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International Agreement Report

TRACE VVER-1000/V-320 Model Validation

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ABSTRACT

This report is developed by the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) and its technical support organization, the State Scientific and Technical Center for Nuclear and Radiation Safety of Ukraine (SSTC NRS), under Implementing Agreement On Thermal-Hydraulic Code Applications And Maintenance Between The United States Nuclear Regulatory Commission and State Nuclear Regulatory Inspectorate of Ukraine (signed in 2014) in accordance with Article III, Section C, of the Agreement.

The report provides results of the validation calculations conducted with application of SSTC NRS model of VVER-1000/V-320 unit for TRACE computer code. The calculation scenarios simulate actual incidents that occurred at Ukrainian NPPs.

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ABBREVIATIONS

AFW	Auxiliary Feedwater System
ARM	Reactor Power Controller, Russian designation
BRU-A	Steam Dump Valve to Atmosphere
BRU-K	Turbine Bypass to Condenser
BRU-SN	House Loads Steam Supply Valve
CAMP	Code Maintenance and Assessment Program
CVCS	Chemical and Volume Control System
ECCS	Emergency Core Cooling System
EFW	Emergency Feedwater System
FASIV	Fast-acting Steam Isolation Valve
HA	Hydroaccumulators
HC	Hydraulic Components
HPIS	High Pressure Injection System
HS	Heat Structure
LOCA	Loss of Coolant Accident
LPIS	Low Pressure Injection System
MFW	Main Feedwater System
MSH	Main Steam Header
MSL	Main Steam Line
NPP	Nuclear Power Plant
PORV	Pilot Operated Relief Valve
PRZ	Pressurizer
PTU	Protective Tubes Unit
PZ-1	Level 1 Reactor Preventive Protection
PZ-2	Level 2 Reactor Preventive Protection
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RNPP	Rivne Nuclear Power Plant
SG	Steam Generator
SNRIU	State Nuclear Regulatory Inspectorate of Ukraine
SPP	Submerged Perforated Plate
SRV	Safety Relief Valve
SSTC NRS	State Scientific and Technical Center for Nuclear and Radiation Safety

UPZ	Fast Reactor Load Shedding Protection, Russian designation
U.S. NRC	United States Nuclear Regulatory Commission
VVER	Pressurized Water Reactor, Russian design
ZNPP	Zaporizhzhya Nuclear Power Plant

1 INTRODUCTION

At the end of 2014 the United States Nuclear Regulatory Commission (USNRC) and the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) signed Implementing Agreement On Thermal-Hydraulic Code Applications And Maintenance (CAMP). In accordance with Article III, Section C, of the Agreement, SNRIU shall submit to the USNRC the in-kind contribution reports providing the code assessment results or other activities results of equivalent value.

In the framework of the Agreement SNRIU and SSTC NRS obtained the state-of the-art TRACE code which provides advanced capabilities for modeling thermal-hydraulic processes and components, control systems and allows coupling with PARCS neutron kinetics code. In 2015 SSTC NRS initiated activities on TRACE code application for evaluation of the results of safety assessments performed for Ukrainian NPPs. The existing SNRIU/SSTC NRS RELAP5 models for VVER-440 and VVER-1000 were converted to TRACE code.

In order to justify capabilities of VVER-1000 model for TRACE code to simulate adequately the plant response during transients and accidents, the validation of the model was performed. This report provides description of validation results.

Section 2 of the report briefly describes main primary and secondary systems of Zaporizhzhya NPP (ZNPP) unit 5 (of VVER-1000/V-320 design) which are important for development of thermal-hydraulic model. Description of ZNPP Unit 5 model for TRACE code is provided in Section 3. The results of TRACE calculations for several scenarios simulating actual incidents that occurred at Ukrainian NPPs are provided in Section 4 of the report.

The incidents simulated include:

- MFW-1 pump trip;
- inadvertent FASIV closure;
- PRZ PORV stuck open during tests.

For each validation calculation the following information is provided:

- brief description of the incident;
- initial and boundary conditions selected for incident simulation;
- sequence of events;
- description of calculation results;
- plots of the main primary and secondary circuit parameters (including calculated and measured data).

2 MAIN FEATURES OF VVER-1000/V-320 DESIGN

2.1 VVER-1000/V-320 Reactor System

VVER-1000/V-320 is 1000 MWe nuclear power plant with pressurized water reactor designed in former Soviet Union. The primary circuit of VVER-1000/V-320 unit consists of reactor and four identical primary circuit loops of Dn 850 mm with horizontal steam generators (SG) PGV-1000M and reactor coolant pumps (RCP) of GCN-195M type. Main design characteristics of VVER-1000/V-320 and general layout of the reactor coolant system (based on Zaporizhzhia NPP Unit 5 data [1]) are provided in Table 2-1 and Figures 2-1, 2-2. Principal diagram of VVER-1000/V-320 reactor coolant system is shown in 2-3.

Table 2-1 Main characteristics of VVER-1000/V-320

Parameter	Value
Reactor thermal power, MWt	3000
Reactor pressure, abs., kgf/cm^2 (MPa)	160±3 (15.7±0.3)
Reactor outlet temperature, °C	320
Coolant heat-up, °C	30.3
Reactor coolant flow, m^3/h	84800+4000-4800
SG steam mass flow, t/h	5880
SG pressure, abs., kgf/cm^2 (MPa)	64±2 (6.28±0.2)
Steam temperature, °C	278.5±2
Feedwater temperature with/without heaters, °C	220±5 164±4
Pressurizer level, mm	8770±150
SG level, mm	2550±50
Grid frequency, Hz	50+0.5-1.0

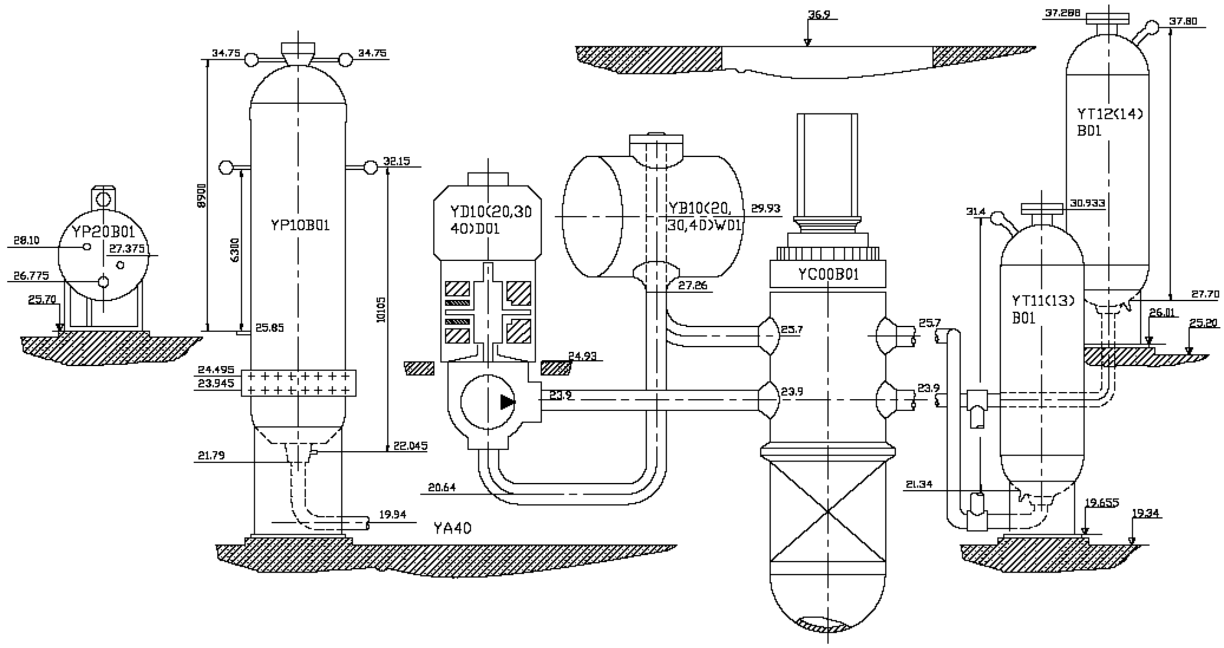

Figure 2-1 VVER-1000/V-320 Reactor System Layout

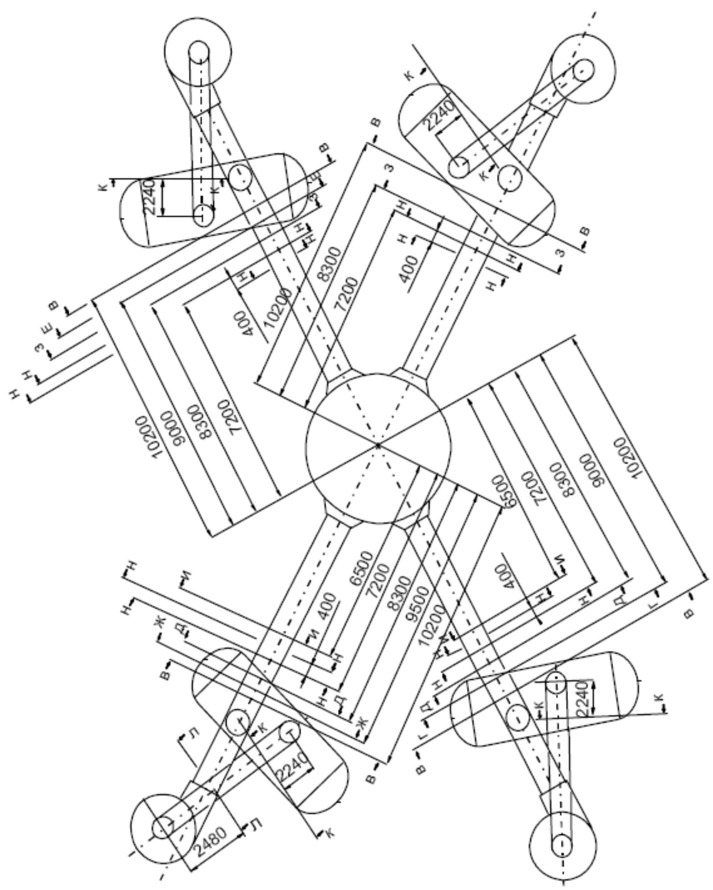

Figure 2-2 VVER-1000/V-320 Reactor System Primary Loops (Top View)

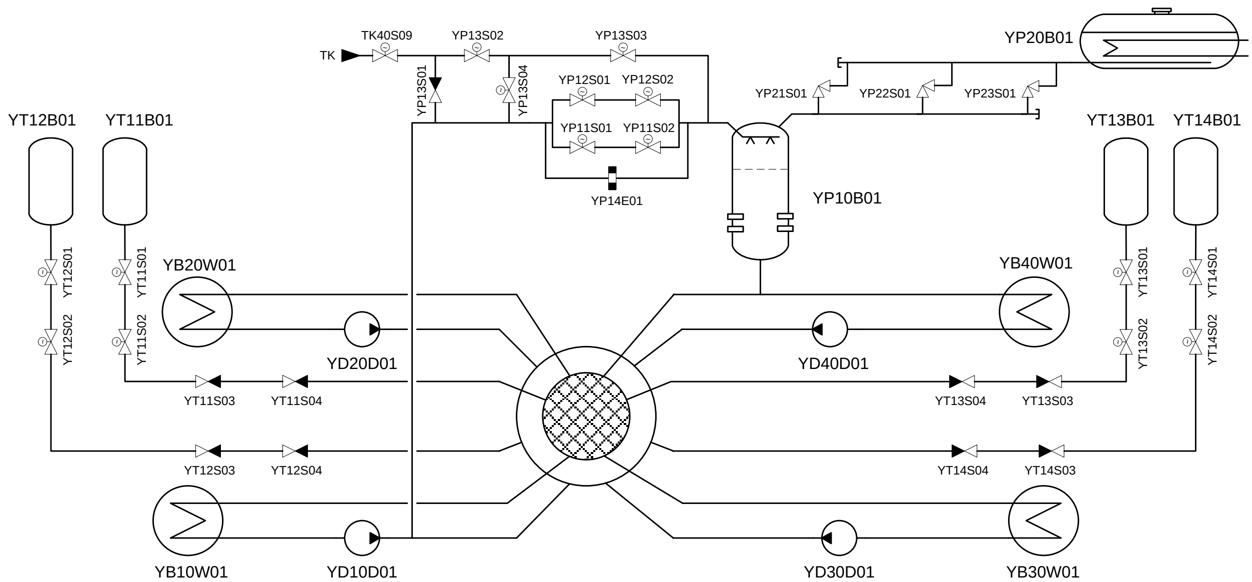

Figure 2-3 Principal Diagram of VVER-1000/V-320 Reactor Coolant System

VVER-1000 reactor consists of the following elements (Figure 2-4, left):

- reactor pressure vessel (pos.1);
- separation ring (flow divider) (pos.2);
- reactor upper head (pos.3);
- core barrel including perforated cylindrical part (pos.4), perforated elliptical bottom (pos.5), grid plate (pos.6) and support tubes (pos.7);
- core baffle (pos.8);
- protective tubes unit (PTU) including upper, medium and bottom (support) plates (pos.9, 10, 11), bulge (pos.12), perforated cylindrical shell (pos.13), I&C shroud (pos.14), I&C and control rods protective tubes (pos.15, 16), thermal and neutron flux detectors shroud (pos.17, 18), control rods shaft (pos.20) and its protective tube (pos.19);
- reactor core with fuel assemblies (pos.23).

Coolant flow paths in the reactor core are shown in Figure 2-4, right. From the reactor inlet nozzles the coolant flows down in the annular gap between reactor vessel and core barrel, and passes through the perforated elliptical bottom of the core barrel. Most of the coolant passes through the orifices in core support tubes and enters the fuel assemblies. Small amount of coolant flows through channels in the core baffle, and in the gap between core baffle and barrel. Heated in the reactor core coolant through the orifices in the lower plate of protective tubes unit and in the protective tubes enters tube space of PTU, and then through the perforation in upper part of the core barrel and reactor outlet nozzles exits the reactor.

The reactor core consists of 163 hexagonal fuel assemblies, 61 of which contain movable control rods. The fuel assembly is composed of fuel rods bundle (312 fuel rods), control rods guide tubes, central tube, top and bottom nozzles. The fuel rods are arranged in a triangular grid and separated by honeycomb-type spacers.

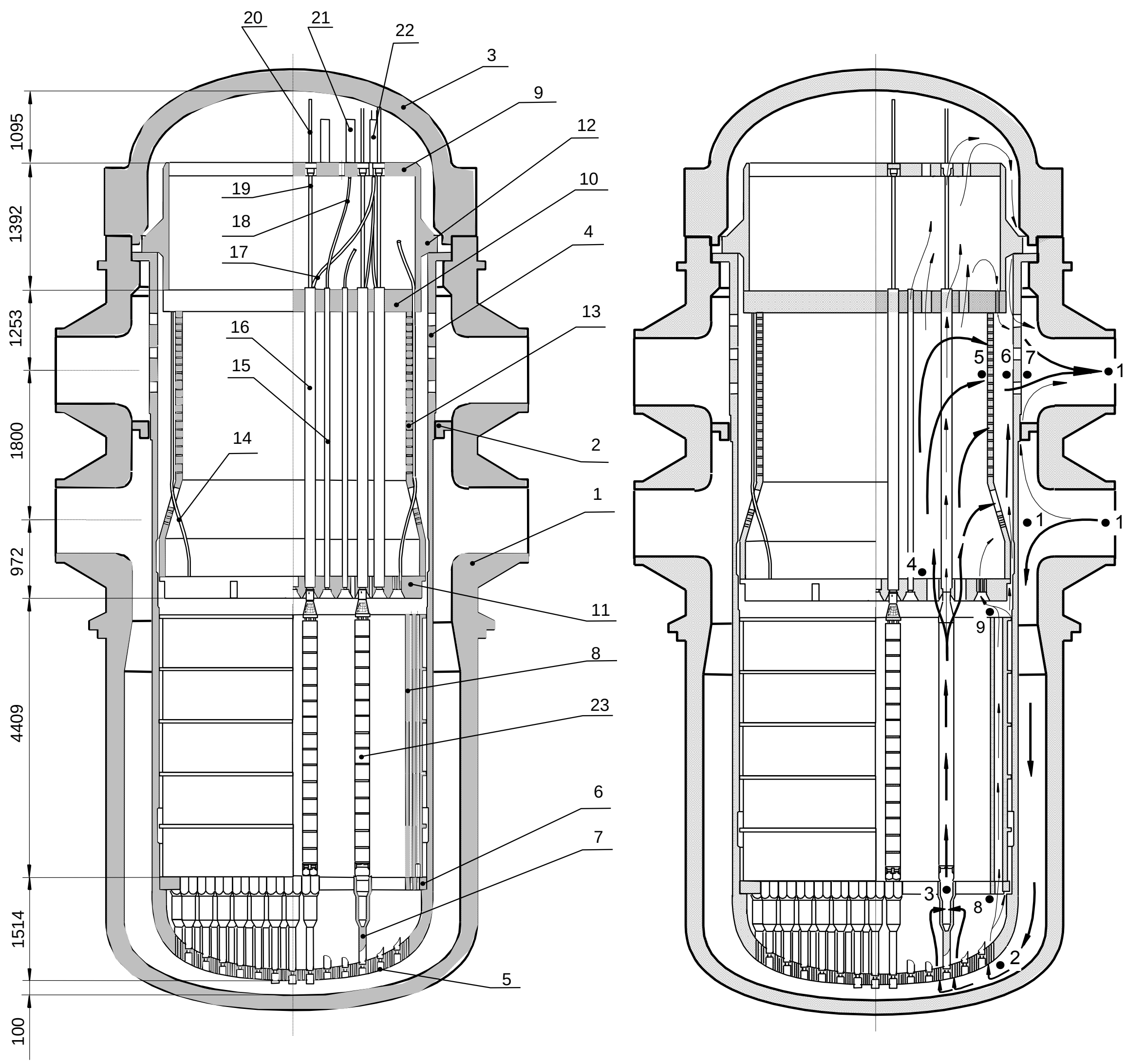

Figure 2-4 VVER-1000 Reactor and Coolant Flow Paths

The main circulating pipeline (Figures 2-1, 2-2) is intended to connect the equipment of the primary circuit and categorized as a normal operation system. Each of four circulating loops consists of hot and cold legs Dn 850 mm. The hot leg connects the reactor outlet nozzle with the steam generator inlet collector. The cold leg connects the SG outlet collector with the reactor coolant pump, and RCP with the reactor inlet nozzle.

Circulation of the coolant through the primary circuit is provided by operation of reactor coolant pumps GCN-195M of a centrifugal type with mechanical shaft sealing. A special flywheel is installed on the pump shaft to provide the desired flow pump coast-down.

PGV-1000 is horizontal steam generator with submerged bundle of horizontal U-shaped tubes and is designed to remove heat from the primary circuit and to produce dry saturated steam. Longitudinal and cross-section view of PGV-213 is provided on Figures 2-5 and 2-6.

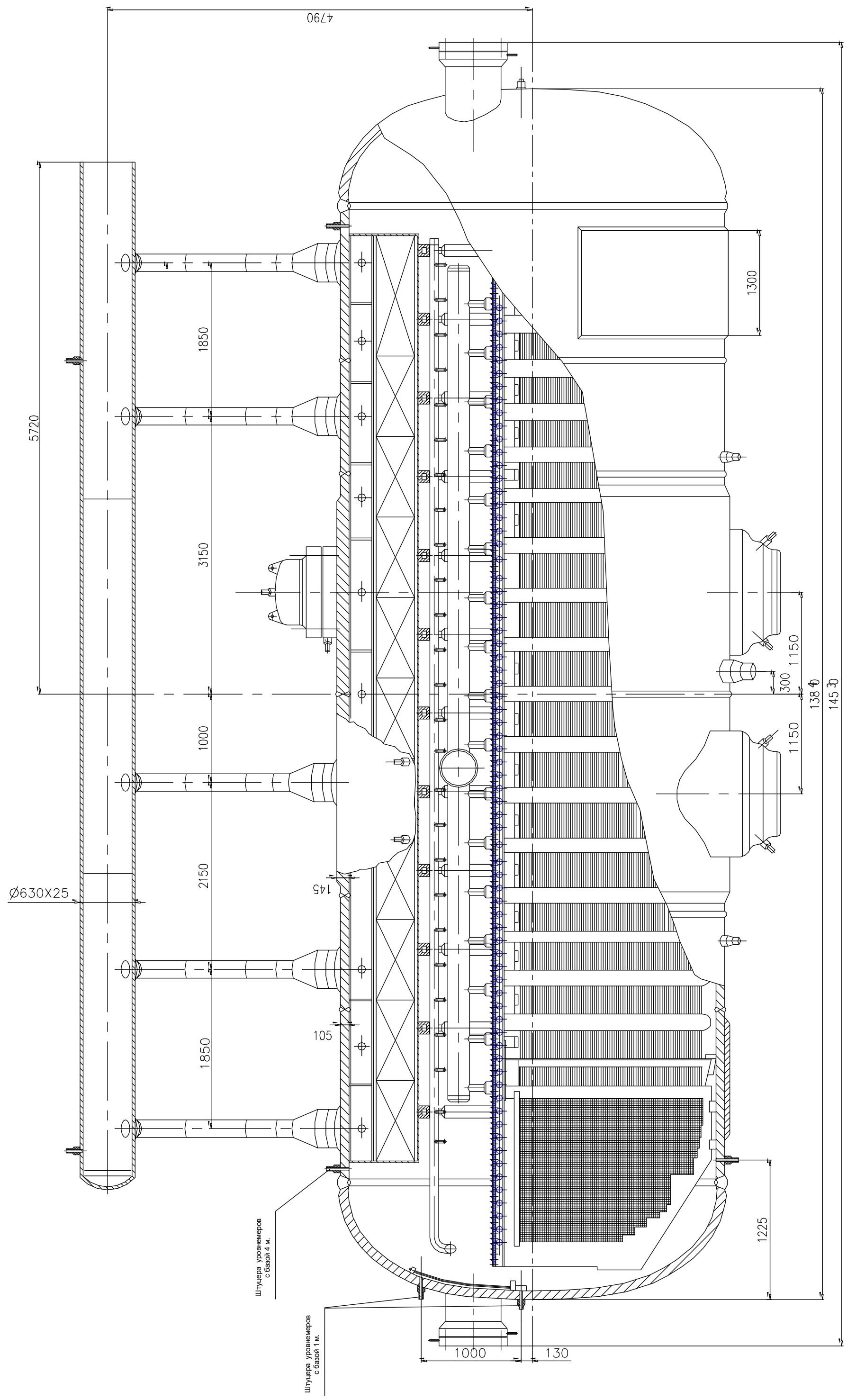

Figure 2-5 PGV-1000 Longitudinal Section

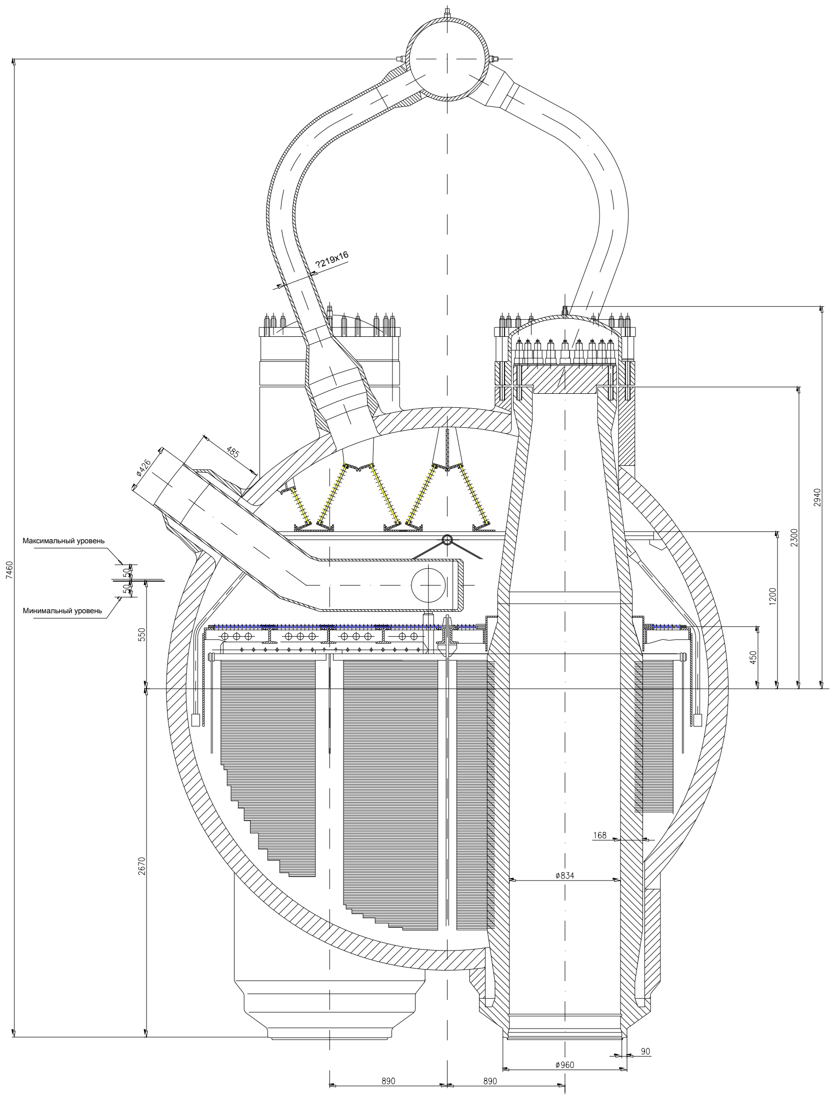

Figure 2-6 PGV-1000 Cross Section

PGV-1000 SG consists of the following components:

- SG vessel;
- two vertical primary circuit collectors with collector covers;
- U-shaped horizontal heat exchange tube bundle with support structures;
- steam pipes and manifold;
- main feedwater manifold and distribution pipeline;
- emergency feedwater manifold and distribution pipeline;
- steam separators;
- submerged perforated plate.

Heat exchange tubes are assembled into two U-shaped tube bundles. The vertical and horizontal step between tubes is 19 mm and 23 mm, respectively. Each bundle has three vertical corridors ensuring controlled hydrodynamics of circulating water.

During normal power operation, startup and reactor cooling down the primary circuit pressure is maintained by primary pressure control system which is categorized as a normal operation system and comprises of pressurizer YP10B01 (Figure 2-7), relief tank YP20B01, pipelines and valves. Pressurizer (PRZ) is connected to the hot leg of loop no.4 with surge line (Figure 2-3).

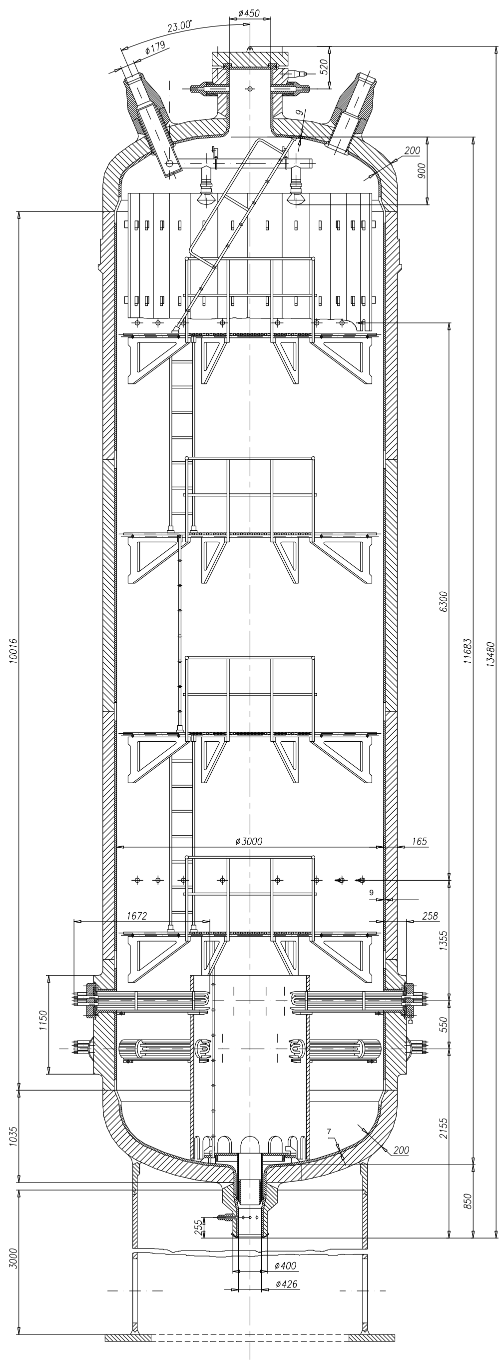

Figure 2-7 Pressurizer

To maintain prescribed reactor coolant system (RCS) pressure the pressurizer is equipped with a spray unit installed at the top of PRZ to inject cold RCS coolant to the steam volume of PRZ and electrical heaters located at bottom part of PRZ. The PRZ spray unit is connected to the cold leg of loop no.1 (downstream RCP) by the spray pipeline. The spray flow is controlled by opening and closure of valves YP11S02, YP12S02 (Figure 2-3). Control valve YP13S02 is used to maintain coolant temperature difference between PRZ and hot legs during reactor start-up or shut-down.

2.2 Technological and Safety Systems Connected to the Primary Circuit

Main technological and safety systems of the primary circuit are:

- chemical and volume control system;
- RCS overpressure protection system;
- emergency gas evacuation system;
- emergency core cooling system.

2.2.1 Chemical and Volume Control System

The chemical and volume control system (CVCS) is intended to control the volume, purity and boric acid content of the reactor coolant during normal operational conditions and transients, including startup, shutdown and changes of the reactor power level. During normal operation the CVCS compensates uncontrolled and controlled leaks from the RCS. Adjustments in coolant volume are made automatically to maintain a predetermined level in the pressurizer.

The design functions of chemical and volume control system are to:

- supply sealing water to RCP seals;
- return purified let-down water to the primary circuit;
- compensate uncontrolled primary coolant leaks;
- control reactivity of the reactor by variation of boron acid concentration;
- provide required chemical content of the primary coolant;
- control primary coolant inventory during loss of coolant accidents (LOCA) compensated by CVCS;
- control parameters of primary circuit during start-up and shutdown.

The system is categorized as a normal operation system and consists of the following subsystems:

- primary coolant deaerating subsystem;
- primary make-up subsystem;
- RCP sealing water subsystem;

- demineralized water supply subsystem;
- primary coolant let-down subsystem.

2.2.2 RCS Overpressure Protection System

RCS overpressure protection system is categorized as a safety system and designed to protect RCS pipelines and equipment from a pressure increase above the design limits.

The system consists of one control and two main pilot operated relief valves (PORV) YP21S01, YP22S01, YP23S01 (Figure 2-8) which open automatically at the preset RCS pressure and dump coolant to the relief tank YP20B01 [2]. Steam from PRZ released by PORV is condensed in the relief tank, and the condensate is pumped out for maintaining nominal level in the tank. If the pressure in the relief tank rises above the safety limit, a rupture disk which is installed at the top of relief tank breaks and coolant is released to the containment.

PORV can be controlled manually to depressurize primary circuit and implement primary feed-and-bleed procedure.

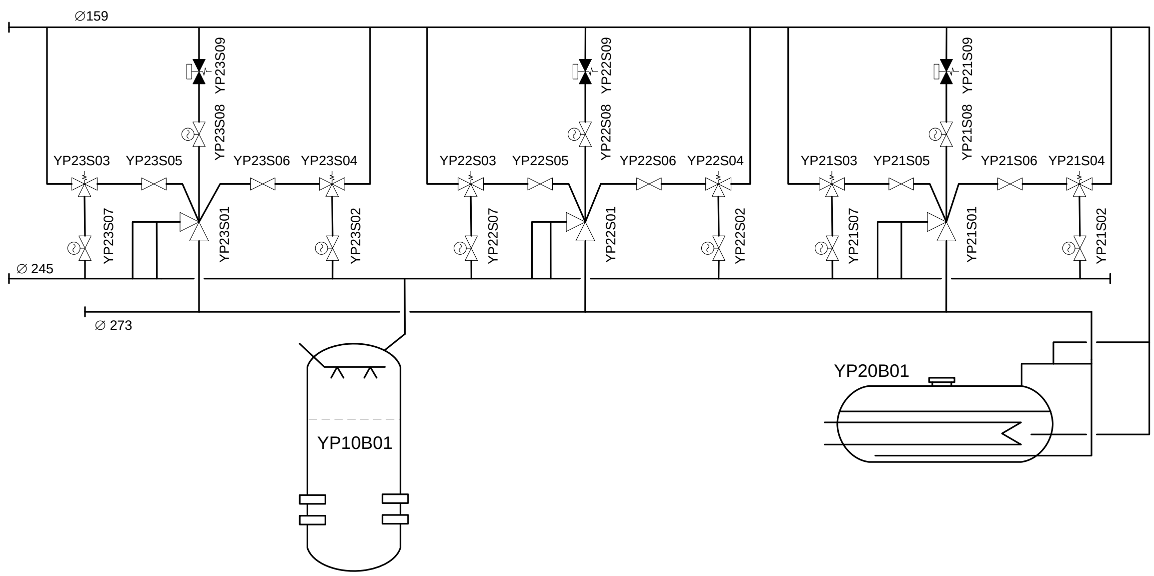

Figure 2-8 Principal Diagram of the Primary Pressure Control System

2.2.3 Emergency Gas Evacuation System

Emergency gas evacuation system is categorized as a safety system and designed to remove steam-gas mixture from the top points of the primary circuit (from the reactor upper head, steam generator collectors and PRZ) in accidents leading to the reactor core uncovery and steam-zirconium oxidation.

Emergency gas evacuation system consists of:

- pipelines from the reactor;
- pipelines from the SG collectors;

- pipelines connected to the pressurizer relief line;
- valves.

The system is operated manually from the control room. Principal diagram of the system is shown in Figure 2-9.

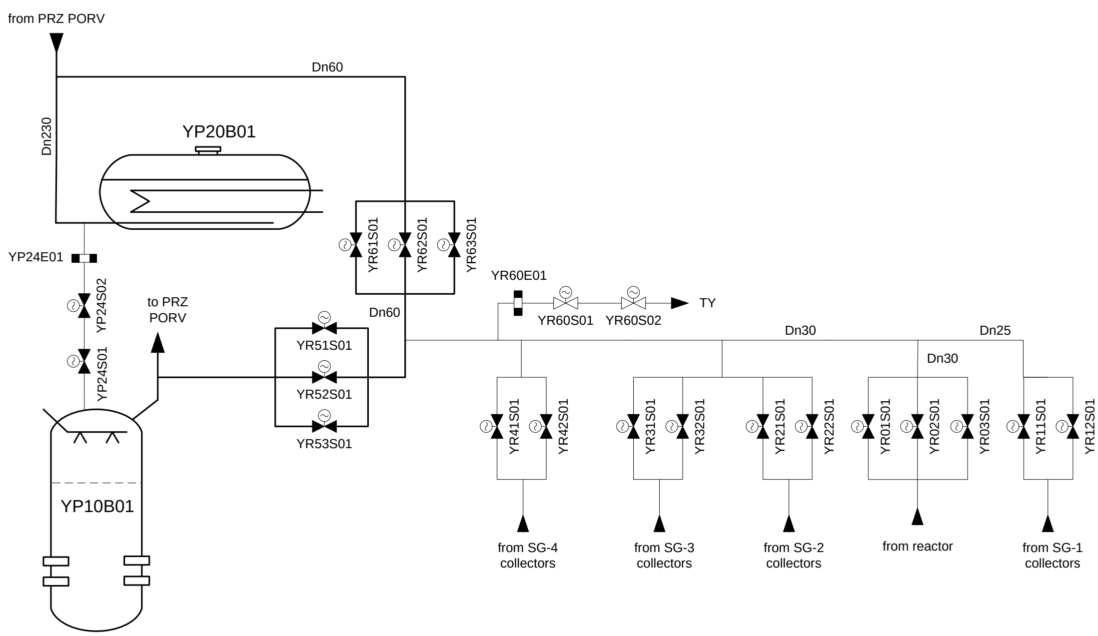

Figure 2-9 Principal Diagram of Emergency Gas Evacuation System

2.2.4 Emergency Core Cooling System

Emergency core cooling system (ECCS) includes:

- 4 hydroaccumulators (HA);
- high pressure injection system (HPIS);
- low pressure injection system (LPIS).

The passive part of emergency core cooling system includes 4 HA YT11B01, YT12B01, YT13B01, YT14B01 (Figure 2-3) with 50 m^3 of boric acid each at the pressure of 60 kgf/cm^2. Two of the hydroaccumulators (YT11B01, YT13B01) are connected to the reactor upper plenum, and two others (YT12B01, YT14B01) are connected to the downcomer. Decrease of RCS pressure causes opening of the check valves at the pipelines connecting HA and reactor, and boric acid from HA is injected to the reactor [2].

HPIS is composed of three identical and independent trains which are connected to the cold legs of RCS loops no.1, 4 and 3 downstream reactor coolant pumps. Principal diagram of the first HPIS train is shown in Figure 2-10.

Each HPIS train consists of the following main equipment [2]:

- high head safety injection pump (TQ14D01, TQ24D01, TQ34D01);

- high pressure injection pump (TQ13D01, TQ23D01, TQ33D01);
- high concentration (40 g/kg of H_3BO_3) boric acid storage tank (TQ14B01, TQ24B01, TQ34B01);
- HPIS storage tank (TQ13B01, TQ23B01, TQ33B01) with boric acid concentration of 16 g/kg;
- pipelines and valves.

The high head safety injection pump characteristics allow to inject 6.3 m^3/h of boron solution at RCS pressure up to 200 bar. HPIS pumps TQ13D01, TQ23D01, TQ33D01 allow to inject ~100 m^3/h of boric acid solution at RCS pressure below 10 MPa. After depletion of HPIS tank the high pressure injection pump operates from LPIS tank (TQ10B01, TQ20B01, TQ30B01).

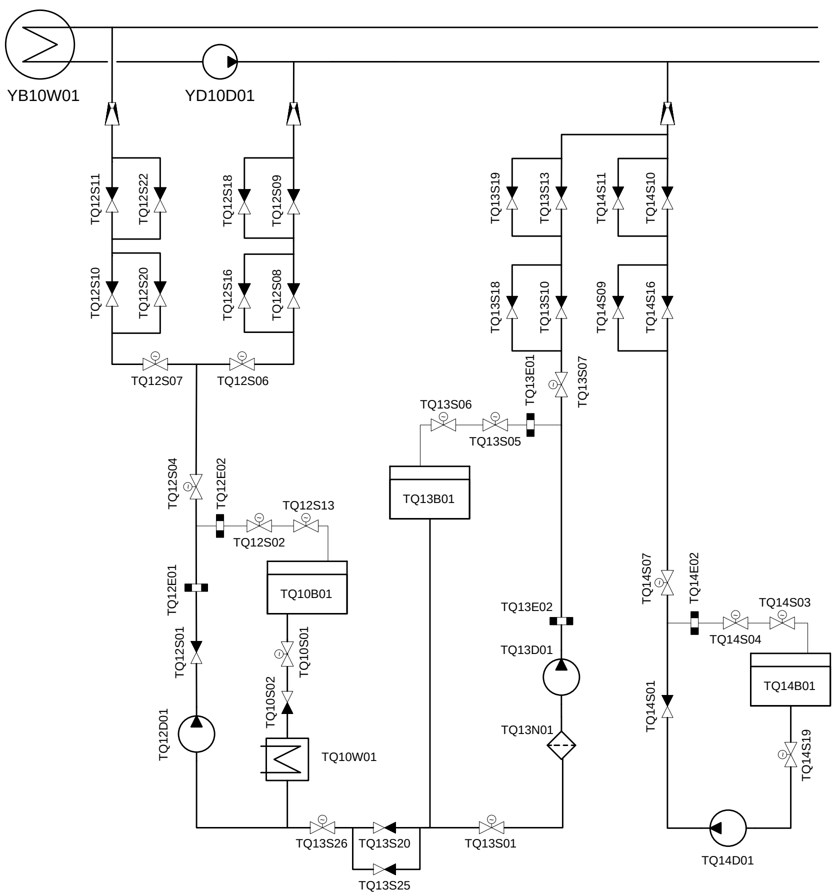

Figure 2-10 Principal Diagram of HPIS and LPIS (1st train)0

LPIS is designed to provide emergency core cooling during large and medium break LOCAs. LPIS is also used for planned reactor cooling down at low RCS pressure and long term residual heat removal during a refueling outage.

LPIS is composed of three identical and independent trains. Principal diagram of the first LPIS train is shown in Figure 2-10.

Each train includes the following main equipment (Figure 2-10) [2]:

- low pressure injection pump (TQ12D01, TQ22D01, TQ32D01);
- ECCS heat exchanger (TQ10W01, TQ20W01, TQ30W01);
- borated water tank (TQ10B01, TQ20B01, TQ30B01), which contains boric acid of 16 g/kg concentration;
- pipelines and valves.

Charging line of the 1st LPIS train is connected to the hot and cold legs of RCS loop no.1. The 2nd and 3rd LPIS trains are connected to the pipelines of HA-3,4 and HA-1,2, respectively.

2.3 Technological and Safety Systems of the Secondary Circuit

The main technological and safety systems of the secondary circuit include:

- main steam lines system;
- turbine bypass to condenser BRU-K;
- steam dump valve to atmosphere BRU-A;
- secondary circuit overpressure protection system;
- main feedwater system;
- auxiliary feedwater system;
- emergency feedwater system.

2.3.1 Main Steam Lines System

The main steam lines (MSL) system is intended for transportation of steam produced in steam generators to the turbine. The principal diagram of the MSLs system is shown in Figure 2-11.

The steam from each steam generator is delivered through the separate pipeline to one of four turbine stop valves. To isolate the steam generators in the case of the steam line breaks the fast-acting steam isolation valve (FASIV) TX50,60,70,80S06 and check valve TX50,60,70,80S07 are installed at each of four main steam lines. Downstream these valves the MSLs are connected to the main steam header (MSH).

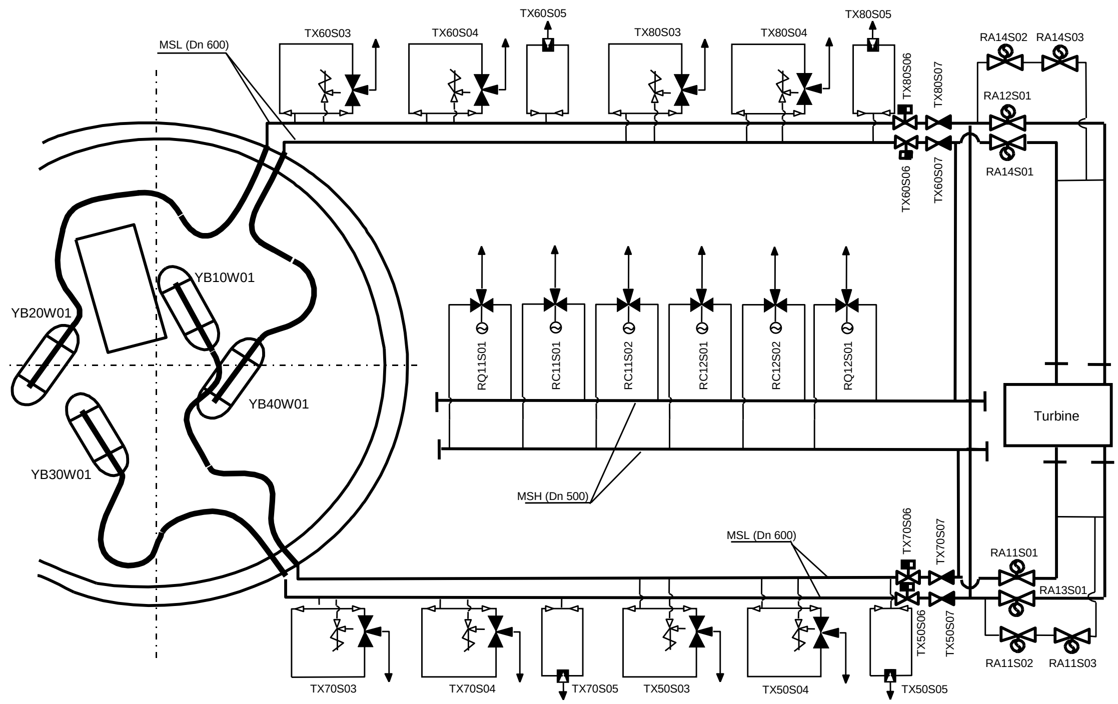

Figure 2-11 Principal Diagram of the Main Steam Lines System

2.3.2 Turbine Bypass to Condenser BRU-K

The system consists of four BRU-K RC11S01, RC11S02, RC12S01, RC12S02 (Figure 2-11) which are connected to the main steam header and intended:

- to maintain the secondary circuit pressure below 68 kgf/cm^2; by steam dump to the turbine condenser;
- to provide the secondary cooling down.

During normal power operation the system is in standby mode with all BRU-Ks closed. The system automatically switches to the pressure maintenance mode at the increase of MSH pressure above 68 kgf/cm^2 and BRU-K controllers start to maintain the pressure in the range of 64 – 68 kgf/cm^2.

At the decrease of the turbine load for 10% with the rate of 33 MW/s or higher BRU-K controllers switch to the load shedding mode with opening of BRU-K valves proportionally to the load decrease value. BRU-K setpoint for further pressure maintenance corresponds to MSH pressure at the time of load shedding mode actuation.

Two other BRU-K controller operation modes are used during reactor start-up and cooling-down and include maintaining of current MSH pressure value and cool-down with a rate of 30°C/h or 60°C/h.

2.3.3 Steam Dump Valves to Atmosphere BRU-A

BRU-As TX50,60,70,80S05 (Figure 2-11) are connected to the MSLs upstream FASIV (one BRU-A per SG) and are intended:

- to prevent secondary circuit pressure increase above 73 kgf/cm^2;
- to remove decay heat from the primary circuit;
- to perform secondary circuit cool-down.

During normal power operation the system is in standby mode with all BRU-As closed. At the increase of MSH pressure above 73 kgf/cm^2 the system automatically switches to the pressure maintenance mode and BRU-A controllers start to maintain pressure in the range of 68 – 73 kgf/cm^2. At the decrease of secondary circuit pressure down to 64 kgf/cm^2 the controller closes BRU-A and after 100 s switches to the standby mode.

Cool-down mode is intended to remove decay heat if cooling down via BRU-K or technological condenser is not possible. Cool-down can be performed with a rate of 30°C/h or 60°C/h. If cool-down mode is started the automatic interlock for positive BRU-A closure at the opening percentage less than 6% is deactivated.

2.3.4 Secondary Circuit Overpressure Protection System

Each MSL is equipped with two SG safety relief valves (SRV) TX50,60,70,80S03 and TX50,60,70,80S04 (Figure 2-11). SG SRV characteristics are selected so as to avoid an increase of the secondary circuit pressure for more than 15% of the design value during design basis transients and accidents. The opening setpoints for the control and main SG SRVs are 84 kgf/cm^2 and 86 kgf/cm^2, respectively. The setpoint for SG SRVs closure is 70 kgf/cm^2. The valves can also be operated manually from the main or reserve control room.

2.3.5 Main and Auxiliary Feedwater Systems

The main feedwater system (MFW) is intended to supply feedwater from secondary circuit deaerators to SGs during normal power operation, start-up and cool-down. The principal diagram of MFW system is shown in Figure 2-12.

The main equipment of the system includes [2]:

- two secondary circuit deaerators (including the tank RL21,22B01 and deaerating towers RL21,22W01,02);
- filters RL31,32N01,02;
- two booster pumps RL31,32D01 and turbine driven pumps RL41,42D01;
- high pressure heaters RL11,12,21,22W01;
- SG feed water distribution unit;
- pipelines and valves.

Upstream SG feed water distribution unit the main feedwater system consists of two identical trains. Feed water distribution unit consists of 4 trains supplying water to each SG. During normal power operation the water flow to SGs is controlled by MFW controllers RL71,72,73,74S02. At the bypasses of the main controllers the start-up/shut-down feed water controllers RL71,72,73,74S04 are installed to maintain required MFW flow at low flow rates.

Auxiliary feedwater system (AFW) is intended for supplying feedwater from deaerators to SGs during:

- reactor start-up and cooling-down;
- house loads steam supply power (< 5% of nominal power);
- malfunctions of MFW.

The system consists of:

- AFW electric pumps RL51,52D01 (Figure 2-12);
- cut-off, control (RL51,52S06) and check valves;
- suction pipelines from the inset of MFW pipelines to AFW pumps;
- charging pipelines from AFW pumps to the insets of MFW charging lines;
- recirculation pipelines to deaerators for feed water heat-up and deaeration during start-up.

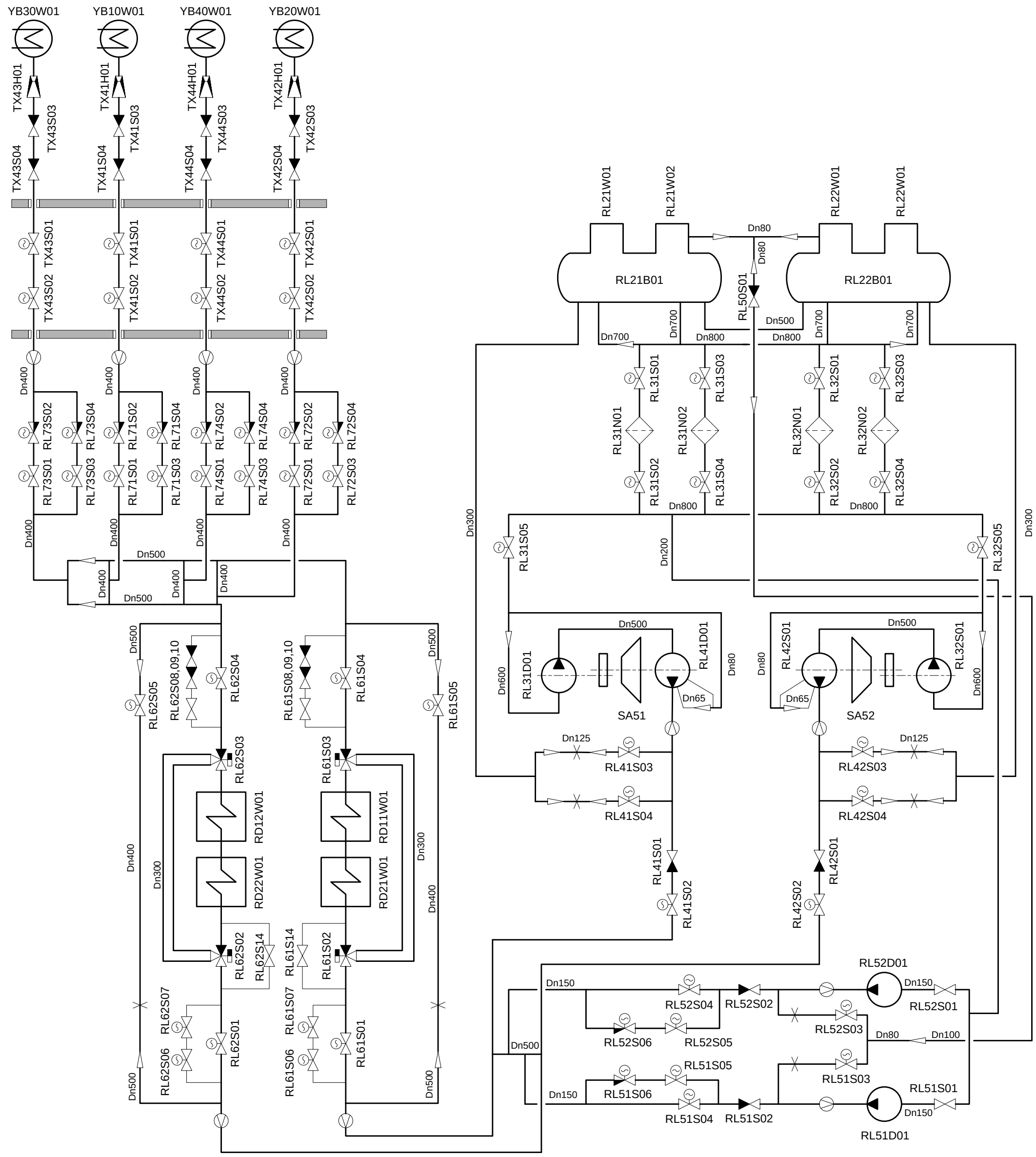

Figure 2-12 Principal Diagram of the Main and Auxiliary Feedwater Systems

2.3.6 Emergency Feedwater System

Emergency feedwater system (EFW) is designed to provide feed water supply to SG during transients and accidents with a loss of MFW and AFW. The principal diagram of EFW system is shown in Figure 2-13 (based on data from [2]).

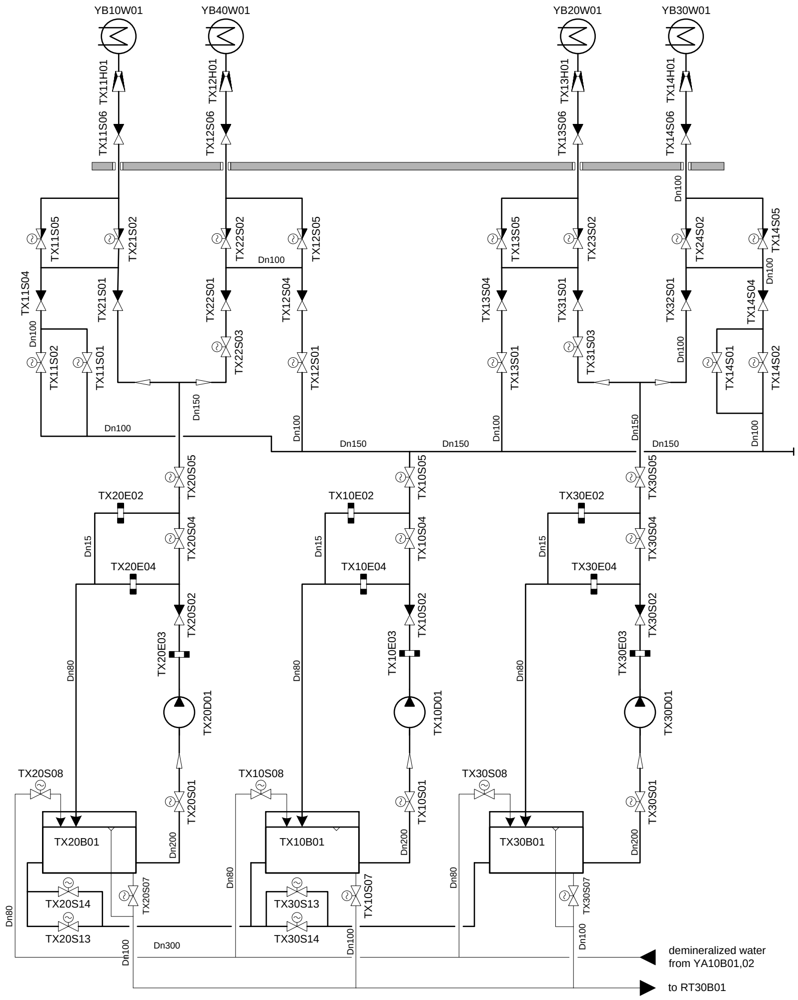

Figure 2-13 Principal Diagram of the Emergency Feedwater System

The system consists of three trains TX10,20,30 and include the following main equipment [2]:

- EFW tanks (TX10,20,30B01) which are connected by pipelines with normally closed cut-off valves TX20,30S13,14;
- EFW pumps TX10,20,30D01;
- pipelines and valves including EFW control valves TX11S05, TX21S02, TX14S05, TX32S02, TX12S05, TX22S02, TX13S05, TX31S02.

EFW pumps TX20D01 and TX30D01 which supply feedwater to SG-1,4 and SG-2,3, respectively. EFW pump TX10D01 is able to provide EFW supply to any of 4 SGs. But normally two of them (YB10,30W01) are isolated from TX10D01 by cut-off valves.

3 BRIEF DESCRIPTION OF TRACE MODEL FOR VVER-1000

Description of the main hydraulic components of TRACE model for VVER-1000 is provided below.

3.1 Reactor Model

Nodalization diagram of WWER-1000 reactor model for TRACE code is shown in Figure 3-1. The hydraulic model consists of the following main components:

- downcomer;
- lower plenum;
- core region;
- core bypass;
- upper plenum;
- upper head.

Reactor model is 4-sectoral and has cross-links to simulate crossflows between the sectors.

The area of inlet and outlet nozzles is divided into 8 equal sectors (hydraulic components 5001 – 5004, 5245 – 5248 for inlet nozzles; 5249 – 5256 for outlet nozzles) modeling annular gap between the core barrel and the reactor pressure vessel. This allows proper flow distribution in scenarios with partial number of reactor coolant pumps in operation.

Hydraulic components (HC) 450, 470, 690, 650 model four sectors of downcomer part below the RCS nozzles region, and HC 380, 410, 700, 660 model a gap between reactor bottom and core barrel.

Lower plenum is modeled by 16 HC (4 sectors, 4 vertical layers) 5013 – 5016; 390, 400, 680, 600; 80, 90, 220, 210; and 420, 440, 570, 500.

The core region is divided into 4 sectors (individual sector for each of 4 reactor coolant system loops). Radial division of core region is not envisaged. Each sector has two channels: for "average" fuel assemblies (HC 320, 350, 580, 510) and for "hot" fuel assembly (HC 5033 – 5036).

Core bypasses are presented by HC 5259, 5332 and 5271.

Upper plenum is modeled by two groups of HC 490, 480, 590, 550, simulating volume inside PTU perforated cylindrical shell and HC 5081, 5242 – 5244, simulating the volume between PTU shell and perforated cylindrical part of core barrel. The reactor upper head is represented by HC 5257 and 5260. The latter one simulates annular gap between the cylindrical shell of PTU (above the medium PTU plate) and reactor upper head.

Heat structures (HS) in the reactor core are modeled by “Fuel Rod” type and have cylindrical geometry with the height of 354 cm. Eight heat structures are modeled in the core, and each heat structure has an appropriate surface multiplier according to the subdivision of reactor core into hydraulic components.

HS of the reactor pressure vessel simulates the heat transfer from the coolant to the containment air through the vessel wall.

HS of reactor internals such as core barrel and baffle model the heat exchange between coolant and these reactor components.

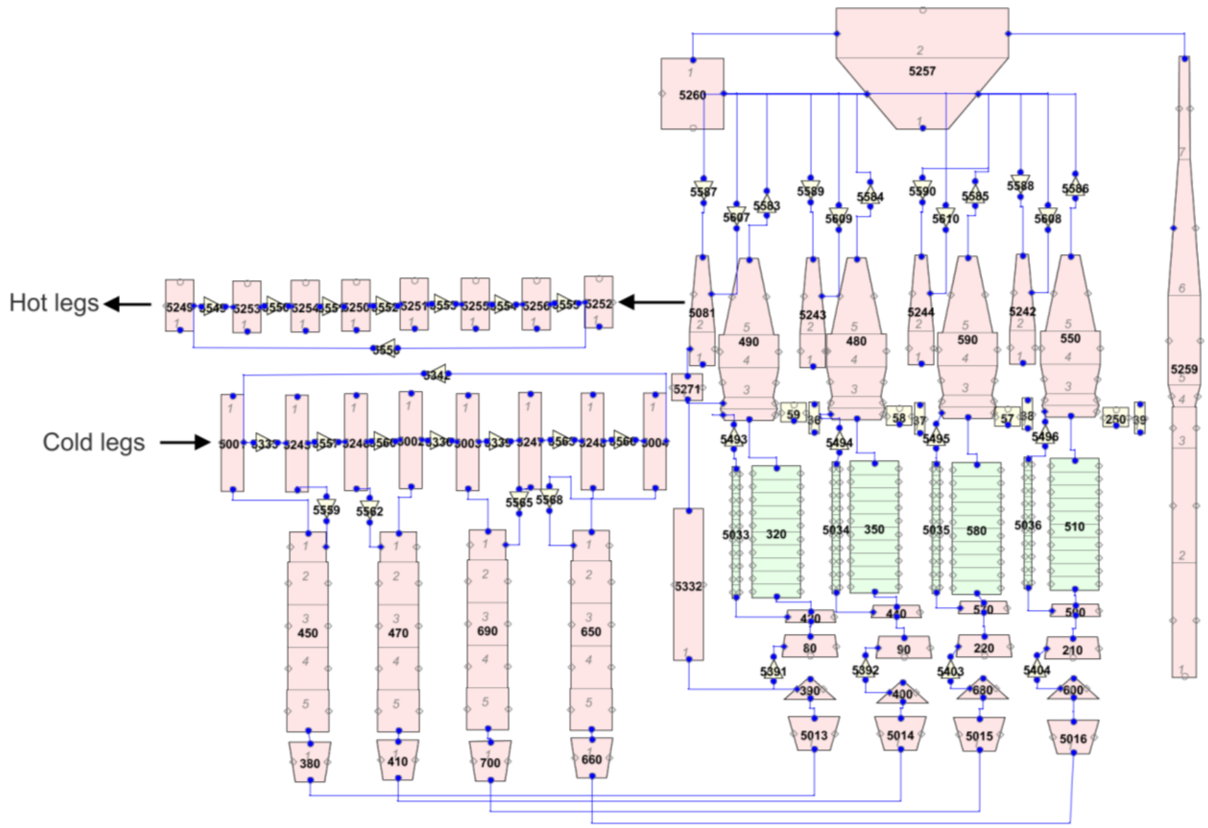

Figure 3-1 Nodalization Diagram of VVER-1000 Reactor

3.2 <u>RCS Loops Model</u>

All four RCS loops are simulated in the model separately. Nodalization diagram of RCS loop no.1 including hot and cold legs, RCPs, as well as SG collectors is shown in Figure 3-2. Other loops are modeled similar to loop no.1.

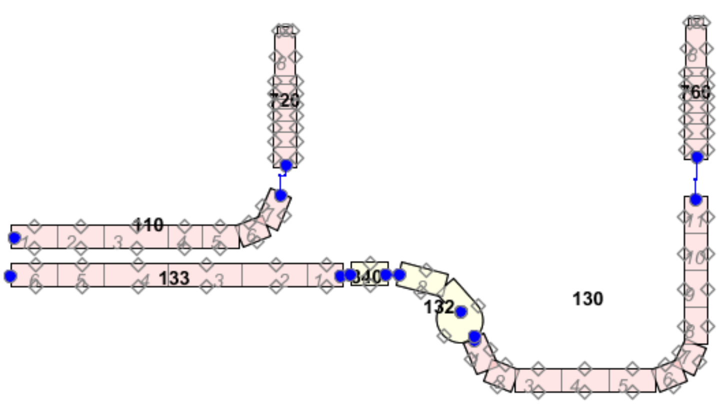

Figure 3-2 Nodalization Diagram of RCS Loop 1

3.3 <u>Pressurizer System Model</u>

Pressurizer model is implemented by HC 421 of PRIZER type and connected to the model of loop 4 hot leg by hydraulic components 200 and 790 that represent PRZ surge line (Figure 3-3). Steam discharge pipe (PIPE HC 50002 and 50003) is connected to the pressurizer top part. HC 605 and 607 of the VALVE type simulate PRZ control and the main PORV, respectively. Operation of these valves is implemented according to opening/closure setpoints of the main and control PORVs. The setpoints are changed at the station blackout after discharge of batteries to simulate spring-controlled PORV operation.

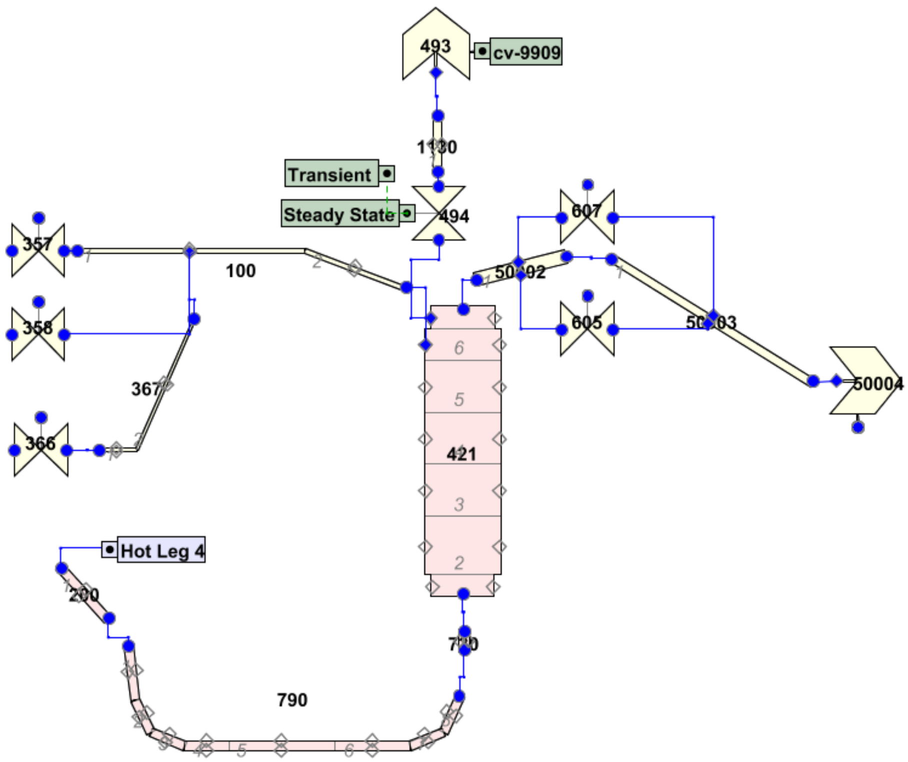

Figure 3-3 Nodalization Diagram of Pressurizer, Surge and Relief Pipes

Nodalization diagram of the pressurizer spray line is shown in Figure 3-4.

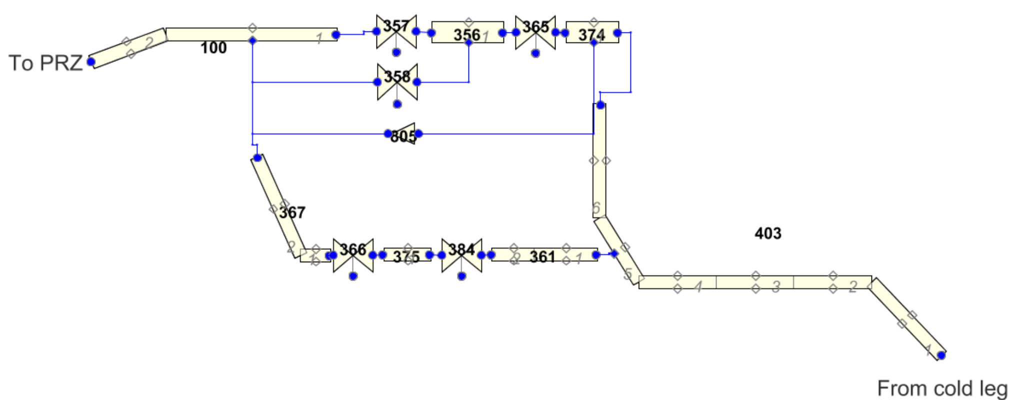

Figure 3-4 Nodalization Diagram of the Pressurizer Spray Line

3.4 Steam Generators Model

SG primary side model (Figure 3-5, left) includes hydraulic components that simulate SG collectors and a tube bundle. In accordance with nodalization of SG secondary side (Figure 3-5, right) the tube bundle is subdivided into five layers. Each layer includes two HC representing U-shaped tubes.

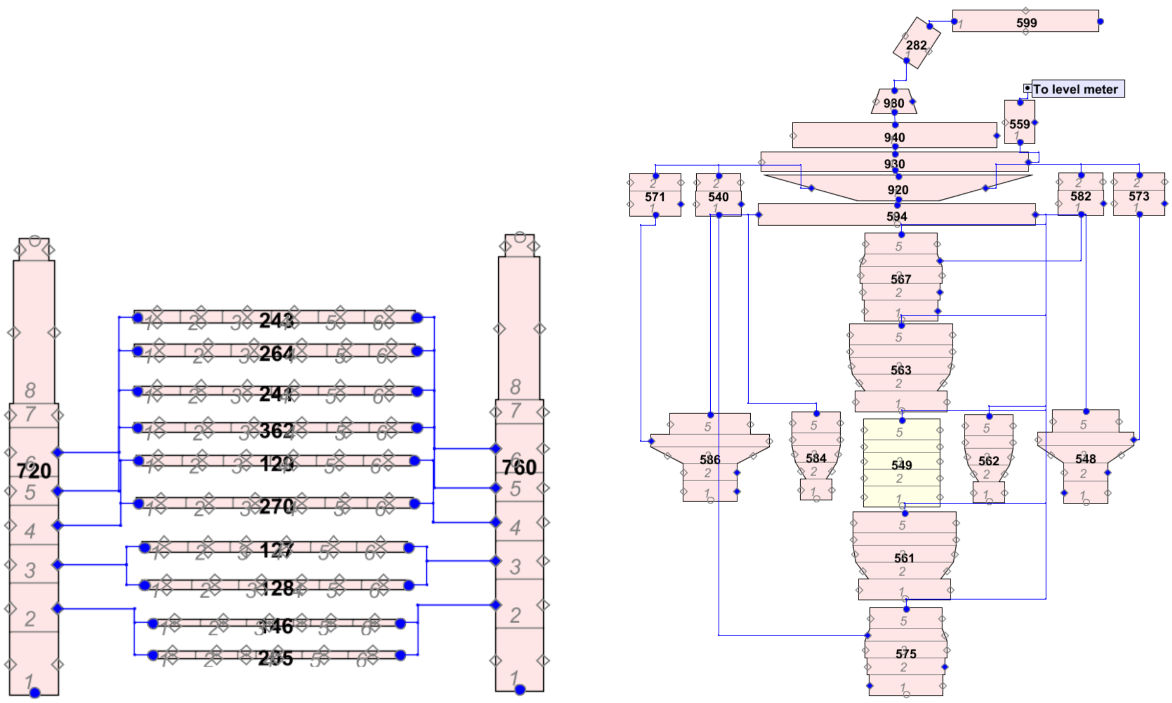

Figure 3-5 Nodalization Diagram of SG Primary and Secondary Side

SG secondary side is modeled using quasi-3D approach. Such approximation was chosen for correct distribution of thermal load between SG secondary side volumes.

HC 563, 561 and HC 562, 584 model SG secondary side volumes of the tube bundle region corresponding to the straight and bended (U-shaped) parts of SG-tubes, respectively. HC 575, 567 (up to the 4th volume) model the secondary side volumes between SG tube bundle and SG vessel, and between external and internal tube bundle packs corresponding to the straight portions of SG tubes. 5th volume of these HC is somewhat smaller and is bounded by submerged perforated plate (SPP) side walls. HC 586 and 548 model similar SG secondary side volumes which correspond to the bended portion of SG tubes. HC 549 represents the central part of SG secondary side between SG tubes packs. HC 540, 571, 582, 573 model the secondary side volume between SPP side walls and SG vessel. The steam volume of SG secondary side and of SG steam header are represented by HC 594, 920, 930, 940, 980 and HC 282, respectively.

3.5 Make-Up and Let-Down Model

The make-up and let-down subsystems of CVCS are modeled at a functional level. Makeup charging lines are connected to the cold legs of all four loops between SG cold collector and RCP. Let-down pipelines are connected to the cold legs of loop 2 and 3 downstream RCP outlet nozzle.

Nodalization diagram of the makeup and let-down is shown in Figure 3-6.

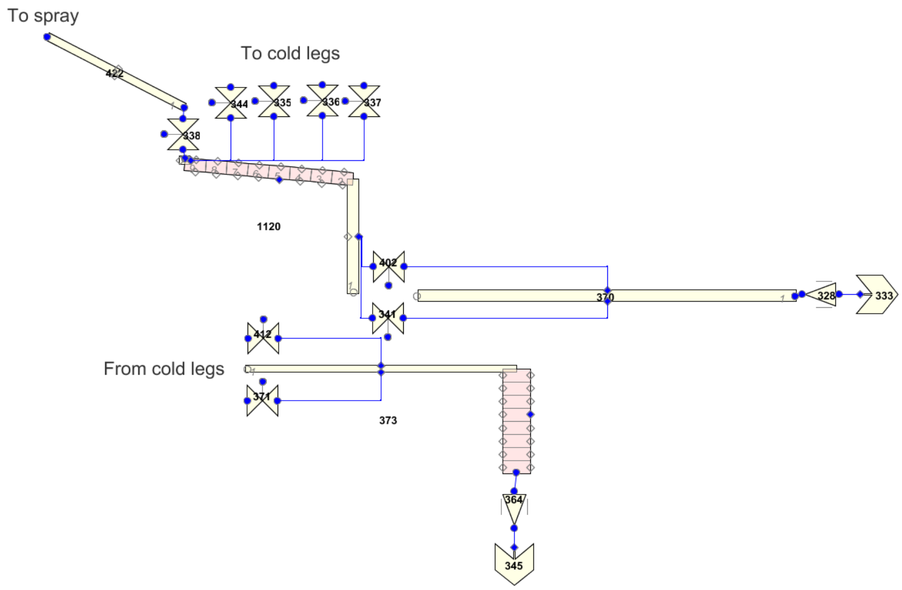

Figure 3-6 Nodalization Diagram of Makeup and Let-down

3.6 Emergency Gas Evacuation System Model

The emergency gas evacuation system is intended to remove steam-gas mixture from the primary system during accidents. The system consists of connecting pipes with cut-off valves installed, which connect reactor coolant system elements with relief tank (Figure 2-9).

Nodalization diagram of the emergency gas removal system is presented on Figure 3-7.

