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International Agreement Report

Analysis with TRACE Code of PKL III Tests G1.2. Study on Heat Transfer Mechanisms in the SG in Presence of Nitrogen, Steam and Water as a Function of the Primary Coolant Inventory in Double Loop Operation

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ABSTRACT

The goal of this report is to explain the main results obtained in the simulations performed with the consolidated thermal-hydraulic code TRACE regarding tests PKL III G1.2. The G1 test series are composed of G1.1, G1.1a and G1.2 tests, all of them are focused on the occurrence of boron dilution processes following the loss of Residual Heat Removal System (RHRS) during ¾-loop operation (primary circuit still closed). Main objective was to provide a data basis for thermal-hydraulics codes for a better understanding of the heat transfer mechanisms in the Steam Generator in presence of Nitrogen, steam and water in the U-tubes and of the coolant transport phenomena observed inside the U-tubes. The differences among G1.1/G1.1a and G1.2 test series are the coolant drain-injections sequences and the loop configuration, 1 loop or 2 loops in operation respectively. The main goal of this report is to analyse the capacity of TRACE V5.0p2 code to precisely simulate thermal stratification and natural circulation of both single and two-phase fluxes inside the whole primary circuit, as well as accurately predicting boron concentration variations.

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:

- assessment of plant modifications (e.g., Technical Specifications, power uprates, etc.);
- analysis of actual transients, incidents and/or start-up tests;
- development and verification of Emergency Operating Procedures;
- providing some elements for the Probabilistic Safety Assessments (e.g., success criteria and available time for manual actions, and sequence delineation) and its applications within the risk informed regulation framework;
- training personnel (e.g., full scope and engineering simulators); and/or
- assessment of new designs.

For that reason, the history of the involvement in Thermalhydraulics of CSN, nuclear Spanish Industry as well as Spanish universities, is long. It dates back to mid 80's when the first serious talks about Spain participation in LOFT-OCDE and ICAP Programs took place. Since then, CSN has paved a long way through several periods of CAMP programs, promoting coordinated joint efforts with Spanish organizations within different periods of associated national programs (i.e., CAMP-España).

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

Many experimental facilities have contributed to the today's availability of a large thermal-hydraulic database (both separated and integral effect tests). However there is a continuous need for additional experimental work and code development and verification, in areas where no emphasis have been made along the past. On the basis of the SESAR/FAP[1] reports *"Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk" (SESAR/FAP, 2001) and its 2007 updated version "Support Facilities for Existing and Advanced Reactors (SFEAR) NEA/CSNI/R(2007)6",* CSNI is promoting since the beginning of this century several collaborative international actions in the area of experimental TH research. These reports presented some findings and recommendations to the CSNI, to sustain an adequate level of research, identifying a number of experimental facilities and programmes of potential interest for present or future international collaboration within the nuclear safety community during the coming decade. The different series of PKL, ROSA and ATLAS projects are under these premises.

CSN, as Spanish representative in CSNI, is involved in some of these research activities, helping in this international support of facilities and in the establishment of a large network of international collaborations. In the TH framework, most of these actions are either covering not

[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).

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 PKL/OECD ROSA/OECD and ATLAS/OECD experiments.
- Analysis of applicability and/or extension of the results and knowledge acquired in these projects to the safety, operation or availability of the Spanish nuclear power plants.

Both objectives are carried out by simulating the experiments and conducting the plant application with the last available versions of NRC TH codes (RELAP5 and/or TRACE).

On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermalhydraulics analysis of accidents in the Spanish nuclear power plants. Nuclear safety needs have not decreased as the nuclear share of the nation's 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

This report presents the main results obtained with the NRC consolidated code TRACE for the simulation of OECD/NEA PKL-2 project test G1.2 conducted at the Primärkreislauf-Versuchsanlage (primary coolant loop test facility) PKL. This facility is owned and operated by AREVA NP and is located in Erlangen, Germany. The PKL-III G test program investigates safety issues relevant for current pressurized water reactor (PWR) plants as well as for new PWR design concepts, focusing on complex heat transfer mechanisms in the steam generators and boron precipitation processes under postulated accident situations. Specifically, the first test series G1 focused in systematically investigating the heat transfer mechanisms in the steam generators in the presence of nitrogen, steam and water, with two experiments being conducted: G1.1 (one loop configuration) and G1.2 (two loop configuration).

The PKL facility models the entire primary side and significant parts of the secondary side of a pressurized water reactor at a height scale of 1:1, with volumes, power ratings and mass flows being scaled with a ratio of 1:145. The experimental facility consists of four primary loops with circulation pumps and steam generators (SGs) arranged symmetrically around the reactor pressure vessel (RPV). The investigations carried out in this facility encompass a very broad spectrum, from accident scenario simulations with large, medium, and small breaks, over the investigation of shutdown procedures after a wide variety of accidents, to the systematic investigation of complex thermohydraulic phenomena, having been in operation since 1977.

The objective of this report is to compare code predictions with experimental data obtained during the PKL tests G1.2 in order to establish TRACE's capability to model heat transfers at atmospheric pressure in presence of nitrogen (modelled as air in the TRACE model, due to its similarity with pure nitrogen), water and steam in the U-tubes and the coolant transfer phenomena inside the U-tubes observed in previous tests, as well as assessing TRACE code precision. As a secondary objective, this report assesses TRACE capability to correctly predict boron concentration variations during several consecutive evaporations and condensations of primary coolant liquid.

During all tests the pressurizer (PRZ) was permanently isolated from the primary circuit, thus not being part of the primary side volumes.

Prior to the transient phase start, a conditioning phase was conducted, during which, test conditions and initial inventory status were arranged. The preliminary test phase started with a complete filling of loops 1 and 2 with subcooled water at a homogeneous boron concentration of 2000 ppm and ambient pressure, then proceeding with a slow drain of the primary inventory down to ¾ loop coupled with a constant feed of N_2 to the primary circuit via the PRZ valve station, thus replacing the void volumes resulting from the drainage. After completing the decrease of primary coolant inventory, rod bundle power was decreased to 200 kW in order to simulate the decay heat in the core, accounting to 0.6% of full load thermal power, including compensation for heat losses. After fixing core thermal power, the system remains under this conditions enough time to reach the steady-state conditions. During this time, core power is removed from the system using the residual heat removal system (RHRS).

Start of test (SOT) begins with the shut-down of RHRS, it causes:
> Heat-up of core inventory.
> Start of steam formation in the core (approximately 10 min after shut-down of RHRS).
> Frothing of core inventory.

Test G1.2 started with stationary conditions and ¾-loop with the RHRS engaged. Throughout the tests, the secondary pressure was controlled at 2 bars by the main steam relief valve (MSRV). Reduction and increase of inventory was accomplished via lower plenum drain line and injection lines into the lower section of the DownComer-pipes (DC-pipes). In this way, the additional coolant was injected into the subcooled fluid and not into steam volumes. Thereby, the steam condensation (and heat transfer in the U-tubes) was left undisturbed by draining/replenishment procedures.

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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. We also thank 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 PKL III facility and many interesting suggestions on the improvement of the model.

ABBREVIATIONS AND ACRONYMS

Abbreviations

B	Boron ([B]: boron concentration, ppm)
CCFL	CounterCurrent Flow Limitation
CI	Coolant Inventory
CL	Cold Leg
CM	Coolant Mass
COMBO	Continuous Measurement of Boron Concentration
CSN	Spanish Nuclear Regulatory Commission
CVCS	Chemical/Volume Control System
DC	Downcomer
DCT	Downcomer Tube/pipe
DCV	Downcomer Vessel
EOT	End Of Test
HL	Hot Leg
ITF	Integral Test Facility
LOCA	Loss of Coolant Accident
MCP	Main Coolant Pump
MS	Main Steam
MSRV	Main Steam Relief Valve
NC	Natural Circulation
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plants
OECD	Organization for Economic Cooperation and Development
PCI	Primary Coolant Inventory
PKL	Test facility, (German acronym for “Primärkreislauf”, means: primary circuit)
PRZ	Pressurizer
PS	Pump Seal
PWR	Pressurized Water Reactor
RC	Reflux-Condenser
RCL	Reactor Coolant Line
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RPV	Reactor Pressure Vessel
SG	Steam Generator
SOT	Start Of Test
STP	Standard Temperature and Pressure conditions
TRACE	TRAC-RELAP Advanced Computational Engine
UNESA	Asociación Española de la Industria Eléctrica
UPTF	Upper Plenum Test Facility

Latin and Greek Symbols

$\dot{m}$	kg/s	Mass flow
p	bar, Pa	Pressure
Δp	bar, Pa	Pressure difference
P	kW	Power
t	s	Time

T	°C, K	Temperature
ρ	kg/m^3	Density

Subscripts

max	Maximum
min	Minimum
prim	Primary
sec	Secondary

1 INTRODUCTION

The present study was performed as a contribution 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 was coordinated by the CSN and a steering committee.

The PKL facility is owned and operated by AREVA NP and is situated in Erlangen, Germany. The analysis of PKL III G1 experiments with the TRACE code were assigned to the “thermal-hydraulics and nuclear engineering group” of the Polytechnic University of Valencia. The PKL tests have been performed in the Primärkreislauf-Versuchsanlage (primary coolant loop test facility) PKL. The TRACE code employed to perform the simulation was the Version 5.0p2 and the SNAP interface version the 2.0.4.

The main objective of test series G1 was the systematic study of heat transfer in the SG U-tubes in presence of nitrogen, and with different primary coolant inventory (PCI) levels in order to establish different water heights inside U-tubes, which will account for different heat transfer mechanisms: single and two-phase coolant flow in U-tubes (being continuous or discontinuous) and blocking of U-tubes due to nitrogen concentration in the upper parts of them. A practical issue which was to be solved by this test series was to support theoretical conclusions regarding stabilization of primary pressure and temperatures, the prevention of boron dilution during LOCAs, and the effectiveness of counter measures to re-establish operation conditions for the Residual Heat Removal System, RHRS (assuming re-availability).

Test G1.2 consists of several stages. The test conditions and the initial inventory status were arranged in the course of the preliminary test phase, which started with a complete filling with subcooled water at a homogeneous boron concentration of 2000 ppm and ambient pressure (p_{prim}~1 bar) of loops 1 and 2. Primary coolant inventory was then slowly drained down to ¾-loop (approximately 1140 kg) coupled with a constant feed of N_2 to the primary circuit via the PRZ valve station, thereby replacing the void volumes emerging from the drainage. The volume of N_2 fed to the primary circuit was approximately 0.8 m^3 at STP.

After the decrease of the primary inventory, the rod bundle power was set to 200 kW (simulation of the decay heat in the core, resembling 0.6 % of full load thermal power, inclusive compensation for heat losses) and kept constant. Until start of test, the core power was removed from the primary circuit via the RHRS engaged in loop 1.

The tests begin with the shut-down of the RHRS. This shut-down causes the heat-up of core inventory and consequently the start of steam formation in the core (approximately 10 minutes after shut-down of the RHRS) and the frothing of core inventory. After 760 seconds (the amount of time deemed necessary to attain steady-state conditions after RHRS shut-down), primary coolant inventory is reduced by 14% of its original value. This establishes a heat transfer mode in U-tubes similar to Reflux-Condenser (RC) operation, with active and passive heat transfer zones. After 1100 seconds, a second coolant drain is performed, accounting for an additional 8.7% of the original coolant inventory.

Heat transfer to the secondary side leads to a temperature and pressure increase on the secondary side. The secondary sides of the active loops were kept constant at a pressure of 2 bars and 12.2 m fill level via MSRV and feedwater injection. Reduction and increase of inventory was accomplished via lower plenum drain line and injection lines into the lower section of the DC-pipes. In this way, additional coolant was injected into already subcooled fluid and not into steam volumes. Thereby, the steam condensation (and heat transfer in the U-tubes) was left undisturbed by draining/replenishment procedures.

