NUREG/IA-0432

International Agreement Report

Analysis of the Test OECD-PKL2 G7.1 with the Thermal-Hydraulic System Code TRACE

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Washington, DC 20555-0001

Manuscript Completed: August 2013
Date Published: September 2013

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

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ABSTRACT

The test OECD-PKL2 G7.1 is a counterpart of the OECD/NEA ROSA-2 Test 3. These tests consisted of a small break loss-of-coolant-accident experiment concurrent with additional system failures, namely the total failure of the High Pressure Safety Injection system combined with the absence of an early manual secondary side cool down. To prevent a core meltdown a manual secondary depressurization (cooldown) is needed to restore the heat sink through fast reduction of the secondary pressure inducing similar primary pressure behavior, and thus leading to passive accumulator injection and low pressure safety injection. The core exit temperature parameter can be used to detect a core heat up and trigger such accident management procedures.

The test OECD-PKL2 G7.1 was performed to assess the reliability of the core exit temperature measurement and its correlation to the peak cladding temperature, and to investigate the physical processes affecting the performance of the core exit temperature measurement.

In this report, the analysis of test OECD-PKL2 G7.1 using the US-NRC thermal-hydraulics best estimate system code TRACE and based on an existing PKL Mark III model is presented and the obtained results are compared with the experimental data. A good agreement with the experimental data is observed, especially for relevant parameters such as the primary pressure. The predicted core exit and peak cladding temperatures evolutions could also capture the experimental data, although some discrepancies in the shapes of the evolution curves are observed.

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ABBREVIATIONS

Acc	Accumulator
AM	Accident Management
CET	Core Exit Temperature
CL	Cold Leg
DC	Downcomer
ECCS	Emergency Core Cooling System
FW	Feed Water
HL	Hot Leg
HPSI	High Pressure Safety Injection
JAEA	Japan Atomic Energy Agency
LP	Lower Plenum
LPSI	Low Pressure Safety Injection
MS-RCV	Main Steam Relief Control Valve
NC	Natural Circulation
PCT	Peak Cladding Temperature
PS	Primary System
PWR	Pressurized Water Reactor
PZR	Pressurizer
RC	Reflux Condensation
RCL	Reactor Coolant Line
RCP	Reactor Coolant Pump
RPV	Reactor Pressure Vessel
SAMG	Severe Accident Management Guidelines
SBLOCA	Small Break Loss Of Coolant Accident
SEPD	Separator
SG	Steam Generator
SOT	Start Of Test
SS	Steady State
TC	Thermocouple
UH	Upper Head
UP	Upper Plenum

ACKNOWLEDGEMENTS

This work was partly funded by the Swiss Federal Nuclear Safety Inspectorate ENSI (Eidgenössisches Nuklearsicherheitsinspektorat) and the Swiss Federal Office of Energy (Bundesamt für Energie), and the Spanish Nuclear Regulatory Body (Consejo de Seguridad Nuclear).

This report contains findings that were produced within the OECD-PKL2 Project. The authors are grateful to the participants and the Management Board of the OECD-PKL2 Project for their consent to this publication.

1 INTRODUCTION

The aim of this work is to evaluate the capability of the US-NRC thermal hydraulic best estimate system code TRACE [4] in properly simulating a SBLOCA in the hot leg of a PWR integral test facility, with special focus on the prediction of the Core Exit Temperature (CET). In various countries, the CET reading is used as a criterion for the initiation of the AM procedures involving emergency operation procedures and/or Severe Accident Management Guidelines (SAMG), since the CET can provide a timely indication of core heat up. The value of the CET set point to be used in the AM procedures may vary among different PWR types, because the performance of the CET depends on the installation position of the thermocouples, the radial and axial power distribution in the core, the scenario conditions (pressure, break size), and the geometry of the upper plenum.

The test OECD-PKL2 G7.1 (Test G7.1) was performed to assess the reliability of the CET measurement and its correlation to the Peak Cladding Temperature (PCT), and more generally to provide information on the physical phenomena responsible for the CET performance. The PKL integral test facility replicates the entire primary system and most of the secondary system of a 1300 MWe Pressurized Water Reactor (PWR) with elevations scaled to 1:1 and diameters reduced by a factor 12. It models the nuclear power plant on a scale of 1:145. The detailed design of the test facility was based to the largest possible extent on specific data of the Philippsburg nuclear power plant, unit2, a Siemens 4-loop design [1].

Test G7.1 [2] is a hot leg (HL) SBLOCA scenario concurrent with additional system failures, namely the total failure of the High Pressure Safety Injection (HPSI) system with no early secondary side cool down. This scenario necessitates Accident Management (AM) measures to prevent a core melt down. The secondary depressurization is intended to restore the heat sink through a fast reduction of the primary pressure, thus leading to passive Accumulators (Acc) injection followed by Low Pressure Safety Injection (LPSI).

The test OECD-PKL2 G7.1 is a counterpart of the OECD/NEA ROSA-2 Test 3, a HL SBLOCA transient conducted at the ROSA/LSTF facility. ROSA is a full-height and 1/48 volumetrically scaled test facility operated by the Japan Atomic Energy Agency (JAEA) for system integral experiments simulating the thermal-hydraulic responses at full pressure conditions of a 1100 MWe-class PWR during SBLOCA accidents and other transients [3]. The working primary-side pressure of the ROSA facility is the same as in a PWR. The ROSA-2 Test 3 is run in two sequences. During the first sequence, a first core uncovery takes place around 80 bars thus allowing for a comparison of the CET and PCT at high pressure. Then, after a transition phase where direct HPSI injection into the Reactor Pressure Vessel (RPV) is used to replenish the water level up to the Steam Generator (SG) inlet plenum, a secondary side cool down is conducted further down to below 45 bars which is the PKL test facility pressure range. A second core uncovery and core heat up sequence is initiated at this pressure and is followed by a secondary side depressurization and Cold Leg (CL) Accs injection. The second phase of the ROSA-2 Test 3 can be compared with the conditions of Test G7.1.

The initial conditions of Test G7.1 are reached following a conditioning phase up to the Start Of Test (SOT). Before the SOT, the inventory in the reactor coolant system is reduced down to bring about a mixture level in the SG inlet chambers thus allowing for stationary reflux condenser conditions. At the SOT, an upwardly oriented small break with a size equal to 1.5% of the CL cross section is opened on the HL. The heaters in the core simulator generate a volumetrically scaled residual heating power equivalent to 1.8% of the reactor total power, and the power level is held constant throughout the conditioning phase and the duration of the test. The four SGs are depressurized through the full opening of two of the four Main Steam Relief Control Valves (MS-RCV) as an AM measure as soon as the CET reaches the safety limit value set in the AM procedure. The maximum steam flow rate in the PKL facility is limited by nozzles of approximately 20 mm orifice diameters, installed in the main steam lines.

The CET parameter is used in many countries as one of the main signals to initiate emergency operating procedures and to switch to severe accident management procedures. The CET measurements are known to have some limitations in detecting core cooling and core uncovery. In fact, if the CET reading indicates superheating, it is in all cases with certain time delay and it is always lower than the actual maximum cladding temperature. It is proposed to evaluate in this study to what extent a best-estimate system code like TRACE is able to capture the time delay and the differences between rod surface temperatures and CET readings under core heat up conditions.

2 OBJECTIVES

The general objectives of the test G7.1 concern the investigation of relevant phenomena occurring under the AM conditions as well as the investigation of scaling effects. Thus, the important phenomena to simulate in this test are:

- core uncovery due to boil-off with generation of superheated steam
- primary-side pressure behavior before and after occurrence of core uncovery
- relation between the maximum cladding temperature and the CET during the core uncovery period
- SG depressurization based on the CET measurement as a core protection action
- interaction between primary pressure and Acc injection
- Accs injection after SG depressurization and influence on core cooling
- loop seal clearing occurring soon after the Acc's injection and effect on the RPV water levels

The calculations are performed using the TRACE code version V5.0rc3 and a PKL facility model that has been previously used in the simulation of various tests ([5], [6]). For the present study, first the test initial and boundary conditions have been introduced based on the boundary descriptions provided in ref. [2] and [7]. The quality of the results obtained is derived from comparison with the available experimental data.

3 TEST FACILITY DESCRIPTION

3.1 Overview

The PKL test facility replicates the entire primary system and most of the secondary system of a 1300 MWe 4-loop PWR plant with elevations scaled to 1:1 and diameters reduced by a factor 12. A layout of the facility where two loops are displayed can be seen in Figure 1. The detailed design of the test facility was based to the largest possible extent on specific data of the Philippsburg nuclear power plant, unit2. The scaling concept aims at simulating the overall thermal-hydraulic behavior of the full-scale power plant. One of the main conditions to meet these requirements is to respect a full scale hydrostatic head correspondence with the reference PWR, in order to properly capture gravitational and natural circulation (NC) conditions.

The power, volume and cross-sectional area scaling factor is 1:145. A full-scale correspondence of the frictional pressure loss distribution in the system is reproduced. The four primary loops are simulated independently to enable all possible primary loop coolant asymmetries during transient conditions. Full height fuel rods, spacers and internal structures are used in the RPV simulator, and the rods in the core are electrically heated. Also a full height U-tube bundle with a scaled number of tubes is used to simulate the primary side of each SG.

