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International Agreement Report

Simulation of LSTF Upper Head Break (OECD/NEA ROSA Test 6.1) with TRACE Code. Application to a PWR NPP Model

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ABSTRACT

An analysis of LSTF Upper Head Break experiment (OECD/NEA ROSA test 6.1) has been performed with TRACE code. This test, included within the OECD/NEA ROSA project, attempts to analyze the phenomenology and different accident management actions after the occurrence of a Upper Head break with failure of High Pressure Safety Injection (HPSI). The comparison between the experimental data and the results obtained with TRACE code shows that, in general, the main phenomena are well reproduced.

Additionally, a broad analysis of Upper Head Small Break Loss of Coolant Accident (SBLOCA) 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 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.

OECD/NEA ROSA test 6.1 (SB-PV-09) was carried out on November 17, 2005 in the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA). This test consisted of a vessel upper-head break equivalent in size to a 1.9% cold leg break, and without availability of High Pressure Safety Injection.

In this report, an extended analysis of Upper Head SBLOCA with HPSI unavailability in a Westinghouse PWR is presented. The analysis has been performed through 2 stages:

1. In a first stage, the *post-test* simulation with TRACE code was performed, and then extensively evaluated through comparision with the experimental results.
2. In a second stage, similar transients to test 6.1 have been simulated with the TRACE model of Almaraz NPP (Westinghouse 3 loop design). This extended analysis takes into account different accident management actions and conditions in order to check their suitability.

The purpose of this analysis is to contribute in the validation of TRACE code and its ability to properly simulate transient conditions. The main findings of the comparison of TRACE results with the OECD/NEA ROSA test 6.1 experiment are:

- Results of OECD/NEA ROSA Test 6.1 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 RFRIC factors and the activation of reflood model, as well as a renodalization of the vessel, cold and hot legs and steam generators.
- Not all improvements added to the LSTF model have lead to good results, as expected. For example, addition of 9 U-tubes in the SG of LSTF model yields worse results than the simplest model, with only 1 U-tube.
- One of the pre-established experimental conditions of OECD/NEA ROSA Test 6.1, was the trip of reactor coolant trip at the same time that break occurs. Sensitivity analysis to RCP trip delay shows that maximum cladding temperature increases as RCP trip delay increases, but only until approximately 1000 seconds after the break.

Beyond that point, PCT decreases slightly, remaining almost constant independently of RCP trip delay.

- Transposition of Test 6.1 into Almaraz NPP requires taking into account respective scaling factors. Once considered this scale factor, both transients look very similar, except some deviations in pressure and PCT. Depressurization is quite similar in both models, until ACCs demand, because of an earlier discharge in Almaraz due to its higher pressure set-point. Cladding temperature begins to rise up earlier in experimental test PCT is significantly lower in Almaraz simulations, due to its earlier ACCs discharge.

- Simulation of Test 6.1 in Almaraz NPP model shows diverse grades of sensitivity to several parameter modifications. Break location, steady state upper head mass flow and friction factors at ACCs exit lead to little change in results. Model exhibits medium sensitivity to discharge coefficients and upper downcomer area, and shows high sensitivity to break area size, RCP trip delay and quantity of ACCs available.

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 Almaraz-Trillo AIE and the Spanish Nuclear Regulatory Body (CSN) for the technical and financial support under the agreement STN/1388/05/748.

ABBREVIATIONS

ACC	Accumulator
AFWS	Auxiliary Feedwater System
AM	Accident Management
CAMP	Code Applications and Maintenance Program
CET	Core Exit Thermocouples
CRDM	Control Rod Drive Mechanism
CSN	Consejo de Seguridad Nuclear (Spanish Nuclear Council)
CSNI	Committee on the Safety of Nuclear Installations
CVCS	Chemical and Volume Control System
DC	Discharge Coefficient
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
FWS	Feedwater System
HPSI	High Pressure Safety Injection
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss Of Coolant Accident
LOFT	Loss of Fluid Tests
LPSI	Low Pressure Safety Injection
LSTF	Large Scale Test Facility
NRC	Nuclear Regulatory Commission
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
OECD	Organisation for Economic Co-operation and Development
PCT	Peak Cladding Temperature
PKL	Primär-Kreis-Lauf
PORV	Pilot Operated Relief Valve
PWR	Pressurized Water Reactor
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RPV	Reactor Pressure Vessel
ROSA	Rig Of Safety Assessment
RV	relief Valve
SBLOCA	Small Break LOCA
SESAR	Senior Group of Experts on Nuclear Safety Research
SG	Steam Generator
SI	Safety Injection
SNAP	Symbolic Nuclear Analisys Package

TH	Thermal-Hydraulics
TMI-2	Three Mile Island NPP - Unit 2
UH	Upper Head
UPM	Universidad Politecnica de Madrid
USNRC	United States Nuclear Regulatory Commission
W-4L	Generic Westinghouse 4-loop PWR

1 INTRODUCTION

In the beginning of PWR transient analysis history, the most limiting accident considered was the surge line break (10 inch), because credit was not given to an RCS large pipe break. In 1966, the Atomic Energy Commission required the break analysis of the larger pipe in the NSSS: the RCS piping double-end break. For many years, it was thought to be the most conservative accident for PWRs.

In 1979, the Small Break Loss of Coolant Accident (SBLOCA) at Three Mile Island 2 NPP, shown that SBLOCA had to be taking into account as the plant behavior is quite different from LBLOCA and it is not fully covered by the classical LBLOCA analysis.

The most limiting SBLOCA since that time is the break at the cold leg, as it drives to the complete loss of one coolant injection path. Later on, the finding of a vessel head wall thinning at the Davis Besse reactor in 2002 showed the possibility of a SBLOCA in the upper head of the reactor vessel due to the circumferential cracking of a CRDM penetration nozzle.

With the aim of simulating PWR behavior during such type of scenarios, several experimental test has been performed in LSTF facility. 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 Project, which started in 2005 by the agreement between JAEA, OECD/NEA and thirteen member countries, determined to conduct a SBLOCA test (Test 6.1, SB-PV-09 in JAEA). This test 6.1 simulates a PWR vessel upper-head break equivalent in size to a 1.9% cold leg break. The objective of the test was to study the effect of accident management (AM) actions and to provide integral test data for assessment and development of advanced analytical codes.

In this report, a post-test analysis of OECD/NEA ROSA test 6.1 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 upper head small break LOCA transients.

2 DESCRIPTION OF LSTF AND OECD/NEA ROSA TEST 6.1

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 NPP Unit 2), 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 actions allow the detailed analysis of several accident scenarios like small break LOCA, steam generator U-tube rupture,main steam line break, etc.

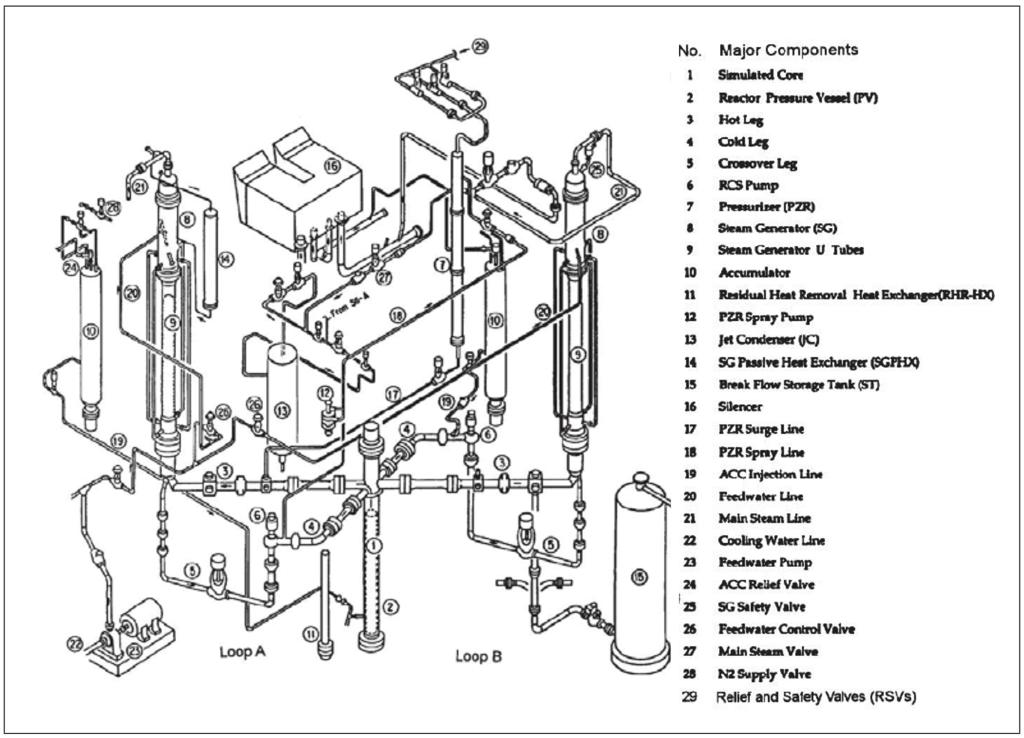


Figure 1: Large Scale Test Facility (LSTF)

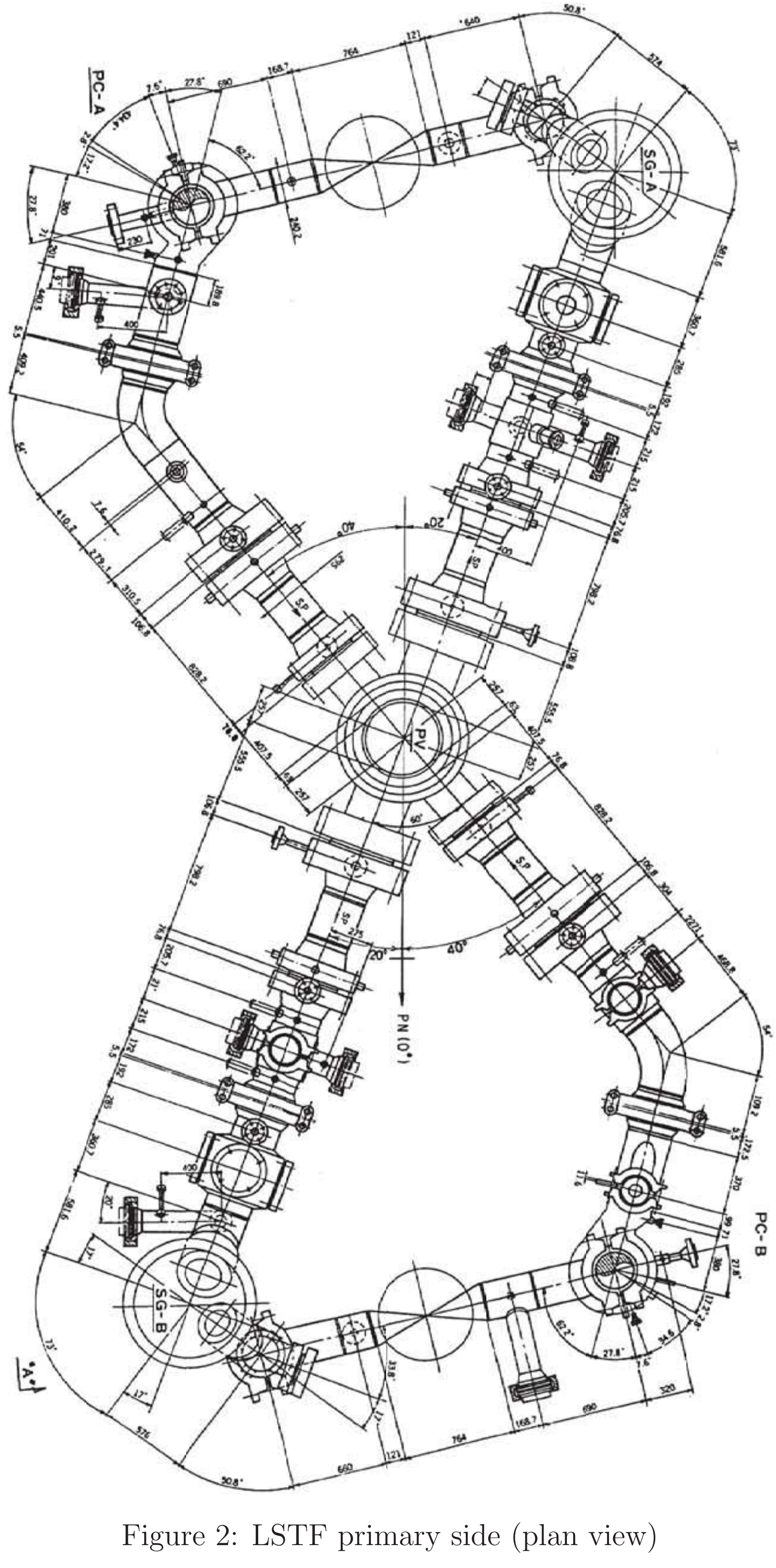


Figure 2: LSTF primary side (plan view)

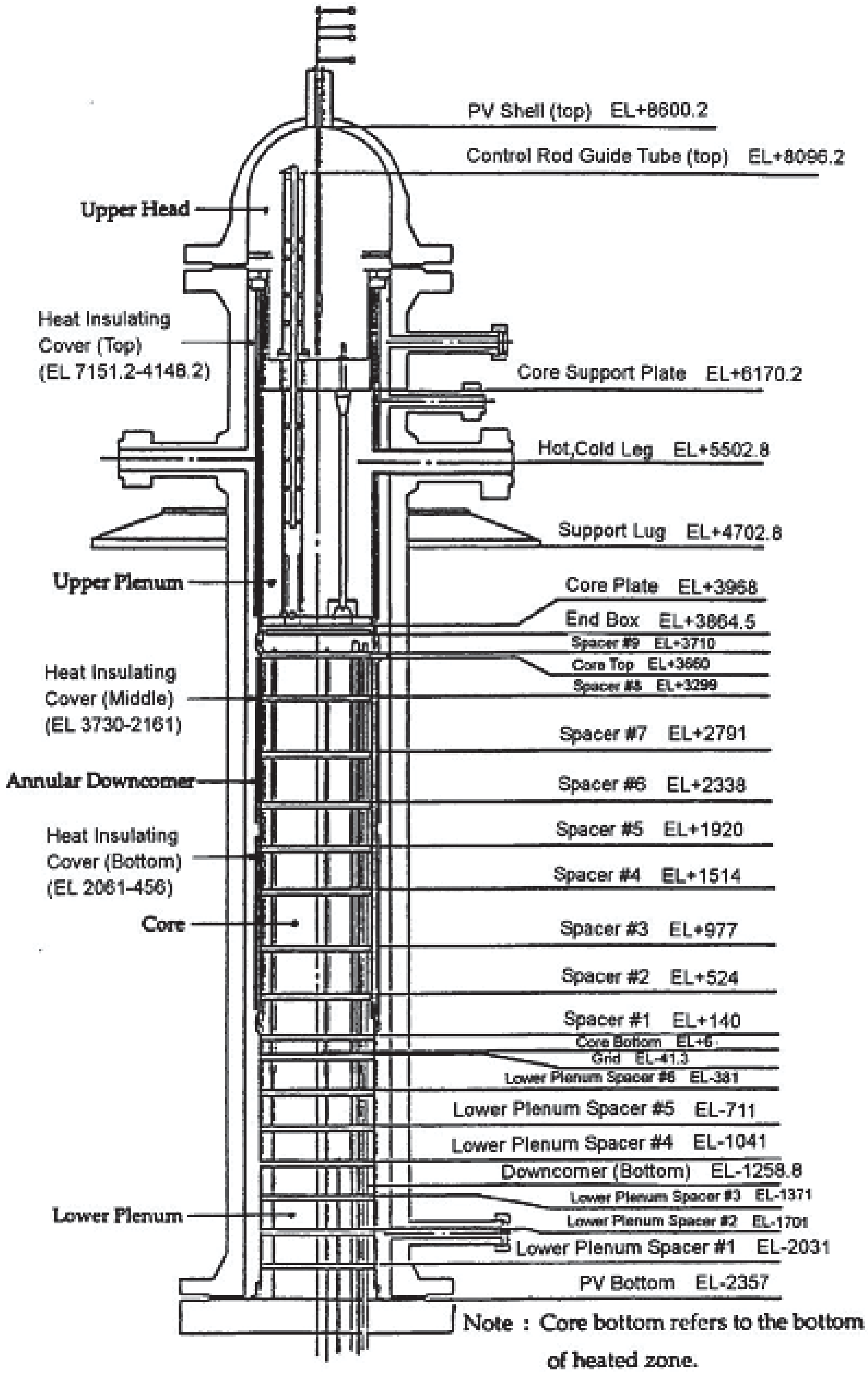


Figure 3: LSTF pressure vessel (elevation view)

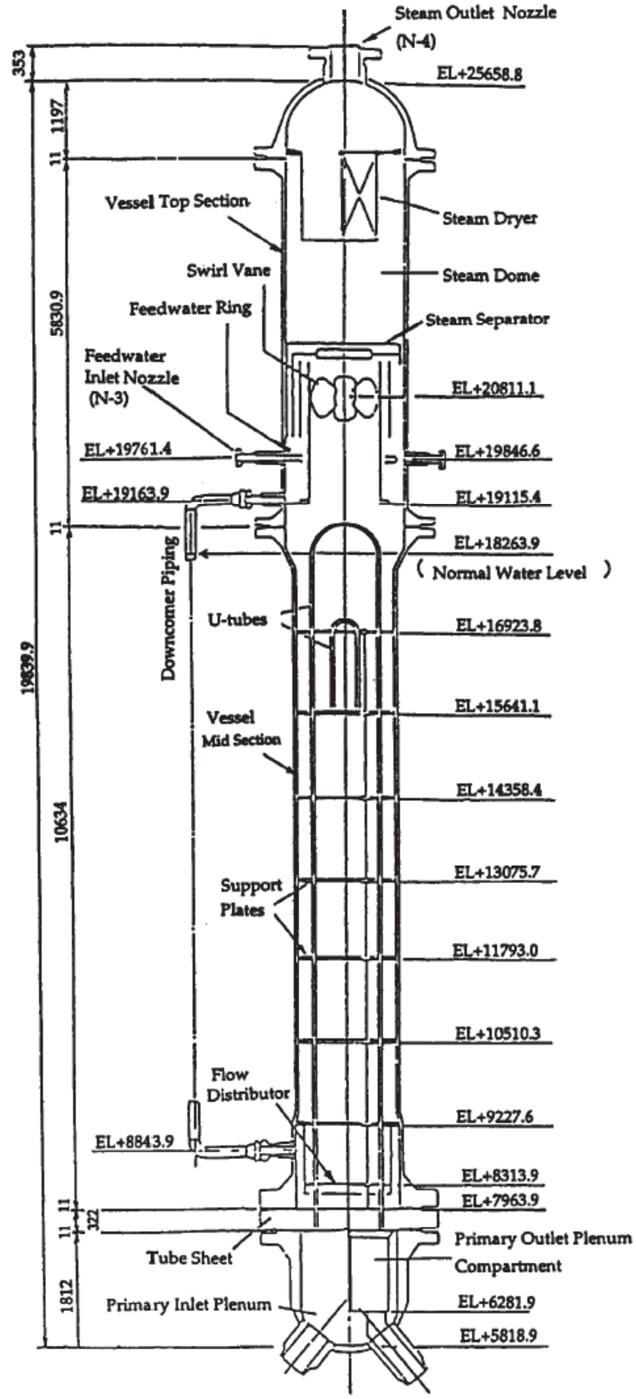


Figure 4: LSTF steam generator (elevation view)

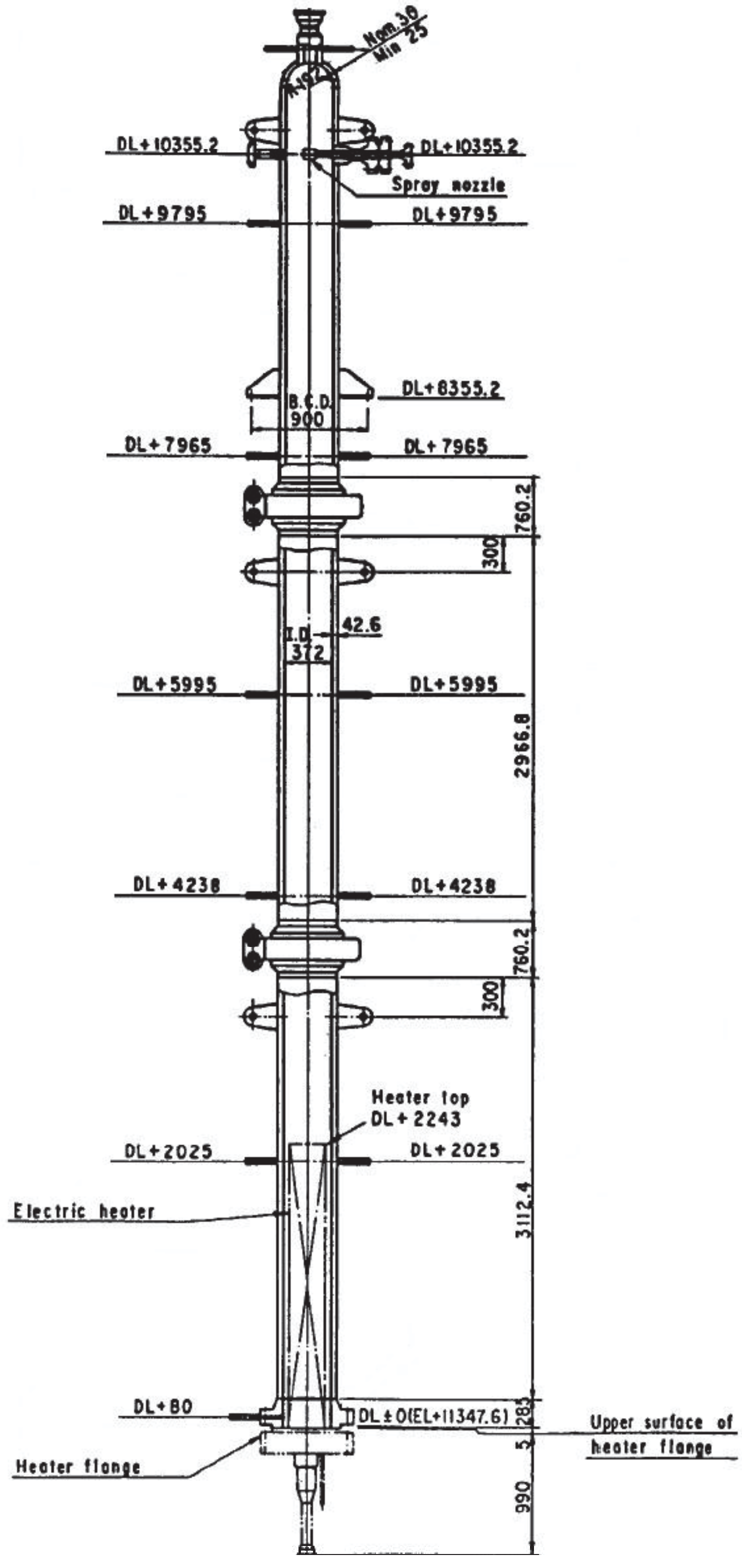


Figure 5: LSTF pressurizer (elevation view)

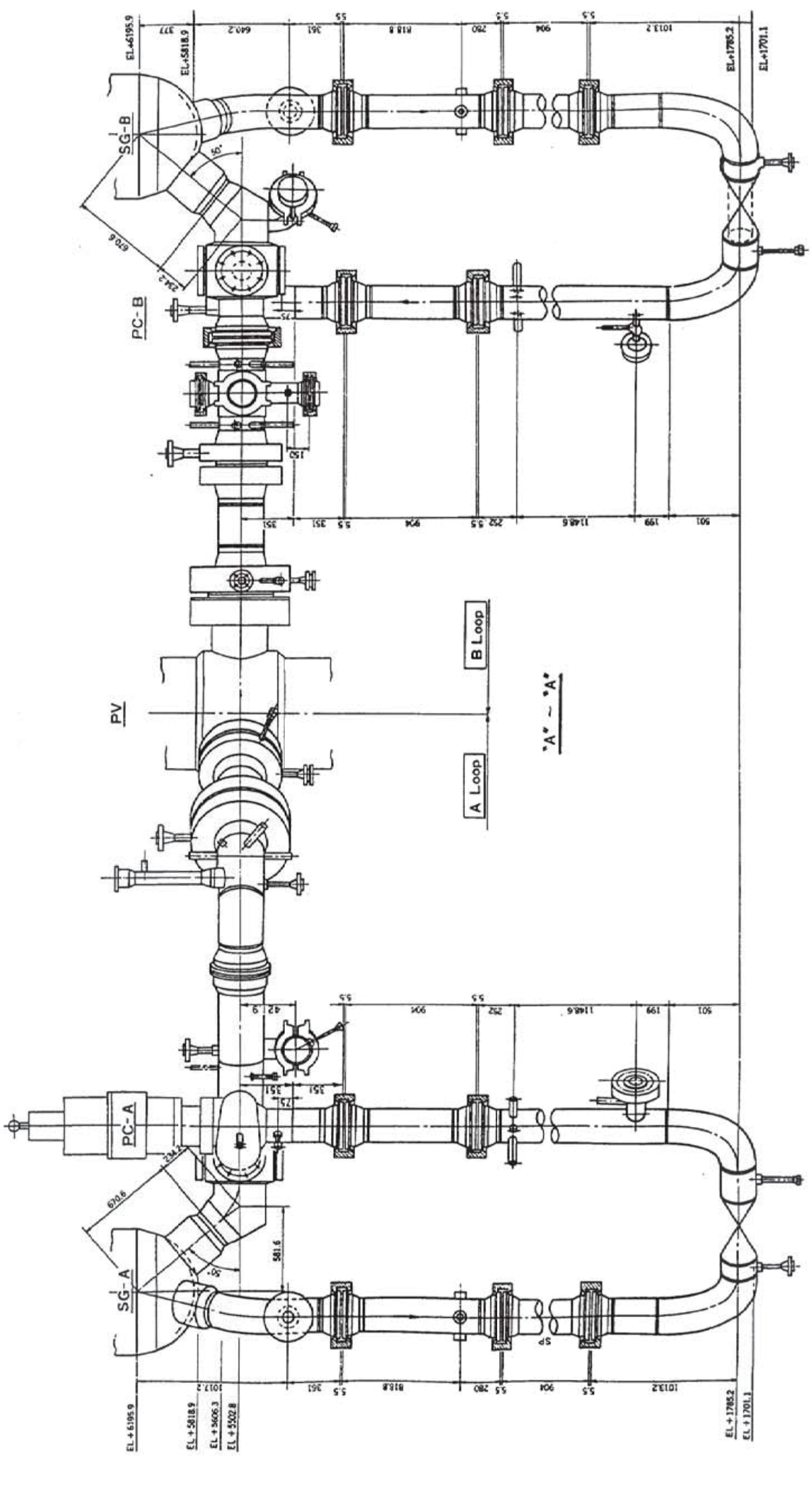


Figure 6: LSTF hot and cold legs (elevation view)

2.2 Description of OECD/NEA ROSA Test 6.1

The main objective of OECD/NEA ROSA Test 6.1 is to analyze the phenomenology of upper head SBLOCA with HPSI failed in a PWR reactor. This test is also useful to obtain experimental thermal-hydraulic data for the assessment of thermal-hydraulic computer codes like TRACE.

