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International Agreement Report

Post-Test Calculation of the ROSA/LSTF Test 3-2 Using RELAP5/Mod3.3

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ABSTRACT

The Thermalhydraulic Studies Group of Technical University of Catalonia (UPC) holds a large background in nuclear safety studies in the field of Nuclear Power Plant (NPP) code simulators. This report analyzes with RELAP5mod3.3 the LSTF Test 3-2, which simulates a loss-of-feedwater transient (LOFW) without scram and with a total failure of the high pressure injection system (HPIS). The simulation of several phenomena related with the high power natural circulation has been object of study as well as the impact of the environment heat losses in the primary coolant discharge across the pressurizer power-operated relief valve (PORV).

FOREWORD

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

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

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

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

Many experimental facilities have contributed to the today's availability of a large thermal-hydraulic database (both separated and integral effect tests). However there is continued need for additional experimental work and code development and verification, in areas where no emphasis have been made along the past. On the basis of the SESAR/FAP[2] reports "*Nuclear Safety Research in OECD Countries:Major Facilities and Programmes at Risk" (SESAR/FAP, 2001)* and its 2007 updated version "*Support Facilities for Existing and Advanced Reactors (SFEAR) NEA/CSNI/R(2007)6"*, CSNI is promoting since 2001 several collaborative international actions in the area of experimental TH research. These reports presented some findings and recommendations to the CSNI, to sustain an adequate level of research, identifying a number of experimental facilities and programmes of potential interest for present or future international collaboration within the safety community during the coming decade.

CSN, as Spanish representative in CSNI, is involved in some of these research activities, helping in this international support of facilities and in the establishment of a large network of international collaborations. In 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.

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1 It's worth to note the emphasis made in the application to actual NPP incidents.
2 SESAR/FAP is the *Senior Group of Experts on Nuclear Safety Research Facilities and Programmes* of NEA Committee on the Safety of Nuclear Installations (CSNI).

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

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

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

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

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

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

Both objectives are carried out by simulating experiments and plant application with the last available versions of NRC TH codes (RELAP5 and TRACE). A CAMP in-kind contribution is aimed as end result of both types of studies. Development of these activities, technically and financially supported by CSN, is being carried out by 5 different national research groups (Technical Universities of Madrid, Valencia and Cataluña). On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermal hydraulics analysis of accidents of the Spanish nuclear power plants.

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

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EXECUTIVE SUMMARY

Experimental research activities are being performed in Japan by the OECD/NEA ROSA project with the aim of obtaining thermal-hydraulic data for the validation of computer codes and models for system integral analyses coupled with detailed analyses of local phenomena. These experiments are carried out at the LSTF test facility.

This report analyses Test 3-2, which simulates a loss-of-feedwater (LOFW) transient without scram under the assumption of a total failure of the high pressure injection system (HPIS) and an actuation of the auxiliary feedwater (AFW). Liquid oscillation in the U-tubes during the natural circulation and coolant carryover from the hot leg into the pressurizer are objective of study in this test.

UPC-INTE LSTF model created for the simulation of LSTF/ROSA Test 3-1 [1] has been used and improved for this test. Calculations have been performed using RELAP5mod3.3 code.

In order to improve the base case, a sensitivity analysis was performed to study the impact of the environment heat losses in the transient. Discrepancies were corrected adjusting the mass discharged through the power-operated relief valve (PORV) and the collapsed liquid level in the pressurizer.

Many other aspects related to the nodalization were adjusted and verified in order to improve results.

The general agreement between model predictions and experimental data qualifies the UPC-INTE LSTF model as an appropriate tool to simulate LSTF “High Power Natural Circulation” Tests. This level of qualification will certainly improve after a more accurate adjustment of the environment heat transfer for each component of the facility.

ACKNOWLEDGMENTS

This paper contains findings that were produced within the OECD-NEA ROSA Project. The authors are grateful to the Management Board of the ROSA Project for their consent to this publication.

ABBREVIATIONS

AFW	auxiliary feed water
ANAV	Asociación Nuclear Ascó-Vandellòs
ATWS	anticipated transient without scram
CCFL	counter current flow limitation
CNV II	Vandellos II NPP
ECCS	emergency core cooling system
HPIS	high pressure injection system
INTE	Institut de Tècniques Energètiques
LOFW	loss of feed water
LPIS	low pressure injection system
LSTF	large scale test facility
MFW	main feed water
MSIV	main steam isolation valve
NEA	Nuclear Energy Agency
NPP	nuclear power plant
OECD	Organization for Economic Cooperation and Development
PORV	power operated relief valve
PZR	pressurizer
PWR	pressurized water reactor
RCP	reactor coolant pumps
RELAP	reactor excursion and leak analysis program
ROSA	rig of safety assessment
SBLOCA	small break loss of coolant accident
SG	steam generator
UPC	Universitat Politècnica de Catalunya (Technical University of Catalonia)

1. INTRODUCTION

Several safety activities have been performed during the last decades under the auspices of the OECD to develop and improve computer codes. They include several experiments at integral test facilities like the ROSA/LSTF Test 3-2, which forms part of the OECD/NEA ROSA project.

1.1 High-Power Natural Circulation Events

High-power events are transients with failure of scram in which core power decrease is due to negative reactivity feedback. Depending on the transient characteristics, this situation can lead to a relatively high core power during a long time.

Natural circulation occurs in transients with gradual loss of mass inventory (SBLOCA or LOFW –losses across the pressurizer relief valve due to overpressure on the primary system-). While there is high core power and water in the loops, vapor and liquid with high velocity exit from the vessel inducing supercritical flow during natural circulation. This phenomenon can modify significantly the coolant distribution affecting the core cooling. Particularly, for a LOFW-ATWS, the high pressure drop in the pressurizer PORV pulls in the supercritical flow in the hot leg at the surge line inlet nozzle, causing a counter-current flow limitation (CCFL) in the bottom of the pressurizer. It avoids that coolant returns to vessel for cooling the core –see figure one-.

