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Assessment of TRACE 5.0 against ROSA Test 6-1, Vessel Upper Head SBLOCA

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ABSTRACT

The purpose of this work is to provide an overview of the results obtained in the simulation of a pressure vessel upper head Small Break Loss-Of-Coolant Accident (SBLOCA) under the assumption of total failure of High Pressure Injection System (HPIS) 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 6-1 (SB-PV-9 in JAEA). Test 6-1 simulated a PWR pressure vessel upper-head SBLOCA with a break size equivalent to 1.9% of the cold leg break. The break size and core uncover caused the primary depressurization. When the accident management (AM) action was initiated by fully opening the steam generator relief valves (detection of high core exit temperature of 623 K) primary pressure was lower than the steam generator secondary-side pressure. A detailed model has been developed following these assumptions.

Results of the simulation with TRACE5 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 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 codes1, 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/FAP2 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

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ABBREVIATIONS

AFW	Auxiliary Feed Water
AM	Accident Management
CAMP	Code Assessment and Management Program
CCFL	Counter-current Flow Limiting
CPU	Central Processing Unit
CSN	Consejo de Seguridad Nuclear (Spanish nuclear regulatory commission)
ECCS	Emergency Core Cooling System
FW	Feedwater
HPI	High Pressure Injection
JAEA	Japan Atomic Energy Agency
kg	kilogram(s)
LSTF	Large Scale Test Facility
l/s	liter per second
m	meter(s)
mm	millimeter(s)
MPa	megapascal
kg/cm^2	kilogram per square centimeter
°C	degrees Celsius
°K	degrees Kelvin
MSIV	main steam isolation valve
MW	megawatt(s)
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
PV	Pressure Vessel
PWR	Pressurized Water Reactor
PZR	Pressurizer
RV	Relief Valve
s	second(s)
SBLOCA	Small Break Loss-Of-Coolant Accident
SG	Steam Generator
SNAP	Symbolic Nuclear Analysis Package
TRACE	TRAC/RELAP Advanced Computational Engine

1 INTRODUCTION

The Loss-Of-Coolant Accident (LOCA) is one of the most important design basis accidents. Many works with the goal of improving safety methodologies, protocols and safety-related operating limits for a plant, can be found in literature [1-8], together with works focused on testing thermalhydraulic codes [9-13]. In the last years, there has been a significant interest in the development of codes and methodologies for "best-estimate" analysis of LOCAs In this frame, improvement of operating efficiencies in Small Break Loss-Of-Coolant Accidents (SBLOCA) is nowadays a concern.

The aim of the present work is to describe the main results achieved by the authors using the thermal-hydraulic code TRACE5 [9, 10], in the frame of OECD/NEA ROSA Project Test 6-1 (SB-PV-9 in JAEA) [14] with the purpose of testing the behavior of the code for this transient. The experiment 6-1 of the OECD/NEA ROSA (SB-PV-9 in JAEA) project was managed during 17th of December 2005 in the Large Scale Test Facility (LSTF) [15] of the Japanese Atomic Energy Agency (JAEA). 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 an SBLOCA in the upper head of the vessel of a Pressurize Water Reactor (PWR) of four loops. The size of the break is equivalent to 1.9% of the diameter of the cold leg.

2 ROSA FACILITY DESCRIPTION

In this section, a brief description of the LSTF facility (in the Tokai Research Establishment of the JAERI) is presented. The primary coolant system consists of the pressure vessel (PV), the primary loop A with the pressurizer (PZR) and the symmetrical primary loop B. Each loop contains a primary coolant pump (PC) and a steam generator (SG). The secondary-coolant system consists of the jet condenser (JC), the feedwater pump (PF), the auxiliary feedwater pumps (PA) and related piping system in addition to two SG secondary systems. The ECCSs consist of 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). The nitrogen gas is supplied to the accumulator tanks and some gas injection locations. The coolant discharged from the primary system is stored in the break flow storage tank (ST). The pressure vessel (PV) 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 and is sufficiently capable to simulate PWR decay heat power after the reactor scram.

Regarding to the steam generators, each of them contains 141 U-tubes which can be classified in different groups 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). U-tubes have an inner diameter of 19.6 mm and an outer diameter of 25.4 mm (with 2.9 mm wall thickness). As a consequence, the total inner and outer surface areas are 171 and 222 m^2, respectively. On the other hand, vessel, plenum and riser of steam generators have an inner height of 19.840, 1.183 and 17.827 m, respectively. The downcomer is 14.101 m in height.

3 TRANSIENT DESCRIPTION

The control logic of the transient is listed in Table 1. The experiment was initiated by quickly opening the break valve (inner diameter of 13.8 mm), at time zero. Simultaneously, rotational speed of primary coolant pumps was increased up to 1500 rpm. A scram signal was generated when the pressurizer pressure dropped to a determined value. This signal produces the initiation of the core power decay curve, calculated by considering the stored heat in the fuel rods and the 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).

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

Break	Time zero
Generation of scram signal	Primary pressure drops to a determined value
Pressurizer (PZR) heater off	Generation of scram signal or PZR liquid level below a
Initiation of core power decay curve simulation	Generation of scram signal
Initiation of primary coolant pump coastdown	Generation of scram signal
Turbine trip (closure of stop valve)	Generation of scram signal
Closure of main steam isolation valve	Generation of scram signal
Termination of main feedwater	Generation of scram signal
Generation of safety injection (SI) signal	Determined value of primary pressure
Initiation of auxiliary feedwater	Generation of SI signal
Initiation of steam generator (SG) secondary-side depressurization as accident management (AM) action by fully opening relief valves	Core exit temperature reaches determined maximum
Initiation of accumulator system	Determined value of primary pressure
Initiation of low pressure injection system	Determined value of PV lower plenum pressure

Table 2 Predetermined core power decay curve

Normalized Time	Normalized Power	Normalized Time	Normalized Power	Normalized Time	Normalized Power
0	1	0.0267	0.3042	0.2	0.1832
0.006	1	0.0333	0.2763	0.266	0.1577
0.0067	0.8150	0.05	0.2423	0.333	0.1487
0.01	0.5366	0.0667	0.2263	0.5	0.1342
0.0133	0.4504	0.1	0.2079	0.666	0.1238
0.0167	0.3906	0.133	0.2000	1	0.1096
0.02	0.3538	0.166	0.1913		

At the same time, the primary coolant pump coastdown is initiated, also using a pre-determined rotational speed curve (Table 3).

Table 3 Pumps relative rotational speed

Normalized Time	Relative rotational speed	Normalized Time	Relative rotational speed	Normalized Time	Relative rotational speed
0	1.000	0.01	0.280	0.026	0.125
0.00066	0.850	0.013	0.220	0.03	0.110
0.0017	0.730	0.016	0.185	0.033	0.100
0.0033	0.540	0.02	0.160	0.083	0.000
0.0067	0.370	0.023	0.140	--	---

A turbine trip is activated by closing the steam generators main steam isolation valves (MSIVs). The closure of MSIVs produces an increasing of SG secondary-side pressure and a temporary rise in the primary pressure, followed by a new decrease due to the core power decay effect. Simultaneously, main feedwater flow of both SGs is stopped. The safety injection signal (SI) is generated when the primary pressure decreases to a determined value. From this moment on, the relief valves (RV) in both steam generators, begin opening and closing in order to maintain the pressure between two fixed values. When the core exit temperature reaches a determined value, the accident management (AM) action is initiated by fully opening relief valves of both steam generators. The core power is automatically decreased by the core protection system when the maximum fuel rod surface temperature exceeds a certain maximum, 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%	958
50%	968
25%	969
10%	970
0%	973

When primary pressure drops to a determined value, the accumulation system starts to inject water in cold legs. In the last part of the experiment, nitrogen gas is injected in both cold legs, throughout the accumulators. Finally, when primary pressure drops to a determined value, the Low Pressure Injection System (LPIS) is initiated.

4 APPLIED METHOD: TRACE5 MODEL OF ROSA FACILITY

TRACE5 code is designed to perform best-estimate analyses of loss-of-coolant accidents (LOCAs) or operational transients. However, it has some limitations in use, for example, model cannot be applied directly to those transients in which one expects to observe thermal stratification of the liquid phase in the 1D components. Furthermore, it is not appropriate for modeling situations in which transfer of momentum plays an important role at a localized level. 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) [16].

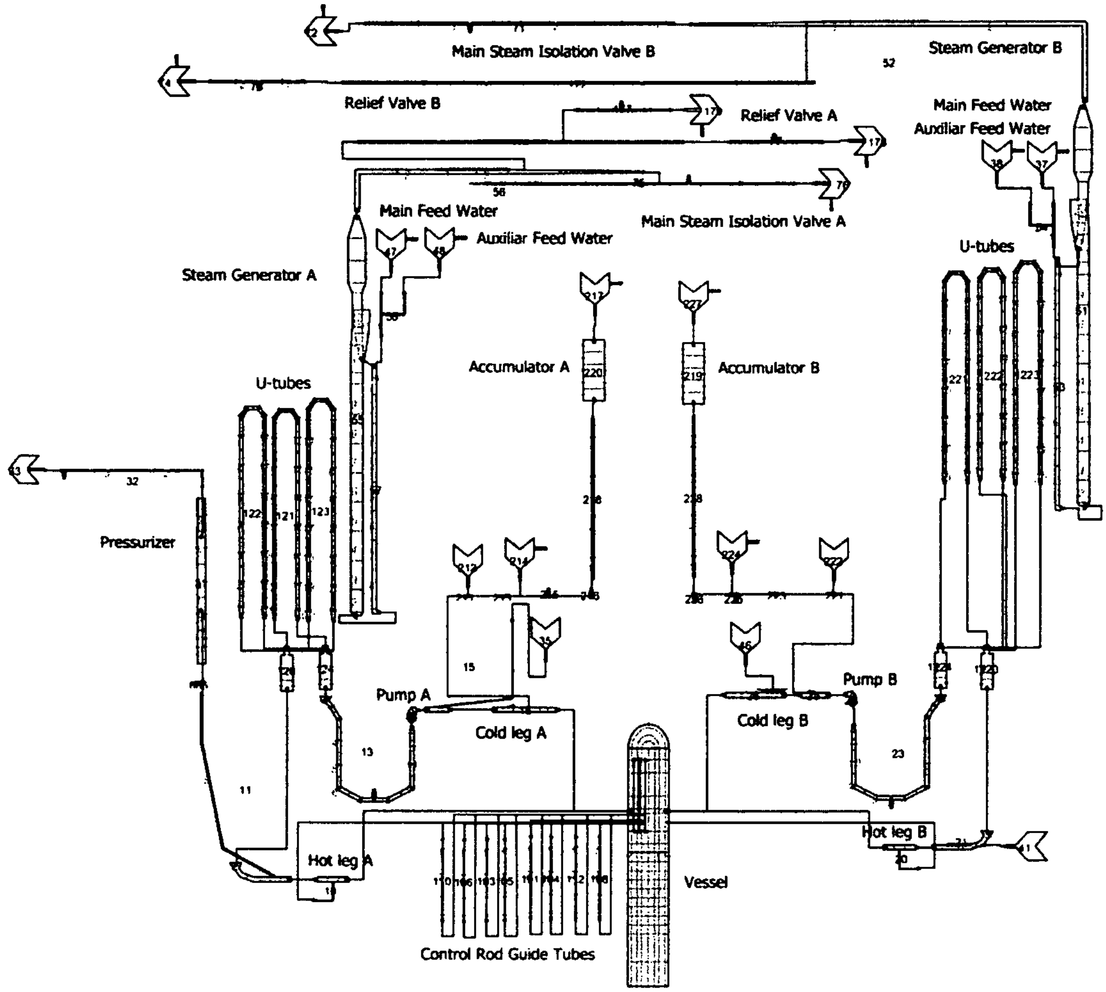

Figure 1 Model nodalization used for simulation

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 [14]. 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. In level 16, the upper core support plate is located. 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.

