NUREG/IA-0449

International Agreement Report

Post-Test Analysis of Upper Plenum 11% Break at PSB-VVER Facility using TRACE V5.0 and RELAP5/MOD3.3 Code

Prepared by:
Petr Heralecky

TES Ltd.
Prazska 597
674 01 Trebic, Czech Republic

K. Tien, NRC Project Manager

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

Manuscript Completed: May 2014
Date Published: July 2014

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

Published by
U.S. Nuclear Regulatory Commission

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:
Address: U.S. Nuclear Regulatory Commission
Office of Administration
Publications Branch
Washington, DC 20555-0001
E-mail: DISTRIBUTION.RESOURCE@NRC.GOV
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—
The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—
American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG-XXXX) or agency contractors (NUREG/CR-XXXX), (2) proceedings of conferences (NUREG/CP-XXXX), (3) reports resulting from international agreements (NUREG/IA-XXXX), (4) brochures (NUREG/BR-XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG-0750).

NUREG/IA-0449

International Agreement Report

Post-Test Analysis of Upper Plenum 11% Break at PSB-VVER Facility using TRACE V5.0 and RELAP5/MOD3.3 Code

Prepared by:
Petr Heralecky

TES Ltd.
Prazska 597
674 01 Trebic, Czech Republic

K. Tien, NRC Project Manager

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

Manuscript Completed: May 2014
Date Published: July 2014

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

Published by
U.S. Nuclear Regulatory Commission

ABSTRACT

The best estimate thermo-hydraulic computer code TRACE V5.0 and RELAP5 MOD3.3 hadbeen assessed using Upper Plenum 11% break experiment at the large-scale test facility PSB-VVER. The PSB-VVER facility is a 1:300 volume scaled model of VVER 1000 NPP located in Electrogorsk, Russia. An extensive TRACE and RELAP5 input decks of PSB-VVER facility were developed including all important components of the PSB-VVER facility: reactor, 4 separated loops, pressurizer, 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) and the RELAP5 code is its predecessor. The TRACE and RELAP5 codes are 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 both codes 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 purpose of performed analysis is to extend the validation of the TRACE and RELAP5 code focused on VVER type of NPPs. The TRACE calculation was performed in the frame of R&D project co-funded by The Ministry of Industry and Trade of Czech Republic. The RELAP5 calculation was performed to support standardization of the RELAP5 code in TES Company.

CONTENTS

Page

FIGURES

TABLES

ACKNOWLEDGMENTS

The project FI-IM5/150 was funded by The Czech Ministry of Industry and Trade in frame of R&D activities.

ABBREVIATIONS

ACC	Hydro-accumulator
CL	Cold Leg
CS	Core Simulator
DC	Downcomer
ECCS	Emergency Core Cooling System
EREC	Electrogorsk Research and Engineering Institute
FRS	Fuel Rod Simulator
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
MCP	Main Coolant Pump
MCP	Main Coolant Pump
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Co-operation and Development
PKL	Experimental Facility in Finland
PCS	Plant Control System
PRZ	Pressurizer
PWR	Pressurized Water Reactor
R&D	Research and Development
RPV	Reactor Pressure Vessel
RPV	Reactor Pressure Vessel
SCRAM	Emergency Shutdown of a Nuclear Reactor
SETF	Separate Effect Test Facility
SG	Steam Generator
TC	Thermo Couple
TH	Thermal Hydraulic
UP	Upper Plenum
VVER	Russian Pressurized Water Type Reactor

1. INTRODUCTION

The assessment of PWR safety codes is mainly performed on the basis of experimental data coming from scaled-down integral or separate test facilities. The TRACE and RELAP5 code validation process was mainly based on the data from experimental facilities or real NPPs of western PWR type as well. There is a significant number of VVER type of reactors operating all over the world and many other are under construction or under preparation as well. VVER reactors are in many aspects similar to western PWRs. Therefore a lot of experimental data measured on PWRs or PWR based test facilities are valuable also for VVER research. On the other hand, the VVER design has several specific features such as larger volumes of primary coolant, horizontal steam generators, different ECCS injection points and so on. Therefore the validation of the thermo-hydraulic codes for VVER types of reactors is often required by national regulators. The purpose of performed analyses is to extend the validation of the TRACE and RELAP5 code focused on VVER type of NPPs. The best estimate thermo-hydraulic computer code TRACE V5.0 and RELAP5/MOD3.3. were assessed using Upper Plenum 11% break experiment at the large-scale test facility PSB-VVER. The PSB-VVER facility is a 1:300 volume scaled model of VVER 1000 NPP located in Electrogorsk, Russia. In order to perform code validation an extensive TRACE and RELAP5 input decks of PSB-VVER facility were developed. Both models include all important components of the PSB-VVER facility: reactor, four separated loops, pressurizer, break unit, main circulation pumps, steam generators, and important parts of secondary circuit.

2. FACILITY AND TEST DESCRIPTIONS

Detail information about the PSB-VVER test facility systems and elements is given in Ref 4. Only a brief description of the PSB-VVER facility is given here. The hardware configuration for UP-11-07 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. The PSB rod simulator bundle has 15 spacer grids with prototypic 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.

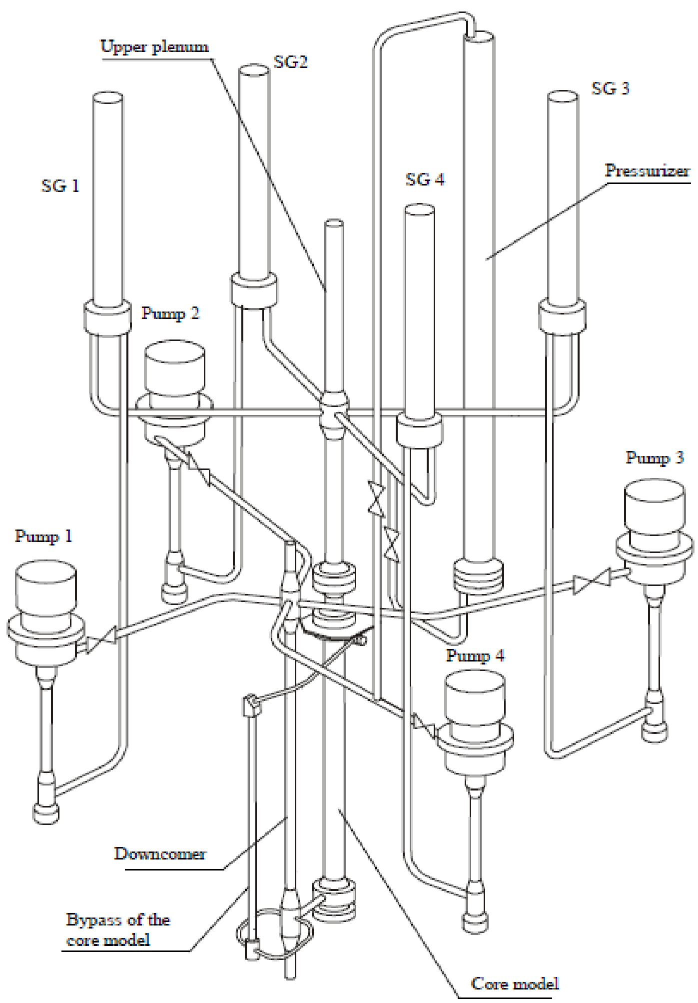
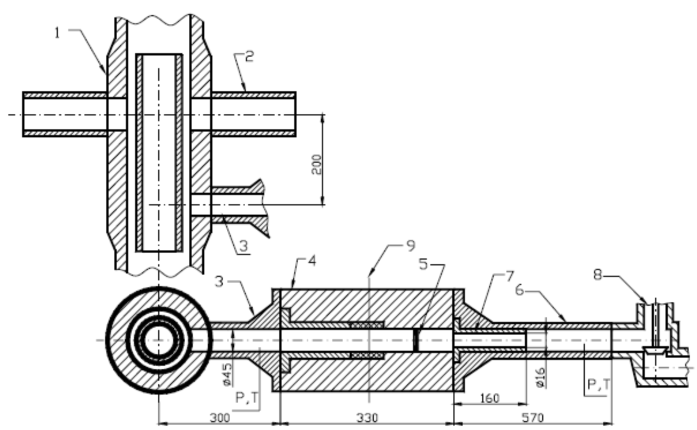


Figure 1: General View of PSB-VVER Facility

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 special break systems to facilitate research of thermal hydraulics during break accidents. There is a special system to simulate accumulator water supplying pipe rupture which is utilized for 11% upper plenum break experiments (UP-11-07 and PSBV1). The break system consists of a break unit, a discharge pipeline with isolating valves and catch tank-condenser. Principal scheme of the break unit is given in Figure 2 and geometric characteristics of the discharge line are represented in Figure 3.



1 – UP; 2 – "hot" leg; 3 – break nozzle; 4 – instrumented spool peace; 5 – drag-screen; 6 – branch pipe; 7 – discharging channel; 8 – shut-off valve; 9 – X-ray path.

Figure 2: Upper Plenum Break Unit

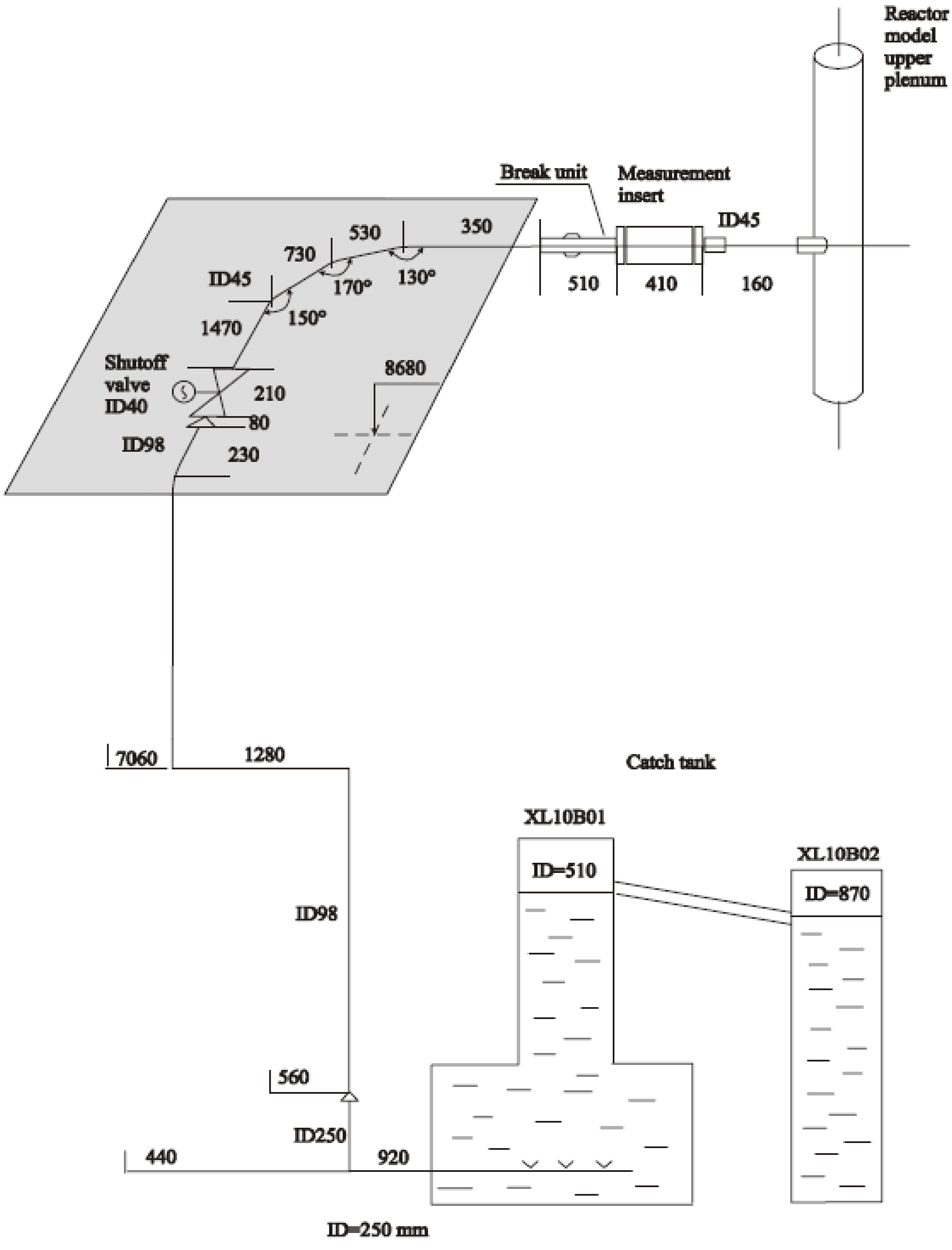


Figure 3: Discharge line

2.2 Experiment UP-11-07

The test UP-11-07 „Upper Plenum Break 11%“ was performed in the PSB-VVER test facility at Electrogorsk Research and Engineering Center (EREC) in Russia. The thermal-hydraulic processes related to upper plenum break 11% were researched.

2.2.1 Facility configuration

The information on the test facility hardware and configuration of the system specific for UP-11-07 test are given in the Table 2.

Table 2: Test Facility Configuration in UP-11-07 Test

Equipment	Status
Pressurizer	Connected to the loop #2
Core by-pass	2 throttles with 2 orifices of diameter 7 mm were installed at inlet and outlet of core by-pass
HPIS	One channel was connected to the cold leg of the loop #1
LPIS	Two channels connected to the cold and hot legs of the loop #1
ACCs	ACCs #1 and 3 were connected to UP. ACC #4 was connected to downcomer. ACC #2 was switched off (isolated)
SGs	All SGs were connected through steam lines
Feed water heater	In use. SG levels under steady-state were maintained by supply of feed water
Large break unit	Break is located in upper plenum. Leak channel is a throttle of 16 mm, L = 160 mm. Horizontal blowdown line of 45 mm is located below hot legs inlet by 200 mm. Coolant discharging is realized from annular chamber between annular screen and UP wall

2.2.2 Initial Conditions

The main initial conditions of UP-11-07 test are given in the Table 3. UP-11-07 test has been performed under reduced initial core power corresponding to approximately 15% of nominal power.

Table 3: Measured initial condition for UP-11-07 test

Parameter	Units	Value
Primary circuit		
Pressure in upper plenum (gauge YC01P16)	MPa	15.744
Coolant temperature (DC inlet/UP outlet - gauges C01T02/YC01T04)	deg	274.6 / 306.6
Primary loops flow rates (gauges YA01÷04F01)	kg/s	2.321 / 2.289 / 2.343 / 2.357
Core power (gauge YC01N01)	kW	1496.5
Core by-pass power (gauge YC01N02)	kW	17.1
Coolant level in PRZ (gauge YP01L02)	m	6.472
Secondary circuit		
Pressure in SGs (gauges RA01÷04P01)	MPa	6.269 / 6.300 / 6.192 / 6.285
Level in SGs (gauges YB01÷04L01)	m	1.694 / 1.692 / 1.835 / 1.664

ECCS		
Pressure in ACCs (gauges TH01÷04P01)	MPa	6.015 / 6.019 / 5.890 / 5.898
Level in ACCs (gauges TH01÷04L01)	m	4.800 / 5.172 / 4.807 / 4.806

2.2.3 Boundary Conditions (test scenario)

Detail information about the UP-11-07 test boundary conditions is given in Ref 5. The main events of UP-11-07 test are described in the Table 4.

Table 4: Main Events During UP-11-07 Test

Event	Time [s]
IE – Break Opening	0
Pressure in UP (YC01P16) < 13.73 MPa $\rightarrow$ conditions for SCRAM - Start of MCP coastdown - Start of core power and core bypass power reduction - Stop of feed water flow supply - Stop of steam removal from steam generators	2 2 3 9 11
HPIS activation	12.0
ACCs activation	110 / / 112 / 109
LPIS activation	383
ACCs empty	504 / - / 499 / 489
LPIS termination	3645
HPIS termination	3696
End of test (FRS power switched off)	5593

The experiment is started with opening of an isolation valve XL10S01 in the leak line.

When the UP pressure P_{UP} = 13.73 MPa PCS automatically cuts the electric power from PRZ heaters and starts to simulate operation of NPP automatics in accordance with the SCRAM signal (station blackout takes place simultaneously), which provides for the following actions to be performed at the test facility:
Power reduction on core simulator and core by-pass is started

- the procedure of MCPs coastdown is started
- stop steam removal from steam generators
- close feed water supply

There was rather nonstandard coastdown of MCPs realized. MCPs rotational speed was temporarily stabilized at revolution corresponding to 29.5% of nominal value for 200 second, and then the standard coastdown continued.

After achievement of value of 150 kW, the core model power is fixed and is the same up to the end of the experiment. At this moment the core by-pass power is also fixed and is the same up to the end of the experiment.

After achievement of two conditions 1) UP pressure decreases to the value PUP = 10.88 MPa and 2) 10 s later the moment when PUP = 13.73 MPa (interval of time to start operation of diesel-generator), start of high pressure injection system operation is simulated. Cooling water from HPIS is supplied to the cold leg of loop #1. HPIS mass flow rate 0.10 kg/s is provided. Temperature of injected cooling water is 28°C in the beginning of test.

The accumulators start operate when pressure in upper plenum decreases below 5.89 MPa (accumulator #2 does not operate in this test). Operation of accumulators is stopped when level in accumulators falls down to 0.100 m.

When UP pressure decreases to the value PUP = 1.76 MPa, regular starting of low pressure ECCS pump is simulated. Cooling water of LPIS is supplied to cold and hot legs of loop #1, mass flow rate is 0.205 kg/s per each line. HPIS and LPIS keep water delivery to primary circuit till total water supply of HPIS and LPIS achieves a value 1.72 m^3 (simulation of emptying of LPIS water tank), when operation of HPIS and LPIS is terminated.

The experiment is stopped when cladding temperature is reaching a value of 1000°C..

.

.

3. THE TRACE AND RELAP5/MOD3.3 CODES

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 RELAP5 computer code is one of four TRACE's predecessor.

Both codes have 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 and RELAP5 are 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.

RELAP5/MOD3.3 uses a one-dimensional, two fluids, nonequilibrium, six equation hydrodynamic model with a simplified capability to treat multi-dimensional flows. This model provides continuity, momentum, and energy equations for both the liquid and the vapor phases within a control volume. The energy equation contains source terms which couple the hydrodynamic model to the heat structure conduction model by a convective heat transfer formulation. The code contains special process models for critical flow, abrupt area changes, branching, crossflow junctions, pumps, accumulators, valves, core neutronics, and control systems. A countercurrent flow limitation model can also be applied at vertical junctions

3.1 <u>The TRACE and RELAP5/MOD3.3 code assessment</u>

Confidence in the computational tools (codes) and establishment of their validity for a given application depends on proper assessment. TRACE and RELAP5, 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 16). The assessment process however, can also indicate potential deficiencies in the code. There are following four sources of data for code assessment (Ref 17):

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

3.2 <u>VVER typical features related to TRACE and RELAP5/MOD3.3 code assessment</u>

The TRACE and RELAP5 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 19):

- 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 line

There are approximately 50 operating units of VVER type (Ref 18). It is a meaningful number in comparison to approximately 216 operating units of PWR reactors (Ref 18). Therefore corresponding attention should be given to code validation for VVER type of reactors.

4. INPUT DECK DESCRIPTION

An extensive TRACE and RELAP5 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. Both input decks were designed on the basis of PSB-VVER facility documentation (Ref 4, 20).

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 32 axial layers, 4 azimuthal theta sector and 6 radial rings). The model also includes by-pass piping between DC to UP and 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. 20. No two-phase degradation was modeled because of no appropriate data.

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 and #4. Low pressure injection system (LPIS) optionally provides flow to the hot and cold leg of loop #1 and #3 (test UP-11-07) and loop#3. 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 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 heat loss from the primary and secondary is represented in the TRACE model by entering the thickness of the insulation on the outside 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 1756 volumes, 3433 junctions, and 1285 heat structures with 4814 mesh points. Standard modeling guidelines were followed in developing the nodalization of the system.

Components Statistic for TRACE model – see the next Table 5.

Table 5: TRACE Components Statistic

TRACE Component		Notes
VESSEL	2	DC+LP; FRS+UP
PIPE	67+66*1	-
HSTR	157+1*1	-
POWER	1+1*1	FRS simulator + By-pass heating
VALVE	25	-
PUMP	4	MCPs
BREAK	4	Upper plenum and Large Break unit + release of secondary steam
FILL	8	HPIS, LPIS, Feedwater
Whole No of Components	336	-

*1 the second number is the number of spawned component

Table 6: 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	-	
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	-	
Steam headers (RA)	+	VALVE + HTSTR + BREAK

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

4.2 The RELAP5 input deck

Nodalization diagrams of the RELAP5 Input Deck are presented in Figure A-3 (reactor + primary circuit) and Figure A-4 (secondary circuit) in the Appendix A. The RELAP5 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. All parts of RPV employ PIPE, BRANCH and SNGLJUN components. The model also includes by-pass piping between DC to UP.

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. 20. No two-phase degradation was modeled because no appropriate data were 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). Relief valves are included on the pressurizer.

Active ECCS are modeled using simple boundary condition - TMDPJUN + TMDPVOL components, accumulators are modeled using ACCUM, PIPE and VALVE components. High pressure injection system (HPIS) optionally provides flow to the hot leg of loop #1 and #4. Low pressure injection system (LPIS) optionally provides flow to the hot and cold leg of loop #1 and #3 (test UP-11-07) and loop #3. 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

The steam generator secondary side is modeled with a single stack of volumes in the tube bundle region. There is no physical barrier between the tube bundle and the outer shell of the steam generator. The secondary side of the steam generators included main and auxiliary feedwater, individual steam lines and the common steam header, and all of the steam lines is equipped with relief valve.

The heat loss from the primary and secondary is represented in the RELAP5/MOD3.3 model by entering the thickness of the insulation on the outside of all the pipes and other system components. Appropriate material properties were input for the insulation. A constant boundary temperature and heat transfer coefficient of outer air is applied.

Choking was turned off at most of the junctions, as recommended in the code user guidelines. Exceptions were the break, at valves, at the pressurizer connection to the surge line and the surge line connections to the hot leg, and at the core outlet. The break piping was attached to the upper plenum one volume below the hot leg connections. The break was modeled as an abrupt area change with user-input loss coefficients.

The RELAP5/MOD3.3 countercurrent flow limitation (CCFL) model was applied at five locations in the model: in the downcomer below the accumulator injection nozzles, at the core outlet, in the upper plenum below the accumulator injection nozzles, at the plate between the upper plenum and the upper head, and at the outlet of the riser section on the secondary side of the steam generators. These junctions were selected because they were vertically oriented, represented places where the flow area changed, and modeled regions of the facility where CCFL might be expected to occur.

The model contains 536 volumes, 559 junctions, and 521 heat structures with 5549 mesh points. Standard modeling guidelines were followed in developing the nodalization of the system.

Table 7: RELAP5 Components Statistic

TRACE Component		Notes
PIPE	61	-
BRANCH	37	-
SNGLVOL	4	-
ANNULUS	2	-
PUMP	4	MCPs
ACCUM	4	Accumulators
VALVE	23	-
SNGLJUN	6	-
MTPLJUN	8	-
TMDPVOL	19	Upper plenum and Large Break unit, release of secondary steam, HPIS, LPIS, relief valves BC
TMDPJUN	10	HPIS, LPIS, Feedwater
HSTR	132	-
Whole No of Components	310	-

Table 8: List of the main systems and components of PSB-VVER RELAP input deck

PSB system	Input deck	Used RELAP components
Rector (YC)		
Downcomer	+	PIPE + BRANCH + HTSTR
Lower plenum	+	PIPE + BRANCH + SNGLJUN + HTSTR
Core	+	PIPE + HTSTR + CONTRL BLOCK
Upper plenum	+	PIPE + BRANCH + SNGLJUN + HTSTR
Core bypass	+	PIPE + SNGLJUN
DC to UP bypass	+	PIPE + VALVE + BRANCH + HTSTR
UP heating	-	
LOOP (YA)		
Hot leg	+	PIPE + HTSTR
Loop seal	+	PIPE + HTSTR
Cold leg	+	PIPE + SNGLVOL + HTSTR
Main cooling pump (YD)		PUMP + HTSTR + CONTRL BLOCK
Presurizer (YP)		
Vessel	+	PIPE + HTSTR
heaters	+	HTSTR + CONTROL BLOCK
Surge line	+	PIPE + BRANCH + VALVE + HTSTR
Relief valve	(+)	VALVE + TMDPVOL
ECCS (TJ, TH)		
HPIS	(+)	TMDPJUN + TMDPVOL + CONTROL BLOCK
LPIS	(+)	TMDPJUN + TMDPVOL + CONTROL BLOCK
Accumulators	+	ACCUMULATOR + PIPE + HTSTR
Steam generators (YB)		
Vessel	+	PIPE/ANNULUS + HTSTRT
Heat exchange tubes	+	PIPE + HTSTR
Primary headers	+	PIPE + BRANCH + MTPLJUN + HTSTRT
Steam lines	+	BRANCH + VALVE
Feedwater	(+)	TMDPJUN + TMDPVOL + CONTROL BLOCK
Relief valve		TMDPJUN + TMDPVOL
Steam headers (RA)	+	PIPE + SNGLJUN + TMDPVOL

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

5. RESULTS

5.1 TRACE Steady-state calculation

In order to achieve stable initial conditions of the UP-11-07 test, the steady state was calculated for 300 s. The following controllers were used for the first 300 s:

- Pressurizer pressure 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 is 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 200 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. Steady state calculation took approximately 30 min. Main calculated and measured parameters are compared in the Table 9.

5.2 RELAP5 Steady-state calculation

In order to achieve stable initial conditions of the UP-11-07 test, the steady state was calculated for 1600 s. The following controllers were used for the first 1600 s:

- Pressurizer pressure controller
- Pressurizer level controller
- Main steam header pressure controller
- Steam generators level controllers
- Main circulation pumps velocity controllers

The same boundary conditions like in the TRACE calculation were used in the RELAP5 calculation. Because of longer steady state calculation, feedwater steady state PI-controller was in operation for 1500 s and for the rest of steady state calculation and for transient calculation SG feedwater flow was entered as a boundary condition in the same way like in TRACE calculation. Steady state calculation took approximately 6 min. Main calculated and measured parameters are compared in the Table 9 .

Table 9: Initial Conditions (TRACE and RELAP5 calculation vs. experiment comparison)

Parameters	Units	accuracy	UP-11-07*2	RELAP 5	TRACE
Upper plenum pressure (YC01P16)	MPa	± 0.16	15.774 ± 0.021	15.765	15.718
Pressure drop at FRS (YC01DP07-DP10)	kPa	± 4.60*1	-28.63	-28.94	-28.32
Hot Leg outlet Coolant Temp. (TA01T03)	°C	± 3	305.4 ± 0.3	305.7	305.8
Hot Leg outlet Coolant Temp. (TA02T03)	°C	± 3	305.2 ± 0.3	306.7	305.8
Hot Leg outlet Coolant Temp. (TA03T03)	°C	± 3	304.3 ± 0.4	306.6	305.8

Parameters	Units	accuracy	UP-11-07*2	RELAP 5	TRACE
Hot Leg outlet Coolant Temp. (TA04T03)	°C	± 3	304.7 ± 0.3	306.7	305.8
Cold Leg outlet Coolant Temp. (YA01T02)	°C	± 3	276.0 ± 0.3	276.3	275.2
Cold Leg outlet Coolant Temp. (YA02T02)	°C	± 3	275.7 ± 0.3	276.2	275.1
Cold Leg outlet Coolant Temp. (YA03T02)	°C	± 3	274.7 ± 0.3	276.1	275.3
Cold Leg outlet Coolant Temp. (YA04T02)	°C	± 3	274.5 ± 0.3	276.2	275.2
Loop-1 flow rate (YA01F01)	kg/s	± 0,05	2.321 ± 0.017	2.299	2.318
Loop-2 flow rate (YA02F01)	kg/s	± 0,05	2.289 ± 0.010	2.316	2.286
Loop-3 flow rate (YA03F01)	kg/s	± 0,05	2.343 ± 0.020	2.322	2.340
Loop-4 flow rate (YA04F01)	kg/s	± 0,05	2.357 ± 0.018	2.323	2.354
FRS power (YC01N01)	kW	± 15	1496.5 ± 11.3	1500.0	1500.0
Core by-pass power (YC01N02)	kW	± 0,4	17.1 ± 0.2	17.0	17.0
Collapsed level in PRZ (YP01L02)	m	± 0.3	6.472 ± 0.051	6.431	6.473
Pressure in SG-1 (YB01P01)	MPa	± 0.05	6.269 ± 0.04	6.136	6.057
Pressure in SG-2 (YB02P01)	MPa	± 0.05	6.300 ± 0.035	6.149	6.073
Pressure in SG-3 (YB03P01)	MPa	± 0.05	6.192 ± 0.066	6.135	6.071
Pressure in SG-4 (YB04P01)	MPa	± 0.05	6.285 ± 0.031	6.134	6.054
Collapsed level in SG-1 (YB01L01)	m	± 0.07	1.694 ± 0.047	1.624	1.689
Collapsed level in SG-2 (YB02L01)	m	± 0.07	1.692 ± 0.037	1.709	1.714
Collapsed level in SG-3 (YB03L01)	m	± 0.07	1.835 ± 0.124	1.864	1.871
Collapsed level in SG-4 (YB04L01)	m	± 0.07	1.664 ± 0.023	1.620	1.683
ACCU-1 pressure (TH01P01)	MPa	± 0.03	4.800 ± 0.002	4.800	4.800
ACCU-2 pressure (TH02P01) switched off	MPa	± 0.03	5.172 ± 0.002	5.172	5.172
ACCU-3 pressure (TH03P01)	MPa	± 0.03	4.807 ± 0.001	4.807	4.807
ACCU-4 pressure (TH04P01)	MPa	± 0.03	4.806 ± 0.001	4.806	4.806
ACCU-1 collapsed level (TH01P01)	m	± 0.07	6.015 ± 0.004	6.016	6.015
ACCU-2 collapsed level (TH02P01)	m	± 0.07	6.019 ± 0.005	6.020	6.019
ACCU-3 collapsed level (TH03P01)	m	± 0.07	5.890 ± 0.005	5.889	5.890
ACCU-4 collapsed level (TH04P01)	m	± 0.07	5.898 ± 0.004	5.898	5.898

*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

5.3 TRACE transient calculation

The post-test calculation of UP-11-07 experimental test at PSB-VVER facility started at the time

0 s with an initiating event – upper plenum break 11 % (to simulate accumulator water supplying pipe rupture). A comparison of calculated and experimental times of the occurrence of main events is presented in the Table 10. 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.

