NUREG/IA-0503

International Agreement Report

Post-Test Analysis of ROSA-2 Test 2 (IBLOCA) with TRACE

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

Universitat Politècnica de València
Instituto Universitario de Seguridad Industrial, Radiofísica y Medioambiental
Camí de Vera s/n
46022 Valencia, SPAIN

K. Tien, NRC Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: April 2018
Date Published: February 2019

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
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ABSTRACT

The purpose of this work is to overview the results obtained by simulating a 17% cold leg Intermediate Break Loss-Of-Coolant Accident (IBLOCA) in the Large Scale Test Facility (LSTF) using the thermal-hydraulic code TRACE5 patch 2. This IBLOCA transient corresponds to the Test 2 (IB-CL-03 in JAEA) and it is performed in the frame of the OECD/NEA ROSA-2 Project. During this transient, the single-failure of the High and the Low Pressure Injection systems (HPI and LPI, respectively) in the broken loop and the total failure of Auxiliary Feedwater (AFW) are assumed.

A detailed model of the LSTF and the chronology of events following these assumptions have been developed with TRACE5 patch 2. A comparison between the simulation and the experimental results is provided throughout different graphs. Acceptable general behavior is observed in the entire transient.

FOREWORD

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

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

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

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

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

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[1] SESARP/FAP is the Senior Group of Experts on Nuclear Safety Research Facilities and Programmers of NEA Committee on the Safety of Nuclear Installations (CSNI)

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

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

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

Both objectives are carried out by simulating the experiments and conducting the plant application with the last available versions of NRC TH codes (RELAP5 and/or TRACE).

On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermalhydraulics analysis of accidents in the Spanish nuclear power plants. Nuclear safety needs have not decreased as the nuclear share of the nations grid is expected to be maintained if not increased during next years, with new plants in some countries, but also with older plants of higher power in most of the countries. This is the challenge that will require new ideas and a continued effort.

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

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EXECUTIVE SUMMARY

The purpose of this work is to test the capability of the thermal-hydraulic code TRACE5 in the simulation of a cold leg Intermediate Break LOCA (IBLOCA) performed in the frame of the OECD/NEA ROSA-2 Project. An IBLOCA is chosen as a Design Basis Event (DBE) for the assessment of the Emergency Core Cooling System (ECCS) effectiveness. Many test facilities have produced experimental data on Small Break LOCA (SBLOCA) for code assessment. However, the experimental data for IBLOCA is quite limited. For the evaluation of safety margins in a realistic manner, detailed data prepared by tests simulating IBLOCA are desirable. Furthermore, the data are useful to understand thermal-hydraulic responses and to validate the latest Best Estimate (BE) computer codes.

Test 2 was conducted in the Large Scale Test Facility (LSTF), which simulates the thermal-hydraulic responses during a PWR 17% cold leg IBLOCA. Single-failure of both High and Low Pressure Injection systems (HPI and LPI, respectively) and the total failure of the Auxiliary Feedwater (AFW) are assumed. A detailed LSTF TRACE5 model has been developed following these assumptions.

The simulation results have been compared with the experimental measurements in different graphs, including primary and secondary pressures, discharged inventory, primary mass flow rate, and collapsed liquid levels (in the pressure vessel, hot leg, steam generators U-tubes, etc.). In general, the simulation results show good agreement with the available experimental data.

ACKNOWLEDGMENTS

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

ABBREVIATIONS AND ACRONYMS

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

1 INTRODUCTION

Many test facilities have produced experimental data on Small Break Loss-Of-Coolant Accident (SBLOCA) for code assessment. However, experimental data for Intermediate Break Loss-Of-Coolant Accident (IBLOCA) are quite limited. Some examples of research on this break size range can be found in literature, for example the work developed in 1986 at ROSA III [1] for BWR simulation.

Detailed thermal-hydraulic data are desirable to understand the thermal-hydraulic responses and to validate the latest Best Estimate (BE) computer codes. Regarding the simulation of these transients, some works have been developed with different codes, such as ATHLET, for example the post-test calculation to 11% break Loss-Of-Coolant Accident in 1997 [2].

The purpose of the present work is to describe the most relevant results achieved by using the thermal-hydraulic code TRACE5 patch 2 [3, 4] to simulate the Test 2 (IB-CL-03) [5], which reproduces an IBLOCA. This transient was developed by the Thermohydraulic Safety Research Group of the Nuclear Safety Research Center (Japanese Atomic Energy Agency, JAEA) in the frame of the OECD/NEA ROSA-2 Project. Test 2 was handled in the Large Scale Test Facility (LSTF) [6], which simulates a PWR reactor, Westinghouse type of four loops and 3423 MW of thermal power, scaled to 1/48 in volume. Test 2 simulates the thermal-hydraulic responses during a PWR 17% cold leg IBLOCA by using a long break nozzle with upwards orientation. In this transient, a single–failure of both High and Low Pressure Injection systems (HPI and LPI, respectively) and the total failure of the Auxiliary Feedwater (AFW) are assumed.

The simulation results are compared with the experimental measurements in several graphs, including primary and secondary pressures, discharged inventory, primary mass flow rates, and collapsed liquid levels (in the pressure vessel, hot legs, steam generators U-tubes, etc.). The comparison between simulation and experimental results shows the TRACE5 capacity to reproduce the experimental behavior. In general, TRACE5 results show good agreement with experimental data.

2 ROSA FACILITY DESCRIPTION

In this section, a brief description of the LSTF facility [6] is presented. LSTF simulates a PWR reactor, Westinghouse type, of four loops and 3423 MW of thermal power. The facility is electrically heated, scaled 1:1 in height and 1:48 in flow areas and volumes, except the loops, which are defined by a scaling factor of 1:24. Figure 1 shows the scheme of the LSTF facility.

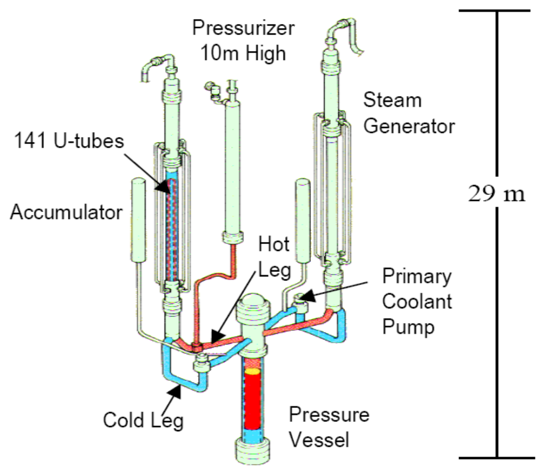

Figure 1 Schematic View of the LSTF Facility

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

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. The vessel has 8 upper head spray nozzles (of 3.4 mm inner-diameter) and 8 Control Rod Guide Tubes (CRGTs). The maximum core power of the LSTF is limited to 10 MW, which corresponds to 14% of the volumetrically scaled PWR core power, being capable to simulate the PWR decay heat after the reactor scram.

Regarding the SGs, each of them contains 141 U-tubes, which can be classified into different groups depending on their length. The U-tubes have an inner diameter of 19.6 mm and an outer diameter of 25.4 mm (with 2.9 mm thickness). On the other hand, the vessel, plenum and riser of the 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

Test 2 starts by opening the break valve in the cold leg of the loop without pressurizer and increasing the rotational speed of the coolant pumps. Few seconds afterwards, the scram signal is generated. This signal produces the initiation of the core power decay curve. The complete control logic of the transient is listed in Table 1.

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

Break.	Time zero
Generation of scram signal.	Primary pressure reaches 12.97 MPa.
Pressurizer (PZR) heater off.	Generation of scram signal or PZR liquid level below 2.3 m.
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 SG main steam stop valve).	Generation of scram signal.
Closure of Steam Generators (SG) Main Steam Isolation Valves (MSIVs).	Generation of scram signal.
Termination of SG Main Feedwater.	Generation of scram signal.
Generation of Safety Injection (SI) signal	Primary pressure reaches 12.27 MPa.
Initiation of High Pressure Injection system (HPI) in loop with PZR only.	12 seconds after SI signal.
Initiation of Accumulator Injection System (AIS) in loop with PZR only.	Primary pressure decreases to 4.51 MPa.
Initiation of Low Pressure Injection system (LPI) in loop with PZR only.	Primary pressure decreases to 1.24 MPa

The scram signal produces the initiation of primary coolant pumps coast down, turbine trip, closure of Main Steam Isolation Valves (MSIV) and termination of Main Feedwater (MFW). When the primary pressure falls below 12.27 MPa, the Safety Injection (SI) signal is generated. The HPI system is activated few seconds after the SI signal generation only in the loop with pressurizer. The Accumulator Injection System (AIS) actuates when primary pressure falls below 4.51 MPa. In order to protect the facility, the core power is decreased by the LSTF Core Protection System when the maximum fuel rod surface temperature reaches 958 K, as it can be seen in Table 2. Finally, at 1.24 MPa, the LPI system actuates following a determined Pressure-Mass flow rate curve. LPI is only activated in the loop with pressurizer.

Test 2 finishes with the closure of the break valve when the primary and secondary pressures are stabilized.

Table 2 Core Protection Logic

Core power to	Maximum fuel rod surface temperature (K)
70%	958 K
35%	961 K
13%	966 K
5%	977 K
0% (core power trip)	1003 K

4 TRACE5 MODEL OF LSTF

LSTF has been modeled with 82 hydraulic components (7 BREAKs, 11FILLs, 23 PIPEs, 2 PUMPs, 1 PRIZER, 22 TEEs, 15 VALVEs and 1 VESSEL). Figure 2 shows the nodalization of the model using the Symbolic Nuclear Analysis Package software (SNAP) [7].

Primary side comprises cold and hot legs, pumps, loop seals, a pressurizer in loop A, the ECCS which includes AIS, HPI and LPI systems, the U-tubes of both SG and the PV. On the other hand, secondary side includes steam separators, downcomers, Safety Relief Valves (SRV), MSIV and FILLs to provide MFW and AFW.

