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Post-Test Analysis of Hot Leg 2x25% Break at PSB-VVER Facility using TRACE V5.0 Code

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ABSTRACT

The best estimate thermo-hydraulic computer code TRACE V5.0 has been assessed using Hot Leg 2x25% break experiment at the large-scale test facility PSB VVER. The PSB-VVER facility is a 1:3000 volume scaled model of VVER 1000 NPP located in Electrogorsk, Russia. 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 unit, main circulation pumps, steam generators, break section 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 code 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 required by regulators. The purpose of performed analysis is to extend the validation of the TRACE code focused on VVER type of NPP's. This work was performed in the frame of R&D project sponsored by The Ministry of Industry and Trade of Czech Republic.

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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
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 code validation process was mainly based on the data from experimental facilities or real NPP's of Western PWR type as well. There is significant number of VVER type of reactors operating all over the world and many other under construction or under preparation as well. VVER reactors are in many aspects similar to Western PWR's. Therefore a lot of experimental data measured on PWR's 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 analysis is to extend the validation of the TRACE code focused on VVER type of NPP's. The best estimate thermo-hydraulic computer code TRACE V5.0 has been assessed using Hot Leg 2x25% break experiment at the large-scale test facility PSB VVER. The PSB-VVER facility is a 1:3000 volume scaled model of VVER 1000 NPP located in Electrogorsk, Russia. In order to perform code validation an extensive TRACE input deck of PSB-VVER facility was developed including all important components of the PSB-VVER facility: reactor, four separated loops, pressurizer, break unit, main circulation pumps, steam generators, break section and important parts of secondary circuit.

2. FACILITY AND TEST DESCRIPTIONS

Detail information about the PSB-VVER test facility's systems and elements is given in Ref 4. Only a brief description of the PSB-VVER facility is given here. The hardware configuration for HL-2x25-02 test is reported below.

2.1 PSB-VVER Facility

The PSB-VVER test facility is a large-scale integral facility, whose structure is similar to that of the primary circuit of VVER-1000 nuclear power plant. The facility consists of four loops, each of them connected to the reactor model. Each loop includes a circulation pump, a steam generator model, and cold and hot legs. Volumetric - power scale of the facility is 1:300. The hot and cold leg elevations including the loop seals of the reference plant have been reproduced in the PSB-VVER. Figure 1 shows an isometric picture of the test facility, its main characteristics are given in Table 1.

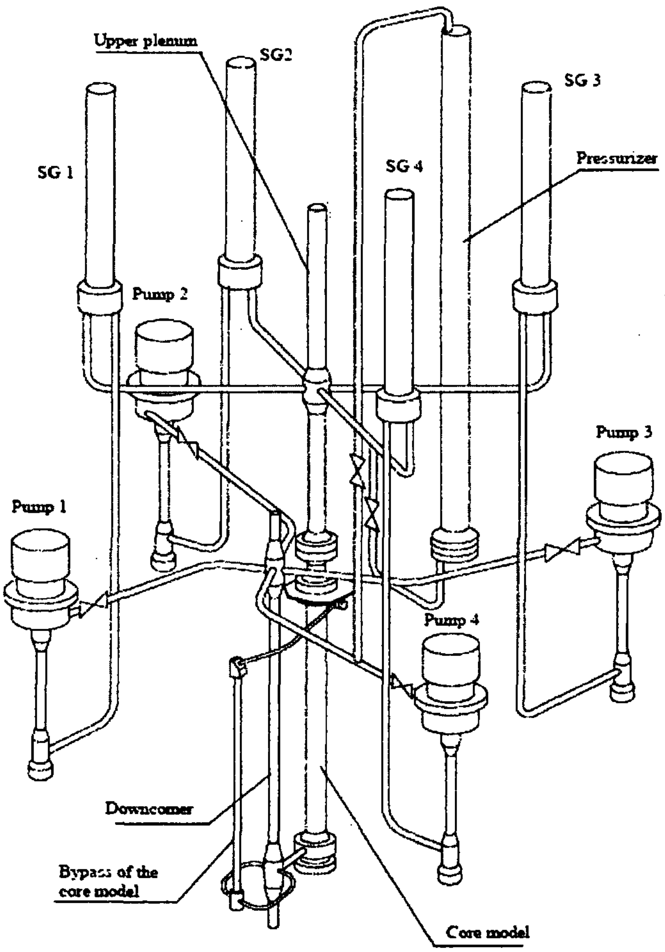

Figure 1: General View of PSB-VVER Facility

Primary and secondary circuits of the PSB-VVER facility are operated under nominal pressure of reactor prototype. The reactor model consists of four elements: external downcomer, core simulator, core by-pass, and upper plenum. The facility also has a pressurizer (PRZ) and ECCS.
ECCS of the PSB-VVER test facility includes, as the reactor VVER-1000, three subsystems: a passive and two active systems – high pressure injection system (HPIS) and low pressure injection system (LPIS). The passive system consists of four accumulators, which are connected in pairs to inlet and outlet chambers of the reactor. Cooling water of active ECCS can be supplied to three loops, both to cold and hot legs.

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.

Heat is removed from PSB-VVER primary circuit by four steam generators (SG's). 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.

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 "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.

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	10
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

To research thermal-hydraulics of large break accidents, there is special unit to reproduce guillotine break of hot leg in PSB-VVER. Principal scheme of this unit is given in Figure 2. There are installed two blowdown lines YE10 and YE20 to remove discharging coolant from each side of the pipe. Fast acting valve YA03S08, located between these two blowdown lines, closes cross section of hot leg during the breaks of special rupture discs.

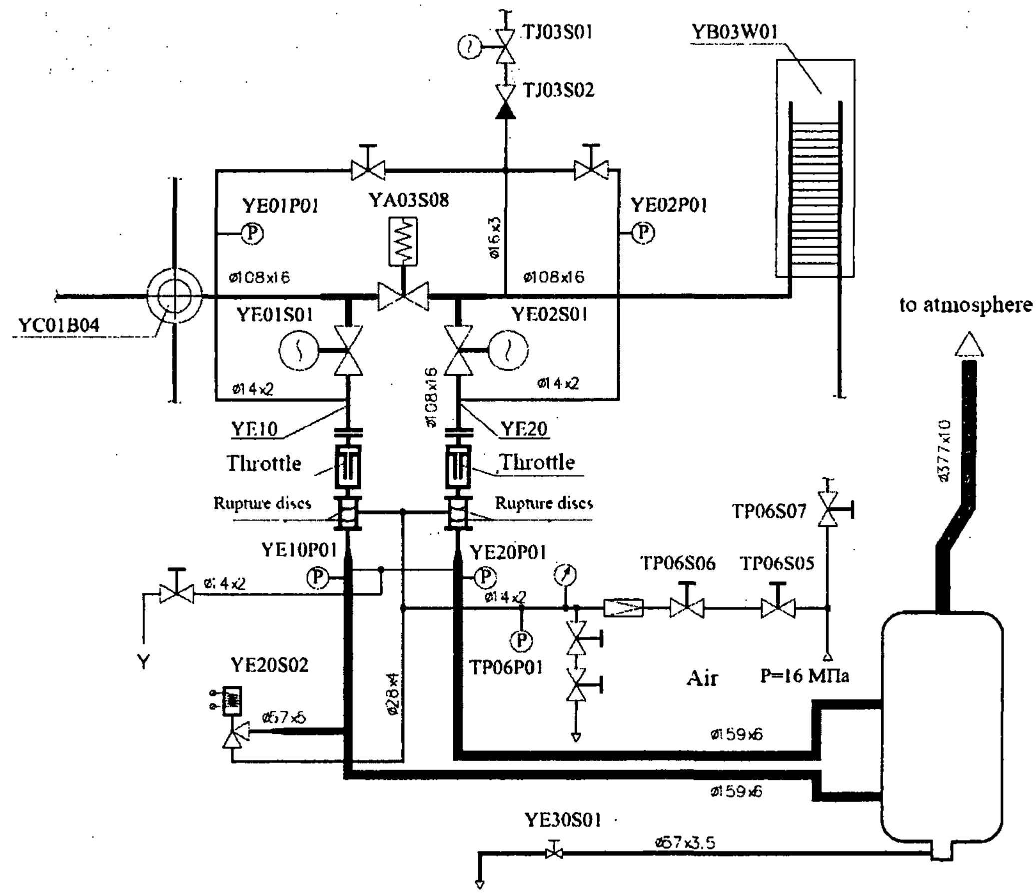

Figure 2: Principal Scheme of PSB-VVER Large Break Unit

To simulate fast break, each blowdown line contains special system of two rupture discs. To produce the break, the valve YE20S02 is opened to remove air from the cavity between discs. As a result, after instantaneous rupture of two discs, coolant begins to discharge in blowdown line. Throttles, determining a size of the break, are installed in the blowdown lines. Valves YE01S01, YE02S01 cut off blowdown lines from primary circuit for period of preparation of experiment. Before beginning of experiment they are fully opened.

2.2 Experiment HL-2x25-2

The test HL-2x25-02 „Hot Leg Large Break 2x25%“ was performed in the PSB-VVER test facility in Electrogorsk Research and Engineering Center (EREC) in Russia. The thermal-hydraulic processes related to hot leg large break 2x25% were researched.

2.2.1 Facility configuration

The information on the test facility hardware and configuration of the system specific for HL-2x25-02 test are given in the Table 2.

Table 2: Test Facility Configuration in HL-2x25-02 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 #3
ACC's	ACC's #1 and 3 were connected to UP. ACC #2 and 4 was connected to downcomer
SG's	All SG's 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	The break unit was located at the hot leg of the loop #3 between UP and SG. Leak channel in each blowdown line is a horizontal throttle of 25 mm, L = 250 mm.

2.2.2 Initial Conditions

The main initial conditions of HL-2x25-2 test are given in the Table 3. The HL-2x25-2 test has been performed under reduced initial core power corresponding to approximately 15% of nominal power.

Table 3: Main Operational Characteristics of PSB-VVER Facility

Parameter	Units	Value
Primary circuit		
Pressure in upper plenum	MPa	15.78
Coolant temperature (DC inlet / UP outlet)	deg	290 / 317
Primary loops flow rates	kg/s	2.32÷2.37
Core power	kW	1520
Core by-pass power	kW	16
Coolant level in PRZ (gauge YP01L02)	m	6.72
Secondary circuit		
Pressure in SG's	MPa	7.84 / 7.84 / 7.84 /7.84
Level in SG's (gauges YB01÷04L01)	m	1.67 / 1.67 /1.67 / 1.67
ECCS		
Pressure in ACC's (ACC 2 disconnected)	MPa	5.88 / - / 5.88 / 5.88

Level in ACC's (gauges TH01÷04P01)	m	4.84 / - / 4.90 / 4.90

2.2.3 Boundary Conditions (test scenario)

Detail information about the HL-2x25-2 test boundary conditions is given in Ref 5. The main events of HL-2x25-2 test are described in the Table 4.

Table 4: Main Events During HL-2x25-2 Test

Event	Time [s]
IE – Break Opening	0
Pressure in UP (YC01P16) < 13.73 MPa → conditions for SCRAM - initiation of MCP coastdown - FW termination to all SG's - initiation of core power and core bypass power reduction - steamline separation from all SG's	0.4 0.4 0.4 3 18
HPIS activation	27
ACC's activation	35 / - / 35 / 35
LPIS activation	78
ACC's empty	151÷172
Break termination (closing of blowdown lines)	335
HPIS termination	473
End of test (FRS power switched off)	1500

The experiment is started with opening of the valve YE20S02 in the air drainage line from the cavity between rupture discs. As a result, air pressure in the cavity drastically drops, and rupture discs are ruptured in two blowdown lines practically simultaneously. Coolant discharging from primary circuit is started. Simultaneously the valve YA03S08 is closed to prevent coolant flow in hot leg of the loop #3 between blowdown lines. Interval of time to close the valve YA03S08 is less 0.1 s. Interval of time to rupture discs is 0.001 – 0.01 s.

When the UP pressure P_{UP} = 13.73 MPa, the facility control system 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 BP is started
- Stop steam removal from steam generators
- Feed water temperature is decreased from 220 to 150°C during 30 s
- The procedure of MCP's coast down is started
- Close feed water supply
- Power of PRZ heater is switched off.

There was rather nonstandard coastdown of MCP's realized. MCP's rotational speed was temporarily stabilized at revolutions corresponding to 24% of nominal value for 173 second, then standard coastdown continued.

After achievement of two conditions: 1) UP pressure decreases to the value P_{UP} = 10.79 MPa and 2) later the moment 22 s after the experiment start (interval of time to start operation of diesel-generator), activation of high pressure injection system is simulated. Cooling water of

HPIS is supplied to cold leg of the loop #1. HPIS mass flow rate 0.12 kg/s is provided. Temperature of cooling water is 28°C.

HPIS operation is continued, while total volume of HPIS water supplied is achieved $0.052m^3$, later HPIS water supply is terminated (simulation of emptying of HPIS water tank).

After upper plenum pressure decreasing below 5.88 MPa, hydro-accumulators start to operate.

After upper plenum pressure decreasing below 2.16 MPa, activation of low pressure injection system is simulated. Cooling water of LPIS is supplied to hot and cold legs of the loop #3, mass flow rate is 0.35 kg/s per each line. LPIS operation continues till the end of the experiment.

After upper plenum pressure decreasing below 0.4 MPa, the valves YE01-02S01 are closed, and coolant discharging from primary circuit is terminated (simulation of containment pressure achievement).

Feed water flow rate provides SG's collapsed levels at nominal value during the experiment.

The experiment is terminated by switching off CS and BP powers and by closing of LPIS.

3. THE TRACE CODE

The TRACE (TRAC/RELAP Advanced Computational Engine) 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 is a finite-volume two-fluid compressible flow code 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 can be directly coupled with the 3 D reactor kinetics code PARCS or with the other TRACE *jobs and 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 non condensable 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. All TRACE calculations presented in frame of this report were performed using TRACE V5.0 code.

3.1 The TRACE code assessment

Confidence in the computational tools (codes) and establishment of their validity for a given application depends on assessment. TRACE, 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 database. 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 VVER typical features related to TRACE code assessment

The TRACE *code validation process is mainly based on the data from experimental facilities or* real NPP's of Western PWR type. VVER reactors are in many aspects similar to Western PWR's. Therefore a lot of experimental data measured on PWR's 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 50 operating of VVER type (Ref 18). It is a meaningful number compare to 216 operating 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 input deck of PSB-VVER facility was developed including all important components of the PSB-VVER facility: reactor, 4 separated loops, pressurizer, break unit, main circulation pumps, steam generators, break section and important parts of secondary circuit. The input deck was designed on the basis of PSB-VVER facility documentation (Ref 4, 20).

Nodalization diagrams of the PSB-VVER facility Input Deck are presented in Figure A-1 (reactor + primary circuit) and Figure A-2 (secondary circuit) in the Appendix A. Nodalization scheme of the primary circuit includes all four loops equipped with MCP's and SG's.

