NUREG/IA-0492

International Agreement Report

Assessment of TRACE V5.0 Patch 4 Code Against PWR PACTEL Loop Seal Clearing Experiment

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Washington, DC 20555-0001

Manuscript Completed: August 2017
Date Published: December 2018

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 TRAC/RELAP advanced computational engine (TRACE) developed by U.S. nuclear Regulatory Commission is one of the main system codes to perform thermal-hydraulic safety analyses of loss-of-coolant accidents, operational transients, and other accident scenarios of light water reactors. This report presents the TRACE calculation model of the PWR PACTEL facility and the calculation results of PWR PACTEL loop seal clearing experiment LSC-03 with TRACE V5.0 patch 4. The PWR PACTEL facility is designed and constructed in 2009 at Lappeenranta University of Technology and used in the safety studies related to thermal-hydraulics of pressurized water reactors with European pressurized water type vertical U-tube steam generators.

The TRACE calculation results were compared to the experimental data. In general the results agreed reasonably well with experimental data. Some discrepancies were found in core peak temperatures, water level predictions, and the pressure and temperature predictions on the secondary side of the steam generators after loop seal clearing. The behavior of the loop seals seemed to be relatively similar to the experiment as one loop seal cleared out and another refilled. However, in the calculation, the loop seal 1 cleared and the loop seal 2 refilled, while in the experiment the behavior was opposite.

TABLE OF CONTENTS

LIST OF FIGURES

EXECUTIVE SUMMARY

The TRAC/RELAP advanced computational engine (TRACE) developed by U.S. Nuclear Regulatory Commission is one of the main system codes to perform thermal-hydraulic safety analyses of loss-of-coolant accidents, operational transients, and other accident scenarios of light water reactors. Therefore, assessing the calculation capability of the TRACE code to predict different thermal-hydraulic transients in the nuclear power plant is essential.

The PWR PACTEL integral test facility is designed and constructed in 2009 by the Nuclear Safety Research Unit at Lappeenranta University of Technology (LUT). It is used in the safety studies related to thermal-hydraulics of pressurized water reactors with European pressurized water type vertical U-tube steam generators. The research focuses on different phenomena in the main circulation loops and vertical steam generators in particular. The TRACE system code is used as a helping tool for experiment planning and result analyzing.

This report presents the TRACE calculation model of the PWR PACTEL facility and the calculation results of PWR PACTEL loop seal clearing experiment LSC-03 with TRACE V5.0 patch 4. The calculation results were compared to the experimental data and, in general, the code predicts the experimental data reasonably well. The main events of the transient were predicted satisfactorily. However, some discrepancies were found in core peak temperatures, water level predictions, and the pressure and temperature predictions on the secondary side of the steam generators after loop seal clearing.

The core peak cladding temperature before the loop seal clearing was relatively well predicted but the second peak before the safety system injection did not occur in the calculation. The reason for absence of second temperature peak was that the core water level recovery by loop seal clearing was overestimated in the calculation. It is possible that the extra water in the core was gathering from different places, like loop seals, tubes and plenums of steam generators, or the pressurizer surge line. However, based on the available measurements, it is difficult to say exactly how the water was actually distributed in the primary side during the experiment. The calculated pressure and temperature on the secondary side of the steam generators after loop seal clearing were substantially higher than measured values. The reason for this difference could be in the modeling of the secondary side heat losses and massive steel structures, for example flanges of the secondary side, which serve as a heat storage. The behavior of the loop seals seemed to be relatively similar to the experiment as one loop seal cleared out and another refilled. In the calculation, the loop seal 1 cleared and the loop seal 2 refilled, while in the experiment the behavior was opposite.

ACKNOWLEDGMENTS

The author acknowledges the financial support from The Finnish Research Programmes on Nuclear Power Plant Safety (SAFIR2014) associated with CAMP program.

ABBREVIATIONS

CCFL	Counter-Current Flow Limitation
ECCS	Emergency Core Cooling System
EPR	European Pressurized Reactor
EXP	Experiment data
HPIS	High Pressure Injection System
LSC	Loop Seal Clearing
LOCA	Loss of Coolant Accident
LPIS	Low Pressure Injection System
LUT	Lappeenranta University of Technology
PACTEL	Parallel Channel Test Loop
PWR	Pressurized Water Reactor
SETS	Stability Enhancing Two-Step Numerical Method
SNAP	Symbolic Nuclear Analysis Package
TRACE	TRAC/RELAP Advanced Computational Engine
US NRC	United States Nuclear Regulatory Commission

1 INTRODUCTION

The TRAC/RELAP advanced computational engine (TRACE) (Ref. 1) is a best-estimate system code developed by U.S. Nuclear Regulatory Commission. The code is developed to perform analyses of loss-of-coolant accidents (LOCAs), operational transients, and other thermal-hydraulic transient scenarios of light water reactors. It can also be used to model thermal-hydraulic phenomena occurring in experimental facilities.

The PWR PACTEL integral test facility (Ref. 2) is designed and constructed in 2009 by the Nuclear Safety Research Unit at Lappeenranta University of Technology (LUT). It is used in the safety studies related to thermal-hydraulics of pressurized water reactors with European pressurized water reactor (EPR) type vertical U-tube steam generators. The research focuses on different phenomena in the main circulation loops and vertical steam generators in particular. The TRACE system code is used as a helping tool for experiment planning and result analyzing.

In the present work the TRACE calculation model of the PWR PACTEL facility and the calculation results of PWR PACTEL loop seal clearing experiment LSC-03 with TRACE V5.0 patch 4 are presented and discussed. The results are compared against the experimental data. The calculation results are used to improve the TRACE model of PWR PACTEL facility and test the applicability of the nodalization.

2 PWR PACTEL TEST FACILITY AND LOOP SEAL CLEARING EXPERIMENT LSC-03

2.1 PWR PACTEL Test Facility

The PWR PACTEL test facility (Ref. 2) is designed and constructed in 2009 to be used in the safety studies related to thermal-hydraulics of pressurized water reactors with EPR type vertical U-tube steam generators. The PWR PACTEL facility consists of a reactor pressure vessel model, two loops with vertical steam generators, a pressurizer, and emergency core cooling systems (ECCSs). Figure 1 presents a general view of the PWR PACTEL facility and inverted U-tube steam generator. Table 1 presents characteristics of the PWR PACTEL facility.

