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International Agreement Report

PACTEL Small Break LOCA Experiment SBL-30 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 to 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 PACTEL small break LOCA experiment SBL-30 with the TRACE V5.0 Patch 4. The PACTEL facility is volumetrically scaled full-height model of a six-loop Russian design VVER-440 PWR. This reactor type has specific features like horizontal steam generators and hot leg loop seals. Although the TRACE code has not been originally developed for the special geometry of the VVER-440 reactor type, it was proven that the code is capable for relatively accurate reproducing the natural circulation phenomena at a satisfactory level.

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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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 to accurately predict temperature and density distributions throughout the system. Specific modeling capabilities are required for heat transfer and two-phase flow phenomena.

The present work introduces the assessment of the PACTEL small break LOCA experiment SBL-30 with the TRACE V5.0 Patch 4. The PACTEL facility is volumetrically scaled full-height model of a six-loop Russian design VVER-440 PWR. This reactor type has specific features like horizontal steam generators and hot leg loop seals. Although the TRACE code has not been originally developed for the special geometry of the VVER-440 reactor type, it was proven that the code is capable for reproduce the natural circulation phenomena satisfyingly.

However, some discrepancies between the predicted variables and the experimental data suggests that further investigation of the TRACE modeling is necessary.

ACKNOWLEDGMENTS

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ABBREVIATIONS

ACCS	Accumulator Core Cooling System
DC	Downcomer
ECCS	Emergency Core Cooling Systems
FLSG	Full Length Steam Generator
HA	Hydro-accumulator
HPIS	High Pressure Injection System
LDSG	Large Diameter Steam Generator
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
VVER	Vodo Vodjanyi Energetitseskij Reaktor

1 INTRODUCTION

This report focuses on the validation of a new thermal hydraulic system analysis code TRACE for VVER-440 reactor applications. 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). TRACE includes six equation two phase flow basically in one-dimensional form, but for certain component, such as vessel, is three-dimensional formalism applied.

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 for VVER-440 reactor related experiment with PACTEL facility. The TRACE code has not been originally developed for the special geometry of the VVER-440 reactor type. 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 PACTEL TEST FACILITY

2.1 <u>Description of the PACTEL Test Facility</u>

The PACTEL facility (see Fig. 2.1) is a 1:305 volumetrically scaled, out-of-pile, full-height model of a six-loop Russian design VVER-440 PWR (Ref. 1, Ref. 2). Volumetric scaling expects that all components volumes have to be equally proportional to the reference volumes of PWR. Scaling factor for component heights and elevations is 1:1. The facility consists of three primary loops, which are nearly symmetric and have identical volume. Each loop equals to two loops in the reference VVER-440. The loop is composed of one horizontal Steam Generator (SG) with 118 heat exchange U-tubes with inner diameter equal to 13 mm, one Primary Coolant Pump (PCP) and two loop seals, in the hot and cold legs. The inner diameter of the legs is 52.5 mm each. (Ref. 3). A simplified schematic of PACTEL test facility is presented in **Figure 1**.

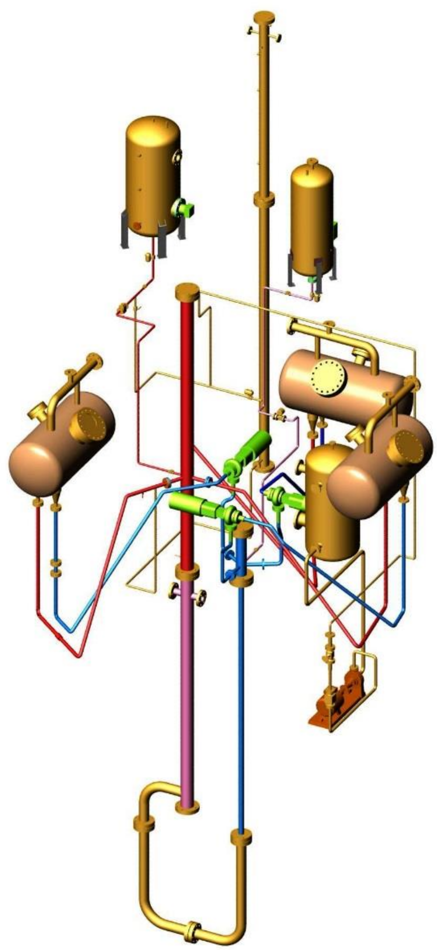

Figure 1 Schematic View of the PACTEL Facility

The PACTEL core geometry is identical to the reference reactor. The core consists of 144 electrical heater rods ordered in three parallel channels in triangular grid. The rod diameter is 9.1 mm, lattice pitch is 12.2 mm and heating length is 2420 mm, which is the same as in VVER 440 hexagonal bundle fuel rods. The amount and construction of the rod spacers are the same as in the reference reactor. The maximum heating power is 1 MW obtained from electric supply. This is approximately 22% of the scaled thermal power of VVER 440 (1375 MW) (Ref. 3).