Two models of the reactor pressure vessel (RPV) with internal structures were developed. Both models of RPV are 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 last part represents core by-pass from DC to UP. Both models employs VESSEL component to model DC + LP (includes 26 axial layers, 1 azimuthal theta sector and 2 radial rings) and 3 PIPE components to model core by-pass. The Fuel Rod Simulator optionally employs one component VESSEL (includes 32 axial layers, 4 azimuthal theta sector and 6 radial rings) or 2 PIPE components, where one represents main part of FRS and the other represents the hot nozzles region.

Notice Results of the model which employs VESSEL component to model FRS is labeled "3D" and the other model which employs 2 component PIPE is labeled "1D".

The pressurizer is modeled as PIPE component equipped with the heaters. It can be connected to loop No 4 or to loop No 2 of the facility (as original facility design). Main circulation pumps are modeled as PUMP components using original characteristics from Ref 20. HPIS and LPIS systems are modeled as easy boundary conditions using FILL components. The large break unit is simulated using BREAK component on the hot leg No 3.

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 SG's 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 SG's and the common steam header are also modeled. The BREAK component simulates the release of secondary steam from steam header.

Components Statistic for both TRACE models – see the next Table 5.

Table 5: TRACE Components Statistic

TRACE Component	3D Model	1D Model	Notes
VESSEL	2	1	DC+LP (1D, 3D); FRS+UP (3D)
PIPE	133	135	FRS+UP (1D model)
HSTR	170	143	-
POWER	2	2	FRS simulator + By-pass heating
VALVE	24	24	-
PUMP	4	4	MCP's
BREAK	3	3	Large Break unit + release of secondary steam
FILL	7	7	HPIS, LPIS, Feedwater
Whole No of Components	345	319	-

5. RESULTS

5.1 Steady-state calculation

In order to achieve stable initial conditions of the HL-2x25-2 test, the steady state was calculated for 1600 s. The following controllers were used for the first 1000 s:

- Pressurizer pressure controller
- Main steam header pressure controller
- Steam generators level controllers

The other controlled parameters (fuel rod simulator power, core bypass power, feedwater temperature) were described as boundary conditions. It was necessary to decrease initial pressure in SG's within 0.5 MPa compared with measured parameters in order to achieve good results with primary temperatures. During the steady-state calculation, all SG's levels were maintained by PI-controller, which was replaced by an original P-controller for the last 120 s to get corresponding behavior of model before beginning of the experiment. All SG' levels controllers were switched off for the last 600 s of the steady state to check the steady-state calculation stability. Steady state calculation took approximately 100 min, main calculated and measured parameters are compared in the next table.

Table 6: Initial Conditions (TRACE calculation vs. experiment comparison)

Parameters	Units	HL-2x25-2	TRACE	
			3D	1D
Primary circuit				
Upper plenum pressure (YC01P16)	MPa	15.79 ± 0.02	15.771	15.770
Pressure drop at FRS (YC01P07-P10)	kPa	29.39	27.25	28.59
Coolant temperature at FRS inlet (YC01T06)	°C	290.6± 0.14	288.9	288.8
Coolant temp. at FRS outlet (YA01T04b)	°C	317.8 ± 0.16	319.6	322.2
Loop-1 flow rate (YA01F01)	kg/s	2.31 ± 0.03	2.275	2.138
Loop-2 flow rate (YA02F01)	kg/s	2.37 ± 0.01	2.280	2.138
Loop-3 flow rate (YA03F01)	kg/s	2.31 ± 0.01	2.284	2.140
Loop-4 flow rate (YA04F01)	kg/s	2.19 ± 0.01	2.285	2.138
FRS power (YC01N01)	kW	1533.0 ± 14.4	1533.0	1533.0
Core by-pass power (YC01N02)	kW	16.1 ± 0.3	16.1	16.1
Collapsed level in PRZ (YP01L02)	m	6.97 ± 0.02	7.125	6.835
Secondary circuit				
Pressure in SG-1 (YB01P01)	MPa	7.839 ± 0.06	7.319	7.326
Pressure in SG-2 (YB02P01)	MPa	7.755 ± 0.07	7.322	7.320
Pressure in SG-3 (YB03P01)	MPa	7.805 ± 0.06	7.310	7.308
Pressure in SG-4 (YB04P01)	MPa	7.734 ± 0.07	7.326	7.323
Collapsed level in SG-1 (YB01L01)	m	1.66 ± 0.01	1.693	1.689
Collapsed level in SG-2 (YB02L01)	m	1.66 ± 0.02	1.684	1.685
Collapsed level in SG-3 (YB03L01)	m	1.67 ± 0.01	1.684	1.685

Collapsed level in SG-4 (YB04L01)	m	1.60 ± 0.03	1.668	1.670
Hydroaccumulators				
ACCU-1 pressure (TH01P01)	MPa	5.924 ± 0.003	5.946	5.946
ACCU-2 pressure (TH02P01) out of service	MPa	5.874 ± 0.004	5.887	5.887
ACCU-3 pressure (TH03P01)	MPa	5.873 ± 0.005	5.889	5.889
ACCU-4 pressure (TH04P01)	MPa	5.890 ± 0.004	5.910	5.910
ACCU-1 collapsed level (TH01P01)	m	4.84 ± 0.002	4.845	4.845
ACCU-2 collapsed level (TH02P01)	m	4.82 ± 0.002	4.822	4.822
ACCU-3 collapsed level (TH03P01)	m	4.91 ± 0.001	4.910	4.910
ACCU-4 collapsed level (TH04P01)	m	4.90 ± 0.001	4.896	4.896

5.2 Transient calculation

The post-test calculation of HL-2x25-2 experimental test at PSB-VVER facility started at the time 0 s with initiating event – double ended break 25% at the hot leg of the Loop No 3. A comparison of calculated and experimental times of the occurrence of main events is presented in the next Table 7.

Table 7: Chronology of main events (TRACE calculation vs. experiment comparison)

	Time [s]		
	PSB	TRACE	
Event	HL-2x25-2	3D	1D
Initiating Event – Leakage opening	0	0	0
Pressure in UP (YC01P16) < 13.73 MPa ⇒ Reactor SCRAM signal	0.4	0.34	0.31
- start of MCP coastdown	0	0.34	0.31
- start of FRS and BP power reduction	3	3.64	3.61
- signal for stopping feed water supply (RL01-04S01)	0.4	0.34	0.31
- signal for stopping steam discharge (RA06S01)	0	0.34	0.31
PRZ emptying (according to YP01L02 / complete emptying)	11 / -	13/28	12/27
Start of HPIS injection into Loop 1	22	28	28
Pressure in UP (YC01P16) < pressure in ACCU's - start of ACCU-1 operation	34	31	32
- start of ACCU-2 operation	34	32	33
- start of ACCU-3 operation	35	32	34
- start of ACCU-4 operation	36	32	34
Start of LPIS injection into Loop 3	65	71	67
Levels in ACCU's (TH01-04L01) < 0.1 m - ACCU's 1-4 separated	129 - 134	138 - 158	144 - 166
Break termination (closing YE01S01 a YE02S01) - containment pressure achievement	335	335	335
HPIS injection termination	473	470	470
End of the experiment – FRS power switched off	1500	1500	1500

The graphs of main measured and calculated parameters are presented bellow. The parameters labeled "3D" came from model which employs VESSEL component to model FRS. The parameters labeled "1D" came from model which employs 2 PIPE components to model FRS.

The following description in this chapter is provided for 3D model results. Comparison of results from 3D vs. 1D model has been done in the next Chapter 5.3.

Calculation started at 0 s by simultaneously opening both blowdown lines and closing valve YA03S08 between both blowdown lines.

Due to high initial break flow (see Figure 21), primary parameters were rapidly decreasing – see primary pressure at Figure 4, pressurizer level at Figure 6 and cladding temperature at Figure 5. At 0.05 s PRZ electric power was switched off.

At 0.34 s UP pressure decreased below 13.73 MPa (Figure 4) and power to core simulator and by-bass started to decrease along to the specified function (Figure 10). The initial rate of PZR pressure decrease was lower than one in FRS, which was caused by presence critical flow rate in the surge line, see Figure 4. PRZ pressure became equal to primary pressure at 28 s (after PZR emptying). Collapsed level in PZR was decreased to 1.885 m (in accordance to the gauge YP01L02).

Detailed analysis of the first period calculation is presented below. There can be identified 3 stages.

First stage of calculation (0 – 1.1 s) is characterized by fast depressurization, primary pressure was very quickly decreased from 15.7 MPa to 10 MPa as a result of a high initial break flow (the initial rate of primary pressure decrease was approximately 5.2 MPa/s.) The first stage of calculation was in very good agreement with experimental data, see Figures 4, 6 to compare primary pressure and collapsed water level in pressurizer – most important parameters, which describes dynamic behavior of the whole primary circuit.

During the second stage of the calculation (1.1 – 28 s) the rate of primary pressure decrease become lower (approximately 150 kPa/s). Primary pressure became lower than secondary one at 16 s (Figures 3, 11). Thus, reverse of heat exchange in SG's took place. Rate of primary pressure decrease after the reverse of heat exchange in SG's was decreased to 90 kPa/s. During the second stage of calculation the value of primary pressure was slightly lower than during experiment, but pressure progression coincide with experimental data very good.

Third stage (28 – 160 s) is characterized by emergency water supply. At 28 s cooling water was started to be supplied to primary circuit from HPIS, mass flow rate was 0.12 kg/s (the same like during PSB test), see Figure 7. HPIS operation was terminated at 470 s, after total volume of HPIS water supplied had achieved 0.052 m^3. At 32 s primary pressure decreased below a value of pressure in ACC's and cooling water from ACC's started to be supplied to DC and UP, see ACC levels in Figure 9. Termination of water supply from ACC's occurred between 138 and 158 s, after level in a particular ACC decreased below 0.1 m (in order to prevent non-condensable gas from penetrating into primary circuit). At 65 s started LPIS and continued in operation until the end of the calculation, mass flow rate was 0.35 kg/s to the cold leg and to the hot leg as well. Due to HPIS and ACC's water supply the rate of primary pressure decrease increased up to 167 kPa/s at 44 s. After termination ACC's operation the primary pressure was fairly stable and the rate of primary pressure decrease went up to very low value until the moment of blowdown line closing at 335 s. Calculated time of beginning acting particular emergency water supply from HPIS, LPIS and ACC's was comparable to experimental data, see Table 7 (the longest time difference equaled to 6 s). Calculated stable primary pressure in the end of ACC's discharge, after 115 s, was slightly higher than during experiment. As a result of higher calculated primary pressure, ACC's discharge time was slightly longer then during experiment, see Figure 9.

Figures 13 – 16 present a comparison of calculated and experimental transients of coolant temperatures at inlet of SGx (YAxxT04) and at LOOPx seal (YAxxT26) in broken Loop No 3 and intact Loop No 4 (behavior of temperatures in other intact loops No 1 and 2 is analogous). Figures show that calculated temperatures in intact loop No4 behaved in a similar manner whereas broken loop No3, where an essential difference was observed due to presence of the break here. All calculated temperatures were in quite a good accordance with experimental ones.

Figure 5 presents cladding temperature at the top of the rod bundle. After 0.34 s when UP pressure decreased below 13.73 MPa, power to core simulator started to decrease along to the specified function so cladding temperature was rapidly decreasing and was near saturation temperature until the location of cladding temperature was filled with cooling water. The calculated and experimental transient was in good agreement during the period when the break was opened, but after break had been terminated (at 335 s - see Figure 33) and core simulator begun to be flooded, calculated transient was slightly faster than experimental one. The reason of this behavior is discussed in more detail in the next paragraphs.

The whole calculated transient can be divided into two main stages. The first stage from 0 to 335 s is characterized by blowdown of primary circuit due to coolant water discharge from break. The second stage from 335 s, when the break was terminated, till end of calculation where primary circuit was intensively filled with water as a result of LPIS operation. Water inventory and its distribution in reactor (DC, FRS and UP) is described as collapsed water in FRS and UP, see Figure 23, and as differential pressures over the reactor, see Figures 17 – 20. The differential pressure measurements location in the reactor are given in Figure B-1 in Appendix B.

During an early time period of blowdown stage 0 to 28 s, when no cooling water was supplied, collapsed water decreased dramatically (see Figure 23) and upper part of DC and UP was emptied - see differential pressures at Figure 15, 16 and Figure 21 – 29 (lower absolute value of pressure drop means lower liquid volume between gauges of differential pressure measurement). After cooling water started to be supplied to primary circuit from HPIS at 28 s, the rate of the reactor collapsed level decrease got lower. After ACC'S discharge begun at 32 s the total mass flow supplied from HPIS and ACC's was higher than the break coolant discharge, see Figure 22. As a result of higher cooling water supply than the break discharge, level in reactor started slightly increasing. In the end of ACC's coolant water discharge and mainly after its termination, coolant water supply from HPIS and LPIS was not able to compensate the break discharge and the reactor collapsed level started again to decrease until termination of break discharge, see Figures 21, 23. All described behavior of collapsed level in the reactor during ACC's discharge are clearly visible on pressure differences in FRS and UP, see Figures 17 - 19.

Calculated reactor water inventory and its distribution over the reactor were in a fairly good accordance with experimental ones, but calculated water distribution slightly differs from experimental one. In case of calculation, more liquid was kept in lower plenum and lower part of FRS than in case of the experiment. Calculated absolute value of pressure differences had higher value than experimental ones - see Figures 17 – 20. As a result such a liquid distribution the calculated break mass flow was lower than during the experiment. Lower break discharge means higher water inventory in the reactor, see Figure 23.

At 335 s, the same time like in experiment, break was terminated and valve YA03S08 between both blowdown lines was opened to make possible water flow through loop No3. After closing of blowdown lines, filling of primary circuit was realized intensively, see reactor collapsed level at Figure 23. Calculated progress of filling of primary circuit was faster than calculated – the reason (higher amount of liquid in lower plenum) was described in previous paragraphs. Filling of particular regions of the reactor is described by rising of absolute value of pressure differences over the reactor. If value of pressure difference is nearly 0, it means emptied region – only vapor on it. Contrariwise, if value of pressure difference is near nominal value and is stable, it means region is filed – only liquid on it, see Figures 17 - 20.
Pressurizer filling started at 1235 s which is good coincidence with experiment where it happened at 1300 s.

