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International Agreement Report

Simulation of the LSTF-PKL Counterpart G7.1 Test at PKL Facility Using TRACE 5

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ABSTRACT

The PKL-2 test program is investigating safety issues relevant for current pressurized water reactor (PWR) plants as well as for new PWR design concepts and will focus on complex heat transfer mechanisms in the steam generators and boron precipitation processes under postulated accident situations.

These issues are being investigated by means of thermal-hydraulic experiments that are conducted at the Primärkreislauf-Versuchsanlage (primary coolant loop test facility) PKL. This facility is owned and operated by AREVA NP and is situated in Erlangen, Germany.

AREVA NP has, for a number of years, conducted valuable experiments on reactor thermal-hydraulics in the PKL facility, including earlier experiments carried out in the framework of the SETH project (2001-2003) and of the PKL-1 Project (2004-2007).

The first category includes tests addressing the heat transfer mechanisms in the steam generators in the presence of nitrogen, steam and water, in both vertical and horizontal steam generators. Cooldown procedures in the case where steam generators have partly dried out on the secondary side have also been covered. A further topic addresses the heat transfer in the steam generators under reflux condenser conditions (e.g. fast secondary side depressurization). Fast cooldown transients (with water filled reactor coolant system) such as main steam line break, completed by tests on mixing of hot and cold water in the RPV downcomer and the lower plenum are also considered in the test program. Further investigation address boron precipitation processes in the core following large break loss of coolant accidents. G7: Counterpart Test with ROSA / LSTF on small break LOCA with Accident Management procedures (one test performed in July 2011).

This paper focuses on the simulation, using the best estimate code TRACE, of the experiment G7.1 conducted at the PKL facility. The OECD-PKL2 test G7.1 is a hot leg Small Break-Loss of Coolant Accident (SB-LOCA) with a total failure of the High-Pressure Safety Injection (HPSI). The PKL test facility simulates a KWU 1300 MWe pressurized water reactor with all elevations scaled 1:1 and with volume and power scaled by a factor of 1:145. This test had been previously performed in ROSA/LSTF facility in order to compare results between facilities and to analyze different plant configurations and scaling effect.

The postulated additional system failures (no HPSI, no automatic secondary-side cooldown) return a course of events that necessitates Action Management (AM) measures to prevent core-melt scenario. A fast secondary-side depressurization initiated after occurrence of core uncovery was employed, as AM measure for restoration of the secondary side heat sink aiming for a fast reduction of the primary pressure. The reduction of the primary pressure down to ACC injection pressure then effectuates the transition to the low-pressure phase with the Low-Pressure Safety Injection (LPSI) active.

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/FAP1 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.

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).

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, 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 PKL2/OECD and ROSA2/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

The PKL-2 test program is one of the projects aimed by the Nuclear Energy Agency focused on analyzing safety issues of pressurized water reactor plants. In particular, in this project complex heat transfer mechanisms in the steam generators and boron precipitation processes are studied. Both issues are investigated by means of thermal-hydraulic experiments, conducted at the Primärkreislauf-Versuchsanlage (primary coolant loop test facility) PKL.

Among the experiments performed at PKL, G7.1 was also conducted at Rig of Safety (ROSA) facility with the objective of performing a counterpart test of the results obtained at both facilities in order to compare and observe if the results are coherent or not between these two different facilities. G7.1 experiment consists of a hot leg small break LOCA with additional failure of safety systems, as the high-pressure injection system. Such situation makes necessary an adequate accident mitigation procedures to prevent the accident would lead to core damage. The efficiency of the accident mitigation measures postulated are analyzed and thereby safety margins are explored. In addition, an assessment of the performance of the Core Exit Temperature (CET) is performed, which is used as criterion for the initiation of accident mitigation measures involving emergency operating procedures and/or severe accident management measures.

One of the main objectives of this experiment is focused on analyzing the relation between the core exit temperature (CET) and peak cladding temperature (PCT) as the CET is the value used to initiate the accident mitigating measures to assure PCT will not violate the safety margin.

To better understand the thermal-hydraulic processes that take place in G7.1 experiment has been simulated using best estimate codes such as RELAP-5 or TRACE. In this document, a model of PKL facility for TRACE thermal-hydraulic code, that makes use of two VESSEL hydraulic components to model the PKL reactor pressure vessel, is proposed. Using the PKL model built, the experiment G7.1 is simulated and, in general, the results obtained agree with the experimental results.

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, and thank the Spanish Nuclear Regulatory Body (CSN) for the technical and financial support under the agreement STN/1388/05/748.

ABBREVIATIONS

ACC	Accumulator
AM	Accident Management
C1D	Courant number 1D
C3D	Courant number 3D
CAMP	Code Assessment and Management Program
CET	Core Exit Temperature
CL	Cold Leg
CPU	Execution time (s)
CSN	Consejo de Seguridad Nuclear (Spanish nuclear regulatory commission)
CT	Cladding Temperature
DC	Downcomer
DT	Total number of time steps
ECCS	Emergency Core Cooling System
HL	Hot Leg
HPSI	High Pressure Safety Injection
JAERI	Japan Atomic Energy Research Institute
LBLOCA	Large Break Loss of Coolant Analysis
LPSI	Low Pressure Safety injection
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NEA	Nuclear Energy Agency
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
RCL	Reactor Coolant Line
RCS	Reactor Coolant System
RELAP	Reactor Excursion and Leak Analysis Program
RHR	Residual Heat Removal System
RT	Transient time (s)
SB-LOCA	Small Break Loss of Coolant Analysis
SG	Steam Generator
SOT	Start Of the Test
SSD	Secondary Side Depressurization
TRACE	TRAC/RELAP Advanced Computational Engine
TH	Thermal-hydraulic
TS	Maximum time step (s)
UNESA	Asociación Española de la Industria Eléctrica

1 INTRODUCTION

Since the beginning of nuclear power plants operation, safety is one of the major fields of study in the development and implementation of nuclear energy. In this field, simulations of accidental sequences play an important role to improve the knowledge about the physical phenomena that take place inside the reactor during a certain transient, and analyze the effectiveness of the emergency systems to guarantee a safe plant situation. Such simulations can be performed using best estimate thermal-hydraulic codes, as RELAP-5, TRAC, CATHARE, ATHLET or TRACE. Among these codes, RELAP-5 and TRAC have traditionally been used to reproduce transients of Pressurized Water Reactors (PWR) and Boiling Water Reactors, respectively. Nowadays, TRACE code (TRAC/RELAP Advanced Computational Engine) is being developed to make use of the more favorable characteristics of RELAP-5 and TRAC codes to simulate both, PWR and BWR, technologies.

Experimental facilities are essential to develop and improve the models implemented in the thermal-hydraulic codes. Thus, the data collected in the experiments is of great importance in the assessment of the capabilities of thermal-hydraulic codes to reproduce the different physical phenomena that may take place inside the reactor in accidental situations. PKL is a test facility located in Germany that represents a typical PWR western design. The PKL test facility simulates a KWU 1300 MWe pressurized-water reactor with all elevations scaled 1:1 and with volume and power scaled by a factor of 1:145. Several experimental programs have been conducted at PKL facility. Thus, programs PKL-1 and PKL-2 were focused on the study of Large Break Loss of Coolant Accidents (LBLOCAs) and Small Break Loss of Coolant Accidents (SBLOCAs) with the objective of best estimate codes test and validation. PKL-3 program started with the objective of studying different transients with and without LOCAs [Ref. 1, Ref. 2, Ref. 3]. The PKL tests results have been used for preparation and verification of procedures described in the operating manuals, for answering questions exposed by regulatory bodies and to perform code calculations assessment.

In particular, in the frame of the OECD/NEA CSNI PKL-2 project, counterpart test between Rig of Safety (ROSA) and PKL test facilities were developed. Thus, a small break loss of coolant in a hot leg (SB-LOCA) with additional failure of safety systems was conducted at ROSA and at PKL to compare and analyze the results obtained in both installations. At PKL facility this transient was named as G7.1 experiment

Thus, when a SB-LOCA occurs the water inventory and pressure of the reactor coolant system decrease and this leads to empty the reactor pressure vessel, and to core uncovering. Therefore, it is necessary the actuation of the safety systems to inject water in the primary circuit, High Pressure Injection System (HPIS) in this case, to maintain the core full of water and cooled. The safety system failures postulated are no HPSI injection and no automatic secondary-side cooldown. This situation leads to core uncovering and the clad temperature increases until core-melt scenario if no action is performed [Ref. 1, Ref. 4]. Therefore, it is necessary to explore the Accident Management (AM) measures necessary to prevent this scenario.

As AM measures proposed to prevent core melting is a fast secondary-side depressurization, initiated after core uncover to re-establish the steam generators secondary side as heat sink aiming for a fast reduction of the primary pressure [Ref. 5]. The reduction of the primary pressure permits the injection through the accumulators and makes possible the Low-Pressure Safety Injection (LPSI) activation.

The efficiency of the accident mitigation measures proposed are analyzed and thereby the safety margins are explored. In addition, an assessment of the performance of the Core Exit Temperature (CET), used as criterion for the initiation of accident mitigation measures involving emergency operating procedures and/or severe accident management measures, is performed [Ref. 6].

