International Agreement Report

Post-Test Analysis of Cold Leg Small Break 4.1% at PSB-VVER Facility using TRACE V5.0

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ABSTRACT

The U.S. NRC best estimate thermo-hydraulic computer code TRACE V5.0 has been assessed against the “4.1 % cold leg break (CL-4.1-03)” experiment at the large-scale test facility PSB-VVER. The PSB-VVER facility is a 1:300 volume-power scaled model of VVER-1000 NPP located in Electrogorsk, Russia. An extensive TRACE input deck of PSB-VVER facility was developed. The model includes all important components of the PSB-VVER facility: reactor, 4 separated loops, pressurizer, HPIS and LPIS ECCS, several break units, main circulation pumps, steam generators, and important parts of secondary circuit. The TRACE (TRAC/ RELAP Advanced Computational Engine) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission in frame of CAMP (Code Application and Maintenance Program). The TRACE code is a component-oriented reactor systems analysis codes designed to analyze light water reactor transients up to the point of significant fuel damage. The original validation of the TRACE code was mainly based on experiments performed on experimental facilities of typical PWR design. There are some different features of VVER design comparing to PWR. Therefore the validation of the thermo-hydraulic codes for VVER types of reactors is often required by national regulators. The presented analysis is the latest in series of TRACE and RELAP5 assessment calculations evaluated at the company TES. The purpose of performed analyses is to extend the validation of the TRACE code focused on VVER type of NPPs and to support applicability of the TRACE code in the Czech Republic.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

The purpose of this work is to contribute to the USNRC thermal-hydraulic codes assessment as agreed in the CAMP agreement. TRACE V5.0 Patch 3 and Patch 4 computer codes are assessed against LOCA test data from the PSB-VVER test facility.

ACKNOWLEDGMENTS

This report and used input model of the Integral Test Facility (ITF) PSB-VVER is based on the outcomes of the project FI-IM5/150 that was funded by The Czech Ministry of Industry and Trade in frame of national R&D activities.

ABBREVIATIONS AND ACRONYMS

ACC	Hydro-accumulator
ACAP	Automated Code Assessment Program
ADS	Atmospheric Dump System
ATWS	Anticipated Transient Without SCRAM
CAMP	Code Application and Maintenance Program
CCFL	Counter Current Flow Limitation
CFD	Computational Fluid Dynamics
CL	Cold Leg
CPU	Central Processing Unit
CS	Core Simulator
DC	Downcomer
ECCS	Emergency Core Cooling System
ECI	Exterior Communications Interface
EREC	Electrogorsk Research and Engineering Institute
FOM	Figure of Merit
FRS	Fuel Rod Simulator
GNU	GNU's Not Unix!
HL	Hot Leg
HPIS	High Pressure Injection System
IE	Initiating Event
ITF	Integral Effect Test Facility
LOCA	Loss of Coolant Accident
LP	Lower Plenum
LPIS	Low Pressure Injection System
LWR	Light Water Reactor
MCP	Main Coolant Pump
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Co-operation and Development
PARCS	Purdue Advanced Reactor Core Simulator
PKL	Experimental Facility in Finland
PCS	Plant Control System
PRZ	Pressurizer
PRISE	Primary to Secondary Leak
PWR	Pressurized Water Reactor
R&D	Research and Development
RPV	Reactor Pressure Vessel
SCRAM	Emergency Shutdown of a Nuclear Reactor
SETF	Separate Effect Test Facility
SETS	Semi-implicit Method and the Stability Enhancing Two-Step

SG	Steam Generator
SNAP	Symbolic Nuclear Analysis Package
TC	Thermocouple
TH	Thermal Hydraulic
UP	Upper Plenum
VVER	Russian Pressurized Water Type Reactor

1 INTRODUCTION

The assessment of LWR best estimate thermo-hydraulic computer codes is mainly based on experimental data from separate and integral test facilities. The TRACE is a member of the western PWR codes family which validation process is mainly based on the data from experimental facilities or real NPPs of western PWR type. There is a significant number of LWR NPPs which utilize VVER design of reactors operating all over the world and many other are under construction or planned as well. The VVER design of LWR is generally similar to western PWRs but there are several features that should be taken into consideration such as larger volumes of primary coolant, horizontal steam generators, different ECCS injection points and so on. While a lot of experimental data measured on PWRs or PWR-based test facilities are valuable also for research of VVER design, the specific validation of thermo-hydraulic codes for VVER design reactors is often required by national regulators. The purpose of performed analyses is to extend the validation of the TRACE code focused on VVER type of NPPs. The best estimate thermo-hydraulic computer code TRACE V5.0 patch03 and patch04 was assessed using Cold Leg 4.1% break experiment at the large-scale test facility PSB-VVER.

The PSB-VVER facility is a 1:300 volume-power scaled model of VVER 1000 NPP located at Electrogorsk, Russia. In order to perform code validation an extensive TRACE input deck of PSB-VVER facility was developed. The TRACE model includes all important components of the PSB-VVER facility: reactor, four separated loops, pressurizer, break units, main circulation pumps, steam generators, and important parts of secondary circuit.

The main goals of the experiment 4.1 % cold leg break (CL-4.1-03) were as follows:

- to study VVER-1000 thermal hydraulics in an accident caused by a small break in the cold leg
- to investigate scale effect
- to investigate design effect of the horizontal SG tube arrangement
- to provide data for code assessment regarding the prediction of a VVER system thermal hydraulic response following the small break loss-of-coolant accident

The test was also intended to be similar to the experiments performed in western integral test facility LSTF (test SB-CL-21), BETHSY (test 6.2 TC), LOBI (test BL-34) and SPES (test SP-SB-03). The time course of the transient that was performed in the LOBI facility was replicated in the test performed in the PSB-VVER facility and may be divide into four main stages:

- blowdown of subcooled coolant and the first heating of fuel assembly
- decreasing the primary coolant mass at high pressure and re-flooding after actuation of the ECCS ACCs
- decreasing the primary coolant mass at low pressure after stopping the operation of the ECCS ACCs and the final heating of the fuel assembly model
- actuation of the LP ECCS and the flooding of the primary circuit

2 FACILITY AND TEST DESCRIPTIONS

Detail information about the PSB-VVER test facility systems and elements is given in Ref 11. Only a brief description of the PSB-VVER facility is given here. The hardware configuration for CL-4.1-03 test is reported below.

2.1 PSB-VVER Facility

PSB-VVER is a large-scale integral test facility which structurally corresponds to primary circuit of NPP with VVER-1000 (V-320 design). The volumetric and power scale is 1:300, and the main equipment elevations correspond to those of the prototype reactor.

The facility consists of four loops linked up to the reactor model. Each loop has a circulation pump, a steam generator, hot and cold legs. One of the loops (loop No.4, "broken") has special branch pipes for connection to primary leakage simulation system. The test facility also includes a pressurizer (PRZ) and ECCS, which has, as in actual VVER-1000, three subsystems: a passive system and two active ones.

The reactor model comprises four elements: an external downcomer, core model, core bypass and an upper plenum. The PSB-VVER core model consists of 168 full-height indirectly electrically heated fuel rod simulators with uniform power distribution. The rod simulator pitch (12.75 mm) and diameter (9.1 mm) are identical to those of the reference reactor. The fuel rod simulators are arranged on a triangular grid. The rod bundle cross section has the shape of regular hexagon with "wrench" size of 168 mm. The core model represents the central part of the reference fuel rod assembly. Along its height, the assembly has 15 spacer grids with natural geometry

PSB-VVER pressurization system includes a pressurizer, surge lines, spray lines, and a relief valve. By means of surge and spray lines the pressurizer can be connected to the “broken” loop (loop #4) or to one of the intact loops (loop #2) of the facility. The PRZ vessel height, the bottom elevation and location of nominal level correspond to the reference ones. An electric heater with a power of up to 80 kW is built in the lower part of the pressurizer vessel.

PSB main circulation pumps are used to provide forced circulation in primary circuit. The circulation pumps are variable-speed ones of vertical centrifugal single-stage type and can operate under two-phase fluid conditions.

The passive ECCS system consists of four accumulators connected in pairs to an inlet and outlet chamber of the reactor pressure vessel. The active ECCS system consists of high pressure injection system (HPIS) and low pressure injection system (LPIS). Cooling water of active ECCS can be supplied to three loops, both to cold and hot legs as original facility design.

The PSB-VVER SG is a vertical vessel with two vertical headers inside. A bundle of horizontal spiral heat-exchanging tubes of full size is mounted between the two headers. The PSB-VVER SG is designed in such way that the reference tube bundle elevations and tube lengths to be conserved, as well as the flow area. Heat transfer surface and secondary fluid volume to be matched the scale factor. On the secondary side, the feed water system and the main steam lines are simulated. The turbine and the condenser are not modeled.

Primary and secondary circuits of the PSB-VVER facility are operated at nominal pressure of a reactor prototype.

Figure 1 depicts an isometric projection of the test facility. Main operational characteristics of the test facility are given in Table 1.

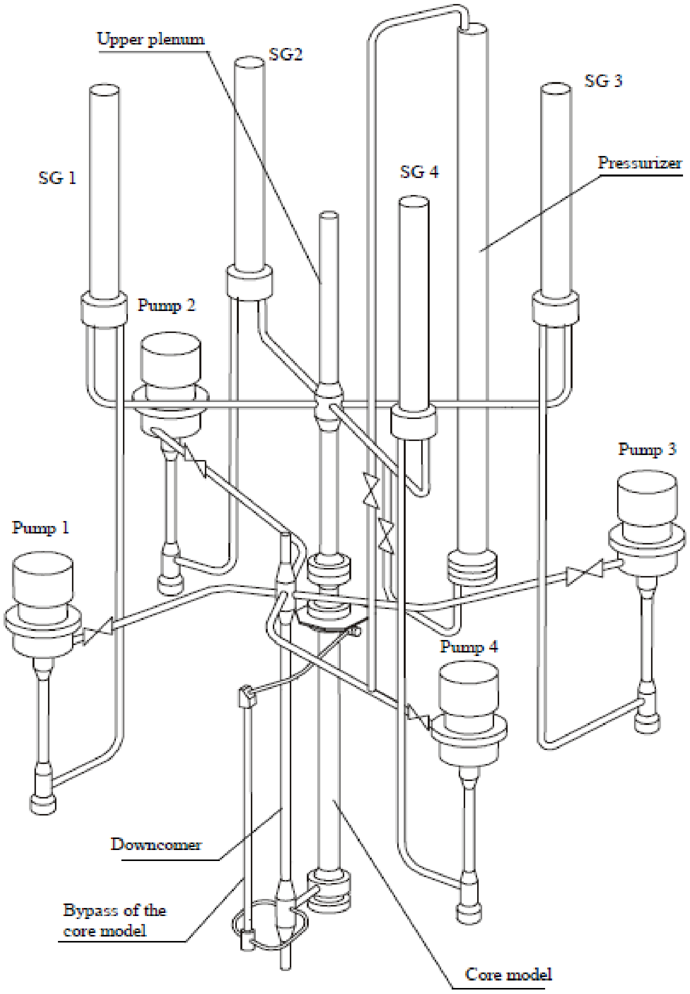

Figure 1 General View of PSB-VVER Facility

Table 1 Main Operational Characteristics of PSB-VVER Comparing to VVER-1000

Parameter	Units	VVER-1000	PSB-VVER
Coolant	-	water	water
Number of circulation loops	-	4	4
Primary circuit			
Pressure	MPa	15.7	15.7
Coolant temperature (hot/cold leg)	deg	290/320	290/320
Coolant flowrate	m^3/h	82485	< 280
Core power	MW	3000	15
Secondary circuit			
Steam generator pressure	MPa	6.3	6.3
Feed water temperature	deg	220	< 270
Thermal power of one SG	MW	750	2.5

The PSB test facility is equipped with several special break systems to facilitate research of thermal hydraulics during break accidents. There is a special system to simulate:

- accumulator water supplying pipe rupture (11% upper plenum break experiments UP-11-07, UP-11-08)
- guillotine break of hot leg (2x25% hot leg large break experiment HL-2x25-02)
- guillotine break of cold leg (100 % double ended cold break experiment CL-2x100-01)
- small cold leg break (4.1 % cold leg break test CL-4.1-03)
- primary to secondary leak (PRISE test Psh-1.4-04)

2.2 Experiment CL-4.1-03

The test CL-4.1-03 ”4.1 % Cold Leg Break test“ was performed in the PSB-VVER test facility at Electrogorsk Research and Engineering Center (EREC) in Russia. The thermal-hydraulic processes related to cold leg break 4.1% were investigated.

2.2.1 Facility Configuration

The information on the test facility hardware and configuration of the system specific for CL-4.1-03 test is given in the Table 2.

Table 2 Test Facility Configuration in CL-4.1-03 Test

Equipment	Status
Pressurizer	Connected to the loop #4
Core by-pass	2 diaphragms with 2 orifices with diameter of 7 mm are installed at inlet and outlet of core by-pass
HPIS	Not used
LPIS	ECCS water is injected in cold legs of loops # 1, 3 and 4
ACCs	Two accumulators are connected to the inlet chamber
SGs	Under steady state all SGs are connected to each other by steam header. The pressure is adjusted by one steam dumping valve RA06S01
ADS	ADS (atmospheric steam dump system) is connected to each SG. In each ADS line, throttle channel (L/d=10) 12.1 mm in diameter is installed
Feed water heater	Under steady-state conditions the SG levels are maintained by means of impulse supply of the feed water with temperature ~ 170 C°.
Large break unit	Upward vertically oriented break line is connected to loop #4 cold leg between MCP-4 and pressure vessel inlet. The leak is limited by a smoothly edged (R=6 mm) break nozzle 10 mm in diameter and 100 mm length.
UP warming-up line	Under steady state the line is open. The warming-up of the UP top part is stopped about 2 min before opening break line.
Warming-up line of break line	Under steady state the line is open. The warming-up of the break line is stopped about 1 min before opening break line.

2.2.2 Initial Conditions

The main initial conditions of CL-4.1-03 test are given in the Table 3. The CL-4.1-03 test has been performed under reduced initial core power corresponding to approximately 15% of nominal power.

Table 3 Measured Initial Condition for CL-4.1-03 Test

Parameter	Units	Value
Primary circuit		
Pressure in upper plenum (gauge YC01P16)	MPa	15.512
Coolant temperature	deg C	282.4 / 309.2

Table 3 Measured Initial Condition for CL-4.1-03 Test (Cont'd)

(DC inlet/UP outlet - gauges C01T02/YC01T04)		
Primary loops flow rates (gauges YA01÷04F01)	kg/s	1.964 / 1.989 / 1.979 / 1.937
Core power (gauge YC01N01)	kW	1128.6
Core by-pass power (gauge YC01N02)	kW	14.9
Coolant level in PRZ (gauge YP01L02)	m	3.073
Secondary circuit		
Pressure in SGs (gauges RA01÷04P01)	MPa	6.878 / 6.913 / 6.936 / 6.886
Level in SGs (gauges YB01÷04L01)	m	1.904 / 1.913 / 1.935 / 1.901
ECCS		
Pressure in ACCs (gauges TH02,04P01)	MPa	4.079 / 4.072
Level in ACCs (gauges TH02,04L01)	m	4.577 / 4.570

2.2.3 **Boundary Conditions (Test Scenario)**

Detail information about the CL-4.1-03 test boundary conditions is given in Ref 14. The main events of CL-4.1-03 test are described in the Table 4.

Table 4 Main Events During CL-4.1-03 Test

Event	Time [s]
Cutting off electric load at PRZ	-0.6
IE – Break Opening	0
Pressure in UP (YC01P16) < 13.73 MPa → conditions for SCRAM	4.1
Stopping of MCPs 1 ÷ 4 ()	7.8
Pressurizer emptying(YP01L02)	10
Feed water valve closure (RL01÷ 04S06)	14
Main steam valve closure (RA06S01)	18
Core and core by-pass power reduction onset (YC01N01, 02)	57.6
Accumulators injection onset : TH02B01 / TH04B01	414 / 406
Accumulators injection termination (L < 0.559 m): TH02B01 / TH04B01	1452 / 1365
LPIS injection onset (TJ01F02 / TJ03F02 / TJ03F02)	2432 / 2432 / 2434
End of test (FRS power switched off)	2593.4

The experiment is started with opening the break valve XL01S01, after that the PSB-VVER PCS gives signal to disconnect the PRZ heaters.

The actions on reaching the primary pressure of value 13 MPa are as follows:

- closure of secondary pressure control valve RA06S01, while the steam generators remain connected with each other by the steam line;
- closure of valves RL01-04S06 and cutoff feed water into steam generators;
- trip of all MCPs, the pump rotor speed drops from its initial value to zero in 4 s
- core power follows the power/time relation given in following table:

Table 5 Time Dependence of Dimensionless Core Power

Time (s)	Power/Initial power (-)	Time (s)	Power/Initial power (-)
0	1	400	0.224
53	1	600	0.196
60	0.941	800	0.186
80	0.783	1000	0.178
100	0.662	1500	0.163
150	0.464	2000	0.148
200	0.359	2375	0.144

When primary pressure drops below 4 MPa, the ECCS accumulators start to inject water in the pressure vessel inlet chamber. Accumulators remain connected to the inlet chamber until the water levels decrease below 1.31 m (0.559 as readings TH0204L01). Then the accumulators are disconnected by valves TH02S05 and TH04S05 in order to avoid air penetration into the primary system.

The low pressure ECCS is activated at the FRS temperature 500 °C, provided that the accumulator water injection is already terminated and the tank are disconnected from the primary system. The water is supplied in cold legs of loops #1, 3 and 4 with the mass flow rate 0.248 kg/s in each line, the flow rate is regulated by the valves TJ01S08, TJ03S08 and TJ04S08.

The SG ADS valves open at secondary pressure 7.4 MPa and close at 7.2 MPa. The open/close set points of the ADS valves are derived from the plots demonstrating the secondary pressure behavior in LOBI facility. Throughout the transient, the steam generators remain connected to a common steam header; therefore the steam generators secondary pressures are expected to be equal. But an insignificant uncertainty in measuring the pressures can lead to opening ADS valve in some steam generator while the others remain closed due to decrease in the secondary pressure. Therefore, all the ADS valves are assumed to be dependent on SG-1 pressure (YB01P01).

The experiment had to be ended on reaching the FRS temperatures to 700 °C or on reaching the steady core cooling conditions after the final core rewet.

3 THE TRACE V5.0 CODE

The TRACE (TRAC/RELAP Advanced Computational Engine) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission for analyzing transient and steady-state neutronic-thermal-hydraulic behavior in light water reactor.

The TRACE code has been widely used by U.S. Nuclear Regulatory Commission (NRC) and other organizations for rulemaking, licensing audit calculations, evaluation of operator guidelines, and as a basis for a nuclear plant analyzer. Specific applications of their capability have included simulations of transients in LWR systems, such as loss of coolant, anticipated transients without scram (ATWS), and operational transients such as loss of feedwater, loss of offsite power, station blackout, and turbine trip. The TRACE is a highly generic code that, in addition to calculating the behavior of a reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of steam, water, noncondensable gases, and solute.

The TRACE code is a component-oriented reactor systems analysis code designed to analyze light water reactor transients up to the point of significant fuel damage. The TRACE code solves a finite-volume two-phase multidimensional compressible flow with one, two and three dimensional flow geometry. The TRACE code can model heat structures and control systems that interact with component models and the fluid solution. The TRACE code has capability to use build-in point reactor kinetics or 3D reactor kinetics through coupling with Purdue Advanced Reactor Core Simulator (PARCS). In addition the TRACE code can be coupled with another TRACE jobs or other codes (CFD, CONTAIN ...) through its exterior communications interface (ECI). TRACE uses what is commonly known as a 6-equation model for two-phase flow (mass equation, equation of motion and energy equation for each phase). Additional equations can be solved for noncondensable gas, dissolved boron, control systems and reactor power. There are five additional closure relationships for field equations: equations of state, wall drag, interfacial drag, wall heat transfer and interfacial heat transfer. These constitutive models are semi empirical equations. There are two numerical methods available in TRACE: semi-implicit method and the stability enhancing two-step (SETS) method.

3.1 The TRACE V5.0 Code Assessment

Confidence in the computational tools (codes) and establishment of their validity for a given application depends on proper assessment. The TRACE code, like other two-fluid codes, is composed of numerous models and correlations. When applied to full scale nuclear power plant conditions, many of these models and correlations can be applied outside of their original scope. By assessing the code against thermal-hydraulic tests, it is possible to show that the code and its constituent model packages can be extended to conditions beyond those for which many of the individual correlations were originally intended (Ref 4). The assessment process however, can also indicate potential deficiencies in the code. There are following four sources of data for code assessment (Ref 21):

- “Fundamental” experiments
- Separate effect test facilities (SETF)
- Integral test facilities (ITF)
- Real plant data

3.2 VVER Typical Features Related to TRACE V5.0 Code Assessment

The TRACE code validation process is mainly based on the data from experimental facilities or real NPPs of Western PWR type. VVER reactors are in many aspects similar to Western PWRs. Therefore a lot of experimental data measured on PWRs or PWR test facilities is valuable also for VVER research. On the other hand, the VVER design has several specific features. From the hardware point of view the main differences between VVER-1000 and PWR are the following (Ref 9):

- Horizontal steam generators with 2 headers
- Lower plenum internal structures
- Fuel assemblies with hexagonal fuel rod arrangements
- ECCS injection points
- Secondary side water volume of the steam generators is larger
- Operational conditions and set points of actuation of ECCS
- Working conditions of secondary side of steam generators and set points for the operation of feedwater and steam lines

There are approximately 60 operating units of VVER type (Ref 20). It is a meaningful number in comparison to approximately 216 operating units of PWR reactors (Ref 20). Therefore corresponding attention should be given to code validation for VVER type of reactors.

4 INPUT DECK DESCRIPTION

An extensive TRACE input deck of PSB-VVER facility was developed including all important components of the PSB-VVER facility: reactor, 4 separated loops, pressurizer, break units, main circulation pumps, steam generators, break sections and important parts of secondary circuit. The input decks were designed on the basis of PSB-VVER facility documentation (Ref 11, 12).

4.1 The TRACE Input Deck

Nodalization diagrams of the TRACE Input Deck are presented in Figure A-1 (reactor + primary circuit) and Figure A-2 (secondary circuit) in the Appendix A. The TRACE model of the reactor pressure vessel (RPV) with internal structures is divided into 3 parts. The first part represents Downcomer (DC) + Lower Plenum (LP), the second part represent Fuel Rod Simulator (FRS) + Upper Plenum (UP) and the third part represents core by-pass from DC to UP. The RPV model employs VESSEL component for DC + LP (includes 26 axial layers, 1 azimuthal theta sector and 2 radial rings), 3 PIPE components for core by-pass and the next VESSEL component for FRS + UP (includes optionally 25 or 26 axial layers, 4 azimuthal theta sector and 3 radial rings). The model also includes the by-pass piping between DC to UP and the UP heating pipe between cold leg of loop #1.

