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# International Agreement Report

# TRACE VVER-440/V-213 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-440 for TRACE computer code. The calculation scenarios simulate actual incidents that occurred at Ukrainian NPPs.

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# ABBREVIATIONS

| | |
|---|---|
| AEFWS | Additional Emergency Feedwater System |
| AFW | Auxiliary Feedwater System |
| ARM | Reactor Power Controller, Russian designation |
| BRU-D | Steam Dump Valve to the Deaerator |
| BRU-K | Steam Dump Valve to the Turbine Condenser |
| CAMP | Code Maintenance and Assessment Program |
| ECCS | Emergency Core Cooling System |
| ECFS | Emergency Core Flooding System |
| EFW | Emergency Feedwater System |
| FA | Fuel Assembly |
| FASIV | Fast-acting Steam Isolation Valve |
| HA | Hydroaccumulators |
| HE | Hydraulic Element |
| HPIS | High Pressure Injection System |
| LOCA | Loss of Coolant Accident |
| LPIS | Low Pressure Injection System |
| MCP | Main Coolant Pump |
| MFW | Main Feedwater System |
| MFWP | Main Feedwater Pump |
| MOV | Motor Operated Valve |
| MSH | Main Steam Header |
| MSIV | Main Steam Isolation Valve |
| MSL | Main Steam Line |
| NPP | Nuclear Power Plant |
| NSSS | Nuclear Steam Supply System |

| | |
|---|---|
| PORV | Pilot Operated Relief Valve |
| PRZ | Pressurizer |
| PTU | Protective Tubes Unit |
| RPL | Reactor Power Limiter |
| RCS | Reactor Coolant System |
| RNPP | Rivne Nuclear Power Plant |
| SG | Steam Generator |
| SNRIU | State Nuclear Regulatory Inspectorate of Ukraine |
| SRV | Safety Relief Valve |
| SSTC NRS | State Scientific and Technical Center for Nuclear and Radiation Safety |
| TG | Turbine Generator |
| U.S. NRC | United States Nuclear Regulatory Commission |
| VVER | Pressurized Water Reactor, Russian design |

# 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-440 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 Rivne NPP (RNPP) unit 1 (of VVER-440/V213 design) which are important for development of thermal-hydraulic model. Description of RNPP Unit 1 model for TRACE code is provided in Section 3. The results of TRACE calculations for several scenarios simulating actual incidents that occurred at Rivne NPP Unit 1 (VVER-440/V-213 design) are provided in Section 4 of the report.

The incidents simulated include:

- Reactor scram caused by concrete slab drop to the connection lines of house loads power supply transformer;
- Reactor scram transient initiated by 6 kV switch short circuit;
- Inadvertent reactor scram.

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;
- plots of the main primary and secondary circuit parameters (calculated and measured data);
- discussion of calculation results evaluation.

# 2 MAIN FEATURES OF VVER-440/V213 DESIGN

## 2.1 Reactor Coolant System

### 2.1.1 General Description of Reactor Coolant System

VVER-440/V213 is 440 MW nuclear power plant with pressurized water reactor designed in former Soviet Union. VVER-440/V213 reactor coolant system (RCS) schematic and general layout of the main RCS equipment is shown on Fig. 1 and Fig. 2, respectively. The primary circuit consists of reactor VVER-440 (pos. 1 on Fig. 2) and six identical loops of Dn 500 mm with horizontal steam generators (pos. 2), main gate valves (pos. 5, 6) which are installed at each of the hot (pos. 4) and cold (pos. 7) legs, and main coolant pumps (MCP) of GCN-317 type (pos. 3). Pressurizer (pos. 8) is connected to one of hot legs with the surge line. To ensure flooding of reactor core during large break loss of coolant accidents there are four hydroaccumulators (pos. 10) with borated water which are connected to the reactor pressure vessel.

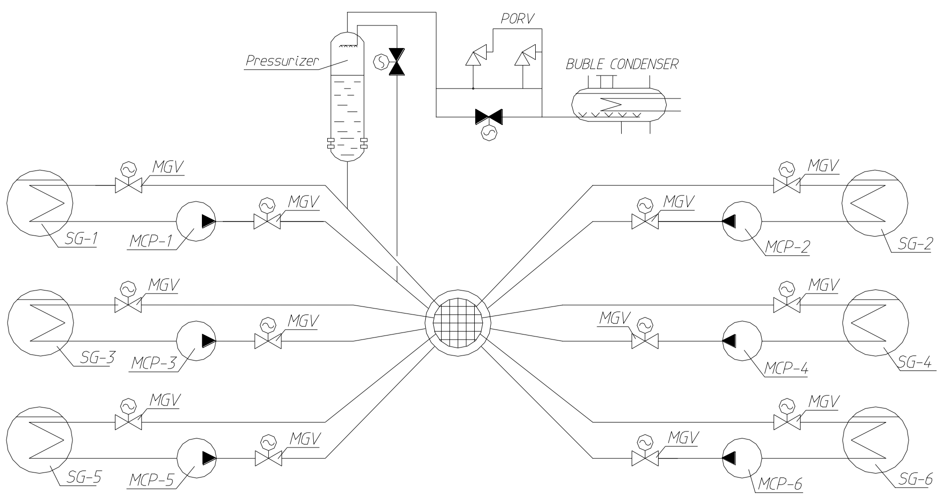



**Figure 1 RNPP Unit 1 Primary Circuit Schematic**

Main characteristics of VVER-440/V213 (according to Rivne NPP Unit 1 Technical Safety Justification [1] and Technical Specification [2]) are provided in Table 1.

**Table 1 Main Characteristics of VVER-440/V213**

| Parameter | Units | Value |
|---|---|---|
| Reactor thermal power | MW | 1375±27 |
| Fuel rod maximal linear heat flux | W/cm | 325.1 |
| Reactor outlet pressure (gauge) | kgf/cm$^2$ | 125±1.2 |
| Coolant temperature at core inlet | °C | ≤267 |
| Coolant temperature at core outlet | °C | 297 |
| Reactor coolant heat-up | °C | 30 |
| Maximal cladding temperature | °C | 335 |

| Parameter | Units | Value |
|---|---|---|
| Reactor coolant volumetric flow | $m^3/h$ | 40700±400 |
| PRZ level | m | 5.96±0.1 |
| SG pressure (gauge) | $kgf/cm^2$ | 46±0.5 |
| SG water level | m | 2.12±0.05 |
| SG steam production | t/h | 450 |
| Main feedwater temperature | °C | 223 |

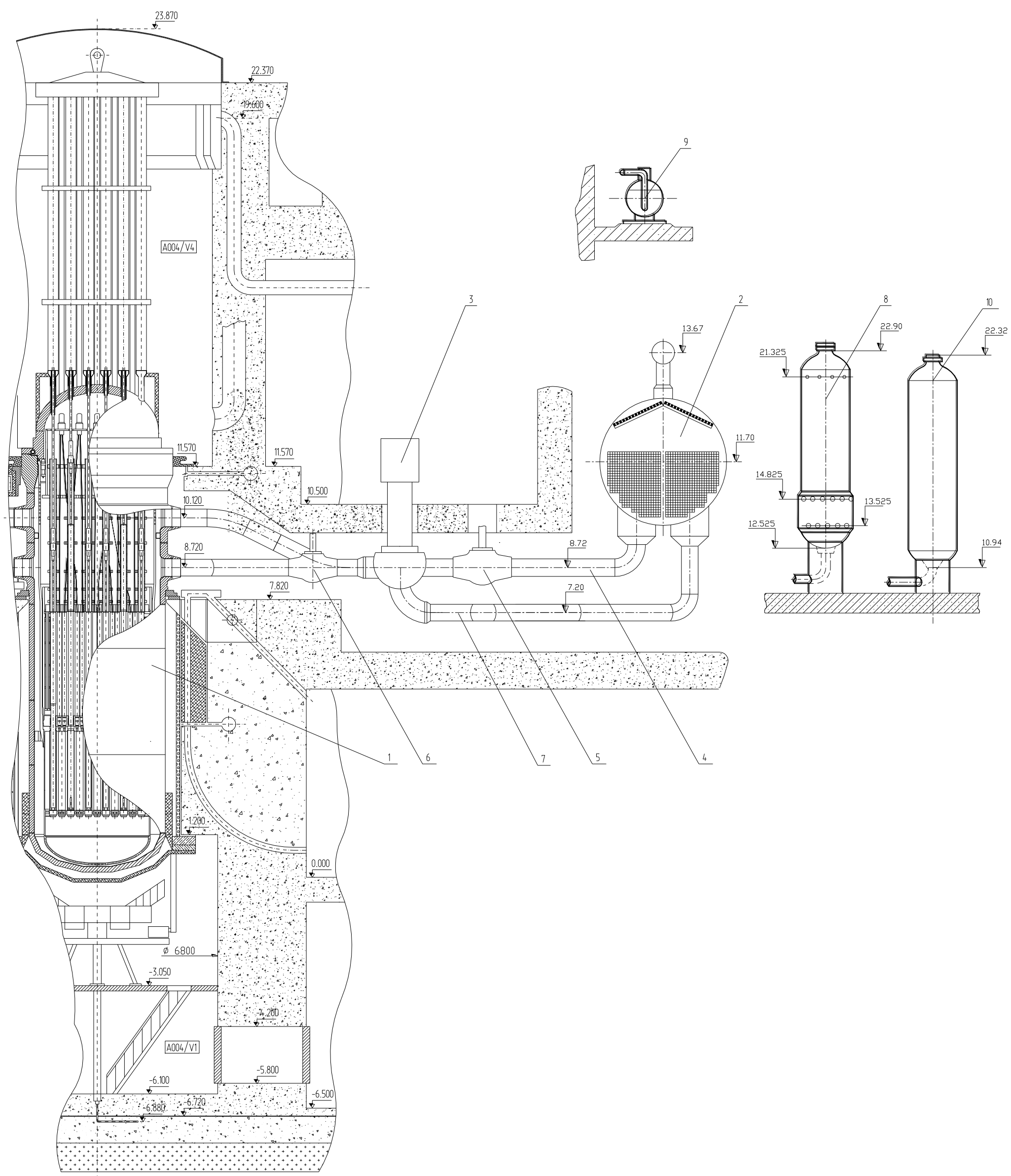



**Figure 2 General Layout of VVER-440/V213 Reactor Coolant System**

### 2.1.2 Reactor

From the reactor inlet nozzles the coolant flows down in the annular gap between reactor vessel (position 1 on Fig. 3) and core barrel (pos. 2), passes through the perforated elliptical bottom of the core barrel (pos. 3) and flows upward through the orifices in the lower plate of the core barrel (pos. 4). The most of the coolant passes through the orifices in upper plate of the core barrel (pos. 5) and enters the bottom shank of fuel assemblies (pos. 6) or dummy assemblies (pos. 7) fixed in the lower plate of core basket (pos. 8). Another part of the coolant through the orifices in protecting and damping tubes (pos. 9) of the core barrel bottom enters fuel or absorber section (pos. 10 and 11) of the control assemblies. The small amount of coolant flows through the gap between core basket (pos. 14) and core baffle (pos. 15) or through the channels of the check test pieces (pos. 16). Heated in the reactor core coolant through the orifices in the lower plate of the protective tubes unit (pos. 17) and in the protective tubes (pos. 18) enters tube space of PTU, and then through the perforation in upper part of the core barrel and reactor outlet nozzles exits the reactor.

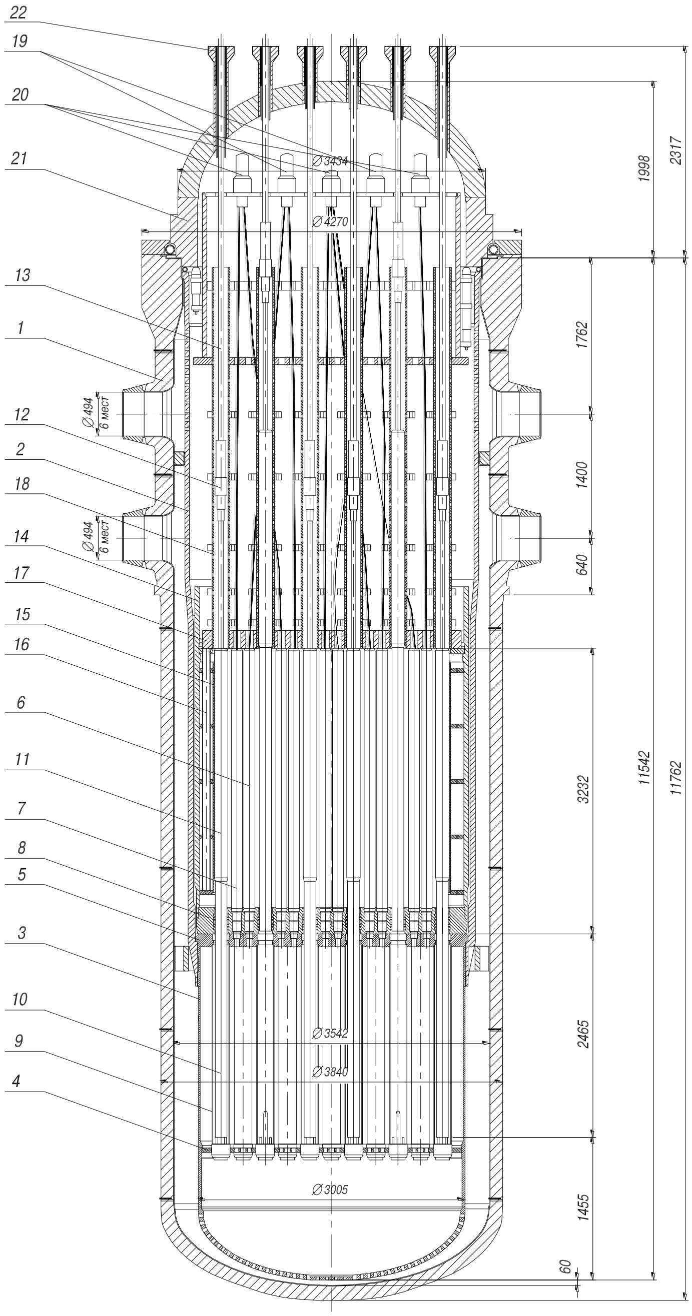



**Figure 3 VVER-440 Reactor**

The core of RNPP Unit 1 is composed of 276 fuel assemblies, 37 movable control assemblies, and 36 dummy assemblies located at the perimeter of the core.

A fuel assembly consists of the fuel rods bundle (126 fuel rods), top nozzle shank, bottom shank and hexagonal shroud tube. The fuel rods are arranged in a triangular grid and spaced by honeycomb-type grids mechanically fixed on the central tube. The flow rate is adjusted by means of throttling orifices in the lower support plate of the core basket.

A control assembly is an executing mechanism of the reactor control and protection system and designed to promptly terminate chain reaction, perform automatic control, and compensate fast changes of reactivity. A control assembly consists of absorber section and fuel section connected by an intermediate shaft. The construction of the fuel section of a control assembly is, with small differences, similar to that of a fuel assembly.

36 dummy assemblies are designed to adjust the core power profile and to reduce neutron flux to the reactor vessel.

### 2.1.3 Primary Circuit Loops

The primary circuit of WWER-440/V213 consists of 6 identical loops. Each loop includes a horizontal steam generator, main coolant pump and two main gate valves. The main circulating pipeline connects RCS equipment and is categorized as a normal operation system. Each of the six circulating loops consists of hot and cold legs (Fig. 4). The hot leg connects the reactor outlet nozzle with the steam generator inlet collector. The cold leg connects the SG outlet collector with the main coolant pump, and MCP with the reactor inlet nozzle.

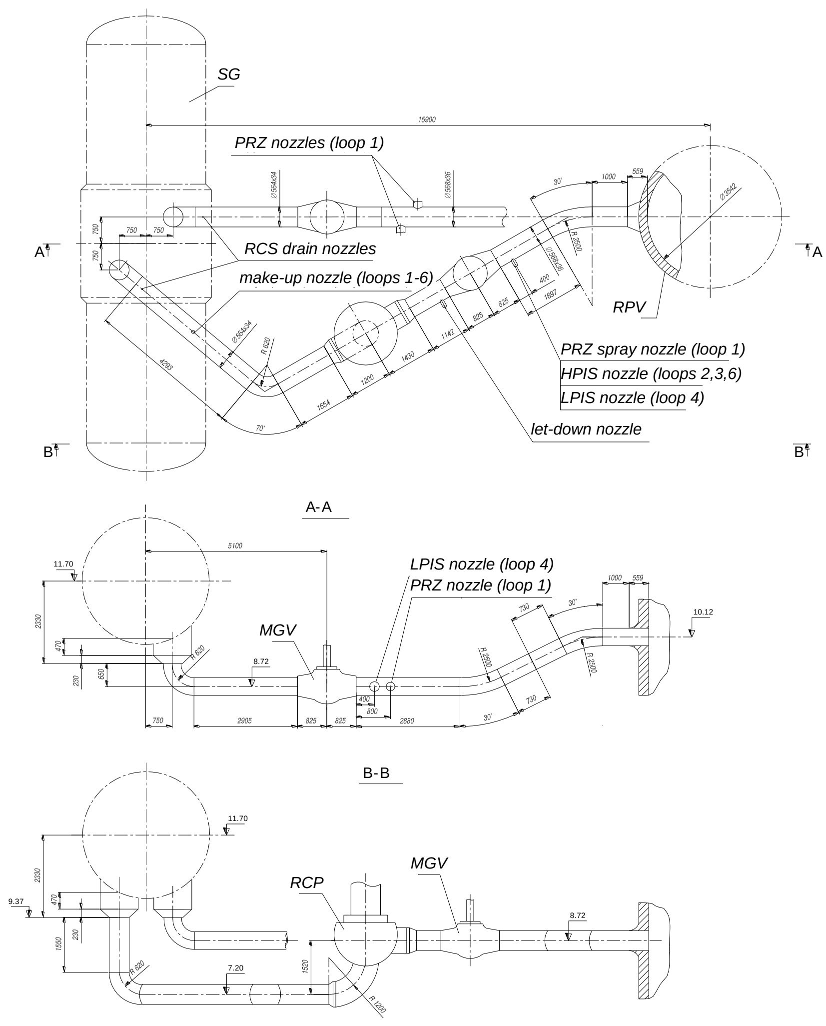
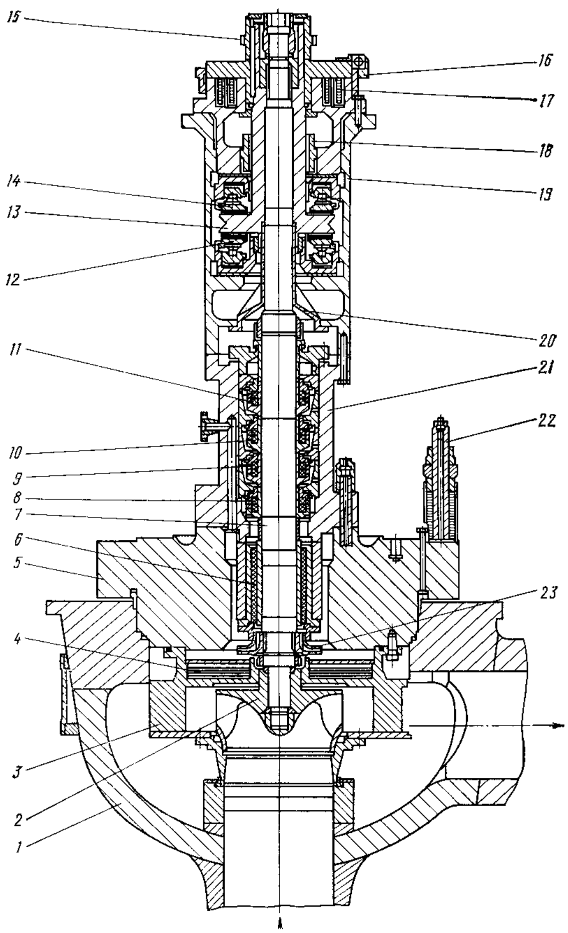



**Figure 4 VVER-440/V213 RCS Hot and Cold Legs Layout**

### 2.1.4 Main Coolant Pump

Circulation of the coolant through the primary circuit is provided by the main coolant pumps GCN-317 of a centrifugal type with mechanical shaft sealing (Fig. 5). A special flywheel is installed on the pump shaft to provide the desired flow pump coast-down. MCP is categorized as a normal operation equipment.




| | | | |
|---|---|---|---|
| 1 | MCP vessel | 12-14 | thrust bearing |
| 2 | impeller | 15 | sleeve |
| 3 | flow guiding unit | 16 | stopper |
| 4 | thermal shield | 17 | electromagnetic unloading unit |
| 5 | flange | 18 | radial bearing |
| 6 | radial bearing | 19 | radial-axial bearing unit |
| 7 | shaft | 20 | tray |
| 8 | MCP sealing separation stage | 21 | sealing unit |
| 9 | MCP sealing main stage | 22 | flange studs |
| 10 | MCP sealing main stage | 23 | auxiliary impeller |
| 11 | MCP sealing ending stage | | |

**Figure 5 Reactor Coolant Pump GCN-317**

### 2.1.5 Main Gate Valves

Main gate valves (Fig. 6) are designed for isolation of a loop for repairing or inspection during the reactor shutdown. The main gate valves can also be used for loop isolation during the main steam line brakes and primary to secondary circuit leaks. Complete isolation of a loop under (i.e., with at least one totally closed main gate valve) during normal plant operation at the nominal or minimal operational power level is not permitted (administrative measure). Main gate valves are categorized as a normal operation equipment.

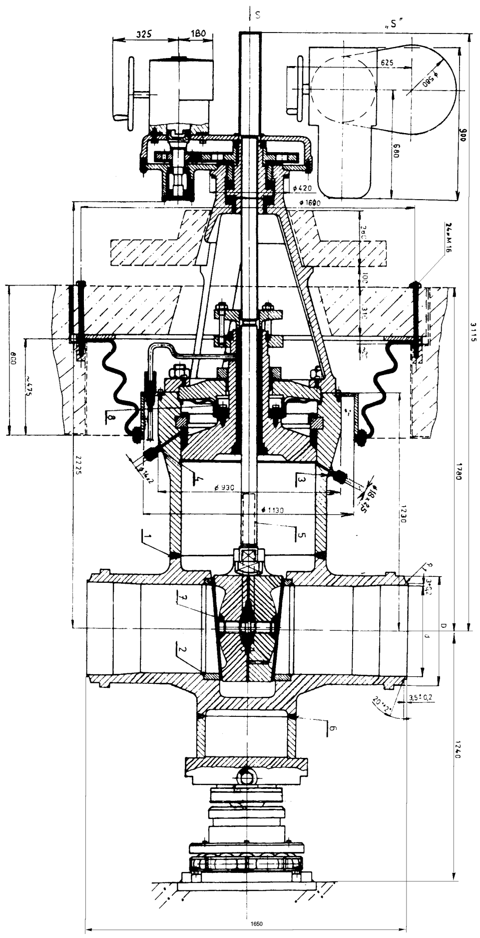


**Figure 6 VVER-440 Main Gate Valve**

### 2.1.6 Steam Generators

The reactor coolant system conducts the heat from the reactor core to horizontal steam generators PGV-213 that provide steam to the turbine generators through main steam lines. Longitudinal and cross-section view of PGV-213 is provided on Fig. 7, 8. Numbers on these figures denote the following SG components:

- SG vessel (pos. 1);
- U-shaped heat-exchange tubes bundle (pos. 13) with support structures (pos. 11, 12);
- primary circuit SG collectors (pos. 2) with collector covers (pos. 3,4);

- steam pipes and manifold (pos. 5, 6);
- main feedwater manifold and distribution pipeline (pos. 9, 10);
- emergency feedwater manifold and distribution pipeline (pos. 14, 15);
- steam separators (pos. 8);
- submerged perforated plate (pos. 7).

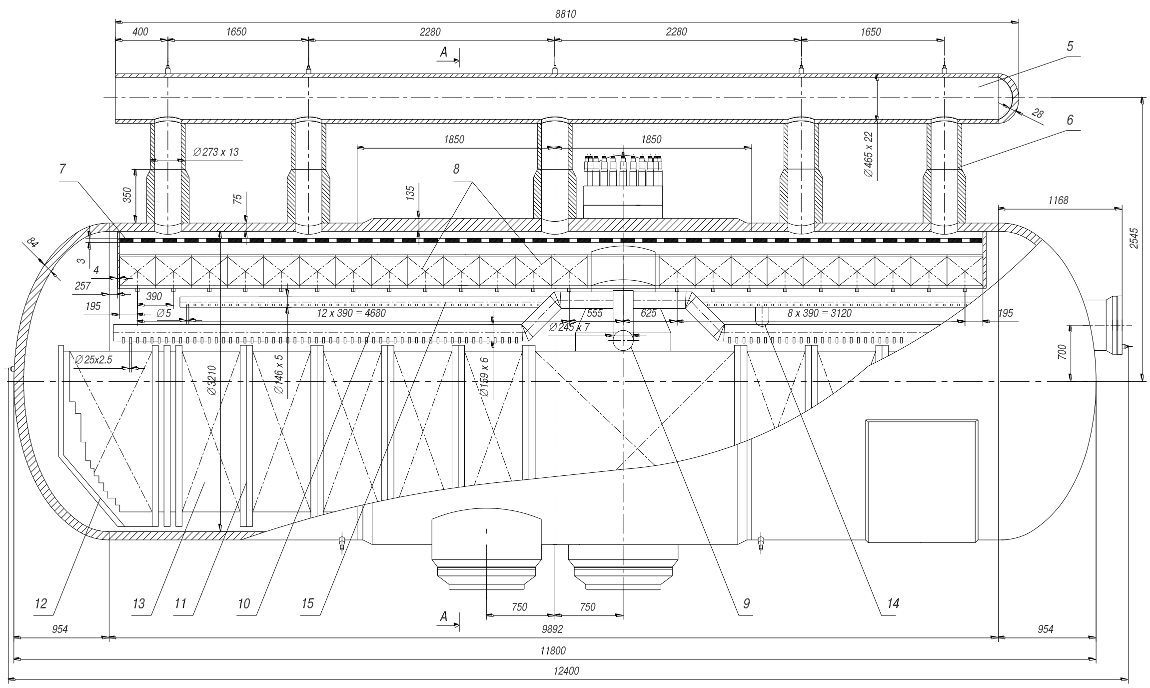


**Figure 7 PGV-213 Longitudinal Section**

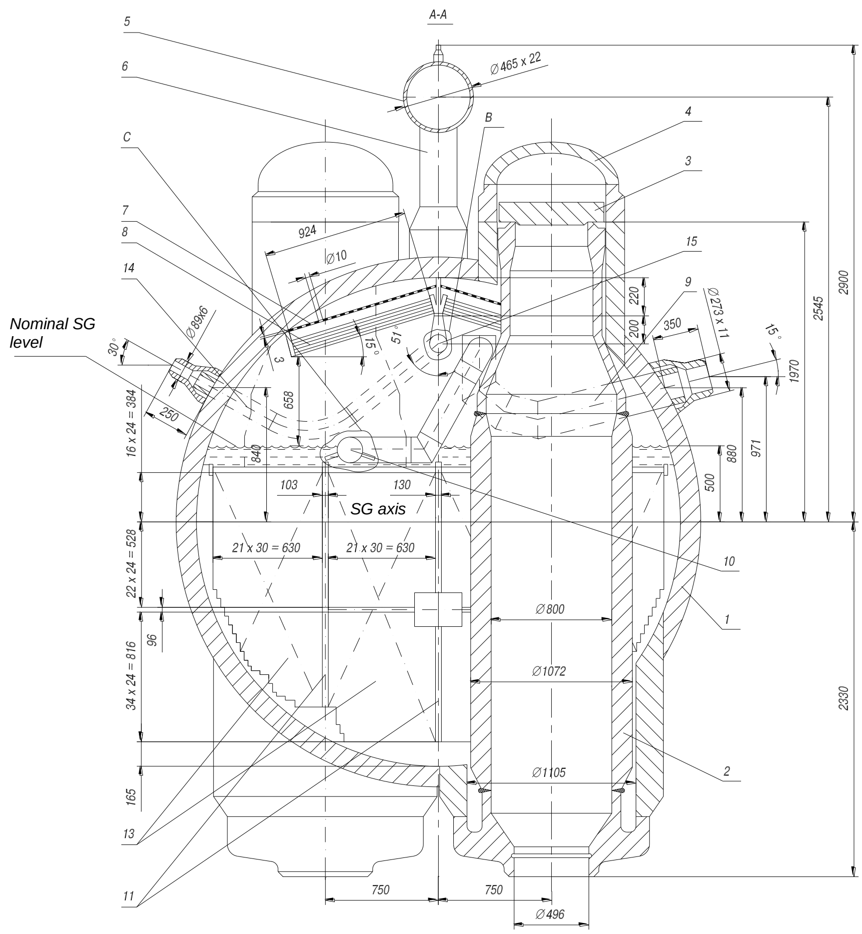



**Figure 8 PGV-213 Cross Section**

### 2.1.7 Pressurizer

During normal power operation, startup and reactor cooling down the primary circuit pressure is maintained by primary pressure control system. This system is categorized as a normal operation system and comprises of pressurizer (Fig. 9), pressurizer (PRZ) heaters, PRZ spray valves, pipelines and valves.

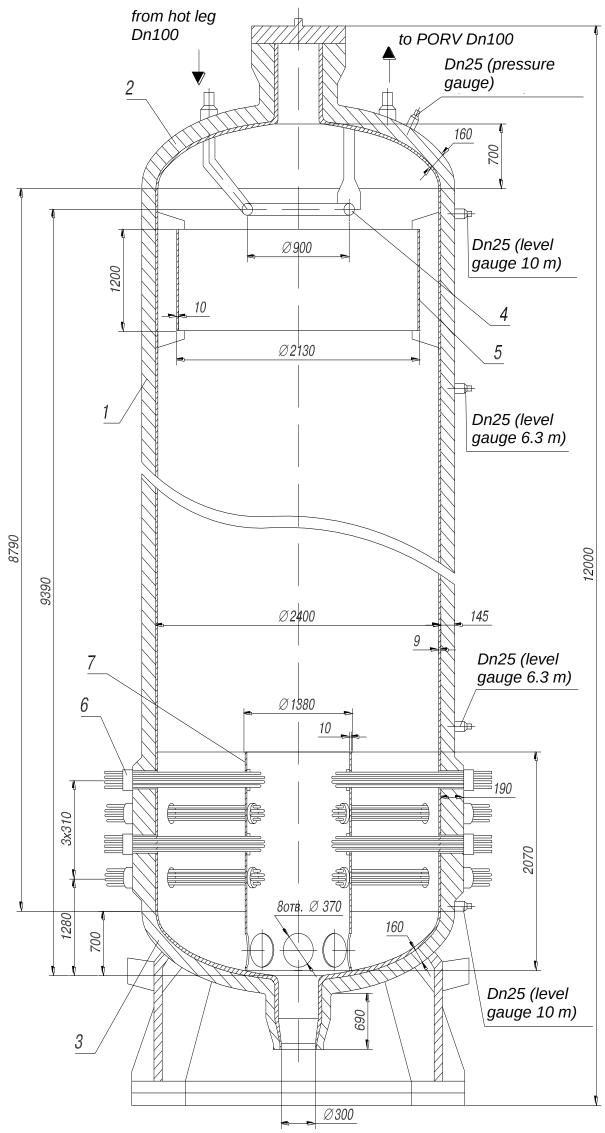



**Figure 9 Pressurizer**

Numbers on Fig. 9 denote the following pressurizer parts: 1 – PRZ vessel cylindrical sections, 2, 3 – elliptical head and bottom; 4 – PRZ spray; 5 – upper shield; 6 – PRZ heaters; 7 – heaters baffle.

## 2.2 Technological and Safety Systems Connected to the Primary Circuit

The major technological and safety systems of the primary circuit are:

- chemical and volume control system;
- emergency gas evacuation system;
- RCS overpressure protection 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 MCP 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 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 let-down subsystem;
- primary coolant deaerating subsystem;
- primary make-up subsystem;
- MCP sealing water subsystem;
- demineralized water supply subsystem;
- primary coolant treatment subsystem;
- boric acid storage and supply subsystem; and
- pressurizer cool-down subsystem.

### 2.2.2 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 and steam generator collectors) 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.

### 2.2.3 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 two identical and independent trains. Each train includes a pilot operated relief valve (PORV) which opens automatically at the preset pressure value and dumps coolant to the relief tank. PORV can also be manually controlled by energizing or de-energizing appropriate pilot valve's solenoid that allows to implement primary feed-and-bleed procedure if the secondary circuit heat removal can not be established.

Additionally bypass pipeline that connects pressurizer with discharge line to the relief tank is installed. The setpoints of bypass valves are lower than ones of PORV that ensures their actuation prior to the operation of PORVs.

### 2.2.4 Emergency Core Cooling System

Emergency core cooling system (ECCS) includes emergency core flooding system (ECFS), high pressure injection system (HPIS) and low pressure injection system (LPIS). ECCS diagram is shown on Fig. 10.

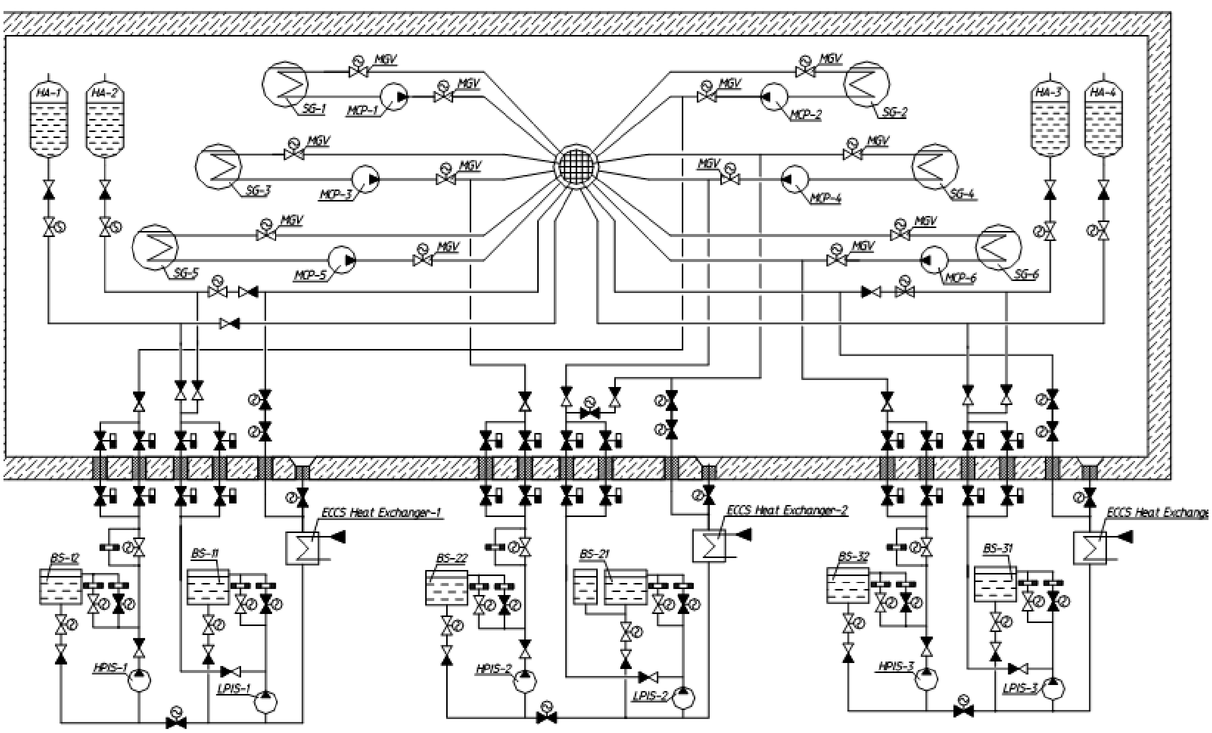



**Figure 10 ECCS Diagram**

ECFS is a passive system designed to inject borated water into reactor core during accidents resulting in RCS pressure drop below 60 bar.

ECFS consists of four hydroaccumulators (HA) directly connected to the reactor pressure vessel by pipelines of Dn 250 mm. Each hydroaccumulator contains 40-50 $m^3$ of boric acid of 16 g/kg concentration at pressure of 60 bar. Two of hydroaccumulators inject water to the upper plenum and two others inject water to the downcomer. Two check valves and motor operated valves (MOV) are located at each of four connecting pipelines.

HPIS is designed to provide borated water to the reactor during accidents in which RCS pressure remains high. The system is composed of three identical trains. Each train is able to fulfill intended function for the whole system and includes the following main equipment:

- high pressure injection pump;
- boric acid storage tank (BS-12, 22, 32), which contains 90 $m^3$ of boric acid of 40 g/kg concentration;
- ECCS heat exchanger;
- pipelines and valves.

HPIS pumps have independent charging lines which separately penetrate containment. Charging lines of HPIS trains 1, 2 and 3 are connected to non-isolatable part of cold legs of RCS loops #2, 3 and 6, respectively.

LPIS is designed to provide emergency core cooling during large and medium LOCAs, RCS cooling down in response to a transient, RCS planned cooling down and long term residual heat removal during a refueling outage.

LPIS is composed of three identical trains. Each train is able to fulfill intended function for the whole system and includes the following main equipment:

- low pressure injection pump (NOR-1, 2, 3);
- ECCS heat exchanger (TOS-1, 2, 3);
- borated water tank (BS-11, 21, 31), which contains 260$m^3$ of boric acid of 12 g/kg concentration;
- pipelines and valves.

LPIS tank (BS-11, 21, 31), pipelines Dn 600 with ECCS heat exchanger and MOVs, as well as suction header are common for particular HPIS, LPIS and spray system train.

LPIS pumps have independent charging lines which separately penetrate containment. Charging line of the 1st LPIS train is connected to the connecting lines of the 1st and 2nd HA between the check valves. Charging line of the 2nd LPIS train is connected to non-isolatable part of cold and hot legs of RCS loop #4. Charging line of the 3rd LPIS train is connected to pipelines of the 3rd and 4th HA between the check valves.

## 2.3 Technological and Safety Systems of the Secondary Circuit

The major technological and safety systems of the secondary circuit are:

- main steam lines system;
- turbine bypass to condenser BRU-K;
- steam dump valves to atmosphere BRU-A;
- secondary circuit overpressure protection system;
- fast acting steam isolation valves;
- main feedwater system;
- auxiliary feedwater system;
- emergency feedwater system;
- additional emergency feedwater system.

### 2.3.1 Main Steam Lines System

The main steam lines system (Fig. 11) is intended for transportation of steam produced in steam generators to turbines and other systems and equipment connected to the steam lines.

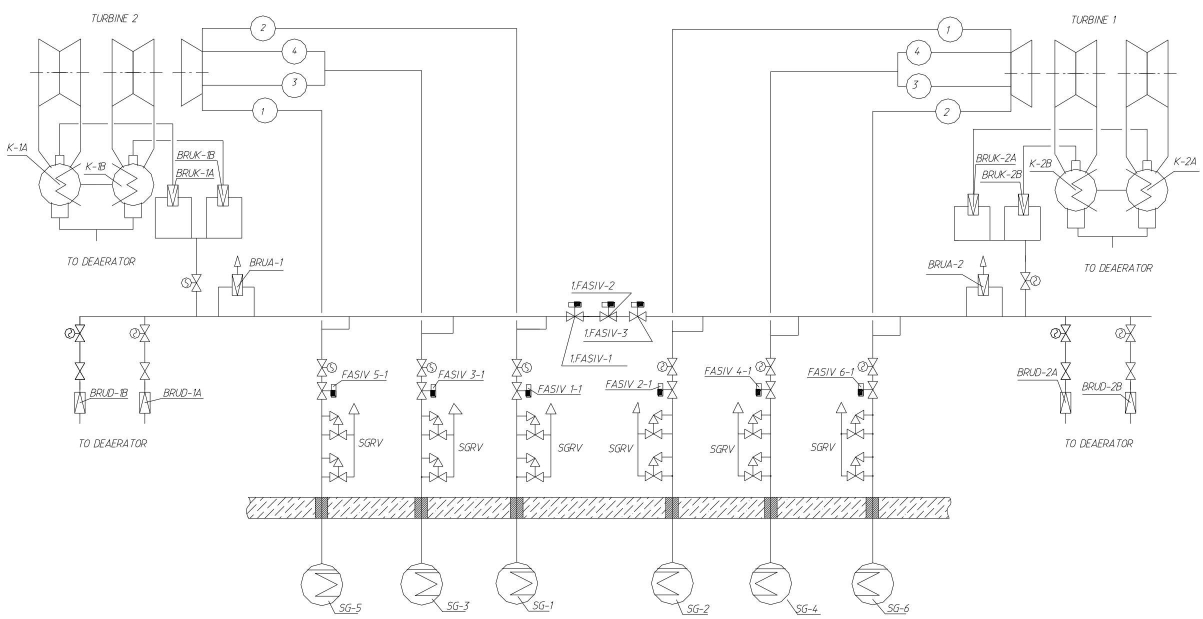



**Figure 11 Main Steam Lines System Diagram**

The steam from each SG is delivered via the separate pipelines to the turbine stop valves. The 1st, 3rd and 5th steam generators are connected to the first turbine. The 2nd, 4th and 6th steam generators are connected to the second turbine of the unit. All steam lines are connected to the main steam header (MSH), which is divided by three fast-acting steam isolation valves (FASIV) into two semi-headers. FASIVs are also installed at each of the main steam lines to isolate the steam generators in the case of the main steam line or main steam header breaks.

### 2.3.2 Turbine Bypass to Condenser BRU-K

Turbine bypass to condenser BRU-K is intended to maintain the secondary circuit pressure below 48 bar by steam dump to the turbine condenser and to provide the secondary circuit cool-down. The system consists of four BRU-K connected to the main steam header (two BRU-K at each semiheader) by pipelines of Dn 400 mm.

During normal power operation all BRU-Ks are in stand-by mode. BRU-Ks open and operate in pressure maintenance mode automatically by control signals from associated BRU-K pressure controllers. Secondary circuit cool-down via BRU-K is performed by operator.

### 2.3.3 Steam Dump Valves to Atmosphere BRU-A

BRU-A are designed to provide controlled steam dump to atmosphere and to prevent secondary circuit pressure increase to SG safety relief valves (SRV) opening setpoints.

The system consists of two BRU-A connected to the main steam header (one BRU-A at each semiheader) by pipelines of Dn 150 mm.

During normal power the system is in a stand-by mode. BRU-As open and maintain secondary circuit pressure automatically by control signals from associated BRU-A pressure controllers. BRU-A maintains steam pressure in the range of 50-52 bar. Forced BRU-A closure is performed if steam pressure decreases below 46 bar. To perform secondary circuit cool-down the operator deactivates interlock for BRU-A closure and remotely control the valve position to achieve desired cooldown rate.

### 2.3.4 Secondary Circuit Overpressure Protection System

Secondary circuit overpressure protection system is categorized as a safety system and designed to protect steam generators and main steam lines from pressure increasing above the design limits.

Each steam generator is equipped with two safety relief valves located at the main steam lines outside the hermetic compartments upstream the fast acting steam isolation valves. Each safety relief valve consists of pilot operated relief valve and pilot valve. Though the pilot valve is designed to open and close automatically, the operator can control it manually energizing or de-energizing pilot valve's solenoid.

During normal power operation the system is in a stand-by mode. If secondary circuit pressure exceeds the pilot valve opening setpoint the valve will automatically open. Once the pilot valve is opened the actuating steam pressure forces opening of correspondent pilot operated valve. The setpoints for opening/closing of one of SRVs are slightly lower than setpoints for the other one (56.7 bar against 57.9 bar for opening, and 49 bar against 51 bar for closing).

### 2.3.5 Fast Acting Steam Isolation Valves

Fast acting steam isolation valves system is categorized as a safety system and designed to isolate SGs and/or main steam header during secondary circuit breaks.

The system includes the following equipment:

- fast acting steam isolation valves P1÷6-1 and cut-off MOVs P1÷6-2 located at the main steam lines from each SG;
- FASIVs P1, 2, 3 dividing the main steam header into two semiheaders.

During normal power operation the system is in a stand-by mode.  FASIVs are automatically closed if the following signals are generated:

- pressure differential between MSH and SG is greater than 5 bar and primary circuit temperature is greater than 150 °C;
- pressure gradient $dp/d\tau$ in MSH is less than -0.7 bar/s (holding for 1s) and primary circuit temperature is greater than 150 °C.  Time delay is 4 s.

On the first signal listed above the FASIV and MOV at the main steam line of affected SG are closed.  Actuation of the signal "MSH pressure gradient $dp/d\tau$ -0.7 bar/s" leads to closure of all FASIVs and MOVs at the main steam lines, as well as FASIVs at the main steam header.

FASIVs can also be manually controlled by operator.

### 2.3.6 Main Feedwater System

Main feedwater (MFW) system is designed for deaeration and transporting of feed water to the steam generators.  The system is categorized as normal operation system.  The system is common for both turbines and comprises of two deaerators, five main feed water pumps, pipelines and valves (Fig. 12).

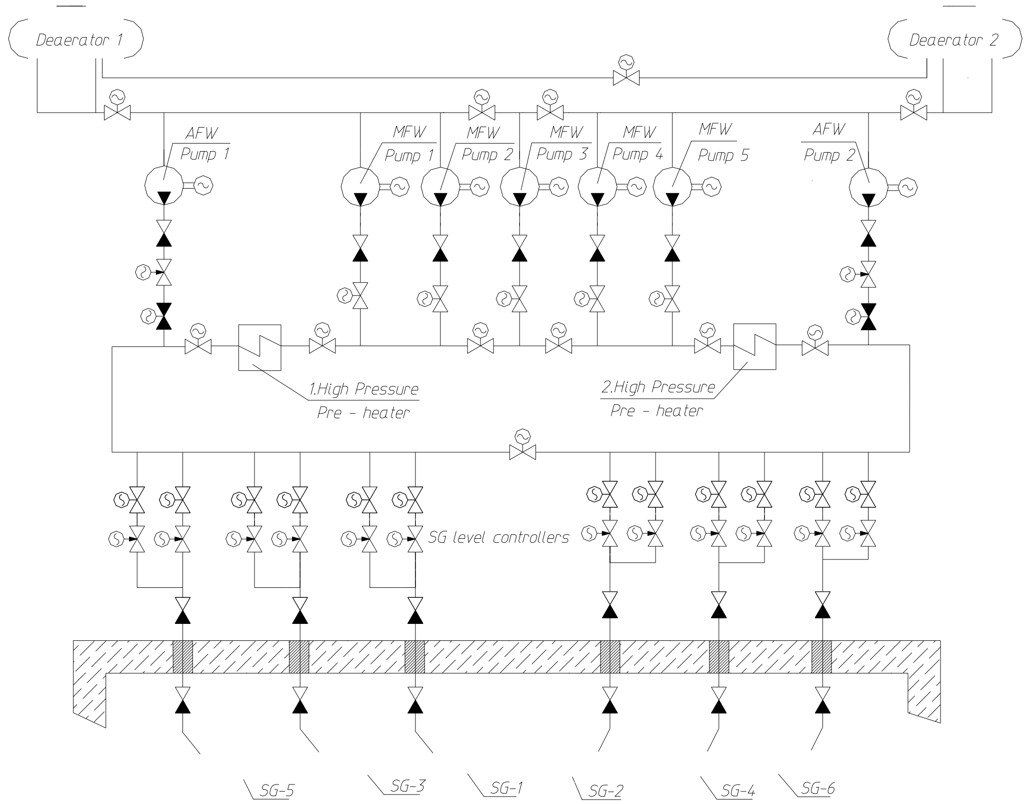



**Figure 12 Main and Auxiliary Feedwater Systems Diagram**

The feed water system is designed so that feed water is distributed from MFW collector to each of the steam generators through individual pipelines. MFW collector is divided by two cut-off valves into two semicollectors. The 1st, 3rd and 5th steam generators are connected to the first feed water semicollector whereas the 2nd, 4th and 6th steam generators are connected to the second semicollector.

### 2.3.7 Auxiliary Feedwater System

Auxiliary feedwater system (AFW) is designed to provide feedwater to steam generators and maintain SG level at the nominal value during accidents with failure of main feedwater system, as well as during reactor start-up and cool-down.

AFW system consists of two independent trains with the identical components sets. The system includes the following equipment:

- AFW pumps (APEN-1, 2);
- cut-off valves located at the bypass lines of MFW control valves;
- start-up feedwater control valves;
- AFW pipelines and valves.

The AFW suction pipelines, feedwater collector and pipelines from feedwater collector to SGs are common for the main and auxiliary feedwater systems.

During normal power operation the system is in a stand-by mode. The system is automatically started if any of the following signals are generated:

- level in 2 out 6 SGs decreases below 140 mm of nominal value;
- loss of house loads power supply.

The system can also be actuated by operator.

### 2.3.8 Emergency Feedwater System

Emergency feedwater system (EFW) is categorized as safety system and designed to provide emergency feedwater to the steam generators and to maintain the SG level at the nominal value during accidents with MFW failure.

EFW system consists of two trains. The system includes the following equipment:

- two demineralized water storage tanks (BZK-1, 2);
- three emergency feedwater pumps (DAPEN-1, 2, 3);
- three demineralized water supply pumps (NBZK-1, 2, 3);
- pipelines and valves.

Demineralized water storage tanks, demineralized water supply pumps and suction collectors of Dn 350 mm are common for EFW systems of RNPP units 1 and 2.

Demineralized water from BZK-1, 2 is supplied to EFW pumps via suction collectors Dn 350 mm. DAPEN-1 and 2 have independent charging lines of Dn 100 mm which are connected to two emergency feedwater semiheaders. DAPEN-3 is connected to the charging lines of DAPEN-1 and 2. One of the emergency feedwater semiheaders is connected to SG-1, 3 and 5, and the other one is connected to SG-2, 4 and 6. Each emergency feedwater pump has its minimum flow bypass line with orifice, normally opened MOV and check valve.

During normal power operation the system is in a stand-by mode. The system is automatically initiated if any of the following signals are generated:

- level in 2 out of 6 SGs decreases below 450 mm of nominal value;
- loss of house loads power supply.

The system can also be actuated by operator.

### 2.3.9 Additional Emergency Feedwater System

Additional emergency feedwater system (AEFWS) is installed at Rivne NPP and intended for maintaining required SG level in order to ensure reactor core decay heat removal thus avoiding inacceptable fuel elements overheating and damage, to provide reactor cool-down and safe transfer to the cold shutdown state in the case of failure of main and emergency feed water systems.

System is common for RNPP Units 1 and 2 and consists of two autonomous subsystems. Each subsystem is dedicated to one of the units and includes:

- one diesel-driven pump RR91(92)D01 with electric generator for powering AEFWS components;
- one chemically demineralized water storage tank RR91(92)B01 of 1000 $m^3$ with inter-connection pipeline allowing to supply water to each of the diesel-driven pumps;
- charging pipeline with connections to other subsystems pipelines;
- diesel-driven pump support pipelines.

Additionally the redundant subsystem that can be connected to the above described subsystems is installed. Redundant subsystem consists of one diesel-driven pump RR93D01 with electric generator for powering AEFWS components, diesel-driven pump support pipelines and valves.

Inside the reactor building of each unit the system includes charging header of Dn100 which is connected by individual pipelines of Dn80 with each SG. Each line to SG has repair manual cut-off valve, control and check valves, cut-off MOV and additional check valve RR10(20-60)S05 inside containment and is connected to correspondent emergency feed water line directly upstream SG.

In normal operation mode the power supply to AEFWS equipment is provided from house loads power supply busbars of 0.4 kV. In accident conditions requiring AEFWS operation the generators mounted at the same (common) shaft as diesel drives provide power supply.

## 2.4 Reactor Control and Protection Systems

Reactor control and protection system consists of the following components (subsystems):

- reactor power controller ARM-5C;
- reactor power limiter ROM-2C;
- reactor protection system;
- system of control assemblies drives;
- system of control assemblies position monitoring;
- neutron flux measurement system.

Reactor power controller ARM-5C is designed to maintain the reactor power at the prescribed level and to maintain correspondence between the reactor and turbine power.

ARM-5C consists of three identical trains. Each train includes two controllers:

- neutron flux controller;
- secondary pressure controller.

Except the controllers mentioned above, each train includes several blocks (e.g., alarm block) common for this particular train. The control signal of the system is generated using “2 out of 3” actuation logic.

Reactor power controller can operate in the following modes:

- MSH pressure maintenance mode (“T” mode);
- reactor neutron flux maintenance mode (“N” mode).

During normal power operation ARM-5C is in neutron flux maintenance mode. ARM-5C automatically switches to “T” mode if MSH pressure increases to more than 2.5 bar above the nominal value. Automatic return to “N” mode is performed on the following signals:

- reactor power is greater than 102% of nominal value;
- actuation of level 3 reactor protection.

Reactor power limiter ROM-2C is designed to reduce reactor power depending on the number of MCPs in operation. ROM-2C consists of three identical trains. Each train includes one temperature measurement train, two neutron flux measurement trains and logic voting scheme.

The principle of ROM-2C operation is based on comparing of reactor neutron and heat power values. Neutron power, in turn, is compared with the setpoint which is constant for the given number of operating MCPs and decreases after MCP trip.

During normal power operation the system is in a stand-by mode. ROM-2C generates control signal to level 3 reactor protection scheme following abrupt increase of neutron power, trip of one or more MCPs, or turbine load-shedding.

Reactor protection system is designed to prevent plant operation parameters from exceeding safety limits in order to preclude unacceptable plant damage and release of radioactive materials to the environment. Depending on the consequences, the reactor protection system responds according to one of the following four levels:

- Level 1 Reactor Protection (AZ-1 or scram) induces the drop of all groups of control assemblies simultaneously with the maximal speed (20…30 cm/s) for fast, complete, and irreversible shutdown of the reactor;
- Level 2 Reactor Protection (AZ-2) induces the sequential insertion of control assemblies groups into the core with the maximal speed (20…30 cm/s) until while the protection signal exists. If the protection signal is terminated, the control assemblies insertion ceases and no further reduction of power takes place;
- Level 3 Reactor Protection (AZ-3) induces the sequential insertion of control assemblies groups into the core with the nominal speed (2…3 cm/s) while the protection signal exists;
- Level 4 Reactor Protection (AZ-4) inhibits control assemblies withdrawal from the core, so as to preclude further increase of reactor power.

# 3 DESCRIPTION OF VVER-440/V213 MODEL FOR TRACE CODE

## 3.1 Reactor Model

Nodalization diagram of WWER-440 reactor model for TRACE code is shown on Fig.13. The hydraulic model consists of the following main components:

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

The upper part of the downcomer at the level of emergency core flooding system baffles is divided into two sectors and represented in the model by hydraulic elements (HE) 20 and 30. HE 20 is connected to the model of cold legs and HE 30 – with ECFS injection pipelines. The upper and lower volumes of these elements model the regions above and below the cold leg nozzles, respectively. The height of the 2$^{nd}$ volume is equal to the inner diameter of the cold leg.

HE 40 models the annular gap of the downcomer between the reactor vessel and core barrel from the lower edge of ECFS baffles down to elliptical reactor pressure vessel (RPV) bottom which is represented by HE 50.

Hydraulic elements 52 and 56 represent the lower plenum portion inside the core barrel up to the lower unheated portion of fixed fuel assemblies (FA).

