



# International Agreement Report

## Simulation of the G3.1 Experiment at PKL Facility Using RELAP5/Mod3.3

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## **ABSTRACT**

When a nuclear power plant is in full conditions of operation, or with the primary of reactor coolant system pressurized, one of the most important accidental situation is when a main steam line break occurs, due to that causes a rapid depressurization of the affected steam generator (SG). This depressurization leads to an increment of heat transfer from the primary to the secondary side that cooldown primary side quickly.

This paper focuses on the simulation, using the best estimate code RELAP5/Mod 3.3, of the experiment G3.1 conducted at the PKL facility that simulate a main steam line break. In the experiment G3.1 the physical phenomena to investigate are the assessment of the reactor pressure vessel (RPV) integrity considering PTS (pressurized thermal shock) and the assessment of potential recriticality following entrainment of colder water into the core, though this work has centered in the thermahydraulic behavior.



## 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

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<sup>1</sup> 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).

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 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.

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



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## **EXECUTIVE SUMMARY**

The experiment G3.1 conducted at the PKL facility consists of a large break in the main steam line (MSLB), at the exit of one of the steam generators secondary side. The MSLB leads to heat transfer increase from the primary to the secondary side through the steam generators and causes a fast cooldown transient on the primary side. In the experiment two phases are distinguished: The first one analyzes the primary circuit cooldown due to the fast secondary side depressurization, and in the second phase the effect of high-pressure injections and the opening of the relief valve of pressurizer is studied.

Among the objectives of this experiment there is a great interest in analyzing the capability of thermal-hydraulic codes against heat transfer from the primary to the secondary side, and to export the lessons from the facility to the commercial plants.

A simulation using RELAP5 code has been performed and the results of such simulation are presented, which show that the most important physical phenomena observed in PKL-G3.1 experiment are well reproduced in the RELAP PKL model used.



## **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
ATWS	Anticipated transient without scram
CAMP	Code Assessment and Management Program
CL	Cold Leg
CSN	Consejo de Seguridad Nuclear (Spanish nuclear regulatory commission)
ECCS	Emergency Core Cooling System
HL	Hot Leg
HPIS	High Pressure injection system
JAERI	Japan Atomic Energy Research Institute
LBLOCA	Large Break Loss of Coolant Analysis
LPIS	Low Pressure injection system
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NEA	Nuclear Energy Agency
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
OECD	Organisation for Economic Cooperation and Development
PWR	Pressurized Water Reactor
RCL	Reactor Coolant Line
RCS	Reactor Coolant System
RELAP	Reactor Excursion and Leak Analysis Program
RHR	Residual Heat Removal System
SBLOCA	Small Break Loss of Coolant Analysis
SG	Steam Generator
TH	Therma-hydraulic



# 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 coded 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. Several experimental programs have been conducted at PKL facility. Thus, programs PKL I and PKL II, 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 III program started with the objective of studying different transients with and without LOCAs [Ref. 1]. One example of a transient considered in this later case is the loss of the residual heat removal system in shutdown conditions [Ref. 3, 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 a main steam line break (MSLB), experiment G3.1, has been conducted at PKL facility. The MSLB leads to increase heat transfer from the primary to the secondary side through the steam generators and causes a fast cooldown transient on the primary side. The main objective of this experiment is focused on analyzing the capability of thermal-hydraulic codes against heat transfer from the primary to the secondary side.

The results obtained in the experiments must also serve to predict and analyze safety issues of commercial nuclear power plants.

The work presented in this document analyzes the simulation of a main steam line break experiment, using RELAP5 code. The analysis of the results show that the most important physical phenomena presented in PKL-G3.1 experiment is well reproduced in the PKL RELAP model presented.

The rest of the document is organized as follows: The PKL facility is briefly described in Section 2. Section 3 is devoted to introduce experiment G3.1, and Section 4 describes the RELAP5/Mod 3.3 model for the PKL facility used to simulate the experiment. In Section 5, the main results obtained from simulation is presented and compared with PKL experimental data. Finally, the run statistics are presented in Section 6 and the main conclusions of this paper are summarized in Section 7.

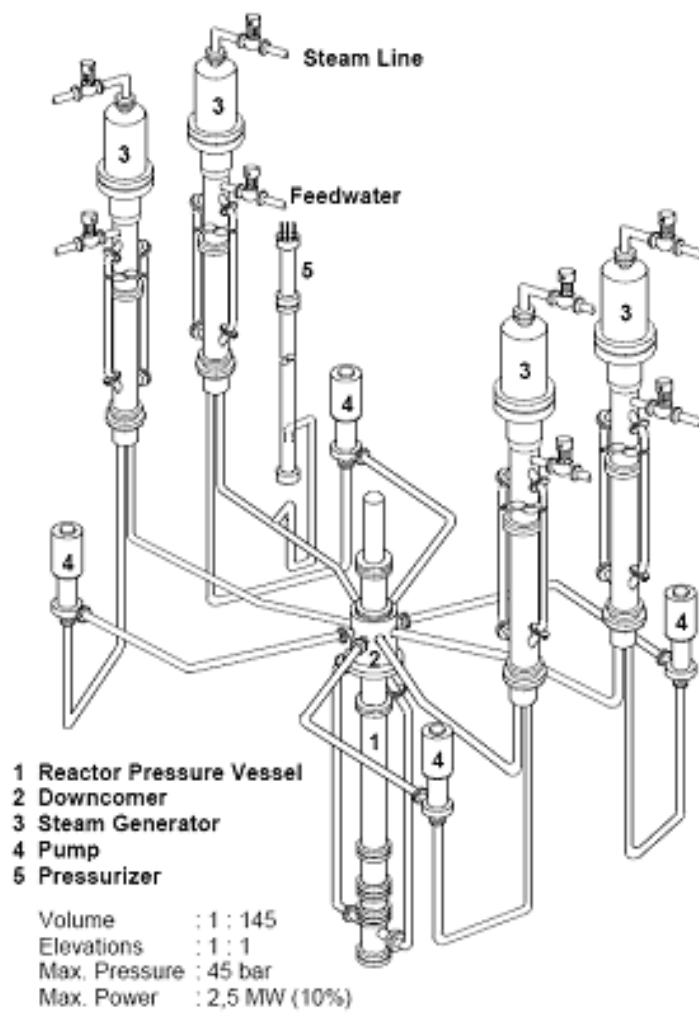


## **2 PKL FACILITY DESCRIPTION**

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 [Ref. 1, Ref. 4].

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. 4].

Three experimental programs have been conducted at PKL facility. Programs PKL I and PKL II, 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 III program started in 1986 [Ref. 1], with the objective of studying different transients with and without LOCAs. The PKL tests results have also been used for preparation and verification of procedures described in the operating manuals and for answering questions exposed by regulatory bodies.



**Figure 1**      **PKL Facility**

### 3 DESCRIPTION OF G3.1 TRANSIENT

The initiating event in G31 experiment is a large break in the main steam line (MSLB), at the connection with steam generator one [Ref. 5]. This leads to increase heat transfer from the primary to the secondary side through all the steam generators and causes a fast cooldown transient on the primary side. After that, injection from high pressure injection system (HPIS) is activated in two of the four loops what produces an increase in the primary pressure. The pressure rises until pressurizer safety valves set point and, when the valves open, steam is lost and the pressure decreases. The most important initial and boundary conditions in the accomplishment of the G3.1 experiment are:

- Primary completely full and in operation at the maximum pressure of installation, 42 bar.
- Power 260 kW, equivalent to 0.8% of the residual power including the compensation for loss of heat.
- Hot pressurizer, approximately 523 °K (250° C).
- The initiating event is a break of 10% of the main steam line area of steam generator one.
- The secondary side of steam generator one is filled reaching a level of 9.2 m. The others are isolated from the break (main steam isolation valves closed) and the other steam generators filled with a level of 12.2 m, and a pressure at the beginning of the transient of 35 bar.
- All steam generators are isolated from the feed water system.
- The primary recirculation stops at the beginning of the experiment.