Each of the four coolant loops comprises of: a hot leg, steam generator, pump suction loop seal piping, main coolant pump, and a cold leg including control valve between MCP discharge and DC.

The pump performance is based on single-phase head and torque characteristic of the pump TsNIS 1620 from Ref. 12. No two-phase degradation was modeled because no appropriate data are available.

The pressurizer is modeled using component PIPE equipped with heaters and with surge lines connected to the hot legs of loop #2 and loop #4. Pressurizer can be optionally connected to loop #2 or loop #4 (as original facility design).

Active ECCS are modeled using simple boundary condition - FILL component, accumulators are modeled using PIPE and VALVE components. High pressure injection system (HPIS) optionally provides flow to the hot leg of loop #1. Low pressure injection system (LPIS) optionally provides flow to the hot leg of loop #1 and #3 and to the cold leg of loop#1, loop#3 and loop#4. Four accumulators provide flow to the downcomer and upper plenum (two to each location), and any of them can be switch off. Cooling water delivery from ECCS depends on hardware configuration of a particular test.

Steam generator is modeled using multi-tube approach. The primary side of SG Input Deck consists of 5 axial layers of heat exchanging tubes and two headers (the original facility SGs consists of 34 tubes). Each axial layer is divided into 15 segments. The SG secondary part is modeled as an original three-channel complex with 10 axial layers (5 of them in the area of exchanging tubes). The feedwater system, the steam lines connected to all SGs and the common steam header are also modeled. The BREAK component simulates the release of secondary steam from steam header. The secondary side of all SGs is equipped with the ADS (atmospheric dump system) using VALVE and BREAK component.

The heat loss from the primary and secondary is represented in the TRACE model by entering the thickness of the insulation on the outer surface of all the pipes and other system components. Appropriate material properties are input for the insulation. A constant boundary temperature and heat transfer coefficient of outer air is applied.

The model contains 1239 volumes, 2176 junctions, and 1114 heat structures with 4174 mesh points. Standard modeling guidelines were followed in developing the nodalization of the system.

Components Statistic for TRACE model – see the next Table 6.

Table 6 TRACE Components Statistic

TRACE Component		**Notes**
VESSEL	2	DC+LP; FRS+UP
PIPE	67+66[*1]	-
HSTR	139+1[*1]	-
POWER	1+1[*1]	FRS simulator + By-pass heating
VALVE	30	-
PUMP	4	MCPs
BREAK	8	Upper plenum, Large and Cold Leg Break units, release of secondary steam, SG ADS
FILL	10	HPIS, LPIS, Feedwater, Make-up system
Total No of Components	330	-

*1 the second number is the number of spawned component

Table 7 List of the Main Systems and Components of PSB-VVER TRACE Input Deck

PSB system	Input deck	Used TRACE components
Rector (YC)		
Downcomer	+	VESSEL + HTSTR
Lower plenum	+	VESSEL + PIPE + HTSTR
Core	+	VESSEL + HTSTR + POWER + CONTROL BLOCK
Upper plenum	+	VESSEL + HTSTR
Core bypass	+	PIPE + HTSTR + POWER + CONTROL BLOCK
DC to UP bypass	+	VALVE + HTSTR
UP heating	+	VALVE + HTSTR
LOOP (YA)		
Hot leg	+	PIPE + HTSTRT
Loop seal	+	PIPE + HTSTRT
Cold leg	+	VALVE + HTSTRT
Main cooling pump (YD)		PUMP + HTSTR + CONTROL BLOCK
Pressurizer (YP)		
Vessel	+	PIPE + HTSTR
heaters	+	HTSTR + CONTROL BLOCK
Surge line	+	VALVE + HTSTR
Relief valve	-	
Make-up system	(+)	FIL + CONTROL BLOCK
ECCS (TJ, TH)		
HPIS	(+)	FILL + CONTROL BLOCK
LPIS	(+)	FILL + CONTROL BLOCK
Accumulators	+	PIPE + VALVES + HTSTRT
Steam generators (YB)		
Vessel	+	PIPE + HTSTR
Heat exchange tubes	+	PIPE + HTSTR
Primary headers	+	PIPE + HTSTR
Steam lines	+	VALVE + HTSTR
Feedwater	(+)	FILL + CONTROL BLOCK
Relief valve	-	
ADS	(+)	VALVE + BREAK + CONTROL BLOCK
Steam headers (RA)	+	VALVE + HTSTR + BREAK

Key: + a fine model
(+) a simplified model
\- not modeled

5 RESULTS

5.1 Steady-State Calculation

In order to achieve stable initial conditions of the CL-4.1-03 test, the steady state was calculated for 1200 s. The following controllers were used for the first 1200 s:

- Pressurizer pressure controller
- Pressurizer level controller
- Steam generators level controllers

The other controlled parameters (fuel rod simulator power, core bypass power, feedwater temperature, main steam header pressure) were entered as boundary conditions. The steam generator pressures are lower than the measured values, because the steam header pressure was adjusted to get the desired reactor vessel inlet temperature (average cold legs temperature) in steady state calculation. A real PSB controller that kept the liquid level in all SGs within a desired band was replaced by a PI-controller for 1000 s of steady-state calculation, for the rest of steady state calculation and for transient calculation was feed water flow to SGs entered as a boundary condition. Reaching steady state conditions took approximately 41 min. Main calculated and measured parameters are compared in the following Table 8.

Table 8 Initial Conditions (TRACE Calculation vs. Experiment Comparison)

Parameters	TAG	Units	accuracy	CL-4.1-03[*2]	TRACE
Primary Circuit					
Upper plenum pressure	YC01P16	MPa	± 0.05	15.512 ± 0.004	15.513
Pressure drop at FRS	YC01DP07-DP10	kPa	± 2.40[*1]	-28.6 ± 0.021	-29.4
Upper Plenum Coolant temperature	YC01T04b	°C	± 4.2	311.5 ± 0.1	308.9
Core Inlet Coolant Temperature	YC01T259	°C	± 3.0	282.5 ± 0.1	282.7
Hot Leg outlet Coolant Temp.	TA01T03	°C	± 3.0	310.4 ± 0.1	308.3
Hot Leg outlet Coolant Temp.	TA02T03	°C	± 3.0	308.1 ± 0.4	309.4
Hot Leg outlet Coolant Temp.	TA03T03	°C	± 3.0	310.5 ± 0.1	309.4
Hot Leg outlet Coolant Temp.	TA04T03	°C	± 3.0	308.5 ± 0.4	309.6
Cold Leg outlet Coolant Temp.	TA0T02	°C	± 3.0	283 ± 0.1	282.7
Cold Leg outlet Coolant Temp.	TA02T02	°C	± 3.0	283.5 ± 0.1	282.7
Cold Leg outlet Coolant Temp.	TA03T02	°C	± 3.0	282.4 ± 0.1	282.7
Cold Leg outlet Coolant Temp.	TA04T02	°C	± 3.	281.8 ± 0.1	282.7
Loop-1 flow rate	YA01F01	kg/s	± 0.1	1.964 ± 0.012	1.929
Loop-2 flow rate	YA02F01	kg/s	± 0.1	1.989 ± 0.007	1.948
Loop-3 flow rate	YA03F01	kg/s	± 0.1	1.979 ± 0.007	1.945
Loop-4 flow rate	YA04F01	kg/s	± 0.1	1.937 ± 0.013	1.913
FRS power	YC01N01	kW	± 15	1128.6 ± 8.3	1128.6
Core by-pass power	YC01N02	kW	± 0,4	14.87 ± 0.02	14.9
Collapsed level in PRZ	YP01L02	m	± 0.3	3.073 ± 0.017	3.073

Table 8 Initial Conditions (TRACE Calculation vs. Experiment Comparison) (Cont'd)

Parameters	TAG	Units	accuracy	CL-4.1-03[*2]	TRACE
Secondary Circuit					
Pressure in SG-1	YB01P01	MPa	± 0.05	6.878 ± 0.002	6.784
Pressure in SG-2	YB02P01	MPa	± 0.05	6.913 ± 0.002	6.783
Pressure in SG-3	YB03P01	MPa	± 0.05	6.936 ± 0.002	6.776
Pressure in SG-4	YB04P01	MPa	± 0.05	6.886 ± 0.002	6.785
Collapsed level in SG-1	YB01L01	m	± 0.08	1.904 ± 0.002	1.904
Collapsed level in SG-2	YB02L01	m	± 0.08	1.913 ± 0.002	1.913
Collapsed level in SG-3	YB03L01	m	± 0.08	1.935 ± 0.002	1.935
Collapsed level in SG-4	YB04L01	m	± 0.08	1.901 ± 0.003	1.901
Accumulators					
ACCU-2 pressure	TH02P01	MPa	± 0.03	4.577 ± 0.001	4.577
ACCU-4 pressure	TH04P01	MPa	± 0.03	4.570 ± 0.001	4.570
ACCU-2 collapsed level	TH02L02	m	± 0.07	4.079 ± 0.000	4.080
ACCU-4 collapsed level	TH04L02	m	± 0.07	4.072 ± 0.001	4.074

*1 sum of accuracy of pressure drop YC01DP07-DP10 (accuracy of YC01DP07,08,10 = ± 1.2 kPa, YC01DP09 = ± 1.0 kPa)

*2 Average value ±standard deviation of measured parameters at initial steady state condition of the test facility

5.2 Transient Calculation

The post-test calculation of the CL-4.1-03 experimental test at PSB-VVER test facility started at the time 0 s simultaneously with an initiating event – Cold Leg break 4.1 % and disconnecting of the PRZ electric heater. A comparison of calculated and experimental times of the occurrence of main events is presented in the Table 9. Time courses of all important parameters and their comparison with experimental data are presented in Appendix C.

A brief overview of the behavior observed in the experiment is provided here, then the comparisons between the measured data and calculations will be presented and discussed.

5.2.1 Time Course of the Transient

The time course of the transient can be divided into four main stages:

- blowdown of subcooled coolant and the first heating of fuel assembly
- decreasing the primary coolant mass at high pressure and re-flooding after actuation of the ECCS ACCs
- decreasing the primary coolant mass at low pressure after stopping the operation of the ECCS ACCS and the final heating of the fuel assembly model
- actuation of the LP ECCS and the flooding of the primary circuit

Immediately after the leak flow onset, the primary pressure rapidly decreased (Figure C-1). At the time 4.1 s the primary pressure decreased down to 13.0 MPa followed by closing RA06S01 valve in steam dumping line and RL01-04S06 valves in feed water lines. At the same time all MCPs were also turned off and they were fully stopped by 8 s. Power loss at the assembly model and by-pass began at 57 s. By 1620 s the power at the fuel rod simulator (FRS) was decreased down to 180 kW and later on it has constant value (Figure C-7).

As a result of the primary pressure decrease, the PRZ coolant boiled up and was completely displaced into hot leg of loop 4. By the 10 s, the PRZ fluid level reached the lower pressure tap elevation mark (Figure C-3, gauge YP01L02).

Mass-flow rates through the circulation loops during first 160 s are presented in Figure C-27. The pressure drops gauge YA01-04DP04 was utilized to show the collapsed level in cold legs above flow meter locations (Figure C-15, C-17, C-19 and C-21). The absolute value of pressure drops close to zero value indicates that after 120 s the collapsed level in cold legs is below the flow meter locations. After that time the YA01-04F01 measurements are not valid.

After the steam dump line valve was closed at the 18 s, there took place rapid growth of pressure (Figure C-6) caused by heat transfer from primary circuit into secondary circuit. The growth of secondary pressure continued until it reached the setpoint and actuated atmospheric steam dump system. The signal to open the valves simulating the ADS operation is conditioned by the SG1 pressure transducer (YB01P01). The valves RA11-14S01 simulating the ADS operation are opened three times in total. The ADS system stopped acting after 150 s, when the primary pressure fell down below the secondary one and reverse heat exchange through SGs took place (SG did not remove heat from primary circuit any more).

On reaching the coolant saturation point and onset of its sub-boiling, phase separation is begun in the primary circuit. Boling of primary coolant was followed by formation of a steam bubble in the upper part of UP at 16 s, in the SG hot headers at 20 s and in the SG cold headers at 36 s. Hotter water discharged from PRZ to the hot leg of loop#4 reached saturation point 7 s earlier than in other steam generators.

Simultaneously with drying of the upper plenum, the levels in the SG headers in the cold and hot legs decreased as well (Figure C-15, C-17, C-19 and C-21). The drying of the SG headers resulted in the primary coolant circulation ending in loop 4 at the 55 s and in other loops at the 80 s (Figure C-27).

As the result of circulation ending in primary loops, there begun intensive flashing of the core model (Figure C-22). By the 100 s the core coolant mass was decreased down to minimal value. The first heating of the fuel rod simulator cladding surface in the top part of the assembly occurred at the 97 s, it was an consequence of that core coolant mass decrease (Figure C-2). The heating of the assembly was stopped at the 105 s as the result of loops 1 and 4 seals water clearance (Figure C-16 and C-22 – gauge YA01,04DP05,06) into downcomer and then into the core model. Clearance of loops 1 and 4 seals were realized completely by the 300 s. Since the 143 s, there was observed presence of steam in the break. This is evident from specific increasing rate of primary pressure reduction (Figure C-2). Since that moment, the primary circuit did not need in steam generators to remove heat and as result of that, there took place reverse of heat exchange between the primary and secondary circuits.

At the 113 s, the coolant in downcomer reached state of saturation as the result of loop seal clearance. By the 124 s the level in downcomer was lowered down to cold leg elevation marks and slowly continued decreasing till the 227 s (Figure C-23). After the 113s, there occurred discharge of the saturated coolant through the break.

The upper plenum was completely dried by the 380 s. After that the core region drainage started (Figure C-22). The second heating of the fuel rod simulator cladding surface begun at the 405 s as the result of their top part dry-out (Figure C-2). The heating was stopped after the primary pressure decreased below pressure in accumulators and begun their discharge into the downcomer.

Actuation of the accumulators TH04B01 and TH02B01 took place at the 406 and 414s respectively (Figure C-5 and C-6). The water from the accumulators TH02B01 and TH04B01 is entering downcomer inlet of the reactor model. Emptying the accumulators and disconnection them from the primary circuit took place at the 1365 s (TH04B01) and at the 1452 s (TH02B01). As both accumulators were supplying water into the same place of the primary circuit, difference in their disconnection time may be explained due to different hydraulic resistances of their connections.

As the result of accumulators operation there took place raising of levels in the reactor model within period of the time 420-800 s (Figures C-23 to C-25). During accumulators discharge the level in the reactor model was kept constant. After termination of operation of accumulators there begun drainage of the upper plenum and downcomer again. At the 1975 s (when the upper plenum was dried almost completely) the level in the fuel core simulator began to decrease. The third heating-up of the fuel rod simulator cladding surface begun at the 2057 s (Figure C-2). It was continued up to the LP ECCS actuation (at the 2432 s) by the signal "overshoot of the FRS cladding surface temperature" (up to 500 °C)

After LP ECCS water supply started with flow rate 0.248 kg/s into cold legs of loops #1, 3 and 4, rapid flooding of the primary circuit begun. By the 2570 s, temperature of simulated fuel element surfaces decreased down to saturation point, and at the 2593 s the experiment was stopped.

5.2.2 Comparison of TRACE Prediction to Reference Data

Comparison of the calculated and measured results showed that calculated values of primary pressure were in a very good agreement with experimental data (Figure C-1). But initial calculated rate of the primary pressure decrease was considerably higher so the set-point 13.0 MPa was reached earlier at 2.2 s unlike in the experiment at 4.1 s. During the time period from the reversal of heat exchange from primary and secondary circuits to the start of the accumulators discharge the calculated primary pressure was decreasing more rapidly than the pressure in the experiment. As the result, the set point of the accumulators is reached sooner (Figure C-6).

To properly predict fuel cladding temperature peaks the option Level tracking was activated in the core and the UP region. As a result of that the accumulators discharge turned to be stepwise. The stepwise discharge was caused by temporally increasing pressure in the core after water from accumulators had been delivered to the core evaporated.

The TRACE code predicted a little bit slower decreasing liquid water in the upper part of the core during the first stage of the experiment so the first temperature peak between the 97 s and 105 s was not predicted. The absolute temperature and time occurrence of the second and the third temperature peak were predicted very well (Figure C-2).

Water distribution (pressure differences) over the reactor vessel and circulation loops (Figure C-15 to C-26) was in a good compliance with experiment data. The decreasing of the absolute value of the measured pressure difference means drainage of a particular region. There are only two regions where the TRACE code predicted different liquid content: the middle part of the core and loop seals (a part including the MCP and the filter). The calculated liquid content of the middle part of the core was somewhat lower than measured one (Figure C-24 and C-25 gauge YC01DP07, 08, 09). The calculated rate of clearance of the loop seals of loops with no break was slower in case the loop#1(Figure C-16) and faster in case the loops #2 and #3 (Figure C-18 and C-20). But these discrepancies did not significantly affected predicted heat-up of the upper part of the core, where it was expected to happen first (Figure C-2). The third temperature peak overcame the FRS cladding temperature set-point 500 °C nearly at the same time in the experiment and calculation followed by acting of LPIS ECCS.

The time course of the secondary pressure including pressure peaks before ADS was activated is in very good agreement with experimental data.

Table 9 Chronology of Main Events (TRACE Calculation vs Experiment Comparison)

Event	Time [s]	
	CL-4.1-03	TRACE
Cutting off electric load to PRZ heaters	-0.6	0
Initiating Event – break opens	0	0
Pressure in UP (YC01P16) < 13.70 MPa – SCRAM signal	4.1	2.2

Table 9 Chronology of Main Events (TRACE Calculation vs Experiment Comparison) (Cont'd)

	7.8	6
Pressurizer emptying (YP01L02)	10.0	9.0
Feedwater valves closure	14	14
Saturation condition at core outlet	16	20
Main steam valve closure	18	3.
Start of core and core by-pass power reduction	57.6	55.2
First core dry-out (peak cladding temperature)	97	-
Two-phase flow break	113	135
Primary to secondary pressure reversal	150	159
Second core dry-out (peak cladding temperature)	405	
Start of ACCU-2 injection	414	342
Start of ACCU-4 injection	406	343
End of ACCU-2 injection	1452	1848
End of ACCU-4 injection	1365	1600
Final core dry-out (peak cladding temperature)	2057	2218
Start of LPIS injection	2432	2408
End of test	2593	2600

5.2.3 Sensitivity Studies Performed

Before final calculation a complex sensitivity study of the impact of various parameters and code options on the prediction of the rod heaters cladding temperature was performed. The parameters with no impact on prediction of the rod heaters cladding temperature are as follows with short explanation:

- CCFL at core outlet plate: The accumulators were connected to the DC and ECCS LPIS were connected to the cold legs so core was flooded from the bottom to the top and such a phenomena as CCFL did not occur.
- Offtake model at the break (the top and bottom orientation of the break): No horizontal stratified flow occurs in the pipe with the break.
- Discharge coefficients at the break: No choking occurred during time period when the TRACE over predicted the break flow (higher the primary pressure rate before accumulators operation).

The parameters affected prediction of the rod heaters cladding temperature are as follows with short explanation:

- The Level Tracking model in the core and UP region.
- The finer discretization of the upper part of the core
- To utilize the latest version TRACE V5.0 patch 04 (at the time when calculations were performed)

The activation of the Level tracking model introduced the presence of the second temperature peak. All other parameters improved time occurrence and absolute value of the second and the third temperature peak.

For the final calculations the finer discretization of the upper part of the core and Level tracking model in the core and UP were applied. The calculation is optionally performed using the TRACE code V5.0 patch03 and patch 04 to access changes between versions of the TRACE ode. Comparison of the test data and calculated results is presented in the appendix C.

Measured data has a circle symbol and calculated results are labeled p03 for TRACE V5.0 patch03 results and p04 for TRACE V5.0 patch04.

5.3 Quantitative Assessment of the Calculations

To quantify agreement of presented TRACE calculations the figure of merit (FOM) was evaluated using software ACAP (Automated Code Assessment Program), which is a part of the software package SNAP. Settings of ACAP was based on Ref 22 including choice of particular metrics and their weighting factors - and see the Table 10.

Table 10 ACAP Metrics Settings

Metric name	Abbreviation	Weighting factor
D'Auria Fast Fourier Transformation	FFT	0.35
Mean Error Magnitude	MEM	0.35
Size-Independent (Pred - Perf) Norm	SI-PMPN	0.15
Degree of Randomness	DOR	0.15

To assess the value of FOM, acceptability criterions were established on the basis of Ref 23, where the FFTB method (Fast Fourier Transform Based Method) is described. FOM acceptability criterions were based on AA_{tot} (total average amplitude). Value of AA_{tot} is transformed to FOM using the equation of D'Auria FFT metric:

$$FOM_{DAURIA} = \frac{1}{\left(\left[AA^2 + \left(\frac{k}{WF}\right)^2\right]^{1/2} + 1\right)}$$

Where *k* is weighted frequency importance factor and value *k* = 0 was applied, which means that pure magnitude error is evaluated using D'Auria FFT metric. The next Table 11 contains values of acceptability criterions range and their meaning.

Table 11 Acceptability Criterions

AA_{tot} range	FOM range	Abbreviation	Color indication
$AA_{tot} \leq 0.30$	$FOM \geq 0.77$	Very good code predictions	green
$0.30 < AA_{tot} \leq 0.50$	$0.67 \leq FOM < 0.77$	Good code predictions	blue
$0.50 < AA_{tot} \leq 0.70$	$0.59 \leq FOM < 0.67$	Poor code predictions	orange
$AA_{tot} > 0.70$	$FOM < 0.59$	Very poor code predictions	red

To assess TRACE and RELAP5 calculation, the representative set of 71 parameters were chosen including:

- Primary pressure
- Fuel cladding temperature
- Pressurizer water level
- RPV Pressure drops

- LOOPs pressure drops and mass flow rates
- Accumulator water levels and pressures
- Steam generators level and pressure
- LPIS mass flow rates

To evaluate overall FOM uniform weighting factors were used for each of parameters.

The CL-4.1-03 experiment was a long lasting transient where many different TH phenomena occurred. In order to carefully assess both TRACE calculations the whole time course was divided into three time windows of interest as follows:

- W1: 0 ÷ 400 s – an early stage of the test when no ECCS flow was provided
- W2: 400 ÷ 1470 s – a middle stage of the test when accumulators flow was provided
- W3: 1470 ÷ 2593 s – the final stage of the test when ECCS LPIS flow was provided

To assess the whole time course of the test FOM calculations with no time segmentation was performed as well. The following table contains all evaluated FOMs for all time windows of interest. To make results more readable color indication mentioned in the Table 11 was applied. FOM = 1 means the best agreement and FOM = 0 the worst agreement. Location of PSB-VVER measurements is depicted in Appendix B for evaluated parameters.