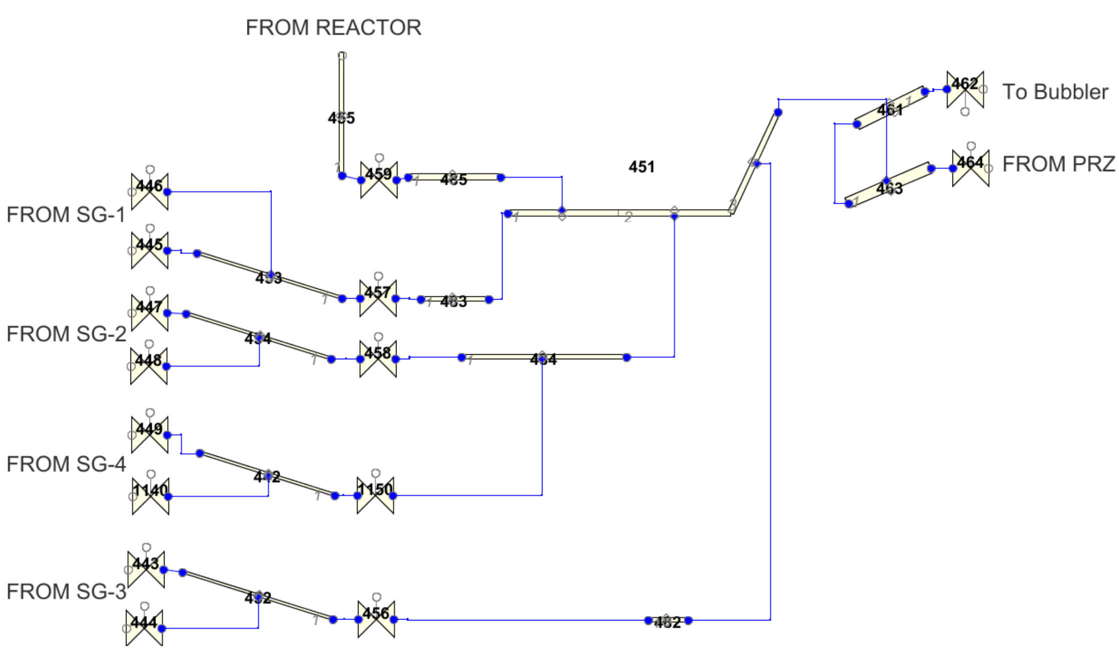

Figure 3-7 Nodalization Diagram of Emergency Gas Removal System

3.7 Main Steam Lines Model

Nodalization diagrams of MSLs are shown in Figures 3-8 – 3-11. Each steam line model includes steam dump valves to the atmosphere BRU-A, SG steam relief valves, main steam isolation valves, turbine stop and control valves.

BRU-As are simulated by VALVE HC 706, 736, 786, 876. Hydraulic components (1-4)60 of the BREAK type model atmospheric conditions downstream BRU-A. BRU-A flow rate is 900 t/h. Time for full opening/closure of BRU-A is 18 sec. The control and the main SG SRVs are modeled similarly to BRU-A.

Turbine is modeled by boundary conditions (downstream turbine stop valves) and represented by HC 574 of the BREAK type. Turbine stop valves are modeled by HC 566, 666, 756, 856 with identical boundary conditions.

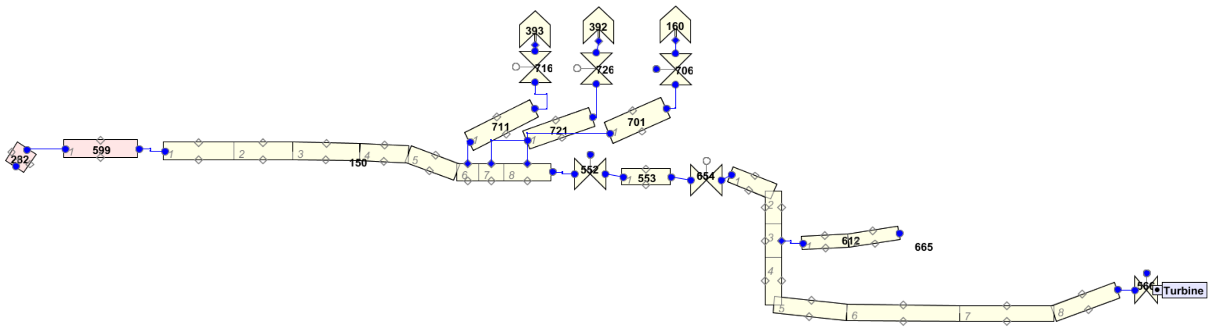

Figure 3-8 Nodalization Diagram of MSL No. 1

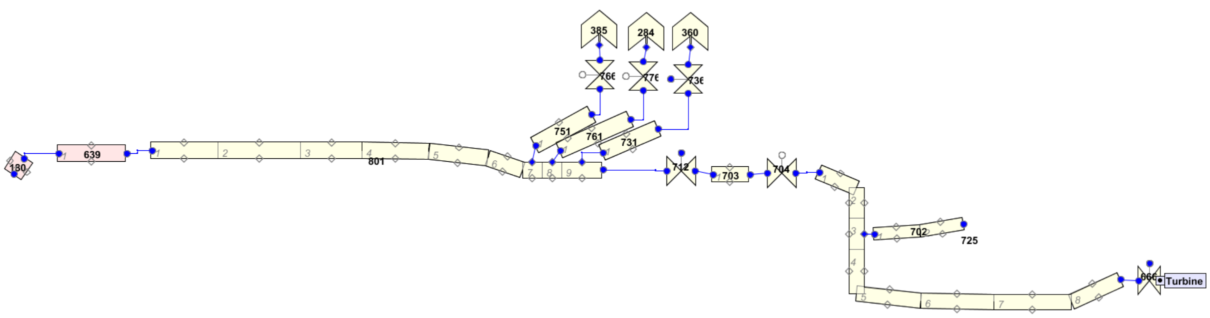

Figure 3-9 Nodalization Diagram of MSL No. 2

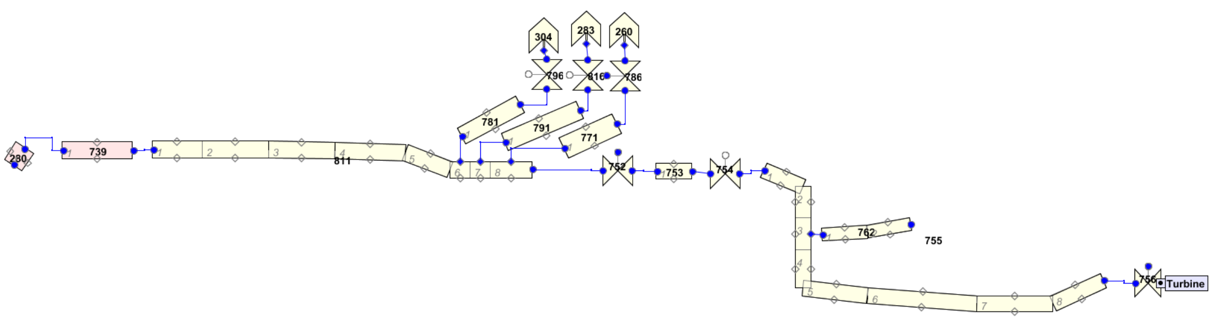

Figure 3-10 Nodalization Diagram of MSL No. 3

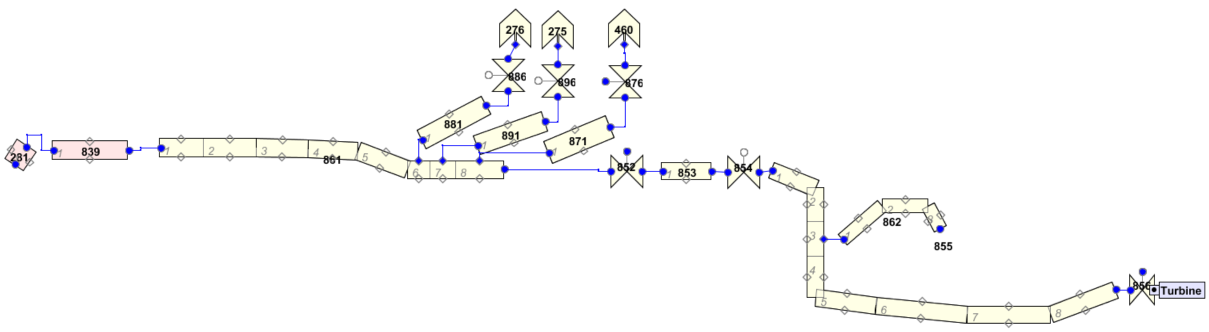

Figure 3-11 Nodalization Diagram of MSL No. 4

3.8 Main Steam Header Model

Main steam header includes two semi-headers, connection lines between them and steam dump valves to the condenser BRU-K (Figure 2-11).

MSH is modeled by PIPE HC 901, 903. HC 906, 907, 926, 927, 908, 918 model connection lines to BRU-K. VALVE HC 911, 921 model BRU-K. One model valve represents two BRU-Ks with equivalent cross section. BREAK HC 558, 589 model turbine condenser. Nodalization diagram of MSH and BRU-K is presented in Figure 3-12.

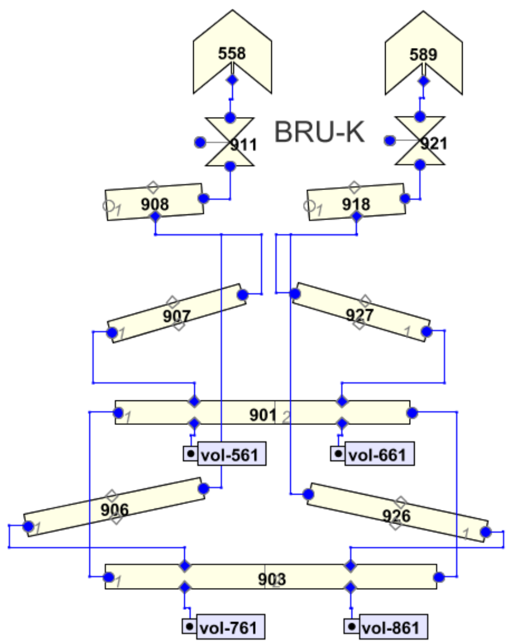

Figure 3-12 Nodalization Diagram of MSH and BRU-K

3.9 Main, Auxiliary and Emergency Feedwater Systems

The nodalization diagrams of the main, auxiliary and emergency feedwater systems are presented in Figures 3-13 and 3-14.

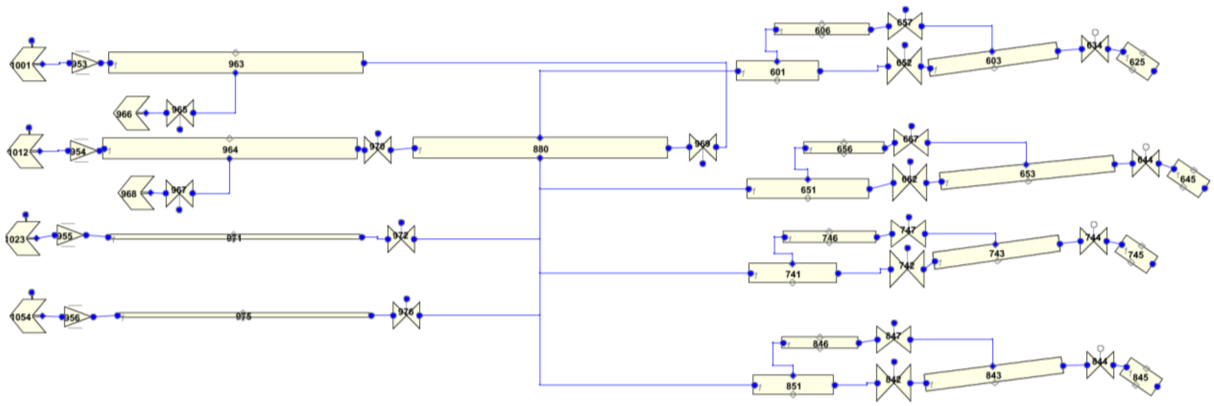

Figure 3-13 Nodalization Diagram of the Main and Auxiliary Feedwater Systems

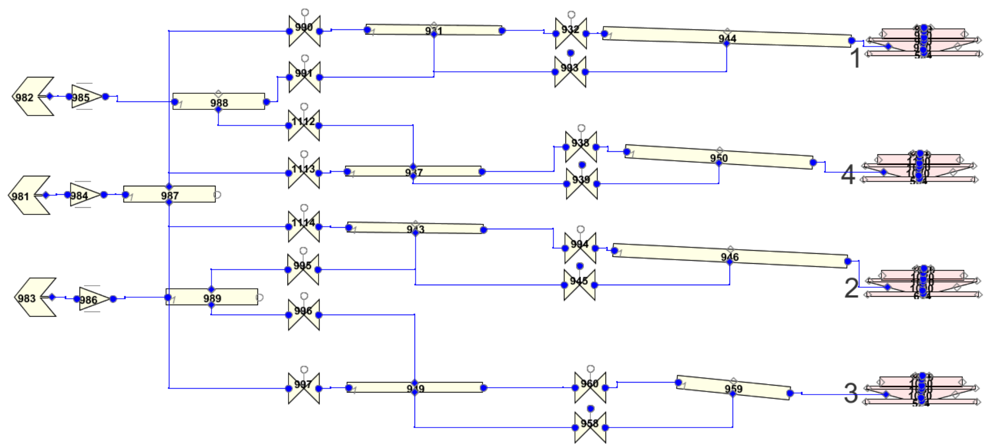

Figure 3-14 Nodalization Diagram of the Emergency Feedwater System

3.10 ECCS Model

Four hydroaccumulators are simulated by PIPE hydraulic components 610, 620, 630, 640 (with Accumulator option). Two of them (HC 610 and 630) are connected to the reactor upper plenum HC 5081 and 5243 (Figure 3-1) via HC 611 – 613, 631 – 633 simulating HA pipelines, and two others (HC 620 and 640) are connected to the downcomer HC 5246 and 5248 via HC 621 – 623, 641 – 643.

LPIS and HPIS models are implemented at the functional level. The charging lines of LPIS trains no.2, 3 are simulated with HC 821 – 824, 831 – 834, respectively and are connected to hydraulic components simulating HA pipelines. The charging lines of the first LPIS train (HC 811 – 814) are connected to the model of hot and cold legs of loop no.1. Hydraulic components 812, 822, 832 of FILL type simulate LPIS pumps.

Each HPIS train is modeled identically and is represented by FILL hydraulic components (HC 913, 914 – 1st train; HC 923, 924 – 2nd train; HC 933, 934 – 3rd train) simulating high head safety injection and high pressure injection pumps, and VALVE components (HC 915, 916; HC 925, 926; HC 935, 936) modelling HPIS charging pipelines.

The nodalization diagrams of ECCS passive part, LPIS and HPIS trains are shown in Figures 3-15, 3-16. Logic diagrams of HPIS and LPIS operation are shown in Figures 3-17 and 218, respectively.

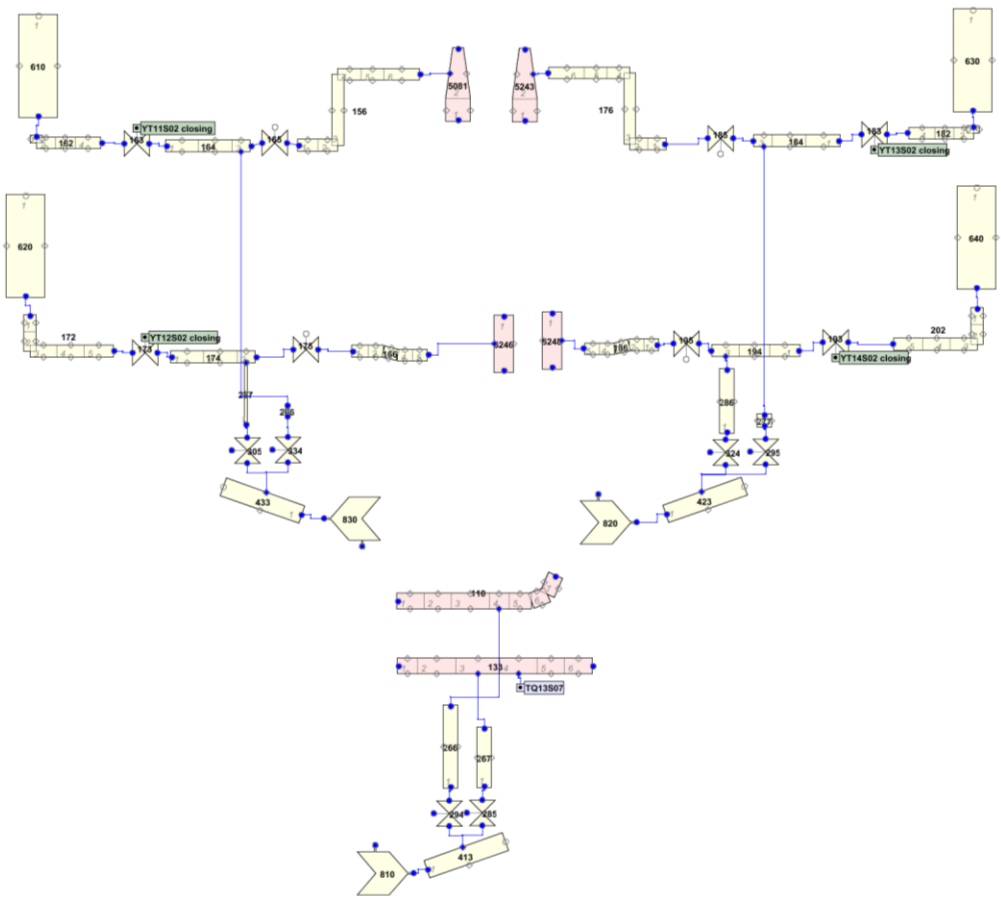

Figure 3-15   Nodalization Diagram of ECCS HA and LPIS Trains

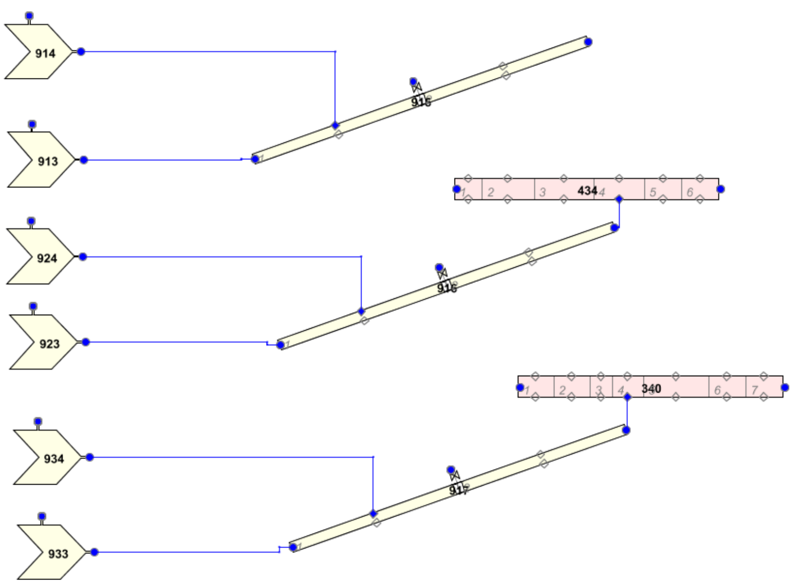

Figure 3-16   Nodalization Diagram of HPIS Trains

3.11 Safeguards and Control Systems Operation Logic

Figures 3-17 – 3-29 present logical diagrams for the main safeguards and control systems. The operation logic for the following systems and elements is provided:

- HPIS and LPIS (Figures 3-17, 218);
- reactor control and protection system (Figures 3-19, 3-22);
- PRZ level control (Figure 3-23);
- make-up pressure difference control (Figure 3-24);
- MFW flow rate controller (Figures 3-25, 3-26);
- SG level controller (Figure 3-27);
- BRU-A controller (Figure 3-28);
- turbine control system (Figure 3-29).

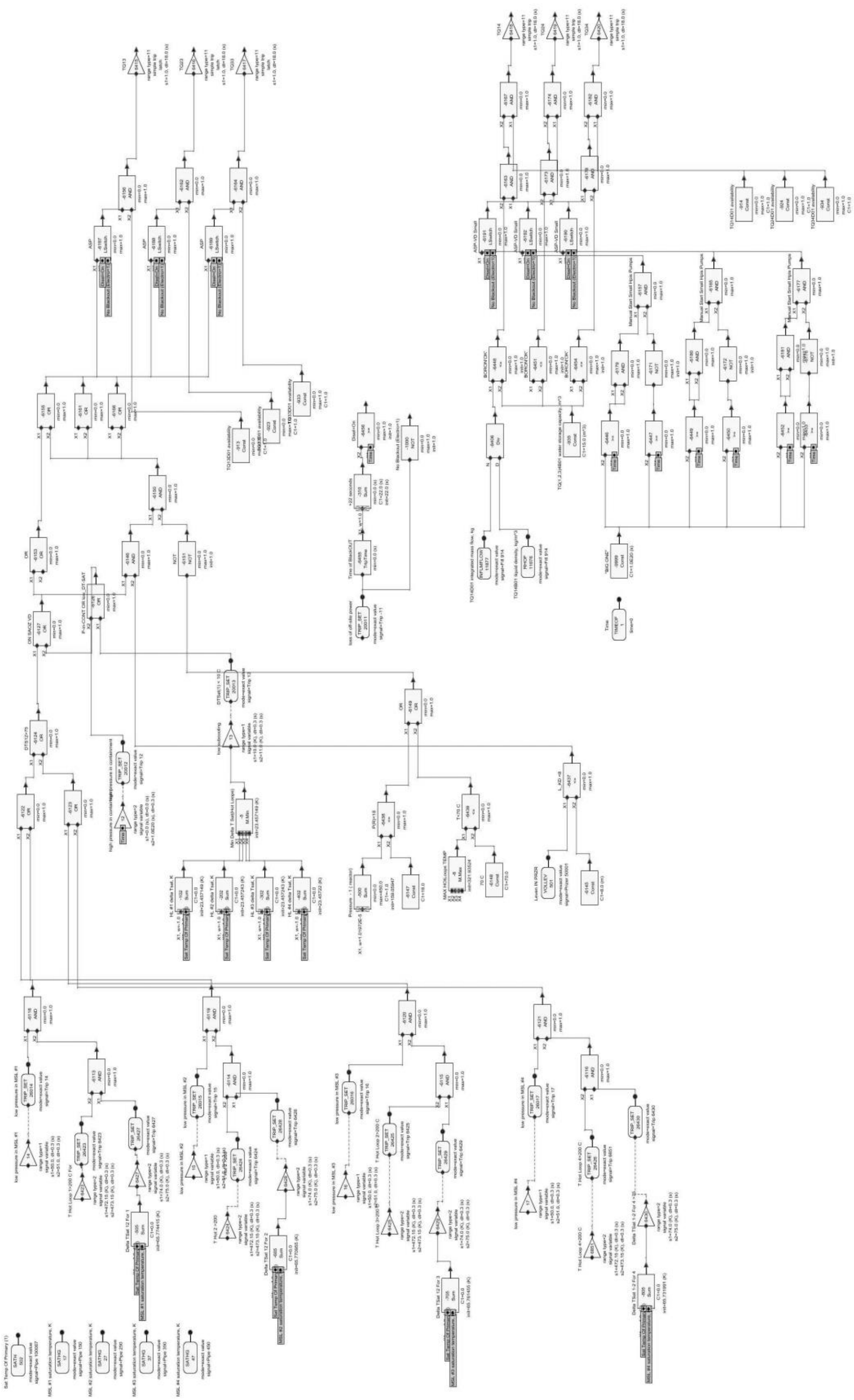

Figure 3-17 Logical Diagram of HPIS Operation

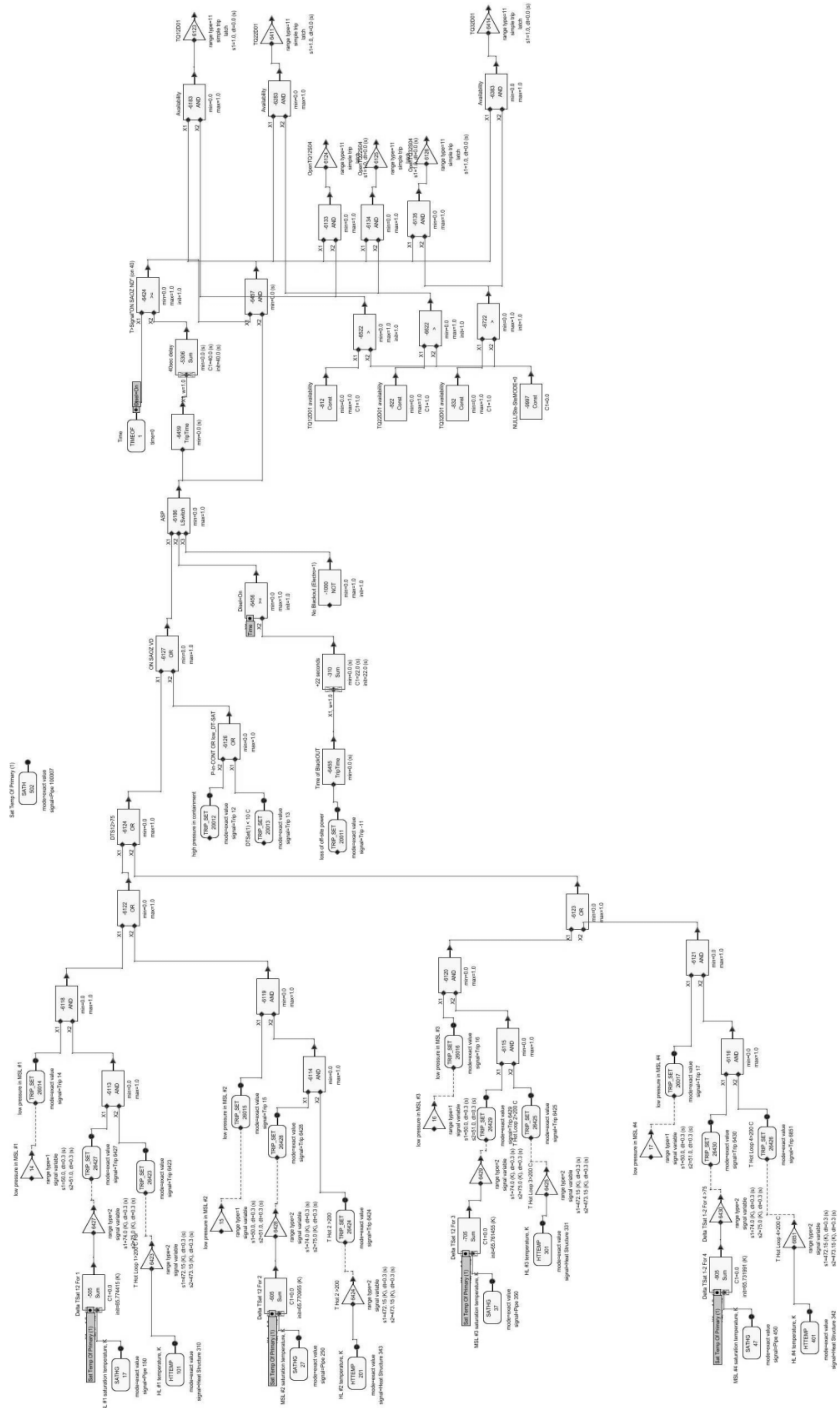

Figure 3-18 Logical Diagram of LPIS Operation

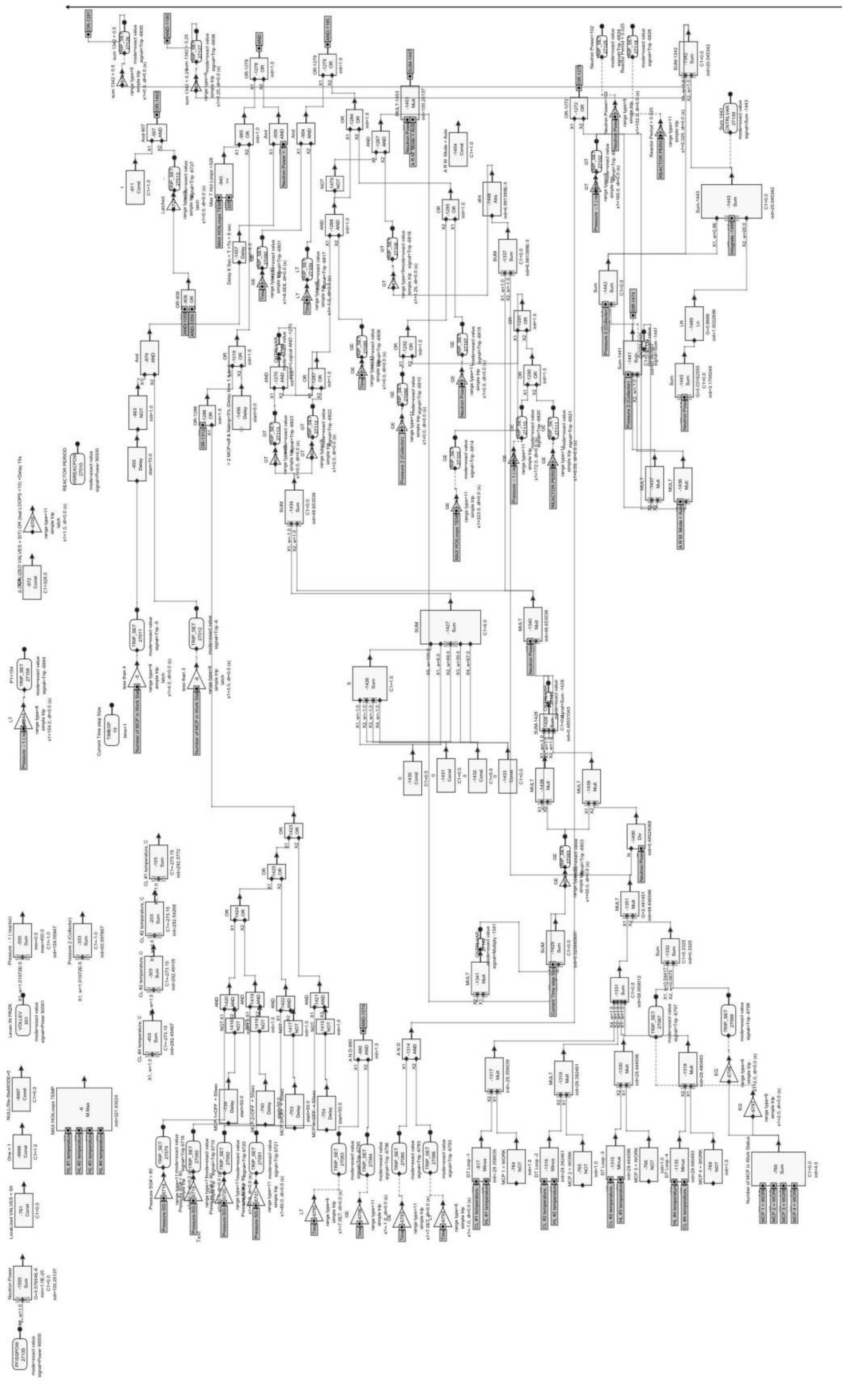

Figure 3-19 Logical Diagram of Control and Protection System (Part 1)

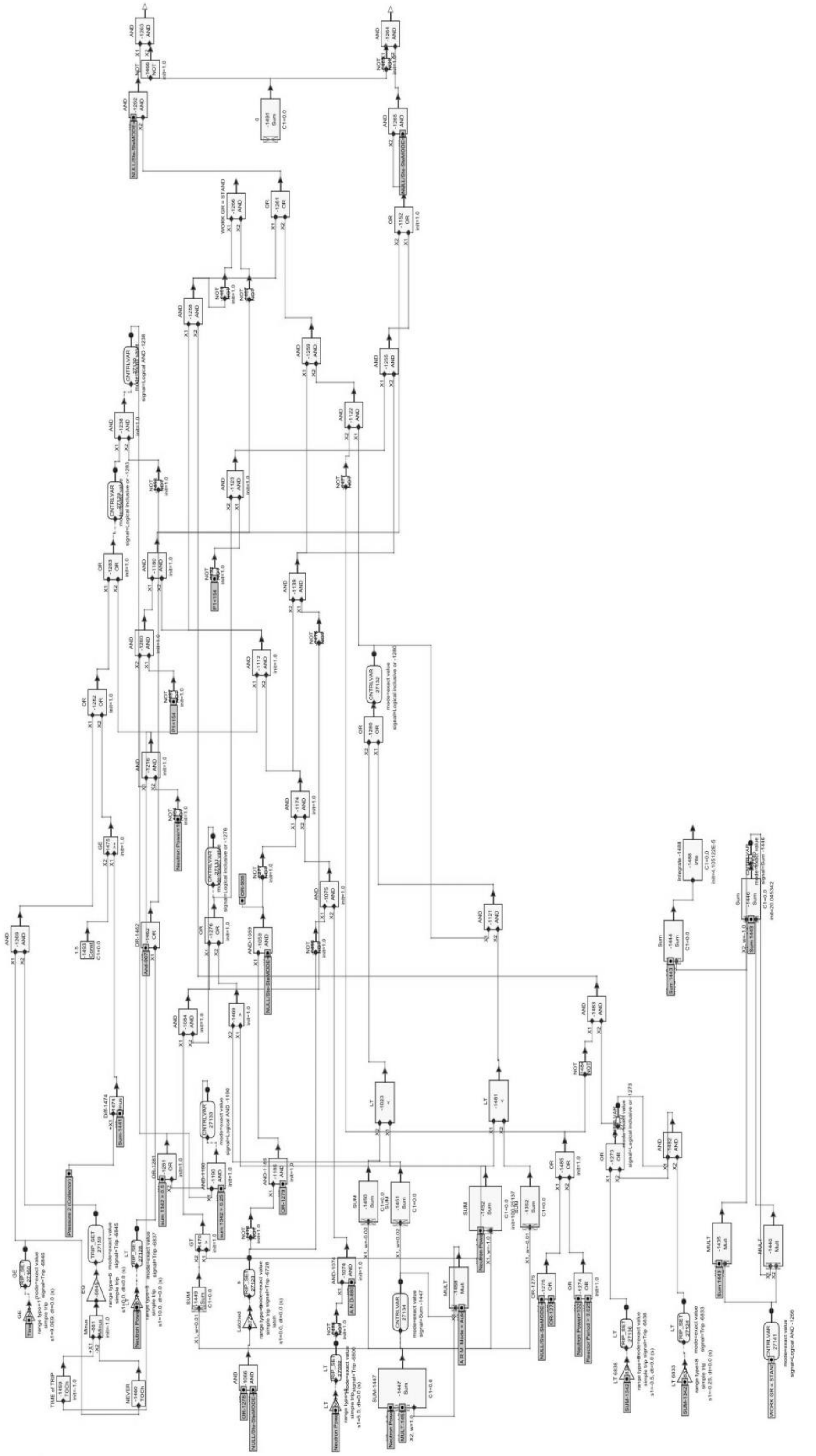

Figure 3-20 Logical Diagram of Control and Protection System (Part 2)

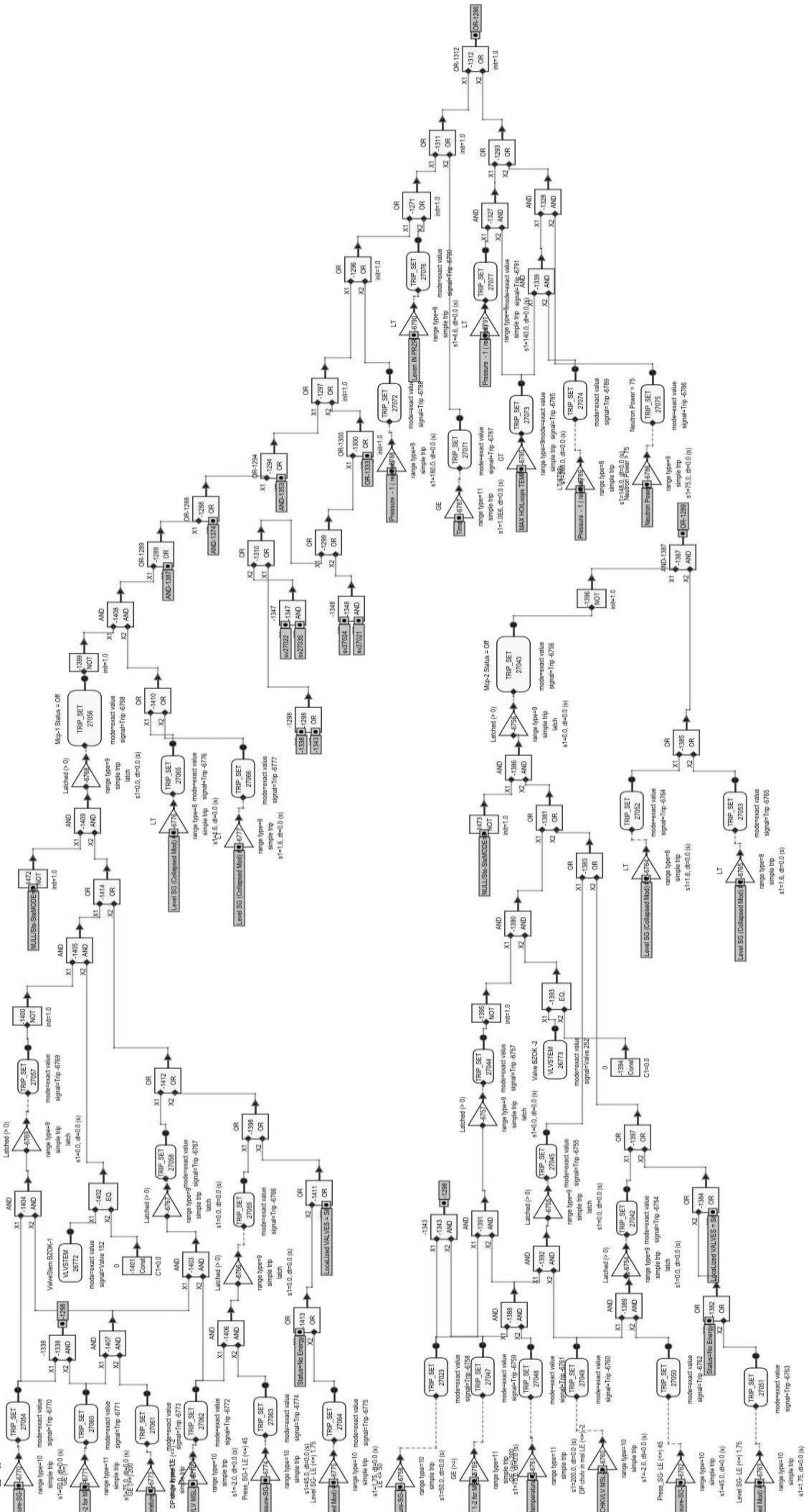

Figure 3-21 Logical Diagram of Control and Protection System (Part 3)

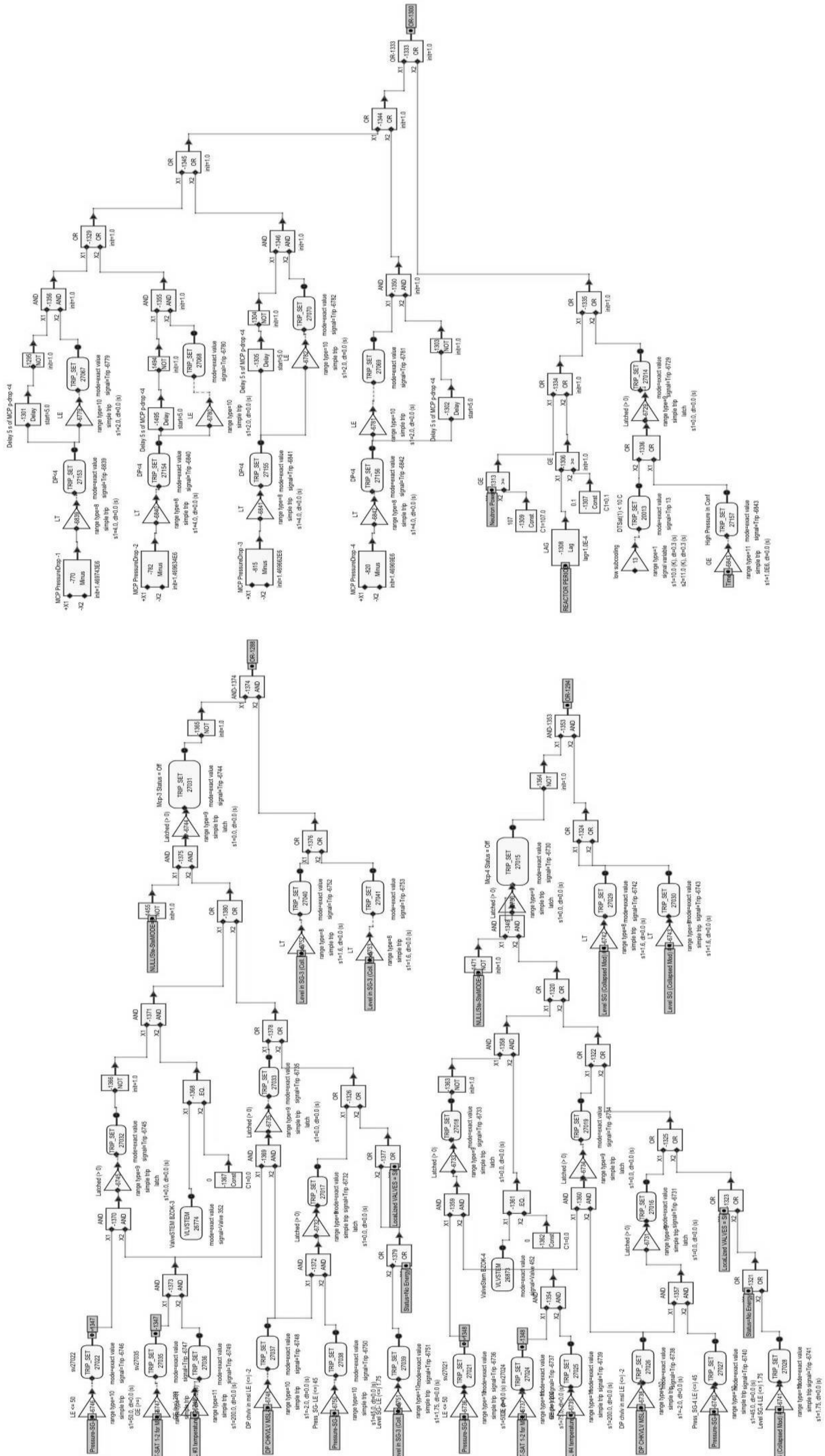

Figure 3-22 Logical Diagram of Control and Protection System (Part 4)

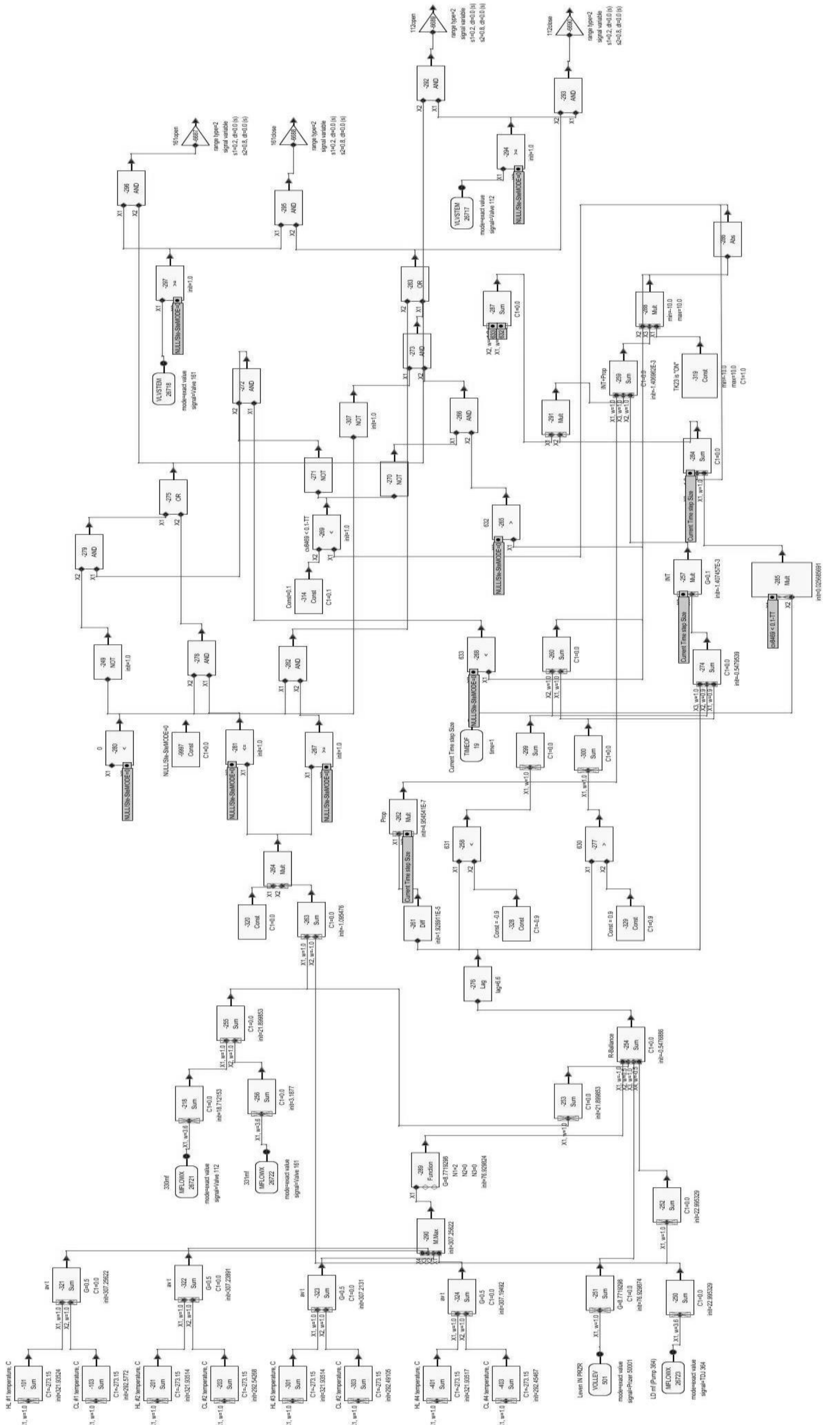

Figure 3-23 Logical Diagram of PRZ Level Control by Makeup and Let-Down

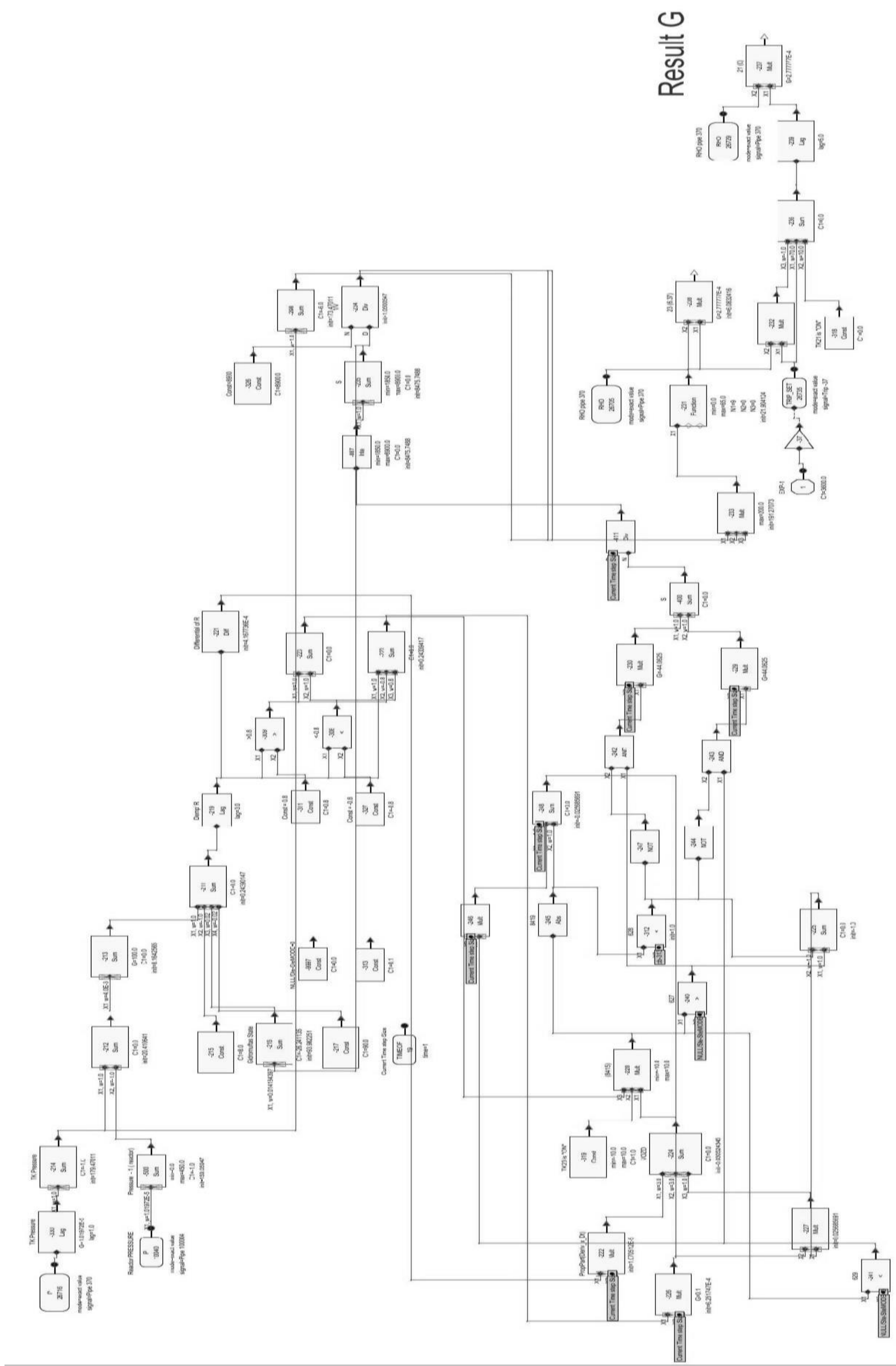

Figure 3-24 Logical Diagram of Make-up Pressure Difference Controller

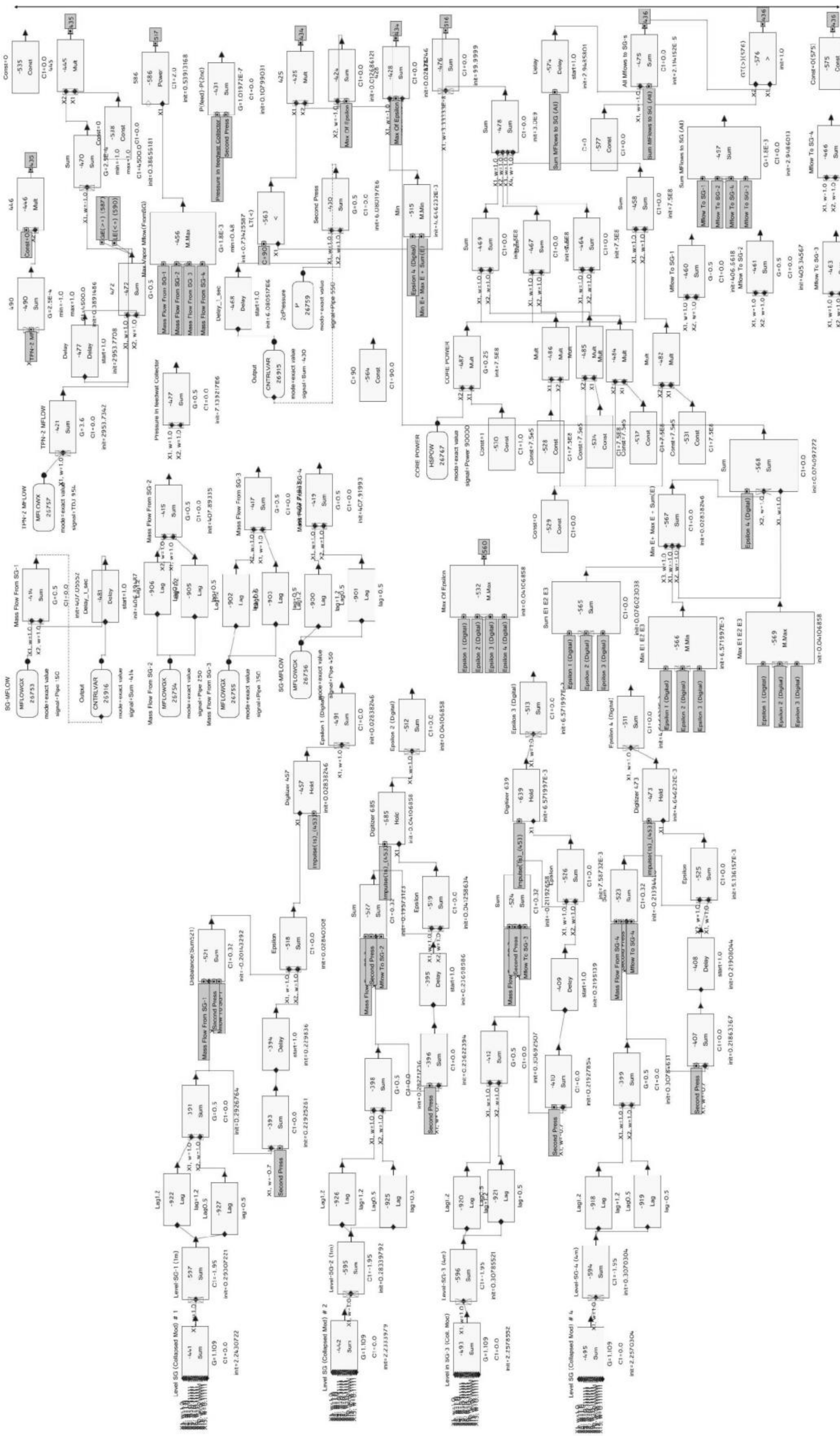

Figure 3-25 Logical Diagram of MFW Flow Rate Controller (Part 1)

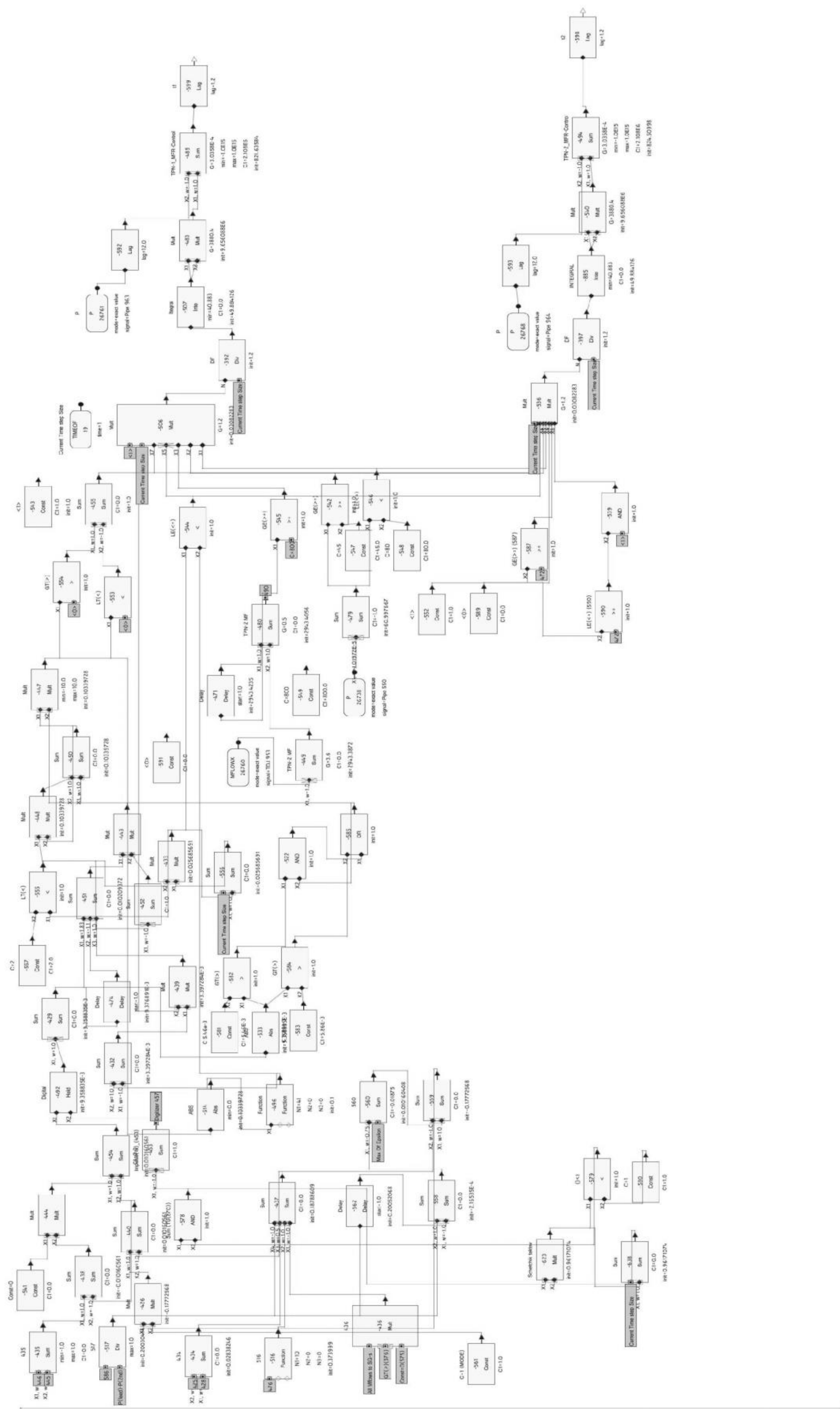

Figure 3-26 Logical Diagram of MFW Flow Rate Controller (Part 2)

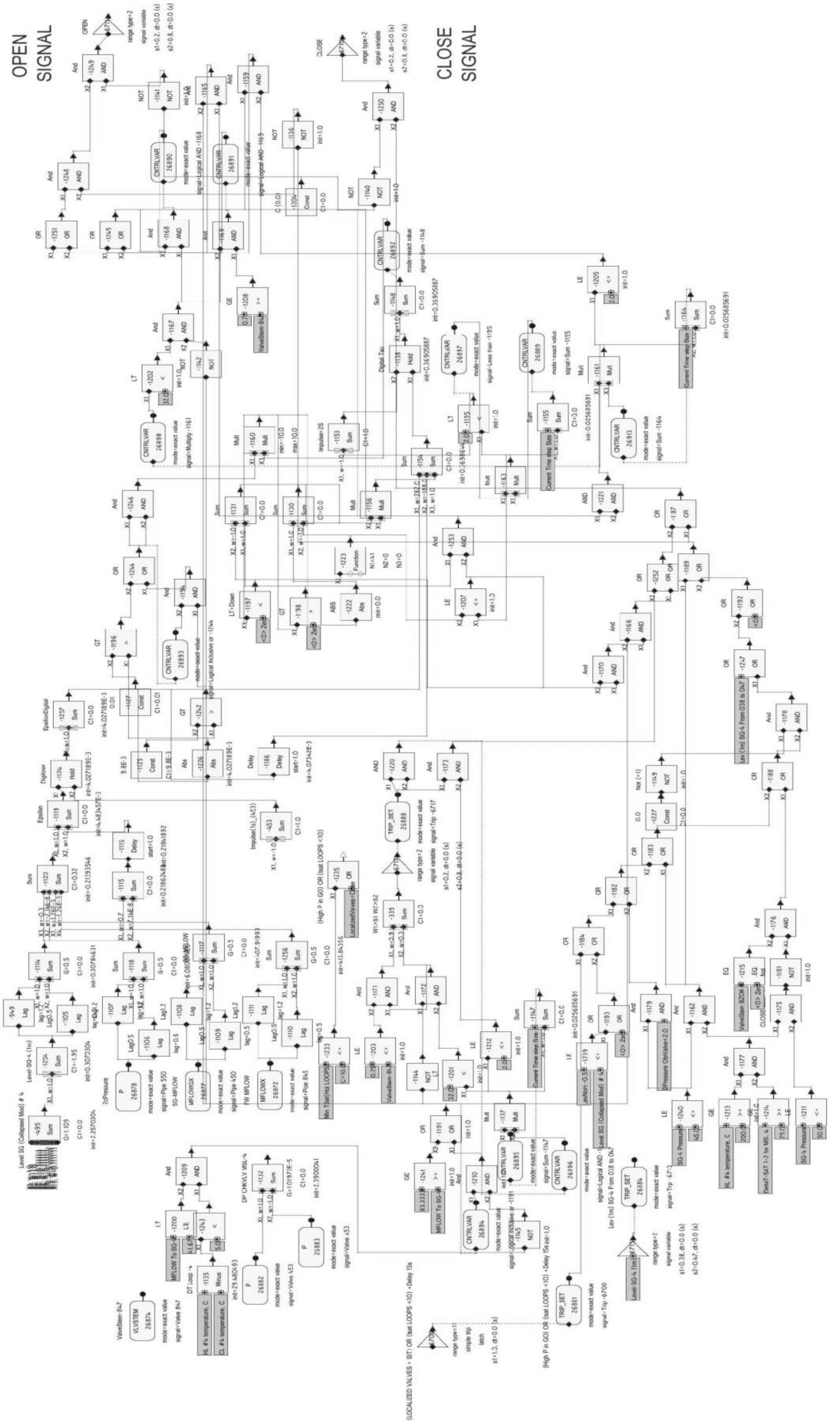

Figure 3-27 Logical Diagram of SG Level Controller

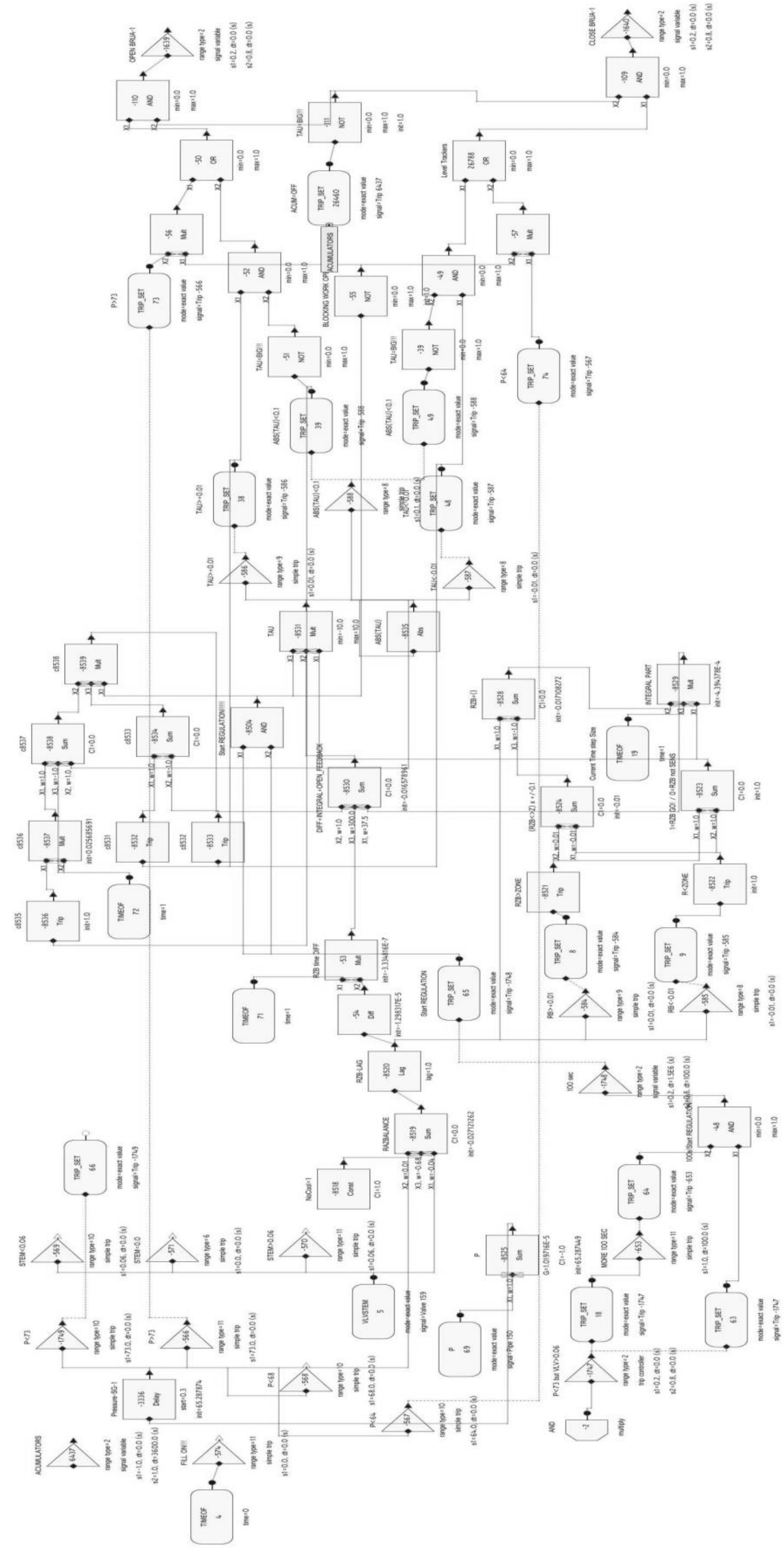

Figure 3-28 Logical Diagram of BRU-A Controller

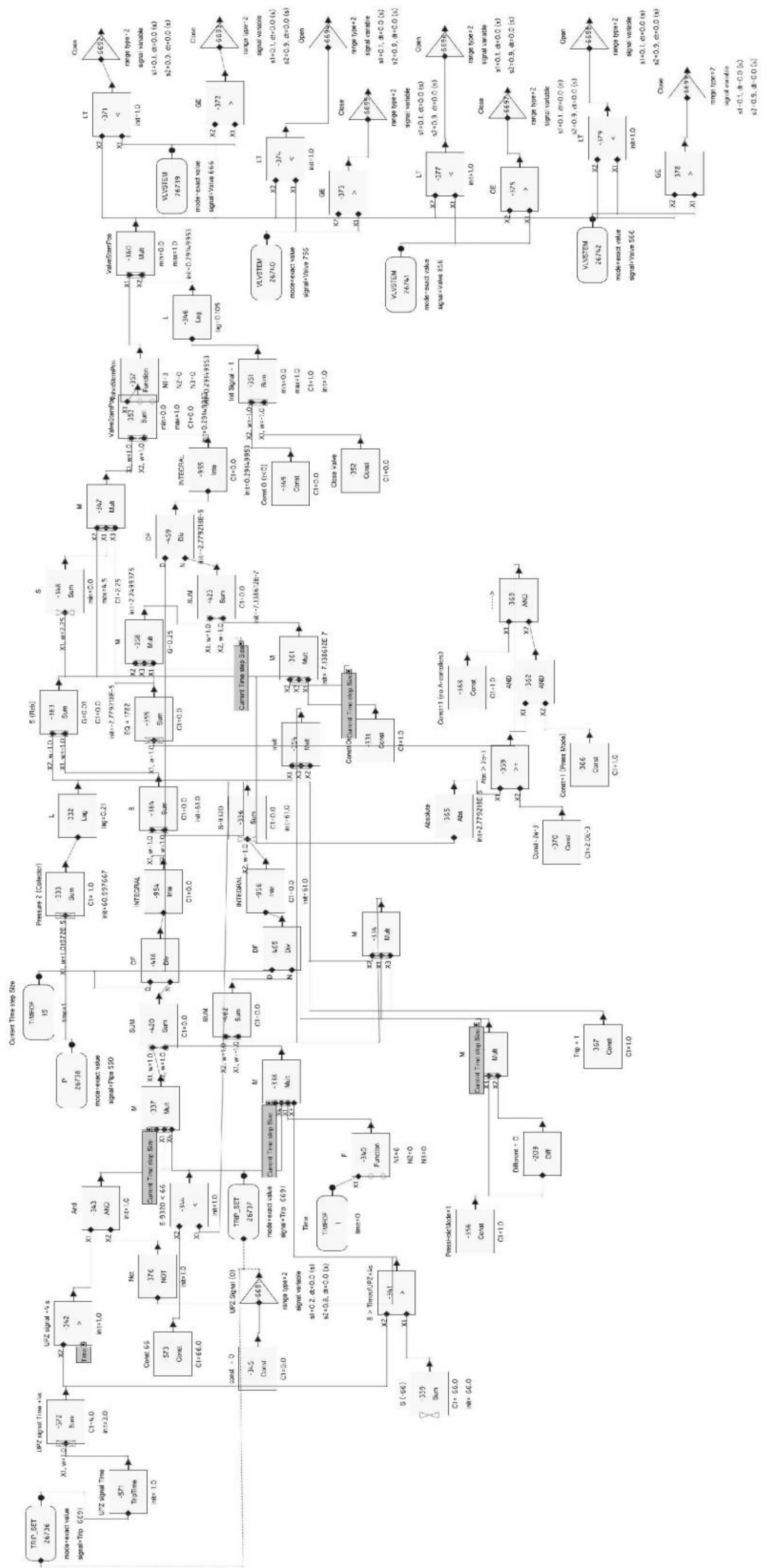

Figure 3-29 Logical Diagram of Turbine Control System

4 CALCULATION OF TRANSIENTS

The following incidents were selected for validation calculations:

- ZNPP Unit 5 MFW-1 pump trip;
- ZNPP Unit 6 inadvertent FASIV closure;
- RNPP Unit 3 PRZ PORV stuck open during tests.

4.1 ZNPP Unit 5 MFW-1 Pump Trip

This incident occurred at ZNPP Unit 5 on November 25, 1998, and was caused by a human error during the regular maintenance of MFW-1 pump steam drive stop valves. Erroneous closure of the stop valve resulted in a trip of MFW-1 pump with subsequent reactor power decrease by operation of fast reactor load shedding protection and reactor power limiter [3].

4.1.1 Initial Conditions

Before the incident the following main equipment was in operation at ZNPP Unit 5: 4 RCPs, 2 MFW pumps, electro-hydraulic turbine control system, MFW flow controllers to individual SGs, MFW pump flow controller, RCS pressure and PRZ level controllers.

Table 4-1 presents the measured and calculated values of the main primary and secondary circuit parameters before the transient.

Table 4-1 Initial Conditions for Validation Transient No.1

Parameter	Units	Nominal value	Calculated value
Reactor thermal power	MW	3028	3000
Generator electric power	WW	1017	–
Reactor outlet pressure	kgf/cm^2	158.6	158.3
MSH pressure	kgf/cm^2	60.2	60.3
Pressure downstream turbine stop valves	kgf/cm^2	55.6	54.7
PRZ level	m	8.85	8.87
Make-up flow rate	t/h	21.9	23.2
Let-down flow rate	t/h	23.8	23.0
Make-up temperature	°C	242.2	238.0
Hot legs coolant temperature	°C		
loop no.1		321.0	319.3
loop no.2		324.0	319.4
loop no.3		322.7	319.3
loop no.4		322.0	319.2

Parameter	Units	Nominal value	Calculated value
Cold legs coolant temperature	°C		
loop no.1		287.5	289.1
loop no.2		288.1	289.2
loop no.3		287.6	289.1
loop no.4		287.8	289.0
SG pressure	kgf/cm^2		
SG-1		62.3	62.3
SG-2		62.6	62.2
SG-3		62.5	62.3
SG-4		61.8	62.3
SG level (narrow range measurement)	m		
SG-1		0.331	0.316
SG-2		0.307	0.297
SG-3		0.313	0.315
SG-4		0.338	0.312
SG level (wide range measurement)	m		
SG-1		2.31	2.32
SG-2		2.33	2.33
SG-3		2.24	2.22
SG-4		2.31	2.30
MFW pumps flow rate	t/h		
MFW pump 1		2859	2928
MFW pump 2		2997	2928
SG feed water flow rate	t/h		
SG-1		1427	1461
SG-2		1463	1464
SG-3		1502	1463
SG-4		1464	1468
Main feedwater temperature	°C	216.8	220.0

The calculated initial state at RCS pressure of 158.3 kgf/cm^2 with the 1st PRZ heaters group in operation is slightly different from the actual plant state were operation of two PRZ heaters groups was observed at RCS pressure of 158.6 kgf/cm^2.

4.1.2 Boundary Conditions

The main assumptions on normal and safety systems operation applied in validation calculation are described below.

1. The flow rate of the main feedwater pump no.1 is defined as a function of time based on actual measured data provided in Table 4-2.

Table 4-2 Measured Flow Rate of MFW Pump No.1

Time, s	Flow rate, t/h
0	2460
11	1621
15	30
20	10
40	0

2. Termination of level 2 preventive protection signal at 253 s of incident is modeled as operator action.
3. Malfunction of SG-4 main feedwater controller (RL74C02) in the course of incident was caused by incorrect and delayed measurement YA40T10 of SG-4 hot and cold legs temperature difference. As the results, RL74C02 controller could not maintain SG-4 level within the prescribed band and the signal for closure of RL74S02 control valve was actuated. In calculation this was modeled by RL74S02 positive closure signal at 39 s (controller malfunction time).
4. Operation of house loads steam supply valve (BRU-SN) is simulated as an additional steam release from MSH. Steam release flow rate is derived from BRU-SN characteristics defined in [4]. BRU-SN stem position as a function of time was set according to plant measured data and specified in Table 4-3.

