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International Agreement Report

TRACE VVER-440/V-213 Model Cross-Code 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 comparison calculations conducted with application of SSTC NRS model of VVER-440/V-213 for TRACE and RELAP5 computer codes. The calculation scenarios analyzed include design basis accidents and transients from several initiating event groups usually evaluated in safety analysis reports.

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EXECUTIVE SUMMARY

This report is developed in the framework of the Implementing Agreement on Thermal-Hydraulic Code Applications and Maintenance between United States Nuclear Regulatory Commission and the State Nuclear Regulatory Inspectorate of Ukraine.

At the previous stages of these activities existing RELAP5 model for VVER-440 was converted to TRACE code format and set of validation calculations based on actual incidents were performed.

This work is aimed at the comparative TRACE and RELAP calculations for selected design basis accident scenarios. Thus, this report contains numeric analyses results of the following initiating events:

- guillotine break of the main steam header;
- loss of vacuum in the condenser of one of the turbines;
- trip of 4 out of 6 reactor coolant pumps;
- uncontrolled withdrawal of control group of control assemblies from the reactor core with a normal operating speed of 20 mm/s;
- break of the pressurizer surge line.

Comparison of the results obtained with TRACE and RELAP5 models indicates some differences in calculated parameters. In particular, the differences in the primary circuit pressure that were observed in some of the scenarios are caused by different mathematical models and correlations for steam condensation, which are used in the special PRESSURIZER model in TRACE, as compared to the pressurizer modelling in RELAP5. Cladding temperature differences are related to the specifics of the heat structure modeling approach and the absence of TRACE correlation options, which does not allow more precise model adjustment to ensure complete convergence with the relevant RELAP5 models.

The results of cross-code validation calculations demonstrate that developed VVER-440/V213 thermal-hydraulic model for TRACE code is able to reproduce adequately the plant response to transients and accidents without core melt that were calculated previously in safety analysis reports using the RELAP5 model. For the majority of plant parameters good correspondence between TRACE and RELAP5 results is obtained.

ABBREVIATIONS AND ACRONYMS

AFW	Auxiliary Feedwater System
ARM	Reactor Power Controller, Russian designation
BRU-A	Steam Dump Valve to Atmosphere
BRU-K	Turbine Bypass to Condenser
CAMP	Code Maintenance and Assessment Program
DBA	Design Basis Accident
DG	Emergency Power Supply Diesel-Generator
ECCS	Emergency Core Cooling System
EFW	Emergency Feedwater System
FASIV	Fast-acting Steam Isolation Valve
HA	Hydroaccumulators
HPIS	High Pressure Injection System
IE	Initiating Event
LOCA	Loss of Coolant Accident
LPIS	Low Pressure Injection System
MFW	Main Feedwater System
MSH	Main Steam Header
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
NPP	Nuclear Power Plant
PRZ	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RNPP	Rivne Nuclear Power Plant
RPL	Reactor Power Limiter
SG	Steam Generator
SLP	Sequential DG Loading Program
SNRIU	State Nuclear Regulatory Inspectorate of Ukraine
SRV	Safety Relief Valve
SSTC NRS	State Scientific and Technical Center for Nuclear and Radiation Safety
USNRC	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 [1], [2] 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.

As the first step of these activities, the existing SNRIU/SSTC NRS RELAP5 model for VVER-440 was converted to TRACE code format.

In order to justify capabilities of VVER-440 model for TRACE code to simulate adequately the plant response during transients, calculations of several events that had actually occurred at Ukrainian NPPs were conducted. Results of TRACE simulations (validation) of these events in comparison to the plant measured data are provided in NUREG/IA-0485 [3]. Since these events do not cover all the phenomena expected to occur during accidents, the VVER-440 input model validation activities were extended by performing the comparative calculations of selected design basis transient and accident scenarios with RELAP5 and TRACE model. This report provides the results of comparative calculations.

Section 2 of the report briefly describes the validation process and refers to the description of the main primary and secondary systems of Rivne NPP (RNPP) unit 1 (VVER-440/V-213 design) which are important for development of thermal-hydraulic model, as well as to a description of RNPP Unit 1 model.

The results of comparative TRACE and RELAP calculations for selected design basis accident (DBA) scenarios are provided in Section 0 of the report.

For each scenario the following information is provided:

- brief description of the scenario;
- initial and boundary conditions selected for calculation;
- sequence of events;
- description of calculation results;
- plots of the main primary and secondary circuit parameters.

2 BRIEF DESCRIPTION OF MODEL VALIDATION PROCESS

After preparation of RNPP unit 1 model for TRACE code and adjustment of steady state calculation several transient calculations were performed simulating the actual incidents that had actually occurred at Rivne NPP and the results of calculations were compared with the plant measurement data. In particular the following incidents were simulated:

- 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.

The results of these validation calculations, as well as a brief description of the model and of the main VVER-440/V-213 design features are presented in NUREG/IA-0485 report [3]. In general, the results demonstrate that calculated behavior of the main primary and secondary circuit parameters is in good agreement with the plant measured data.

However, since simulated incidents do not cover all phenomena that are important for accident analysis and allow to check correctness of modelling for the limited number of plant systems only it was decided to extend TRACE model validation by performing comparative calculations of several scenarios with TRACE and RELAP codes. For this purpose the following DBA scenarios at full power operation are simulated:

- main steam header (MSH) break;
- loss of turbine condenser vacuum;
- trip of 4 out of 6 reactor coolant pumps (RCP);
- uncontrolled withdrawal of control rods group;
- pressurizer (PRZ) surge line break.

These scenarios cover the majority of DBA initiating events groups, including loss of coolant accidents and secondary circuit breaks that can not be evaluated otherwise due to lack of correspondent plant incidents or reliable measured data.

For each scenario the identical initial and boundary conditions were specified for TRACE and RELAP calculations, and the results obtained with these models were compared.

3 RESULTS OF COMPARATIVE CALCULATIONS

3.1 MSH Break

3.1.1 Brief Description of Initiating Event

This initiating event (IE) assumes postulated guillotine break of the main steam header with a diameter of 465×16 mm, that leads to a sharp increase of heat removal by the secondary circuit. According to the expected frequency of occurrence the IE is categorized as a design basis accident.

3.1.2 Initial Conditions

The initial conditions selected for calculation correspond to normal full power plant operation (taking into account allowances due to plant control systems operation) and are presented in Table 3-1.

Table 3-1 Results of Steady State Calculation

Parameter	Units	Design value	Calculated value
Core thermal power	MWt	1375±27	1402
Reactor outlet pressure	kgf/cm^2	125	124
Reactor inlet temperature	°C	267	267
Coolant temperature at reactor outlet	°C	297.9	297
Coolant heating in the reactor	°C	30.9	30
Maximum temperature of fuel cladding external surface	°C	335	327
Reactor coolant flow rate	m^3/h	40600±400	40200
PRZ level	m	5.96	5.97
SG pressure	kgf/cm^2	47	45.9-46.5
SG level	m	2.105	2.1-2.11
SG steam production	t/h	450	444 - 467
SG feedwater temperature	°C	223	223
Water temperature in ECCS tanks	°C	55-60	60.0
Water temperature in ECCS HA	°C	60.0	60.0

3.1.3 Boundary Conditions

The following assumptions on systems availability and configuration are selected in the analysis.

The break occurs at MSH semi-header connected to turbine No.1 and is located in close proximity to fast-acting steam isolation valves (FASIV) separating MSH into two semi-headers. Instant opening of the valves simulating MSH break is modeled.

No operator actions are considered.

Operation of reactor power controller (ARM) and reactor power limiter (RPL) to decrease power is not accounted. At the moment of IE occurrence ARM operates in "T" mode (MSH pressure maintenance), and automatic switching to "N" mode (neutron power maintenance) is not considered.

Conservatively, loss of normal power supply is postulated to occur simultaneously with reactor scram. Operation of Level 4, Level 3 and Level 2 emergency reactor protection is not taken into account.

A failure of one out of three emergency feedwater (EWF) pumps (namely, EFWP-1) to start is postulated as a single failure.

3.1.4 Calculation Results

Sequence of events for this accident is presented in Table 3-2.

Table 3-2 Sequence of Events for MSH Break Accident

TRACE Time, s	RELAP5 Time, s	Event	Description
0.0	0.0	Guillotine break of MSH semi-header No.1	Start of double-ended steam discharge from MSH
0.07	0.07	Maximal cladding temperature 330 °C (TRACE) and 334 °C (RELAP5)	
0.1	0.1	Closure of turbine no.1 stop valves	MSH-1 pressure decrease to 38 kgf/cm^2
0.2	0.2	Closure of turbine no.2 stop valves	MSH-2 pressure decrease to 40 kgf/cm^2
0.70	0.74	Scram signal	Due to closure of the stop valves of last operating turbine
0.70	0.74	Loss of normal power supply	Boundary condition
1.70	1.74	Start of control assemblies drop due to reactor scram	Delay of 1 s from scram signal is assumed
2.70	2.74	Actuation of ECCS safeguard and start of emergency power supply diesel-generators (DG)	Due to loss of normal power supply
5.70	5.74	Trip of all RCPs Switching-off of all groups of PRZ electric heaters Trip of main feedwater (MFW) pumps	Due to loss of normal power supply
5.70	5.74	Signal to close main steam isolation valves (MSIV) dividing MSH and MSIV at the main steam lines (MSL)	Due to signal "Increase of pressure drop rate in each MSH semi-header to 0.7 kgf/cm^2/s at cold legs temperature >150 °C in 2 out of 6 loops holding for 1 s"
10.70	10.74	Full closure of all MSIVs	Closure time is 5 s
17.70	17.74	Start of DG sequential loading program (SLP)	DG start-up time of 15 s is assumed
35.0	34.0	Minimum pressure at the reactor outlet (115.2/116.5 kgf/cm^2)	
35.0	34.0	Minimum PRZ level is 4.7 m	
45.0	42.74	Start of boric acid supply to the reactor coolant system (RCS) by high pressure injection system (HPIS)	Due to ECCS signal taking into account a transport delay
180.0	170.0	Maximum pressure at reactor outlet is 136 kgf/cm^2	
201.0	200.0	End of RCP coast-down	
400.0-2000.0	400.0-2000.0	Periodic actuation of SG safety relief valves (SRV)	Secondary pressure change within 48-56 kgf/cm^2
2000.0	2000.0	End of calculation	Stabilization of parameters

After MSH break, abrupt pressure drop occurs in the main steam header (Figure 3-32 and Figure 3-33). Within 0.3 s due to decrease of pressure in MSH semi-headers the turbines' stop valves are closed, both turbines are tripped and due to a closure of 2/4 stop valves of last operating turbine the reactor scram signal is generated at 0.74 s of the accident (Figure 3-40).

According to the scenario the loss of normal power supply is postulated simultaneously with the scram that leads to a trip of all RCPs, MFW pumps and PRZ electric heaters.

At 1.03 s (with a delay of 1.0 s) a “MSH break” signal is formed due to increase of pressure drop rate in MSH, and then a signal to open cutoff valves at HPIS charging lines, as well as a signal to close all MSIV (at MSH and MSLs) are actuated.

At 45.0/42.7 s of the calculation, after postulated delays for DG and HPIS start-up (after loss of normal power supply) and HPIS transport delay, boric acid injection to the primary circuit by HPIS pumps is initiated (Figure 3-45 – Figure 3-47). Due to HPIS injection RCS pressure and PRZ level recover and injection rate decreases.

Up to 200.0 s, RCP coast-down ends and natural circulation in all RCS loops is established (Figure 3-16 – Figure 3-21).

After closure of all MSIVs, the secondary coolant loss is terminated. Due to HPIS injection and RCS coolant heat-up the primary pressure increases up to 136.0 kgf/cm^2 at 180.0 s of the accident. Lack of heat removal from the secondary side results in an increase of secondary circuit pressure that leads to the opening of control SRVs at all SGs at ~400.0 s (Figure 3-26 – Figure 3-31).

The maximum secondary pressure reached in the calculation is 57.0 kgf/cm^2, and the maximal cladding temperature after scram is 309.0/301.0 °C.

The plots of the main parameters of calculation are presented below on Figure 3-1 – Figure 3-51.

