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International Agreement Report

Post-Test Calculation of the PKL-2 Test G7.1 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 G7.1 PKL Test. 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 Core Exit Temperature (CET) and the Peak Cladding Temperature (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 enough investigated safety issues and phenomena (e.g., boron dilution, low power and

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[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).

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 Germany by the OECD PKL 2 project with the aim of addressing thermal-hydraulic safety issues for current PWR and new PWR design concepts. These experiments are carried out at the PKL III test facility.

This report analyses the experiment G7.1 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 PKL Relap5mod33 nodalization, created for the simulation of the PKL boron dilution 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 PKL 1D nodalization, with just one core channel; and a second, the UPC Pseudo 3D nodalization, with concentric 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 PKL-2 Project. The authors are grateful to the Management Board of the PKL-2 Project for their consent to this publication.

ABBREVIATIONS AND ACRONYMS

ACC	accumulator
AM	accident management
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
SBLOCA	small break loss of coolant accident
SG	steam generator
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 G7.1, which forms part of the OECD PKL-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 critical 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
- Relationship between PCT and CET

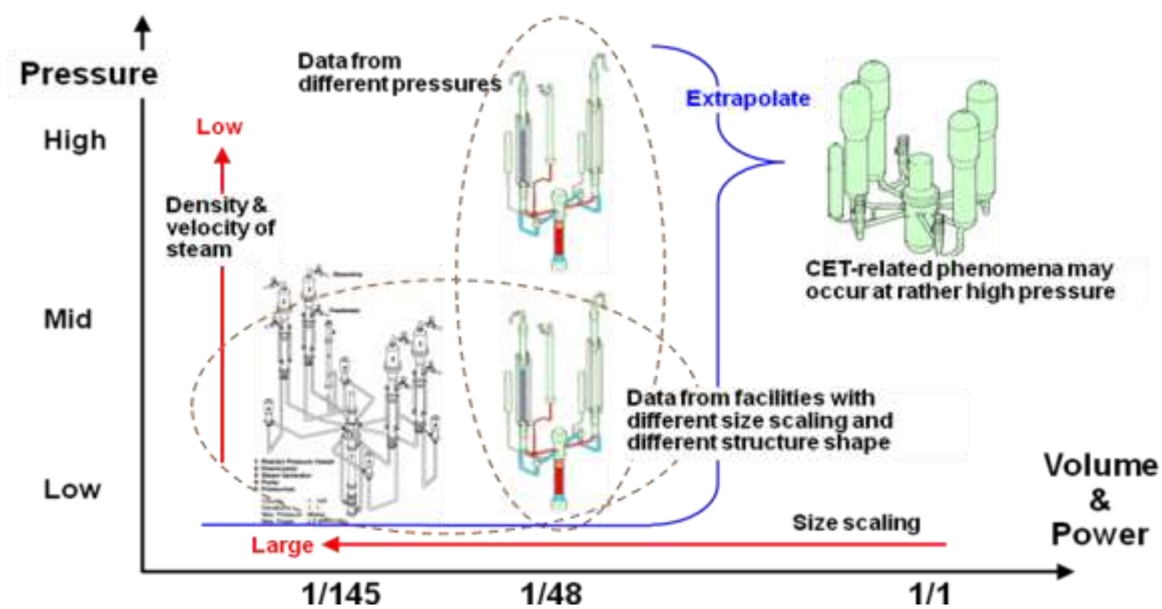

Figure 1 Diagram with the Different Counterpart Conditions (Courtesy of the OECD/NEA ROSA-2 Group)

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 PKL 2 Project</u>

The OECD PKL 2 is an international project carried out between 2008 and 2011 that includes several types of integral experiments (see figure two) on PKL III, ROCOM and PMK test facilities. The aim of the project is to address thermal-hydraulic safety issues for current PWR and new PWR design concepts. It also provides a wide database for the validation of computer codes and models for system integral analyses coupled with detailed analyses of local phenomena.