The goal of this study is to compare the experimental results with the predictions of the TRACE code and to analyse the set of phenomena that take place at the primary and secondary sides of the Steam Generators (SG). Special emphasis is devoted to the analysis and comparison of the code results and the experimental data.

This study has been divided into four sections: section 1 is a general introduction, section 2 is a description of PKL facility, test G.1.2, and the applied boundary conditions, section 3 deals with the study of the transient sequence and the comparison of TRACE results for the main physical magnitudes versus the experimental data and, finally, section 4 is devoted to present the final conclusions.

2 INITIAL AND BOUNDARY CONDITIONS

2.1 Description of the PKL Facility

The integral test facility PKL (Figure 2-1), which is operated at the Technical Center of Framatome ANP, is a mock-up of a 1300MW class PWR (Kremin, Limprecht et al., 2001). It is used for research into the behavior of the thermal-hydraulic system under accident situations with and without loss of coolant. The test facility simulates the entire primary side with four loops and the essential parts of the secondary side. In view of the importance of gravity during accident situations, all elevations of the test facility correspond to actual reactor dimensions. The overall volume and power scaling factor is 1:145. In order to account for important phenomena in the hot legs such as flow separation and counter current flow limitation, the design of the hot leg bases on the conservation of the Froude number whereas also the results of the experiments in the 1:1 scaled UPTF were taken into consideration.

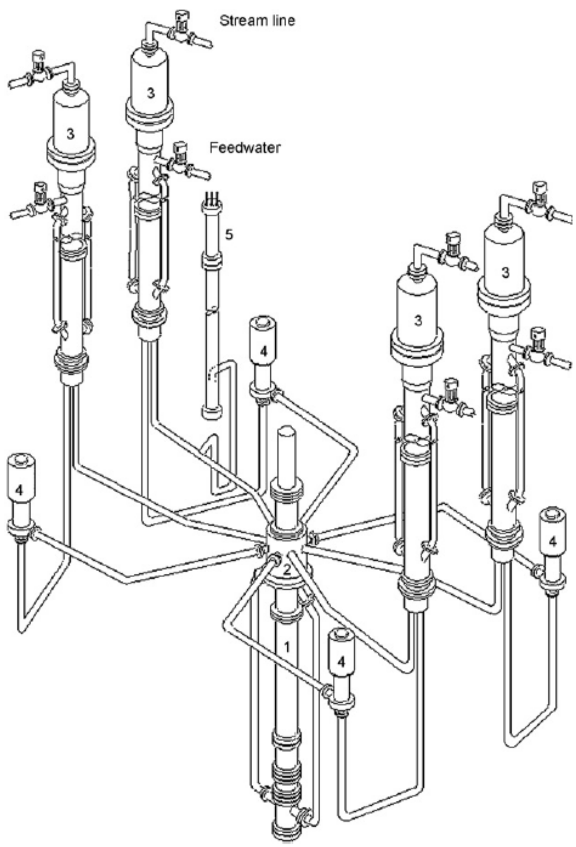

Figure 2-1 PKL facility. (1) Reactor Pressure Vessel; (2) Downcomer; (3) Steam Generator; (4) Pump; (5) Pressurizer. Volume: 1:145; Elevations: 1:1; max. pressure: 45 bars; max. power 2.5MW

The reactor core and the steam generators are simulated as a “section” from the actual system, in other words, full-scale rods and U-tubes are used. The number of rods and tubes has been scaled. The reactor pressure vessel has been modeled by scaling the cross sectional area preserving the full height of the core and the upper and lower plenum. The core is modeled by a bundle of 314 electrically heated rods with a total power of 2.5MW corresponding to 10% of the scaled nominal power. The reactor pressure vessel (RPV) downcomer is modeled as an annulus in the upper region and continues as two stand pipes connected to the lower plenum. This configuration provides symmetrical connection of the four cold legs to the RPV, reliable determination of flow rates, preservation of frictional pressure losses and does not unacceptably distort the volume/surface ratio. The symmetrical arrangement of the four loops around the RPV means that the requirement for identical piping lengths and hence recirculation period is fulfilled. This configuration enables the individual effects of multiple system failures to be studied as well as other events. Experiments on the behavior of a 3-loop (2-loop) plant can also be conducted by simply isolating one (two) loop(s).

Each of the primary-side loops contains active coolant pumps which are equipped with speed controllers to enable any pump characteristics to be simulated. The four fully scaled steam generators are equipped with prototype tubing (diameter, wall thickness, differing lengths) and tube sheet.

By preserving the frictional pressure losses in the steam generators and in the core region, the integral pressure loss for the entire primary system is also very similar to that of the actual plant. The maximum operating pressure of the PKL facility is 45 bars on the primary side and 60 bars on the secondary. This allows simulation over a wide temperature range.

PKL is also equipped with all relevant safety and operational systems on both the primary and secondary side. On the primary side the following are all simulated: four independent high- and low-pressure safety injection systems connected to both the hot and cold legs, the residual heat removal system, eight accumulators, the pressurizer pressure control system and the chemical and volume control system. On the secondary side, the feedwater system, the emergency feedwater system and the main steam lines, with all control features of the original systems are modeled. For the realistic simulation of secondary-side bleed-and-feed procedures, special care was taken to correctly model the feedwater lines and the feedwater tank with respect to the volume (1:145), the elevations (1:1) and the friction losses (1:1). All these features allow the simulation of a wide spectrum of accident scenarios involving the interaction between the primary and secondary side in combination with various safety and operational systems.

The facility is extensively instrumented with more than 1300 measuring points. Besides conventional measurements (temperature, pressure, etc.), two-phase flow measurements can also be made. In addition, for the test series PKL III E, F and the current series PKL III G, special devices for the detection of boron concentration were installed.

To summarize, the PKL facility is a full-height Integral Test Facility (ITF) that models the entire primary system (four loops) and most of the secondary system (except for turbine and condenser) of a 1300-MW PWR.

The facility includes:
- Reactor Coolant System (RCS)
- Steam Generators (SG's)
- The interfacing systems on the primary and secondary side and the break.

The RCS includes:
- The upper head plenum, which is cylindrical, full-scale in height and 1:145 in volume.
- The upper plenum, full-scale in height and scaled down in volume.

- The upper head bypass, represented by four lines associated with the respective loops to enable detection of asymmetric flow phenomena in the RCS (e.g., single-loop operation).
- The reactor core model, consisting of 314 electrically heated fuel rods and 26 control rod guide thimbles. The maximum electrical power of the test bundle is 2512 kW.
- The reflector gap, located between the rod bundle vessel and the bundle wrapper (the barrel in the real plant).
- The lower plenum, containing the 314 extension tubes connected with the heated rods. The down-comer pipes are welded on the lower plenum bottom in diametrically opposite position. Two plates are located in this zone: the Fuel Assembly Bottom Fitting and the Flow Distribution Plate.
- The down-comer modeled as an annulus in the upper region and continues as two stand pipes connected to the lower plenum. This configuration, as already mentioned above, permits symmetrical connection of the 4 Cold Legs (CL) to the RPV, preserves the frictional pressure losses.
- The (four) hot legs, designed taking into account the relevance of an accurate simulation of the two phase flow phenomena, in particular CounterCurrent Flow Limitation (CCFL), in the hot leg piping as in the reactor.
- The (four) cold legs, connecting the SG to the Main Coolant Pump (MCP) through the loop seal and the MCP to the DownComer (DC) vessel. The hydrostatic elevations of the loop seals are 1:1 compared with the prototype NPP.
- The (four) MCP, which are vertical single-stage centrifugal pumps.
- The PRZ, full-height and connected through the surge line to the hot leg #2.
- The SG primary side, modeled with vertical U-tube bundle heat exchangers like in the prototype NPP. The scaling factor has been preserved by reducing the number of tubes (28 tubes with seven different lengths).
- The SG (secondary side) is constituted by the tube bundle zone, seal welded hollow fillers (below the shortest tubes), the DC (with the upper zone annular containing the FW ring, the central zone modeled by two tubes outside of the SG housing and the lower zone with annular shape) and the uppermost part of the SG that models the steam plenum.

2.2 Stationary Initial Test Conditions and Test Phase

2.2.1 Conditioning Phase

The SNAP interface version used to launch TRACE simulations was the 2.0.4 and the code employed to perform the simulations was the TRACE Version 5.0p2.
The TRACE 1-D model of the PKL III E2.2 test conditions (developed by “Grupo de Análisis Dinámico de Sistemas Energéticos del Instituto de Técnicas Energéticas de la Universidad Politécnica de Cataluña”) is the employed model as a starting point. Therefore, first step is to deal with the evolution from E2.2 test conditions to the initial G1 test series conditions. Table 2-1 presents the main characteristics of test E2.2 and G1.2 tests.

Table 2-1 E2.2 vs. G1.2 Test Conditions

Test	E2.2	G1.2
Primary Pressure (bar)	42	≈1
Secondary Pressure(bar)	28	≈1
Rod Bundle Power (kW)	530	≈200
Coolant Inventory (kg)	2250	1140
Boron Concentration (ppm)	1000	2000

As can be seen from Table 2-1 test conditions are quite different in E2.2 and G1.2 test, so thoroughgoing work has been needed to reach the initial G1.2 test conditions. Besides, not only the previous differences shown in Table 2-1 were present: mass flow rates, temperatures, pressures, extraction and injection events, valves adjustment, control systems and many other elements can be modeled or modified. After having achieved the initial steady state, initial G1.2 test conditions, the pre-test phase is ended. Test phase can begin, that phase starts with the shut-down of the RHRS, that sudden lock of the RHRS produces a quick rise of core temperature. A big difference in the temperature evolution experimental and modeled can be appreciated. This contradiction is produced by the own model design, one dimensional components are not able to reproduce mixing processes in natural circulation regime. With the aim of increase the mixing processes a collection of small by-passes (3% of the core average flow area) have been introduced among the different core cells and with the adjacent components (see NUREG "Analysis with TRACE Code of PKL III Tests G1.1 & G1.1a." for further information).

2.2.2 Initial Test Conditions

The test conditions and the initial inventory status were arranged in the course of the preliminary test phase. Two temporal scales are in present tests, the general time scale which begins at the preliminary test phase and the after start of test scale (SOT) which selects as t = 0 the beginning of the test phase.

The preliminary test phase started with a complete filling of the entire loops 1 and 2 with subcooled water at a homogeneous boron concentration of 2000 ppm and ambient pressure (p_{prim} ~ 1 bar). The slow drain of the primary inventory down to ¾-loop (approximately 1140 kg of residual inventory) was attended by a constant feed of N_2 to the primary circuit via the PRZ valve station, thereby replacing the void volumes emerging from the drainage. The volume of N_2 fed to the primary circuit is approximately 0.8 m^3 at Standard Temperature and Pressure conditions (STP) in both test runs.

After the decrease of the primary inventory, the rod bundle power was set to 200 kW (simulation of the decay heat in the core, resembling 0.6 % of full load thermal power, inclusive compensation for heat losses) and kept constant. Until start of test, the core power was removed from the primary circuit via the residual heat removal system engaged in loop 1.

These initial conditions were the operating ones prior to the start of test, SOT. Table 2-2 and Figure 2-2 present this test initial facility configuration.

In test G1.2, only the primary and secondary sides of loops 1 and 2 were in operation during the test. In the primary side, pressurizer system was isolated, Chemical/Volume Control System (CVCS) injected cold water with a boron concentration of 2000 ppm into the lower parts of both RPV DC-pipes, while RPV drain was done via a drain line located in the lower plenum of RPV (in the TRACE model, the drain line was used for both extraction and injection of cold water), and the RHRS was active until SOT, then it was shut-down until the end of test. In the secondary side, feedwater system and main steam system were in operation for the entire duration of the test, in order to maintain pressure and water level constant.

