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International Agreement Report

Simulation of the PKL-G7.1 Experiment in a Westinghouse Nuclear Power Plant Using RELAP5/Mod3.3

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ABSTRACT

This paper focuses on the simulation in a Westinghouse design nuclear power plant of the experiment G7.1 conducted at the PKL facility. The test G7.1 consists of a Small Break-Loss of Coolant Accident (SB-LOCA) in the Hot Leg with a total failure of the high-pressure injection system (HPIS). The PKL 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, so some assumptions regarding the scaling and design features of the Westinghouse reactors should be taken into consideration to undertake the simulation.

The postulated additional system failures (no HPIS and no automatic secondary-side cooldown) make necessary to determine the Action Management (AM) measures to prevent core-melt scenario. Under such conditions, the accident mitigation procedure proposed is the manual depressurization of the secondary side steam generators followed by injection from accumulators. The simulation of the transient for a Westinghouse reactor has been performed using RELAP-5 thermal-hydraulic code.

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,

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).

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. G7.1 experiment consists of a hot leg small break LOCA with additional failure of safety systems, as high pressure injection system. Such situation makes necessary an adequate accident mitigation procedure to prevent the accident would lead to core damage. The efficiency of the accident mitigation measures proposed 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. Other objective is focused on performing such experiment on different facilities to export the lessons learnt to commercial nuclear power plants. In the Universitat Politècnica de València we are working on safety analysis of Westinghouse reactors, and there exist a great interest in transferring the knowledge from the experiments performed at PKL facility to this kind of reactors, different from the PKL plant of reference.

In this frame, the work presented analyzes the differences that exist between PKL and a Westinghouse reactor and the assumptions made to adapt the conditions of the experiment to the commercial plant technology.

To simulate the experiment a volume scaling with full height full pressure approach has been used. This approach preserves time scale, important for a fast response as the fast cooldown of this transient, and other scaling ratios as length, hydraulic diameter, velocity, heat generation rate/volume, fluid temperature and non-dimensional characteristics of pumps and valves.

Apart from the scaling considerations, the difference in the technology leads to different configurations of the safety systems available as well as the actuation of such safety systems.

A simulation using RELAP5 code has been performed and the result of such simulation has been presented, which show that the most important physical phenomena presented in PKL-G7.1 experiment is well reproduced in a Westinghouse design reactor.

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 (BWR), 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 Large Scale Test Facility/Rig of Safety (LSTF/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 water inventory and pressure of the reactor coolant system decrease and this leads to empty the reactor pressure vessel, and to core uncovering. So, it is necessary the safety systems actuation 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, no HPSI and no automatic secondary-side cooldown, leads to core uncovering and the clad temperature increases until core-melt scenario if no action is done [Ref. 1, Ref. 4]. It makes necessary Accident Management (AM) measures to prevent this scenario.

A fast secondary-side depressurization initiated after core uncovery should be performed as AM measure to reestablish 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].

The results obtained in the experiments should be used to predict and analyze safety issues of commercial nuclear power plants. Thus, other important objective of this study is focused on exporting the lessons learnt from the analysis of the experimental results to a commercial Spanish nuclear power plant. In this frame, the Universitat Politècnica de València, in collaboration with the Consejo de Seguridad Nuclear and UNESA, is developing a safety analysis project to transfer the results of PKL facility experiments to a three loop Westinghouse reactor design.

The work presented in this document analyzes the differences that exist between PKL and a Westinghouse reactor designs and the assumptions made to adapt the conditions of the experiment to the commercial plant technology. A simulation of a small break loss of coolant experiment, using RELAP5 code, has been performed and the results of such simulation are presented. The analysis of the results shows that the most important physical phenomena presented in PKL-G7.1 experiment are well reproduced in general [Ref.7, Ref. 8, Ref. 9], and in a Westinghouse design reactor, in this work.

The rest of the document is organized as follows: The PKL facility and the Westinghouse design reactor are briefly described in Section 2 and 3 respectively. Section 4 is devoted to introduce experiment G7.1 and the assumptions made to simulate such experiment in the plant of reference. Section 0 describes the RELAP5/Mod 3.3 model for the Westinghouse design reactor used and introduces the adjustment to the conditions of the commercial plant. In Section 6, the main results obtained from simulations are presented and compared with PKL experimental data. Finally, the run statistics are presented in Section 7 and the main conclusions of this paper are summarized in Section 8.

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. 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].

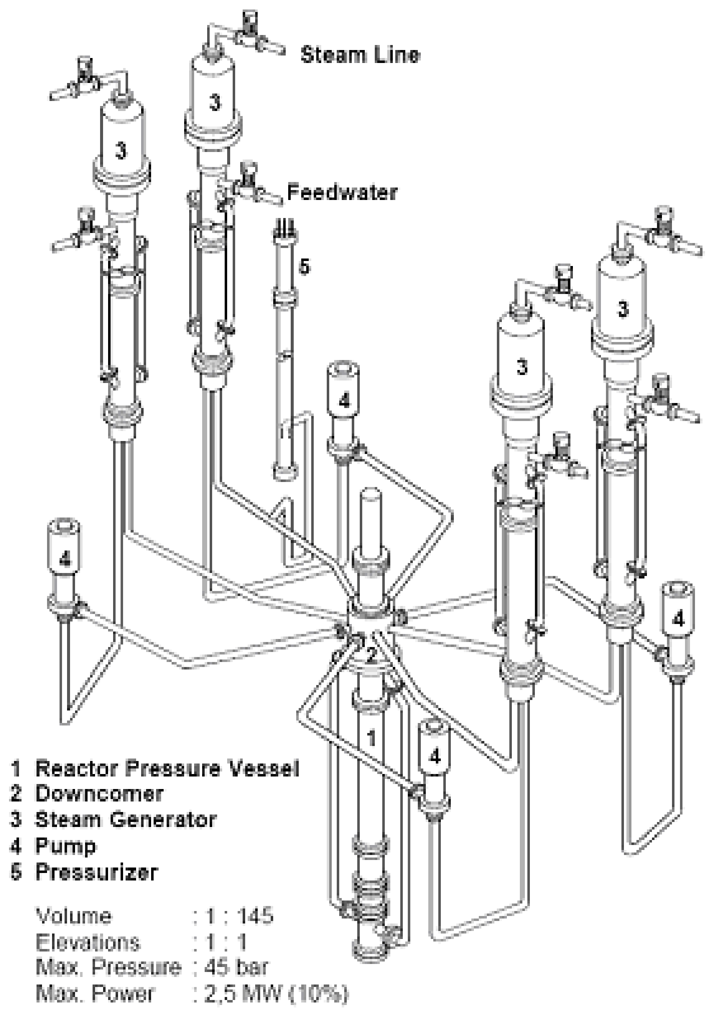

Figure 1 PKL Facility

3 WESTINGHOUSE DESIGN REACTOR AND RELAP5 MODEL DESCRIPTION

The plant of reference used in this study is a three loop Westinghouse Pressure Water Reactor (PWR) design. The system is designed to guarantee a power of 2900 MWt. In each one of the three loops, there is a steam generator to transfer the heat generated in the core from primary to secondary side. The pressurizer is connected to the hot leg of loop one and maintains a pressure of 157 bars during full power operation. Figure 2 presents the RELAP-5 nodalization for the primary circuit and secondary side, respectively.