A U-tube construction of the PACTEL consists of upper plenum (UP), lower plenum (LP), core and downcomer (DC). Volumes and elevations of components was determined in accordance with general scaling factors. The UP is composed of one tube with three connections for hot legs. Diffusers in hot and cold connections limit direct flow of ECCS water from the ACCS and HPIS to the loops (Ref. 3).

Secondary side of the steam generators has a common steam line connecting the three steam generators. After this common steam line the steam is released to the atmosphere. There are separate feed water injection systems for all three steam generators in the secondary side. Above the heat exchange tube bundle and in the middle of the bundle in each steam generator there are two separate feed water lines. It is possible to control the pressure in all steam generators with a separate PI controller. One of the suitable means to control the pressure is to use controller relating to the common steam line. There is possibility to control pressure and feed water injection in each steam generator separately, due to this feature experiments with asymmetric secondary side behavior can be implemented (Ref. 1).

The secondary-side volume in the PACTEL steam generators is larger than in the reference VVER-400 because the distance between steam generator tube rows is doubled. Thus, the water volume of the secondary side of one steam generator in the PACTEL facility equals to three volumetrically scaled volume of the secondary side of two reference steam generators (Ref. 1).

2.2 Pactel Experiment SBL-30

In the SBL-30 experiment, all three loops of the PACTEL facility were in use. The SBL-30 focused on the behaviour of new Large Diameter Steam Generator (LDSG), and it was a comparison experiment for SBL-7, which was carried out earlier with the Full Length Steam Generators (FLSG) (Ref. 3). The main circulation pumps were not running during the whole recording period of the SBL-30 experiment; 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 7.4 MPa and 4.2 MPa, respectively. The core power set-point was 160 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-30 experiment before the opening of the break are presented in **Table 1**.

The break was located vertically at the bottom of the loop 2 cold leg near 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 PACTEL cold leg cross-sectional area. Due to the scaling method used, this break size corresponds to 0.1 % in the reference reactor. The transient was initiated by opening the blowdown valve downstream of the break orifice at time 1000 s. At the same time the pressurizer heaters were switched off. The pressurizer was disconnected from the rest of the primary system as the break was opened by closing an

Table 1 Initial Conditions of the SBL-30 Experiment Before the Break Opening

Parameter	
Primary pressure [MPa]	7.31
Secondary pressure [MPa]	4.19
Loop 1 / Loop 2 / Loop 3 [kg/s]	0.44 / 0.43 / 0.46
SG1 / SG2 / SG3 feed water flow [l/min]	1.97 / 0 / 1.97
Core inlet temperature [ºC]	257
Core outlet temperature [ºC]	269
Pressurizer level [m]	5.2
SG1 / SG2 / SG3 level [cm]	69.2 / 79.1 / 78.3

isolation valve in the pressurizer line. All the natural circulation modes, single-phase, two-phase and boiler-condenser modes were observed. Due to hot leg loop seals, two-phase and boiler-condenser modes were intermittent. The PACTEL operators terminated the experiment, when the primary circuit liquid inventory had depleted to the point where the core outlet temperatures started to rise. The timing of the main events are presented in **Table 2**. The operators controlled manually the feedwater flow to the steam generators. The purpose was to keep the collapsed level constant at the set point of 75 cm. Therefore the control method was an on/off procedure.

Table 2 Main Events in SBL-30

Time, s	Event
1000	Blowdown initiated, PRZ isolated, PRZ heaters switched off
3360	Loop flows stagnated, primary pressure build up started
3665	Loop seals cleared, flows resumed
3350	Core power off
3430	Core power on
3470	Core power off
3640	Core power on
10170	Void at the top of the DC
11010	Break flow changed from single-phase to two-phase flow
12150	Core heat up first observed
12301	Cladding temp. exceeded 300 °C, experiment terminated

3 TRACE INPUT MODEL OF THE PACTEL FACILITY

The TRACE model of the 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. The modelling of the PACTEL facility with the TRACE code resembles the guidelines adopted in the RELAP5 modelling for PACTEL. New versions 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.