This document focuses on the simulation, using the best estimate code TRACE 5, of the experiment G7.1 conducted at the PKL facility. The analysis of the results shows that the most important physical phenomena presented in PKL-G7.1 experiment are well reproduced by TRACE model [Ref. 7, Ref. 8, Ref. 9].

The rest of the work is organized as follows: The PKL facility is briefly described in Section 2. Section 3 is devoted to introduce the experiment G7.1; Section 4 is devoted to describe the TRACE 5 model for the PKL facility used to simulate the experiment. In Section 5, the main results obtained from simulation are presented and compared with the experimental data. Finally, the run statistics are presented in Section 6 and the main conclusions of this work are summarized in Section 7.

2 PKL FACILITY DESCRIPTION

The PKL test facility represents a typical western design PWR with a volume and power scale of 1:145, while all the components height on the primary and secondary side correspond to real plant dimensions. It models the entire primary system and the relevant parts of the secondary side. In order to investigate the influence of non-symmetrical boundary conditions on the system behavior, PKL facility is equipped with four primary loops symmetrically arranged around the reactor pressurized vessel. Each loop contains a reactor coolant pump and a steam generator. This facility simulates a maximum of 10% of the nominal power scaled and works with a maximum pressure of 45 bars [Ref. 1,Ref. 10].

The facility also models all the important safety and auxiliary systems as eight accumulators, one in each of the hot legs and one in each of the cold legs, four independent injections from the high and low pressure injection system, the residual heat removal system and the pressure control in the pressurizer. Figure 1 shows an overview of PKL test facility [Ref. 10].

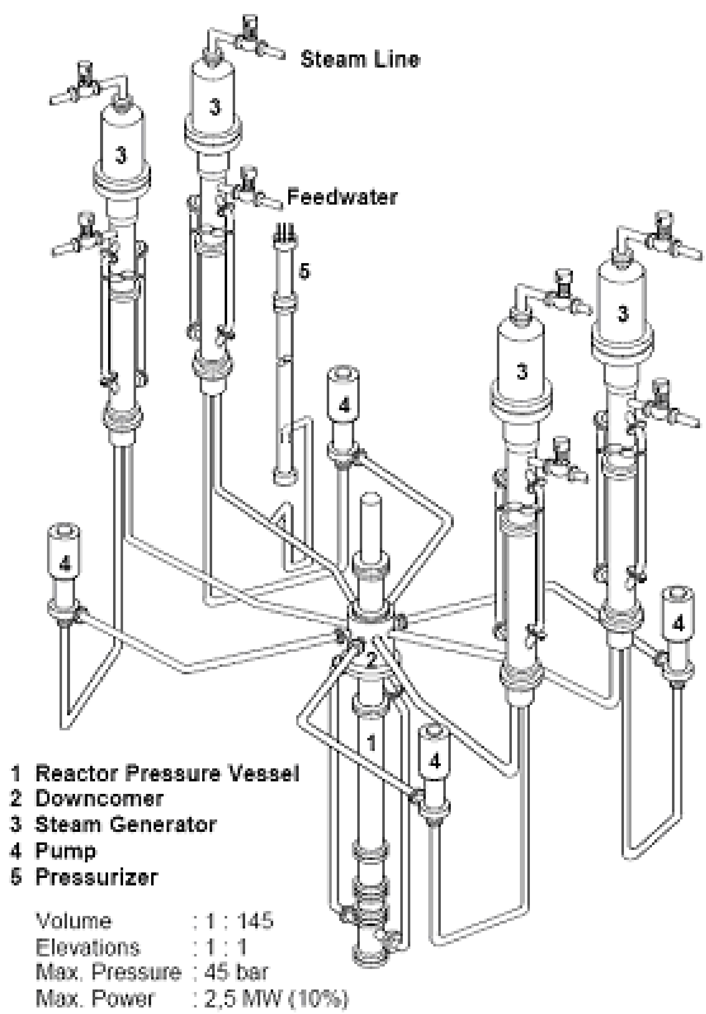

Figure 1 PKL Facility

3 DESCRIPTION OF G7.1 INITIAL AND TRANSIENT CONDITIONS

G7.1 transient consists of a small break LOCA in the hot leg of the primary side followed by a total failure of the high-pressure injection system (HPIS) together with the failure of the automatic steam generator secondary-side cooldown. In such conditions, the accident mitigation procedure proposed is the manual depressurization of the secondary side steam generators, followed by the injection from the accumulators. Test G7.1 was designed to investigate the core heat-up sequence and the effectiveness of depressurization proposed to permit the actuation of ACCs and LPIS.

Under these boundary conditions (small break LOCA in the hot leg of the primary side), the SB-LOCA is countered by safety injection systems being triggered as soon as the setpoints of the different systems are met, being HPIS the first system being activated. In case of a total failure of HPIS a heat up of the core is produced and, depending on the break size, the primary pressure evolution would not permit the Accumulators and LPSI actuation, as the pressure set point will never be reached. In this situation for the most PWR designs, the design-basis course of events foresees a parallel cooldown of the primary side via the steam generators secondary side to reach cold shutdown conditions. This cooldown process may be initiated automatically or manually and may be performed either partially or completely. In G7.1 experiment a secondary side depressurization is proposed as AM to reduce the primary pressure until appropriated values to allow ACCs and LPIS safety injections. The signal to trigger the depressurization procedure is the core exit temperature (CET). Thus, when this variable reaches 623 K secondary side depressurization starts.

This experiment has been conducted in two different facilities, ROSA and PKL, in order to study the relevant phenomena occurring during the transient evolution and the possible scaling effects. As the facilities are different a conditioning phase before the start of the transient is needed to obtain similar boundary conditions. Figure 2 shows the evolution at PKL facility of primary pressure, temperature and primary side coolant inventory for the conditioning and test phases [Ref. 11].

The most important initial and boundary conditions considered to simulate the G7.1 experiment are the following:

- Primary circuit pressured at 45 bars, which is the maximum pressure of the facility. The coolant level in the primary circuit is at midloop.
- Power generation of 565 kW, equivalent to 1.8% of the residual power including the compensation for heat losses.
- The initiating event is a break of 1.5% upwards in hot leg of loop 1.
- The steam generators secondary side are filled with water, with a level of 7.7 m, and in operation. The secondary side pressure at the beginning of the transient is 43.7 bar.
- All steam generators are connected via the main steam header.
- A total failure of the HPIS is produced.
- No automatically initiated secondary-side cooldown is considered.
- Steam generators secondary side depressurization is considered as AM measure, when CET ≥ 623 K (350 °C) what makes possible:
    - Cold-leg Accumulators injection at p = 26 bar
    - Cold-leg Low Pressure injection at p = 8 bar

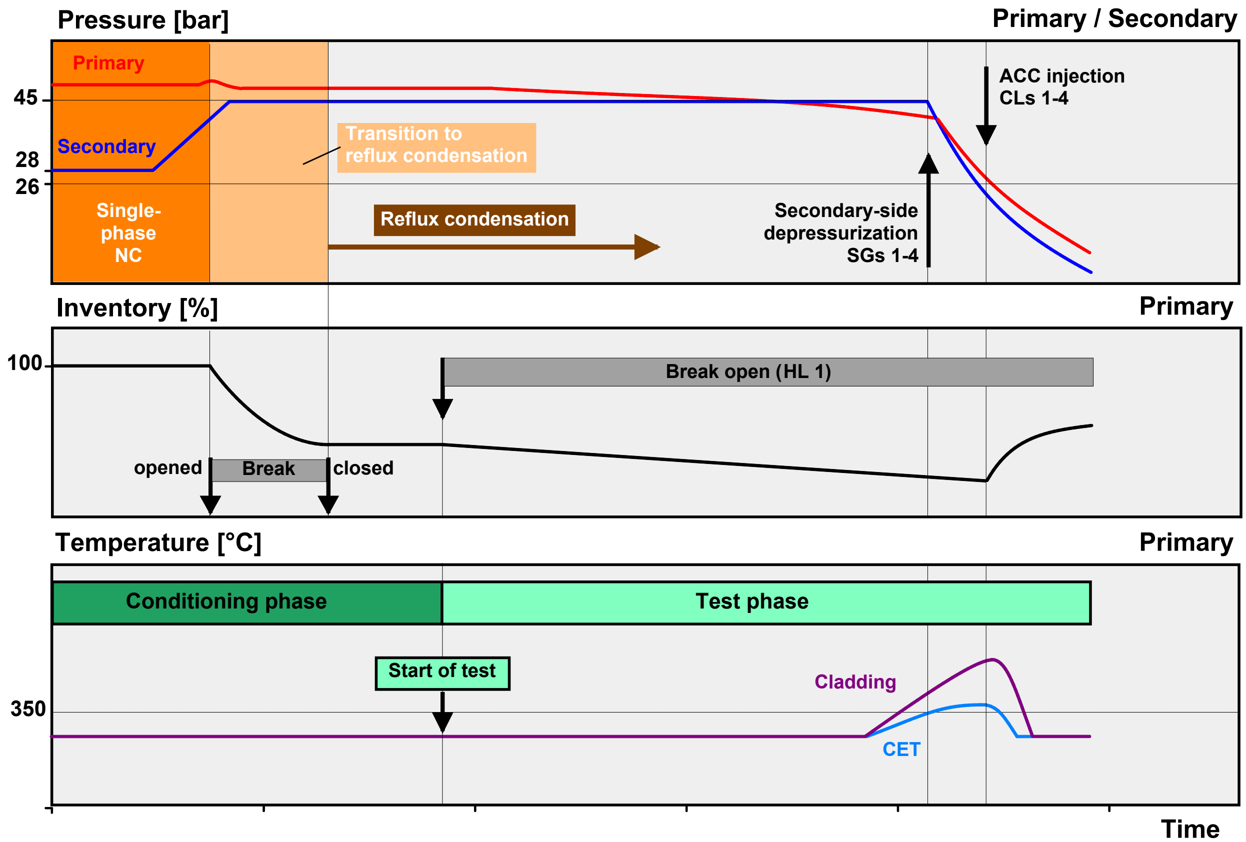

Figure 2 Primary Pressure, Temperature and Inventory for the Conditioning and Test Phases