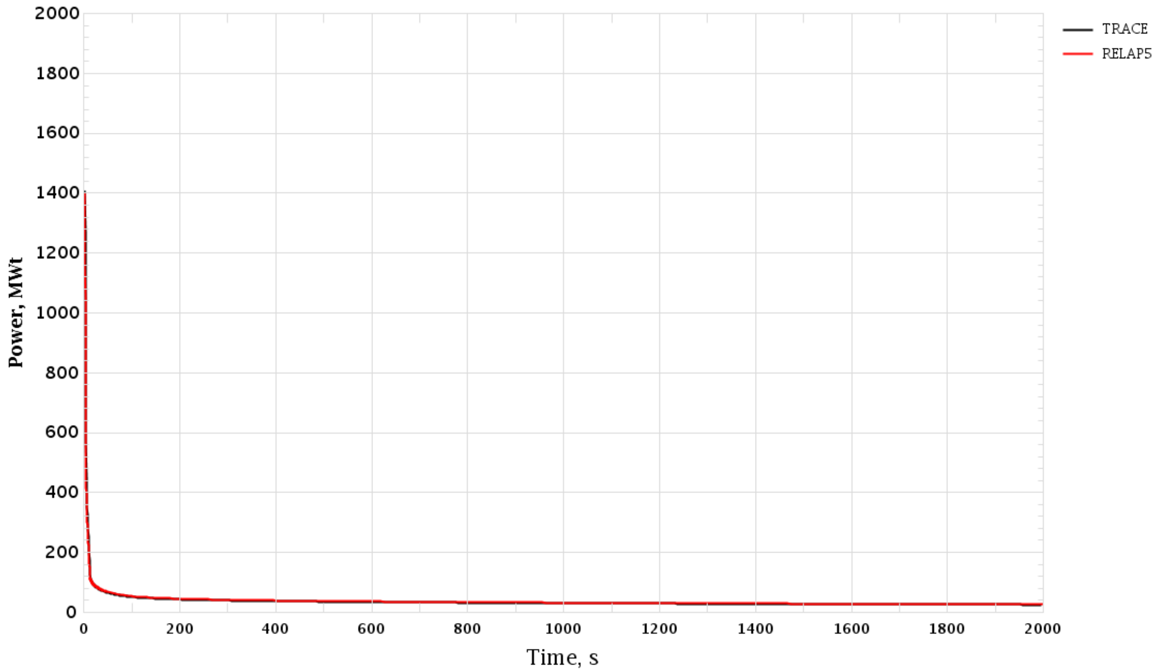

Figure 3-1 MSH Break. Core Thermal Power

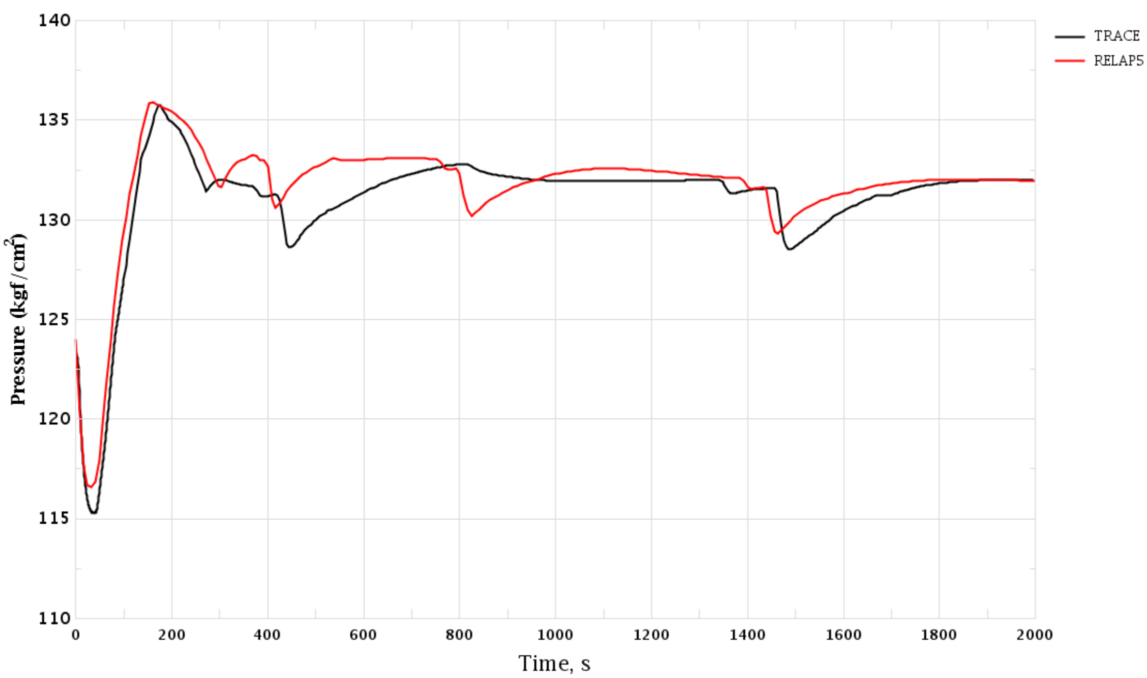

Figure 3-2 MSH Break. RCS Pressure

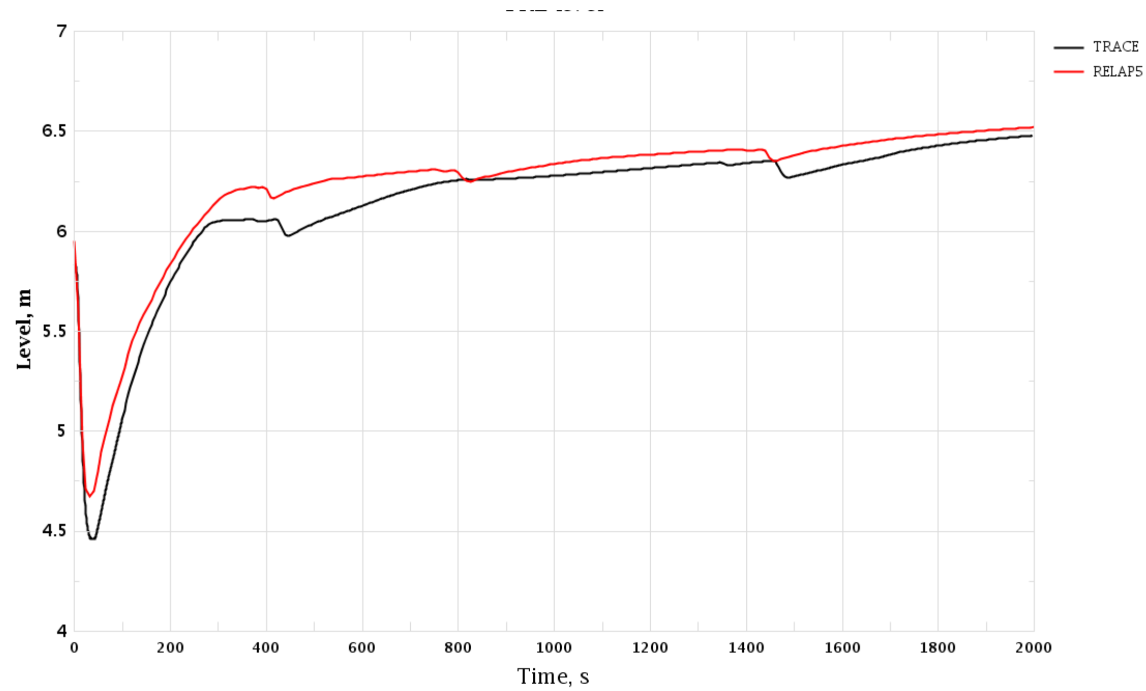

Figure 3-3 MSH Break. Pressurizer Level

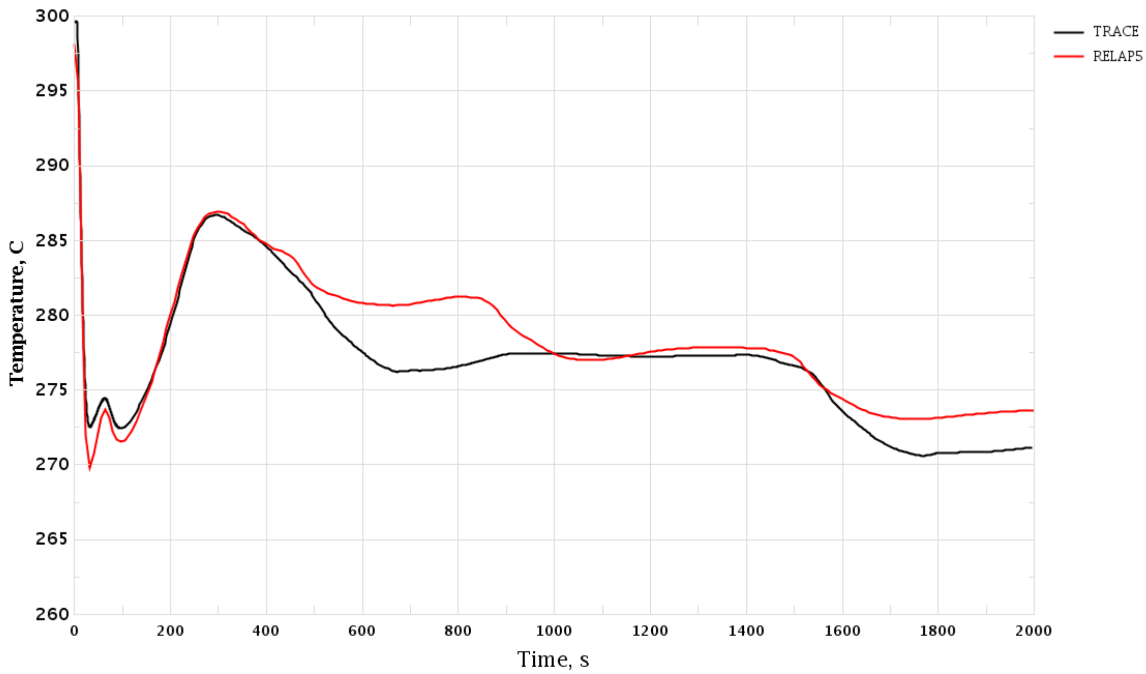

Figure 3-4 MSH Break. Coolant Temperature in Hot Leg, Loop 1

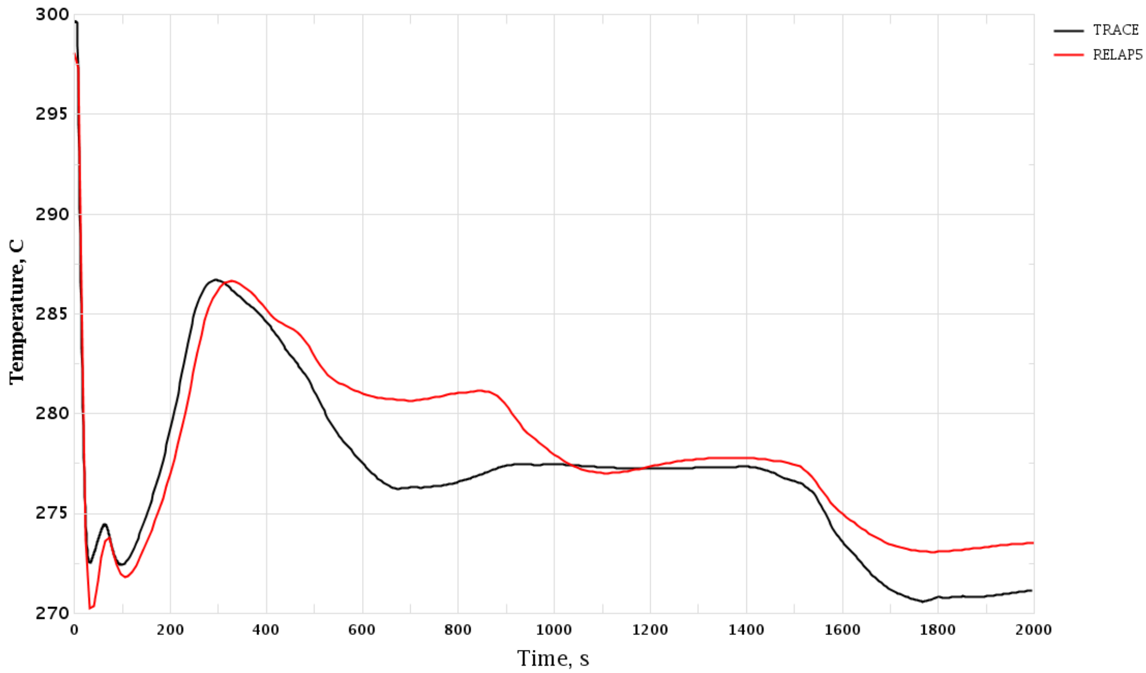

Figure 3-5 MSH Break. Coolant Temperature in Hot Leg, Loop 2

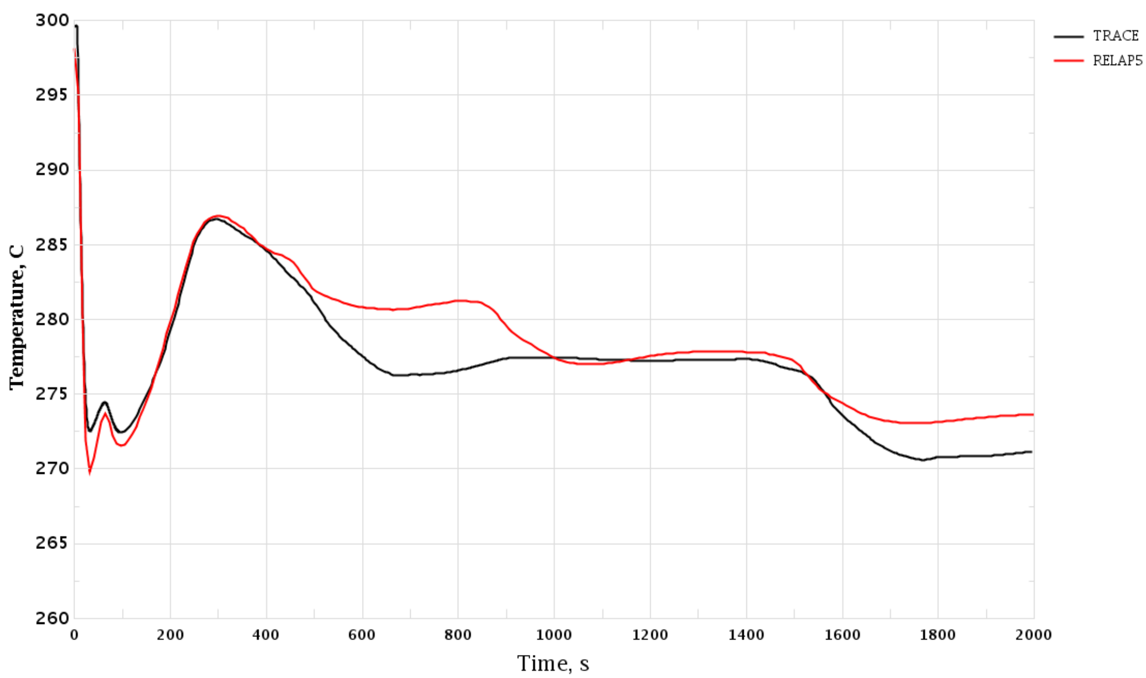

Figure 3-6 MSH Break. Coolant Temperature in Hot Leg, Loop 3

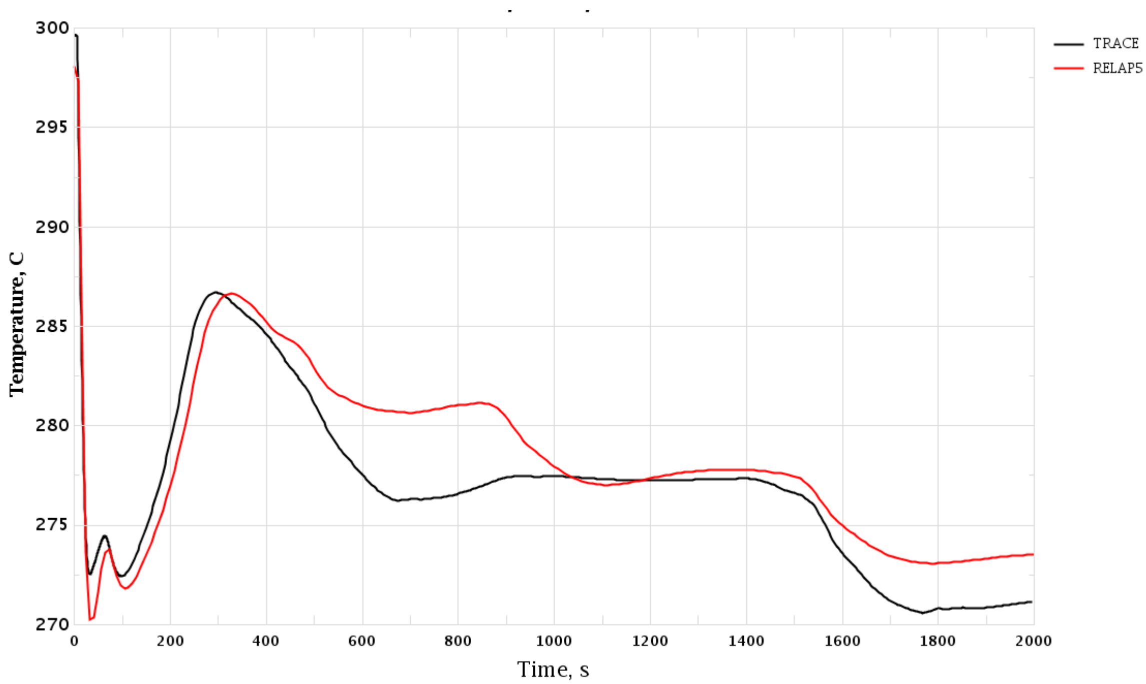

Figure 3-7 MSH Break. Coolant Temperature in Hot Leg, Loop 4

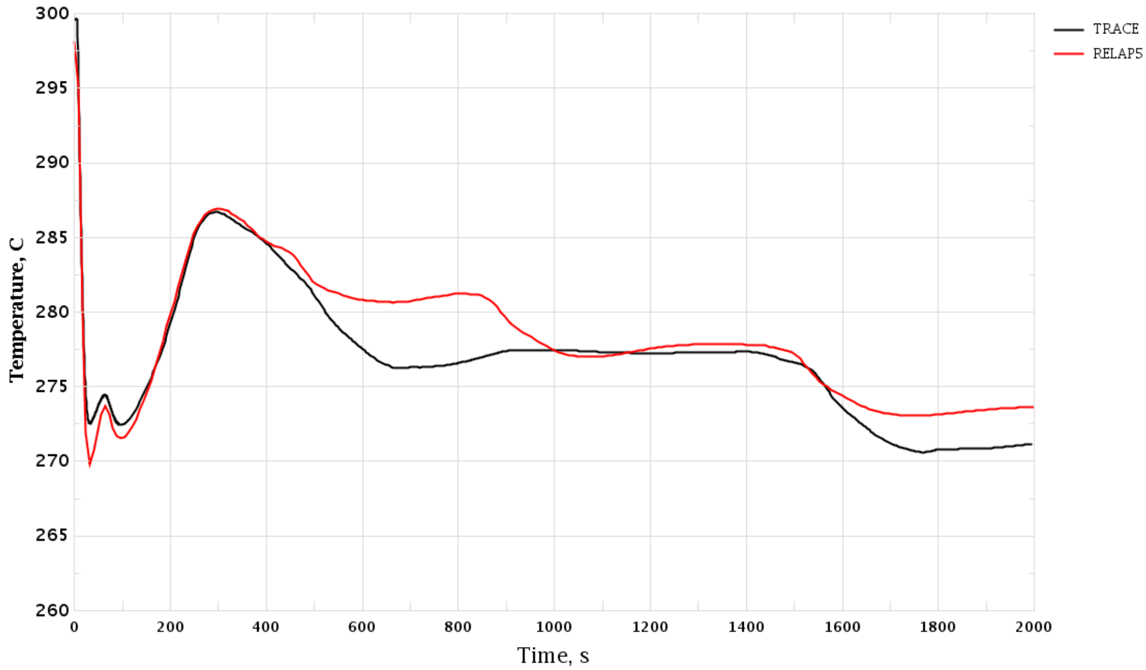

Figure 3-8 MSH Break. Coolant Temperature in Hot Leg, Loop 5

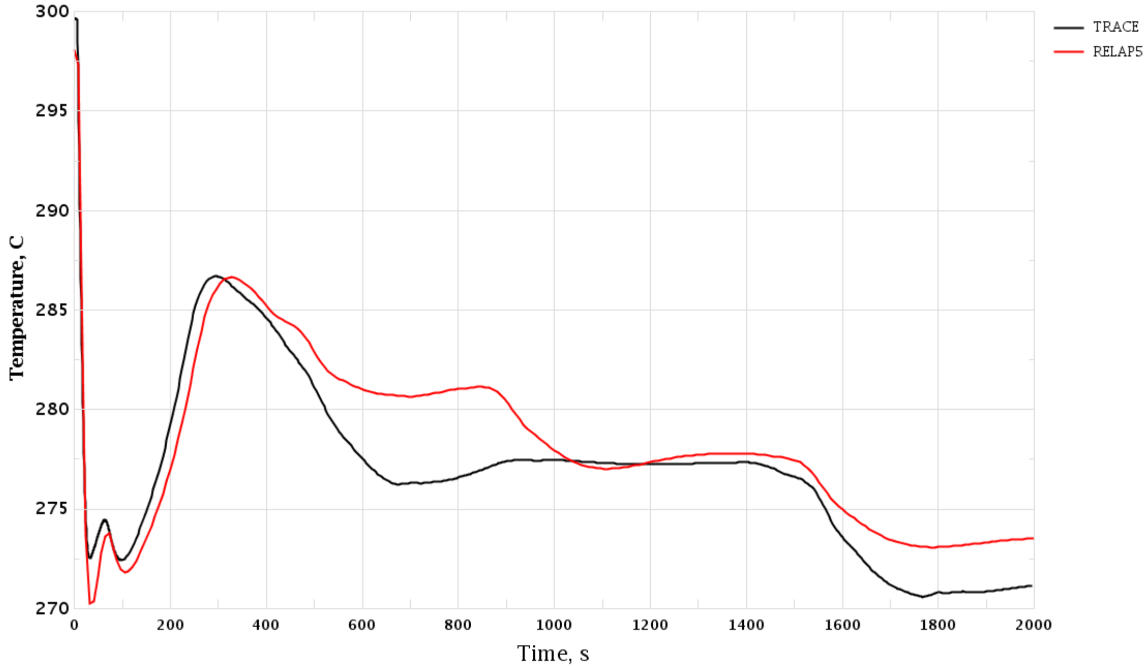

Figure 3-9 MSH Break. Coolant Temperature in Hot Leg, Loop 6

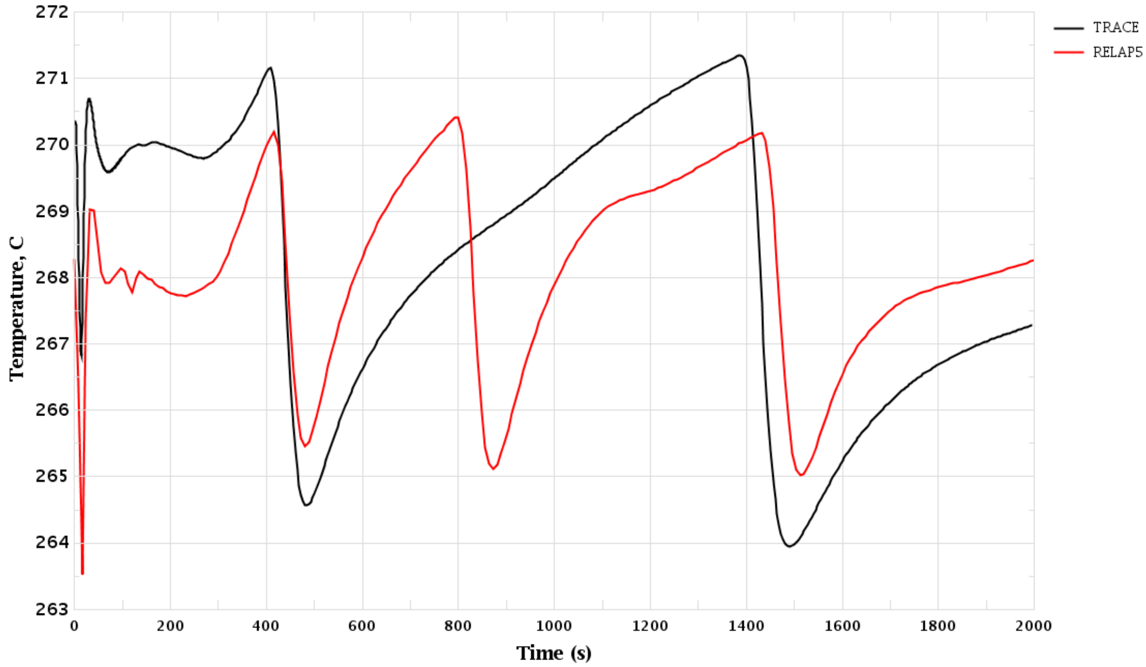

Figure 3-10 MSH Break. Coolant Temperature in Cold Leg, Loop 1

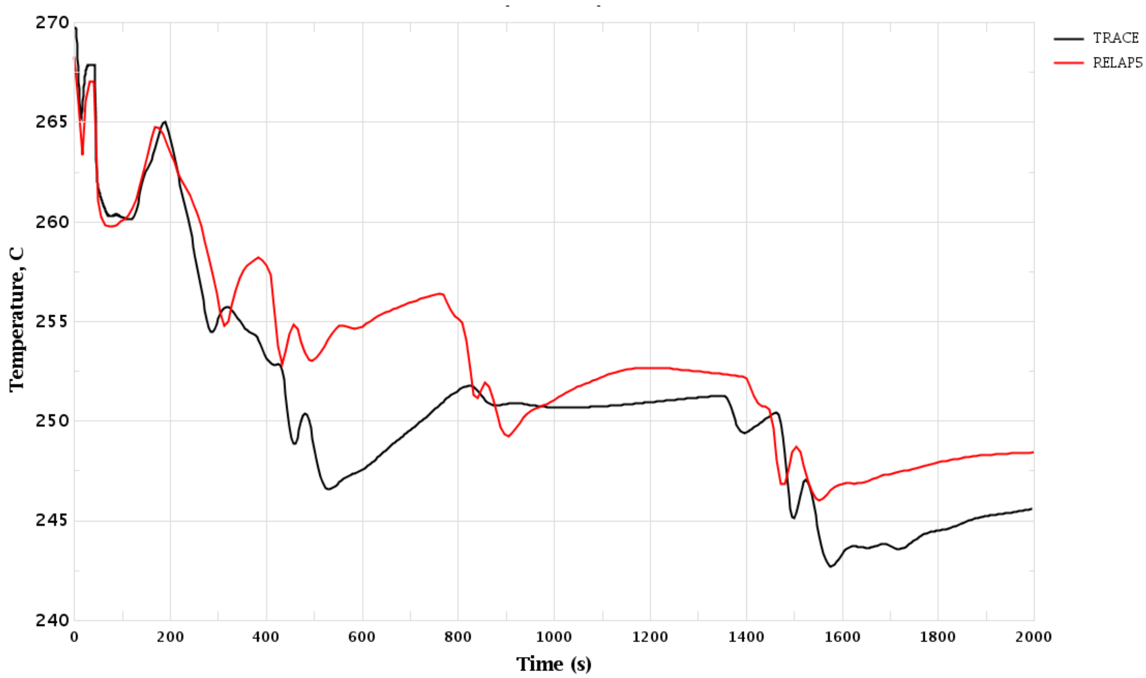

Figure 3-11 MSH Break. Coolant Temperature in Cold Leg, Loop 2

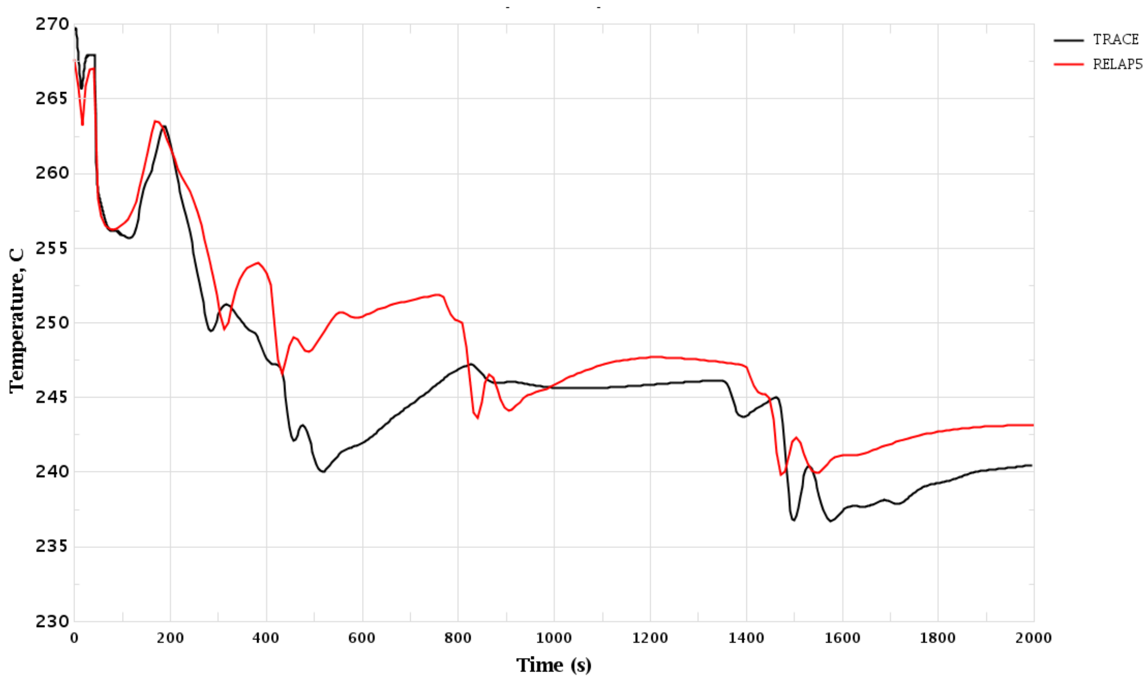

Figure 3-12 MSH Break. Coolant Temperature in Cold Leg, Loop 3

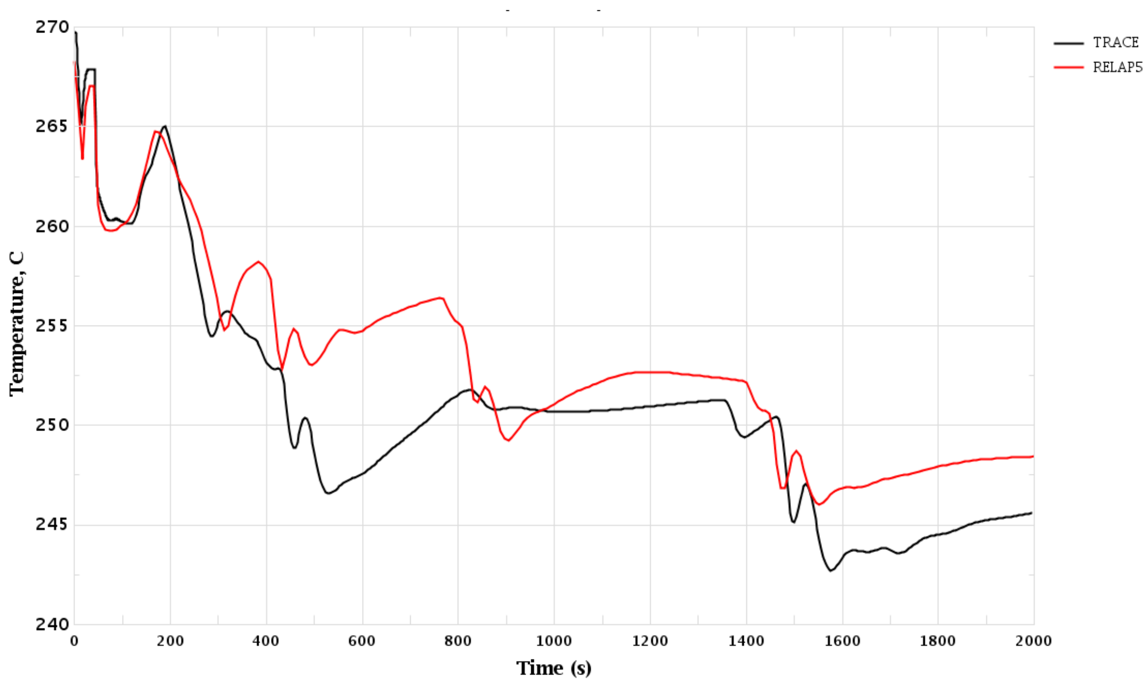

Figure 3-13 MSH Break. Coolant Temperature in Cold Leg, Loop 4

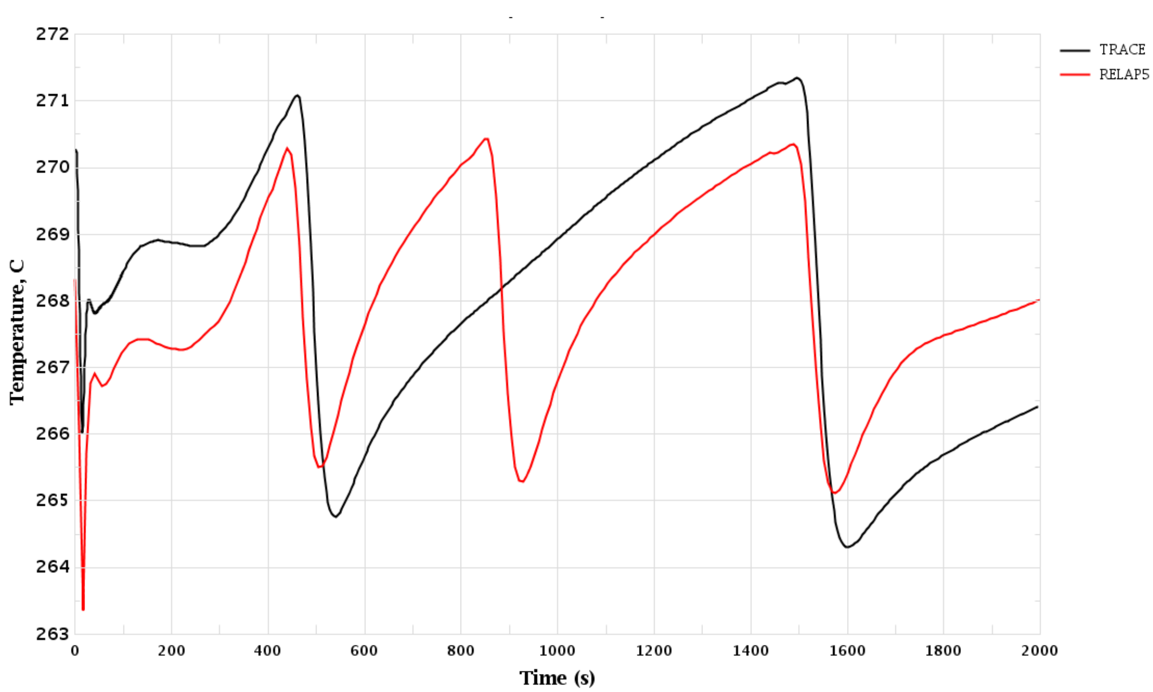

Figure 3-14 MSH Break. Coolant Temperature in Cold Leg, Loop 5

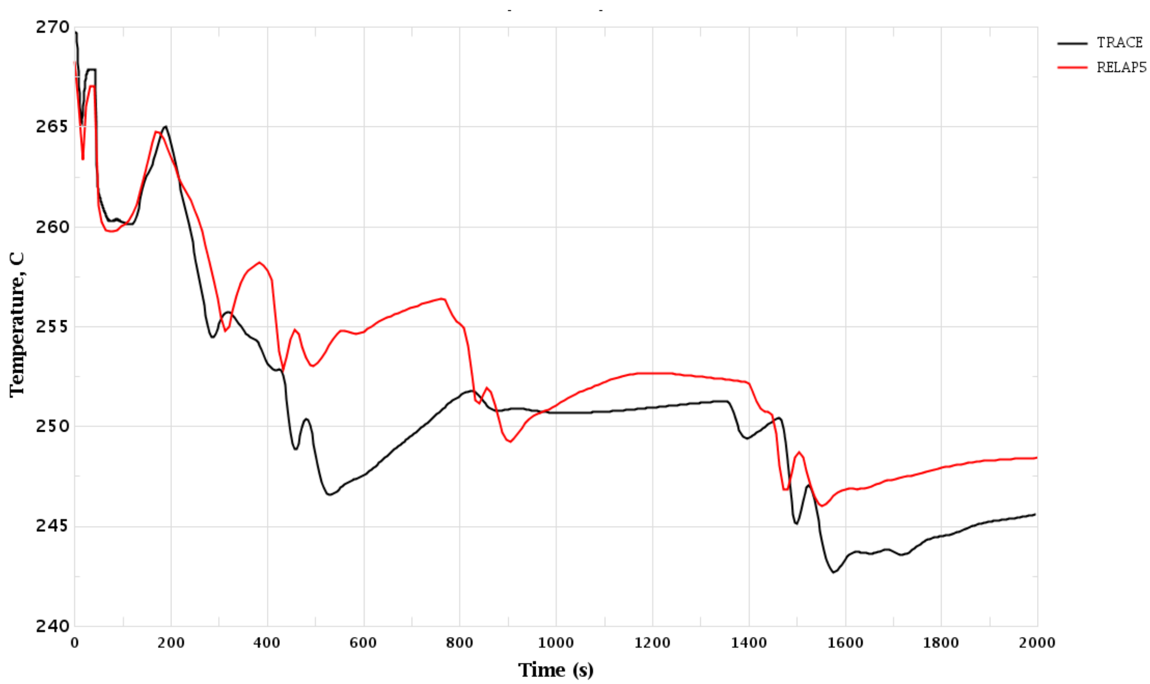

Figure 3-15 MSH Break. Coolant Temperature in Cold Leg, Loop 6

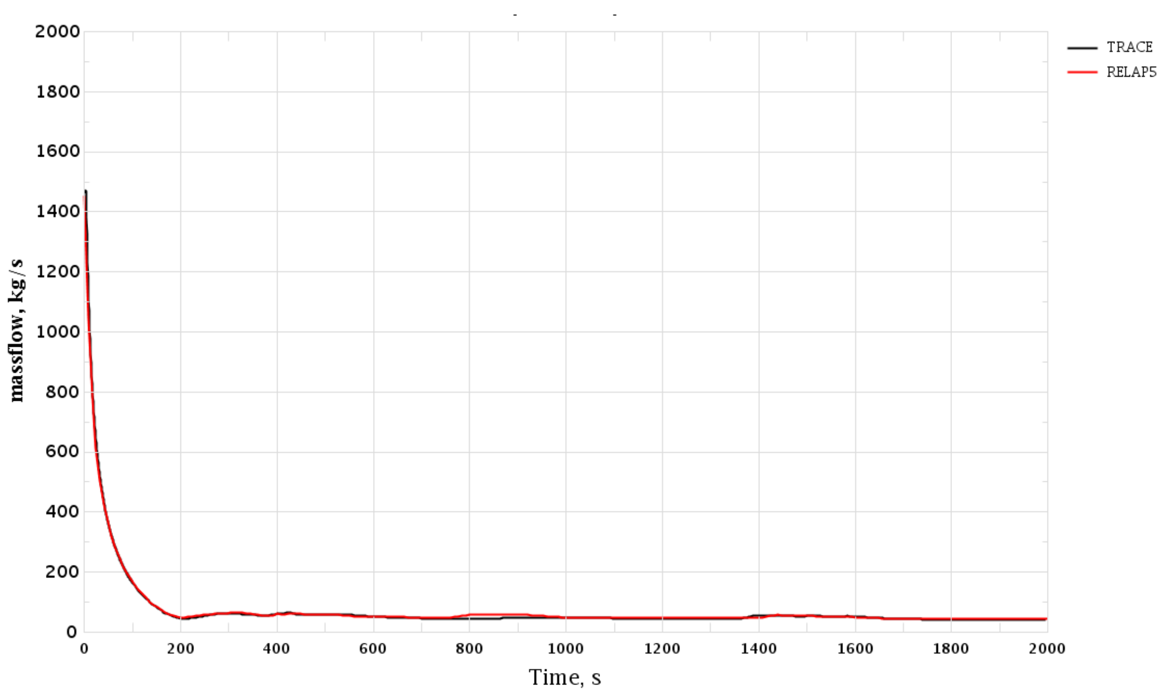

Figure 3-16 MSH Break. RCS Loop 1 Mass Flow Rate

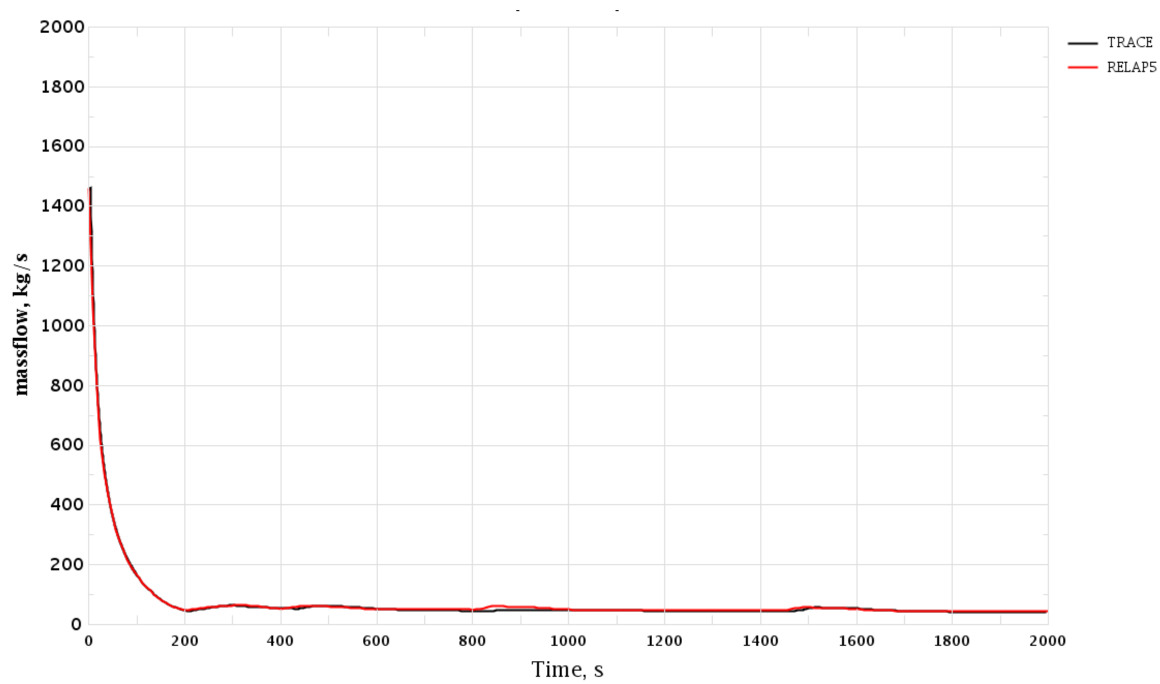

Figure 3-17 MSH Break. RCS Loop 2 Mass Flow Rate

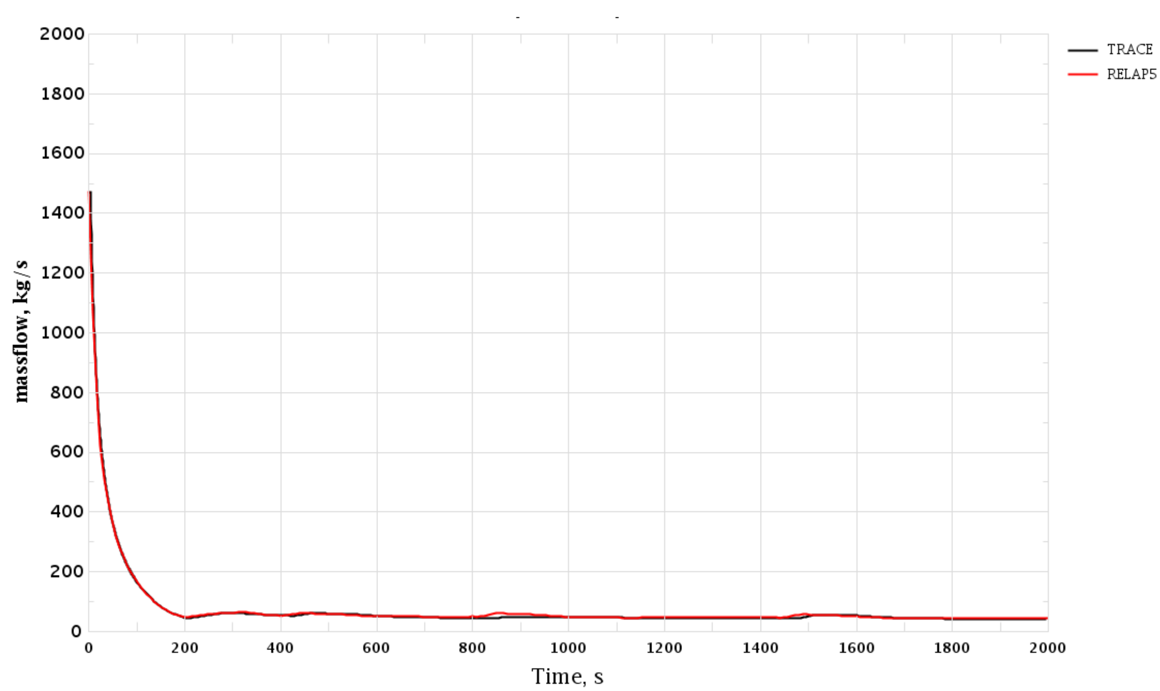

Figure 3-18 MSH Break. RCS Loop 3 Mass Flow Rate

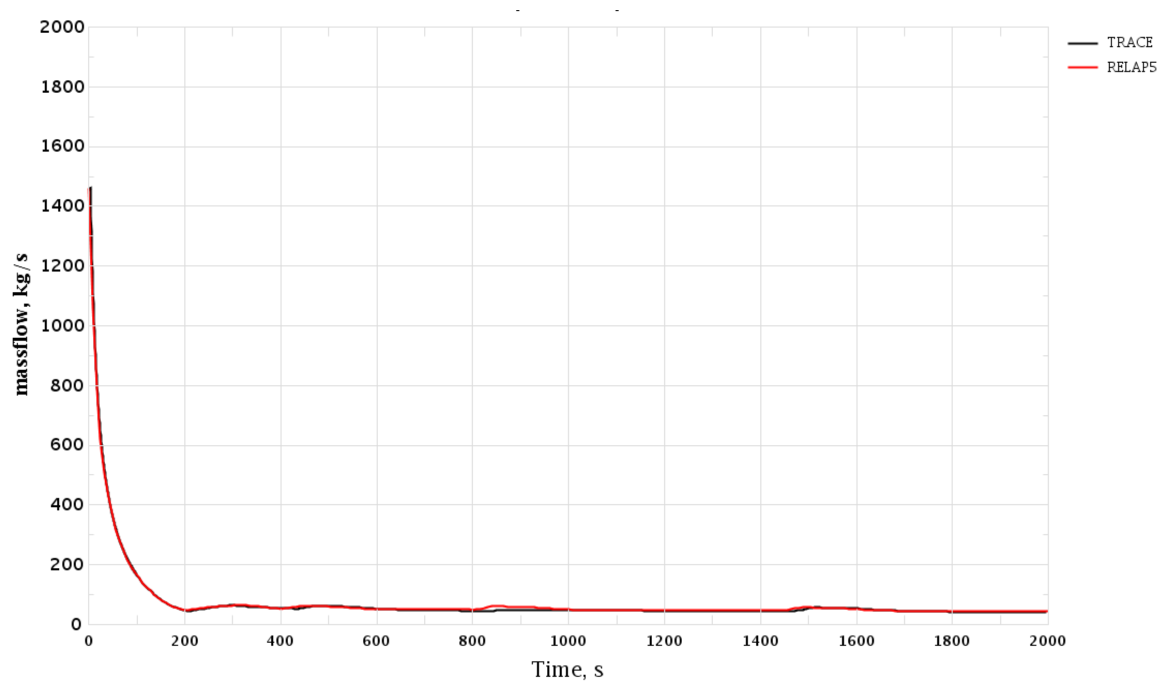

Figure 3-19 MSH Break. RCS Loop 4 Mass Flow Rate

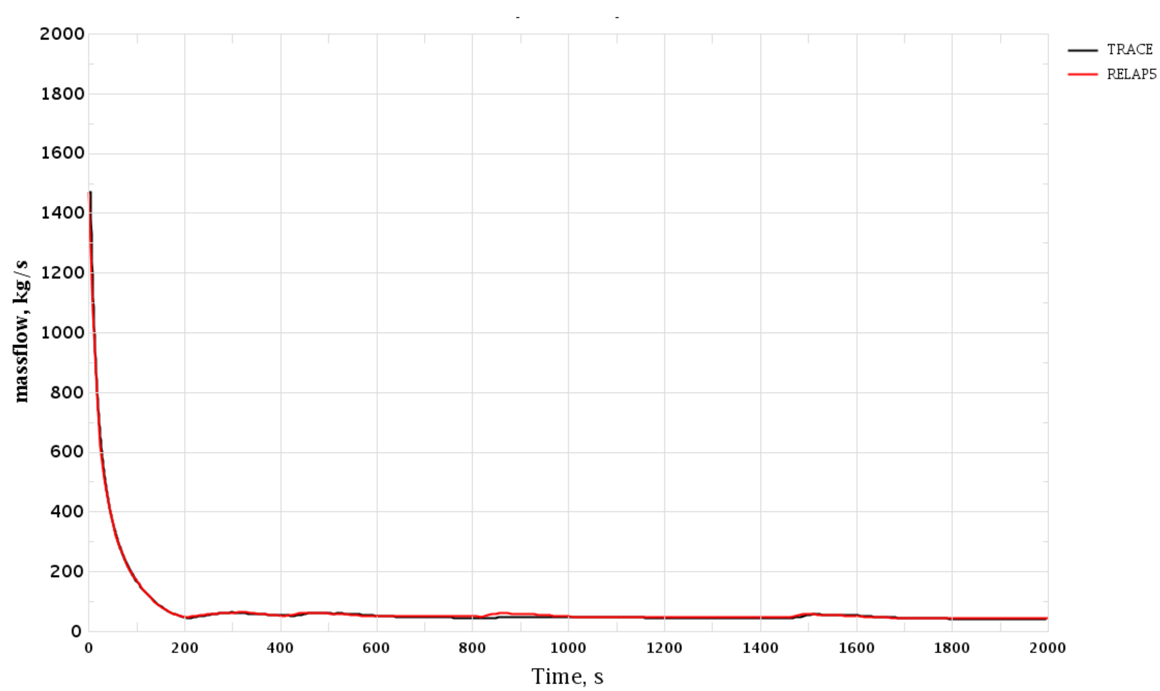

Figure 3-20 MSH Break. RCS Loop 5 Mass Flow Rate

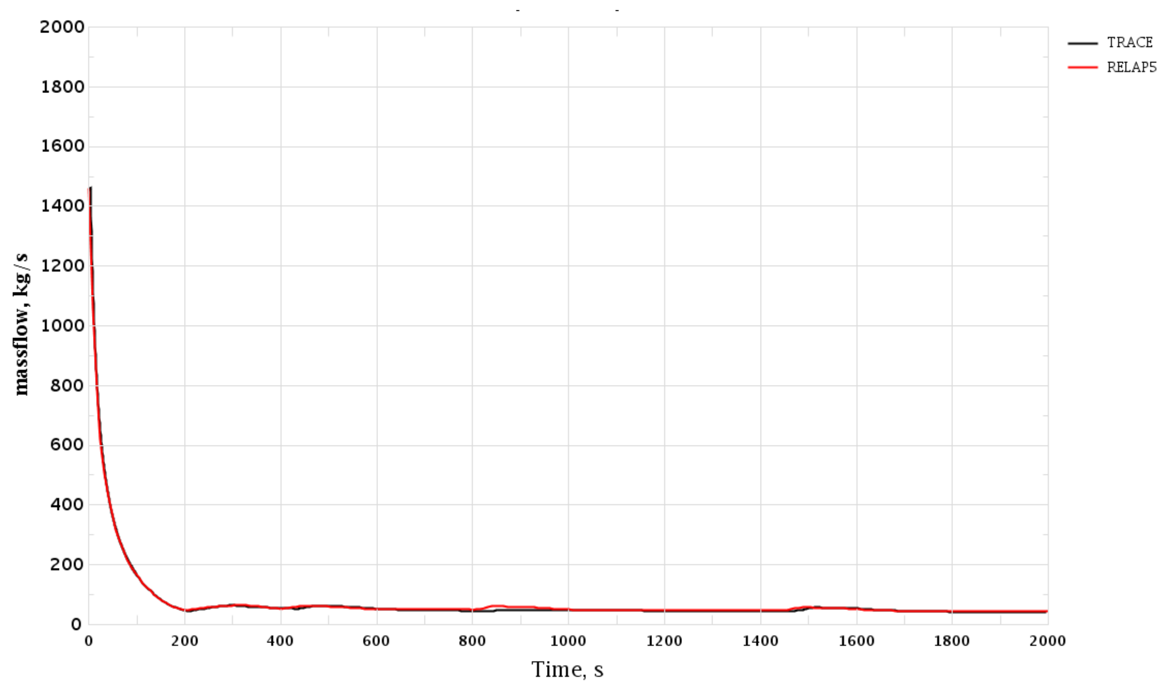

Figure 3-21 MSH Break. RCS Loop 6 Mass Flow Rate

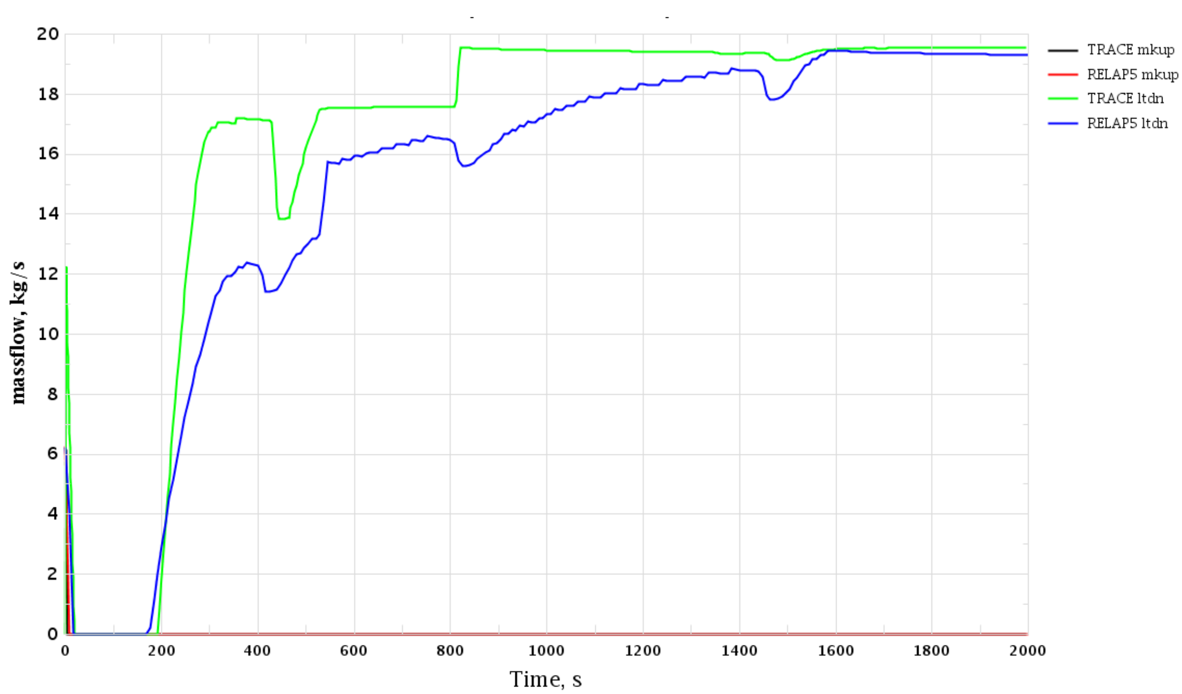

Figure 3-22 MSH Break. Make-Up and Let-Down Mass Flow Rate

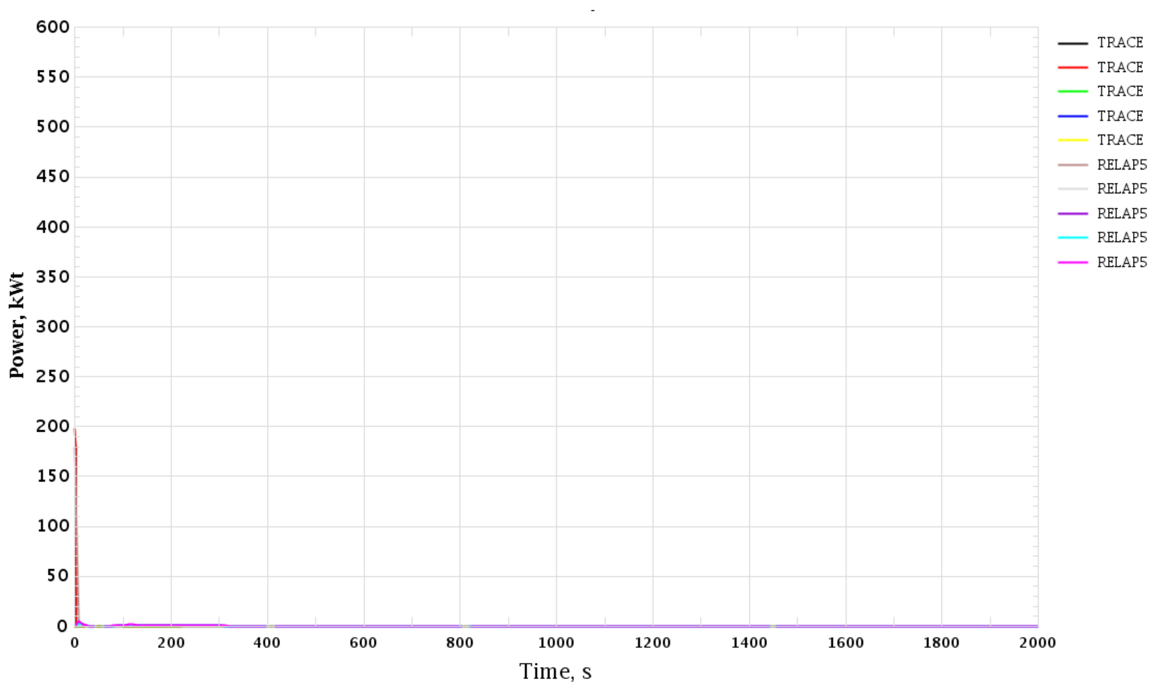

Figure 3-23 MSH Break. PRZ Heaters Power

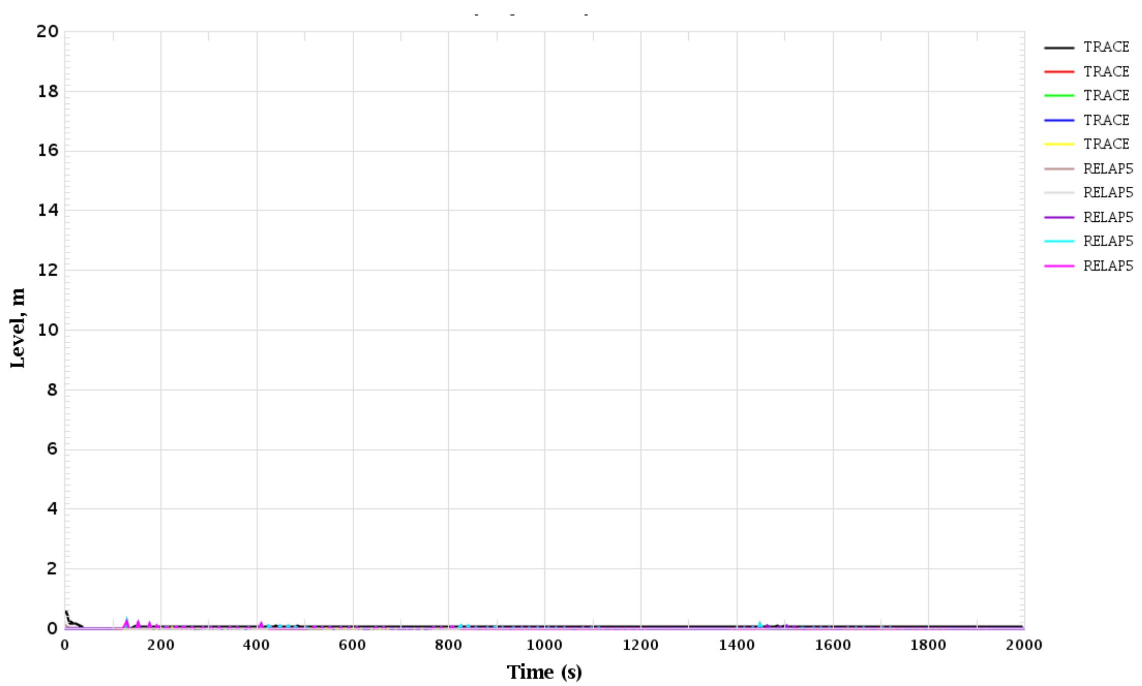

Figure 3-24 MSH Break. PRZ Spray Mass Flow Rate

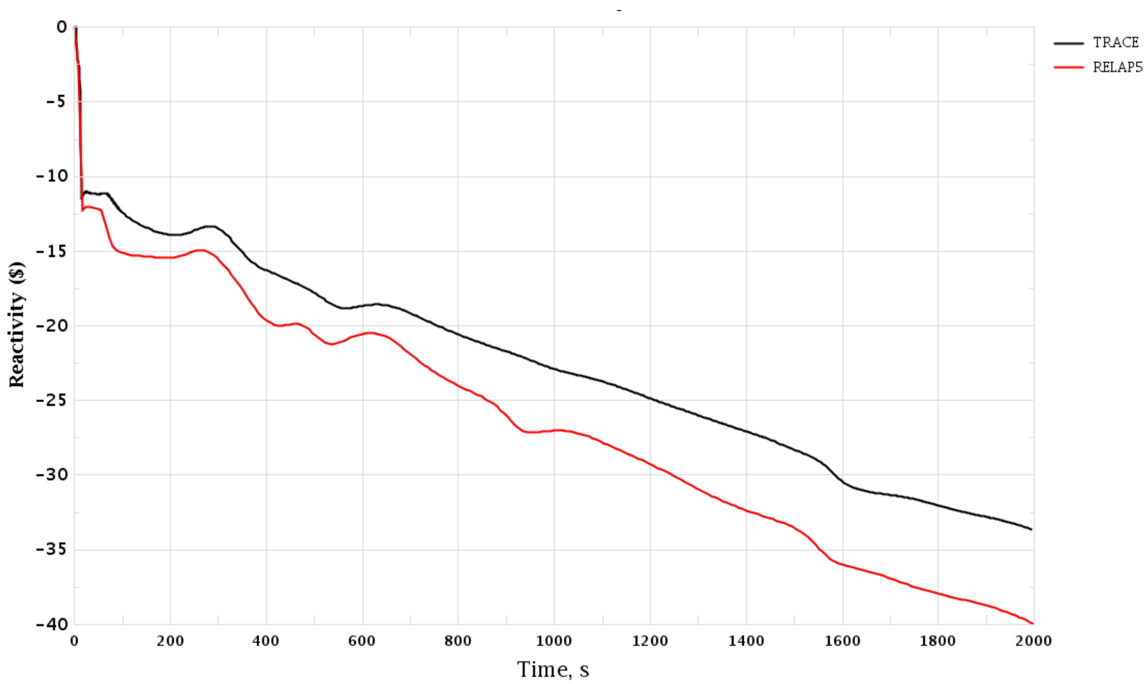

Figure 3-25 MSH Break. Core Reactivity

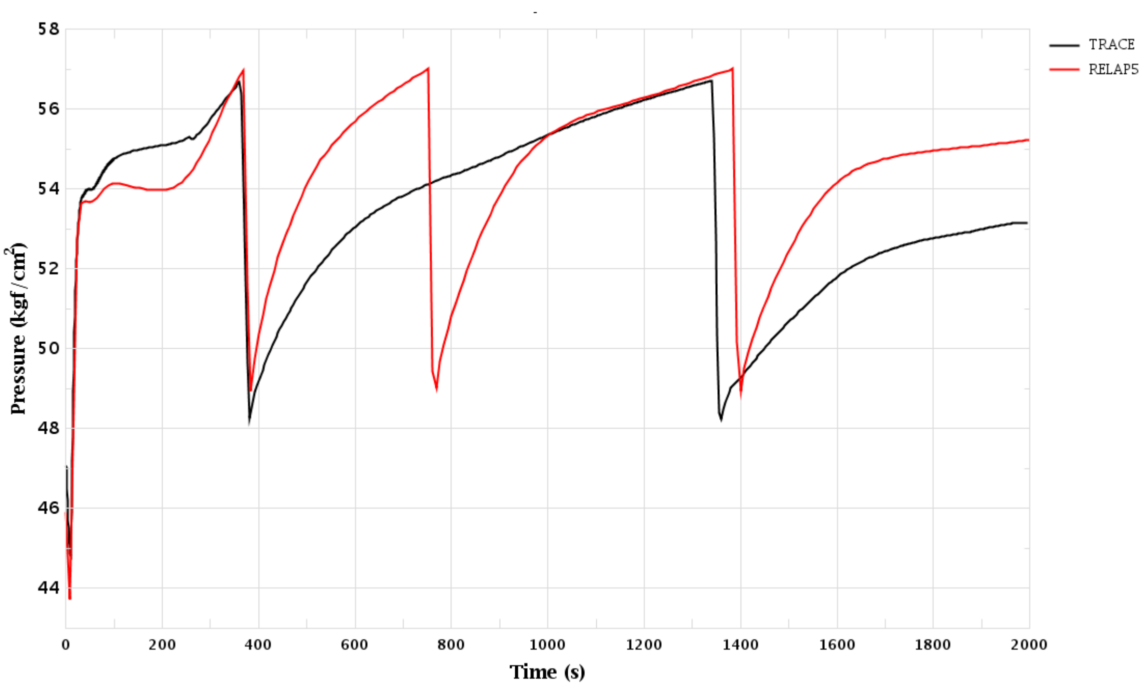

Figure 3-26 MSH Break. SG-1 Pressure

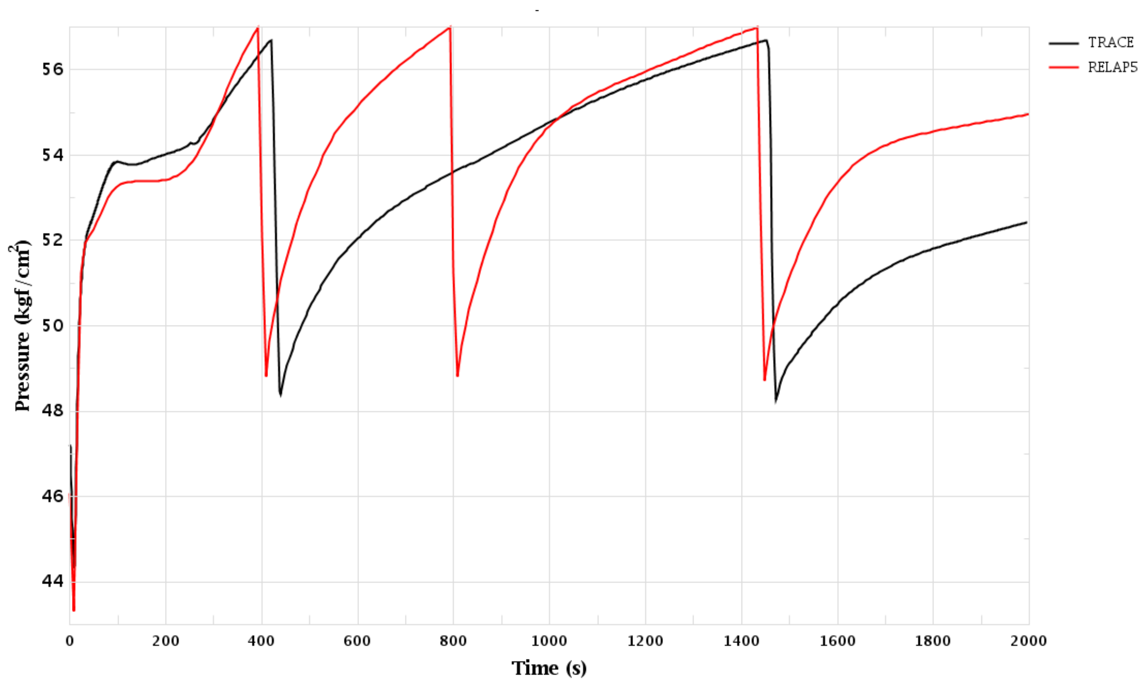

Figure 3-27 MSH Break. SG-2 Pressure

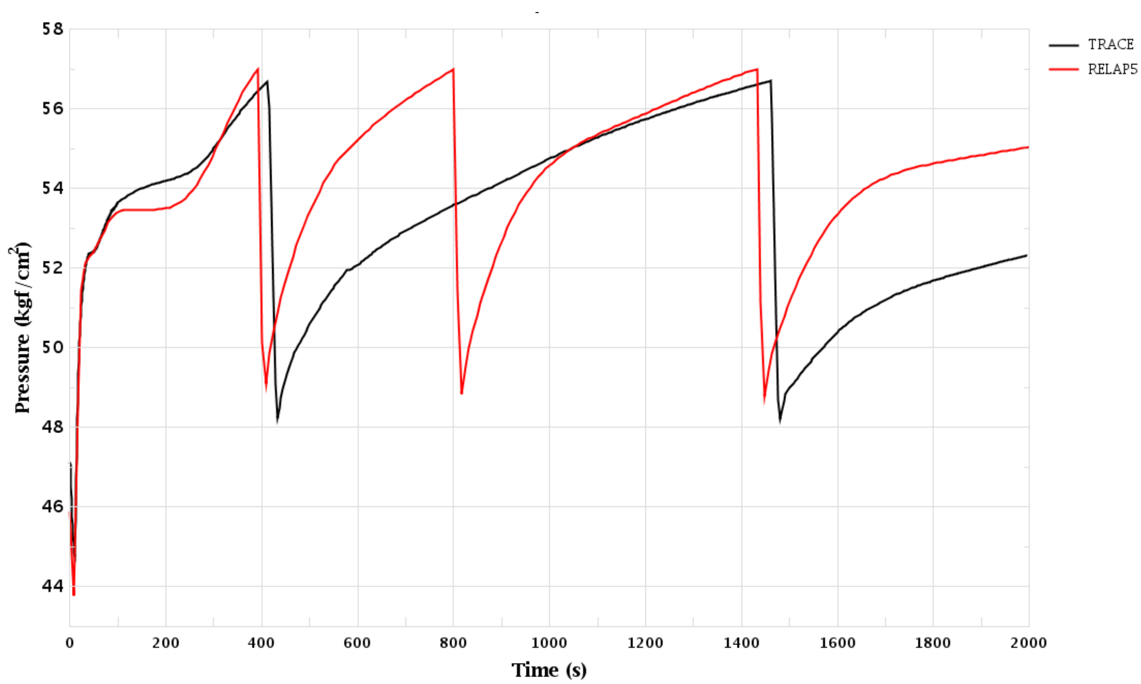

Figure 3-28 MSH Break. SG-3 Pressure

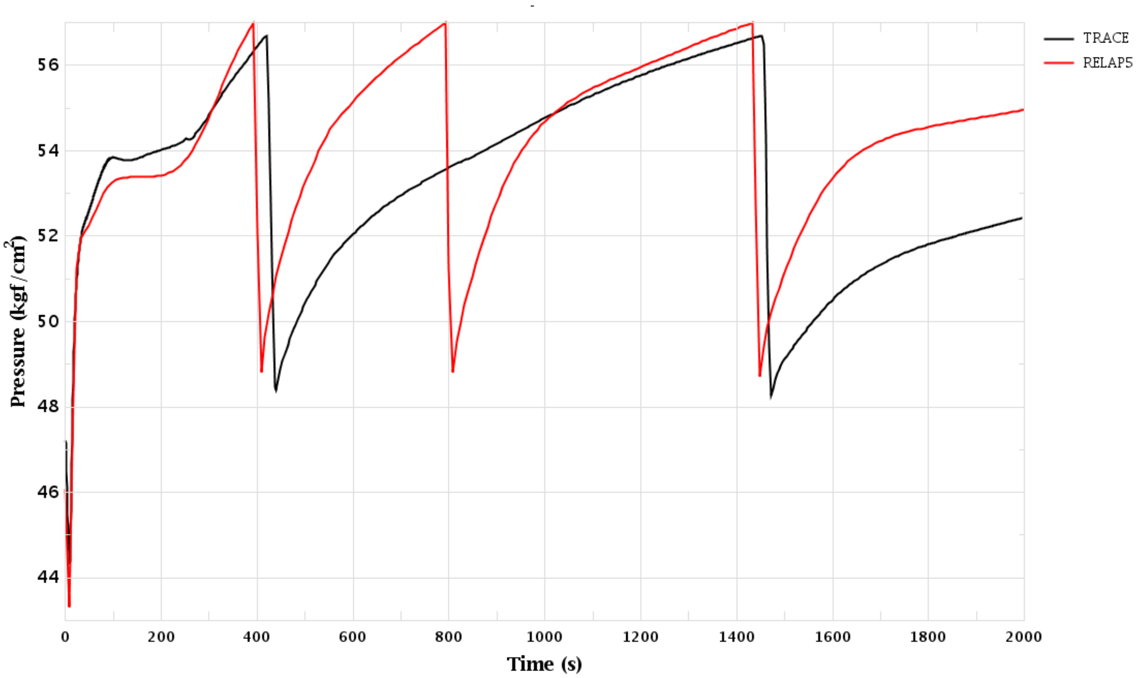

Figure 3-29 MSH Break. SG-4 Pressure

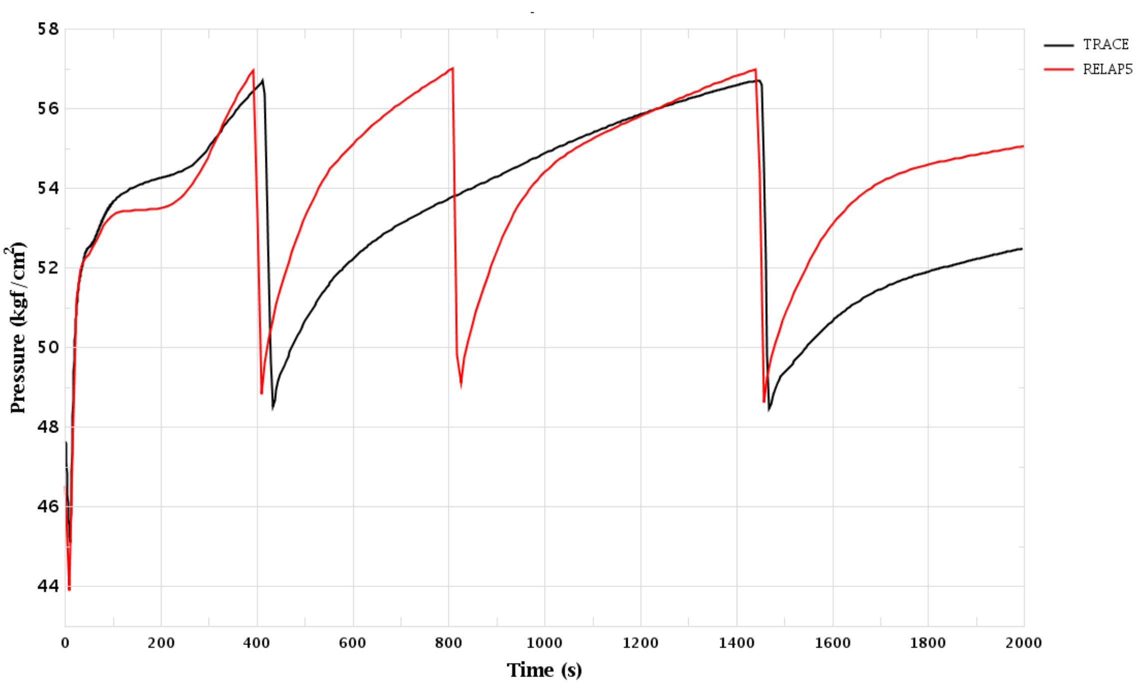

Figure 3-30 MSH Break. SG-5 Pressure

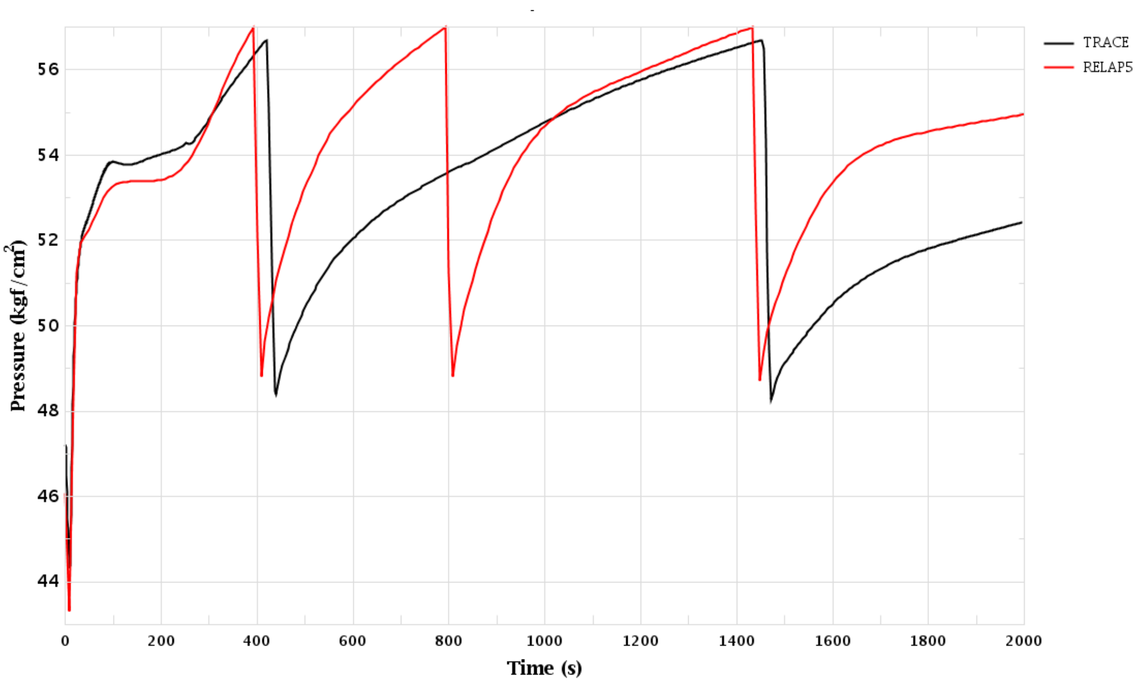

Figure 3-31 MSH Break. SG-6 Pressure

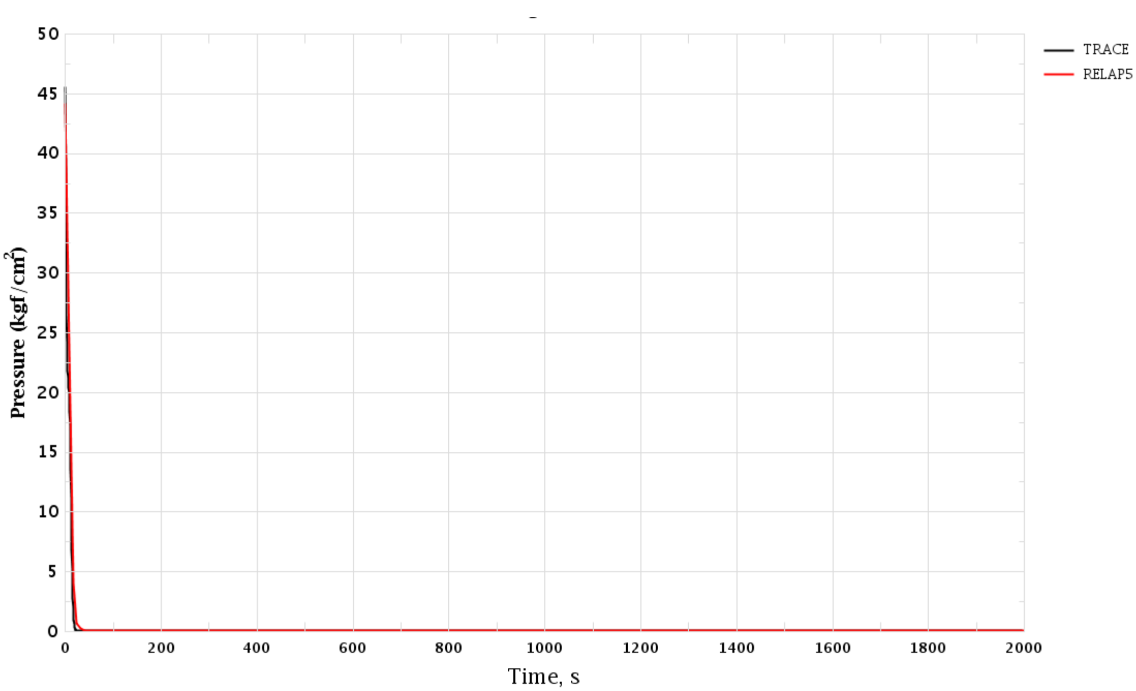

Figure 3-32 MSH Break. MSH-1 Pressure

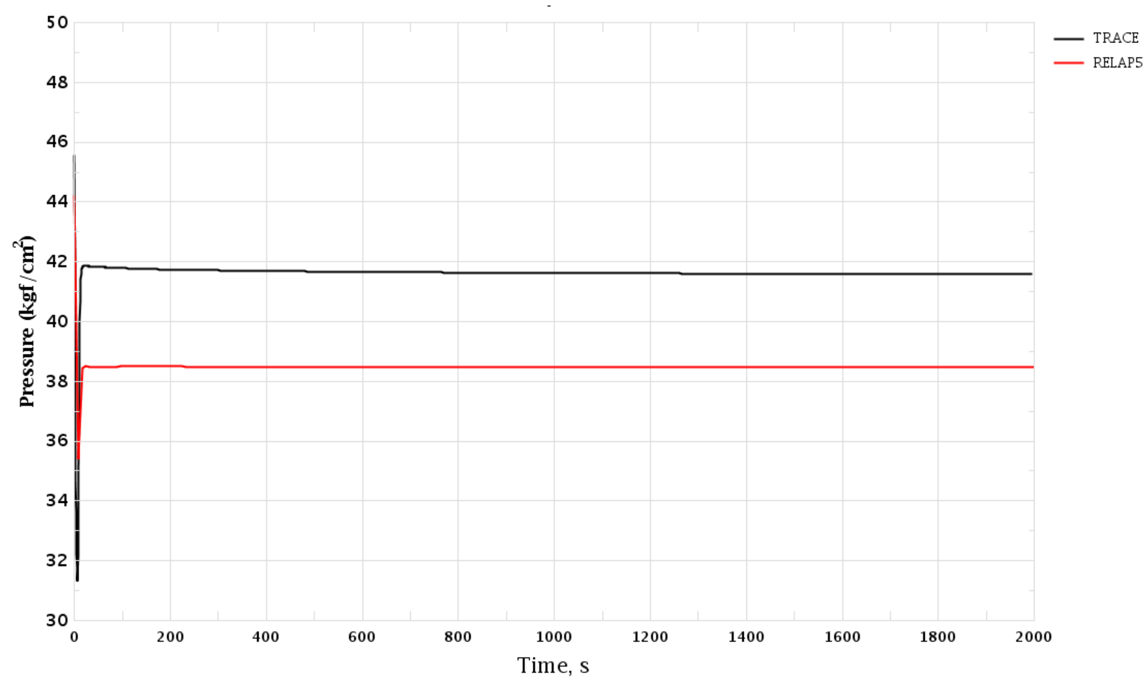

Figure 3-33 MSH Break. MSH-2 Pressure

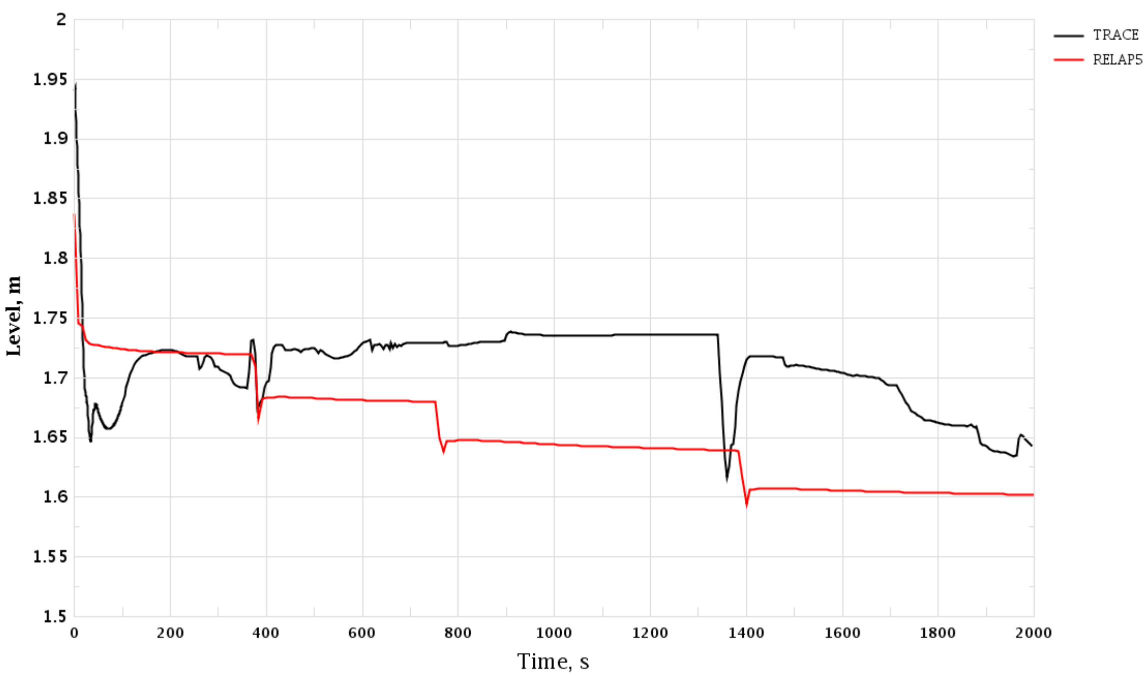

Figure 3-34 MSH Break. SG-1 Level (Wide Range)

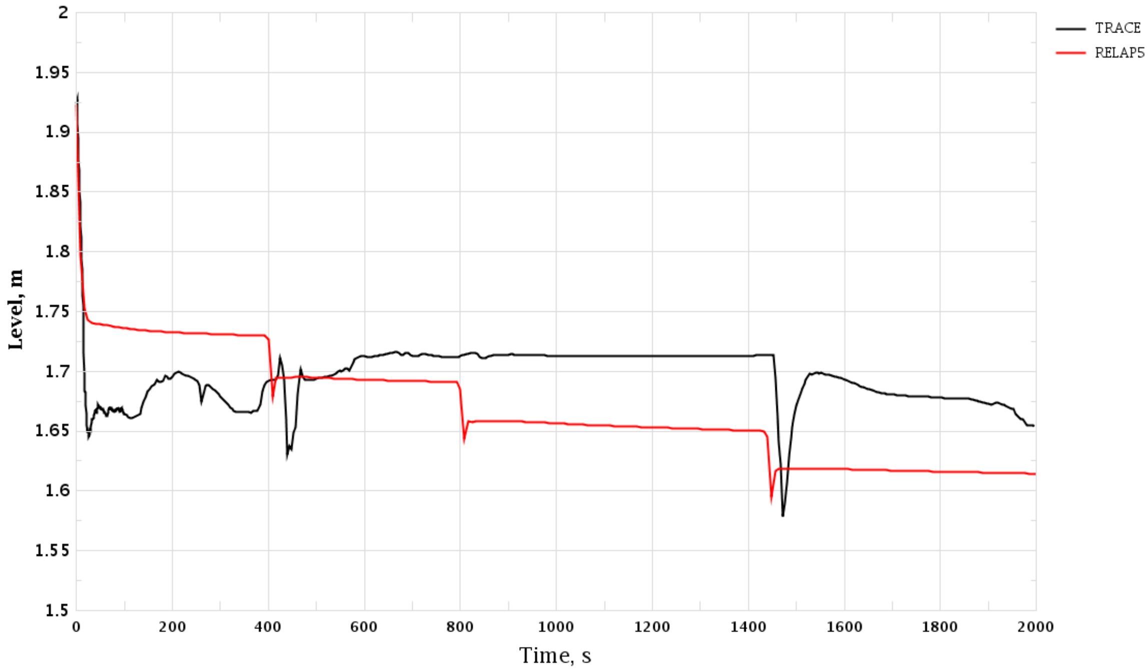

Figure 3-35 MSH Break. SG-2 Level (Wide Range)

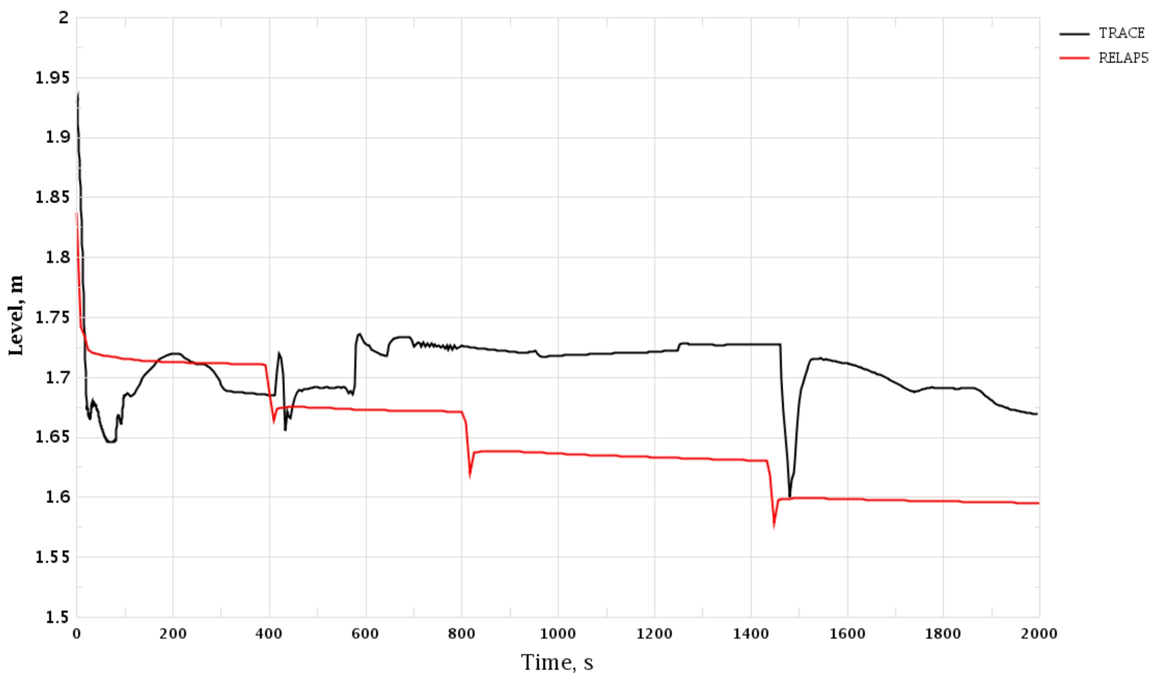

Figure 3-36 MSH Break. SG-3 Level (Wide Range)

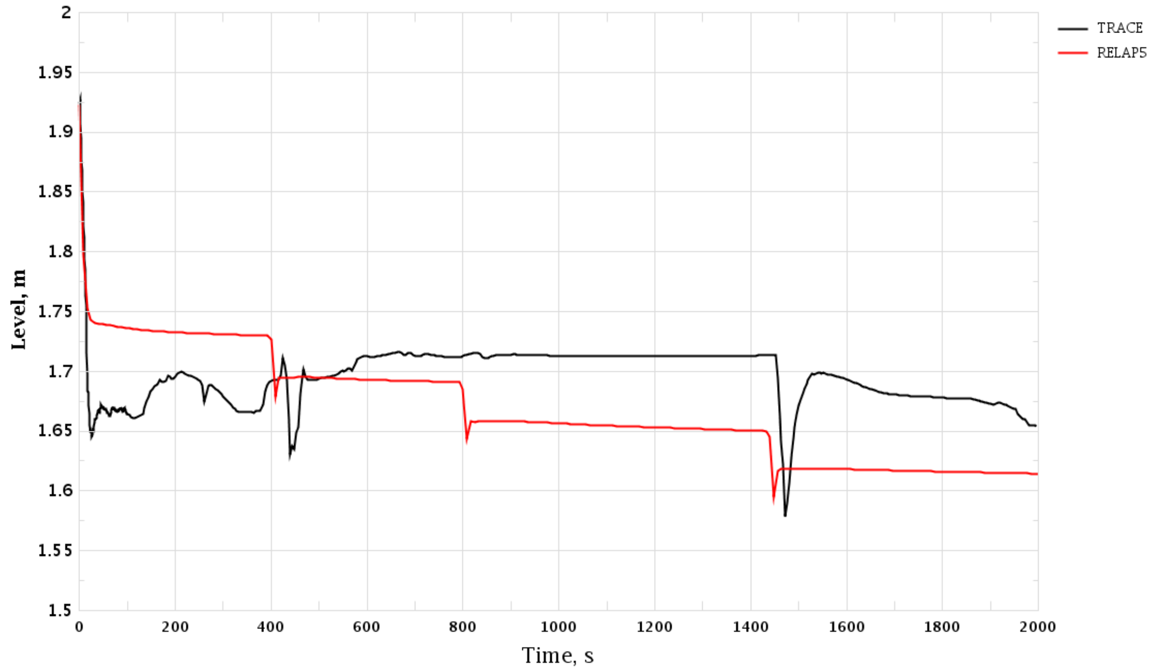

Figure 3-37 MSH Break. SG-4 Level (Wide Range)

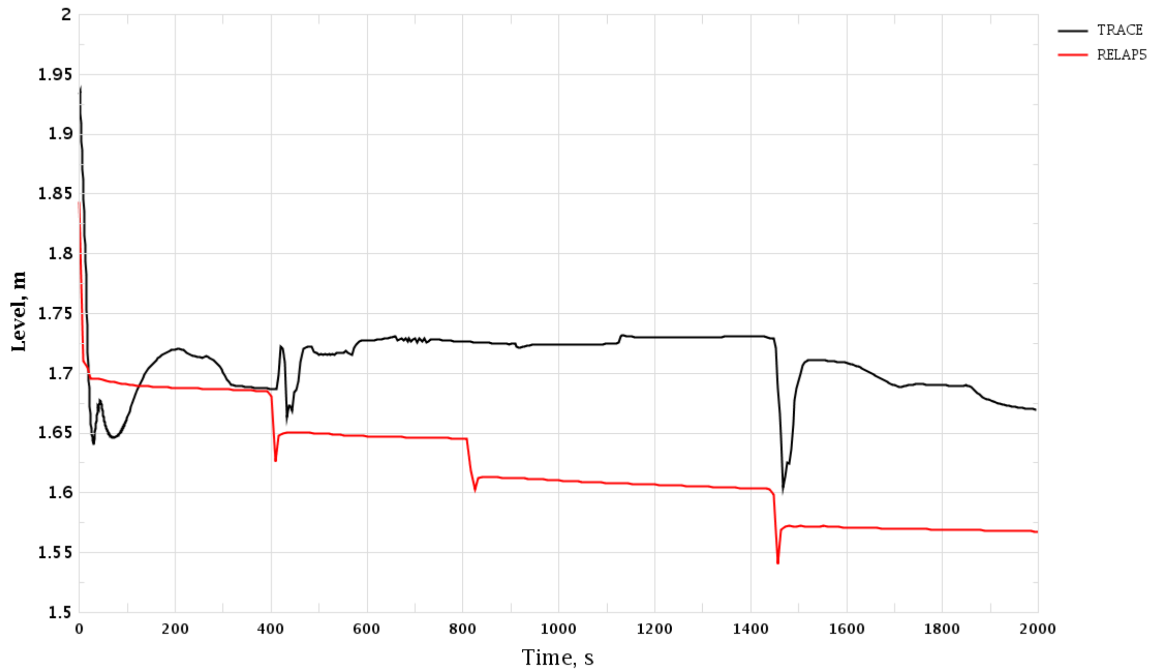

Figure 3-38 MSH Break. SG-5 Level (Wide Range)

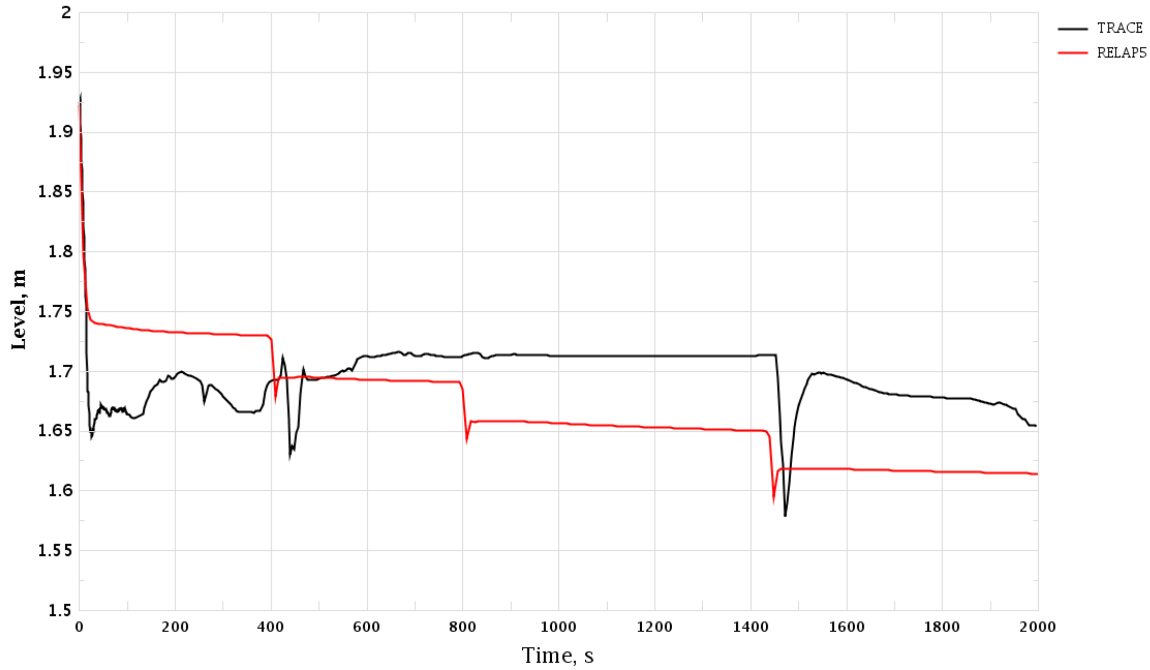

Figure 3-39 MSH Break. SG-6 Level (Wide Range)

