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International Agreement Report

Assessment of TRACE 5.0 Against ROSA Test 3-2, High Power Natural Circulation

Prepared by:
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ABSTRACT

The purpose of this work is to provide an overview of the results obtained in the simulation of a pressurized water reactor (PWR) high-power natural circulation due to failure of scram following a Loss-Of-Feed Water (LOFW) transient under the assumption of total failure of High Pressure Injection System (HPIS), but actuation of Auxiliary Feed Water (AFW).

Simulation of the experiment conducted in the Large Scale Test Facility (LSTF) is performed via the thermal-hydraulic code TRACE5. This work is developed in the frame of OECD/NEA ROSA Project Test 3-2 (TR-LF-13 in JAEA). AFW was actuated when the steam generator (SG) secondary-side collapsed liquid level decreased to a determined value, providing a continuous primary-to-secondary heat removal. The primary and secondary pressures were maintained almost constant by cycle opening of pressurizer Power-Operated Relief Valve (PORV) and SG relief valves till the end of the test.

A detailed model has been developed with TRACE5 following these assumptions. Results of the simulation are compared with the experimental in several graphs, observing an acceptable general behaviour in the entire transient. In conclusion, this work represents a good contribution for assessment of the predictability of thermalhydraulic computer codes such as TRACE5.

FOREWORD

Extensive knowledge and techniques have been produced and made available in the field of thermal-hydraulic responses during reactor transients and accidents, and major system computer codes have achieved a high degree of maturity through extensive qualification, assessment and validation processes. Best-estimate analysis methods are increasingly used in licensing, replacing the traditional conservative approaches. Such methods include an assessment of the uncertainty of their results that must be taken into account when the safety acceptance criteria for the licensing analysis are verified.

Traditional agreements between the Nuclear Regulatory Commission of the United States of America (USNRC) and the Consejo de Seguridad Nuclear of Spain (CSN) in the area of nuclear safety research have given access to CSN to the NRC-developed best estimate thermalhydraulic codes RELAP5, TRAC-P, TRAC-B, and currently TRACE. These complex tools, suitable state-of-the-art application of current two-phase flow fluid mechanics techniques to light water nuclear power plants, allow a realistic representation and simulation of thermalhydraulic phenomena at normal and incidental operation of NPP. Owe to the huge required resources, qualification of these codes have been performed through international cooperation programs. USNRC CAMP program (Code Applications and Maintenance Program) represents the international framework for verification and validation of NRC TH codes, allowing to:

- Share experience on code errors and inadequacies, cooperating in resolution of deficiencies and maintaining a single, internationally recognized code version.
- Share user experience on code scaling, applicability, and uncertainty studies.
- Share a well documented code assessment data base.
- Share experience on full scale power plant safety-related analyses performed with codes (analyses of operating reactors, advanced light water reactors, transients, risk-dominant sequences, and accident management and operator procedures-related studies).
- Maintain and improve user expertise and guidelines for code applications.

Since 1984, when the first LOFT agreement was settled down, CSN has been promoting coordinated joint efforts with Spanish organizations, such as UNESA (the association of Spanish electric energy industry) as well as universities and engineering companies, in the aim of assimilating, applying, improving and helping the international community in the validation of these TH simulation codes[1], within different periods of the associated national programs (e.g., CAMP-España). As a result of these actions, there is currently in Spain a good collection of productive plant models as well as a good selection of national experts in the application of TH simulation tools, with adequate TH knowledge and suitable experience on their use.

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 continued 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[2] 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 2001 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 safety community during the coming decade.

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

1 It's worth to note the emphasis made in the application to actual NPP incidents.
2 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).

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), or enlarging code validation and qualification data bases incorporating new information (e.g., multi-dimensional aspects, non-condensable gas effects). In particular, CSN is currently participating in the PKL and ROSA programmes.

The PKL is an important integral test facility operated by of AREVA-NP in Erlangen (Germany), and designed to investigate thermal-hydraulic response of a four-loop Siemens designed PWR. Experiments performed during the PKL/OECD program have been focused on the issues:

- Boron dilution events after small-break loss of coolant accidents.
- Loss of residual heat removal during mid-loop operation (both with closed and open reactor coolant system.

ROSA/LSTF of Japan Atomic Energy Research Institute (JAERI) is an integral test facility designed to simulate a 1100 MWe four-loop Westinghouse-type PWR, by two loops at full-height and 1/48 volumetric scaling to better simulate thermal-hydraulic responses in large-scale components. The ROSA/OECD project has investigated issues in thermal-hydraulics analyses relevant to water reactor safety, focusing on the verification of models and simulation methods for complex phenomena that can occur during reactor transients and accidents such as:

- Temperature stratification and coolant mixing during ECCS coolant injection
- Water hammer-like phenomena
- ATWS
- Natural circulation with super-heated steam
- Primary cooling through SG depressurization
- Pressure vessel upper-head and bottom break LOCA

This overall CSN involvement in different international TH programmes has outlined the scope of the new period of CAMP-España activities focused on:

- Analysis, simulation and investigation of specific safety aspects of PKL/OECD and ROSA/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 experiments and plant application with the last available versions of NRC TH codes (RELAP5 and TRACE). A CAMP in-kind contribution is aimed as end result of both types of studies.

Development of these activities, technically and financially supported by CSN, is being carried out by 5 different national research groups (Technical Universities of Madrid, Valencia and Cataluña). On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermal hydraulics analysis of accidents of the Spanish nuclear power plants.

Francisco Fernández Moreno, Commissioner
Consejo de Seguridad Nuclear (CSN)

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EXECUTIVE SUMMARY

The purpose of this work is to provide an overview of the results obtained in the simulation of a Pressurized Water Reactor (PWR) high-power natural circulation due to failure of scram during a Loss-Of-Feed Water (LOFW) transient under the assumption of total failure of High Pressure Injection System (HPIS), but with actuation of Auxiliary Feed Water (AFW), in the Large Scale Test Facility (LSTF) via the thermal-hydraulic code TRACE5. The work is developed in the frame of OECD/NEA ROSA Project Test 3-2 (TR-LF-13 in JAEA).

The experiment was initiated by termination of the Main Feed Water (MFW) and opening the PORV valve at time zero. The scram signal was generated few seconds after. Along with the scram, a turbine trip is produced by closing the Steam Generators Main Steam Isolation Valves (MSIVs). The Safety Injection (SI) signal is generated when the secondary-side collapsed liquid level decreased to a determined value (0.5 m approximately). From this moment on, the Relief Valves (RV) of both SGs, begin opening and closing in order to maintain the pressure almost constant. AFW was actuated when the Steam Generator (SG) secondary-side collapsed liquid level decreased to a determined value, providing a continuous primary-to-secondary heat removal. The primary and secondary pressures were maintained almost constant by cycle opening of pressurizer Power-Operated Relief Valve (PORV) and SG relief valves till the end of the test.

A detailed model has been developed with TRACE5 (RELEASE CANDIDATE 3) following these assumptions. Results of the simulation are compared with the experimental in several graphs, observing an acceptable general behaviour in the entire transient.

ACKNOWLEDGEMENTS

This paper contains findings that were produced within the OECD-NEA ROSA 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

AFW	auxiliary feedwater
AM	accident management
ATWS	anticipated transient without scram
CCFL	counter-current flow limiting
CRGT	control rod guide tubes
DP	differential pressure
ECCS	emergency core cooling system
HPI	high pressure injection
JAEA	Japan Atomic Energy Agency
LOFT	loss of fluid test
LOFW	loss-of-feedwater
LSTF	large scale test facility
MFW	main feed water
MSIV	main steam isolation valve
RV	relief valve
PCT	peak cladding temperature
PORV	power-operated relief valve
PV	pressure vessel
PWR	pressurized water reactor
PZR	pressurizer
SBLOCA	small break loss-of-coolant accident
SG	steam generator
ST	storage tank
SV	safety valve

1 INTRODUCTION

There is an ongoing interest in the research and development of codes and methodologies for "best-estimate" analysis of Anticipated Transients Without Scram (ATWS) even though not much has been reported on these type of transients with high-power natural circulation, due to the difficulties found in simulating these events [1 - 9].

During a Loss-Of-Feed Water (LOFW) accident in a Pressurized Water Reactor (PWR) with scram, high-power natural circulation occurs in general when Auxiliary Feed Water (AFW) actuates providing a heat sink. Such phenomena include natural circulation with high core power with liquid entrainment in the hot leg at the surge line inlet nozzle and top of pressurizer (PZR), and Counter-Current Flow Limiting (CCFL) at the pressurizer bottom that may hold a large amount of coolant in the pressurizer.

In the transient following LOFW, PZR Power-Operated Relief Valve (PORV) may keep continuous cycle, resulting in continuous loss of primary coolant inventory and beginning natural circulation in a very early stage of the transient.

In this frame, an important phenomenon is investigated when the core boiling starts and the U tubes liquid level significantly oscillates. The main goal of the experiment is to study whether the core is properly cooled before the pressures reach nearly-equilibrium condition, after the initiation of the automatic power decrease procedure. Also, thermal-hydraulic data related to high-power natural circulation is studied for the validation of computer codes and models for integral system analyses.

The present work describes the main results achieved by the authors using the thermal-hydraulic code TRACE5 [10, 11], in the frame of OECD/NEA ROSA Project Test 3-2 (TR-LF-13 in JAEA) [12] with the purpose of testing the behaviour of the code in this case. A post-test analysis was performed with the main objective of assessing the code's capability in predicting the high-power natural circulation phenomena.

The experiment 3-2 of the OECD/NEA ROSA project was managed during 8th and 9th of November, 2007 in the Large Scale Test Facility (LSTF) of the Japanese Atomic Energy Agency (JAEA) [13]. The LSTF simulates a PWR reactor, Westinghouse type, of four loops and 3423 MW of thermal power, scaled to 1/48 in volume and two loops. The experiment simulates a LOFW-induced ATWS with high-power natural circulation under the assumption of total failure of HPIS but actuation of AFW.

2 ROSA FACILITY DESCRIPTION

This section consists in a sketched description of the LSTF facility (in the Tokai Research Establishment of the JAERI) [13]. Two loops compose the primary coolant system: the primary loop A with the pressurizer (PZR) and the symmetrical primary loop B. Both include a primary coolant pump (PC) and a steam generator (SG). On the other hand, the secondary-coolant system includes a jet condenser (JC), a feed water pump (PF), the auxiliary feed water pumps (PA) and two SG secondary systems with a related piping system.

The Emergency Core Cooling System (ECCS) consists of the following sub-systems: the high pressure charging pump (PJ), the high pressure injection pump (PL), the residual heat removal (RHR) system and the primary gravity injection tank (PGIT). A break flow Storage Tank (ST) stores the discharged coolant from the primary system.

The pressure vessel (PV) is composed of the following elements: The upper head located above the upper core support plate; the upper plenum situated between the upper core support plate and the upper core plate; the core; the lower plenum and the downcomer annulus region which surrounds the core and upper plenum. LSTF vessel is structured with 8 spray nozzles (of 3.4 mm inner-diameter) at the Upper Head, and 8 Control Rod Guide Tubes (CRGTs) which lead the flow path between the Upper Head and the Upper Plenum.

Each Steam Generator (SG) contains 141 U-tubes grouped depending on their length (an average length of 19.7 m can be considered, with a maximum height of 10.62 m and a minimum height of 9.156 m). All the U-tubes are characterized with an inner diameter of 19.6 mm and an outer diameter of 25.4 mm (2.9 mm of wall thickness). The total inner and outer surface areas are therefore 171 and 222 m2, respectively. Regarding to the vessel, plenum and riser of steam generators, the inner heights are 19.840, 1.183 and 17.827 m, respectively. The downcomer is 14.101 m.

3 TRANSIENT DESCRIPTION

The control logic of the transient is listed in Table 1. The break unit is emulated by the PORV, which effectuates a cycle opening. After the break started, the primary coolant is discharged through the break and accumulated in the Storage Tank (ST).

The experiment was initiated by termination of the Main Feed Water (MFW) and opening the PORV valve at time zero. The scram signal was generated few seconds after. Along with the scram, a turbine trip is produced by closing the Steam Generators Main Steam Isolation Valves (MSIVs). Simultaneously, rotational speed of primary coolant pumps was increased up to 1500 rpm. The Safety Injection (SI) signal is generated when the secondary-side collapsed liquid level decreased to a determined value (0.5 m approximately). From this moment on, the Relief Valves (RV) of both SGs, begin opening and closing in order to maintain the pressure almost constant.

Table 1 Control logic and sequence of major events in the experiment.

Event	Condition
Termination of main feedwater	Time zero.
Generation of scram signal	Few seconds after.
Pressurizer heaters off.	Generation of scram signal or PZR liquid level below 2.3 m.
Initiation of the core power decay curve.	Generation of scram signal.
Initiation of primary pumps stopping curve.	Generation of scram signal.
Turbine signal (turbine trip)	Generation of scram signal.
Main Steam Isolation Valve (MSIV) closure.	Generation of scram signal.
Initiation of auxiliary feed water	Generation of safety injection SI signal.

The scram signal produces the initiation of the core power decay curve, calculated by considering the stored heat in fuel rods and delayed neutron fission power, as it can be seen in Table 2. The initial core power corresponds to 14% of the nominal power of a PWR volumetrically scaled (1/48). When liquid level pressurizer is lower than 2.3 m, heaters (backup and proportional) are turned off.

Table 2 Predetermined core power decay curve (normalized to the Steady State power).