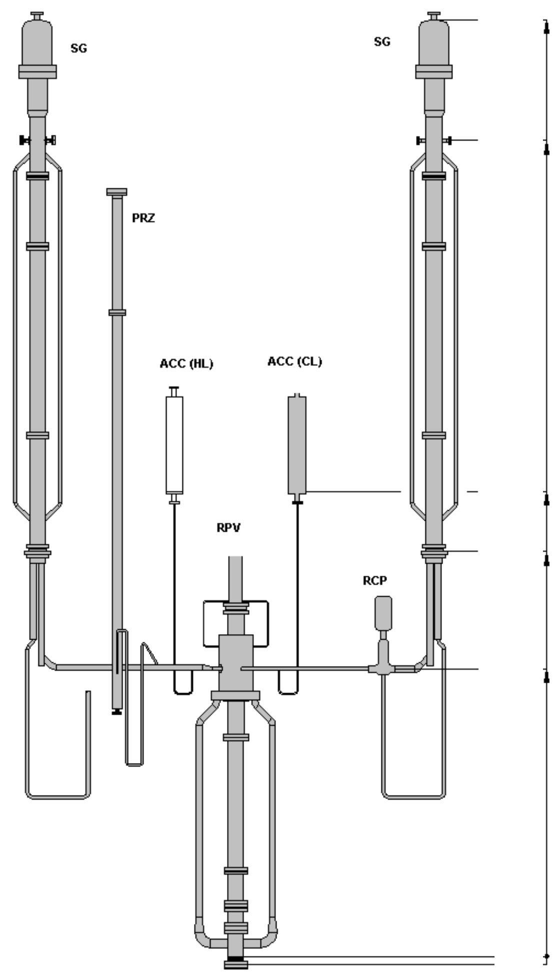

Figure 1 PKL integral test facility layout [1]

The next subsections provide a brief description of the main components of the PKL test facility. Further information can be found in ref. [1].

3.2 Reactor Coolant System (RCS)

The RCS is mainly composed of the RPV containing a heater bundle simulating the PWR core; the downcomer (DC) model; the four loops equipped with a Reactor Circulation Pump (RCP) a SG and the connecting pipes, mainly the HL, the CL and the pump seal (PS). The pressurizer is connected to the HL of the first loop by the surge line.

3.3 Rod Bundle Vessel

The vessel models the Upper Head (UH) plenum, the Upper Plenum (UP) both above and below the Reactor Coolant Line (RCL), the reactor core, the reflector gap and the Lower Plenum (LP). The maximum allowable working pressure and temperature of this component are 50 bars and 573 K, respectively.

For the assessment of the CET performance, the vessel is equipped with multiple thermocouples (TC) positioned at the core exit, on different radial locations and installation configurations, as a replication of a typical PWR instrumentation (TC with and without shield tube). See Figure 2 for more details.

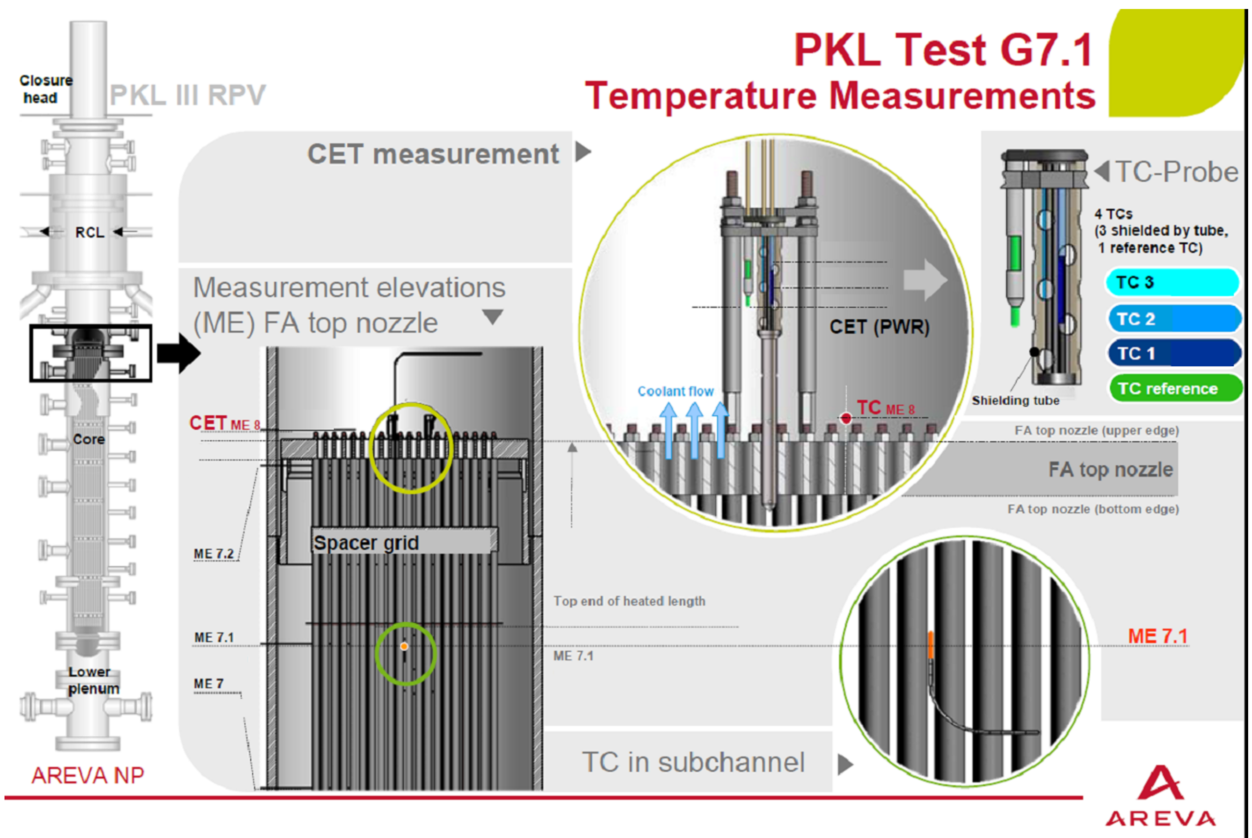

Figure 2 Description of the CET measurements [7]

3.4 Upper Head and Upper Plenum

The upper head and upper plenum are modeled at full scale in height and 1:145 in volume with respect to the reference PWR. The internals in the UP are simulated by means of seal welded tubes and a RPV level detector is installed in the center. The top of the sensor is located at the mid-loop level of the RCL.

3.5 Upper Head Bypass

The upper head bypass is modeled with four parallel bypass lines associated with their respective primary loop to enable asymmetric flow distribution in the RCS; for example a single loop operation. The bypass pressure loss is controlled via an orifice in each by-pass line, in order to reproduce a core bypass flow equal to 0.5% of the total primary mass flow rate.

3.6 Test Bundle

The PKL simulated core consists of 314 electrically heated fuel rods. The core also contains 26 control rod guide thimbles. These are held within an assembly by a top end fitting and the spacer grids. The fuel rods are arranged into three concentric zones powered independently. This enables the simulation of a radial power profile. However, for Test 7.1 all zones are equally powered.

Sixteen rods are equipped with chromel-alumel-sheathed thermocouples. The six sheathed thermocouples per rod are brazed into slots distributed over the heater rod length. The thermocouples have an outside diameter of about half a millimeter and the measuring junctions are insulated from the outer sheath.

4 TRACE MODEL

The TRACE nodalization of the PKL test facility used in this study has been converted and further developed from a well tested RELAP5 model developed at the Technical University of Catalonia [8]. The model used in this work is consistent with the latest configuration Mark III (three) of the facility, which was utilized to conduct Test G7.1. A diagram of the nodalization is shown in Figure 3. The discretization used in the nodalization consists of 438 hydraulic volumes and 1356 heat structures mesh points. All the walls and surfaces are modeled by heat structures to take into account any heat loss, or heat exchange between the structures and the coolant. The core fuel rods and other heaters (upper head and pressurizer) are modeled by means of heat structures, as well. Most auxiliary, safety and control systems are modeled by means of FILL and BREAK components.

The reactor pressure vessel is represented by PIPE components. The core region is nodalized by a single PIPE with 7 axial cells. The core bypass is modeled by a single PIPE. The DC is modeled with two parallel PIPEs in order to allow capturing recirculation flow in the DC (this pattern has been observed in some experiments). On the other hand, the downcomer of the SGs are simulated with a single PIPE as the required level of detail in the secondary system is not as high as in the primary side. The four bypass pipes from the DC to the UH are described with two different PIPEs. In each PIPE, the flow area is equivalent to the cross section of two PKL facility by-passes. The PIPE components reproduce the bypass orifice using a hydraulic diameter restriction.

The four PKL primary loops are modeled with PIPE components. Only one PIPE component is used to model the U-tube bundle. The hydraulic diameter of this PIPE component is equal to the diameter of a single U-tube while a multiplication factor is used to set the right total heat exchange area of the U-tubes. The Emergency Core Cooling System (ECCS) of each loop is connected to the cold leg. Each ECCS is composed of an Acc, a HPSI and a LPSI. The Acc is represented by a single PIPE component and a valve component. The HPSI and LPSI are represented by FILL components and the injection characteristics are imposed as a mass flow as function of the primary system pressure. For Test G7.1, only the four Accs and the four LPSI systems are available.

The PKL integral test rig pressurizer (PZR) is represented in the TRACE model by a single PIPE component. VALVE and BREAK components are used to represent the PZR relief valve and relief line boundary conditions. A FILL component is connected at the bottom of the PZR to control the water level. The PRZ heaters are linked to the bottom cell of the PZR and are used to control the system pressure during the steady state (SS). The PZR is connected to Loop 1 by the surge line.

The SG secondary side includes a separator TRACE component (SEPD), a downcomer and riser PIPE component and a steam line PIPE component. The four steam lines are connected to a common steam header. A constant emergency Feed Water (FW) flow rate is imposed after the CET limit trip is activated and this flow is kept constant until the end of the transient.