With these boundary conditions, the initial conditions at the beginning of the transient are shown in Table 1.

**Table 1      G3.1 Initial Conditions**

	Initial Conditions
<b>Primary side:</b>	
<b>Coolant inventory</b>	Full of water
<b>Presurizer Level</b>	7.4 m
<b>Temperature at core outlet</b>	519 K
<b>Pressure</b>	42 bar
<b>Power</b>	260 kW, aprox 0.8 % of residual power
<b>Flow rate loops</b>	34 kg/s Symmetrical heat removal through 4 steam generators with forced circulation of subcooled water by 4 loops.
<b>Secondary side</b>	
<b>Level</b>	4 SG filled with water; 8.8-12.5 m
<b>Temperature</b>	473-513 K
<b>Pressure</b>	35 bar

Under the conditions exposed in Table 1, the development of the transient G3.1 consists of two different phases. The first phase begins with the break in the main steam line and includes the emptying of the secondary side of the affected SG. The second phase starts with the initiation

of high-pressure safety injection to retrieve the reactor coolant circuit inventory and includes the pressure limitation (control) by the pressurizer safety valve and the reduction of the HPIS flow. Table 2 presents a summary of the major facts that occur during the transient.

**Table 2 Major Events of the Transient G3.1**

<b>Time (s)</b>	<b>Event</b>
<b>0</b>	<b>Phase I</b> Start of the transient: <ul style="list-style-type: none"> <li>• Main steam line break;</li> <li>• Gradual stop of reactor coolant pumps</li> </ul>
210	Blocking of reactor coolant pumps;
<b>1000</b>	<b>Start of Phase II</b>
1030	High-pressure safety injection (HPIS) at loops 1 and 4 starts.
1420	Pressurizer relief valve opening
2150	Safety high-pressure injection flow is reduced.
4410	End of the transient

Following the actions exposed in Table 2, the expected evolution of primary and secondary side pressures and the main temperatures of the reactor coolant circuit during the different phases of transient are shown in Figure 2 [Ref. 5]. In this figure, the fast depressurization in steam generator one due to the break opening is observed. Primary pressure follows secondary side pressure decrease during phase one, as shown in Figure 2. Then, HPIS injection starts and pressure rises again until control pressure activation is triggered.

Regarding reactor pressure vessel temperatures, Figure 2 shows a decrease in the temperatures at the break opening, which is more evident in loop one. Once the pressure is at the lowest level, the temperature starts to increase, as a result of the residual heat generated in the core, until the pressure control activation, which is the measure proposed to evacuate the residual heat.

For each phase of the transient the following phenomena were investigated.

#### **PHASE I:**

- Behavior of steam generator affected by break.
- Evolution of pressure, temperature, level and flow.
- Heat transfer primary-secondary and boiling at the steam generator affected by the break during fast depressurization.
- The natural circulation and influence on the coolant temperature at the vessel inlet in each loop.



## PHASE II:

- Behavior of the pressurizer.
- Heat transfer between primary and secondary sides.
- Influence of the high-pressure injection system on natural circulation and temperature at the entrance of the vessel's reactor.

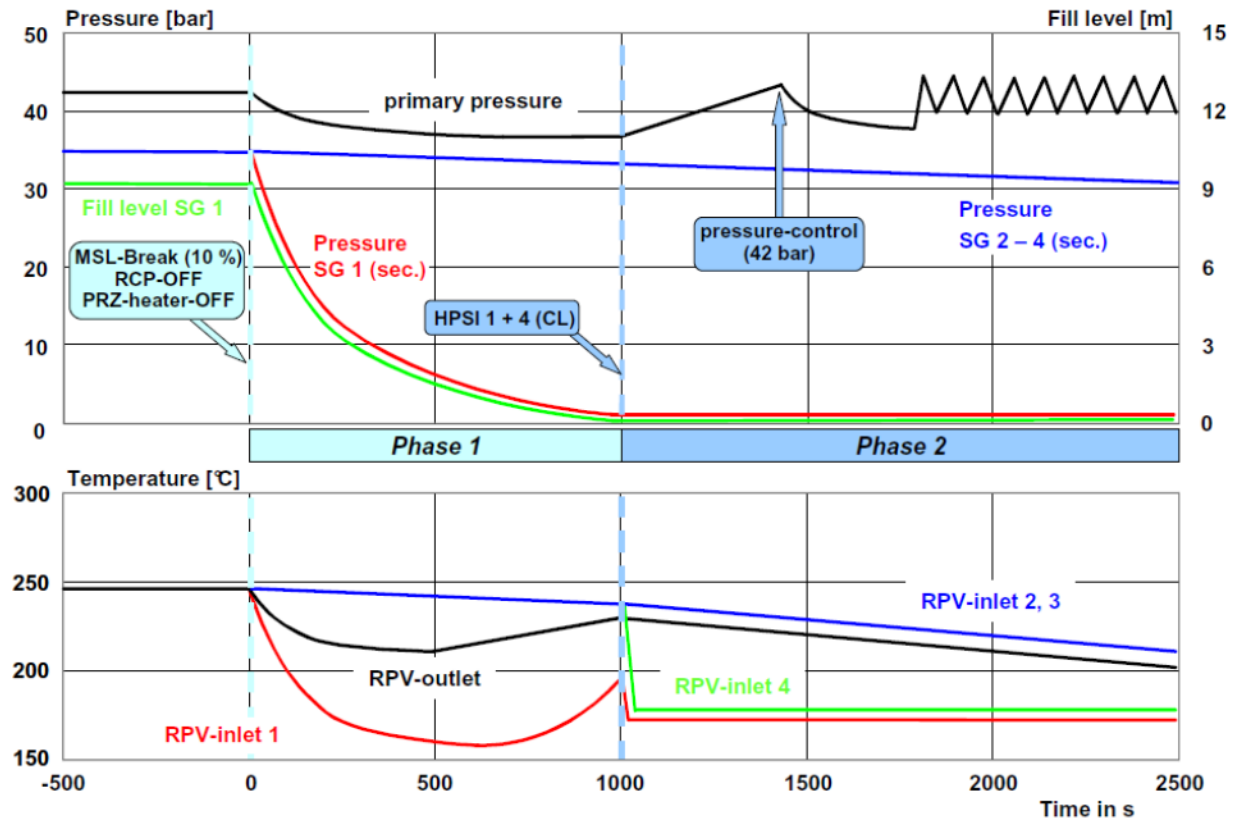


Figure 2 Primary and Secondary Pressure, Level and Temperature in Test G3.1



## 4 RELAP5 MODEL OF PKL FACILITY

The transient simulation has been performed using RELAP5-Mod 3.3 code [Ref. 6] and SNAP [Ref. 7]. The RELAP5 model used consists of 600 hydraulic volumes, 622 junctions and 512 heat structures. This model has been adapted to simulate test conditions, from the PKL model provided by the facility [Ref. 8]. Figure 3 outlines the nodalization used in the simulation.