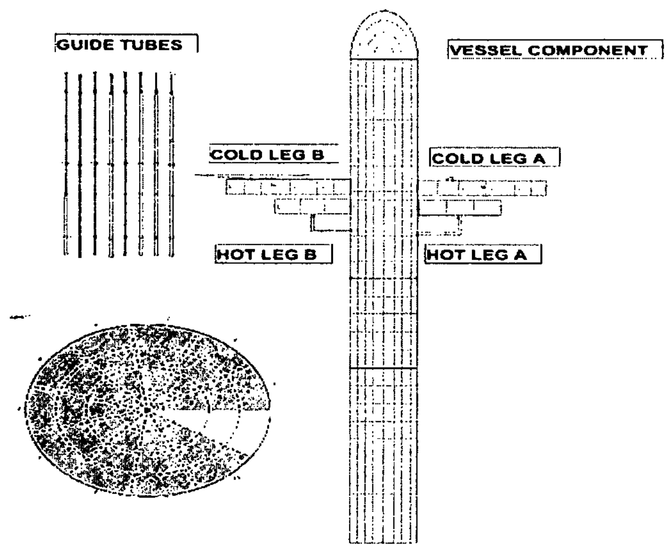

Figure 2 3D Vessel nodalization and connections visualized with SNAP

30 HTSTRs simulate the fuel assemblies in the active core. A POWER component 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 it can be seen in Table 5.

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 modelled 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 1022 at a determined cell edge, allowing only to flow through the cell interface gas phase. 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]. Inner and outer surface boundary conditions for each axial level, has 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.

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

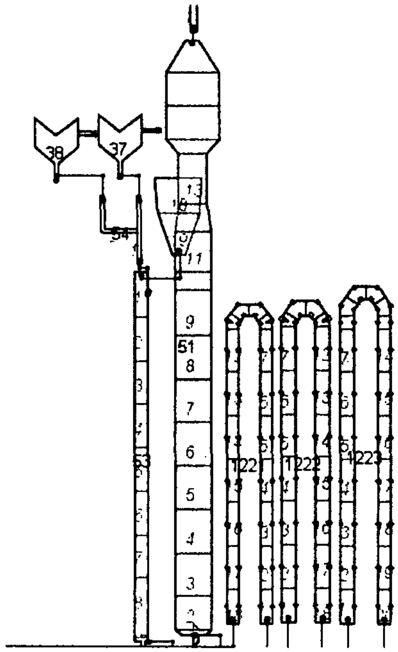

Figure 3 Steam generator nodalization

Regarding to the break simulation, it is important to take into account the necessity of activating the Choke flow model in the break when critical flow conditions are expected to appear. Choke model predicts for a given cell the conditions for which choked flow is expected to occur, providing three different models in one: subcooled-liquid, two-phase and single-phase vapor model. The break has been simulated by means of a VALVE component connected to a BREAK component in order to establish the boundary conditions. This BREAK has been modelled following the recommendations of the TRACE5 user's manual [9]. In this case, since the break is simulated to discharge in a big volume space (the storage tank), a dxin=1.0*10^{-6} (length) and a volin=1.0*10^{6} (volume) has been selected with the purpose of providing a large area.

5 RESULTS AND DISCUSSION

Steady-state conditions achieved in the simulation were in reasonable agreement with the experimental values, as it can be seen in Table 6.

Table 6 Steady state conditions. Comparison between experiment and TRACE5

Item	Relative Error (%) (loop without PZR)
Pressure vessel	
Core power (MW)	0.00
Primary loop	
Hot leg fluid temperature (K)	0.20
Cold leg fluid temperature (K)	0.07
Mass flow rate (kg/s / loop)	2.80
"downcomer"-to-hot leg bypass (kg/s)	2.00
Pressurizer	
Pressure (MPa)	0.50
Liquid level (m)	4.10
Accumulator system	
Pressure (MPa)	0.20
Temperature (K)	0.50
Steam generators	
Secondary-side pressure (MPa)	0.70
Secondary-side liquid level (m)	5.90
Steam flow rate (kg/s)	4.10
Main feedwater flow rate (kg/s)	0.70
Main feedwater temperature (K)	0.10
Auxiliary feedwater temp (K)	0.06

Table 7 lists the chronology sequence of events during the transient and the comparison in Normalized Time between the experiment and TRACE5 results.

Table 7 Chronological sequence of events. Comparison between experiment and TRACE5

Event	Experiment Normalized Time	TRACE5 Normalized Time
Valve open	0.0	0.0
Scram signal (determined value of primary pressure)	0.007	0.006
S.I. signal (determined value of primary pressure)	0.008	0.01
Phase change (one phase liquid to two-phase) in the break	0.01	0.02
Pumps stop	0.09	0.09
Phase change (two-phase to one phase vapor) in the break	0.24	0.23
Primary pressure lower than in the secondary	0.27	0.25
Initiation of secondary depressurization (determined exit temperature of the "upper plenum")	0.37	0.37
Initiation of core protection system (determined vessel temperature)	0.41	0.43
Accumulators start (determined value of primary pressure)	0.44	0.44
Initiation of LPIs (determined value of lower plenum pressure)	1	0.93

Variables presented in this section follow the requirements for an exhaustive analysis of the transient. The most important parameters that will be studied in this paper 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 liquid level in the accumulators.

5.1 System pressures

Figures 4 and 5 compare the primary and secondary pressures. The primary pressure begins to decrease at time zero (when the break is produced). In the experiment, the scram signal is generated at 0.007 NT after the break, when the primary pressure decreases to a determined value. The generation of the scram signal causes the main steam valve (MSIV) of steam generators to close and the beginning of the primary coolant pumps coastdown. The SI signal is generated at 0.008 NT when the primary pressure decreases to the determined value. The secondary pressure rapidly increases after the closure of the MSIVs. From this moment on, the secondary pressure starts to oscillate by means of opening and closing the relief valves (RV) of steam generators until 0.37 NT, when the depressurization of steam generators secondary-side begins.

The relatively big size of the break results in a rapid depressurization of the primary side, especially until 0.24 NT when the break flow changes to single-phase steam. The pressure in the primary decreases below the pressure of the secondary in the steam generator at 0.27 NT, almost simultaneously to the core uncover. Anyway, according to the experiment report [14], the depressurization of the secondary is not effective until 0.27 NT, when the pressure of the primary becomes lower than the pressure of the secondary.

The first part of the transient (between 0 and 0.24 NT) shows a perfect agreement between experimental and simulated values. Discrepancies appear in the second part of transient, when two-phase break flow turns to single-phase steam flow (Figures 4 and 5). Deviation between both primary pressures might be due to an improper treatment of the steam critical flow at the break by TRACE5, as it will be explained in the following section. The accumulator system is initiated at about 0.44 NT when the primary pressure decreases to a determined value, producing an increasing in the primary pressure of experimental values. The entrance of water coming from the accumulators produces a peak of pressure, also coincident with the refill of the pressure vessel. This effect is not registered in the simulation. The coolant injection from the accumulator system is finished when the primary pressure decreases to the determined value, followed by the discharge of nitrogen gas from the accumulator tanks.

TRACE5 adequately reproduces the liquid level decrease of accumulators (as it will be shown in other section) and the entrance of nitrogen gas. Once accumulators are empty of water, both primary pressures of experimental and TRACE5 become almost the same. When vessel lower plenum pressure reaches the determined value, the low pressure injection (LPI) system actuates at about 1 NT in the experiment, in agreement with TRACE5, which registers this event slightly before.

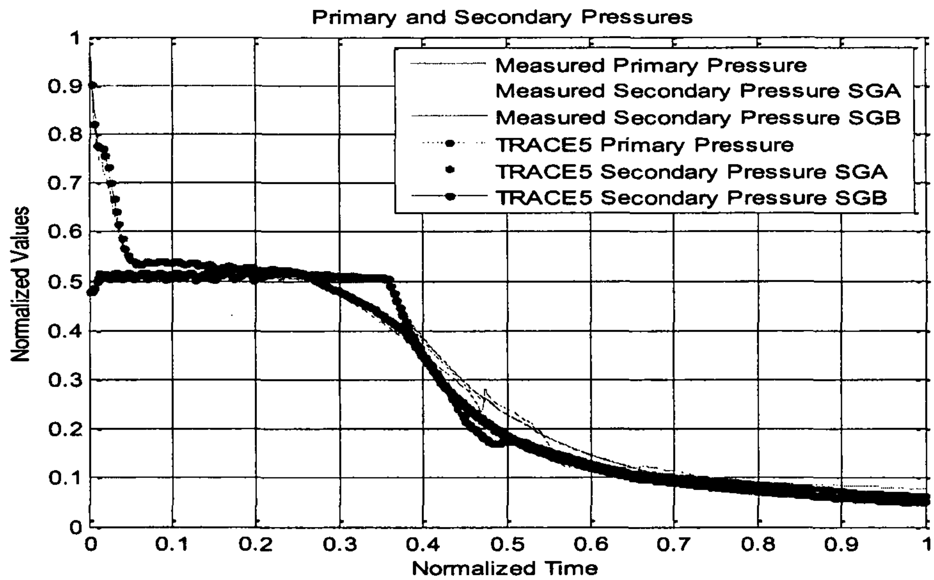

Figure 4 Primary and secondary pressures (0 to 1 NT)