The SG water level during the conditioning phase is controlled using four TRACE FILL components and their associated control system. The FILL components are located in secondary downcomer of the SGs.

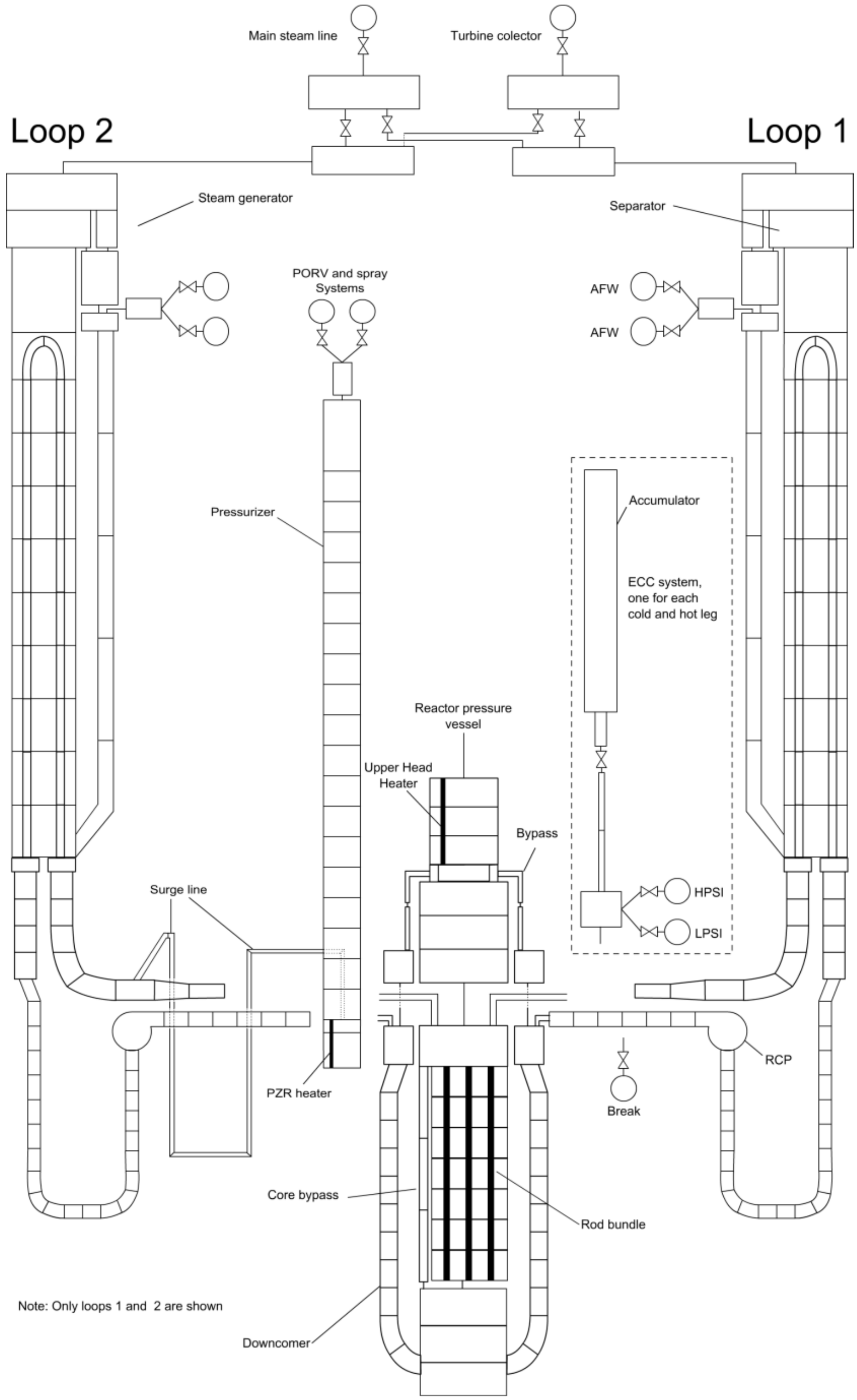

Figure 3 TRACE nodalization of the PKL test facility (only two loops are shown)

5 BOUNDARY CONDITIONS OF TEST G7.1

5.1 Test Procedure

The conditioning phase aims at adjusting the required initial conditions prior to the SOT.

The first phase consists of establishing a primary side full of coolant at a pressure slightly above 40 bars and operating a stable SS under NC conditions. The scaled core power is set to 1.8% and the RPV heat losses are compensated. The secondary side initial working conditions are around 25 bars.

The second phase consists of bringing the PKL facility to the counterpart OECD/NEA ROSA-2 Test 3 conditions. First, the steam dump valve is closed to enable a pressurization of the secondary side up to more than 40 bars and to decrease the subcooling in the primary side. Meanwhile, the break is opened to achieve the desired collapsed levels in the primary side of the SG U-tubes. Then, the break is closed to enable SS conditions prior to the SOT. The required secondary side pressure is obtained by controlling the operation of the MS-RCVs.

The third phase corresponds to the case study transient of a HL SBLOCA concurrent with additional failures: total failure of the HPSI combined to no secondary side manual cool down. At the end of the conditioning phase the break is opened at the SOT and is kept open until the end of the transient phase.
An overview of the test procedure including the conditioning phase is shown in Figure 4.

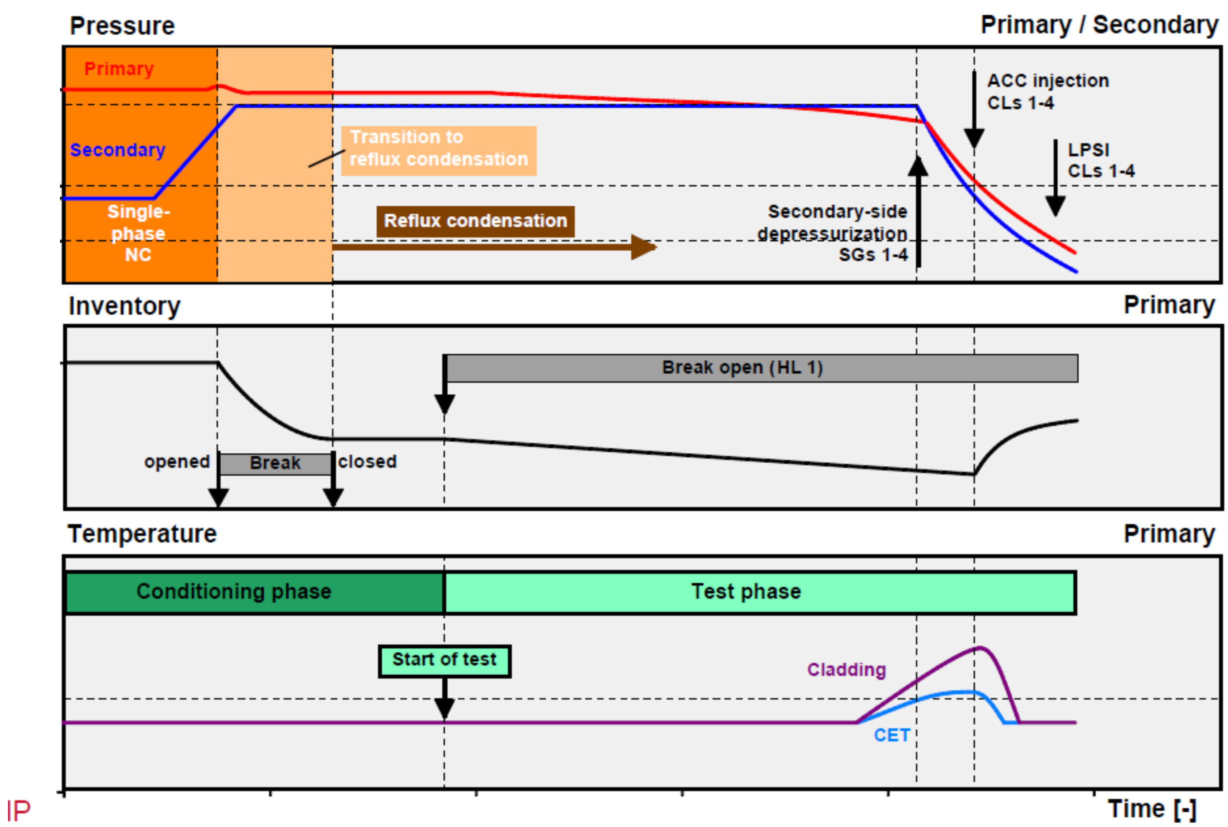

Figure 4 Test procedure including the conditioning phase [7]

5.2 Core Power

The power of the core is 1.8% of the scaled PWR power. In addition, power is generated in the system to compensate for the PKL facility heat losses [9]. The core simulator contains 314 electrically heated elements subdivided into three core zones of 63, 118 and 133 fuel elements respectively with a flat axial core power profile. These zones are modeled independently by three different heat structures in the TRACE model. However, they are all connected to the same hydraulic channel.

5.3 Secondary Pressurization

The first action in bringing the PKL facility to the counterpart OECD/NEA ROSA-2 Test 3 conditions is the secondary pressurization up to more than 40 bars by closing the steam dump valve (VALVE component in the TRACE model). This action is done after reaching the initial conditions of a full primary side under SS NC conditions. The secondary pressure is then controlled with MS-RCVs. The new SS obtained allows SG reflux condenser (RC) cooling conditions.

5.4 Break Conditions

The small break is located on the HL of Loop 1 and is oriented upwards. The size of the break is 1.5% of the CL cross section area. The break is first used to bring the facility to the specified initial inventory prior to the SOT. The break is opened for a fixed time duration to bring the collapsed level in the SG U-tubes down to the entrance chambers. The HL SBLOCA transient itself is started with the break opening after stationary RC conditions have been reached.