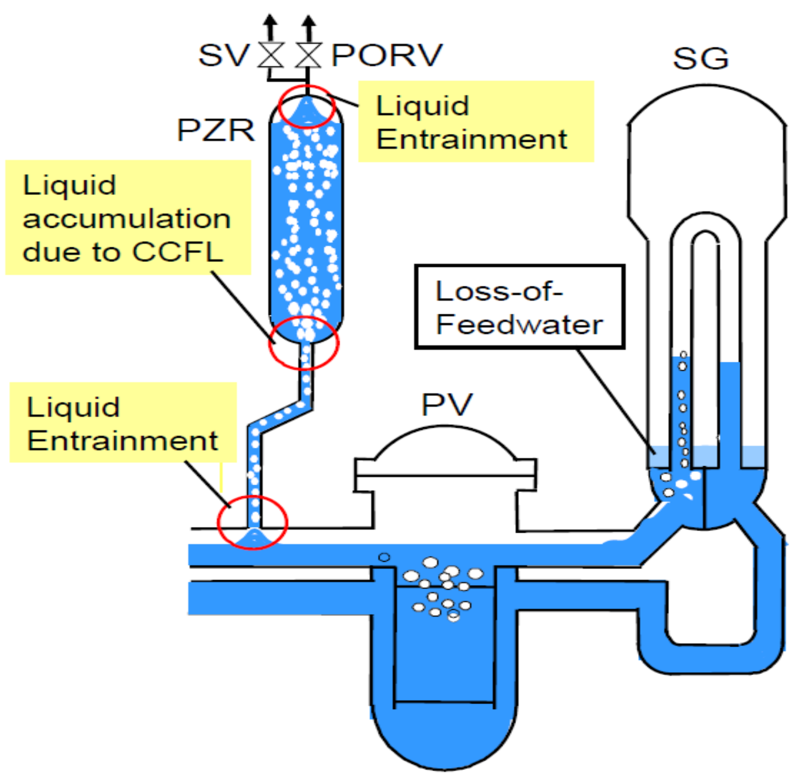

Figure 1 Thermal-hydraulic phenomena during LOFW without scram (Courtesy of the OECD/NEA ROSA group)

1.2 The OECD/NEA ROSA Project

The OECD/NEA ROSA project includes several types of experiments (see figure two) on ROSA/LSTF test facility with the aim of providing a wide database for the validation of computer codes and models for system integral analyses coupled with detailed analyses of local phenomena. The main phenomena to study in these experiments are multi-dimensional mixing, stratification, parallel flows, unstable flows, convection and influences of non-condensable gas. Test 3-2 is included in the group of “High power natural circulation experiments”.

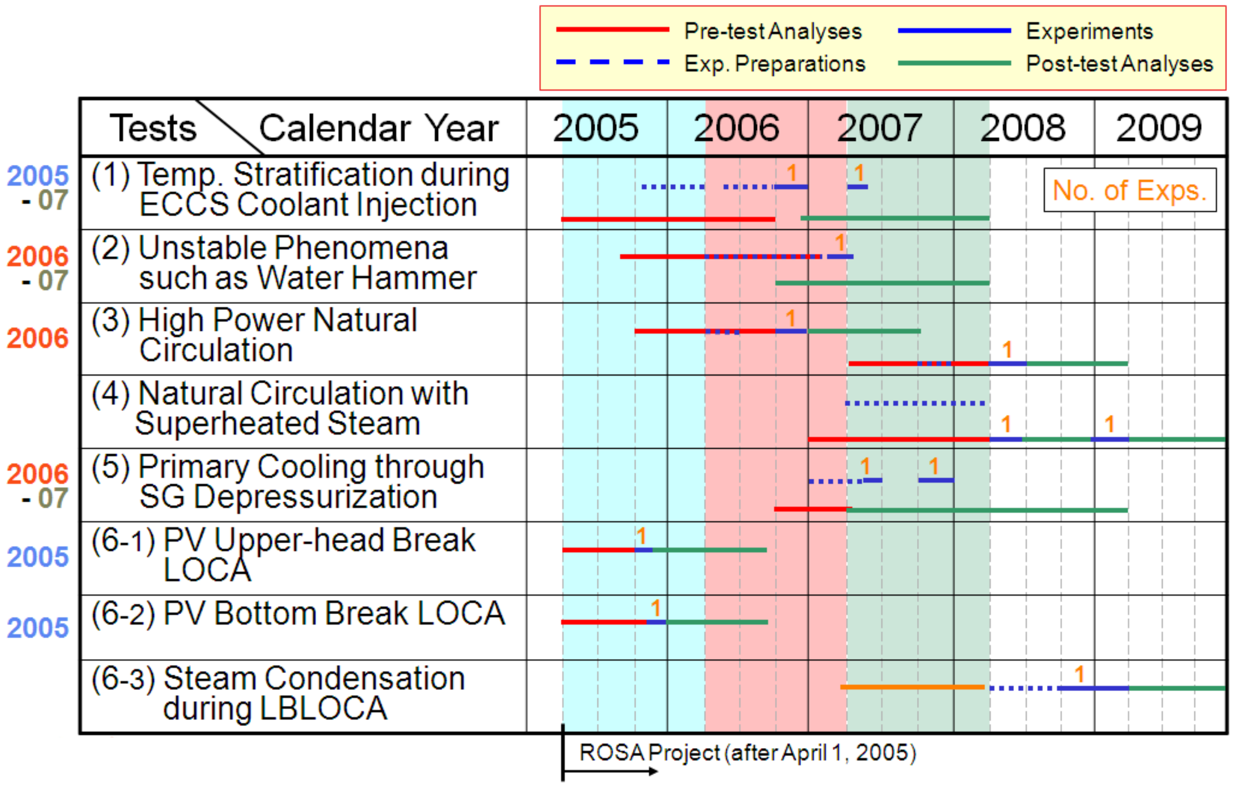

Figure 2 OECD/NEA ROSA project experiments (Courtesy of the OECD/NEA ROSA group)

2. FACILITY AND TEST DESCRIPTION

2.1 LSTF Test Facility

LSTF (see figure three) is an experimental facility designed to simulate a Westinghouse-type 4-loop 3,420 MWth PWR under accidental conditions. It is a full-height and 1/48 volumetrically-scaled two-loop system with a maximum core power of 10 MW (14 % of the scaled PWR nominal core power) and pressures scaled 1:1. Loops are sized to conserve volumetric factor (2/48) and to simulate the same flow regime transitions in the horizontal legs (respecting L/√D factor).