Following the break opening, the system pressure decreased rapidly. Measured and calculated pressures in the upper plenum are presented in Figure C-1. At the time 2 s upper plenum pressure decreased bellow 13.73 MPa and SCRAM signal was triggered simulated. The SCRAM initiated core and by-pass power reduction along to the specified function, that represents simulation of core decay heat after the SCRAM and MCP coastdown, with temporary stabilization at 24 % of nominal speed. The depressurization slowed near 40 s, as liquid began to boil in the entire core. As the pressure continued to decrease, ECCS injection begun: HPIS injection at 18 s was followed by accumulators injection approximately at 108 s and finally by LPIS injection at 383 s. As the system pressure decreased the total ECC injection could compensate decreasing break flow. Collapsed level in vessel achieved minimal value approximately at +600 s, see pressure differences in Figure C-18 and C-19, collapsed level in vessel slowly increased after this time until ECC injection was terminated at +3697 s. After ECC injection was terminated drainage of the reactor vessel begun again and top portion of the core started heat up at 5323 s, see Figure C-3. When cladding temperature in the top portion of the core exceeded 1000°C, electric power supply to core was switched off and experiment was terminated.

During first seconds of calculation calculated primary system depressurization rate was higher than measured one despite the fact that calculated break flow was slightly higher than measured one, see Figure C-2 and C-21. But overall agreement between both parameters (primary system pressure and break flow) during the whole transient course was very good (according to Ref. 4 - error of measured break flow can be up to 20 – 30 %).

After break was opened pressurizer heaters automatically tried to prevent the system pressure from decreasing and increased heating to maximum value. But at +1.0 s the system pressure decreased below heaters setpoint 13.73 MPa and heaters were switched off.

At 1.0 s the system pressure decreased below 13.73 MPa, thus SCRAM signal was simulated and MCP coast down begun. SCRAM signal was followed by the core and by-pass power reduction at +2.5 s. Time courses of power reduction of core and by-pass and MCP coast down were specified as boundary conditions based on experiment data see Figure C-9, C-10, and C-6.

In the beginning of the experiment the high break flow was not compensated and the primary system was emptying. Calculated emptying rate of pressurizer was nearly the same like measured. The part of pressurizer above heaters (gauges YP01L02) was emptied at +15 s in calculation whereas in experiment at 16 s (see Figure C-4).

The pressure in steam generators is presented in Figure C-11. The measured and calculated pressure increased rapidly to a peak 6.86 MPa in experiment and 6.81 in calculation when isolation valve at common part steam header RA06S01 was closed. The secondary system pressure did not achieve a value of BRU-A activation neither in experiment nor in calculation. Calculated absolute value of the secondary pressure peak agree with experiment very well. Calculated pressure change from steady state level pressure to pressure peak was 20% higher, because of lower calculated steady state level pressure before pressure peak (approximately 0.15 MPa lower than measured one). It was probably due to the used heat structure correlation that is not capable to perfectly model the coil tube bundle in the steam generator. During the rest of calculation secondary pressure decreased more quickly than in experiment. It was probably caused by overshooting of heat losses of SG vessels and steam lines (primary loops were emptied and there were no heat exchange between primary and secondary side of SG).

When the primary system pressure dropped below the secondary system pressure (between 80 and 90 s in the experiment and approximately at 60 s in the calculation), steam generator did not remove primary heat any more. As the primary pressure continued to decrease below 10.88 MPa and HPI injection begun at 18 s in the experiment and in 16 s in calculation, mass flow 0.105 kg/s was injected to the hot leg loop #1, see Figure C-5.

Primary system pressure was continuously decreasing, so the primary system pressure dropped below accumulators pressure at +108 s in the experiment and at +121 s in the calculation.. Condensation of some of the steam in the system by the cold ECC liquid caused the depressurization rate to increase. When levels in accumulators fell down to 0.1 m they were cut off at +489-500 s in experiment and at +579-632 s in calculation. The accumulators injection was followed by LPI injection at + 383 s in experiment and at + 347 s in calculation, so the mass flow 0.2 kg/s was provided to the hot and cold leg of loop #1. As the system pressure decreased, the total ECC injection could compensate decreasing break flow. Collapsed level in the reactor vessel achieved minimal value approximately 600 s (see Figure C-22) after this time slowly increased until ECC injection was terminated at +3697 s in experiment and at +3730 s in calculation. HPI and LPI injection were terminated when total water delivery achieved 1.72 m^3 (simulation of emptying ECC tanks). After ECC injection was terminated drainage of the reactor vessel begun again, see pressure differences in core portion of vessel (YC01DP07 – DP10) in Figure C-18 and C-19.

Deviation of calculated cladding temperature from experiment data was up to 5 °C approximately until +2600 s – see Figure C-3. During following time course deviation increased up to 18 °C (accuracy of measured cladding temperature channel YC01T10 was ±10.8°C). Increasing deviation of calculated cladding temperature from the experiment was caused by lower calculated average void fraction in the core region of vessel – it means that TRACE calculated less liquid in core than was measured in experiment, see pressure differences in core portion of vessel (YC01DP07 – DP10) Figure C-18 and C-19. Possible reason of undershooting of core liquid inventory are that CCFL at the core exit is preventing liquid from falling back into the core region in the calculation

In the calculation, the heat-up of top portion of the core started at +4950 s, it was about approximately 350 s earlier than in test, see Figure C-3. Calculated earlier start of heat-up was caused by less calculated water inventory in the core than in the test - possible reasons are mentioned in a previous paragraph. Calculated heat-up rate was higher than those in the test, see Figure C-3. Similar behavior was observed in other post-test calculations of an upper plenum small break performed in the code RELAP5/MOD3.2. (Ref. 7, 8) and also in the code CATHARE (Ref. 11).

At +5593 s power supply to core and by-pass was switched off and experiment was terminated.

5.4 RELAP5 transient calculation

The post-test calculation of UP-11-07 experimental test at PSB-VVER facility started at the time 0 s with initiating event – upper plenum break 11 % (to simulate accumulator water supplying pipe rupture). A comparison of calculated and experimental times of the occurrence of main events is presented in the Table 10. Time courses of all important parameters and their comparison with experimental data are presented in Appendix D.

The comparisons between the measured data and calculations will be presented and discussed here.

Calculation started at 0s by opening the valve on leakage line from upper plenum.

The calculated break flow was little more higher than was measured immediately after the break was opened so the system pressure decreased a little more rapidly, see Figure D-2 and D-21. But overall agreement both parameters during the whole transient course with measured data were very good (Ref. 5 - error of measured break flow can be up to 20 – 30 %).

After the break was opened pressurizer heaters automatically tried to prevent the system pressure from decreasing and increased heating to maximum value. But at +3.5 s the system pressure decreased below heaters setpoint 13.73 MPa so heaters were switched off.

At 3.5 s the system pressure decreased bellow 13.73 MPa and SCRAM signal was simulated. SCRAM signal was followed by the core and by-pass power reduction and MCP coast down. Time course of power reduction of the core and by-pass and MCP coast down were specified as boundary conditions based on experimental data, see Figure D-9, D-10, and D-6.

In the beginning of the experiment the high break flow was not compensated so the primary system was emptying. Calculated higher primary system depressurization rate caused quicker emptying of pressurizer. The pressurizer portion above heaters (gauges YP01L02) was emptied at +13 s in the calculation whereas in experiment at 16 s.

The pressure in steam generators is presented in Figure D-11. The measured and calculated pressure increased rapidly to a peak 6.86 MPa in experiment whereas to 7.10 MPa in the calculation after isolation valve at common part steam header RA06S01 was closed. The secondary system pressure did not achieve a value of BRU-A activation neither in experiment nor in calculation. Calculated secondary pressure decrease was slightly higher than in the experiment. It was probably due to the used heat structure correlation that is not capable to perfectly model the coil tube bundle in the steam generator.

When the primary system pressure dropped below the secondary system pressure (between 80 and 90 s in the experiment and approximately at 55 s in the calculation), steam generator did not remove primary heat any more. The end of removing primary heat to secondary system was followed by temporary decreasing primary system depressurization rate and decreasing primary system cooldown rate see Figure D-1.

As the primary pressure continued to decrease below 10.88 MPa and HPI injection begun (at 18 s in the experiment and in 13 s in calculation), mass flow 0.105 kg/s was injected to hot leg loop #1, see Figure D-5.

Primary system pressure was continuously decreasing so the primary system pressure dropped below accumulators pressure at +108 s in the experiment and at +100 s in the calculation.. Condensation of some of the steam in the system by the cold ECC liquid caused the depressurization rate to increase. When level in accumulators fell down to 0.1 m accumulators were cut off at +489-500 s in experiment and at +528-535 s in calculation. The accumulators injection was followed by LPI injection at + 383 s in experiment and at + 405 s in calculation, mass flow 0.2 kg/s was provided to hot and cold leg loop #1. As the system pressure decreased, the total ECC injection could compensate decreasing break flow. Collapsed level in vessel achieved minimal value approximately 600 s (see Figure D-22) after this time slowly increased until ECC injection was terminated at +3697 s in experiment and at +3770 s in calculation. HPI and LPI injections were terminated when total water delivery achieved 1.72 m^3 – simulation of emptying ECC tanks. After ECC injection was terminated drainage of vessel begun again, see pressure differences in core portion of vessel (YC01DP07 – DP10) in Figure D-18 and D-19.

Deviation of calculated cladding temperature from experiment data was up to 5 °C approximately until +2600 s. During following time course deviation increased up to 20 °C. Increasing deviation of calculated cladding temperature from the experiment was caused by lower calculated average void fraction in the core region of vessel – it means that RELAP5 calculated less liquid in the core than was measured in experiment, see pressure differences in the core portion of the reactor vessel (YC01DP07 – DP10) Figure D-18 and D-19. Possible reason of under prediction of core liquid inventory were that CCFL at the core exit was preventing liquid from falling back into the core region in the calculation, that inter-phase drag was over predicted at the core exit, again preventing liquid from separating from vapor flow and draining the lower portion of the reactor vessel.

In the calculation, the heat-up of top portion of core started at +4640 s, it was approximately 700 s earlier than in the UP-11-07 test, see Figure D-3. Calculated earlier start of heat-up was caused by less calculated water inventory in the core than in the test - possible reasons are mentioned in a previous paragraph. Calculated heat-up rate was higher than those in the test, see Figure D-3. Similar behavior was observed in other post-test calculations of an upper plenum small break performed in the code version RELAP5/MOD3.2. (Ref. 7, 8) and also in the code CATHARE (Ref. 11).

At +5593 s power supply to core and by-pass was switched off and experiment was terminated.

Table 10: Chronology of main events (TRACE and RELAP5 calc. vs. comparison)

Event	Time [s]		
	UP-11-07	RELAP5	TRACE
Initiating Event – break opens	0	0	0
Pressure in UP (YC01P16) < 13.73 MPa – SCRAM signal	2	3.4	1
Start of MCP coastdown	2	3.5	1
Start of core and core by-pass power reduction	3	3.9	2.5
Stop of feedwater flow supply	9	11	10
Stop of steam removal from steam generators	11	11	10
Pressure in UP (YC01P16) < 10.88 MPa	11	10	8

Event	Time [s]		
	UP-11-07	RELAP5	TRACE
Start of HPIS injection into Loop #1	18	13	17
PRZ empty (according to YP01L02)	16	13	15
UP pressure < SG pressure	84	55	60
Pressure in UP/DC < pressure in ACCUs	108 ÷ 110	100 ÷ 110	121 ÷ 129
Start of ACCU-1 injection	110	100	121
Start of ACCU-2 injection (switched off)	-	-	-
Start of ACCU-3 injection	108	109	129
Start of ACCU-4 injection	108	110	127
Start of LPIS injection into Loop #1	383	405	347
End of accumulators injection (TH01-04L01 < 0.1 m)	489 ÷ 500	528 ÷ 535	579 ÷ 632
End of injection of LPIS injection into hot leg of loop #1	3648	3770	3730
End of injection of LPIS and HPIS injection into hot leg of loop #1	3697	3770	3730
Heat up of top portion of core (YC01T10)	5323	4640	4950
End of the experiment	5593	5593	5593

5.5 Quantitative Assessment of the Calculations

To quantify agreement of presented TRACE and RELAP5 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 25 including choice of particular metrics and their weighting factors - and see the Table 11.

Table 11: 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 26, 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 12 contains values of acceptability criterions range and their meaning.

Table 12: Acceptability criterions

AA_{tot} range	FOM range	Abbreviation	Color indication
$AA_{tot} \leq 0.30$	FOM ≥ 0.77	Very good code predictions	green
$0.30 < AA_{tot} \leq 0.50$	0.67 ≤ FOM < 0.77	Good code predictions	blue
$0.50 < AA_{tot} \leq 0.70$	0.59 ≤ 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 65 parameters were chosen including:

- Primary pressure
- Fuel cladding temperature
- Pressurizer water level
- Break flow
- RPV Pressure drops
- LOOPs pressure drops and mass flow rates
- Accumulator water levels and pressures

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

The UP-11-07 experiment was a long lasting transient where many different TH phenomena were expected. In order to carefully assess both TRACE and RELAP5 calculations the whole time course isdivided into three time windows of interest as follows:

- W1: 0 ÷ 100 s – an early stage of the test when only HPIS flow was provided
- W2: 100 ÷ 3800 s – a middle stage of the test when HPIS, LPIS and accumulators flow was provided
- W3: 3800 ÷ 5600 s – the final stage of the test when no ECCS 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 12 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 13: Evaluation of FOM for UP-11-07 test

		RELAP				TRACE			
		W0	W1	W2	W3	W0	W1	W2	W3
Parametr	TAG	0 – 5596	0 – 100	100 – 3800	3800-5596	0 – 5596	0 – 100	100 – 3800	3800-5596
Water level in accumulator of ECCS TH01B01	TH01L01	0.967	0.820	0.965	N/A	0.939	0.827	0.935	N/A
Pressure of water in accumulator of ECCS TH01B01	TH01P01	0.955	0.751	0.951	N/A	0.920	0.747	0.912	N/A
Water level in accumulator of ECCS TH02B01	TH02L01	0.945	0.921	0.947	N/A	0.945	0.922	0.946	N/A
Pressure of water in accumulator of ECCS TH02B01	TH02P01	0.935	0.811	0.935	N/A	0.941	0.808	0.944	N/A
Water level in accumulator of ECCS TH03B01	TH03L01	0.946	0.897	0.943	N/A	0.936	0.905	0.932	N/A
Pressure of water in accumulator of ECCS TH03B01	TH03P01	0.974	0.880	0.971	N/A	0.942	0.890	0.936	N/A
Water level in accumulator of ECCS TH04B01	TH04L01	0.952	0.929	0.950	N/A	0.937	0.929	0.932	N/A
Pressure of water in accumulator of ECCS TH04B01	TH04P01	0.977	0.888	0.975	N/A	0.947	0.892	0.942	N/A
Break line flow rate	XL01F01	0.757	0.821	0.678	0.530	0.762	0.813	0.684	0.539
Press drop in HL of loop 1 (elevation part)	YA01DP02	0.597	0.841	0.593	0.653	0.770	0.852	0.751	0.652
Press drop in CL of loop 1 (from SG outlet to the flow meter)	YA01DP04	0.729	0.894	0.662	0.640	0.755	0.913	0.688	0.532
Press drop in CL of loop 1 (from flow meter to mechanical filter)	YA01DP05	0.732	0.887	0.724	0.791	0.704	0.808	0.681	0.803
Press drop in CL of loop 1 (section with a mechanical filter, pump and regulating valve)	YA01DP06	0.638	0.829	0.611	0.635	0.692	0.845	0.660	0.824
Press drop on the filter in CL of loop 1	YA01DP08	0.747	0.763	0.742	0.677	0.755	0.739	0.754	0.651
Press drop on MCP of loop 1	YA01DP09	0.646	0.886	0.609	0.805	0.678	0.916	0.641	0.598
Press drop in "cold" header of SG 1 YB01W01	YA01DP13	0.871	0.878	0.713	0.433	0.853	0.816	0.704	0.431
Press drop in "hot" header of SG 1 YB01W01	YA01DP14	0.694	0.913	0.590	0.613	0.893	0.916	0.774	0.618
Flow rate in the pump circulation loop YD01D01	YA01F01	0.698	0.784	N/A	N/A	0.705	0.805	N/A	N/A
Press drop in HL of loop 2 (elevation part)	YA02DP02	0.627	0.709	0.359	0.752	0.843	0.848	0.748	0.751
Press drop in CL of loop 2 (from SG outlet to the flow meter)	YA02DP04	0.884	0.847	0.761	0.748	0.861	0.818	0.764	0.731
Press drop in CL of loop 2 (from the flow meter to mechanical filter)	YA02DP05	0.751	0.905	0.710	0.726	0.740	0.841	0.704	0.694

		RELAP				TRACE			
		W0	W1	W2	W3	W0	W1	W2	W3
Parametr	TAG	0 – 5596	0 – 100	100 – 3800	3800-5596	0 – 5596	0 – 100	100 – 3800	3800-5596
Press drop in CL of loop 2 (section with mechanical filter, pump and regulating valve)	YA02DP06	0.744	0.919	0.734	0.648	0.734	0.922	0.719	0.648
Press drop on filter in CL of loop 2	YA02DP08	0.700	0.718	0.706	0.597	0.716	0.701	0.731	0.594
Press drop on MCP of loop 2	YA02DP09	0.758	0.860	0.750	0.572	0.721	0.884	0.703	0.571
Press drop in "cold" header of SG 2 YB02W01	YA02DP13	0.905	0.891	0.790	0.733	0.853	0.781	0.715	0.732
Press drop in "hot" header of SG 2 YB02W01	YA02DP14	0.773	0.879	0.464	0.530	0.898	0.869	0.779	0.534
Flow rate in the pump circulation loop YD02D01	YA02F01	0.668	0.819	N/A	N/A	0.653	0.826	N/A	N/A
Press drop in HL of loop 3 (elevation part)	YA03DP02	0.805	0.815	0.707	0.678	0.807	0.796	0.737	0.674
Press drop in CL of loop 3 (from SG outlet to the flow meter)	YA03DP04	0.839	0.852	0.757	0.722	0.826	0.806	0.756	0.720
Press drop in CL of loop 3 (from the flow meter to mechanical filter)	YA03DP05	0.712	0.901	0.687	0.390	0.740	0.848	0.671	0.723
Press drop in CL of loop 3 (section with mechanical filter, pump and regulating valve)	YA03DP06	0.687	0.909	0.689	0.478	0.727	0.915	0.712	0.654
Press drop on filter in CL of loop 3	YA03DP08	0.667	0.687	0.613	0.574	0.712	0.677	0.661	0.638
Press drop on MCP of loop 3	YA03DP09	0.710	0.898	0.717	0.410	0.732	0.923	0.717	0.751
Press drop in "cold" header of SG 3 YB03W01	YA03DP13	0.863	0.877	0.773	0.743	0.867	0.898	0.762	0.739
Press drop in "hot" header of SG 3 YB03W01	YA03DP14	0.913	0.895	0.790	0.759	0.910	0.877	0.789	0.755
Flow rate in the pump circulation loop YD03D01	YA03F01	0.616	0.790	N/A	N/A	0.616	0.837	N/A	N/A
Press drop in HL of loop 4 (elevation part)	YA04DP02	0.652	0.822	0.531	0.563	0.671	0.829	0.551	0.562
Press drop in CL of loop 4 (from SG outlet to the flow meter)	YA04DP04	0.917	0.896	0.743	0.547	0.853	0.798	0.738	0.539
Press drop in CL of loop 4 (from the flow meter to mechanical filter)	YA04DP05	0.694	0.916	0.669	0.532	0.687	0.841	0.661	0.525
Press drop in CL of loop 4 (section with mechanical filter, pump and regulating valve)	YA04DP06	0.679	0.883	0.689	0.518	0.677	0.891	0.687	0.513
Press drop on filter in CL of loop 4	YA04DP08	0.708	0.534	0.715	0.612	0.707	0.533	0.716	0.602
Press drop on MCP of loop 4	YA04DP09	0.727	0.879	0.731	0.586	0.724	0.900	0.727	0.584
Press drop in "cold" header of SG 4 YB04W01	YA04DP13	0.889	0.883	0.720	0.457	0.857	0.825	0.685	0.455

		RELAP				TRACE			
		W0	W1	W2	W3	W0	W1	W2	W3
Parametr	TAG	0 – 5596	0 – 100	100 – 3800	3800-5596	0 – 5596	0 – 100	100 – 3800	3800-5596
Press drop in "hot" header of SG 4 YB04W01	YA04DP14	0.818	0.928	0.465	0.568	0.825	0.816	0.446	0.543
Flow rate in the pump circulation loop YD04D01	YA04F01	0.622	0.816	N/A	N/A	0.615	0.774	N/A	N/A
Press drop on DC (upper part with inlet chamber)	YC01DP01	0.816	0.413	0.780	0.594	0.853	0.587	0.809	0.595
Press drop on DC (inlet in the middle part)	YC01DP02	0.753	0.875	0.741	0.757	0.713	0.837	0.703	0.755
Press drop on DC (middle part)	YC01DP03	0.643	0.938	0.563	0.666	0.684	0.915	0.614	0.682
Press drop on DC (outlet from the middle part)	YC01DP04	0.667	0.899	0.694	0.438	0.710	0.869	0.747	0.447
Press drop in lower DC chamber	YC01DP05	0.772	0.890	0.773	0.592	0.804	0.929	0.803	0.628
Press drop between DC LC and CS	YC01DP06	0.734	0.888	0.721	0.609	0.723	0.899	0.703	0.674
Press drop in the channel with fuel rod simulators (lower part)	YC01DP07	0.619	0.842	0.619	0.596	0.615	0.857	0.619	0.571
Press drop in the channel with fuel rod simulators (inlet in the middle part)	YC01DP08	0.542	0.837	0.549	0.523	0.607	0.855	0.610	0.571
Press drop in the channel with fuel rod simulators (outlet from the middle part)	YC01DP09	0.618	0.872	0.523	0.687	0.694	0.867	0.607	0.727
Press drop in the channel with fuel rod simulators (upper part)	YC01DP10	0.587	0.871	0.554	0.610	0.691	0.938	0.662	0.683
Press drop between CS and UP	YC01DP11	0.651	0.886	0.555	0.692	0.731	0.780	0.670	0.722
Press drop in UP (on the insert)	YC01DP12	0.724	0.922	0.675	0.754	0.749	0.819	0.715	0.770
Press drop in UP (lower part)	YC01DP13	0.763	0.889	0.751	0.745	0.752	0.744	0.742	0.740
Press drop in UP (middle part)	YC01DP14	0.791	0.857	0.673	0.545	0.777	0.828	0.658	0.566
Press drop in UP (top piece)	YC01DP15	0.899	0.870	0.745	0.516	0.882	0.784	0.731	0.516
Press drop between DC and UP	YC01DP16	0.698	0.899	0.661	0.698	0.701	0.823	0.666	0.698
Press drop on CS BP	YC01DP17	0.581	0.938	0.563	0.609	0.637	0.910	0.621	0.638
Pressure of coolant in UP	YC01P16	0.966	0.948	0.967	0.533	0.945	0.935	0.950	0.570
Fuel rod simulator wall temperature (protection)	YC01T10	0.820	0.828	0.946	0.772	0.777	0.949	0.949	0.677
Water level in pressurizer YP01B01	YP01L02	0.948	0.948	N/A	N/A	0.944	0.938	N/A	N/A
FOM avg	-	0.764	0.853	0.715	0.618	0.779	0.841	0.738	0.637

6. RUN STATISTICS

The transients were calculated on calculation server with Intel Xeon 5440 processor 2.83 GHz under GNU/Linux Debian 5.0 Lenny x64. The run statistics is shown in the following Table 14. The TRACE calculation run substantially slower than real time due to the application of 3-D vessel components.

Table 14: Run statistics

	RELAP5	TRACE
Number of components	310	336
Number of time steps	280 092	381 186
Transient time	5 600 s	5600 s
CPU time	2 818 s	114 806s
CPU time / Transient time	0.503	20.5

7. CONCLUSIONS

The main goal of these analyses was to assess the TRACE TH code and its predecessor the RELAP5/MOD3.3 using the 11% upper plenum break UP-11-07 in the large scale test facility PSB-VVER. The second reason of using two TH codes, RELAP5 and TRACE, is to compare overall capability of the new code to its predecessor to catch all important phenomena that take place/occur during investigated transients. 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 in an environment of assessed codes.

Comparisons of both post-test TRACE and RELAP5 calculations with experiment data proved that both TRACE and RELAP5 codes are capable to model PSB-VVER integral system effects reasonably. The calculated time courses of the main facility parameters was similar to that of the test, indicating that all of the significant events that occurred in the test were present in the calculation.

To quantify errors/deviation of presented TRACE and RELAP5 calculations the figure of merit (FOM) was evaluated using software ACAP. FOMs of 65 main measured and calculated parameters were evaluated analogously for TRACE and RELAP5 calculations. The following table shows the final average FOM evaluated for both calculations at pre-defined time windows of interest.

	FOM avg			
	W0	W1	W2	W3
Time	0 – 5596	0 – 100	100 – 3800	3800-5596
RELAP5	0.76 good prediction	0.85 very good prediction	0.72 good prediction	0.62 poor prediction
TRACE	0.78 very good prediction	0.84 very good prediction	0.74 good prediction	0.64 poor prediction

Presented overall final FOMs (time window W0) prove that the both codes predicted behavior of test facility during the whole transient acceptable, although TRACE 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 corresponds to the duration of the UP-11-07 test (5600 s), that was longer than “common” LOCA tests with duration of tenths of seconds. The accumulation of minor deviations (e.g. void distribution in the primary circuit in the UP-11-07 test) might lead to gradual increasing of deviations of main calculated parameters.

However the quantitative assessment gave mainly good or very good predictions of the selected main parameters (both for TRACE and RELAP5), following particular discrepancies were identified. Both codes predicted different liquid distribution over reactor vessel during the ECC injection part of experiments comparing to the measured data. TRACE and RELAP5 calculated less liquid in the core region of the vessel and more uniform core axial void profile than was measured. Possible reason of under prediction of core liquid inventory might be following: CCFL at the core exit was preventing liquid from falling back into the core region in the calculation, that inter-phase drag was over predicted at the core exit, again preventing liquid from separating from vapor flow and draining the lower portion of the reactor vessel. The different liquid inventory in the core caused faster heat-up of top portion of the core comparing to the measured data.

The calculation cost of the TRACE calculation was much higher than RELAP5 calculation. Worse time efficiency of the TRACE calculation was caused by using a very fine discretized VESSEL component representing FRS and UP. VESSEL component consists of 768 cells whereas the full RELAP input deck consists of 536 cells. If number of cells of VESSEL component is reduced to 40% it will increase calculation speed approximately 4 times, based on our experience.