Table 12 Evaluation of FOM for CL-4.1-03 Test

		TRACE V5 patch3				TRACE V5 patch4			
		all	W1	W2	W3	all	W1	W2	W3
Parameter	TAG	0 - 2593	0 - 400	400 - 1470	1470 - 2593	0 - 2593	0 - 400	400 - 1470	1470 - 2593
Water level in accumulator of ECCS TH02B01	TH02L01	0.930	0.650	0.886	0.369	0.926	0.686	0.892	0.343
Pressure in accumulator TH02B01	TH02P01	0.893	0.467	0.851	0.422	0.901	0.488	0.867	0.431
Water level in accumulator TH04B01	TH04L01	0.917	0.742	0.865	0.370	0.923	0.769	0.874	0.474
Pressure in accumulator TH04B01	TH04P01	0.885	0.667	0.848	0.403	0.896	0.700	0.863	0.421
Flow rate in active ECCS system	TJ01F02	0.737	0.752	0.706	0.710	0.794	0.752	0.706	0.789
Flow rate in active ECCS system	TJ03F02	0.734	0.756	0.756	0.704	0.799	0.756	0.756	0.790
Flow rate in active ECCS system	TJ04F02	0.737	0.738	0.743	0.710	0.790	0.738	0.743	0.782
Pressure drop in HL of loop 1 (elevation part)	YA01DP02	0.771	0.791	0.708	0.726	0.771	0.794	0.709	0.727
Pressure drop in CL of loop 1 (from SG outlet to the flow meter)	YA01DP04	0.857	0.840	0.616	0.567	0.878	0.862	0.617	0.569
Pressure drop in CL of loop 1 (from flow meter to mechanical filter)	YA01DP05	0.854	0.804	0.619	0.674	0.858	0.810	0.617	0.679
Pressure drop in CL of loop 1 (section with a mechanical filter, pump and regulating valve)	YA01DP06	0.706	0.662	0.313	0.583	0.699	0.664	0.304	0.556
Pressure drop on the filter in CL of loop 1	YA01DP08	0.602	0.659	0.303	0.494	0.588	0.663	0.299	0.466
Pressure drop on MCP of loop 1	YA01DP09	0.715	0.657	0.277	0.482	0.705	0.658	0.267	0.432
Pressure drop in “cold” header of SG 1 YB01W01	YA01DP13	0.875	0.849	0.646	0.720	0.872	0.842	0.632	0.743
Pressure drop in “hot” header of SG 1 YB01W01	YA01DP14	0.889	0.895	0.665	0.644	0.890	0.896	0.665	0.648
Flow rate in the pump circulation loop YD01D01 (I 5)	YA01F01	0.607	0.593	0.514	0.652	0.608	0.593	0.514	0.654
Pressure drop in HL of loop 2 (elevation part)	YA02DP02	0.835	0.829	0.671	0.681	0.836	0.829	0.673	0.680
Pressure drop in CL of loop 2 (from SG outlet to the flow meter)	YA02DP04	0.864	0.849	0.679	0.711	0.869	0.854	0.679	0.713
Pressure drop in CL of loop 2 (from the flow meter to mechanical filter)	YA02DP05	0.667	0.676	0.588	0.513	0.667	0.677	0.588	0.517
Pressure drop in CL of loop 2 (section with mechanical filter, pump and regulating valve)	YA02DP06	0.703	0.818	0.661	0.668	0.714	0.820	0.671	0.679
Pressure drop on filter in CL of loop 2	YA02DP08	0.698	0.775	0.702	0.739	0.702	0.779	0.713	0.737
Pressure drop on MCP of loop 2	YA02DP09	0.689	0.828	0.658	0.523	0.701	0.827	0.664	0.537
Pressure drop in “cold” header of SG 2 YB02W01	YA02DP13	0.863	0.835	0.644	0.635	0.852	0.820	0.638	0.634
Pressure drop in “hot” header of SG 2 YB02W01	YA02DP14	0.917	0.903	0.725	0.729	0.922	0.908	0.734	0.728
Flow rate in the pump circulation loop YD02D01 (I 6)	YA02F01	0.649	0.739	0.430	0.341	0.652	0.744	0.430	0.340
Pressure drop in HL of loop 3 (elevation part)	YA03DP02	0.816	0.817	0.665	0.718	0.816	0.818	0.665	0.718

Table 12 Evaluation of FOM for CL-4.1-03 Test (Cont'd)

		TRACE V5 patch3				TRACE V5 patch4			
		all	W1	W2	W3	all	W1	W2	W3
Parameter	TAG	0 - 2593	0 - 400	400 - 1470	1470 - 2593	0 - 2593	0 - 400	400 - 1470	1470 - 2593
Pressure drop in CL of loop 3 (from SG outlet to the flow meter)	YA03DP04	0.869	0.861	0.726	0.756	0.863	0.853	0.727	0.755
Pressure drop in CL of loop 3 (from the flow meter to mechanical filter)	YA03DP05	0.658	0.671	0.603	0.635	0.656	0.673	0.604	0.629
Pressure drop in CL of loop 3 (section with mechanical filter, pump and regulating valve)	YA03DP06	0.695	0.817	0.664	0.703	0.714	0.815	0.675	0.722
Pressure drop on filter in CL of loop 3	YA03DP08	0.729	0.792	0.689	0.731	0.733	0.793	0.706	0.727
Pressure drop on MCP of loop 3	YA03DP09	0.721	0.820	0.655	0.706	0.734	0.817	0.667	0.720
Pressure drop in “cold” header of SG 3 YB03W01	YA03DP13	0.852	0.855	0.615	0.649	0.853	0.853	0.619	0.649
Pressure drop in “hot” header of SG 3 YB03W01	YA03DP14	0.922	0.905	0.740	0.717	0.924	0.903	0.754	0.711
Flow rate in the pump circulation loop YD03D01 (I 7)	YA03F01	0.653	0.734	0.501	0.410	0.653	0.734	0.501	0.410
Pressure drop in HL of loop 4 (elevation part)	YA04DP02	0.765	0.794	0.725	0.731	0.757	0.810	0.718	0.709
Pressure drop in CL of loop 4 (from SG outlet to the flow meter)	YA04DP04	0.852	0.831	0.712	0.759	0.862	0.842	0.712	0.758
Pressure drop in CL of loop 4 (from the flow meter to mechanical filter)	YA04DP05	0.856	0.832	0.725	0.747	0.862	0.841	0.726	0.748
Pressure drop in CL of loop 4 (section with mechanical filter, pump and regulating valve)	YA04DP06	0.816	0.808	0.615	0.722	0.823	0.819	0.611	0.725
Pressure drop on filter in CL of loop 4	YA04DP08	0.846	0.836	0.686	0.724	0.851	0.841	0.695	0.727
Pressure drop on MCP of loop 4	YA04DP09	0.817	0.805	0.513	0.647	0.824	0.814	0.509	0.651
Pressure drop in “cold” header of SG 4 YB04W01	YA04DP13	0.834	0.808	0.603	0.705	0.838	0.812	0.605	0.711
Pressure drop in “hot” header of SG 4 YB04W01	YA04DP14	0.834	0.841	0.754	0.756	0.830	0.838	0.752	0.733
Flow rate in the pump circulation loop YD04D01 (I 8)	YA04F01	0.650	0.692	0.525	0.629	0.650	0.693	0.525	0.630
Water level in SG1	YB01L01	0.908	0.886	0.797	0.689	0.902	0.875	0.776	0.692
Pressure of fluid in SG YB01W01	YB01P01	0.954	0.952	0.949	0.892	0.953	0.946	0.941	0.897
Water level in SG2	YB02L01	0.884	0.875	0.786	0.767	0.879	0.871	0.772	0.759
Pressure of fluid in SG YB02W01	YB02P01	0.958	0.937	0.960	0.911	0.958	0.949	0.956	0.917
Water level in SG3	YB03L01	0.927	0.920	0.846	0.837	0.926	0.919	0.833	0.842
Pressure of fluid in SG YB03W01	YB03P01	0.960	0.926	0.965	0.925	0.962	0.942	0.964	0.931
Water level in SG4	YB04L01	0.853	0.884	0.688	0.565	0.854	0.876	0.688	0.565
Pressure of fluid in SG YB04W01	YB04P01	0.955	0.951	0.952	0.897	0.955	0.948	0.945	0.903
Pressure drop on DC (upper part with inlet chamber)	YC01DP01	0.842	0.817	0.754	0.760	0.841	0.813	0.756	0.761
Pressure drop on DC (inlet in the middle part)	YC01DP02	0.751	0.904	0.584	0.716	0.822	0.911	0.662	0.733
Pressure drop on DC (middle part)	YC01DP03	0.722	0.885	0.689	0.731	0.739	0.888	0.683	0.777
Pressure drop on DC (outlet from the middle part)	YC01DP04	0.757	0.817	0.776	0.772	0.768	0.817	0.781	0.790
Pressure drop in lower DC chamber	YC01DP05	0.815	0.892	0.826	0.830	0.814	0.885	0.828	0.828

Table 12 Evaluation of FOM for CL-4.1-03 Test (Cont'd)

		TRACE V5 patch3				TRACE V5 patch4			
		all	W1	W2	W3	all	W1	W2	W3
Parameter	TAG	0 - 2593	0 - 400	400 - 1470	1470 - 2593	0 - 2593	0 - 400	400 - 1470	1470 - 2593
Pressure drop between DC LC and CS	YC01DP06	0.732	0.762	0.750	0.772	0.734	0.788	0.734	0.781
Pressure drop in the channel with FRS (lower part)	YC01DP07	0.667	0.753	0.661	0.664	0.658	0.724	0.644	0.660
		TRACE V5 patch3				TRACE V5 patch4			
		all	W1	W2	W3	all	W1	W2	W3
Parameter	TAG	0 - 2593	0 - 400	400 - 1470	1470 - 2593	0 - 2593	0 - 400	400 - 1470	1470 - 2593
Pressure drop in the channel with FRS (inlet in the middle part)	YC01DP08	0.691	0.756	0.648	0.687	0.689	0.751	0.646	0.679
Pressure drop in the channel with FRS (outlet from the middle part)	YC01DP09	0.674	0.759	0.602	0.672	0.674	0.762	0.602	0.672
Pressure drop in the channel with FRS (upper part)	YC01DP10	0.753	0.837	0.735	0.705	0.757	0.835	0.731	0.719
Pressure drop between CS and UP	YC01DP11	0.691	0.686	0.700	0.678	0.703	0.724	0.707	0.687
Pressure drop in UP (on the insert)	YC01DP12	0.710	0.756	0.717	0.702	0.716	0.781	0.732	0.700
Pressure drop in UP (lower part)	YC01DP13	0.716	0.725	0.684	0.701	0.737	0.762	0.705	0.726
Pressure drop in UP (middle part)	YC01DP14	0.804	0.812	0.684	0.690	0.815	0.831	0.685	0.694
Pressure drop in UP (top piece)	YC01DP15	0.797	0.747	0.571	0.439	0.854	0.801	0.609	0.441
Pressure drop between DC and UP	YC01DP16	0.714	0.717	0.545	0.674	0.713	0.723	0.535	0.671
Pressure drop on CS BP	YC01DP17	0.723	0.829	0.672	0.681	0.719	0.842	0.637	0.689
Pressure of coolant in UP	YC01P16	0.962	0.950	0.855	0.859	0.965	0.955	0.871	0.888
Water level in PRZ	YP01L02	0.945	0.942	0.515	0.582	0.944	0.942	0.515	0.582
Pressure of fluid in PRZ YP01B01	YP01P01	0.962	0.948	0.850	0.855	0.966	0.954	0.866	0.896
FOM total avg		0.797	0.802	0.684	0.674	0.804	0.807	0.686	0.681

6 RUN STATISTICS

The transients were calculated on calculation server with Intel Xeon E5-2643 v2 processor 3.50 GHz under GNU/Linux CentOS 7.4.1708. The run statistics is shown in the following Table 13. The TRACE patch04 calculation run twice faster than patch03 did due more stable calculation mainly upon transient conditions.

Table 13 Run Statistics

	TRACE V5.0 patch03	TRACE V5.0 patch04
Number of components	330	330
Number of time steps	249 194	141 812
Transient time	3 800 s	3 800 s
CPU time	24 909 s	12 632s
CPU time / Transient time	0.153	0.301

7 CONCLUSIONS

The main goal of these analyses was to assess the TRACE code using the cold leg break 4.1% CL-4.1-03 in the large scale test facility PSB VVER. The second reason was to assess changes of the code TRACE V5.0 from version patch03 to patch04. Before doing the final calculation a complex sensitivity study of the impact of various parameters and code options on the prediction of the rod heaters cladding temperature was performed. For the final calculations the following settings were applied:

- The level tracking model in the core and UP region was activated
- The finer discretization of the upper part of the core (the height of two top layers of the core region is half of the rest of the core)
- TRACE V5.0 patch03 and patch04 were used

A part of these analyses is quantitative assessment of agreement of the calculations against the experiment data that can help identify pros and cons of an applied way of modeling integral test facility.

Comparisons of post-test TRACE calculations with experimental data proved that both version of the TRACE code are capable to model PSB-VVER integral system effects reasonably. The calculated time courses of the main facility parameters were similar to that of the test, indicating that all of the significant events that occurred in the test were reproduced by in the calculation

To quantify errors/deviations of presented TRACE calculations a set of the figure of merit (FOM) was evaluated using software ACAP. FOMs of 71 main measured and calculated parameters were evaluated analogously for both TRACE calculations. The following table shows the final average FOM evaluated for both calculations at pre-defined time windows of interest.

- W0: 0 ÷ 2593 s – the whole transient
- W1: 0 ÷ 400 s – an early stage of the test when no ECCS flow was provided
- W2: 400 ÷ 1470 s – a middle stage of the test when accumulators flow was provided
- W3: 1470 ÷ 2593 s – the final stage of the test when ECCS LPIS flow was provided

Table 14 Final Average FOM Evaluated for Pre-defined Time Windows

	FOM avg			
	W0	W1	W2	W3
Time	0 – 2593	0 – 400	400 – 1470	1470-2593
TRACE V5.0 patch03	0.797 very good prediction	0.802 very good prediction	0.684 good prediction	0.674 good prediction
TRACE V5.0 patch04	0.804 very good prediction	0.807 very good prediction	0.686 good prediction	0.681 good prediction

Presented overall final FOMs (time window W0) prove that the both code versions correctly predicted behavior of test facility during the whole transient, although the version patch 04 prediction seems “slightly” better. It is clearly visible that better agreements were reached during the early stage of the transient whereas the worst agreements were identified in the end transient. These results correspond to the duration of the CL-4.1-03 test (2593 s), that was longer than “common” LOCA tests with duration of tens of seconds. The accumulation of minor deviations might lead to gradual increasing of deviations of main calculated parameters.

Despite the quantitative assessment gave mainly good or very good predictions of the selected main parameters following particular discrepancies were identified. The TRACE code predicted a little bit slower decrease of liquid water in the upper part of the core during the first stage of the experiment so the first temperature peak between the 97 s and 105 s was not predicted. Although the calculated water distribution over the reactor vessel and circulation loops was in a good compliance with experiment data, there are two regions where the TRACE code predicted different liquid content: the middle part of the core and loop seals (a part including the MCP and the filter). The calculated liquid content of the middle part of the core was somewhat lower than measured one. The calculated rate of clearance of the loop seals of loops with no break is slower in case the loop#1 and faster in case the loops #2 and #3. But these discrepancies did not significantly affected predicted heat-up of the upper part of the core, where it was expected to happen first.

The calculation cost of the TRACE patch03 calculation was twice higher than TRACE patch04 calculation. The better performance of the TRACE patch04 calculation was influenced by new enhanced features e.g. AUTO backup time step so time coarse of time step was more stable than in case using version patch03 than does not includes these features.

