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International Agreement Report

PWR PACTEL Small Break LOCA Experiment SBL-50 Calculation with TRACE Code

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ABSTRACT

TRACE is one of the main codes used for performing nuclear power plant thermal hydraulic safety analysis at present. Therefore, the importance of assessing the TRACE code capability to predict various thermal hydraulic transients in reactor systems becomes evident. One such transient that can occur small break loss-of-coolant-accident. The natural circulation is of particular interest for code assessment, as it requires the system code accurately predict temperature and density distributions throughout the system. Specific modeling capabilities are required for heat transfer and two-phase flow phenomena.

This research presents the assessment of the PWR PACTEL small break LOCA experiment SBL-50 with the TRACE V5.0 Patch 4. The PWR PACTEL facility is a modified version of the original PACTEL facility utilizing some parts of the original facility but also including completely new parts, i.e. loops and vertical steam generators (SG). The research focus with PWR PACTEL is set on the loop and vertical steam generator behavior in natural circulation conditions during small break LOCA event.

TRACE code was able to reproduce natural circulation phenomenon and small break LOCA conditions rather well. However, some discrepancies between the predicted variables and the experimental data suggests that further investigation of the TRACE modeling is necessary.

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EXECUTIVE SUMMARY

The present work introduces the assessment of the PWR PACTEL test facility small break LOCA experiment SBL-50 with the TRACE V5.0 Patch 4. The PWR PACTEL facility is 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 at LUT as a helping tool for experiment planning and result analyzing. TRACE is one of the main codes used for performing nuclear power plant thermal-hydraulic safety analysis at present. Therefore, the importance of assessing the TRACE code capability to predict various thermal-hydraulic transients in reactor systems becomes evident. One such transient that can occur is small break loss-of-coolant-accident. The natural circulation is of particular interest for code assessment, as it requires the system code accurately predict temperature and density distributions throughout the system. Specific modeling capabilities are required for heat transfer and two-phase flow phenomena.

Overall matching of the main parameters like primary and secondary pressures along with temperatures was rather good. However, some discrepancies between the predicted variables and the experimental data suggests that further investigation of the TRACE modeling is necessary.

ACKNOWLEDGMENTS

The author acknowledges the support from The Finnish Research Programmes on Nuclear Power Plant Safety (SAFIR2010 and SAFIR2014) associated with CAMP program.

ABBREVIATIONS AND ACRONYMS

DC	Downcomer
ECCS	Emergency Core Cooling Systems
HA	Hydro-accumulator
HPIS	High Pressure Injection System
LP	Lower Plenum
LPIS	Low Pressure Injection System
PACTEL	Parallel Channel Test Loop
PCP	Primary Coolant Pump
PRZ	Pressurizer
PWR	Pressurized Water Reactor
SBLOCA	Small Break Loss Of Coolant Accident
SG	Steam Generator
SNAP	Symbolic Nuclear Analysis Package
TRACE	TRAC/RELAP Advanced Computational Engine
UP	Upper Plenum
US NRC	United States Nuclear Regulatory Commission

1 INTRODUCTION

This report focuses on the validation of a new thermal hydraulic system analysis code TRACE for PWR PACTEL integral test facility, which has been designed and constructed in 2009 by the Nuclear Safety Research Unit at Lappeenranta University of Technology. The PWR PACTELfacility 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 original PACTEL facility is an out-of-pile integrated experiment facility, one of the largest facilities of its kind. It was originally designed to model the thermal hydraulic behavior of a VVER-440 type pressurized water reactor (PWR). The PWR PACTEL facility is a modified version of the original PACTEL facility utilizing some parts of the original facility but also including completely new parts, i.e. loops and vertical steam generators (SG). The research focuses on different phenomena in the main circulation loops and vertical steam generators in particular.

The TRACE code has been developed in the United States by the U.S. Nuclear Regulatory Commission (NRC) for the generic thermal hydraulic safety analysis of light water reactors (LWRs). The Finnish interest in TRACE stems from the authority requirement to maintain diverse safety analysis tools for the safety analysis of Finnish reactors. The report presents the results of the TRACE validation effort related to PWR PACTEL facility Small Break LOCA experiment SBL-50.

Thermal hydraulic modelling is always an optimization task: different modelling options have to be evaluated and decisions have to be made to reach the most applicable solution. As always in numerical modelling, the model accuracy is competing with the need to have reasonable computing times.

2 PWR PACTEL TEST FACILITY

2.1 Description of the PACTEL Test Facility

The PWR PACTEL facility consists of a reactor pressure vessel model, two loops with vertical steam generators, a pressurizer, and emergency core cooling systems (ECCS). The pressure vessel model comprises a U-tube construction modelling a downcomer, lower plenum, core, and upper plenum. A significant design and construction basis of the facility is the utilization of parts of the original PACTEL facility, i.e. the pressure vessel model, pressurizer, and ECCSs. Hence, those parts are not direct models of the reference EPR parts. Completely new parts, i.e. two loops and vertical steam generators of the EPR style construction, have been introduced to the PWR PACTEL architecture to enable the fulfilling of the specific facility research purpose. Figure 1 shows a general view of the PWR PACTEL facility.

As the PWR PACTEL is a modified version of the PACTEL facility, the pressurizer, pressure vessel parts and ECCSs are maintained as if they were constructed in the original PACTEL facility /1/, /4/. The principal difference between the constructions of the original PACTEL and PWR PACTEL facilities is in the loop and steam generator design. The PWR PACTEL facility consists of the two loops both containing a vertical steam generator (Figure 1).