Norm. Time	Norm. Power	Norm. Time	Norm. Power	Norm. Time	Norm. Power	Norm. Time	Norm. Power
0.0000	1.000	0.0041	0.498	0.0097	0.451	0.0212	0.530
0.0034	1.000	0.0045	0.399	0.0110	0.500	0.0345	0.525
0.0035	0.919	0.0057	0.300	0.0144	0.551	0.0390	0.525
0.0036	0.808	0.0072	0.300	0.0158	0.570	1.0000	0.525
0.0038	0.701	0.0078	0.350	0.0177	0.570		
0.0039	0.615	0.0088	0.401	0.0188	0.550		

At the same time, the primary coolant pump coastdown is initiated, also using a pre-determined rotational speed curve (Table 3). Pumps are completely stopped some minutes after scram signal generation.

Table 3 Pumps relative rotational speed (normalized values).

Normalized Time	Relative rotational	Normalized Time	Relative rotational	Normalized Time	Relative rotational
0.00000	1.000	0.00060	0.280	0.00160	0.125
0.00004	0.850	0.00080	0.220	0.00180	0.110
0.00010	0.730	0.00100	0.185	0.00200	0.100
0.00020	0.540	0.00120	0.160	0.00500	0.000
0.00040	0.370	0.00140	0.140	--	---

Primary and secondary pressures are maintained almost constant by cycle opening of PORV and RVs till the end of the test.

The core power is automatically decreased by the core protection system when the maximum fuel rod surface temperature excess 873 K, as it can be seen in Table 4.

Table 4 Core protection system logic.

Control of core power to	Maximum fuel rod surface temperature (K)
75%	873
50%	893
25%	903
10%	913
0%	923

4 ROSA FACILITY MODEL: TRACE5 GENERAL CONSIDERATIONS

In this work, the LSTF has been modeled with 88 hydraulic components (7 BREAKs, 13 FILLs, 29 PIPEs, 2 PUMPs, 1 PRIZER, 21 TEEs, 14 VALVEs and 1 VESSEL). In order to characterize the heat transfer processes, 48 Heat Structure components (Steam Generator U-tubes, core power, pressurizer heaters and heat losses) have been considered. Figure 1 shows the nodalization of the model using SNAP (Symbolic Nuclear Analysis Package software).

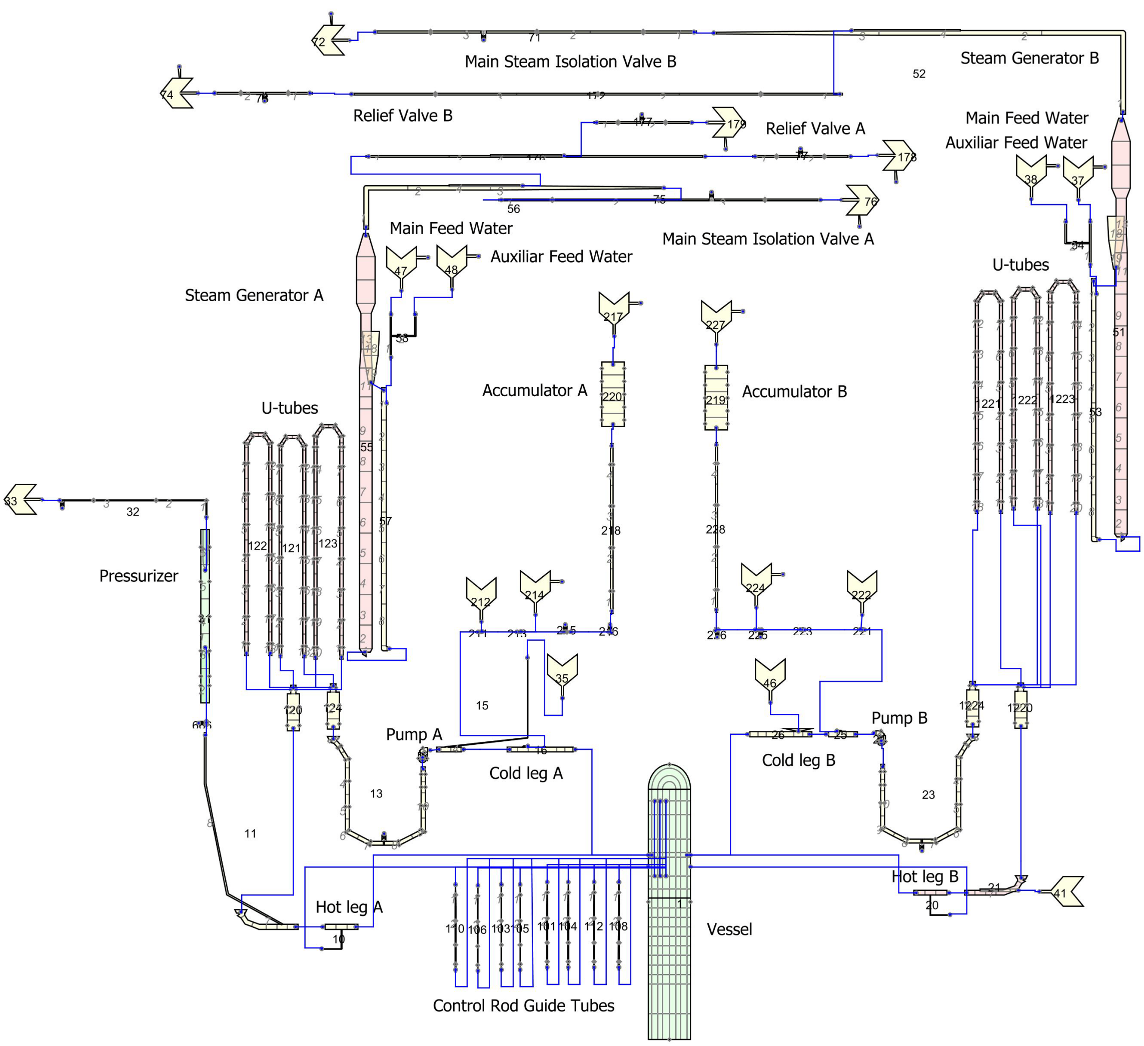

Figure 1 Model nodalization used for simulation.

In order to model the pressure vessel, a 3D–VESSEL component has been considered (Figure 2). A nodalization consisting of 19 axial levels, 4 radial rings and 10 azimuthal sectors has been selected. This nodalization characterizes with an acceptable detail the actual features of the LSTF vessel. Increasing the number of axial levels, azimuthal sectors or radial rings, does not improve significantly the agreement with experimental results, but increases CPU time. For each axial level, volume and effective flow area fractions have been set according to technical specifications provided by the organization [13]. Active core is located between levels 3 and 11. Level 12 simulates the upper core plate. Levels 13 to 15 characterize the vessel upper plenum. The upper core support plate is located in level 16. Finally, upper head is defined between levels 17 to 19. 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 PIPEs components, connecting levels 13 and 19 and allowing the flow between upper head and upper plenum.

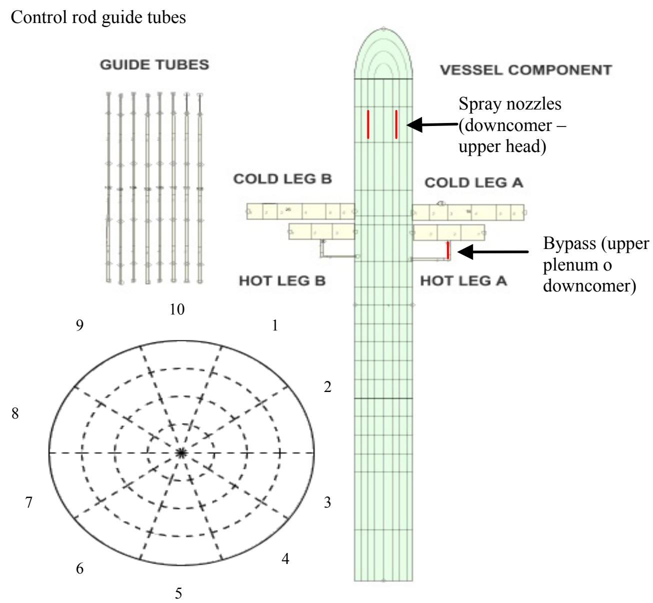

Figure 2 3D Vessel nodalization and connections visualized with SNAP.

30 HTSTRs simulate the fuel assemblies in the active core. The component power manages the power supplied by each HTSTR to the 3D-VESSEL. Fuel elements (1008 in total) were distributed into the 3 rings: 154 elements in ring 1, 356 in ring 2 and 498 in ring 3 and also characterized by HTSTR components. In both axial and radial direction, peaking factors were considered. The power ratio in the axial direction presents a peaking factor of 1.495. On the other hand, depending on the radial ring, different peaking factors were considered (0.66 in ring 1, 1.51 in ring 2 and 1.0 in ring 3). The number of fuel rod components associated with each heat structure has been determined from the technical documentation given, taking into account the distribution of fuel rod elements in the vessel, as can be seen in Table 5.

Table 5 Number of heaters per heat structure.

HTSTR	Number of heaters	HTSTR	Number of heaters	HTSTR	Number of heaters
310	17	320	44	330	60
311	17	321	40	331	54
312	10	322	23	332	32
313	12	323	32	333	45
314	20	324	40	334	56
315	17	325	42	335	61
316	16	326	38	336	57
317	12	327	26	337	31
318	14	328	30	338	45
319	17	329	39	339	57

A detailed model of SG (geometry and thermal features) has been developed, due to the fact that TRACE5 does not include any pre-determined steam generator component. A representation of the SG nodalization can be seen in Figure 3. Both boiler and downcomer components of secondary-side have been modeled by TEEs components. U-tubes have been classified into three groups according to each average length and heat transfer features. Steam-separator model can be invoked 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 by using HTSTR components. Cylindrical-shape geometry has been used to best fit heat transmission. Critical heat flux flag has been set in order to use an AECL-IPPE table, calculating critical quality from Biasi correlation [10, 11]. Inner and outer surface boundary conditions for each axial level have been set to couple HTSTR component to hydro components (primary and secondary fluids). 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 U-tubes without drastically increasing CPU time, a 3-group configuration was finally chosen. Heat losses to environment have been added to secondary-side walls.

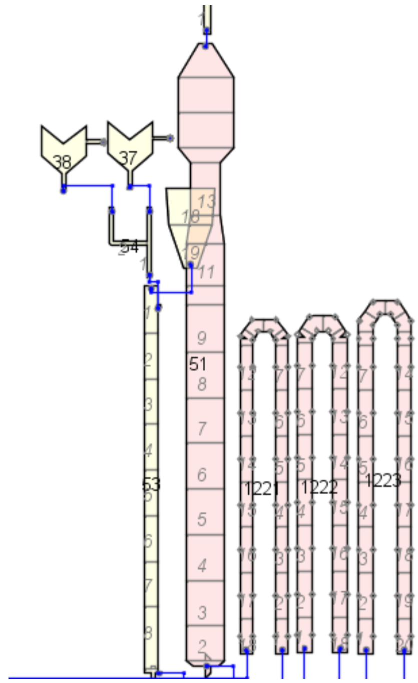

Figure 3 Steam generator nodalization.

5 RESULTS AND DISCUSSION

Steady-State conditions achieved in the simulation were in reasonable agreement with the experimental values. In Table 6, 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 5%. In order to achieve the steady state conditions, the duration of simulation was stated to 3000 seconds.

Table 6 Steady-state condition. Comparison between experiment and TRACE.

Item	Relative Error (%) (Loop with PZR)
Pressure Vessel	
Core Power	0.0
Primary Loop	
Hot Leg Fluid Temperature	0.2
Cold Leg Fluid Temperature	0.1
Mass Flow Rate	5.0
“Downcomer”-to-Hot Leg bypass	5.0
Pressurizer	
Pressure	0.6
Liquid Level	3.8
Steam Generators	
Secondary-side Pressure	0.8
Secondary-side liquid level	
Steam Flow Rate	4.0
Main Feedwater Flow Rate	4.0
Main Feedwater Temperature	0.0
Auxiliary Feedwater Temperature	0.0

Regarding to the SG U-tubes, simulated fluid temperature with TRACE5 has been compared with experimental values. As can be seen in Figure 4 and Figure 5 (group 2, up-flow and down-flow sides of both SGs), the relative error between the experiment and calculated values is lower than 2% in any case.

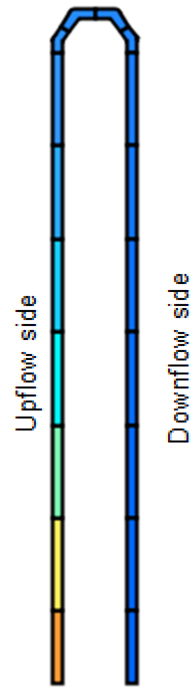

Figure 4 Fluid temperature. Relative error Exp./TRACE5 in %. SG A. U-tube group 2.

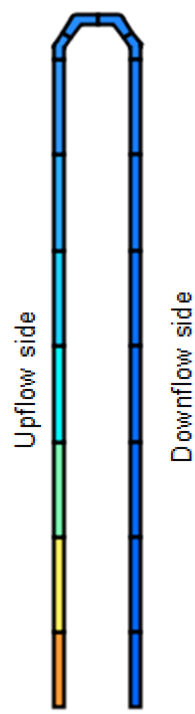

Figure 5 Fluid temperature. Relative error Exp./TRACE5 in %. SG B. U-tube group 2.