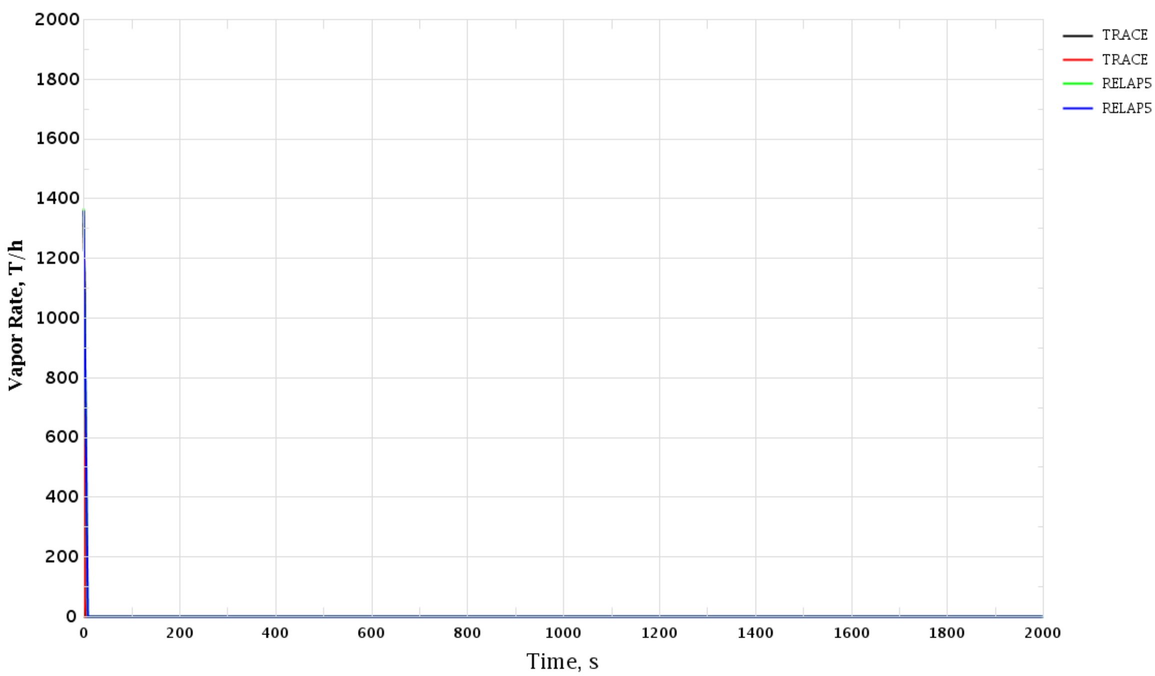

Figure 3-40 MSH Break. Turbines Mass Flow Rate

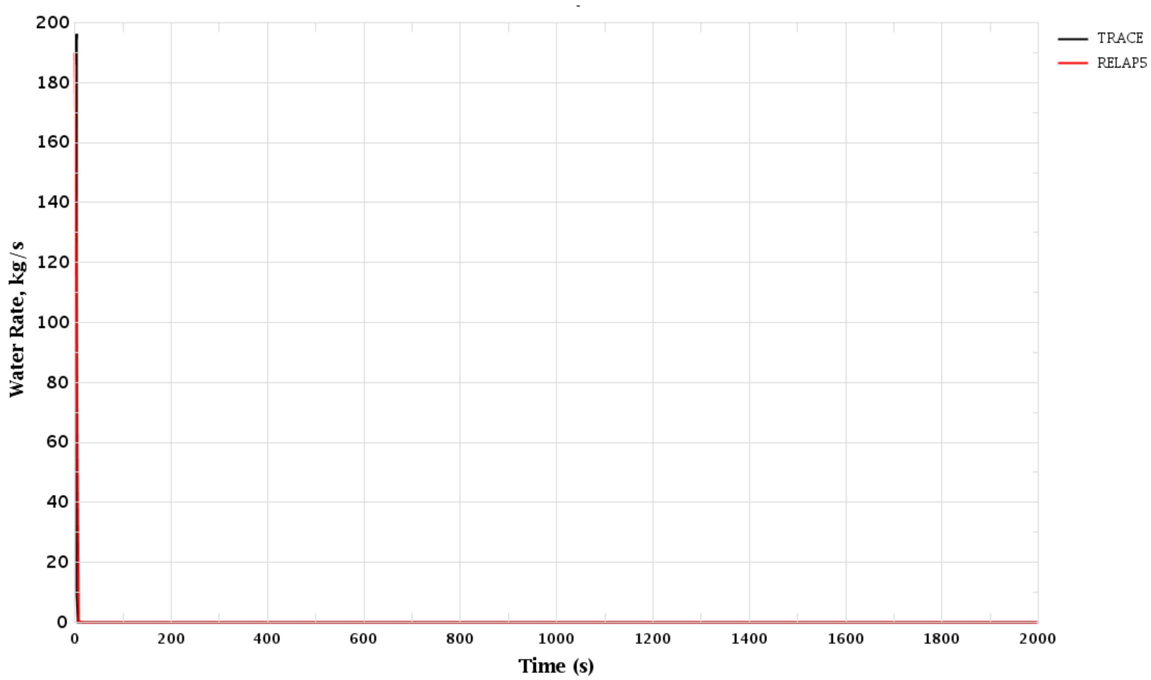

Figure 3-41 MSH Break. MFW Pump No.1 Mass Flow Rate

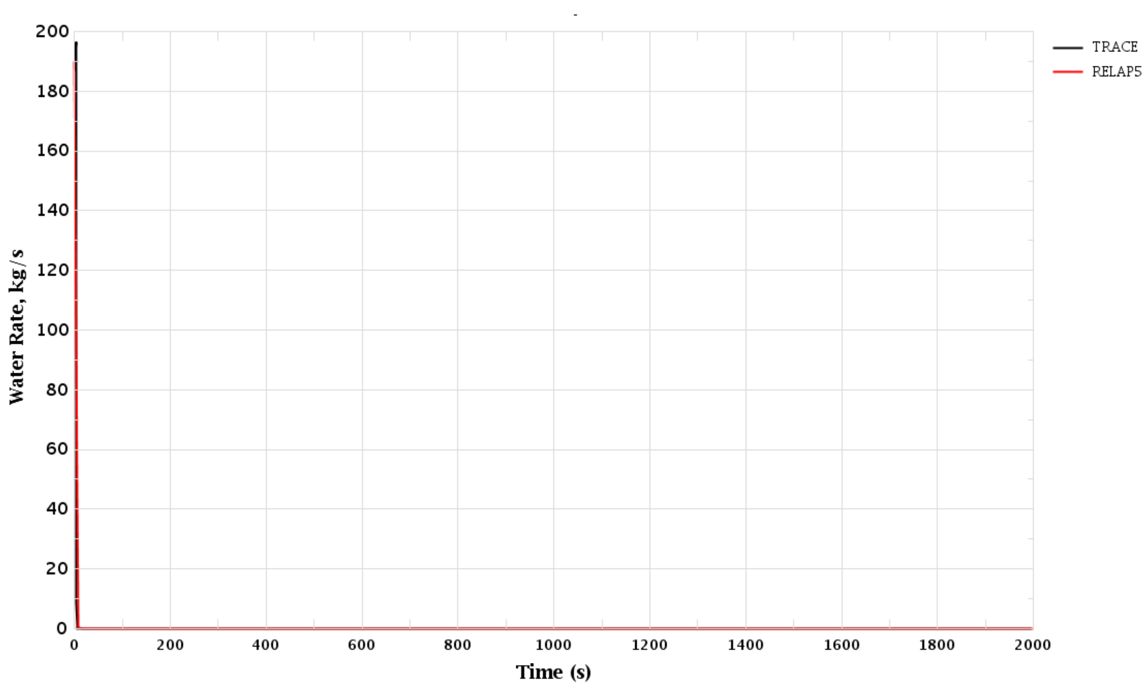

Figure 3-42 MSH Break. MFW Pump No.2 Mass Flow Rate

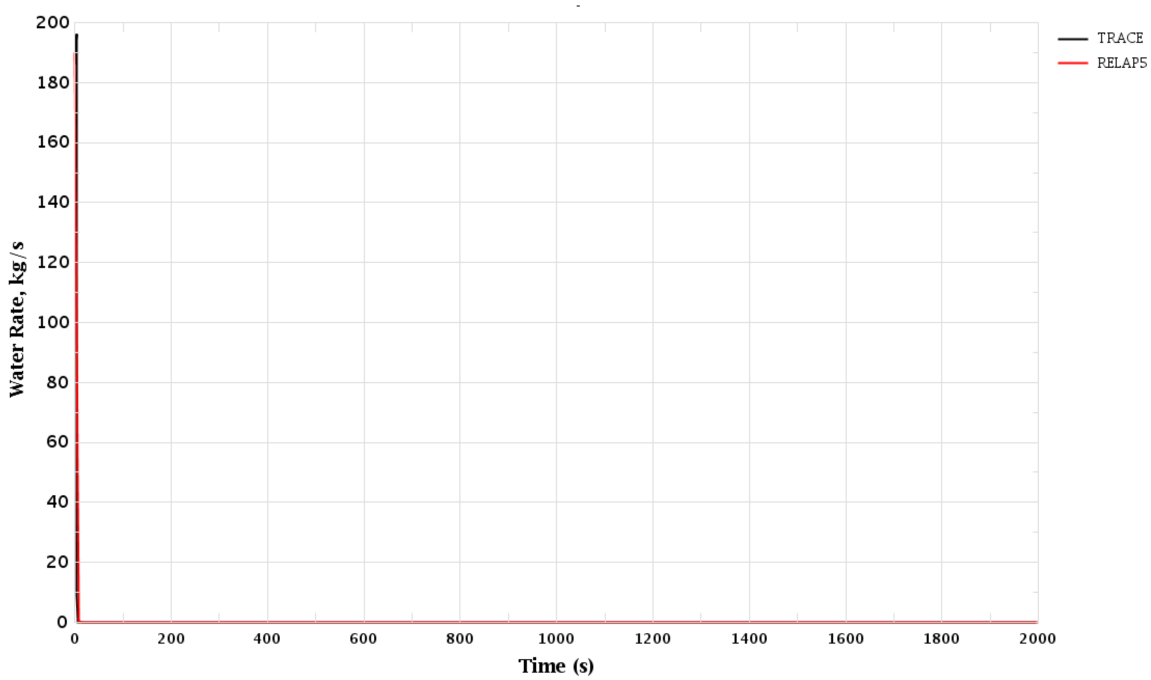

Figure 3-43 MSH Break. MFW Pump No.3 Mass Flow Rate

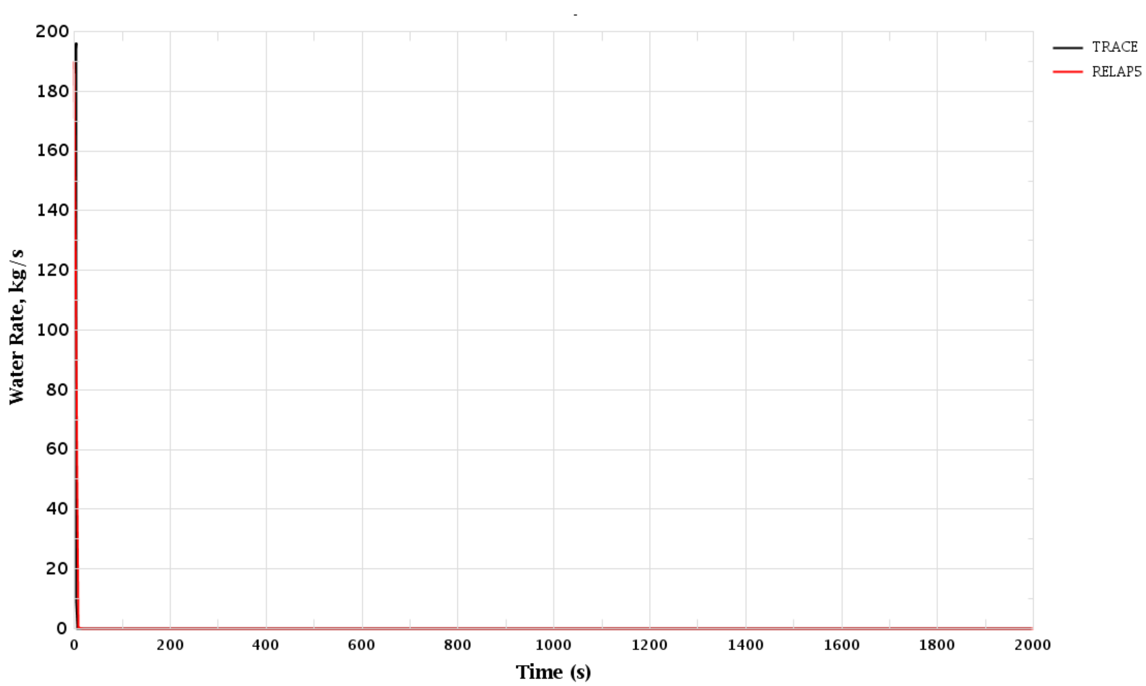

Figure 3-44 MSH Break. MFW Pump No.4 Mass Flow Rate

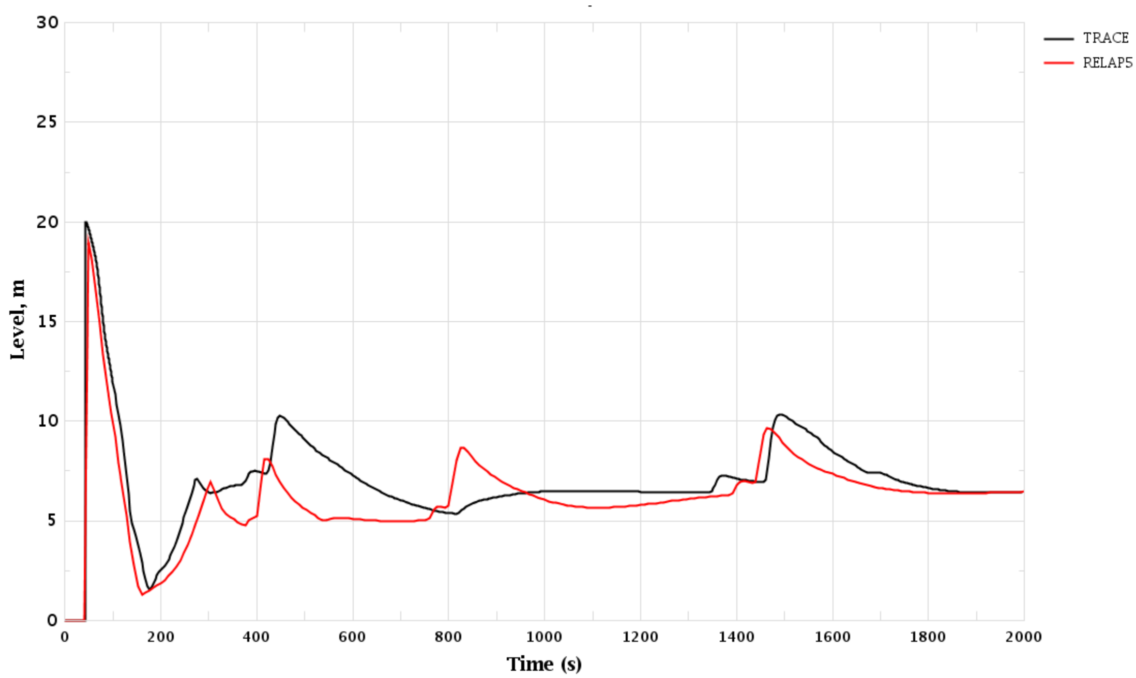

Figure 3-45 MSH Break. HPIS-1 Mass Flow Rate

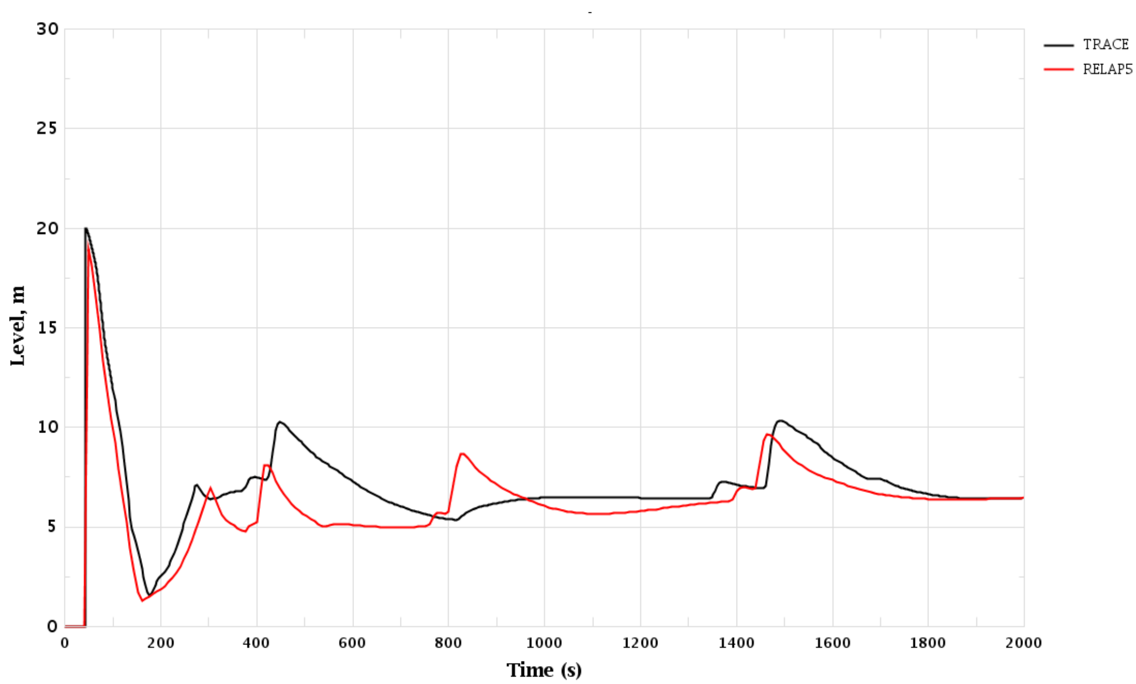

Figure 3-46 MSH Break. HPIS-2 Mass Flow Rate

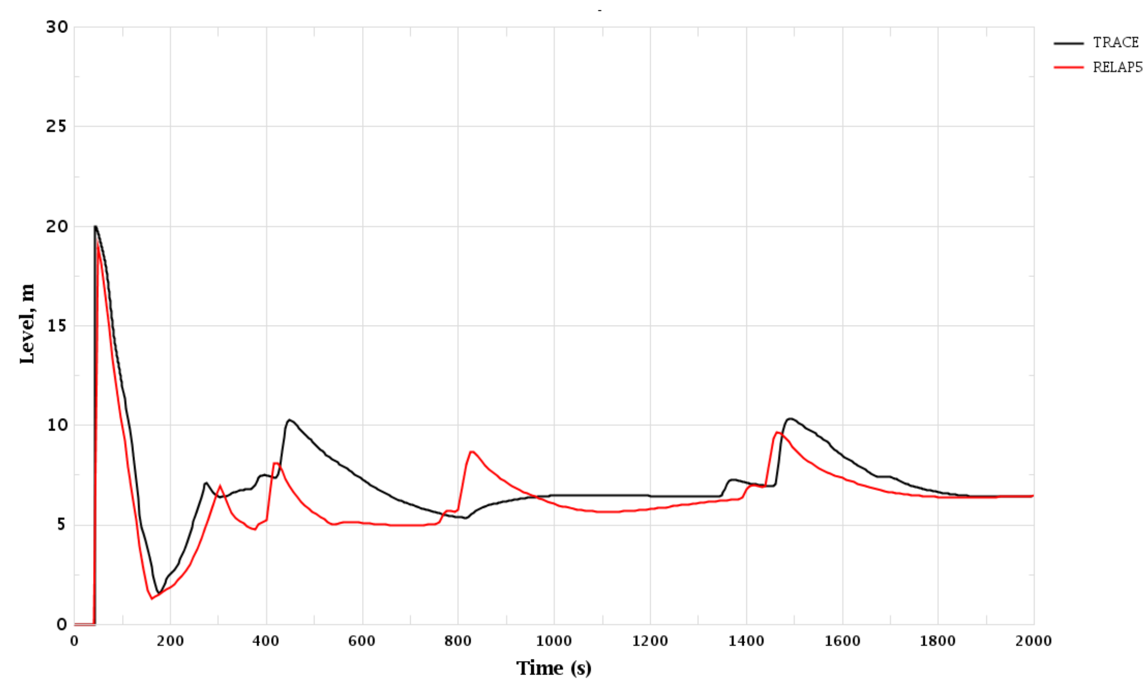

Figure 3-47 MSH Break. HPIS-3 Mass Flow Rate

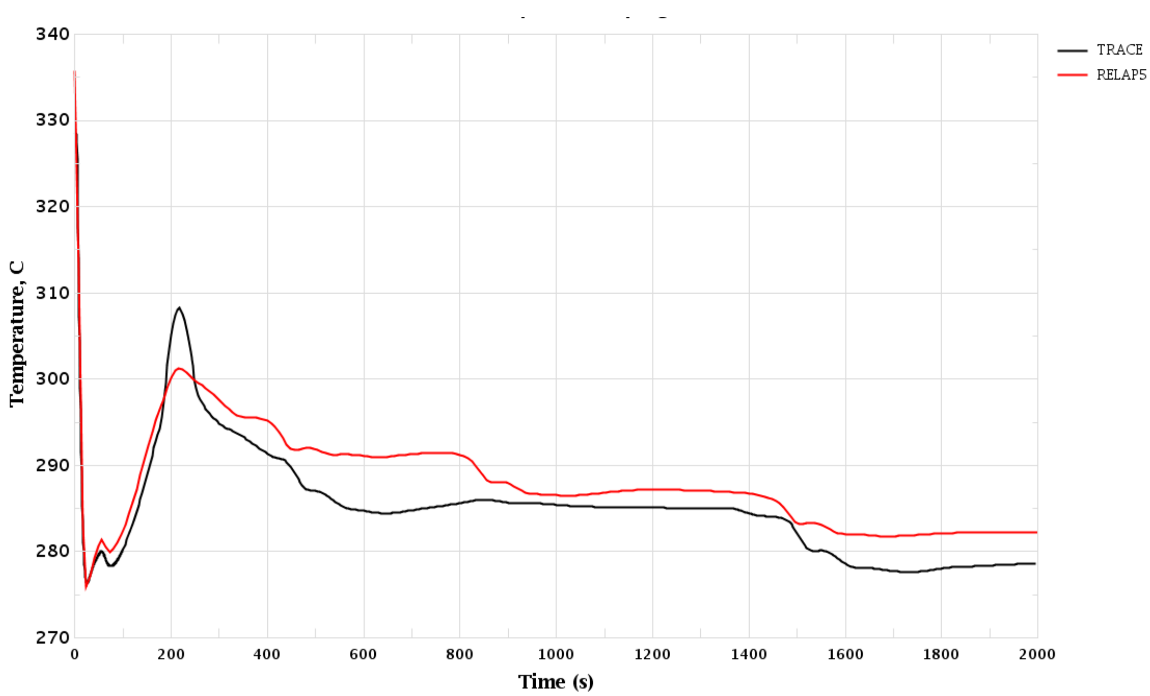

Figure 3-48 MSH Break. Maximal Cladding Temperature

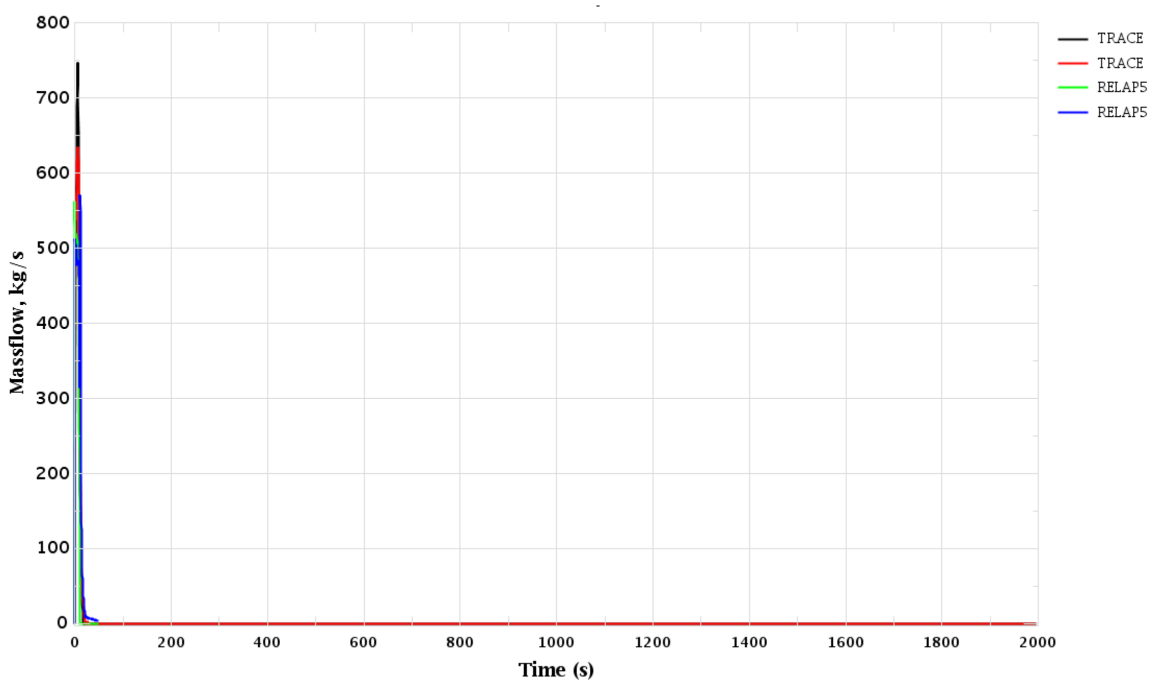

Figure 3-49 MSH Break. Break Mass Flow Rate

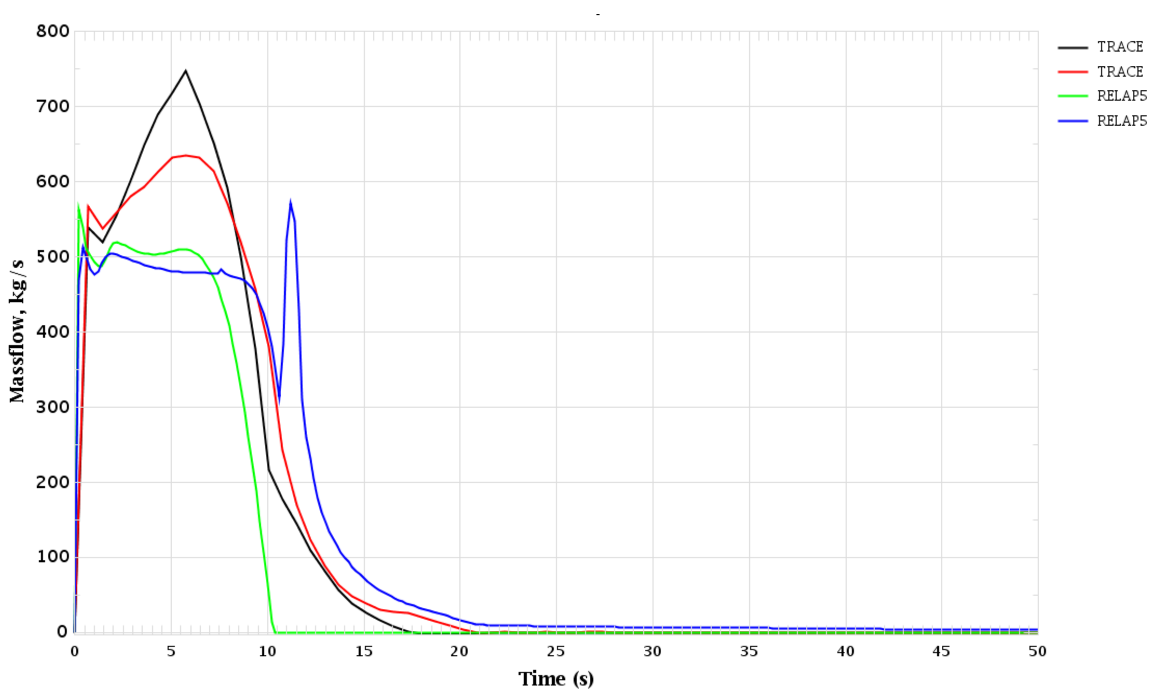

Figure 3-50 MSH Break. Break Mass Flow Rate (Fragment)

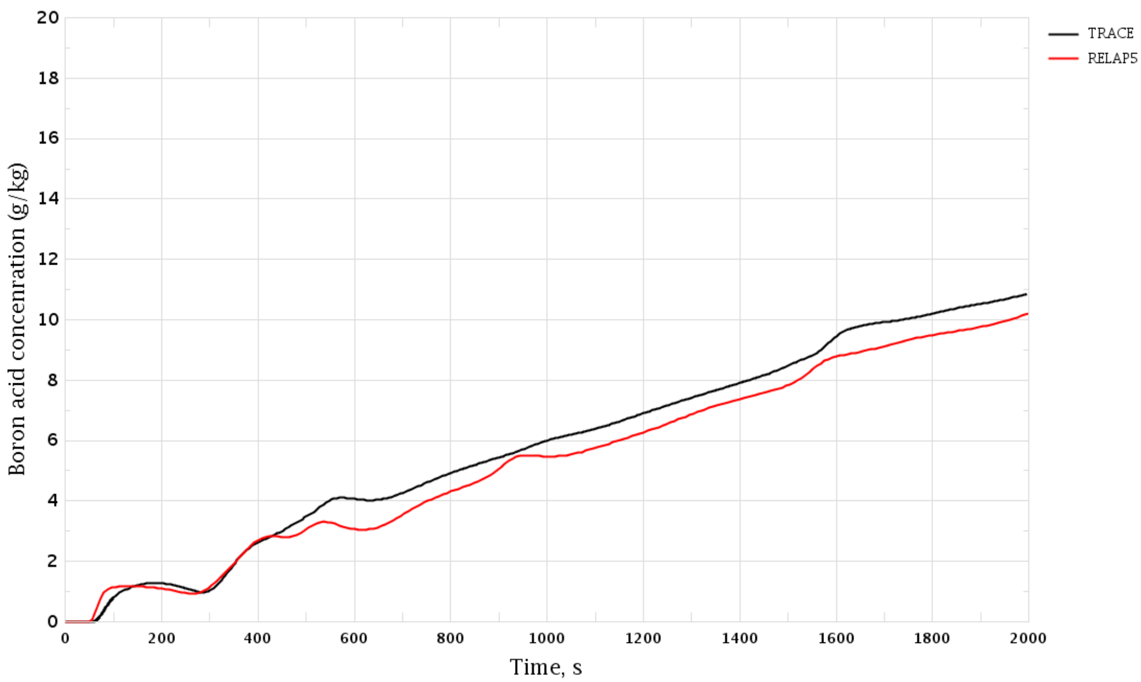

Figure 3-51 MSH Break. Boric Acid Concentration in the Core

3.2 Loss of Turbine Condenser Vacuum

3.2.1 Brief Description of Initiating Event

This initiating event postulates loss of vacuum in the condenser of one of the turbines that leads to a trip of this turbine with closure of its stop valves. According to the expected frequency of occurrence the initiating events is categorized as a postulated transient and leads to a decrease of heat removal by the secondary circuit.

3.2.2 Initial Conditions

The initial conditions selected for transient calculation correspond to those specified in Table 3-1.

3.2.3 Boundary Conditions

Assumptions on the systems availability that are considered in calculation of the transient are specified below.

No operator actions are simulated in the scenario.

Control of reactor power, primary pressure, PRZ and SG levels, and MSH pressure is performed by automatic controllers. Operation of RPL and automatic switching of ARM to "T" mode (MSH pressure maintenance mode) due to increase of MSH pressure is not considered.

Operation of PRZ spray, as well as operation of the make-up and let-down system is also not taken into account.

RCP coast-down is performed according to pump characteristics.

Conservatively, operation of Level 4, Level 3 and Level 2 emergency reactor protection is not accounted.

The 1.0 s delay is postulated for start of control assemblies drop to the reactor core after reactor scram initiation.

Steam pressure in MSH is maintained by operation of steam dump to atmosphere (BRU-A) no.1 controller. BRU-A-2 operation is not taken into account.

3.2.4 Calculation Results

Sequence of events for this transient is presented in Table 3-3.

Table 3-3 Sequence of Events for Loss of Turbine Condenser Vacuum Transient

TRACE Time, s	RELAP5 Time, s	Event	Description
0.0	0.0	Closure of turbine no.1 stop valves	Due to loss of vacuum in a condenser of turbine no.1
1.1	1.0	Maximal cladding temperature is reached: 333.0 °C (TRACE) and 339 °C (RELAP5)	
9.0	10.0	Switching off of PRZ electric heaters' group no.1	PRZ pressure increase
10.5 – 99.5	10.5 – 99.5	Periodic signals to withdraw control assemblies from ARM	ARM operation in "N" (neutron power maintenance) mode
30.0	30.0	Actuation of BRU-A automatic control mode	MSH semi-header pressure increase up to 52.0 kgf/cm^2
44.0	43.0	Complete opening of BRU-A valve	Steam dump to atmosphere
100.0-1200.0	100.0-1200.0	ARM operation in "N" mode	
1230.0	1240.0	MFW pumps trip	Due to decrease of secondary circuit deaerator down to 0.5 m
1220.0	1240.0	Start of EFW pump and start of water supply to SG	Cutoff valve at charging line is opened at level decrease in 2 out of 6 SGs for 0.4 m from the nominal value
1216.0	1232.0	Reactor scram signal	Level decrease in 2 SGs by 0.45 m from the nominal value and correspondent RCPs are in operation
1216.0	1232.0	RCPs trip	SG level decrease for 400 mm from the nominal value (holding for 20 s)
1217.0	1233.0	Start of control assemblies drop by scram signal	
1416.0	1433.0	End of RCPs coast-down	Establishment of natural circulation in RCS loops
1500.0	1440.0	Maximal RCS pressure (136 kgf/cm^2)	
3600.0	3600.0	End of calculation	Stabilization of main parameters

In the initial period of the transient after closure of stop valves of turbine no.1 (Figure 3-91), the steam flow to the turbine that remains in operation is not sufficient to remove heat transferred from primary circuit to SGs, and the secondary pressure reaches the setpoints of BRU-A actuation at 30 s of the calculation time (Figure 3-92). Increase in secondary circuit pressure causes increase in the primary circuit parameters: pressure (Figure 3-53) and temperature (Figure 3-55 – Figure 3-66).

The calculation scenario does not take into account automatic ARM switching to "T" (secondary circuit pressure maintenance) mode at the increase MSH pressure, so ARM controller continues to maintain a specified power setpoint till the moment of scram actuation.

Uncompensated loss of the secondary circuit coolant through BRU-A results in a decrease of secondary circuit deaerators level that causes trip of MFW pumps (Figure 3-93 – Figure 3-96). Consequential level decrease for 0.45 m from the nominal in 2 SGs with correspondent RCPs in operation leads to actuation of reactor scram at 1216.0/1232.0 s. Due to a drop of all control assemblies into the reactor core the reactor power decreases to the decay value. At the same time excessive (compared to decay heat) steam flow to the operating turbine and steam loss via BRU-As causes decrease of secondary circuit pressure, that in turn results in a primary pressure decrease. BRU-A controllers remain in MSH pressure maintenance mode throughout the calculation (Figure 3-92).

Upon SG level decrease for 400 mm from the nominal value (holding for 20 s), all operating RCPs are tripped (Figure 3-67 – Figure 3-72).

Decrease of SG level (Figure 3-85 – Figure 3-90) causes actuation of EFW pumps (Figure 3-97) and their operation restores SG levels.

The maximal primary circuit pressure reached in calculation is 136.0 kgf/cm^2 (Figure 3-53).

The plots of the main parameters of calculation are presented below on Figure 3-52 – Figure 3-98.