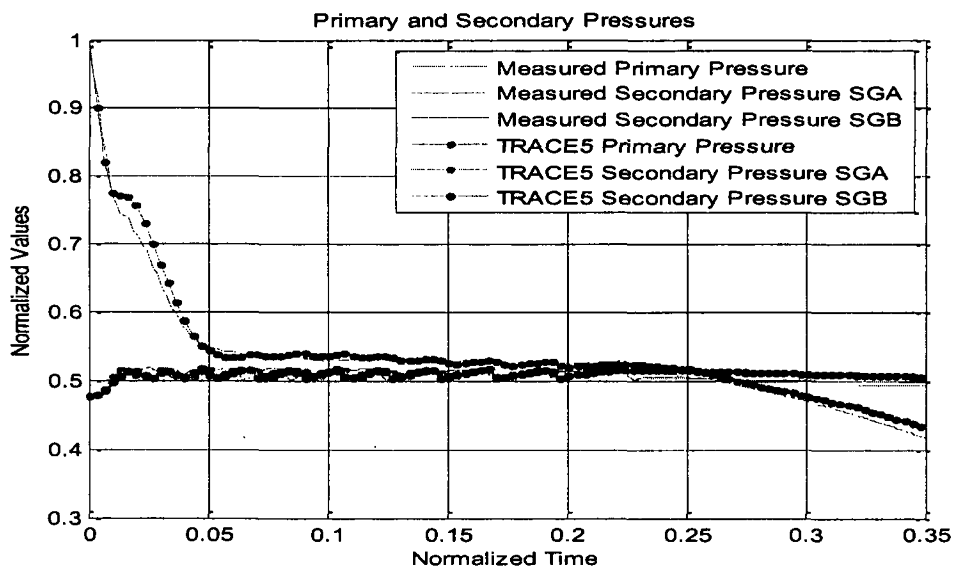

Figure 5 Primary and secondary pressures (0 to 0.35 NT)

5.2 Break

Break flow rate is shown in Figure 6. Figure 7 shows the discharged inventory. The primary pressure and the inventory decrease sharply until the saturation pressure of the coolant is reached. Then, the pressure slowly decreases as the break flow changes to saturated two-phase flow. TRACE5 adequately reproduces the flow through the break, and so the discharged inventory in both cases (single and two phases). From 0.25 NT on, the upper plenum begins to empty and so the two-phase flow through the guide tubes connecting the upper plenum and the upper head ends. Thereafter, the code overpredictes the break mass flow. This fact can be seen in Figure 7 for time up to 0.24 NT. TRACE5 allows to apply Choke flow [9] conditions and use discharge coefficients (one liquid single-phase and two-phase coefficients).

Between 0.25 and 0.4 NT TRACE predicts single-phase vapour, but during this period the total mass flow rate calculated through the break is slightly higher than the experimental measurement. In other time intervals (for example between 0.65 and 0.75 NT) a two-phase liquid-vapor mass flow is leaving through the break, but before this time period, the total discharged inventory was higher than the experimental measurement. This permits to conclude that a single vapour phase discharge coefficient is needed in the simulation. Other codes, such as RELAP, include the possibility to have into account this discharge coefficient for one-phase vapour.

On the other hand, to best simulate the break mass flow rate it is necessary to adequately model all the flowpaths reaching the break. In the simulation three flowpaths have been studied: control rod guide tubes connecting the upper head and the upper plenum, the spray nozzles between the upper plenum and the downcomer and the bypass between the upper plenum and the downcomer through the hot leg. The bypass between the upper plenum and the downcomer has been modeled by means of a TEE component. Friction and flow are of bypass has been adjusted during the steady state calculations in order to achieve a mass flow rate through the bypass similar to the obtained in the experiment. NFF factors have been set to -1 in the secondary side of this TEE component. Regarding to the control rod guide tubes, results that better fits experimental measurements are obtained adjusting the flow are of the control rod guide tubes to the actual flow area and considering NFF=-1. Finally, spray nozzles between upper plenum and upper head have been modeled taking into account the technical specifications provided by JAEA.

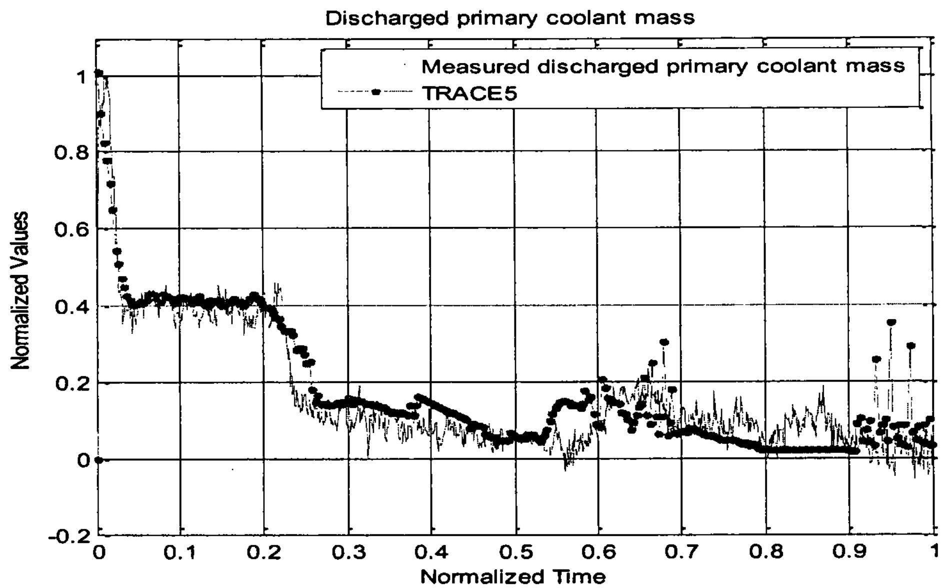

Figure 6 Break mass flow rate

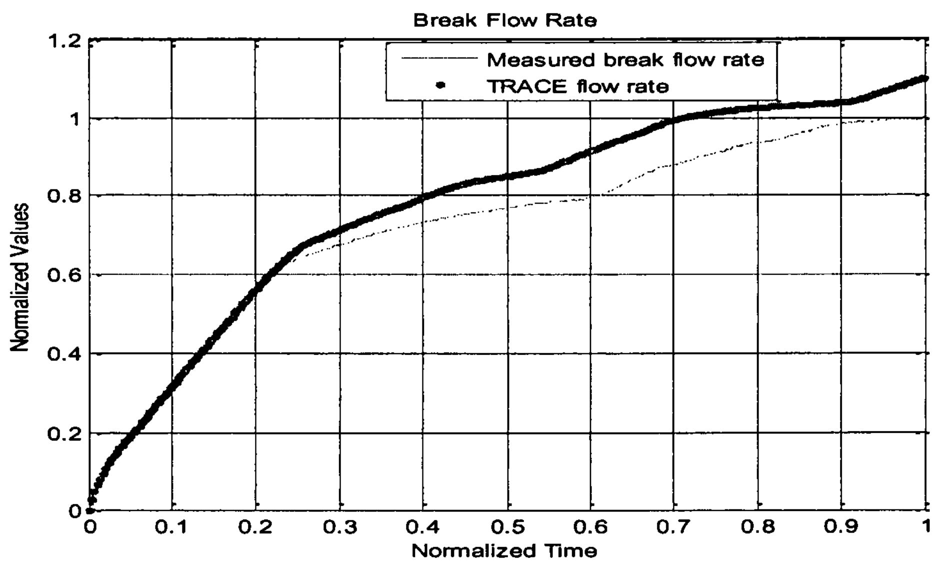

Figure 7 Discharged inventory through the break

5.3 Primary loops mass flow rates

In general, mass flow through both loops is adequately reproduced, as it can be seen in the following Figures. Figures 8 and 9 show the mass flow of loop A (measured in TRACE5 at pump position). In addition, Figures 10 and 11 show the loop B mass flow. Almost symmetric flow rates were observed among both loops during the entire transient. According to these results, TRACE5 successfully reproduces the natural flow circulation in primary loops under conditions of upper plenum SBLOCA.

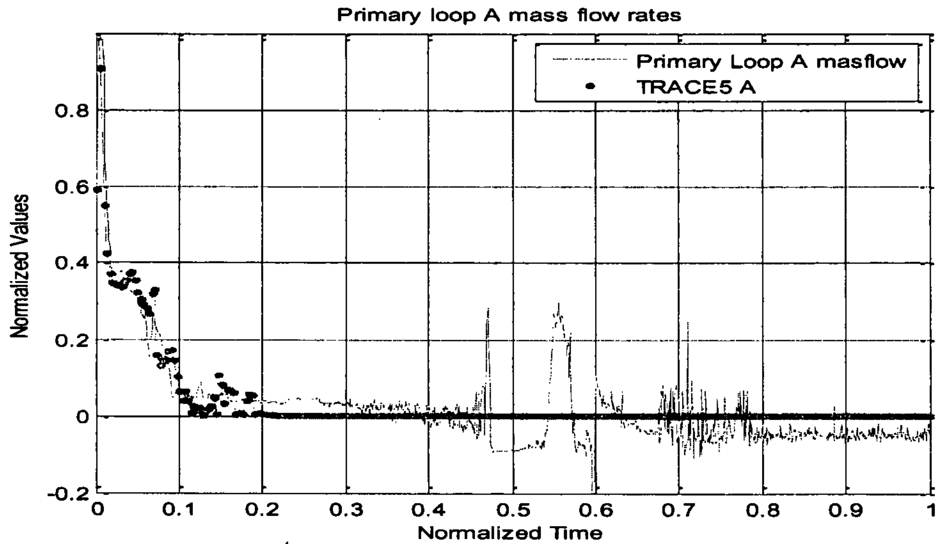

Figure 8 Primary loop A mass flow (0 to 1 NT)

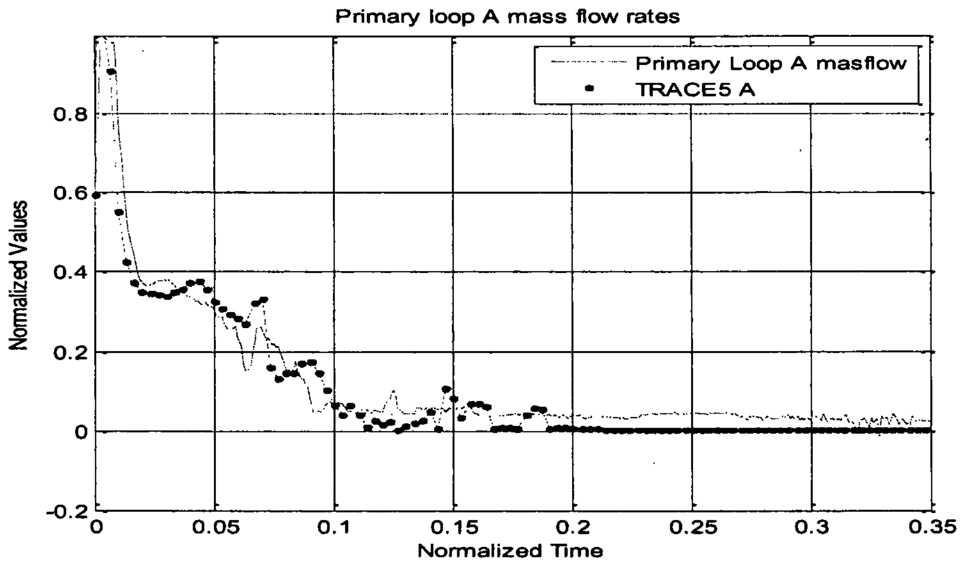

Figure 9 Primary loop A mass flow (0 to 0.35 NT)