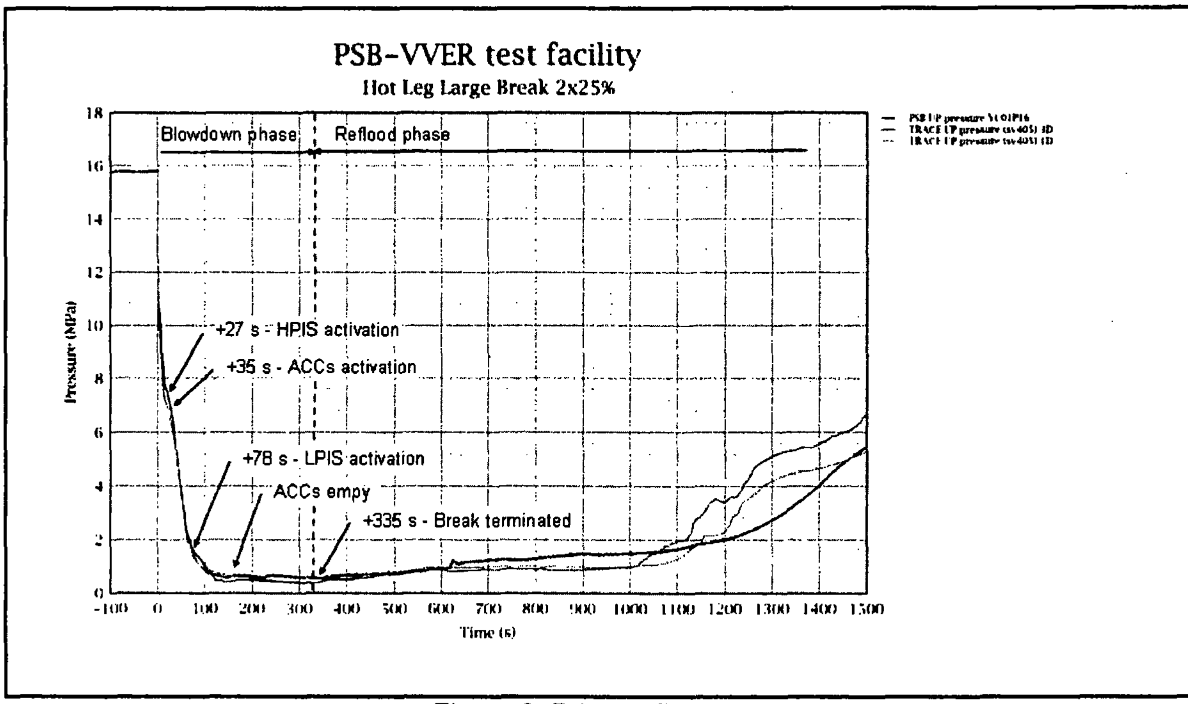

Figure 3: Primary Pressure

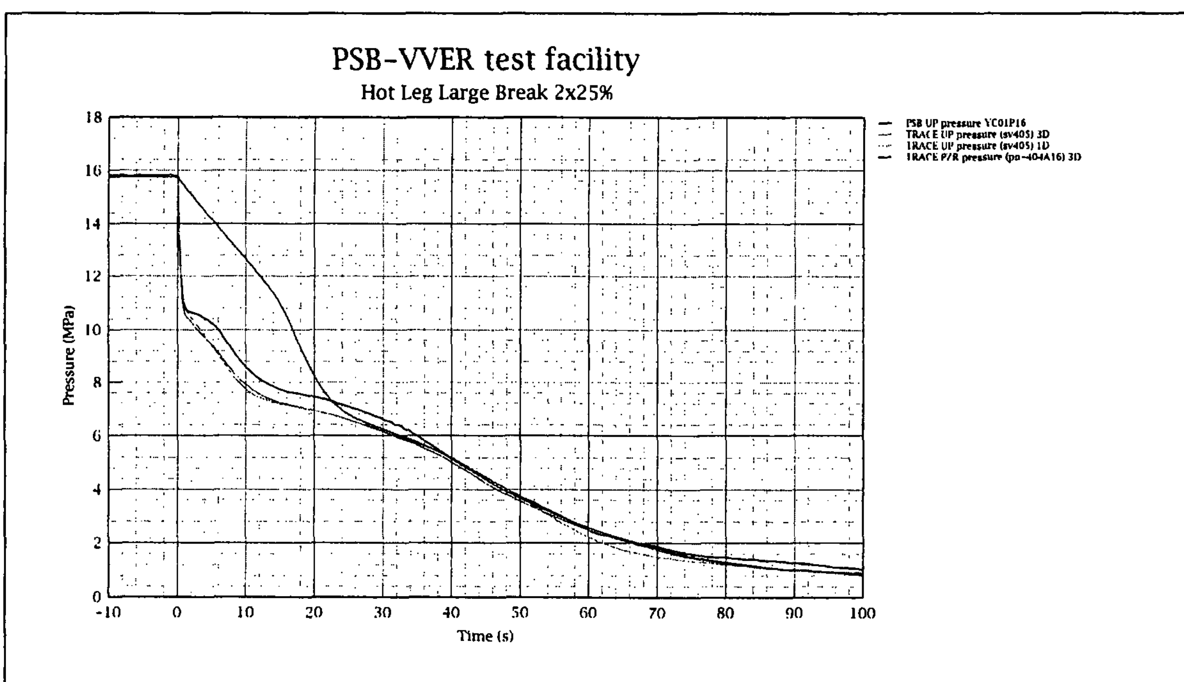

Figure 4: Primary Pressure (detail)

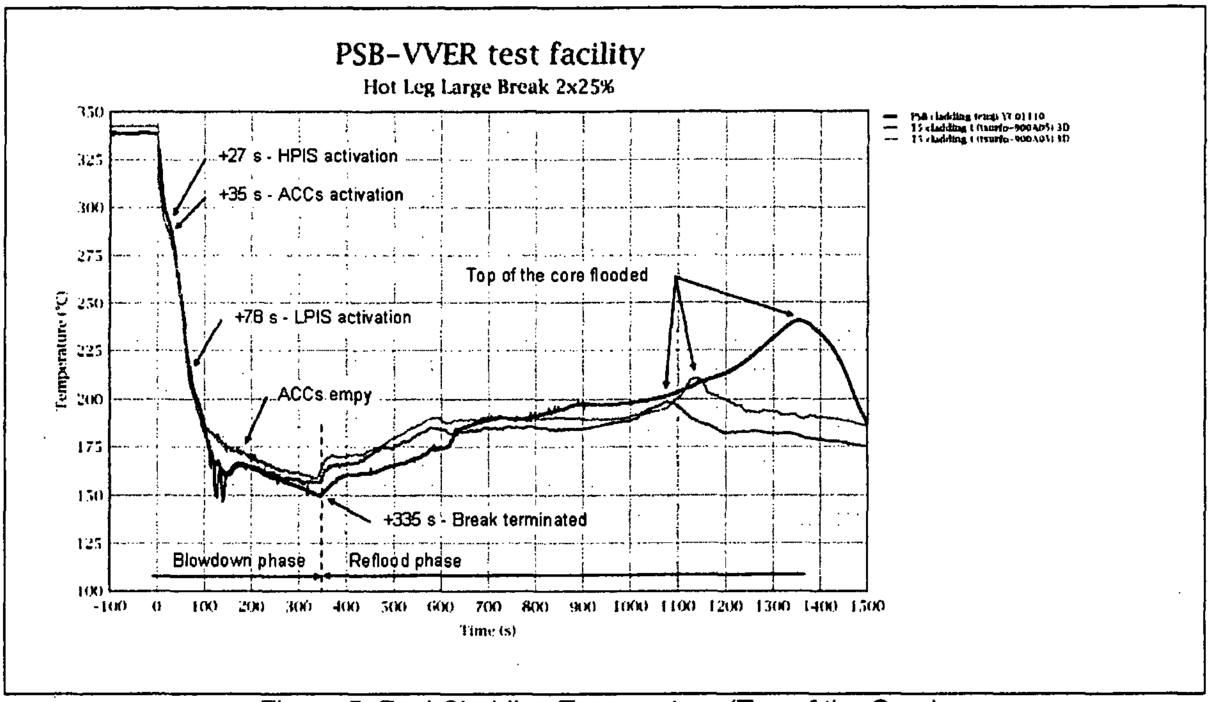

Figure 5: Fuel Cladding Temperature (Top of the Core)

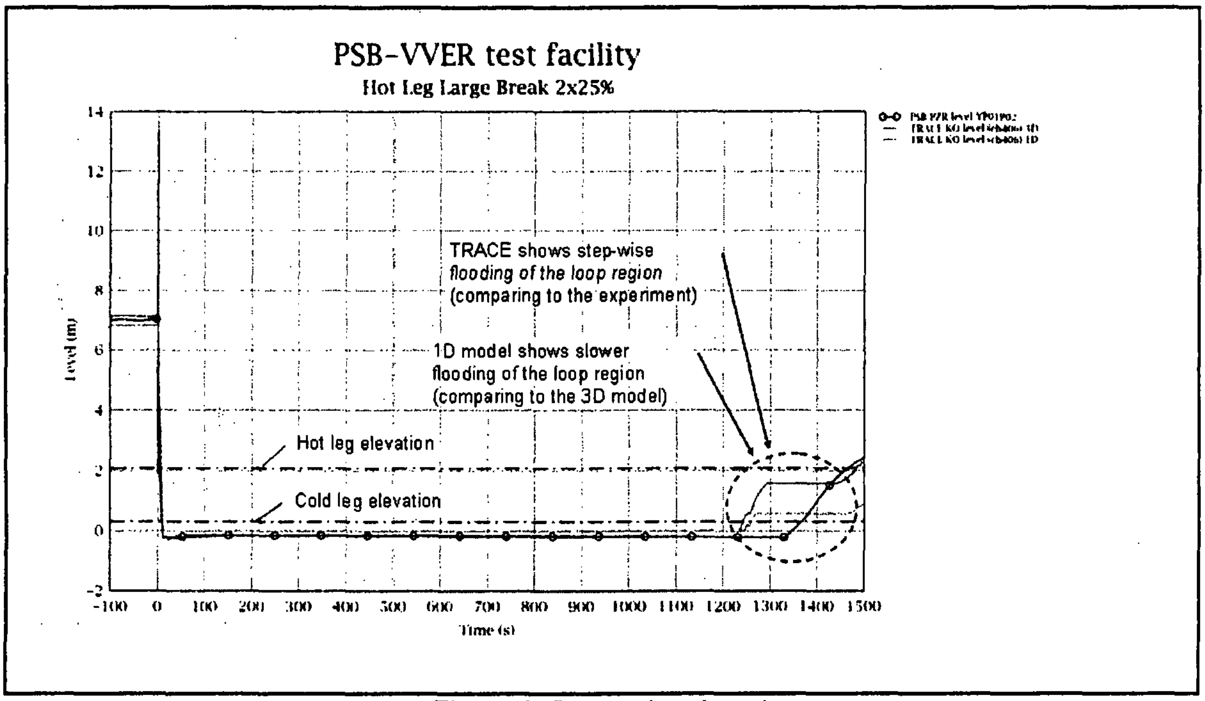

Figure 6: Pressurizer Level

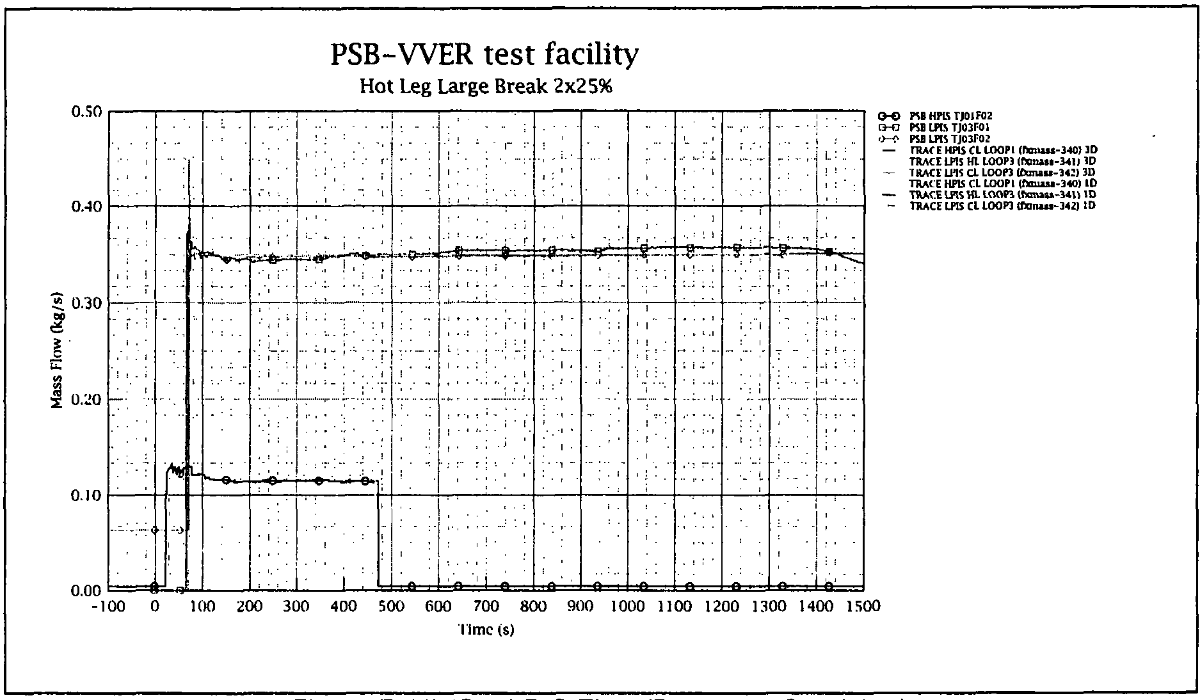

Figure 7: HPIS + LPIS Flow (Boundary Condition)

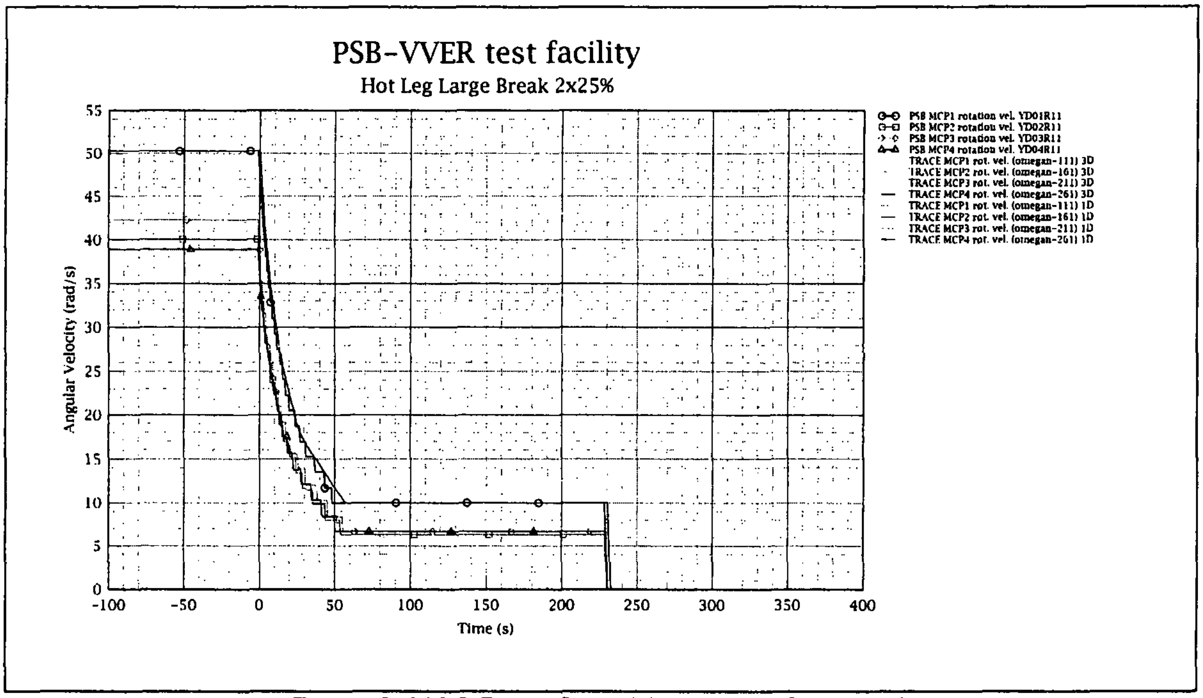

Figure 8: MCP Rotor Speed (Boundary Condition)