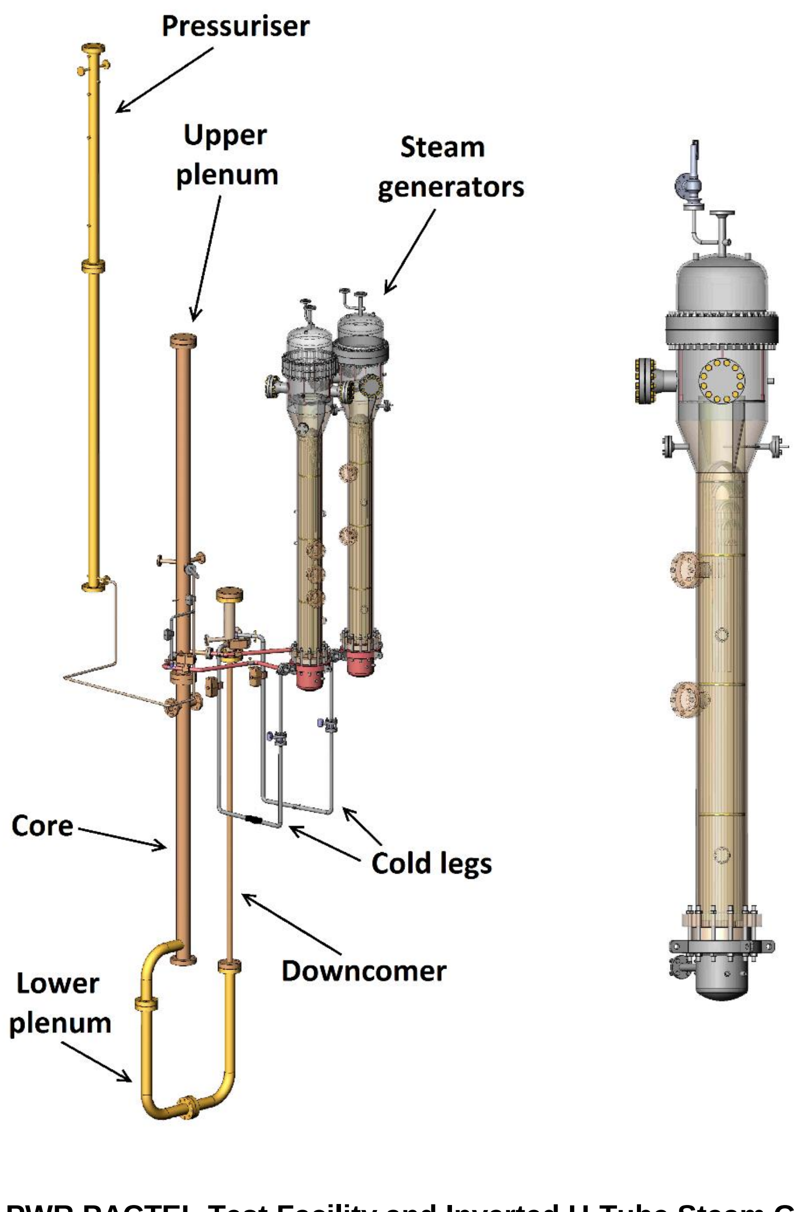

Figure 1 PWR PACTEL Test Facility and Inverted U-Tube Steam Generator

The pressure vessel model comprises a U-tube construction modeling a downcomer, lower plenum, core, and upper plenum. The core part is simulated with a rod bundle of 144 electrically heated fuel rod simulators distributed in three parallel channels. The maximum core power is 1 MW, which corresponds roughly to the scaled residual heating power of the EPR reactor. The pressurizer is connected to the hot leg of loop 2. ECCSs in the PWR PACTEL facility include the high and low pressure safety injection system pumps (HPIS and LPIS) and two separate accumulators.

Both loops with the vertical U-tube steam generators are designed to simulate the behavior of one reference EPR type primary loop. As there are four primary loops in the EPR, half of the rated EPR capacity is simulated with the PWR PACTEL facility. Compared to the reference steam generator, the height of the PWR PACTEL steam generator scales down with a ratio of 1:4. The heat transfer area of the steam generator U-tube bundle and the primary side volume of both steam generators are scaled with a ratio of 1:400. Both steam generators include 51 heat exchange tubes with an average length of 6.5 m. The tubes are arranged in a triangular grid and in five groups with different lengths. Figure 2 presents the configuration of heat exchanger tubes in the PWR PACTEL steam generator. The secondary side of the steam generators is divided into several volumes. The annular type downcomer surrounds the riser area in which the heat exchange tube bundle is located. The downcomer is divided into hot and cold compartments. The lower riser area is also divided into hot and cold compartments with a divider plate. The upper part of the steam generators is the steam volume from where steam is conveyed to steam lines.

Table 1 PWR PACTEL Facility Characteristics

Characteristics	**PWR PACTEL**
Reference power plant (loops and steam generators)	PWR (EPR)
Volumetric scale: pressure vessel, steam generators, pressurizer	1:405, 1:400, 1:562
Height scale: pressure vessel, steam generators, pressurizer	1:1, 1:4, 1:1.6
Number of primary loops	2
Maximum core heating power [MW]	1
Number of fuel rod simulators	144
Outer diameter of fuel rod simulators [mm]	9.1
Heating length of fuel rod simulators [m]	2.42
Axial power distribution of the core	Chopped cosine
Maximum fuel rod simulator cladding temperature [ºC]	750
Maximum design primary / secondary pressure [MPa]	8.0 / 4.65
Maximum design primary / secondary temperature [ºC]	300 / 260
Steam generator heat exchange tube diameter / thickness [mm]	19.05 / 1.24
Average steam generator heat exchange tube length [m]	6.5
Number of heat exchange tubes in steam generator	51
Maximum accumulator pressure [MPa]	5.5
Maximum HPIS/LPIS water pressure [MPa]	8.0 / 0.7
Main material of components	Stainless steel (AISI 304)
Insulation material	Mineral wool (aluminum cover)

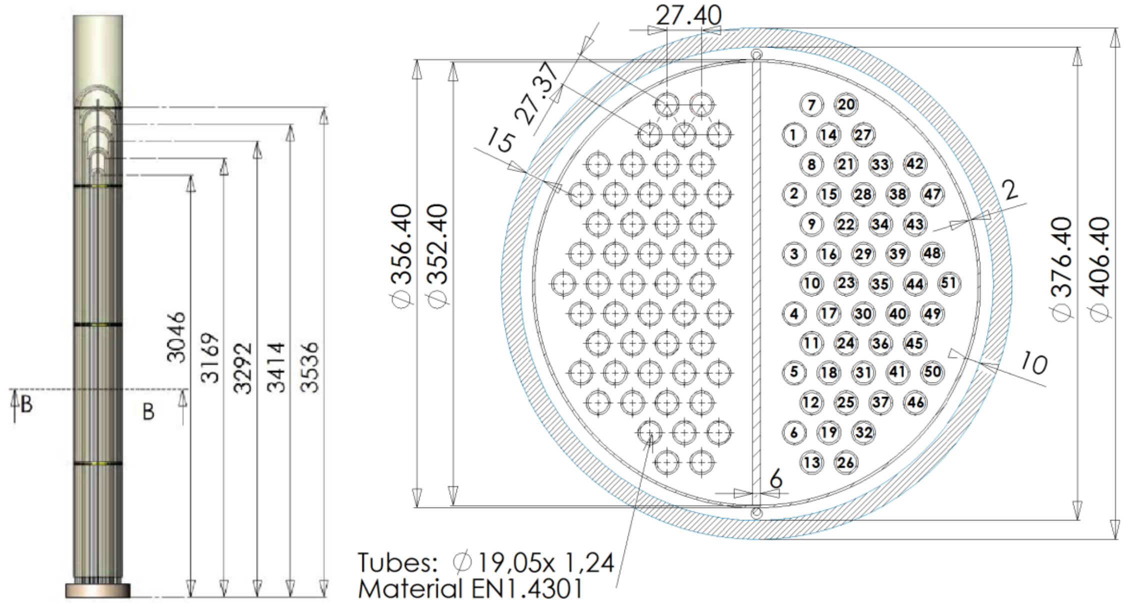

Figure 2 Heat Exchange Tubes of the PWR PACTEL Steam Generators

2.2 Loop Seal Clearing Experiment LSC-03

The LSC-03 experiment with the PWR PACTEL test facility is one of the three experiment to determine the effect of break size on loop seal clearing (LSC) phenomenon (Ref. 3). In the experiment the reactor pressure vessel model, both loops, the pressurizer, and two ECCSs, i.e. safety injection system with pump and the accumulator, were used. Both of these ECCSs injected water to the top part of the downcomer. The break was located in the cold leg in the loop 2 between the loop seal and the downcomer on the top side of the pipe. Figure 3 presents the break line configuration. An orifice plate with the diameter of 6 mm was used to simulate the break. The break size is about 1.3% of the PWR PACTEL cold leg cross-sectional area and corresponds the break size of 115 cm^2 in the EPR scale. The two-phase mixture leaked from the primary system was condensed and collected in a separate tank. An important detail regarding the LSC phenomenon is that there was no by-pass between the upper plenum and the downcomer. This means that the pressure difference between the downcomer and the upper plenum can only stabilize through the loop seals or the lower plenum.

Because there were no main circulation pumps in the PWR PACTEL facility at the time when the LSC-03 experiment was carried out, the facility was operated with natural circulation. The experiment was conducted by first establishing steady state operation at full inventory for 1000 seconds and then opening a break. Table 2 presents the initial conditions of the LSC-03 experiment after the 1000 second steady-state period. When the break was opened the water level in the pressurizer began to decrease. The pressurizer heaters were switched off when the water level in the pressurizer reached the top of the heaters. The safety injection system pump was started manually when the primary pressure reached at 38.5 bar. The accumulator injection started at 30 bar. The secondary side water level was maintained at a constant level during the whole experiment.

No other actions were taken during the experiment. The experiment was continued until LSC occurred, the HPIS and accumulator was injecting, and the water level in the core was restored. The volume of the tank, that collected the leaked water mass of the system, was about 700 l. The adequacy of this volume also limited the duration of the experiment.