The initial conditions at the beginning of the transient are shown in Table 1. Departing from the conditions exposed in Table 1, the development of the transient G7.1 consists of different phases. The transient begins when the break in hot leg is produced and the primary circuit empties. Once the temperature at the core exit (CET) reaches 623 K (350 ºC) secondary side depressurization, through two main steam valves, starts. As all four SG are connected, the secondary side depressurization is homogenous for all SG, and permits to reduce the primary pressure to reach the accumulators pressure set point, so this system injects water in the reactor coolant line and, later on, the primary pressure reaches LPIS activation set point and coolant injection is produced. These actions lead the plant to a safe condition to the end of the transient. Table 2 presents a summary of milestones.

Table 1 G7.1 Initial Conditions

Test Conditions	***PKL-3 G7.1 OECD-PKL***
PRIMARY	
Flow Pattern	Reflux-condenser conditions
Coolant inventory (kg)	2250 kg
Heater rod bundle power (decay heat)	455kW + 110kW (compensation for heat losses)
Primary pressure	4.5 MPa
Core outlet temperature	530 K
Subcooling at core outlet	0 K
Pressurizer level	0.8 m
Accumulators	Liquid level: 1.62 m N2 Volume: 0.099 m3 Temperature: ACCs 1/4 306 K; ACCs 2/3 304 K Pressure: 2.66 MPa
SECONDARY	
Secondary side	Filled with water and in operation
Main Steam pressure	4.37 MPa
Secondary temperature	529 K
Level	7.7 m
Feed water temperature	346 K

Table 2 Development of the Transient G7.1 Based on Experimental Data at PKL

Time (s)	**Milestones**
0	Start of the transient: • Break in hot leg 1
1360	Secondary Side depressurization at CET~ 623 K;
1500	Accumulator injections at p=26.6 bar.
2060	LPIS injection at p=7.7 bar
5685	End of the transient

4 PKL FACILITY TRACE 5 MODEL DESCRIPTION

The transient simulation has been performed using TRACE 5 V2 code [Ref. 12] and SNAP [Ref. 13]. The TRACE 5 model used consists of 136 hydraulic volumes, including two 3D VESSEL components, and 35 heat structures. This model has been adapted from the RELAP PKL model provided by the facility [Ref. 10, Ref. 14]. Figure 3 and Figure 4 outline the nodalization used in the simulation.

PKL reactor pressure vessel, Figure 1, has been modelled as a combination of two VESSEL hydraulic components: core vessel and downcomer vessel, as shown in Figure 3. Using this nodalization the model built is closer to the real PKL reactor pressure vessel. The core vessel models the lower and upper plenums, the core, and the head of the PKL reactor pressure vessel, while the downcomer vessel represents the part of the reactor pressure vessel downcomer and upper head bypasses.

The core vessel has been divided in 19 axial levels, 1 radial rings and 2 azimuthal sectors. In the axial direction, the active core is modeled by 7 levels (6 to 12), and the hot legs are connected at level 15 in radial direction.

The downcomer vessel has been divided in 5 axial levels, 3 radial levels and 2 azimuthal sectors. The head bypasses are also modeled connecting both VESSEL components, as shown in Figure 3. For both vessels, each axial level, volume and effective flow area fractions have been set according to PKL technical specifications provided by the organization [Ref. 10].

The four primary loops are modelled with a pump and a steam generator in each loop using PIPE and PUMP components. The U-tubes of the steam generators are lumped into three PIPE components of different heights. The heat transfer between the primary and secondary systems is simulated using three heat structures, one for each of the three pipes that simulate the steam generators U-tubes.

The injections from the accumulators performed at the transient (see Figure 2) have been simulated using a PIPE component connected to the loop by a VALVE. The locations of the injections from the accumulators are shown in Figure 3. A LPIS injection has been simulated in each cold leg (see Figure 3). The model of these injections has been simplified by using FILL components connected with the loops. Finally, the SB-LOCA has been simulated with a BREAK component connected to the hot leg 1.

Figure 4 shows the nodalization of the steam generators secondary side, which are connected at the exit with different breaks through their corresponding valves to simulate depressurization control considered as accident mitigation measure. This depressurization is produced by the relieve valves (RV). In addition, it has been simulated the main steam valves (MSV) connected to the turbine.