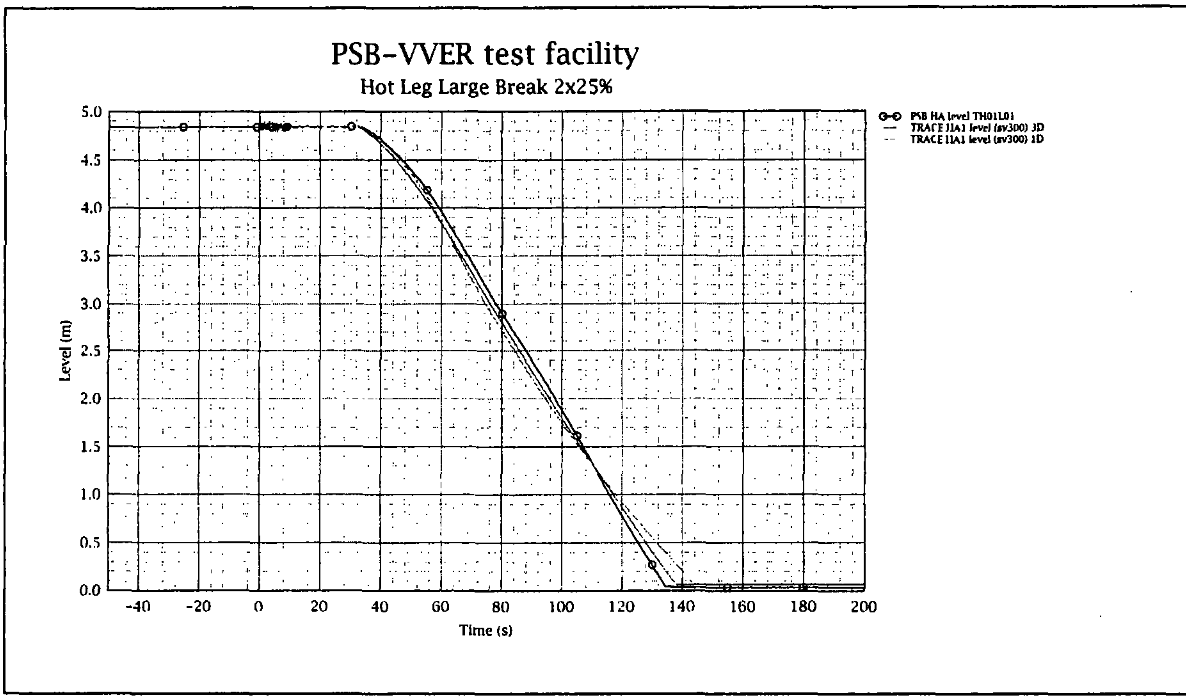

Figure 9: Hydroaccumulators Levels

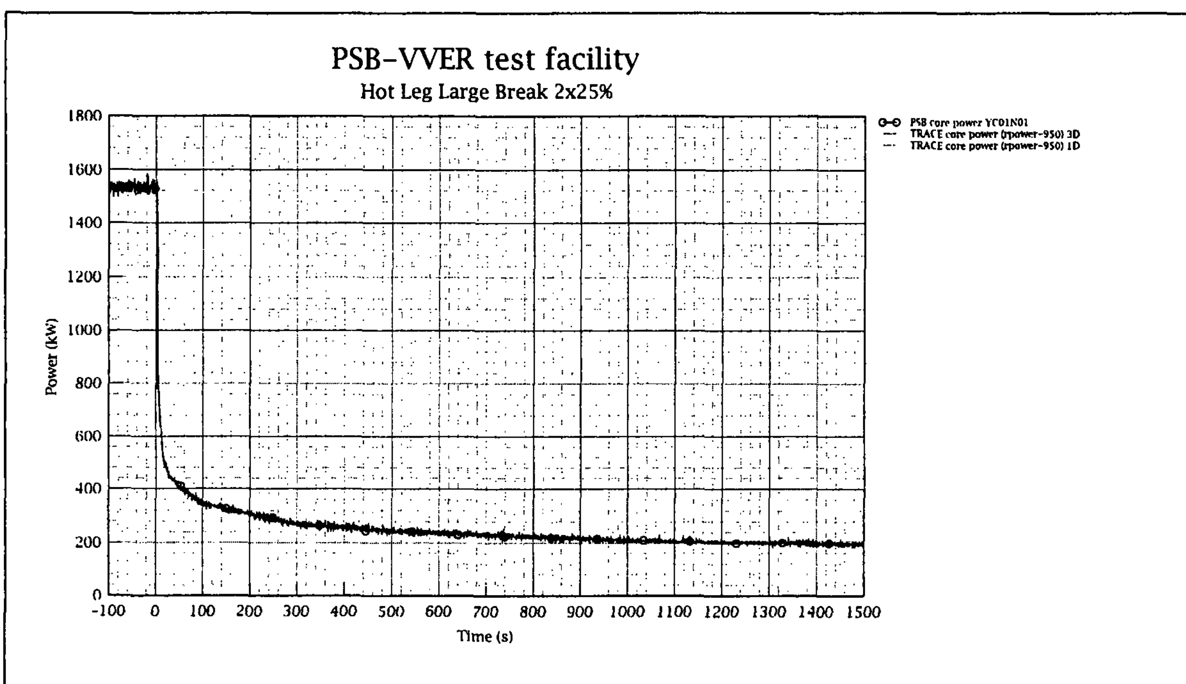

Figure 10: Fuel Rod Simulator Power (Boundary Condition)

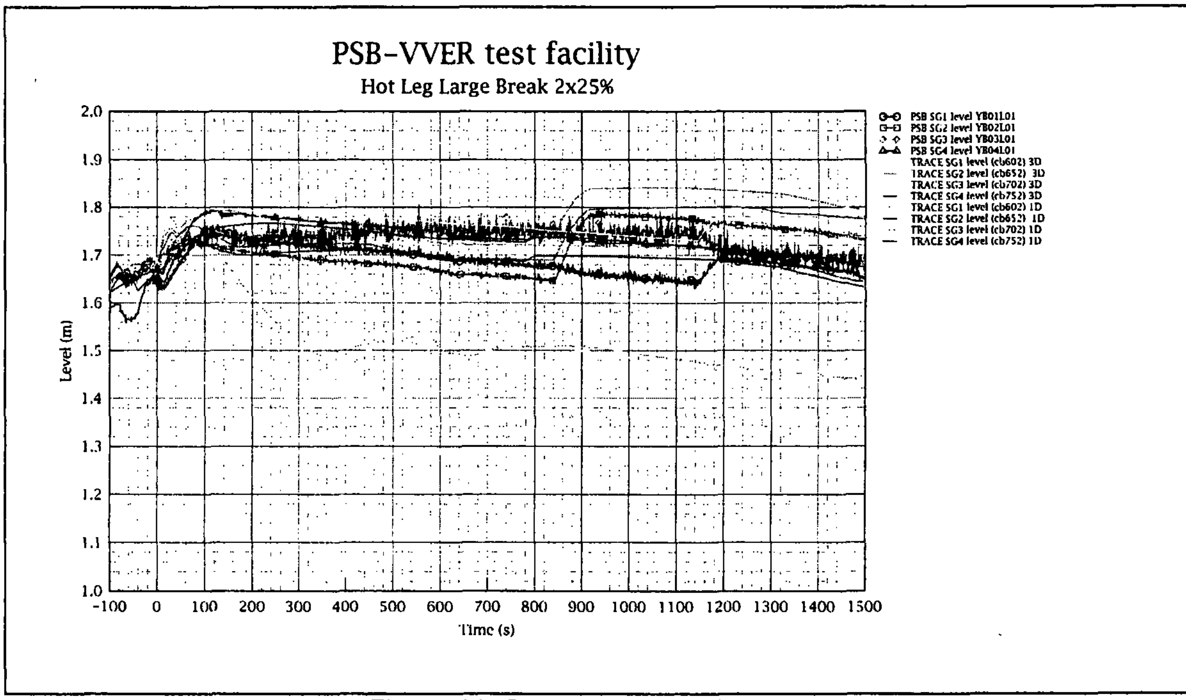

Figure 11: Secondary Side Pressures

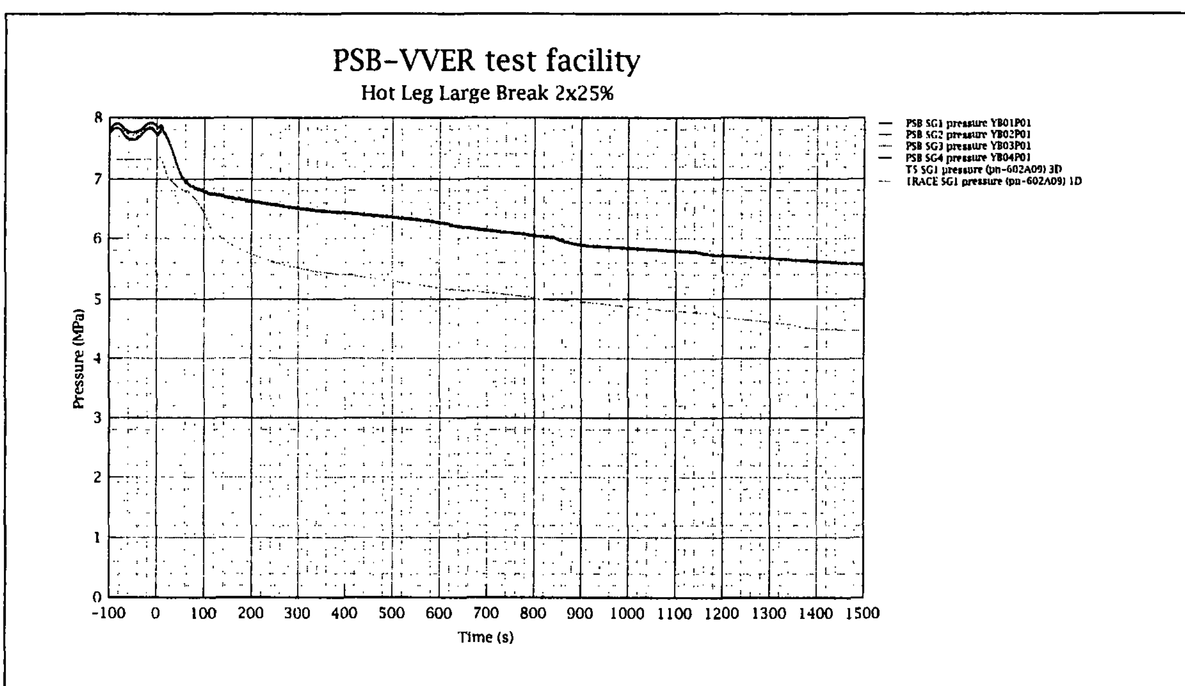

Figure 12: Steam Generators Levels

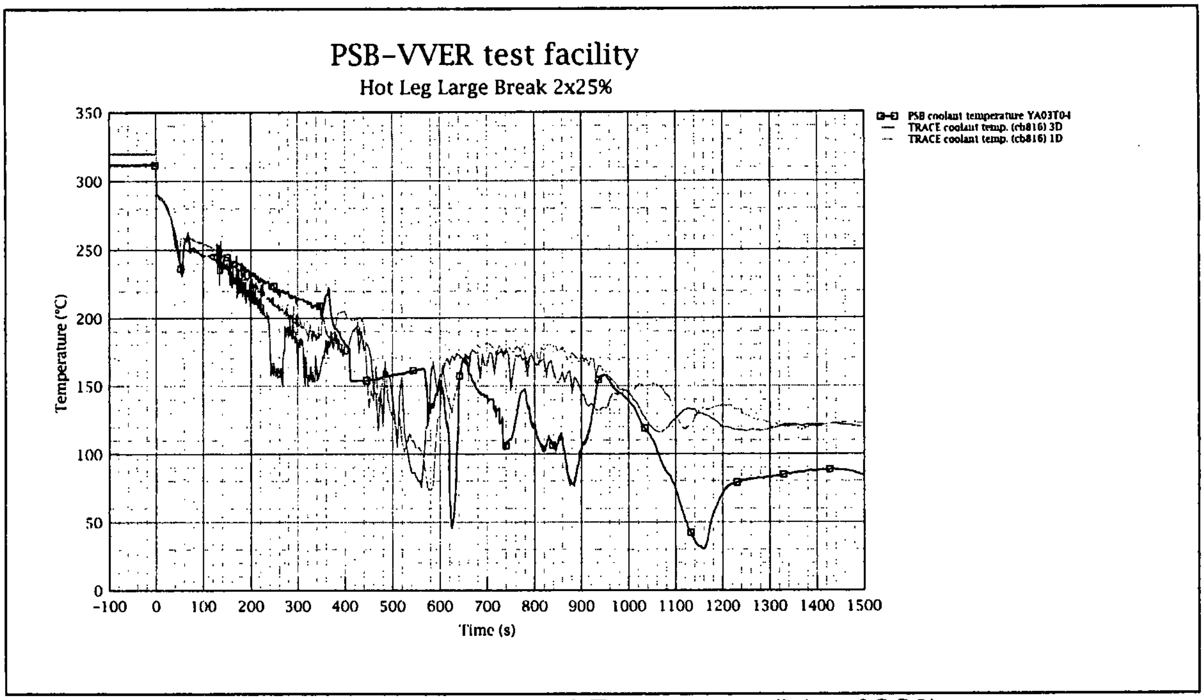

Figure 13: Broken Loop 3 Temperatures (Inlet of SG3)

