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International Agreement Report

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

Prepared by:
V Martinez-Quiroga, F. Reventos, C. Pretel

Institute of Energy Technologies
Technical University of Catalonia
ETSEIB, Av. Diagonal 647, Pav. C
08028 Barcelona
Spain

Kirk Tien, NRC Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
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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 ROSA-2 Test 3. This experiment is part of a Counterpart Test performed in LSTF and PKL Test Facilities within the framework of the OECD/NEA ROSA-2 and PKL-2 projects. Detailed core nodalizations and Pseudo 3D modeling have been object of study as well as the capabilities of the code for reproducing the correlation between the CET and the PCT.

FOREWORD

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

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

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

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

Many experimental facilities have contributed to the today's availability of a large thermal-hydraulic database (both separated and integral effect tests). However there is a continuous need for additional experimental work and code development and verification, in areas where no emphasis have been made along the past. On the basis of the SESAR/FAP[1] 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 the beginning of this century several collaborative international actions in the area of experimental TH research. These reports presented some findings and recommendations to the CSNI, to sustain an adequate level of research, identifying a number of experimental facilities and programmes of potential interest for present or future international collaboration within the nuclear safety community during the coming decade. The different series of PKL, ROSA and ATLAS projects are under these premises.

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

[1] 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).

enough investigated safety issues and phenomena (e.g., boron dilution, low power and shutdown conditions, beyond design accidents), or enlarging code validation and qualification data bases incorporating new information (e.g., multi-dimensional aspects, non-condensable gas effects, passive components).

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

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

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

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

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

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EXECUTIVE SUMMARY

Experimental research activities are being performed in Japan by the OECD ROSA 2 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 the experiment Test 3 of the LSTF and PKL Counterpart Test, which was carried out as a part of the OECD/NEA ROSA-2 and PKL-2 projects. The aim of this international synergy was to analyze the effectiveness of Core Exit Temperature measurement in Accident Management strategies as well as the scaling effects that appear between counterpart transients performed at different sizes and designs.

UPC LSTF Relap5mod33 nodalization, created for the simulation of the LSTF ATWS experiments, has been used and improved for this test. Two different nodalizations were prepared in order to check the capabilities of the code for simulating properly the correlation between the PCT and CET: one, the UPC LSTF 1D nodalization, with just one core channel; and a second, the UPC LSTF Pseudo-3D nodalization, with cartesian core channels and activating the transversal momentum equations.

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

In general, the results of the simulation demonstrated that 1D nodalizations are good enough for describing the general behavior of the transient as well as the main events and phenomena. On the other hand, for simulating accurately the correlation between the CET and PCT, Pseudo 3D modeling is necessary.

ACKNOWLEDGMENTS

This paper contains findings that were produced within the OECD-NEA ROSA-2 Project. The authors are grateful to the Management Board of the ROSA-2 Project for their consent to this publication.

ABBREVIATIONS AND ACRONYMS

ACC	accumulator
AM	accident management
ATWS	anticipated transient without scram
CET	core exit temperature
ECCS	emergency core cooling system
EOP	emergency operational procedure
HPIS	high pressure injection system
HS	heat structure
LPIS	low pressure injection system
LSTF	large scale test facility
MS	main steam
MSIV	main steam isolation valve
MSLB	main steam line break
NEA	Nuclear Energy Agency
NPP	nuclear power plant
OECD	Organization for Economic Cooperation and Development
PCT	peak cladding temperature
PKL	Primärkreislauf
PZR	pressurizer
PWR	pressurized water reactor
RELAP	reactor excursion and leak analysis program
ROSA	rig of safety assessment
RV	relief valve
IBLOCA	intermediate break loos of coolant accident
SBLOCA	small break loss of coolant accident
SG	steam generator
SGTR	steam generator tube rupture
UP	upper plenum
UPC	Universitat Politècnica de Catalunya (Technical University of Catalonia)
UT	u-tubes

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 Test 3, which forms part of the OECD/NEA ROSA-2 project.

1.1 PKL-2 and ROSA-2 Counterpart Test

In 2011 a Counterpart Test was performed in LSTF and PKL Test Facilities as a part of the OECD/NEA ROSA-2 and PKL-2 projects (Test 3 and test G7.1 respectively). The objective of both tests was devoted to analyze two aspects:

- Core Exit Temperature measurement effectiveness in Accident Management of NPP's
- Scaling effects between PKL and LSTF Test Facilities

In Accident Management strategies, core exit temperature measurement becomes crutial for detecting core dryout and for avoiding that PCT rises until safety limits. In the Counterpart Test, the relationship between CET and PCT was object of study in order to analyze EOP set points.