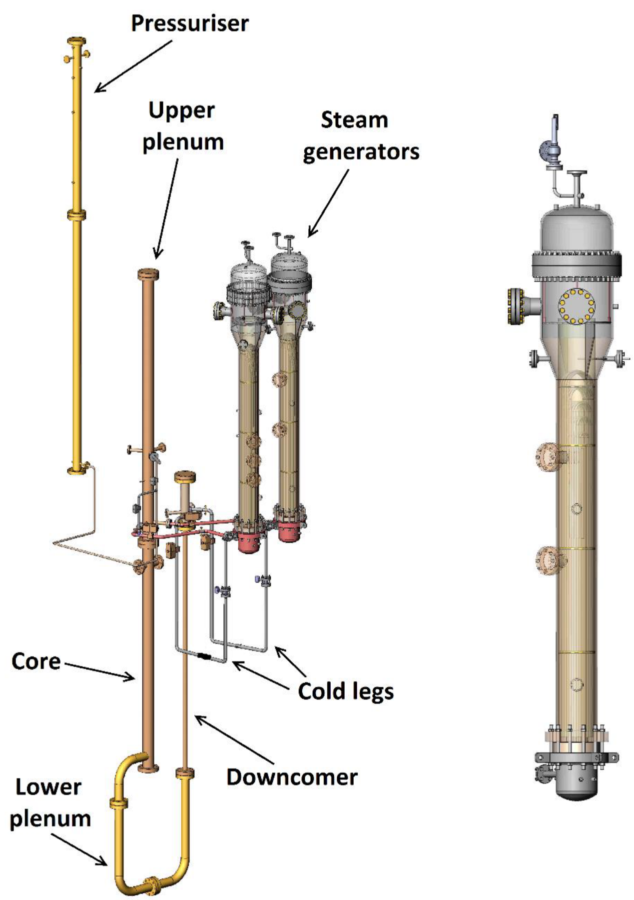

Figure 1 Schematic View of the PWR PACTEL Facility

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)

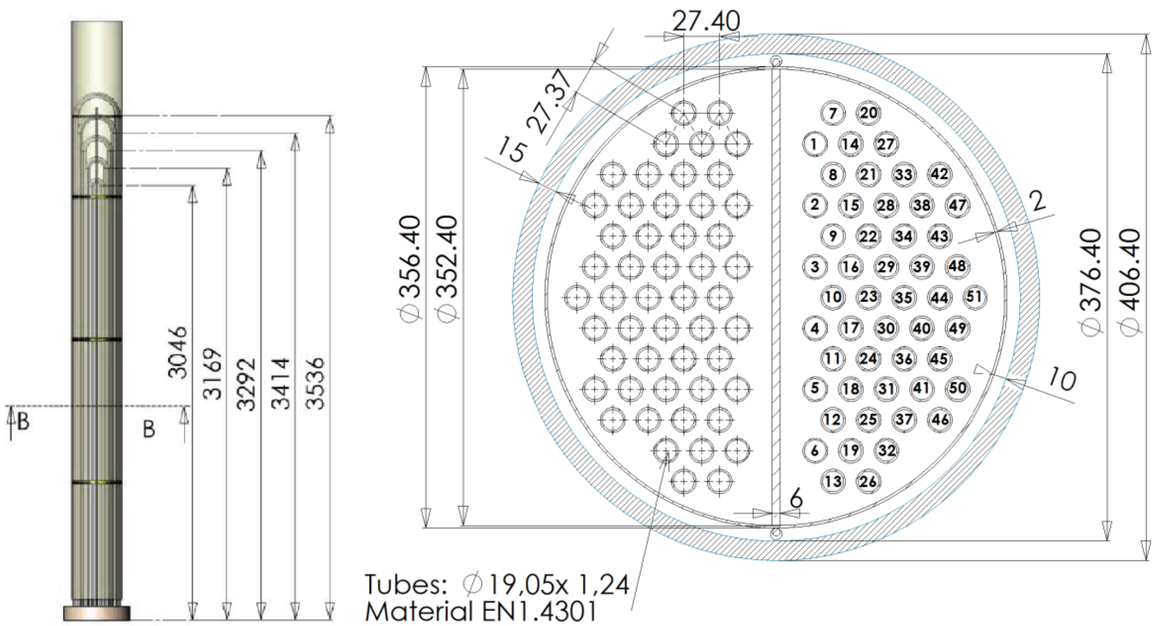

Figure 2 Heat Exchange Tubes of the PWR PACTEL Steam Generators

2.2 PWR Pactel Experiment SBL-50

The SBL-50 experiment was focused on behavior of natural circulation flow during small break loss of coolant accident. Both loops of the PWR PACTEL facility were in use. There were no main circulation pumps attached, when the experiment was performed; hence, all the flows were induced by natural circulation. In the beginning of the experiment, the primary side flows were single-phase natural circulation. The initial primary and secondary side pressures were about 7.5 MPa and 4.2 MPa, respectively. The core power set point was 155 kW. The secondary side inventory was held as constant as possible during each experiment. A steady-state period of 1000 s was recorded before the transient phase began. The initial conditions of the SBL-50 experiment before the opening of the break are presented in Table 2.

The break was located at the loop 2 between cold leg and the downcomer. A sharp-edged orifice (1 mm diameter) simulated the break. The flow area of the orifice in this experiment corresponded to 0.04 % of the cold leg cross-sectional area. The transient was initiated by opening the blowdown valve downstream of the break orifice at time 1000 s. At the same time the pressurizer was isolated from the rest of the primary system. No specific operator actions were introduced for the primary side during the experiment. However, the break valve was closed at about 53 % inventory because of a malfunction in the valve control system. This was discovered and rectified about 20 minutes later. Therefore, there was also an unintentional quasi steady state period in the experiment. The total recorded duration of the experiment was 10640 s. It was ended after the core started to dry out and thus temperatures in the core began to rise substantially.