Other conditions and specifications for the test are the following:

1. Total failure of HPSI system when demanded.
2. Loss of off-site power coinciding with scram.
3. AFW system initiated immediately after SI signal, at primary pressure of 12.27 MPa (when required AM action to maintain SG secondary-side water level about U-tubes).
4. Initiation of steam generator (SG) secondary-side depressurization by fully opening relief valves, as accident management (AM) action when core exit temperature reaches 623 K.
5. Break size is 1.9% cold leg equivalent (diameter of 13.8 mm).
6. The core power is automatically decreased by the core protection system when the maximum fuel rod surface temperature exceeds a certain maximum. Threshold temperature for the LSTF core protection system set as follows: 958 K = 75%, 968 K = 50%, 969 K = 25%, 970 K = 10% and 973 K = 0% of pre-determined value.

Steady-state conditions of the test can be seen in Table 1. Chronology of major events in Test 6.1 is shown in Table 2 and Figure 7.

The relatively big size of the break results at the beginning of the transient, in a fast depressurization and loss of inventory in the RCS, leading to an early core uncovery. Subsequent core overheating triggers the AM action of depressurizing the secondary side when the Core Exit Thermocouples (CETs) detect a high temperature (T>623K). However, the fact that primary pressure was much lower than the SG secondary-side pressure (see Figure 7) indicates that detection of high CET was late in time, and so subsquent AM action of opening SG reielf valves.

This AM action was ineffective in the early stage,as the LSTF core protection system automatically had to decrease the core power down to 10% of the decay power level as the maximum fuel rod surface temperature exceeded the core protection limit (T>958 K).

The test had to be terminated prematurely to avoid excessive overheating of the core. Results showed that the core uncovery had started significantly early before the CET thermocouples indicated superheating and that the temperature increase rate was higher in the core than in the CET. The results suggested that the response of the CET thermocouples could be inadequate to initiate the relevant AM actions.

PARAMETER	**VALUE**
Power (MW)	10.0 ± 0.07
Hot Leg temperature (K)	598.1 ± 2.75
Cold Leg temperature (K)	562.4 ± 2.75
PZR pressure (MPa)	15.5 ± 0.108
PZR level (m)	7.2 ± 0.25
Mass flow per loop (kg/s)	24.3 ± 1.25
Secondary pressure (MPa)	7.3 ± 0.054
Steam mass flow (kg/s)	2.74 ± 0.10
Feed water mass flow (kg/s)	2.74 ± 0.05
ACCs pressure (MPa)	4.51 ± 0.054
LPSI pressure (MPa)	1.24 ± 0.108

Table 1: Initial conditions in OECD/NEA ROSA Test 6.1

TIME (s)	**EVENT**
2000	Break valve open
2022	SCRAM signal (Primary Pressure = 12.97 MPa)
2028	SI signal (Primary pressure = 12.27 MPa)
2050	Break flow from single-phase liquid to two-phase flow
2276	Primary coolant pumps stop
2700	Break flow to single-phase vapor
$\sim$ 2800	Primary pressure lower than SG secondary-side pressure, Core uncovery
3090	Initiation of SG secondary-side depressurization (full opening of relief valves, core exit temperature = 623 K)
$\sim$ 3200	Core power decrease by LSTF core protection system (Max. fuel rod surface temperature = 970 K)
$\sim$ 3300	Initiation of accumulator system (Primary pressure = 4.51 MPa)
$\sim$ 4300	Inflow of nitrogen gas from accumulator tank into primary loop
$\sim$ 4900	Initiation of LPSI system (RPV lower plenum pressure = 1.23 MPa)
5265	Break valve closure

Table 2: Chronology of major events in OECD/NEA ROSA Test 6.1

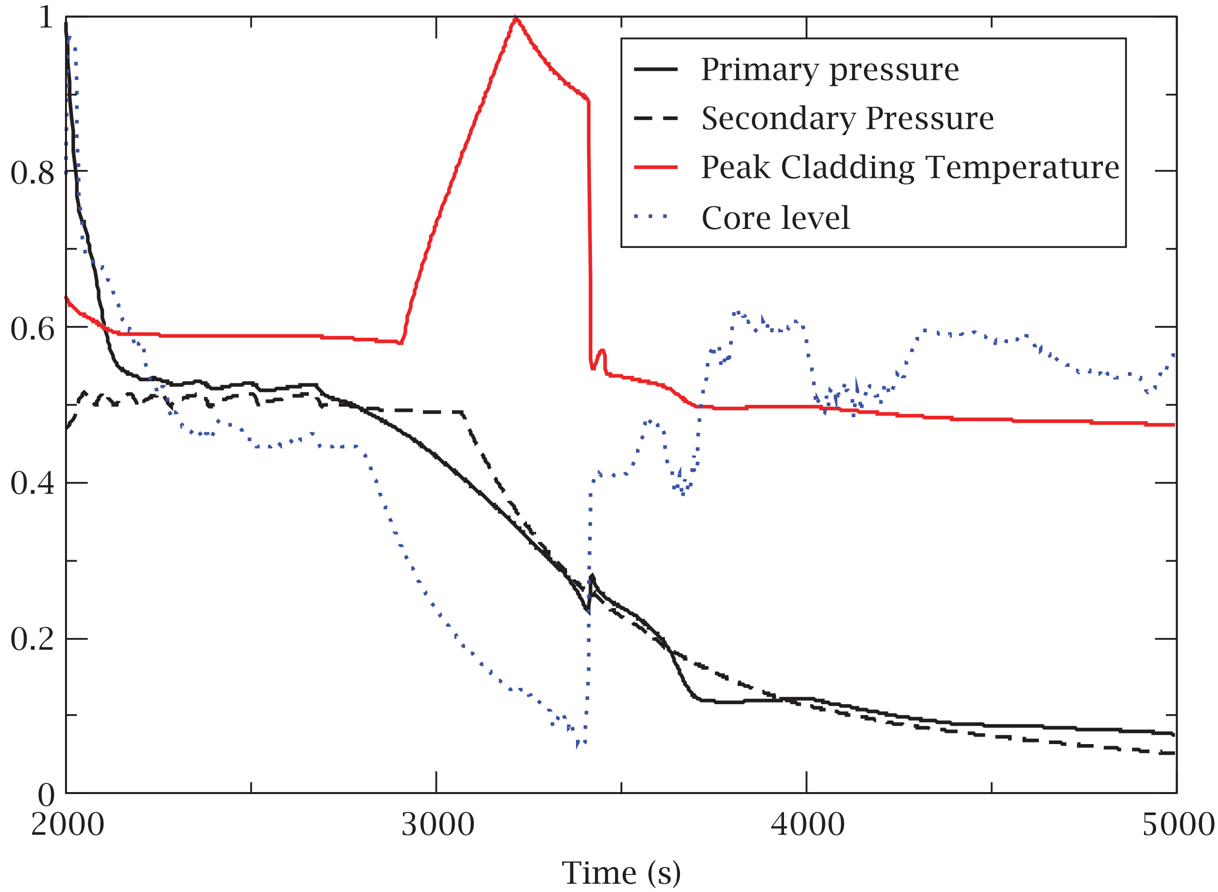


Figure 7: Evolution of main variables in OECD/NEA ROSA Test 6.1

3 DESCRIPTION OF TRACE MODEL OF LSTF

The group of Universidad Politecnica de 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.

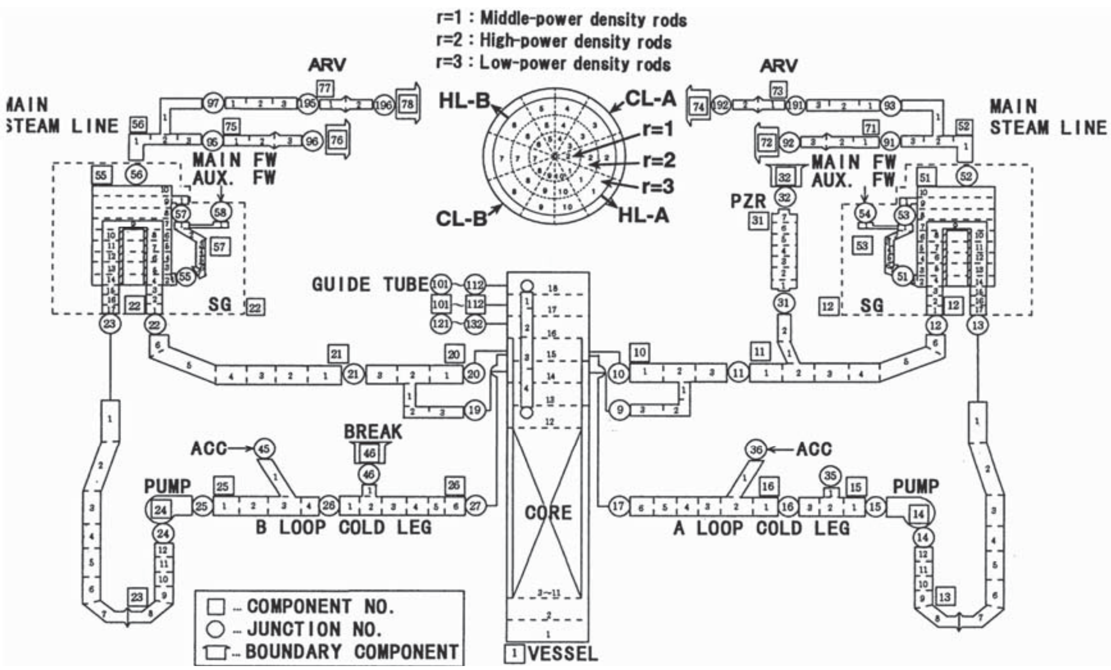


Figure 8: TRAC-PF1/MOD1 model of LSTF (JAERI)

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 had to be adapted to TRACE code. Later on, in a second stage, since 2006 many improvements have been included. This section contains a briefly description of both phases.

3.1 First Stage: LSTF model translation from TRAC-PF1 to TRACE

LSTF model for TRAC-PF1 code (Figure 8) was migrated to TRACE code.Main tasks related with the translation of the model were:

1. Old VESSEL component was translated to TRACE: VESSEL component from TRAC-PF1 model was divided into hydraulic and thermal components, creating new HTSTR to substitute the core heaters included in component VESSEL (see Figure 9).
2. Old STGEN component was translated to TRACE model as a set of components (TEEs and PIPEs), preserving volumes and lengths (see Figure 10).
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.