The selected scenario was a hot leg SBLOCA. System failures as no high pressure safety injection and no automatic secondary-side safety cooldown were imposed. The particular test conditions for the PKL-2 Test G7.1 are described in section 2.2. The main phenomena of interest were:

- Core boil-off with steam generation
- Steam flow toward hot-leg break
- Realistic pressure during core boil-off
- Relantionship between PCT and CET

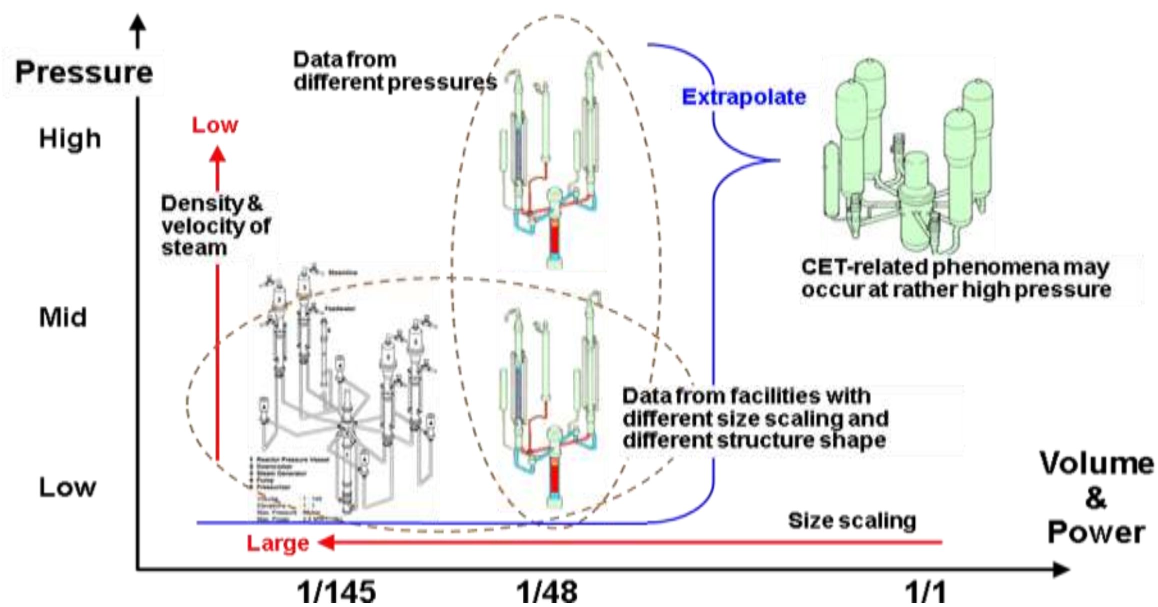

Figure 1 Diagram with the Different Counterpart Conditions (Courtesy of the OECD/NEA ROSA-2 Group)

About the scaling issue, core dryout and AM actions were simulated under counterpart conditions at different sizes (PKL -1:145- and LSTF -1:48-) and pressures (LSTF high and low pressure transient phases) in order to check how the scaling affects the CET and its related phenomena, and in order to sound out possible strategies for extrapolating the results to a commercial NPP (see figure one). The direct comparison of both transients was established between the whole transient of the PKL Test G7.1, and the low pressure transient phase of the LSTF Test 3.

1.2 <u>The OECD/NEA ROSA 2 Project</u>

The OECD/NEA ROSA-2 is an international project carried out between 2009 and 2012. It includes several types of integral 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 the complex flows that may appear during reactor accidents and transients such as intermediate break LOCAs and SGTR with accident management mitigation and under influences of MSLB.