Table 2 Initial Conditions of the SBL-50 Experiment at Time 0 s

Parameter	
Core power [kW]	155 ± 6
Primary side pressure [MPa]	7.5 ± 0.1
Secondary side pressure [MPa]	4.2 ± 0.06
Mass flow rate Loop 1 & Loop 2 [kg/s]	0.6 ± 0.14
Core inlet temperature [ºC]	252
Core outlet temperature [ºC]	276
Steam generator collapsed level SG1 & SG2 [m]	3.9 ± 0.12

3 TRACE INPUT MODEL OF THE PWR PACTEL FACILITY

The TRACE model of the PWR PACTEL facility has been developed at Lappeenranta University of Technology. The TRACE code has been developed in the United States by the U.S. Nuclear Regulatory Commission (NRC) for the generic thermal-hydraulic safety analysis of light water reactors (LWRs). The Finnish interest in TRACE stems from the authority requirement to maintain diverse safety analysis tools for the safety analysis of Finnish reactors. The model was constructed from scratch with the aim to cover finally all the main parts of the primary and secondary sides of the facility. New versions of the TRACE code have been adopted as they have become available. The latest version in use has been TRACE 5.0 patch 4. The TRACE modelling was conducted using Symbolic Nuclear Analysis Package (SNAP). The model editor and animation tool of the SNAP applications were used to help in the TRACE model preparation.

Since the PWR PACTEL is a modification of the PACTEL facility, the common parts such as the pressure vessel model, pressurizer, and ECCSs were maintained the same as with the original PACTEL facility model [4]. The computational model of PWR PACTEL facility is represented as an aggregate of TRACE code elements conceptually repeating the construction of real facility. The model contains such major elements of the primary side as reactor core, upper plenum, pressurizer, hot legs, heat exchange tubes, cold legs, downcomer, lower plenum, and elements of the secondary side: feed water input, secondary side downcomer, risers, and steam dome of the steam generators.

3.1 Primary Side

In the designed TRACE model of the PWR PACTEL facility, all of the elevation points of different parts coincide with those of experimental construction. The highest elevation point of the facility is considered to be at 16.64 m, which is the top point of the pressurizer as in the facility, thus in the computational model. As it was mentioned before, there are some simplifications had to be made, in order to streamline computing. Thus, in the primary side, the most perceptible simplifications refer to the core and heat exchange U-tubes.

Regarding to the core of the TRACE model, it is represented as the four-pipe assembly. Three of those pipe elements represent core heat section each. They are marked with green color filling, as it shown on the Figure 3. The same heat element structure belongs to the pressurizer. In such a manner, the first three nodes of the pressurizer pipe component are filled with green color, which means that they are heat elements. Each node is responsible for one out of three groups of the pressurizer's heating elements.

The steam generator heat exchange tubes are set in the triangular pitch. There is length difference between the different tubes. Formally, to be accurate, the tubes need to be divided in ten groups, because of their length. However, in computational model there are only five groups. As long as TRACE code does not allow round shapes, there are compromises were found. Thus, U-shape bends in each tube are made by turning nodes on 45 degrees angle. This modelling step is made by founding compromises between such three points as total length, length of U-shape bend and tube ceiling dimensioning.

The upper plenum has a “pseudo-annular” channel, which was created between hot leg connection and upper plenum with pipe components. Upper plenum consist of two parallel flow channels, which connected to each other (Figure 3).

3.2 Secondary Side

The secondary side contains such main parts as feedwater inlet, downcomer, riser, steam dome and steam outlet (Figure 4). The feed water inlet system is modelled using a fill component. This component allows adjusting only a velocity inlet, and therefore mass flow is handled by level controller element. The connection between the fill component and downcomer is arranged by a pipe component. The downcomer is divided into hot and cold sections, and the connection of the feedwater pipe is implemented to the cold side. Cold side of the downcomer is contemplated for feedwater downwards flow. The hot downcomer is connected from the top to the steam dome and from the bottom to the hot riser. The hot downcomer is brings down the condensation water through holes from the steam dome. Accordingly, the cold downcomer is joined to the feedwater inlet system on the top and to the cold riser at the bottom. Riser parts are also implemented as pipe components. There are five riser parts: hot and cold risers, one common element with heat exchange tubes, and one riser element with the top points of U-shape bands of tubes and on riser element without tubes.

