is terminated after the second PORV closure.
Finally, when the pressure vessel lower plenum pressure is lower than 1.24 MPa, the Low Pressure Injection system is activated. The transient finishes when nearly-equilibrium in primary and secondary pressures is reached.

Table 1 Control Logic and Sequence of Major Events in the Experiment

MSLB concurrent with SGTR.	Time zero
Generation of scram signal.	Safety Injection (SI) signal or primary pressure reaches 12.97 MPa.
Generation of SI signal.	Broken SG secondary-side pressure (4.24 MPa) or primary pressure reaches 12.27 MPa.
Pressurizer (PZR) heater off.	Generation of scram signal or SI signal or PZR liquid level below 2.3 m.
Initiation of core power decay curve.	Generation of scram signal.
Initiation of Primary Coolant Pumps coastdown.	Generation of scram signal.
Turbine trip.	Generation of scram signal.
Closure of steam generators (SG) Main Steam Isolation Valves (MSIVs).	Generation of scram signal.
Termination of steam generator Main Feedwater (MFW).	Generation of scram signal.
Initiation of Auxiliary Feedwater (AFW) in the broken loop.	Generation of SI signal.
Initiation of High Pressure Injection (HPI) system in both loops.	PV lower plenum pressure reaches 12.27 MPa
Initiation of intact SG secondary-side depressurization by fully opening Relief Valves (RVs) as AM action.	30 minutes after SI signal
Initiation of AFW in intact loop.	Simultaneously with AM action
Power-Operated Relief Valve (PORV) open (twice).	Primary pressure stagnates at full and half of HPI flow rate capacity.
PORV closure (twice).	PZR liquid level reaches 1 m after PORV open
Termination AFW in broken loop.	SG secondary-side collapsed liquid level reaches the initial liquid level.
Reduction from full to half of HPI flow rate capacity.	1st PORV closure.
Termination of HPI system in both loops.	2nd PORV closure.
Initiation of Low Pressure Injection (LPI) system in both loops.	PV lower plenum pressure lower than 1.24 MPa.

4 TRACE5 MODEL OF LSTF

The LSTF has been modeled using TRACE5 with 83 hydraulic components (7 BREAKs, 11FILLs, 23 PIPEs, 2 PUMPs, 1 PRIZER, 22 TEEs, 15 VALVEs and 1 VESSEL). Figure 2 shows the nodalization of the model using the Symbolic Nuclear Analysis Package software (SNAP) [11].

The LSTF model contains the two loops of the facility, each one provided with primary and secondary sides. The primary side comprises cold and hot legs, pumps, loop seals, a pressurizer in loop A, the Emergency Core Cooling System (ECCS) which includes Accumulator Injection System (AIS), High Pressure Injection (HPI) and Low Pressure Injection (LPI) systems, the U-tubes of both Steams Generators and the Pressure Vessel. On the other hand, the secondary side includes steam separators, downcomers, Safety Relief Valves (SRV), Main Steam Isolation Valves (MSIV) and FILLs to provide Main and Auxiliary Feedwater (MFW and AFW, respectively).

Heat transfer between primary and secondary side, pressurizer heaters and heat losses have been performed by using 50 Heat Structure (HTSTR) components. Cylindrical-shape geometry has been used to best fit heat transmission. The power supplied to the vessel from 1008 fuel elements present in the LSTF has been simulated using a POWER component.
The pressure vessel has been modelled using a 3D–VESSEL component. The VESSEL consists of 20 axial levels, 4 radial rings and 10 azimuthal sectors. This nodalization characterizes with in sufficient detail the actual features of the LSTF vessel. For each axial level, volume and effective flow area fractions have been set according to technical specifications provided by the organization [10]. Levels 1 and 2 simulate the lower plenum. Active core is located between levels 3 and 11. Level 12 simulates the upper core plate. Levels 13 to 16 characterize the vessel upper plenum. The upper core support plate is located in level 17. The upper head is defined between levels 18 to 20. The 3D-VESSEL is connected to different 1D components: 8 Control Rod Guide Tubes (CRGT), hot leg A and B (level 15), cold leg A and B (level 15) and a bypass channel (level 14). Control rod guide tubes have been simulated by PIPE components, connecting levels 14 and 19 and allowing the flow between upper head and upper plenum.

The power ratio in the axial direction presents a peaking factor of 1.495, while the radial power profile is divided into three power zones using the first three radial rings. Depending on the radial ring, different peaking factors have been considered (0.66 in ring 1, 1.51 in ring 2 and 1.0 in ring 3).

30 HTSTR components simulate the fuel assemblies in the active core. A POWER component manages the power supplied by each HTSTR to the 3D-VESSEL. Fuel elements (1008 in total) are distributed into the 3 rings: 154 elements in ring 1, 356 in ring 2 and 498 in ring 3. The number of fuel rod components associated with each heat structure has been determined from the technical documentation given, considering the distribution of fuel rod elements in the vessel. A detailed model of the steam generators (geometry and thermal features) has been developed. Boiler and downcomer have been modeled by TEE components. The U-tubes have been classified into three groups according to their length.

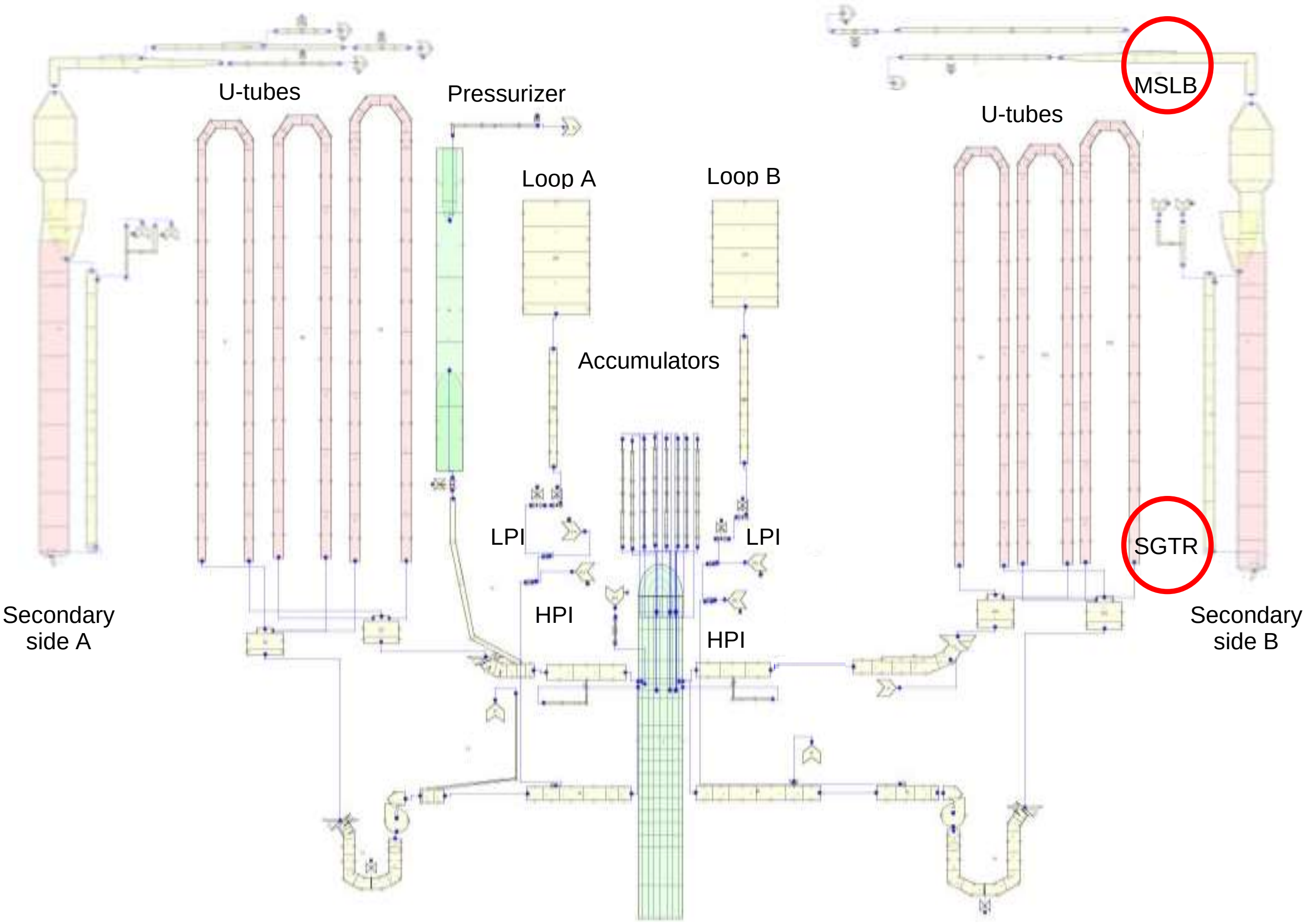

Figure 2 Model Nodalization Used for Simulation

The steam separator model can be activated in TRACE5 setting a friction coefficient (FRIC) greater than 10^{22} at a determined cell edge, allowing only gas phase to flow through the cell interface. Heat transfer between primary and secondary sides has been performed using HTSTR components. Cylindrical-shape geometry has been used to best fit heat transmission. Different models varying the number of U-tube groups were tested (1, 3 and 6 groups). It was found that results do not apparently change, using these models. However, in order to best fit the collapsed liquid level in the U-tubes without drastically increasing CPU time, a 3-group configuration was chosen. Heat losses to the environment have been added to the secondary-side walls.