The core region occupied by fixed fuel assemblies and fuel sections of the control assemblies is simulated by HE 60, 62 and 64, where:

- HE 60 is the hot fuel assembly;
- HE 62 is the inner part of the reactor core (approx. 2/3 of fuel assemblies, including fuel portions of 19 control assemblies);
- HE 64 is the peripheral (outer) part of the reactor core (approx. 1/3 of fuel assemblies including fuel portions of 18 control assemblies).

Hydraulic elements 66 and 68 simulate core bypass including bypass between FA shrouds and bypass between structural elements (baffles, basket, barrel) including the dummy assemblies.

HE 70, 72, 80 and 82 represent the part of the upper plenum between the upper edge of the core basket and the hot leg nozzles. HE 70, 72 and HE 80, 82 correspond to the inner and peripheral parts of the core, respectively.

The top portion of the upper plenum (above the hot leg nozzles up to protection tubes top edge) is simulated by HE 75, 86, and the upper head volume is represented by HE 88.

HE 96 simulates annular gap between core barrel and RPV in the cold leg nozzles region. Coolant flow through protection tubes is modeled by HE 77.

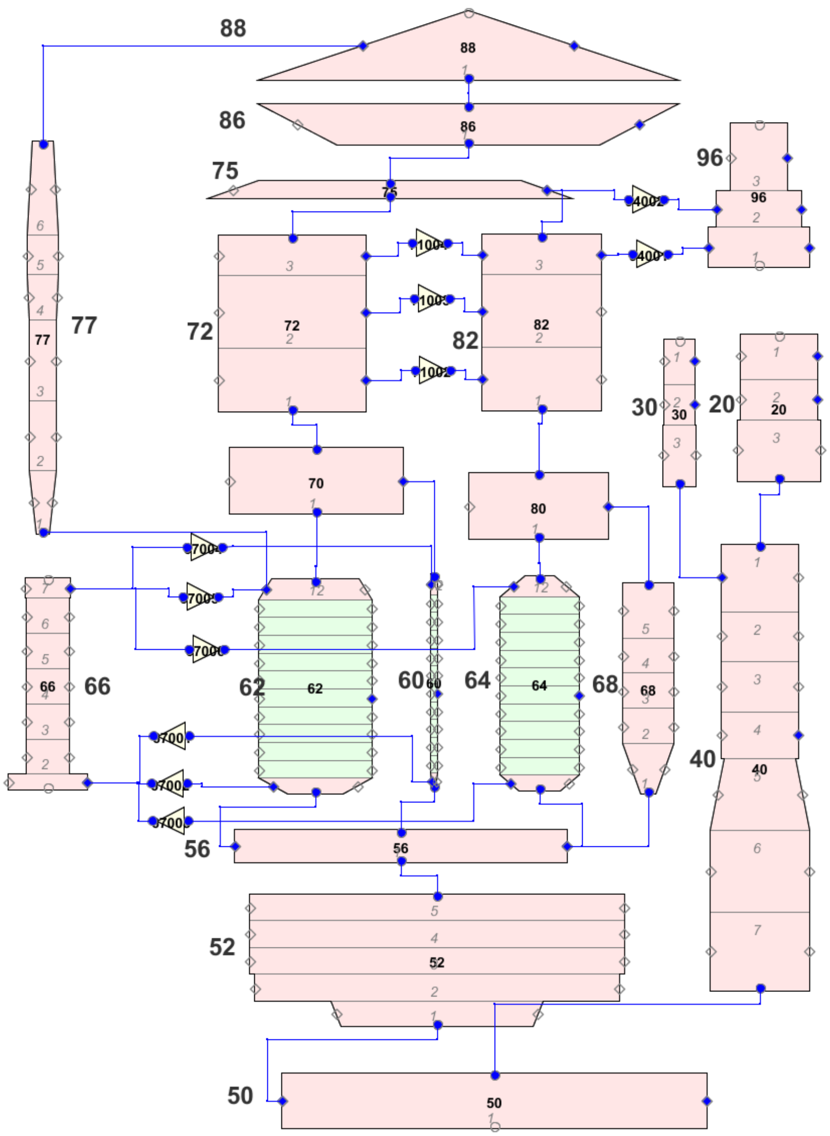



**Figure 13 Nodalization Diagram of WWER-440 Reactor**

Heat transfer from the reactor fuel to the primary circuit coolant, between reactor internals and the coolant, as well as heat losses from RPV are modeled using correspondent heat structures.

## 3.2 Main Circulation Pipelines Model

Six loops of VVER-440 reactor coolant system are represented by 4 model loops: loops 1, 3, 5 are simulated by 3 "single" loops, and loops 2, 4, 6 are simulated by "triple" model loop. Subdivision to hydraulic elements is similar for all model loops. Nodalization diagram of RCS loop 1 is shown on Fig. 14.

Hot leg model of loop 1 includes reactor exit nozzle and RCS pipeline portion (HE 100, 101, 103) up to the connection with hot SG collector (HE 110 on Fig. 14) including the main gate valve (HE 102 on Fig. 14). The cold leg model represents RCS pipeline portion (HE 140, 142, 144, 145) from cold SG manifold (HE 130) to the reactor inlet nozzles (including the inner volume of nozzles) and includes MGV (HE 143) as well as internal volume of reactor coolant pump (HE 141). Elements 1÷5 on Fig. 14 simulate double-ended guillotine break.

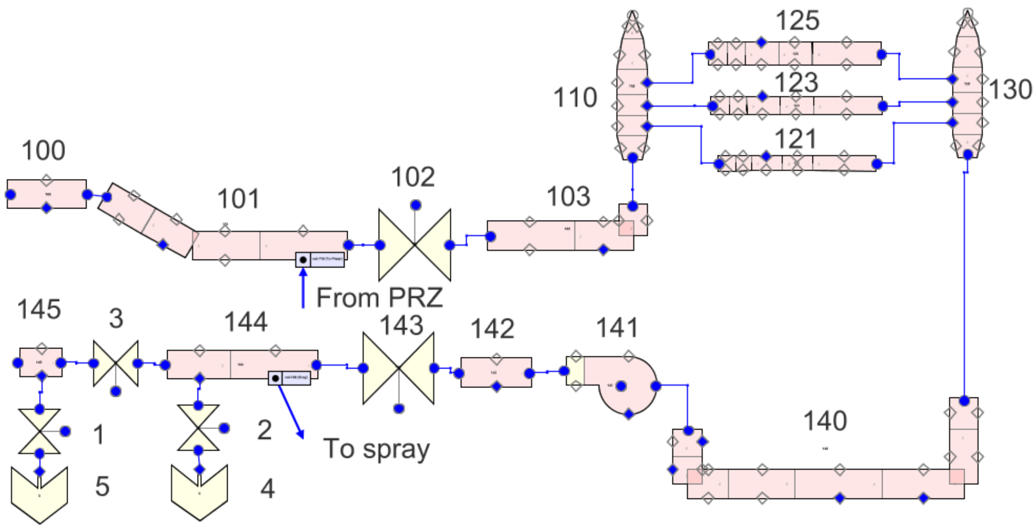



**Figure 14 RCS Loop 1 Nodalization Diagram**

To simulate heat losses from the RCS pipelines correspondent heat structures are introduced in the model.

## 3.3 Steam Generators Model

SG model includes:

- SG cold and hot collectors;
- tubes bundle;
- SG vessel;
- main and emergency feedwater nozzles;
- steam separator.

According to RCS loops modeling, six SGs are represented in the model by 3 "single" SGs and one "triple" SG. SG model nodalization diagram is shown on Fig. 15.

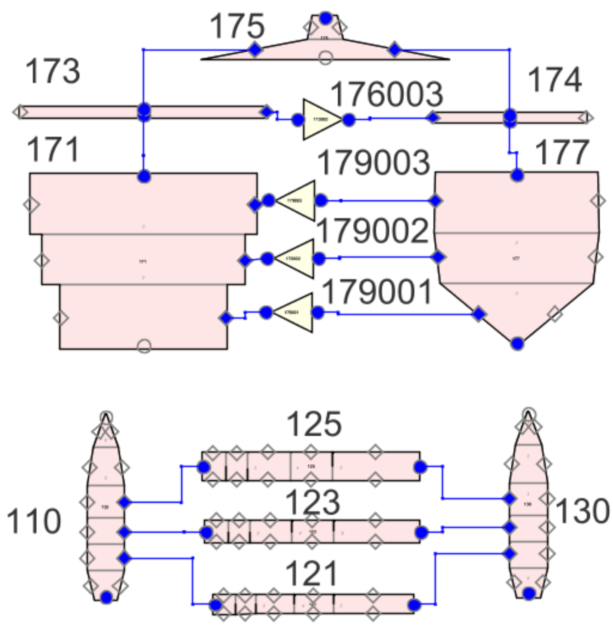



**Figure 15 SG-1 Nodalization Diagram**

HE 110 and 130 simulate hot and cold SG collectors, respectively. The tube bundle of each SG (consisting of 5536 heat exchange tubes) is subdivided into three vertical layers:

- bottom layer of 24 lower rows with 1270 SG tubes;
- middle layer of 24 rows with 1722 SG tubes;
- upper later of 29 rows with 2544 SG tubes.

Each SG tube layer consist of 5 volumes with the length ratio of 1:1:2:2:4 (from hot to cold SG collectors) in order to obtain nearly the same coolant temperature difference between each of the volumes.

At the secondary side the tube bundle region is simulated by HE 171 and 177, where the first one represents free area between SG tube bundle sections (non-heated portion), and HE 177 represents the coolant flow inside SG tube bundles sections (heated portion). This modeling approach allows to simulate circulation of the secondary coolant inside SG. Nodalization of SG secondary side is selected according to SG primary side subdivision into SG layers (e.g., height of HE 171, 177 volumes correspond to height of SG tube bundle layers).

HE 173,174 and 175 represent the secondary side volume above SG tubes bundle. During normal power operation SG level is located inside HE 173, 174, while HE 175 corresponds to the steam volume of SG secondary side.

## 3.4 Primary Pressure Control System Model

The model of primary pressure control system include:

- pressurizer (PRZ) vessel with heaters;

- surge pipeline;
- spray pipeline with PRZ spray valves.

Nodalization diagram of the system is shown on Fig. 16. The valves (PORVs and MOVs) of RCS overpressure protection system (HE 781-783) are also shown on this diagram.

PRZ is modeled by hydraulic element 700 of PRIZER type and is subdivided in 14 volumes. The first and the last of these volumes represent PRZ bottom and head, respectively, while the others simulate cylindrical part of PRZ vessel.

Two surge pipelines are represented by "double" model surge pipe (HE 710). HE 715 is the vertical part of tee-junction connecting surge line with PRZ bottom.

The spray pipeline is modeled by HE 720 and 726, and spray valves are simulated with HE 722, 723, 724, 725. Orifice at the minimal flow spray line is represented with HE 721.

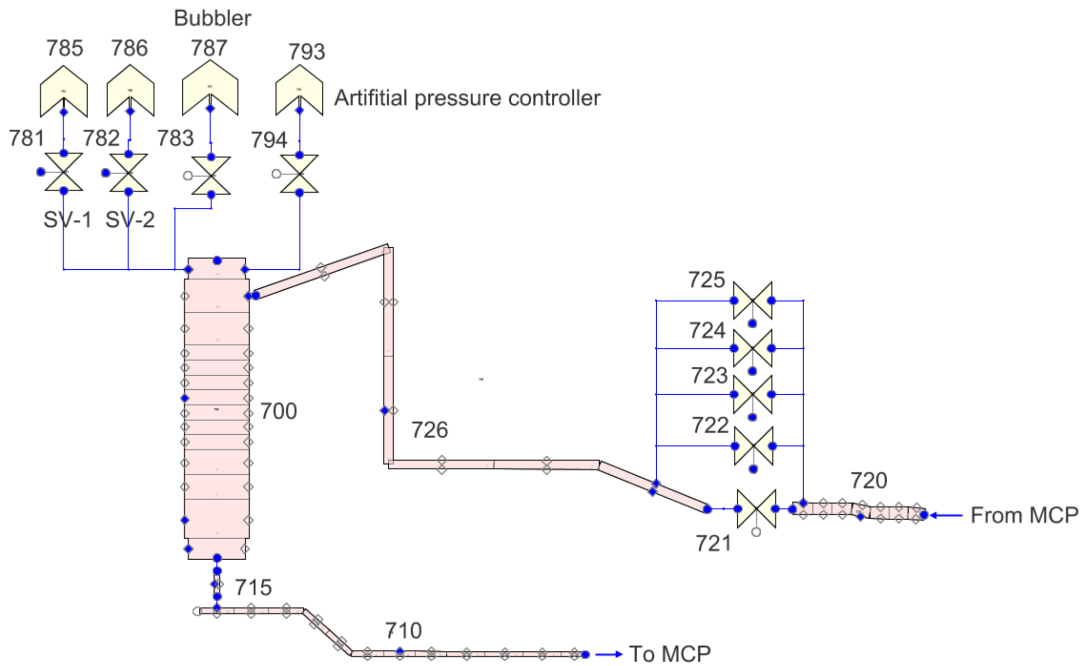



**Figure 16 Primary Pressure Control System Nodalization Diagram**

## 3.5 <u>Emergency Core Flooding System</u>

ECFS consisting of 4 hydroaccumulators and correspondent pipelines. Nodalization diagram of ECFS is shown on Fig. 17.

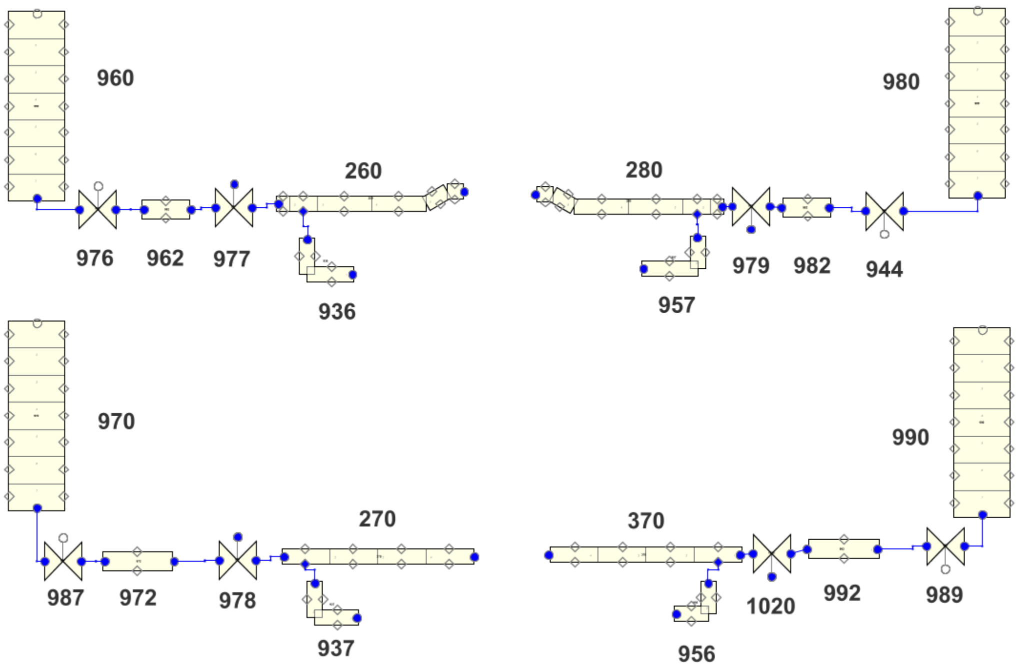



**Figure 17	ECFS Nodalization Diagram**

HE 960 and 980 representing HA-1 and HA-4, respectively, are connected with surge lines (HE 962, 260 and 982, 280) to the reactor upper plenum. HE 970 and 990 modeling HA-2 and HA-3, respectively, are connected with the surge lines (HE 972, 270 and 992, 370) and the downcomer.

## 3.6 High Pressure Injection System

All three trains of high pressure injection system are represented in the model.

HE 901 and 903 simulate HPIS and LPIS tanks, respectively. Each HPIS pump is modeled by two HE (the first one simulates injection from HPIS tank, while the other one allows to provide injection from LPIS tank model): HE 911 and 914 for HPIS-1, HE 912 and 915 for HPIS-2, HE 913 and 916 for HPIS-3. The injection lines are modeled by HE 921, 922, 923. HE 120 simulates connection of the 1$^{st}$ HPIS train to the cold leg of 2$^{nd}$ RCS loop and HE 160, 170 connect 2$^{nd}$ and 3$^{rd}$ HPIS trains to the cold leg of 4$^{th}$ (triple) RCS loop.

Nodalization diagram of HPIS is shown on Fig. 18.

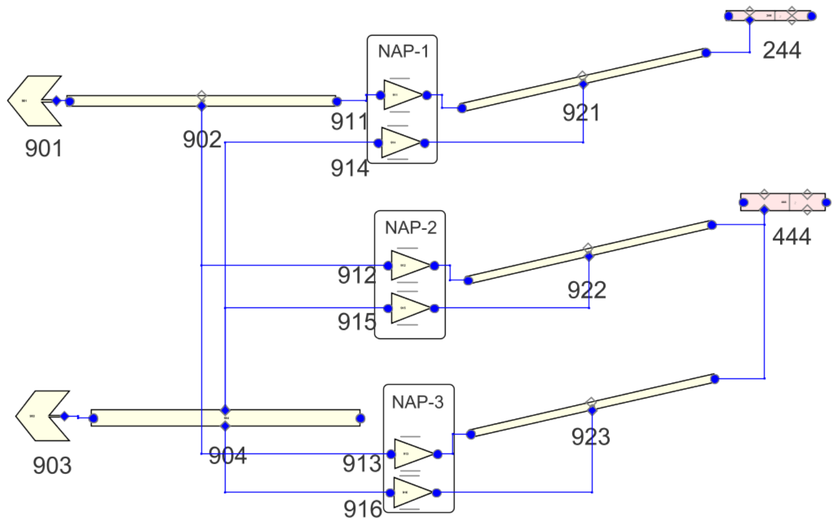



**Figure 18 HPIS Nodalization Diagram**

## 3.7 Low Pressure Injection System

LPIS model includes all three system trains. HE 903 represent three LPIS tanks. LPIS pumps are modeled by HE 917, 918, 919. Correspondent injection lines are simulated with HE 931, 941 and 951. Connection of the 1st and 3rd LPIS trains injection lines (HE 931 and 951) to HA surge lines is performed with HE 936,937 and HE 956, 957. Injection line of the second LPIS train (HE 941) is connected to the hot and cold legs of 4th RCS loop via HE 942, 943.

LPIS nodalization diagram is shown on Fig. 19.

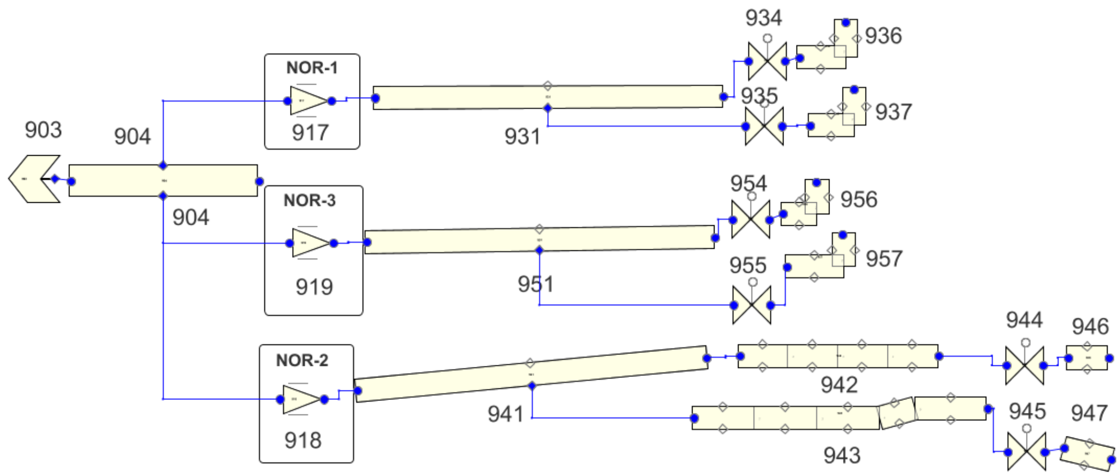



**Figure 19 LPIS Nodalization Diagram**

## 3.8 Make-Up and Let-Down

Nodalization diagram of make-up and let-down subsystems of CVCS is shown on Fig. 20.

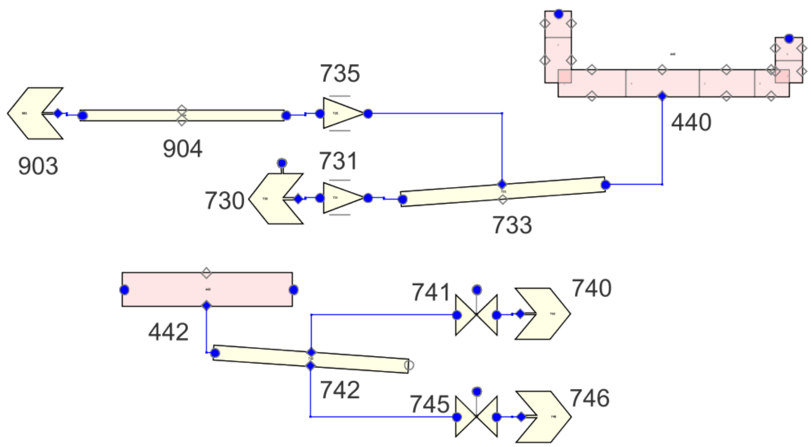



**Figure 20 Make-Up and Let-Down Nodalization Diagram**

Make-up pipeline is modeled by HE 904, 733 and 180 of the PIPE type and connected upstream MCP of the 4$^{th}$ RCS loop. Make-up flow is controlled by HE 731 and 735 simulating make-up from deaerator (HE 730) or from LPIS tanks (HE 903).

Let-down line is connected downstream MCP of the 4$^{th}$ RCS loop and is modeled by HE 712.

## 3.9 Steam Lines and the Main Steam Header

The main steam lines (MSL) and the main steam header model includes:

- four model MSLs (3 single lines and 1 triple line) from SG to the turbine stop valves;
- main steam header;
- connections between MSLs and MSH;
- turbine bypass to condenser BRU-K and their connection lines;
- steam dump valves to atmosphere BRU-A and their connection lines;
- SG steam relief valves;
- fast acting steam isolation valves;
- turbine with stop and control valves.

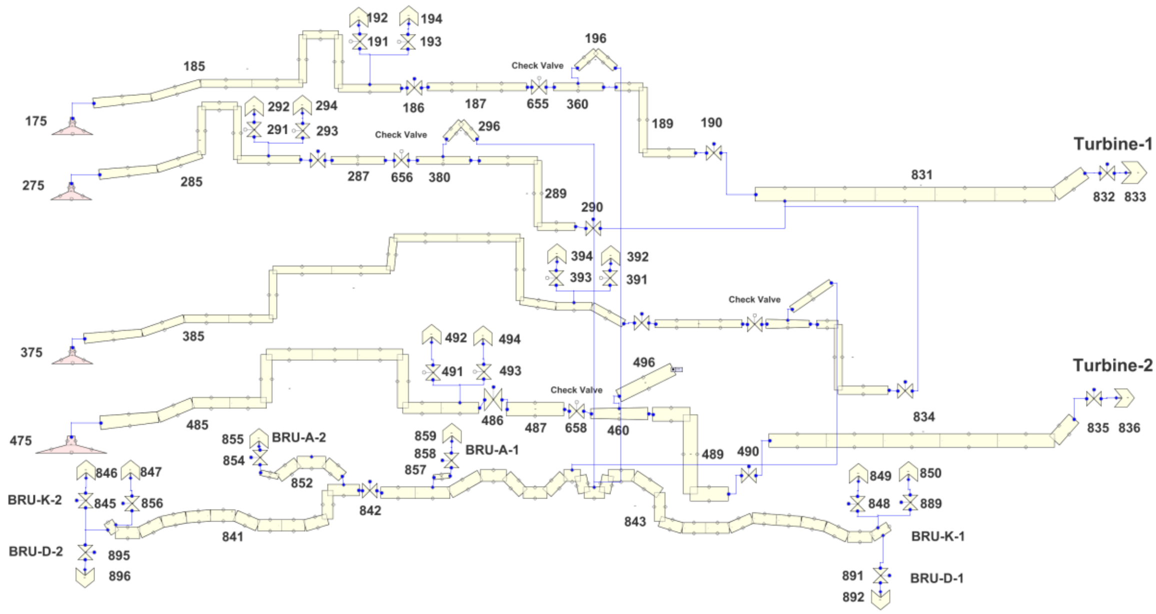



**Figure 21 MSL and MSH Nodalization Diagram**

HE 185, 285, 385, 485 simulate parts of MSLs from SGs to FASIVs (HE 186, 286, 386, 486). Next MSL sections from FASIVs to the main steam valves are modeled with HE 187, 287, 387, 487 and 189, 289, 389, 489. Hydraulic elements 196, 296, 396 and 496 connect MSLs with MSH which is subdivided by FASIVs (HE 842) into two semi-headers. HE 831 and 834 represent steam lines sections from the main steam valves to the turbine stop valves. HE 831 is the triple model line connecting MSLs from SG-1, 3, 5 with turbine No. 1, and HE 834 is the triple line connecting MSLs from SG-2, 4, 6 with turbine No. 2.

HE 832 and 835 represent turbine stop and control valves of turbines No.1 and No.2. Turbines are simplistically simulated by HE 833 and 836 with predefined boundary conditions.

SG SRVs are simulated by HE (1–4)91 and (1–4)93.

MSH semi-headers are represented by HE 841, 843, 852 and connected to BRU-K (HE 845 and 848) and BRU-A models (HE 854 and 858).

## 3.10 Main and Auxiliary Feed Water Systems

The main feed water and auxiliary feed water systems nodalization is shown on Fig. 22.

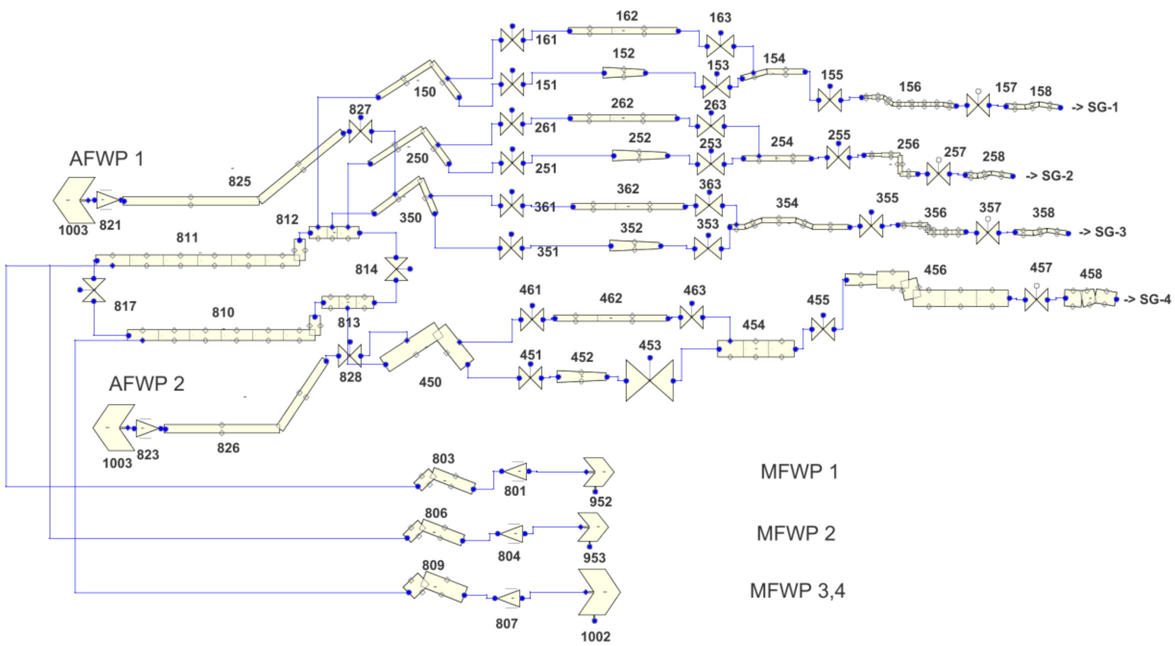



**Figure 22 MFW and AFW Nodalization Diagram**

MFW pipelines from the secondary side deaerators to MFW collector (HE 810, 811, 812, 813) are simulated by three pipes (HE 803, 808, 809). The pipelines from MFW collector to correspondent SGs are represented by 4 model pipelines (three single and one triple pipeline) consisting of HE (1-4)50, 52, 62, 54, 56 and (1-4)58. HE (1-4)51, 61 and (1-4)53, 63 represent cut-off and control MFW valves, and HE (1-4)57 simulate the check valves at MFW lines.

HE 825 and 826 which are shown on Fig.22 model the injection lines from auxiliary feed water system.

## 3.11 Emergency Feed Water System

Emergency feedwater system pumps are modeled with HE 861, 863, 865. EFW charging pipelines from EFW pumps No.2, 3 up to horizontal sections at elevation of +15.0 m are modeled by HE 867 and 868, respectively. HE 869 represents charging line section from EFW pump No. 1 up to connection with EFW pumps No. 2, 3 charging line.

Horizontal sections of EFW pipelines at elevation of +15.0 m up are represented by HE 876 and 877 which consist of 3 volumes each. EFW flow control valves are modeled by HE 871, 872, 873, and HE 878, 879 model the check valves.

EFW pipelines to individual SGs are represented with HE (1-4)64,65. EFW SG level controllers and check valves which are installed at these pipelines are modeled with HE (1-4)67 and (1-4)66, respectively.

Nodalization diagram of EFW system is shown on Fig. 23.

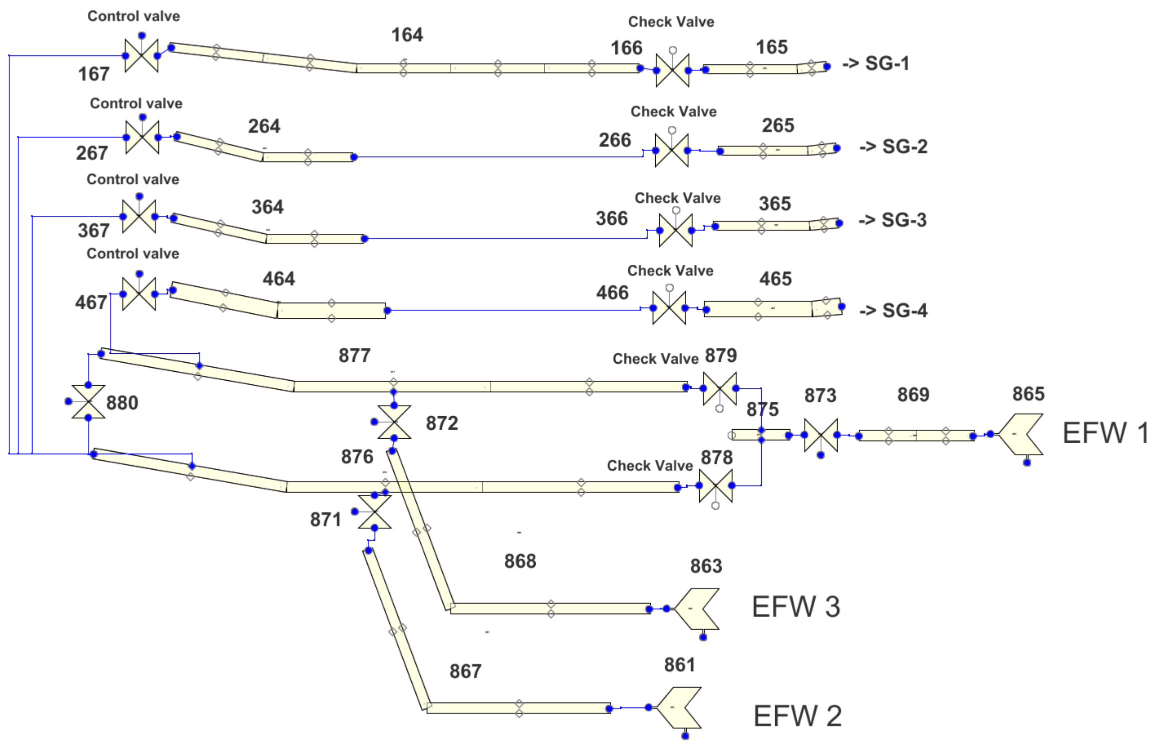



**Figure 23 EFW System Nodalization Diagram**

## 3.12 Main Control and Protection Systems

Main control and protection systems simulated in TRACE model for WWER-440/V213 include:

- reactor protection system (AZ-1, AZ-2, AZ-3, AZ-4);
- automatic start of safety systems logic;
- ECCS (HPIS and LPIS) safeguards;
- PRZ heaters, spray valves and PRZ PORV operation logic;
- MCP protection safeguards;
- MGV closure signals;
- MFW, AFW and EFW systems operation logic (including safeguards and controllers);
- SG SRVs and steam dump valves (BRU-K, BRU-A) operation logic (including BRU-K and BRU-A controllers);
- FASIVs and turbine stop valves closure logic;
- make-up and let-down controllers.

# 4 CALCULATION OF TRANSIENTS

The following RNPP-1 incidents were selected for validation calculations:

- reactor scram caused by concrete slab drop to the connection lines of house loads power supply transformer;
- reactor scram transient initiated by 6 kV switch short circuit;
- inadvertent reactor scram.

## 4.1 Reactor Scram Caused by Concrete Slab Drop to the Connection Lines of House Loads Power Supply

According to [3] this incident was caused by mechanical damage of the house loads power supply line A that resulted in disconnection of house loads power supply transformers from correspondent busbars of 6kV, connection of standby transformer to energize these busbars and automatic restart of connected equipment. During equipment restart a trip of condenser cooling water pumps and MFW pumps #1,5 trip occurred (caused by overload protection) with closure of turbine #1 (TG-1) stop valves, while turbine #2 (TG-2) continued to operate in no-load mode. Further automatic equipment restart was only successful for one of two condenser cooling pumps of TG-2. Steam produced in SG was dumped via turbine bypass to condenser (BRU-Ks in load shedding mode) until TG-2 condenser vacuum was lost due to failure of circulation pumps restart and TG-2 stop valves were closed. Trip of the last operating turbine initiated reactor scram signal actuation.

The analysis of available NPP data shows that sharp decrease of secondary circuit pressure from 50 to 35 kgf/cm$^2$ and significant steam flow rate from SG (70-80 t/h) was observed from 130 to 370 sec. This may be explained by a failure of one BRU-K to close after beginning of steam dump to the condensers with subsequent forced closure of failed BRU-K by operators.

The transient may be divided into three phases:

- phase 1 (from 0 to 123 sec) is characterized by TG-1 stop valves closure, switching of TG-2 to idle run, secondary pressure increase and start of BRU-Ks operation, reactor power decrease by the reactor power controller ARM in secondary circuit pressure maintenance mode;
- phase 2 (123-370 sec) is characterized by closure of TG-2 stop valves and stop of BRUK-2A,B operation followed by reactor scram, failure of BRUK-1B to close, secondary pressure decrease (down to 35 kgf/cm$^2$), primary pressure and temperature decrease, as well as PRZ level decrease;
- phase 3 (after 370 sec) is characterized by closure of BRUK-1B and stabilization of primary and secondary parameters.

Based on the description of actual transient and specifics of developed VVER-440 model for TRACE code the following calculation scenario was developed.

At the beginning of transient simulation (0 sec), TG-2 stop valves closure is modeled, at the same time TG-1 is switched to no-load operation mode. Simultaneously, two operating main feedwater pumps MFWP-1 and MFWP-2 are switched off and at the 4$^{th}$ second of the problem time MFWP-2 is switched on (trip of two operating MFWP and automatic start of stand-by MFWP according to the actual timing of events).

At 123 sec of problem time, TG-1 stop valves and BRU-K2 (double) are closed.

According to the scenario, it is postulated that BRUK-1B fails to close and remains in open position until it is closed by the operator at 370 sec. Also the operator actions on transferring SG feed from the main feedwater supply to auxiliary feedwater supply through startup/shutdown feedwater control valves is modeled following the decrease of main feedwater massflow rate (with predefined delay).

Operation of other main unit equipment complies with the design characteristics and set points. The overall transient calculation time is 1000 sec.

### 4.1.1 Initial Conditions

The nominal values of the main primary and secondary circuit parameters are selected in accordance with the design values prescribed for normal power operation in Rivne NPP Unit 1 Technical Safety Justification [1] and Technical Specification [2].

Before the transient the unit was in nominal power operation for about 10 months since the beginning of the 18th fuel cycle. In validation calculation the reactor core characteristics which correspond to the end of the 22th fuel cycle [4] were applied. The decay heat is calculated based on ANS-79 standard [5] with 1.0 multiplier.

Table 2 presents the measured and calculated values of the main primary and secondary circuit parameters before the transient.

**Table 2 Initial Conditions for Validation Calculation**

| Parameter | Units | Nominal value | Calculated value |
|---|---|---|---|
| Reactor thermal power | MW | 1375±27 | 1375 |
| Fuel rod maximal linear heat flux | W/cm | 325.1 | 289 |
| Reactor outlet pressure (gauge) | kgf/cm$^2$ | 125±1.2 | 124.1 |
| Coolant temperature at core inlet | °C | ≤267 | 267 |
| Coolant temperature at core outlet | °C | 297 | 297 |
| Reactor coolant heat-up | °C | 30 | 30 |
| Maximal cladding temperature | °C | 335 | 326.4 |
| Reactor coolant volumetric flow | m$^3$/h | 40700±400 | 40600 |
| PRZ level | m | 5.96±0.1 | 5.97 |
| SG pressure (gauge) | kgf/cm$^2$ | 46±0.5 | 45.2 – 45.7 |
| SG water level | m | 2.12±0.05 | 2.08-2.11 |
| SG steam production | t/h | 450 | 444 – 467 |
| Main feedwater temperature | °C | 223 | 223 |

### 4.1.2 Boundary Conditions

The following assumptions on normal and safety systems operation are applied:

- scram is actuated by the first initiating signal according to the design set points;
- primary makeup system operates according to the design logic;
- primary pressure control system operates in compliance with the design set points;
- TG trip is simulated as described in the incident report [3] (one of the turbines is tripped at the beginning of transient and the second turbine is tripped at 123 s);
- operating MFW pumps are tripped at the beginning of transient and stand-by MFWP starts 4 seconds later in compliance with the incident description. Feedwater flow rate is controlled by the main feedwater controllers;
- BRU-Ks operate according to design set points. Closure of two of four BRU-Ks is modelled simultaneously with turbine trip at 123 s. One of the remaining BRU-Ks is assumed to stuck in open position at 140 s and is closed (by operator) at 370 s.

### 4.1.3 Calculation Results

Table 3 provides comparison of calculated and actual timing of events occurred in the course of the incident.

**Table 3 Sequence of Events**

| Time, sec | | Event | Note |
|---|---|---|---|
| Incident | Calculation | | |
| 0 | 0 | TG-2 stop valves closure | Boundary condition |
| | | TG-1 controller switches to the no-load mode | |
| | | MFWP-1 trip | |
| | | MFWP-2 trip | |
| 0.2 | 0.4 | BRU-K2 controller switching to the load shedding mode | Decrease of TG-2 load for more than 20% |
| 0.4 | 0.6 | BRU-K1 controller switching to the load shedding mode | Decrease of TG-1 load for more than 20% |
| 2.5 | 2.8 | ARM interlock to switch to the secondary circuit pressure maintenance mode | MSH pressure increase for 2.5 kgf/cm$^2$ |
| 2.54 | 2.88 | Control assemblies insertion into the core with normal operating speed | ARM signal |
| 4.2 | 5.5 | Signal for opening of BRU-K-1A, BRU-K-1B | |
| 4,45 | 4,45 | Signal for opening of BRU-K-2 (double) | According to controller operation logic |
| 7.1 | 8.5 | PRZ heaters group 1 is OFF | PRZ pressure increase above 126 kgf/cm$^2$ |
| 9.6 | 11.0 | Pressurizer spray valve 1 open signal | PRZ pressure increase above 128 kgf/cm$^2$ |

| Time, sec | | Event | Note |
|---|---|---|---|
| Incident | Calculation | | |
| 11.1 | 12.1 | Pressurizer spray valve 2 open signal | PRZ pressure increase above 129 $kgf/cm^2$ |
| 13.3 | 14.4 | Pressurizer spray valve 3 open signal | PRZ pressure increase above 130 $kgf/cm^2$ |
| 18.6 | 19.1 | Pressurizer spray valve 3 close signal | PRZ pressure decrease down to 129 $kgf/cm^2$ |
| 20.1 | 21.5 | Pressurizer spray valve 2 close signal | PRZ pressure decrease down to 128 $kgf/cm^2$ |
| 21.5 | 22.6 | Pressurizer spray valve 1 close signal | PRZ pressure decrease down to 127 $kgf/cm^2$ |
| 25.9 | 30.1 | PRZ heaters group 1 is ON | PRZ pressure decrease down to 124 $kgf/cm^2$ |
| 28.1 | 32.0 | PRZ heaters group 2 is ON | PRZ pressure decrease down to 123 $kgf/cm^2$ |
| 32.1 | 42.0 | PRZ heaters group 3 is ON | PRZ pressure decrease down to 122 $kgf/cm^2$ |
| 46.5 | 52.0 | PRZ heaters group 4 is ON | PRZ pressure decrease down to 121 $kgf/cm^2$ |
| - | 107.0 | PRZ heaters group 5 is ON | PRZ pressure decrease down to 118 $kgf/cm^2$ |
| 67.8 | 63.0 | PRZ level controller switch to automatic mode | PRZ level decrease for 0.3 m below nominal |
| 123 | 123 | TG-1 stop valves closure | Boundary condition |
| | | BRU-K2 valve closure | |
| 123.1 | 123.1 | Scram signal | Due to a closure of stop valves of the last operating turbine |
| | | BRU-K1 load shedding mode is off | By scram signal |
| 139 | 105 | ARM power decrease prohibition interlock | Reactor outlet pressure decrease to 120 $kgf/cm^2$ |
| 140 | 140 | Failure of BRU-K-1B to close after opening | Boundary condition |
| 148 | 148 | PRZ heaters group 5 is ON | Pressure decrease below 118 $kgf/cm^2$ |
| 276 | 276 | MSH-2 pressure decrease down to 40 $kgf/cm^2$ | |
| 320 | 320 | MSH -1 pressure decrease down to 38 $kgf/cm^2$ | |
| 370 | 370 | BRU-K-1B closure (operator action) | |
| 397 | 384 | PRZ heaters group 5 is OFF | PRZ pressure increase above 121 $kgf/cm^2$ |
| 412 | 395 | PRZ heaters group 4 is OFF | PRZ pressure increase above 122 $kgf/cm^2$ |
| 425 | 412 | PRZ heaters group 3 is OFF | PRZ pressure increase above 123 $kgf/cm^2$ |
| 437 | 426 | PRZ heaters group 2 is OFF | PRZ pressure increase above 124 $kgf/cm^2$ |

| Time, sec | | Event | Note |
|---|---|---|---|
| Incident | Calculation | | |
| 465 | 450 | PRZ heaters group 1 is OFF | PRZ pressure increase above 126 kgf/cm$^2$ |
| 490 | 470 | Start of PRZ spray valve 1 operation | PRZ pressure increase above 128 kgf/cm$^2$ |
| 760 | 800 | End of PRZ spray valve 1 operation | PRZ pressure increase above 127 kgf/cm$^2$ |
| 1000 | 1000 | End of calculation | |

As it can be seen from the results' plots (see Fig. 24, Fig. 44, Fig. 51), the main calculated primary and secondary parameters' values correlate with incident measured data. Initial phase of the incident is characterized by sharp increase of secondary circuit pressure up to 51.6 kgf/cm$^2$ (see Fig. 25, Fig. 26) due to TG-2 stop valves closure and switching of TG-1 to no-load run. At the same time, BRUK-1, 2 are immediately opened (see Fig. 64, Fig. 65, Fig. 66) due to TG load shedding signal actuation.

After secondary pressure increase, the primary pressure also increases up to 131.1 kgf/cm$^2$ (in the calculation) (see Fig. 24). Calculated RCS pressure peak is slightly higher than the one in the actual incident and is explained by differences in the initial RCS parameters values.

Increase in the secondary circuit pressure causes switching of ARM to the pressure maintenance mode with insertion of control group of control assemblies into the core thus decreasing the reactor power (see Fig. 85). After TG-1 stop valves closure (by the loss of condenser vacuum signal) reactor scram is actuated and reactor power decreases to decay heat.

Decrease of RCS coolant temperature (see Fig. 33 – Fig. 44) after scram results in the decrease of calculated primary circuit pressure below 116.5 kgf/cm$^2$ (see Fig. 24) and of PRZ level below 4.30 m (see Fig. 51). The rate of pressure decrease is slowed down by sequential PRZ heater groups operation (see Table 3). Secondary circuit pressure quickly decreases to 36.2 kgf/cm$^2$ (see Fig. 25 – Fig. 32) due to failure of BRU-K-1B to close until it is closed by the operator at 370 sec. After this, cooldown via BRU-K is terminated, primary circuit pressure increases and stabilizes at 127-129 kgf/cm$^2$.

The actual incident data of reactor coolant flow are not available but qualitatively the flow rate trend may be assessed by pressure difference across MCPs (see Fig. 79 – Fig. 84). The calculated coolant flow rate in RCS loops is presented in Fig. 86.

At the third phase of accident the difference between calculated and measured SG levels data can observed (see Fig. 52 – Fig. 63): the plant data show continuous SG level increase that can be explained by leakage through the main feedwater control valves. The level increase stops only after complete forced closure of MFW valves (during documented incident period this is observed in two SGs).

Calculated feedwater temperature (Fig. 45 – Fig. 50) follows (with a delay) the reactor power and reflects changes in feedwater heaters operation.