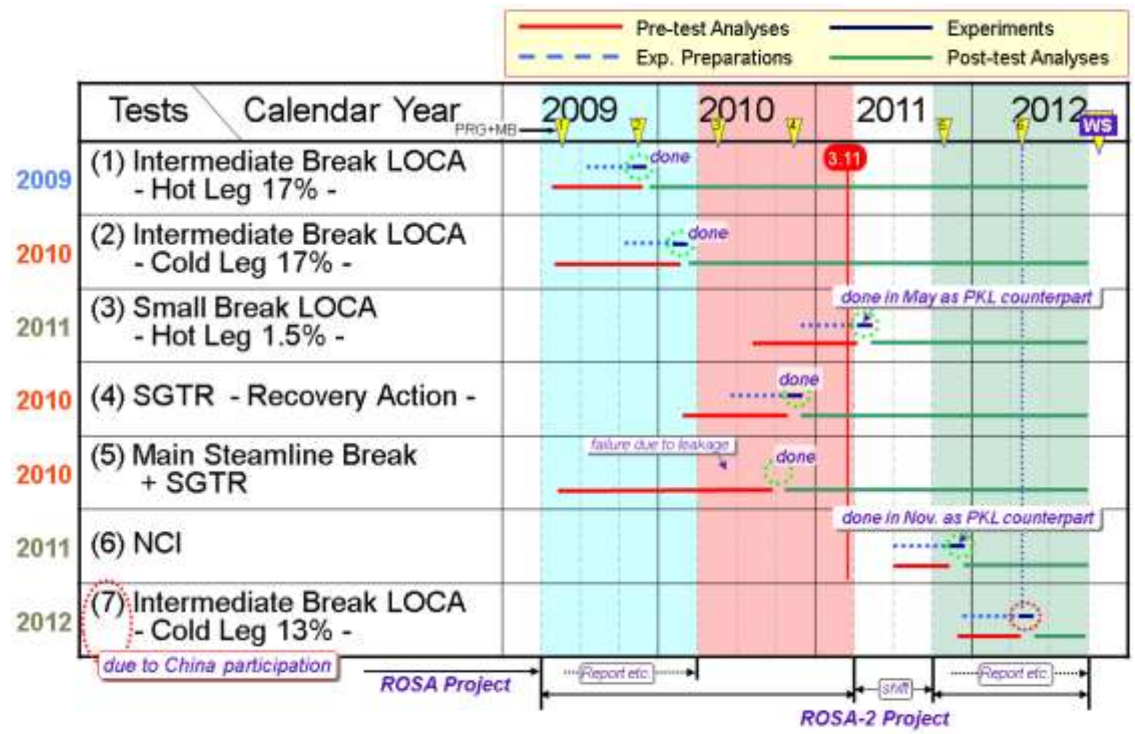

Figure 2 OECD/NEA ROSA-2 Project Experiments (Courtesy of the OECD/NEA ROSA-2 Group)

Seven experiments were performed along the project divided in three main groups: IBLOCAs, SGTRs, and counterparts tests. The test 3, described in this report, is included in the counterpart test and linked with the PKL-2 Test G7.

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…-).

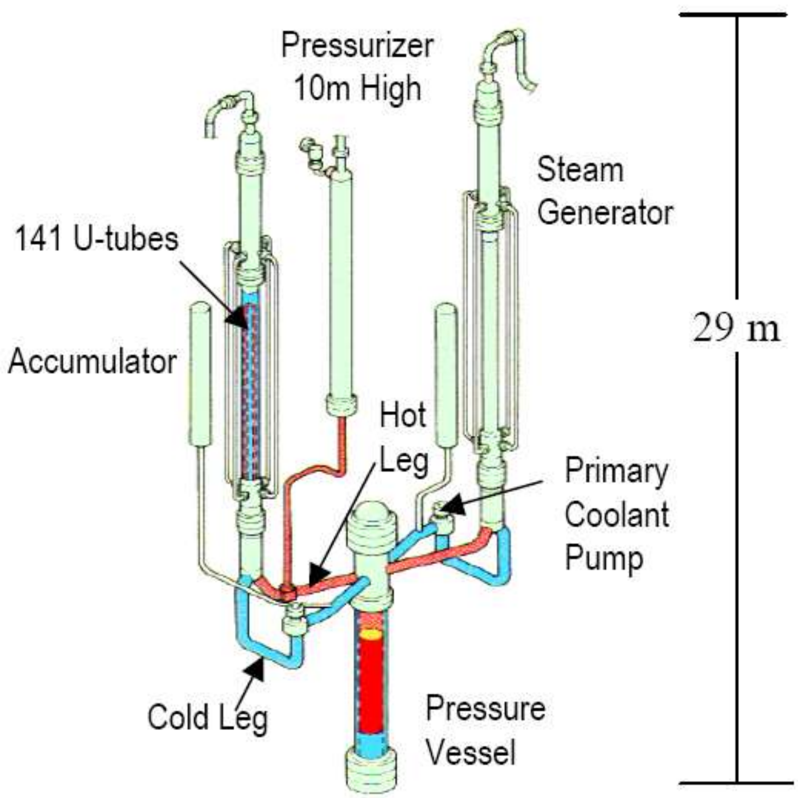

Figure 3 LSTF Test Facility (Courtesy of the OECD/NEA ROSA-2 Group)