The PV has been modelled using a 3D–VESSEL component. The VESSEL consists of 20 axial levels, 4 radial rings and 10 azimuthal sectors. For each axial level, volume and effective flow area fractions have been set according to technical specifications provided by the organization [6]. Table 3 shows the vessel nodalization in the axial direction.

Table 3 Vessel Nodalization

Levels	Parts of the vessel
1-2	Lower plenum
3-11	Core
12	Upper core plate
13-16	Upper plenum
17	Upper core support plate
18-20	Upper head

The 3D-VESSEL is connected to different 1D components: 8 control rod guide tubes (CRGT), hot leg A and B (level 15), cold leg A and B (level 15) and a bypass channel (level 14). Control rod guide tubes have been simulated using PIPE components, connecting levels 14 and 19 and allowing the flow between upper head and upper plenum.

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) were distributed into the 3 rings: 154 elements in ring 1, 356 in ring 2 and 498 in ring 3. In both axial and radial direction, different peaking factors have been considered. The power ratio in the axial direction presents a peaking factor of 1.495, while the radial power profile is divided into three power zones using the first three radial rings. Depending on the radial ring, different peaking factors have been considered (0.66 in ring 1, 1.51 in ring 2 and 1.0 in ring 3). 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.

A detailed model of SG (geometry and thermal features) has been developed. Both boiler and downcomer components of the secondary-side have been modeled by TEE components. U-tubes have been classified into three groups according to their length.

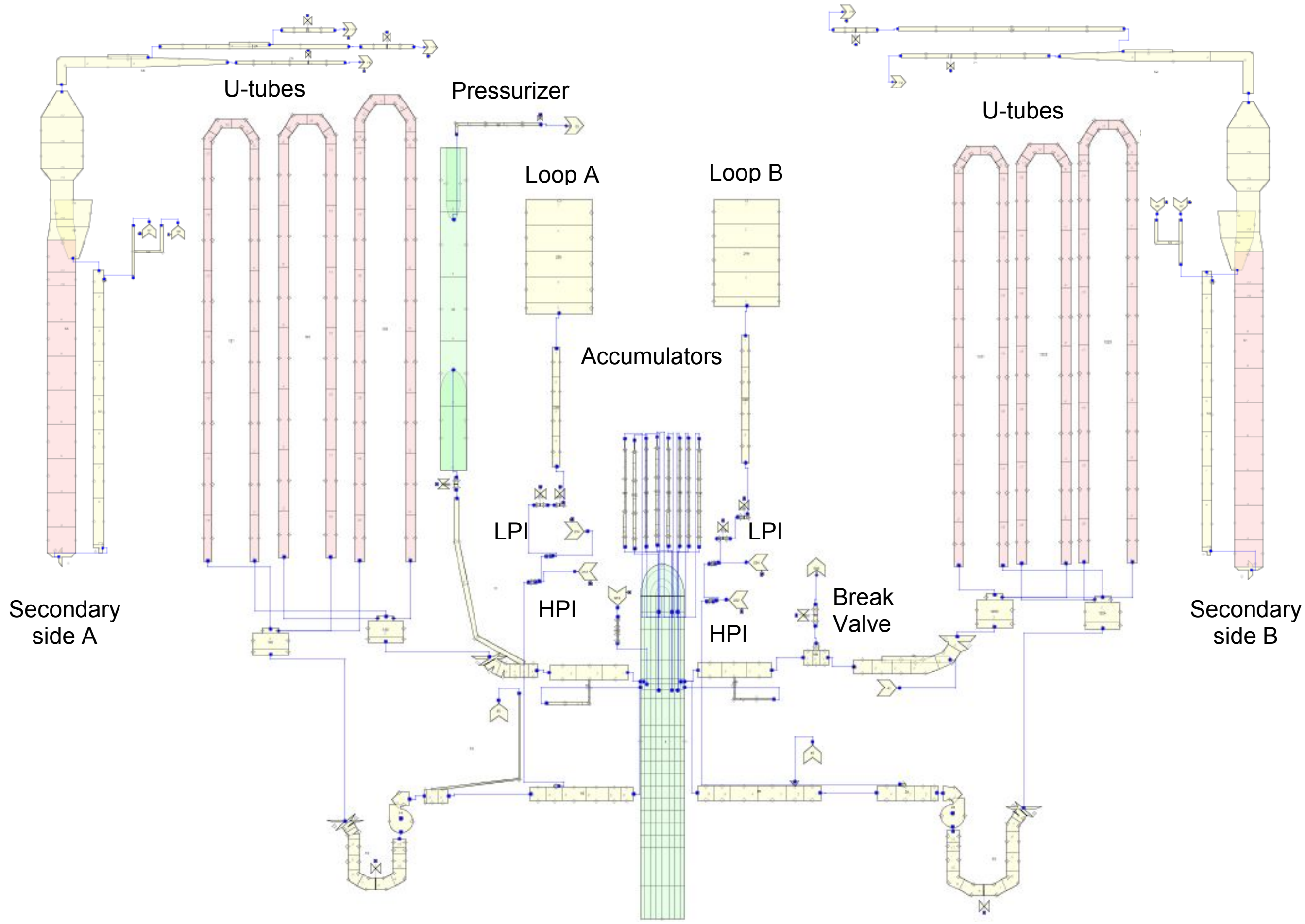

Figure 2 Model Nodalization

The separator can be simulated in TRACE5 setting a friction coefficient (FRIC) greater than 10^{22} at a determined cell edge, allowing only gas phase to flow through the cell interface. Heat transfer between primary and secondary sides has been performed using HTSTR components. Cylindrical-shape geometry has been used to best fit heat transmission. Inner and outer surface boundary conditions for each axial level have been set to couple HTSTR components to hydro components (primary and secondary fluids). Heat losses to environment have been considered in the secondary-side walls. Choke model predicts for a given cell the conditions for which choked flow is expected to occur, providing three different models: subcooled-liquid, two-phase and single-phase vapor model. TRACE5 patch 2 code allows to fix the subcooled-liquid and two-phase coefficients. In this case, the default values (1.0) have been selected. The break has been simulated by means of a VALVE component connected to a BREAK component in order to establish the boundary conditions.

5 RESULTS AND DISCUSSION

5.1 Steady-State

Steady-state conditions achieved in the simulation were in reasonable agreement with the experimental values. Table 4 shows the relative errors (%) between experimental and simulated results for different items. The maximum difference between experiment and simulation is lower than 5%.

Table 4 Steady-State Conditions. Comparison between Experimental and Simulated Values

Item (Loop with PZR)	Relative Error (%)
Core Power	0.00
Hot leg Fluid Temperature	0.30
Cold leg Fluid Temperature	-0.21
Mass Flow Rate	-2.16
Pressurizer Pressure	-0.39
Pressurizer Liquid Level	-2.89
Accumulator System Pressure	0.22
Accumulator System Temperature	0.22
SG Secondary-side Pressure	-1.23
SG Secondary-side Liquid Level	-4.78
Steam Flow Rate	-4.55
Main Feedwater Flow Rate	-0.37
Main Feedwater Temperature	0.36
Auxiliary Feedwater Temperature	0.00

5.2 Transient

Table 5 lists the chronology sequence of events during the transient and the comparison in time between the experiment and TRACE5.

Table 5 Chronological Sequence of Events. Comparison between Experiment and TRACE5

Event	Experiment Time (s)	TRACE5 Time (s)
Break valve open	0	0
Scram signal	7	4
SI signal	9	5
Closure of steam generators (SG) Main Steam Isolation Valves (MSIVs)	10	5
Initiation of decrease in liquid level in SG U-tube	About 10	10
Initiation of coastdown of primary coolant pumps	11	10
Termination of SG main feedwater	13	5
Open of SG relief valves (RVs)	About 27-57	20
Initiation of core power decay	29	20
Activation of HPI system in loop with PZR only	About 35	20
Primary pressure becomes lower than SG secondary side pressure	About 55	56
Activation of Accumulator Injection System (AIS) in loop with PZR only	About 110	116
Core power decrease by LSTF core protection system when maximum fuel rod surface temperature reached 958 K	About 140	155
Maximum fuel rod surface temperature about 978 K	About 150	159
Primary coolant pumps stop	260	250
Termination of AIS in loop with PZR only	About 280	250
Activation of Low Pressure Injection (LPI) system in loop with PZR only	About 290	295
Break valve closure	500	500

5.3 System Pressures

A comparison between primary and secondary pressures is presented in Figure 3. At time 0, the primary pressure starts to decrease due to the loss of coolant discharged through the break. After the closure of the MSIVs, the secondary pressure of Steam Generators increases, reaching its maximum value at 25 s. Due to the secondary pressure increase, the SG Relief Valves are opened about 30 s. Simultaneously, the primary pressure is smoothed temporarily due to the MSIVs closure and then starts again to decrease.