To perform safety injections, there is an accumulator in each one of the three cold legs of primary side. Accumulators are prepared to inject borated water into the primary circuit to a pressure of 44.7 bars and with a temperature of 323 ºK (50 ºC) if the plant is working at full power operation. The other path to inject water into the primary circuit is via the residual heat removal (RHR) system. This system has two identical trains and can work in recirculation or injection modes. When working in the recirculation mode the RHR system extracts water from hot legs of loop 1 and 2, which is cooled and injected again in the cold legs of the three loops. Figure 2 shows the location of the RHR suctions and injections. High pressure safety injection (HPIS) and low-pressure safety injection (LPIS) are also performed using the RHR trains just changing the pump used. In this mode of operation, the water from the Reactor water storage tank (RWST) is injected into the three loops of the primary circuit at the locations showed in Figure 2.

In the RELAP 5 model built to simulate the transient, HPIS and LPIS are modeled as boundary conditions using a time dependent junction connected with a time dependent volume in each one of the tree loops. The same model has been used to model the main and auxiliary feed water system in the steam generators secondary side.

The transient has been simulated using RELAP5-Mod 3.3 code [Ref. 11] and SNAP [Ref. 12]. The RELAP5 model used consists of 865 hydraulic volumes, 890 junctions and 512 heat structures. This model has been adapted to simulate the G7.1 conditions.

The core is modelled using a pipe component of six volumes that contains the fuel rods, which are simulated using a heat structure component that generates the residual power corresponding to the transient initial conditions. The vessel downcomer is simulated in the RELAP model by means of an external pipe. Cold legs of all three loops are built pipe and branch components, which are connected to the vessel downcomer and upper plenum by two branches. The facility has a by-pass in the vessel upper head too.

The three 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 break is simulated in this model by a VALVE component of 1.5% of the area of adjacent pipe connected to a time dependent volume which simulates containment conditions. The valve opens at start of the transient activated by a trip. The break is located in the hot leg of loop 1 as shown in Figure 2.

Depressurization occurs in the steam head acting simultaneously in the three steam generators. RELAP model has been simulated with the trip valves 686, 786 and 886 (Figure 2) opening the three generators with the same rate of depressurization. The feedwater system is modeled as a boundary condition using a time dependent volume and a time dependent junction, stopping the water injection when the break is produced.

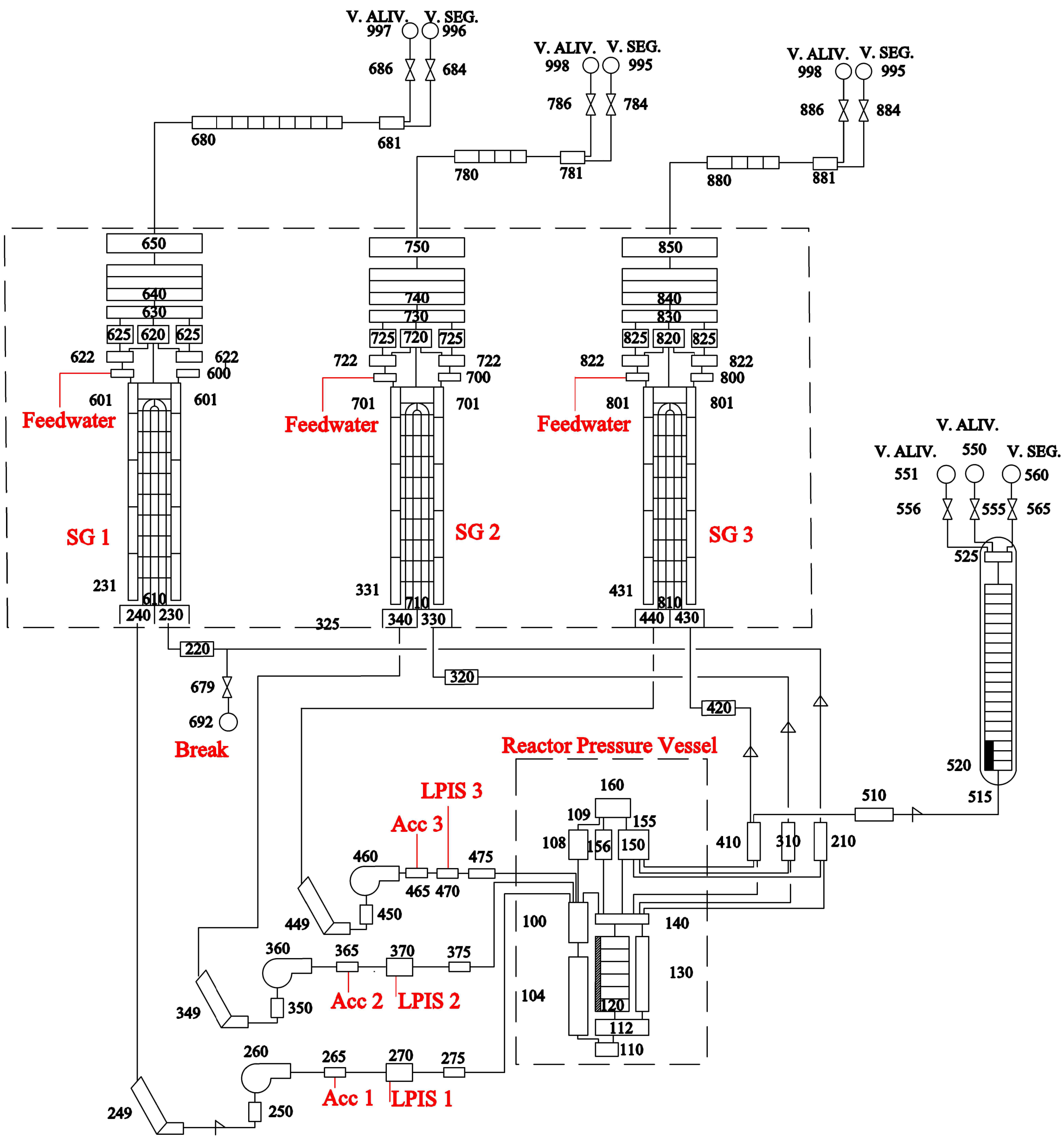

Figure 2 Westinghouse Design Reactor RELAP Model

4 DESCRIPTION OF G7.1 INITIAL AND TRANSIENT CONDITIONS

The initiating event in G7.1 experiment conducted at PKL facility [Ref. 13] is a small break LOCA in the hot leg of the reactor coolant system followed by a total failure of the high-pressure injection system (HPIS) together with the failure of the automatic steam generator secondary-side cooldown. Under such conditions the accident mitigation procedure proposed is the manual depressurization of the secondary side steam generators, followed by the injection from accumulators.