The construction of the PACTEL facility TRACE input model was started by creating the different parts of the facility and testing them separately. All main parts of PACTEL; the pressure vessel part, main circulation loops, steam generators and the pressurizer; were created and tested separately. The input creating procedure contains geometry definitions, the decision of nodalization formulation (density, location) and the degree of lumping needed.

The nodalization of the TRACE model was largely derived according to the guidelines approved for the RELAP5 input deck of the PACTEL facility (Ref. 4). In some cases, exceptions in the TRACE nodalization following the RELAP5 nodalization had to be made. For example, the different approach between RELAP5 and TRACE in the handling of bending pipelines causes a totally different nodalization. The basic rule in the nodalization is that it should be in balance between coarse and fine nodalization. The nodalization has to be fine enough to be able to describe the essential changes throughout the system and coarse enough to perform the simulation calculation within a reasonable time frame.

At first, a stand-alone horizontal steam generator model was prepared. In this phase, different modelling options were tested against the loss-of-feedwater (LOF-10) PACTEL experiment. The main parameter defining the applicability of the model was the propagation of the collapsed level on the steam generator secondary side. The main used modelling options were the number of layers of the heat exchange tubes (14 in the facility). The first attempt was implemented with four rows to model the heat exchanger tube bank. The second phase was to increase the row number to five. The calculation results indicated that the proper modelling of the tube packages requires eight layers at the minimum to bring out the level behaviour similar to the experiment result (Ref. 5).

The full model with all three loops was then constructed and modelled (Ref. 6). The functionality of the model was tested with calculations of pressure and heat loss experiments. During the development process of the model, it was modified and expanded due to the needs of different validation cases. The model was also updated step by step according to the recommendations of the code developers.

The upper plenum part of the PACTEL facility has a diffuser structure, which consists of two nested concentric pipes preventing the direct flow of the ECC water from the hydro-accumulators (HA) or from the HPI system to the loops. For the TRACE simulation model, a similar structure was built using two parallel pipes, which were connected together with single components. This modelling structure clearly improved the upper plenum behavior especially during events with decreasing level. The similar structure exists also on the downcomer side of the PACTEL facility. This structure was modelled to the input. The pressurizer was modelled using a standard pipe component. The core section was divided into three parallel lines, and the heat production in the core as well as in the pressurizer heaters was implemented with POWER components, which can be controlled with time dependent functions and trips. As the primary tube bank in each horizontal steam generator is divided into eight rows, the three top rows represent one tube row of the real PACTEL facility. Other tube rows represent two rows of the real facility. The secondary side is divided into the riser and downcomer sections. The model is shown in Figure 2 representing the layout of the main parts of the PACTEL facility containing Loop 1 (SNAP, Model Editor). The other two loops and auxiliary systems are drawn in separate windows.

In the SBLOCA experiments, the break flow was set to Cold leg 2 near the downcomer. The break orifice set-up was realized with a single junction component connected with a cross-flow junction to the cold leg. The break valve takes care of the break initiation and leads the flow to the break component, which acts as a pressure boundary (see Figure 3).