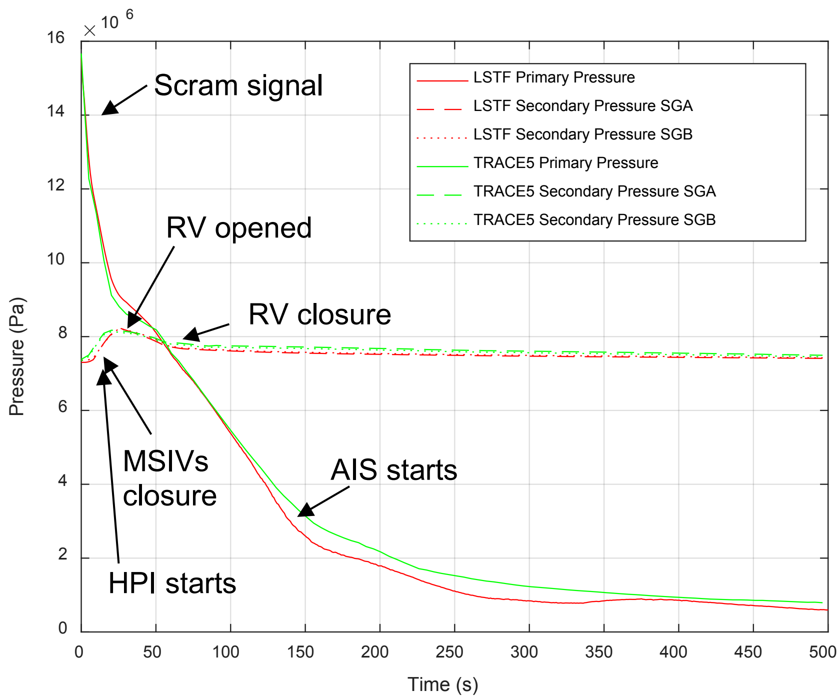

Figure 3 Primary and Secondary Pressures

The primary pressure soon becomes lower than SG secondary–side pressure (at about 60 s). From this moment on, the SGs no longer serve as the heat sink. The accumulator system is initiated at about 110 s when the primary pressure decreases to 4.51 MPa. At this moment, the depressurization becomes effective due to steam condensation caused by the coolant injection in one cold leg.

The main discrepancies in the primary pressure between simulation and experimental measurements are observed when the injection of the accumulator begins (at 125 s approximately). As it can be seen, TRACE5 overpredicts the primary pressure. These discrepancies in pressure can be attributed to the different distribution of coolant inventory in the primary system.

Regarding to the secondary pressure, TRACE5 adequately reproduces the pressure increase during the interval between 0 – 25 s. During this period, the SG relief valves are completely opened. At 25 s, the RVs are closed and the loss of pressure is due to the heat losses in the SGs secondary side.

In general, both primary and secondary-side pressures are successfully reproduced by TRACE5 in the whole transient, disagreeing only during a certain time interval (between 150 and 350 s) due to different distribution of coolant inventory in the primary system.

5.4 Break

Figure 4 and Figure 5 show the mass flow rate and the discharged coolant inventory through the break, respectively. A sensitivity analysis varying the discharge coefficients of the Choked Flow model [4] has been performed to adjust the break mass flow rate and the discharged primary coolant inventory with TRACE5. From this analysis it has been stated that the discharge coefficients fixed to 1.0 (default value), for single-phase liquid and two-phase, fit successfully the experimental data.

TRACE5 estimates a change from liquid single-phase to two-phase in a very short time after the break, at around 10 s. The two-phase fluid regime is maintained until the hot legs are empty (about 150 s, Figures 14 and 15), when the fluid regime changes to one-phase vapor. As it can be seen, a good agreement between TRACE5 results and experimental data is achieved.

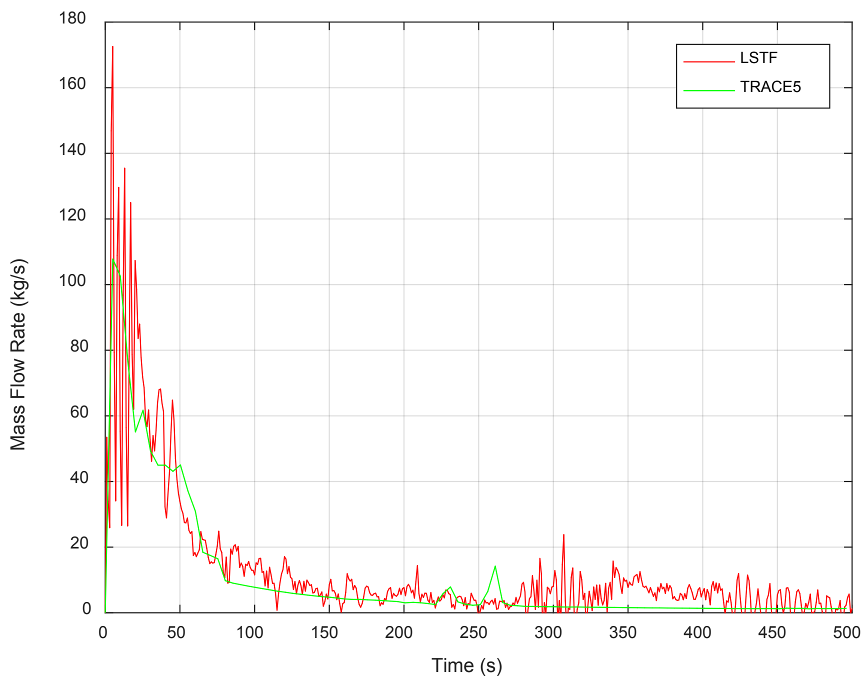

Figure 4 Break Mass Flow Rate

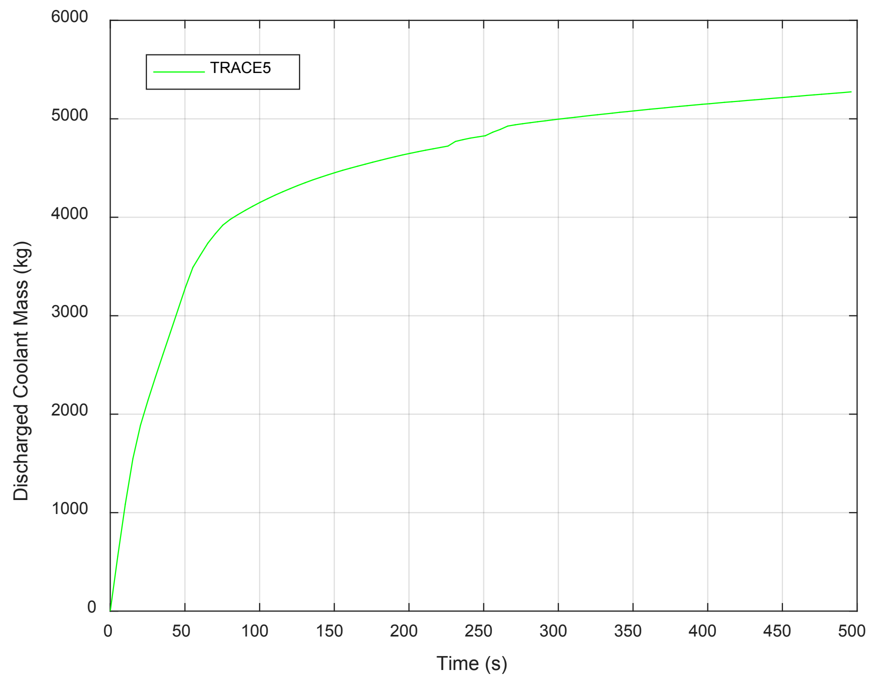

Figure 5 Discharged Inventory through the Break

5.5 Primary Loop Mass Flow Rates

Primary loop A and B mass flow rates are shown in Figures 6 and 7, respectively. In the first part of the transient, the primary mass flow increases due to the higher angular speed of the pumps. At 10 s, it starts to decrease simultaneously with the pumps coast down. After that, only natural circulation occurs and it contributes to cool down the core during the first part of the transient, until 30 s, when the SG RVs are fully closed.

TRACE5 underpredicts the primary mass flow rate in both loops, between 50 and 300 s. Furthermore, the effect of the accumulator injection is only observed in loop A.

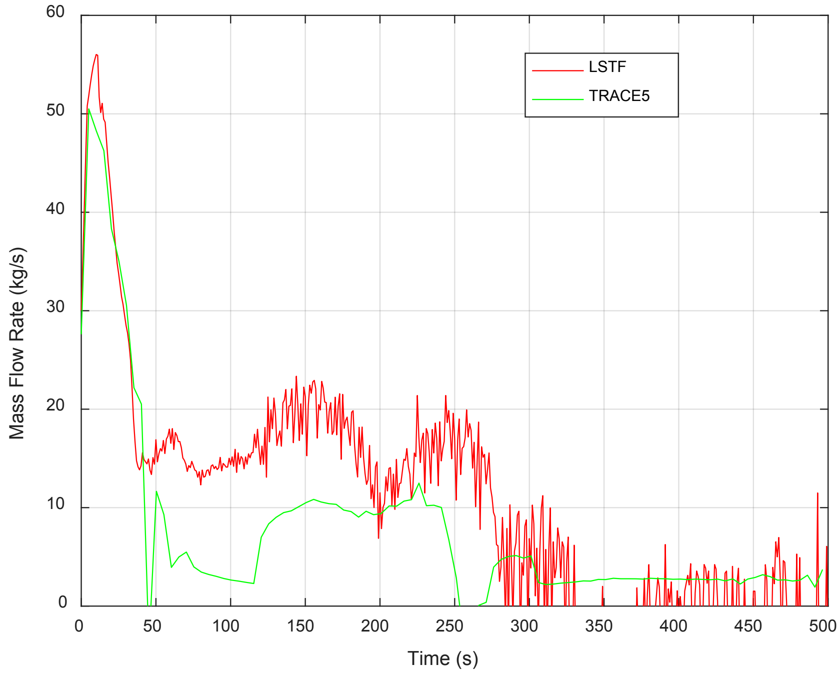

Figure 6 Primary Loop A Mass Flow Rate

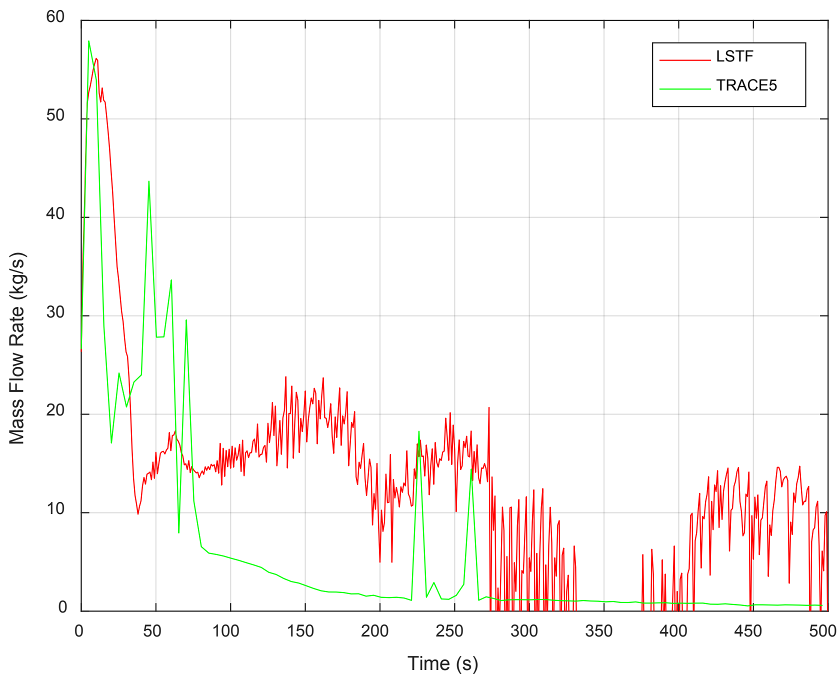

Figure 7 Primary Loop B Mass Flow Rate

5.6 Vessel Collapsed Liquid Levels

The following Figures (8, 9 and 10) show a comparison between the collapsed liquid levels in core, upper plenum and downcomer, respectively, for both experimental and TRACE5 results. All the tendencies are successfully reproduced by TRACE5. However, some discrepancies are observed.