Table 4-3 BRU-SN stem position (S) as a function of time

Time, s	S, %	Time, s	S, %	Time, s	S, %	Time, s	S, %
0	0	196	14,1	352	35,1	500	51,0
20	0	200	16,0	356	41,1	504	56,3
24	14,7	204	18,0	360	46,5	512	56,3
28	19,3	216	18,0	372	46,5	516	45,4
32	0	220	15,1	376	42,0	520	45,4
36	2,4	232	15,1	380	41,9	524	48,9
40	53,3	236	18,1	384	49,7	528	55,7
44	45,2	252	18,1	388	52,8	540	55,7
48	25,5	256	21,8	412	52,8	544	48,9
52	16,4	260	30,0	416	54,9	548	46,0
56	16,4	272	30,0	420	57,5	552	46,0
60	18,5	276	32,3	432	57,5	556	50,3
64	22,7	280	38,4	436	44,1	560	50,4
80	22,7	288	38,4	440	44,1	568	50,4
84	18,3	292	41,2	444	51,1	572	43,8
88	15,0	304	41,2	448	57,8	596	43,8
96	15,0	308	45,1	456	57,8	600	45,8

Time, s	S, %	Time, s	S, %	Time, s	S, %	Time, s	S, %
100	22,1	316	45,1	460	47,8	612	45,8
104	25,0	320	41,5	472	47,8	616	41,6
116	25,0	324	38,9	476	51,7	620	38,9
120	16,9	328	38,9	480	57,1	624	44,2
132	16,9	332	42,9	484	57,1	1280	44,2
136	6,4	336	45,3	488	49,5		
144	6,4	344	45,3	492	46,0		
148	14,1	348	35,1	496	46,0		

5. Control rods effectiveness and reactivity coefficients applied in calculation correspond to the end of ZNPP Unit 5 10th fuel cycle (248.5 eff.days) due to unavailability of data for actual incident date (105.1 eff.days).

4.1.3 Calculation Results

Table 4-4 provides comparison of calculated and actual timing of events occurred in the course of the incident.

Table 4-4 Sequence of Events for Validation Transient No.1

Time, sec		Event
Incident	Calculation	
0	0	Start of calculation
3	3	Trip of MFW pump no.1
4	4	Actuation of fast reactor load shedding protection (UPZ) signal
5	5	Actuation of level 1 preventive reactor protection (PZ-1) signal
7	7	Actuation of level 2 preventive reactor protection (PZ-2) signal
8	11	PRZ heaters group no.3 is on
8	17	PRZ heaters group no.4 is on
39-80	39-79	Start of SG-4 MFW control valve closure
85	105	Termination of PZ-1 signal
93	93	RCP-4 trip
106	123	Start of SG-4 start-up/shutdown feedwater control valve operation
253	253	Termination of PZ-2 signal
307	297	Actuation of PZ-1 signal
312	427	PRZ heaters group no.4 is off
315	307	Termination of PZ-1 signal
332	499	PRZ heaters group no.3 is off
588	605	PRZ heaters group no.2 is off
–	1000	End of calculation

Trip of MFW pump no.1 caused an actuation of UPZ signal with a drop of selected control rods group and further decrease of reactor power (Figure 4-1) by PZ-1 (sequential insertion of control rods groups into the reactor core with a rate of 20 mm/s).

Operation of PZ-1 in calculation lasts longer (5-105 s) than in the incident (5-85 s), that can be explained by redistribution of assemblies power in the reactor core which cannot be modeled with point kinetics. Differences in coolant temperature reactivity feedback and power reactivity feedback during power decrease are compensated in calculation by control group of control rods.

After decrease the reactor power is stabilized at 46.4% according to plant measurement data and at 46.3% in calculation.

Measured and calculated relative turbine power (Figure 4-28) maintained by the turbine control system in MSH pressure maintenance mode are in good agreement. Faster decrease of turbine steam flow at the beginning of incident is explained by higher reactor power decrease rate following a drop of UPZ control rods group. After decrease the turbine power is stabilized at 41.6% of nominal value, while correspondent value in calculation is equal to 39.6%.

Initial RCS pressure drop at 4 s (Figure 4-2) is also caused by UPZ and PZ-1 operation. During decrease of reactor power the RCS pressure is determined by the following factors:

- reactor power decrease rate;
- operation of PRZ heaters groups;
- decrease of heat removal by the secondary side during increase of MSH pressure and decrease of SG levels.

After decrease of reactor power at the time interval from 100s to 300 s higher calculated RCS pressure can be explained by:

- stabilization of reactor power in calculation while measured power continues to decrease;
- termination of PZ-2 signal (prohibition of reactor power increase) at 253 s of calculation that causes an increase of reactor power from 43.2% to 47.8% by the reactor power controller (ARM) operation.

Minimal RCS pressure reached during decrease of reactor power is 150.2 kgf/cm^2 in calculation and 149.4 kgf/cm^2 as measured at the plant.

PRZ level (Figure 4-14) is controlled by the changes in make-up and let-down flow. Minimal calculated collapsed PRZ level is 6.79 m which is in good agreement with the measured data (6.80 m).

Higher calculated make-up flow (Figure 4-12) results in slightly faster restoration of PRZ level and RCS pressure than it is observed in the incident. Maximal calculated make-up flow of 38.6 t/h is reached at 205 s, while measured value at this moment is 33.4 t/h.

Decrease in make-up temperature (Figure 4-11) is determined by difference between make-up and let-down flow and changes in make-up heating due to the decrease of cold legs temperatures (Figures 4-7 – 4-10). Incorporation of make-up/let-down heat exchanger into the VVER-1000/V-320 model for TRACE code allowed to track changes in make-up temperature during the incident. Minimal calculated make-up temperature is 184.2°C while the measured value is 188.3°C.

After RCP-4 trip at 93 s of incident and its coast-down the coolant flow in loop 4 reverses and hot leg temperature in this loop decreases down to 277°C (Figure 4-6). Calculated and measured difference in temperatures of hot and cold legs of loops 2 and 3 (which are opposite to loop 4) decreases to 11.2, 12.0°C and 11.1, 12.8°C, respectively. Correspondent difference in loop 1 (adjacent to loop 4) decreases for 15.7°C in calculation and for 16.8°C according to measured data, which is higher than for loops 2, 3 and caused by less intense coolant mixing between loops 2, 3 and loop 4 comparing to mixing between loops 1 and 4.

Cold legs temperature (at the reactor inlet) in loops 1 – 4 decreases to 283.2, 285.6, 284.9, 283.9 °C in calculation and to 282.0, 284.3, 283.8, 282.8 °C according to the plant measurements (Figures 4-7 – 4-10).

Comparison of calculated and measured pressure differences at RCPs (Figures 4-19 – 26-22) before the incident (6,28 – 6,30 kgf/cm^2 vs 5.59 – 5.71 kgf/cm^2) and lower measured coolant heat-up (34.7°C vs calculated and nominal values of 30.2°C and 30.3°C) suggest that reactor coolant flow at the plant was lower than the design flow and can be explained by lower grid frequency. Calculated RCP-4 pressure difference decrease rate after RCP-4 trip is in agreement with measured data that confirms correct modelling of RCP coast-down time.

Actuation of the load shedding signal at the beginning of incident causes switching of the turbine control system to the MSH pressure maintenance mode. Following initial decrease of MSH pressure caused by decrease of reactor power by UPZ the maximal calculated and measured MSH pressure values (Figure 4-23) are 61.2 and 61.3 kgf/cm^2, respectively, that corresponds to the turbine control system target value of 61.0±0.5 kgf/cm^2.

Trip of MFW pump no.1 at 3 s resulted in the increase of pump no.2 flow from 2980 t/h to 4300 t/h (Figure 4-38) with following decrease due to decrease of reactor power. This process is adequately reproduced by MFW pump model. Average calculated MFW pump no.2 flow after decrease of reactor power is 2570 t/h, and the measured value is 2500 t/h.

The main feedwater flow to SG-1, 2, 3 is maintained by correspondent MFW controllers. Malfunction of SG-4 MFW controller and closure of RL74S02 valve results in significant decrease of SG level (Figure 4-36) and operation of start-up/shutdown feedwater controller RL74S04.

Calculated and measured SG levels behavior (Figures 4-29 – 4-36) corresponds to the changes in feedwater flow. Minimal values of calculated SG1–4 levels (narrow range) are 0.107, 0.116, 0.107, 0.008 m (Figures 4-29 – 36-32). Correspondent measured values are 0.153, 0.087, 0.055 and 0.006 m. Maximal SG level values after the reactor power decrease are 0.372, 0.374, 0.373, 0.315 m in calculation and 0.353, 0.436, 0.447, 0.349 m according to measurement data.

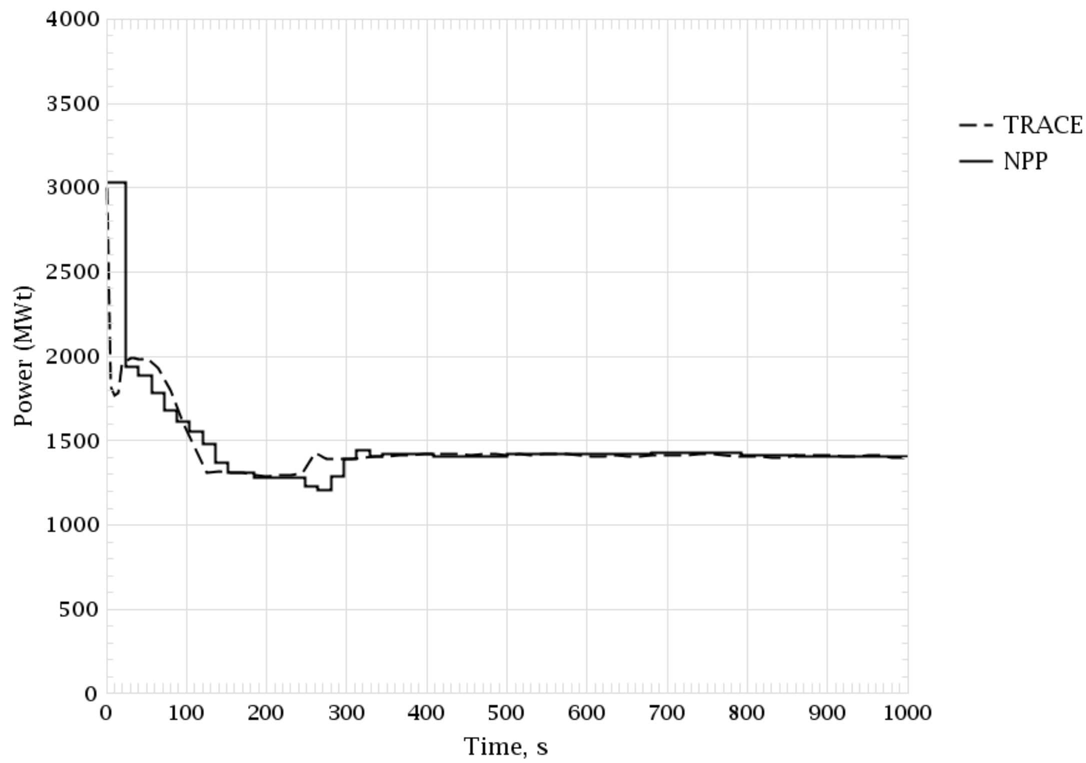

Figure 4-1 Reactor Power

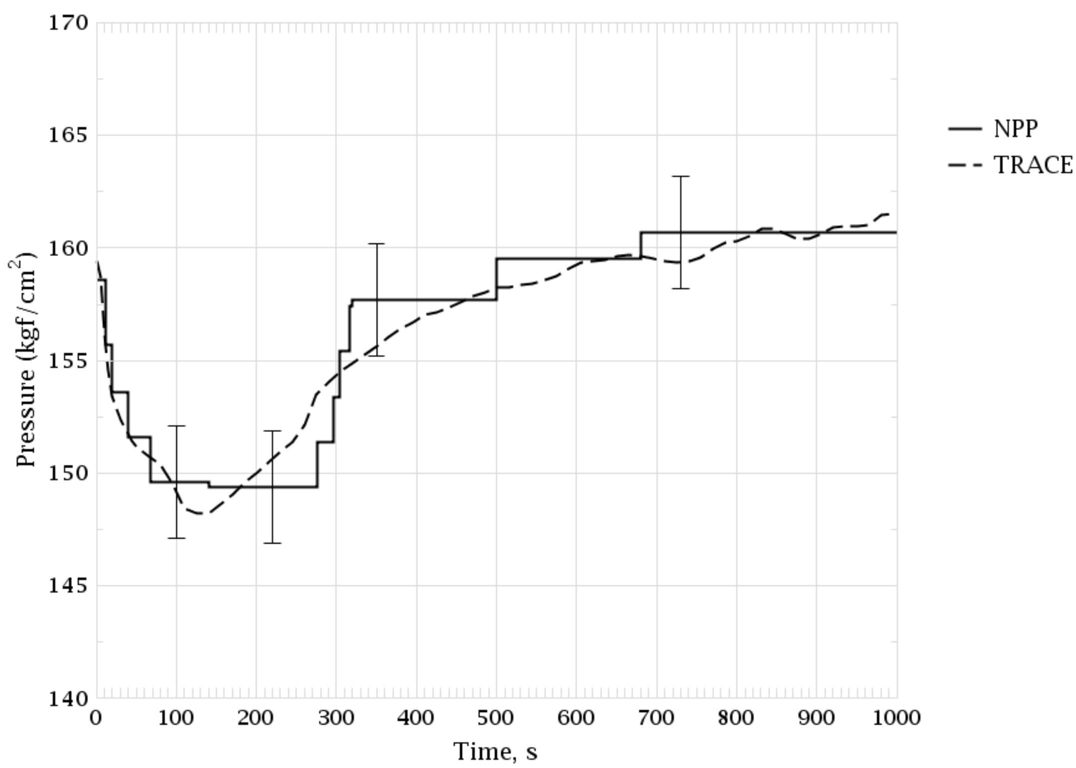

Figure 4-2 RCS Pressure

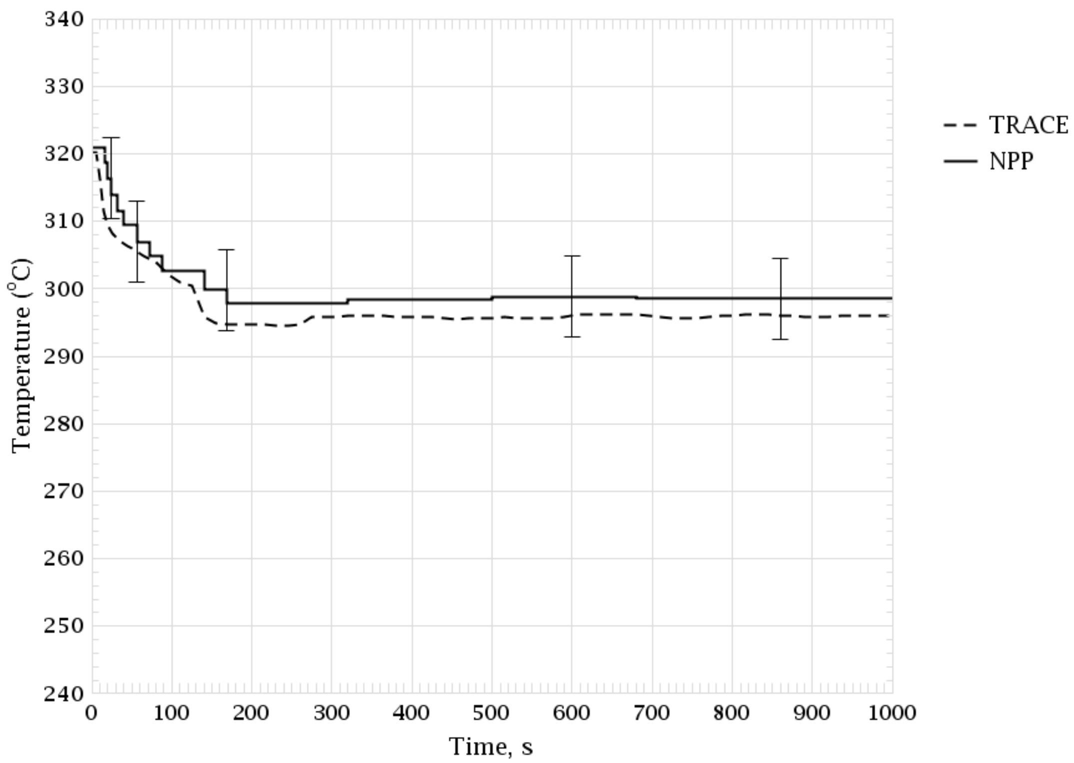

Figure 4-3 Hot Leg Coolant Temperature, Loop No.1

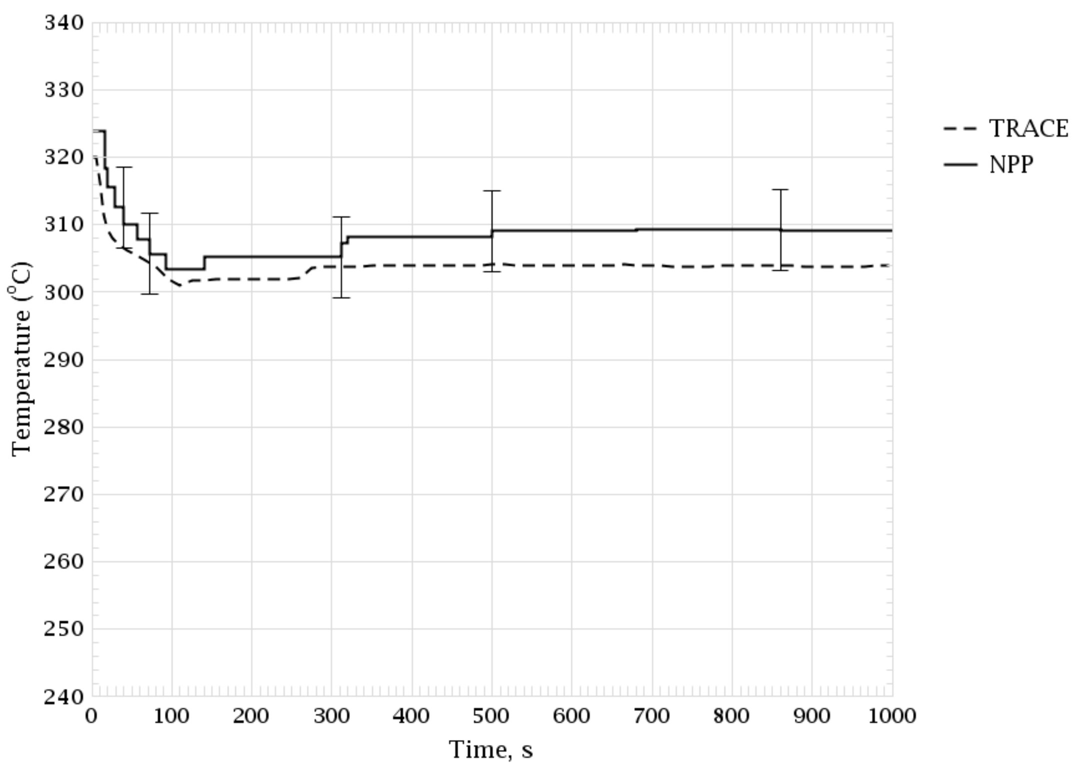

Figure 4-4 Hot Leg Coolant Temperature, Loop No.2

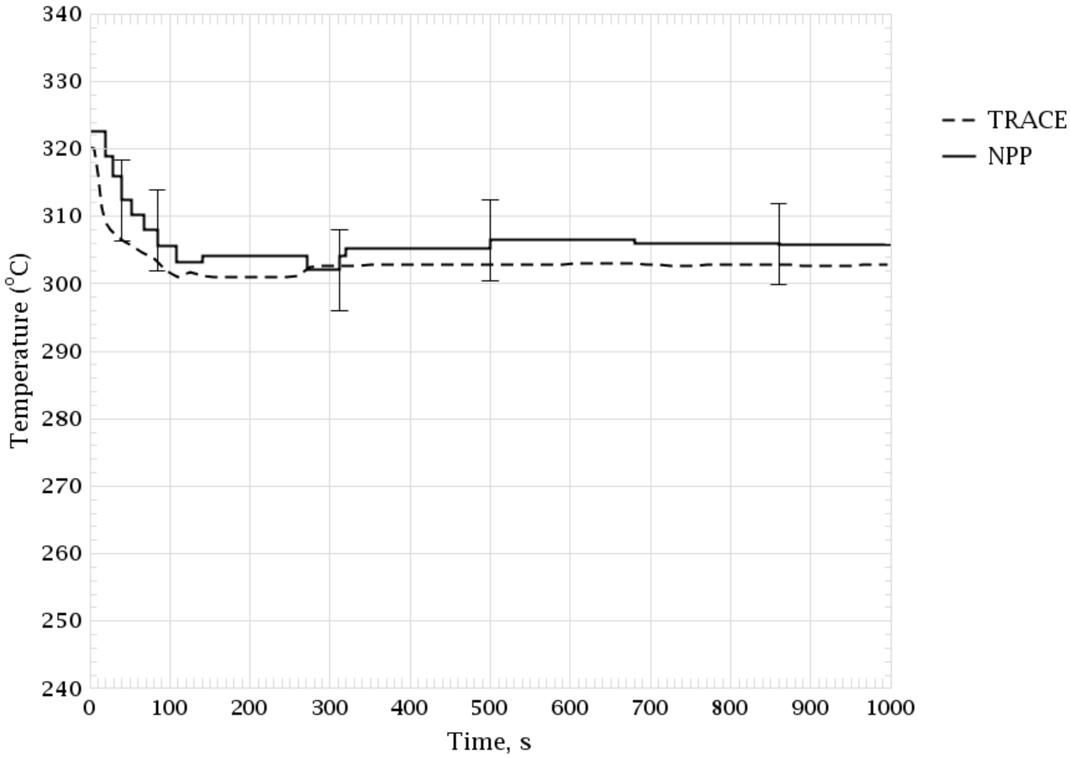

Figure 4-5 Hot Leg Coolant Temperature, Loop No.3

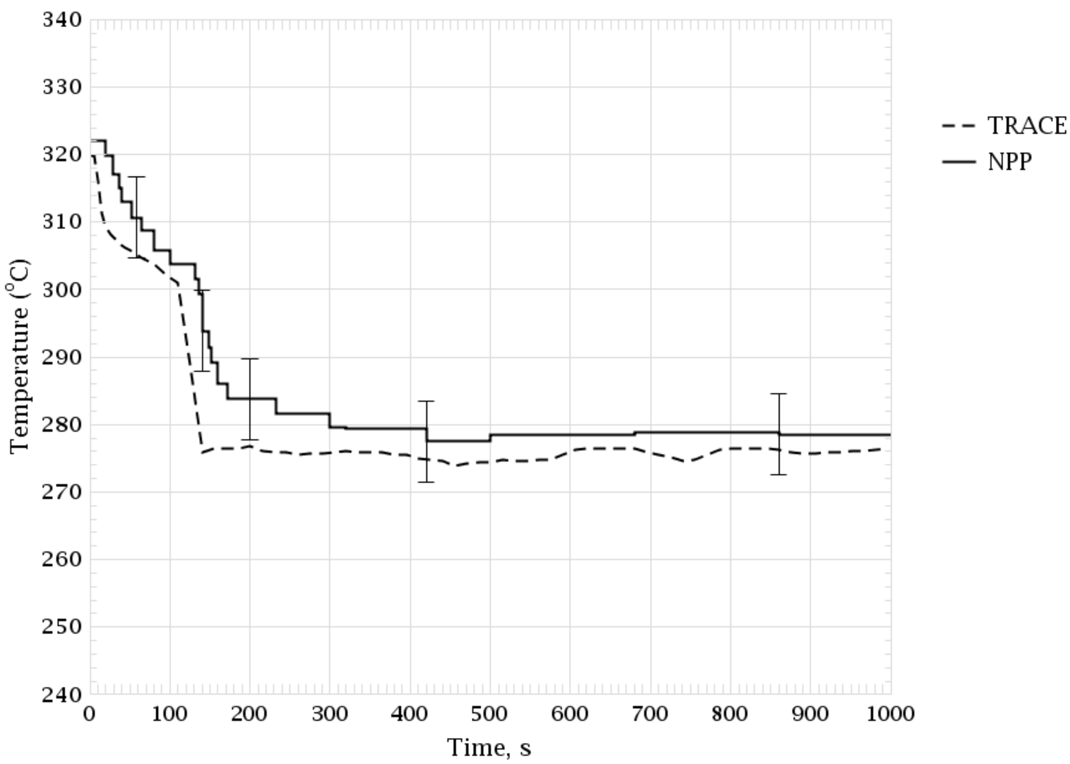

Figure 4-6 Hot Leg Coolant Temperature, Loop No.4

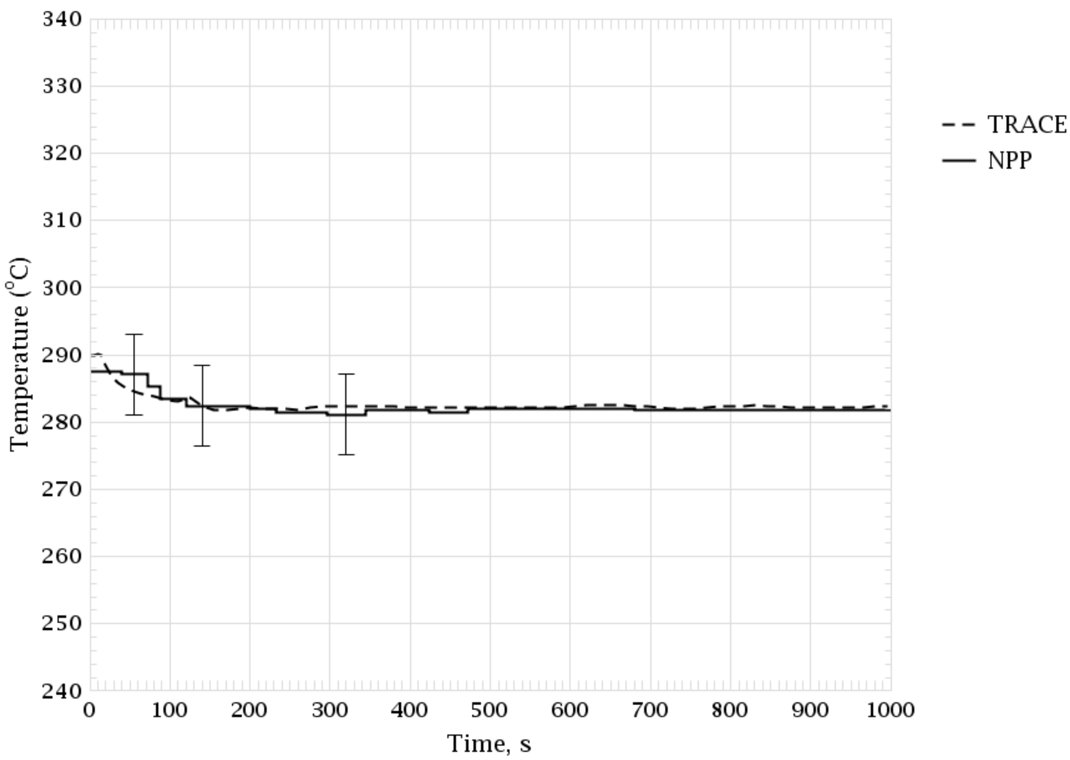

Figure 4-7 Cold Leg Coolant Temperature, Loop No.1

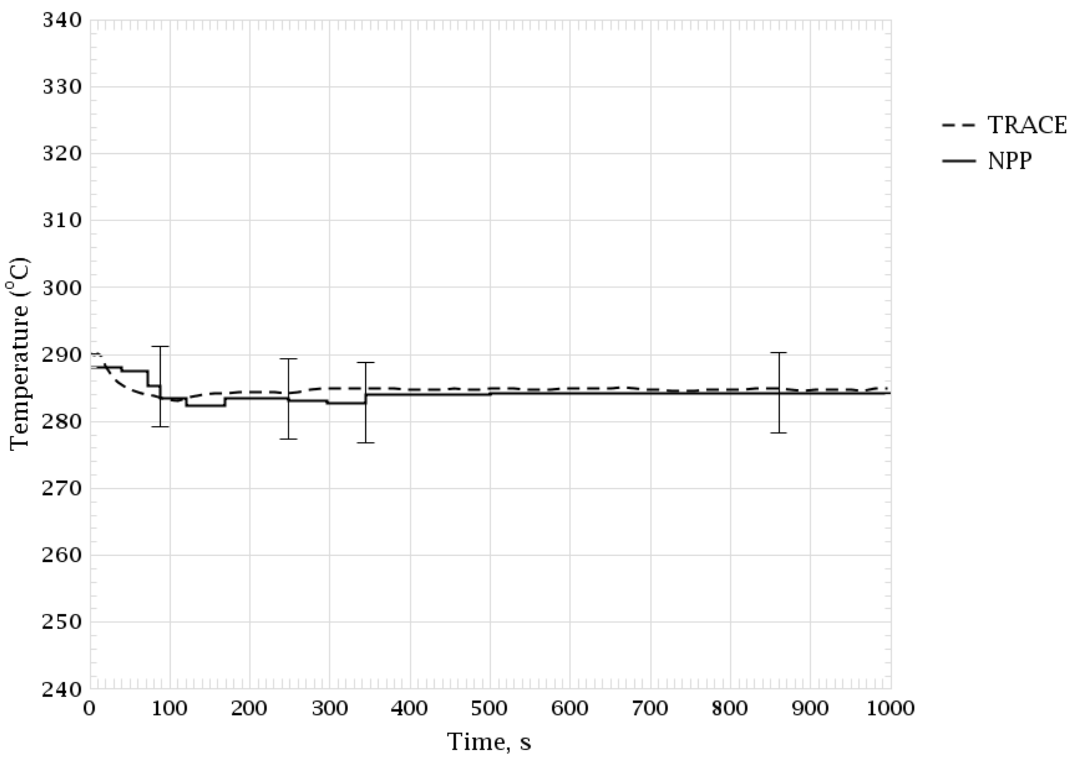

Figure 4-8 Cold Leg Coolant Temperature, Loop No.2

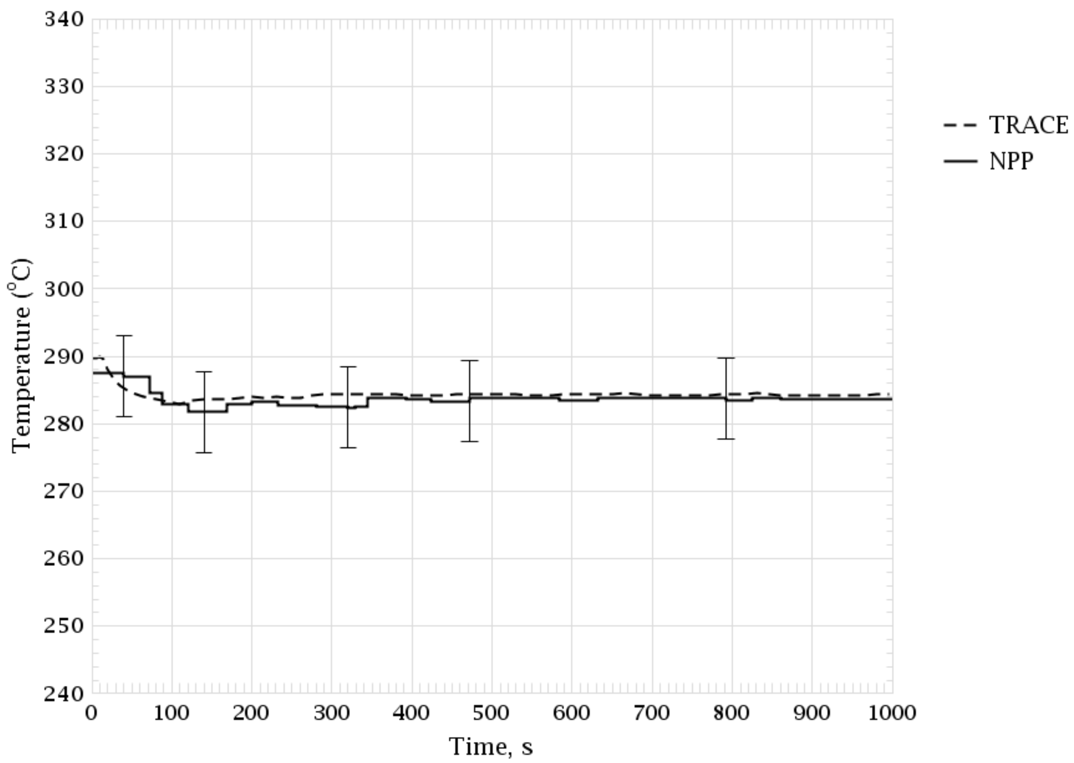

Figure 4-9 Cold Leg Coolant Temperature, Loop No.3

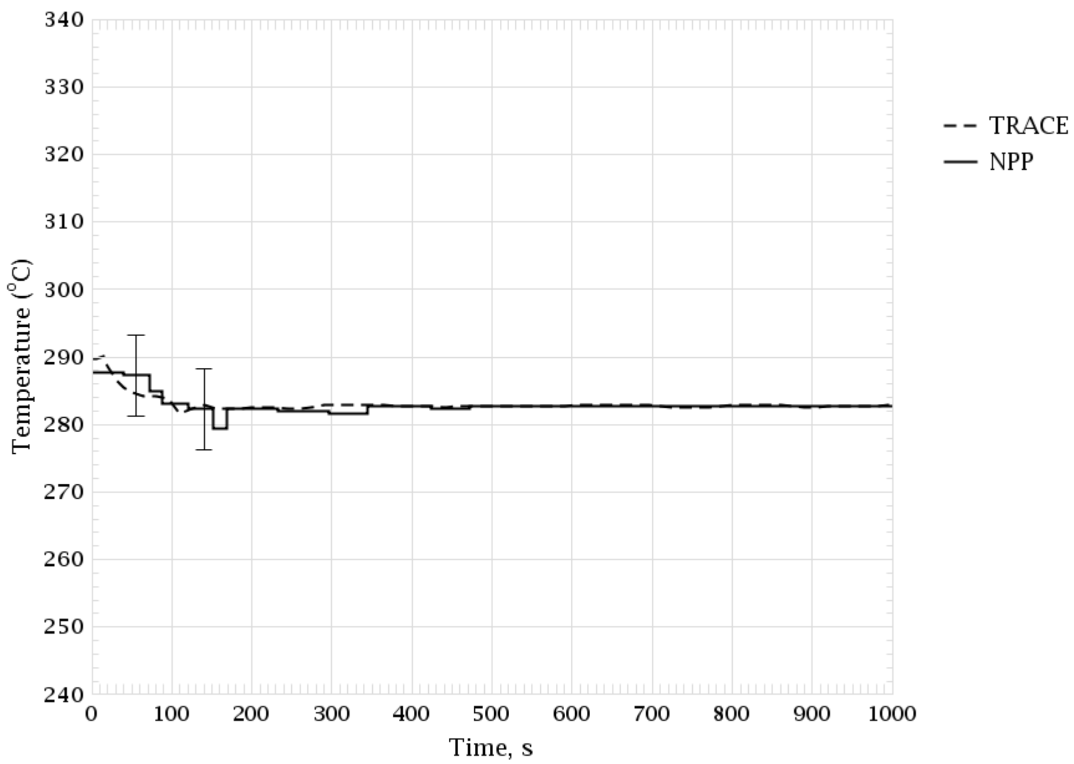

Figure 4-10 Cold Leg Coolant Temperature, Loop No.4

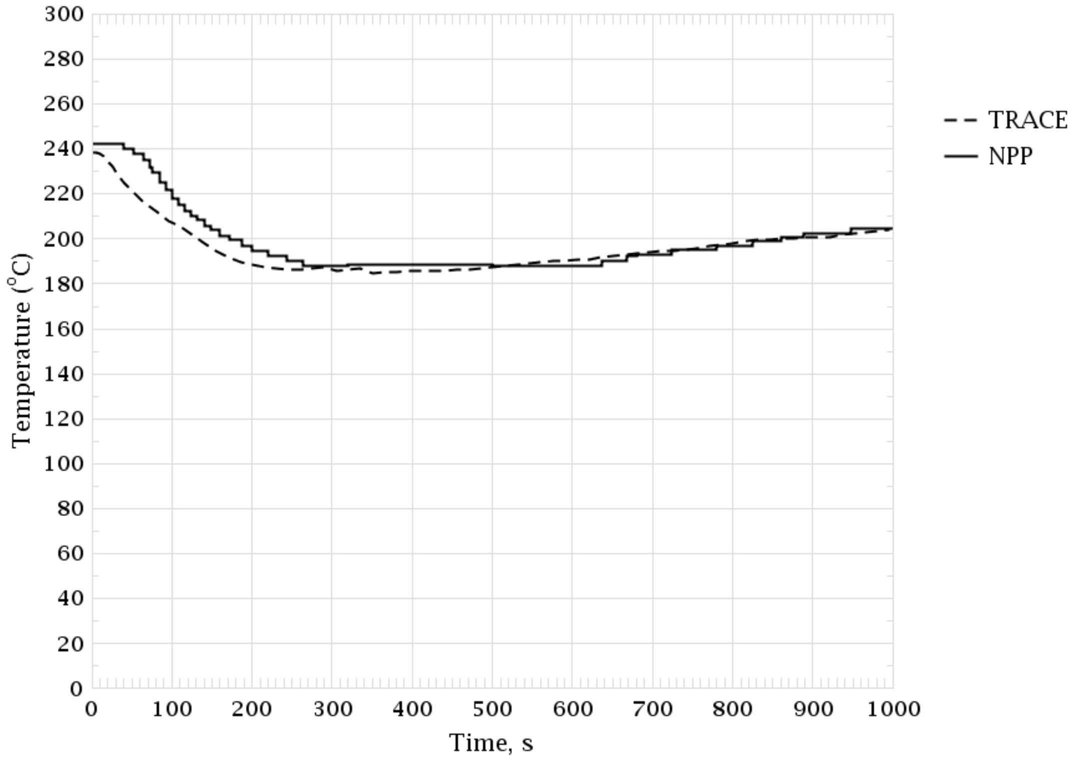

Figure 4-11 Make-up Temperature

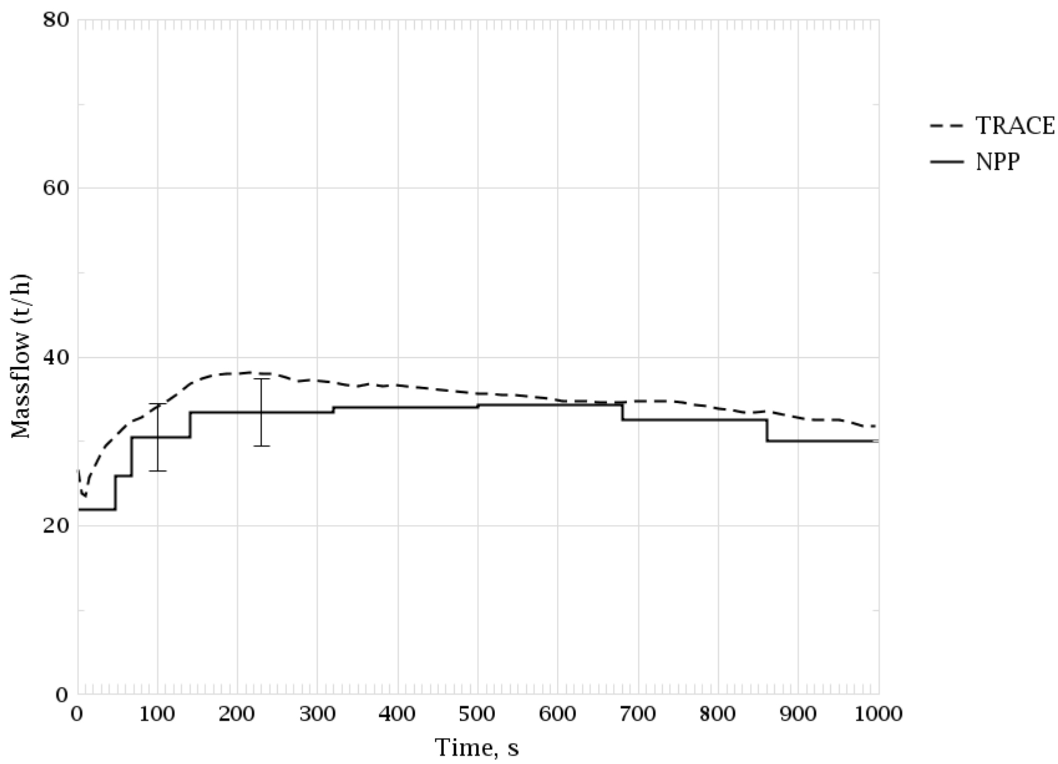

Figure 4-12 Make-up Mass Flow Rate

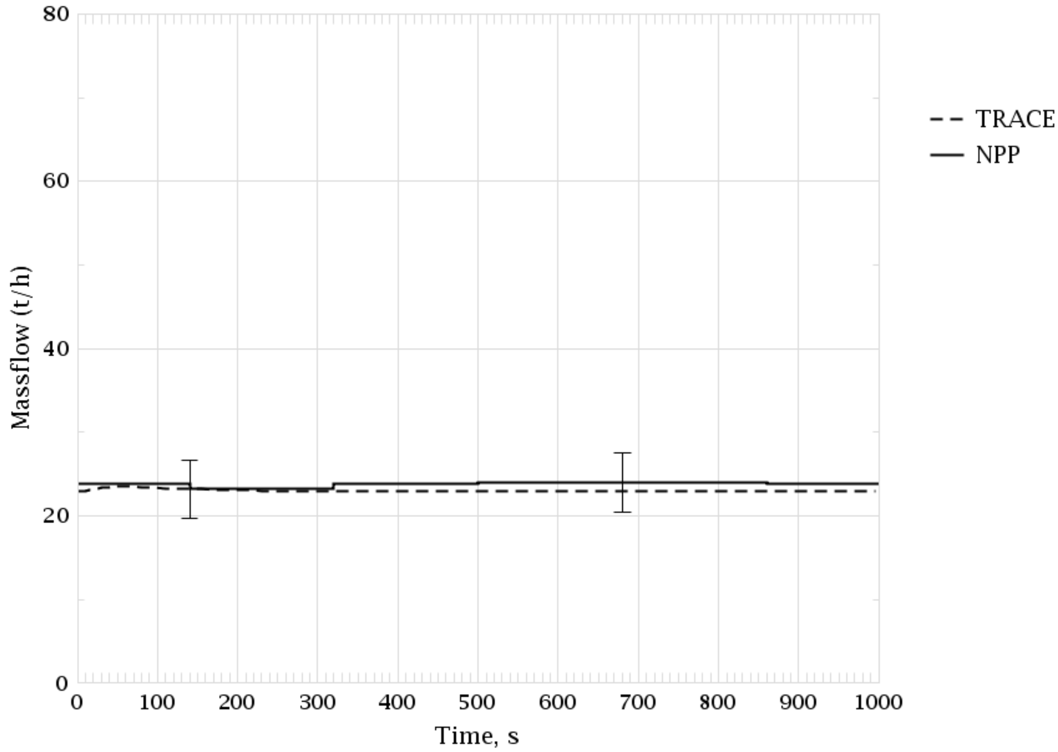

Figure 4-13 Let-down Mass Flow Rate

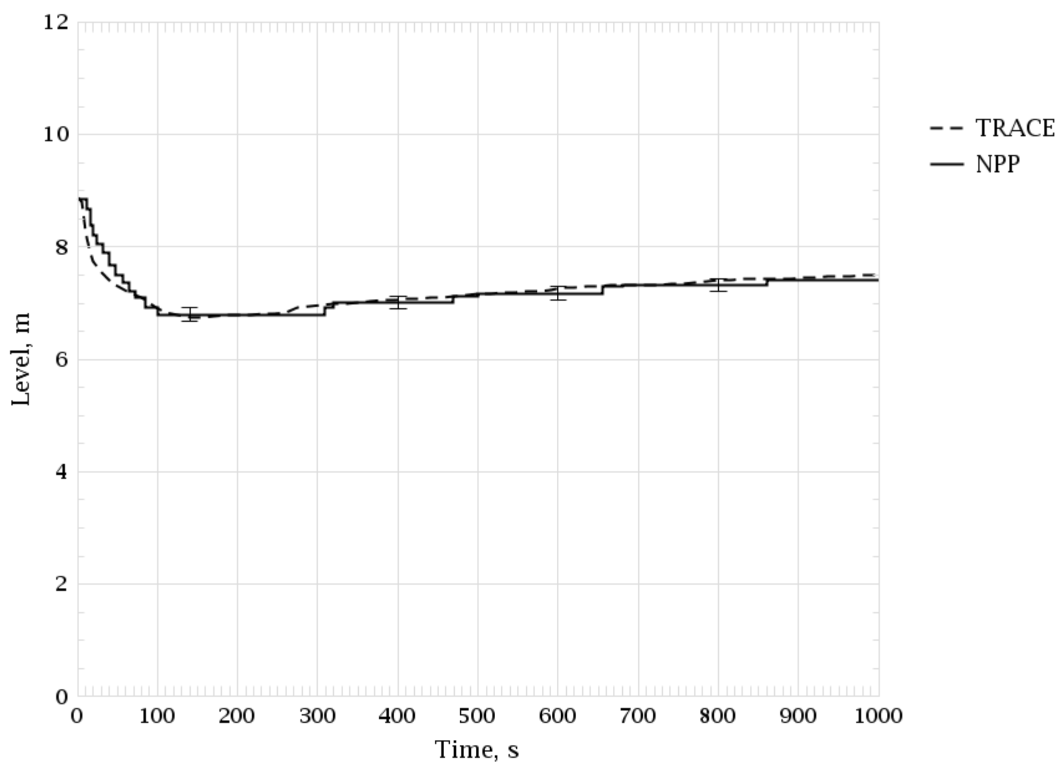

Figure 4-14 PRZ Level

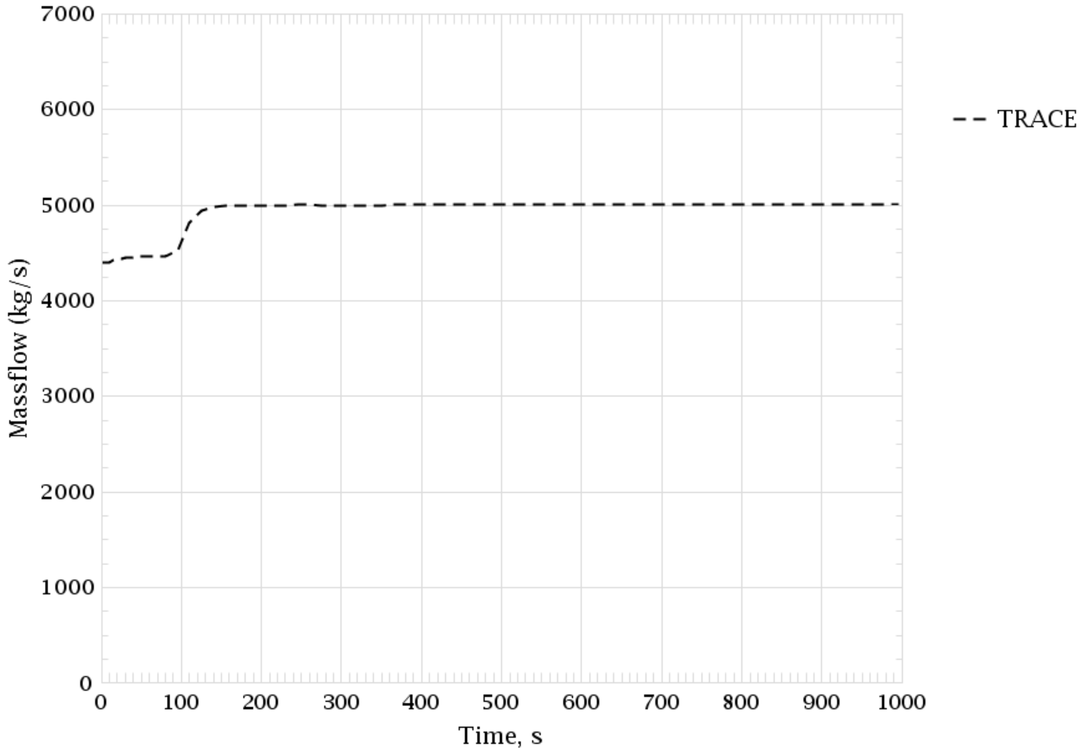

Figure 4-15 RCP Mass Flow Rate, Loop No.1

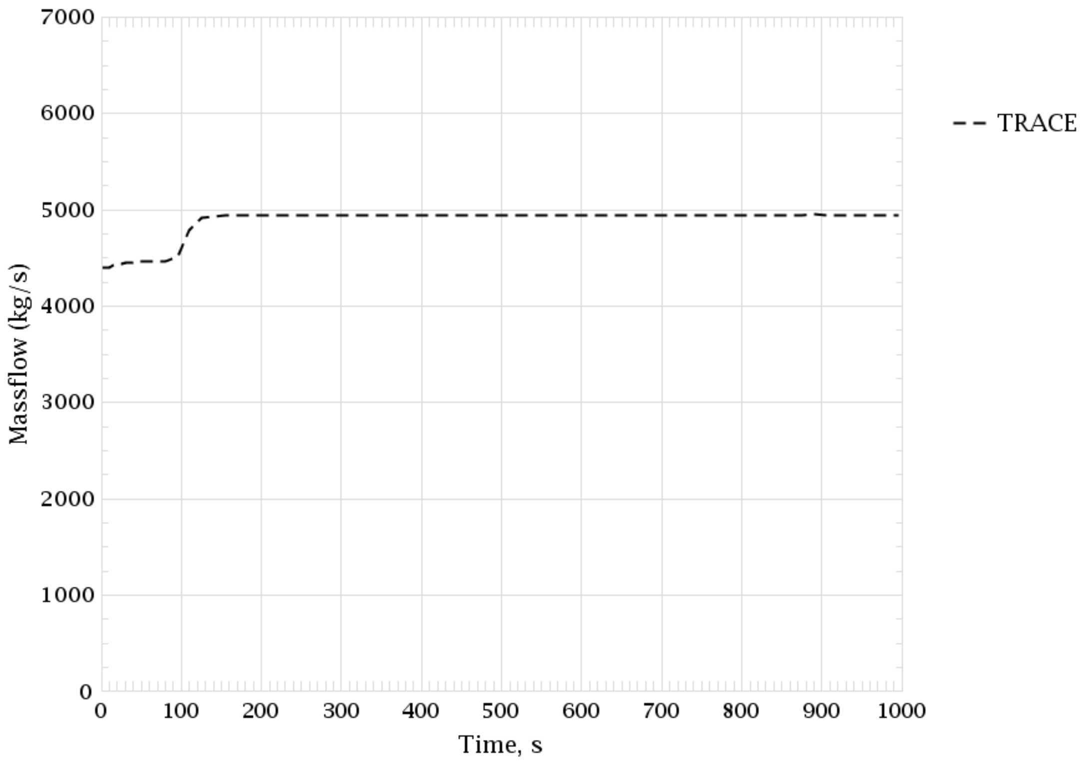

Figure 4-16 RCP Mass Flow Rate, Loop No.2

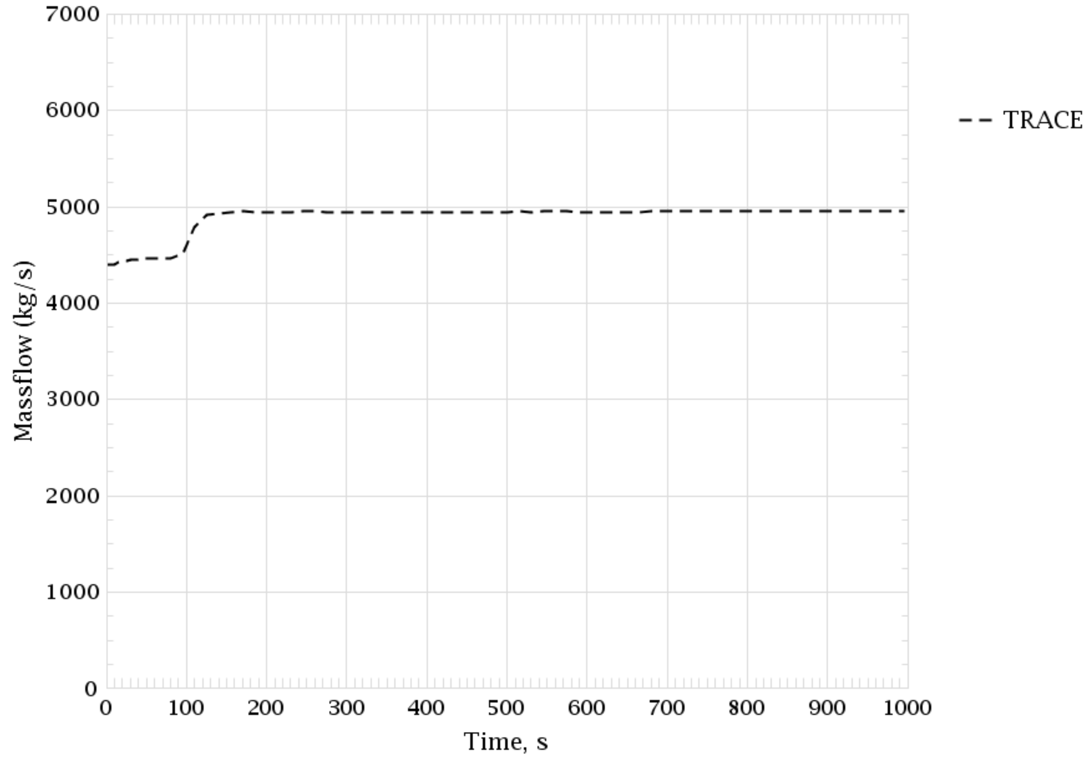

Figure 4-17 RCP Mass Flow Rate, Loop No.3

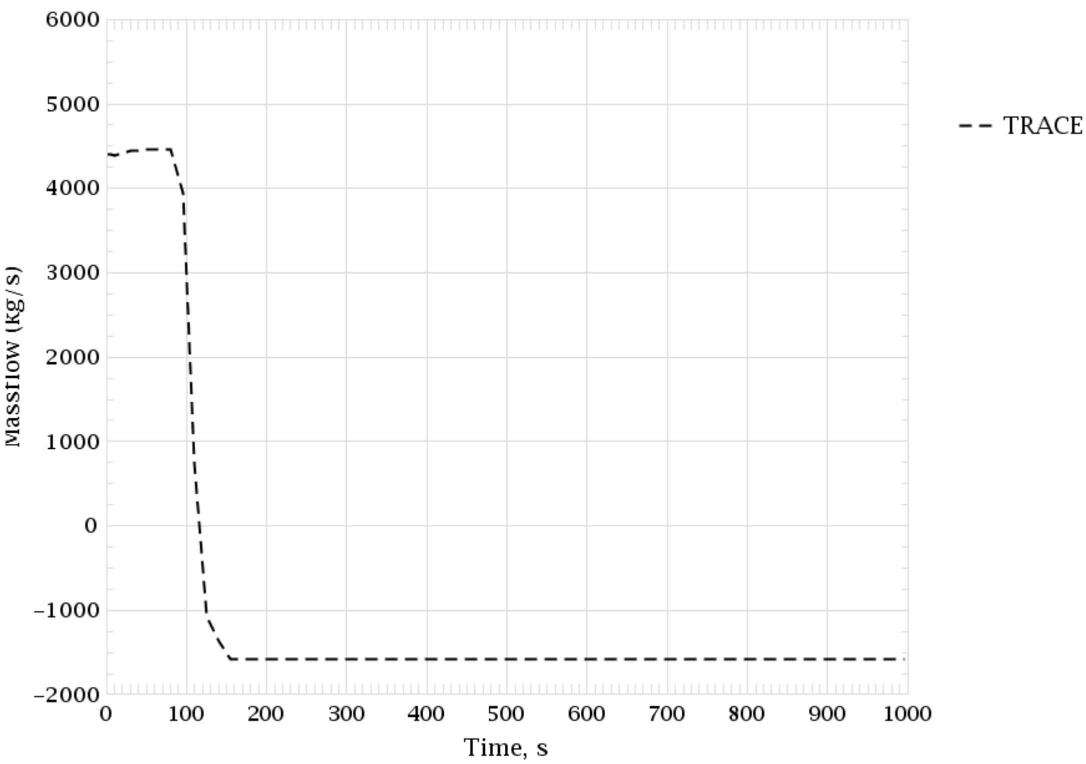

Figure 4-18 RCP Mass Flow Rate, Loop No.4

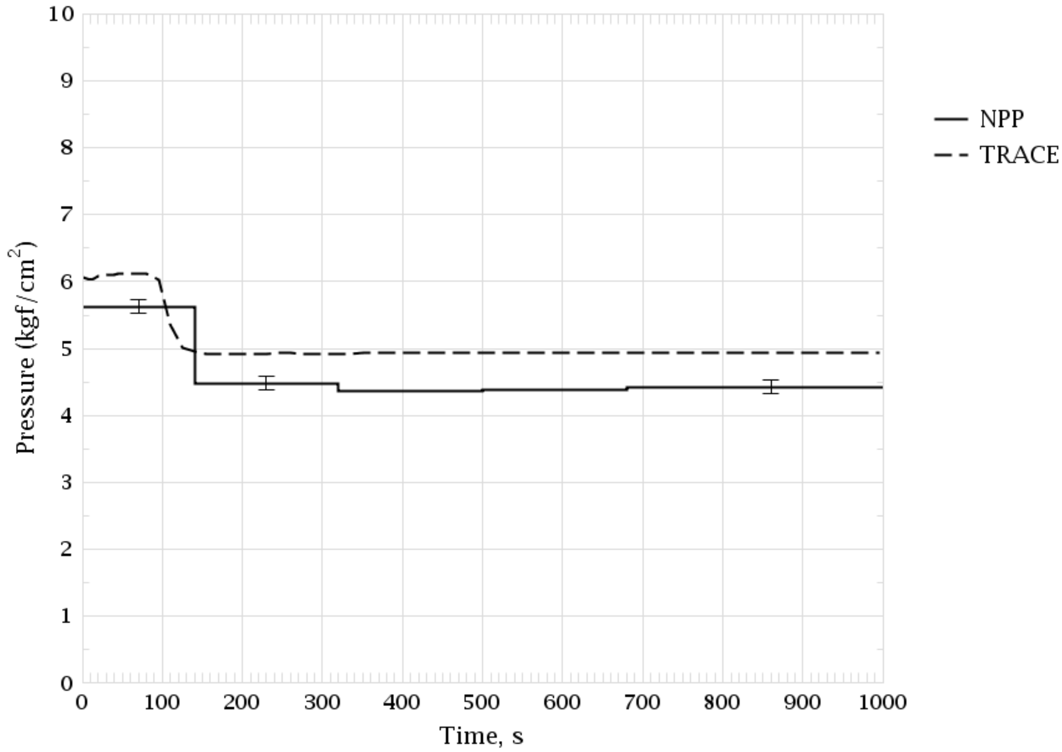

Figure 4-19 Pressure Difference at RCP-1

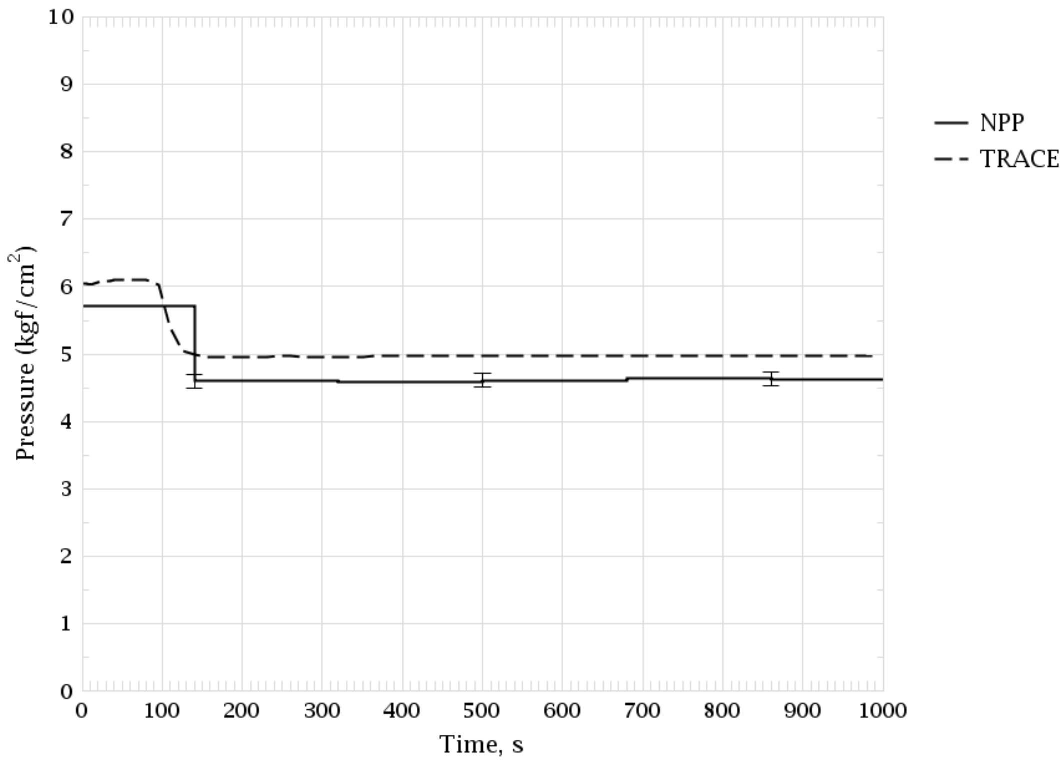

Figure 4-20 Pressure Difference at RCP-2

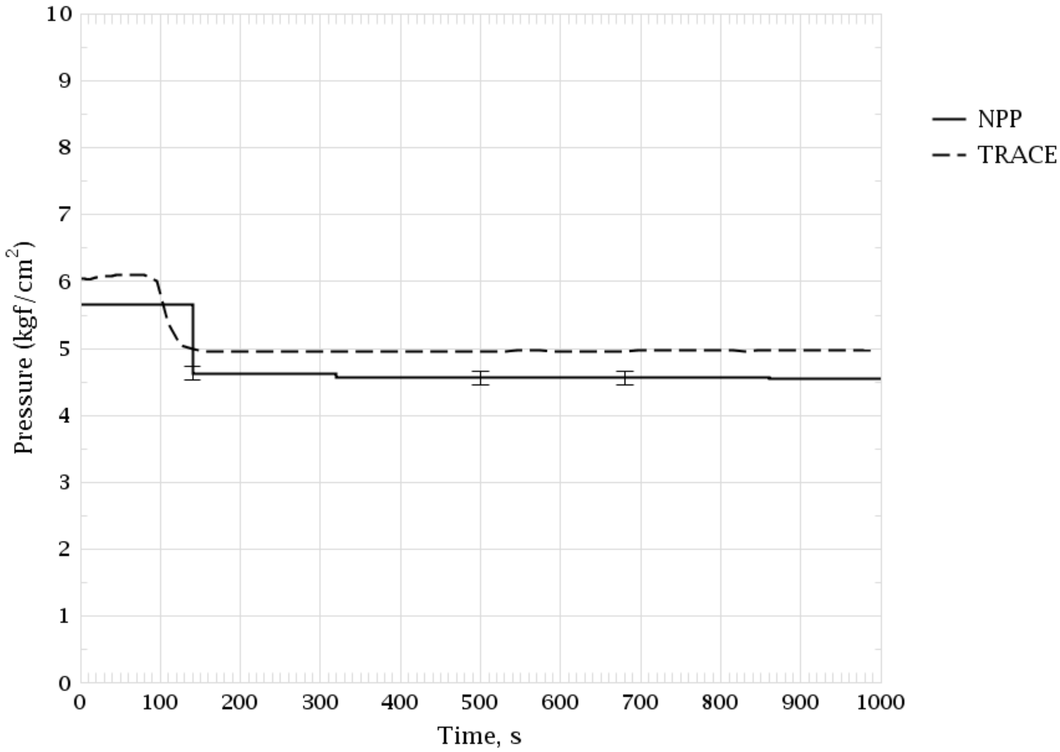

Figure 4-21 Pressure Difference at RCP-3

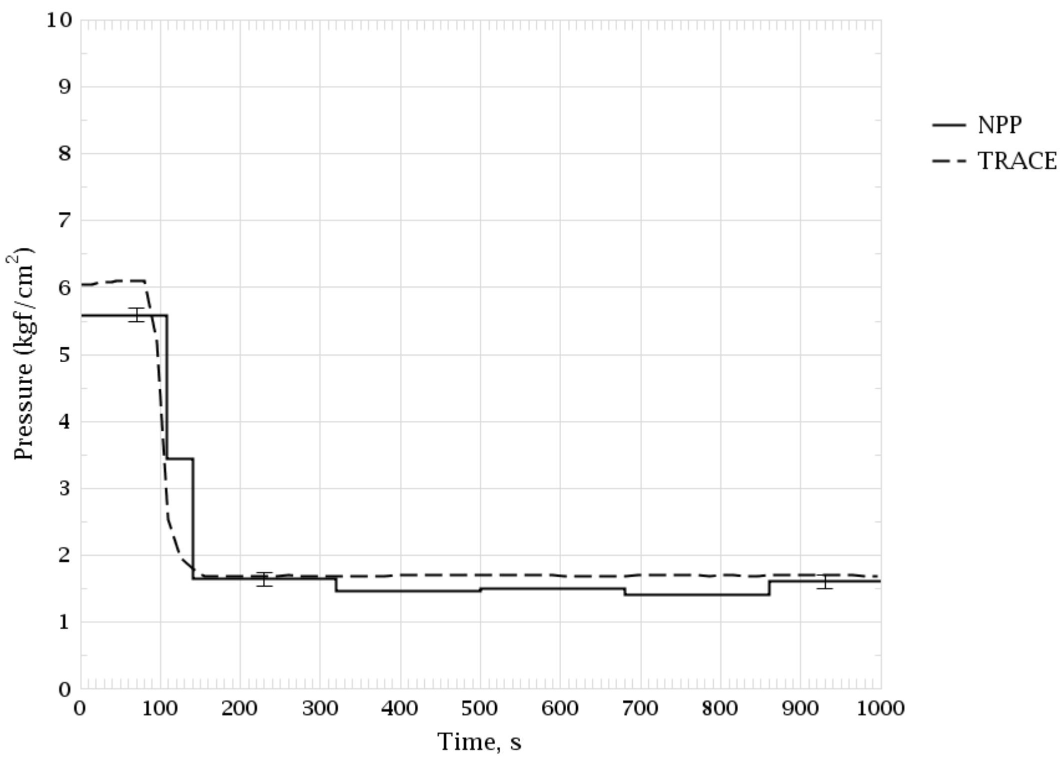

Figure 4-22 Pressure Difference at RCP-4

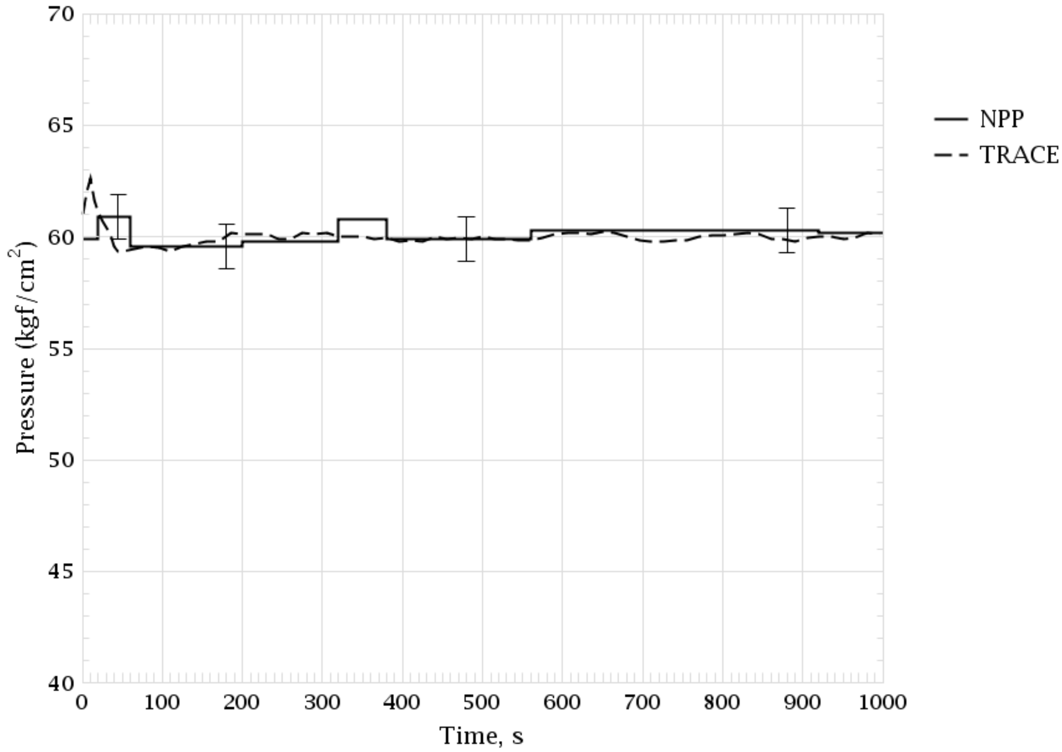

Figure 4-23 MSH Pressure

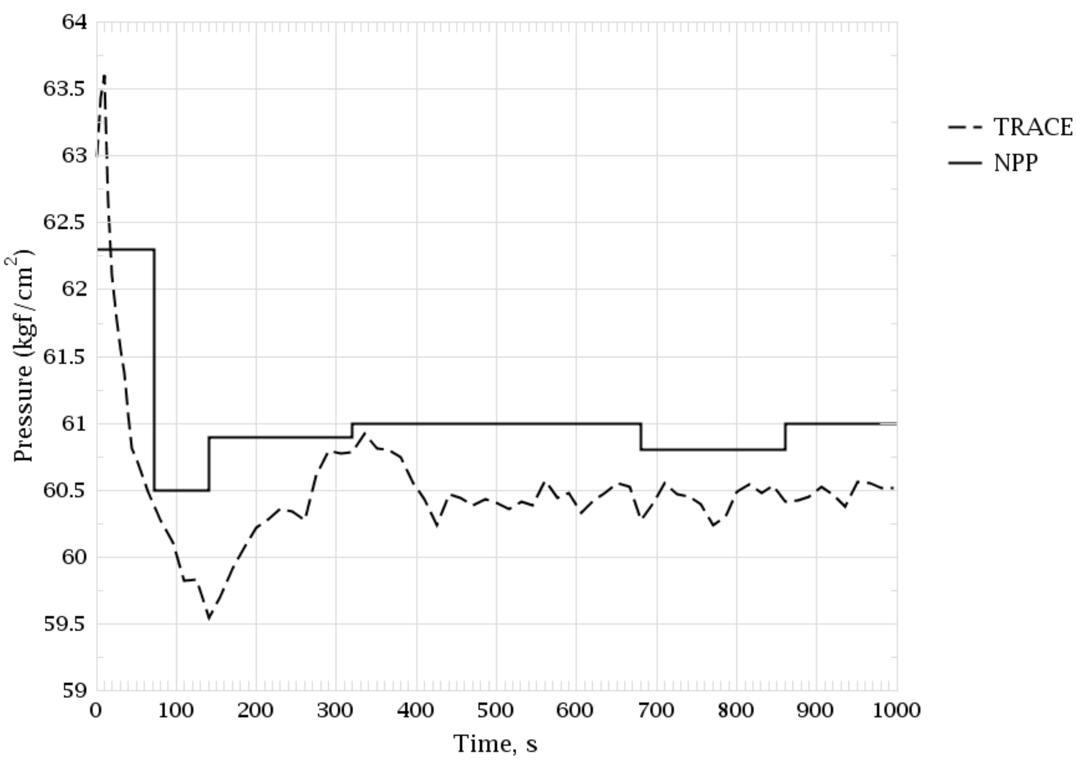

Figure 4-24 SG-1 Pressure

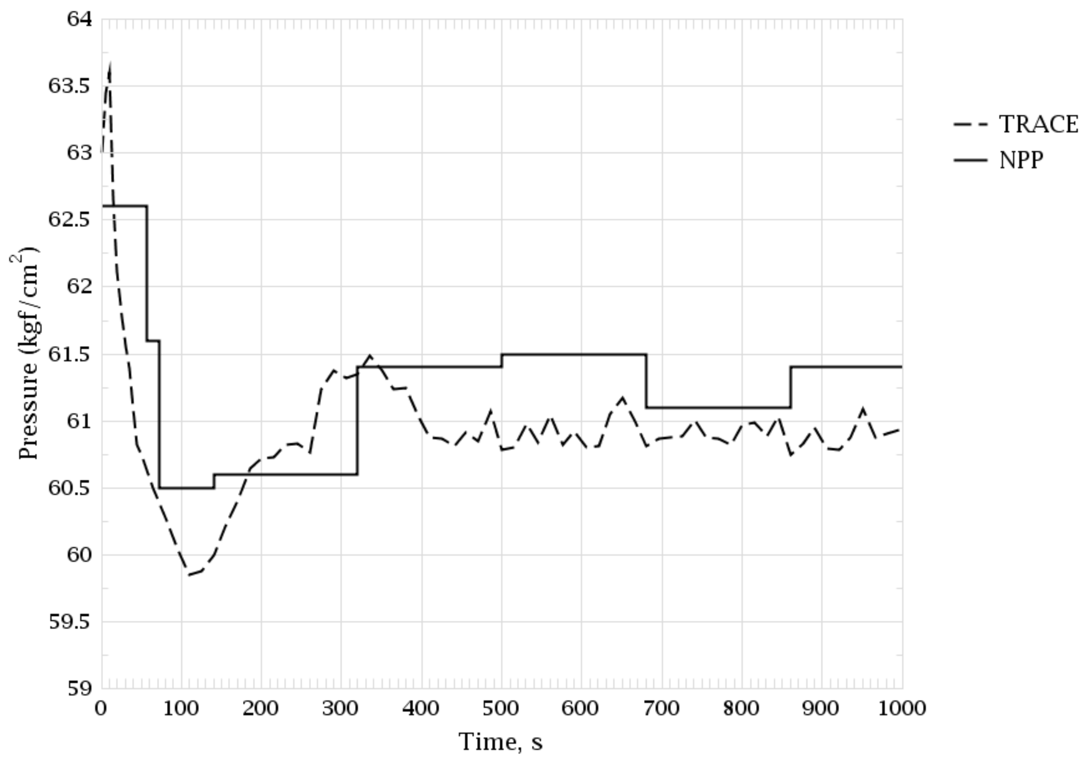

Figure 4-25 SG-2 Pressure

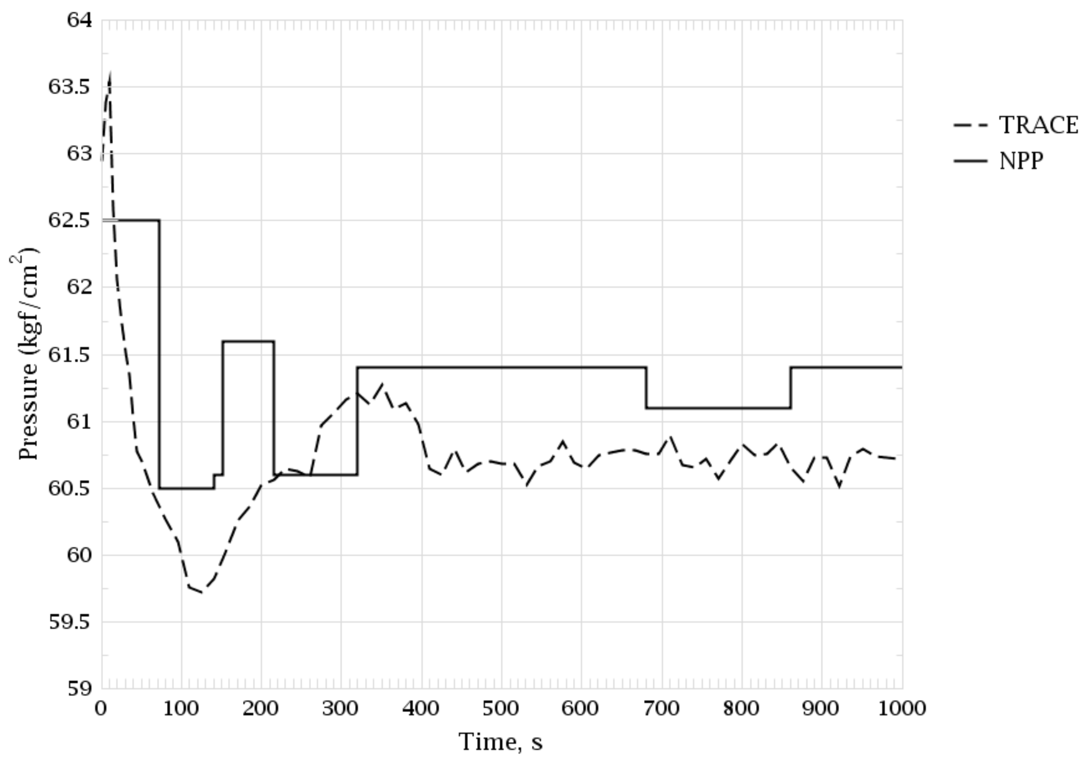

Figure 4-26 SG-3 Pressure

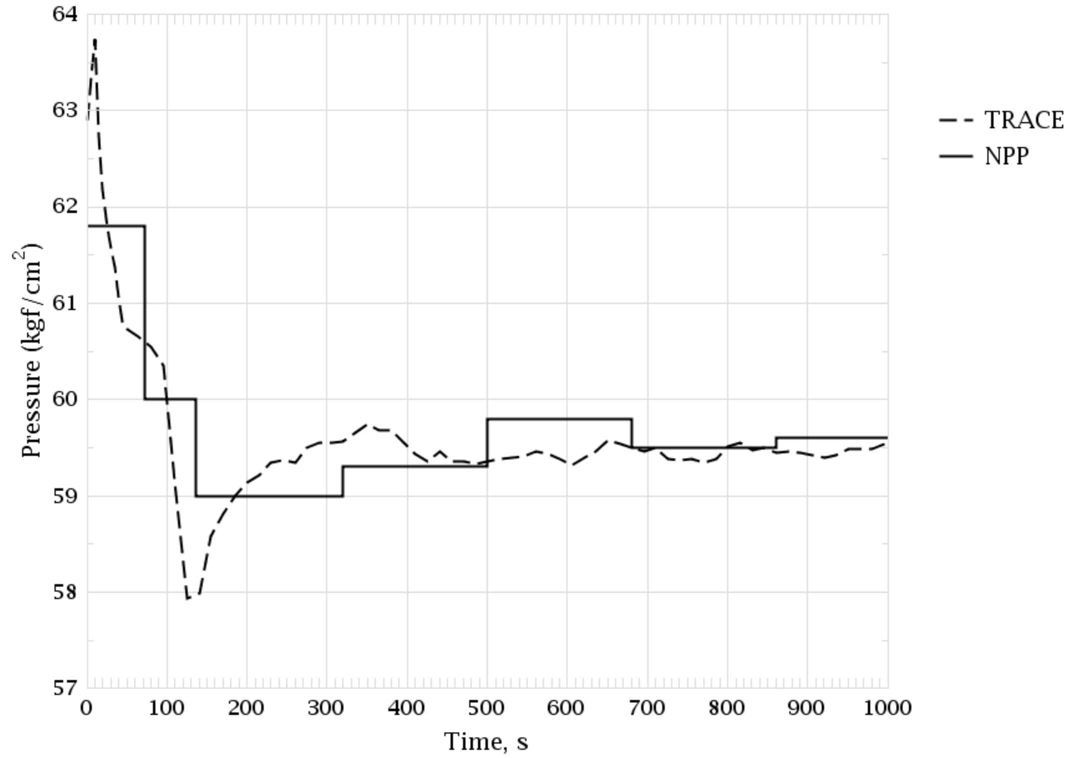

Figure 4-27 SG-4 Pressure

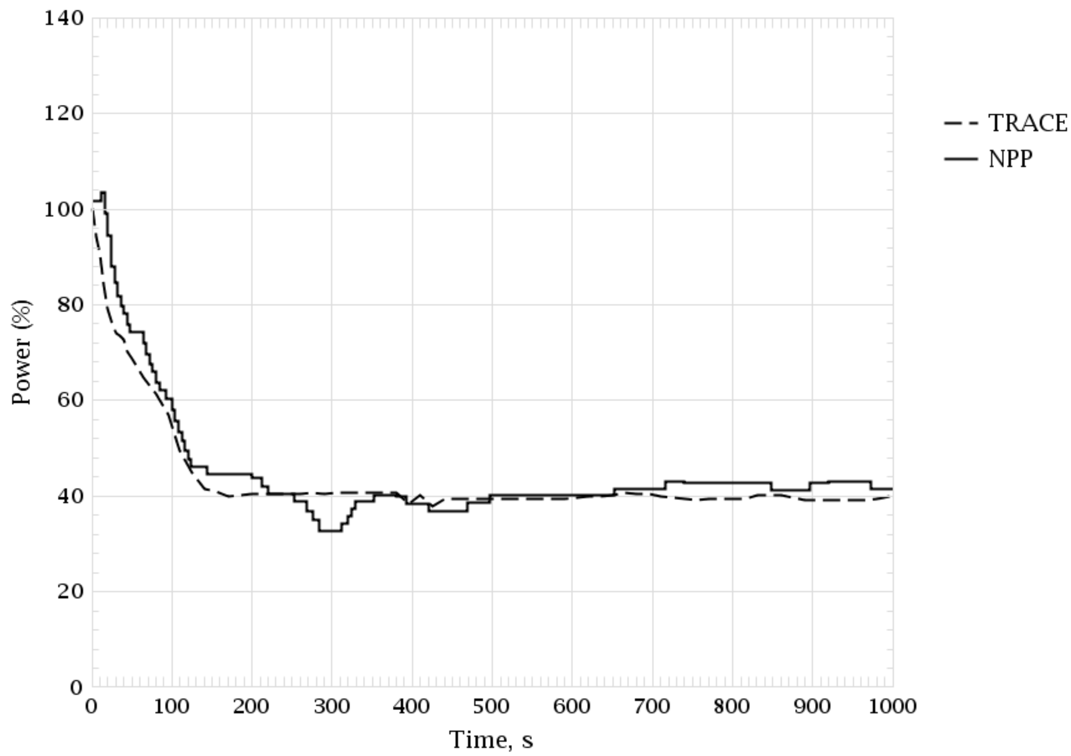

Figure 4-28 Relative Turbine Power

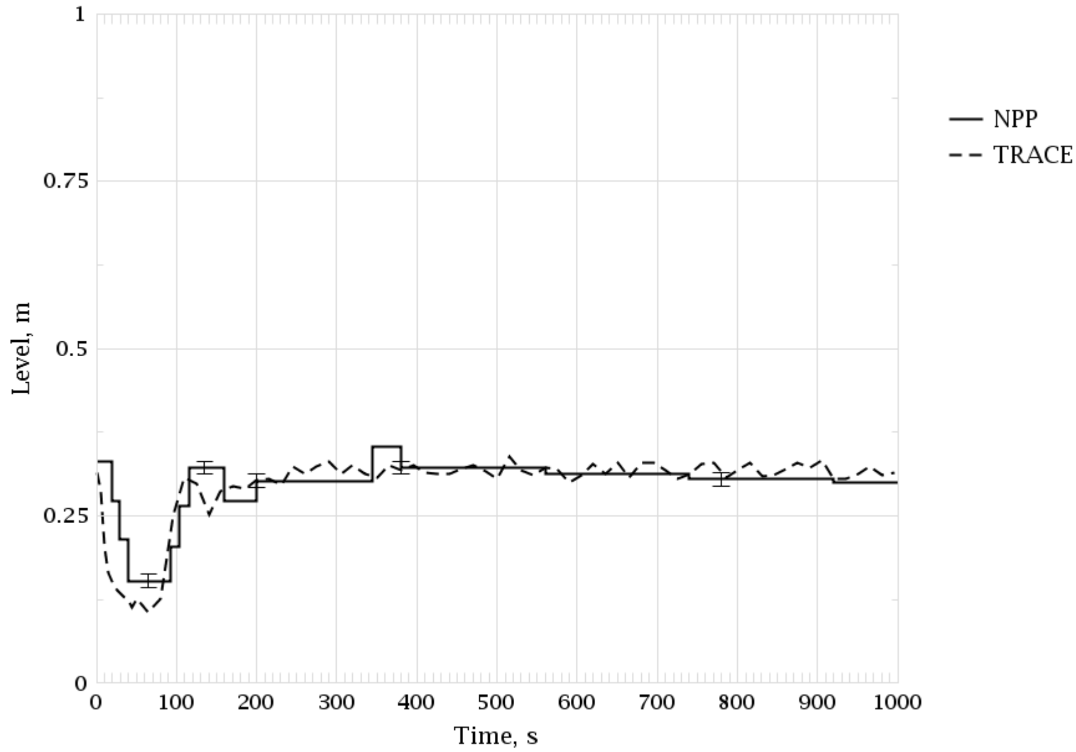

Figure 4-29 SG-1 Level (Narrow Range)