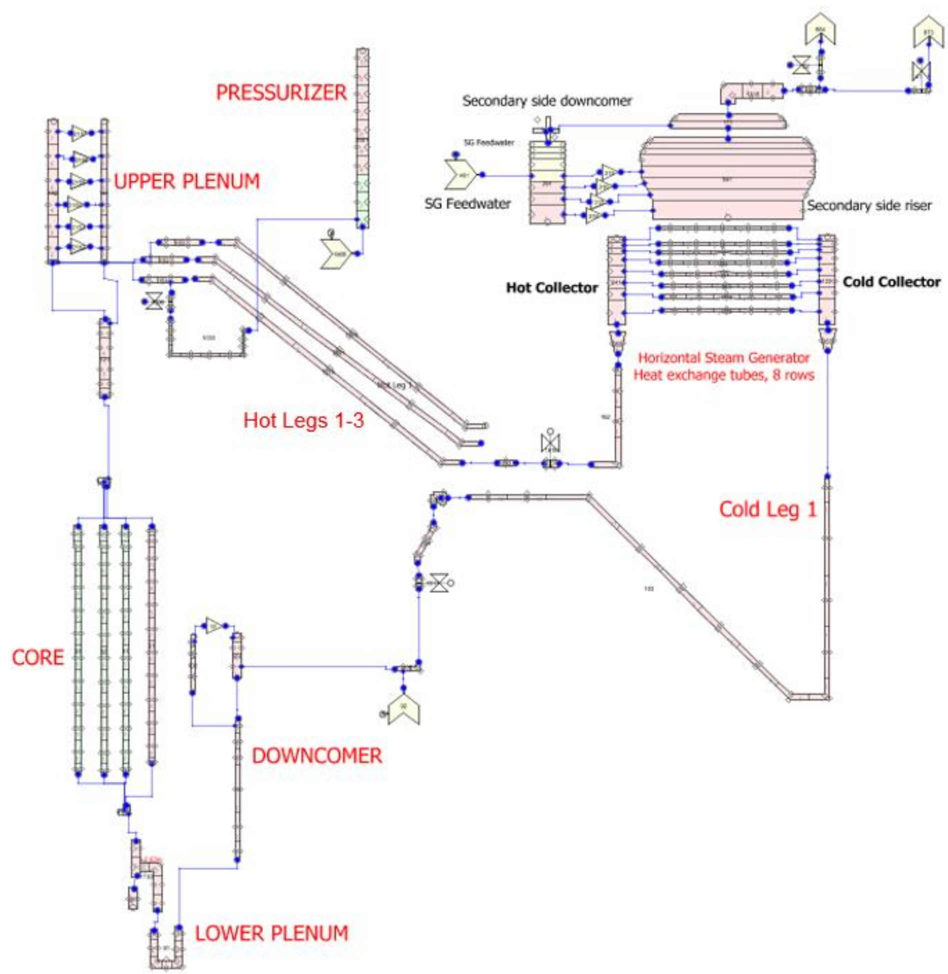

Figure 2 **TRACE/SNAP Model of PACTEL for Calculation of SBL-30-Experiment (Loop 1 totally presented, Loops 2-3, only first parts of hot legs)**

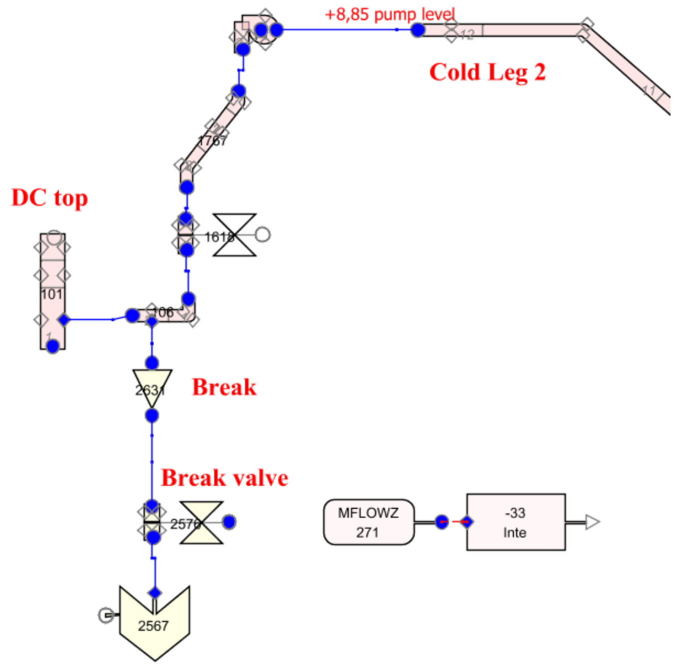

Figure 3 **TRACE/SNAP Model Set-Up for Break in SBL-30-Experiment. Break is Located at Cold Leg 2.**

3.1 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 centre 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.

3.2 Heat Loss Definition

The heat losses for the PACTEL facility were defined using the heat-up and cool-down method. The heat losses for the TRACE model of the PACTEL facility were defined using the cool-down method only. The heat-up procedure contained operator actions that could not be reproduced in the calculation without having very large uncertainties in the procedure. The pressurizer heat losses were defined with separate test calculations. The singular heat losses were adjusted according to the data from ISP-33, and the pump heat losses from a separate data. The overall heat losses were verified against the data from PACTEL experiment HL-22. 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. The heat losses of the primary circulation pumps are large, almost one third of the total heat losses at the nominal PACTEL conditions. The pumps are not insulated, and thus the casing material was used for heat loss adjustment.

4 TRACE CALCULATION RESULTS OF THE EXPERIMENT SBL-30

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 similar to the experiment data. This was verified using cumulative break mass (Figure 4). The break flow changed from single-phase to two-phase flow slightly earlier in the calculation than in the experiment (time ~11000 s). The main features of the experiment were found also from the calculation. Figures 4 - 11 present the comparison of representative experiment and calculation results. The time period of the presented results is from 0 to 12000 seconds. The calculation was continued until 12500 s, but no significant findings were met after 12000 s.