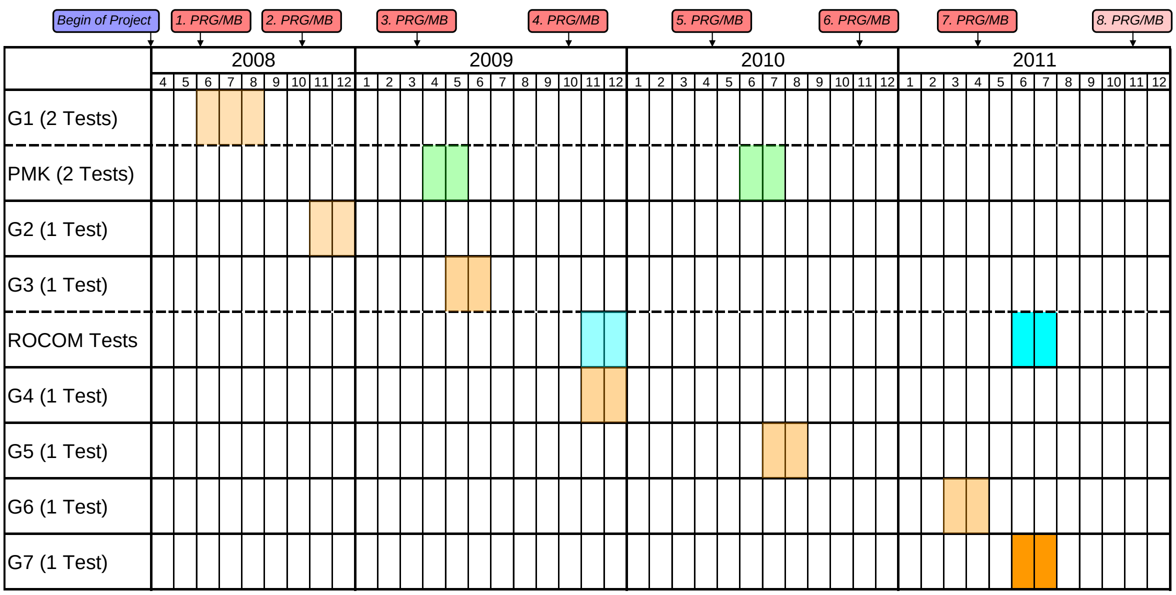

Figure 2 OECD PKL 2 Project Experiments (Courtesy of the OECD/NEA PKL-2 Group)

The topics of the experiments are:

G1 Systematic investigation of the heat transfer mechanisms in the SGs in presence of nitrogen, steam and water
G2 Cooldown procedures with SGs isolated and emptied on the secondary side
G3 Fast cooldown transients (main steam line break)
G4 Systematic study on heat transfer in the SG under reflux condenser conditions
G5 Boron precipitation processes after LB – LOCA
G6 RCS cooldown with void formation in RPV upper head
G7 Counterpart Test with the OECD/NEA ROSA-2 Test 3 (hot leg SBLOCA)

2 FACILITY AND TEST DESCRIPTION

2.1 PKL III Test Facility

PKL is an experimental power facility plant designed to simulate pressurized water reactors (PWR) under accidental conditions. The plant, situated in Erlangen (Germany), has been the scenario of several experiments in the last 25 years. The PKL facility replicates the entire primary system and most of the secondary system (except for the turbine and condenser) of a 1300-MW PWR plant, with elevations scaled 1:1 and diameters reduced by a factor of 12. Therefore volumes and power have been reduced by a factor of 145. The number of rods in the core and the U-tubes in the steam generator has been divided by 145 too. The core has been modeled by 314 electrical heater rods. Unlike many experimental facilities with only two available loops (one for the break loop and one to simulate the other three loops) PKL simulates all four loops separately. This is very important in order to analyze asymmetrical transients, e.g. with injection in two out of four loops. A diagram of the PKL test facility is shown in Figure 3.