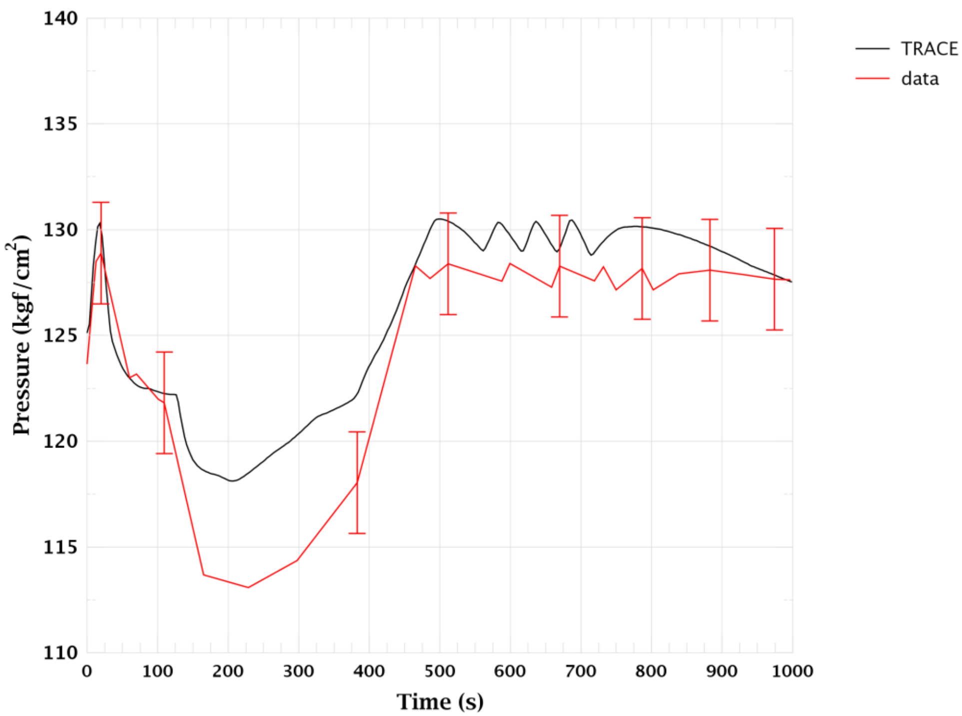



**Figure 24 Reactor Outlet Pressure**

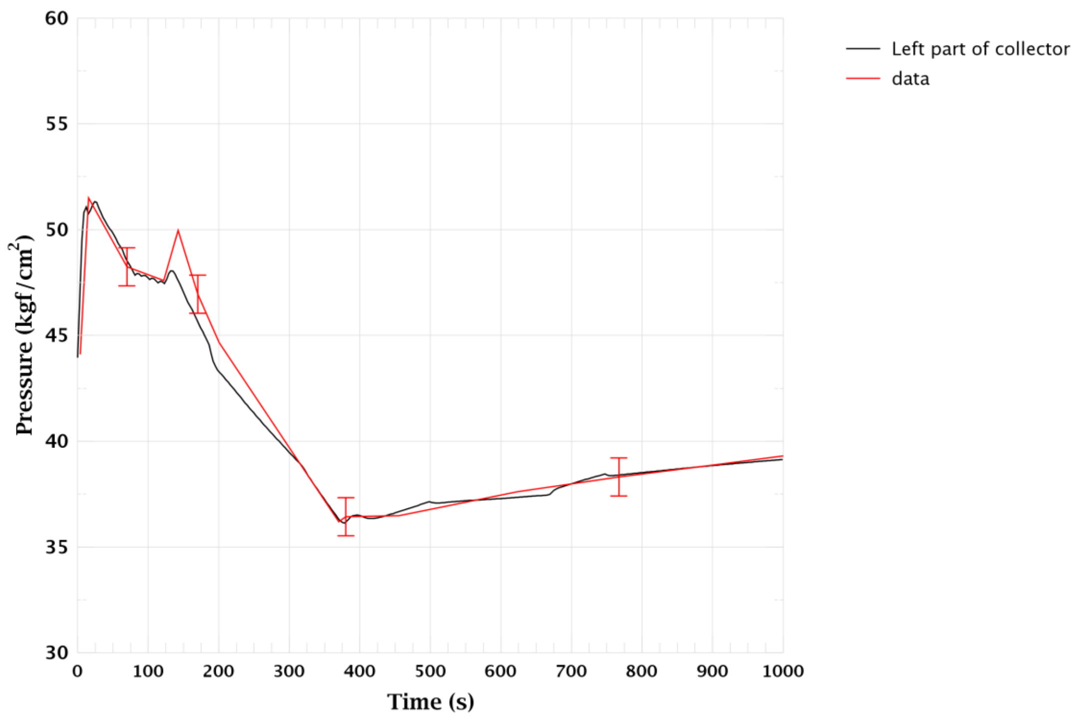



**Figure 25 Left MSH Semi-Header Pressure**

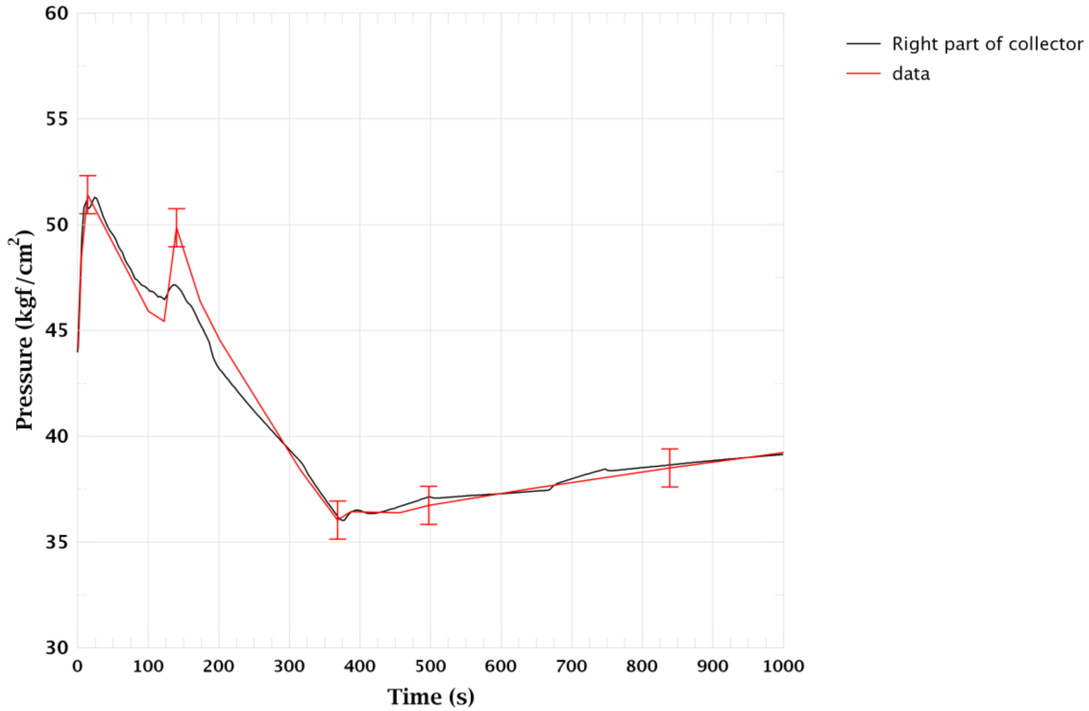



**Figure 26 Right MSH Semi-Header Pressure**

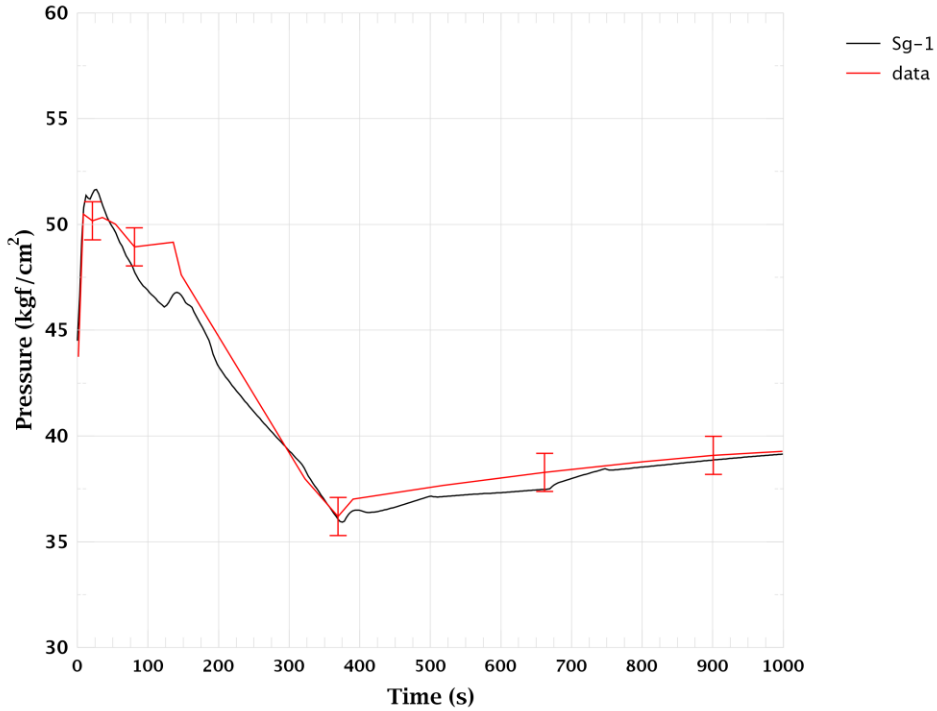



**Figure 27 SG-1 Pressure**

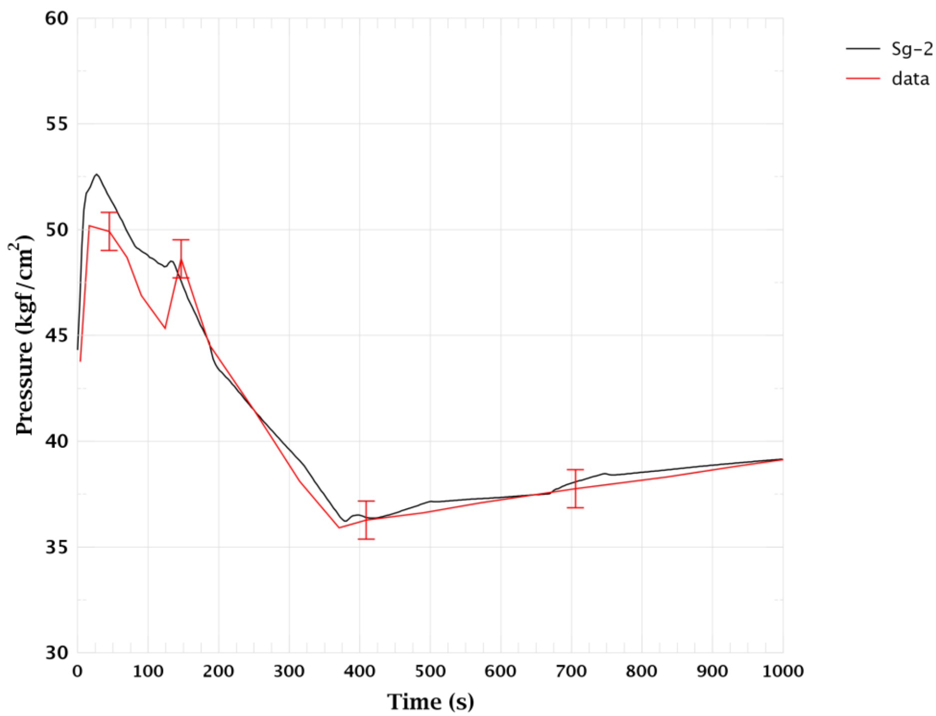



**Figure 28 SG-2 Pressure**

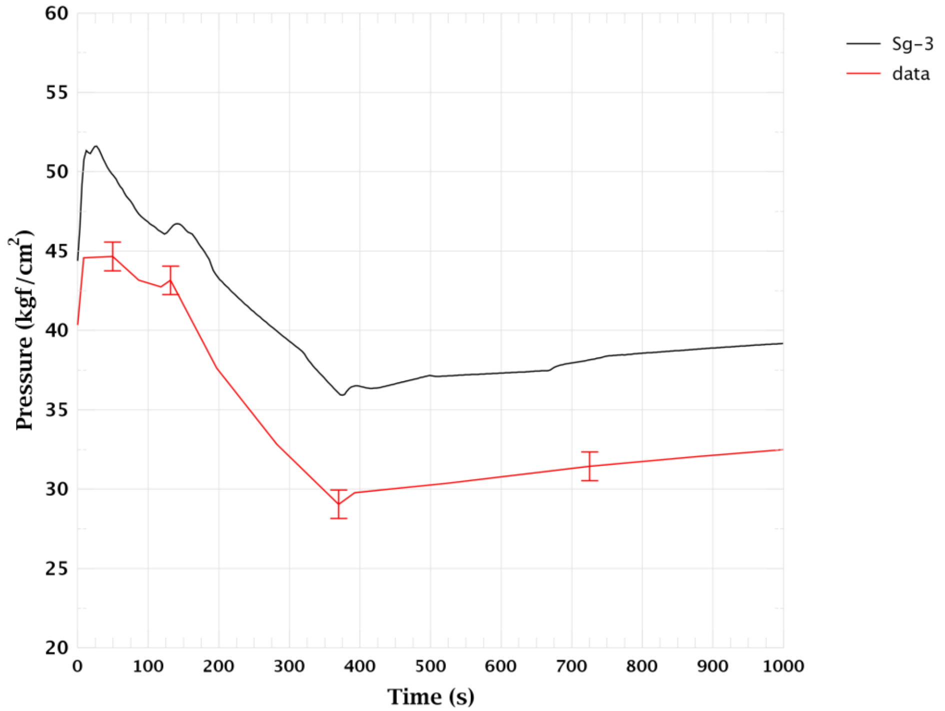



**Figure 29 SG-3 Pressure**

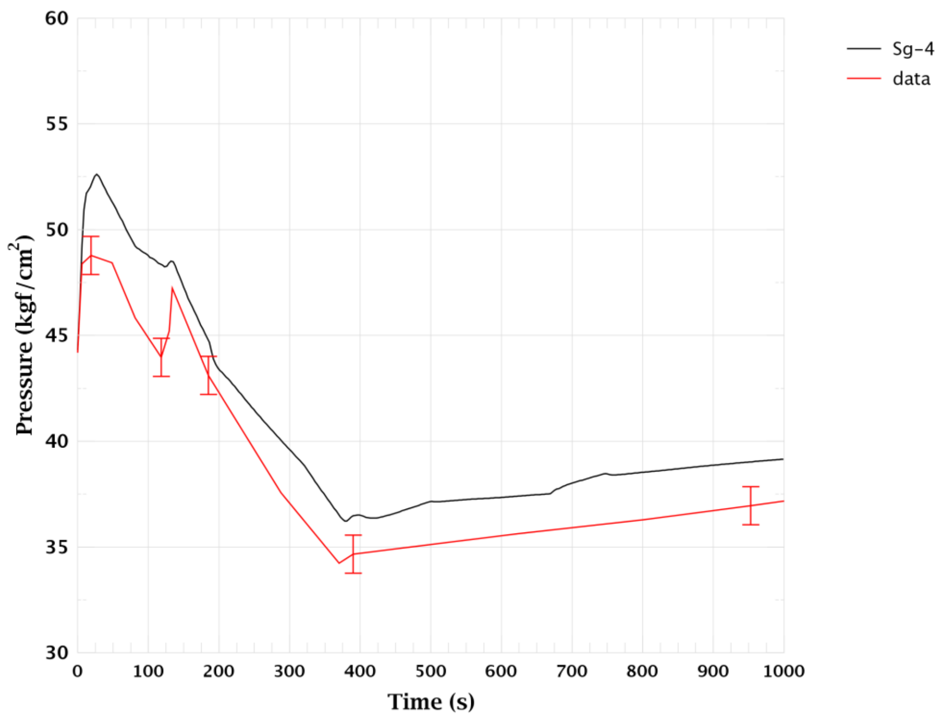



**Figure 30 SG-4 Pressure**

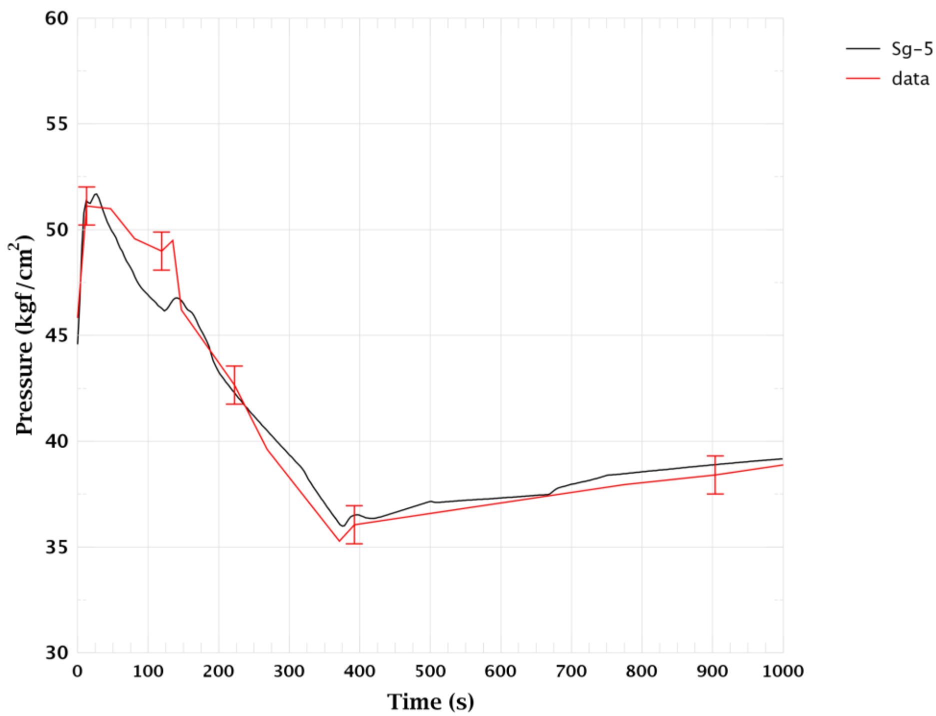



**Figure 31 SG-5 Pressure**

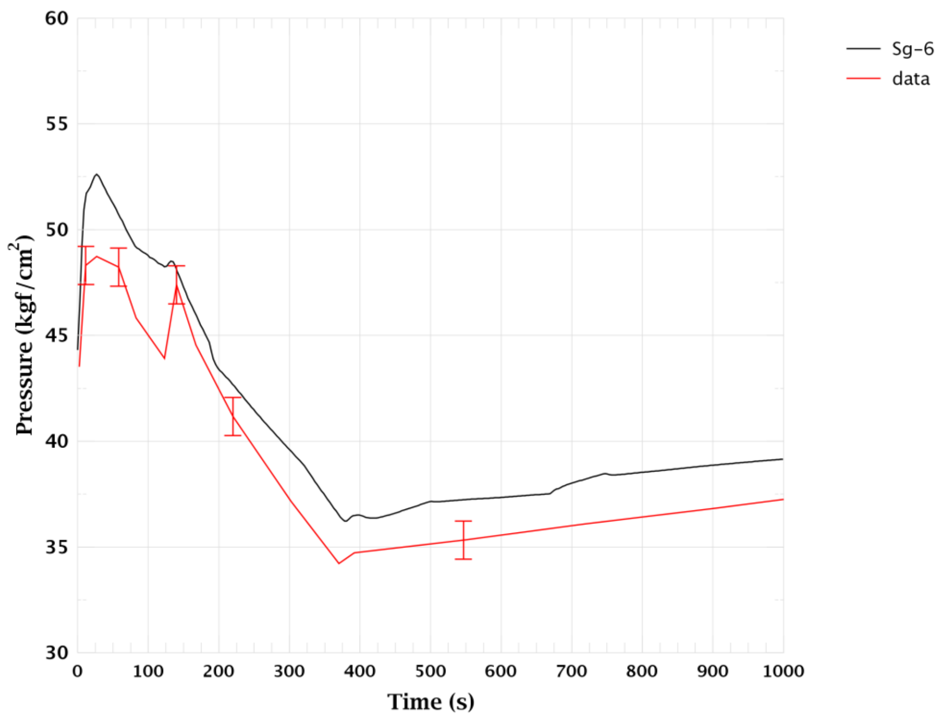



**Figure 32 SG-6 Pressure**

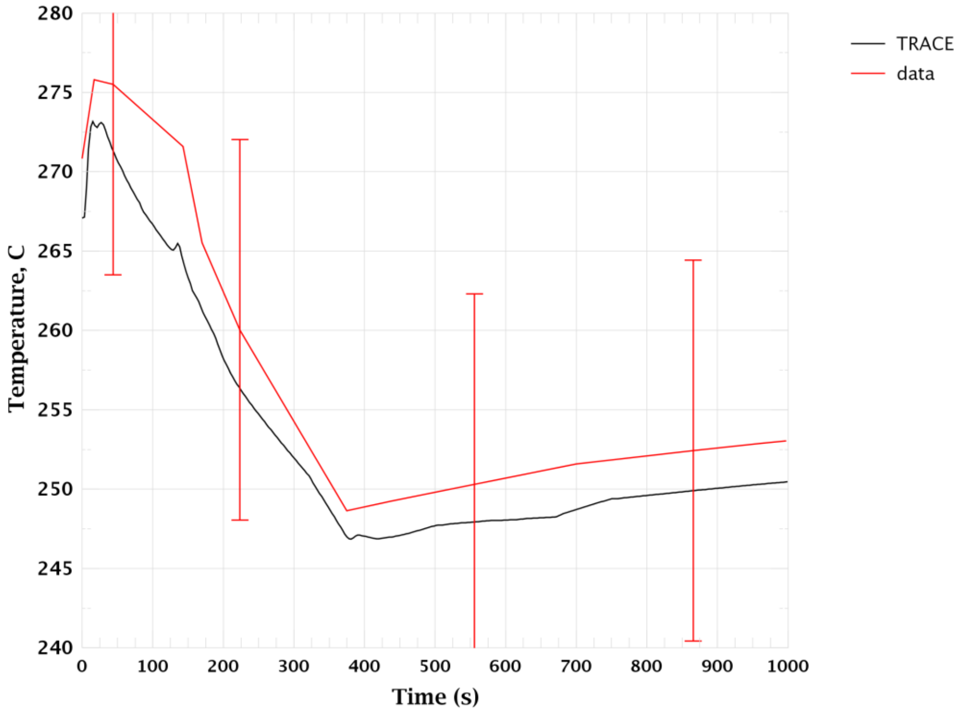



**Figure 33 Cold Leg No. 1 Temperature**

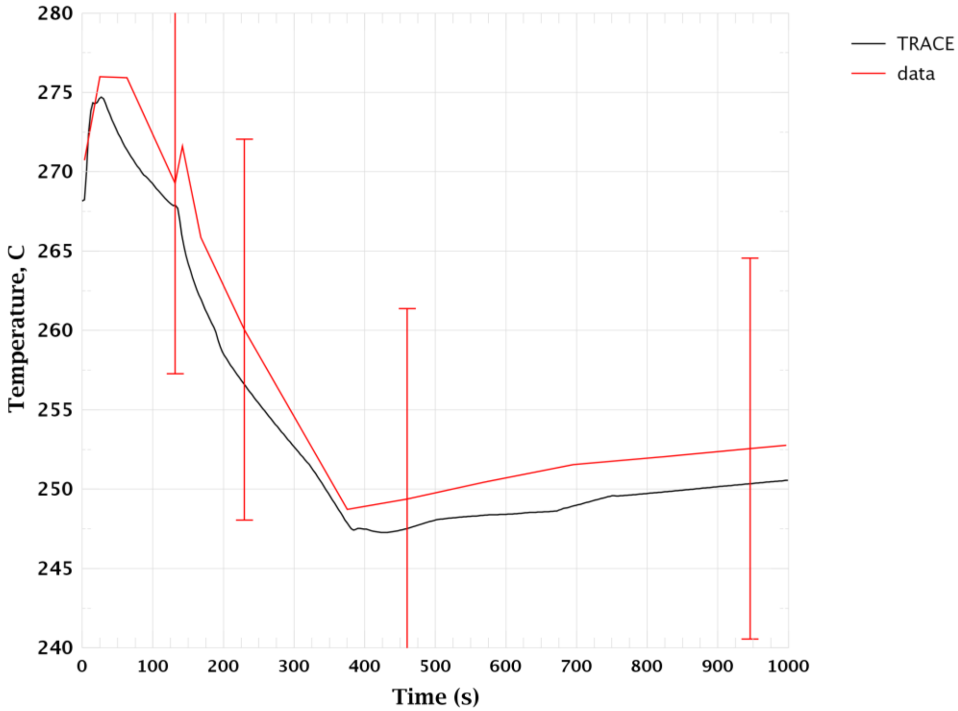



**Figure 34 Cold Leg No. 2 Temperature**

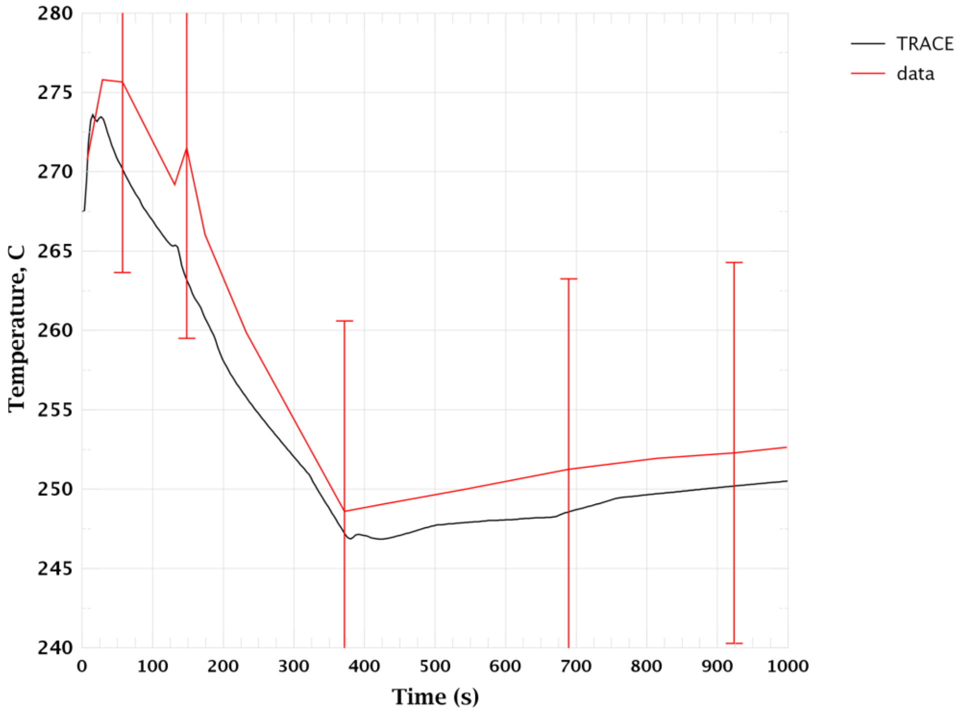



**Figure 35 Cold Leg No. 3 Temperature**

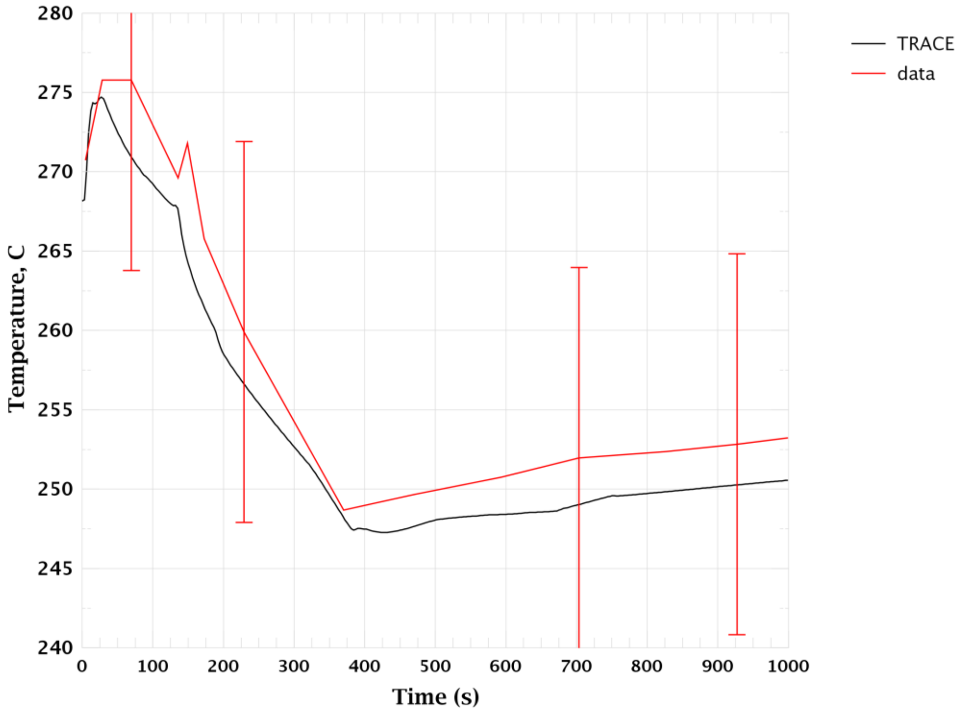



**Figure 36 Cold Leg No. 4 Temperature**

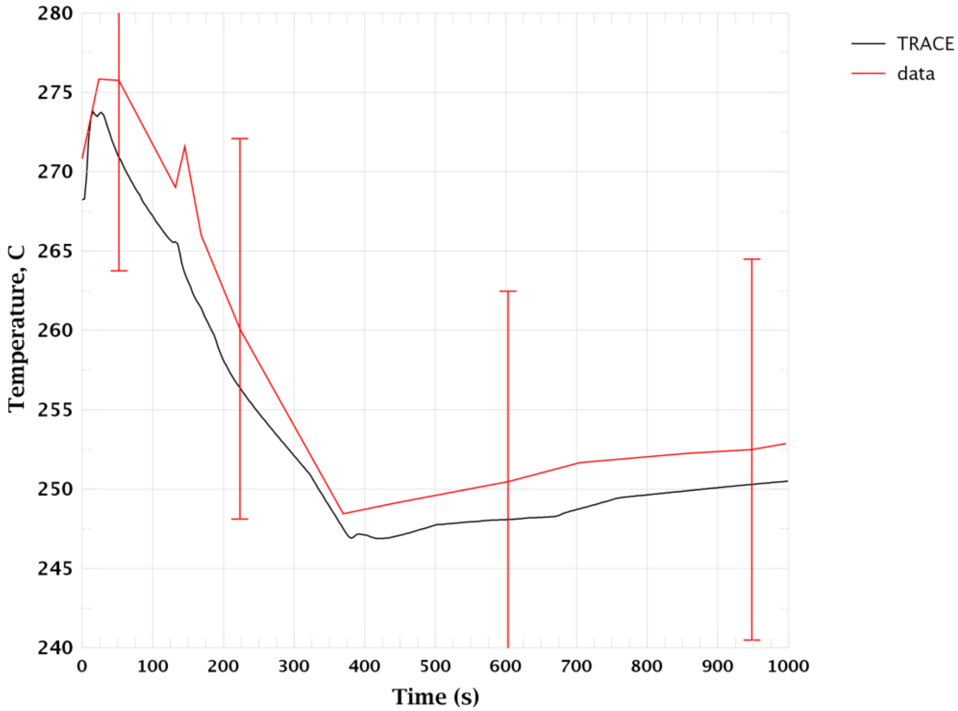



**Figure 37 Cold Leg No. 5 Temperature**

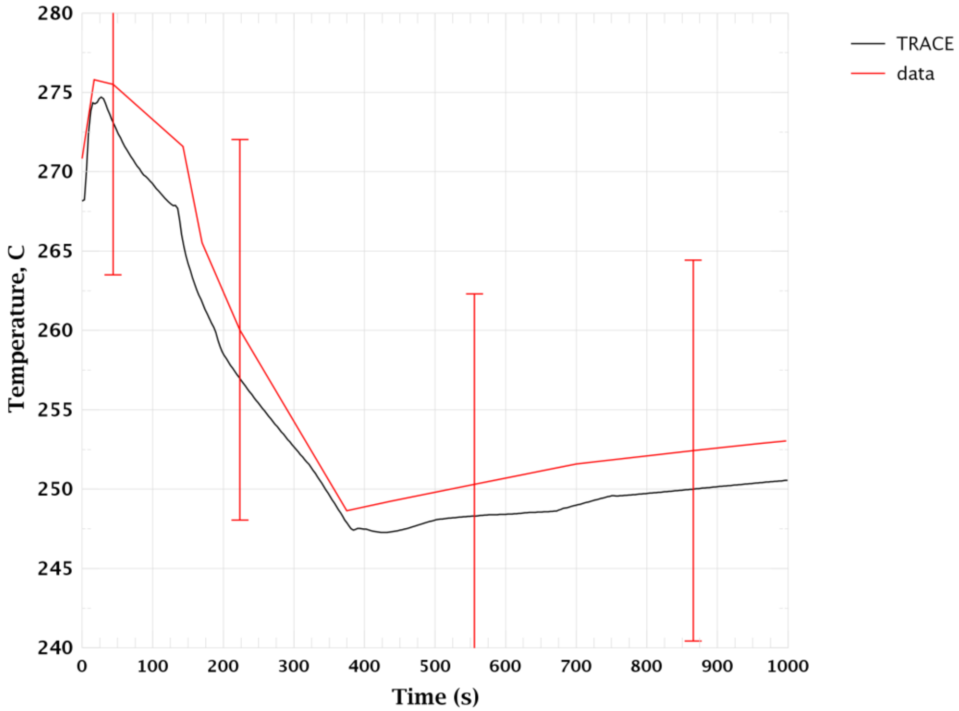



**Figure 38 Cold Leg No. 6 Temperature**

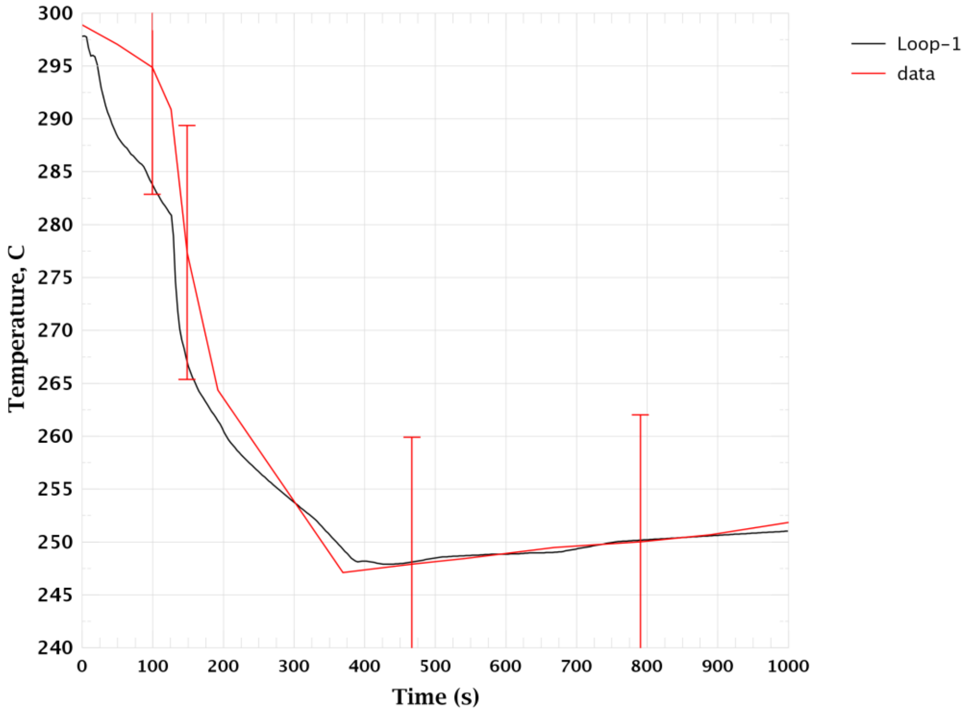



**Figure 39 Hot Leg No. 1 Temperature**

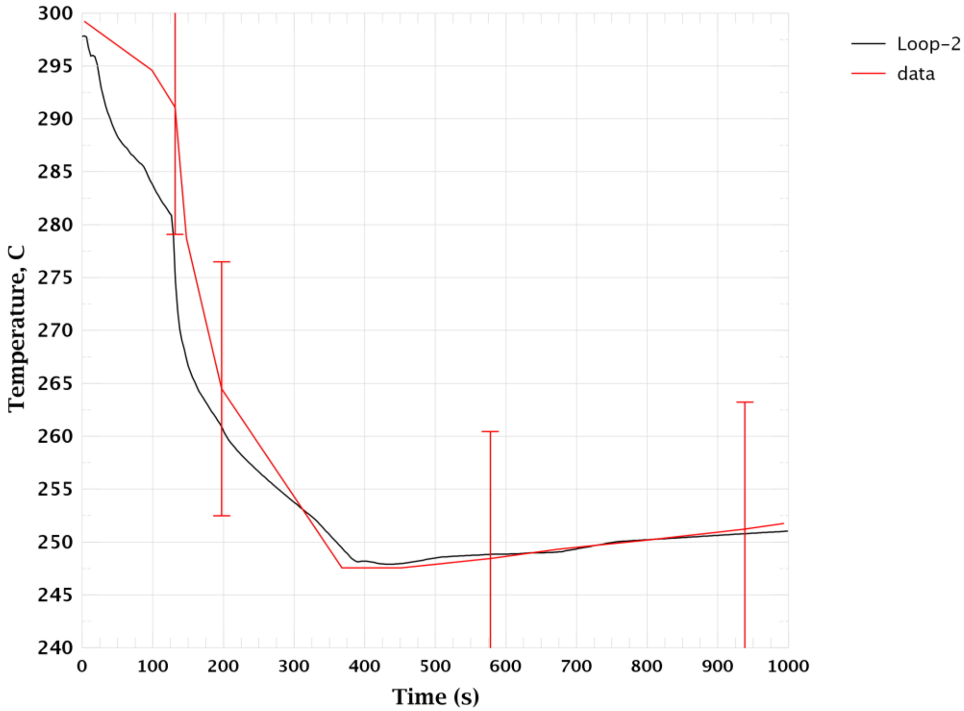



**Figure 40 Hot Leg No. 2 Temperature**

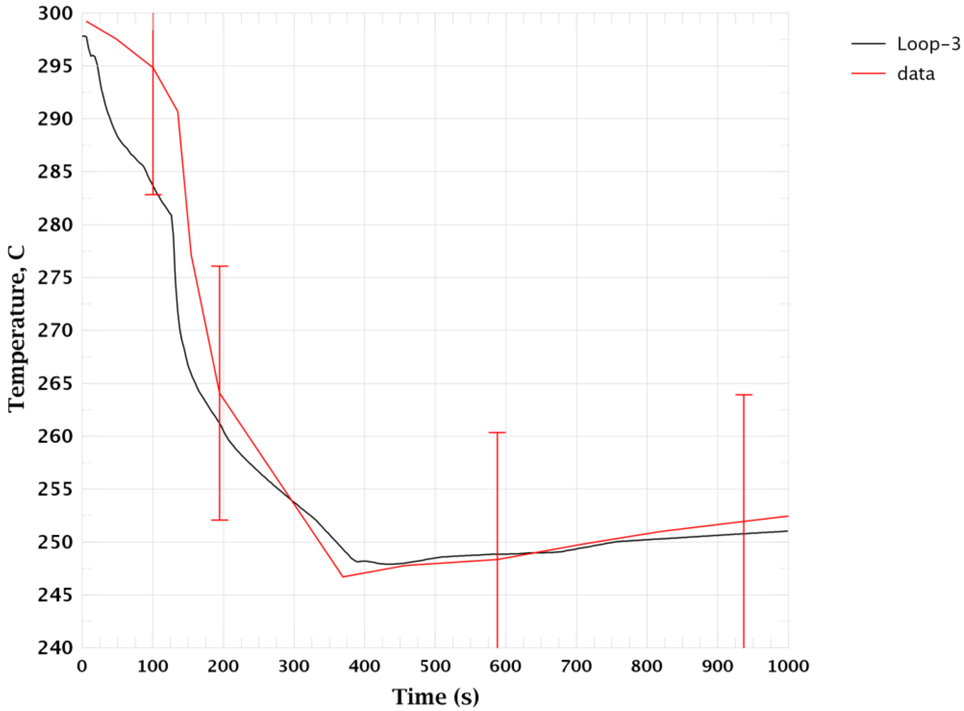



**Figure 41 Hot Leg No. 3 Temperature**

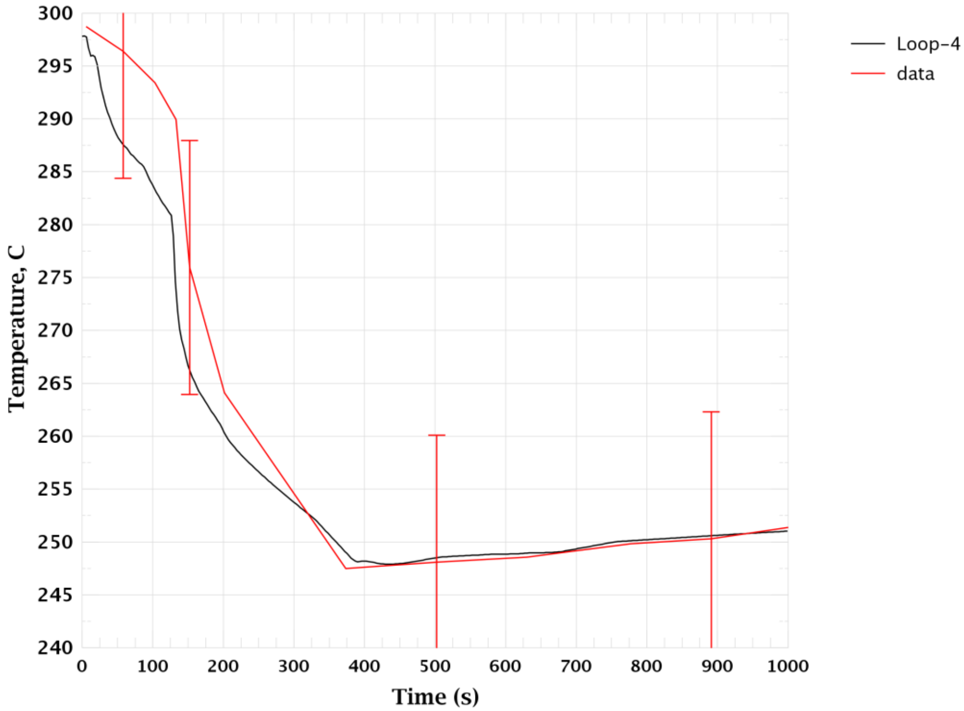



**Figure 42 Hot Leg No. 4 Temperature**

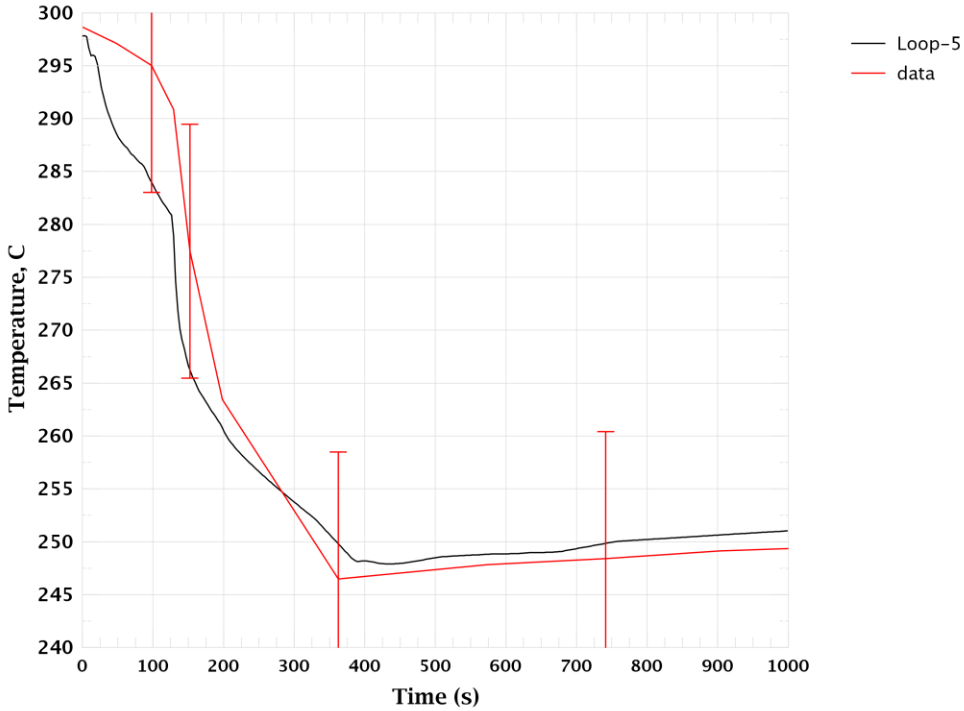



**Figure 43 Hot Leg No. 5 Temperature**

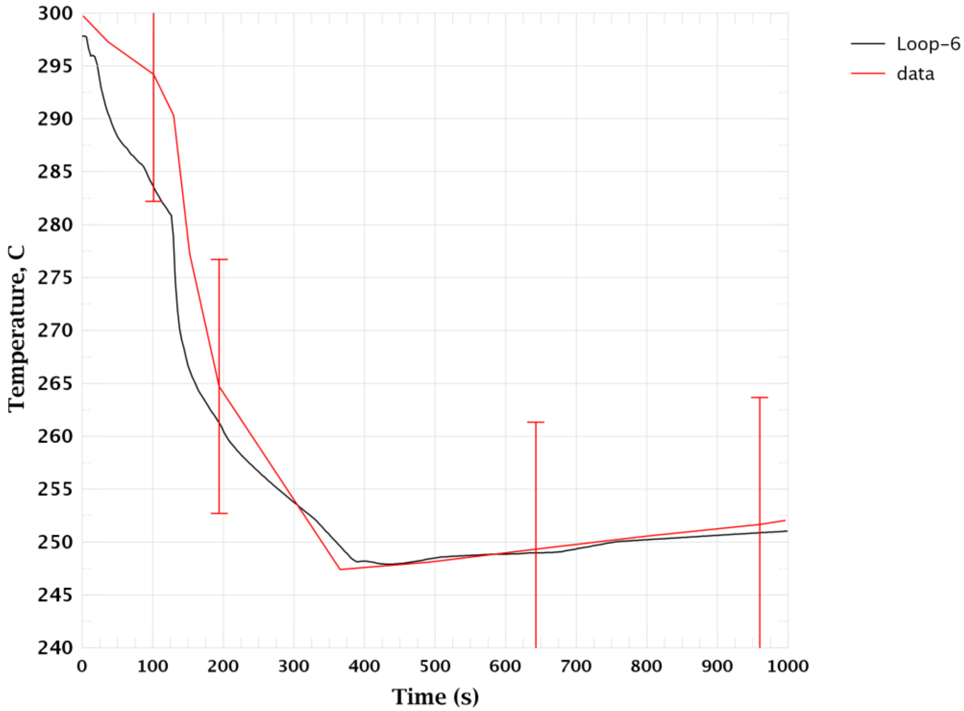



**Figure 44 Hot Leg No. 6 Temperature**

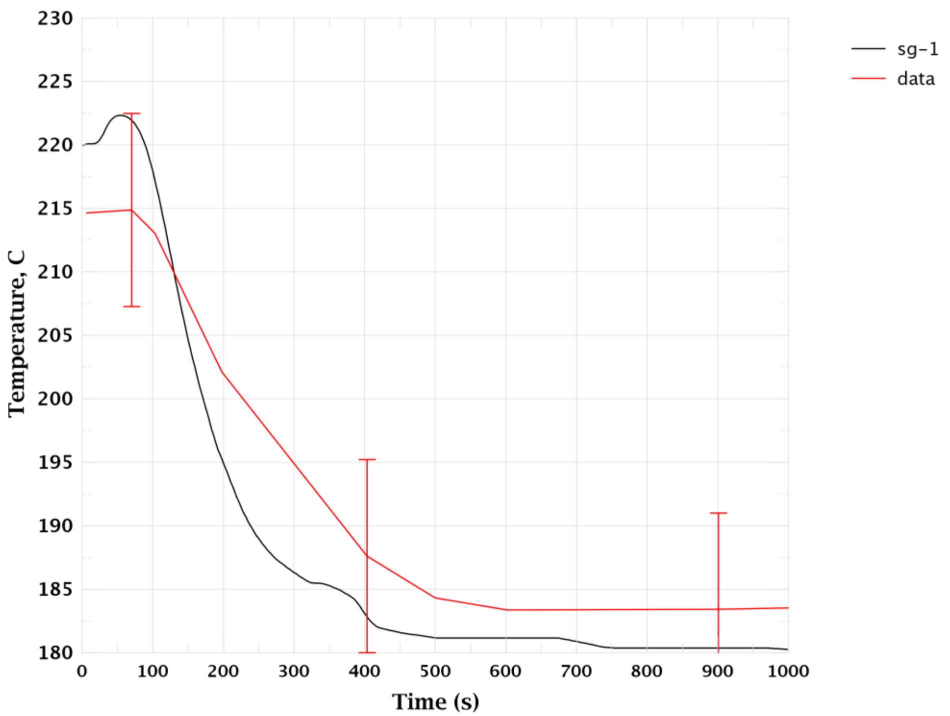



**Figure 45** **SG-1 Feedwater Temperature**

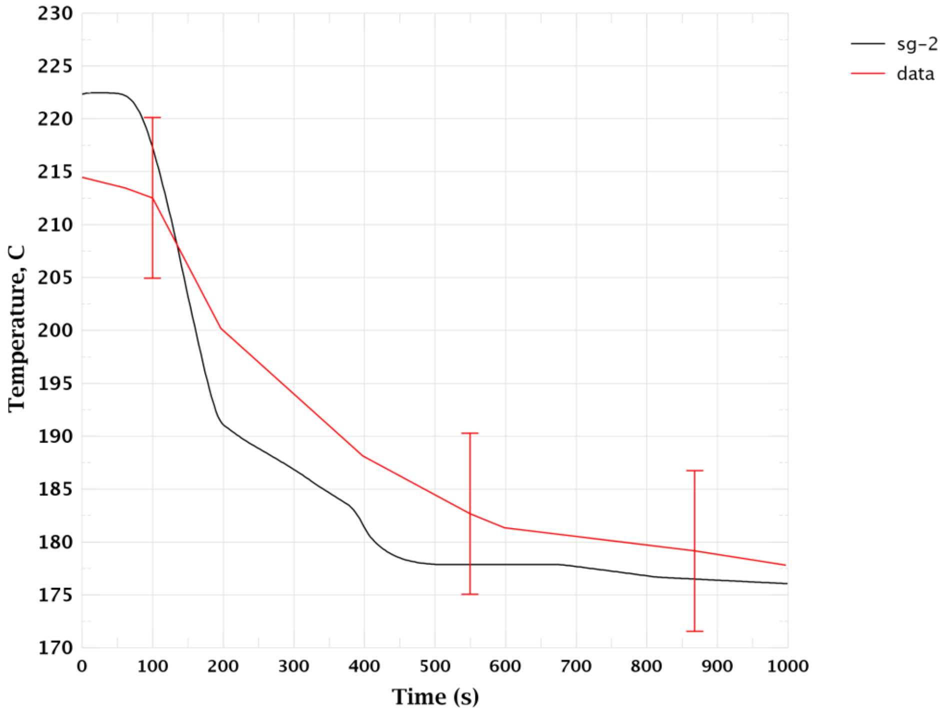



**Figure 46 SG-2 Feedwater Temperature**

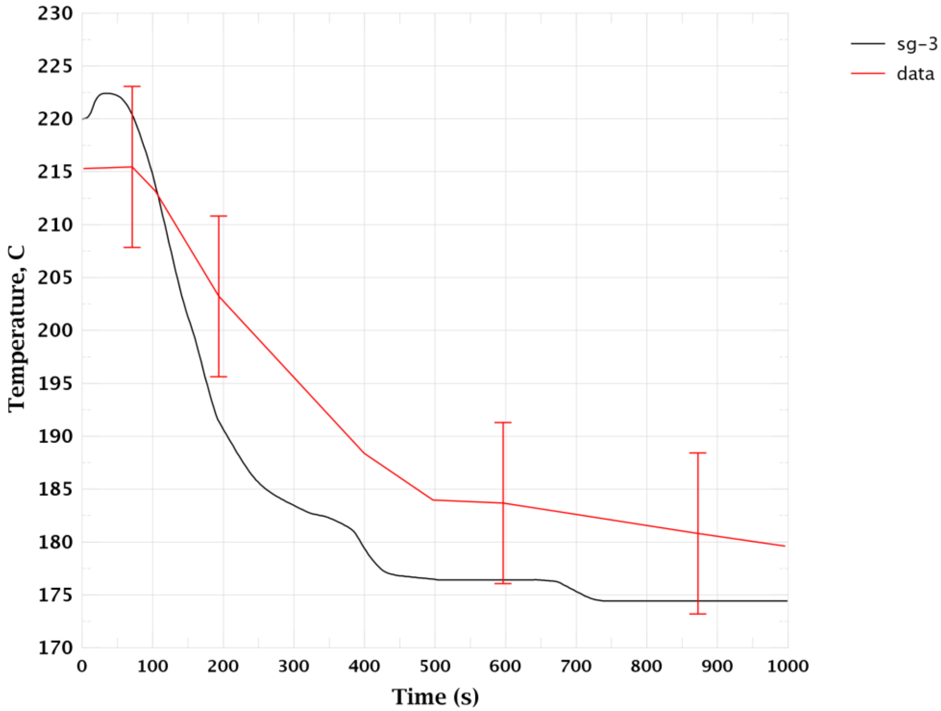



**Figure 47 SG-3 Feedwater Temperature**

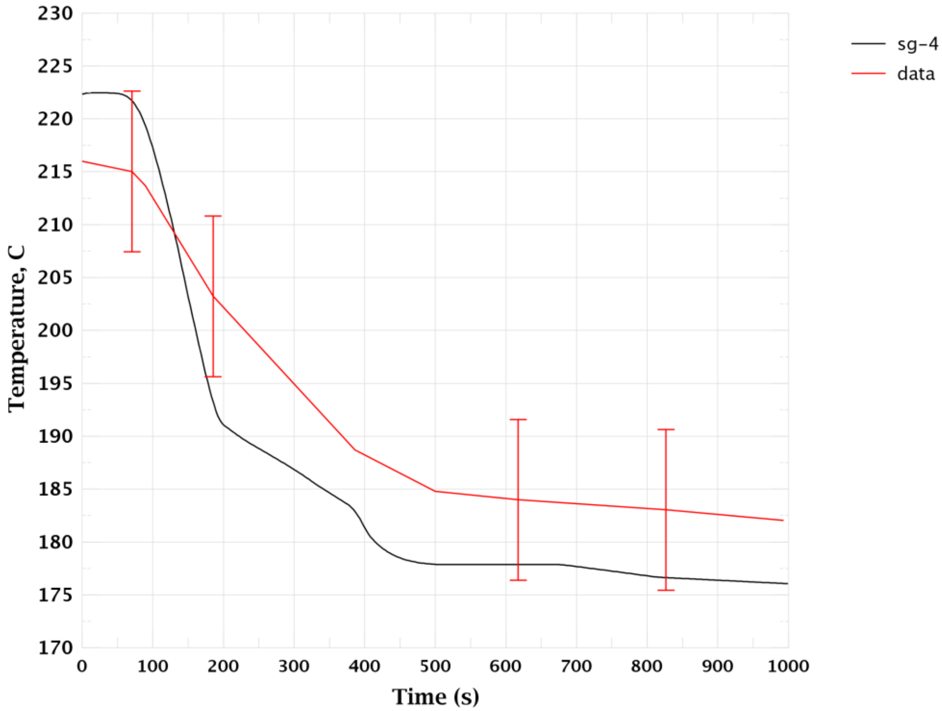



**Figure 48 SG-4 Feedwater Temperature**

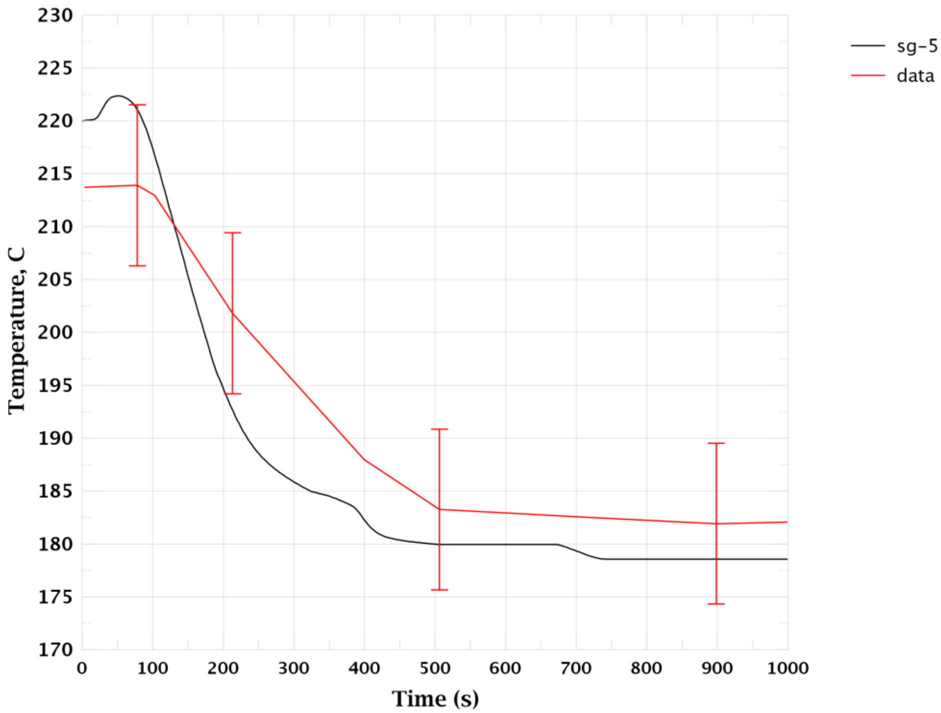



**Figure 49 SG-5 Feedwater Temperature**

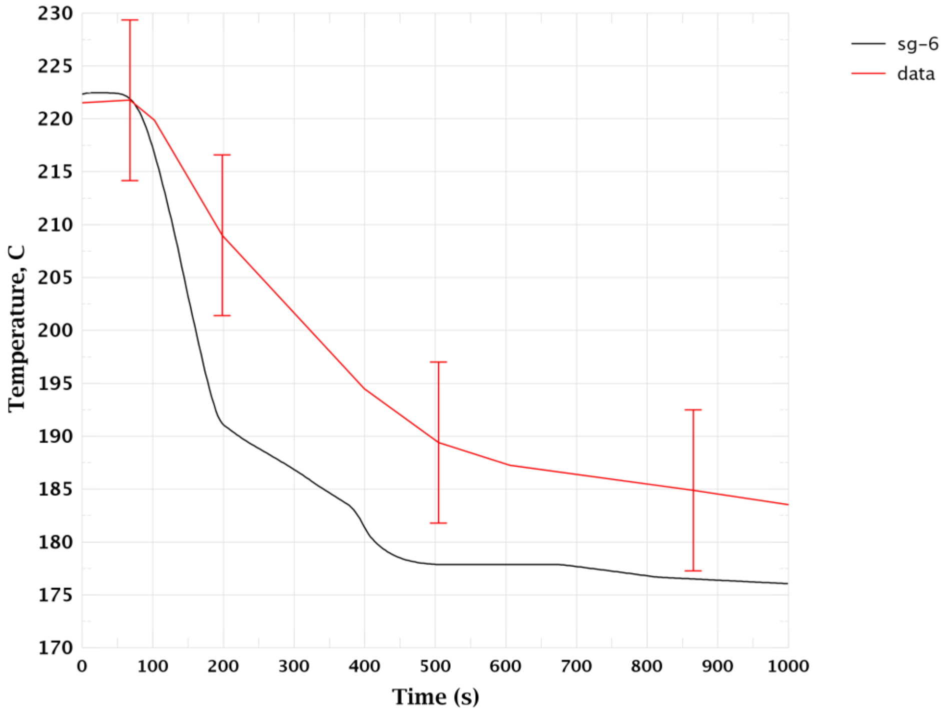



**Figure 50 SG-6 Feedwater Temperature**

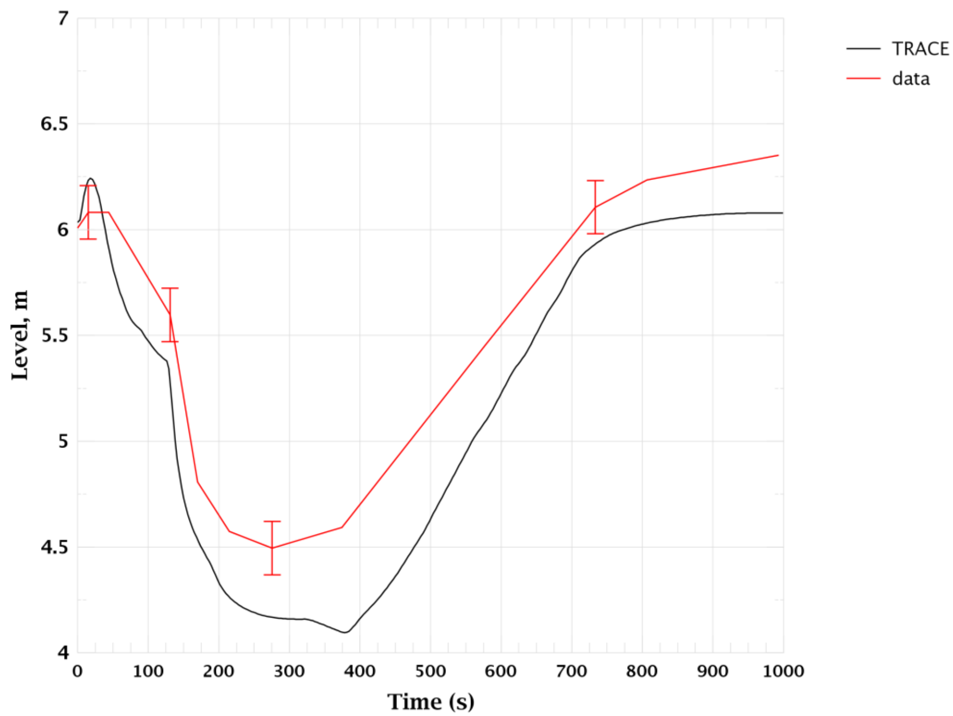



**Figure 51 PRZ Level**

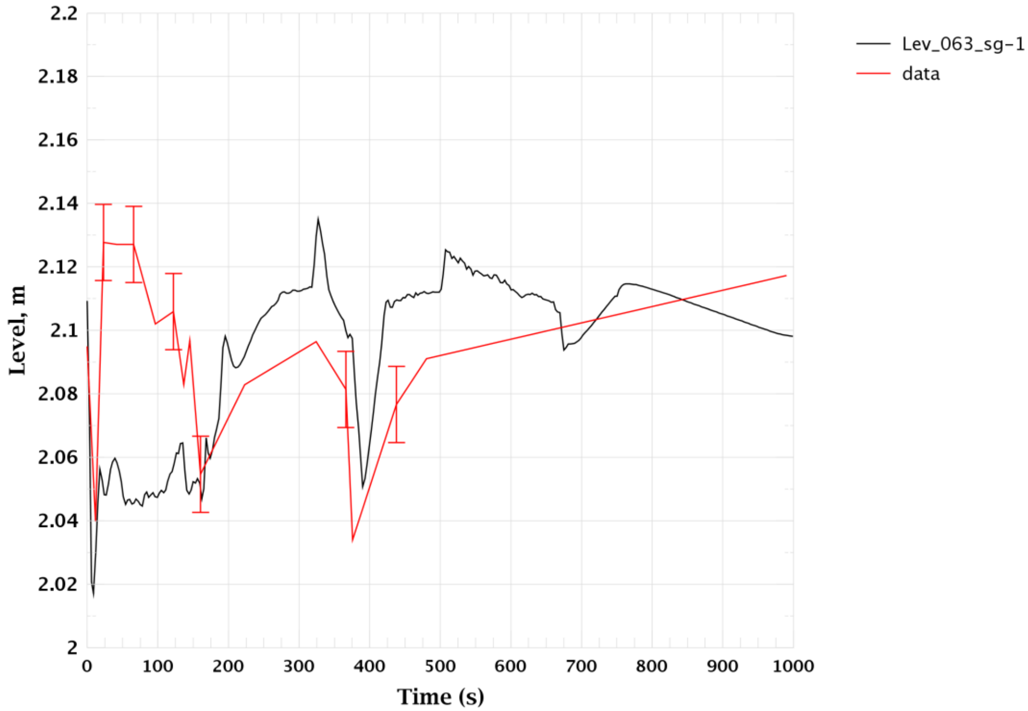



**Figure 52 SG-1 Level (narrow range measurement)**

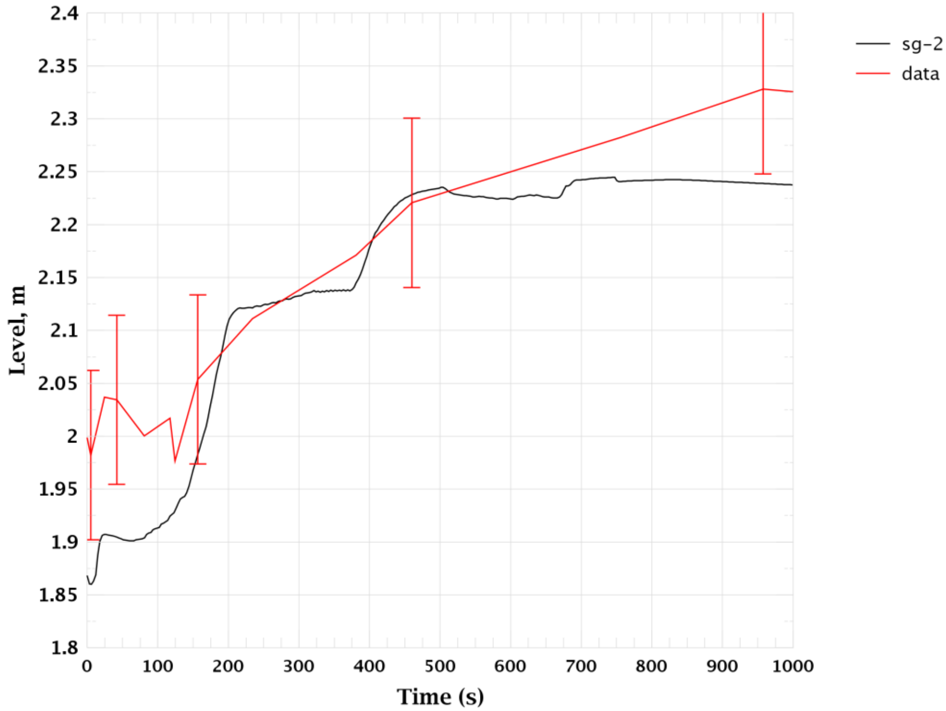



**Figure 53 SG-2 Level (narrow range measurement)**

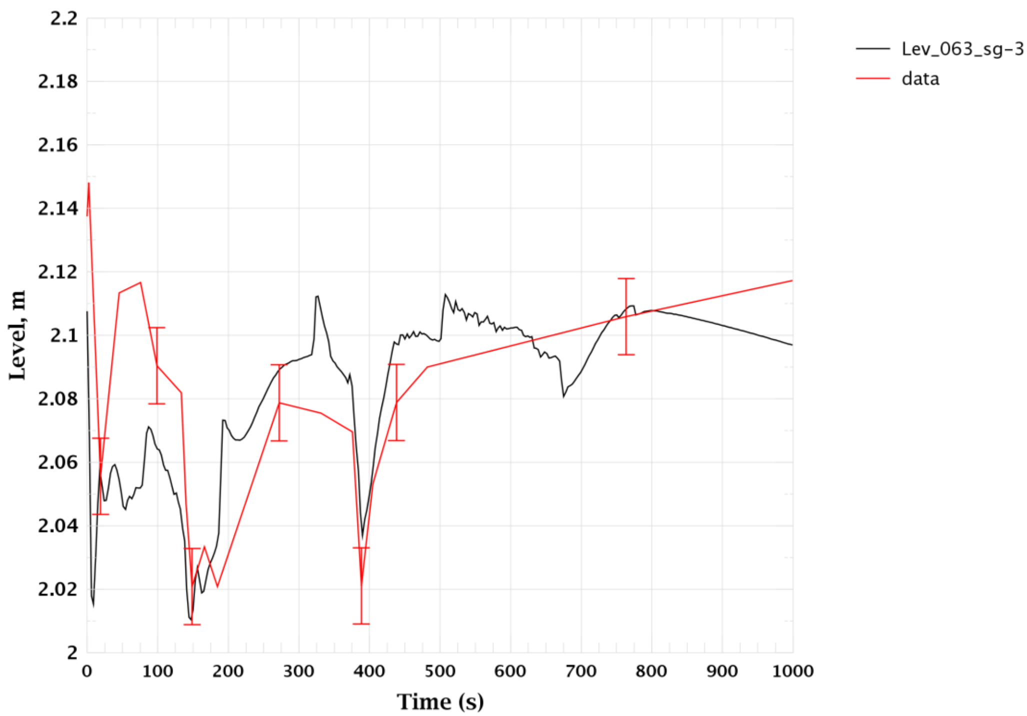



**Figure 54 SG-3 Level (narrow range measurement)**

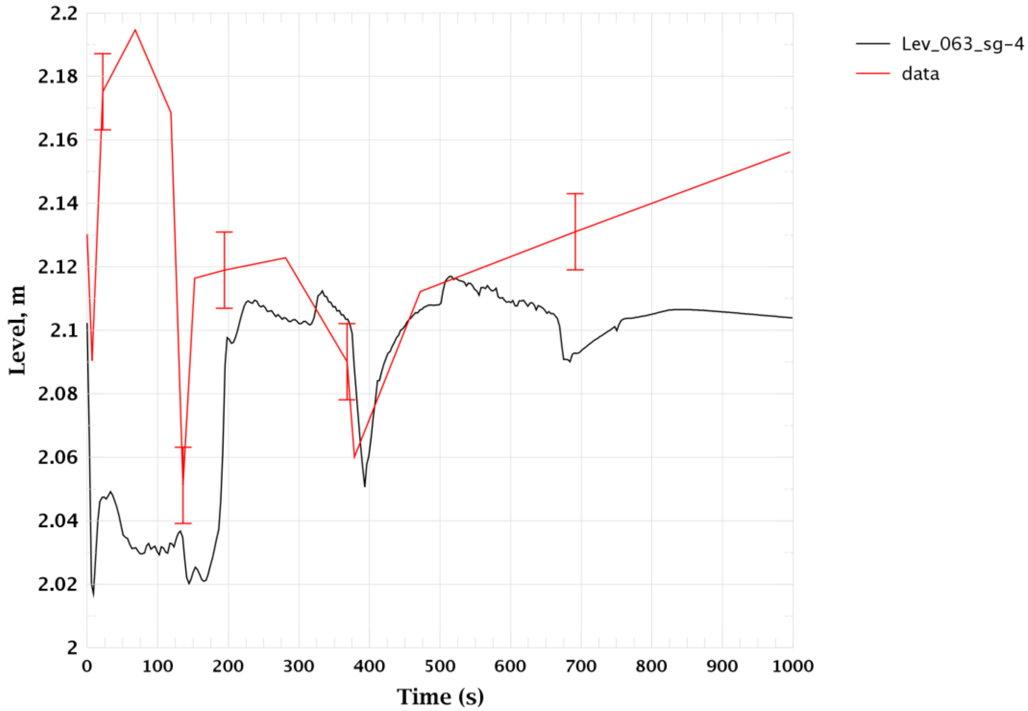



**Figure 55 SG-4 Level (narrow range measurement)**

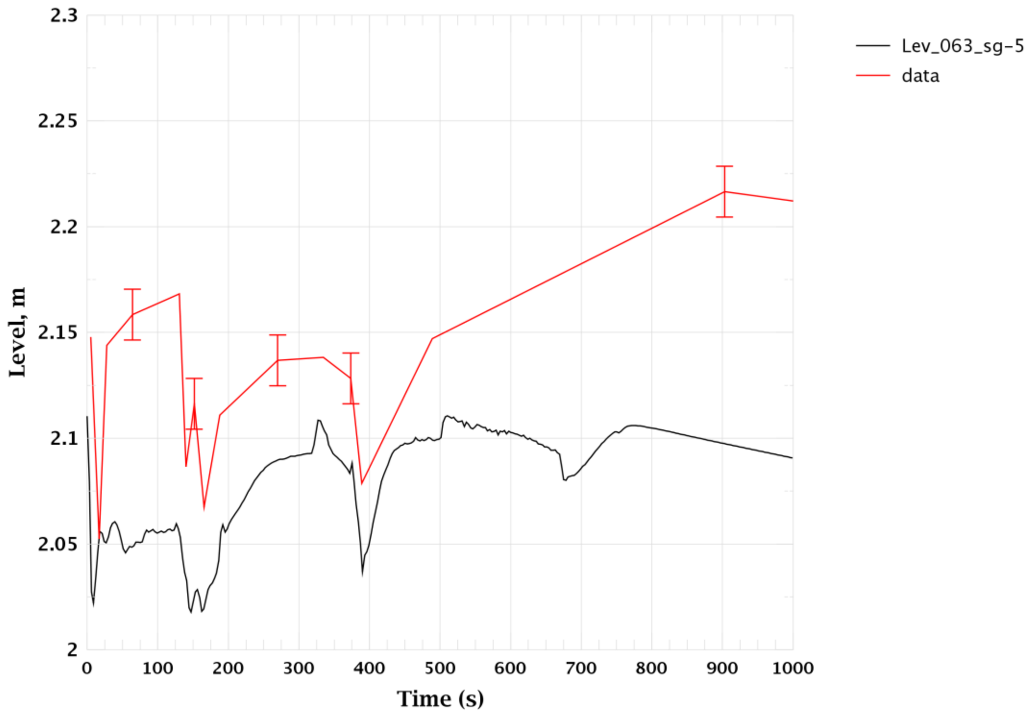



**Figure 56 SG-5 Level (narrow range measurement)**

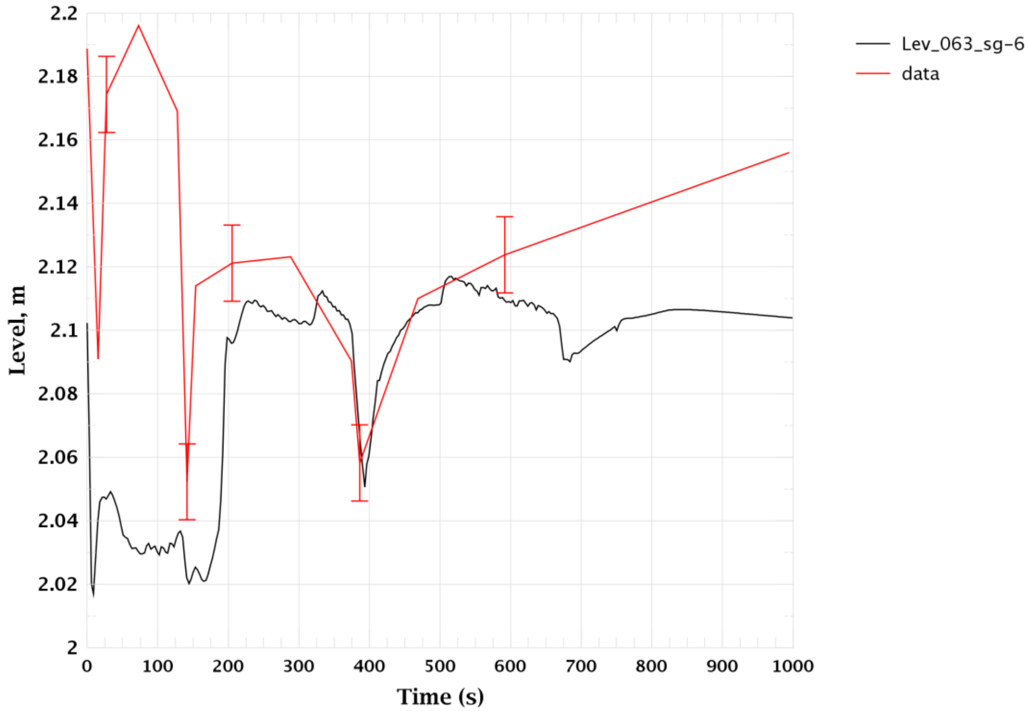



**Figure 57 SG-6 Level (narrow range measurement)**

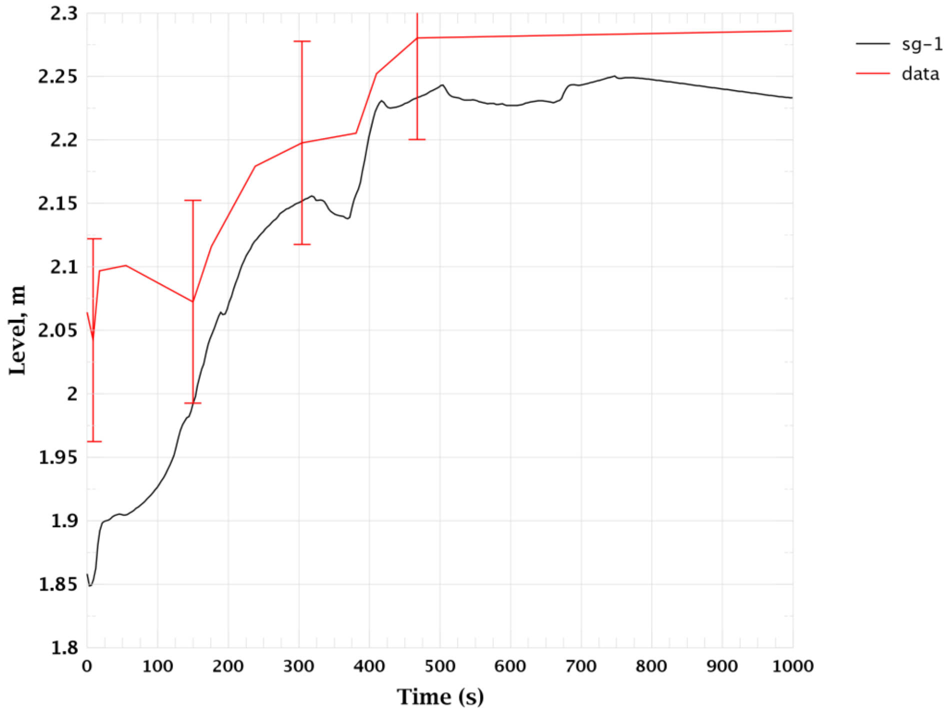



**Figure 58 SG-1 Level (wide range measurement)**

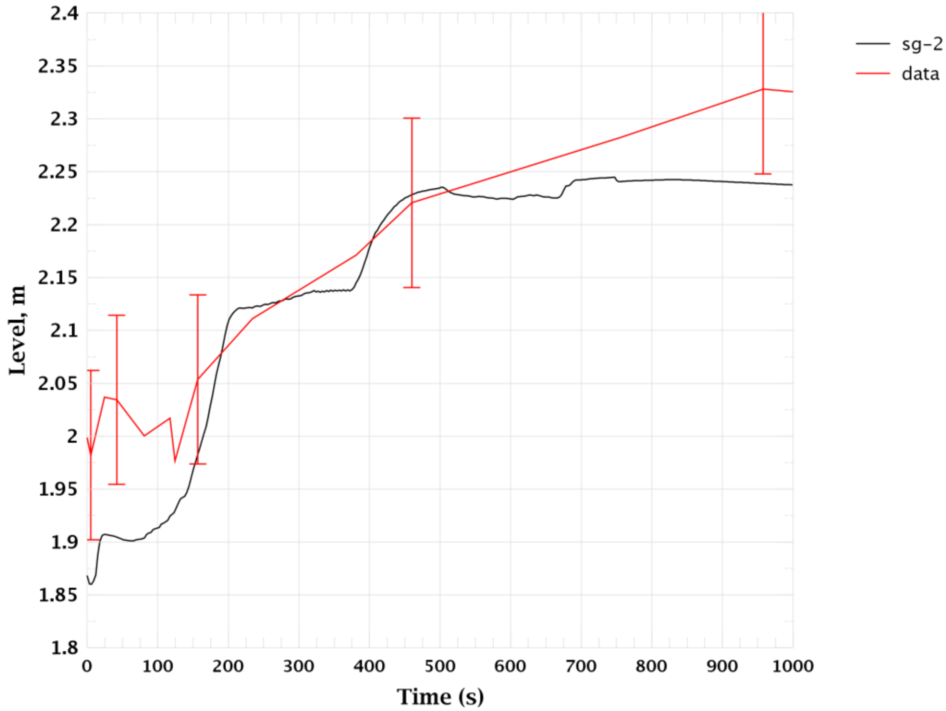



**Figure 59 SG-2 Level (wide range measurement)**

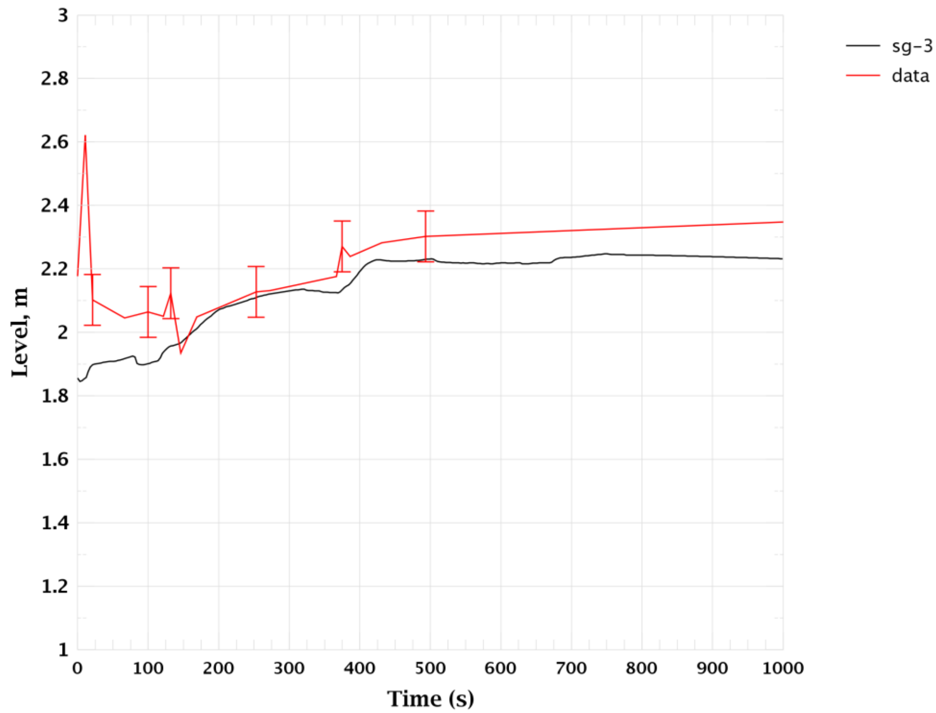



**Figure 60 SG-3 Level (wide range measurement)**

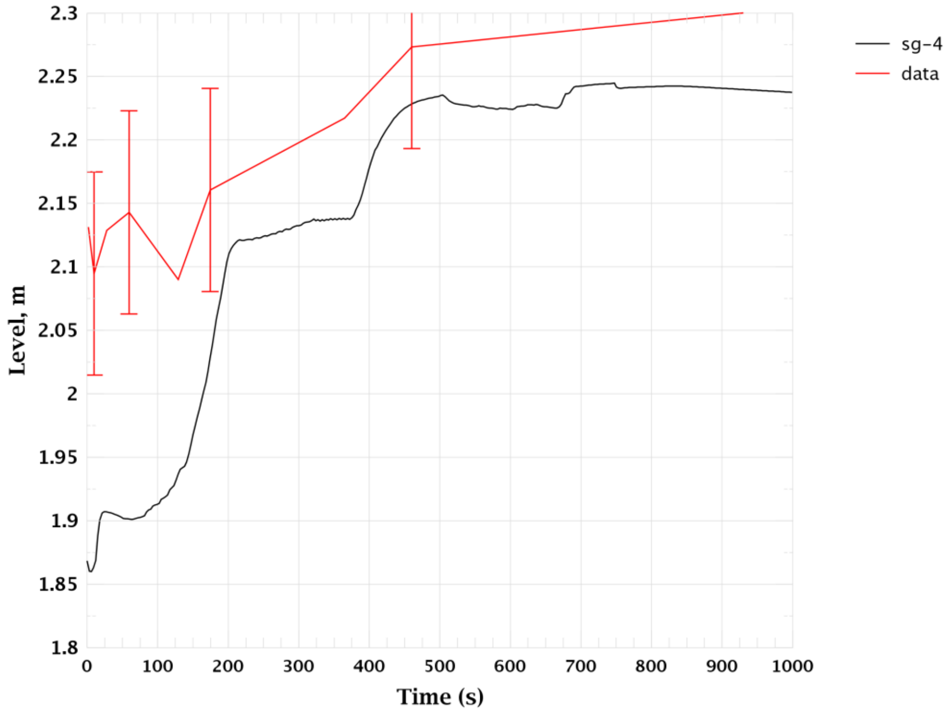



**Figure 61 SG-4 Level (wide range measurement)**

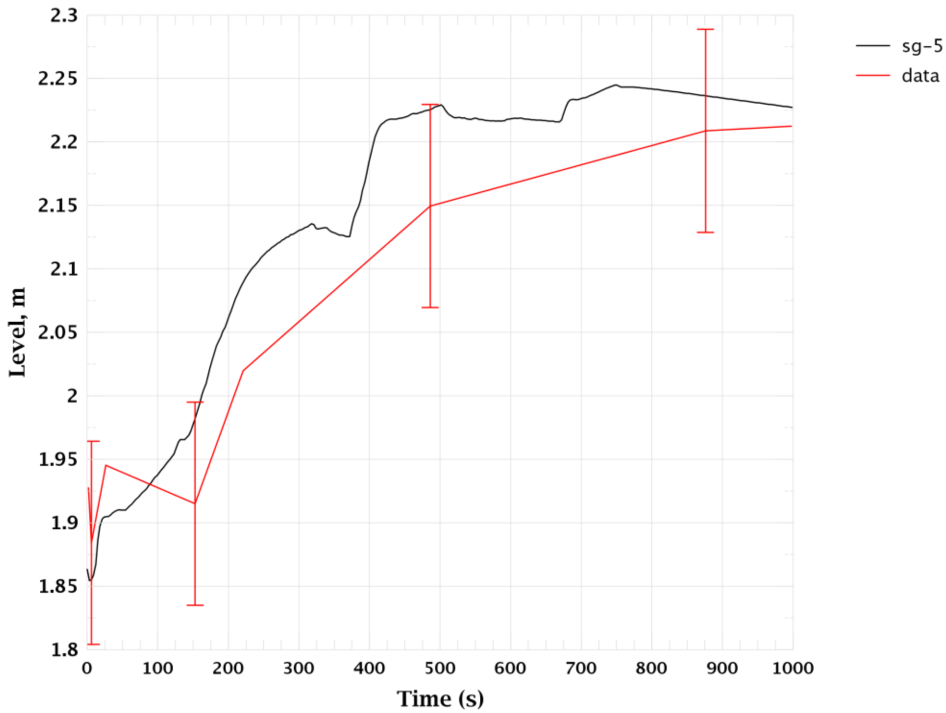



**Figure 62 SG-5 Level (wide range measurement)**

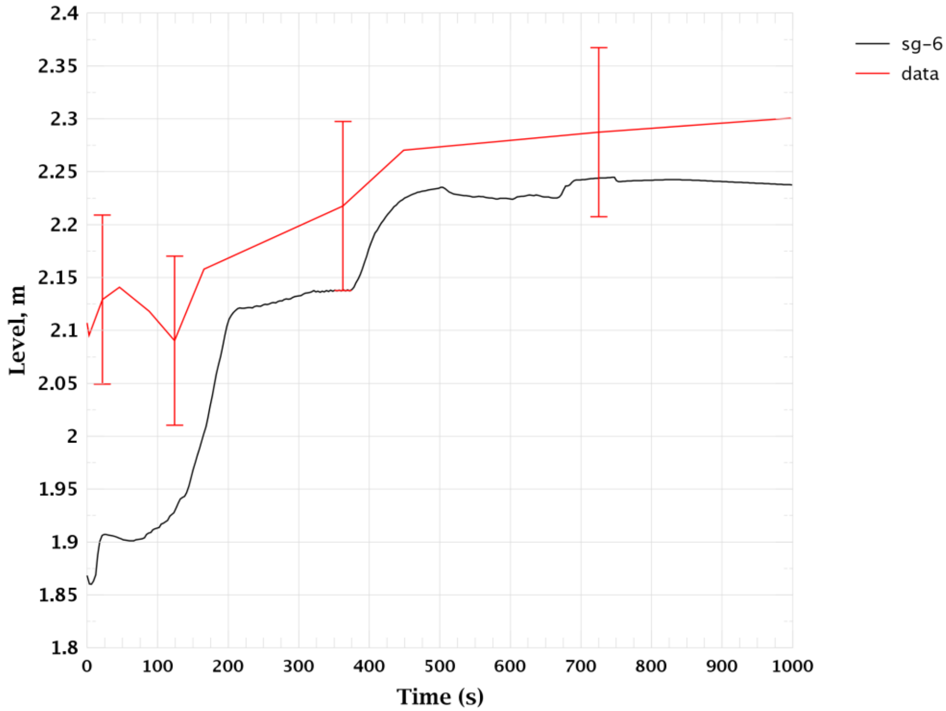



**Figure 63 SG-6 Level (wide range measurement)**

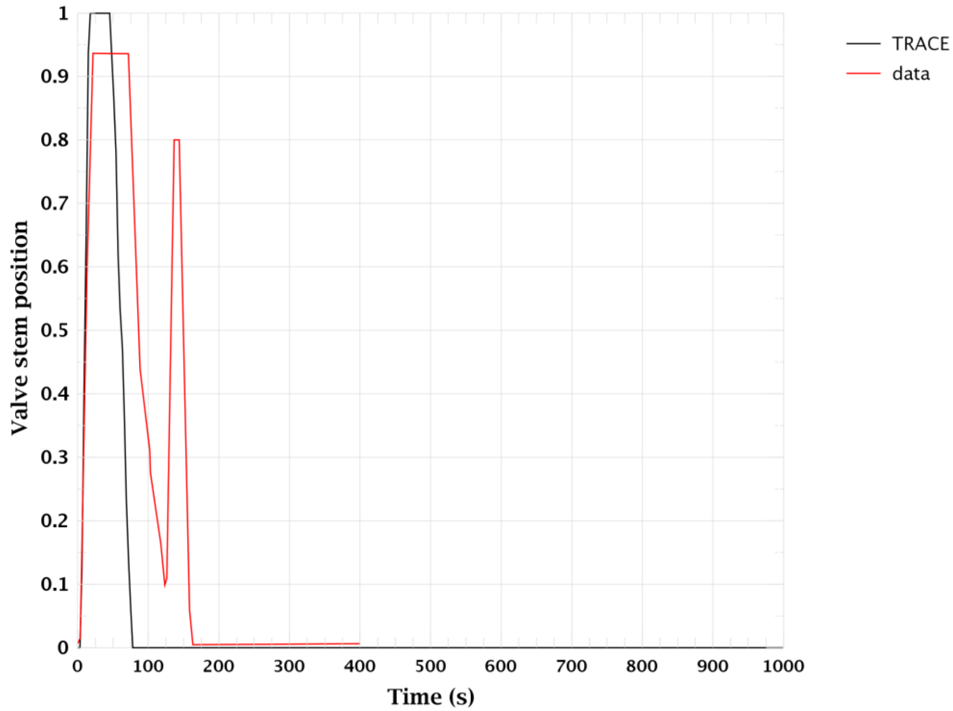



**Figure 64 BRU-K-1A Valve Stem Position**

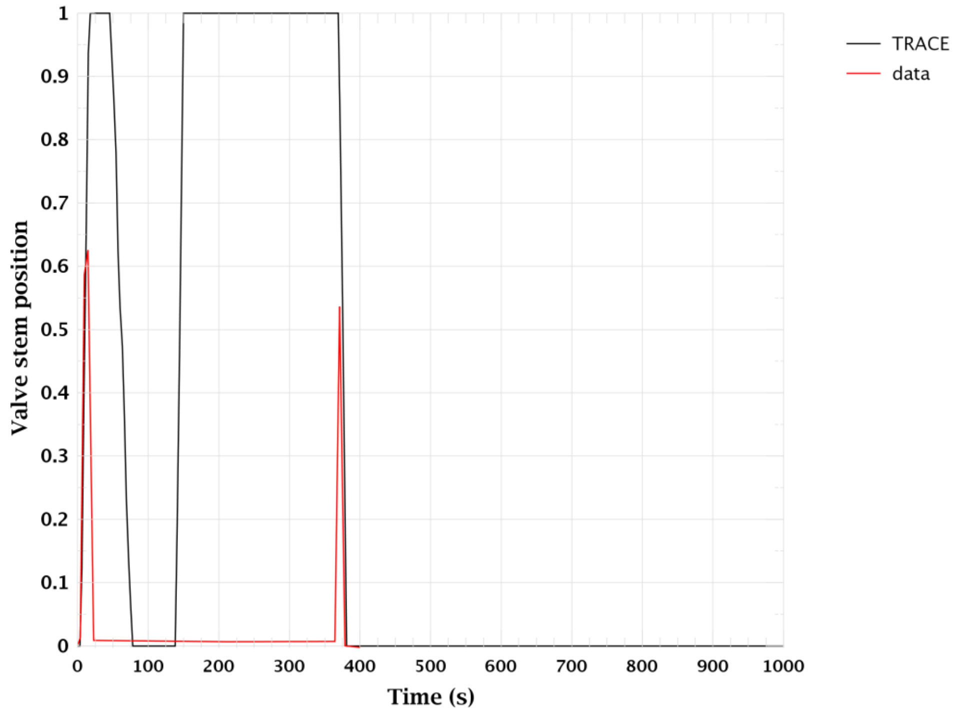



**Figure 65 BRU-K-1B Valve Stem Position**

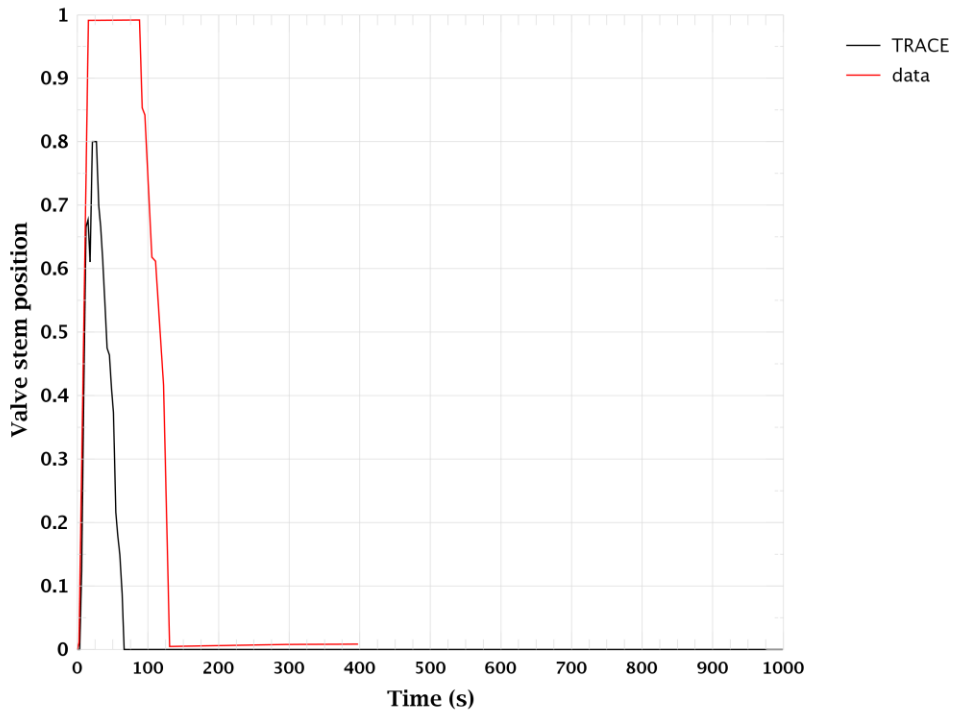



**Figure 66 BRU-K-2 Valve Stem Position**

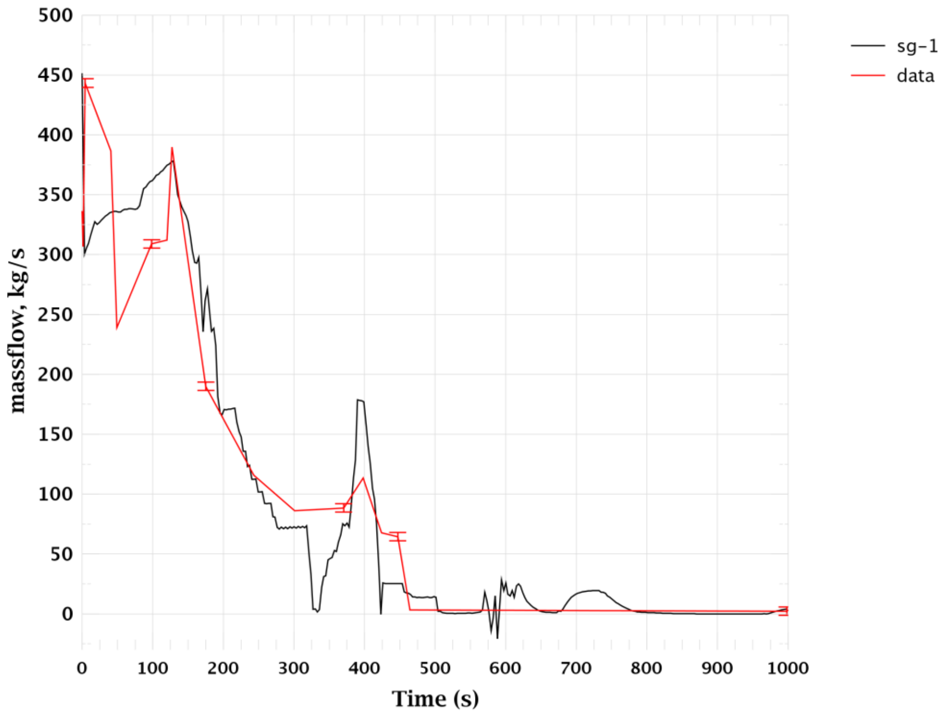



**Figure 67 SG-1 Feedwater Mass Flow**

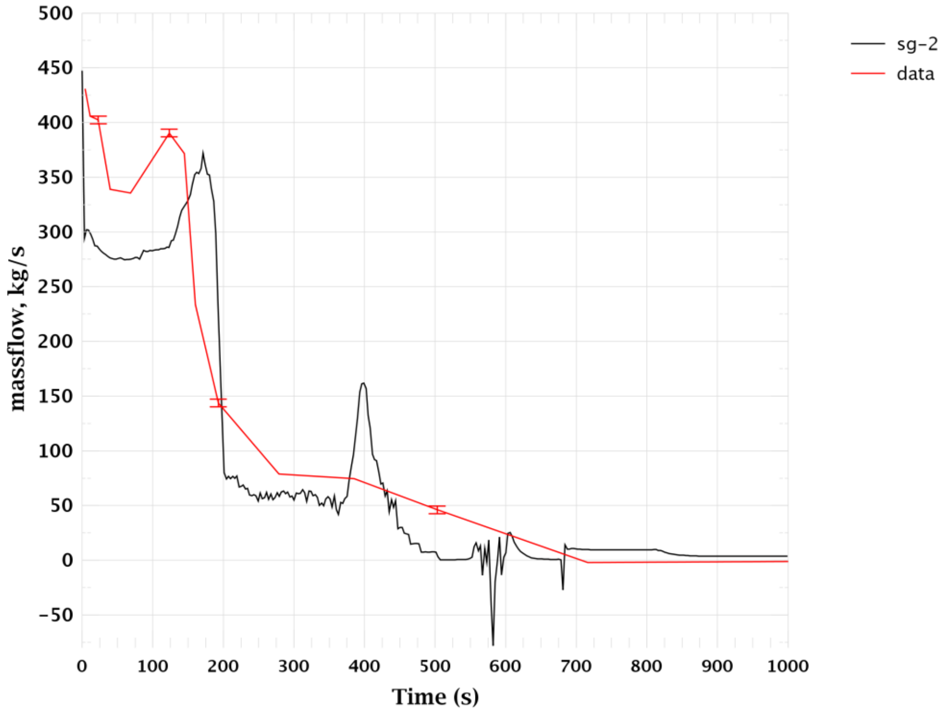



**Figure 68 SG-2 Feedwater Mass Flow**

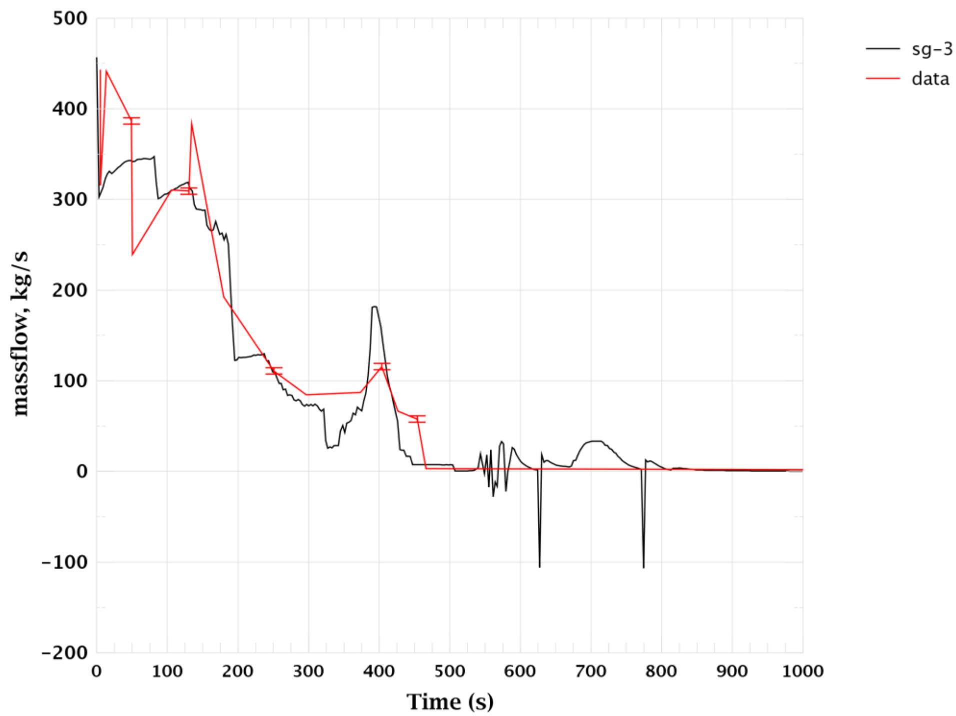



**Figure 69 SG-3 Feedwater Mass Flow**

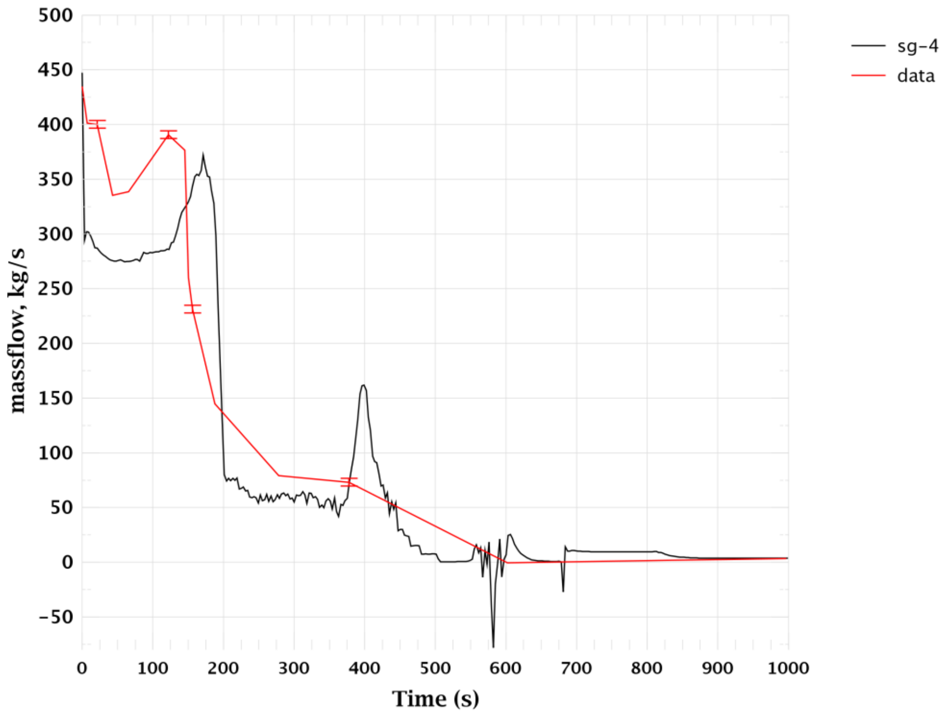



**Figure 70 SG-4 Feedwater Mass Flow**

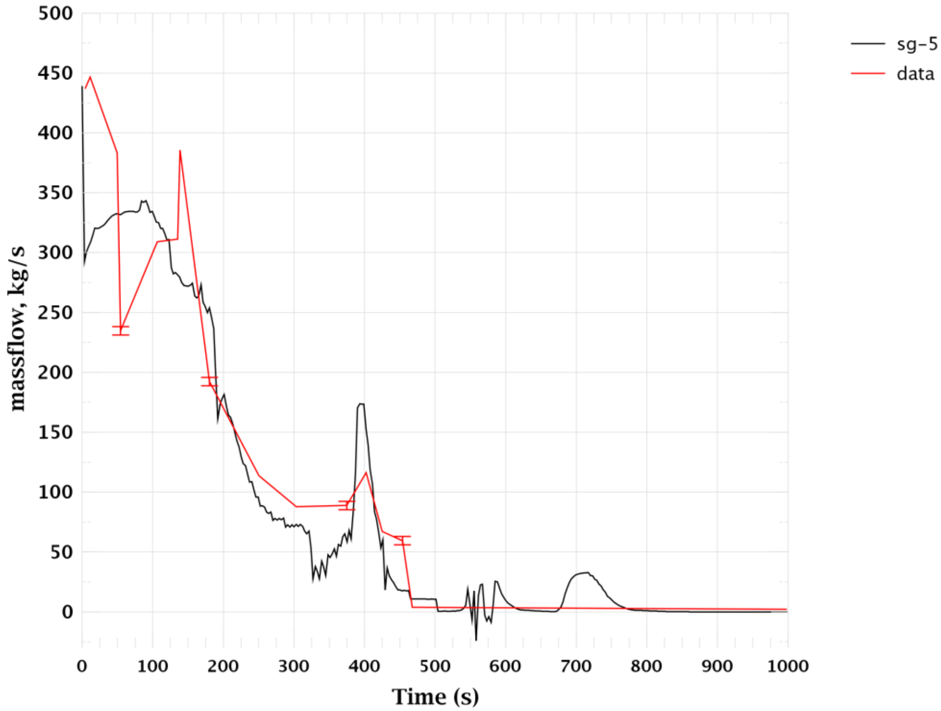



**Figure 71 SG-5 Feedwater Mass Flow**

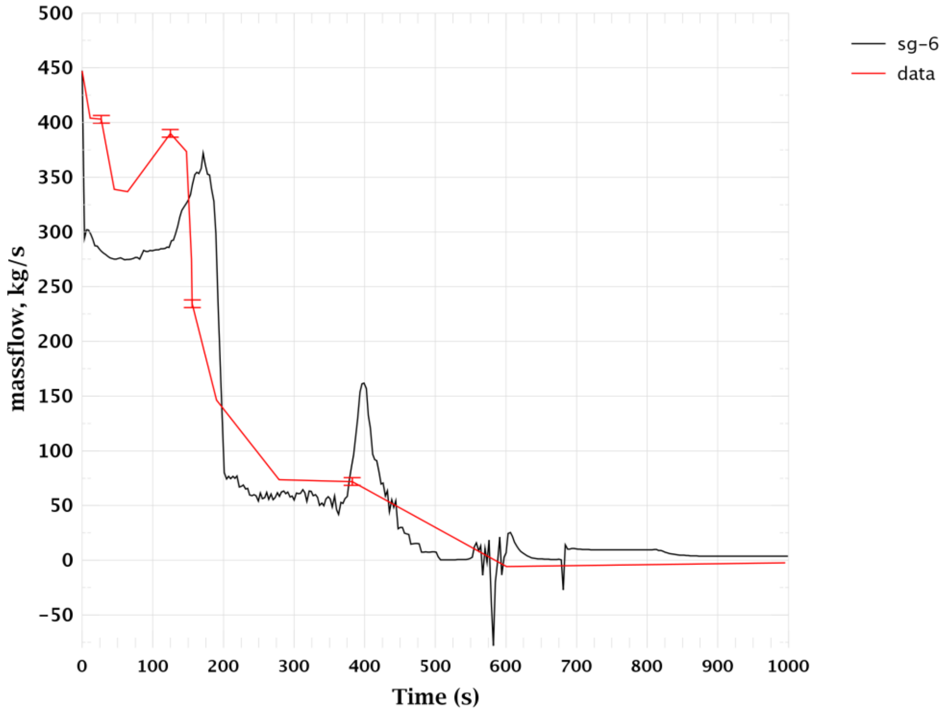



**Figure 72 SG-6 Feedwater Mass Flow**

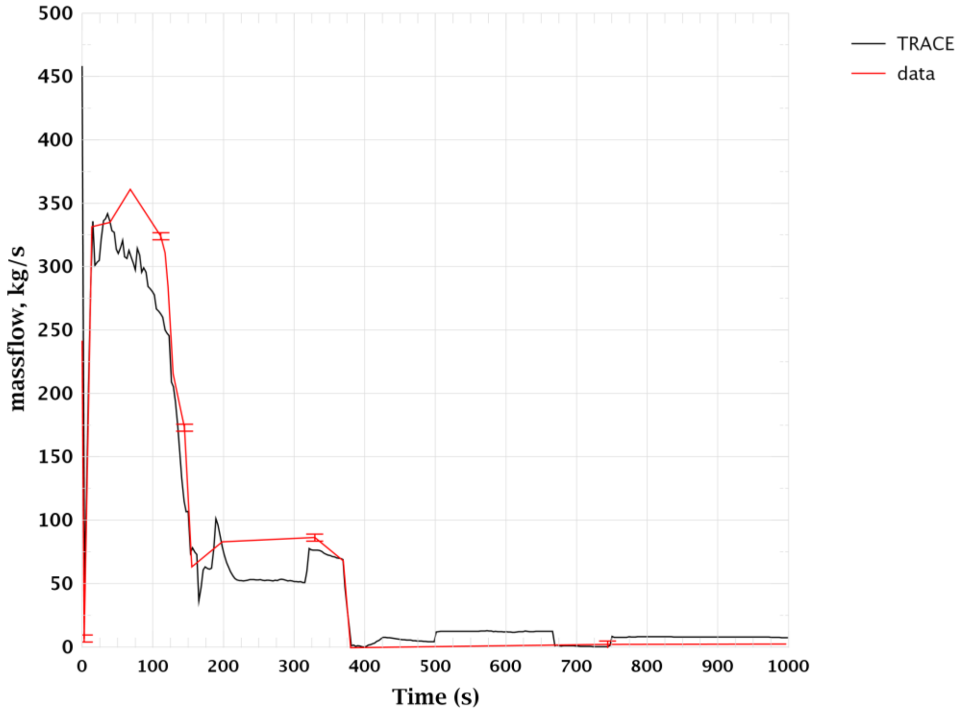



**Figure 73 SG-1 Steam Flow Rate**

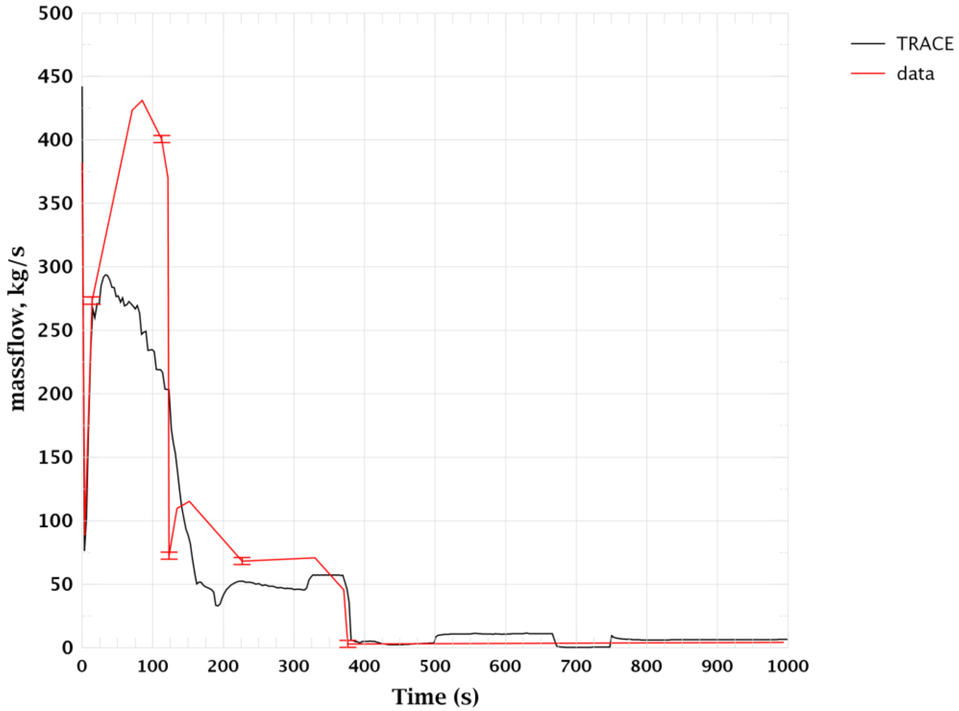



**Figure 74 SG-2 Steam Flow Rate**

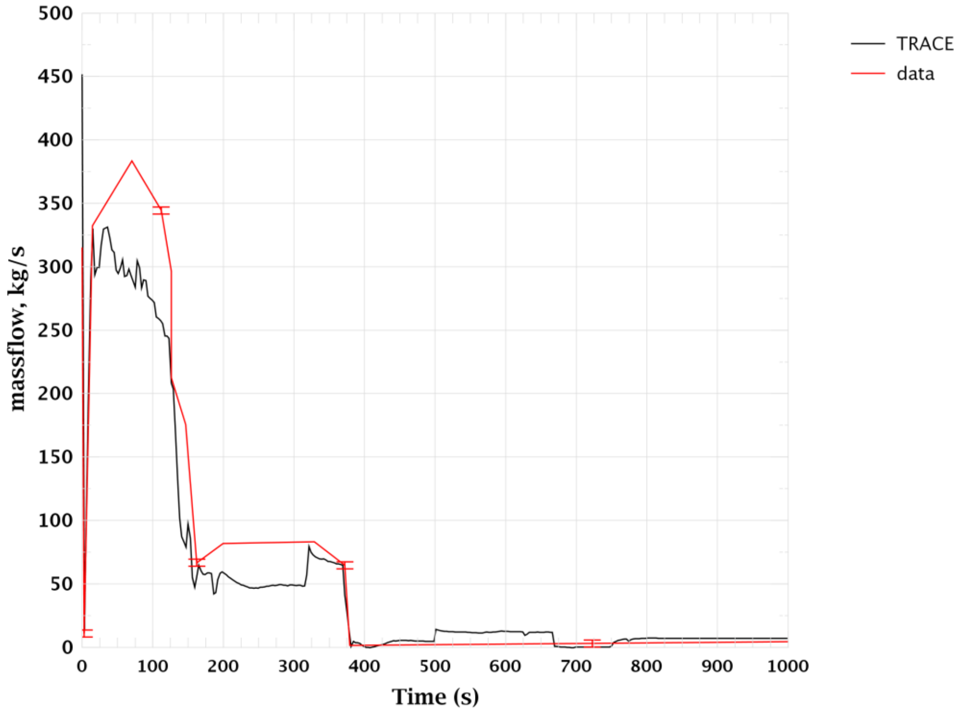



**Figure 75 SG-3 Steam Flow Rate**

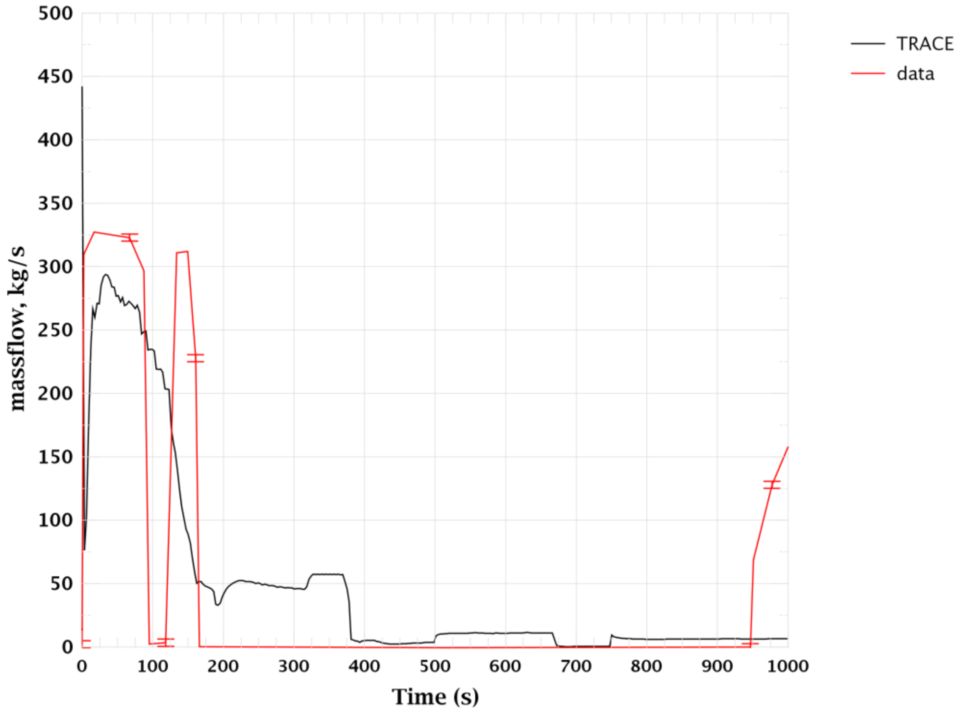



**Figure 76 SG-4 Steam Flow Rate**

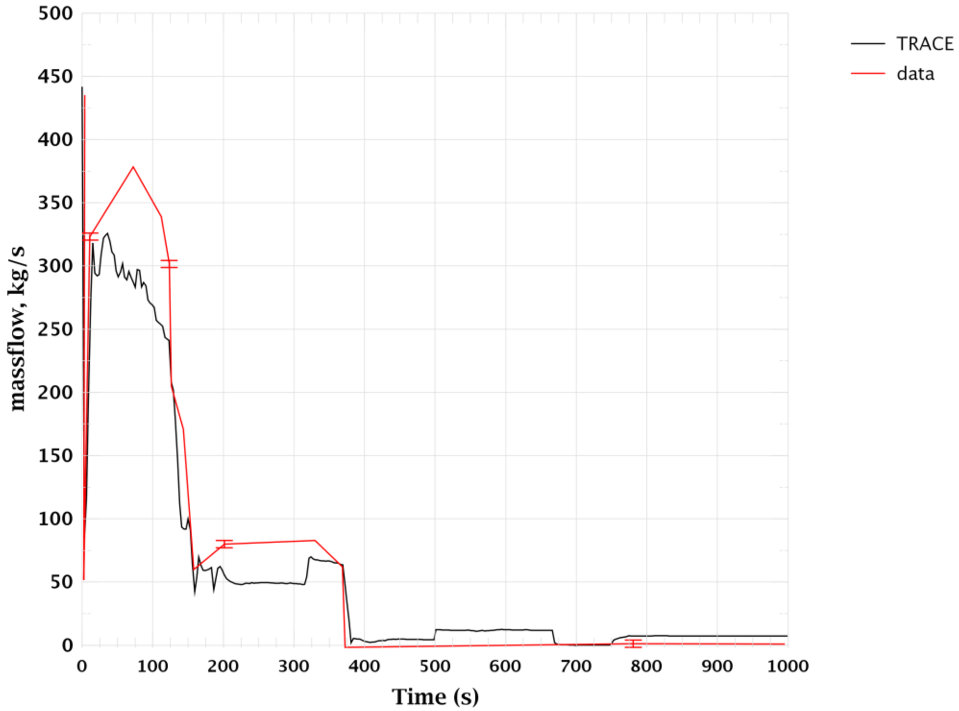



**Figure 77 SG-5 Steam Flow Rate**

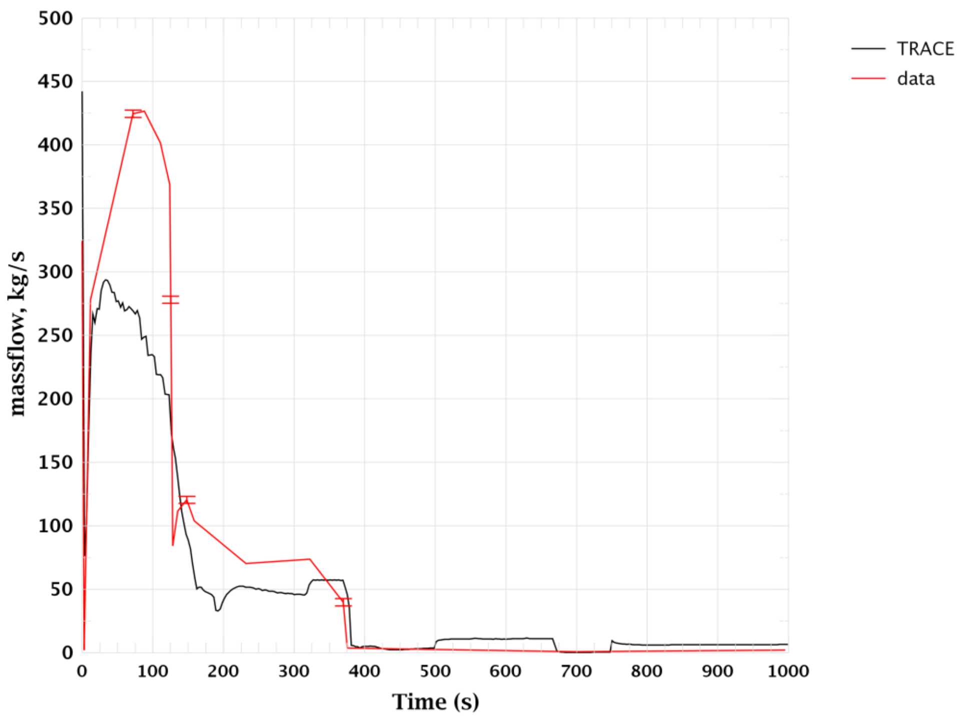



**Figure 78 SG-6 Steam Flow Rate**

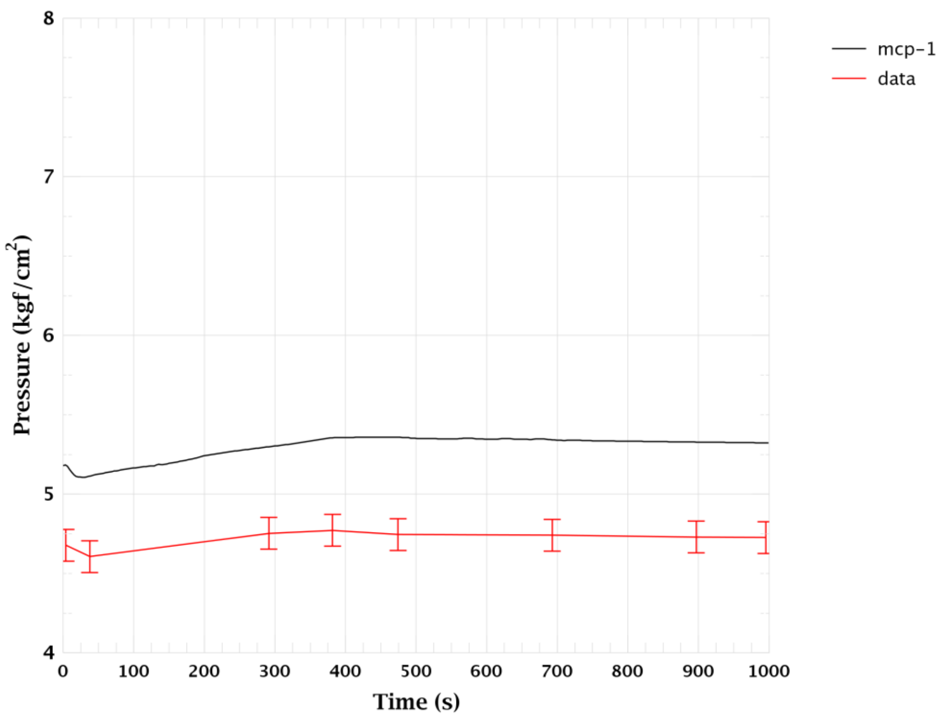



**Figure 79 MCP-1 Pressure Drop**

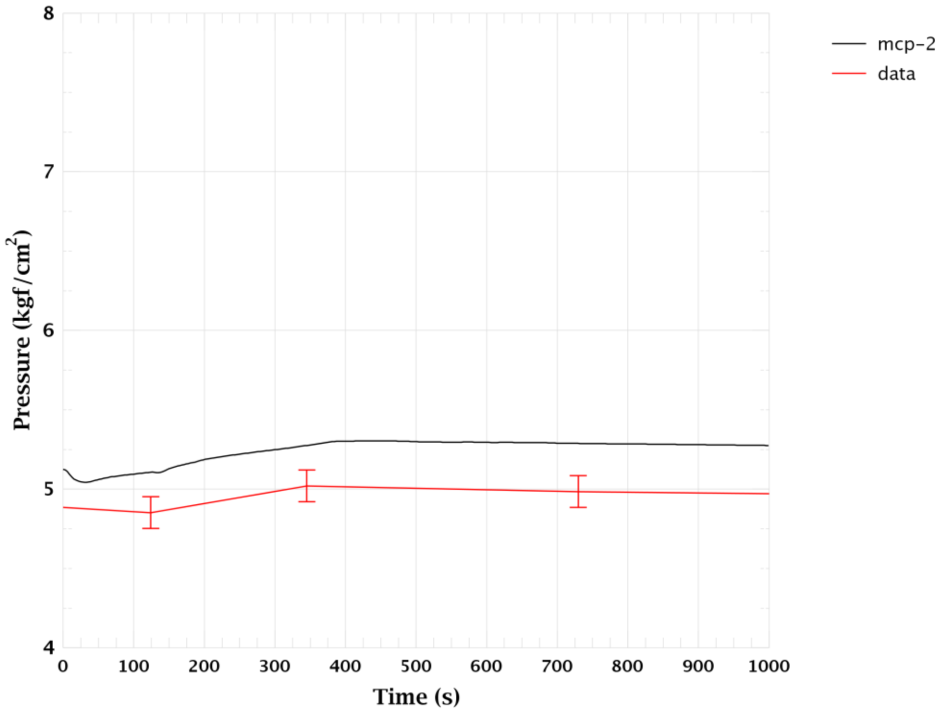



**Figure 80 MCP-2 Pressure Drop**

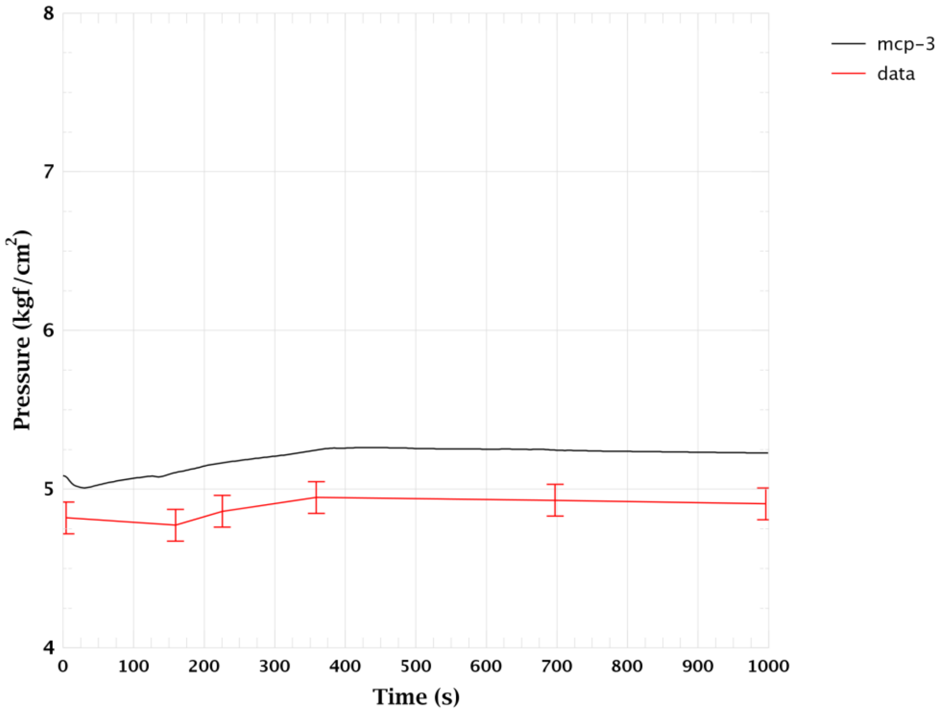



**Figure 81 MCP-3 Pressure Drop**

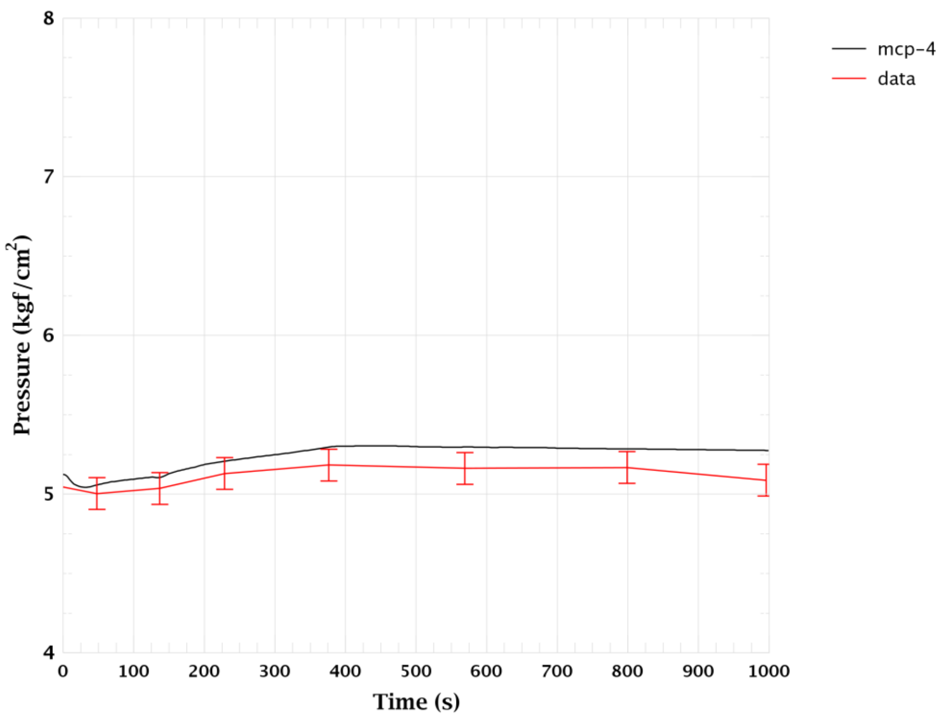



**Figure 82 MCP-4 Pressure Drop**

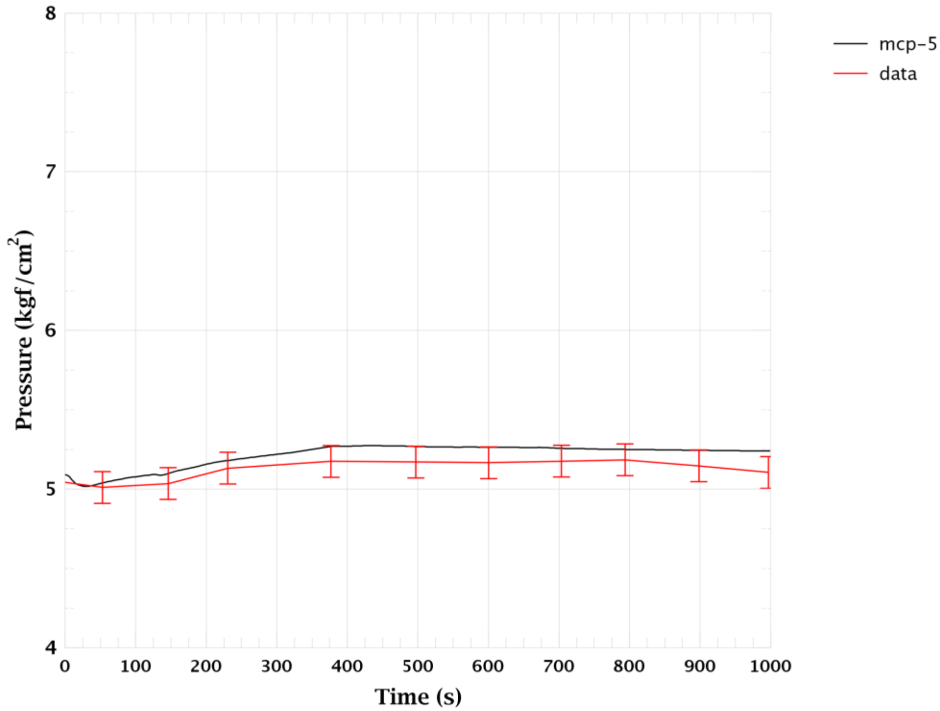



**Figure 83 MCP-5 Pressure Drop**

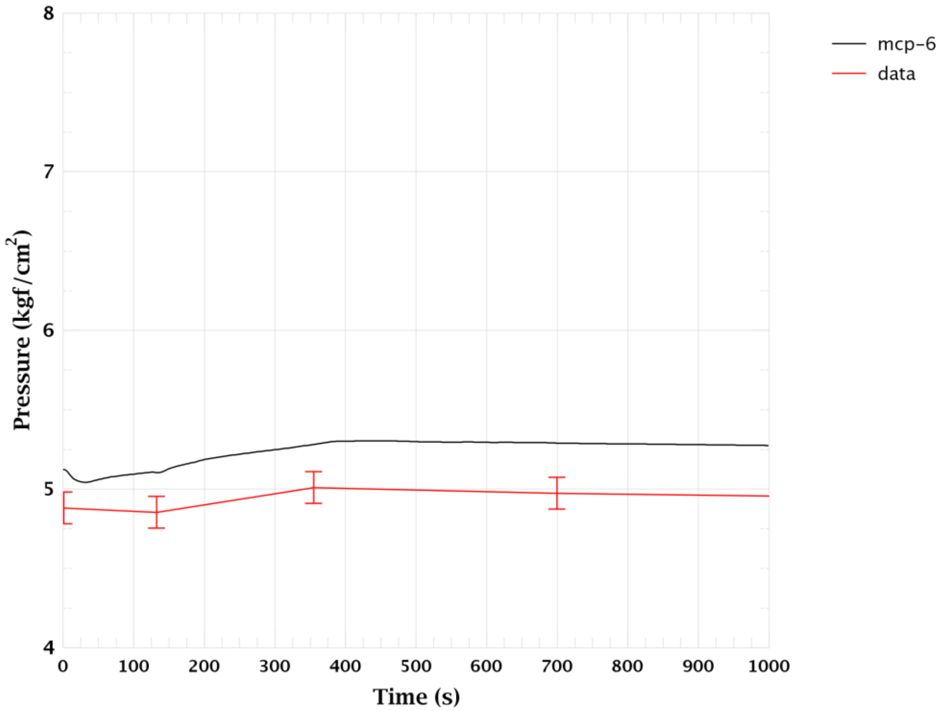



**Figure 84 MCP-6 Pressure Drop**

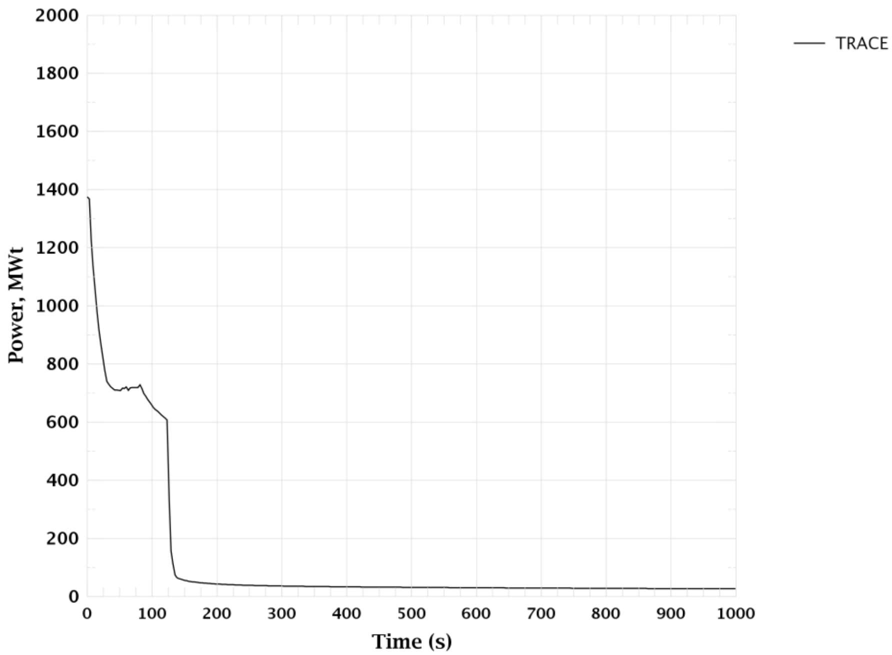



**Figure 85 Reactor Thermal Power**

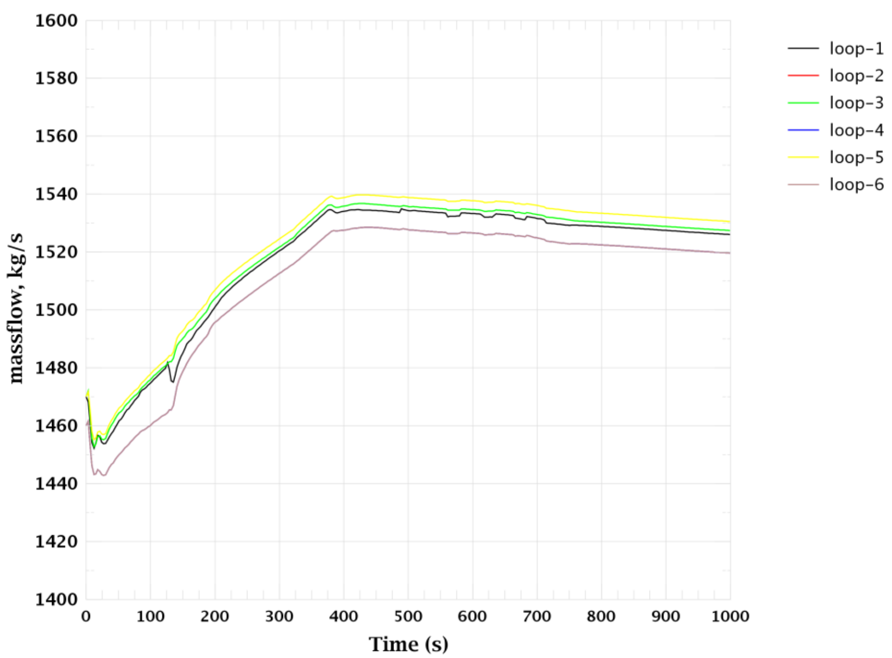



**Figure 86 MCP Mass Flow Rate**

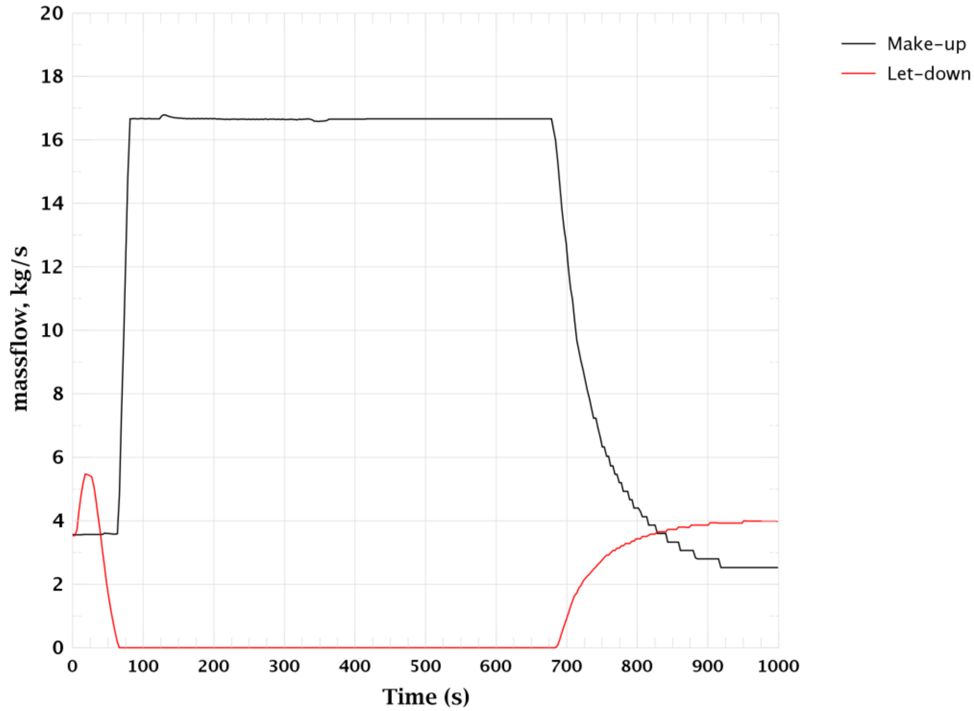



**Figure 87 Makeup – Letdown Mass Flow Rate**

### 4.1.4 Conclusion

The comparative computer simulation of "Reactor scram caused by concrete slab drop to the connection lines of house loads power supply" incident demonstrates that calculated parameters behavior is in general in good compliance with measured data.

It shall be noted that some engineering assumptions were introduced to achieve better agreement between calculated and measured data. Thus, for example, the failure of one BRU-K to close and its further forced closure by operator was modeled taking into account that the decrease of secondary pressure was observed during the incident.

The difference between calculation results and incident data is also observed in the behavior of SG level. According to NPP experts, continuous SG level increase at the third phase of incident is explained by leakage through the main feedwater control valves which was terminated (in two SGs) after forced closure of MFW valves.

Simulation of selected incident allowed to validate correctness of modelling of the main primary and secondary circuit equipment and controllers. Particularly, correctness of the following models was assessed:

- primary makeup and letdown controllers;
- primary pressure control system (PRZ heaters and spray valves);

- ARM in secondary circuit pressure maintenance mode;
- BRU-K controllers;
- MFW controllers.

## 4.2 Reactor Scram Transient Initiated by 6 kV Switch Short Circuit

In general, the incident progression is similar to the one described in Section 4.1. Before the incident, the unit operated at nominal thermal power and electric power was 380 MWe. Normal operation systems and equipment, main electric circuit and house loads power supply as well as all protections and interlocks were in design configuration for nominal power operation mode. Safety system trains No.1, 2, 3 were in standby mode. Main feed water supply was provided by pumps 1MFWP-1, 1MFWP-2, 1MFWP-4, 1MFWP-5 while MFWP-3 was under repair.