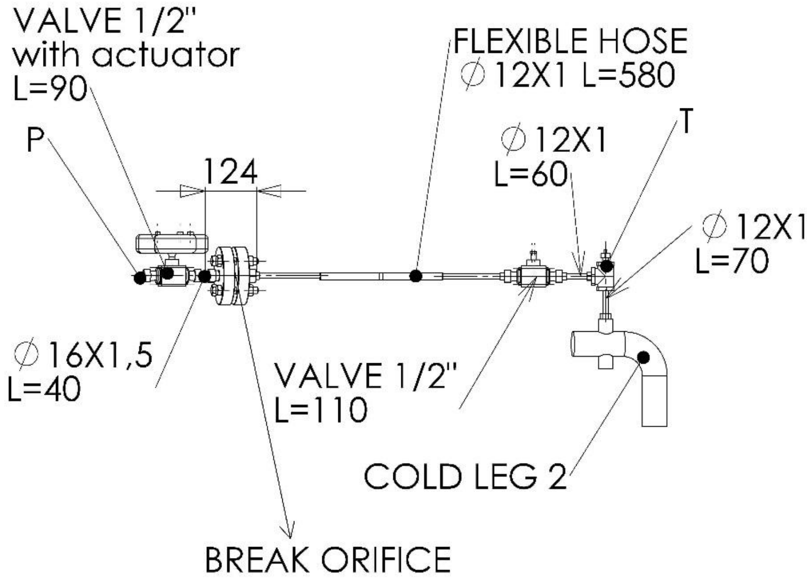

Figure 3 **Break Line Configuration in LSC-03 Experiment in the PWR PACTEL Facility**

Table 2 **Initial Conditions of the LSC-03 Experiment and TRACE Calculation After 1000 Second Steady State Period**

Parameter	LSC-03 experiment	TRACE calculation
Primary side pressure	75 ± 1 bar	74.4 bar
Secondary side pressure SG1/SG2	40.0 ± 0.6 bar	40.0/40.1 bar
Core power	180 ± 6 kW	180 kW
Downcomer mass flow rate	1.3 ± 0.3 kg/s	1.32 kg/s
Core inlet temperature	249 °C ± 3 °C	250 °C
Core outlet temperature	277 °C ± 2 °C	277 °C
Pressurizer level	5.7 ± 0.2 m	5.62 m
Steam generator collapsed level SG1/SG2	3.90 ± 0.12 m	3.89/3.90 m
Feed water temperature	22 ± 1 °C	21 °C
Accumulator pressure	30 ± 1 bar	30.0 bar
Accumulator temperature	54 ± 4 °C	50 °C
Safety injection system flow rate	6.0 ± 0.3 l/min	6.0 l/min
Safety injection system temperature	19 ± 2 °C	20 °C

3 TRACE INPUT MODEL OF PWR PACTEL

The TRACE code version 5.0 patch 4 was used in the calculation of the LSC-03 experiment. Symbolic nuclear analysis package (SNAP) version 2.5.2 was used to create the TRACE nodalization of the PWR PACTEL model. The nodalization is totally 1-dimensional and includes all the main components of PWR PACTEL. The nodalization includes totally of 121 hydraulic components, 121 heat structures, 6 power components, 185 control blocks, 296 signal variables, and 3 trips. Figure 4 presents the nodalization of the pressure vessel and accumulator of PWR PACTEL.

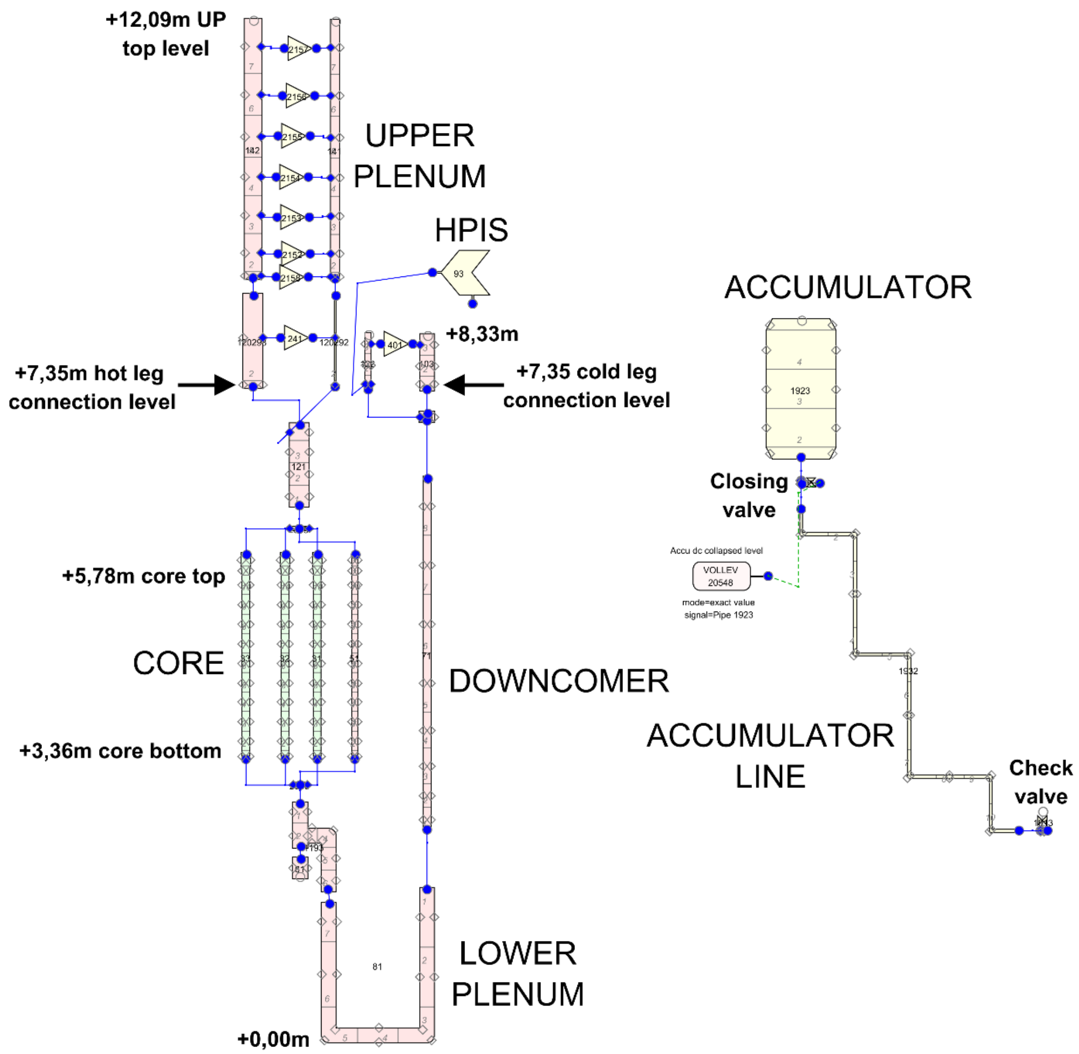

Figure 4 Nodalization of Pressure Vessel and Accumulator of PWR PACTEL Facility. The Accumulator is Connected in Same Cell as HPIS.

The core region is modeled with three separate heated channels and one core bypass, based on the PWR PACTEL construction.

In PWR PACTEL, the core section is divided into different axial power sections to model cosine axial power distribution in the core. In the nodalization, the node lengths of the core section are based on these power sections and power axial distribution is set according to the facility values. The upper plenum and the upper part of the downcomer include two parallel pipelines with multiple horizontal connections following the facility construction (there are diffusers, i.e., hollow pipes with holes inside the upper plenum and upper part of the downcomer of PWR PACTEL) and allowing a water circulation in these parts. This nodalization construction results in more accurate temperature calculation in upper plenum and prevents direct HPIS and accumulator water flow to the cold legs in the upper part of downcomer. The U-shaped lower plenum and downcomer region are modeled with pipe components. The accumulator and related piping are fully modeled. The accumulator is modeled with pipe component using a non-spherical accumulator model option. Accumulator water injection is controlled with a pressure check valves. The HPIS is simply modeled with fill component by injecting the desired water amount to the primary circuit, i.e., there is no actual HPIS pump in the model.

Figure 5 presents the nodalization of loop 2 with the inverted U-tube steam generator and pressurizer. Loop 1 is modelled with same principles as loop 2. In the U-shaped loop seals, practicable small and almost equal sized cells are used in the down-flow and up-flow sides. The pressurizer is modeled with a single vertical pipe component. The pressurizer heaters and the water injection are included in the nodalization allowing pressure and water level control of the primary system.