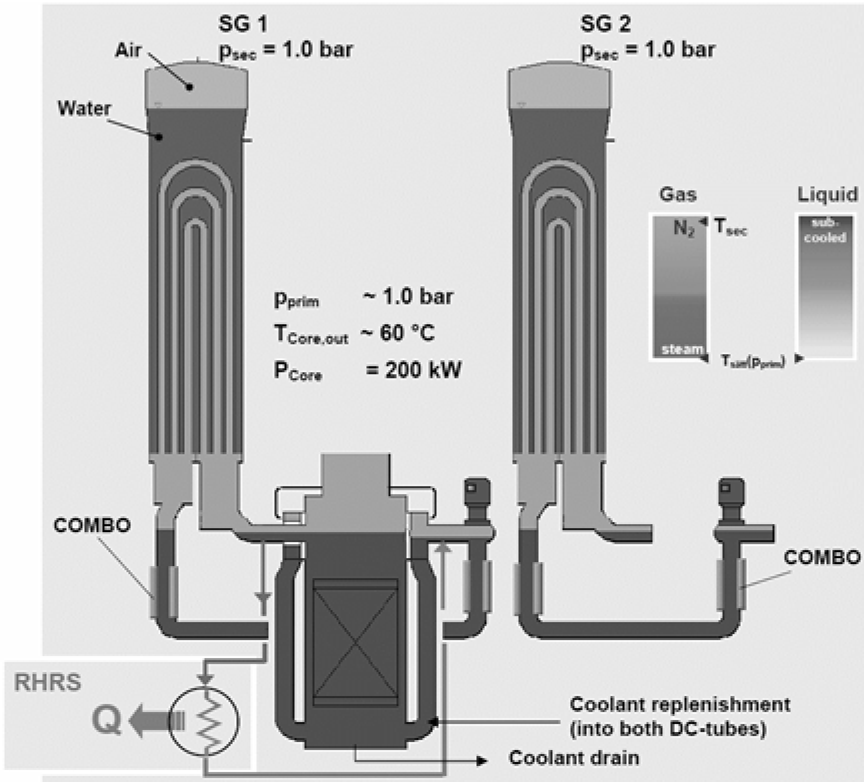

Figure 2-2 Initial (SOT, t=0) Test Facility Configuration for both Test Runs

Table 2-2 Initial Conditions for Test G1.2

Primary side	
General conditions of flow and heat transfer	Cold shutdown conditions. No flow. Loops 1 & 2 filled with water up to ¾-loop, N_2 above. Remaining 2 loops isolated by blank flange close to RPV outlet/inlet. RHRS active in loop 1.
Coolant inventory	1140 kg (PRZ isolated)
Boron concentration	2000 ppm
Heater rod bundle power	200 kW
Pressure	≈ 1 bar (atmospheric pressure)
Fluid temperature at core outlet	≈ 333 K
Subcooling at core outlet	≈ 40 K
Pressurizer fluid temperature	PRZ isolated throughout the whole test
Pressurizer level	
Flow conditions	No flow
Secondary side	
Secondary pressure in SG (remaining SGs not in operation)	≈ 1 bar (atmospheric pressure), MSRV closed
Secondary temperature in SG 1	≈ 298 K
Water levels in SG 1	≈ 12.2 m (air above)

Some differences has been found between the stated conditions at PKL III G1.2, Quick Look Report (Schollenberger, 2009), and the real measured experimental tests conditions at the SOT, all of them are reported in Table 2-3.

Table 2-3 Initial Tests Condition, Stated vs. Real Experimental Test Conditions

	Stated Conditions	**Test Conditions (G1.2)**
General conditions of flow and heat transfer	Cold shutdown conditions. No flow. Loops 1 & 2 primary filled with water up to ¾-loop, N_2 above. Remaining 2 loops isolated by blank flange close to RPV outlet/inlet. RHRS active in loop 1.	Almost no flow, with only slight variations at SG1 Outlet (0.1 kg/s around SOT). 20% of Loops 1 & 2 filled with water, N_2 above. Everything else as stated.
Coolant inventory	1060 kg (PRZ isolated)	No data
Boron concentration	2000 ppm	2250 ppm
Heater rod bundle power	200 kW	223 kW
Pressure	~ 1 bar (atmospheric pressure)	0.94 bar
Fluid temperature at core outlet	~ 60 °C	61.7 °C
Subcooling at core outlet	~ 40 K	~ 37 K
Pressurizer fluid temperature	PRZ isolated throughout the whole test	As stated
Pressurizer level		
Flow conditions	No flow	No flow

2.2.3 Tests Run Conditions

Test G1.2 was started at steady state conditions at SOT time scale t = 0 (5500 s of the general scale time) with the initial conditions described in the above tables (see Table 2-1 and Figure 2-2). The G1.2 test run started from ¾-loop with the RHRS still active. The test starts with the shut-down of RHRS. As a consequence a heat-up of core inventory is produced, starting the steam formation in the core after approximately 10 minutes after the shut-down and consequently the frothing of the core inventory.

The reduction of primary coolant inventory is caused in order to establish a swell level in the SG inlet chamber below the tube sheet. This established a heat transfer mode in U-tubes similar to RC operation, with active and passive heat transfer zones. After that reduction of the primary coolant inventory a gradual increase is carried out. Previously to the coolant inventory increase a (quasi-) steady state conditions have been establishment. During the whole test run heat transfer to the secondary side leads to a temperature and pressure increase on the secondary side. But the secondary side of loops 1 and 2 were kept constant at 2 bars pressure and 12.2 m fill level via the main steam relief valve (MSRV) and feed water injection. The procedure returned a sequence of phases at steady-state operating conditions.

Reduction and increase of inventory was accomplished via lower plenum drain line and injection lines into the lower section of the DC-tubes. In this way, additional coolant was injected into already subcooled fluid and not into steam volumes. Thereby, the steam condensation (and heat transfer in the U-tubes) was left undisturbed by draining/replenishment procedures.
In Table 2-4 there are chronologically displayed the changes of coolant inventory and significant events during test phase for the PKL III facility in the course of test G1.2. The evolution of the main parameters are presented in Figure 2-3.

Table 2-4 Test Run G1.2: Changes of Coolant Inventory and Significant Events

General Time [s]	Time after SOT [s]	Measures / Events	Primary coolant inventory [kg] +/- 20kg
0		Preliminary Test Phase	1140
5500	0	Start of Test (SOT)	1140
6260	760	Start of coolant drain	1140
6735	1235	End of coolant drain	980
7865	2365	Start of coolant drain	980
8175	2675	End of coolant drain	880
19520	14020	Start of coolant injection	880
20015	14515	End of coolant injection	920
26070	20570	Start of coolant injection	920
26705	21205	End of coolant injection	960
30340	24840	Start of coolant injection	960
31115	25615	End of coolant injection	1010
35125	29625	Start of coolant injection	1010
35545	30045	End of coolant injection	1035
39350	33850	Start of coolant injection	1035
39895	34395	End of coolant injection	1070
43250	37750	Start of coolant injection	1070
44800	39300	End of coolant injection	1125
50095	44595	Start of coolant injection	1125
51360	45860	End of coolant injection	1205
54570	49070	Start of coolant injection	1205
56035	50535	End of coolant injection	1300
59200	53700	End of test	1300

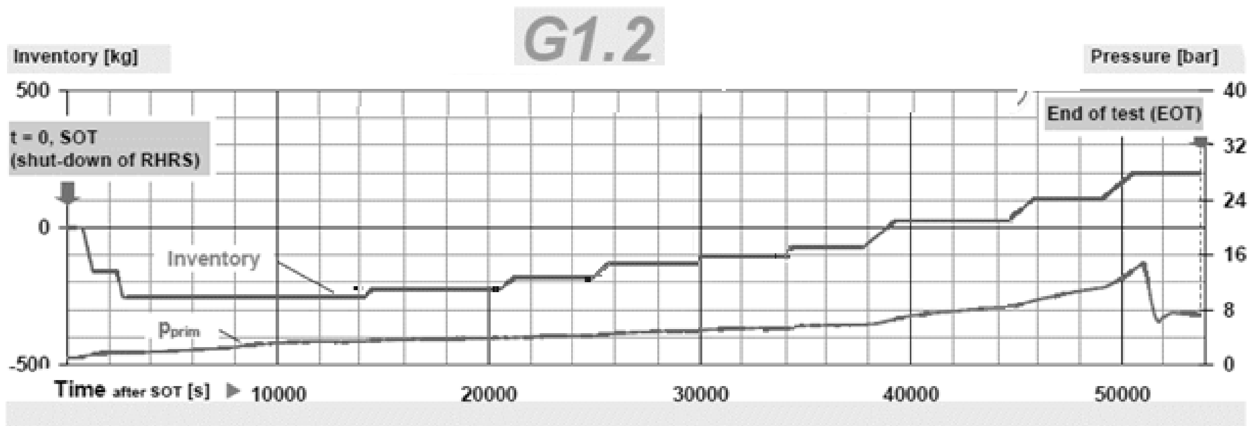

Figure 2-3 Evolution of Main Parameters (Inventory, Primary Pressure)

Some differences has been found between the stated conditions at PKL III G1.2 Quick Look Report and the real measured experimental tests conditions during the tests run, all of them are presented in next figures. Figure 2-4 presents the variation of the rod bundle power during test run. It can be seen that it remains constant at a higher output level than the stated 200 KW. A value of around 223 KW was used in the TRACE model.

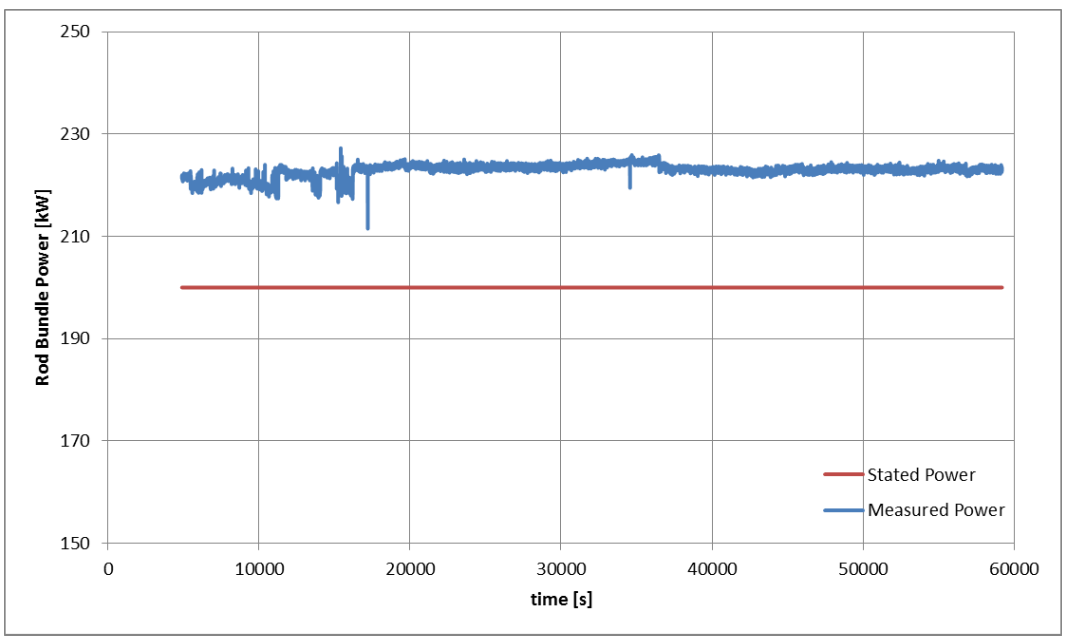

Figure 2-4 Rod Bundle Power Variation vs. Time throughout G1.2 Test Run (MST 2713) and the Stated at Quick Look Report

Refilling of coolant inventory is stated to be made by injection of “cold” water with [B] = 2000 ppm into lower parts of both RPV DC-pipes, the temporal evolution of this injection temperature is presented in Figure 2-5.