There is one steam generator (SG) for each loop respecting the same scaling factors. They have 141 full-size U-tubes, inlet and outlet plenum, steam separator, steam dome, steam dryer, main steam line, four downcomers and other internals.

All emergency systems are represented and have a big versatility referred to their functions and positions. Many break locations (20) are available too.

LSTF test facility has about 1,760 measurement points that allow an exhaustive analysis of the tests. There are two types of data or measurements of interest: directly measured quantities (temperature, pressure, differential pressure), and derived quantities (from the combination of two or more direct measured quantities –coolant density, mass flow rate…-).

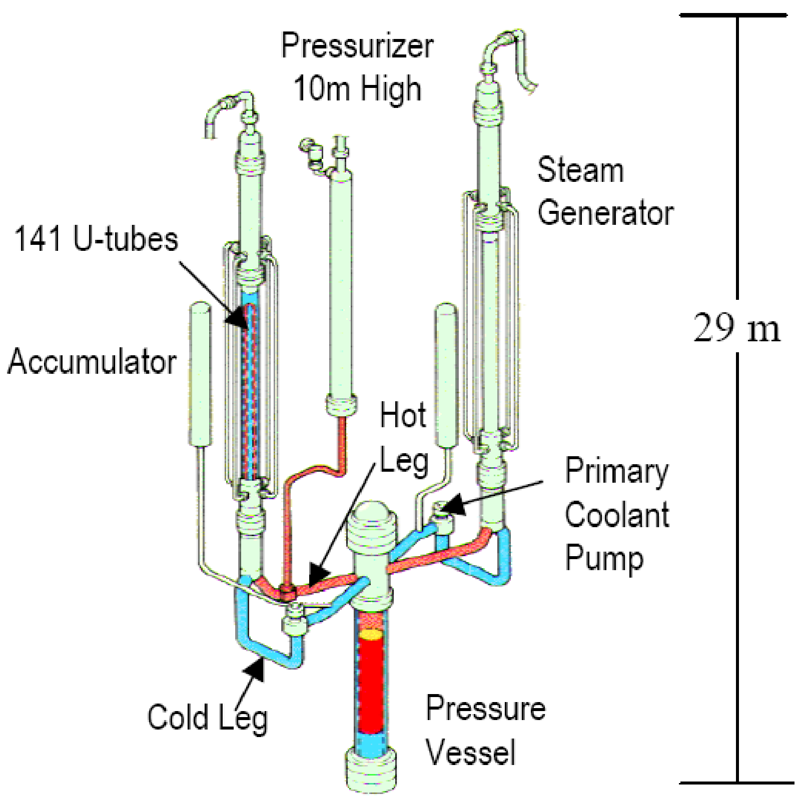

Figure 3 LSTF Test Facility (Courtesy of the OECD/NEA ROSA group)

Test 3-2 simulates a LOFW transient without scram under the assumption of a total failure of the HPIS system and an actuation of the AFW. Liquid accumulation in the U-tubes during the natural circulation and coolant carryover from the hot leg into the pressurizer are objective of study in this test.

2.2 Experimental Conditions

The hardware configuration of LSTF is described in references [2] and [3]. Some important points are the following:

- *ECCs:* HPI and LPI unavailable simulating loss of off-site power.
- *AFW:* initiated when SG collapsed liquid level is less than its 3%.
- *Core power curve:* pre-determined from a previous volumetrically scaled analysis performed with RELAP5 code using one-point-kinetics model, which reproduces the transient in a commercial PWR. (see reference [3] and [4] for more detailed information). As LSTF core power is limited to the 14% of the scaled reference plant nominal power, the portion higher than 10 MW is cut-off (see first 200 s of figure four).
- *LSTF core protection system:* Core power is modified according to the maximum fuel rod surface temperature
- *PZR heaters:* shut off with the scram signal (proportional heater) and when the PZR level becomes lower than its 20% (backup heater).

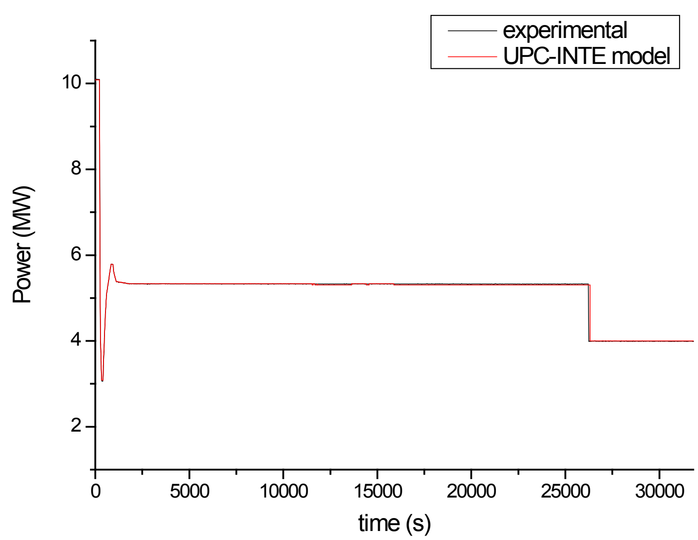

Figure 4 LSTF core power

2.3 Initial Conditions

Initial steady-state conditions were fixed according to the reference PWR conditions. Because of the LSTF initial core power (14 % of the scaled PWR nominal core power), core flow rate was set to 14 % of the scaled nominal flow rate to obtain the same PWR temperatures, and secondary pressure was raised to limit the primary-to-secondary heat transfer rate to 10 MW.

2.4 Test Phase

The transient starts at t=0 seconds with the loss of main feedwater in the secondary system. After 50 seconds the scram signal is generated causing the closure of the MSIV and the stop valve (turbine trip); pressurizer proportional heater is switched off and coastdown of the primary coolant pumps is initiated. Primary and secondary pressure raise to the set point of the PORV and steam generators (SG) relief valves initiating a continuous cycle opening at 330 seconds. Thirty seconds before approximately, pumps stop and natural circulation is started, changing from single to two phase flow about the 1,150 seconds. During this period, SG collapsed liquid level decreases below its 3 % switching on the AFW and achieving a semi-equilibrium energy balance in the test facility until the core dry out (core power and SG levels keep more or less constant until 26,000 seconds during PZR PORV and SG RV valves continuous opening).