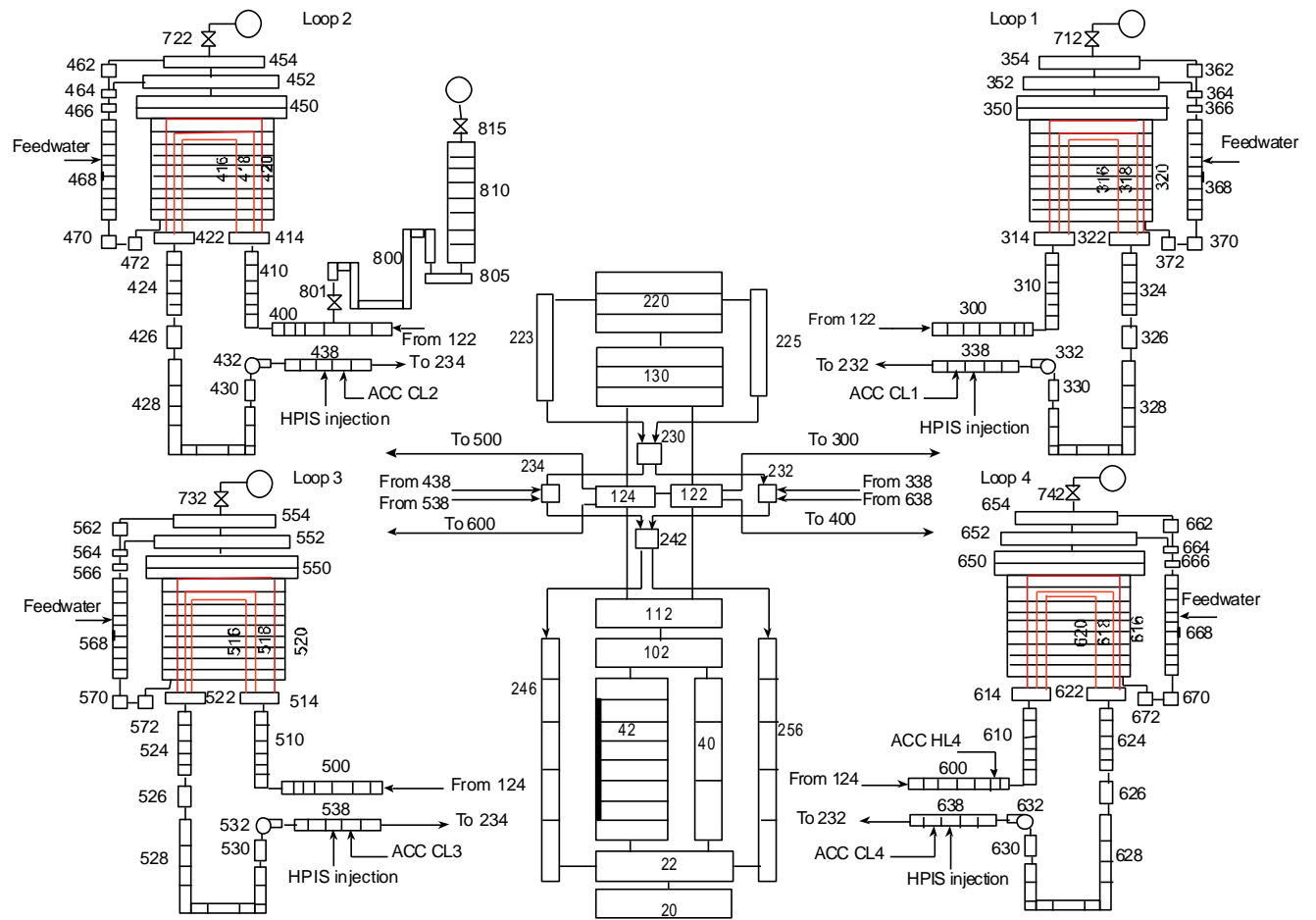
The core is simulated using a pipe component of eight volumes. Six of these volumes contain the fuel rods, which are simulated using a heat structure component that generates the residual power established in Table 1. The vessel of the PKL facility has two external downcomers (see Figure 1) simulated in the RELAP model by means of two external pipes. The cold legs of all four loops are nodalized using pipe and branch components, which are connected to two branches, volumes 232 and 234, which in turn are connected to the downcomer vessel. The facility has four by-passes in the vessel upper head which have been collapsed in this model into two branch components, 223 and 225.

The four primary loops are modelled with a pump and a steam generator in each loop using pipe, pump and branch 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 main steam line break is simulated in this model by a trip valve connected to a time dependent volume that opens at the beginning of the transient.

The different injections from the HPIS at the end of the transient have been simulated using time dependent volumes connected with time dependent junctions that establish the amount of water to be injected. All these injections are made by time according to specifications of test.

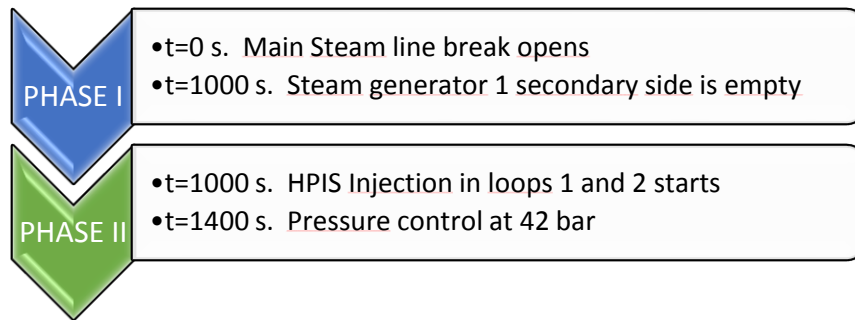
The pressurizer relief valve is modeled using a check valve with a set point of 42 bars to open when pressurizer is full of water later the injections.



**Figure 3** PKL RELAP Model

## 5 MAIN STEAM LINE BREAK SIMULATION RESULTS

The simulation has been performed using the initial conditions exposed in Section 3 and the RELAP model for PKL explained in Section 4. In the evolution of the transient, two phases are clearly distinguished. In the first one, the main objective is to study how the fast depressurization on the steam generator secondary side affects the primary pressure. In the second phase, once the secondary side of the steam generator affected by the break is empty, the HPIS injections are performed, and the effect of such injections are analyzed. In Figure 4, the main actions occurring during the transient evolution and their approximate timing are summarized.

