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Post-Test Analysis of PKL III Test H2.2 Run 2 (SBO) with TRACE

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ABSTRACT

This document presents and explains the main results obtained in the simulations performed with the TRACE consolidated thermal-hydraulic code in relation to the PKL III H2.2 Run 2 test. The H2.2 Run 2 test deals with a Station Blackout (SBO) in which three main phases take place. The first one in which the secondary circuit is depressurized, the second one in which the primary side is depressurized and the third one in which the emergency feedwater system (EFWS) is activated. The main goal is to analyze the capability of the TRACE V5.0 p4 code to accurately simulate the events and phenomena encountered throughout the studied transient. The activation of the different systems and the values of the main parameters will be used to assess the capabilities of the PKL model developed with the TRACE code.

FOREWORD

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

- Assessments 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;
- Analytical information in support of 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
- Assessments 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-80s and comes to the current days through several periods of USNRC CAMP programs. During this long history, CSN has promoted coordinated joint efforts with Spanish organizations through different periods of the so-called CAMP-España, the associated national program.

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

Many experimental facilities have contributed to 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 has been made in the past. On the basis of the SESAR/SFEAR[1] reports "*Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk" (SESAR/FAP, 2001),* "*Support Facilities for Existing and Advanced Reactors (SFEAR) NEA/CSNI/R(2007)6"*, and 2019 updated SESAR/SFEAR2 report, 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 several experimental facilities and programmes of potential interest for present or future international collaboration within the nuclear safety community during the coming

1 SESAR/SFEAR is the *Senior Expert Group on Safety Research / Support Facilities for Existing and Advanced Reactors* of NEA Committee on the Safety of Nuclear Installations (CSNI).

decade. The different series of PKL, ROSA, ATLAS, and RBHT projects are under these premises.

CSN, as the 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 databases 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:

- The analysis, simulation, and investigation of specific safety aspects of PKL/OECD and ATLAS/OECD experiments.

- The analysis of applicability and/or extension of the results of 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, TRACE, and/or PARCS).

An additional goal of CSN is to assure and 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 nation's grid for the next coming years is expected to be maintained with plants of extended life and/or higher power. This is the challenge that will require continued effort.

Javier Dies Llovera, Commisioner

Nuclear Safety Council (CSN) of Spain

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EXECUTIVE SUMMARY

This document analyzes the main results obtained with the NRC TRACE consolidated code for the simulation of the OECD/NEA PKL III project H2.2 Run 2 test of the OECD/NEA PKL III project performed at the PKL Primärkreislauf-Versuchsanlage (primary coolant loop test facility). This facility is owned by AREVA NP and is located in Erlangen (Germany). The main objective of the PKL III H test program is to clarify open questions on the issues of nuclear power plant safety assessment with a focus on the events that happened at the Japanese Fukushima power plant. The subject complex "accidents/sequences of events beyond the design basis with significant core heating" therefore constitutes a central focus within the PKL III H test program.

The PKL facility consists of the entire primary side and the most significant parts of the secondary side of a pressurized water reactor at a height scale of 1:1, with volumes, powers and mass flows at a scale of 1:145. The experimental setup consists of four primary loops, each with a reactor coolant pump (RCP) and a steam generator (SG), so that these loops are arranged symmetrically around the reactor pressure vessel (RPV). The spectrum of scenarios, investigations and transients that are reproduced in this facility cover a wide range of studies. Simulations of accident scenarios with large, medium, and small ruptures are reproduced, shutdown procedures after a wide variety of accidents are investigated, there are studies of different complex thermal-hydraulic phenomena, etc.

The results obtained in the simulations carried out with the model designed for the PKL installation using the TRACE code are presented in relation to the PKL III H2.2 Run 2 test. The test in question reproduces a Station Blackout (SBO) scenario. Prior to the execution of the test, a conditioning process of the installation is performed and the test is started under fully stable conditions. After finding the appropriate conditions, the test has three distinct phases. In the first phase the secondary circuit is depressurized, in the next phase the primary side is depressurized and in the third phase the emergency feed water system (EFWS) is activated. The TRACE V5.0 p4 code is used to analyze the simulation capacity of the events and phenomena encountered during the transient under study. The activation of the different systems and the values of the main parameters will be used to evaluate the capabilities of the PKL model developed in TRACE.

ACKNOWLEDGMENTS

The thermal-hydraulic and nuclear engineering group of the UPV is indebted to the management board of the PKL III project and to the people of the Thermal-Hydraulics Safety Research Group of AREVA. The authors are grateful to the CSN, who financed this study and help to develop this report, and to the “Grupo de Análisis Dinámico de Sistemas Energéticos del Instituto de Técnicas Energéticas de la Universidad Politécnica de Cataluña”, who provided the TRACE model of the original PKL III facility from which the current model has evolved.

ABBREVIATIONS AND ACRONYMS

ACC	Accumulator
CET	Core Exit Temperature
CVCS	Chemical and Volumetric Control System
EFWS	Emergency Feedwater System
FWS	Feedwater System
HPSI	High Pressure Safety Injection System
LPSI	Low Pressure Safety Injection System
PCT	Peak Cladding Temperature
PDE	Primary Side Depressurization
PKL	Primaerkreislauf (Primary Circuit)
PRZ	Pressurizer
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RPV	Reactor Pressure Vessel
RV	Relief Valve
SBO	Station Blackout
SDE	Secondary Side Depressurization
SG	Steam Generator
SNAP	Symbolic Nuclear Analysis Package
SOT	Start Of Test
SV	Safety Valve
TRACE	TRAC/RELAP Advanced Computational Engine
UPTF	Upper Plenum Test Facility

1 INTRODUCTION

The present study consists of a Station Blackout (SBO) case simulation, a transient based on the loss of power supply in a nuclear power plant. This test is divided into four distinct phases, the first one of pre-test or conditioning phase and three others of the transient itself, referred to as A, B and C from this point of the document. The test has been reproduced in the PKL experimental facility, and its behavior has been simulated using the TRACE thermal-hydraulic code, modeling the plant and configuring it to simulate the test in question. Finally, a comparison is made between the results obtained in the experimental measurements and those obtained with TRACE.

This work-study contributed to the OCDE international collaborative research project PKL. The Spanish contribution was coordinated by the Spanish Nuclear Regulatory Commission (CSN) with the contribution of the Spanish Electricity Producers Association (UNESA). A consortium formed by the CSN, several Spanish Technical Universities and UNESA developed the Spanish participation in the project that the CSN and a steering committee coordinated.

The PKL facility is owned and operated by AREVA NP in Erlangen, Germany. The analysis of the experiment PKL H2.2 Run 2 [1] with the TRACE code was assigned to the "thermal-hydraulics and nuclear engineering group" of the Polytechnic University of Valencia. The PKL test have been performed in the "Primärkreislauf-Versuchsanlage" (primary coolant loop test facility) PKL. The TRACE code employed to perform the simulation was the Version 5.0 p4 and the SNAP interface version 2.6.1.

This work aims to compare the experimental results with the predictions of the TRACE code and analyze the set of phenomena that take place at the primary and secondary sides of the Steam Generator (SG). Particular emphasis is devoted to analyzing and comparing the code results and the experimental data [2] [3].

This study has been divided into five sections: section 1 is a general introduction, section 2 is a description of PKL facility and test H2.2 Run 2, section 3 defines the model of the installation implemented in TRACE, and section 4 deals with the study of the transient sequence and the comparison of TRACE results for the primary physical magnitudes versus the experimental data and section 5 is devoted to present the conclusions.

2 INITIAL AND BOUNDARY CONDITIONS

2.1 Description of the PKL Facility

The PKL experimental plant (Figure 2-1) is a scaled-up installation of a commercial PWR-type reactor. It models the entire primary system with the four loops and the essential parts of the secondary circuit of a 1300 MW (electrical) nuclear power plant at a scale of 1:145; specifically, it is based on unit 2 of the Philippsburg plant [4].

This type of installation is used to investigate the behavior of the thermal-hydraulic system under accident situations. The accidents that can be studied are with and without loss of coolant. In accidents with accidental loss of coolant, the power generated in the reactor core is more difficult to extract due to the reduction of the amount of coolant and as a consequence the fuel temperature rises and the established safety limits may be exceeded. If conditions prevail for long enough, temperatures that lead to core meltdown can be reached. On the other hand, when an accident is not a consequence of coolant loss, control rod insertions can also occur due to increased core reactivity. In these cases there can be many initiating events that can cause a disturbance in the neutron balance and the consequent uncontrolled increase in power. Due to the importance in transients, certain design parameters have to be contemplated. As essential considerations, all elevations in PKL correspond to the actual reactor dimensions. The Froude number has been maintained to take into account all important phenomena in hot branches, such as flow separation in the hot branch experiment design conditions. In addition, the experiments performed in the UPTF facility have been taken into account in the design of the PKL facility. The UPTF consists of a full-size upper plenum and the experimental results from these 1:1 scale in height facilities can significantly reduce the uncertainty of geometric scaling with respect to experimental results obtained in downscaled facilities. In the UPTF facility, countercurrent flow limit, fluid mixing and the development of fluid and wall temperature fields in the cold leg and downcomer region of a PWR phenomena have been studied and events such as LOCAs or valve and pump blockages have been reproduced. Then, events reproduced in scaled-down facilities, such as PKL or UPTF, should be extrapolated to the reactor scale to evaluate the thermal-hydraulic behavior of the full-size reactor. However, there is some uncertainty in this process, so it is necessary to continue analyzing different transients in these facilities at scale and check as far as possible their applicability to real plants. This, given the impossibility of performing the transients in real plants, is often carried out through the use of codes, such as TRACE.

In the full-scale experimental facilities, the behavior of the overall thermal hydraulics of the commercial power plant should be simulated as realistic as possible. A broader set of more specific and detailed features that serve to meet this requirement are specified below:

- Full-scale hydrostatic head.
- Power, volume and cross-sectional area with a scale factor of 1:145.
- Full-scale friction pressure loss for single-phase flow.
- Simulation of the four loops.

- The core and steam generators are simulated as a "section" of the actual systems; in other words, full-scale dimensions of bars, U-tubes and spacers are used. However, the number of bars and tubes is reduced.

- The hot legs' design geometry is based on Froude number conservation and full-scale UPTF experiments.

- The vessel downcomer is modeled as a ring in the upper region and continues with two pipes connecting to the lower plenum. This configuration allows the symmetrical connection of the four cold legs to the RPV, preserving frictional pressure losses without affecting the reference plant's volume and surface area ratio.

The operating pressure of the PKL plant is limited to 50 bar on the primary side and 56 bar on the secondary side. These settings allow simulation over a wide temperature range (250°C), particularly applicable to the investigated cooling procedures.

In Figure 2-1, the schematic of the experimental PKL installation with its main components is represented.

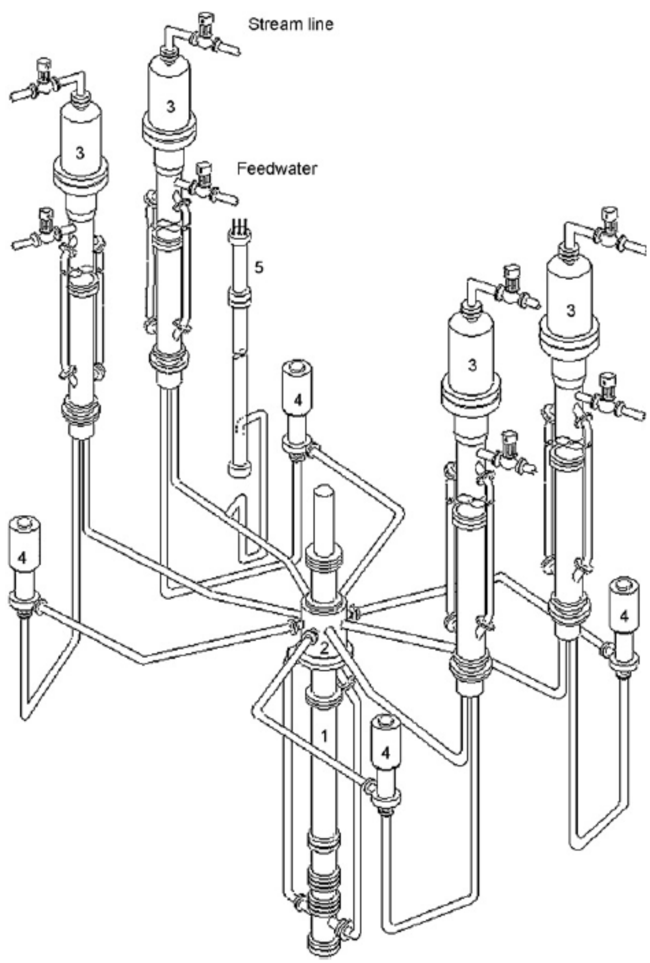

Figure 2-1 PKL Integral Installation Schematic: (1) Reactor Vessel; (2) Downcomer; (3) Steam Generators; (4) Pumps; (5) Pressurizer. Scaling: Volume 1:145 and Elevations 1:1; Maximum power 2.5 MW

2.1.1 Main Emergency Cooling, Waste Heat Removal and Pressure Relief Systems of the PKL Facility

During the course of the Station Blackout (SBO) transient, which is the one to be analyzed in this facility, different emergency systems are put into operation to ensure core cooling to shut down the reactor, maintaining this shutdown situation and preventing the release of radioactive material.