Secondary feed water system is stated in Quick Look Report to be at 298 K for the whole tests run, whereas the experimental data shows temperature changing constantly in every single test, see Figure 2-6.

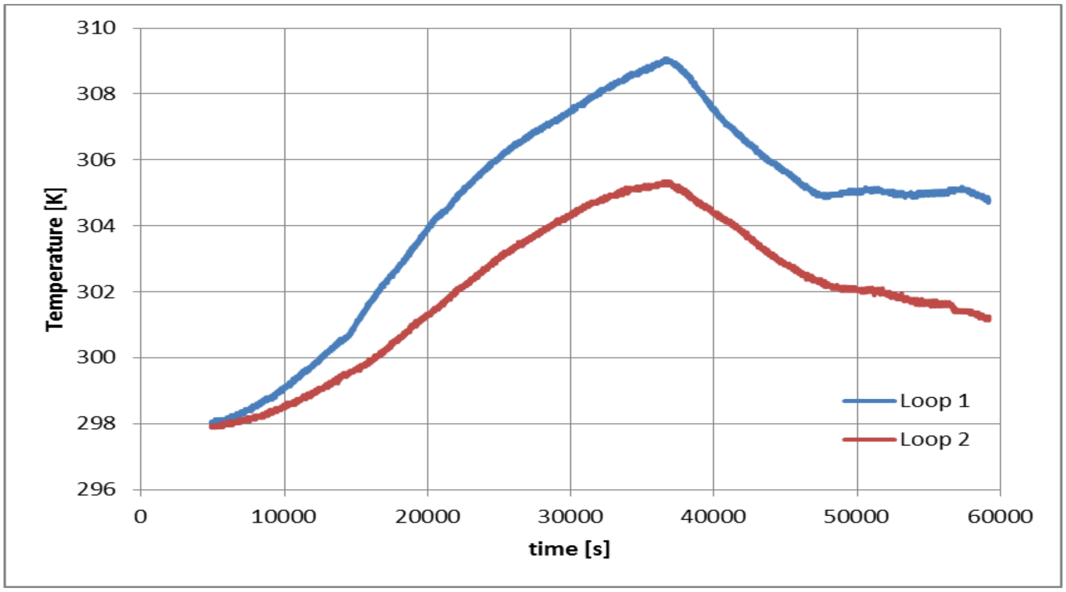

Figure 2-5 Coolant Inventory Replenishment Temperature vs. Time along G1.2 Test Run (MST 1830&1831)

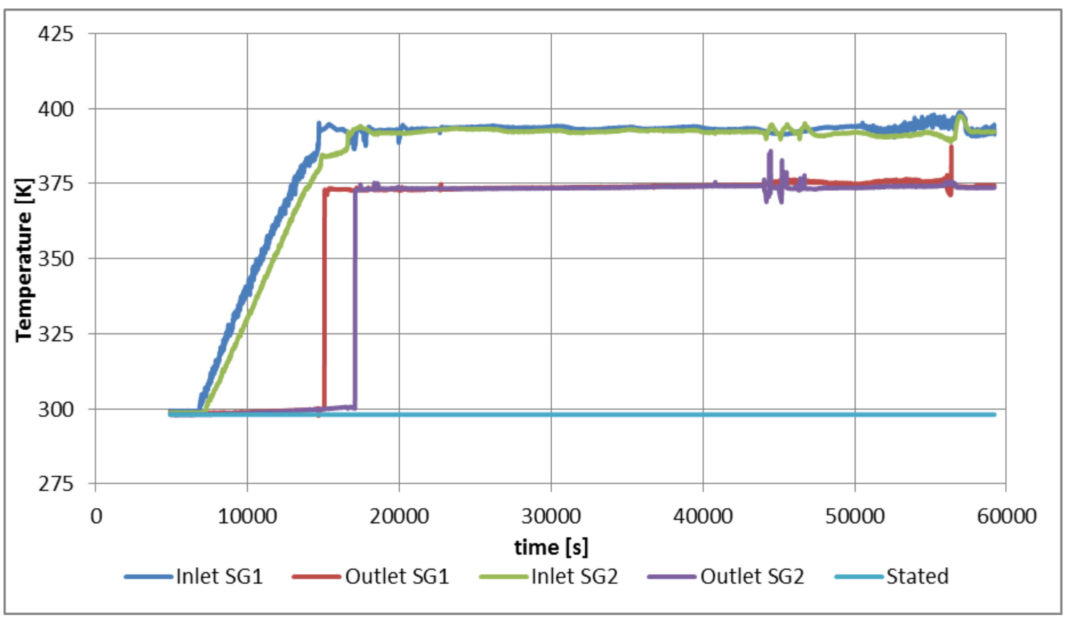

Figure 2-6 Water Temperature of the Steam Generator Secondary Side vs. Time throughout G1.2 Test Run (Inlet SG1 MST 817 & Outlet SG1 835, Inlet SG2 MST 919 & Outlet SG2 917) and the stated at Quick Look Report

3 TEST RESULTS AND COMPARISON WITH TRACE

This section is devoted to review the main experimental results and to discuss the results obtained with the TRACE code. The comparison of the main characteristics curves between experimental PKL facility and TRACE simulation results to G1.2 are presented in the next paragraphs. First graphic presented is the comparison of experimental measures versus TRACE simulation results for vessel liquid level. Next figures presented are the comparative temperature curves in the most significant components, such are the core, downcomer, lower plenum, hot line, steam generator and cold leg. Finally the pressure curve in the upper plenum is presented.

Figure 3-1 shows the comparison between experimental PKL facility data and TRACE calculations of the water level in the vessel. The calculations performed by TRACE provide a quite good correlation of the water level into the vessel, although, in some periods of the transient the level calculated differs from the experimental data. For instance, the initial decrease in level at the end of first extraction is more pronounced in the experimental test, while the opposite occurs in the second one. Besides the level fluctuations in the simulation are usually more marked during the initial part of the transient, while the opposite takes place during the rest of it. It is also worth noting that the water level modeled with TRACE was maintained below the experimental measurements throughout the transient.

The experimental data versus TRACE simulation results of the core temperature are shown in Figure 3-2. Both temperatures evolve in a similar way until the second 9000 approximately, with the abrupt initial increase caused by the shut-down of the RHRS. But from that moment on, the increase in the TRACE simulation temperature curve has a bigger slope than the measured in the experimental facility. At about 21000 seconds of transient the temperature of the TRACE simulation stabilizes among 450 K approximately, while the experimental data curve continues is increase throughout the transient, reaching almost until 475 K near the end of the transient.

Figure 3-3 presents the experimental versus the TRACE code results for the lower plenum temperatures. This figure is very similar to the previous one, in both is shown the sharp initial increase caused by the shut-down of the RHRS, the temperature with the TRACE simulation code increases gradually until approximately 25000 seconds, then stabilizes around 450 K. Although it has small temperature falls of about 15-25 K coinciding with the coolant injections but with some delay. Whereas experimental values increase sharply at the beginning, with a greater slope than the modeled with TRACE, then continue with a consistent and smooth rising along the test. The experimental measurements end with several abrupt falls.

The comparison among the downcomer temperatures at several elevations are presented in Figure 3-4. All four TRACE simulation temperatures evolve in a similar way, initial abrupt increase, with a subsequent stabilization, fluctuations between 425-450K, whereas the experimental measures present a lower temperature increase. The top and middle downcomer cells temperatures evolve in a similar way, first a constant increasing slope followed by several falls and recoveries, but with a smaller slope for the middle section. Meanwhile in the bottom measurement nothing happens until almost the end of the test, where a small increase in temperature appears.

As far as hot leg temperatures concerns (see Figures 3-5 & 3-6) we can say that they (both loop1 & 2) evolve in a very similar way than the core temperature, Figure 3-2, and lower plenum temperature, Figure 3-3.

Figures 3-7 & 3-8 present the steam generators (Loops 1 & 2) inlet level measurements. The experimental values remain in cero until 27000 seconds approximately whereas the TRACE simulation starts its increase at approximately 7000 seconds. The TRACE level simulation increase continues for approximately 10000 seconds, after this period of time stabilizes at about 6 meters for both cases until the prior to the unexpected termination of the simulation. However, the experimental values have an almost constant slope, producing an almost constant increase of the SG's level throughout the duration of the tests, although it ends with a sharp fall.

Values of the steam generators experimental data versus TRACE simulation temperatures values at two elevations are presented in Figures 3-9 & 3-10. A sudden temperature rise in all four cases takes place almost at the start of the transient, being much more pronounced in the case of the SG 1 for the experimental measures than for the simulated values. Whereas for SG 2 both, the experimental and the TRACE results, are almost equal. In all cases this sudden rise are followed by an almost flat profile, but with wide fluctuations in some cases, being wider in the experimental data at 2 meters of the SG inlet. Note that for the SG2 the experimental and the TRACE simulations curves are overlapped.

The cold leg temperatures are shown in Figures 3-11 & 3-12. TRACE simulation presents a quick increase from about 11000 to 20000 seconds, rising from 300 to 425 K approximately, followed by a gradual decrease until the unexpected interruption of the test simulation, being quite similar in all the CL regions. Whereas for the experimental measures of the CL pipes (both loops 1 & 2) this abrupt increase takes place in a similar way than in the TRACE simulation., but the abrupt temperature rise is followed by an almost constant increase, not decrease as in the simulation results predicted by TRACE. The inlet pump and pump seal temperature measurements remain constant, until approximately 17000 and 39000 seconds respectively. At these moments both of them present a gradual increase along 2000 seconds. This small rise in temperature is followed by a sharp increase, along 3 or 4 thousand seconds, until the experimental CL pipes temperatures are reached. Then all of them evolve with a progressive increase, ending with an abrupt decrease approximately at the second 56000.

Pressure in the RPV upper plenum is shown in Figure 3-13. TRACE simulation matches the experiment almost perfectly from SOT (5500 seconds) until the end of the second coolant drain (at 8176 seconds). From this point on, a steady growth in the TRACE model pressure takes place, this increase exceeds for long the measurements of the experimental facility. This increase continues steadily until arriving at a maximum of 10 bars at around 25000 seconds, only minor decreases appears following injections at 19000 and 26000 seconds. From this point onwards the pressure fluctuates slightly for approximately 10000 seconds around this point, increasing the fluctuations from 41000 seconds until the abnormal simulation termination at second 44000 approximately. Meanwhile the experimental measurements continue evolving with a constant slope, but always with lower values than those of the TRACE simulation. This growth continues until the 55000 seconds approximately, but with a slope increase from around 45000 seconds, reaching a pressure of 14 bars. The experimental measurements end with a sharp fall down to 6 bars.