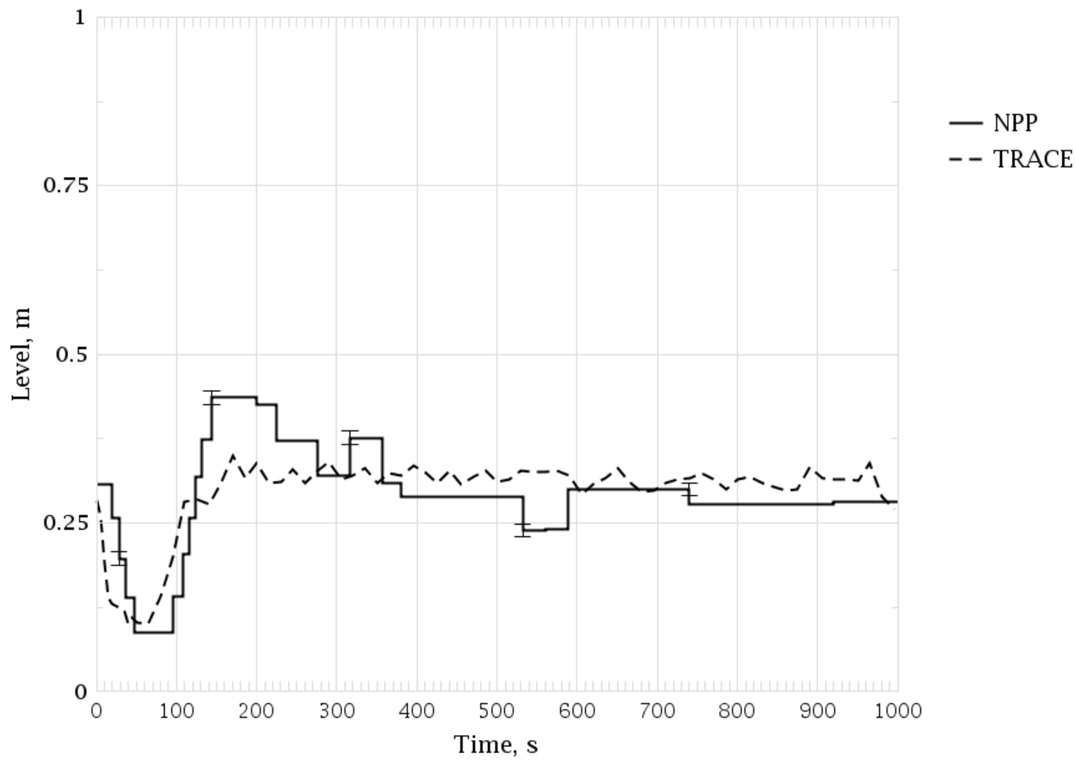

Figure 4-30 SG-2 Level (Narrow Range)

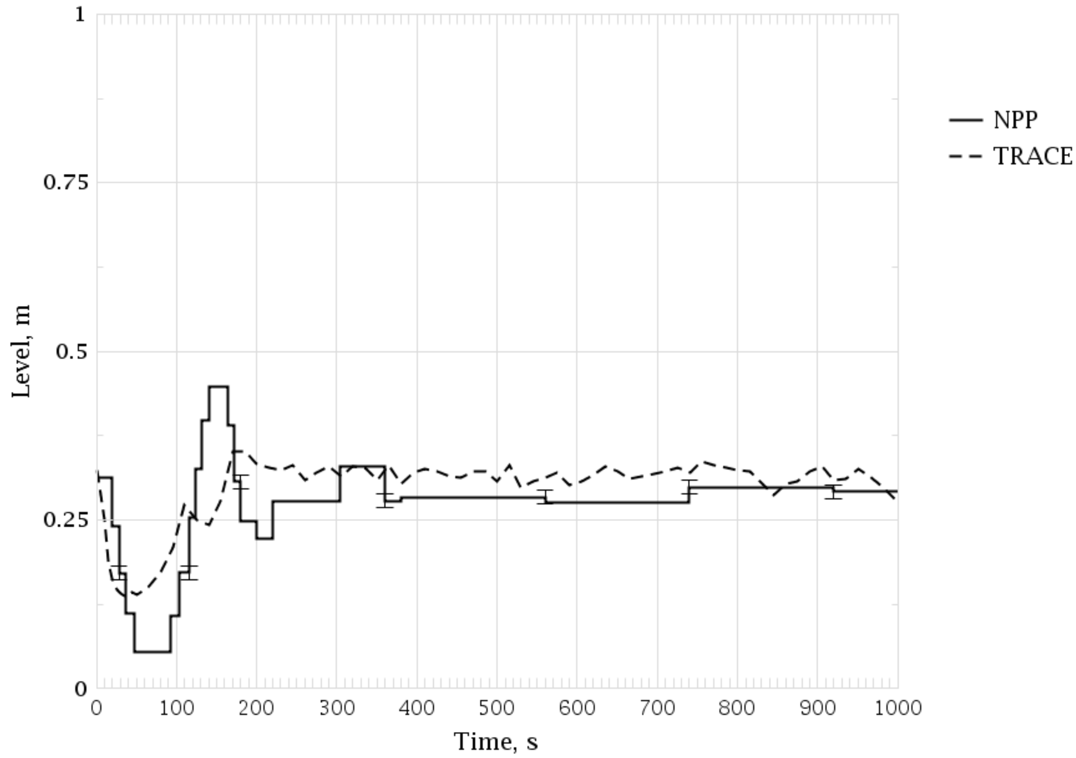

Figure 4-31 SG-3 Level (Narrow Range)

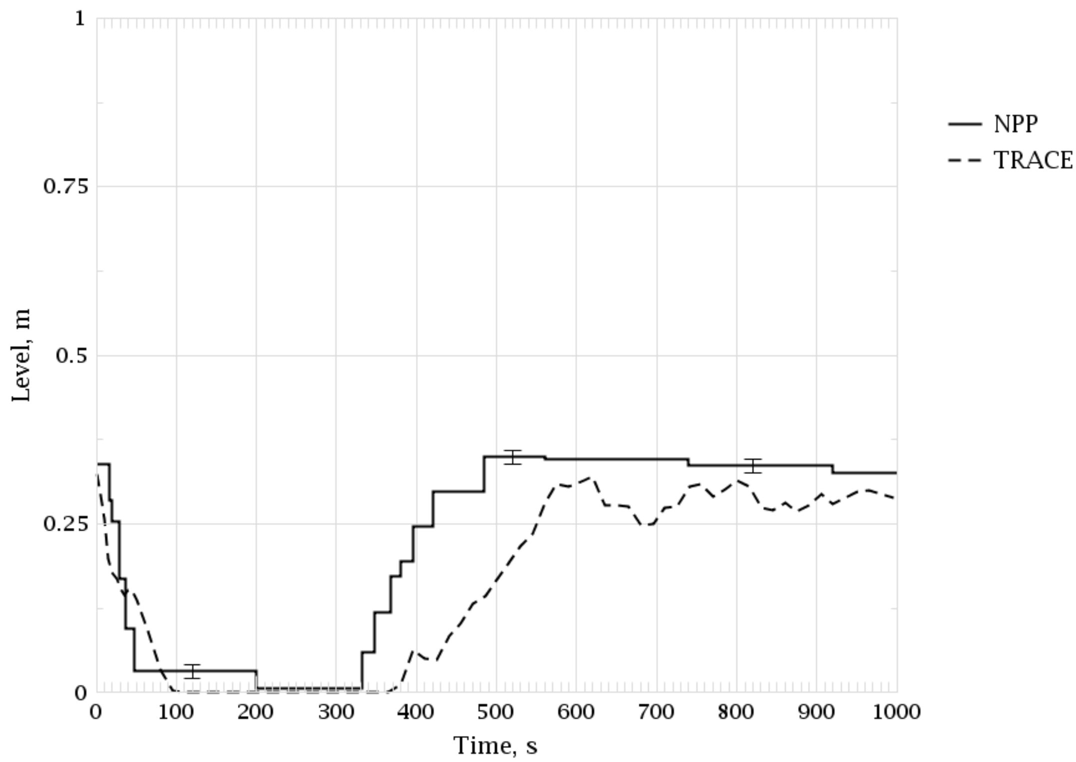

Figure 4-32 SG-4 Level (Narrow Range)

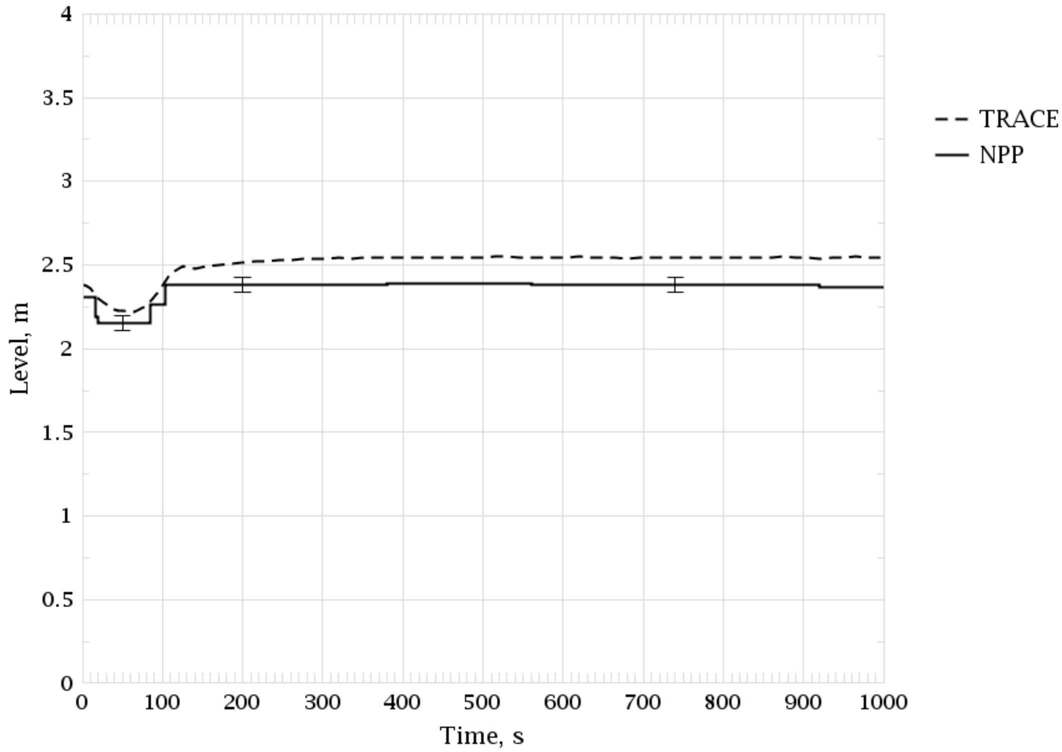

Figure 4-33 SG-1 Level (Wide Range)

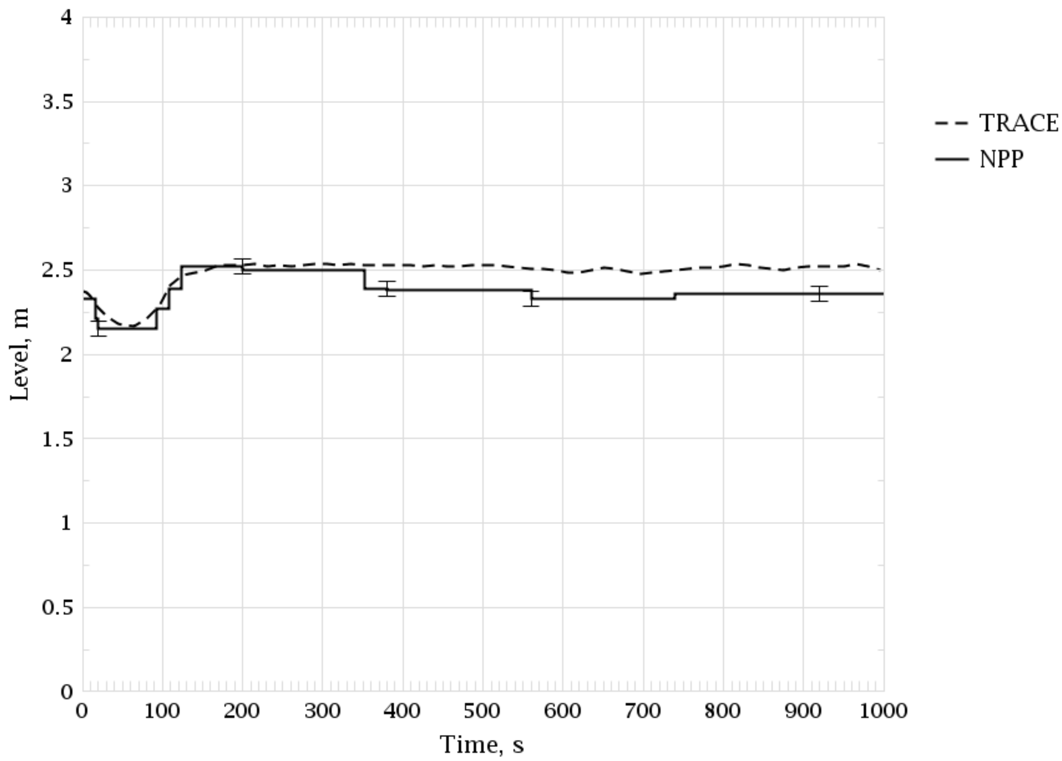

Figure 4-34 SG-2 Level (Wide Range)

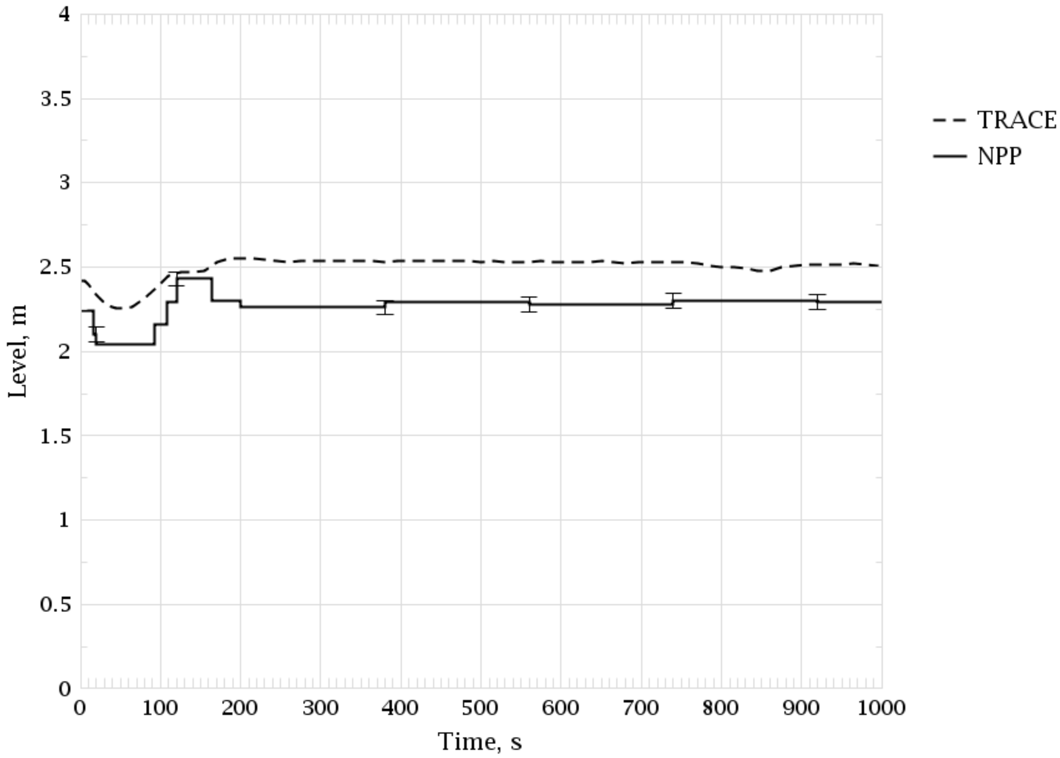

Figure 4-35 SG-3 Level (Wide Range)

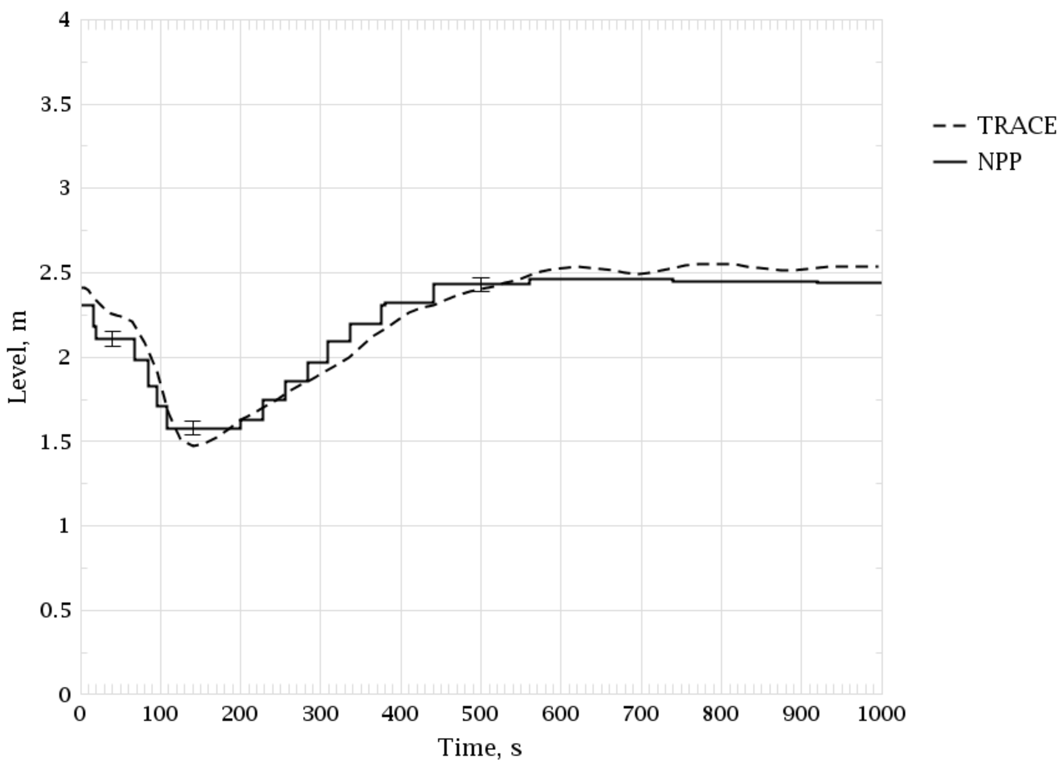

Figure 4-36 SG-4 Level (Wide Range)

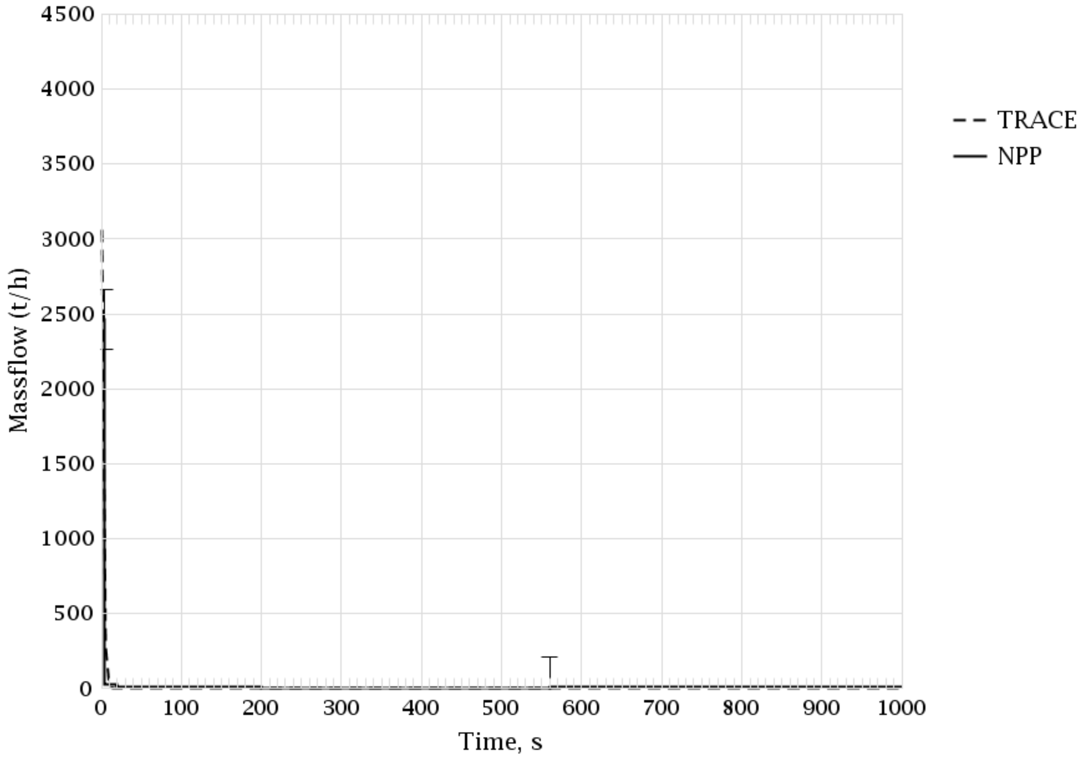

Figure 4-37 MFW Pump No.1 Flow Rate

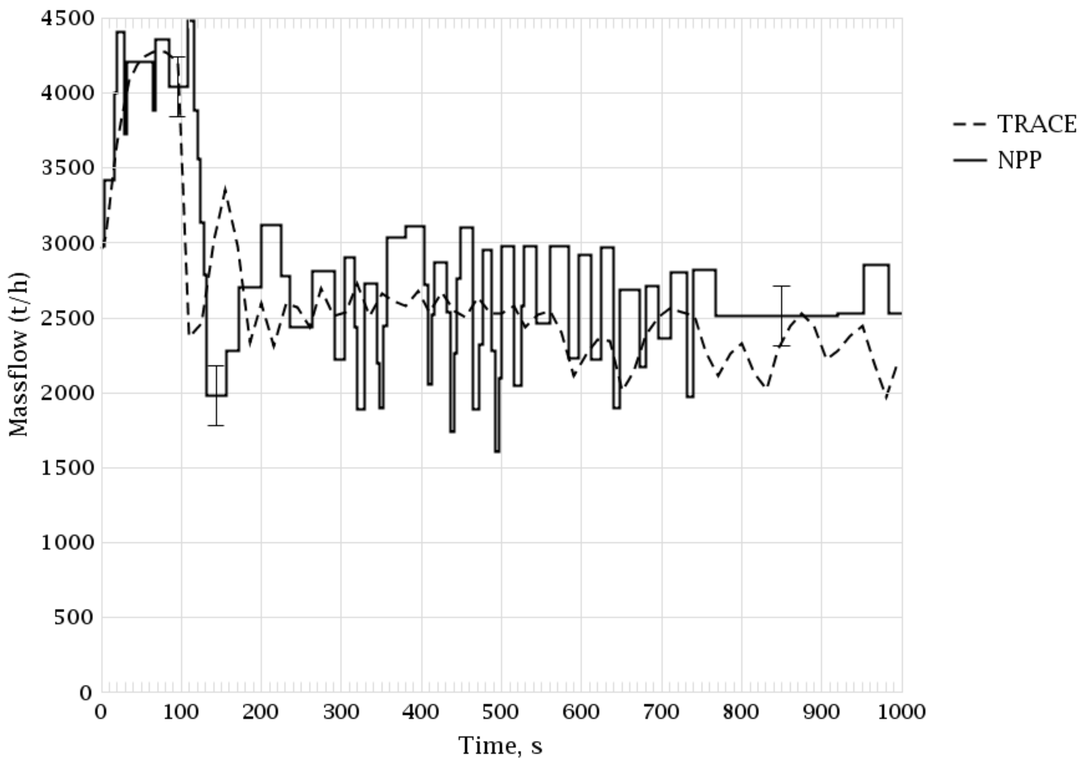

Figure 4-38 MFW Pump No.2 Flow Rate

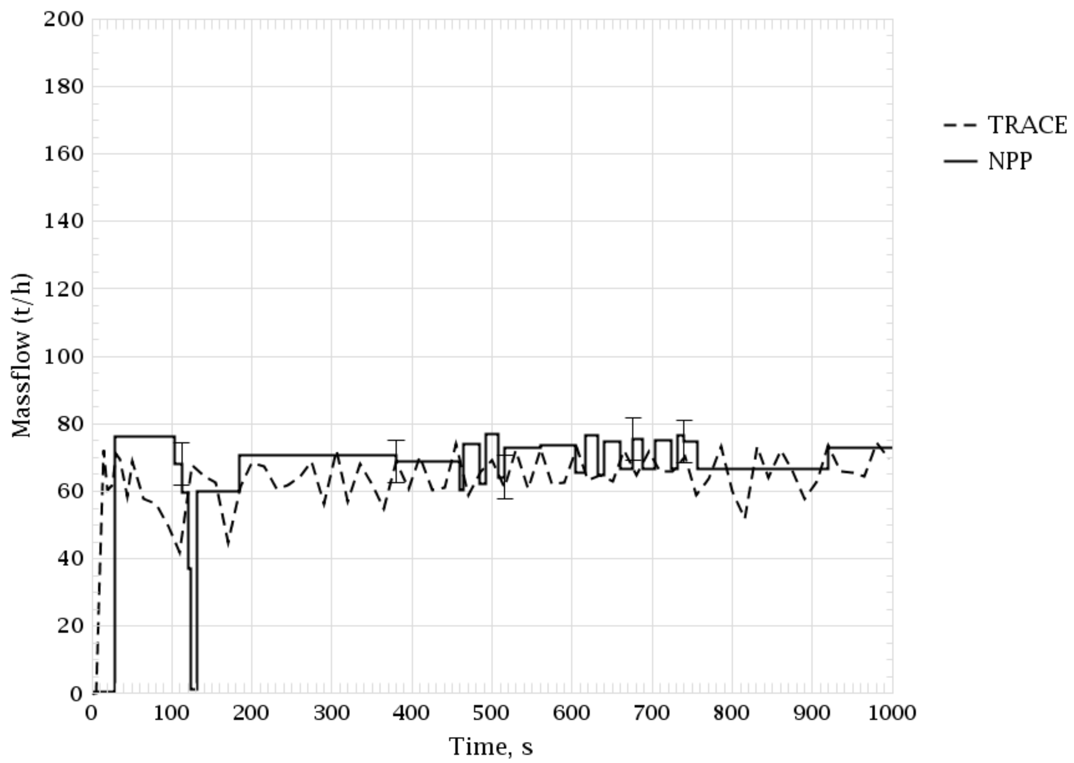

Figure 4-39 AFW Pump No.1 Flow Rate

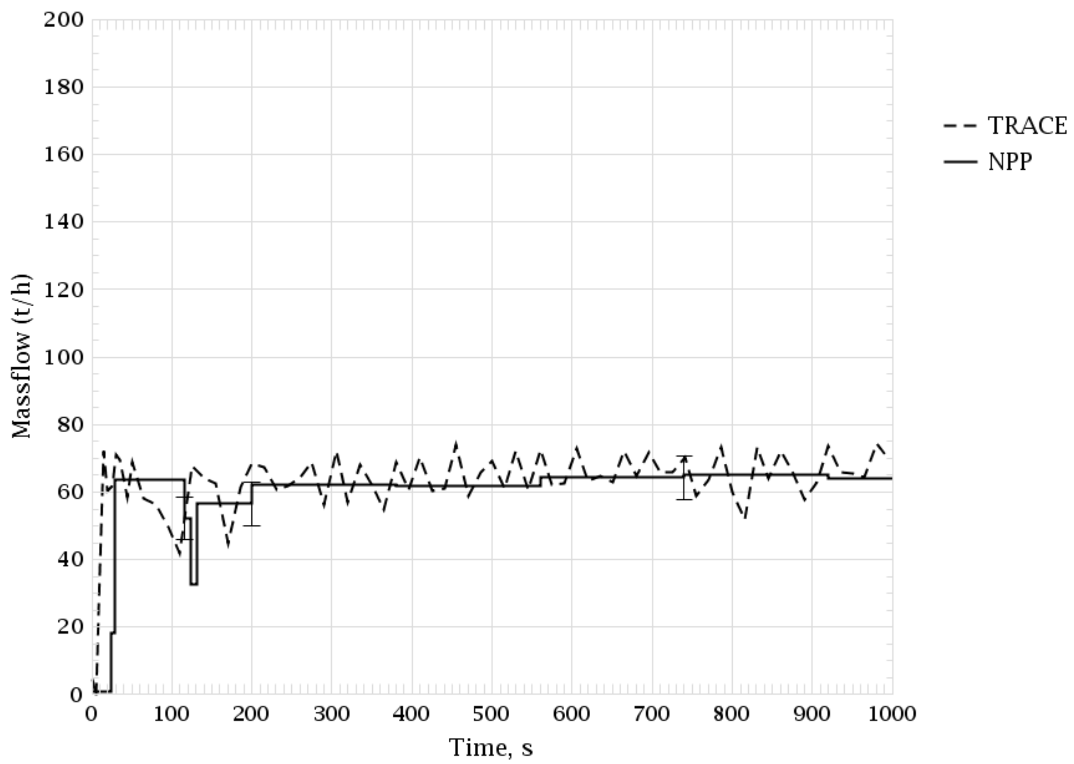

Figure 4-40 AFW Pump No.2 Flow Rate

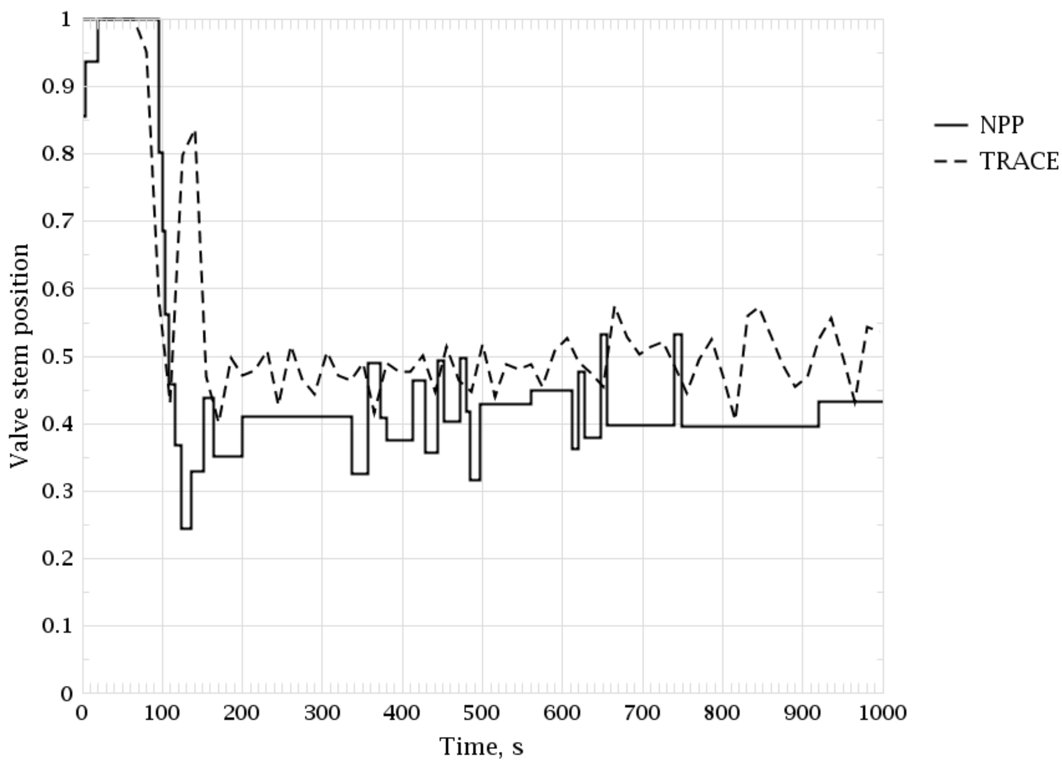

Figure 4-41 Main Feed Water Control Valve No.1 Stem Position

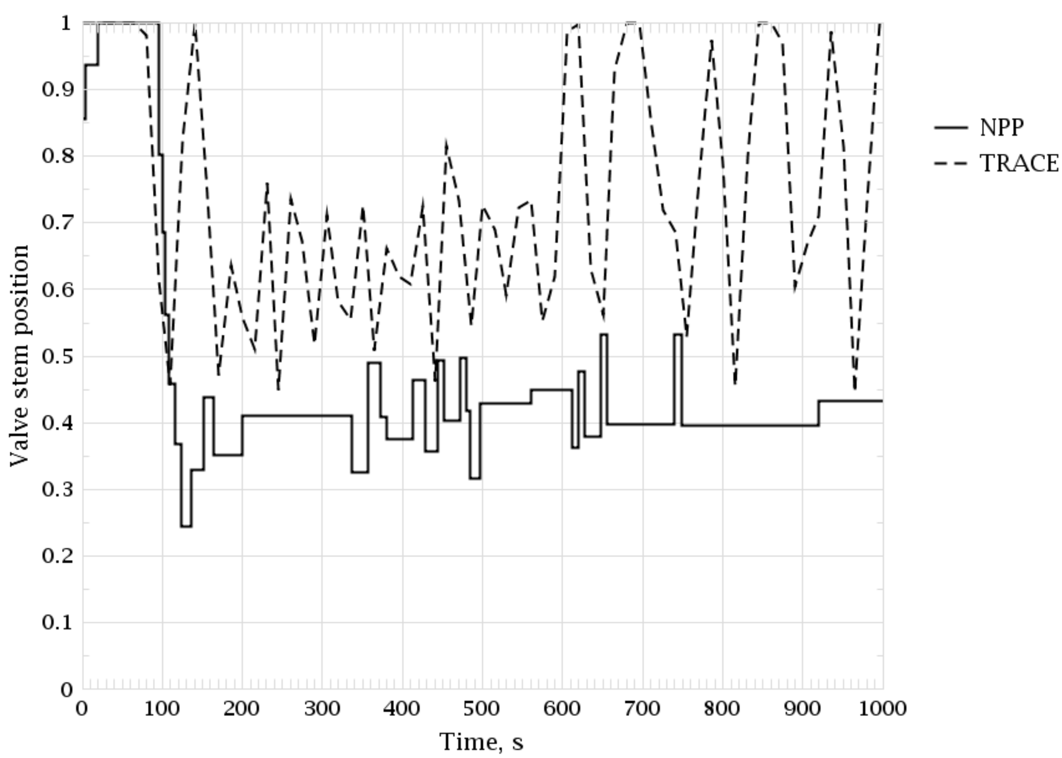

Figure 4-42 Main Feed Water Control Valve No.2 Stem Position

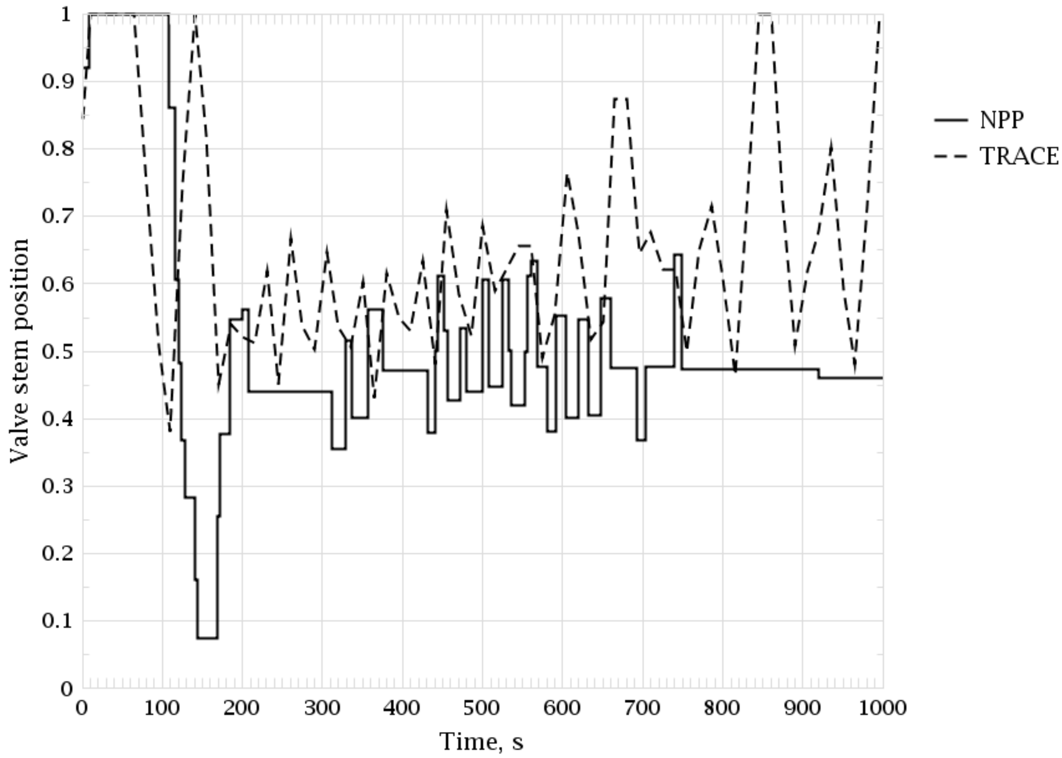

Figure 4-43 Main Feed Water Control Valve No.3 Stem Position

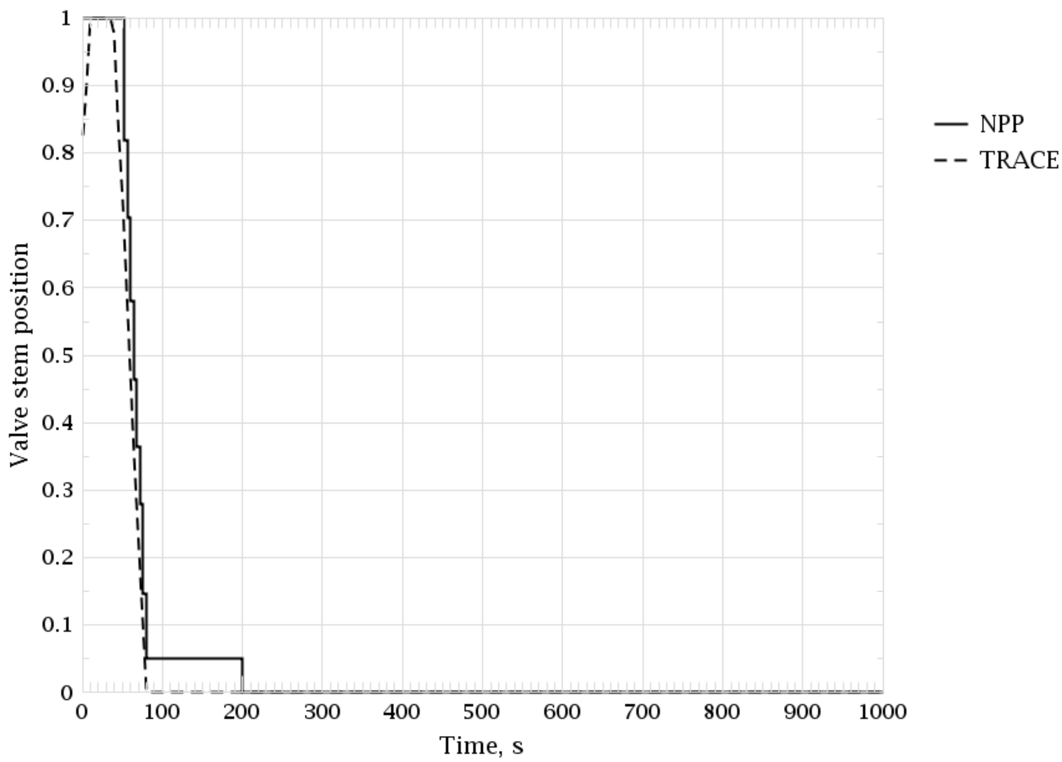

Figure 4-44 Main Feed Water Control Valve No.4 Stem Position

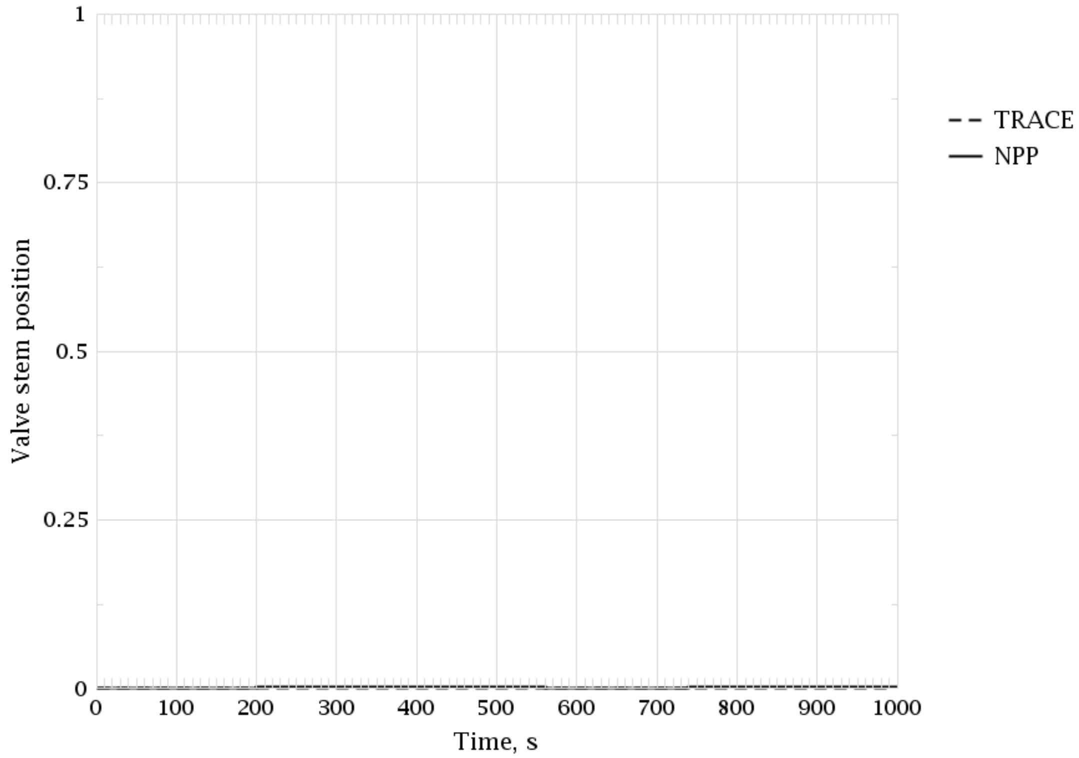

Figure 4-45 Start-up/Shutdown Feed Water Control Valve No.1 Stem Position

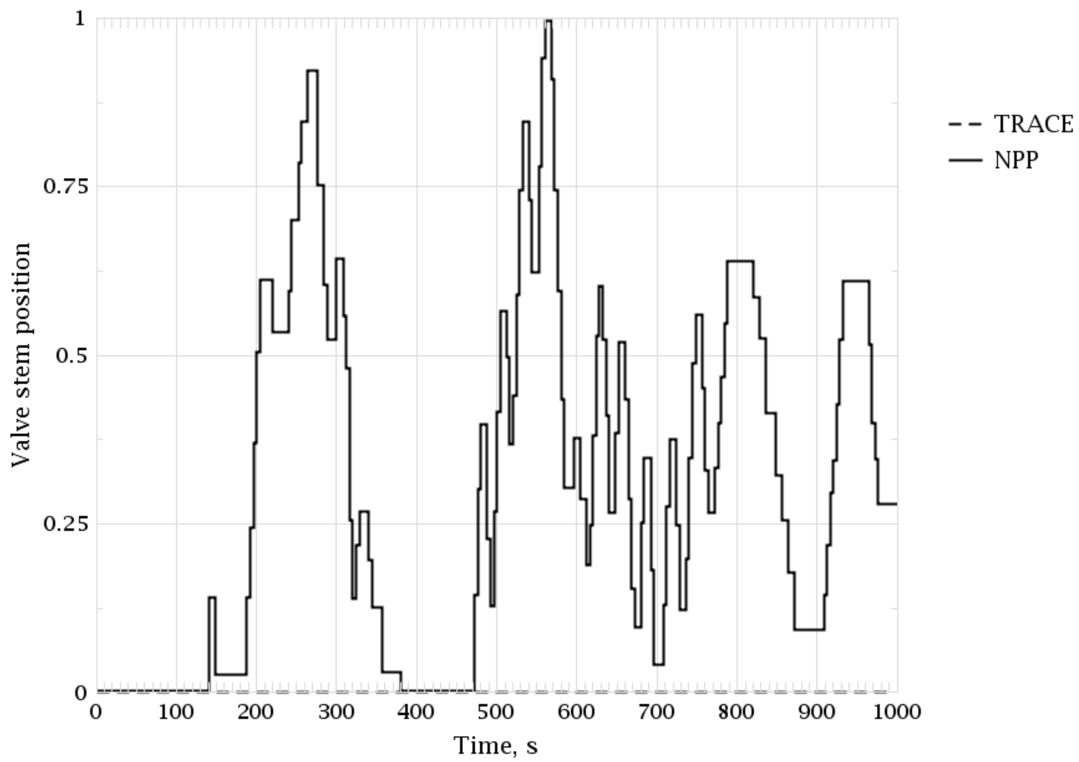

Figure 4-46 Start-up/Shutdown Feed Water Control Valve No.2 Stem Position

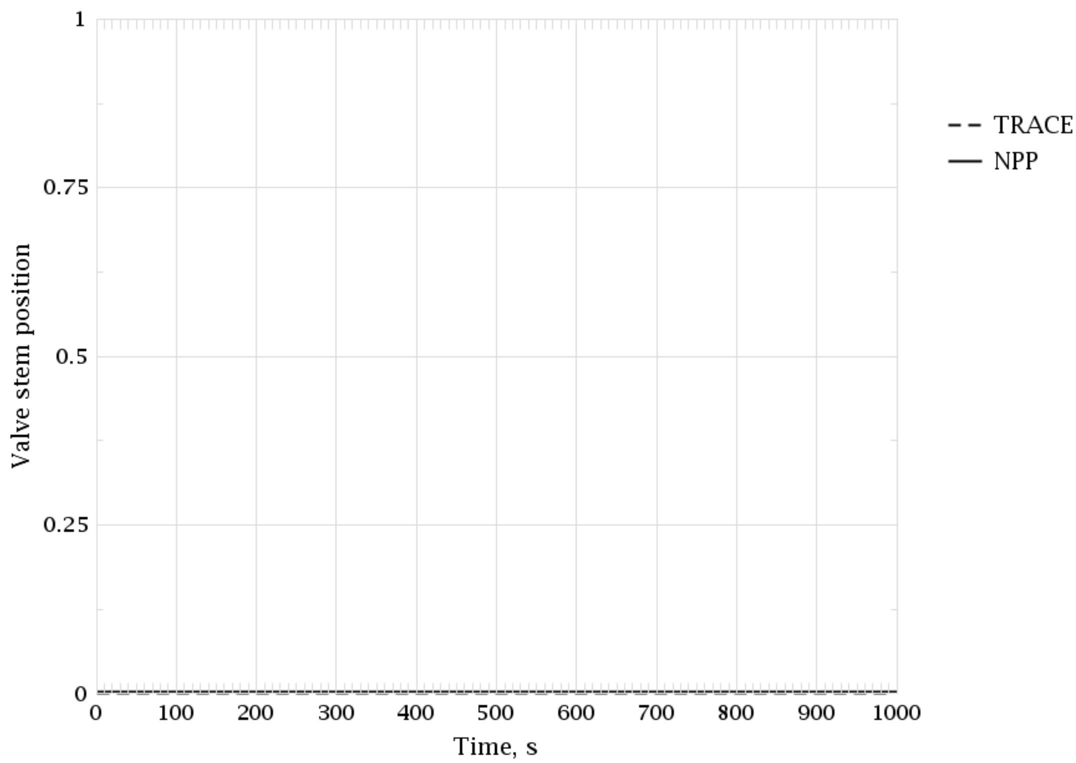

Figure 4-47 Start-up/Shutdown Feed Water Control Valve No.3 Stem Position

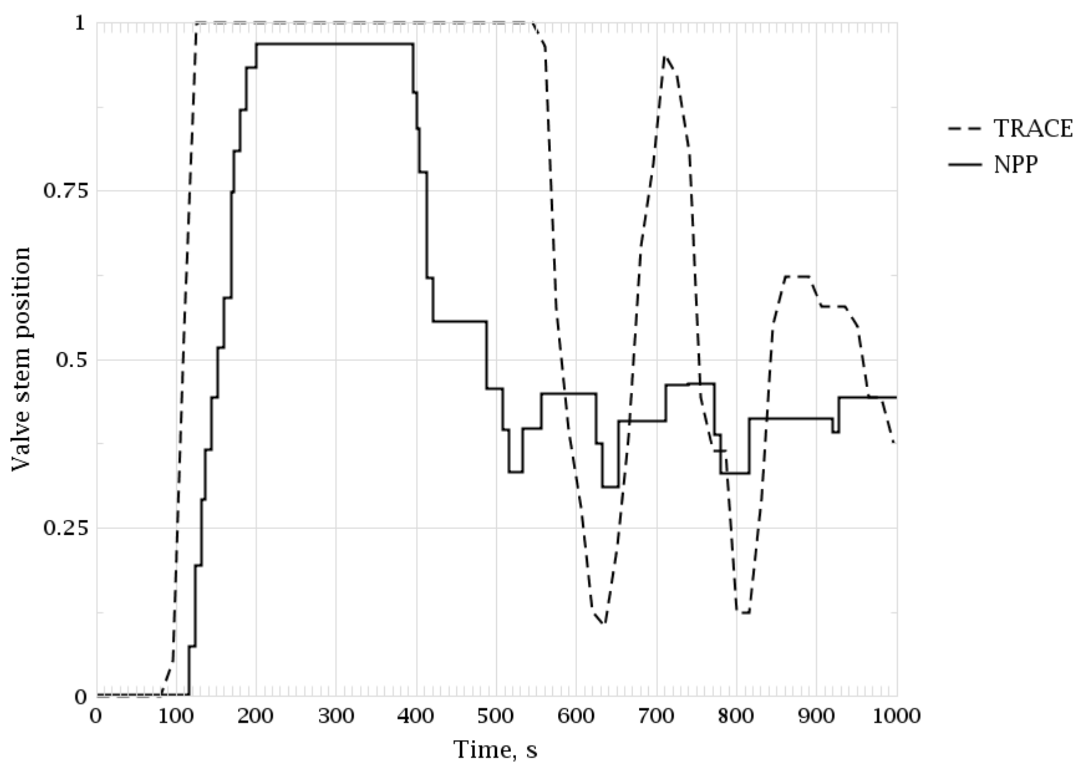

Figure 4-48 Start-up/Shutdown Feed Water Control Valve No.4 Stem Position

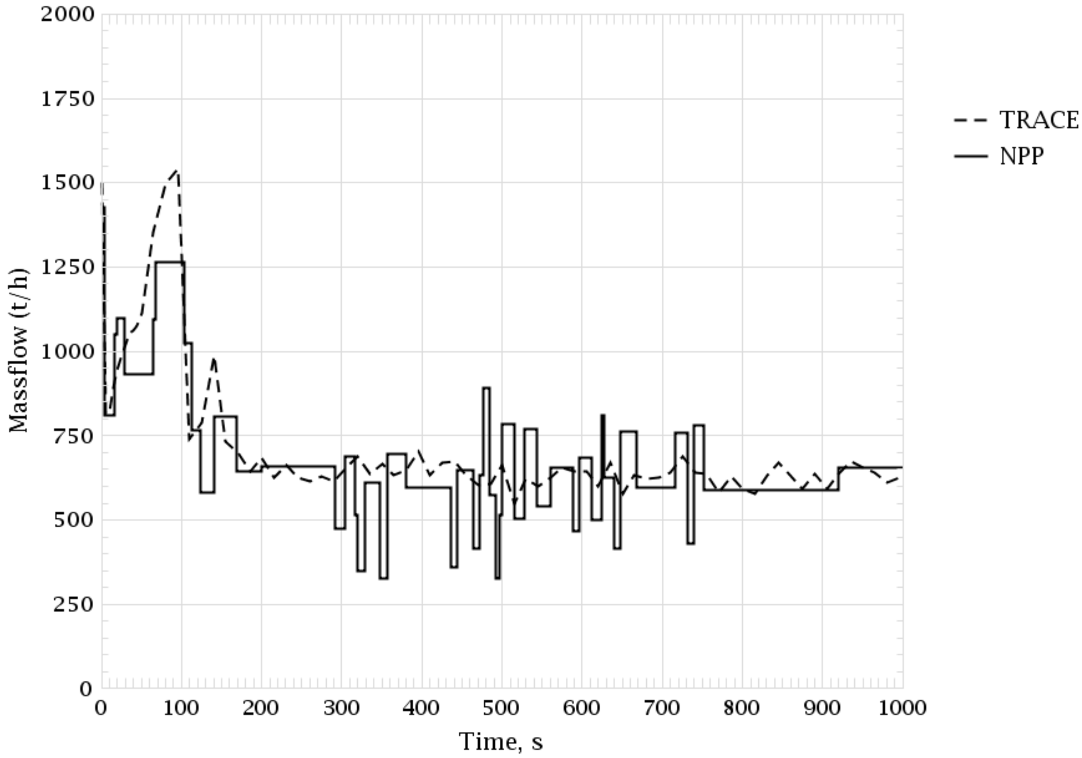

Figure 4-49 SG-1 Feed Water Flow

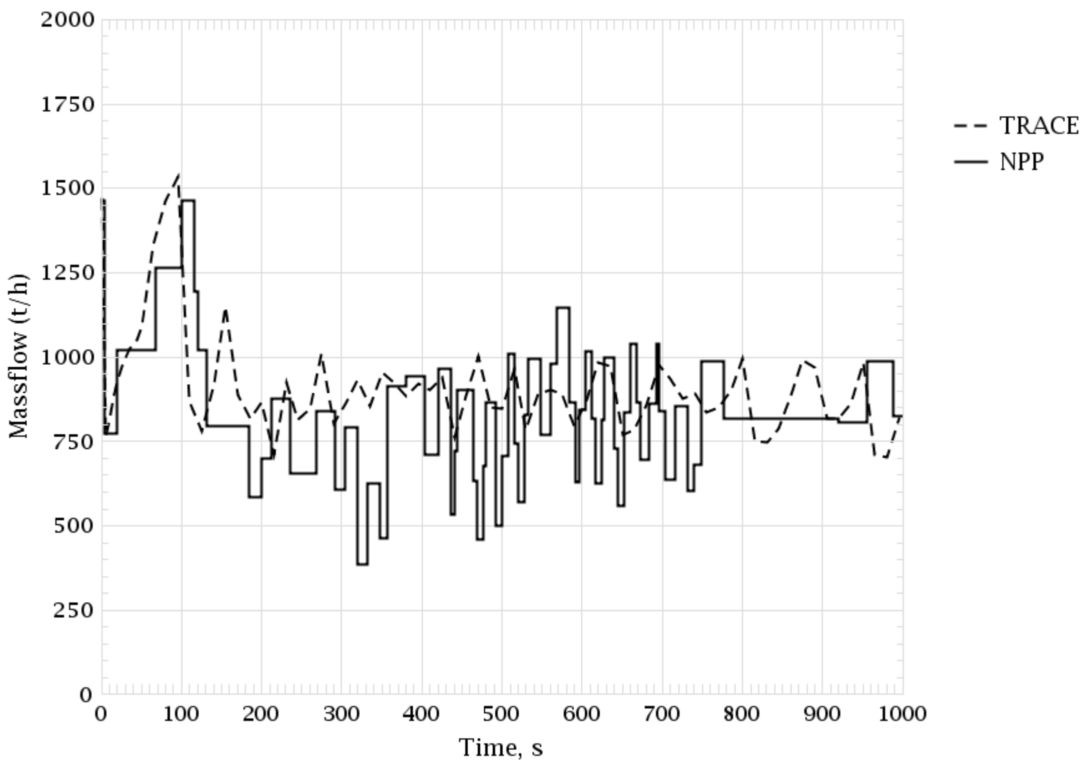

Figure 4-50 SG-2 Feed Water Flow

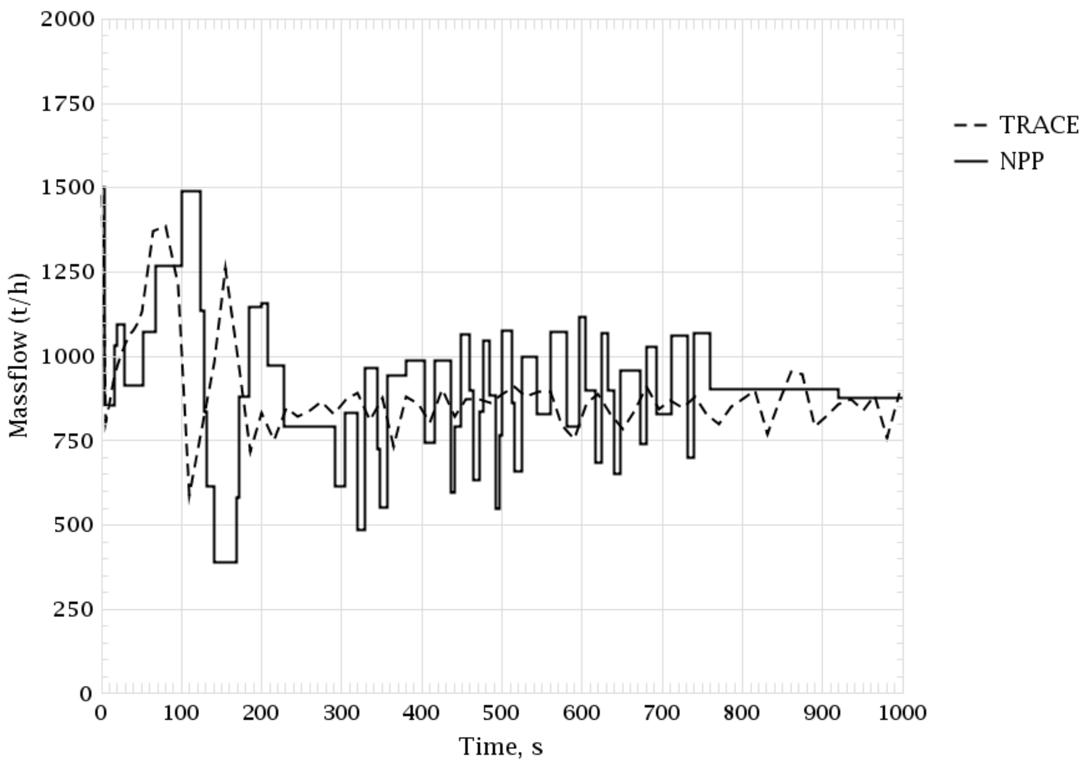

Figure 4-51 SG-3 Feed Water Flow

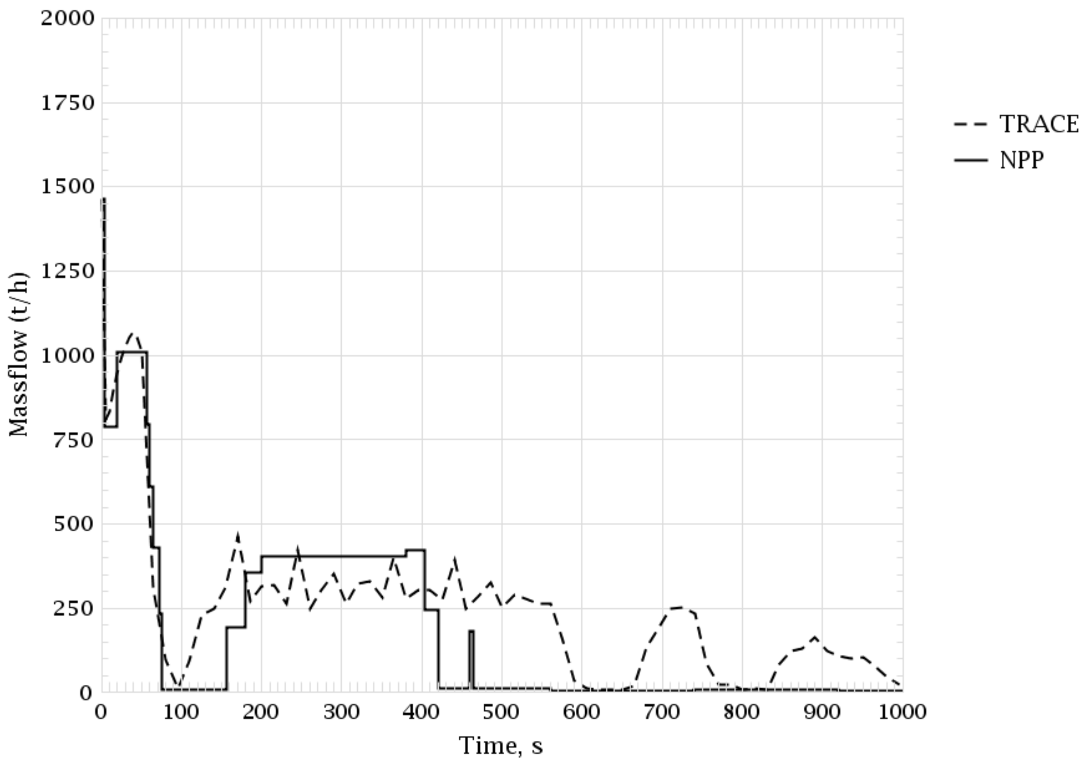

Figure 4-52 SG-4 Feed Water Flow

4.2 ZNPP Unit 6 Inadvertent FASIV Closure

This incident occurred at ZNPP Unit 6 on November 08, 2015, and was caused by inadvertent closure of FASIV at nominal power operation [5]. Closure of FASIV at the main steam line (MSL) No.1 resulted in automatic trip of reactor coolant pump at correspondent RCS loop and in increase of MSL pressure with actuation of steam dump to atmosphere. At MSL-1 pressure above 80 kgf/cm^2 reactor scram was actuated at 14 sec of incident. Turbine and MFW-1 pump trip was initiated by operator at 27 sec and 355 sec of transient, respectively.

4.2.1 Initial Conditions

Before the incident the following main equipment was in operation at ZNPP Unit 6: 4 RCPs, 2 MFW pumps, electro-hydraulic turbine control system, MFW flow controllers to individual SGs, MFW pump flow controller, RCS pressure and PRZ level controllers.

The initial conditions selected for transient calculation correspond to those specified in Table 4-1.

4.2.2 Boundary Conditions

The boundary conditions applied in validation calculation are described below.

1. As the initiating event the closure of FASIV is modeled.
2. Closure of turbine stop valves is performed by operator 27 s after actuation of reactor scram. Correspondent action is simulated in calculation.
3. From plant measurement data it can be seen that during the incident only one out of two AFW pumps (pump no.2) was in operation. Therefore trip of AFW pump no.1 is modeled in calculation.
4. At 355 s after incident occurrence operator switches off MFW pump no.1. Same action is modeled in calculation.
5. Operation of BRU-SN is simulated as an additional steam release from MSH. Steam release flow rate is derived from BRU-SN characteristics defined in [4]. BRU-SN stem position as a function of time was set according to plant measured data and specified in Table 4-3.