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APPENDIX A

INPUT DECK NODALISATION SCHEMES

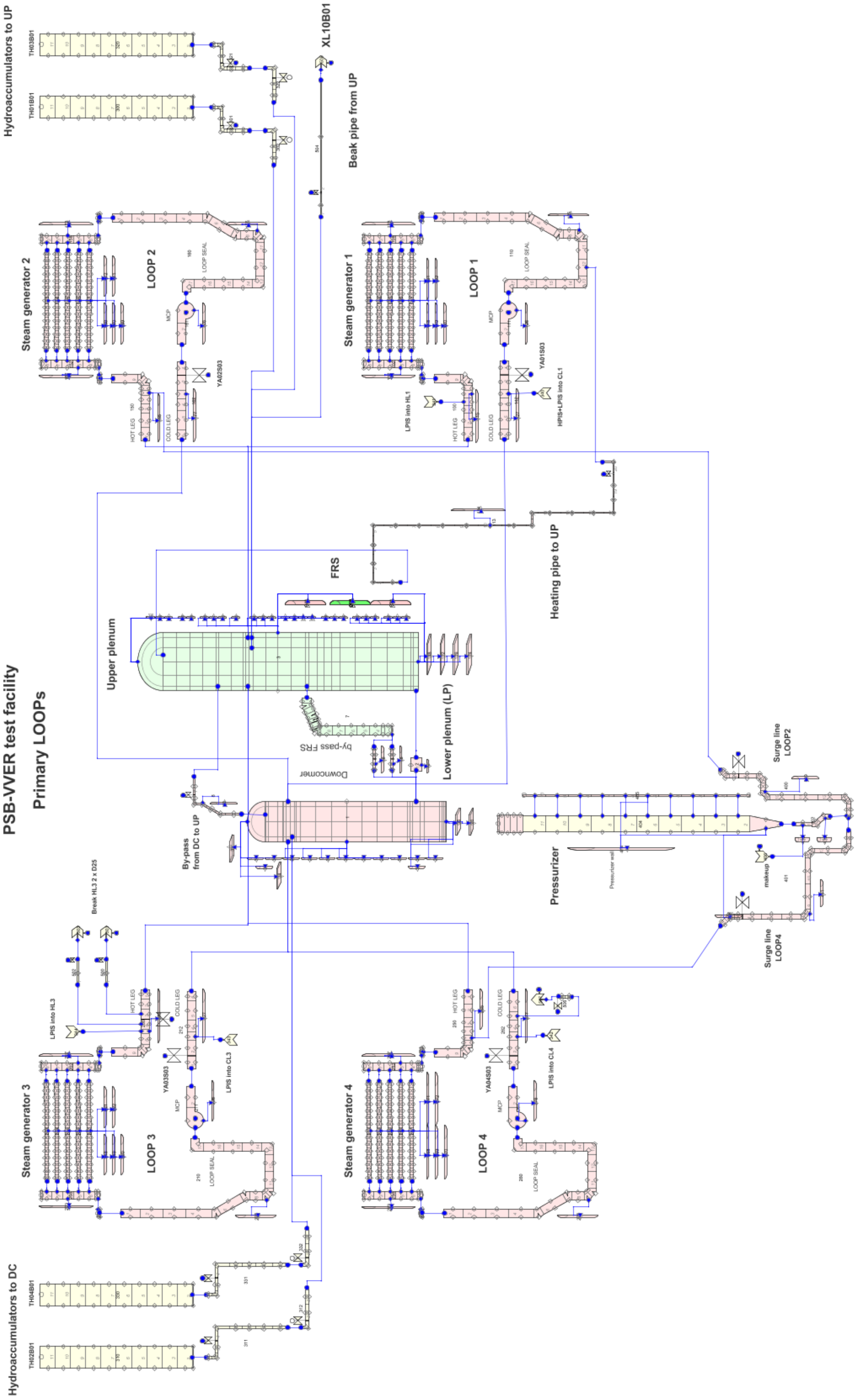

Figure A-1 TRACE Nodalization Scheme of Primary Circuit of PSB-VVER Facility

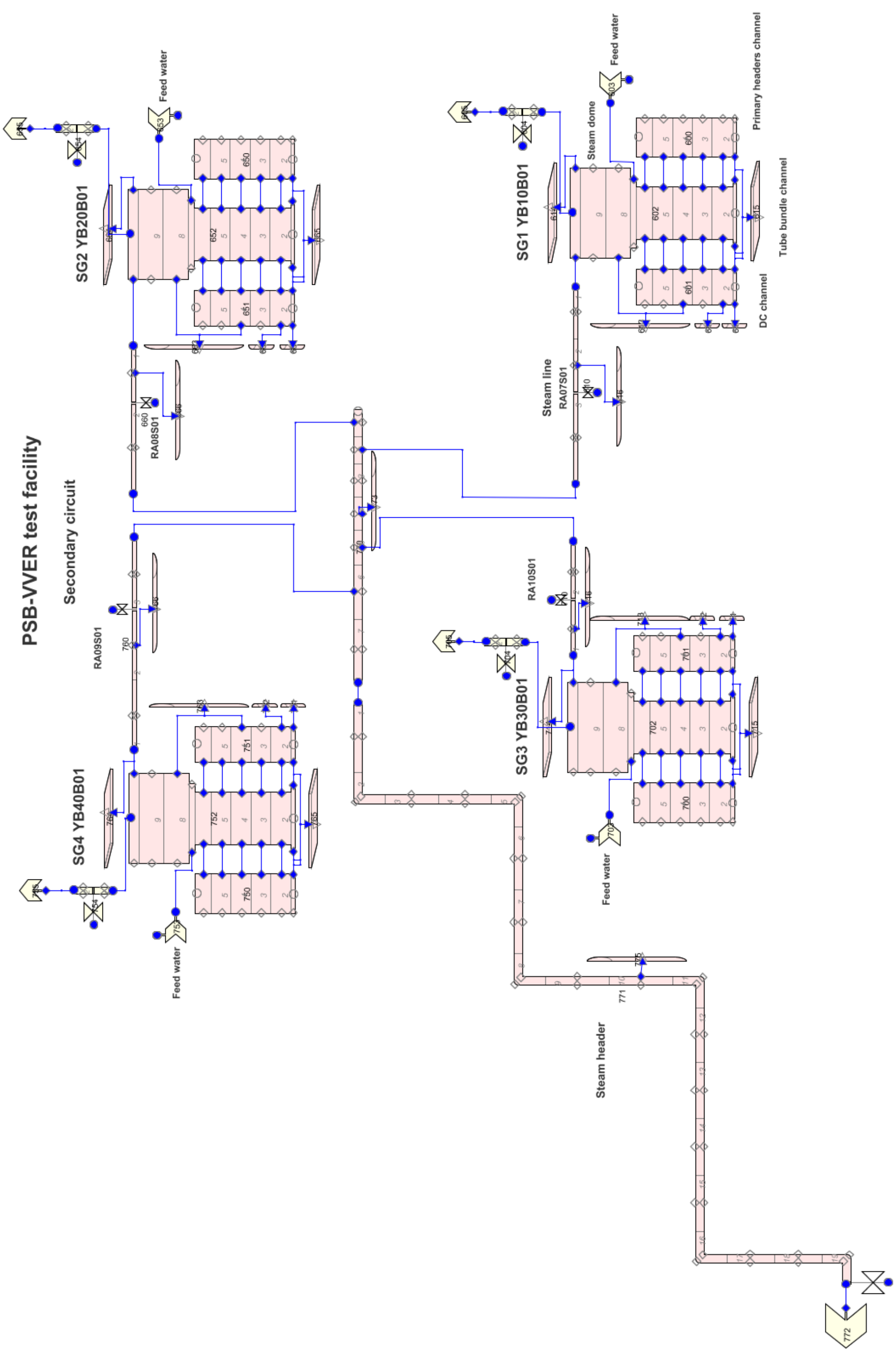

Figure A-2 TRACE Nodalization Scheme of Secondary Circuit of PSB-VVER Facility

APPENDIX B

THERMAL-HYDRAULIC DIAGRAM AND MEASUREMENT LOCALISATION AT PSB-VVER FACILITY

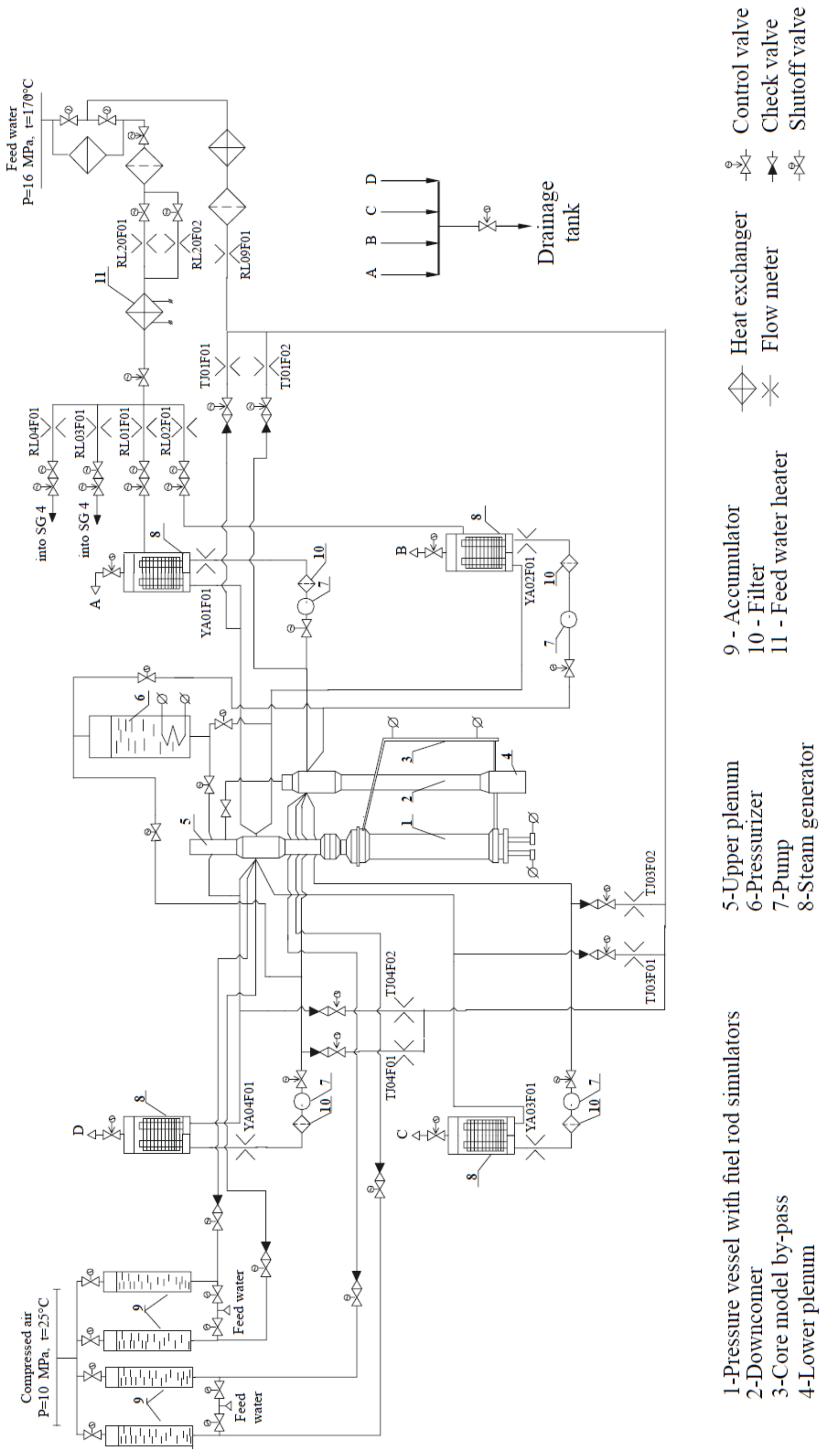

Figure B-1 PSB-VVER Thermal-Hydraulic Diagram

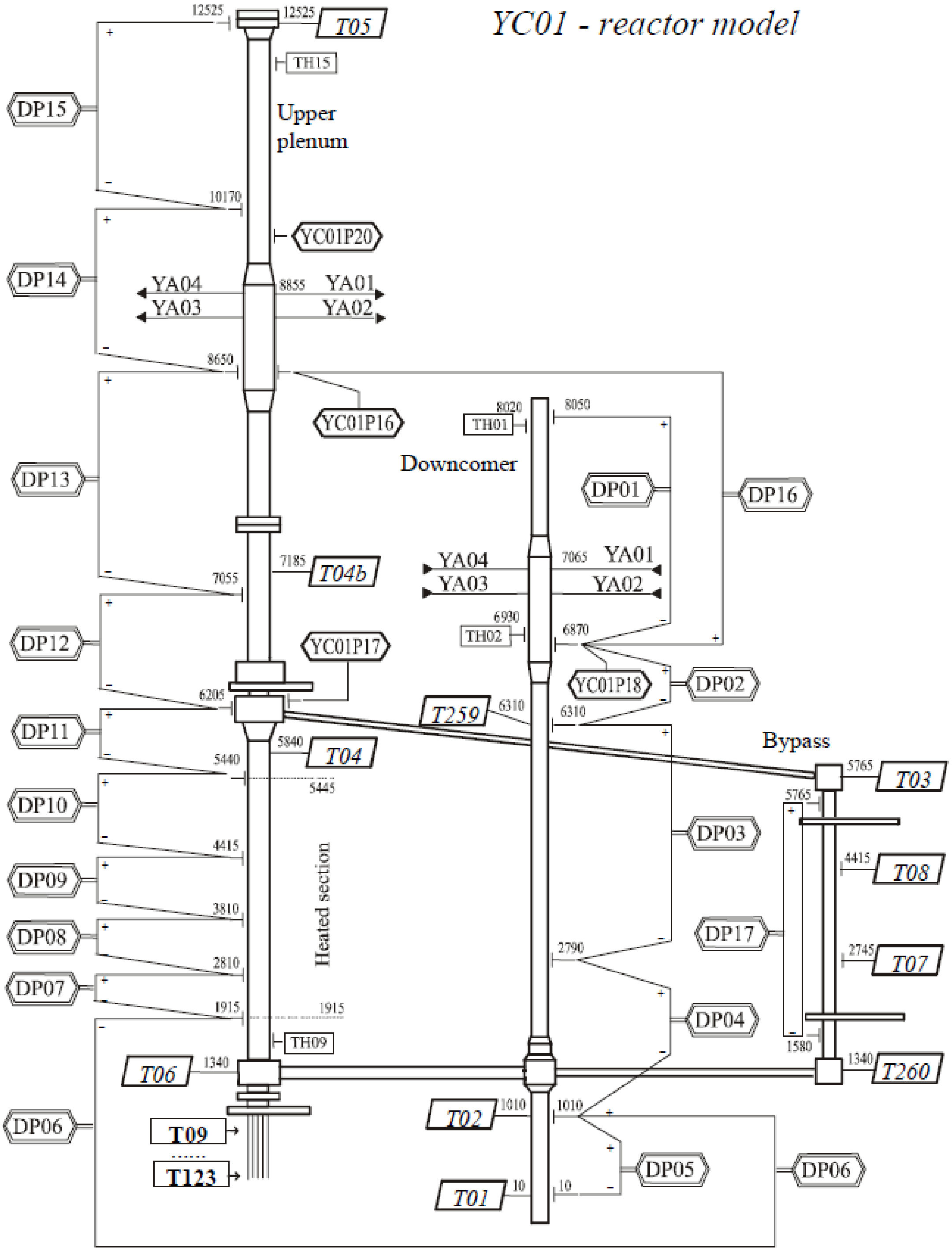

Figure B-2 PSB-VVER Reactor Model Measurements

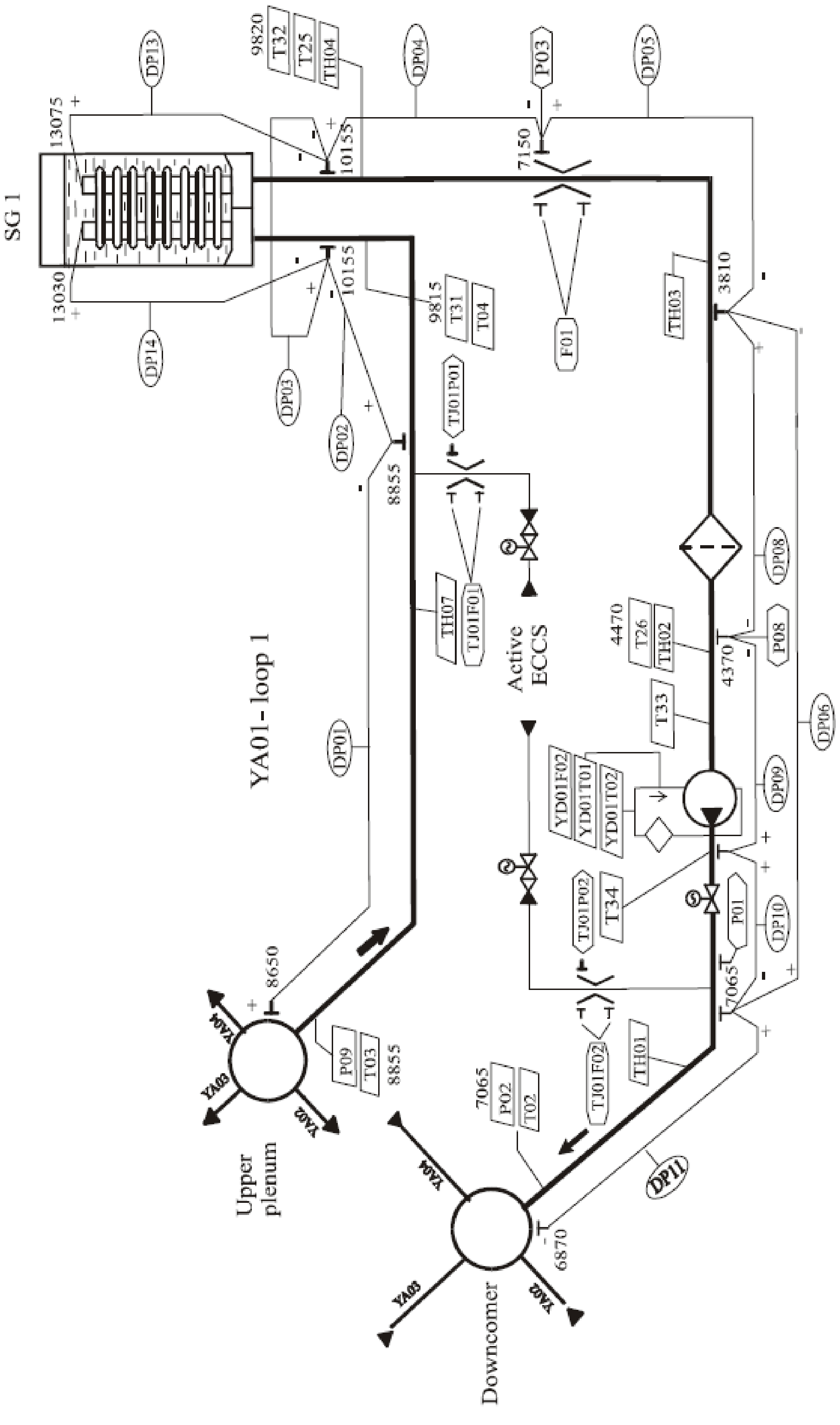

Figure B-3 PSB-VVER Loop 1 and SG-1 Model Measurement

APPENDIX C

COMPLETE SET OF COMPARISON PLOTS FOR TRACE CALCULATION

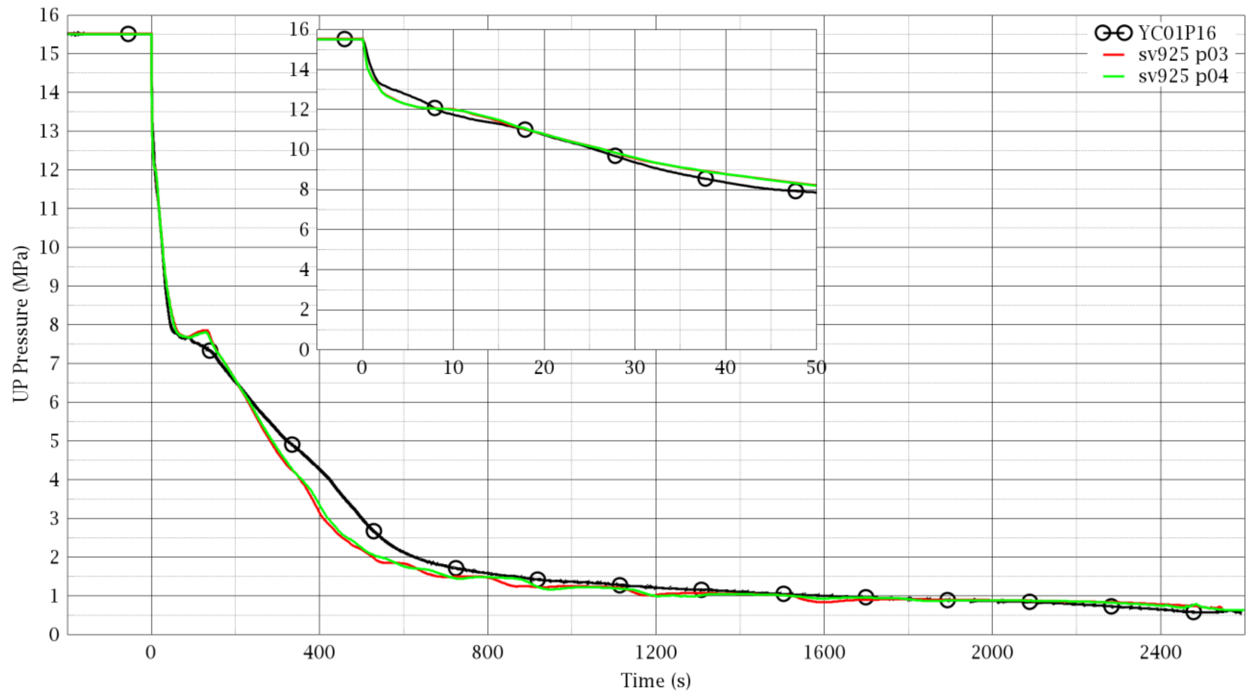

Figure C-1 Primary Pressure

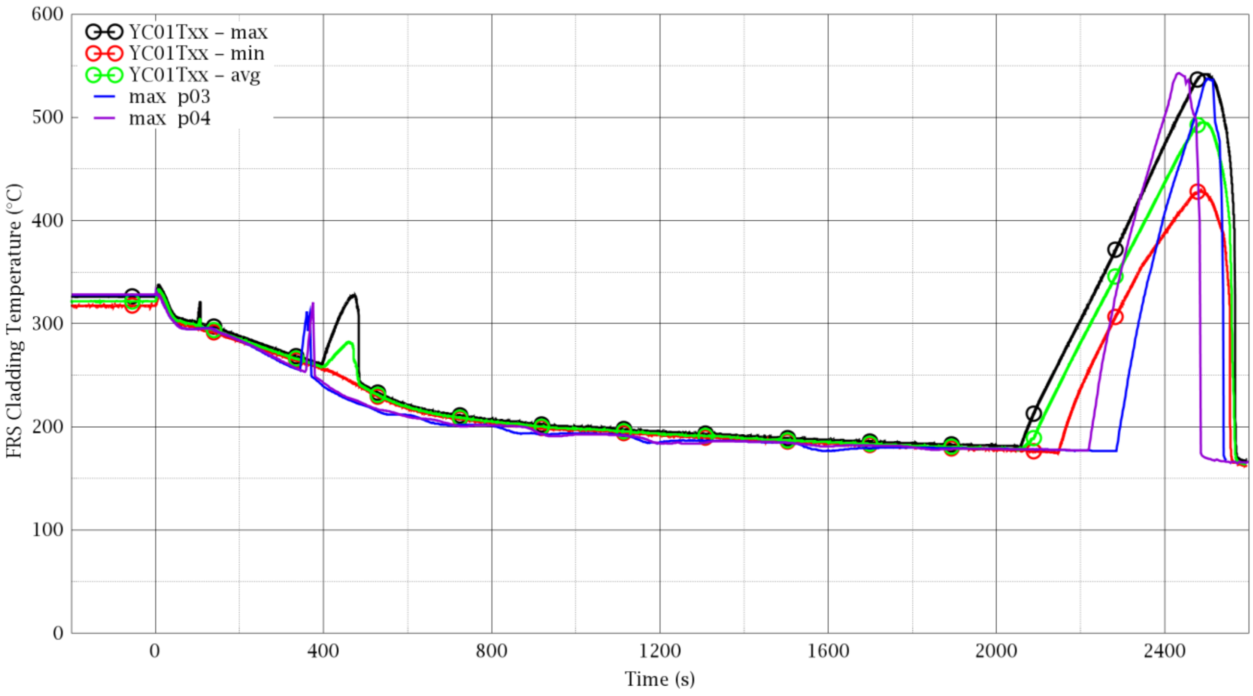

Figure C-2 Fuel Cladding Temperature (Top of the Core)

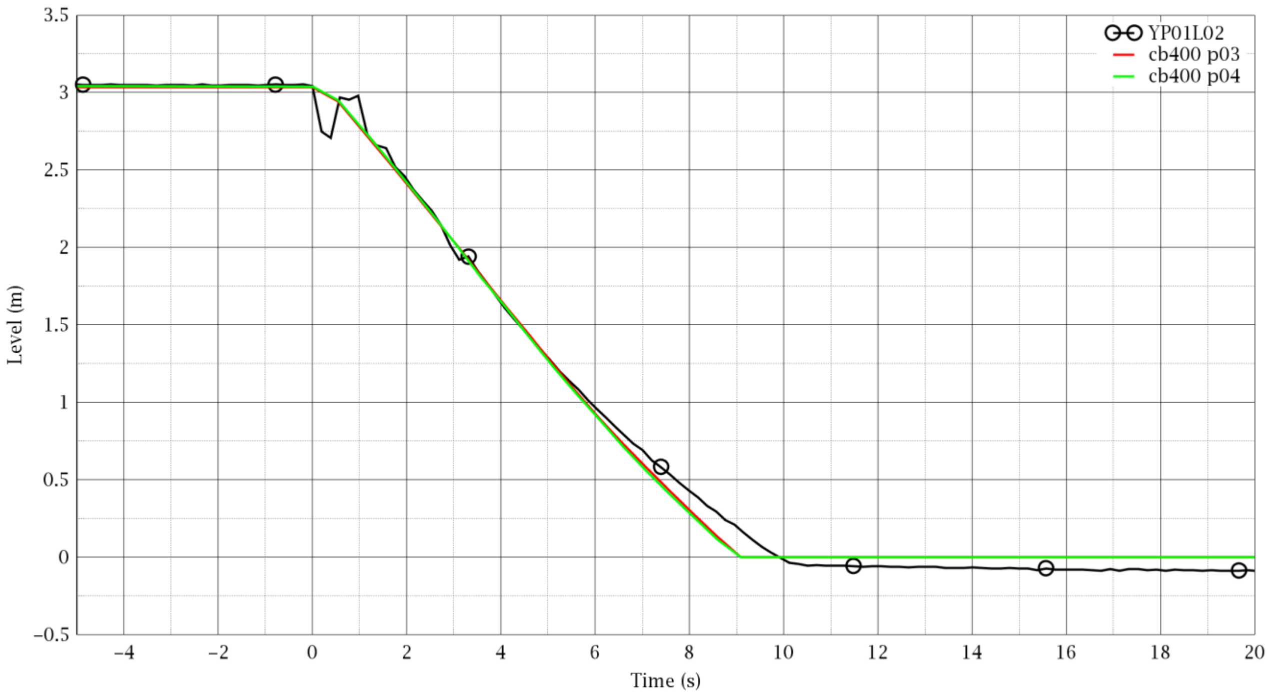

Figure C-3 Pressurizer Level

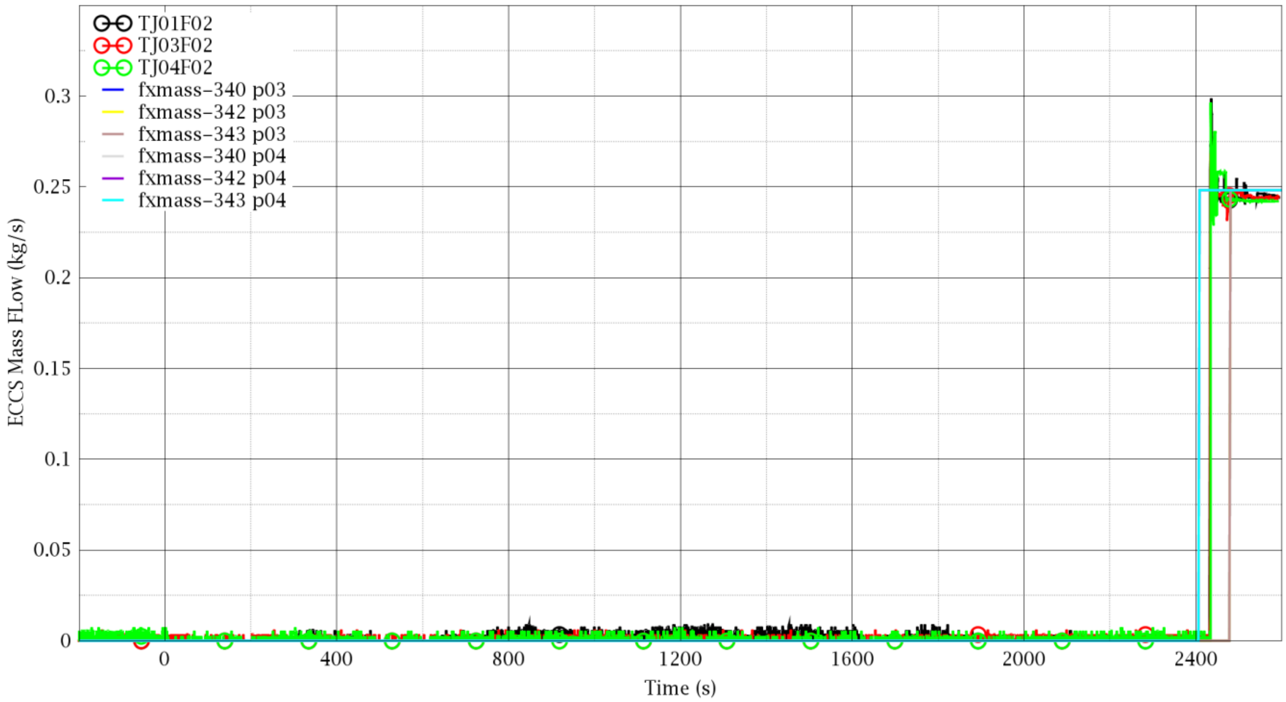

Figure C-4 LPIS Flow (Boundary Condition)