Until 12,000 seconds, natural circulation increases gradually, time in which mass flow rate drops as a result of the whole core boiling and a large voiding in the SG U-tubes. Then, liquid levels of the U-tubes start to oscillate asymmetrically affecting the primary mass flow rate, the primary-to-secondary heat transfer and the discharge across the PORV valve. About 15,800 seconds, pressurizer becomes empty of liquid. Natural circulation keeps until 24,000 seconds approximately, time in which SG inlet liquid levels drops and reflux condensation begins.

About 25,700 seconds, significant increase occurs in the fuel rod surface temperature due to core dry out, initiating the LSTF core protection system about 26,000 seconds. It causes core quenching and the primary system depressurization. At 32,000 seconds approximately, SG are refilled, AFW is closed and pressures of both systems become constant, so transient is finished. The main events are described in table one:

Table 1 Chronology of the main events in Test 3-2

Event	Time [s]
Loss of feedwater	0
SCRAM signal: · Turbine trip and closure MSIV · PZR proportional heater off · Closure of MSIVs · Start coast-down of primary coolant pumps	50
Primary coolant pumps stop	300
Initiation of the AFW	1030
Natural circulation from single phase to two phase	1150
Significant level oscillation begin in SG U-tubes (whole core boiling)	About 12000
PZR became empty of liquid	15800
Initiation of reflux and condensation phase	23500
Core power decrease by LSTF core protection system	25700
Max. fuel rod surface temperature	26220
Termination of auxiliary feedwater	32100

3. CODE INPUT MODEL DESCRIPTION

3.1 Nodalization

The RELAP5 version used to perform the post-test calculations is RELAP5/MOD3.3. The input deck used to simulate the transient was developed for the Pos-test calculation of the LSTF/ROSA Test 3-1 (reference [1]). Some improvements have been performed with the aim of obtaining a more realistic nodalization. So, five U-tube models have been designed to reproduce the nine LSTF U-tube types, one simulating the shortest one, and the others averaging of pair wise the rest of the lengths. Steam generators have been modified in the secondary side too, increasing the number of volumes in the riser for tuning the slope of temperature, and reproducing both annulus in the top and the bottom of the downcomer. Furthermore, surgeline has been re-nodalized, keeping lengths and heights as well as pressure drops, location and orientation of its inlet nozzle in the hot leg, and reproducing the multiple orifices in the inlet of the pressurizer. Finally, the line between the pressurizer and the storage tank has been modelled taking into account suggestions referred in [5] about the discharge lines nodalization.

3.2 Preliminary Calculations

Preliminary calculations of the system energy balance showed an important disagreement in the environment heat losses between the experimental data and the simulation, particularly during the conditioning phase of the transient. A sensitivity analysis was performed to evaluate and to adjust the heat losses, showing a significant impact in the behavior of the PZR collapsed liquid level and the mass inventory discharged across the PORV valve. Results of this sensitivity analysis are developed in more detail in 4.2.1.

Table two shows the main parameters in steady state conditions for the UPC-INTE LSTF model. Values are normalized to the measured steady state conditions.

Table 2 Steady state conditions

	UPC-INTE LSTF model (loops w / wo PZR)
Core power	1.01
Hot leg temperature	1.003 /1.003
Cold leg temperature	1.001 / 1.001
Mass flow rate (x loop)	0.98 / 0.965
Downcomer-to-hot-leg bypass	1.001 / 1.001
Pressurizer pressure	1.0001
Pressurizer liquid level	0.999
Secondary-side pressure	0.998 / 0.998
Secondary-side liquid level	1.000 / 0.9998
Main feedwater temperature	1.0 / 1.0
Auxiliar feedwater temperature	1.0
Main feedwater flow rate	1.038 / 1.047
Accumulators pressure	1.0
Accumulators temperature	1.0 / 1.0
Steam flow rate	1.034 / 1.035

4. RESULTS

4.1 Test Phase

Table three shows the chronology of the main events occurred in Test 3-2, comparing the experimental values with the calculated ones.

Table 3 Comparison between experimental and simulated main events

Event	Experimental [s]	UPC-INTE LSTF model [s]
Loss of feedwater	0	0
SCRAM signal: · Turbine trip and closure MSIV · PZR proportional heater off · Closure of MSIV valves · Start coast-down of coolant pumps	50	20
Primary coolant pumps stop	300	300
Initiation of the AFW	1030	1080
Natural circulation from single phase to two phase	1150	1100
Significant level oscillation begin in SG U-tubes (whole core boiling)	About 12000	About 12500
PZR became empty of liquid	15800	17100
Initiation of reflux and condensation phase	23500	23030
Core power decrease by LSTF core protection system	25700	26090
Max. fuel rod surface temperature	26220	26320
Termination of auxiliary feedwater	32100	32100

As shown in figure five, primary and secondary pressure in the UPC-INTE LSTF model have good agreement with the experimental data until maximum fuel clad temperature is reached at 26,220 seconds (see figure six), simulating correctly primary-to-secondary cooldown and SG collapsed liquid levels (see figure seven) during the actuation of the AFW (see figure eight).