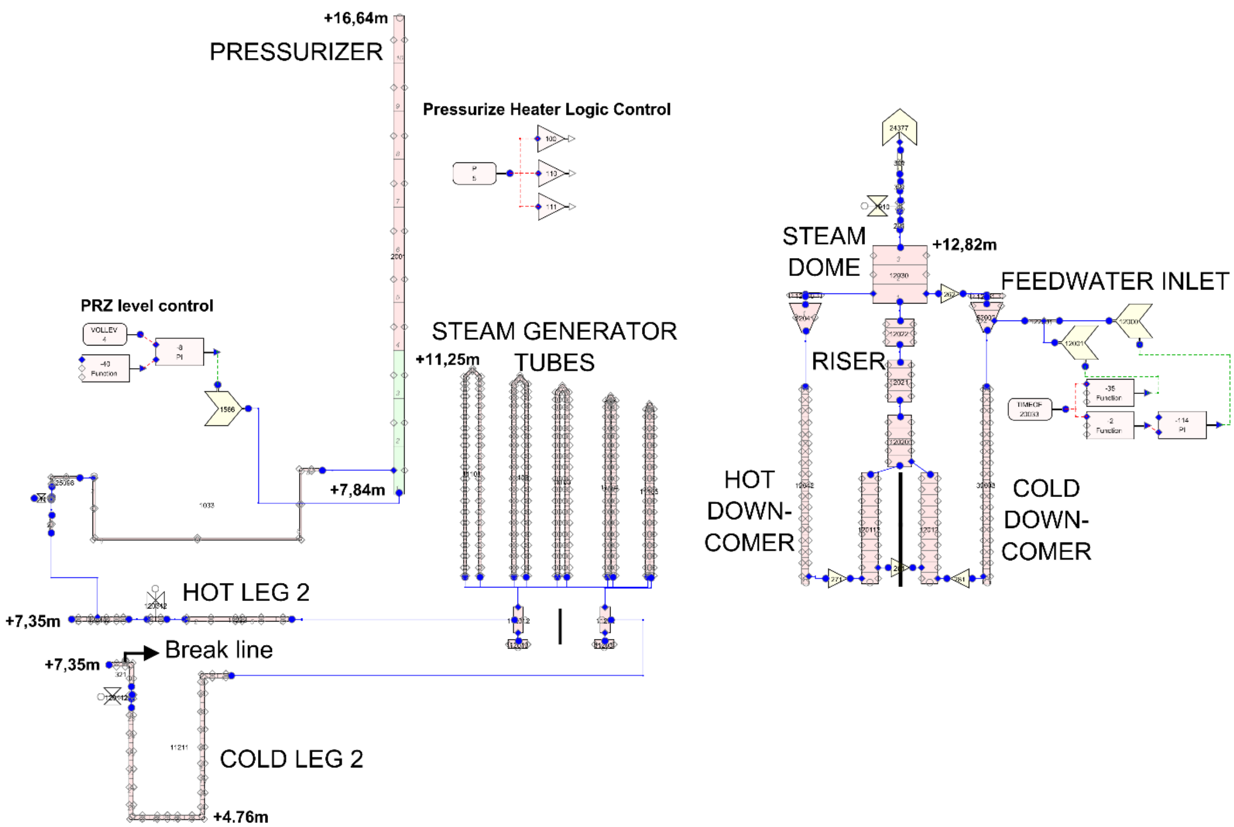

Figure 5 **Nodalization of Loop 2 With Inverted U-Tube Steam Generator and Pressurizer**

The 51 steam generator tubes in the PWR PACTEL steam generator are lumped into five tubes in the nodalization according to the heights of the tubes. The steam generator tube nodalization is constructed partly based on the facility set-up and the locations of the instrumentation. The cells are mainly equal size and the same cell length is used also on the secondary side of the steam generator to easy the modeling of the primary to secondary side heat transfer. The hot and cold plenums are modeled with two nodes allowing water retention in the plenums during the LOCA calculation. The secondary side nodalization is relatively detailed including the various volumes, e.g. the hot and cold downcomer, the riser, the steam dome, and the steam pipe lines. The secondary side pressure is controlled with motor control valve and the water level is adjusted with feed water control systems.

Figure 6 presents the nodalization of the break line in the cold leg 2. The break line is presented relatively accurately in the nodalization. The break valve 1 is used to control opening of the break. The flexible hose and the pipe with the orifice plate is modeled with pipe components. The pipe with the orifice plate is modelled with three cells where the middle cell represents the orifice plate. The choke flow is defined in both edge of this cell. After the break valve 2, break component is used to model the boundary conditions.

The pressure and heat losses for the TRACE model are defined according to the separate pressure and heat loss experiments. The pressure losses are defined separately for the different parts of the model and for normal and reversed flow direction with different mass flow rates. The heat losses are defined in different steady-state conditions by changing the core power and the primary and secondary side pressures. Heat losses are adjusted by changing a thermal conductivity of mineral wool.

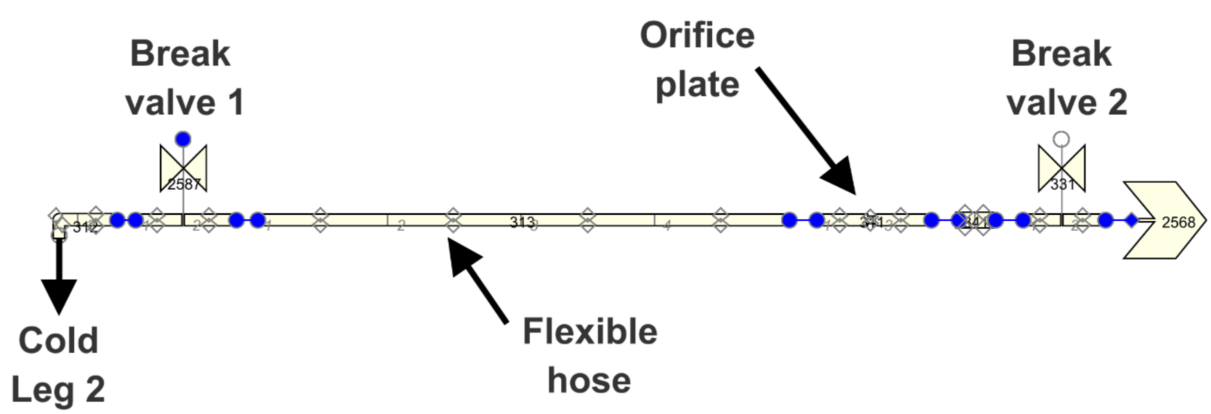

Figure 6 Nodalization of Break Line

4 CALCULATION RESULTS

4.1 TRACE Calculation Results of LSC-03 Experiment of PWR PACTEL

Figures 7 to 13 present the measured data from LSC-03 experiment and the calculation data of TRACE code calculation. In these figures the experimental data is labeled as “EXP” and calculation data as “TRACE”. The 1000 second steady-state period before the experiment is also presented in the figures. The timing of main events in the experiment is presented with vertical dash lines. Table 2 presents the initial conditions of the experiment and the code calculation after the 1000 second steady-state period. The initial conditions in the calculation match relatively well with the experiment values.

The timing and the sequence of the main events in the experiment and calculation is presented in Table 3. As Table 3 shows, in the calculation LSC occurs almost correct time but the HPIS injection, the secondary side pressure decreasing below primary side pressure, and the accumulator injection starts earlier than in the experiment. These discrepancies result from the difference between the measured and calculated primary side pressures, as explained later.

Table 3 Sequence of Events in LSC-03 Experiment and TRACE Calculation

Event	LSC-03 experiment	TRACE calculation	Timing difference
Start of the transient	1000 s	1000 s	0 s
Loop seal clearing	1614 s	1619 s	+ 5 s
HPIS injection	1837 s	1789 s	- 48 s
Primary pressure < secondary pressure	1923 s	1684 s	- 239 s
Accumulator injection	2255 s	2060 s	- 195 s

Figure 7 presents the drained water volume through the break in the experiment and calculation. In the TRACE nodalization, the constant form loss coefficient of the break orifice plate is adjusted so that the break flow rate and the total break water inventory of the calculation match the experiment as well as possible. In the calculation, the total amount of leaked water is about 6 liters higher than in the experiment before LSC. The total amount of leaked water in the experiment before LSC was about 373 liters; the error is thus about 1.6%.

Figure 8 presents the volume flow rate of the break in the experiment and calculation. After LSC, the break flow rate stabilizes to the level of 5–10 l/min in the experiment. When the HPIS injection of 6 l/min starts, it is roughly sufficient to compensate the break flow rate. The water inventory of the primary system thus remains nearly constant or slightly increases after the HPIS injection begins. After LSC, the calculated break flow rate corresponds relatively well with the value of the experiment.

Figure 9 presents the primary and secondary side pressures in the experiment and calculation.