Table 4-5 BRU-SN stem position (S) as a function of time

Time, s	S, %	Time, s	S, %	Time, s	S, %	Time, s	S, %
0,0	0,0	44,0	65,1	124,0	62,9	192,0	15,2
20,0	5,3	48,0	60,2	132,0	51,7	196,0	4,5
24,0	19,7	56,0	56,3	140,0	49,4	200,0	1,4
28,0	40,6	64,0	48,6	152,0	41,0	276,0	1,4
32,0	55,3	76,0	46,2	156,0	26,6	456,0	1,4
36,0	63,9	96,0	46,2	160,0	24,5	636,0	0,0
40,0	68,1	120,0	55,5	164,0	18,4	996,0	0,0

4.2.3 Calculation Results

Table 4-6 provides comparison of calculated and actual timing of events occurred in the course of the incident. Plots of calculated and measured parameter values during the incident are shown in Figures 4-53 – 4-113.

Table 4-6 Sequence of Events for Validation Transient No.2

Time, sec		Event
Incident	Calculation	
0	0	SG-1 MSL FASIV closure
3	3	RCP-1 trip by interlock YBF15
4	4	Actuation of UPZ signal
11	11	BRU-A opening due to increase of MSL pressure up to 73 kgf/cm^2
12	14	Reactor scram
30	20	Start of AFW operation
27	27	Closure of turbine stop valves
50	45	Start of NRU-K operation
355	355	Trip of MFW pumps
–	1000	End of calculation

After reactor scram due to increase of secondary circuit pressure the reactor power decreases down to decay heat (Figure 4-53). Sharp decrease of reactor power results in a decrease of RCS pressure at 30 s (Figure 4-54). Further RCS pressure behavior is determined by the following factors:

- operation of PRZ heaters groups;
- decrease of heat removal by the secondary side during increase of MSH pressure and decrease of SG levels.

Minimal RCS pressure during the incident is 138.0 kgf/cm^2 vs 140.0 kgf/cm^2 according to measured data.

Decrease of average coolant temperature (Figures 4-64 – 4-67) causes decrease of PRZ level setpoint. PRZ level (Figure 4-55) is controlled by the changes in make-up and let-down flow (Figures 4-68, 4-69). Minimal calculated collapsed PRZ level is 5.2 m which is in good agreement with the measured data (5.4 m).

Maximal calculated make-up flow of 40.0 t/h (Figure 4-68) is reached at 100 s, while measured value at this moment is 45.4 t/h.

Cold and hot legs temperatures are shown in Figures 4-56 – 4-67. After RCP-1 trip at 3 s of incident and its coast-down the coolant flow in loop 1 reverses and hot leg temperature in this loop decreases down to 273.3°C (Figure 4-56). Calculated and measured difference between temperatures of hot and cold legs of loops 2 – 4 decreases to 0.73, 0.76 and 0.64°C (Figures 4-64 – 4-67).

The turbine stop valves are closed by operator at 27 s of incident. However the expected increase of MSH pressure is prevented by BRU-SN operation. Maximal calculated and measured MSH pressure values (Figure 4-84) are 60.5 and 58.8 kgf/cm^2, respectively. Maintenance of MSH pressure is performed by BRU-K controller at 60.1±0.5 kgf/cm^2 (according to controller operation logic the target value is set to MSH measured pressure at the moment of controller actuation).

Closure of FASIV at MSL-1 and subsequent increase of SG-1 pressure above 73 kgf/cm^2 (Figure 4-85) results in opening of BRU-A. Further behavior of SG-1 pressure is determined by heat transfer from the primary circuit and heat losses from SG vessel. SG-2,3,4 pressure (Figures 4-86 – 4-88) corresponds to MSH pressure which is controlled by BRU-K and BRU-SN operation.

After reactor scram and closure of turbine stop valves the steam production in SGs decreases and MFW flow to SGs is decreased by closing of MFW control valves (Figures 4-102 – 4-105). At 30 s operation of AFW pump no.2 begins resulting in an increase of MFW collector pressure. Operator switches off MFW pump no.1, while MFW pump no.2 starts to operate through the minimal bypass flow line.

After closure of MFW control valves the feedwater flow to SGs is maintained by operation of start-up/shutdown control valves RL71-74S04 (Figures 4-104 – 4-109).

Calculated and measured SG levels behavior (Figures 4-93 – 4-100) corresponds to the changes in feedwater flow. Minimal calculated SG1–4 level (narrow range) is 0.0 m (Figures 4-47 – 4-100) and 0.04 m according to the measured data. Maximal calculated and measured SG level is 0.4 m and 0.47 m, respectively.

Correspondent minimal/maximal SG–1 level values according to wide level meter (Figure 4-93) are 2.1/2.7 m in calculation and 2.3/2.65 m as measured at the plant.