In Table 7 the chronological sequence of events are listed, comparing Normalized Time (normalized to the total transient time) between experimental and simulated results.

Table 7 Chronological sequence of events. Comparison between experiment and TRACE.

Event	Experiment. Normalized Time	TRACE. Normalized
Signal for termination of feed water	0.000	0.000
Scram signal	0.002	0.002
Primary coolant pumps stopped	0.009	0.009
Initiation of Auxiliary Feedwater	0.032	0.032
Natural circulation from single-phase to two-phase	0.036	0.037
PZR became empty of liquid	0.492	0.495
Core power decrease by LSTF core protection system, max. fuel rod surface	0.816	0.829
Termination of Auxiliary Feedwater	1.000	0.997

Variables presented in this section follow the requirements for an exhaustive analysis of the transient. The most important parameters studied in this report are the following: Pressures at both primary and secondary circuits, mass flow rate and inventory at the break, primary mass flow, vessel collapsed-liquid levels, maximum fuel rod surface temperature, core exit temperature, collapsed-liquid levels in hot and cold legs, mass flow in SG relief valves, liquid level in SG secondary-side and temperatures in hot legs, downcomer and upper head.

5.1 System pressures

Figure 6 compares primary and secondary pressures during the first part of the transient (Normalized Time between 0 and 0.025). In this time period primary pressure increases due to the high core power after the MSIVs closure following the scram signal. The SG secondary-side pressure rapidly increases due also to the high core power after the closure of the MSIVs, keeping the RVs open for a while (0.005 to 0.01 NT) in both steam generators. The secondary pressure begins to fluctuate afterwards, between the fixed predetermined values by cycle opening of RVs, as it can be seen in Figures 6 and 7.

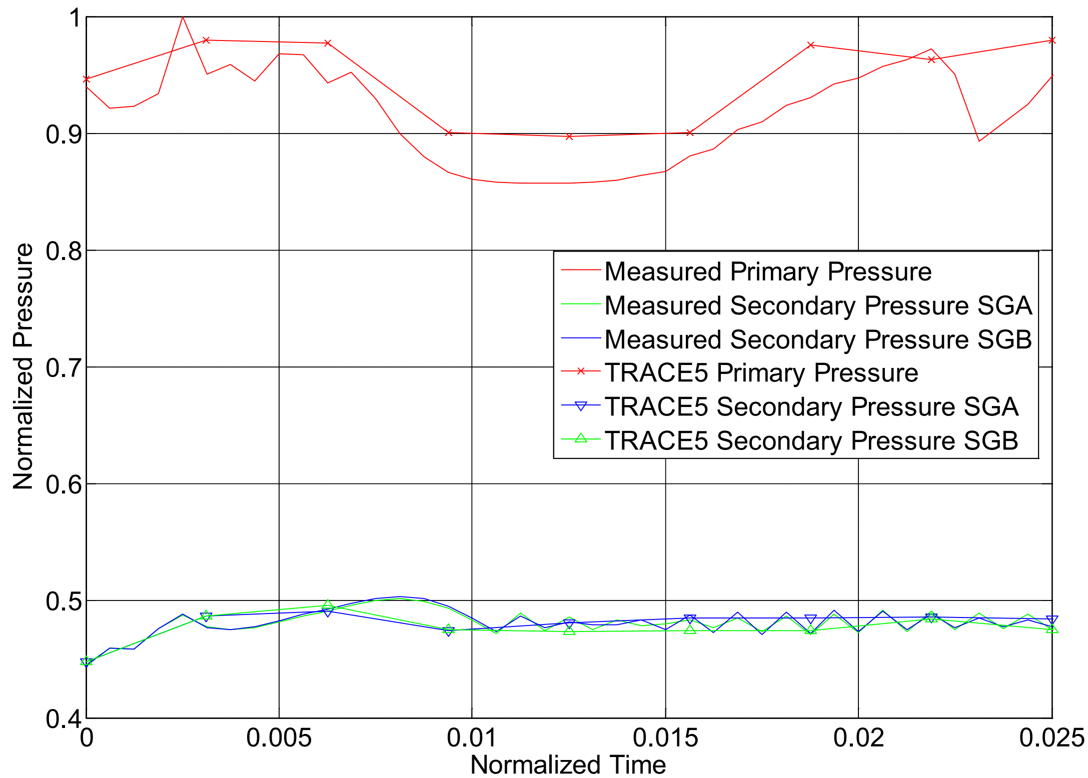

Figure 6 Primary and secondary pressures (normalized time 0 to 0.025).

When the collapsed liquid level of SG secondary-side decreases below 0.5 m the AFW system is activated. Primary pressure is kept almost constant by cycle opening of the PORV. The secondary-side pressure of the steam generator is also kept constant by cycle opening of the RVs. This behavior is maintained until near 0.84 NT, as can be seen in Figure 7.

At 0.84 NT, the core protection system actuates due to the high temperature of the core. In this moment, core power is reduced according to logic control (Table 2), decreasing the primary and secondary pressure. Then, the primary and secondary pressures become almost constant, thus reaching a nearly-equilibrium condition with well-cooled core at 1 NT

TRACE5 adequately reproduces the general behavior of both primary and secondary pressures during the whole transient, even the drop of pressure produced by the core protection actuation (between 0.8 and 1.0 NT). Primary and secondary pressures turn almost constant almost reaching the quasi-equilibrium condition with the core properly cooled at the end of the transient.

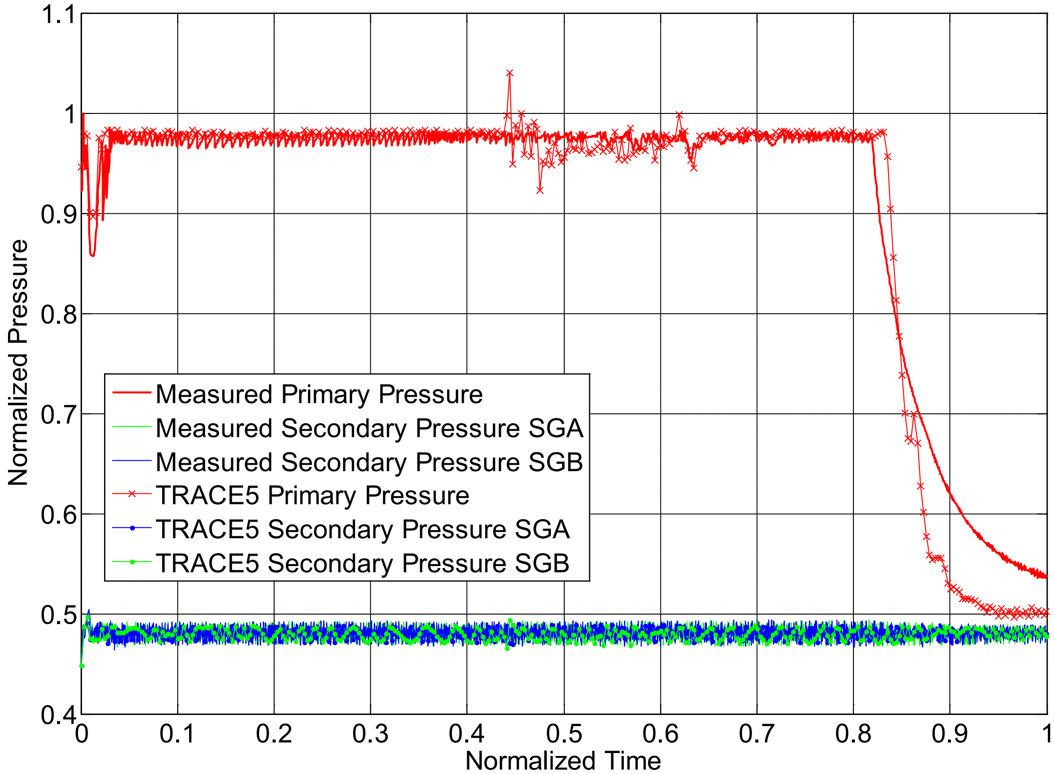

Figure 7 Primary and secondary pressures (normalized time 0 to 1).

5.2 Coolant discharge through PORV

Figure 8 shows the coolant inventory discharged through the pressurizer PORV. Experimentally, mass flow rate through PORV is obtained by measuring the liquid level in the break flow Storage Tank (ST). In the experiment, the ST liquid level starts to decrease after the first opening of the PORV probably due to condensation of the steam in the discharge line after discharging air that initially occupied, and turned to increase after the fourth opening of the PORV. At approximately 0.01 NT a change from two-phase to single-phase vapor through the PORV is produced.

The model is capable of reproducing the general tendencies and behavior of the transient. However, some discrepancies are observed during the time interval simulated. As can be seen in Figure 8a, in the simulated discharged primary coolant mass through the PORV, there are two abrupt increases, which are not observed in the experiment. The first increase is produced when the SG U-tubes liquid level starts to fluctuate (at 0.45 NT), as can be seen in Figure 8b. In this moment, heat transfer between the primary and secondary side in both SGs is drastically reduced. At 0.6 NT, U tubes are completely empty, and heat transfer between the primary and the secondary sides is again decreased. The poor heat transfer during the time period when U-tubes level fluctuates, produces an increase in the pressure at 0.45 and 0.6 NT, as observed in Figure 7. The increase in the primary pressure produces an increase of the discharged coolant through the PORV.

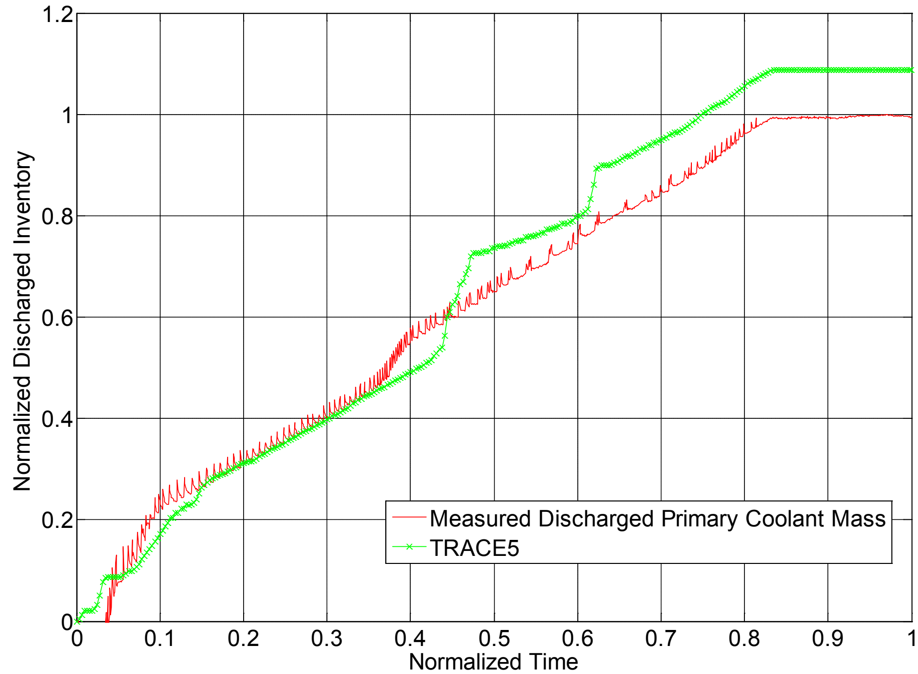

Figure 8a Discharged inventory through PORV.

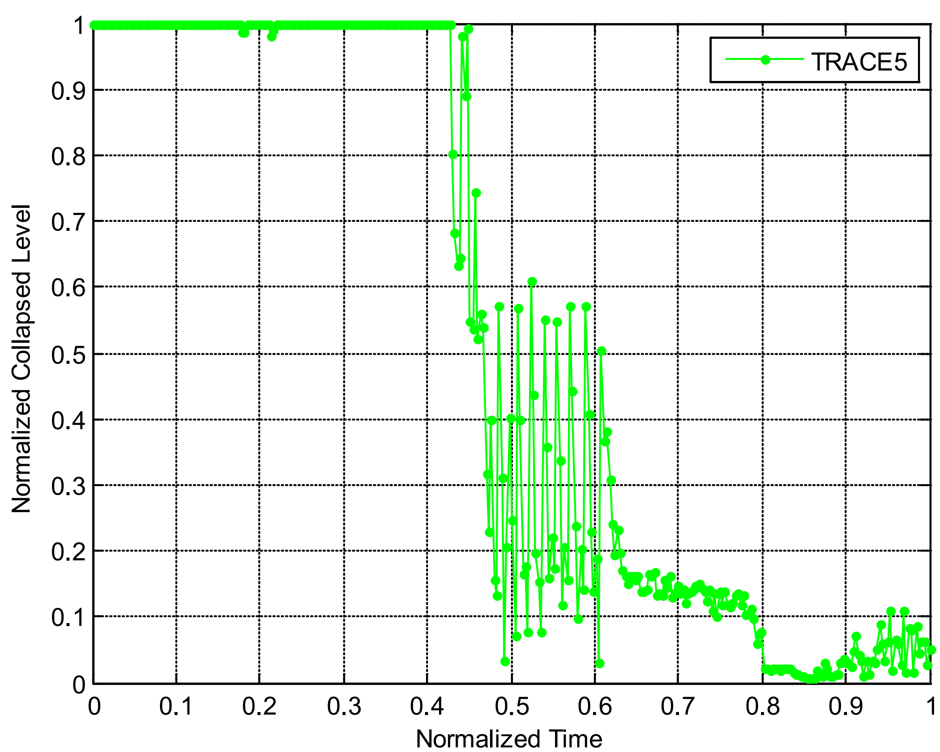

Figure 8b U-tubes collapsed liquid level.