The SGTR reproduces a U-Tube rupture in the steam generator B (flow area = 30.19 mm^2 and hydraulic diameter = 6.2 mm), which means a communication between primary and secondary sides. This break is simulated with a PIPE component, between the base of the U-tube bundle and the secondary side. In the model, three new components (2 TEEs and 1 VALVE) have been added to simulate the connection between the primary and secondary sides. When this new valve is open, both TEE components are connected simulating the nozzle used in the experiment.

On the other hand, the MSLB is reproduced by a Break Unit (BU), which starts in the steam line of the steam generator B and discharges into a storage tank. In the model three new components (1 TEE, 1 VALVE and 1 BREAK) have been added. The TEE is connected to the BREAK using the VALVE. The BREAK simulates the atmospheric conditions and the valve simulates the break when it is open. Figure 3 shows a diagram with the nodalization used to reproduce the SGTR and the MSLB in the model.

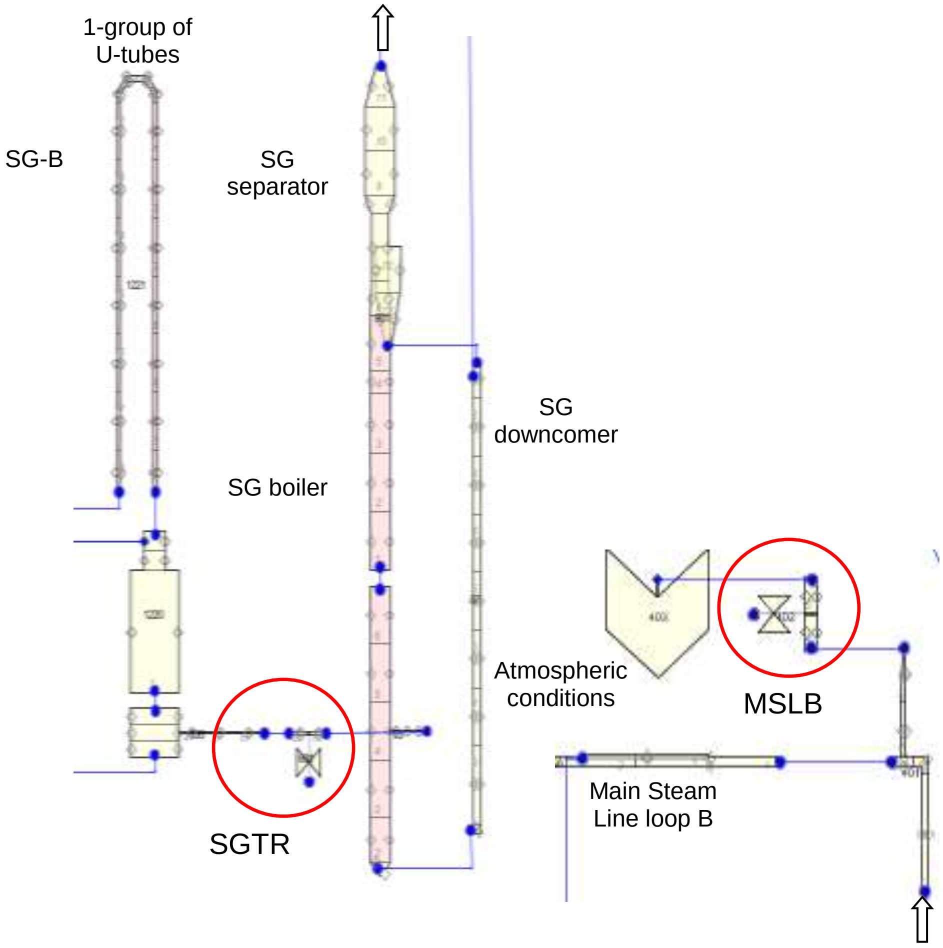
1-group of U-tubes
SG-B
SG separator
SG downcomer
SG boiler
Atmospheric conditions
MSLB
SGTR
Main Steam Line loop B
1221
1220
400
402
401

Figure 3 SGTR and MSLB Nodalization

5 RESULTS AND DISCUSSION

5.1 Steady-state

The steady-state conditions achieved in the simulation are in reasonable agreement with the experimental values. It can be seen in Table 2, where the relative errors (%) between experimental and simulated results for different items are listed. It is important to remark that in any case, the maximum difference between experiment and simulation is lower than 5%.

Table 2 Steady-State Condition Comparison between Experimental and Simulated Values

Item (Loop with PZR)	Relative Error (%)
Core Power.	0.00
Hot leg Fluid Temperature.	0.03
Cold leg Fluid Temperature.	0.07
Mass Flow Rate.	0.77
Pressurizer Pressure.	0.00
Pressurizer Liquid Level.	3.56
Accumulator System Pressure.	0.00
Accumulator System Temperature.	0.00
SG Secondary-side Pressure.	0.37
SG Secondary-side Liquid Level.	3.47
Steam Flow Rate.	2.14
Main Feedwater Flow Rate.	0.00
Main Feedwater Temperature.	0.00
Auxiliary Feedwater Temperature.	0.35

5.2 Transient

Table 3 lists the chronological sequence of events during the transient and the comparison between the experiment and TRACE5 results.

Table 3 Chronological Sequence of Events Comparison between Experiment and TRACE5

Event	Experiment Time (s)	TRACE5 Time (s)
Open of break valves for both MSLB and SGTR.	0	0
Both scram and SI signals.	49	48
Closure of SG MSIVs.	52	48
Initiation of coastdown of primary coolant pumps.	53	48
Termination of SG main feedwater.	58	48
Initiation of core power decay.	69	65
Initiation of AFW in broken loop.	About 80	47
Initiation of HPI system in both loops.	About 145	145
Primary coolant pumps stop.	300	300
Initiation of intact SG secondary-side depressurization (full opening RV, some minutes after SI signal).	About 1850	1847
Initiation of AFW in intact loop.	About 1860	1847
1st PORV open (stagnation of primary pressure at full of HPI flow rate capacity).	About 3050	3000
1st PORV closure.	About 3070	3010
Reduction from full to half of HPI flow rate capacity.	About 3130	2800
Termination of AFW in broken loop.	About 3200	2630
2nd PORV open (stagnation of primary pressure at half of HPI flow rate capacity).	About 5600	5500
2nd PORV closure.	About 5660	5510
Termination of HPI system in both loops.	About 6100	5500
Initiation of LPI system in both loops.	About 80	6026
Closure of break valves for both MSLB and SGTR.	3983	6500

In this section, some variables obtained with the LSTF model are compared with the experimental results. These variables include pressure in the primary and secondary sides, mass flow rate and inventory discharged through the break, primary mass flow, core power and collapsed-liquid levels in the pressure vessel, hot and cold legs, etc.

5.3 System Pressures

After opening the break valves that simulate the SGTR and MSLB, primary and secondary pressures start to decrease (Figure 4).

Due to the rapid depressurization of the secondary side of the broken steam generator, the secondary pressure decreases to 4.24 MPa in few seconds. However, in the intact steam generator (loop A), an increasing in the secondary pressure is produced due to the MSIVs closure. At this moment on, the relieve valve of the steam generator A actuates by cyclic openings to maintain the pressure almost constant. At 240 s the relief valve is definitively closed, being the slight loss of pressure, measured and simulated, due to the heat losses.

The HPI system is activated when the pressure vessel lower plenum pressure is lower than 12.27 MPa. Some minutes after the SI signal generation, the intact steam generator is depressurized by full opening the relief valve. During the time interval defined between 0 and the beginning of the secondary depressurization, at 1800 s approximately, the pressures calculated by TRACE5 are in good agreement with the experimental ones. Furthermore, during this interval the heat extraction in the intact steam generator is effective due to the primary pressure is higher than the secondary one. After the intact steam generator depressurization, the primary pressure stagnates at about 3000 s in the experiment. However, this stagnation is not well reproduced with TRACE5.