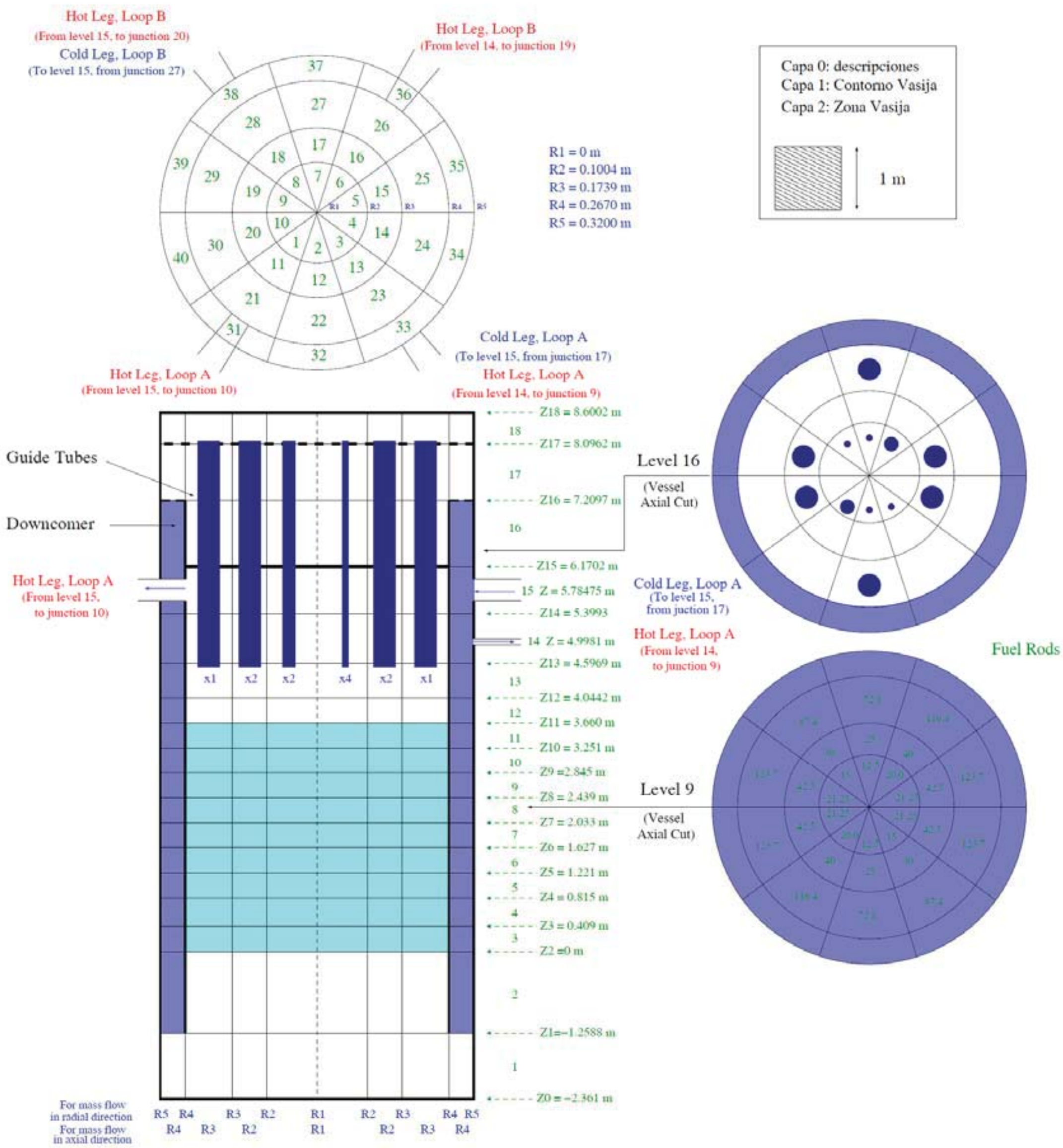


Figure 9: TRACE model of LSTF Vessel

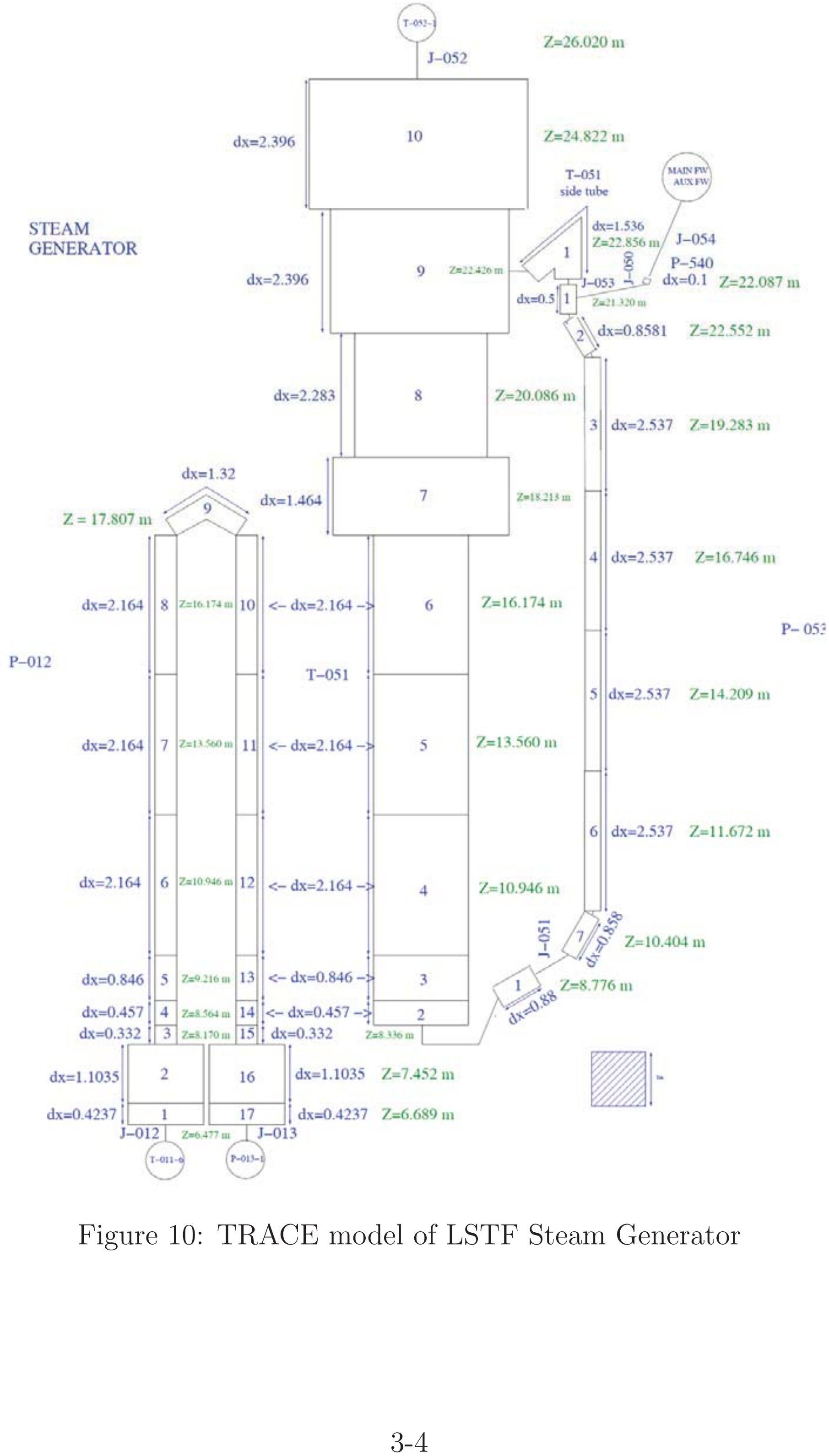


Figure 10: TRACE model of LSTF Steam Generator

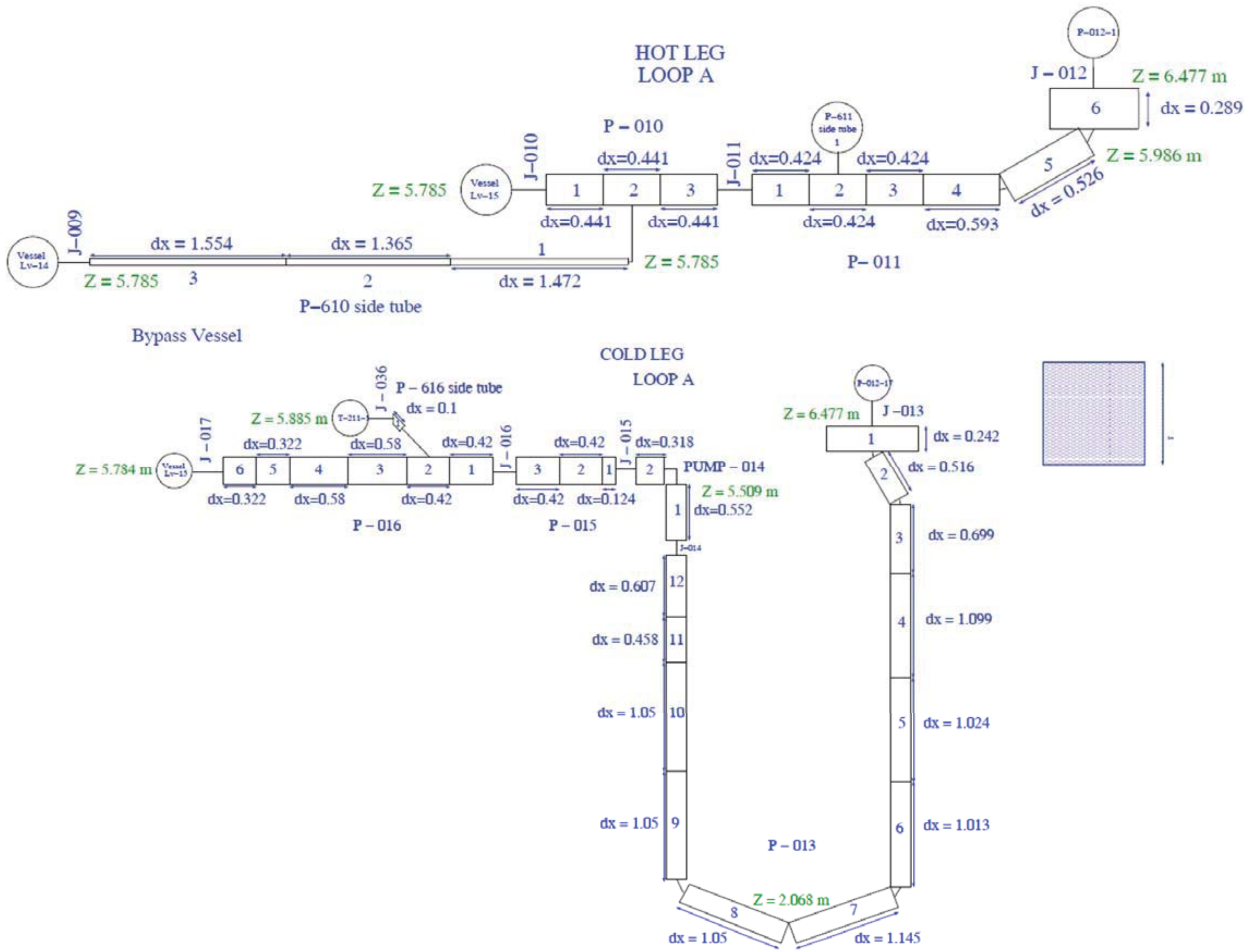


Figure 11: TRACE model of LSTF Hot and Cold Legs

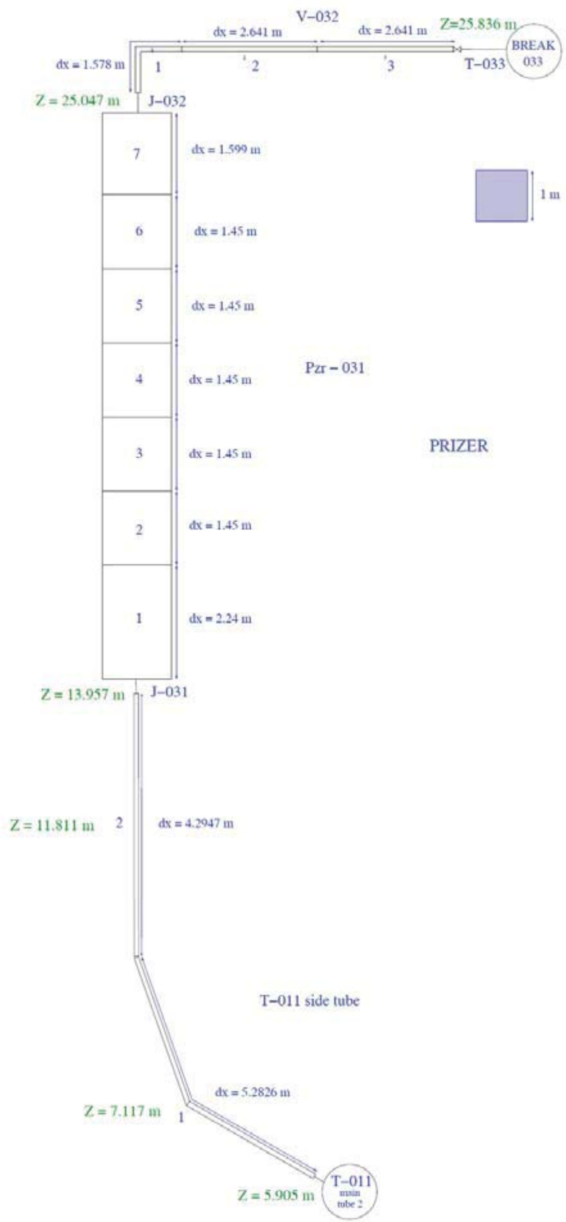


Figure 12: TRACE model of LSTF Pressurizer

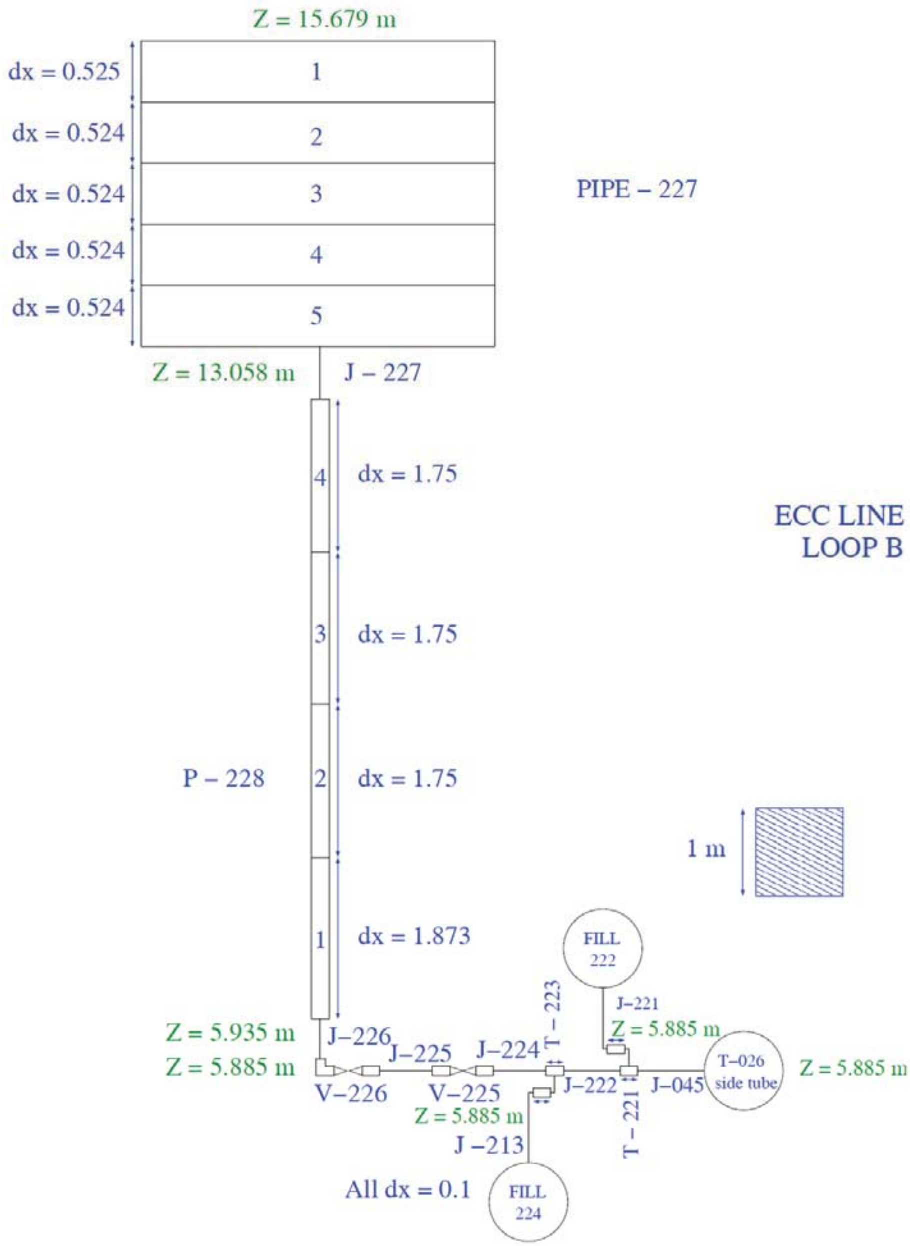


Figure 13: TRACE model of LSTF ECCS

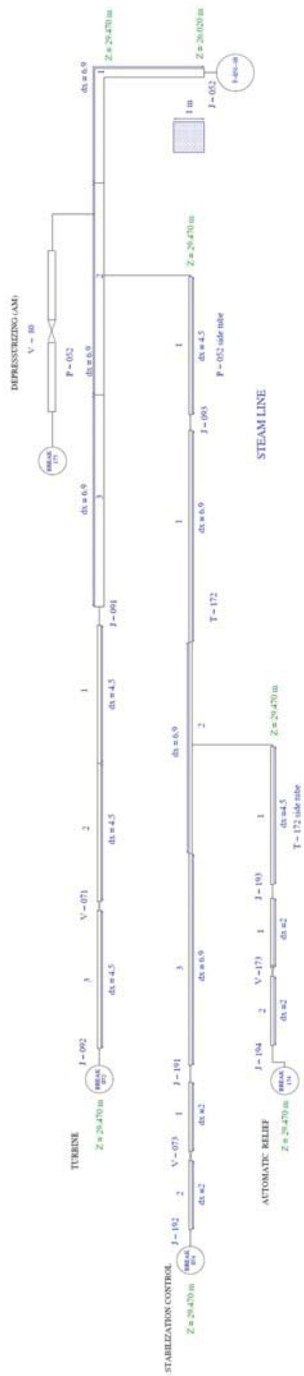


Figure 14: TRACE model of LSTF Steam Lines

3.2 Second Stage: LSTF TRACE model improvements

Later on, in a second stage, LSTF TRACE model was upgraded with the addition of several improvements to the model:

1. New 2D pressurizer model (see Figure 15) was included 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 (see Figure 16).
2. Pressurizer level and pressure control systems were added to set the steady state more adequately.
3. New detailed proportional and base heaters with more detail were also added in the pressurizer.
4. Heat losses of the whole model were adjusted (see Table 3).
5. The mass flow rate from the downcomer to the upper head of the vessel was adjusted to the specified one (0.3% of the downcomer vessel total mass flow).
6. The temperature in the Upper Head of the Vessel was adjusted to the measured one (about 586 K).
7. ΔP along the model was revised with reasonable results.
8. Control blocks, signal variables and trips were renumbered to avoid misunderstandings reading output data.
9. New signal variables to measure heat losses, surface temperature in core heaters, liquid level in upper head, core and PZR.
10. Several masses of HTSTR components were corrected, i.e. U-tube support plate.
11. The OFFTAKE model was activated in the connections of the valves that simulate breaks in different localizations of LSTF model.
12. Several models of break rupture discharge valve were tested for simulation of test 6.1.
13. New SG 9 heights U-tube model, with a more detailed nodalization (see Figure 20, and Table 4).
14. Adjustment of several RFRIC factors.
15. Reflood model was activated.
16. Improved nodalization and model dimensions for better correlation between model height/volume and LSTF height/volume (see Table 5):

- Modified vessel with 19 levels, improved nodalization and more adjusted lower and upper plenum, core, lower and upper head and downcomer (see Figure 17).
- Modified cold and hot legs in loop A and loop B (See Figure 18).
- Improved steam generators modelling (See Figure 19).

17. Finally, an animation mask was created with SNAP tool (see Figure 27). This mask allows performing videos of the simulations, which allows an easy interpretation of the transient behavior.

Modifications 1 to 12 lead to a first model (MODEL-1). Later, a second model with nine U-tube elevations was obtained in order to check the impact of several heights in the results (MODEL-2). As it will be shown later, this model did not improve results of MODEL-1. Two new modificactins (14 and 15), included in MODEL-3, improved the results although maintining slight deviations. Finally, a full review of nodalization and model dimensions was performed obtaining the last model (MODEL-4) that provided the best results. Nowadays, the TRACE model of ROSA/LSTF facility (see Figure 26) has 178 thermal-hydraulic components (2 VESSEL, 45 PIPE, 8 TEE, 2 SEPD, 22 VALVE, 2 PUMP, 9 FILL, 15 BREAK, 70 HTSTR and 3 POWER), 1013 Signal Variables, 167 Control Blocks and 20 Trips.

Component	**Temp(K)**	**LSTF (kW)**	**TRACE (kW)**
Pressurizer	620	15.0	15.41
Pressure Vessel	600	58.6	59.5
Primary Loop A	600	19.0	19.5
Primary Loop B	600	19.0	19.5
SG-A Plena	600	2.6	2.96
SG-B Plena	600	3.2	2.96
SG-A Sec. Vessel	560	24.1	28.05
SG-B Sec. Vessel	560	31.1	28.05
SG-A Downcomer Piping	560	3.5	3.27
SG-B Downcomer Piping	560	2.2	3.27

Table 3: Heat losses of the ROSA/LSTF System. Comparison between LSTF and TRACE model

Type	**R (mm)**	**L (mm)**	**Number**
1	50,8	9439,9	21
2	83,3	9590,7	19
3	115,8	9741,2	19
4	148,3	9891,7	19
5	180,8	10042,2	17
6	213,3	10192,7	15
7	245,8	10343,2	13
8	278,3	10493,7	11
9	310,8	10644,2	7
		total	**141**

Table 4: Added U-tubes geometry in steam generator model

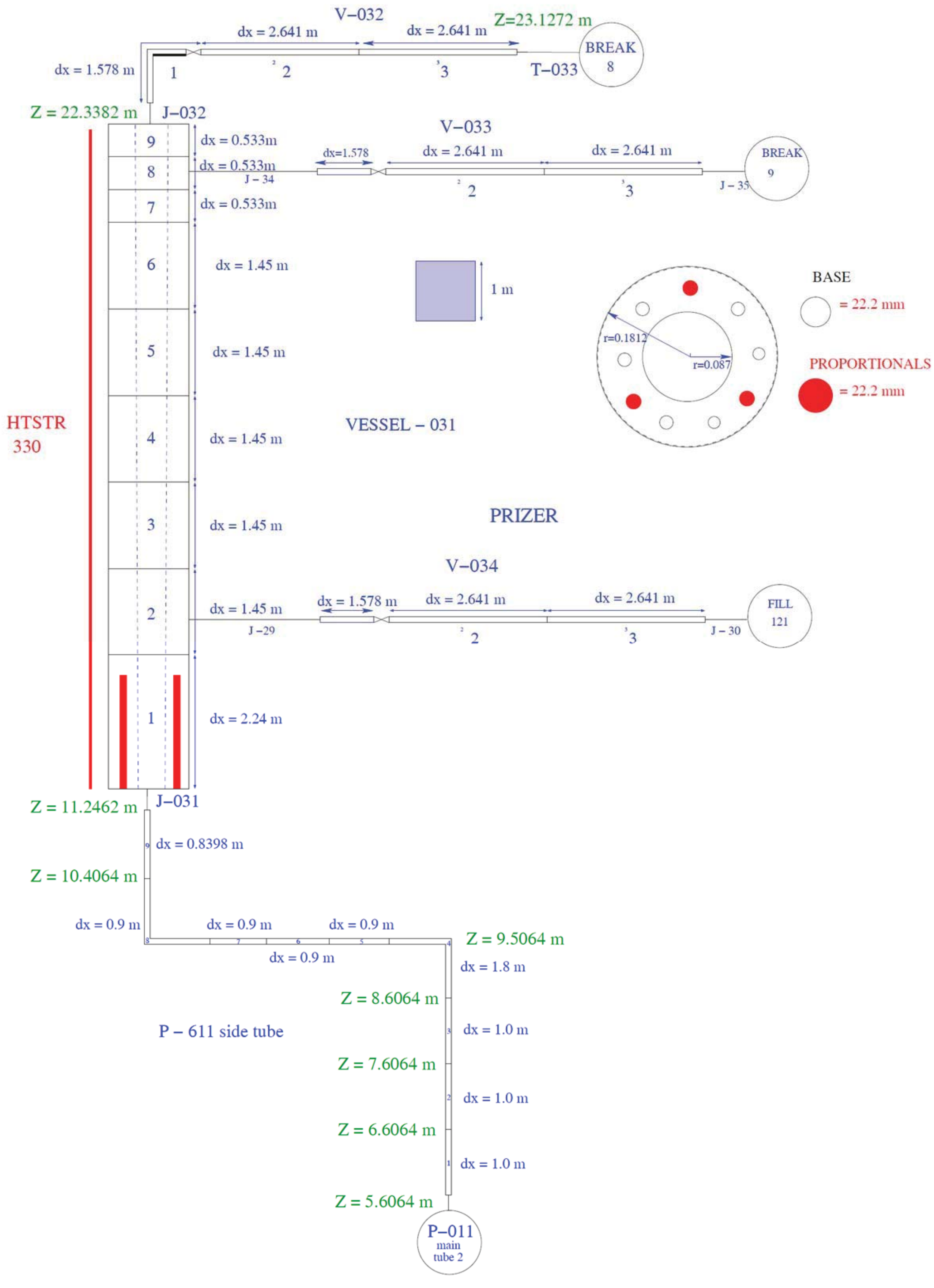
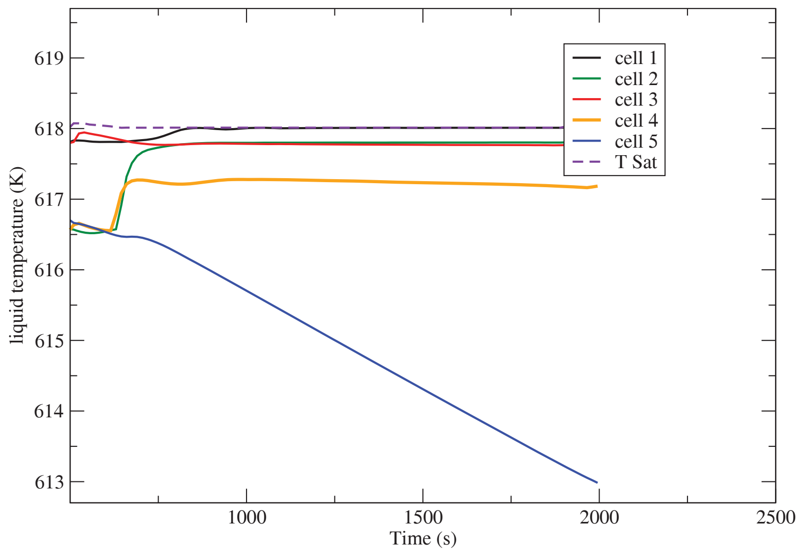