Figures 3-14 & 3-15 present the evolution of the experimental boron concentrations versus the TRACE simulation results. First comment that the initial boron concentration is about 2250 ppm, compared to the 2000 assumed in the previous report. However, both experimental curves (loops 1 and 2) evolve similarly up to 25000 seconds, from this point the measured boron concentration below the pump sharply falls to zero, in about 2-3 thousand seconds. Meanwhile below the SG remains constant at approximately 2250 ppm until the second 32000, decreasing to 1000 ppm at about second 45000. The test ends with a sharp rise up to the initial value, approximately about 2400 ppm. As far as the TRACE simulation is concerned, for SG1 both curves obtained for the two measuring points evolve similarly. Namely, since the initial value of 2000 ppm, a decrease until 1250 ppm from 10000 to 16000 seconds approximately, with large fluctuations below the pump, reaching up to 0. From this moment on, the boron concentration remain almost constant for about 18000 seconds, although with a peak about the second 18000 for the pump seal calculations and large fluctuations below the pump reaching 0. Unexpectedly, for SG2 the initial value of boron concentration has decreased at 1500 ppm, it remains constant until second 10000, moment in which boron concentration decreases to 1000 ppm at 15000 seconds. It remains constant for 17000 seconds approximately. It is followed by a small fall of 200 ppm approximately, followed by a sharp increase up to 2000 ppm in 3000 seconds, after which boron concentration stabilizes in the vicinity of 2000 ppm with certain variations. Moments before the end of the simulation with a sudden interruption, the boron concentration falls 500 ppm.

Regarding the comments on the TRACE simulation results, it can start by saying that, although up to approximately 10000 seconds the main experimental measurements and the obtained with the simulation are quite similar, such as temperatures and pressures in the vessel region. From this point on, there are a major break between experimental values and the results obtained with the TRACE simulation. Even though, acceptable results beyond the aforementioned point are obtained for the temperature values of several variables, such as the lower plenum, the SG and HL. But it does not happens for the temperatures in the downcomer and CL, the upper plenum pressures and the SG levels. The main conclusion that can be drawn is the fact that the experimental curves and the TRACE simulations are very different, namely, the code cannot simulate the experimental conditions.

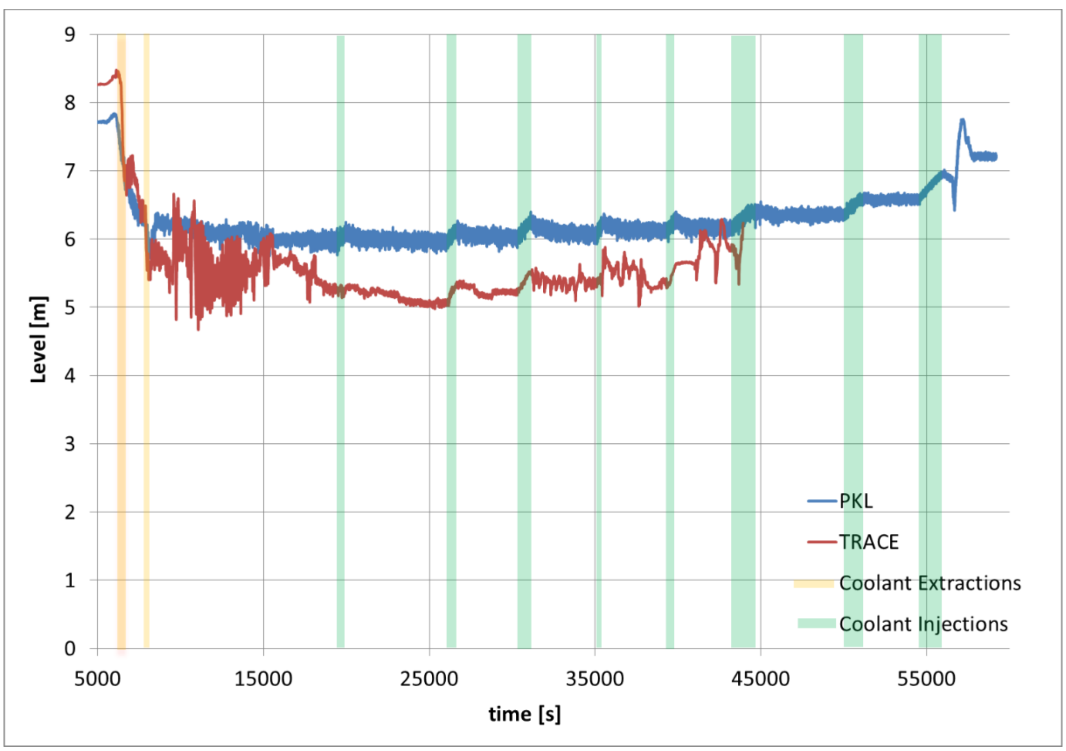

Figure 3-1 Test G1.2, Experimental Vessel Level of PKL Facility (MST 45) vs. TRACE Simulation

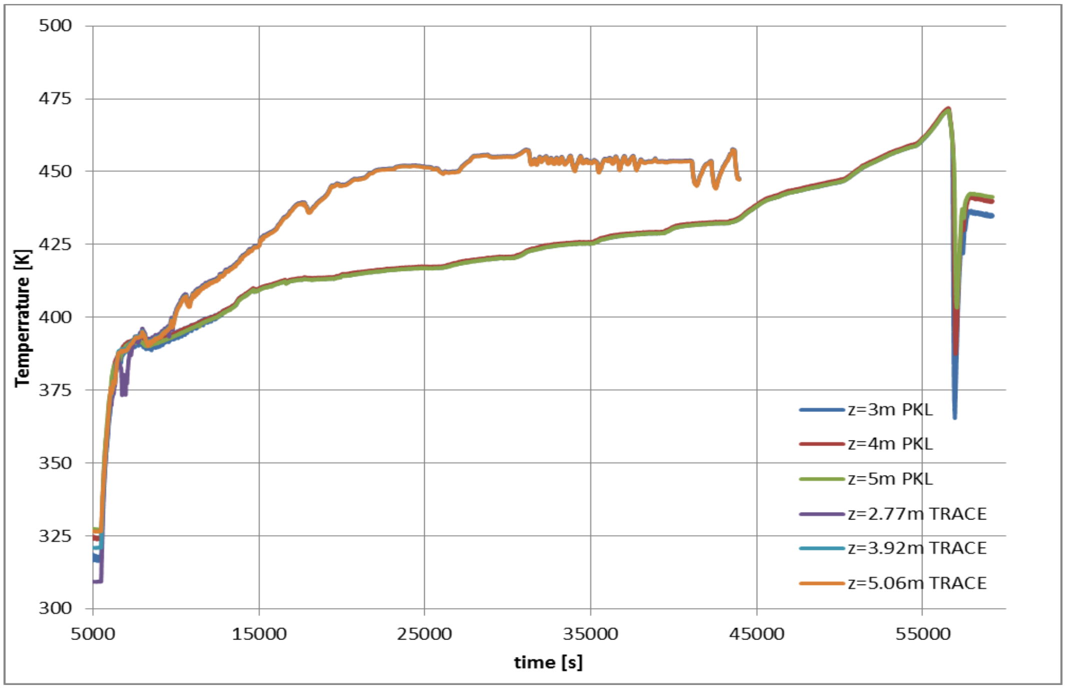

Figure 3-2 Test G1.2, Experimental Core Temperature of PKL Facility (MST 612, 574 & 575 respectively) vs. TRACE Simulation (Pipe 120, cells 2, 4 & 6 respectively)

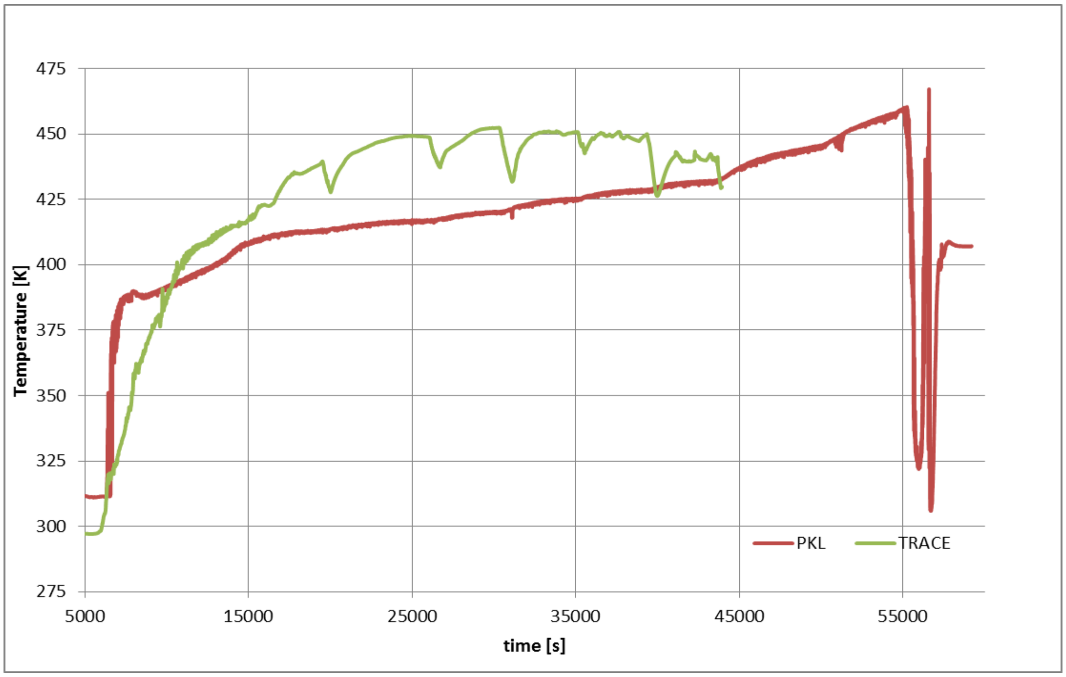

Figure 3-3 Test G1.2, Experimental Lower Plenum Temperature of PKL Facility (MST 636) vs. TRACE Simulation (Plenum 110)

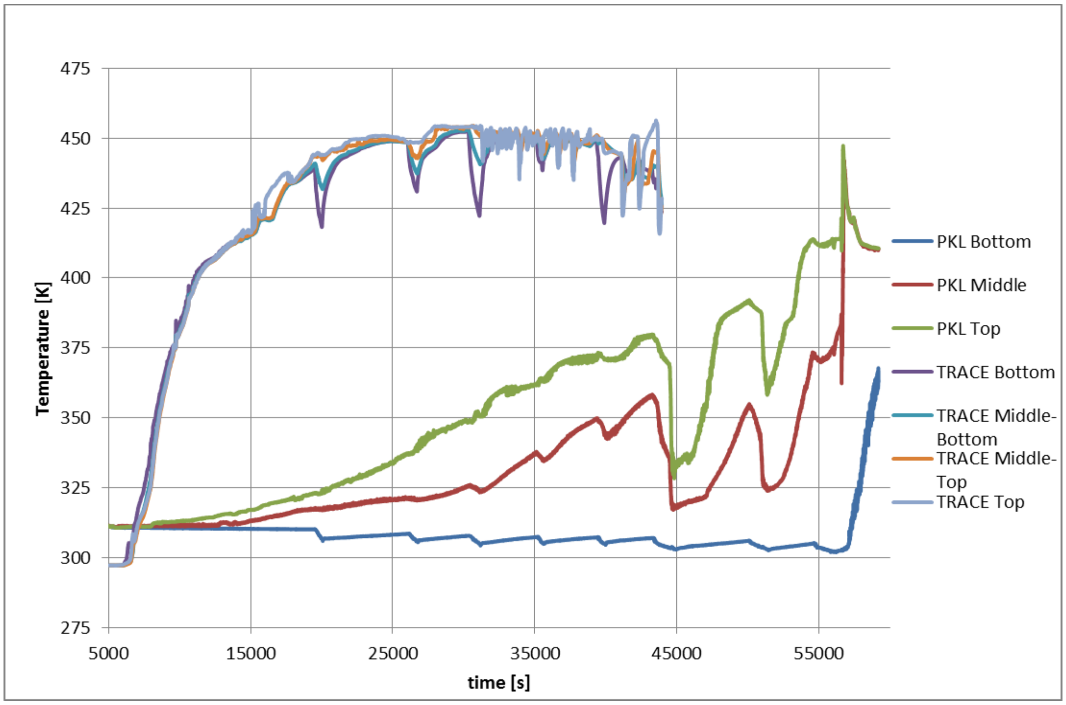

Figure 3-4 Test G1.2, Experimental Downcomer Temperature of PKL Facility (MST 1153, 1155 & 1157 respectively) vs. TRACE Simulation (Pipe 104, cells 4 to 1 respectively)