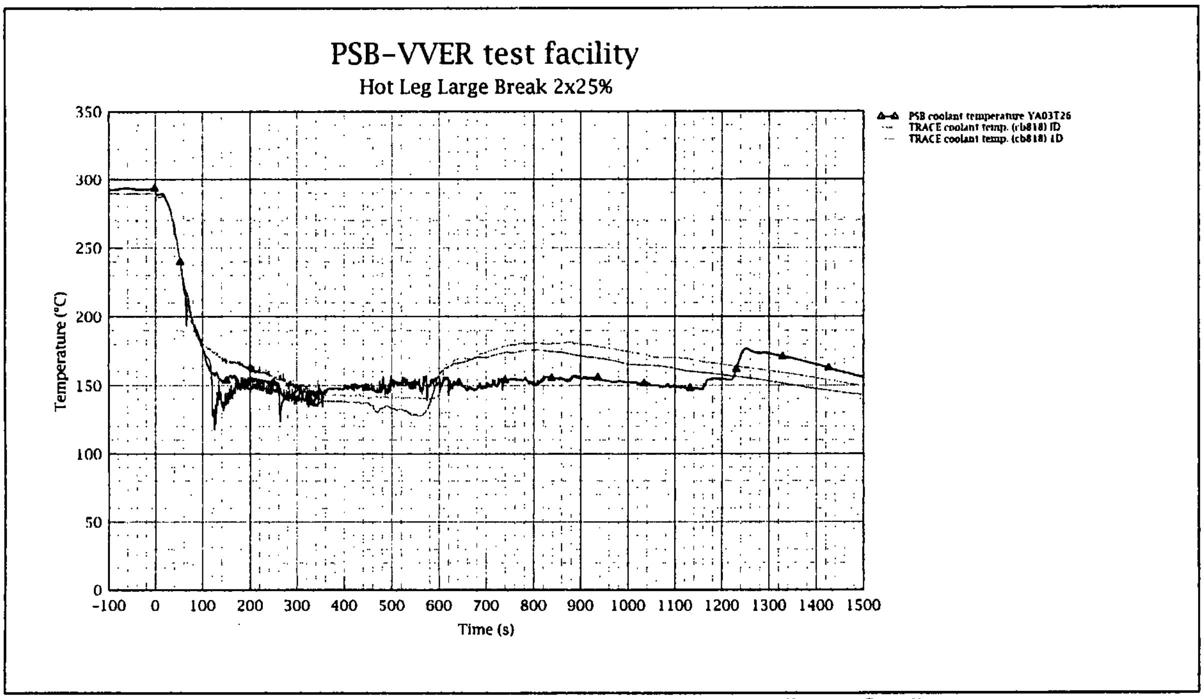

Figure 14: Broken Loop 3 Temperatures (Loop Seal)

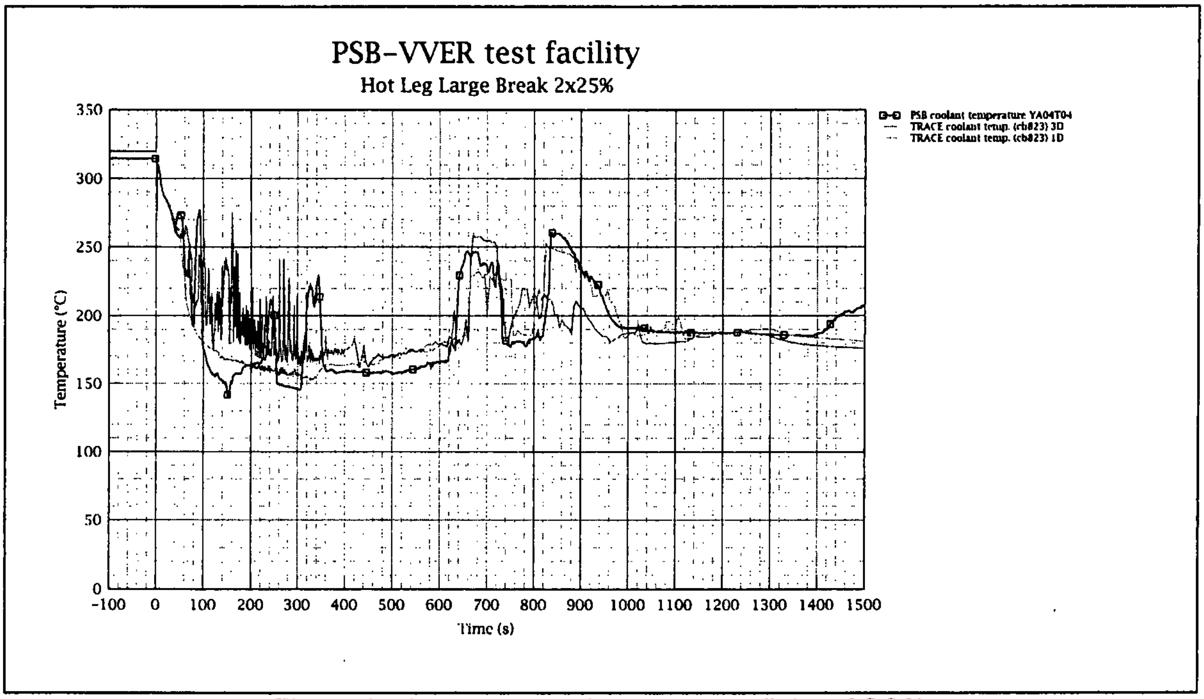

Figure 15: Intact Loop 4 Temperatures (Inlet of SG3)

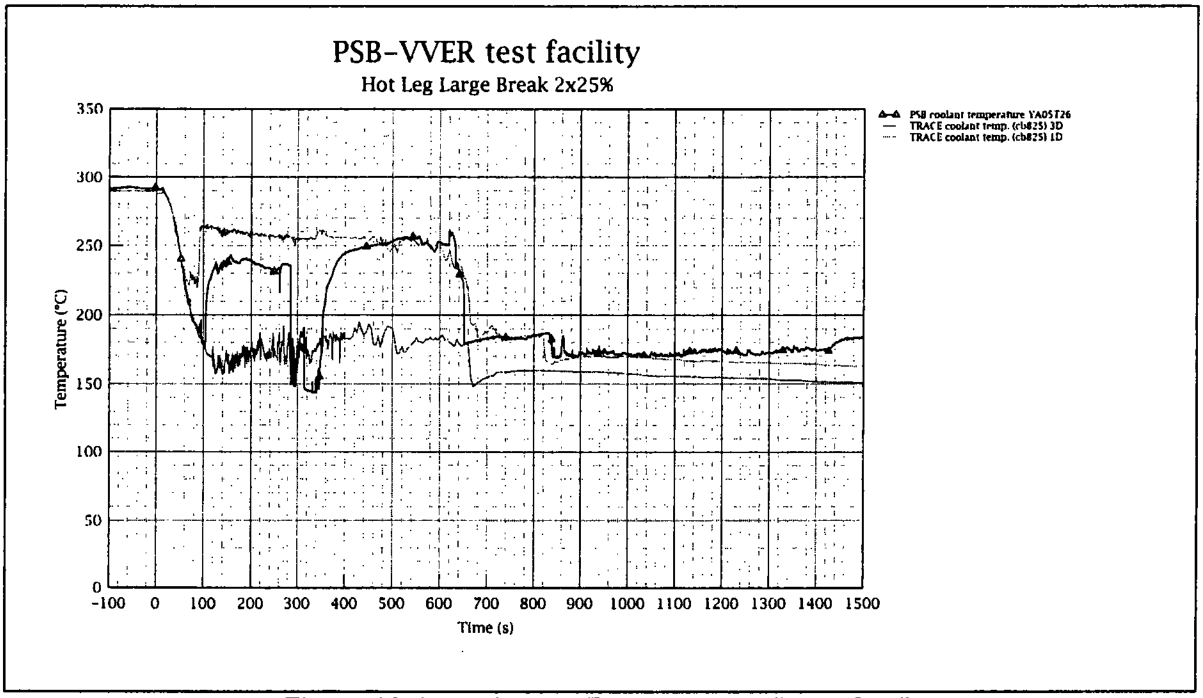

Figure 16: Intact Loop 4 Temperatures (Loop Seal)

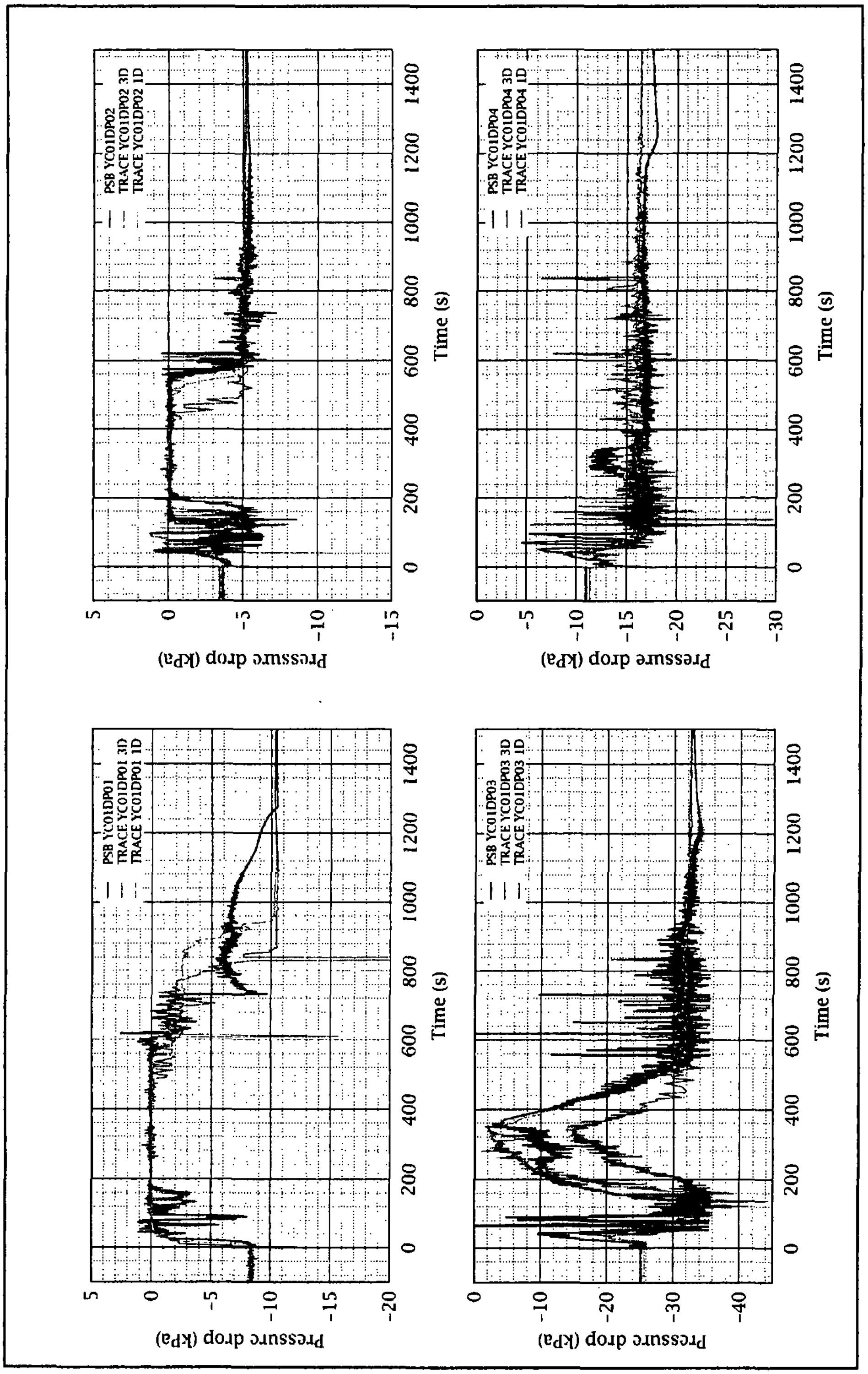

Figure 17: Pressure Differences DP01-DP04 (Downcomer)

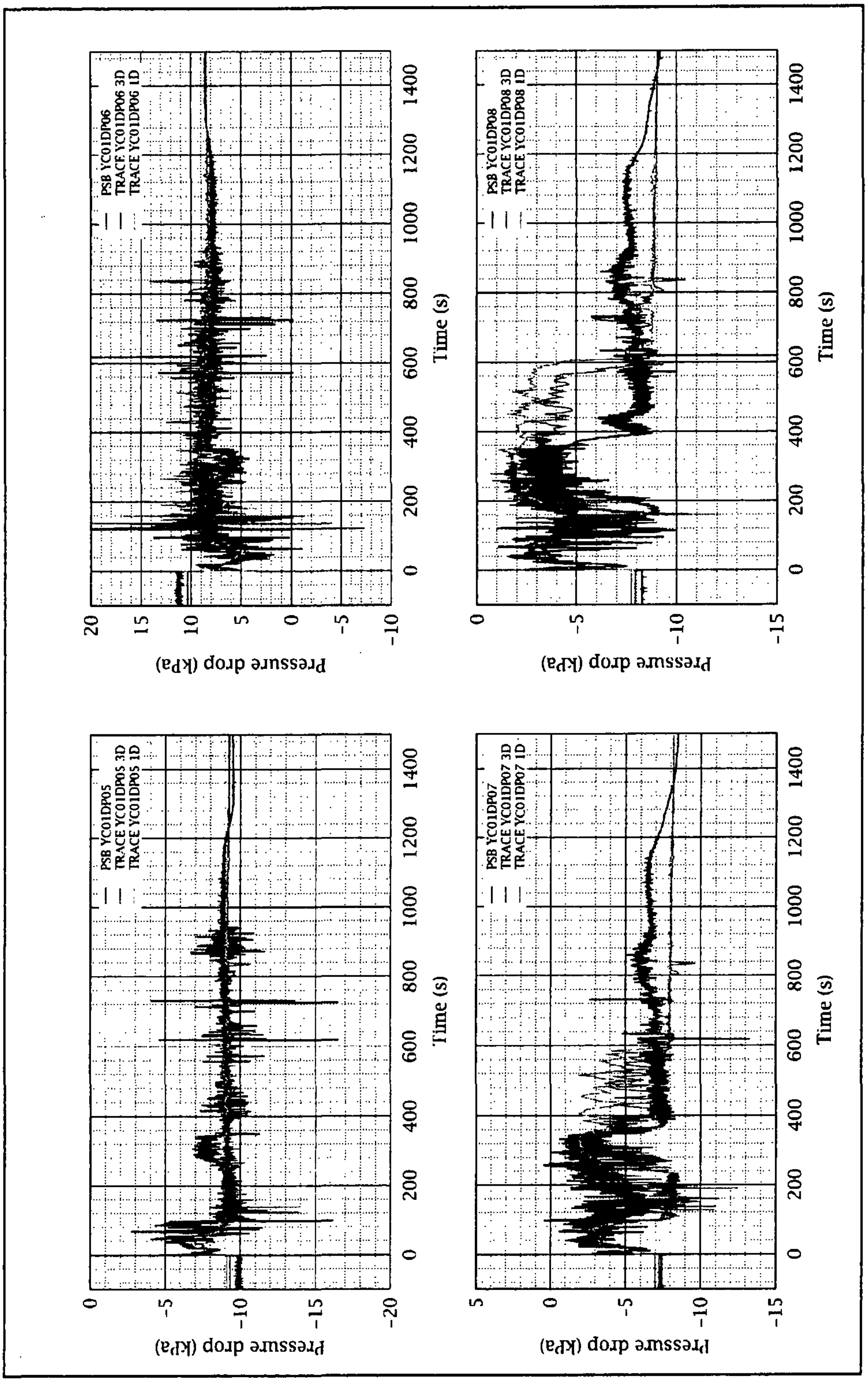

Figure 18: Pressure Differences DP05-DP08 (Lower Plenum + Lower Part of FRS)