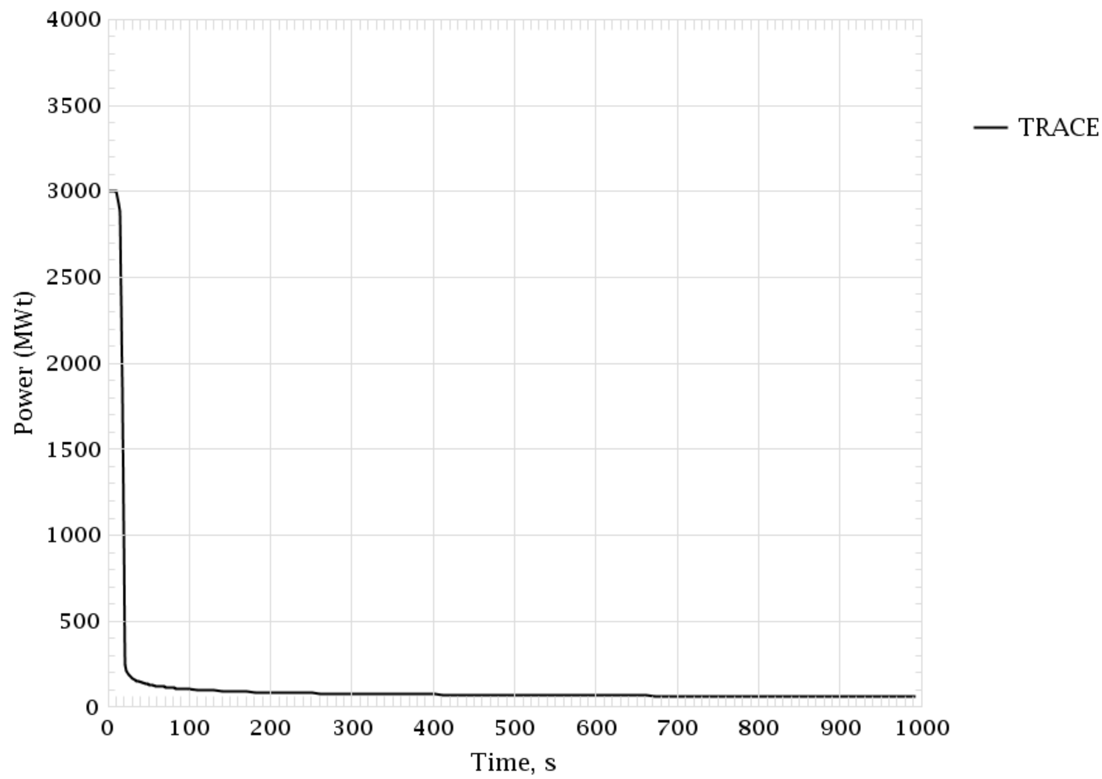

Figure 4-53 Thermal Reactor Power

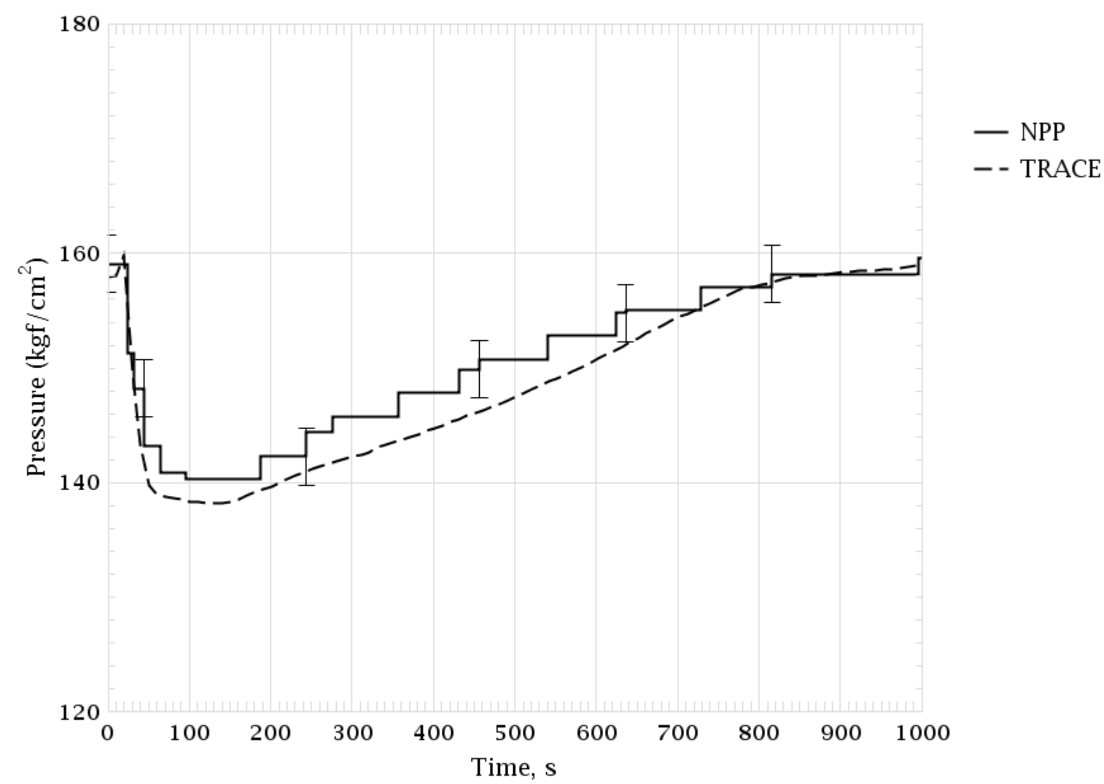

Figure 4-54 RCS Pressure

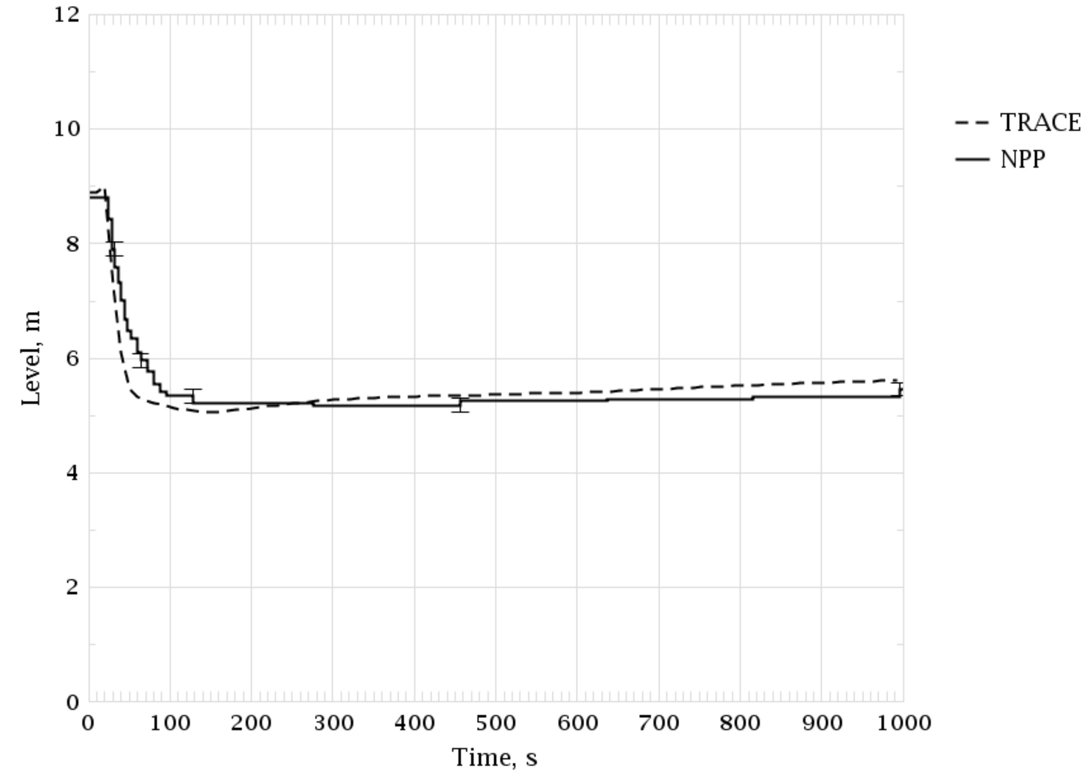

Figure 4-55 PRZ Level

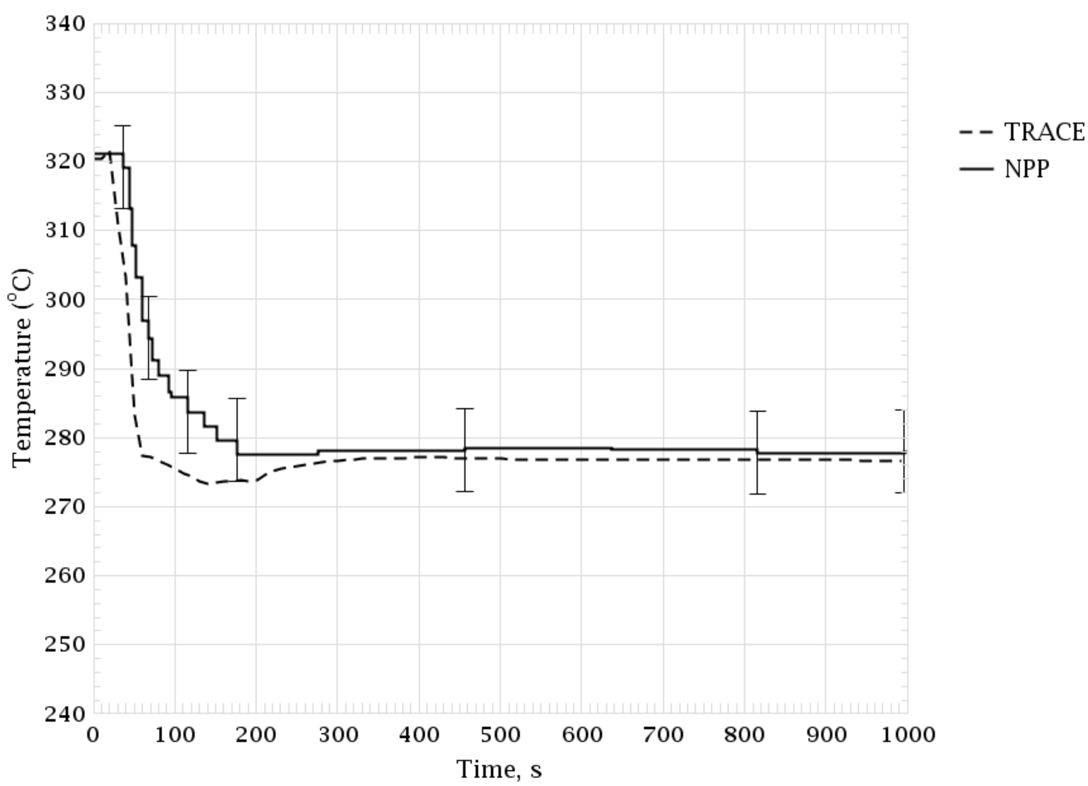

Figure 4-56 Hot Leg Coolant Temperature, Loop No.1

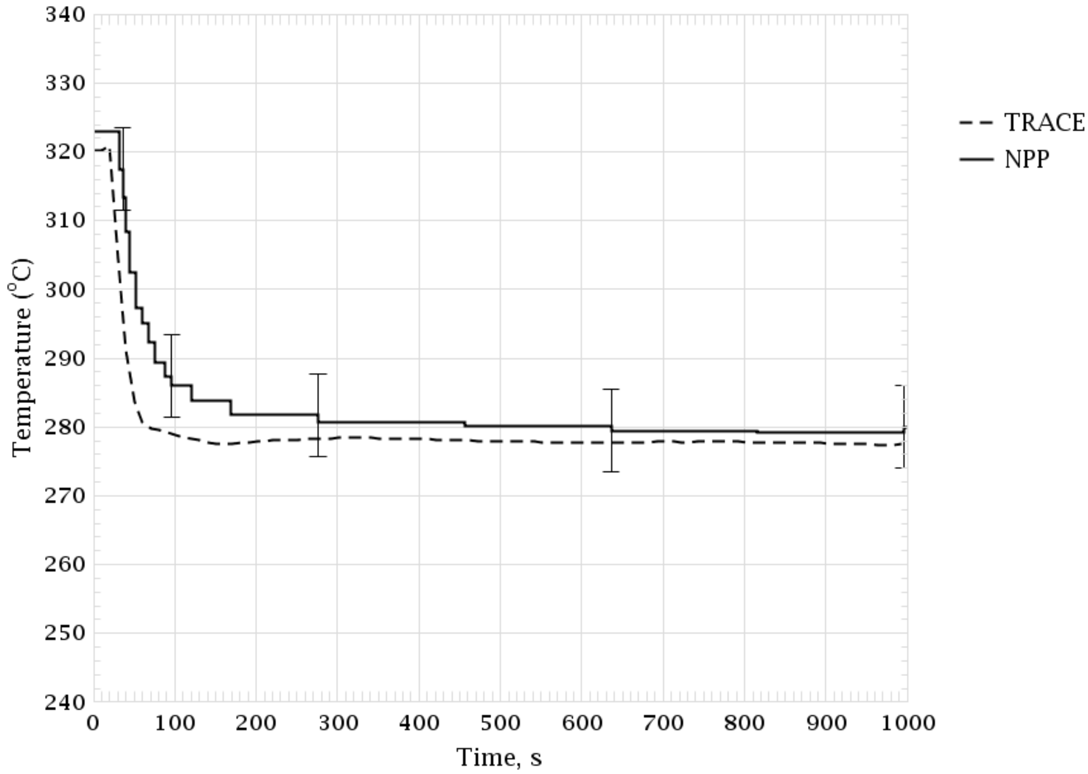

Figure 4-57 Hot Leg Coolant Temperature, Loop No.2

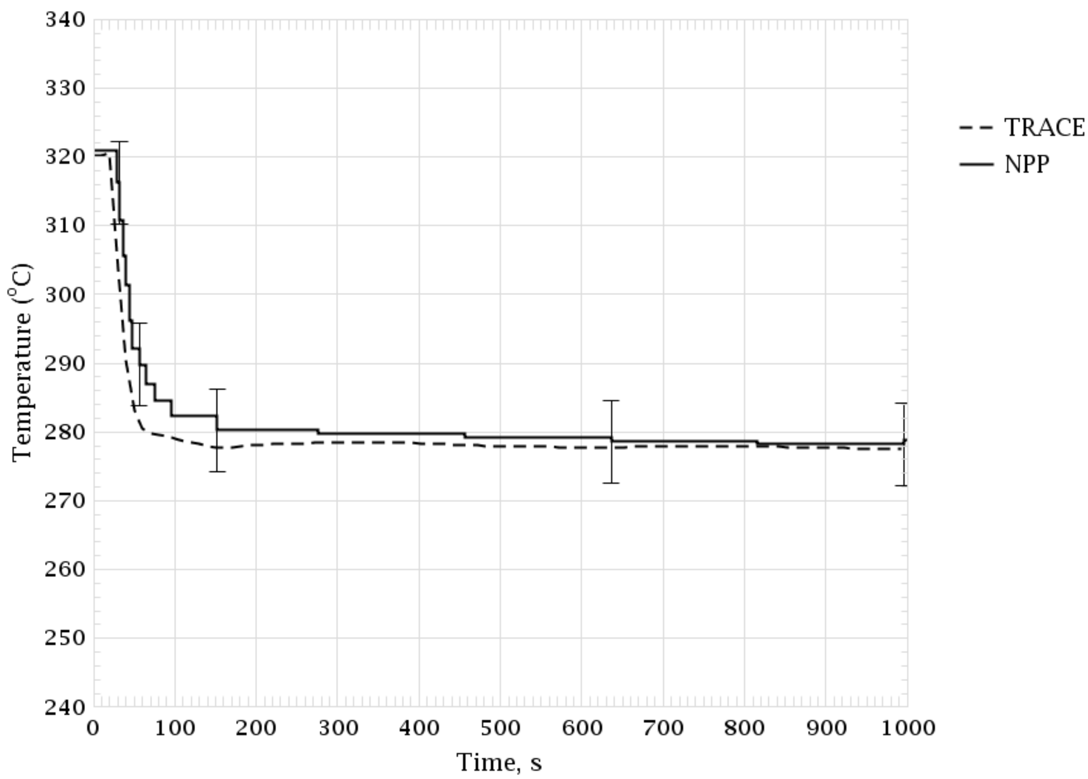

Figure 4-58 Hot Leg Coolant Temperature, Loop No.3

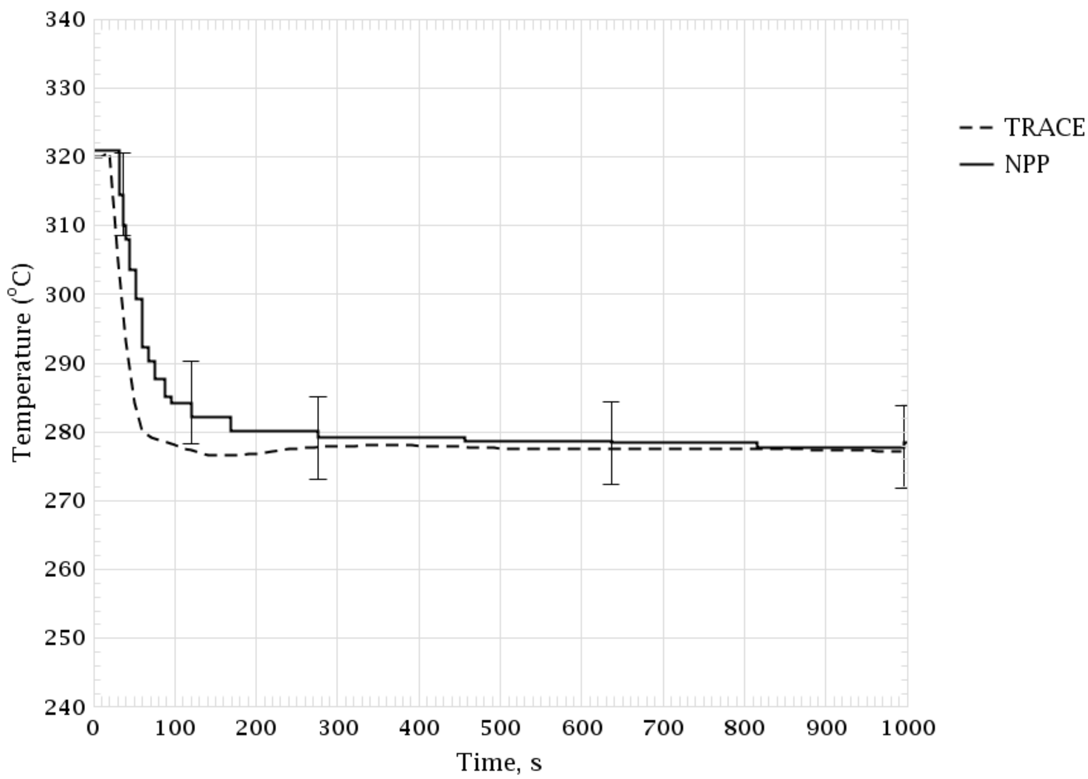

Figure 4-59 Hot Leg Coolant Temperature, Loop No.4

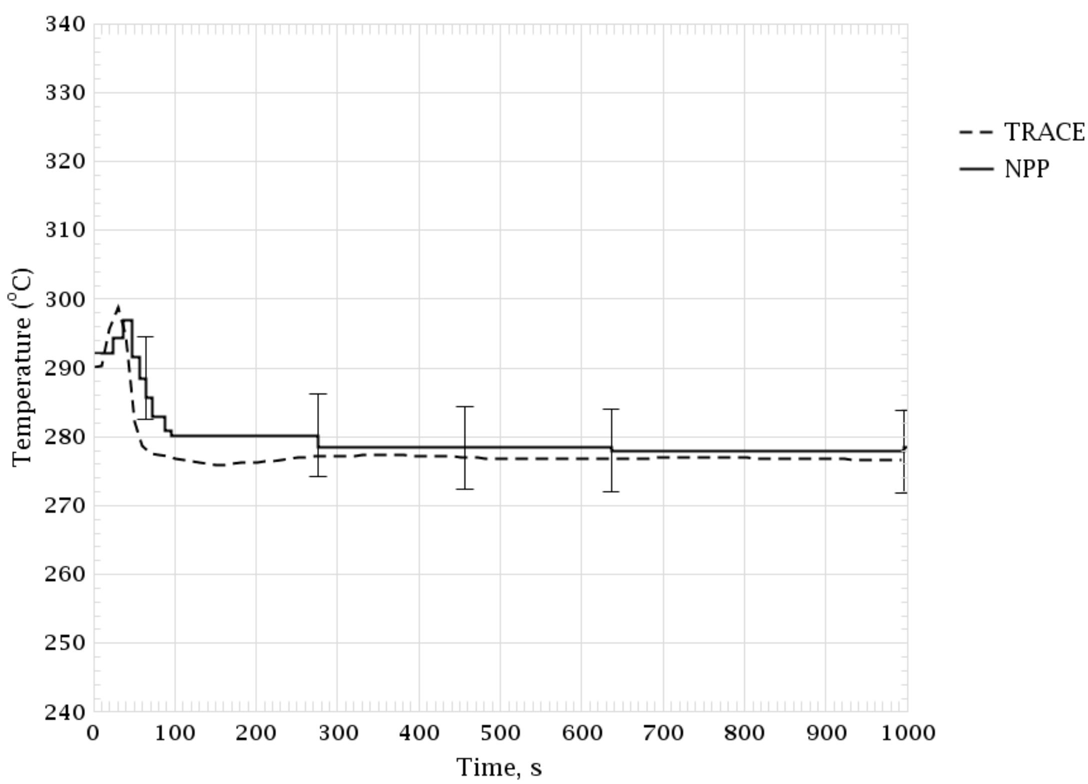

Figure 4-60 Cold Leg Coolant Temperature, Loop No.1

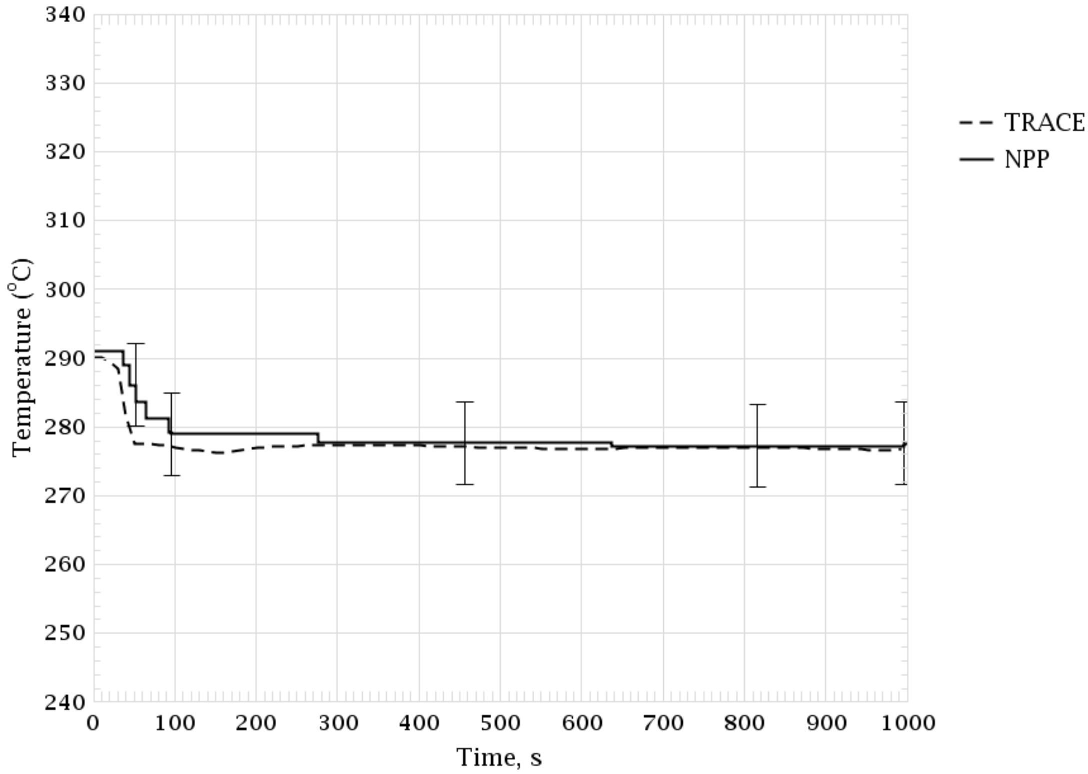

Figure 4-61 Cold Leg Coolant Temperature, Loop No.2

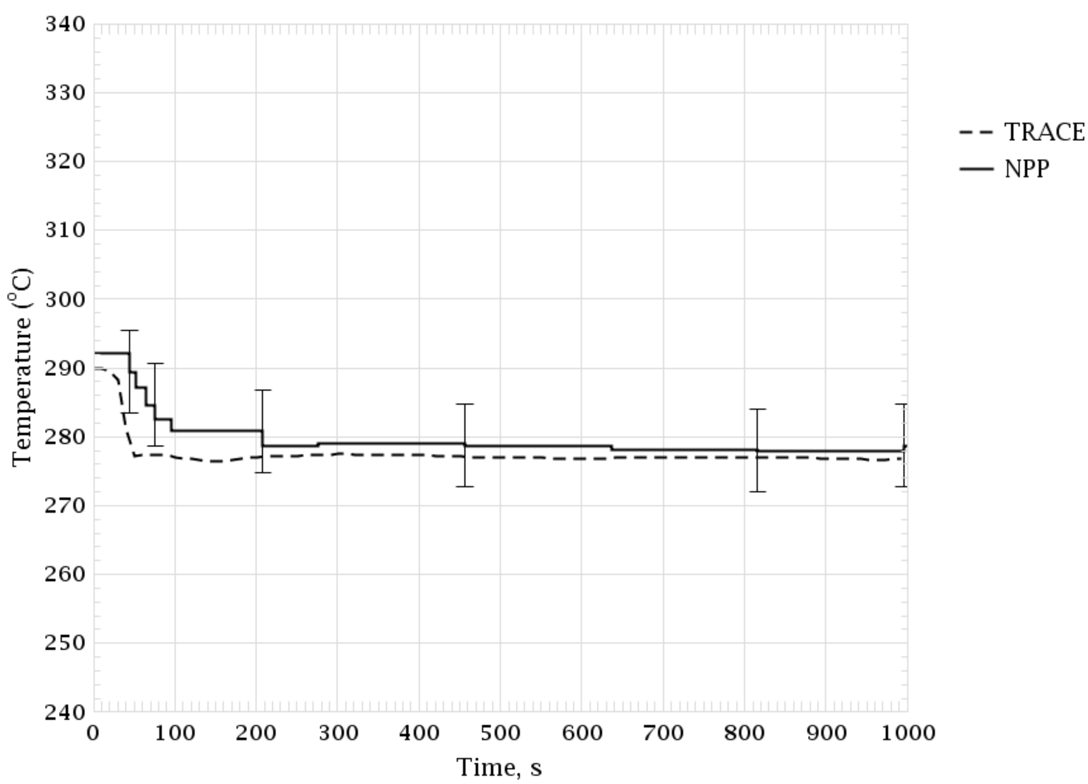

Figure 4-62 Cold Leg Coolant Temperature, Loop No.3

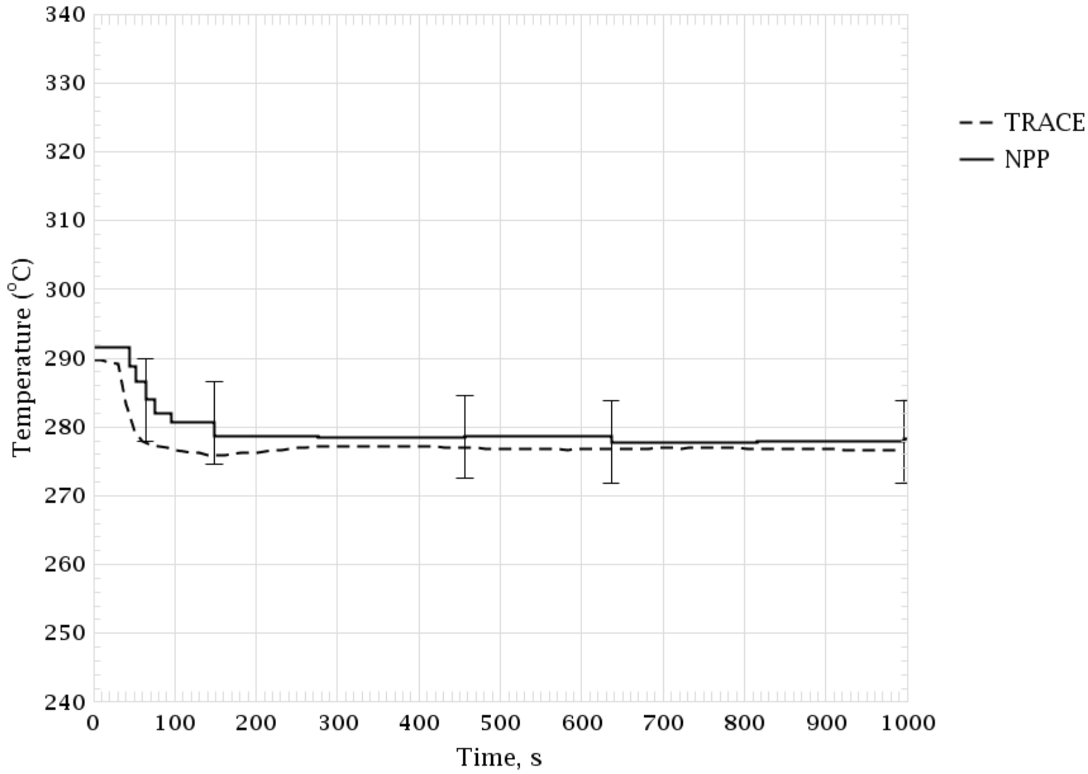

Figure 4-63 Cold Leg Coolant Temperature, Loop No.4

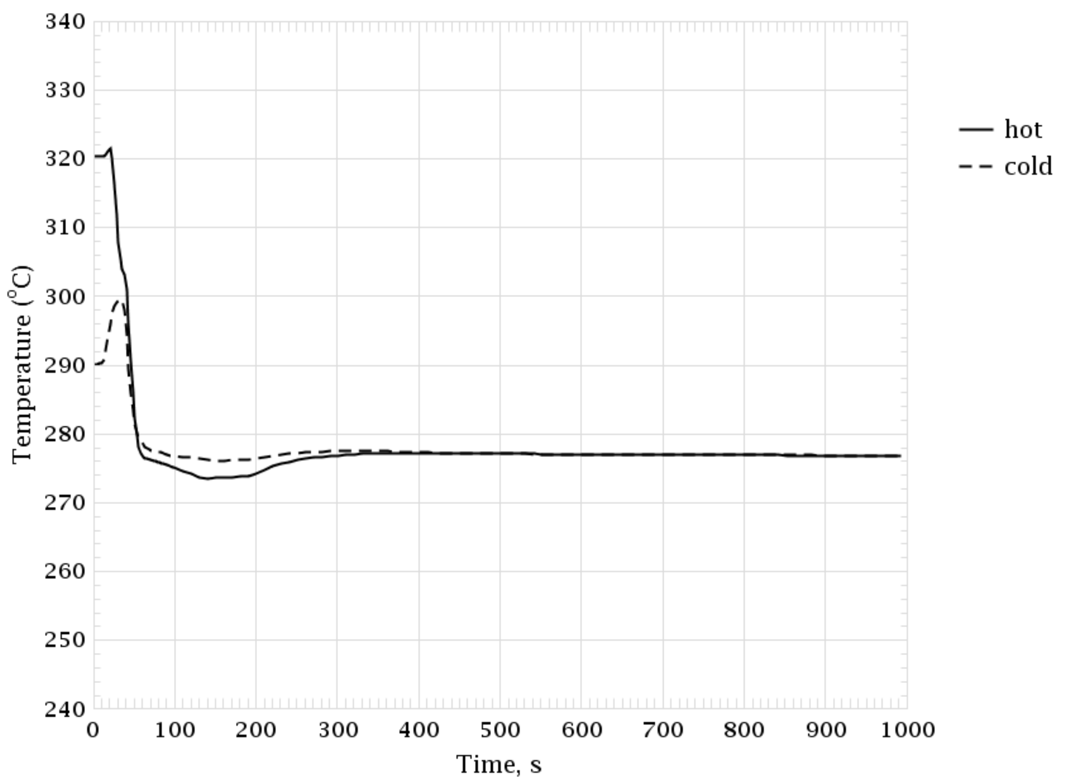

Figure 4-64 Cold and Hot Legs Temperature in Loop No.1

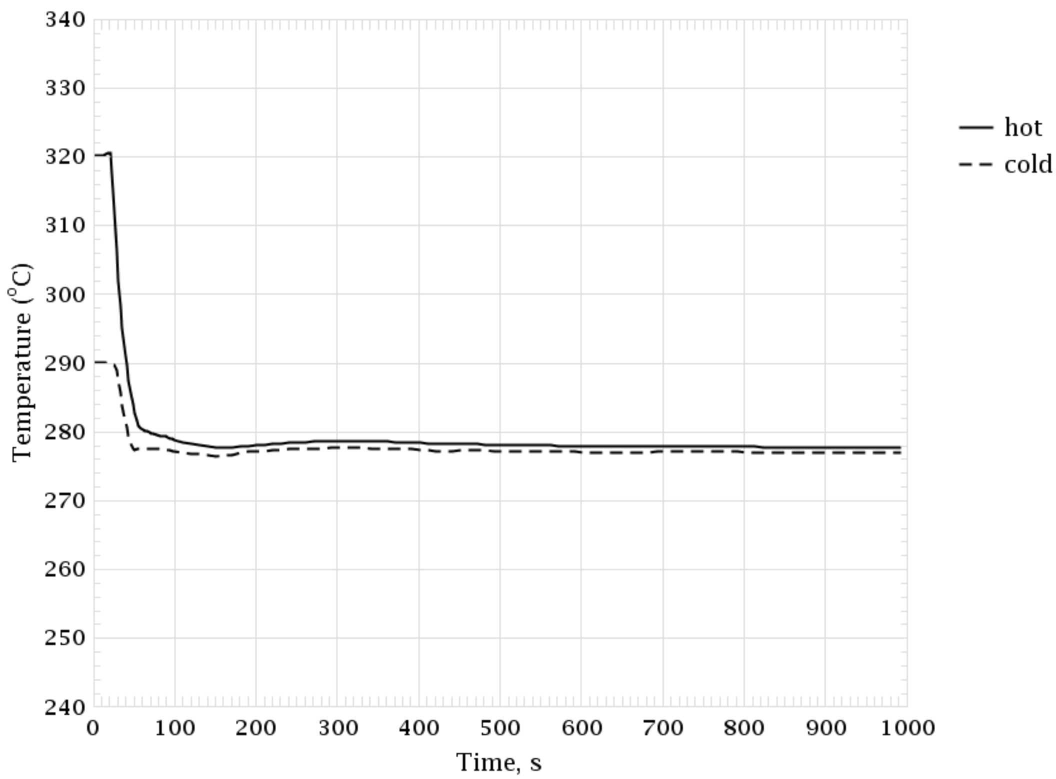

Figure 4-65 Cold and Hot Legs Temperature in Loop No. 2

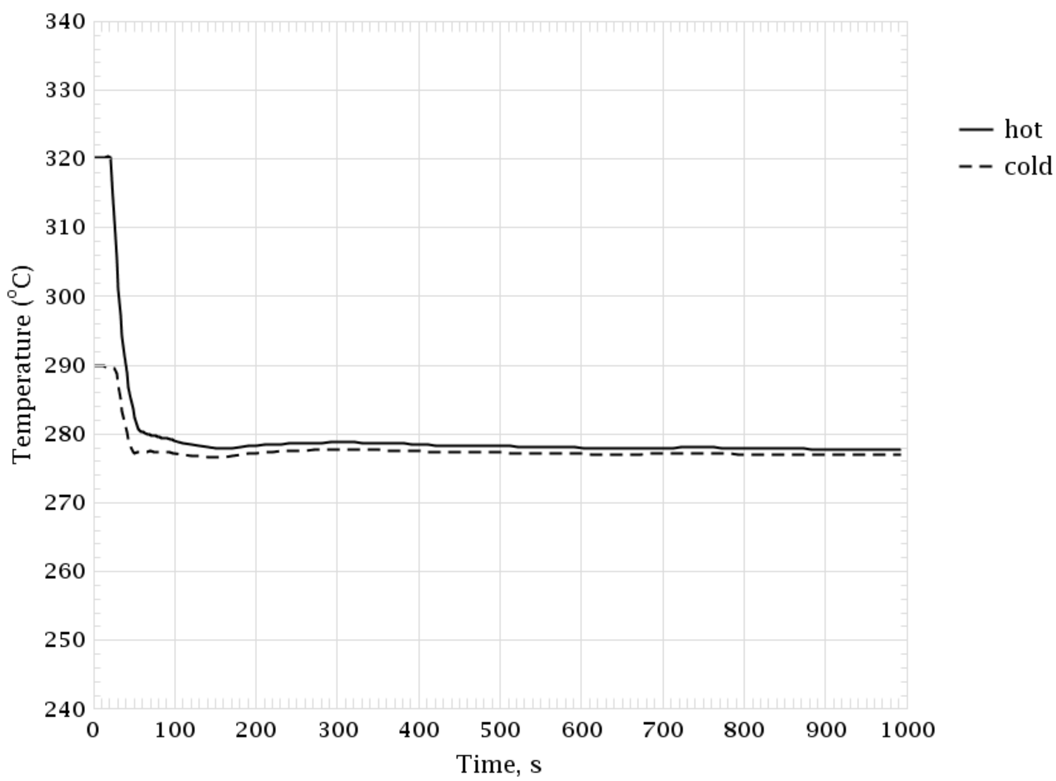

Figure 4-66 Cold and Hot Legs Temperature in Loop No. 3

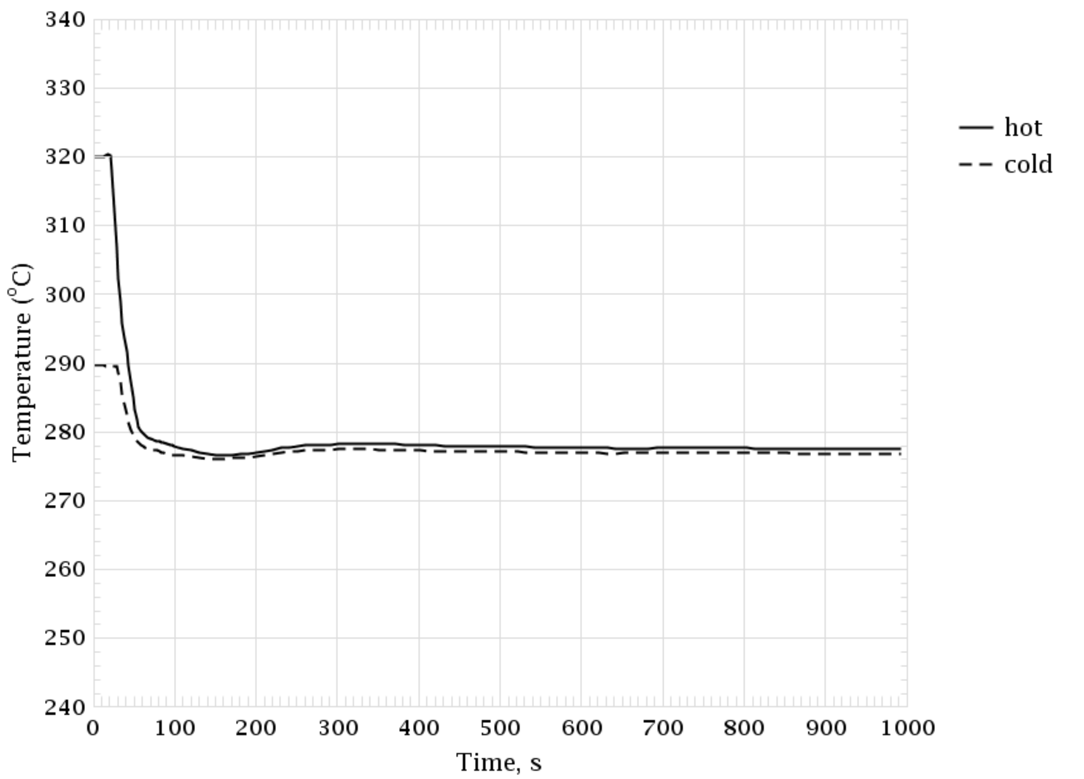

Figure 4-67 Cold and Hot Legs Temperature in Loop No. 4

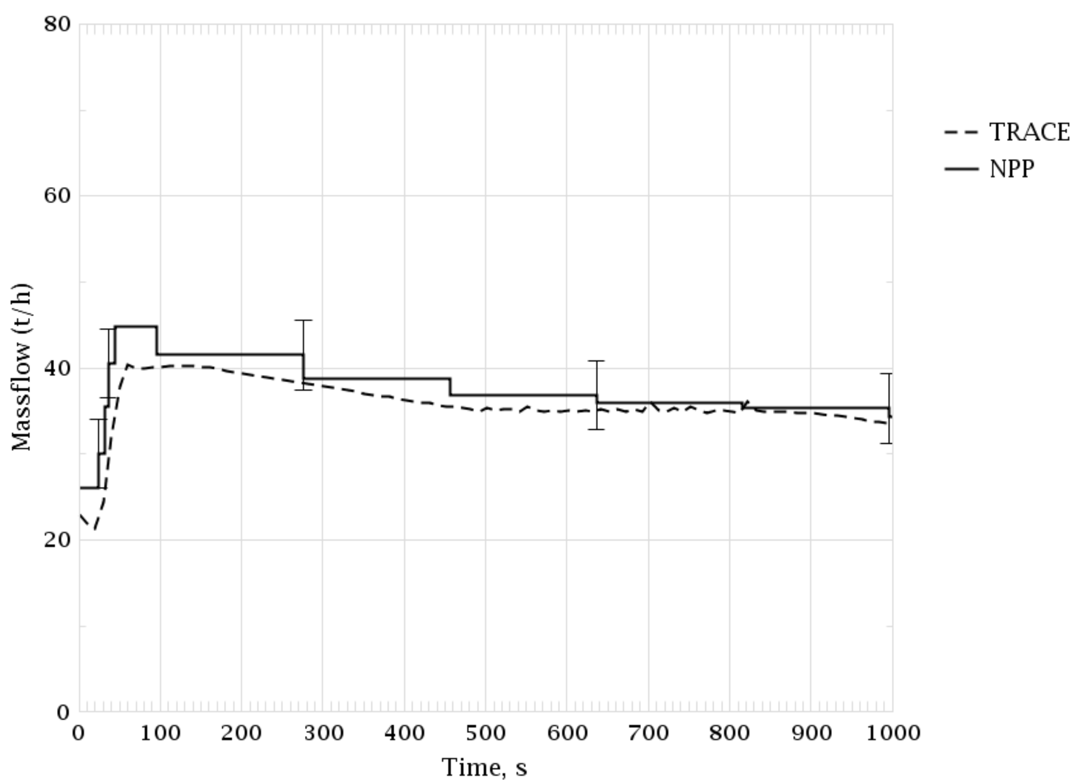

Figure 4-68 Make-up Mass Flow Rate

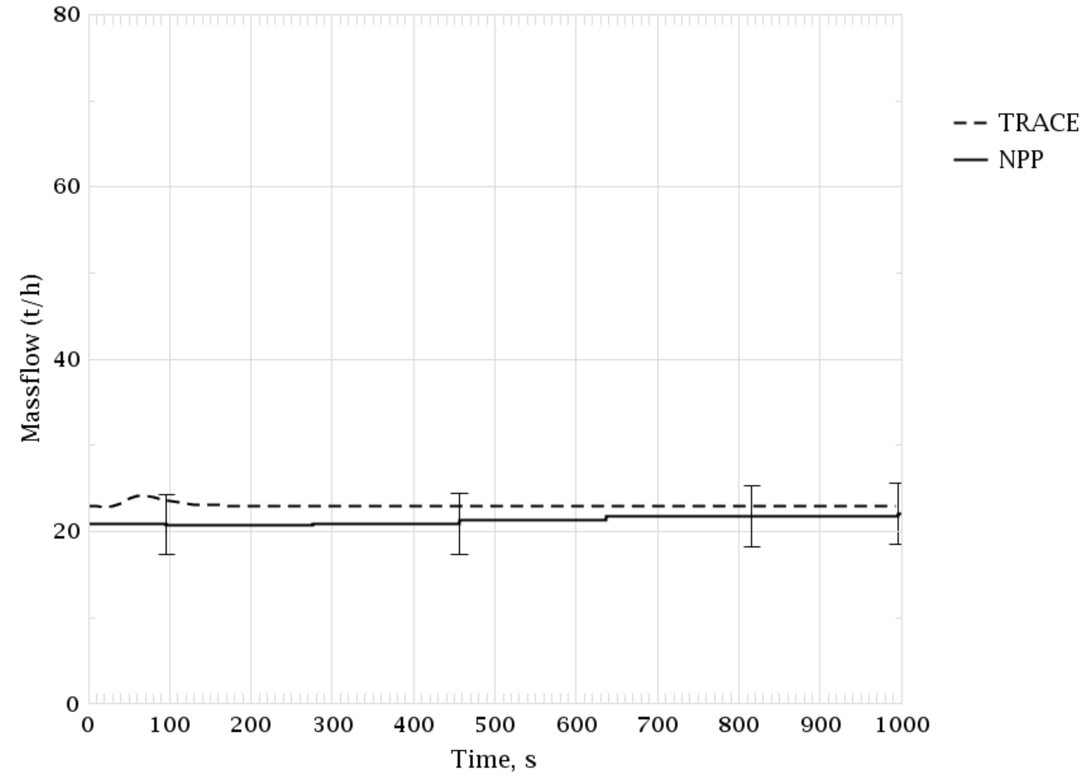

Figure 4-69 Let-down Mass Flow Rate

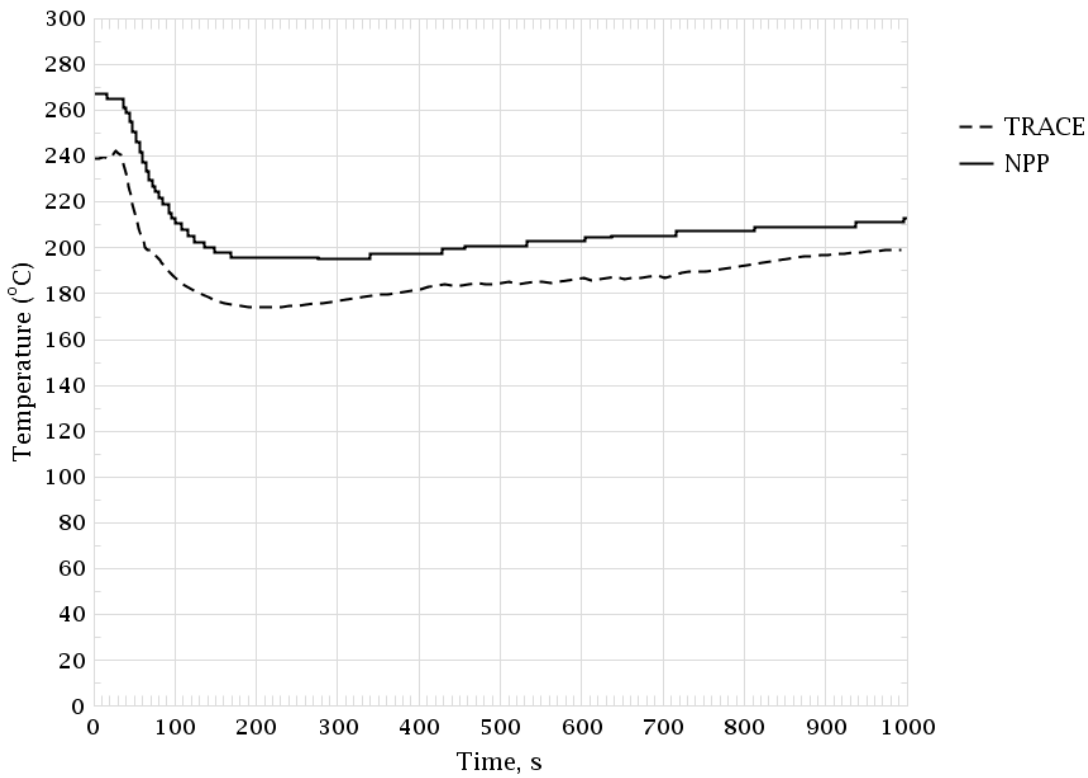

Figure 4-70 Make-up Temperature

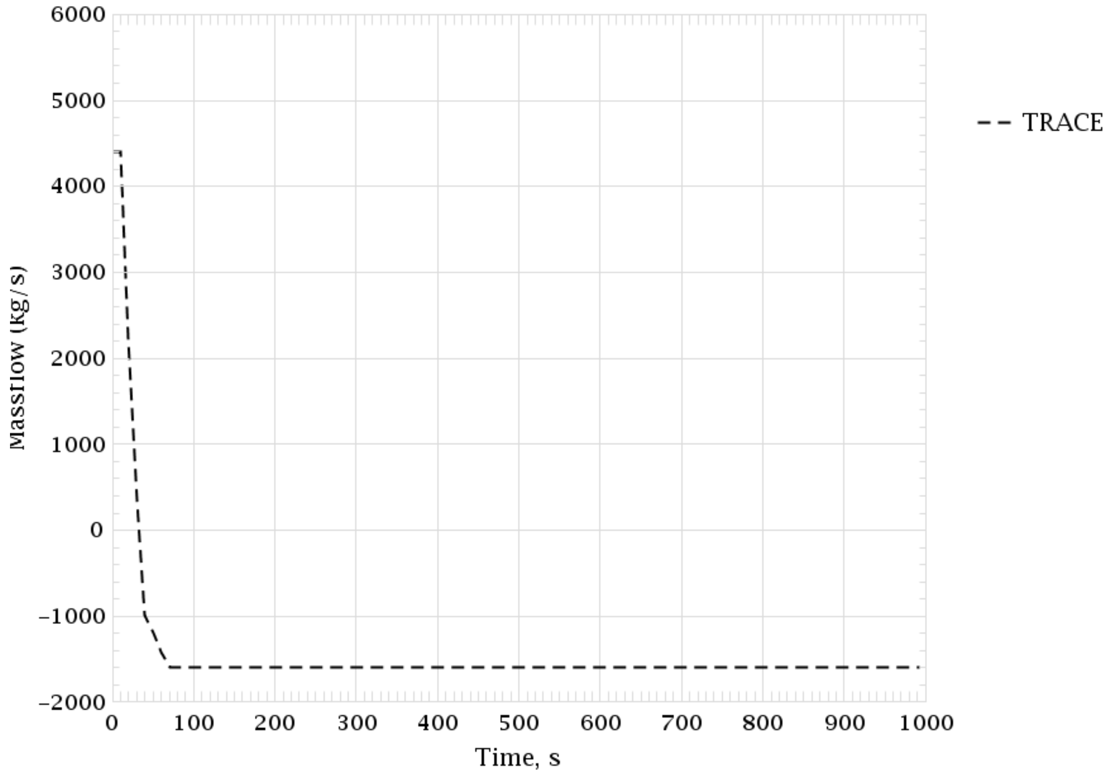

Figure 4-71 RCP-1 Mass Flow Rate

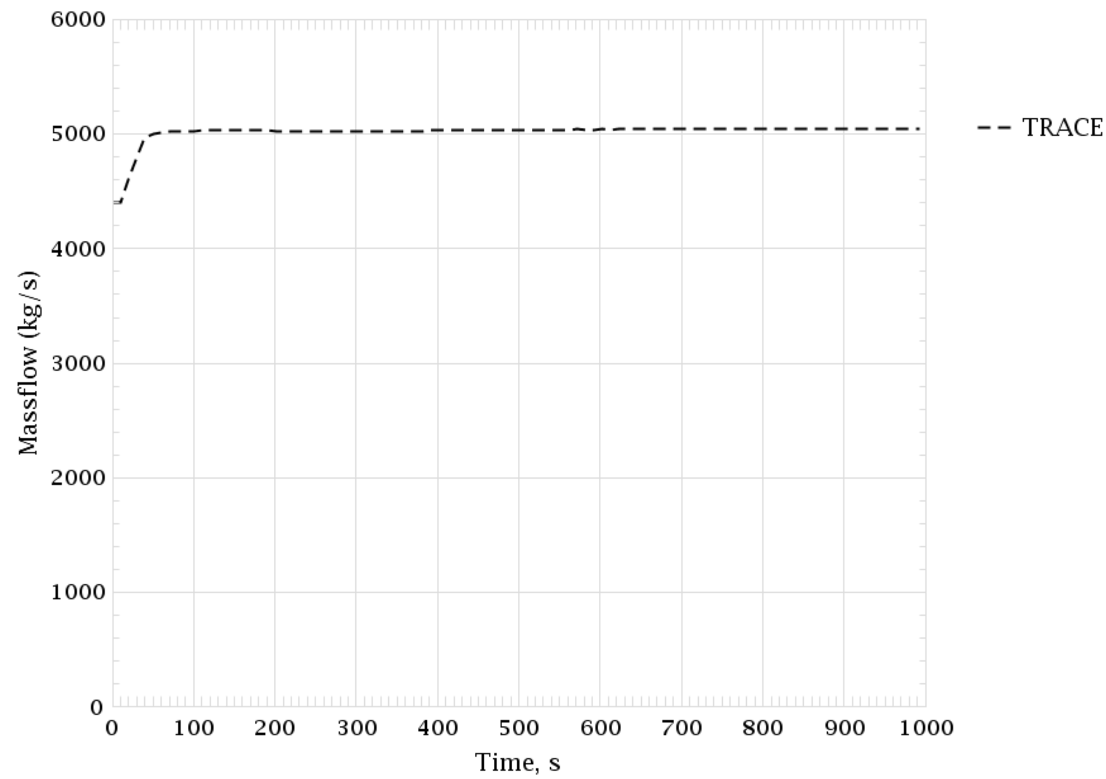

Figure 4-72 RCP-2 Mass Flow Rate

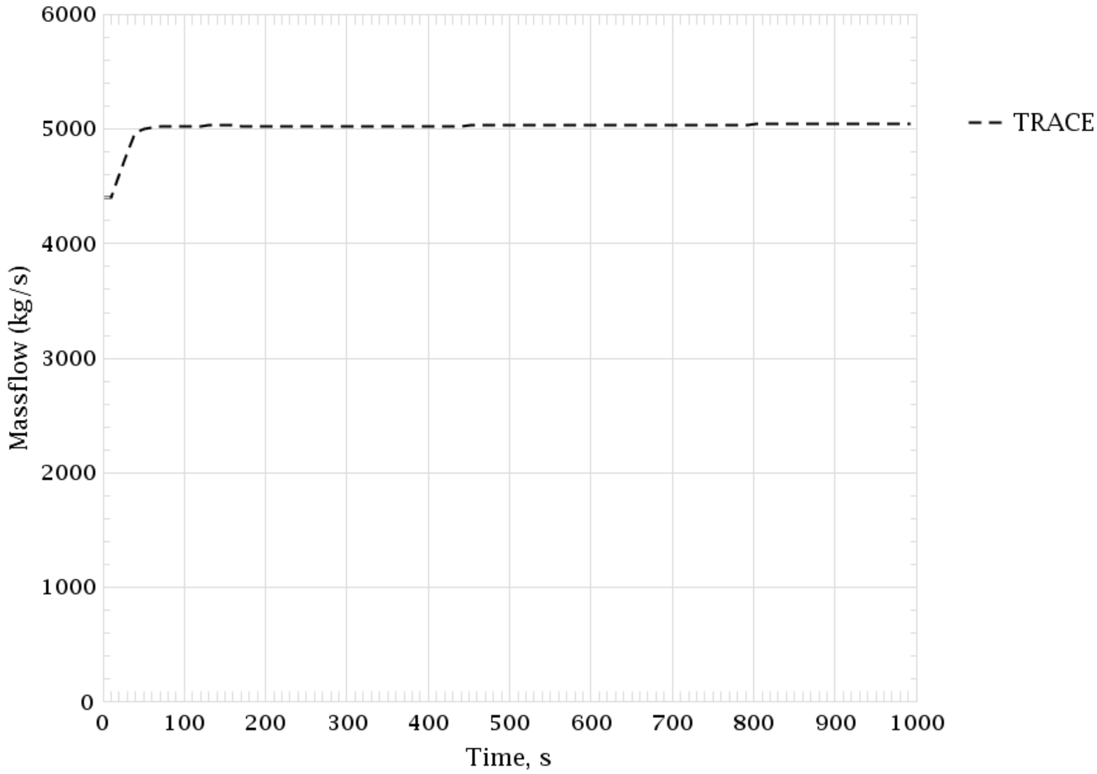

Figure 4-73 RCP-3 Mass Flow Rate

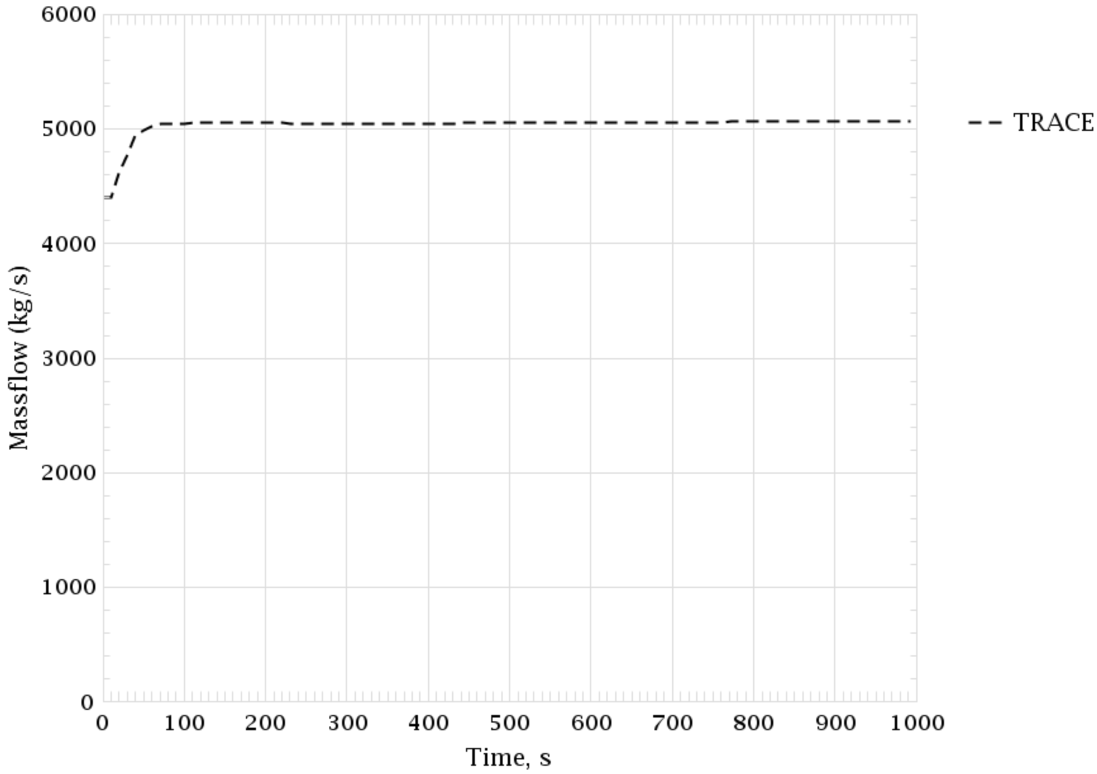

Figure 4-74 RCP-4 Mass Flow Rate

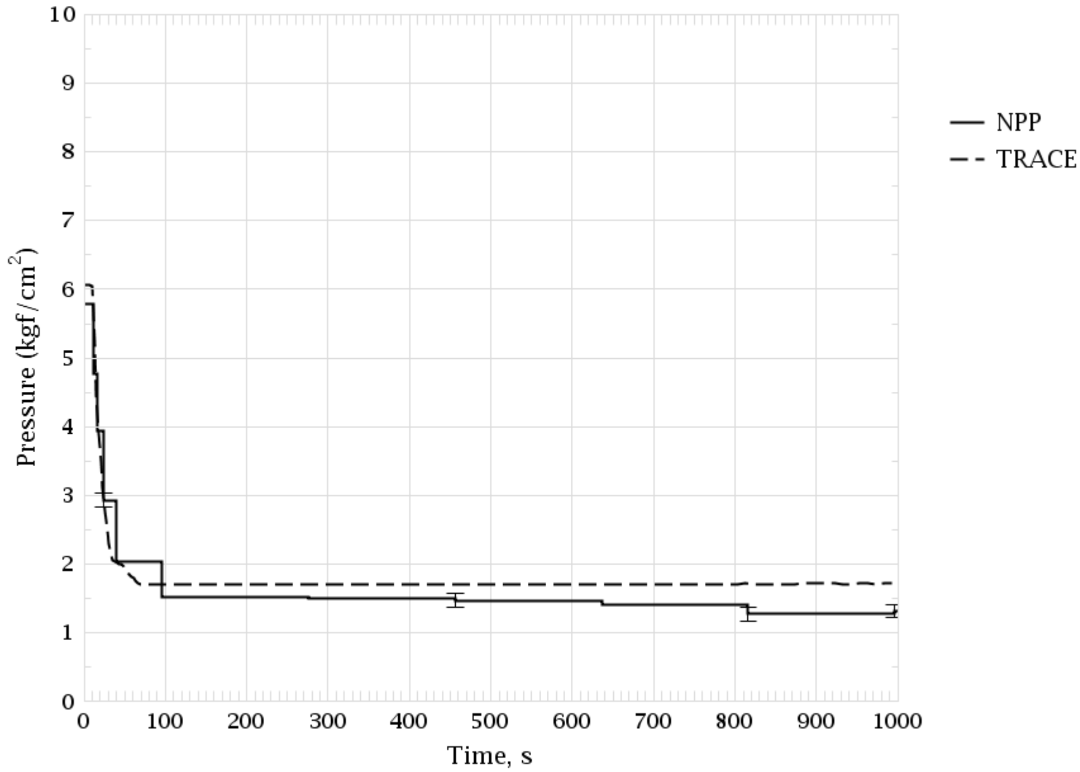

Figure 4-75 Pressure Difference at RCP No.1

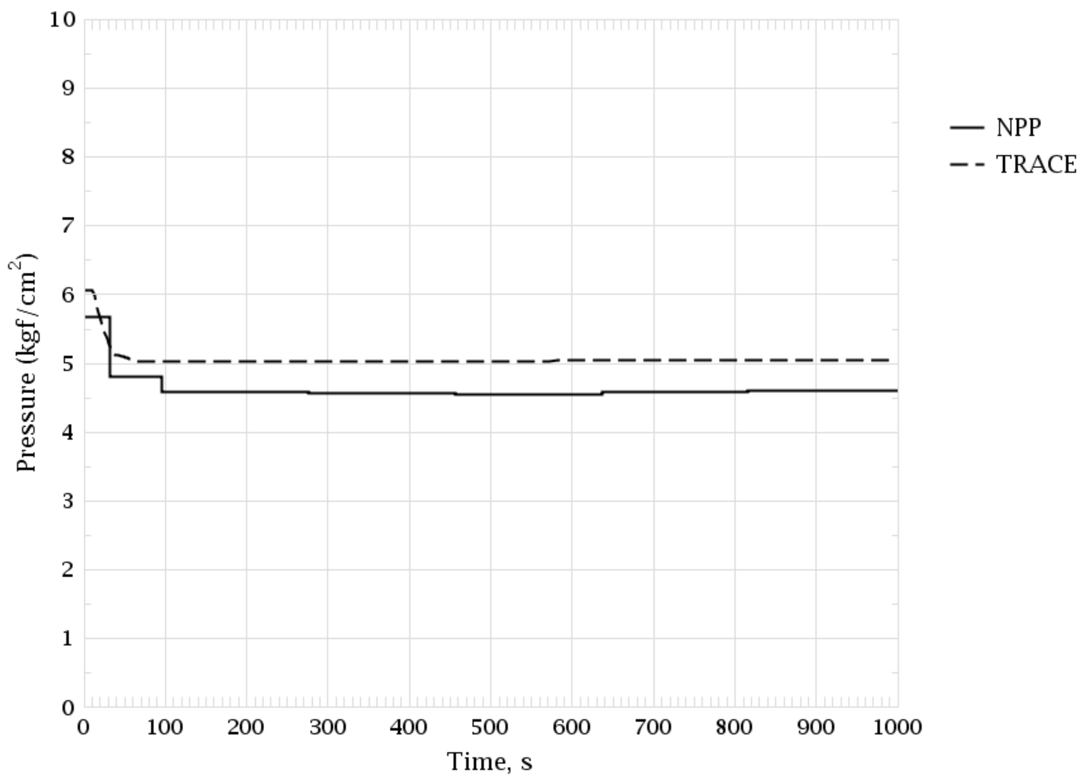

Figure 4-76 Pressure Difference at RCP No.2

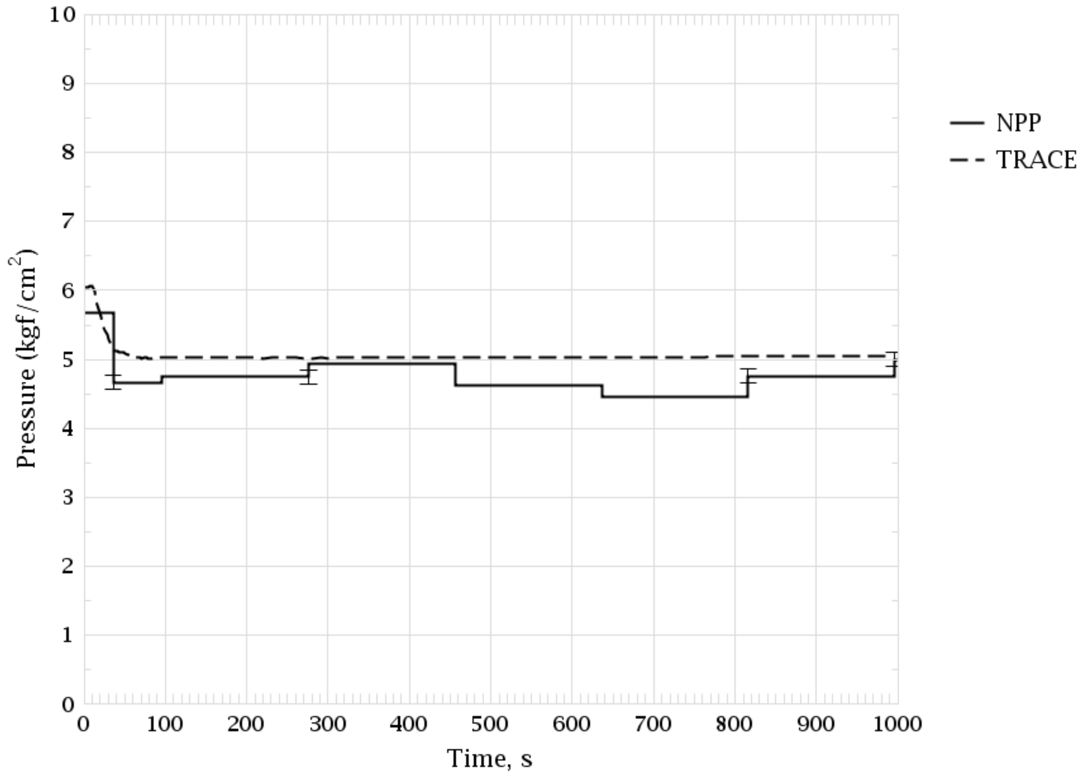

Figure 4-77 Pressure Difference at RCP No.3

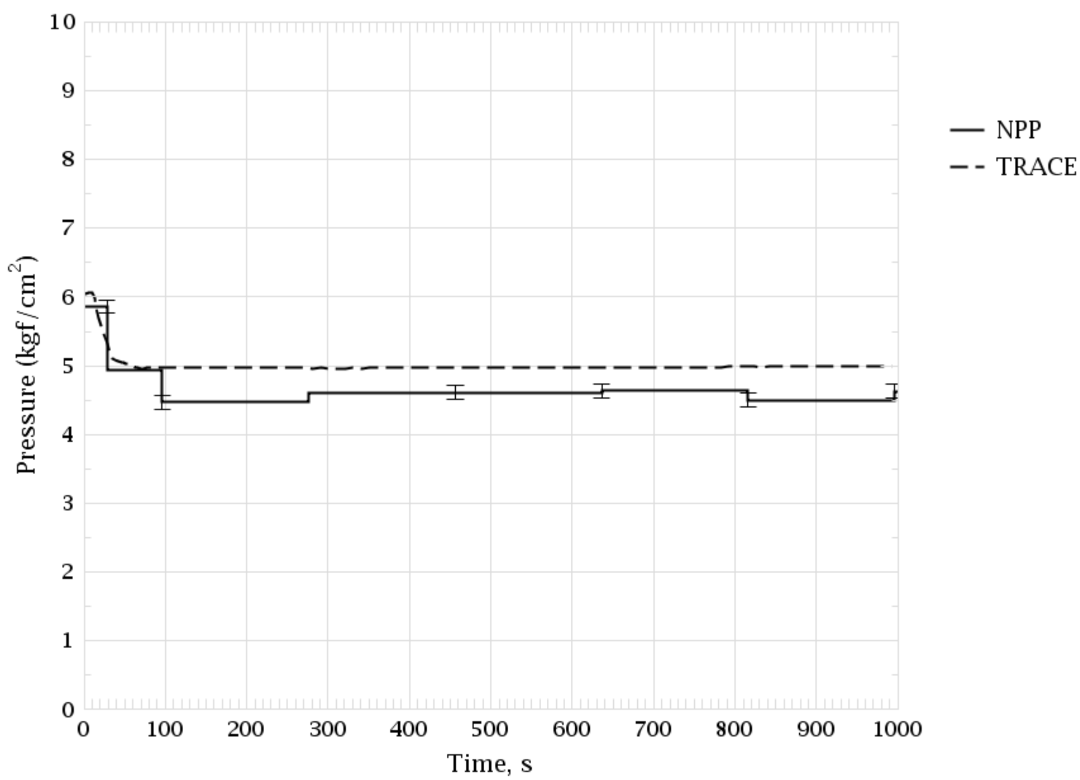

Figure 4-78 Pressure Difference at RCP No.4

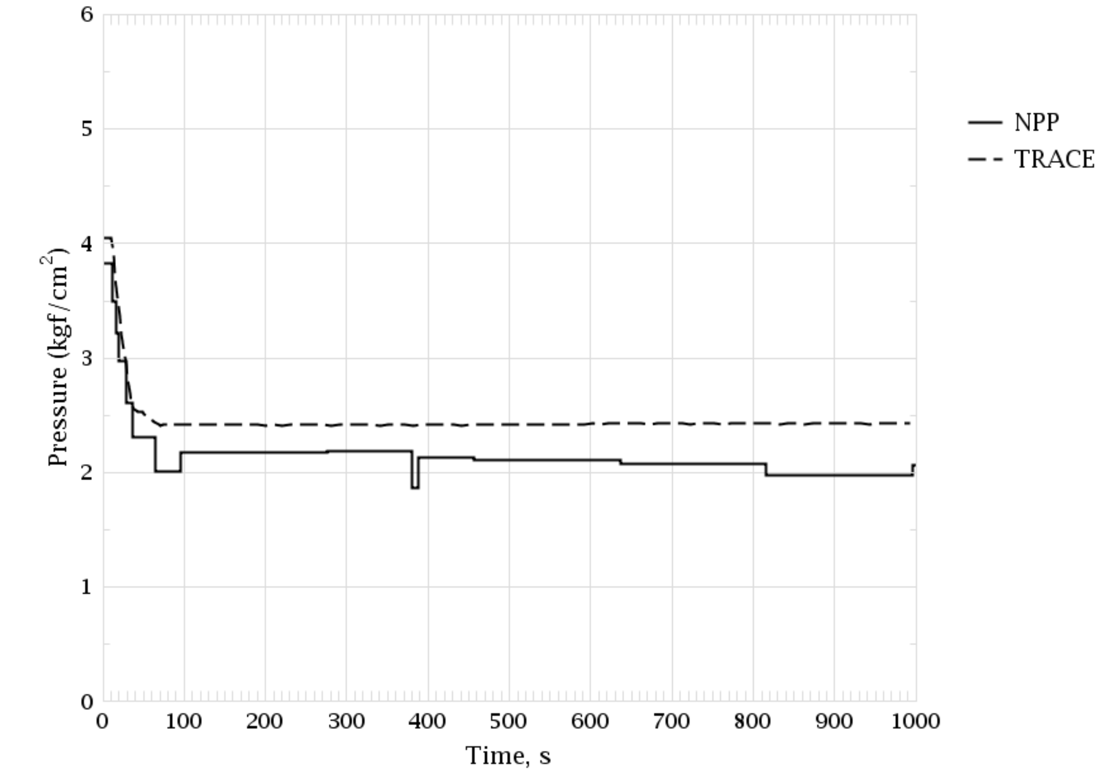

Figure 4-79   Pressure Drop at the Reactor

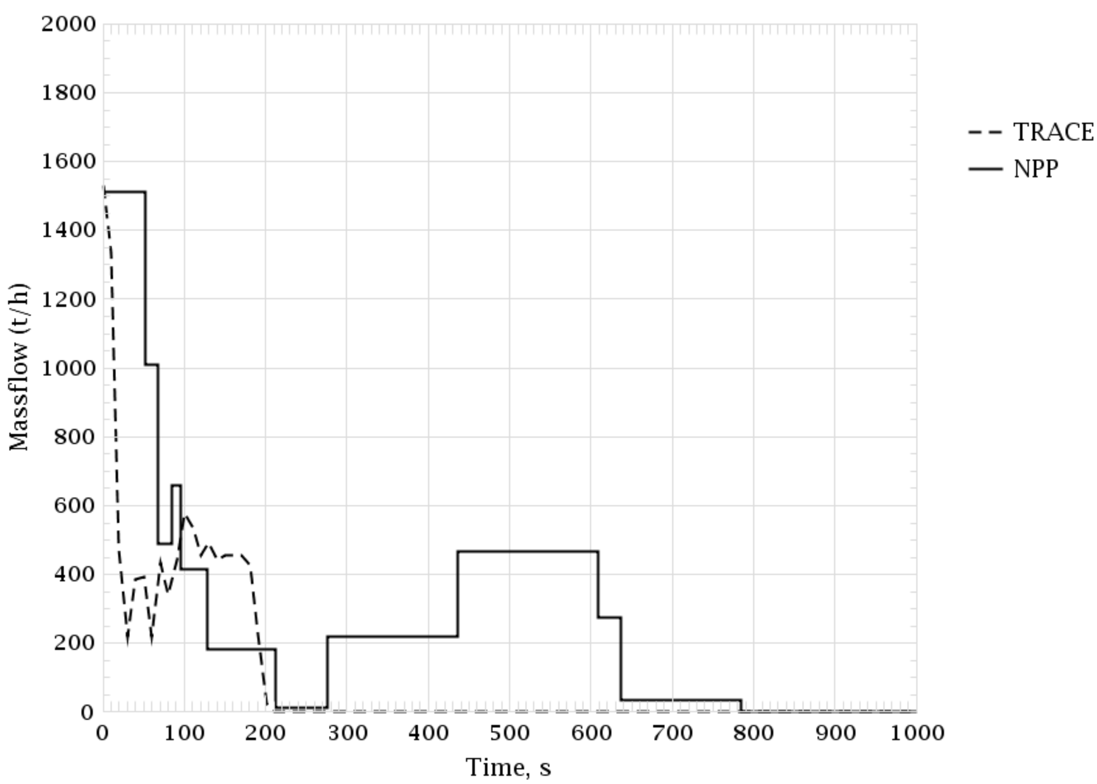

Figure 4-80   SG-1 Feed Water Flow

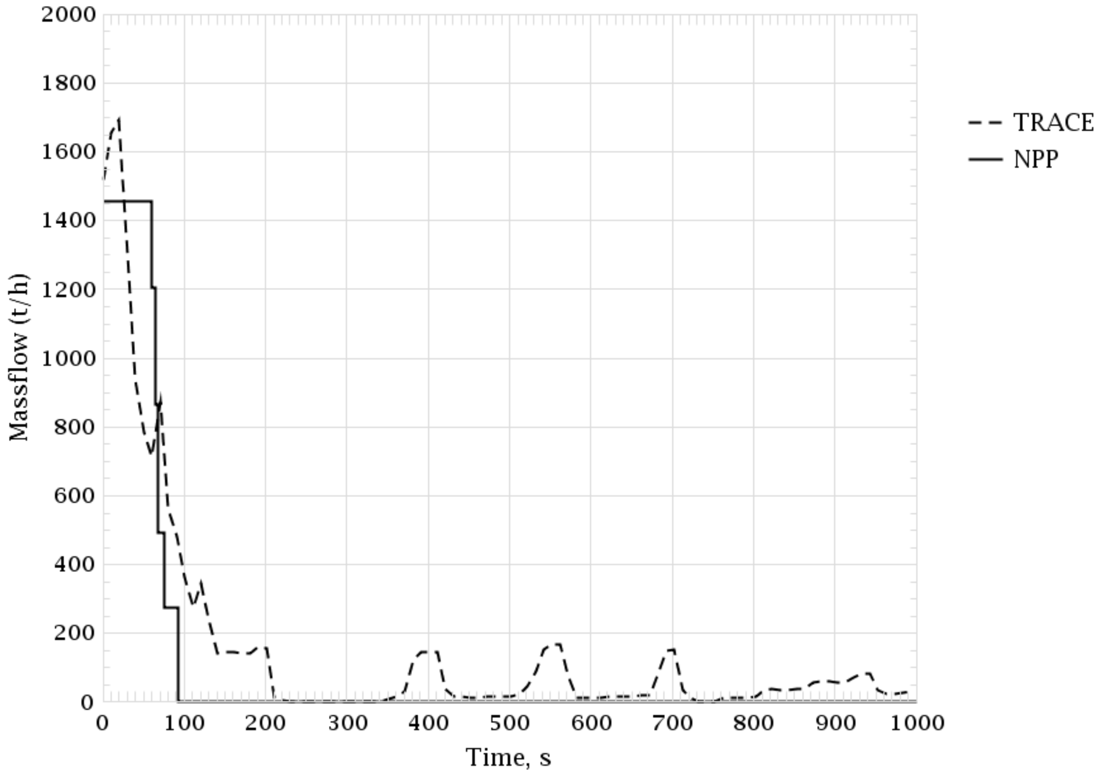

Figure 4-81 SG-2 Feed Water Flow

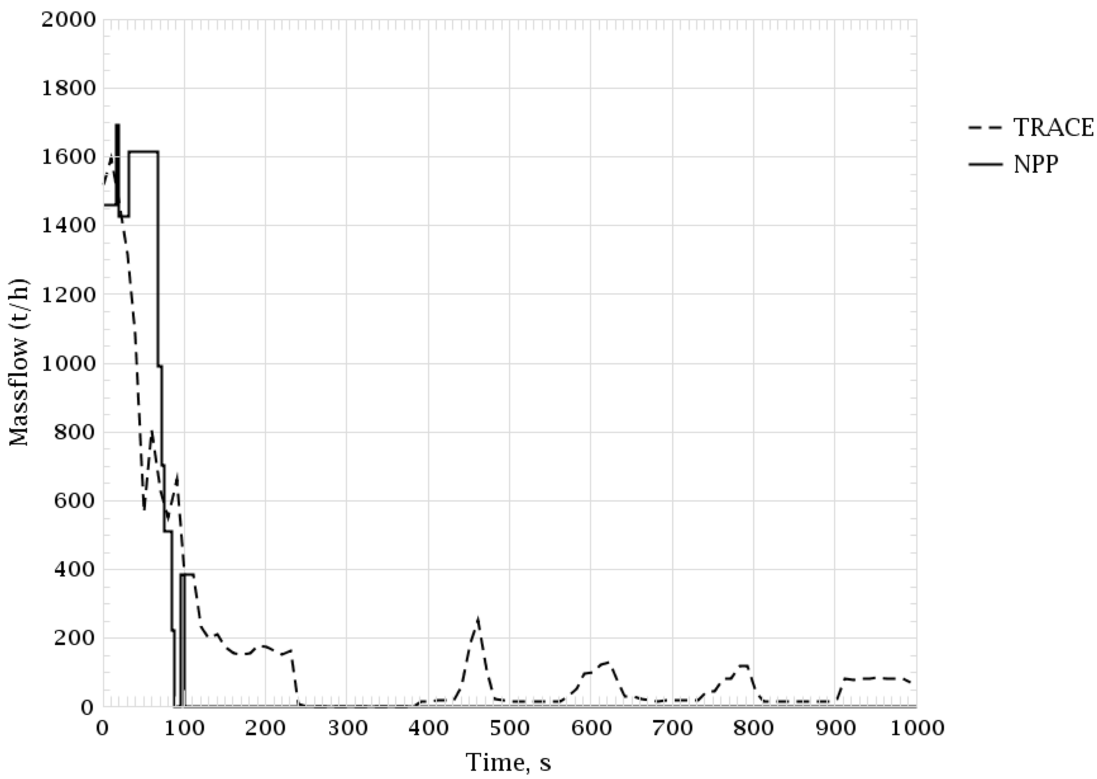

Figure 4-82 SG-3 Feed Water Flow

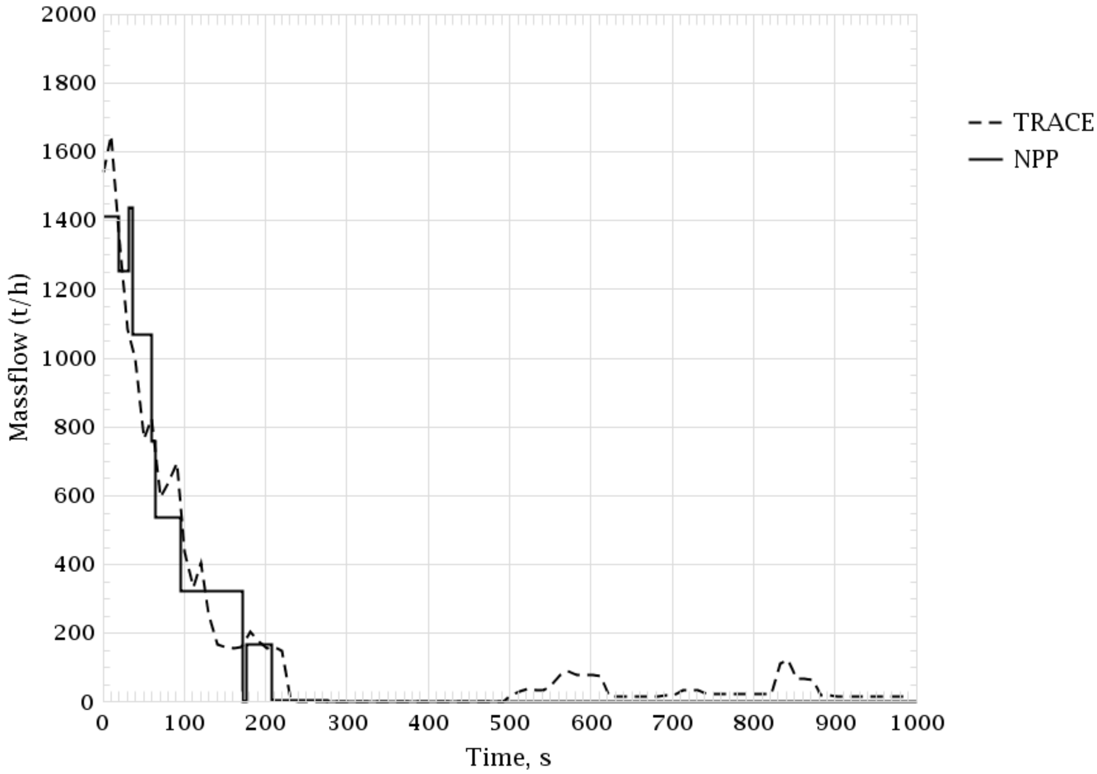

Figure 4-83 SG-4 Feed Water Flow

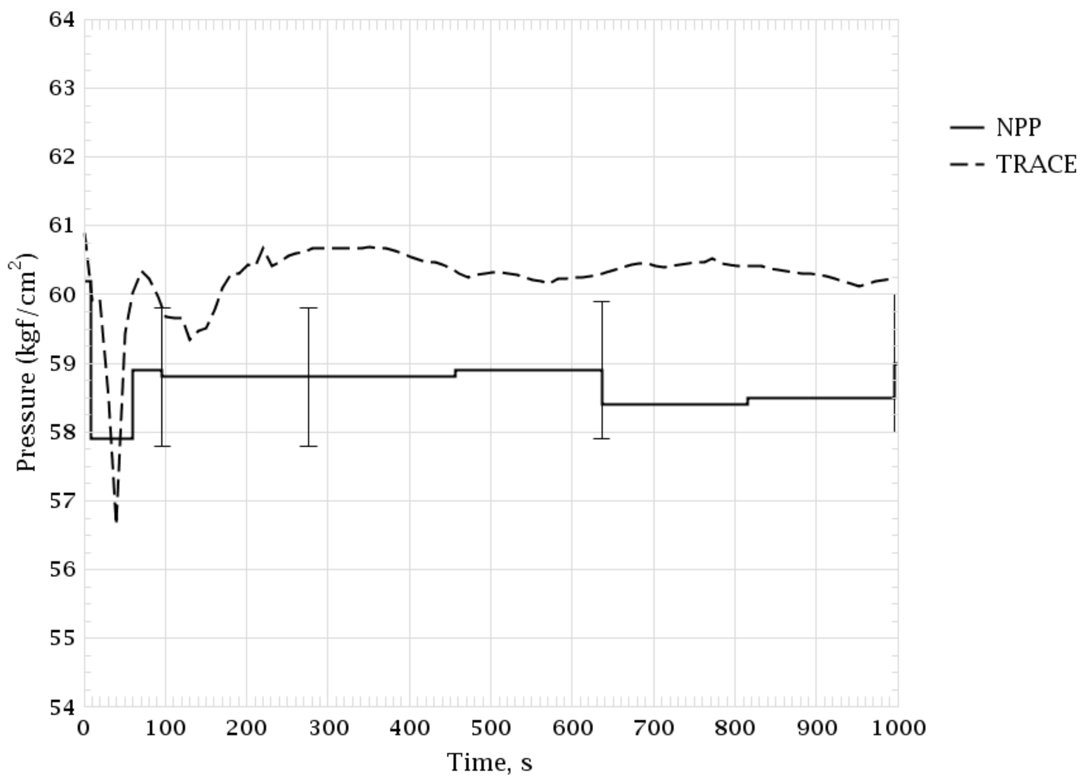

Figure 4-84 MSH Pressure

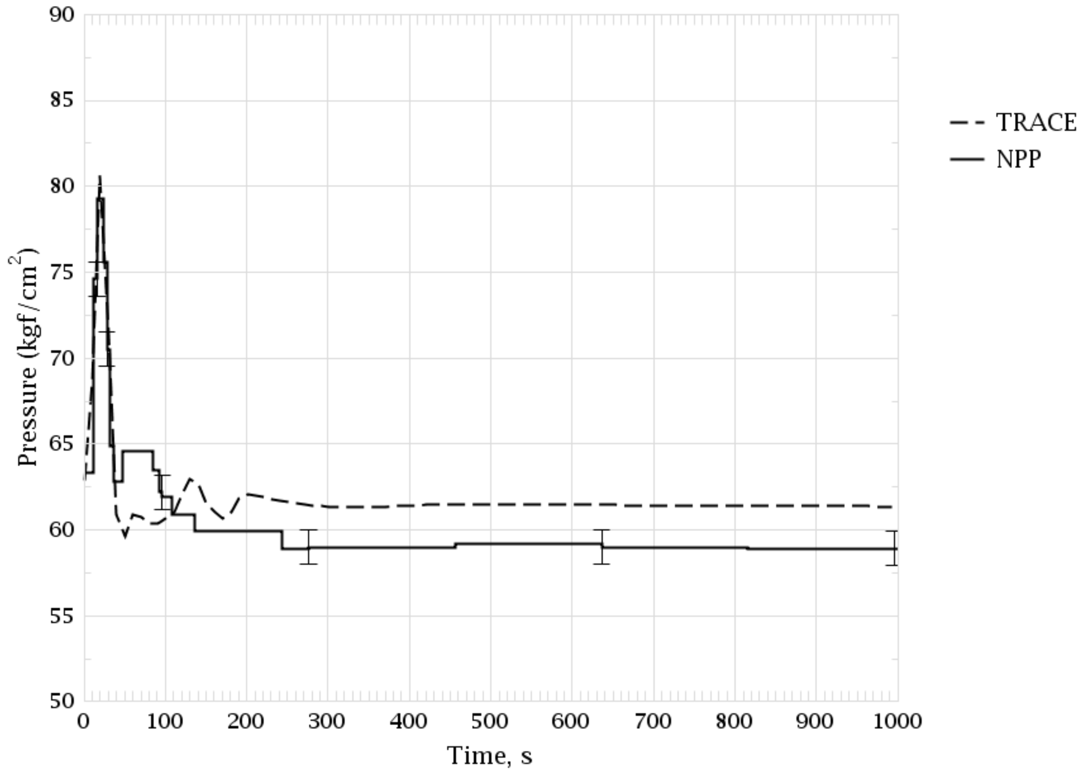

Figure 4-85 SG-1 Pressure

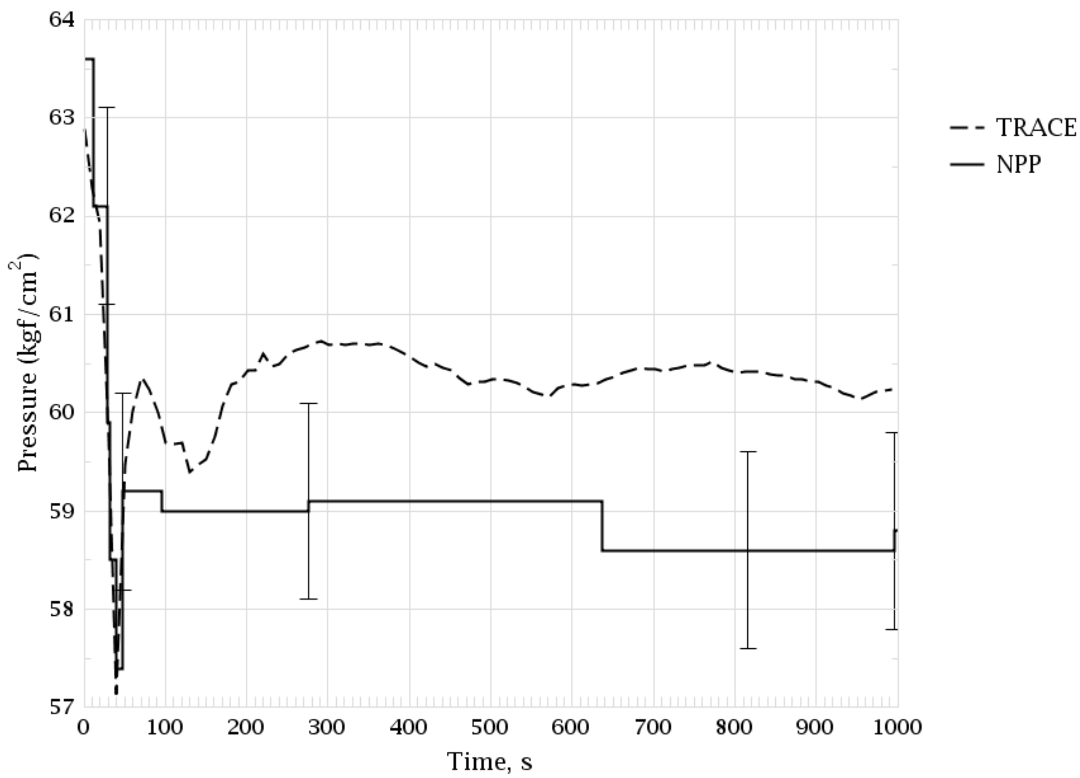

Figure 4-86 SG-2 Pressure

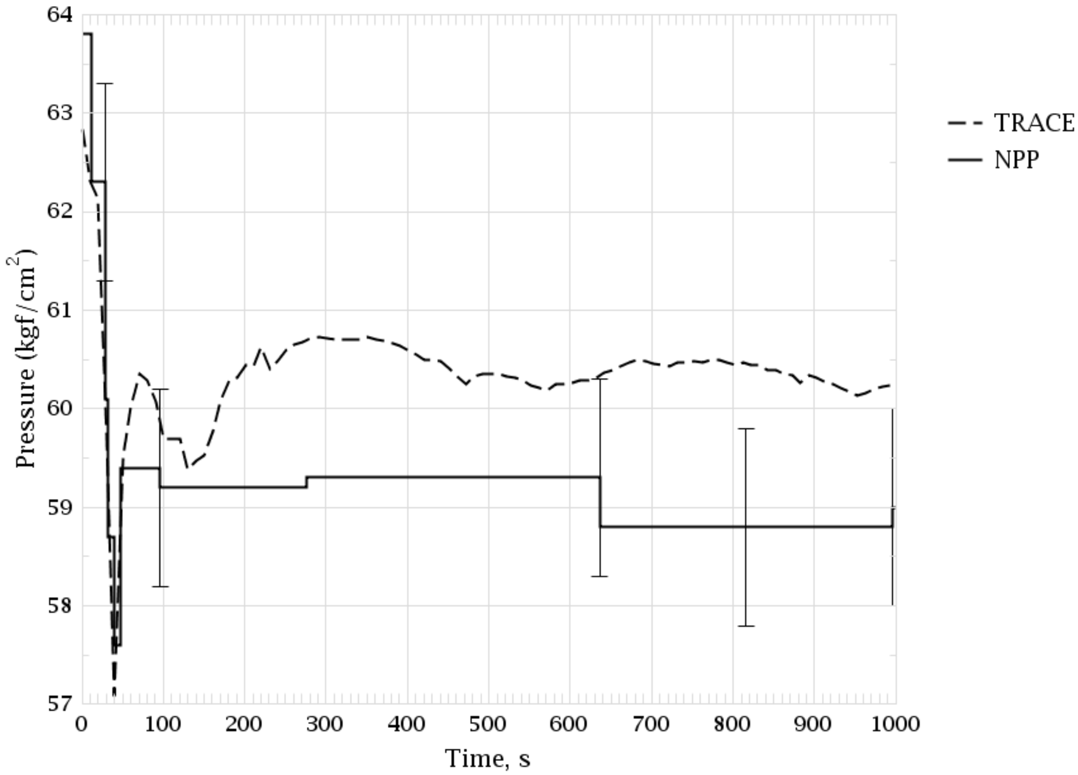

Figure 4-87 SG-3 Pressure

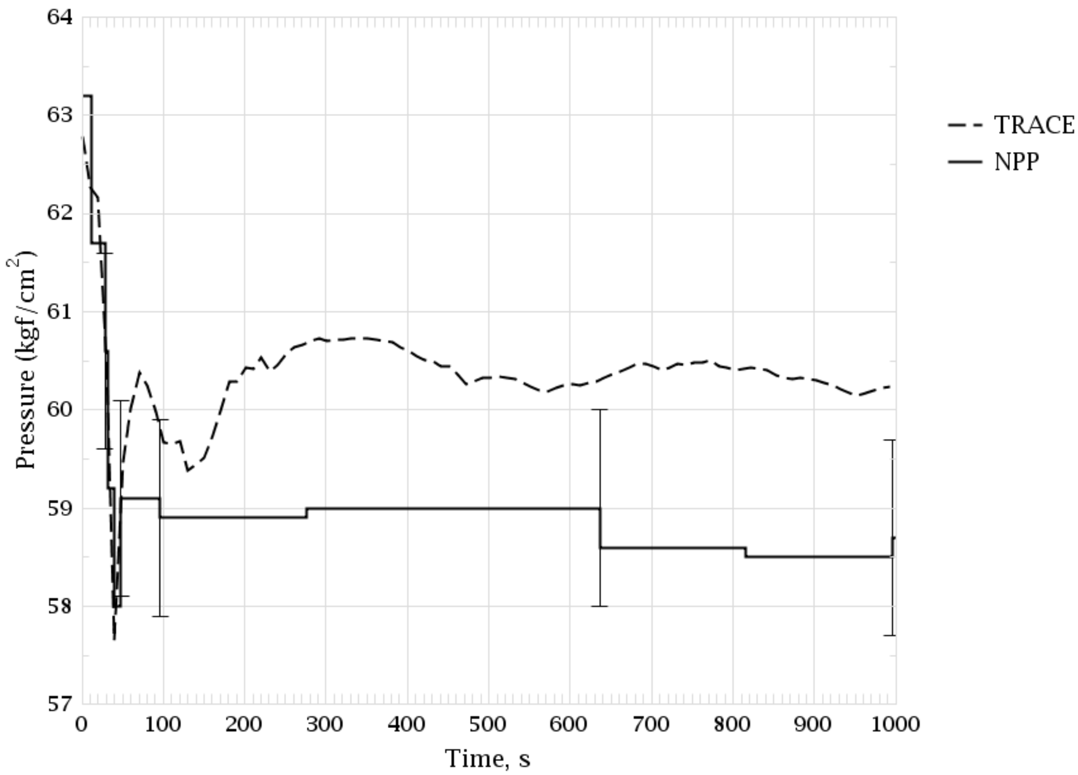

Figure 4-88 SG-4 Pressure

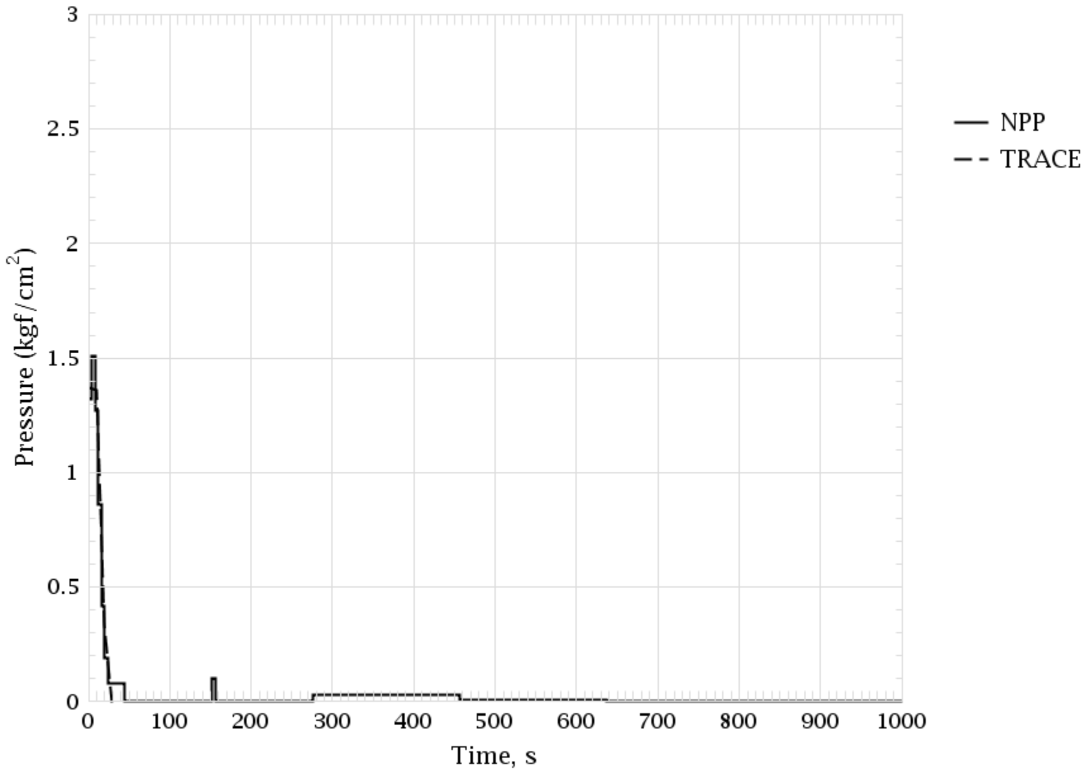

Figure 4-89 Pressure Loss at SG-1 Primary Side

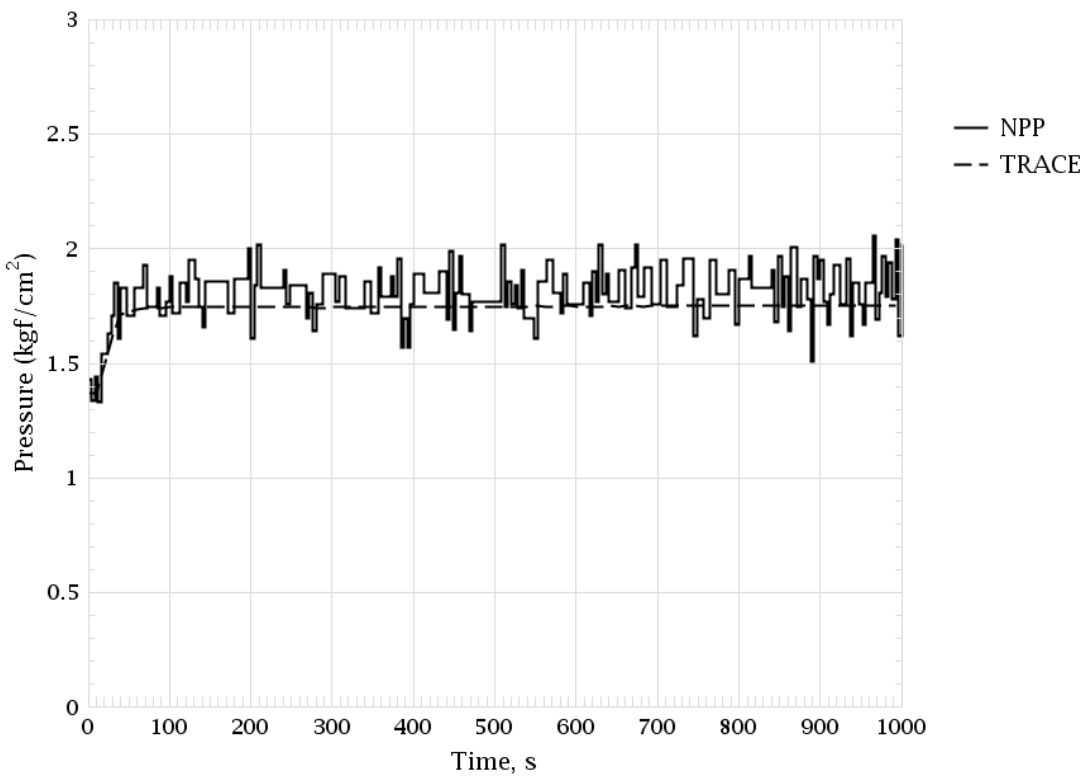

Figure 4-90 Pressure Loss at SG-2 Primary Side

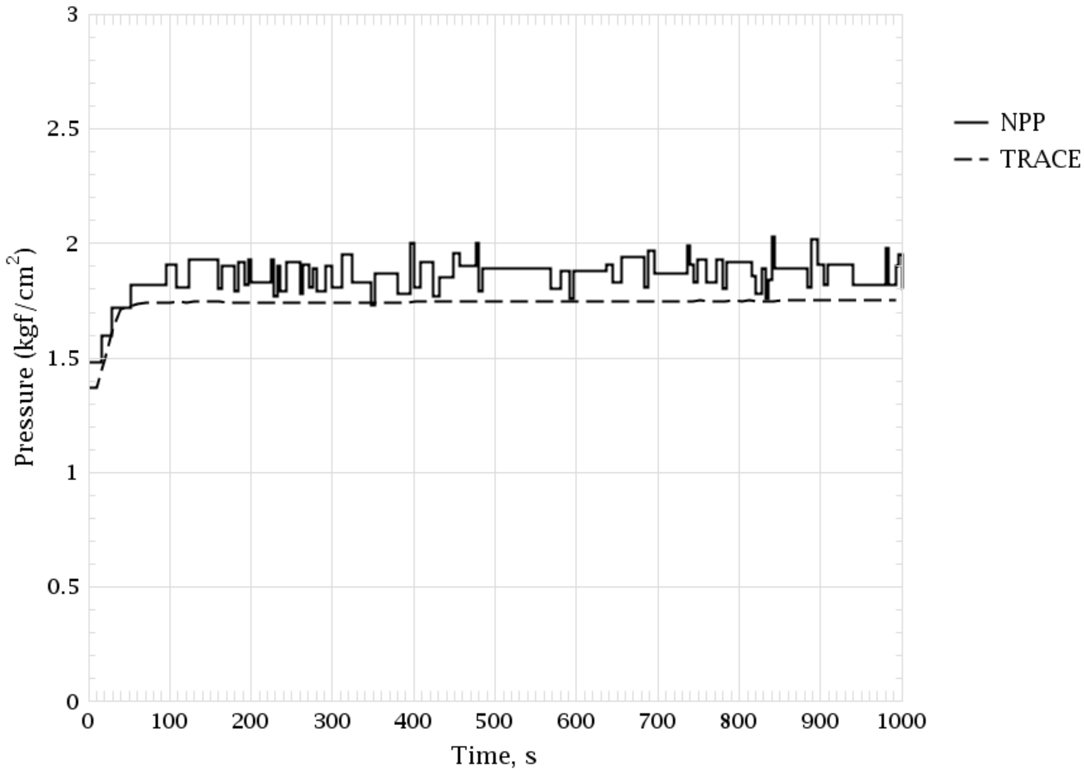

Figure 4-91 Pressure Loss at SG-3 Primary Side

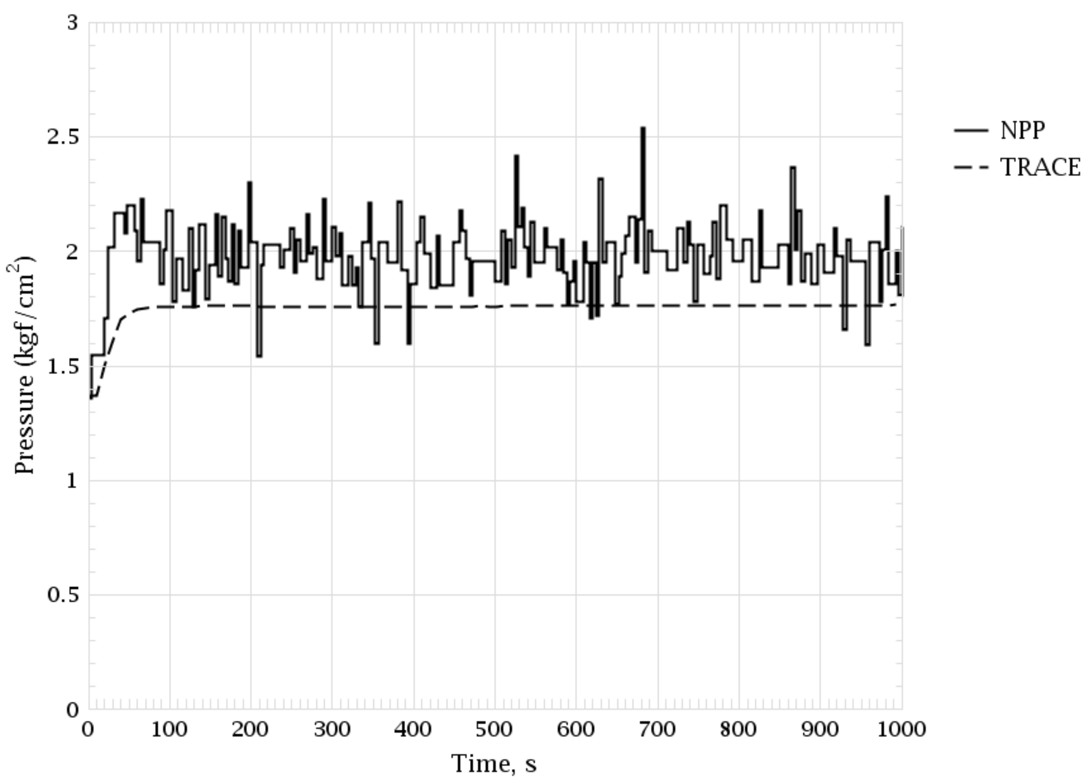

Figure 4-92 Pressure Loss at SG-4 Primary Side

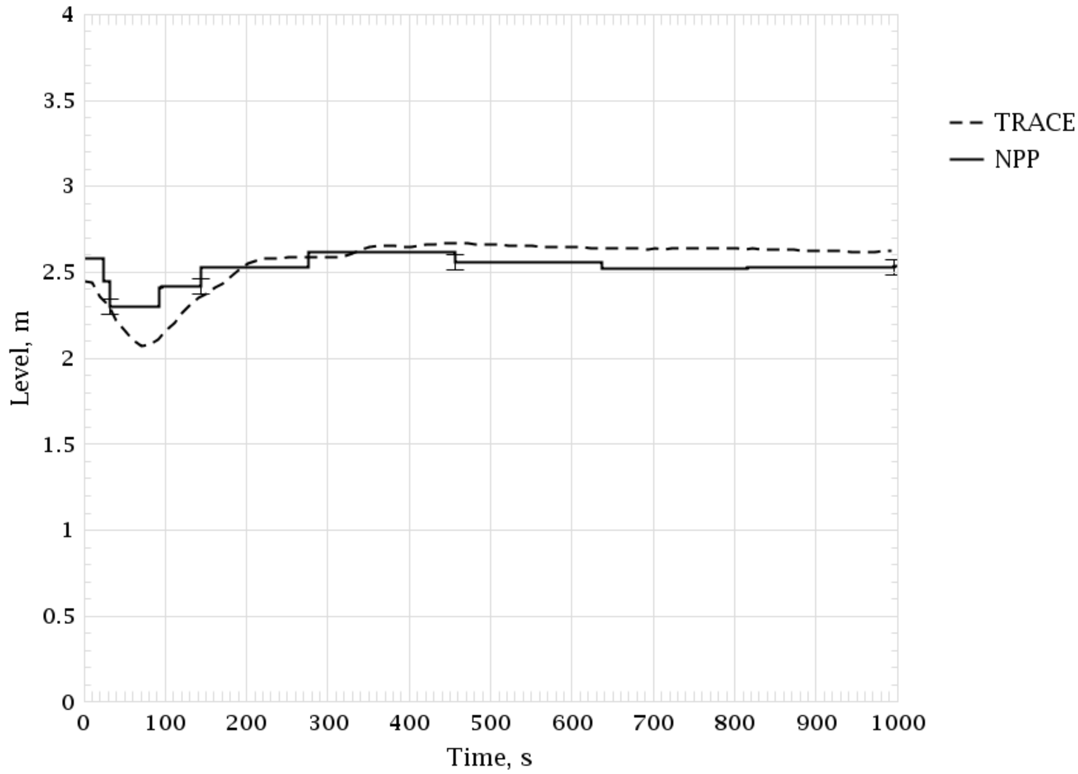

Figure 4-93 SG-1 Level (Wide Range)

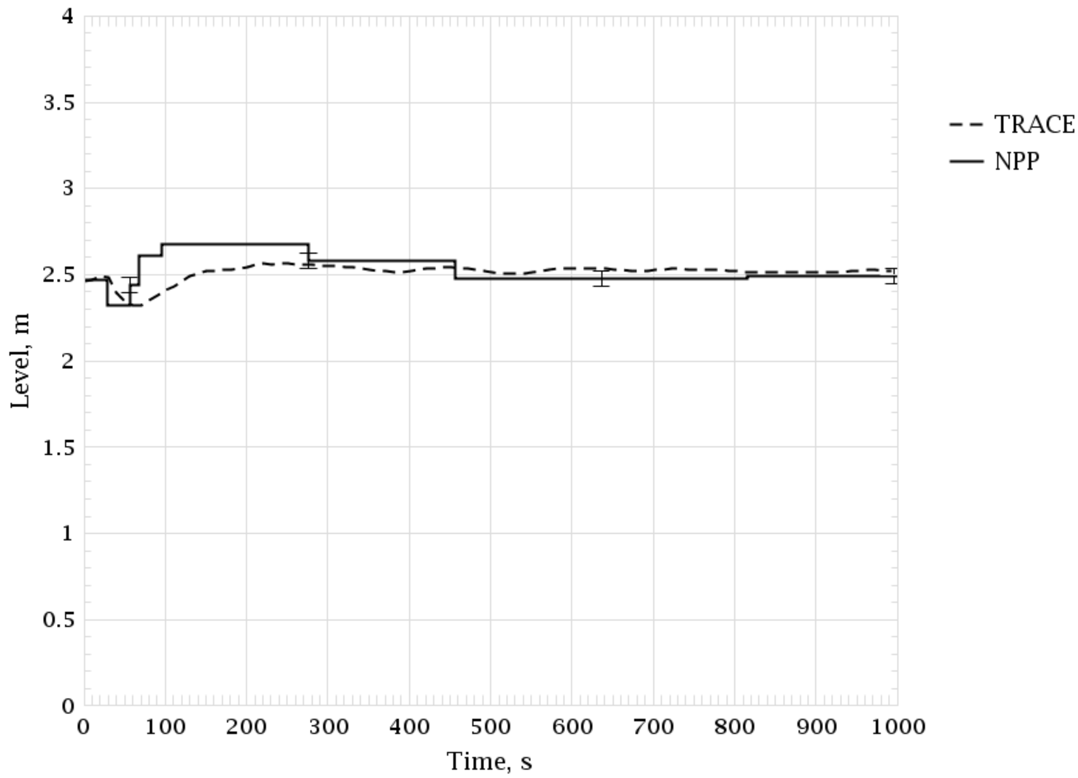

Figure 4-94 SG-2 Level (Wide Range)

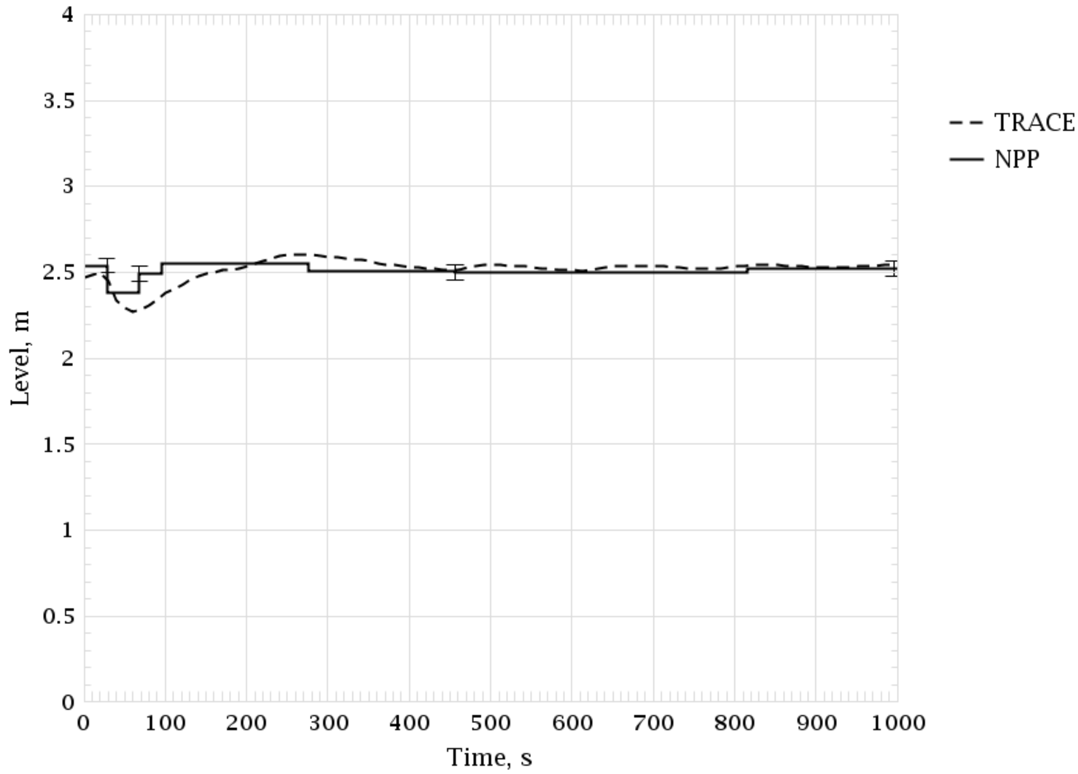

Figure 4-95 SG-3 Level (Wide Range)

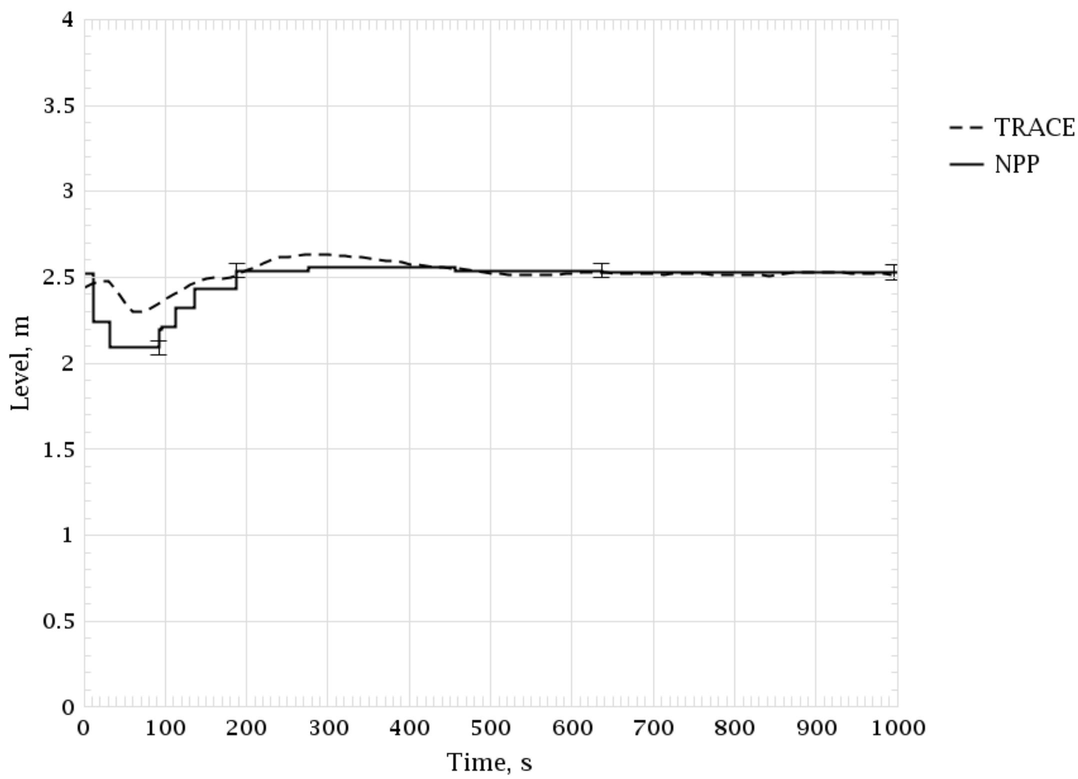

Figure 4-96 SG-4 Level (Wide Range)

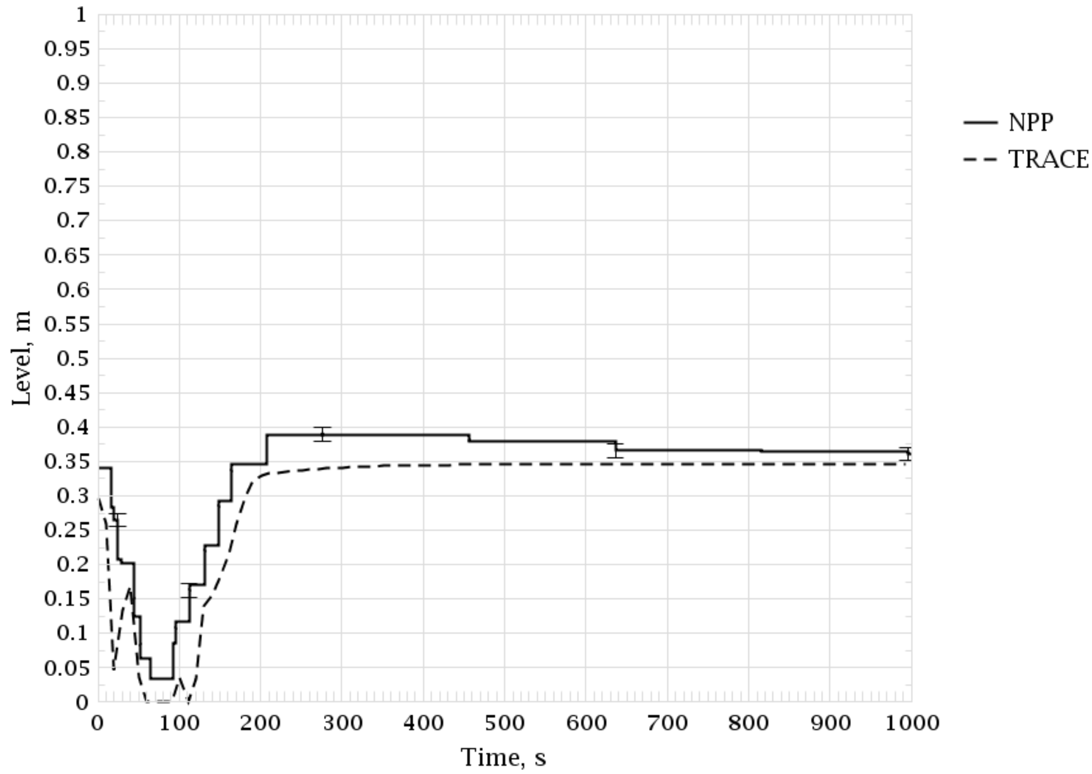

Figure 4-97 SG-1 Level (Narrow Range)

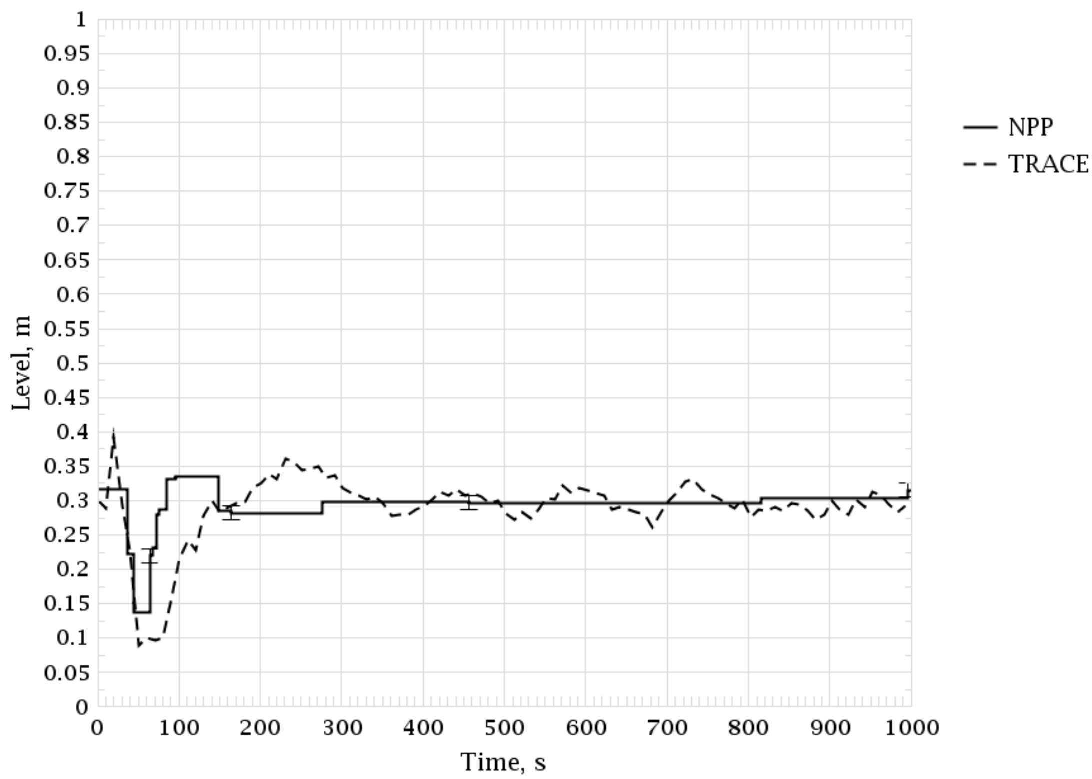

Figure 4-98 SG-2 Level (Narrow Range)

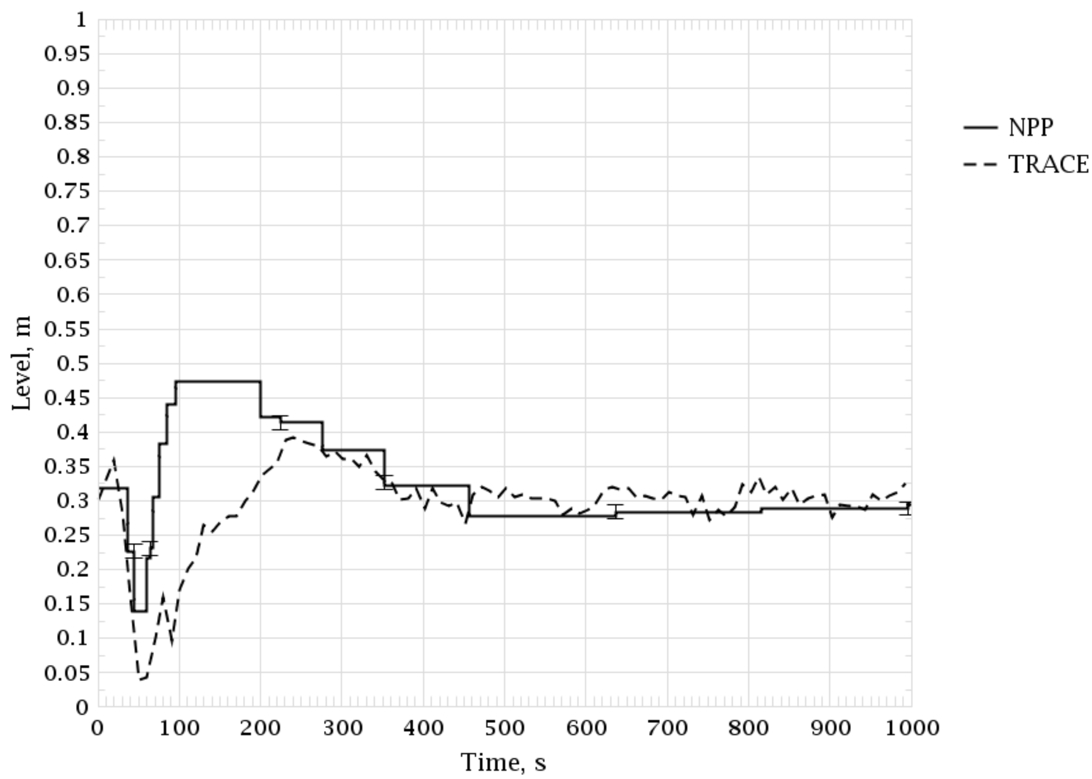

Figure 4-99 SG-3 Level (Narrow Range)

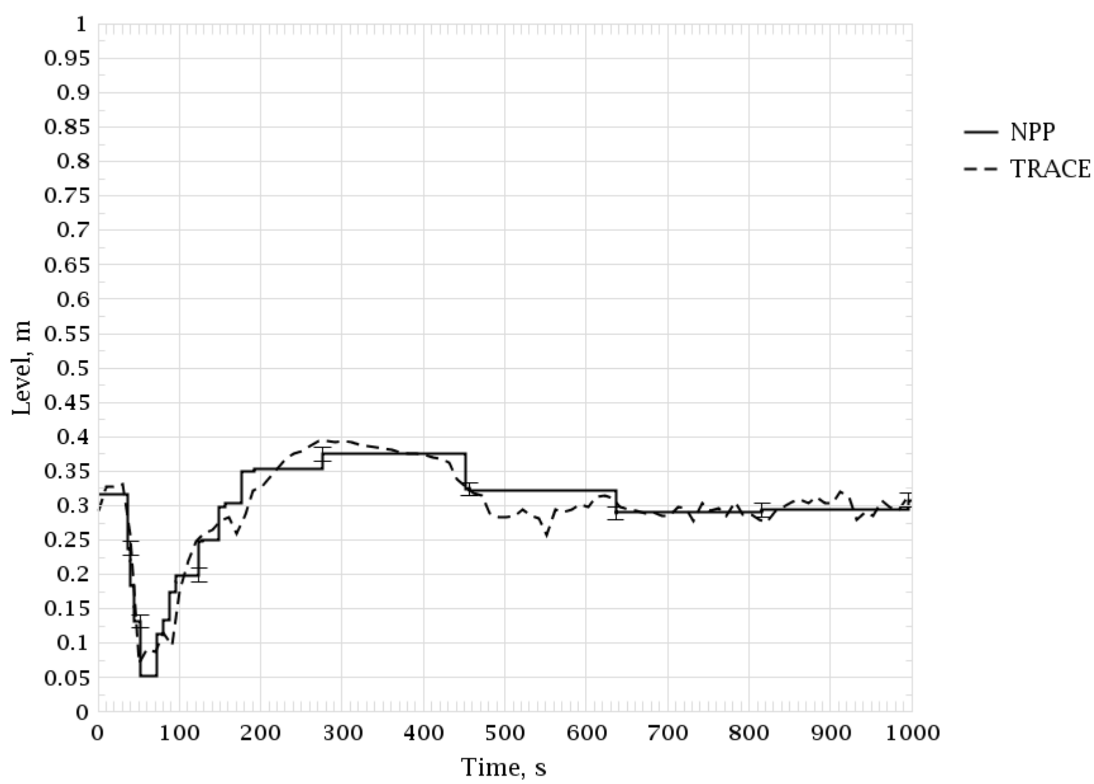

Figure 4-100 SG-4 Level (Narrow Range)