Active core and upper plenum liquid levels sharply decrease after the break due to the flashing of fluid because of a fast primary depressurization, but temporarily increase when the break mass flow rate starts to decrease, reaching a relative maximum at 25 s approximately. In Figure 8, it is shown that the flashing predicted by TRACE5 produces a more abrupt core liquid level reduction in comparison to the experimental data.

At 110 s, the pressure measured in the pressure vessel drops below 4.51 MPa and the AIS in the loop with pressurizer is activated. From this moment on, the refill of the active core is produced. The entrance of cold water injected in one of the cold legs produces the vapor condensation in the upper plenum.

As it can be seen in Figure 9 and 10, discrepancies in upper plenum and downcomer are observed during the refill. In the experiment, the refill is produced at the same time, but higher condensation produces a faster liquid level increasing.

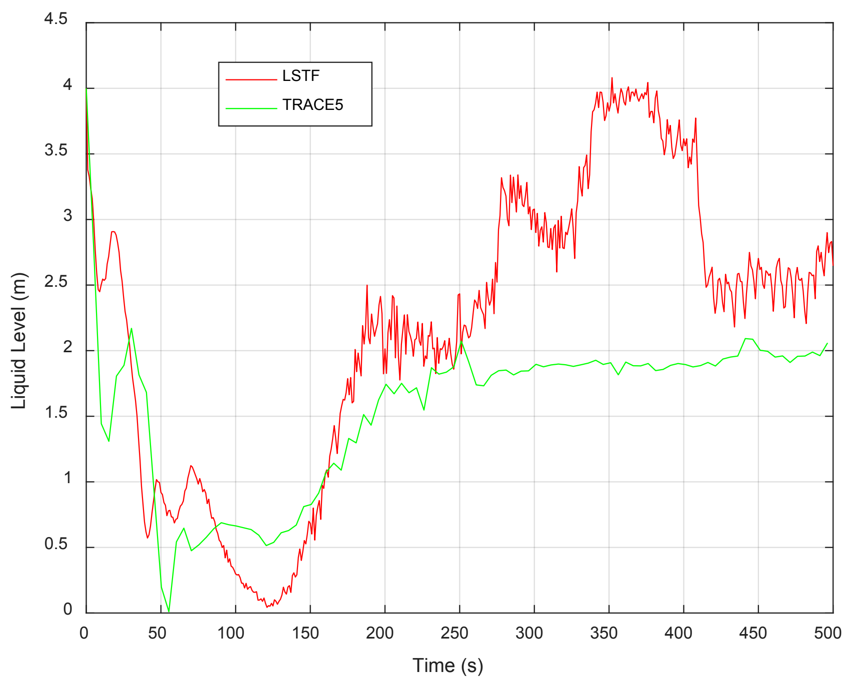

Figure 8 Core Collapsed Liquid Level

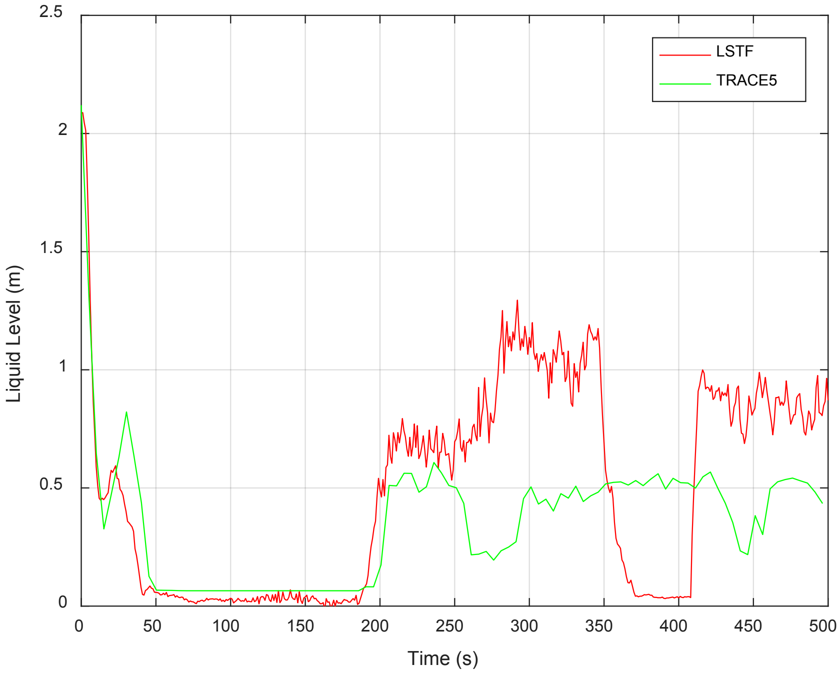

Figure 9 Upper Plenum Collapsed Liquid Level

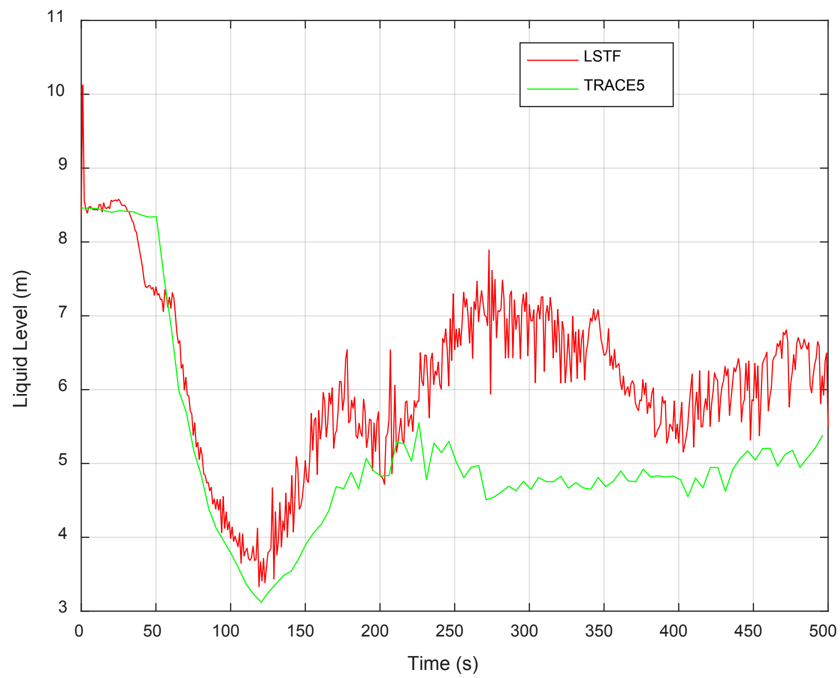

Figure 10 Downcomer Collapsed Liquid Level

5.7 Maximum Fuel Rod Surface and Core Exit Temperatures

The maximum Peak Cladding Temperature (PCT) and the Core Exit Temperature (CET) are shown in Figure 11. The maximum PCT starts to increase when the core clearance takes place and the upper plenum becomes almost empty (at 50 s, approximately). TRACE5 predicts such increment some seconds later. However, the maximum values are similar and they are reached almost at the same time. These maximum values produce the activation of the LSTF core protection system, which automatically decreases the core power (Figure 20). It is observed that the maximum CET is produced at the same time that the maximum PCT.

Figure 12 shows the fuel rod temperatures obtained with TRACE5 at nine different axial positions, in which the core is divided. Figure 13 shows the fuel rod temperature for different HTSTRs measured in the fifth axial level of the core. As it can be seen, the maximum fuel rod temperatures correspond to HTSTR 320 and 325, which are located in the second ring.