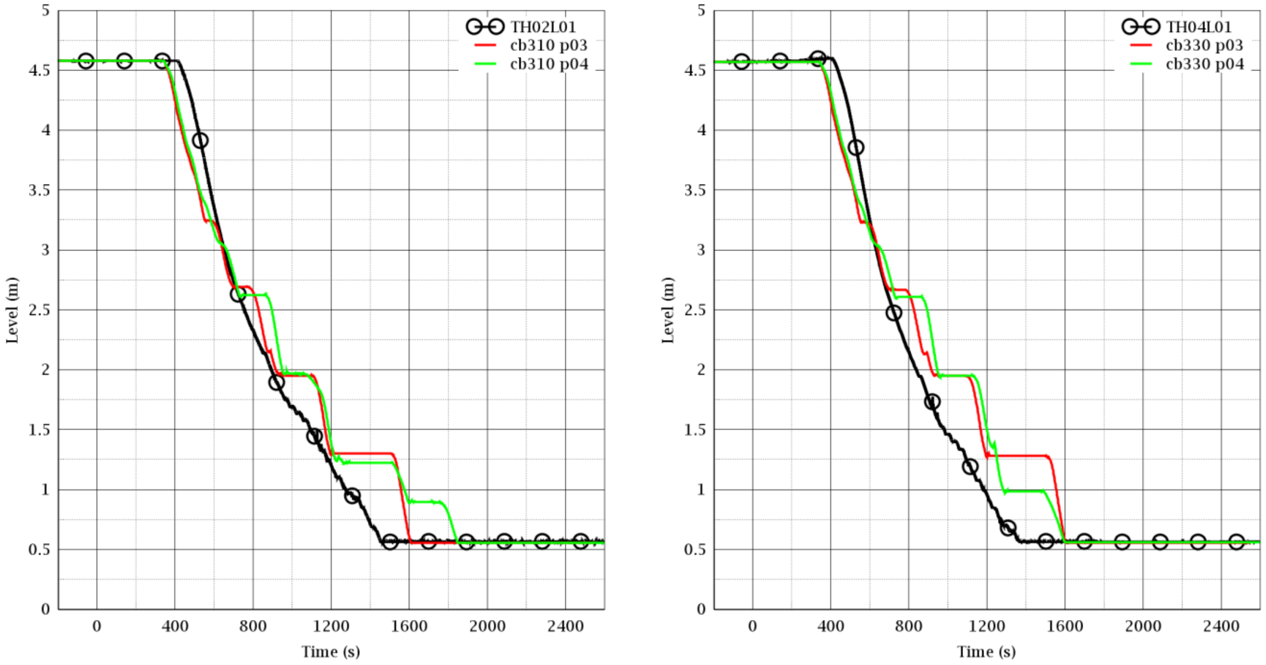

Figure C-5 Accumulators Levels

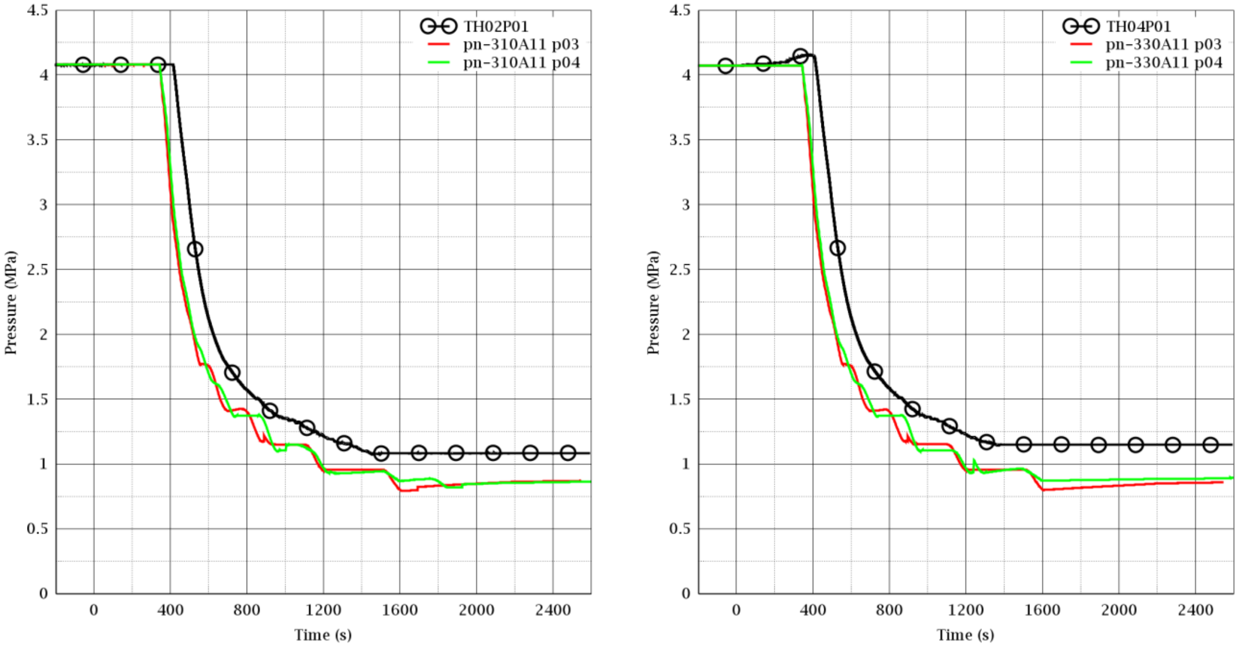

Figure C-6 Accumulators Pressures

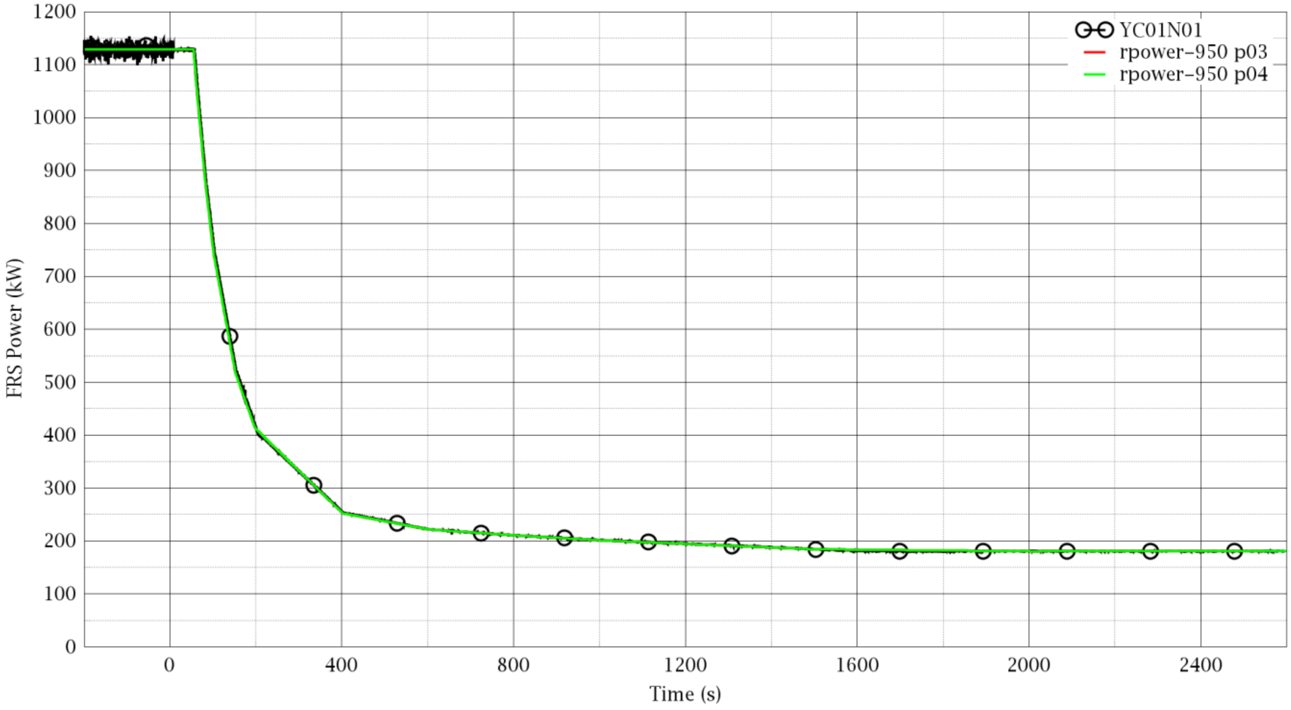

Figure C-7 Fuel Rod Simulator Power (Boundary Condition)

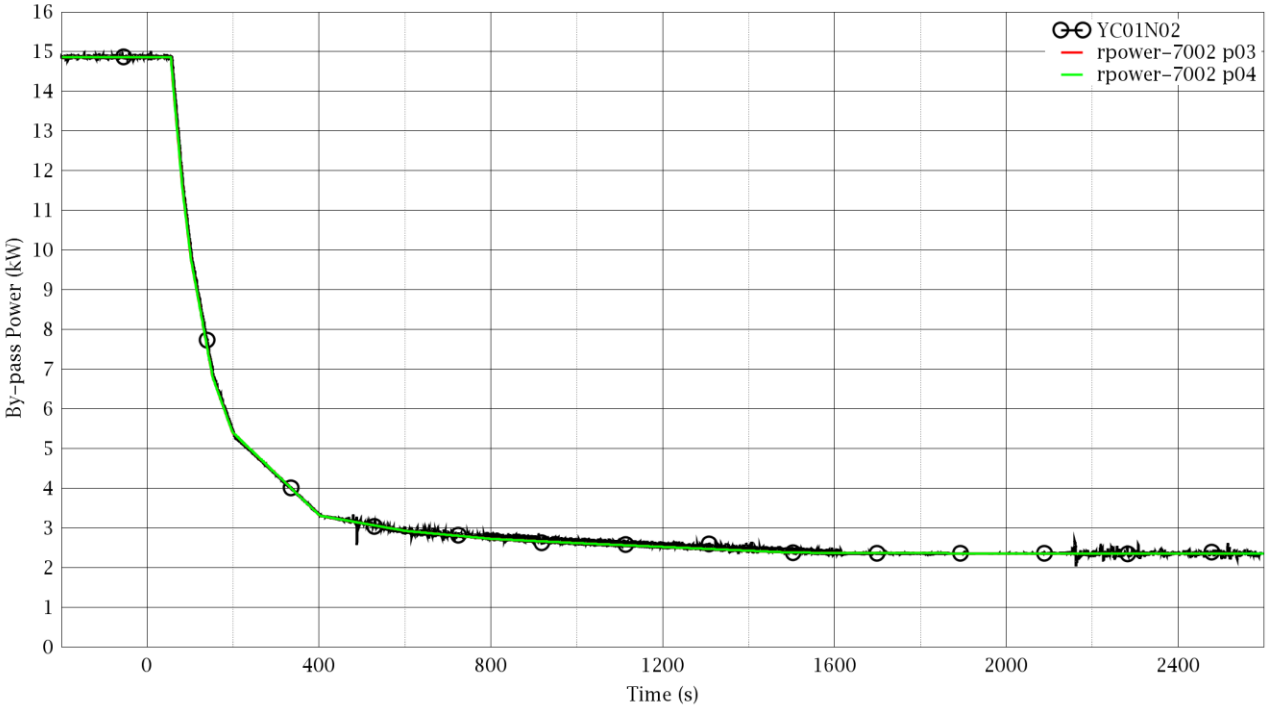

Figure C-8 Core By-pass Power (Boundary Condition)

