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International Agreement Report

Simulation of LSTF Hot Leg Break (OECD/NEA ROSA-2 Test 1) with TRACE Code: Application to a PWR NPP Model

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ABSTRACT

An analysis of LSTF Surge Line Break experiment (OECD/NEA ROSA-2 test 1) has been performed with TRACE code. This test, included within the OECD/NEA ROSA-2 project and performed in Large Scale Test Facility (LSTF), it is attempted to analyze the phenomenology and different accident management actions after the occurrence of a surge line (SL) break with failure of High Pressure Safety Injection (HPSI). The comparison between experimental data and the results obtained with TRACE code shows that, in general, the main phenomena are well reproduced.

Additionally, a broad analysis of Surge Line Medium Break Loss of Coolant Accident with HPSI failed in a Westinghouse PWR has been performed taking into account different accident management actions and conditions in order to check their suitability.

These works has been performed in the framework of OECD/NEA ROSA-2 and CAMP projects.

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 thermal-hydraulic 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 thermal-hydraulic 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, 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 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 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.

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 OECD/NEA PKL 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 OECD/NEA ROSA 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 OECD/NEA PKL and OECD/NEA ROSA 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 Large Scale Test Facility (LSTF) is a full-height, full-pressure and 1/48 volumetrically scaled simulator for a Westinghouse-type 4-loop (3423 MWt) pressurized water reactor (PWR) with primary and secondary coolant systems including an electrically-heated simulated core, emergency core cooling system (ECCS) and control systems for accident management (AM) actions. The maximum core power of 10 MWt is equivalent to 14% of the 1/48-scaled PWR rated power covering the scaled PWR decay heat after the scram.

The experiment 1 (IB-HL-01) of the OECD/NEA ROSA-2 project was conducted on November 19, 2009 at the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA). This test consisted of a surge line double ending guillotine break (DEGB) equivalent in size to a 17% cold leg break, and without High Pressure Safety Injection [7].

In this report, an extended analysis of surge line MBLOCA with HPSI unavailability for Westinghouse PWRs is presented. The analysis has been performed through 3 stages:

1. In a first stage, the *pre-test* simulation with TRACE code was performed, in order to obtain a first estimation of main parameters during transient.
2. In a second stage, the *post-test* simulation with TRACE code was performed, and then extensively evaluated through comparison with the experimental results.
3. Finally, similar transients of OECD/ROSA-2 test 1 have been simulated with the TRACE model of Almaraz NPP. This analysis takes into account different accident management actions and conditions in order to check their suitability.

The purpose of this work is to contribute to the validation of TRACE code and its ability to properly simulate transient conditions.

Both pre-test and post-test analysis of OECD/NEA ROSA-2 Test 1 have been carried out by menas of TRACE 5.0, as well as the simulation of such a test in the TRACE model of Almaraz NPP. The main findings of the comparison of TRACE results with the OECD/NEA ROSA-2 test 1 experiment are:

- Results of OECD/NEA ROSA-2 Test 1 have been well reproduced through successive improvements and modifications in the LSTF model. Main changes in order to obtain an adequate post-test simulation were: ACC nodalization; activation of offtake model in the surge line; and the adjustments of discharge coefficients, friction factors in the break line and the ACC discharge pipe.

- The results obtained from the simulations of Almaraz NPP show a similar primary and secondary pressure transient with respect to ROSA-2 Test 1, but it must be noticed that there are not observed Inadequate Core Cooling (ICC) conditions in the simulation of Almaraz NPP as in ROSA-2 test 1. This can be due to the design differences between LSTF and Almaraz NPP.

- Simulation of ROSA-2 Test 1 in Almaraz NPP model shows diverse grades of sensitivity to several parameter modifications. In order to perform the sensitivity analysis with respect to the beginning of secondary side depressurization and break area, damage domain (defined as the region of the space of uncertain parameters of interest where damage conditions are exceeded) has been built up. Damage domain is an useful tool in order to analyze regions of the space of uncertain parameters where damage or success conditions are achieved. This kind of analysis is part of a more wide methodology called Integrated Safety Analysis Methodology (ISA) which has been developed by the Modelization and Simulation area (MOSI) of Spanish Nuclear Safety Council (Consejo de Seguridad Nuclear, CSN).

ACKNOWLEDGMENTS

This work 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

ACC	Acumulator
AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
AM	Accident Management
CAMP	Code Applications and Maintenance Program
DD	Damage Domain
DEF	Damage Exceedance Frequency
DEGB	Double Ending Guillotine Break
DET	Dynamic Event Tree
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
FW	Feedwater
HPSI	High Pressure Injection System
ICC	Inadequate Core Cooling
ISA	Integrated Safety Assessment
JAEA	Japan Atomic Energy Agency
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LPSI	Low Pressure Injection System
LSTF	Large Scale Test Facility
MBLOCA	Medium Break Loss of Coolant Accident
MSIV	Mean Steam Isolation Valve
NSSS	Nuclear Steam Supply System
PCT	Peak Cladding Temperature
PD	Previous Damage
PV	Pressure Vessel
PWR	Pressurized Water reactor
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RV	Relief Valve
SG	Steam Generator
SL	Surge Line
UPM	Universidad Politecnica de Madrid
USAEC	US Atomic Energy Commission

1 INTRODUCTION

The Large Scale Test Facility (LSTF) is a full-height, full-pressure and 1/48 volumetrically scaled simulator for a Westinghouse-type 4-loop PWR of 3423 MWt with primary and secondary coolant systems including an electrically-heated simulated core, ECCS and control systems for accident management actions. The maximum core power of 10 MW is equivalent to 14% of the 1/48-scaled PWR rated power covering the scaled PWR decay heat after the scram.

The OECD/NEA ROSA-2 Project, which started in 2005 by the agreement between JAEA, OECD/NEA and thirteen member countries, determined to conduct a MBLOCA test (Test 1, IB-HL-01 in JAEA, [7]). Test 1 simulates a Hot Leg MBLOCA assuming a total failure of the HPSI with a break size equivalent to 17% cold leg break. The objective of the test was to study the effect of accident management (AM) action and to provide integral test data for assessment and development of advanced analytical codes.

In this report, a pre-test and post-test analysis of OECD/NEA ROSA-2 Test 1 using TRACE 5.0 code is presented. A description of the model inputs is given, and the comparison of measured and calculated results is discussed. Simulation results of equivalent transient conditions implemented in the model of a Westinghouse-design Spanish 3-loop plant, are also discussed in the report.

The purpose of this analysis is to contribute in the validation of TRACE code and its ability to properly simulate hot leg medium break LOCA transients.

2 DESCRIPTION OF LSTF AND OECD/NEA ROSA-2 TEST 1

2.1 Description of LSTF

LSTF is a large scaled-down model of a four loop Westinghouse PWR design of 3423 MWt (Reference plant: Tsuruga unit-2 NPP), Figures 1 to 6. Its main characteristics are:

- LSTF has 2 loops, instead the 4 loops of the reference PWR.
- Elevations are scaled 1/1 while the volumes, power and mass flows are scaled 1/48 for the Reactor Coolant System (RCS) and 1/145 for the secondary system.
- The reactor core is modeled by a bundle of electrically heated rods with a maximum power of 10 MWt (14% of Tsuruga-2 rated power).
- Maximum operating pressure is 180 bar.
- Each steam generator has 141 U-tubes (4.2% of Tsuruga-2 U-tubes for SG), of original size and material.

As LSTF is full scale in height, the natural circulation phenomena during accident conditions is suitable to reproduce, so most of the different behaviors of a PWR during accident can be estimated with fidelity.

One of the most interesting studies done usually in the facility are those related with the role of operators in the accident management. As the control systems are implemented, the strategies are quite much the same as in a real plant transient, so the impact of the different timing and implementation of the manual actions can be measured *on line* as the plant status is monitored during the transients.

Both, the ability to reproduce the thermal-hydraulic phenomena and the operator action allow the detailed analysis of several accident scenarios like small break LOCA, steam generator U-tube rupture, main steam line break, etc.

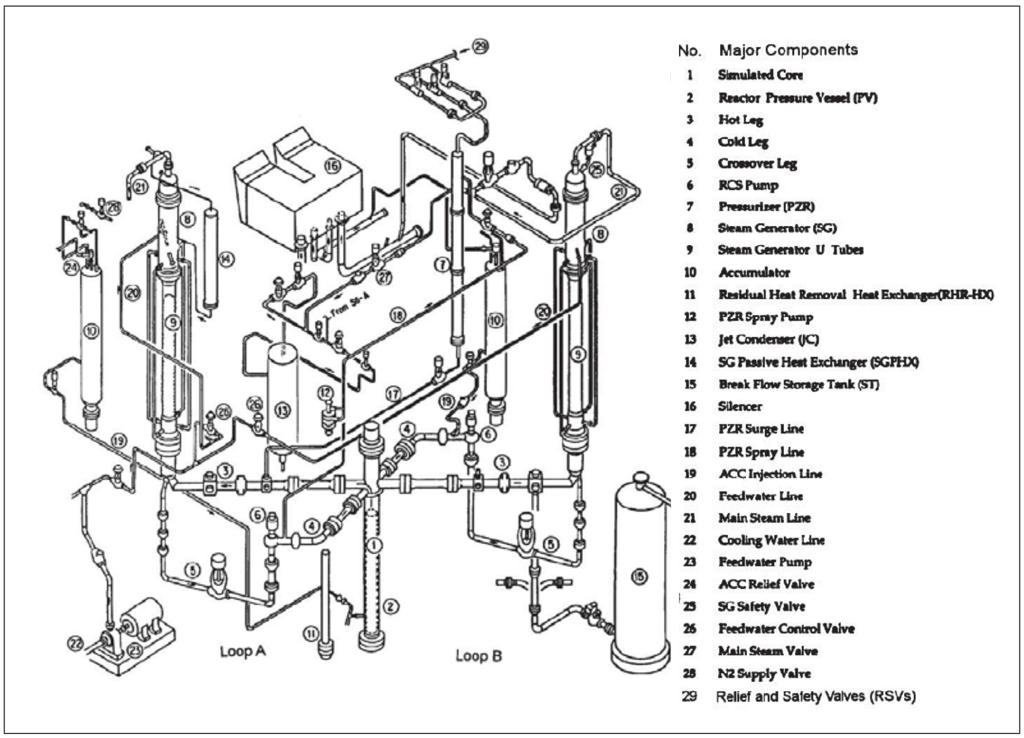


Figure 1: Large Scale Test Facility (LSTF)

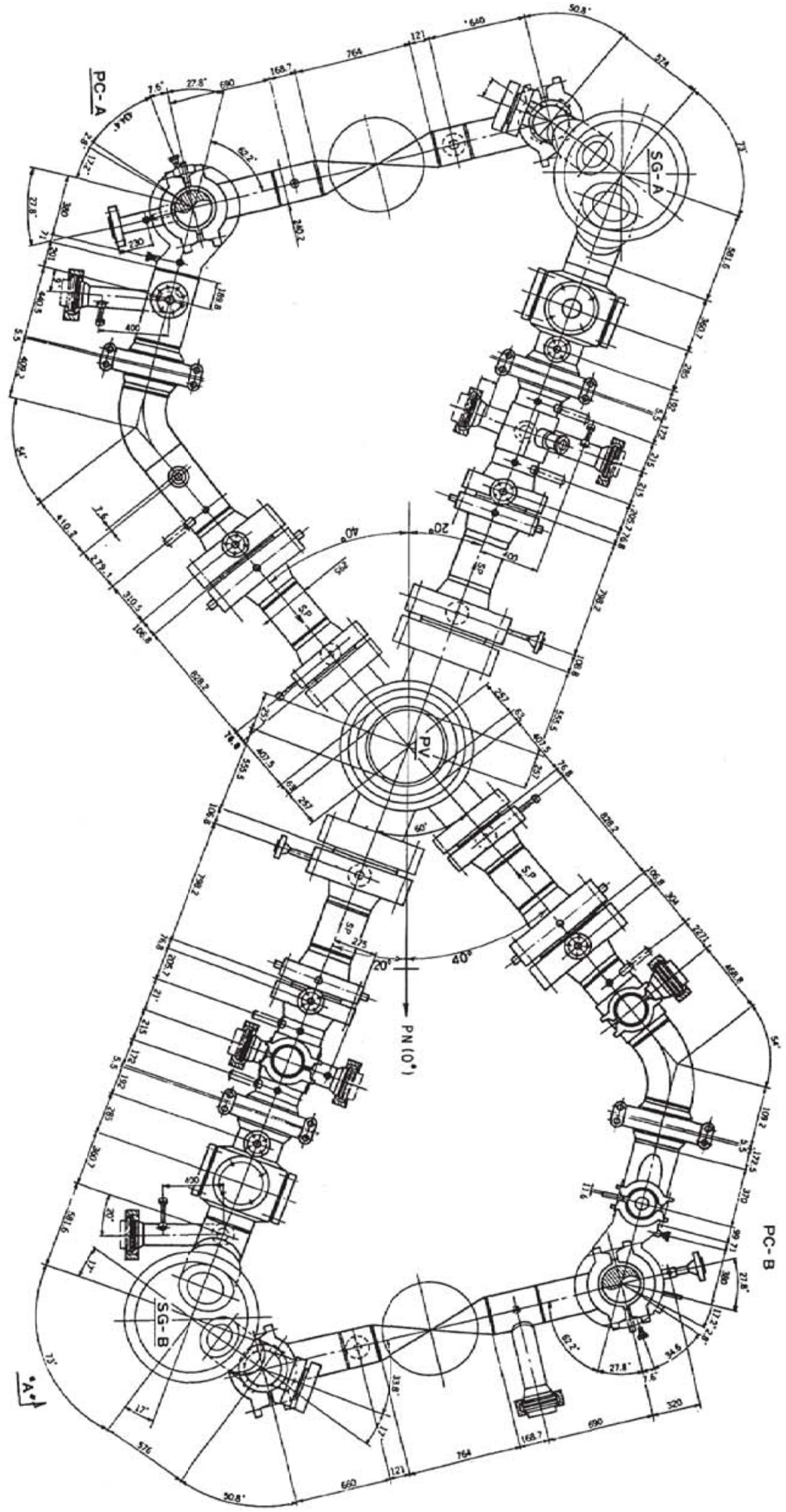


Figure 2: LSTF primary side (plan view)

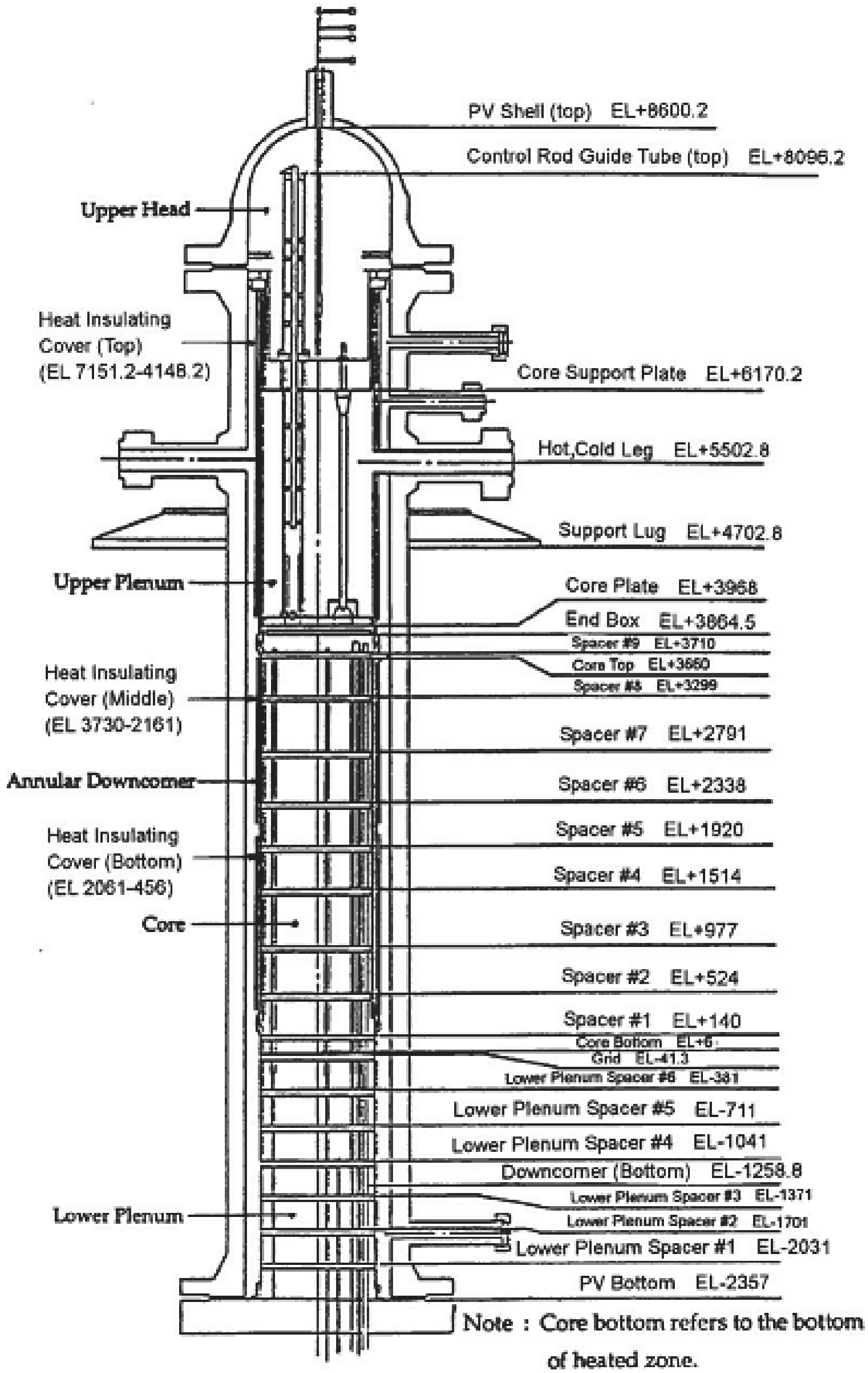


Figure 3: LSTF pressure vessel (elevation view)