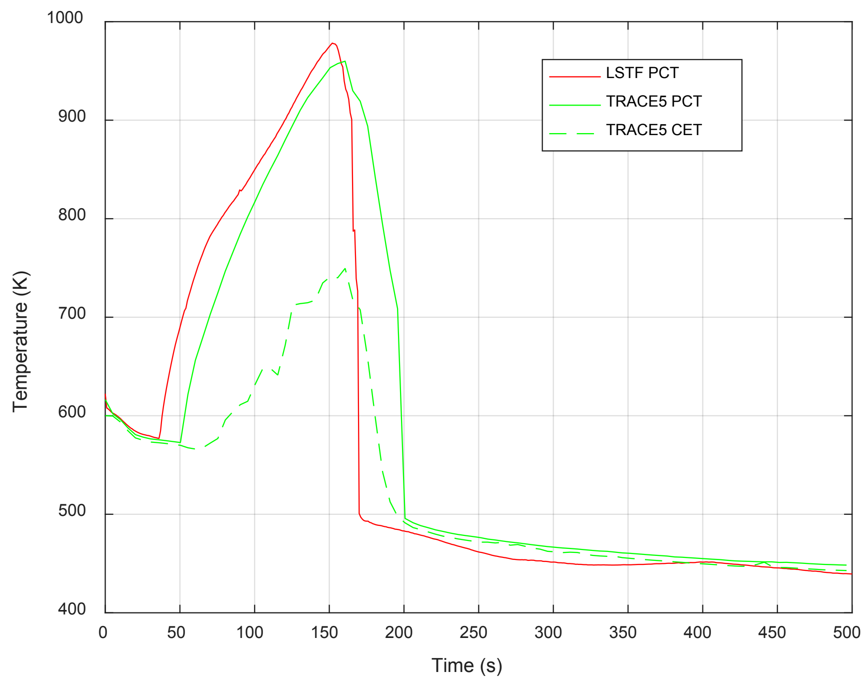

Figure 11 Maximum Fuel Rod Surface and Core Exit Temperatures

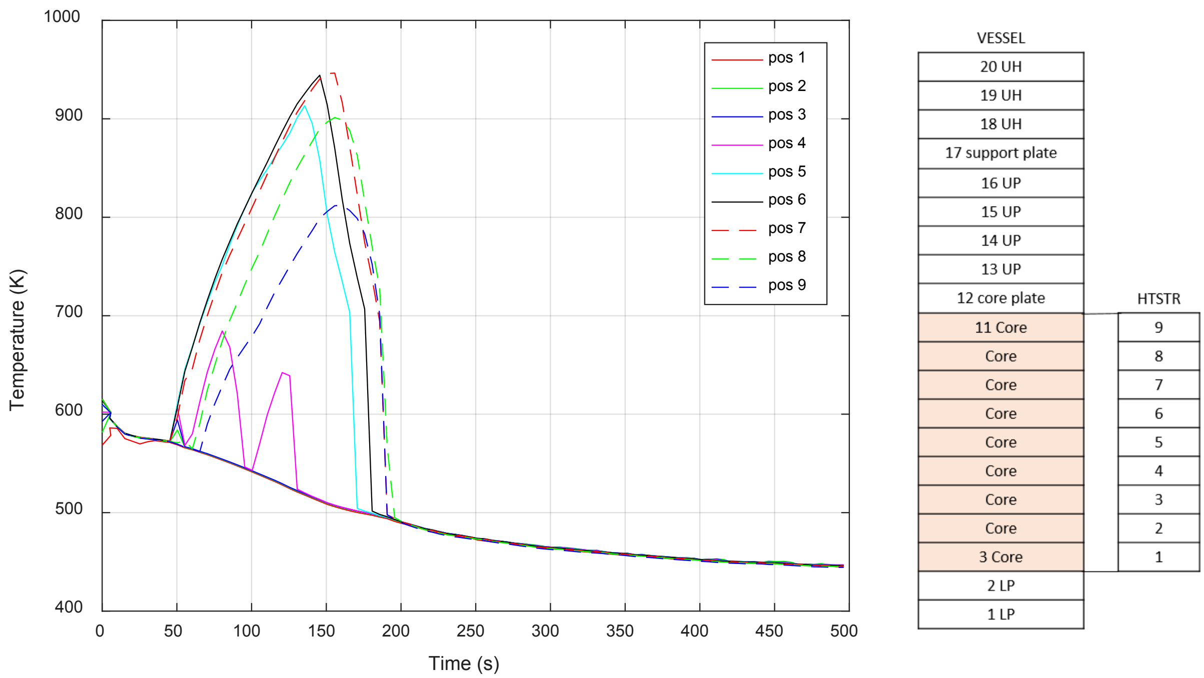

Figure 12 Fuel Rod Surface Temperatures at Different Axial Positions

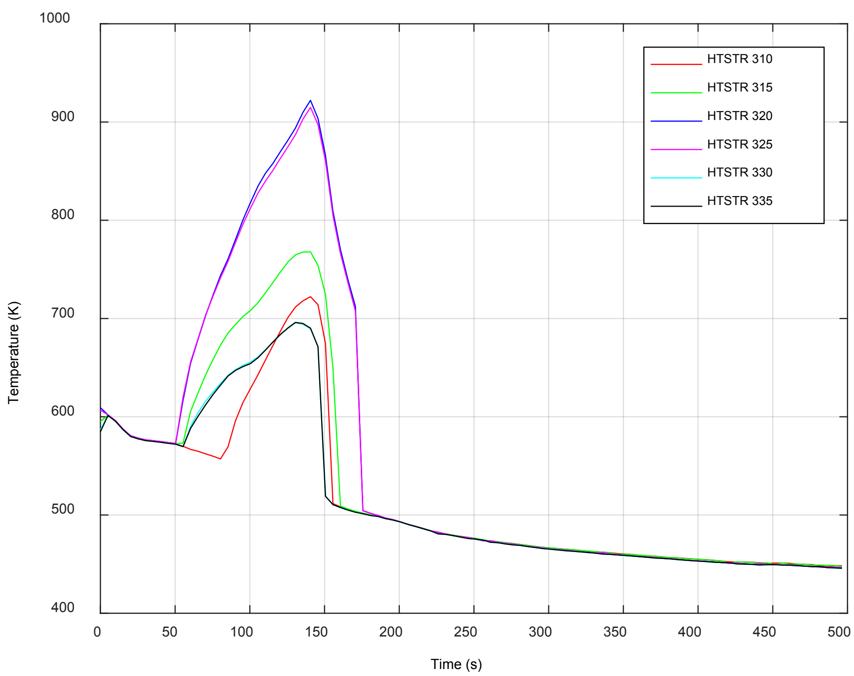

Figure 13 Fuel Rod Surface Temperatures for Different HTSTR in the Axial Level 5

5.8 Hot and Cold Legs Collapsed Liquid Levels

Figures 14 and 15 show the liquid level in the hot legs. Just after the break, the fast primary depressurization produces the flashing of fluid. As a consequence, the liquid levels in both hot legs fall, as it can be seen. The main discrepancies are produced between 50 and 150 s, when both hot legs are empty (in TRACE5). During the refill, from 250 to 350 s, TRACE5 does not reproduce the experimental behavior.

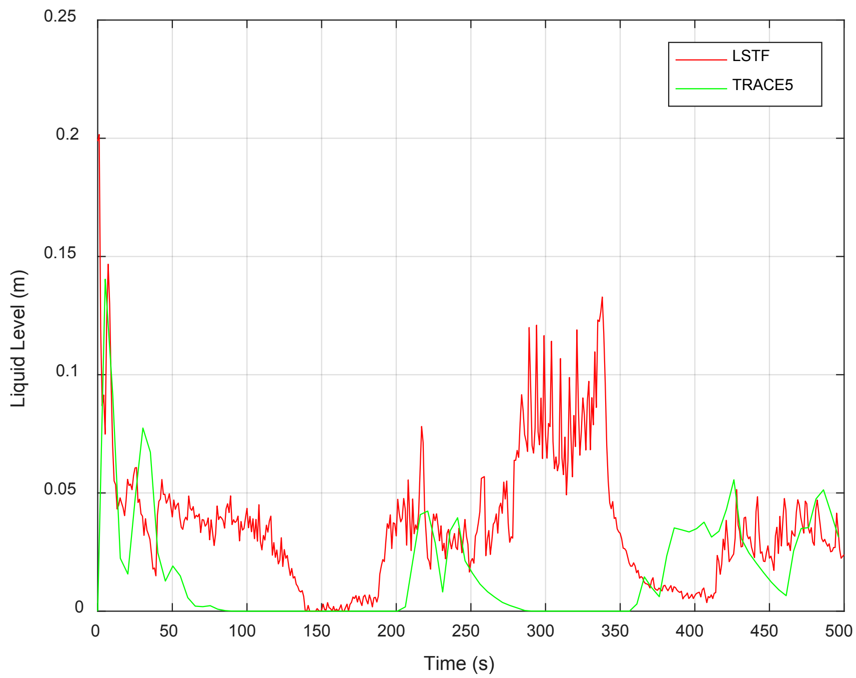

Figure 14 Collapsed Liquid Level in the Hot Leg A

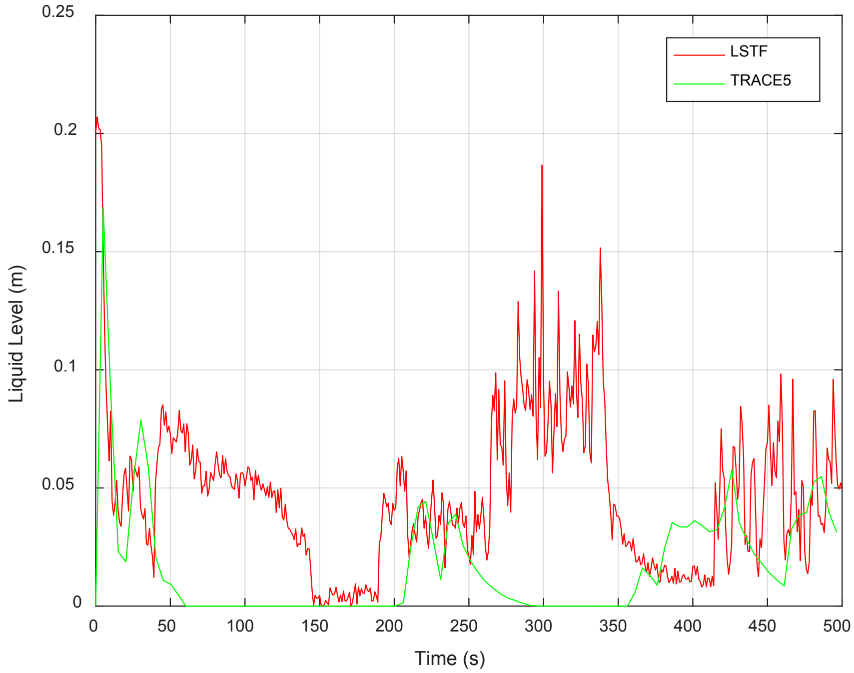

Figure 15 Collapsed Liquid Level in the Hot Leg B

5.9 Emergency Core Cooling Systems

The HPI system is activated almost simultaneously with the core dry out (about 20 s), as it can be seen in Figure 16. However, it is ineffective on the core cooling. A large temperature excursion in the core induces the automatic core power system actuation to protect the LSTF core. In general, a good reproduction has been achieved with TRACE5 model.