When the primary pressure stagnates, the pressurizer PORV is opened. TRACE5 reproduces a drastic drop of pressure in comparison to the experiment. This abrupt drop can be attributed to different upper head fluid conditions achieved during the steady state simulation. The hydraulic paths to the upper head such as Control Rod Guide Tubes (CRGTs) have been tested, varying friction coefficients and flow areas. However, relevant changes have not been found.
After the change to half of HPI flow rate, the primary pressure reaches another stagnation at 5000 s. Few seconds later, the PORV is opened again. In this case, the simulated pressure drop is more like the experimental one.

In the last part of the transient, the LPI system actuates. After the LPI system actuation and the core power-off, Test 5 terminates.

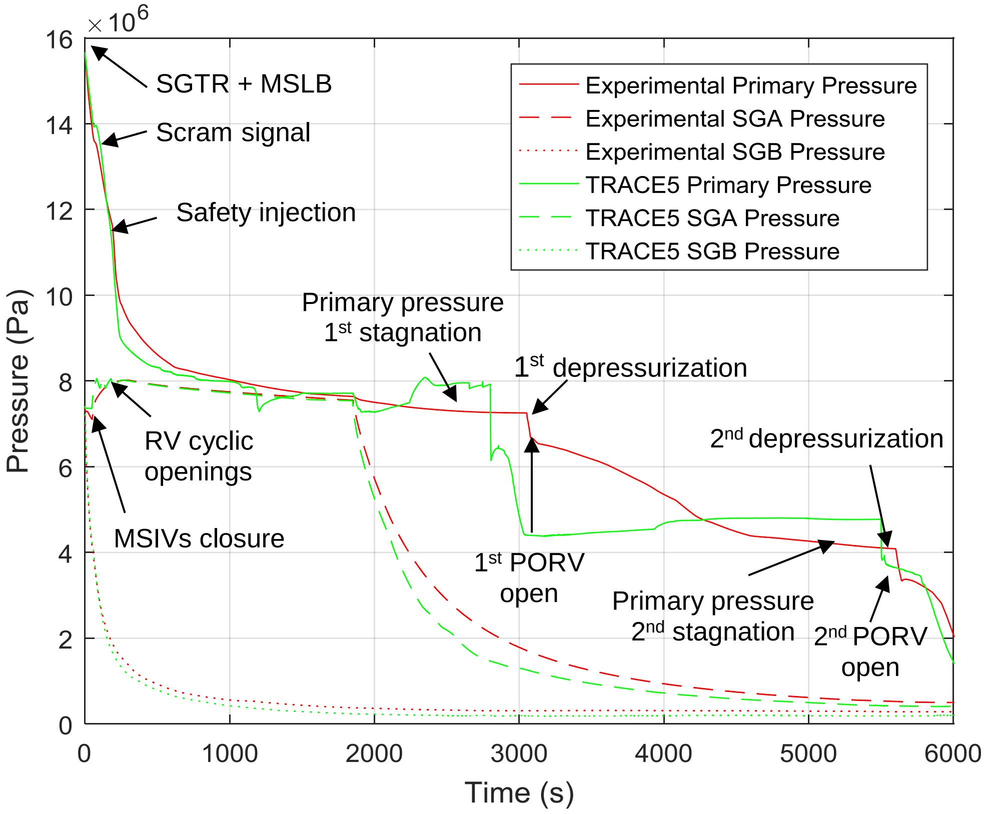

Figure 4 Primary and Secondary Pressures

5.4 Break

Figure 5 shows the mass flow rate through the steam line in the broken loop. TRACE5 reproduces its rapid fall successfully. The SGTR and the MSLB produce an asymmetrical coolant distribution in the primary system. This fact is evident when the primary mass flow rates in both loops are represented (Figure 7).

The SGTR mass flow rate is shown in Figure 6. As it can be seen, the simulated SGTR mass flow rate is slightly lower in comparison to the experiment. After the first depressurization, the SGTR mass flow rate is steeply decreased due to the abrupt primary pressure drop. When the HPI mass flow rate (Figure 19) is reduced to the 50%, a pseudo-equilibrium between SGTR and HPI mass flow rates is reached in both experimental and simulation cases. This equilibrium produces a second stagnation of the primary pressure at 5000 s (Figure 4).

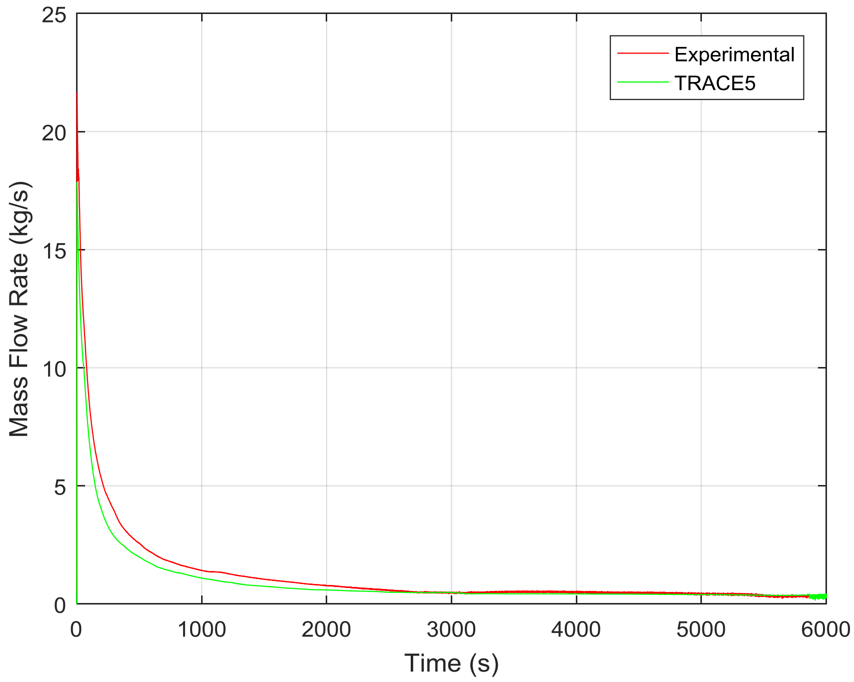

Figure 5 MSLB Mass Flow Rate

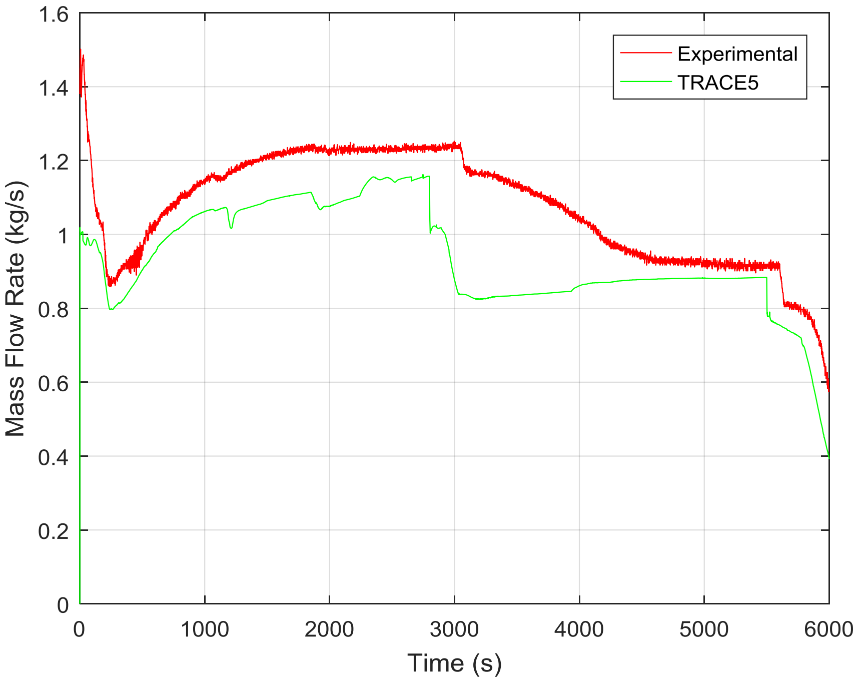

Figure 6 SGTR Mass Flow Rate

5.5 Primary Loop Mass Flows

Figure 7 shows the primary mass flow rate in both loops. As it has been said, the SGTR and MSLB produce an asymmetrical coolant distribution in the primary system. In the transient, the natural circulation is early finished in the intact loop (at 600 s). Nevertheless, in the broken loop B the situation is quite different, the natural circulation is observed during the whole transient due to the SGTR + MSLB. The mass flow rate in loop B is strongly decreased according to the drop in the secondary pressure. During the pressure stagnation, the natural circulation mass flow is almost constant.