The primary side pressure behavior is predicted relatively well in the calculation until LSC occurs. A small perturbation in the primary side pressure during LSC, probably caused by water oscillation in the loop seals, can also be seen in the calculation. The calculation predicts that the primary pressure drops slightly lower level after the perturbation compared to the experiment. This lower pressure causes too early HPIS injection in the calculation (Table 3). Later, it seems that the HPIS injection cools down the primary side too fast in the calculation and eventually causes also the accumulator injection to start too early.

The secondary side pressures (Figure 9) are limited to 40 bars by controlling the steam valves of the steam generators. After LSC, the primary side pressure decreases below the secondary side pressure and therefore, the heat transfer between the primary and secondary side is reversed. Heat transfers from the secondary side to the primary side and decreases the secondary side pressure. This decrease is clearly steeper in the experiment than in the calculation. The reason for this discrepancy could be in the modeling of the secondary side heat losses and massive steel structures, for example flanges of the secondary side, which serve as a heat storage. Another explanation could be the cross flows between the hot and cold downcomer on the secondary side. In the PWR PACTEL steam generators, there is a gap between these downcomers that is not modeled in detail in the nodalization. The lack of this mixing flow prevents the water circulation on the secondary side between the hot and cold downcomers, and thus could reduce the temperature difference between the primary and secondary side, experienced by the ascending and descending U-tube sections. In the calculation, the steam flow from the core retains water in the steam generator 1 tubes after LSC, which improves the heat transfer from the secondary side to primary side causing faster secondary side pressure decrease in steam generator 1 compared to steam generator 2 after LSC.

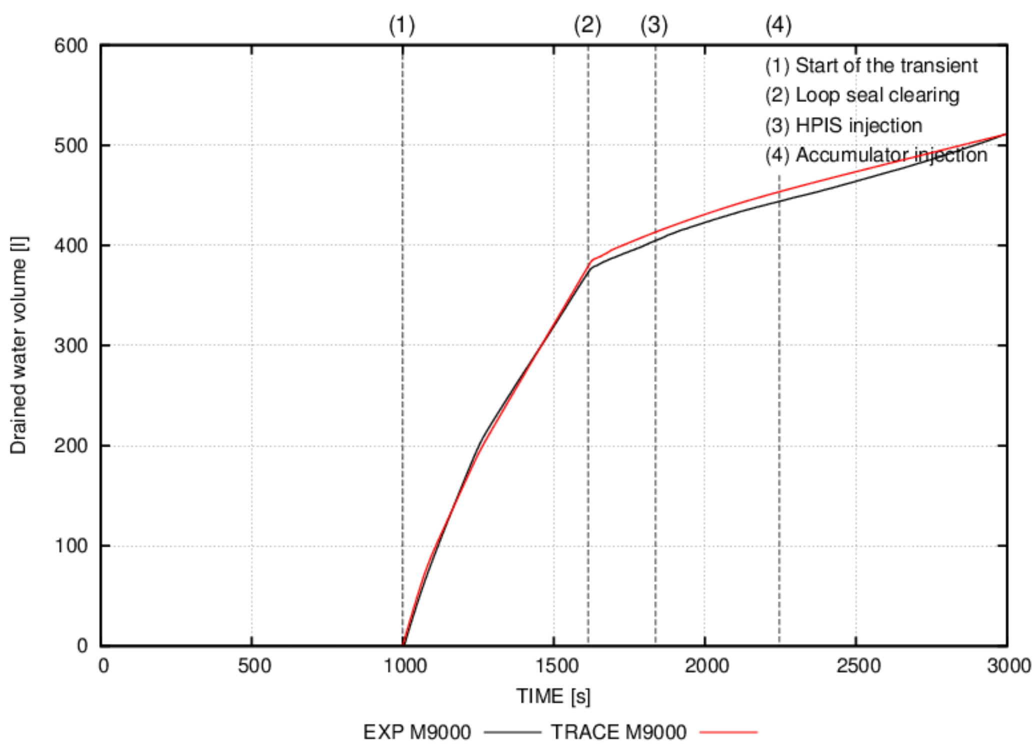

Figure 7 Drained Water Volume in LSC-03 Experiment and TRACE Calculation

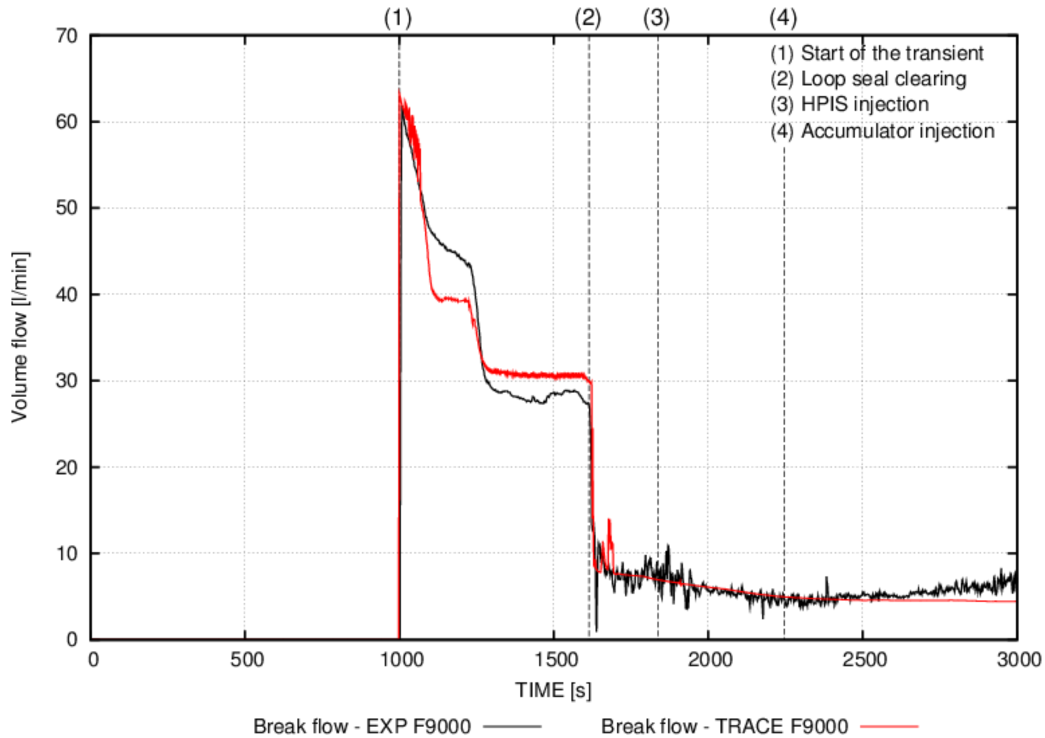

Figure 8 Volume Flow Rate of the Break in LSC-03 Experiment and TRACE Calculation

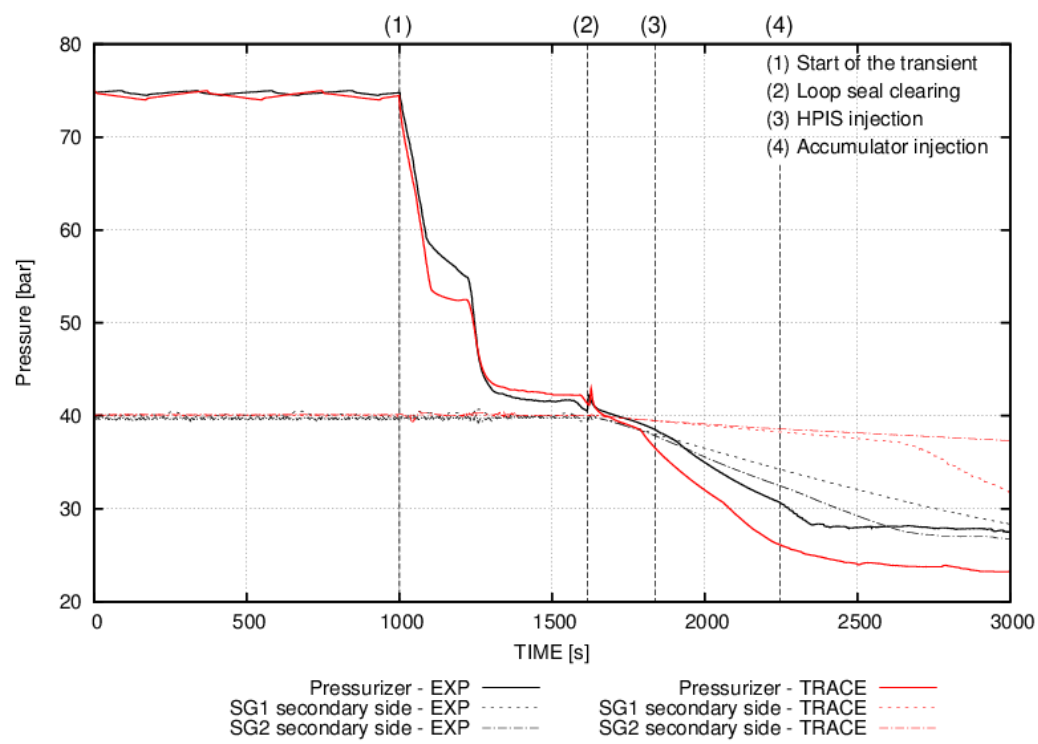

Figure 9 Primary Side and Secondary Side Pressure in LSC-03 Experiment and TRACE Calculation

In the experiment, a core temperature excursion occurs first time before LSC, and for the second time at the top part of the core before the HPIS injection (Figure 10). Both excursions are caused by the decrease of the water level in the core (Figure 11) and the depressurization of the primary system. The reason for the first water level decrease is the manometric effect caused by the closed loop seals and for the second excursion the water inventory depletion on the primary side. The first excursion is halted by LSC which allows the core water level to recover. The second temperature excursion is interrupted by the HPIS injection.