Throughout the accident, different mechanisms with particular characteristics are activated. The four emergency cooling systems that operate in the PKL installation will be described, which are the residual heat removal system (RHRS), the high-pressure safety injection system (HPSI), the accumulator injection system and the low-pressure safety injection system (LPSI). The work performed by the pressure valves, both the safety valve (PRZ-SV) and the pressure relief valve (PRZ-RV), also play an essential role in the current test.

2.1.1.1 Residual Heat Removal System (RHRS)

In the PKL experimental facility, the residual heat removal system serves two main functions: removing decay heat in the core, heat during shutdown and residual heat from the core after a LOCA-type accident.

To meet these objectives, the RHRS mainly consists of the following components:

- A waste heat removal pump with a maximum flow rate of 9.66 kg/s and a working capacity at 75 meters of water column pressure.
- A heat exchanger with thermal power of 1500 kW.
- Two control valves (DN 80).
- Vortex flow meters, manual isolation valves, lift check valves and piping.

This system is used for operational and safety purposes and is subdivided into four trains. Each suction line is connected to a hot leg injection line, and each injection line is connected to a nozzle on the reactor coolant piping on the cold leg (between the RCPs and the vessel). The extracted coolant is cooled in the RHR heat exchanger keeping the total flow rate constant and reducing it according to the number of trains in operation. The temperature of the injected coolant is controlled to establish a given cooling rate (the control variable is the measured core outlet temperature).

2.1.1.2 High Pressure Safety Injection System (HPSI)

In nuclear power plants, the High Pressure Safety Injection (HPSI) system provides emergency core cooling mainly in LOCAs when the pressure in the system drops below 110 bar. The PKL installation is designed for a working pressure of only 50 bar. The HPSI system is modeled to start injecting emergency coolant into the core at 45 bar. The length of the injection period depends on the size of the rupture or the amount of coolant lost. If the primary side pressure drops below the limiting pressure for the LPSI signal during a high-pressure injection, the low-pressure injection is started in parallel to the HPSI.

At PKL, the HPSI system consists of the following main components:

- A safety injection pump whose main characteristics are a maximum flow rate of 1.72 kg/s and a capacity of up to 500 meters of water column pressure.

- A 10 m³ borated water storage tank, maintained at a temperature between 30 and 80°C at atmospheric pressure.

- Eight control valves (DN 15).

- Several orifices for characteristic flow measurements, lift check valves and piping.

Each train of the HPSI system is connected on one side to the hot branch injection line or suction line and on the other side to the cold branch injection nozzle of the waste heat removal system. The injection into the hot and/or cold branches is produced by 1, 2, 3 or 4 pumps, according to the reduced flow rates simulated in the PKL installation.

2.1.1.3 Accumulator Injection

PKL is equipped, like the reference plant, with eight accumulators. Two accumulators per reactor cooling loop are connected to the injection lines, one to the hot branch and one to the cold branch. The accumulators are designed for an operating pressure of 50 bar, causing injection to the primary circuit automatically when the primary circuit pressure drops below 26 bar.

2.1.1.4 Low Pressure Safety Injection System (LPSI)

With the LPSI signal in the course of a loss of coolant accident, all pumps, including the one in this system, are started for reactor waste heat removal in the experimental facility.

The pump starts injecting when the primary circuit is at a pressure of 10 bar. It operates according to its mass flow/supply pressure operating curves.

The following components constitute the LPSI system in the PKL facility:

- A low pressure safety injection pump whose main characteristics are a mass flow rate of 9.66 kg/s working at a pressure of up to 94 meters water column.

- A 10 m³ borated water storage tank, maintained at a temperature between 30 and 80°C at atmospheric pressure.

- Eight control valves (DN 25).

- Several orifices for characteristic flow measurements, lift check valves and piping.

As in the high-pressure system, each train of the LPSI system is connected on one side to the hot leg injection line or suction line and on the other to the cold leg injection nozzle of the waste heat removal system. Injection into the hot and/or cold legs is produced by 1, 2, 3 or 4 pumps according to the reduced flow rates simulated in the PKL Facility.

2.1.1.5 Pressure Relief and Safety Valves (PRZ-SV and PRZ-RV)

In order to keep the pressure of the primary circuit within an acceptable range, the pressurizer has different means to reduce or increase the pressure. The operation of the relief systems has particular importance for this study. They have a noticeable impact in cases where an overpressure is created in the installation.

The sprinkler system is intended to reduce the pressure by condensing the vapor cushion at the top. By converting this vapor into liquid water, the pressure inside this component is lowered. There is a distinction between operational spraying, from a signal from the volume control system, and auxiliary spraying, from a signal from the boron injection system. Operational spraying can only be performed when the pumps are running, while the two auxiliary functions can be performed even with the reactor coolant pumps stopped.

However, in high pressure situations or when the pressurizer is filled with liquid coolant, a relief system is needed to eliminate overpressures in more extreme cases than the spray system can control. The entire pressurizer installation comprises a separator tank to which part of the inventory is sent in situations where the safety valves (SV) and relief valves (RV) are used. Figure 2-2 shows the distribution of the different elements that make up the relief system.

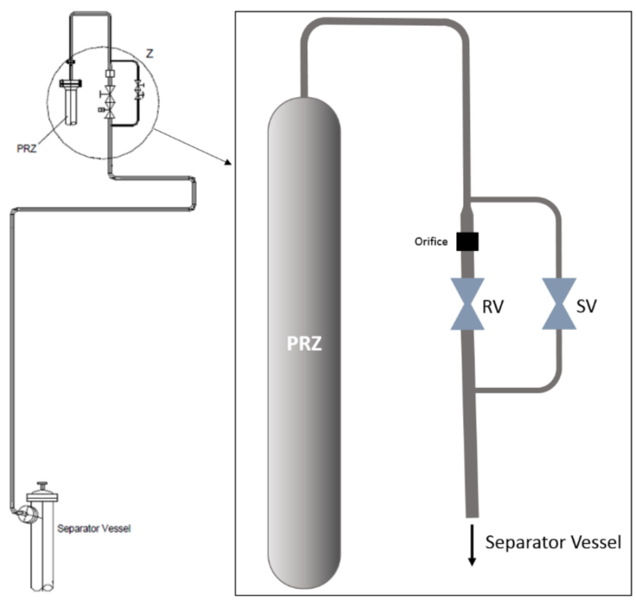

Figure 2-2 Pressurizer Relief Line

The use of the safety valve depends on the pressure found in the primary circuit; when exceeding 47.3 bar the SV opens and, after this action, it closes when returning below 46.6 bar. However, the relief valve acts when the temperature at the core outlet exceeds 300Cº. Closing occurs when the pressure drops below 12 bar.

The discharge rate after the RV signal is limited to a large extent by the bushing above the valve (illustrated in black in Figure 2-2), as it has a smaller cross-section than the rest of the line, including the parallel line where the safety valve is located.

This system plays a fundamental role during the station blackout of the test of this study.

2.2 Type of Transient. Description and Characteristics of the Test

2.2.1 General Characteristics of a Station Blackout (SBO)

The accident known as Station Blackout (SBO) consists of the loss of power supply to the plant, leading to a severe core cooling problem [5]. Many power plant components require electricity supply for their correct activity, such as pumps or pressurizer heaters. In situations of loss of power supply, an imbalance is created between heat production in the reactor core and its evacuation.

It is not a transient that is expected to occur. Therefore, the power lines are specially protected at the production and transport levels. However, it is studied because this situation endangers the reactor, and the effects of this incident could be severe since a possible consequence would be the release of radioactive material from the vessel.

Several degrees of loss of power supply can be studied, from the partial loss of power supply, temporarily in a part of the plant to the complete shutdown of all electrical equipment without recovery. For these situations, there are autonomous auxiliary emergency systems, such as diesel generators, which ensure the minimum power supply and preserve the plant's integrity. They allow the extraction of the residual heat providing energy to the most critical components of the system, thus minimizing the impact of the SBO until a regular electrical operation can be recovered.

This type of transient study has increased significantly since the Fukushima incident. On March 11, 2011, an earthquake of magnitude 9.0 on the Richter magnitude scale occurred off the northwest coast of Japan. This event resulted in the automatic shutdown of reactors 1, 2 and 3 of the plant, stopping electrical production. The external power lines were also damaged due to the earthquake, so diesel generators were started up to supply the plant's critical elements. A tsunami stopped the diesel engines from operating after the earthquake and prevented the emergency cooling systems from functioning. All three reactors suffered core meltdowns.

2.2.2 Specifications of the Transient Under Study

The transient to be analyzed, reproduced in the PKL III installation (described in the following chapter), is test H2.2 Run 2. It consists of studying a Station Blackout initiated by the supply water cut-off in the secondary circuit. It consists of three distinct phases, plus a previous conditioning phase that is carried out to adjust the plant conditions to those at the start of the test.

In the conditioning phase, the fundamental events are the shutdown of the primary pumps and the lowering of the secondary water level of the steam generators. Once the initial conditions are reached, the test is divided into three stages:

- Phase A: There is depressurization of the secondary and a period of pressure control in the primary by multiple discharges by the safety valves.
- Phase B: There is a depressurization of the primary circuit that activates the injection of the accumulators.
- Phase C: Two steam generators are refilled by employing the auxiliary pumping system.

The evolution of the plant parameters in the course of the transient is developed in more detail below, with graphical contributions.

2.2.2.1 Conditioning Phase

A process previous to the test itself, it consists of adjusting certain conditions and maintaining the plant in stable operation for sufficient time to ensure the correct stationary activity of the installation. This lapse of time is a phase called conditioning since it is outside the course of the SBO to be studied.

The experimental measurement begins to be performed with specific parameters outside the values needed at the beginning of the test. Therefore, some procedures are performed during the conditioning phase to achieve the necessary adjustments. The parameters to be modified are the temperature at the reactor core outlet (CET), the pressure in the secondary circuit, and the steam generators' level.

First, the secondary circuit supply is shut down to reduce the level of the steam generators. This level, which had a value of twelve meters, is reduced to four meters. When this point is reached, the secondary water circulation is reestablished. On the other hand, when the level of the steam generators reaches eight meters, a signal is produced that causes the pumps of the primary circuit to stop, which produces an increase in the core temperature. The CET is subcooled to about five Kelvin from the saturation temperature by adjusting the pressure in the secondary circuit, bringing the temperature up to about 255ºC. The secondary circuit pressure is set to a value of 24 bar at the start of the test.

The programmed evolution of the main parameters, including those highlighted above, in this conditioning phase, is shown in Figure 2-3.