5.3 Primary loops mass flow rate

Primary mass flow rate in loop A and B are shown in Figures 9 to 12. The mass flow rate in the primary loops is measured via a Venturi flow meter in each of the pumps of the primary loops. The mass flow rapidly drops following the beginning of the pumps coast down, and natural circulation begins when the pumps are completely stopped (at 0.01 NT, as can be seen in Figures 9 and 11). The primary loop flow turned into two-phase natural circulation at about 0.04 NT. The primary loop mass flow rate gradually increased thereafter and started to decrease with some oscillation after around 0.36 NT when the liquid level in the hot leg became lower than the half height (Figures 10 and 12).

In general, all tendencies of the mass flow are well reproduced by TRACE5 in both loops, as the following figures show. The most significant difference is registered in the oscillations between 0.4 NT and 0.6 NT, and in an overestimated mass flow in the first part of the transient (0.05 NT to 0.45 NT), as it can be seen in Figures 10 and 12.

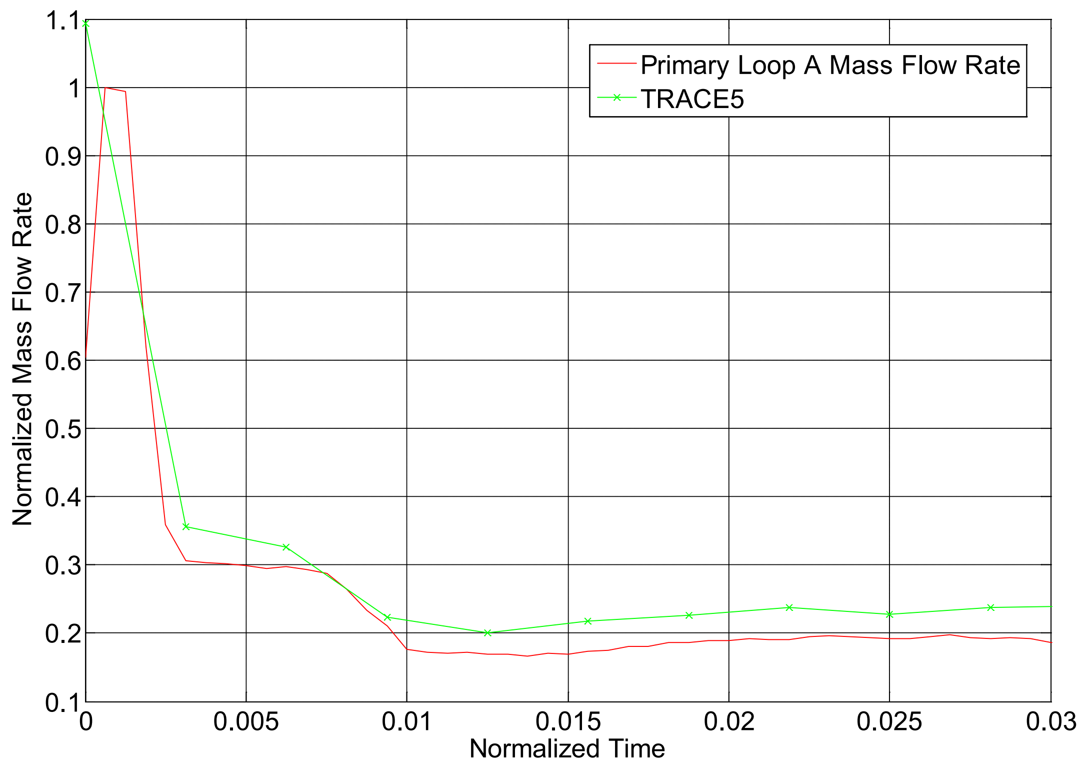

Figure 9 Primary mass flow, loop A (normalized time 0 to 0.03).

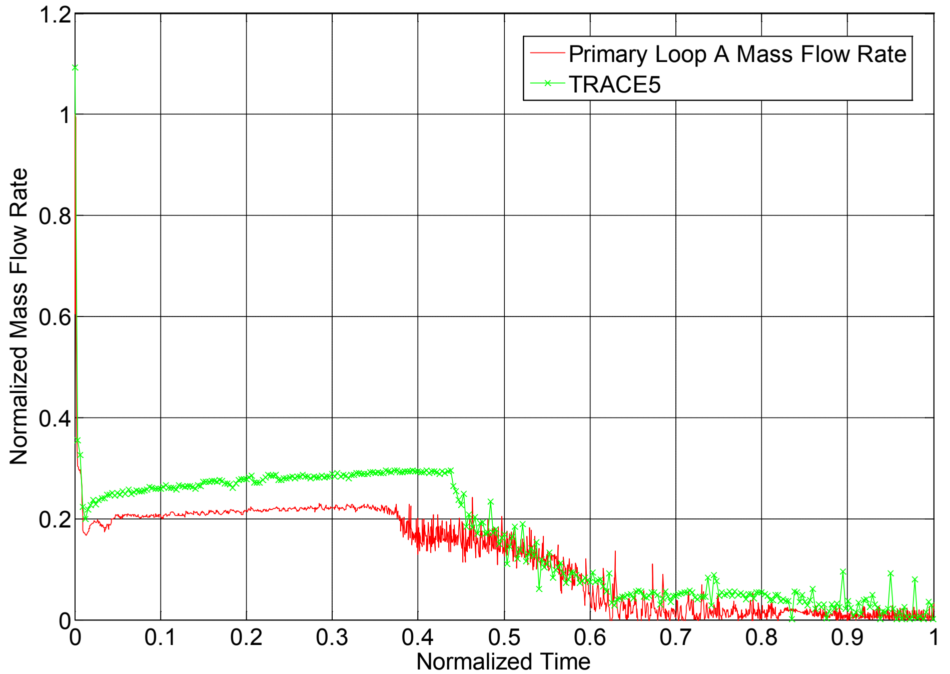

Figure 10 Primary mass flow, loop A (normalized time 0 to 1).

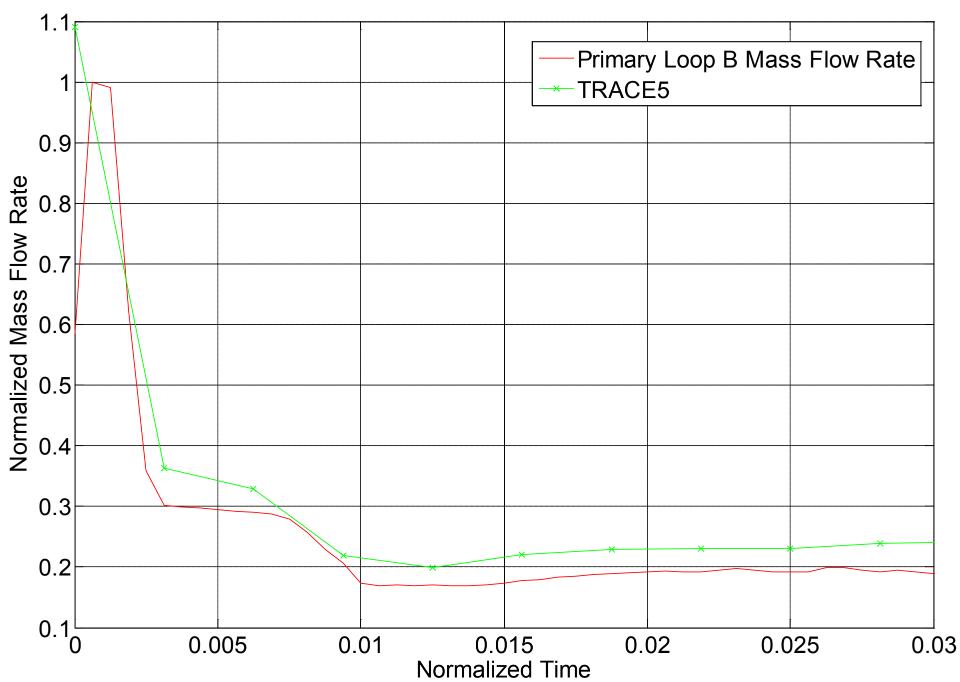

Figure 11 Primary mass flow, loop B (normalized time 0 to 0.03).

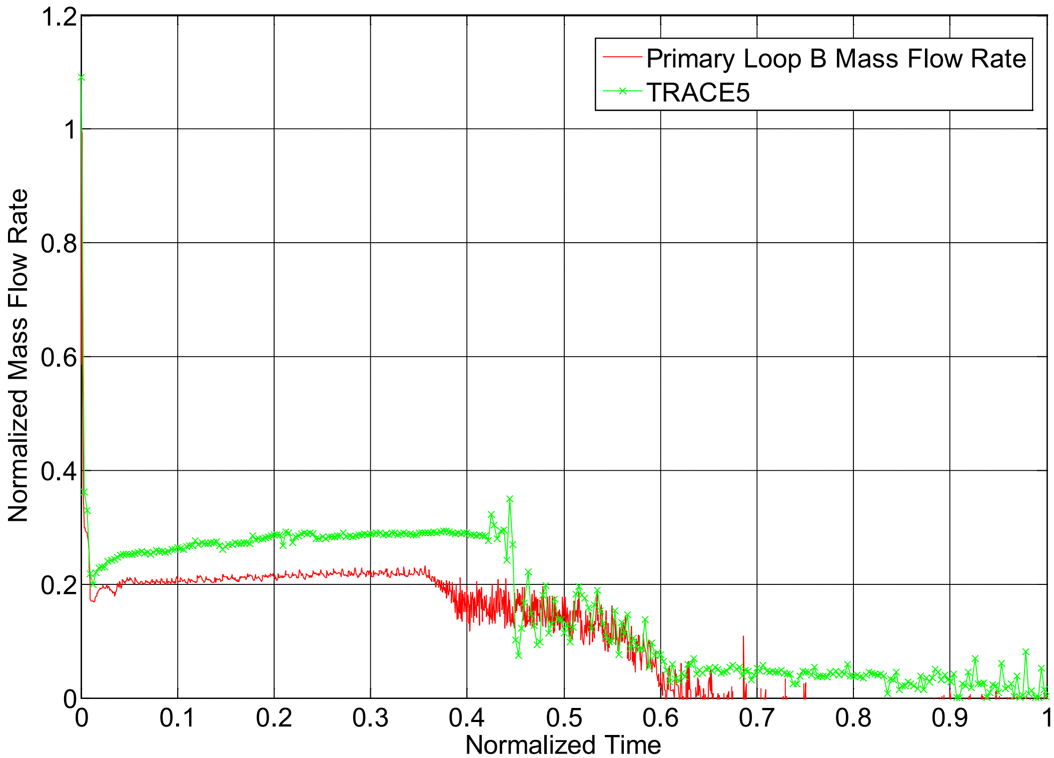

Figure 12 Primary mass flow, loop B (normalized time 0 to 1).

5.4 Vessel collapsed liquid levels

Figures 13, 14 and 15 show a comparison of the collapsed liquid levels in the upper plenum, core and downcomer, respectively between experimental and TRACE results. In the experiment, the collapsed liquid level is computed from differences in pressure between the upper and lower parts of each region, and the coolant densities. The collapsed liquid level in the upper plenum changes in response to the liquid level of the hot leg.

TRACE adequately reproduces all collapsed liquid levels during the first and second part of the transient. However, the core liquid level is slightly lower compared to the experimental during the first part of the transient until 0.5 NT. For the upper plenum and the downcomer, the first part of the transient is properly adjusted, whilst the level drop is more abrupt with TRACE than in the experiment.

When U-tubes are completely empty (at 0.6 NT approximately), a sudden vaporization is predicted by TRACE5, which is not observed in the experiment. The effect of this vaporization can be observed in the Upper Plenum liquid level at 0.6 NT (Figure 13), in the Downcomer liquid level (Figure 15) and in the Active Core (Figure 14). At 0.6 NT, no change in the Maximum Fuel Rod Surface Temperature is observed in the experiment (Figure 17). However, for the same Active Core liquid level simulated by TRACE5, an increase in the Maximum Fuel Rod Surface Temperature is observed at 0.6 NT. The reason for such disagreement between experimental and simulated values can be due to the flashing conditions predicted by TRACE5, producing an abrupt loss of liquid level in the pressurized vessel.