The TRACE code predicts the maximum temperature in the middle of the core before LSC relatively well (maximum temperature is about 5 °C higher than in the experiment) but in the top part of the core the TRACE code underestimates the maximum temperature by 25 °C. The reason for the lower temperature at the top part of the core is the reversed flow from the hot legs to the core before LSC. The countercurrent flow limitation model (CCFL) with the default coefficient values is used in the choke plate above the core. The CCFL model and the choke plate with upward flowing steam causes a water hold-up above the core during the reverse flow. This hold-up is not completely water tight and water leaks to the highest nodes of the core. This leaking causes that the core dry out occurs about the same time in the top and in the middle part of the core. In the experiment the core dry out occurs in the top part earlier than in the middle part of the core. This later dry out in the calculation explains why the peak temperature is lower in the top of the core. The selected nodalization of the pressure vessel above the core or the selected CCFL coefficients might affect the peak temperature; however, this is not tested.

The TRACE code overestimates the collapsed water level recovery in the core caused by LSC (Figure 11). The collapsed water level is overestimated by around 0.35 m, which corresponds to around 6–7 liters of water. A reason for the overestimated water level could be that in the calculation too much water is flowing from steam generator 2 tubes and plenums into the downcomer immediately after LSC, increasing the water inventory in the core, lower plenum, and downcomer section. It is also possible that extra water in the core is gathering from loop seals or the pressurizer surge line. With the TRACE code, the effect of water retention in the plenums to the core water level is tested by splitting up the plenums in two cells (see Figure 5, initially the plenums were modeled with a single cell horizontal pipe component). In the PWR PACTEL facility, the elevations of the hot/cold leg connections in the steam generator plenums are halfway up the plenum height, and about 10 liters of water can get stuck in each plenum. In the changed nodalization, the volume of separate plenum dead-ends is about 1/3 of the whole plenum volume. However, this did not explain the too high water level in the core in the calculation. Based on the available measurements, it is difficult to say exactly how the water is actually distributed in the primary side during the experiment.

In the calculation, too high water level in the core after LSC and too early HPIS and accumulator injection compared to experiment interrupts the second core water level decrease and maintains the water level near or over the top of the core for rest of the calculation (Figure 11). Because of this too high core water level, the second temperature excursion of the experiment do not occur in the calculation (Figure 10). After the accumulator injection, the collapsed water levels in the core and downcomer in calculation stabilize to the measured level.

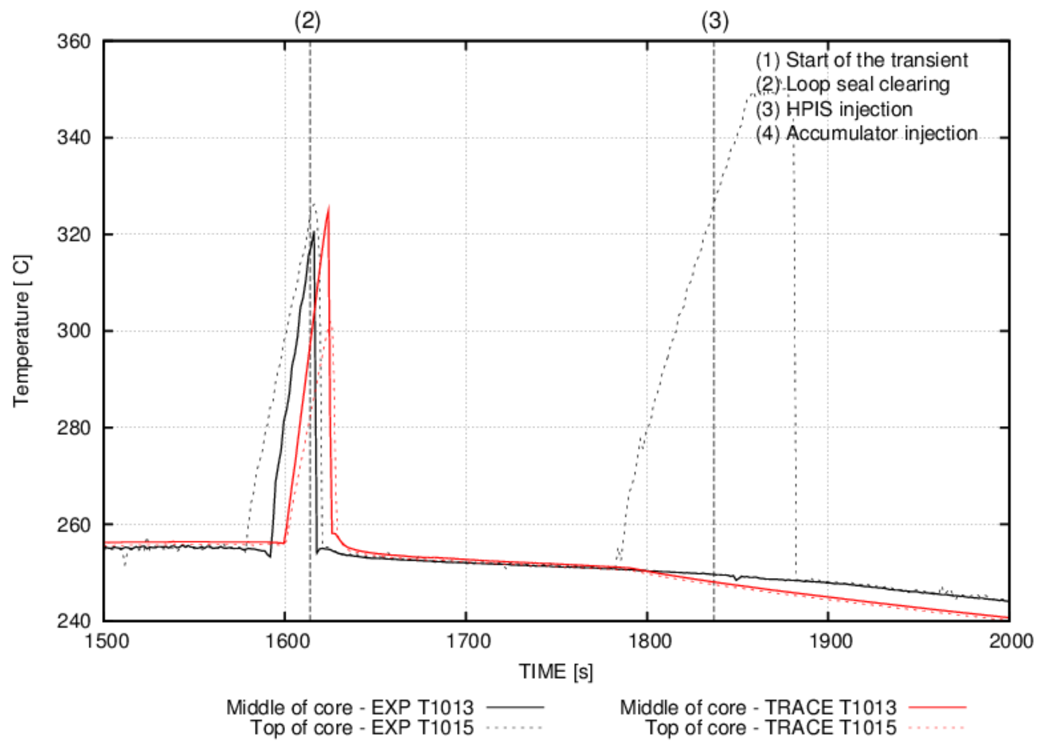

Figure 10 **Rod Temperatures at the Middle and Top of the Core in LSC-03 Experiment and TRACE Calculation**

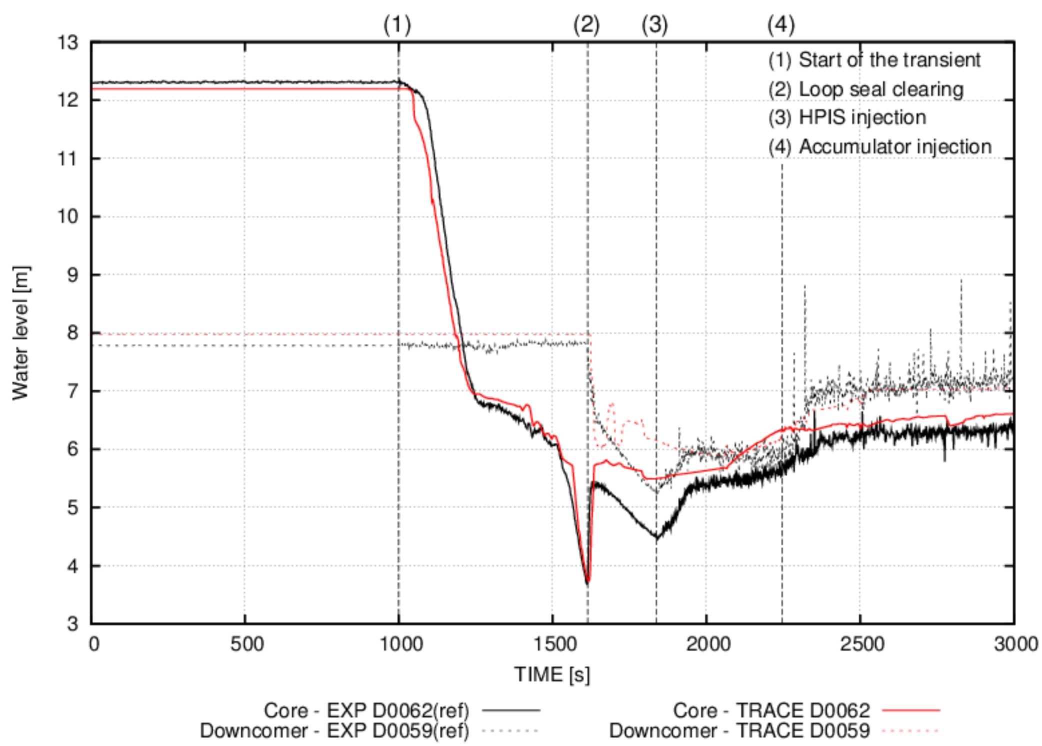

Figure 11 **Core and Downcomer Collapsed Water Levels in LSC-03 Experiment and TRACE Calculation**

LSC is a consequence of hydrostatic forces and the local conditions of the loop. The phenomenon can generally occur in any loop of a primary system, and it is unlikely that all loop seals will clear together (Ref. 4). Figure 12 and Figure 13 present the water level behavior of loop seal 1 and loop seal 2, respectively, in the experiment and in the TRACE calculation. As can be seen from these figures, in the experiment, the water level in the down-flow side in both loop seals starts to decrease slowly after about 500 seconds from the break opening. Both these water levels reach the bottom around 610 seconds after the break opening. After that, the vapor in hot leg 2 escapes via the bottom of loop seal 2, clears the water from the loop seal (the collapsed water level of the up-flow side decreases fast), and relieves the pressure difference between the loops. At the same time, in loop seal 1, the water level of down-flow side recovers because of the disappearance of the pressure difference over the loop seal.