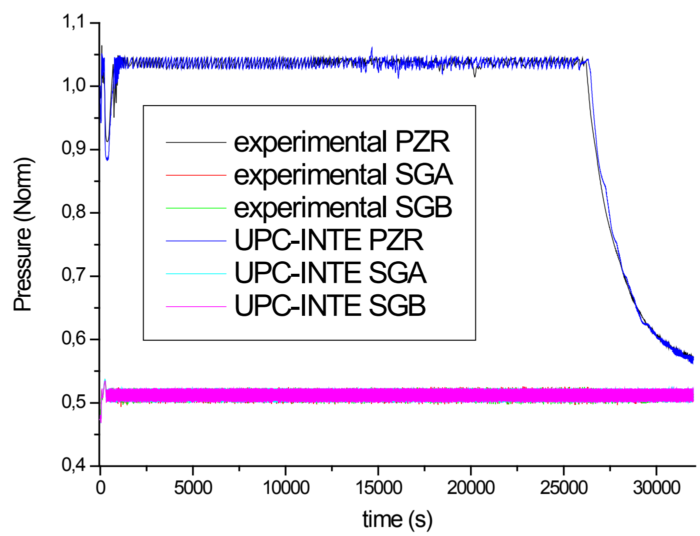

Figure 5 Primary and secondary pressure

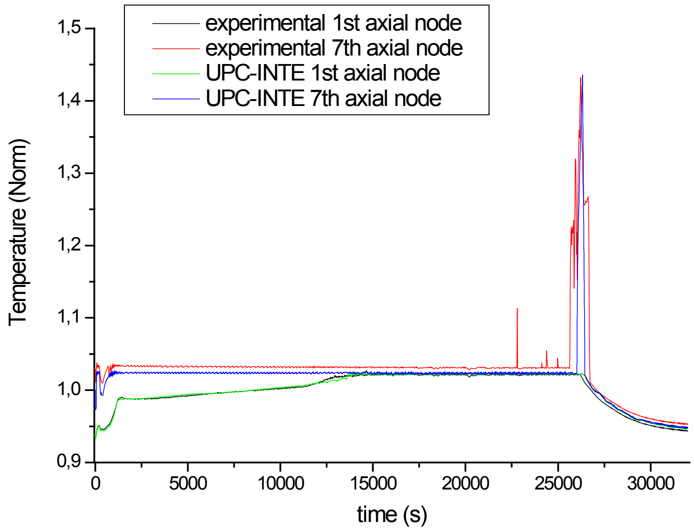

Figure 6 Maximum and minimum clad temperatures

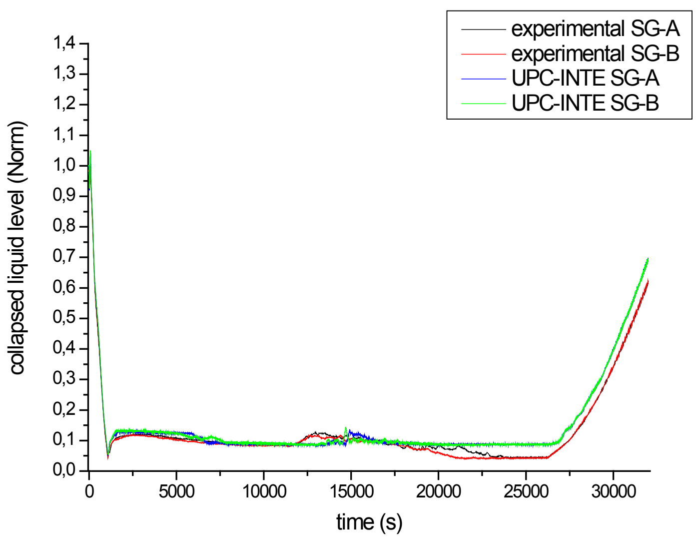

Figure 7 SG riser collapsed liquid level

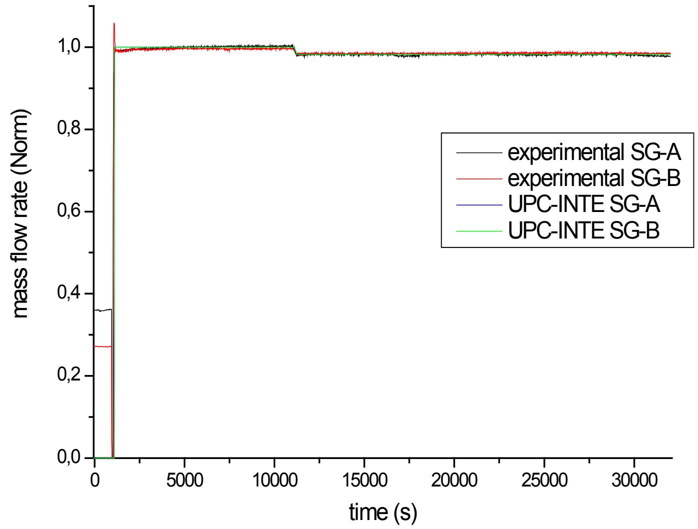

Figure 8 Auxiliary feedwater

Figure six shows UPC-INTE LSTF model reproduces quite well the whole core boiling and so, the beginning of U-tube liquid levels oscillation (see figure nine).

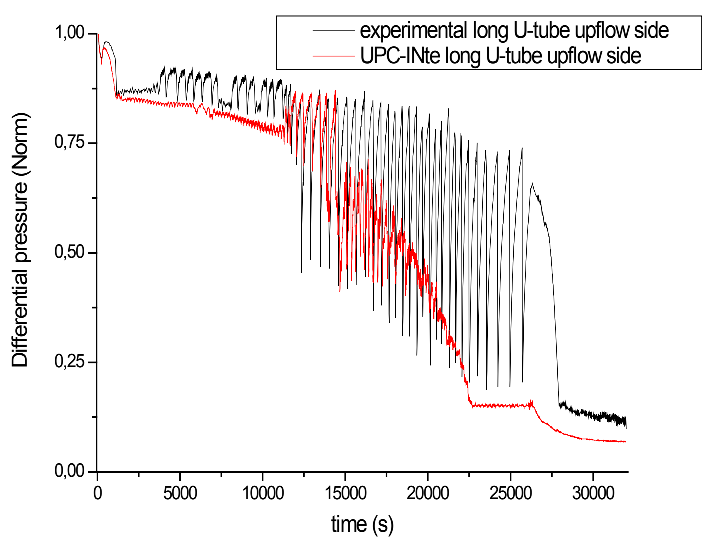

Figure 9 U-tubes upflow side differential pressure

However, although simulation reproduces primary mass flow rate drop related with the whole core boiling (see figure 10), it exists a relative delay between UPC-INTE LSTF model and experimental data that affects the mass flow rate into the pressurizer and its partial refilling (see figure 11). Its phenomenon holds up the emptying of the pressurizer, but even in this case, all these events are qualitatively well reproduced in the simulation.