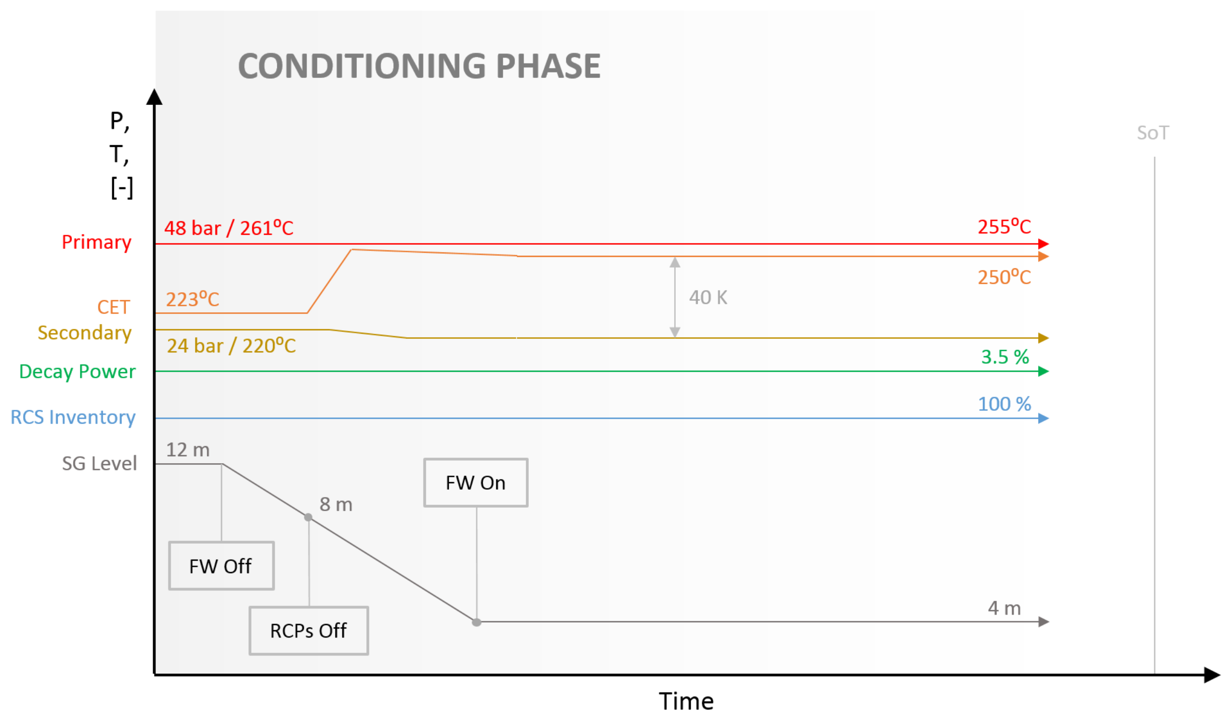

Figure 2-3 Expected Evolution of the PKL Installation During the Refurbishment Phase

After the conditioning phase has been completed and all adjustments have been made, the system reaches a steady state.The conditions of the installation at the beginning of the test (SOT) are presented in Table 2-1:

Table 2-1 Initial Conditions of Test H2.2 Run 2

Primary Side	
Reactor Pressure	46.6 bar
Core Exit Temperature (CET)	527.3 K
PRZ Level	7.7 m
Thermal Power	981 kW
Secondary Side	
Secondary Pressure	21.3 bar
SG Secondary Temperature	489 K
SG Level Water	4.18 m

2.2.2.2 H2.2 Run 2 Test

Once the conditioning phase has been completed and all parameters have been set, the test begins with the interruption of the water supply in the secondary circuit. Consequently, the level in the steam generators drops until the liquid is exhausted. In addition, associated with the start of the test is a power decay that causes a pressure and temperature drop in the primary circuit. The sequence of events that occurs is shown in Figure 2-4.

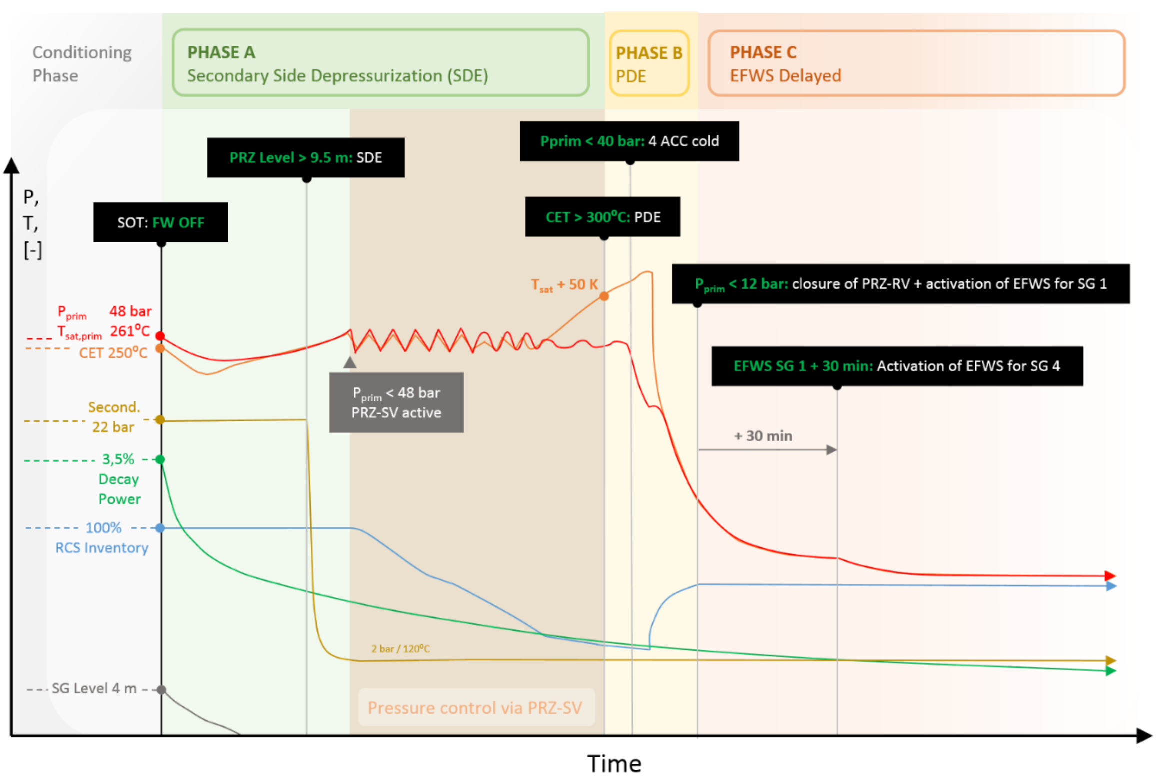

Figure 2-4 Predicted Evolution of the PKL System During the Test H2.2 Run 2

In support of Figure 2-4, the development of the test and the events that occurred from the beginning of the test are described below. Three different phases can be identified, already mentioned earlier: A, B, and C.

Phase A: Depressurization of the secondary circuit.

At the start of transient (SOT) several events take place, the power begins to drop following the decay curve, the feed water to the steam generators (SG) is cut off, and the pressurizer heaters' activity is interrupted. As a result, there is a slight drop in the core outlet temperature and pressure of the primary circuit. In contrast, in the secondary side, the SG's are completely drained (having a level of 4 m at the beginning of the test). This leads to the loss of the main heat sink and, therefore, the primary circuit coolant temperature starts to rise due to the degradation of the heat transfer between the primary and secondary sides of the SGs. This event in the secondary circuit causes the saturation temperature to be reached at the core outlet, forming a vapor head at the top of the vessel. The coolant inventory then moves from the reactor vessel to the pressurizer, reaching 9.5 m above the latter's level, which causes a control signal to depressurize the secondary.

The SG secondary side is empty at this instant, so extracting the decay power is impossible. Consequently, the average temperature of the coolant and the average pressure of the primary continue to rise, resulting in the formation of steam which causes the pressurizer to overpressure, forcing the safety valve (SV) to open when the pressure reaches approximately 48 bar. This opening is performed intermittently (valve cycling), thus stabilizing the temperature at the core outlet for a certain time. However, there comes a time when this measure is not enough and the CET rises over 300ºC, initiating the depressurization of the primary circuit, which marks the beginning of phase B of this test.

Phase B: Depressurization of the primary circuit.

With the completion of the cycling of the safety valve (SV) there is an increase in the temperature at the core outlet until it reaches 300 C, a value associated with the relief valve control (RV), which causes the depressurization of the primary circuit (PDE). When the pressure drops to 40 bar, the emergency coolant injection is activated by emptying the accumulators (ACC).

The pressure continues to drop and at 12 bar, the last phase begins.

Phase C: Activation of the auxiliary pumping system.

When the pressure in the primary circuit finally drops below 12 bar in all four loops, the pressure relief valve (RV) closes, and the auxiliary water supply (EFWS) is activated for steam generator 1.

After reactivation of the feed water in the secondary of the SGs, the test's main events are concluded.

As a summary of all the events observed during the transient, Table 2-2 includes the actual time by which it occurs and its startup criterion. Each event triggers an action to preserve or restore the normal operation of the power plant. It also contains the information corresponding to the conditioning phase.

Table 2-2 Sequence of Relevant Events Throughout H2.2 Run 2 Test

Test Phase	Time after SOT	Criteria	Action
Conditioning phase	-10800		Shutdown of Feedwater System (FWS) for all SGs
	-6115	Level less than 8 m in all SGs	Shutdown of all RCPs
	-6020	Level in SGs 2 and 3 of 4 m	Reactivation of the FWS for steam generators 2 and 3
	-4875	Level of 4 m in SG1	Reactivation of the FWS for steam generator 1
	-4010	Level of 4 m in SG4	Reactivation of FWS for steam generator 4
Phase A	0	Start of Test (SOT)	Shutdown of FWS, pressurizer heaters, CVCS, steam pressure control at 22 bar
	4550	Pressurizer level at 9.5 m	Secondary circuit depressurization by means of steam relief train
	5700	Primary pressure above 47 bar	Pressure control of the primary circuit via the pressurizer safety valve (PRZ-SV)
Phase B	9525	Reactor Core Exit Temperature (CET) higher than 300 C	Depressurization of the primary circuit by means of the pressurizer relief valve (PRZ-RV)
	9630	Primary pressure lower than 40 bar	Start of injection by the accumulators in all loops in the cold legs
Phase C	14230	Primary pressure lower than 12 bar	Start of emergency power supply (EFWS) on SG1. Closing of PRZ-RV
	14660		ACC of loop 4 empty. Start of N_2 discharge at RCS
	14810		Loop 1 and 3 ACC empty. Start of N_2 discharge in the RCS
	14840		ACC of loop 2 empty. Start of N_2 discharge in the RCS
	16035	After 30 minutes from the start of EFWS for SG1	Start of EFWS for SG4
	18150		End of test

3 TRACE MODEL OF PKL

The PKL model has been modeled in TRACE with 181 hydraulic components, including the vessel and the pressurizer, designed with pipe elements. The SNAP (Symbolic Nuclear Analysis Package) interface was used to develop this model.

The primary side consists of the vessel and the four loops of the facility with all the elements that integrate it, the pressurizer is included in loop 2. Furthermore, the secondary side includes the steam generators and the necessary elements in a more simplified way, so that the appropriate actions are produced to carry out the simulations correctly.

3.1 <u>Nodalization of Main Components</u>

The pressure vessel has been modeled using pipe elements. Each pipe serves the function of the different vessel parts with a configuration that favors the circulation between the components. For each axial level, the volume fractions and effective flow area have been established according to the technical specifications provided by the organization [6].

The two downcomers and the two bypasses of the upper head of the facility have been modeled. All the pressure vessel components are shown in Figure 3-1. The lower plenum or upper plenum are also defined, using pipe elements that allow specifying the geometry of each component. The height and volume of the vessel have the values determined by the facility's specifications [7].

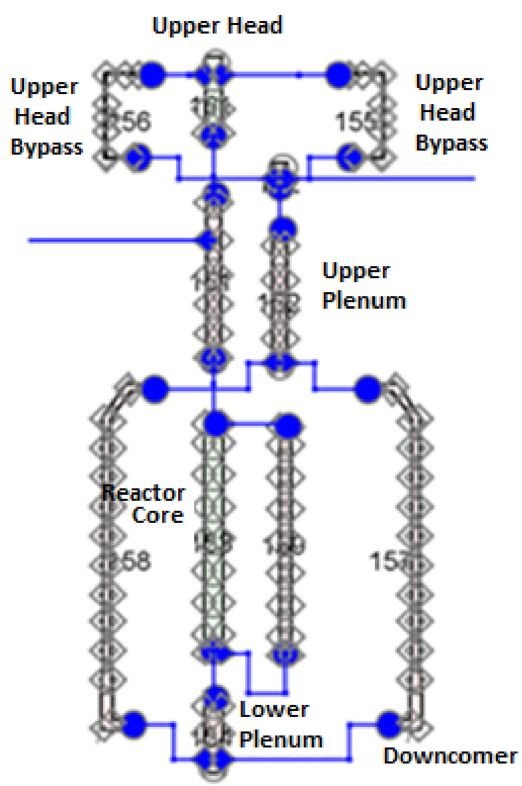

Figure 3-1 Nodalization of the PKL Model Vessel

The set of elements of the pressurizer is as shown in Figure 3-2. The discharge line starts from the pressurizer, where the safety and relief valves (SV and RV) are located. This discharge line consists of several pipes. There is a bifurcation; SV on the one hand and RV on the other hand. After the valves, another pipe leads to the separator vessel. This component finally discharges to the outside (simulated with a Break).

The SV and RV are used for:

- The safety valve (PRZ-SV) controls the primary pressure limit. It opens when the pressure exceeds the maximum value specified in the PKL facility. If the pressure drops enough, the valve closes again to continue the plant's regular operation.

- The relief valve (PRZ-RV) depressurizes the primary circuit after opening in cases of pressure peaks higher than the SV can control.