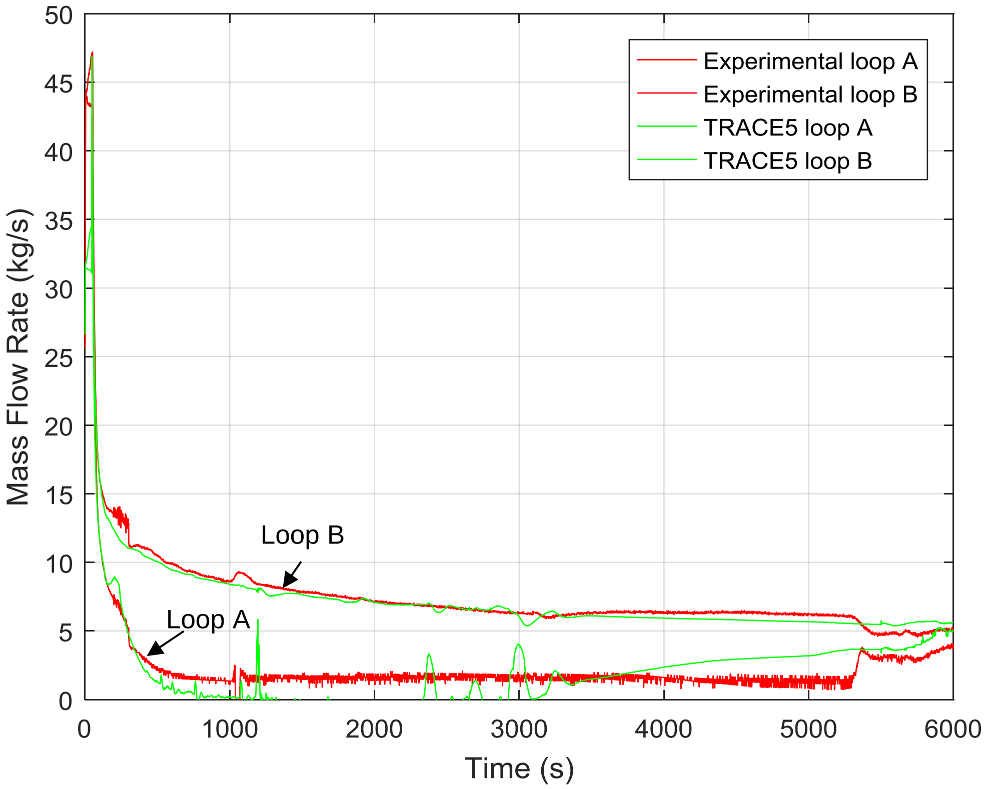

Figure 7 Primary Loop Mass Flow Rate

5.6 Vessel Collapsed Liquid Levels

Figures 8, 9 and 10 show the pressure vessel collapsed liquid levels. The core collapsed liquid level obtained with TRACE5 is shown in Figure 8. As it can be seen, the core is full during the transient. The upper plenum collapsed liquid level is shown in Figure 9. The simulated liquid level in the upper plenum is higher than the experimental values. As it can be predicted, at the end of the transient, steam starts to appear in the upper plenum in the experiment and in the simulation results. Figure 10 shows the simulated downcomer collapsed liquid level. The downcomer is full of liquid during the entire transient. The experimental collapsed liquid level in the downcomer is not available, as happens with the core.