In the TRACE calculation, the timing of LSC matches the experiment well which indicates that the volume of the system and drained flow rate of the break are well predicted. The loop seal water levels on the down-flow side decrease in both loops, as in the experiment, and reach the bottom at almost the same time. After that the behavior of the loop seals seems to be relatively similar to the experiment as one loop seal clears out and another refills. However, in the calculation, the loop seal 1 clears and the loop seal 2 refills, while in the experiment the loop seal 2 clears and the loop seal 1 refills.

In the calculation, part of the water from the refilled loop seal 2 flows to the break after LSC and the down-flow side water level stays lower level than in the experiment (Figure 12 and Figure 13). This water level draining from the loop seal to the break may cause that larger amount of water preserves in the core, lower plenum, and downcomer section in the calculation and could partly explain why the calculated water level in the core is higher after LSC than the measured water level.

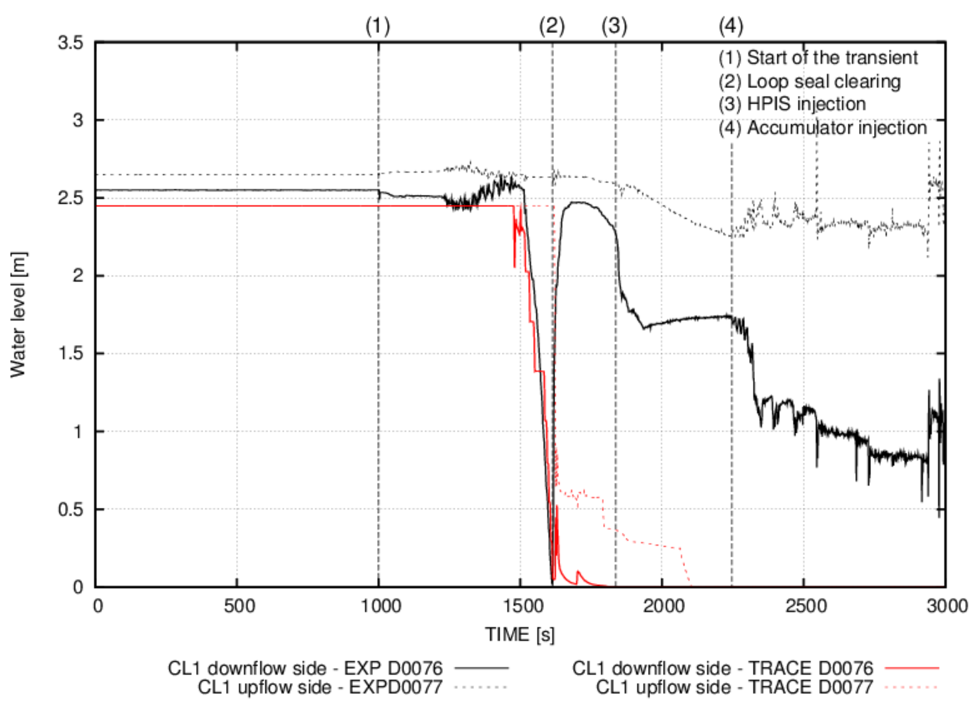

Figure 12 Collapsed Water Levels in Loop Seal 1 in LSC-03 Experiment and TRACE Calculation

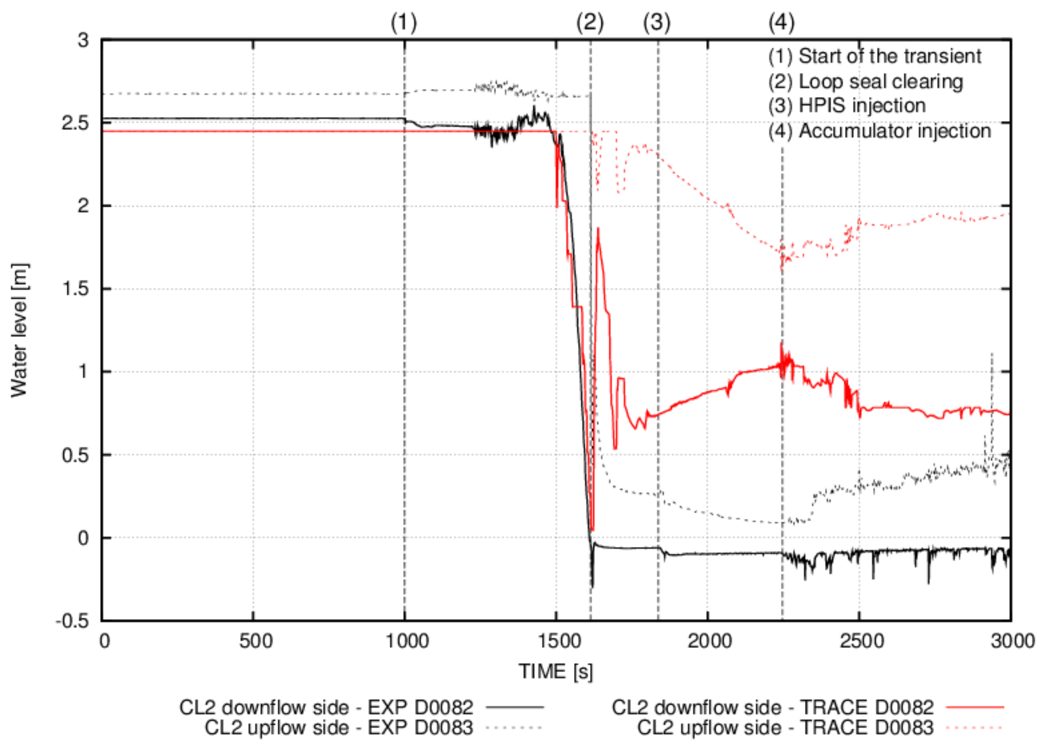

Figure 13 Collapsed Water Levels in Loop Seal 2 in LSC-03 Experiment and TRACE Calculation

4.2 SNAP Animation Model

A SNAP animation model is developed to graphically displaying the evolution of LSC-03 calculation. The animation model is developed by copying the necessary components from the SNAP calculation model and pasting them in the animation model.

Figure 14 presents water distribution (void fraction in each cell) on the primary side before LSC in the TRACE calculation. At this point, the water level of the core is decreased to lowest point and the water levels in the down-flow sides of the loop seals are decreased to the bottom of the loop seals. Water from the hot legs, hot plenums, and steam generator tubes is flowing to the core but does not wet the upper part of the heat rods because upward flowing steam holds the water above the core and guide it to the core bypass.

Figure 15 presents water distribution on the primary side 20 seconds after LSC. At this point, the water level of the core is already increased to the top level of the core. The loop seal 1 is opened and emptied from water, and the loop seal 2 is almost refilled. Water from hot leg 2 and steam generator 2 is draining to the core, and steam generated in the core is flowing through the loop 1 to the break.