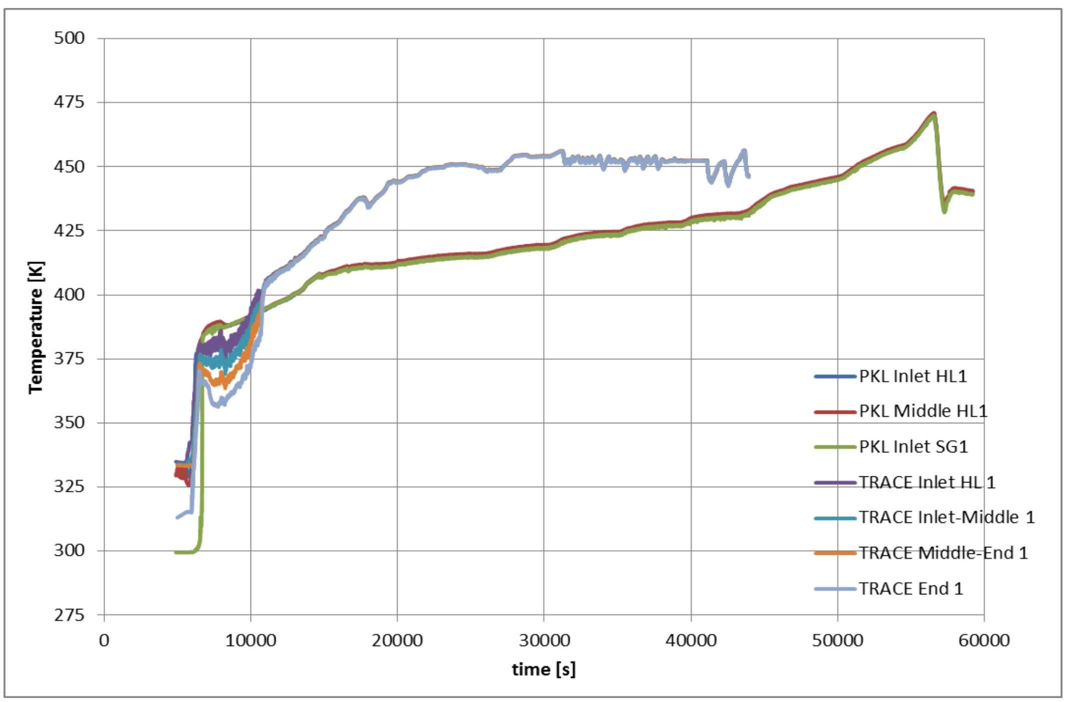

Figure 3-5 Test G1.2, Experimental Hot Leg 1 Temperature of PKL Facility (MST 1167,1170 & 1194 vs. TRACE Simulation (Pipe 210 Inlet GV, cells 1-4)

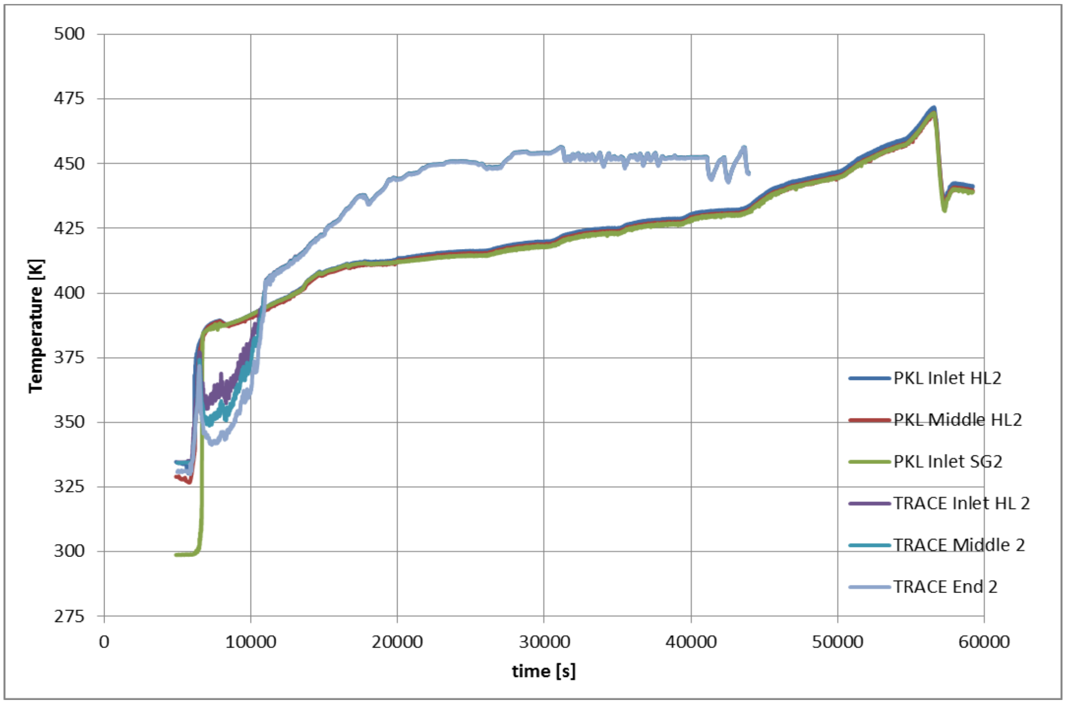

Figure 3-6 Test G1.2, Experimental Hot Leg 2 Temperature of PKL Facility (MST 1174, 1177 & 1154 vs. TRACE Simulation (Valve 81 Inlet GV, cells 1-3)

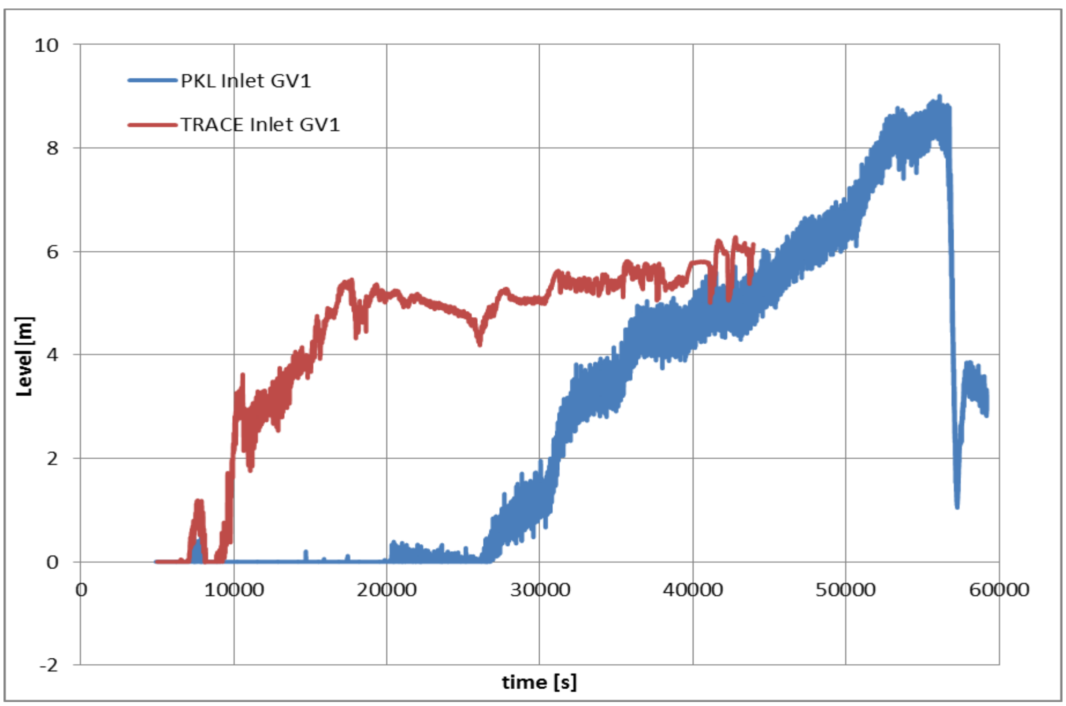

Figure 3-7 Test G1.2, Experimental Steam Generator 1 Level of PKL Facility (MST 71) vs. TRACE Simulation

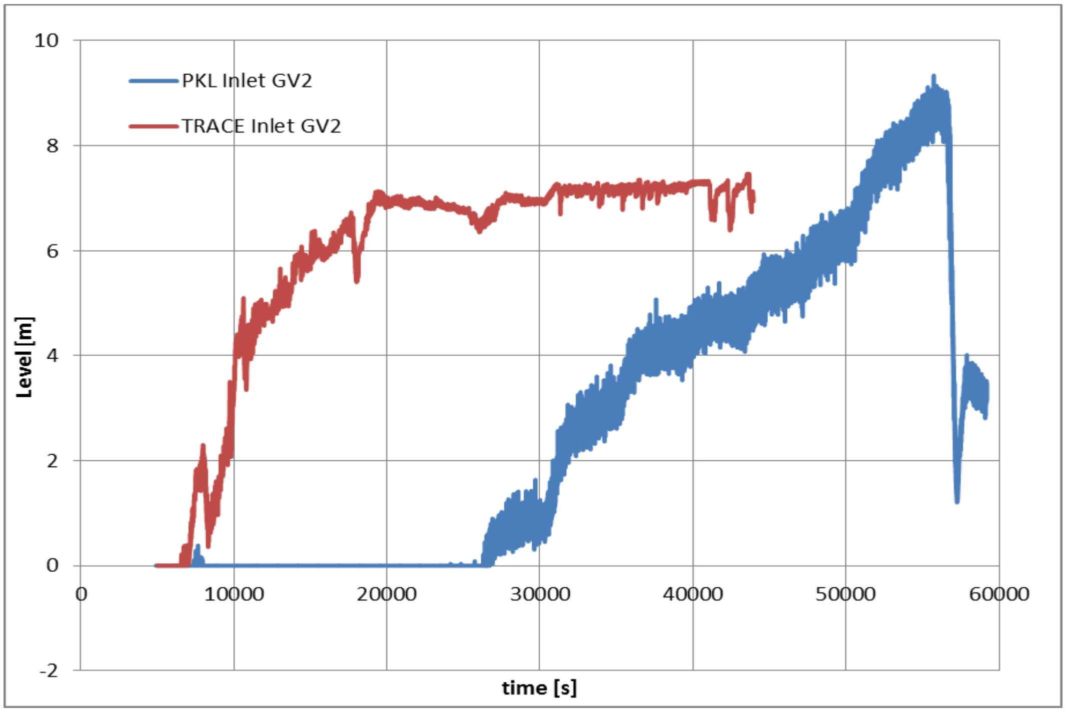

Figure 3-8 Test G1.2, Experimental Steam Generator 2 Level of PKL Facility (MST 81) vs. TRACE Simulation