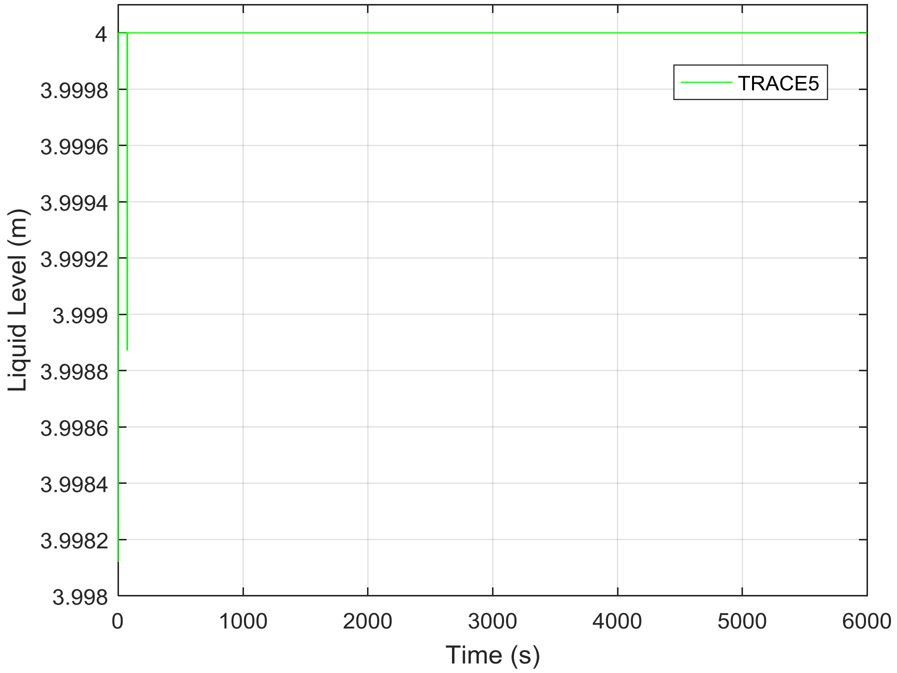

Figure 8 Core Collapsed Liquid Level

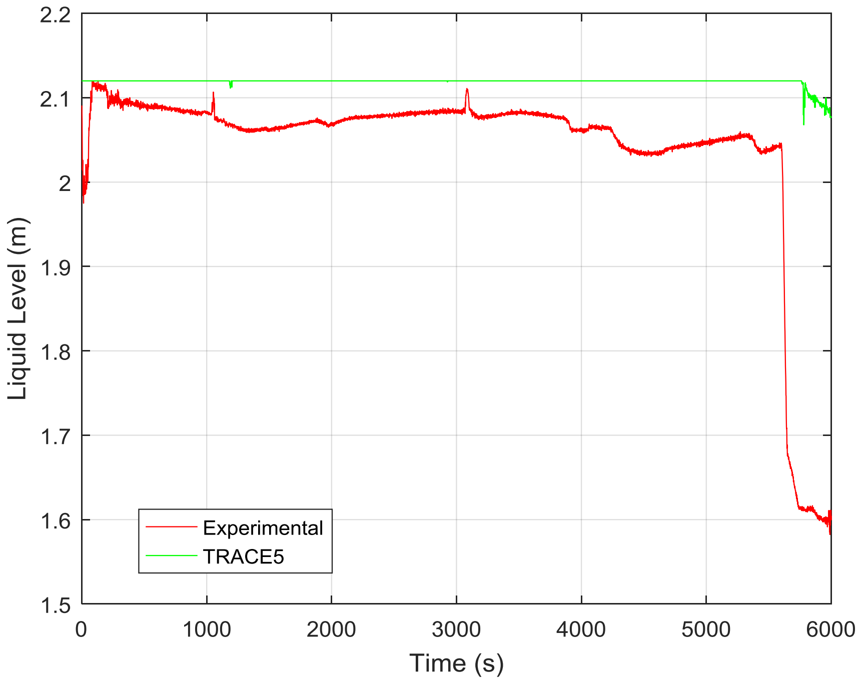

Figure 9 Upper Plenum Collapsed Liquid Level

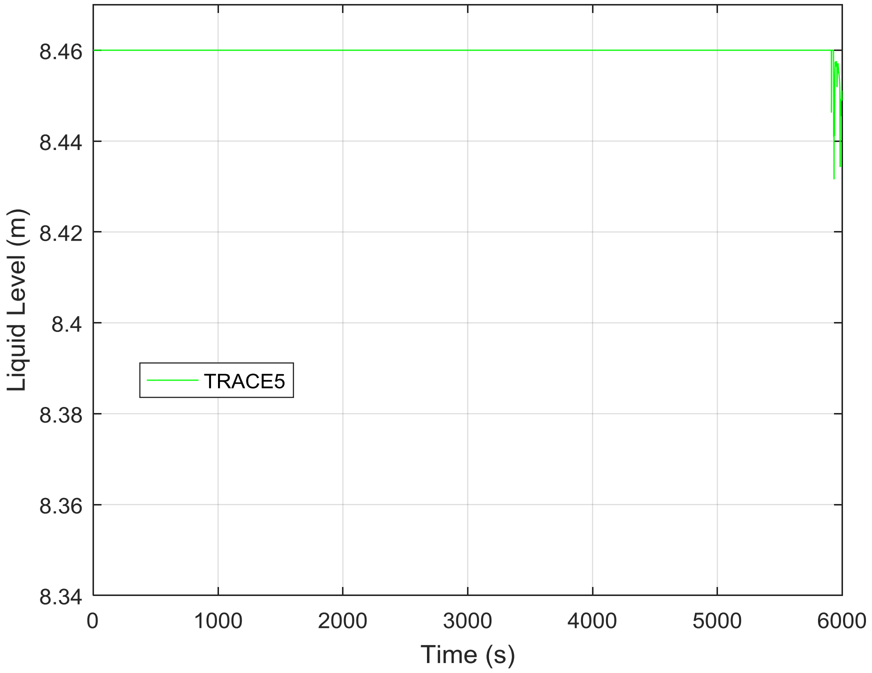

Figure 10 Downcomer Collapsed Liquid Level

5.7 Fluid Temperatures

Temperature in hot and cold legs of the broken loop are shown in Figures 11 and 12, respectively. As it can be seen, at the end of the transient the hot and cold leg fluid temperatures are about 430 K and 400 K, respectively. In Figures 13 and 14 show the downcomer and lower plenum fluid temperatures. The fluid temperature at the end of the transient is about 410 K in the downcomer and 400 K in the lower plenum.