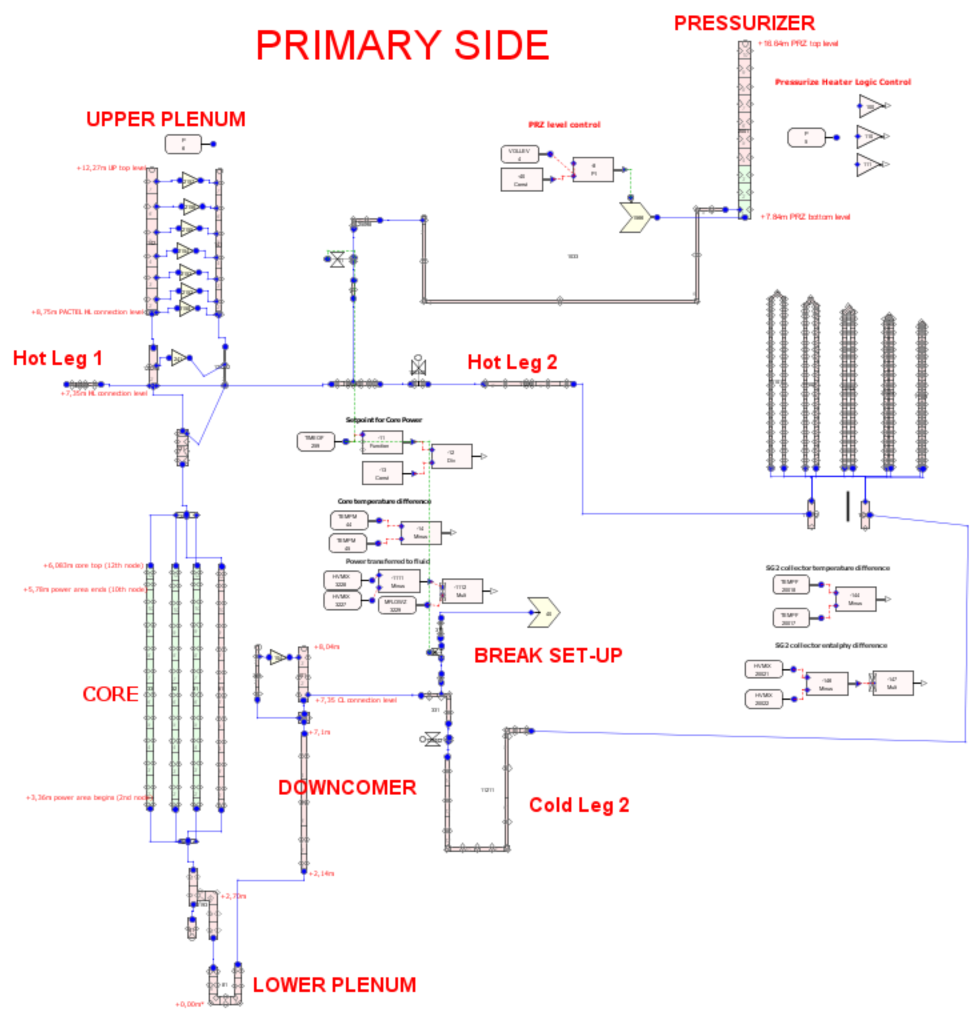

Figure 3 Primary Side of the PWR PACTEL TRACE/SNAP Model

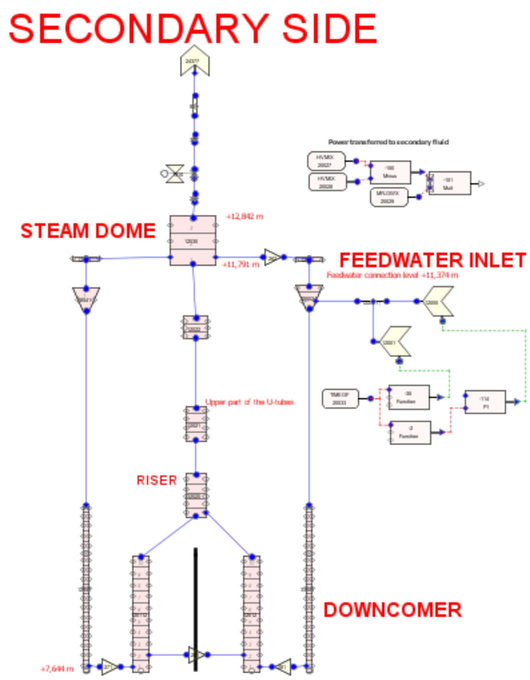

Figure 4 Secondary Side of the PWR PACTEL TRACE/SNAP Model

3.3 Pressure Loss Definition

The pressure losses were defined separately for the different parts of the full TRACE model. At this phase, the model nodalization was rechecked to correspond to the locations of the pressure difference measurement taps. As stated in the staggered grid method, the pressure is implemented into the center elevation of the node. The exact match of the locations of the measurement in the facility and in the calculation model was not possible in all cases since the node length would have become too short and caused time step problems. In most cases, the correspondence of the locations in the facility and in the model was accurate.

3.4 Heat Loss Definition

The heat losses for the TRACE model of the PWR PACTEL facility were defined in steady-state conditions using different parameters at nominal conditions, and at lower pressure and temperature conditions. The main varied parameter used for the adjustments was the thermal conductivity of the insulation material. Several user-defined materials were created to set the heat loss distribution in detail.

4 TRACE CALCULATION RESULTS OF THE EXPERIMENT SBL-50

To reach steady state conditions comparable to the experiment the calculation was started with pre-transient steady state period of 5000 s. The time was then reset and the duration of the actual transient simulation was set equal to the experiment time. The actual experiment period was started with 1000 s period resembling the experiment steady state. Before the initiation of the blowdown initiation, the parameters in the calculation model and in the experiment were close to each other.

The break flow adjustment was carried out by testing different additive loss factors in the single junction component simulating the break orifice. The break mass flow rate in the calculation was almost perfectly matched to the experiment data. Cumulative break mass verifies this observation (Figure 5). An unexpected event took place in the experiment when a malfunction in the break valve control closed the valve at time 7700 s. The break valve remained closed for 1260 s. Then the break valve was opened again and the transient proceeded as planned until the end. The stagnation in the break flow was set also to the TRACE modelling.

The main features of the experiment were also found in the calculation results. Figures 5 - 11 present the comparison of representative experiment and calculation results. The period of the presented results is from 0 to 11000 seconds.