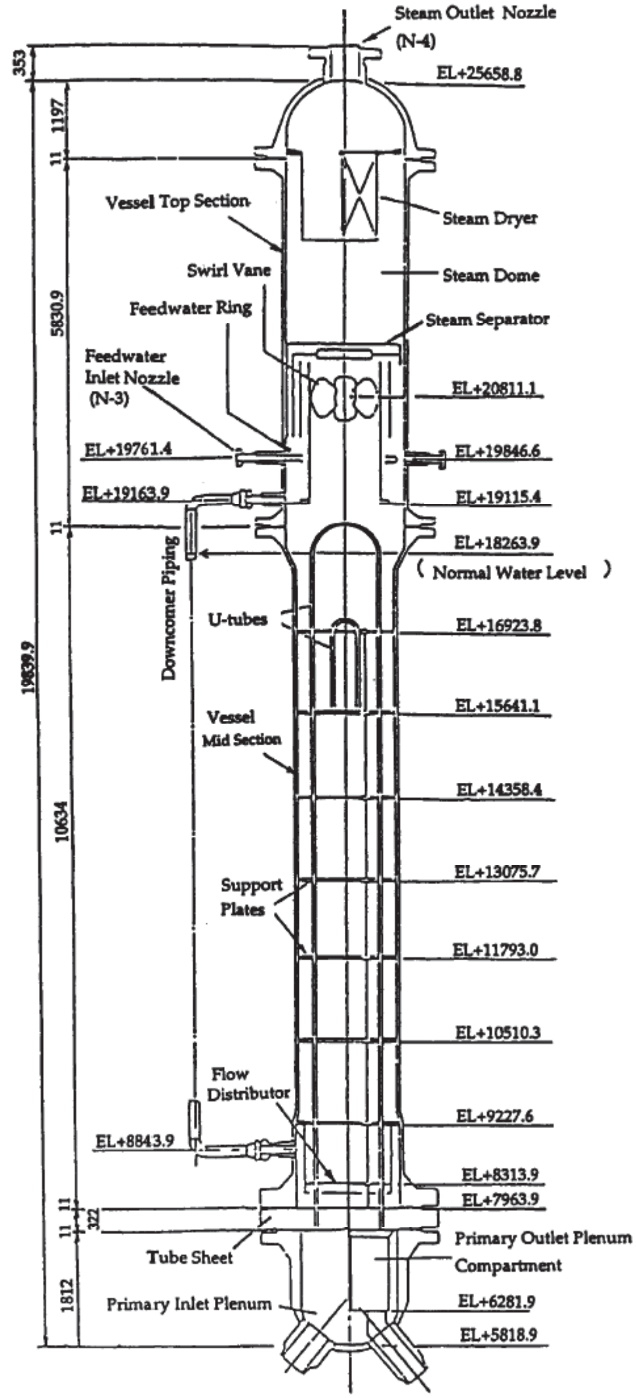


Figure 4: LSTF steam generator (elevation view)

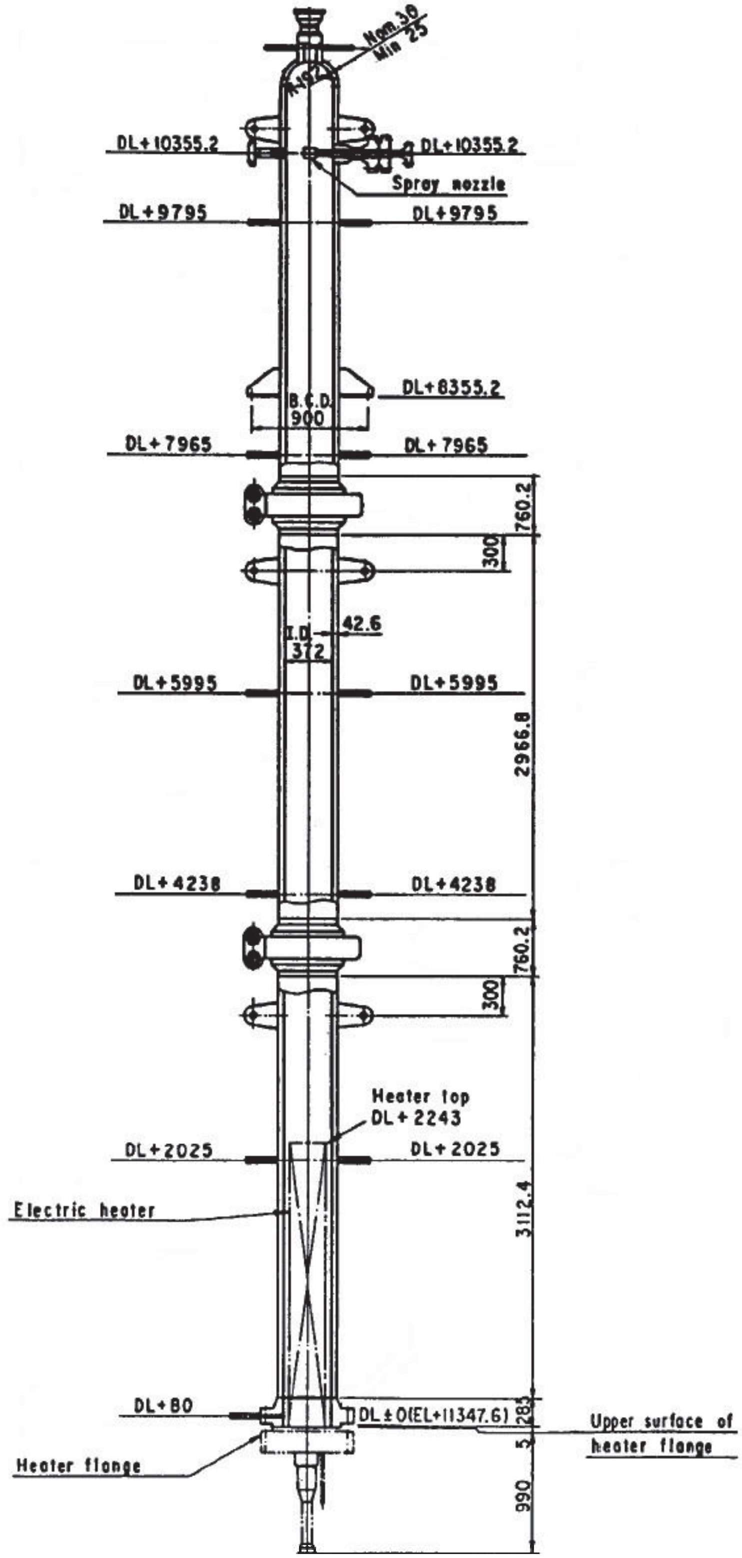


Figure 5: LSTF pressurizer (elevation view)

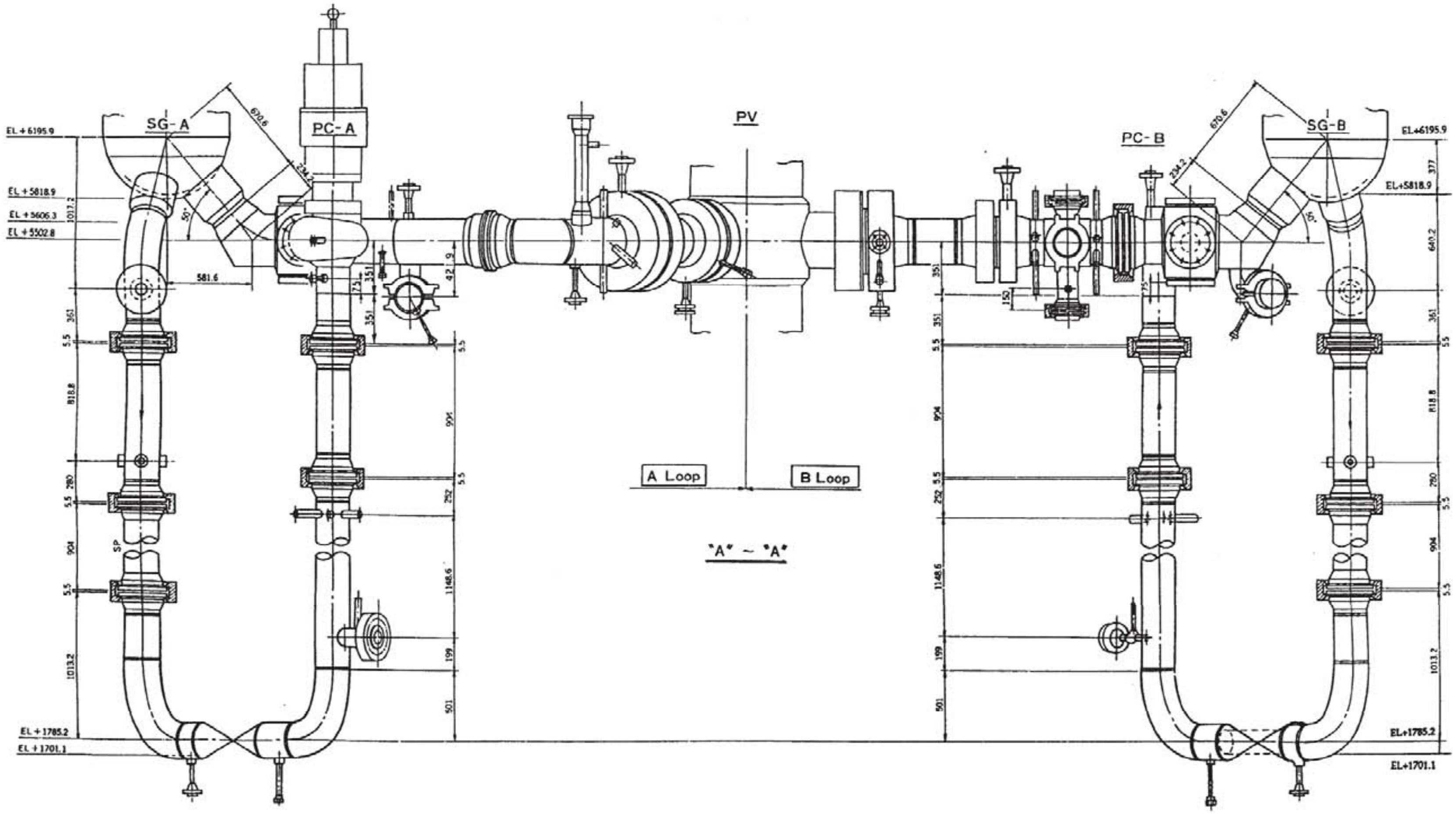


Figure 6: LSTF hot and cold legs (elevation view)

2.2 Description of OECD/NEA ROSA-2 Test 1

The main objective of OECD/NEA ROSA-2 Project was to analyze thermal-hydraulic issues relevant to light water reactors (LWR) safety by using Large Scale Test Facility (LSTF). As a part of this project ROSA-2 test1 was performed in LSTF during 2009. This test consists of a double ended guillotine break in the surge line of the pressurizer, with unavailable high pressure safety injection (HPSI) and reactor coolant pumps being tripped simultaneously to reactor scram.

Other conditions and specifications for the test are the following ones:

1. Loss of off-site power coinciding with scram.
2. Total failure of HPSI system when demanded.
3. Auxiliary feedwater (AFW) system totally failed to actuate.
4. ACCs and LPSI fully actuated.
5. Non-condensable gas inflow from ACCs.
6. Set point pressures for opening and closure of SG relief valves (RVs) in both loops are, respectively, 8.03 MPa and 7.82 MPa.
7. Break size is 17% cold leg equivalent (diameter of 41.28 cm).

Steady-state conditions of the test are summarized in Table 1. Chronology of major events in Test 1 is shown in Table 2 and Figure 7. It must be noticed that ROSA data are shown normalized in figures because they are proprietary at the present time.

Previously to the test simulation, the PZR was isolated and therefore pressure could not be measured in PZR. Due to that fact, scram signal was assumed to be generated at 4 seconds after the break opening. Scram signal triggered RCPs and turbine trips, power decay simulation and the isolation of main steam line and main feedwater (FW).

The relatively big size of the break results in a fast depressurization and loss of inventory in the RCS, leading to early core uncovery (see Figure 7 and Table 2). PCT started to increase at 10 seconds after the actuation of ACCs (154 seconds after break) and a maximum PCT of 607 K was reached at 182 seconds. However, the fast actuation of ACCs (and later the LPSI injection) avoids higher PCT values.

Items	Specified
Pressure Vessel	
Core power(MW)	10±0.07
Primary Loop	
Hot Leg Fluid Temperature(K)	598.1±2.75
Cold Leg Fluid Temperature(K)	562.4±2.75
Mass Flow Rate (kg/s / loop)	24.3±1.25
Downcomer to Hot Leg bypass (kg/s)	0.049±0.01
Pressurizer	
Pressure(MPa)	15.5±0.108
Liquid Level(m)	7.2±0.25
Accumulator System	
Pressure(MPa)	4.51±0.054
Temperature (K)	320±2.3/2.4
Secondary loop	
Secondary-Side Pressure (MPa)	7.3±0.054
Secondary-Side Liquid Level(m)	10.3±0.38
Steam Flow Rate (kg/s)	2.74±0.1
Main Feedwater Flow Rate (kg/s)	2.74±0.05
Main Feedwater Temperature(K)	495.2±2.63
Auxiliary Feedwater Temperature(K)	310±2.37

Table 1: Steady State Conditions. OECD/NEA ROSA-2 test 1

Event	**Time (s)**
Isolation of PZR	1940
Break valve opening	2000
Scram signal	2004
Initiation of RCP coastdown	2005
Termination of main FW	2005
Initiation of core power decay simulation	2020
Initiation of ACCs discharge	2154
Maximum cladding temperature	2182
Initiation of LPSI	2504

Table 2: Chronology of main events in OECD/NEA ROSA-2 test 1

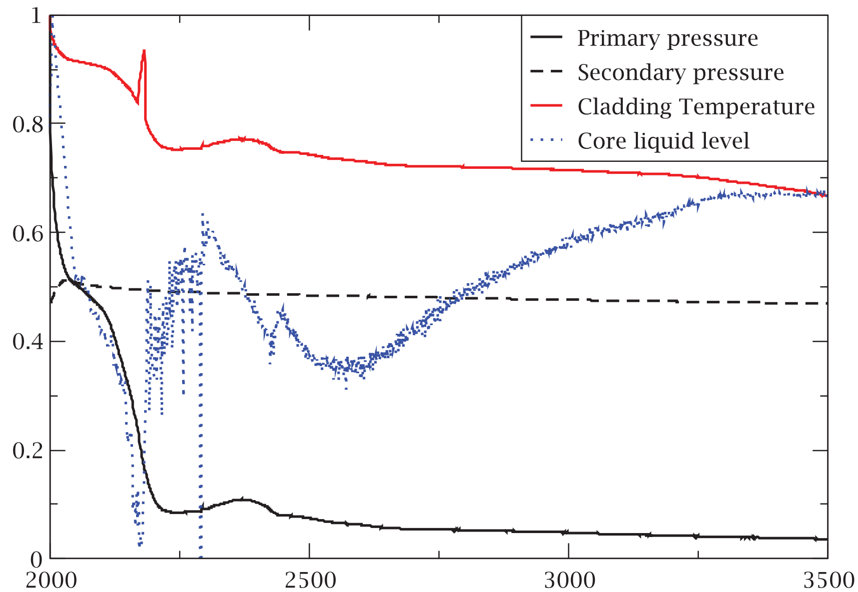


Figure 7: Evolution of main variables in OECD/NEA ROSA-2 Test 1

3 DESCRIPTION OF TRACE MODEL OF LSTF

The group of Technical University of Madrid (UPM) is working with TRACE model of LSTF facility since February 2006. The development of the TRACE model of LSTF is based on the TRAC-PF1 model (see Figure 8) delivered by the Japan Atomic Energy Research Institute (JAERI) to the participants of OECD/NEA ROSA project.