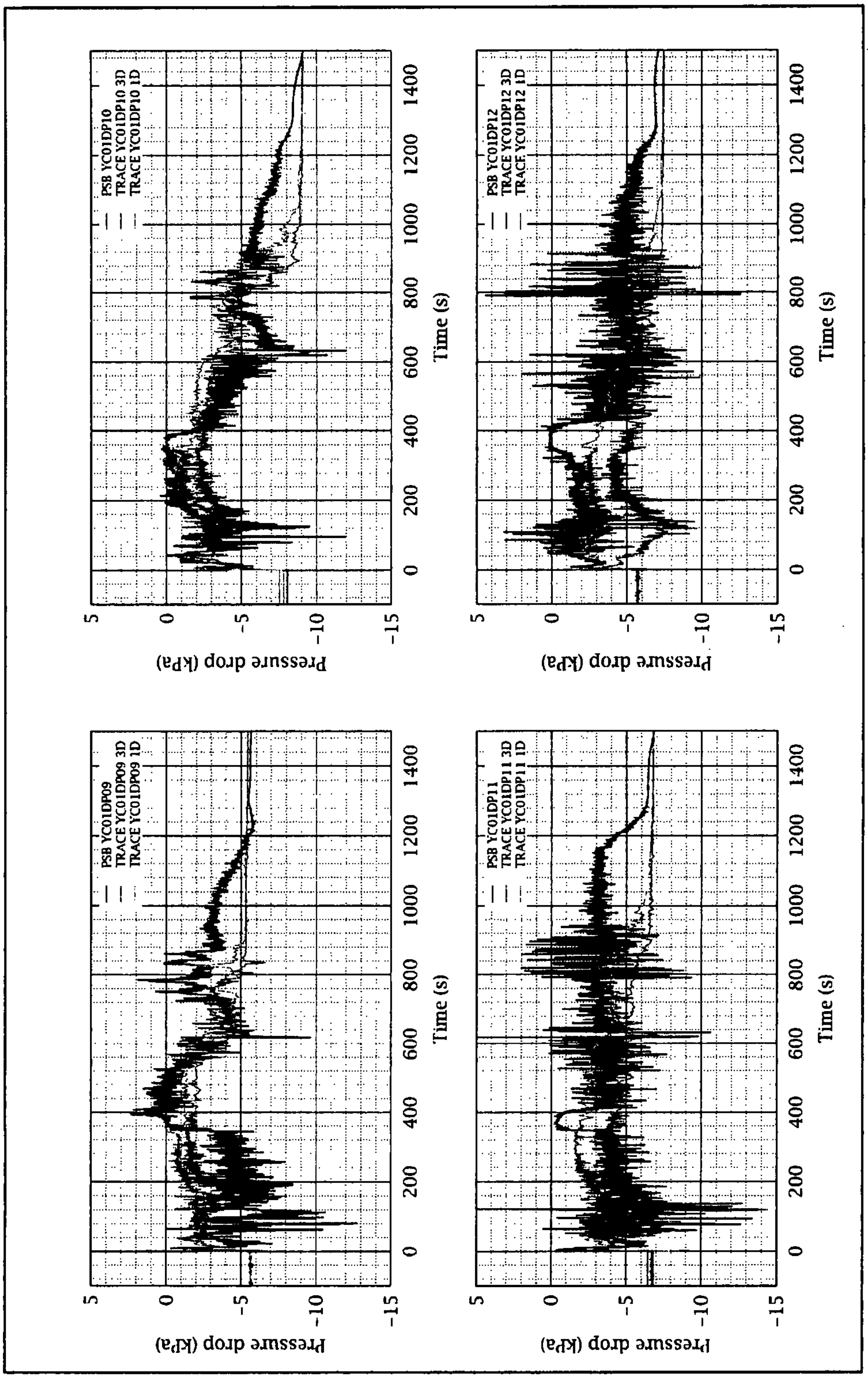

Figure 19: Pressure Differences DP09-DP12 (FRS + Lower Part of Upper Plenum)

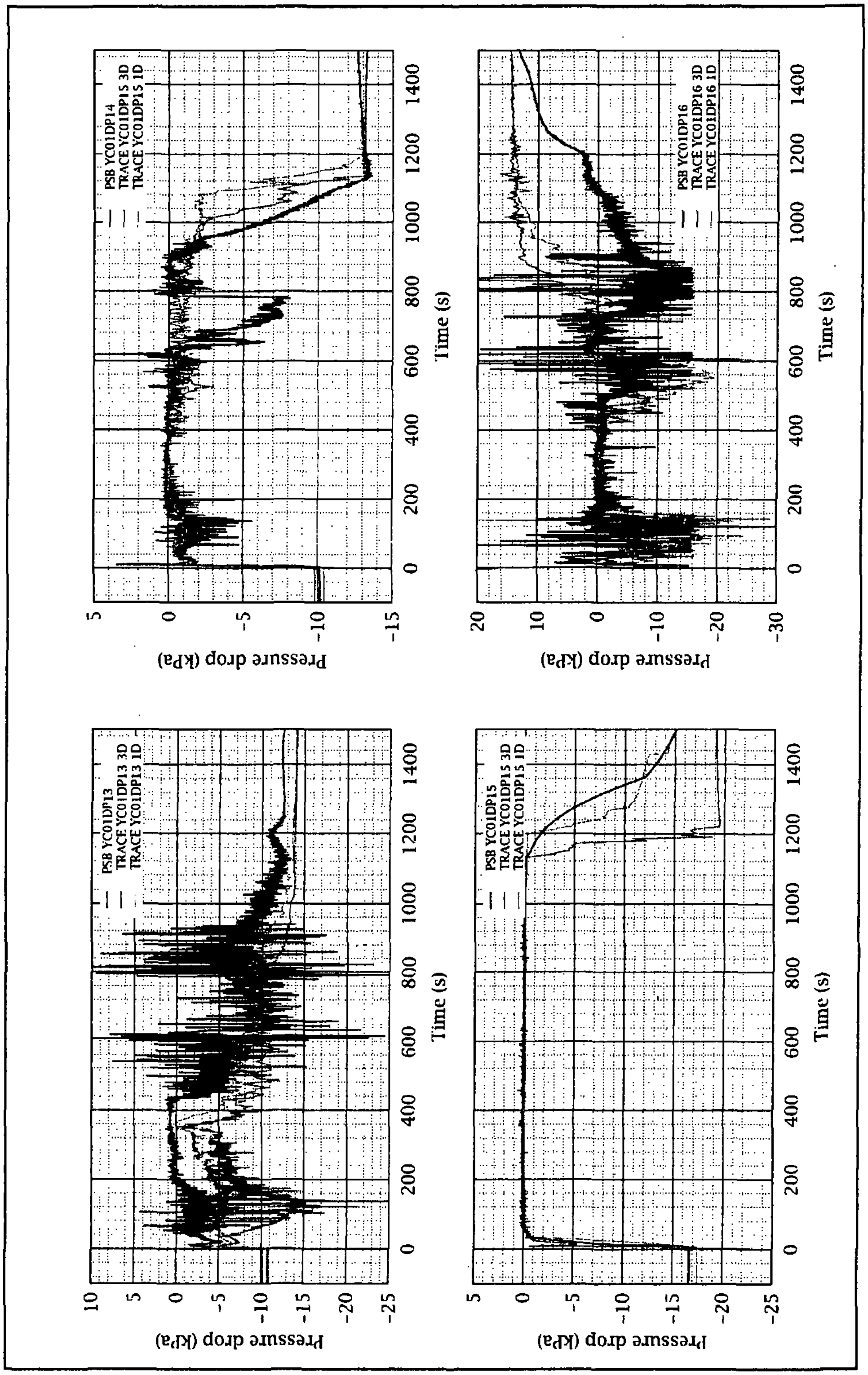

Figure 20: Pressure Differences DP13-DP16 (Upper Part of Upper Plenum)