According to [6], the incident was caused by short circuit in V-11T-B switch of 6kV which resulted in trip of TG-1 and switching of TG-2 to no-load mode. Decrease of steam flow to the turbines led to an increase of the main steam header pressure up to BRU-K opening setpoint and start of steam dump to the turbine condensers. During connection to stand-by power supply MFW pumps were tripped by low busbar voltage protection that caused decrease of SG levels, increase of the primary and secondary pressure and of PRZ level.

Reactor power was initially decreased by the automatic power controller (in secondary circuit pressure maintenance mode) and by level 3 reactor protection after decrease of SG level. Manual restart of MFW pumps No.1, 2 and 5 caused decrease of secondary and primary circuit pressure with actuation of level 2 reactor protection, closure of TG-2 stop valves and reactor scram.

From the validation standpoint, this scenario is interesting since it allows to check the models of:

- the steam dump valves (BRU-A, BRU-K) and correspondent controllers;
- reactor power limiter (RPL), ARM, reactor scram and primary circuit pressure controllers (PRZ heaters and spray valves);
- PRZ level controller (primary makeup and letdown controllers).

### 4.2.1 Initial Conditions

The initial conditions selected for transient calculation correspond to those specified in Table 2.

Before the transient the unit was in nominal power operation for about 2 months since the beginning of the 22th fuel cycle. In validation calculation the reactor core characteristics which correspond to beginning of the 22th fuel cycle [4] were applied. The decay heat is calculated based on ANS-79 standard [5] with 1.0 multiplier.

### 4.2.2 Boundary Conditions

The boundary conditions for transient simulation include automatic and manual actions. Taking into account that the state of certain systems and equipment which should be in operation according to the design (e.g., secondary circuit drains, AFW pumps, SG blow-down) is not clearly described in the incident report [6], an influence of these systems operation on the main documented parameters was evaluated by performing preliminary calculations and incorporating correspondent assumptions into the final transient scenario.

Thus, for example, to match incident description provided in [6] the additional 10 sec stand-by time was introduced in ARM operation logic after the first impulse of ARM and before further insertion of the control group of control assemblies into the reactor core. This allowed to minimize discrepancies in calculated and actual ARM behavior during the accident caused by the differences in calculated and measured secondary circuit pressure response.

Moreover, the minimal primary circuit pressure reached in calculation was 115.7 kgf/cm$^2$, which is just 0.7 kgf/cm$^2$ higher (i.e., within RCS pressure measurement error) than level 2 reactor protection setpoint. In order to comply with the actual incident scenario it was assumed that level 2 reactor protection signal is formed when the minimal calculated RCS pressure value is reached.

The incident description also indicates differences in BRU-K1A,B and BRU-K2A,B operation. Thus BRU-K1A,B valve behavior corresponds to the design algorithm of BRU-K controller operation on maintaining the secondary circuit pressure within 48-50 kgf/cm$^2$ while BRU-K2A closure occurred at lower pressure. This suggests that BRU-K2A closure was initiated by forced closure interlock. Therefore BRU-K2 controller failure after first opening of the valve was postulated in transient simulation.

To take into account operation of steam dump valve to deaerator (BRU-D) an additional steam release from MSH was modeled in calculation.

As for the operation of AFW pumps and secondary circuit drains, the preliminary calculations indicated that their operation does not affect simulation results significantly and was not taken into account in final validation calculation.

Other assumptions on operation of safety systems and main equipment applied in final validation calculation are described below:

- operation of reactor control and protection system is started by the first initiating signal in compliance with design set points. ARM and level 2 reactor protection operation was corrected according to the actual incident data;
- design operation of the primary makeup/letdown and primary pressure control system is modeled;
- TG trip is modeled according to the incident scenario: TG-1 stop valves are closed due to a short circuit in V-11T-B switch of 6kV (initiating event) and TG-2 is switched to no-load operation. Further closure of TG-2 stop valves occurs according to the design set points;
- MFW pumps are initially tripped at the beginning of transient. Further manual restart of three MFWP was modeled according to the incident description;
- SG feedwater flow is controlled by operation of MFW controllers according to their design characteristics;
- AFW pumps operation is not considered;

- BRU-Ks operation according to design set points and adjustment coefficients is modeled. Failure of BRU-K2 controller is postulated after its first opening. Further operation of BRU-K2 is controlled by forced closure/opening interlocks;
- BRU-D operation is simulated as additional steam release from MSH.

### 4.2.3 Calculation Results

Transient simulation of selected scenario was performed for 360 s from the initiating event occurrence.

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

**Table 4 Sequence of Events**

<table>
<tr><th colspan="2">Time, sec</th><th rowspan="2">Event in the incident</th><th rowspan="2">Event in calculation</th></tr>
<tr><th>Incident</th><th>Calculation</th></tr>
<tr><td>0.0</td><td>0.0</td><td>Short circuit in V-11T-B switch of 6kV. TG-1 stop valves closure by protection without time delay. TG-2 switches to no-load mode. Secondary pressure started to increase</td><td>Closure of TG-1 stop valves.<br>TG-2 switches to no-load mode.<br>Boundary condition</td></tr>
<tr><td>1.0</td><td>1.0</td><td>Trip of 1MFWP-1, 2, 4, 5 (1MFWP-3, which according to the design had to remain in operation was under repair)</td><td>MFWP trip. Boundary condition</td></tr>
<tr><td>2.0</td><td>3.0</td><td colspan="2">ARM interlock to switch to the secondary circuit pressure maintenance mode. Start of control assemblies insertion into the core with normal operating speed</td></tr>
<tr><td>3.0</td><td>2.0 – 2.5</td><td colspan="2">Start of BRU-K1A, B and BRU-K2A, B operation</td></tr>
<tr><td>5.0</td><td>6.1</td><td colspan="2">PRZ heaters groups no.1,2 are OFF (RCS pressure <126 kgf/cm$^2$)</td></tr>
<tr><td>9.0</td><td>9.0</td><td colspan="2">Opening of PRZ spray valve no.1 (KO10-1) at RCS pressure increase up to 128 kgf/cm$^2$</td></tr>
<tr><td>12.0</td><td>12.0</td><td>At 87.1% of thermal power ARM was switched to standby mode according to the design algorithm</td><td>Modeled as a boundary condition to match incident description provided in [6]</td></tr>