8. REFERENCES

1. TRACE V5.0 THEORY MANUAL, Field Equations, Solution Methods, and Physical Models; Division of Systems Analysis, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; May 2012
2. TRACE V5.0 USER'S MANUAL, Volume 1: Input Specification; Division of Systems Analysis; Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; December 2011
3. TRACE V5.0 USER'S MANUAL, Volume 2: Modeling Guidelines; Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; July 2011
4. O. I. Melikhov, V. I. Melikhov; Analysis of LOCA thermal-hydraulics of VVER 1000. Description of PSB-VVER Test Facility; Moscow Power Engineering Institute (Technical University), NPP Department; Moscow; 2008
5. O. I. Melikhov, V. I. Melikhov; Analysis of LOCA thermal-hydraulics of VVER 1000. Upper Plenum Break 11%; Moscow Power Engineering Institute (Technical University), NPP Department; Moscow; 2008
6. P. Heralecký; Engineering Handbook modelu PSB-1.0; TES-Z-09-175; TES s.r.o. Třebíč; prosinec 2009
7. P.D. Bayless, O. Melikhov, V. Melikhov, Y. Parfenov, O. Gavritenkova, I. Elkin, I. Lipatov; RELAP5/MOD3.2 Assessment Using INSC SP-PSBV1; Seventh International Information Exchange Forum On SAFETY ANALYSIS FOR NUCLEAR POWER PLANTS OF VVER AND RBMK TYPES; Slovakia, Piešťany; 28-30 October 2003
8. B. Neykov, M. Cherubini, F.D. Auria; Relap 5 Thermal Hydraulic analysis 11% upper plenum leakage for PSB-WWER test facility and WWER NPP
9. A. Del Nevo, F.D. Auria, M. Mazini, M. Vylov, I.V. Elkin, A. Suslov; The Design of PSB-VVER Experiments Relevant to Accident Management; Journal of Power and Energy Systems; Vol2, No. 1, 2008
10. V. Blinkov, O.I. Melikhov; Effect of Steam Generátor Nodalization on RELAP5 Simulation of Loss-of-Feed-Water Transient in PSB-VVER Facility; Seventh International Information Exchange Forum On SAFETY ANALYSIS FOR NUCLEAR POWER PLANTS OF VVER AND RBMK TYPES; Slovakia, Piešťany; 27-31 October 2003
11. L. Sabotinov, P. Chevrier; Post-Test Analysis of 11% Break at PSB-VVER Experimental Facility using Cathare 2 Code; Journal of Power and Energy Systems; Vol2, No. 2, 2008
12. Thermal-Hydraulic Codes News, Volume 09, No.3&4; U.S. Nuclear Regulatory Commission; December 2009
13. P. Heralecký; Srovnávací výpočet testu PSB-VVER UP-11-07 v prostředí kódu TRACE 5.0; TES-Z-09-179; TES s.r.o.; Třebíč; prosinec 2009
14. Preliminary Safety Assessment Report, Rev.2 - Input Data for Safety Analysis of NPP Dukovany, Revision V5. 01-March-2007

15. Staudenmeier J. TRACE Code Modelling Capabilities. Proceedings of TRACE/SNAP User Workshop. Potomac, Maryland, 27 – 29 March 2006

16. Bajorek S. at al. TRACE V5.0 Assessment Manual – Main Report. Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission

17. Petruzzi A. and Auria F. Thermal-Hydraulic System Codes in Nuclear Reactor Safety and Qualification Procedures. Science and Technology of Nuclear Installations Volume 2008, Article ID 460795

18. WNA Reactor Database. March 3, 2008

19. OECD Support Group on the VVER Thermal-Hydraulic Code Validation Matrix. Validation Matrix for the Assessment of Thermal-Hydraulic Codes For VVER LOCA and Transients. NEA/CSNI/R(2001)4. 01 June 2001, Paris, France

20. O. I. Melikhov, V. I. Melikhov; Analysis of LOCA thermal-hydraulics of VVER 1000. Single-phase characteristics of the PSB-VVER main circulation pumps; Moscow Power Engineering Institute (Technical University), NPP Department; Moscow; 2008

21. RELAP5/MOD3.3 CODE MANUAL VOLUME I: Code Structure, System Models, and Solution Methods; Information System Laboratories, Inc., Rockville Maryland, Idaho Falls, Idaho; October 2010

22. RELAP5/MOD3.3 CODE MANUAL VOLUME II: User's Guide and Input Requirements; Information System Laboratories, Inc., Rockville Maryland, Idaho Falls, Idaho; October 2010

23. RELAP5/MOD3.3 CODE MANUAL VOLUME V: User's Guidelines; Information System Laboratories, Inc., Rockville Maryland, Idaho Falls, Idaho; October 2010

24. P. Heralecký; RELAP5/MOD3.3 Assessment Using Experiments in the PSB-VVER Facility; TES-Z-12-049; TES s.r.o. Třebíč; září 2012

25. Matthew D. Lazor; Evaluation of Assessment Techniques for Verification and Validation of the TRACE Nuclear Systems Code; The Pennsylvania State University, College of Engineering; December 2004

26. A. Prošek, B. Mavko; Quantitative Code Assessment with Fast Fourier Transformation Based Method Improved by Signal Mirroring; NUREG/IA-0220; Jožev Stefan Institute; Slovenia; Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission; Washington DC, 20555-0001; December 2009