2.2 Experimental Conditions

The hardware configuration of LSTF is described in references [1] and [2]. Some important points are the following:

- *Break*: upward oriented SBLOCA (1.5 %) at Hot Leg without PZR.
- *ECCS:* HPIS full failure.
  ACCs and LPIS set points fixed at 2.6 and 1.0 MPa respectively.
  ACC level and LPIS mass flow adjusted according to PKL final conditions.
- *Core power curve:* pre-determined from a previous volumetrically scaled analysis. For counterpart phases, the power is fixed to an 1.8 % of the nominal power.
- *SG depressurization:* fully opening of 2 SG relief valves when CET achieves 623 K.
- *Steam generator relief valves:* modified considering the LSTF/PKL scaling ratio.

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.

The transient was divided in three phases: a high pressure phase, reproducing the NPP scenario at full pressure; a low pressure phase, reproducing the same scenario at counterpart conditions with PKL; and finally, an intermediate phase, with the purpose of conditioning the LSTF conditions at the end of the high pressure phase to the PKL counterpart test conditions. Table one shows the list of imposed conditions for each phase.

Table 1 Test 3 Test Conditions

Event	Condition	HP phase	Cond. phase	LP phase
Break valve opened	$t = 0$ s			
Low pressure scram signal	$P_{prim} < 12.96$ MPa			
Secondary system isolation	scram signal			
Initiation of primary coolant pump coastdown	scram signal			
Initiation of core power decay curve simulation	scram signal			
Initiation of HPI coolant injection into PV UP	$T_{PCT} > 750$ K			
Break valve closed	$P_{prim} < 5$ MPa			
Power constant	$P_{prim} < 5$ MPa			
Termination of HPI coolant injection into PV UP	$H_{HL} \cong ½ \cdot H_{HL}$			
SG's RV depressuritzation	$H_{HL} \cong ½ \cdot H_{HL}$			
Secondary system isolation	$P_{prim} < 3.9$ MPa			
Break valve re-opened	$P_{prim} > 4.5$ MPa			
SG depressurization as AM action	$T_{CET} > 623$ K			
Initiation of AFW in both loops	AM action signal			
Initiation of ACC system in both loops	$P_{prim} < 2.6$ MPa			
Termination of ACC system in both loops	$P_{prim} < 1.2$ MPa			
Initiation of LPI system in both loops	$P_{prim} < 1$ MPa			

Several parameters were adjusted from the PKL test conditions with the aim of having analogous behaviors during the low pressure phase of the transient. A scaling factor K_v=2.55 was applied for calculating the opening area of the SG's relief valves, the water volumes of the accumulators, and the injection rates of the low pressure injection (LPI) system. Pressures in the primary and secondary systems were adjusted 1:1 to the PKL test conditions during the conditioning phase, and the mass inventory was reinstated in order to have a similar hot leg liquid level as expected to occur in the PKL test. Reflux and condenser conditions were established in both facilities at the beginning of the counterpart phase. Accumulators set point was reduced to 2.6 MPa for including their actuation in both facilities.

2.4 Test Description

The main events are described in table two:

Table 2 Chronology of the Main Events of Test 3

Event	**Experimental data [s]**	**UPC LSTF 1D nodalization [s]**	**UPC LSTF Pseudo 3D nodalization [s]**
High pressure transient phase	-	-	-
Break	0	0	0
SCRAM signal: · Turbine trip and closure MSIV · PZR heater off · Termination main feedwater	25	33	33
Initiation of coastdown of primary coolant pumps	31	39	39
Termination of continuous opening of SG RVs, termination of two-phase natural circulation, break flow from single-phase liquid to two-phase flow	600	1238	1185
Core liquid level starts to decrease (core uncovered)	1545	1460	1475
End high pressure transient phase (PCT > 750 K)	1840	1852	1778
Condititioning phase	-	-	-
HPI system activated	1850	1862	1788
Break valve closed	2163	2169	2161
HPIS closed	2852	2852	2852
SG depressurization	2880	2892	2880
Termination of SG depressurization	3024	3008	3012
End of Conditioning phase (break valve re-opening)	3323	3323	3323
Low pressure transient phase	-	-	-
Break valve re-opening	3323	3323	3323
Primary pressure lower than SG secondary pressure	4108	4085	4105
SG depressurization (CET > 623 K)	4392	4297	4388
Initiation of ACC system (primary pressure = 2.6 MPa)	4505	4419	4488
Initiation of LPIS (primary pressure = 1.0 MPa)	5005	4660	4741
End of the transient	5500	5500	5500