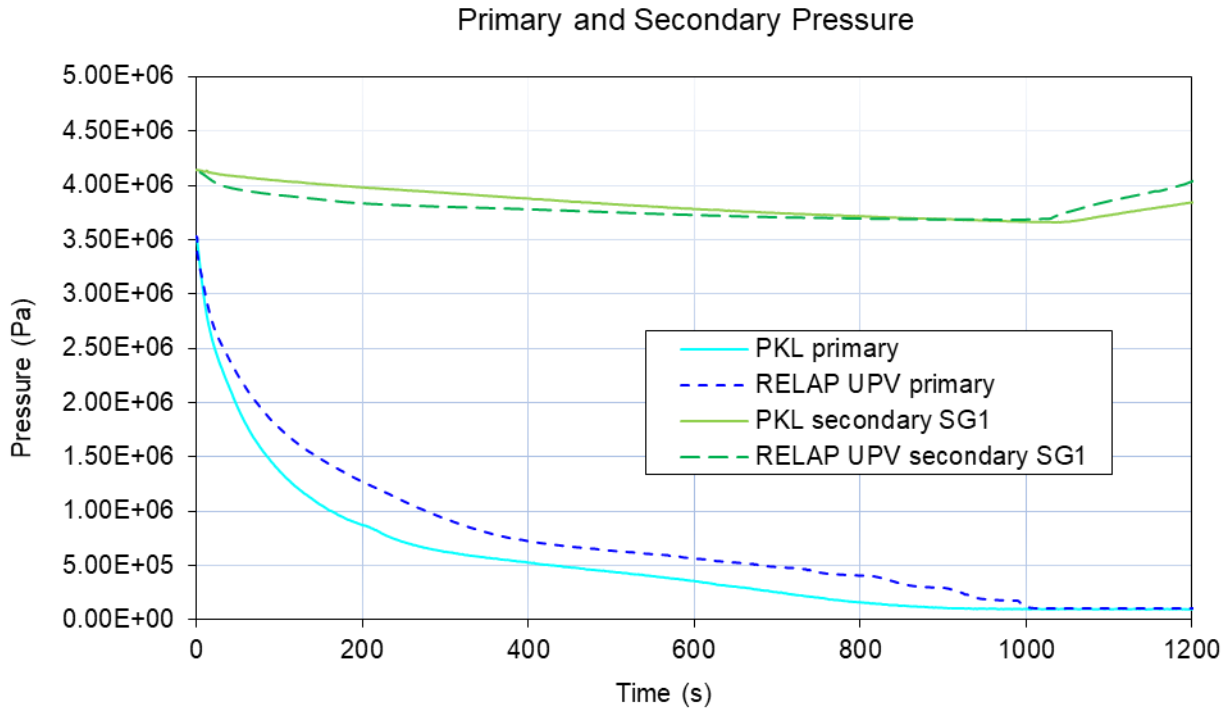


**Figure 4 Main Milestones During the Transient Evolution**

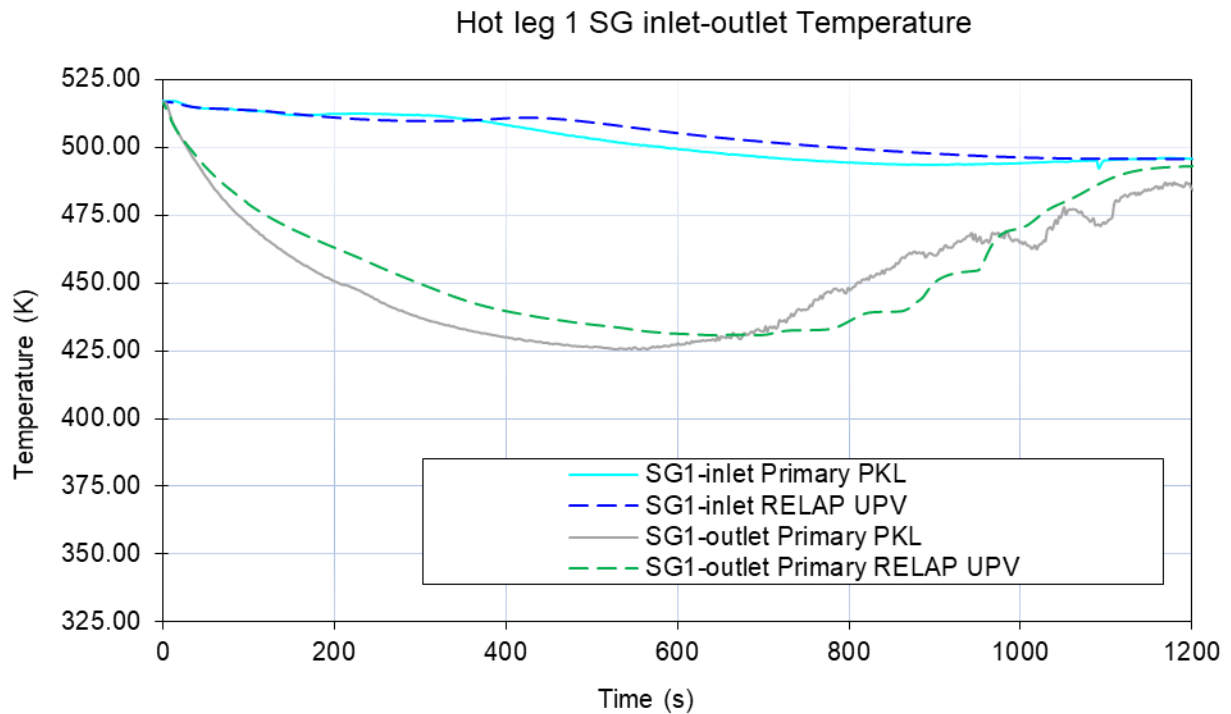
Once the initial conditions are reached, the transient is initiated by a break in the main steam line of steam generator opening the valve connected to the component TMDPVOL that represents the break simulated.

### **5.1 Phase I (From t=0 sec. until t=1000 sec.)**

Once the Main Steam Line break is produced, the pressure of steam generator 1, starts to decrease very fast, as can be observed in Figure 5. As all the steam generators are isolated, the rest of steam generators maintain their initial pressure. The sharp depressurization in the secondary affects the pressure in the primary pressure, which decreases quite fast, as shown in Figure 5. Consequently, primary side temperature decreases as observed in Figure 6. This situation is maintained while there is enough coolant inventory in the steam generator to allow heat transfer (Phase I).



**Figure 5 Primary and Secondary SG1 Pressure. Phase I**



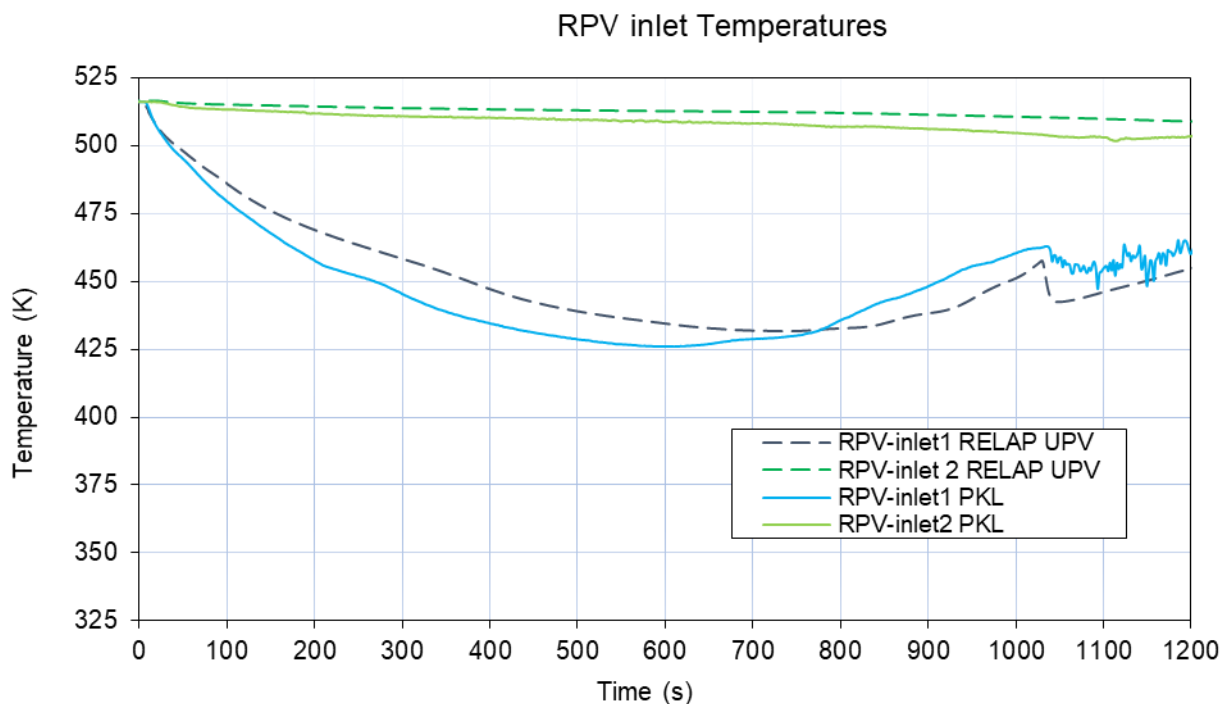
**Figure 6 Primary SG1 Inlet-Outlet Temperature. Phase I**

RELAP calculations predict a lower depressurization in the steam generator secondary side than the one observed in the experiment as shown in Figure 5. However, primary side pressure decreases faster in the calculations than in the experiment. Thus, the RELAP overpredicts the effect of secondary side depressurization on the primary pressure.