The plant emergency procedure in case of a small break LOCA (SBLOCA) require safety injection, where usually the High-Pressure Injection System starts at first, before the actuation of accumulators and finally the low-pressure Injection system is activated when pressure set points are met. Moreover, the emergency procedures of PWR designs during this kind of accident foresees a parallel cooldown of the primary side via steam generators secondary side to achieve cold shut-down condition. This cooldown process may be initiated automatically or manually and may be performed either partially or completely, depending on PWR plant design.

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

- Primary midloop inventory, primary closed and in operation at the maximum pressure of facility, 45 bar.
- Power 565 kW, equivalent to 1.8% of the residual power including the compensation for loss of heat.
- The initiating event is a break of 1.5% upwards in hot leg of loop 1.
- The secondary side of steam generators are filled with water and in operation with a level of 12.5 m, and a pressure at the beginning of the transient of 43.7 bar.
- All steam generators are connected via the main steam header.
- 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 ET 0 a ake ssible:
  - Cold-leg Accumulators injection at p = 26 bar
  - Cold-leg Low Pressure injection at p = 8 bar

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, 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 during the transient.

Table 1 G7.1 Initial Conditions Based on Experimental Data at PKL

	Initial Conditions
Primary side	
Coolant inventory	995 kg aprox.
PRZ Level	0.8 m
Temperature at core outlet	530 K
PRZ Pressure	45 bar
Core Power	455 kW, aprox 1.8 %
RCS Flow rate	Reflux-condenser conditions
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 side	
SG Level	4 SG Filled with water (12.5 m) and in operation
SG Temperature	529 K
SG Pressure	43.7 bar

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

Time (sec)	**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

The expected evolution of primary and secondary side pressures and the temperatures at the exit of the core and the maximum clad temperature during the transient are shown Figure 3 [Ref. 13]. In this figure, the primary depressurization due to the break opening is observed, and also the increase of core exit temperature and the start of secondary depressurization, which start when primary pressure is lower than secondary side and primary temperature reach 623 K. The secondary side depressurization makes the CET and PCT values decrease and the conditions to start accumulators and LPIS injections are met and the plant is maintained under safe conditions.

In particular, in G7.1 transient the following phenomena and actions were investigated.

- Effectiveness of Accidental Management measures.
- Core uncovery due to boil-off with generation of superheated steam.
- Primary-side pressure behavior before and after occurrence of core uncover.
- SG depressurization based on CET and influence on primary pressure / ACC injection.
- ACC injection after SG depressurization and influence on core cooling.
- Relation between PCT and CET during these processes.

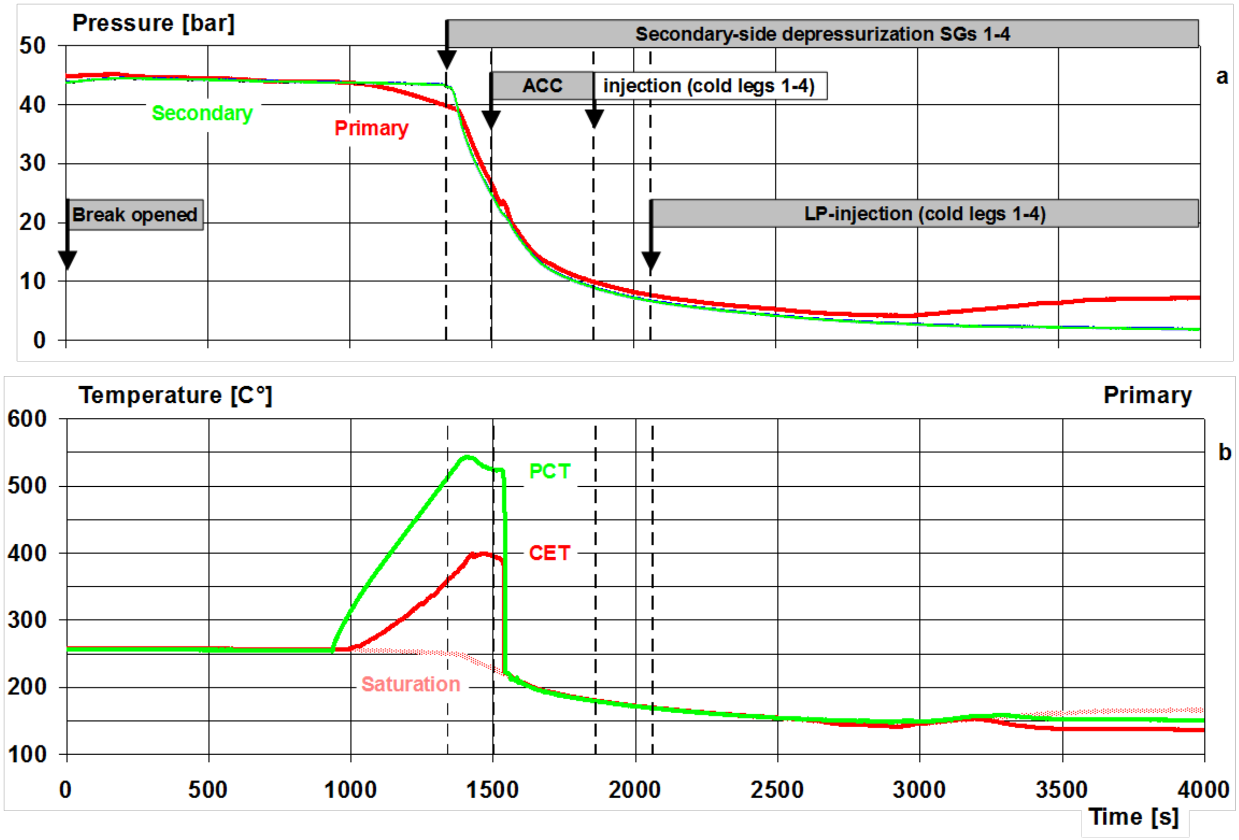

Figure 3 Primary and Secondary Pressure and CET and PCT Temperature in Test G7.1

5 RELAP5 MODEL OF WESTINGHOUSE DESIGN REACTOR. ADJUSTEMENT TO THE CONDITIONS OF TEST

The PKL test facility represents a KONVOI design 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. On the other hand, the plant considered in this study is a three loop Westinghouse design. In order to adjust the conditions of experiment G71 performed at PKL facility to the Westinghouse plant, some assumptions regarding scale and design aspects must be taken into consideration.