APPENDIX A INPUT DECK NODALISATION SCHEMES

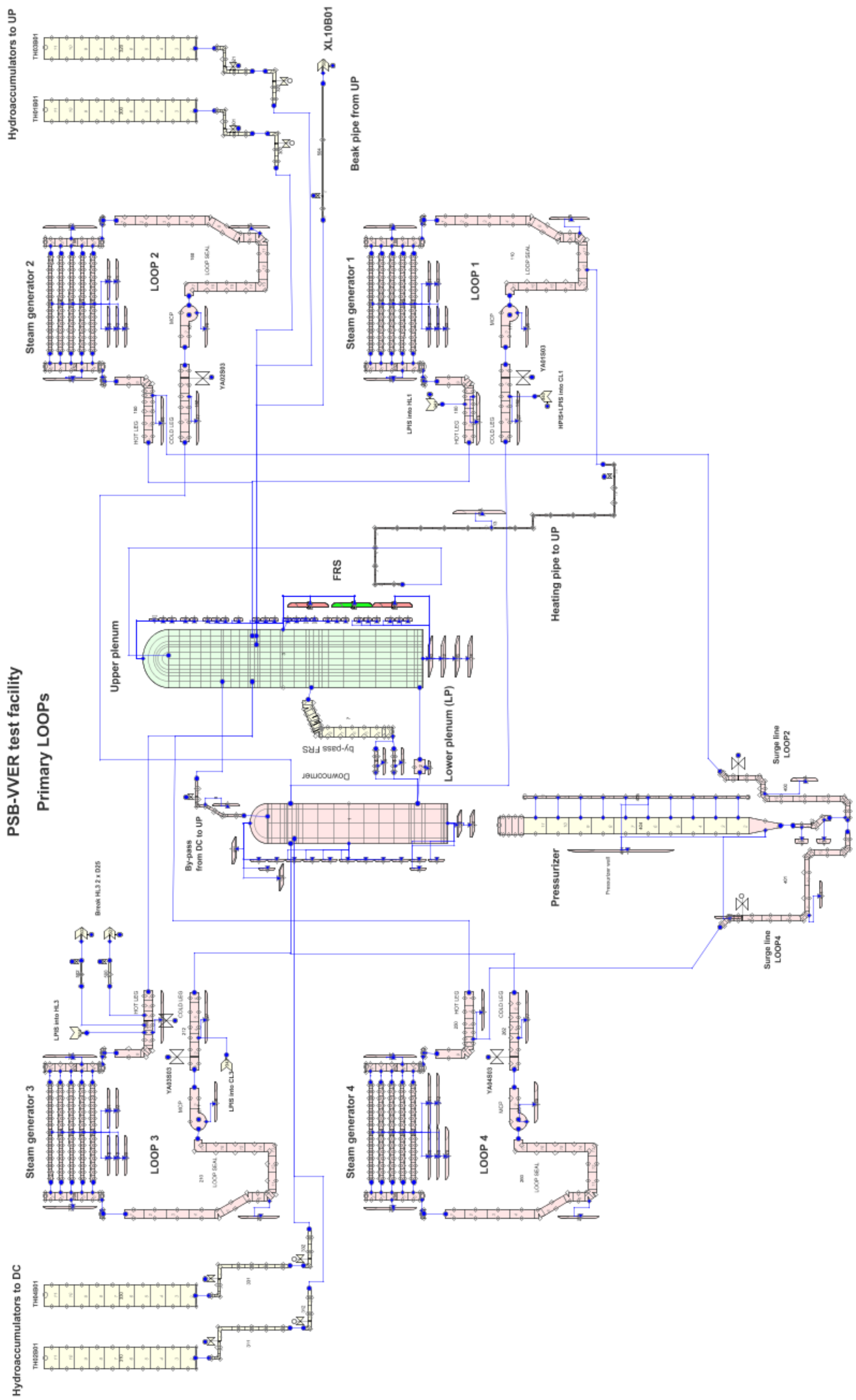


Figure A-1: TRACE Nodalization Scheme of Primary Circuit of PSB-VVER Facility

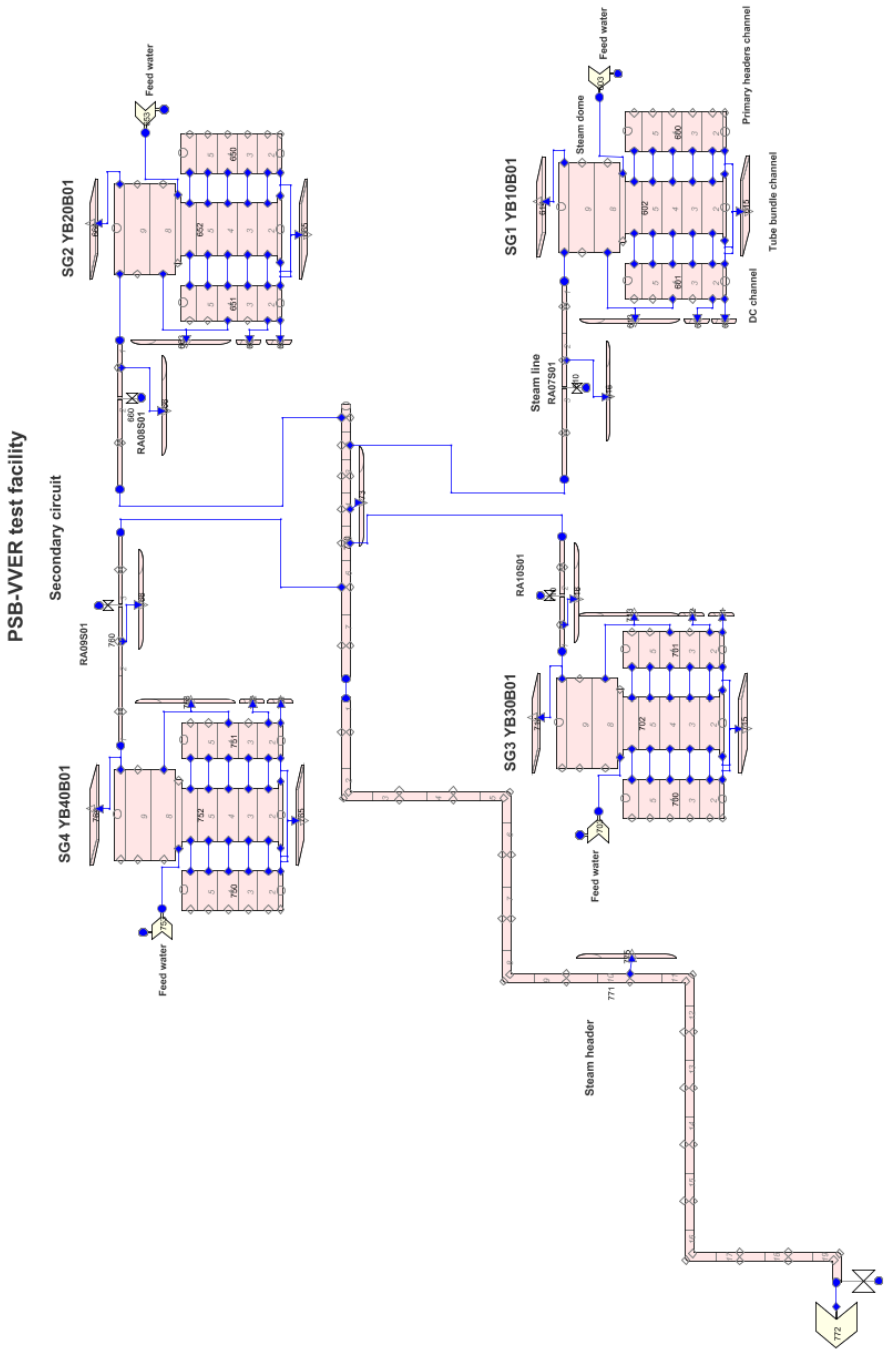


Figure A-2: TRACE Nodalization Scheme of Secondary Circuit of PSB-VVER Facility

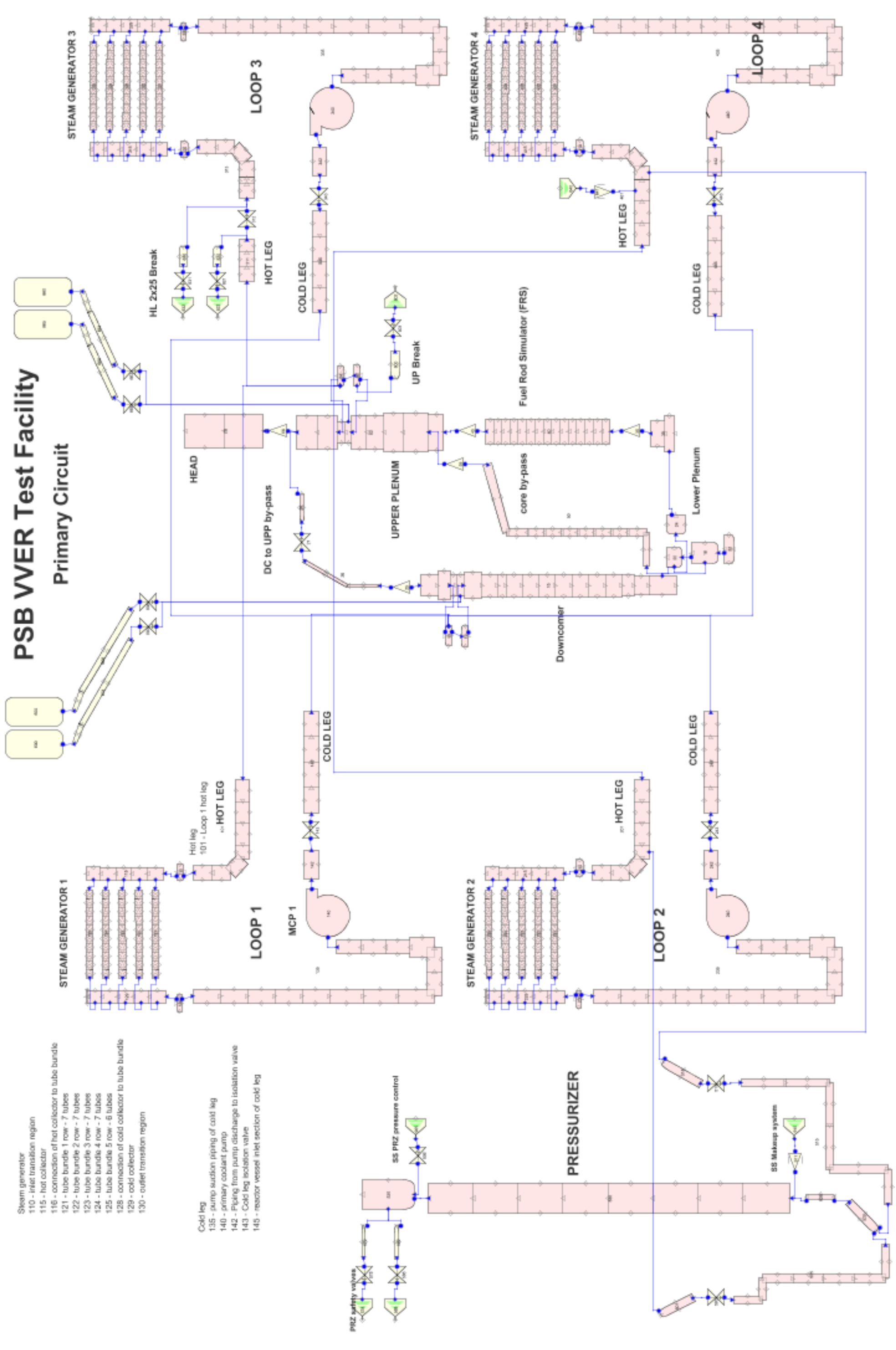


Figure A-3: RELAP5 Nodalization Scheme of Primary Circuit of PSB-VVER Facility

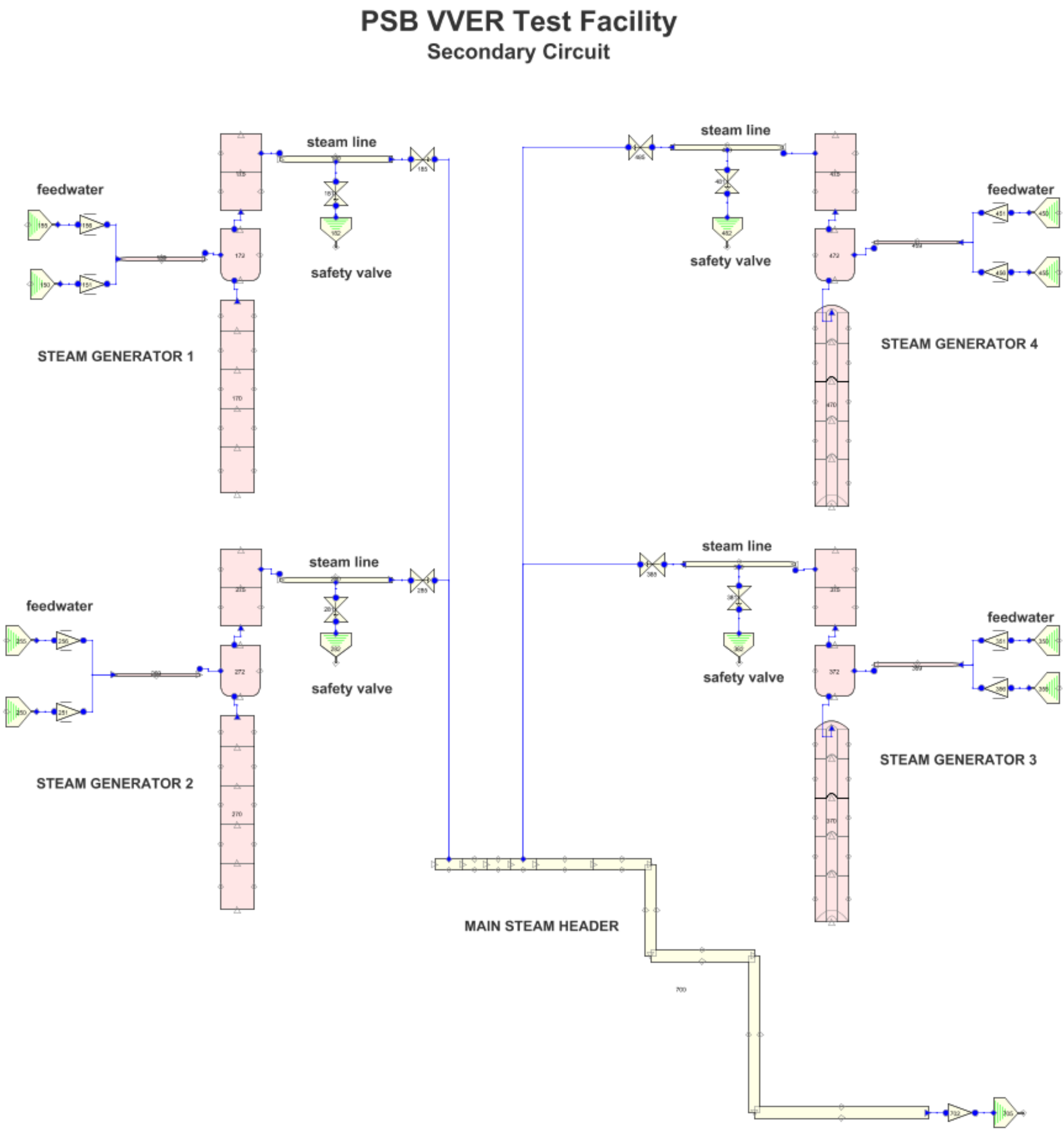


Figure A-4: RELAP5 Nodalization Scheme of Secondary Circuit of PSB-VVER Facility

APPENDIX B MEASUREMENT LOCALISATION AT PSB-VVER FACILITY

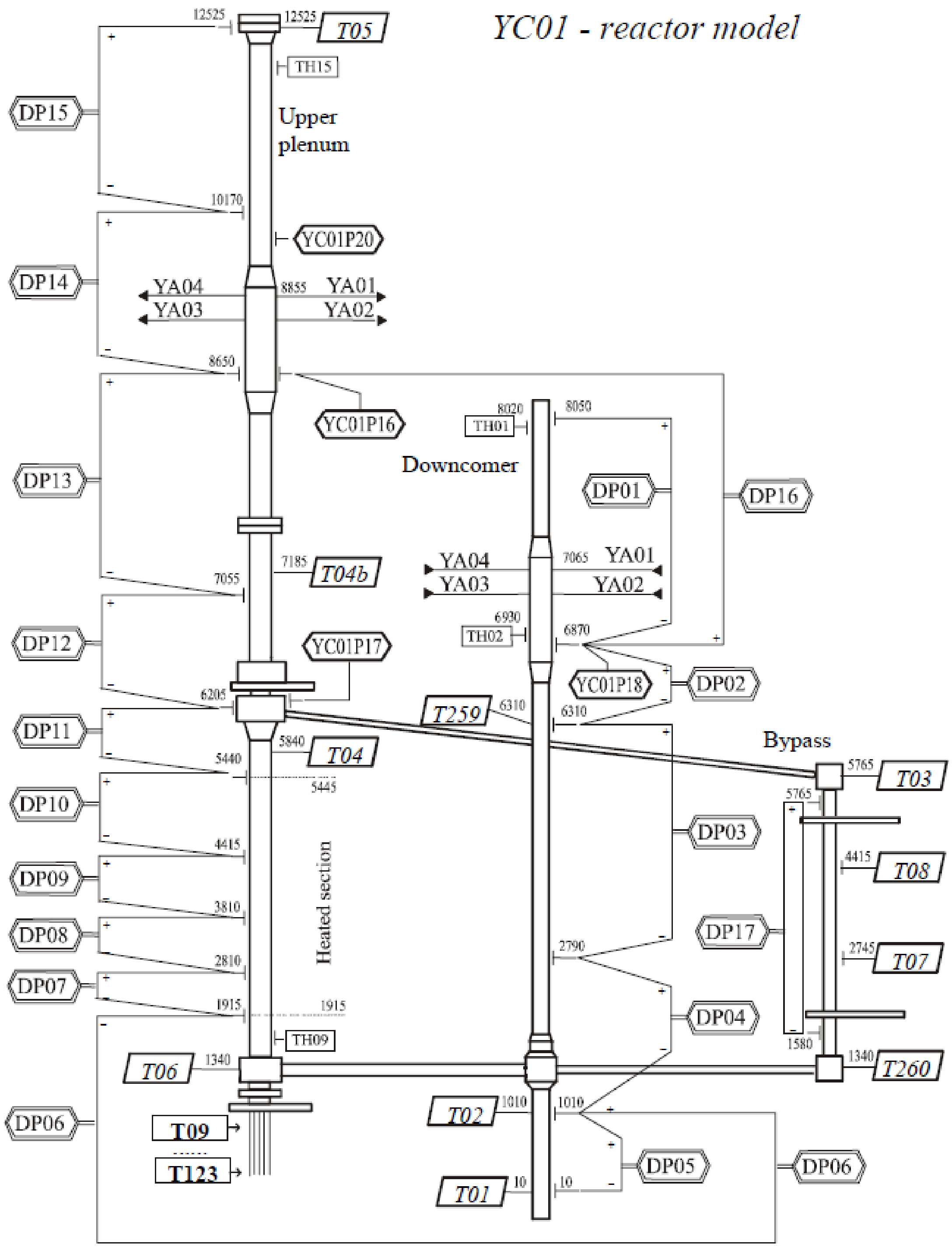


Figure B-1: PSB-VVER Reactor Model Measurements

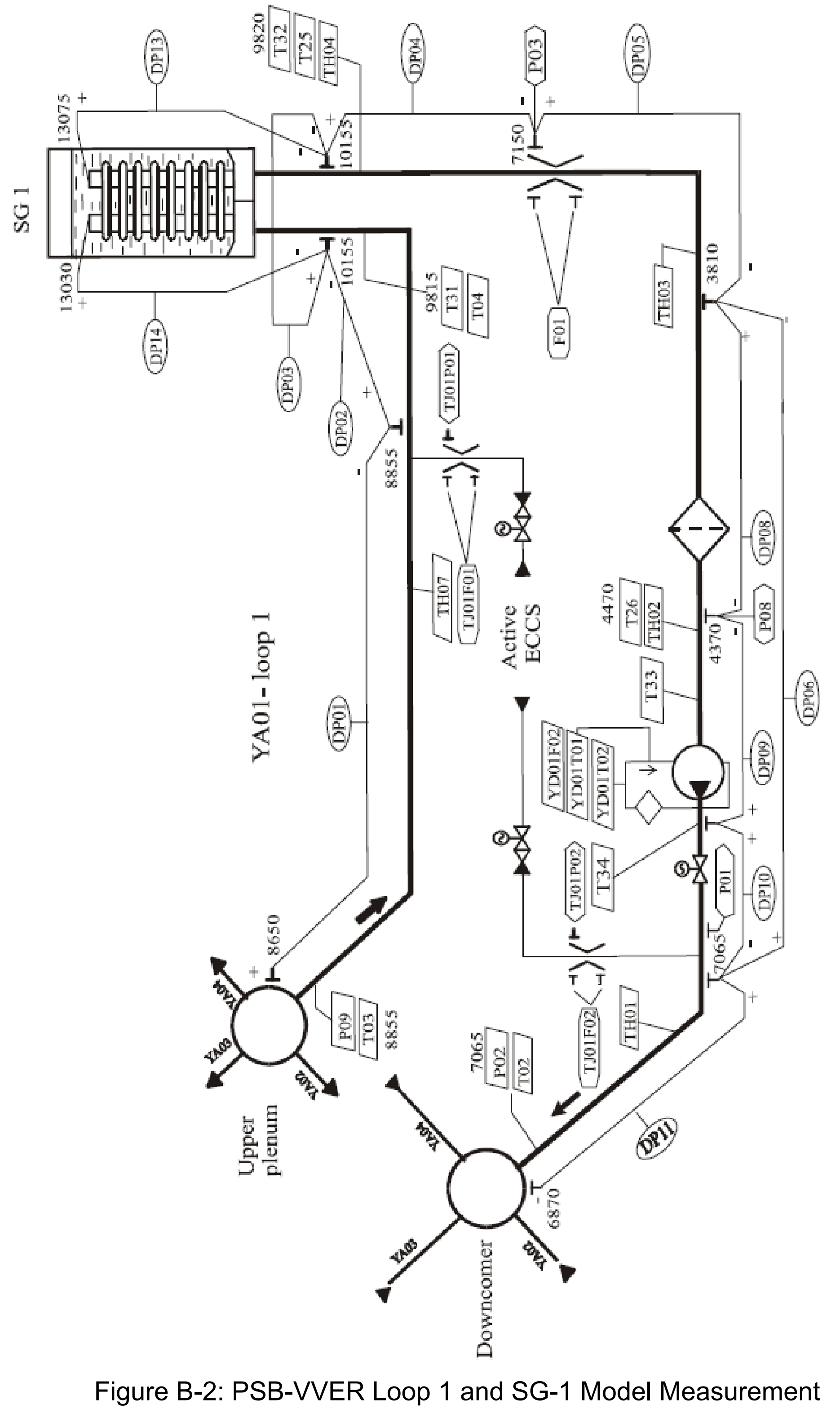


Figure B-2: PSB-VVER Loop 1 and SG-1 Model Measurement

APPENDIX C COMPLETE SET OF COMPARISON PLOTS FOR TRACE CALCULATION

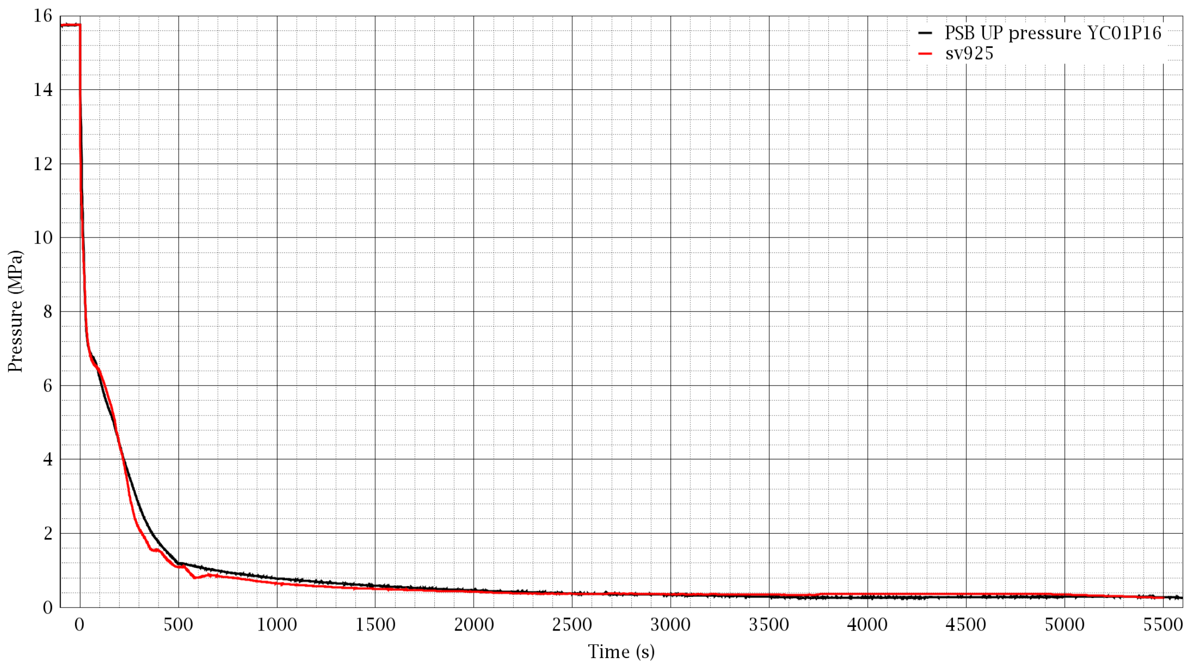


Figure C-1: Primary Pressure

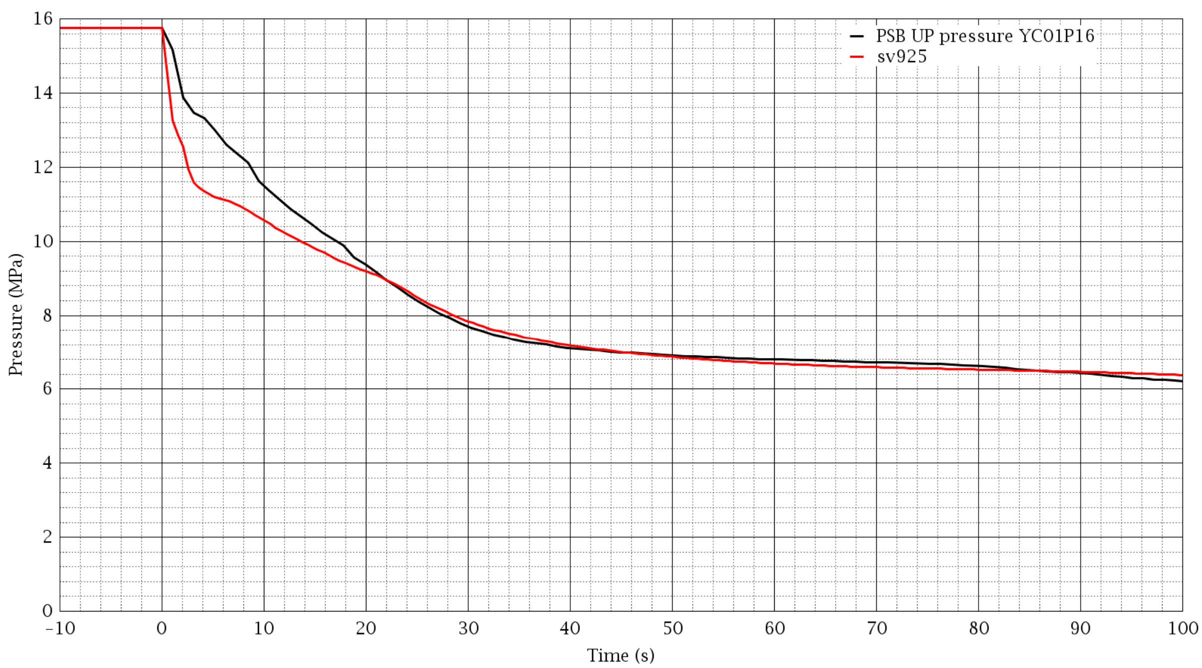


Figure C-2: Primary Pressure (detail)

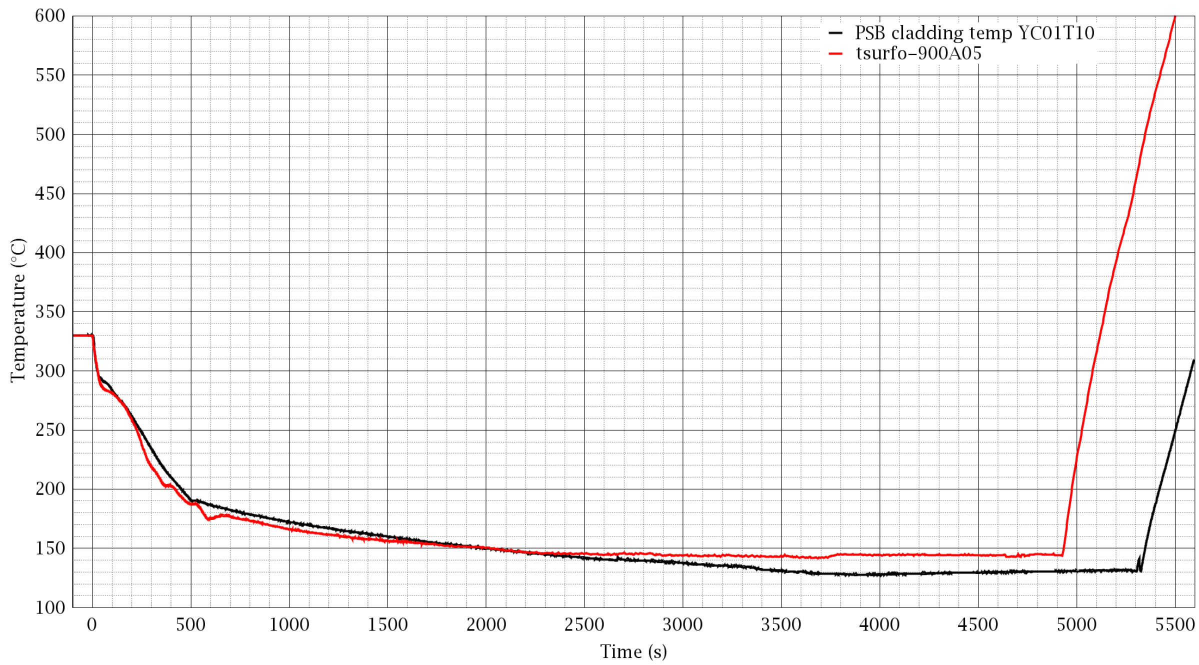


Figure C-3: Fuel Cladding Temperature (Top of the Core)

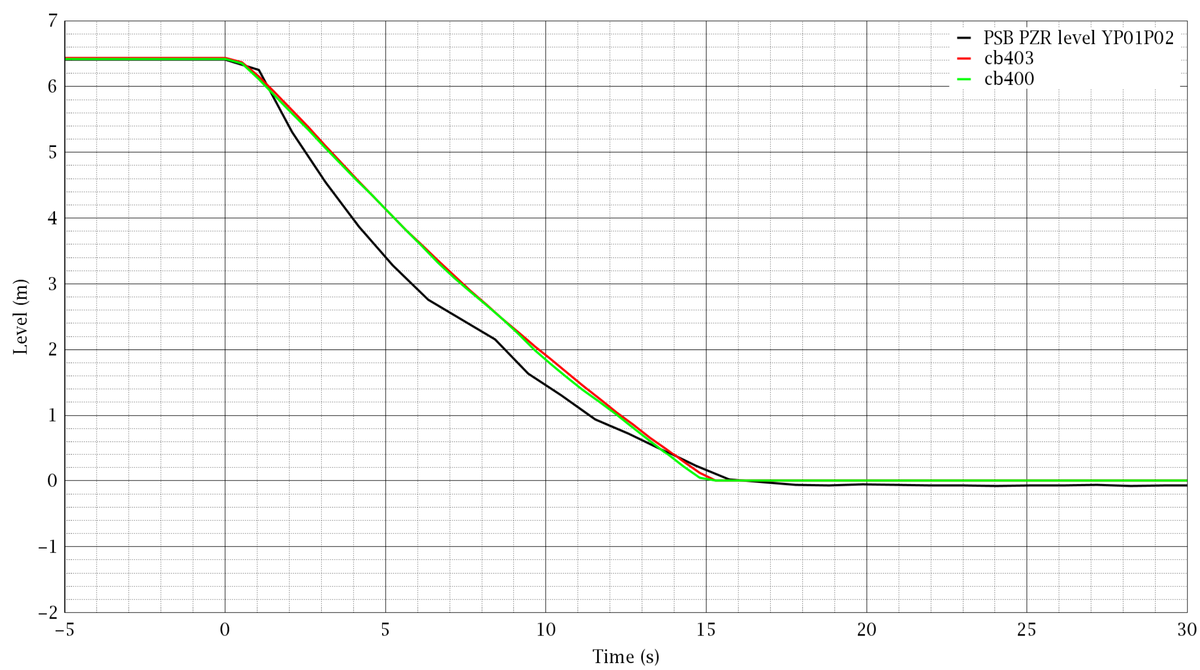


Figure C-4: Presurizer Level

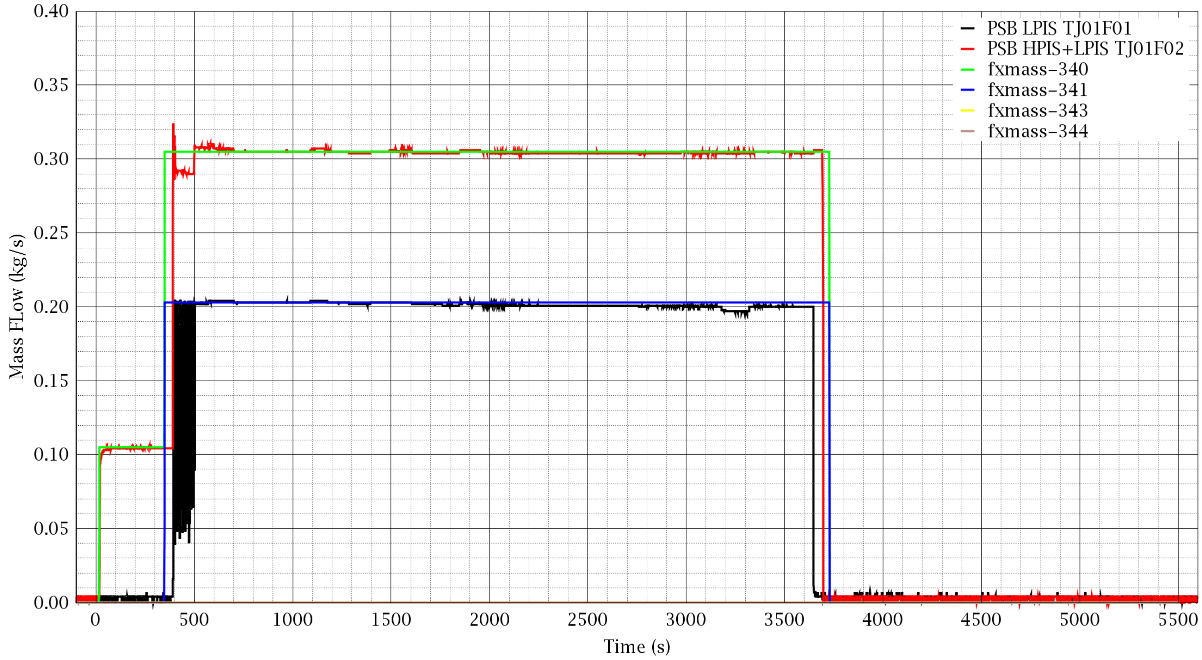


Figure C-5: HPIS + LPIS Flow (Boundary Condition)

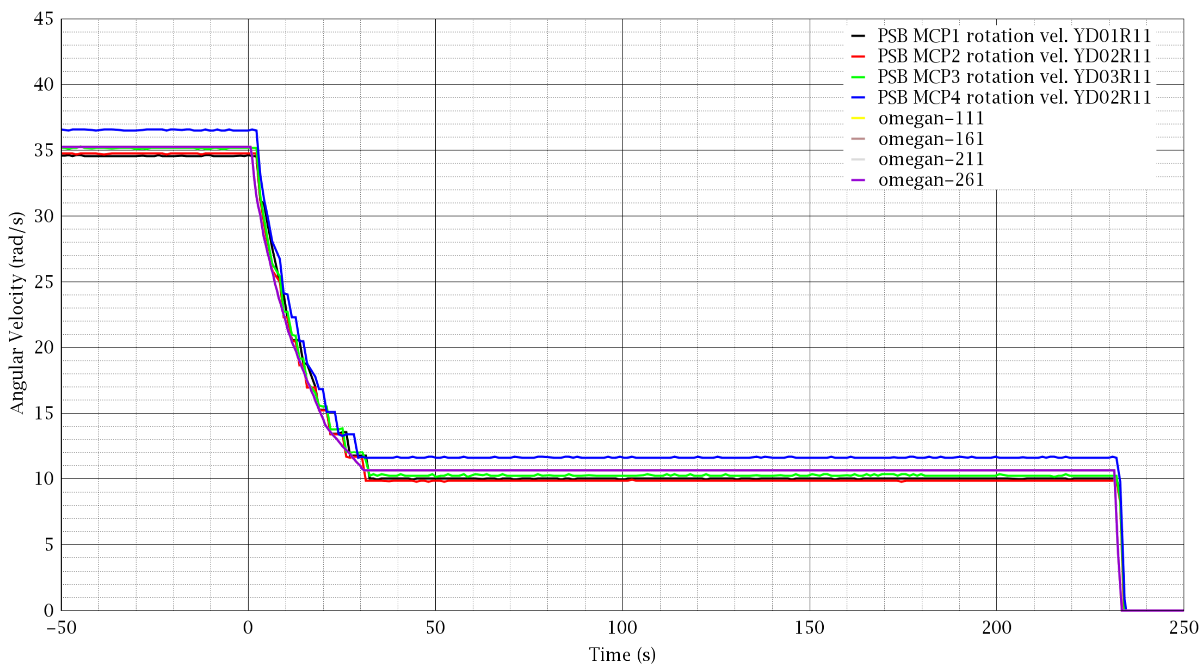


Figure C-6: MCP Rotor Speed (Boundary Condition)

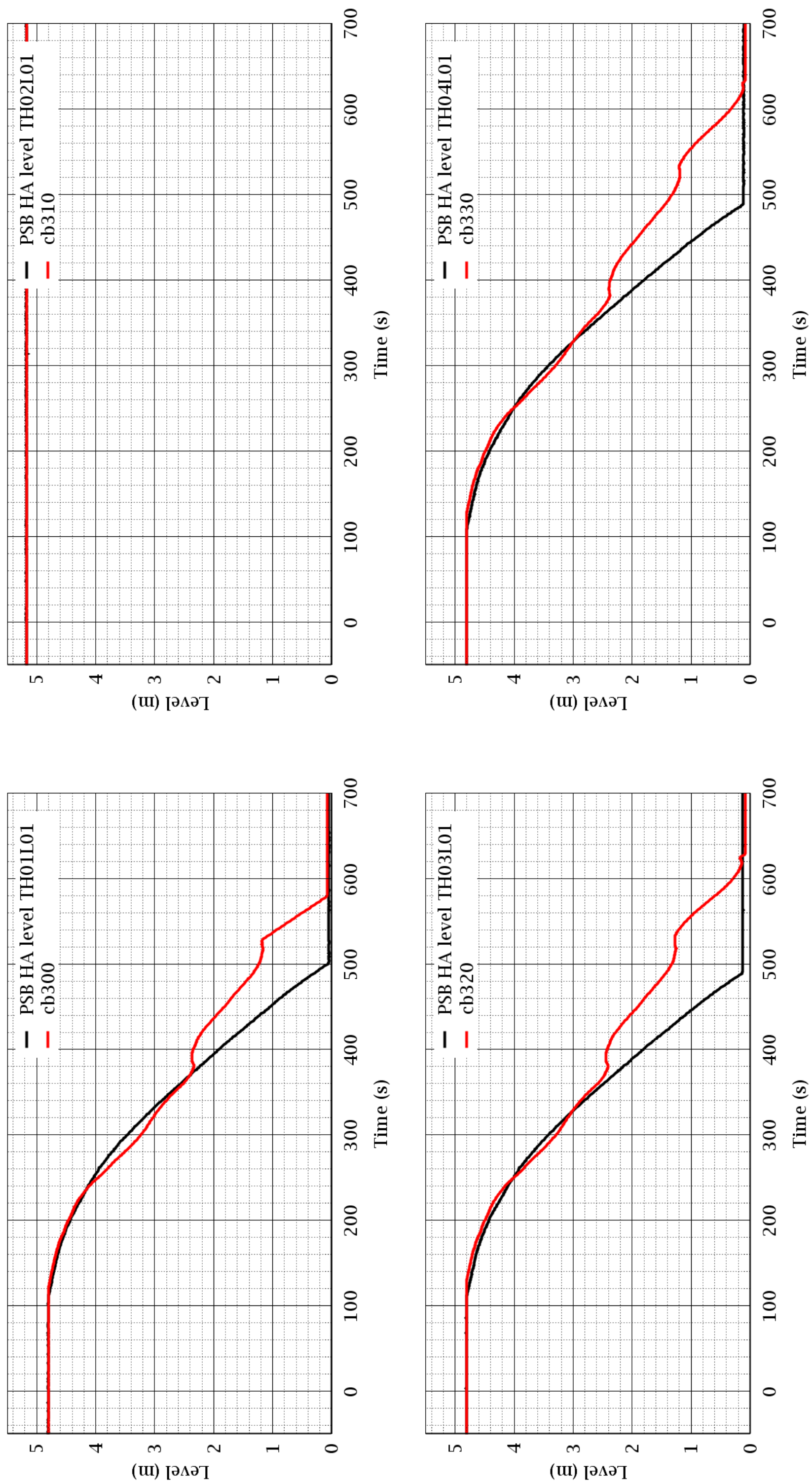


Figure C-7: Accumulators Levels

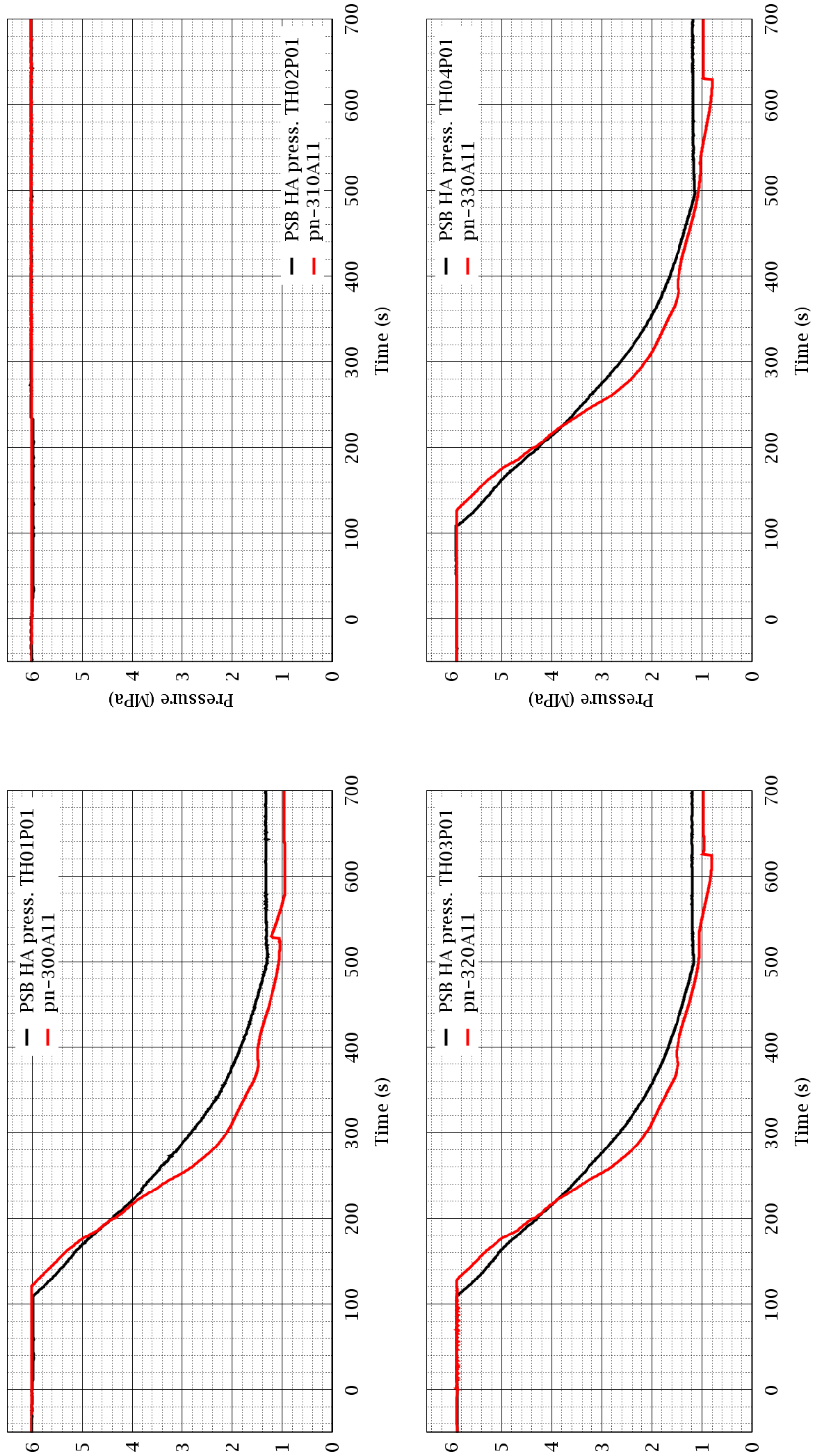


Figure C-8: Accumulators Pressure

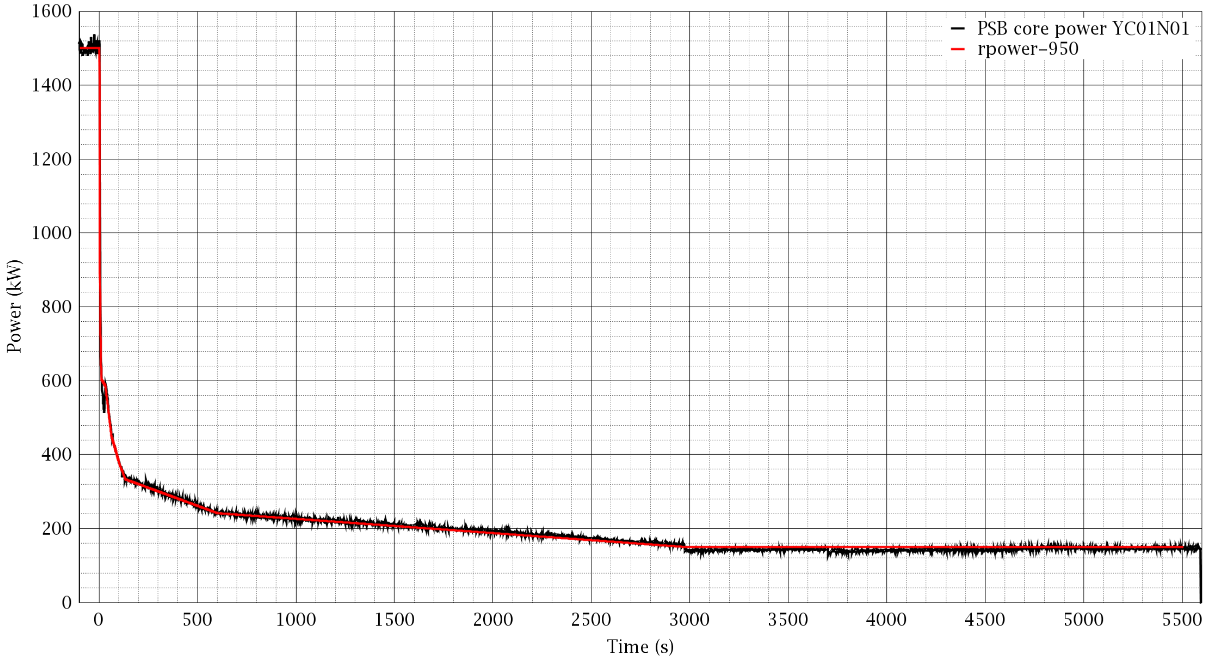


Figure C-9: Fuel Rod Simulator Power (Boundary Condition)

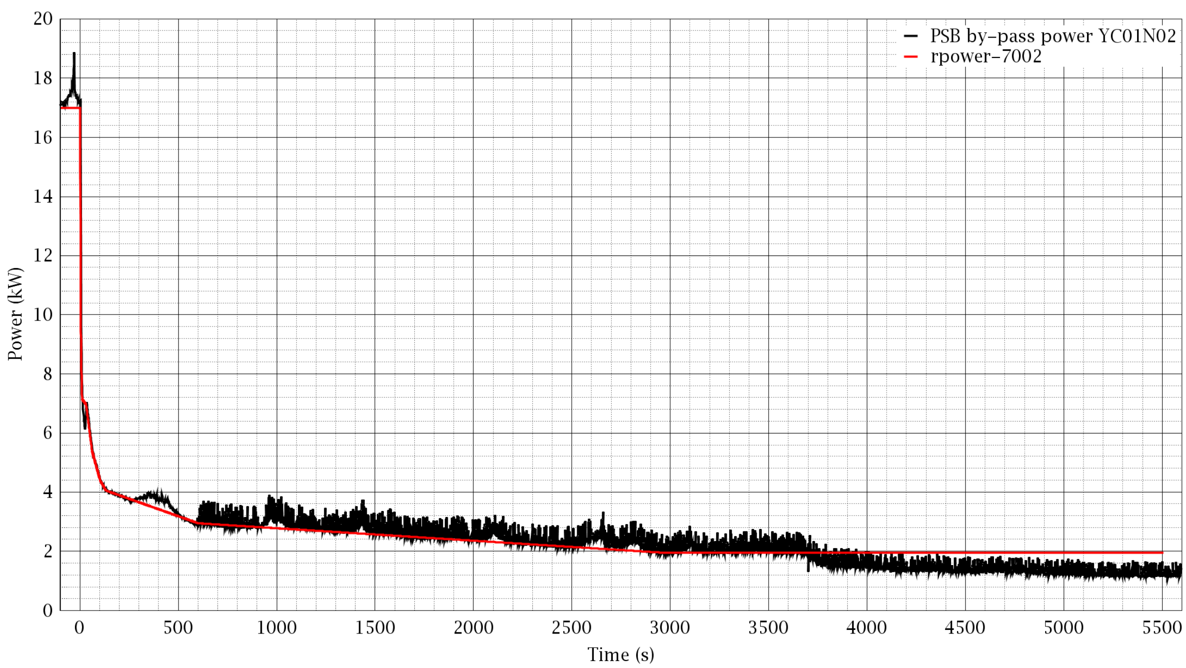


Figure C-10: Core By-pass Power (Boundary Condition)