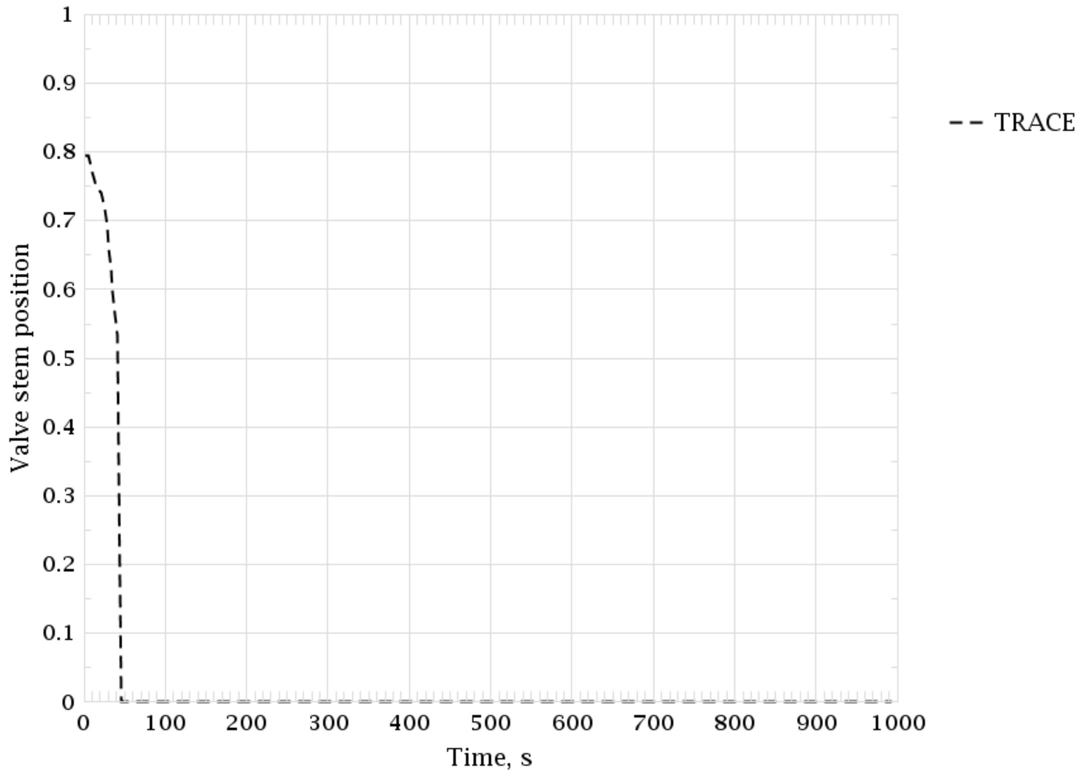

Figure 4-101 Turbine Stop Valves Stem Position

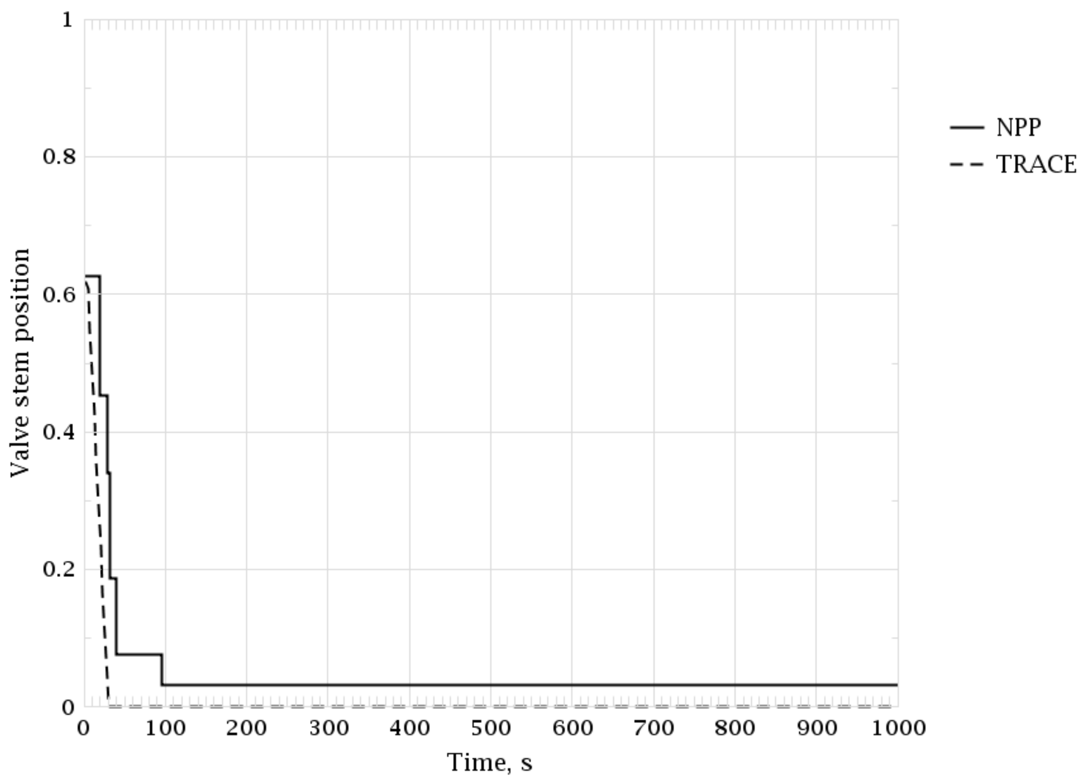

Figure 4-102 Main Feed Water Control Valve No.1 Stem Position

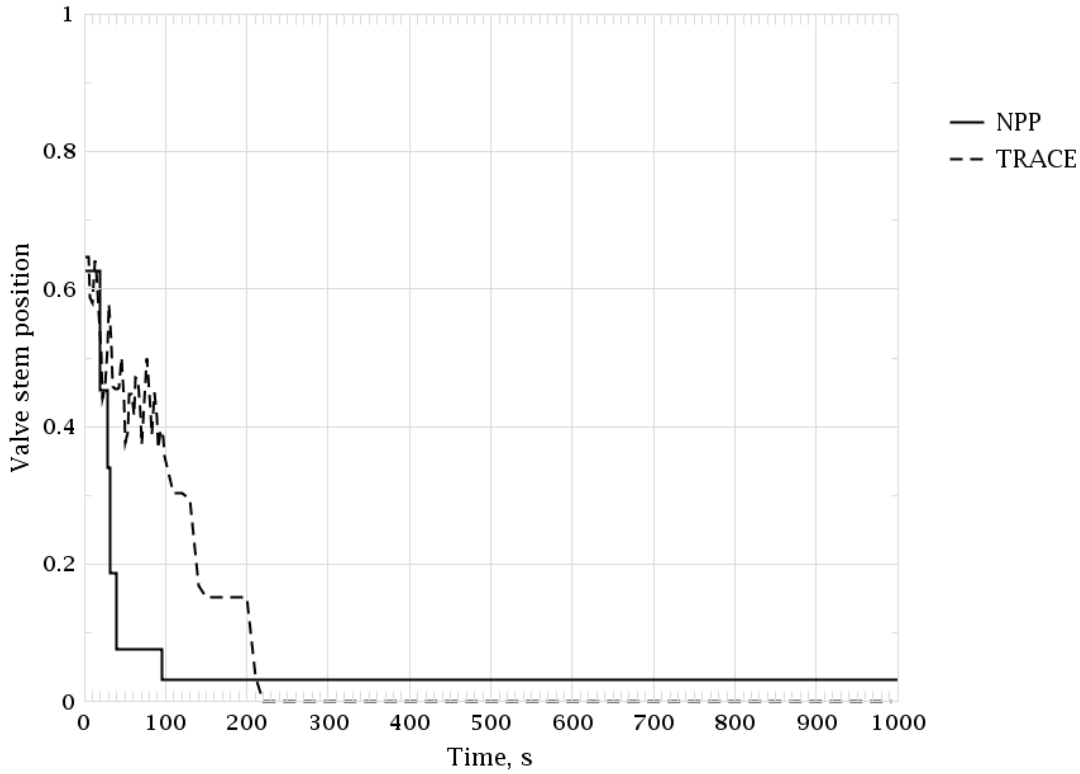

Figure 4-103 Main Feed Water Control Valve No.2 Stem Position

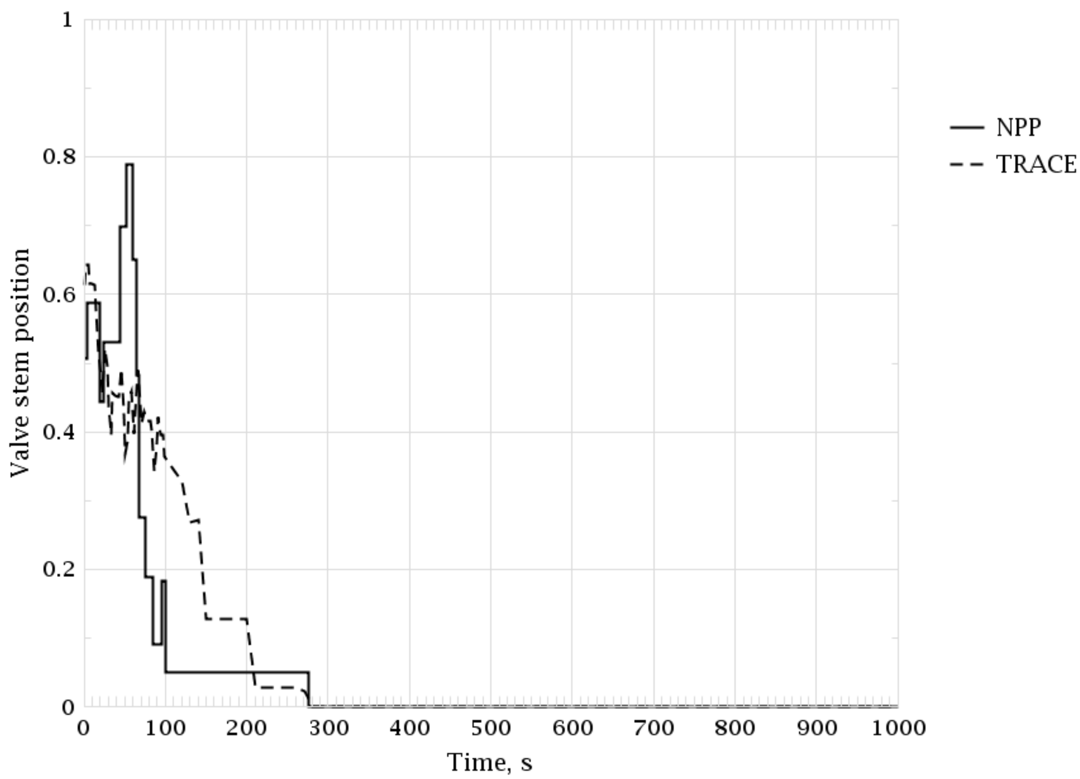

Figure 4-104 Main Feed Water Control Valve No.3 Stem Position

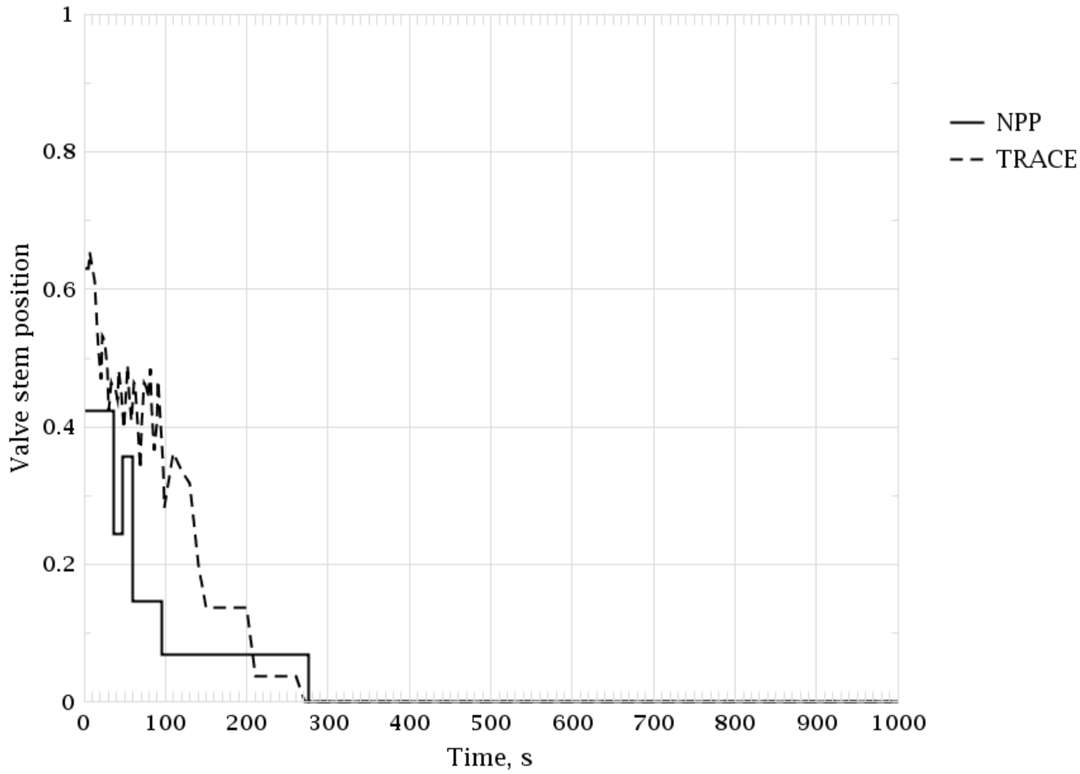

Figure 4-105 Main Feed Water Control Valve No.4 Stem Position

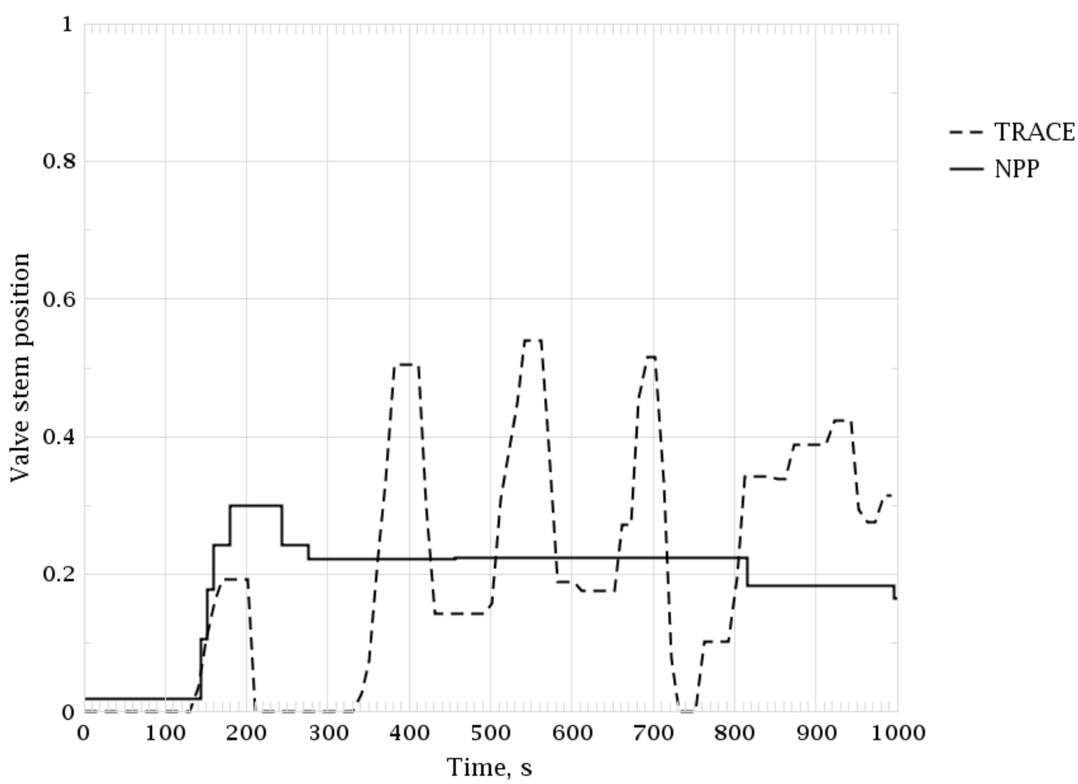

Figure 4-106 Start-up/Shutdown Feed Water Control Valve No.1 Stem Position

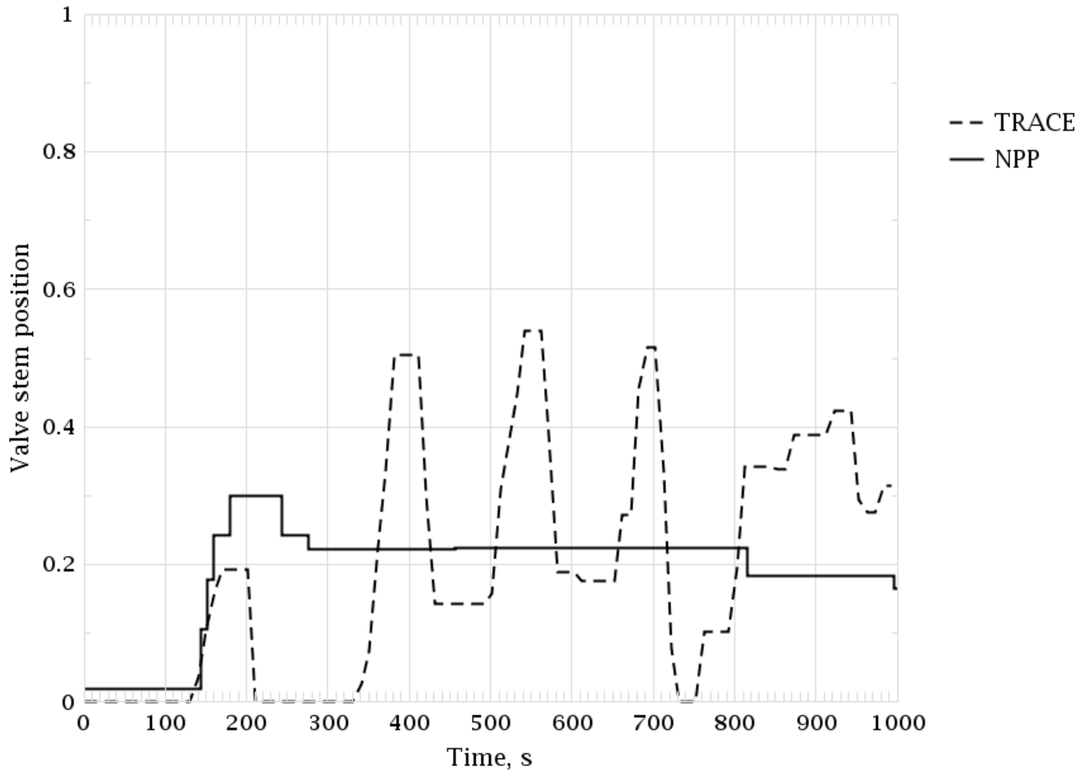

Figure 4-107 Start-up/Shutdown Feed Water Control Valve No.2 Stem Position

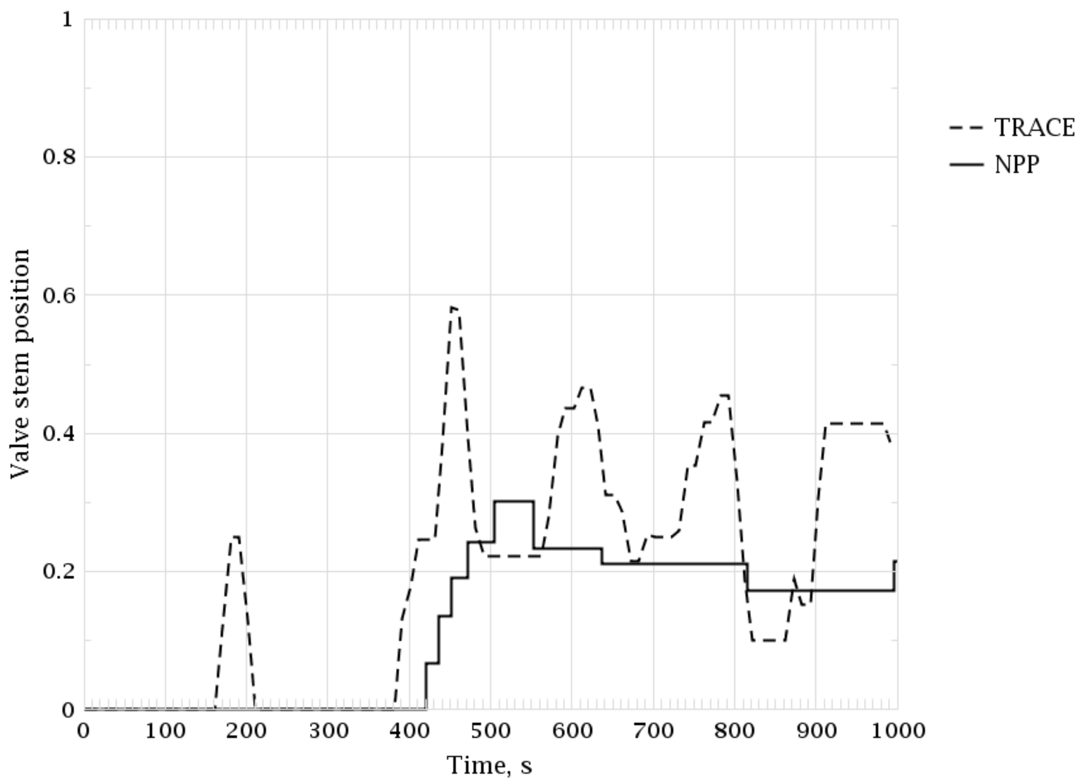

Figure 4-108 Start-up/Shutdown Feed Water Control Valve No.3 Stem Position

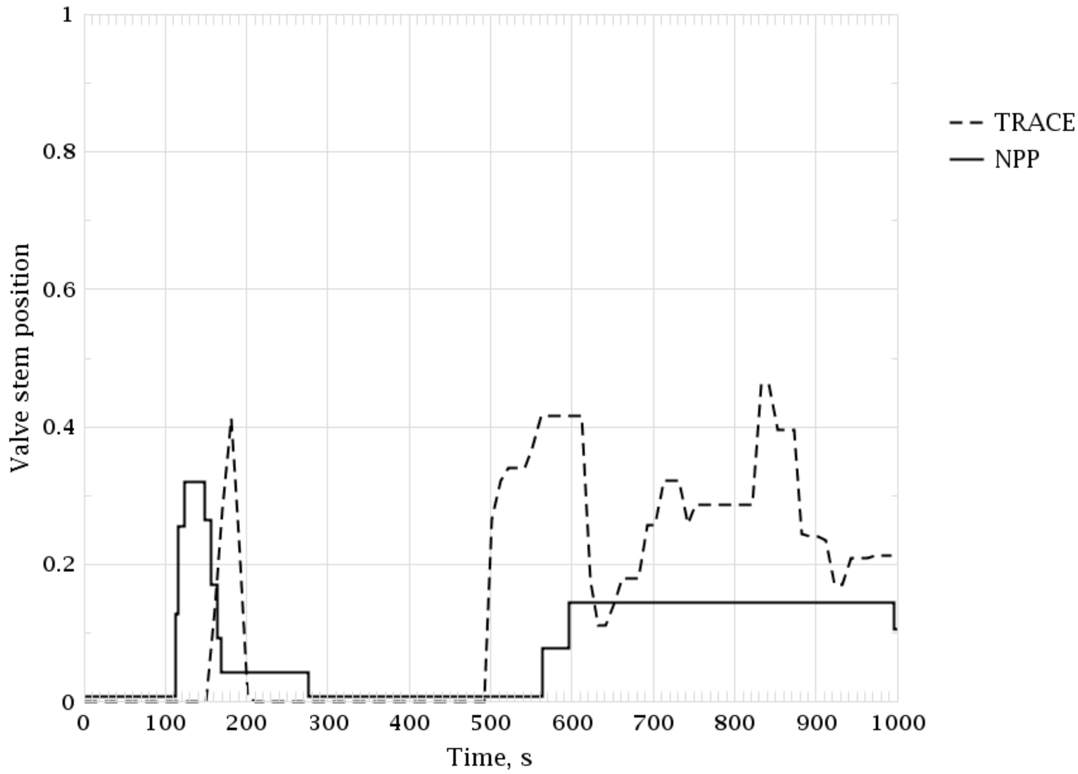

Figure 4-109 Start-up/Shutdown Feed Water Control Valve No.4 Stem Position

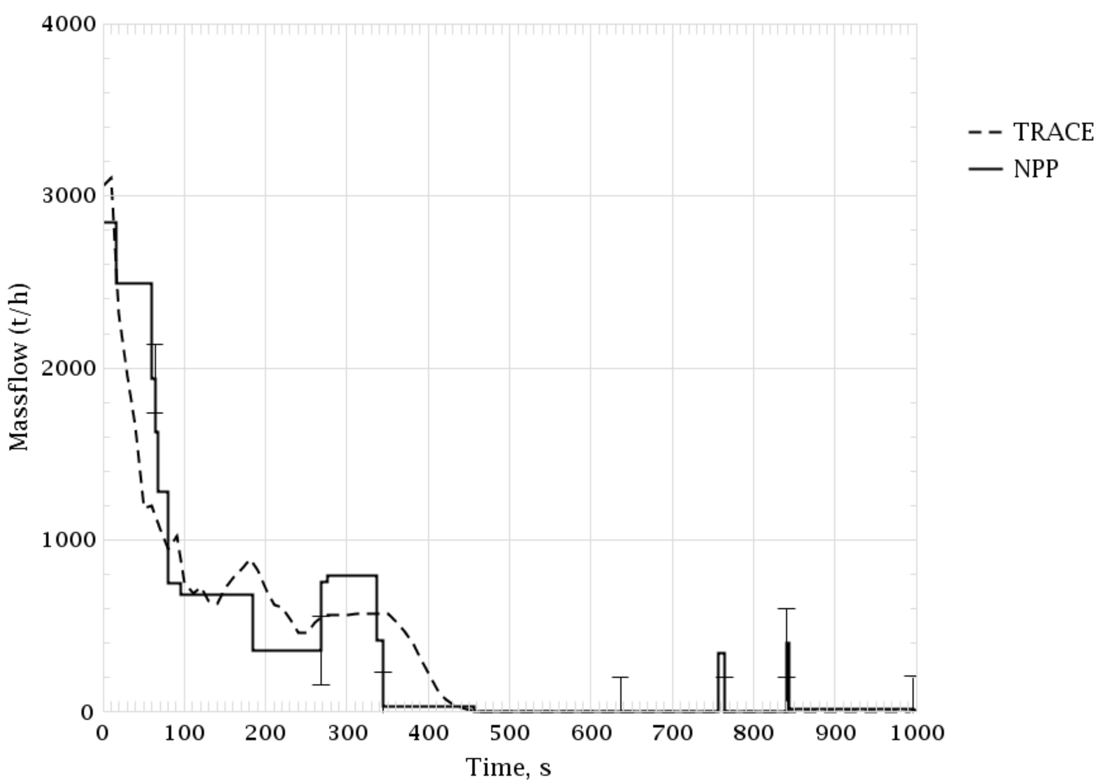

Figure 4-110 MFW Pump No.1 Flow Rate

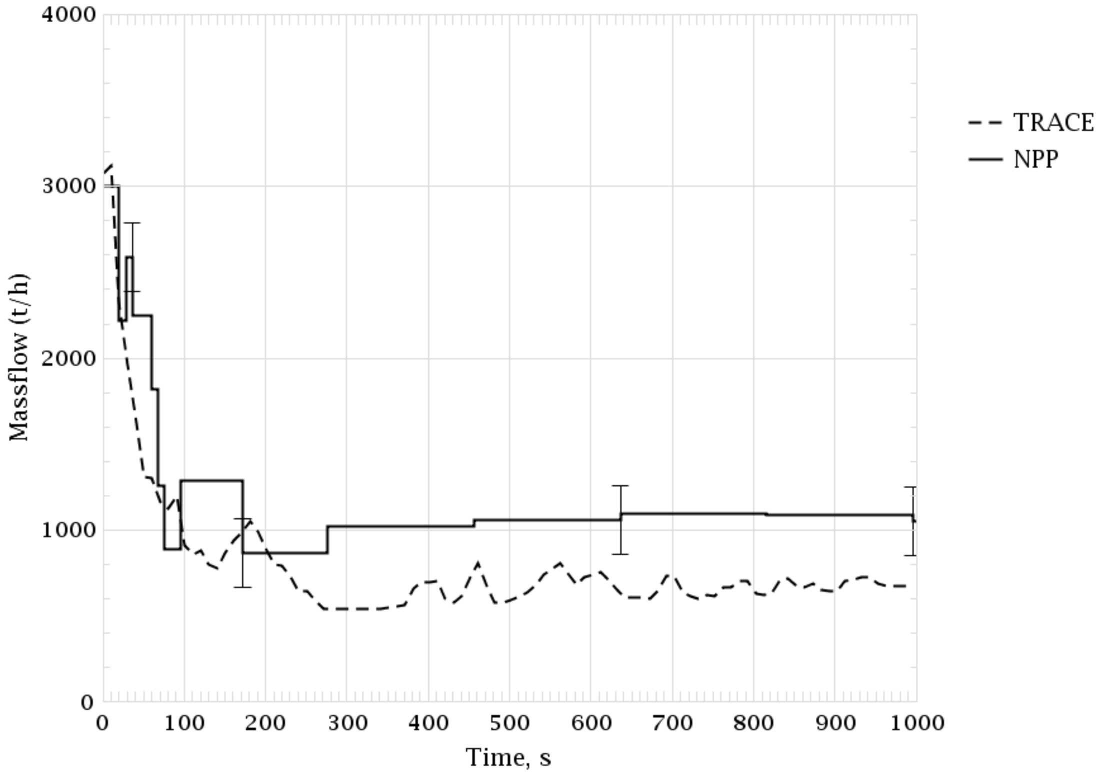

Figure 4-111 MFW Pump No.2 Flow Rate

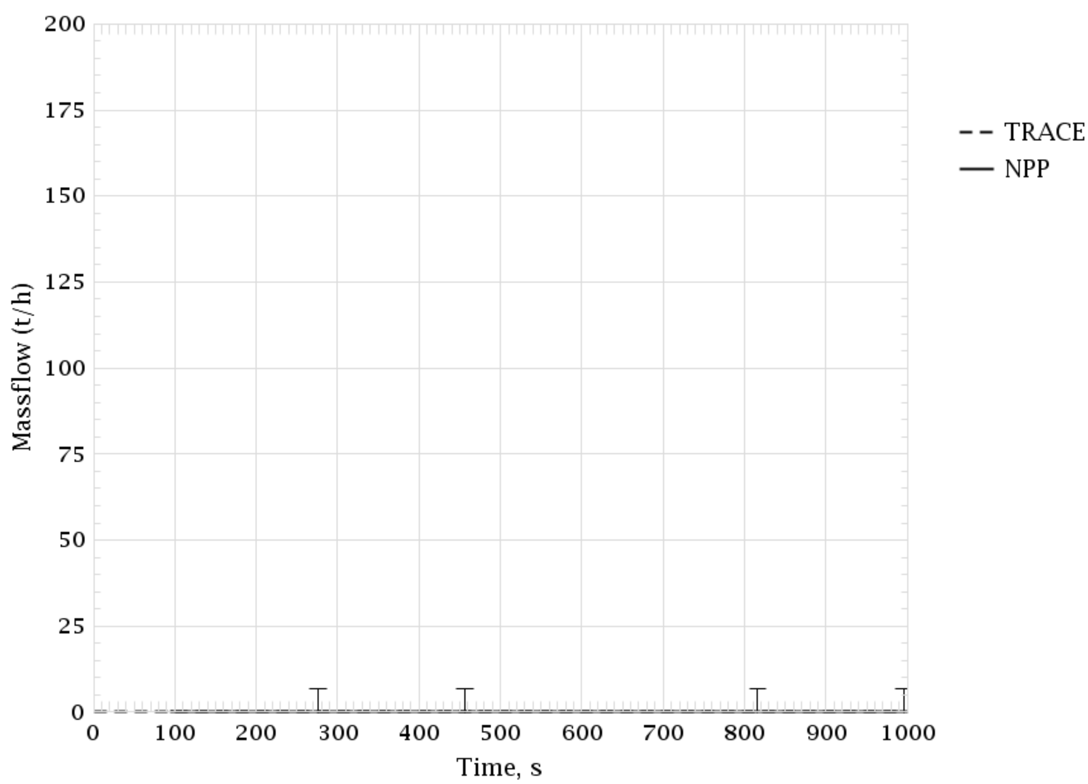

Figure 4-112 AFW Pump No.1 Flow Rate

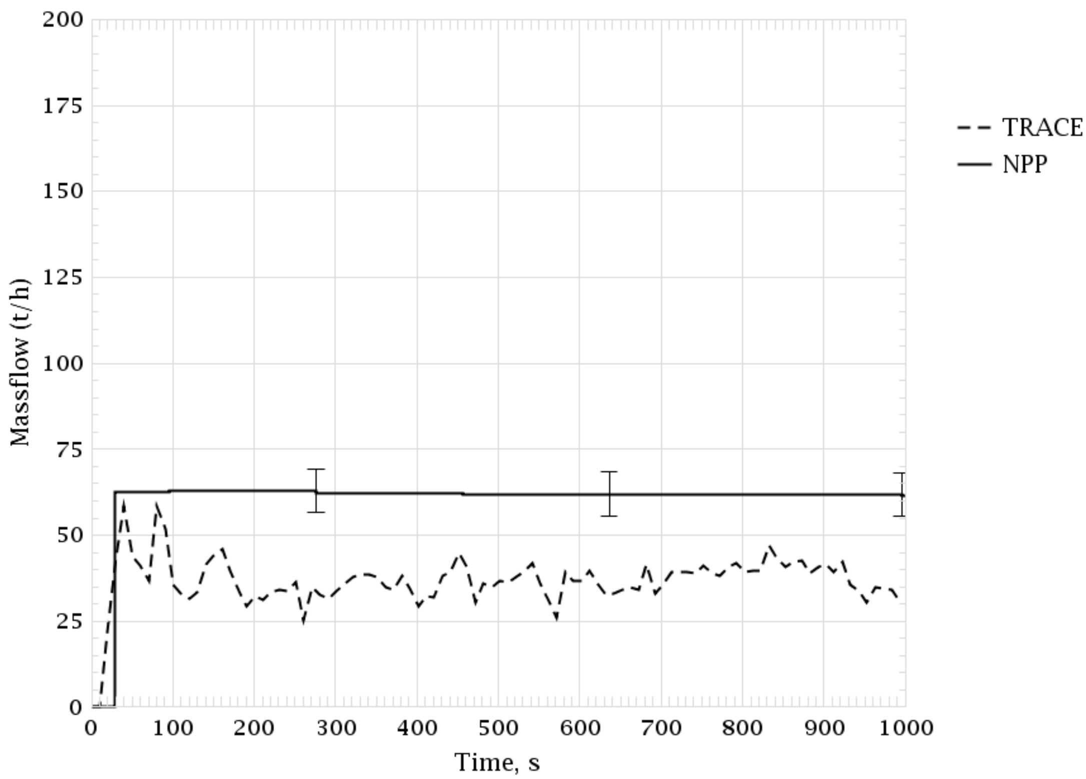

Figure 4-113 AFW Pump No.2 Flow Rate

4.3 RNPP Unit 3 PRZ PORV Stuck Open During Tests

This incident occurred at RNPP Unit 3 in 2009 during regular testing of PRZ PORV operation by RCS pressure increase before unit start-up [6]. Malfunction of the pilot valve after PRZ PORV opening resulted in continuous loss of RCS coolant with actuation of high pressure and low pressure injection pumps, and depletion of hydroaccumulators. Main operator actions during transient were to control safety injection pumps operation and to cool-down the secondary circuit.

4.3.1 Initial Conditions

PRZ PORV opening is selected as the beginning of incident (0 s of problem time). To reach the required initial state the following events are modeled at the steady state calculation:

- trip of all RCPs;
- reactor scram;
- reactor power is set to 1.67 MW [6];
- MFW trip and start of AFW pumps;
- PRZ pressure is increased up to 186.5 kgf/cm^2 by changing of pressurizer operation options and switching PRZ heaters on;
- operation of control PRZ PORV is blocked;
- "subcooling < 10°C" signal is deactivated in ECCS safeguards operation logic;

- feedwater temperature is set equal to 167.1°C according to plant data;
- MSH drains (Dn 20 mm) are modeled to simulate SG pressure decrease during the incident. The drain remains open during the transient simulation;
- PRZ level setpoint is set to 5.28 m;
- HA temperature is set according to plant data, nominal HA pressure (60 kgf/cm^2) is assumed;
- HA geometry and elevation are changed for this calculation in accordance with RNPP Unit 4 data [2].

Measured and calculated parameter values before the incident are provided in Table 4-7.

Table 4-7 Initial Conditions for Validation Transient No.3

Parameter	Units	Measured value [6]	Calculated value
Reactor thermal power	MW	1.67	1.67
Reactor outlet pressure	kgf/cm^2	186.5	186.7
Reactor inlet temperature (cold legs)	°C		
loop no.1		268.0	268.7
loop no.2		267.0	268.7
loop no.3		267.0	268.7
loop no.4		267.5	268.7
Reactor outlet temperature (hot legs)	°C		
loop no.1		271.0	272.5
loop no.2		271.0	272.5
loop no.3		271.0	272.5
loop no.4		271.0	272.5
Average coolant heat-up in the reactor core	°C	3.6	3.8
PRZ level (YP10L06)	m	5.28	5.26
SG pressure	kgf/cm^2		
SG-1		56.2	55.8
SG-2		56.0	55.8
SG-3		56.0	55.8
SG-4		56.1	55.8
SG level (narrow range measurement)	m	–	
SG-1			0.298
SG-2			0.321
SG-3			0.318
SG-4			0.305

Parameter	Units	Measured value [6]	Calculated value
SG level (wide range measurement) SG-1 SG-2 SG-3 SG-4	m	 2.47 2.51 2.63 2.53	 2.54 2.56 2.56 2.55
Main feedwater temperature	°C	167.1	167.1
SG feed water flow rate SG-1 SG-2 SG-3 SG-4	t/h	 1427 1463 1502 1464	 1461 1464 1463 1468
Make-up flow rate	m^3/h	28.1	28.4
Let-down flow rate	m^3/h	18.1	18.1
Make-up temperature	°C	170.0	176.0
HA temperature	°C	61.2 57.4 51.6 62.6	61.2 57.4 51.6 62.6

4.3.2 Boundary Conditions

The boundary conditions applied in validation calculation are described below.

1. Opening of PRZ PORV is modeled at 0 s of calculation. PRZ PORV steam discharge factor is adjusted in accordance with plant data. For two-phase steam-water mixture and superheated steam the discharge factor of 0.81 is used.
2. Relief tank pressure as a function of time is defined based on plant measured data.
3. PRZ heaters are switched on for steady state calculation. In transient calculation PRZ heaters operate according to design logic. Interlock for blocking the PRZ heaters operation after restoration of PRZ level above 4.2 m is implemented in the model.
4. Make-up control valves position is fixed after beginning of transient. Let-down flow is constant up to closure of let-down valves at 120 s.
5. BRU-K-1 stem position is defined as a function of time according to plant measured data. Cross sectional area of BRU-K depending on stem position is adjusted to match secondary circuit pressure measured data.
6. BRU-A-4 stem position is defined as a function of time according to plant measured data. Cross sectional area of BRU-A-4 depending on stem position is adjusted to match SG-4 pressure measured data.
7. High head safety injection pumps TQ14,24,34D01 are switched on/off at 385/2115 s.
8. HPIS pumps flow characteristics as the functions of RCS pressure are defined according to data measured during the incident. Initial HPIS tanks temperature is set to 31.3, 30.7 and

31.4°C for TQ13, 23, 33 trains, respectively. Correspondent temperatures during operation from containment sump are 35.0, 43.3 and 29.2°C. Mass injected by pumps from HPIS tanks is estimated based on tank dimensions from [2] and is equal to 8 m^3, 9.2 m^3 and 8.7 m^3 for TQ13, 23, 33 trains, respectively.

9. LPIS pumps characteristics of base VVER-1000/V-320 model for TRACE code were corrected in the range of 21 – 24 kgf/cm^2 in accordance with measured data. Pump head values were adjusted to account hydrostatic pressure losses due to the differences in elevation of pump outlet and RCS connection points. LPIS tanks (sump) temperatures are selected based on incident measured data.
10. Start-up/shutdown feedwater control valves stem positions are defined as a function of time to simulate operator actions on SG level increase up to 3.8 m.
11. High containment pressure (> 0.3 kgf/cm^2 gauge) ECCS signal is actuated at 2068 s according to incident data.

4.3.3 Calculation Results

Table 4-8 provides comparison of calculated and actual timing of events occurred in the course of the incident.

Table 4-8 Sequence of Events for Validation Transient No.3

Time, sec		Event
Incident	Calculation	
0	0	Start of incident, opening of PRZ PORV
62-65	60	All PRZ heaters groups are on (decrease of RCS pressure down to 158.5-153.0 kgf/cm^2)
120	120	Closure of RCS let-down
135	125	All PRZ heaters groups are switched off due to decrease of PRZ level below 4.2 m
300	315	Minimal subcooling is less than 10°C
315	330	Opening of HA check valves, start of HA injection (RCS pressure is less than 60 kgf/cm^2)
385	385	Start of TQ14-34D01 injection. HPIS pumps are aligned for operation through the minimal bypass flow line
845	845	Start of TQ13D01 injection, operator action
915	915	Start of SG filling in
971	980	Start of TQ13D01 operation from containment sump (TQ10B01)
1045	1045	Termination of TQ13D01 injection
1205	1205	Start of TQ23D01 injection, operator action
1347	1360	Start of TQ23D01 operation from containment sump (TQ20B01)
1669	1669	Start of BRU-K-1 opening
1765	1765	Start of TQ33D01 injection, operator action
1899	1910	Start of TQ33D01 operation from containment sump (TQ30B01)

Time, sec		Event
Incident	Calculation	
2068	2068	Containment pressure increase above 0.3 kgf/cm^2 gauge. Containment isolation. Start of LPIS operation via minimal bypass flow line, start of containment spray pumps injection, termination of primary circuit make-up
2085	2085	Start of TQ13D01 injection (ECCS safeguards)
2115	2115	Termination of TQ14-34D01 injection (operator action)
2160	2160	Termination of TQ33D01 injection (operator action)
2440	2440	Termination of TQ23D01 injection (operator action)
3525	3525	Termination of TQ13D01 injection (operator action)
4485	3900	Start of LPIS pumps TQ12,22,32D01 injection
4019	4019	BRU-K-1 closure
4505	4505	BRU-K-1 opening
4935	4935	Termination of TQ22D01 injection (operator action)
5055	5055	Termination of TQ32D01 injection (operator action)
5475	5800	SG levels are 3.8 m
6375	-	Hot legs temperature decrease down to 70°C
–	7200	End of calculation. Minimal reactor inlet temperature is 40°C, reactor outlet temperature is 63°C, RCS pressure is 23 kgf/cm^2

After PRZ PORV opening the initial steam flow is ~50 kgf/cm^2 (Figure 4-156). Loss of the primary coolant causes RCS pressure decrease (Figure 4-114) and start of PRZ heaters operation at 60 s. After decrease of PRZ level (Figure 4-115) below 4.2 m at 125 s all PRZ heaters groups are switched off with prohibition for operation.

Operator isolates let-down at 120 s (Figure 4-129) and at 195 s establishes boron supply from TB10 tanks to the suction line of make-up pumps. Since let-down is terminated, the make-up temperature reaches 62°C at 300 s and continues to decrease.

Since RCS temperature decreases faster than the secondary circuit temperature, starting from 400 s SGs start to heat the primary circuit. At 315 s of calculation (300 s of actual incident) the ECCS setpoint "Subcooling < 10°C" is reached (Figure 4-127) that has to result in containment isolation and start of ECCS pumps. However this setpoint is deactivated in accordance with PRZ PORV testing program.

At 330 s the primary circuit pressure decreases below 60 kgf/cm^2 and HAs start to inject boric acid solution to the primary circuit. At 385 s the high head safety injection pumps TQ14-34D01 operation to the primary circuit starts while HPIS pumps TQ13-33D01 operate through the minimal bypass flow lines.

At ~400 s the saturation temperature is reached in the primary circuit (Figure 4-127) that leads to SG collector tops voiding and formation of the upper head bubble. The latter results in significant increase of PRZ level (Figure 4-115) and starting from 480 s two-phase mixture discharge from PORV begins. RCS pressure is stabilized at ~53 kgf/cm^2 (Figure 4-114).

At 845 s operator realigns TQ13D01 for RCS injection (Figure 4-137). Filling in of SGs is initiated at 915 s (simulated by changing of feed water flow controllers setpoints). At 1045 s operator terminates TQ13D01 injection to the primary circuit and arranges injection from TQ23D01 pump at 1205 s (Figure 4-138).

To speed-up secondary circuit pressure decrease operator opens BRU-K-1 at 1669 s (Figure 4-155). However this does not affect RCS parameters since the core exit temperature at this moment is ~196°C (Figure 4-116) which corresponds to saturation pressure of ~14.5 kgf/cm^2 abs. that is lower than the secondary circuit pressure (Figures 4-144 – 4-148).

At 1765 s operator initiates injection to the primary circuit from TQ33D01 and at 1950 s RCS pressure starts to increase. Starting from 2000 s the discharge of subcooled water from PRZ PORV begins and continues till the end of incident.

ECCS safeguard "Containment pressure > 0.3 kgf/cm^2 gauge" is actuated at 2068 s resulting in containment isolation, termination of primary circuit make-up, start of LPIS operation via minimal bypass flow lines and start of containment spray pumps injection. This safeguard also causes restart of TQ13D01 pump in addition to operating TQ23,33D01 pumps and RCS pressure increases up to 74 kgf/cm^2 (Figure 4-114).

After that operator sequentially trips HPIS pumps TQ33D01 (at 2160 s), TQ23D01 (at 2440 s) and TQ13D01 (at 3525 s). Calculated core exit temperature after trip of the last operating HPIS pump is 105°C, while measured value is ~90°C (Figure 4-116).

In calculation the cold leg coolant temperatures (by thermocouples model) decrease to the measured values at ~3000, 4800, 3000 and 4300 s for loops no.1-4 (Figures 4-123 – 4-126). The difference between calculated and measured values is explained by one-dimensional representation of the cold legs with averaging of parameters in hydraulic components, while in the actual incident the thermal stratification is observed.

Due to continuous loss of coolant at terminated HPIS injection RCS pressure decreases to LPIS shut-off head (at 3900 s in calculation and at 4485 s in the incident) and LPIS operation stabilizes RCS pressure at 22-23 kgf/cm^2 (Figure 4-114). It shall be noted that measured LPIS flow in Figures 138 – 140 indicates the flow upstream the minimal bypass flow line (i.e. includes flow through bypass line and flow to the primary circuit). Trip of LPIS pumps TQ22D01 and TQ32D01 at 4935 s and 5055 s, respectively, does not affect significantly RCS pressure or core exit temperature, that continues to decrease slowly (Figure 4-116). Starting from 6200 s the calculated loss of coolant via PRZ PORV is compensated by LPIS operation (Figure 4-156).

Since BRU-K remains open (Figure 4-155) SG pressure reaches ~25 kgf/cm^2 at 7200 s.