First, boundary and initial conditions of the transient to be simulated at the Westinghouse reactor must be defined. A scheme of the procedure used is shown in Figure 4. Thus, taking into consideration the scaling between PKL and the KONVOI reactor the values for the initial conditions for this latter plant have been obtained. After that, considering the nominal values of KONVOI and Westinghouse plants scaling relationships between them were obtained, so the values for the initial conditions of the Westinghouse reactor were established using such relationships.

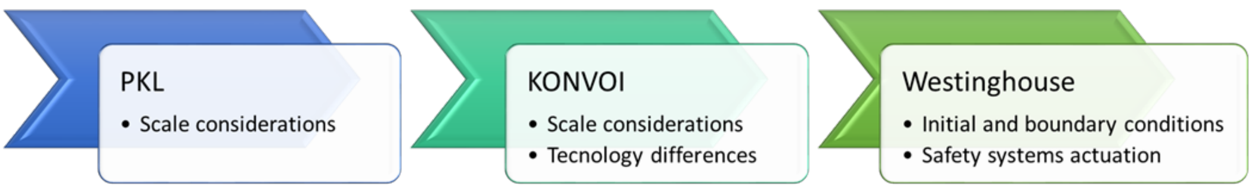

Figure 4 Procedure to Obtain the Initial and Boundary Conditions for a Westinghouse Reactor

As said before, the power relation between PKL facility and its reference power plant is 1:145. Thus, as PKL full power is 25000 KW, the nominal power for the KONVOI plant is 3625 MW of thermal power. Westinghouse nuclear power plant has a value nominal of 2940 MW of thermal power, so a relation between KONVOI and Westinghouse design plants of 0.811 for power is obtained. This relationship is used to determine Westinghouse reactor power during the transient.

The relationship in terms of pressure, drop of pressure and heights are maintained 1:1. Thus, to determine the initial values for pressure and temperature in Westinghouse nuclear power plant the 1:1 relation is used, as shown in Table 3. Finally, it is necessary to obtain a volume scaling with full height full pressure approach to preserve time scale of physical phenomena [Ref. 14, Ref. 15]. This is especially important for a fast response phenomenon and for the establishment of other variables as length, hydraulic diameter, velocity, heat generation rate/volume, fluid temperature and non-dimensional characteristics of pumps and valves. PKL and KONVOI is 1:145 in volume, and the values for the KONVOI plant can be easily obtained. However, to obtain the values of Westinghouse nuclear power plant the differences in the reactor technology should be considered.

Other differences in the technology leads to different configurations of the safety systems available in both reactors as well as the actuation of such safety systems. This technological difference is important to define the accident management measures that actuate along the transient, as for example the safety injections mass flow rate, so, in order to preserve the specific characteristics of Westinghouse design power plant the following assumptions were made:

- Number of loops. At PKL facility, the reactor coolant system consists of four loops, while Westinghouse nuclear power plant is a three-loop design.
- Steam generators. Because of the different number of loops, and consequently the different number of steam generators, the following assumptions were made:
    - Depressurization of secondary side is performed through three valves. PKL use two relieve valves (2x19.2 mm). To obtain the equivalent area, it has been taken into account the volume (area) conversion factor and in addition, due to the different number of loops, an additional factor to assure a correct distribution of massflow is contemplated maintaining the equivalent depressurization rate (factor of 4/3). The valve obtained matches the specifications of Westinghouse PWR.
- Residual power. In G7.1 experiment the residual power generated is 1.8 % of the nominal power for the experiment, which corresponds to 53505.725 kW for Westinghouse nuclear power plant. This residual power remains constant during all transient.
- Accumulators. The working pressure of accumulators have been adapted to conditions of experiment from 45 bars in normal conditions to 26.6 for this case.
- Low Pressure Injection System. For the Westinghouse NPP three injections were considered. The mass flow rate injected by LPIS in the Westinghouse NPP has been adjusted based on the volumetric conversion factor between PKL and Westinghouse.
- Break. As the initiating event is a break of 1.5% upwards in hot leg of loop 1, the total break area has been adjusted based on the number of loop and diameter conversion factor to obtain an equivalent area.

The values of the initial conditions for the transient experiment in PKL and the corresponding values for the Westinghouse simulation are presented in Table 3.

Table 3 Initial Conditions of PKL vs. Westinghouse NPP

	PKL	***KONVOI***	***Westinghouse***
Full Power (kW)	25000	3625000	2940000
Residual Power 1.8% (kW)	455	65975	53505.725
PRIMARY			
PRZ Pressure (bar)	45	45	45
Core exit Temperature (ºK)	530	530	530
Pressurizer Level (m)	0.8	0.8	0.8
SECONDARY			
SG Pressure (bars)	43.7	43.7	43.7
SG Temperature (K)	529	529	529
SG Level (m)	12.5	12.5	12.5

6 SMALL BREAK LOCA IN HOT LEG SIMULATION RESULTS

The transient evolution can be divided in four phases. In the first one, the main objective is to study core uncover and the primary pressure drop below the secondary pressure and the relationship between CET and PCT. In the second phase, once the CET reaches 623 K (350 ºC) secondary side depressurization is studied. The decrease in the primary side pressure makes possible the injection through accumulators in the third phase, and finally in the fourth phase the LPIS injection is activated. In Figure 5, the main milestones during the transient evolution and their approximate timing are summarized.

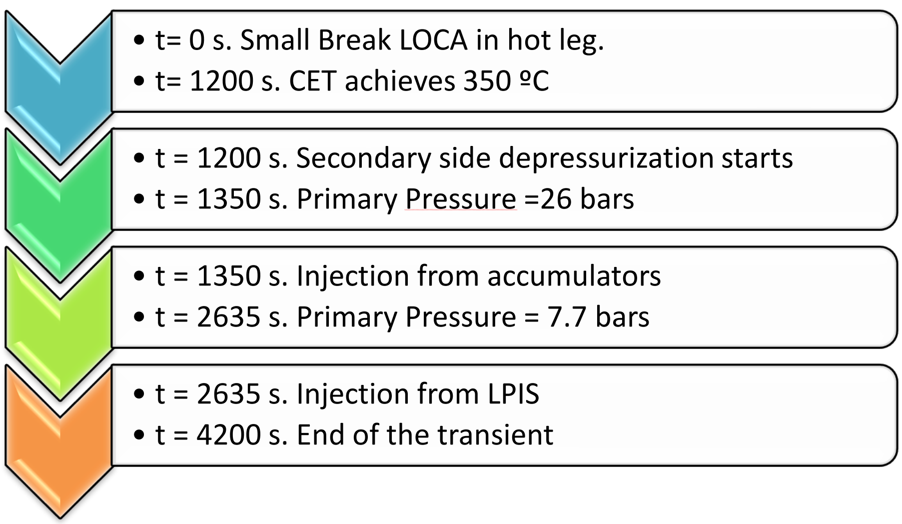

Figure 5 Main Milestones During the Transient Evolution

Once the small break LOCA occurs the pressure of primary starts to decrease due to the loss of inventory through the break, first slowly because the secondary side can still act as a heat sink, but when the primary pressure is lower than the secondary pressure, the primary pressure drops with a higher rate than before, as shown in Figure 6. Approximately, since 500 s heat transfer from primary to secondary side through steam generators is interrupted, and the only heat sink is the loss of coolant through the break.