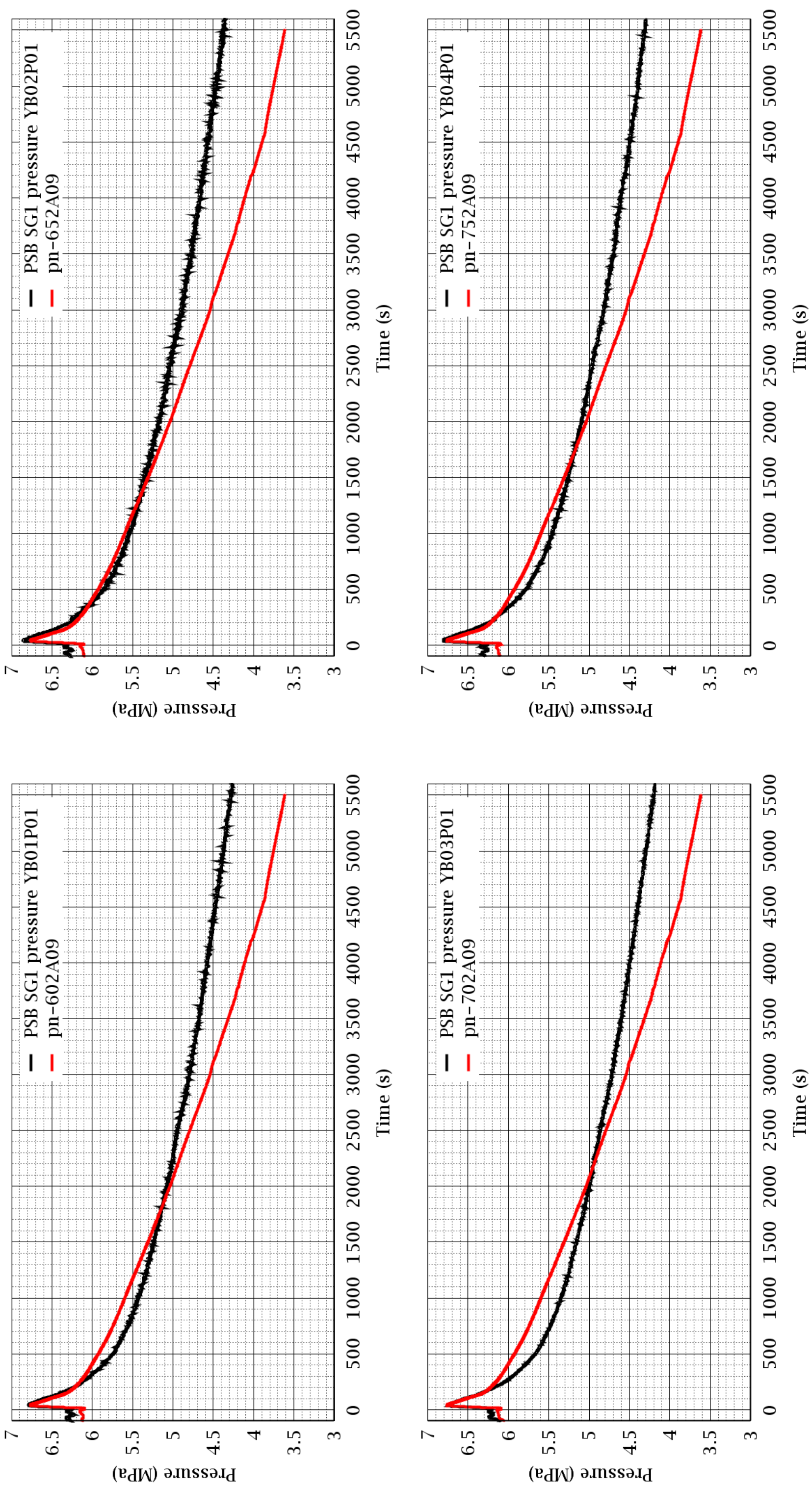


Figure C-11: Secondary Side Pressures

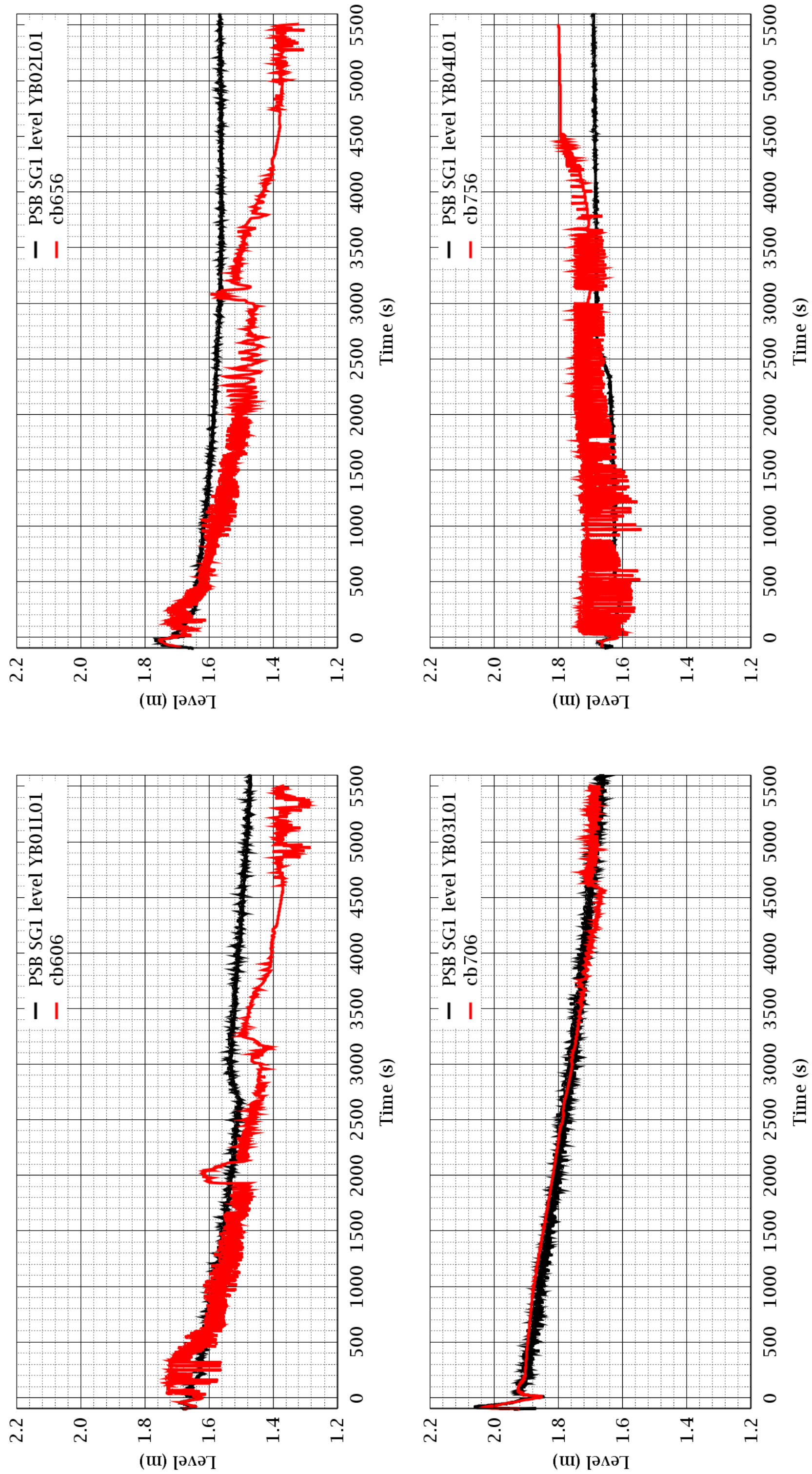


Figure C-12: Steam Generators Levels

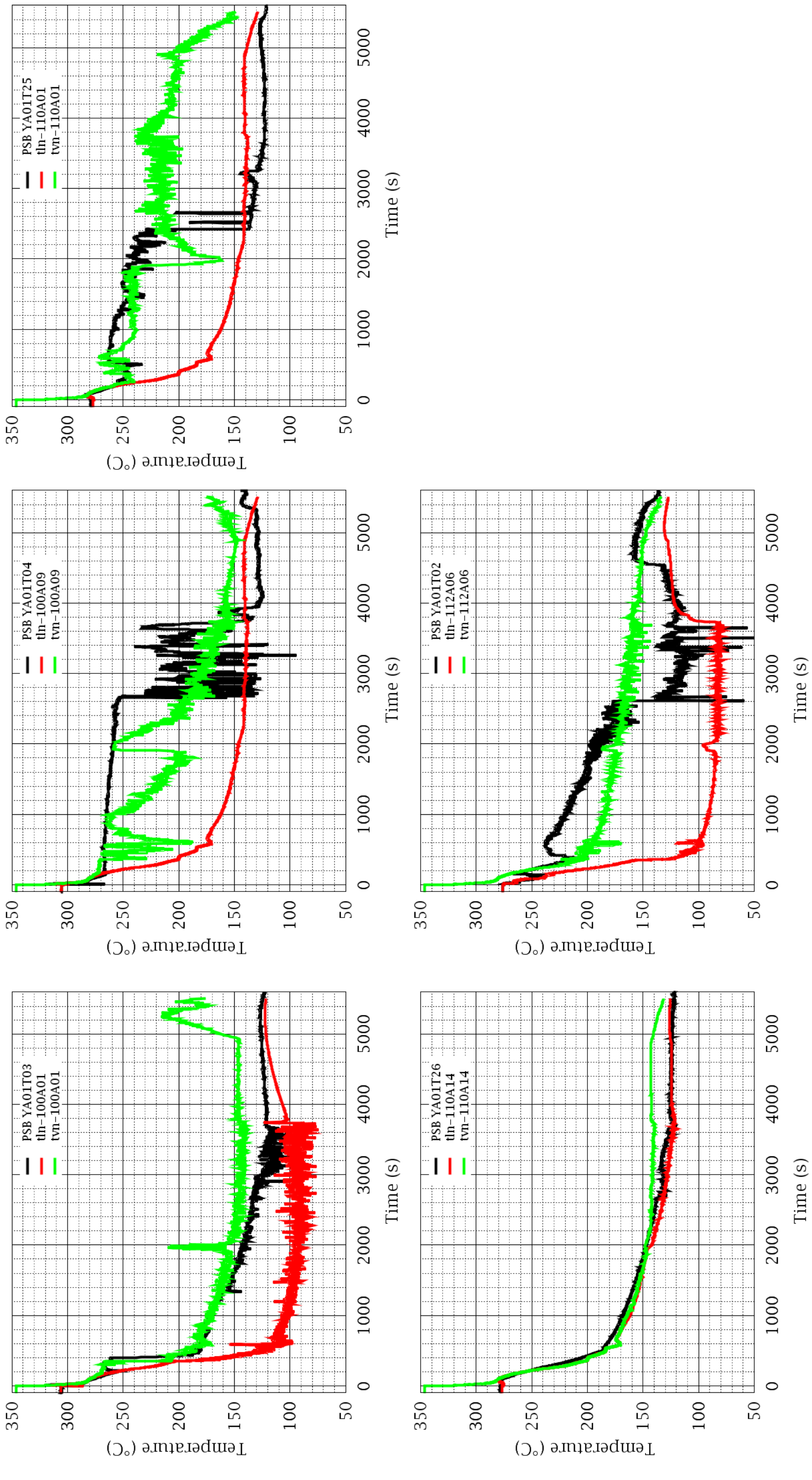


Figure C-13: Loop 1 Temperatures

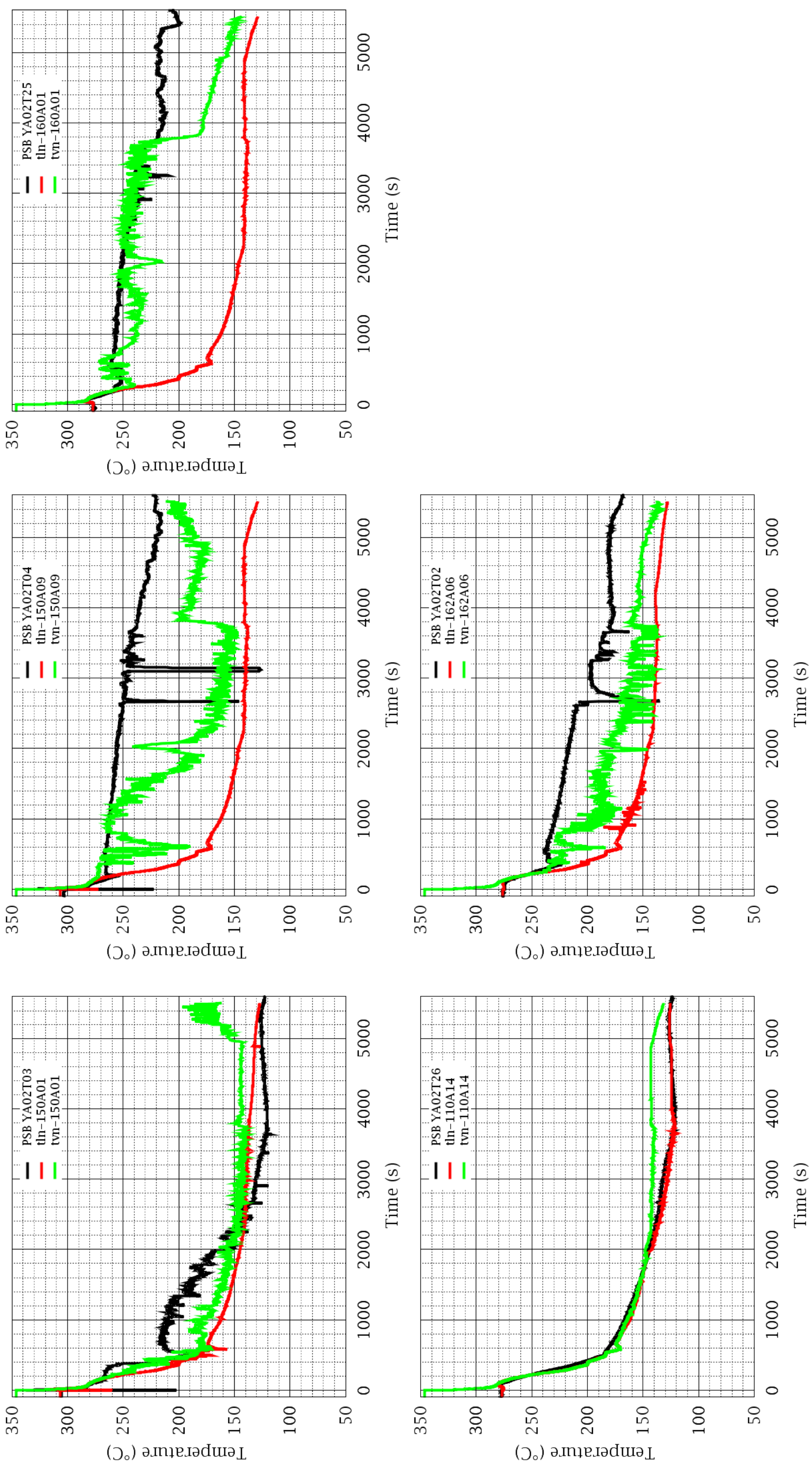


Figure C-14: Loop 2 Temperatures

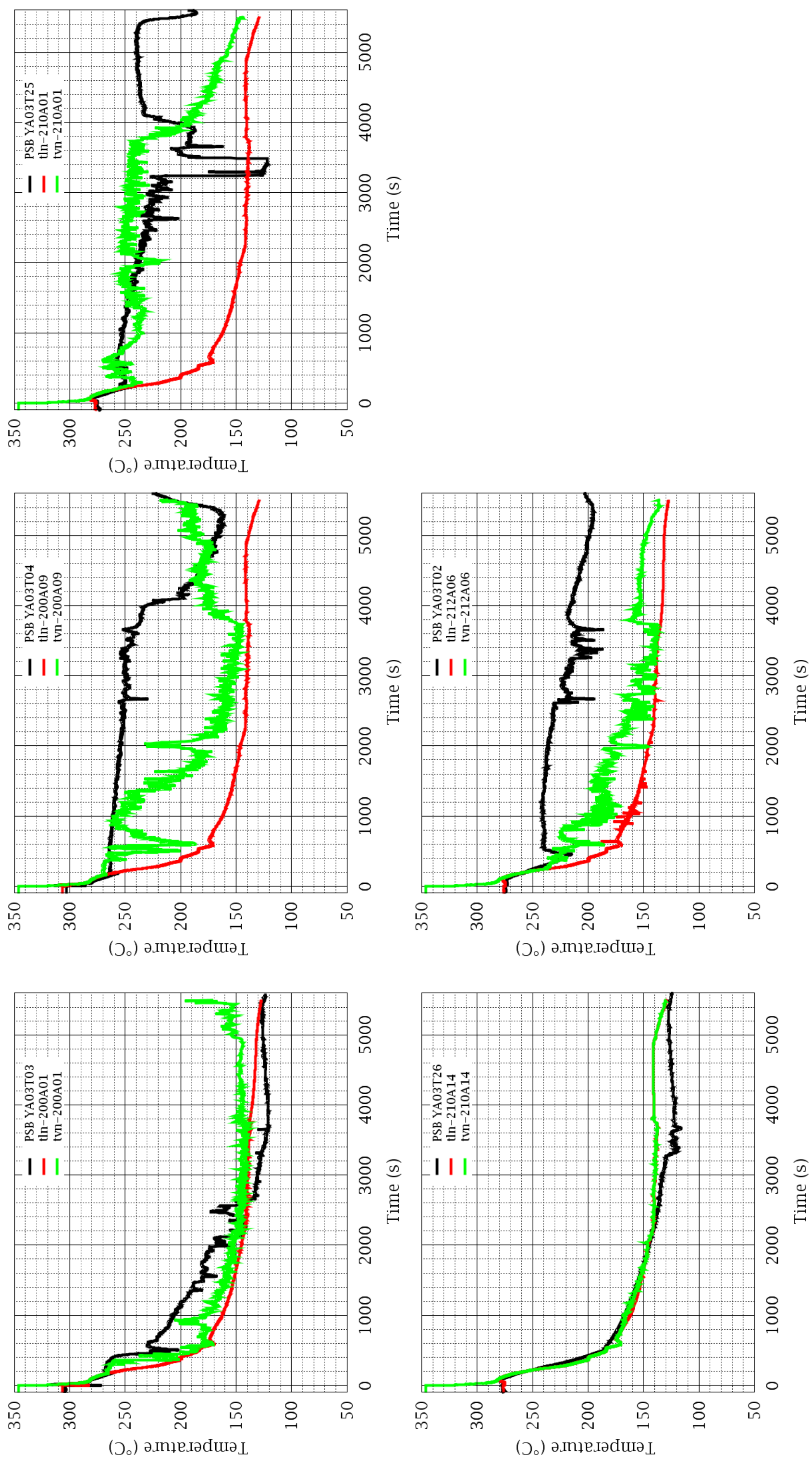


Figure C-15: Loop 3 Temperatures

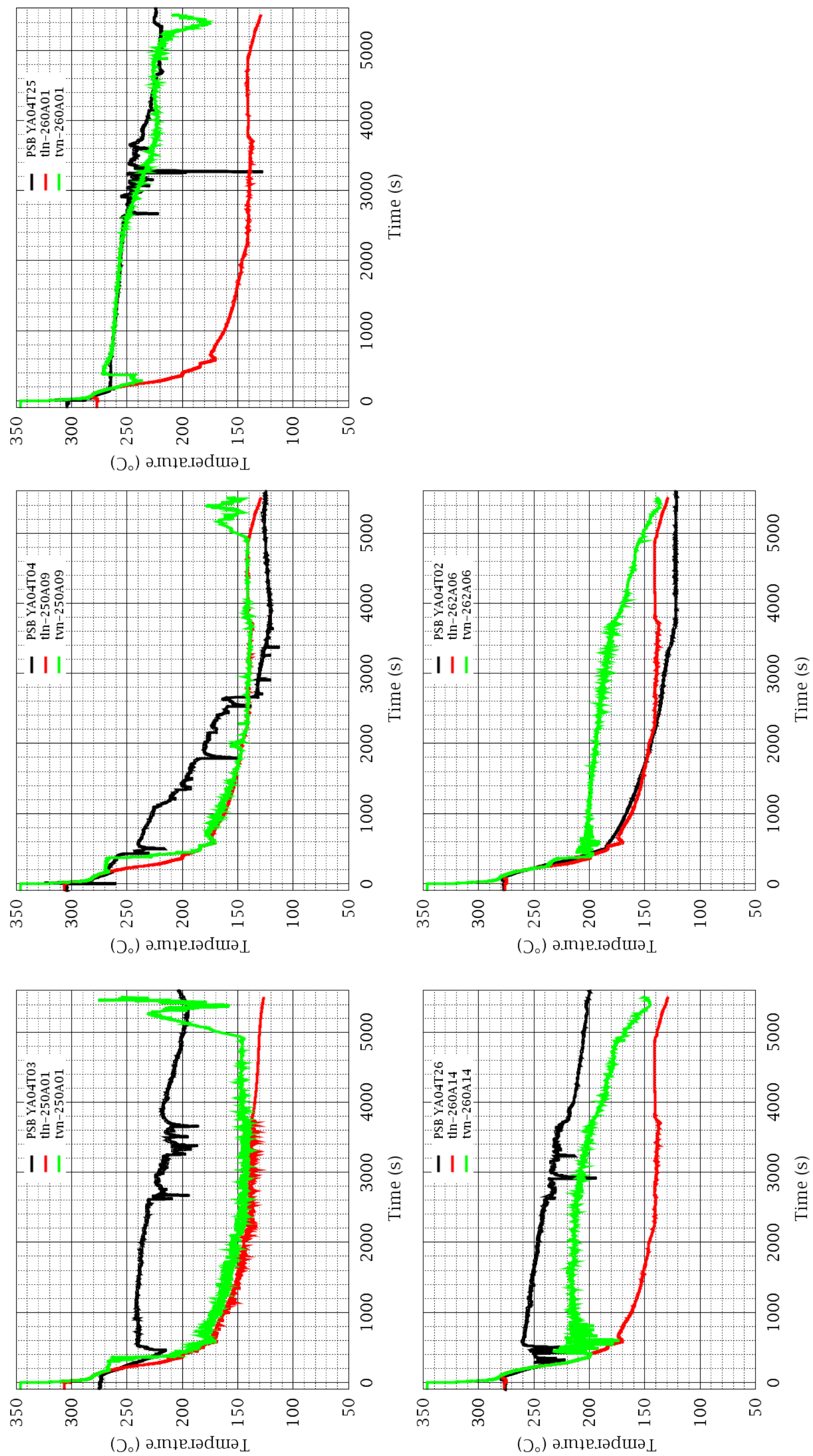


Figure C-16: Loop 4 Temperatures

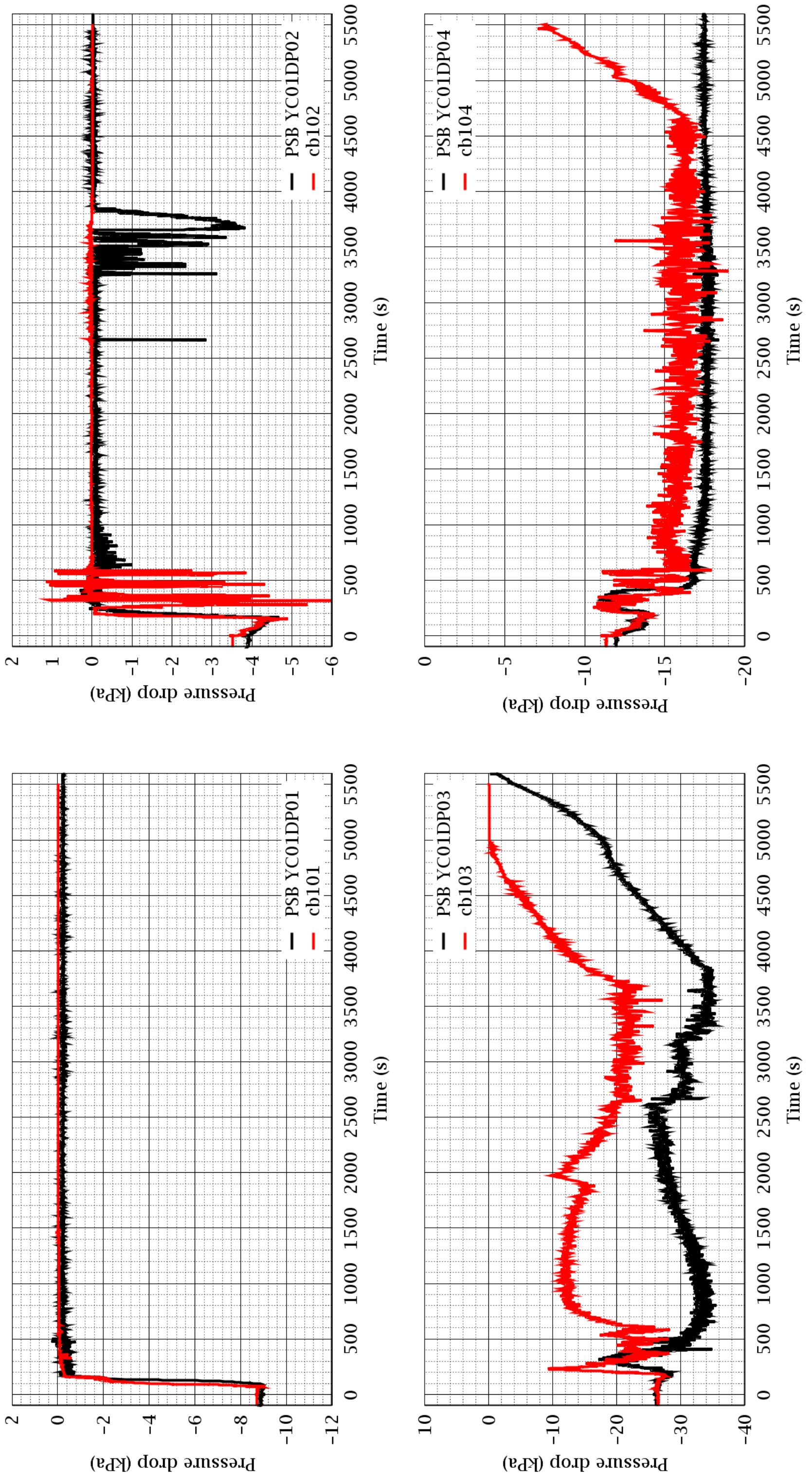


Figure C-17: Pressure Differences DP01-DP04 (DC)

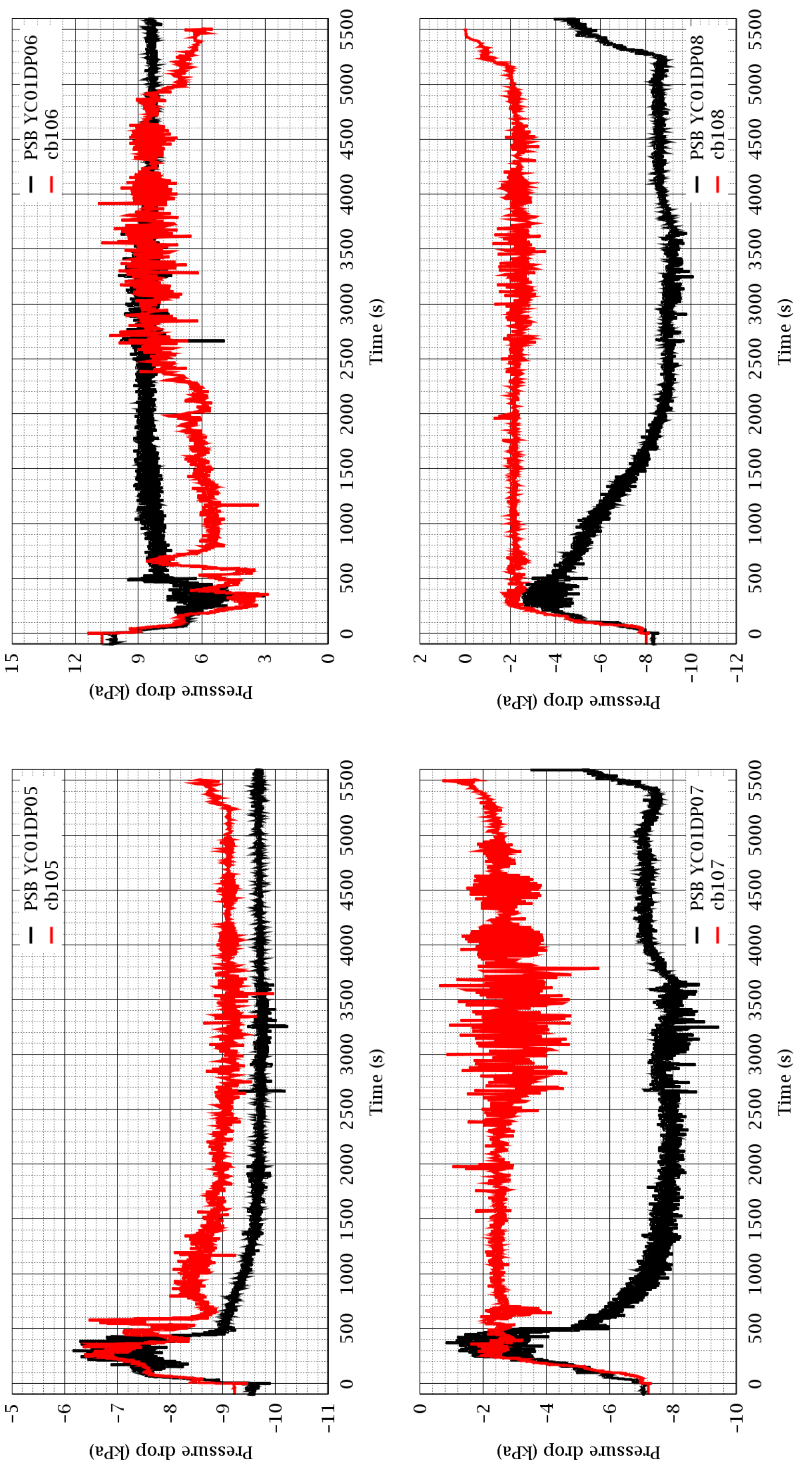


Figure C-18: Pressure Differences DP05-DP08 (Lower Plenum + Lower Part of FRS)