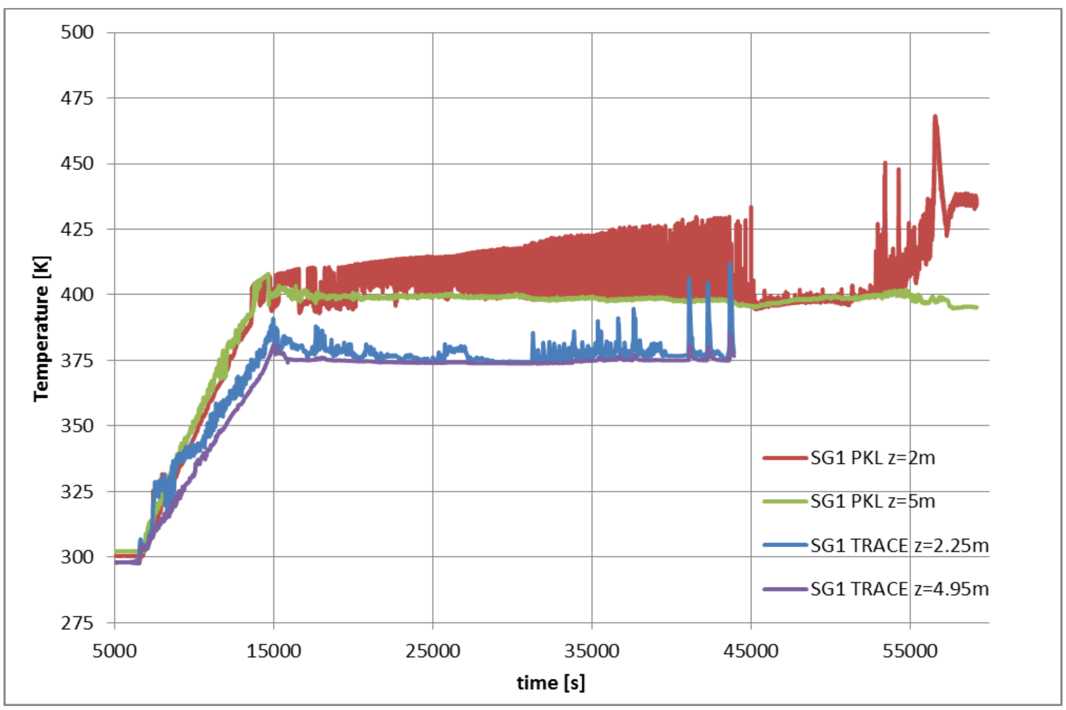

Figure 3-9 Test G1.2, Experimental Steam Generator 1 Temperature of PKL Facility (MST 745 & 753) vs. TRACE Simulation (Pipe 235, cells 3 & 6)

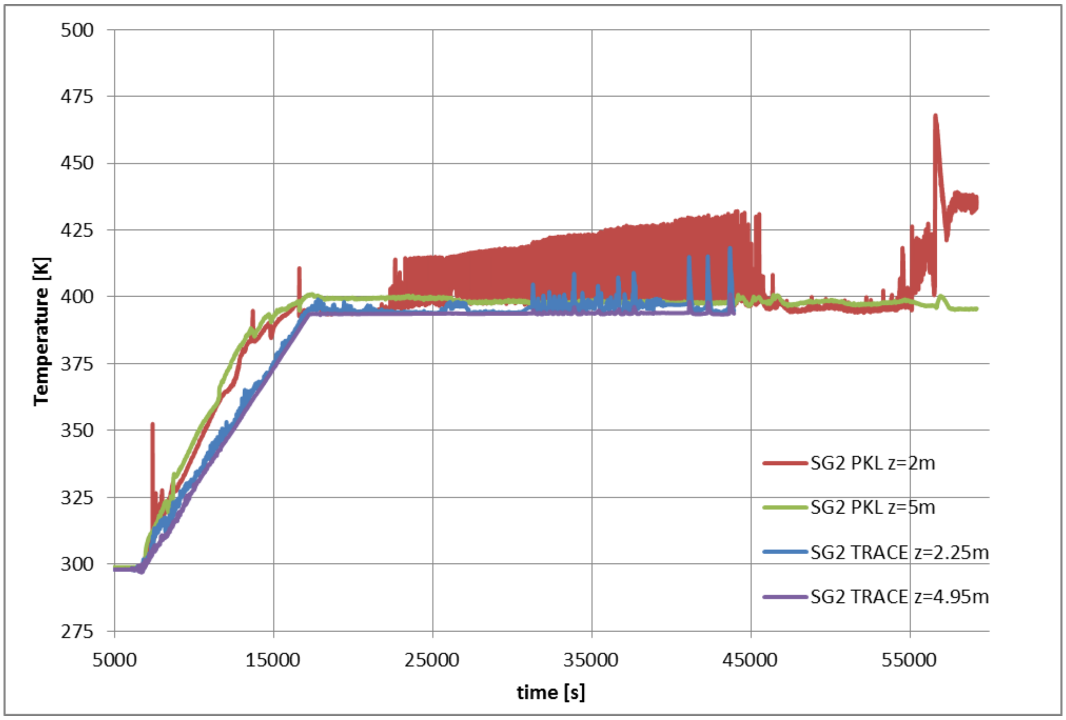

Figure 3-10 Test G1.2, Experimental Steam Generator 2 Temperature of PKL Facility (MST 872 & 876) vs. TRACE Simulation (Pipe 335, cells 3 & 6)

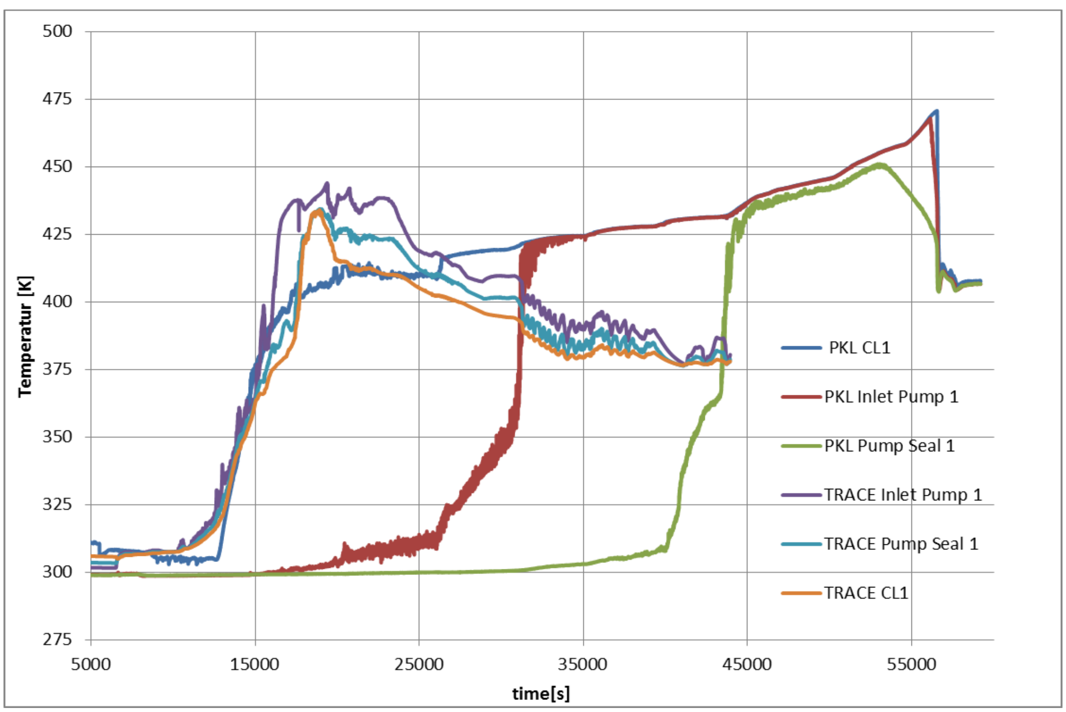

Figure 3-11 Test G1.2, Experimental Cold Leg 1 Temperature of PKL Facility (MST 1373, 1161 & 1207) vs. TRACE Simulation (Pipe 249)

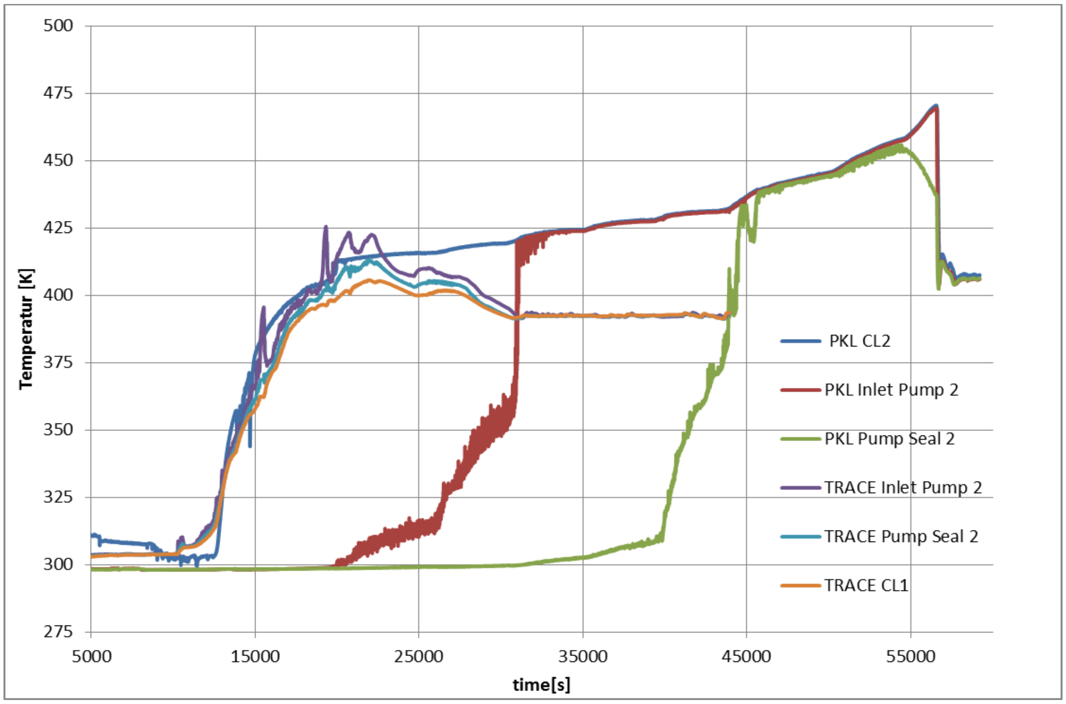

Figure 3-12 Test G1.2, Experimental Cold Leg 2 Temperature of PKL Facility (MST 1230, 1162 & 1209) vs. TRACE Simulation (Pipe 349)

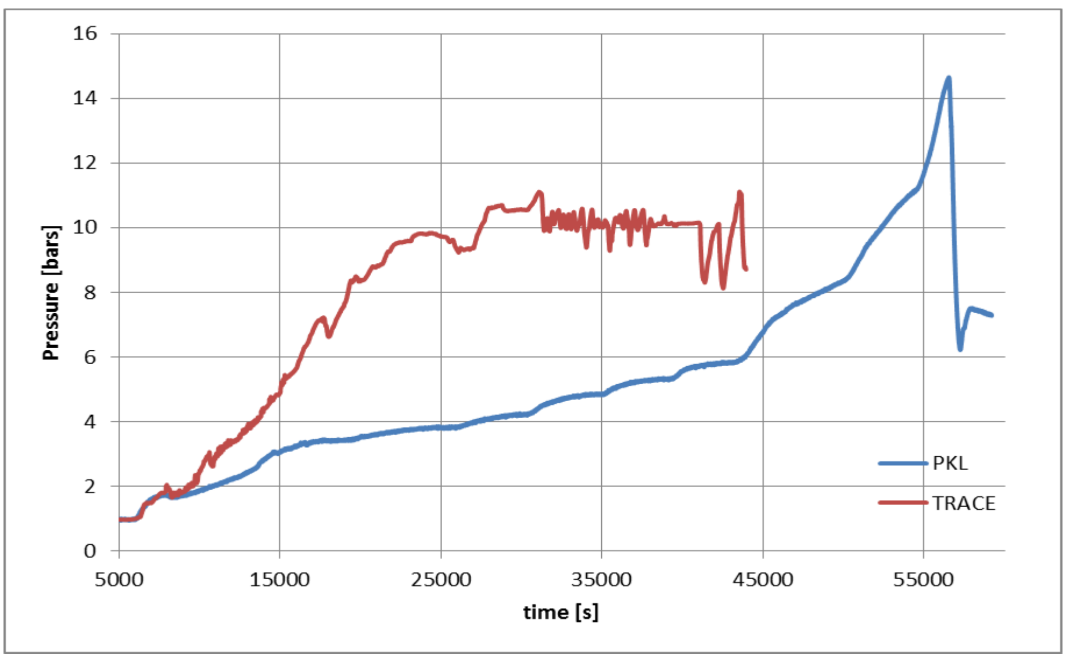

Figure 3-13 G1.2, Experimental Upper Plenum Pressure of PKL Facility (MST 243) vs. TRACE Simulation (Plenum 160)