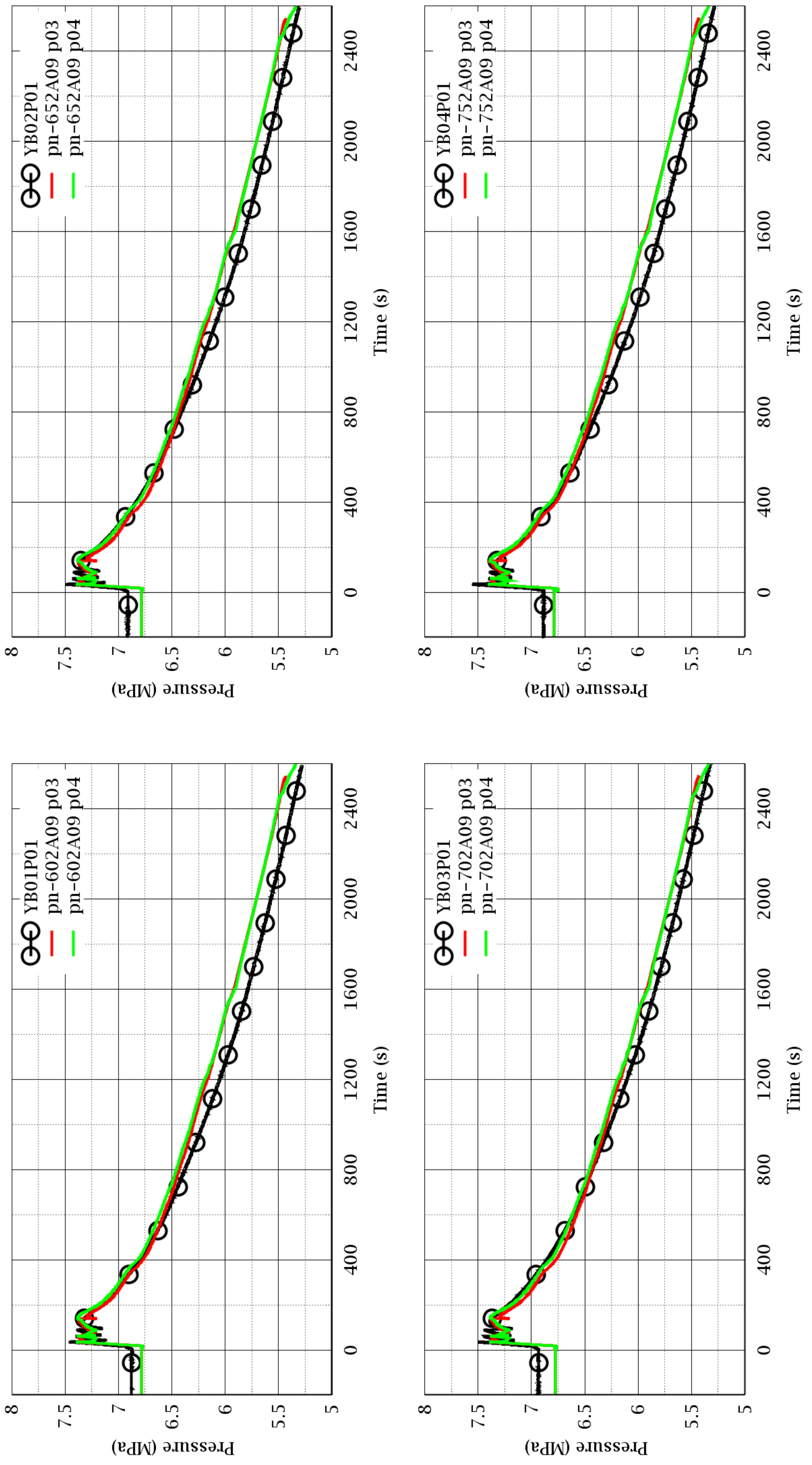

Figure C-9 Secondary Side Pressures

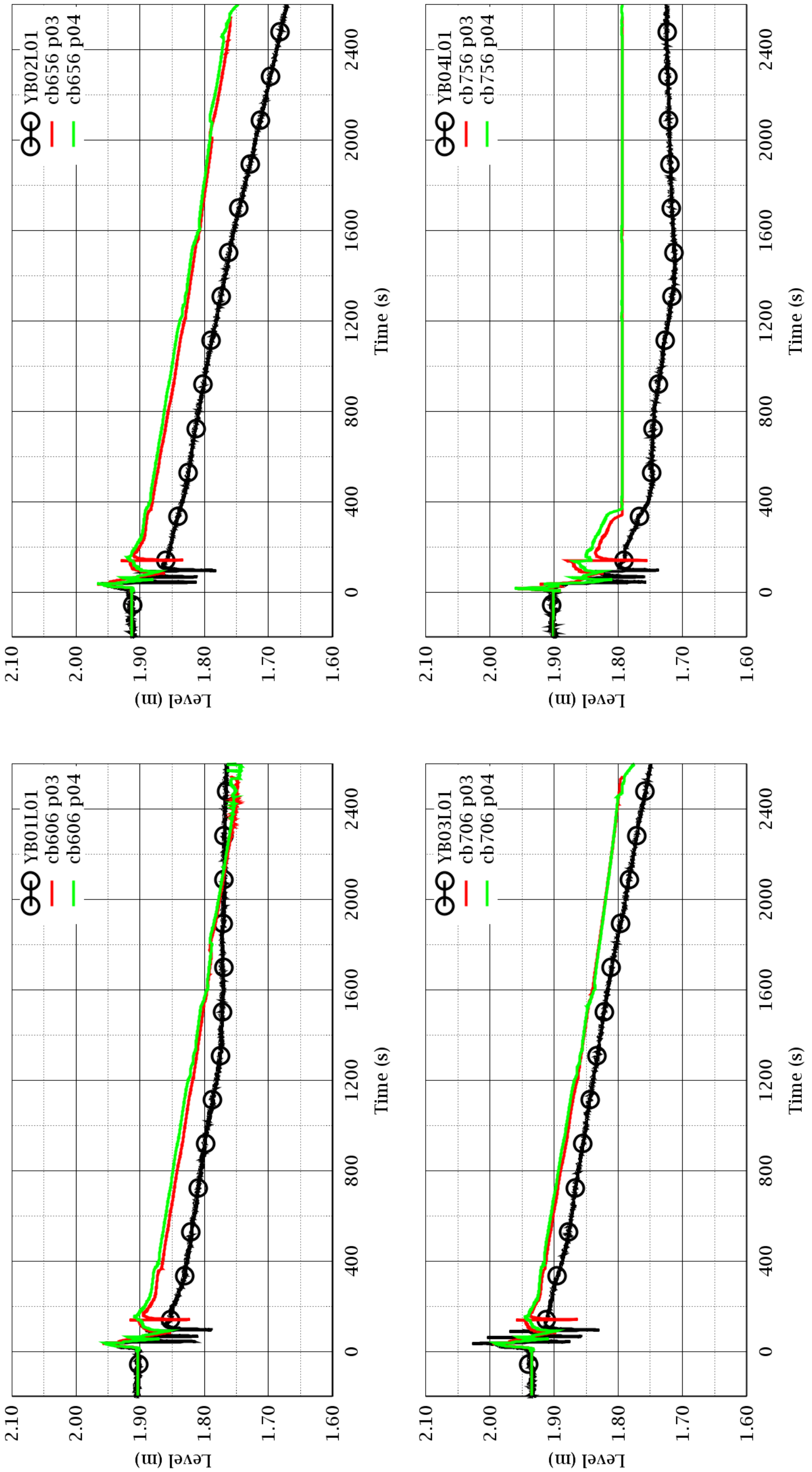

Figure C-10 Steam Generators Levels

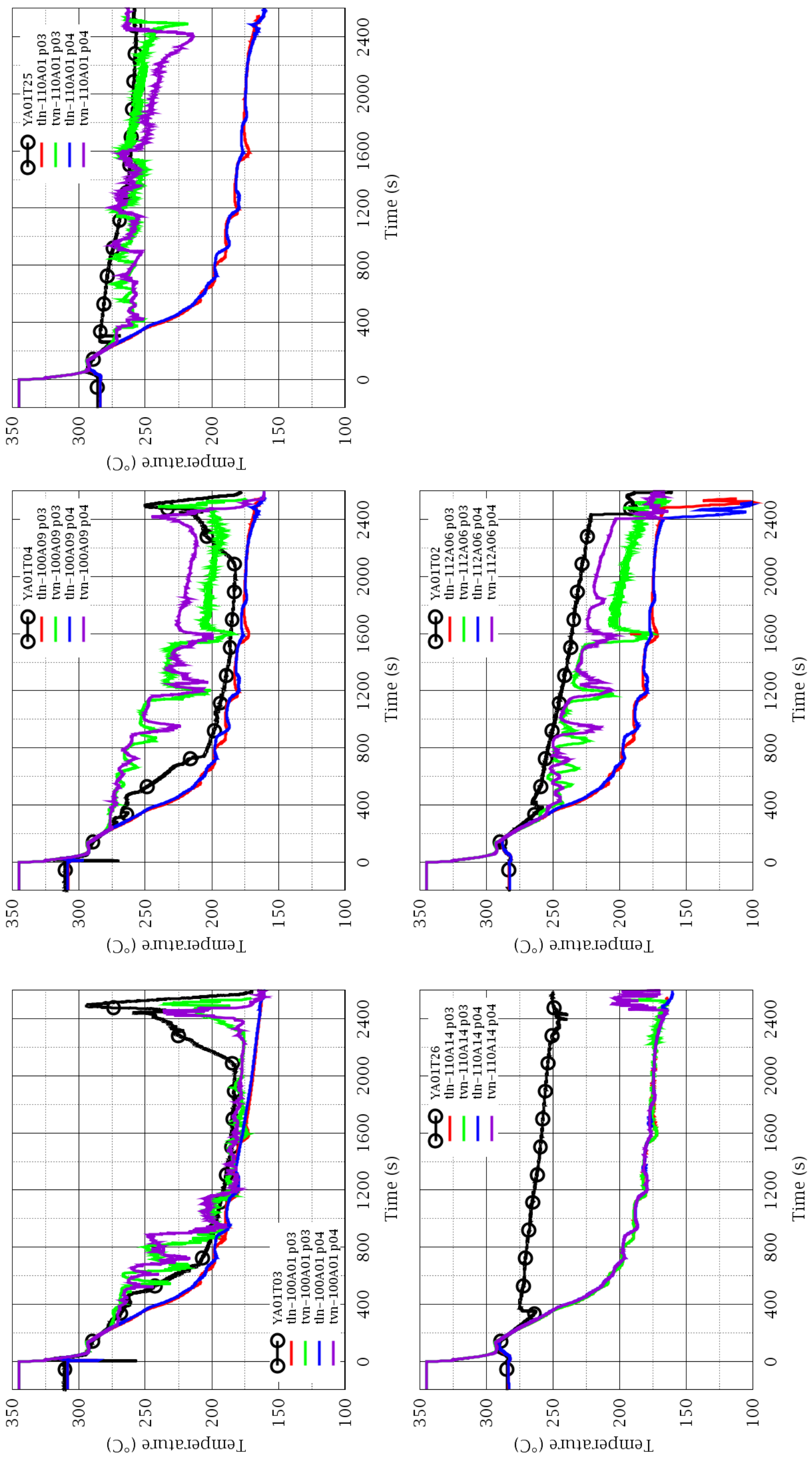

Figure C-11 Loop 1 Temperatures

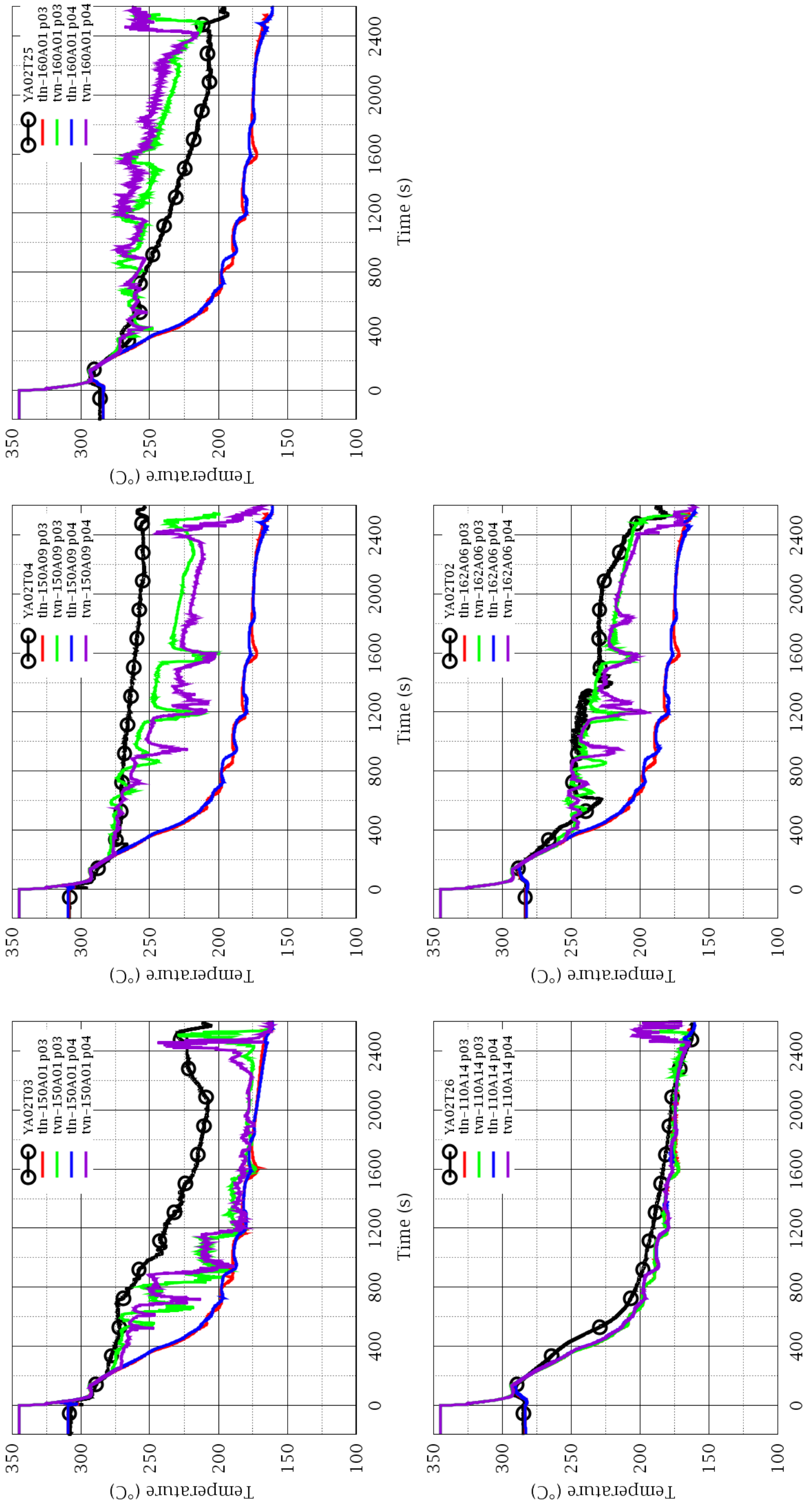

Figure C-12 Loop 2 Temperatures

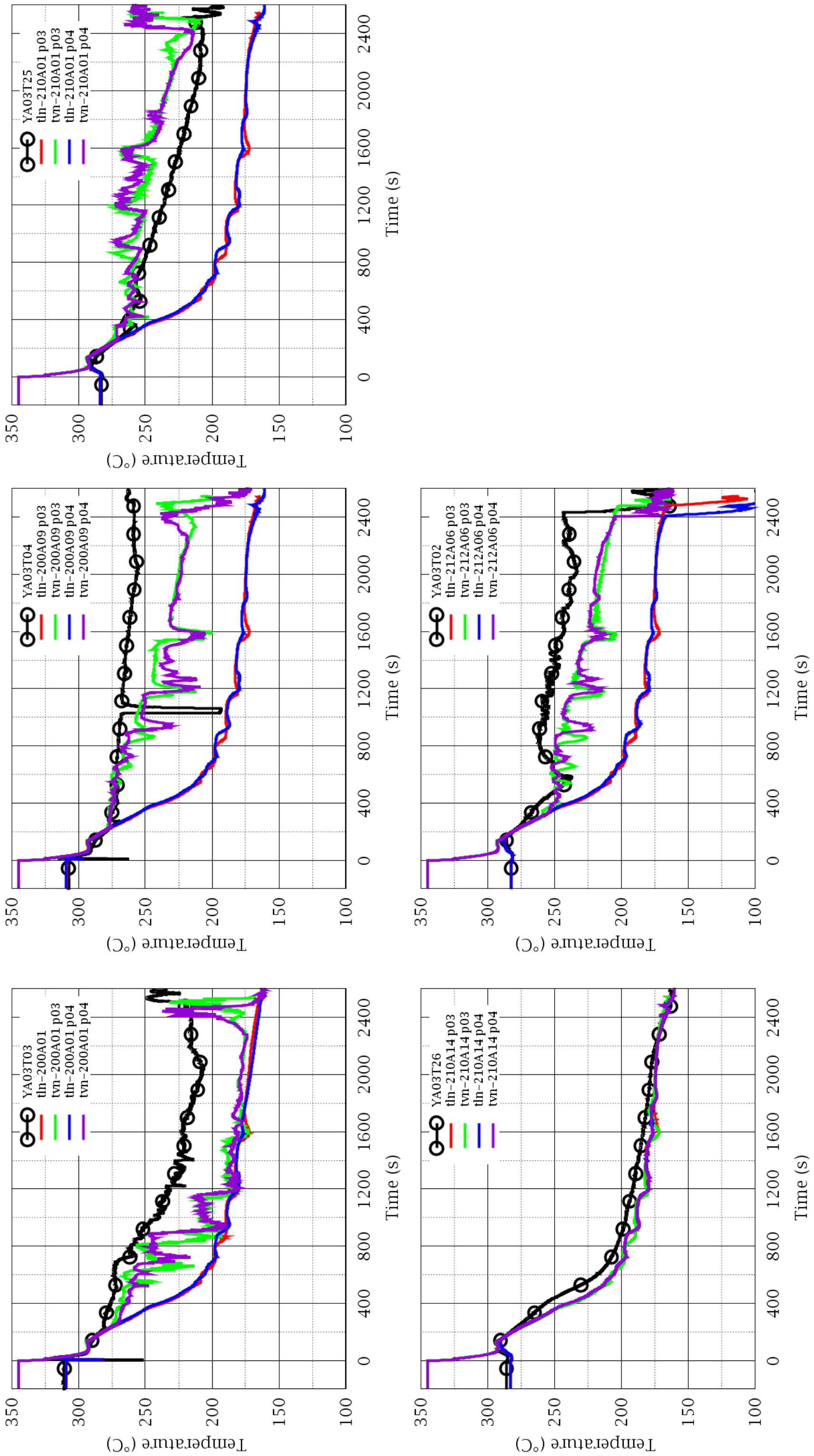

Figure C-13 Loop 3 Temperatures

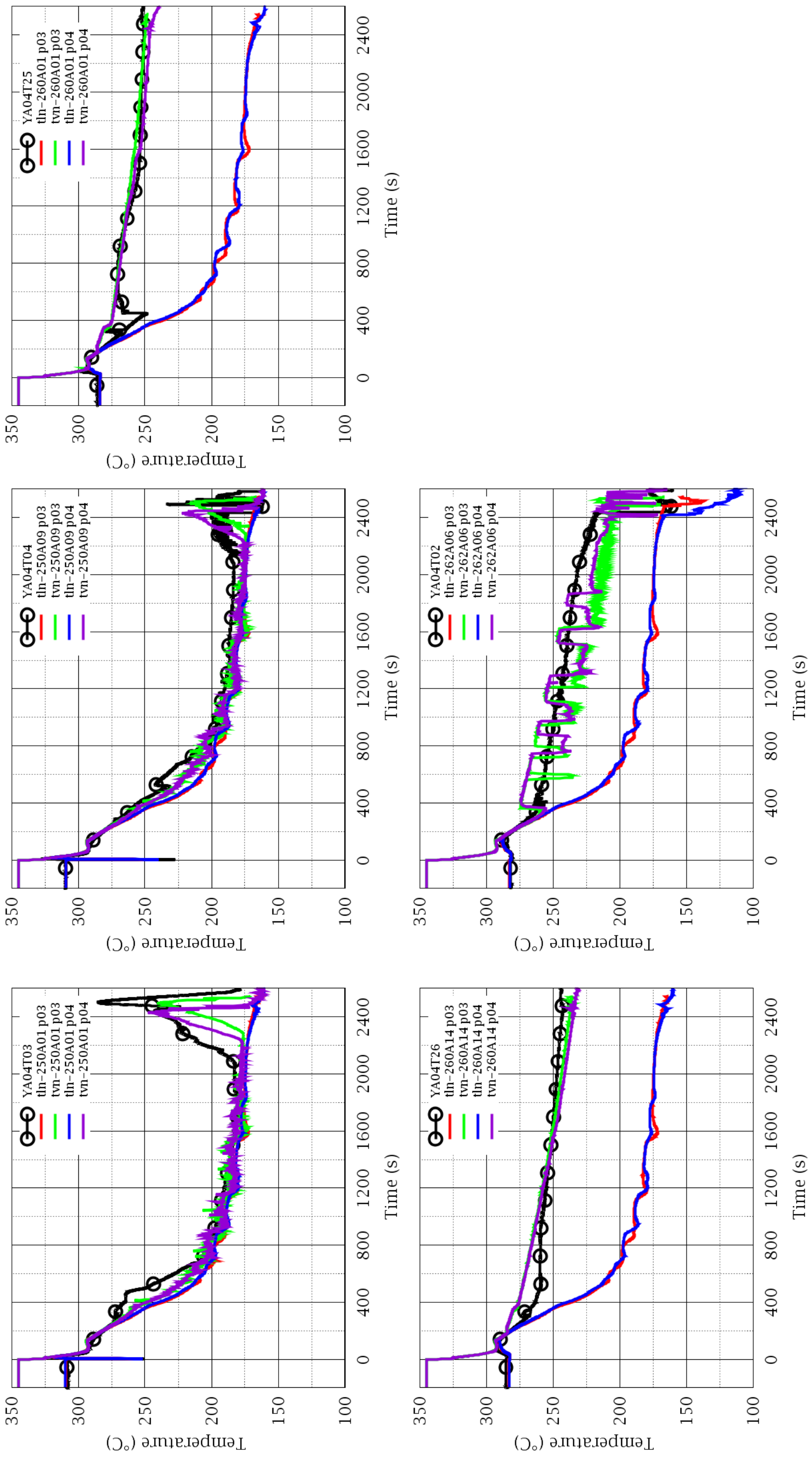

Figure C-14 Loop 4 Temperatures

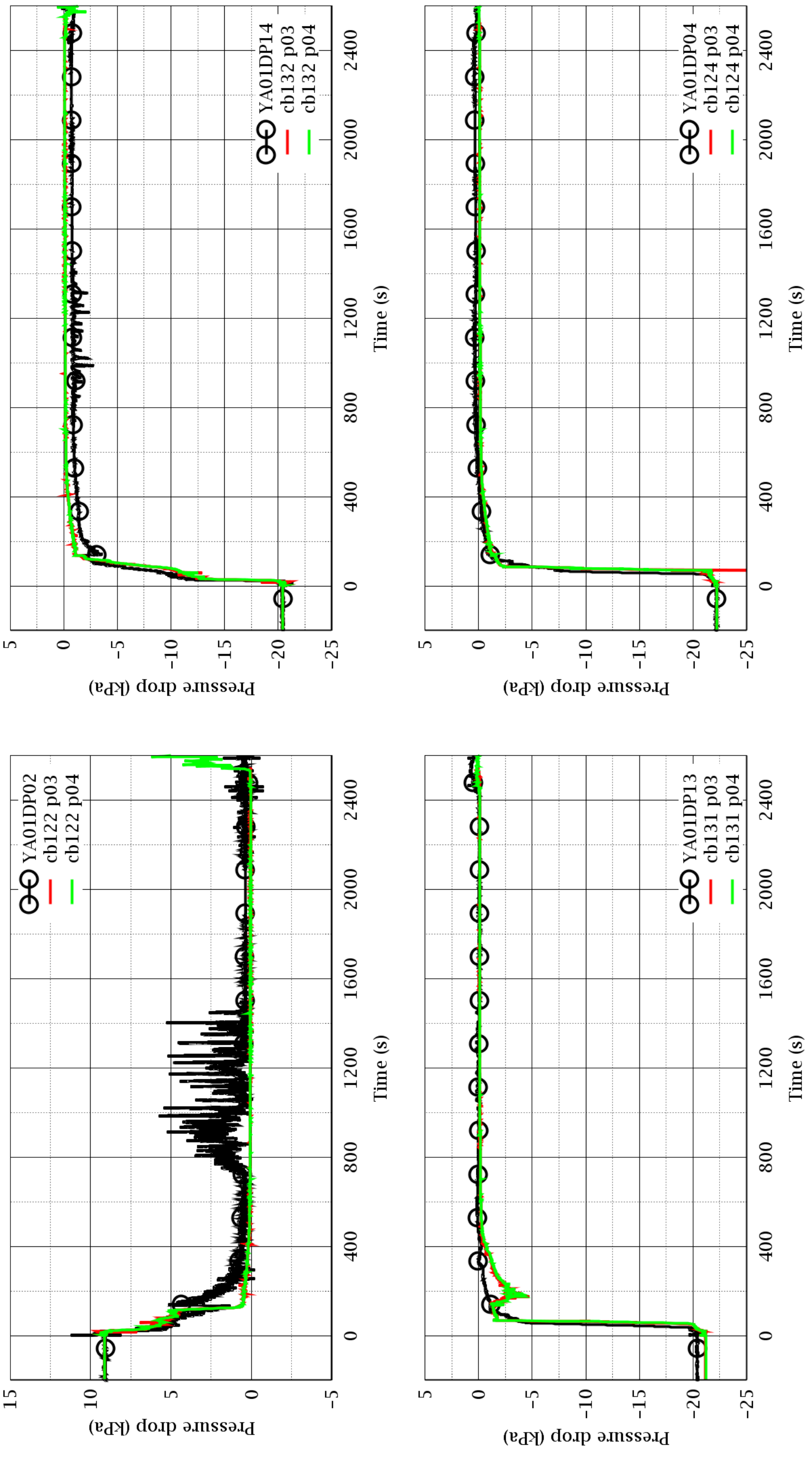

Figure C-15 Loop 1 Pressure Differences 1

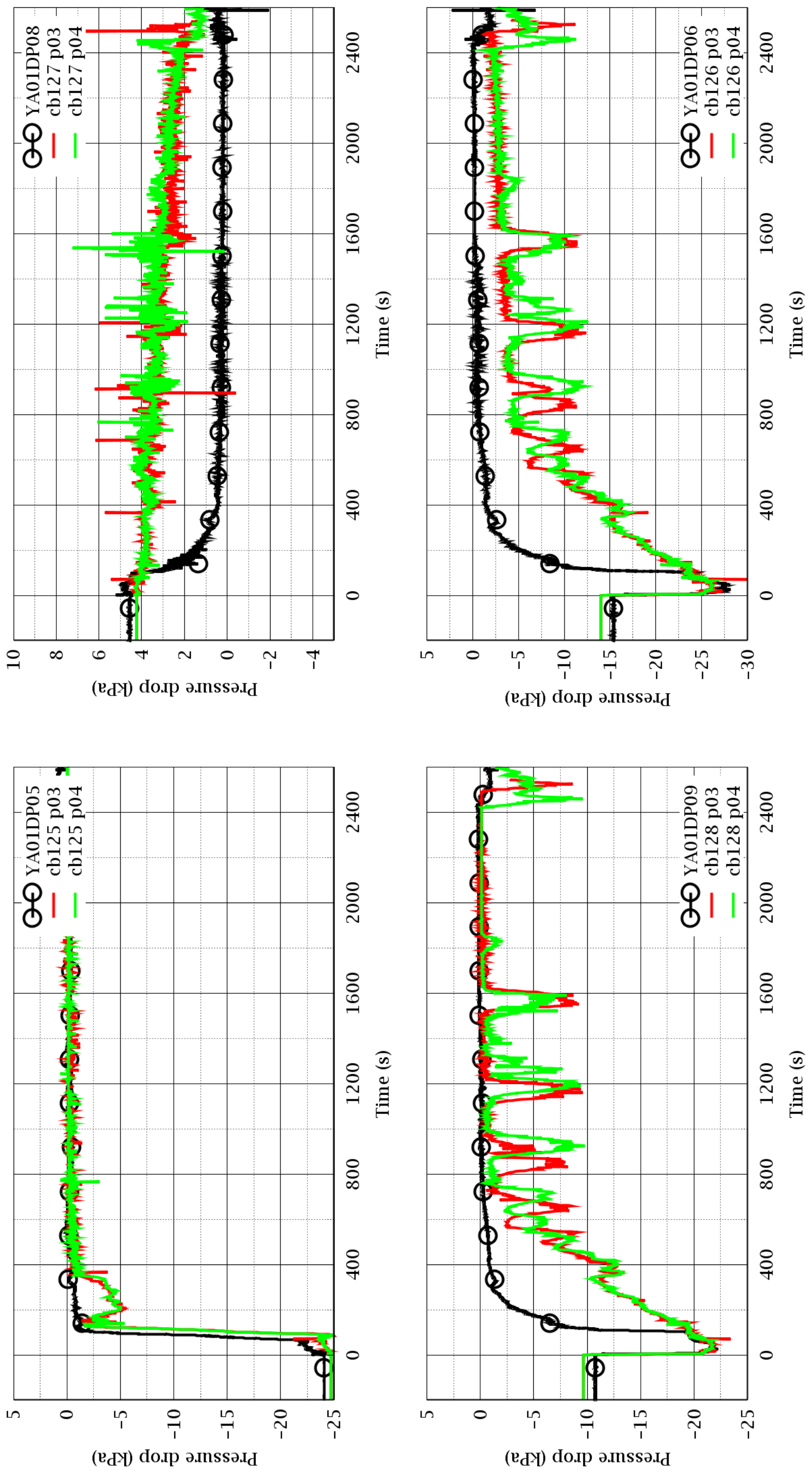

Figure C-16   Loop 1 Pressure Differences 2