<tr><td>12.0</td><td>10.0</td><td colspan="2">Opening of PRZ spray valve no.2 (KO10-2) at RCS pressure increase up to 129 kgf/cm$^2$</td></tr>
<tr><td>13.0</td><td>12.0</td><td colspan="2">Opening of PRZ spray valve no.3 (KO10-3) at RCS pressure increase up to 130 kgf/cm$^2$</td></tr>
<tr><td>-</td><td>14.0</td><td></td><td>Opening of PRZ spray valve no.4 (KO10-4) at RCS pressure increase up to 133 kgf/cm$^2$</td></tr>
<tr><td>15.0</td><td>42.0</td><td colspan="2">Actuation of "Level decrease for 140 mm from nominal" signal for SG-1</td></tr>
</table>

| Time, sec | | Event in the incident | Event in calculation |
|---|---|---|---|
| Incident | Calculation | | |
| 18.0 | 44.0 | Actuation of "Level decrease for 140 mm from nominal" signal for SG-2, SG-6 | |
| 22.0 | - | The 2nd impulse from ARM (duration of 1 sec) for decrease of reactor power | Not modeled in calculation (see boundary conditions) |
| 23.0 | - | At 85.2% of thermal power ARM was switched to standby mode according to the design algorithm | |
| 24.0 | 42.0 | Actuation of "Level decrease for 140 mm from nominal" signal for SG-3 | |
| 29.0 | 43.0 | Actuation of "Level decrease for 140 mm from nominal" signal for SG-4 | |
| 29.0 | 30.0 | Actuation of level 3 reactor protection at Nт = 84.2% due to level decrease in 2 of 6 SGs (SG-2 and SG-4) for 200 mm below nominal (downward sequential insertion of control assemblies' groups into the core with operating speed) | Insertion of control group of control assemblies into the core by ARM, with further actuation of level 3 reactor protection (at ~50 s) due to level decrease in 2 of 6 SGs for 200 mm below nominal |
| - | 55.0 | | Closure of PRZ spray valve no.4 (KO10-4) at RCS pressure decrease down to 130 kgf/cm$^2$ |
| 34.0 | 57.0 | Closure of PRZ spray valve no.3 (KO10-3) at RCS pressure decrease down to 129 kgf/cm$^2$ | |
| 37.0 | 58.0 | Closure of PRZ spray valve no.2 (KO10-2) at RCS pressure decrease down to 128 kgf/cm$^2$ | |
| 37.0 | 62.0 | Manual start of MFWP-1 | Boundary condition |
| 39.0 | 46.5 | Closure of PRZ spray valve no.1 (KO10-1) at RCS pressure decrease down to 127 kgf/cm$^2$ | |
| 39.0 -42,0 | 49.5 | Manual start of MFWP-2, MFWP-5 | Modeled as restart of "double" MFWP |
| 45 - 48 | 58 | PRZ heaters groups no.1 and 2 are ON (decrease of RCS pressure down to 124 kgf/cm$^2$) | |
| 51 | 62-79 | PRZ heaters group no.3 is ON (decrease of RCS pressure down to 122 kgf/cm$^2$) | PRZ heaters groups no.3, 4, 5 are ON (decrease of RCS pressure down to 122, 121 and 118 kgf/cm$^2$) |
| 52 | 52 | BRU-D2B opened upon pressure decrease in deaerator | In calculation, operation of BRU-D2B was modeled by additional steam release from MSH |

| Time, sec | | Event in the incident | Event in calculation |
|---|---|---|---|
| Incident | Calculation | | |
| 58 | 107 | BRU-K1A, B closure at MSH pressure decrease down to 49 kgf/cm$^2$ | In the calculation, BRU-K1A, B closure occurred due to controller operation according to its static characteristics 48-50 kgf/cm$^2$ |
| 64 | 70 | Closure of RCS let-down control valve VV2-6 due to decrease of PRZ level | |
| 66 | 73 | End of level 3 reactor protection operation. Reactor power is ≈34.6 % | End of level 3 reactor protection operation. Reactor power is ≈26.6 % |
| 66 | 63 | PRZ heaters group no.4 is ON (decrease of RCS pressure down to 121 kgf/cm$^2$) | |
| 76 | 78 | Opening of RCS make-up control valve (at decrease of PRZ level for 300 mm below nominal) | |
| 89 | 89 | PRZ heaters group no.5 is ON (decrease of RCS pressure down to 118 kgf/cm$^2$) | |
| 94 - 97 | 112 | BRU-K2A, B closure at MSH pressure decrease down to 47 kgf/cm$^2$ | Closure of BRU-K2A, B by interlock at MSH pressure of 46 kgf/cm$^2$ |
| 118 | 140 | Actuation of level 2 reactor protection (set no.1) due to decrease of RCS pressure down to 115 kgf/cm$^2$ | Actuation of level 2 reactor protection signal at minimal RCS pressure (115.7 kgf/cm$^2$) |
| 119 | | Actuation of level 2 reactor protection (set no.2) due to decrease of RCS pressure down to 115 kgf/cm$^2$ | |
| 179 | 179 | Due to pressure decrease in deaerators, BRU-D1A,B opened at MSH pressure of 44 kgf/cm$^2$ | In calculation, operation of BRU-D1A,B was modeled by additional steam release from MSH |
| 328 | 328 | Manual closure of all BRU-D due to decrease of MSH pressure | Termination of additional steam release from MSH. Boundary condition |
| 329 | 328 | TG-2 trip due to decrease of pressure upstream the turbine control valves below 40 kgf/cm$^2$ | |
| 338 | 328 | Reactor scram due to closure of last operating turbine stop valves. Start of RCS boration by operators. | In the calculation, scram signal was generated upon closure of stop valves of the last operating turbine without time delay |
| 360 | 360 | Stabilization of the main primary and secondary parameters | End of calculation |

Beginning of transient is characterized by a decrease of steam flow to the turbines and increase of the secondary circuit pressure that resulted in opening of BRU-Ks (Fig. 119) and switching of ARM to the secondary circuit pressure maintenance mode. Disbalance between generated and removed energy causes increase of RCS pressure (Fig. 88), temperature (Fig. 92 – Fig. 103) and of PRZ level (Fig. 106). As RCS pressure increases the 1$^{st}$ group of PRZ heaters switches off (Fig. 121) and PRZ spray valves no.1,2,3 are sequentially opened (Fig. 122). Trip of MFW pumps causes SG level decrease (see Fig. 107 – Fig. 112) and actuation of level 3 reactor protection. Reactor power decrease by ARM and level 3 reactor protection as well as spray operation result in RCS pressure decrease, closure of PRZ spray valves and sequential actuation of PRZ heater groups (Fig. 121). Since the average coolant temperature decreases an initial increase of PRZ level is changed to decreasing trend (Fig. 106) that leads to RCS letdown valves closure and opening of makeup valves (Fig. 115).

After manual start of 1MFWP-1, 2, 5 (see Fig. 117) restoration of level in all SGs begins and level 3 reactor protection signal is deactivated. Combined effect of reactor power decrease, continuous steam dump and MFW restoration causes decrease of secondary circuit pressure with closure of BRU-K (Fig. 119).

Since RCS pressure decrease could not be terminated by operation of PRZ heaters the setpoint of level 2 reactor protection (RCS pressure decrease down to 115 kgf/cm$^2$) was reached leading to successive insertion of control assemblies' groups into the reactor core. As it is described in Section 4.2.2, correspondent pressure setpoint was not reached in calculation and level 2 reactor protection was modeled at the minimal calculated RCS pressure value of 115.7 kgf/cm$^2$. Difference between this value and the actual setpoint lies within the pressure measurement error band.

After operation of level 2 reactor protection the reactor power decreases to the decay heat level (Fig. 113) that causes decrease in MSH pressure (Fig. 90) and subsequent closure of TG-2 stop valves. Finally, after manual closure of BRU-D valves the secondary circuit pressure is stabilized at ~40 kgf/cm$^2$.

Increasing and stabilization of RCS pressure (Fig. 88) following its decrease from actuation of level 2 reactor protection is caused by operation of PRZ heaters and of make-up system.