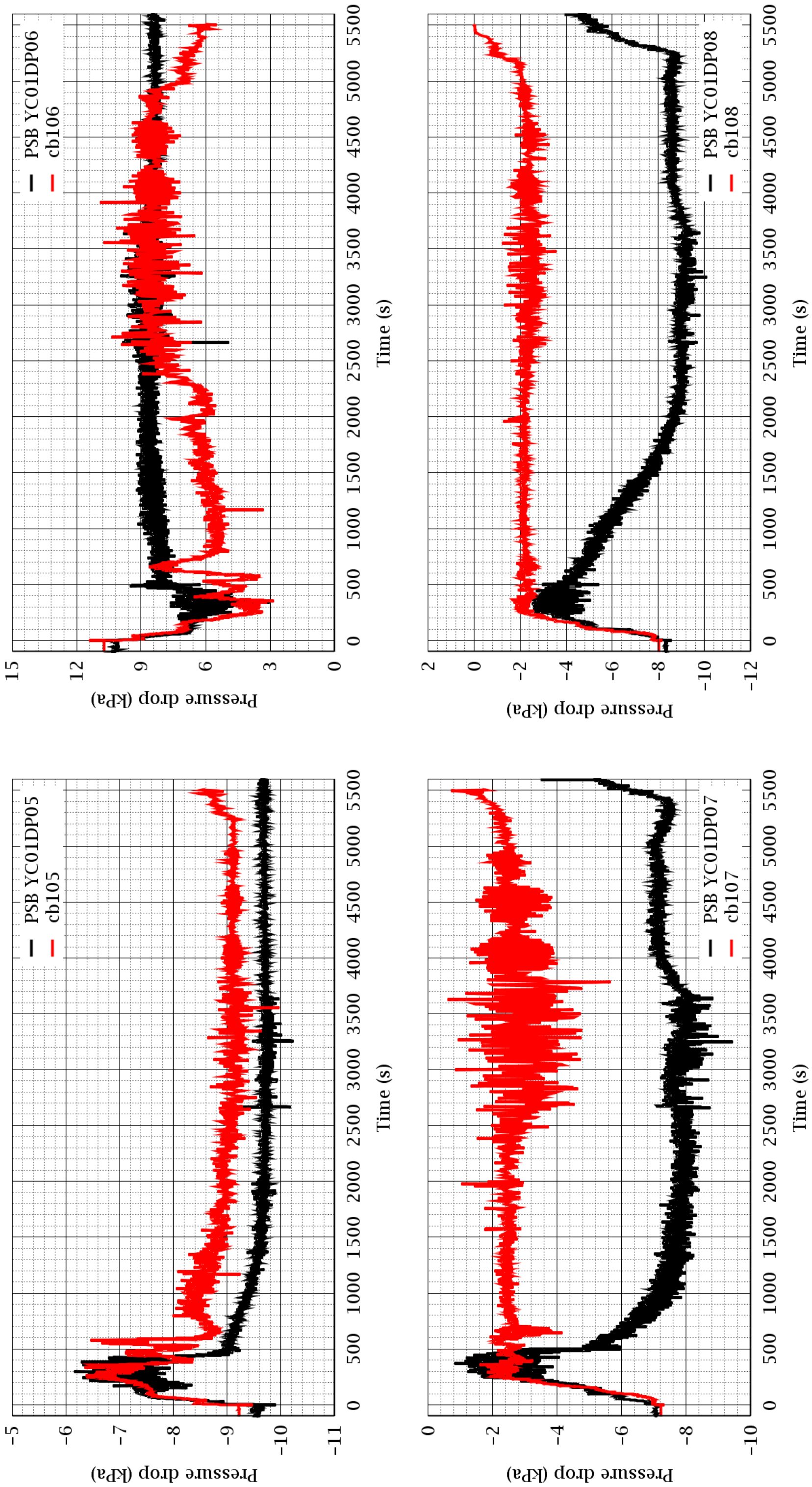


Figure C-19: Pressure Differences DP09-DP12 (FRS + Lower Part of Upper Plenum)

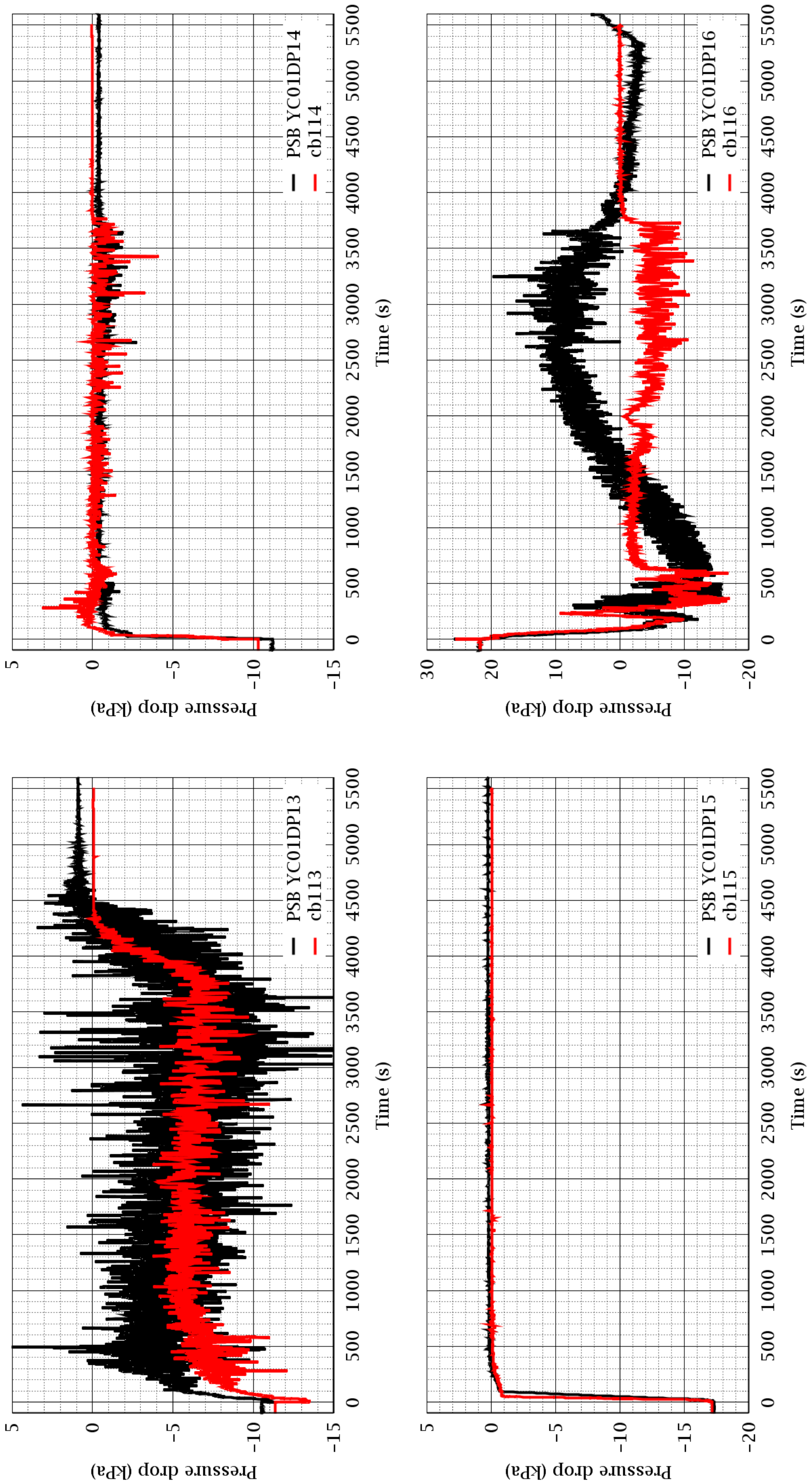


Figure C-20: Pressure Differences DP13-DP16 (Upper Part of Upper Plenum)

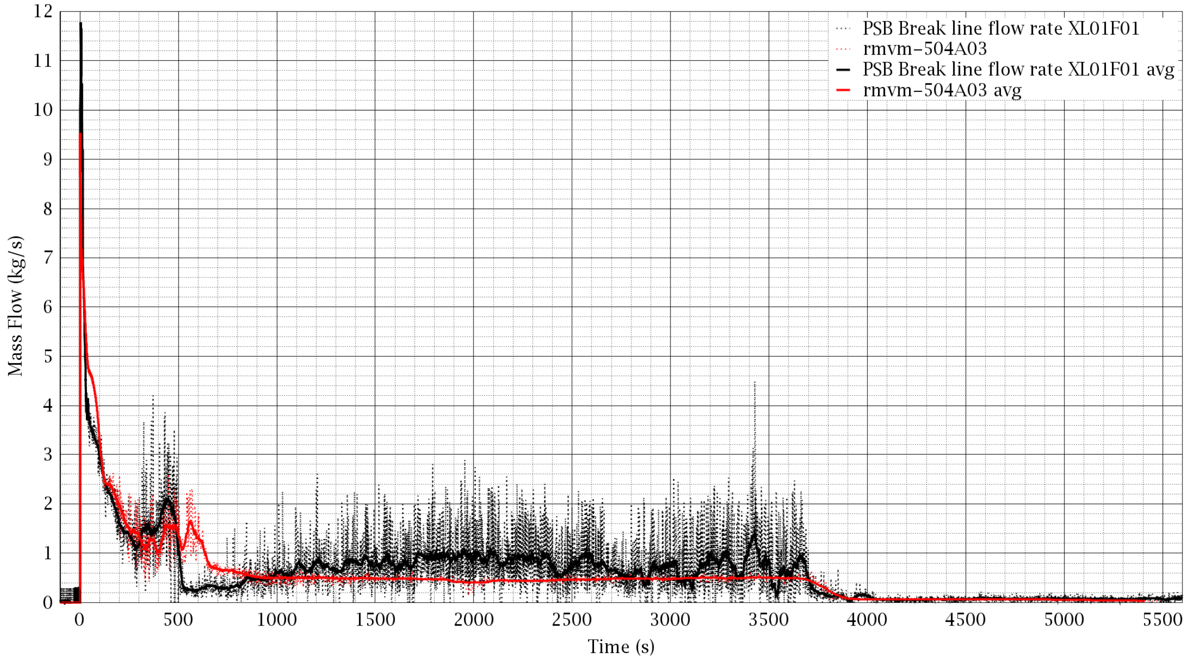


Figure C-21: Break Flow

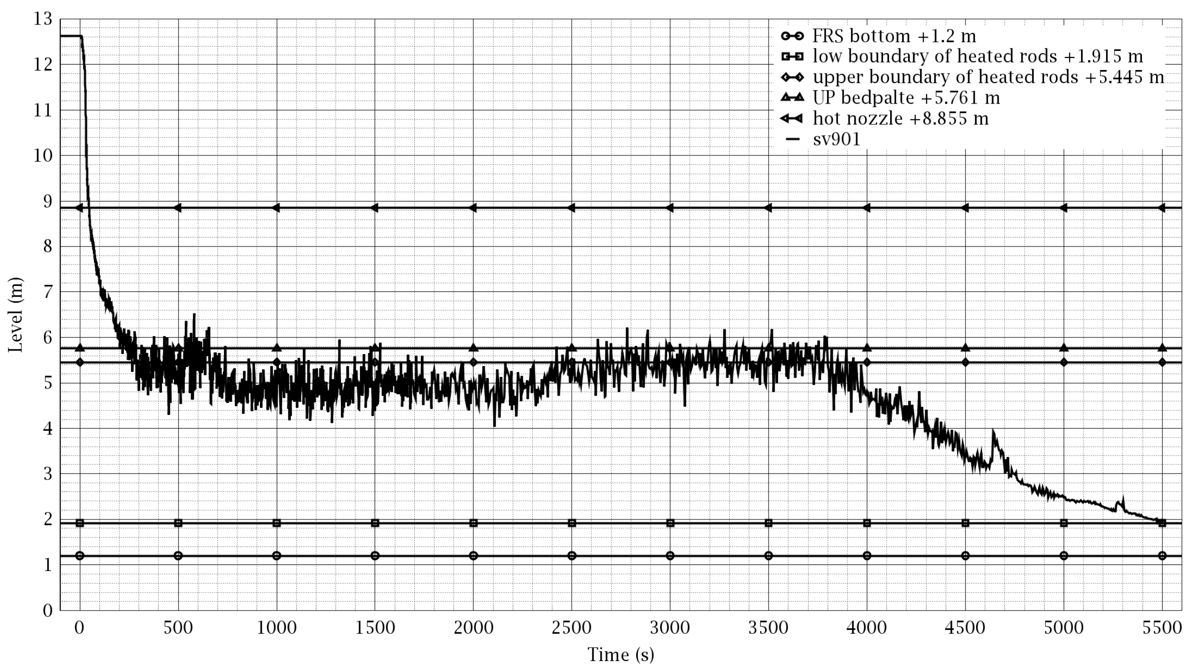


Figure C-22: Reactor Collapsed Level

APPENDIX D COMPLETE SET OF COMPARISON PLOTS FOR RELAP5 CALCULATION

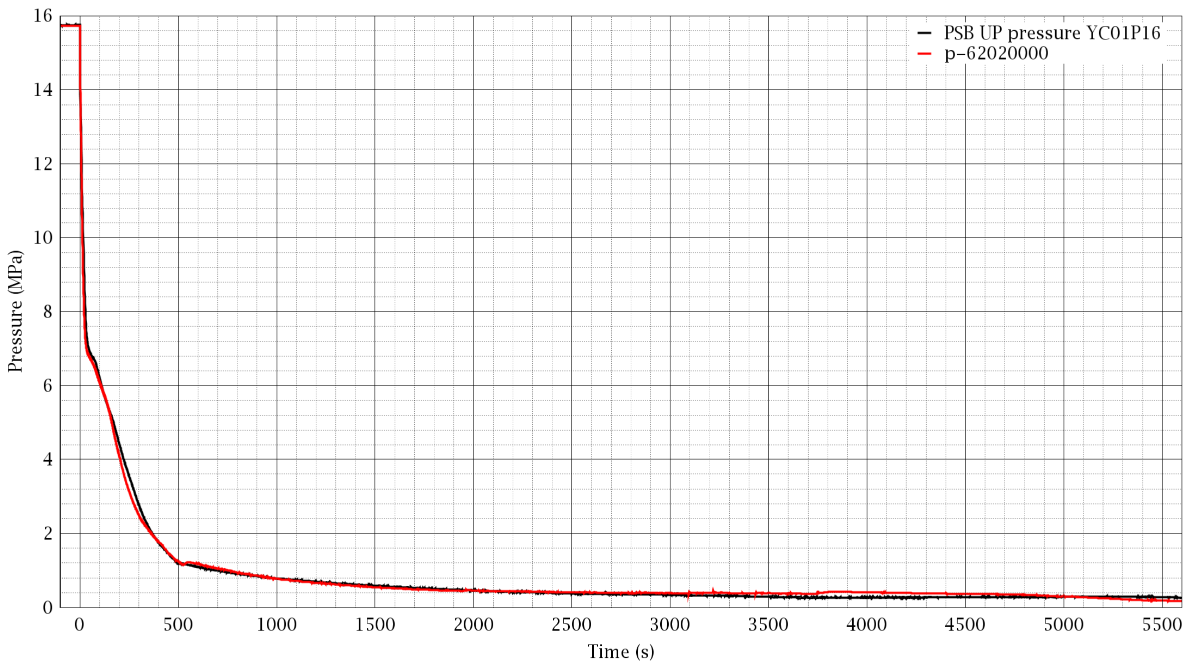


Figure D-1: Primary Pressure

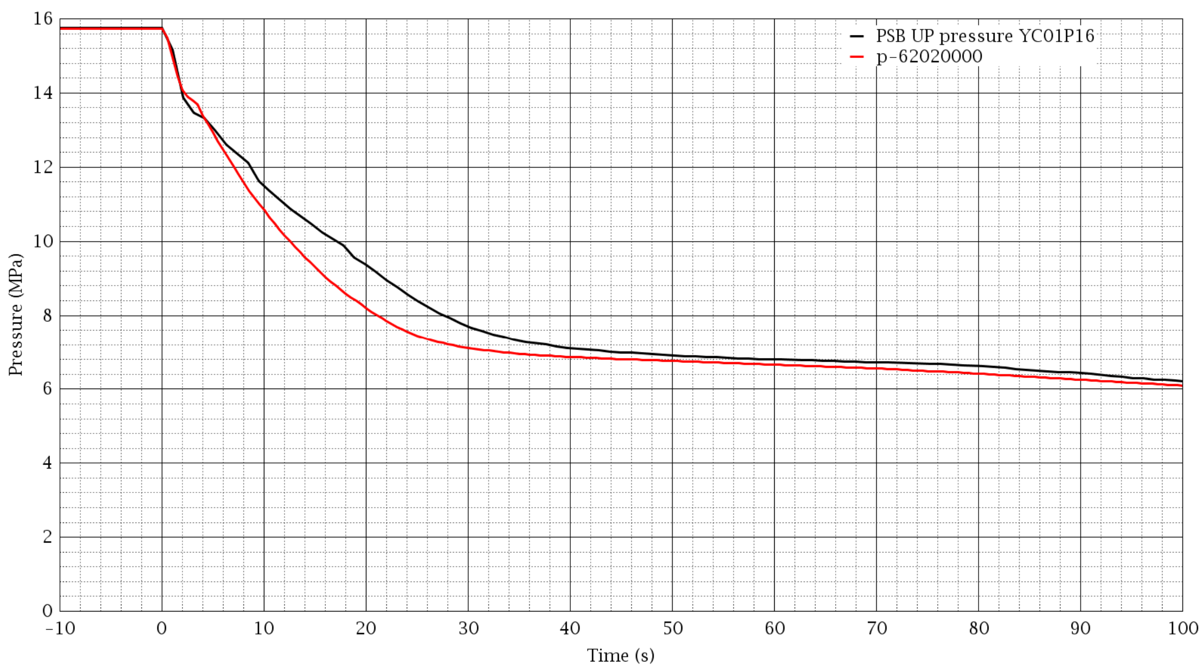


Figure D-2: Primary Pressure (detail)

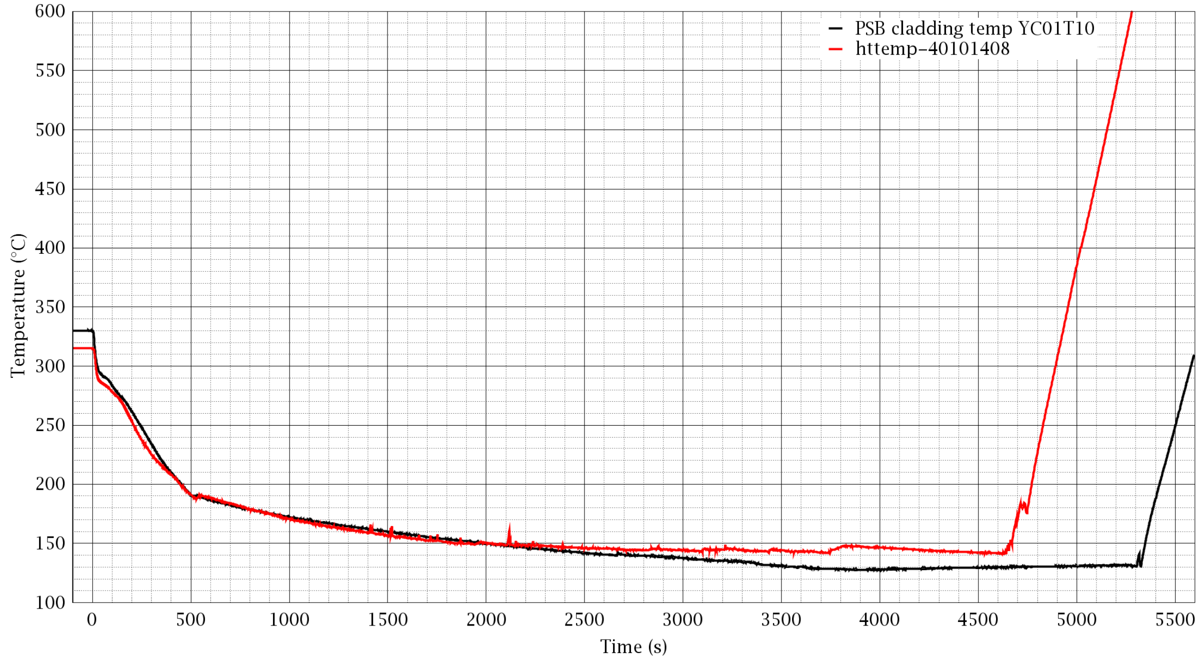


Figure D-3: Fuel Cladding Temperature (Top of the Core)

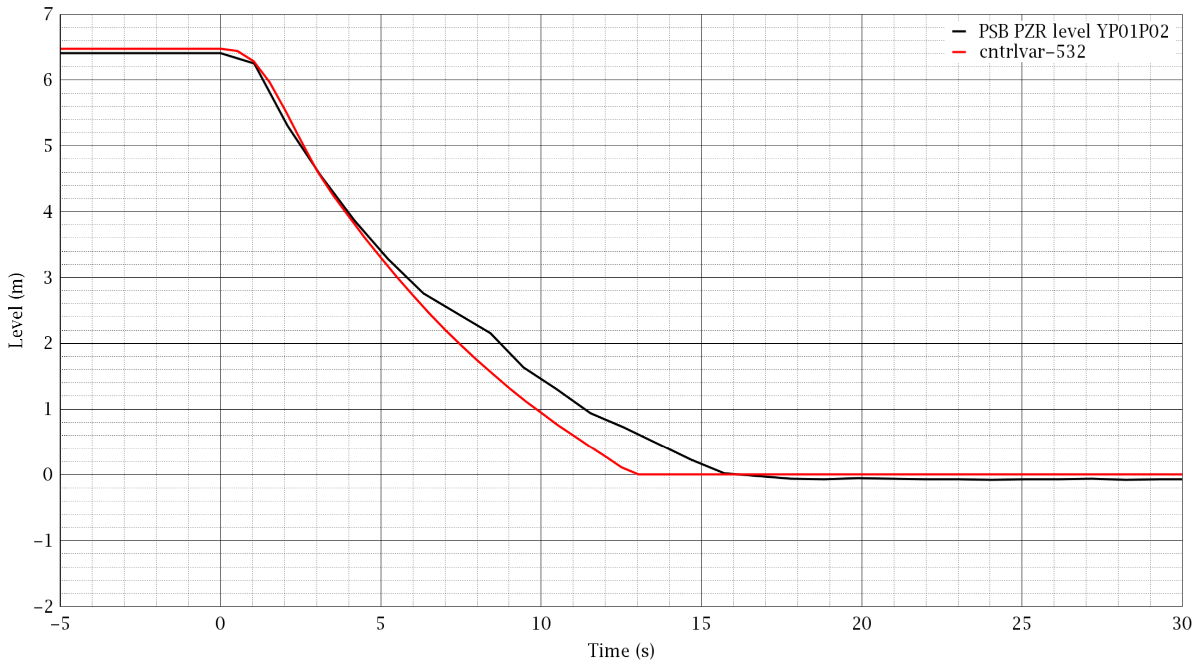


Figure D-4: Presurizer Level

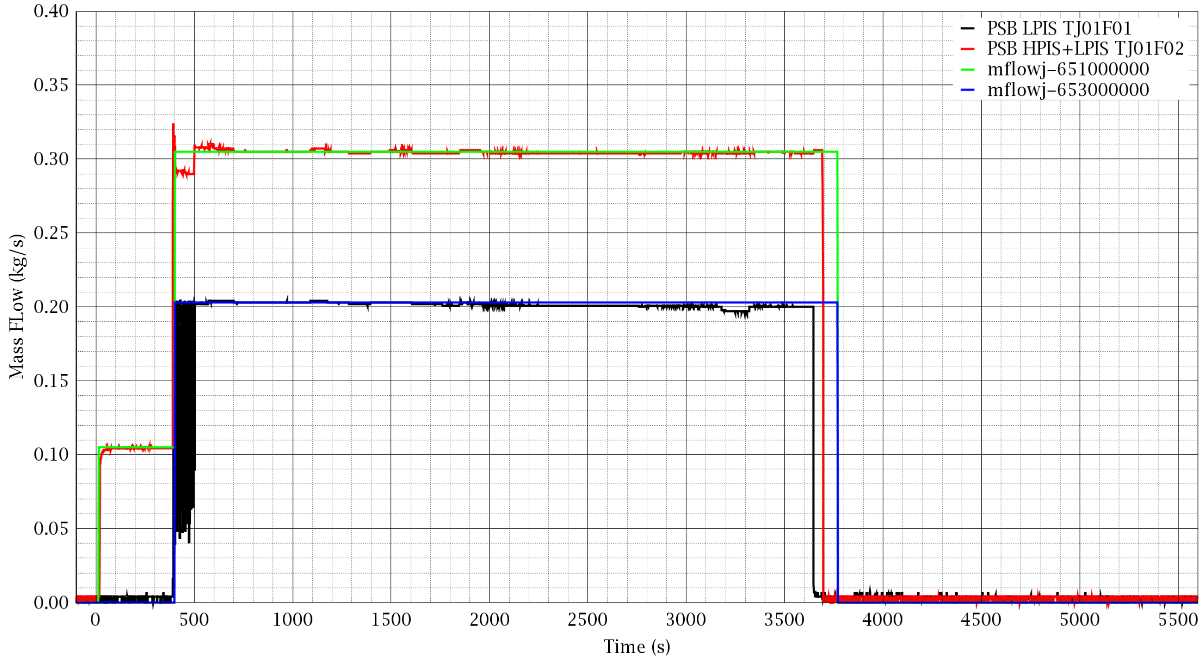


Figure D-5: HPIS + LPIS Flow (Boundary Condition)

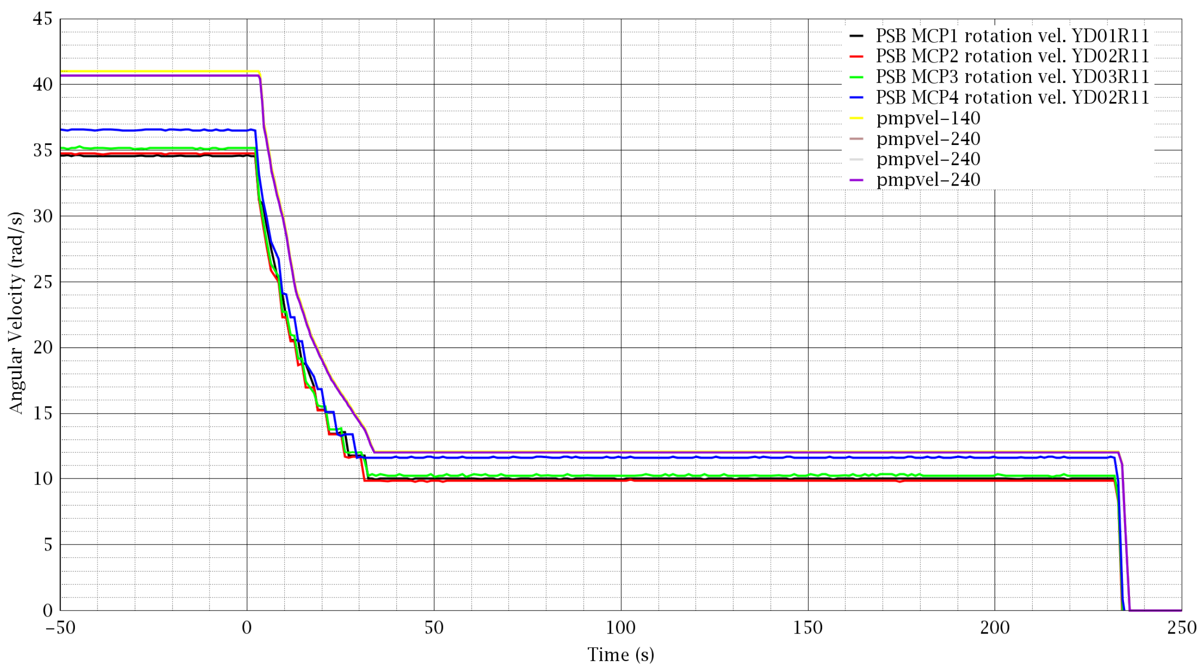


Figure D-6: MCP Rotor Speed (Boundary Condition)