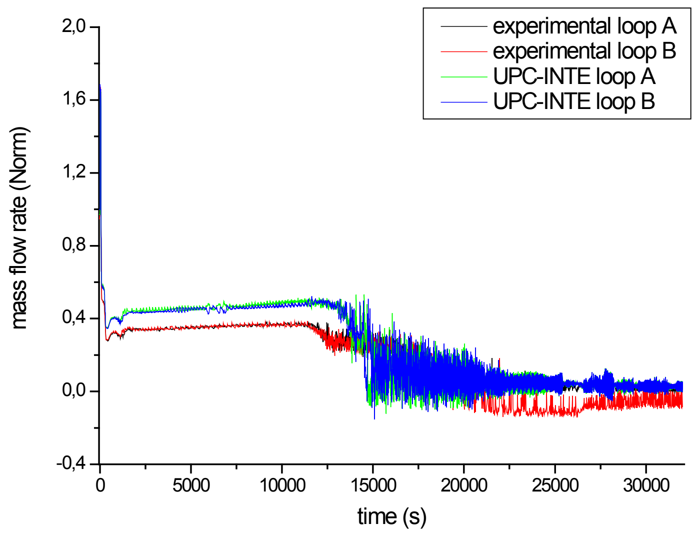

Figure 10 Primary mass flow rate

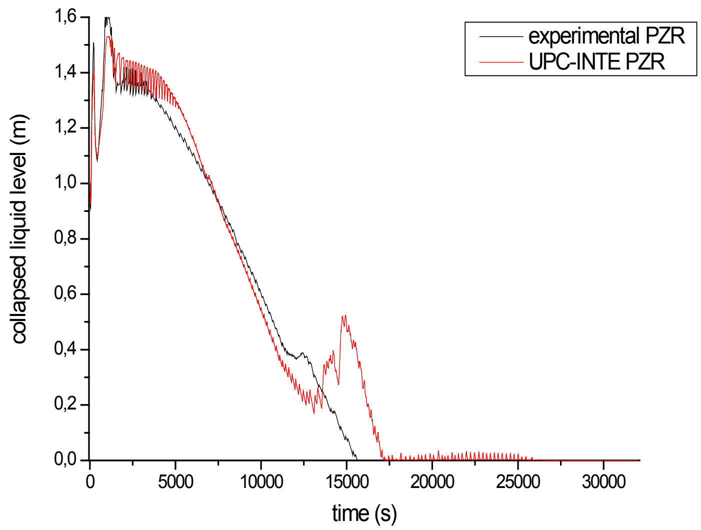

Figure 11 PZR liquid level

Finally, the UPC-INTE LSTF model shows a quite good agreement simulating the partial emptying of the SG inlet and hot leg, and the initiation of reflux condensation (see figure 12).

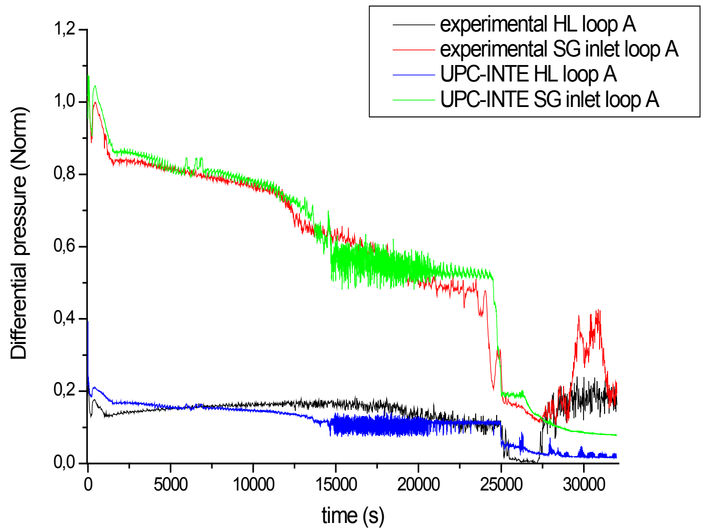

Figure 12 Differential pressures in the SG inlet and hot leg of loop A

4.2 <u>Local Phenomena</u>

4.2.1 U-tubes Liquid Level Oscillation During Natural Circulation

Significant liquid level oscillations start in the U-tubes when the whole core reaches the saturation temperature. Although UPC-INTE LSTF model reproduces this phenomenon, it is worth mentioning their amplitudes decrease faster than in the experimental data (Figure nine). Figure 13 compares U-tubes liquid temperatures with the saturation temperature in the core exit and in the SG riser. Comparison shows that in the simulation, U-tubes liquid temperatures oscillate less in this range, generating lower condensation in the steam generator. This fact justifies liquid level amplitudes in the U-tubes decrease faster in the UPC-INTE LSTF model.

Figures 14 and 15 show the frequency of the U-tubes liquid oscillations during natural circulation and reflux condensation respectively. Comparison with the experimental data and primary and secondary pressures shows UPC-INTE LSTF model reproduces oscillations of liquid in the U-tubes during the natural circulation. Coupling between PORV valve continuous opening and U-tubes liquid level oscillations becomes evident when reflux and condensation is achieved (see figure 15).