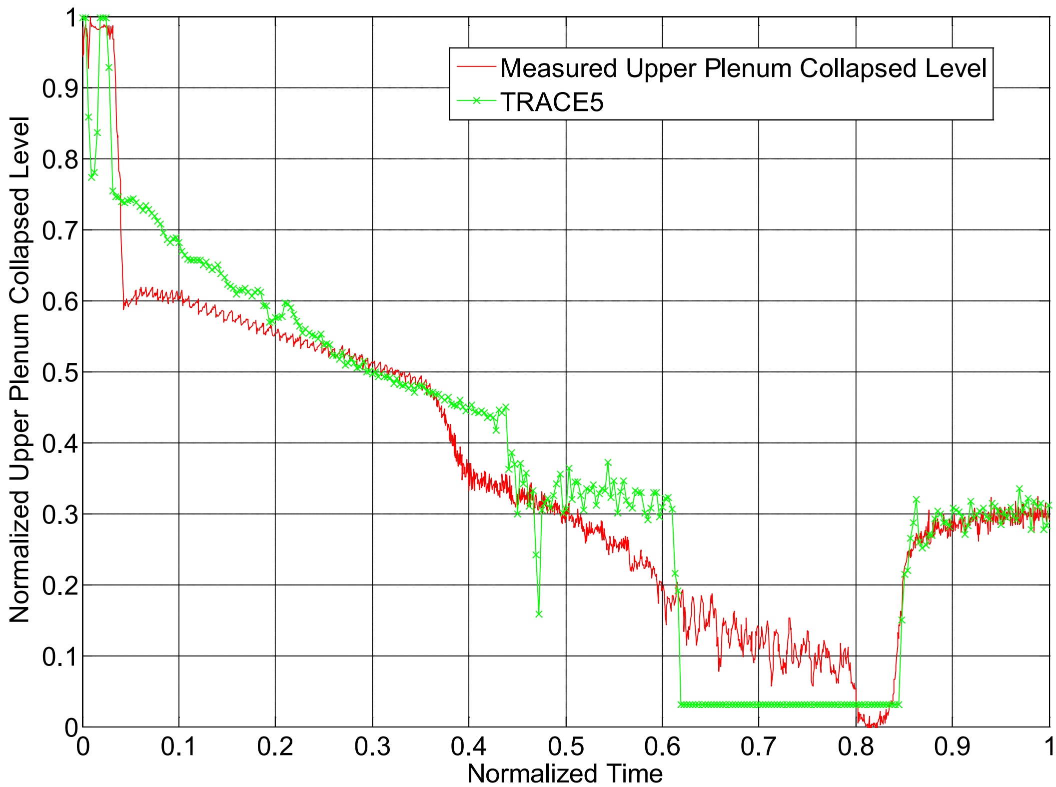

Figure 13 Upper plenum collapsed liquid level.

In the last part of the transient, at 0.82 NT, the core protection system is activated, drastically reducing the core power. From this moment on, the core, upper plenum and downcomer liquid level increases.

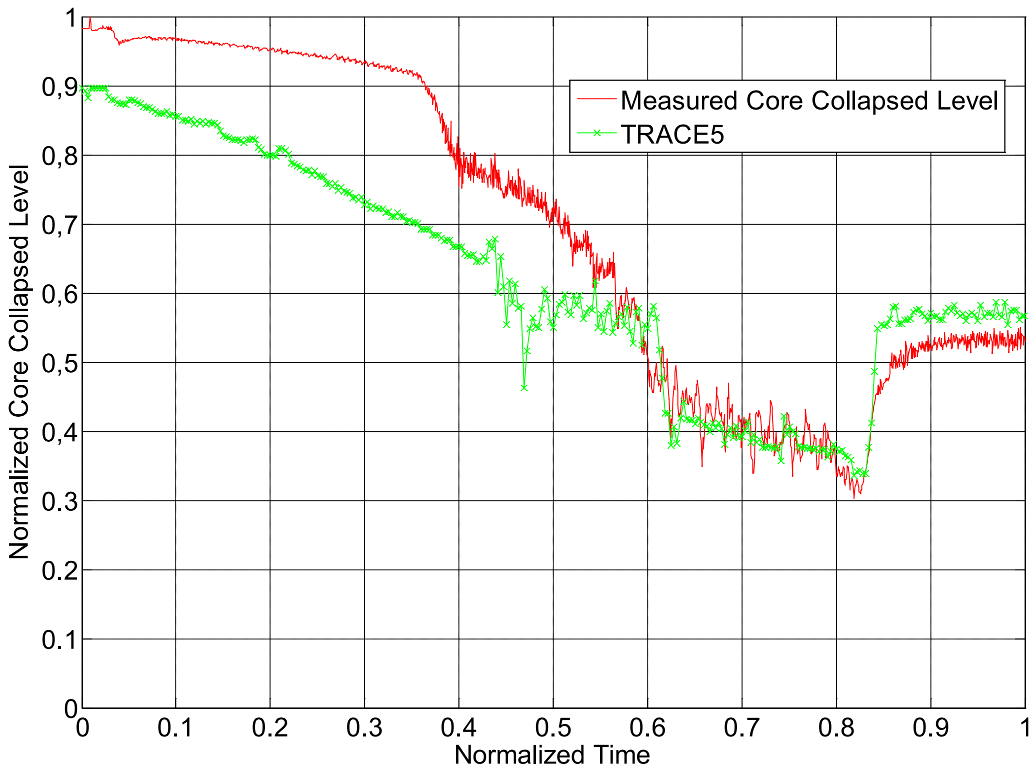

Figure 14 Core collapsed liquid level.

Simultaneously with the TRACE upper plenum drop, also downcomer starts to decrease its liquid level. In the last part of the transient, at 0.82 NT, the core protection system is activated, drastically reducing the core power. From this moment on, the core, upper plenum and downcomer liquid level increases.

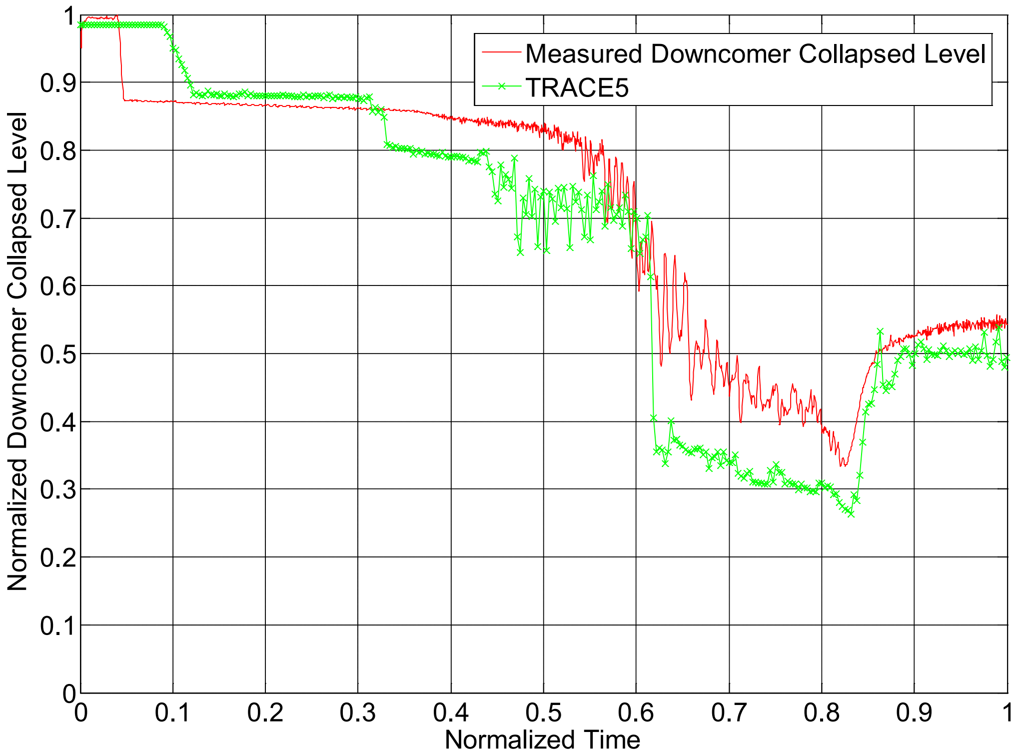

Figure 15 Downcomer collapsed liquid level.

5.5 Maximum fuel rod surface temperature

TRACE5 estimates the evolution of the maximum fuel rod temperature in the core properly. In the beginning, the upper half of the core is producing vapour under saturation conditions, whilst the lower half is subcooled until the whole core begins to boil at about 0.35 NT (Figure 14). At 0.65 NT a temporal increase of the rod surface temperature begins, although liquid level in the upper plenum still exists. At 0.75 NT a significant increase in the fuel rod surface temperature occurs because the whole liquid in the core boils when the upper plenum is completely empty. The core power automatically decreases to the 75% by the core protection system at about 0.8 NT when the maximum fuel rod surface temperature reaches its maximum (Figure 18). Most of the core was quenched at about 0.84 NT being followed by a decrease in the primary pressure and increase in the core collapsed liquid level.

The main disagreement between experimental and TRACE simulation is the gradual increase of temperature predicted by TRACE5 between 0.6 and 0.8 NT, which is not observed in the experiment, as it can be seen in Figure 17.

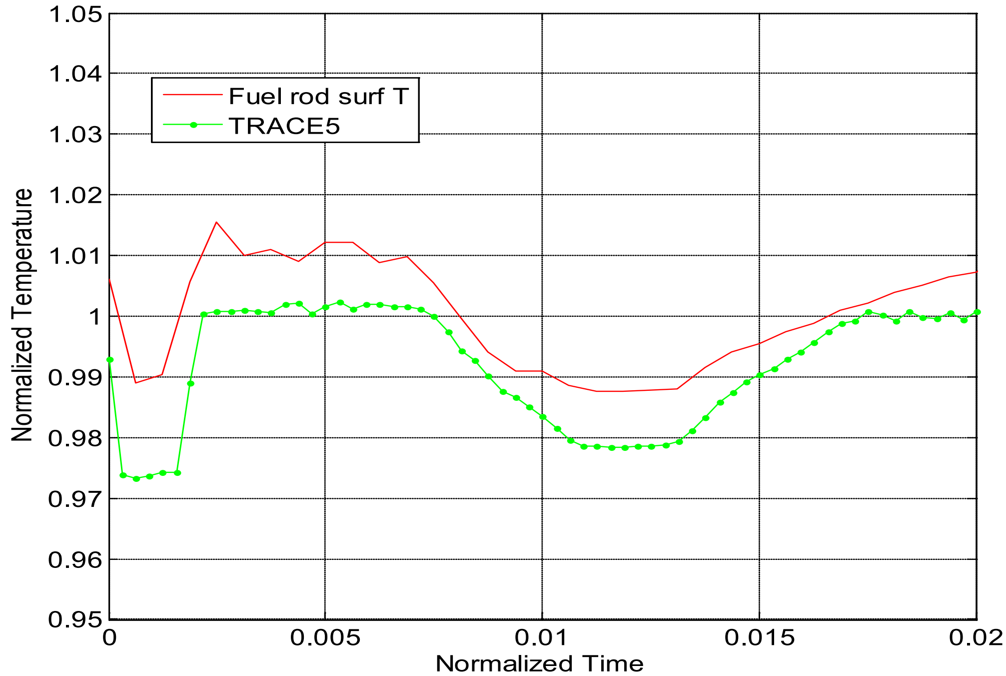

Figure 16 Maximum fuel rod surface temperature (normalized time 0 to 0.02).

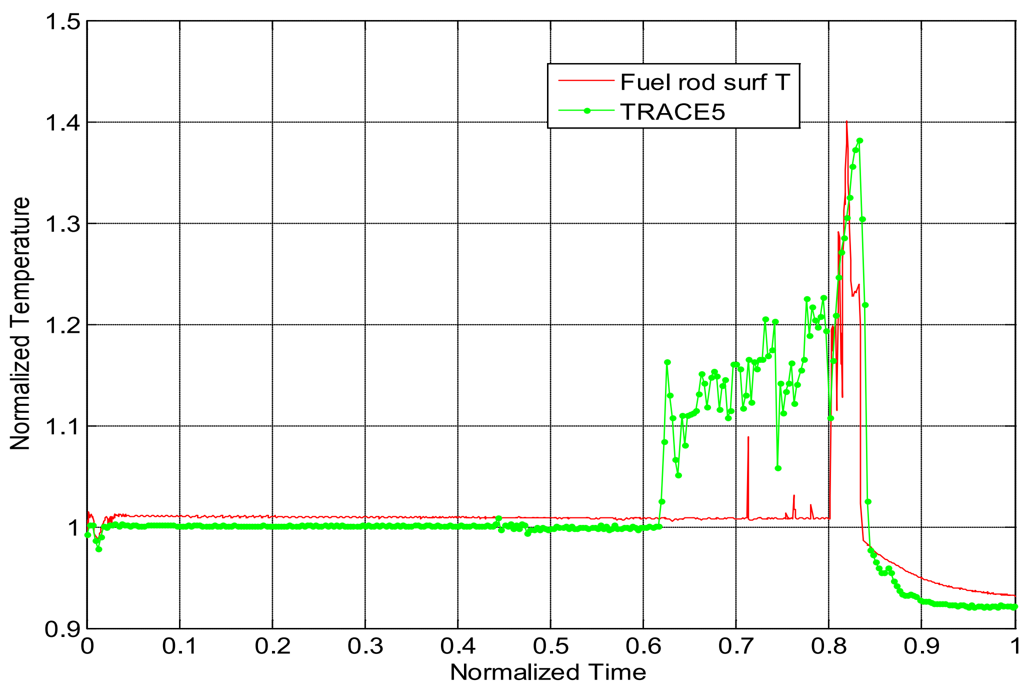

Figure 17 Maximum fuel rod surface temperature (normalized time 0 to 1).