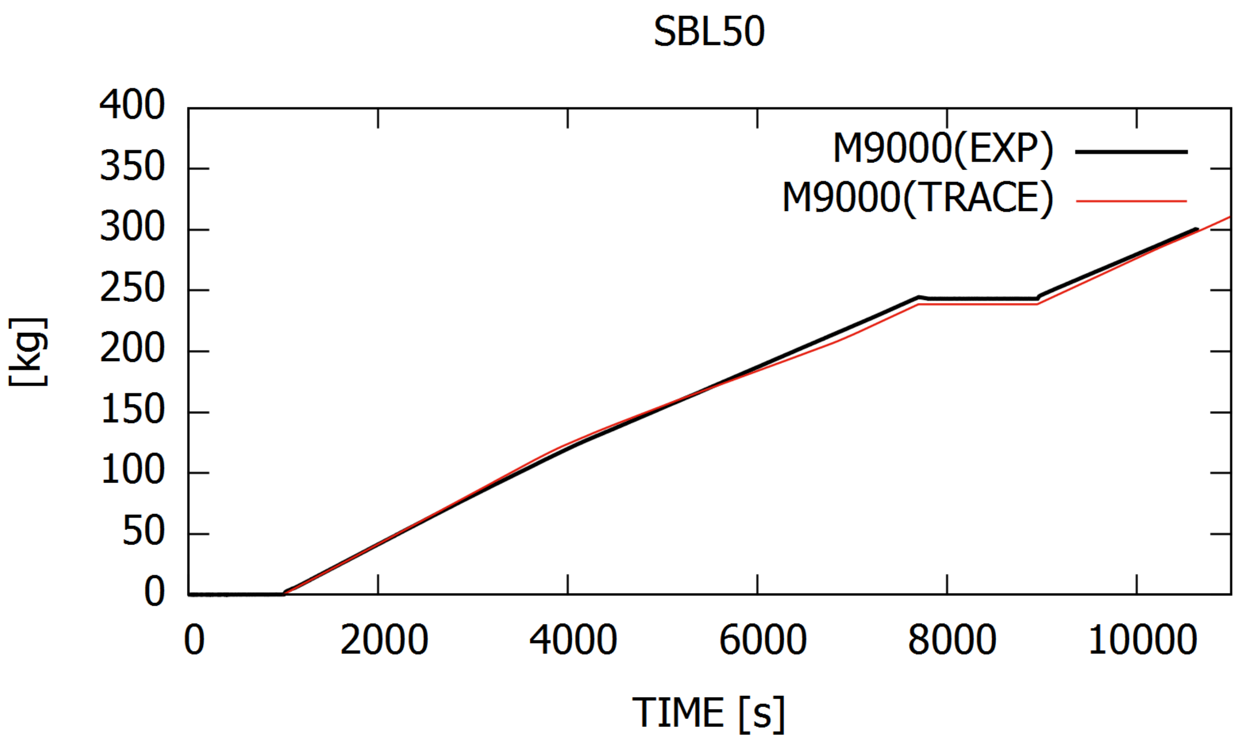

Figure 5 **Cumulative Break Flow in the PWR PACTEL Experiment SBL-50 vs. TRACE Calculation**

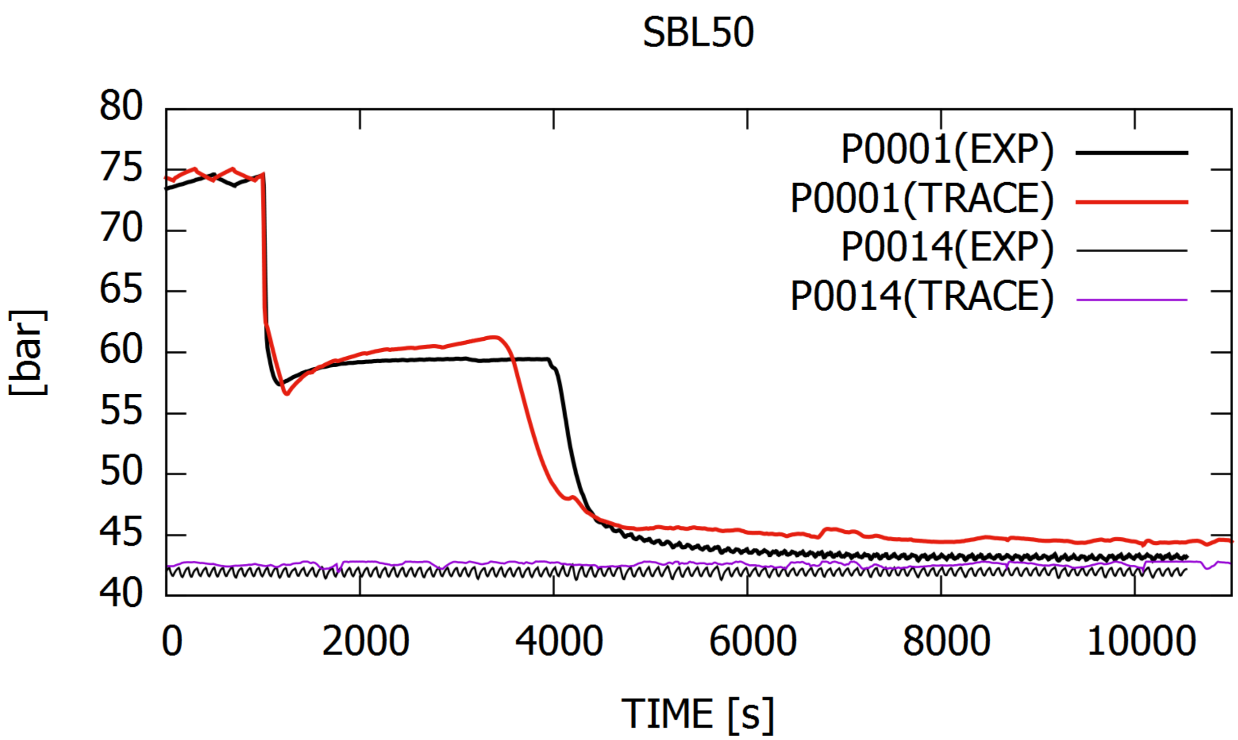

Figure 6 Primary and Secondary Side Pressures in the PACTEL Experiment (Exp) SBL-50 vs. TRACE Calculation

The calculated primary side pressure followed accurately the experiment value during the rapid depressurization and single-phase natural circulation period (Figure 6). In addition, the periods, when the primary pressure started to rise and then fall down again due to natural circulation flow deterioration and hot leg loop seal clearance, were accurately calculated. From time 3700 s onwards, when the two-phase natural circulation began, some discrepancies appeared and as a result, the calculation slightly overestimated the primary side pressure and temperature until the end of the simulation. The calculated primary inventory reduction was similar with the experiment. The collapsed level of the upper plenum was calculated satisfactorily (Figure 7) although the calculated level started to decrease earlier and was faster than in the experiment. The core inlet and outlet temperatures followed the primary pressure behavior (Figure 8). Thus, inconsistencies appeared according to pressure behavior.