5.5 Steam Generator Depressurization

At the end of the conditioning phase, the water level in the SG secondary side is reduced down to a value similar to the one obtained in the OECD/NEA ROSA-2 Test 3 (see the SG water levels evolution in the bottom part of Figure 5).

The fast cool down AM measure is executed using two of the four MS-RCVs. The maximum MS-RCVs steam flow rates are controlled by elliptically shaped nozzles to limit the flow according to the specifications of the reference PWR. After the SOT, the secondary depressurization is activated by the CET signal when it reaches the safety limit value in the AM procedure.

The four SGs of the facility are connected via the main steam header. The steam lines are controlled by 4 VALVE components. Since only two of the four MS-RCVs are operated during the test, the four valves are kept open in order to balance the pressures in the four SGs during the initial phase, the conditioning phase and the transient.

During the transient phase, only two of the four available steam generator relief valves are available. For this reason, two of the MS-RCVs of the TRACE model are associated to the CET trip signal to enable the secondary depressurization as an AM measure, whereas the two other MS-RCVs are reserved for secondary pressurization pressure control during the conditioning phase. This choice is dictated by the different maximum valve opening rates needed for the control of the secondary pressure. When controlling the secondary pressure, the valves open only to one third of their total flow area with a relatively slow opening rate compared to the safety relief function after the CET limit has been reached (then the valves can open much faster). This modeling configuration enabled the MS-RCVs results to better reproduce the experiment procedure.

5.6 Emergency Core Cooling Systems, ECCS

In Test G7.1, all the HPSI systems are assumed to fail. Both the Accs and the LPSI systems inject water in the cold legs of the four PKL primary loops. The initial pressure in the Accs is above 25 bars. The LPSI can inject water in the system only after the primary pressure has fallen well below 10 bars.

6 ANALYSIS RESULTS

6.1 Overview

As previously described, Test G7.1 presents three distinct phases: the steady state, the conditioning phase and the transient test phase. In Figure 5 the RPV, PZR and SG secondary collapsed water levels are shown for the three phases. The evolution of the pressure in the primary and secondary circuits is shown in Figure 6 along with indications of the main events occurring during the test. The actuation of the break valve is shown in Figure 7 and the calculated mass inventory can be seen in Figure 8. The steam and total core mass flows are shown in Figure 9 and the mass flow in the loops is presented in Figure 10. These figures provide an overview of the actions performed during all the phases. In Table 1, the chronology of the main events is shown.

For convenience, the obtained results are presented in five sub-sections according to the most relevant situations during the whole test.