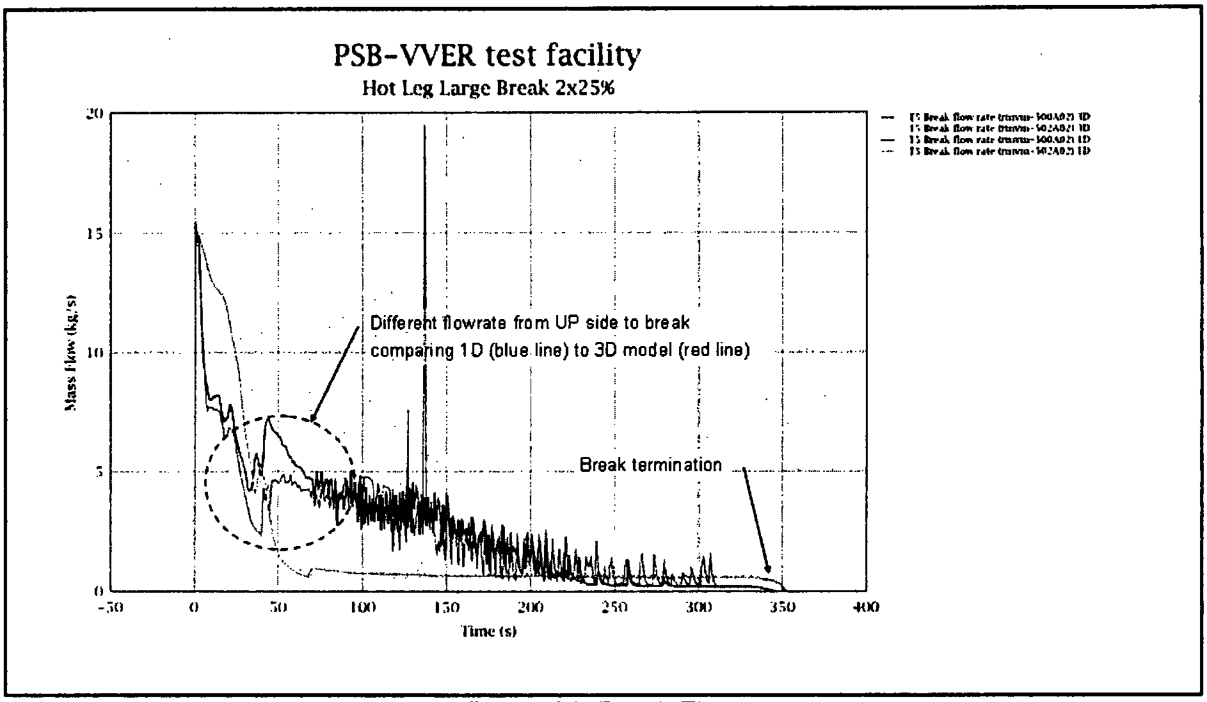

Figure 21: Break Flow

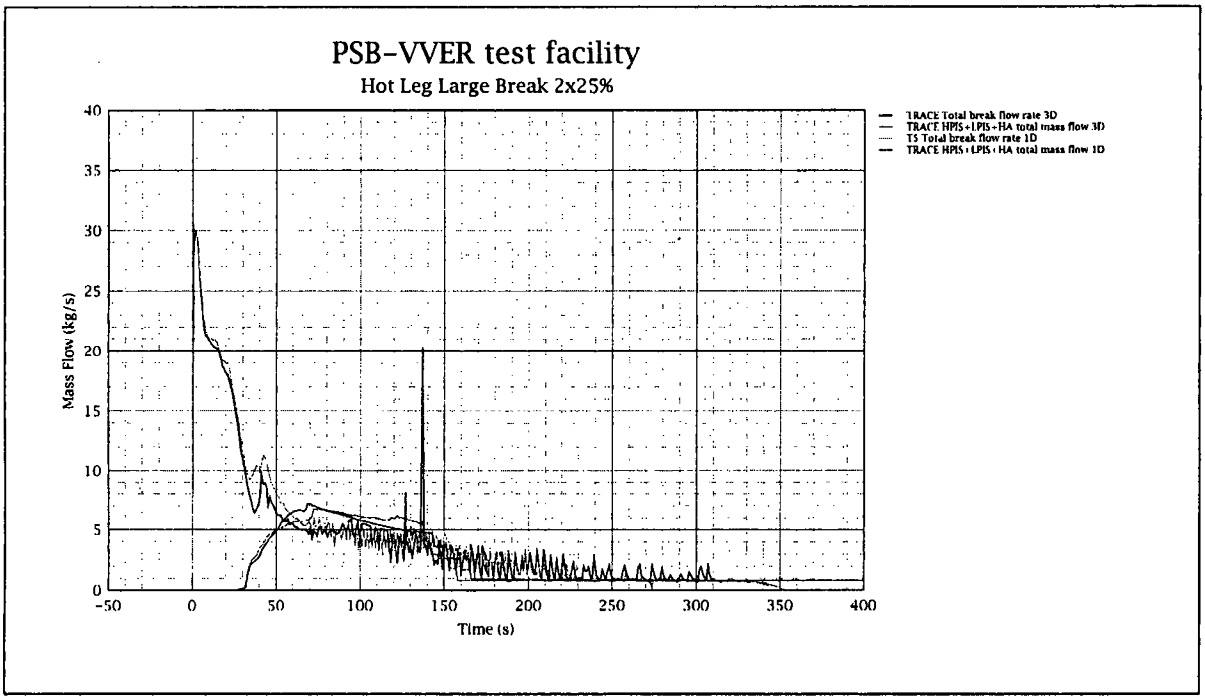

Figure 22: Primary Circuit Flow Balance

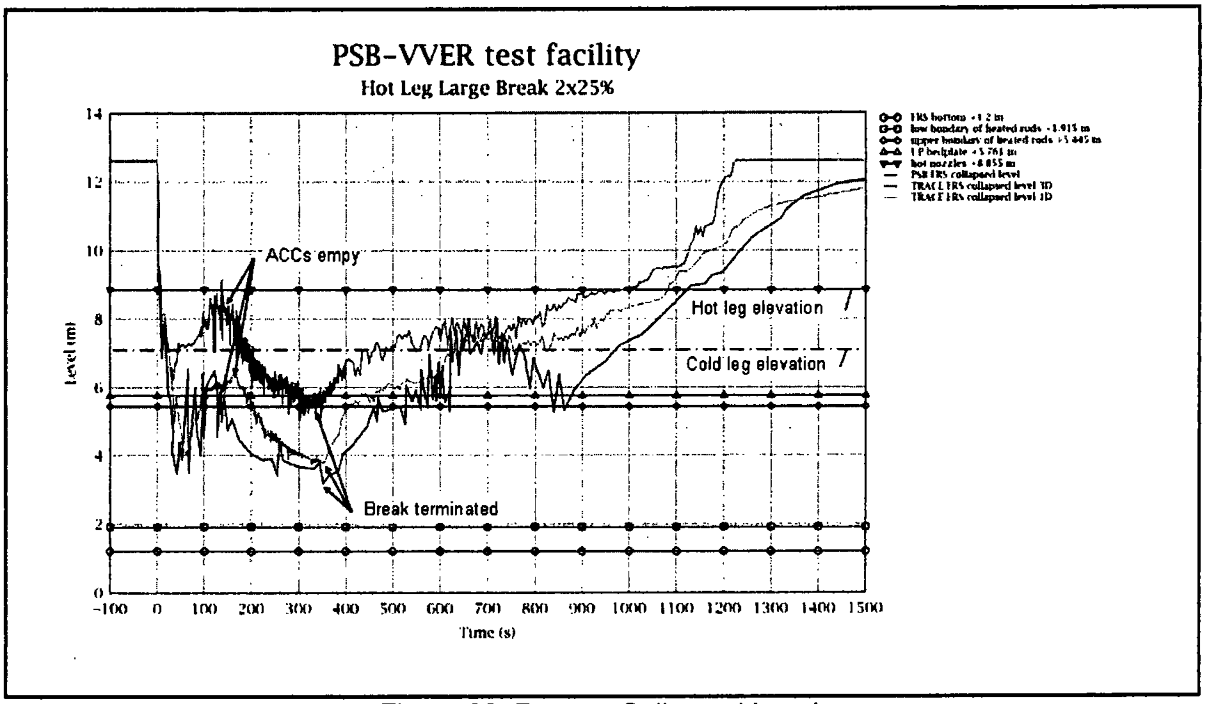

Figure 23: Reactor Collapsed Level

5.3 Nodalisation study - comparison of 1D x 3D model results

The progress of transient using 1D model was very similar to transient with 3D model, but noticeable improvement of water inventory and liquid distribution over the reactor was observed, see reactor collapsed levels at Figure 23 and pressure differences at Figures 17 – 20. That improvement was caused by higher break mass flow with 1D model - see Figure 21. Higher break mass flow had induced lower amount of liquid phase in the core (FRS) which led to better results.

The amount of break discharge was influenced by fluid temperature at break donor cell. In case "1D" model, the temperature at the break donor cell (from the UP side) was significantly lower (see Figure 24) so the break mass flow was higher as well. That temperature difference at the break was caused by different flow pattern of both models. In case of "3D" model cold water injected from ACC's to 21-st level of VESSEL component flowed through central rings of the VESSEL component downwards to the core region, where liquid was heated by hotter fluids coming from core. At the bottom of UP the heated liquid flow gradually changed its direction from central to outer rings and from downwards to upwards to the hot nozzle region where the broken LOOP3 was. Whereas in case of "1D" model of FRS such flow pattern was not observed, in fact the 1D nodalisation of FRS excludes this behavior, and cold water injected from ACC's to 21-st level of PIPE component flowed to the next lower volume and then directly to hot nozzles region, so there was nearly no heating period.

Comparison of both calculated transients shows better agreement of calculation that employs "1D" model of FRS. The comparison leads us to a conclusion that in case PSB-VVER test facility, which reactor vessel diameter is relatively small (180 mm), 3D flow pattern is not significant. Thus simpler FRS modeled with 2 PIPE components gives a reasonably good results for analysed LOCA transient. But it has to be mentioned that significantly different situation might be in case a real reactor vessel of VVER 1000, which inner diameter is approximately 4 m and 3D flow pattern could be very important.

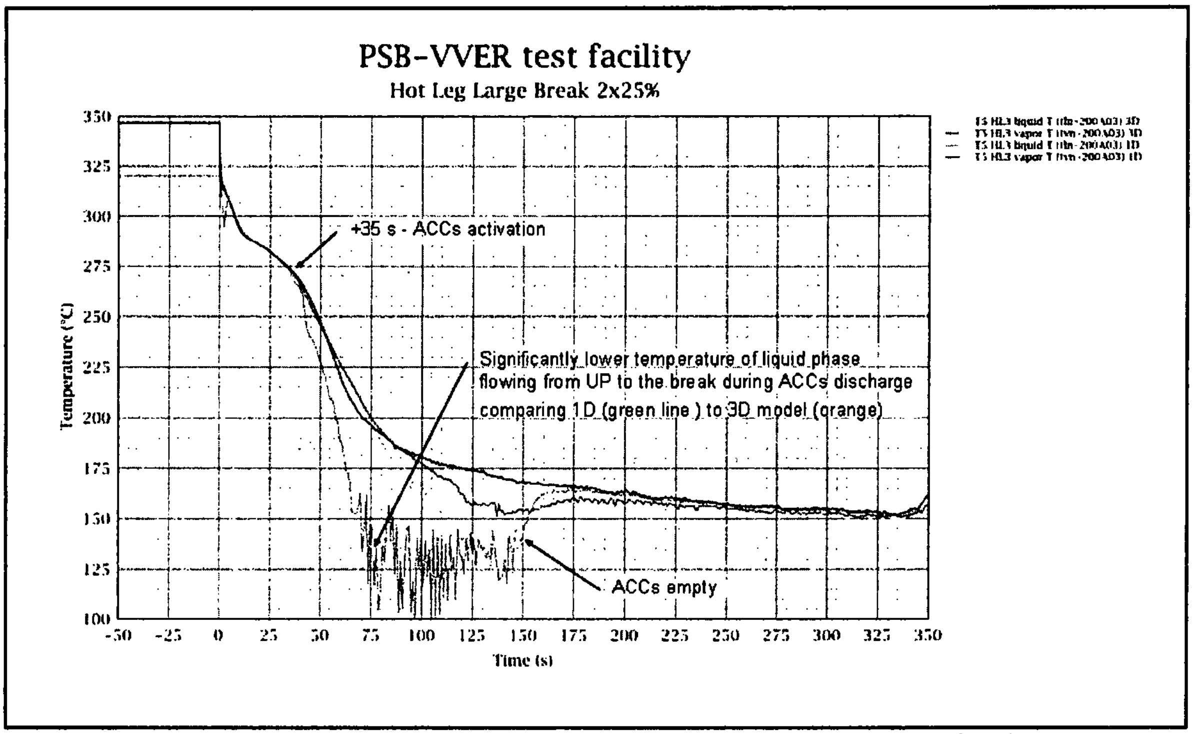

Figure 24: Temperature at the Break Donor Cell (From the UP Side)

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 Table 8. The calculations run substantially slower than real time.

Table 8: Run statistics

	1D Model	3D Model
Number of components	319	345
Number of time steps	56 626	130 832
Transient time	1 500 s	1 500 s
CPU time	8 548 s	31 219 s
CPU time / Transient time	5.7	20.8

7. CONCLUSIONS

The R&D program FI-IM5/150 "Validation of the TRACE Code for VVER Reactors" was introduced . In frame of this program was the PSB-VVER facility evaluated as the most suitable and qualified for the TRACE validation concentrated on VVER-1000 specific design and the data from test HL-2x25-2 (Hot Leg Large Break LOCA 25%) was chosen as proper basis for validation of the TRACE code, especially for reflood phase of LOCA.

Two TRACE 5.0 input decks of PSB-VVER experimental facility were introduced in this paper. The VESSEL component was applied to model fuel rod simulator (FRS) and upper plenum (UP) in the first Input Deck (labeled 3D). Two PIPE components were applied to model FRS and UP in the second Input Deck (labeled 1D). The reason for variant modelling of the PSB-VVER reactor vessel was the aspiration to discover the best approach before modelling the real VVER-1000 plant.

Comparative post-test calculations of HL-2x25-2 test were performed using both (1D and 3D) Input Decks. Relatively good results was achieved comparing primary pressure and cladding temperatures at the top of the core during first - blowdown phase (for both models).

Agreement between measured and calculated data was not satisfactory during second - reflood phase of the test. TRACE 5.0 calculations predicted faster reflood of the core (for both models). As the main reason of such behavior was identified different distribution of liquid phase over the primary circuit (especially loop region) comparing to the HL-2x25-2 test data.

Significant deviations between results from 1D and 3D model were identified analysing partial parameters. 1D Model (using PIPE component to model FRS and UP) gave better results comparing to using 3D model with VESSEL component for FRS and UP. The key reason of such behavior was investigated. We have realised, that different flowpaths of liquid phase from ACC's and LPIS injection into UP to break leads to significant difference of the temperature of the liquid phase discharging through break. Thus the integral break flow is higher using 1D model.

The comparison leads us to a conclusion that in case PSB-VVER test facility, which reactor vessel diameter is relatively small (180 mm), 3D flow pattern is not significant. Thus simpler reactor model gives a reasonably better results for analysed LOCA transient. But it has to be mentioned that significantly different situation might be in case a real reactor vessel of VVER 1000, which inner diameter is approximately 4 m and 3D flow pattern could be very important.

Although a lot of work had been done in frame of this project, further investigation is necessary to asses the TRACE code application for VVER-1000 design. We are ready to recalculate HL-2x25-2 test after issuing proposed TRACE Code upgrades (improved spacer grid and level tracking models). Moreover, we are finishing similar work using darta from UP-11-07 test (Upper Plenum 11% Small Break LOCA) from PSB-VVER experimental facility in these days. We are ready to submit results in frame of CAMP the same way as HL-2x25-2 test results.

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APPENDIX A

INPUT DECK NODALISATION SCHEMES

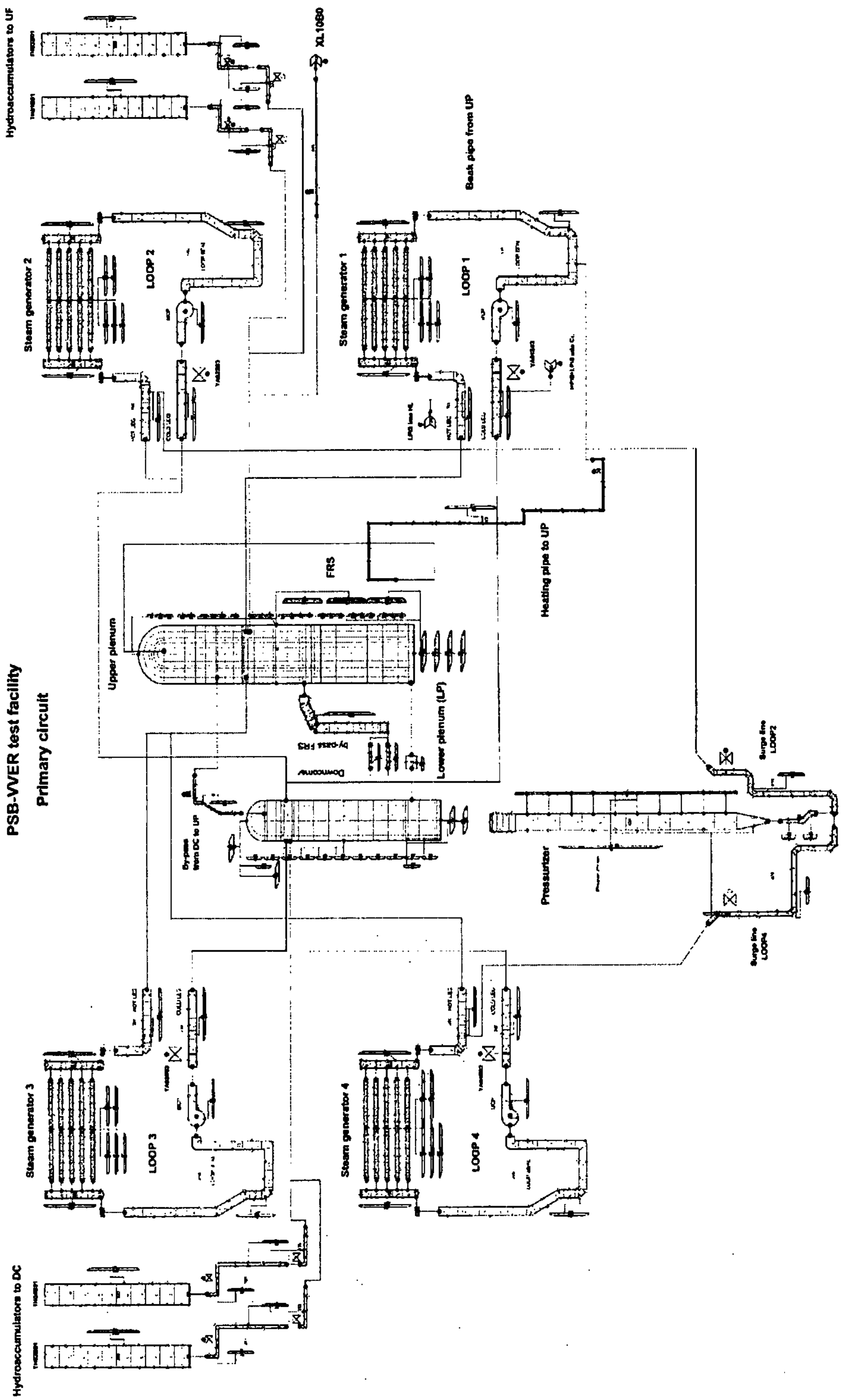

Figure A-1 Nodalisation Scheme of Primary Circuit of PSB-VVER Facility (3D Model)