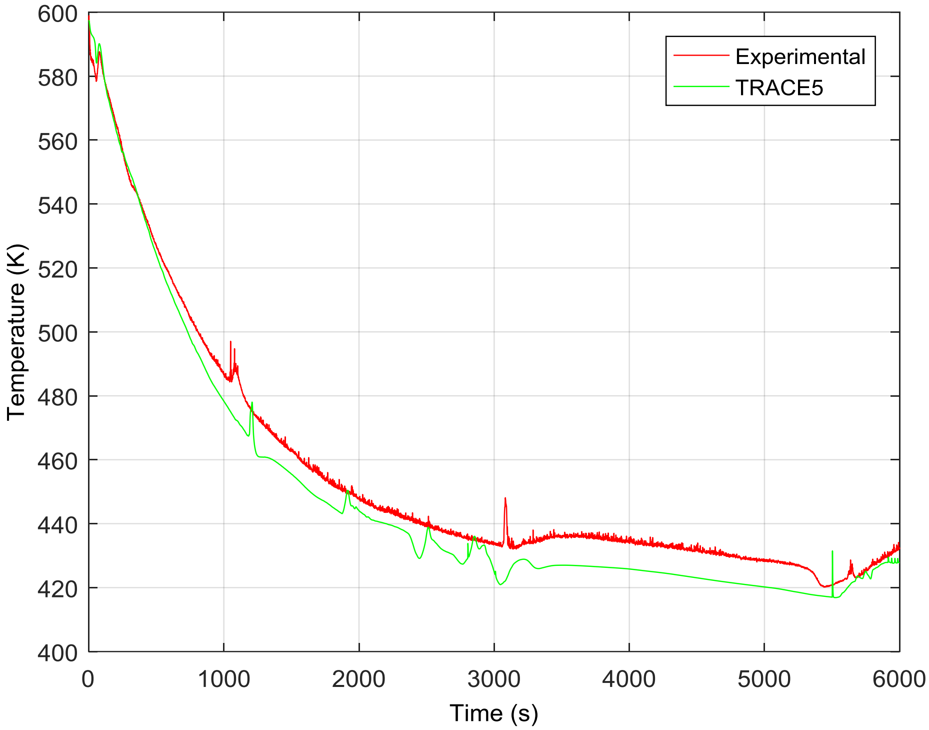

Figure 11 Hot Leg B Fluid Temperature

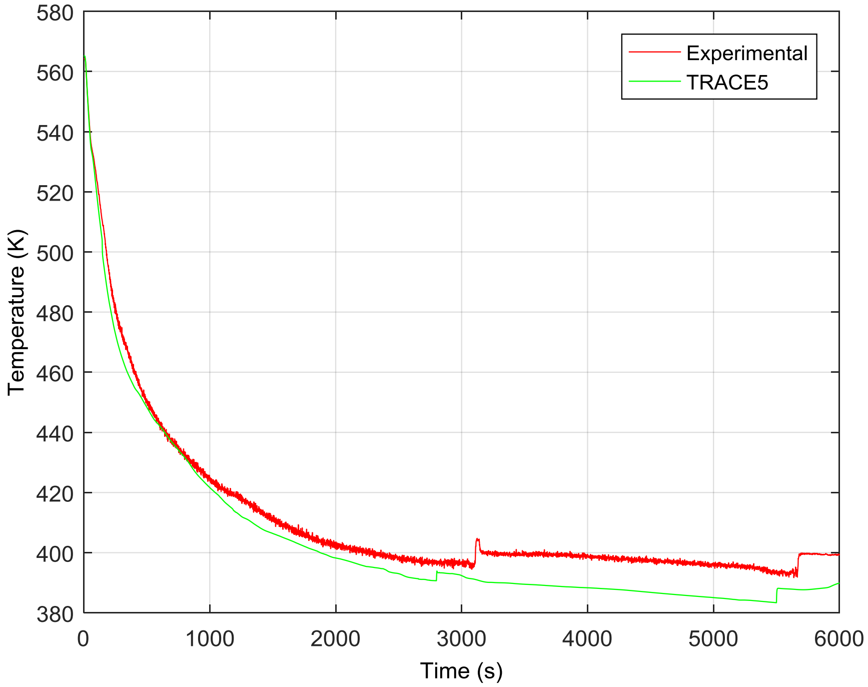

Figure 12 Cold Leg B Fluid Temperature

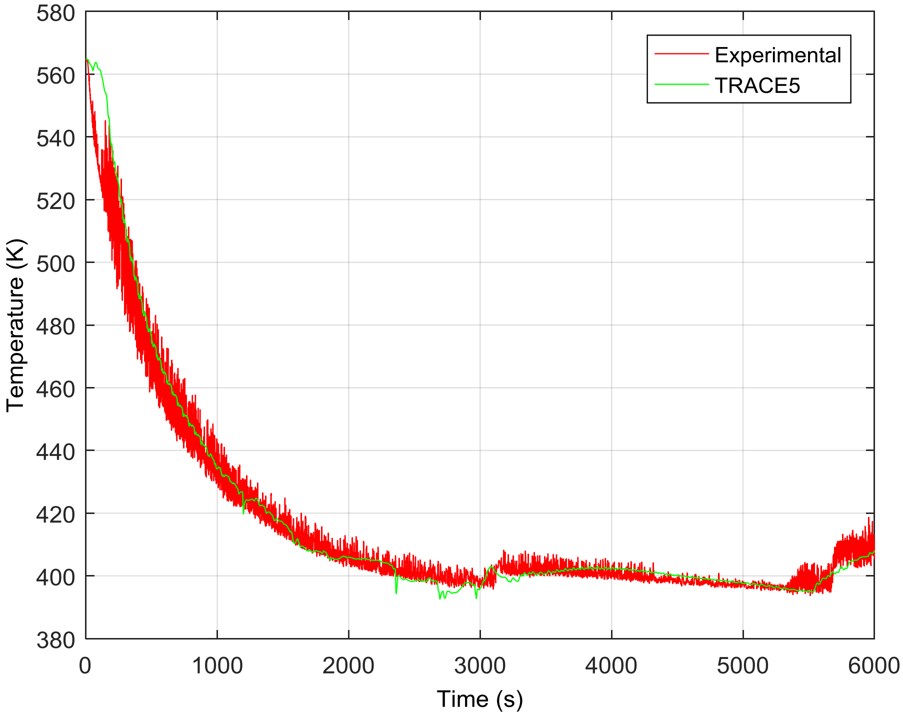

Figure 13 Downcomer Fluid Temperature

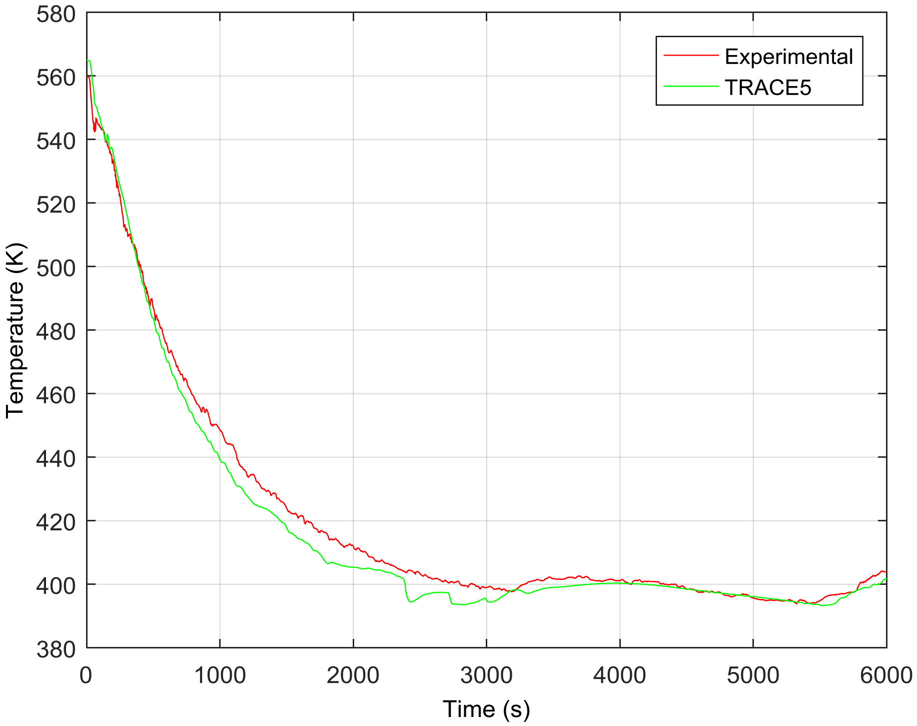

Figure 14 Lower Plenum Fluid Temperature

5.8 Hot and Cold Legs Liquid Levels

Figures 15, 16, 17 and 18 show the collapsed liquid levels in hot and cold legs of both loops, respectively. They are completely full of liquid during the transient.