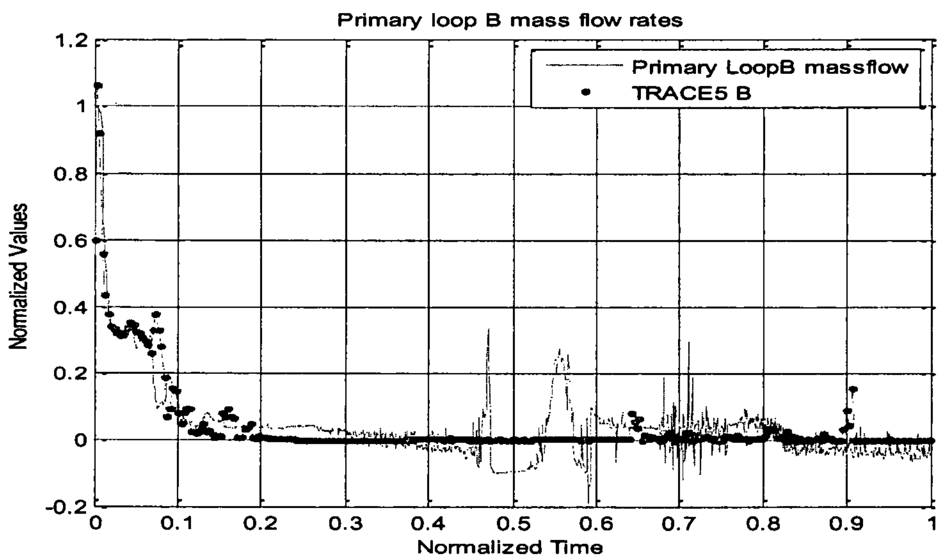

Figure 10 Primary loop B mass flow (0 to 1 NT)

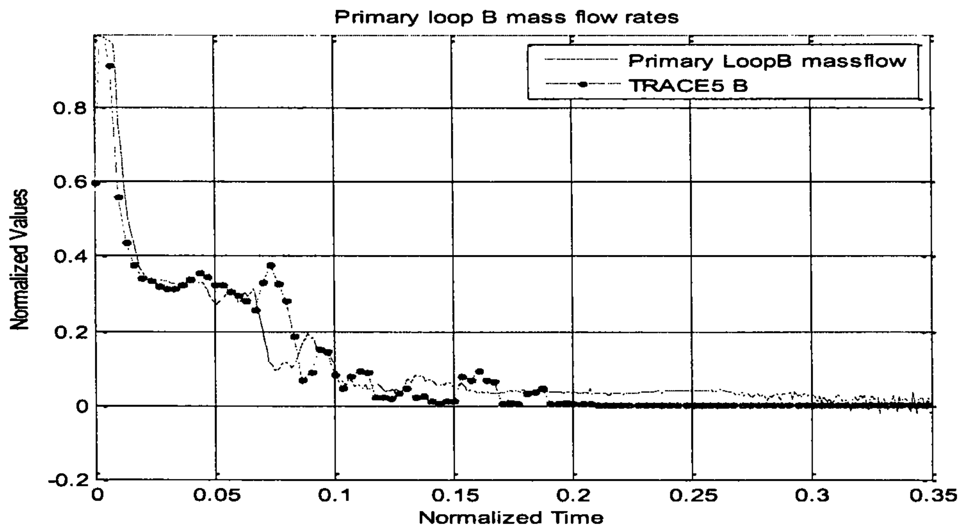

Figure 11 Primary loop B mass flow (0 to 0.35 NT)

5.4 Vessel collapsed liquid levels

The following Figures (12, 13 and 14) show a comparison between the collapsed liquid levels in the upper plenum, core and downcomer, respectively for both experimental and TRACE5 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 water level decreased due to a loss of the RCS coolant inventory through the break. Coolant in the upper plenum entered the upper-head through control rod guide tubes (CRGTs) until the entrance holes at the CRGTs bottom are exposed to steam. In Figure 12 it can be seen that the experimental upper plenum liquid level remains almost constant until 0.24 NT. TRACE5 reproduces this phenomenon delaying the entrance of steam approximately 0.03 NT (to 0.27 NT). Immediately after, core liquid level starts to decrease, producing the core uncover until 0.44 NT (Figure 13). Simultaneously with the core empty, the downcomer liquid level decreases, until the entrance of water coming from accumulators (0.48 NT, approximately) begins (Figure 14). In the experiment, the core and downcomer liquid levels indicate temporal manometric fluctuations during the coolant injection from the accumulator since the local pressure decreases in the cold legs. The liquid level decreases in the core and increases in the downcomer. This phenomenon is not reproduced with TRACE5.

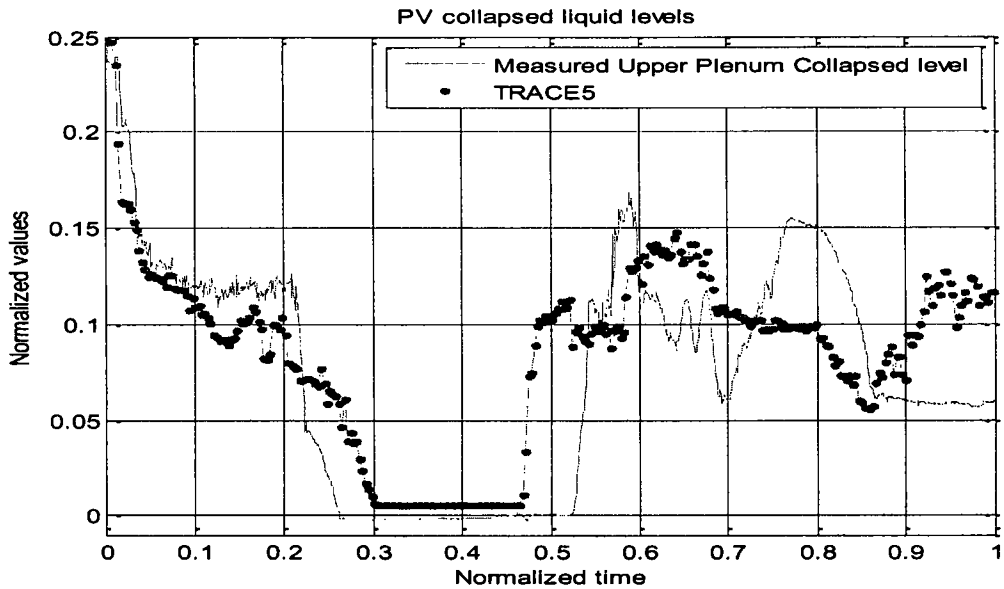

Figure 12 Upper plenum collapsed liquid level

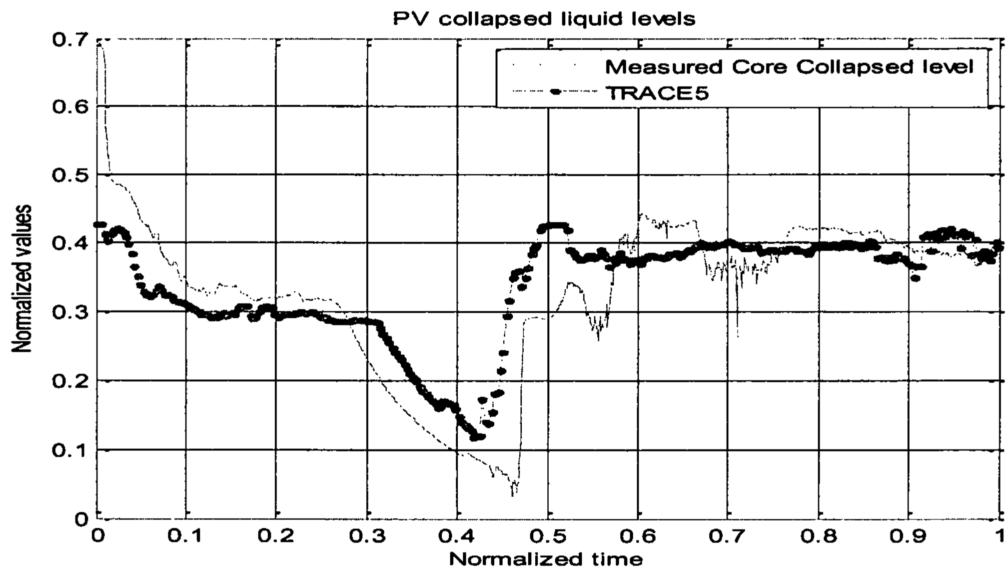

Figure 13 Core collapsed liquid level

The most important discrepancy is observed in the last stretch of the transient (between 0.68 and 1 NT) in the downcomer liquid level. In the model, a decrease in the liquid level occurs, which is not registered in the experiment. This phenomenon has a straight relationship with the mass discharged inventory.

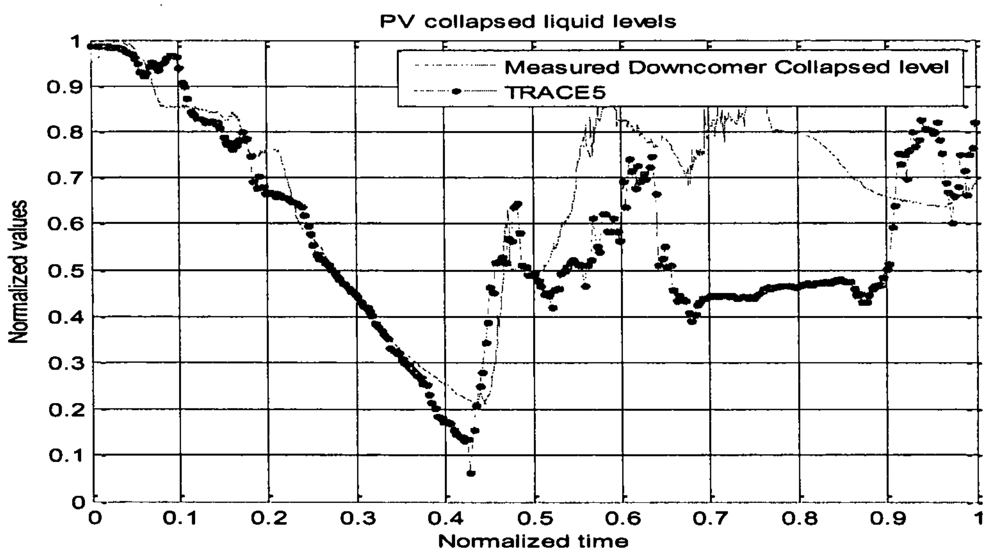

Figure 14 Downcomer collapsed liquid level

Upper head liquid level is shown in Figure 15. A good agreement can be seen between experimental and TRACE5 results. It is important to remark that mass flow rate is conditioned by the upper head liquid level as it can be observed in Figure 16.