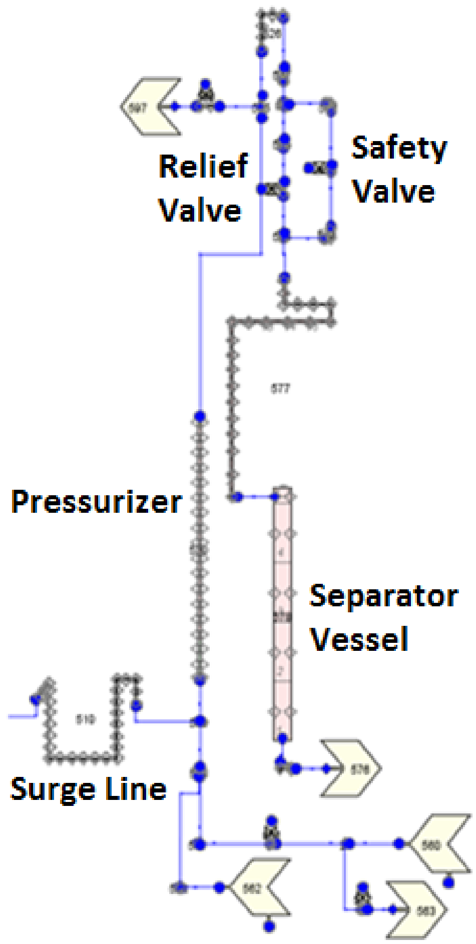

Figure 3-2 Nodalization of the PKL Model Pressurizer

The primary side of the installation is distributed into four loops. All loops are equal except for loop 2, which is connected to the pressurizer. Each of them represents the following course, they start at the outlet of the vessel, then, the hot leg leads the coolant to the inlet water box of the U-tubes of the steam generator.

After passing through the steam generators, the heat is exchanged with the secondary side. Next, in the outlet water box, the U-tubes join, leading the coolant to the pump seal. The main pump drives the water to the cold leg, in this section, the accumulator is also located in this section. These elements have been modeled with pipe elements too. The actual geometry and location with the corresponding water level of the installation elements have been adopted. The free space of all accumulators is occupied by pressurized nitrogen. The discharge is carried out by opening a valve located at their bottom, it is controlled by the pressure signal of the accumulator injection system activation. In the case of the H2.2 Run 2 test, this set point is at 40 bars.

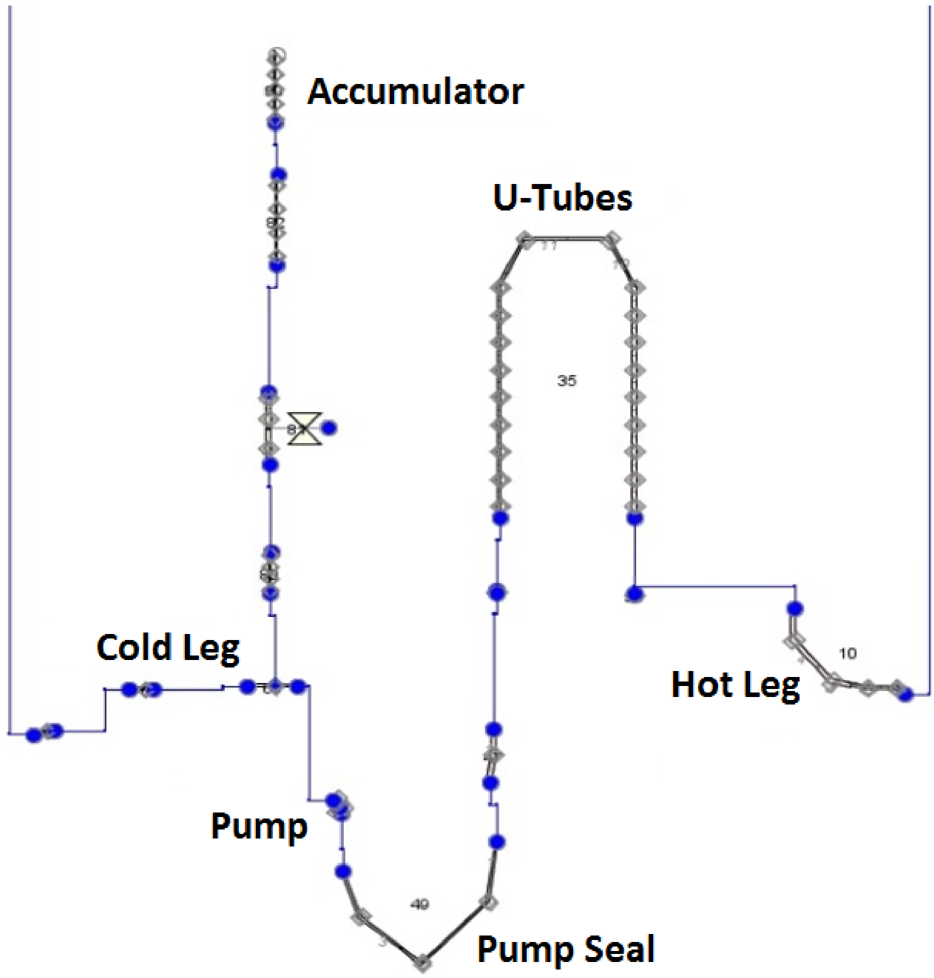

Figure 3-3 Nodalization of the PKL Model Loops

The secondary side is modeled assuming simplifications, since the study is usually focused on the behavior of the primary circuit. The rest of the secondary circuit is modeled through the use of Fill and Break components to introduce and extract the fluid respectively. The main elements of the steam generators are modeled using pipes, being these elements mainly the riser, the downcomer and the dryers and different connections.

With these elements, the conditions on the secondary side of the steam generators can be correctly reproduced. A heat structure simulates the heat transfer between the cells of the U-tubes to the corresponding cells of the riser [8]. The appearance of the steam generators at the SNAP interface is shown in Figure 3-4.

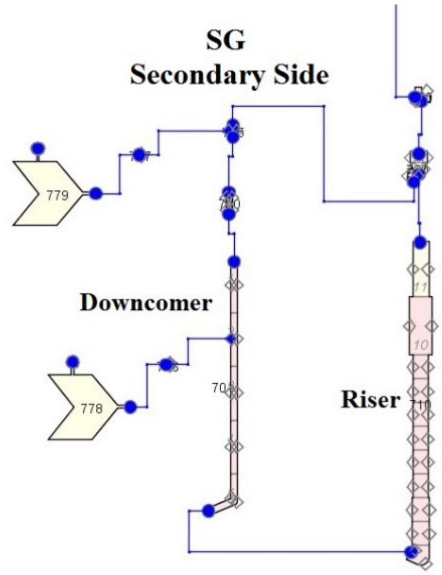

Figure 3-4 Nodalization of the PKL Model SGs

3.2 Global Characteristics of the Model

The main characteristics of the PKL model for the TRACE code (Figure 3-5) are as follows:

- Vessel composed of pipes. Two downcomer components and two upper head bypasses are included with the same volume as the original elements of the installation. In addition, the upper plenum is doubled to promote circulation in the core and separate the inlet and outlet to the loops.

- Four loops in which different parts can be differentiated, such as; the hot leg, U-tubes (where a single tube is modeled respecting the total volume), the pump seal, the pump and the cold leg. The emergency injection systems are also linked in their respective positions with an updated configuration of the accumulators.

- Pressurizer formed by a pipe. Safety and relief valves are included with a vertical arrangement, a forced pressure and level stabilization system, special care has been taken with the modelization of the surge line, which is connected with loop 2.

- Simplified secondary circuit, with four loops consisting of the steam generator and the steam lines. Volume distribution in the risers of the generators with a slight enlargement in the lower zone. Several fills, breaks, and valves help simulate this circuit's behavior for the required conditions.

- Control system, which governs the logic and impositions of the H2.2 Run 2 test, from which the input and output signals for the operation of all active components are obtained.

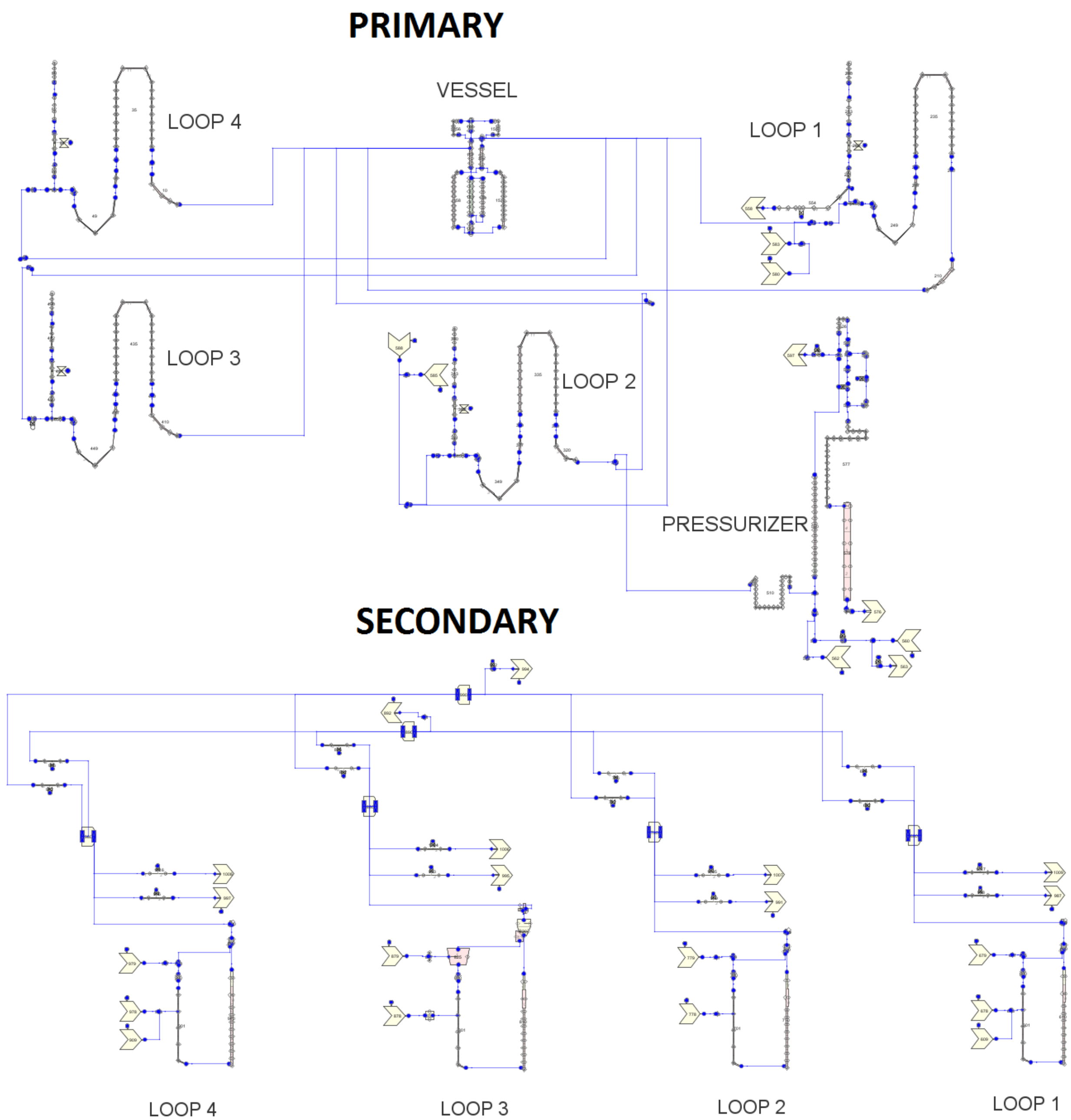

Figure 3-5 Complete Hydraulic Model of the PKL System at TRACE

4 GUIDELINES FOR MODELING AND UNCERTAINTY ANALYSIS

In the simulation of a Station Blackout scenario such as the H2.2 Run 2 test, where the impulse sources are very small, temperature and pressure losses are of great importance. For the execution of this test, it has been necessary to modify the nodalization of some important elements such as the vessel. Additionally, the performance of sensitivity studies focused on particular events such as the mass discharged in the pressurizer during the depressurization of the primary circuit has also been needed. The most significant sources of uncertainty for this scenario are the following:

- Heat losses. In plants built to scale, heat losses have a greater impact [9]. The process of scaling in volume and power to 1:145 means that heat losses are 12 times more influential in the scaled facility than in the reference plant [10]. The evaluation of this parameter has a remarkable impact.

- Pressure losses: It is convenient to carry out sensitivity studies in different stationary conditions to verify the correct behavior of the model. An adequate adjustment of the pressure losses by sectioning the primary circuit by sections provides the correct geometrical dimensioning and the values assigned to the friction coefficients of the components that make up the model.

- Modeling of tanks and enclosures: Components such as the vessel, pressurizer, accumulators or vapor generators, where liquid can accumulate and stagnant conditions can occur, it is necessary to ensure that the balances between cells are not impaired under transient conditions. The elements must allow circulation between cells and mixing to reach logical temperature and pressure conditions. When using 1D elements to model these types of components, stratification and distorted pressure and temperature values can occur. Using 3D elements eliminates problems of this type, but significantly increases the computational cost for a long duration test such as an SBO. In this model, two of the components mentioned above, the pressurizer and the vessel, had to be modified. The pressurizer was initially modeled by means of a pipe component, in which an inadmissible stratification appeared for the execution of the test. As there was no fluid inlet and outlet in the pressurizer, the fluid circulation was impaired and this element had to be replaced by a 3D vessel. The vessel itself, also made of 1D pipes, had to be restructured in order to facilitate the calculation of the balances between cells, the fluid circulation and the exchange between parallel channels to promote convective flows.