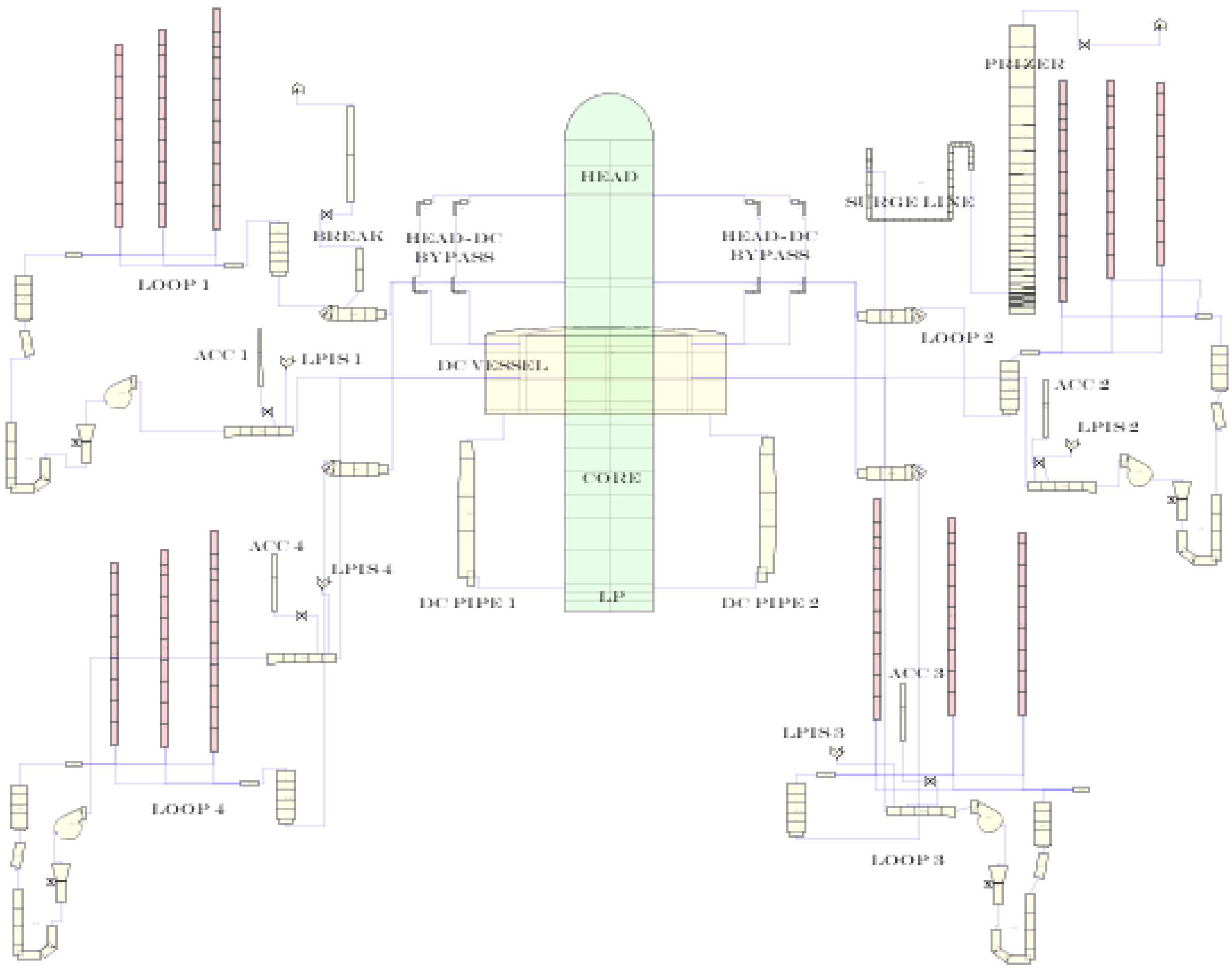

Figure 3 TRACE5 Nodalization of PKL Facility. Primary Side

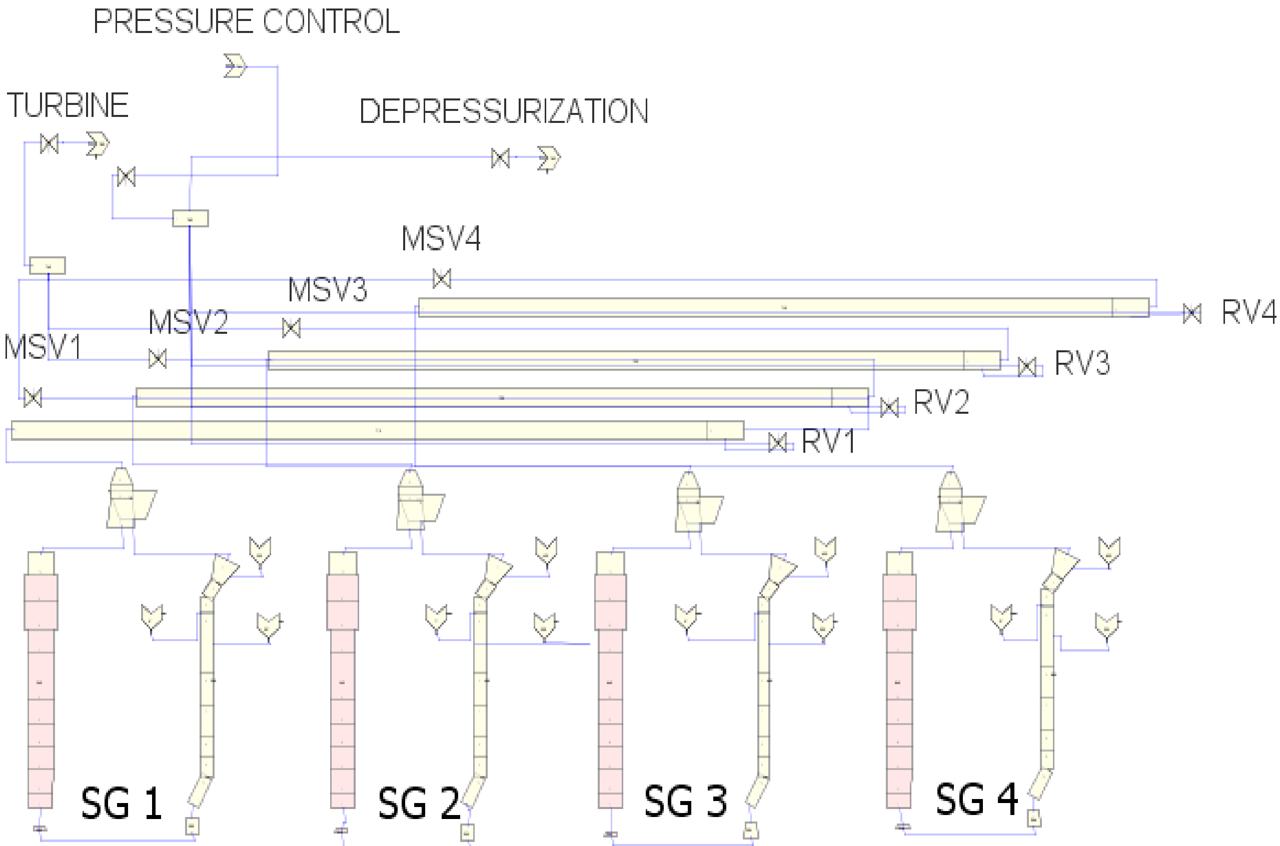

Figure 4   TRACE5 Nodalization of PKL Facility. Secondary Side

5 SMALL BREAK LOCA IN HOT LEG SIMULATION RESULTS

Figure 5 shows the main milestones during the transient evolution.

- t=500 s. Small Break LOCA in hot leg.
- t=1860 s. CET achieves 350 ºC

- t = 1860 s. Secondary side depressurization starts
- t = 2000 s. Primary Pressure =26 bars

- t = 2000 s. Injection from accumulators
- t = 2560 s. Primary Pressure = 7.7 bars

- t = 2560 s Injection from LPIS
- t = 3000 s end of the transient

Figure 5 Main Milestones During the Transient Evolution

Once the small break opens, the coolant flows out through the break what reduces the water inventory in the core. The inventory reduction and the heat generated in the core leads to core uncovering and the primary coolant starts to heat up. The void formation in the core produces a rise in the primary side temperatures, until the core exit temperature (CET) reaches 623K (≈1860s). At this moment, the fast depressurization on secondary side, which is established as an accident management measure, starts, reducing the primary pressure value until the accumulators (ACC) pressure set point is achieved (≈2000s). Such injection stops when 1MPa is reached in cold leg.

After the injection from the accumulators, the only heat sink available is the steam discharge through the break, until primary pressure reaches 0.77 MPa (≈2400s). With this condition, LPIS is activated injecting in all four cold legs. Table 3 shows the timing of the main milestones at the experiment and in the TRACE calculation. In Table 3, it can be observed that TRACE prediction is slightly advanced but close to the experimental data.

The agreement between the experimental data and TRACE calculations is also observed in the evolution of the main plant variables. Thus, Figure 6 shows the evolution of the primary and secondary side pressures. The pressure keeps relatively constant in primary and secondary side until the core is uncovered. After the occurrence of a partial core uncover the primary pressure drops below the secondary pressure, since following the core uncover not all the heat from the core leads to formation of steam but also to heat-up of the uncovered structures (increase in cladding temperatures). TRACE calculated primary pressure is slightly lower than experimental primary pressure during the period between 1600 s until the start of the depressurization. In addition, there is an advance of 50 s respect to experimental data in the depressurization. The accumulator injection begins when the primary pressure is lower than 2.6 MPa at 2000 s, there is only 80 seconds of difference between experimental measurements and TRACE calculation as shown in Table 3.

Table 3   Chronological Events for G7.1 Test

Event	*PKL*	
	Experiment	*TRACE 5*
Break valve open	500	500
CET>623K Depressurization	1860	1780
Accumulator Injection System starts (Primary pressure < 2.6 MPa)	2000	1920
Terminate Accumulator Injection (Primary pressure < 1 MPa)	2360	2226
Low pressure injection starts (Primary pressure < 0.77 MPa)	2560	2436

In Figure 7 we can observe the evolution of the core exit temperature and the peak cladding temperature in the transient. The temperatures remain constant until the core uncover. When the beginning of core uncover for t > 1440 s a continuous rise of the clad temperature is observed. When CET is greater than 623 K (1780 s approximately), the depressurization at the secondary side of steam generators occurs, simultaneously with the activation of the auxiliary water mass flow rate of 0.18 kg/s in each steam generator (simulated as a component FILL). Up to the point in time with the initiation of the secondary side depressurization the cladding temperature has raised 275K approximately. In the experiment, a delay of 50 s. is observed between the raise of the CET and PCT values. However, in TRACE calculations both variables began to increase at the same time, as shown in Figure 7.

The maximum peak cladding temperature (PCT) is well simulated in time. PCT and CET are well reproduced, although the CET is slightly higher in TRACE calculation than the experimental data and, as mentioned above, it is advanced 50 s. When secondary side depressurization is initiated, the CET is about 160 K below the maximum measured cladding temperature in both cases (experimental and calculated).