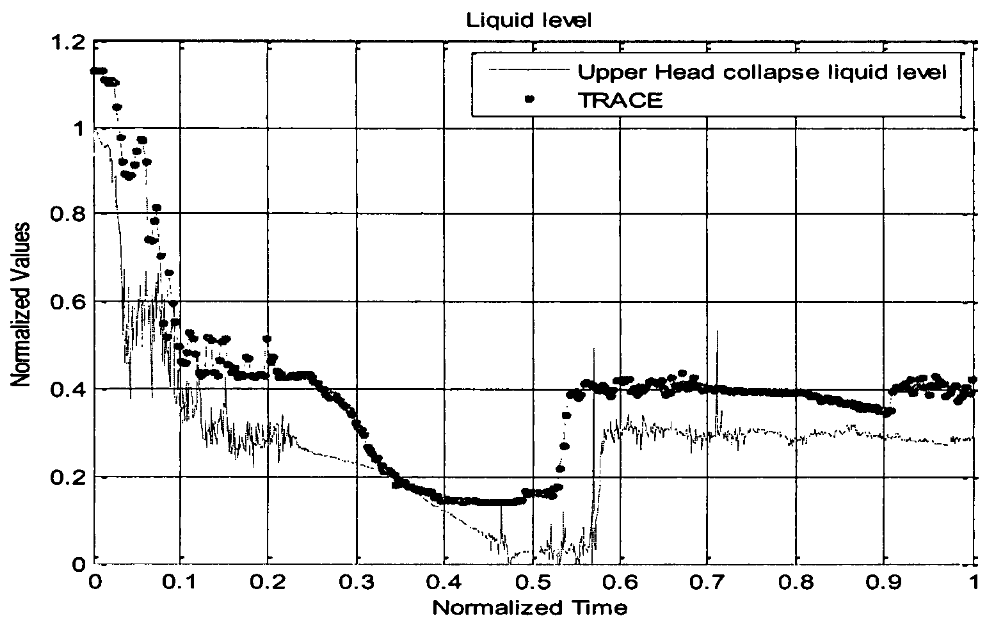

Figure 15 Upper head collapsed liquid level

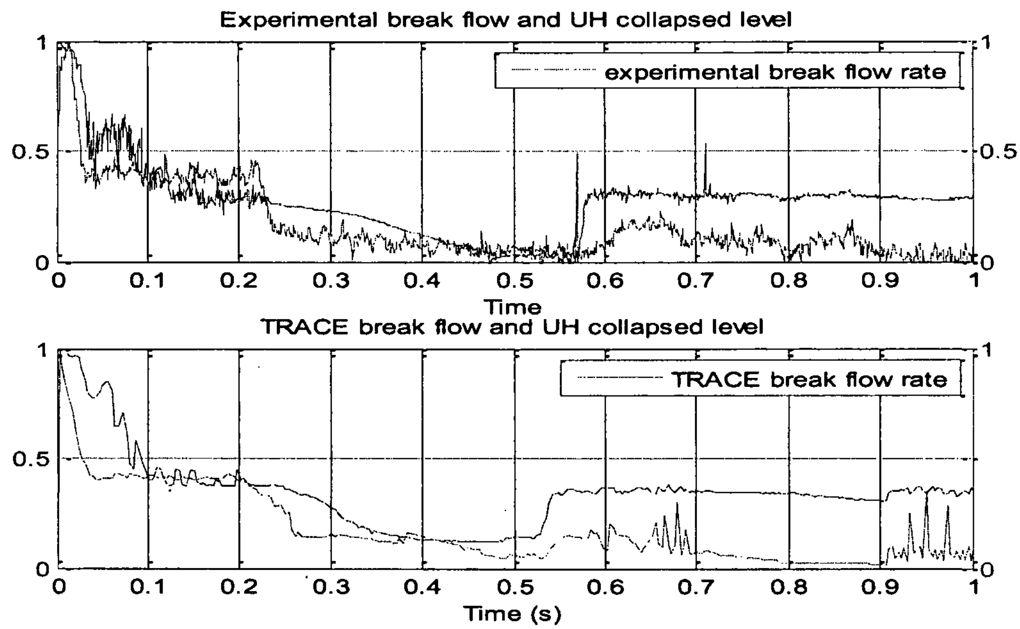

Figure 16 Relation between the Upper head collapsed liquid level and break mass flow rate. Experimental and TRACE5 results

5.5 Maximum fuel rod surface temperature

TRACE5 reproduces precisely the evolution of the maximum fuel rod temperature in the core. The maximum temperature is reached at 0.41 NT. At this moment, the core protection system activation is produced, reducing the core power according to a programmed decay power curve (Figure 18 and Table 4). After the activation of this system and the entrance of water from accumulators, TRACE5 fuel rod surface temperature is in good agreement with experimental values (Figure 17).

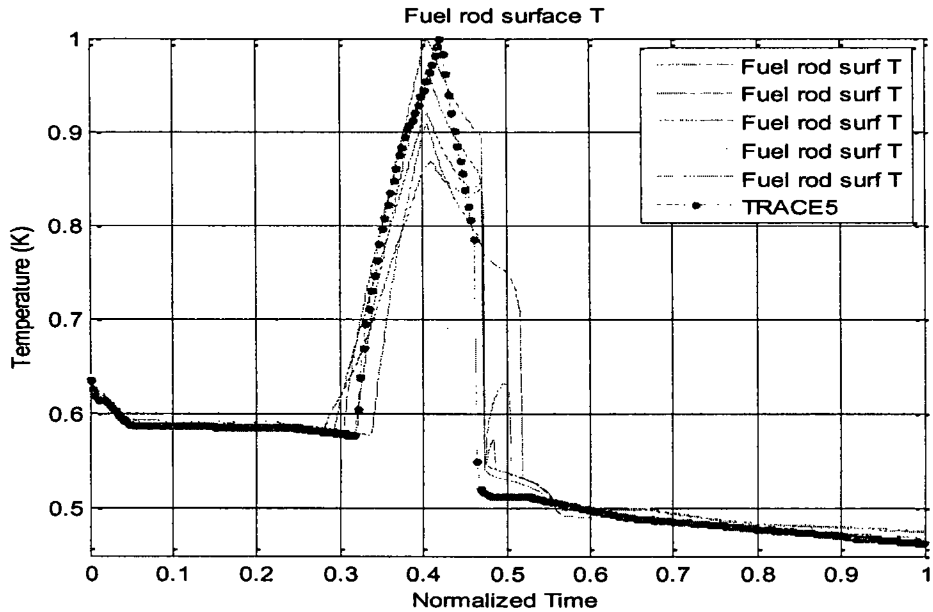

Figure 17 Maximum fuel rod surface temperature

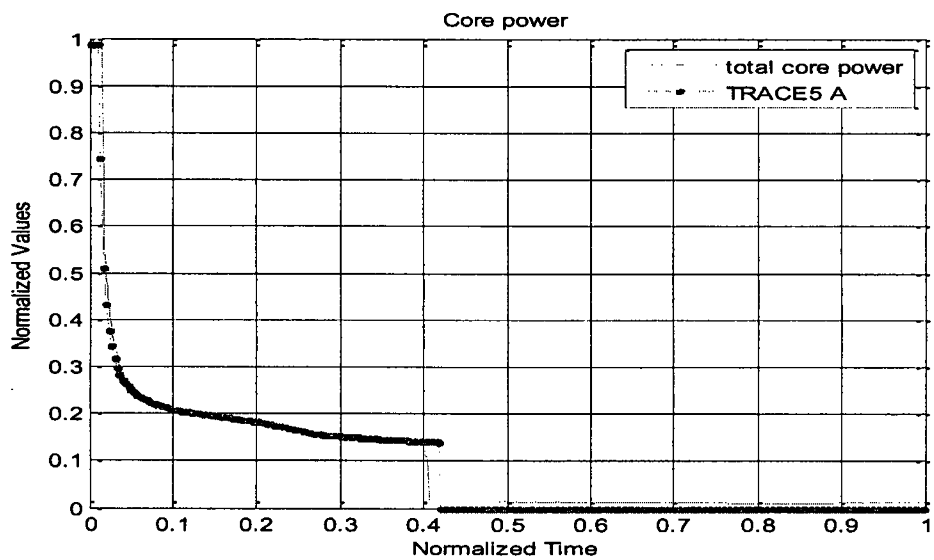

Figure 18 Core power

5.6 Maximum core exit temperature

Figure 19 shows a comparison between TRACE5 and the experimental core-exit fluid temperature. Experimentally, core exit temperatures are observed to keep the saturation temperature until 0.31, even after the beginning of the core uncover, partly due to the condensation produced in the hot leg. It is also shown that core exit temperatures depend on the radial position. In the experiment, the temperature peak occurs in the centre, while the exit temperatures of the middle ring are almost the same to those of the outer ring. The accident management action begins at 0.37 NT (experimentally and with TRACE5) immediately after the core exit temperature reaches a maximum. In TRACE5, thermocouples have been located in the level 13 of the vessel (in the upper plenum region).