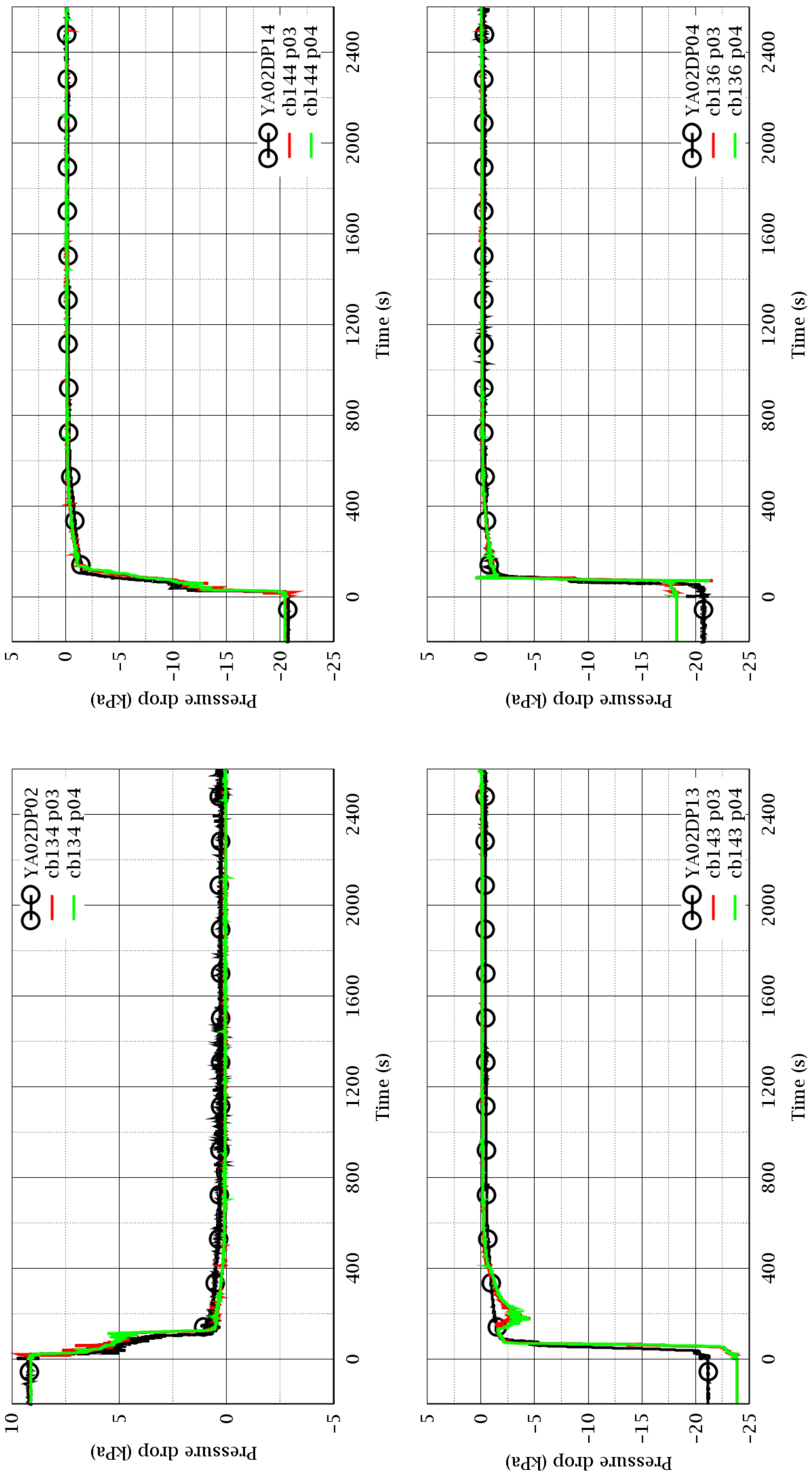

Figure C-17 Loop 2 Pressure Differences 1

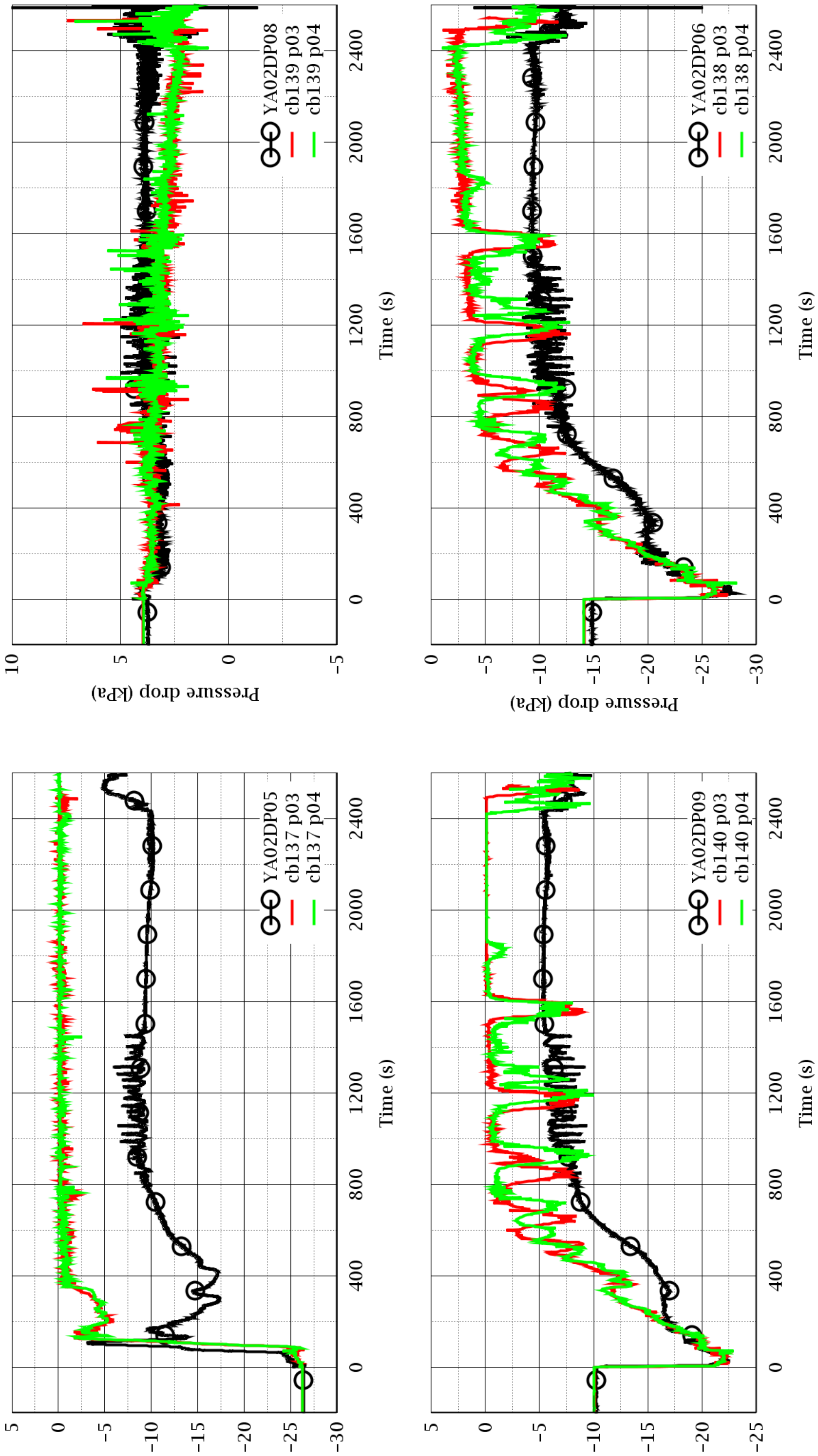

Figure C-18   Loop 2 Pressure Differences 2

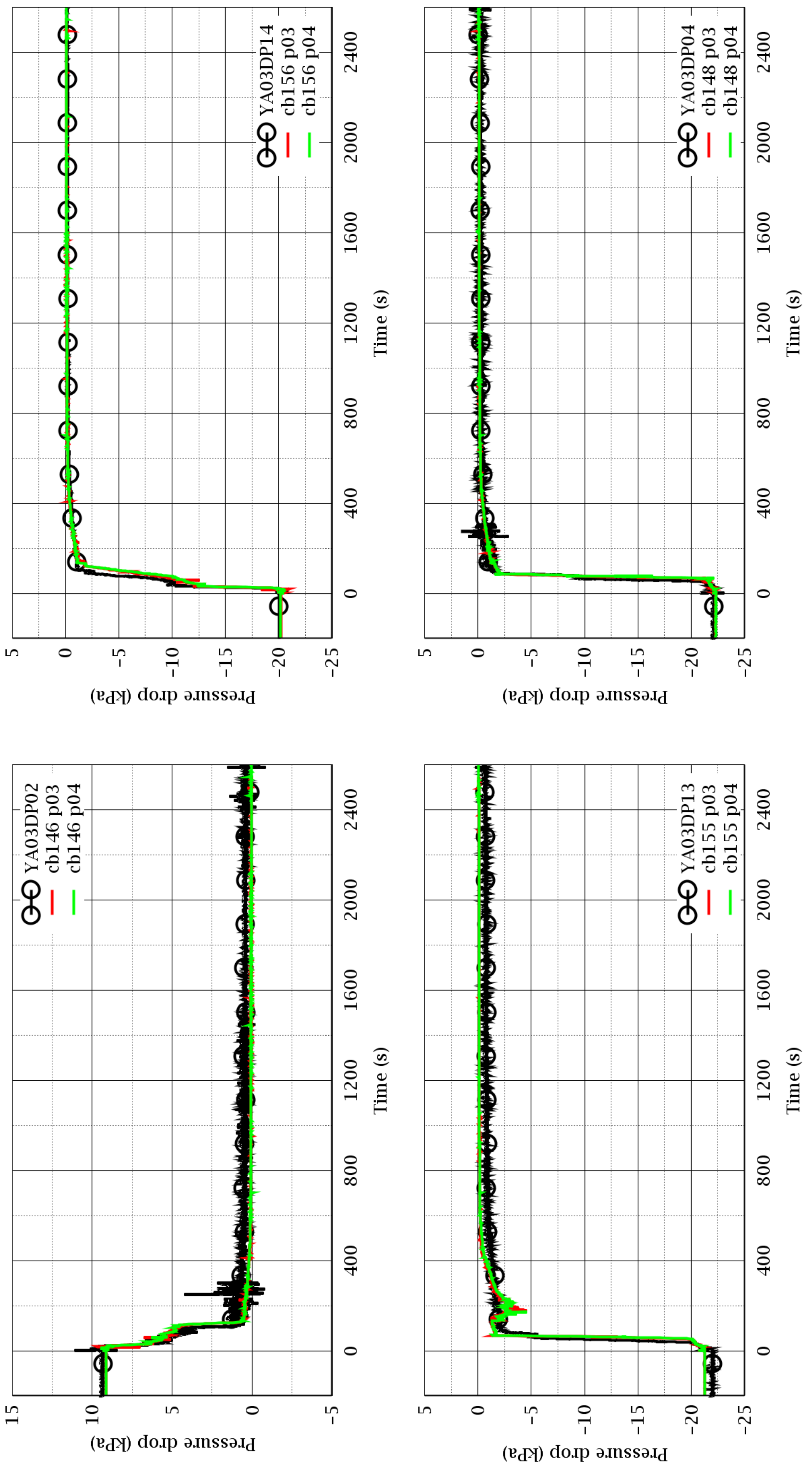

Figure C-19 Loop 3 Pressure Differences 1

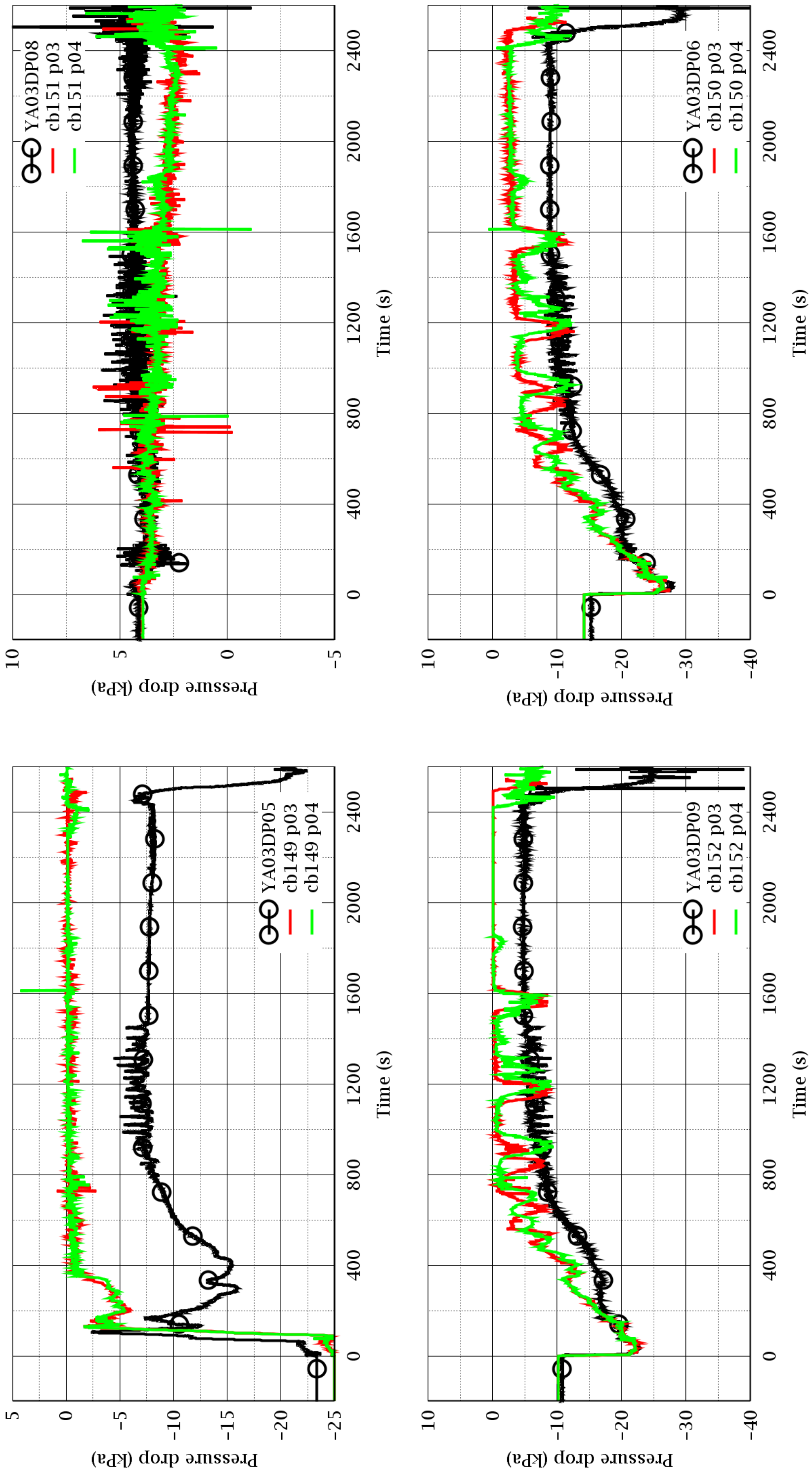

Figure C-20 Loop 3 Pressure Differences 2

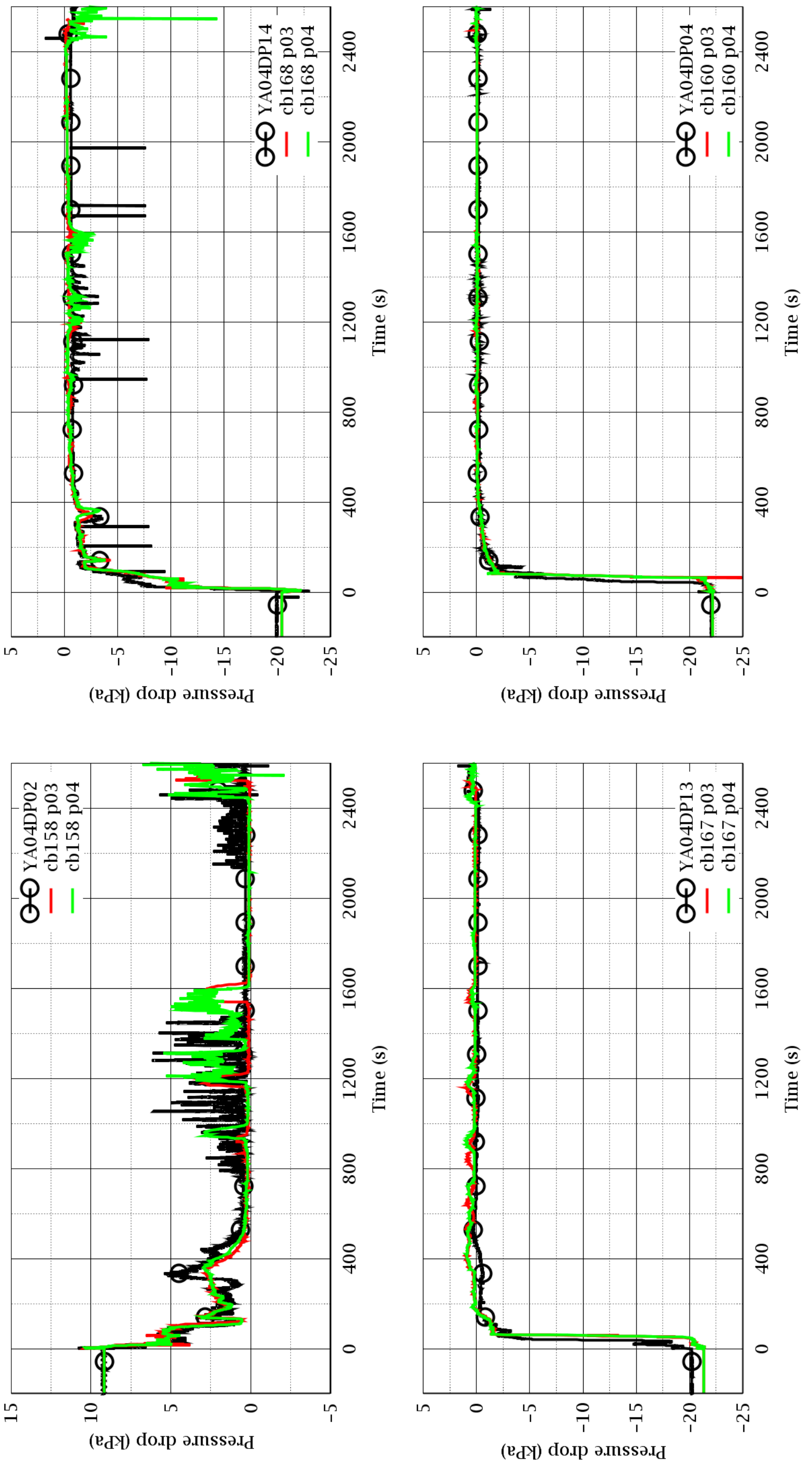

Figure C-21 Loop 4 Pressure Differences 1

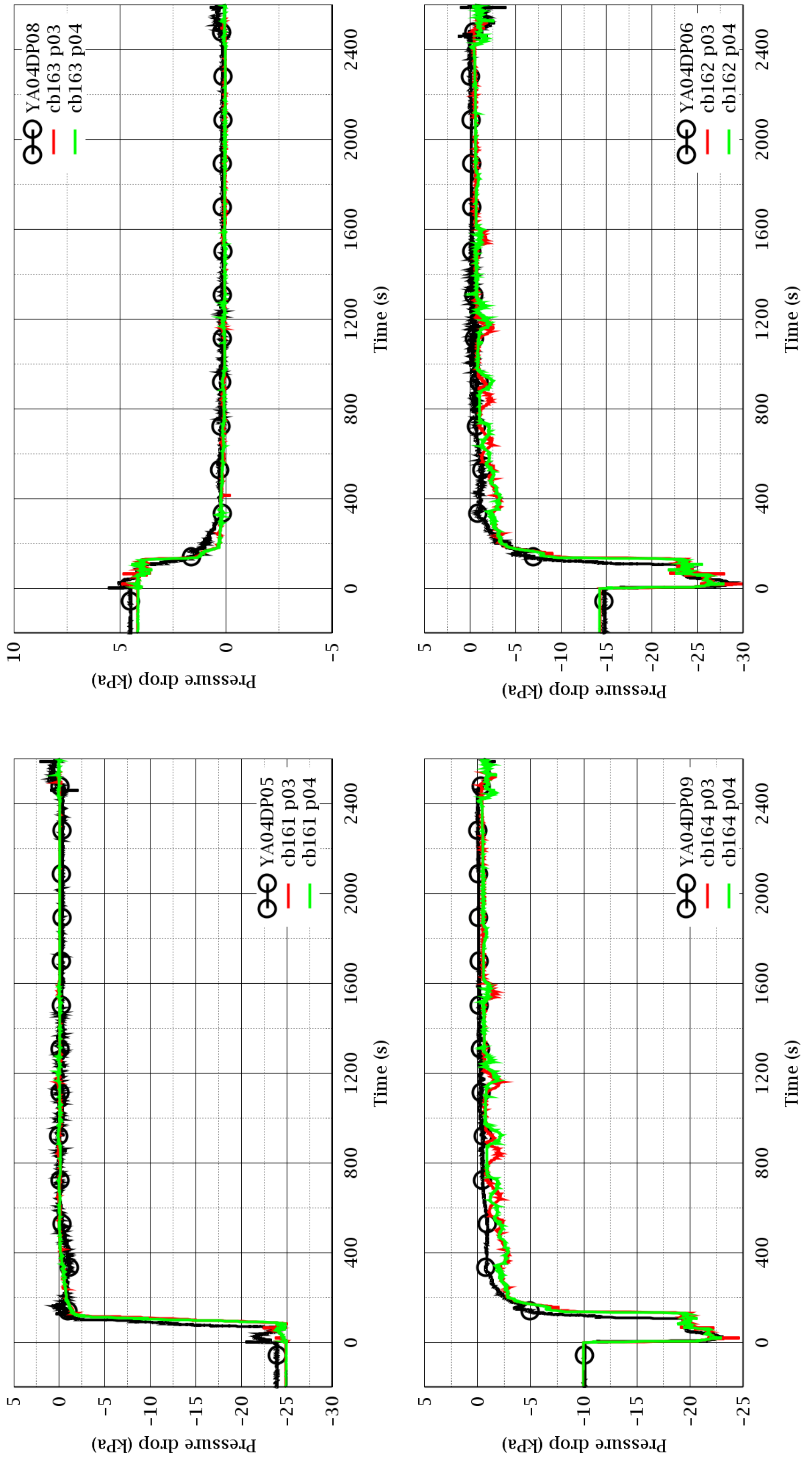

Figure C-22 Loop 4 Pressure Differences 2

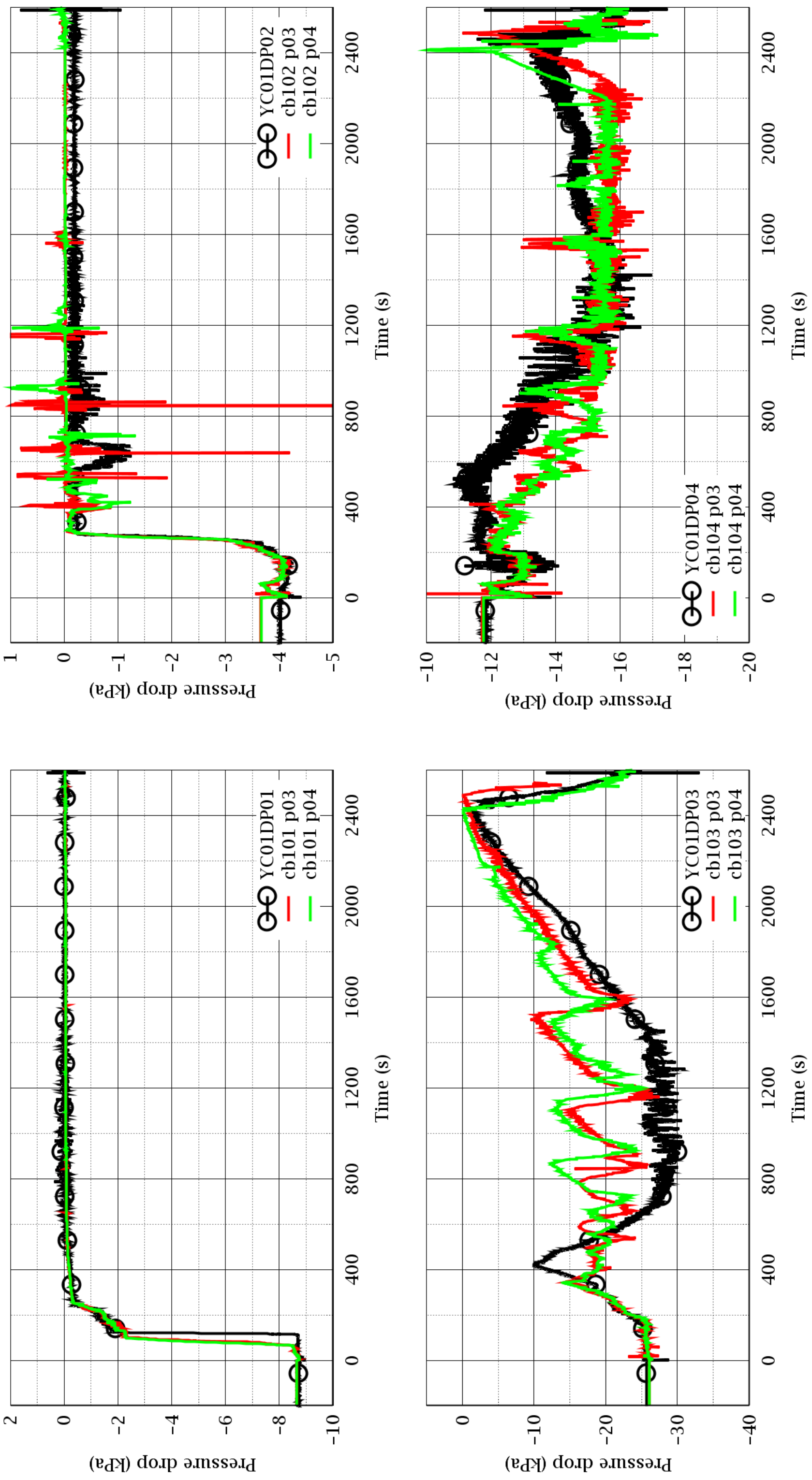

Figure C-23 Pressure Differences DP01-DP04 (DC)