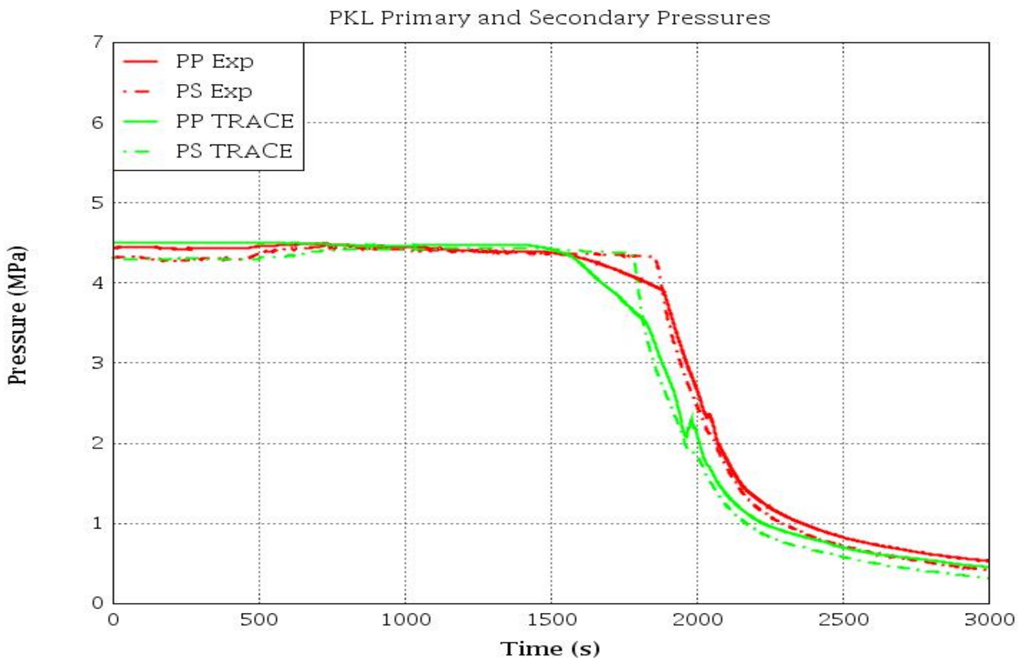

Figure 6   Pressures in the Primary and the Secondary Side

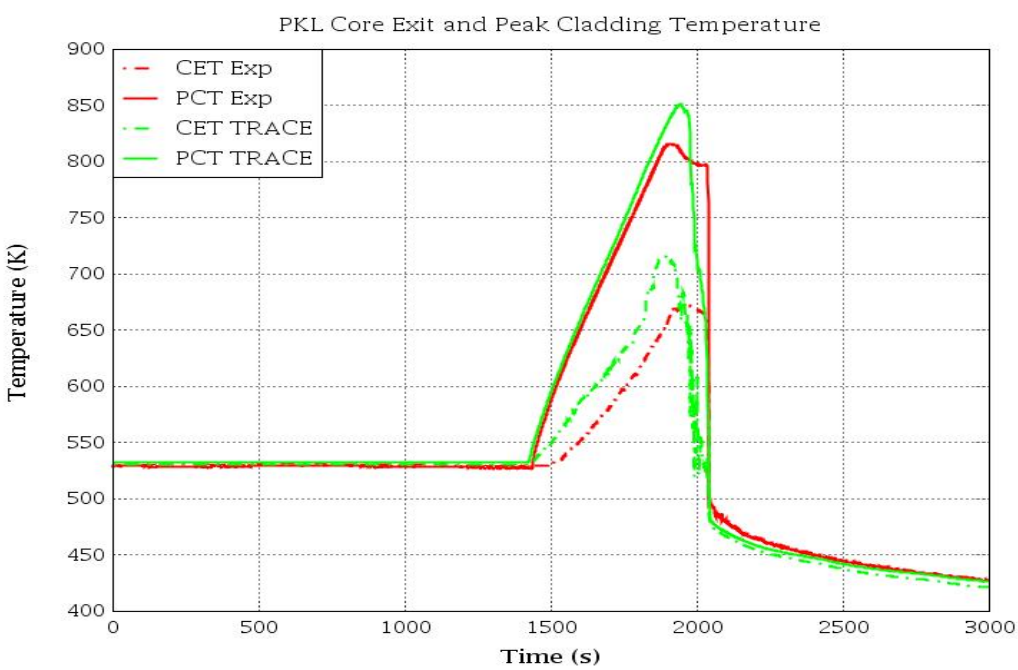

Figure 7 Core Exit and Peak Cladding Temperature

In Figure 8, the break mass flow rate is reproduced. At the start of the transient, as the hot-side coolant swell level reaches into the SG inlet chambers, the break flow is a saturated 2-phase coolant/steam mixture that quickly changes to only steam discharge as soon as the swell level decreases down the hot legs. The TRACE results for the mixture agree very well with the experimental ones.

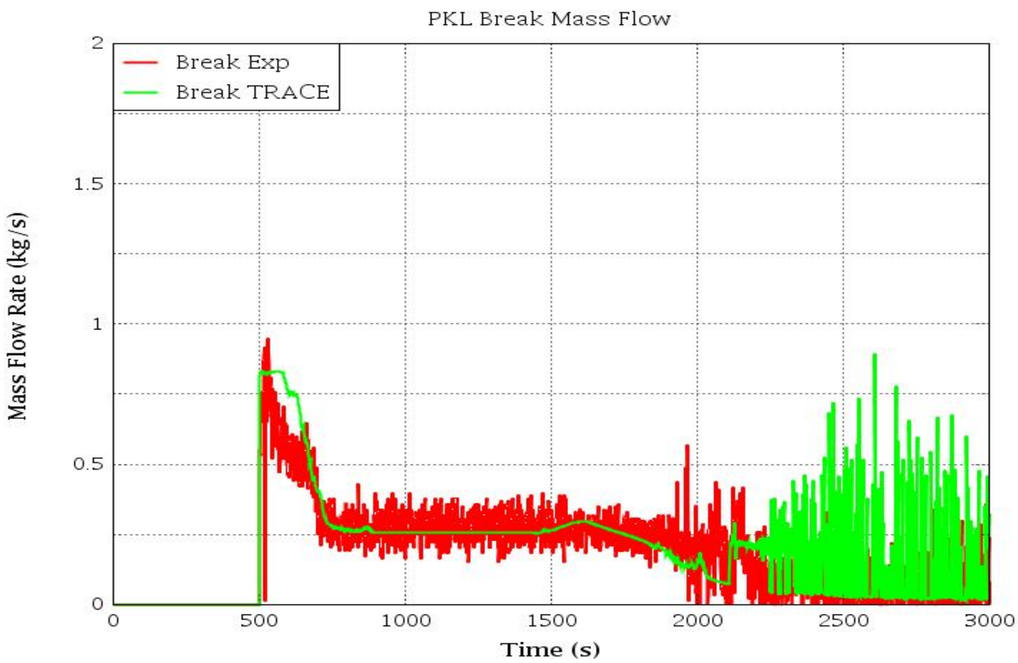

Figure 8   Break Mass Flow Rate

Figure 9 shows the injection of the accumulators (the sum of the all four ACC). These injections start when the primary pressure is lower than 2.6 MPa, and end when the primary pressure is lower than 1 MPa. It can be observed that TRACE results are advanced in time respect to the experimental data, but the total amount of liquid discharged is practically the same. It is worth pointing out that the flooding and the restoration of the core cooling is possible by means of the accumulators injection. Thus, Figure 11 presents the core level evolution that increases at 2000 s due to the accumulators injections.

After the stop of the accumulators injection, the system comes into a phase where the primary inventory and pressure present a slow decrease until LPIS injection, as observed in Figure 11. The LPIS starts when the primary pressure is below 0.77 MPa. The LPIS at each loop is shown in Figure 10 which shows as TRACE simulation is very similar to the observed in the experiment.