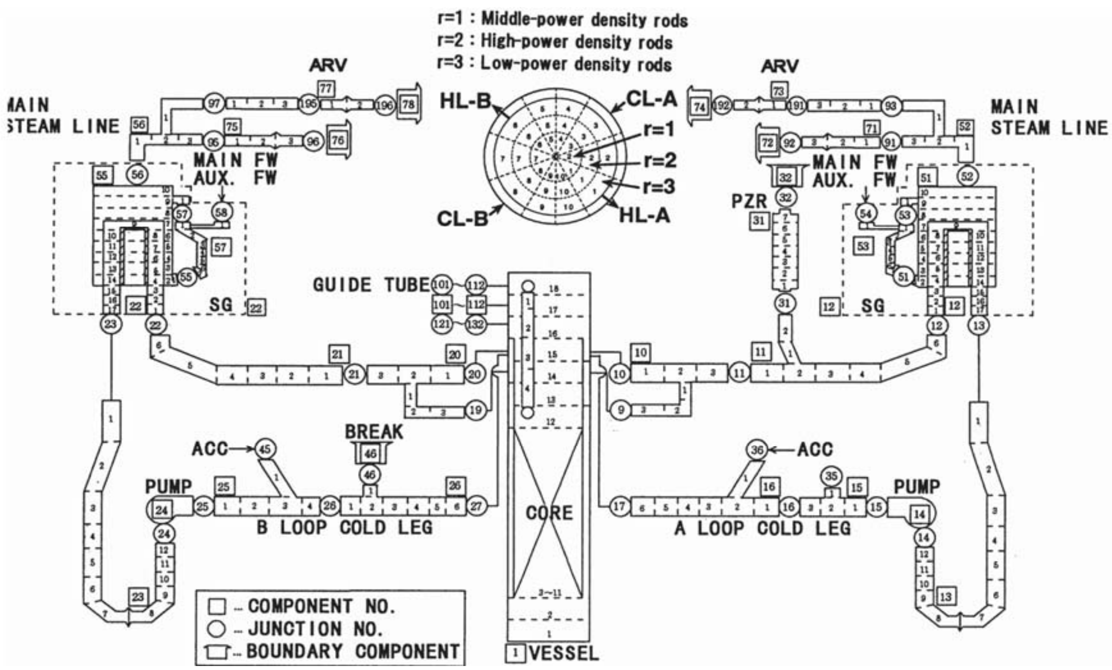


Figure 8: TRAC-PF1/MOD1 model of LSTF (JAERI)

Nowadays, TRACE model of LSTF contains following systems and equipments:

- Primary system:
  - Vessel
  - Steam generators U-tubes (primary-side)
  - Hot legs, intermediate legs and cold legs
  - Pressurizer and surge line
  - Reactor coolant pumps
- Secondary system:
  - Steam generators (secondary-side)

  - Main steam lines

- Safeguard systems:
  - Accumulators system
  - High and low pressure safety injection systems
  - Auxiliary feedwater system

- Instrumentation and control systems:
  - Primary: Reactor trip logic; Reactor coolant pump trip control; Pressurizer level and pressure control; High and low pressure safety injection fail control and Accumulators signal control.
  - Secondary: Turbine trip logic; Secondary-side pressure and level control; Relief valves control (for secondary depressurization purposes); Main feedwater system activation control and Auxiliary feedwater system activation control.

Initially, in a first stage, LSTF TRAC-PF1 model was adapted to TRACE code. Later on, in a second stage, many improvements were included. The final LSTF model for TRACE is showed in Figure 9. A briefly description of changes and upgrades performed in both stages for obtaining the final model can be found in Ref. [17]. Main changes among those performed in the LSTF model were:

1. Old VESSEL component of TRAC-PF1 model was migrated to TRACE.
2. Old STGEN component of TRAC-PF1 model was migrated to TRACE model as a set of components (TEEs and PIPEs).
3. The steam generators recirculation ratio was adjusted.
4. Total mass flow was adjusted in primary loops using FRIC parameters and rated head in RCP.
5. Volume vs. height plots were checked with respect the facility data and some discrepancies were corrected.
6. New 2D pressurizer model was created in order to simulate adequately the natural circulation inside the pressurizer and also to avoid excessive cooling in upper cells of the model during long quasi steady state transients.
7. Pressurizer level and pressure control systems were added to fix the steady state more adequately.
8. New proportional and base heaters with more detail were also added in the pressurizer.

9. Heat losses of the whole model were adjusted.

10. The mass flow rate from the downcomer to the upper head of the vessel has been adjusted to the specified one (0.3% of the downcomer vessel total mass flow).

11. The temperature in the Upper Head of the Vessel has been adjusted to the measured one (about 586 K).

12. ΔP along the model was revised with reasonable results.

13. Control blocks, signal variables and trips were renumbered to avoid misunderstandings reading output data.

14. New signal variables to measure heat losses, liquid level in upper head, core and PZR, and to measure surface temperature in core heaters.

15. Several masses of HTSTR components have been corrected, i.e. U-tube support plate.

16. The OFFTAKE model was activated in the connections of the valves that simulate breaks in different localizations of LSTF model.

17. Adjustment of several RFRIC factors.

18. REFLOOD model was activated.

19. Improved nodalization and model dimensions for better correlation between model height/volume and LSTF height/volume:

    - Modified vessel with 19 levels, improved nodalization and more adjusted lower and upper plenum, core, lower and upper head and downcomer.
    - Modified cold and hot legs in loop A and loop B.
    - Improved steam generators model.

20. An animation mask was created with SNAP tool (Refs. [2] and [1]) (see Figure 10). This mask allows performing videos of the simulations, which allows interpreting easily the transient behavior.

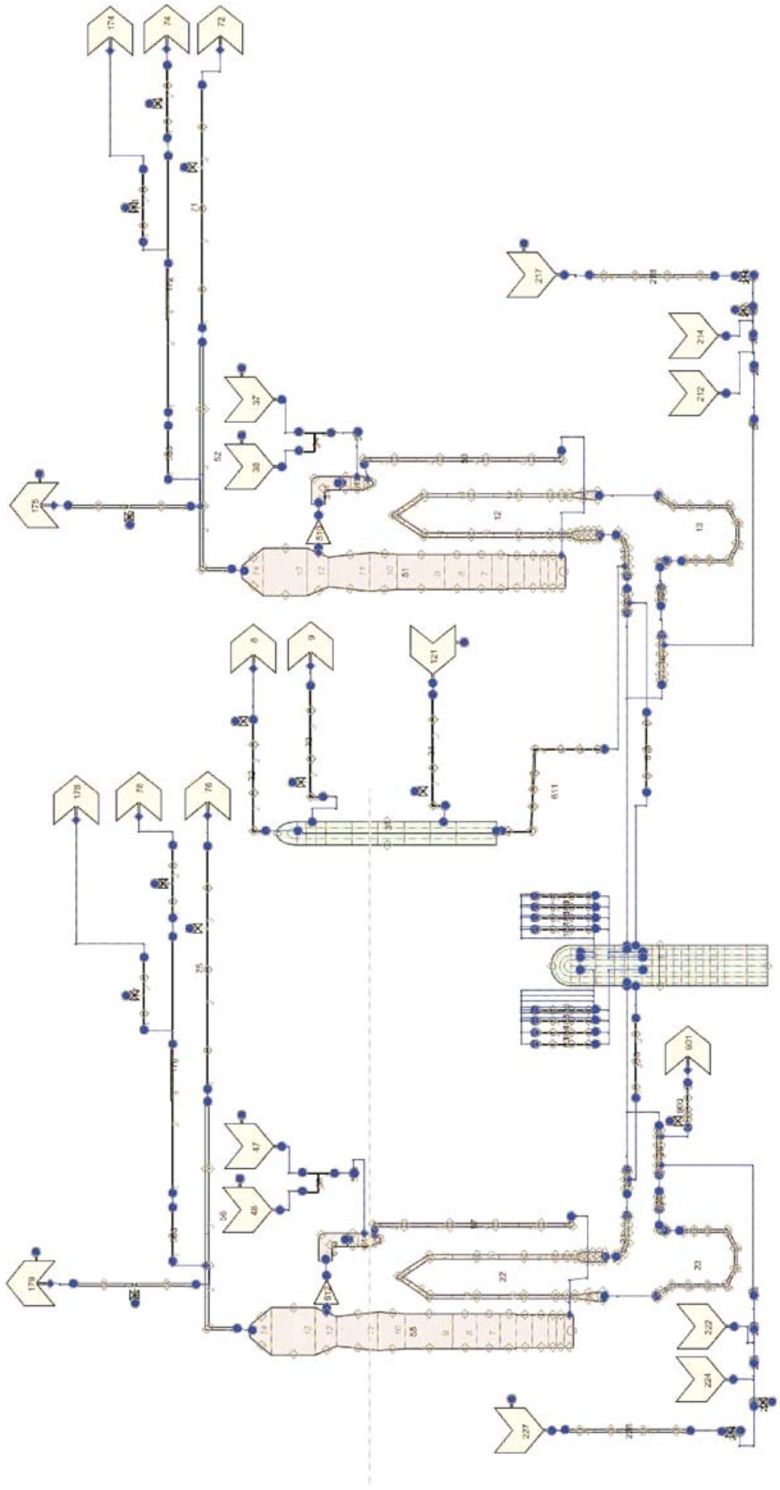


Figure 9: TRACE model for LSTF

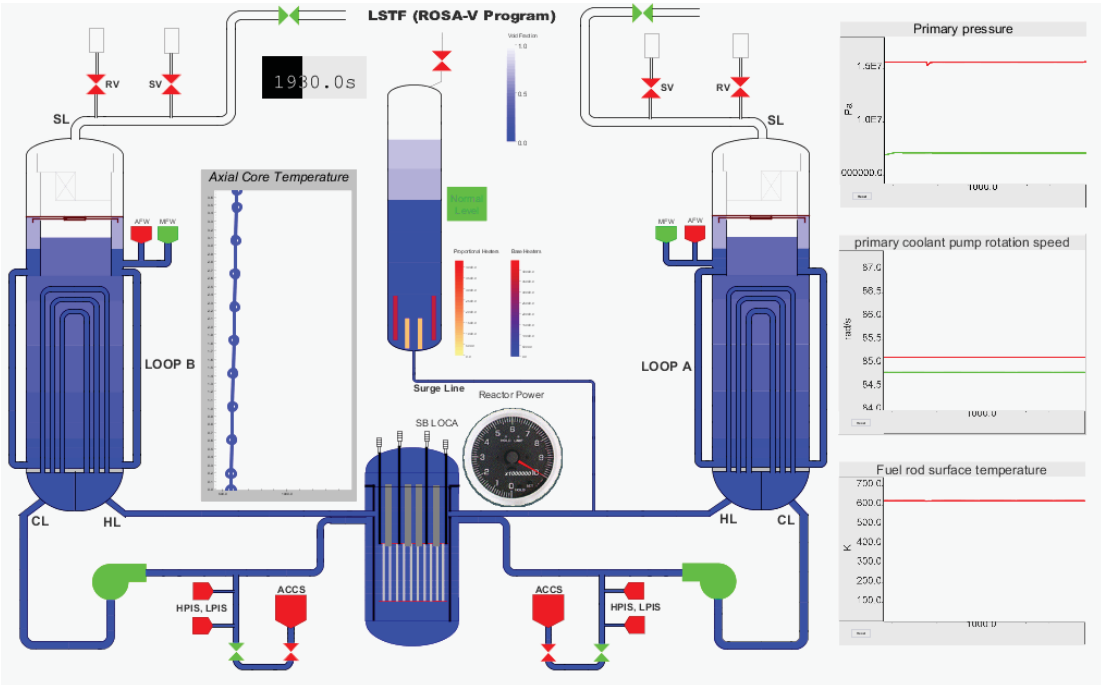


Figure 10: SNAP mask for LSTF TRACE model

4 COMPARISON OF EXPERIMENTAL RESULTS AND SIMULATIONS OF OECD/NEA ROSA-2 TEST 1

First stage of the OECD/NEA ROSA-2 test 1 simulation consists of the blind test simulation, which was performed before test 1 was carried out in LSTF, on November 19, 2009. Once experimental results were obtained, comparison between pre-test and experimental data was carried out. Later on, model was revised and improved in order to reproduce better experimental results in a post-test simulation, considering new assumptions which lead to improve the transient simulation.

4.1 Pre-test simulation of ROSA-2 test 1

In order to obtain expected steady state values, Table 1, adjustment of friction factors was performed for better reproducing the bypass flow between downcomer and hot leg. Steady state values obtained with modified LSTF model adjust very well to ROSA-2 test 1 initial and boundary conditions, see Table 3.

Break in experimental test was simulated with a nozzle of 0.041 m inner diameter oriented upward. In TRACE model, break line included consists of 9 cells (with valve itself between second and third cell), see Figure 11 for more details.

Comparison between main parameters of pre-test and experimental data are showed in Figures 12 to 14, where results are depicted after being normalized. The results of pretest simulation show that RCS depressurizes slightly earlier than in experimental test, see Figure 12. This fact is due to excessive flow mass discharge through the break (see Figure 13), slightly higher at first in pre-test simulation. As result of this difference, the sequence evolves in a different way, ACC discharge takes place earlier and the temperature does not undergo a sudden increase as a result of core uncovery, as shown in Figure 14.

This difference was corrected in next stage, post-test simulation, which is described in next section.

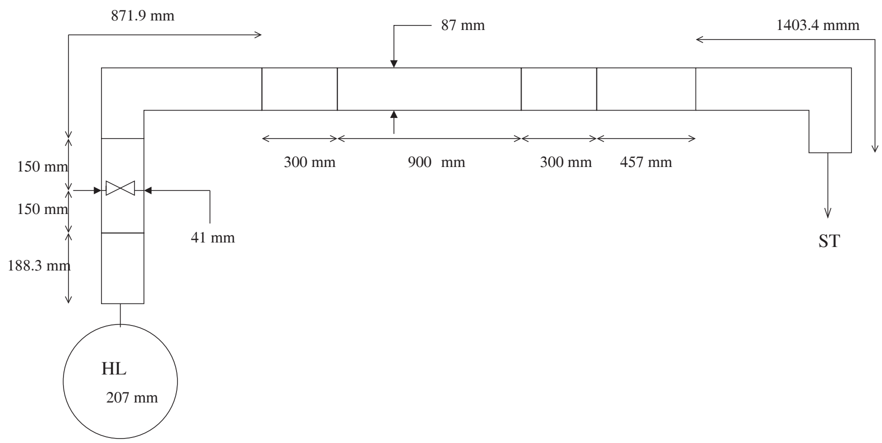


Figure 11: Break line scheme for pre-test in LSTF TRACE model

Items	Specified	TRACE (with/without PZR)
Pressure Vessel		
Core power(MW)	10±0.07	10.00
Primary Loop		
Hot Leg Fluid Temperature(K)	598.1±2.75	599.1/599.0
Cold Leg Fluid Temperature(K)	562.4±2.75	561.7/561.78
Mass Flow Rate (kg/s / loop)	24.3±1.25	23.91/23.9
Downcomer To Hot Leg Bypass (kg/s)	0.049±0.01	0.049/0.049
Pressurizer		
Pressure(MPa)	15.5±0.108	15.5
Liquid Level(m)	7.2±0.25	7.2
Accumulator System		
Pressure(MPa)	4.51±0.054	4.51/4.51
Temperature	320±2.3/2.4	320.0/320.0
Secondary loop		
Secondary-Side Pressure (MPa)	7.3±0.054	7.14/7.14
Secondary-Side Liquid Level(m)	10.3±0.38	10.3/10.3
Steam Flow Rate (kg/s)	2.74±0.1	2.70/2.70
Main Feedwater Flow Rate (kg/s)	2.74±0.05	2.74/2.74
Main Feedwater Temperature(K)	495.2±2.63	495.2/495.2
Auxiliary Feedwater Temperature(K)	310±2.37	310/310

Table 3: Comparison between pretest and proposed ROSA-2 test 1 initial conditions

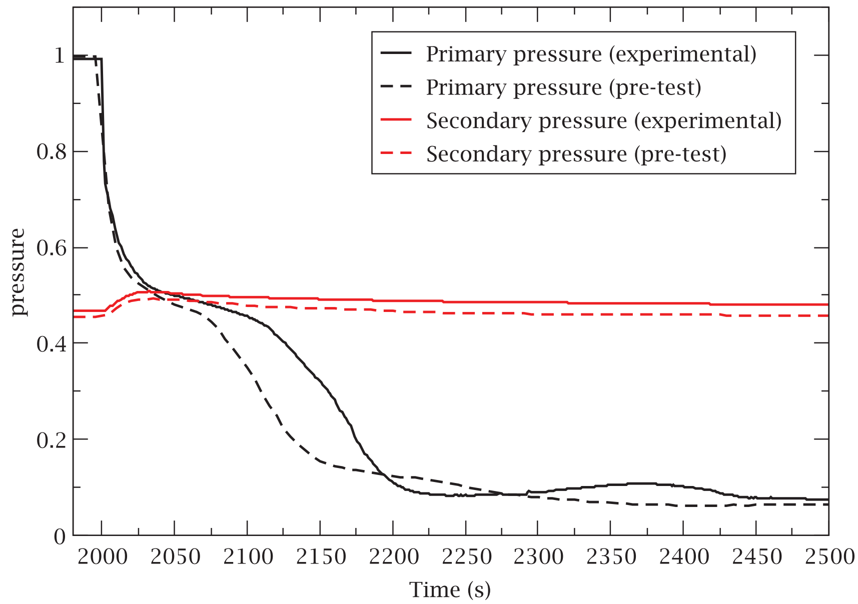


Figure 12: RCS and SG pressure (intact loop). Experimental and pre-test simulation