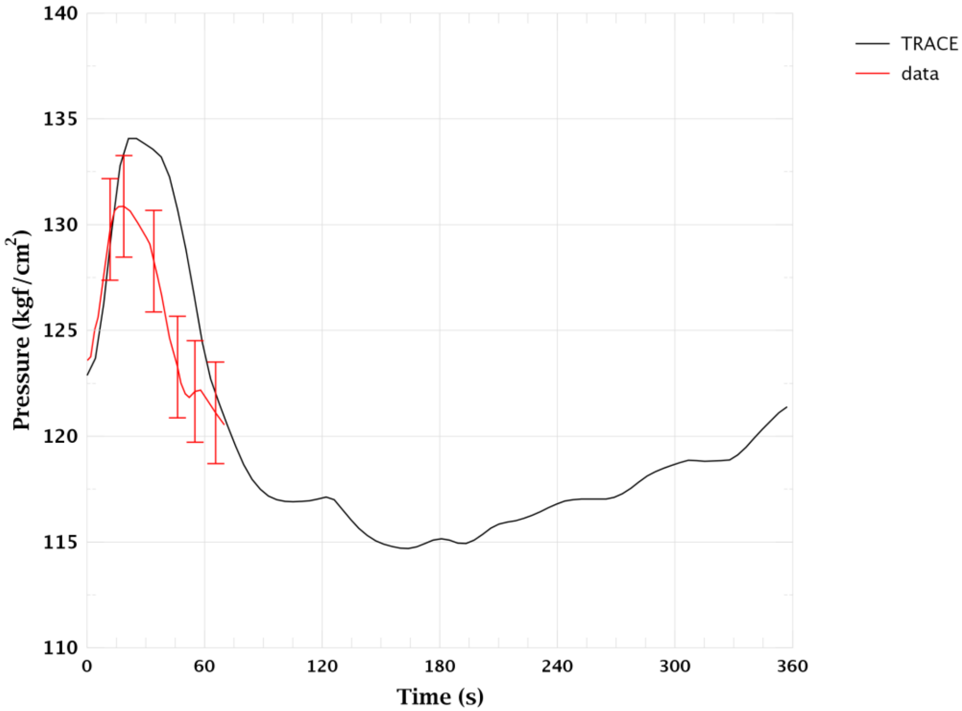



**Figure 88 PRZ Pressure**

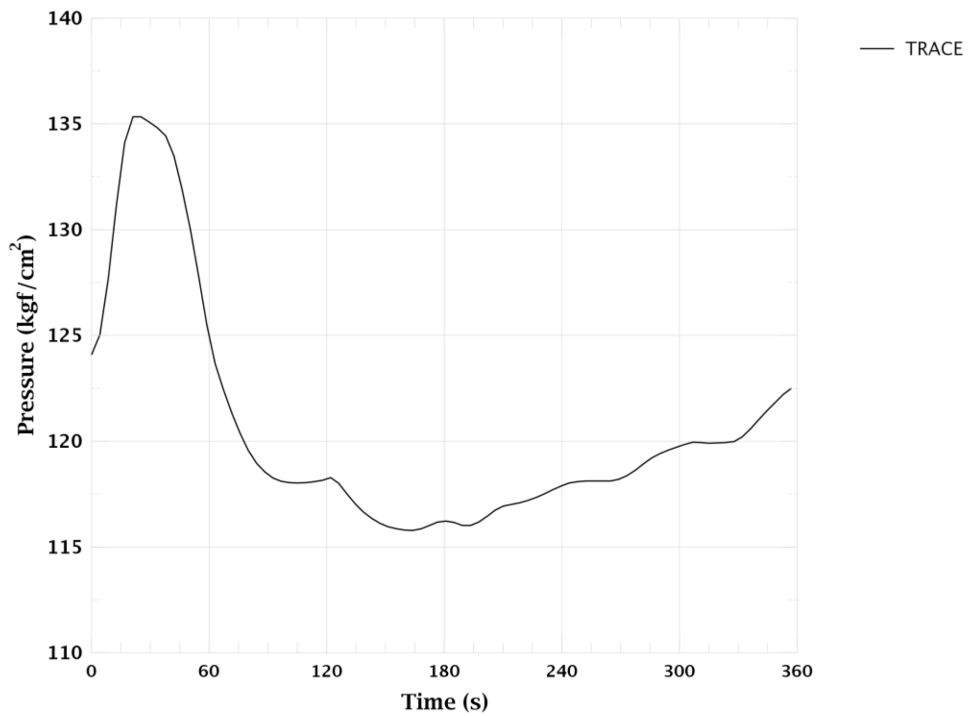



**Figure 89 Reactor Outlet Pressure**

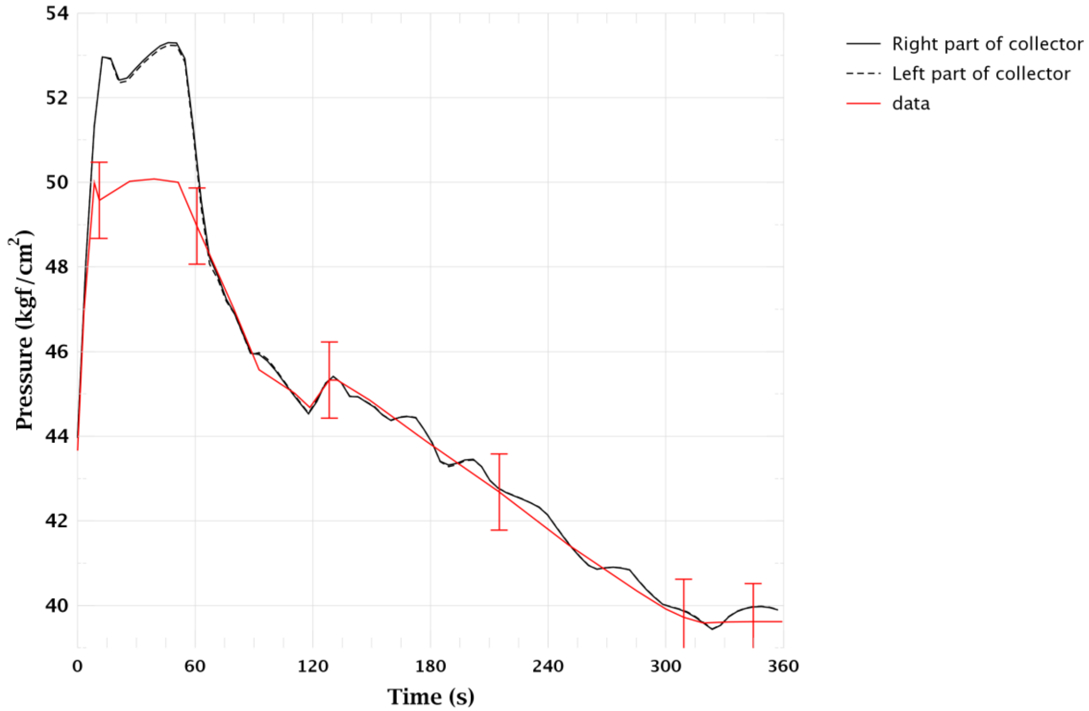



**Figure 90 MSH Semi-Headers Pressure**

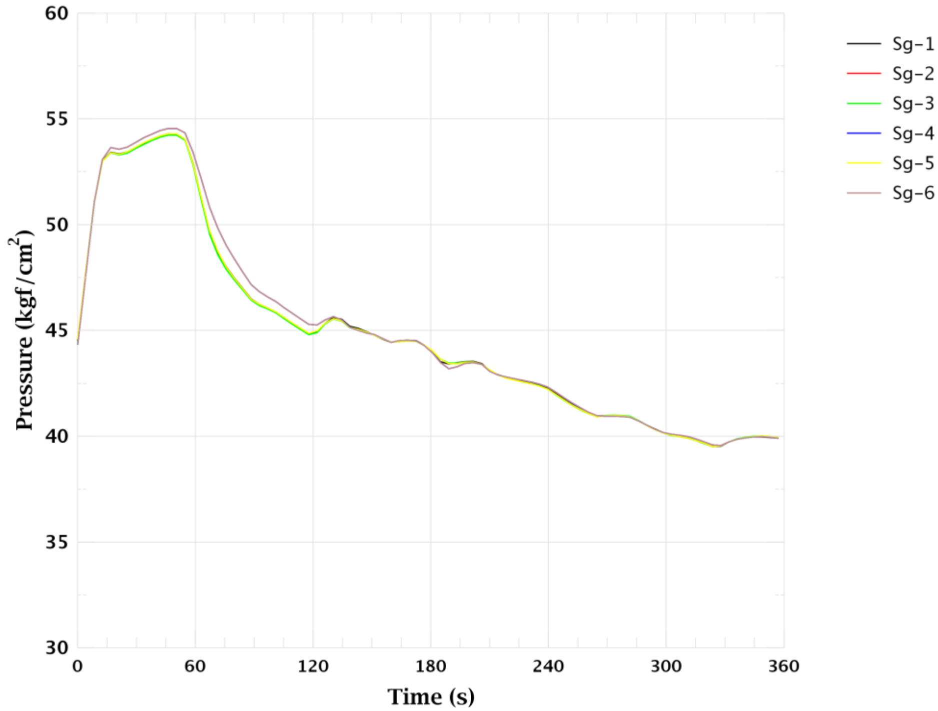



**Figure 91 SG Pressure**

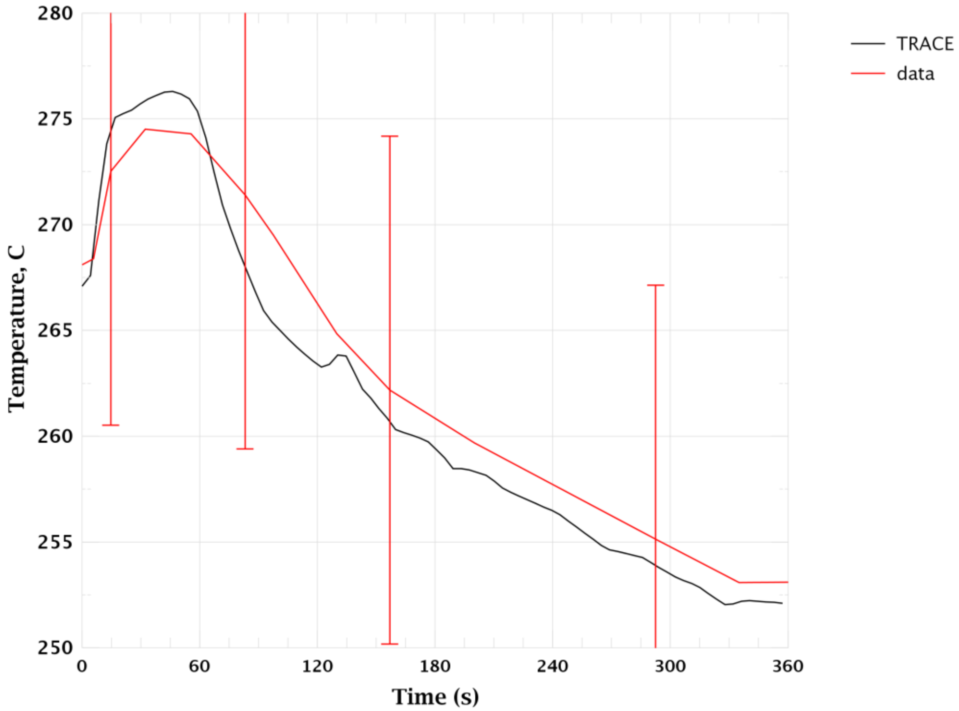



**Figure 92 Cold Leg No. 1 Temperature**

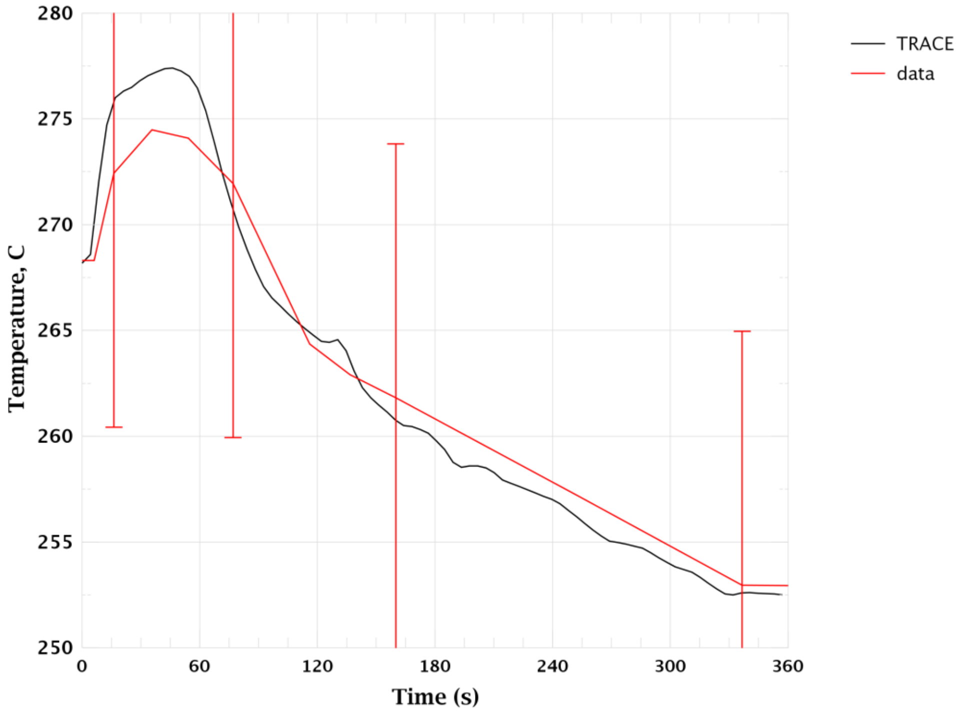



**Figure 93 Cold Leg No. 2 Temperature**

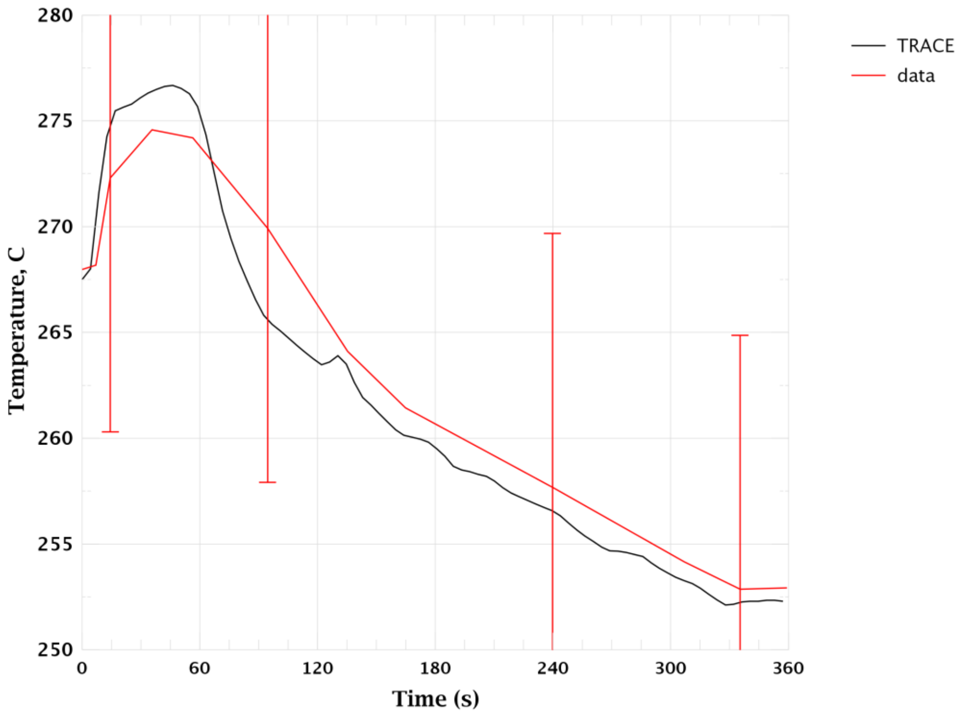



**Figure 94 Cold Leg No. 3 Temperature**

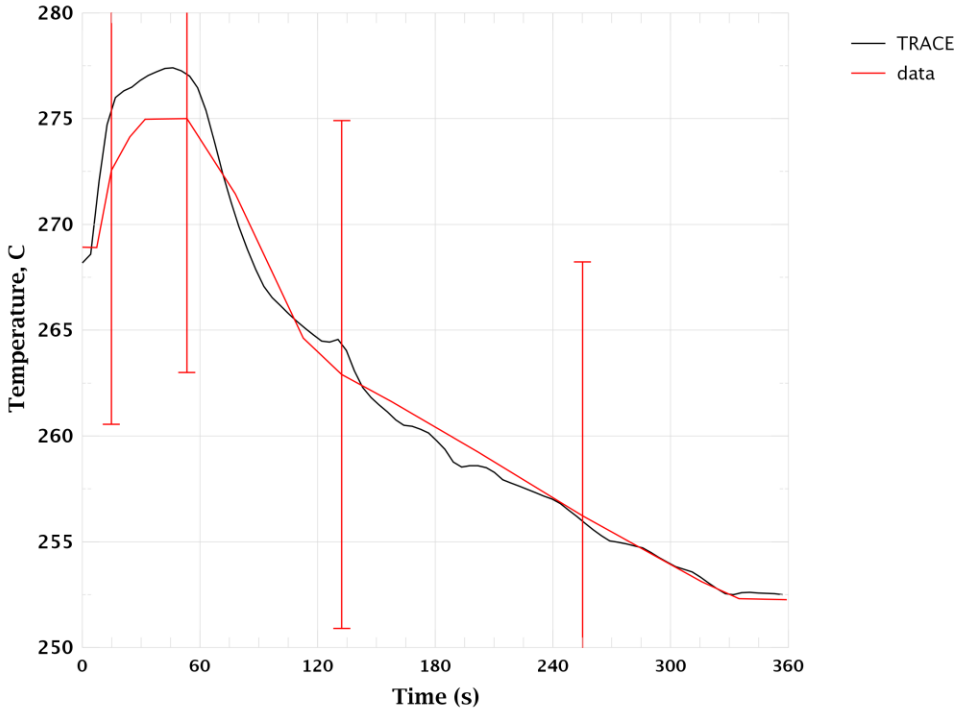



**Figure 95 Cold Leg No. 4 Temperature**

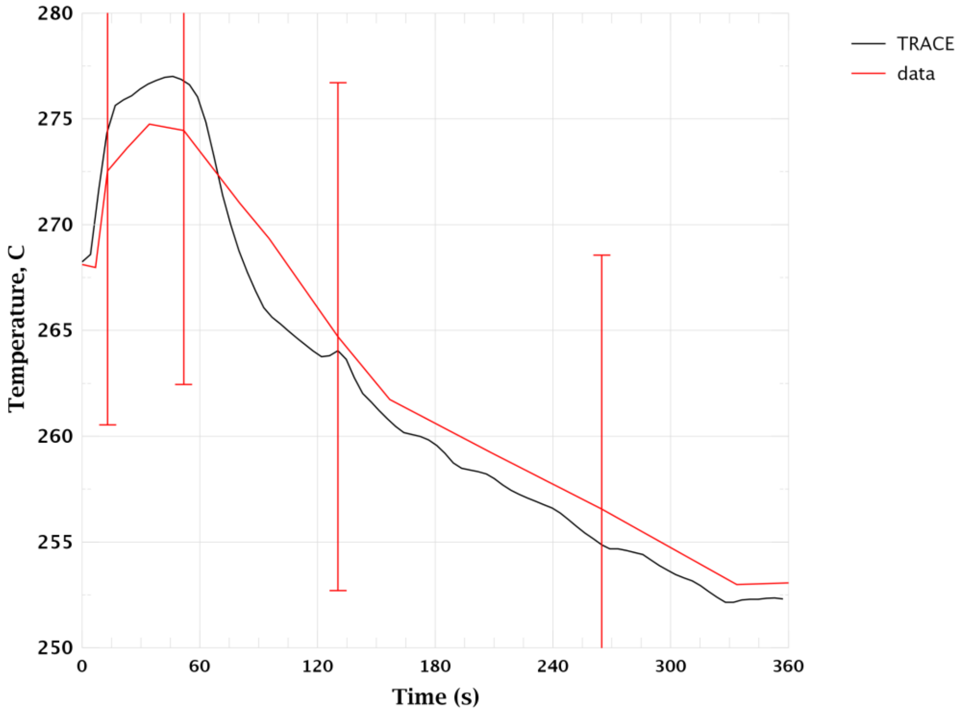



**Figure 96 Cold Leg No. 5 Temperature**

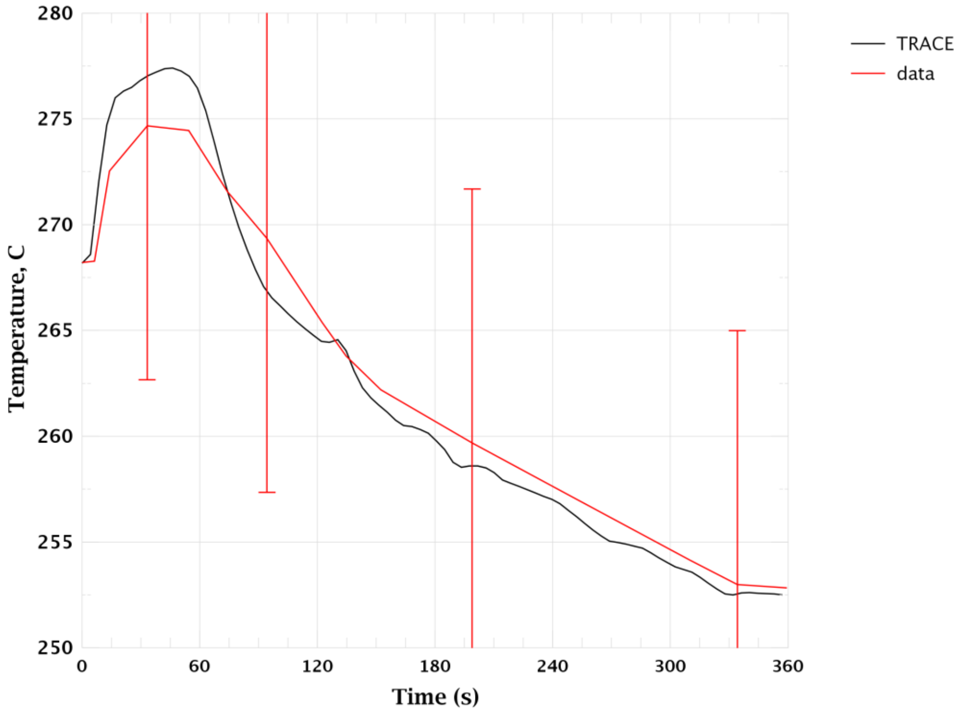



**Figure 97 Cold Leg No. 6 Temperature**

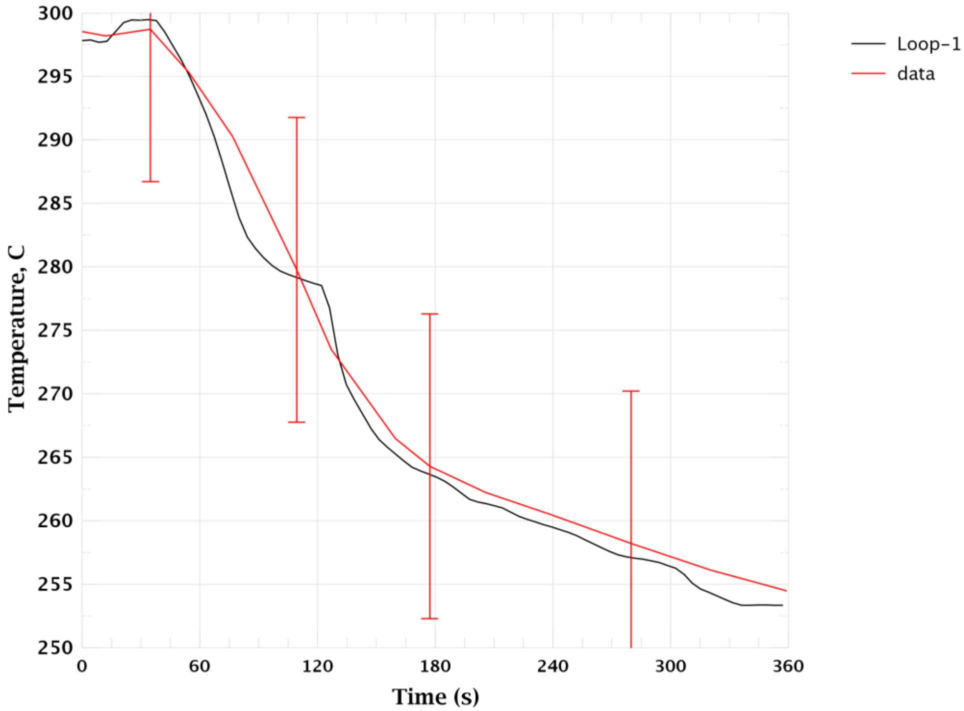



**Figure 98 Hot Leg No. 1 Temperature**

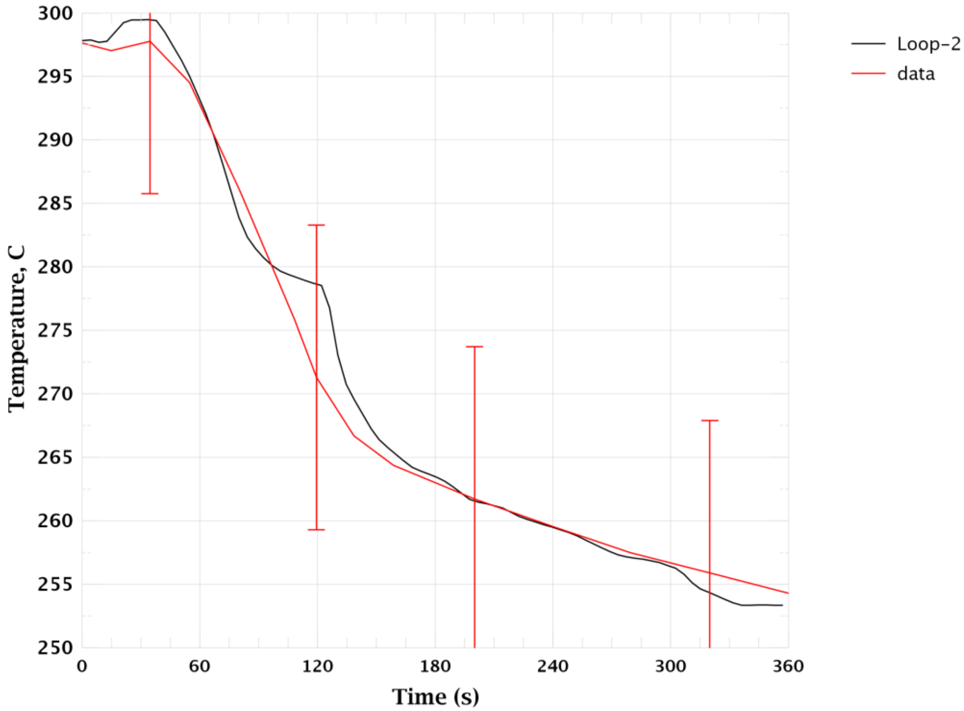



**Figure 99 Hot Leg No. 2 Temperature**

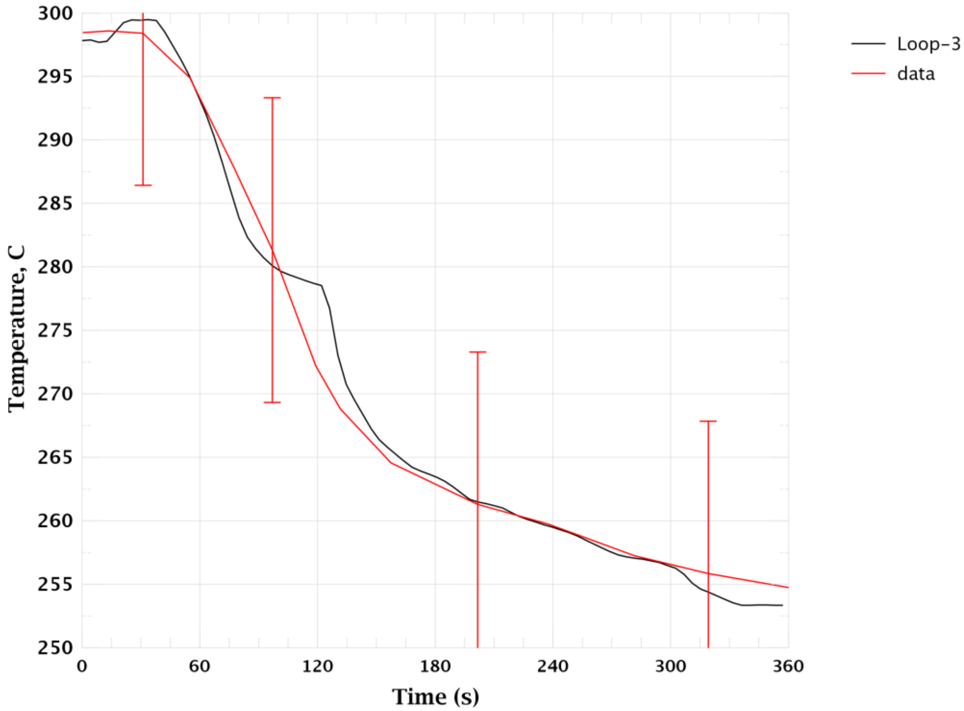



**Figure 100 Hot Leg No. 3 Temperature**

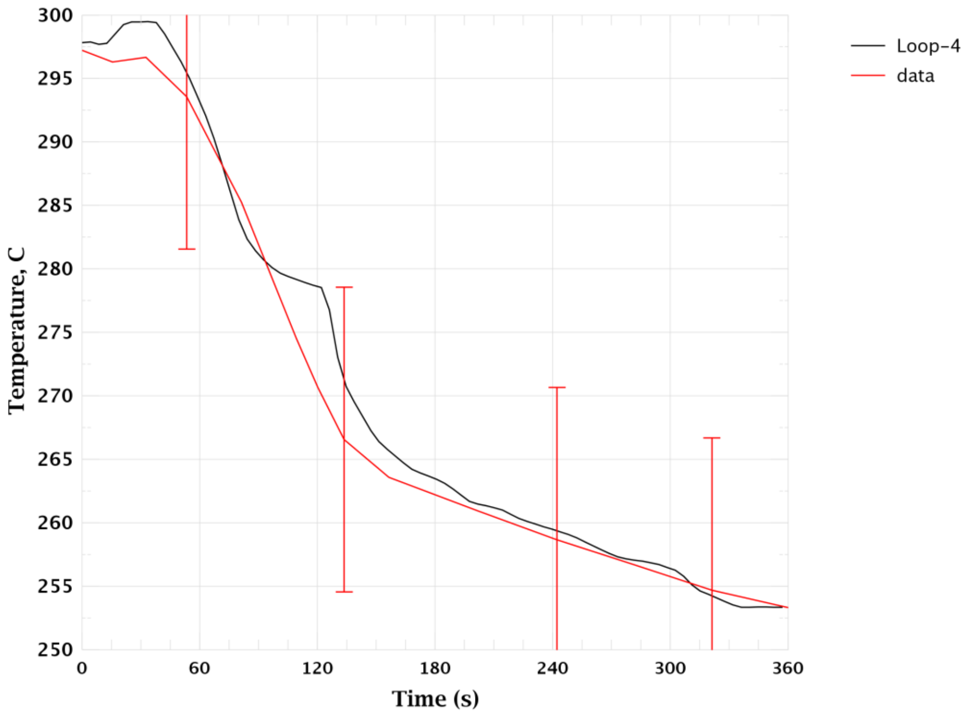



**Figure 101 Hot Leg No. 4 Temperature**

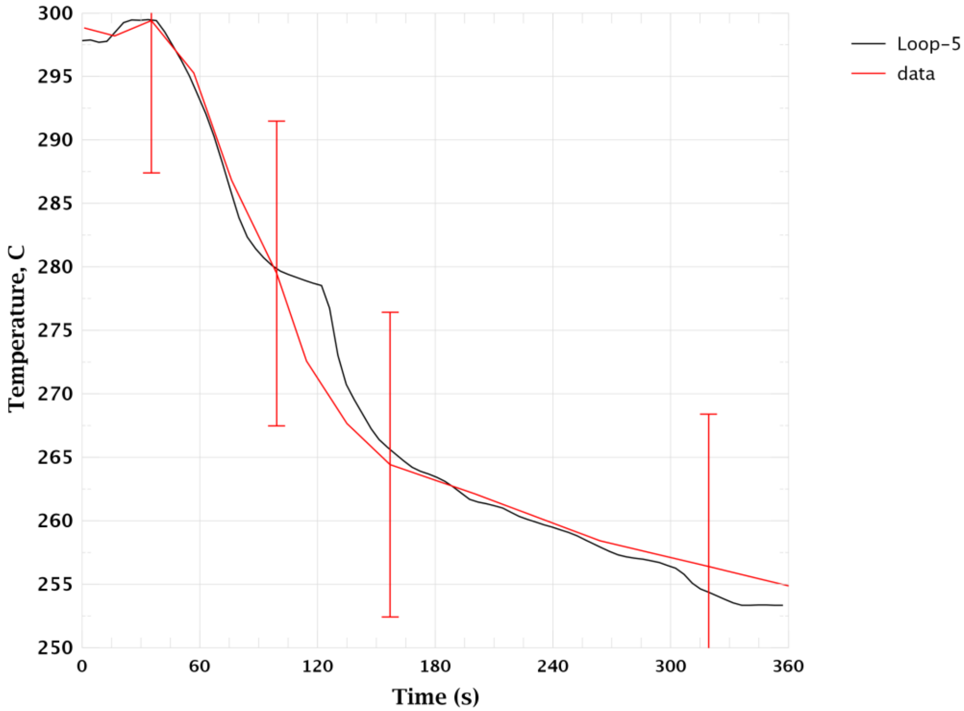



**Figure 102 Hot Leg No. 5 Temperature**

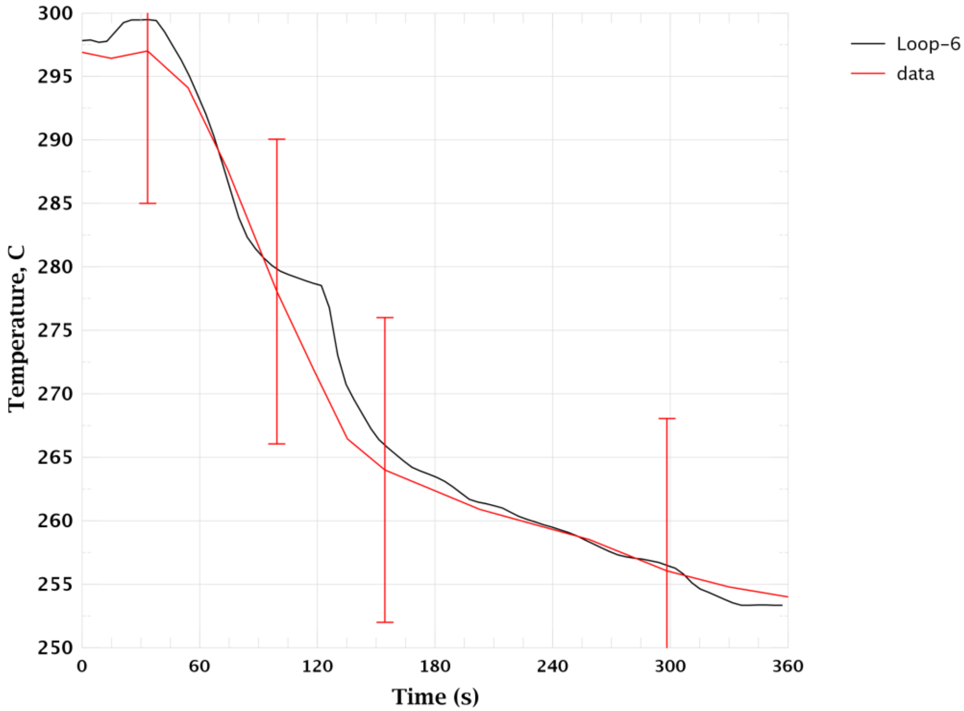



**Figure 103 Hot Leg No. 6 Temperature**

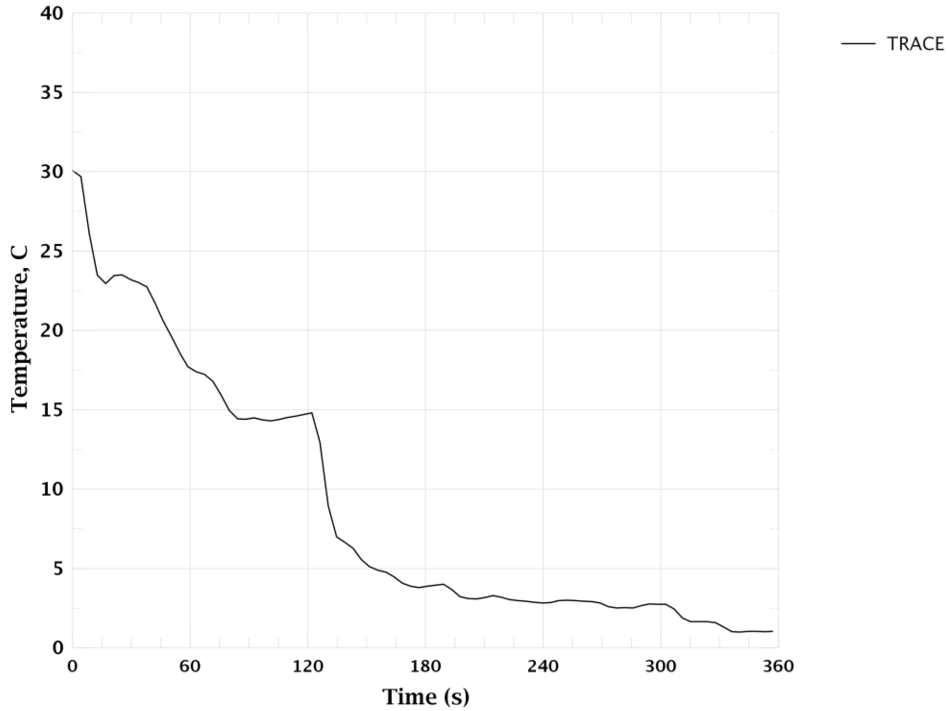



**Figure 104 Reactor Coolant Heat-Up**

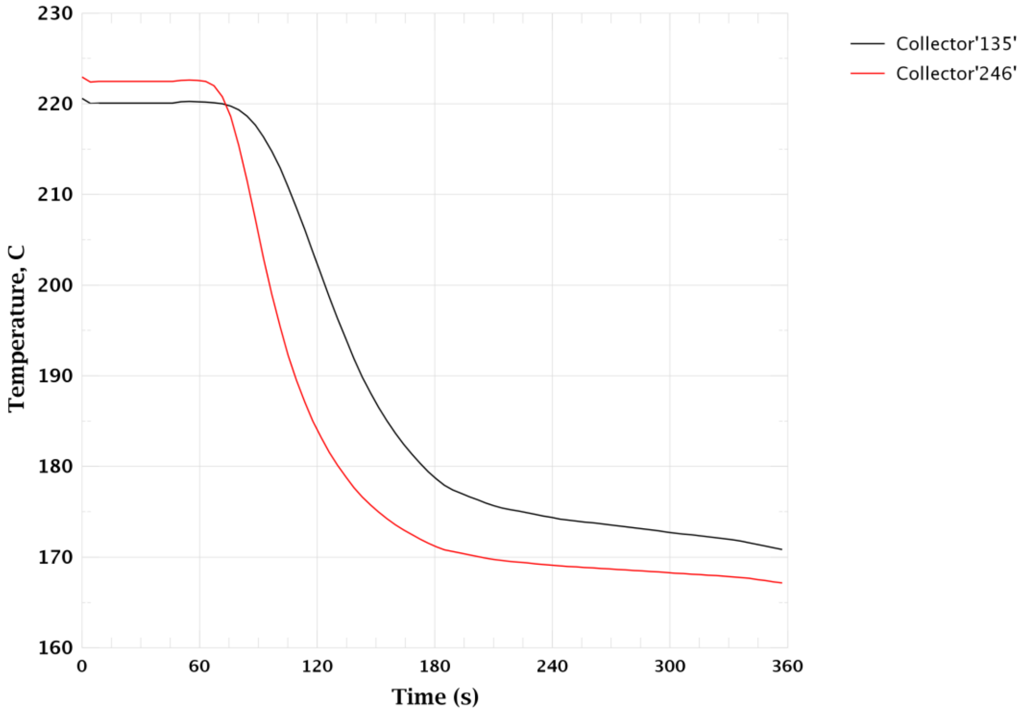



**Figure 105 Feedwater Temperature in MFW Semi-Headers**

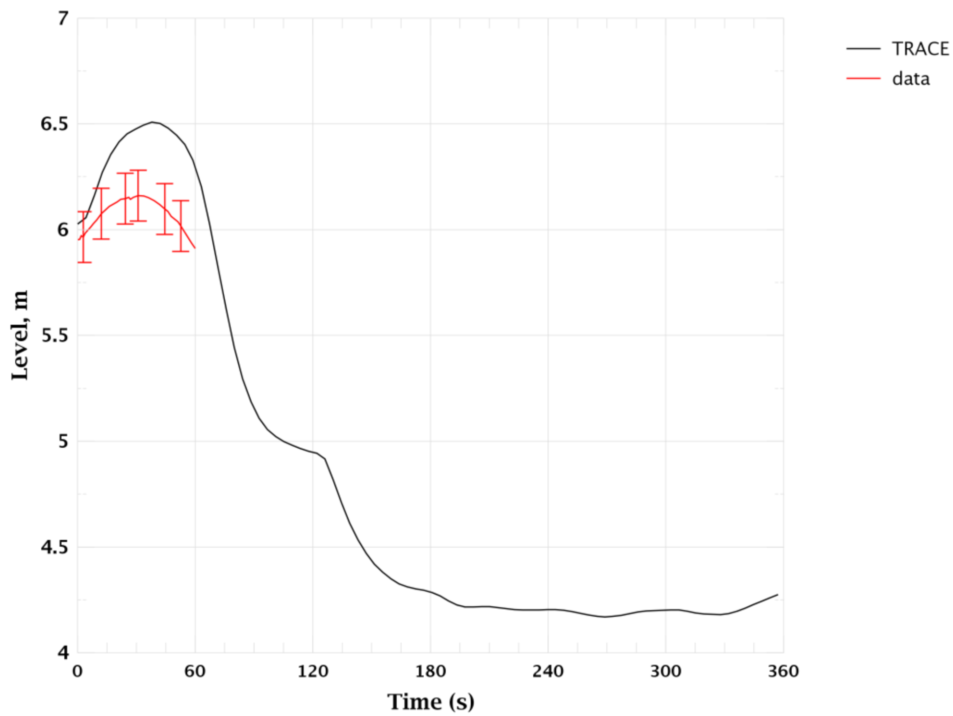



**Figure 106 PRZ Level**

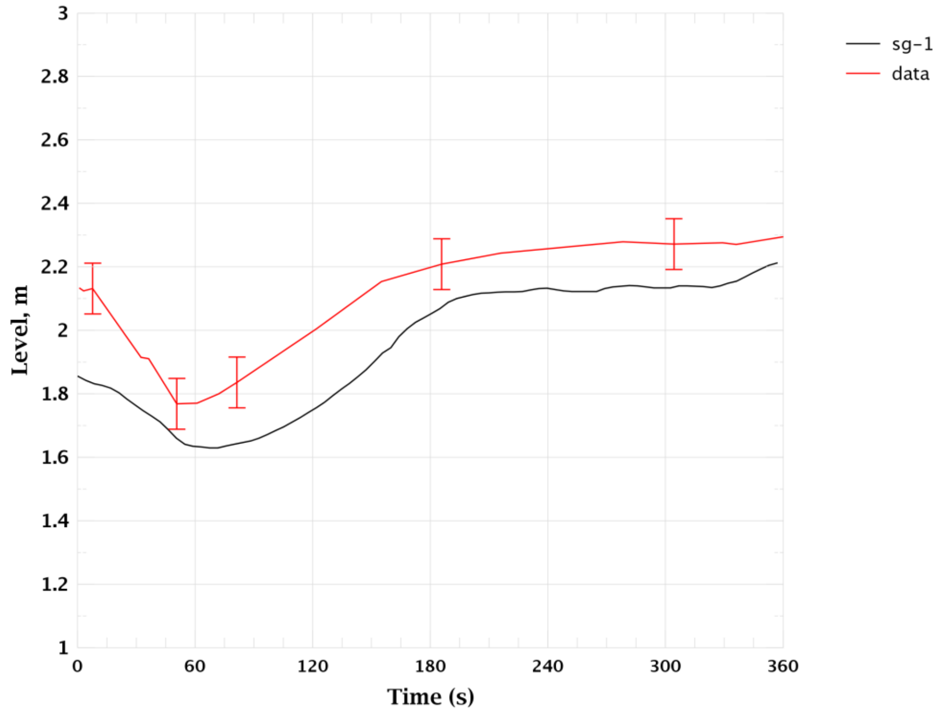



**Figure 107 SG-1 Water Level (wide range measurement)**

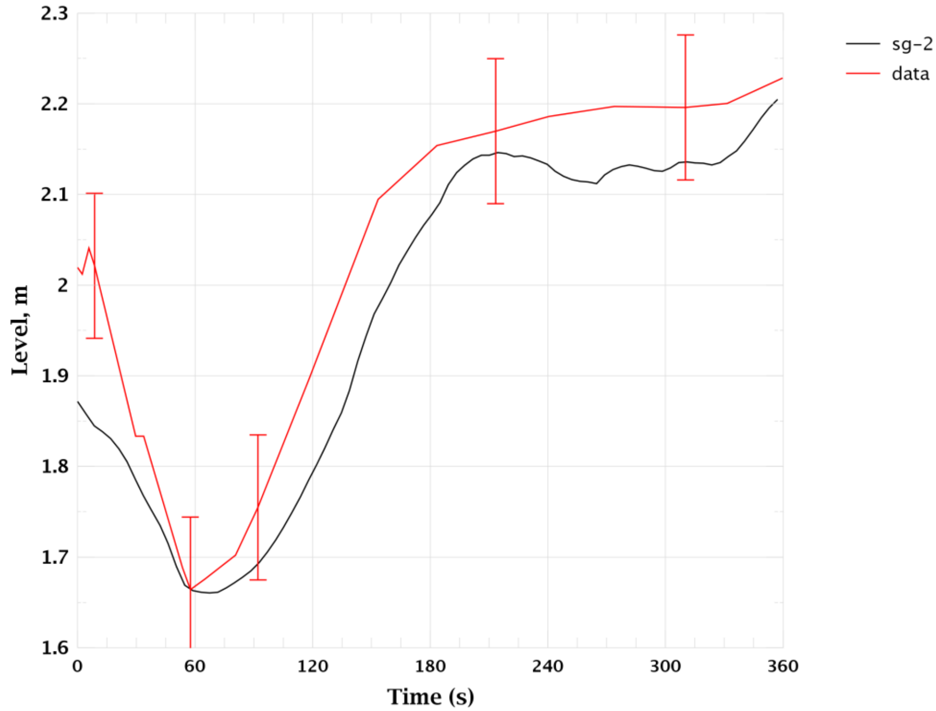



**Figure 108 SG-2 Water Level (wide range measurement)**

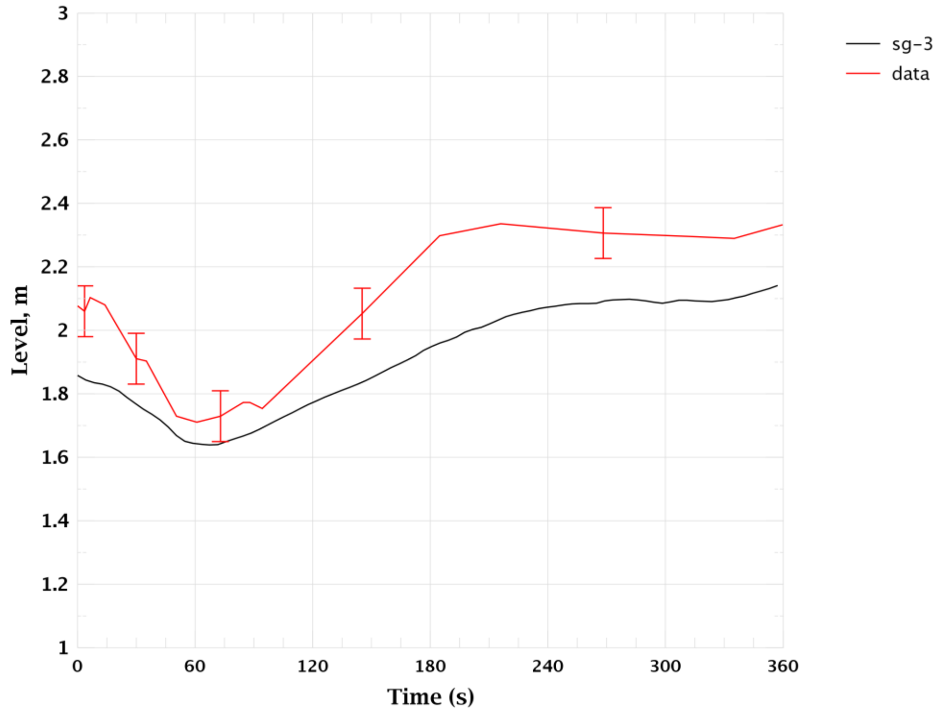



**Figure 109 SG-3 Water Level (wide range measurement)**

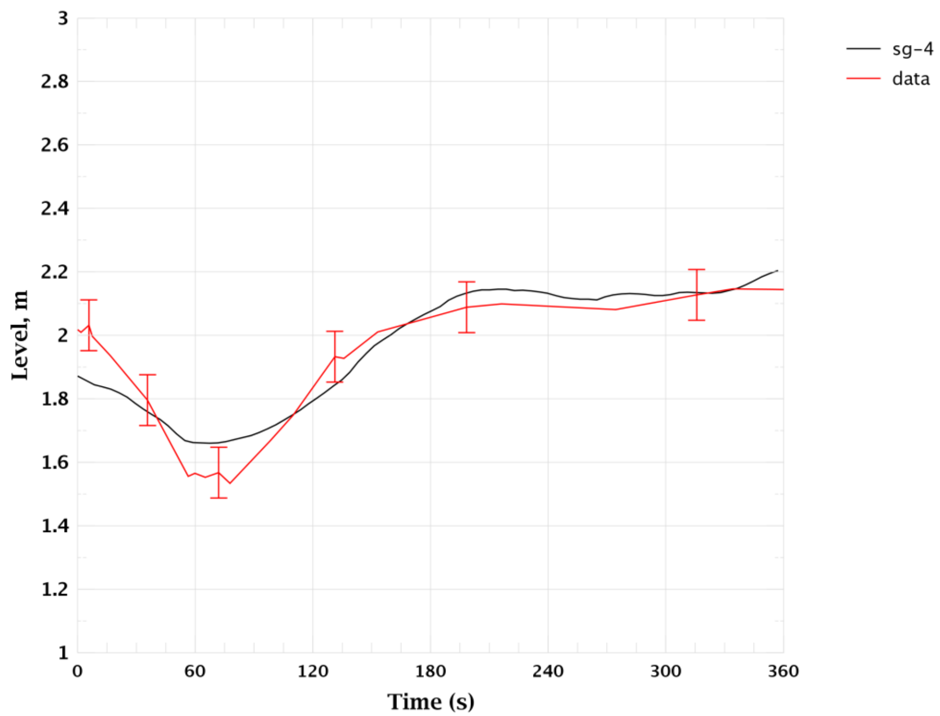



**Figure 110 SG-4 Water Level (wide range measurement)**

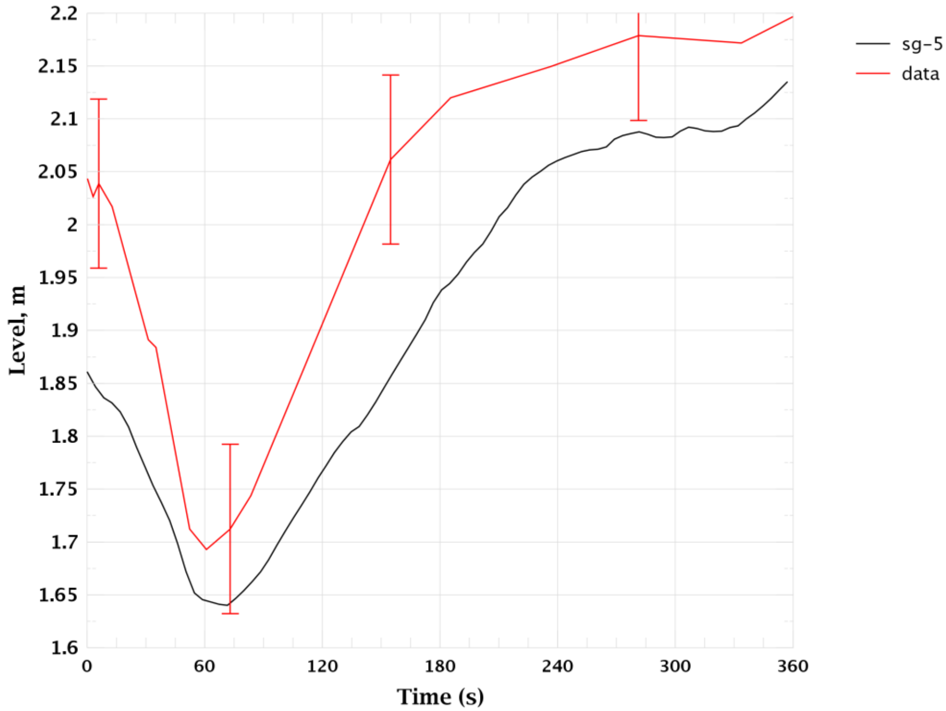



**Figure 111 SG-5 Water Level (wide range measurement)**

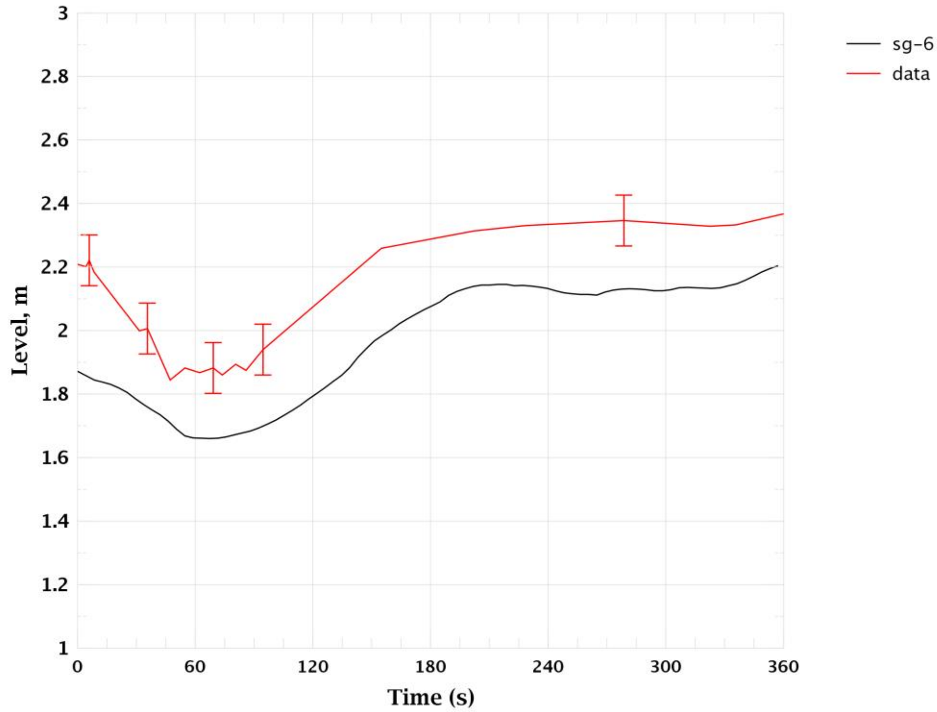



**Figure 112 SG-6 Water Level (wide range measurement)**

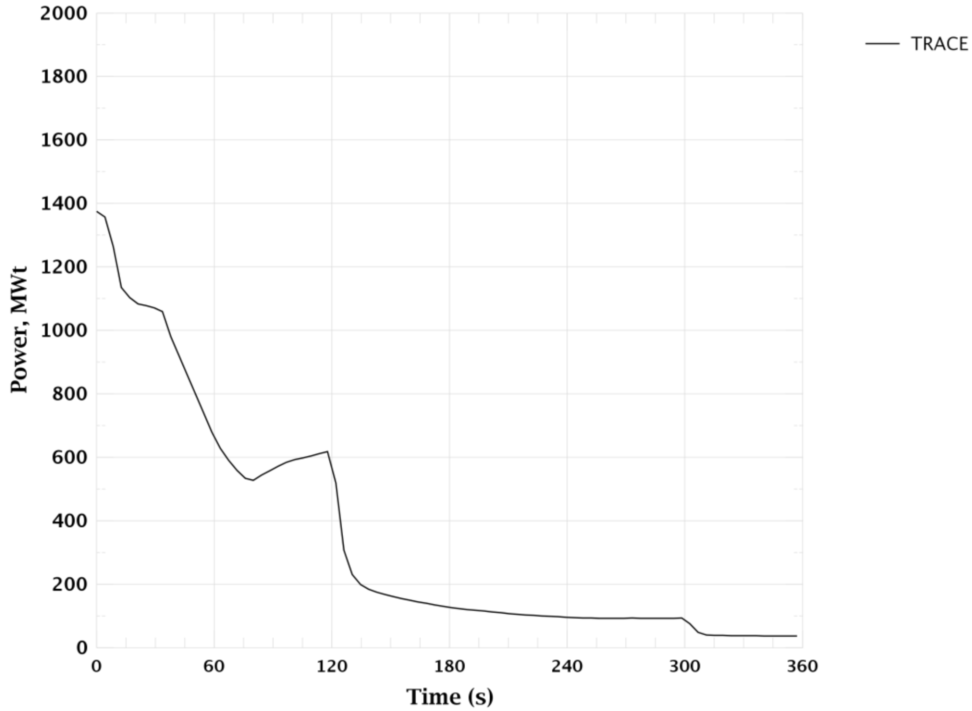



**Figure 113 Reactor Thermal Power**

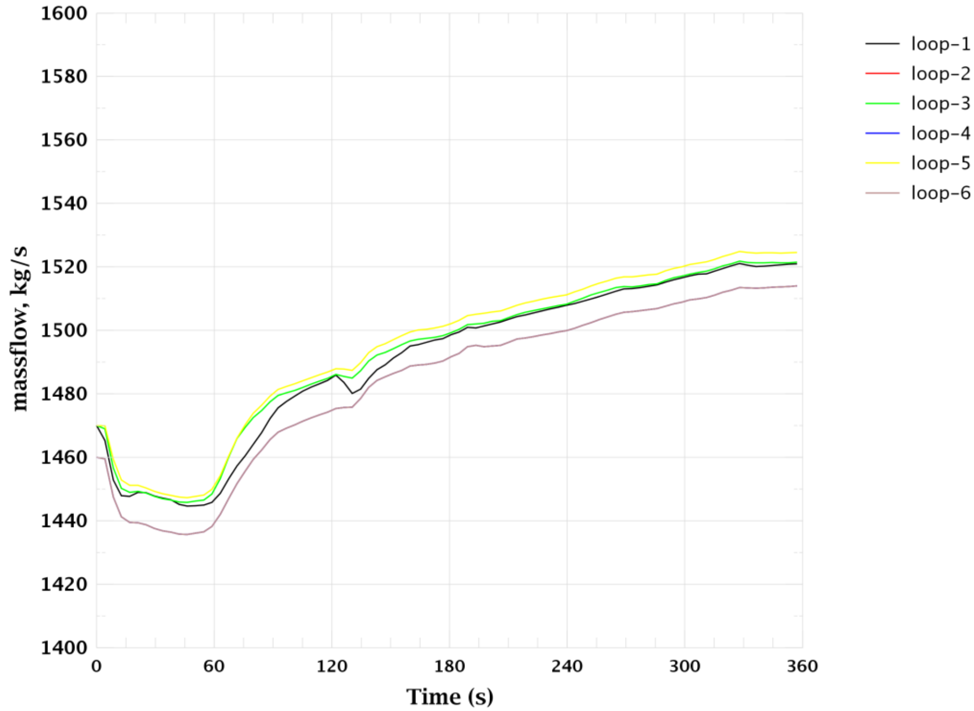



**Figure 114 RCS Loops Mass Flow Rate**

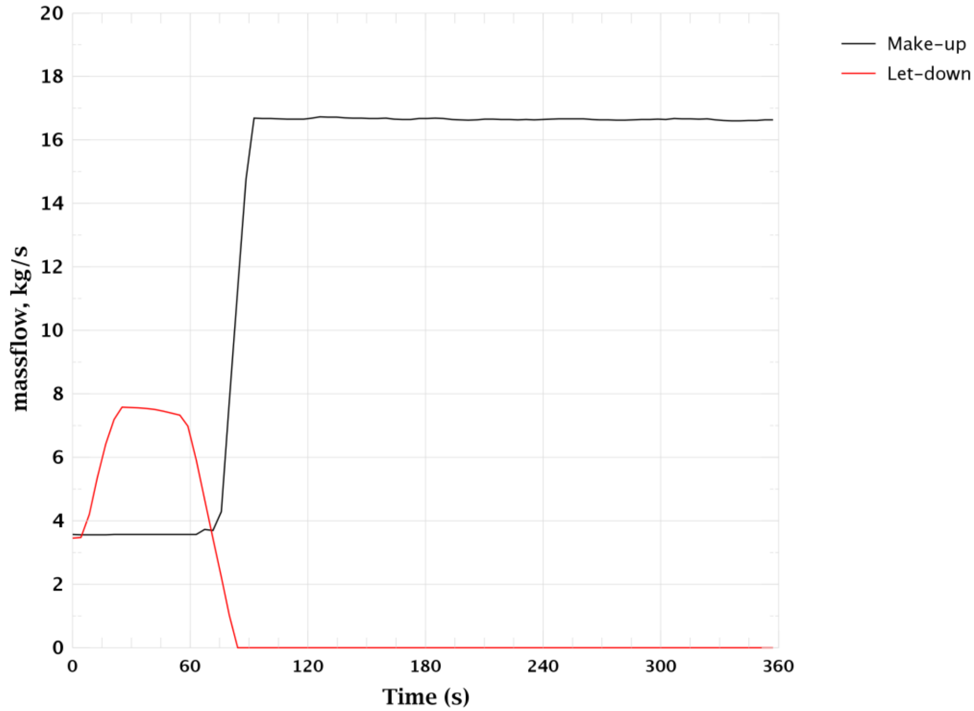



**Figure 115 Makeup-Letdown Mass Flow Rate**

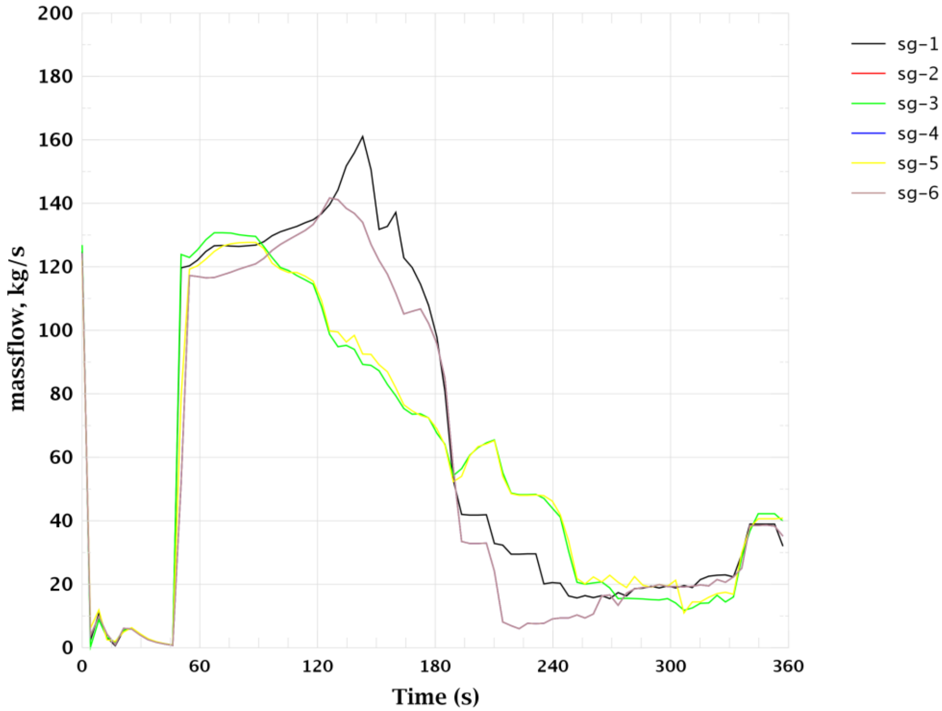



**Figure 116 Feedwater Mass Flow Rate to SG**

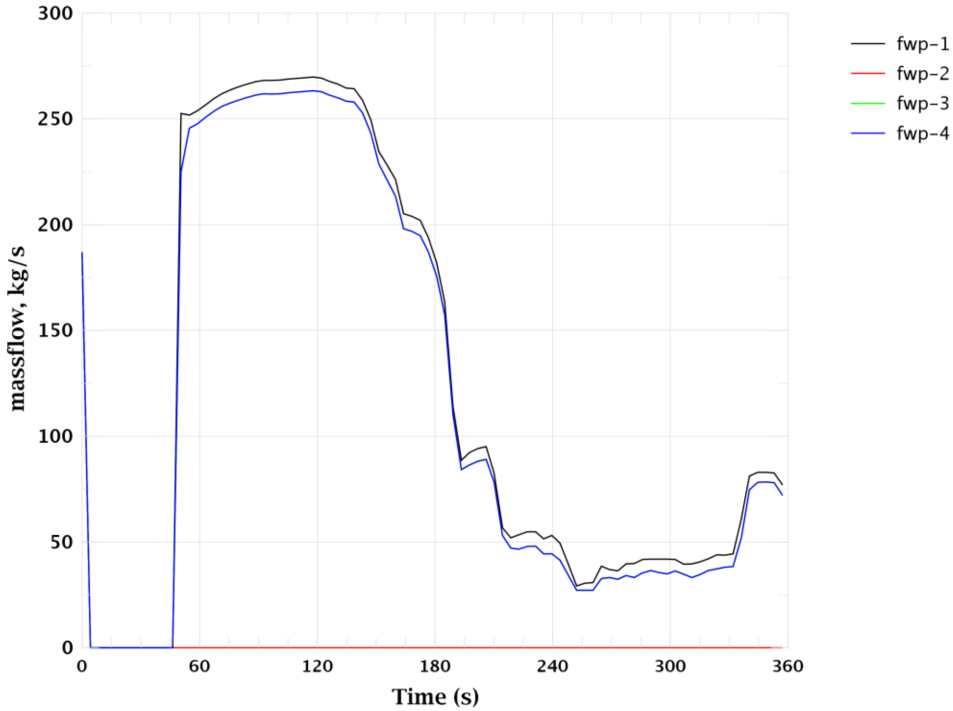



**Figure 117 MFW Pumps Mass Flow Rate**

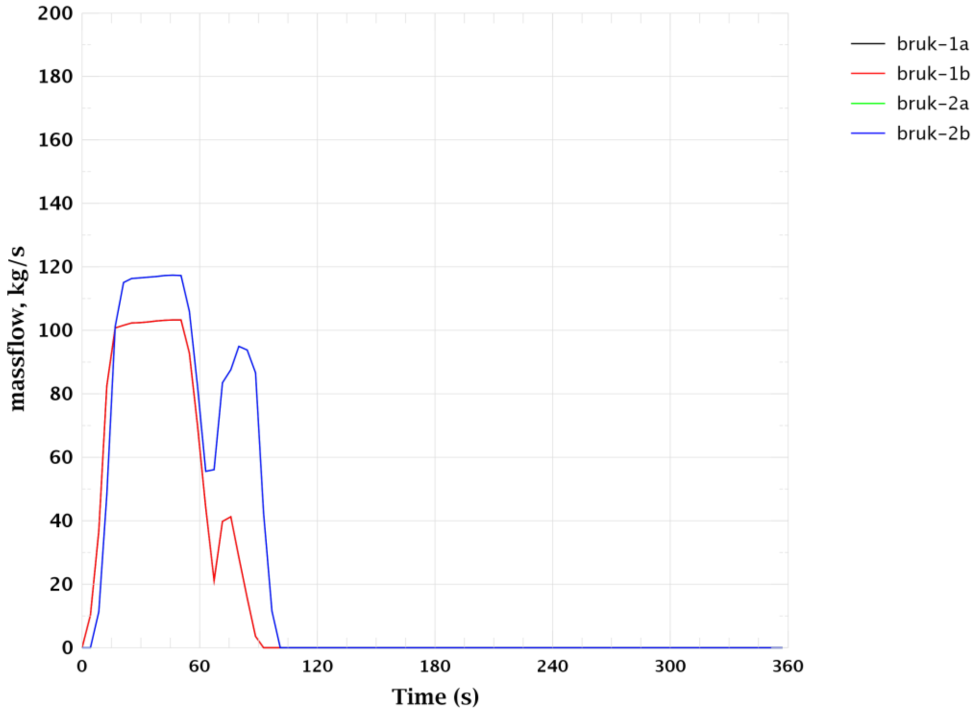



**Figure 118 BRU-K Mass Flow Rate**

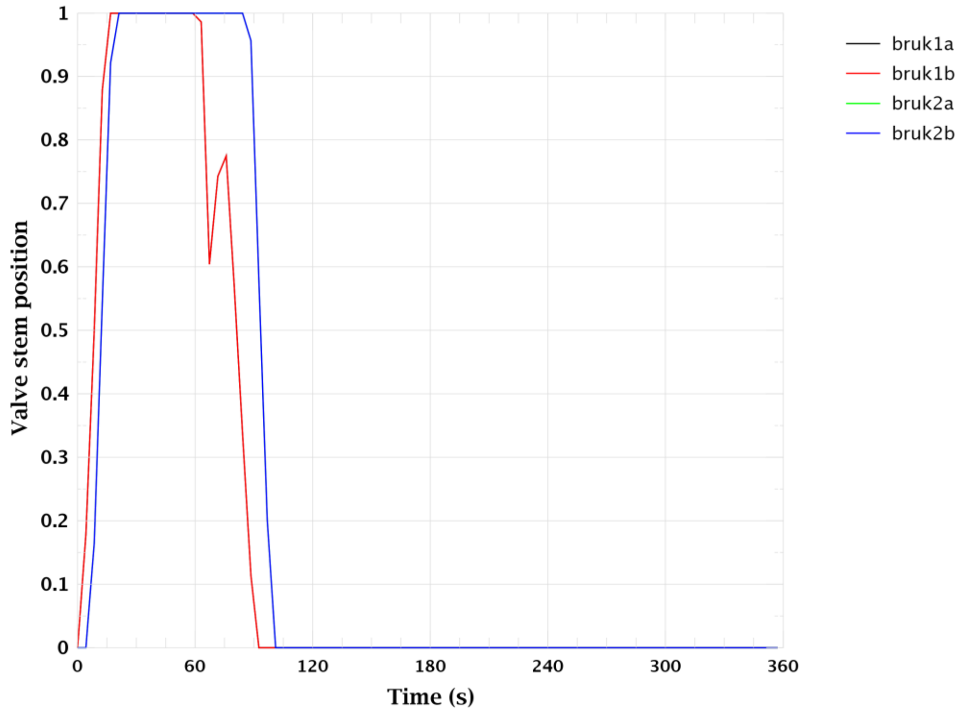



**Figure 119 BRU-K Valve Stem Position**

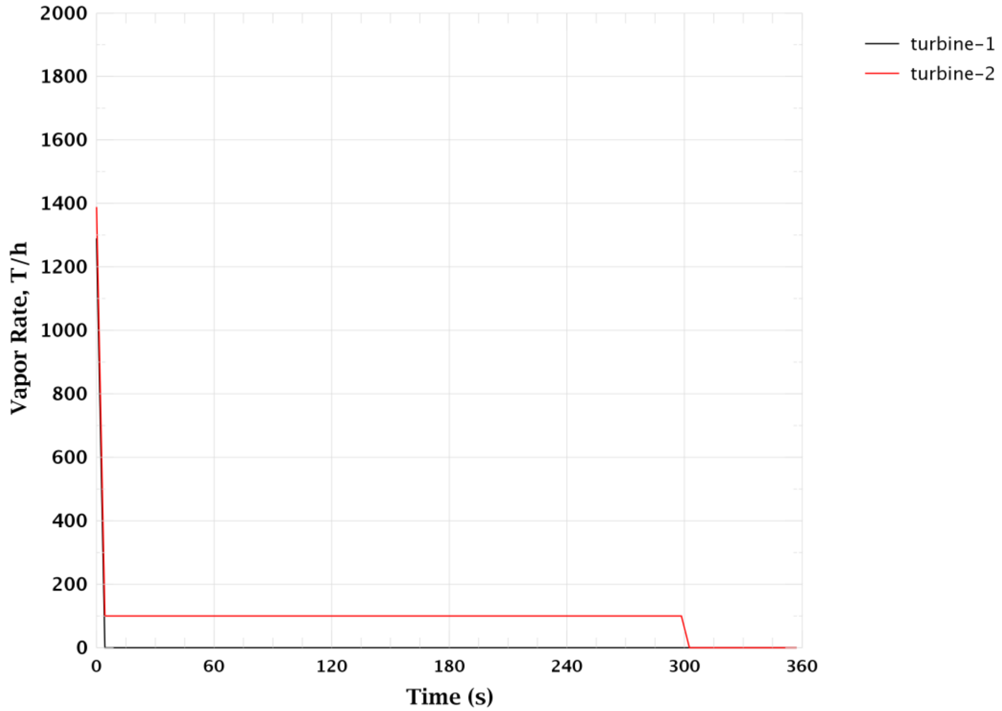



**Figure 120 TG Mass Flow Rate**

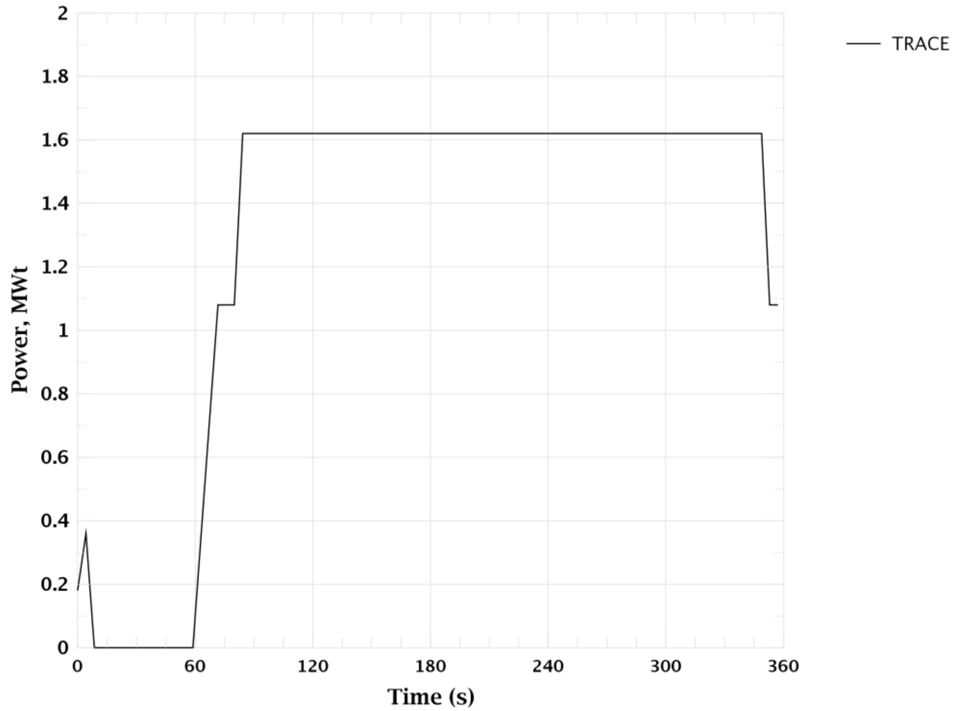



**Figure 121 PRZ Heaters Power**

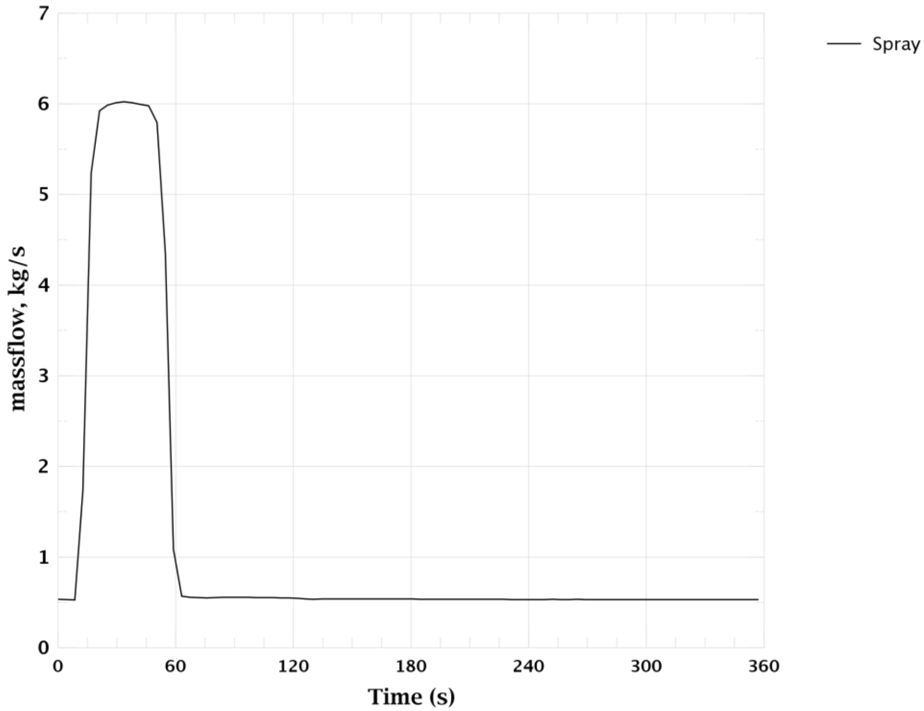



**Figure 122 Spray Valves Massflow**

### 4.2.4 Conclusion

The comparative computer simulation of "Reactor scram transient initiated by 6 kV switch short circuit" incident with RNPP-1 model for TRACE code demonstrates that calculated parameters behavior is in general in good compliance with measured data.