The steam generator one depressurization increases the heat transfer from primary to secondary side. Figure 6 shows the U-tube inlet and outlet temperatures of steam generator 1. In this figure, it can be observed that while inlet temperature remains constant, the outlet temperature decreases what means that the heat transferred is increases.

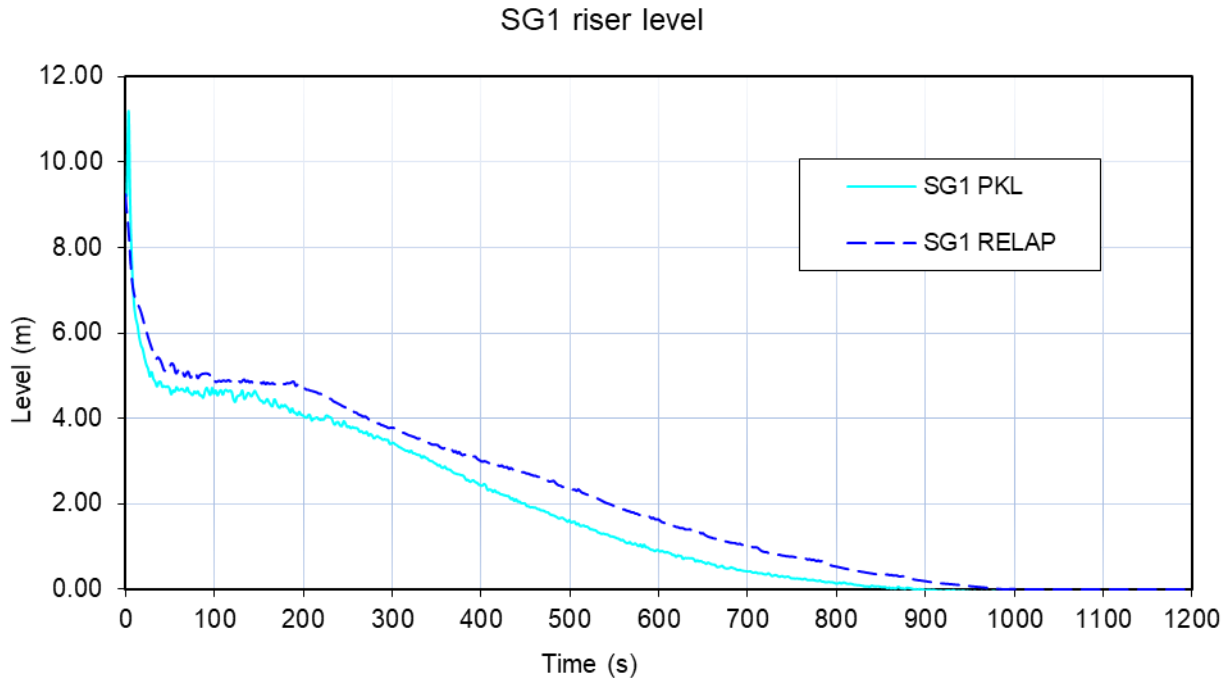
RELAP predicts an outlet temperature higher than PKL data, following the behavior observed in the pressure evolution.

As heat is transferred through the steam generator affected by the break, there is an asymmetry behavior in the temperatures at the inlet of the reactor pressure vessel. Thus, the coolant temperature at the inlet reactor pressure vessel through loop 1 is considerably lower than the temperature of the coolant entering the vessel from the other loops as can be observed in Figure 7.



**Figure 7 RPV Inlet Temperature. Phase I**

Such heat transfer from primary to secondary side of steam generator 1 produces a consequent boiling of water in the secondary side and the loss of inventory (steam) through the break, reducing the secondary side coolant level, as shown in Figure 8. As there is no feedwater injection, at 850 s of the start of the transient steam generator 1 secondary side empties and the steam generator is lost as heat sink, and no further reduction of primary pressure is possible, so it remains constant until the end of phase I, as can be observed in Figure 5.



**Figure 8 Secondary SG1 Level. Phase I**

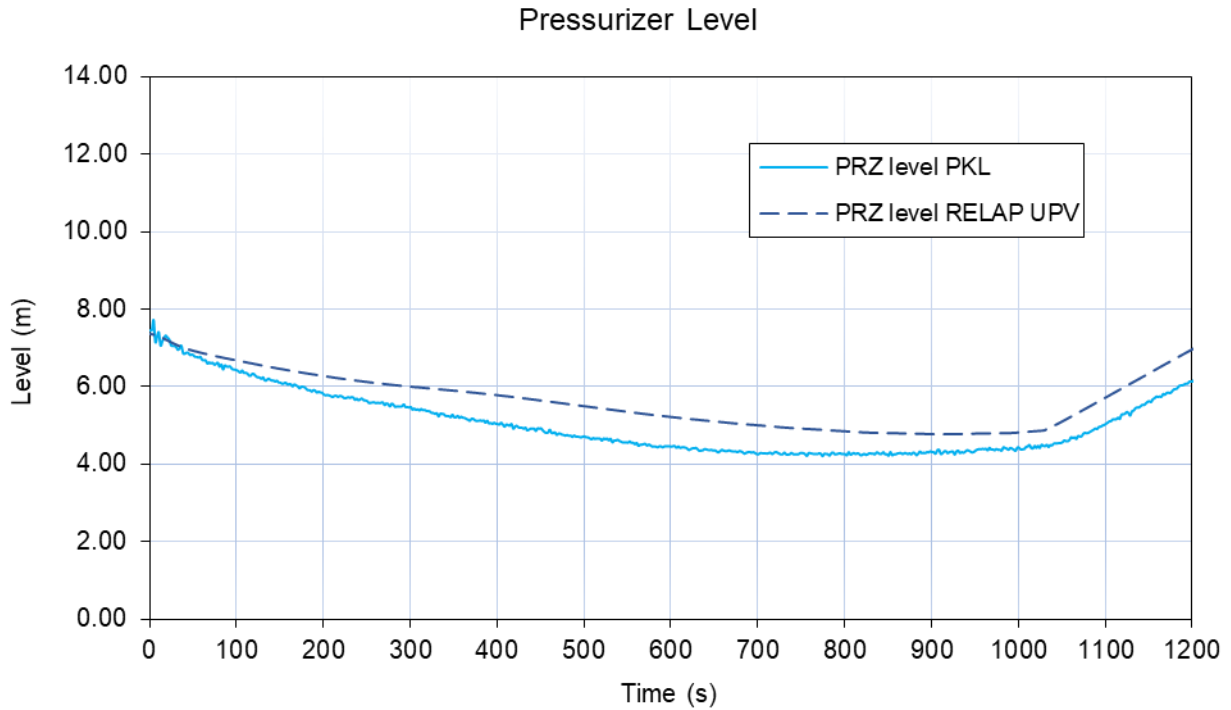
In this case, spite of the higher depressurization rate predicted by RELAP, steam generator level calculated by RELAP is higher than the experimental data, though both get empty at the same time.