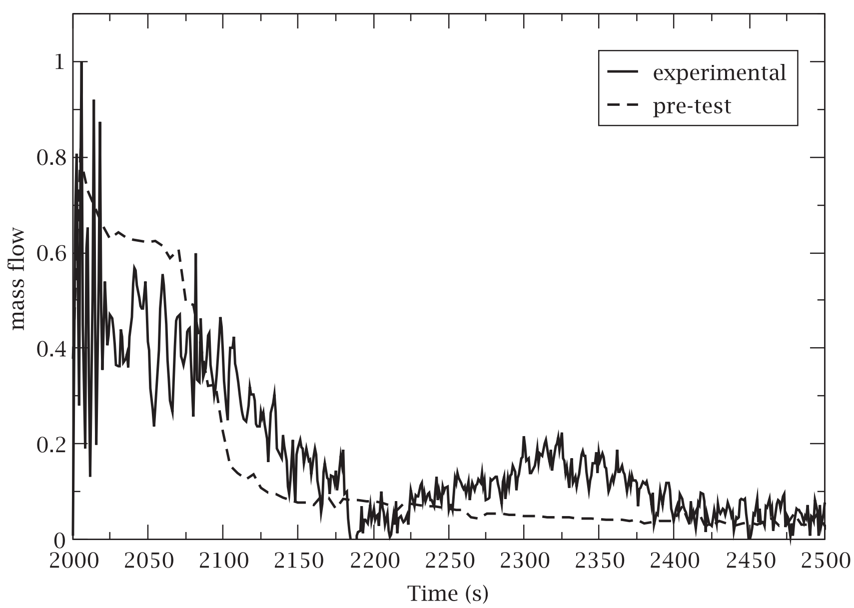


Figure 13: Discharge mass flow. Experimental and pre-test simulation

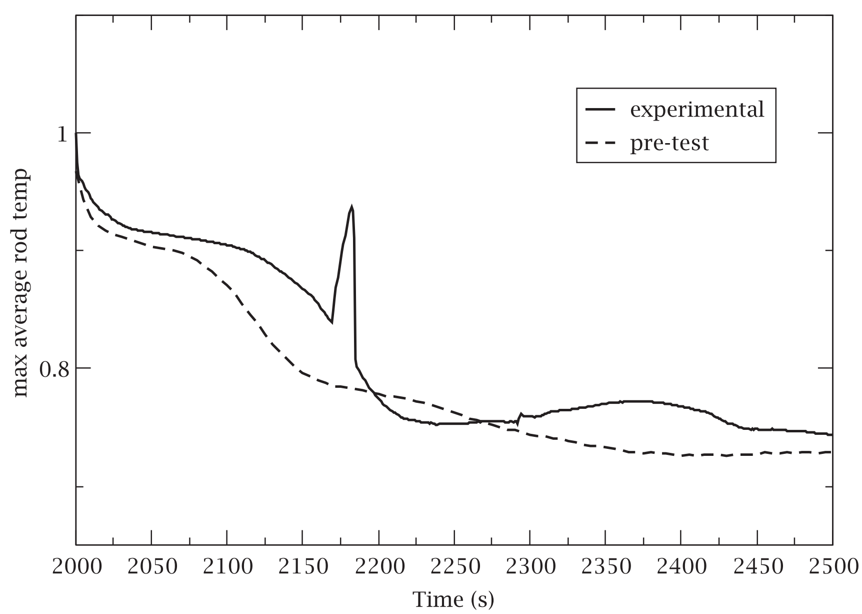


Figure 14: PCT. Experimental and pre-test simulation

4.2 Post-test simulation of ROSA-2 test 1

The aim of second stage, post-test simulation, is to minimize the discrepancies between experimental and pre-test results (those differences are shown in Figures 12 to 14).

With the intent to minimize those differences, the following modifications have been tried out in the model, provided they significantly affect simulations:

- Amendment of **two phase and single-phase coefficients**,
- Adjustment of **friction factors in the break line**,
- Activation of the stratified flow model (**offtake model**) in the surge line.
- Adjustment of **friction factors in the ACC discharge line**,
- **ACC renodalization**.

First step in post-test amendment consisted of adjusting break flow by means of **offtake model** activation, modification of **two-phase and single-phase discharge coefficients**, and modification of **friction factors** in the break line. The overall effect of activating **offtake model** and different **friction factors** is shown in Figure 15, where it can be noticed that simulated pressure becomes similar to the experimental one when offtake model and friction is adjusted in the discharge line.

It is found out that break mass flow is better simulated than in the pre-test case, although ACC discharge still takes place earlier than in the experiment, thus avoiding the PCT increase observed in the experiment. In order to correct this fact, ACC model was **renodalized**, and **friction factor** in the ACC discharge line was adjusted.

Final results of post-test simulation show a good agreement between post-test simulation and experimental data, see Table 4 and Figures 16 to 23. Therefore, these simulation results allow to confirm the capability of TRACE 5.0 in order to simulate this kind of transients. These results show a high sensitivity to ACC model, proving that this component need to be modeled carefully for practical applications.

Several snapshots of the ROSA-2 Test 1 video obtained through SNAP with the LSTF mask and the last TRACE simulation are shown in Figures 24 and 25, in order to provide a better understanding of sequence evolution. In these figures, void fraction is shown in blue-white scale on plant components, green/red color indicates on/off status of pumps, valves and main systems, and a red/blue scale shows the power of PZR heaters. Additionally, primary/secondary pressure, pump rotational speed and cladding temperature profile are depicted for each instant of the transient. The six pictures of those two snapshots correspond to the main phases of the sequence:

1. Steady-state (previous to break).
2. Surge line break and beginning of depressurization in RCS with isolated PZR and voiding inside vessel.
3. Advanced depressurization in RCS and emptying of SG U-tubes.
4. Core uncovery and ICC conditions.
5. Core reflood due to accumulators discharge.
6. LPSI injection (end of the simulation).

In the following section, an equivalent transient to ROSA-2 Test 1 is performed in Almaraz-NPP model, in order to compare results.

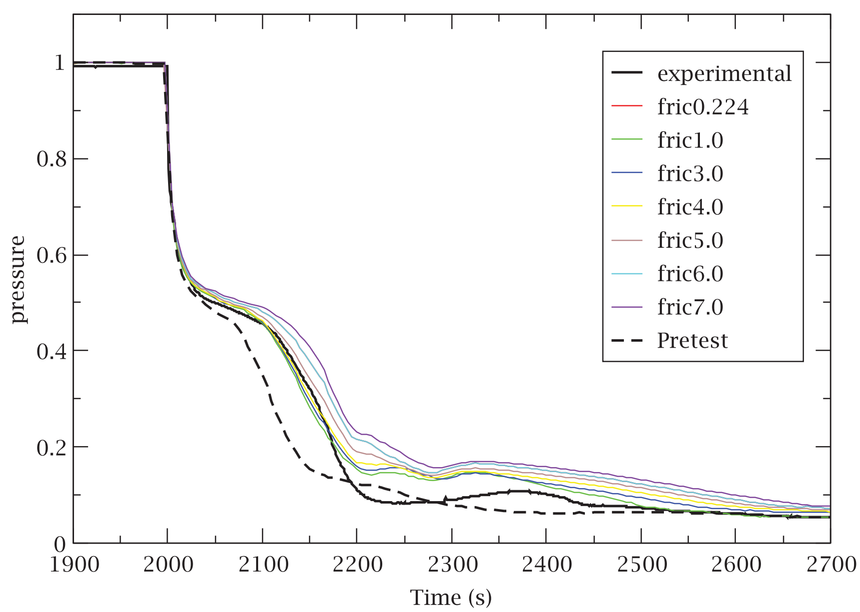


Figure 15: Normalized primary pressure. Friction factor sensitivity

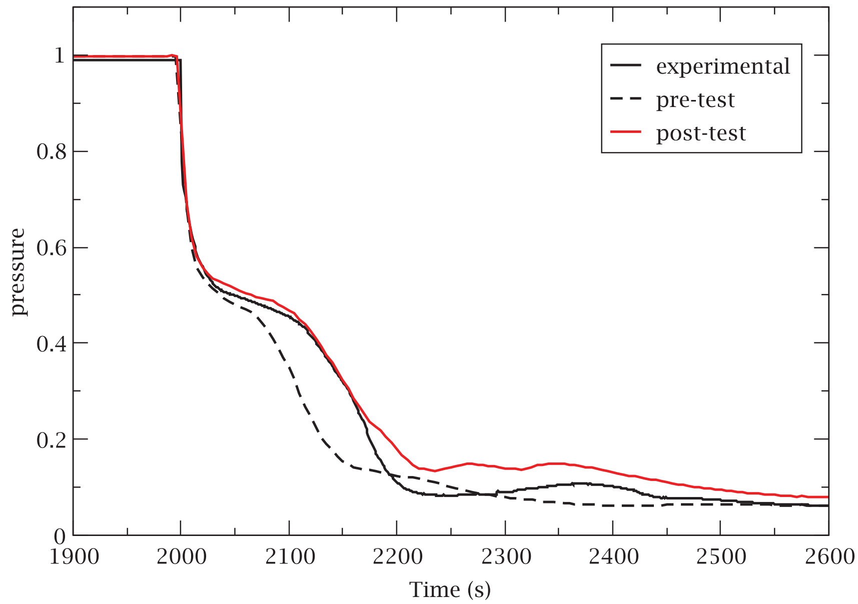


Figure 16: Normalized primary pressure. Pre-test, post-test and test data

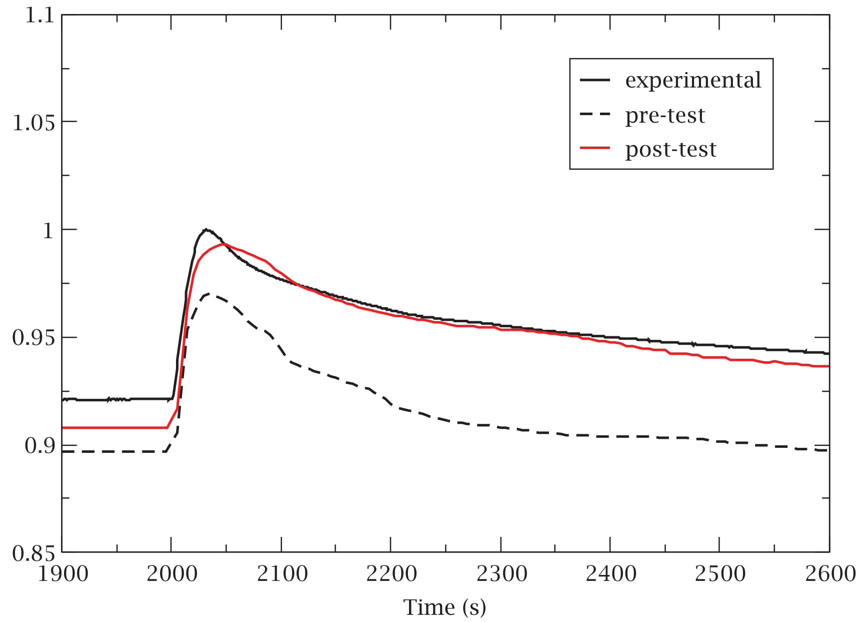


Figure 17: Normalized intact-loop secondary pressure. Pre/post-test and test data

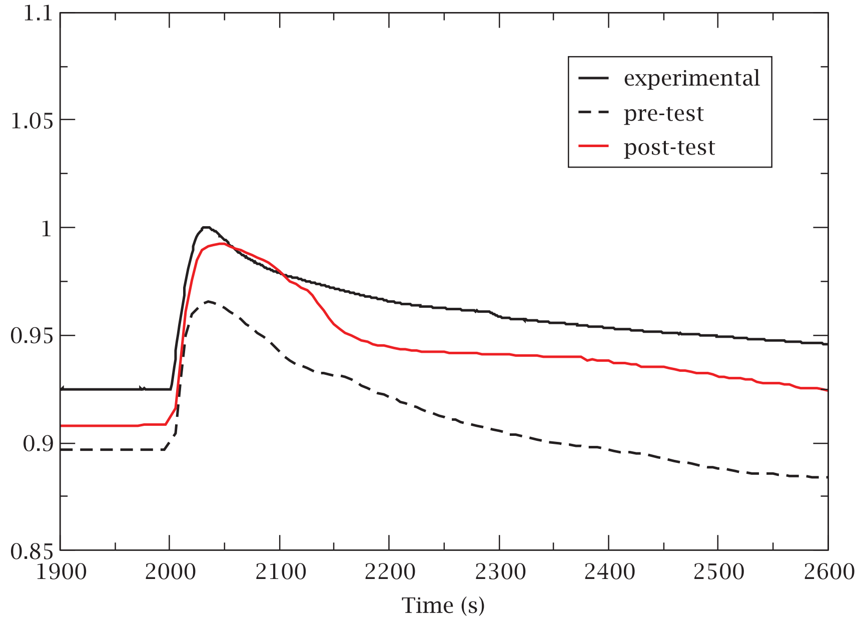


Figure 18: Normalized broken-loop secondary pressure. Pre/post-test and test data