The continuous loss of coolant, and the residual heat generated leads to core uncover at 1000 s of the start of the transient approximately. At this time, the temperature of the coolant at the core exit (CET) and the maximum fuel clad temperature (PCT) increase significantly, as can be observed in Figure 7.

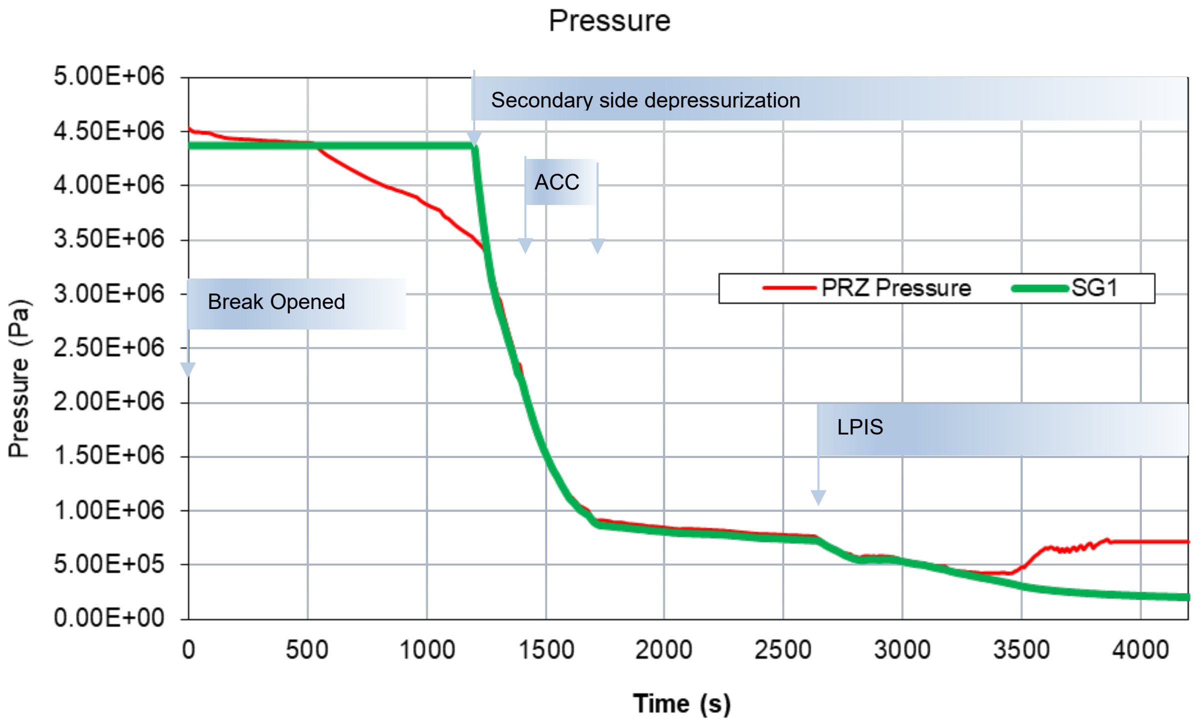

Figure 6 Pressure Pressurizer and SG1

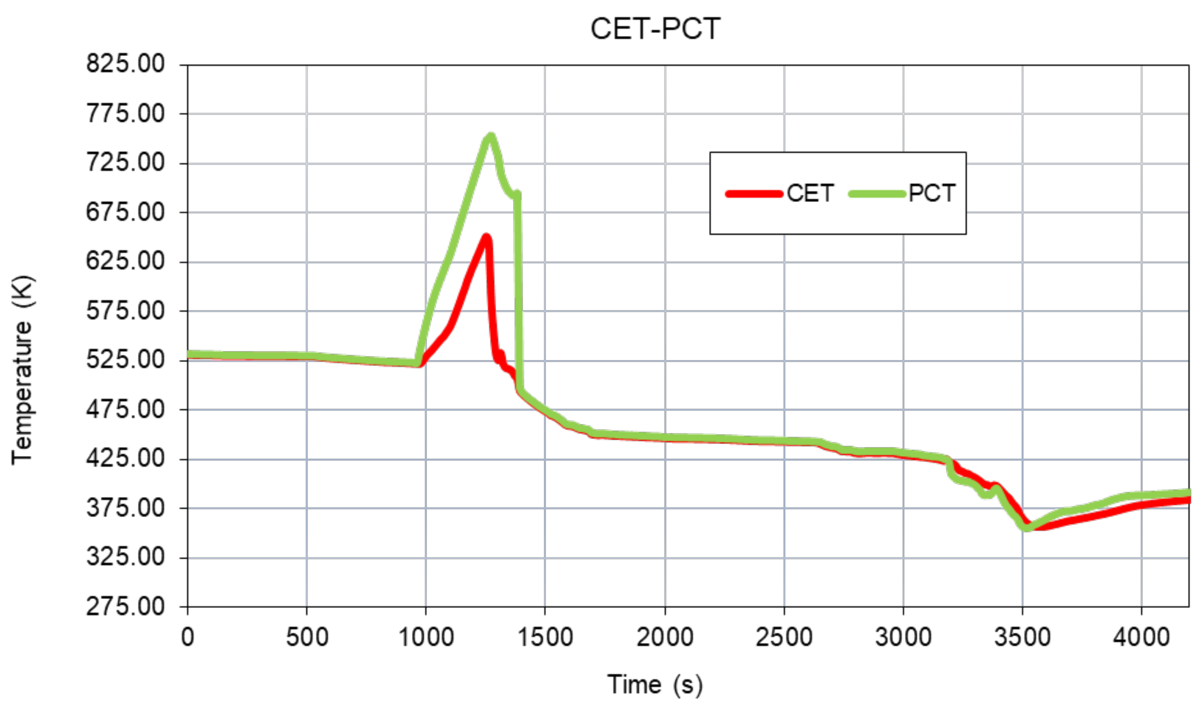

Figure 7 Core Temperatures

A secondary side depressurization is considered as accident management measure, with the objective of decreasing such CET and PCT, what makes possible to reestablish steam generators secondary sides as heat sink. This depressurization starts when CET reaches 623 K (350 ºC) as the measurement of the real value of PCT is difficult to obtain. As observed in Figure 6, secondary side depressurization stars at 1200 s, and produces a reduction of primary side pressure (see Figure 6) and a fast decrease of CET and PCT values (see Figure 7). The depressurization occurs simultaneously with the manual activation of the auxiliary feedwater system in each SG to guarantee there is enough coolant in the steam generators secondary side to remove the residual heat.