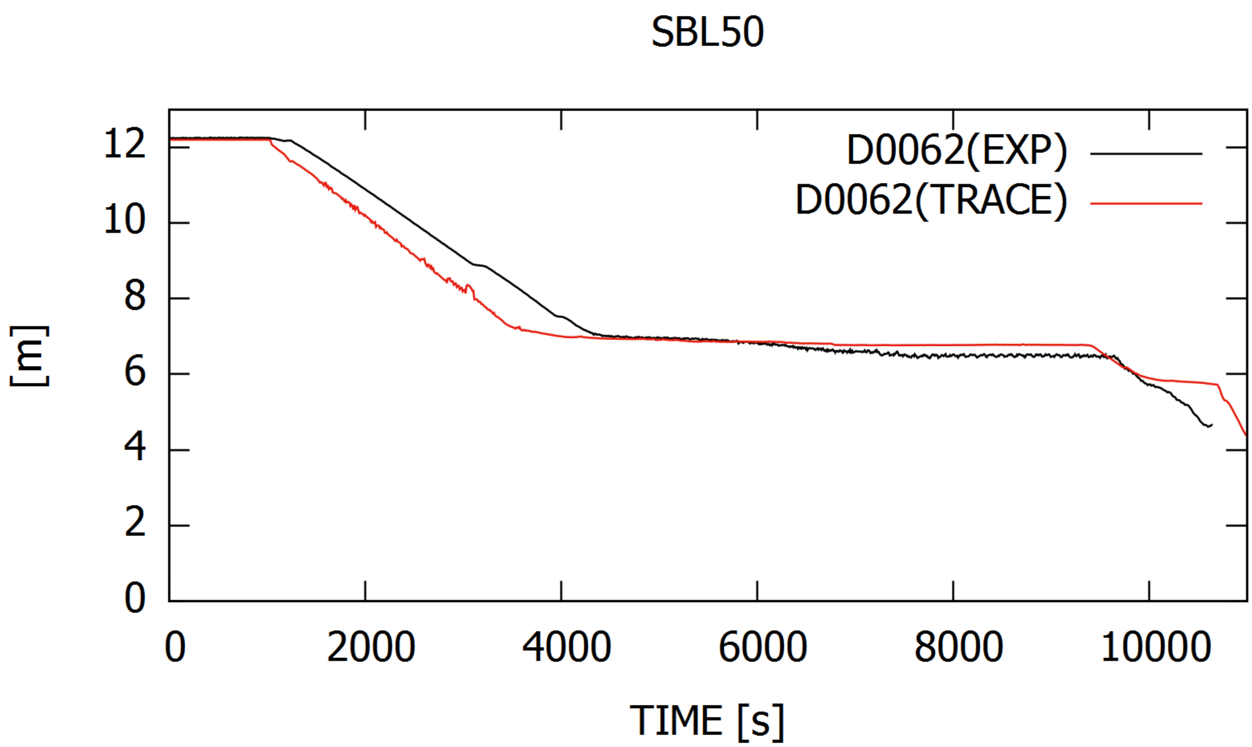

Figure 7 Upper Plenum Collapsed Level in the PWR PACTEL Experiment (Exp) SBL-50 vs. TRACE Calculation

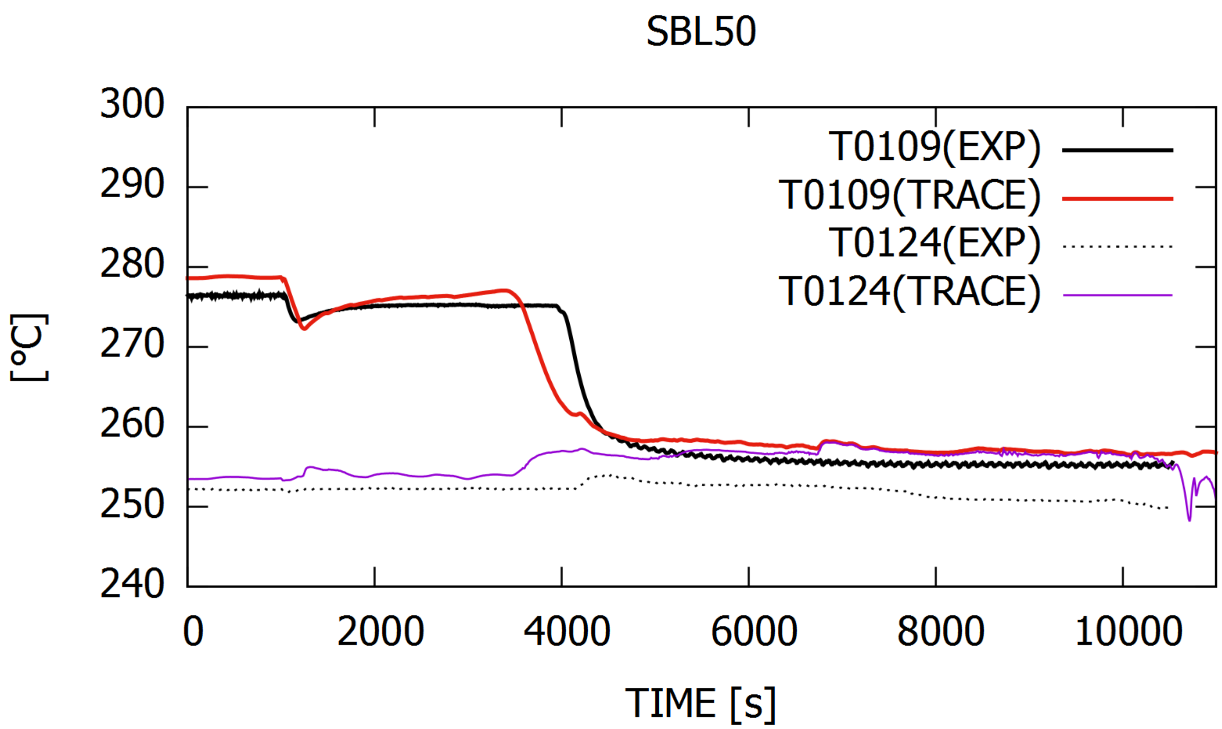

Figure 8 Core Inlet and Outlet Temperatures in the PWR PACTEL Experiment SBL-50 vs. TRACE Calculation

The initiation of a loop flow can be very sensitive to the appearance of small pressure or temperature differences and to the mass balance between water and steam. In addition, the reliability of measurements, when there is a possibility for the presence of two-phase flow, is lower than in a pure single-phase case. The combined mass flow rate at the downcomer resembled better the experiment result (Figure 11) but still remained lower than the measured value.