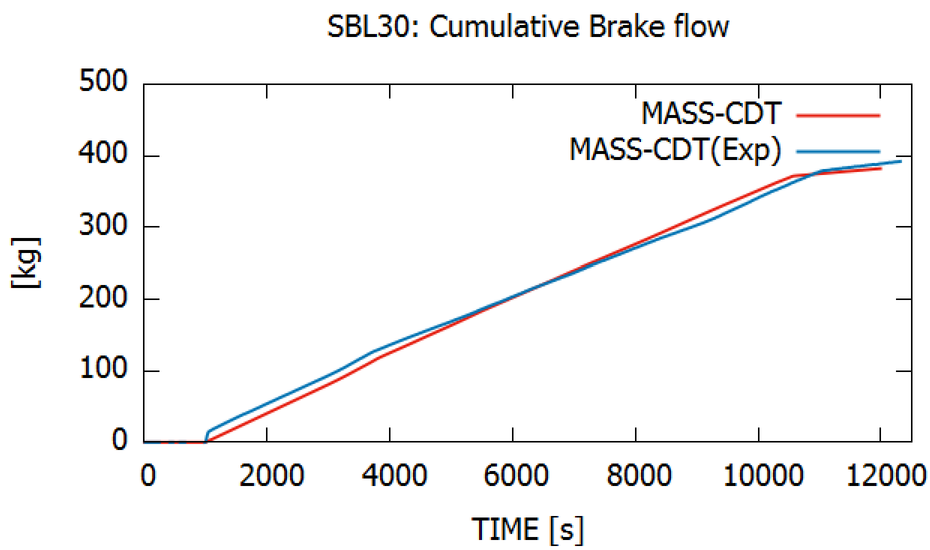

Figure 4 Cumulative Break Flow in the PACTEL Experiment (Exp) SBL-30 vs. TRACE Calculation

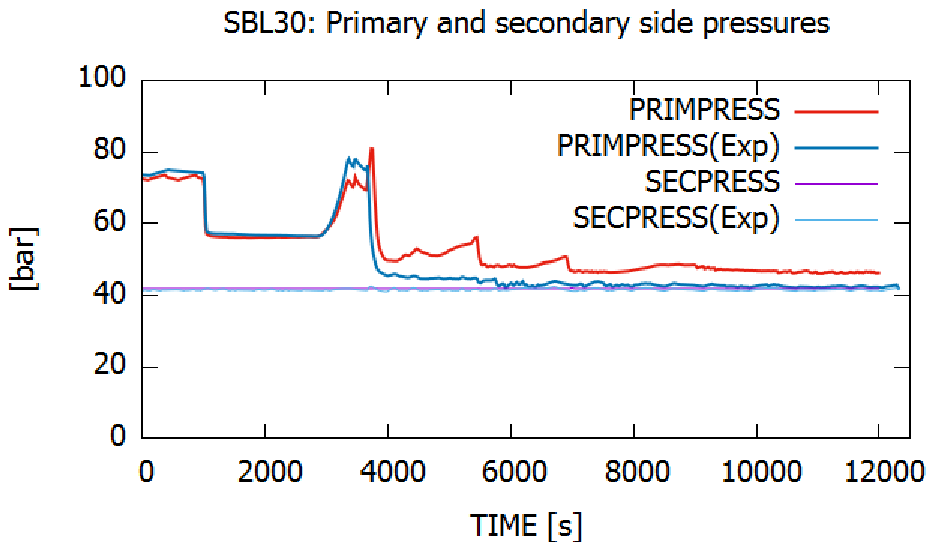

Figure 5 Primary and Secondary Side Pressures in the PACTEL Experiment (Exp) SBL-30 vs. TRACE Calculation

The calculated primary side pressure followed accurately the experiment value during the rapid depressurization and single-phase natural circulation period (Figure 5). Also 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 cumulative break flow in the calculation (Figure 4) followed the experiment data curve quite accurately and also the change from single- to two-phase flow was quite well-timed. The calculated primary inventory reduction was similar with the experiment. The collapsed level of the upper plenum as well as voiding of the downcomer top found in experiment was calculated satisfactorily (Figure 6). The core inlet and outlet temperatures followed the primary pressure behavior (Figure 7).