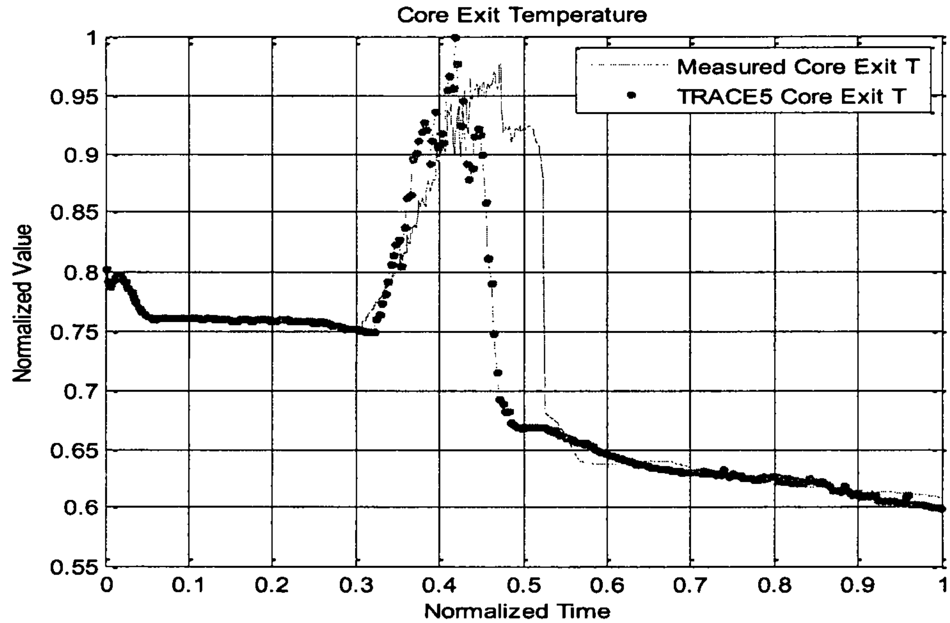

Figure 19 Maximum exit core fluid temperature

5.7 Hot and cold legs liquid levels

The following Figures (20, 21, 22 and 23) show the liquid level in hot and cold legs, respectively. Experimentally, liquid level was obtained with a three gamma ray beam densitometer. The hot leg liquid level is kept constant in a value until 0.24 NT, suddenly decreasing from this time on. Hot legs remain empty until the entrance of the liquid from the accumulators. Experimentally, this phenomenon takes place at 0.55 NT. Regarding the collapsed liquid level in cold legs, it is important to remark the good agreement between TRACE5 and the experiment, until the entrance of accumulators (0.44 NT, approximately).

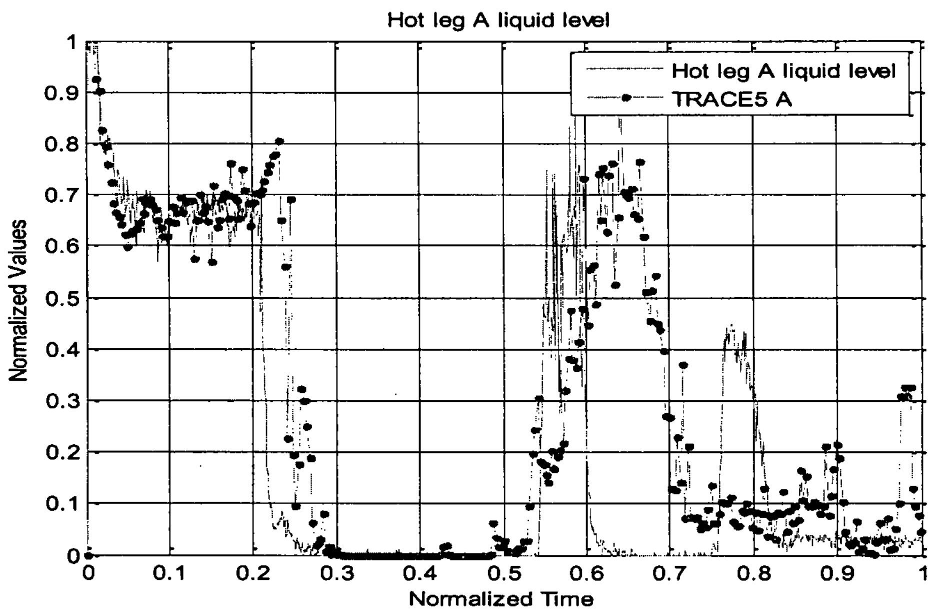

Figure 20 Collapsed liquid level in the hot leg A

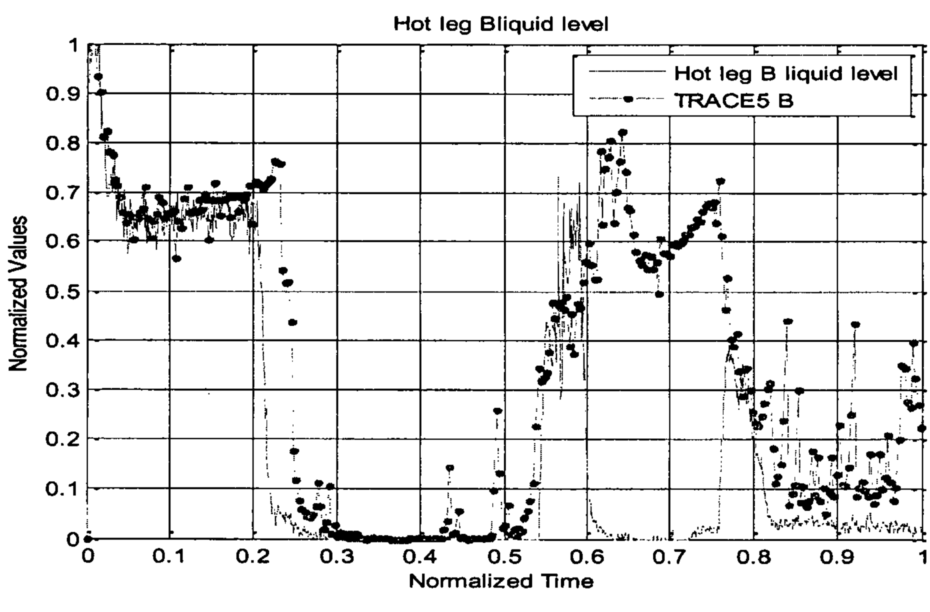

Figure 21 Collapsed liquid level in the hot leg B

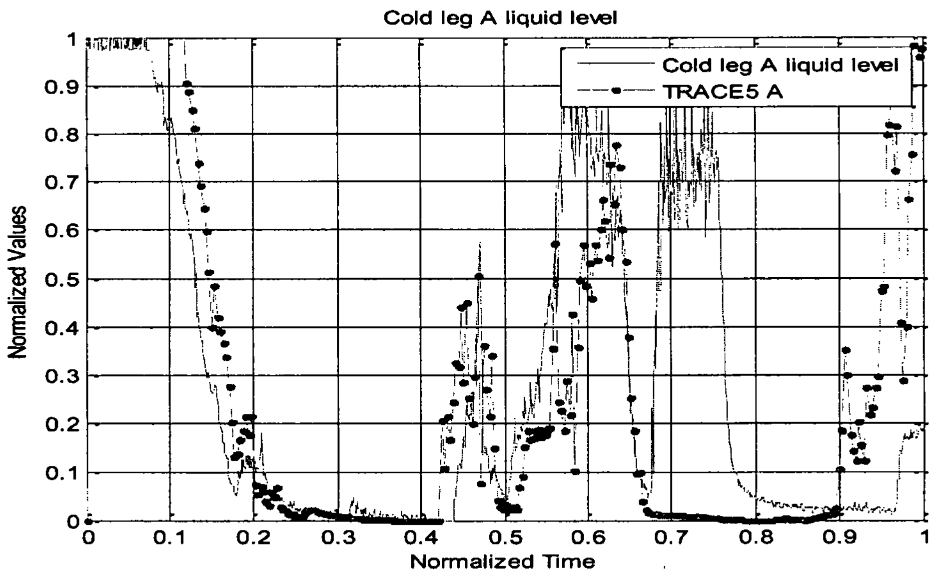

Figure 22 Collapsed liquid level in the cold leg A

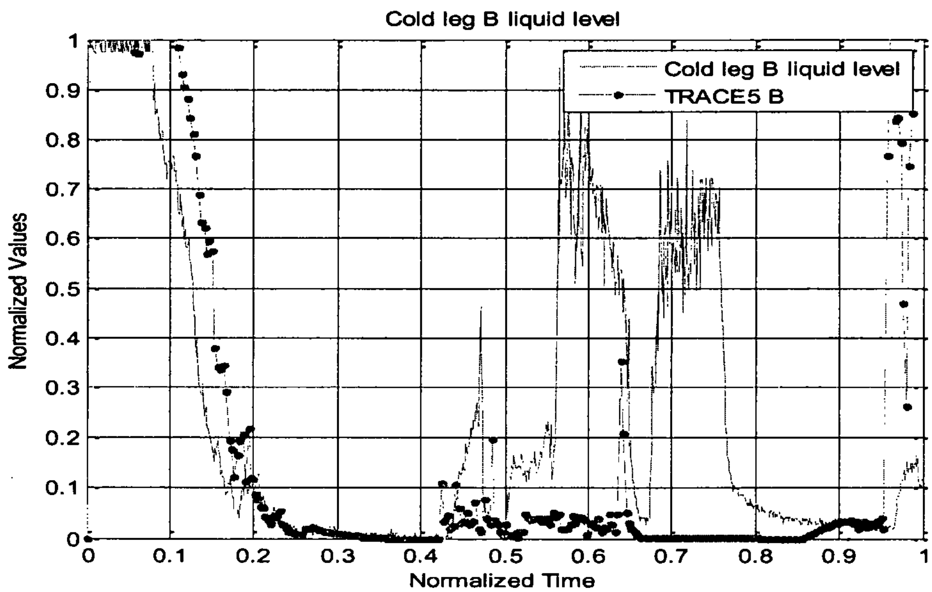

Figure 23 Collapsed liquid level in the cold leg B

5.8 Steam Generator relief valve flow rate

A good agreement has been achieved between TRACE5 and the experiment, as it can be observed in Figures 24 and 25. These Figures perfectly show the periods corresponding to the relief valves actuation and the continuous depressurization of secondary-sides.