2.4.1 High Pressure Transient Phase

The transient starts at t=0 seconds with the opening of the break valve. Primary pressure drops rapidly as a result of the break mass losses, and low pressure scram signal occurs at 25 seconds. At this point, secondary system is isolated and secondary pressure starts to increase until achieving SG RVs set points. During the first 600 seconds, two phase natural circulation occurs and there is a continuous SG RVs opening as a result of the efficient heat transfer from primary-to-secondary side. Suddenly, liquid level stratifies in the U-tubes of the broken loop and reflux and condensation starts. This phenomenon keeps constant until the stratified liquid level of the broken loop starts to decrease (1200 seconds). At that time, vapor mass flow ratio across the break increases and primary pressure drops below secondary pressure as results of higher residual heat removal. At t=1545 seconds core uncover occurs and PCTs start to increase, When the temperature achieves 750 K, HPIS starts to add water in the UP and high pressure transient phase is finished.

2.4.2 Conditioning Phase

Conditioning phase starts with the initiation of the HPIS. At that moment, power is fixed constant to 1.8 % of the nominal power. During this phase, core temperatures are quenched instantly while primary pressure carries on dropping. When 5 MPa is achieved (2163 s), break valve is closed and HPIS is kept active in order to restore the mass inventory with similar conditions as in the PKL-2 Test G7.1. At 2852 seconds, HPIS is closed and SG depressurization is induced. The aim is to reduce system pressures to 4.5 MPa as in the PKL G7.1 boundary conditions. When reflux and condensation is restored at that pressure, break valve is re-opened and low pressure transient phase is started.

2.4.3 Low Pressure Transient Phase

During the first 785 seconds, core is cooled under saturated conditions and reflux and condensation occurs in the UTs of the SGs. In this phase of the transient, primary pressure keeps constant over the 4.4 MPa of the isolated secondary side. After that, core uncovery starts, and few seconds later (4150 seconds), primary pressure drops below secondary pressure. In this second phase, there is vapor superheating in the core, and the CET and the PCT rise above the temperature of saturation (with a delay of 160 seconds between both).

When the CET temperature achieves the 623 K (4392 seconds), SGs are depressurized by relief valves opening. System pressures drop rapidly inducing a complete core quenching. Finally, when primary pressure drops below 1.0 MPa (5005 seconds), LPIS starts to inject water, compensating break losses and keeping constant plant parameters. At 5500 seconds, break valve is closed and the end of the transient is declared.

3 CODE INPUT MODEL DESCRIPTION

Two different nodalizations were prepared by making use of the UPC LSTF RELAP5mod3.3 nodalization. This nodalization had been previously qualified with the simulation of the OECD/NEA ROSA tests 3-1 and 3-2 (see references [3], [4]). Particular Counterpart nodalizations were:

- UPC LSTF 1D nodalization
- UPC LSTF Pseudo-3D nodalization

Differences between both models are exclusively related with the charactesteristics of the vessel modeling.

3.1 UPC LSTF 1D Nodalization

Main characteristics of the vessel model in the UPC LSTF 1D nodalization are:

- Core: 1 channel
- Fuel: 1 Heat structure (average of High, Medium, Low power axial profiles)
9 axial levels
- Other HS: Control rods
Core Barrel
Instrumentation
Environment Heat losses
Upper core support plate
- UP: 1 channel

3.2 UPC LSTF Pseudo-3D Nodalization

Pseudo 3D nodalization was implemented in order to check code capabilities for simulating core exit and peak cladding temperatures. In that sense, core channel of the UPC LSTF 1D nodalization was divided in 13 channels with 18 axial levels (see figure four). The low, medium and high core power axial profiles were simulated, arranging them in each channel as in the experimental radial power distribution. Cartesians crossflows were used for organizing them radially and transversal momentum equations were activated in order to take into account the possible radial ΔP's. Passive HS's were split according to the geometries. Finally, the UP was modified simulating it with two channels, one hot channel, connected to the outlet of the hottest core channel, and another one simulating the rest of the plenum. Transversal momentum equations were also activated in order to consider ΔP's in the vessel that could affect flow path to the hottest channel during the UP HPIS injection.

On the other hand, HPI mass flow was modified in the conditioning phase in order to mach primary mass inventory at the beginning of the low pressure phase. This correction was justified for the analysis of the following counterpart phases in which the achievement of specified initial conditions are necessary.