- Discharge through the PRZ safety and relief valves: The mass discharged during the activation of the SV and RV valves of the pressurizer is affected by the entrainment experienced in the connection of the vessel element to the pipe element that models the conduct where the valves are located. The mass discharged in the simulation is higher and several sensitivities have been performed in order to correct this discrepancy. The problem encountered is that in order to maintain the pressure in the primary circuit during phase A, in particular during the valve cycling, the mass discharge must be higher. This indicates that more liquid phase is being entrained being a parameter with a possible margin for improvement within the model.

- Boundary conditions: There are parameters that have shown an important impact on the SBO scenario studied. Boundary conditions that must be adjusted with special attention because they have a significant influence on the simulation of the transient.

The most important parameters are the pressure on the secondary side of the SGs, the ACC conditions (pressure, level and temperature), the initial water level of the PRZ and the discharge pressure of the safety system and the pressurizer relief.

5 TEST RESULTS AND COMPARISON WITH TRACE

After adapting the model and explaining all the plant characteristics and the transient, the results comparing the simulations run with TRACE patch 4 code model and the experimental measurements taken in the actual installation are presented below. Recall that the test requires a conditioning phase of the steady-state operation (some adjustments are made before the start of the transient as above-mentioned). The H2.2 Run 2 test starts when the supply of feedwater in the secondary circuit is interrupted.

5.1 Conditioning Phase

A conditioning or pre-test phase modifies specific plant parameters before the actual test. Once the conditions are stable and match the plant's measurements, three factors must be modified to establish the status in which the test begins to be reproduced: the level of the steam generators, the temperature at the core outlet and the secondary circuit pressure.

First, Figure 5-1 shows the level change of the steam generators. The manipulation of fill components that feed the steam generator forces its level to drop. As previously mentioned, the design of the secondary circuit does not present as much weight as the primary one in the PKL installation, since the studies focus on the behavior of the primary circuit. Consequently, the model is adjusted using these components to have the parameters as near to the experimental data as possible at the start of the test.

The temperature behavior at the core outlet, or CET, is included in Figure 5-2. This parameter is perfectly adjusted to the rise caused around 6000 seconds before the start of the test through the relevant adjustment in the steam generators exchange conditions, this adjustment has been carried out through the modification of the secondary circuit pressure, which is shown in Figure 5-3.

As a consequence of the previous adjustment, the pressure of the secondary circuit does not precisely match the experimental data, since its modification directly affects other factors, such as the temperature at the core outlet, variable which has been considered of major importance. Thus, the pressure in the secondary side at the beginning of the test is not accurately replicated, prioritizing the primary circuit's status. This discrepancy is due to the secondary circuit's simplicity and the continual adaption of all model components. There may be discrepancies between each model component's geometry and loss coefficients.

To summarize, it can be specified that the adaptations suffered in the conditioning phase led to start the simulation of the test with TRACE in practically the same conditions as the installation was at the time of taking the experimental measurements. The temperature at the core outlet (CET) and the pressure in the primary circuit are directly related, being able to modify the first of the variables named based on the other, so the pressure in the secondary has been raised slightly (2 bar) to adjust the value of CET.

Other important parameters, such as the pressure in the primary or the pressurizer level, have been matched quite accurately before the beginning of the transient, but in this phase, they do not undergo any modification, so no graphs showing their evolution are presented.

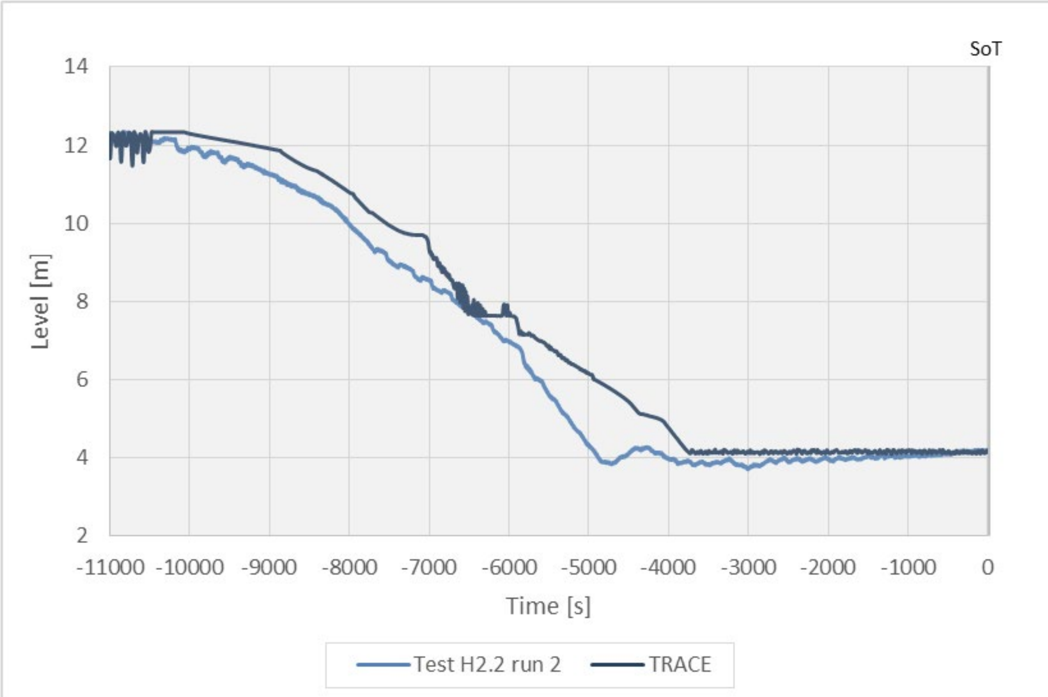

Figure 5-1 Evolution of the Level of the Steam Generators in the Conditioning Phase

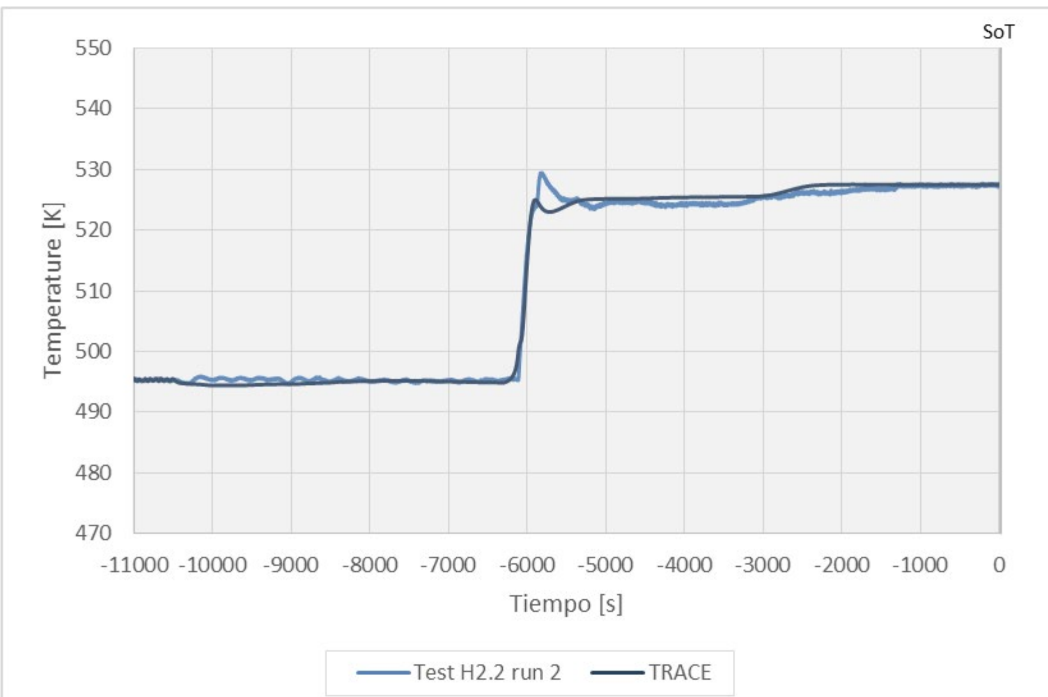

Figure 5-2 Evolution of Core Exit Temperature in the Conditioning Phase

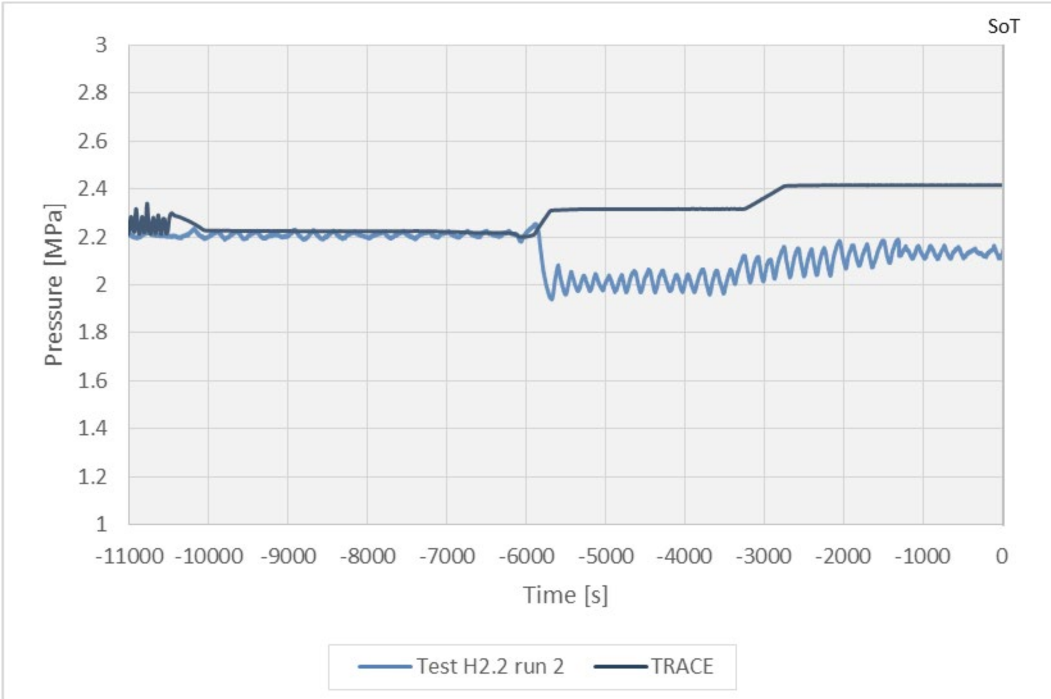

Figure 5-3 Pressure Evolution in the Secondary Circuit in the Conditioning Phase

5.1.1 Test H2.2 Run 2

The initial conditions of the H2.2 test are reached by modifying the status of some parameters. The simulation is carried out by replicating the sequence of events shown in Table 2-2.

A good reproduction of phase A is achieved. However, some parameters deviate from the correct values or are advanced in the simulation with regard to the experiment. For this reason, phases B and C show some discrepancies in core outlet temperature (CET) or pressurizer level. In this way, the test simulation provides good results and very similar evolutions during the transient duration.

First, one boundary condition parameter considered is the reactor power (Figure 5-4). This model has been introduced through a simplified time-dependent table. The power drops rapidly in the first moments of the transient, but this drop is a little more controlled and tends to stabilize after a while. In the last part of the transient, the power has a value of around 300 kW, continuing the decay trend. This value is approximately 30% of the power used by the installation in steady state.

Figure 5-5 shows the pressure in the upper plenum during the 18150 seconds of the H2.2 test. The events described in Table 2-2 (sequence of events) are easily distinguished in the graph drawn until the beginning of phase B, when the pressure drops after the cycling period (where the sawtooth pattern appears in the graph).

The evolution of the pressure in the simulation throughout the transient turns out to have a remarkably faithful behavior compared with the data measured in the plant. The pressure drop overlaps with the lines in the graph at about the 4000th second. The pressure recovery from this point onwards makes it possible to differentiate between the simulation and the actual data. However, the time evolution of the simulation pressure with respect to the experimental data is very similar until the arrival of the valves cycling. These cycles start to occur about 200 seconds earlier in the simulation, insignificant for such a prolonged duration as that of an SBO. These cycles are produced by the action of the safety valves of the pressurizer, set at 47 bar opening and 46.6 bar closing.

From this point on, phase B takes place. The completion of the cycling gives rise to the start of the depressurization of the primary circuit. As commented above, this event is about 200 seconds earlier in the TRACE simulation, but the evolution of this variable is very similar, especially from the 10000th second, where the pressure drop takes a similar slope as in the actual plant. Due to the pressure loss, the accumulators' injection process after the pressure drop to less than 40 bar is observed [11].