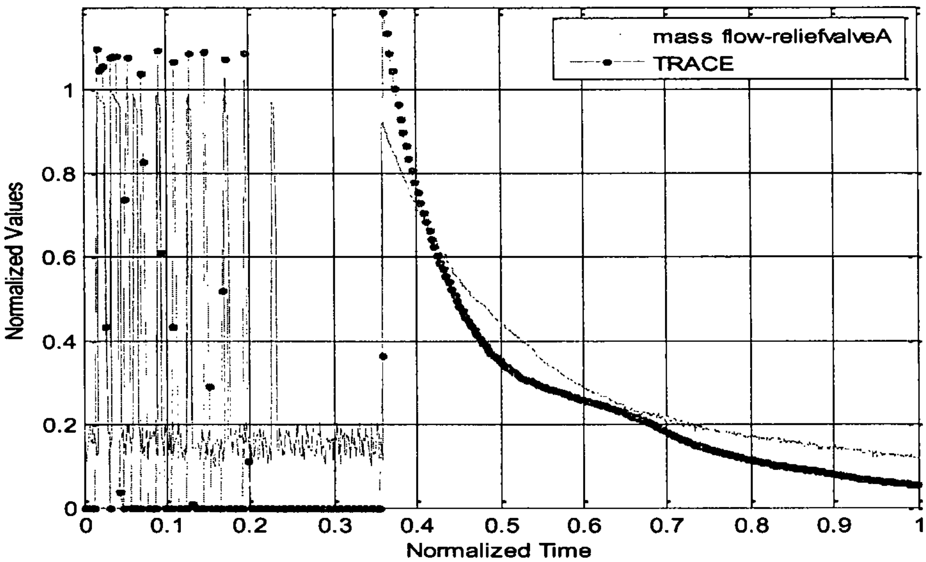

Figure 24 SG A relief valve mass flow

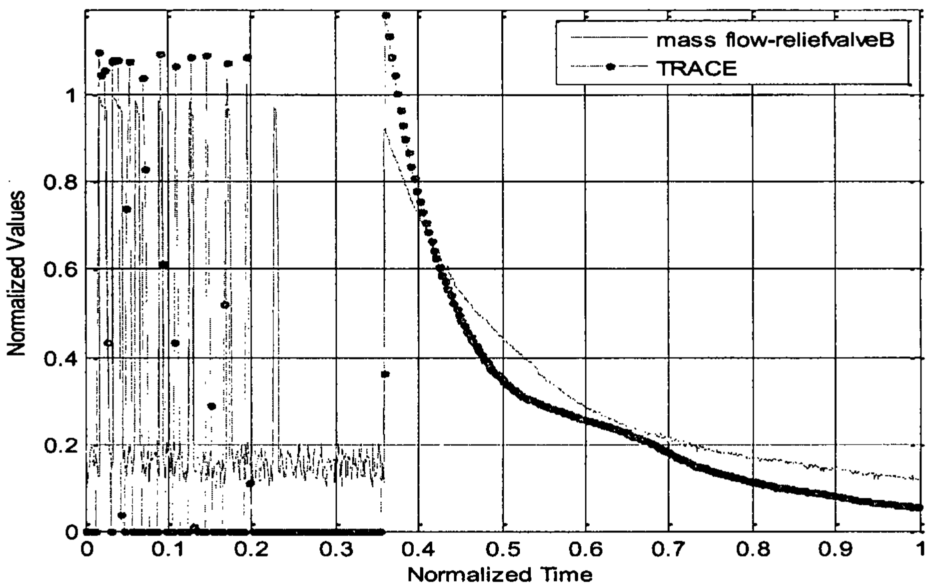

Figure 25 SG B relief valve mass flow

5.9 U-tubes collapsed liquid level

U-tubes have been grouped according to similar lengths. Experimentally six different types were considered. In the TRACE5 simulation and due to the calculation time cost together with the obtained results, a 3-group classification has been adopted. Collapsed liquid levels obtained with TRACE5 are satisfactory, properly reproducing the clearance of the tubes. In both experimental and TRACE5 results, collapsed liquid levels in up-flow and down-flow sides are balanced. The natural two-phase circulation ends about 0.17 NT before the U-tubes lose the collapsed liquid level in the up-flow side. Results are shown in Figures 26, 27, 28 and 29.

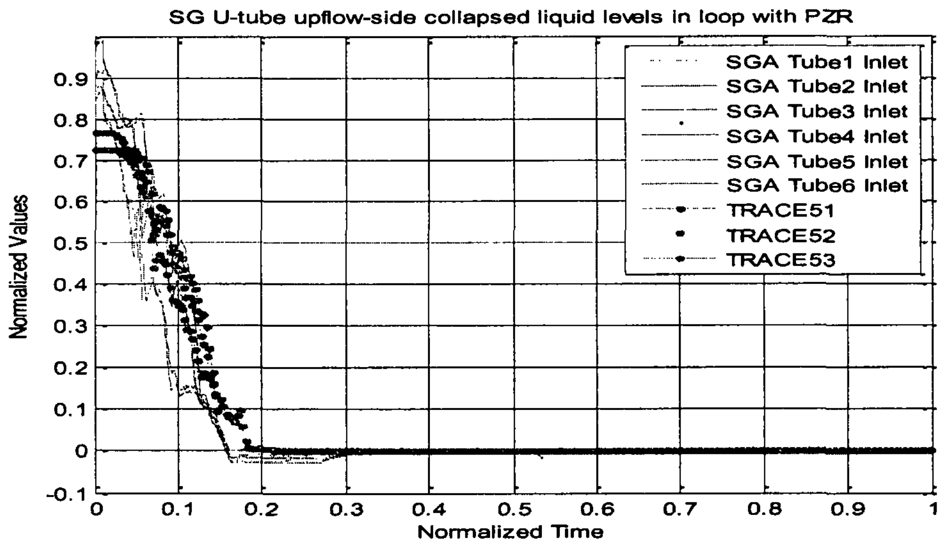

Figure 26 SG U-tube up-flow side collapsed liquid levels in loop with PZR

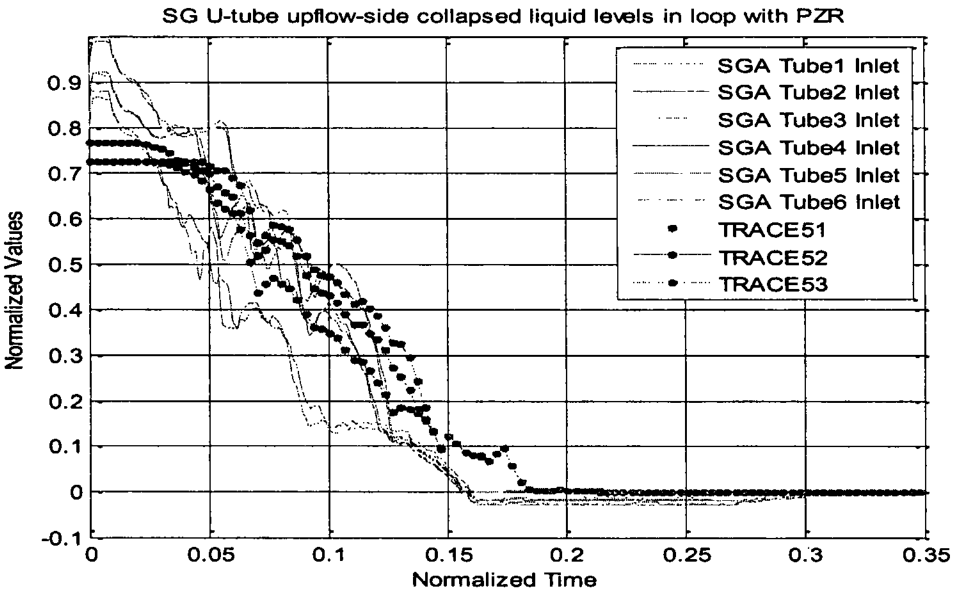

Figure 27 SG U-tube up-flow side collapsed liquid levels in loop with PZR

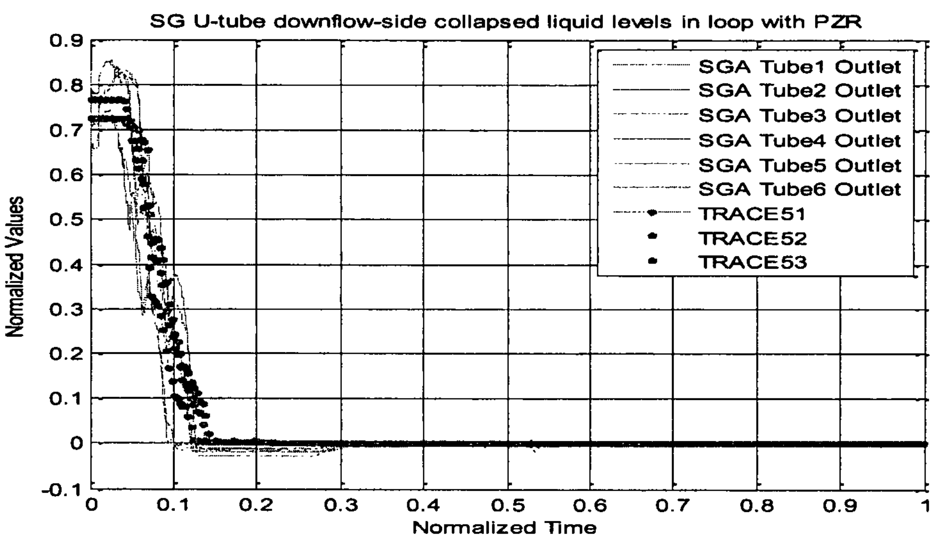

Figure 28 SG U-tube down-flow side collapsed liquid levels in loop with PZR

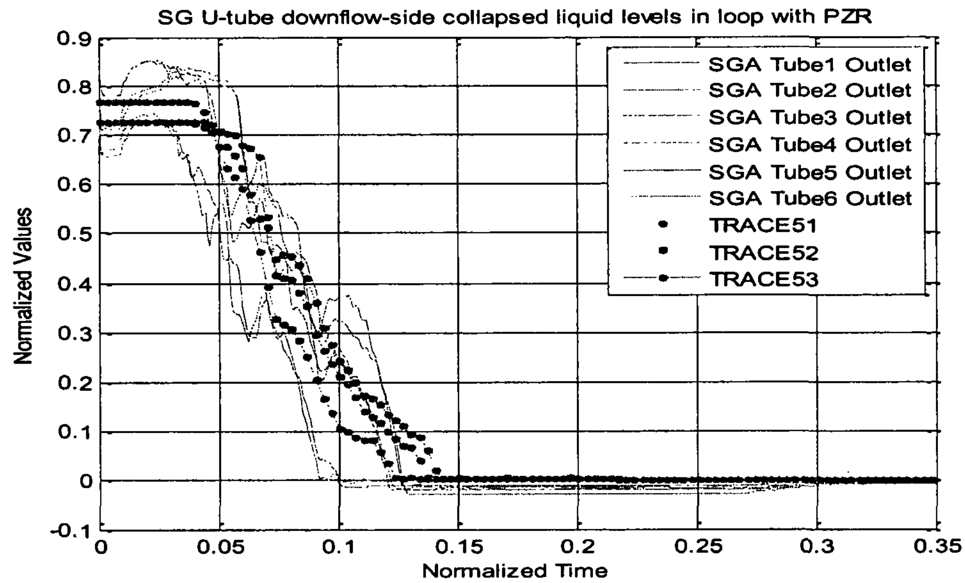

Figure 29 SG U-tube down-flow side collapsed liquid levels in loop with PZR

5.10 Steam generators secondary-side liquid level

The following Figures (30 and 31) show the collapsed liquid level of the secondary side of the steam generator. Experimentally, the liquid level is kept above a determined value so that the U-tubes are covered. The liquid level starts to decrease immediately after the initiation of the AM action. The most relevant discrepancy between the experimental values and those obtained with the model, is that with TRACE5 the sharp drop of level registered in the experiment in each relief valve actuation has not been reproduced.