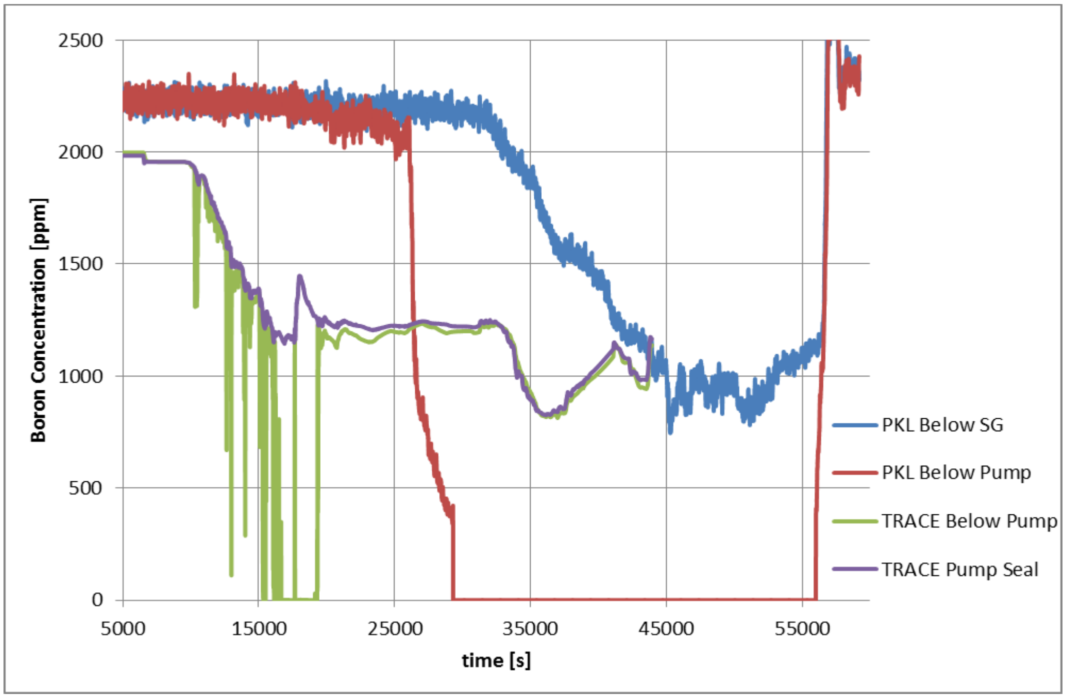

Figure 3-14 G1.2, Experimental Boron Concentration in Loop 1 of the PKL Facility (MS1557 & 1556) vs. TRACE Simulation (Pipe 249, cells 3 & 2)

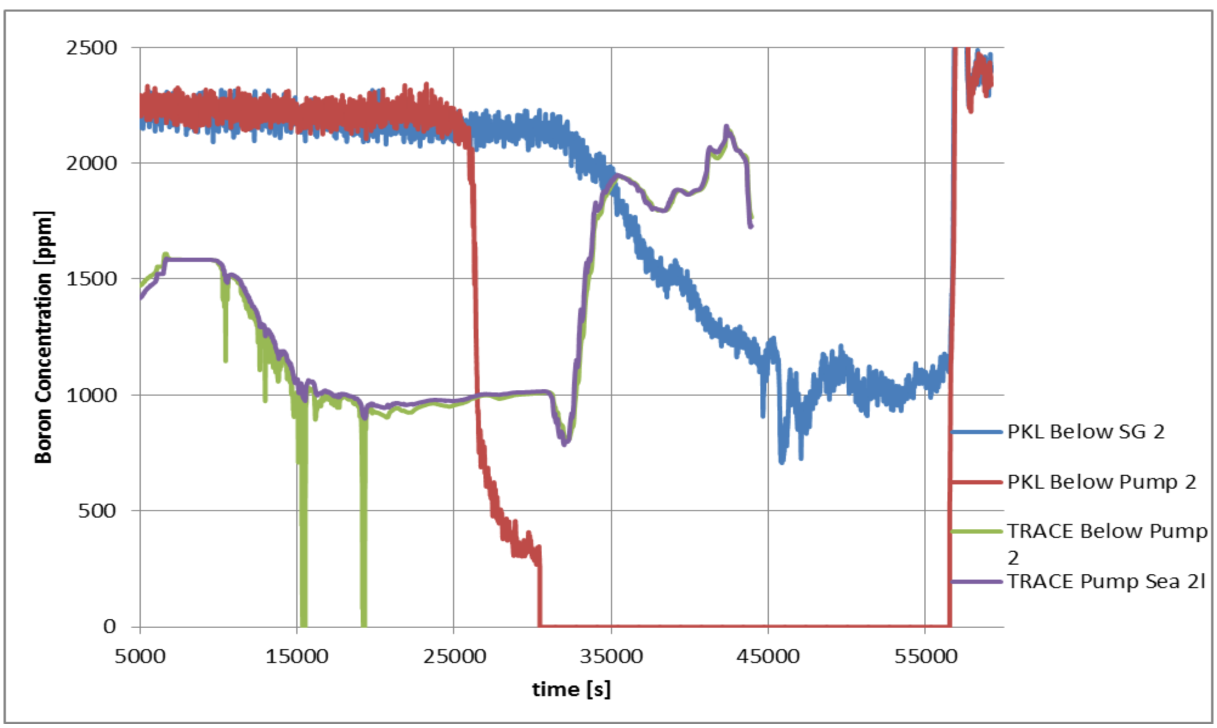

Figure 3-15 G1.2, Experimental Boron Concentration in Loop 2 of the PKL Facility (MST 1555 & 1554) vs. TRACE Simulation (Pipe 349, cells 3 & 2)

4 CONCLUSIONS

The PKL III test facility simulates a typical 1300 MWe pressurized water reactor of Siemens / KWU design. During the test run, the primary coolant inventory was at 3/4-loop level and thus, the heat transfer mechanism in the steam generator in presence of nitrogen, steam and water as a function of the primary coolant inventory in single loop operation has been investigated.

In this document, a post-test analysis of PKL III G1.2 test run using TRACE code, TRACE version 5.0p2, has been presented. The comparison of measured values in the PKL III facility and calculated results by a TRACE model is discussed.

In these experiments, a PKL III facility model created with TRACE code has been used to reproduce the whole tests run performance. Initial steady-state conditions were achieved by maintaining every variable of the model at the intended initial value for 5500 seconds in order to assure that every relevant variable stabilized at a value close to the intended, experimental ones. After this preliminary period, the test starts and runs for around 38000 seconds, until the simulation stops with a “Fatal Error”.

First important aspect to report on is the implementation of several by-passes in the core and adjacent components to “reproduce” the mixing and natural flow processes, which TRACE model of the PKL facility is unable to simulate. This artifice arises from inconvenience that the code is not intended to take into account mixture processes under natural circulation, i.e., TRACE code has not implemented turbulence models. As explained above, the cause of such inability is the fact that natural circulation processes cannot be well reproduced with a 1-dimensional vessel, as our TRACE model of the PKL facility has implemented up to this moment, but even with 3-dimensional components an accurate reproduction of this kind of behaviour could not be assured. Because 3-dimensional components (vessel components in the core region and/or pressurizer) probably would not be able to reproduce accurately the experimental measurements, unless fine mesh and turbulence models would be introduced, which are out of the code aims. Consequently, the current document displays the transients simulation results using only 1-D components, information which is valuable because explores the TRACE V5.0patch2 code behaviour beyond its design limits.

Regarding the simulation results, it has to be pointed out that TRACE code presented a very close match to the experimental results from start of test up to the end of the second drain, diverging from that point onwards, although reaching a maximum not far away from the experimental maximum reached almost at the end of test. Divergences seem to be lower at the core that at other parts of the facility, but it is difficult to make a statement about these differences, as they could be accounted for differences of heat losses between TRACE model and the PKL facility, the larger proportion of subcooled water vs. steam and nitrogen, and so on.

As a result of the previous comments, we can conclude that the actual model of PKL facility implemented in the TRACE code is unable to reproduce natural circulation coupled with two-phase flow (which is even more apparent after the program reaching an abnormal termination in spite of the bypasses). Although, it has some capability to cope with this kind of phenomena when there is only a small percentage of steam and NC gases present, as shown by the good results obtained until the end of the second coolant extraction.

The main reason of these differences, as explained through the document, arises from the fact that natural circulation processes are the dominant mechanisms during practically the whole the transient. Specifically by their increase in predominance as the difference in temperature among different levels of the liquid phase in the core region accentuates, maximum difference which is reached at the end of the core head up. Then, from this moment on, these natural circulation processes are even more dominant, as TRACE code cannot reproduce them appropriately, the simulations results differs from the experimental data. Differences which are especially evident when using 1-D components in the core region. Consequently, in order to be able to capture the evolution of these processes, the code should had be programmed with at least a simple turbulence model and with the possibility to implement a finest meshing of the different components, mainly those of the core region. Consequently, not even with a 3-D vessel in the core region the code would probably be able to capture these phenomena, as TRACE was not intended originally to capture this turbulence phenomenon. Added to this predominance of the long-term natural circulation processes, boron dilution and diffusion processes are also present, which lead to important differences between the experimental data and the simulation results, caused by turbulence phenomena. Consequently, the current document confirms the difficulties that TRACE code has in simulating these long-term experiments in which forced circulation has no longer importance, these huge difficulties are common to all the different thermal-hydraulic codes. In addition, a so extremely long transient, with almost sixty thousand seconds, and which evolves under natural circulation and diffusion processes are beyond the initial objectives of TRACE design. But, despite the very high difficulty of this transient simulation, at the same time, this simulation is useful to find and explore the calculation limits of the different TRACE code versions. In particular, this document mainly explores the capacity of TRACE V5.0 patch 2 to reproduce natural circulation processes in two long-term transients, displaying the code behaviour beyond the design limits of the code.

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NRC FORM 335 (12-2010) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) **NUREG/IA-0521**
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10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

The goal of this report is to explain the main results obtained in the simulations performed with the consolidated thermal-hydraulic code TRACE regarding tests PKL III G1.2. The G1 test series are composed of G1.1, G1.1a and G1.2 tests, all of them are focused on the occurrence of boron dilution processes following the loss of Residual Heat Removal System (RHRS) during ¾-loop operation (primary circuit still closed). Main objective was to provide a data basis for thermal-hydraulics codes for a better understanding of the heat transfer mechanisms in the Steam Generator in presence of Nitrogen, steam and water in the U-tubes and of the coolant transport phenomena observed inside the U-tubes. The differences among G1.1/G1.1a and G1.2 test series are the coolant drain-injections sequences and the loop configuration, 1 loop or 2 loops in operation respectively. The main goal of this report is to analyze the capacity of TRACE V5.0p2 code to precisely simulate thermal stratification and natural circulation of both single and two-phase fluxes inside the whole primary circuit, as well as accurately predicting boron concentration variations.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
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