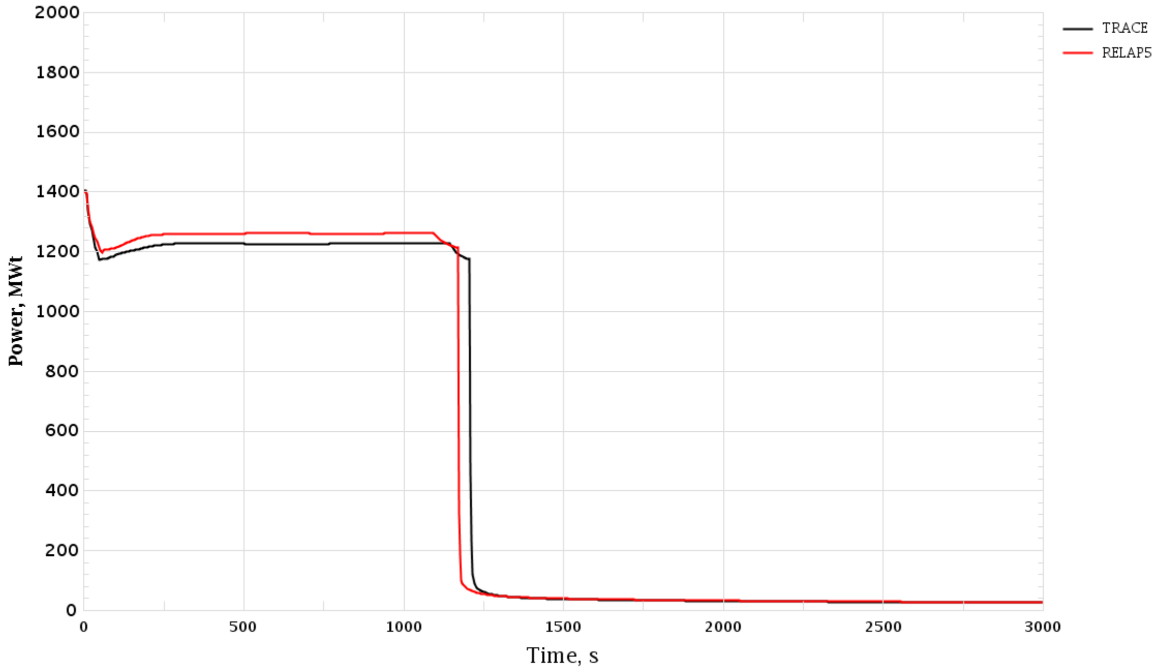

Figure 3-52 Loss of Turbine Condenser Vacuum. Core Thermal Power

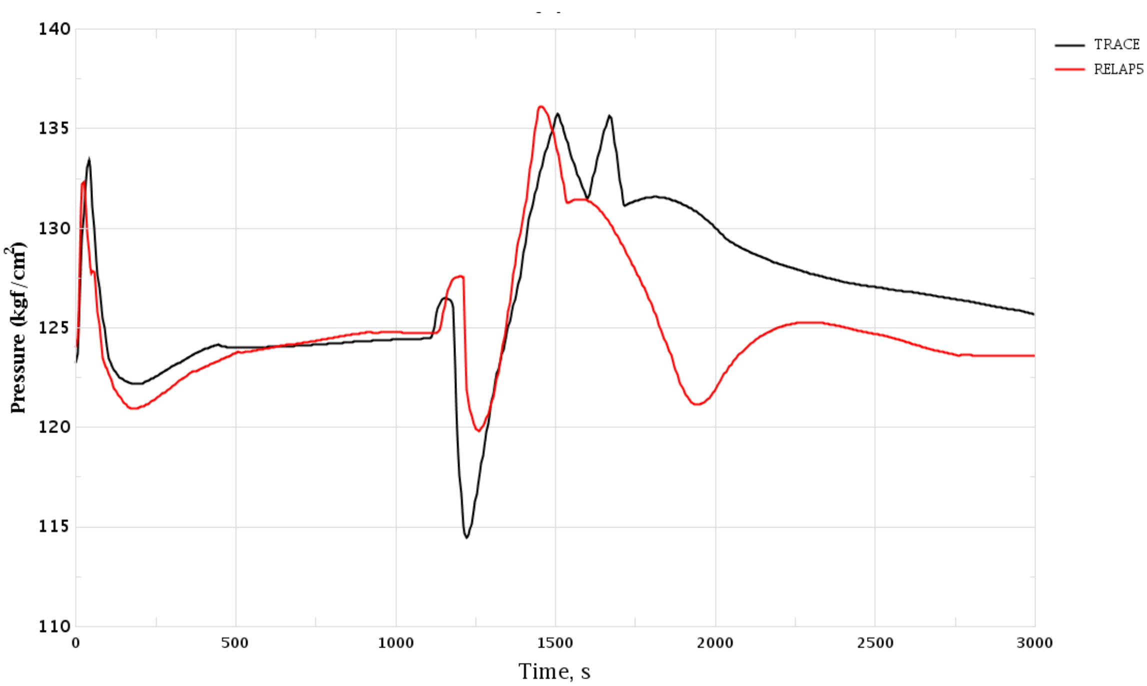

Figure 3-53 Loss of Turbine Condenser Vacuum. RCS Pressure

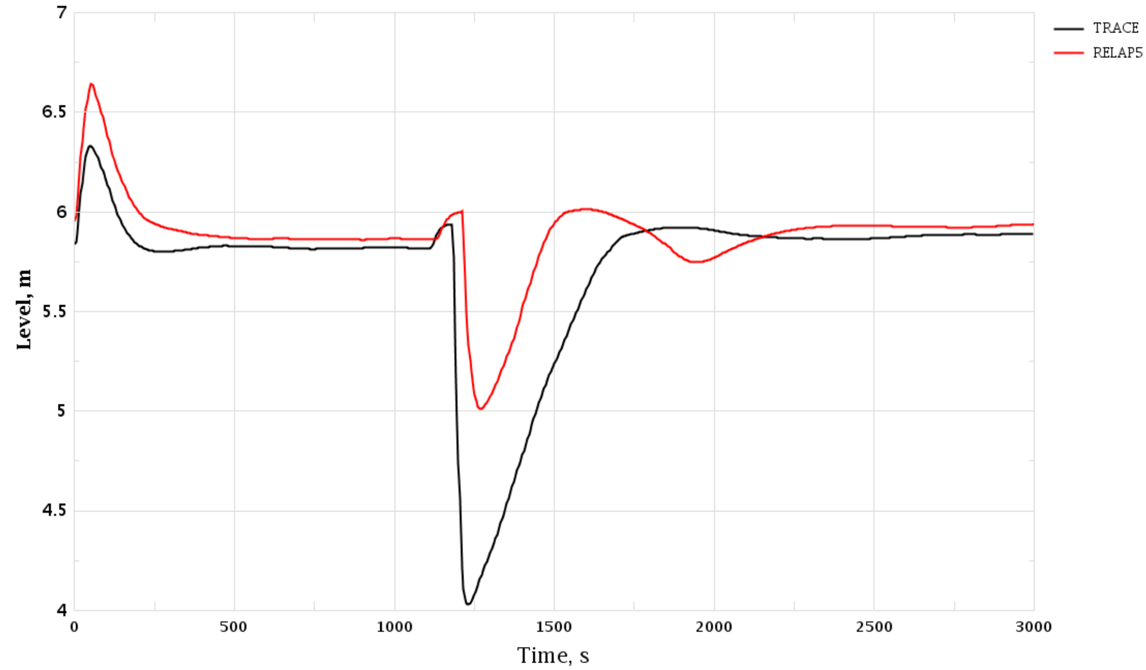

Figure 3-54 Loss of Turbine Condenser Vacuum. Pressurizer Level

Figure 3-55 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 1

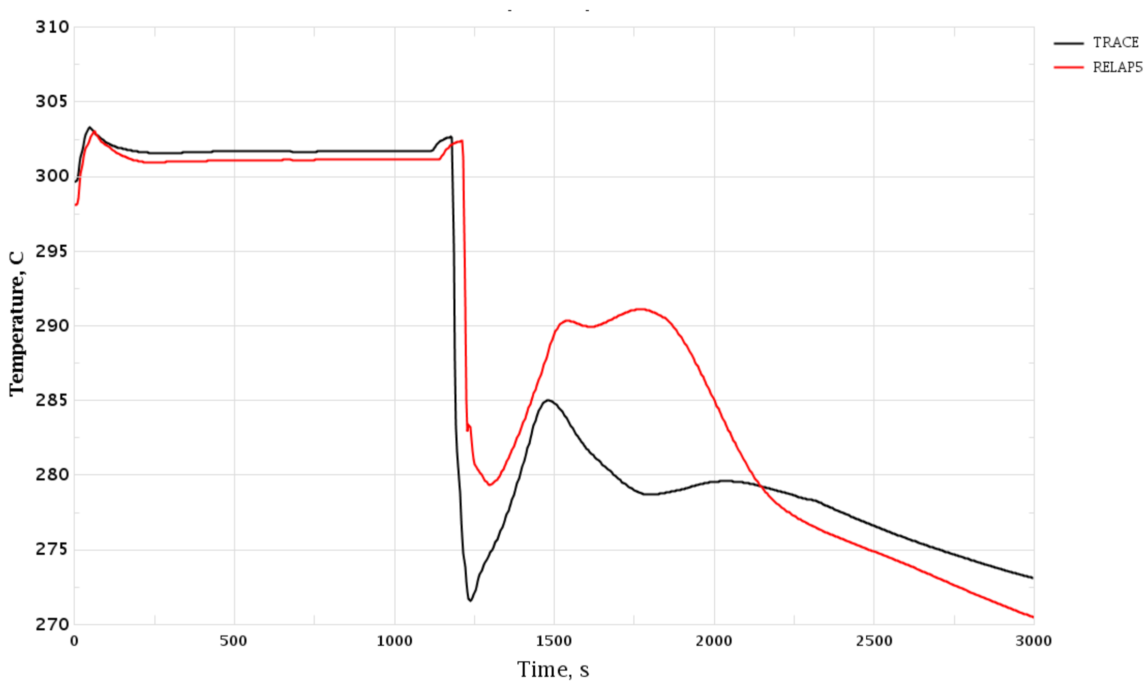

Figure 3-56 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 2

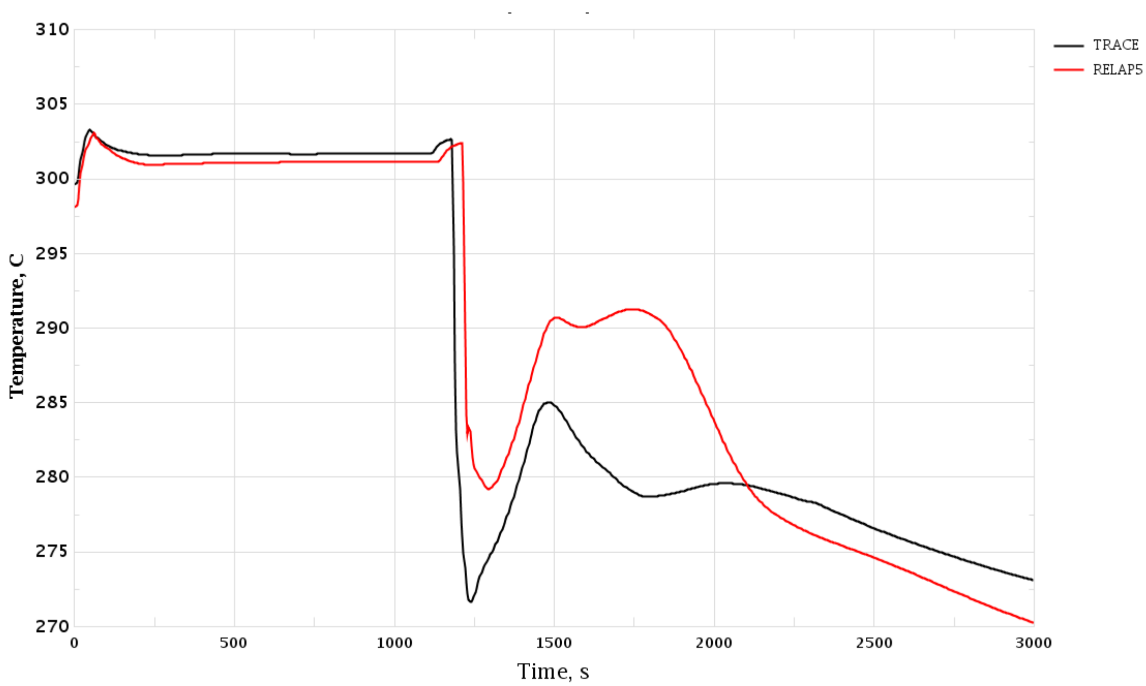

Figure 3-57 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 3

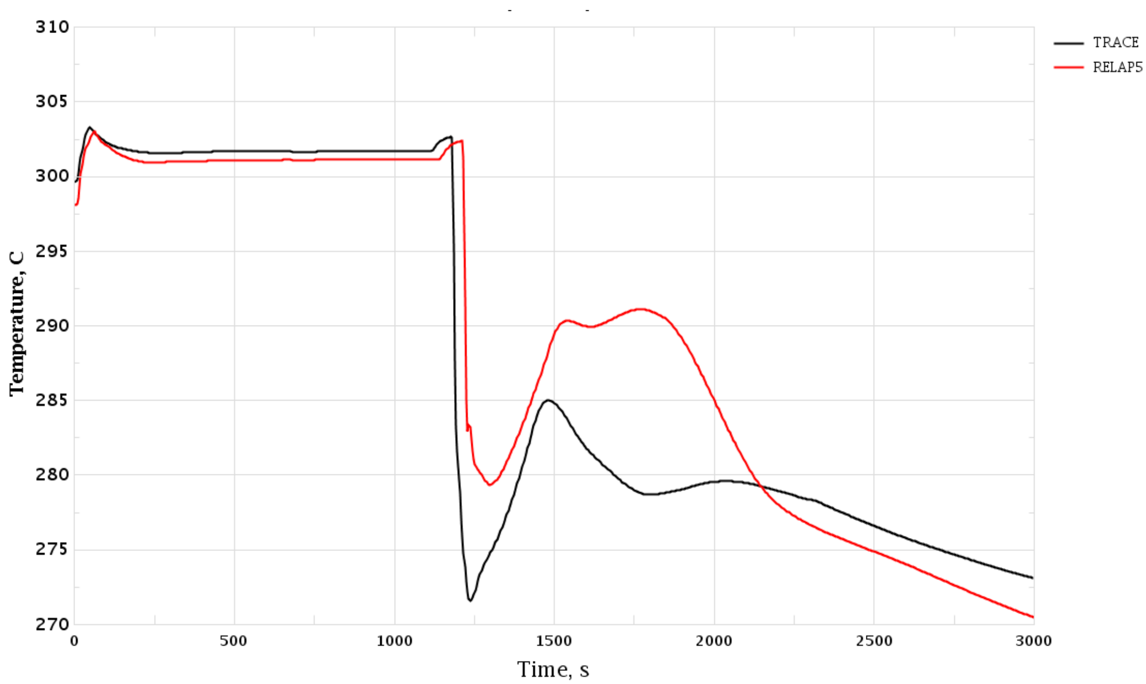

Figure 3-58 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 4

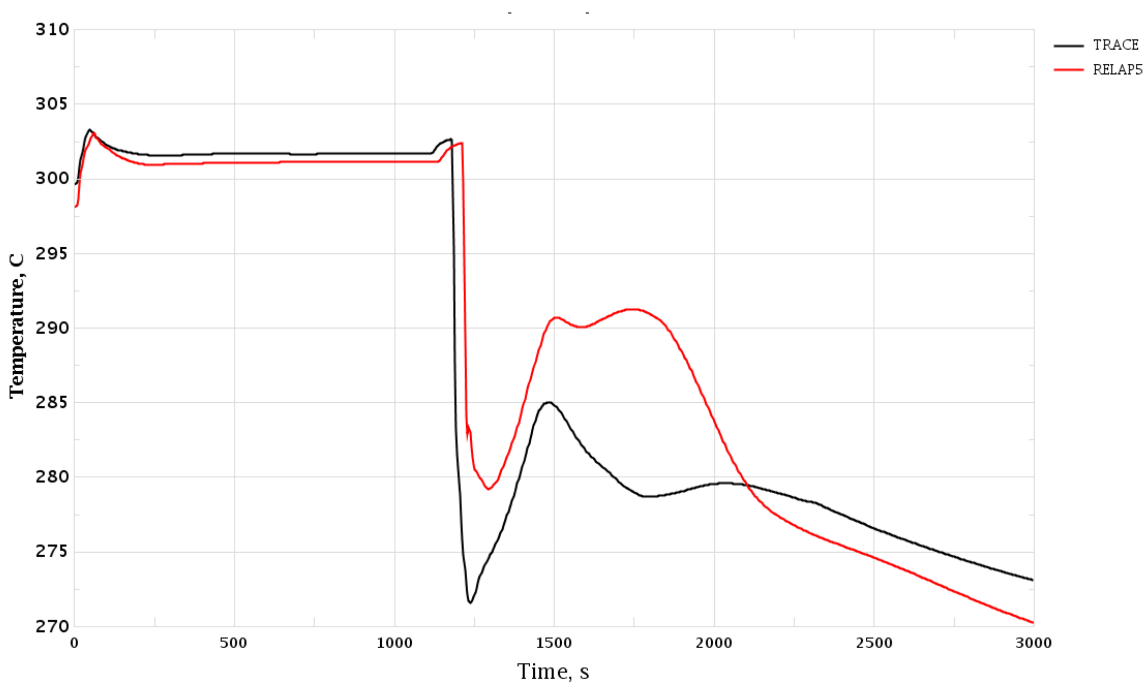

Figure 3-59 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 5

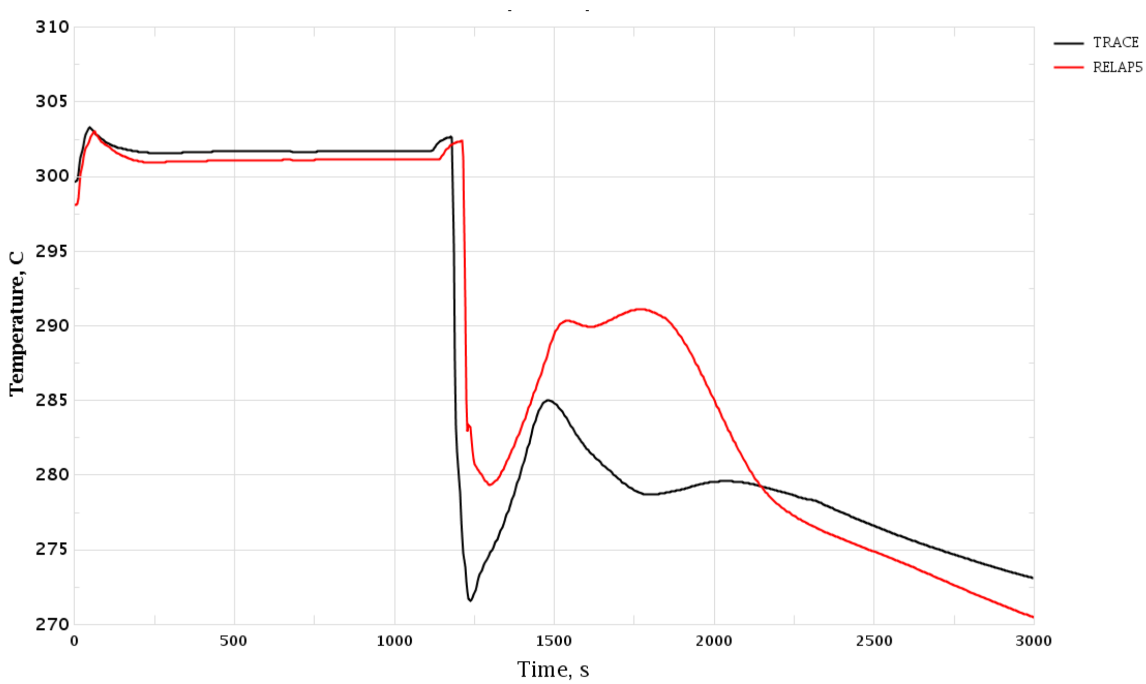

Figure 3-60 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 6

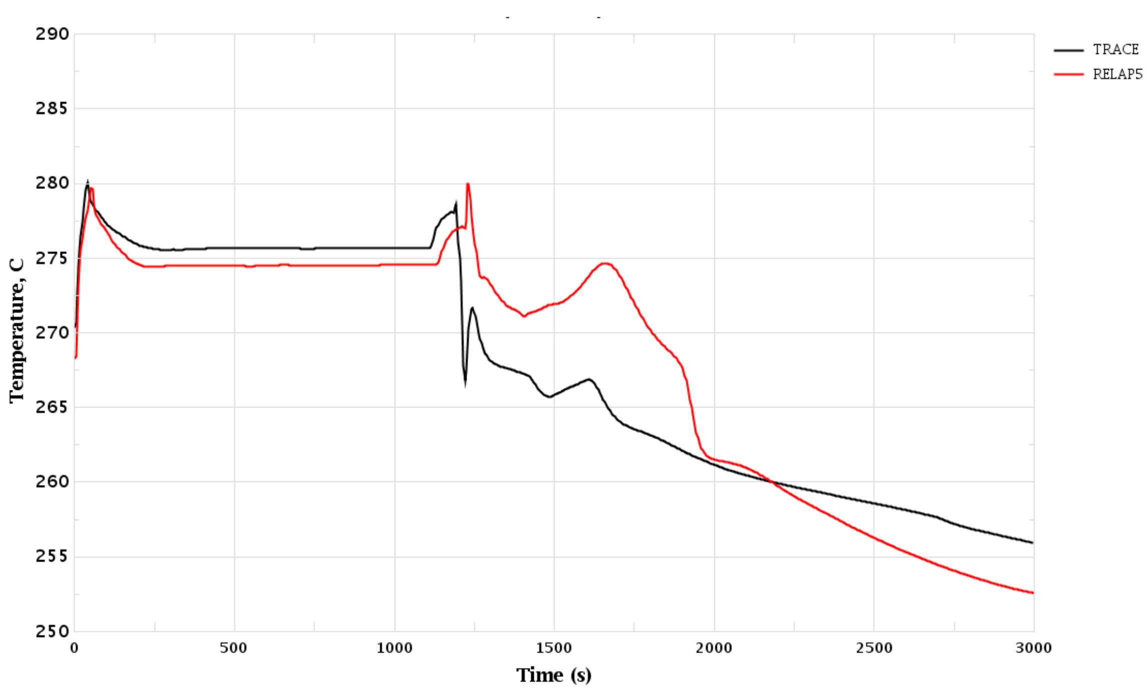

Figure 3-61 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 1

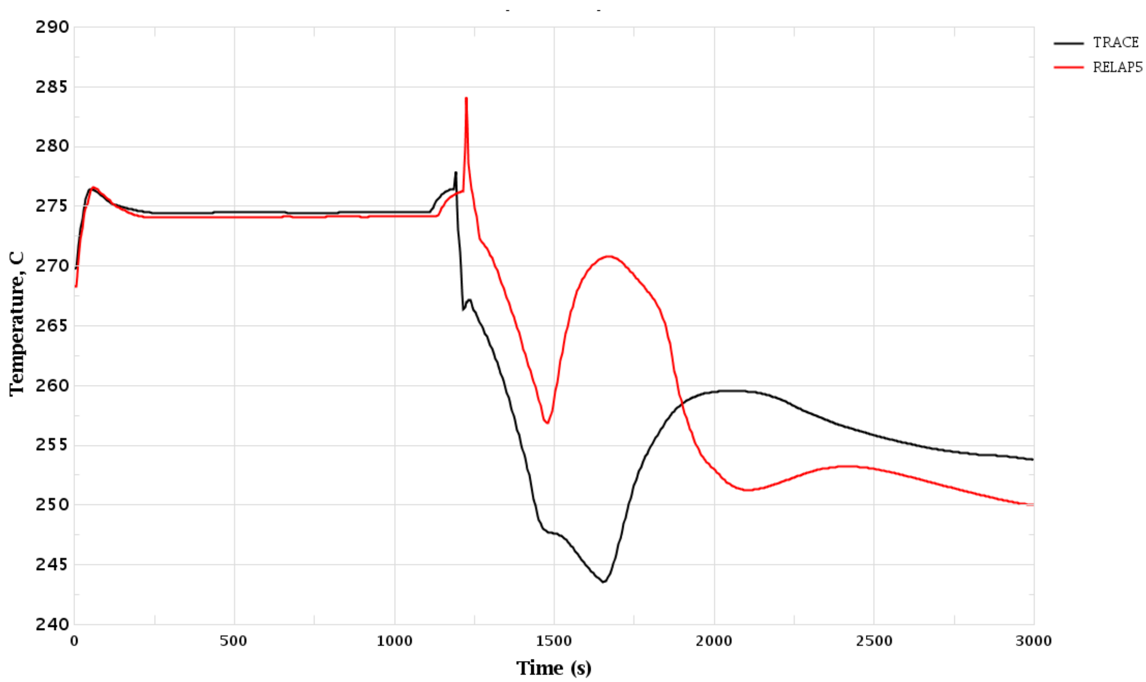

Figure 3-62 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 2

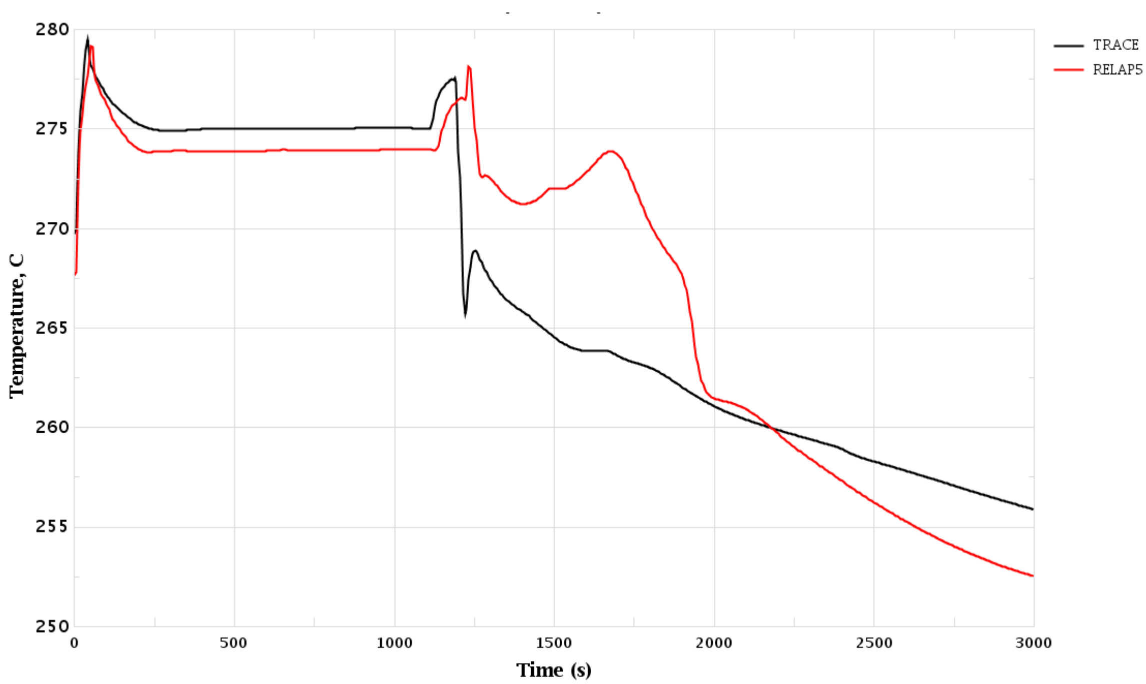

Figure 3-63 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 3

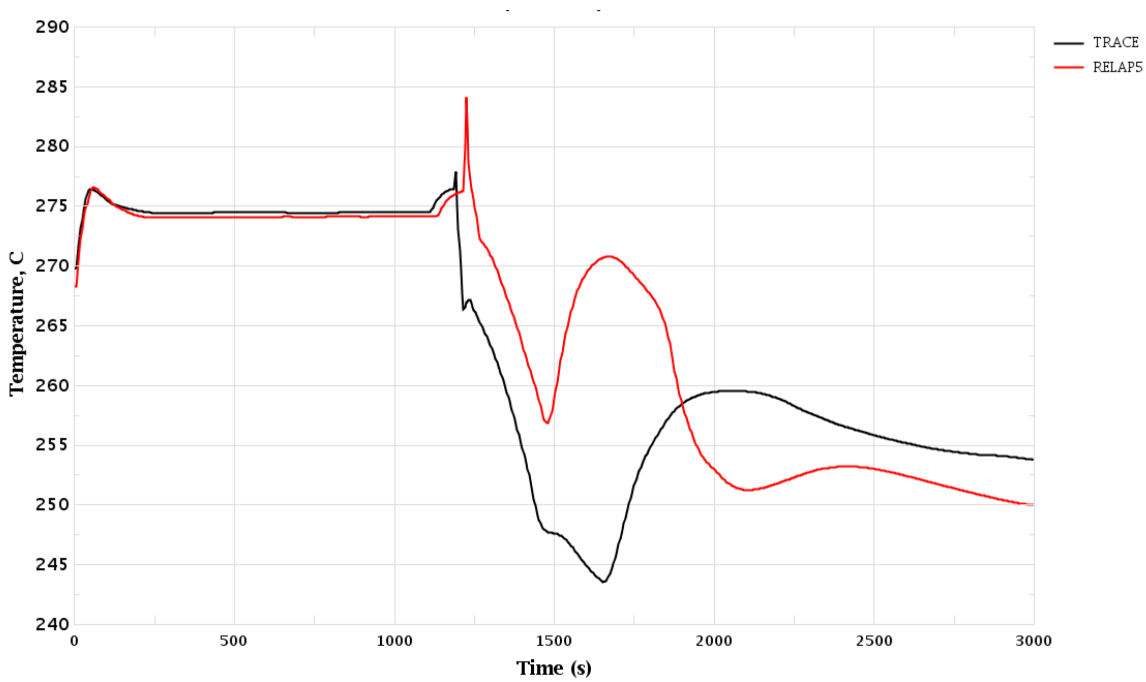

Figure 3-64 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 4

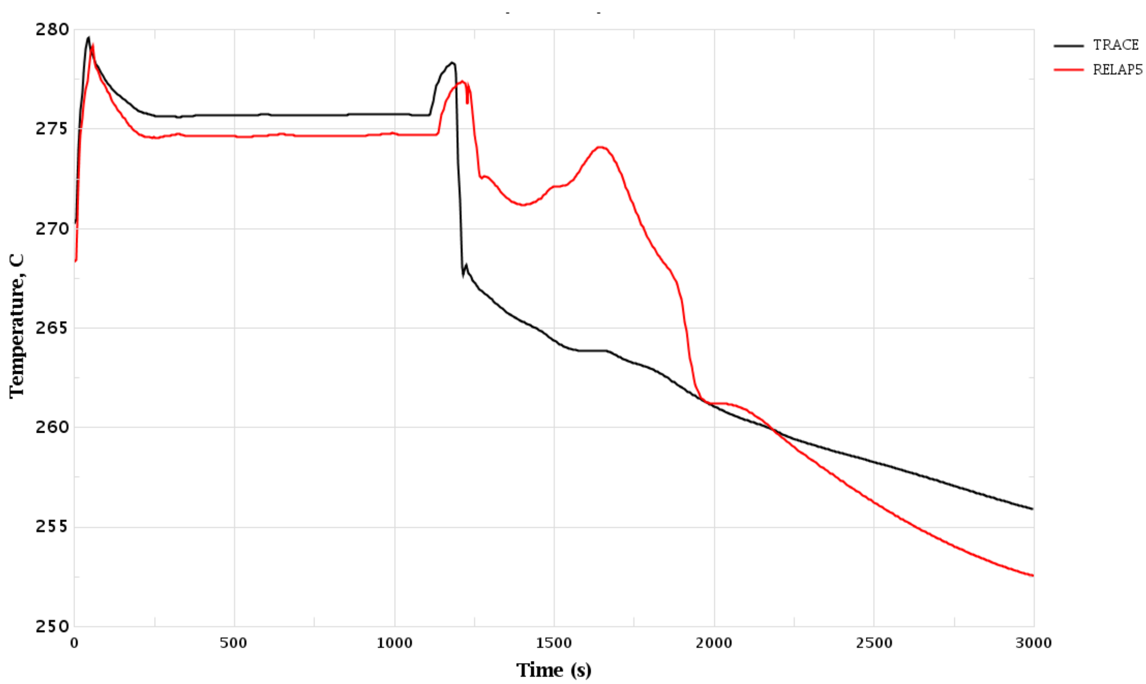

Figure 3-65 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 5

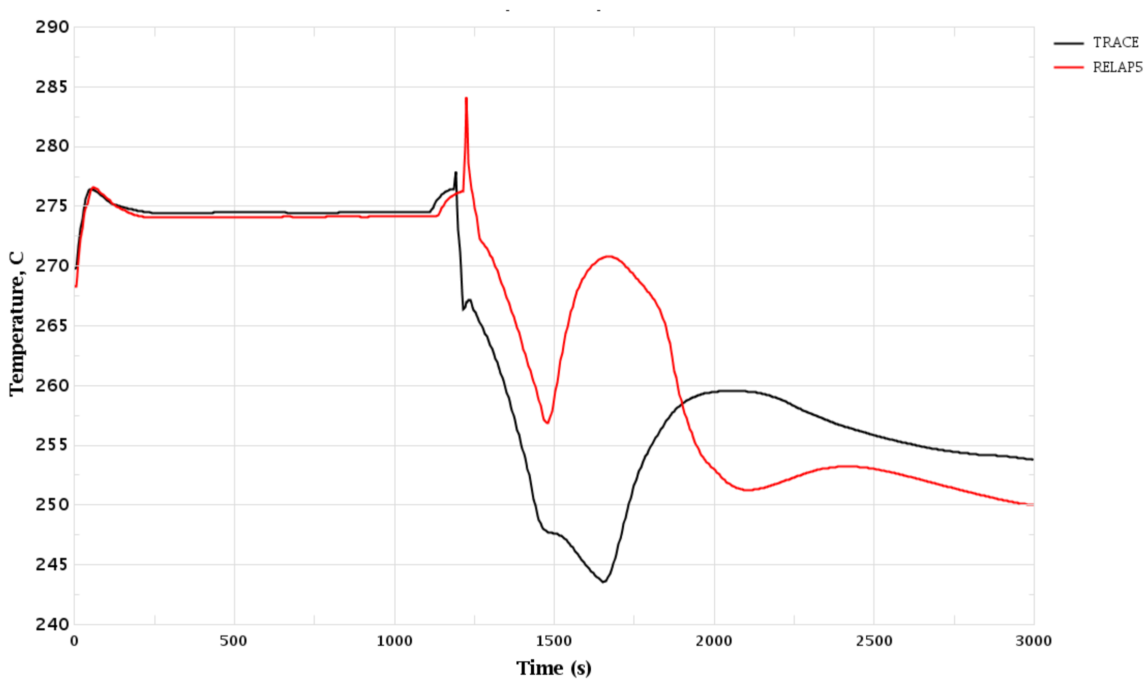

Figure 3-66 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 6

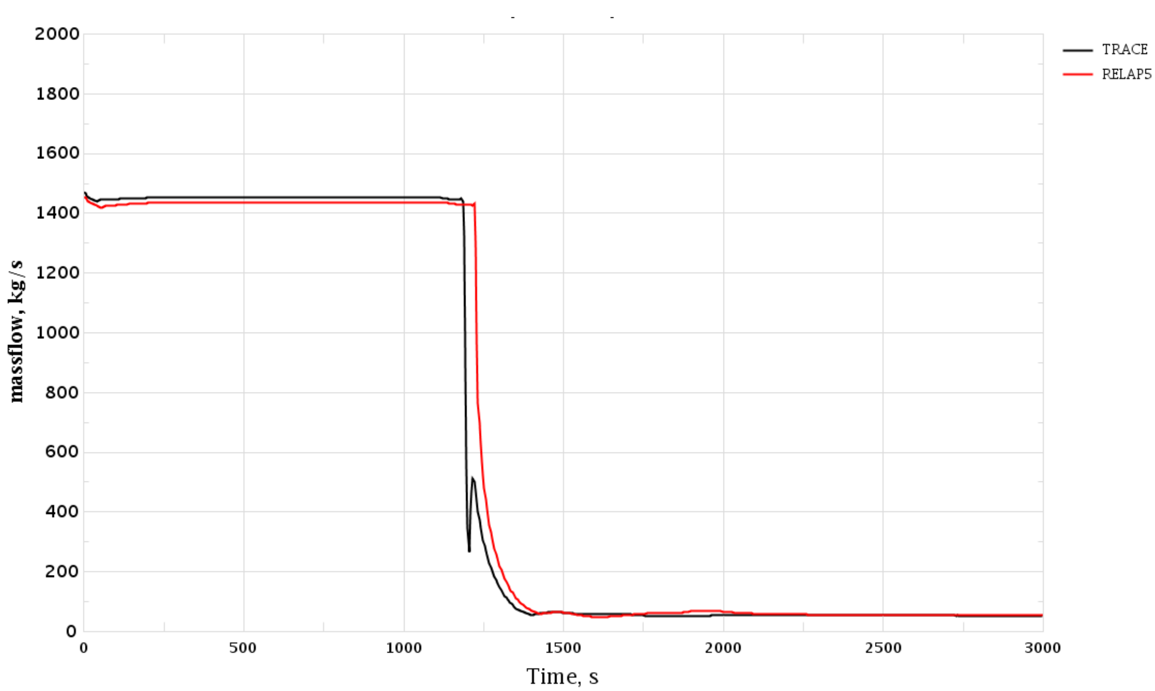

Figure 3-67 Loss of Turbine Condenser Vacuum. RCS Loop 1 Mass Flow Rate

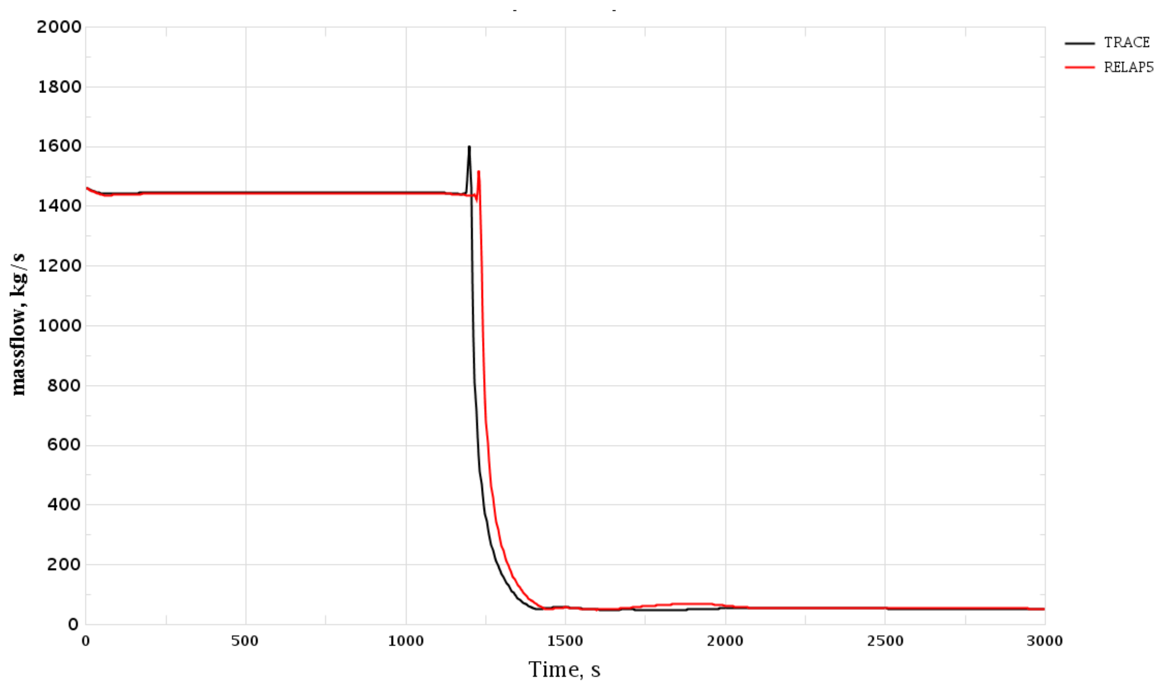

Figure 3-68 Loss of Turbine Condenser Vacuum. RCS Loop 2 Mass Flow Rate

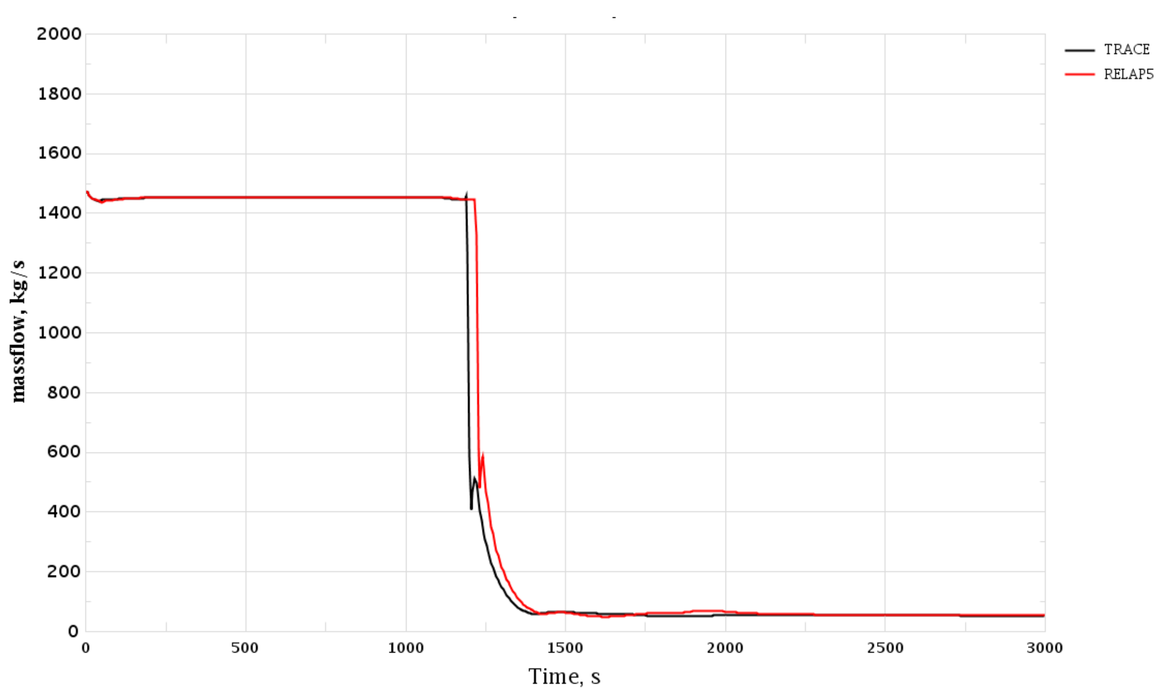

Figure 3-69 Loss of Turbine Condenser Vacuum. RCS Loop 3 Mass Flow Rate

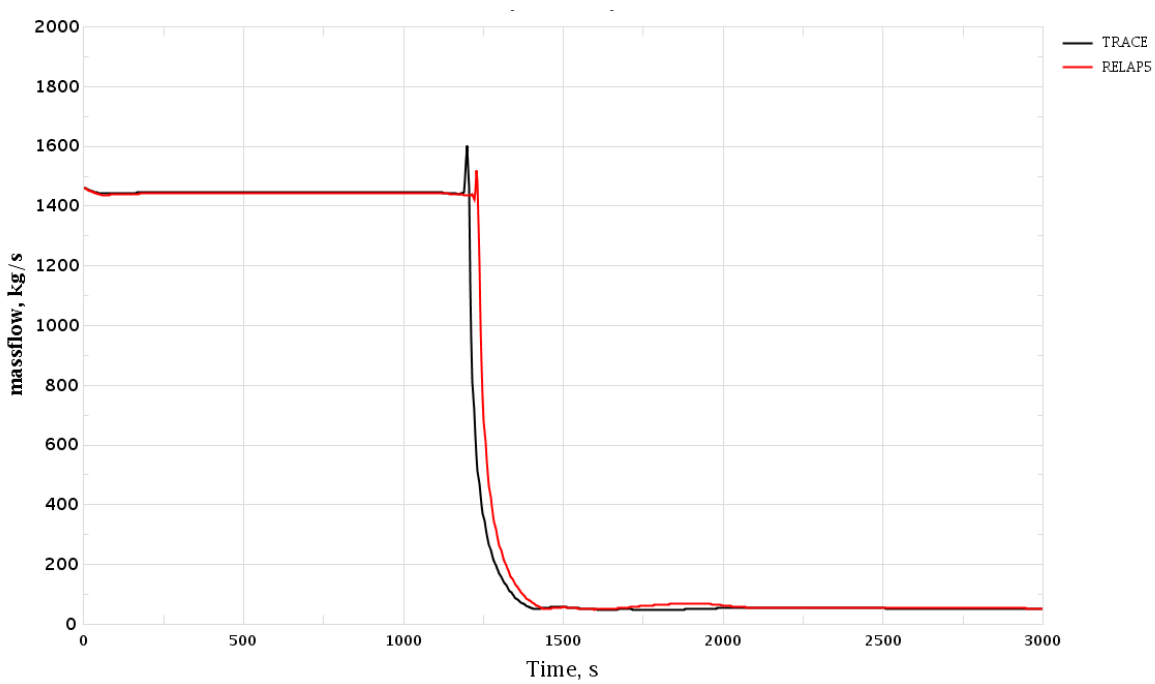

Figure 3-70 Loss of Turbine Condenser Vacuum. RCS Loop 4 Mass Flow Rate

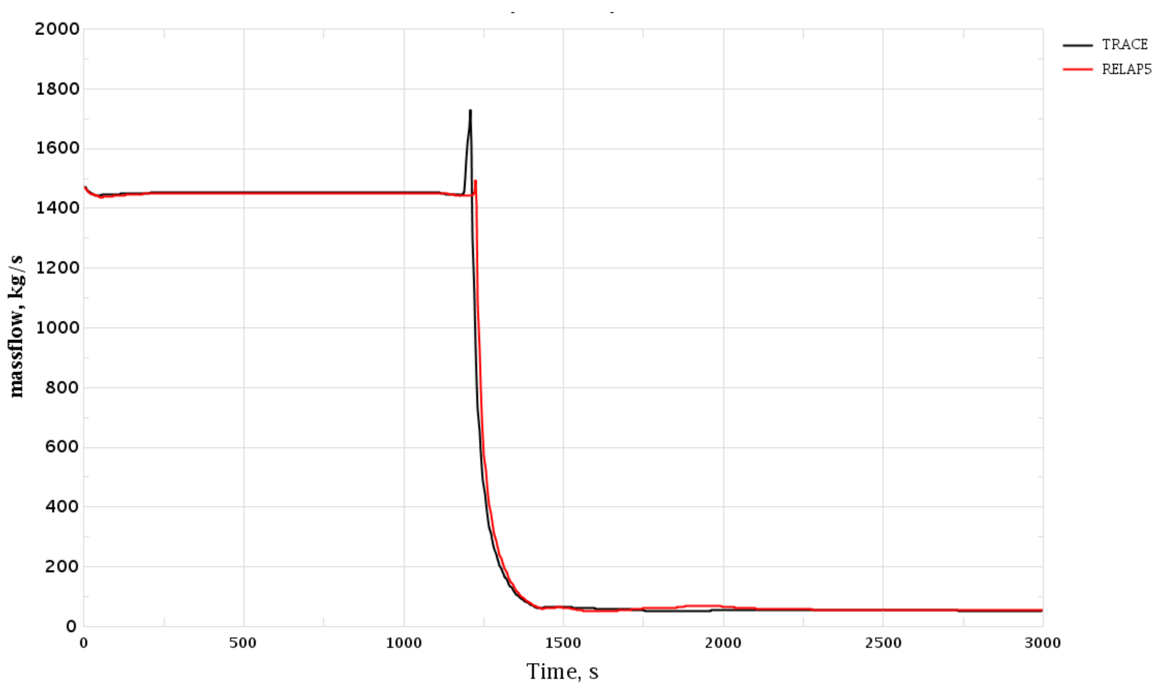

Figure 3-71 Loss of Turbine Condenser Vacuum. RCS Loop 5 Mass Flow Rate

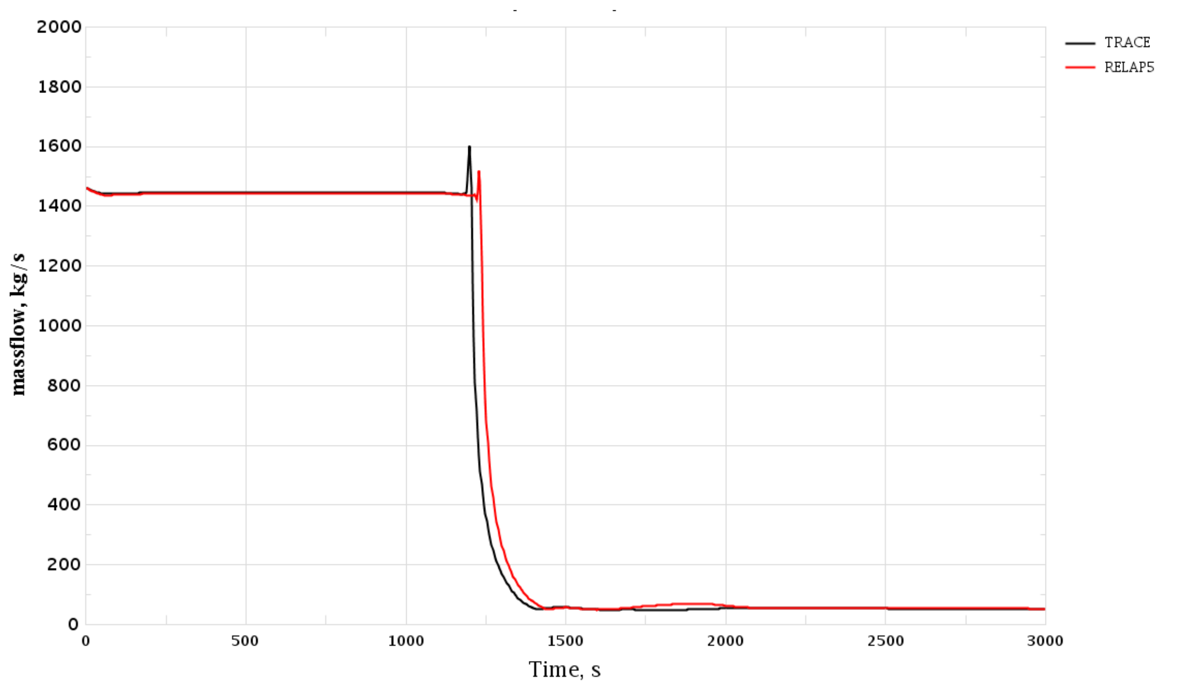

Figure 3-72 Loss of Turbine Condenser Vacuum. RCS Loop 6 Mass Flow Rate

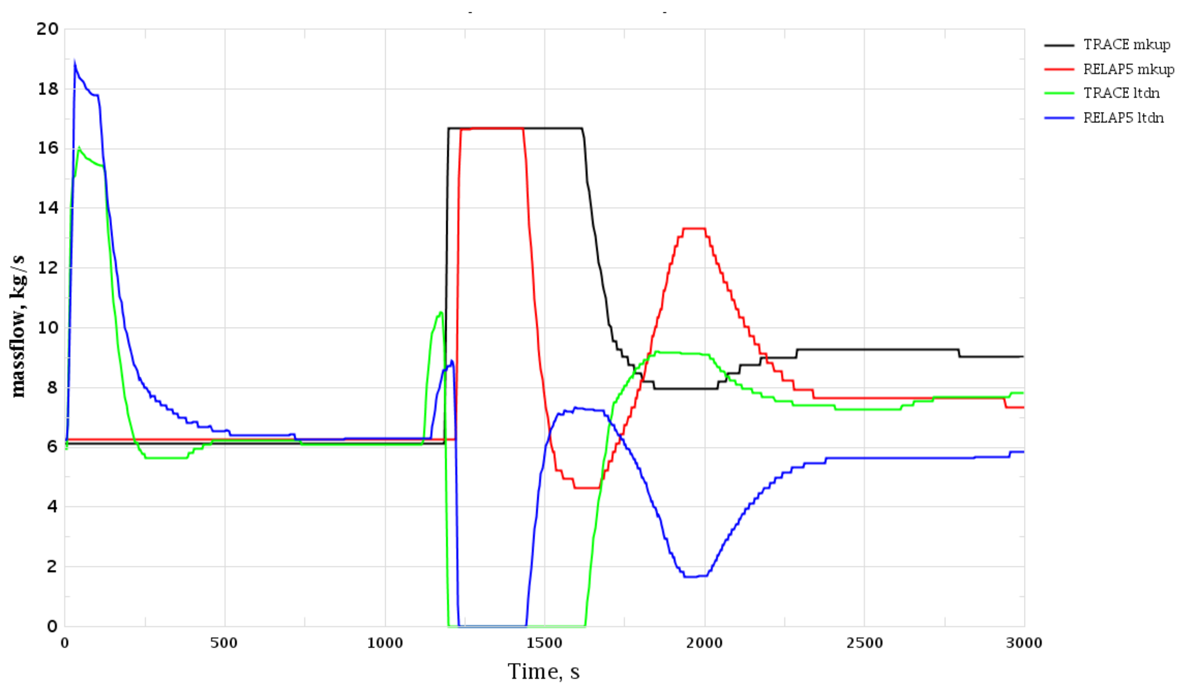

Figure 3-73 Loss of Turbine Condenser Vacuum. Make-Up and Let-Down Mass Flow

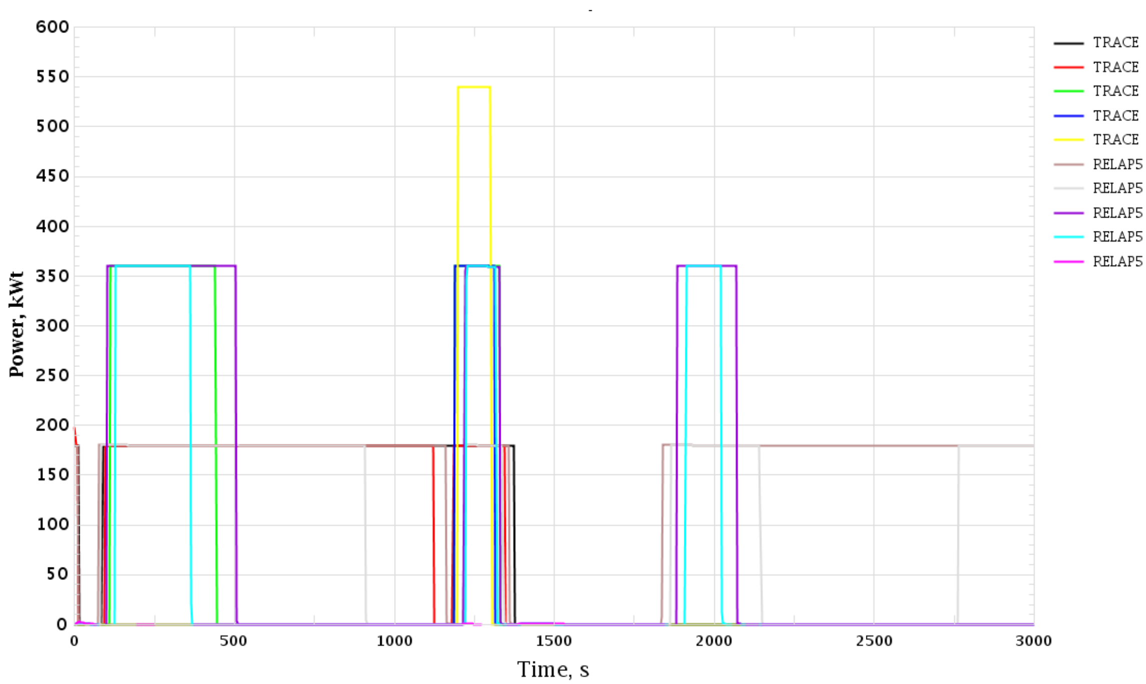

Figure 3-74   Loss of Turbine Condenser Vacuum. PRZ Heaters Power

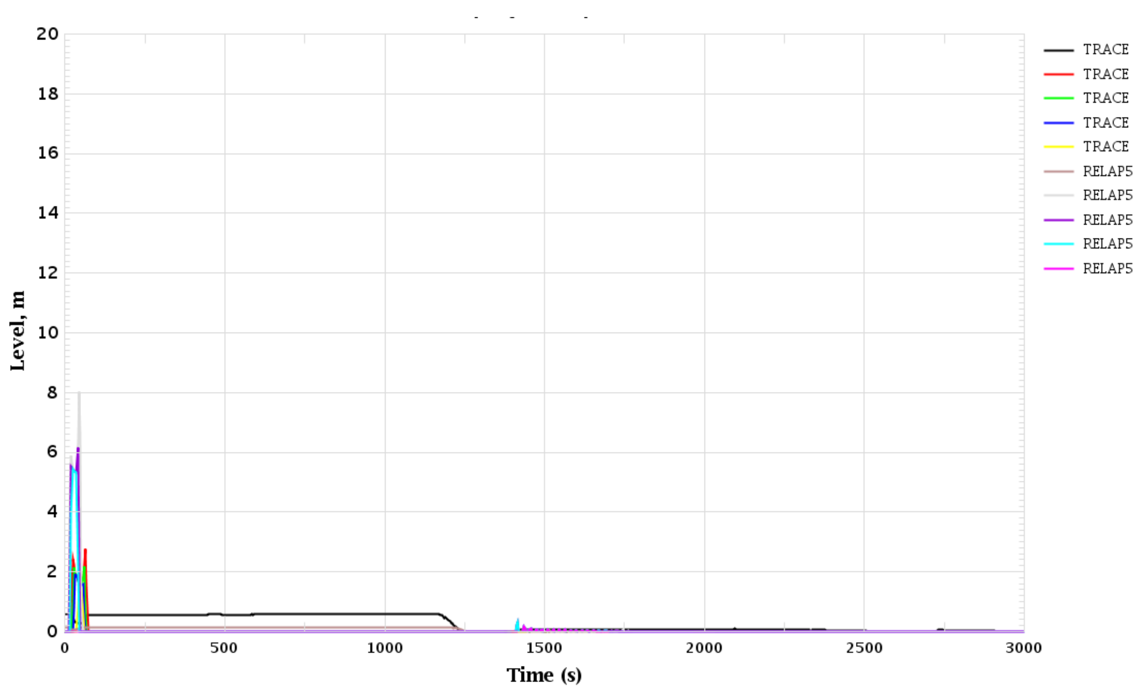

Figure 3-75   Loss of Turbine Condenser Vacuum. PRZ Spray Mass Flow Rate

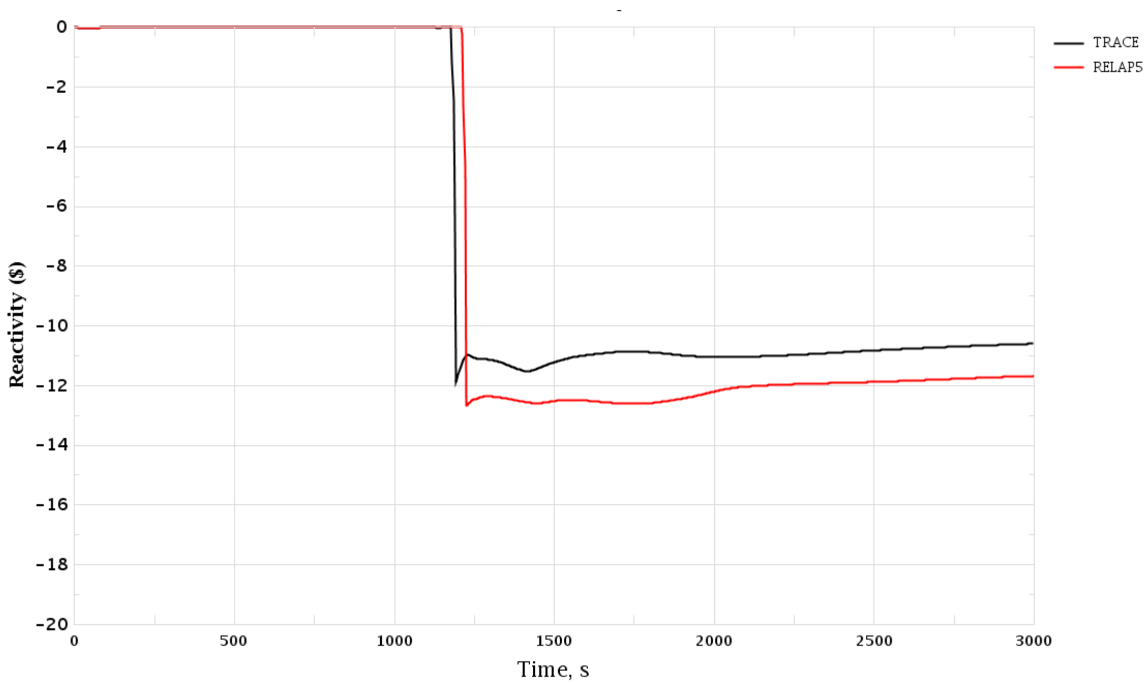

Figure 3-76 Loss of Turbine Condenser Vacuum. Core Reactivity

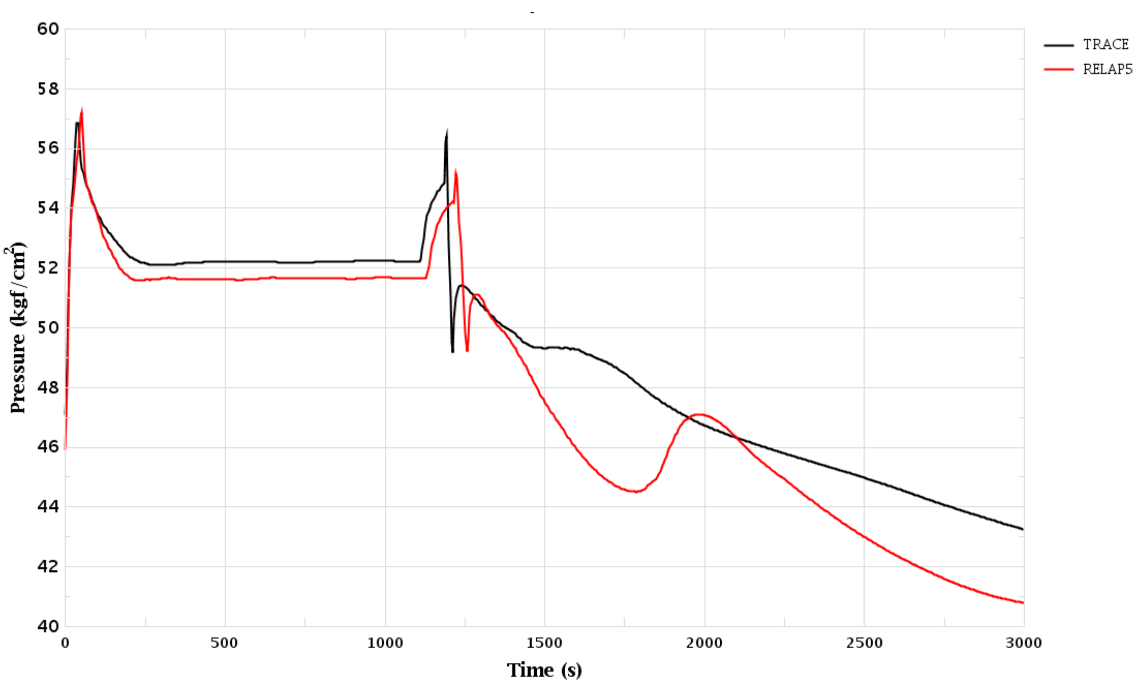

Figure 3-77 Loss of Turbine Condenser Vacuum. SG-1 Pressure

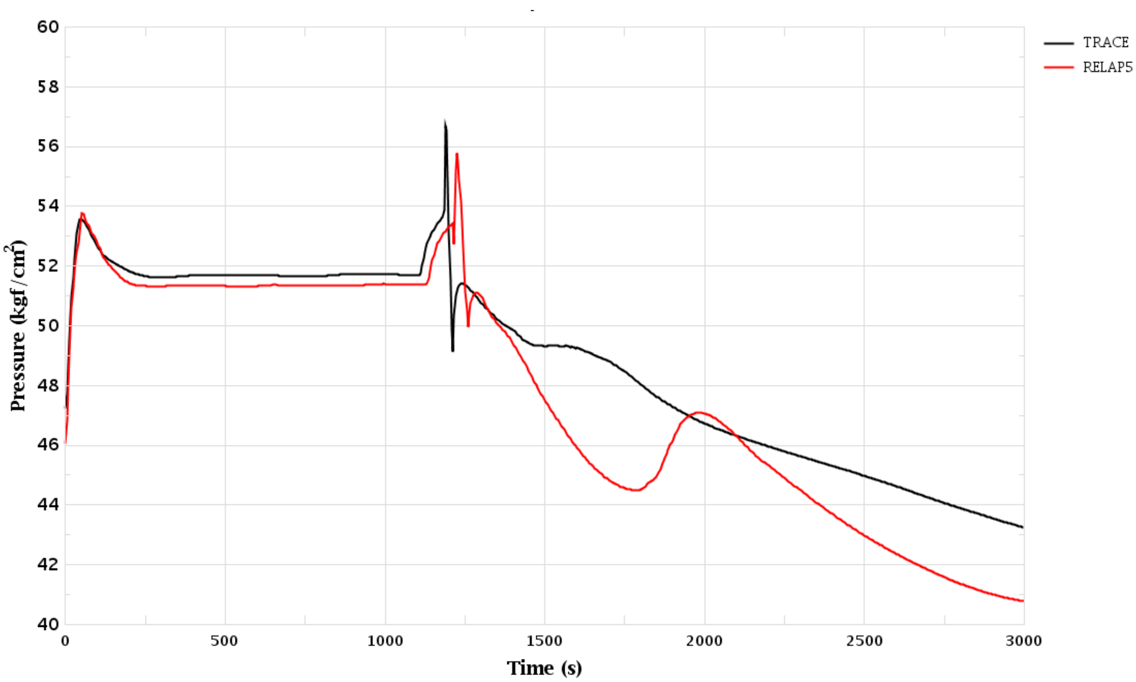

Figure 3-78 Loss of Turbine Condenser Vacuum. SG-2 Pressure

Figure 3-79 Loss of Turbine Condenser Vacuum. SG-3 Pressure

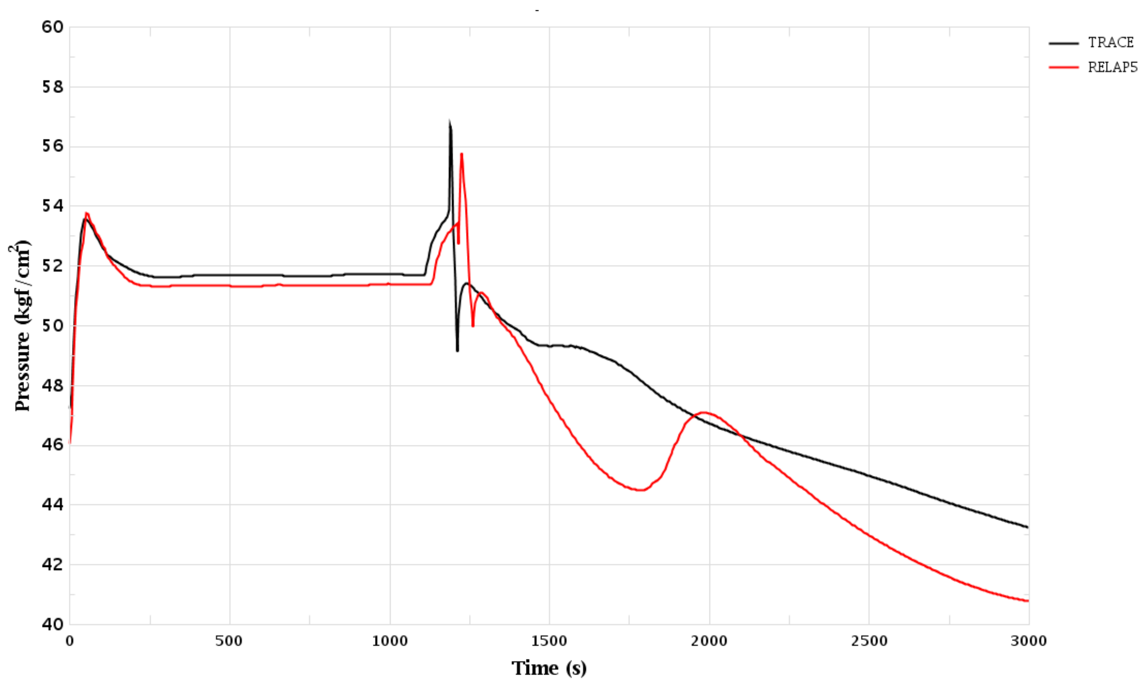

Figure 3-80 Loss of Turbine Condenser Vacuum. SG-4 Pressure

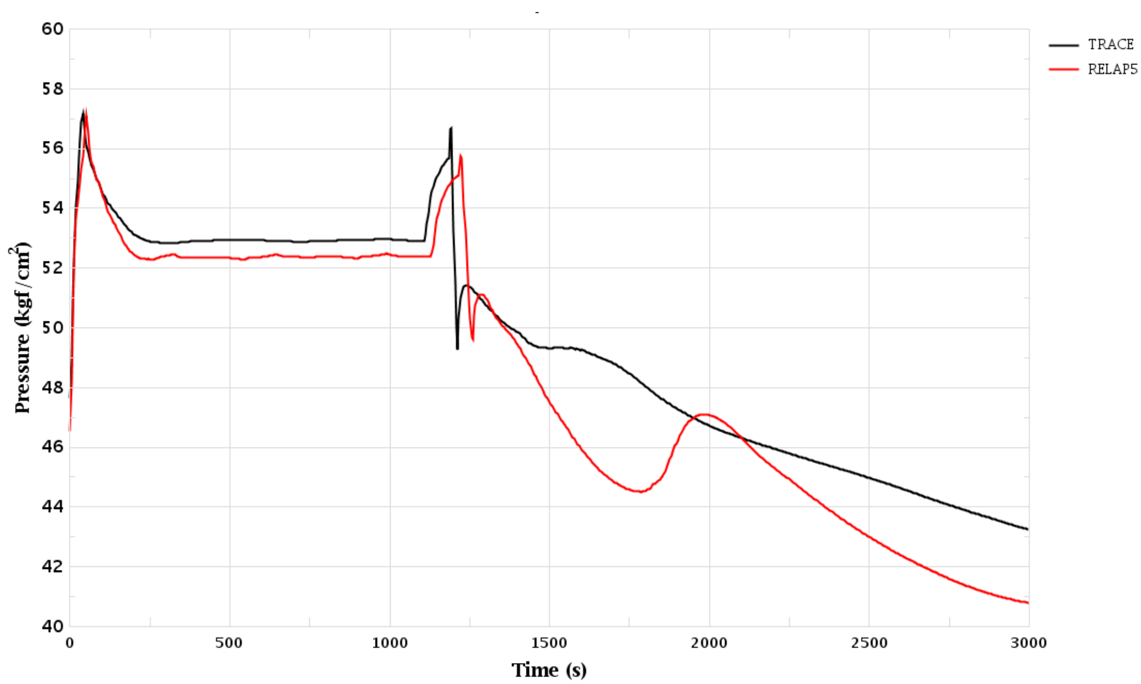

Figure 3-81 Loss of Turbine Condenser Vacuum. SG-5 Pressure

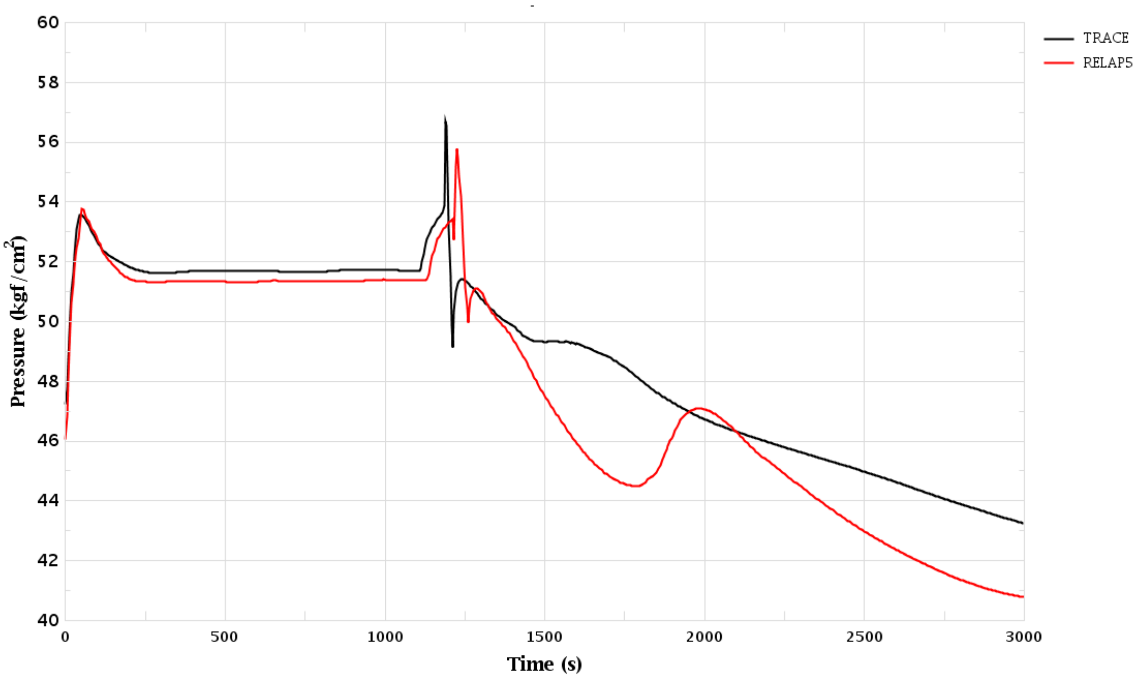

Figure 3-82 Loss of Turbine Condenser Vacuum. SG-6 Pressure

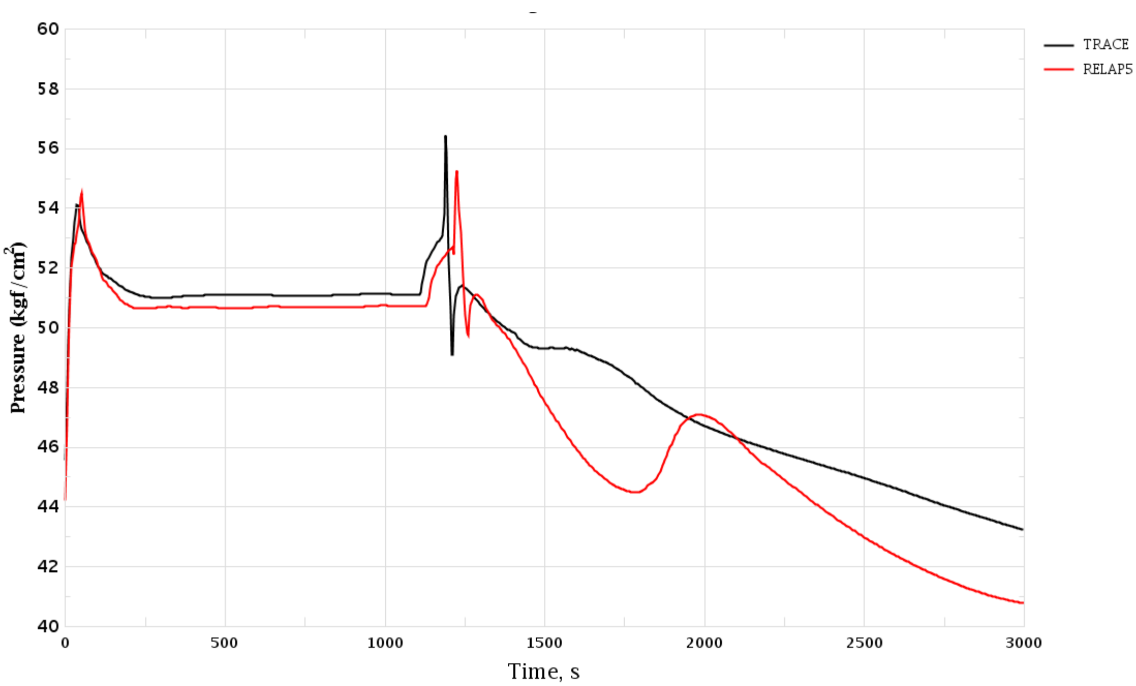

Figure 3-83 Loss of Turbine Condenser Vacuum. MSH-1 Pressure

Figure 3-84 Loss of Turbine Condenser Vacuum. MSH-2 Pressure

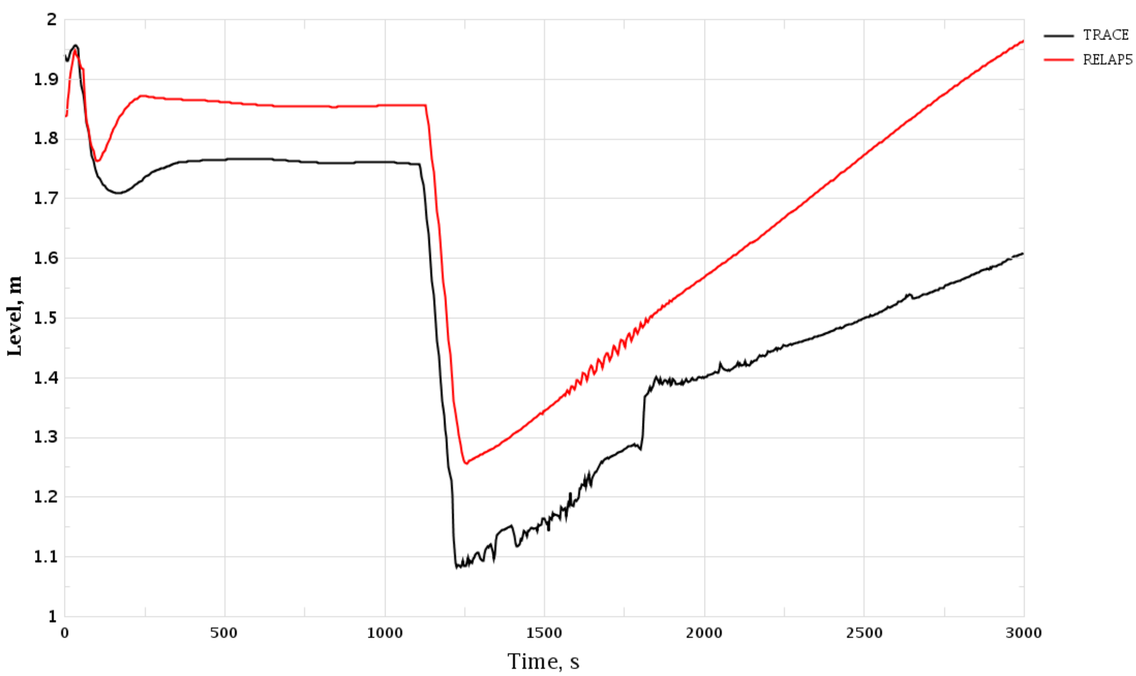

Figure 3-85 Loss of Turbine Condenser Vacuum. SG-1 Level (Wide Range)

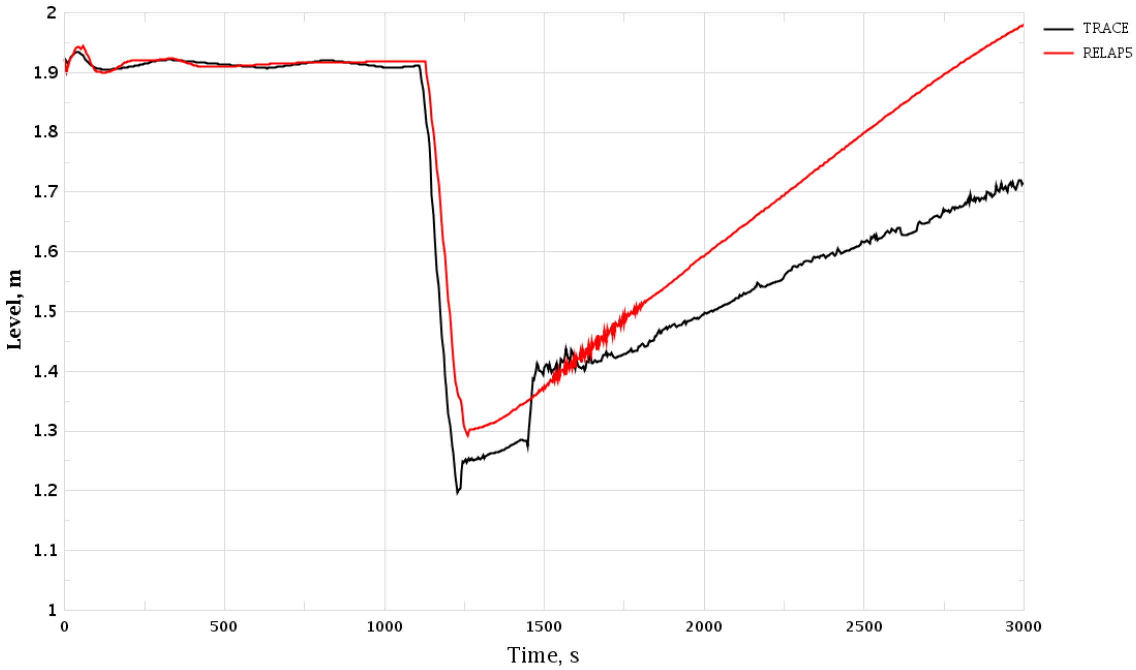

Figure 3-86 Loss of Turbine Condenser Vacuum. SG-2 Level (Wide Range)

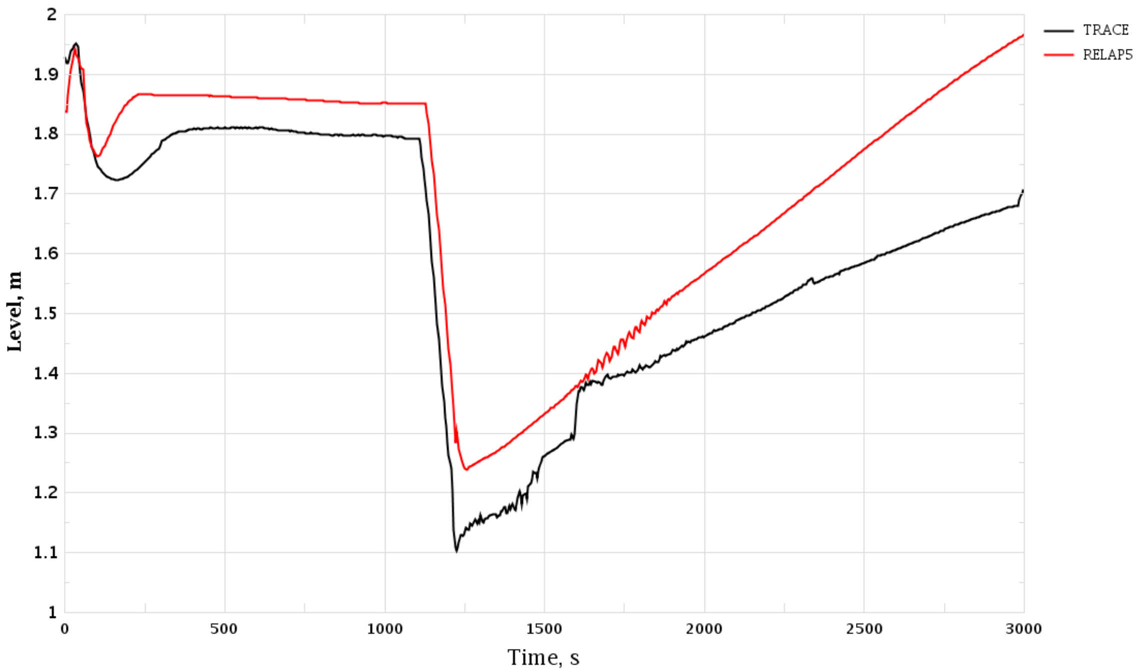

Figure 3-87 Loss of Turbine Condenser Vacuum. SG-3 Level (Wide Range)

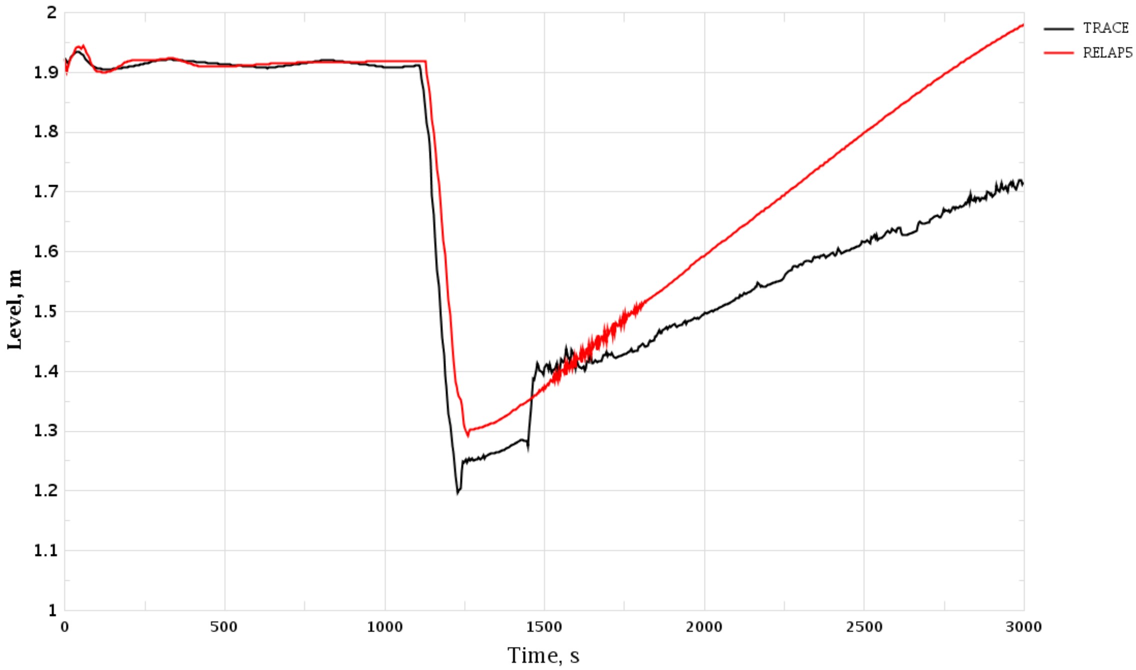

Figure 3-88 Loss of Turbine Condenser Vacuum. SG-4 Level (Wide Range)

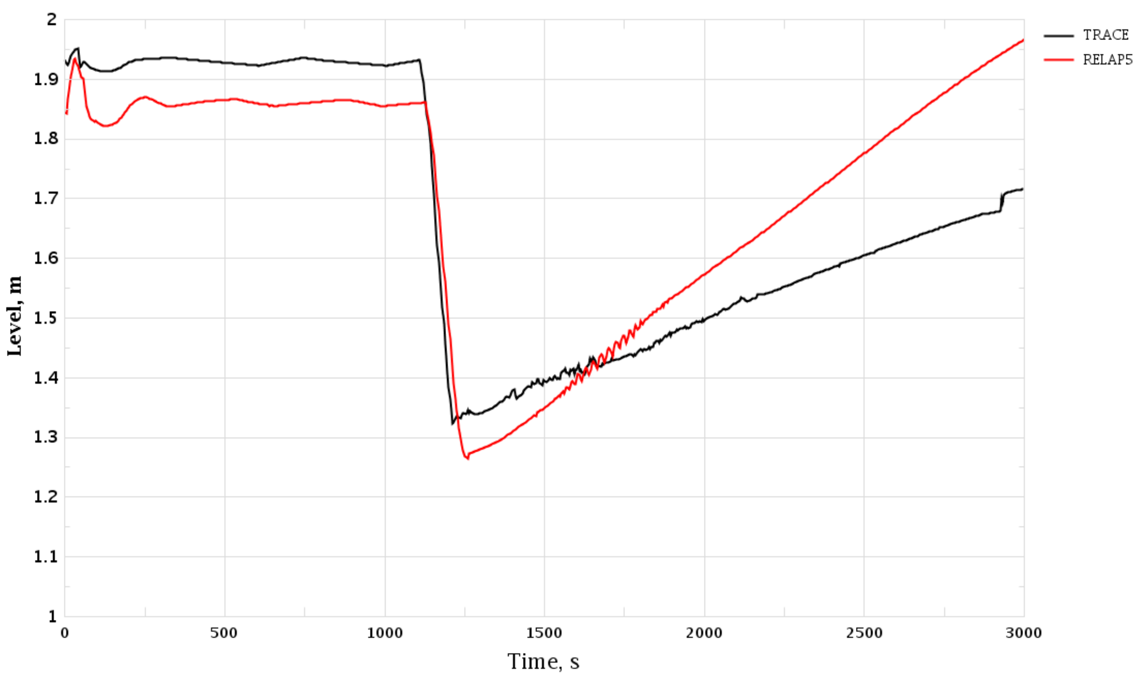

Figure 3-89 Loss of Turbine Condenser Vacuum. SG-5 Level (Wide Range)

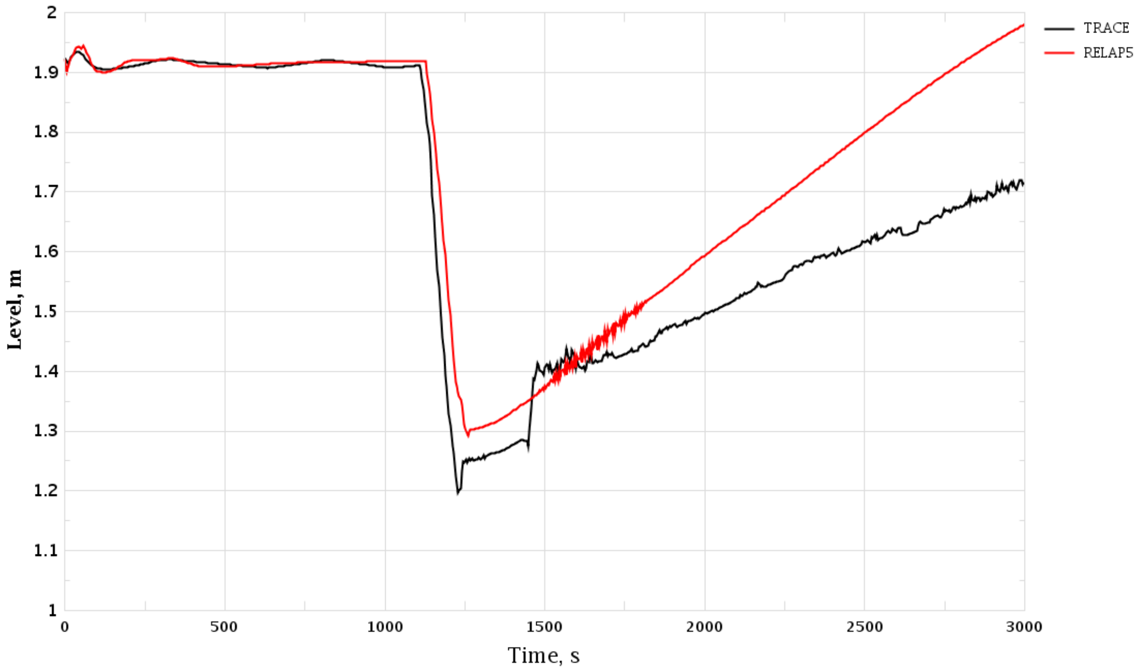

Figure 3-90 Loss of Turbine Condenser Vacuum. SG-6 Level (Wide Range)