Figure 15: TRACE 2D model of LSTF pressurizer



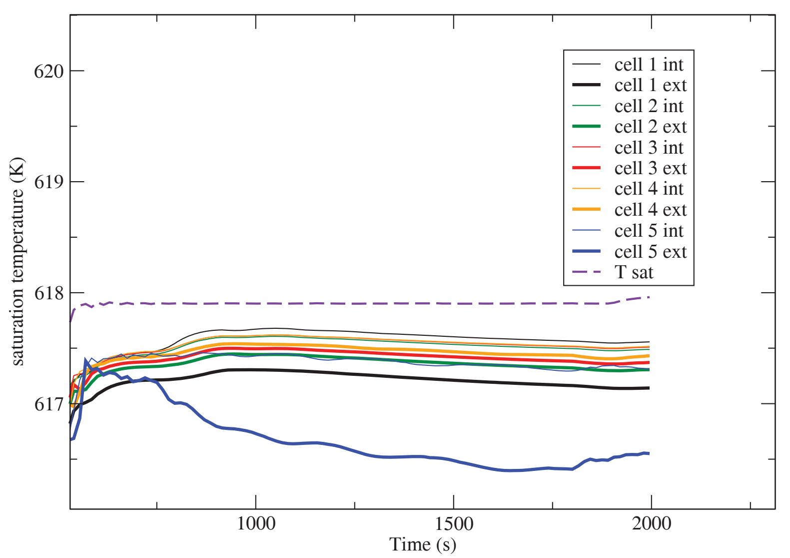


Figure 16: Liquid temperatures: 1D and 2D pressurizer models

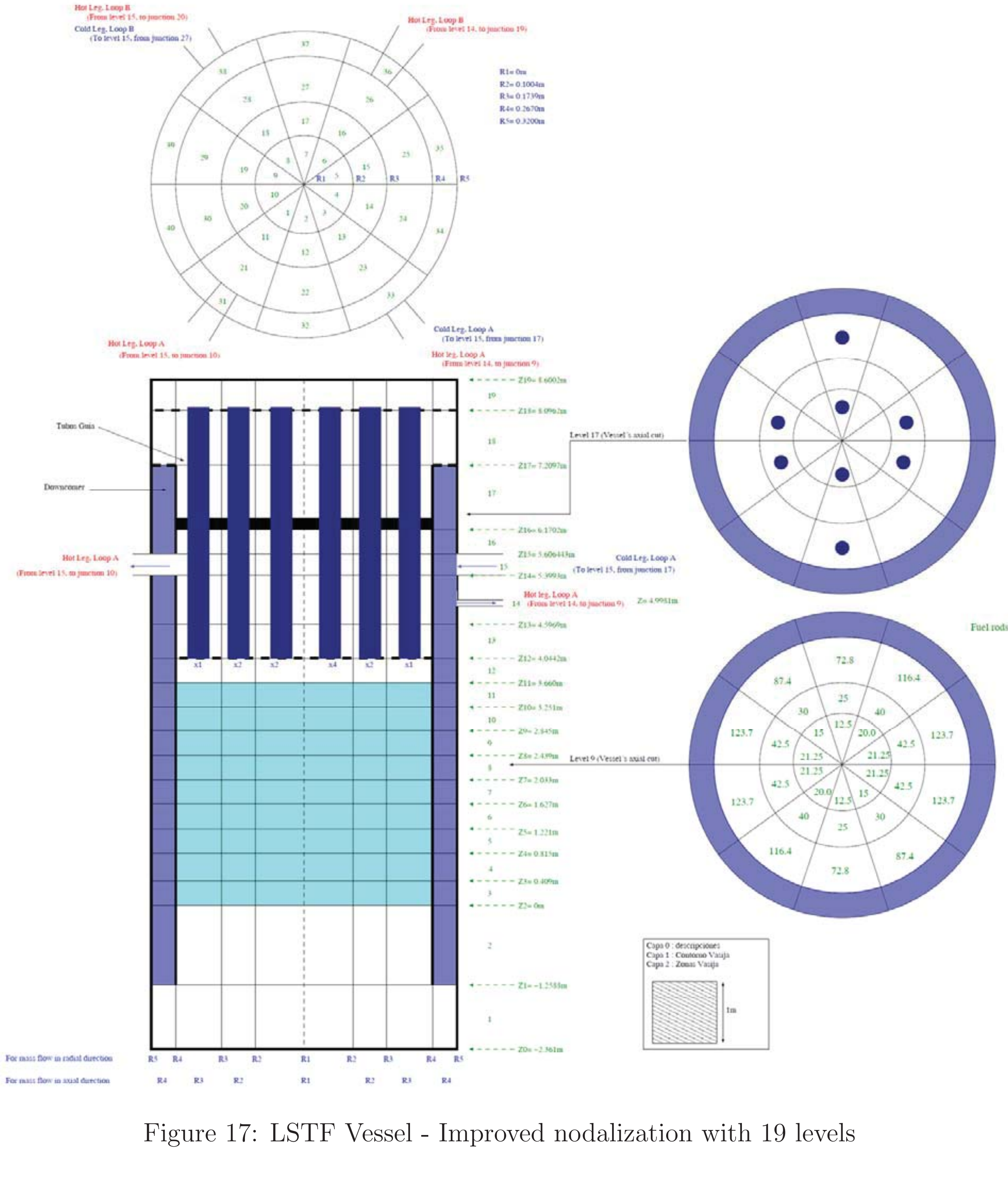


Figure 17: LSTF Vessel - Improved nodalization with 19 levels

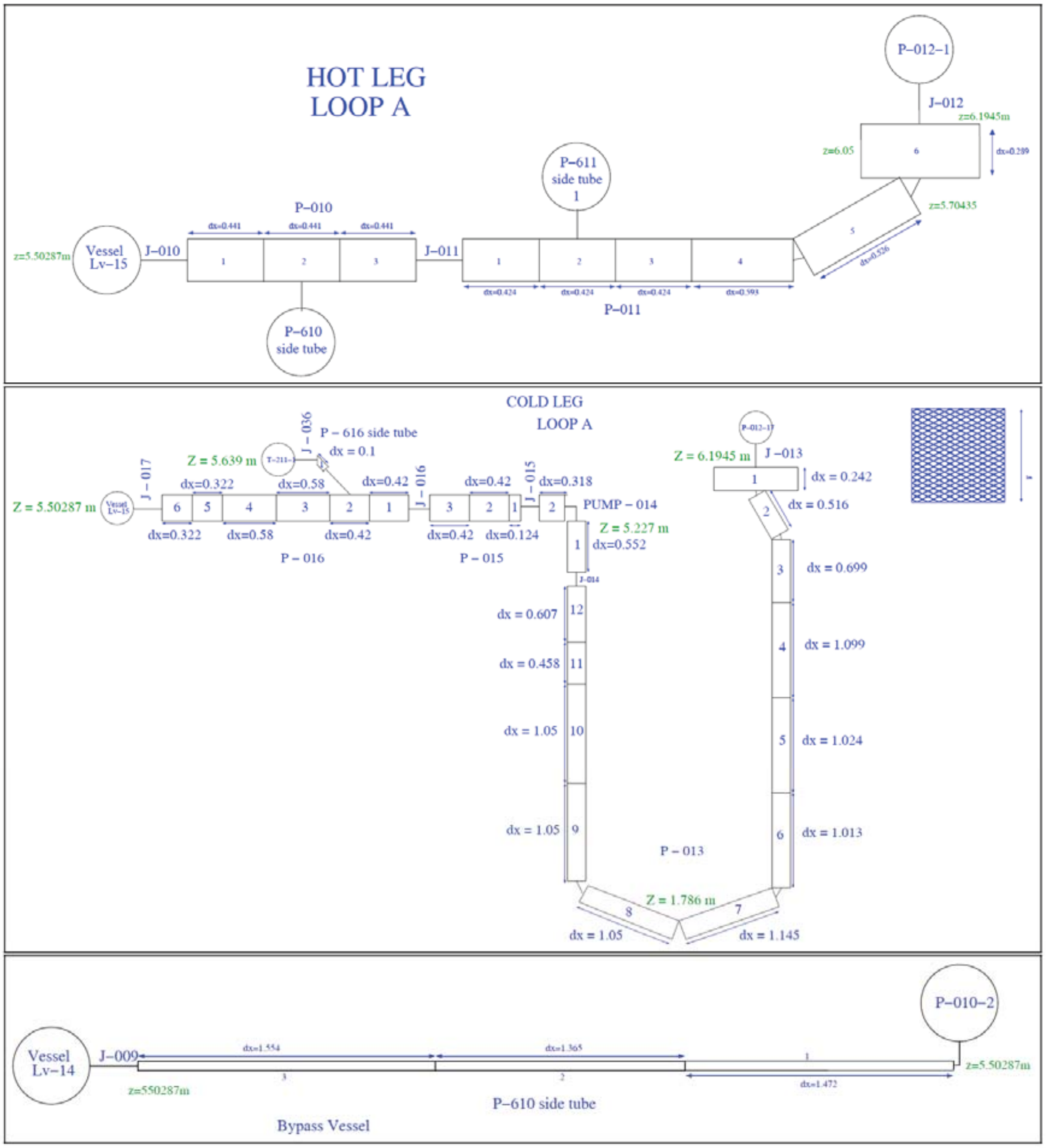


Figure 18: LSTF Hot and cold legs, and vessel bypass of improved model

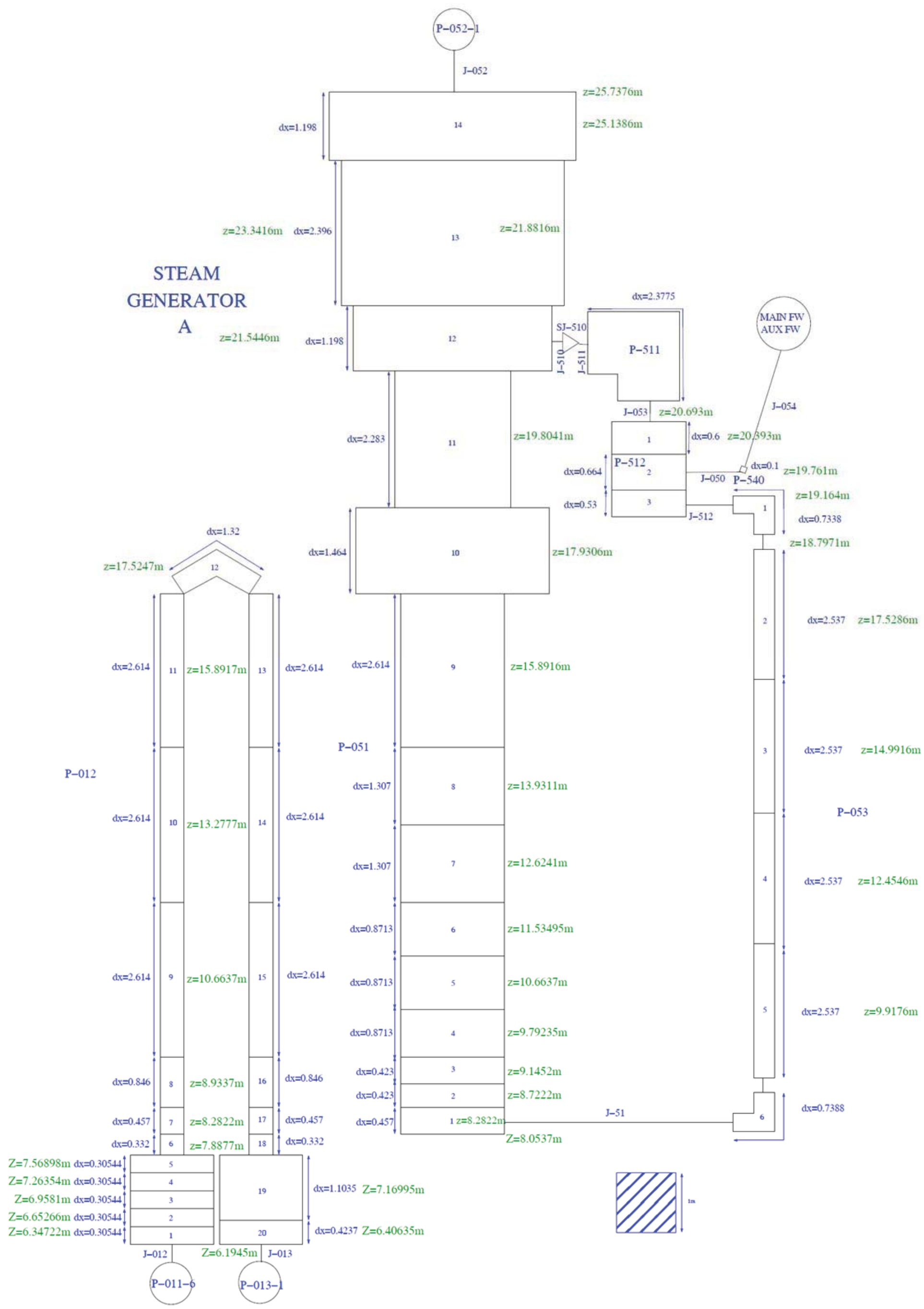


Figure 19: Improved LSTF steam generator model

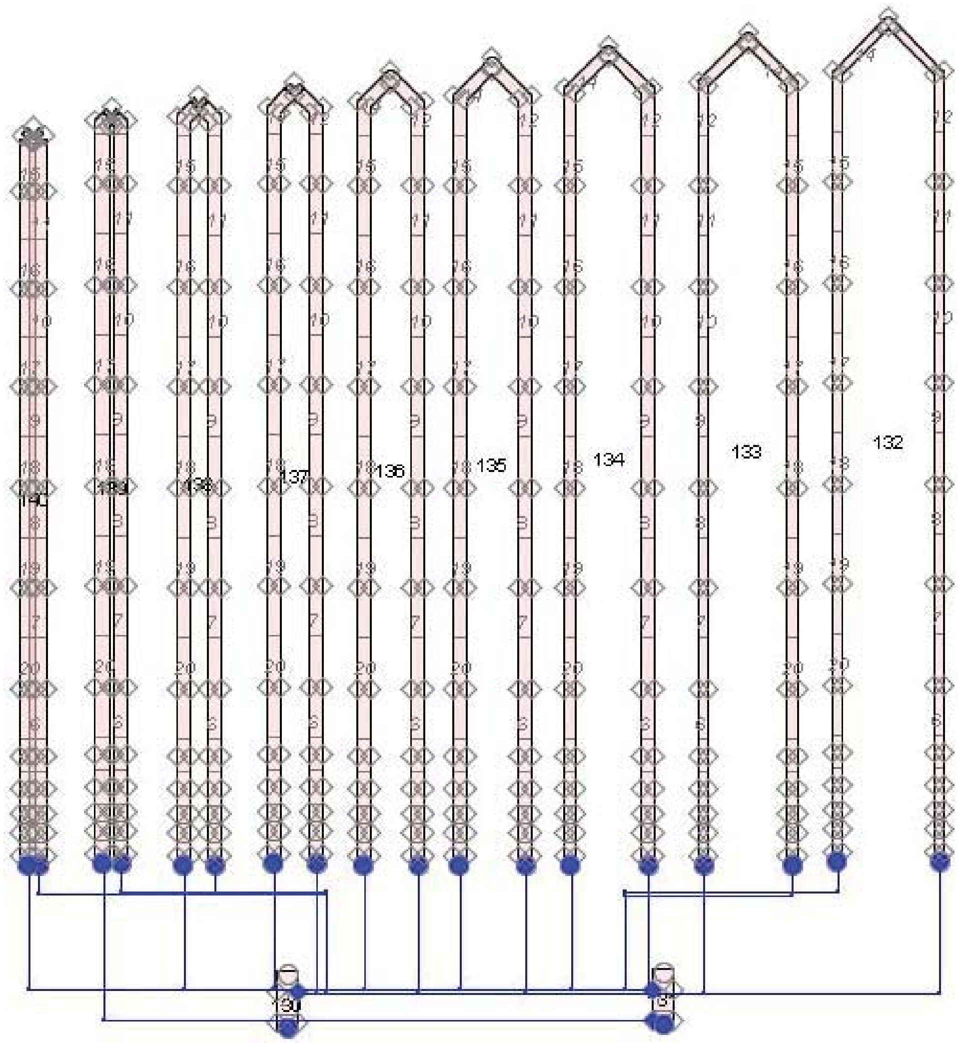


Figure 20: TRACE model of LSTF steam generator with 9 U-tubes

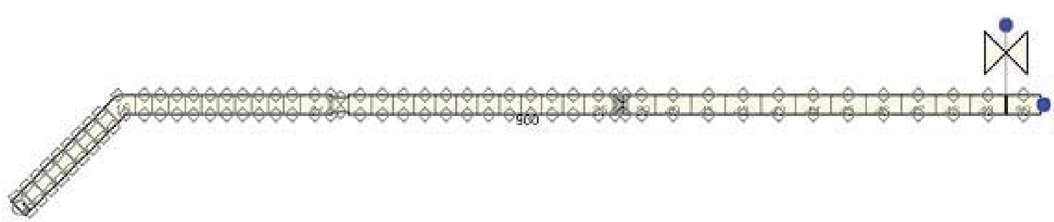
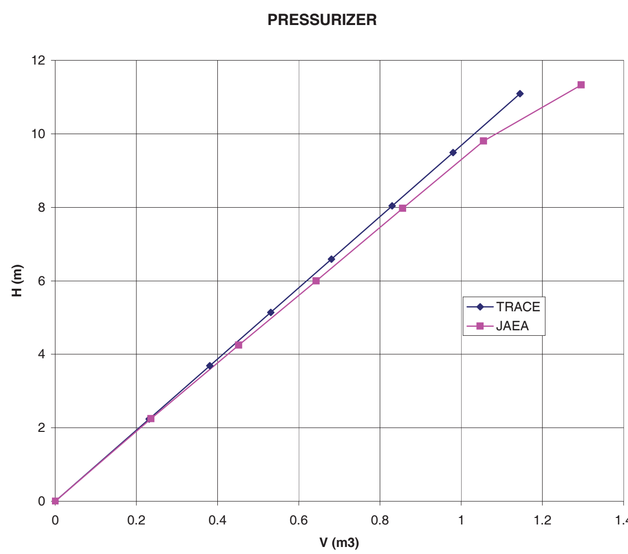


Figure 21: TRACE Model of LSTF break rupture discharge valve



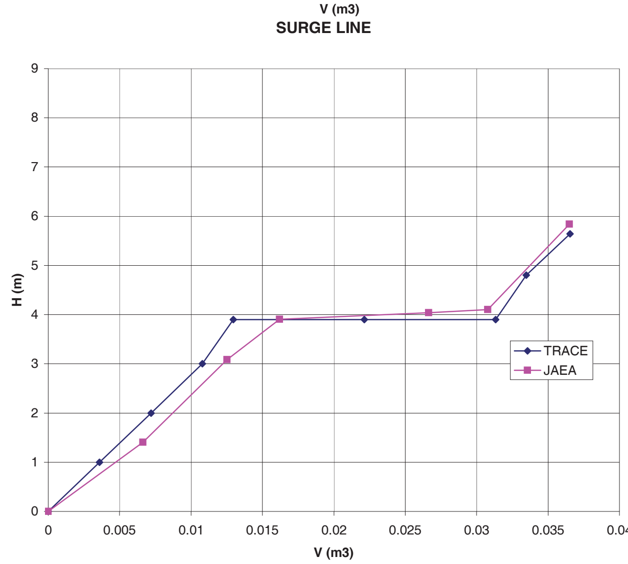


Figure 22: Comparison between volumes of PZR TRACE improved model and PZR of LSTF

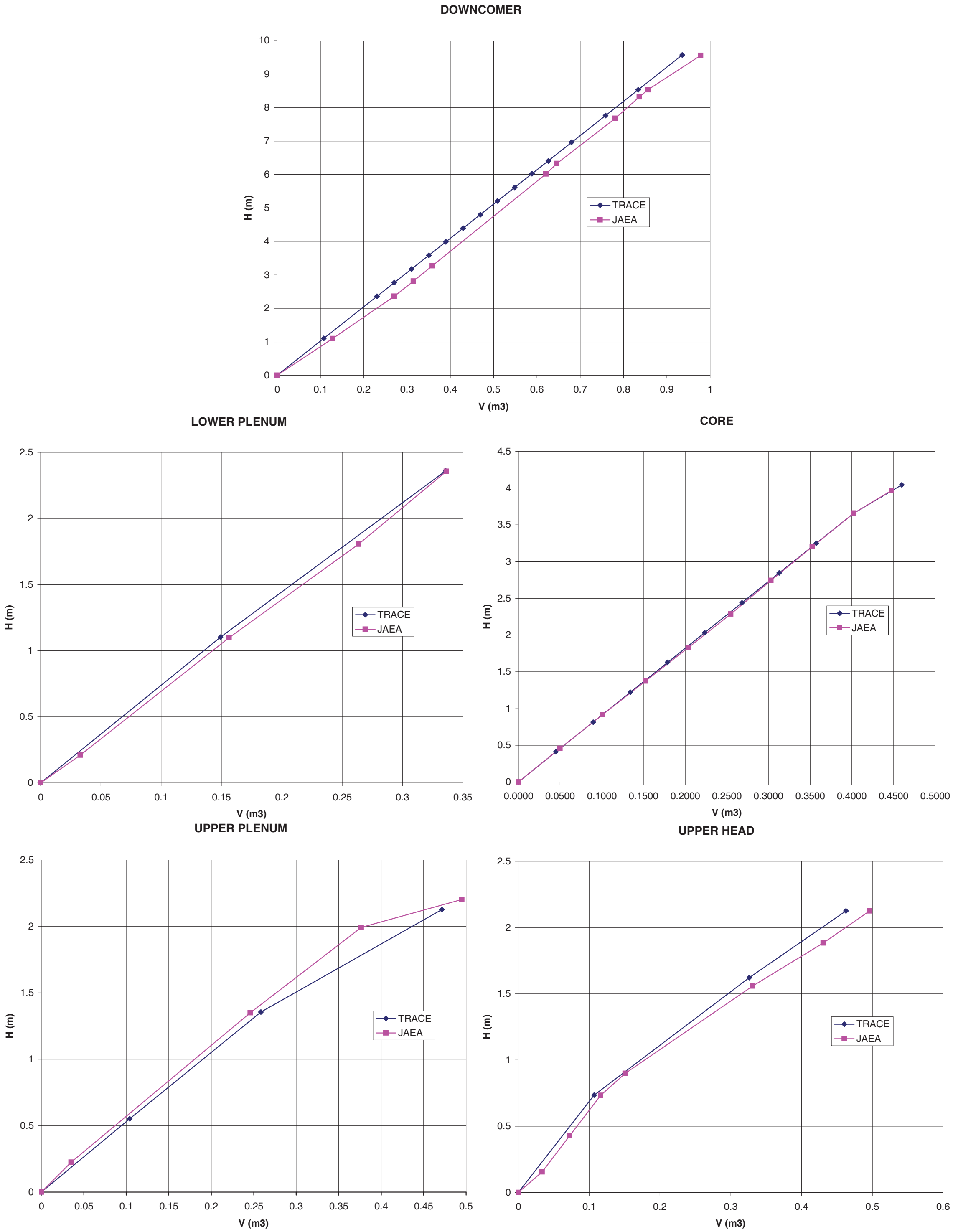


Figure 23: Comparison between volumes of TRACE improved model and LSTF for vessel components

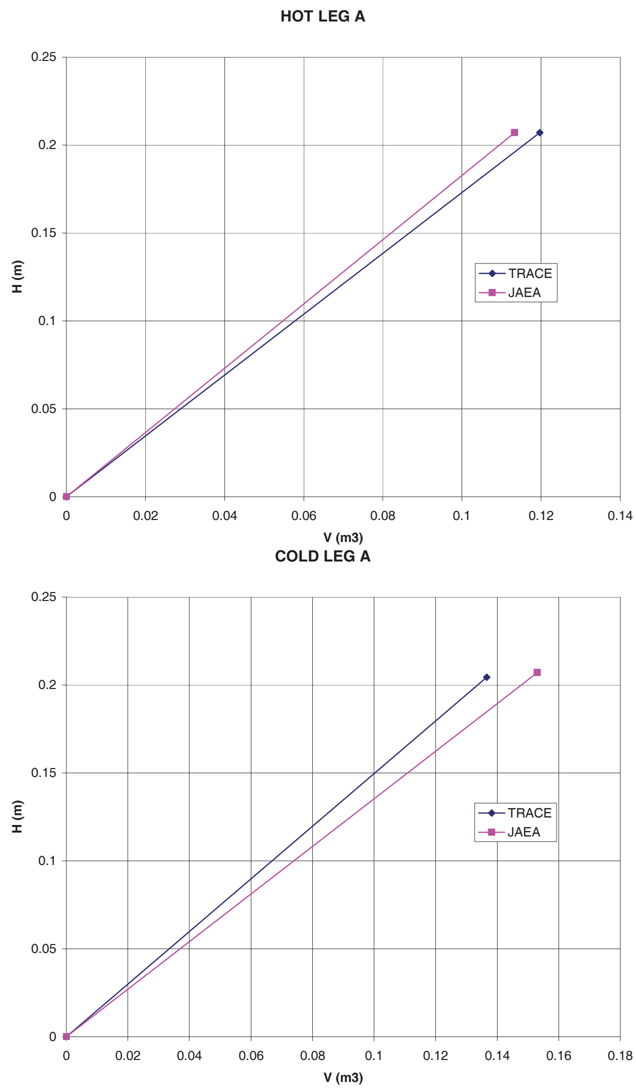


Figure 24: Comparison between volumes of TRACE improved model and LSTF for hot and cold legs

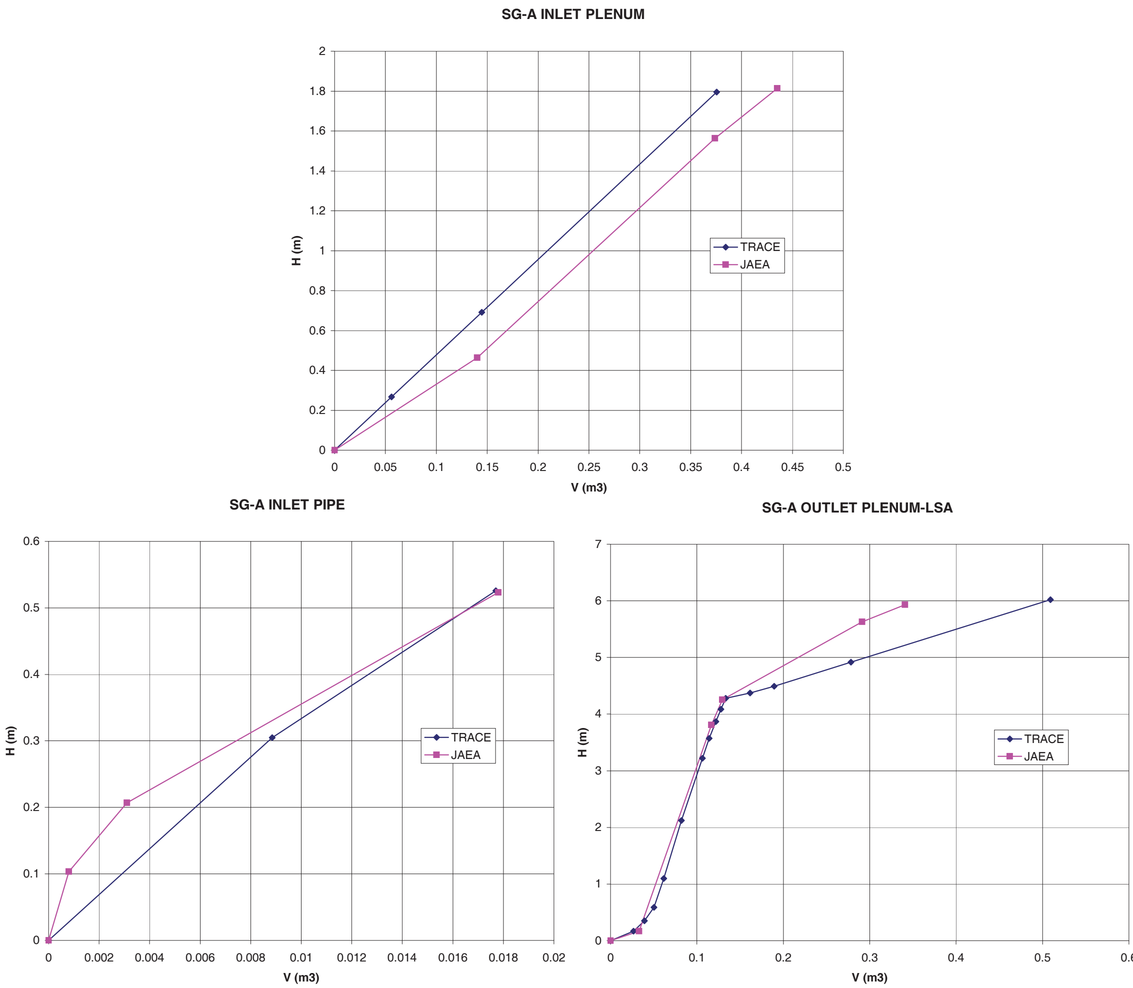


Figure 25: Comparison between volumes of TRACE improved model and LSTF for SG components

Component	**Volume** TRACE (m^3)	**Volume** JAEA (m^3)	Error (m^3)	Relative Error
Lower Plenum	0.3358	0.3364	-0.0006	-0.19%
Core	0.4598	0.4477	0.0121	2.7%
Upper Plenum	0.4714	0.4950	-0.0236	-4.77%
Upper Head	0.4963	0.4963	0.0	0.0%
Downcomer	0.9354	0.9784	-0.043	-4.4%
Total RPV	**2.6987**	**2.7538**	**-0.05513**	**-2.0%**
Prizer	1.2955	1.2955	0.0	0.0%
Surge Line	0.0365	0.0365	0.000003	0.08%
Total PZR	**1.332003**	**1.332**	**0.000003**	**0.0002%**
Hot Leg A	0.1133	0.1133	0.0	0.0%
Hot Leg B	0.1155	0.1156	-0.0001	-0.086%
Cold Leg A	0.1366	0.1531	-0.0165	-10.75%
Cold Leg B	0.1366	0.1617	-0.0250	-15.5%
LSA-PCA	0.1100	0.0921	0.0179	19.43%
LSB-PCB	0.1100	0.0926	0.0174	18.79%
Total loops	**0.72208**	**0.7284**	**-0.00632**	**-0.86%**
SGA Inlet Pipe	0.0177	0.0178	-0.0001	-0.56%
SGB Inlet Pipe	0.0177	0.0178	-0.0001	-0.56%
SGA Inlet Plenum	0.4306	0.4351	-0.0045	-1.03%
SGB Inlet Plenum	0.4306	0.4371	-0.0065	-1.49%
SGA U-tubes (IN)	0.43175	0.43346	-0.0017	-0.39%
SGB U-tubes (IN)	0.43175	0.43346	-0.0017	-0.39%
SGA U-tubes (OUT)	0.43175	0.43346	-0.0017	-0.39%
SGB U-tubes (OUT)	0.43175	0.43346	-0.0017	-0.39%
SGA Outlet Plenum-LSA	0.3416	0.3409	0.000737	0.22%
SGB Outlet Plenum-LSB	0.3416	0.3395	0.002137	0.63%
Total SGs primary	**3.3069**	**3.3222**	**-0.0153**	**-0.46%**
TOTAL PRIMARY	**8.0596**	**8.1364**	**-0.0768**	**-0.94%**