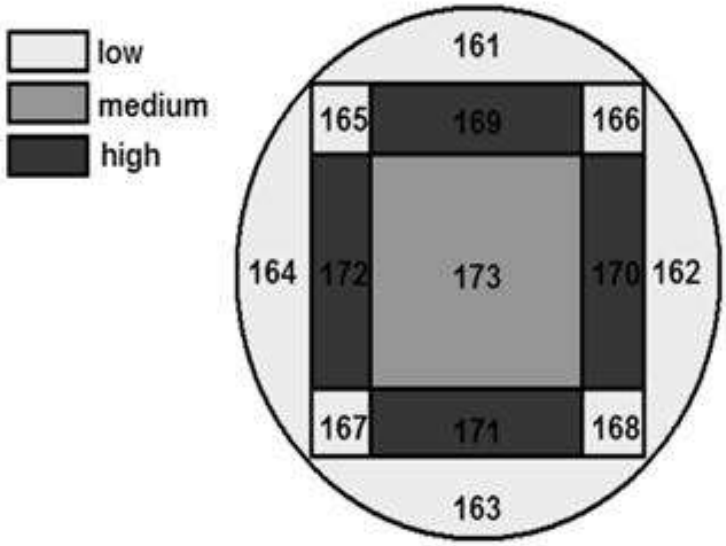

Figure 4 Core Channels of the UPC LSTF Pseudo-3D Nodalization

4 RESULTS

A base case calculation was performed using the UPC LSTF 1D nodalization. Results showed a quite good agreement for reproducing initial steady state conditions and main events of three phases of the transient (see tables three and two). There was a slight overestimation of the break mass losses when stratification appeared in the hot leg, so the break was in two-phase discharging mode (see figure five from 750 s to 1500 s). As a result of this, core uncovery occurred slightly earlier for both phases of the transient (see table two), and consequently, the SG depressurization signal related with the CET was activated in advance as well (see table two and figure six). In any case, the main parameters were consistently reproduced (see figures seven, eight and nine).

Table 3 Initial Conditions of Test G7.1

	Experimental data (loops w/wo PZR)	**UPC LSTF 1D nodalization (loops w / wo PZR)**
Core power (Norm.)	1	1
Hot leg temperature (Norm.)	1	1.001
Cold leg temperature (Norm.)	1	1.003
Mass flow rate (x loop) (Norm.)	1	1.002 / 0.998
Downcomer-to-hot-leg bypass (Norm.)	1	1
Pressurizer pressure (Norm.)	1	1
Pressurizer liquid level (Norm.)	1	1.014
Secondary-side pressure (Norm.)	1 / 1	1.004 / 0.997
Secondary-side liquid level (Norm.)	1	0.995
Main feedwater temperature (Norm.)	1	1
Main feedwater flow rate (Norm.)	1	1.007 / 1.004
Accumulators pressure (Norm.)	1	1
Accumulators temperature (Norm.)	1	1
LPI pressure (initiation of system) (Norm.)	1	1.24
LPI temperature (Norm.)	1	1
Steam flow rate (Norm.)	1	1.007 / 1