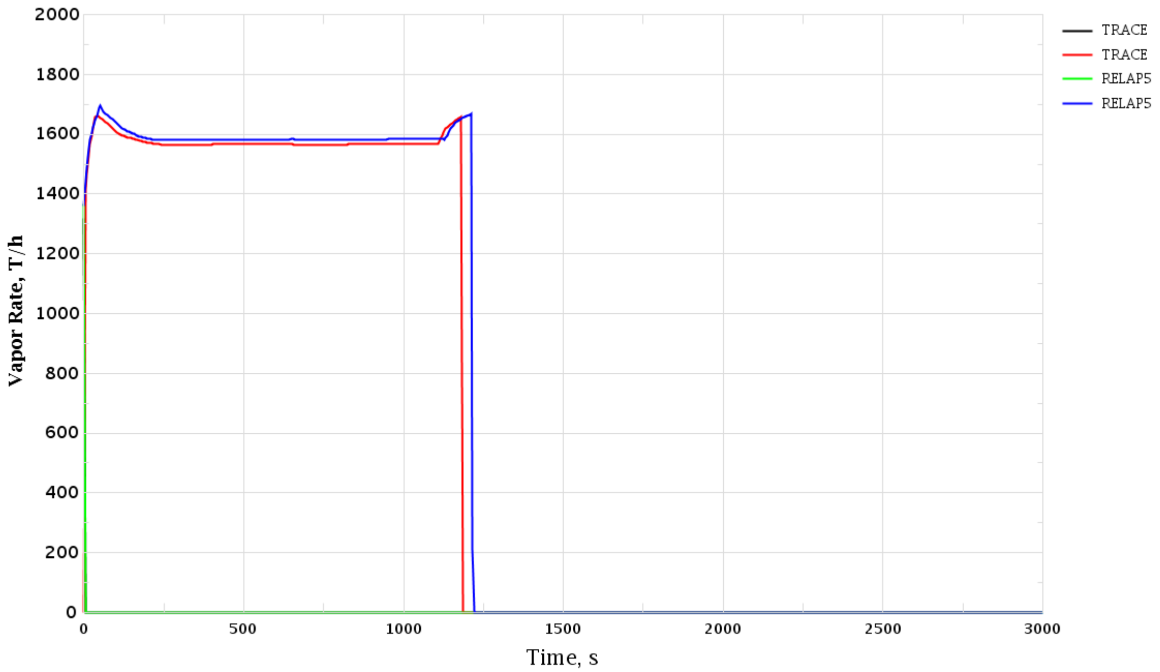

Figure 3-91 Loss of Turbine Condenser Vacuum. Turbines Mass Flow Rate

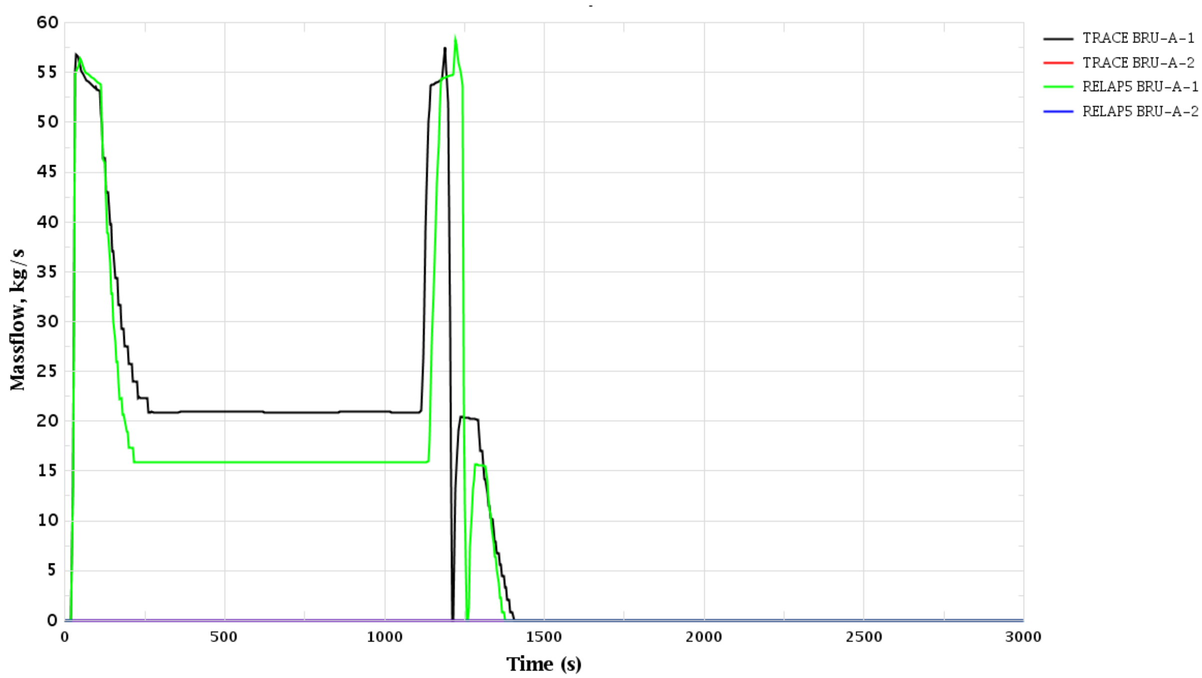

Figure 3-92 Loss of Turbine Condenser Vacuum. BRU-A Steam Mass Flow Rate

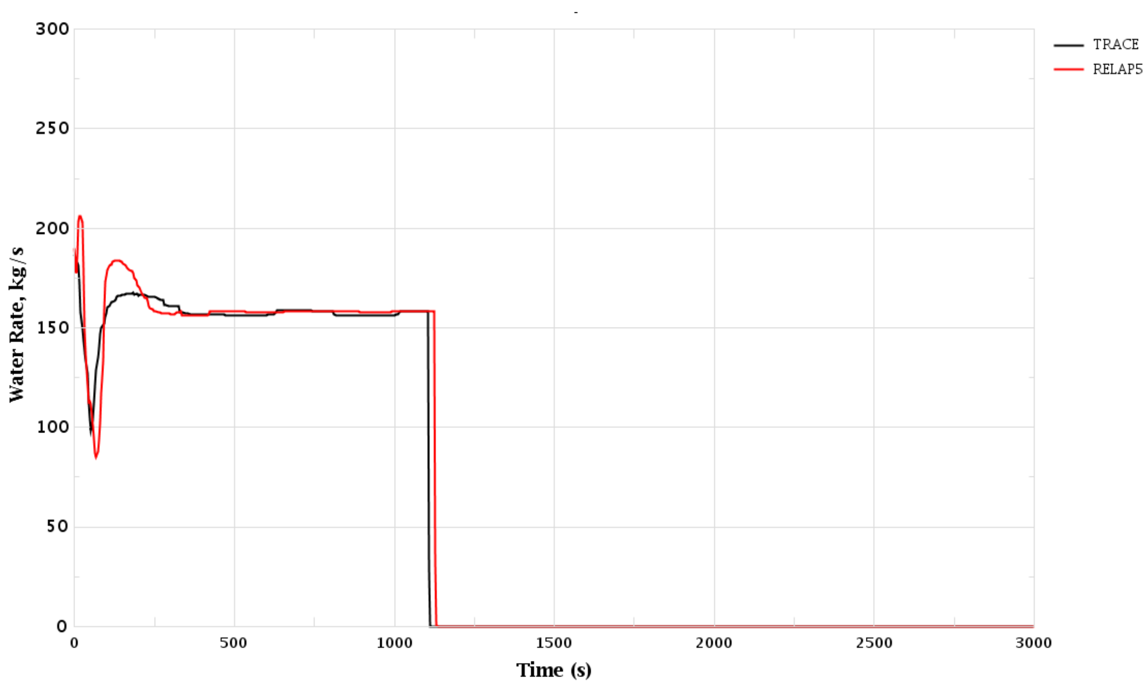

Figure 3-93 Loss of Turbine Condenser Vacuum. MFW Pump No.1 Mass Flow Rate

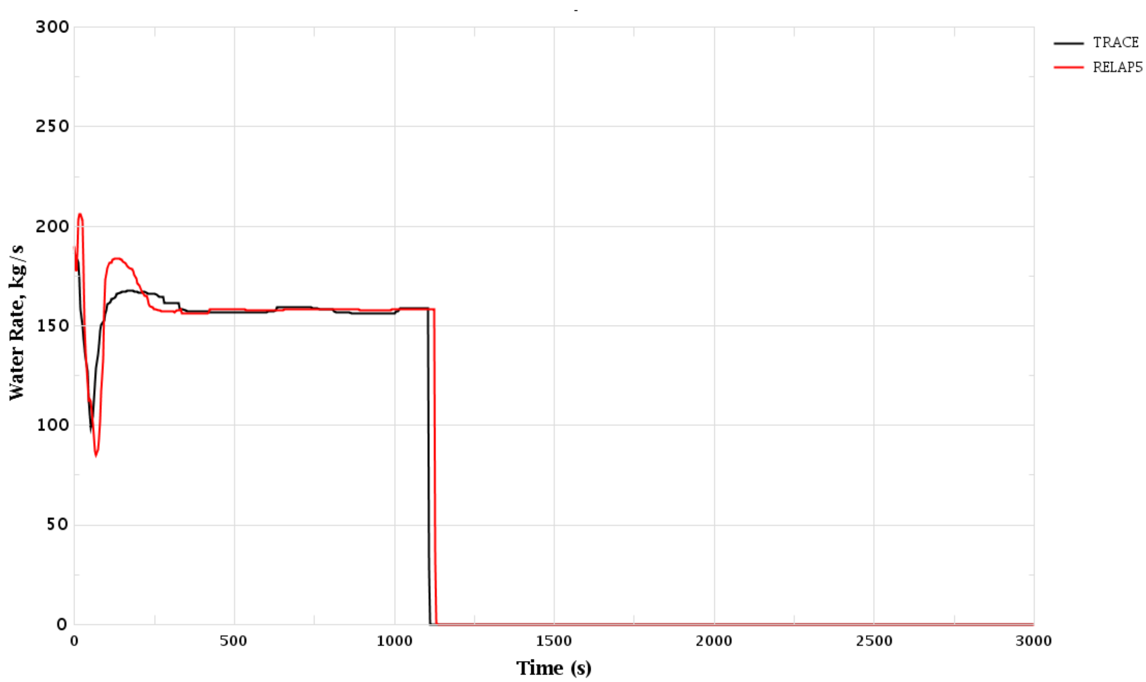

Figure 3-94 Loss of Turbine Condenser Vacuum. MFW Pump No.2 Mass Flow Rate

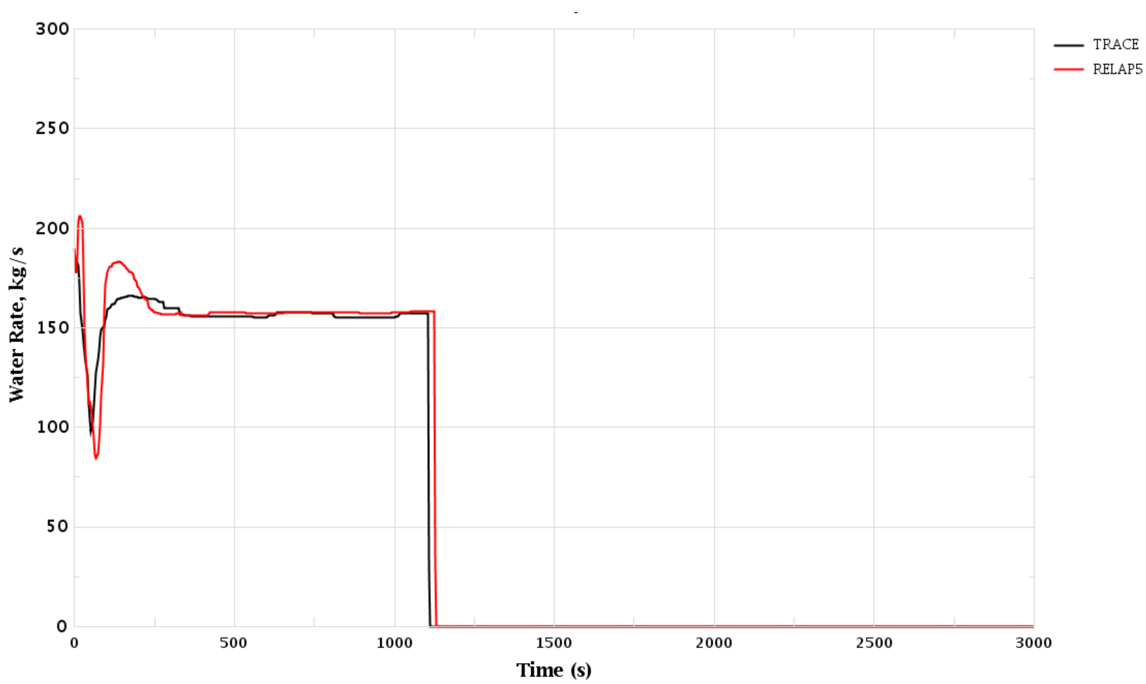

Figure 3-95 Loss of Turbine Condenser Vacuum. MFW Pump No.3 Mass Flow Rate

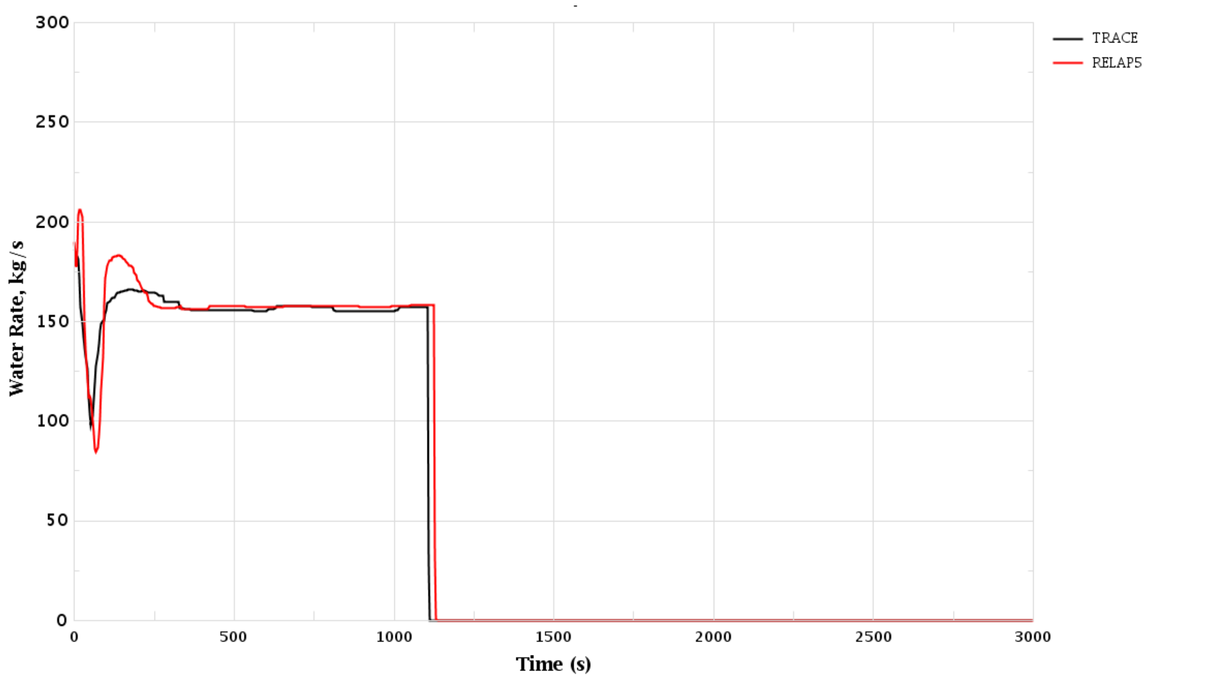

Figure 3-96 Loss of Turbine Condenser Vacuum. MFW Pump No.4 Mass Flow Rate

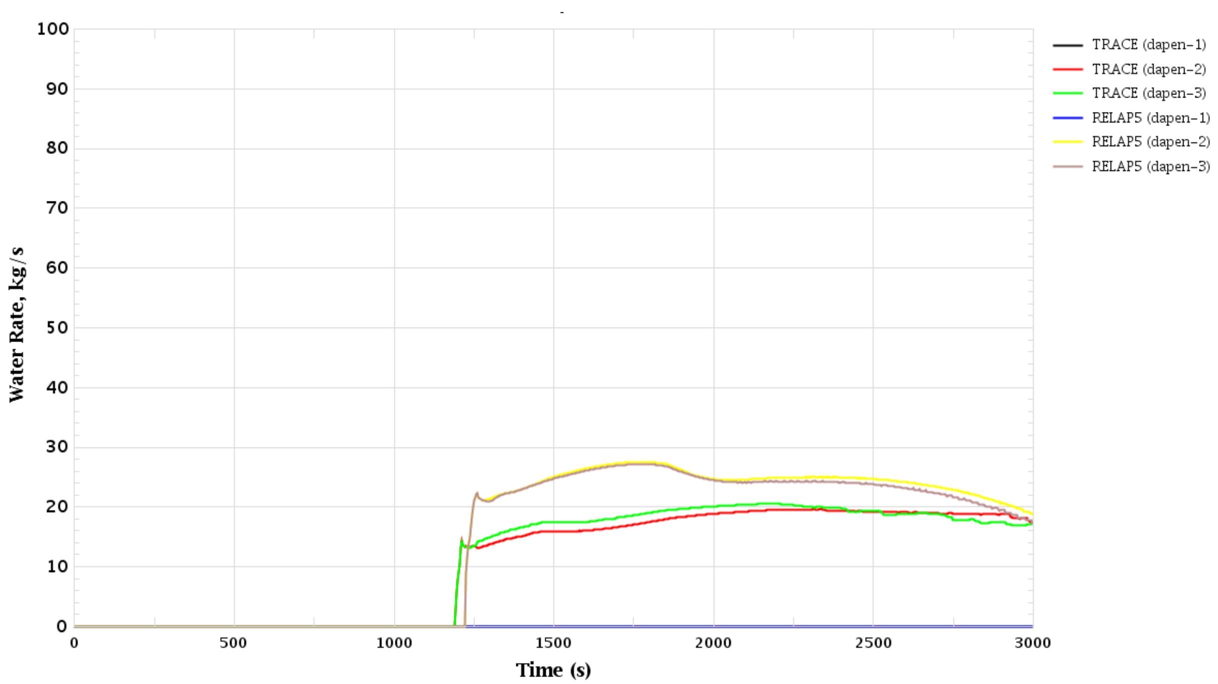

Figure 3-97 Loss of Turbine Condenser Vacuum. EFW Pumps Mass Flow Rate

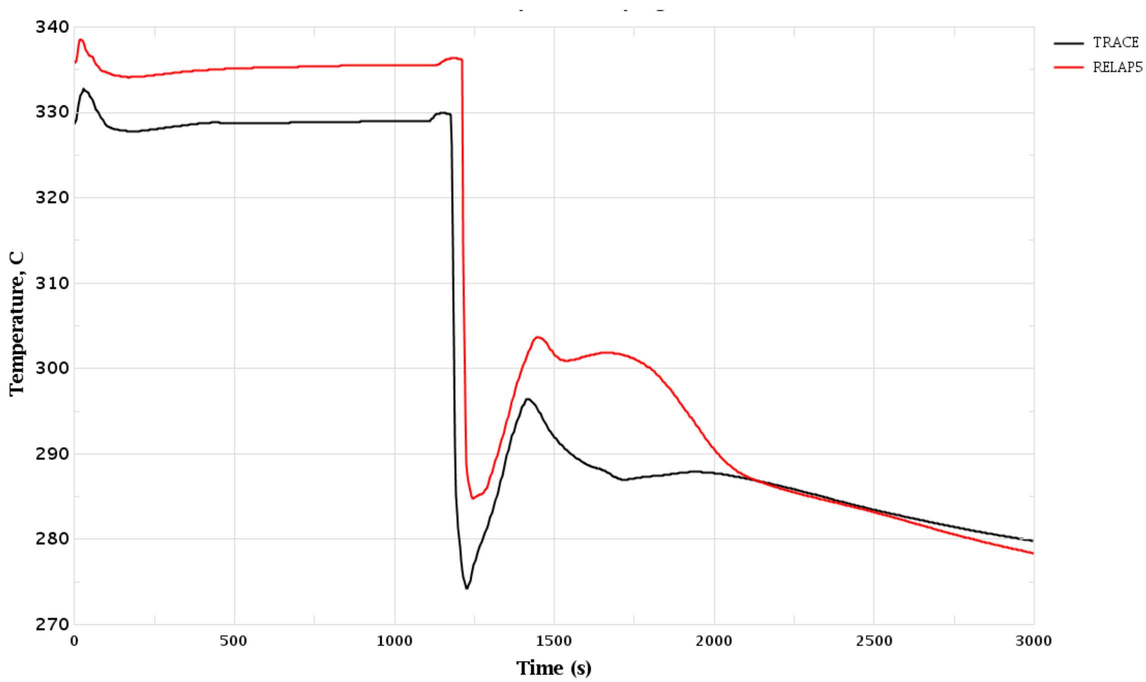

Figure 3-98 Loss of Turbine Condenser Vacuum. Maximal Cladding Temperature

3.3 Trip of 4 Out of 6 RCPs

3.3.1 Brief Description of Initiating Event

This initiating event assumes simultaneous trip of 4 out of 6 RCPs. According to the expected frequency of occurrence the initiating events is categorized as transient and leads to a decrease of reactor coolant flow.

3.3.2 Initial Conditions

The initial conditions selected for transient calculation correspond to those specified in Table 3-1.

3.3.3 Boundary Conditions

Assumptions on the systems availability that are considered in calculation of the transient are specified below.

No operator actions are simulated in the scenario.

Control of reactor power, primary pressure, PRZ and SG levels, and MSH pressure is performed by automatic controllers taking into account their characteristics.

Conservatively the operation of RPL, Level 4, Level 3, Level 2 emergency protection is not considered. This allows ARM to operate in neutron power maintenance mode without a prohibition to withdraw the control group of control assemblies and prolong reactor operation at the increased power level (comparing to the one that corresponds to number of operating RCPs). Make-up system is assumed to fail at the moment of IE occurrence.

The 1.0 s delay is postulated for start of control assemblies drop to the reactor core after reactor scram initiation.

Steam pressure in MSH is maintained by operation of BRU-A. Operation of turbine bypass to condenser (BRU-K) is not taken into account.

RCP coast-down is performed according to pump characteristics.

3.3.4 Calculation Results

Sequence of events for this transient is presented in Table 3-4.

Table 3-4 Sequence of Events for Trip of 4/6 RCP Transient

TRACE Time, s	RELAP5 Time, s	Event	Description
0.0	0.0	Trip of 4/6 RCPs (RCP-2, 4, 5, 6)	IE occurrence
0.0	0.0	Failure of make-up system Failure of BRU-K	Boundary conditions
3.0	3.0	Reactor scram signal	Scram actuation due to a trip of 4 (or more) RCPs with 3 s delay
4.0	4.0	Start of control assemblies drop due to actuation of scram	1 s delay after scram signal initiation is postulated
7.5	7.5	Maximal primary circuit pressure of 124.1/124.8 kgf/cm^2 is reached	
9.0	9.0	Closure of the stop valves of both turbines	Due to scram actuation (with time delay)
17.0	16.6	Closure of let-down valves	PRZ level decrease for 0.300 m from the nominal level
24.0	26.7	Opening of BRU-A1, BRU-A2	Automatic operation of BRU-A controllers in the pressure maintenance mode
45.0	25.2	Maximal steam lines pressure of 52.3 kgf/cm^2 (TRACE) and 52.7 kgf/cm^2 (RELAP) is reached	
200.0	199.4	End of RCP-2,4,5,6 coast-down	
1000.0	1000.0	End of calculation	Stabilization of parameters

After a trip of RCP-2, 4, 5, 6 the coolant flow rate in the loops with tripped RCPs rapidly decreases, while flow rate in the loops with operating RCPs increases due to a decrease of total pressure losses (Figure 3-114 – Figure 3-119). Reactor scram signal is generated with 3.0 s time delay by the "Trip of 4/6 RCPs" signal. In 1 s after the scram signal actuation (additional delay for signal transmission and disconnection of control assemblies' drives) the control assemblies start to drop into the reactor core. By this time, neutron reactor power is already decreased below the nominal due to the temperature reactivity feedback (Figure 3-99).

After scram actuation core power decreases down to decay heat, and hot legs temperature (Figure 3-102 – Figure 3-107) and RCS pressure (Figure 3-100) rapidly decrease. Coolant temperature decrease and coolant shrinkage cause rapid drop of PRZ level (Figure 3-101).

Closure of the stop valves of both turbines that occurs at 9 s of calculation (with a delay after reactor scram) causes sharp increase of the secondary circuit pressure (Figure 3-130, Figure 3-131). BRU-As open automatically and start to dump steam in order to maintain MSH pressure according to their operation logic (Figure 3-139). The maximum pressure reached in MSH semi-headers is 52.3/52.7 kgf/cm^2.

Decay heat power decrease and stable coolant circulation (from operating RCPs) causes decrease and subsequent stabilization of hot legs temperature (Figure 3-102 – Figure 3-107). Initial (within first 50 s of transient) decrease of SG level (Figure 3-132 – Figure 3-137) is compensated by operation of MFW controllers and to the 150-200 s of transient the SG levels are restored.

Reaching the balance between core decay heat and heat removed by dumping steam via BRU-A (Figure 3-139) the main parameters of the primary and secondary circuit are stabilized.

The plots of the main parameters of calculation are presented below on Figure 3-99 – Figure 3-144.