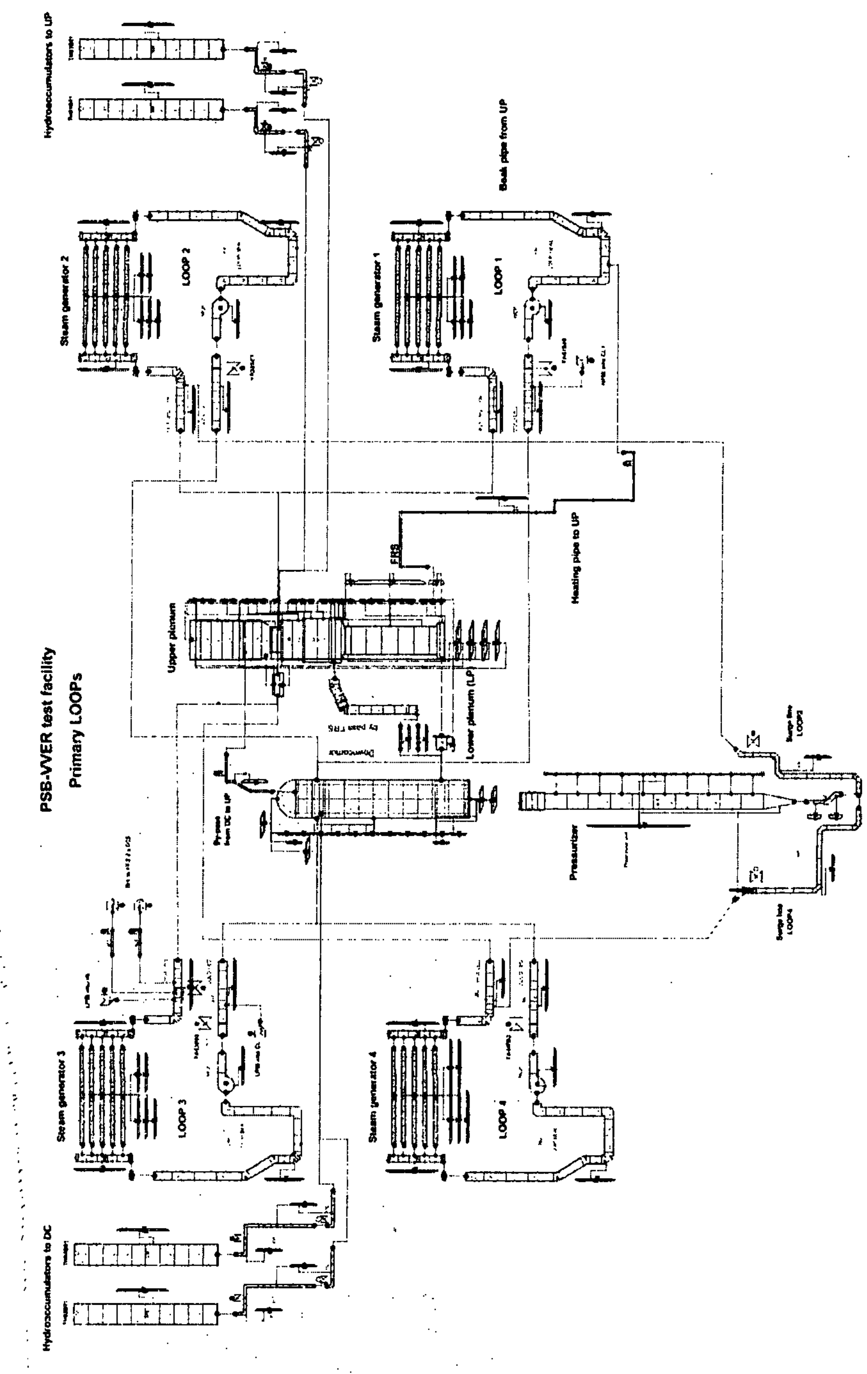

Figure A-2 Nodalisation Scheme of Primary Circuit of PSB-VVER Facility (1D Model)

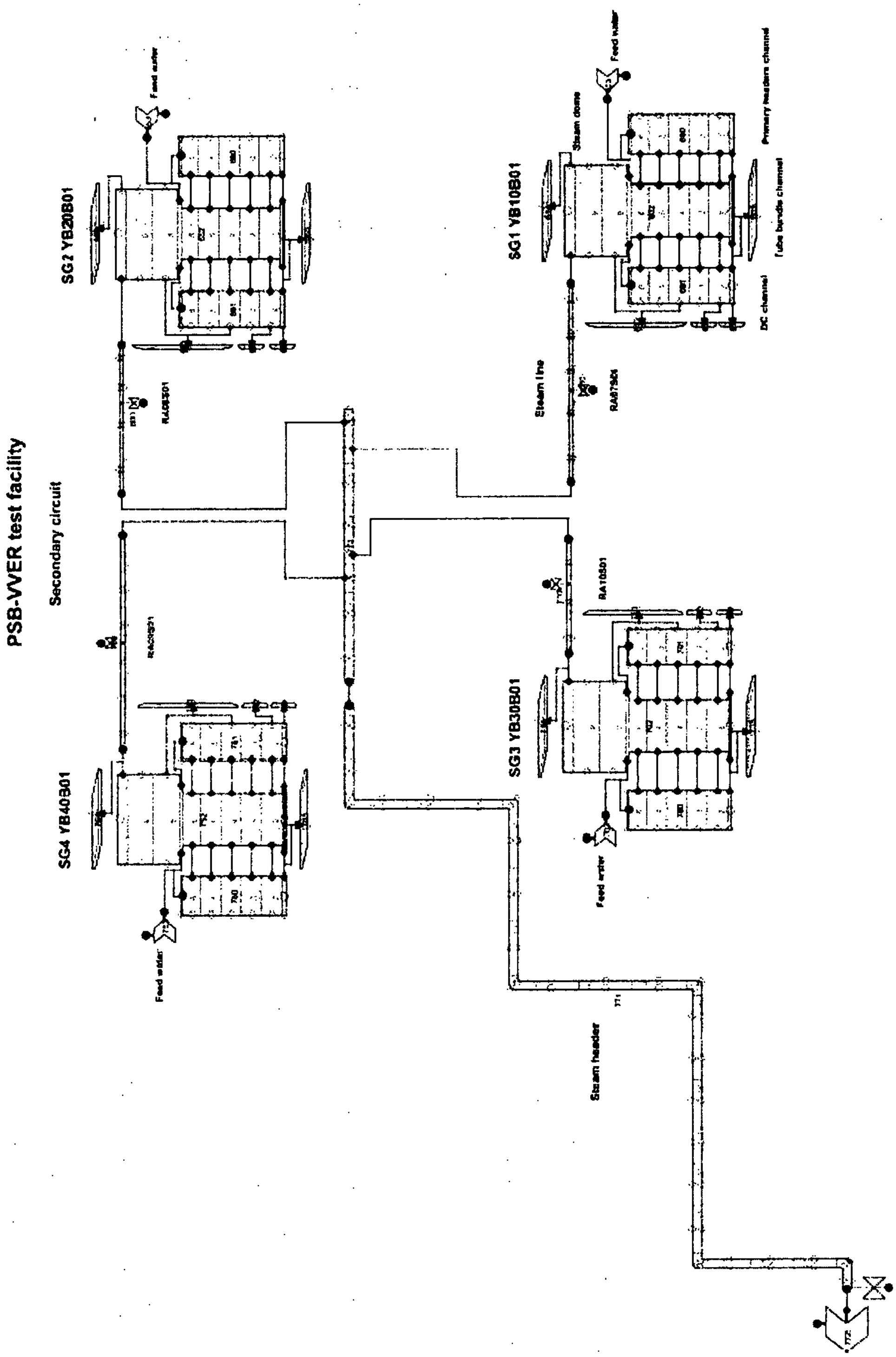

Figure A-3 Nodalisation Scheme of Secondary Circuit of PSB-VVER Facility (for both models)

APPENDIX B

MEASUREMENT LOCALISATION AT PSB-VVER FACILITY

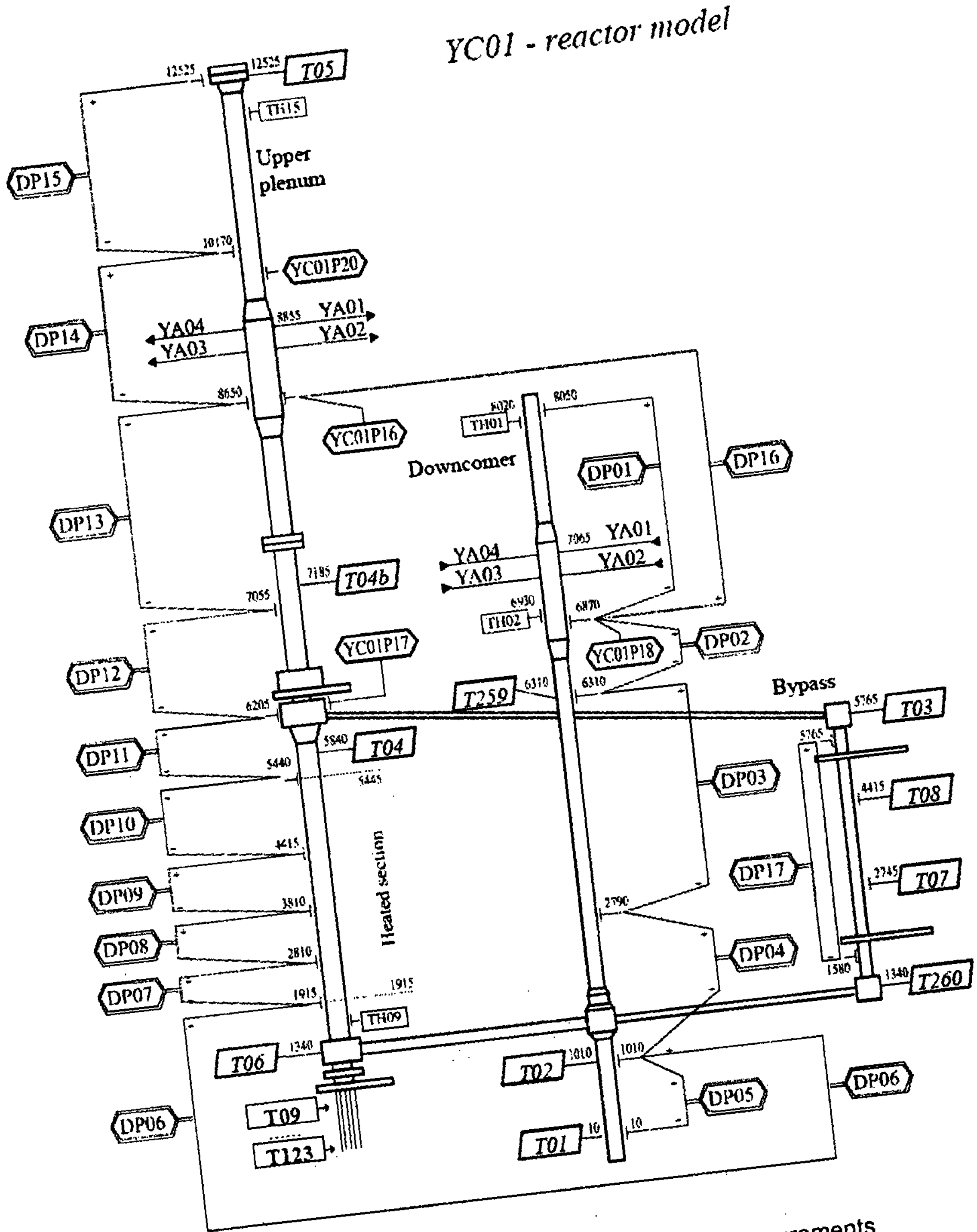

Figure B-1 PSB-VVER Reactor Model Measurements

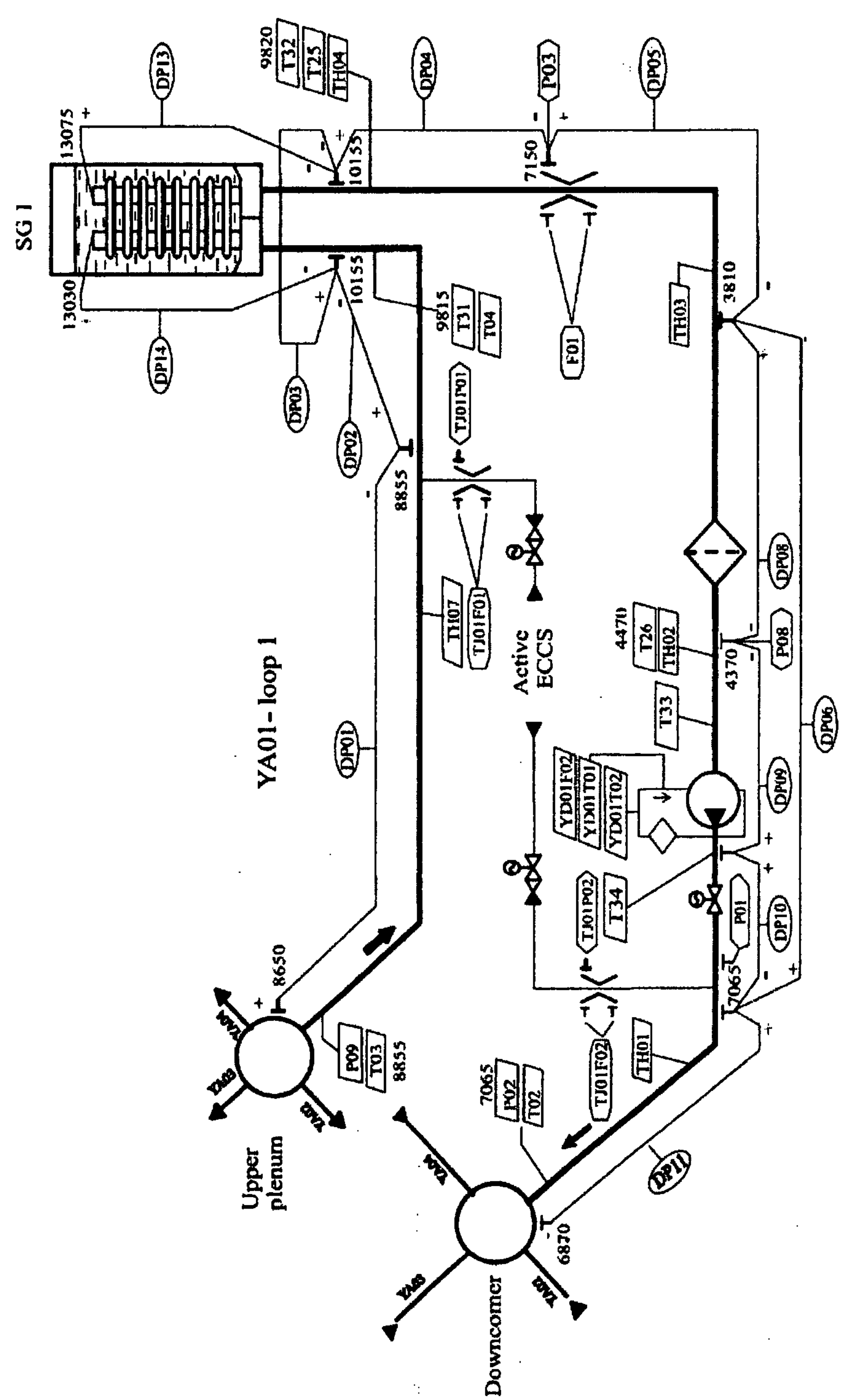

Figure B-2 PSB-VVER Loop 1 and SG-1 Model Measurement

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10. SUPPLEMENTARY NOTES
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11. ABSTRACT *(200 words or less)*
The best estimate thermo-hydraulic computer code TRACE V5.0 has been assessed using Hot Leg 2x25% break experiment at the large-scale test facility PSB VVER. The PSB-VVER facility is a 1:3000 volume scaled model of VVER 1000 NPP located in Electrogorsk, Russia. 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 unit, main circulation pumps, steam generators, break section 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 code 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 required by regulators. The purpose of performed analysis is to extend the validation of the TRACE code focused on VVER type of NPPs. This work was performed in the frame of R&D project sponsored by The Ministry of Industry and Trade of Czech Republic.

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