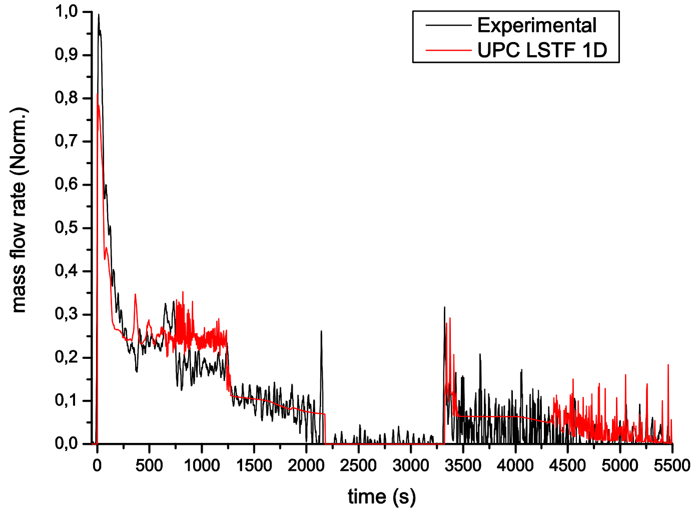

Figure 5 Break Mass Flow

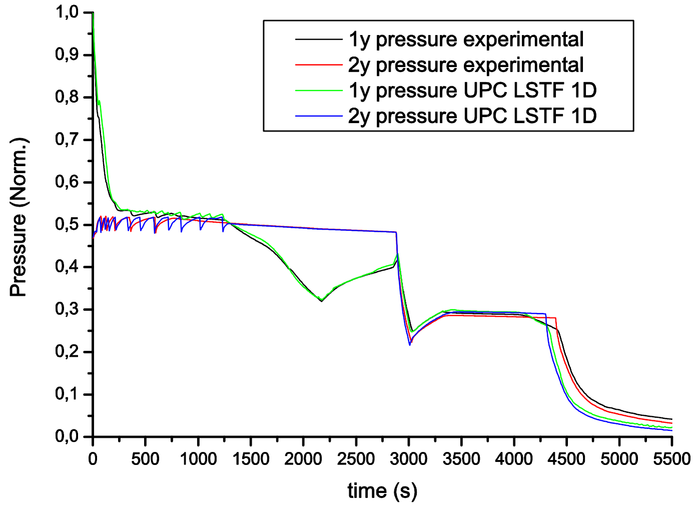

Figure 6 Core System Pressures

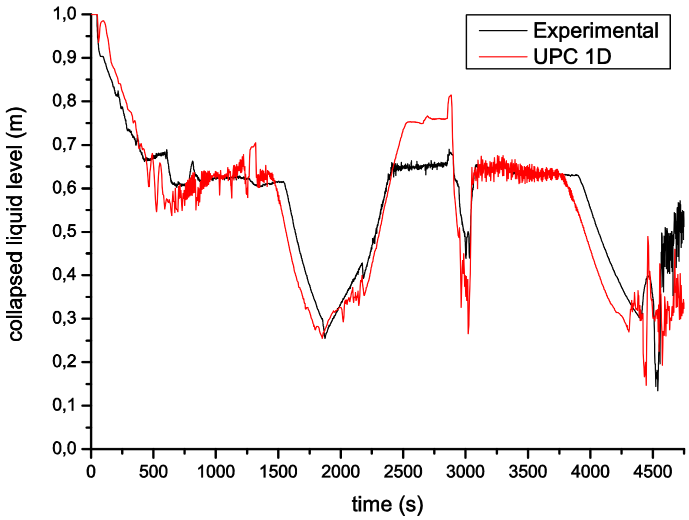

Figure 7 Core Level

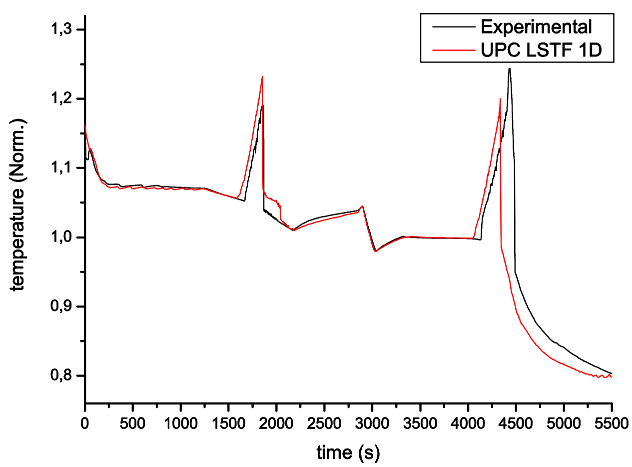

Figure 8 Core Exit Temperatures

Focusing on the relation between the CET and the PCT, results showed a disagreement in the slope of the plots (see figures nine and 10). These results suggested that UPC LSTF nodalization should be improved following a similar approach as the one (Pseudo 3D modeling) applied in the PKL analysis [5].