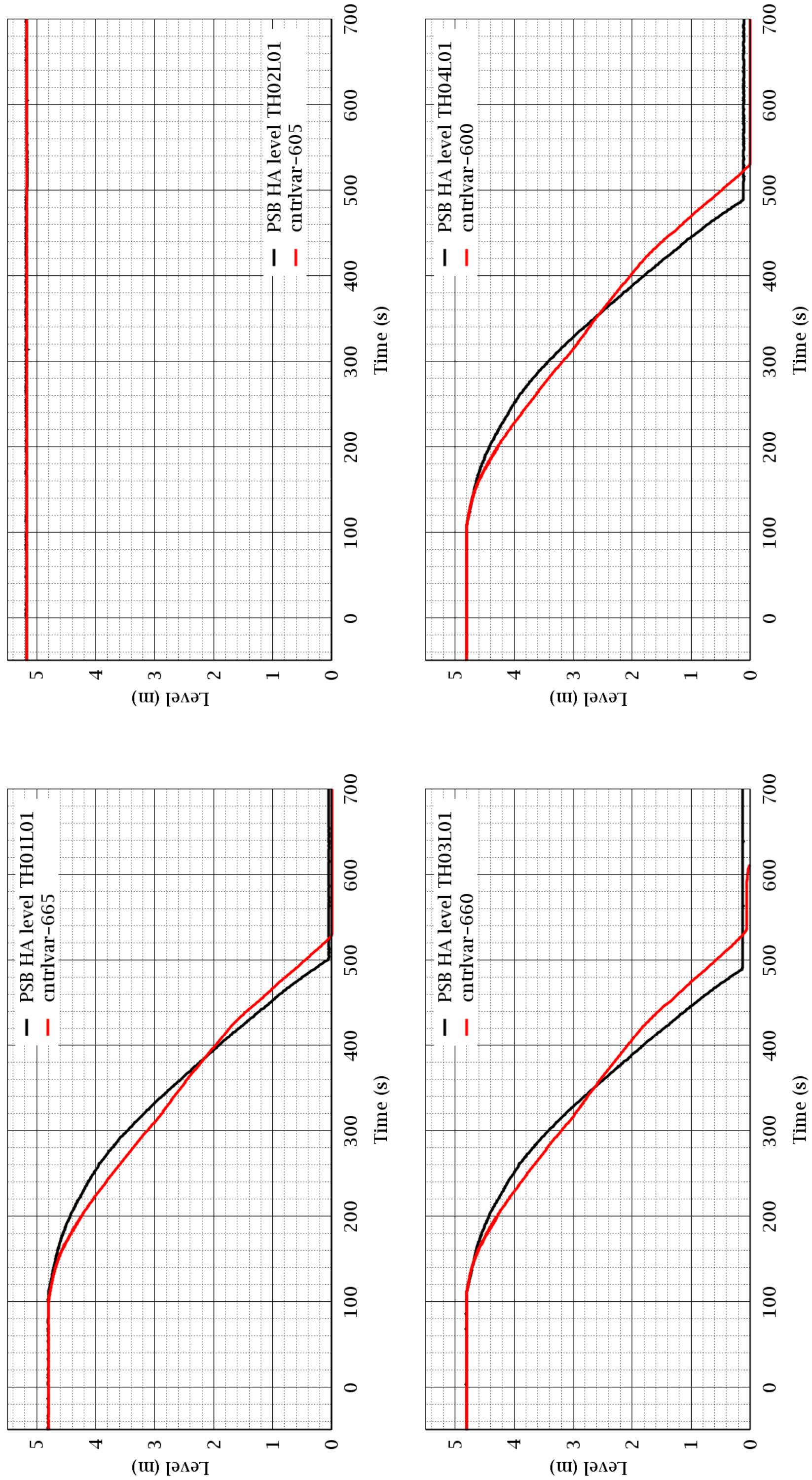


Figure D-7: Accumulators Levels

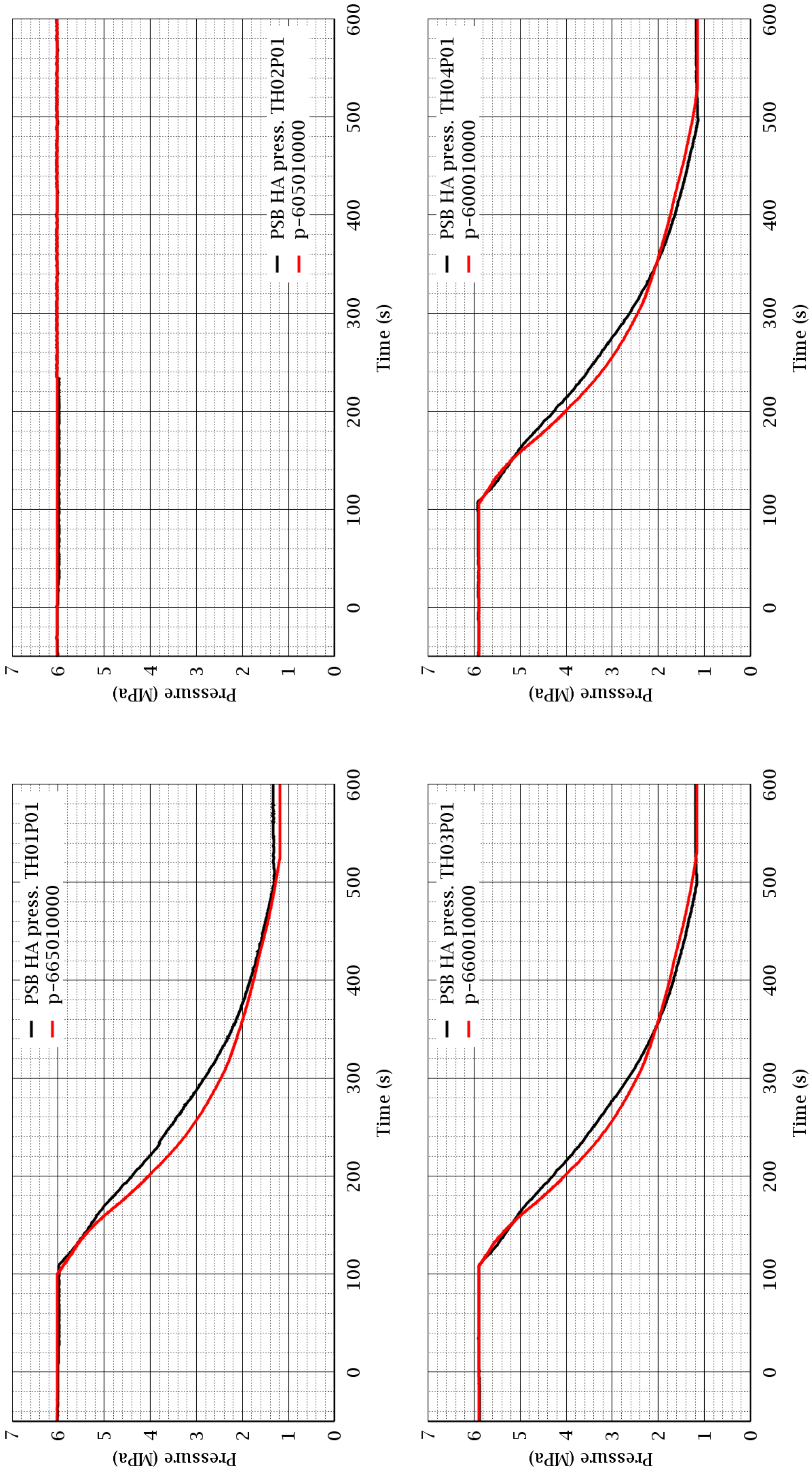


Figure D-8: Accumulators Pressure

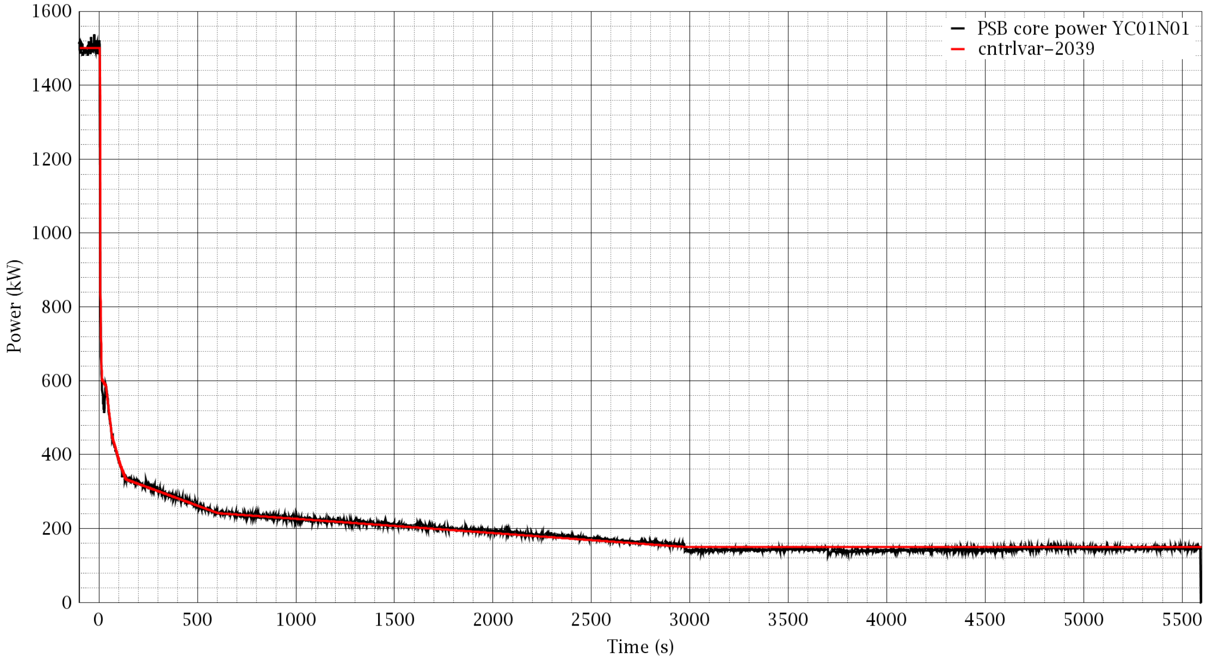


Figure D-9: Fuel Rod Simulator Power (Boundary Condition)

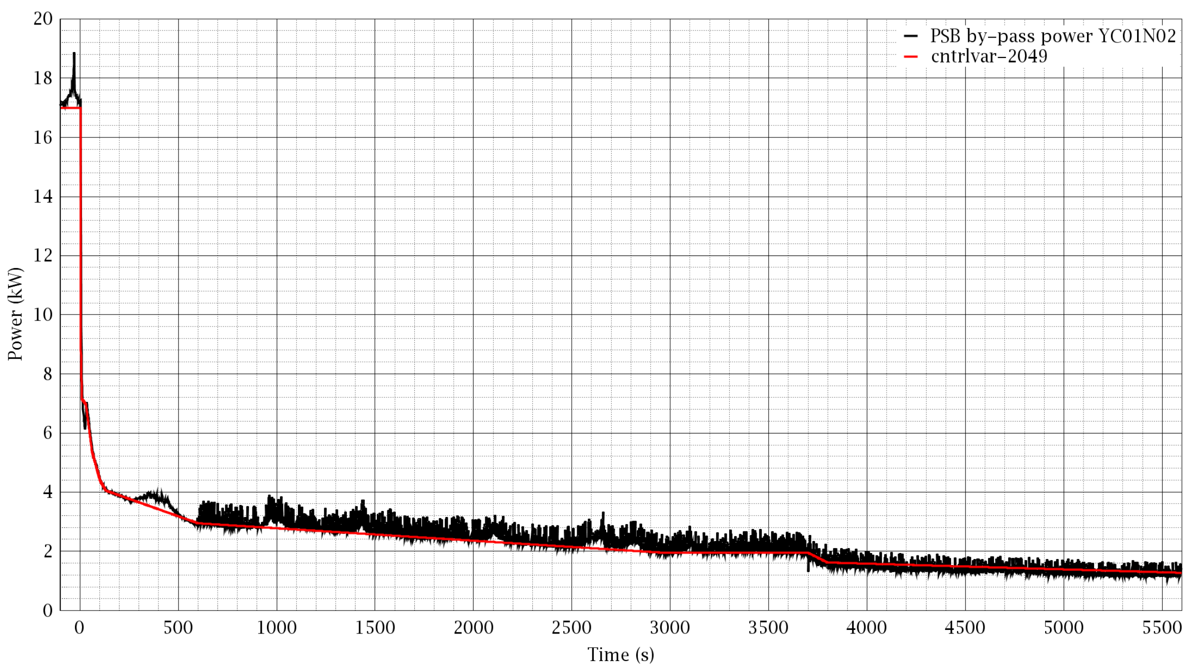


Figure D-10: Core By-pass Power (Boundary Condition)