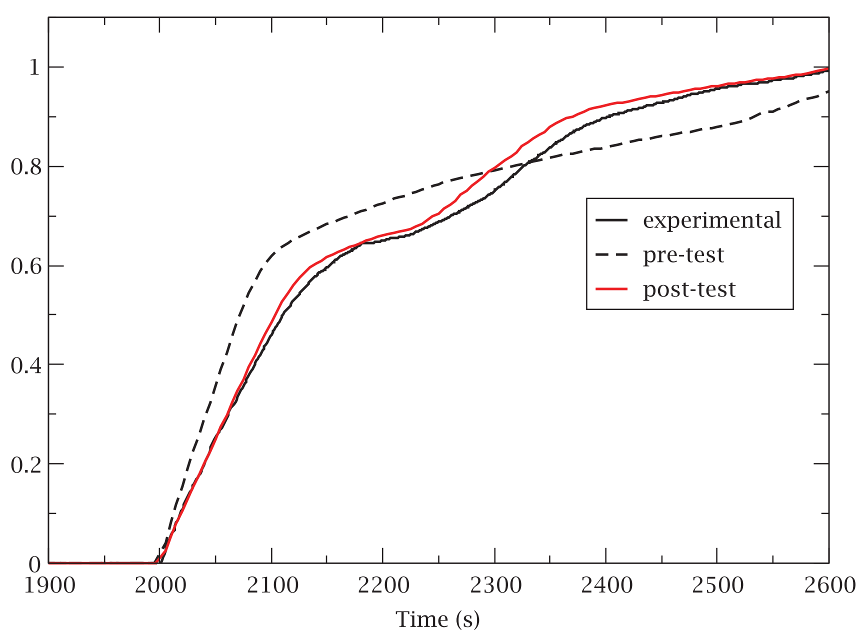


Figure 19: Normalized integrated break mass flow. Pre/post-test and test data

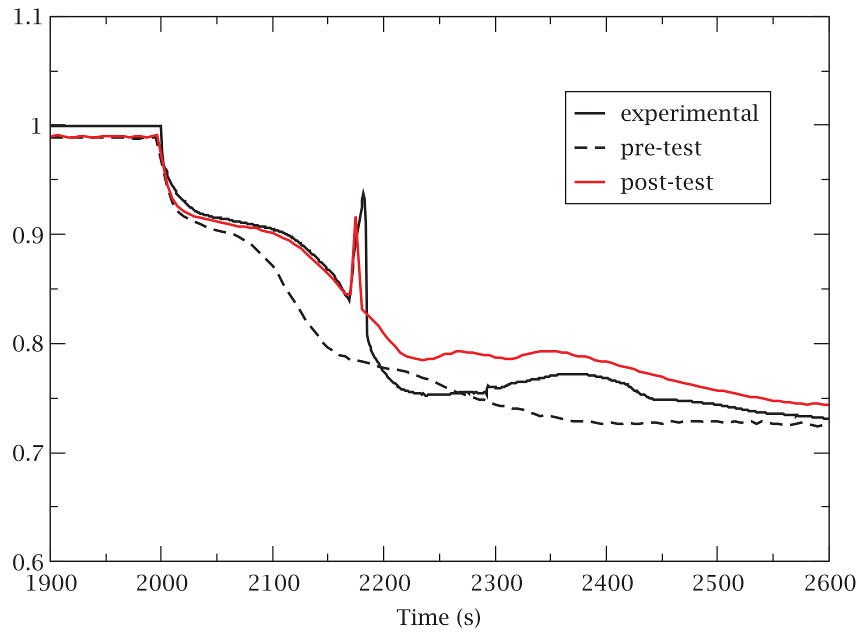


Figure 20: Normalized cladding temperature. Pre-test, post-test and test data

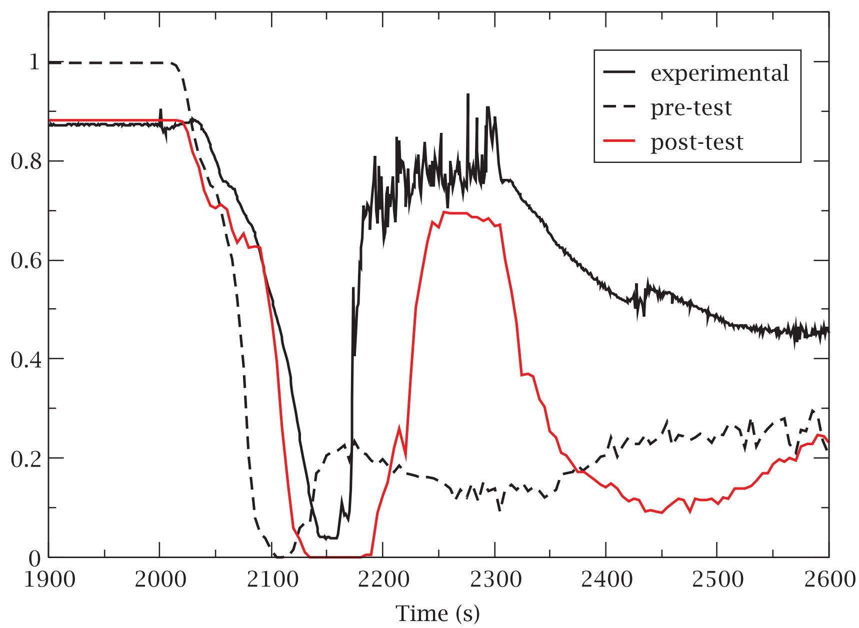


Figure 21: Normalized downcomer level. Pre-test, post-test and test data

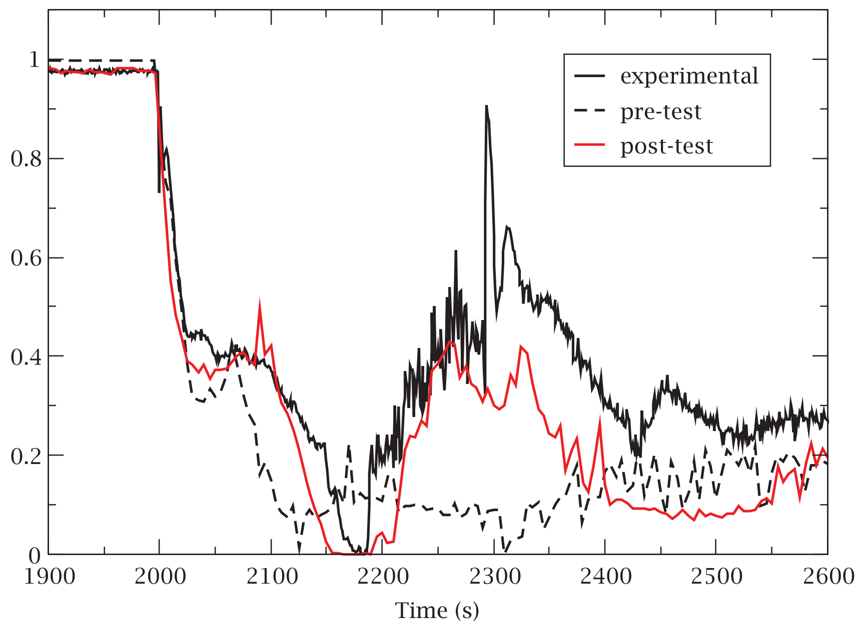


Figure 22: Normalized upper plenum level. Pre-test, post-test and test data

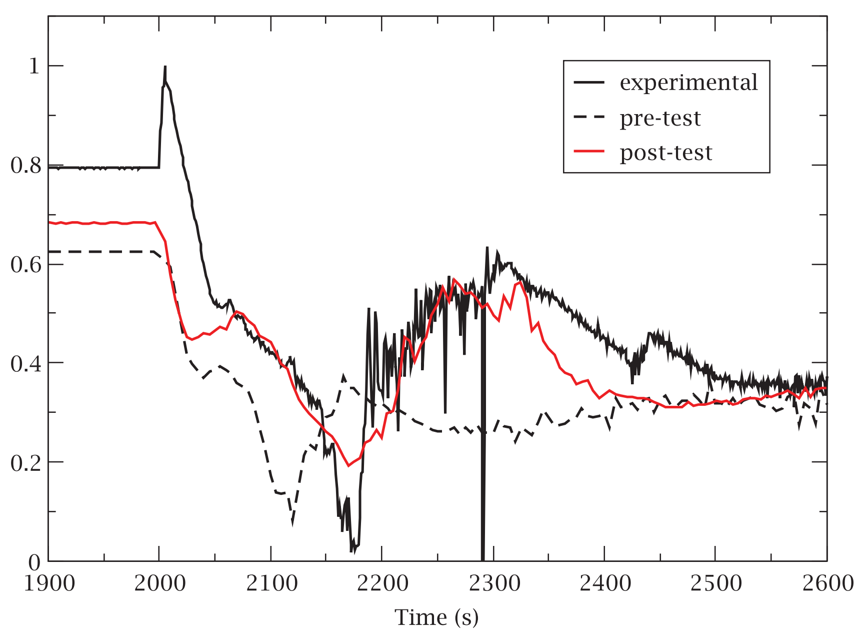
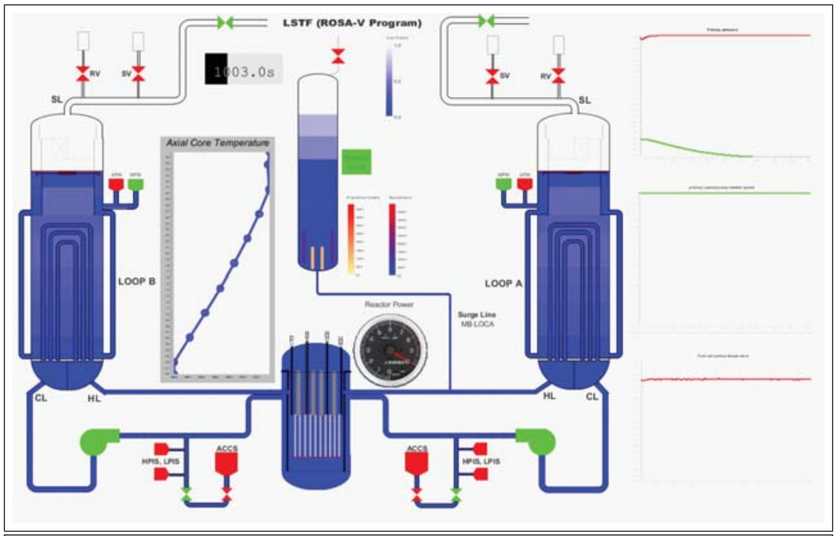
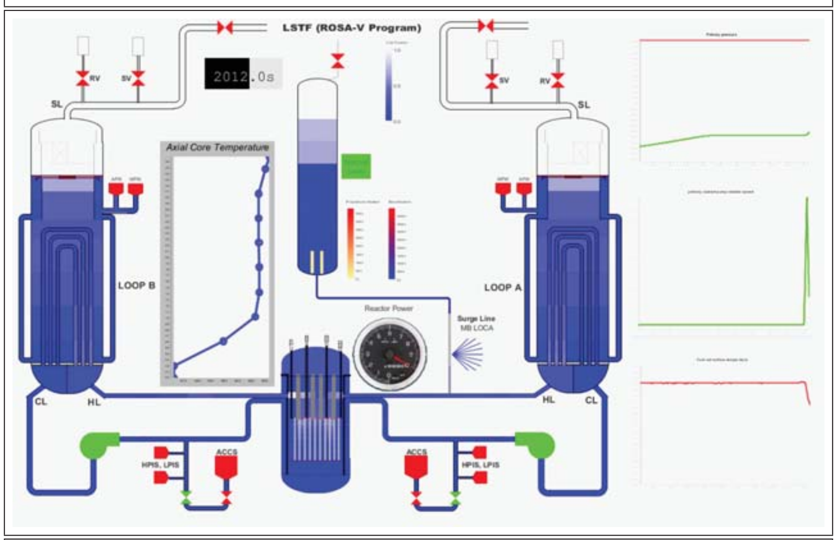


Figure 23: Normalized core level. Pre-test, post-test and test data

EVENT	**experimental time (s)**	**pre-test time (s)**	**post-test time (s)**
PZR isolation	1940	1940	1940
Break valve opening	2000	2000	2000
Scram signal	2004	2004	2004
RCPs coastdown beginning	2005	2005	2005
Termination of main feedwater	2005	2004	2004
Decay power initiation	2020	2022	2022
Beginning of ACCs discharge	2154	2110	2170
Beginning of ICC	2164	-	2165
Peak cladding temperature	2182	-	2175
Beginning of LPSI	2504	2300	2400

Table 4: Chronology of main events. Pre-test, post-test and experimental data





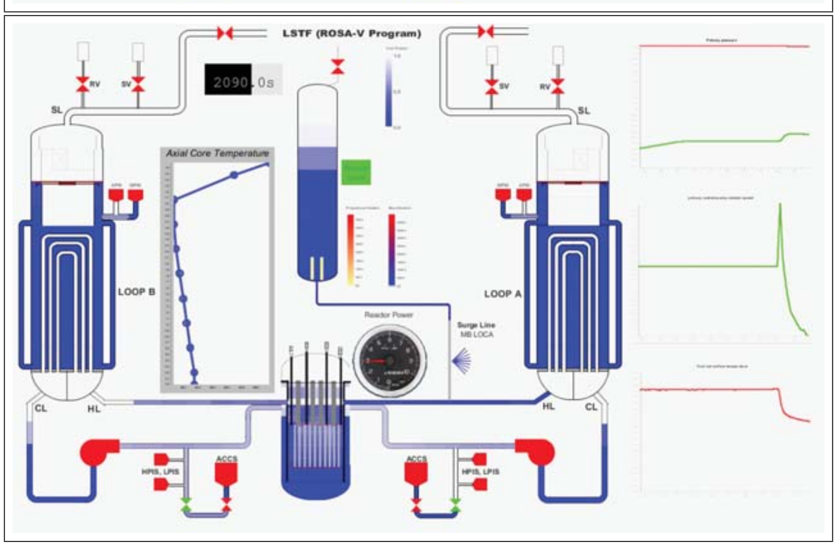


Figure 24: Void fractions. Snapshots of SNAP video simulation (1/2)

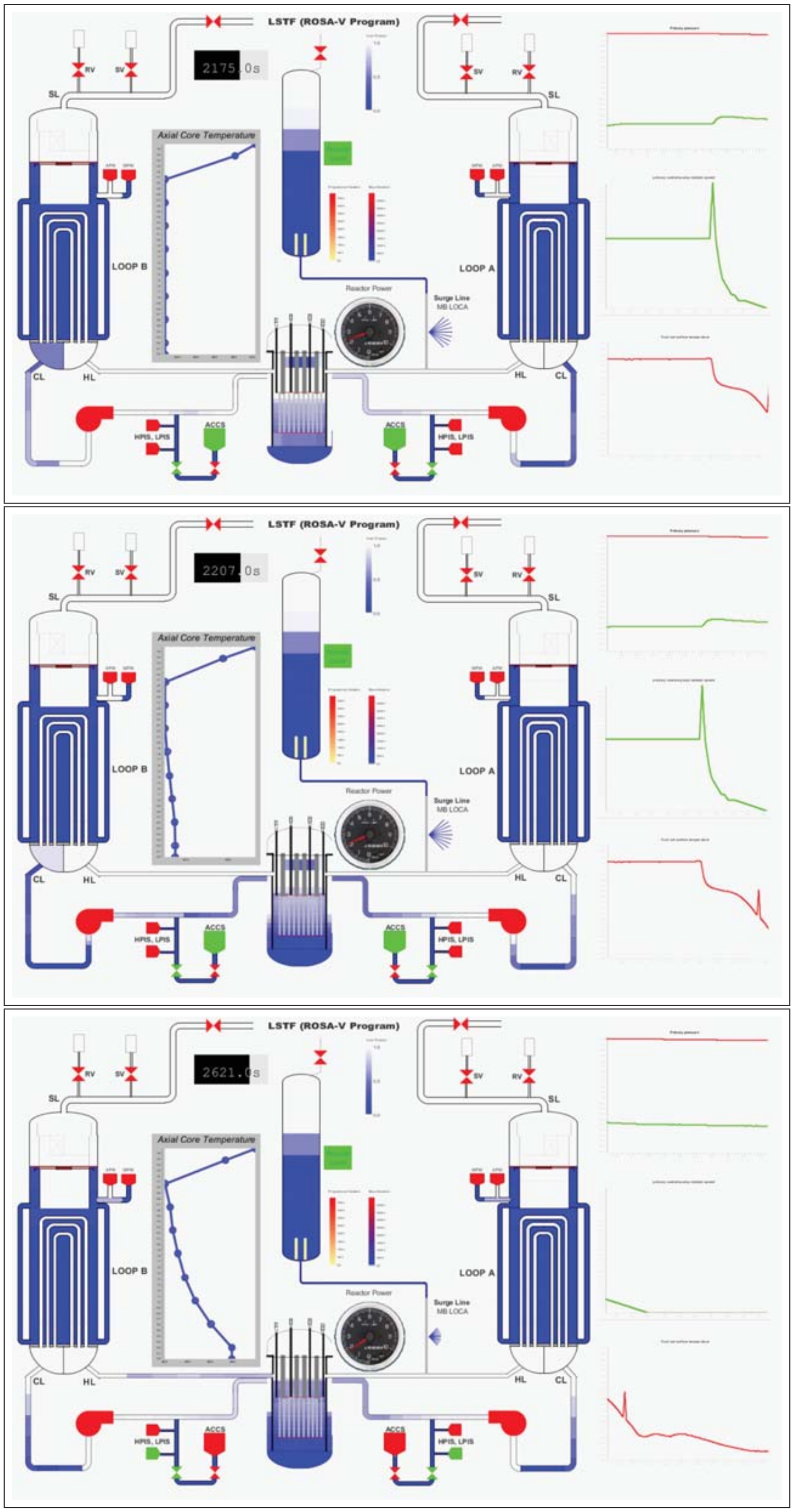


Figure 25: Void fractions. Snapshots of SNAP video simulation (2/2)

5 SIMULATION OF SURGE LINE MBLOCA IN ALMARAZ NPP

Almaraz NPP consists of two PWRs located in Caceres (Spain) and it is owned by a consortium of three Spanish utilities: Iberdrola (53%), Endesa (36%) and Union Fenosa (11%). Commercial operation started in April 1981 (Unit I) and September 1983 (Unit II). Each unit has a PWR Westinghouse with three loops, as shown in Figure 26, and two turbines (high and low pressure turbines) in tandem compound. The nominal power is 2947 MWt for unit 1 and 2729 MWt for unit 2. It is equipped with three steam generators Siemens KWU 61W/D3. Reactor coolant pumps are type single stage, centrifugal model W-11011-Al (93-D) designed by Westinghouse. The AFWS consists of one turbine driven pump and two motor driven pumps.

Relative positions and heights and diameters of RCS components are shown in Figures 26 and 27. Table 5 shows the main operating parameters for each unit.

This section presents main simulation results of similar transients to ROSA-2 test 1 with the TRACE model of Almaraz NPP, taking into account different accident management actions and conditions.