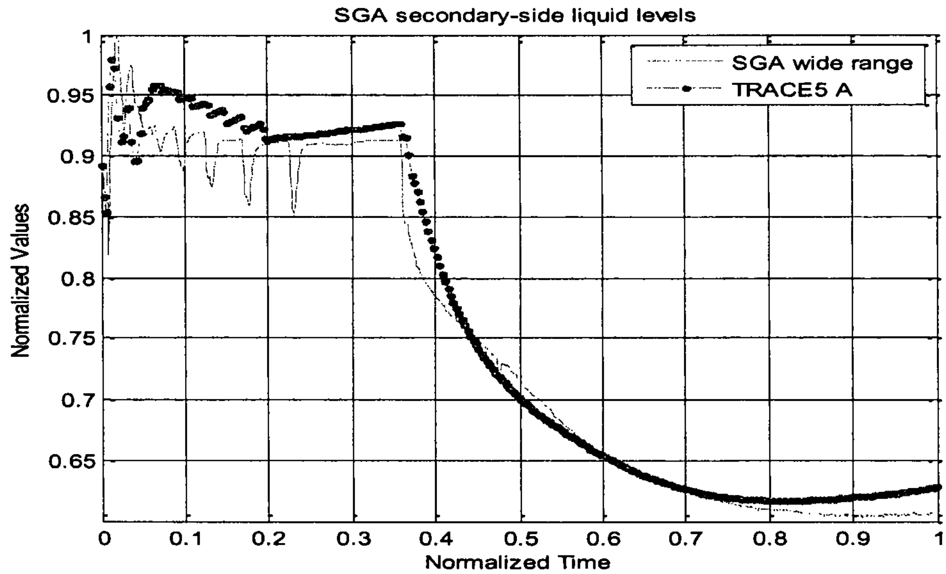

Figure 30 Steam generator A. Secondary-side collapsed liquid level

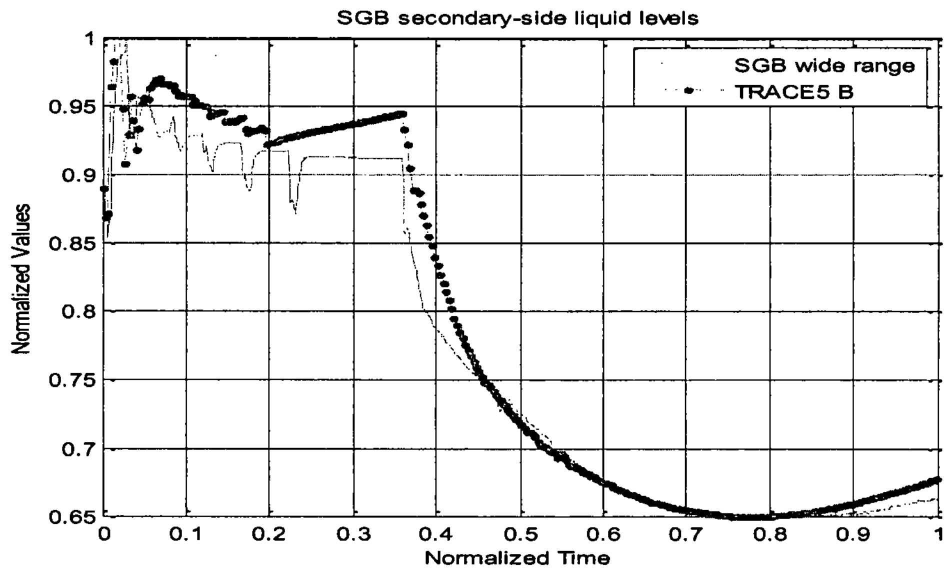

Figure 31 Steam generator B. Secondary-side collapsed liquid level

5.11 Pressurizer liquid level

Figure 32 shows the pressurizer water level. Liquid level starts to decrease immediately after the break. Experimentally, it becomes completely empty at 0.01 NT. TRACE5 reproduces the experiment adequately, but during the period between 0.01 and 0.03 NT, it predicts a liquid level slightly higher than the experimental.

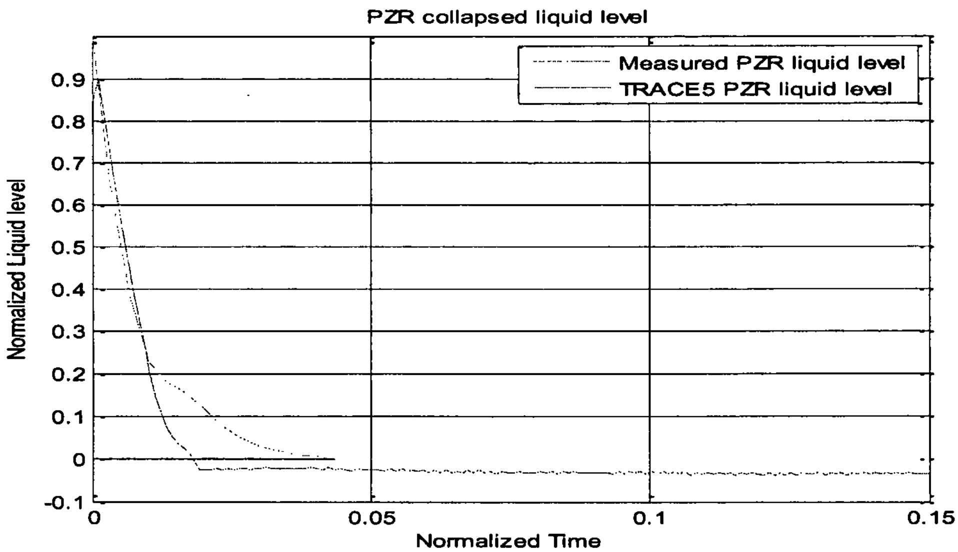

Figure 32 Pressurizer liquid level

5.12 Accumulator liquid level

Figure 33 shows the accumulator tank of loop A collapsed liquid level. In this case, there is no experimental data available. Accumulators start to empty at 0.43 NT, a little bit after the maximum fuel surface temperature is reached.

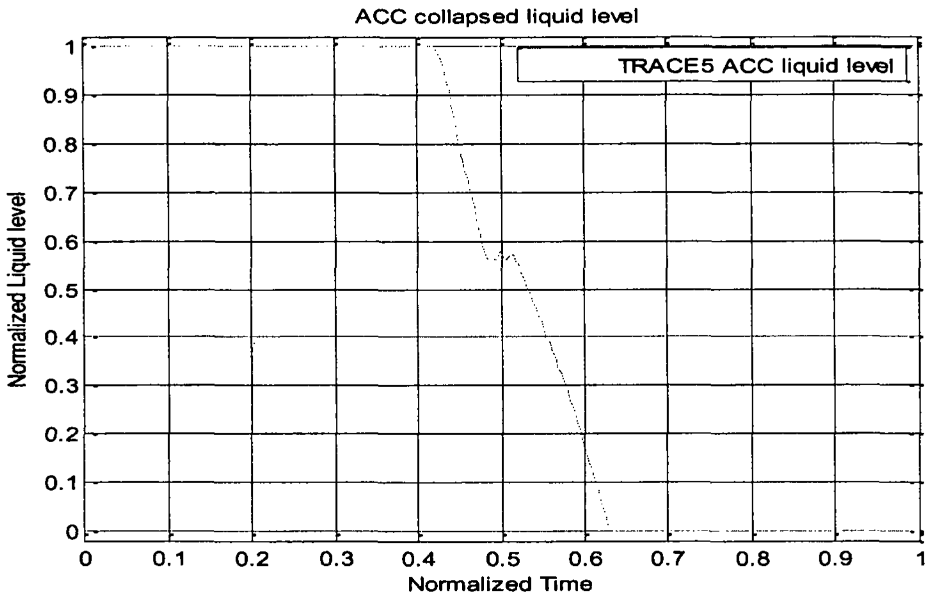

Figure 33 Accumulator tank liquid level

6 CONCLUSIONS

This paper contains results obtained in the simulation of the OECD/NEA ROSA Project Test 6-1 with the code TRACE5. Experimental results show that liquid level in the upper head controls the break flow rate. Phase change from two-phases to one single-phase steam is produced when coolant flow through control rod guide tubes is over. TRACE5 corroborates this fact. Depressurization produced by the break causes the primary pressure to become lower than SG secondary-side pressure at 0.27 NT. At this moment steam generators heat extraction starts to be ineffective. Approximately at 0.37 NT the accident management (AM) action is initiated, consisting in the steam generators secondary-side depressurization (activated when core fluid temperature exceeds a maximum temperature). This action appears to be insufficient to avoid the fuel surface rod temperature excursion (at 0.41 NT). Core protection system decreases the core power automatically at 0.41 NT. TRACE5 successfully reproduces all these events but it is necessary to take into account a limitation that needs to be improved in the future: the lack of a single-phase steam discharge coefficient. Comparison between experimental and TRACE5 results shows that discharged mass inventory through the break predicted by TRACE5 is well reproduced when fluid in upper head is one single liquid phase or two-phase in critical flow conditions. However, when fluid turns into steam single-phase, discharge mass flow is approximately 10% higher than the experiment. This overestimation of TRACE5 causes a lower primary pressure than the experimental. At 0.44 NT, water coming from accumulator tanks is injected in the cold leg of both loops. Upper plenum, core and downcomer refill is well reproduced by TRACE5. The main discrepancy after that is the liquid level decrease observed in the downcomer of the pressure vessel.

Finally, TRACE5 produces the low injection system activation at 0.93 NT, approximately, when primary pressure drops to a determined value, in good agreement with the experiment.

Variable	Meaning	Code	File
n_var	Total stored data: total number of the control variables that will move the control rod banks	PARCS v2.7	pdmr_varM.f
var_leidas(1000)	It stores the read data, that is, the ID's fo the RELAP5 control variables which will move the control rod banks in PARCS v2.7.	PARCS v2.7	pdmr_varM.f

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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 pressure vessel lower plenum Small Break Loss-Of-Coolant Accident (SBLOCA) under the assumption of total failure of High Pressure Injection System (HPIS) 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 6-2 (SB-PV-10 in JAEA). An asymmetrical steam generator secondary-side depressurization is produced as an accident management action at the steam generator in the loop without PZR, after the generation of the safety injection signal in order to achieve a determined depressurization rate in the primary system. 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 behavior in the entire transient. In conclusion, this work represents a small contribution for TRACE5.

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