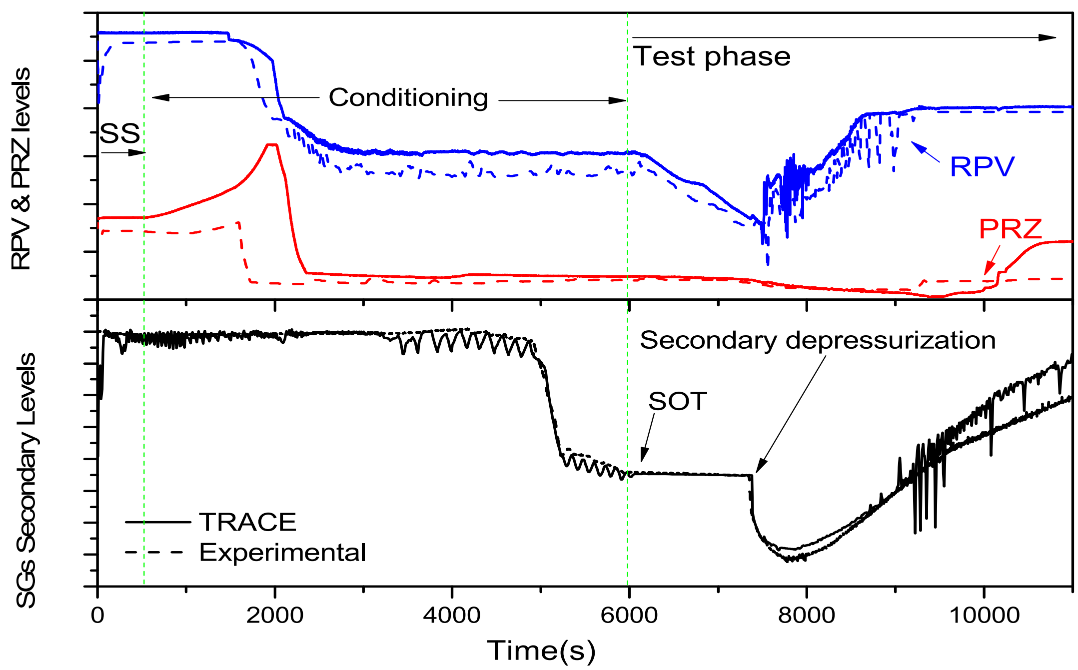

Figure 5 PKL G7.1 evolution of the RPV (blue), PZR (red) and SG (black) levels

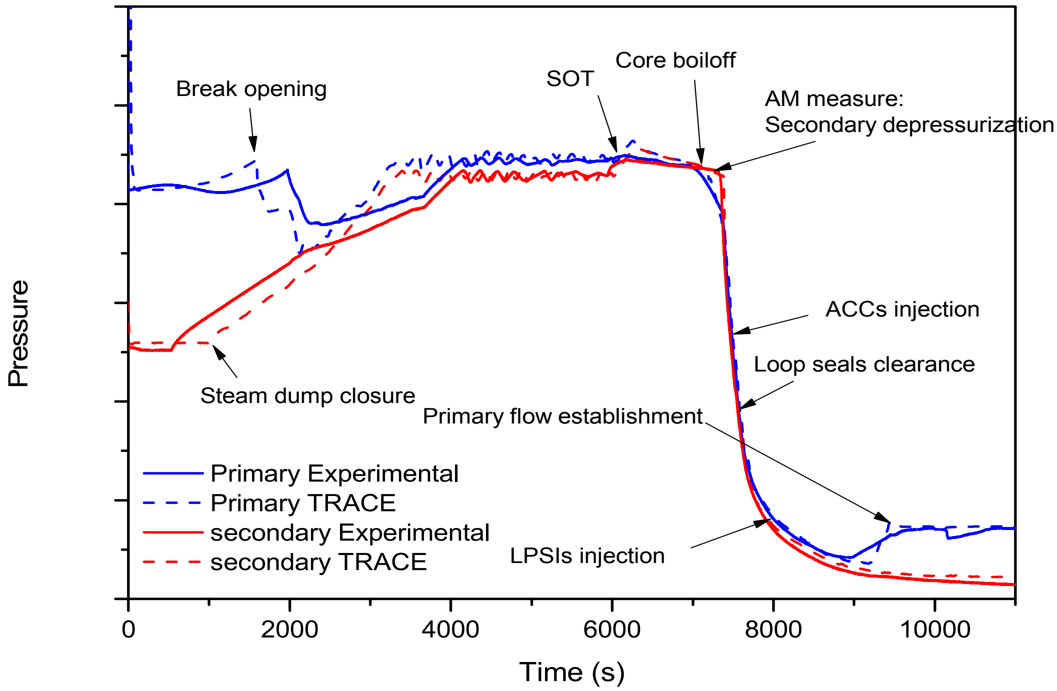

Figure 6 **PKL G7.1 Test pressures evolution**

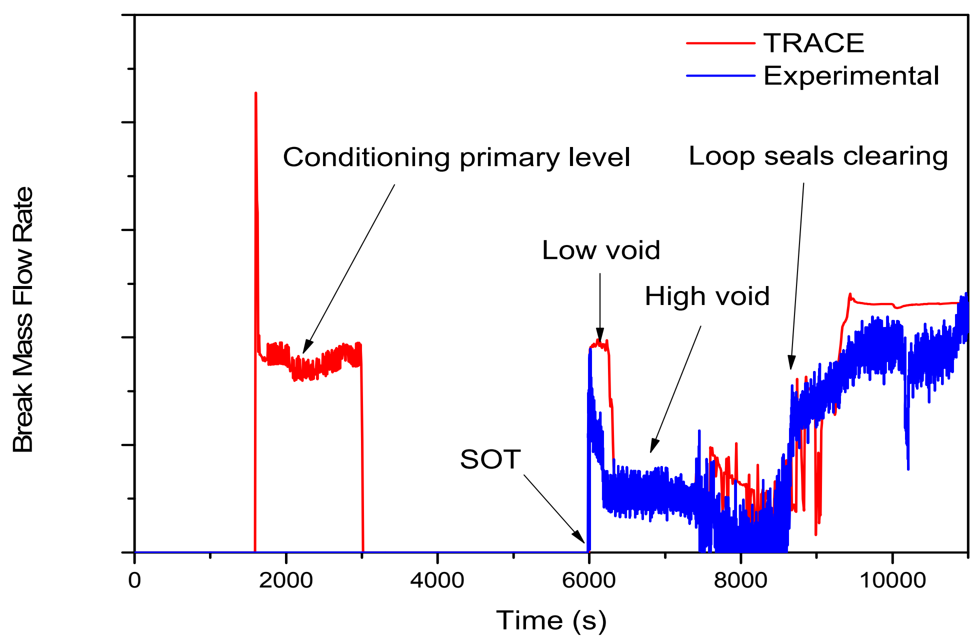

Figure 7 **PKL G7.1 small break flow rate evolution**

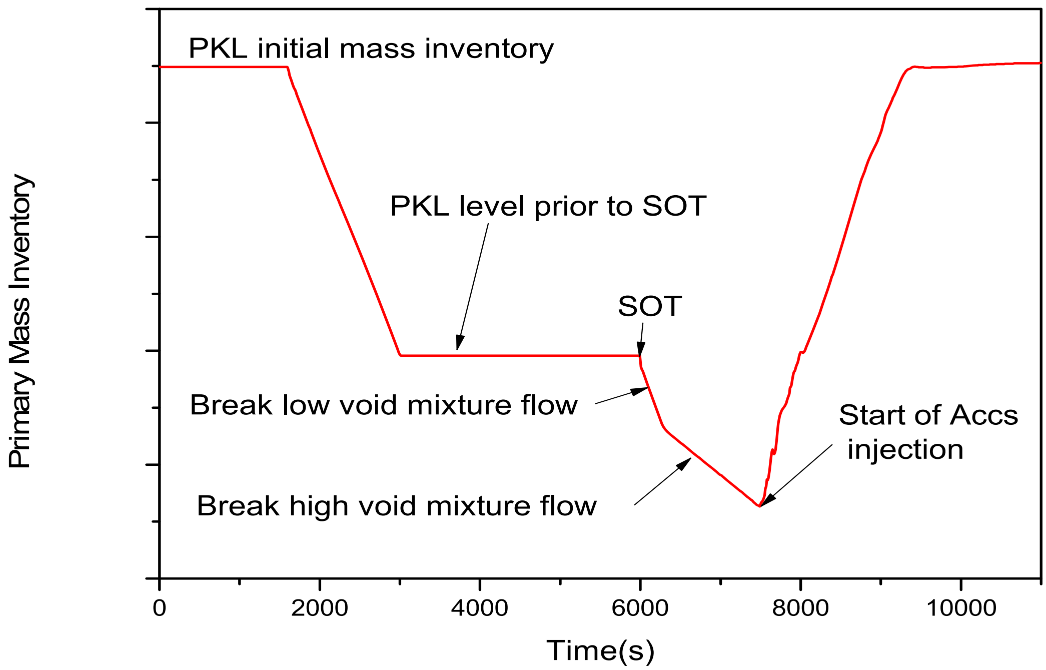

Figure 8 **PKL G7.1 mass inventory evolution in the TRACE calculation**

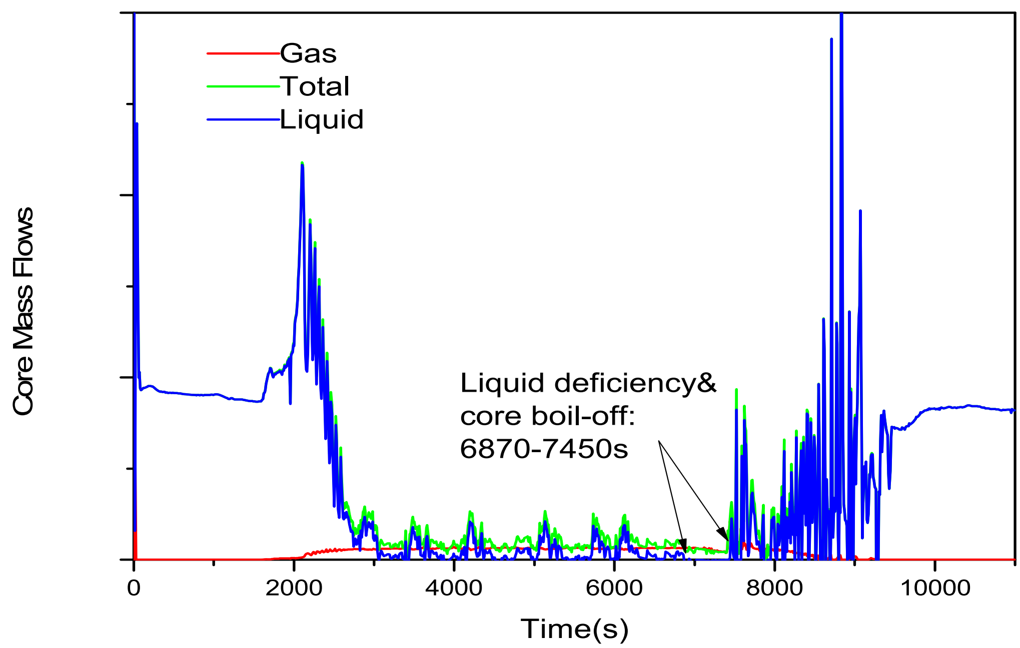

Figure 9 **PKL G7.1 core mass flows evolution as calculated by TRACE**

Table 1 PKL G7.1 test - sequence of events (Deviation from experiment is indicated for relevant events)

Accident event	Calculation (s)	Experiment (s)
SS ends (Steam dump closure)	991	528
Mass inventory reduction starts	1591	-
Mass inventory reduction ends	3012	-
SG level reduction starts	4915	4915
SG level reduction ends	5231	5231
SOT (break opens)	6000	6000
Low void break flow (end off-take)	6320	6177
Start of core uncovery	7050	6990
Core boil off (p-prim < p-sec)	7150 (-0.1 bar)	7010
CET limit is reached	7380	7302
Secondary depressurization	7385 (-1.2 bar)	7317
PCT peak maximum value	7410 (-83 K)	7389
CET peak maximum value	7420 (-43 K)	7457
Core vg increase/early mitigation	7400 (0.7 to 3.0 m/s)	-
Accs injections start	7490	7483
CET peak end	7450	7520
Minimum core level	7560 (-1.3 m)	7503
End of void break flow	7570	-
Accs injection, end	8020	7842
LPSI	8030	7866
Loop seal clearing, end	9221 (-0.9 bar)	9022
Primary flow establishment	9431 (+0.8 bar)	9635

6.2 Situation at the Beginning of the Conditioning Phase

The conditioning phase starts after the TRACE model has reproduced the PKL facility initial conditions the experiment. The four loops are totally filled with primary coolant and the core residual power is evacuated to the secondary system through NC. The electrically heated rod bundle in the RPV is supplied with a constant power. Compensation to the PRZ and UH heat losses is also locally supplied.

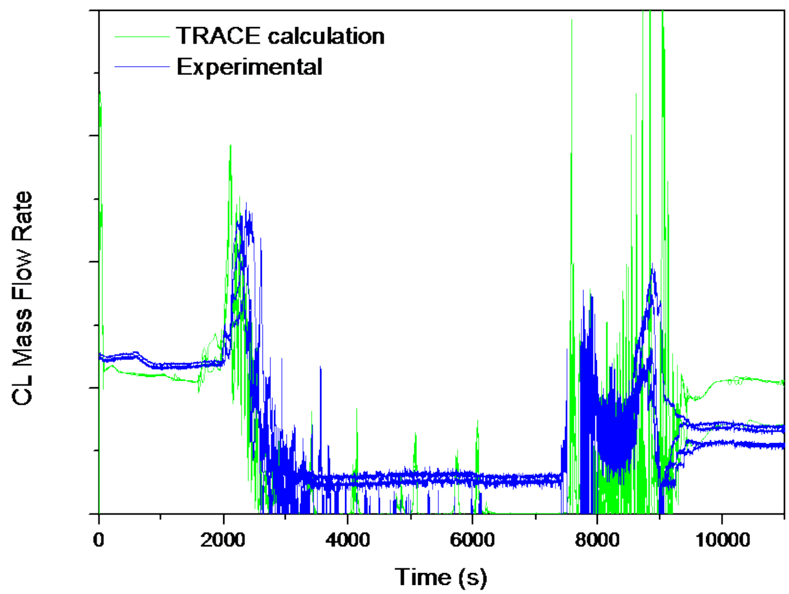

Figure 10 PKL G7.1 loops flow rate evolution

The main steam dump is open and the water levels in the PRZ and SG are stable. The loops primary flow rates are in the order of 1 kg/s. The pump impeller rotational speed is null. Table 2 summarizes the main obtained results. These are in close agreement with the experiment except for a slightly higher CET.

Table 2 Results at the beginning of the conditioning phase

Primary side	**Deviation from experiment**
Core power	0 kW
CET	+3.2 K
Subcooling at core outlet	-2 K
Pressurizer level	-0.3 m
Primary pressure	-0.2 bar
Loop flow rate	0 kg/s
Secondary side	
Main steam pressure	+1 bar
Secondary temperature	-0.5 K
SG water level	0 m
FW temperature	+4 K

6.3 Initial Test Conditions

The initial test conditions are reached after the pressurization of the secondary side is completed (Figure 6). The collapsed water level in the primary system is reduced by temporarily opening the break (Figure 7) down to the intended primary coolant mass inventory at the SOT (Figure 8). The water level in the secondary side of the SGs is also brought down (Figure 5) to match initial conditions consistent with the ones of the OECD/NEA ROSA-2 Test 3. The obtained SS conditions prior to the SOT (Table 3) correspond to stable RC conditions as displayed in Figure 11. As shown in Table 3, the SOT conditions were close to the experiment.