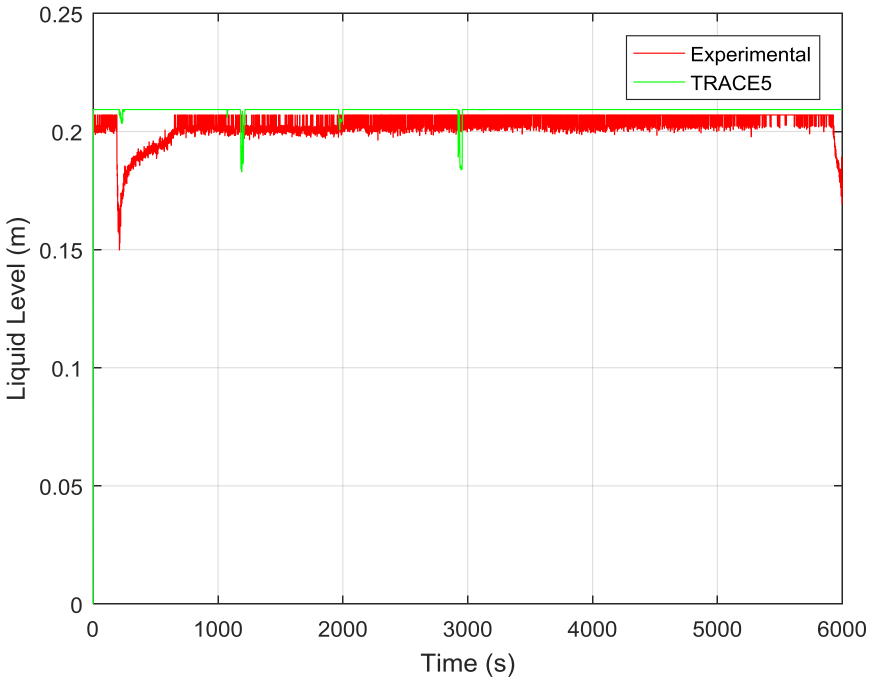

Figure 15 Collapsed Liquid Level in the Hot Leg A

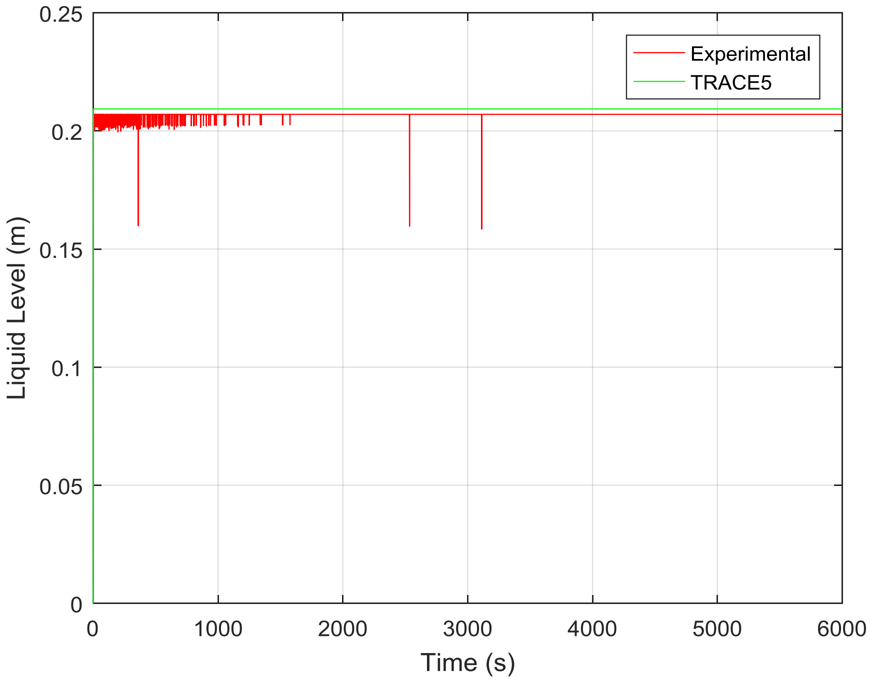

Figure 16 Collapsed Liquid Level in the Hot Leg B

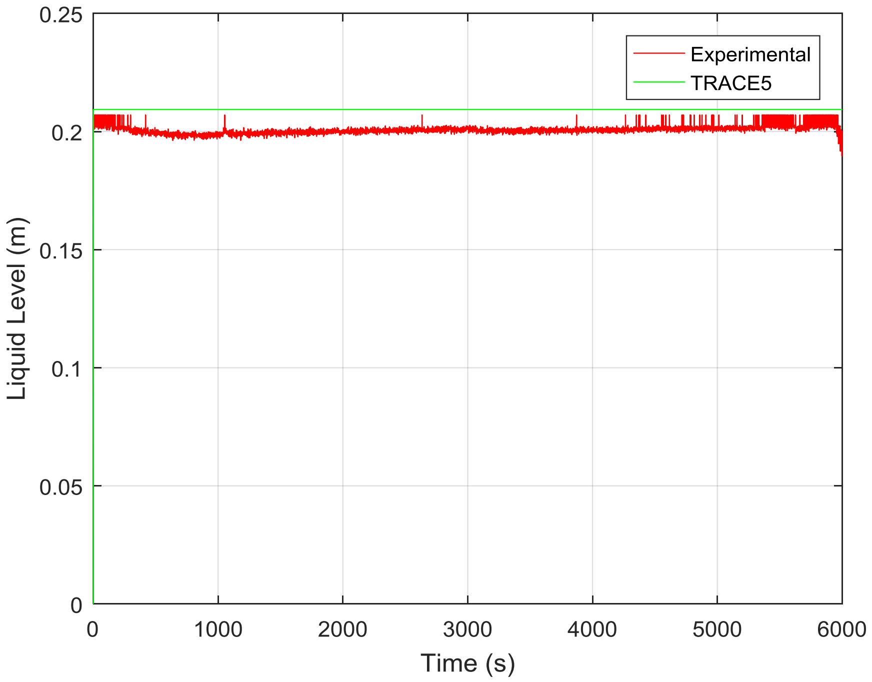

Figure 17 Collapsed Liquid Level in the Cold Leg A

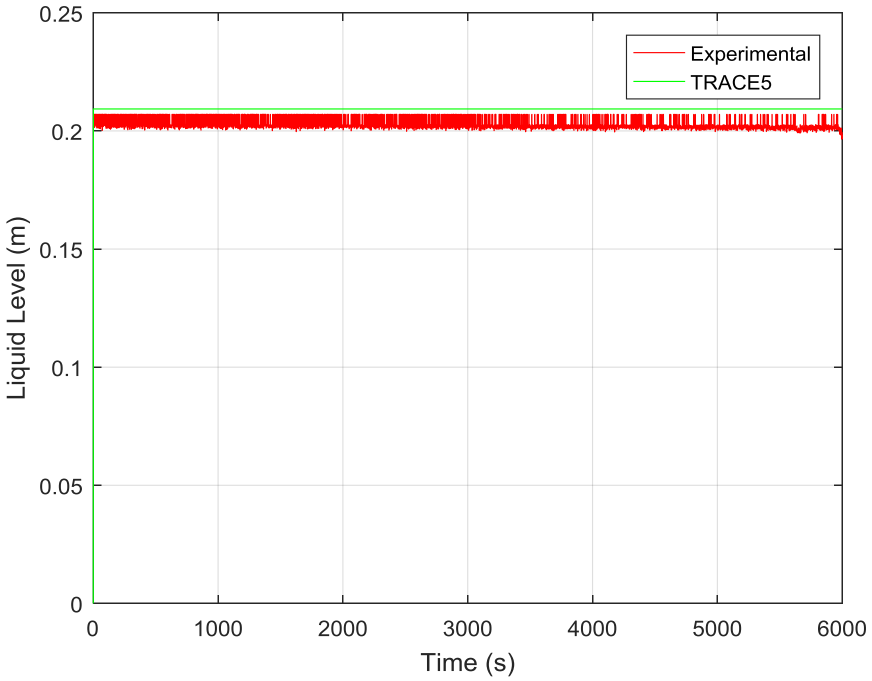

Figure 18 Collapsed Liquid Level in the Cold Leg B

5.9 Emergency Core Cooling Systems Mass Flow Rates

The high pressure injection mass flow rate is shown in Figure 19. It is activated when the vessel lower plenum pressure is lower than 12.27 MPa, according to a pre-determined mass flow rate curve. It is a function of the primary pressure, which depends on the drop of pressure and the loss of coolant produced by the SGTR

When the HPI mass flow rate is reduced to the 50%, a pseudo-equilibrium between SGTR (Figure 4). This equilibrium produces a second stagnation of the primary pressure at 5000 s. Figure 20 shows the LPI coolant flow rate. When the primary pressure is lower than 1.24 MPa, the LPI system is activated in both loops.

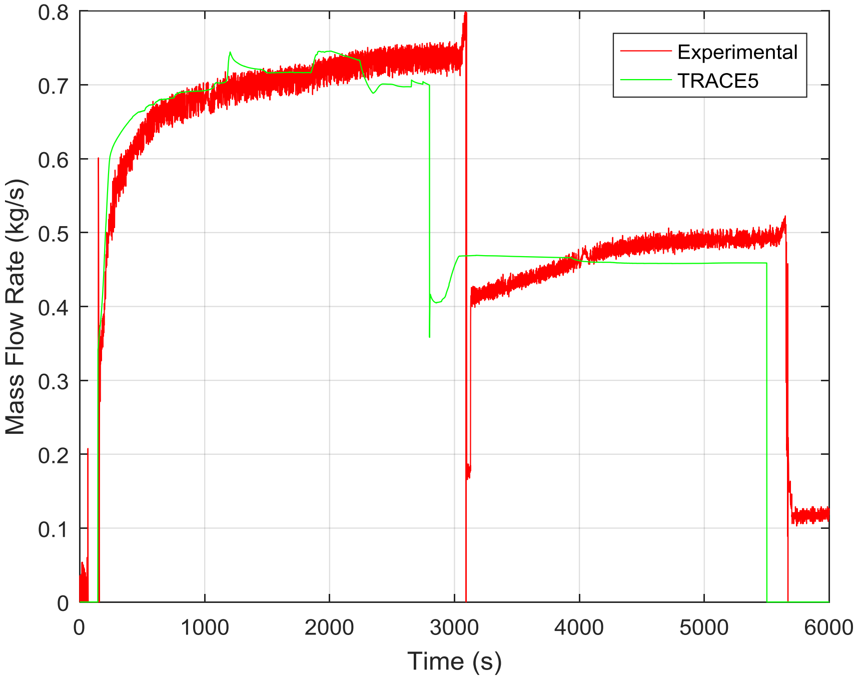

Figure 19 High Pressure Injection System Mass Flow Rate

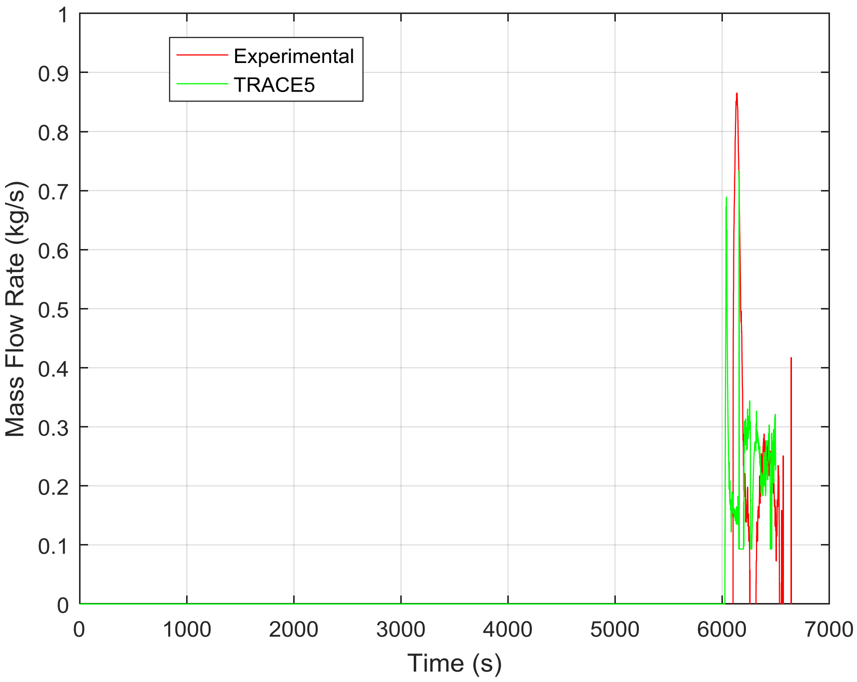

Figure 20 Low Pressure Injection System Mass Flow Rate

5.10 Collapsed Liquid Levels

The sudden drop of the primary pressure after the MSLB, concurrent with the SGTR, causes a rapid emptying of the pressurizer. At 3000 s and 5600 s, the PORV is open, producing a fast increasing of the pressurizer collapsed liquid level (Figure 21). This fast increasing of the collapsed liquid level is not reproduced in the simulation.

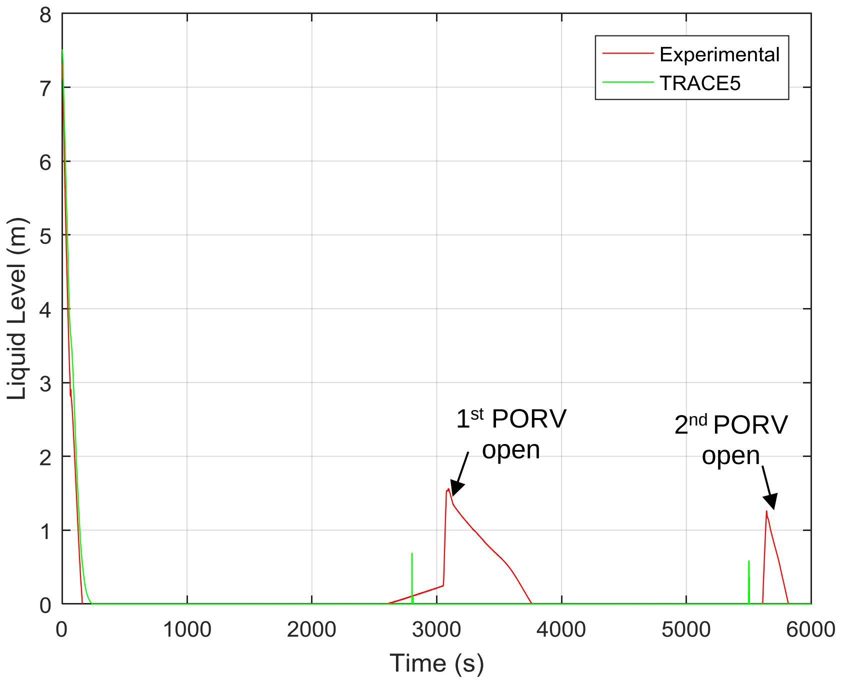

Figure 21 Collapsed Liquid Level in Pressurizer

The secondary–side collapsed liquid levels of both steam generators are shown in Figure 22. The loss of liquid in broken SG-B is evident during the first part of the transient. Intact SG secondary liquid level decreases after depressurization. Both liquid levels are recovered due to the AFW action. In the broken SG, AFW is activated with the SI signal until it reaches the initial liquid level. In intact SG, the AFW is activated with the initiation of the intact SG secondary-side depressurization at 1800 s.