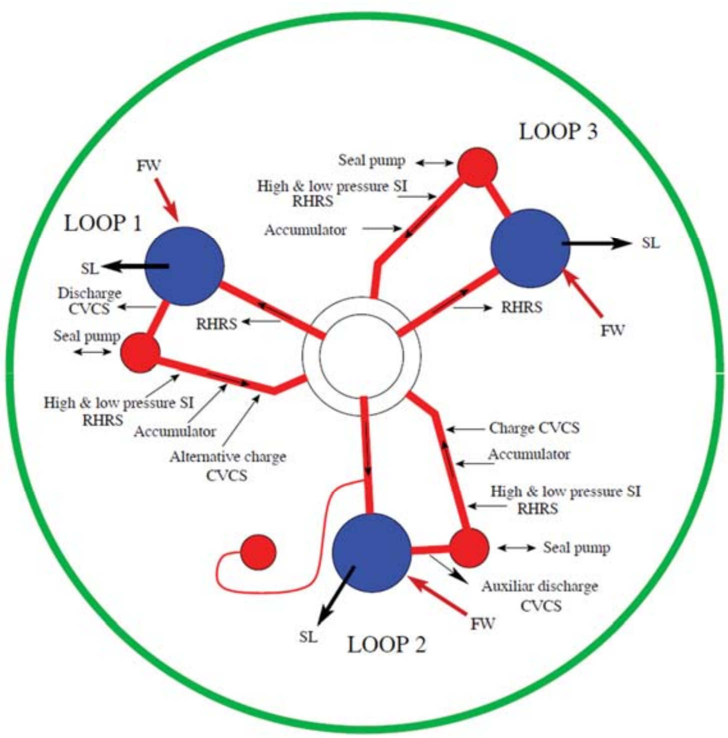


Figure 26: Scheme of Almaraz NPP (plan view)

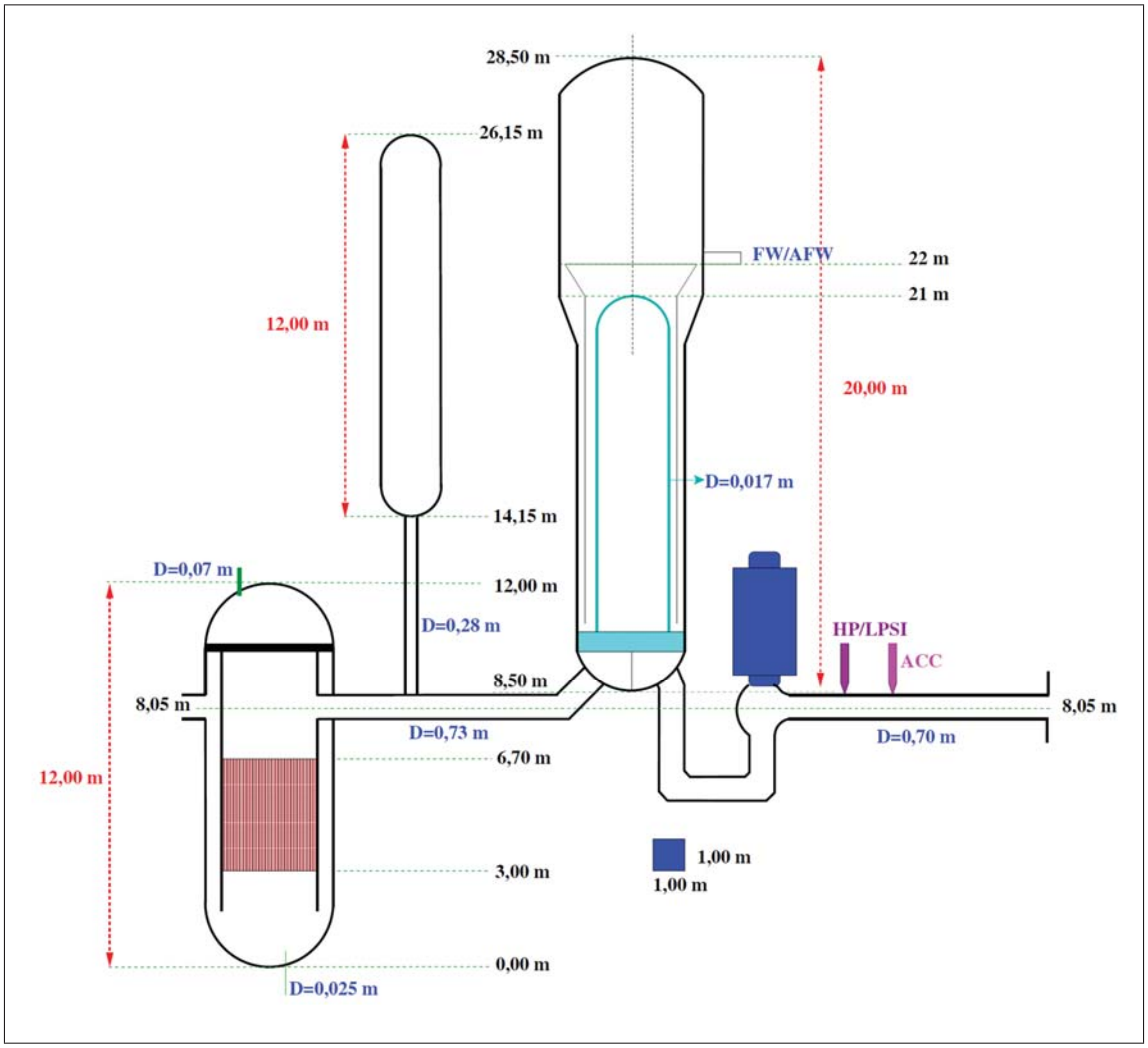


Figure 27: Scheme of Almaraz NPP (elevation view)

Table 5: Main operating parameters of the Almaraz I NPP

Description	**Value**
Thermal reactor power (Unit 1/Unit 2)	2947/2729 MWt
Fuel	UO_2 + GdO_2
Number of assemblies	157
Number of loops	3
Reactor operating pressure	155.017 bar
Coolant averaged temperature	
Zero load	564.9 K
100 %	580.8 K
Steam generators	Siemens KWU 61W/D3
Number of tubes (per SG)	5130
Total tube length (per SG)	108294.3 m
Tube inner diameter	17.96 mm
Tube material	INCOLOY 800
Pumps type	Centrifugal model W-11011-Al (93-D)
Pump discharge head	86.26 m
Design flow rate	6.27 m^3/s
Pump speed	155.509 rad/s
Primary volumes	
Vessel	100.81 m^3
Hot leg (x3)	3.18 m^3
Steam generator (x3)	32.28 m^3
Cross leg (x3)	3.6 m^3
Reactor coolant pump (x3)	4.02 m^3
Cold leg (x3)	3.23 m^3
Surge line	1.14 m^3
Pressurizer	39.64 m^3
Spray lines	0.45 m^3
TOTAL	280.97 m^3
Number of PZR relieve / safety valves	2 / 3
Number of PZR spray valves	2
Heaters capacity (proportional/backup)	(377 kW / 1023 kW)
Maximum spray flow	0.022 $m^3/s\cdot$valve
Steam mass flow rate at 100%	
SG1	489 kg/s
SG2	486 kg/s
SG3	500 kg/s

5.1 Description of Almaraz NPP model

Almaraz I NPP TRACE model has 255 thermal-hydraulic components (2 VESSEL, 73 PIPE, 43 TEE, 54 VALVE, 3 PUMP, 12 FILL, 33 BREAK, 32 HEAT STRUCTURE and 3 POWER component), 740 SIGNAL VARIABLES, 1671 CONTROL BLOCKS and 58 TRIPS.

Figure 28 shows a schematic diagram of the TRACE model of Almaraz NPP. Regarding the primary and secondary circuits, the following components have been modeled:

- Reactor vessel, modeled by a VESSEL component, Figure 29, which includes the core region, guide tubes, support columns, core bypass, and the bypass to the vessel head via downcomer and via guide tubes.
- Nuclear core power is modeled with axial and radial cosine power shape distributions. Core power is distributed into nine HEAT STRUCTURE components located each one in one core sector.
- Primary circuit, including steam generators and pressurizer in loop 2 (containing heaters, relief/safety valves and pressurizer spray system).
- Chemical and Volume Control System (CVCS).
- Emergency Core Cooling System (ECCS): safety injection system and accumulators (ACCs).
- The steam lines up to the turbine stop valves, with the relief, safety and isolating valves.
- The steam dump with the 8 valves.
- FW and AFW systems. Feed water pumps coastdown and auxiliary mass flows are included as boundary conditions.

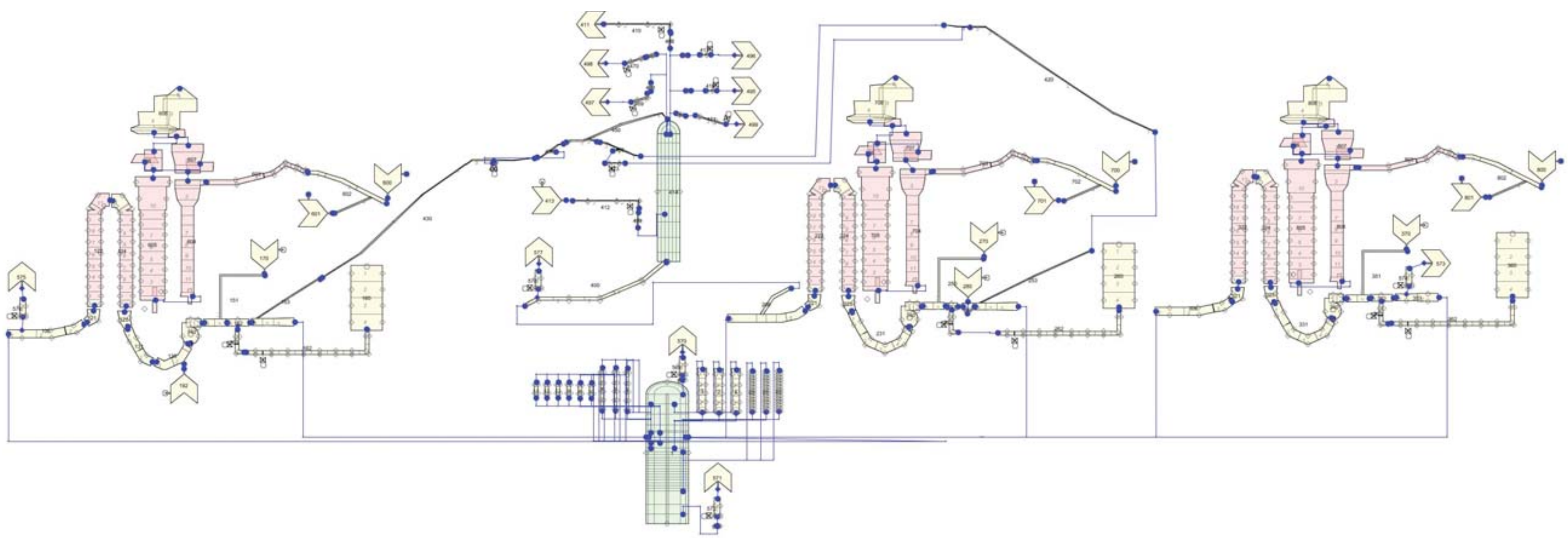


Figure 28: TRACE model of Almaraz NPP (SNAP nodalization with mean components)

The control, protection and engineering safeguard systems and signals modeled are:

- Pressurizer level control:
  - CVCS isolating discharge signal.
  - CVCS charge flow and heaters.
- Pressurizer pressure control:
  - Proportional and backup heaters.
  - Spray lines.
  - PORVs.
- Steam generators level control system.
- Steam dump control.
- Turbine control.
- Protection and engineering safeguard system-signals:
  - Emergency shutdown system (SCRAM).
  - Safety injection.
  - Pressurizer safety valve logic.
  - Auxiliary feedwater system activation and control
  - Relief, safety and isolating valve logic of steam lines.
  - Normal feedwater system isolation.
  - Turbine trip.
  - Pump trip.

This model has been validated with steady and transient conditions and verified with a large set of transients. See References [3], [4], [5], [8], [9], [10], [11], [12], [13], [14] and [15].

5.2 Comparison between Almaraz NPP and LSTF

Several considerations had to be made in order to undertake transposition to Almaraz NPP of ROSA-2 test 1 conditions due to differences between the NPP and the facility. Tables 6 and 7 show main differences between LSTF, Tsuruga NPP and Almaraz NPP.

As can be seen in Figure 29, RPV differences between Almaraz NPP and LSTF are cause of slightly differences in vessel models for both facilities. RPV models are quite similar, with analogous modelization: both cores are nodalized with four radii and three sectors, although Almaraz NPP core is nodalized with five more axial levels. The other components of the vessel are modeled roughly equal in both cases: downcomer, lower and upper heads and guide tubes show similar distribution and slight differences in nodalization.

	LSTF	**TSU**	**TSU/LSTF**	**ALM**	**ALM/LSTF**
Primary Volume (m^3)	8.14	347	42.6	281	34.52
RPV total volume (m^3)	2.754	131.7	47.8	100.81	34.65
Upper head volume (m^3)	0.4963	24.6	49.6	11.81	23.80
Upper plenum volume (m^3)	0.4950	28.4	57.4	28	56.56
Core volume (m^3)	0.4477	17.5	39.1	14.10	31.49
Lower plenum volume (m^3)	0.4644	29.62	63.8	20.20	43.47
Downcomer volume (m^3)	0.8504	31.58	37.1	20	23.52
Hot Leg area (m^2)	0.03365	0.4261	12.66	0.42616393	12.66
Cold Leg area (m^2)	0.03365	0.3831	11.39	0.38321039	11.39
Control rod drive mechanism area (m^2)	0.0006394	0.007280	11.38	0.003832	5.99
Instrumentation penetration area (m^2)	8.04E-6	5.07E-4	63	5.07E-4	63
Core Area (m^2)	0.113	4.75	41.89	3.87	34.27
Downcomer Area (m^2)	0.086	3.38	39.39	2.53	29.41

Table 6: Comparison among LSTF, Almaraz NPP and Tsuruga NPP (1/2)

	LSTF	**TSU**	**TSU/LSTF**	**ALM**	**ALM/LSTF**
Pressure (MPa)	16	16	1	16	1
Temperature (K)	598	598	1	598	1
Number of fuel rods	1008	50952	50.55	41448	41.12
Core height (m)	3.66	3.66	1	3.66	1
Power (MW)	10	3423	342	2686	268.6
Core inlet flow (m^3/s)	0.0488	16.7	342	18.7	383
Number of loops	2	4	2	3	1.5
Number of U-tubes for SG	141	3382	24.0	5130	36.38
Mean U-tube length (m)	20.2	20.2	1	21.11	1.04
Hot Leg diameter (m)	0.207	0.737	3.56	0.73	3.55
Cold Leg diameter (m)	0.207	0.6985	3.374	0.698	3.372
Control rod drive mechanism. Inner diameter (mm)	13.8	102	7.4	69.85	5.1
Instrumentation penetration. Outer diameter (mm)	3.2	25.4	7.94	25.4	7.94
Hot Leg length (m)	3.69	6.99	1.89	7.25	1.96

Table 7: Comparison among LSTF, Almaraz NPP and Tsuruga NPP (2/2)

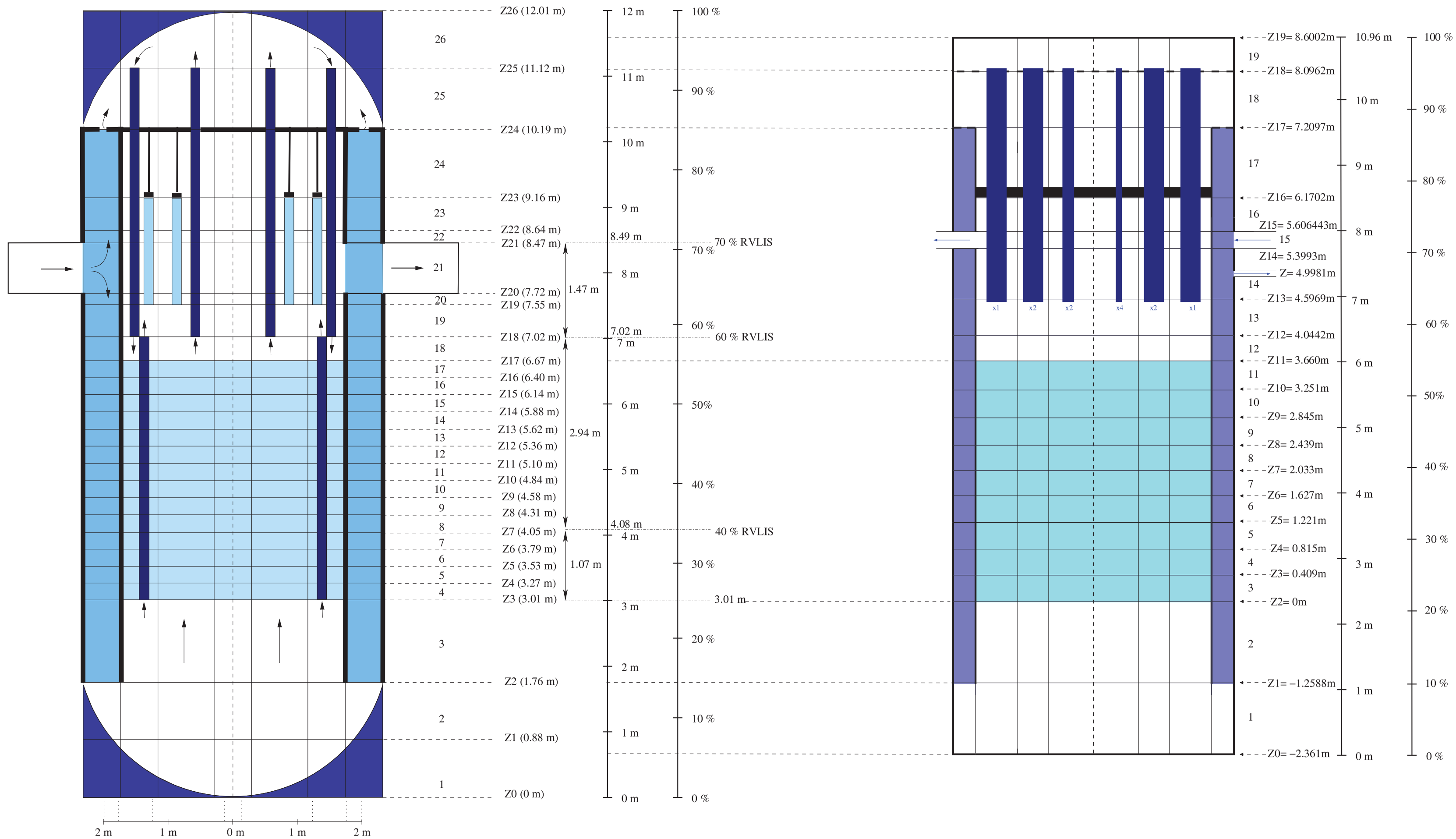


Figure 29: Comparison between LSTF and Almaraz NPP VESSEL components

5.3 Comparison between Almaraz NPP model and LSTF experimental results

First stage of the analysis of surge line MBLOCA without HPSI performed for a commercial PWR model consists of simulating sequence of ROSA-2 test 1 in Almaraz NPP TRACE model. The following conditions have been imposed to Almaraz NPP model:

- LOCA in surge line with an equivalent break size according to ROSA-2 Test 1.
- Unavailable HPSI (similar to ROSA-2 Test 1).
- RCP trip at the beginning of the transient (similar to ROSA-2 Test 1).
- No manual depressurization through SG is assumed (similar to ROSA-2 Test 1).
- 3 out of 3 ACCs are available.
- Mean Steam Isolation Valve (MSIV) closed due to high pressure conditions inside containment.
- 1 out of 2 trains of LPSI is available.
- Low and high discharge coefficients (for liquid and two phases flow) have been tested in order to obtain a bounding band (*Low:*0.75/1.0 and *High:*1.0/1.1).