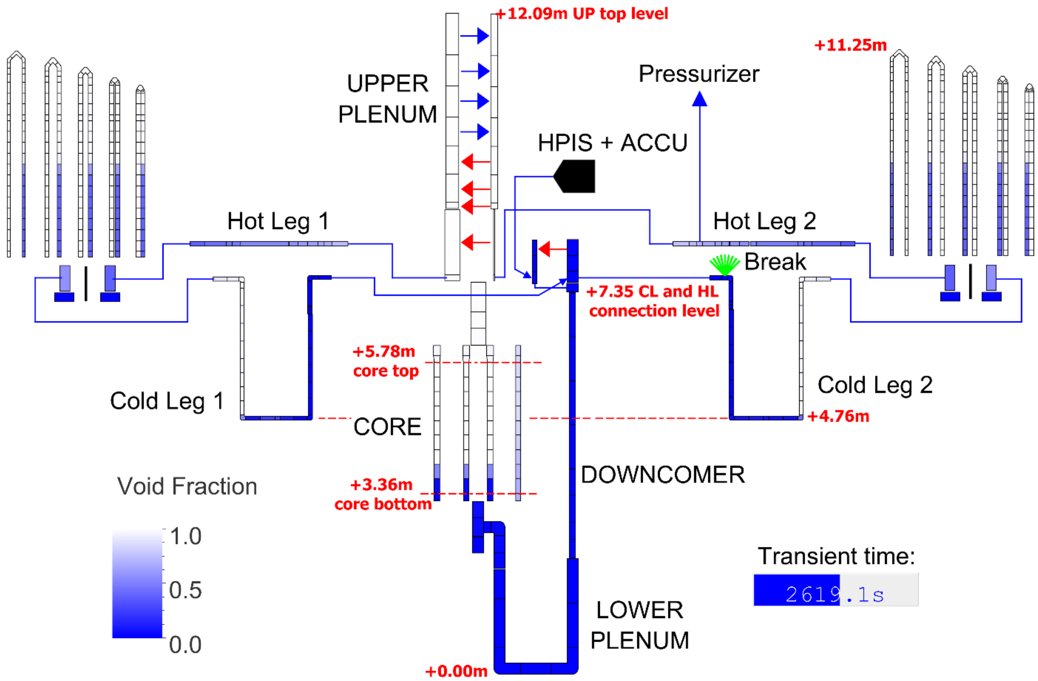

Figure 14 View of SNAP Animation Model of the PWR PACTEL Facility During LSC-03 Calculation, Before Loop Seal Clearing

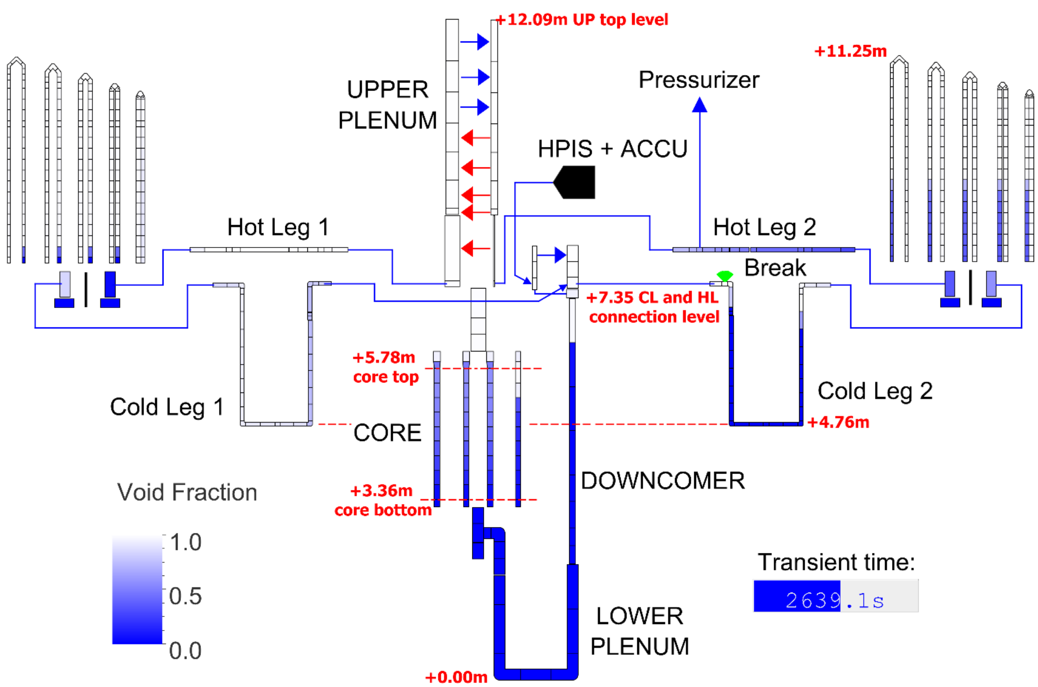

Figure 15 View of SNAP Animation Model of the PWR PACTEL Primary Side During LSC-03 Calculation, 20 Seconds After the Loop Seal Clearing

5 RUN STATISTICS

The calculation was performed using Intel® Core™ i7-2600 @ 3.4 GHz processor. The operating system was Windows 7 Enterprise. Table 4 shows the run statistics for the calculation. The used TRACE code version was TRACE Patch 4. Transient time contains 2000 second steady state period before actual LOCA transient. The default stability enhancing two-step (SETS) numerical method was used in the calculation.

Table 4 Run Statistics

Code	Transient Time (s)	CPU Time (s)	CPU/Transient Time	Number of Time Steps
TRACE Patch 4	5000.0	18511	3.70	514255

6 CONCLUSIONS

The loop seal clearing experiment LSC-03 of the PWR PACTEL test facility is calculated with TRACE v5.0 patch 4. The created TRACE nodalization for the PWR PACTEL facility is totally 1-dimensional and included all the main components of the PWR PACTEL facility.

The calculation results were compared to the experimental data. In general, the TRACE calculation agreed reasonably well with experimental data. The main events of the transient, that is, the decrease of the core water level, the depressurization of the primary circuit, and the water seal formation and clearing in the loop seal were predicted satisfactorily.

The core peak cladding temperature before the LSC was relatively well predicted but the second peak before the HPIS injection was not occurred because the core water level recovery by LSC and the water inventory in the core, lower plenum, and downcomer section was overestimated in the calculation. It is possible that extra water in the core was gathering from different places of the primary system, like loop seals, tubes and plenums of steam generators, or the pressurizer surge line. Based on the available measurements, it was difficult to say exactly how the water was actually distributed in the primary side during the experiment.

The calculated pressure and temperature on the secondary side of the steam generators after LSC were also substantially overestimated compared to measured values. The reason for this discrepancy could be in the modeling of the secondary side heat losses and massive steel structures, for example flanges of the secondary side, which served as a heat storage.

The behavior of the loop seals seemed to be relatively similar to the experiment as one loop seal cleared out and another refilled. In the calculation, the loop seal 1 cleared and the loop seal 2 refilled, while in the experiment the behavior was opposite.

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NRC FORM 335 (12-2010) NRCMD 3.7 — U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
BIBLIOGRAPHIC DATA SHEET *(See instructions on the reverse)*	**NUREG-IA-0492**

2. TITLE AND SUBTITLE

Assessment of TRACE V5.0 Patch 4 Code Against PWR PACTEL Loop Seal Clearing Experiments

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	2018

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Otso-Pekka Kauppinen

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.)

Lappeenranta University of Technology
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FI-53851 Lappeenranta, Finland

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The TRAC/RELAP advanced computational engine (TRACE) developed by U.S. Nuclear Regulatory Commission is one of the main system codes to perform thermal-hydraulic safety analyses of loss-of-coolant accidents, operational transients, and other accident scenarios of light water reactors. This report presents the TRACE calculation model of the PWR PACTEL facility and the calculation results of PWR PACTEL loop seal clearing experiment LSC-03 with TRACE V5.0 patch 4. The PWR PACTEL facility is designed and constructed in 2009 at Lappeenranta University of Technology and used in the safety studies related to thermal-hydraulics of pressurized water reactors with European pressurized water type vertical U-tube steam generators. The TRACE calculation results were compared to the experimental data. In general the results agreed reasonably well with experimental data. Some discrepancies were found in core peak temperatures, water level predictions, and the pressure and temperature predictions on the secondary side of the steam generators after loop seal clearing. The behavior of the loop seals seemed to be relatively similar to the experiment as one loop seal cleared out and another refilled. However, in the calculation, the loop seal 1 cleared and the loop seal 2 refilled, while in the experiment the behavior was opposite.

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

Parallel Channel Test Loop (PACTEL)
Lappeenranta University of Technology (LUT)
Low Pressure Injection System (LPIS)
Loop Seal Clearing (LSC)
European Pressurized Reactor (EPR)
Counter-Current Flow Limitation (CCFL)
Stability Enhancing Two-Step Numerical Method (SETS)

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMSSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

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NUREG/IA-0492

Assessment of TRACE V5.0 Patch 4 Code Against
PWR PACTEL Loop Seal Clearing Experiments

December 2018