Table 3 Results at the start of the test

Primary side	Deviation from experiment
Power	0 kW
Pressure	-0.7 bar
CET	-1.3 K
Loop seal flow rate	0 kg/s
Secondary side	
SG pressure	-0.9 bar
SG water level	0 m
Temperature	-2 K
Feed water temperature	0 K

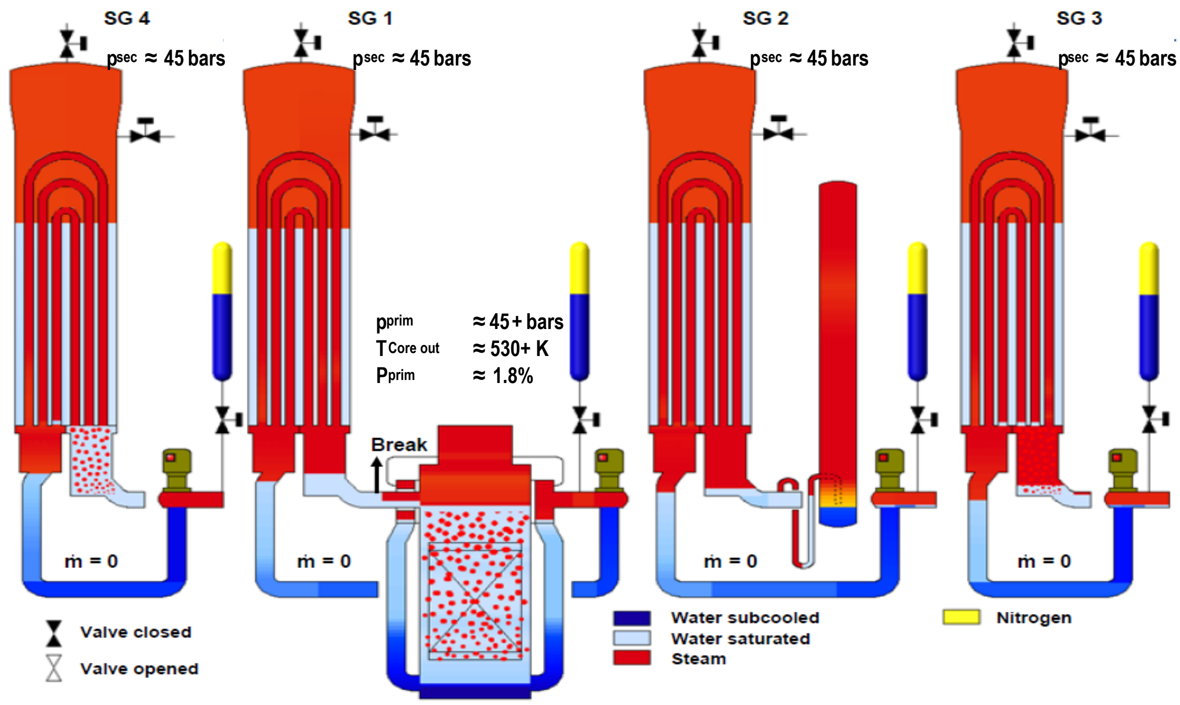

Figure 11 Initial test conditions for Test G7.1

6.4 Situation Prior to Secondary-Side Depressurization

The test is started by opening the break valve. The entire duration of the conditioning phase was 4000 s.

One of the specifics of this test is the particularly small size of the break. This leads to a relatively higher steam generation rate in the reactor core compared to the steam flow rate pulled through the break prior to core uncovery (Figure 7 and Figure 8). This keeps the primary pressure to a value close to the secondary pressure thus establishing a heat sink to the secondary system through RC. The residual core power is partly evacuated in the form of steam flow at the break and the other part is transferred to the SGs. This enables the primary pressure to be governed by the secondary system until the core starts boiling-off (Figure 6).

At the start of the test transient phase, the liquid coolant phase dominates the break leak flow rate. Phase separation in the HL connected to the upward oriented break (controlled by the off-take model in TRACE) essentially sets the duration of this phase. Figure 12 shows the total and steam flows at the break. More power of the generated steam rate is transferred to the heat sink as less steam is pulled through the break.

The next sequence corresponds to the break leak flow rate dominated by vapor phase as the liquid level collapses due to the continuous decrease of the mass inventory (no HPSI is available). The choked flow conditions at the break govern the duration of this sequence. The steam generated in the core is removed through the break steam leak flow rate and the heat sink RC conditions. The secondary pressure evolution gets more stable and the pressure evolution is fairly constant up to the secondary depressurization. The primary pressure stays above the secondary pressure as long as the steam mass flow rate at the break is not sufficient to overcome the steam generation rate in the core.

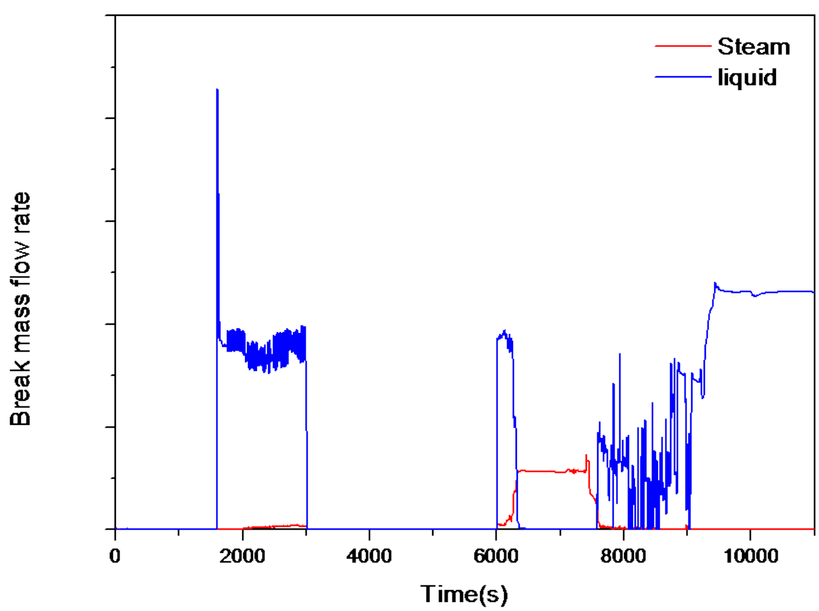

Figure 12 PKL G7.1 break gas and liquid mass flows evolution

As a result of the continuous reduction of the primary mass inventory, the core uncovery eventually starts as a result of the decrease of the water level in the core. The dried upper core structures heats up and the corresponding power is no longer used for steam generation, instead a growing part of core power is used to superheat the generated steam. The corresponding loss of steam generation is not compensated by the volumetric expansion of the superheated steam and this affects the pressure equilibrium, thus causing the primary pressure to drop below the secondary pressure following (Figure 6). Further core uncovery leads to more structure heat up and brings about an excursion of the PCT high enough to produce more superheated steam and causing in the end the CET to reach the safety limit (Figure 13). As an immediate consequence of the AM procedure, the two available MS-RCVs are fully opened and initiate the secondary side depressurization (Figure 5). The total steam mass flow rate in the operated MS-RCVs is shown in Figure 14, along with the previous flow rates during the controlled conditioning phase.

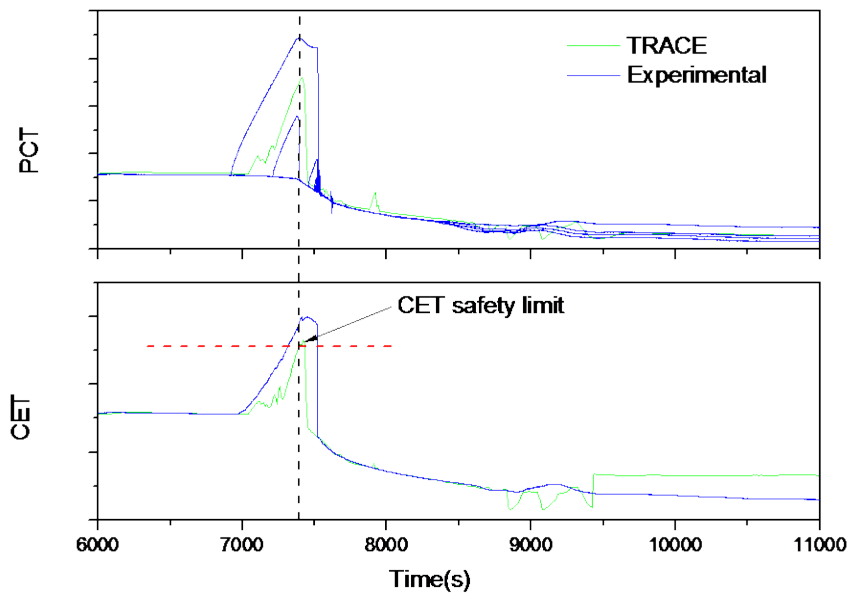

Figure 13 PKL G7.1 PCT & CET evolution

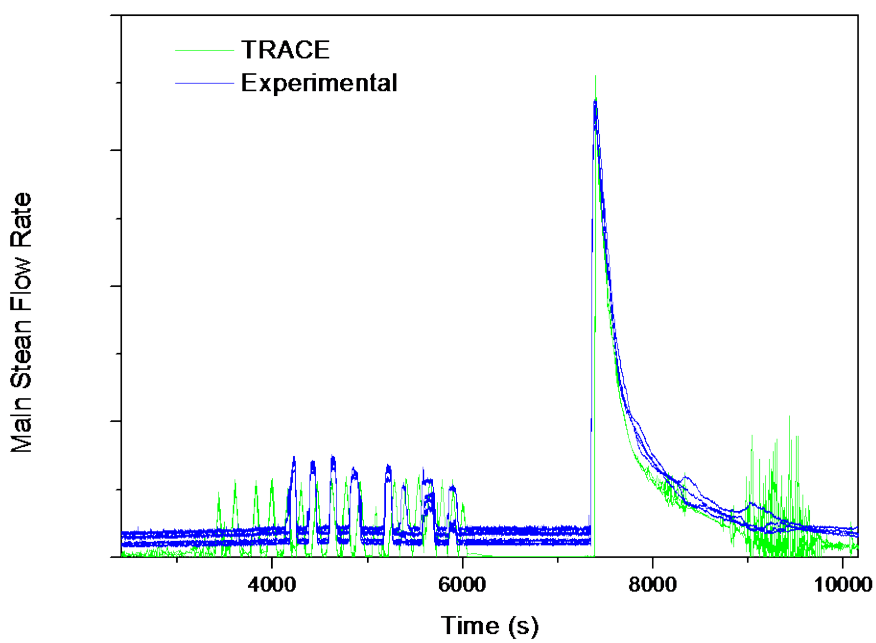

Figure 14 PKL G7.1 MS-RCVs steam flow rate evolution

As can be seen in Figure 6, experimental data show a nearly horizontal time evolution of the secondary pressure compared to the slightly more inclined pressure predicted by from 6000 s to 7000 s. This might be an effect of the secondary heat losses compensation system in the PKL facility, which injects water through a special bypass and is suppressed at the start of the secondary depressurization event. The TRACE model does not introduce this power compensation due to lack of detailed information on this system. Actually, an effort in minimizing the simulation secondary heat losses helped shaping the primary pressure curve closer to the experimental one after the SOT, but also in the period from 7000 s to 7500 s, up to the initiation of the secondary depressurization.