During the rest of the transient the secondary side act as heat sink again, controlling the pressure of the primary circuit until the injections through accumulators and LPIS are produced. Such injections are able to recover the primary inventory and the plant reaches a safe situation (see Figure 6).

In Figure 7, a significant difference between CET and PCT values is observed. It is of great importance to know the relationship between CET and PCT as CET measurement is used to trigger the emergency systems actuation to prevent core damage, which is evaluated using the PCT value. Thus, cladding oxidation, hydrogen production and core melting depend on the cladding temperature that has to be maintained below safety limits.

As the axial power profile is not homogeneous the most conservative axial level should be selected to compare the CET values with the corresponding PCT values. Thus, Figure 8 presents the PCT evolution at different axial levels and shows that the most limiting values are obtained for the sixth axial level, corresponding to the upper part of the fuel assembly. In this figure, it can be observed that CET set point is reached at 1200 s approximately, depressurization starts at this time but the effect of the AM measure is not evidenced until 50 s later when the CET reaches the maximum value of 650K (1250 s). Regarding the PCT evolution, the maximum value of 754 K and is reached 70 s after the secondary side depressurization starts (1270 s), as shown in Figure 8.

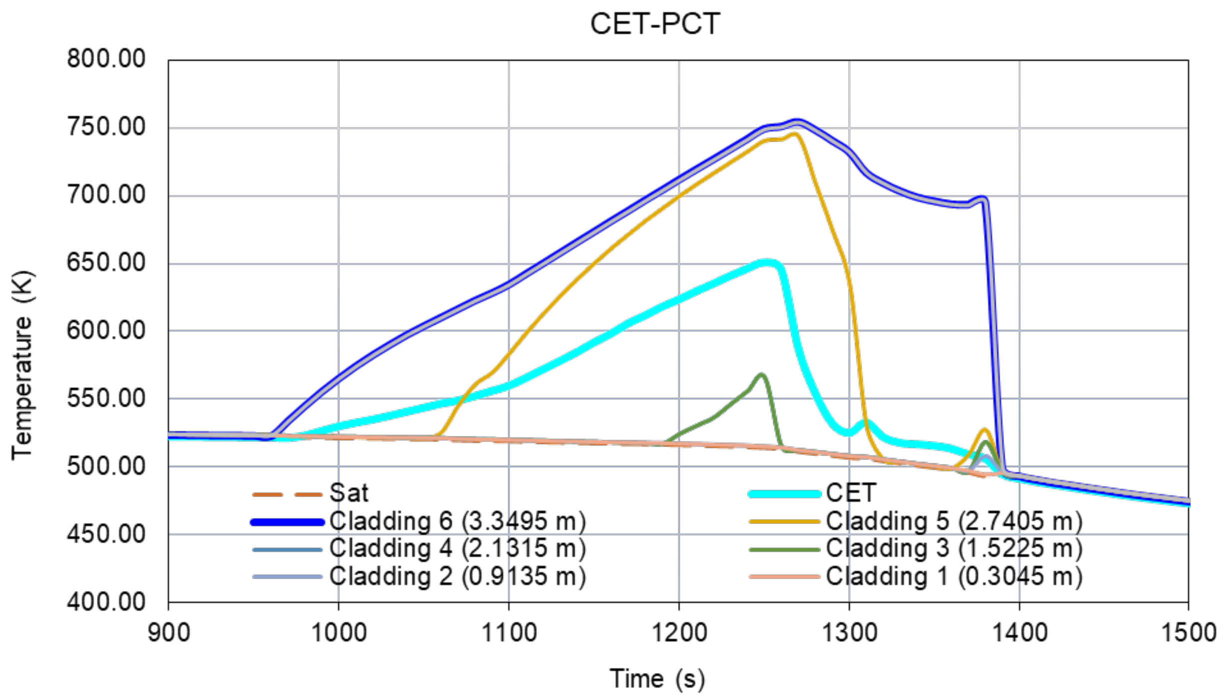

Figure 8 CET and Cladding temperatures

As said before, it is important to obtain the relationship between CET and PCT to guarantee the AM measures will maintain the plant under safe conditions. Figure 9 shows this relationship until the maximum values are reached. This figure confirms the quasi-linear relationship between these two parameters and permits the use of CET as key risk indicator instead of PCT.

Regarding the secondary side, the depressurization is produced through the main steam line (MSL) valves of the three steam generators at 1200 sec, MSL1 MSL2 and MSL3 in Figure 10, and this leads to a sharp decrease in the steam generators secondary side water level. However, the feedwater system actuation is able to recover the inventory lost through the MSL.