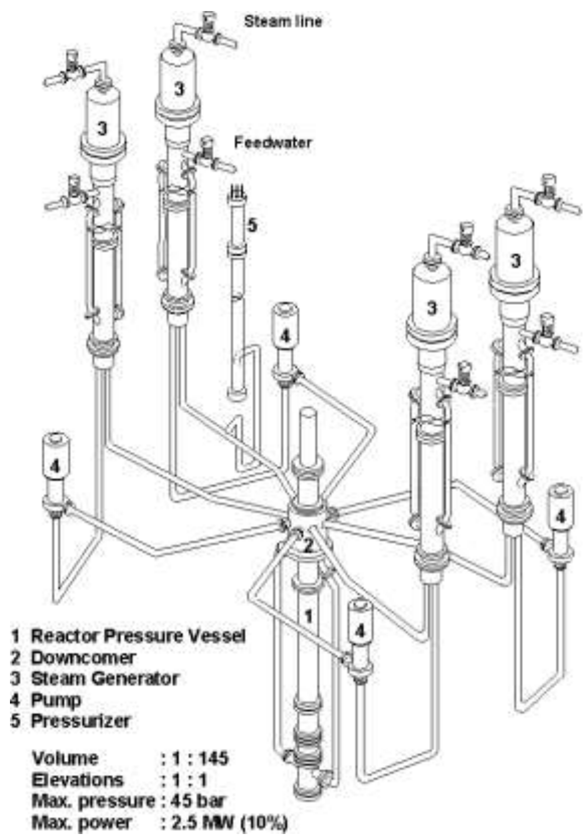

Figure 3 PKL Test Facility (Courtesy of the OECD/NEA PKL Group)

The operating pressure of the PKL facility is limited to 45 bars on the primary side and to 56 bars on the secondary side. This allows simulation over a wide temperature range (322K to 522K) that is particularly applicable to the cooldown procedures investigated.

All emergency systems are represented and have a wide versatility referred to their functions and positions. There are 8 accumulators (one for each cold and hot leg). The pump injection system is available in all the hot and cold legs. Many break locations are available too.

PKL test facility has about 1500 measurement points that permit an exhaustive analysis of the tests. There are measurement devices for cladding, wall and fluid temperature, absolute and differential pressure, one and two phase mass flow, density and boron concentration.

Sixty of the measurement devices are identical to those that are used in a commercial plant to simulate what an operator would control in case of accident.

2.2 Experimental Conditions

The hardware configuration of PKL 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 26 bars and 8 bars respectively.
  ACC level and LPIS mass flow adjusted according to LSTF final conditions.
- *Core power curve:* constant at 1.8 % of the nominal power. Additional heat was added in order to compensate the differences between the environment losses of LSTF and PKL.
- *SG depressurization:* fully opening of 2 main steam relief valves when CET achieves 350 ºC. The depressurization of the 4 SGs was simulated connecting them via main steam header.
- *Main steam relief valves:* modified for having same ratio of LSTF valves.

2.3 Initial Conditions

The PKL initial conditions were adjusted at its maximum pressures in order to reproduce as realistic as it can PWR SBLOCA reflux and condensation, core dry-out and Accident Management phases.

In relation to scaling, the mass inventory in the secondary side was adjusted using K_v factor in order to have the same ratio between liquid and energy storage in the SG’s. In the primary system, mass inventory was tuned for having the same reflux condenser and break discharging initial conditions than the low pressure transient phase of the LSTF Test 3.

2.4 Test Phase

The transient starts at t=0 seconds with the opening of the break valve. During the first 940 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 44 bars of the isolated secondary side. After that, core uncovery starts, and few seconds later (at 1020 seconds), primary pressure drops below secondary pressure. In this second phase, there is vapour superheating in the core, and the CET and the PCT rise above the temperature of saturation (with a delay of 270 seconds between both).

When the CET temperature achieves the AM temperature set point (1360 seconds), SGs are depressurized by MS relief valves opening. System pressures drop rapidly without a complete

core quenching, so CET and PCT temperatures do not drop to saturated conditions until accumulator injection system is activated (1500 seconds).
Finally, when primary pressure drops below 8 bars (2060 seconds), LPIS starts to inject water, compensating break losses and keeping constant plant parameters. At 5685 seconds, break valve is closed and the end of the transient is declared.