As it can be seen from Fig. 90 at the initial phase of the incident the calculated secondary pressure increase rate is almost identical to the measured one. However the maximal calculated pressure is ~53 kgf/cm$^2$ while correspondent measured value is ~50 kgf/cm$^2$, that can be explained by simplified modeling of BRU-D. More precise modelling of additional steam releases from MSH is precluded by absence of measured data on BRU-D operation during the incident.

Deviation in secondary circuit pressure affects operation of ARM (see description of boundary conditions in section 4.2.2 of this report) as well as RCS pressure behavior (Fig. 88). At the initial phase of the incident the pressure increase rate is in good agreement with measured data while the calculated pressure increase time is ~10 sec longer that results in higher maximal pressure value.

With few exceptions the calculated primary circuit temperatures (Fig. 92 – Fig. 103) lie within the measurement error band comparing to measured values. Qualitative behavior of SG levels (Fig. 107 – Fig. 112) is also in good agreement with measured data. At the same time the initial SG level values and level decrease rate in the calculation were lower than in the incident.

Similar to the analysis of incident described in Section 4.1 this transient simulation allowed to check correctness of the main primary and secondary circuit equipment and controllers modelling.

## 4.3 Inadvertent Reactor Scram

According to [7], the incident was caused by malfunction in scram actuation logic that resulted in spurious actuation of reactor scram signal “Pressure difference across the reactor core > 2.8 $kgf/cm^2$ at 5 reactor coolant pumps in operation”. This signal leads to a trip of MCP-1, 3, 5 with subsequent decrease of coolant flow.

Before the incident, the unit operated at nominal thermal power and electric power was 400 MWe. Normal operation systems and equipment, main electric circuit and house loads power supply as well as all protections and interlocks were in design configuration for nominal power operation mode. Safety system trains No.1, 2, 3 were in standby mode.

Spurious actuation of scram resulted in simultaneous drop of all control assemblies groups with a rate of 20 cm/sec and decrease of reactor power down to decay heat.

Trip of MCP-1,3,5 trips caused decrease of coolant flow through the reactor. Further flow reversal in the loops with tripped MCPs caused asymmetric behavior of parameters in RCS loops and SG secondary side.

From the validation standpoint this incident allows to evaluate correctness of reactor pressure losses simulation and modelling of:

- MCP coastdown;
- turbine controllers;
- primary pressure control system (PRZ heaters and spray valves);
- PRZ level controller (RCS makeup and letdown).

### 4.3.1 Initial Conditions

The initial conditions selected for transient calculation correspond to those specified in Table 2.

Before the transient the unit was in nominal power operation for about two months since the beginning of the 21th fuel cylce, which started on June 3, 2001. In validation calculation the reactor core characteristics which correspond to beginning of the 22th fuel cycle [4] were applied.

### 4.3.2 Boundary Conditions

The boundary conditions for transient simulation include automatic and manual actions. Taking into account that the state of certain systems and equipment which should be in operation according to the design (e.g., secondary circuit drains, AFW pumps, SG blow-down) is not clearly described in the incident report [7], an influence of these systems operation on the main documented parameters was evaluated by performing preliminary calculations and incorporating correspondent assumptions into the final transient scenario.

As an initiating event, spurious reactor scram actuation from “Pressure difference across the reactor core > 2.8 kgf/cm$^2$ at 5 reactor coolant pumps in operation” signal was modeled.

According to incident description in [7] the trip of both turbines occurred 10 seconds after the scram, however this statement is not confirmed by actual measurement data. Thus closure of turbines stop valves should cause early increase of secondary circuit pressure with further steam dump through BRU-K. Contrary to that the measurement data indicate MSH pressure decrease for the first 40 seconds of incident. This suggests that closure of TG-1,2 stop valves occurred due to decrease of MSH pressure to 38/40 kgf/cm$^2$. Therefore in calculation the signal for turbines trip after the scram (with 10 sec delay) was blocked.

To take into account operation of steam dump valve to deaerator (BRU-D) an additional steam release from MSH was modeled in calculation. Incompliance of thermal (440 MW) and electrical ($N_e$ = 400 MW) reactor power suggests that such additional steam release actually existed during the incident.

Other assumptions on operation of safety systems and main equipment applied in final validation calculation are described below:

- operation of reactor control and protection system is started by the first initiating signal in compliance with design set points;
- design operation of the primary makeup/letdown and primary pressure control system is modeled;
- trip of MCP-1,3,5 trips was modeled after spurious actuation of scram signal;
- design operation of MFW pumps is assumed. Operation of AFW pumps is not considered;
- SG feedwater flow is controlled by operation of MFW controllers according to their design characteristics;
- BRU-D operation is simulated as additional steam release from MSH.

### 4.3.3 Calculation Results

Transient simulation of selected scenario was performed for 160 s from the initiating event occurrence.

Table 5 provides comparison of calculated and actual timing of events occurred in the course of the incident.

**Table 5 Sequence of Events**

| Time, sec | Event in the incident | Event in calculation |
|---|---|---|
| 0.0 | Reactor scram by “Pressure difference across the reactor core > 2.8 kgf/cm$^2$ at 5 MCPs in operation” signal.<br>MCP-1, 3, 5 trip | Start of all control assemblies drop to the reactor core with a rate of 20 cm/sec.<br>MCP-1, 3, 5 trip. |
| 0.1 | | Switching of TG-2 turbine controller to automatic mode |

| Time, sec | Event in the incident | Event in calculation |
|---|---|---|
| 9.2 | | Let-down line valve closure |
| 6.75 | | PRZ heaters group 3 is ON due to RCS pressure decrease to 122 kgf/cm$^2$ |
| 9.2 | | PRZ heaters group 4 is ON due to RCS pressure decrease to 121 kgf/cm$^2$ |
| 10.0 | Closure of turbines stop valves | Not modeled (boundary condition, see Section 4.3.2) |
| 15.7 | | PRZ heaters group 5 is ON due to RCS pressure decrease to 118 kgf/cm$^2$ |
| 16.1 | | Primary makeup control valves open due to PRZ level decrease for 300 mm from nominal |
| 27.0 | | TG-2 stop valves closure due to decrease of turbine's upstream pressure below 40 kgf/cm$^2$ |
| 38.2 | | TG-1 stop valves closure due to decrease of turbine's upstream pressure below 38 kgf/cm$^2$ |
| 160.0 | | End of calculation |

Beginning of transient is characterized by a decrease of reactor power (Fig. 151) and decrease of coolant temperature in hot and cold legs (Fig. 132 – Fig. 143) that causes decrease of steam production in SGs and decrease of secondary circuit pressure (Fig. 124 – Fig. 125).  Due to decrease of average coolant temperature PRZ level (Fig. 144) and RCS pressure (Fig. 123) start to decrease.  This causes closure of letdown valves, opening of make-up valves (Fig. 154) and sequential actuation of PRZ heaters groups.

Turbine controllers switch from standby to automatic control mode attempting to maintain prescribed secondary circuit pressure by decreasing steam flow to the turbines.  As MSH pressure continues to decrease the setpoints for turbines stop valves closure (40 kgf/cm$^2$ and 38 kgf/cm$^2$ for TG-2 and TG-1, respectively) are reached.  After closure of TG stop valves, secondary circuit pressure starts to increase (Fig. 124 – Fig. 125).

Decrease of RCS coolant flow due to a trip of MCP-1,3,5 decreases pressure drop across the reactor (Fig. 157) and affects behavior of correspondent loops temperatures.  As it can be seen from Fig. 132 – Fig. 143 the initial temperature decrease rate in loops #1,3,5 slows down while flow through these loops decreases during MCP coastdown.  The behavior of temperature in RCS loop #1 (Fig. 138) differs from other loops temperatures which is caused by entrance of hot PRZ coolant to this loop as coolant volume shrinks at the decrease of average temperature.

At SG secondary side the initial decrease of SG level (Fig. 145 – Fig. 150) is compensated by opening of MFW control valves, and after restoration of SG level MFW control valves start to close.

At the end of transient the calculated cold legs temperatures are stabilized at ~253 °C which is close to the measured values.