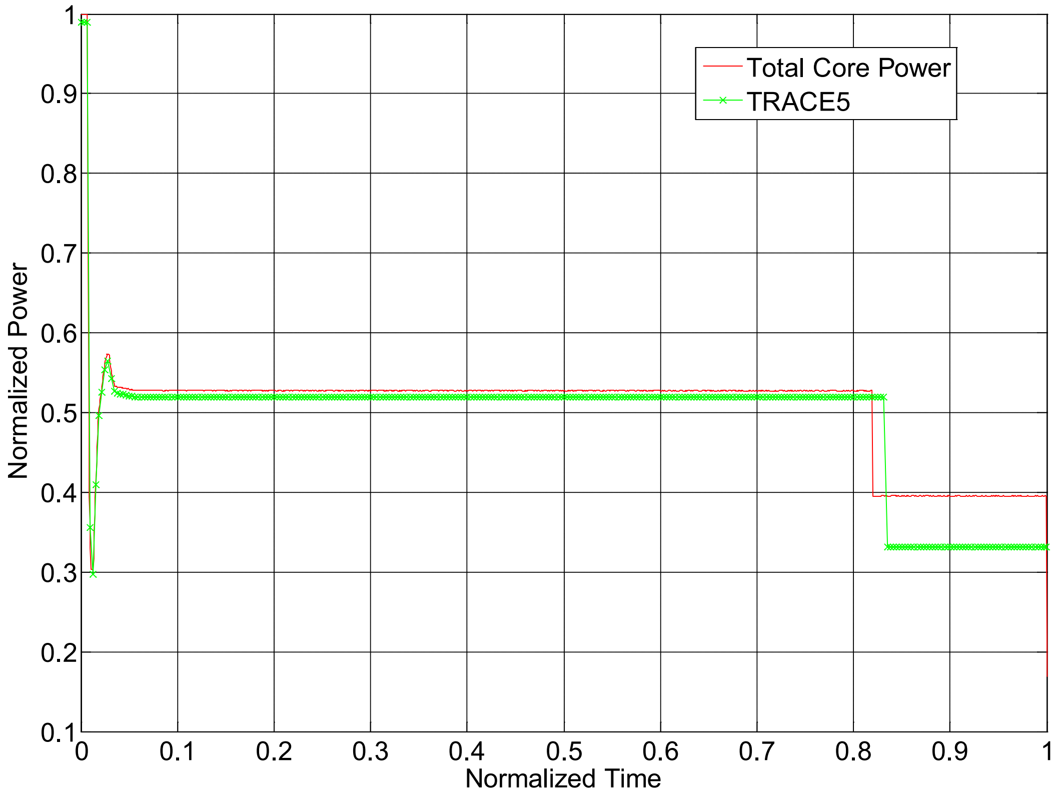

Figure 18 Core power.

5.6 Hot legs liquid levels

Figures 19 and 20 show the liquid level in the hot leg A and B, respectively. Experimentally, liquid level was obtained with a three gamma ray beam densitometer and the saturated coolant densities. The behaviors of hot legs are almost symmetric in both loops. The flow in the hot leg changes from single-phase liquid to two-phase at about 0.035 NT.

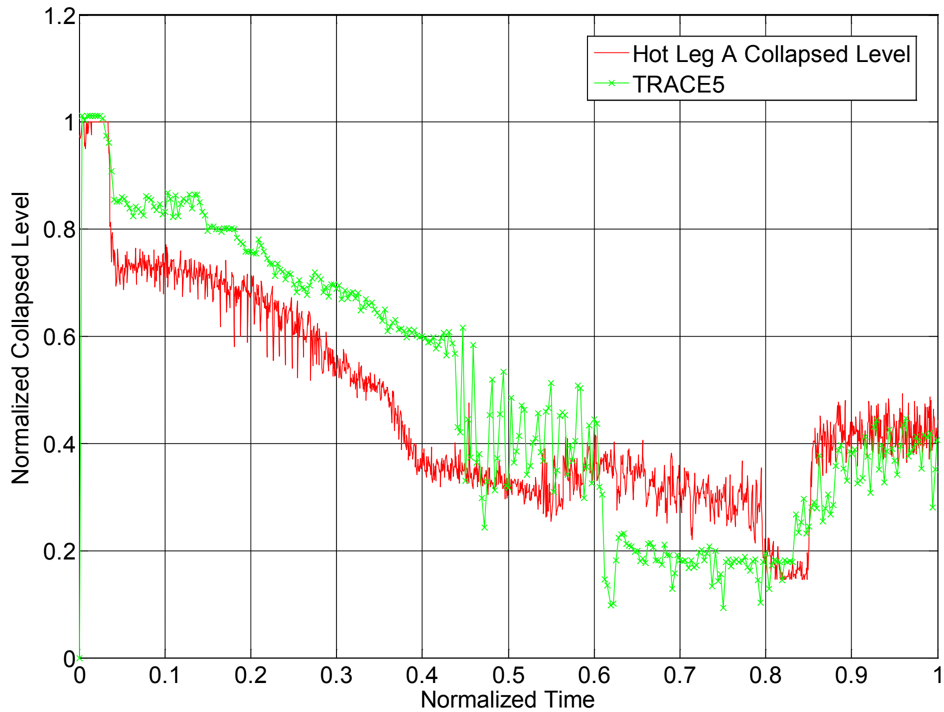

Figure 19 Collapsed liquid level in hot leg A.

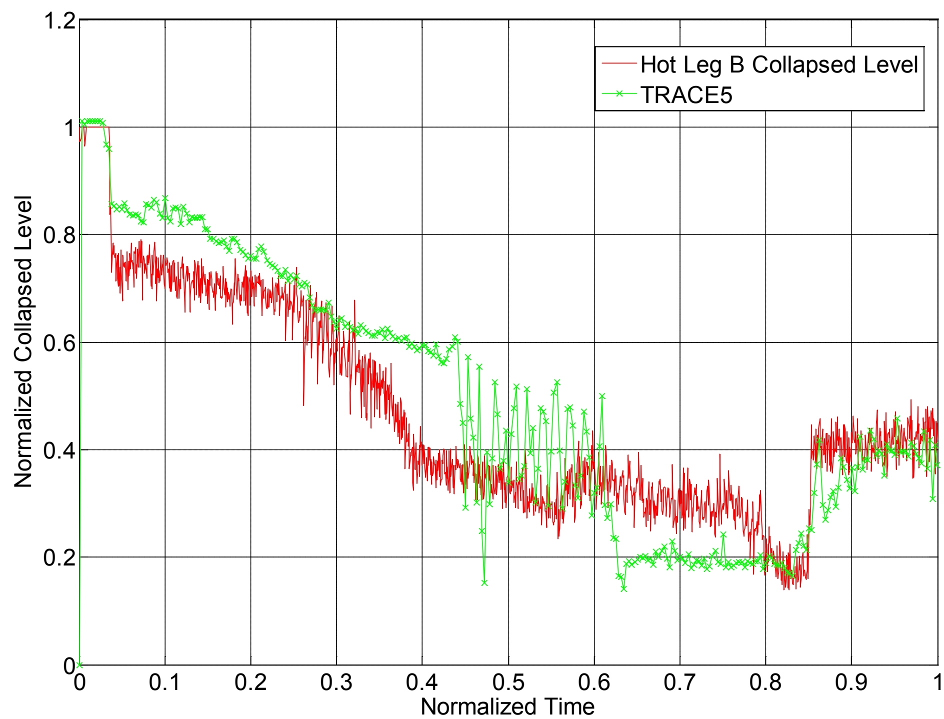

Figure 20 Collapsed liquid level in hot leg B.

5.7 Steam Generator relief valve flow rate

The Relief Valves are kept open for a while when the secondary pressure of the steam generators rapidly increases, due to the high core power. Afterwards, at around 0.01 NT, the pressure of the secondary starts to oscillate by cycle opening of the relief valves (between the two fixed values). In general, a good reproduction of the opening and closing intervals is achieved, as it can be seen in Figures 21-24.

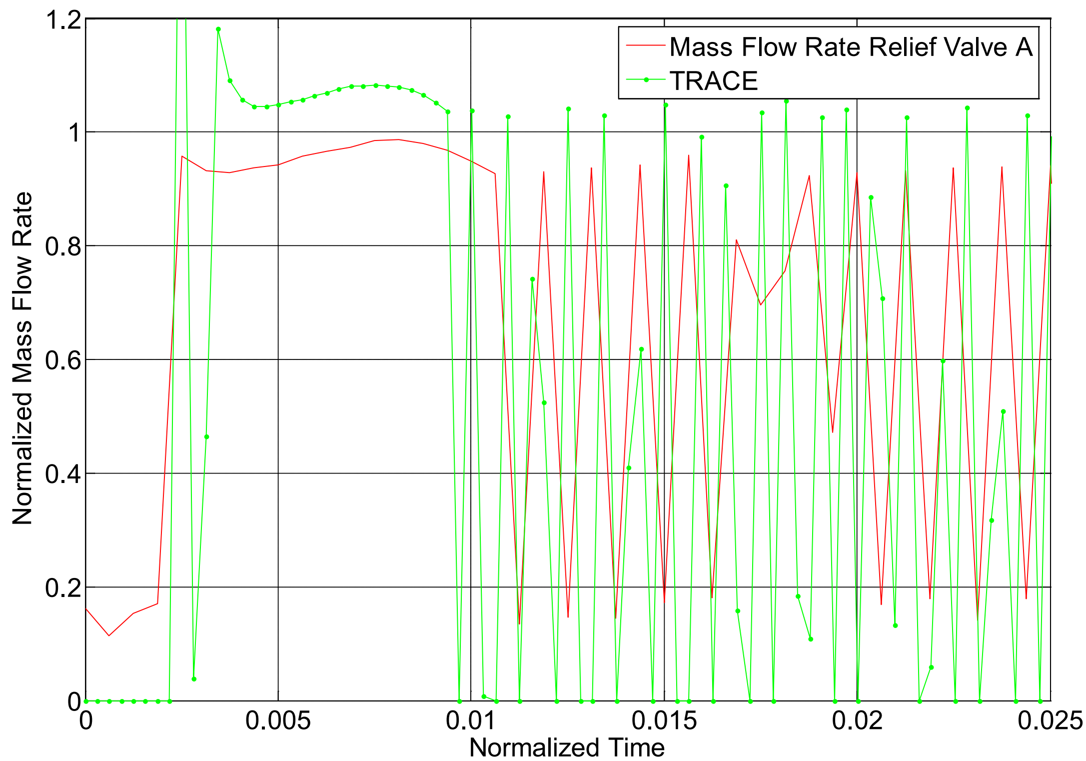

Figure 21 SG A relief valve mass flow rate (normalized time 0 to 0.025).

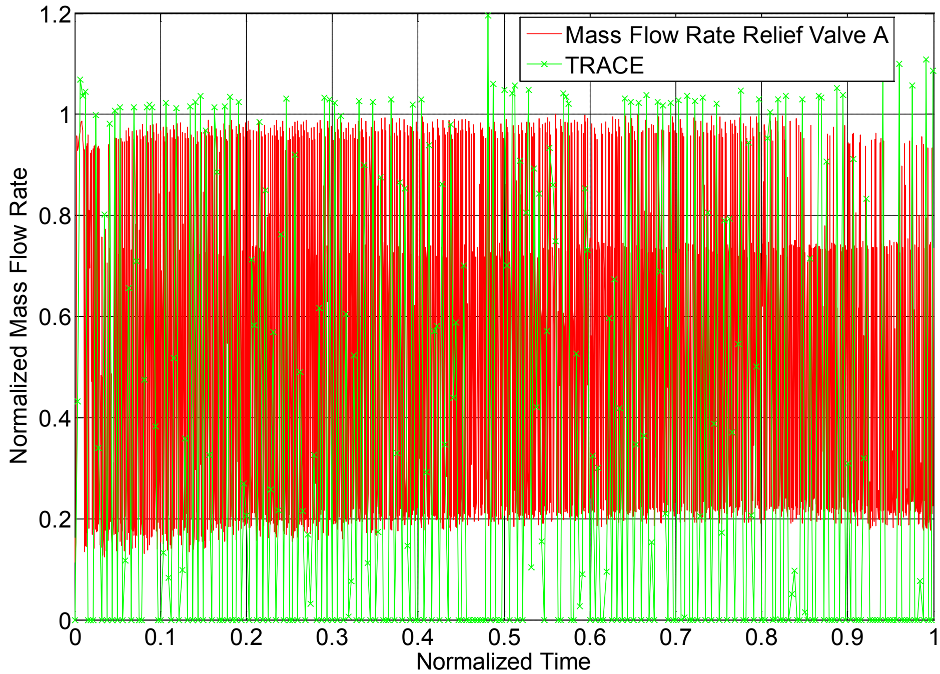

Figure 22 SG A relief valve mass flow rate (normalized time 0 to 1).

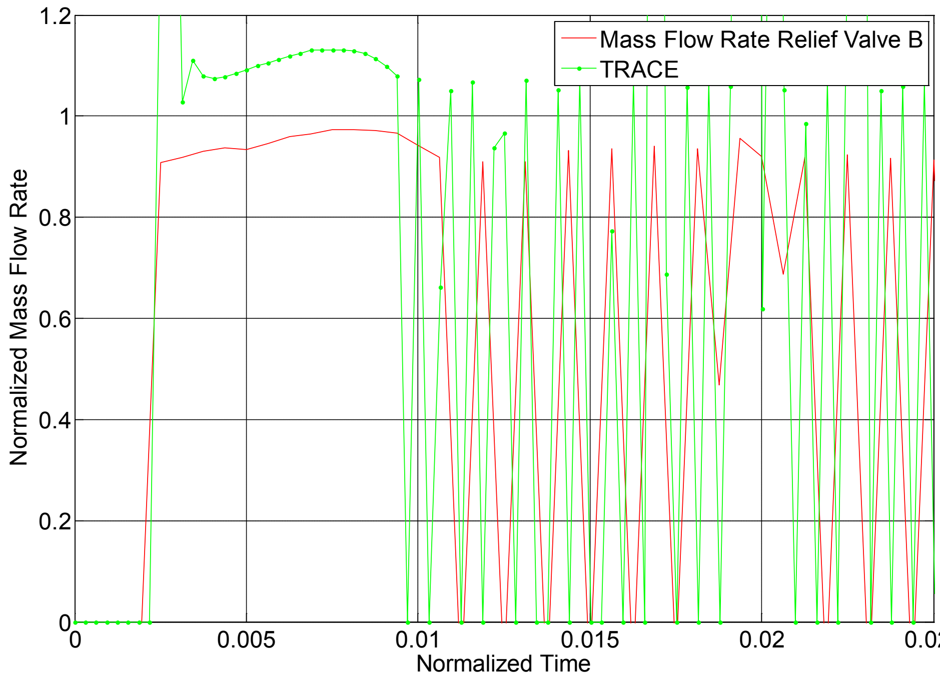

Figure 23 SG B relief valve mass flow rate (normalized time 0 to 0.025).