This discharge maintains the approximate time difference of 200 seconds between the simulation and the plant data. As the depressurization continues, the end of this phase is marked by the closure of the pressurizer relief valve (RV) and the activation of the EFWS injection (for the loop 1 steam generator) after dropping below 12 bar.

The beginning of the last defined period, phase C, starts with about 600 seconds delay in the simulation concerning the in-plant evolution. The loop 4 steam generator starts to be filled by injection thirty minutes after the start of the phase, both for the simulation and the actual PKL plant.

Once described, the transient evolution focused on the pressure variation and exposing the general events and their performance, a particular focus on the behavior of the core exit temperature (CET) and the peak cladding temperature (PCT) and the discrepancies obtained in the reproduction of the test with the TRACE code is made. The CET and PTC are represented in Figure 5-6 and Figure 5-7 respectively.

For these parameters, it can be seen that more significant differences appear in the times at which the different events occur. In principle, it is apparent that outstanding results have been achieved for phase A, reproducing the plant's behavior in the simulation very reliably. The end of this phase is brought forward, as explained above. Due to the mentioned parameter, the beginning of phase B is marked by the temperature represented in Figure 5-6, which, on reaching 300 ºC, gives rise to the start of the depressurization of the primary circuit with the opening of the pressurizer circuit relief valve.

After the start of the primary circuit depressurization, the CET and PCT evolve in a somewhat erratic manner over time since the same evolution is observed, with the same temperature peaks appearing as those measured but advanced. The maximum temperature peak at the core exit has similar values, with the highest value having an error of only about 5%, but the phenomenon is too early. The discrepancy appears as a consequence of the liquid level in the core. It can be seen in Figure 5-8 that the temperature peaks appear at times when the core level drops below approximately 5 m. The first peak is damped due to the start of the accumulator discharge, however, the simulation experiences a more progressive accumulator discharge (Figure 5-9) due to the primary pressure dropping slightly softer. This results in the core being uncovered at the top because less water is injected than in the experiment and, therefore, a premature temperature spike appears in the simulation.

The behavior of the accumulator discharge and the pressure in the primary side also explains why the second peak of the experiment does not occur in the simulation. At the time when this second peak occurs, the experiment is in a fairly stable pressure phase which causes the accumulator discharge to stagnate. After a period of time without a significant water supply from the accumulator, the level in the core drops and, as a consequence, a temperature peak appears. In the simulation performed with the TRACE model, there is no stagnation in the discharge of the storage tanks. The water supply has a constancy that does not allow the level in the core to drop, so CET and PCT do not spike.

The slightly smoother pressure drop in the simulation may be due to the fact that the model has a vessel designed with 1D elements. The behavior is good but the direction of motion may be limited by the orientation of the pipes and the exchange cells. A 3D vessel would favor fluid mixing and remove motion limitations, but the computational cost would increase significantly in a transient that has a long duration.

The TRACE reproduction of the loop 1 accumulator level, shown in Figure 5-9 displays a level pattern comparable.to that detected in the PKL facility. The accumulators are kept full until the pressure in the primary circuit drops below 40 bar. This fact occurs with a small advance in the simulation. This lead-in can be seen in Figure 5-5, when the end of the valve cycling events that define the final of phase A occurs. This advance does not have too much importance for the achievement of the events of the transient. This more progressive behavior does not allow to reproduce the step that appears in the last 3000 seconds of discharge. The above reasoning on the evolution of the primary side pressure based on vessel 1D explains the discrepancies in the discharge of the accumulators.

Once the accumulators have started to discharge, the rest of the level of this element in the TRACE simulation has a duration that is fairly comparable to that of the plant. At around 15000 seconds of transient duration, the component is completely drained. The simulation performed with the TRACE model represents a more progressive emptying than that of the plant data, but it can be observed that the behavior is correctly reproduced.

Due to the depressurization circuit's tremendous influence on the simulation, the pressurizer's level is an interesting parameter. It can indicate the action of the safety and relief valves and the state of the working fluid in the primary circuit. The evolution of the pressurizer level is shown below in Figure 5-10.

As has been the case with the above variables, phase A is reproduced with significant reliability. The level begins the transient falling slightly and progressively rises until it reaches the top value briefly before the safety valves (SV) act. The cycling period produces a varying level drop until relative stability is achieved at about nine meters, reproduced about one meter below.

With the onset of phase B, slightly earlier in the simulation, the similarity of the level evolution is broken. It has already been noted that the behavior may be different due to possible mismatches of loss coefficients in the vessel. The level drops exponentially until after 13000 seconds in the simulation, while the experimental measurements show continuous fluctuations, but despite this, it remains at a value close to nine meters. This is due to the discharge of the SV. As can be seen in Figure 5-11, during the cycling controlled by this valve, the mass discharge is higher in the simulation with TRACE than in the experiment performed in the PKL facility.

A quick emptying happens in phase C, which was detected in the simulation with an advance of around 1500 seconds and from a significantly lower level (3.46 m compared to the actual 8.66 m). This event occurs at approximately 13400 seconds when it should be shortly before 15000, resulting in the complete discharge of the component, which maintains level values corresponding to the height at which the surge line is placed inside the pressurizer tank.

Another of the main variables, shown in Figure 5-12, is the pressure in the secondary circuit: As has been the case so far with the above variables, phase A is reproduced with significant reliability. The level at the start of the transient starts falling slightly and progressively rises until it reaches the top value briefly before the safety valves (SV) act. The cycling period produces a varying level drop until relative stability is achieved at about nine meters, reproduced about one meter below.

Before this sharp pressure drop, the parameter's value is about 24 bar in the simulation, but experimental measurements indicate that it should be somewhat lower (21 bar).The point at which this pressure drop happens is replicated reasonably accurately in time, around 4600 seconds after the transient begins. After this, the simulation is stabilized at 2 bar and the installation at 1 bar until the end of the test. Some unmentioned fluctuations appear after 14000 seconds, which coincide with the points where the pressurizer is discharged, or when the maximum PCT occurs. This alteration is not reflected in TRACE case, being already at the end of the test where the simulation loses accuracy.

On the other hand, other variables that can illustrate the operation of the SGs are shown in Figure 5-13, Figure 5-14 and Figure 5-15. They correspond to the level and temperatures in the high and low zone of the steam generators. The data represented in these figures correspond to the values in steam generator 1.

Figure 5-13 shows how the level of the steam generators drops until this element is empty after about 4000 seconds. This effect is perfectly reproduced in the simulation with TRACE. The level remains at zero until refilling occurs again in the last phase of the transient. On the other hand, the temperature in the high and low zones (shown in Figure 5-14 and Figure 5-15) has some discrepancies. After emptying the accumulators, the temperature of around 530 K is reached. In the simulation with TRACE, a lower level is calculated. This discrepancy arises from the component's empty state and the code's difficulties in accurately reproducing the experimental data. After 9000 seconds of transient, the temperature, which had reached relative stability, decreased its value with erratic behavior. Although the simulation has discrepancies with the plant measurements, the temperature drop occurs with similar behavior.