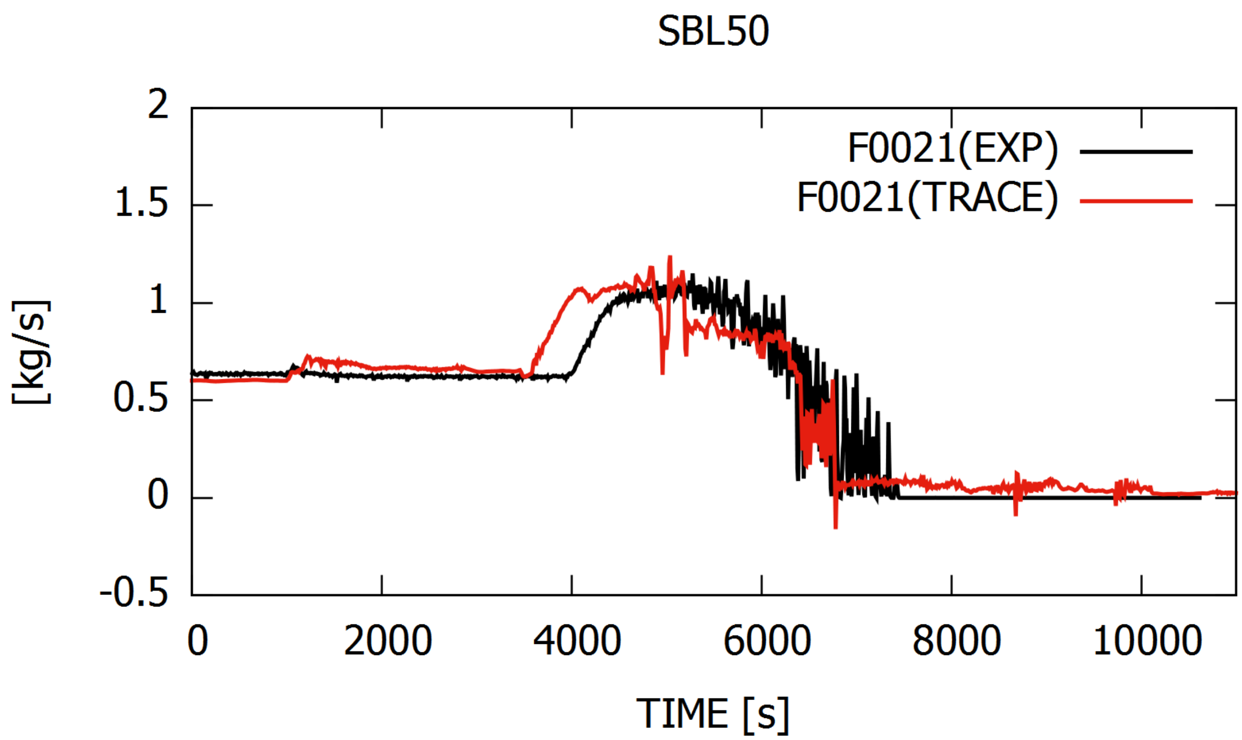

Figure 9 Mass Flow Rate in Cold Leg 1 in the PWR PACTEL Experiment SBL-50 vs. TRACE Calculation

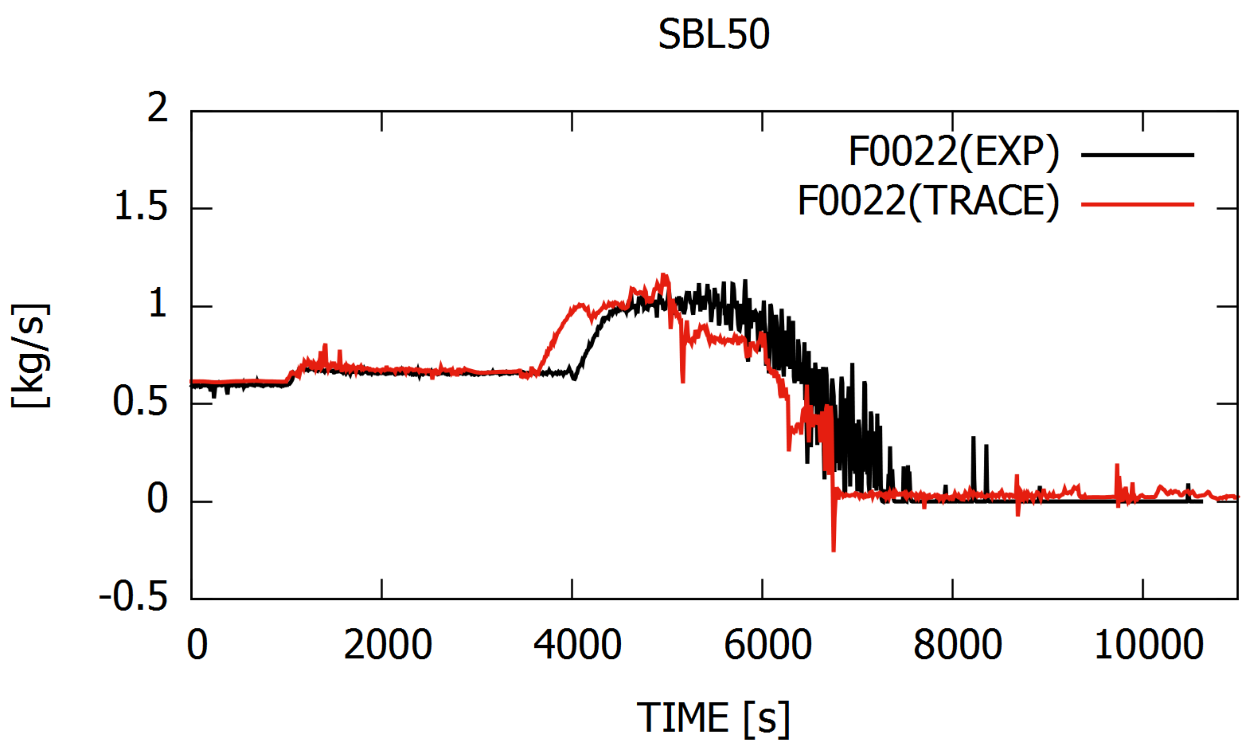

Figure 10 Mass Flow Rate in Cold Leg 2 in the PWR PACTEL Experiment SBL-50 vs. TRACE Calculation