Due to VVER-440 geometry in the PACTEL facility the natural circulation flow is affected by the loop seal effect, which induces asymmetrical flow stagnations between different loops. These phenomena were also found in the SBL-30 experiment and in the calculation. The first flow stagnation appeared when the water inventory decreased to the level where the hot legs are connected to the upper plenum. Then steam could pass to the hot legs. The flow stagnated in all three loops and caused a rapid rise in primary pressure. The pressure started to decrease when the loop seals cleared and the flow resumed. The calculated mass flow rates in the loops did not match with the experiment values during the two-phase flow period from 3700 s to 8000 s (see Figures 8 - 10). However, the chaotic behavior of mass flow distribution between the loops during this time phase was repeated also in the calculation, even that e.g. the flow behavior in loops 2 and 3 seemed to have changed places with each other, i.e. the calculated flow in loop 3 resembled more the situation in loop 2 in the experiment. After this phase the calculation showed quite a good agreement with the experiment, when the cold legs started to run out of water after 8000 s and natural circulation changed to boiler-condenser mode.

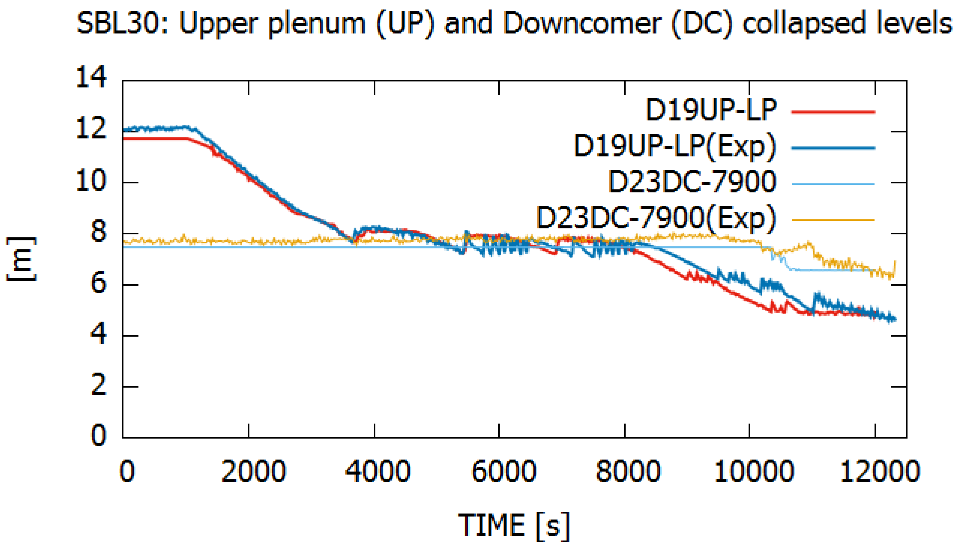

Figure 6 **Upper Plenum and Downcomer Collapsed Levels in the PACTEL Experiment (Exp) SBL-30 vs. TRACE Calculation**

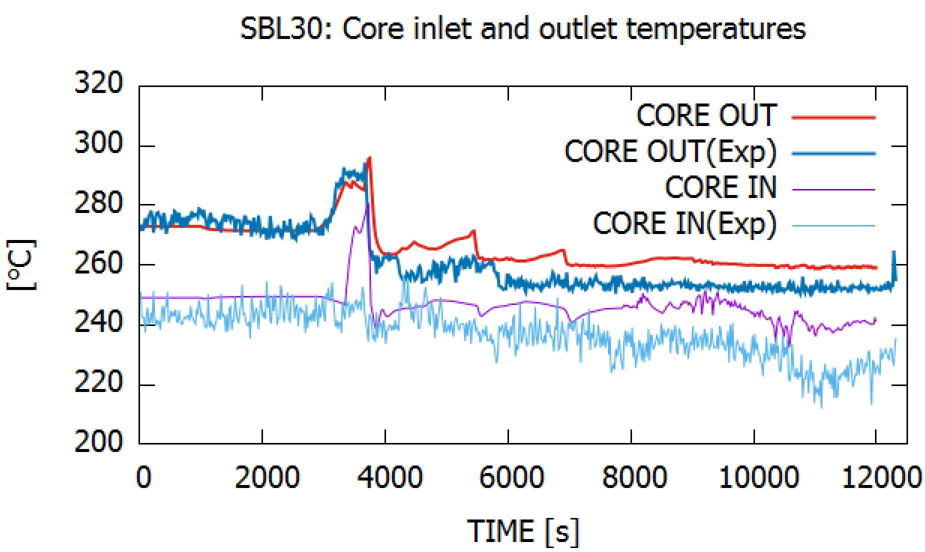

Figure 7 **Core Inlet and Outlet Temperatures in the PACTEL Experiment (Exp) SBL-30 vs. TRACE Calculation**

An accurate modeling of the asymmetric loop flow behavior is a very difficult task, since there are many uncertainties in the experiment situation, which cannot be taken into account in the calculations. 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. Also, 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.