The pressurizer level also follows the same trend, there is a decrease in the level due to the depressurization of the primary system and from 700 s to 1023 s the level remains constant at a very low value, as shown in Figure 9.

As occurs with the level in the secondary side, the level calculated by RELAP is higher than the experimental values, but both present the same evolution.

The first phase of the transient ends at 1030 s, once the secondary side of steam generator 1 is empty and the capacity of removing the residual heat generated in the core is lost. It is also impossible to remove such heat through the other steam generators, because the conditions of their secondary sides do not allow heat transfer. At this moment, the injection from HPIS starts and phase II of the transient begins.

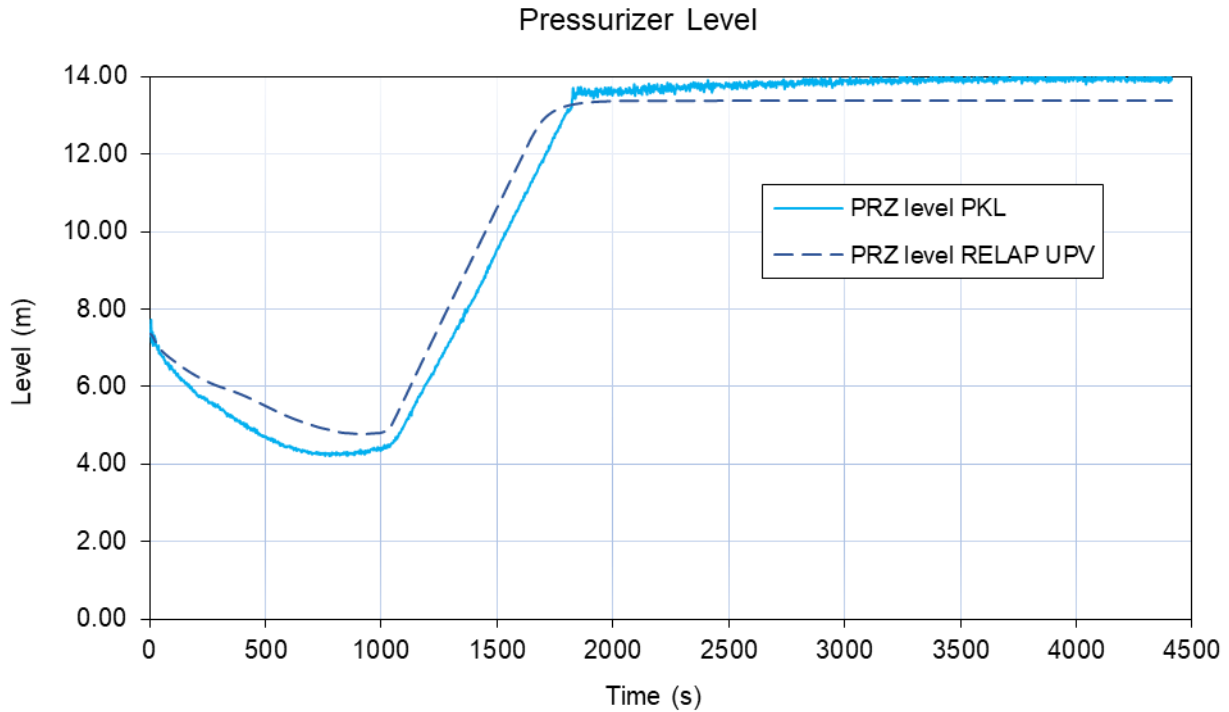




**Figure 9 Pressurizer Level. Phase I**

## 5.2 Phase II (From 1000 sec to the end of the transient)

The second phase starts with the injection of the high-pressure injection system and, consequently, the primary circuit is refilled. Figure 10 shows the evolution of the pressurizer level and the effect of the injection is observed. Thus, at the beginning of phase II the pressurizer is empty and at 1800 s it is completely filled with water. HPIS injects a constant mass flow until the end of the transient.



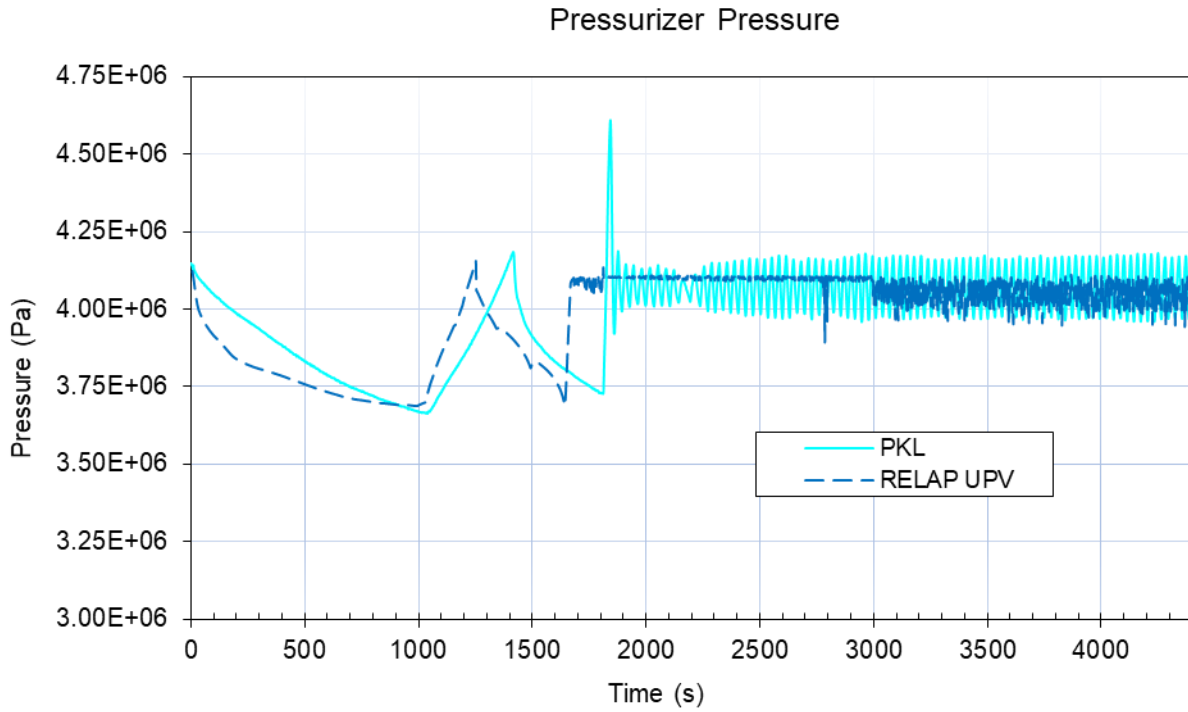
**Figure 10 Pressurizer Level. Phase II**

Due to initial level at the beginning of phase II, the refill of pressurizer is advanced respect to experiment around 100 s (Figure 10).