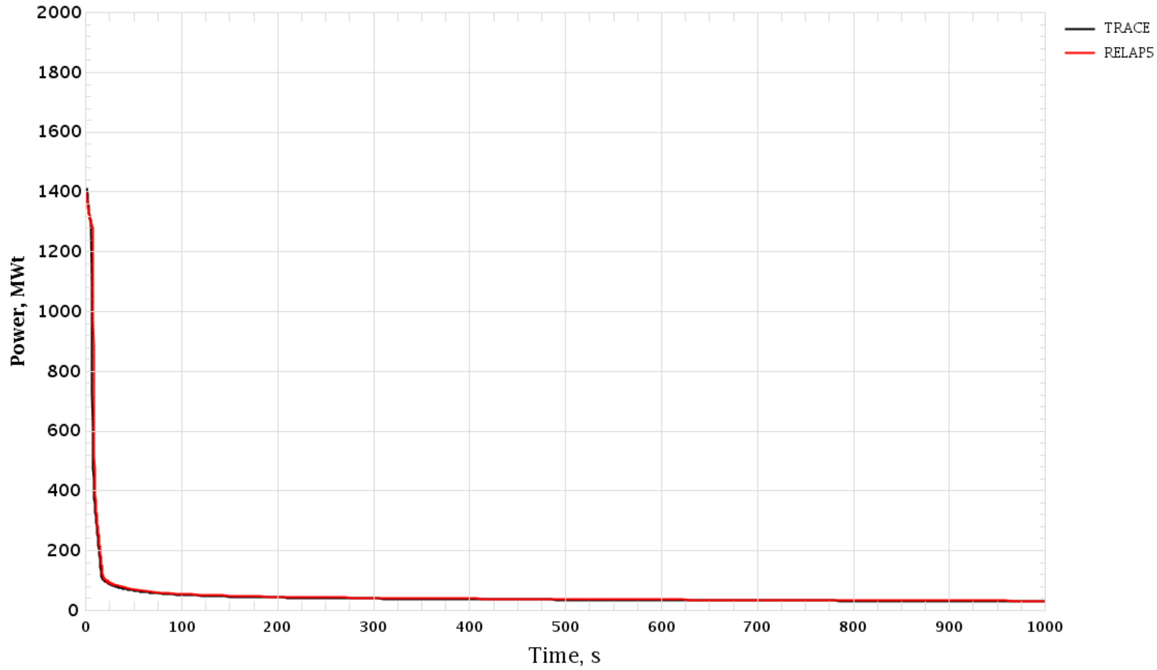

Figure 3-99 Trip of 4/6 RCPs. Core Thermal Power

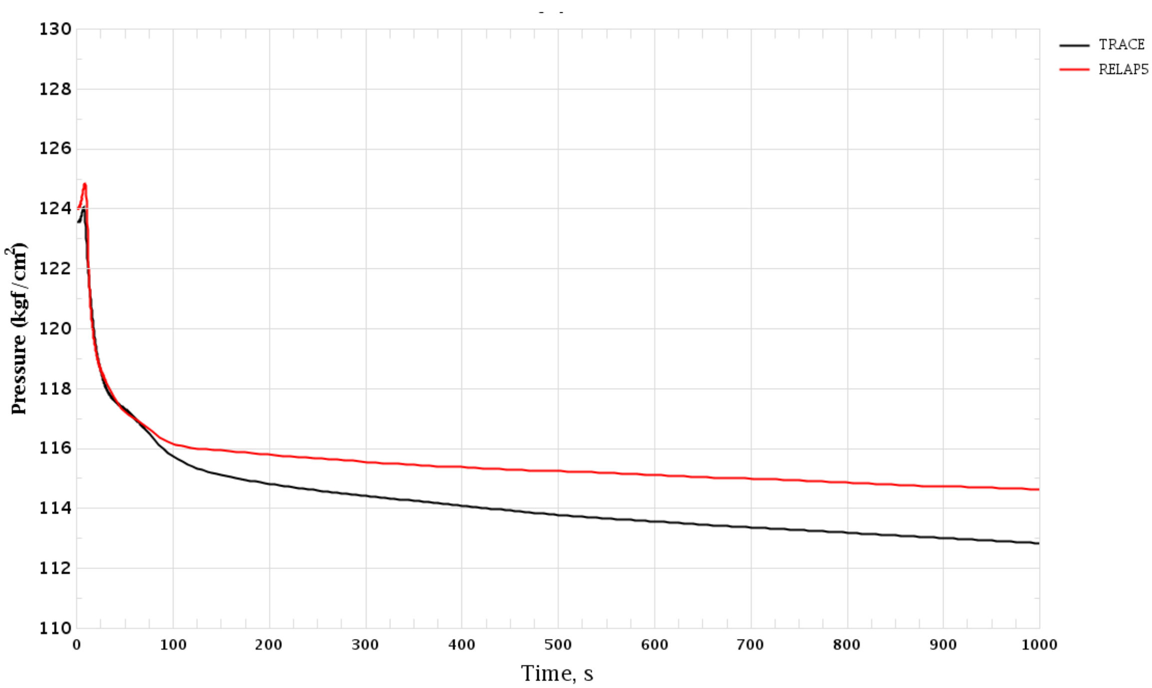

Figure 3-100 Trip of 4/6 RCPs. RCS Pressure

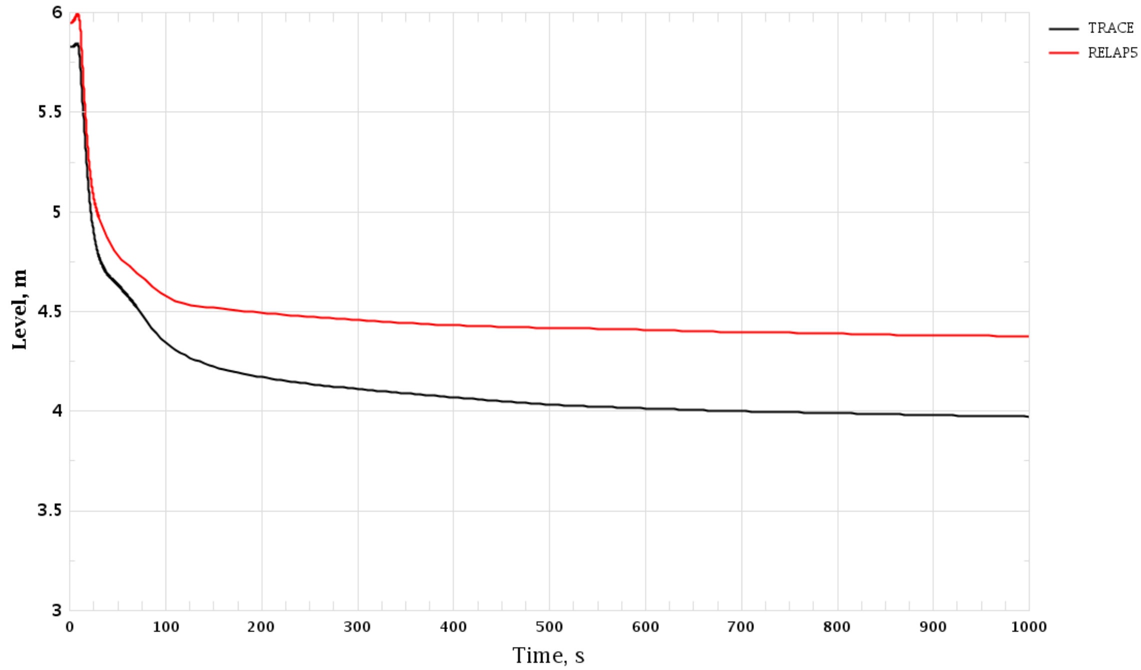

Figure 3-101 Trip of 4/6 RCPs. Pressurizer Level

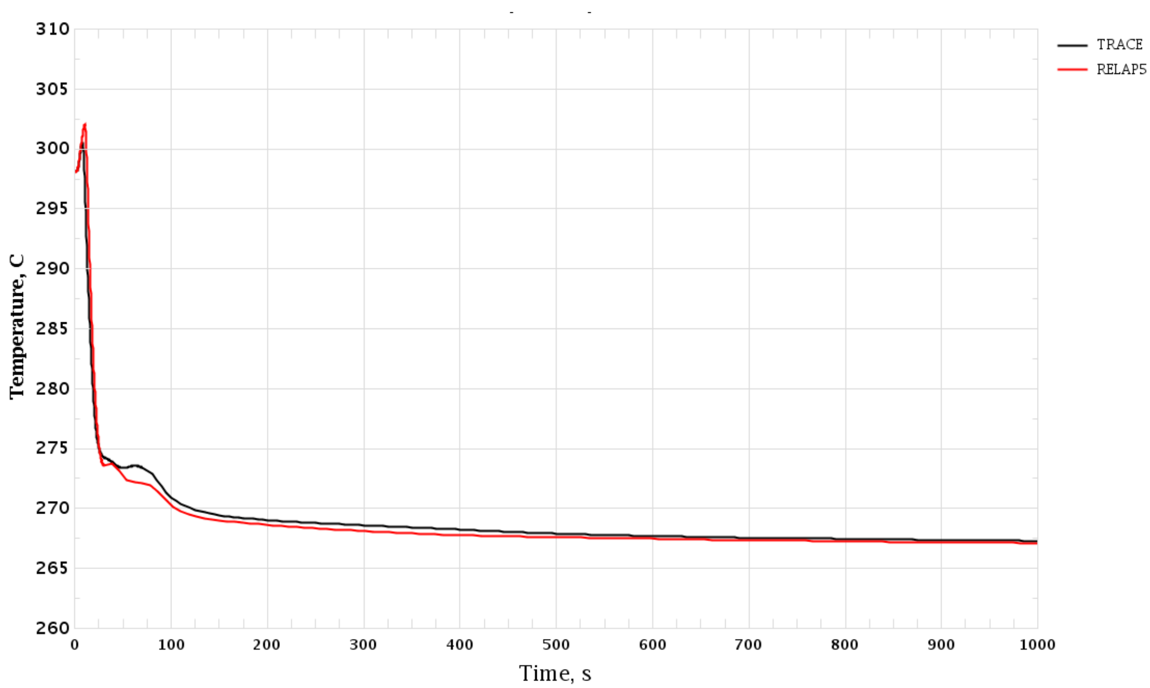

Figure 3-102 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 1

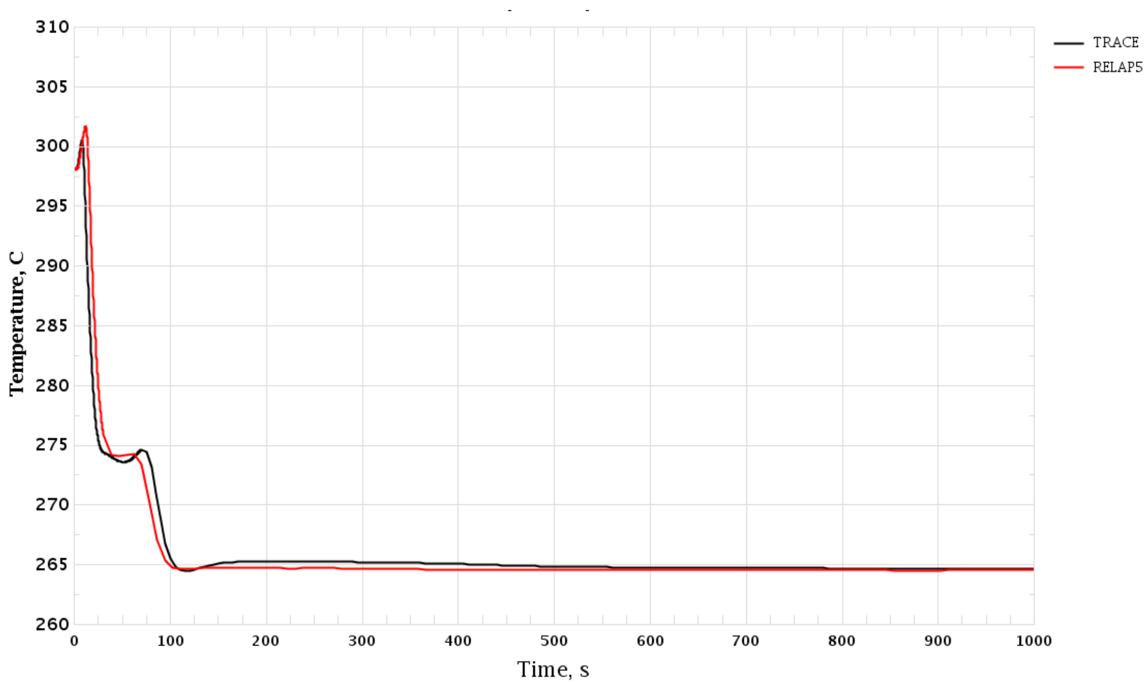

Figure 3-103 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 2

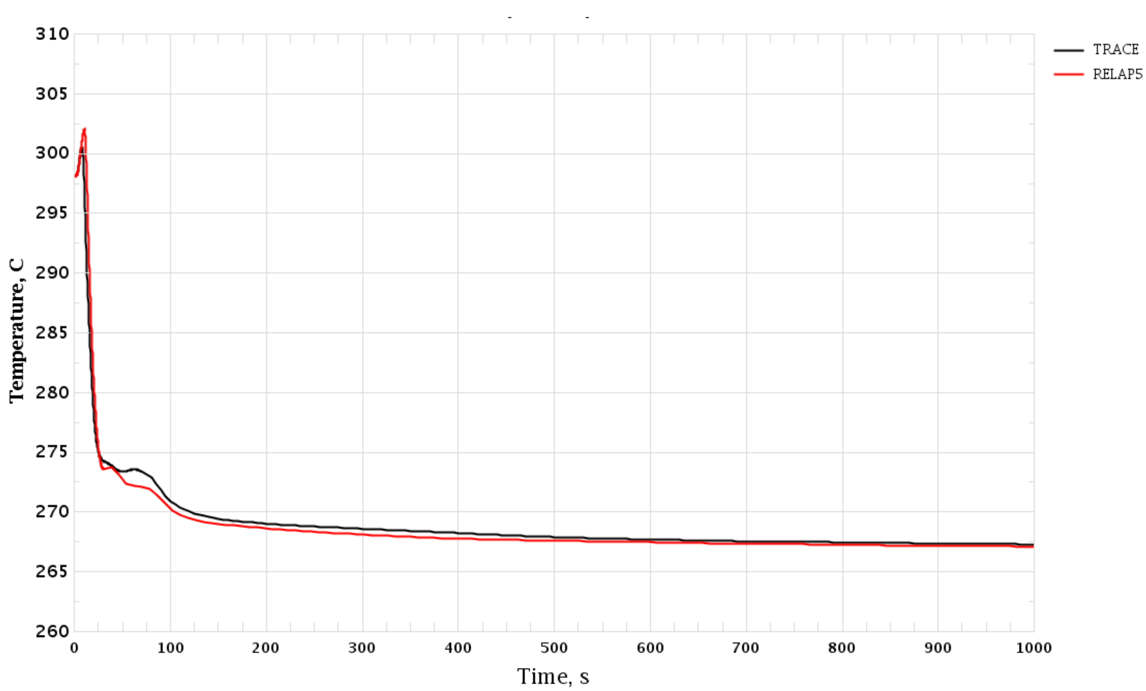

Figure 3-104 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 3

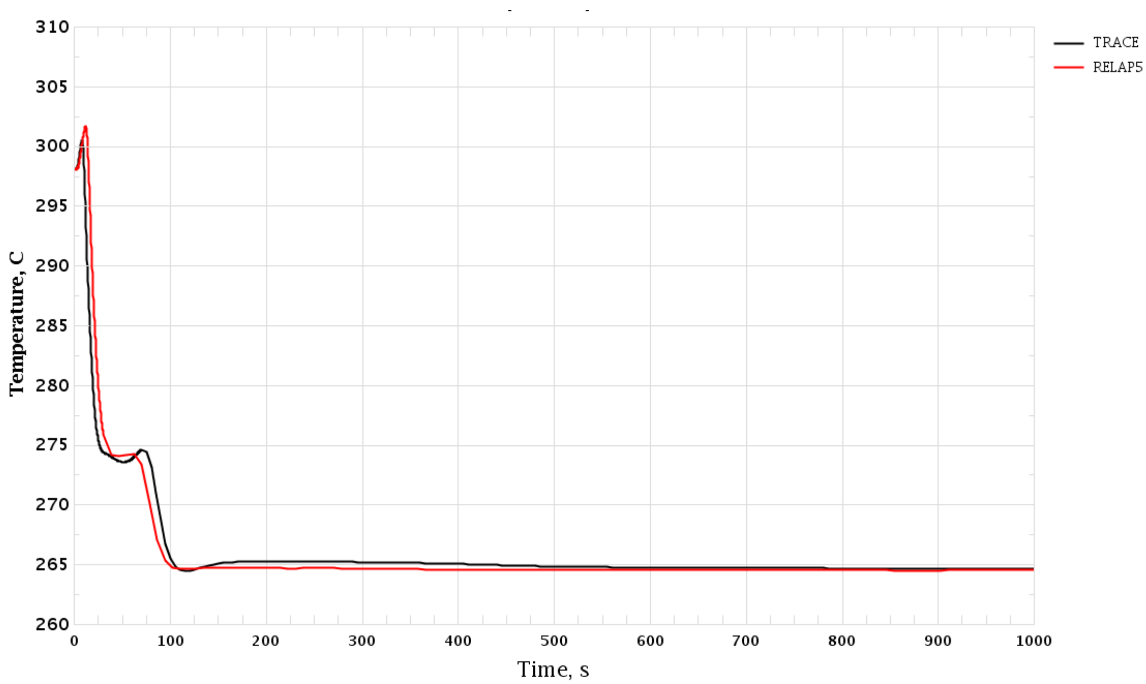

Figure 3-105 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 4

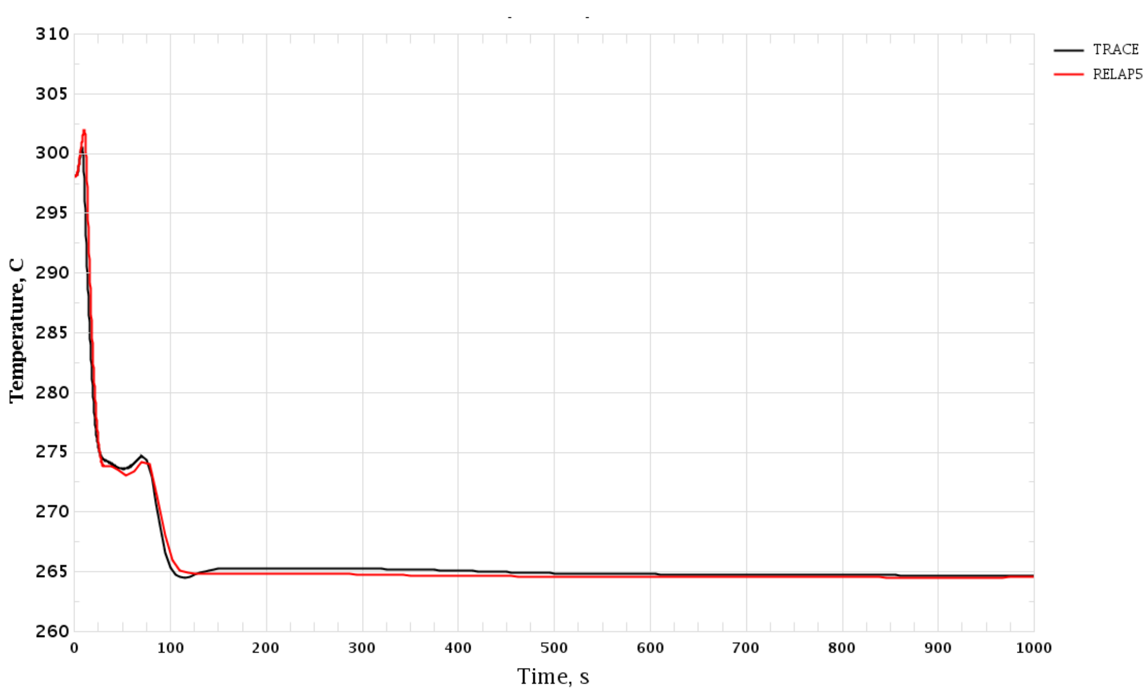

Figure 3-106 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 5

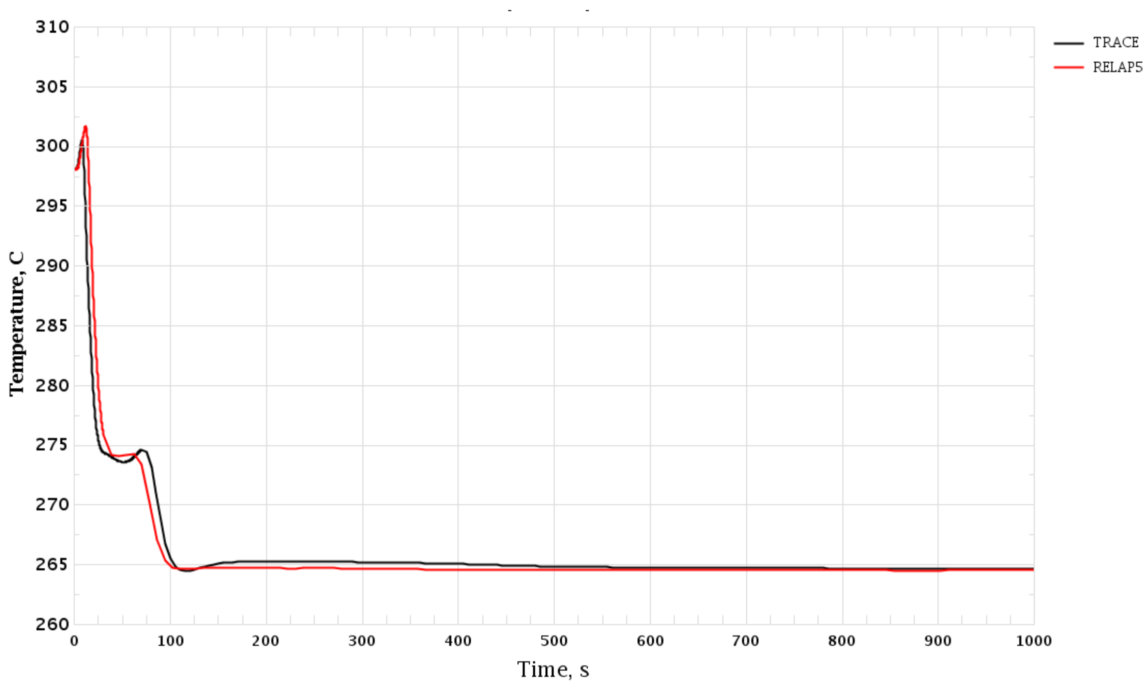

Figure 3-107 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 6

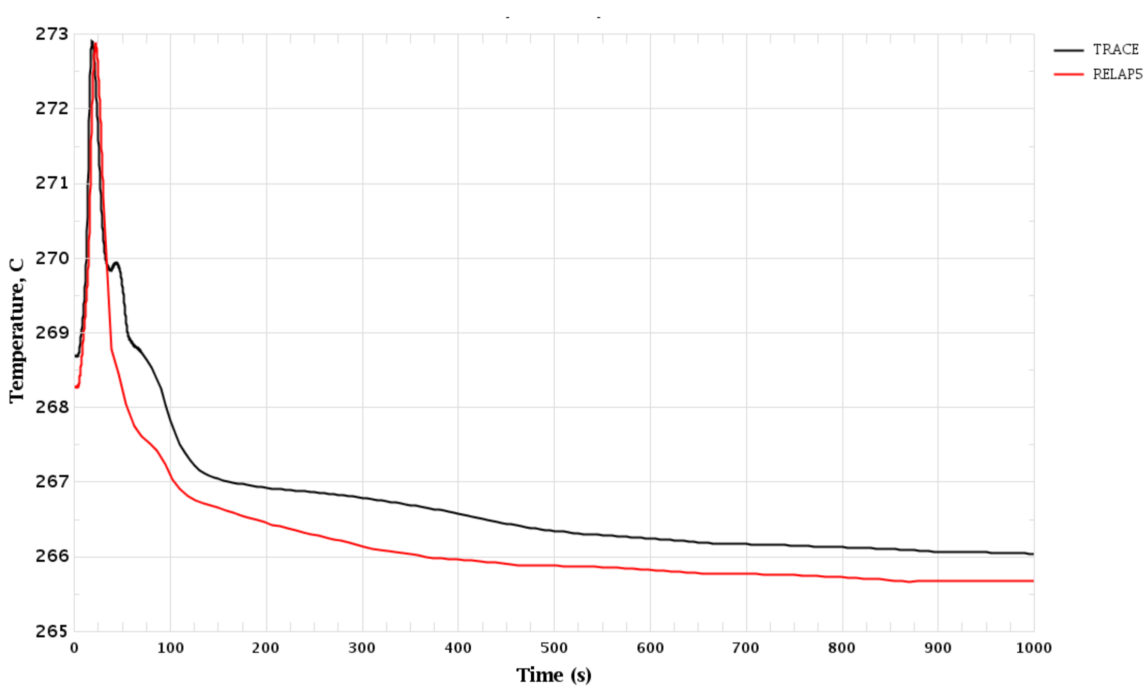

Figure 3-108 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 1

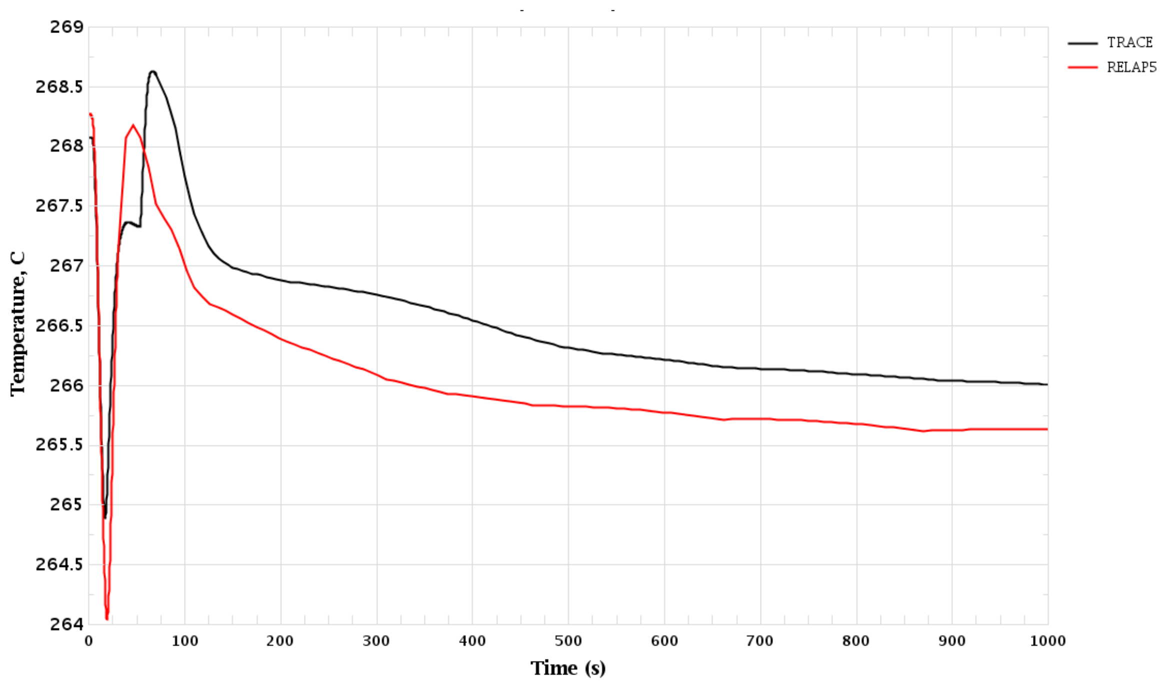

Figure 3-109 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 2

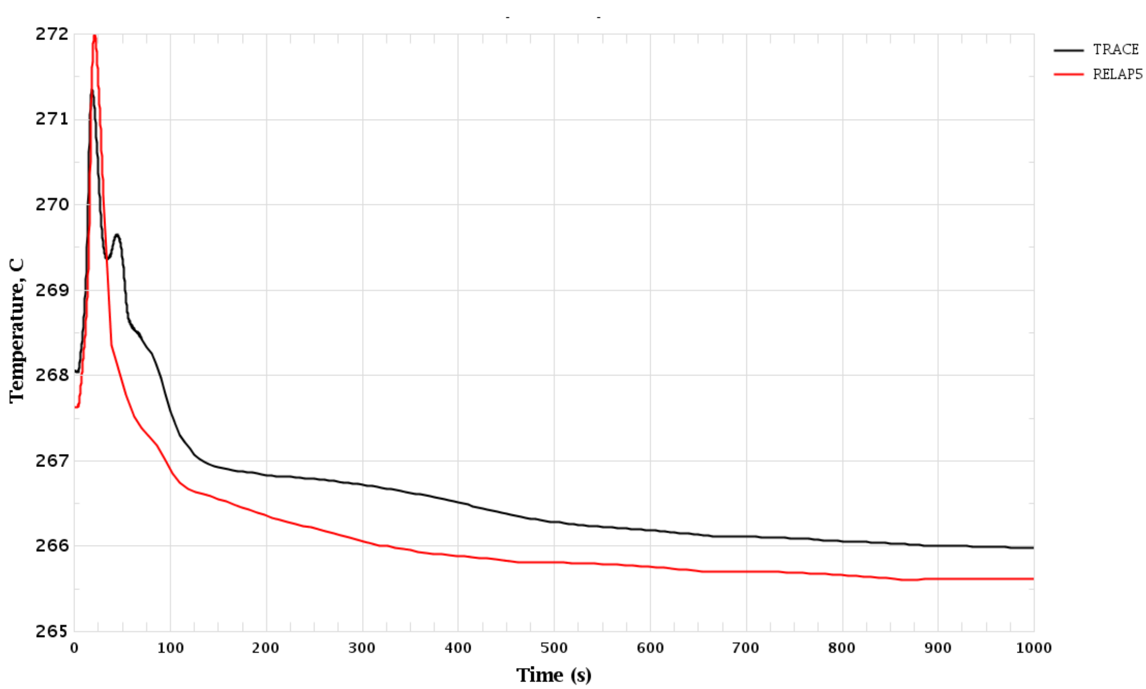

Figure 3-110 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 3

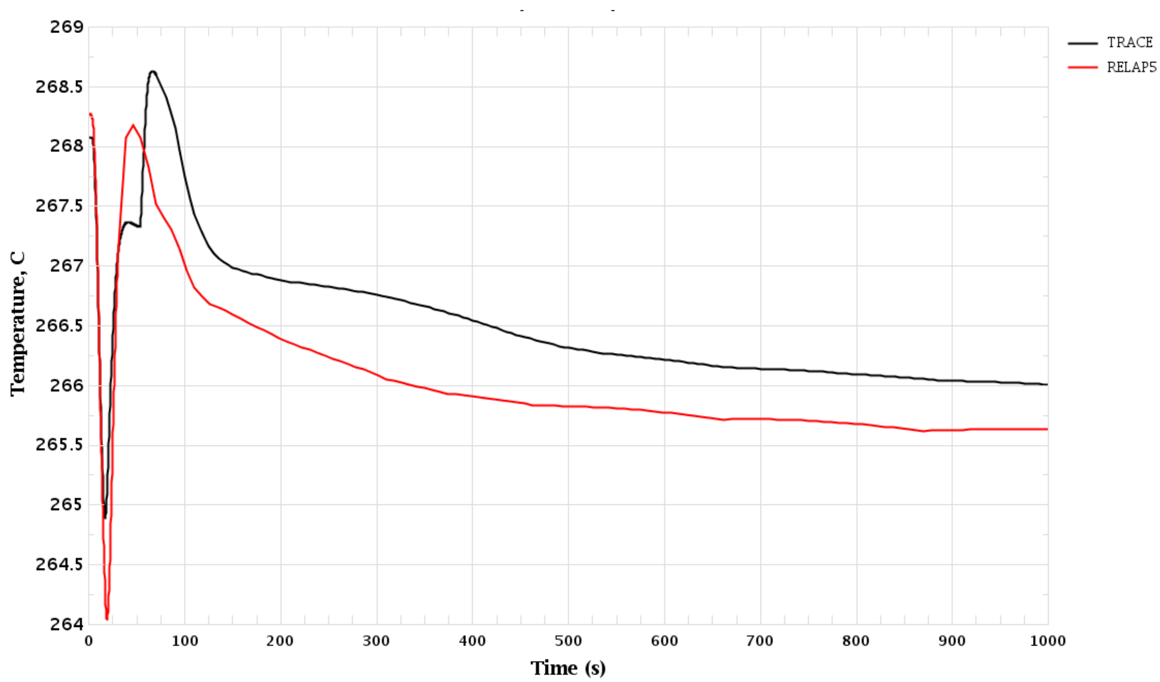

Figure 3-111 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 4

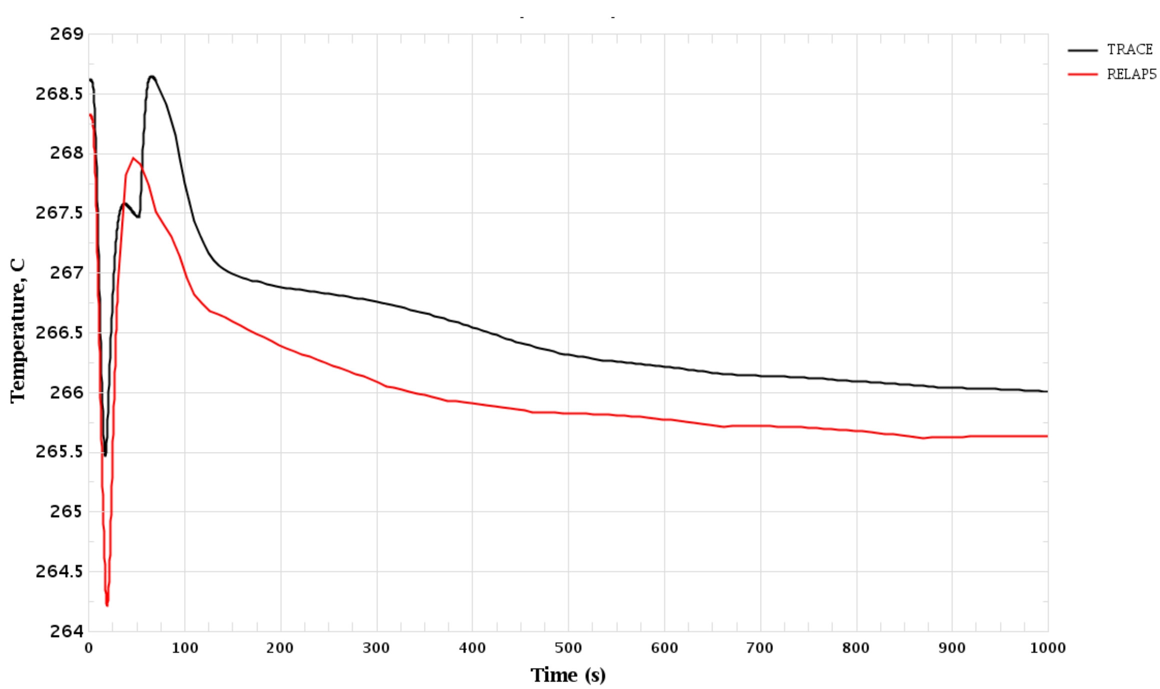

Figure 3-112 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 5

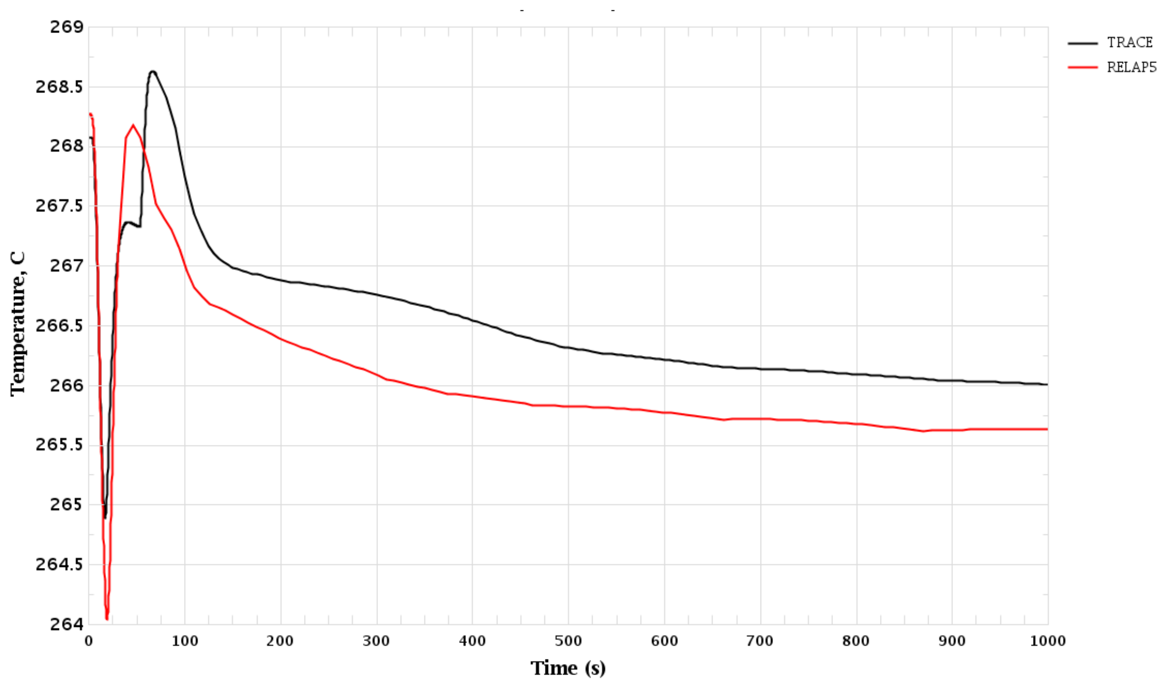

Figure 3-113 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 6

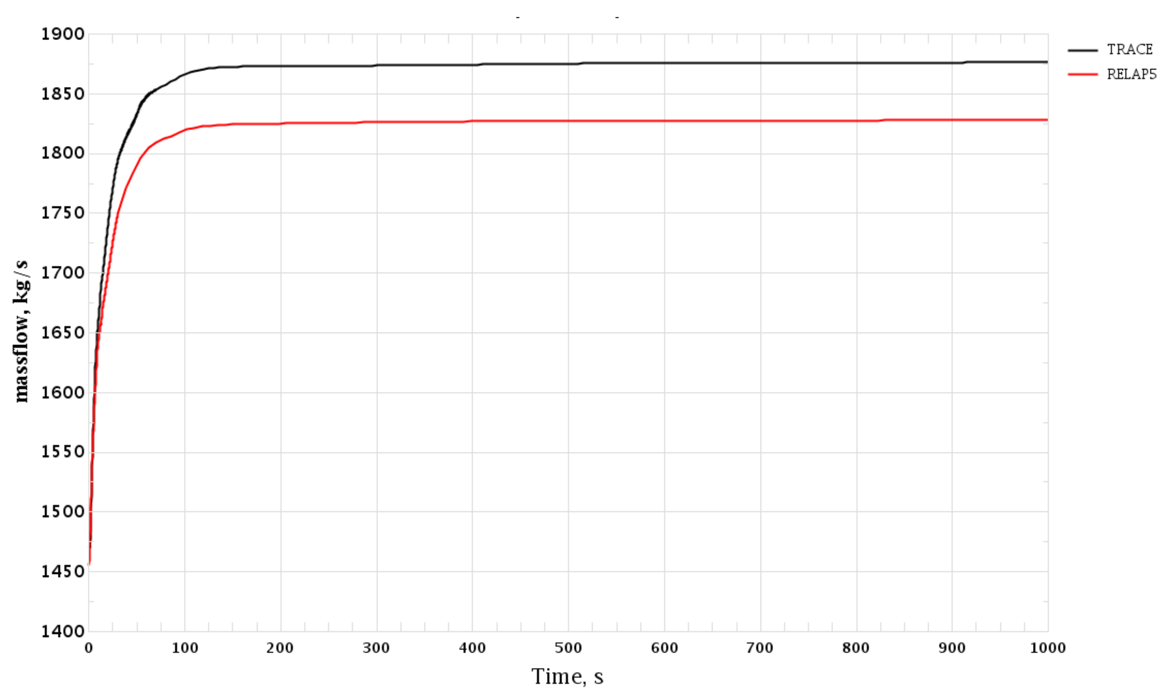

Figure 3-114 Trip of 4/6 RCPs. RCS Loop 1 Mass Flow Rate

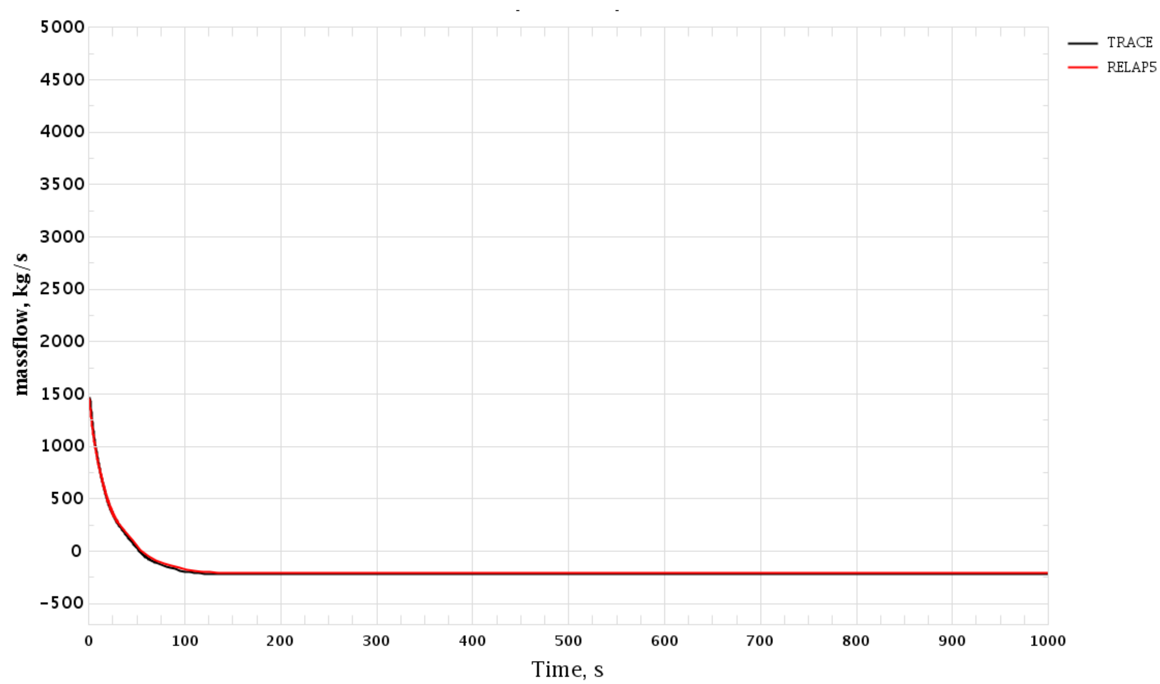

Figure 3-115 Trip of 4/6 RCPs. RCS Loop 2 Mass Flow Rate

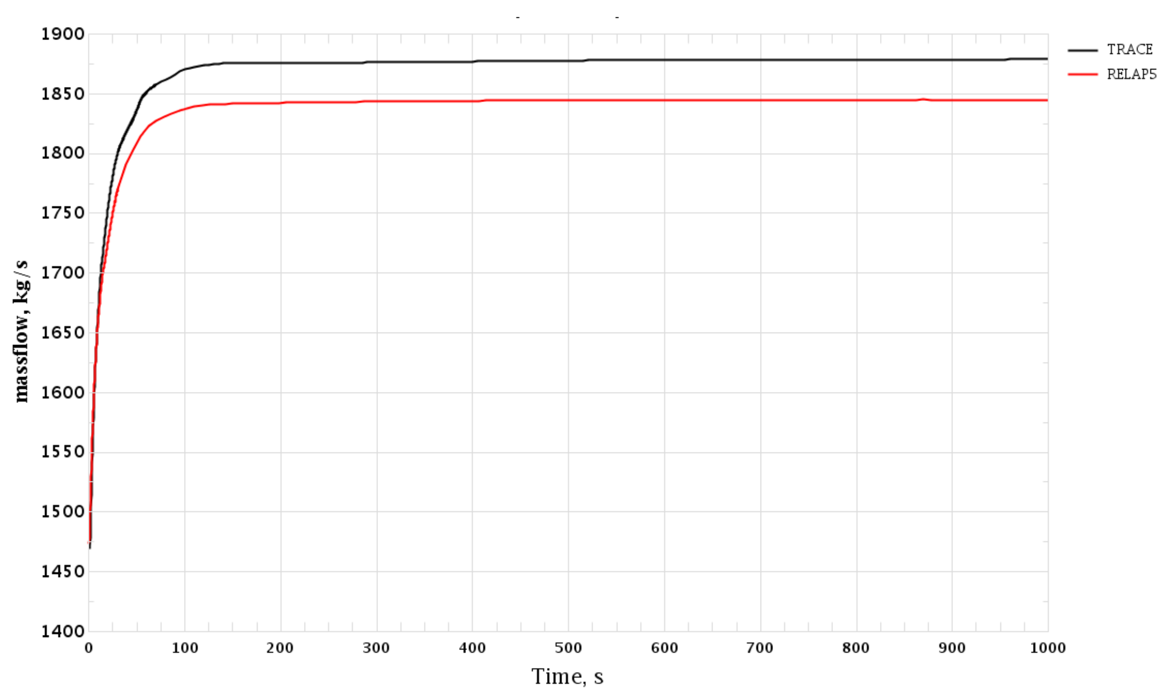

Figure 3-116 Trip of 4/6 RCPs. RCS Loop 3 Mass Flow Rate

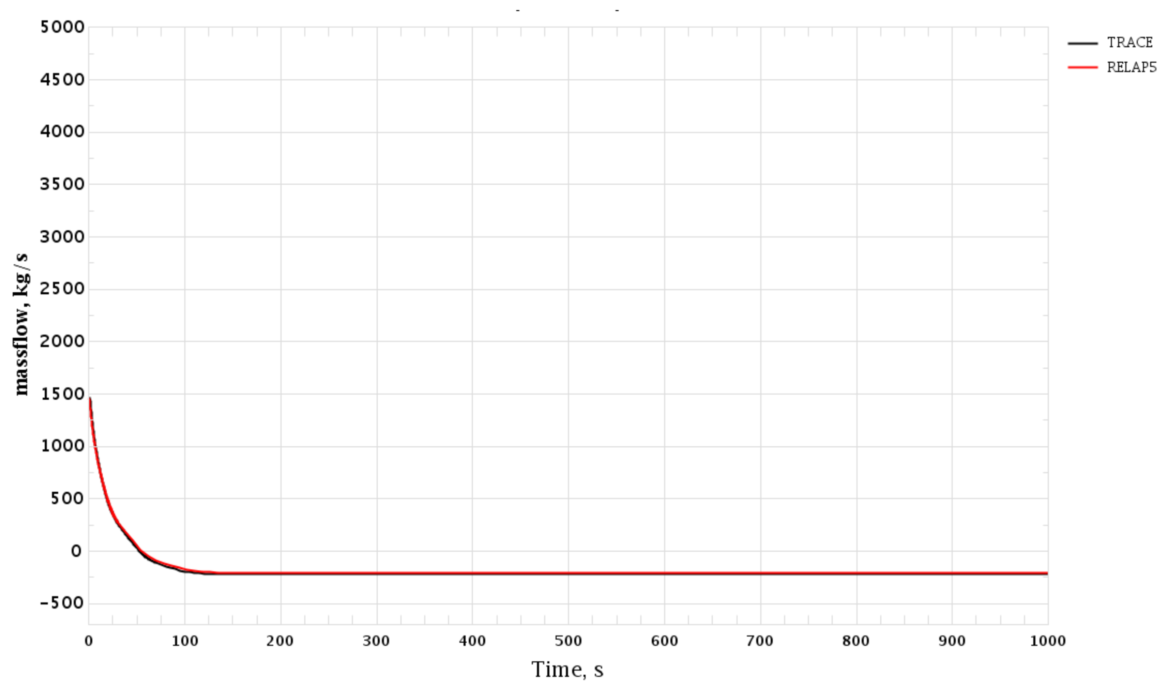

Figure 3-117 Trip of 4/6 RCPs. RCS Loop 4 Mass Flow Rate

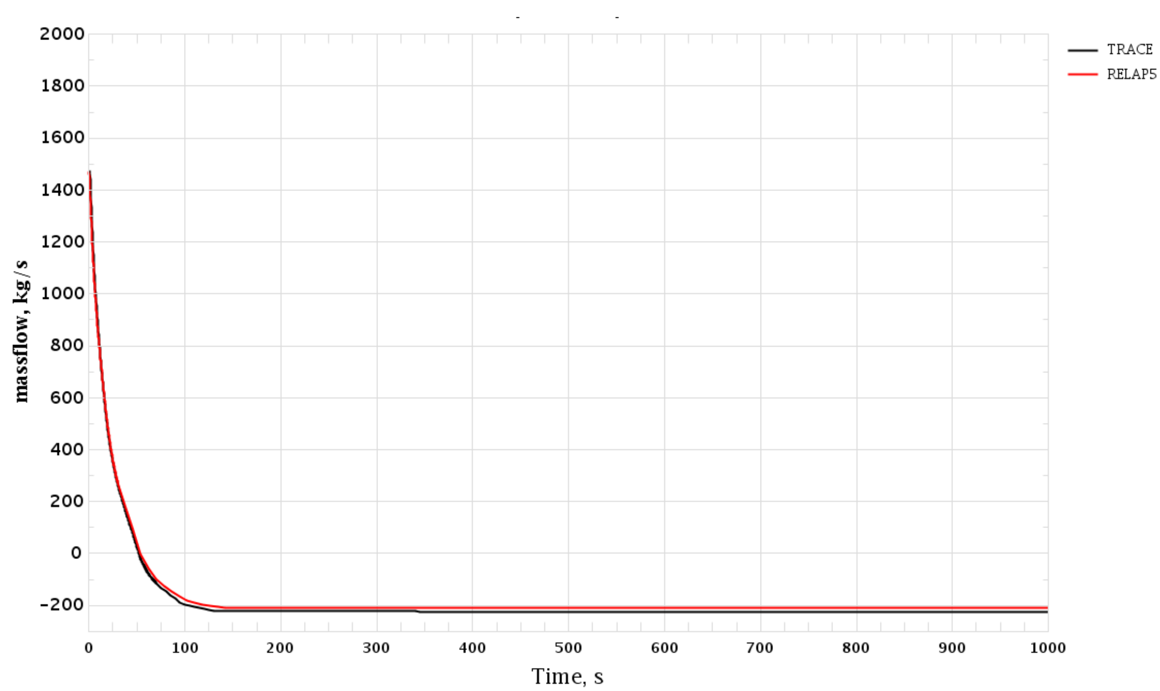

Figure 3-118 Trip of 4/6 RCPs. RCS Loop 5 Mass Flow Rate

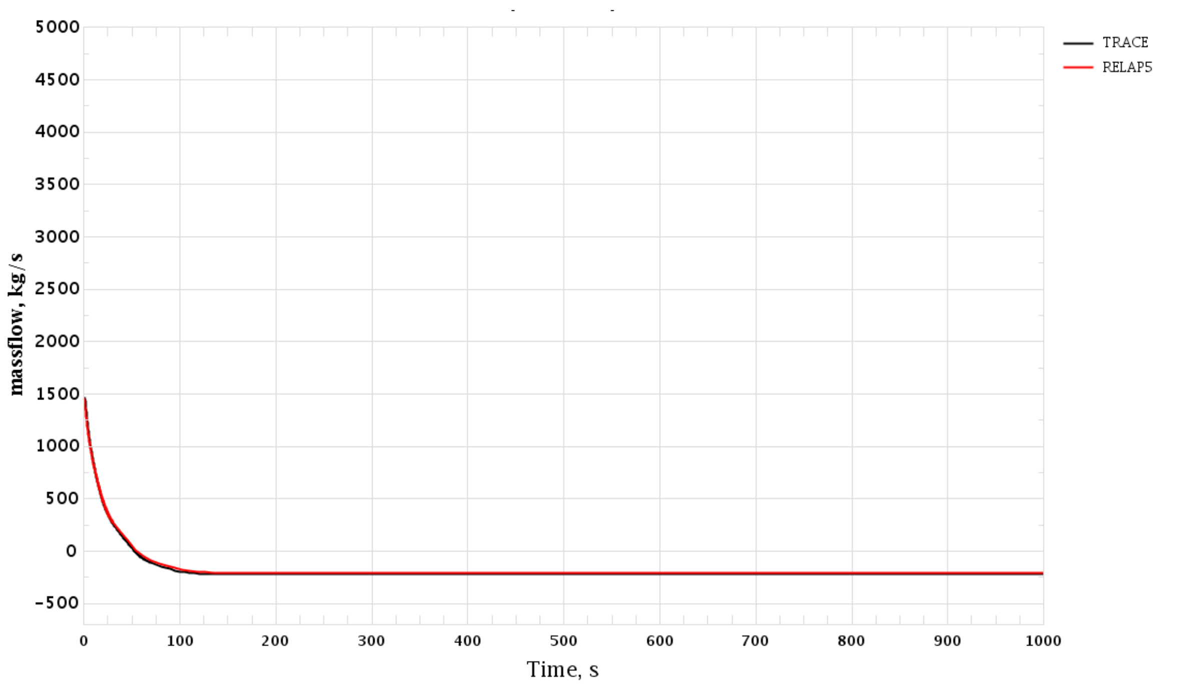

Figure 3-119 Trip of 4/6 RCPs. RCS Loop 6 Mass Flow Rate

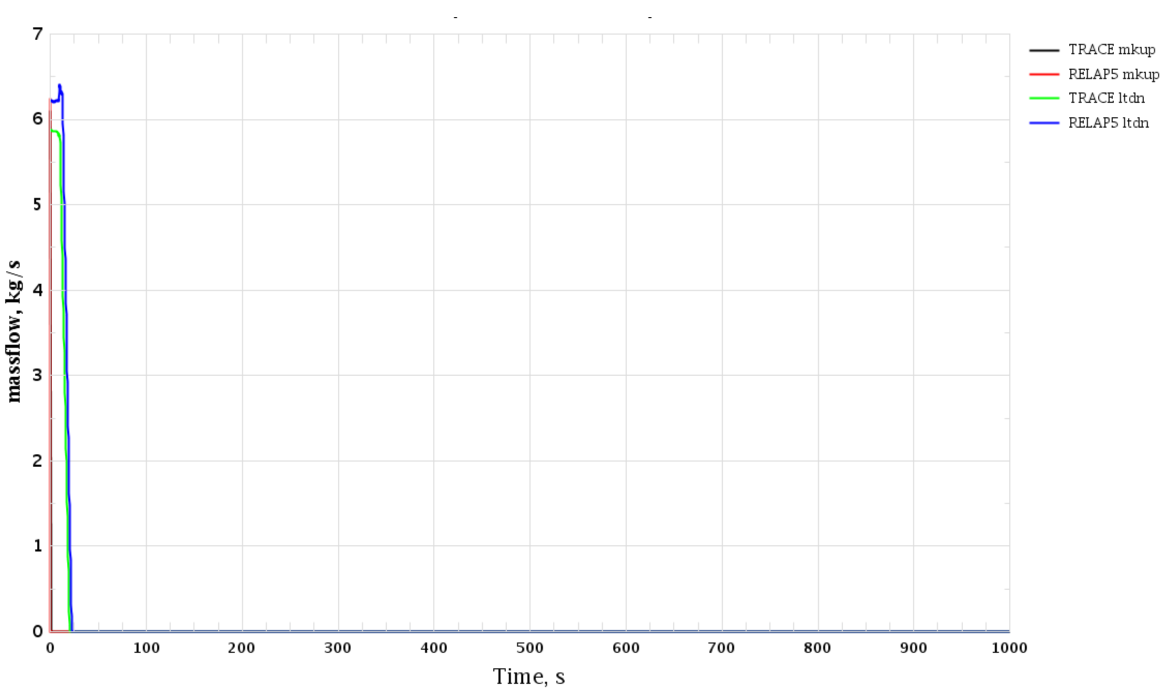

Figure 3-120 Trip of 4/6 RCPs. Make-Up and Let-Down Mass Flow

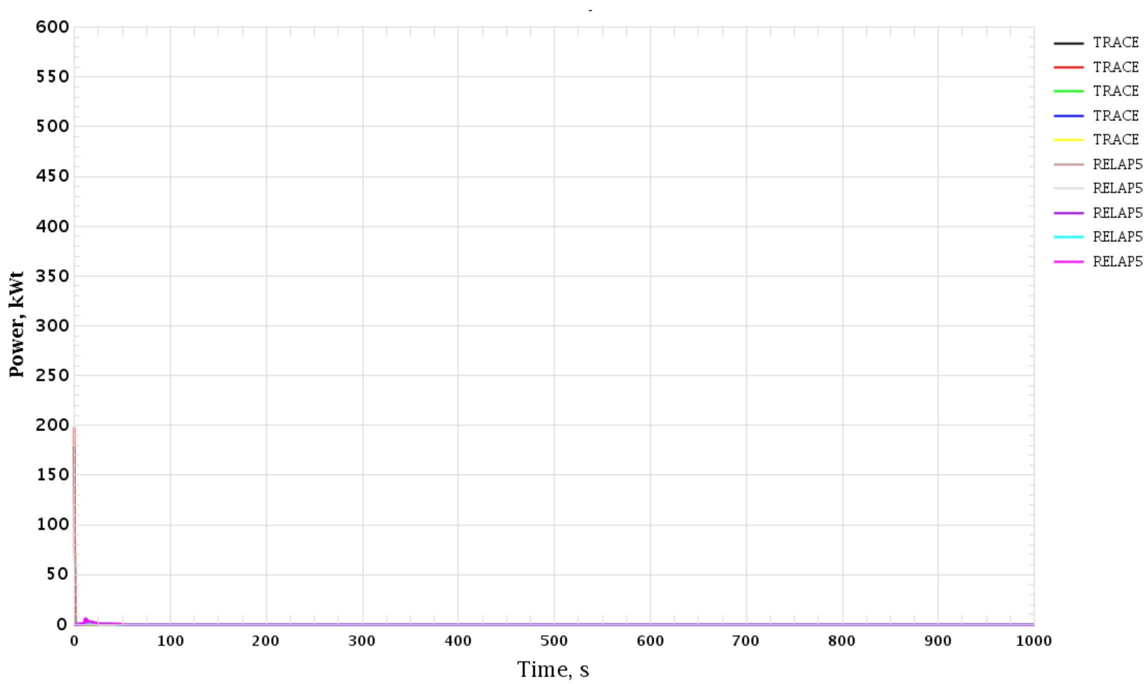

Figure 3-121 Trip of 4/6 RCPs. PRZ Heaters Power

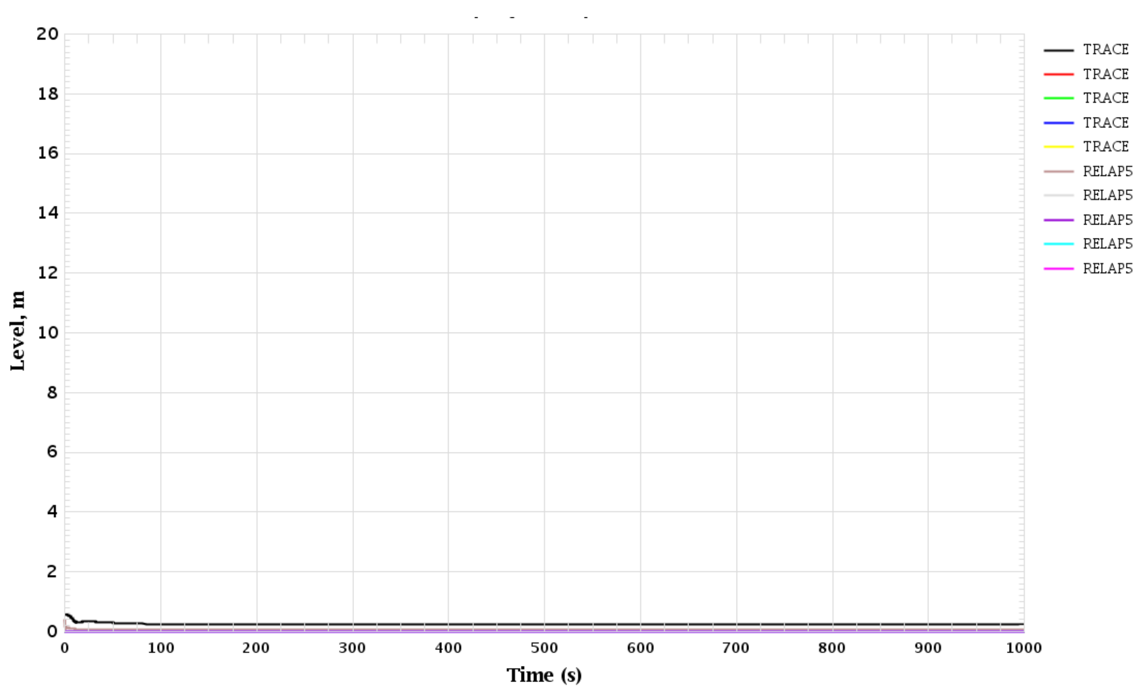

Figure 3-122 Trip of 4/6 RCPs. PRZ Spray Mass Flow Rate

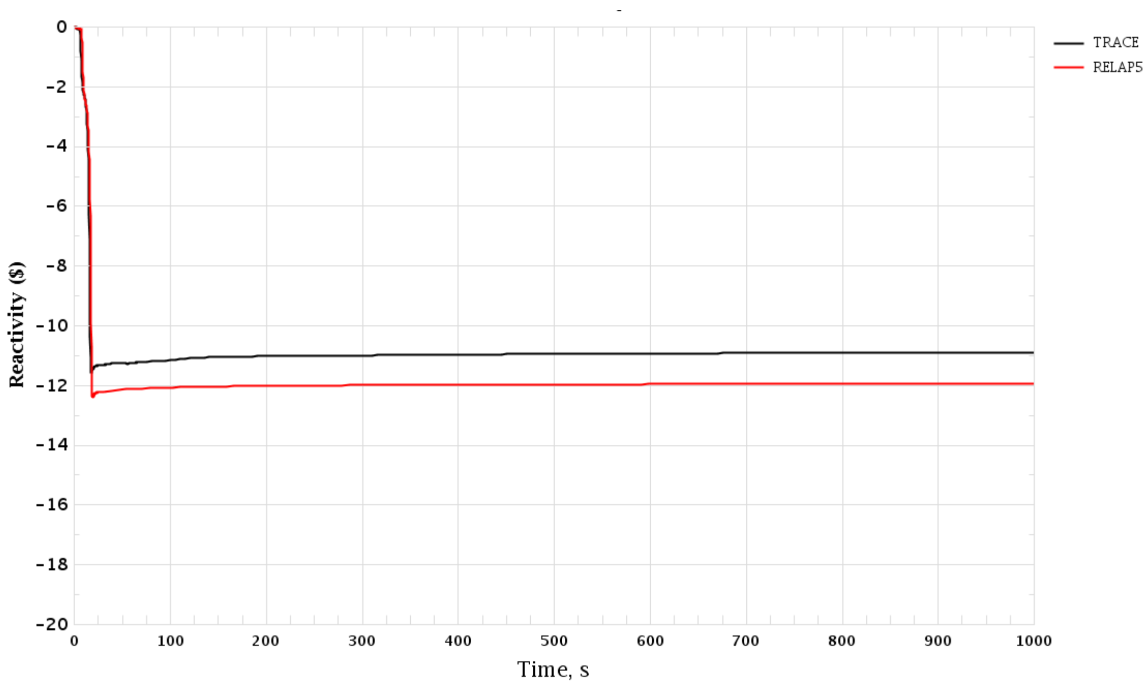

Figure 3-123 Trip of 4/6 RCPs. Core Reactivity

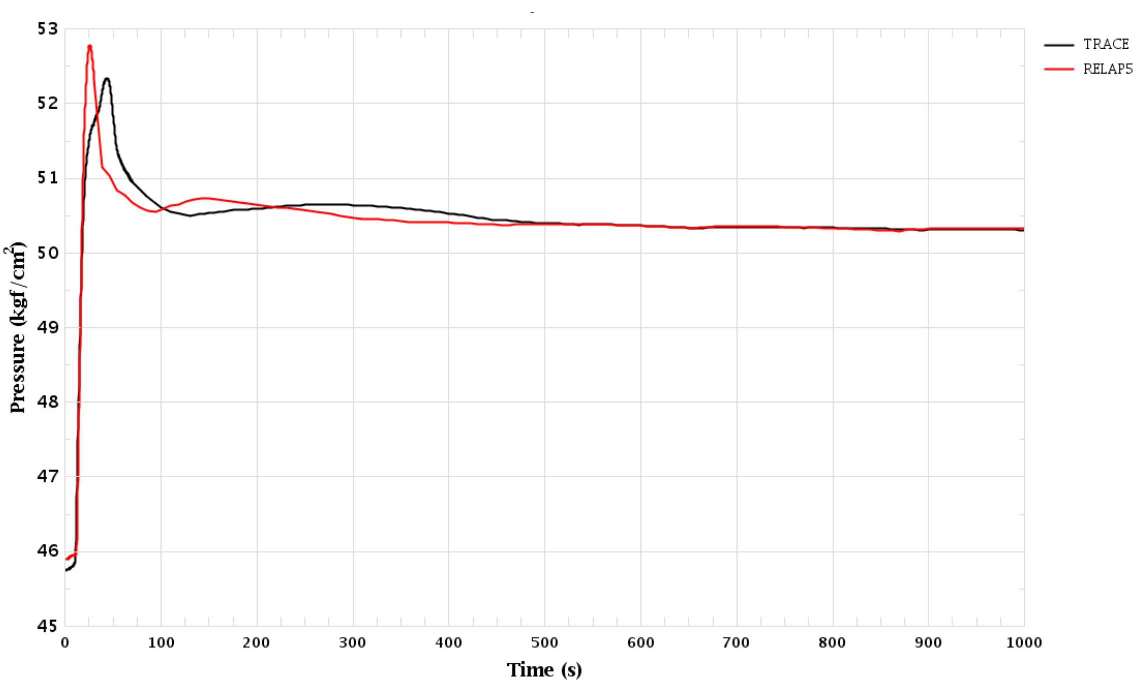

Figure 3-124 Trip of 4/6 RCPs. SG-1 Pressure

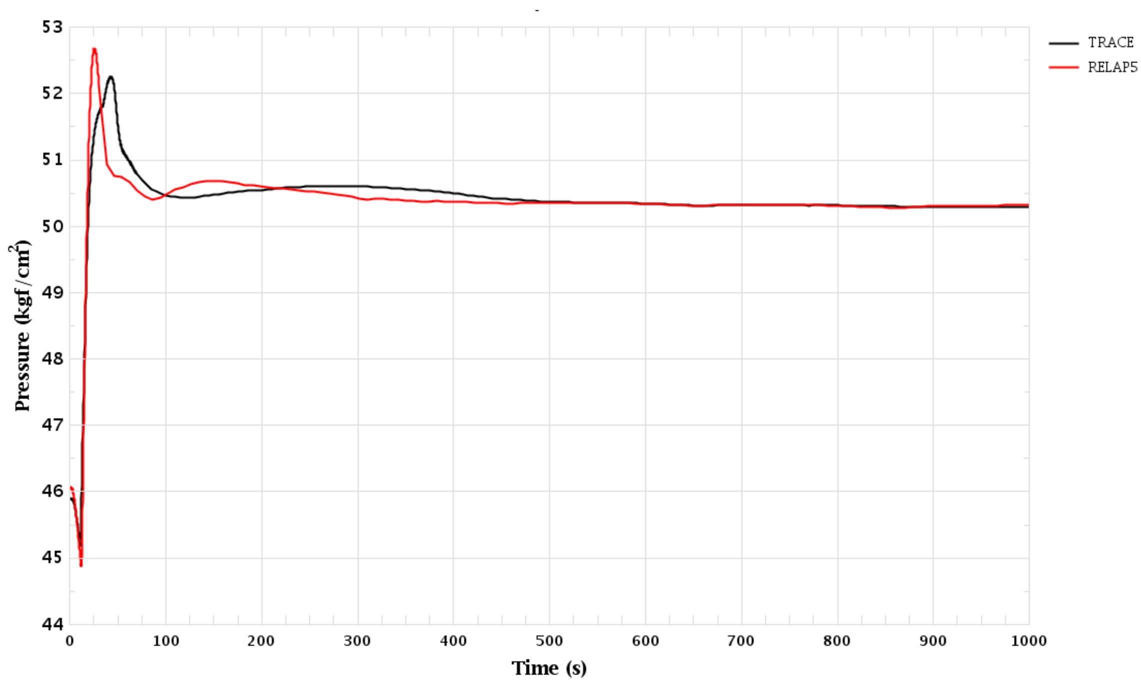

Figure 3-125 Trip of 4/6 RCPs. SG-2 Pressure

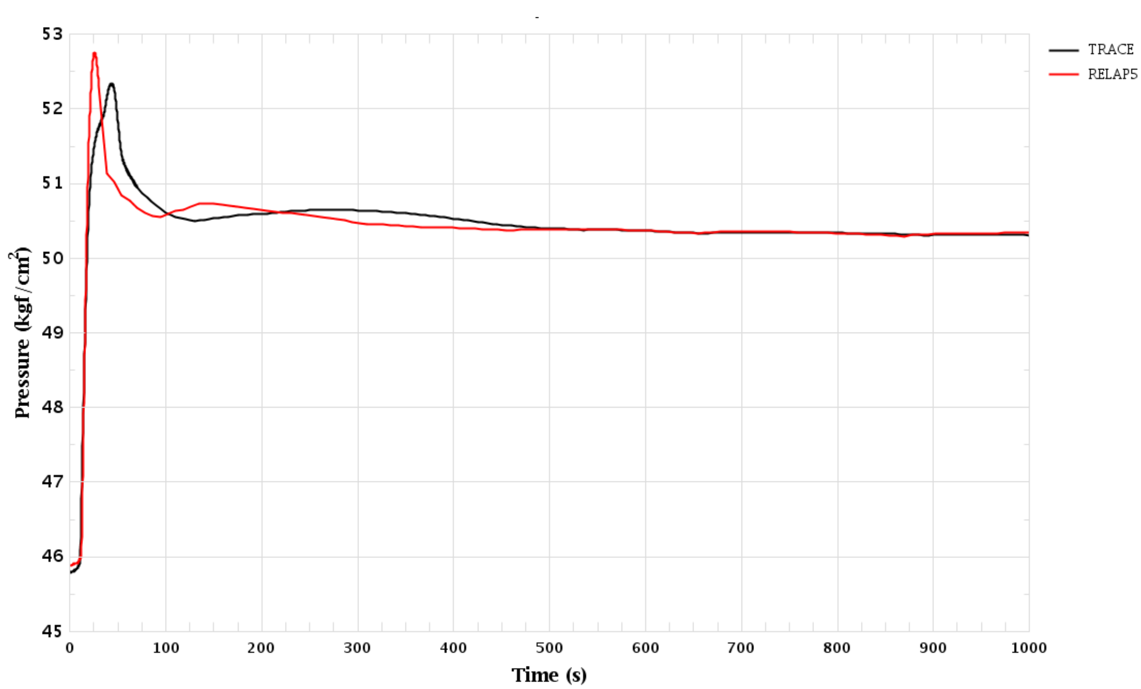

Figure 3-126 Trip of 4/6 RCPs. SG-3 Pressure

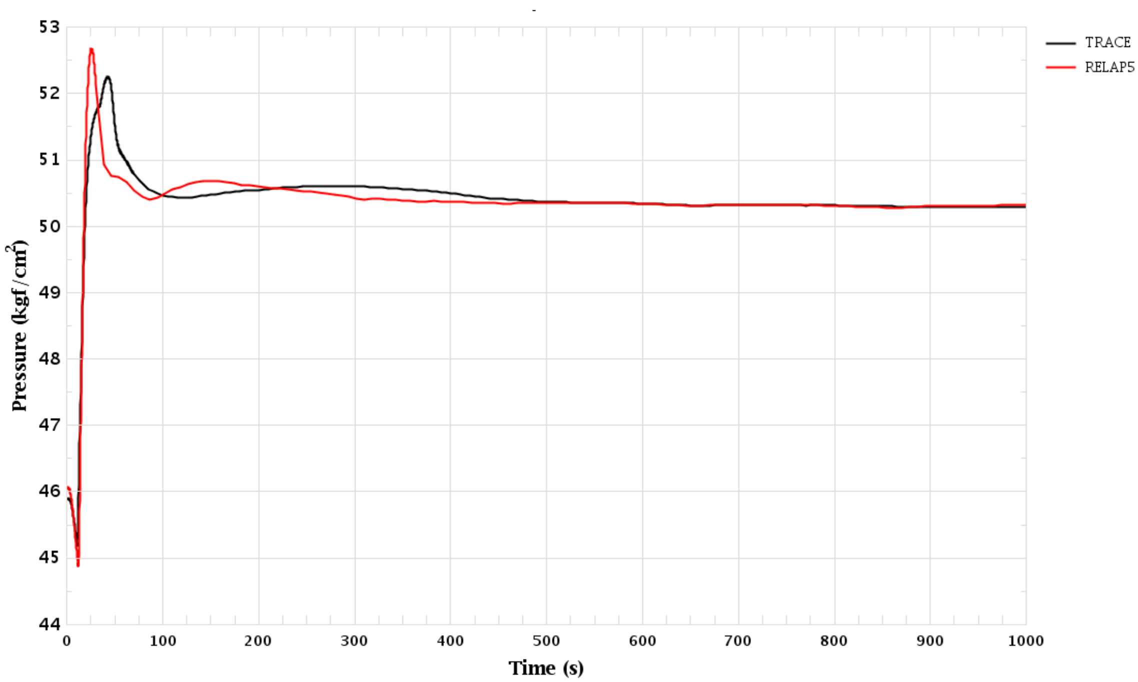

Figure 3-127 Trip of 4/6 RCPs. SG-4 Pressure

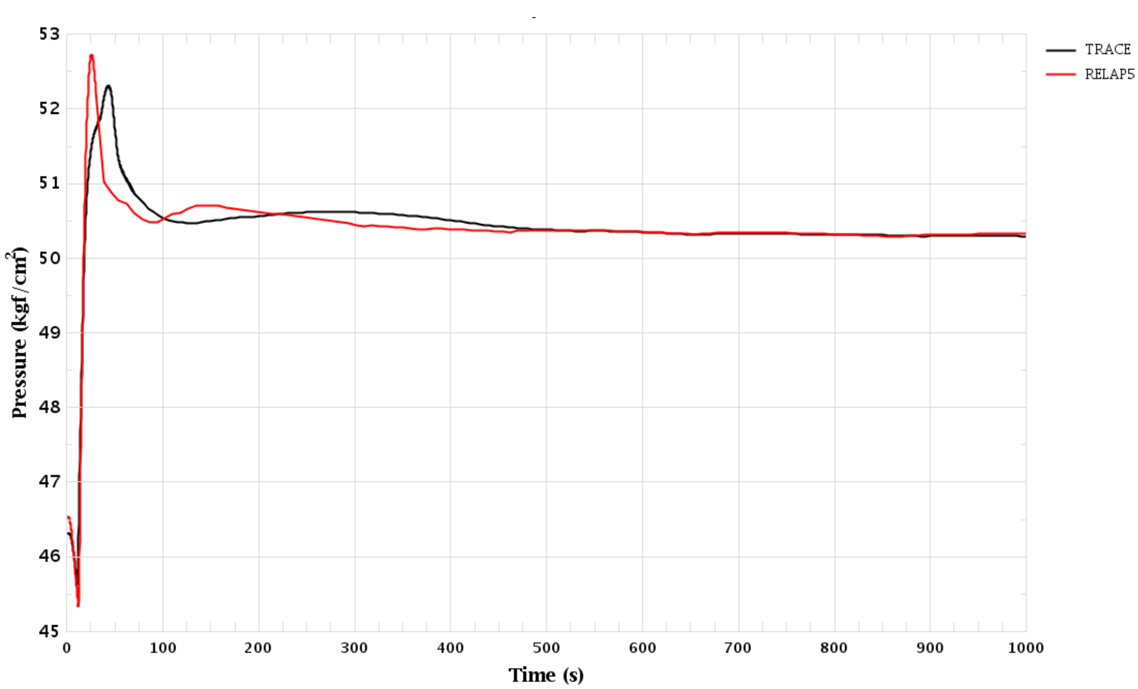

Figure 3-128 Trip of 4/6 RCPs. SG-5 Pressure

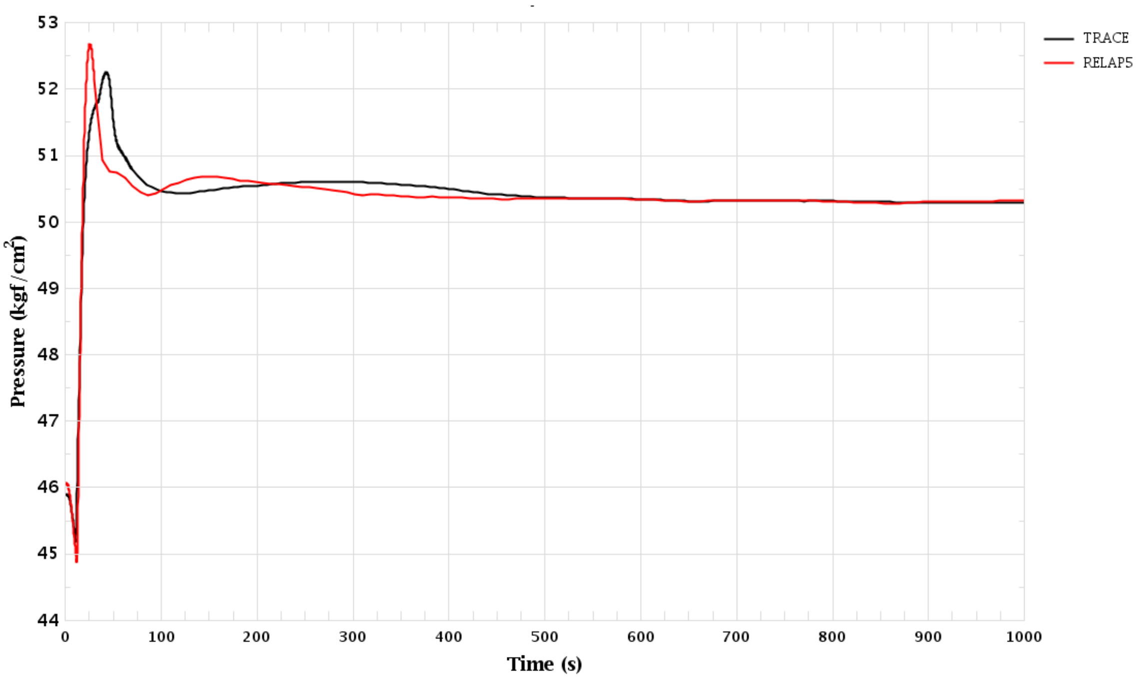

Figure 3-129 Trip of 4/6 RCPs. SG-6 Pressure

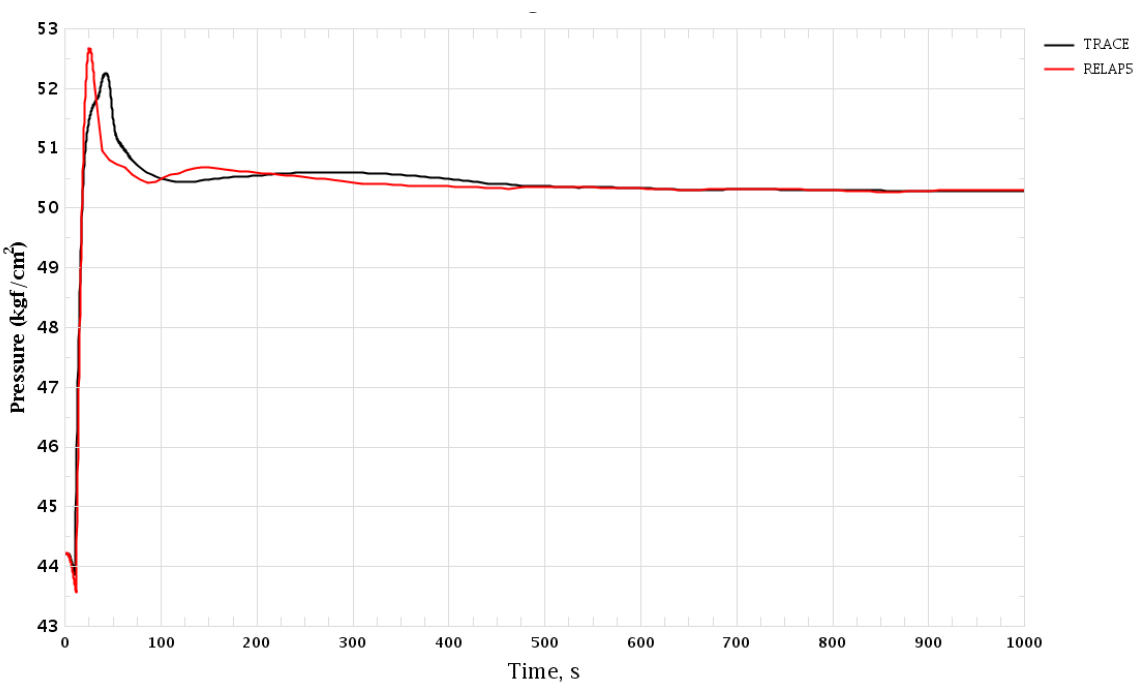

Figure 3-130 Trip of 4/6 RCPs. MSH-1 Pressure

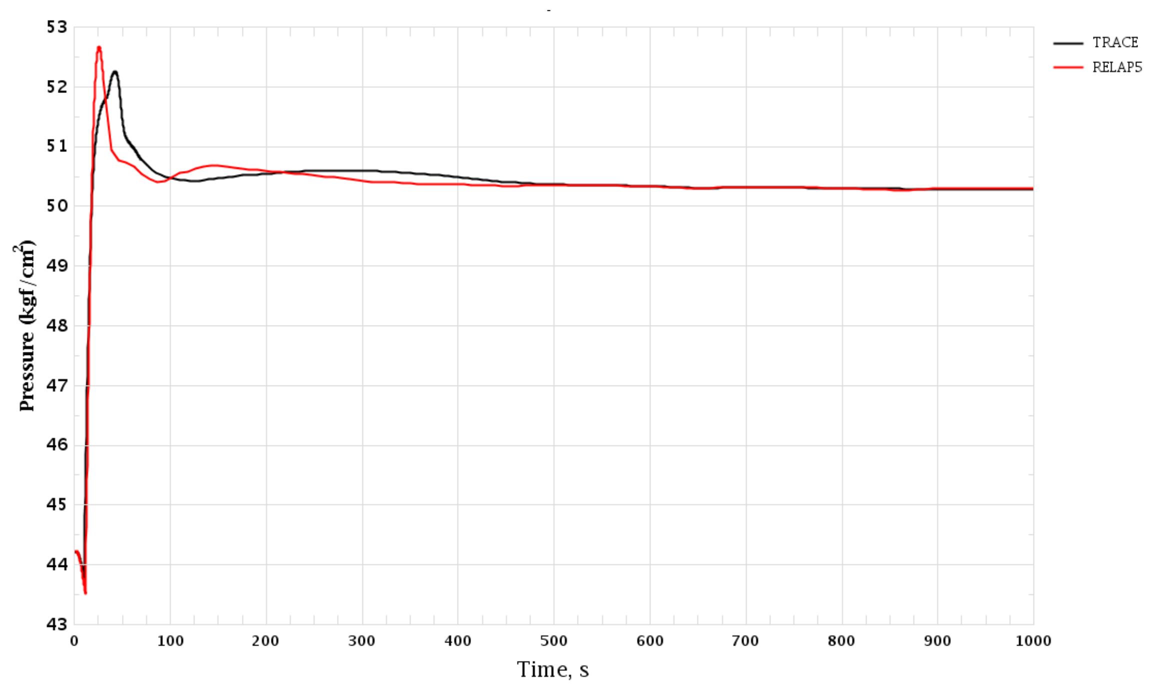

Figure 3-131 Trip of 4/6 RCPs. MSH-2 Pressure

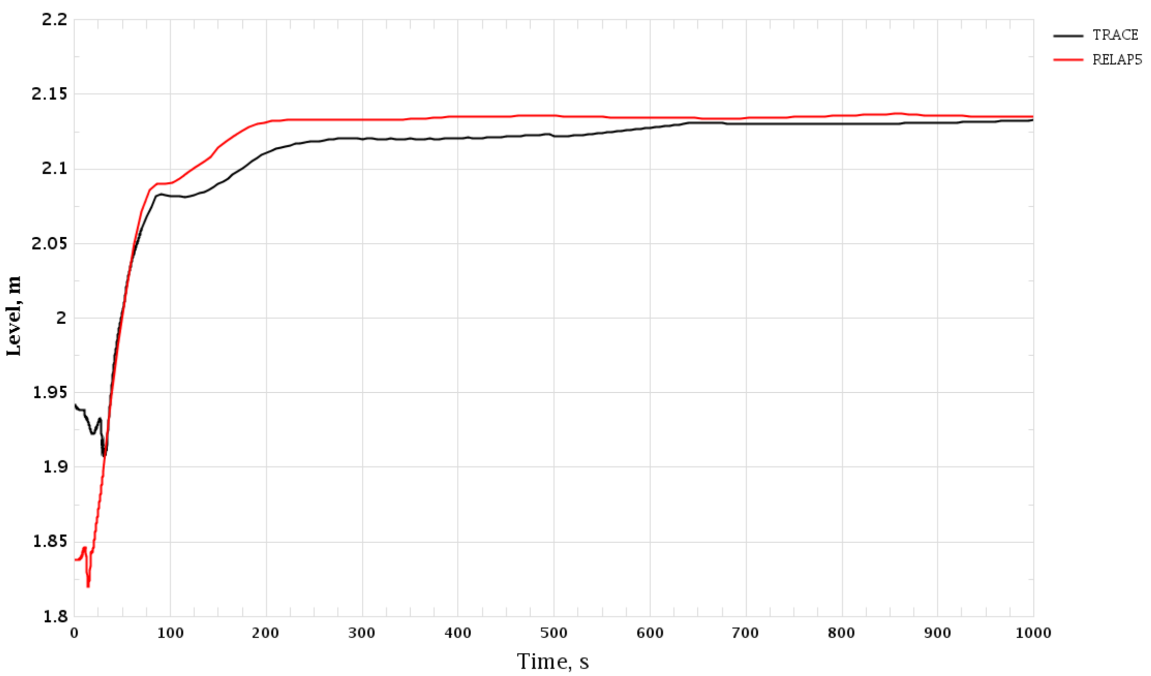

Figure 3-132 Trip of 4/6 RCPs. SG-1 Level (Wide Range)

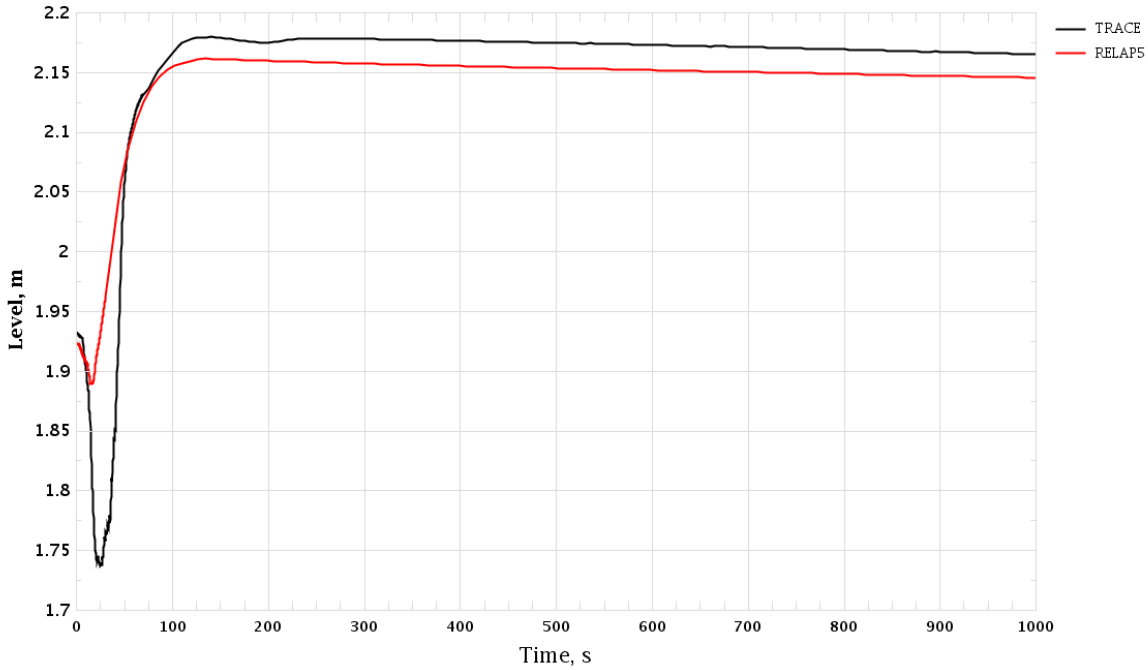

Figure 3-133 Trip of 4/6 RCPs. SG-2 Level (Wide Range)

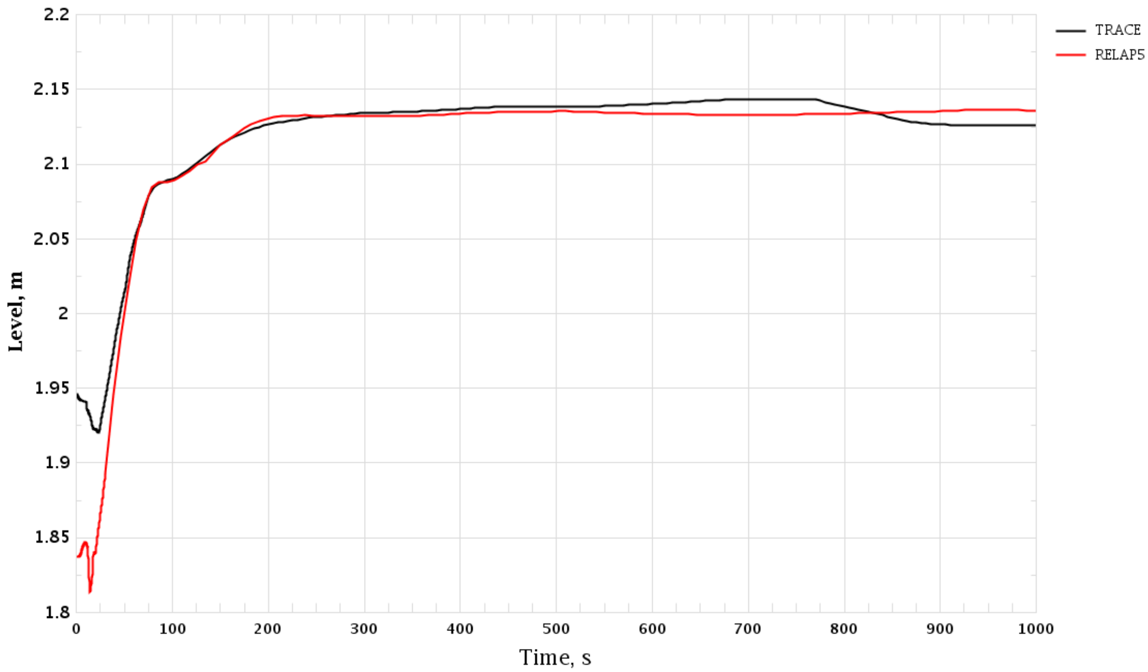

Figure 3-134 Trip of 4/6 RCPs. SG-3 Level (Wide Range)

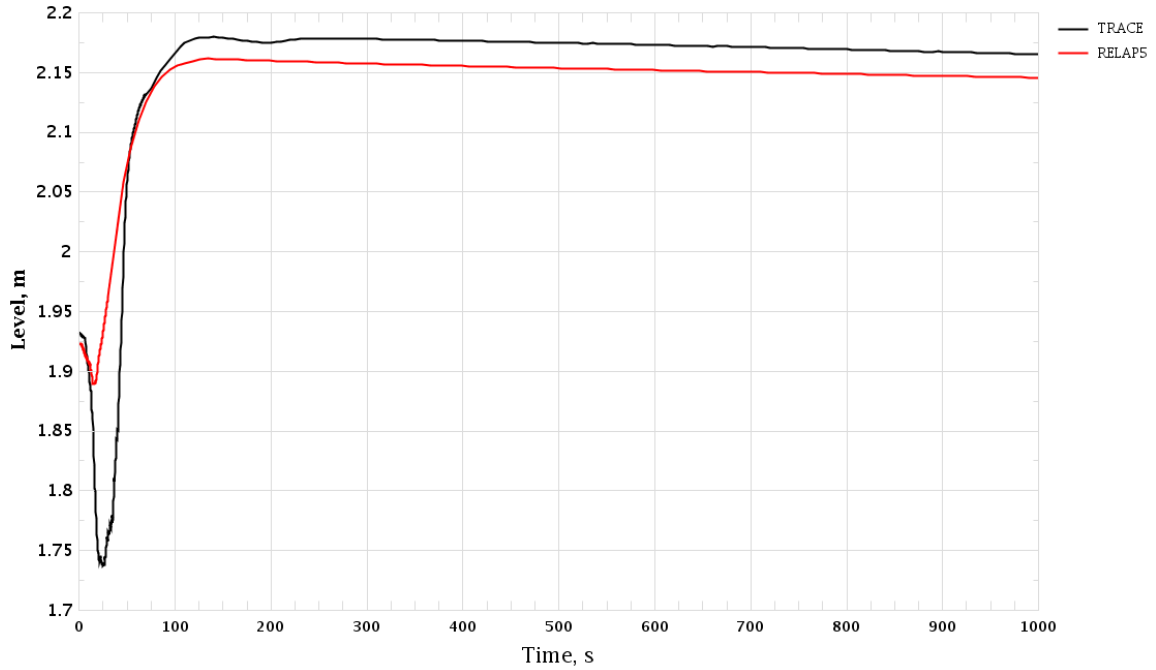

Figure 3-135 Trip of 4/6 RCPs. SG-4 Level (Wide Range)

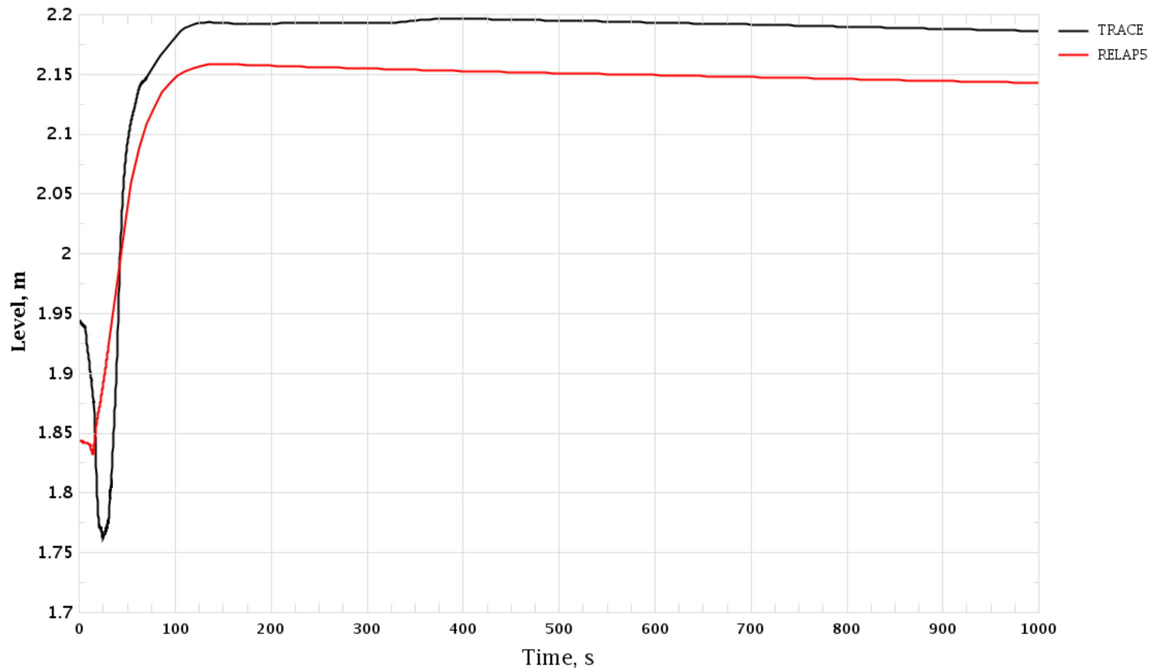

Figure 3-136 Trip of 4/6 RCPs. SG-5 Level (Wide Range)

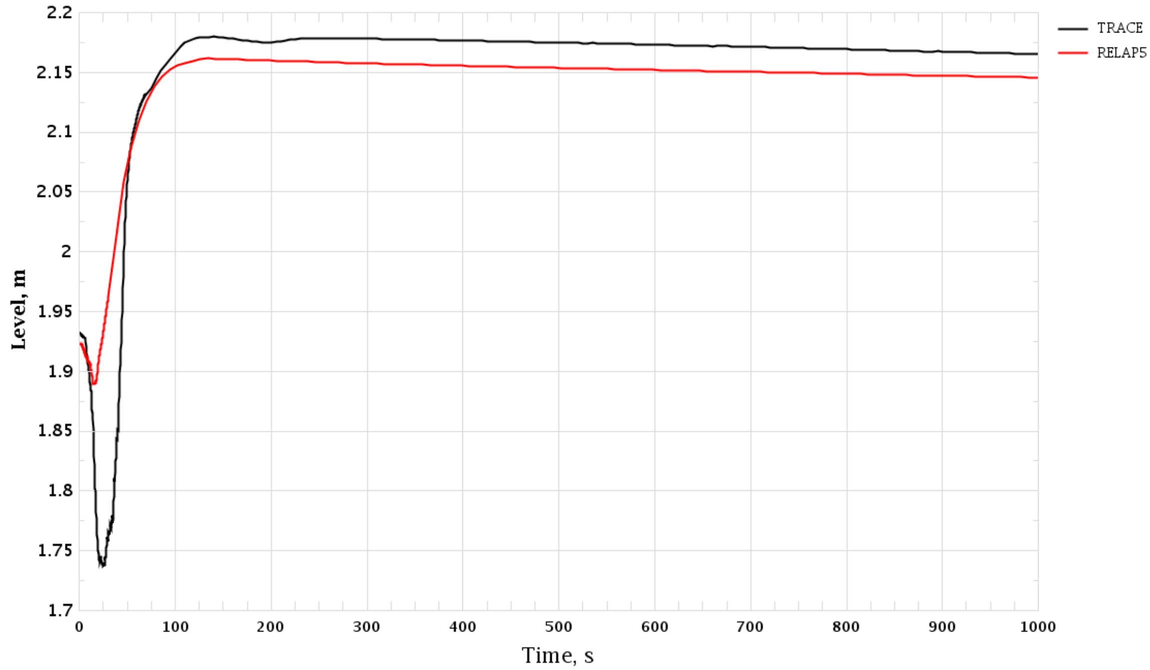

Figure 3-137 Trip of 4/6 RCPs. SG-6 Level (Wide Range)