At this point, three main issues related to simulations with Almaraz NPP model must be noticed:

1. ROSA-2 Test 1 consisted of a surge line break whose size was 17% of cold leg flow area (1.64 inch, 4.1 cm). Firstly, equivalent-break area was obtained by considering the relationship between volume of primary system in LSTF and Almaraz NPP, which nominally is 34.52 (see Table 6). The break area obtained by this method was then adjusted in order to fit the pressure evolution for both the ROSA-2-Test-1 transient and the Almaraz-NPP-equivalent transient. Adjusted equivalent-break diameter was 8 inches (20.3 cm).
2. RCP trip during a LOCA with unavailable HPSI is a well-known concern since the TMI accident, due to several experimental tests and simulations that were performed for Westinghouse reactors for SBLOCA sequences with and without availability of HPSI. Particularly, the EOPs of Westinghouse applied in LOCA sequences (EOP E-1 and ES-1.2, see Figure 30 and Ref. [18]) include two conditions in order to perform the RCP trip: loss of subcooling and availability of HPSI. However, in simulation coolant pumps have been tripped at the same time the break occurs, in order to maintain similarity to ROSA-2 Test 1.

3. Regarding manual depressurization through secondary system carried out by operator, Westinghouse EOPs establish in step 6 of ES-1.2 that manual depressurization must be performed by operator (see Figure 30). However, in this case secondary-side depressurization is not needed, due to the fact that a 8-inch-diameter break is large enough to depressurize RCS rapidly by means of loss of inventory.

Simulation results show a similar primary and secondary pressure transient with respect to ROSA-2 test 1, as shown in Figure 31, where primary and secondary pressures are normalized.

Although pressure evolution is similar in both LSTF and Almaraz NPP cases, it must be pointed out that there is no any cladding-temperature increase in Almaraz-NPP model, as can be observed in Figure 32. Therefore, it was decided to perform a sensitivity analysis in Almaraz NPP model in order to analyze the differences between LSTF and Almaraz NPP results. Such analysis is described in next section.

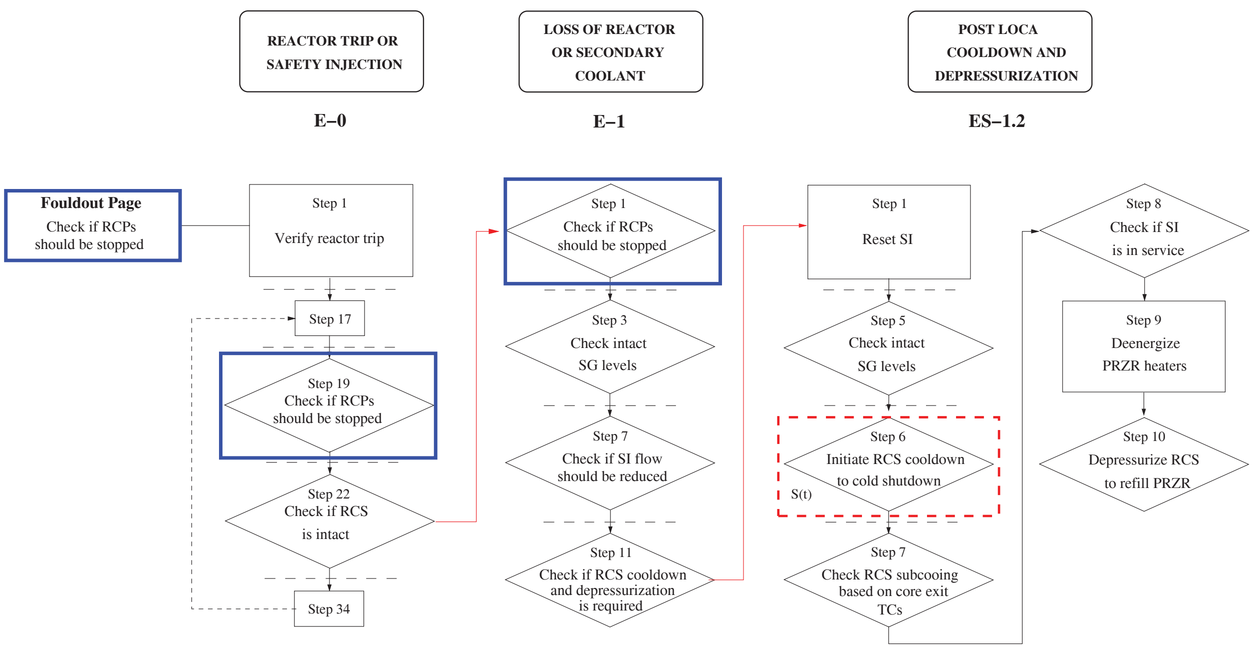


Figure 30: Main steps of the EOPs related with MBLOCA

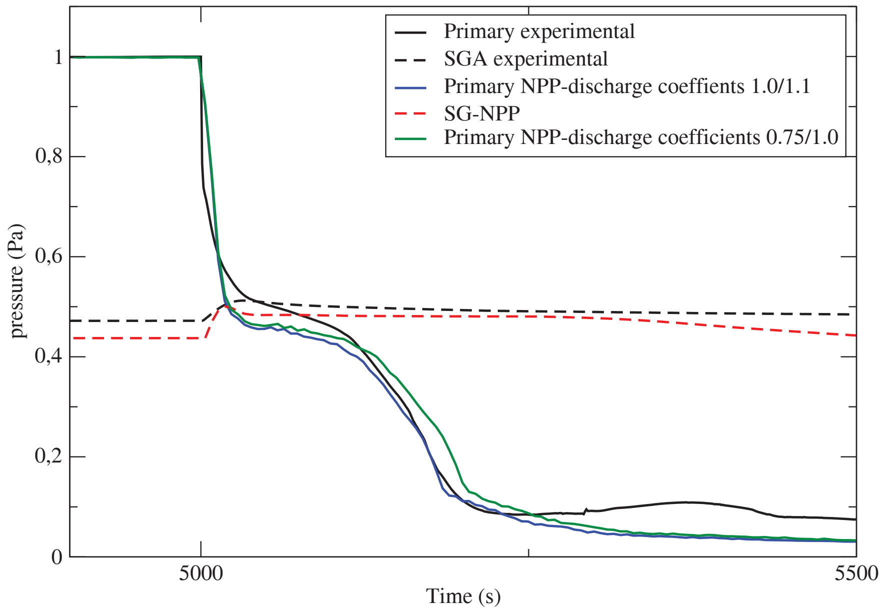


Figure 31: Primary pressure. Comparison between LSTF and Almaraz NPP

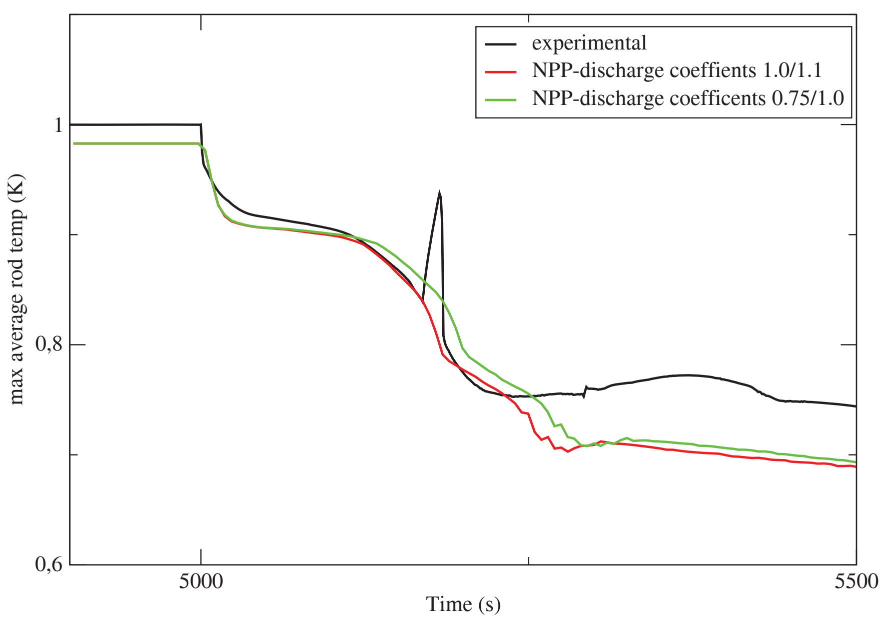


Figure 32: Cladding temperature. Comparison between LSTF and Almaraz NPP

5.4 Sensitivity analysis in Almaraz NPP model

As complementary task, it was decided to perform a broad sensitivity analysis with TRACE 5.0 Almaraz NPP model, in order to analyze the influence of two main parameters in this kind of sequences:

- **Break area:** A wide range of break areas has been analyzed, with break diameter from 0.0254 m to 0.2794 m (1 to 11 inches).
- **Beginning of secondary side cooling (55 K/h as RCS-cooling rate):** From minimum time (600 seconds) on.

The main objective of this analysis is to check the impact of the break size and the effectiveness of secondary side cooling during a surge line SBLOCA/MBLOCA with unavailable HPSI. This analysis has been performed by means of ISA methodology.

The ISA methodology aims at providing with an adequate method to perform a general uncertainty analysis, with emphasis in those sequences where some events occur at uncertain times. For a given safety limit or damage limit (PCT within this work), the numerical result of this methodology consists of the exceedance frequency of that limit (not obtained in this work), generically referred to as DEF, for the sequences originated from a relevant set of initiating events. This is done by means of the identification of the DD of the sequences that contribute to the total DEF. The DD of a sequence is defined as the region of the space of uncertain parameters of interest (break size and secondary-side-depressurization beginning time for the case analyzed in this work) where the limit is exceeded. ISA methodology has been successfully applied in other analysis (Loss of Component Cooling System, Upper and Lower Head SBLOCA, Hot Leg LOCA, Total Loss of Feedwater System, Station blackout, Steam generator tube rupture and hydrogen concentration inside containment), see for example [4], [5], [6], [10] and [16] for more details.

The calculation process is described below (30 cases were simulated with TRACE code):

1. Firstly, failure of S header (no manual depressurization in secondary side) is assumed. A transient (path) is simulated for each considered break diameter (1", 1.25", 1.5", 2", 2.5", 2.75", 3", 4", 5", 6" and 11"), see Figures 33 and 34. As shown in Figure 33, where the horizontal line labeled as 1477 K (2200 F) represents the acceptance criterion for PCT, several of the cases reach the *damage condition*. In those sequences, damage will be reached at a certain time (t_{dam}), which sets the minimum time for the beginning of manual depressurization. As evident, starting depressurization later than t_{dam} is not useful to avoid damage, because damage was reached at t_{dam}. These time points form the line of *Previous Damage* (PD) shown in Figure 37.

2. Furthermore, analyzing the evolution of the average temperature in the reactor coolant for the transients corresponding to PD line, it is possible to check from which time the manual depressurization is inefficient, i.e. the cooling rate due to depressurization is higher than 55 Kelvin per hour without human actions. This is due to the fact that depressurization caused by loss of coolant through the break is, at a certain point, large enough so that the operator can not perform manual depressurization without violating the limit of 55 Kelvin per hour. Therefore the results of PD could be extended until the inefficiency depressurization line, see Figure 37.

3. As shown in Figure 37, a set of transients were simulated with different delay times for the beginning of depressurization (being t=0 the time when break occurs), always below the efficiency line. Some of the transients exceed the damage condition (red diamond) while other paths do not reach it (green circle). Clearly, all possible transients which lay above the inefficiency line reach damage condition (if there is PD for its break size) or are successful transients (if there is not). So, as shown in Figure 37, all possible transients which are set above the inefficiency line undergo damage for break diameter less or equal than 10.8 cm (4.25 inches); being successful for larger sizes. It should be noticed that the provided minimum delay time for manual depressurization is 600 seconds, as the minimum time the operator of a Westinghouse PWR needs to reach the step 6 of the EOP ES-1.2, which leads him to depressurization actions (see Figure 30 and Ref. [18]). As an example, Figures 35 and 36 show the cladding temperature and the primary pressure for three different transients, all of them corresponding to a 2-inch-break diameter, with the only difference of beginning of depressurization. It is clear that beginning of depressurization affects significantly success of transient, as shown in Figure 35, where must be noticed that larger delay in beginning of depressurization leads in general to earlier damage and greater PCT.

4. With these results, plotted in Figure 37, it is possible to obtain the DD for this sequence, as shown in Figure 38, by connecting the first depressurization time that leads to damage for each break size. In this DD it can be noted that there are damage transients with accumulator demanding and others without it. This difference must be taken into account to calculate probabilities or frequencies.

It must be noticed that DD depicted in Figures 37 and 38 corresponds to that one obtained from a 55 Kelvin per hour cooling, as established in EOPs. However, there is also another depressurization option in Westinghouse procedures: **full opening of relief valves** when one of the following logical options is satisfied:

1. Measured core exit thermocouples temperature (T_{CET}) is greater than 921 K, **OR**

2. T_{CET} is greater than 649.13 K, **AND** measured vessel level is lower than 40%.

This second depressurization mode, which has not been considered in this work but amends the cooldown limitation of 55 Kelvin per hour in the case of ICC conditions, would minimize the DD, as analyzed in Ref. [16] for a vessel upper head SBLOCA in sequences with unavailable HPSI.

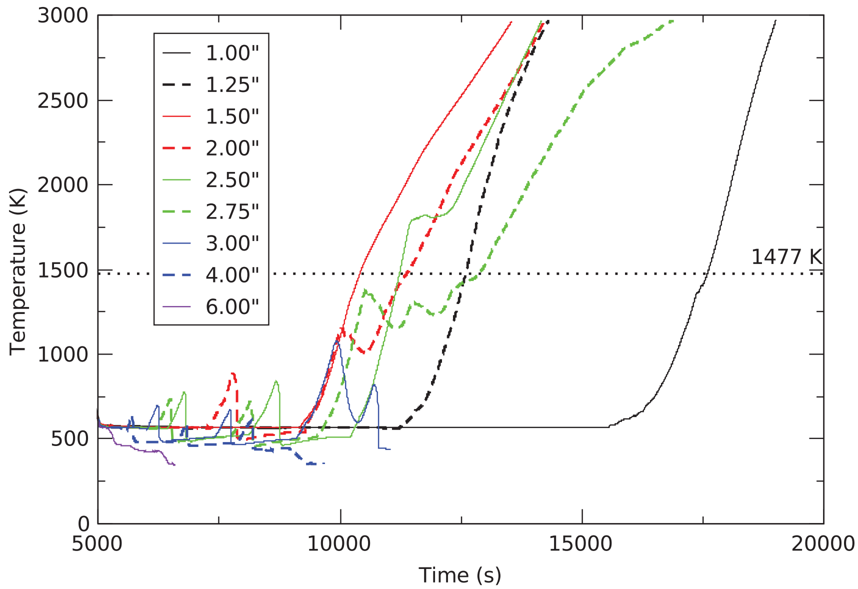


Figure 33: PCT. Sensitivity to break size. No SG depressurization.