After the secondary depressurization has been initiated, the Accs injection (Figure 15 for the pressure and Figure 16 for the mass flow rate) is affected by the slight discrepancy in the primary pressure evolution. The injection phase is longer in the calculation and the final pressure in the Accs is lower compared to the experimental data.

Table 4 presents the main TRACE calculation results compared to the experimental data.

Table 4 Results prior to the secondary side depressurization phase

Primary side	**Deviation from experiment**
Power	0 kW
Pressure	-0.4 bar
CET peak	+14.6 K
Loop seal flow rate	0 kg/s
Secondary side	
Pressure	0 bar
SG water level	0 m

6.5 Acc Injection

The obtained Accs pressure at the end of injection is lower than in the experiment. During the injection phase, the behavior of the Accs is affected by the steam condensation rate in the cold leg, and by the expansion of the nitrogen dome (the Accs are initially pressurized with nitrogen). The Accs reach the end point of operation when the liquid level has been depleted down to a specified value. Then the Accs valves close and the pressure remains equal to the corresponding primary pressure at the end of injection. In Figure 16, the injected mass flow is more irregular than in the experiment. This may hint at the need to improve the condensation heat and mass transfer models used in TRACE under Accs injection conditions. Moreover, the minimization of the heat losses mentioned in the previous sub-section partly contributed to the longer Acc injection time predicted by the TRACE model.

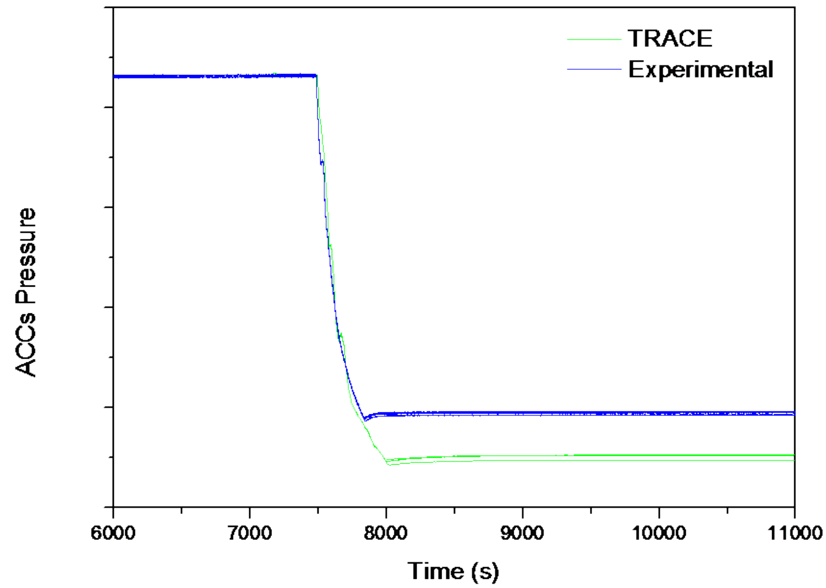

Figure 15 PKL G7.1 Accs pressure evolution

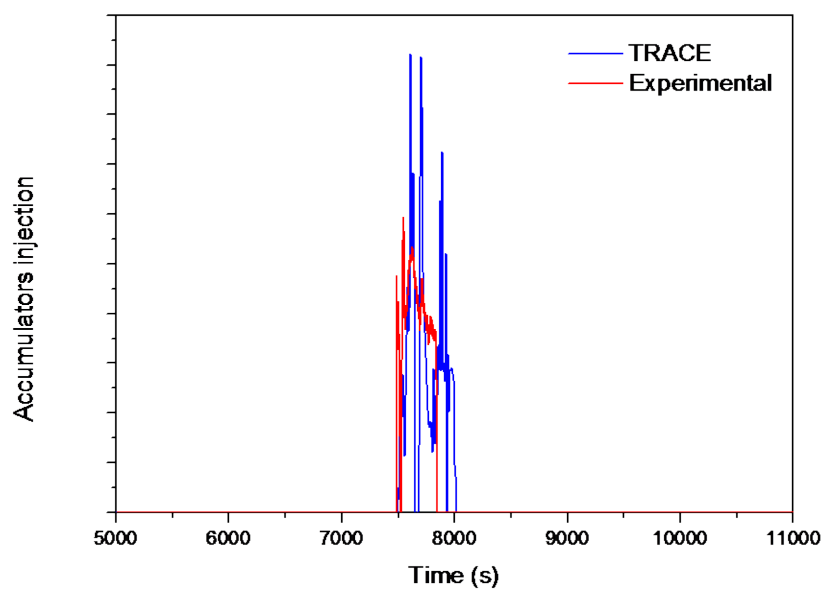

Figure 16 PKL G7.1 Accs injection evolution

The simulated loop seal clearing sequence takes place over one minute after the Accs injection have started and is followed by a sudden core level increase through an agitated phase. However, the TRACE simulation anticipates the mitigation of the CET peak over one minute before the start of the Accs injection (see Figure 13 where the dotted vertical line indicates the time of CET inversion in TRACE, which is sharp and happens more than a minute before Accs injection). This discrepancy can be explained by a different steam generation rate in the lower part of the core, resulting from the fast depressurization in the primary system. Steam production reduces the average coolant density in the lower part of the core while raising the mixture level higher up, which provides an earlier mitigation of the heat up in the hottest section of the core. Figures 17 and 18 show for each core elevation the void fraction and the density, respectively.

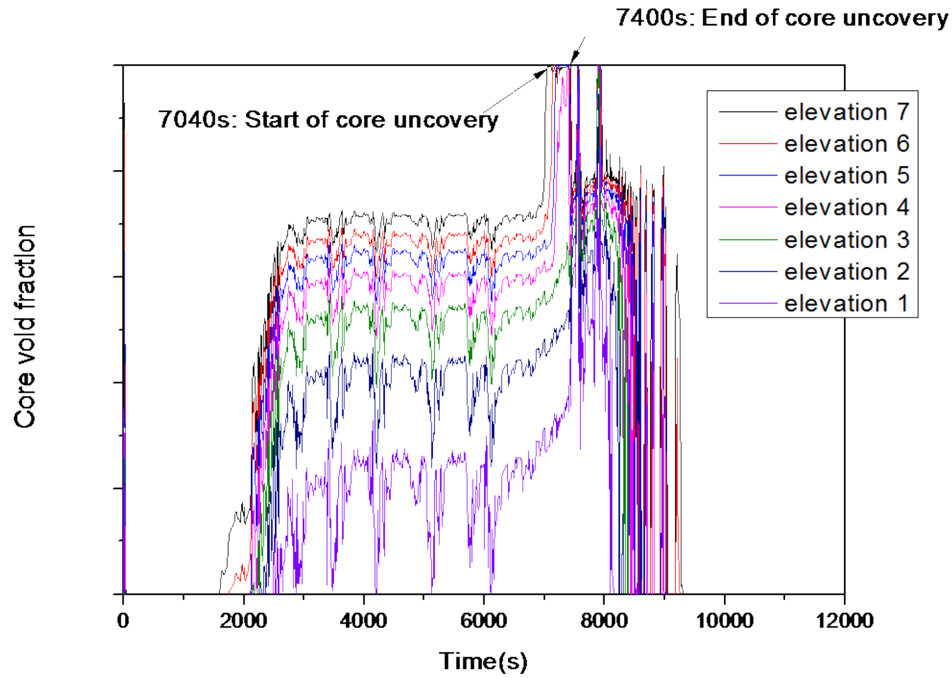

Figure 17 PKL G7.1 core void fraction evolution (calculated by TRACE)

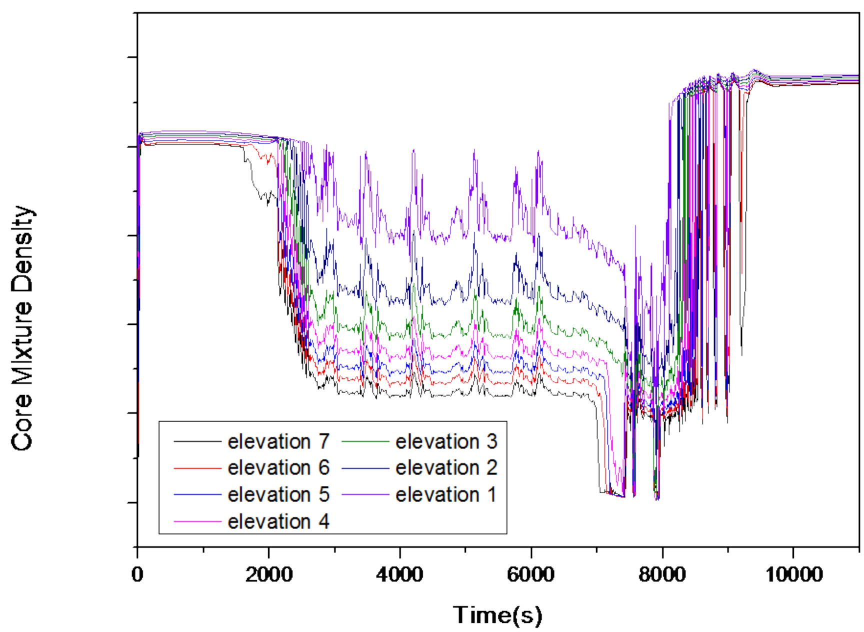

Figure 18 PKL G7.1 core density evolution (calculated by TRACE)

The early mitigation phenomenon can also be explained by looking at the calculation results in Figure 19 where the core gas and liquid velocities are shown. A sudden increase in the core exit vapor flow can be clearly observed after 7400 s, when the core exit vapor velocity increases by a factor 4. These higher steam velocities can also explain for a higher entrainment of water up into the core. This is not observed in the experimental data, where the mitigation of the CET and PCT excursions is not as immediate as in the TRACE simulation. This different behavior results in a relatively larger and flatter maximum experimental peaks compared to the sharp edged calculated peaks (Figure 13). Yet, these more rounded shapes of the experimental temperature peaks could also be seen as an early mitigation effect prior to Accs injection and only due to the secondary depressurization effect, although not to the same extent as in the calculation results.