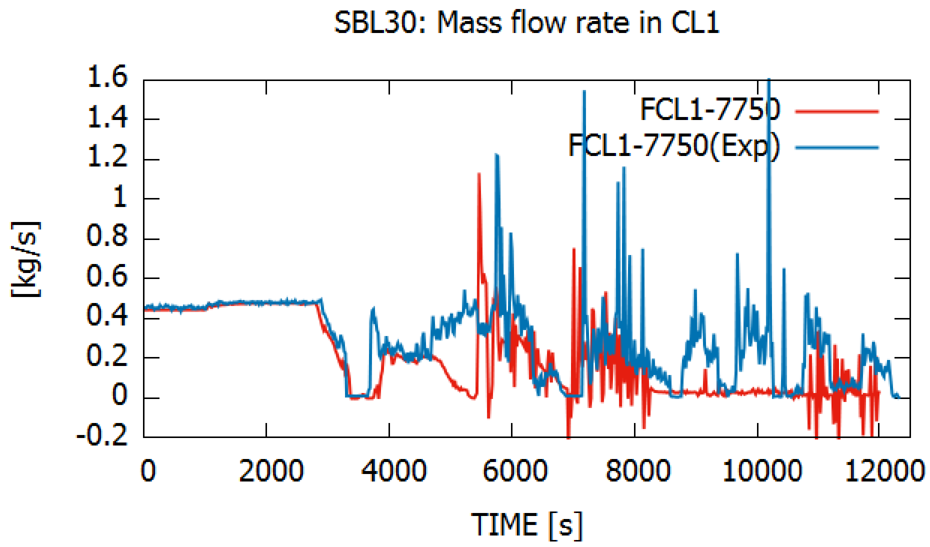

Figure 8 **Mass Flow Rate in Cold Leg 1 in the PACTEL Experiment (Exp) SBL-30 vs. TRACE Calculation**

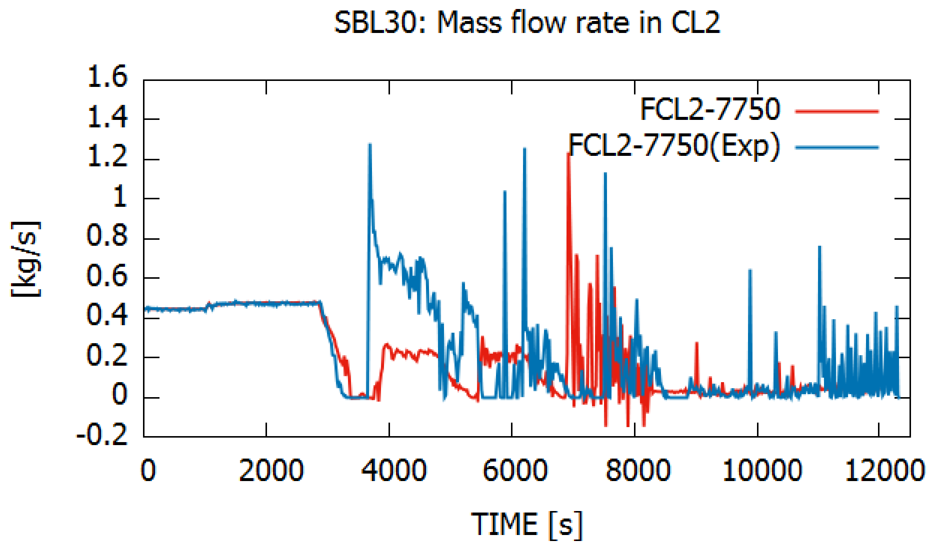

Figure 9 **Mass Flow Rate in Cold Leg 2 in the PACTEL Experiment (Exp) SBL-30 vs. TRACE Calculation**