Table 5: Comparison between volumes of JAEA data and ROSA improved TRACE model

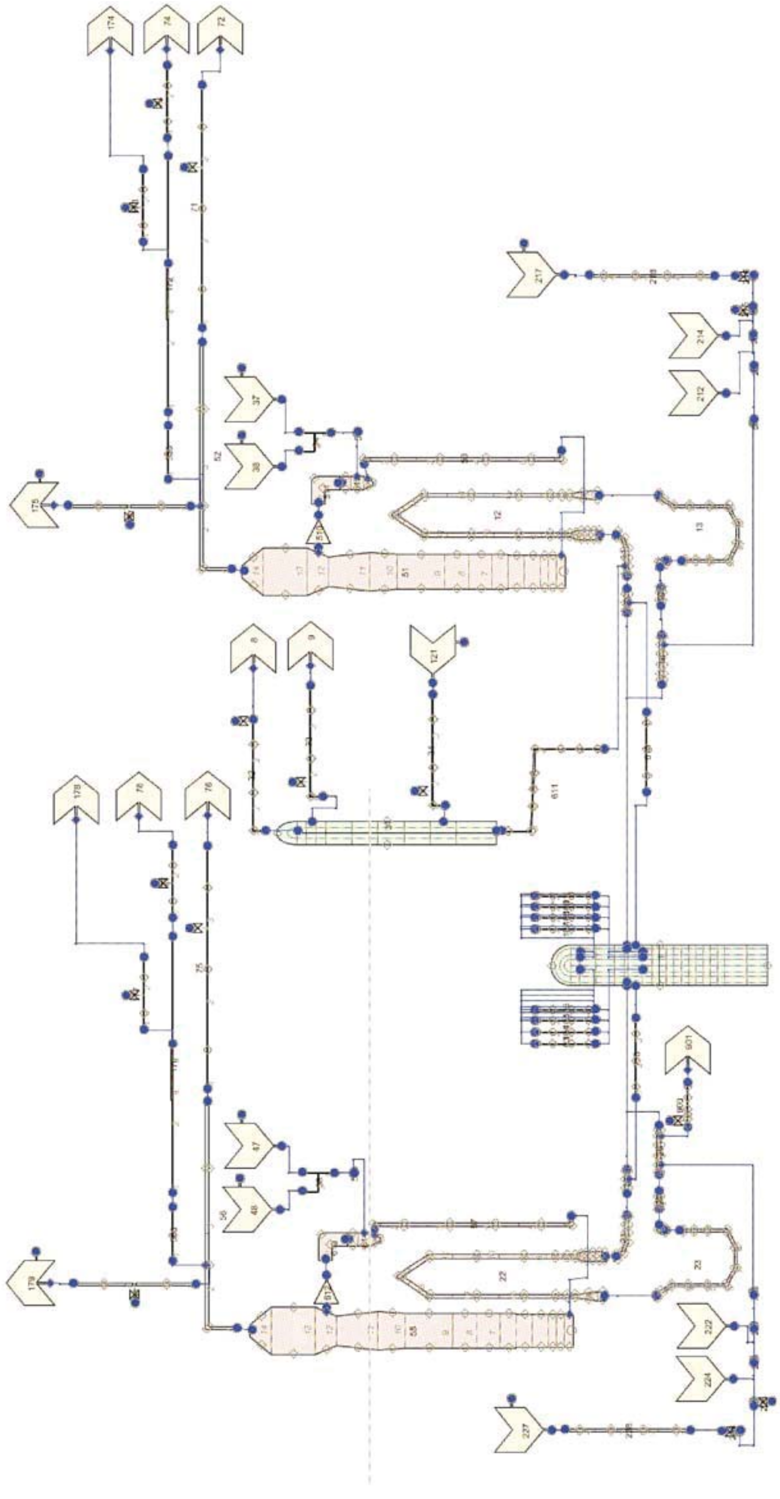


Figure 26: TRACE model of LSTF

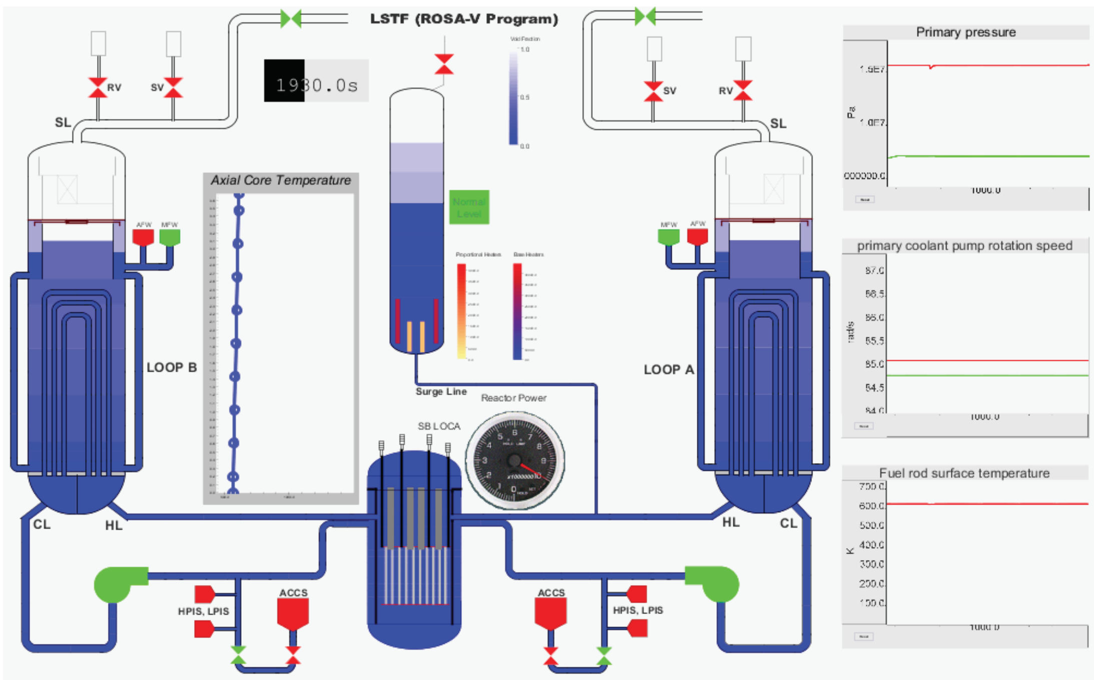


Figure 27: SNAP mask of TRACE model of LSTF

4 COMPARISON OF EXPERIMENTAL RESULTS AND SIMULATIONS OF OECD/NEA ROSA TEST 6.1

In a first step of the OECD/NEA ROSA test 6.1 simulation, the initial conditions were achieved. The obtained values were in reasonable agreement with experimental values, as shown in Table 6.

In a second step, several simulations with MODEL-1 (see page 3-10) were performed. Results of these simulations agreed very well with experimental results, with deviation in some TH parameters. Although phenomenology was well reproduced, a small delay in cladding and core exit temperatures (Figures 29 and 30) was obtained.

Therefore, another model was developed including a nine U-tube heights in steam generators (MODEL-2, see page 3-10). Nevertheless, the simulations performed with MODEL-2 provided similar results than MODEL-1, see Figures 28 to 32 for more details.

In order to capture the delay in cladding and core exit temperatures new modifications were tested (MODEL-3, see page 3-10). Firstly, several RFRIC factors were adjusted in the U-tube and later the reflood model option was activated. These modifications diminished the delay time between experimental and simulated data, as shown in Figure 33.

Finally, the test was simulated with the later and current TRACE model (MODEL-4, see page 3-10), which gives better results, as shown in Figures 34 to 38. The primary and secondary pressures matches fairly well with the experimental result. The evolution of the core uncovery has the same behavior as in the test as well as the CET temperature. There was only a small delay in primary pressure comparing to the test results. See Figures 34 to 38. As shown in Table 7, cronology of major events in the final TRACE simulation with MODEL-4 matchs fairly well with experimental results.

Several snapshots of the Test 6.1 video obtained through SNAP with the LSTF mask and the last TRACE simulation are shown in Figures 39 and 40, where void fraction is showed 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,
2. Upper head break and depressurization in RCS with PZR empty and beginning of

voiding inside vessel,

3. Coastdown of reactor coolant pumps and emptying of SG U-tubes,
4. Beginning of core uncovery,
5. Maximum core uncovery and peak cladding temperature,
6. Accumulators discharge and core reflood (end of the simulation).

Additionally to the post-test simulations, a sensitivity analysis of RCP trip has been performed. In Test 6.1 the RCP trip occurs at the same time than the opening of break valve, nevertherless the standard emergency operating procedures (EOP) of Westinghouse plants (as the reference of LSTF), used in this kind of LOCA sequences (EOP E-1 and ES-1.2) include two conditions in order to perform the RCP trip: loss of subcooling and availability of HPSI. In the test there was not availability of HPSI and therefore the RCP would nor be tripped in the case of EOP application. Next section present main conclusions and results of the impact of RCP trip delay.

Items	Specified	Measured (w/ (w/o) PZR)	TRACE (w/ (w/o) PZR)
Pressure Vessel			
Core power(MW)	10±0.07	10.12	10.00
Primary Loop			
Hot Leg Fluid Temperature(K)	598.1±2.75	598.0/597.7	598.88/598.84
Cold Leg Fluid Temperature(K)	562.4±2.75	563.5/563.3	561.91/561.88
Mass Flow Rate (kg/s / loop)	24.3±1.25	24.9/24.88	23.91/23.77
Downcomer to Hot Leg Bypass (kg/s)	0.049±0.01	0.05/0.045	0.048/0.048
Pressurizer			
Pressure(MPa)	15.5±0.108	15.51	15.5
Liquid Level(m)	7.2±0.25	7.18	7.30
Accumulator System			
Pressure(MPa)	4.51±0.054	4.52/4.51	4.55/4.55
Temperature	320±2.3/2.4	321.6/321.9	320.0/320.0
Secondary loop			
Secondary-Side Pressure (MPa)	7.3±0.054	7.33/7.33	7.09/7.08
Secondary-Side Liquid Level(m)	10.3±0.38	10.25/10.23	10.84/10.84
Steam Flow Rate (kg/s)	2.74±0.1	2.65/2.60	2.75/2.74
Main Feedwater Flow Rate (kg/s)	2.74±0.05	2.76/2.65	2.73/2.72
Main Feedwater Temperature(K)	495.2±2.63	495.9/495.1	495.2/495.2
Auxiliary Feedwater Temperature(K)	310±2.37	309.8	310/310

Table 6: Initial and boundary conditions for OECD/NEA ROSA test 6.1

EVENT	Test 6.1 TIME (s)	Simulation TIME (s)
Break valve open	2000	2000
SCRAM signal	2022	2019
SI signal	2028	2027
Primary coolant pumps stop	2276	2000
Primary pressure lower than SG secondary-side pressure	$\sim$ 2800	$\sim$ 2600
Core uncovery	$\sim$ 2800	$\sim$ 2800
Initiation of SG secondary-side depressurization	3090	3190
Core power decrease by LSTF core protection system	$\sim$ 3200	$\sim$ 3400
Initiation of accumulator system	$\sim$ 3300	$\sim$ 3600
Initiation of LPSI system	$\sim$ 4900	$\sim$ 5200

Table 7: Chronology of major events: comparison between OECD/NEA ROSA Test 6.1 and TRACE simulation

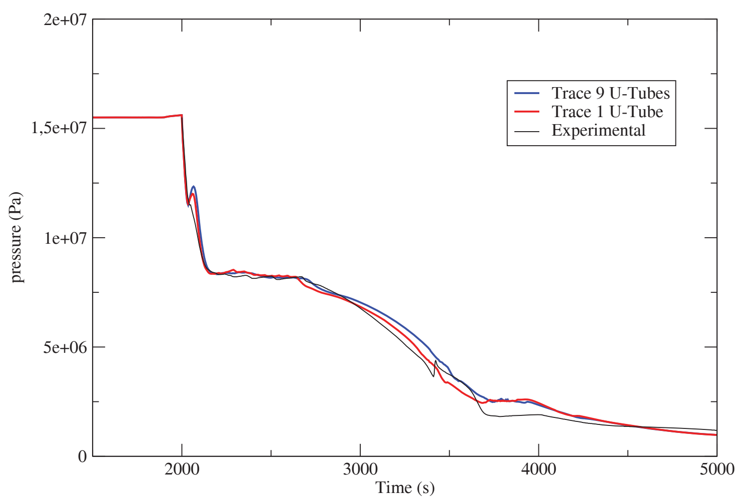


Figure 28: Primary pressure. Comparison between experimental and 1/9 U-tubes models (MODEL-1 and MODEL-2)

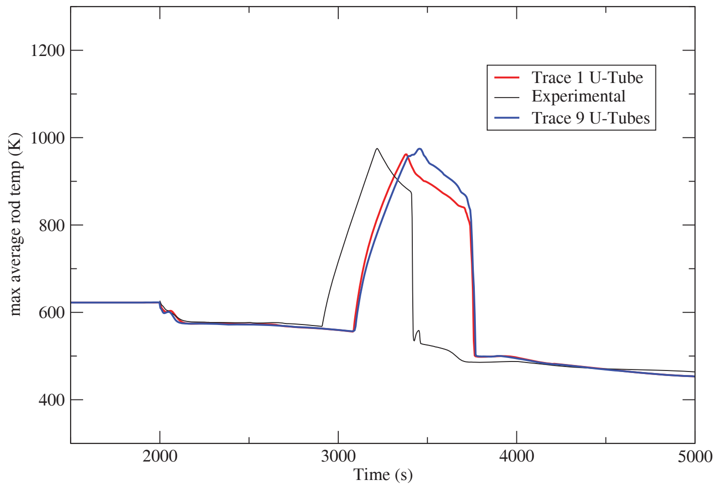


Figure 29: Peak cladding temperature. Comparison between experimental and 1/9 U-tubes models (MODEL-1 and MODEL-2)

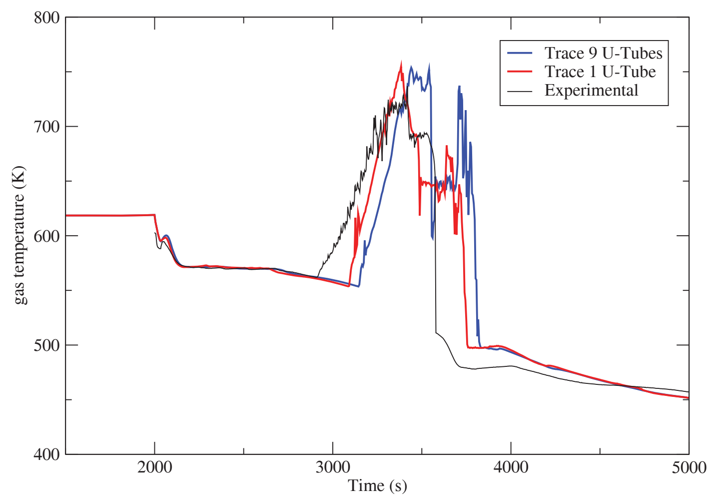


Figure 30: Core exit thermocouple temperature. Comparison between experimental and 1/9 U-tubes models (MODEL-1 and MODEL-2)

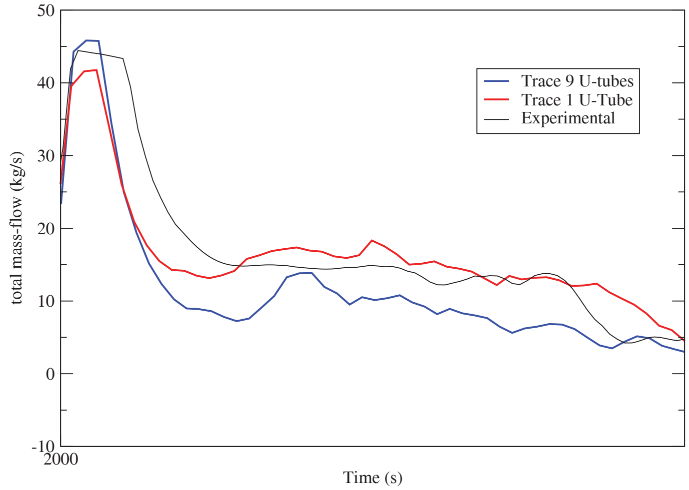


Figure 31: Loop B mass flow. Comparison between experimental and 1/9 U-tubes models (MODEL-1 and MODEL-2)

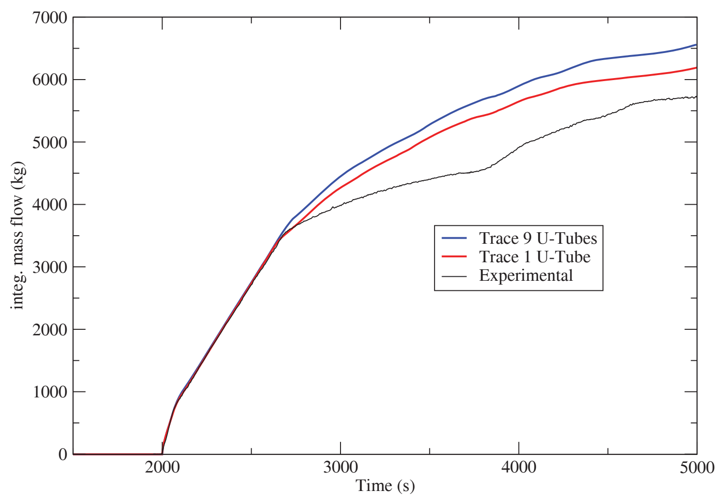


Figure 32: Break integrated mass flow. Comparison between experimental and 1/9 U-tubes models (MODEL-1 and MODEL-2)

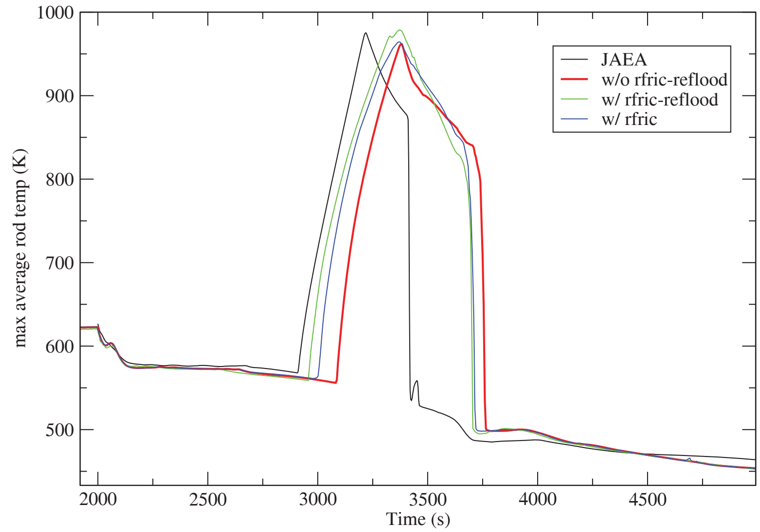


Figure 33: Peak Cladding Temperature. Comparison between experimental data and models with adjusted RFRIC and/or reflood option (MODEL-3)

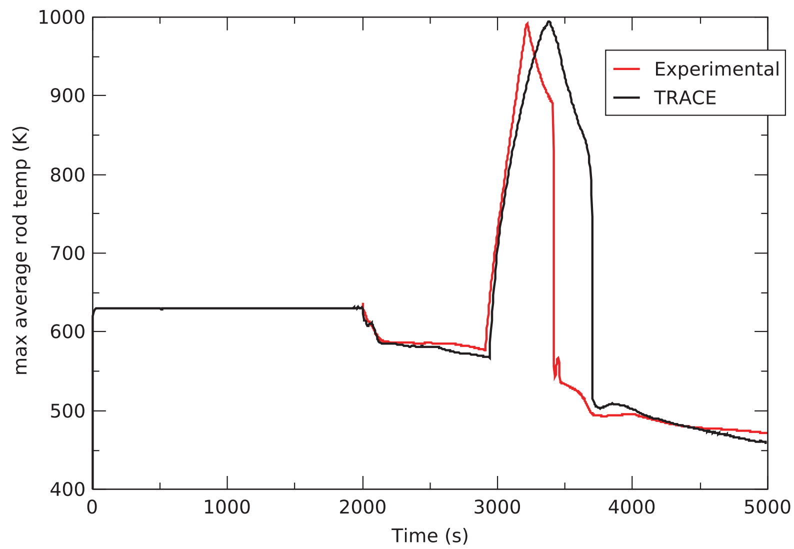


Figure 34: PCT for test 6.1 (MODEL-4)

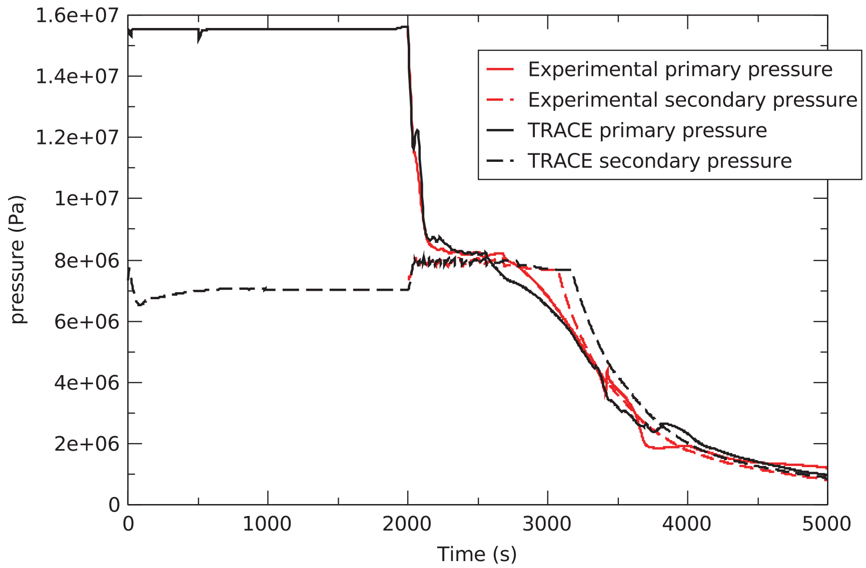


Figure 35: Primary and secondary pressures for test 6.1 (MODEL-4)

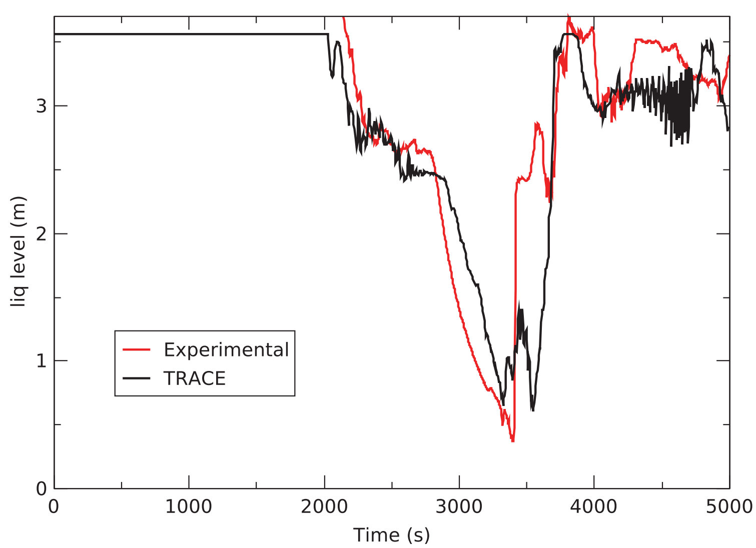


Figure 36: Core liquid level for test 6.1 (MODEL-4)

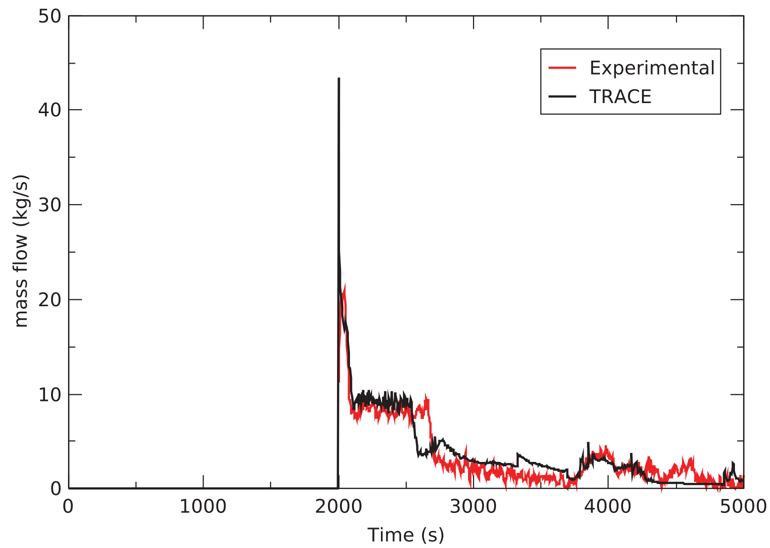


Figure 37: Break mass flow for test 6.1 (MODEL-4)

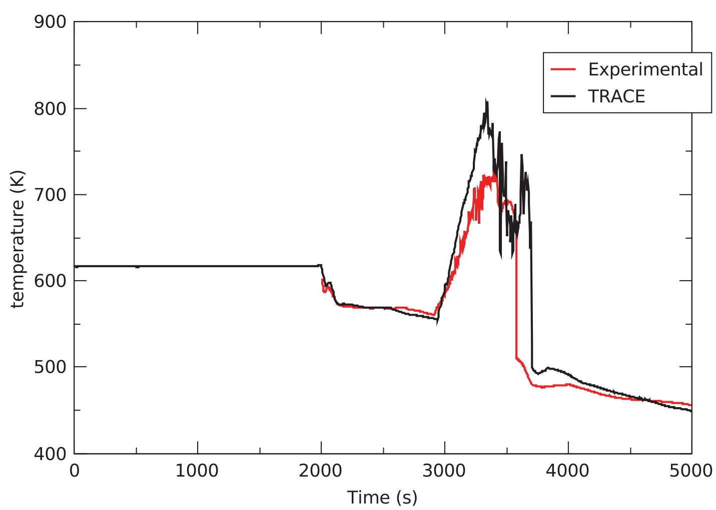


Figure 38: Temperature at core exit for test 6.1 (MODEL-4)