The accumulator mass flow rate is shown in Figure 17. In general, TRACE5 reproduces the accumulator injection similar to the experiment, although the total mass injected is 600 kg lower.

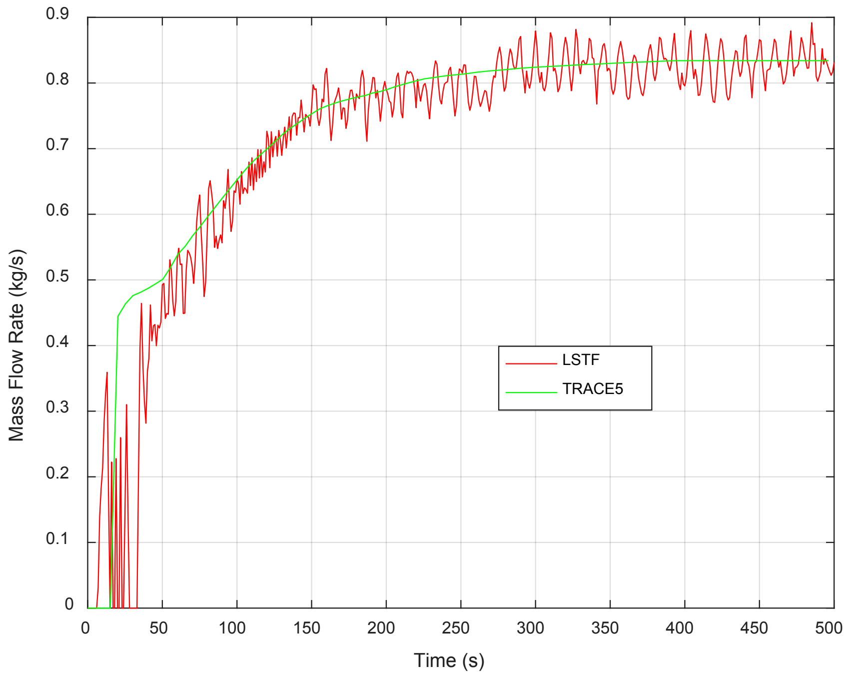

Figure 16 High Pressure Injection System Mass Flow Rate

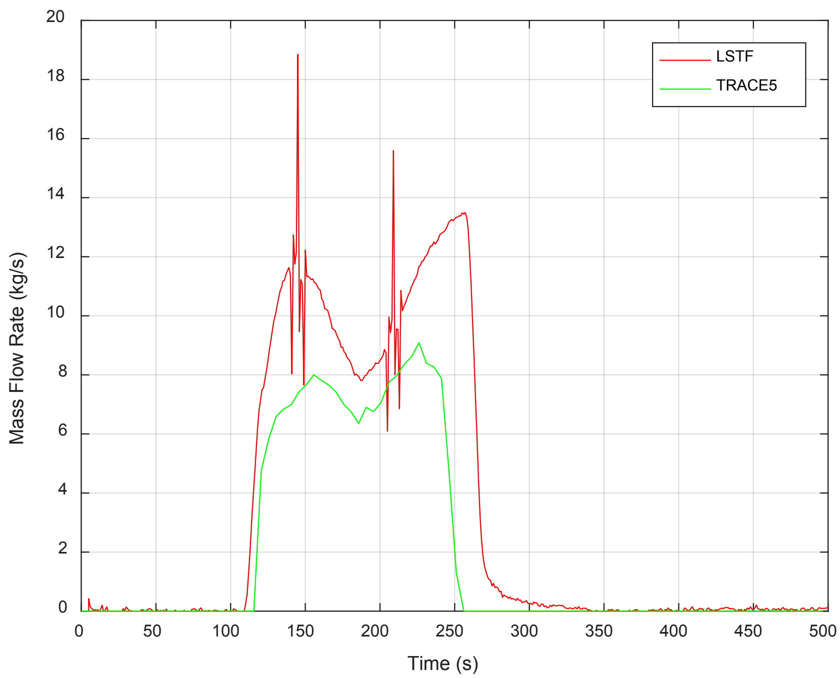

Figure 17 Accumulator Injection System Mass Flow Rate

5.10 U-tubes Collapsed Liquid Level

Figures 18 and 19 show the U-tubes collapsed liquid level. As it can be seen, at the beginning of the transient, the U-tubes start to empty. At 100 s approximately, the U-tubes are completely empty and the heat transfer between primary and secondary sides is finished.

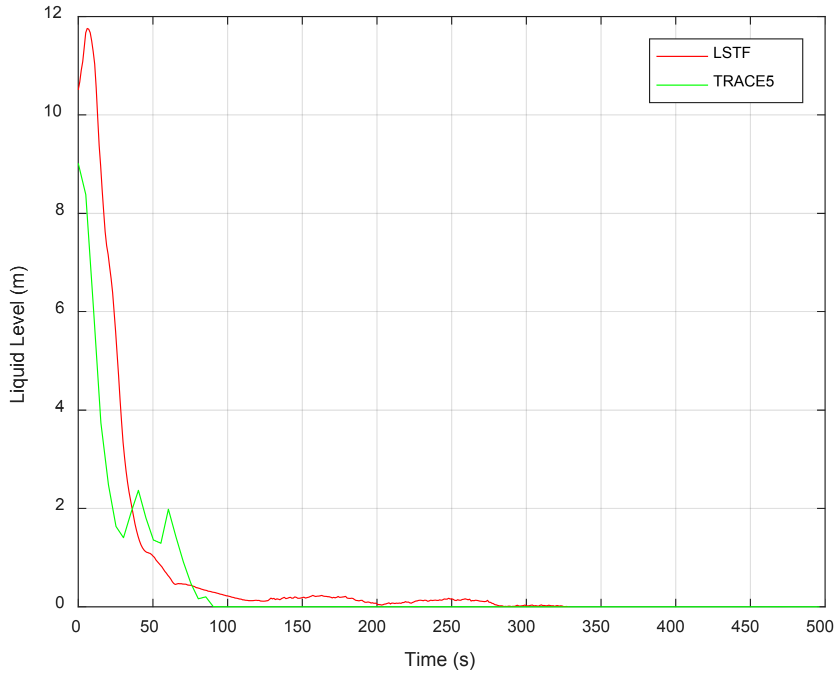

Figure 18 SG U-tube Up-Flow Side Collapsed Liquid Levels in Loop with PZR

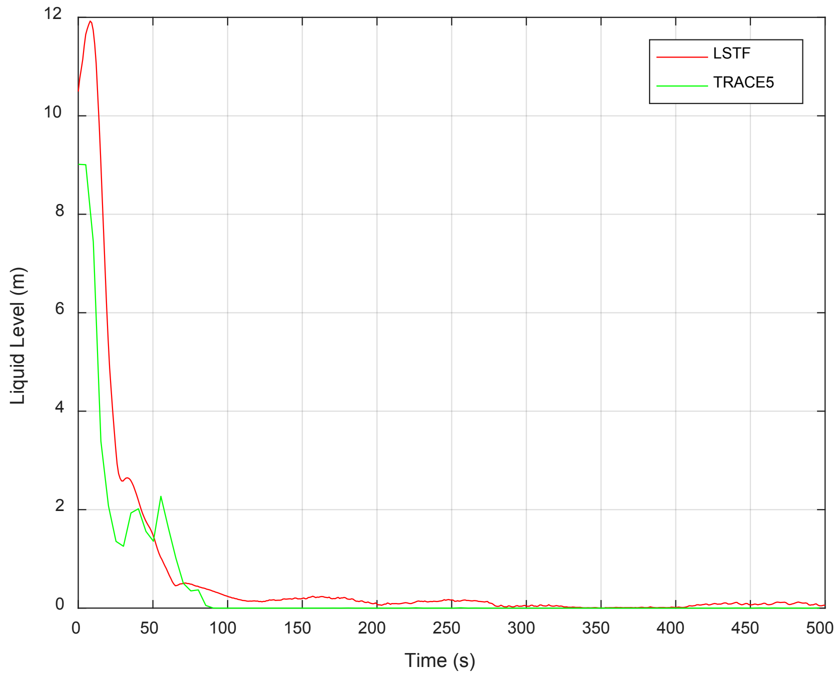

Figure 19 SG U-tube Up-Flow Side Collapsed Liquid Levels in Loop with/o PZR

5.11 Core Power

Figure 20 shows the core power curves and the effect of the core protection system activation. The core power curves are modified by the actuation of the LSTF core protection system. The differences in the power reduction are due to discrepancies in the maximum PCT value (Figure 11).