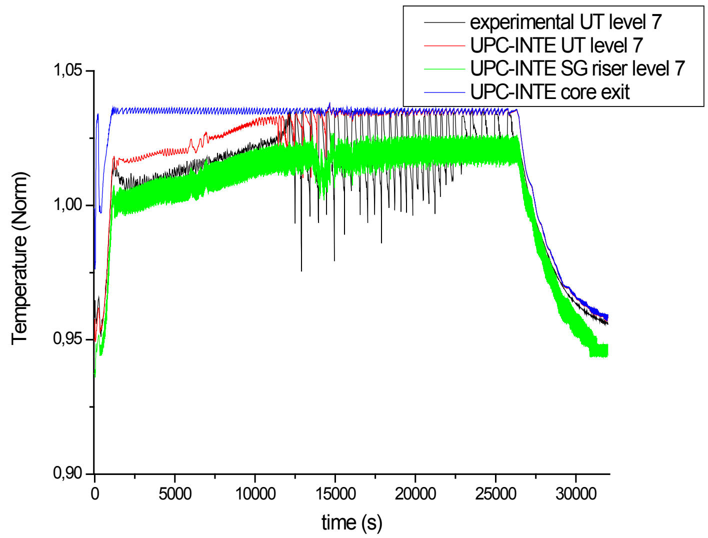

Figure 13 U-tube temperatures

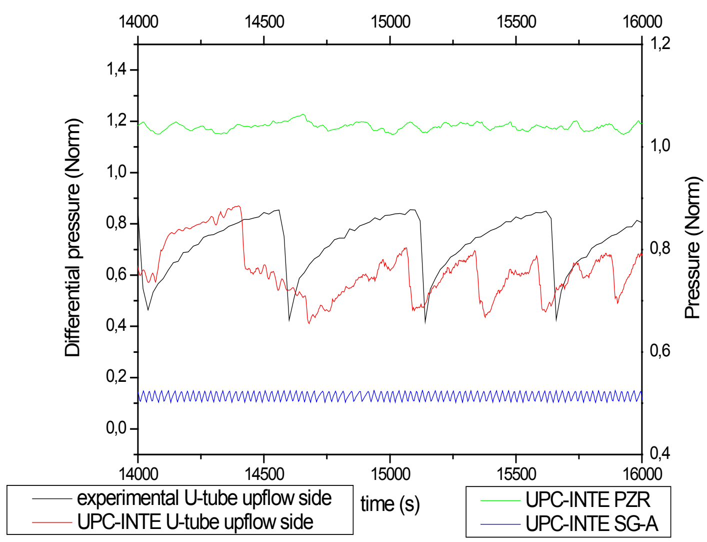

Figure 14 U-tubes differential pressure during natural circulation

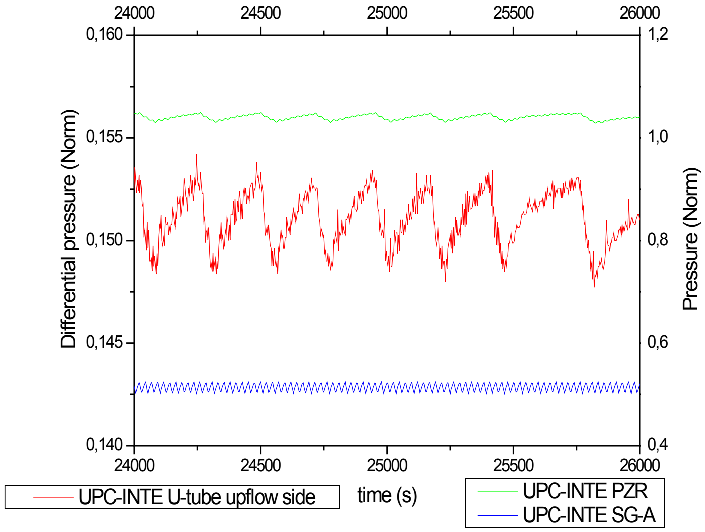

Figure 15   U-tubes differential pressure during reflux condensation

4.3 Sensitivity Analysis

4.3.1 Impact Of The Environment Heat Losses

Preliminary simulations taking into account experimental procedures and boundary conditions of LSTF/ROSA Test 3-2 showed an unexpected increase of the pressurizer primary pressure during the conditioning phase (see figure 16). Liquid temperatures reached saturation quickly suggesting an overheating in the pressurizer (see figure 17). Moreover, calculation of the system energy balance showed an important disagreement in the environment heat losses between the experimental data and the simulation during the conditioning phase of the transient.

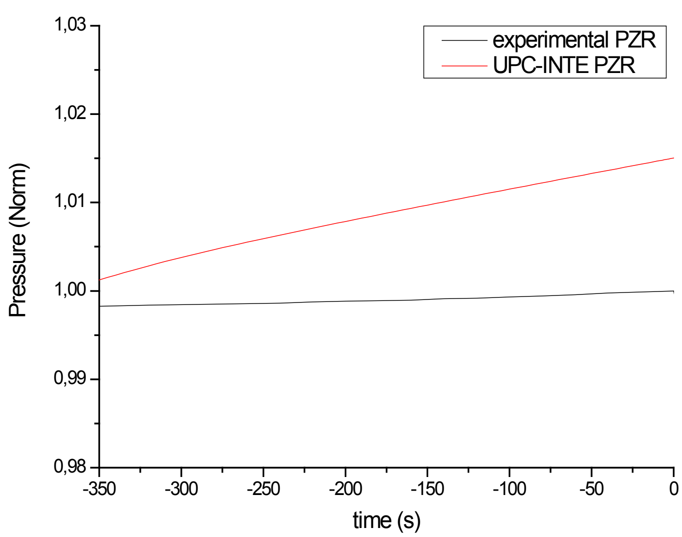

Figure 16   PRZ pressure during conditioning phase

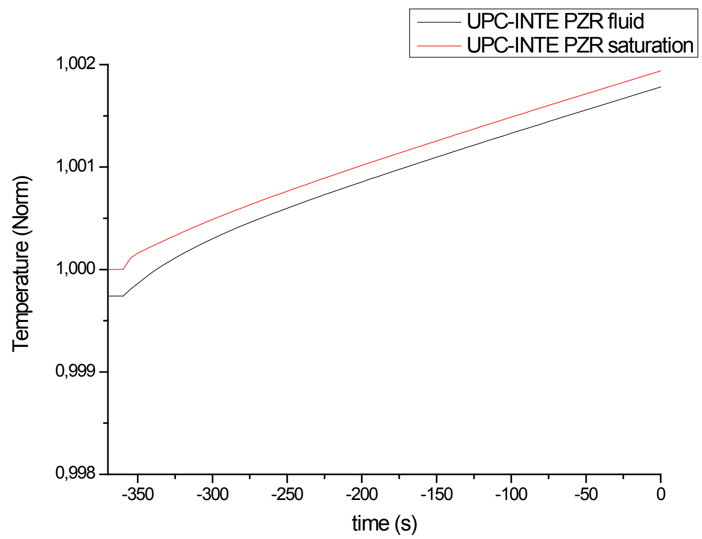

Figure 17 Fluid temperature in the pressurizer

Sensitivity analysis was performed over preliminary simulations to analyze and fix the environment heat losses. A set of simulations were modeled increasing gradually heat losses, especially in the pressurizer. Results showed environment heat transfer affects significantly the pressurizer collapsed liquid level (see figure 18) and the PORV discharge mass flow rate (see figure 19), delaying the dryout of the core (see figure 20) and so, the depressurization of the primary system as a result of the initiation of the LSTF core protection system. The UPC-INTE LSTF model was taken as a base case.