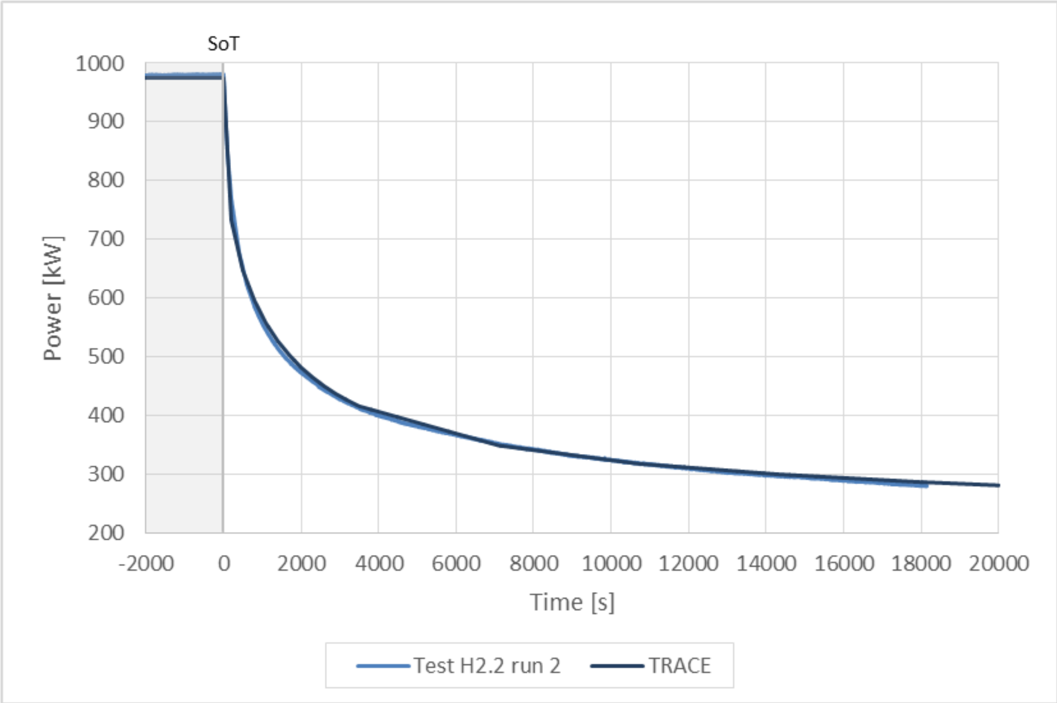

Figure 5-4 Core Power Evolution

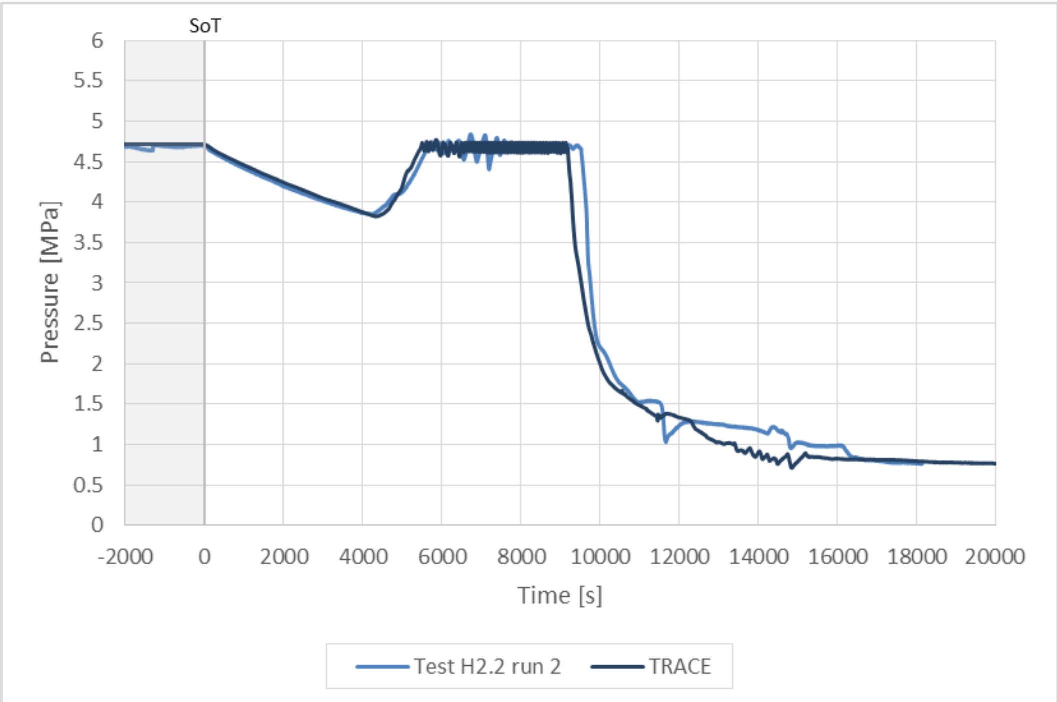

Figure 5-5 Pressure Evolution in the Upper Plenum

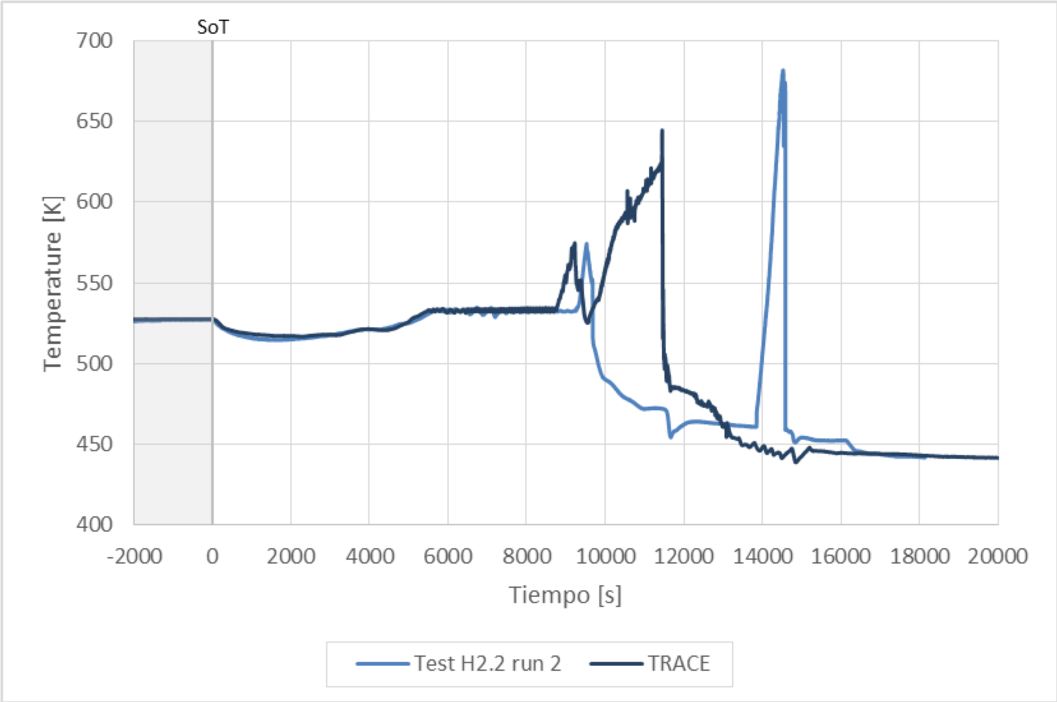

Figure 5-6 Core Exit Temperature Evolution

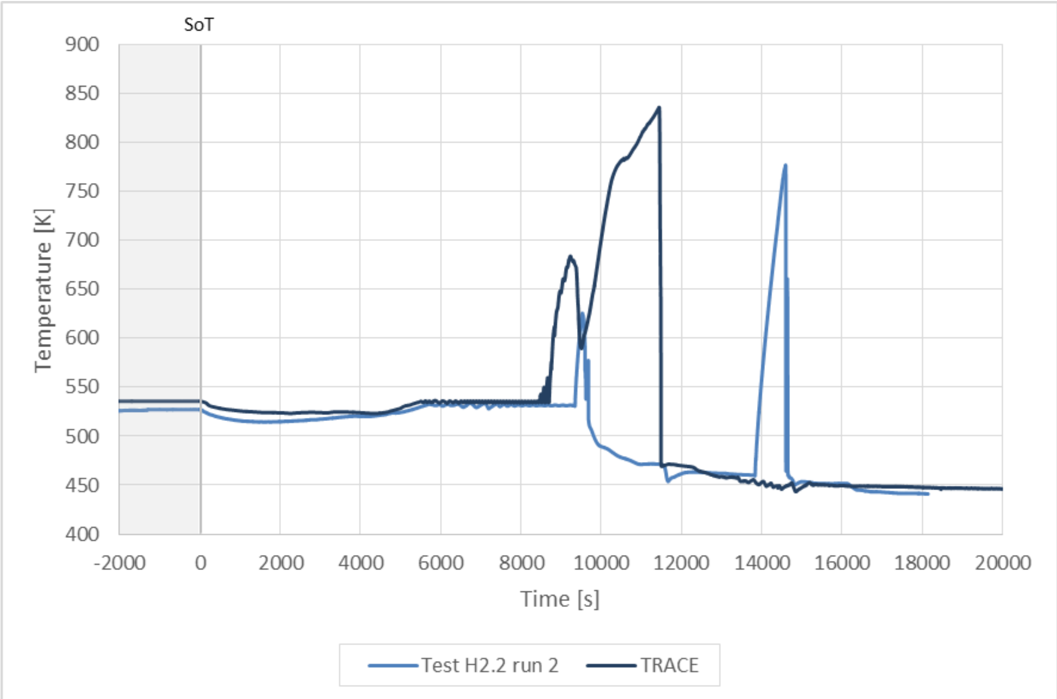

Figure 5-7 Peak Cladding Temperature

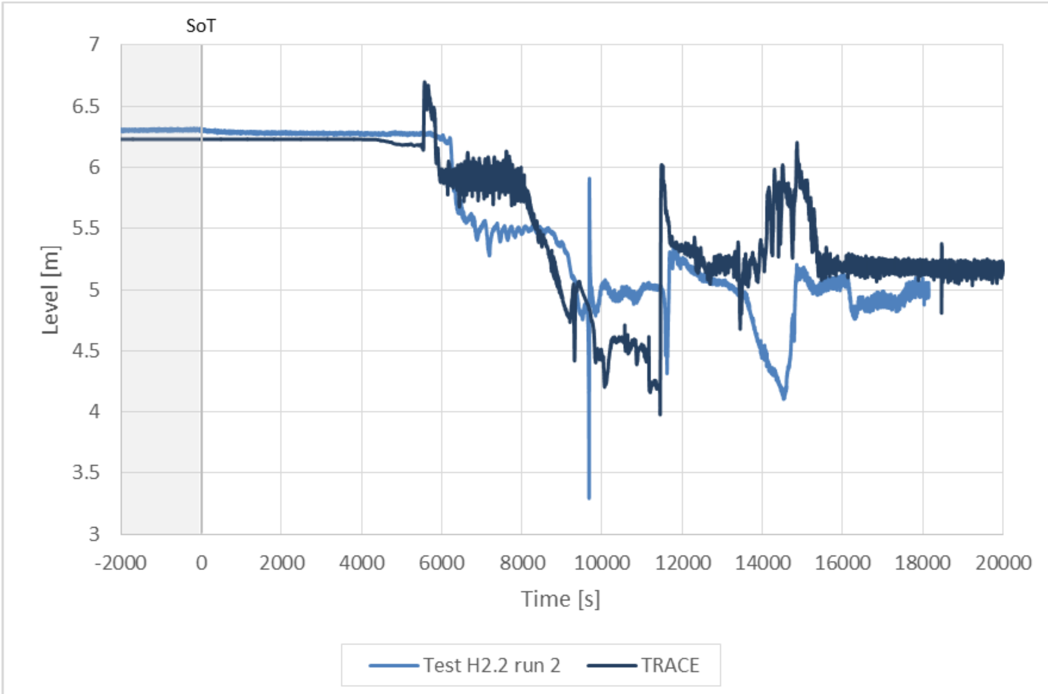

Figure 5-8 Evolution of the Core Level

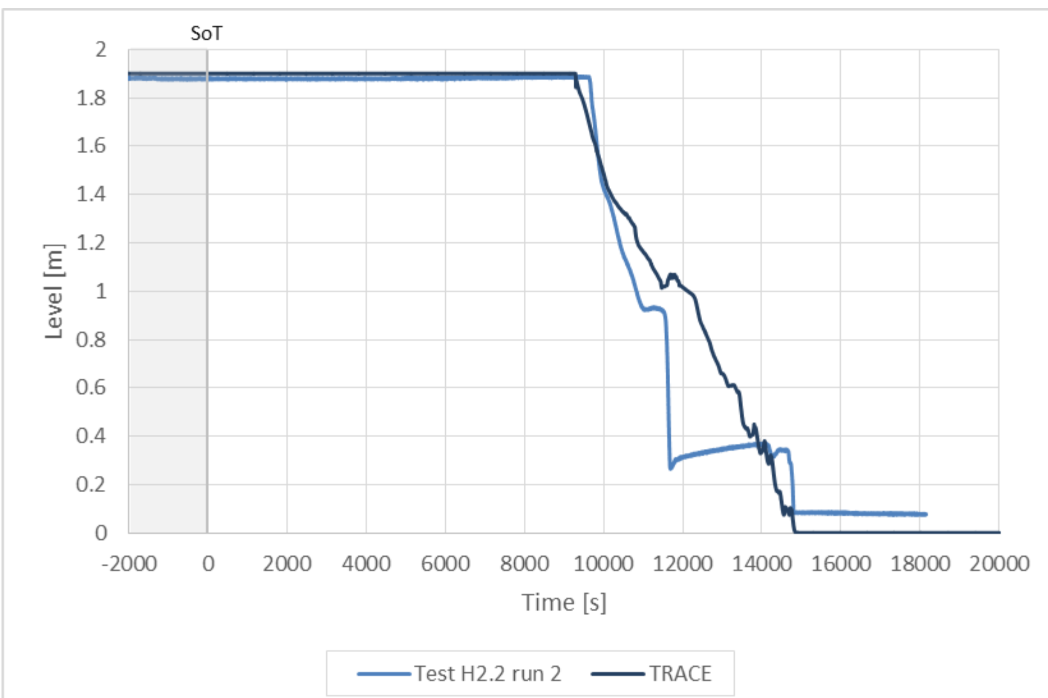

Figure 5-9 Evolution of Loop 1 Accumulator Level

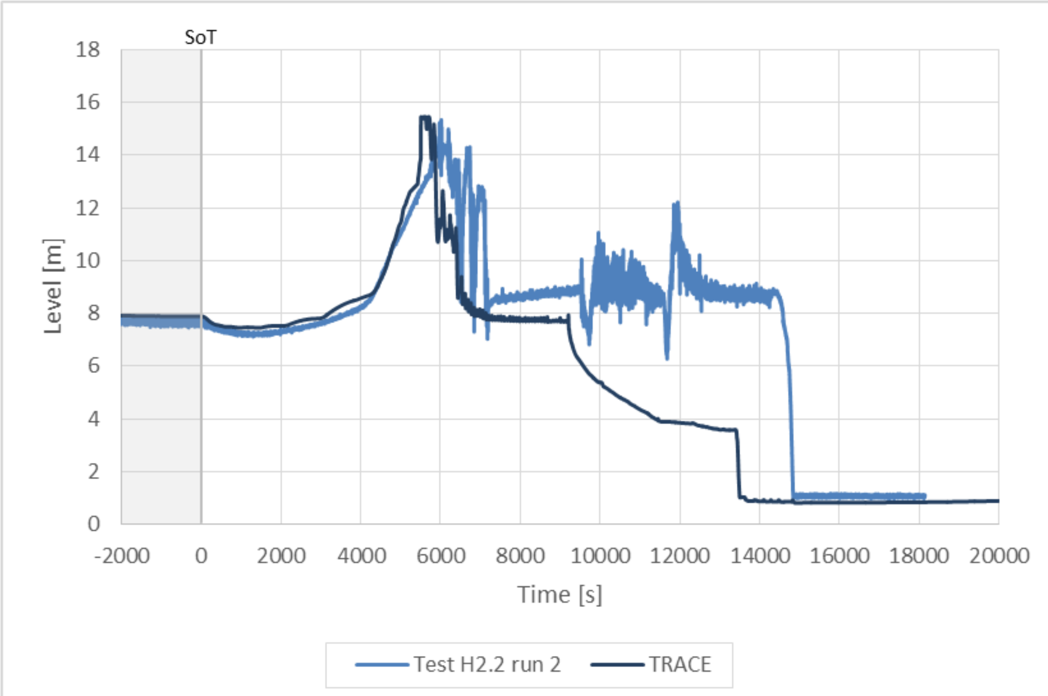

Figure 5-10 Evolution of the Pressurizer Level

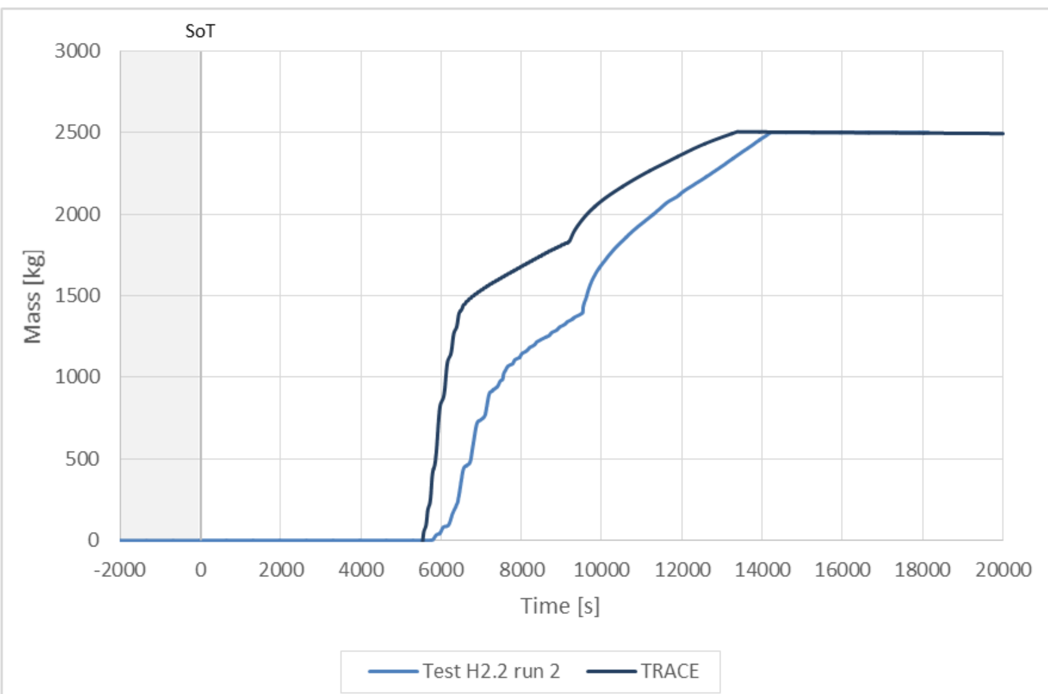

Figure 5-11 Pressurizer Discharge Mass

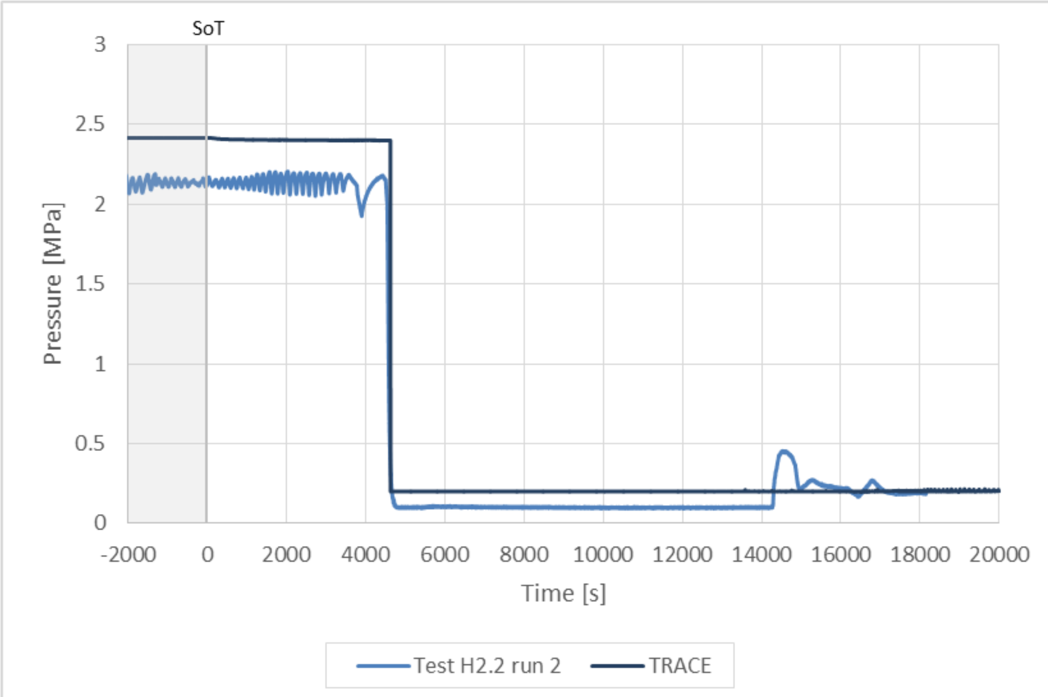

Figure 5-12 Pressure Evolution in the Secondary Circuit

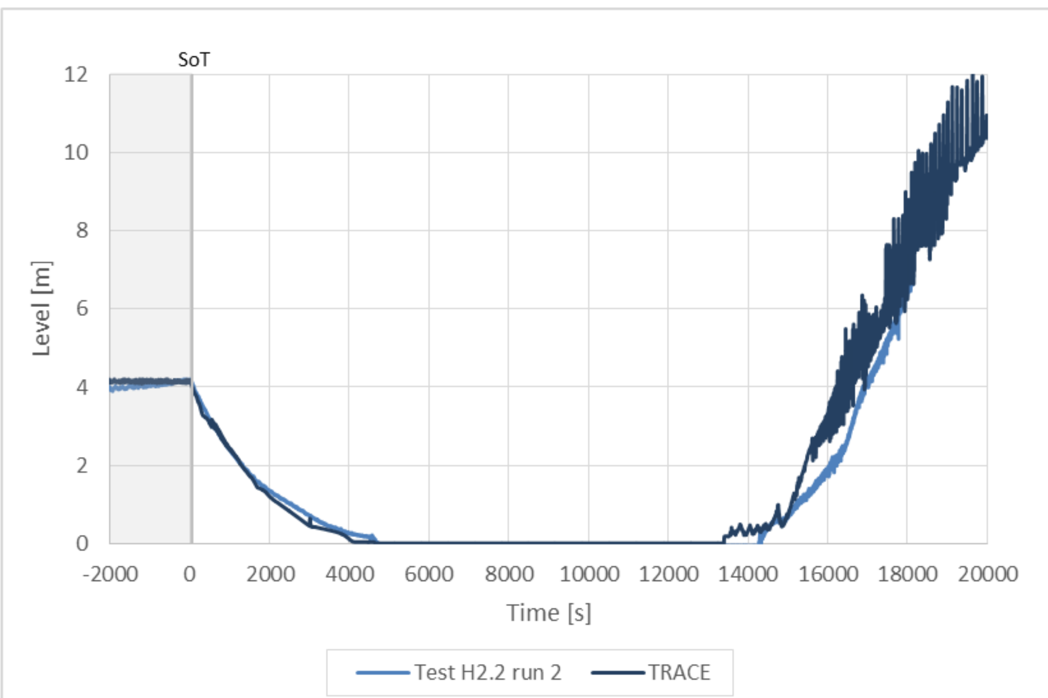

Figure 5-13 Level Evolution in Steam Generator 1

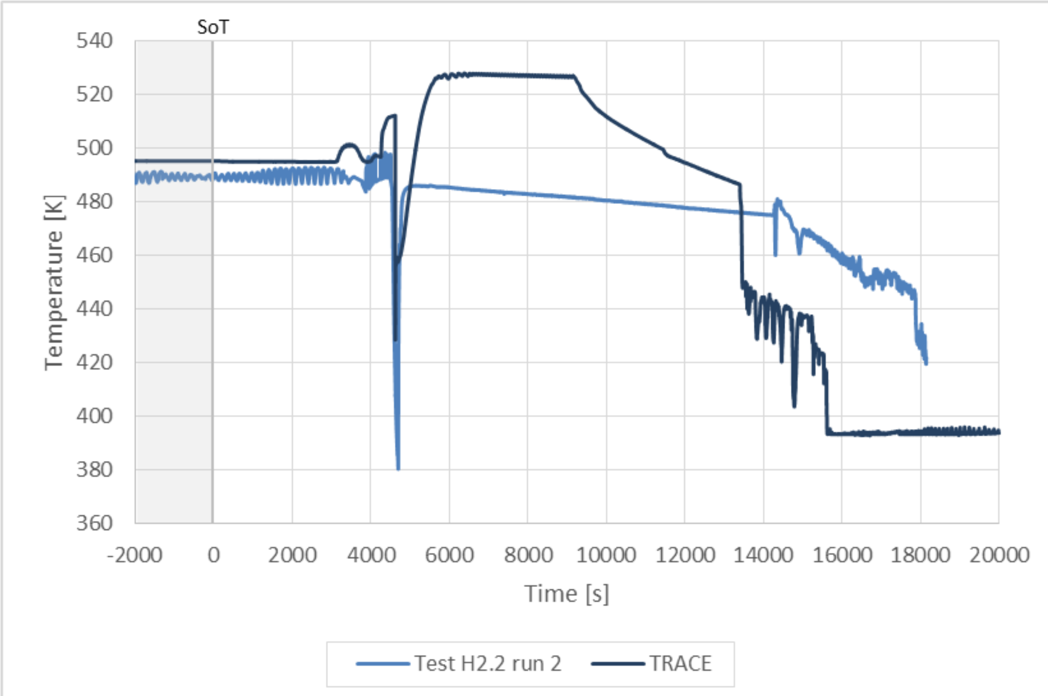

Figure 5-14 Temperature Evolution in the Upper Zone of Steam Generator 1

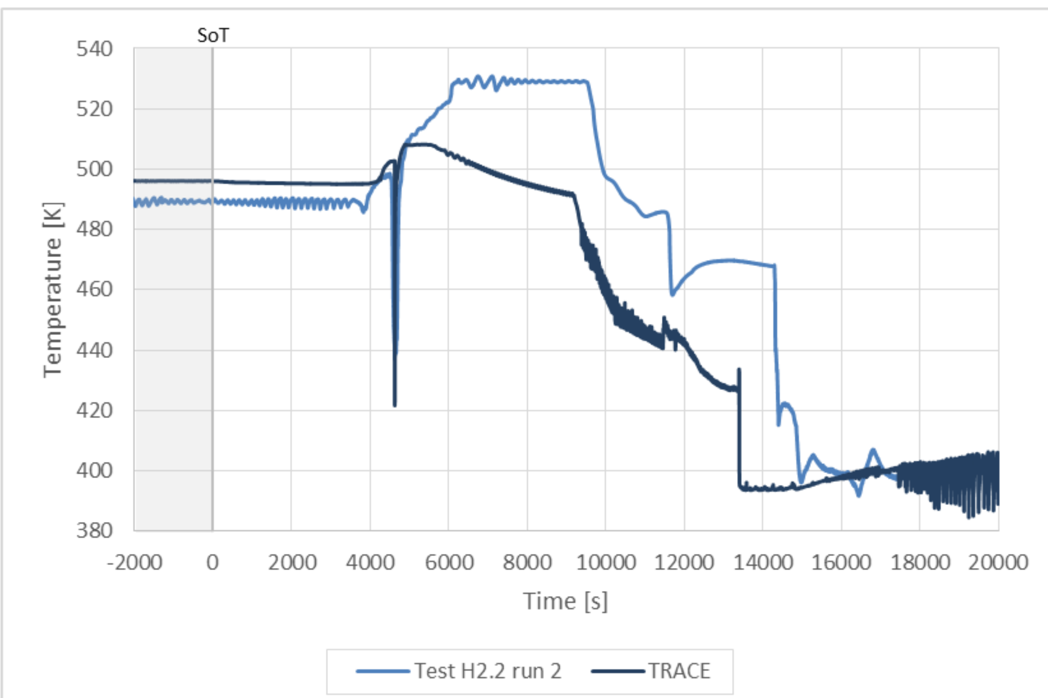

Figure 5-15 Temperature Evolution in the Lower Zone of the Steam Generator 1

6 CONCLUSIONS

The results obtained with the SNAP v2.4.1 interface and the TRACE v5.0 p4 code have been presented. The objectives are oriented to investigate the reproducibility of a Station Blackout transient reproduced in the PKL III facility, which simulates a 1300 MWe PWR reactor, testing the capabilities of the TRACE code and the developed model.

The PKL III H2.2 experimental test run results can be considered satisfactory. The TRACE simulation has replicated a complete Station Blackout test. A reasonably sequence of events is obtained during the whole transient for the TRACE code simulation, with acceptable values for the different magnitudes, since all of them are close to the experimental measurements. Particularly, the evolution of phase A is very accurate, given that all the values of the TRACE variables are very close to the experimental ones.