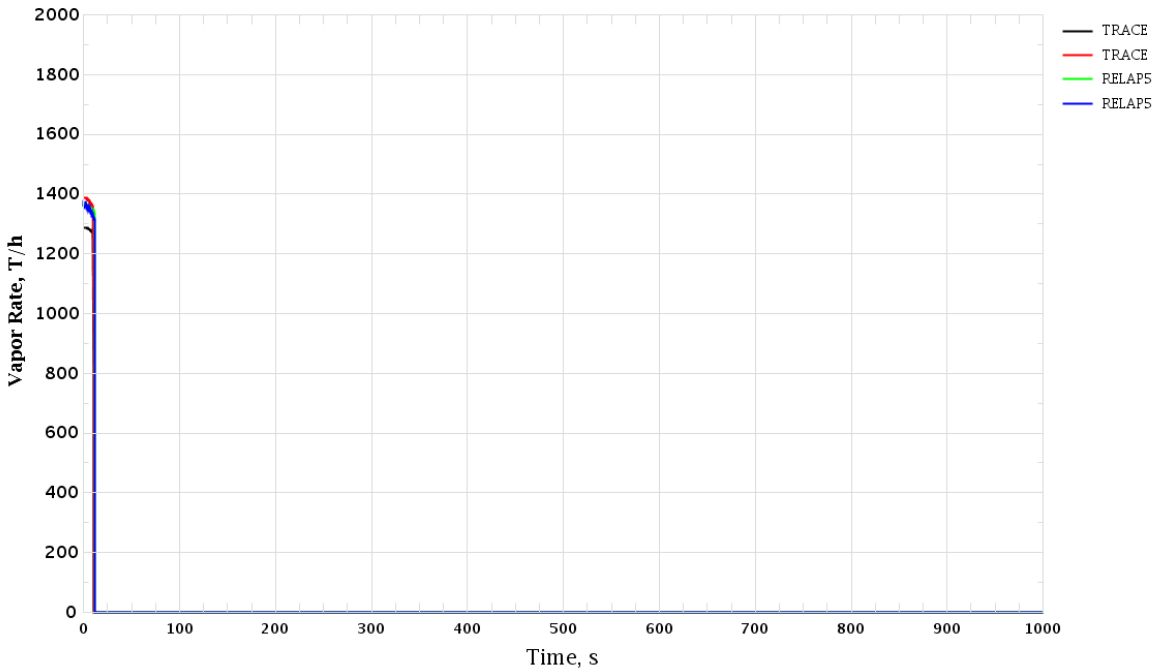

Figure 3-138 Trip of 4/6 RCPs. Turbines Mass Flow Rate

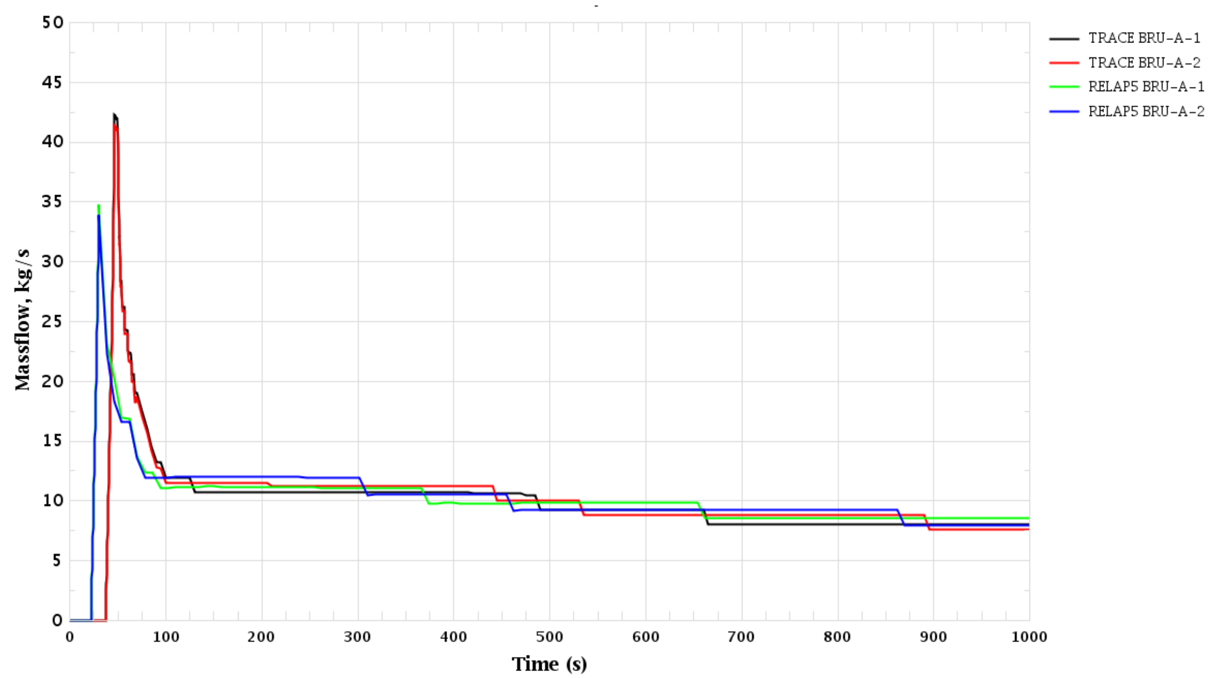

Figure 3-139 Trip of 4/6 RCPs. BRU-A Steam Mass Flow Rate

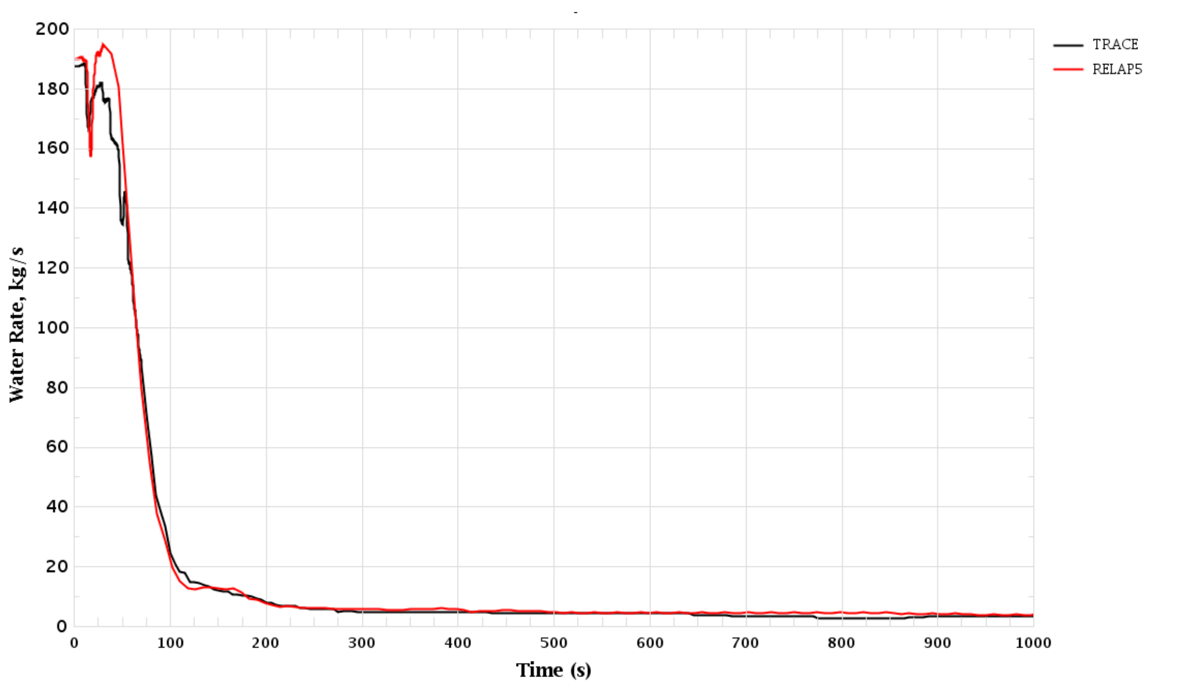

Figure 3-140 Trip of 4/6 RCPs. MFW Pump No.1 Mass Flow Rate

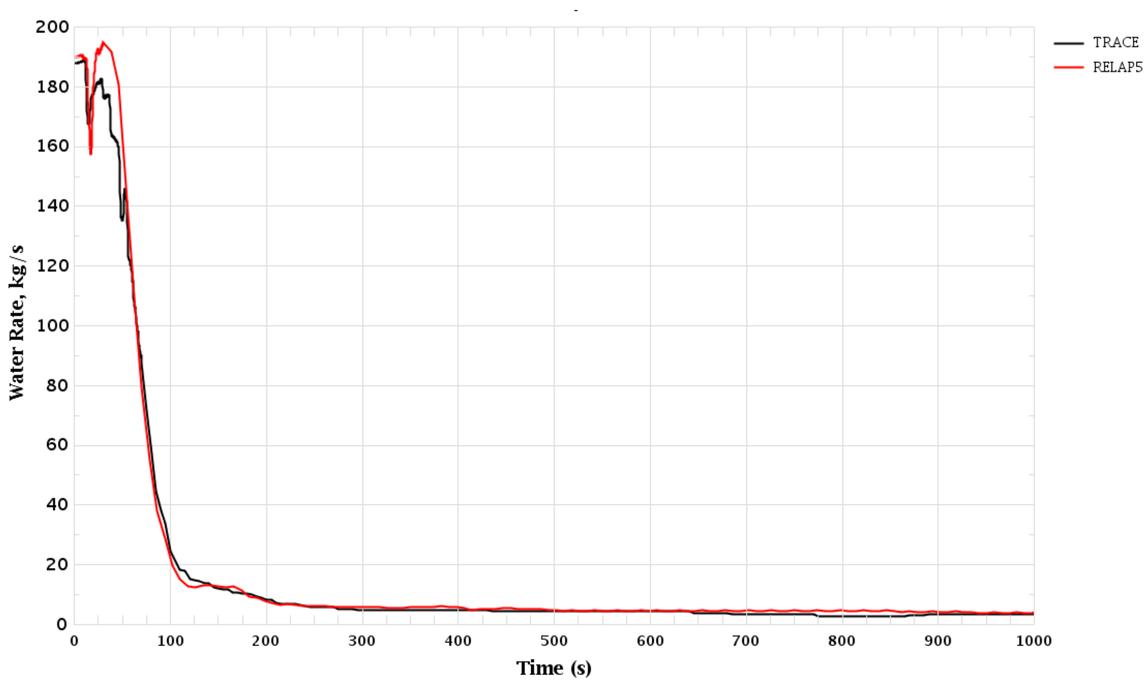

Figure 3-141 Trip of 4/6 RCPs. MFW Pump No.2 Mass Flow Rate

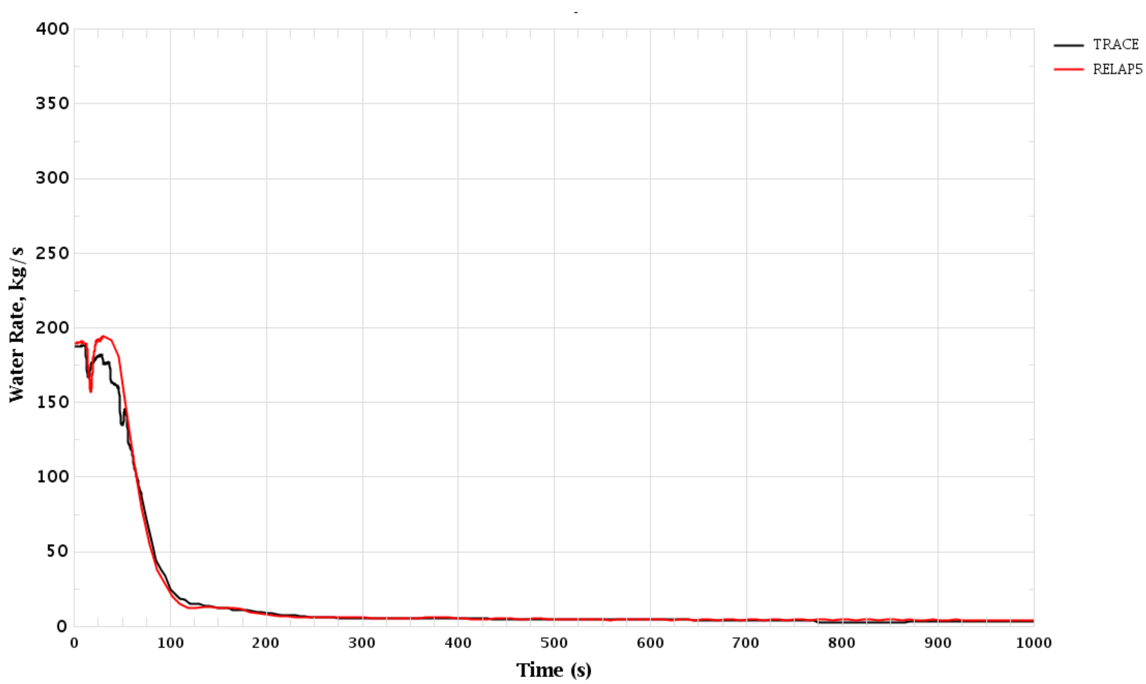

Figure 3-142 Trip of 4/6 RCPs. MFW Pump No.3 Mass Flow Rate

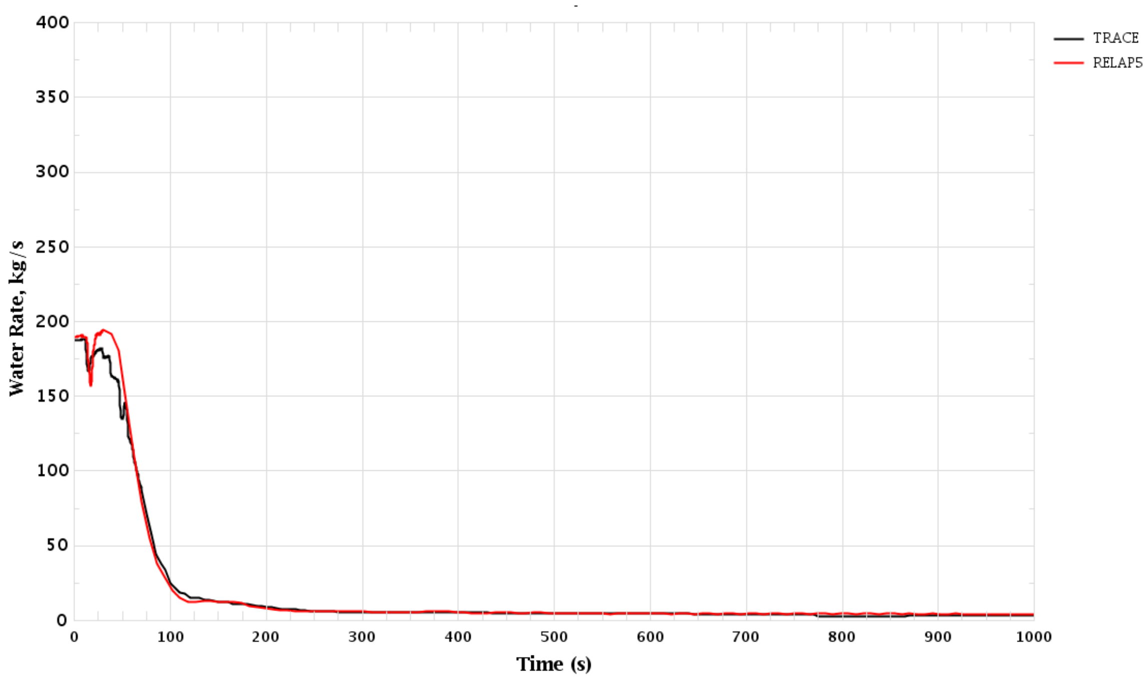

Figure 3-143 Trip of 4/6 RCPs. MFW Pump No.4 Mass Flow Rate

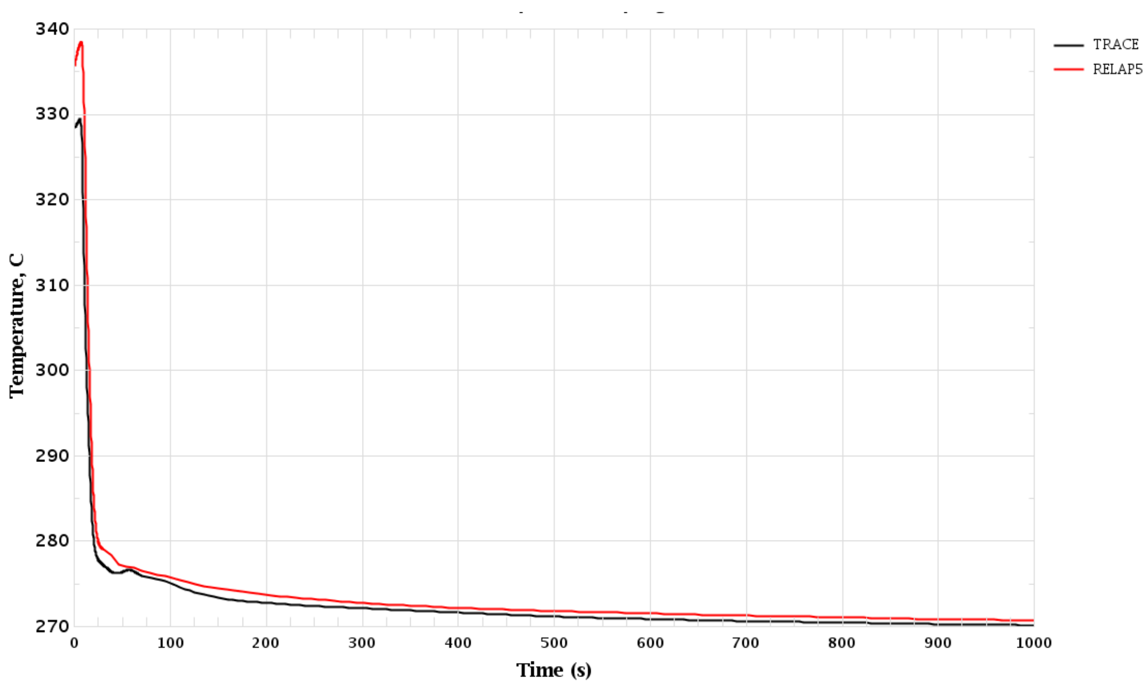

Figure 3-144 Trip of 4/6 RCPs. Maximal Cladding Temperature

3.4 Uncontrolled Withdrawal of Control Assemblies Group

3.4.1 Brief Description of Initiating Event

This initiating event assumes uncontrolled withdrawal of control group of control assemblies from the reactor core with a normal operating speed of 20 mm/s that can be caused by a malfunction of the reactor power control system. According to the expected frequency of occurrence the initiating events is categorized as a postulated transient that leads to an unintended increase of reactor power at the beginning of transient. The IE pertains to the IE group of anomalies in reactivity and power distribution in the reactor core.

3.4.2 Initial Conditions

The initial conditions selected for transient calculation correspond to those specified in Table 3-1.

3.4.3 Boundary Conditions

Assumptions on the systems availability that are considered in calculation of the transient are specified below.

No operator actions are simulated in the scenario.

ARM, RPL, Level 2, Level 3, and Level 4 emergency protection are assumed to be inoperable to allow maximal withdrawal of control group of control assemblies.

3.4.4 Calculation Results

Sequence of events for this transient is presented in Table 3-5.

Table 3-5 Sequence of Events for Uncontrolled Withdrawal of Control Assemblies Group Transient

TRACE Time, s	RELAP5 Time, s	Event	Description
0.0	0.0	Start of uncontrolled withdrawal of control group of control assemblies	
4.0	2.5	Reactor scram	Reactor scram due to high reactor power (>110%) signal actuation
9.0	7.5	Closure of stop valves of both turbines	Turbine trip due to scram with 5.0 s delay
10.0-240.0	10.0-240.0	Operation of PRZ heaters groups	Maintenance of RCS pressure according to design setpoints
14.0	15.0	Maximal SG pressure is 51.9-52.0 kgf/cm^2	
15.0	15.0	Opening of BRU-K due to pressure increase in correspondent MSH semi-header	Automatic operation of BRU-K controllers in the pressure maintenance mode
-	15.0-27.0	Opening of BRU-A1, BRU-A2 due to pressure increase in correspondent MSH semi-header over 52.0 kgf/cm^2	Automatic operation of BRU-A controllers in the pressure maintenance mode
20.0-440.0	20.0-390.0	Full opening of make-up and closure of let-down control valves	PRZ level maintenance
50.0	60.0	Minimal PRZ level is 4.0 m (TRACE) and 4.4 m (RELAP5)	
250.0	220.5	Start of PRZ spray valves operation	Maintenance of RCS pressure according to design setpoints
250.0	230.0	Maximal RCS pressure is 130.5 kgf/cm^2 (TRACE) and 128.0 kgf/cm^2 (RELAP5)	
440.0	420.0	Start of primary pressure decrease	Restoration of nominal RCS pressure by PRZ spray operation
650.0	840.0	Operation of PRZ heaters groups	Maintenance of RCS pressure according to design setpoints
1800.0	1800.0	End of calculation	

Uncontrolled withdrawal of the control group of control assemblies with a normal operating speed of 20 mm/s causes insertion of positive reactivity and thus, the reactor power increase (Figure 3-145).

At 4.0/2.5 s, the reactor neutron power reaches 110.0% of the nominal value (Figure 3-145) that causes scram signal actuation. Increase of coolant temperature (Figure 3-148 – Figure 3-159) at the beginning of transient due to initial reactor power increase is quickly terminated after reactor scram, and the temperature decreases rapidly.

Closure of stop valves of both turbines after the reactor scram causes sharp increase of the secondary circuit pressure (Figure 3-176, Figure 3-177). This causes actuation of BRU-K (Figure 3-185) and BRU-A in pressure maintenance mode that automatically decrease the secondary circuit pressure according to a design algorithm.

Decrease of primary coolant temperature (Figure 3-148 – Figure 3-159) within first 100 s of transient and correspondent coolant shrinkage result in a decrease of PRZ level (Figure 3-147), which is gradually restored at 450 s by operation of PRZ level controllers that adjust make-up and let-down flow (Figure 3-166).

After reaching the minimal value at ~60.0 s RCS pressure (Figure 3-146) starts to recover by PRZ heaters operation (Figure 3-167). At 250.0/220.5 s the primary pressure reaches PRZ spray valves opening setpoints (Figure 3-168), and after slow decrease is maintained close to a nominal value by PRZ heaters operation (Figure 3-167).

Within 1800 s of transient all primary and secondary circuit parameters are stabilized. The decay heat is removed in forced circulation mode by the secondary circuit via BRU-K and MFW pumps operation.

The plots of the main parameters of calculation are presented below on Figure 3-145 – Figure 3-191.

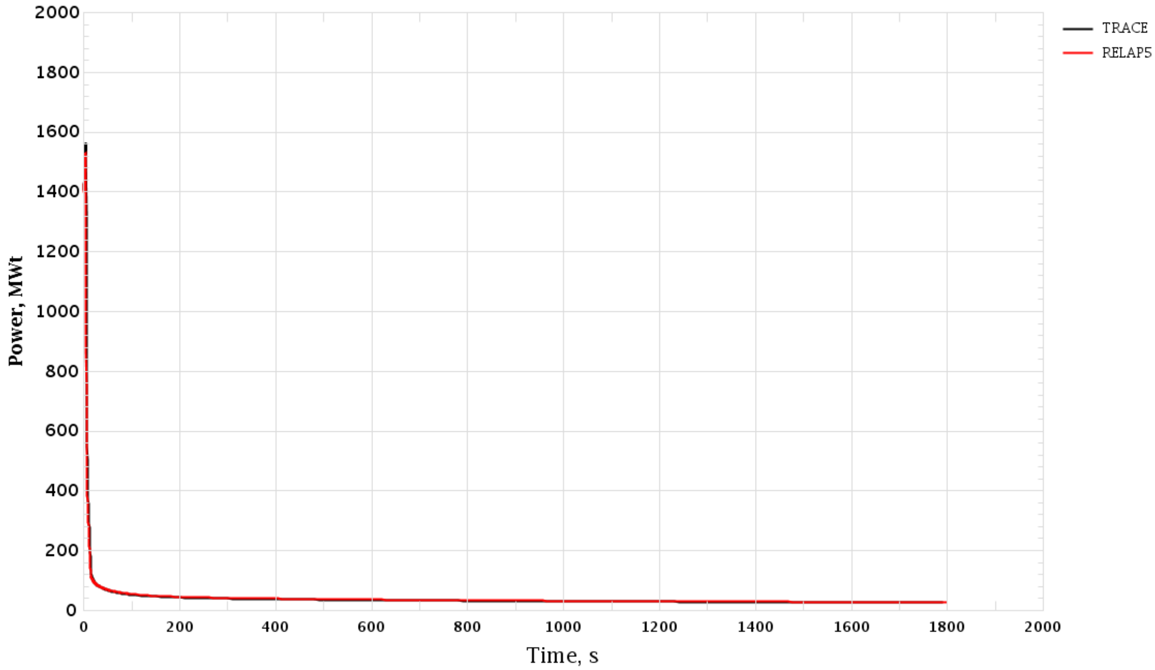

Figure 3-145 Uncontrolled Withdrawal of Control Assemblies Group. Core Thermal Power

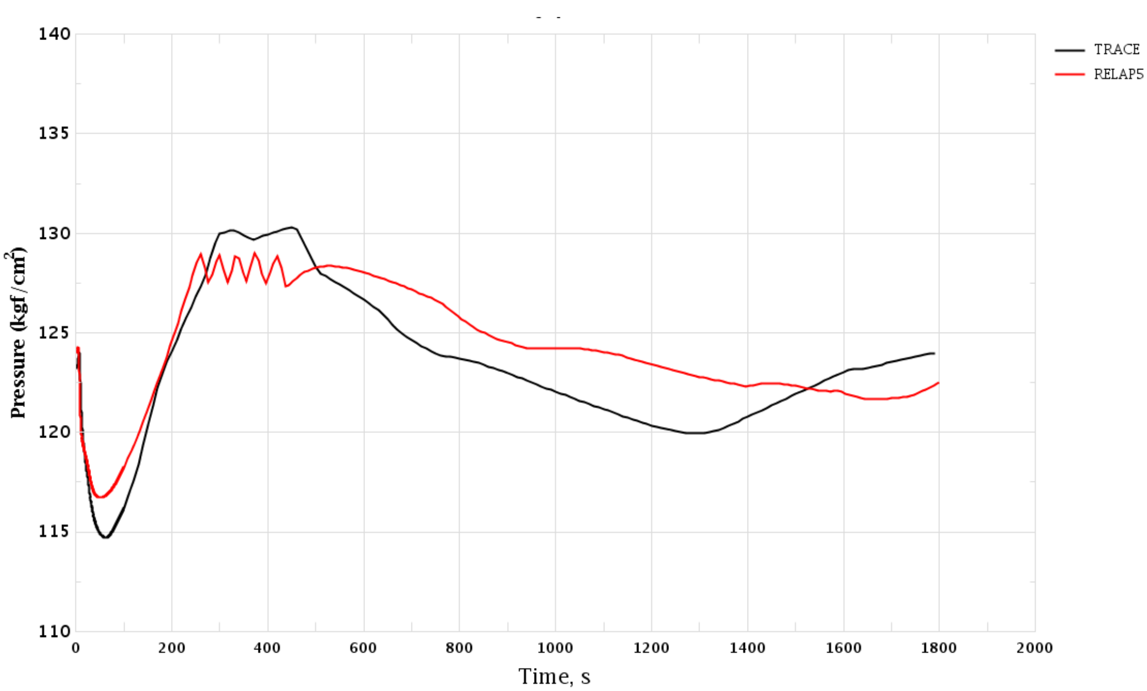

Figure 3-146 Uncontrolled Withdrawal of Control Assemblies Group. RCS Pressure

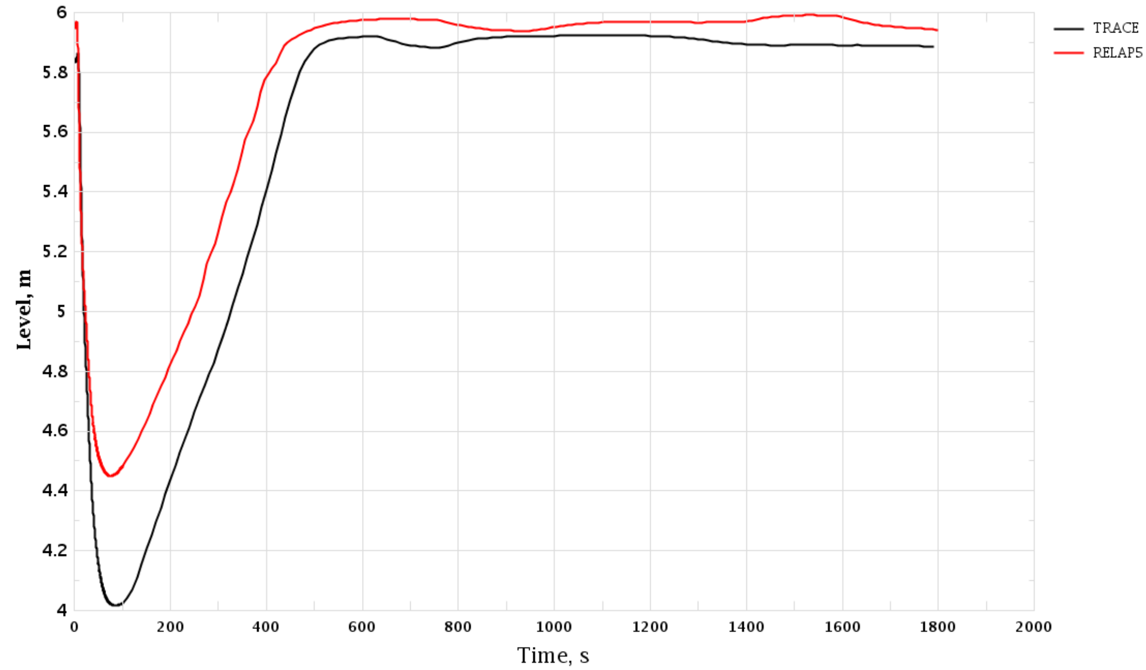

Figure 3-147 Uncontrolled Withdrawal of Control Assemblies Group. Pressurizer Level

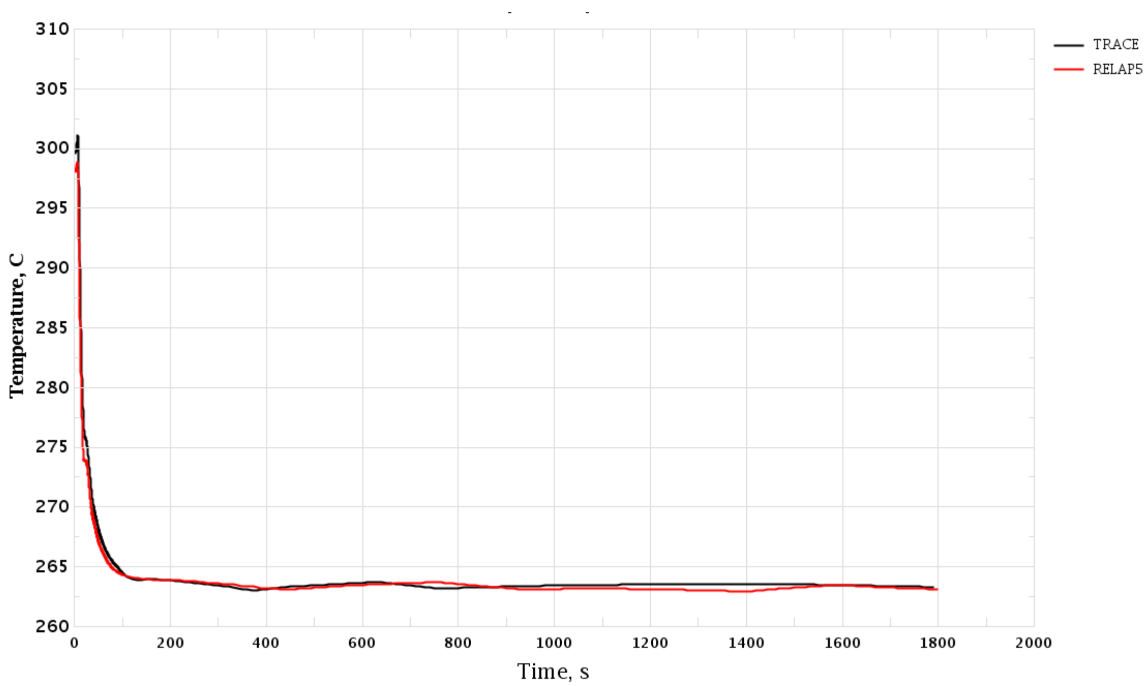

Figure 3-148 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 1

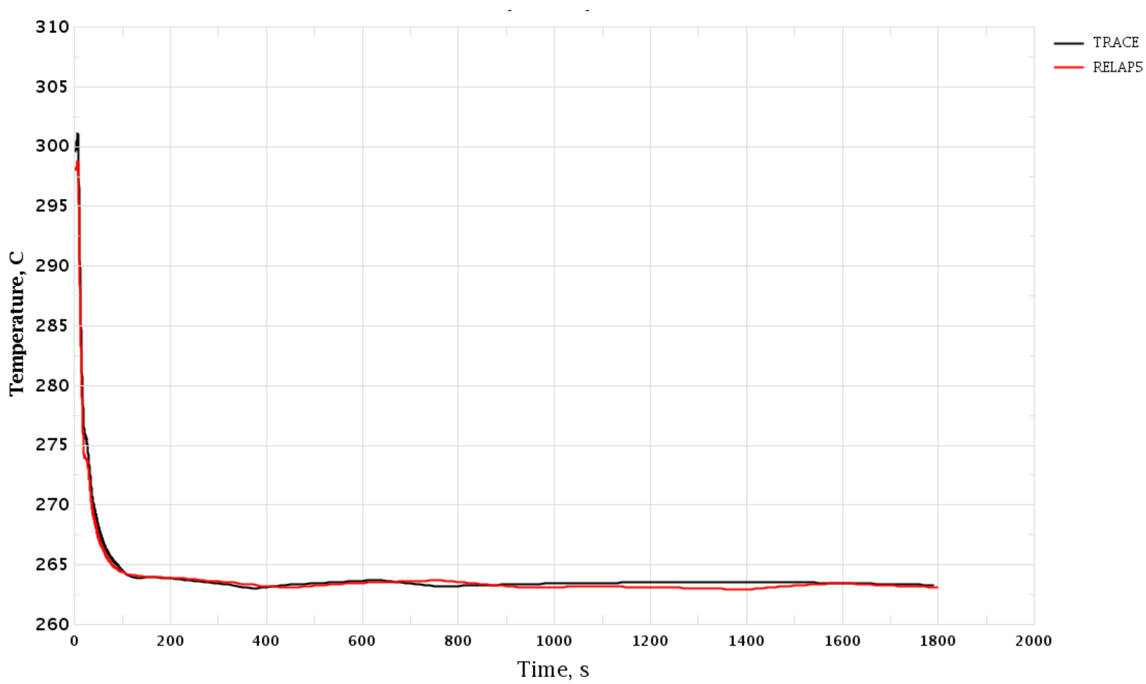

Figure 3-149 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 2

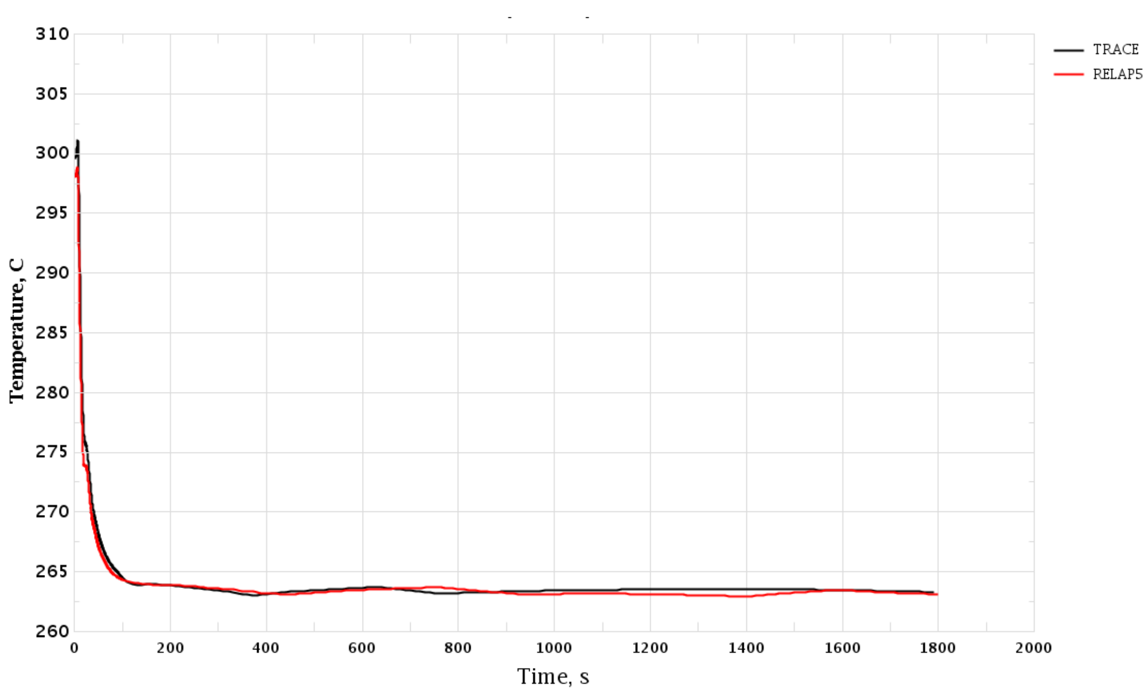

Figure 3-150 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 3

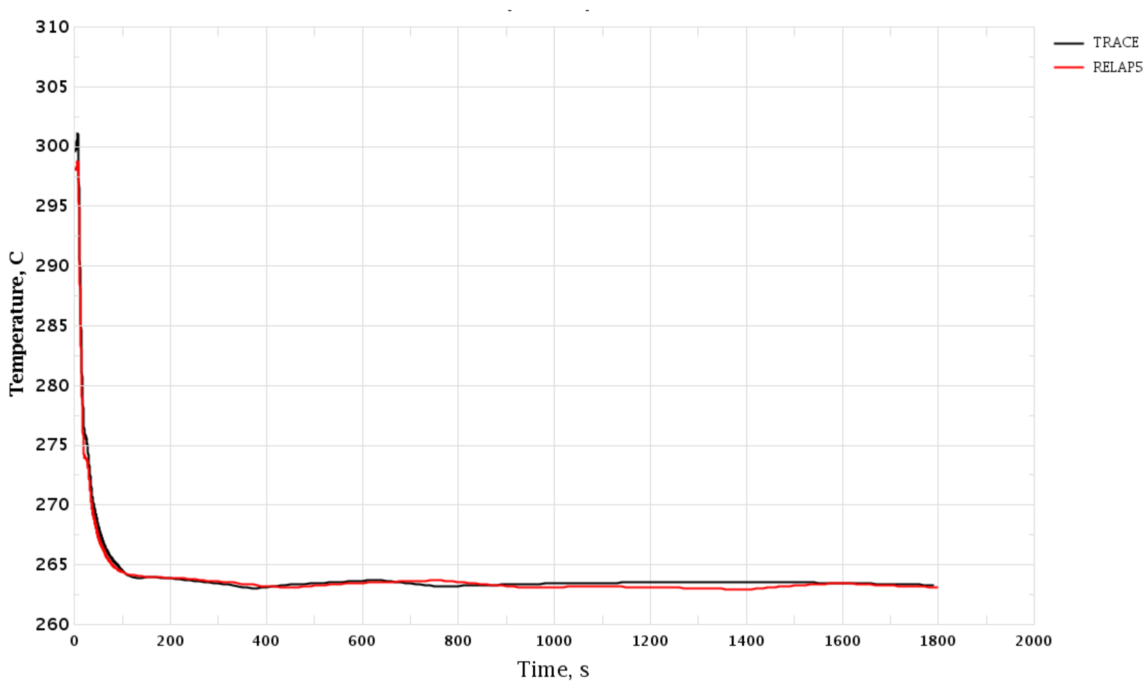

Figure 3-151 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 4

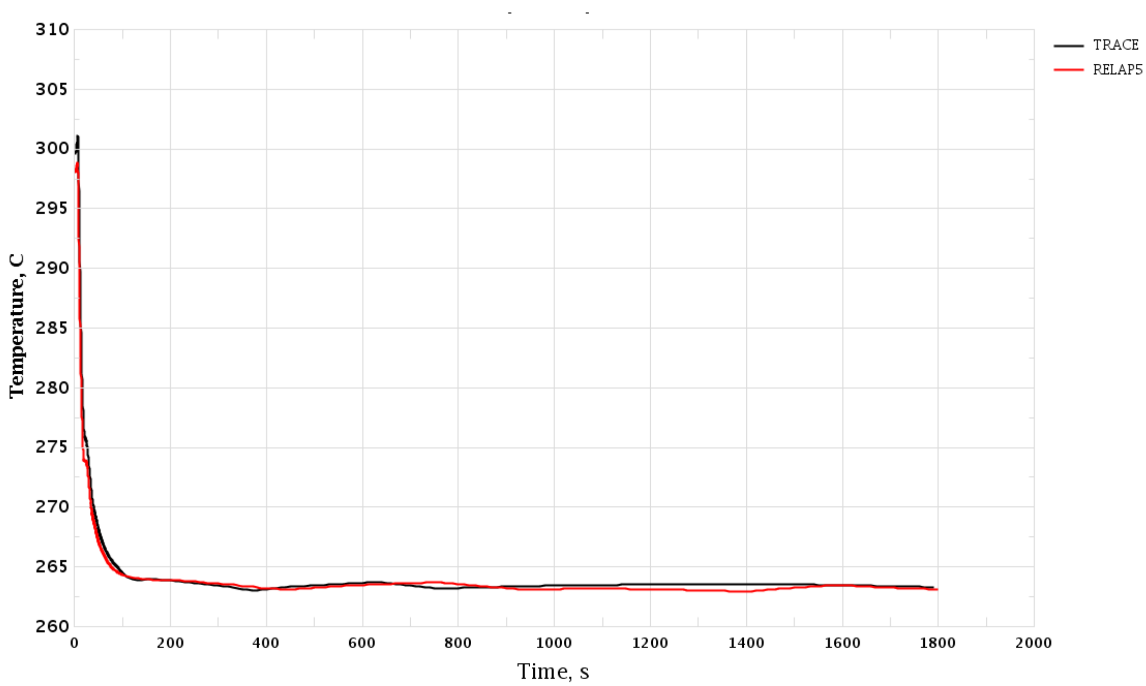

Figure 3-152 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 5

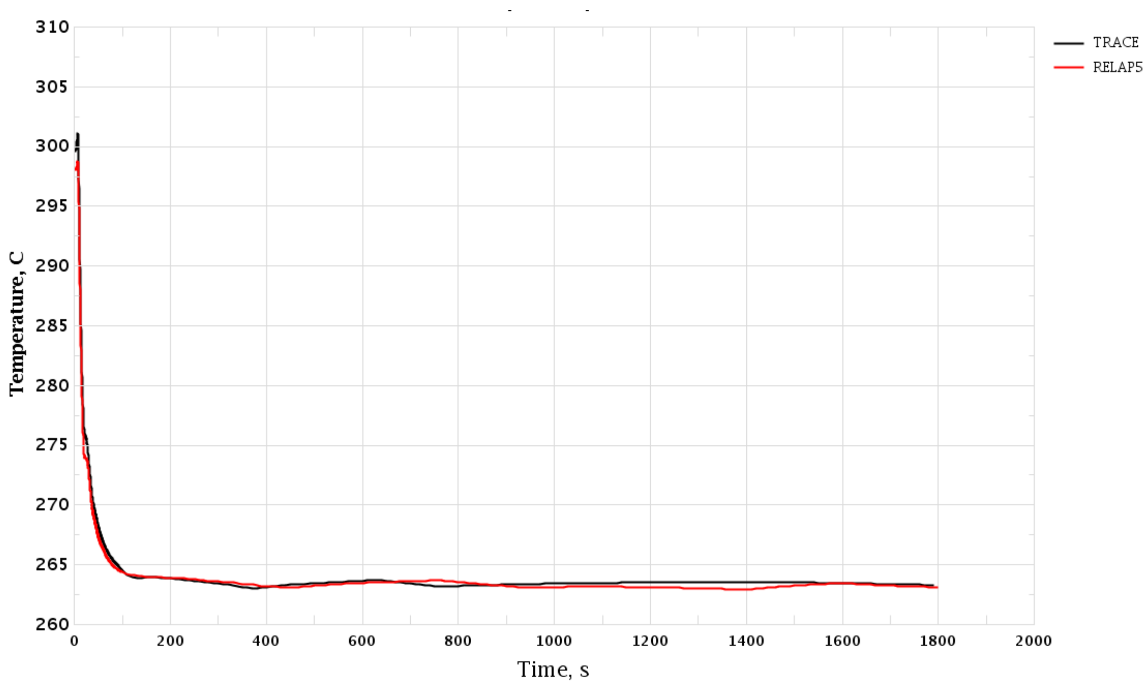

Figure 3-153 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 6

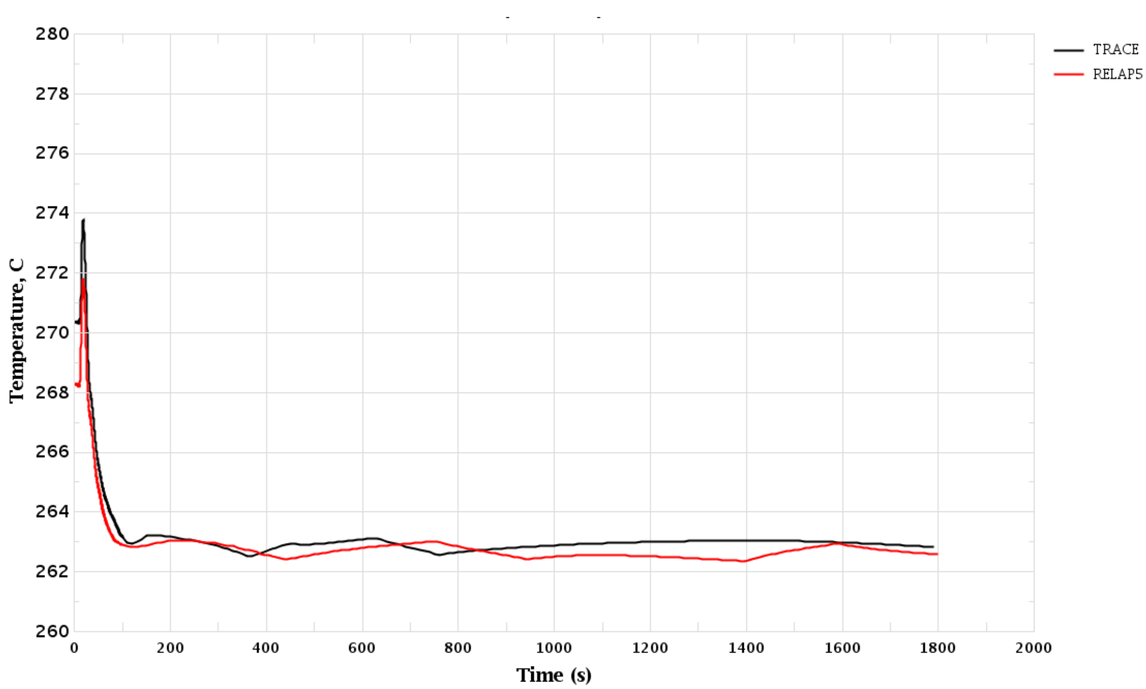

Figure 3-154 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No.1

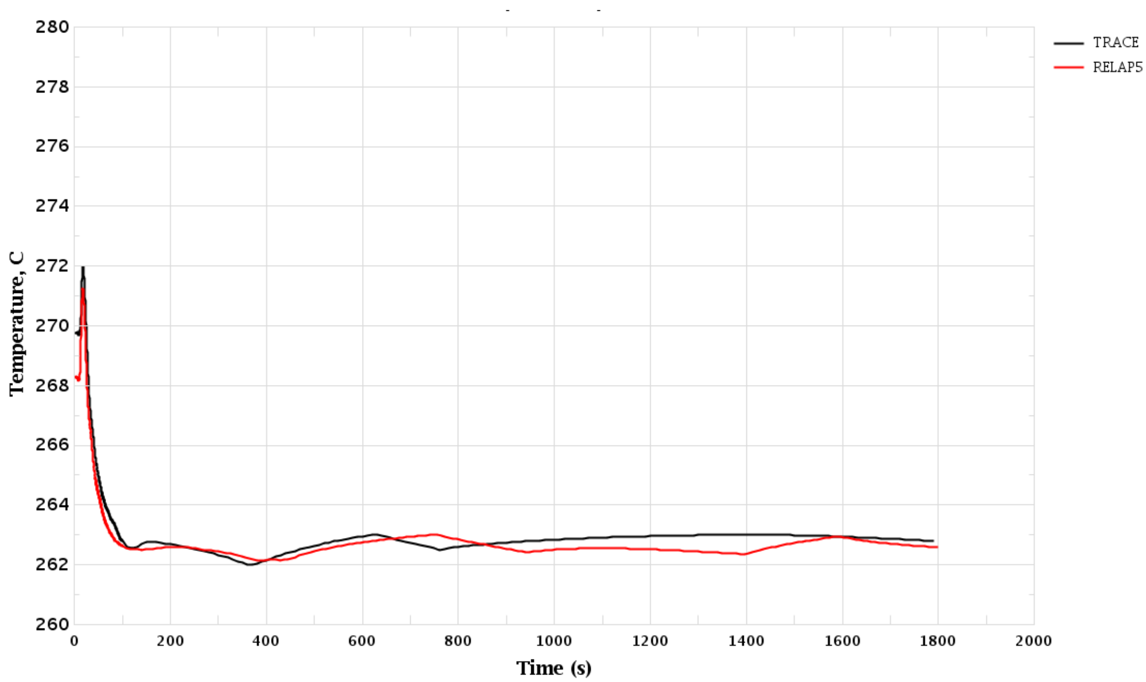

Figure 3-155 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No. 2

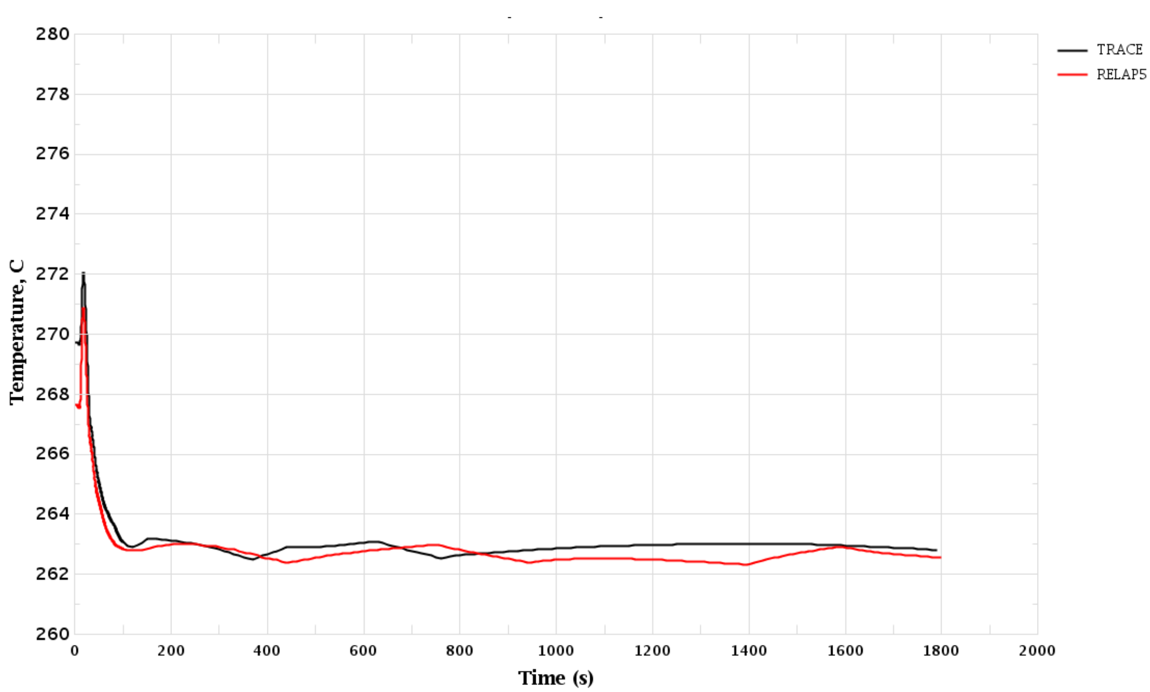

Figure 3-156 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No. 3

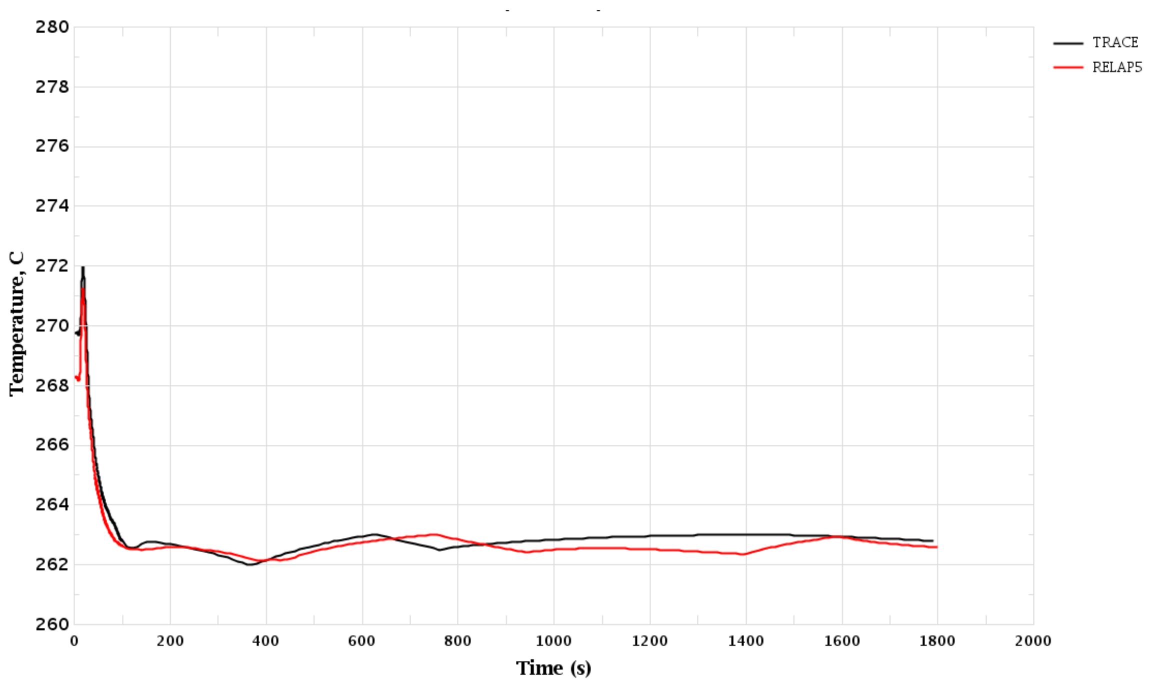

Figure 3-157 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No.4

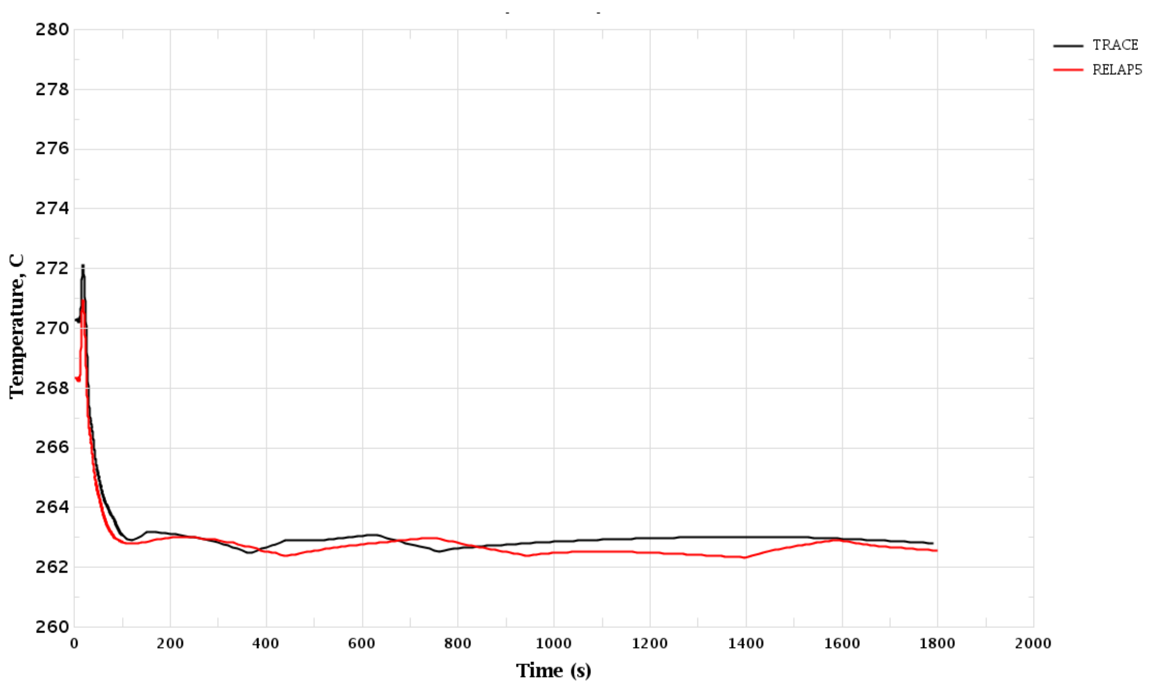

Figure 3-158 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No.5

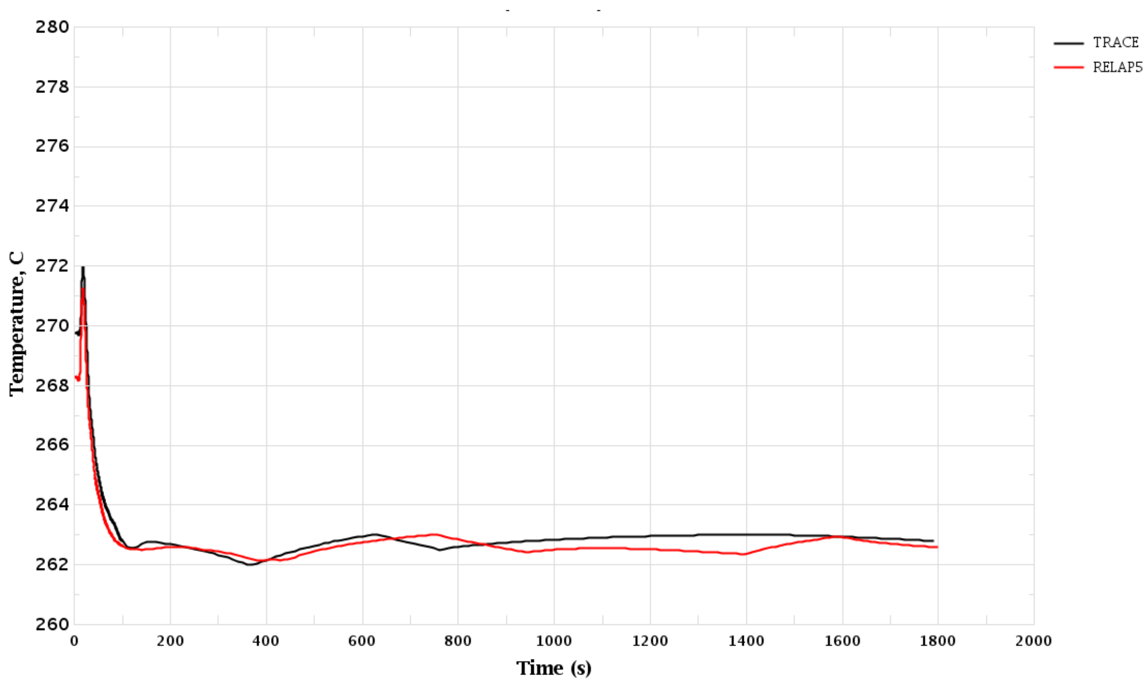

Figure 3-159 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No. 6

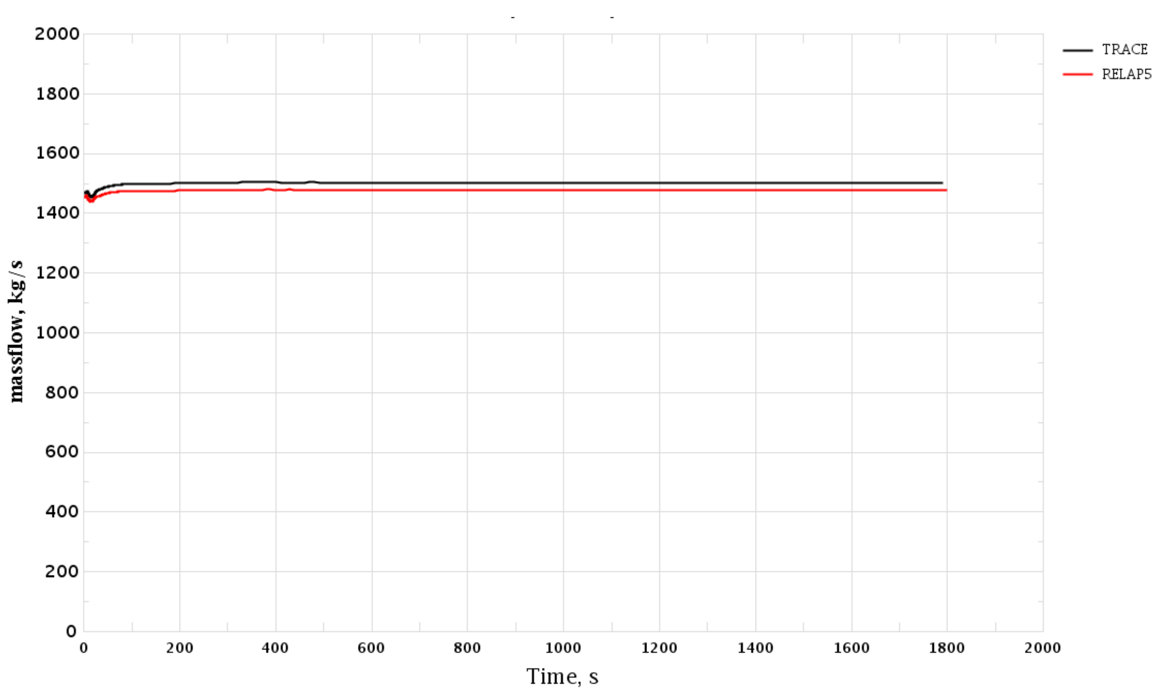

Figure 3-160 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 1 Mass Flow Rate