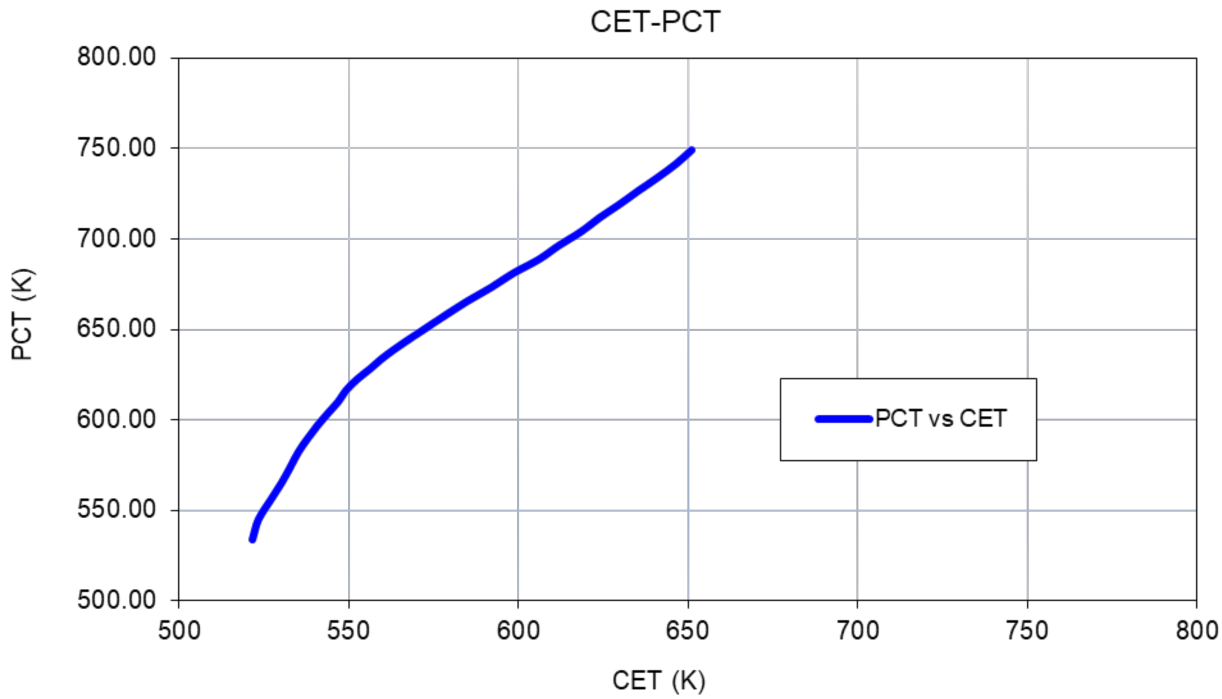

Figure 9 CET vs PCT

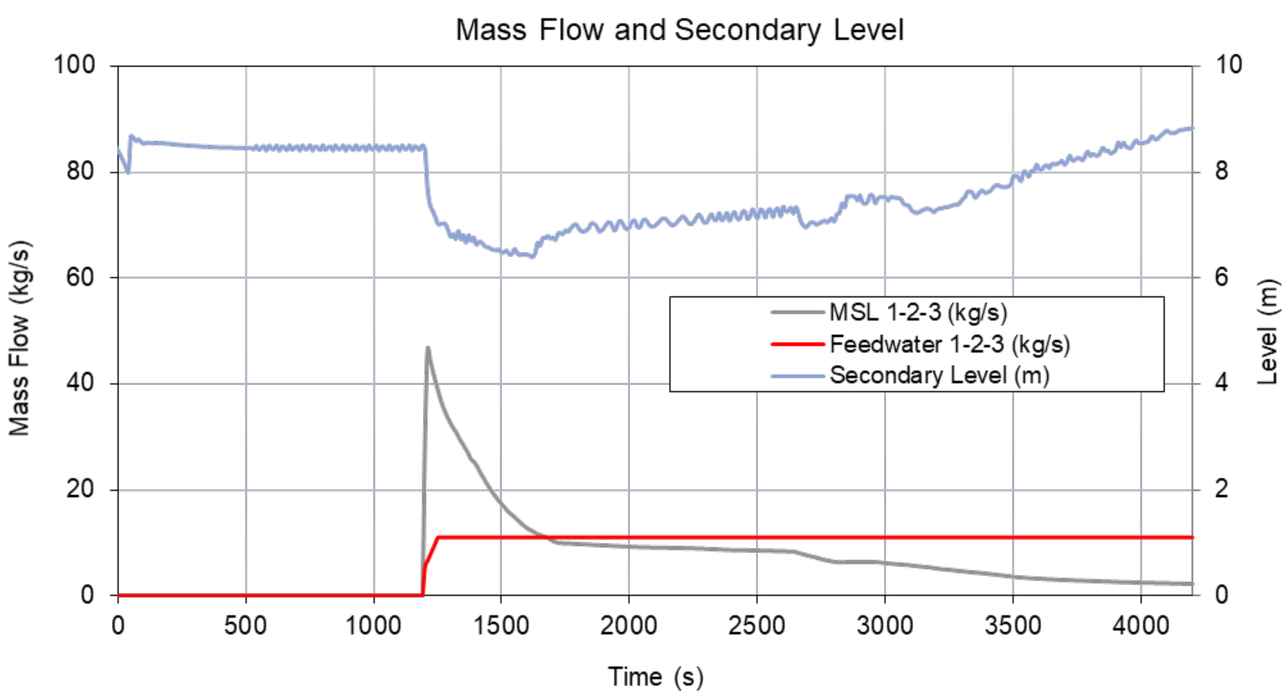

Figure 10 MSL, Feedwater massflows and Secondary Level

The coolant inventory in the primary circuit also decreases due to the water lost through the break, as can be observed in Figure 11. The secondary side depressurization produces the primary pressure to decrease (see Figure 6) until the conditions for safety injections are met. Thus, around 1300 s the accumulators pressure set point of 26 bar is reached and safety injection starts and the coolant inventory in the primary circuit rises until 1700 s and descends again (see Figure 6 and Figure 11). As coolant continues to flow through the break the coolant mass descends until primary pressure reaches 7.7 bar (see Figure 6), when LPIS starts injecting water in all three cold legs recovering the coolant inventory, as shown in Figure 11, and maintaining the temperature in a stable situation (see Figure 7).