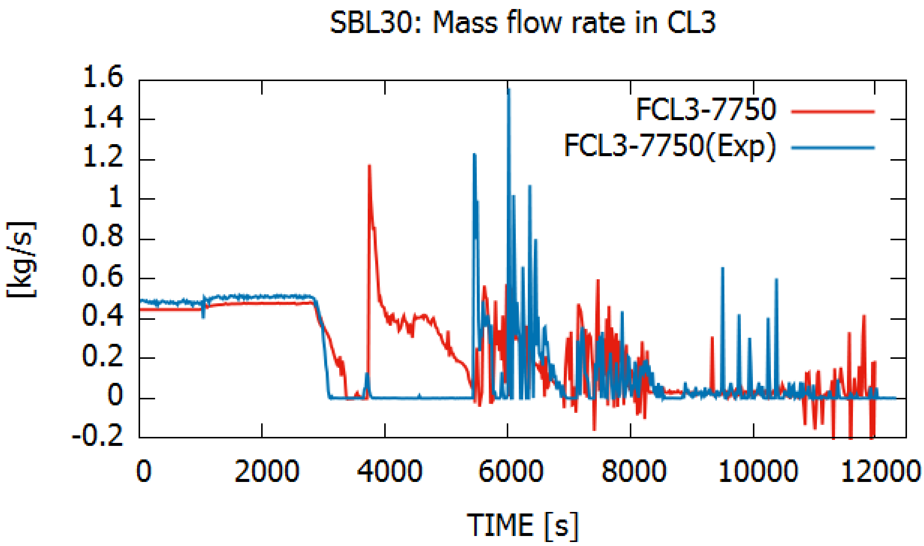

Figure 10 **Mass Flow Rate in Cold Leg 3 in the PACTEL Experiment (Exp) SBL-30 vs. TRACE Calculation**

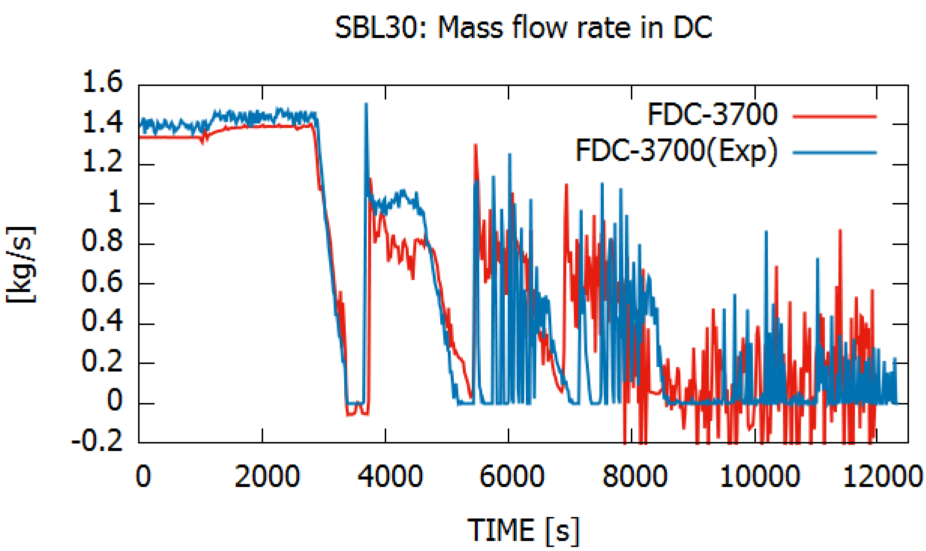

Figure 11 **Mass Flow Rate in Downcomer in the PACTEL Experiment (Exp) SBL-30 vs. TRACE Calculation**

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 3 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	17000	10736	0.632	613782

6 CONCLUSIONS

The PACTEL facility small break loss-of-coolant-accident experiment, SBL-30, was calculated using TRACE V5.0 Patch 4. The calculation results were compared to the experimental data. In general the TRACE calculations agreed reasonably well with experimental data.

A full simulation model of the PACTEL test facility, modeling a VVER-440 type nuclear plant, was prepared with the TRACE thermal hydraulic code. The PACTEL experiment SBL-30 was then calculated using the TRACE model. In the SBL-30 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. In primary pressure and loop flow behavior there were some discrepancies between the calculation and experiment results but the overall tendency with several stagnations and resumes of natural circulation flow agreed well with the experiment. Loop seal clearing taking place in correct succession is very difficult to calculate due to nature of this phenomenon being unstable in reality. The main differences between the simulated and experiment results were probably due to inaccuracies in the definition of the heat loss distribution in the calculation model.

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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 is small break loss-of-coolant-accident. The natural circulation is of particular interest for code assessment as it requires the system code to accurately predict temperature and density distributions throughout the system. Specific modeling capabilities are required for heat transfer and two-phase flow phenomena. The present work introduces the assessment of the PACTEL small break LOCA experiment SBL-30 with the TRACE V5.0 Patch 4. The PACTEL facility is volumetrically scaled full-height model of a six-loop Russian design VVER-440 PWR. This reactor type has specific features like horizontal steam generators and hot leg loop seals. Although the TRACE code has not been originally developed for the special geometry of the VVER-440 reactor type, it was proven that the code is capable for reproduce the natural circulation phenomena satisfyingly. 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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