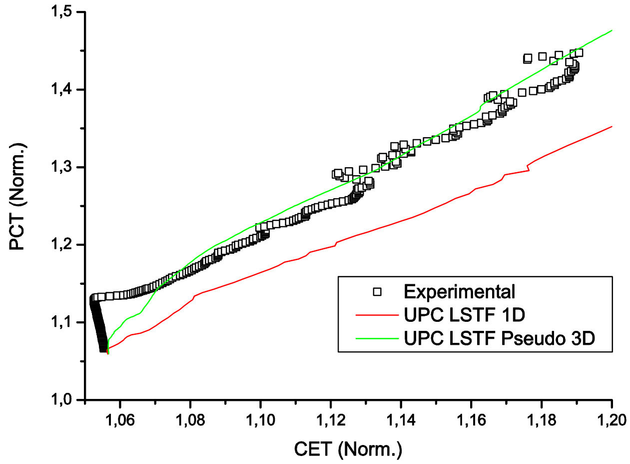

Figure 9 PCT vs CET Curve During High Pressure Transient Phase

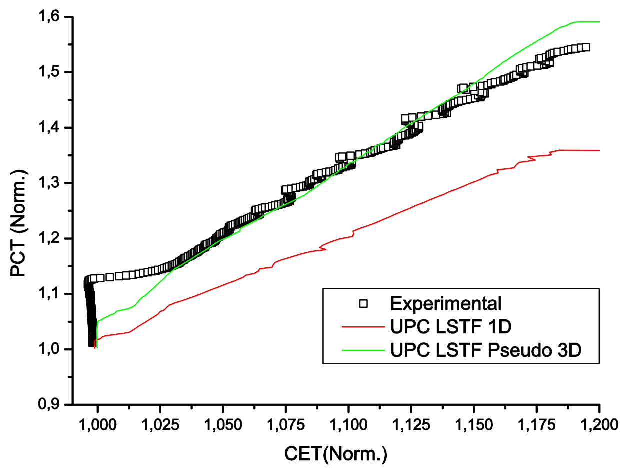

Figure 10 PCT vs CET Curve During Low Pressure Transient Phase

Results of the UPC LSTF Pseudo 3D nodalization kept the consistency in the simulation of the main events (see table two) On the other hand, they showed a close agreement in the simulation of the CET vs PCT relationship compared to the results of the UPC 1D nodalization (see figures nine and 10). For both high and low pressure transient phases, Pseudo 3D nodalization reproduces the same slope and correlation with a margin of 50 and 70 K respectively over AM signal established for this particular scenario ($T_{CET} > 623$ K). Regarding to core uncovery, the adjustment of initial mass inventory at the beginning of the counterpart phase solves the delay (see low pressure transient phase of table two), obtaining a closer agreement in the simulation of reflux condenser, vapour superheating and AM phases (see figure 11).

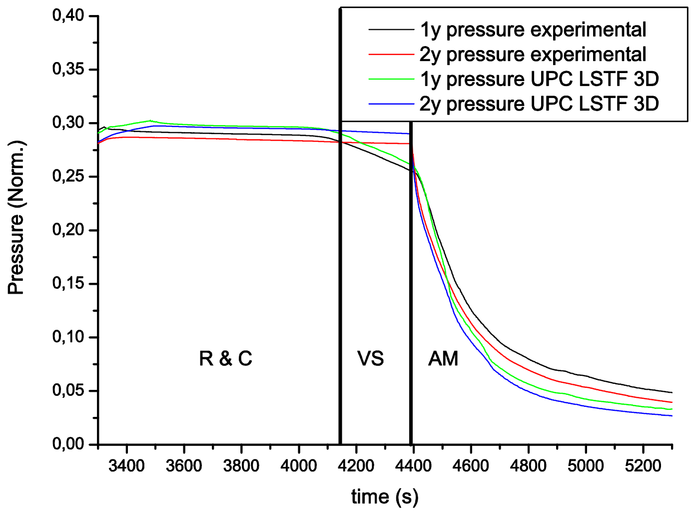

Figure 11 System Pressures

5 RUN STATISTICS

The calculations were performed on a Personal Computer with 3.0 GHz Intel Core Duo processor, 1.97 GB of RAM and Windows XP Service Pack 3 OS.

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

Table 4 Run Statistics

	Transient time (s)	**CPU time (s)**	**Mass error ratio (emass/tmass)**
UPC LSTF 1D nodalization	11000.0	2787.0	$2.8937 \cdot 10^{-5}$
UPC LSTF Pseudo-3D nodalization	11000.0	7690.0	$2.3385 \cdot 10^{-5}$

6 CONCLUSIONS

The UPC LSTF Relap5mod33, that was qualified for the ATWS experiments of the ROSA project, has been adjusted to the ROSA-2 Test 3 proving its suitability to simulate the behavior of this transient. Results showed code and nodalization capabilities to reproduce main phenomena of the transient.

Regarding core modeling, Pseudo 3D vessel modeling has shown to be a good tool for simulating core dryout and CET vs PCT correlation. On the other hand, results of the 1D channel nodalization have demonstrated the necessity of finer core nodalizations for analyzing CET effectiveness in AM strategies.

Closer results on CET vs PCT curve are a good starting point for later scaled plant applications.

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ETSEIB, Av. Diagonal 647, PAv. C
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K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

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 ROSA-2 Test 3. This experiment is part of a Counterpart Test performed in LSTF and PKL-2 Test Facilities within the framework of the OECD/NEA ROSA-2 and PKL-2 projects. Detailed core nodalizations and Pseudo 3D modeling have been object of study as well as the capabilities of the code for reproducing the correlation between the CED and the PCT.

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