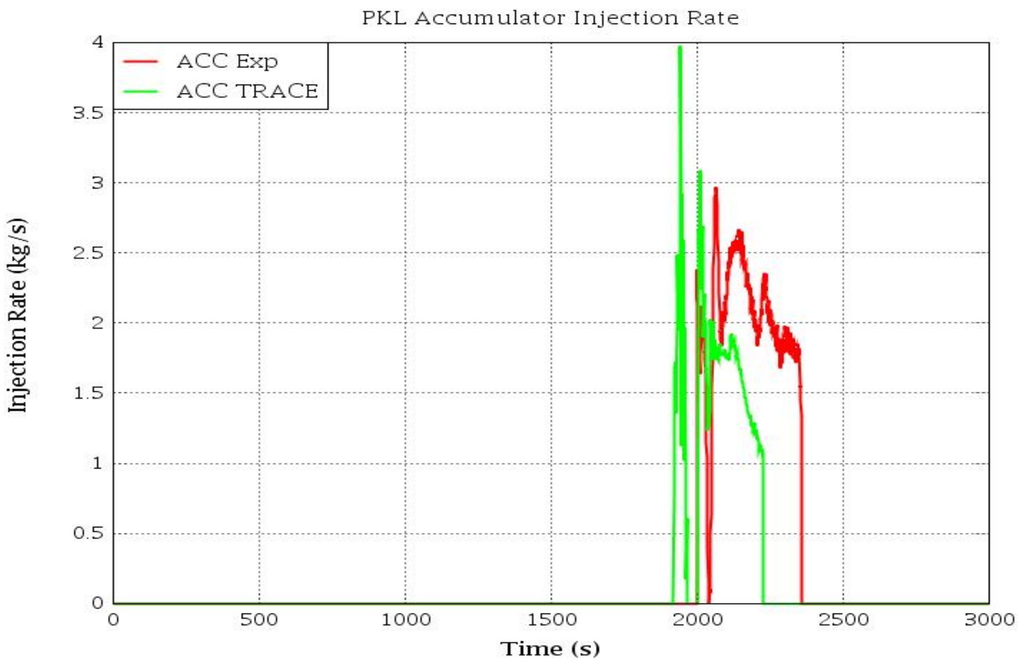

Figure 9 Accumulator Injection Rate

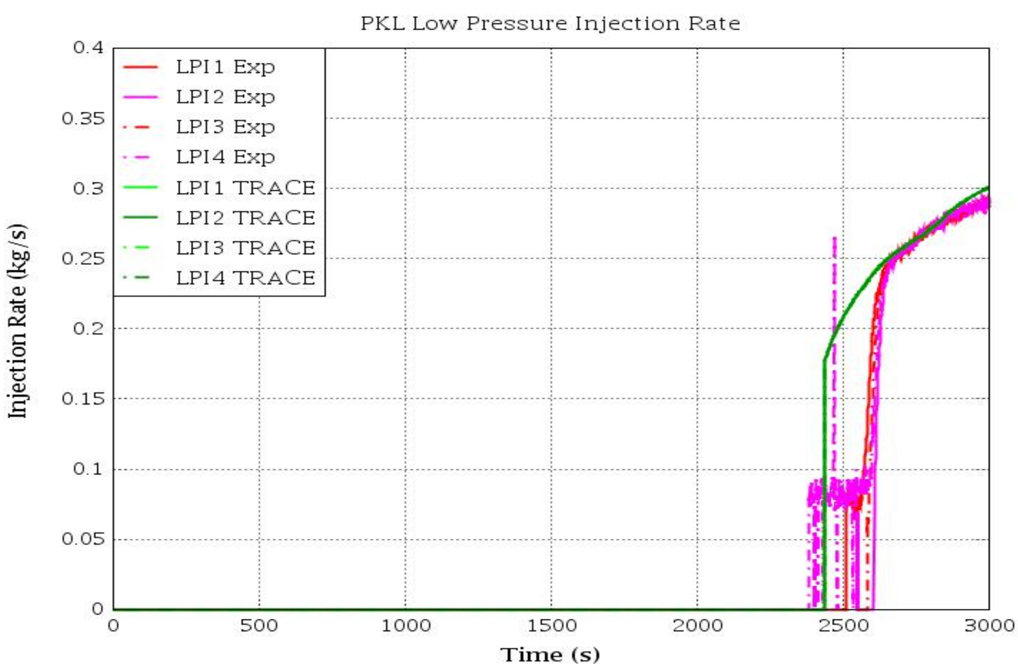

Figure 10 LPSI Rate

By activation of the LPIS the reactor coolant system (RCS) was filled (Figure 11) and achieves subcooled state at end of the transient (3000 s) thanks to the temperature of the water injected. Figure 11 and Figure 12 present the level of the reactor core and the level of the upper plenum of the reactor pressure vessel. The accumulator injections lead to condensation effects in the cold legs. By the condensation effects, a temporal displacement of water from the core to the downcomer happens (2000s).

Then, both the reactor coolant from the loop seals and the injected ACC water rewets the entire core region. In general, a good reproduction of liquid levels (core, upper plenum and downcomer) respect to experimental measurements is observed.

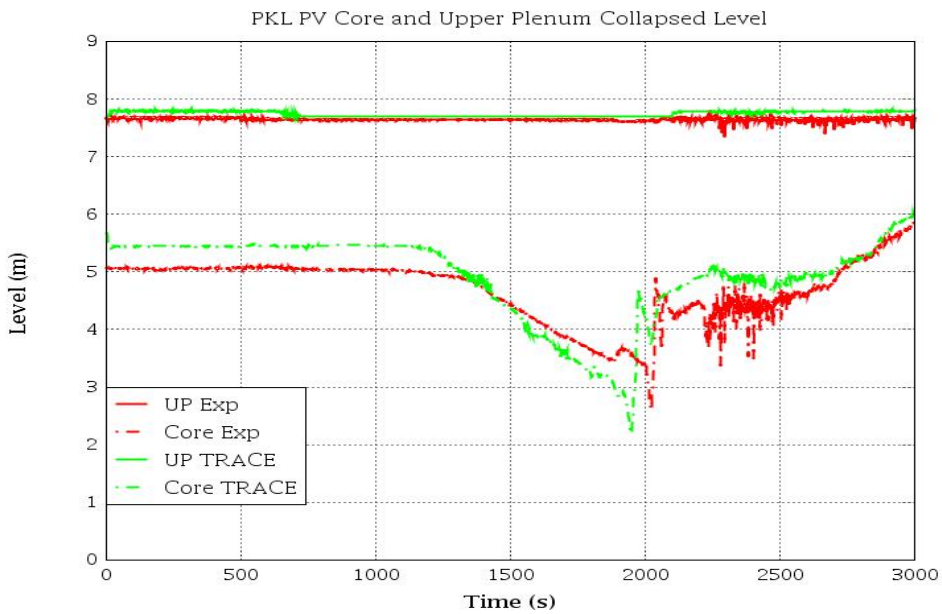

Figure 11 Core and Upper Plenum Collapsed Levels

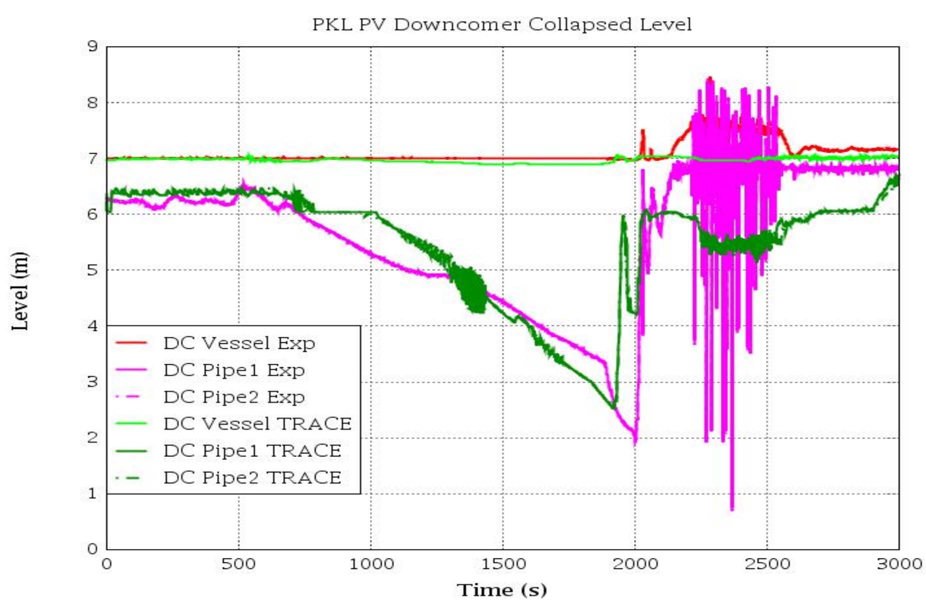

Figure 12 RPV Downcomer Collapsed Levels

The hot leg levels and the cold leg levels are presented in Figure 13 to Figure 16. In those figures the effect of the accumulators injection in all the loops are observed. There are some discrepancies in the hot legs, especially in hot leg 1 (the broken leg). Regarding to the cold legs liquid levels, all tendencies are well simulated with TRACE5. Cold legs remain empty until the entrance of water coming from the accumulators (at 2000 s).