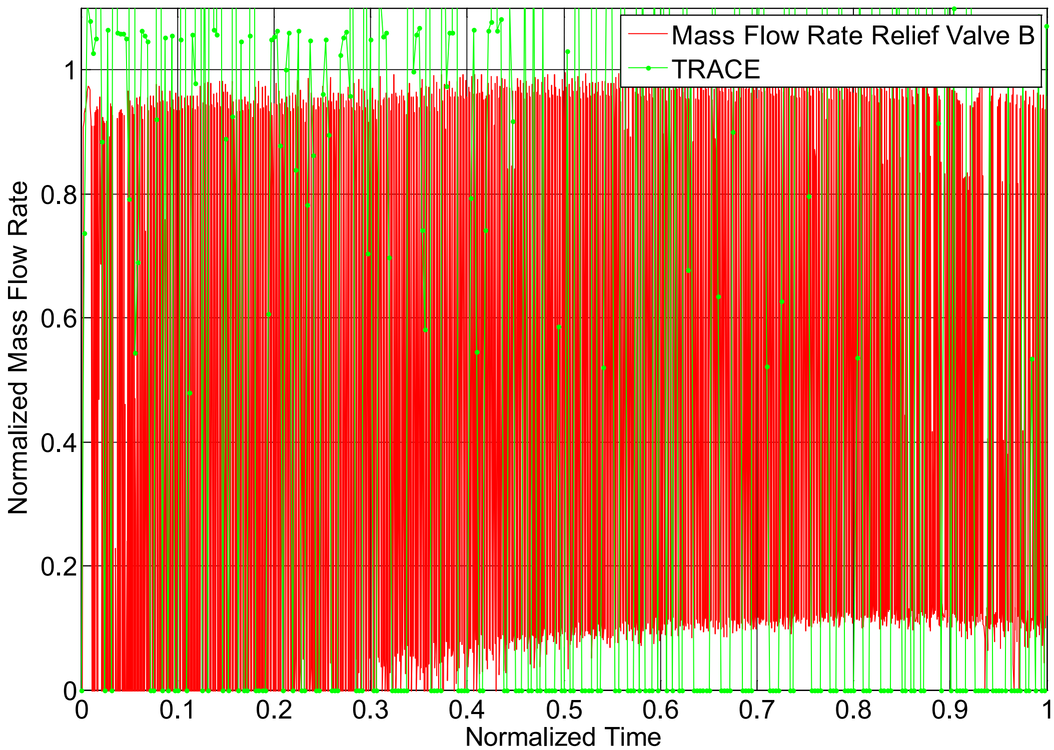

Figure 24 SG B relief valve mass flow rate (normalized time 0 to 1).

5.8 Main Steam Generator Isolation Valves mass flow rate

In Figures 25 and 26 it can be seen the MSIV mass flow rate in both SG. MSIV are closed early in the transient (at 0.0025 NT). The negative mass flow rate in the experimental measurement should not considered.

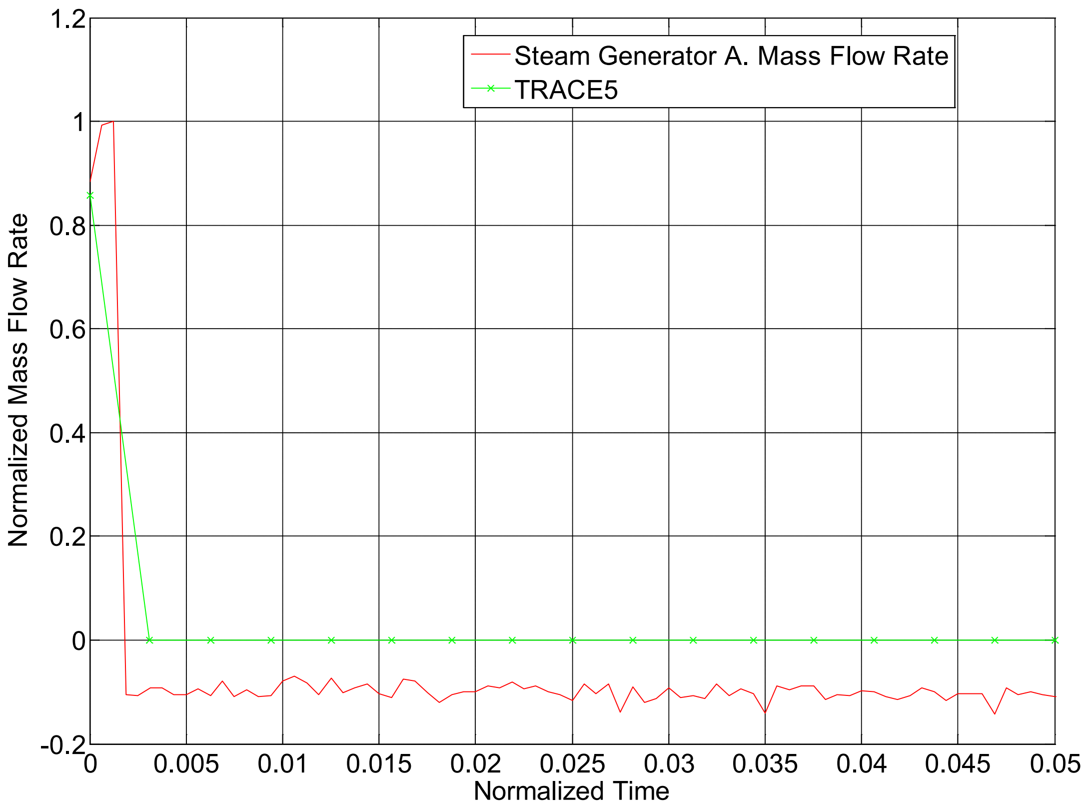

Figure 25 SG A relief valve mass flow rate (normalized time 0 to 0.05).

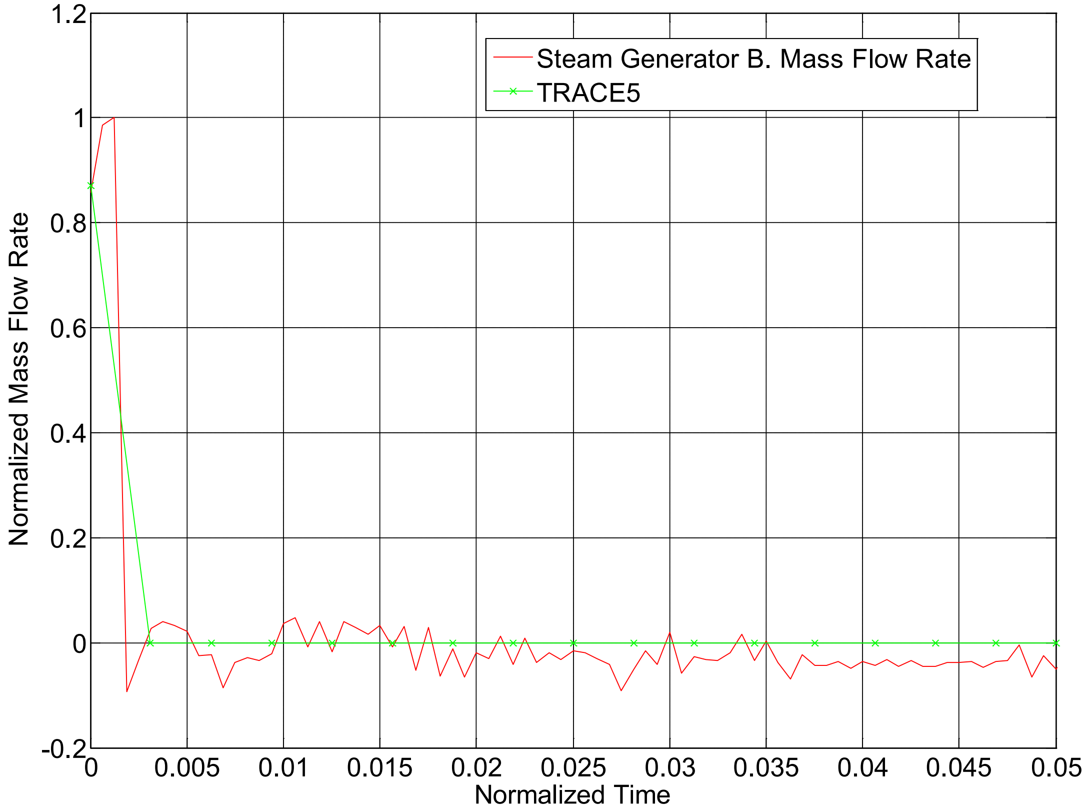

Figure 26 SG B relief valve mass flow rate (normalized time 0 to 0.05).

5.9 Steam generators secondary-side liquid level

The following Figures (27 and 28) show the collapsed liquid level of the secondary side of the steam generator. The actuation of the Auxiliary Feedwater (AFW) starts at 0.032 NT, when the collapsed liquid level of the secondary side of the steam generator decreases below 0.07 Normalized Value of liquid level (NV). The liquid level stays in the interval from 0.04 to 0.12 NLL after the beginning of the AFW until 0.82 NT, when the automatic core power decay curve begins.

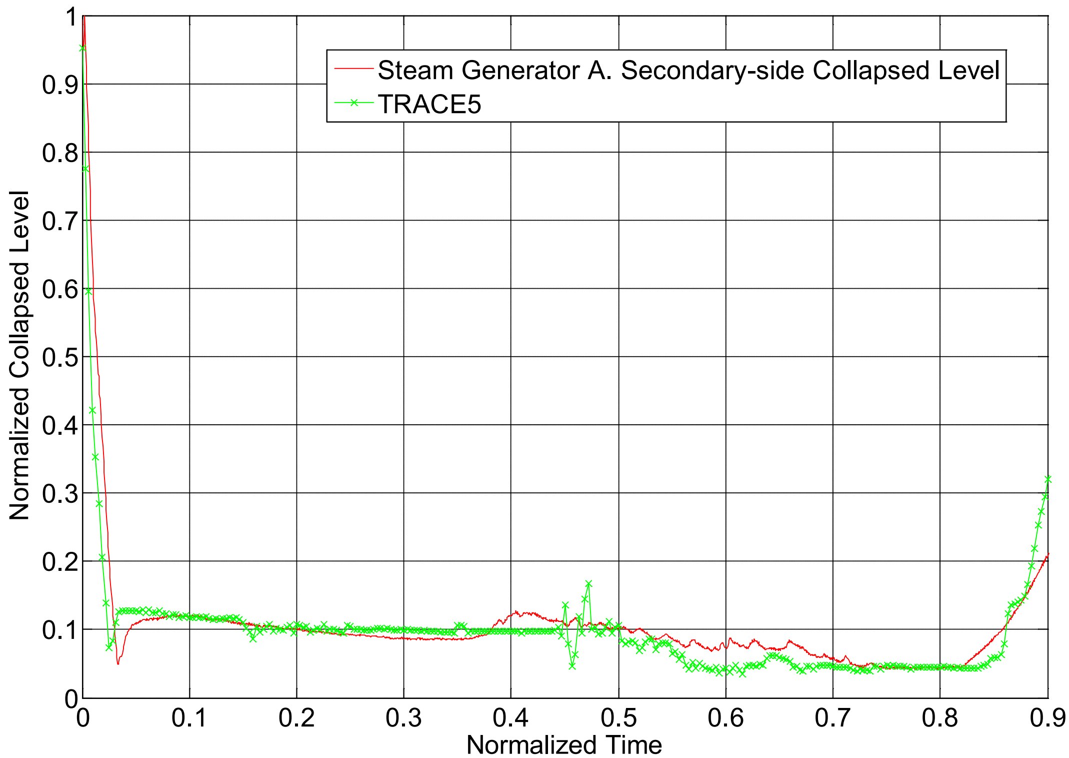

Figure 27 Steam generator A. Secondary-side collapsed liquid level.