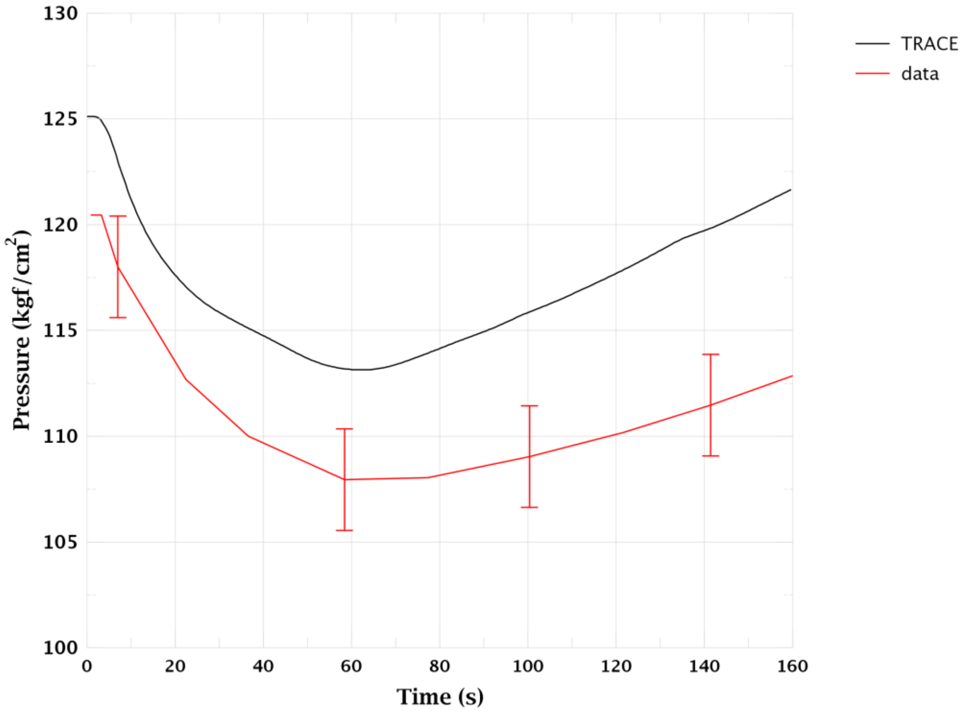



**Figure 123 Reactor Outlet Pressure**

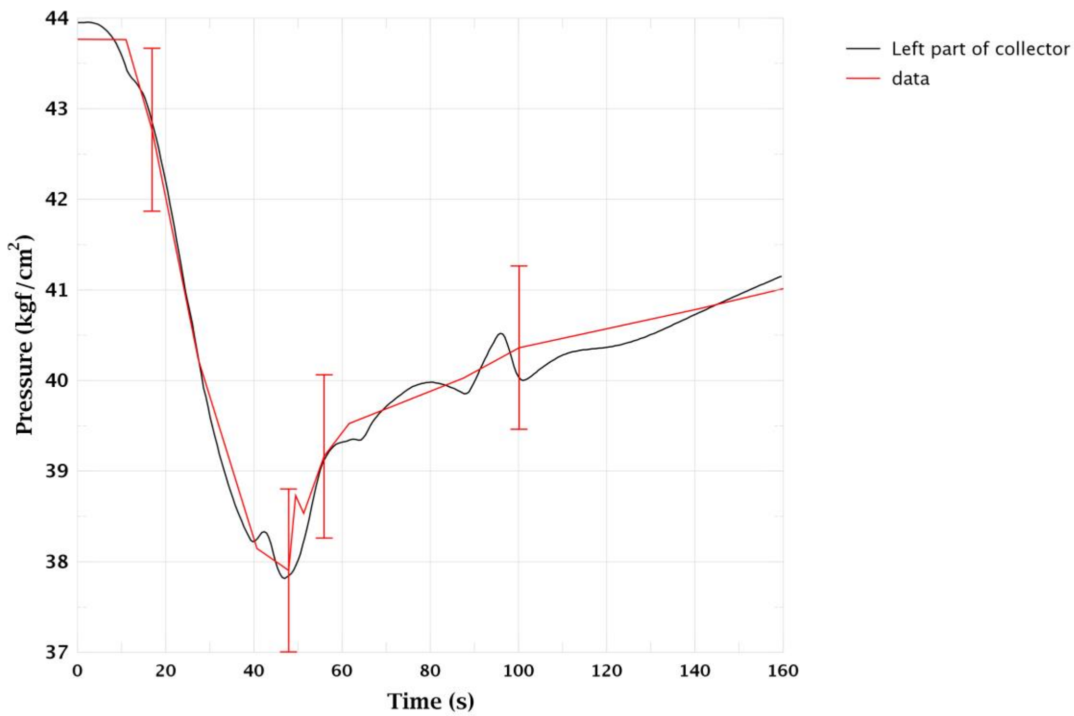



**Figure 124 Left MSH Semi-Header Pressure**

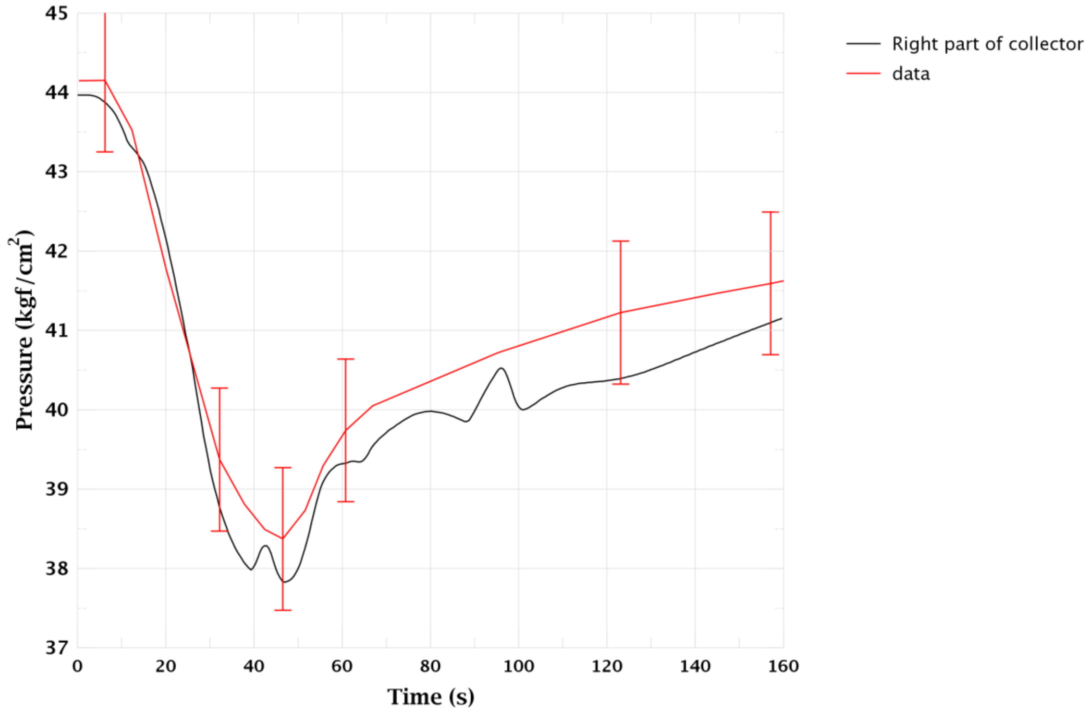



**Figure 125 Right MSH Semi-Header Pressure**

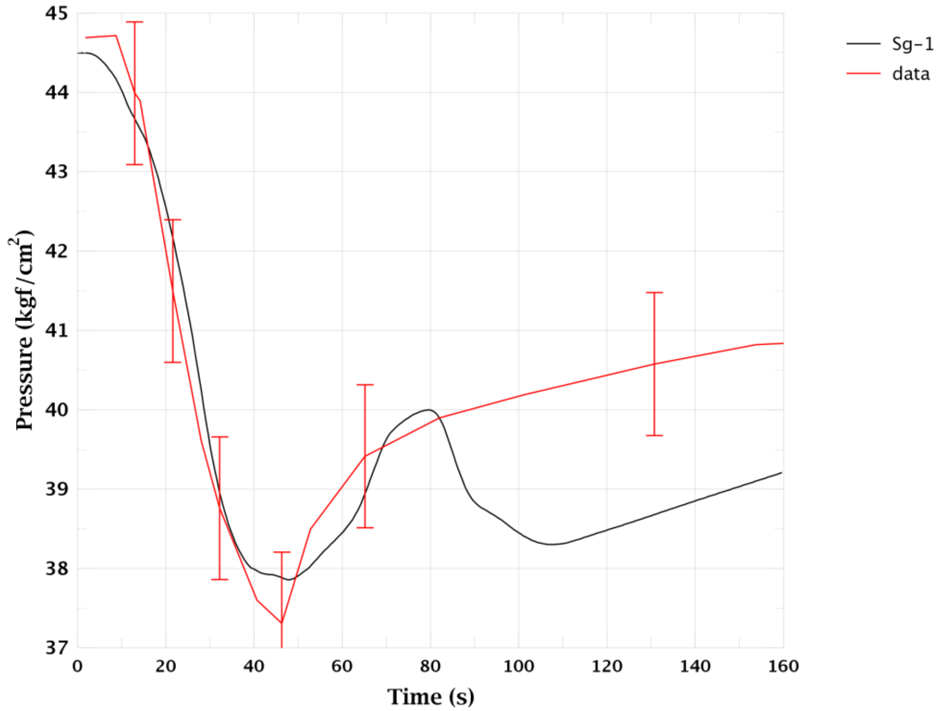



**Figure 126 SG-1 Pressure**

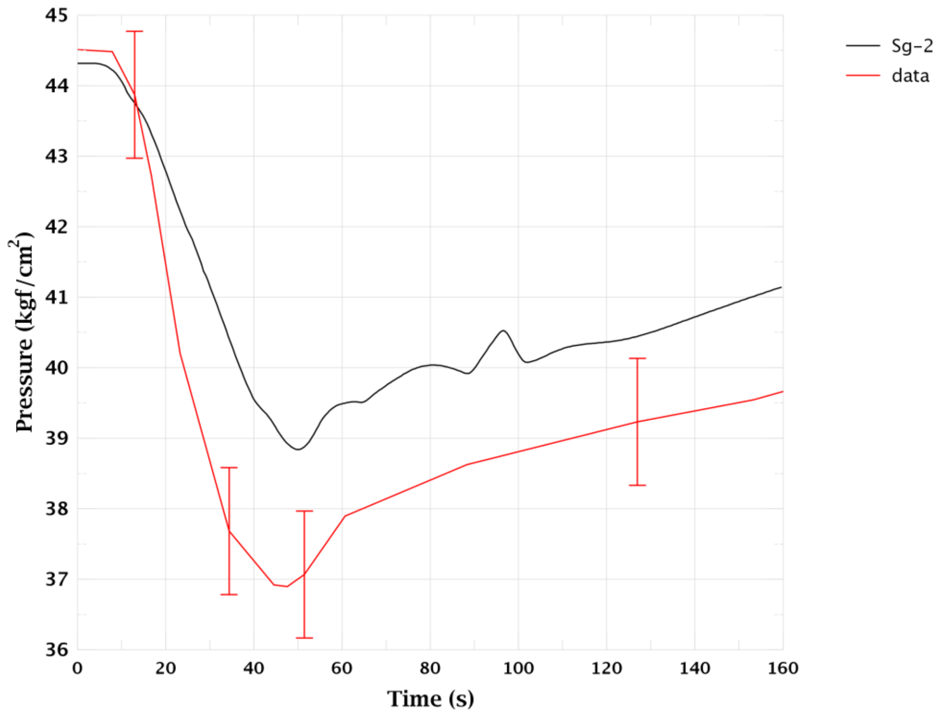



**Figure 127 SG-2 Pressure**

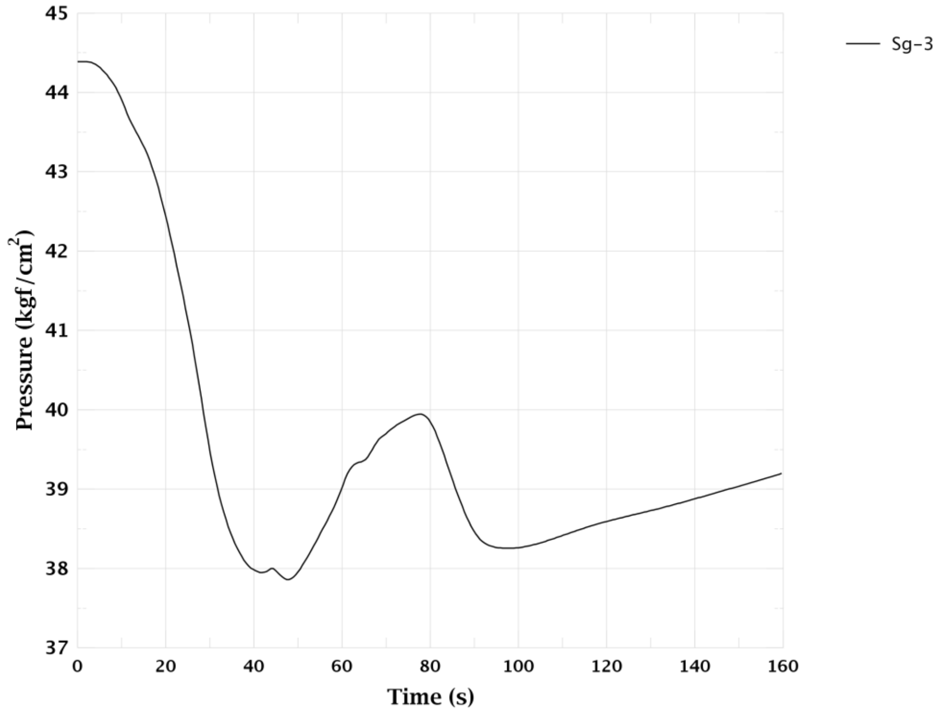



**Figure 128 SG-3 Pressure**

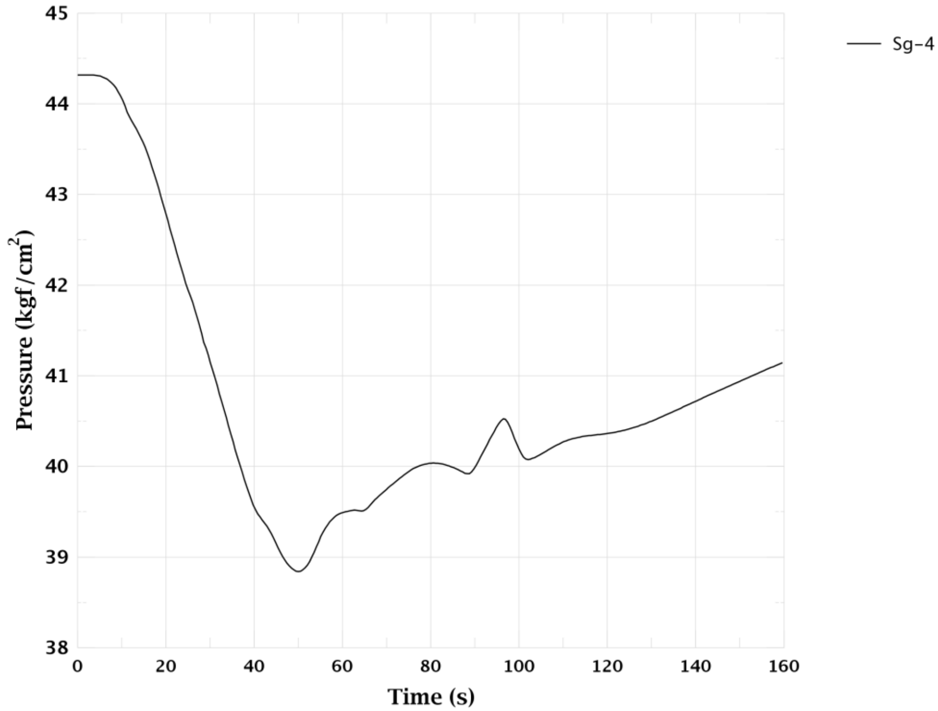



**Figure 129 SG-4 Pressure**

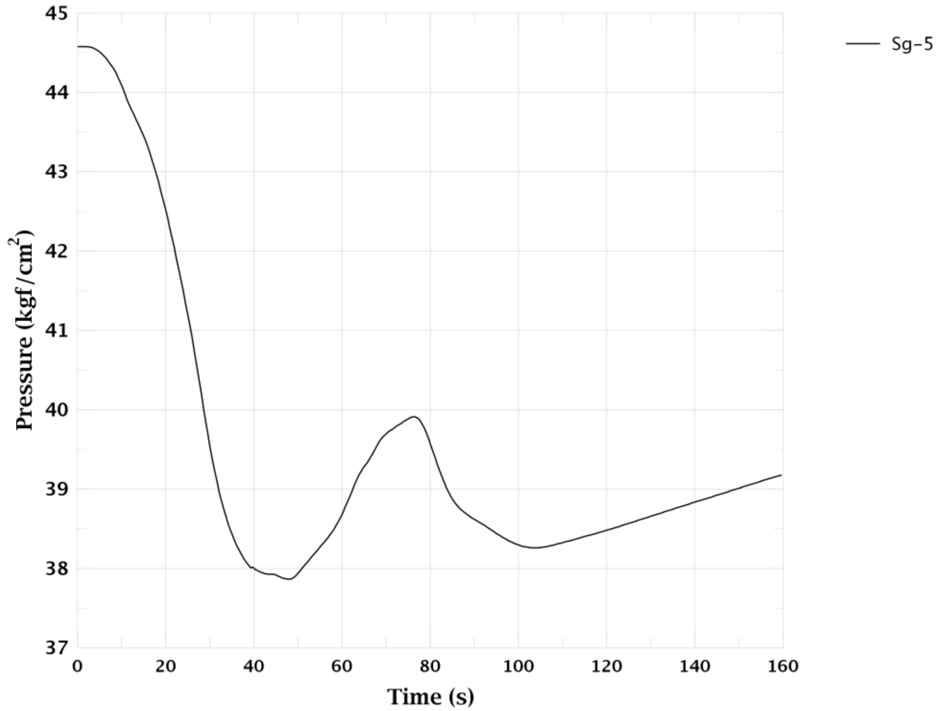



**Figure 130 SG-5 Pressure**

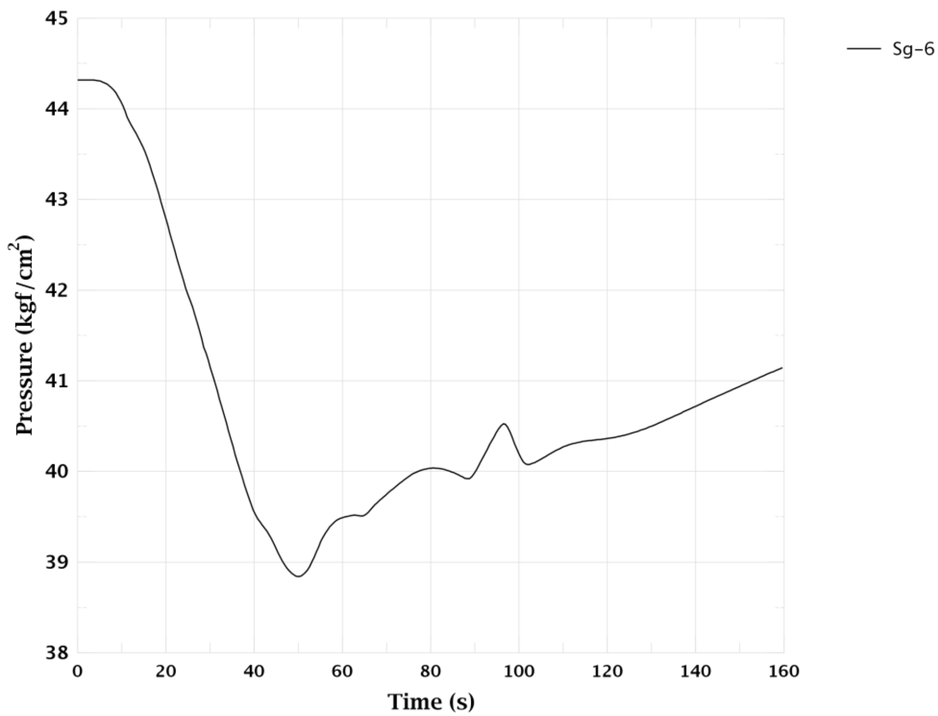



**Figure 131 SG-6 Pressure**

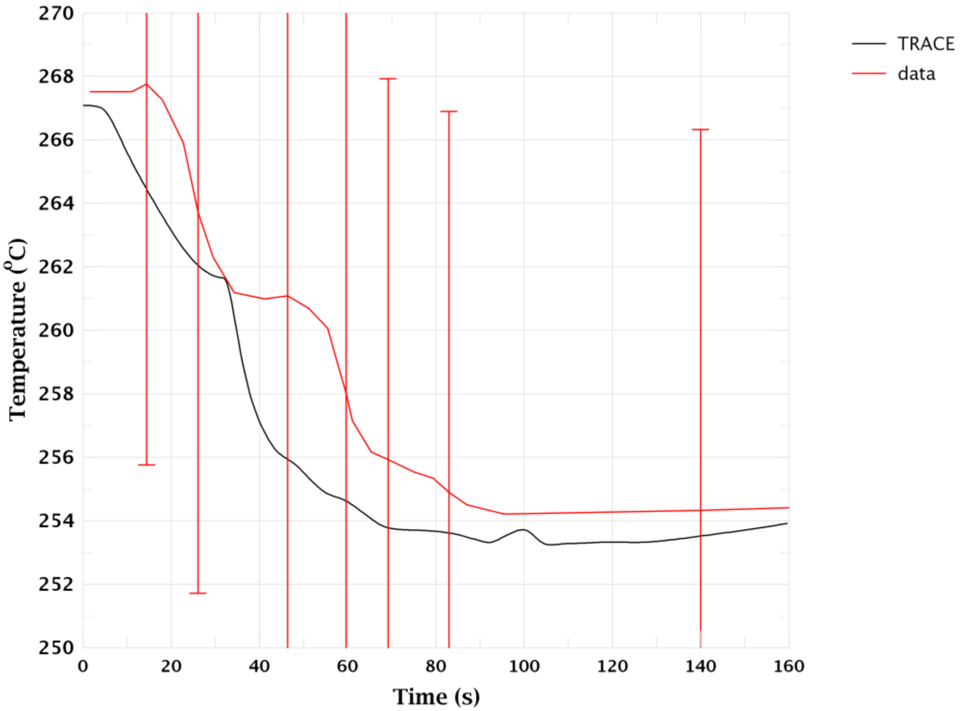



**Figure 132 Cold Leg No. 1 Temperature**

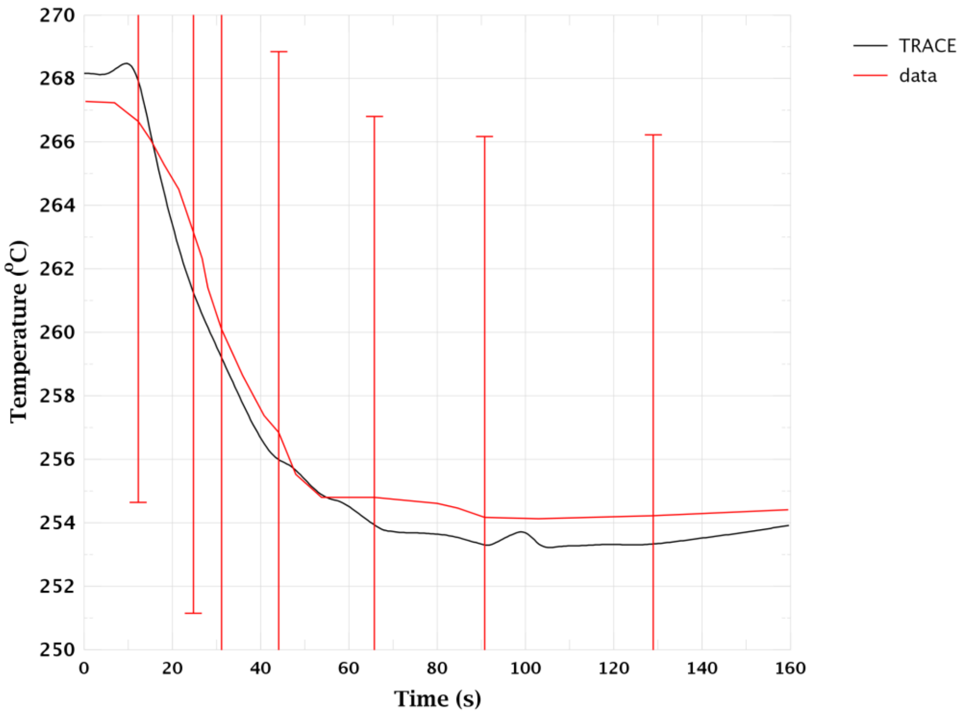



**Figure 133 Cold Leg No. 2 Temperature**

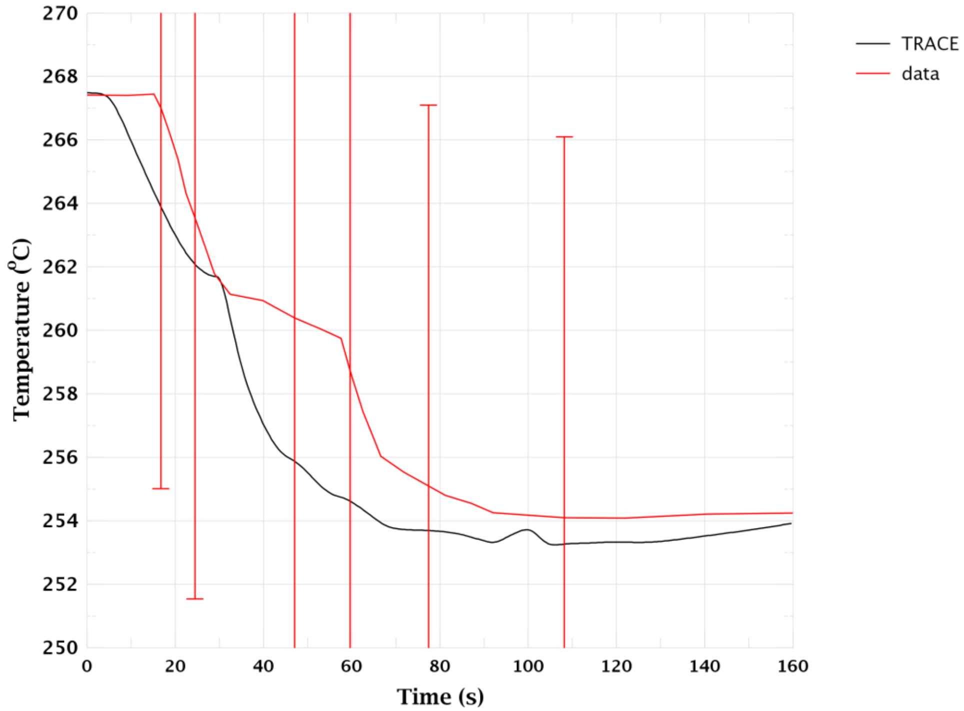



**Figure 134 Cold Leg No. 3 Temperature**

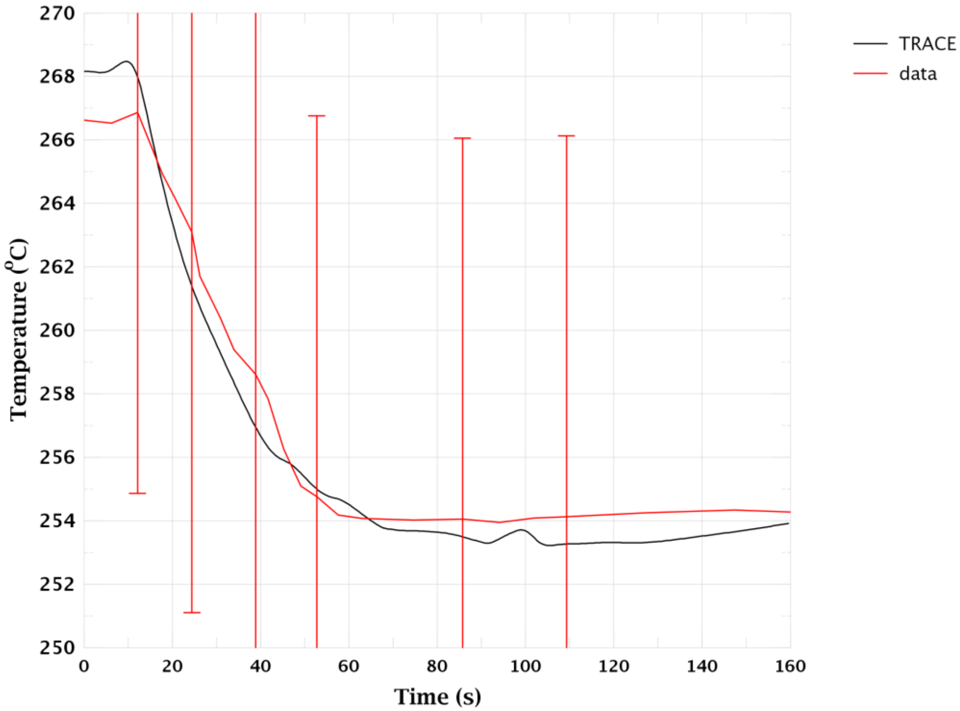



**Figure 135 Cold Leg No. 4 Temperature**

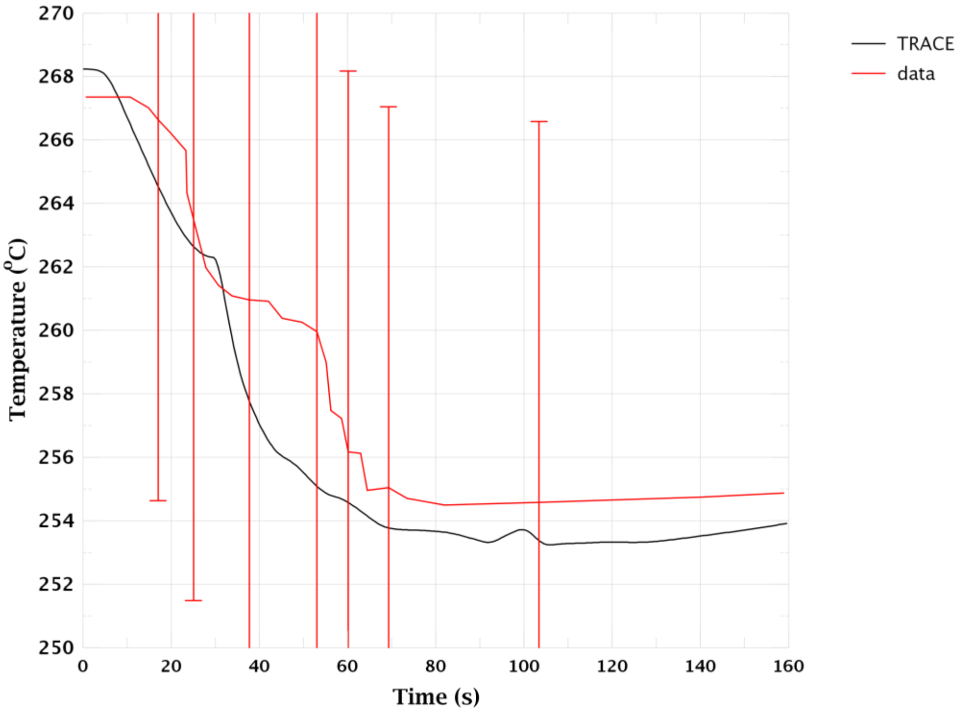



**Figure 136 Cold Leg No. 5 Temperature**

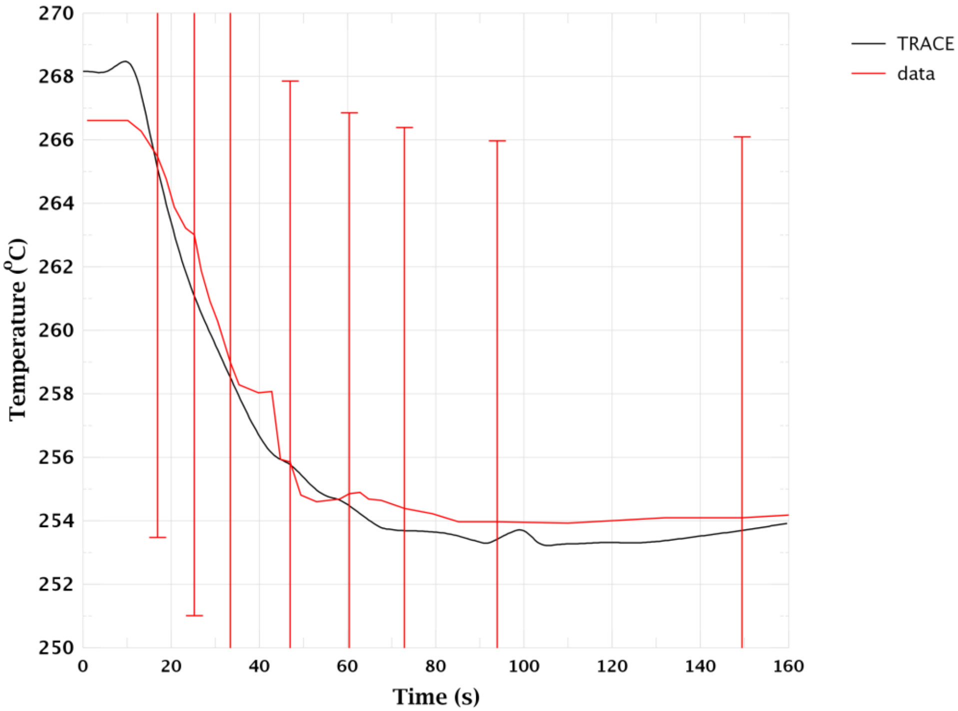



**Figure 137 Cold Leg No. 6 Temperature**

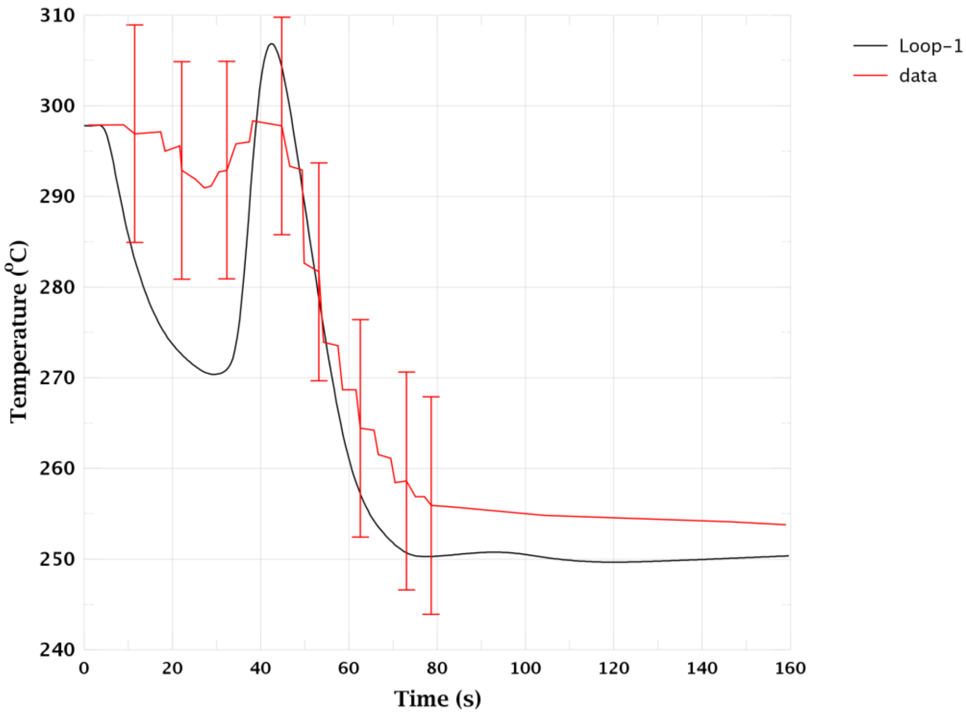



**Figure 138    Hot Leg No. 1 Temperature**

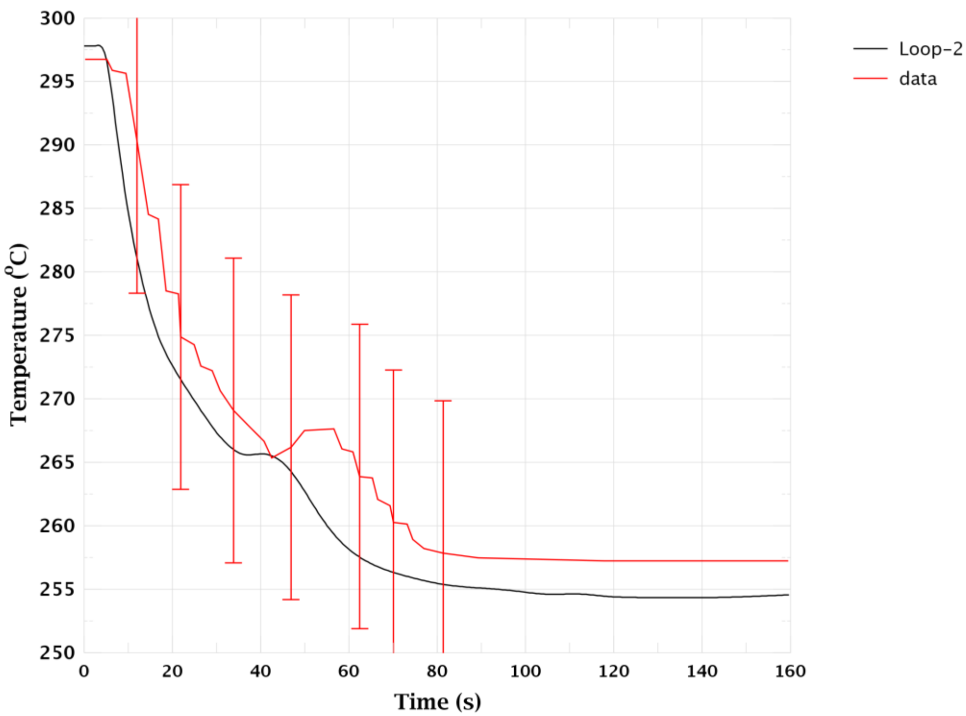



**Figure 139 Hot Leg No. 2 Temperature**

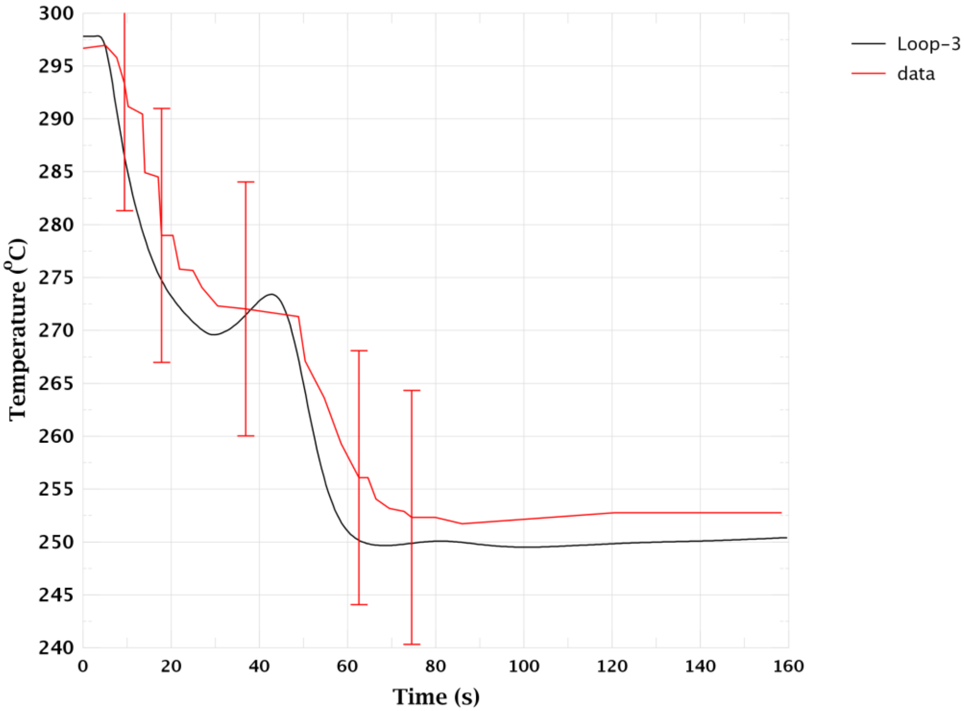



**Figure 140 Hot Leg No. 3 Temperature**

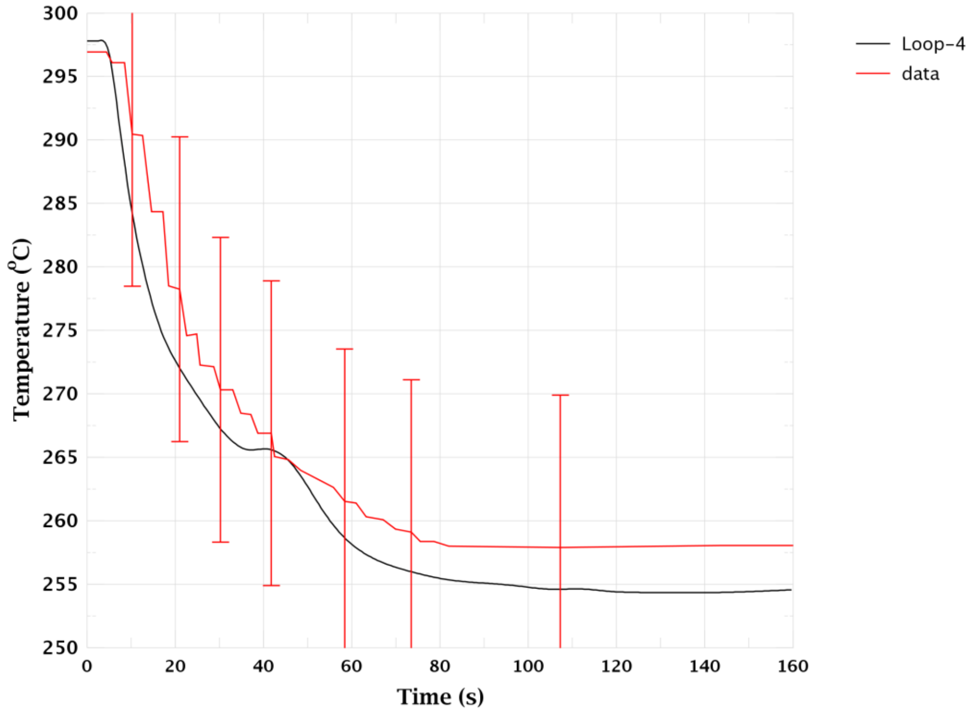



**Figure 141 Hot Leg No. 4 Temperature**

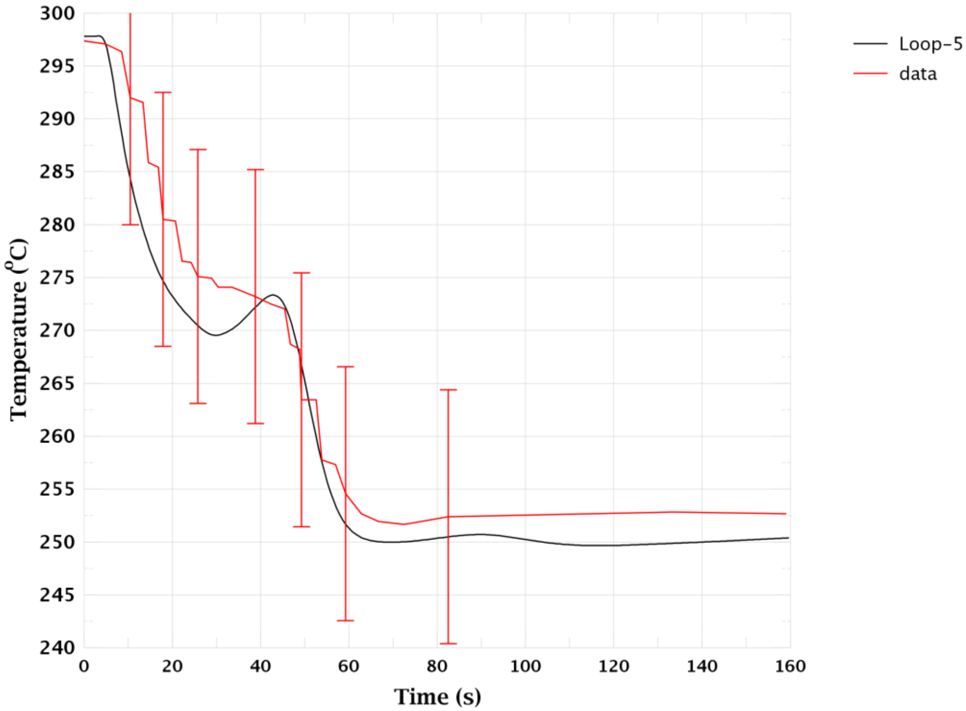



**Figure 142 Hot Leg No. 5 Temperature**

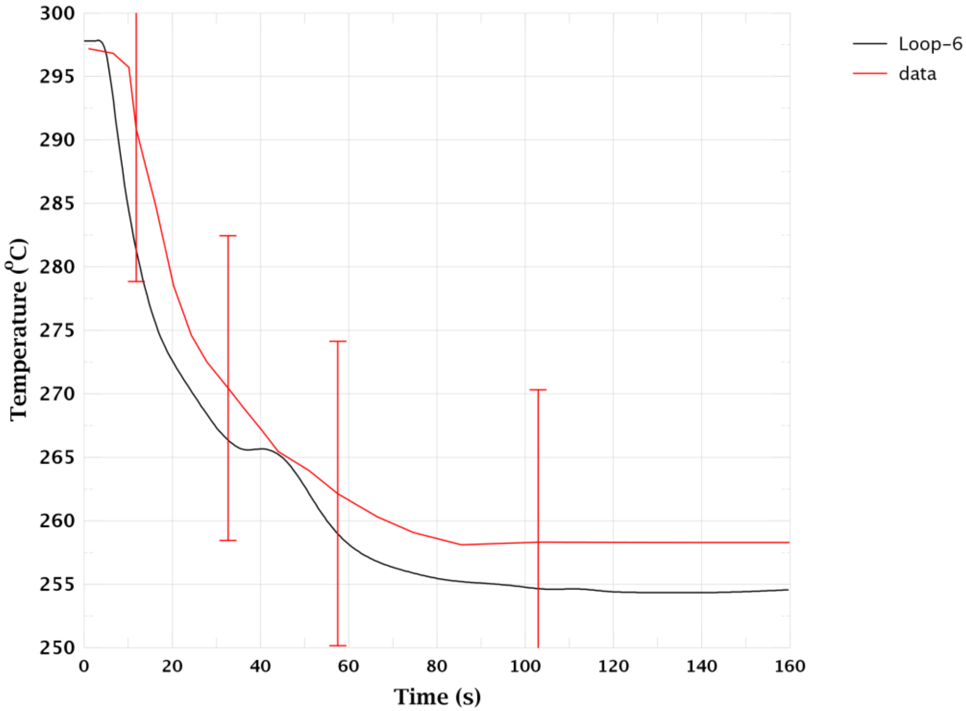



**Figure 143 Hot Leg No. 6 Temperature**

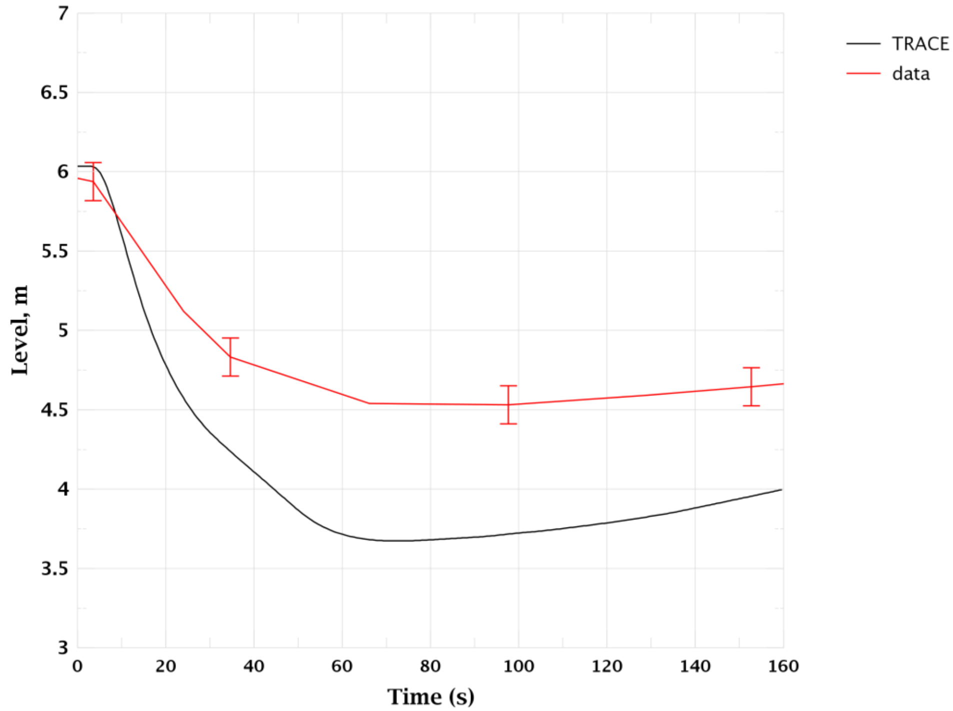



**Figure 144 PRZ Level**

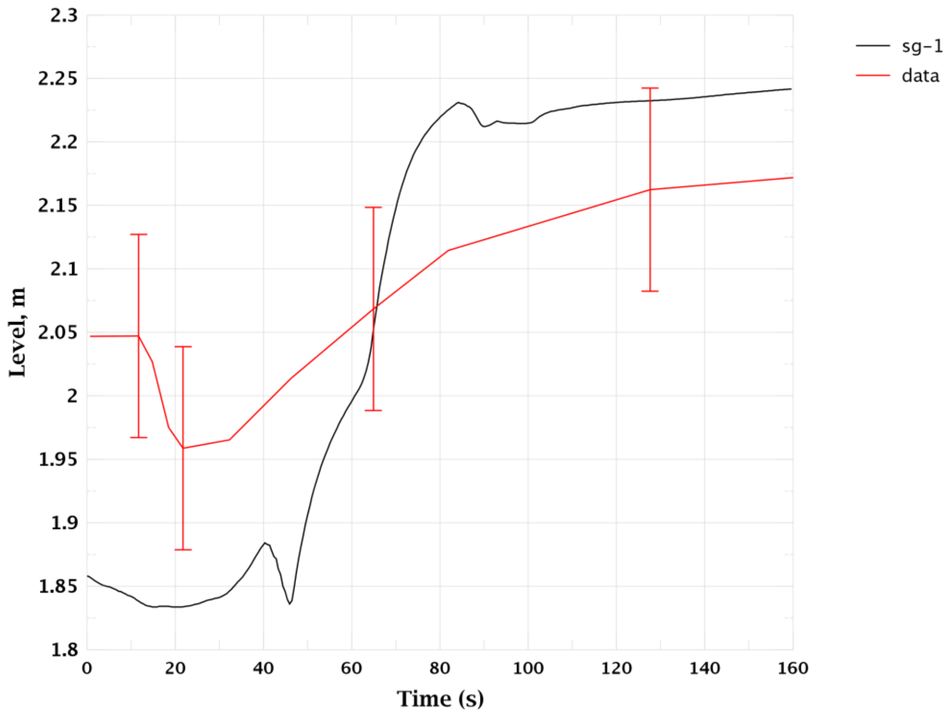



**Figure 145 SG-1 Level (wide range measurement)**

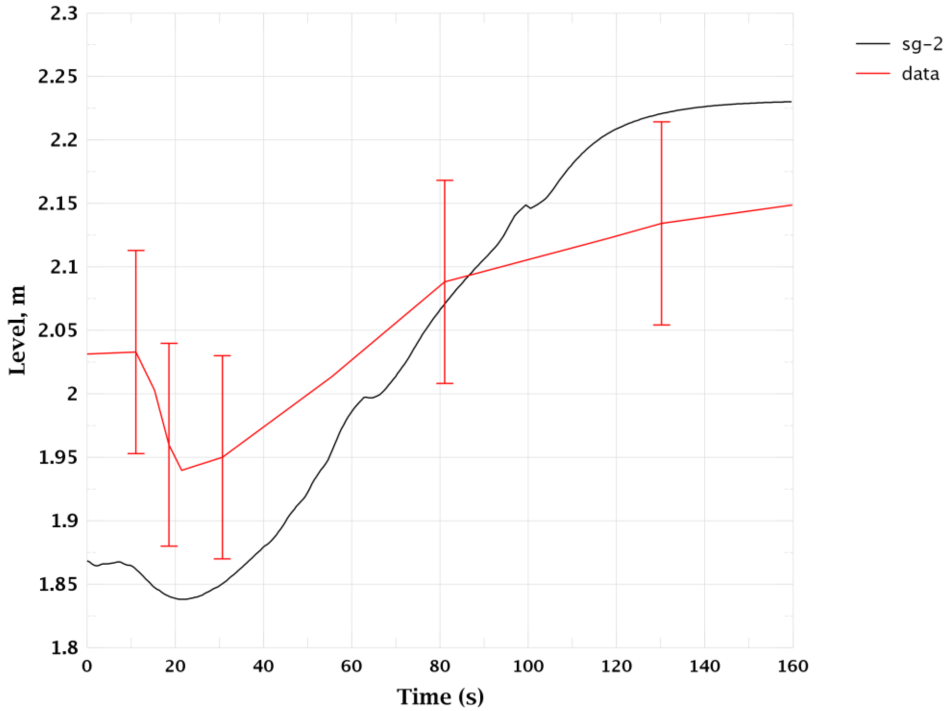



**Figure 146 SG-2 Level (wide range measurement)**

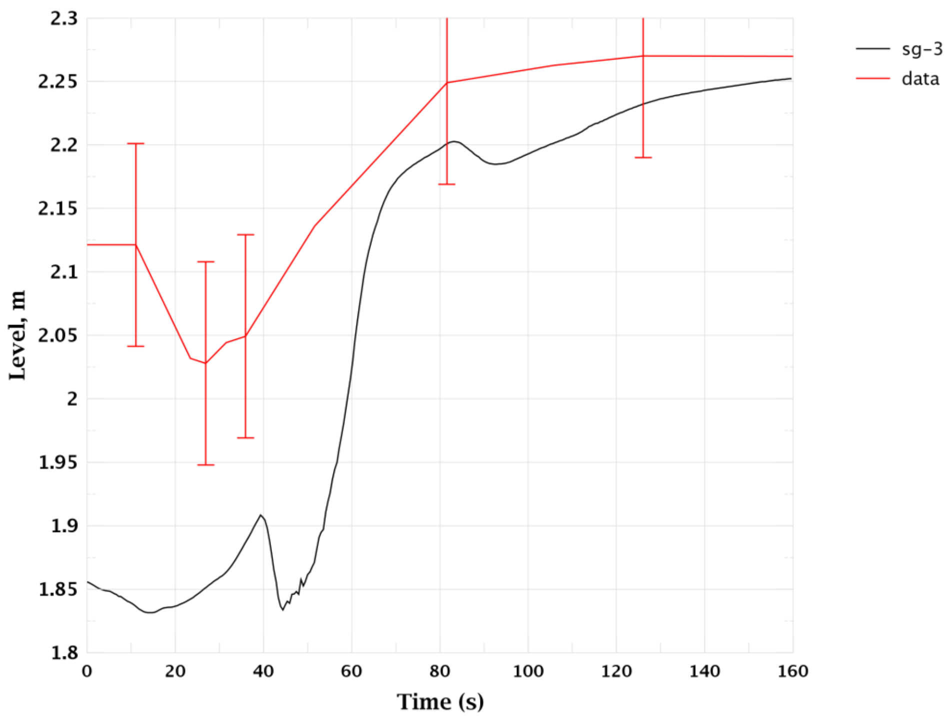



**Figure 147 SG-3 Level (wide range measurement)**

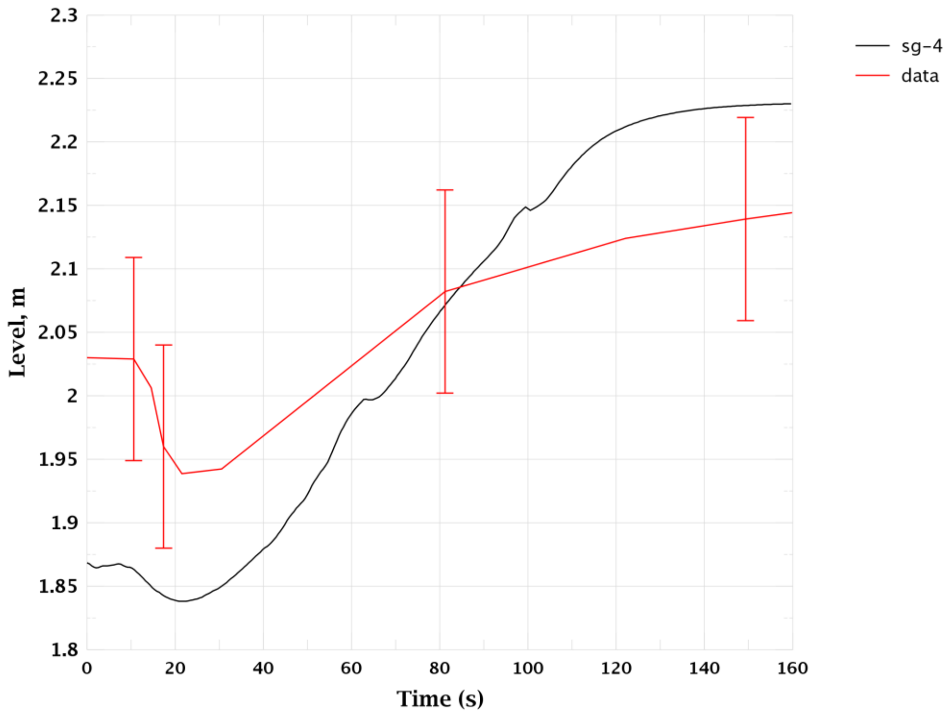



**Figure 148 SG-4 Level (wide range measurement)**

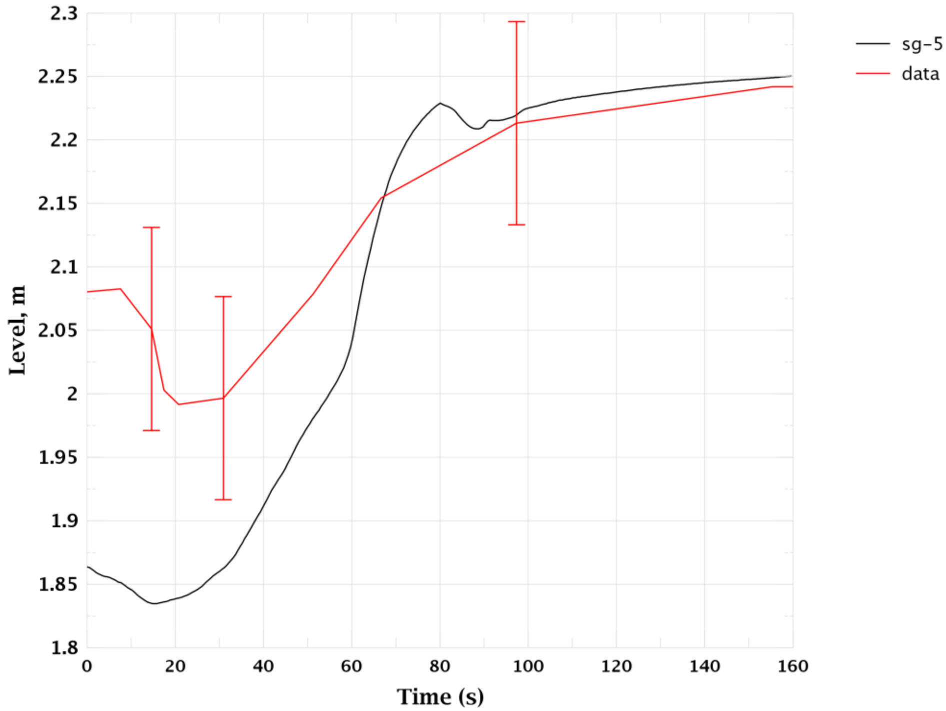



**Figure 149 SG-5 Level (wide range measurement)**

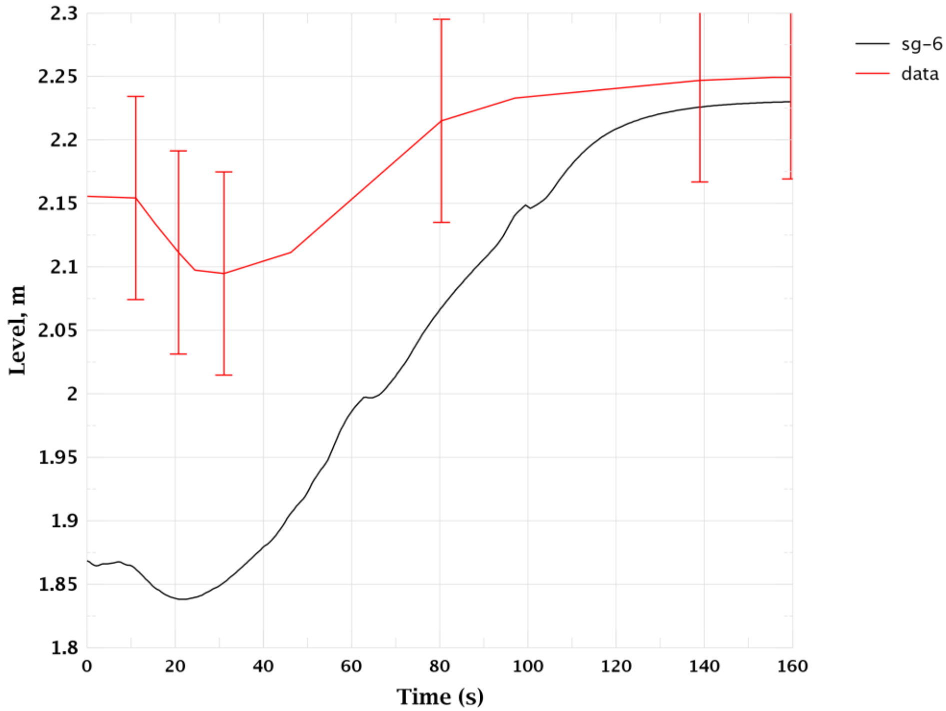



**Figure 150 SG-6 Level (wide range measurement)**

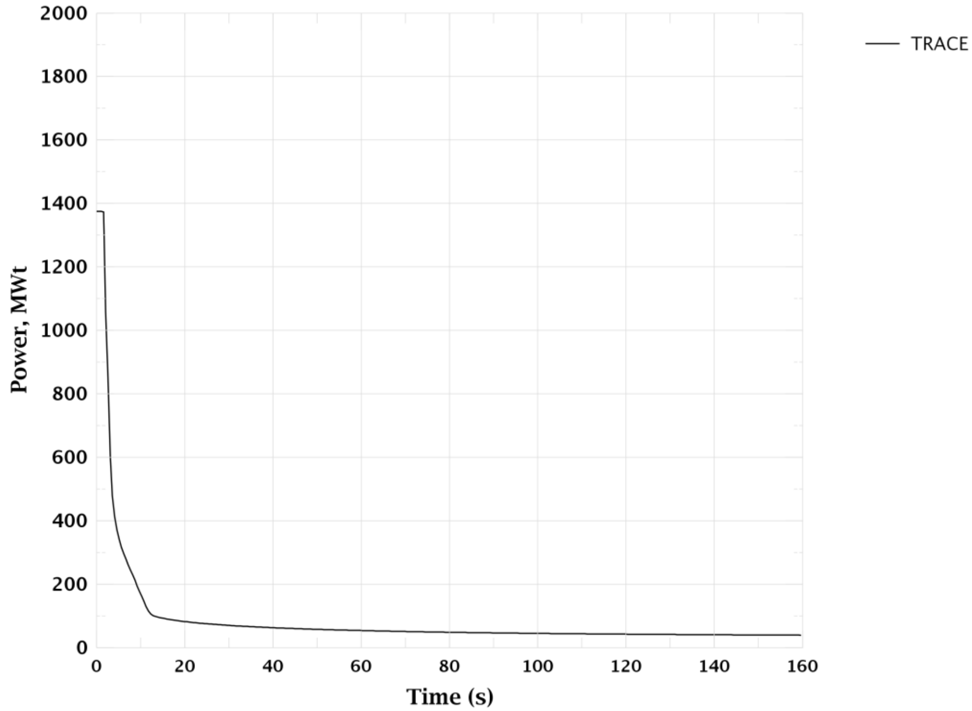



**Figure 151 Reactor Power Thermal Power**

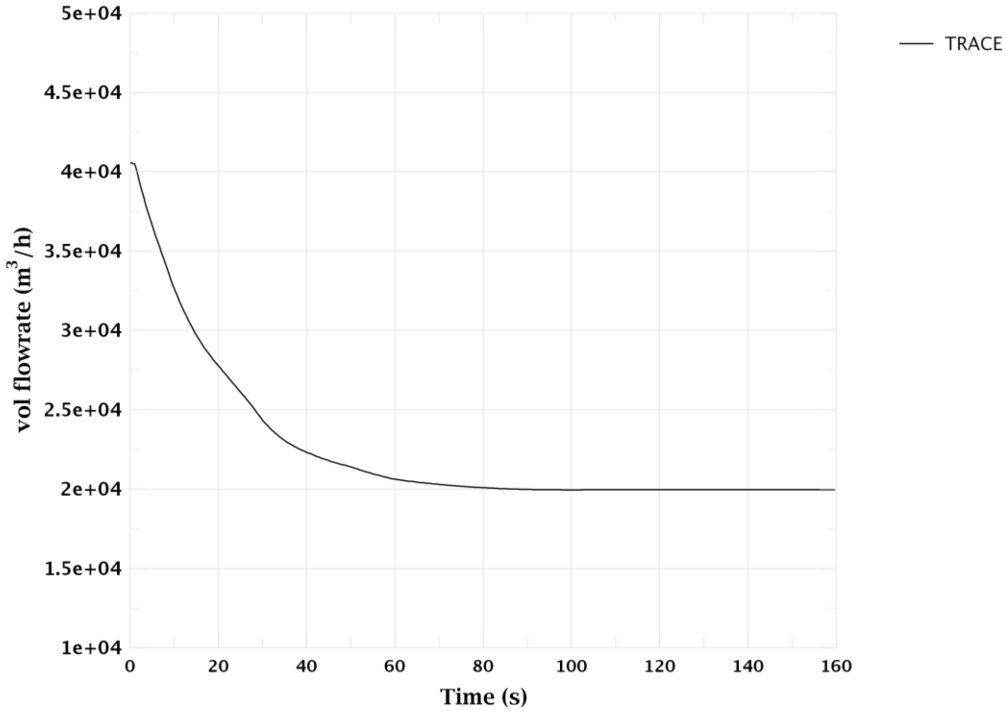



**Figure 152 Volumetric Reactor Coolant Flow Rate**

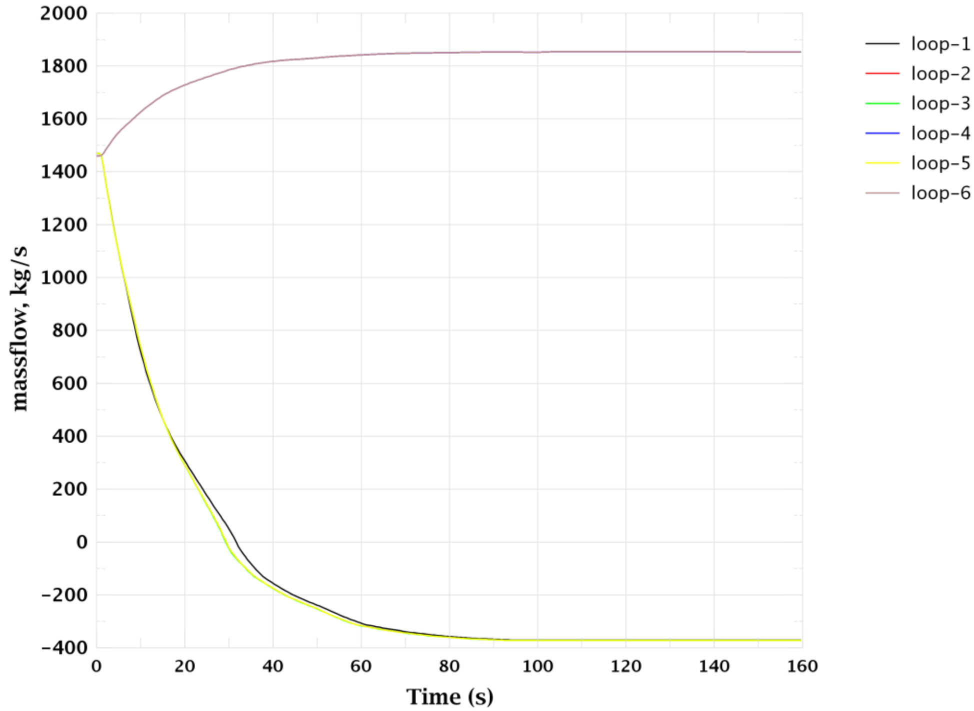



**Figure 153 RCS Loops Mass Flow Rate**

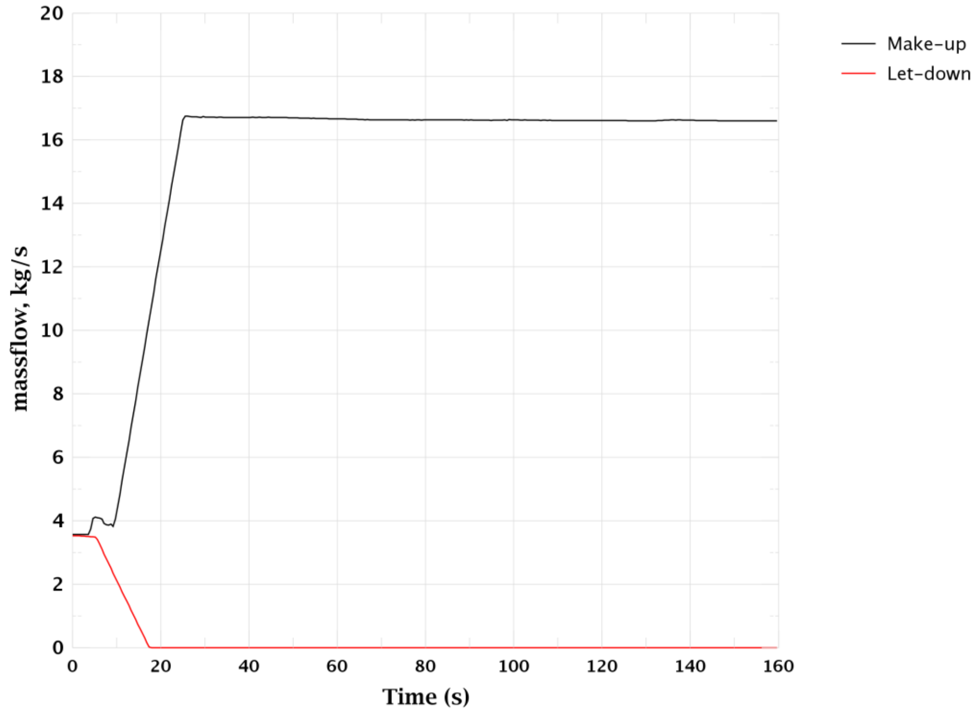



**Figure 154 Makeup-Letdown Mass Flow Rate**

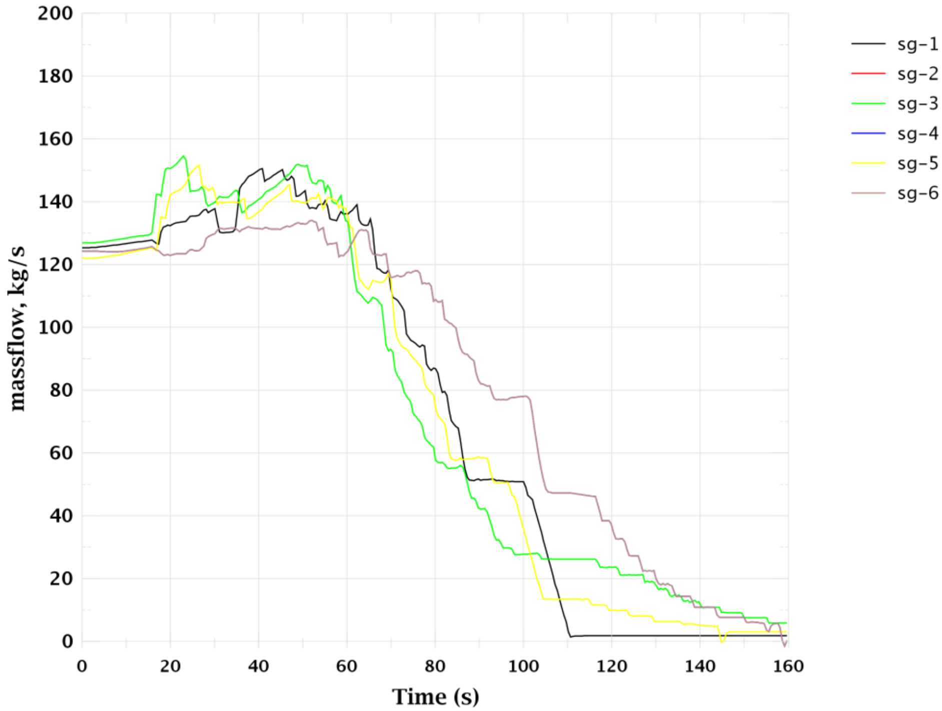



**Figure 155 Feedwater Mass Flow Rate to SG**

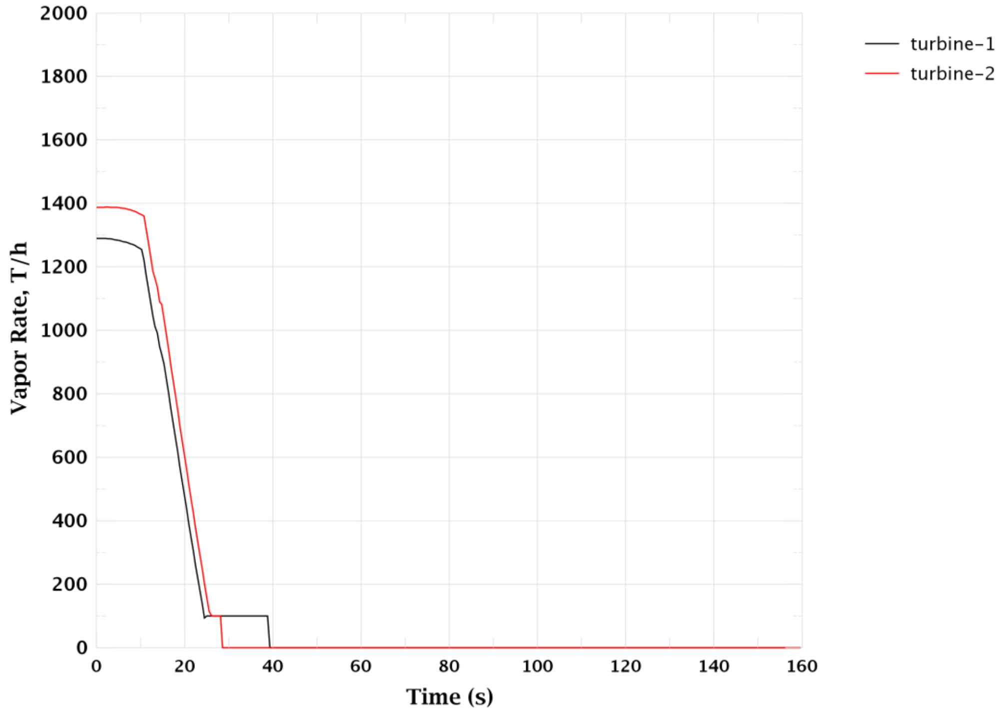



**Figure 156 TG Mass Flow Rate**

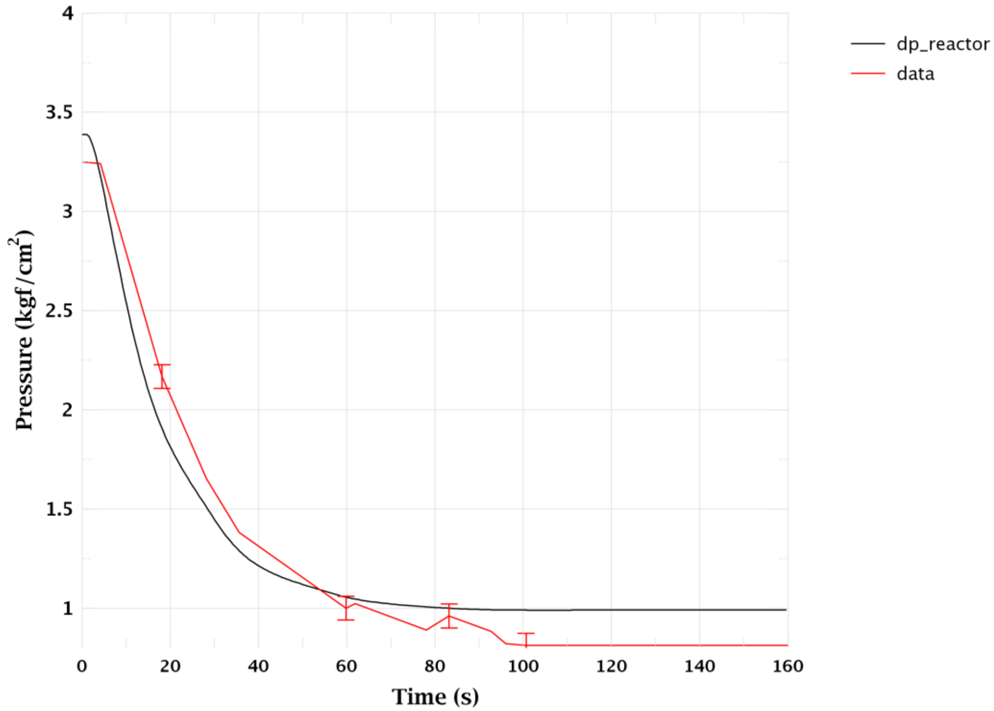



**Figure 157 Reactor Pressure Drop**

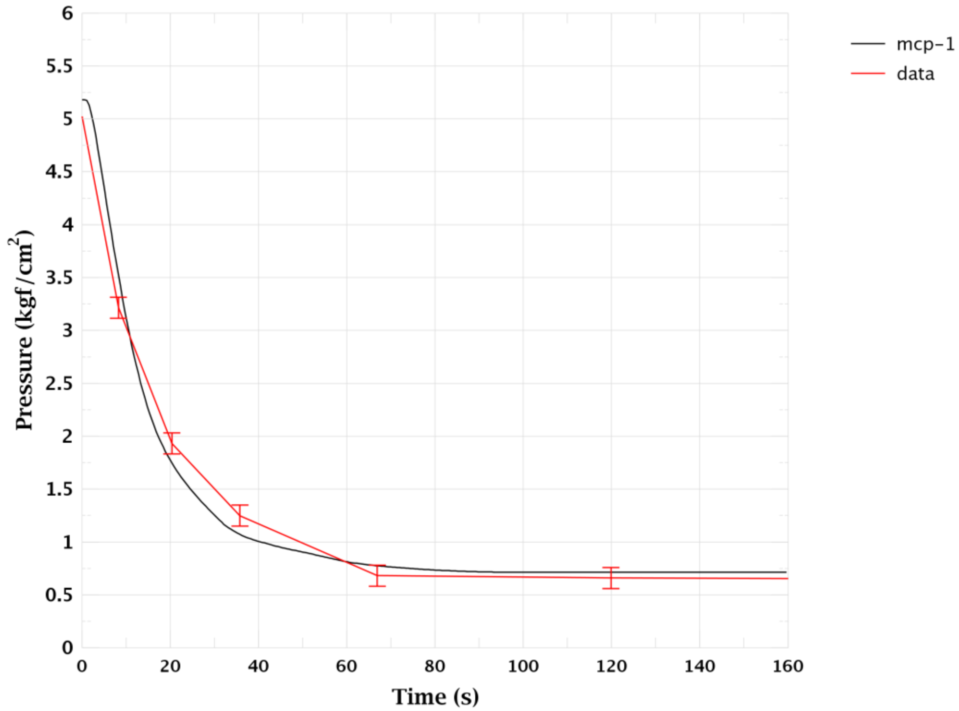



**Figure 158 MCP-1 Pressure Drop**

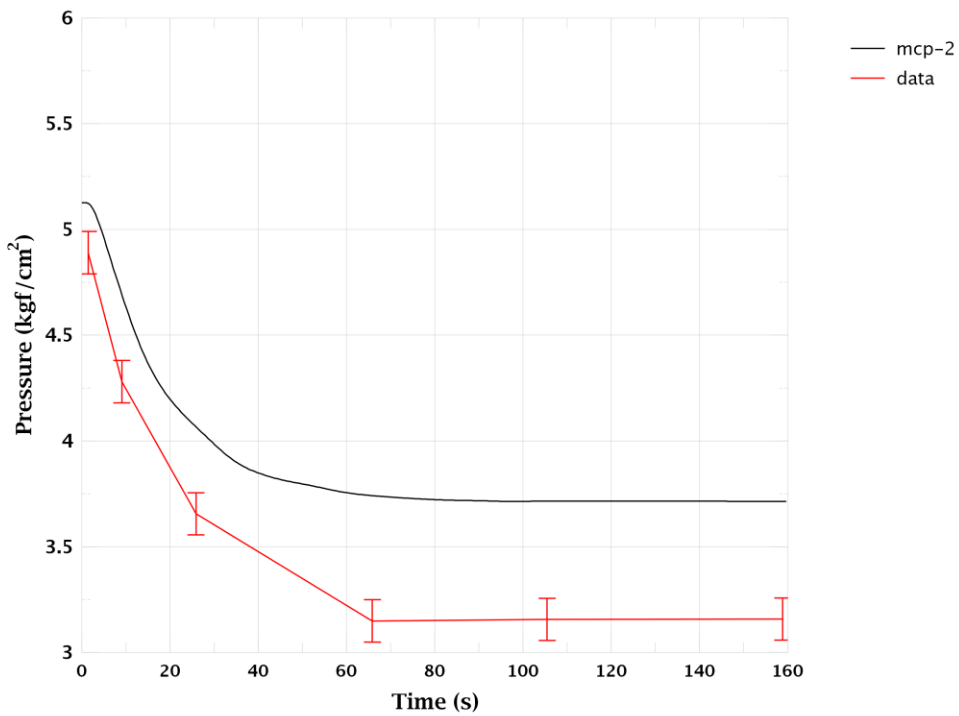



**Figure 159 MCP-2 Pressure Drop**

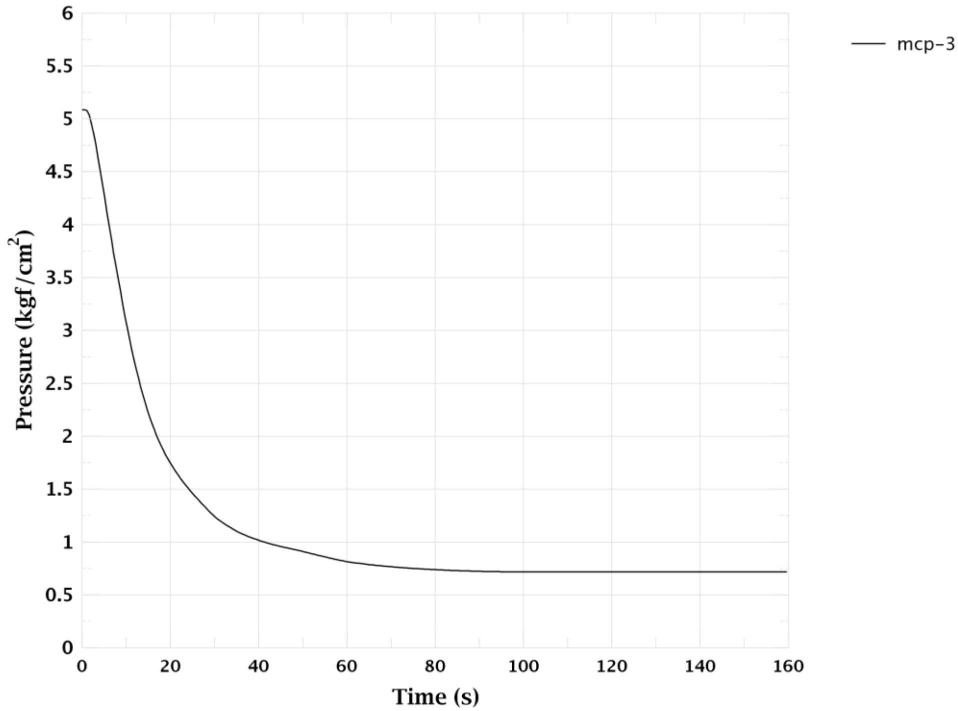



**Figure 160 MCP-3 Pressure Drop**

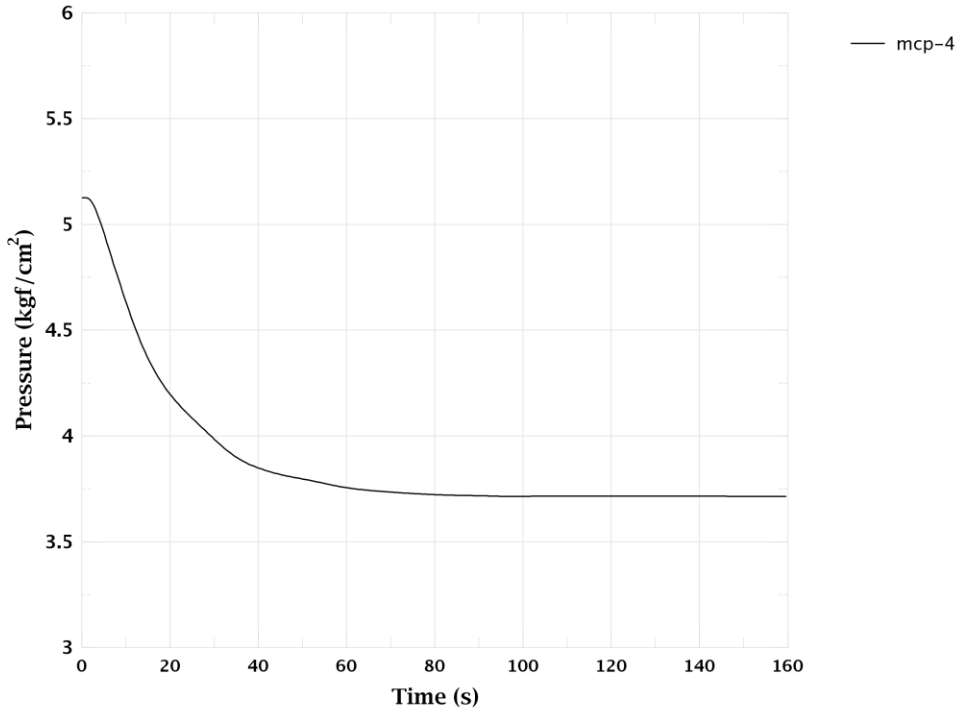



**Figure 161 MCP-4 Pressure Drop**

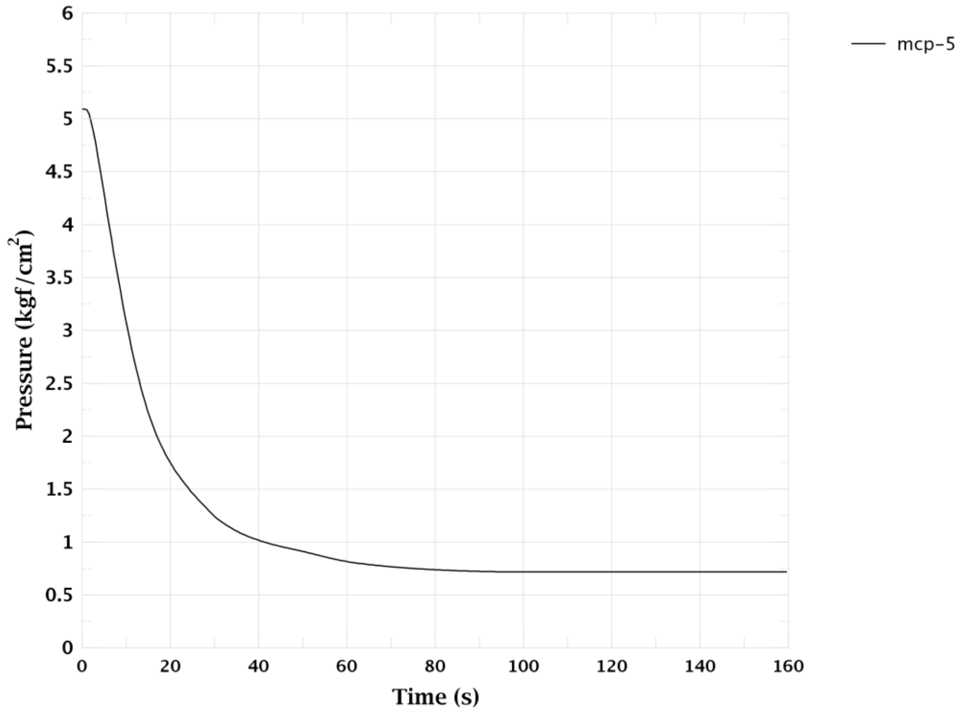



**Figure 162 MCP-5 Pressure Drop**

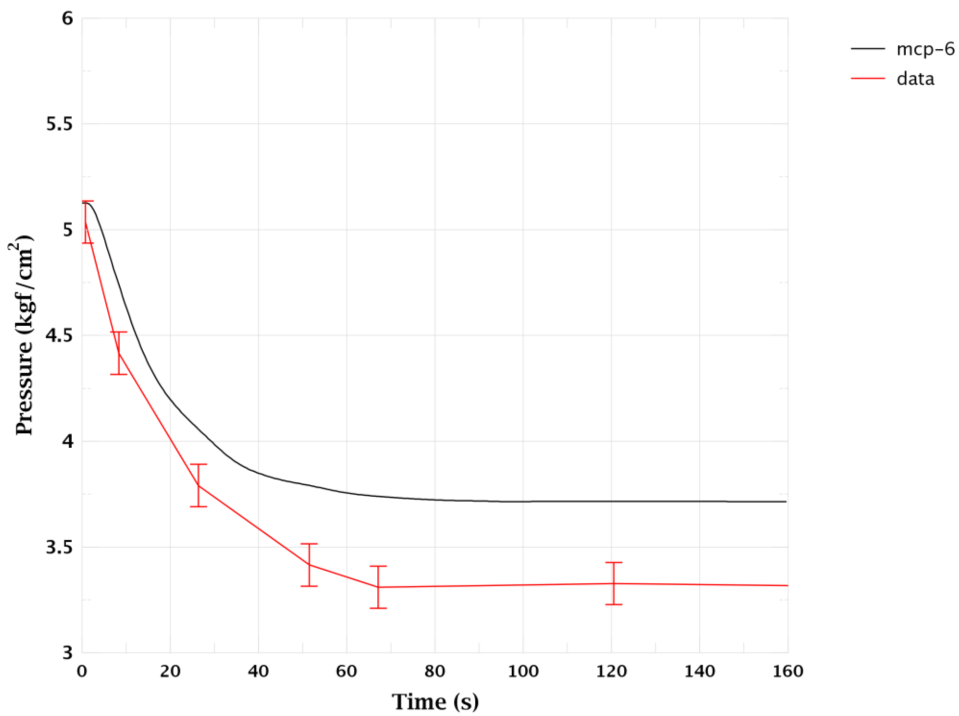



**Figure 163 MCP-6 Pressure Drop**

### 4.3.4 Conclusion

The comparative computer simulation of "Inadvertent reactor scram" incident with RNPP-1 model for TRACE code demonstrates that calculated parameters behavior is in general in good compliance with measured data.

As it can be seen from Fig. 123 at the beginning of incident the measured primary circuit pressure is ~120 kgf/cm$^2$ which is lower than nominal value of 125±1.2 kgf/cm$^2$ and according to PRZ heaters setpoints should results in operation of 4 out of 5 heaters groups. Since the reason for this deviation is not clearly understood, correspondent adjustment of calculated initial RCS pressure to match measured data was not performed and nominal value was used instead. This difference in calculated and measured RCS pressure slightly increases in the course of incident and by the end of calculation reaches ~10 kgf/cm$^2$. Nevertheless the qualitative RCS behavior is satisfactorily reproduced in calculation.

Calculated RCS temperatures (Fig. 132 – Fig. 143) are in good correspondence with measured data and with the exception of hot leg temperature of loop #1 lay within the measurement error band.

Good correspondence is also achieved for pressure difference across the reactor (Fig. 157) and MCPs during their coastdown (Fig. 158) while for running MCPs the calculated values at the end of transient are somewhat higher than the measured ones (Fig. 159 and Fig. 163).

The specific event sequence of the selected incident allowed to validate pump coastdown, operation of make-up and letdown controllers and interlocks, PRZ heaters operation logic, as well as evaluate correctness of reactor pressure losses.

# 5 CONCLUSIONS

After development of VVER-440/V-213 thermal-hydraulic model for TRACE code the validation calculations of several incidents which occurred at Rivne NPP Unit 1 were performed in order to justify capabilities of this model to simulate adequately the plant response during transients and accidents.

Since incident reports do not provide all information needed for correct incident simulation, some assumptions were introduced in calculation scenarios based on the results of preliminary calculations. Evaluation of incident reports also revealed several inconsistencies in the description related to equipment state/operation. Therefore in some cases the calculation scenario was altered comparing to the incident description.

Results of validation analyses demonstrate that calculated behavior of main primary and secondary circuit parameters is generally in good agreement with plant measured data. However, since simulated incidents do not cover all phenomena important for design basis accident analysis it is planned to extend model validation by performing comparative calculations of LOCA and secondary circuit break scenarios with application of VVER-440/V213 thermal-hydraulic models for TRACE and RELAP codes.

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K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)


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 440 model for TRACE code to simulate adequately the plant response during transients and accidents, the validation of the model was performed.

The report provides results of the validation calculations conducted with application of SSTC NRS model of VVER-440 for TRACE computer code. The calculation scenarios simulate actual incidents that occurred at Ukrainian NPPs.
The incidents simulated include:
• Reactor scram caused by concrete slab drop to the connection lines of house loads power supply transformer;
• Reactor scram transient initiated by 6 kV switch short circuit;
• Inadvertent reactor scram.


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TRACE
Reactor Power Controller, Russian designation(ARM)
Fast-acting Steam Isolation Valve(FASIV)
Code Application & Maintenance Program (CAMP)
Steam Dump Valve to the Deaerator(BRU-D)
Steam Dump Valve to the Turbine Condenser(BRU-K)
Emergency Core Flooding System(ECFS)
Protective Tubes Unit(PTU)

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