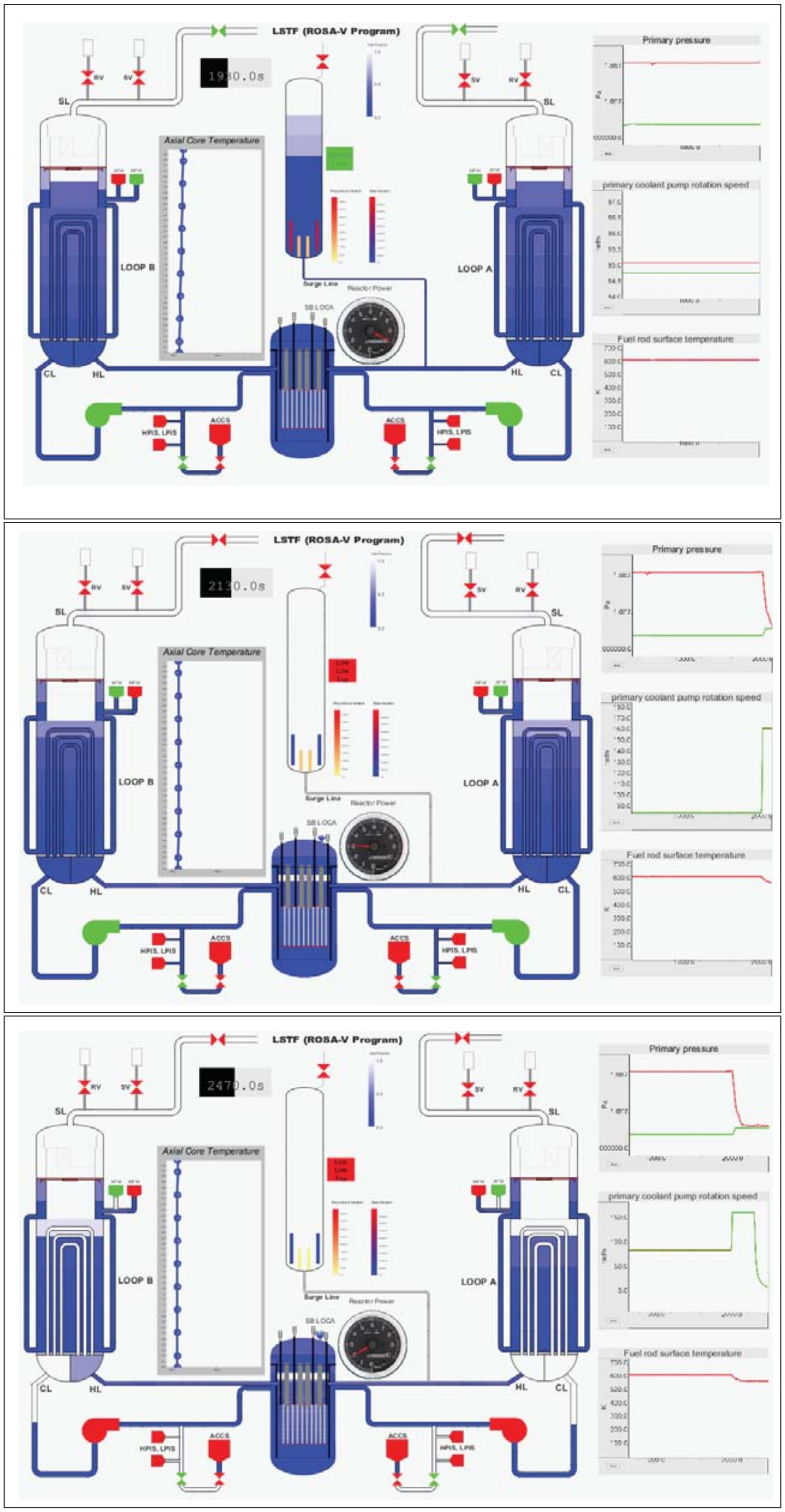


Figure 39: Void fraction in primary and secondary sides. Snapshots of SNAP video simulation (1/2)

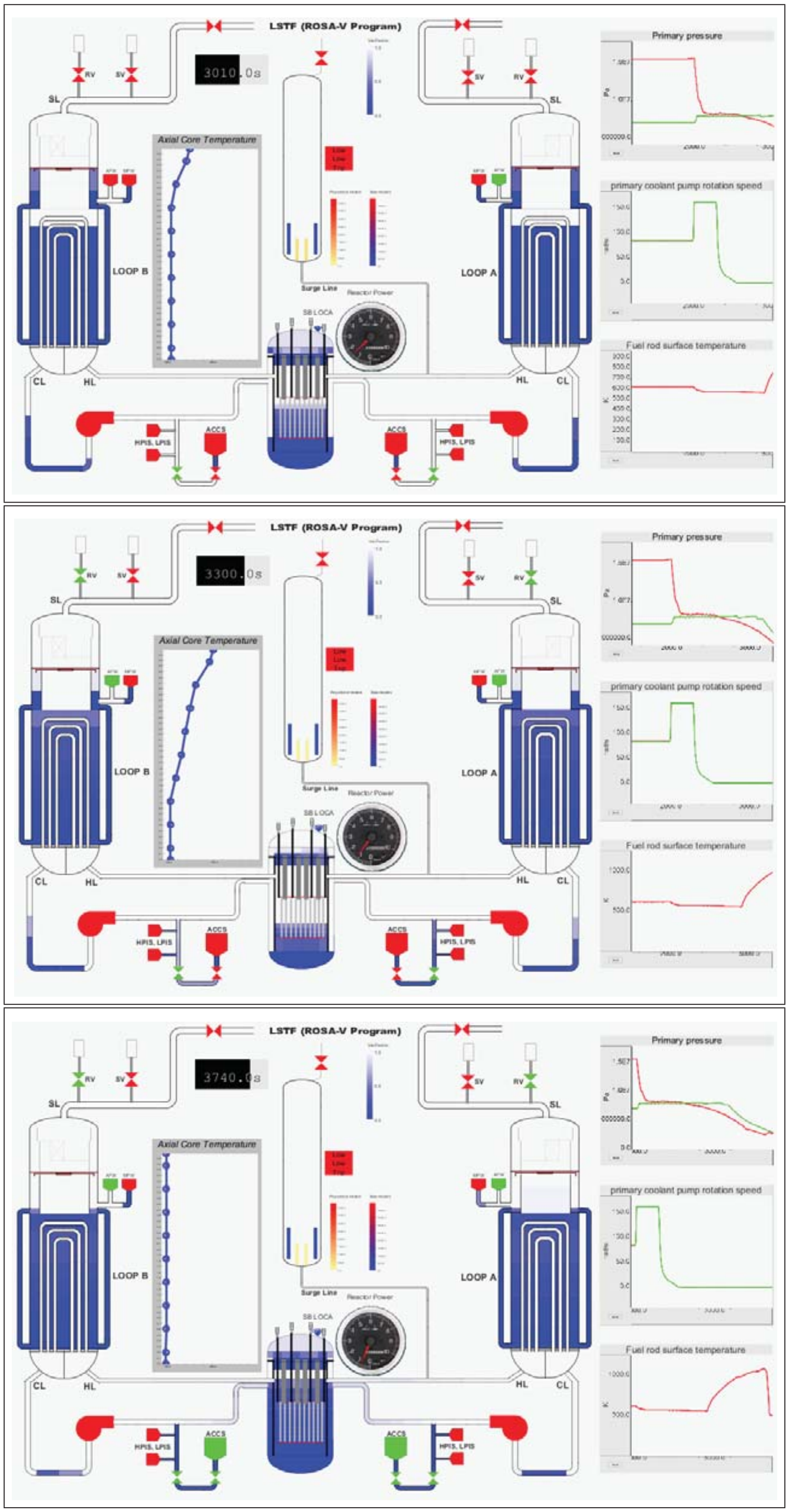


Figure 40: Void fraction in primary and secondary sides. Snapshots of SNAP video simulation (2/2)

4.1 Sensitivity analysis to RCP trip delay

This section describes the main results and conclusins of a sensitivity analysis with repect the instant of RCP trip, trying to identify the most penalizing conditions. Figures 42 to 47 include the results of simulations for different RCP trip times. In all cases, transient is simulated with core protection system activated and deactivated. The sensitivity analisys shows that the worst results are obtained for intermediate times of RCP trip, as shown in Figure 47, where PCT is depicted as a function of RCP trip time (with and without core protection system). These results prove that this task included in Westinghouse EOPs has a great impact in the evolution of the transient. This issue is well known since TMI accident due to several experimental tests and simulations that were performed for Westinghouse reactors for SBLOCA sequences with and without availability of HPSI. The conclusion of this sensitivity analysis is that if the objective of the experimental test is to analyze the expected phenomenology in present Westinghouse nuclear power plants then it could be interesting to perform a similar test without RCP trip. This kind of test could also be interesting in order to analyze the behavior of the core exit thermocouple in sequences with similar conditions to present Westinghouse EOPs.

So, in general, before conducting an experiment would be important to review the EOPs of the different vendors that participate in the project in order to decide which management actions are more adequate for implementing in the experiment. Particularly, the emergency operating procedures (EOP) of Westinghouse used in LOCA sequences (EOP E-1 and ES-1.2, see Figure 41) include two conditions in order to perform the RCP trip: loss of subcooling and availability of HPSI.

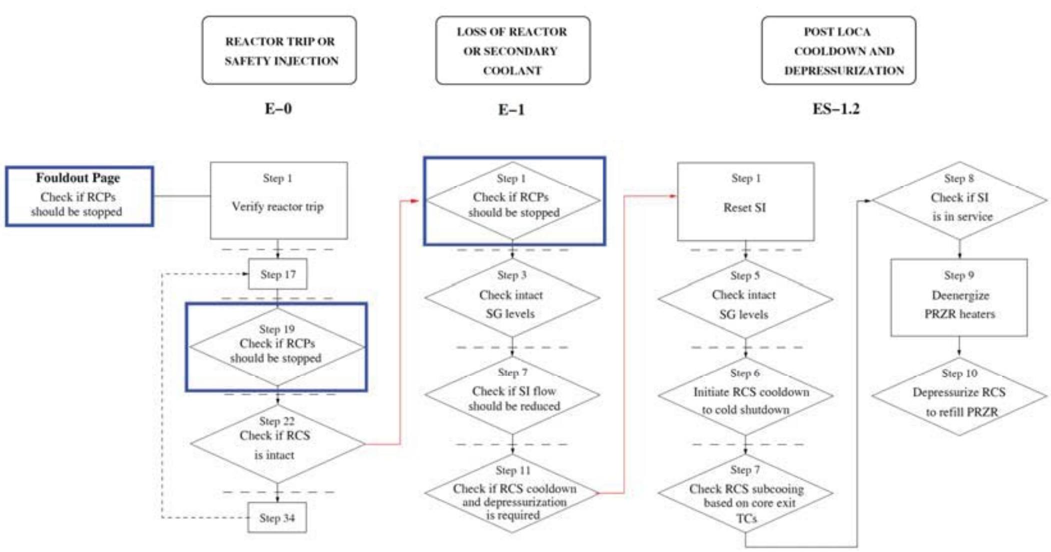


Figure 41: Simplified scheme of standar Westinghouse EOPs in LOCA scenaries

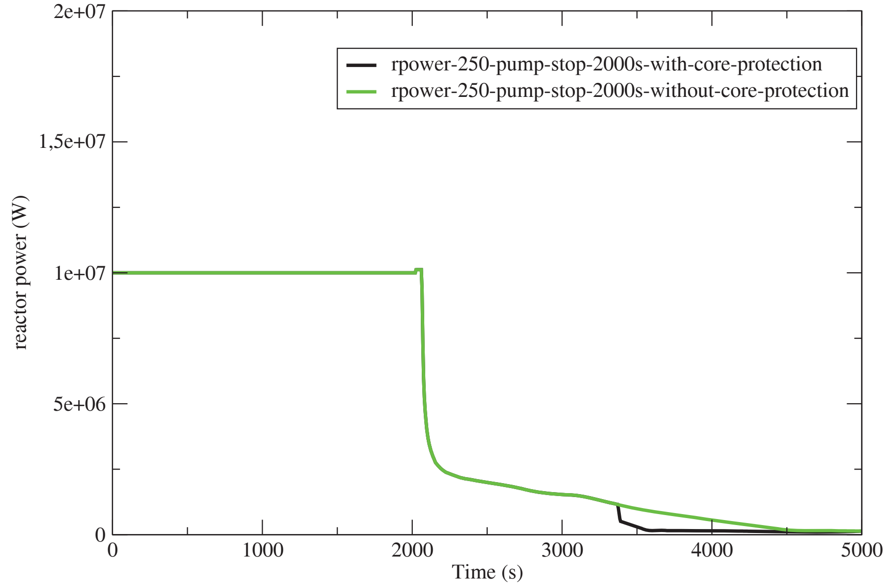


Figure 42: Core power with and without core protection system

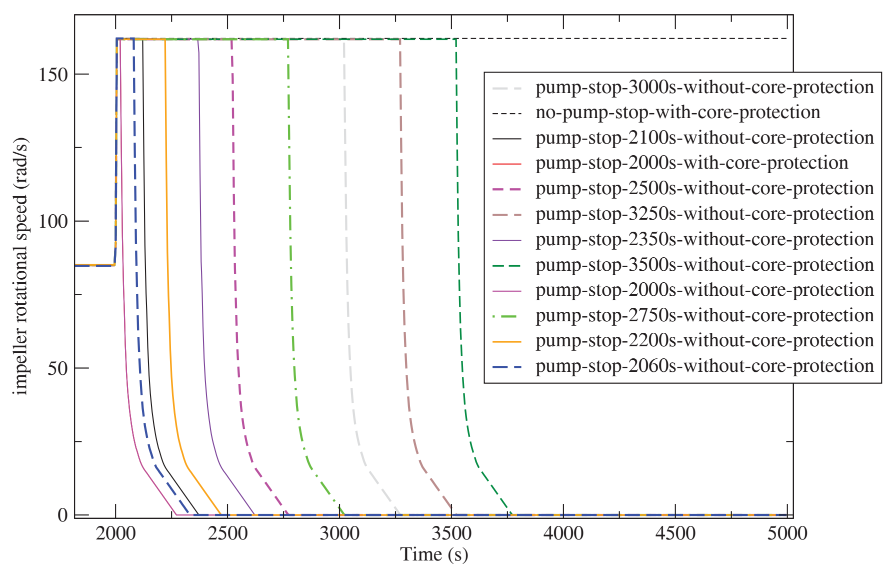


Figure 43: Pump rotational speed for different RCP trip times

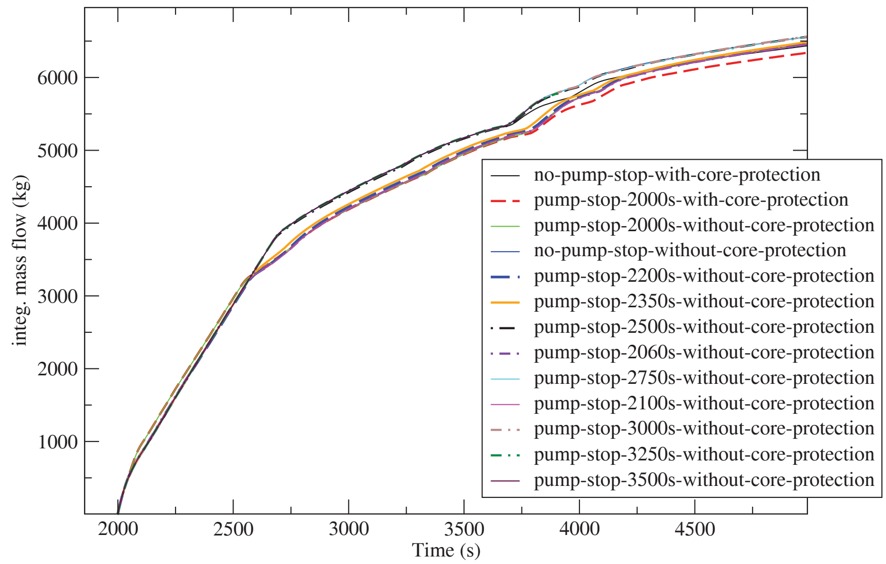


Figure 44: Integrated break mass flow for different RCP trip times

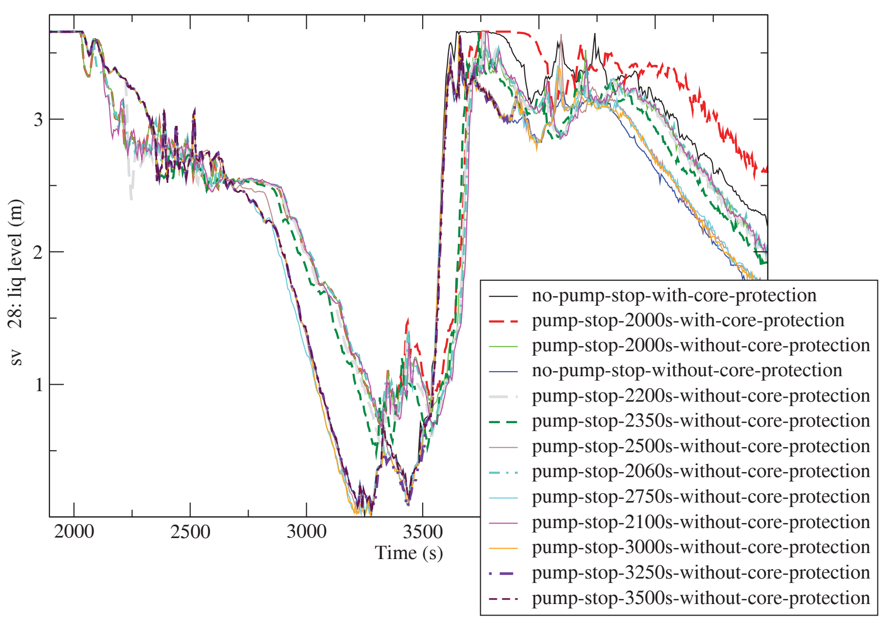


Figure 45: Core level for different RCP trip times

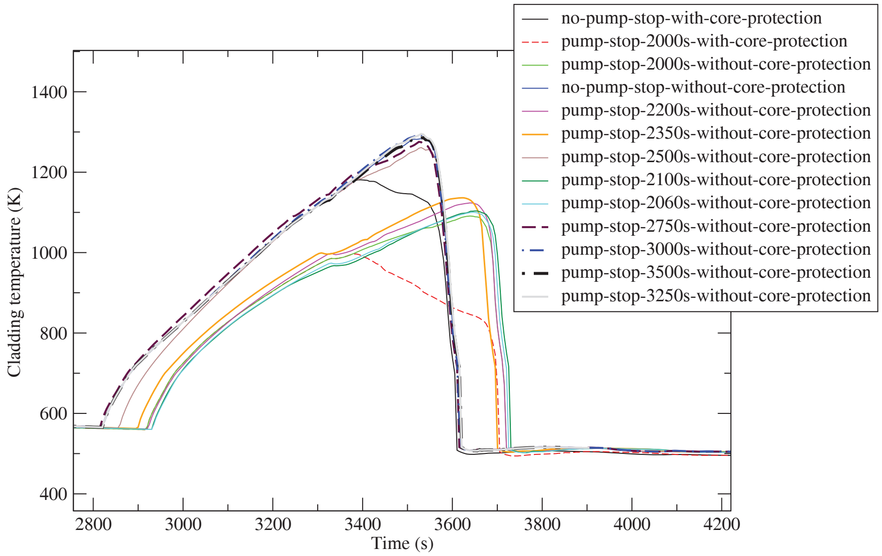


Figure 46: PCT for different RCP trip times

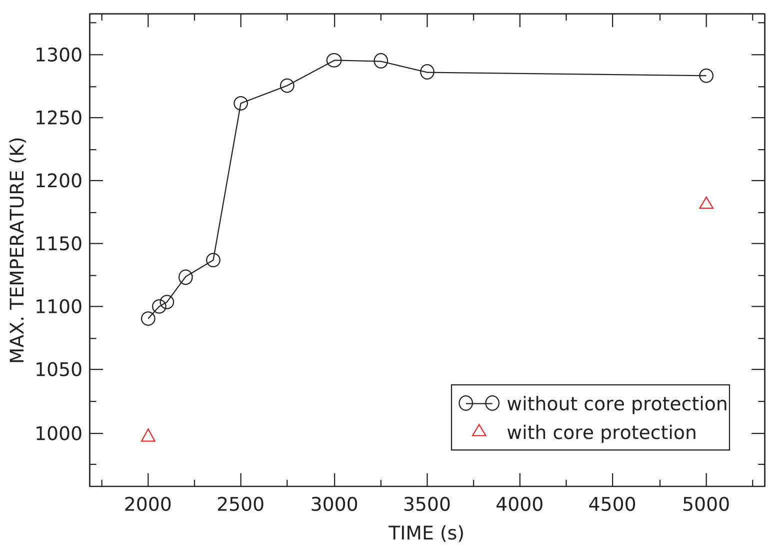


Figure 47: PCT, with and without core protection system. Values at 5000 s represent no RCP trip.

5 SIMULATION OF UPPER HEAD SBLOCA 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 Gas Natural Fenosa (11%). Comertial 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 48, 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 48 and 49. In Table 8 can be read the main operating parameters for both units.

This section presents main simulation results of similar transients to test 6.1 with the TRACE model of Almaraz NPP, taking into account different accident management actions and conditions.

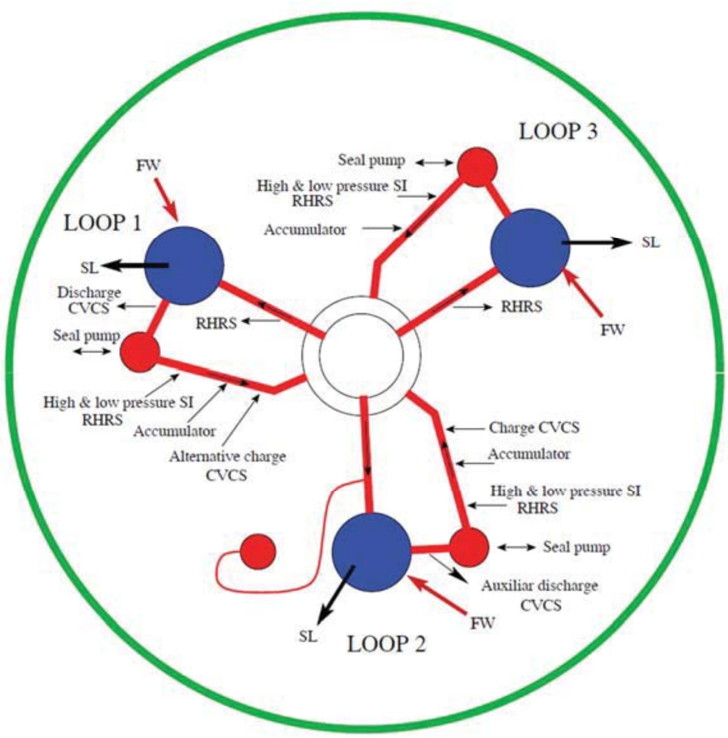


Figure 48: Scheme of Almaraz NPP (plan view)

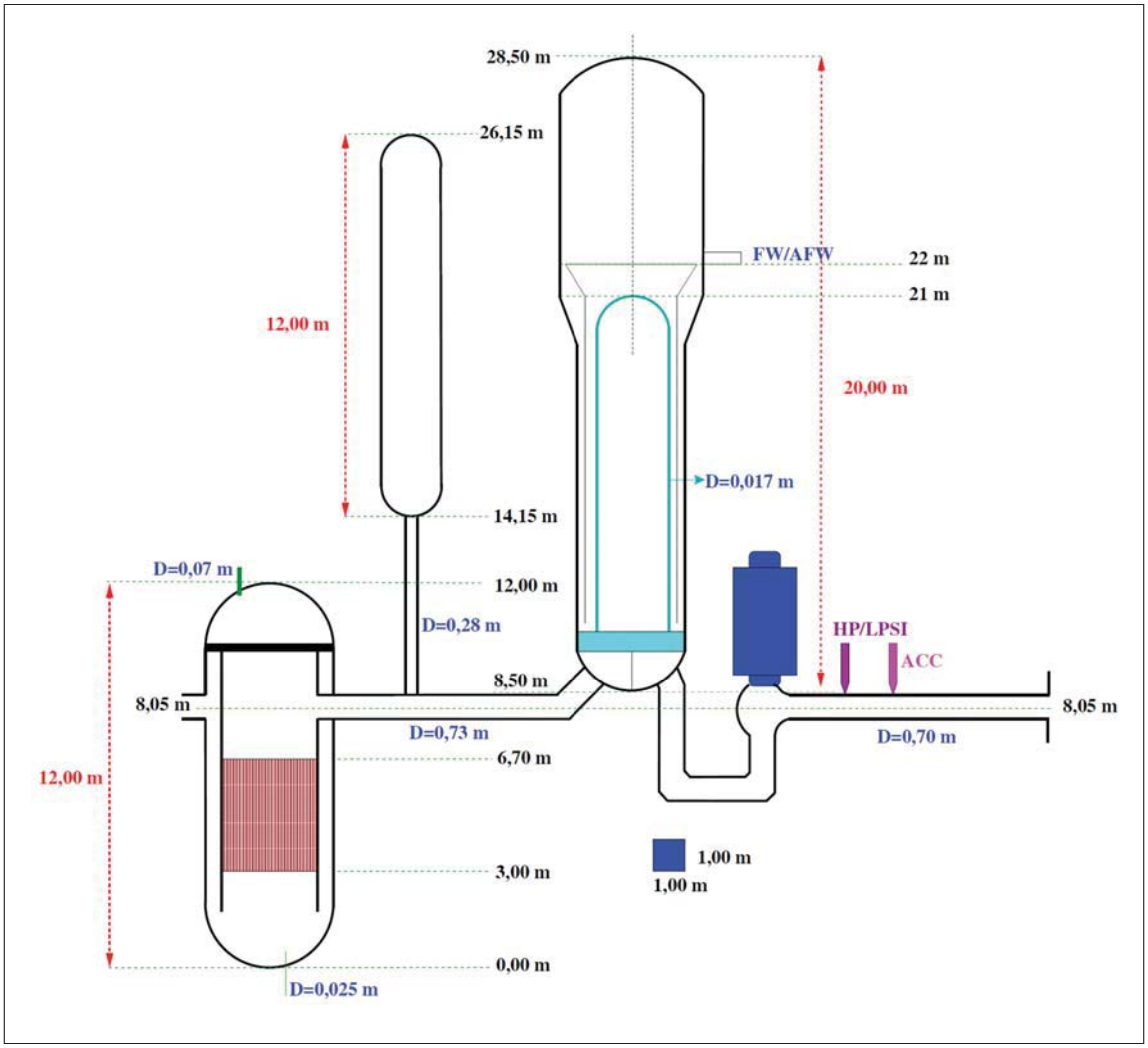


Figure 49: Scheme of Almaraz NPP (elevation view)

Table 8: 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 50 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 51, 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, and the steam dump with the eight valves.
- FW and AFW systems. Feed water pumps coastdown and auxiliary mass flows are included as boundary conditions.