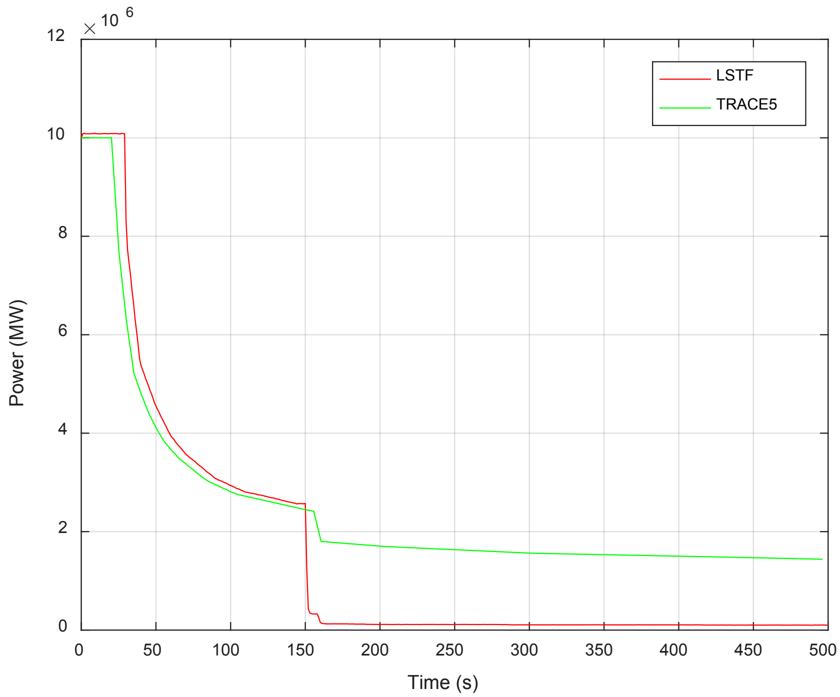

Figure 20 Core Power

5.12 Void Fraction

Figures 21, 22, 23 and 24 show the void fraction achieved using the model during the transient. Figure 21 shows the void fraction in the LSTF at the beginning of the test. As it can be seen, primary and secondary systems are plenty of liquid. Then, the situation at 25 s is shown in Figure 22, when the pressurizer is emptied. At this moment, the liquid is located in the secondary side and in the loop seals. The core region is almost empty. Figure 23 shows the situation when the accumulator of loop A is emptied at 251 s. In this case, the liquid is located in the secondary side. Upper head, upper plenum and core region of the PV are emptied. Figure 24 shows the void fraction at the end of the transient. At this moment, the primary side is completely empty, although some liquid is located in the PV.