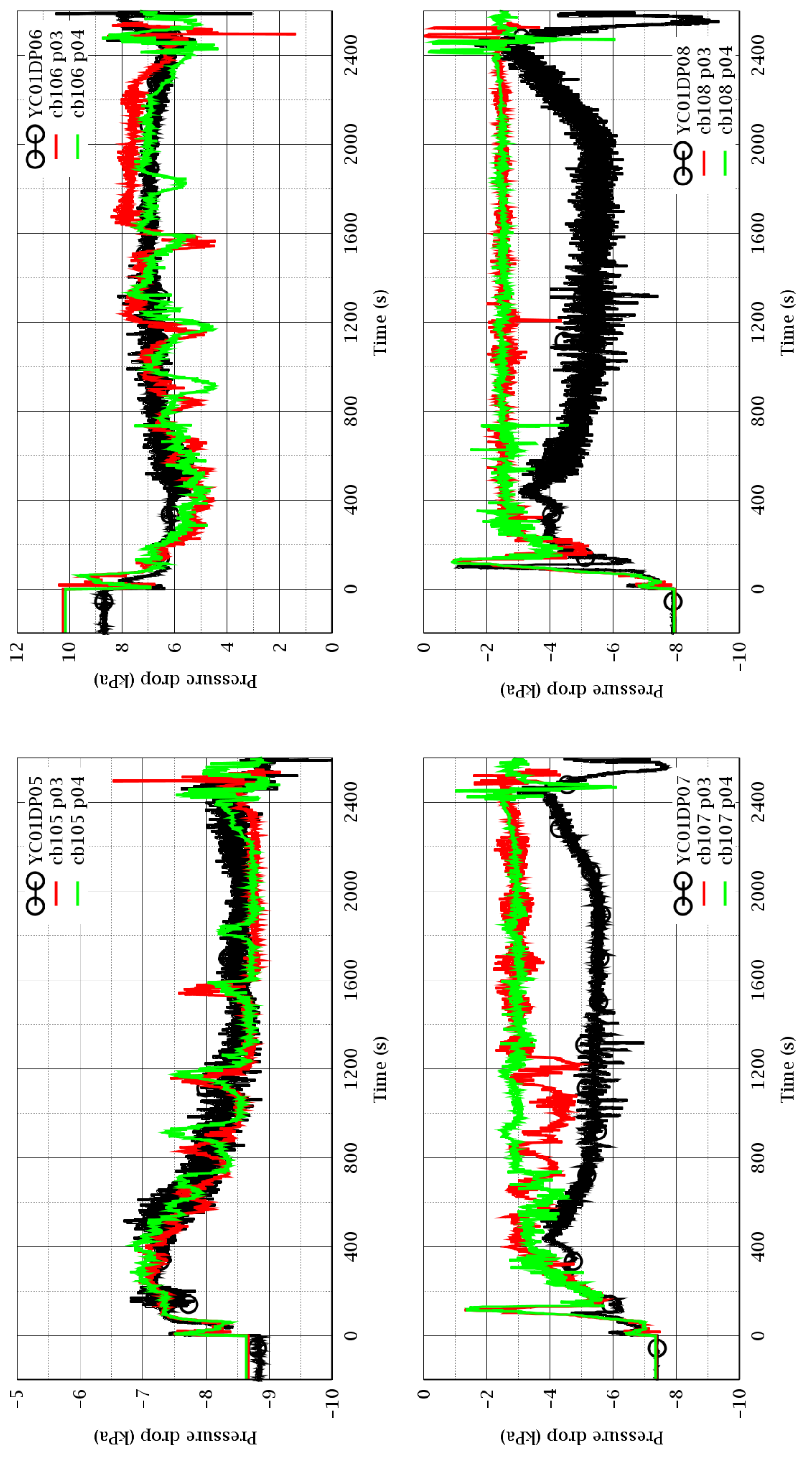

Figure C-24 Pressure Differences DP05-DP08 (Lower Plenum + Lower Part of FRS)

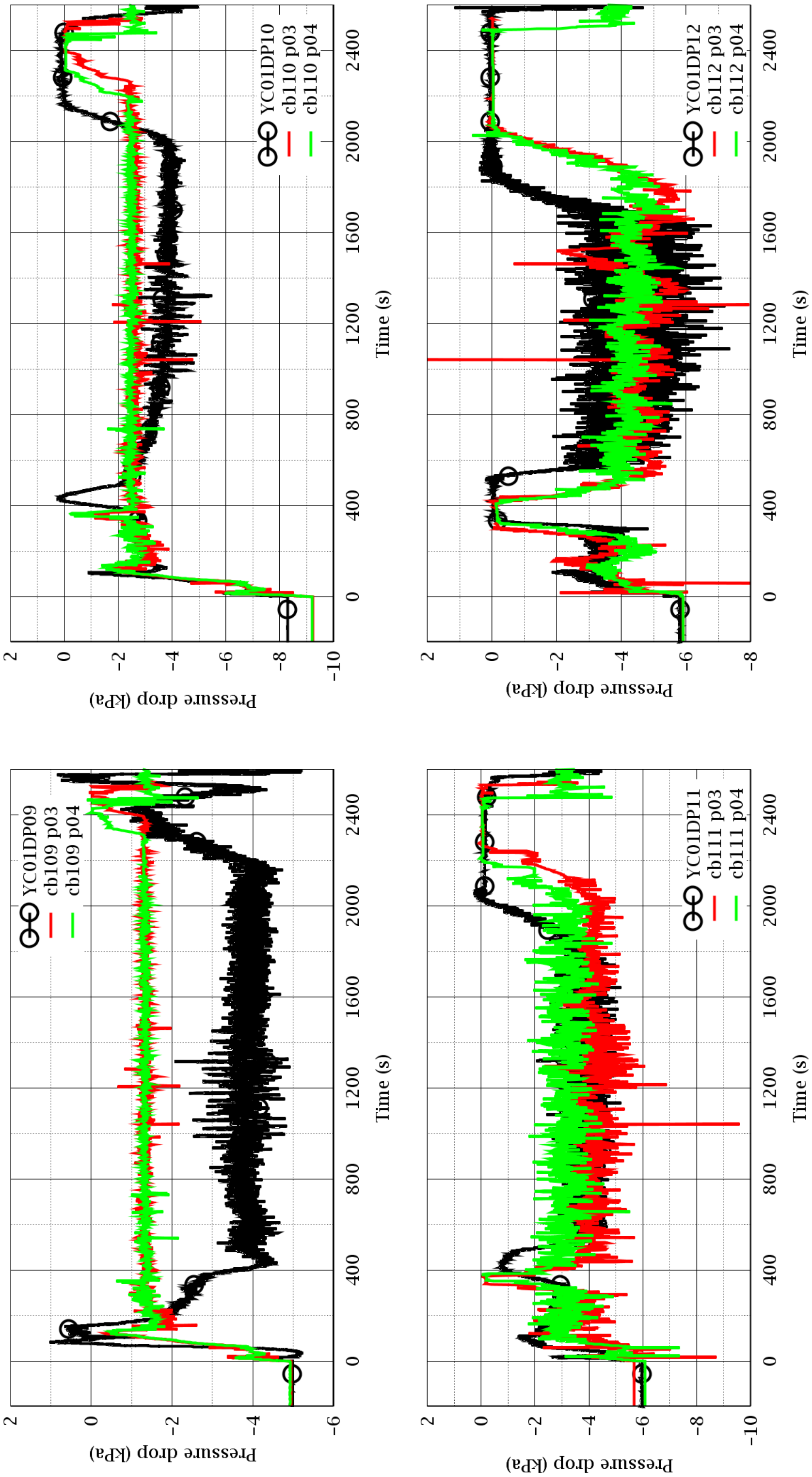

Figure C-25 Pressure Differences DP09-DP12 (FRS + Lower Part of Upper Plenum)

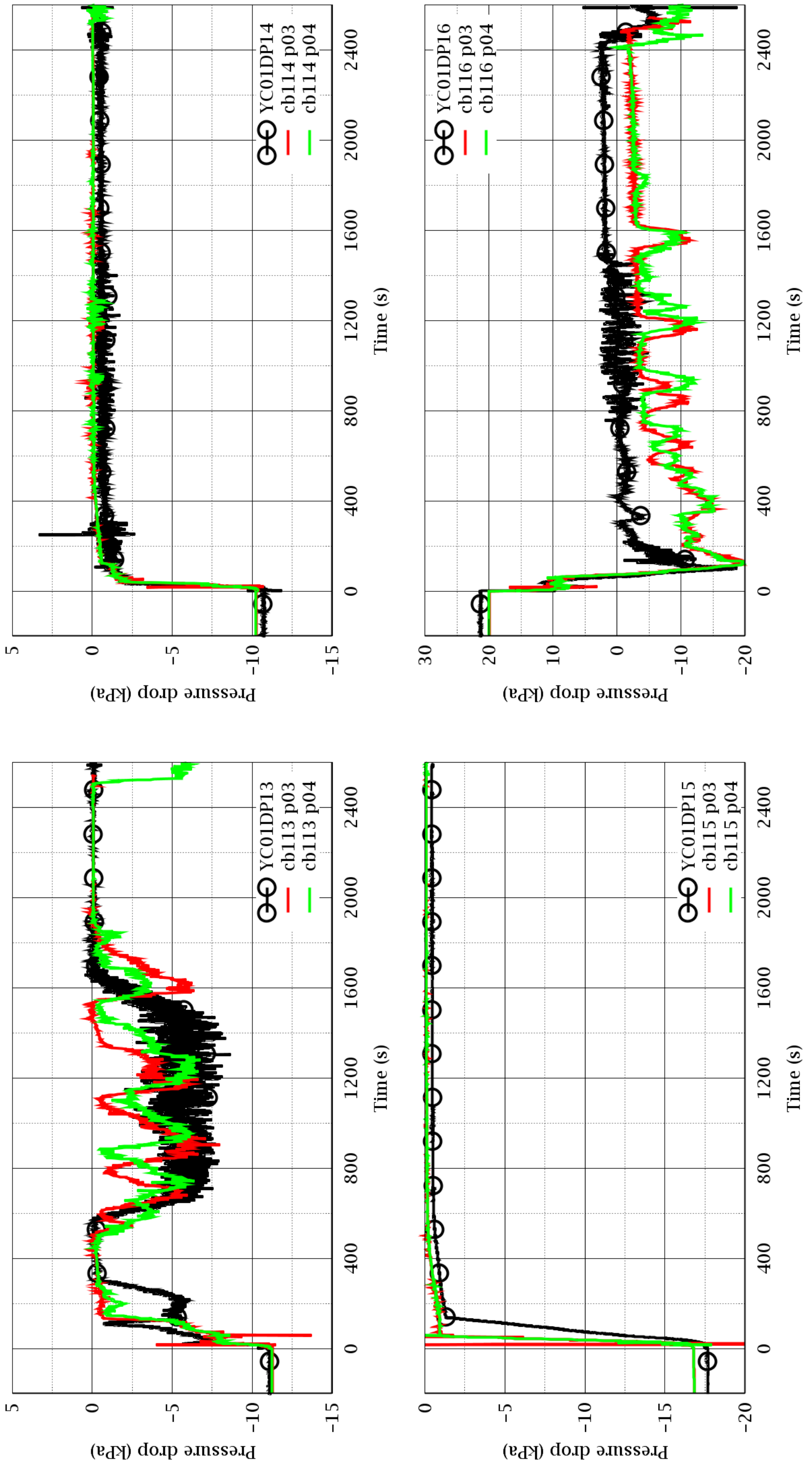

Figure C-26 Pressure Differences DP13-DP16 (Upper Part of Upper Plenum)

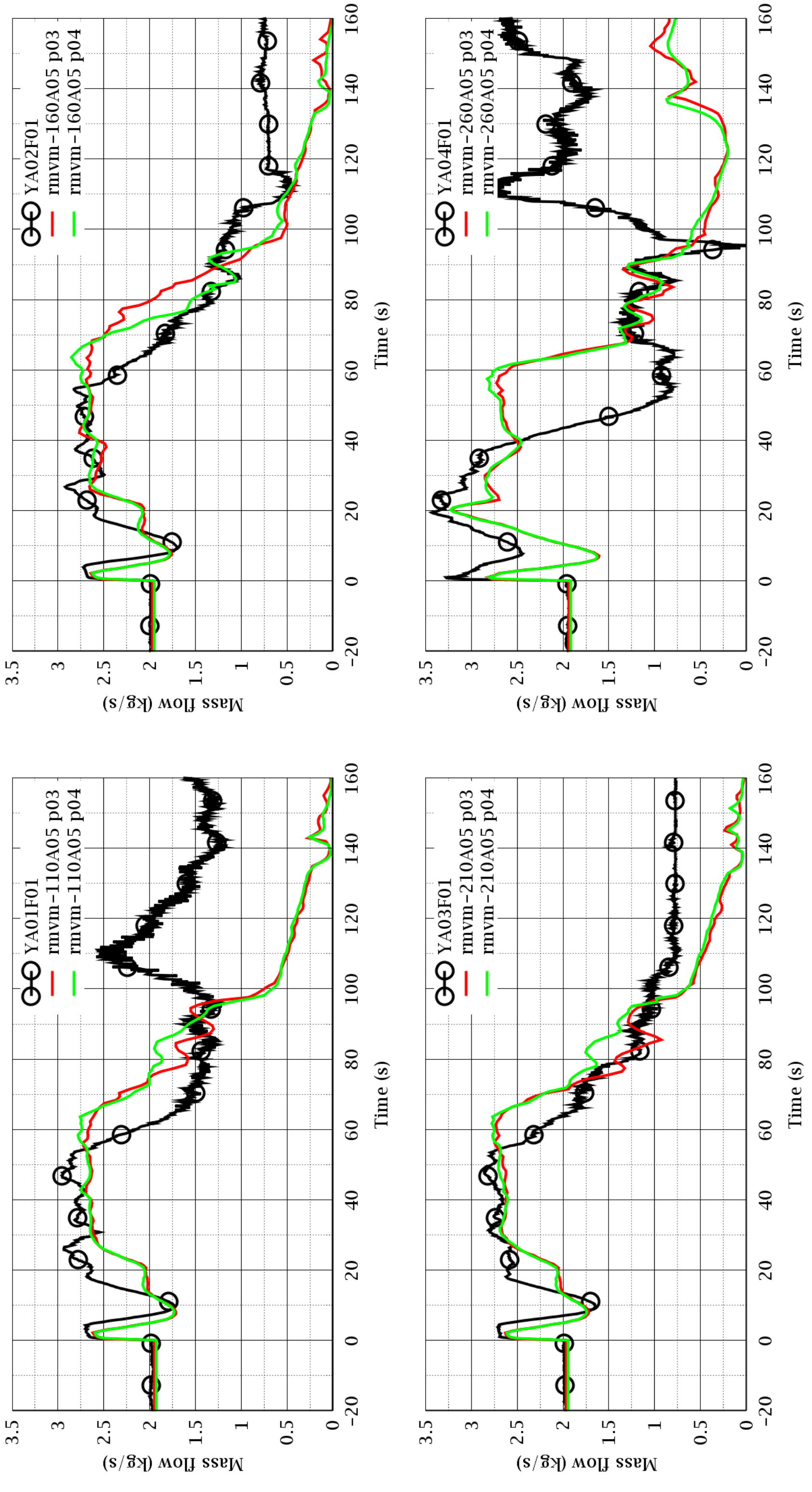

Figure C-27 Primary Loops Flow

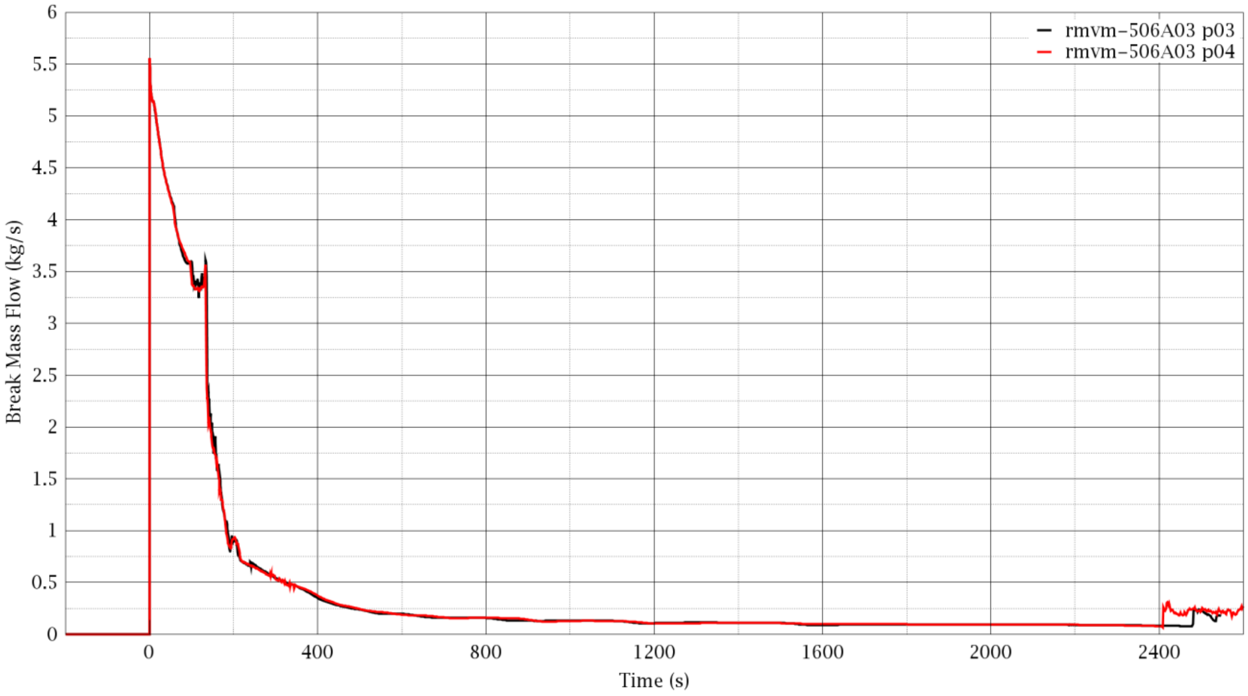

Figure C-28 Break Flow

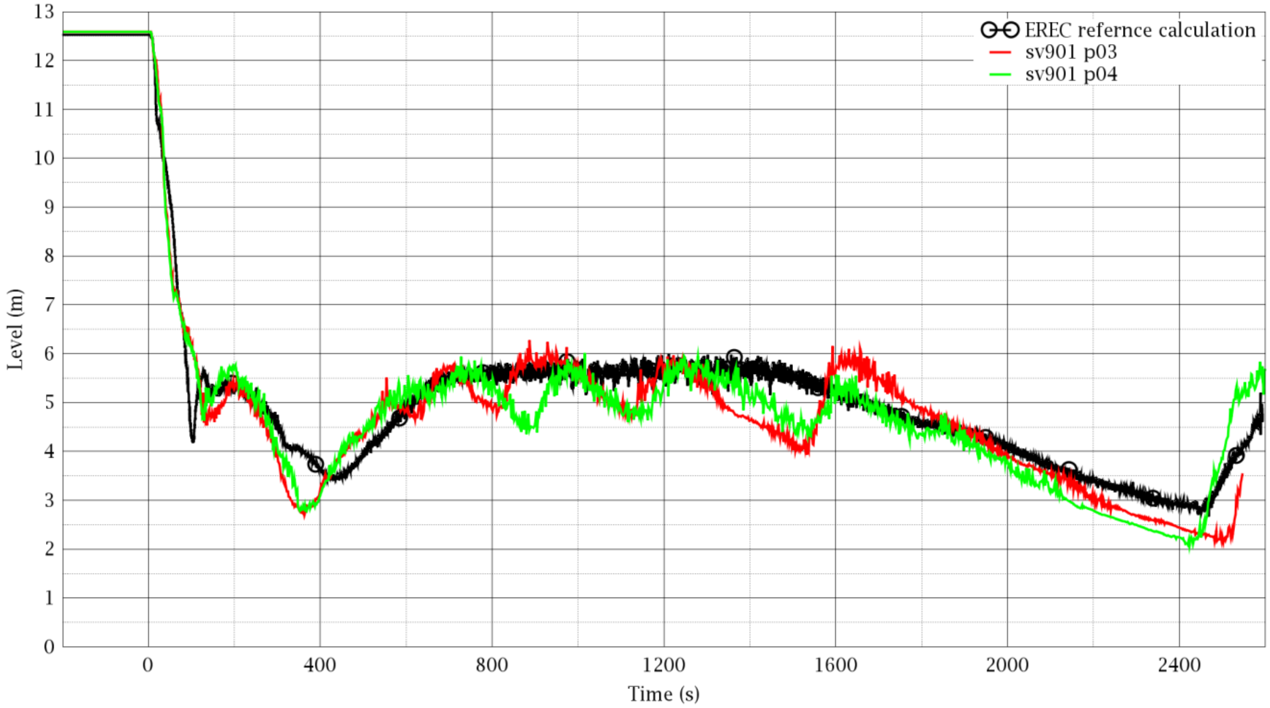

Figure C-29 Reactor Collapsed Level

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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The U.S. NRC best estimate thermo-hydraulic computer code TRACE V5.0 has been assessed against the “4.1 % cold leg break (CL-4.1-03)” experiment at the large-scale test facility PSB-VVER. The PSB-VVER facility is a 1:300 volume-power scaled model of VVER-1000 NPP located in Electrogorsk, Russia. An extensive TRACE input deck of PSB-VVER facility was developed. The model includes all important components of the PSB-VVER facility: reactor, 4 separated loops, pressurizer, HPIS and LPIS ECCS, several break units, main circulation pumps, steam generators, and important parts of secondary circuit. The TRACE (TRAC/RELAP Advanced Computational Engine) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission in frame of CAMP (Code Application and Maintenance Program). The TRACE code is a component-oriented reactor systems analysis codes designed to analyze light water reactor transients up to the point of significant fuel damage. The original validation of the TRACE code was mainly based on experiments performed on experimental facilities of typical PWR design. There are some different features of VVER design comparing to PWR. Therefore the validation of the thermo-hydraulic codes for VVER types of reactors is often required by national regulators. The presented analysis is the latest in series of TRACE and RELAP5 assessment calculations evaluated at the company TES. The purpose of performed analyses is to extend the validation of the TRACE code focused on VVER type of NPPs and to support applicability of the TRACE code in the Czech Republic.

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

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SNAP (Symbolic Nuclear Analysis Program)
Research Centre Řež, TSO
PSB-VVER test facility
OECD PSB project
Electrogorsk Research and Engineering Institute (EREC)
Russian Pressurized Water Type Reactor (VVER)
OECD/NEA/CSNI/WGAMA PSB Project.

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