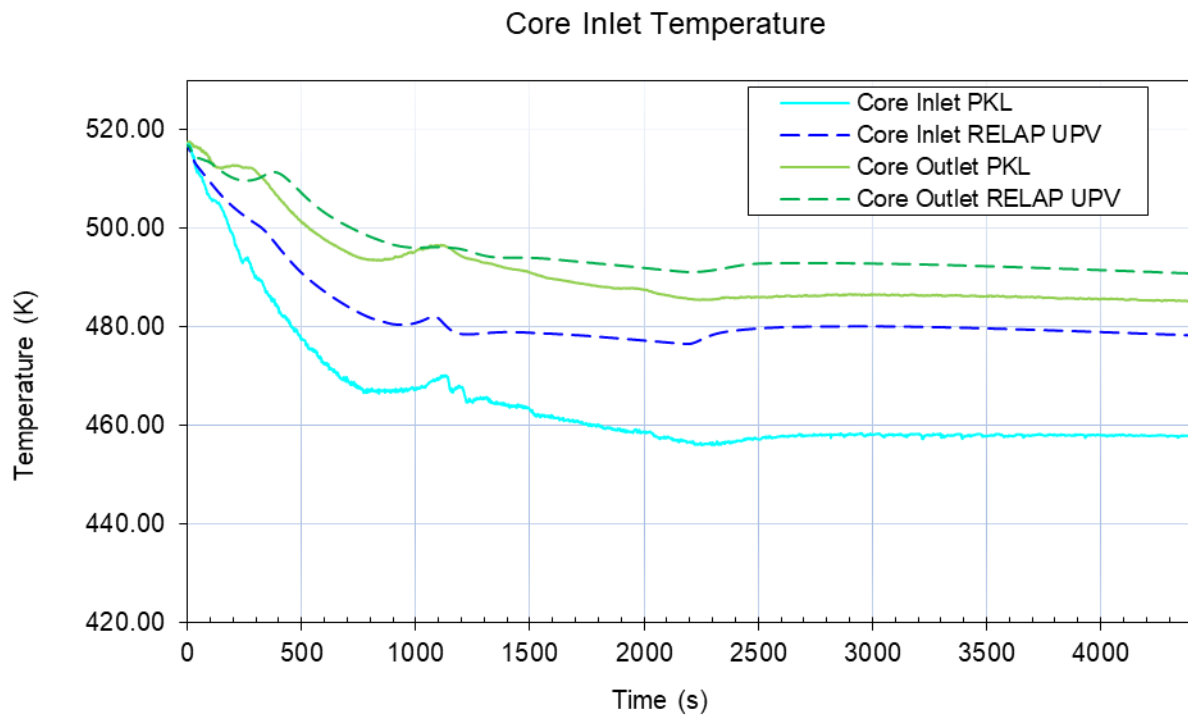
The water is injected in the primary circuit and pressure starts to increase until it reaches 42 bar, which is the pressurizer relief valves set point, what happens at 1250 s, as can be also observed in Figure 11. When the set point is met, the valves open to maintain such pressure. However, the coolant inventory does not decrease due to the HPIS injection.

This feed and bleed of coolant maintains the temperatures at the inlet and outlet of core, as can be observed in Figure 12, obtaining a plant stable situation.

When the pressure reaches the pressurizer relief valve set point, steam is discharged and a sharp depressurization is produced, as can be observed in Figure 11, until the High-Pressure Injection fills up the pressurizer, rising the pressure again until the valve set point. Then the pressure is maintained by the relief valve.

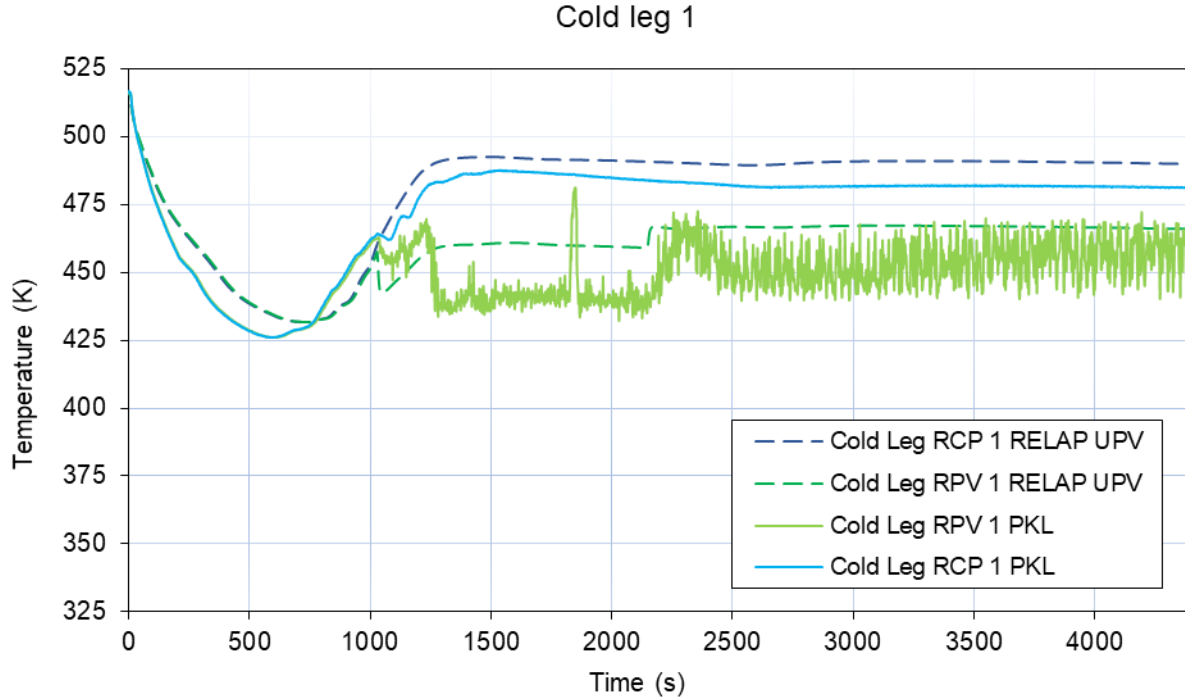


**Figure 11** Pressure Pressurizer. Phase II



**Figure 12** Core Inlet-Outlet Temperatures. Phase II

Figure 13 shows the effectiveness of HPIS injection. This figure presents the evolution of the coolant temperature in the reactor coolant line before HPIS the injection (in the reactor coolant pump, RCP) and inside the reactor pressure vessel (RPV). Temperature decreases in this second location due to the safety injection.