The main events are described in Table 1:

Table 1 Chronology of the Main Events of Test G7.1

	Experimental (s)	UPC 1D nodalization (s)	UPC Pseudo-3D nodalization (s)
Start of the transient	0	0	0
Begin of core uncovery	940	800	940
Primary pressure below secondary pressure	1020	920	1010
Secondary side depressurization	1360	1190	1295
Start of accum. Injection	1500	1304	1450
ACC injection finished	1860	1712	1752
LPIS started	2060	1966	1993
End of the test	5685	5685	5685

3 CODE INPUT MODEL DESCRIPTION

Two different nodalizations were prepared by making use of the UPC PKL RELAP5mod3.3 nodalization. This nodalization had been previously qualified with the simulation of the OECD/NEA PKL I tests F1.1 and F1.2, and the OECD/SETH tests E2.1, E2.2 and E2.3 (see references [3], [4] and [5] and [6]). Particular Counterpart nodalizations were:

- UPC PKL 1D nodalization
- UPC PKL Pseudo-3D nodalization

Several improvements were done for both nodalizations taking advantage of the particular specifications of the transient. In this way, main steam system was modified following the sketch shown in Figure 4. Moreover, realistic bypass heaters were added to the PZR and the SGs (as in Figure 5) in order to have a proper power balance between the environment heat losses and the power supplied by the exchangers.

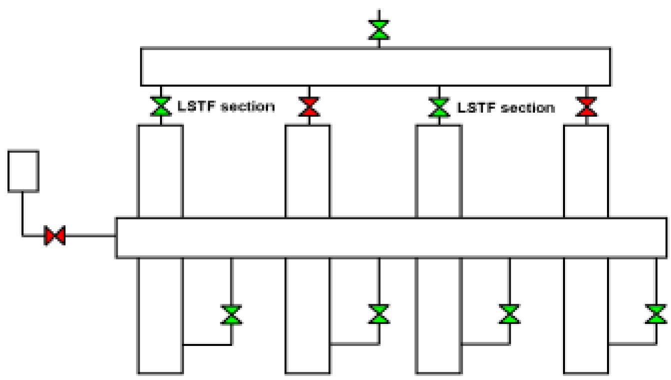

Figure 4 Nodalization of the Main Steam System

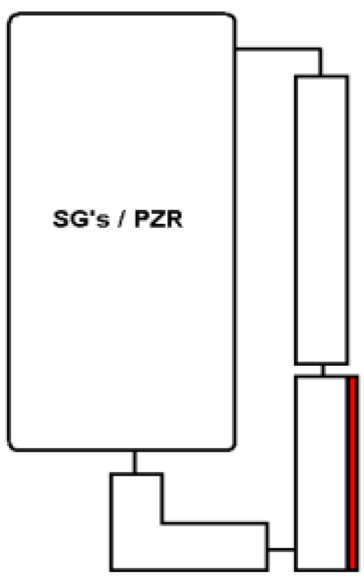

Figure 5 Nodalization of the Bypass Heaters

The differences between both nodalizations were exclusively related to the core and upper plenum (UP) modeling. UPC PKL 1D nodalization simulated them with one channel (in addition to the core bypass), having fuel and all passive heat structures (core barrel and unheated rods) linked to the same volumes. The fuel was modeled with three HS's, with the same power ratio and divided in 7 axial levels.

UPC PKL Pseudo-3D nodalization had the core and UP (until the CET thermocouple level) divided in three radial channels (see Figure 6), with one fuel HS for each channel. The HSs for the passive internal metal structures were split for each channel proportionally to the flow path of each one, and the core barrel was linked to the outer zone. The radial flow paths between cells were modeled and transversal momentum equations were activated following the recommendations of reference [7].The total number of core axial meshes was increased to 14 and the UP cell heights were adjusted so that the center of the node coincided with the elevation of the thermocouples in the test facility.