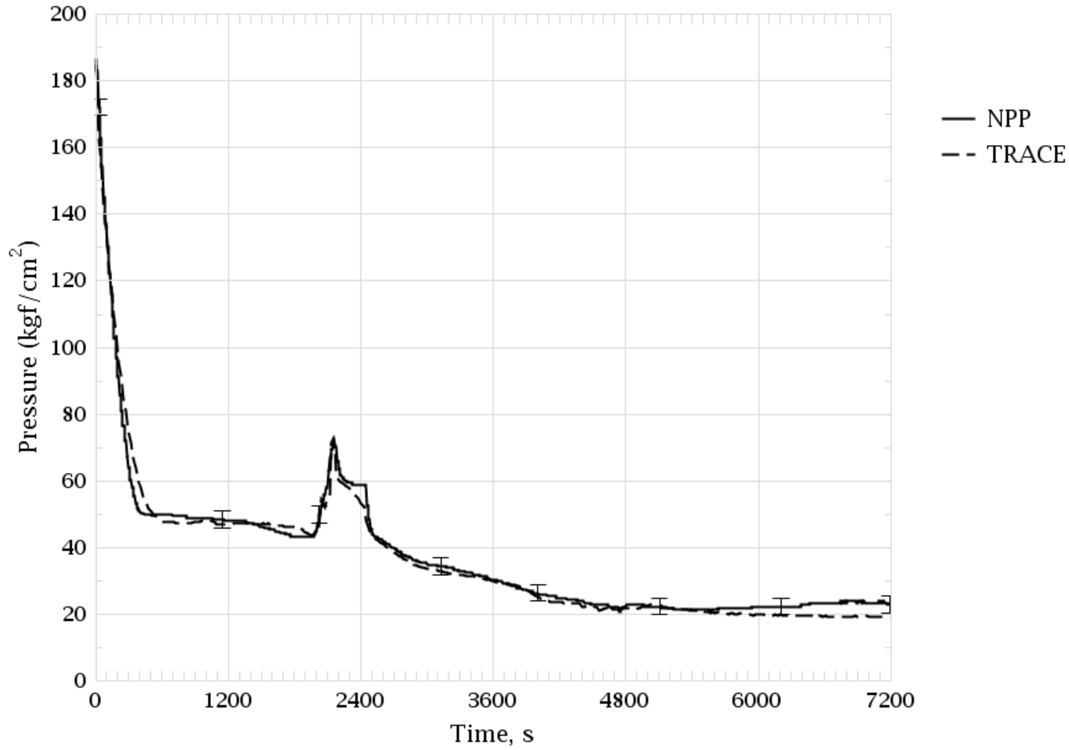

Figure 4-114 RCS Pressure

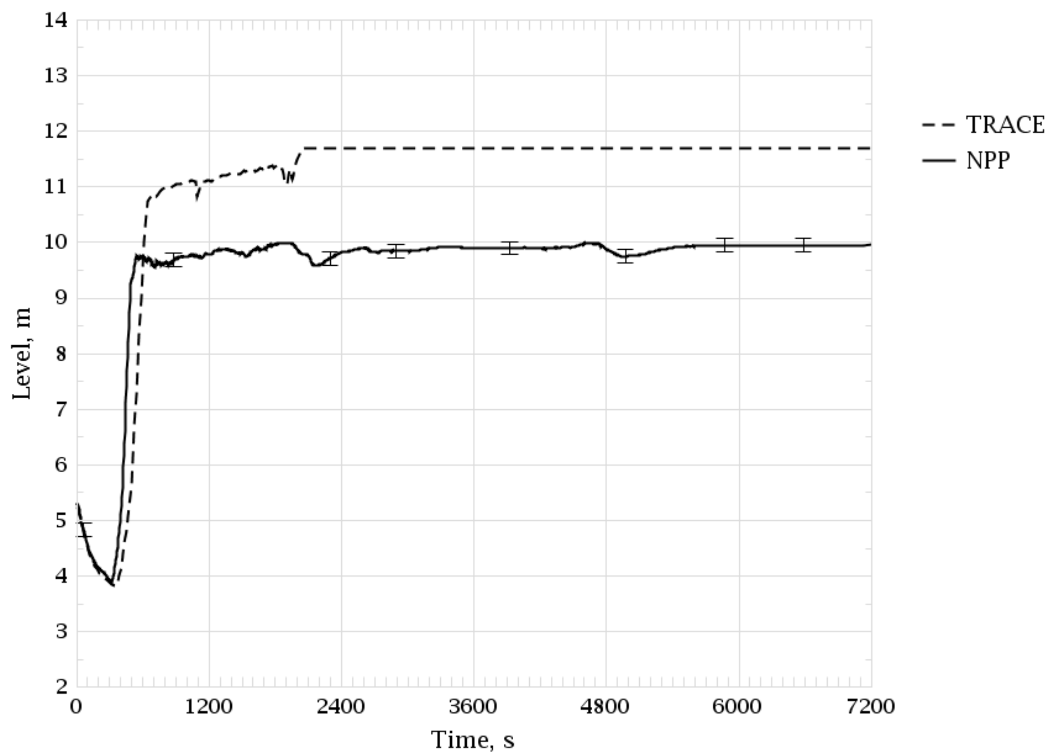

Figure 4-115 PRZ Level

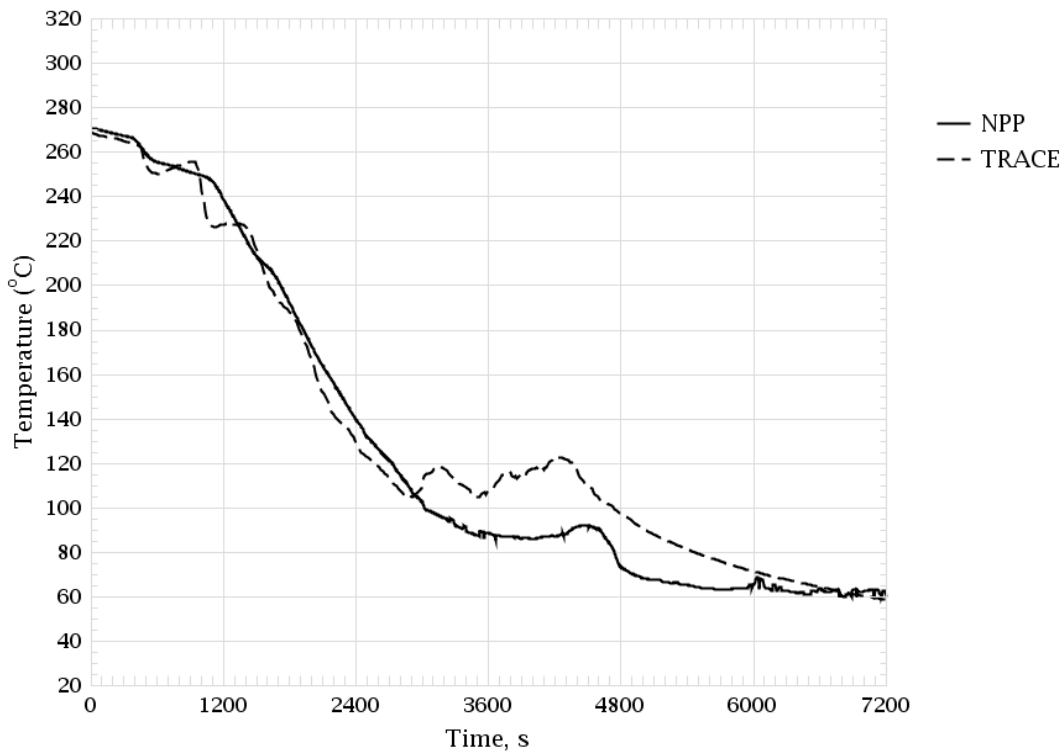

Figure 4-116 Core Exit Temperature

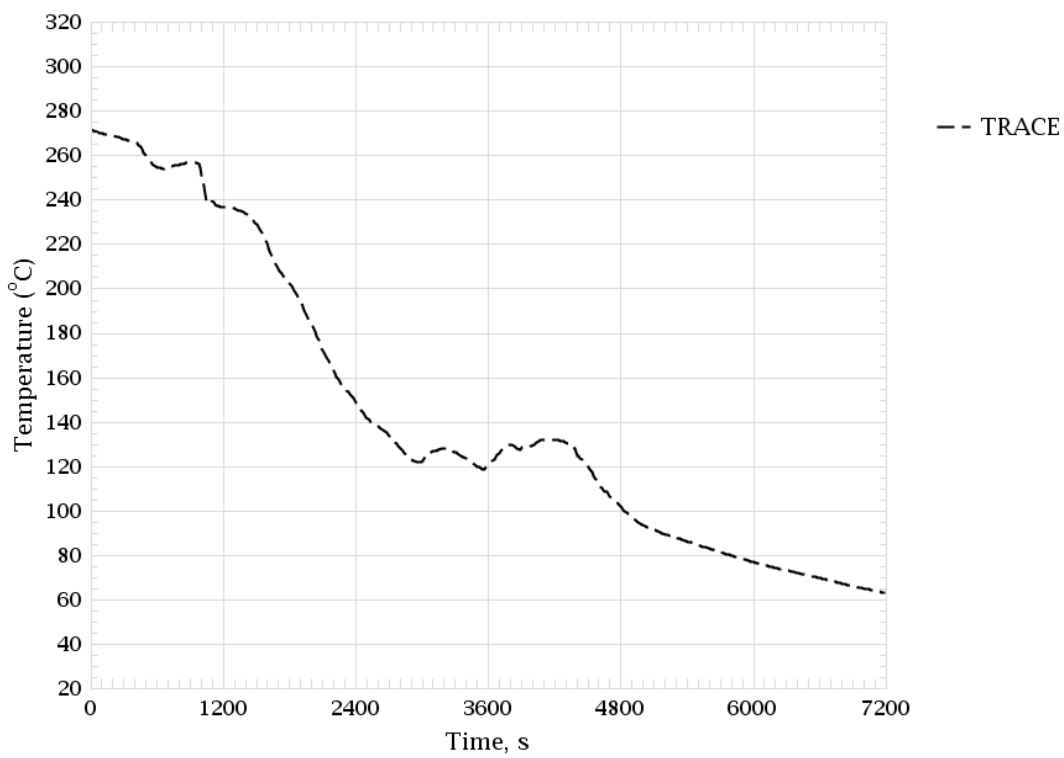

Figure 4-117 Peak Cladding Temperature

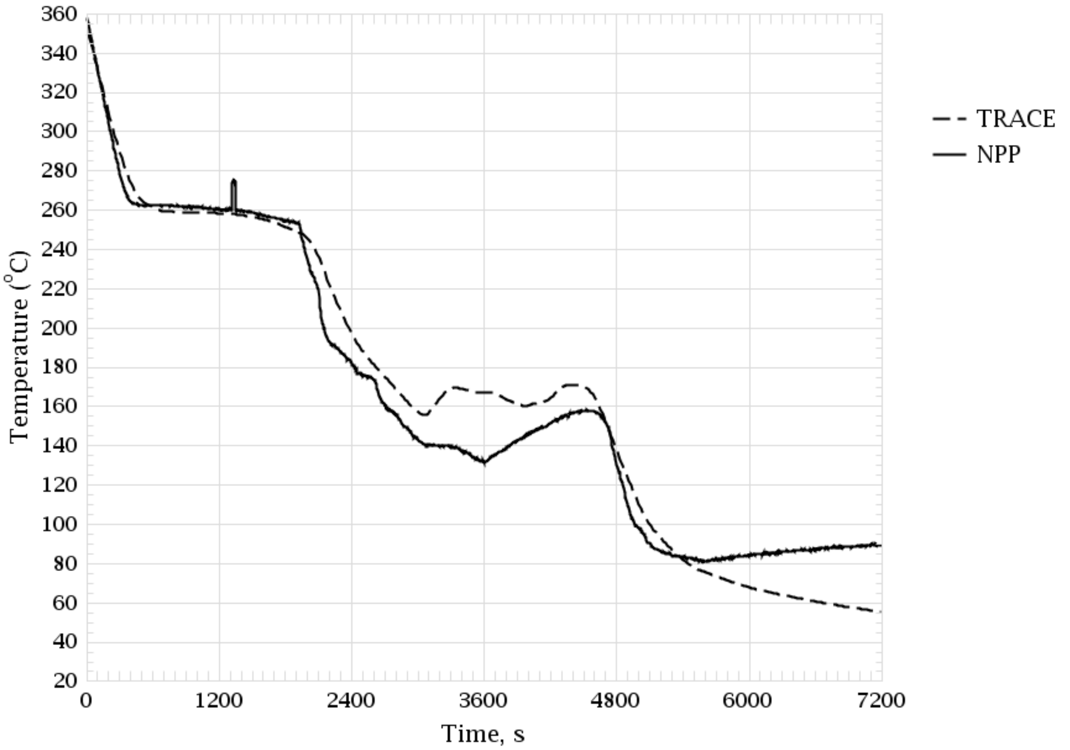

Figure 4-118 PRZ Coolant Temperature

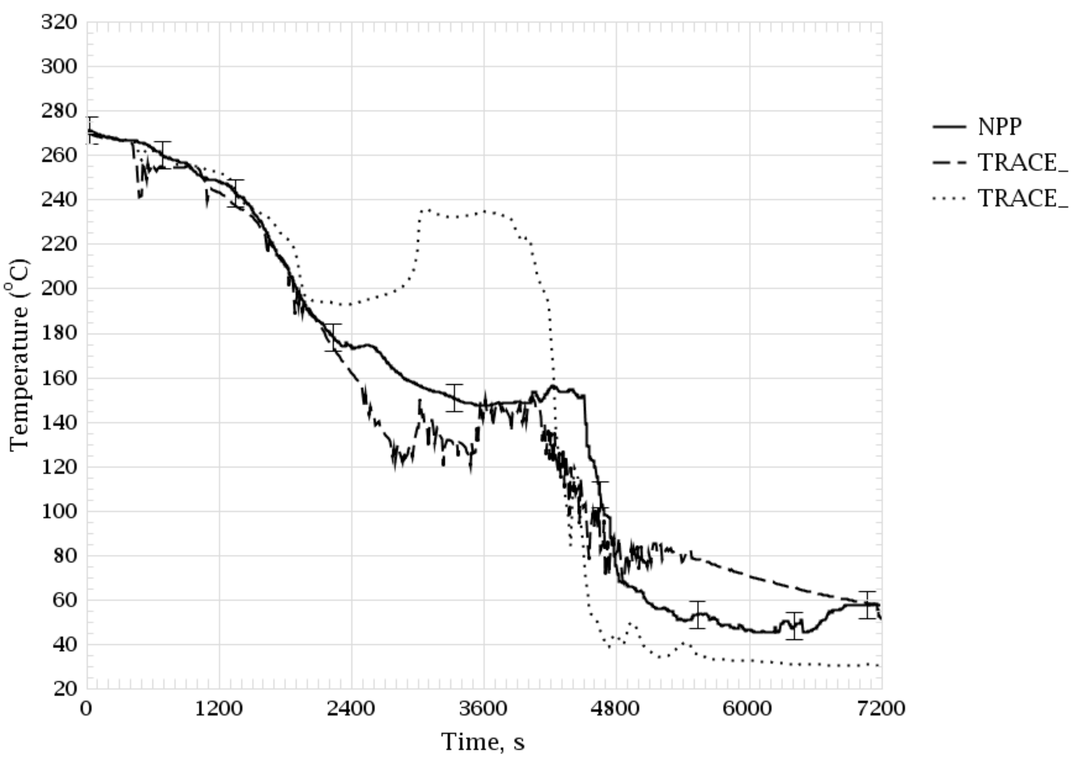

Figure 4-119 Coolant Temperature in Hot Leg No.1 and at the Core Exit

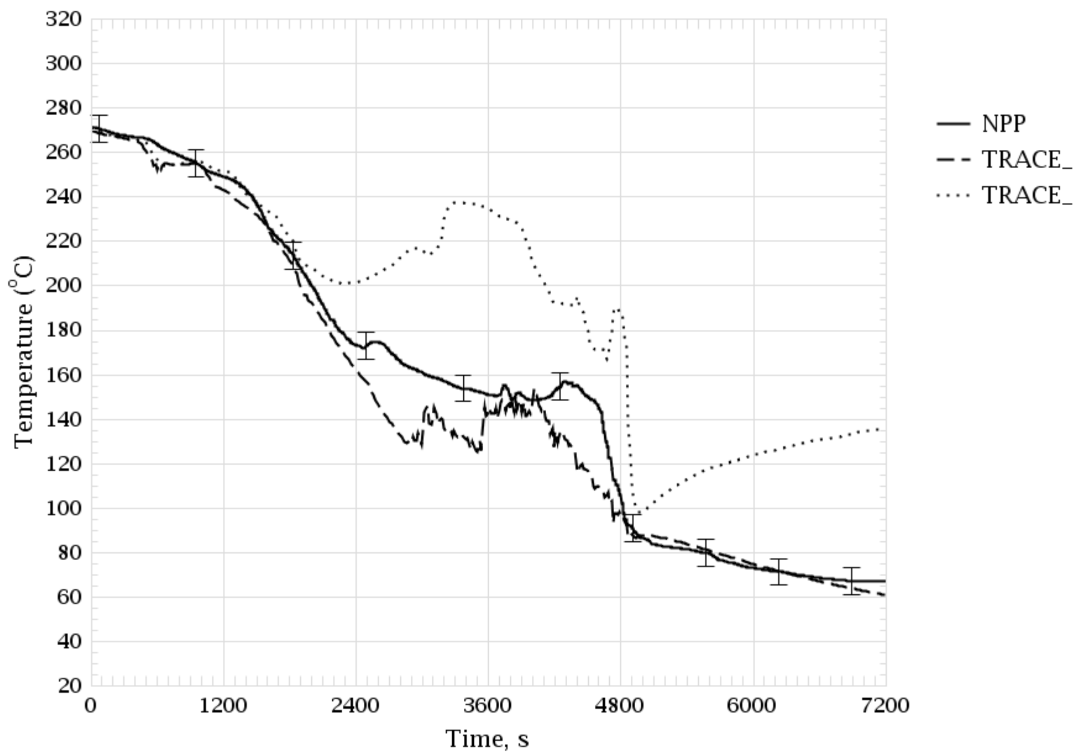

Figure 4-120 Coolant Temperature in Hot Leg No.2 and at the Core Exit

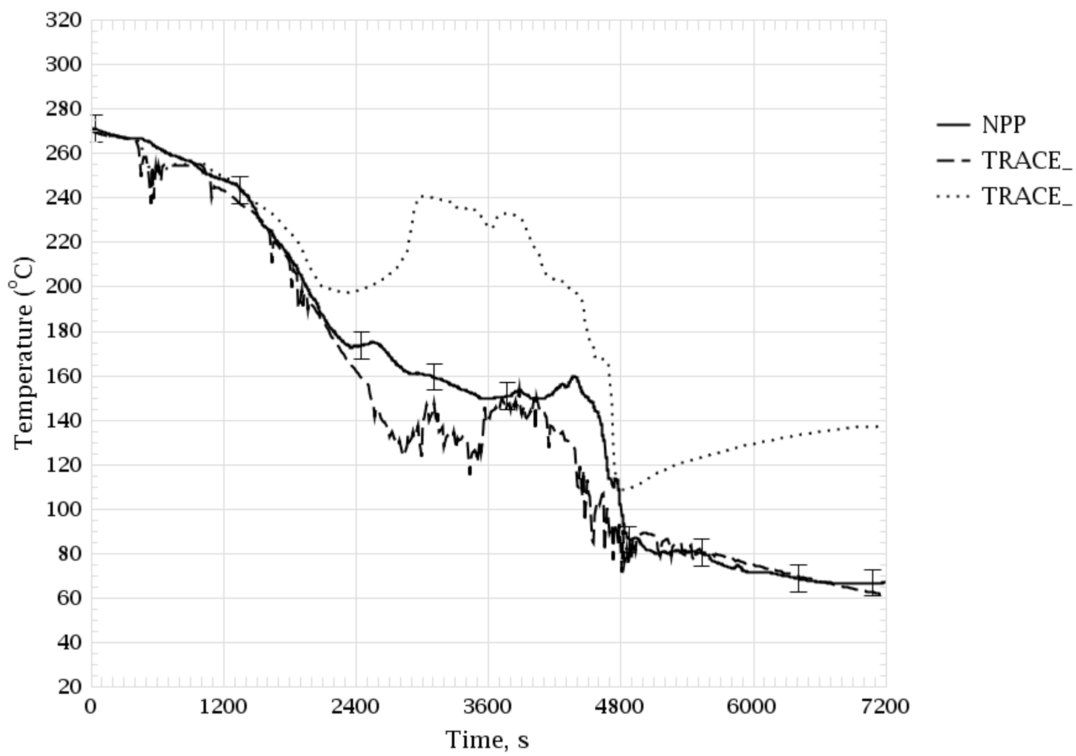

Figure 4-121 Coolant Temperature in Hot Leg No.3 and at the Core Exit

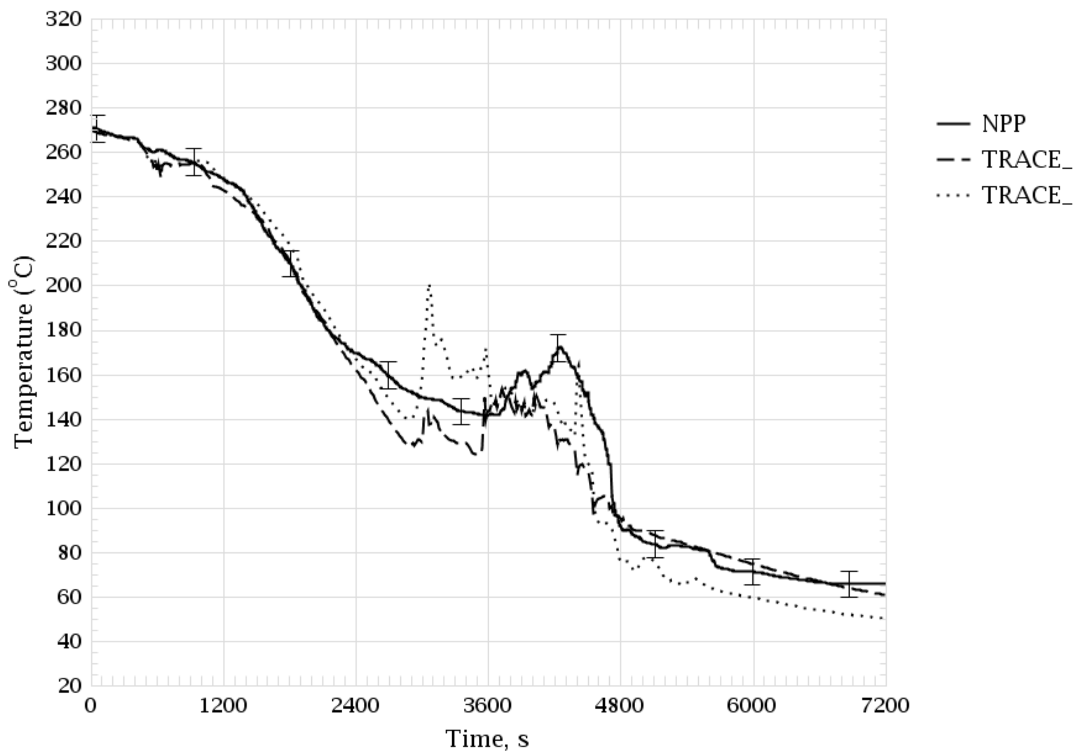

Figure 4-122 Coolant Temperature in Hot Leg No.4 and at the Core Exit

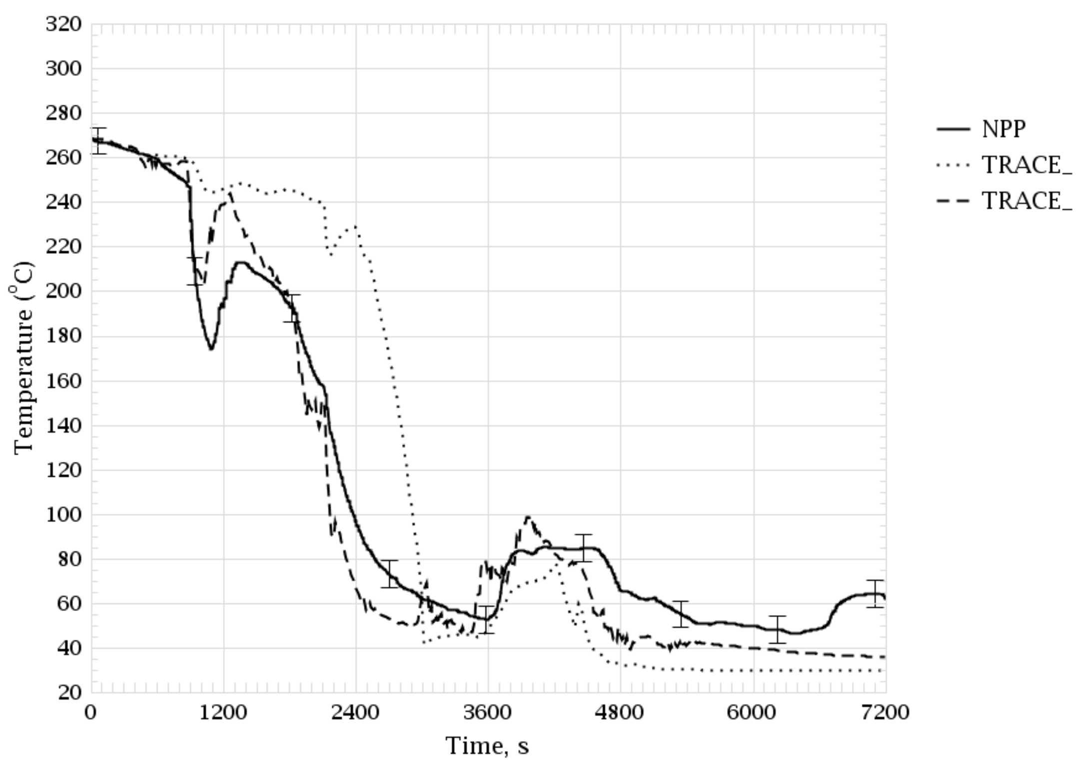

Figure 4-123 Coolant Temperature in Cold Leg No.1 and at the Reactor Inlet

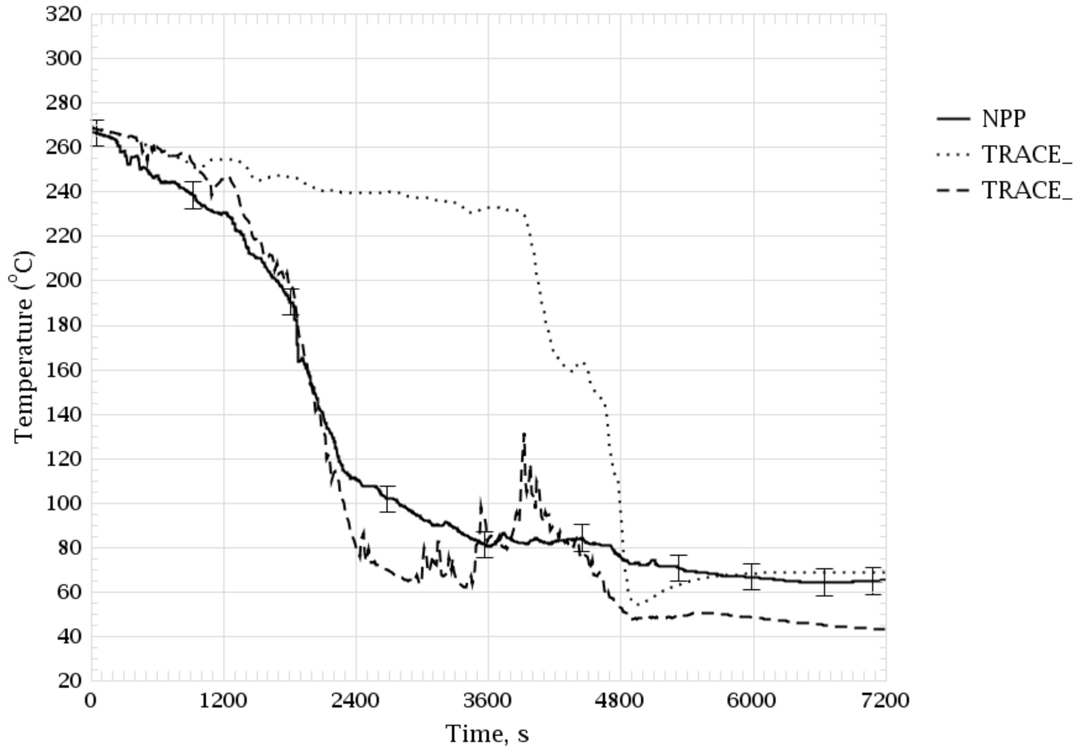

Figure 4-124 Coolant Temperature in Cold Leg No.2 and at the Reactor Inlet

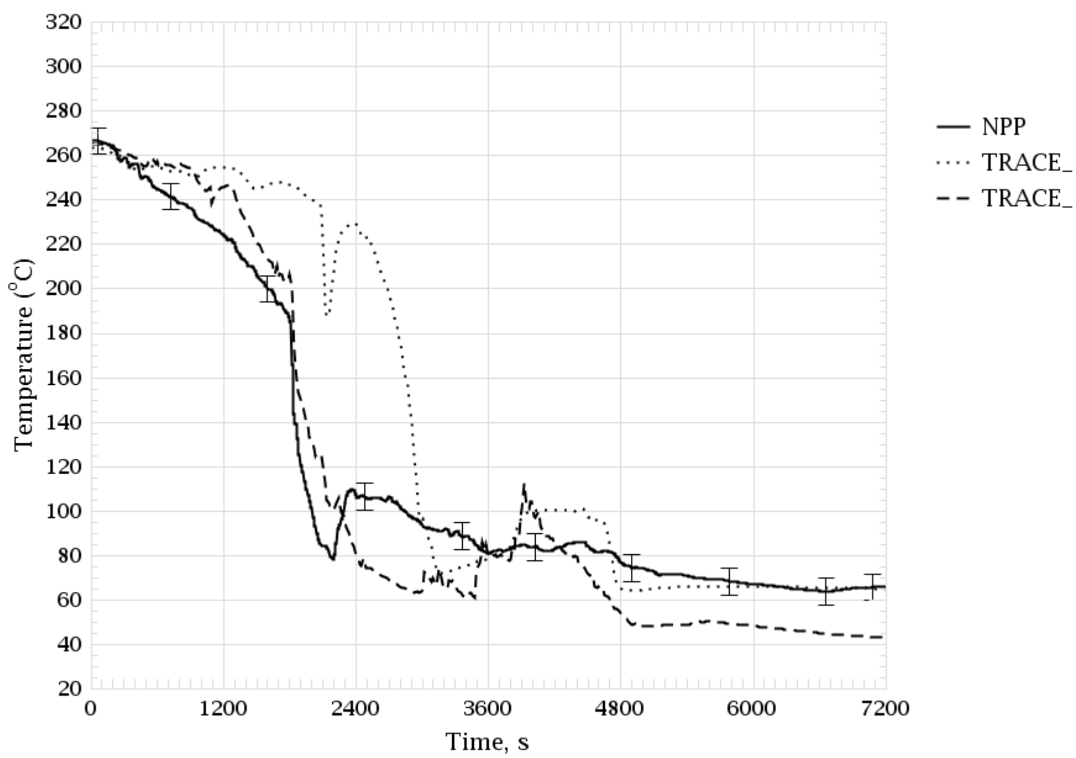

Figure 4-125 Coolant Temperature in Cold Leg No.3 and at the Reactor Inlet

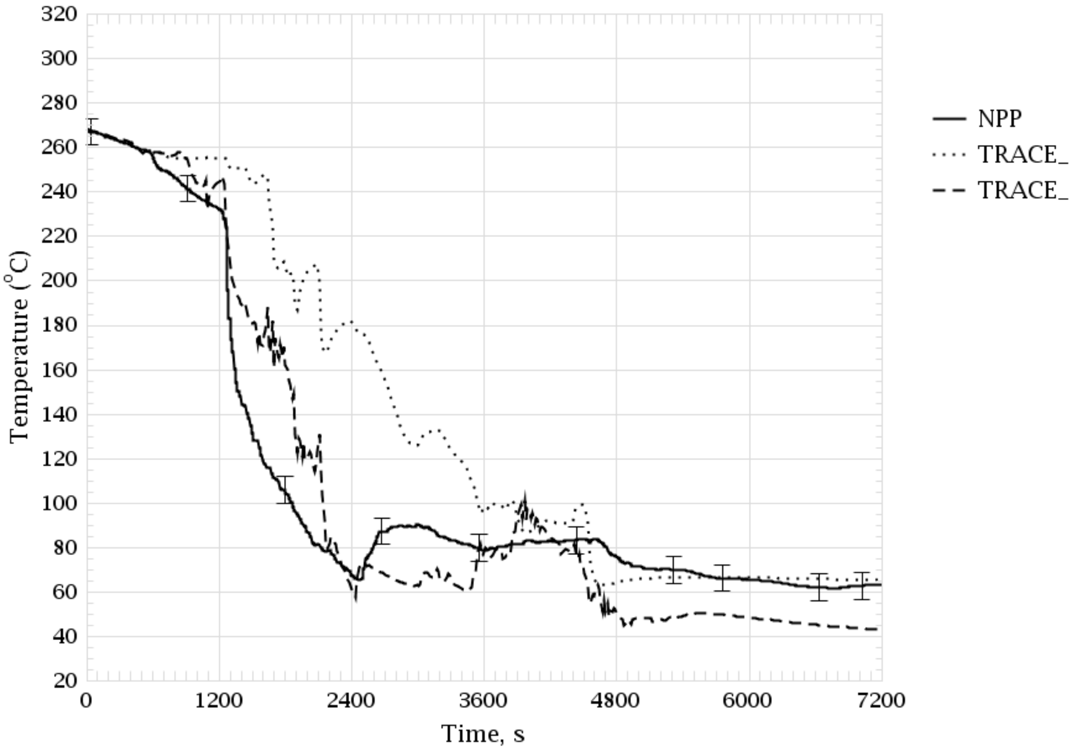

Figure 4-126 Coolant Temperature in Cold Leg No.4 and at the Reactor Inlet

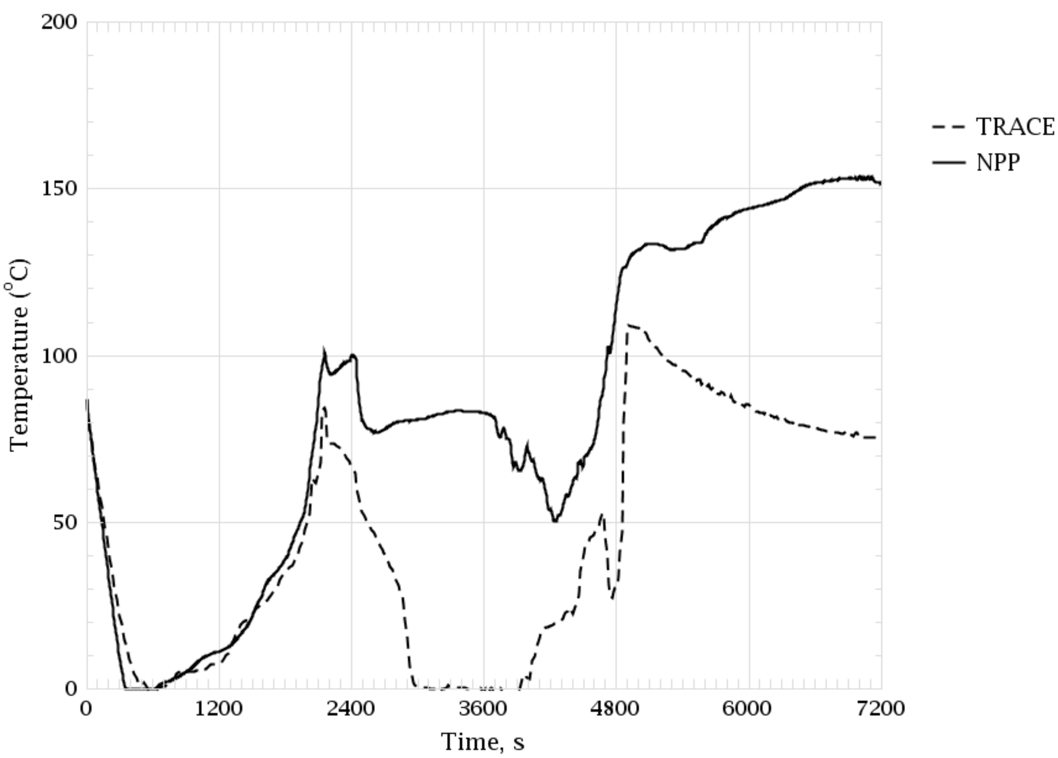

Figure 4-127 Subcooling

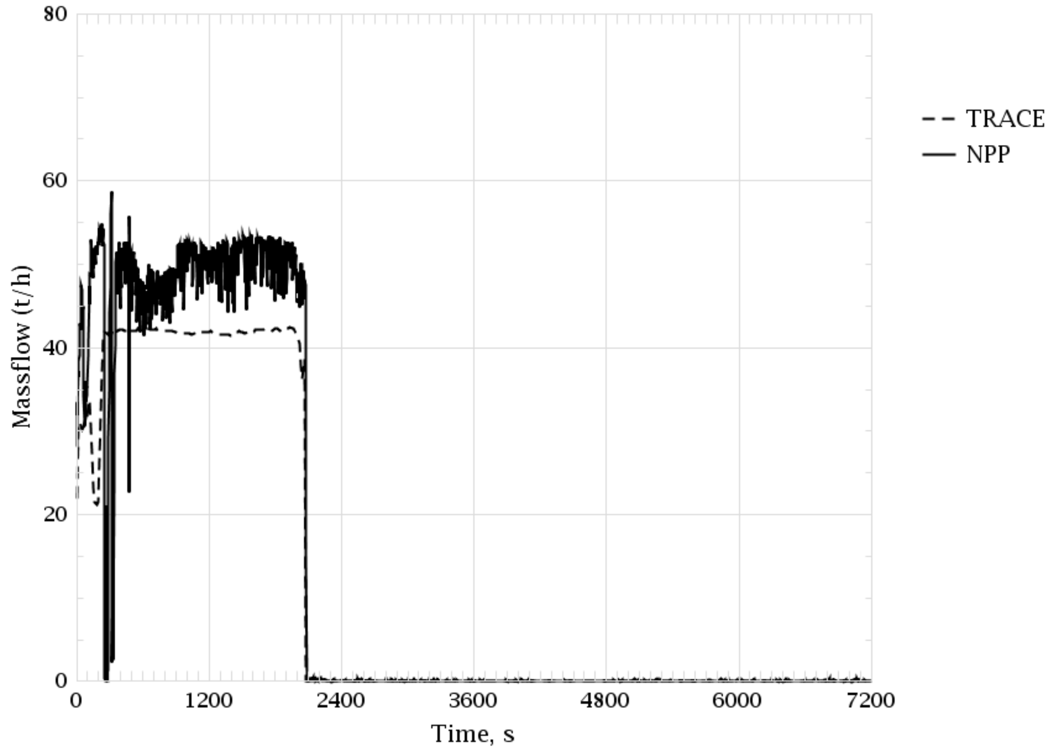

Figure 4-128 Make-up Mass Flow Rate

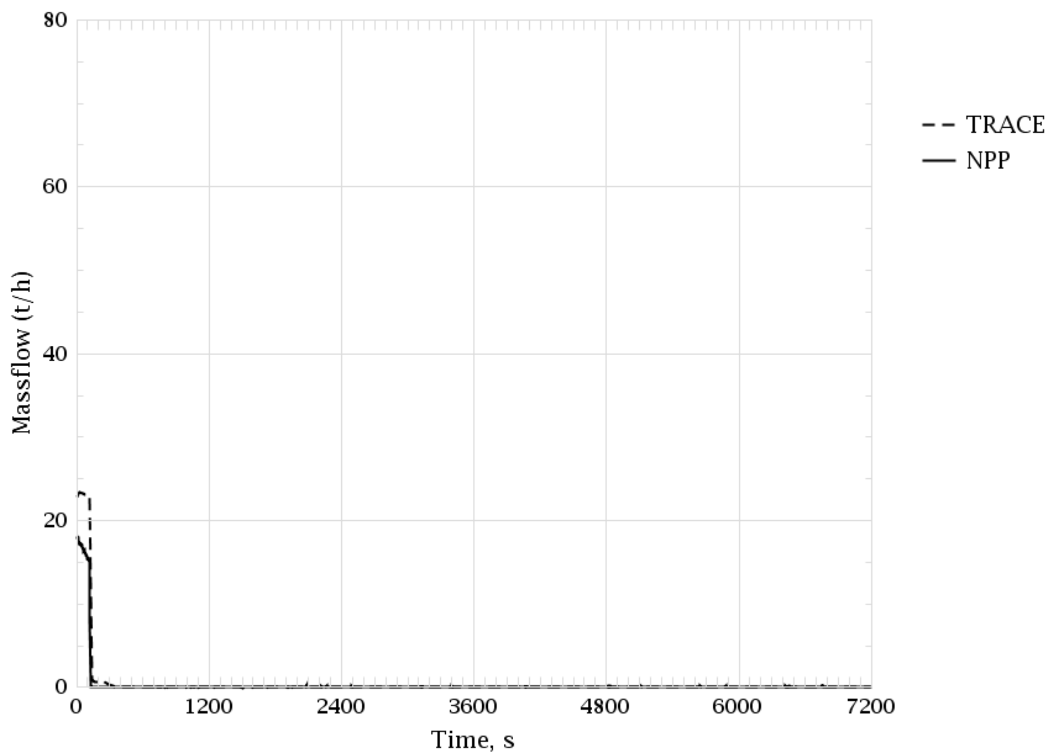

Figure 4-129 Let-down Mass Flow Rate

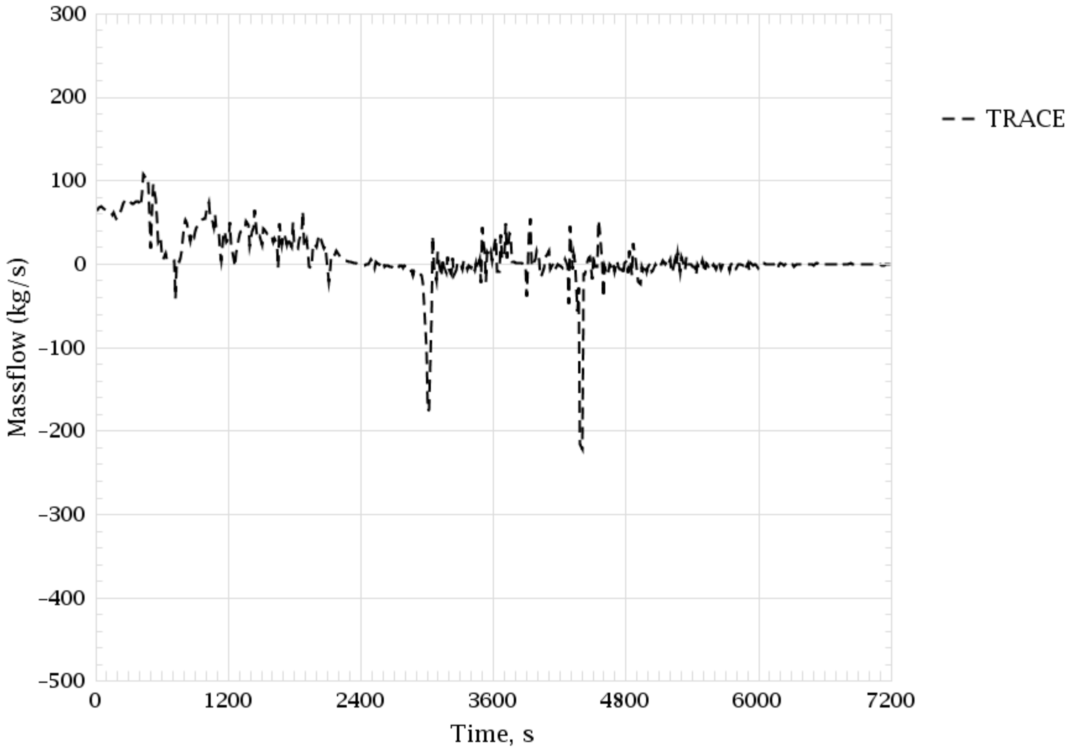

Figure 4-130 RCP Mass Flow Rate, Loop No.1

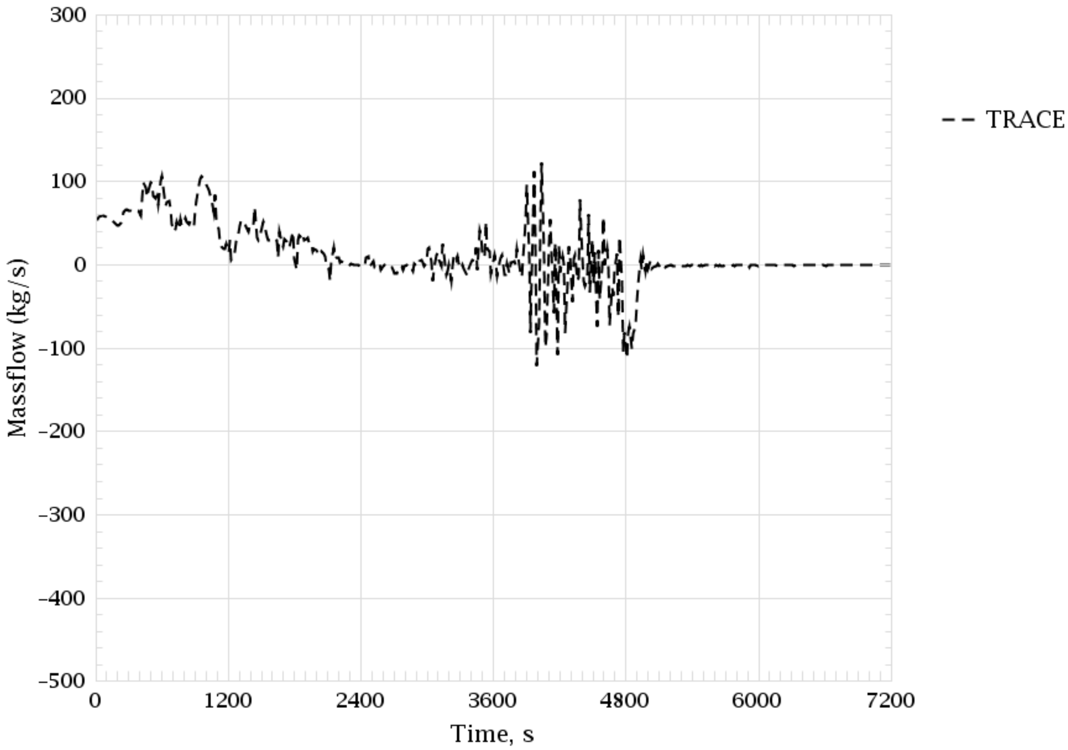

Figure 4-131 RCP Mass Flow Rate, Loop No.2

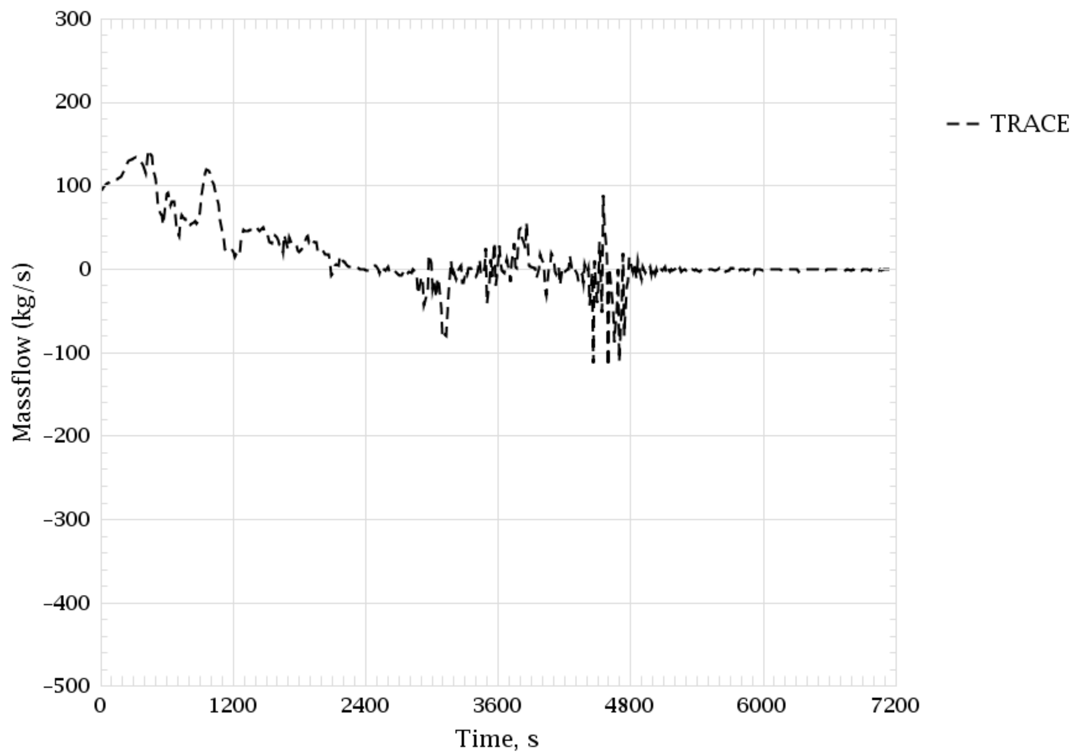

Figure 4-132 RCP Mass Flow Rate, Loop No.3

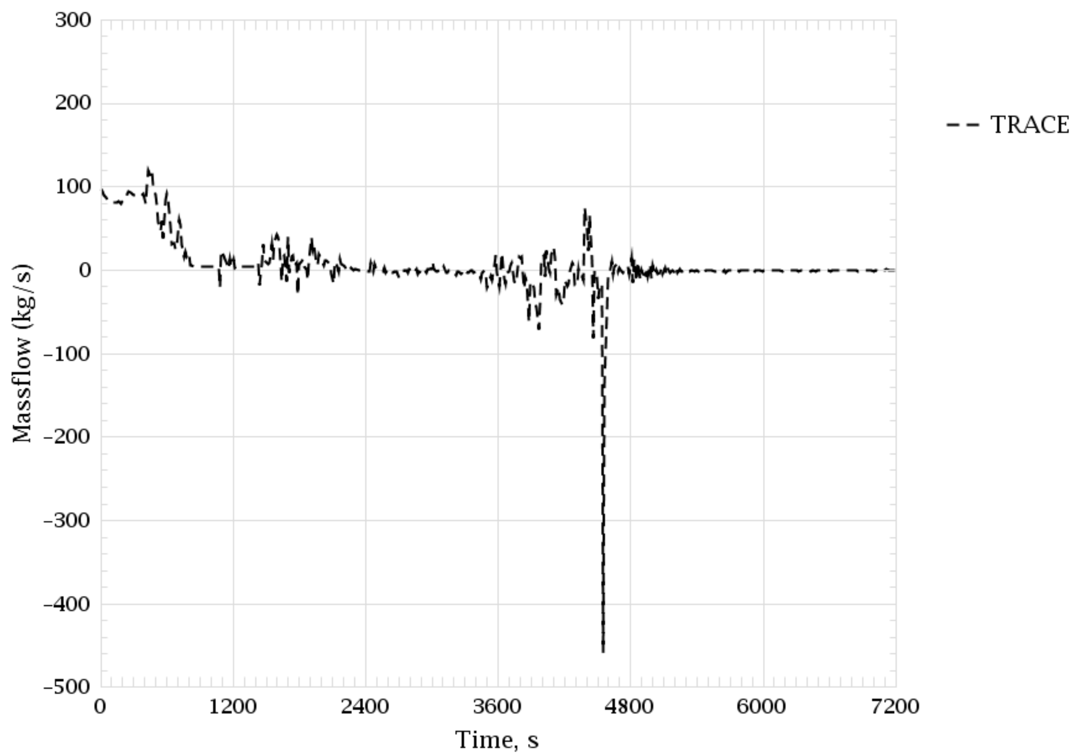

Figure 4-133 RCP Mass Flow Rate, Loop No.4

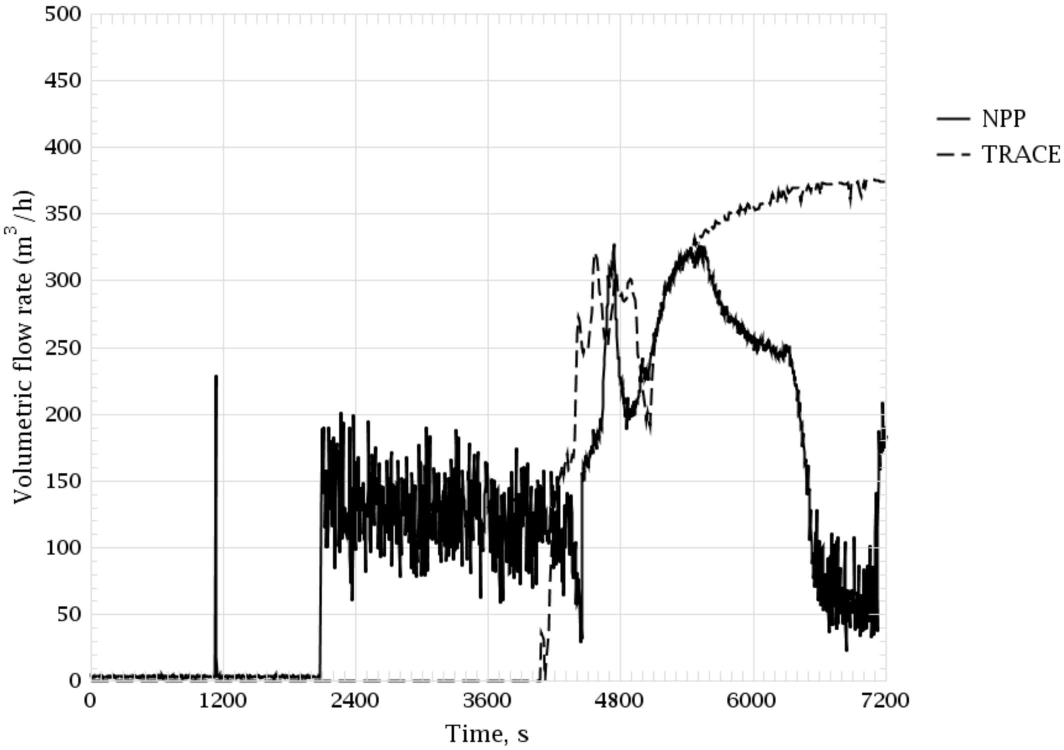

Figure 4-134 TQ12 LPIS Mass Flow Rate

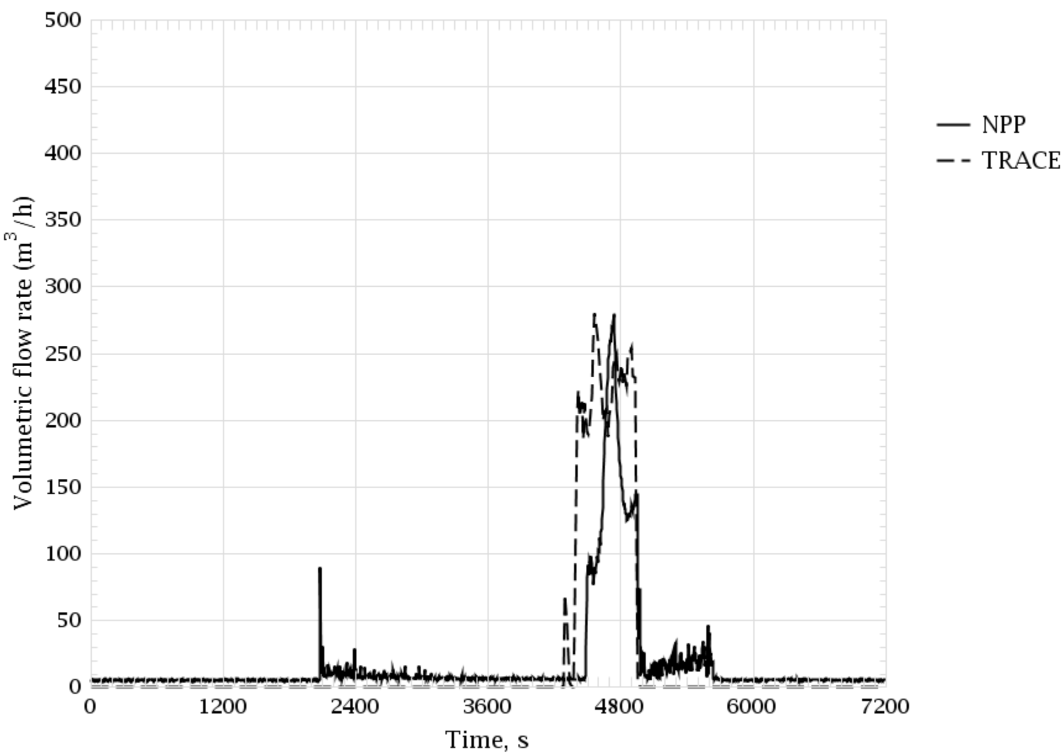

Figure 4-135 TQ22 LPIS Mass Flow Rate

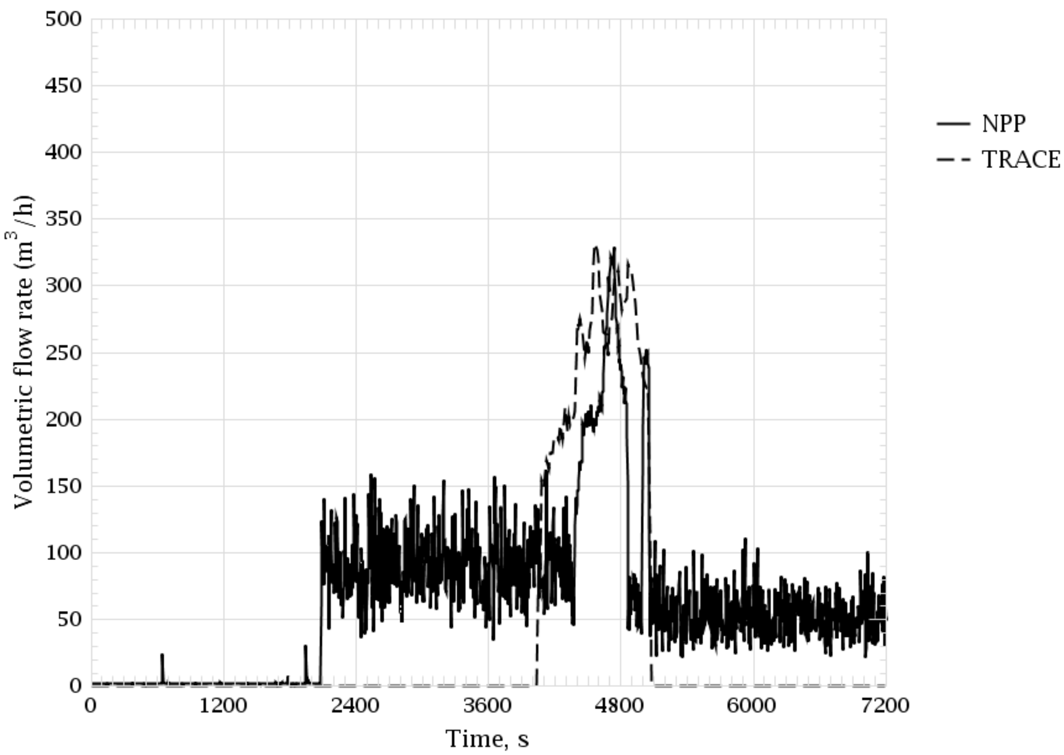

Figure 4-136 TQ32 LPIS Mass Flow Rate

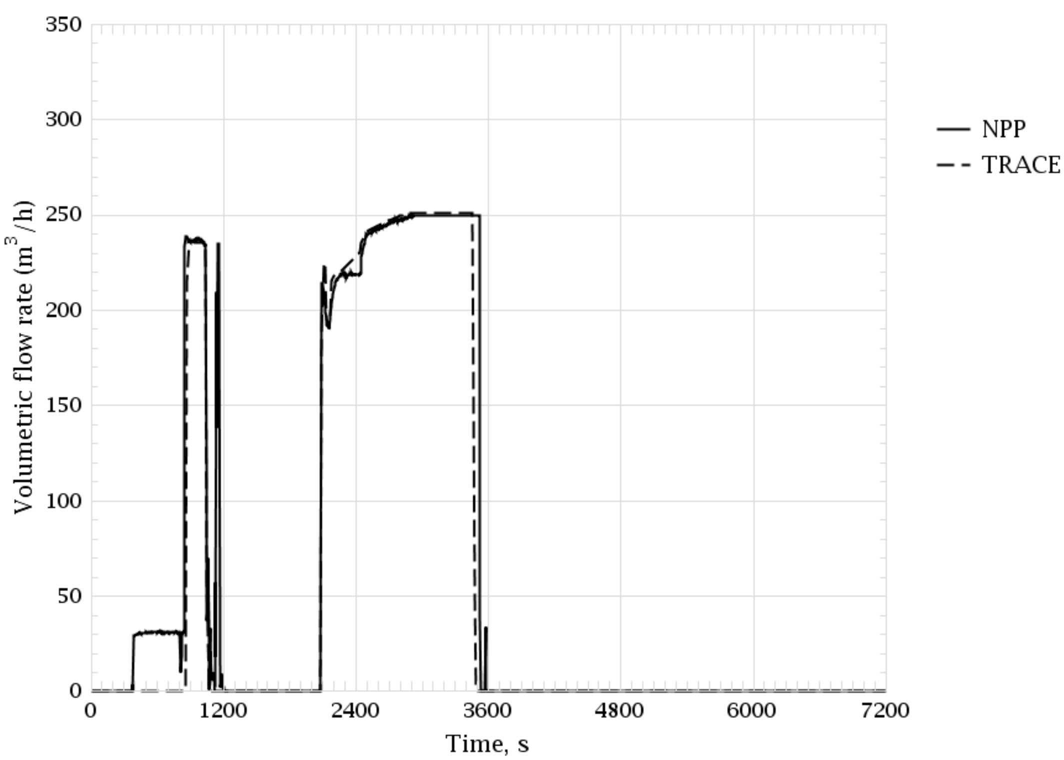

Figure 4-137 TQ13 HPIS Mass Flow Rate

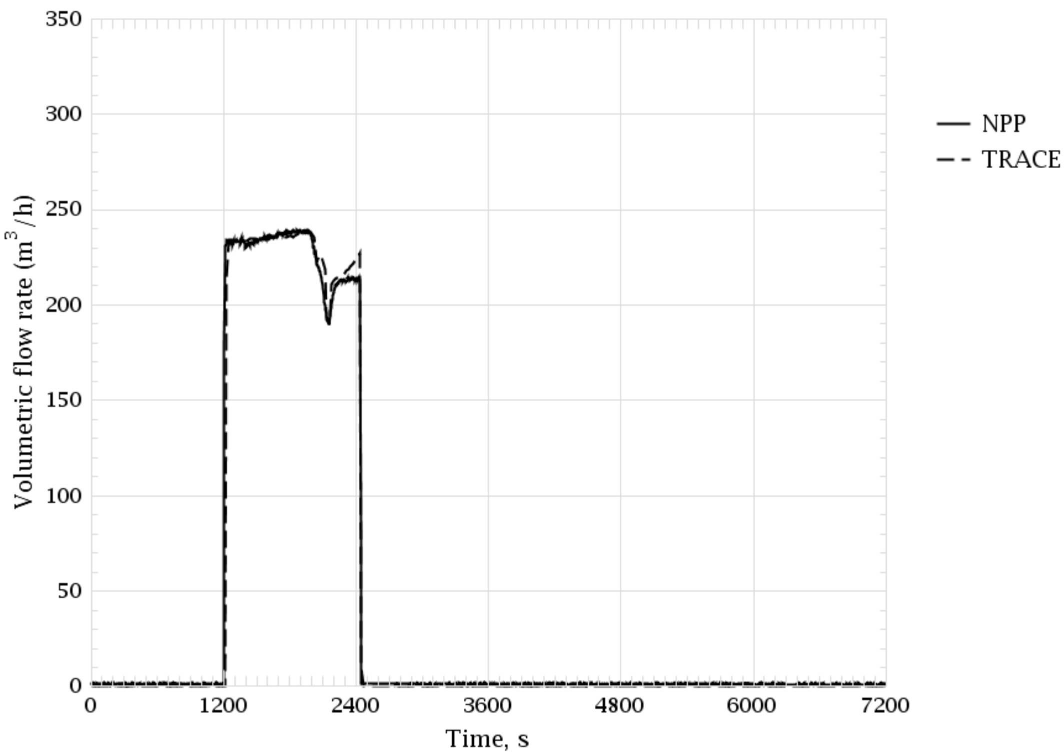

Figure 4-138 TQ23 HPIS Mass Flow Rate

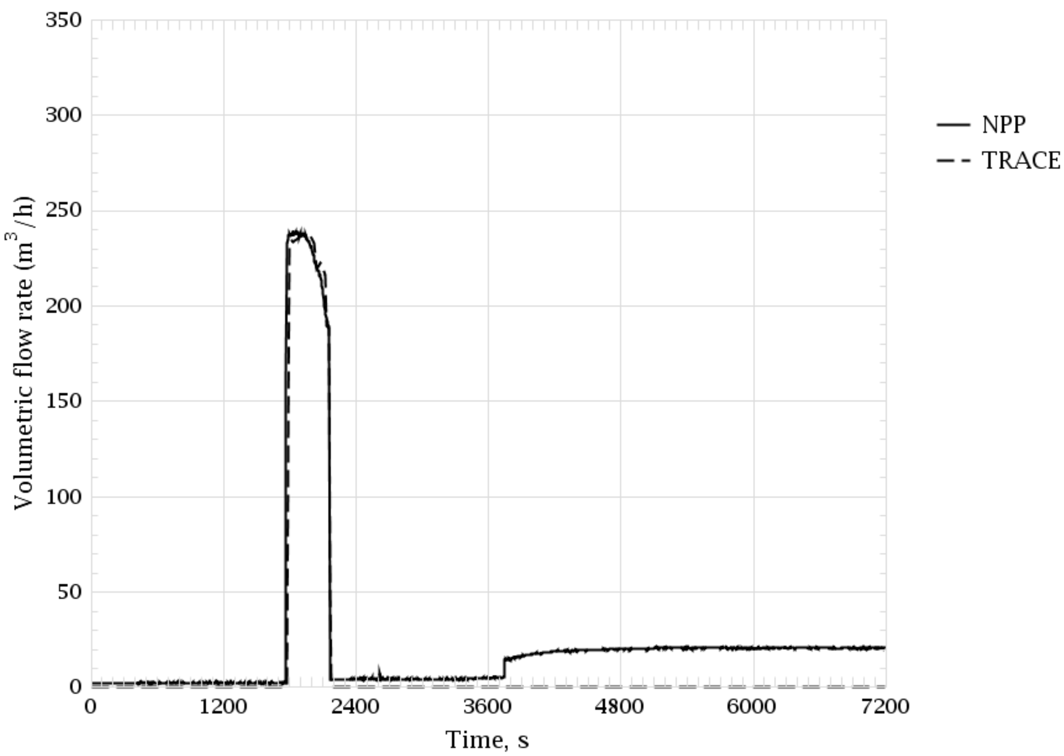

Figure 4-139 TQ33 HPIS Mass Flow Rate

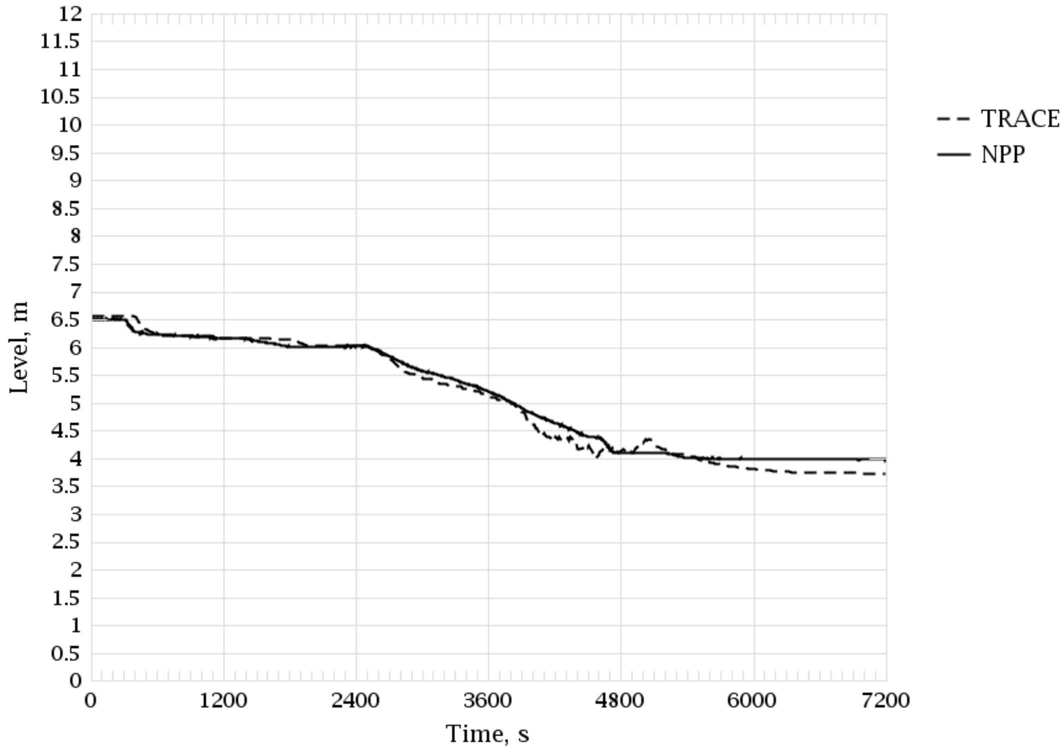

Figure 4-140 HA-1 Level

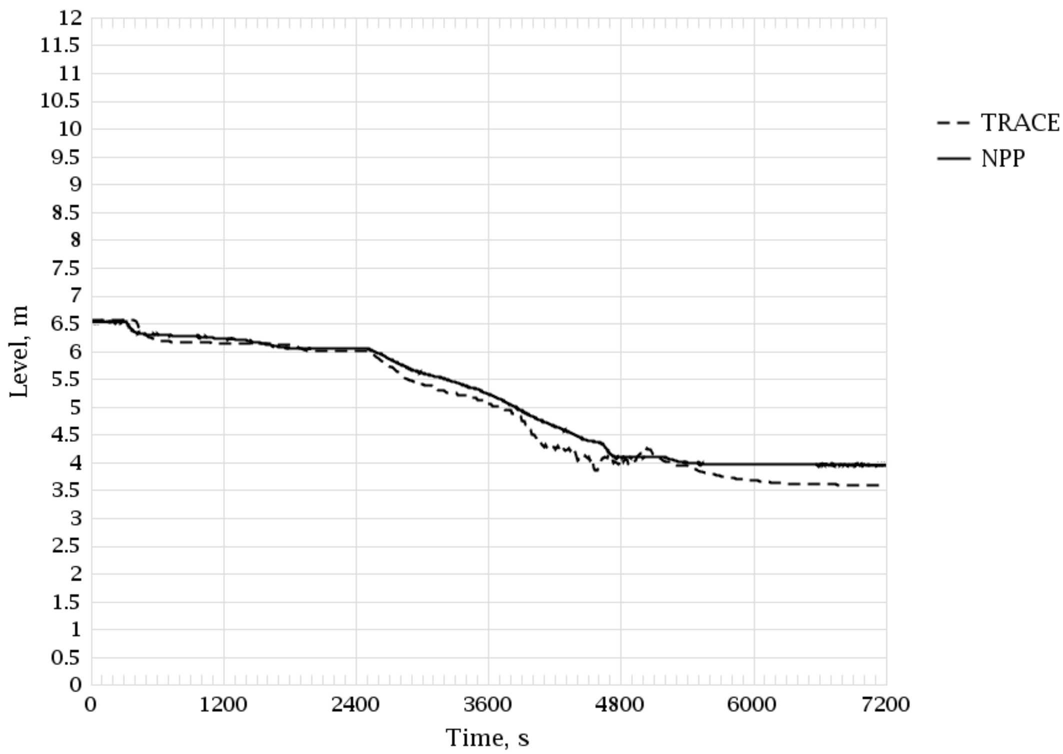

Figure 4-141 HA-2 Level

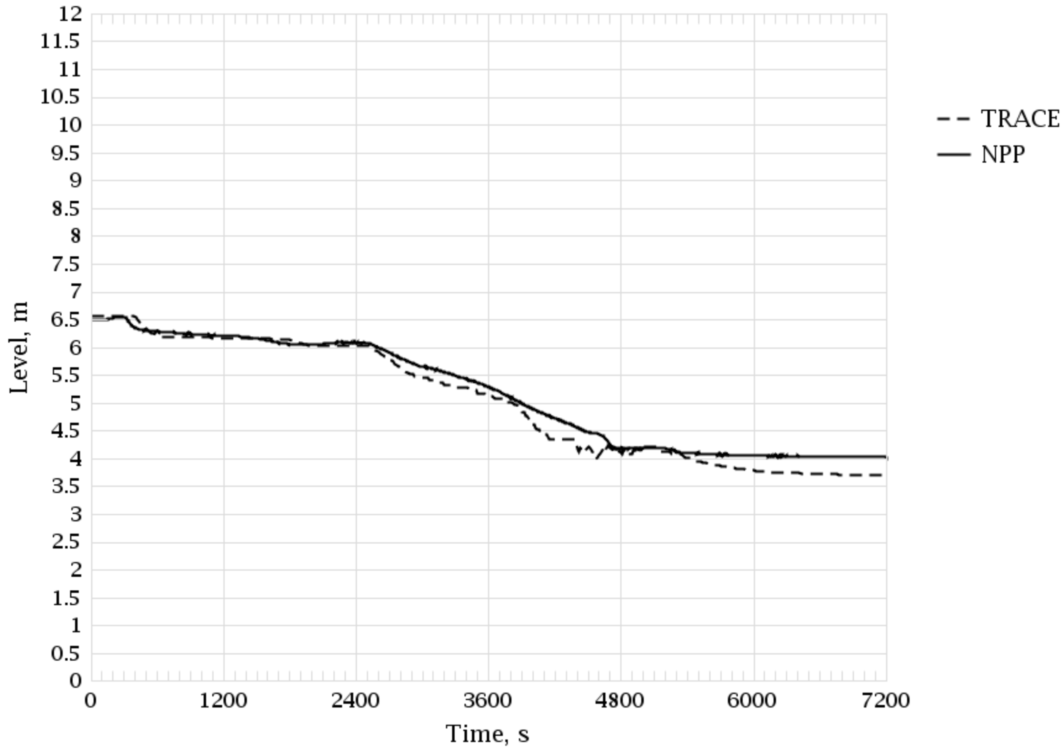

Figure 4-142 HA-3 Level

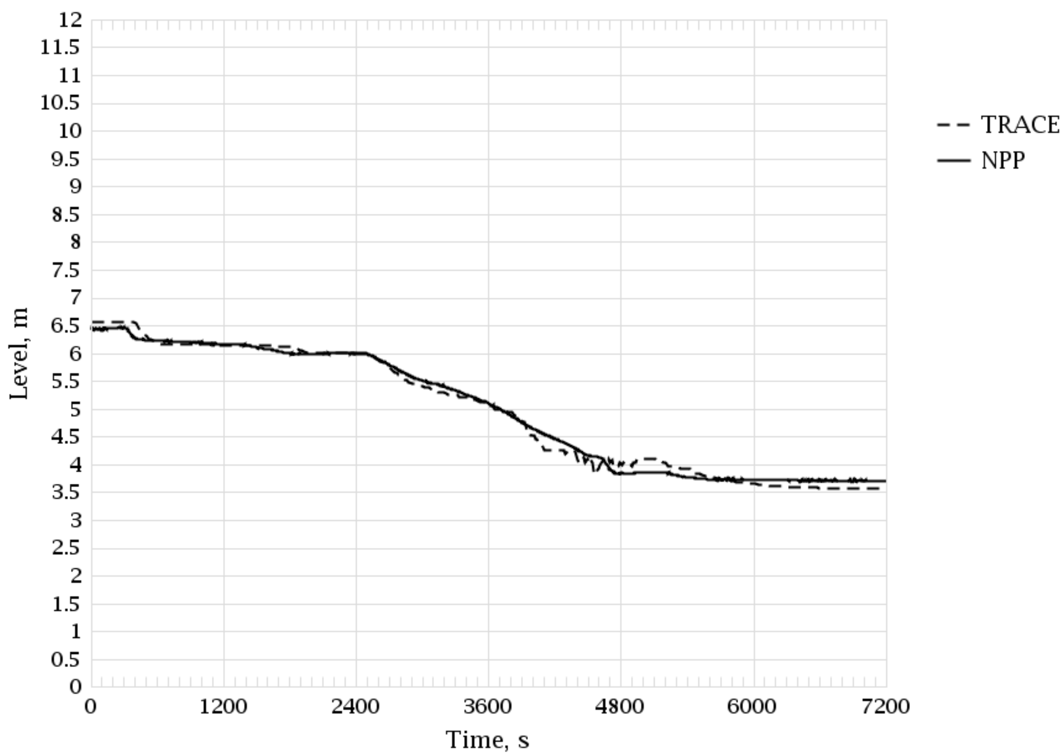

Figure 4-143 HA-4 Level

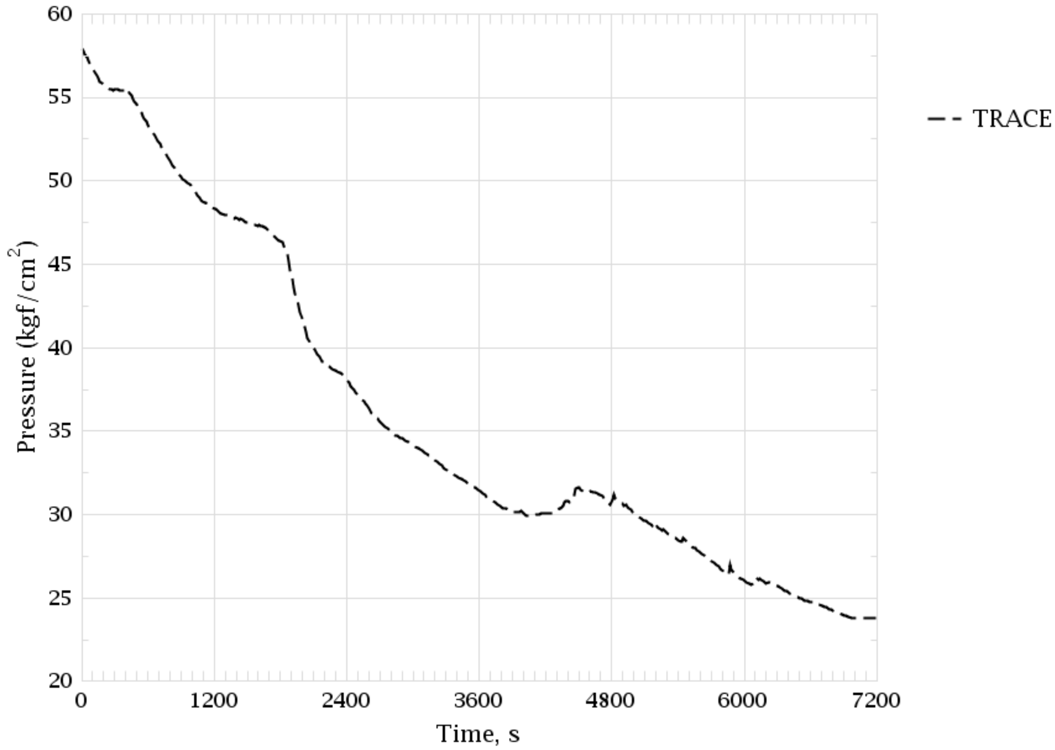

Figure 4-144 MSH Pressure

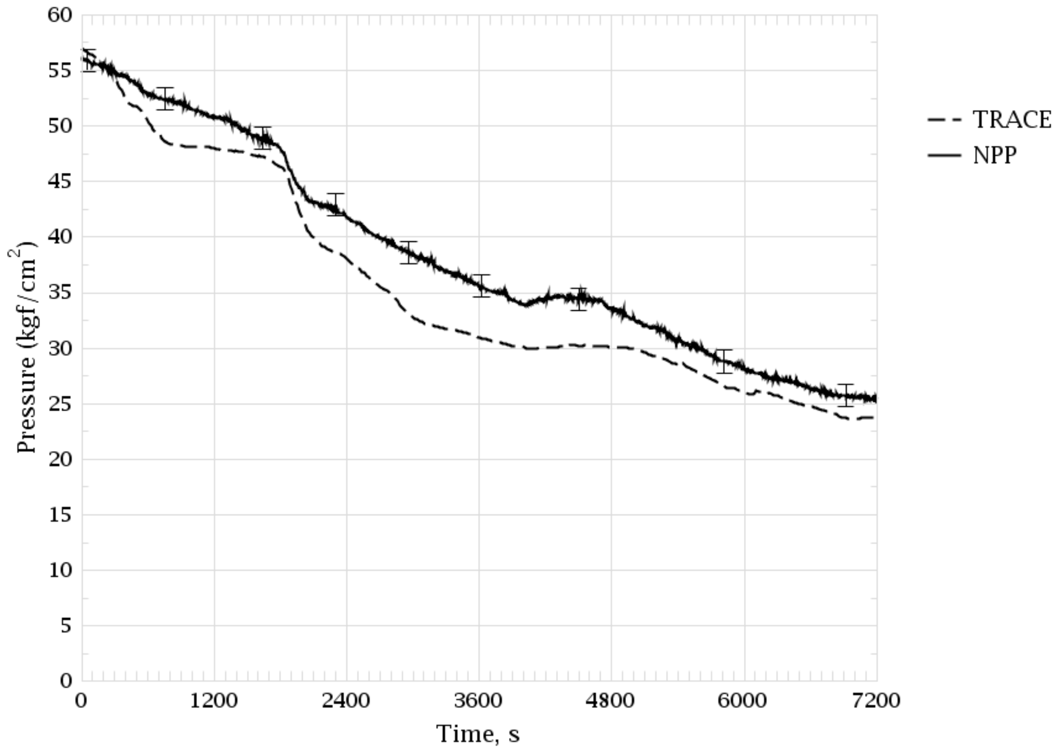

Figure 4-145 SG-1 Pressure

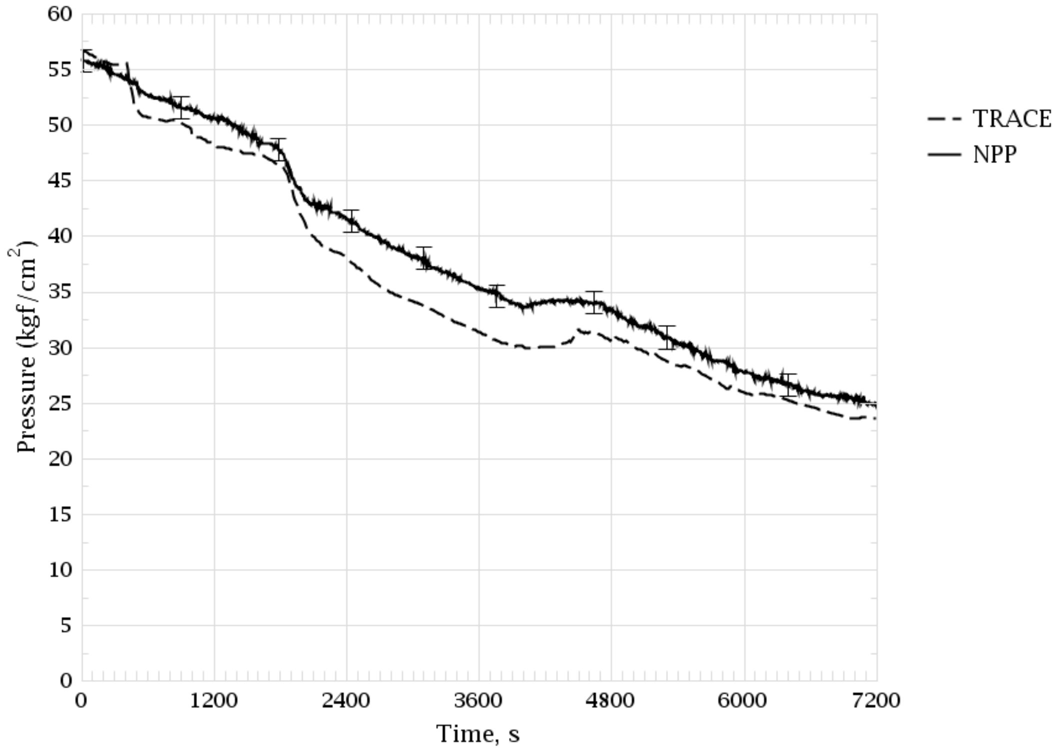

Figure 4-146 SG-2 Pressure

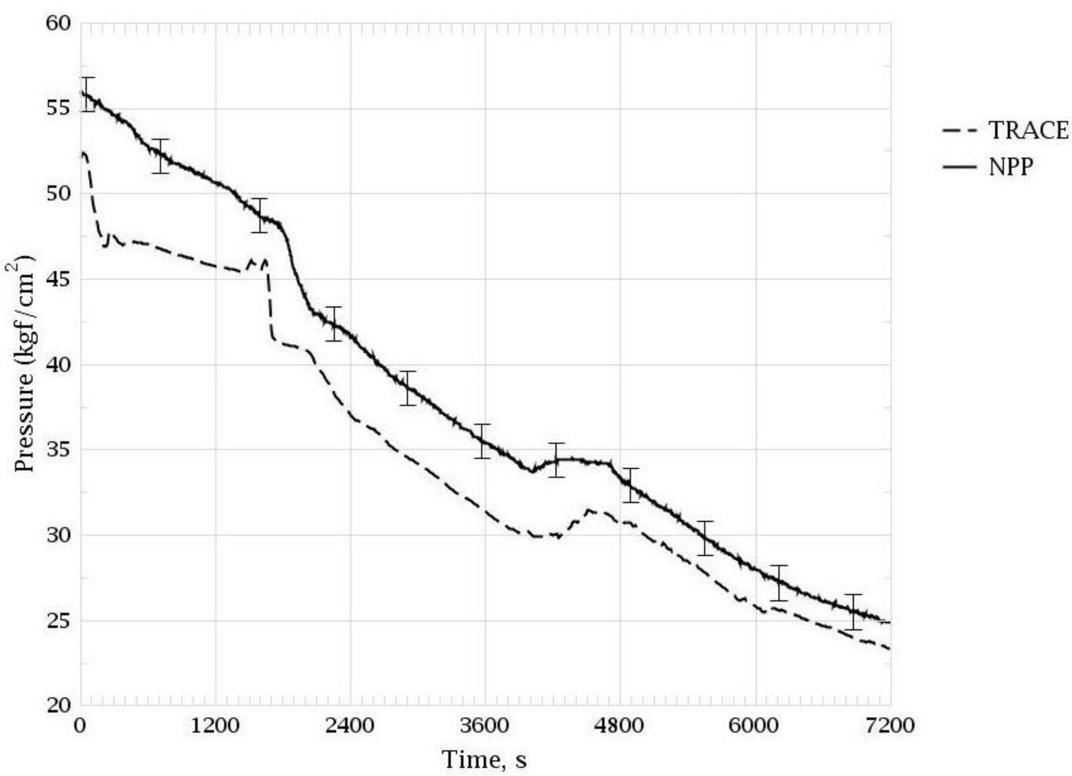

Figure 4-147 SG-3 Pressure

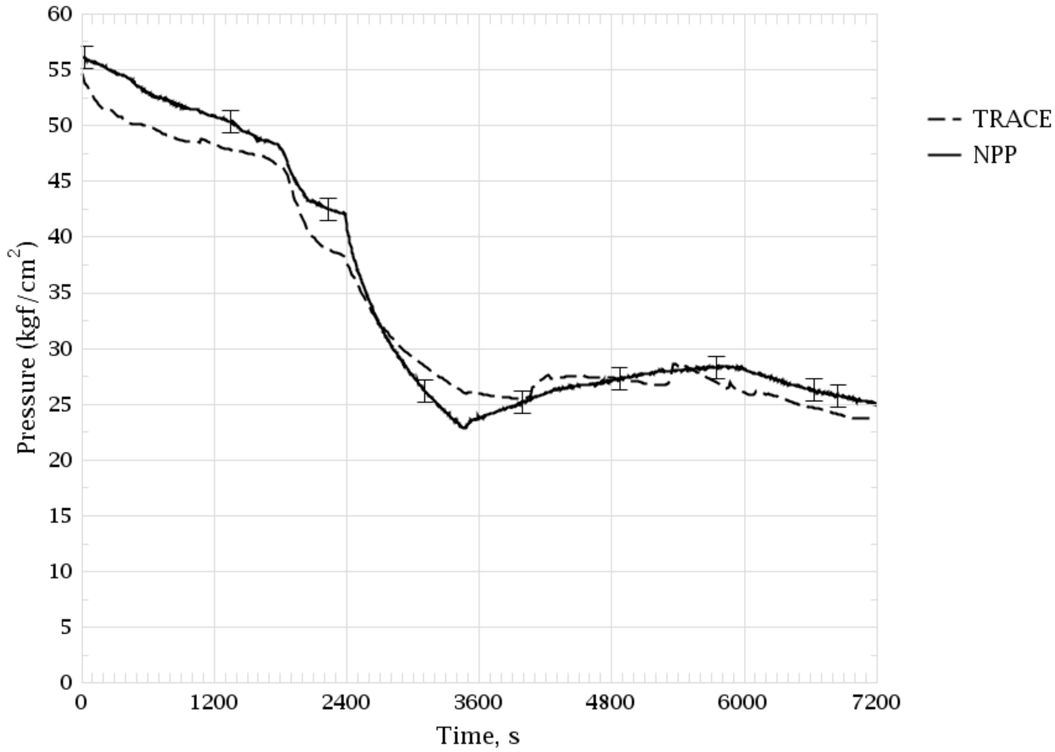

Figure 4-148 SG-4 Pressure

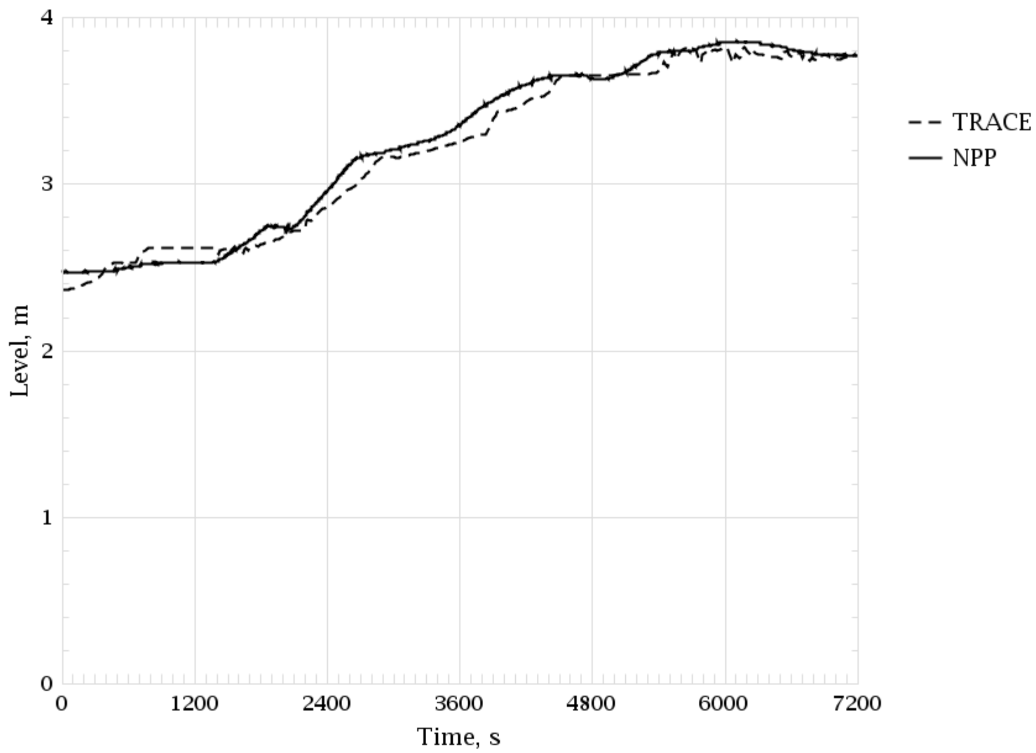

Figure 4-149 SG-1 Level (Wide Range)

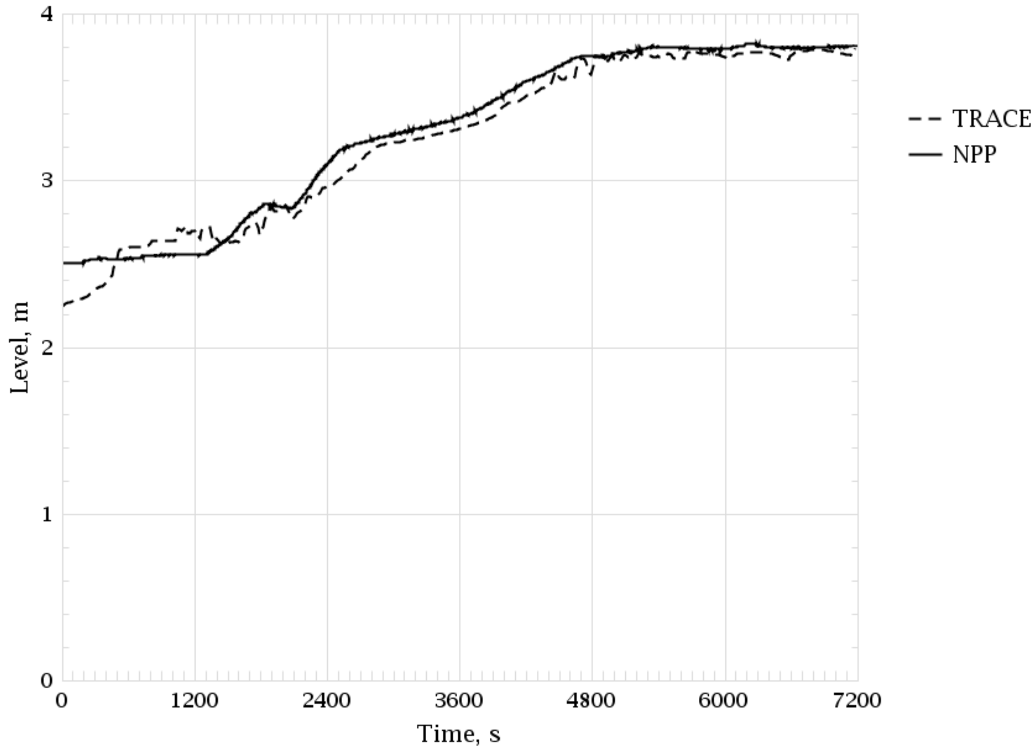

Figure 4-150 SG-2 Level (Wide Range)

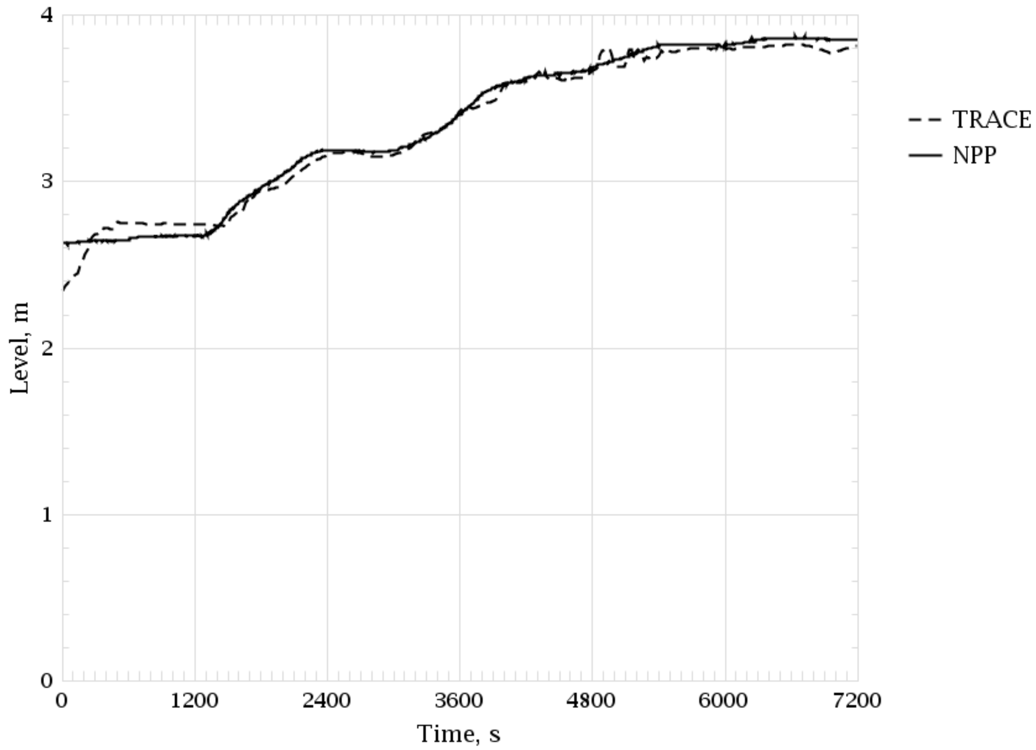

Figure 4-151 SG-3 Level (Wide Range)

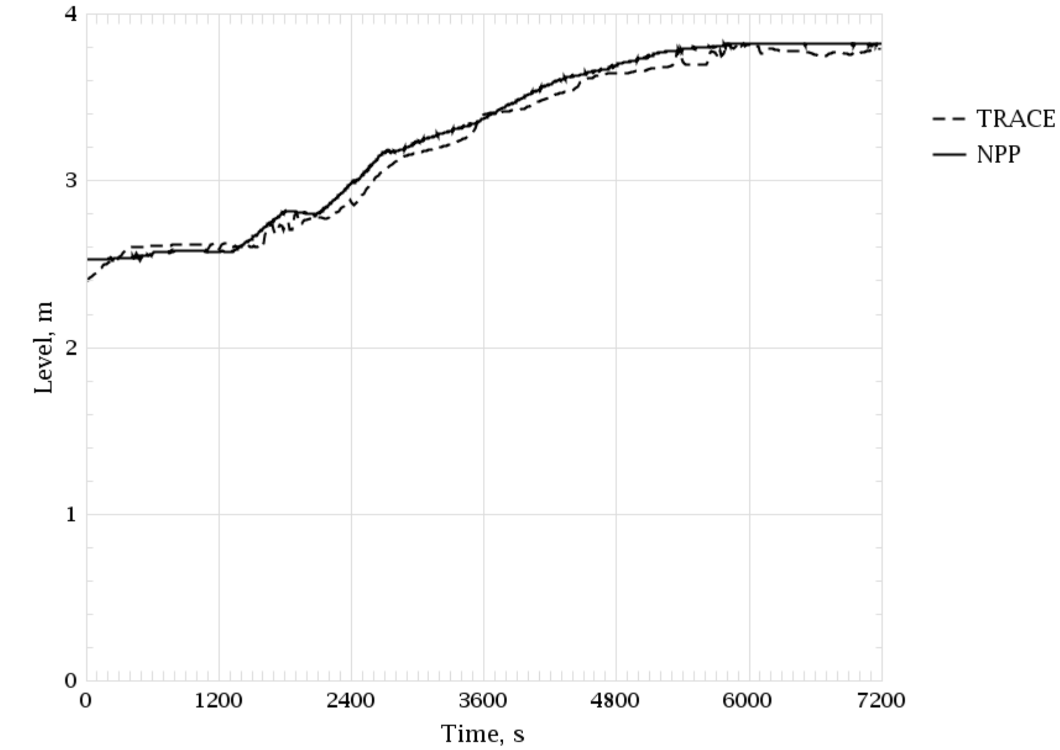

Figure 4-152 SG-4 Level (Wide Range)

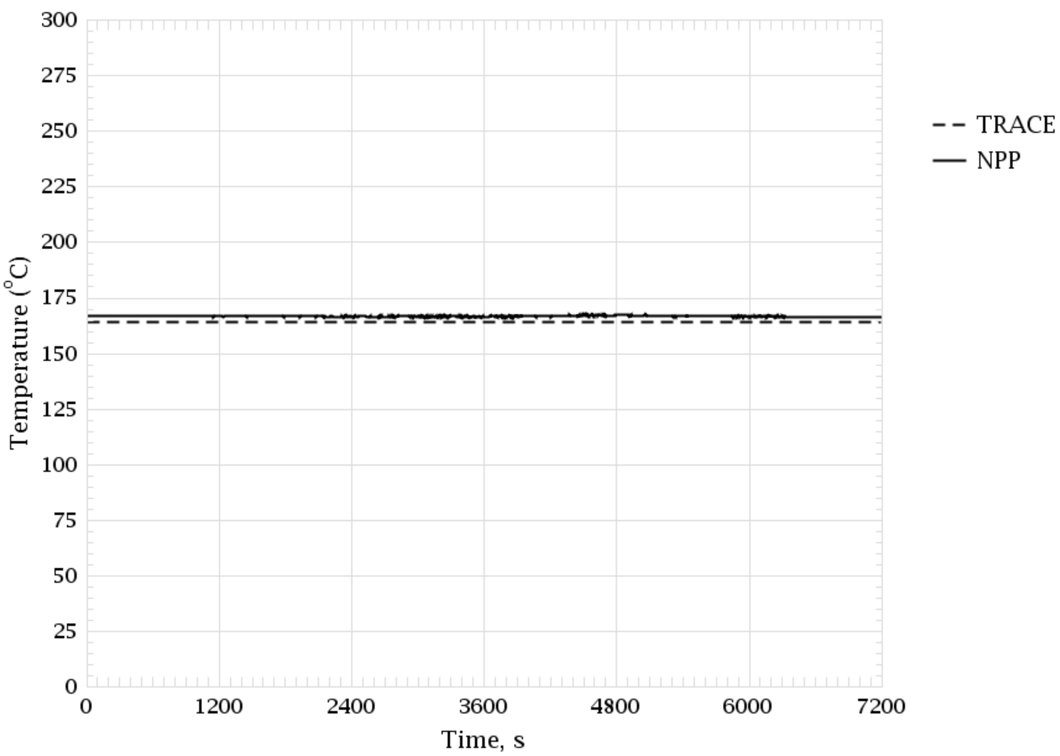

Figure 4-153 Feedwater Temperature

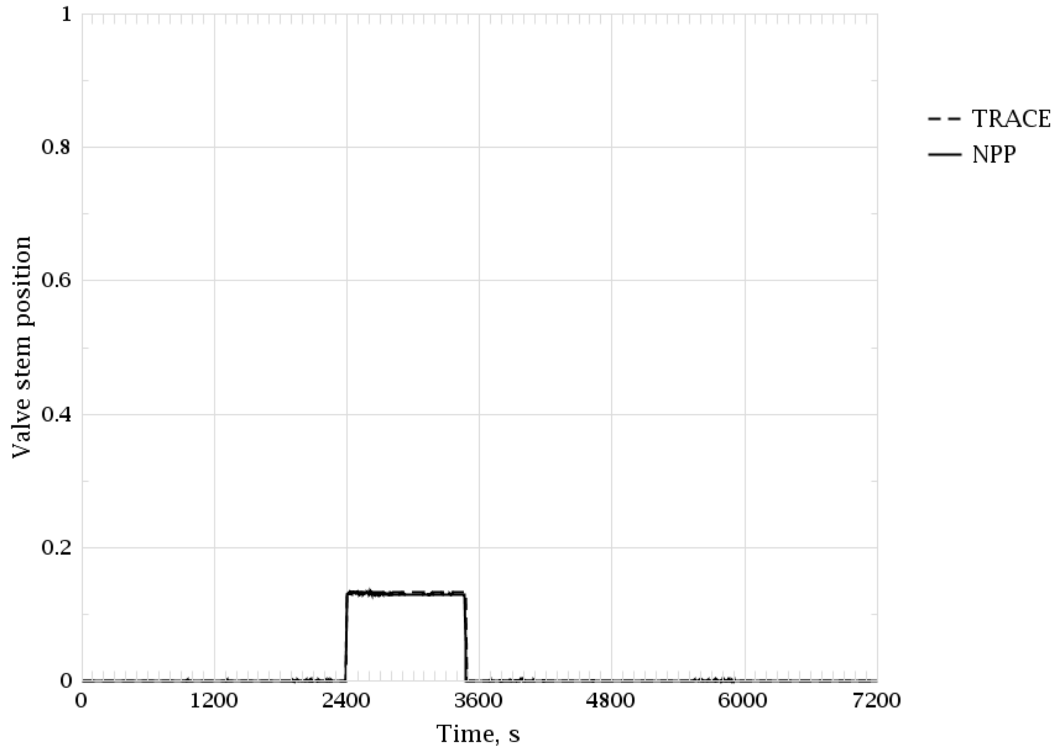

Figure 4-154 BRU-A No.4 Stem Position

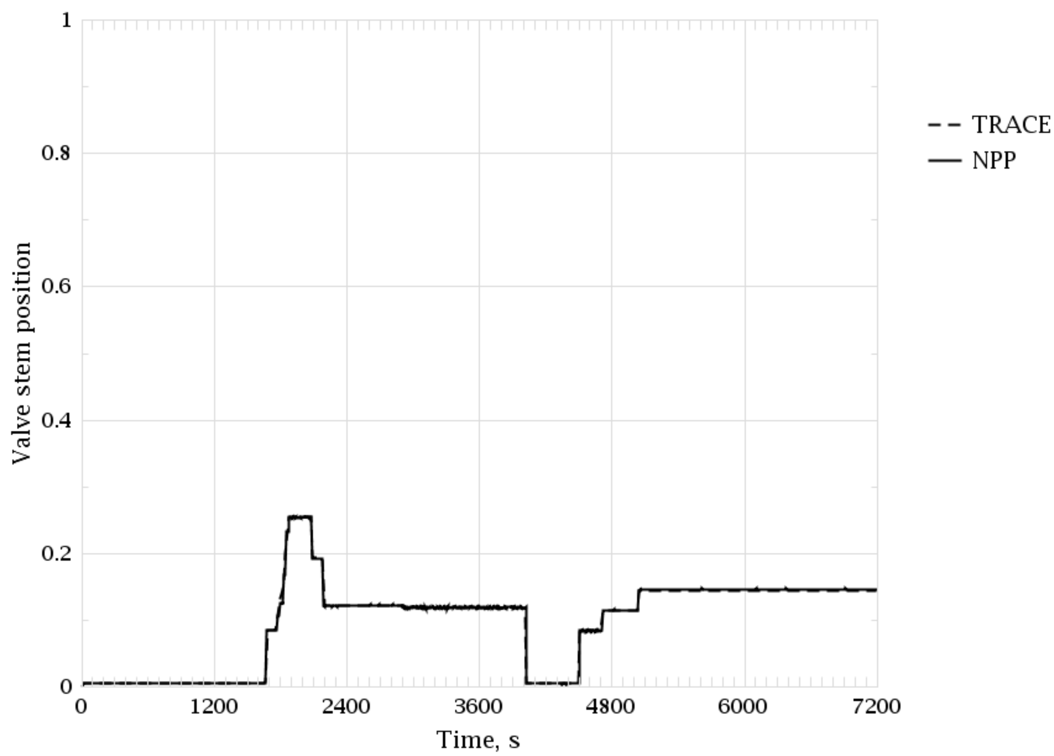

Figure 4-155 BRU-K No.1 Stem Position

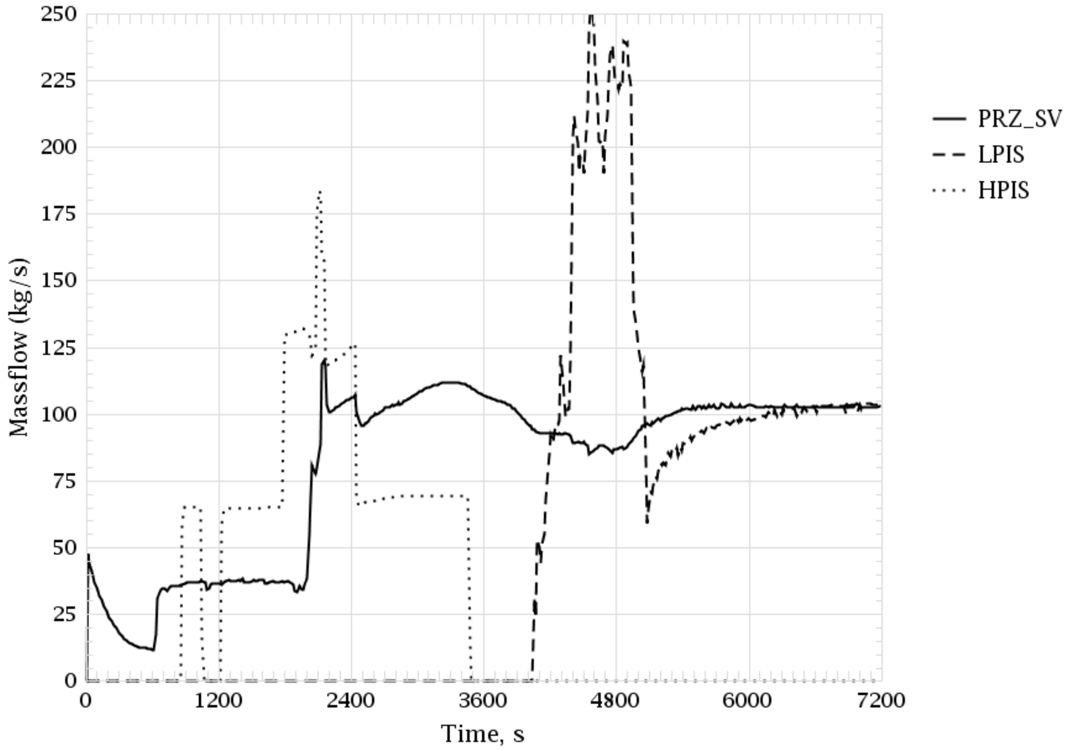

Figure 4-156 Coolant Mass Flow Rate from PRZ PORV, HPIS and LPIS

5 CONCLUSIONS

After development of VVER-1000/V-320 thermal-hydraulic model for TRACE code the validation calculations of several incidents which occurred at Ukrainian NPPs were performed in order to justify capabilities of this model to simulate adequately the plant response during transients and accidents.

The incidents simulated include:

- Zaporizhzhya NPP Unit 5 MFW-1 pump trip;
- Zaporizhzhya NPP Unit 6 inadvertent FASIV closure;
- Rivne NPP Unit 3 PRZ PORV stuck open during tests.

The results of validation calculations demonstrate that developed WWER 1000/V 320 thermal hydraulic model for TRACE code is able to reproduce adequately NPP transient response. For the majority of plant parameters good correspondence between calculated and measured data is obtained in transient scenarios evaluated.

It shall be noted that in RNPP-3 PRZ PORV stuck open transient the noticeable deviations in calculated response of coolant temperature in RCS legs comparing to the measurement data can be observed after start of safety injection. Such deviations are caused by thermal stratification of coolant in RCS loops which cannot be reproduced in one-dimensional model representation. Nevertheless even in this scenario the overall transient response is adequately reproduced by TRACE model.

Based on the results of validation it can be concluded that WWER-1000/V-320 thermal hydraulic model for TRACE computer code which was developed by SSTC NRS can be used for calculations of transients and accidents in support of regulatory review of safety analyses documentation.

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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-0490**
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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

This report is developed by the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) and its technical support organization, the State Scientific and Technical Center for Nuclear and Radiation Safety of Ukraine (SSTC NRS), under Implementing Agreement On Thermal-Hydraulic Code Applications And Maintenance Between The United States Nuclear Regulatory Commission and State Nuclear Regulatory Inspectorate of Ukraine (signed in 2014) in accordance with Article III, Section C, of the Agreement.

The report provides results of the validation calculations conducted with application of SSTC NRS model of VVER-1000/V-320 unit for TRACE computer code. The calculation scenarios simulate actual incidents that occurred at Ukrainian NPPs.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	
Pressurized Water Reactor, Russian Design (VVER) TRACE Reactor Power Controller, Russian designation(ARM) Fast-acting Steam Isolation Valve(FASIV) Hydroaccumulators (HA) Steam Dump Valve to Atmosphere(BRU-A) Steam Dump Valve to the Turbine Condenser(BRU-K) Fast Reactor Load Shedding Protection, Russian designation(UPZ) Protective Tubes Unit(PTU) Submerged Perforated Plate(SPP)	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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