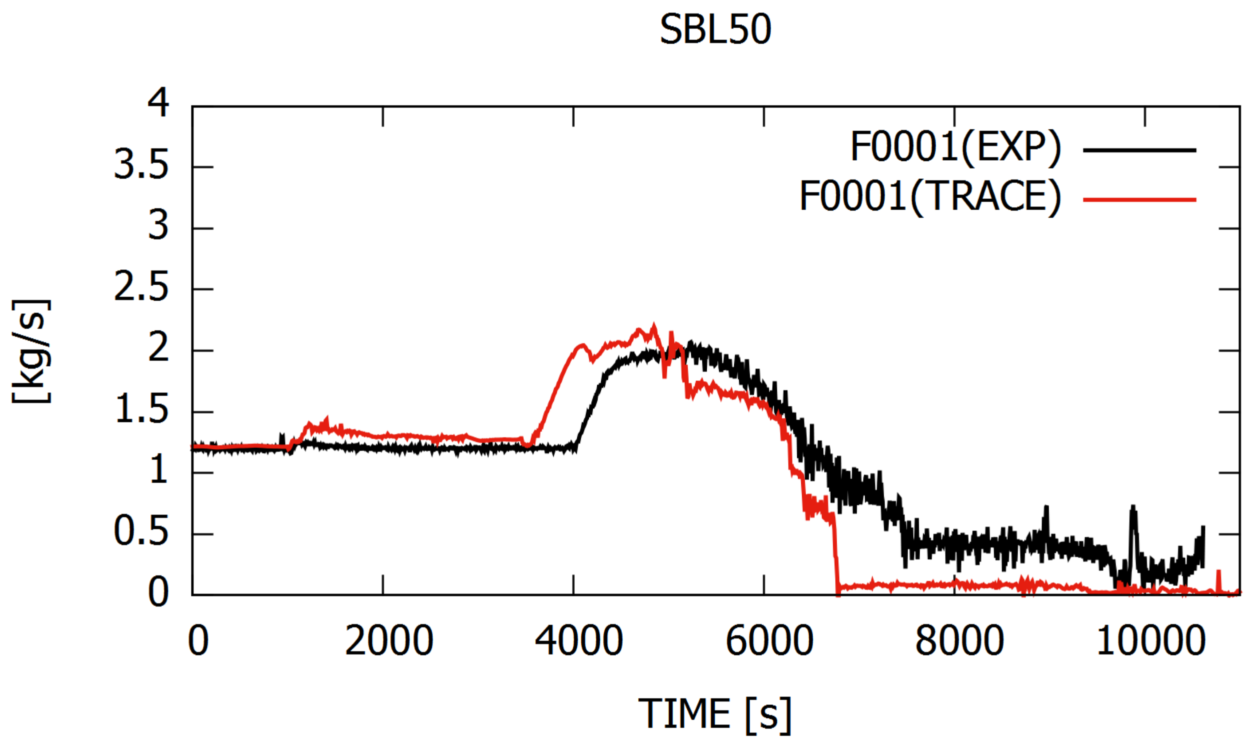

Figure 11 Mass Flow Rate in Downcomer in the PACTEL Experiment (Exp) SBL-50 vs. TRACE Calculation

5 RUN STATISTICS

The calculations were performed using Intel® Core™ i7-6820HQ CPU @ 2.70 GHz processor. The operating system is Windows 7 Enterprise.

Table shows the run statistics for the codes TRACE Patch 4.

Table 3 Run Statistics

Code	Transient Time (s)	CPU Time (s)	CPU/Transient Time	Number of Time Steps
TRACE Patch 4	16000	5869	0.3668	225118

6 CONCLUSIONS

The PWR PACTEL facility small break loss-of-coolant-accident experiment, SBL-50, was calculated using TRACE V5.0 Patch 4. The calculation results were compared to the experimental data. The TRACE calculations agreed satisfactorily with experimental data.

A full TRACE code thermal hydraulic simulation model was prepared for the PWR PACTEL facility. The model was partly originated from the earlier PACTEL facility model of VVER-440 type nuclear plant. The PWR PACTEL experiment SBL-50 was calculated using the TRACE model. In the SBL-50 experiment, a 1 mm break was introduced and the primary inventory was let to decrease until the cladding temperatures started to rise. Modeling of the break flow succeeded quite well also during the difficult two-phase flow period. Even though the mass balance was modeled successfully, the primary pressure and level behavior showed discrepancy in timing between calculation and experiment. This apparently led in earlier start of the two-phase natural circulation period in the calculation model. Despite of the timing discrepancies between the calculation and experiment results the overall tendency with several stagnations and resumes of natural circulation flow agreed well with the experiment.

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10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

TRACE is one of the main codes used for performing nuclear power plant thermal hydraulic safety analysis at present. Therefore, the importance of assessing the TRACE code capability to predict various thermal hydraulic transients in reactor systems becomes evident. One such transient that can occur small break loss-of-coolant-accident. The natural circulation is of particular interest for code assessment, as it requires the system code accurately predict temperature and density distributions throughout the system. Specific modeling capabilities are required for heat transfer and two-phase flow phenomena. This research presents the assessment of the PWR PACTEL small break LOCA experiment SBL-50 with the TRACE V5.0 Patch 4. The PWR PACTEL facility is a modified version of the original PACTEL facility utilizing some parts of the original facility but also including completely new parts, i.e. loops and vertical steam generators (SG). The research focus with PWR PACTEL is set on the loop and vertical steam generator behavior in natural circulation conditions during small break LOCA event. TRACE code was able to reproduce natural circulation phenomenon and small break LOCA conditions rather well. However, some discrepancies between the predicted variables and the experimental data suggests that further investigation of the TRACE modeling is necessary.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
Parallel Channel Test Loop (PCTEL) Small Break Loss of Coolant Accident (SBLOCA) Hydro Accumulator (HA) Vertical Steam Generator (SG) Symbolic Nuclear Analysis Package (SNAP) Code Assessment and Validation	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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