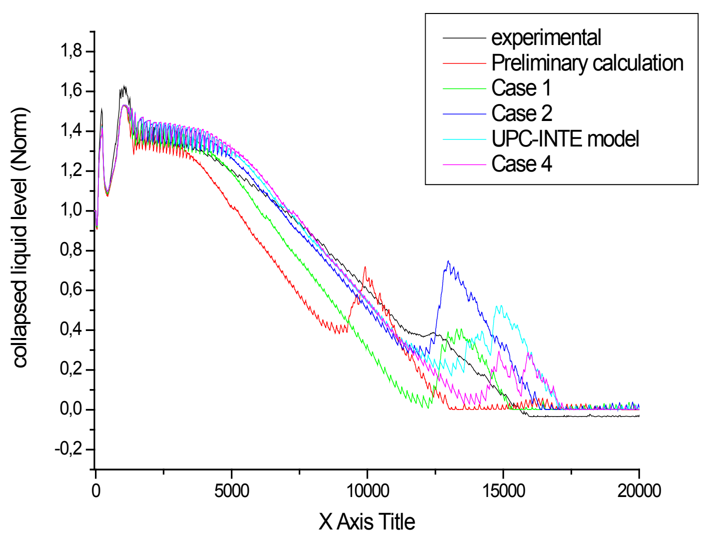

Figure 18 PZR collapsed liquid level

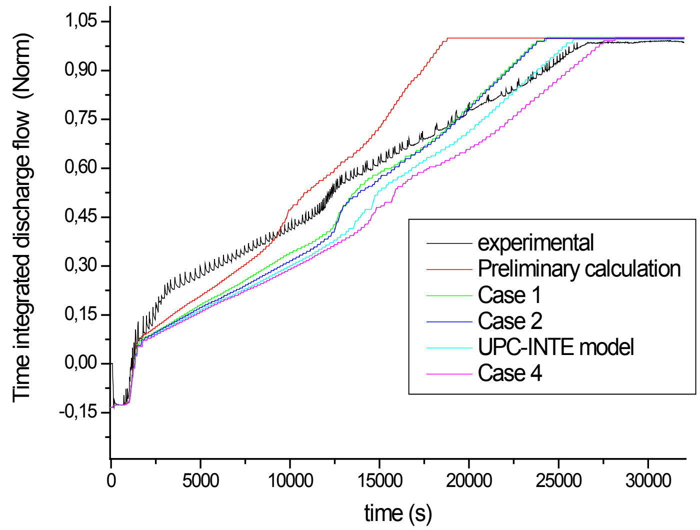

Figure 19 Time-integrated discharge flow through PORV

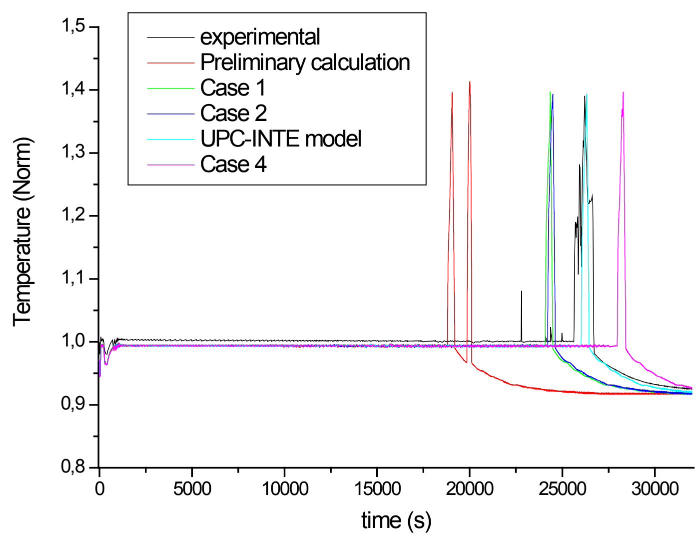

Figure 20 Maximum clad surface temperature

5. RUN STATISTICS

The calculations were performed on a Personal Computer with 3.4 GHz Pentium IV processor, 512 MB of RAM and Windows XP Service Pack 2 SO.

Table four shows main run statistics for all calculations performed in this report:

Table 4 Run Statistics

	Transient time	CPU time	Mass error ratio (emass/tmass)
Preliminary calculation	35900.0	5373.34	$4.3515 \cdot 10^{-5}$
Case 1	35900.0	5358.19	$4.11722 \cdot 10^{-5}$
Case 2	35900.0	5567.07	$4.3036 \cdot 10^{-5}$
UPC-INTE LSTF model	35900.0	5362.62	$4.3789 \cdot 10^{-5}$
Case 4	35900.0	5246.41	$4.4801 \cdot 10^{-5}$

6. CONCLUSIONS

UPC-INTE LSTF model developed for LSTF Test 3-1 [1] has been adjusted to the LSTF Test 3-2 proving its suitability to simulate the behavior of this facility.

Model predictions for Test 3-2 were in quite good agreement with the available experimental data. Several conclusions have been obtained from the study of local phenomena and sensitivity analysis. First, that RELAP5/mod3.3 reproduces U-tubes liquid level oscillations during High-Power Natural Circulation. Although condensation in U-tubes, which affects to the amplitude of the oscillations, is slightly under predicted by the code, the main events of the transient don't change significantly. On the other hand, sensitivity analysis shows the significance of the environment heat losses simulation. For long transients with a relatively high-core power, an incorrect implementation of the environment heat transfer causes discrepancies in the core power heat removal. Particularly, for Test 3-2, they modify wrongly the mass discharge through the PORV.

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10. SUPPLEMENTARY NOTES

A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The Thermal hydraulic Studies Group of Technical University of Catalonia (UPC) holds a large background in nuclear safety studies in the field of Nuclear Power Plant (NPP) code simulators. This report analyzes with RELAP5mod3.3 the LSTF Test 3-2, which simulates a loss-of-feedwater transient (LOFW) without scram and with a total failure of the high pressure injection system (HPIS). The simulation of several phenomena related with the high power natural circulation has been object of study as well as the impact of the environment heat losses in the primary coolant discharge across the pressurizer power-operated relief valve (PORV).

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