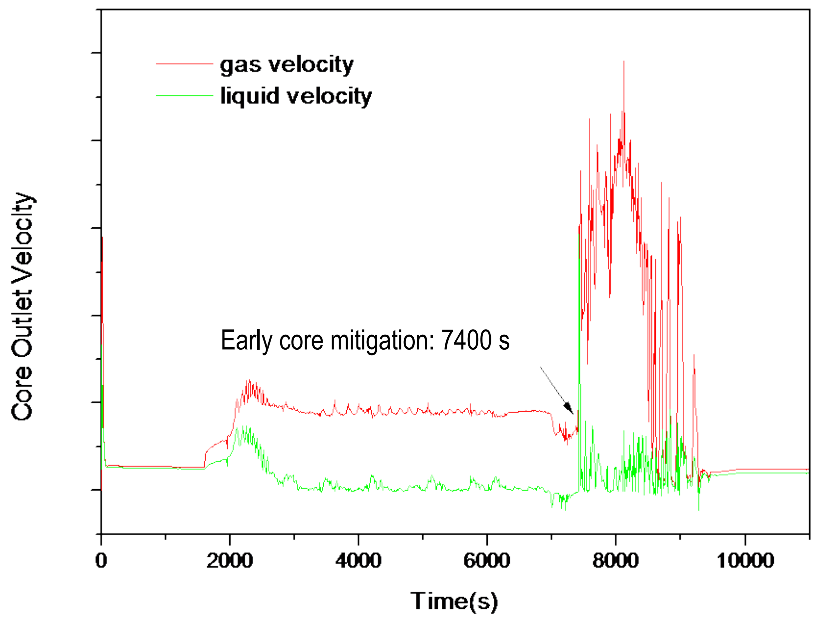

Figure 19 PKL G7.1 core gas and liquid velocities evolution (calculated by TRACE)

6.6 Situation After Acc Injection

The core has been successfully reflooded with the injection of the Accs and the LS clearance, while the secondary depressurization has been completed. As the MS-SRVs remain open, the primary system keeps depressurizing following the secondary system and the LPSI set-point is reached. The simulated LPSI injection is compared to the experimental data in Figure 20. The LPSI injection successfully replenishes the primary system and NC conditions are met again as shown in Figure 10.

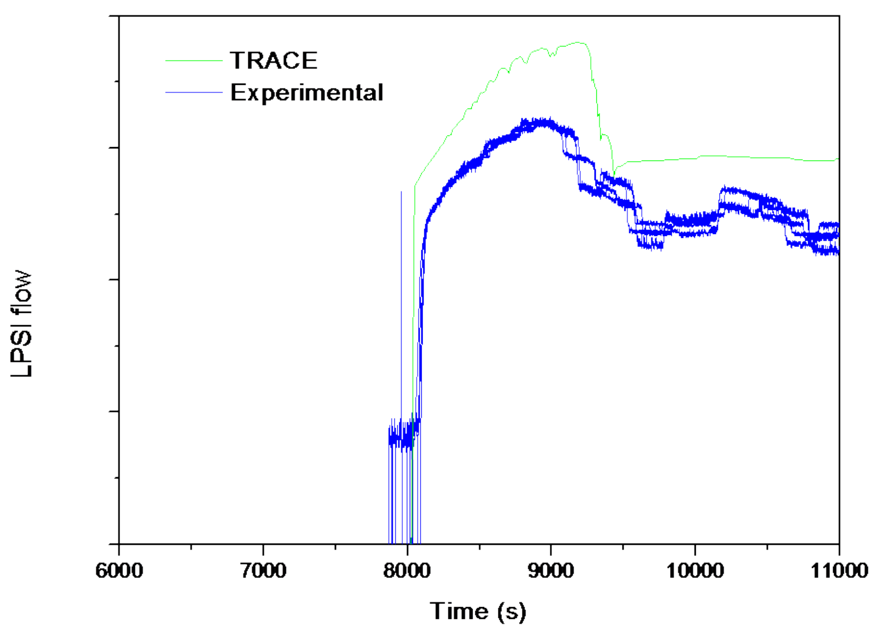

Figure 20 PKL G7.1 MS-LPSI injections evolution

7 CONCLUSIONS

A post-test analysis using TRACE of the test OECD-PKL2 G7.1 has been conducted. The test was based on a 1.5% SBLOCA scenario without HPSI and without operator action to initiate the secondary depressurization. The break was located on the top of the HL and vertically oriented.

The PKL TRACE model, using the version V5.0rc3 of the code, could reproduce well the main physical phenomena involved in Test G7.1, and in good agreement with the experimental data. The calculation showed however some discrepancies with the experimental data, such as in the shapes of the CET and PCT time evolutions, the Accs injection and the phase separation upstream of the break (off-take model), the overall pressure evolution and the progression of the different phenomena could be captured with a good level of accuracy. Another important conclusion is that the PKL TRACE model could replicate both qualitatively and quantitatively the effectiveness of the secondary side depressurization as AM measure to mitigate the core heat-up after boil-off.

Finally, in respect to the performance of the TRACE models in simulating the important phenomena exhibited by Test G7.1, one can draw the following conclusions:

- The break model could accurately capture the early maximum liquid mass flow rate at the time of the SOT, while a slight over-prediction of the break mass flow was observed at low void. Very good agreement was obtained at high void.

- The primary-side and secondary-side pressure behaviors could be particularly well reproduced. To allow for such results, slight adaptations to the heat losses of the system had to be made (within the bounds of the specifications from the test facility).

- The time evolution of the water level in the core after the SOT could be captured with good accuracy, and throughout the defining sequences of the transient, such as the depressurization phase, the Accs injection and the loop seal clearance (e.g. the time of core uncovery could be captured within a minute accuracy).

- During the core uncovery period, the maximum values of the CET and PCT calculated by the TRACE model were relatively lower compared to the corresponding experimental values, as a result of an early mitigation of the core heat up, prior to the Accs injection. This early mitigation effect as a direct consequence of secondary depressurization will need to be further evaluated and is possible confirmed by more related experimental data.

- During the Accs injection phase, the injected mass flow proved less regular than in the experiment. This may hint at the need to improve the condensation heat and mass transfer models used in TRACE under Accs injection conditions, and more sensitivity studies should be conducted to better characterize the issue.

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NRC FORM 335 (9-2004) 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-0432
2. TITLE AND SUBTITLE Analysis of the Test OECD-PKL2 G7.1 with the Thermal-hydraulic System Code TRACE	3. DATE REPORT PUBLISHED MONTH: September YEAR: 2013 4. FIN OR GRANT NUMBER
5. AUTHOR(S) Salah Belaïd Jordi Freixa Omar Zerkak	6. TYPE OF REPORT Technical 7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

K. Tien, Project Manager

11. ABSTRACT *(200 words or less)*

The test OECD-PKL2 G7.1 is a counterpart of the OECD/NEA ROSA-2 Test 3. These tests consisted of a small break loss-of-coolant-accident experiment concurrent with additional system failures, namely the total failure of the High Pressure Safety Injection system combined with the absence of an early manual secondary side cool down. To prevent a core meltdown a manual secondary depressurization (cooldown) is needed to restore the heat sink through fast reduction of the secondary pressure inducing similar primary pressure behavior, and thus leading to passive accumulator injection and low pressure safety injection. The core exit temperature parameter can be used to detect a core heat up and trigger such accident management procedures.

The test OECD-PKL2 G7.1 was performed to assess the reliability of the core exit temperature measurement and its correlation to the peak cladding temperature, and to investigate the physical processes affecting the performance of the core exit temperature measurement.

In this report, the analysis of test OECD-PKL2 G7.1 using the US-NRC thermal-hydraulics best estimate system code TRACE and based on an existing PKL Mark III model is presented and the obtained results are compared with the experimental data. A good agreement with the experimental data is observed, especially for relevant parameters such as the primary pressure. The predicted core exit and peak cladding temperatures evolutions could also capture the experimental data, although some discrepancies in the shapes of the evolution curves are observed.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
LOCA, Small Break, PKL, Thermal-hydraulic, TRACE, Core Exit Temperature	unlimited
	14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified
	15. NUMBER OF PAGES
	16. PRICE

UNITED STATES
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NUREG/IA-0432 **Analysis of the Test OECD-PKL2 G7.1 with the Thermal-Hydraulic System Code TRACE** **September 2013**