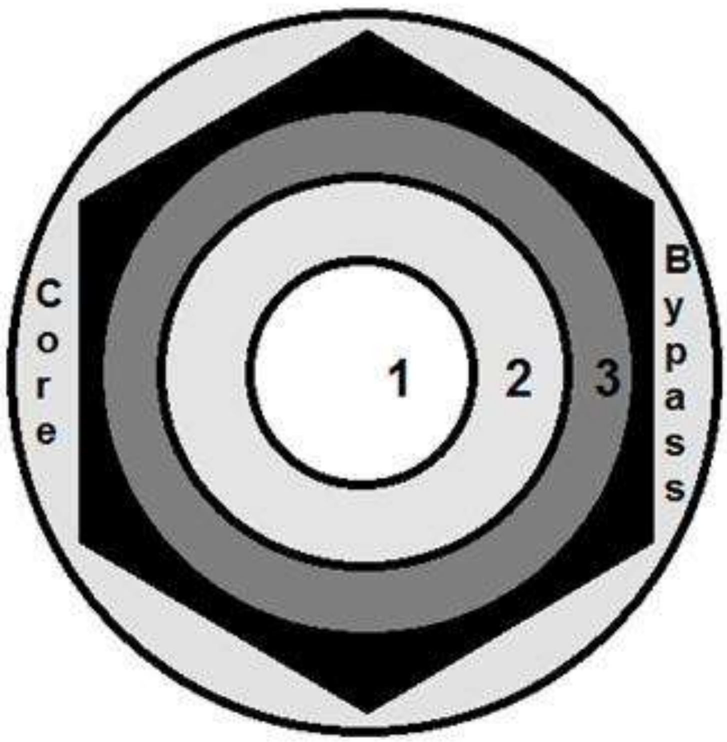

Figure 6 UPC PKL Pseudo 3D Nodalization Core Channels

4 RESULTS

The results obtained for both nodalizations showed a quite close agreement reproducing the initial conditions (see Table 2) as well as the general behavior of the transient (see Figures 7 and 8). The Pseudo 3D nodalization provided closer results for the main events because it reduced the delay in the core uncovery (see Table 1). This was seen to be a consequence because of a higher vapor generation in the 1D nodalization during the phases of reflux and condensation and vapor superheating. It implied that, for similar break mass losses, liquid mass inventory decreased faster and core uncovery started before. In Figure 9, the differences between vapor generation and break mass flows are compared (the differences are calculated by substracting the values of the Pseudo 3D nodalization to the 1D nodalization).

Table 2 Initial Conditions of Test G7.1

	Experimental data	UPC 1D nodalization	UPC Pseudo 3D nodalization
Core power (Norm.)	1	0,996	0,996
Pressurizer pressure (Norm.)	1	1	1
Pressurizer liquid level (Norm.)	1	0,7	0,7
Secondary-side pressure (Norm.)	1	1	0.998
Secondary-side liquid level (Norm.)	1	0.998	1
Main feedwater temperature (Norm.)	1	1	1
Accumulators pressure (Norm.)	1	1	1
Accumulators temperature (Norm.)	1	1	1
LPIS pressure (initiation of system) (Norm.)	1	1	1
LPIS temperature (Norm.)	1	1	1