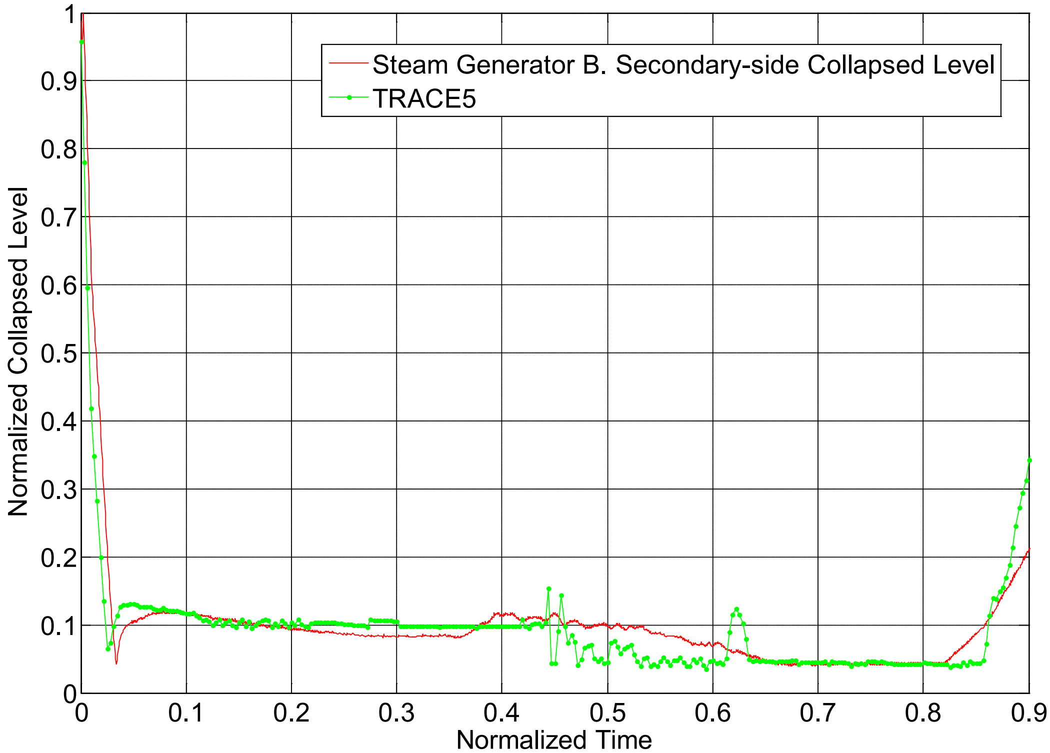

Figure 28 Steam generator B. Secondary-side collapsed liquid level.

5.10 Steam generators U-tubes liquid level

Figures 29 to 32 show the collapsed liquid level in U-tubes of both Steam Generators. In the period defined between 0.45 and 0.6 NT, U-tubes liquid level present strong oscillations. The main effect of this phenomenon is the lack of heat transfer in Steam Generators. Liquid level oscillation started at 0.45 NT in the three U-tube groups causing a single-phase and two-phase natural circulation among the tubes. The liquid levels changed similarly in both upflow-side and downflow-side of each U-tube due to counter balance of water head.

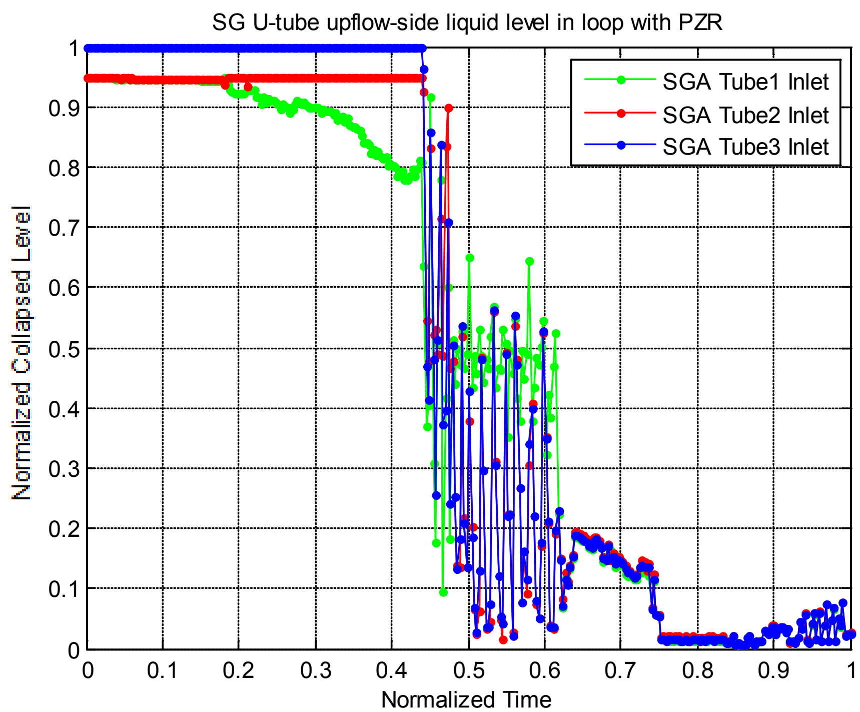

Figure 29 Steam generator A. U-tube upflow side liquid level.

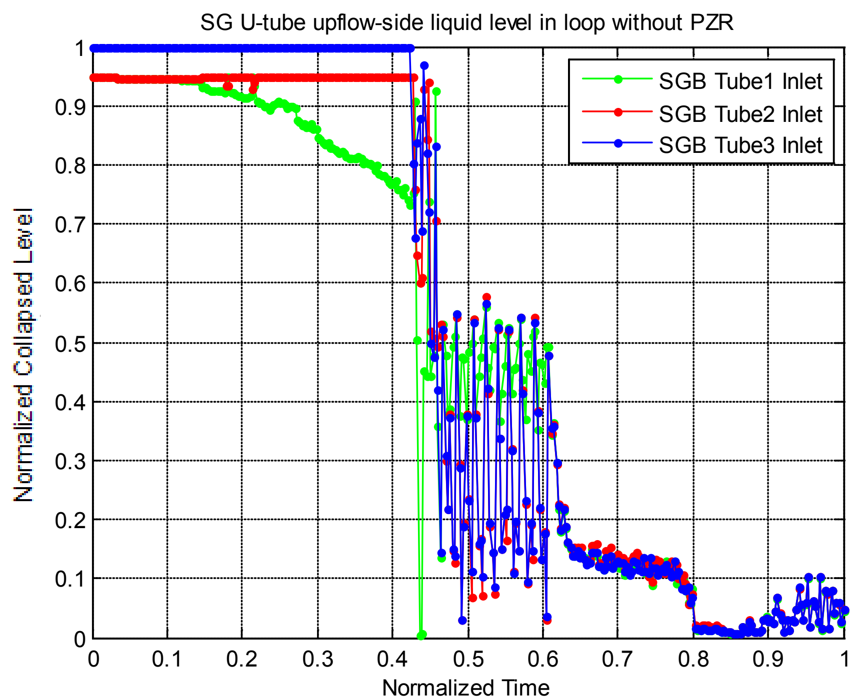

Figure 30 Steam generator B. U-tube upflow side liquid level.

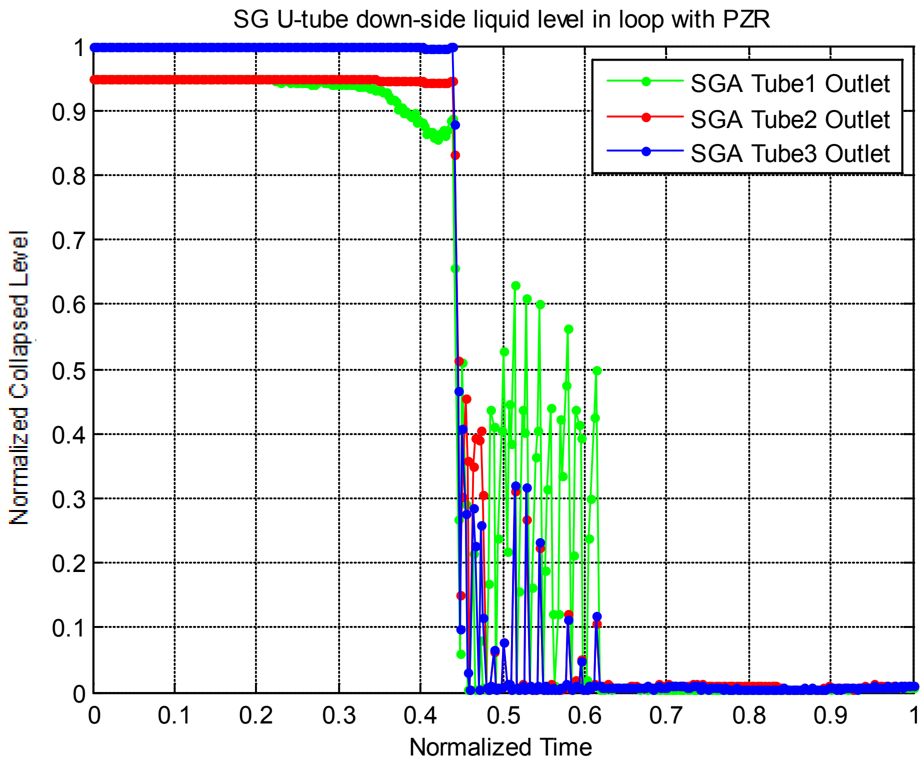

Figure 31 Steam generator A. U-tube downflow side liquid level.

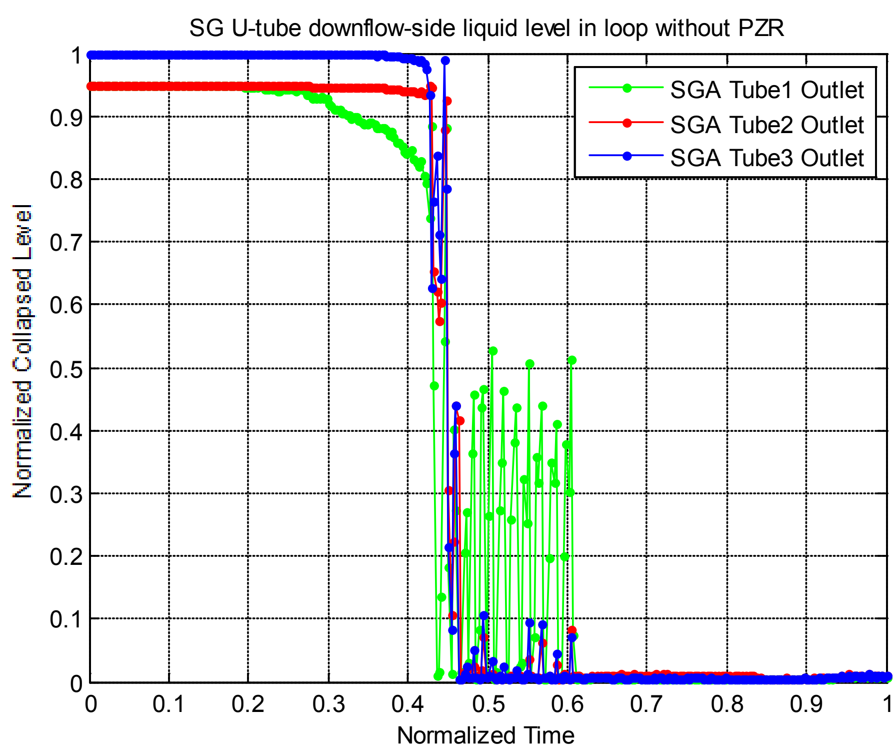

Figure 32 Steam generator B. U-tube downflow side liquid level.

6 CONCLUSIONS

This report contains results obtained in the simulation of the OECD/NEA ROSA Project Test 3-2 with the code TRACE5. One of the goals of the work is to investigate the effectiveness of the high-power natural circulation due to failure of scram during a loss-of-feedwater, assuming a total failure of high pressure injection system but actuation of the Auxiliary Feedwater system, using experimental data from the integral test facility LSTF together with TRACE5 code analyses.

Results show that TRACE5 can reproduce complicated conditions of natural circulation, when a break flow through a PORV valve is produced. TRACE5 adequately predicts the coolant distribution in primary and secondary circuits. However, oscillations in U-tubes reproduced with TRACE are smoothed. During the time period of liquid oscillation in U-tubes, heat transfer is not properly done by TRACE5.

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10. SUPPLEMENTARY NOTES

A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The purpose of this work is to provide an overview of the results obtained in the simulation of a pressurized water reactor (PWR) high-power natural circulation due to failure of scram following a Loss-Of-Feed Water (LOFW) transient under the assumption of total failure of High Pressure Injection System (HPIS), but actuation of Auxiliary Feed Water (AFW). Simulation of the experiment conducted in the Large Scale Test Facility (LSTF) is performed via the thermal-hydraulic code TRACE5. This work is developed in the frame of OECD/NEA ROSA Project Test 3-2 (TR-LF-13 in JAEA). AFW was actuated when the steam generator (SG) secondary-side collapsed liquid level decreased to a determined value, providing a continuous primary-to-secondary heat removal. The primary and secondary pressures were maintained almost constant by cycle opening of pressurizer Power-Operated Relief Valve (PORV) and SG relief valves till the end of the test.

A detailed model has been developed with TRACE5 following these assumptions. Results of the simulation are compared with the experimental in several graphs, observing an acceptable general behaviour in the entire transient. In conclusion, this work represents a good contribution for assessment of the predictability of thermalhydraulic computer codes such as TRACE5.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)* Consejo de Seguridad Nuclear (CSN) Thermal-hydraulic CAMP-Spain program TRAC/RELAP Advanced Computational Engine (TRACE) code Loss-Of-Feed Water (LOFW) Large Scale Test Facility (LSTF) OECD/NEA ROSA Project Test 3-1 (SB-CL-38 in JAEA) Power-Operated Relief Valve (PORV) Pressurized water reactor (PWR)	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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