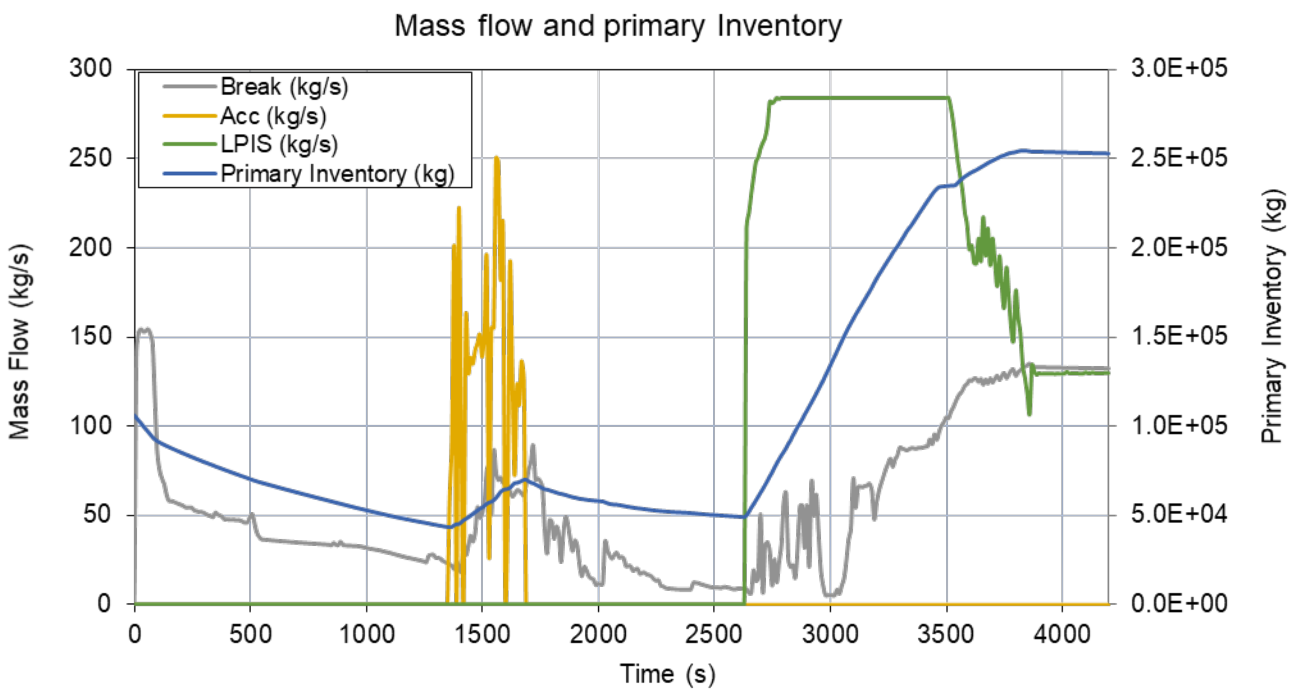

Figure 11 Break, Accumulators, LPIS Mass Flows and RCS Inventory

Finally, when the secondary side depressurization starts flashing is observed inside the core region as there is an evaporation for expansion. This situation is quickly resolved by accumulator injections that recovers the core region, shown in Figure 12 where is represented the density inside the core at different levels (core 1, lower, to core 6, higher).

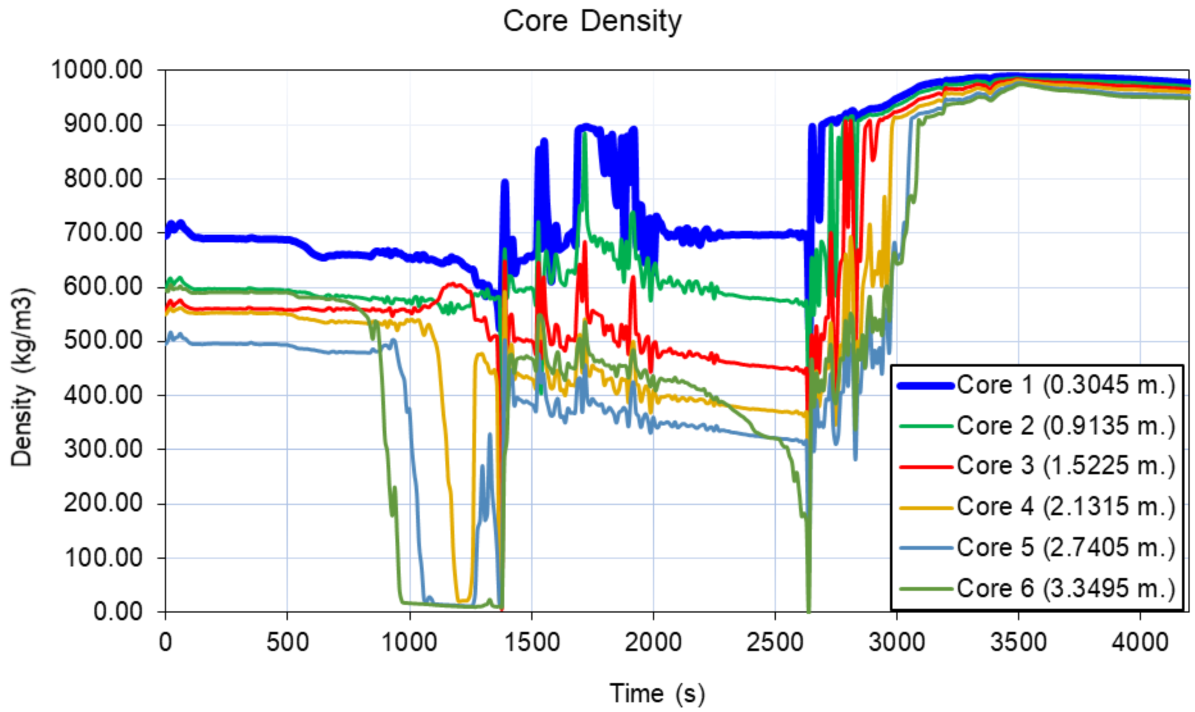

Figure 12   Core Density

7 RUN STATISTICS

The simulation of G7.1 experiment applied to a Westinghouse NPP has 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.

Table 4 summarizes the relevant parameters of the run statistics of the simulation of experiment G7.1 Plant Application.

Table 4 Run Statistics

	RT	CPU	TS	CPU/RT	C	DT	GT
Depressurization Secondary Side	1200	73687.47	0.01	13.1585	600	835641	0.1470
Accumulator Injections	1350	120626.38	0.01	12.0626	600	1349621	0.1490
LPIS injections	2640	224533.27	0.01	10.3663	600	2519785	0.1485
End of transient	4200	908540.71	0.005	26.1300	600	12520282	0.1209

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*10^3/(C*DT))

8 CONCLUSIONS

This work presents the results of the simulation of a small break loss of coolant accident with primary circuit closed and midloop inventory for a three loop Westinghouse NPP with HPIS failure. This experiment has been previously developed in the PKL Test Facility (G7.1 experiment).

It has been necessary to take into account a number of considerations and changes with regards to the G7.1 experiment due to the different scale in terms of volume and power and especially due to the different technology. Thus, PKL represents a four loop KONVOI NPP while Westinghouse design reactor considered has three loops.

From the results obtained in the previous sections, it can be concluded that the main phenomenology is reproduced in the Westinghouse NPP application, where the Primary and Secondary pressures, levels and temperatures follow the evolution of the experiment.

It has been appreciated a great influence of secondary-side depressurization as a measure to consider depressurizing the primary, avoiding high temperatures in the core and permitting the actuation of accumulators and LPIS.

A certain flashing in the core has been observed for the fast depressurization of primary.

Once the accumulators and LPIS act it is established a stable situation with a rewetting of the core.

Respect the differences between PCT and CET, there is a time delay of 120 s (related to 350 ºC) and a maximum temperature difference of 103 K in the Westinghouse simulation, while in the PKL experiment the differences were of 270 s and 160 K respectively.

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

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10. SUPPLEMENTARY NOTES
K.Tien, NRC Project Manager

11. ABSTRACT (200 words or less)
This paper focuses on the simulation in a Westinghouse design nuclear power plant of the experiment G7.1 conducted at the PKL facility. The test G7.1 consists of a Small Break-Loss of Coolant Accident (SB-LOCA) in the Hot Leg with a total failure of the high-pressure injection system (HPIS). The PKL 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, so some assumptions regarding the scaling and design features of the Westinghouse reactors should be taken into consideration to undertake the simulation. The postulated additional system failures (no HPIS and no automatic secondary-side cooldown) make necessary to determine the Action Management (AM) measures to prevent core-melt scenario. Under such conditions, the accident mitigation procedure proposed is the manual depressurization of the secondary side steam generators followed by injection from accumulators. The simulation of the transient for a Westinghouse reactor has been performed using RELAP-5 thermal-hydraulic code.

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) Asociación Española de la Industria Eléctrica (UNESA)	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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