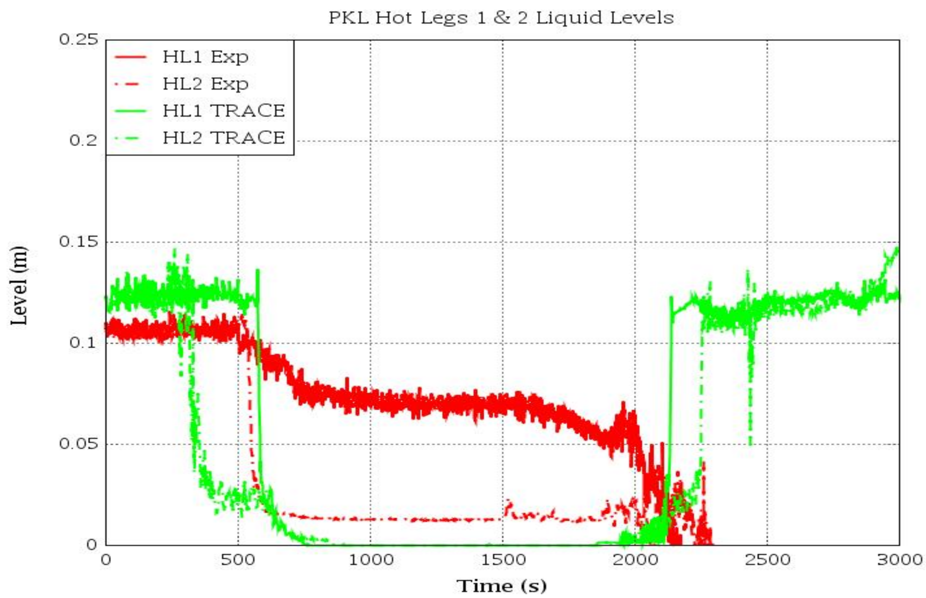

Figure 13 Hot Legs 1 and 2 Collapsed Levels

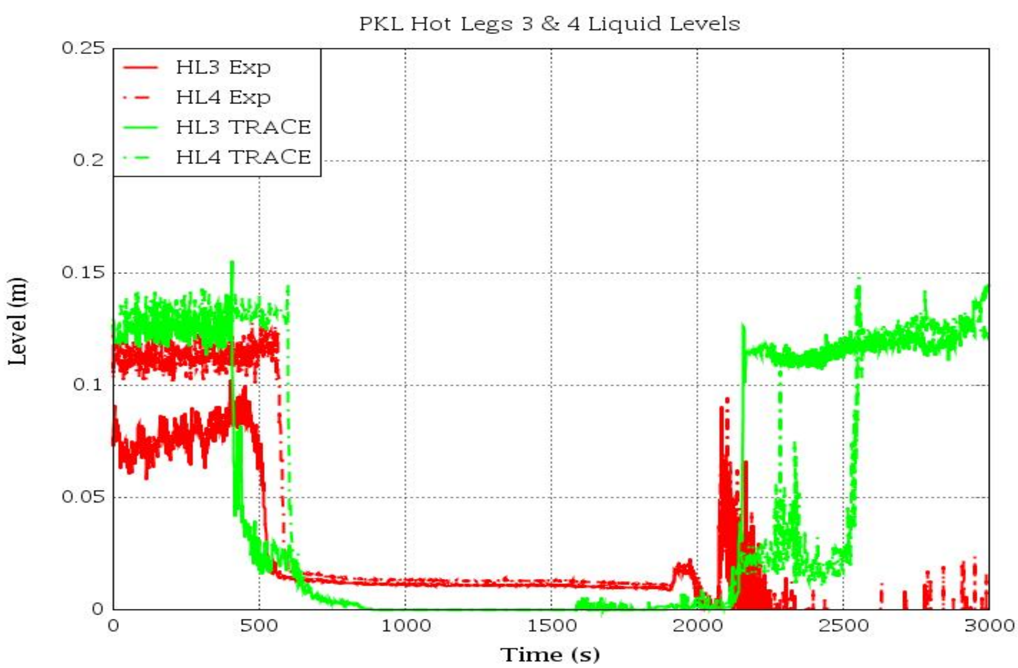

Figure 14 Hot Legs 3 and 4 Collapsed Levels

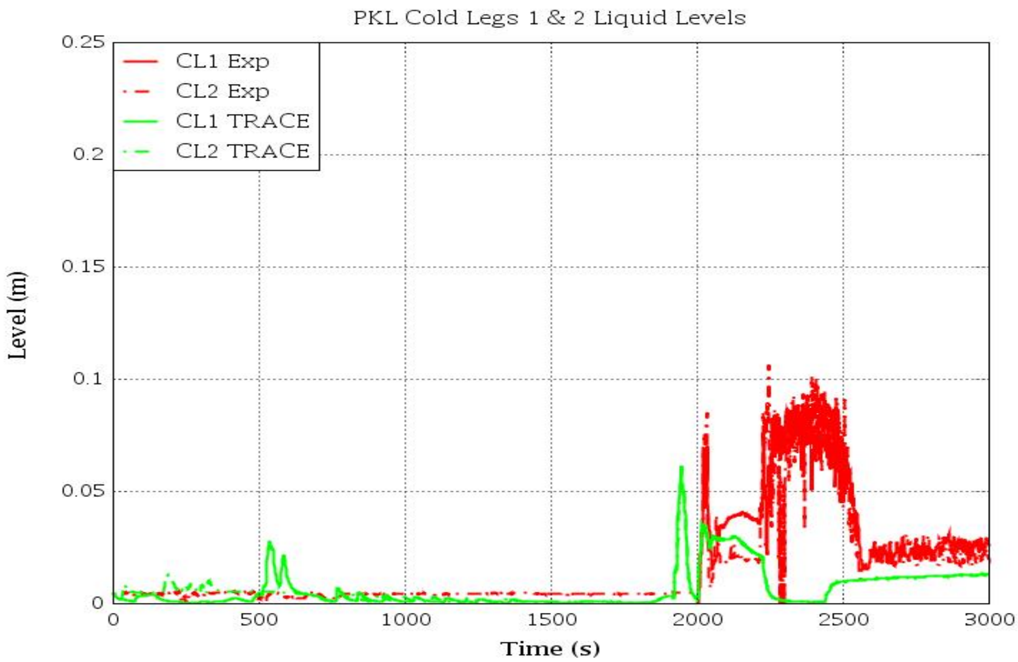

Figure 15   Cold Legs 1 and 2 Collapsed Levels

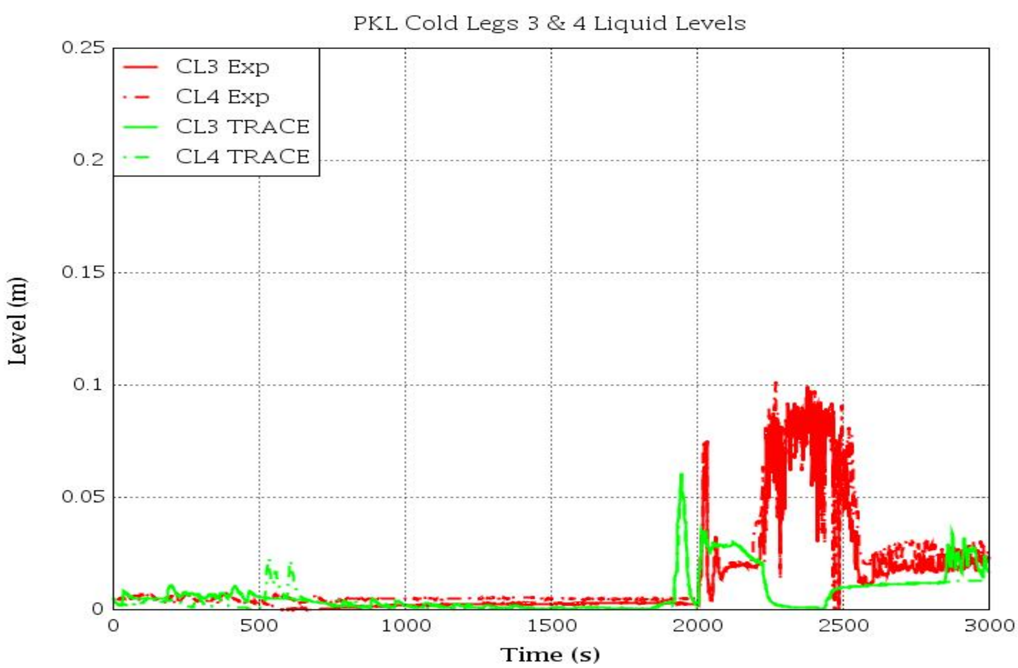

Figure 16   Cold Legs 3 and 4 Collapsed Levels

Finally, in Figure 17, Figure 18 and Figure 19 the relationship between the CET and PCT is analyzed. This relation is very important due to the AM is produced via the CET measure, and the objective of the AM (depressurization at secondary side) is preventing core damage, which is evaluated with the PCT. Figure 17 shows the difference between PCT and CET, Figure 18 shows the quotient between PCT and CET, and Figure 19 shows the PCT vs the CET values.

The results simulated with TRACE5 predict well the experiment relations. In Figure 17, it can be observed that the maximum difference between PCT and CET regarding to the rise of temperatures is about 160 K for experimental data and about 130 K for TRACE results. Thus, in Figure 18, the maximum value of the quotient between PCT and CET is 1.25 and 1.2 for experimental and TRACE results respectively. So, the TRACE results present lower differences between the two measures, what had been observed in Figure 7 too, where for an instant determined, the PCT is very well estimated but the CET is over predicted in comparison with the experimental result.

Finally, in Figure 19 we can observe the values of PCT for each CET values during the rise of temperature. TRACE results underpredict the experimental ones.