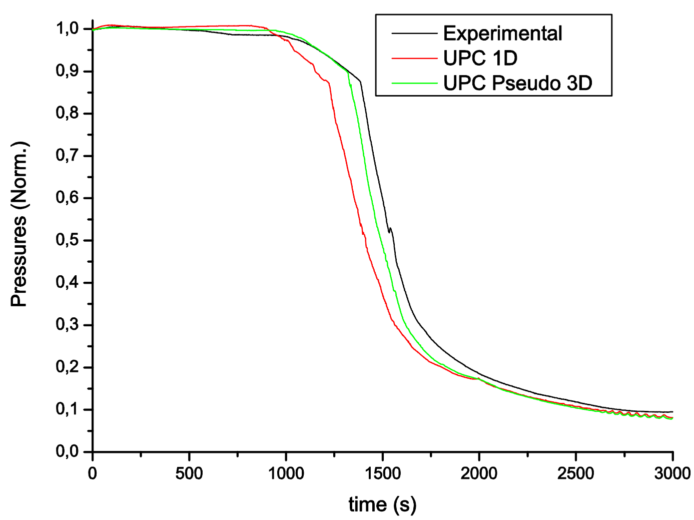

Figure 7 Primary Pressure

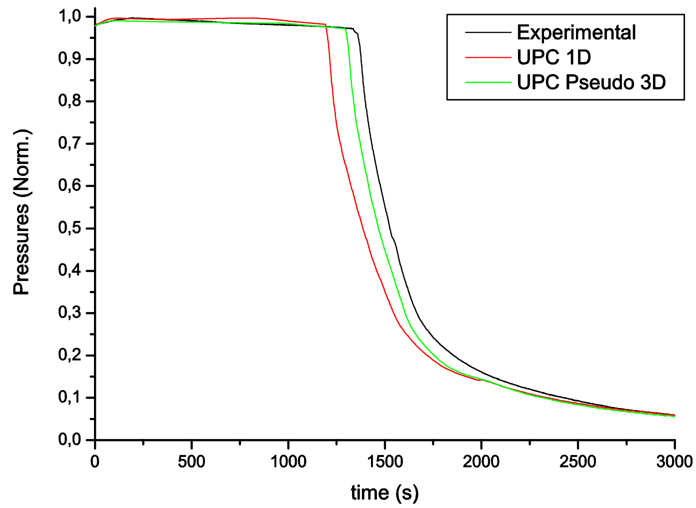

Figure 8 Secondary Pressure

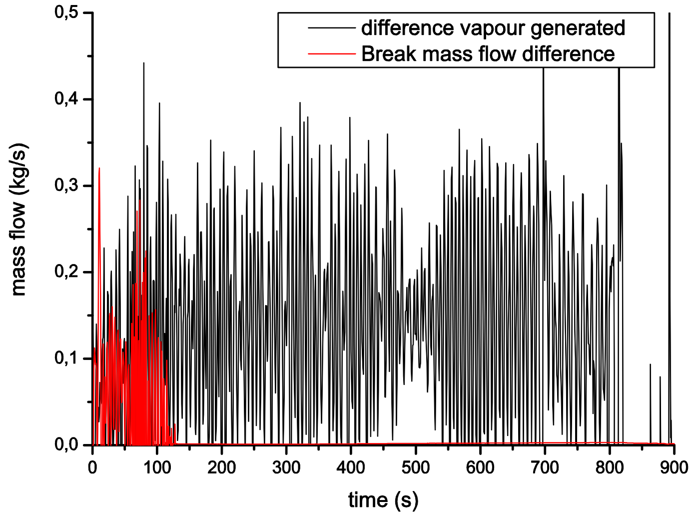

Figure 9 Mass Flow Comparison

Finally, the UPC PKL Pseudo 3D nodalization was qualified for reproducing the relationship between the CET and the PCT. This nodalization solved instabilities in the simulation of the overheated CET (Fig. 10), obtaining close results in the CETvsPCT curve (Fig. 11). The Pseudo 3D nodalization reproduced the same slope of the experimental data as well as the initial increase of the PCT. This agreement in the initial increase of the PCT is as consequence of reproducing qualitatively the reported delay between both temperatures (Fig. 12).