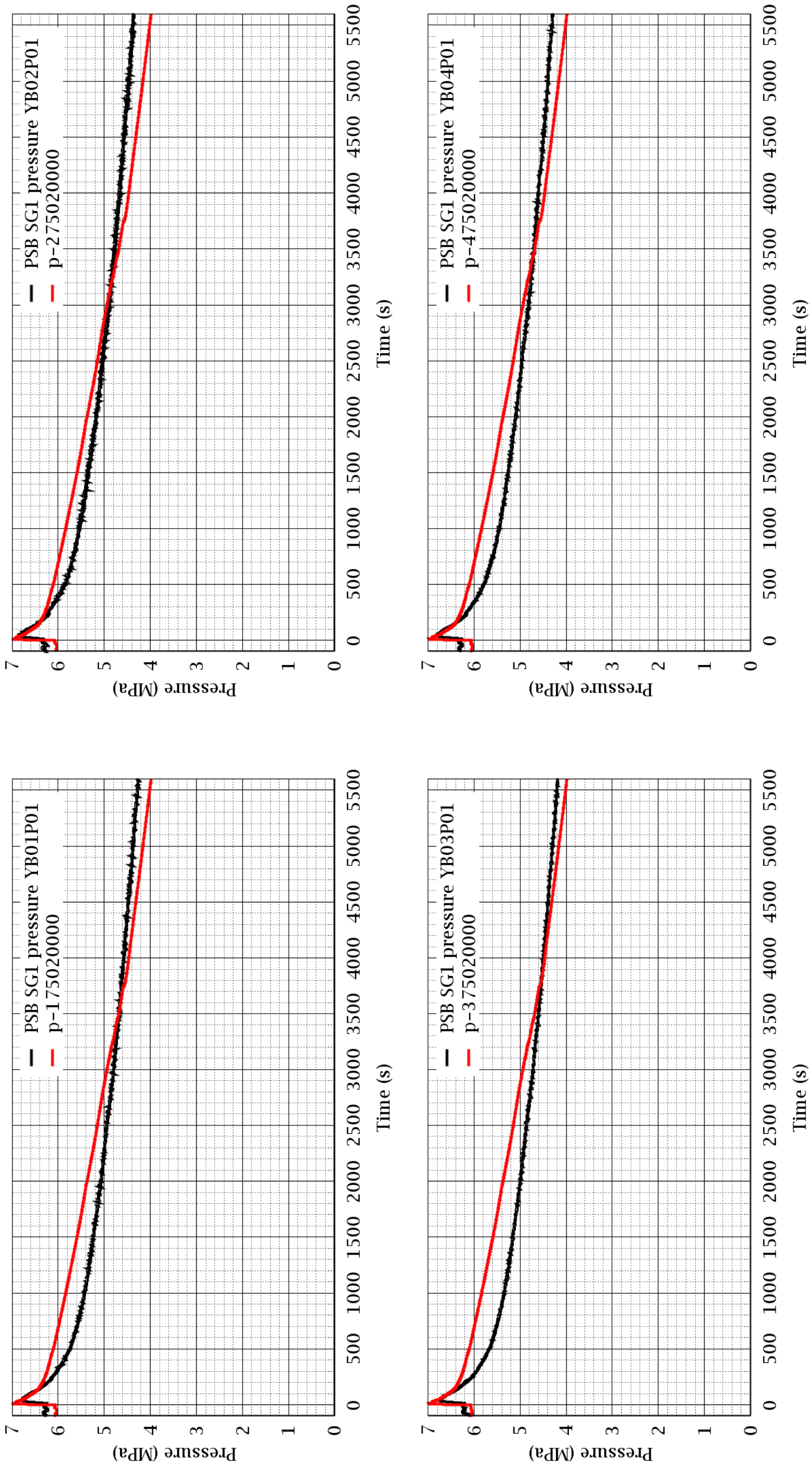


Figure D-11: Secondary Side Pressures

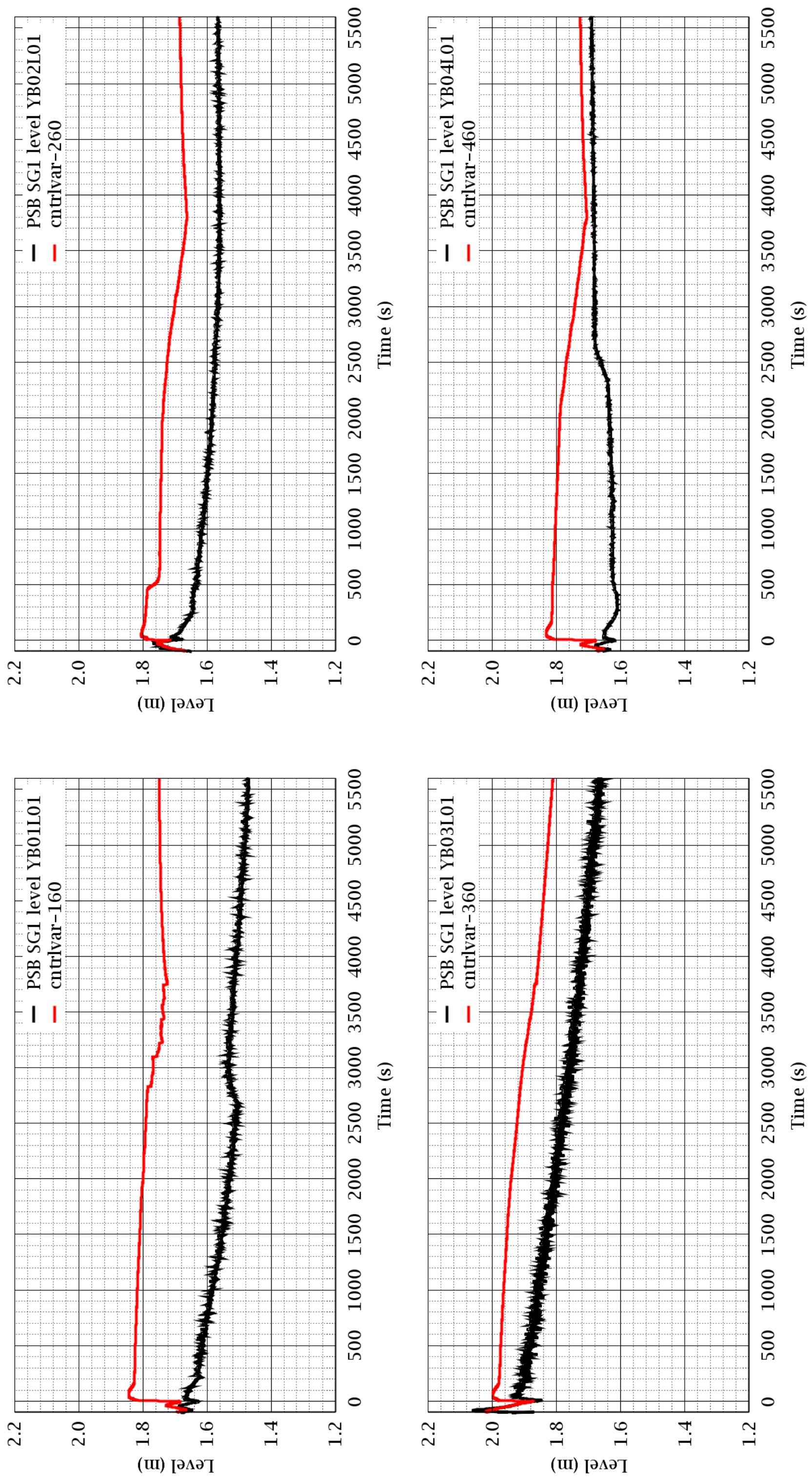


Figure D-12: Steam Generators Levels

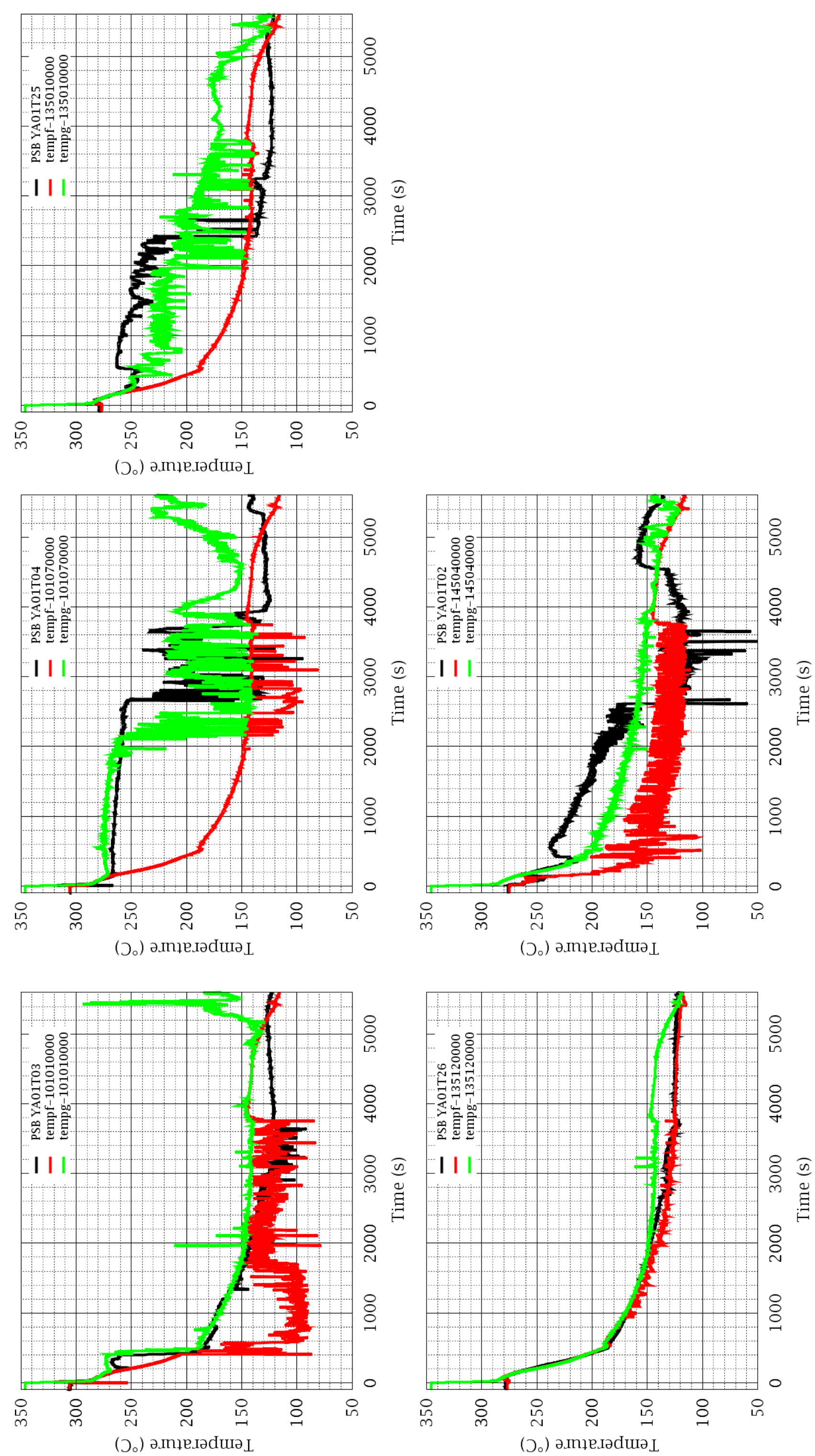


Figure D-13: Loop 1 Temperatures

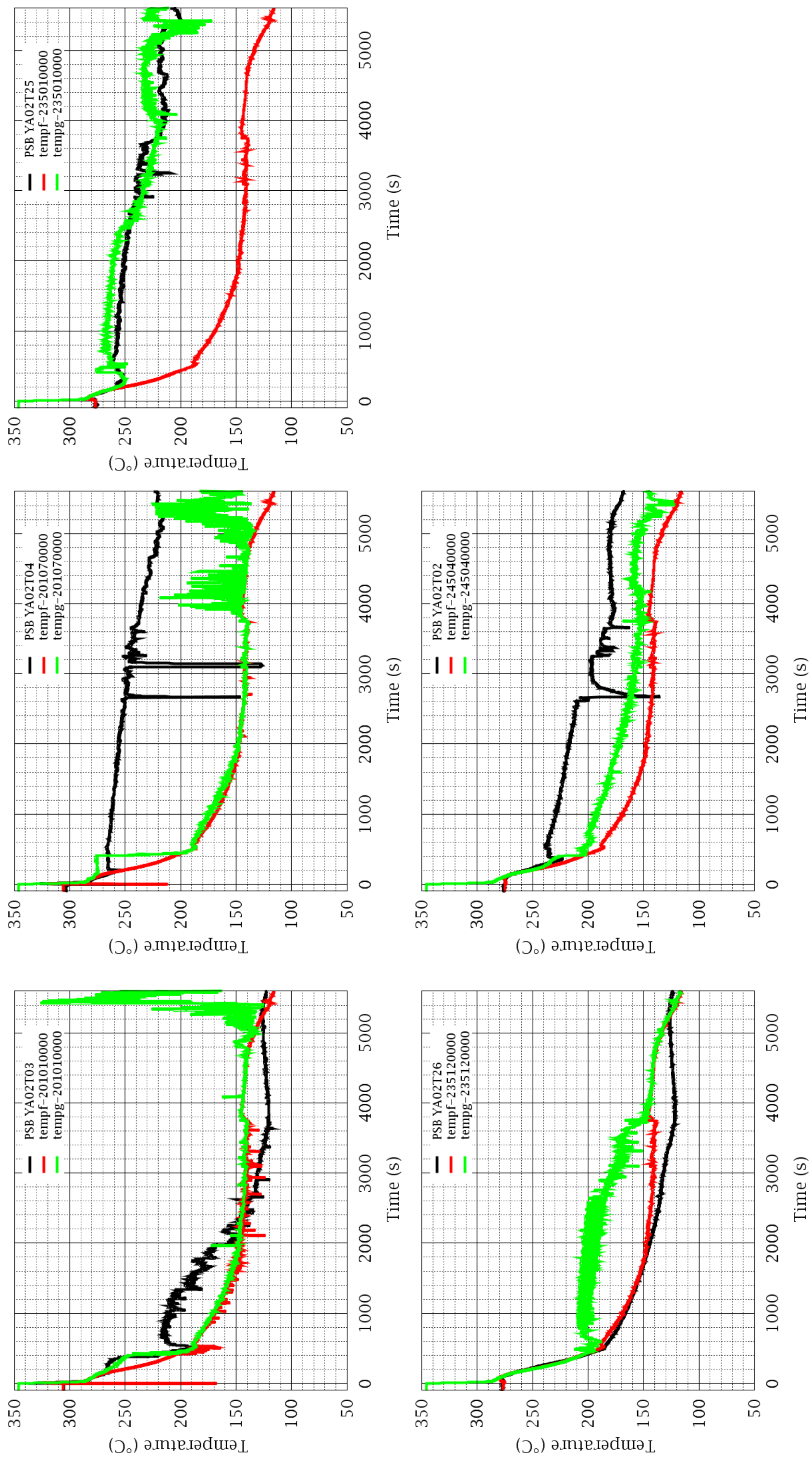


Figure D-14: Loop 2 Temperatures

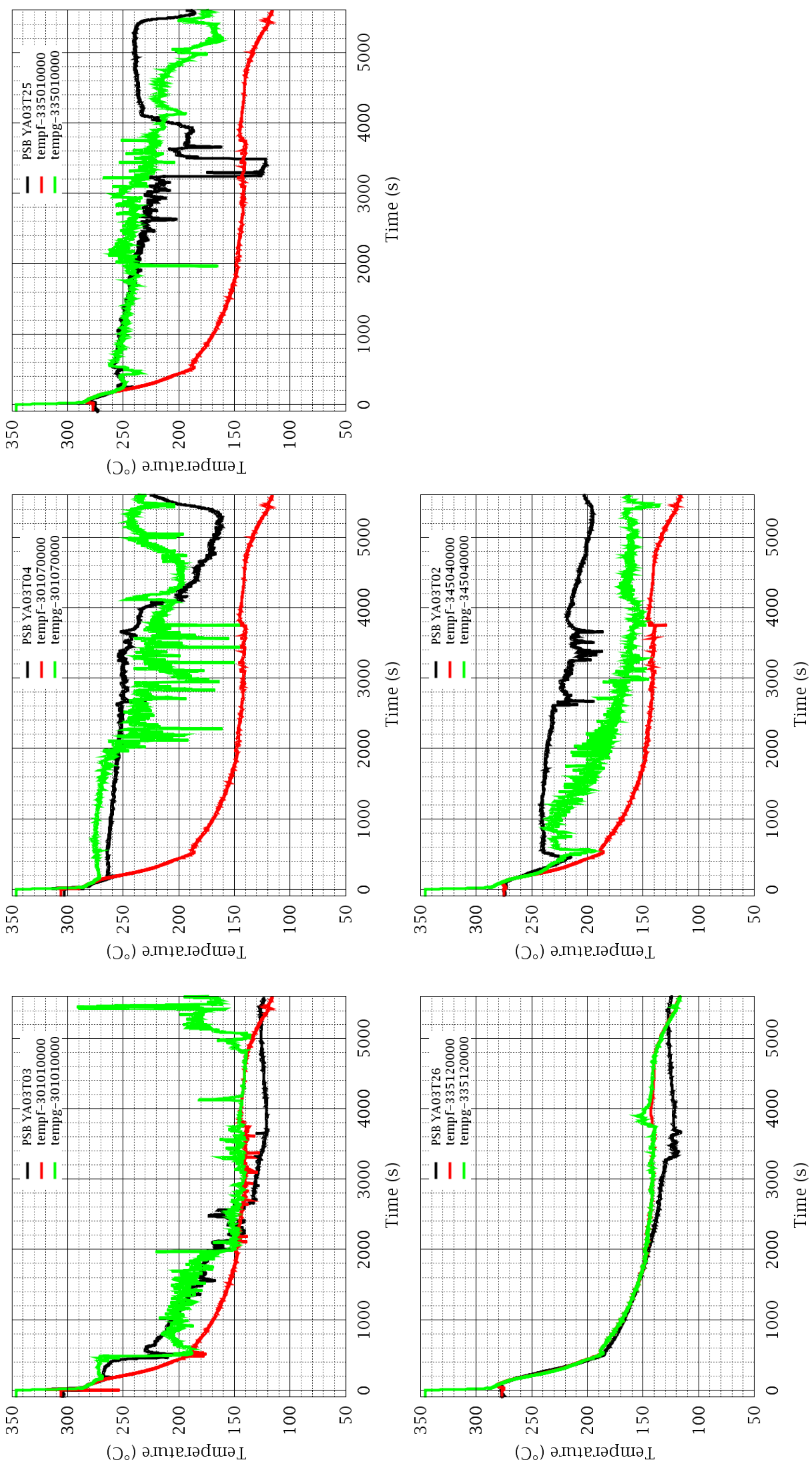


Figure D-15: Loop 3 Temperatures

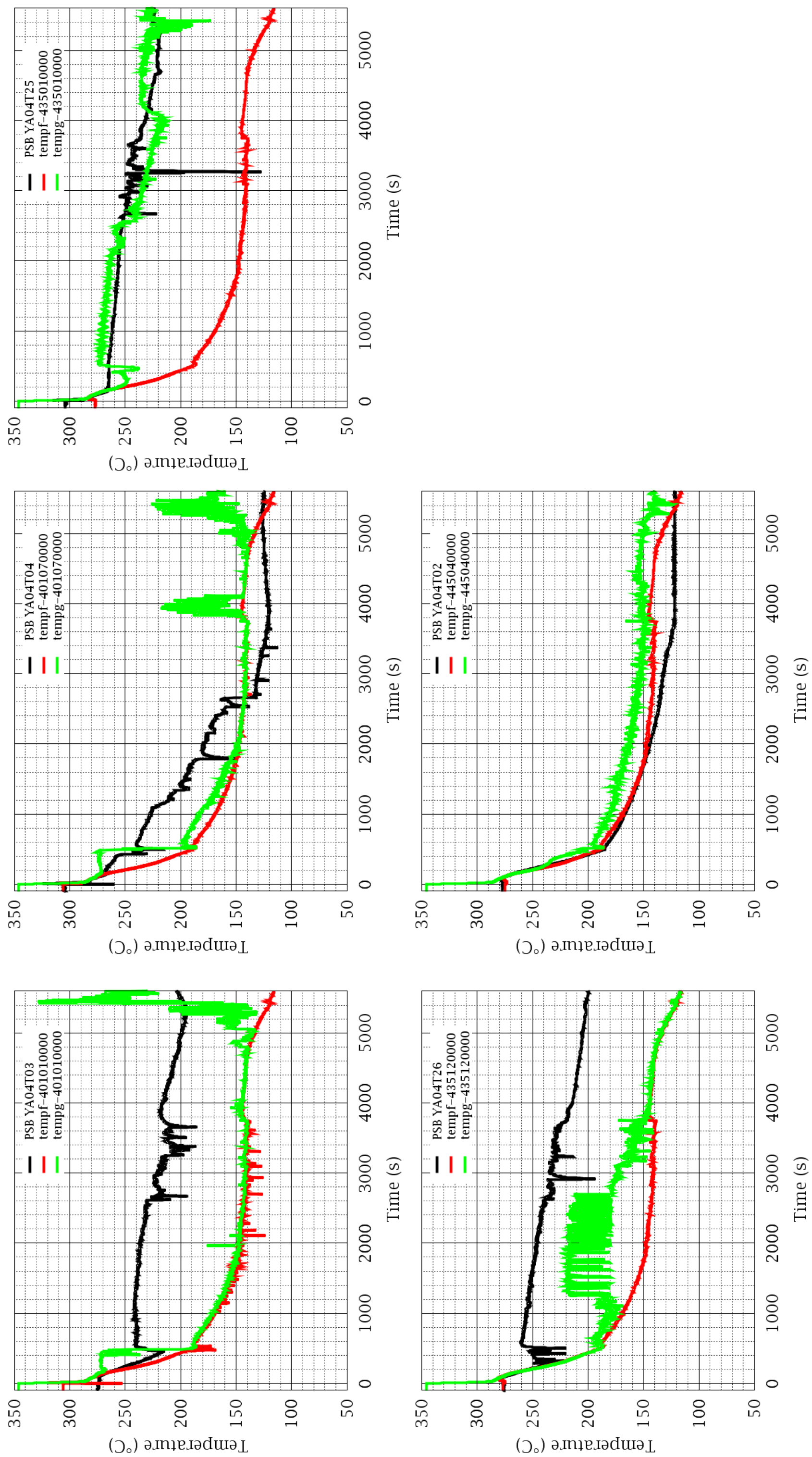


Figure D-16: Loop 4 Temperatures

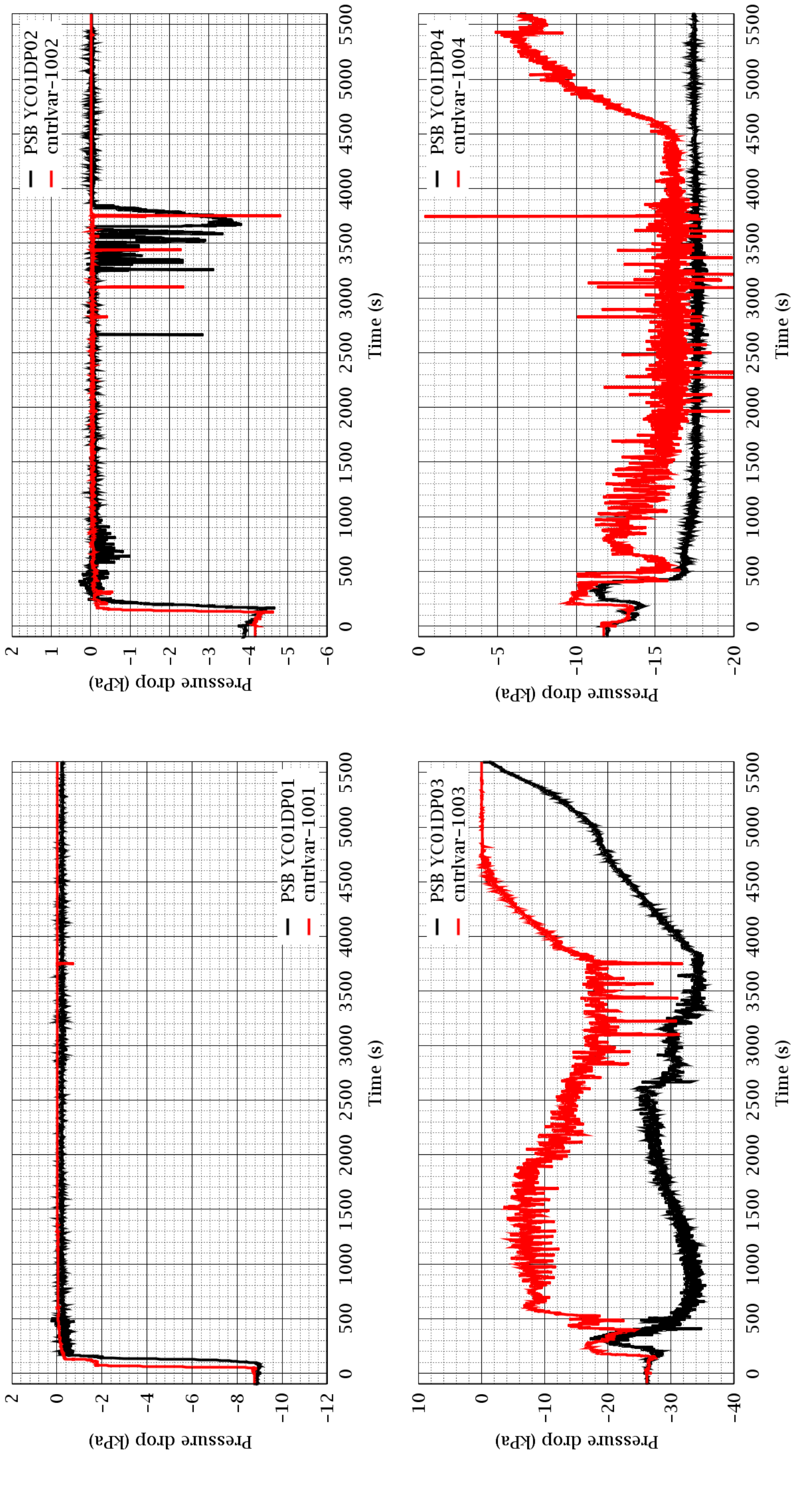


Figure D-17: Pressure Differences DP01-DP04 (DC)

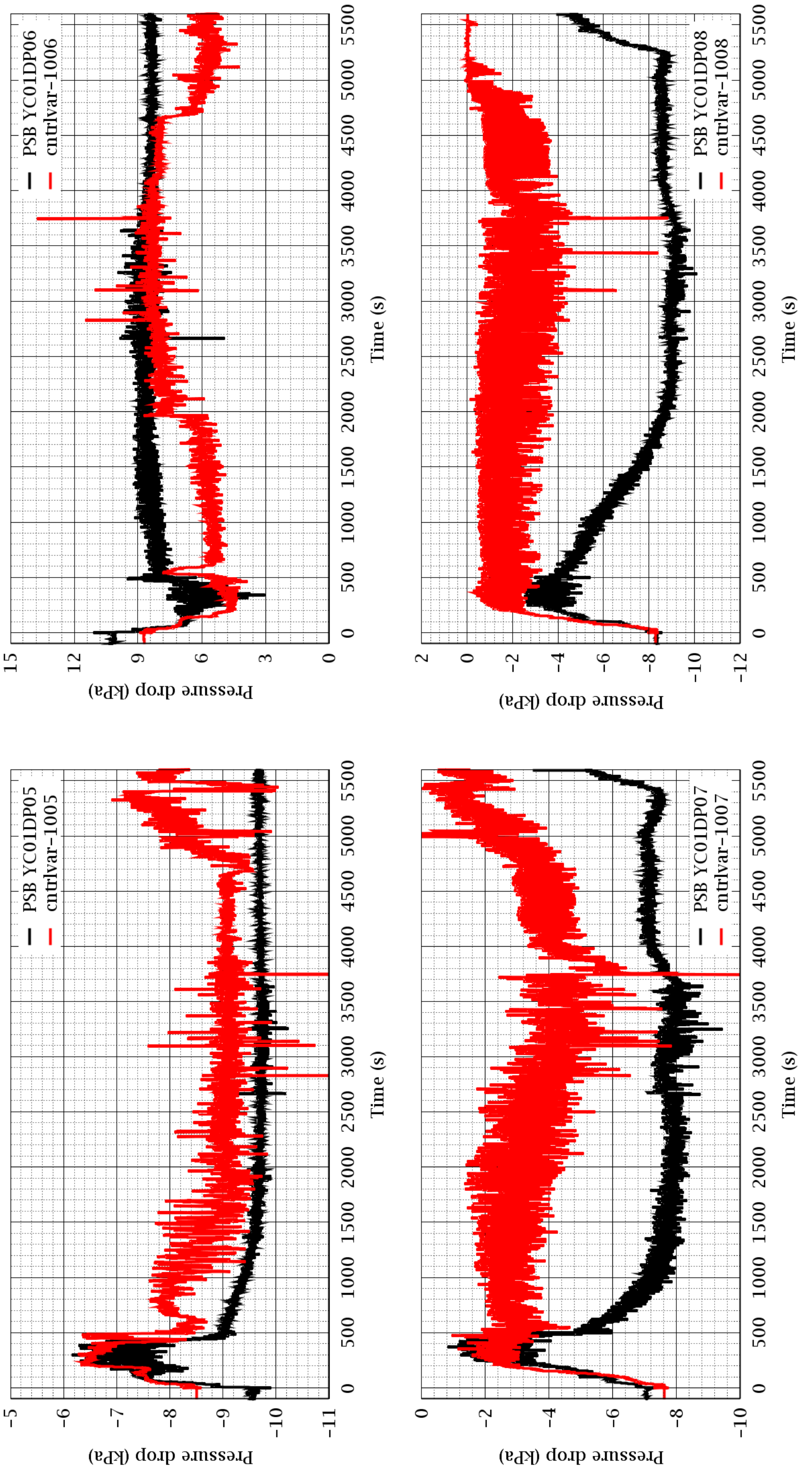


Figure D-18: Pressure Differences DP05-DP08 (Lower Plenum + Lower Part of FRS)

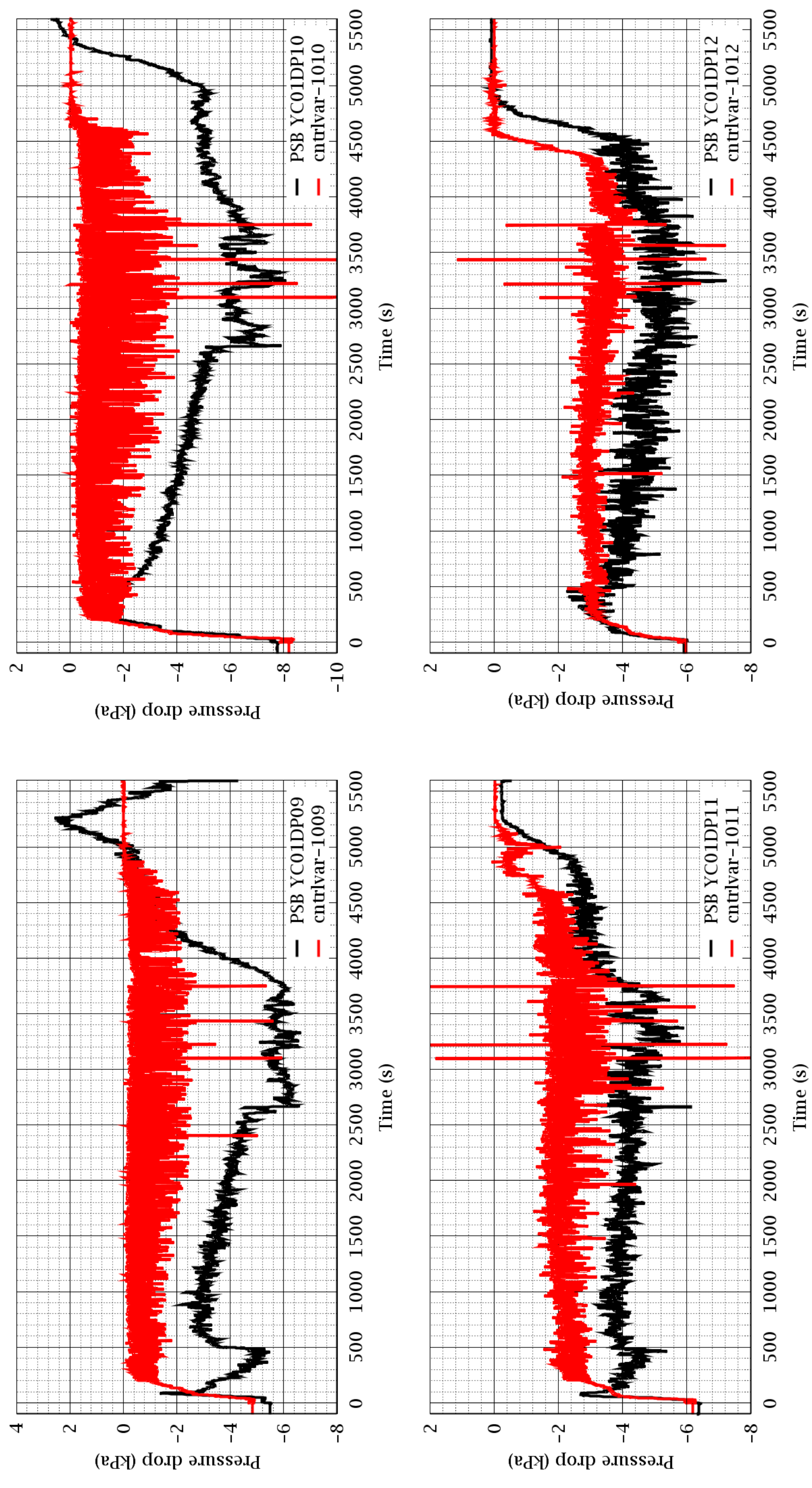


Figure D-19: Pressure Differences DP09-DP12 (FRS + Lower Part of Upper Plenum)

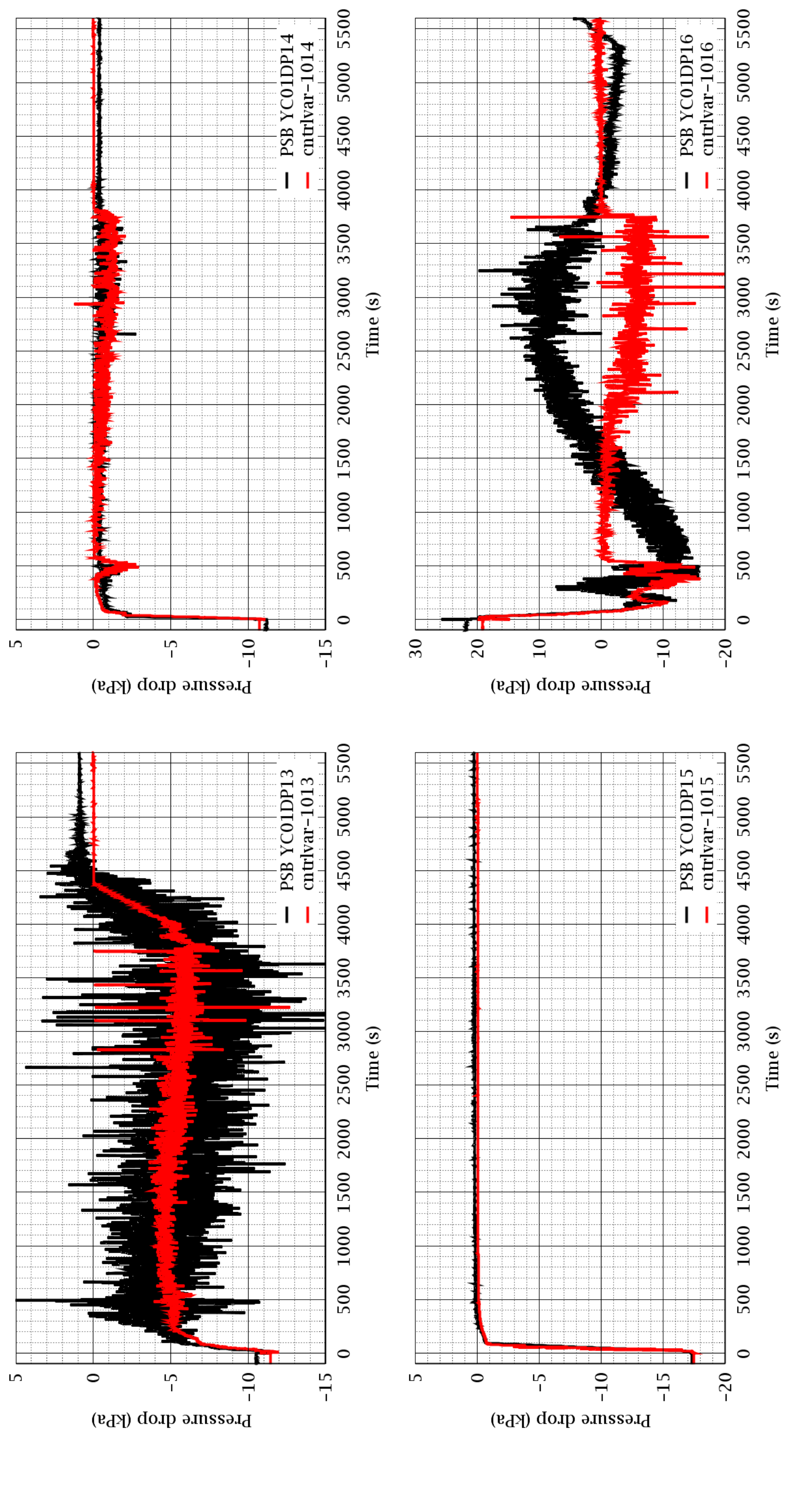


Figure D-20: Pressure Differences DP13-DP16 (Upper Part of Upper Plenum)

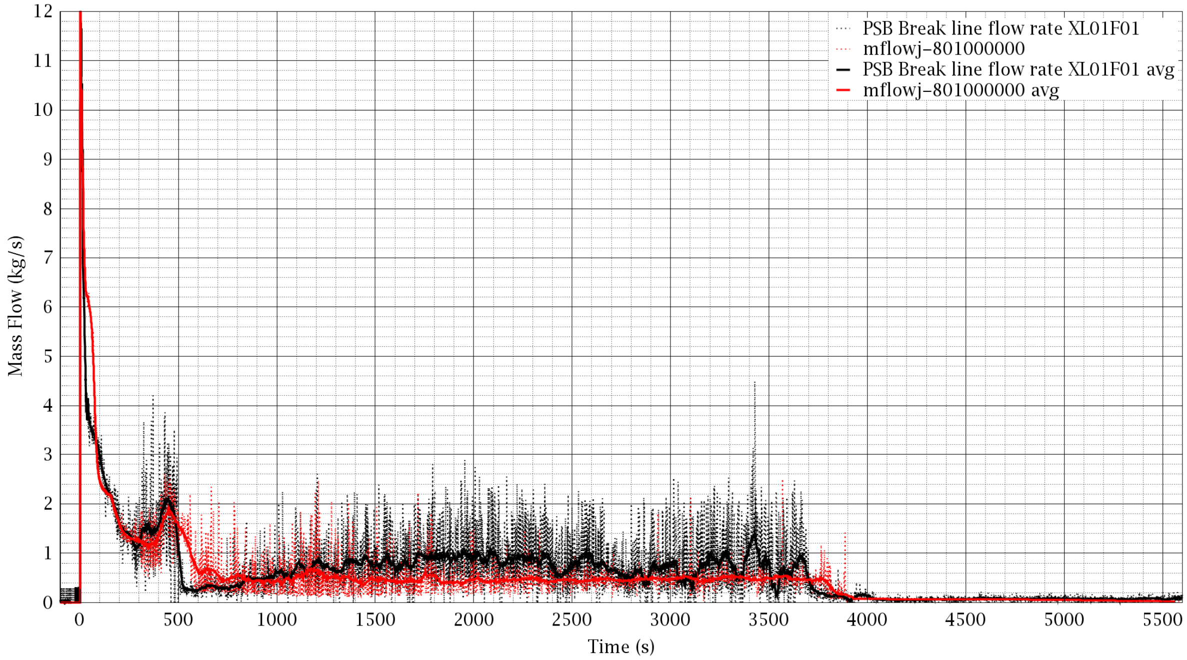


Figure D-21: Break Flow

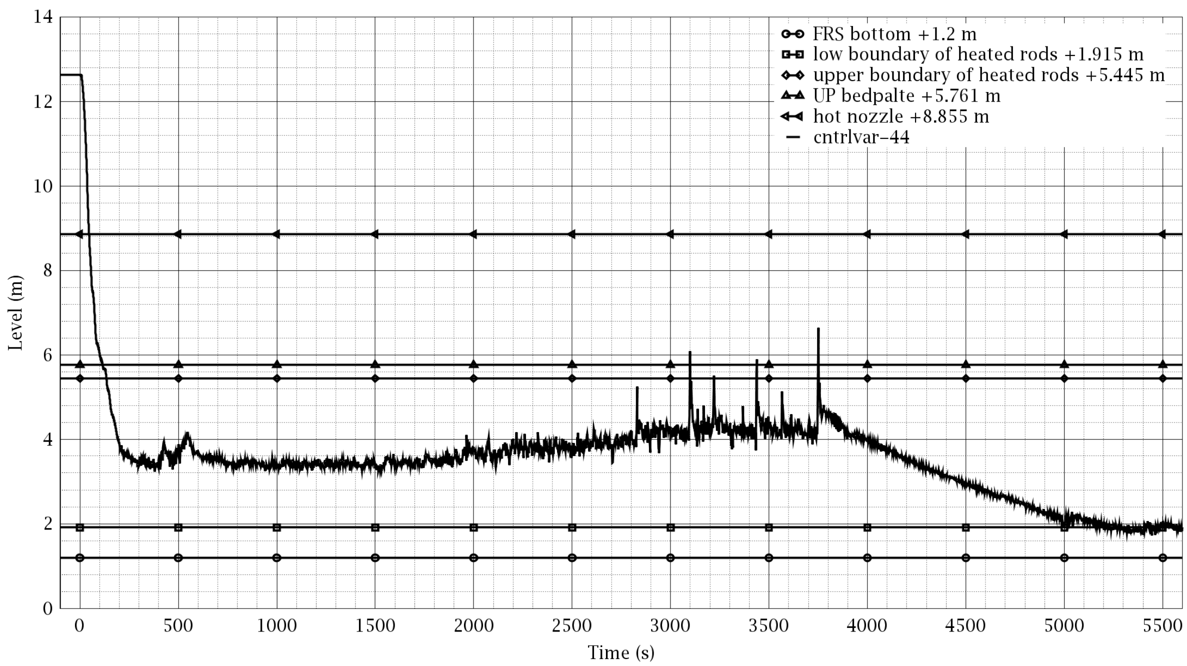


Figure D-22: Reactor Collapsed Level

NRC FORM 335 (12-2010) NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
NUREG/IA-0449

2. TITLE AND SUBTITLE
Post-Test Analysis of Upper Plenum 11% Break at PSB-VVER Facility using TRACE V5.0 and RELAP5/MOD3.3 Code

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	2014

4. FIN OR GRANT NUMBER

5. AUTHOR(S)
Petr Heralecky

6. TYPE OF REPORT
Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)
TES Ltd.
Prazska 597
674 01 Trebic, Czech Republic

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)
Division of Systems Analysis
Office of Nuclear Regulartory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
K. Tien , NRC Project Manager

11. ABSTRACT (200 words or less)
The best estimate thermo-hydraulic computer code TRACE V5.0 and RELAP5 MOD3.3 had been assessed using Upper Plenum 11% break experiment at the large-scale test facility PSB-VVER. The PSB-VVER facility is a 1:300 volume scaled model of VVER 1000 NPP located in Electrogorsk, Russia. An extensive TRACE and RELAP5 input decks of PSB-VVER facility were developed including all important components of the PSB-VVER facility: reactor, 4 separated loops, pressurizer, 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) and the RELAP5 code is its predecessor. The TRACE and RELAP5 codes are 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 both codes 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 purpose of performed analysis is to extend the validation of the TRACE and RELAP5 code focused on VVER type of NPPs. The TRACE calculation was performed in the frame of R&D project co-funded by The Ministry of Industry and Trade of Czech Republic. The RELAP5 calculation was performed to support standardization of the RELAP5 code in TES Company

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)
RELAP5/MOD3.3
Upper Plenum Break
VVER Russian Type Presurrized Water Reactor
Low Pressure Injection System (LPIS)
Integral Effect Test Facility (ITF)
TRACE
High Pressure Injection System (HPIS)
Electrogorsk Research and Engineering Institute (EREC)
Fuel Rod Simulator(FRS)

13. AVAILABILITY STATEMENT
unlimited

14. SECURITY CLASSIFICATION
(This Page) unclassified
(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

NUREG/IA-0449

Post-Test Analysis of Upper Plenum 11% Break at PSB-VVER Facility using TRACE V5.0 and RELAP5/MOD3.3 Code

July 2014