The U-tubes collapsed liquid level of steam generator B is shown in Figure 23. At 1200 s a sudden evaporation is predicted by TRACE5, reducing more than 20% the liquid level in the U-tubes. At the same time, the drop of pressure causes that the primary pressure is lower than the secondary one avoiding the heat transfer to the steam generator momentarily.

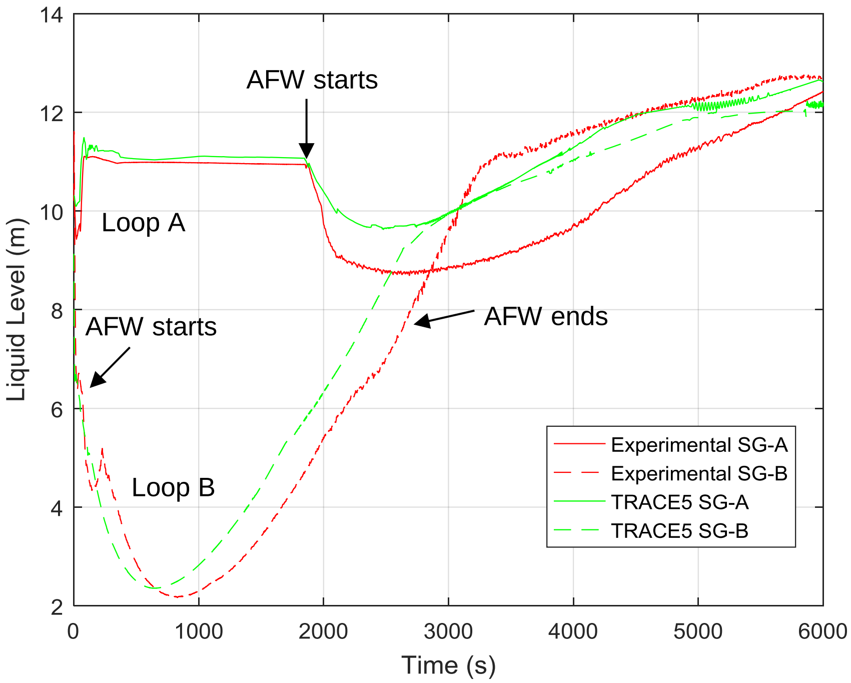

Figure 22 Collapsed Liquid Levels in Both Steam Generators

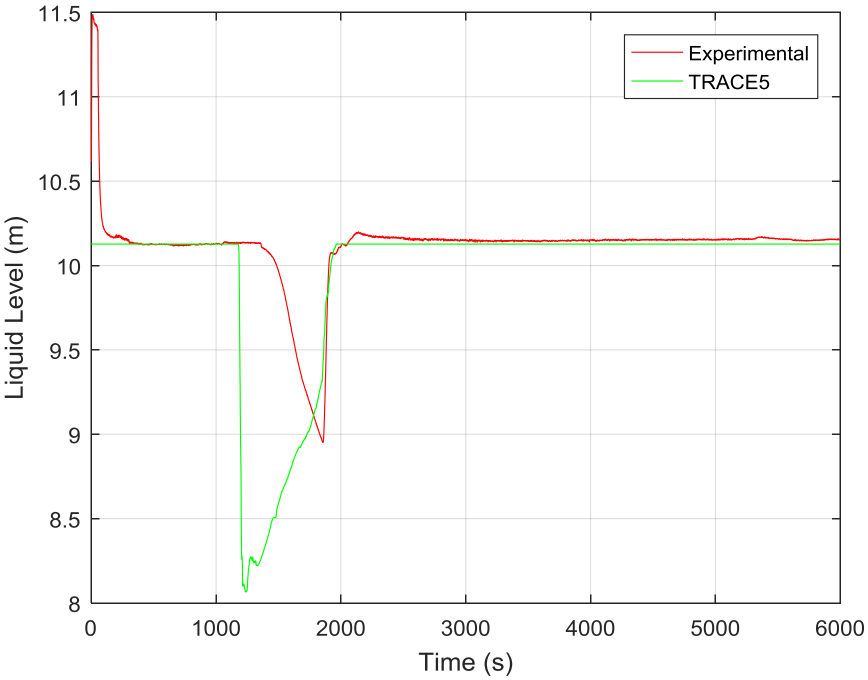

Figure 23 Collapsed Liquid Levels in U-tubes of SG-B

5.11 Relief Valve Mass Flow Rate of Intact Loop

The Relief Valve mass flow rate of the intact steam generator is shown in Figure 24. At the beginning of the transient, the relief valve actuates according to cyclic openings to maintain the secondary pressure almost constant. After that, the relief valve is closed till 1800 s when the depressurization starts.

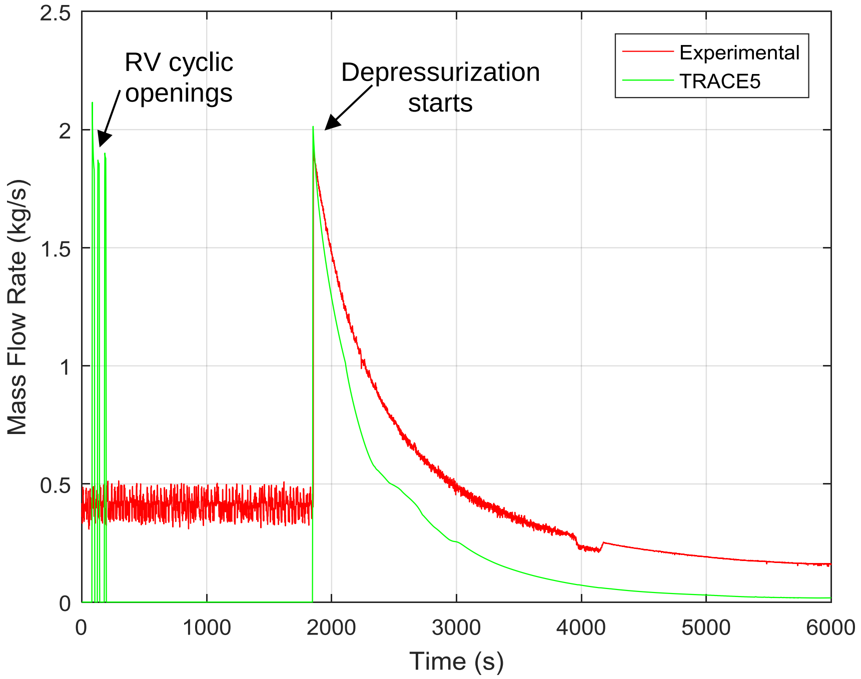

Figure 24 Relief Valve SG-A Mass Flow Rate

6 CONCLUSIONS

The objective of this report is to describe the most relevant results achieved by using the thermal-hydraulic code TRACE5 in the simulation of the OECD/NEA ROSA-2 Project Test 5 (SB-SG-14 in the Japan Atomic Energy Agency). This test reproduces a PWR Steam Generator Tube Rupture (SGTR) induced by a Main Steam Line Break (MSLB).

The simulation results show that TRACE5 can reproduce qualitatively the main thermal-hydraulic phenomena that occur during the transient. However, some important discrepancies between experimental data and the simulation results are found during the first depressurization. The primary pressure stagnation is not well simulated. Furthermore, an abrupt drop in the primary pressure is reproduced after the PORV opening in comparison to the experiment.

These discrepancies could be attributed to the different fluid conditions achieved in the upper head of the pressure vessel during the steady state simulation.

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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The purpose of this work is to overview the results obtained by the simulation, using the thermal-hydraulic code TRACE5, of Test 5 (SB-SG-14) in the frame of the OECD/NEA ROSA-2 Project. This test, conducted at the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA), simulates the thermal hydraulic responses after a PWR Steam Generator Tube Rupture (SGTR) induced by a Main Steam Line Break (MSLB). The result of these simultaneous breaks is a depressurization in both primary and secondary systems because they are connected through the SGTR. The actuation of the Accumulator Injection System was suppressed to keep primary coolant discharge to the Steam Generator secondary-side as low as possible.

The STGR is considered one of the main accidents in nuclear safety due to steam generator reliability and performance are serious concerns in the PWR operation. Through several studies, it has been reported that the severe accident management procedures such as foresee flooding and the primary system depressurization are used to minimize the release from the affected steam generator. These actions may significantly reduce the source term in SGTR accidents.

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Consejo de Seguridad Nuclear (Spanish Nuclear Regulatory Commission, CSN)
Large Scale Test Facility (LSTF)
High Pressure Charging Pump (PJ)
High Pressure Injection Pump (PL)
Intermediate Break Loss-of-Coolant-Accident (IBLOCA)
Accumulator Injection System (AIS)

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