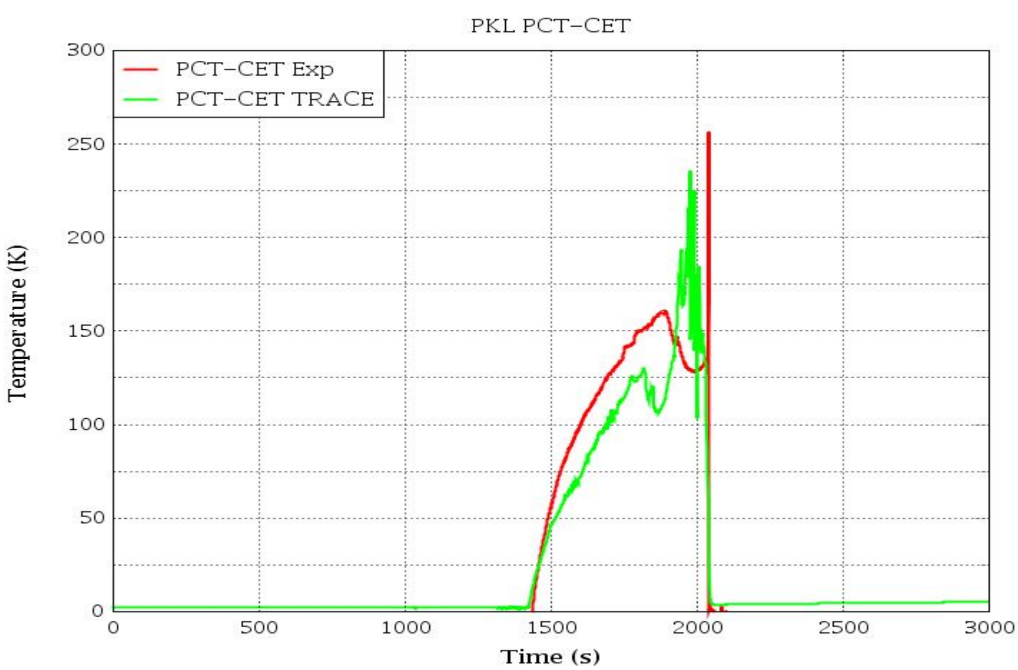

Figure 17 PCT-CET

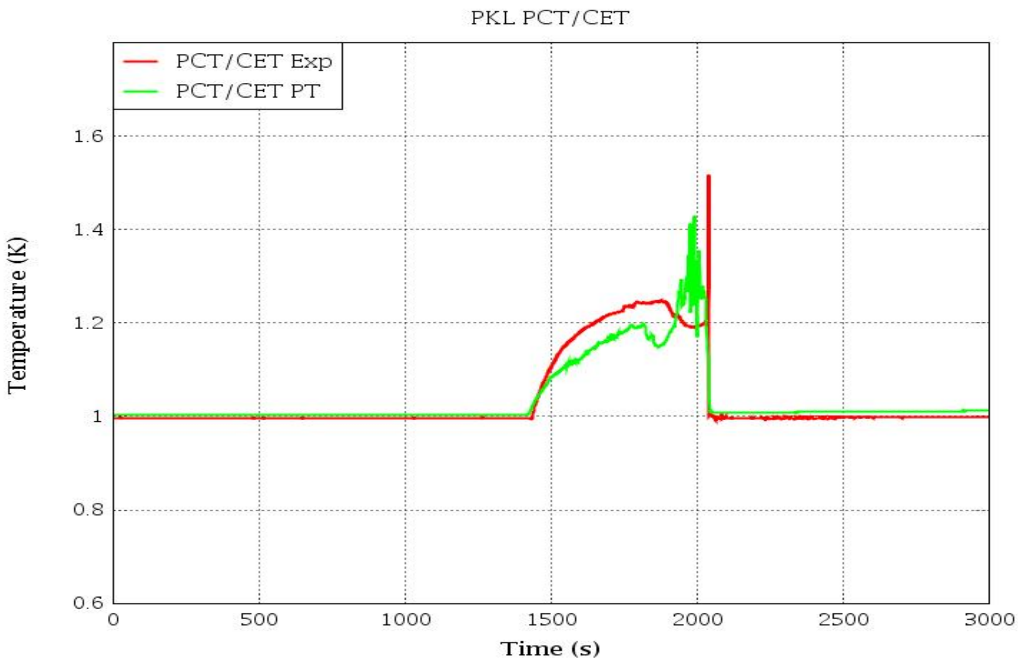

Figure 18 PCT/CET

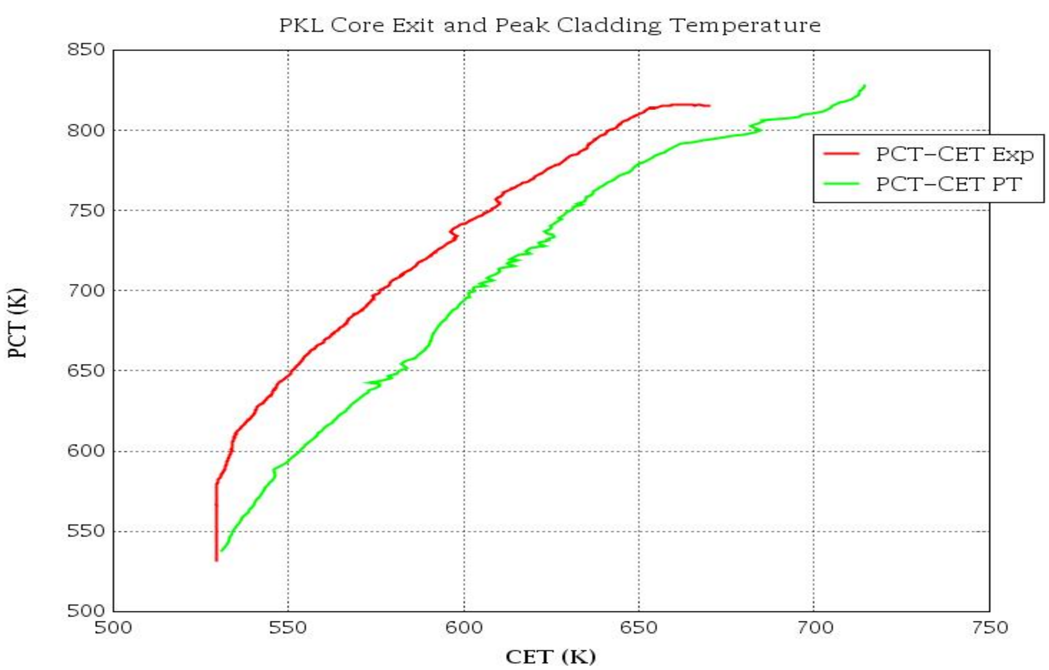

Figure 19 PCT VS CET

6 RUN STATISTICS

The calculation of G7.1 have been performed using a server of Cluster IBM 1350 with a biprocessor Intel Xeon with the following characteristics:

- x335 2.40GHz/100MHz/512KB L2, 512MB Memory, 331W, HS Open bay.
- x335 Processor 2.4GHz/512KB Upgrade.
- 1GB PC2100 CL2.5 ECC DDR SDRAM RDIMM.
- 18.2GB 10K-RPM ULTRA 160 SCSI Hot-Swap SL HDD.
- Remote Supervisor Adaptor.

In, Table 4 there are exposed the relevant parameters of the run statistics of the simulation of experiment G7.1

Table 4 Run Statistics G7.1

	RT	CPU	TS	CPU/RT	DT
Break Valve Opening (SOT)	500	1165.078	0.1	2.33	9835
Depressurization (CET>623K)	1780	3154.000	0.1	1.77	24275
ACC discharge	1920	11232.719	0.01	5.85	38190
ACC injection finished	2226	16111.359	0.01	7.24	69470
LPSI injection	2436	19049.203	0.01	7.82	90885
End of transient	3000	31233.594	0.01	10.41	148855

RT: Transient time (s)
CPU: Execution time (s)
TS: Maximum time step (s)
C: Total number of volumes
DT: Total number of time steps

7 CONCLUSIONS

This work presents the results of the simulation by TRACE of a small break loss of coolant accident with primary circuit closed and pressurized, and at midloop coolant inventory with a HPIS failure developed in the PKL Test Facility (G7.1 experiment). A TRACE model for PKL facility based on two VESSEL components to represent PKL reactor vessel is presented.

From the results obtained in the previous sections, it can be concluded that the main phenomenology observed in the experiment is reproduced in TRACE simulation, where the primary and secondary pressures, levels and temperatures follow the evolution of the experimental measurements.

A secondary-side depressurization postulated as an accident mitigation measure has succeeded to depressurize the primary circuit, permitting the actuation of accumulators and LPIS, which avoid high temperatures in the core. Once the safety injections via the accumulators and the LPIS are produced a stable situation of the plant is achieved with a rewetting of the core.

Regarding the differences between PCT and CET, there is a maximum temperature difference of 130 K in the TRACE5 simulation, while in the PKL experiment the difference is of 160 K. The difference in relation with the behavior of the CET, implies to improve the measurement of that temperature and a better study about why is greater than the experimental.

In general, TRACE5 results for the PKL are coherent with PKL experimental data.

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10. SUPPLEMENTARY NOTES

K.Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The PKL-2 test program is investigating safety issues relevant for pressurized water reactor (PWR) plants at the Primärkreislauf-Versuchsanlage (PKL) facility situated in Erlangen, Germany. AREVA NP has, for a number of years, conducted valuable experiments on reactor thermal-hydraulics in the PKL facility, including tests addressing the heat transfer mechanisms in the steam generators in the presence of nitrogen, steam and water, in both vertical and horizontal steam generators.
This paper focuses on the simulation, using the best estimate code TRACE, of the experiment G7.1 conducted at the PKL facility. The OECD-PKL2 test G7.1 is a hot leg Small Break-Loss of Coolant Accident (SB-LOCA) with a total failure of the High-Pressure Safety Injection (HPSI). The PKL test facility simulates a KWU 1300 MWe pressurized water reactor with all elevations scaled 1:1 and with volume and power scaled by a factor of 1:145. This test had been previously performed in ROSA/LSTF facility in order to compare results between facilities and to analyze different plant configurations and scaling effect. The postulated additional system failures (no HPSI, no automatic secondary-side cooldown) return a course of events that necessitates Action Management (AM) measures to prevent core-melt scenario. A fast secondary-side depressurization initiated after occurrence of core uncovery was employed, as AM measure for restoration of the secondary side heat sink aiming for a fast reduction of the primary pressure. The reduction of the primary pressure down to ACC injection pressure then effectuates the transition to the low-pressure phase with the Low-Pressure Safety Injection (LPSI) active

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
High Pressure Safety Injection (HPSI) Secondary Side Depressurization (SSD) Consejo de Seguridad Nuclear (Spanish nuclear regulatory commission, CSN) Core Exit Temperature (CET) Accident Management (AM) Primärkreislauf-Versuchsanlage (primary coolant loop test facility,PKL)	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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