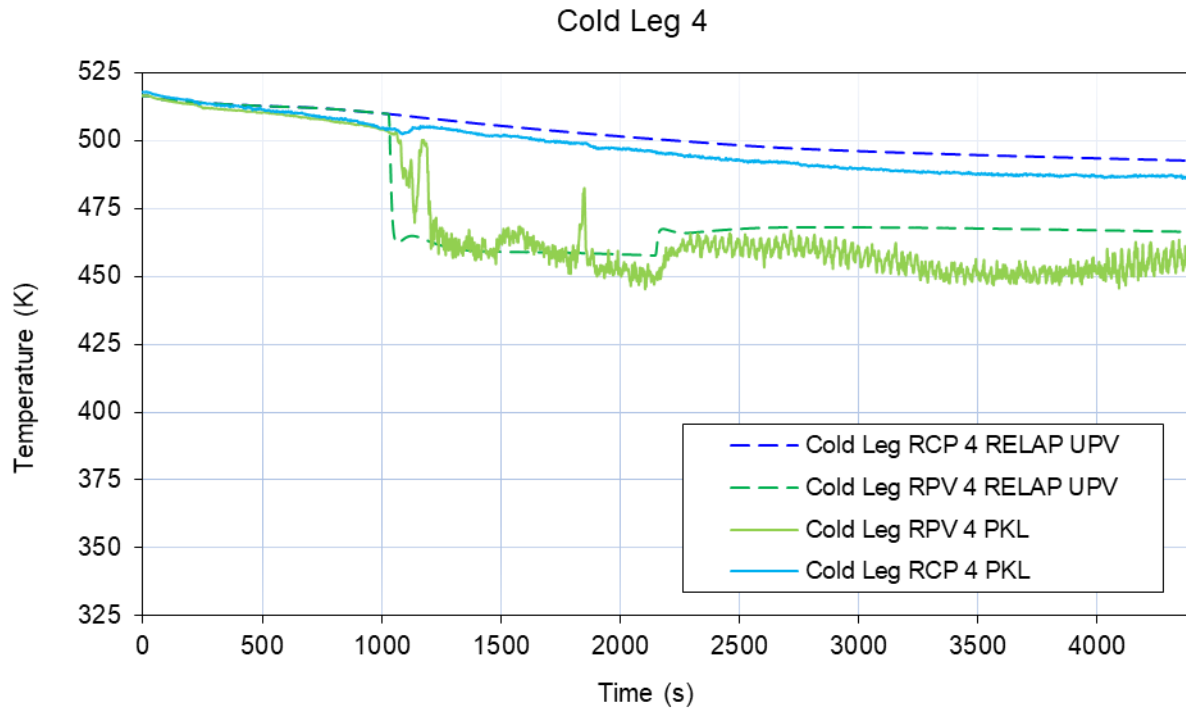


**Figure 13 Cold Leg 1 Temperatures. Phase II**

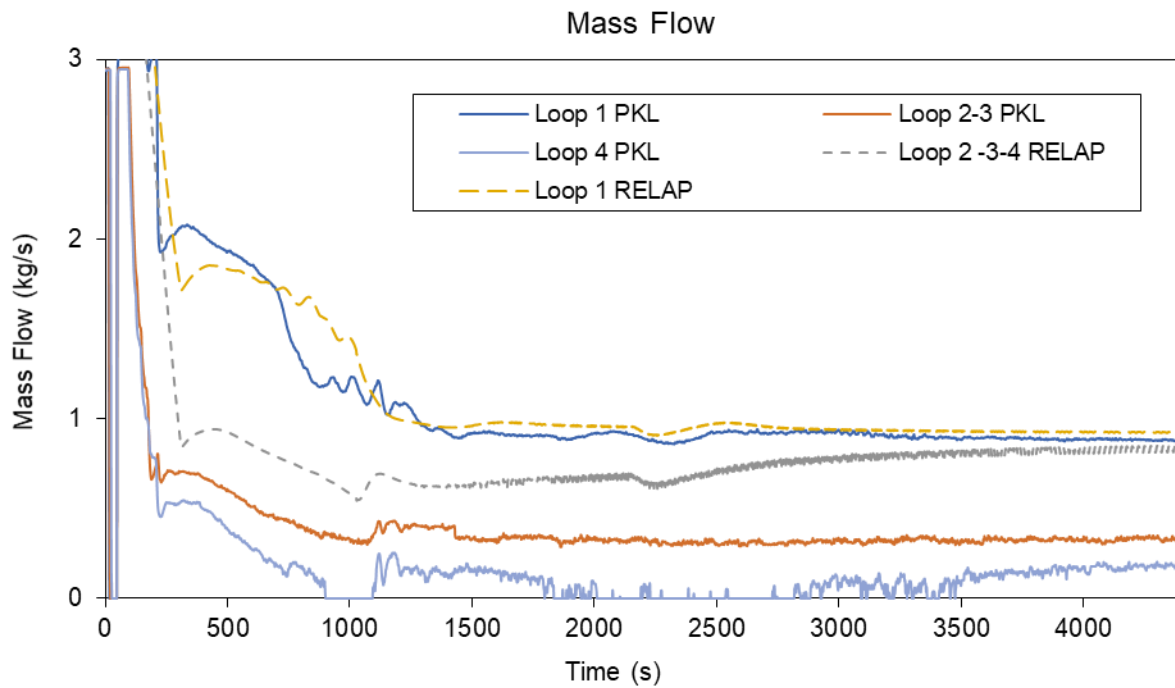
In this case, the difference between results are in the same order of magnitude than in phase I, showing the same difference before and later the injection, showing the efficacy of it.

As it occurred in phase I, a difference in the coolant temperature is observed, depending on the location of safety injections. Figure 14 presents the temperature in the cold leg of loop 4, without HPIS injection and inside the reactor pressure vessel. In the loop, the temperature remains constant and higher than the temperature inside the reactor pressure vessel due to the effectiveness of HPIS injection.

Finally, natural circulation observed in the experiment is also reproduced by RELAP calculations. Thus, Figure 15, shows the mass flow in all four loops. In the experimental data in loop 1 the mass flow is higher than in the others due to the HPIS injection, with a value of 1 kg/s of coolant and in loops 2 and 3 the mass flow rate is maintained at 0.5 kg/s, while in loop 4 the value is around 0.25 kg/s. RELAP results show that at the final of transient, the mass flow rates in loops 2, 3 and 4 due to natural circulation are the same, although there exists a difference with the experimental.



**Figure 14 Cold Leg 4 Temperatures. Phase II**



**Figure 15 Mass Flow Rates Loops. Phase II**



## 6 RUN STATISTICS

The calculations of G3.1 test applied to a PKL RELAP model 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 3 there is exposed the relevant parameters of the run statistics of the simulation of experiment G3.1.

**Table 3      Run Statistics**

	RT	CPU	TS	CPU/RT	C	DT	GT
Reactor coolant pumps blocked	210	2091	0.01	9.96	600	22477	0.155
SG1 secondary side empty	1000	105767	0.01	105.77	600	1237460	0.143
PRZ relief valves open	1290	107901	0.01	83.64	600	1277461	0.141
End of transient	4500	130828	0.01	29.07	600	1587460	0.137

RT: Transient time (s)

CPU: Execution time (s)

TS: Maximum time step (s)

C: Total number of volumes

DT: Total number of time steps

GT:  $GT = (CPU \cdot 10^3 / (C \cdot DT))$





## 7 CONCLUSIONS

This work presents the results of the simulation of a main steam line break with primary circuit closed and full of water. This simulation has been previously developed in the PKL Test Facility [Ref. 5].

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

However, a higher depressurization is predicted by RELAP simulation, while calculated levels and temperatures are overestimated, though the time that the secondary side is empty is very well reproduced.

The injection from HPIS allows maintain the primary in safe situation though it is necessary to make a bleed to avoid the overpresurization of primary side.

RELAP results shown as at the final of transient the natural circulation trends to be equal in four loops to the loops 1 of experimental result, not presenting differences between loops as in the experiment occurs.

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



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K.Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

When a nuclear power plant is in full conditions of operation, or with the primary of reactor coolant system pressurized, one of the most important accidental situation is when a main steam line break occurs, due to that causes a rapid depressurization of the affected steam generator (SG). This depressurization leads to an increment of heat transfer from the primary to the secondary side that cooldown primary side quickly.

This paper focuses on the simulation, using the best estimate code RELAP5/Mod 3.3, of the experiment G3.1 conducted at the PKL facility that simulate a main steam line break. In the experiment G3.1 the physical phenomena to investigate are the assessment of the reactor pressure vessel (RPV) integrity considering PTS (pressurized thermal shock) and the assessment of potential recriticality following entrainment of colder water into the core, thought this work has centered in the thermahydraulic behavior.

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