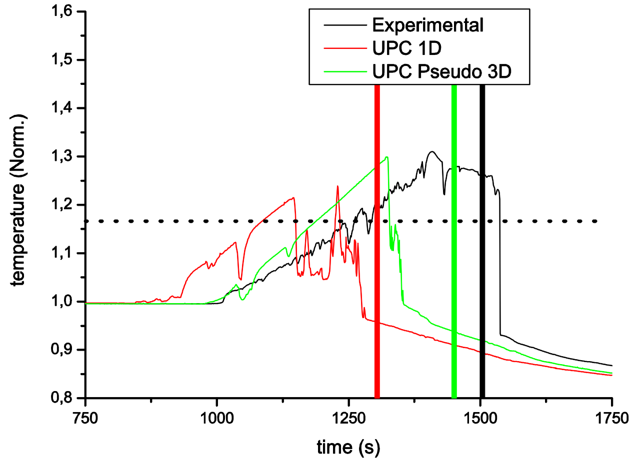

Figure 10 Core Exit Temperature

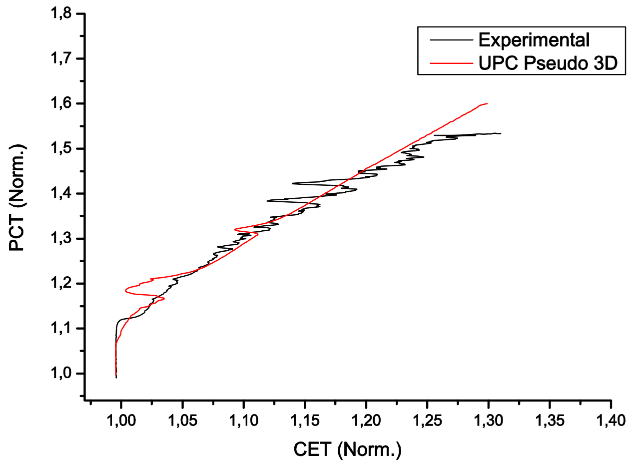

Figure 11 CET vs PCT Correlation

On the other hand, the Pseudo 3D nodalization was not qualified for reproducing closely the core quenching after SG depressurization action. Despite core refilling was simulated, in the calculation quench front achieved the top level of the active core before accumulators' injection, showing a discrepancy with experimental results. In Fig. 10, each CET curve is associated with a vertical line that indicates the time in which the accumulators' injection starts. The comparison shows that for both simulations, the temperatures dropped before accumulators' injection, unlike experimental data, in which it did not occur.

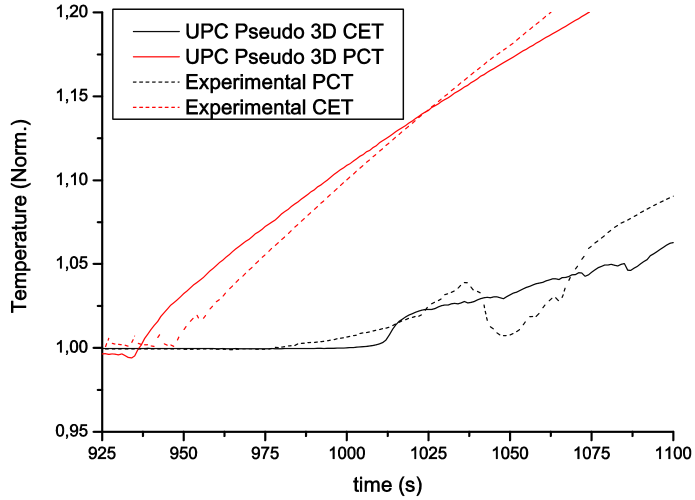

Figure 12 CET and PCT versus time

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

Table 3 Run Statistics

	Transient time (s)	**CPU time (s)**	**Mass error ratio (emass/tmass)**
UPC PKL 1D nodalization	15450.0	2325.66	$7.5097 \cdot 10^{-4}$
UPC PKL Pseudo-3D nodalization	15450.0	2920.2	$5.6616 \cdot 10^{-4}$

6 CONCLUSIONS

The UPC PKL Relap5mod33, that was qualified for the PKL I boron dilution experiments, has been adjusted to the PKL-2 Test G7.1 proving its suitability to simulate the behavior of this transient. Results showed code and nodalization capabilities to reproduce main phenomena of the transient. Otherwise, some limitations were detected for reproducing condensation and core quenching during the fast depressurization induced by the steam generators valve opening.

Regarding core modeling, Pseudo 3D vessel modelling has shown to be a good tool for simulating core dryout and CET vs PCT correlation. On the other hand, some discrepancies have been detected with 1D modelling, as higher vapour generation and instabilities in the core exit temperatures during vapour superheating.

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

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11. ABSTRACT (200 words or less)

The Thermalhydraulic Studies Group of Technical University of Catalonia (UPC) hold a large background in nuclear safety studies in the field of Nuclear Power Plan (NPP) code simulators. This report analyses with RELAP5 mod3.3 the G7.1 PKL Test. 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 Core Exit Temperature (CET) and the Peak Cladding Temperature (PCT). This report analyses the experiment F7.1 of the LSTF and PKL Counterpart Test, which was carried out as 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.

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