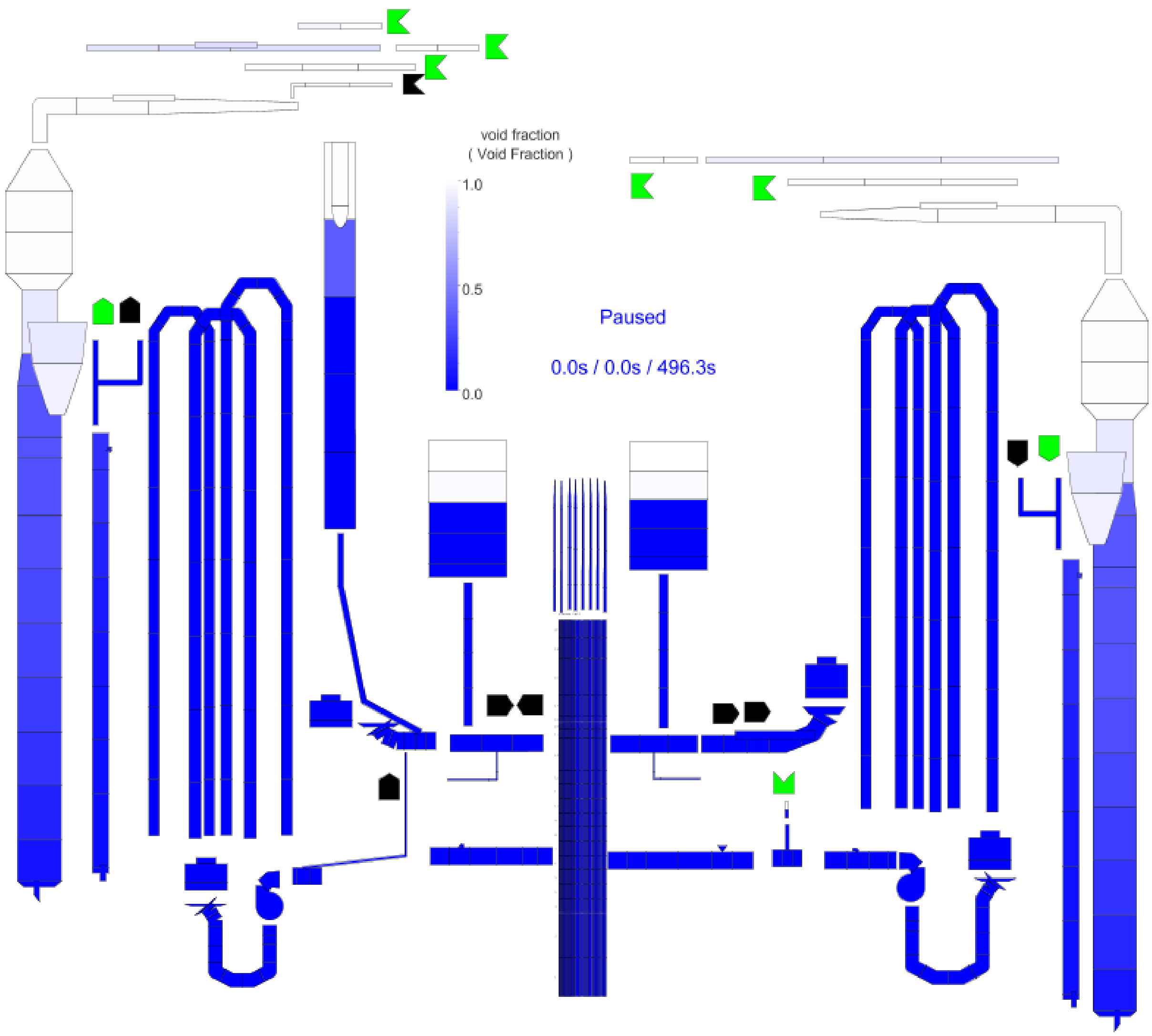

Figure 21 Void Fraction in the LSTF at 0 s

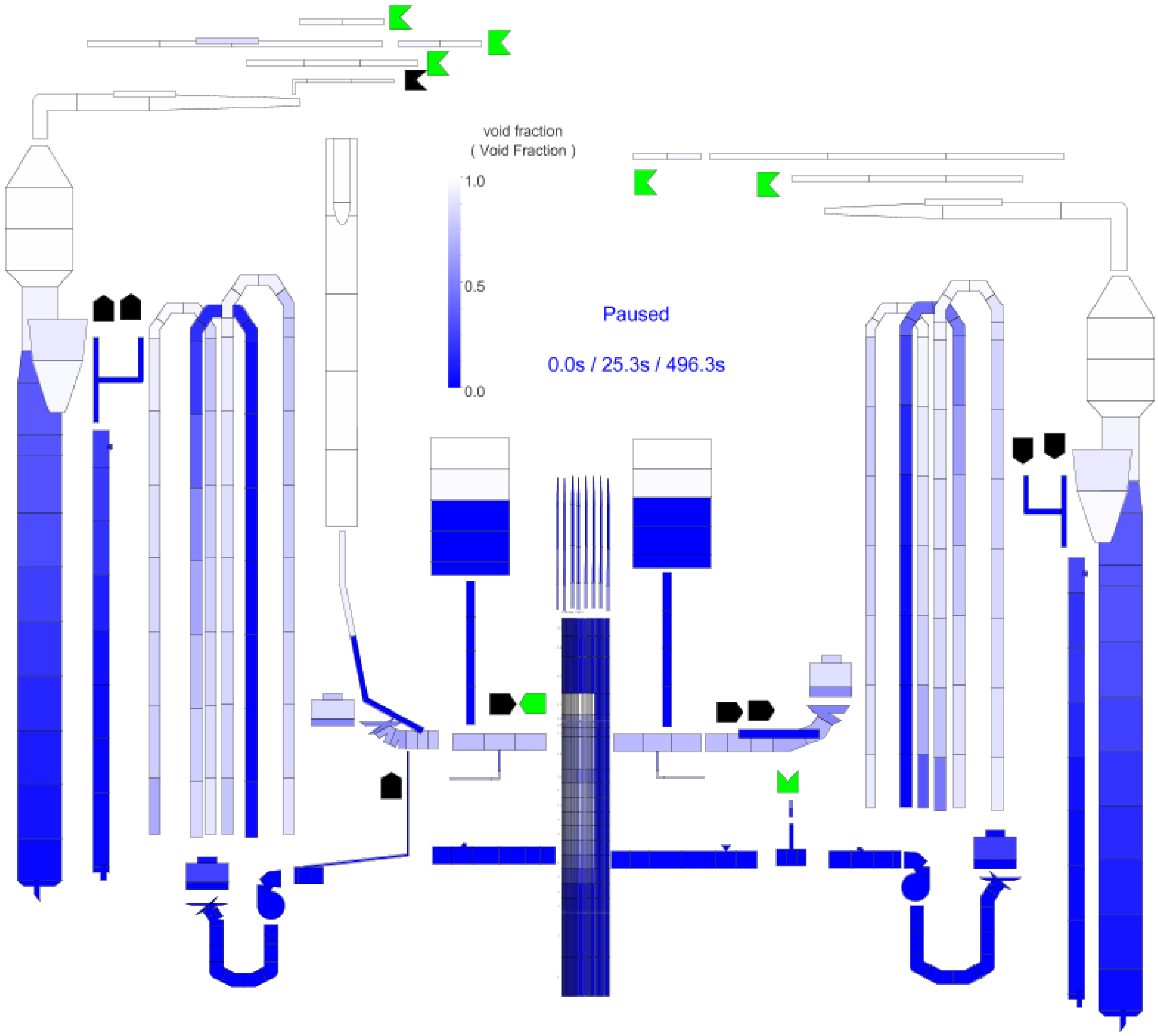

Figure 22 Void Fraction in the LSTF at 25 s

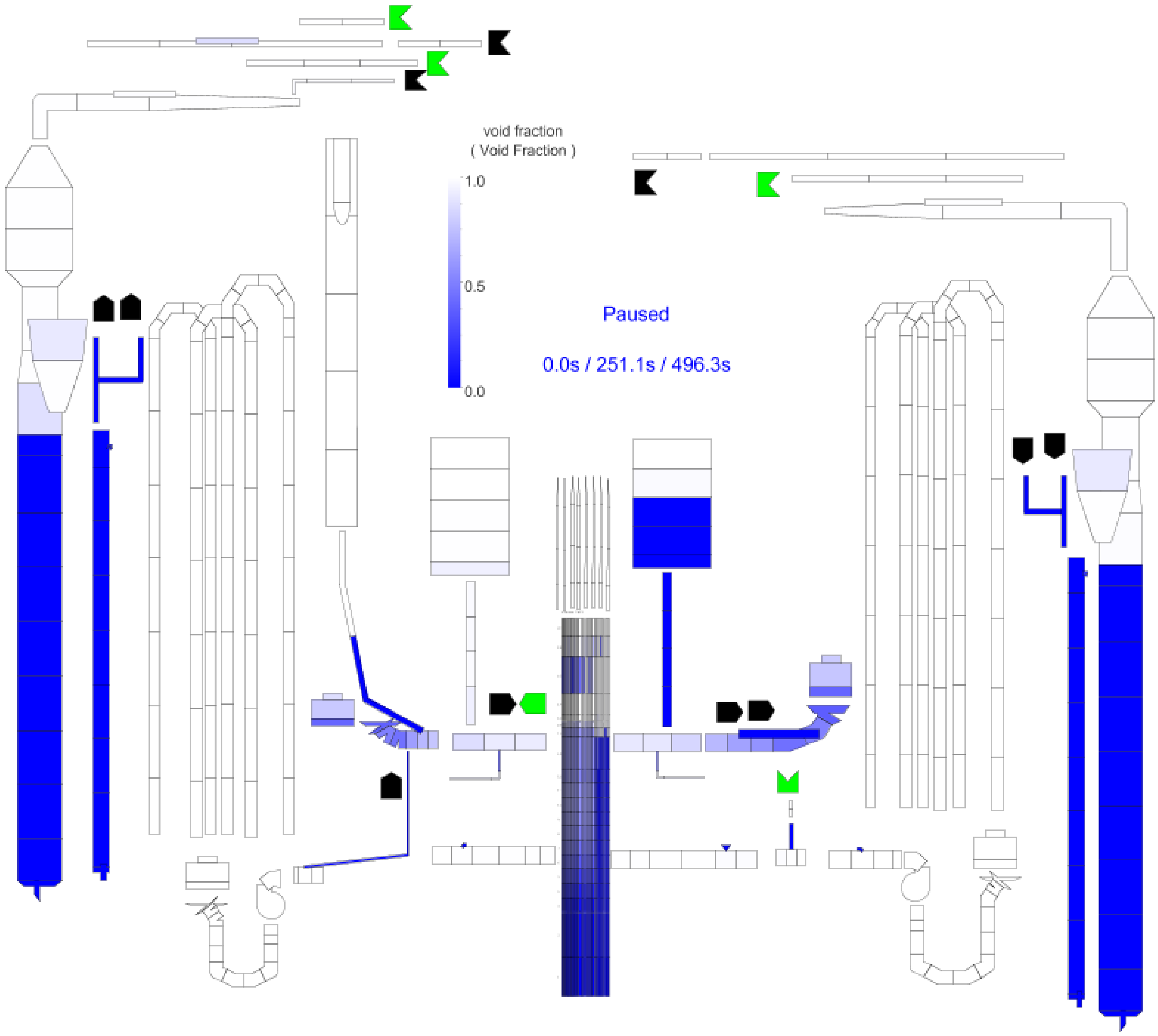

Figure 23 Void Fraction in the LSTF at 251 s

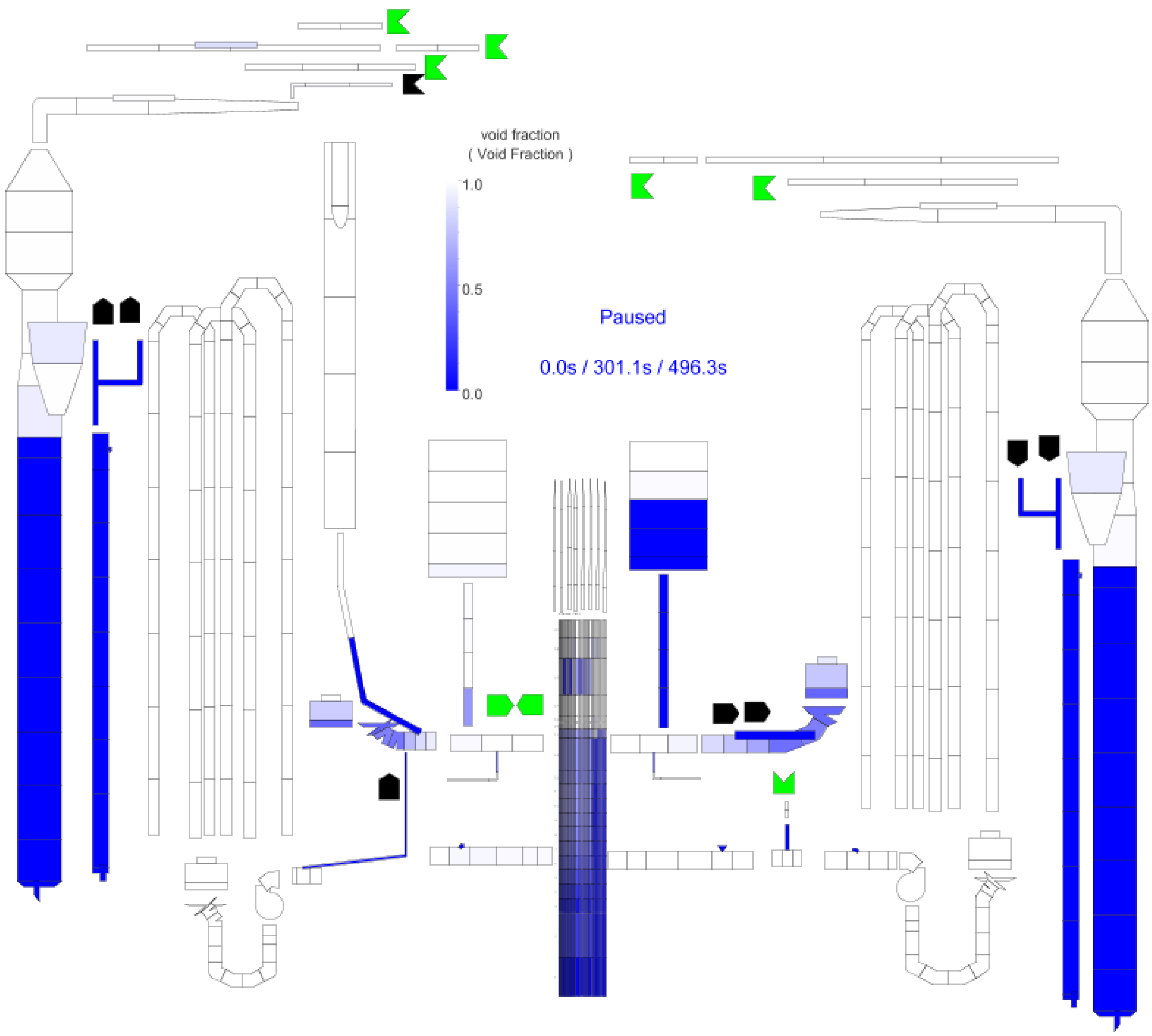

Figure 24 Void Fraction in the LSTF at the End of the Transient

6 CONCLUSIONS

In this work the Test 2, performed in the frame of the OECD/NEA ROSA-2 Project, has been analyzed. Test 2 was conducted in the Large Scale Test Facility (LSTF) and simulates the thermal-hydraulic responses during a PWR 17% cold leg Intermediate Break Loss-Of-Coolant-Accident.

The simulation results show that TRACE5 patch 2 can successfully reproduce the conditions of this IBLOCA. The main system variables are well reproduced in agreement with experimental data. The main thermal-hydraulic phenomena have been reproduced:

- Relatively large size of break results in a fast primary depressurization;
- Break flow turns from single-phase liquid to two-phase flow in a very short time after the break;
- High break flow rate produces a flashing effect in the core, which rapidly decreases its liquid level;
- Large temperature excursion appears in the core inducing the actuation of the core power protection system.

The main discrepancies between the simulation and the experimental results are observed from 50 to 150 s, when both hot legs are empty (in TRACE5). During the refill (from 250 to 350 s), TRACE5 does not reproduce the experimental behavior. These discrepancies can be attributed to differences in the coolant inventory distribution.

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NRC FORM 335 (12-2010) NRCMD 3.7 — U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) **NUREG/IA-0503**
2. TITLE AND SUBTITLE **Post-Test Analysis of ROSA-2 Test 2 (IBLOCA) with TRACE**	3. DATE REPORT PUBLISHED MONTH: **February** YEAR: **2019** 4. FIN OR GRANT NUMBER
5. AUTHOR(S) S. Gallardo, A. Querol, M. Lorduy and G. Verdú	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Universitat Politècnica de València
Instituto Universitario de Seguridad Industrial, Radiofísica y Medioambiental
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The purpose of this work is to overview the results obtained by simulating a 17% cold leg Intermediate Break Loss-Of-Coolant Accident (IBLOCA) in the Large Scale Test Facility (LSTF) using the thermal-hydraulic code TRACE5 patch 2. This IBLOCA transient corresponds to the Test 2 (IB-CL-03 in JAEA) and it is performed in the frame of the OECD/NEA ROSA-2 Project. During this transient, the single-failure of the High and the Low Pressure Injection systems (HPI and LPI, respectively) in the broken loop and the total failure of Auxiliary Feedwater (AFW) are assumed.

A detailed model of the LSTF and the chronology of events following these assumptions have been developed with TRACE5 patch 2. A comparison between the simulation and the experimental results is provided throughout different graphs. Acceptable general behavior is observed in the entire transient.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
Accident Management (AM) Committee on the Safety of Nuclear Installations (CSNI) Nuclear Safety Council, Spain (CSN) Large Scale Test Facility (LSTF) High Pressure Charging Pump (PJ) High Pressure Injection Pump (PL) Intermediate Break Loss-of-Coolant Accident (IBLOCA)	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMSSION
WASHINGTON, DC 20555-0001

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