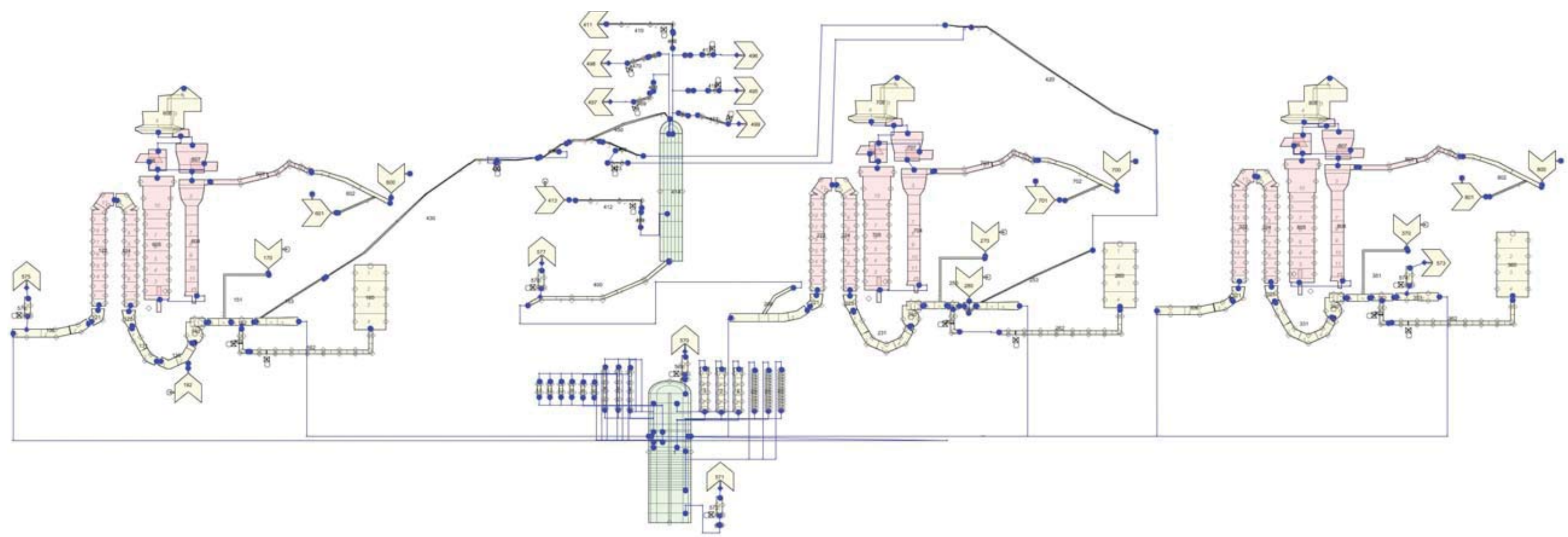


Figure 50: TRACE model of Almaraz NPP (SNAP mask with mean components)

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 [1], [2], [3], [4], [5], [6] and [7].

5.2 Comparison between Almaraz NPP and LSTF

Several considerations had to be made in order to undertake the transposition to Almaraz NPP of test 6.1 conditions, due to differences between the NPP and the facility. Tables 9 and 10 show main differences between LSTF, Tsuruga NPP and Almaraz NPP.

As can be seen in Figure 51, 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 modelling: 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	36.60
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 9: Comparison among LSTF, Almaraz NPP and Tsuruga NPP. Volumes and areas.

	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 10: Comparison between LSTF, Almaraz NPP and Tsuruga NPP. TH parameters and lengths

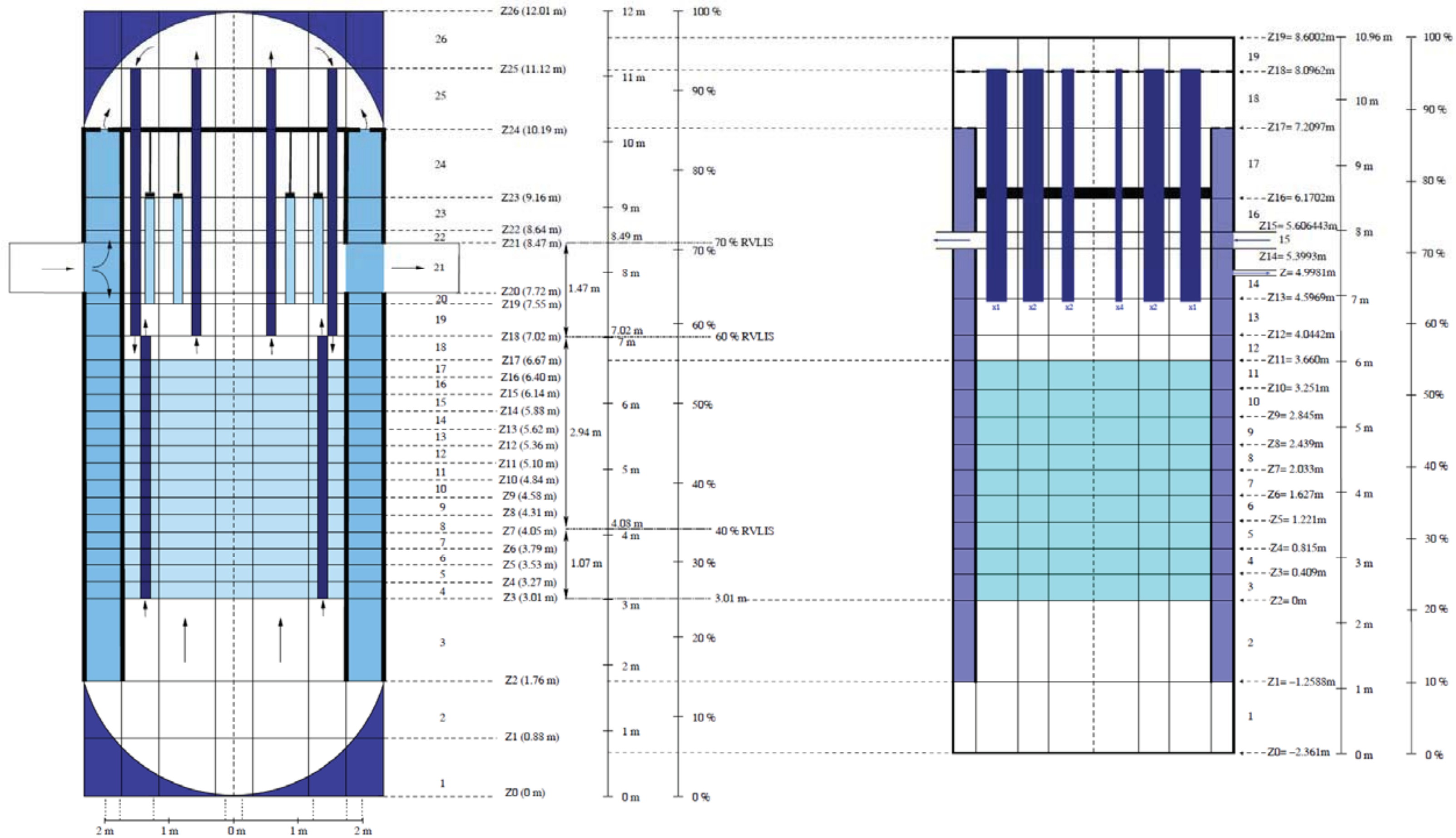


Figure 51: Comparison between Almaraz NPP and LSTF VESSEL components

5.3 Comparison between Almaraz NPP model and LSTF experimental results

Several modifications in both sets of parameter models have attempted in order to fit results obtained with Almaraz NPP model and LSTF. Section 5.4 will show further details of these results.

In a first attempt, the upper head break area for LSTF was scaled-up for Almaraz model, taking into account the volume scaling between vessel of LSTF and Tsuruga-2 NPP (1/48 volume ratio, see Table 10). Results show that the discharged coolant mass was greater than the experimental one, as shown in Figure 52. Due to this difference the pressure decreases quicker in Almaraz NPP model than in LSTF, see Figure 53 for more details.

In a second attempt, the break area was scaled in Almaraz NPP model taking into account that the volume ratio between Almaraz NPP and LSTF is 1/34.52 instead of 1/48 (see Table 10). Transient and TH parameters result to be better adjusted than simulations with volume factor 1/48 (scale factor between LSTF and Tsuruga-2), because scale factor of 1/34.52 is more realistic for Almaraz NPP. Due to that, this volume factor was considered acceptable for ongoing analisys. This volume factor was considered for subsequent analyses. Results obtained for the integrated break flow is slightly lower for Almaraz NPP, as depicted in Figure 55. On the other hand, Figure 56 shows that the depressurization of the primary does not occur at the same time, but with some delay, probably because less flow is discharged through break. Peak cladding temperature rise in Almaraz NPP is lower and later than in LSTF (see Figure 57).

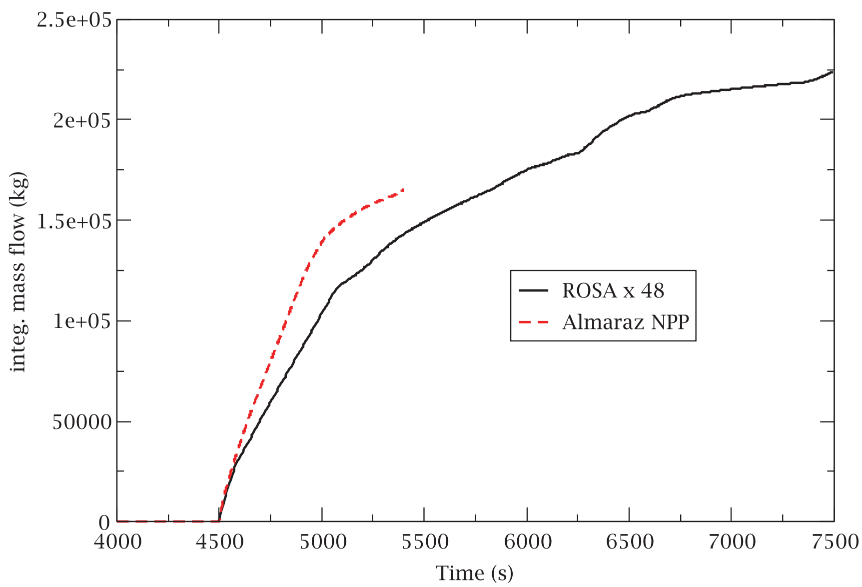


Figure 52: Integrated break flow. Almaraz NPP vs 1/48 scaled LSTF

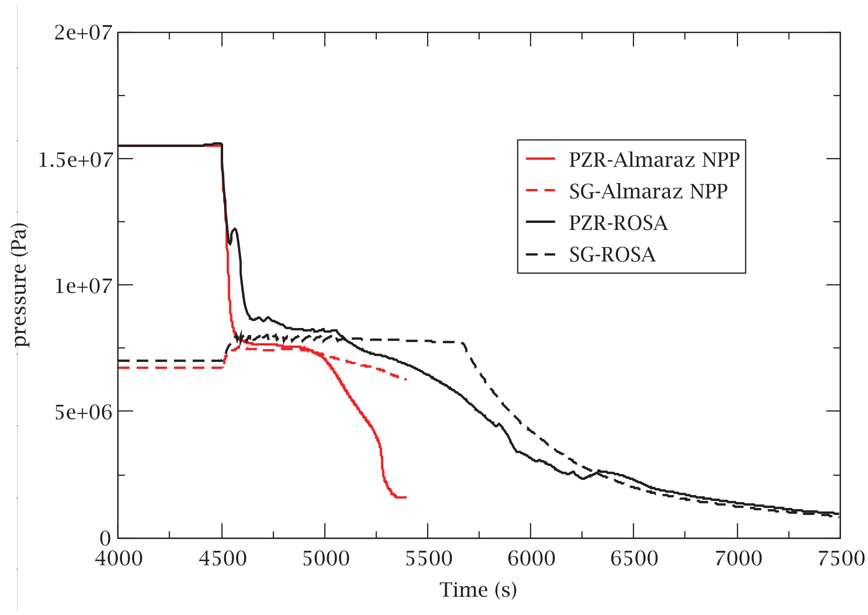


Figure 53: Primary and secondary pressure. Almaraz NPP vs 1/48 scaled LSTF

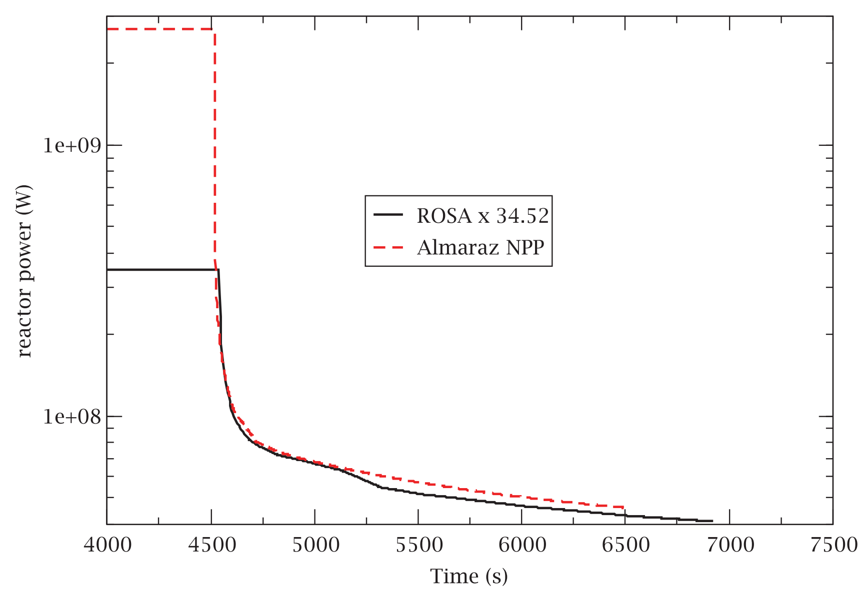


Figure 54: Core power comparison. Almaraz NPP vs 1/34.52 scaled LSTF

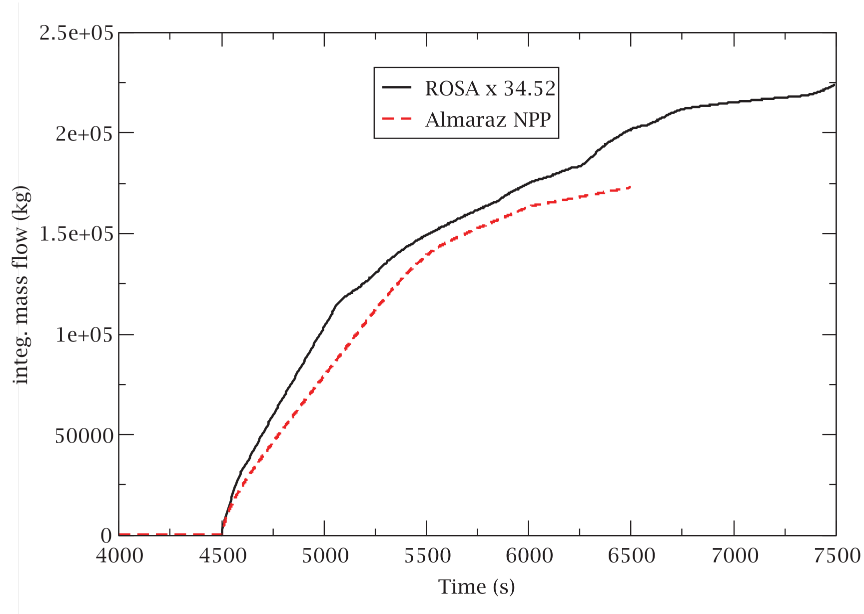


Figure 55: Integrated break flow. Almaraz NPP vs 1/34.52 scaled LSTF

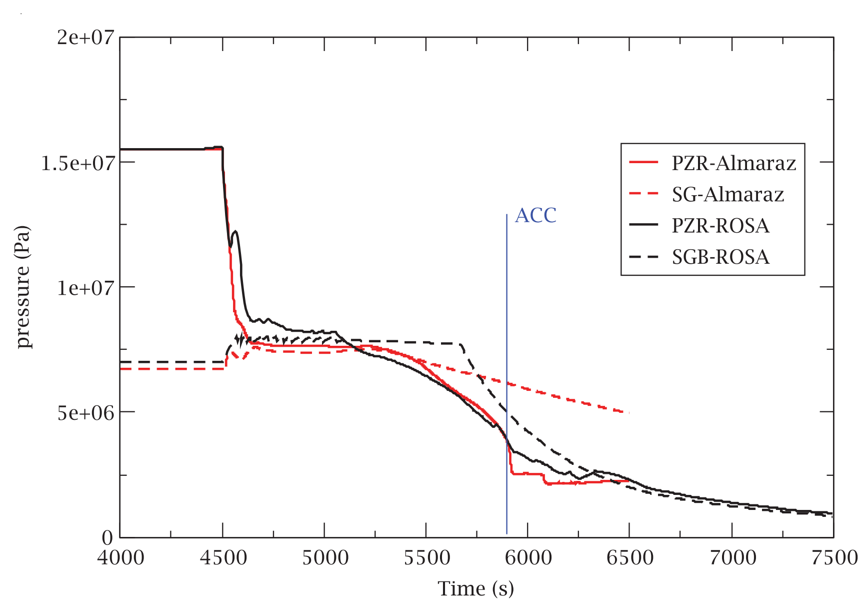


Figure 56: Primary and secondary pressure. Almaraz NPP vs 1/34.52 scaled LSTF

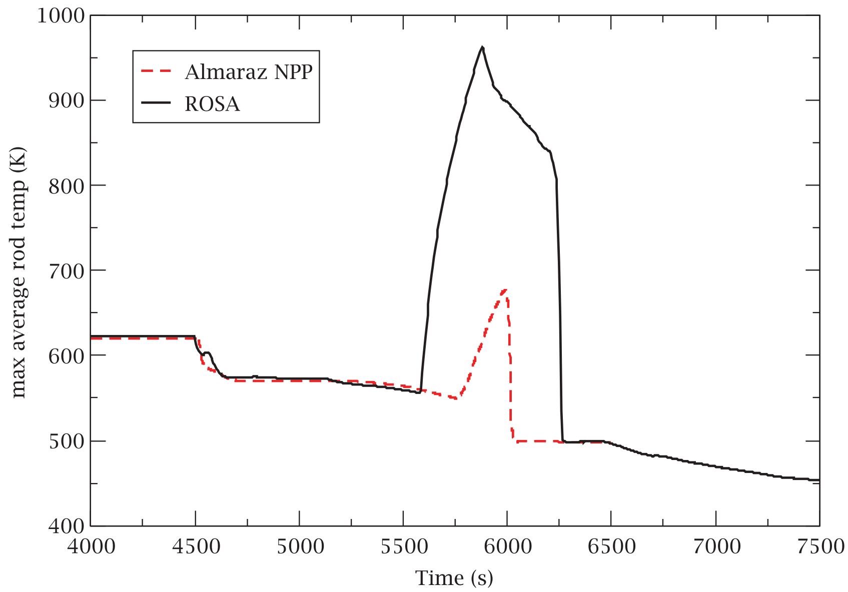


Figure 57: Peak cladding temperature. Almaraz NPP vs 1/34.52 scaled LSTF

5.4 Sensitivity analysis in Almaraz NPP model

A broad sensitivity analysis has been also preformed in order to understand the differences betrween LSTF and NPP transients. A broad spectrum of parametres have been explored in order to find the most limiting cases.

1. Discharge coefficient (sensitivity examined after scaling LSTF model).
2. Break area.
3. Break localization in different radial sectors of the upper head model.
4. Friction factors in ACCs exit.
5. Initial accumulators pressure.
6. Accumulators flow (scaled from test ROSA 6.1).
7. Upper downcomer area.
8. Friction factors in guide tubes (FRIC and RFRIC).

Only results corresponding to relevant parameters are shown in the sequel.

As shown in Figure 55, flow discharged through break is slightly lower than in the case of test 6.1. Discharge coefficients (**chm12** and **chm22** in TRACE input, for subcooled and two-phase flow, respectively; ranging from 0.8 to 1.3) were adjusted to fit break discharge flow; results are depicted in Figure 58. Discharge coefficients **chm12** = 1.1 and **chm22** = 1.2, allow to improve the starting time for rising of peak cladding temperature (see Figure 59), although are not able to capture experimental maximum value in PCT.

In a second stage, a sensitivity analisys of break area was performed, varying break area between -10% and +75% of original break area[1]. Results obtained from simulations show that there is a sligth dependence on break area of time when cladding temperature starts to rise, but there is almost no dependence of maximum temperature reached by cladding (see Figure 60, where integrated mass flow through break is depicted; and Figure 61, which shows peak cladding temperature).

In a third stage, a modification in the model, varying break position in upper head was carried out. UH break was varied from the center to the perimeter of the vessel, trying

[1] although several hypothetical break areas have been examined, it should be noted that the maximum diameter of UH break in Almaraz NPP is only 6.99cm, according to size of control rod drive mechanism in plant.

four different radial positions. Results of that sensitivity analisys are depicted in Figure 62, where PCT is depicted for each radial position. Although there were not expected significant differences between position, it can be noted that when simulation is carried out with UH break near the perimeter of the vessel PCT maximum is much lesser than in other cases. Figures 63 and 64 show that accumulators behaviour can explain this unexpected difference. In all cases it can be noted very little dependence on position of integrated mass flow through break.

In a fourth stage, different friction factors have been checked for accumulators outlet. As shown in Figure 65, integrated discharged flow from accumulators is greatly modified through variation of friction factor, being the model of accumulators outlet without friction the most similar to experimental data of LSTF. The results show that peak cladding temperature has very little dependence on accumulators outlet friction factor (see Figure 66).

Afterwards, the transient was simulated with two different downcomer areas: one with an area 1/5 of original area and the other without upper downcomer area. Results of PCT are depicted in Figure 67, where it can be noted that an area 1/5 times the original one has very little differences in PCT. Only the case where upper downcomer is closed has significant differences: in that case rise up time of cladding temperature is similar to experimental one, as well as PCT maximum, which increases to almost 90% of PCT maximum in test 6.1, at the same time. This modification causes another PCT local maximum, because of accumulators discharge interruption in transient.

Figure 68 shows depressurization dependence on friction factors of vessel guide tubes. As it can be noted, accumulators discharge is significantly affected by friction factors in guide tubes, which causes meaningful differences in depressurization of RCS.

In summary, results of peak cladding temperature sensitivity cases show:

- **Low sensitivity** to break location, friction factors at ACCs exit and steady state upper head mass flows.
- **Medium sensitivity** to discharge coefficients and upper downcomer area.
- **High sensitivity** with respect to break area size, RCP trip delay and number of ACCs available.

For sensitivity analysis on other parameters, interested reader is referred to other works of authors, e.g. Accident Management Actions in an Upper Head SBLOCA with HPSI Failed [8], where a broad analysis of Upper Head SBLOCA with HPSI failed is performed taking into account different accident management actions and conditions in order to check their suitability.