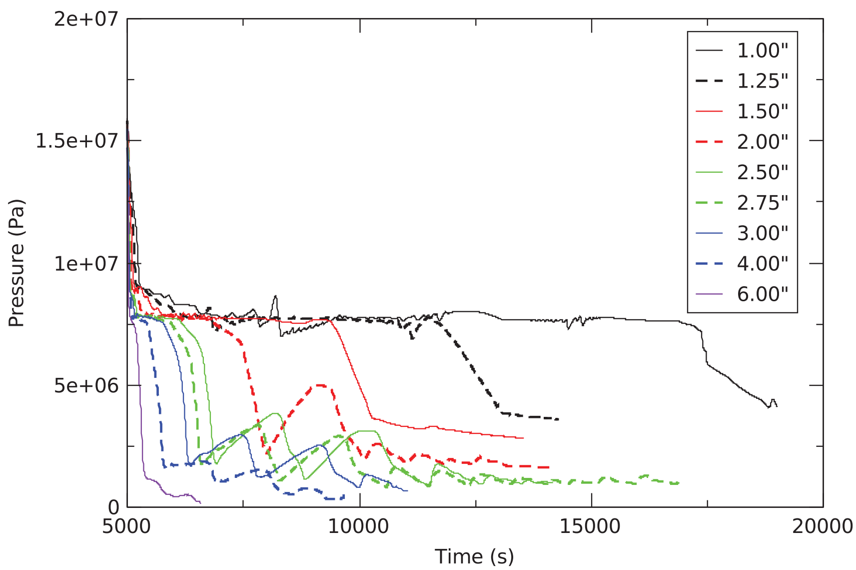


Figure 34: Primary pressure. Sensitivity to break size. No SG depressurization.

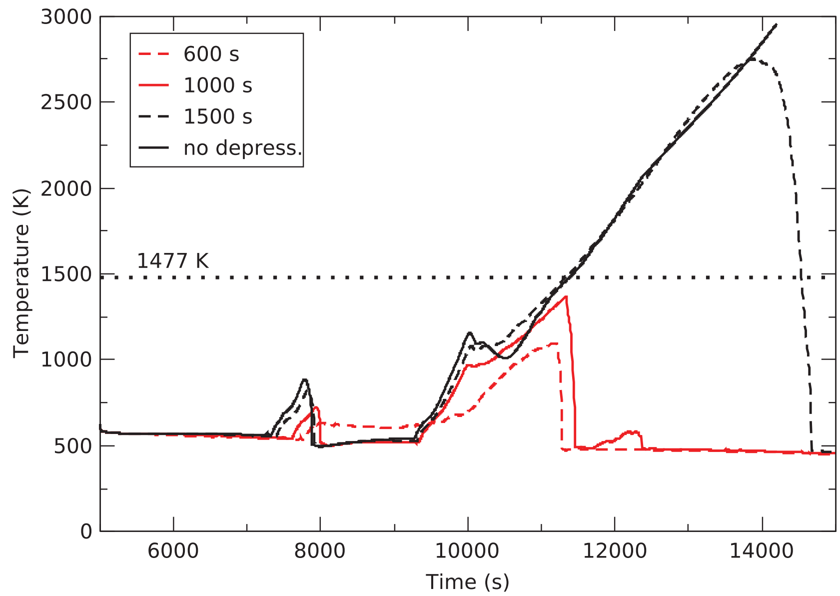


Figure 35: PCT. Sensitivity to SG depressurization. Break diameter: 2”.

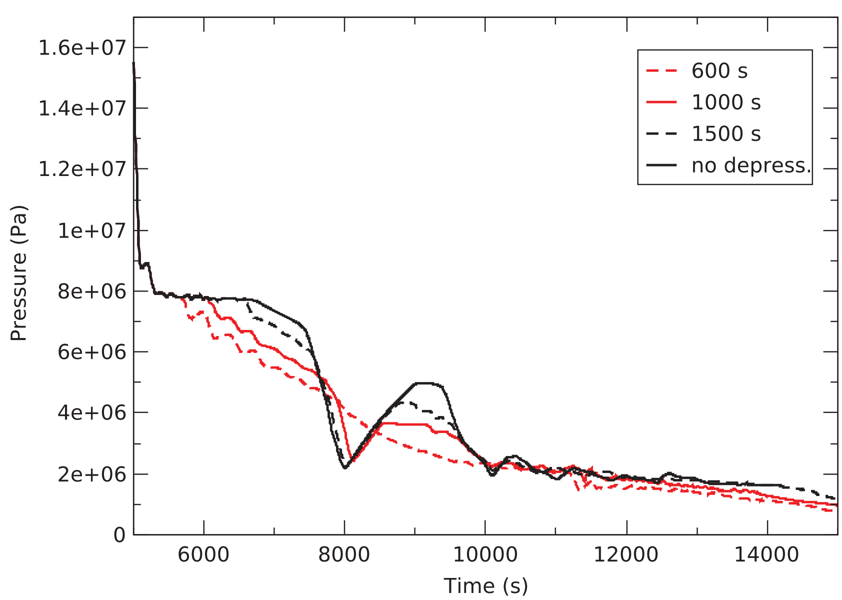


Figure 36: Primary pressure. Sensitivity to SG depressurization. Break diameter: 2”.

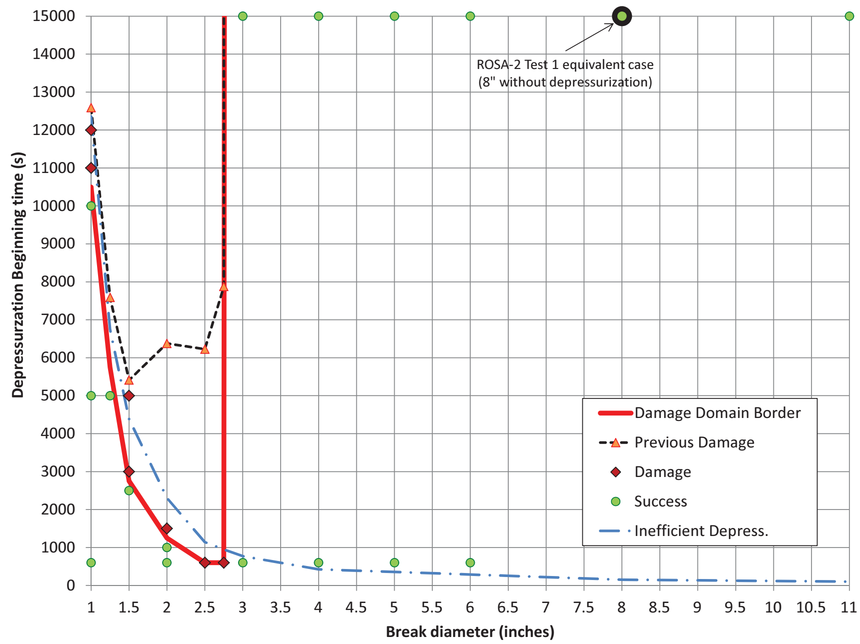


Figure 37: Success and damage transients. Surge line MBLOCA with HPSI failed.

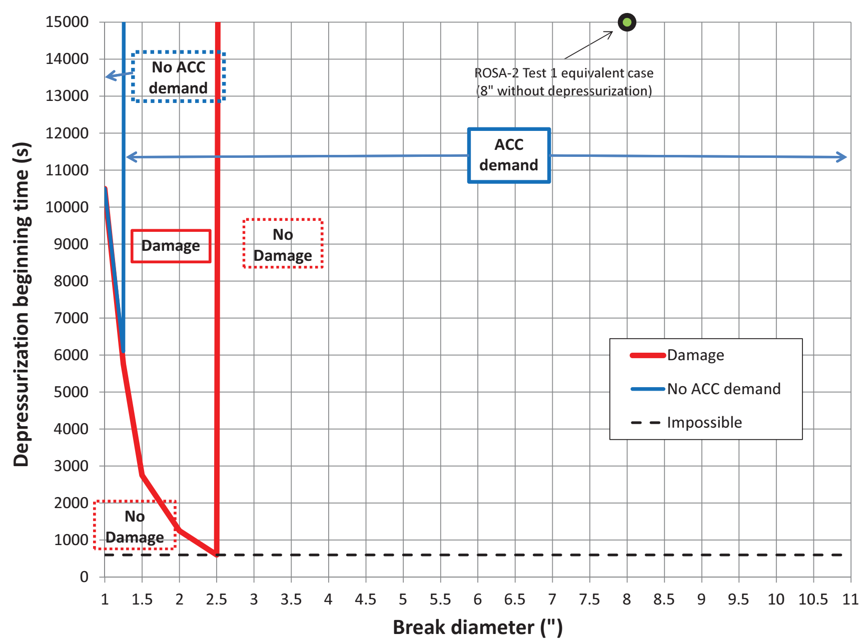


Figure 38: Damage domain of surge line MBLOCA with HPSI failed.

6 EXECUTION STATISTICS

The simulations have been run in AMD Opteron Dual Core Processors 180 & 1222 under Debian, both with 32 and 64 bits precompiled executables provided by NRC. No significant differences where found between runs executed in Windows and Debian systems, and between 32 and 64 bits code versions.

As shown in Figure 39, CPU effort for simulation of ROSA-2 Test 1 with LSTF model can be divided into three main stages (summarized in Table 8):

1. High performance calculation: 0-2000 seconds of simulation (steady-state until break).

2. Medium performance calculation: From ∼2000 to ∼2950 seconds. Corresponding to main stage of transient.

3. Lower performance calculation: Corresponding to low pressure calculations and maximum LPSI inflow.

Nevertheless, simulation of equivalent sequence with Almaraz NPP model shows different CPU performance (see Figure 40), mainly due to differences between complexity and completeness of both models (nodalization, system modeling, etc.). Performance is summarized in Table 9.

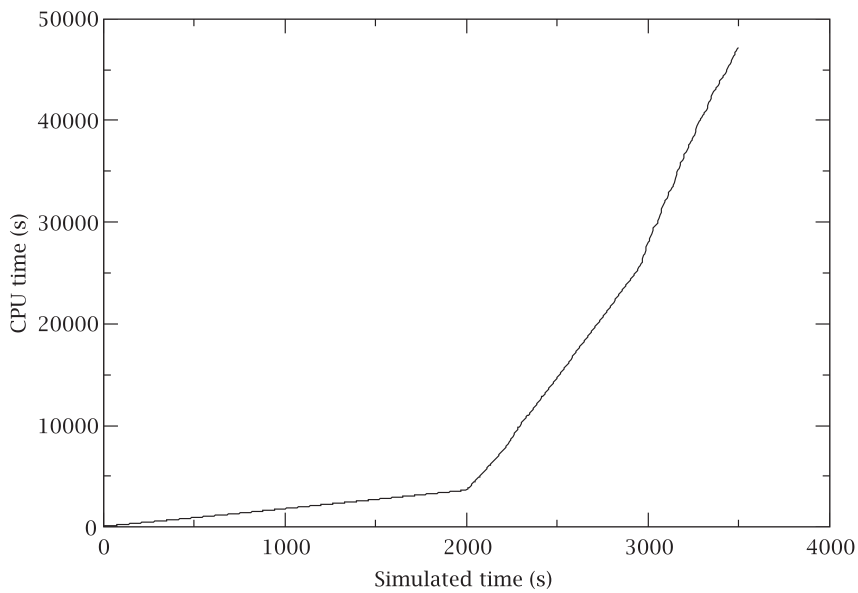


Figure 39: CPU performance for ROSA-2 test 1 simulation with LSTF model

Computing Stage	LSTF simulation performance (%)
Stage 1	54.9% (0-2000 s)
Stage 2	4.3% (2000-2950 s)
Stage 3	2.5% (2950-3500 s)

Table 8: CPU performance for LSTF simulations of ROSA-2 Test 1

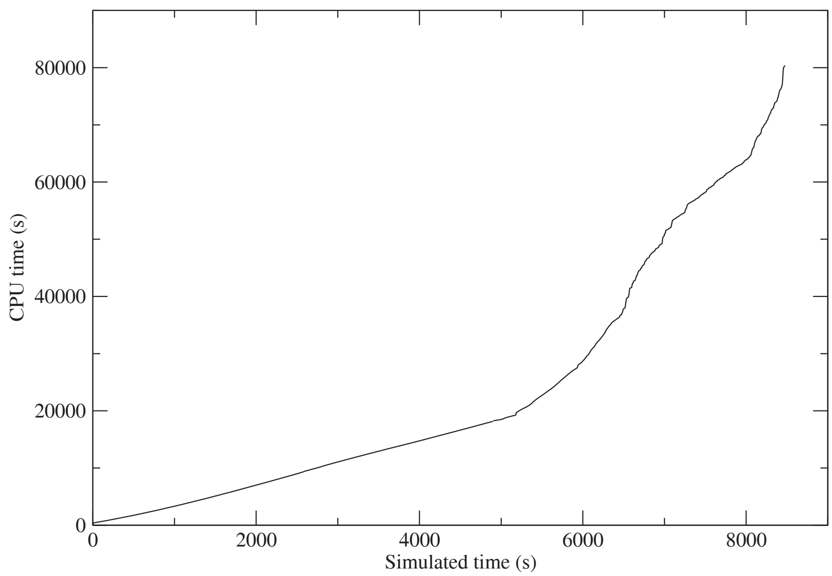


Figure 40: CPU performance for ROSA-2 test 1 simulation with Almaraz NPP model

Computing Stage	Almaraz NPP simulation performance (%)
Stage 1	46.7% (0-5000 s)
Stage 2	5.8% (5000-8500 s)

Table 9: CPU performance for Almaraz NPP simulation of ROSA-2 Test 1

7 CONCLUSIONS

The Large Scale Test Facility replicates the primary and secondary coolant systems of a Westinghouse 4-loop reactor of 3423 MWt, including ECCS and control systems for accident management actions. In OECD/NEA ROSA-2 Test 1, phenomenology of cold leg MBLOCA with HPSI failed was investigated, in order to obtain experimental thermal-hydraulic data for the assessment of thermal-hydraulic computer codes like TRACE, and models for plant integral analysis.

In this report, a post-test analysis of OECD/NEA ROSA-2 Test 1 using TRACE 5.0 has been presented, as well as the simulation of such a test in the TRACE model of a commercial nuclear power plant (Almaraz NPP). Main findings were:

- The comparison between ROSA-2 Test 1 and the simulations performed in TRACE code show that the primary and secondary pressures match fairly well with the experimental results until accumulators discharge. The results also give a good prediction of the core uncovering, as well as the PCT, being this one quite sensitive to the actuation pressure of accumulators. It can be concluded that, although ROSA-2 Test 1 is roughly well reproduced, LSTF model still needs improvements.
- On the other hand, the results obtained from the simulations of Almaraz NPP show a similar primary and secondary pressure transient with respect to ROSA-2 Test 1, but it must be noticed that there is no core uncovering in the simulation for Almaraz NPP. This can be due to the scale and design differences between LSTF and Almaraz NPP, since LSTF is based in a 4-loop Westinghouse. Nevertheless, TRACE code has shown a good capability for simulating similar sequences to ROSA-2 test 1.
- In reference to the complementary task (sensitivity analysis with respect to the beginning of secondary side depressurization and break area) the results show that the sequences with intermediate or large break diameters, from 0.0762 meters on (>3") manual depressurization has not any effect on sequence result, due to the fact that break is large enough to depressurize RCS before reaching damage (without HPSI and with RCP trip). In SBLOCA sequences, from 0.0254 m to 0.0635 m (1" to 2.5"), there is an available time in order to avoid core damage. However, for break diameters between 0.0635 m and 0.07 m (2.5" and 2.75"), manual depressurization is not effective, because DD border lays at a depressurization beginning time of 600 seconds, which is considered the minimum time to depressurize.
- Damage domain is an useful tool to analyze accidental sequences in nuclear power plants. This kind of analysis is part of a more wide methodology called Integrated Safety Analysis Methodology (ISA) which has been developed by the Modeling and Simulation area (MOSI) of Spanish Nuclear Safety Council (Consejo de Seguridad Nuclear, CSN).

As a final remark, it should be emphasized that the simulations performed for ROSA-2 test 1 and its application to Almaraz NPP with TRACE code meet the objectives of the work. As it has been shown throughout this document, TRACE is an adequate code to simulate this kind of experiments, and allows a deeper comprehension of thermal-hydraulics for complex transients in complex models.

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10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

An analysis of LSTF Surge Line Break experiment (OECD/NEA ROSA-2 test 1) has been performed with TRACE code. This test, included within the OECD/NEA ROSA-2 project and performed in Large Scale Test Facility (LSTF), it is attempted to analyze the phenomenology and different accident management actions after the occurrence of a surge line (SL) break with failure of High Pressure Safety Injection (HPSI). The comparison between experimental data and the results obtained with TRACE code shows that, in general, the main phenomena are well reproduced. Additionally, a broad analysis of Surge Line Medium Break Loss of Coolant Accident with HPSI failed in a Westinghouse PWR has been performed taking into account different accident management actions and conditions in order to check their suitability. These works has been performed in the framework of OECD/NEA ROSA-2 and CAMP projects.

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