Given the limitations of the 1D components of which the vessel is made, reliable performance is achieved. For its modeling, the inlet and outlet paths to each element have been taken into account to avoid conflicts in the flow directions. In this way, recirculation is favored in elements such as the core where conflicts could arise. A 3D vessel would favor fluid mixing and eliminate motion limitations, but the computational cost would increase significantly when simulating long-duration transients such as the SBO of test H2.2 Run 2. The differences that appear after the end of phase A are probably due to the slightly softer pressure drop on the primary side. The major discrepancies found in the test are the CET and PCT evolutions, which show a premature second temperature peak. These can be explained by the discharge behavior of the accumulators, which is directly related to the primary pressure. The coolant supply from the accumulators is much more progressive than in the experiment and fails to fill the core sufficiently after the cycling phase. Consequently, the second temperature peak appears in the CET and PCT. In addition, local effects such as load losses and heat transfer discrepancies between the experimental installation and the TRACE model must be taken into account.

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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

This document presents and explains the main results obtained in the simulations performed with the TRACE consolidated thermal-hydraulic code in relation to the PKL III H2.2 Run 2 test. The H2.2 Run 2 test deals with a Station Blackout (SBO) in which three main phases take place. A first one in which the secondary circuit is depressurized, a second one in which the primary side is depressurized and the third one in which the emergency feedwater system (EFWS) is activated. The main goal is to analyze the ability of the TRACE V5.0 p4 code to accurately simulate the events and phenomena encountered throughout the studied transient. The activation of the different systems and the values of the main parameters will be used to assess the capabilities of the PKL model developed in TRACE.

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