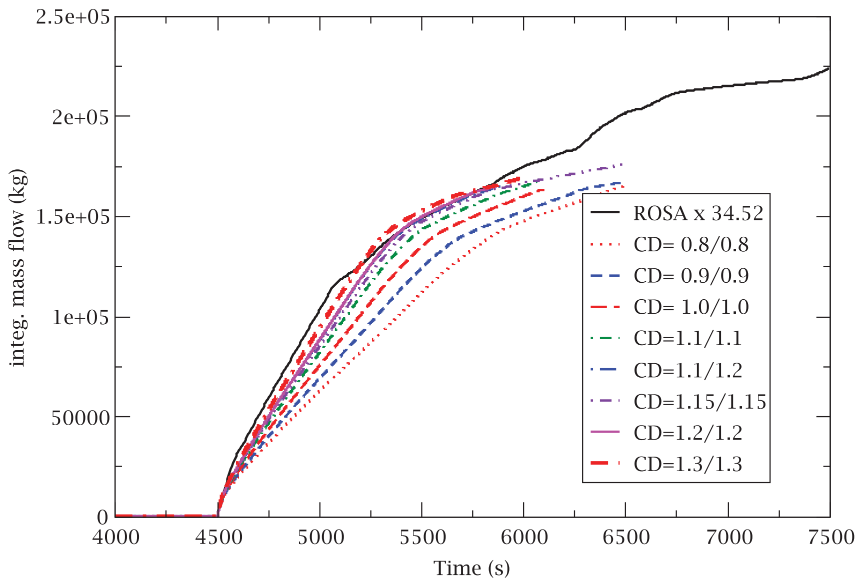


Figure 58: Integrated break flow. Almaraz NPP vs 1/34.52 scaled LSTF with different discharge coefficients

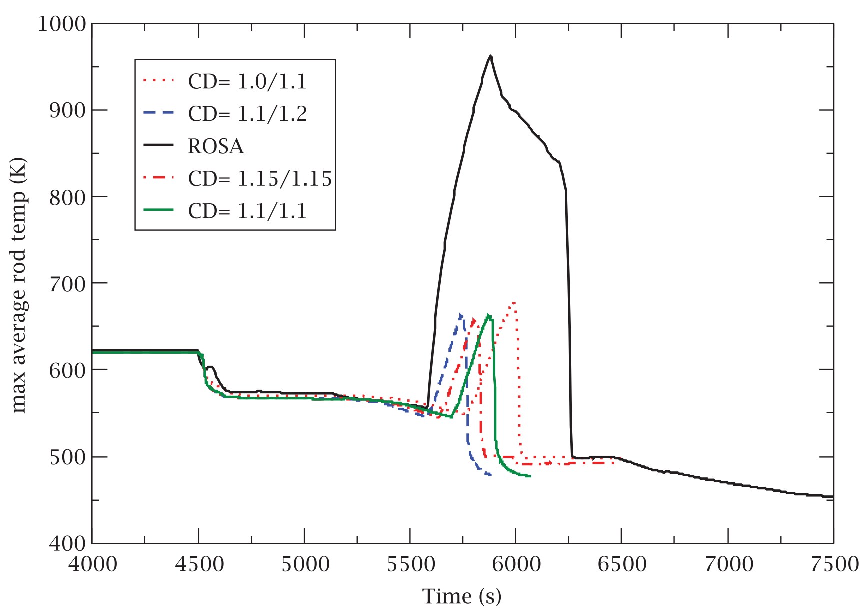


Figure 59: Peak Cladding Temperature. Almaraz NPP vs 1/34.52 scaled LSTF with different discharge coefficients

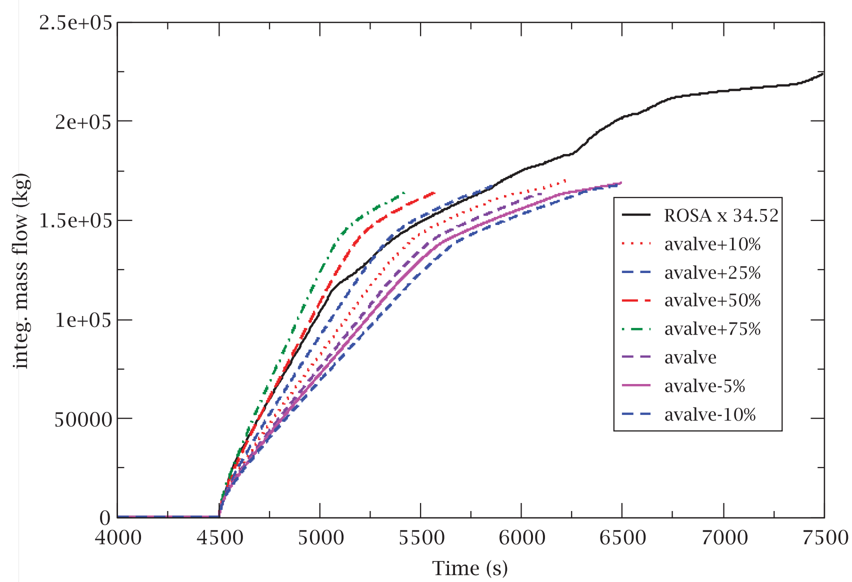


Figure 60: Integrated break flow. Almaraz NPP vs 1/34.52 scaled LSTF with different break areas

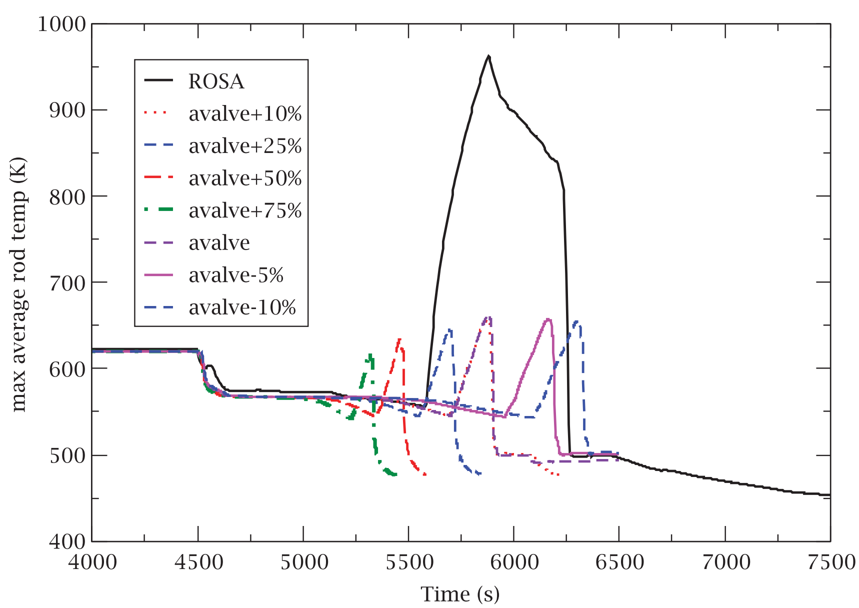


Figure 61: PCT sensitivity to break area

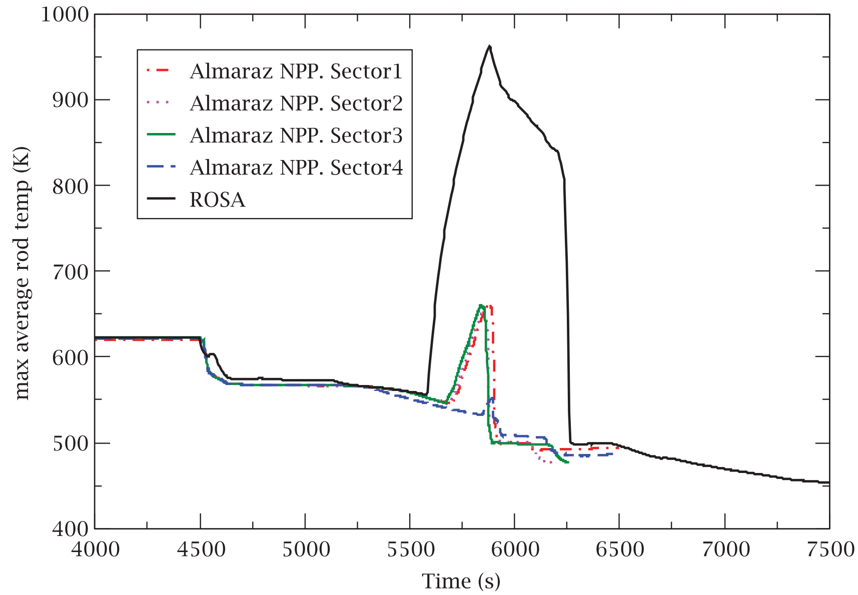


Figure 62: PCT sensitivity to break location in different radial sectors

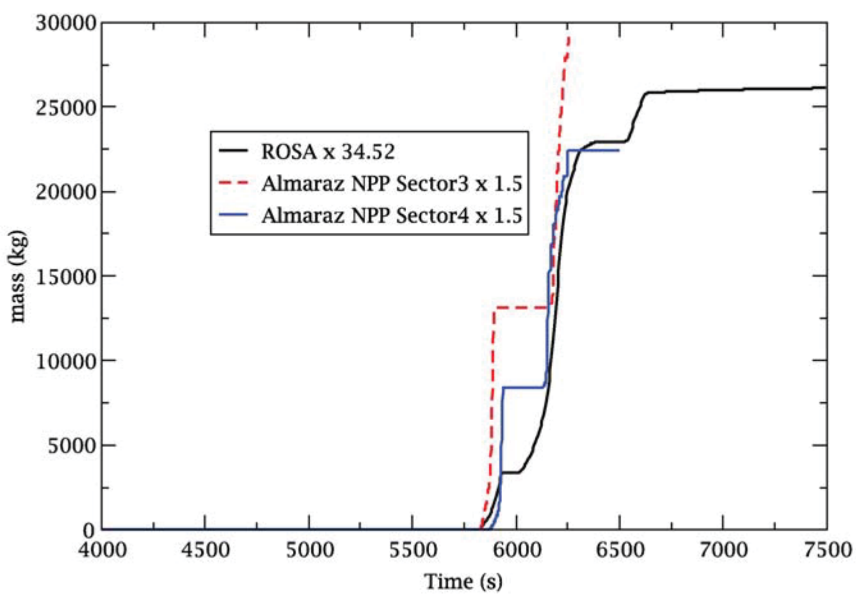


Figure 63: Integrated discharged flow from accumulators. Almaraz NPP vs 1/34.52 scaled LSTF. Cases with break in radial sectors 3 and 4

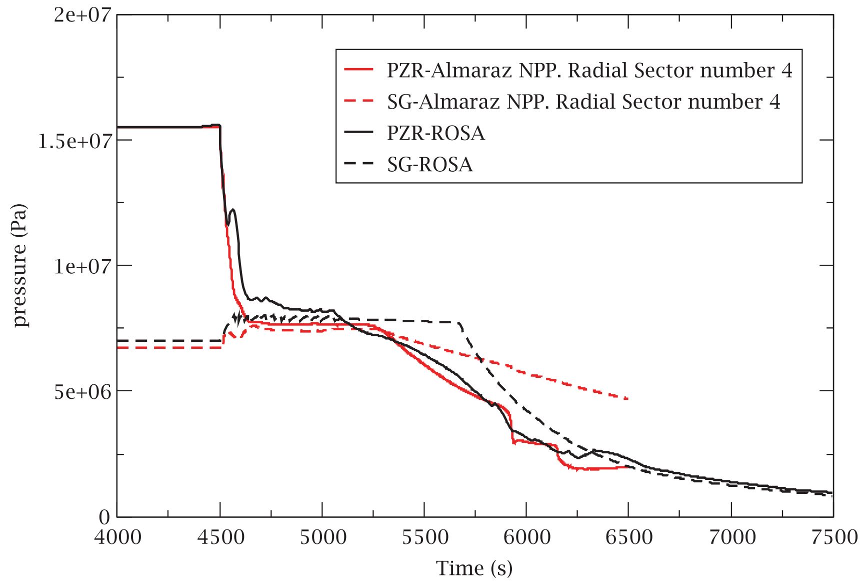


Figure 64: Primary and secondary pressures. Almaraz NPP vs 1/34.52 scaled LSTF with upper break in radial sector number 4

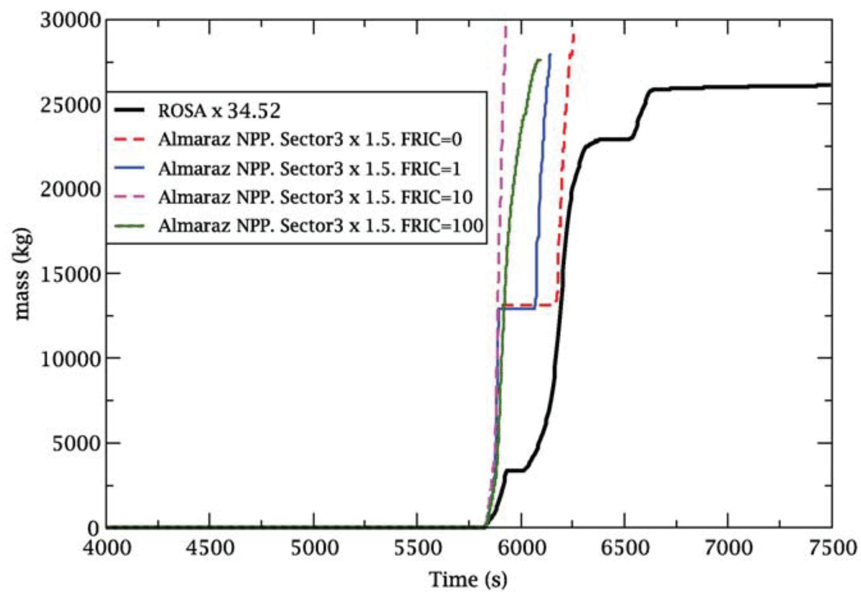


Figure 65: Integrated discharged flow from accumulators. Almaraz NPP vs 1/34.52 scaled LSTF. Cases with different friction factors at ACC outlet

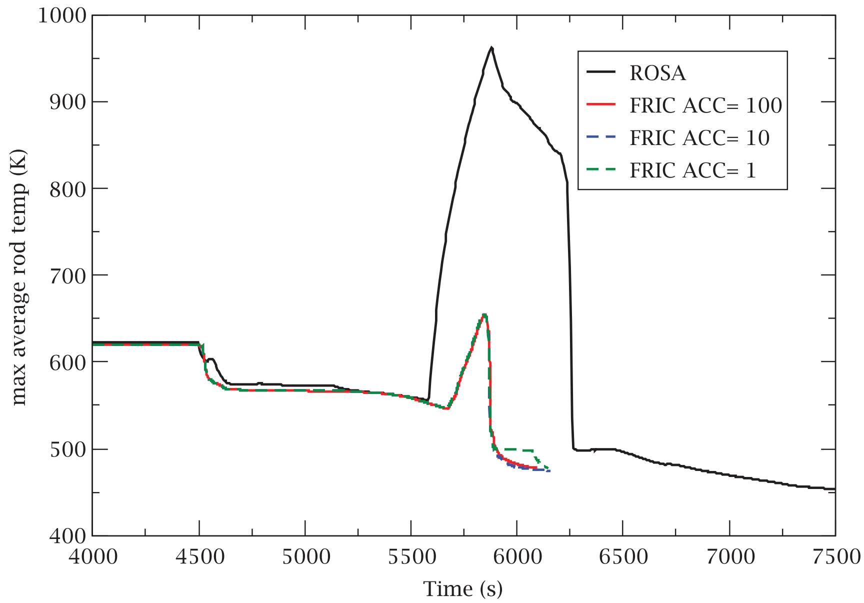


Figure 66: PCT sensitivity to friction factors at ACC outlet

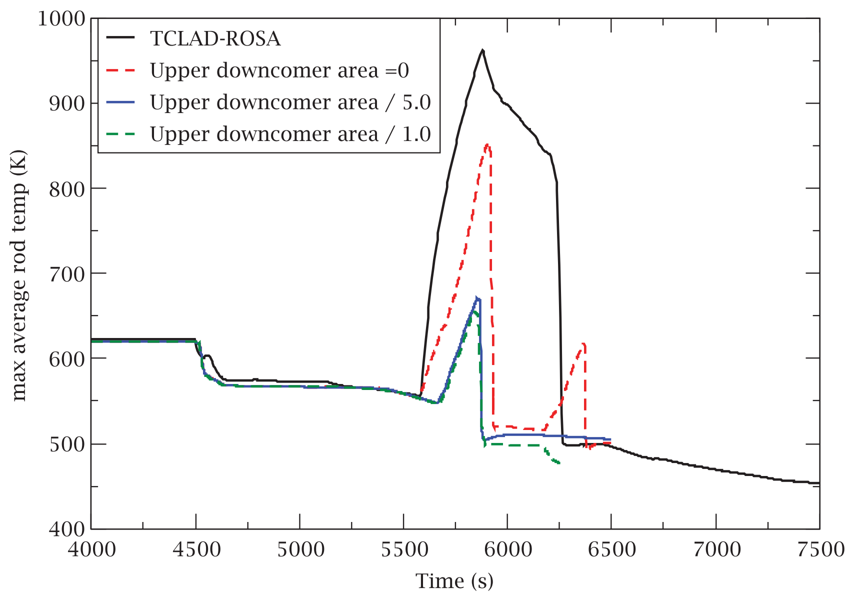


Figure 67: PCT sensitivity to friction Upper Downcomer Area

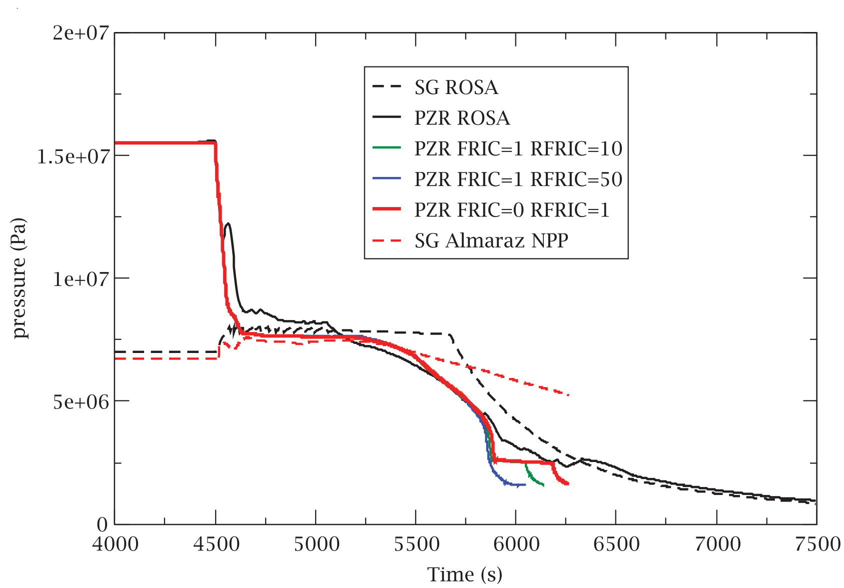


Figure 68: Primary and secondary pressures. Sensitivity to friction factors at guide tubes

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 figures, CPU effort can be divided into two main stages: Figure 69 shows that there is an increase in computing time at 2600 seconds of simulation, after coolant pumps are tripped (2276 s) and before break flow becomes single-phase vapor (2700 s). On the other hand, simulation with Almaraz model (Figure 70) shows an increase of computing time at the same time the break occurs (4650 s). CPU time performance for both phases and in both cases is summarized in Table 11.

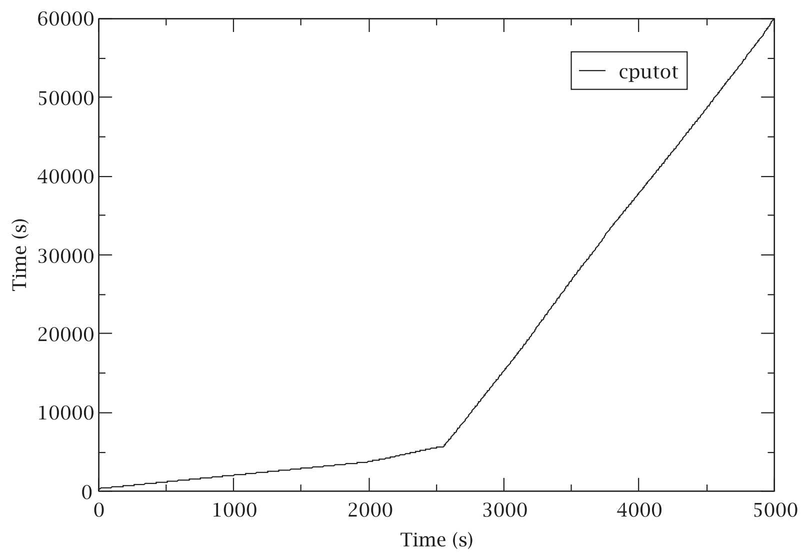


Figure 69: CPU time performance for ROSA/LSTF test 6.1 simulation

Computing Stage	**LSTF simulation performance (%)**	**Almaraz NPP simulation performance (%)**
Stage 1	45.4% (0-2600 s)	40.0% (0-4650 s)
Stage 2	4.4% (2600-5000 s)	4.0% (4650-80000 s)

Table 11: CPU performance for LSTF and Almaraz NPP simulations of test 6.1

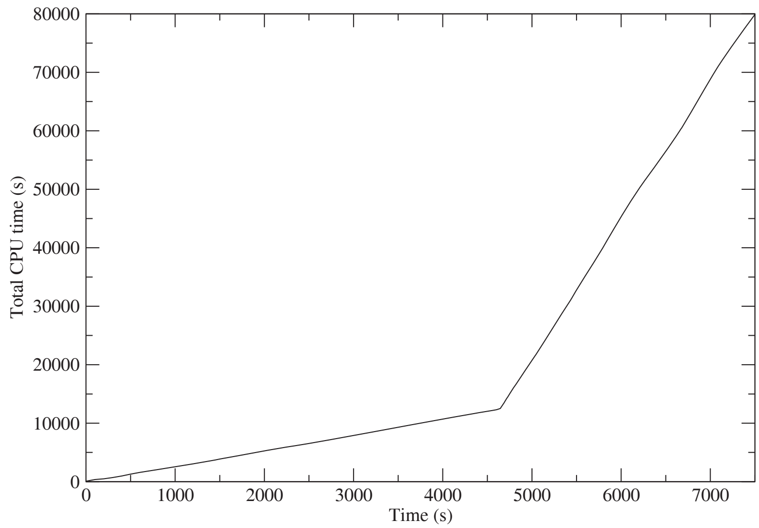


Figure 70: CPU time performance for Almaraz NPP model simulation of test 6.1

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 Test 6.1, phenomenology of upper head SBLOCA with HPSI failed was analyzed, 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 analisys of OECD/NEA ROSA Test 6.1 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:

- Results of TRACE post-test for OECD/NEA ROSA Test 6.1 have been well reproduced only through sucessive improvements and modifications in the LSTF model. Main changes in order to obtain an adequate post-test simulation were RFRIC factors and the activation of reflood model, as well as a renodalization of the vessel, cold and hot legs and steam generators.

- Not all improvements added to the LSTF model have led to good results, contrary to expectations. For example, the addition of 9 U-tubes in the SG of LSTF model yields worse results than the simplest model, with only 1 U-tube.

- During OECD/NEA ROSA Test 6.1, one of the pre-established experimental conditions was the coolant pump trip at the same time that break occurs. Sensitivity analisys to RCP trip delay shows that maximum cladding temperature increases as RCP trip delay increases, but only until approximately 1000 seconds after the break. Beyond that point, PCT decreases slightly, remaining almost constant independently of RCP trip delay.

- Simulation of OECD/NEA ROSA Test 6.1 in Almaraz NPP TRACE model requires taking into account the scale factor between LSTF and Almaraz NPP. Considering this scale factor, transient in both models seems very similar, with some deviations. Depressurization is quite similar in both models until ACCs discharge, whose pressure is higher in Almaraz NPP, and therefore discharge before. Cladding temperature begins to rise up earlier in experimental test. Also, PCT is significantly lower in Almaraz NPP simulations, due to ACCs discharge, which yields to core reflood.

- Simulation of Test 6.1 in Almaraz NPP model shows diverse grades of sensitivity to several parameter modifications. Break location, steady state upper head mass flow and friction factors at ACCs exit lead to little change in results. Model exhibits medium sensitivity to discharge coefficients and upper downcomer area, and shows high sensitivity to break area size, RCP trip delay and quantity of ACCs available.

As a general conclusion, it should be noted that TRACE code is adequate for simulating upper head SBLOCA sequences.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

An analysis of LSTF Upper Head Break experiment (OECD/NEA ROSA test 6.1) has been performed with TRACE code. This test, included within the OECD/NEA ROSA project, attempts to analyze the phenomenology and different accident management actions after the occurrence of a Upper Head break with failure of High Pressure Safety Injection (HPSI).
The comparison between the experimental data and the results obtained with TRACE code shows that, in general, the main phenomena are well reproduced. Additionally, a broad analysis of Upper Head Small Break Loss of Coolant Accident (SBLOCA) 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 and CAMP projects.

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