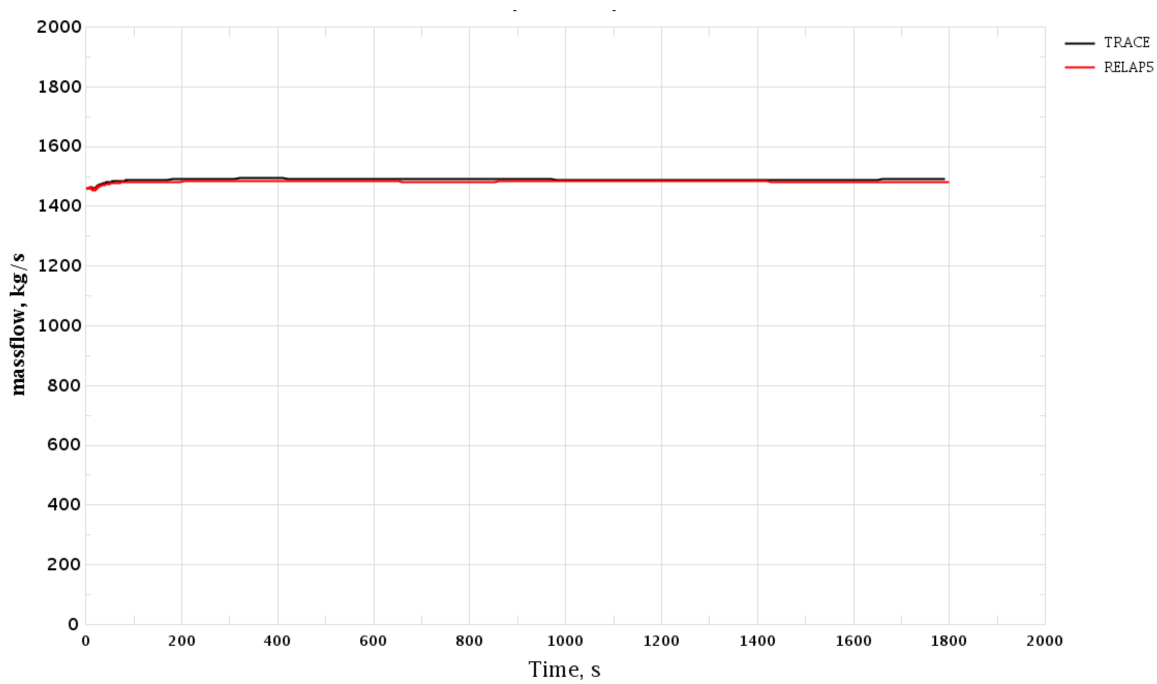

Figure 3-161 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 2 Mass Flow Rate

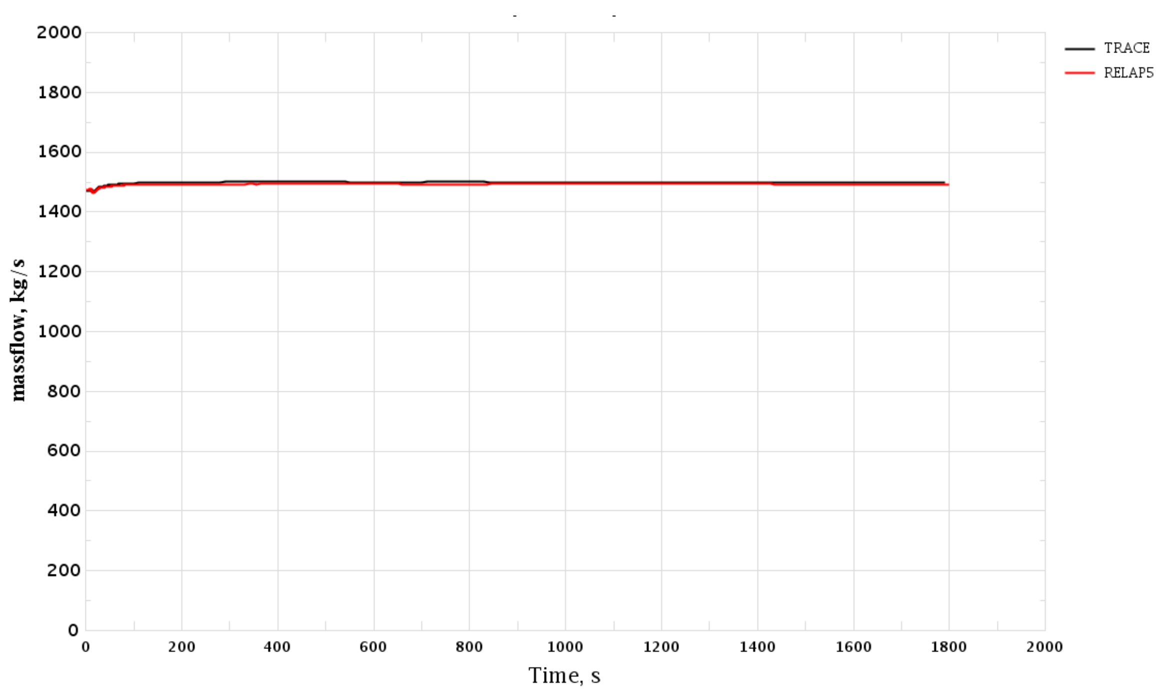

Figure 3-162 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 3 Mass Flow Rate

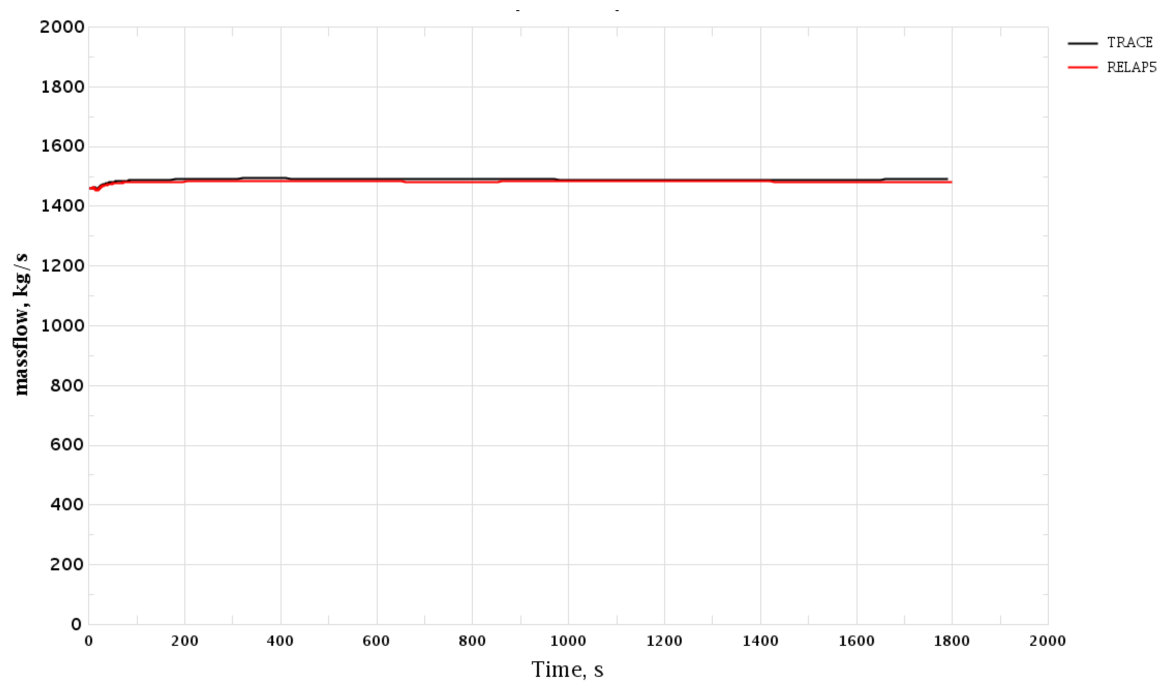

Figure 3-163 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 4 Mass Flow Rate

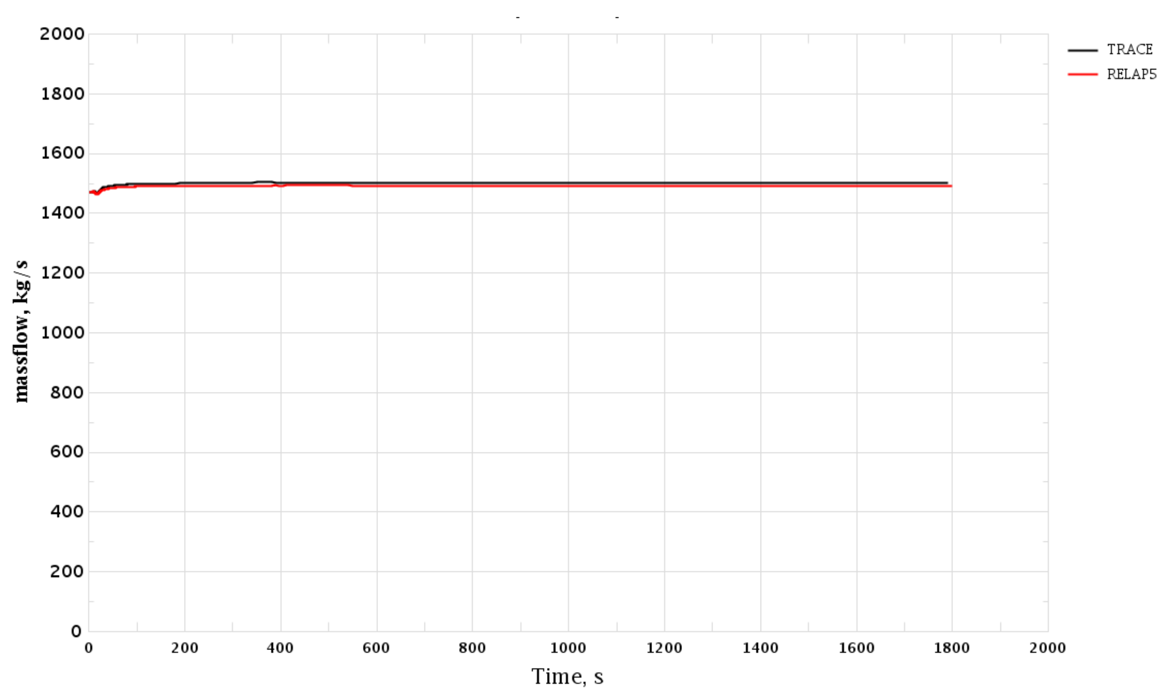

Figure 3-164 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 5 Mass Flow Rate

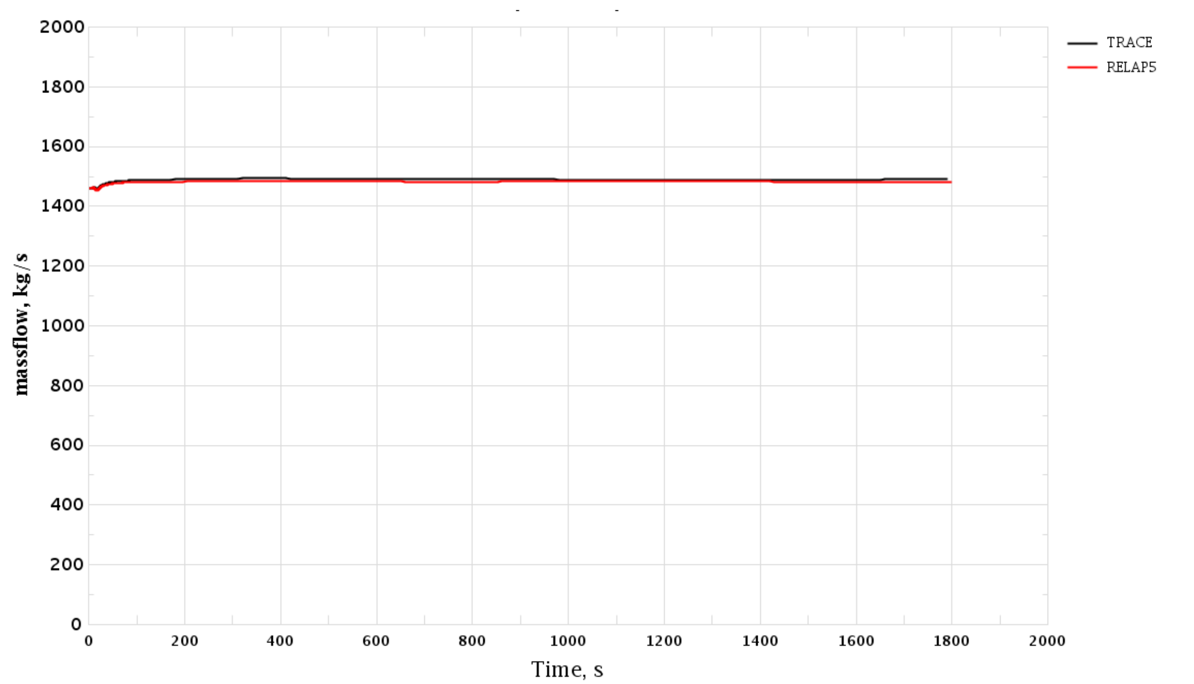

Figure 3-165 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 6 Mass Flow Rate

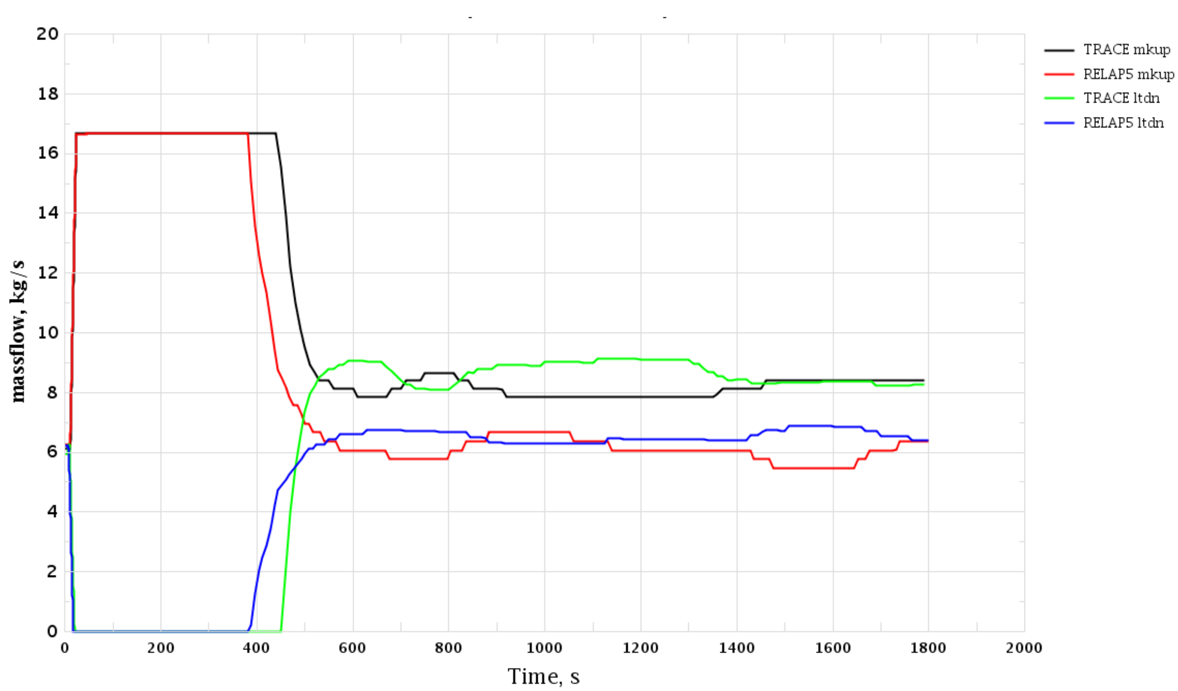

Figure 3-166 Uncontrolled Withdrawal of Control Assemblies Group. Make-Up and Let-Down Mass Flow

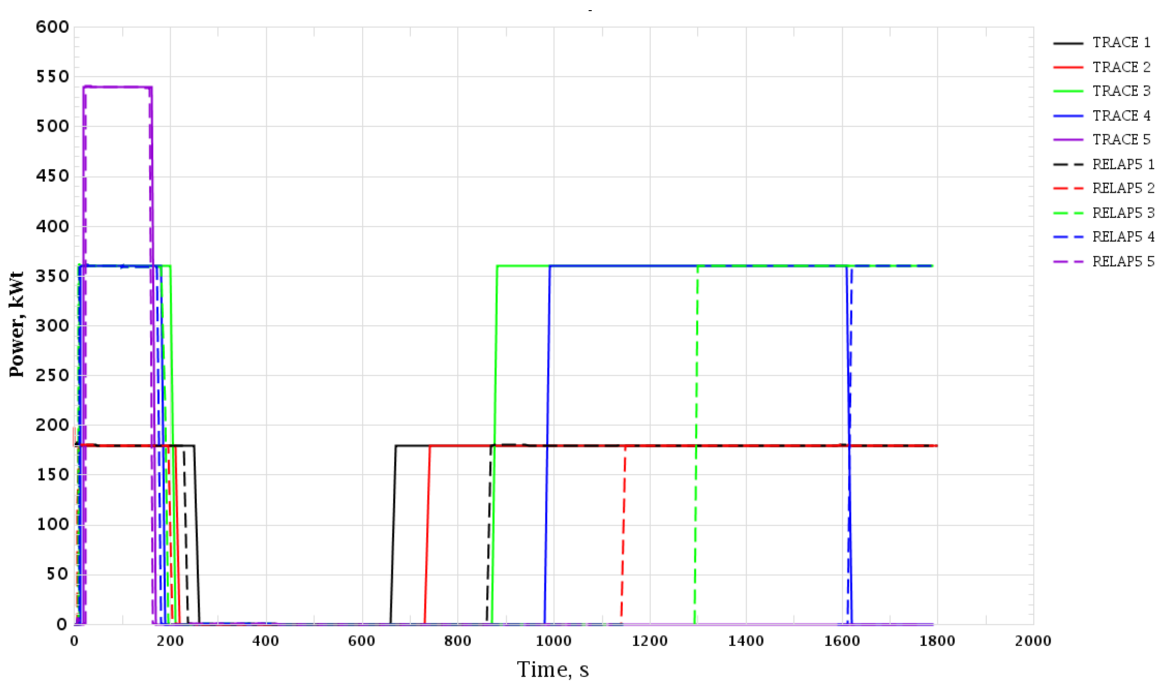

Figure 3-167 Uncontrolled Withdrawal of Control Assemblies Group. PRZ Heaters Power

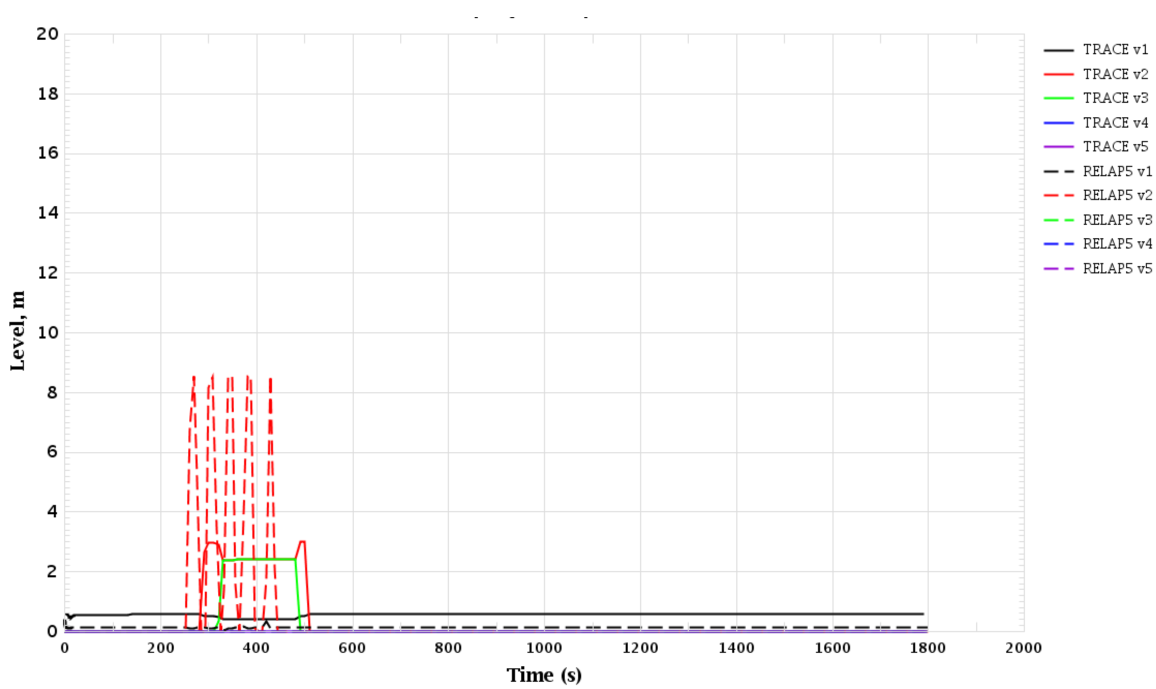

Figure 3-168 Uncontrolled Withdrawal of Control Assemblies Group. PRZ Spray Mass Flow Rate

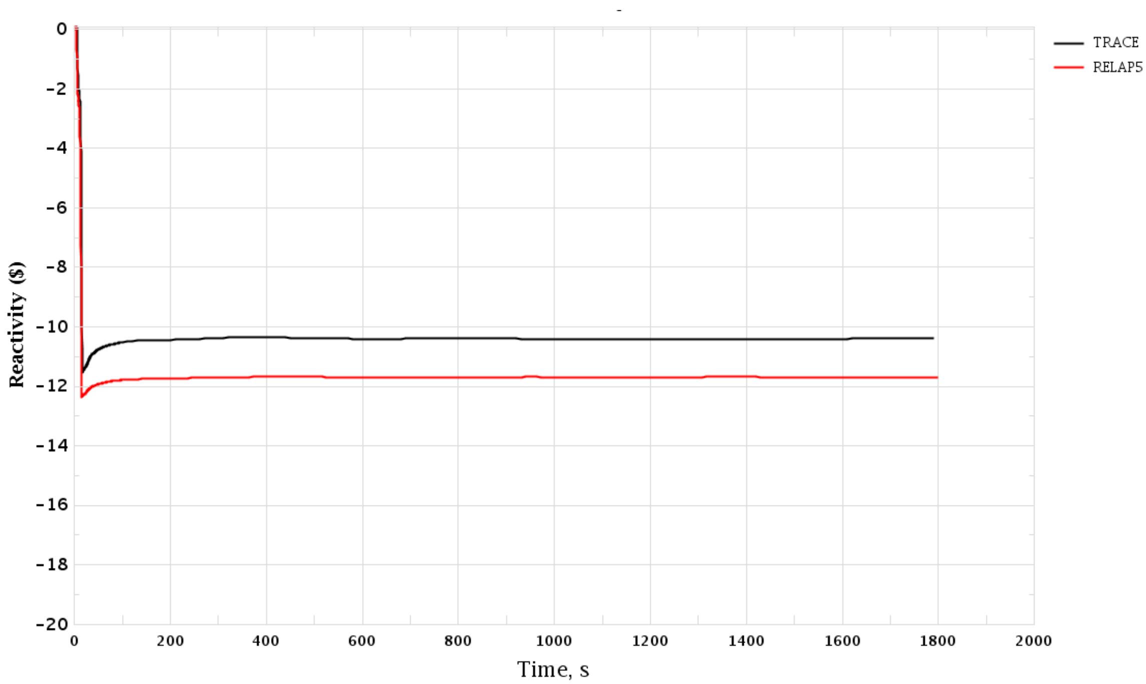

Figure 3-169 Uncontrolled Withdrawal of Control Assemblies Group. Core Reactivity

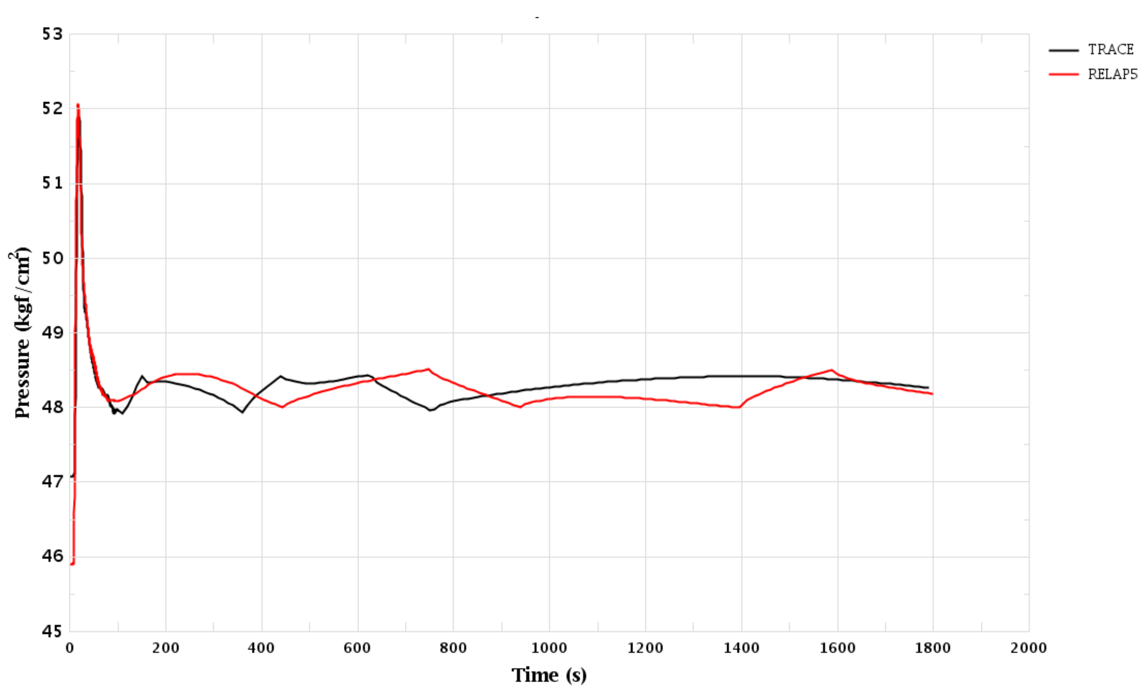

Figure 3-170 Uncontrolled Withdrawal of Control Assemblies Group. SG-1 Pressure

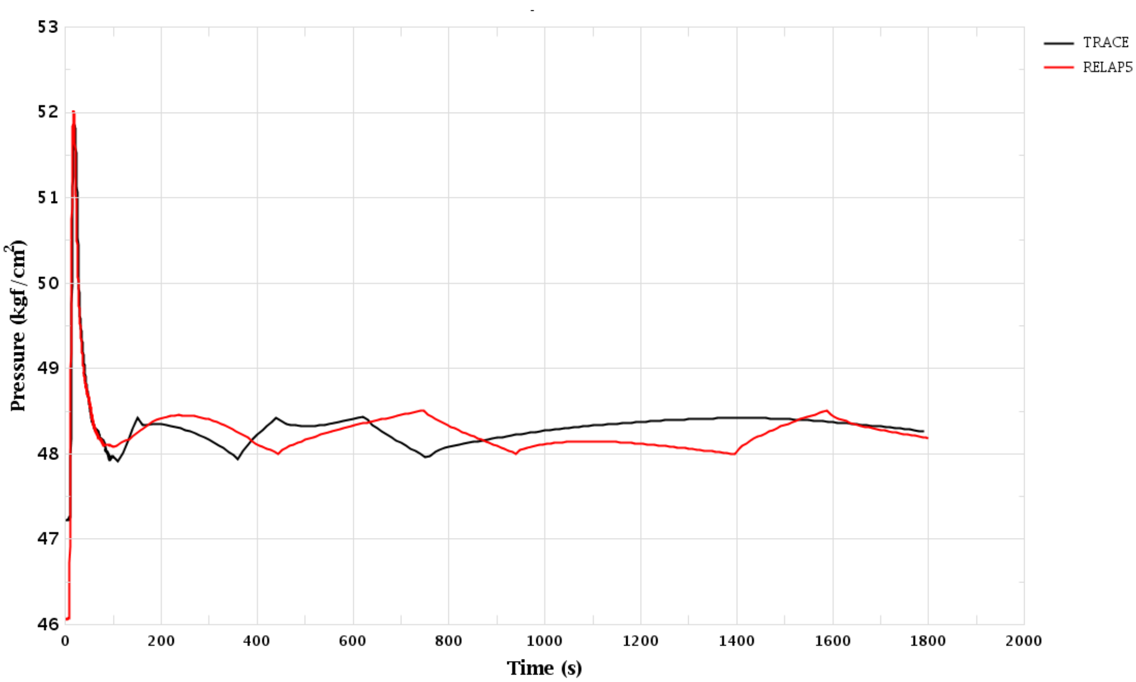

Figure 3-171 Uncontrolled Withdrawal of Control Assemblies Group. SG-2 Pressure

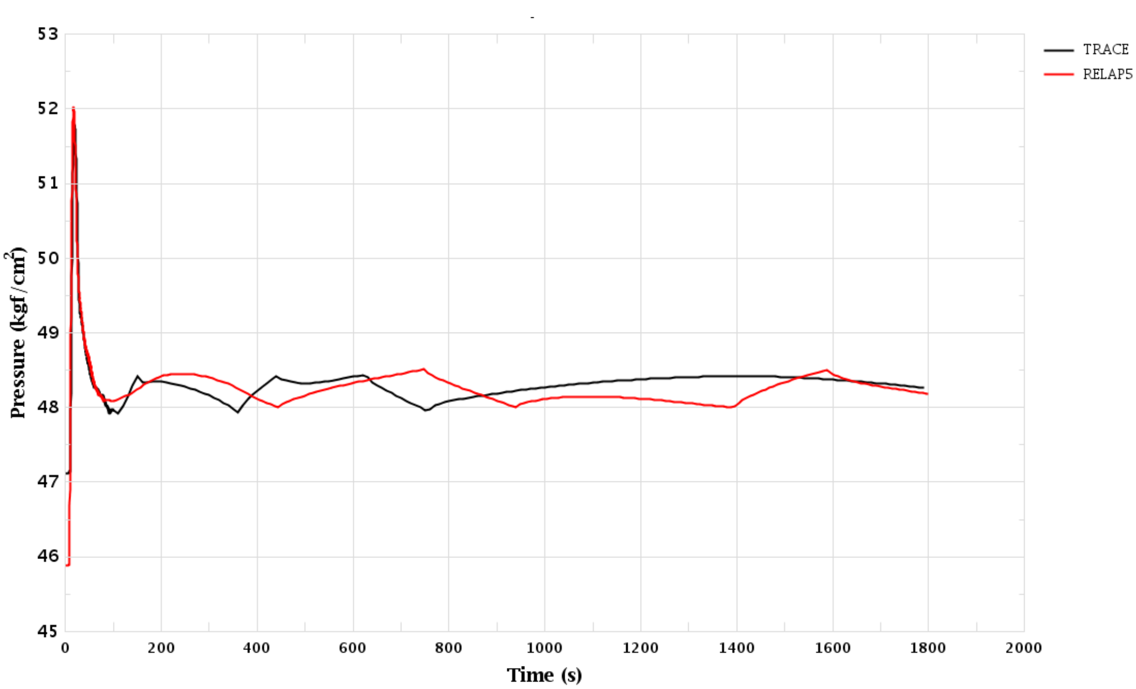

Figure 3-172 Uncontrolled Withdrawal of Control Assemblies Group. SG-3 Pressure

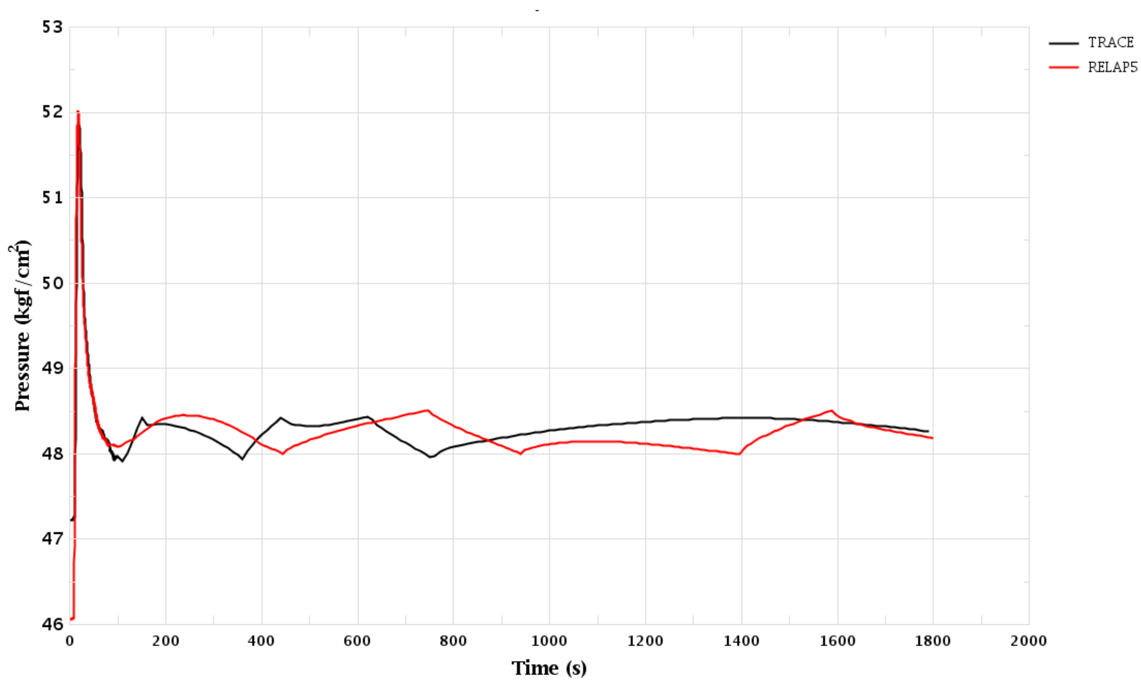

Figure 3-173 Uncontrolled Withdrawal of Control Assemblies Group. SG-4 Pressure

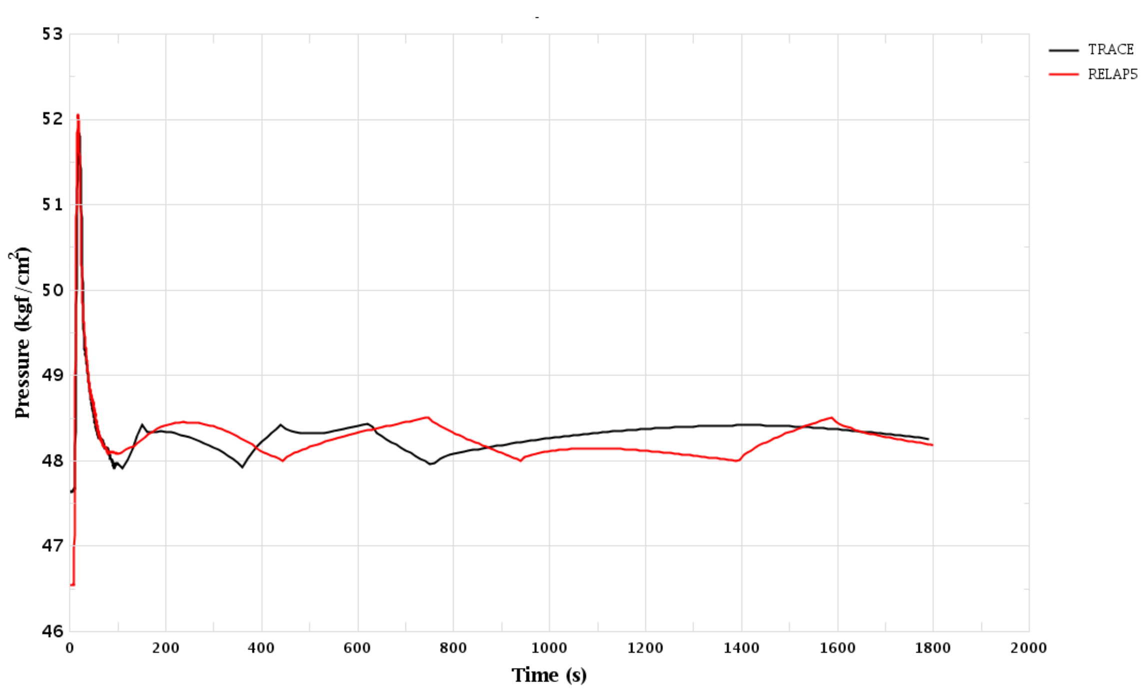

Figure 3-174 Uncontrolled Withdrawal of Control Assemblies Group. SG-5 Pressure

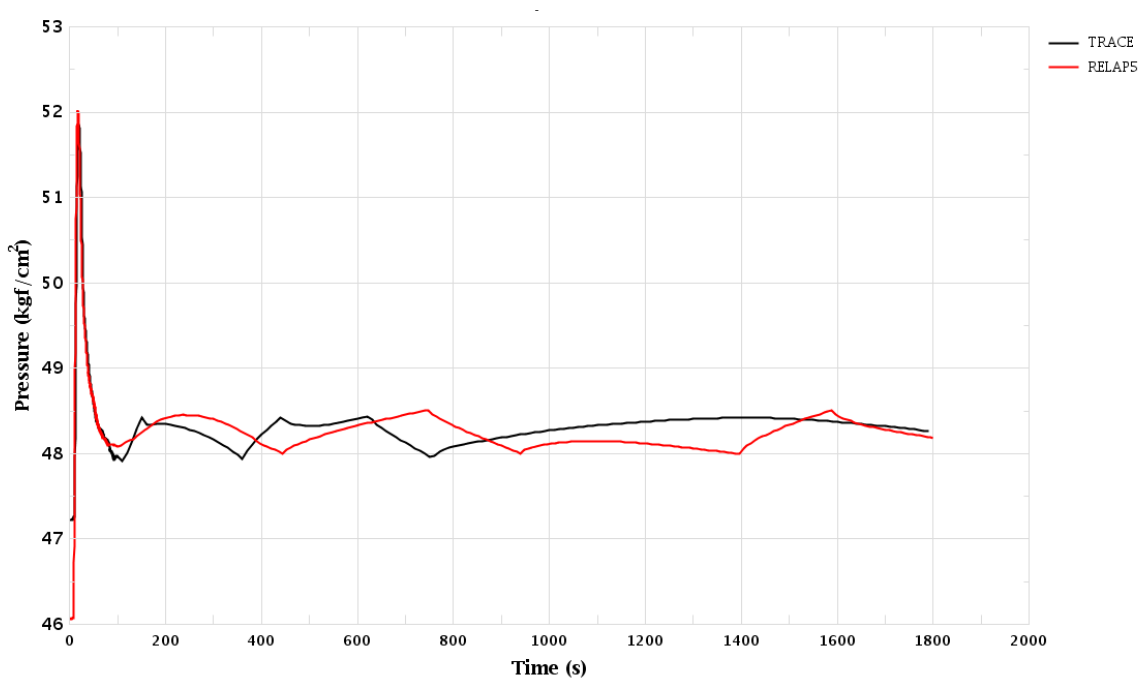

Figure 3-175 Uncontrolled Withdrawal of Control Assemblies Group. SG-6 Pressure

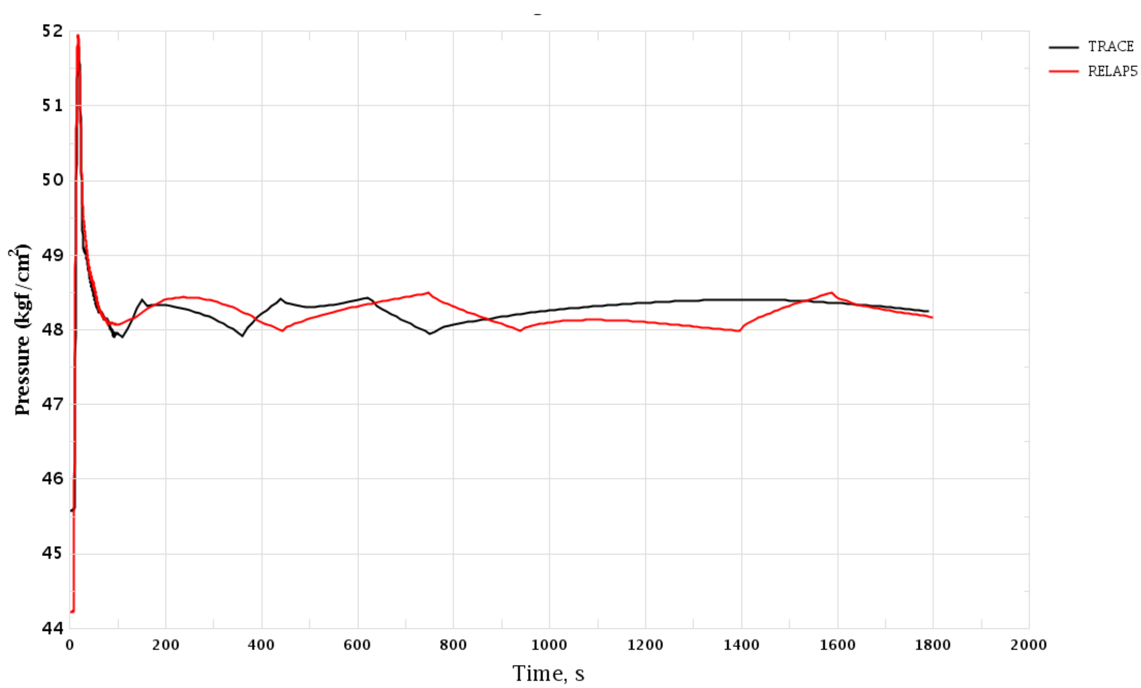

Figure 3-176 Uncontrolled Withdrawal of Control Assemblies Group. MSH-1 Pressure

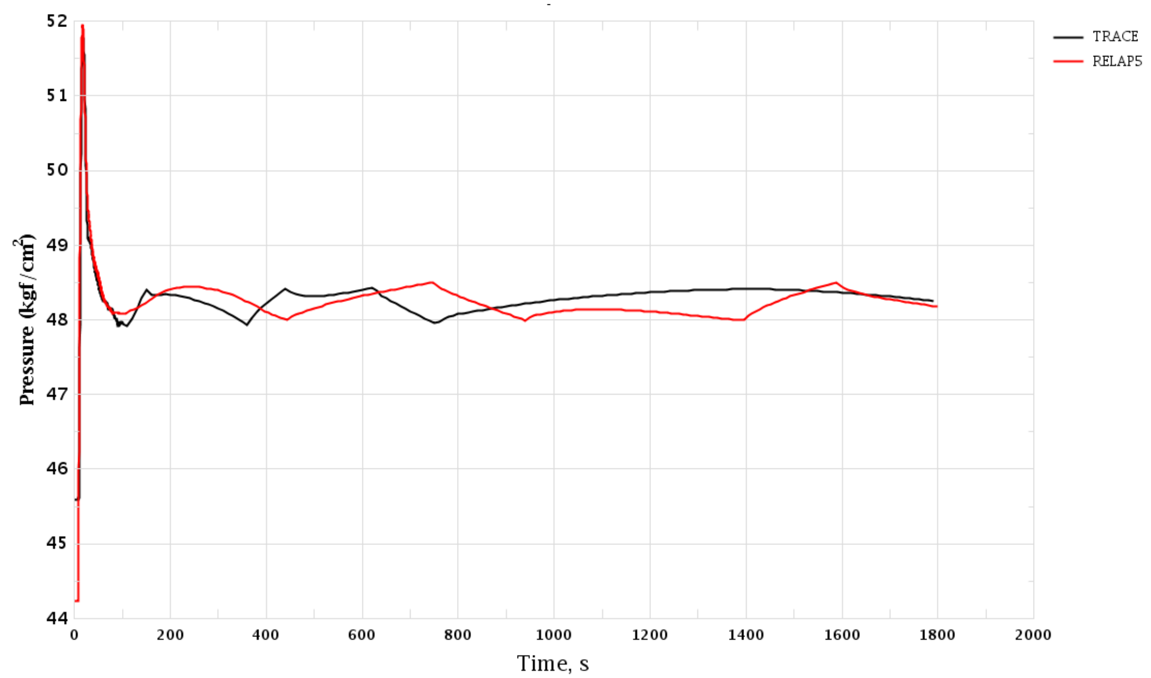

Figure 3-177 Uncontrolled Withdrawal of Control Assemblies Group. MSH-2 Pressure

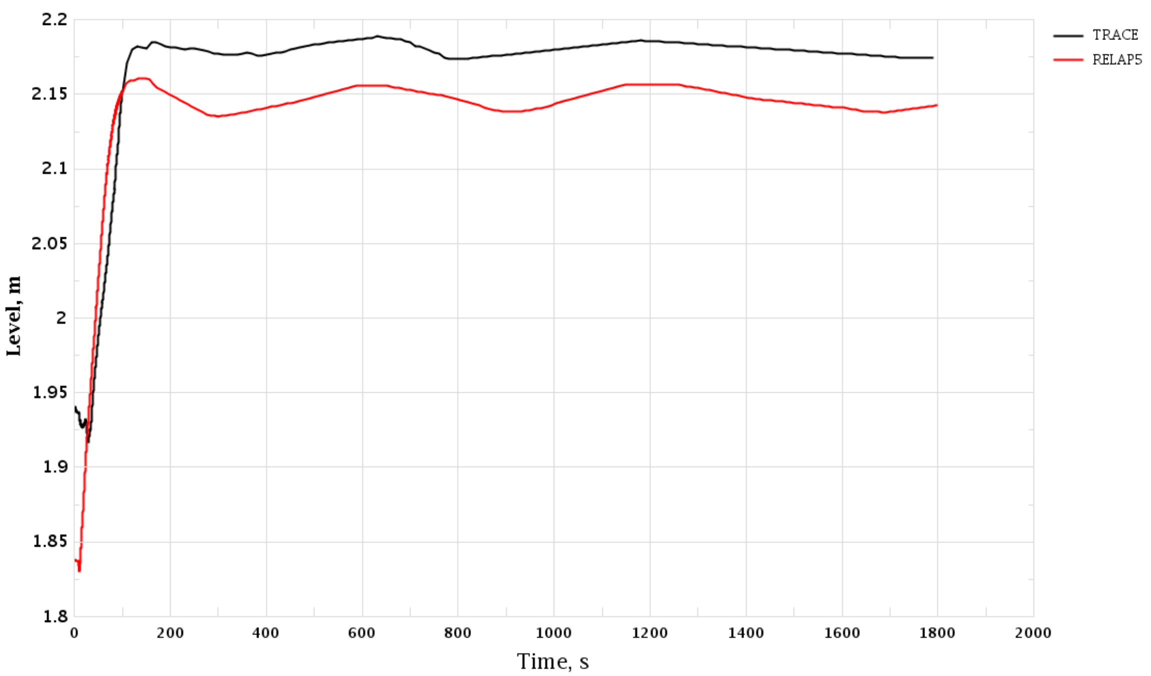

Figure 3-178 Uncontrolled Withdrawal of Control Assemblies Group. SG-1 Level (Wide Range)

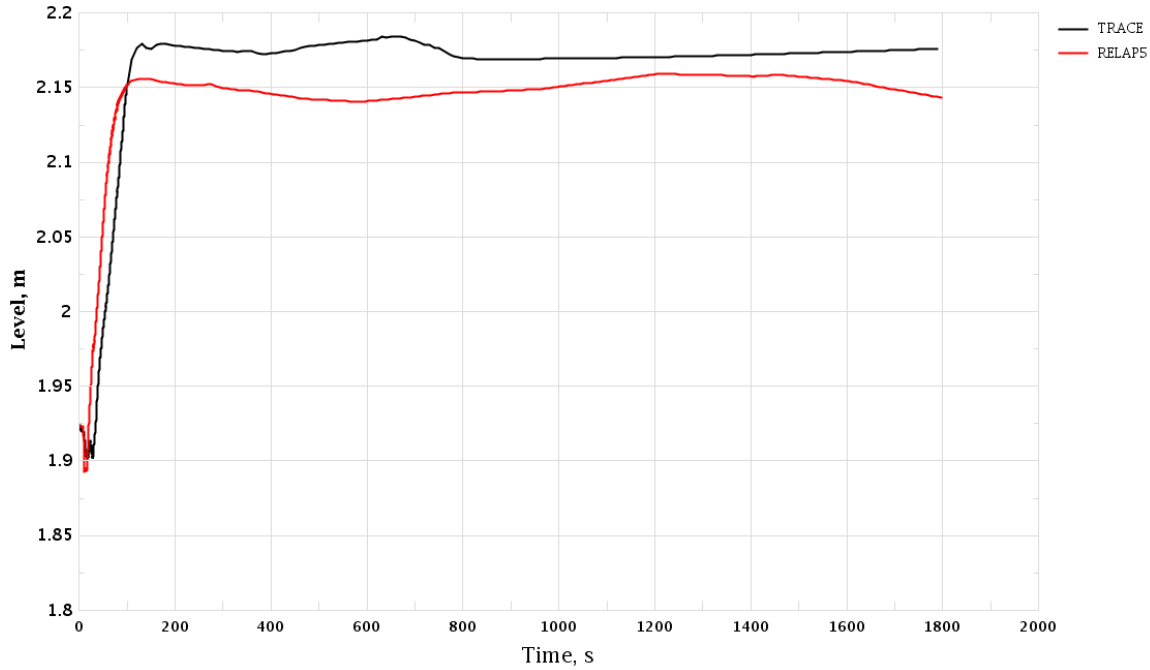

Figure 3-179 Uncontrolled Withdrawal of Control Assemblies Group. SG-2 Level (Wide Range)

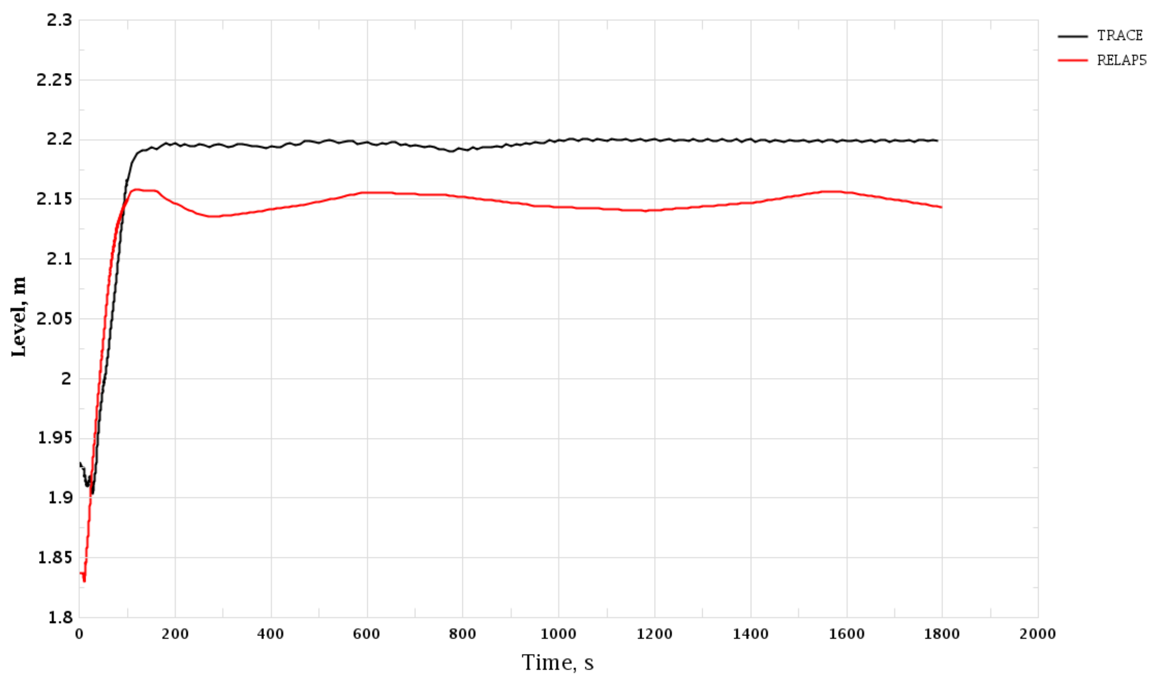

Figure 3-180 Uncontrolled Withdrawal of Control Assemblies Group. SG-3 Level (Wide Range)

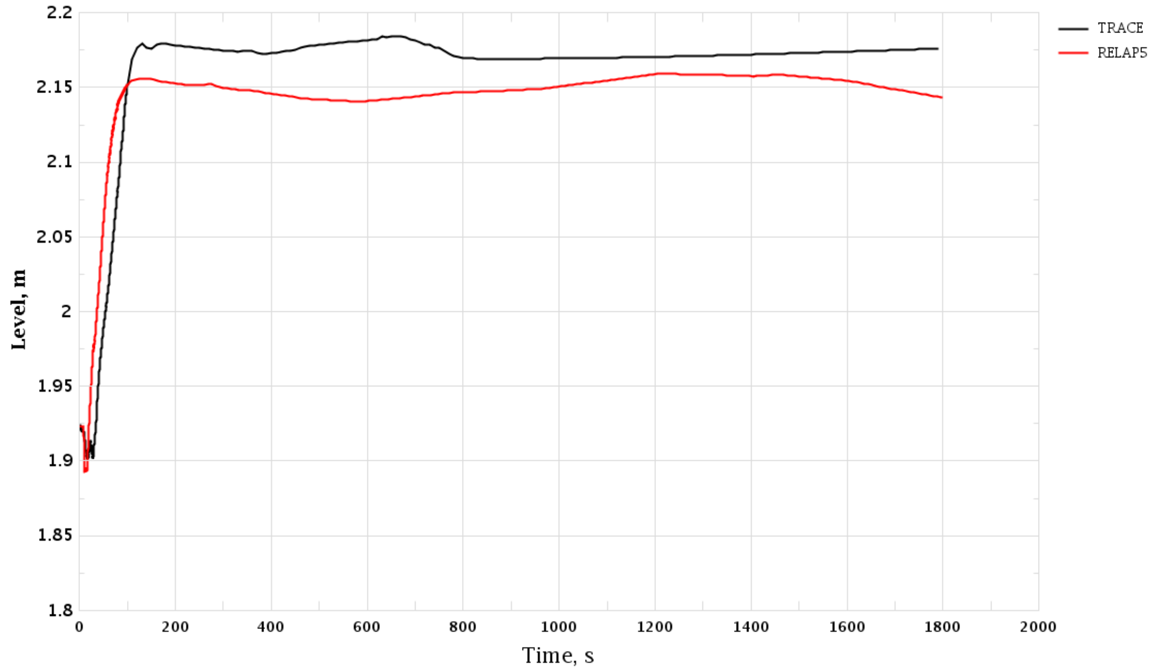

Figure 3-181 Uncontrolled Withdrawal of Control Assemblies Group. SG-4 Level (Wide Range)

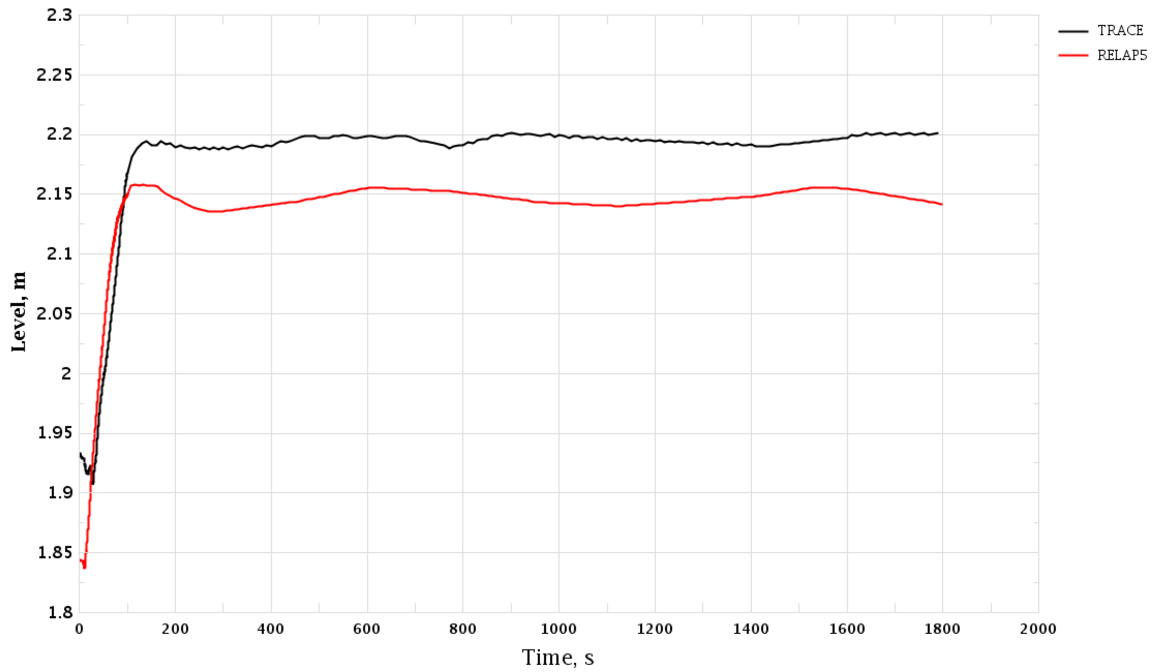

Figure 3-182 Uncontrolled Withdrawal of Control Assemblies Group. SG-5 Level (Wide Range)

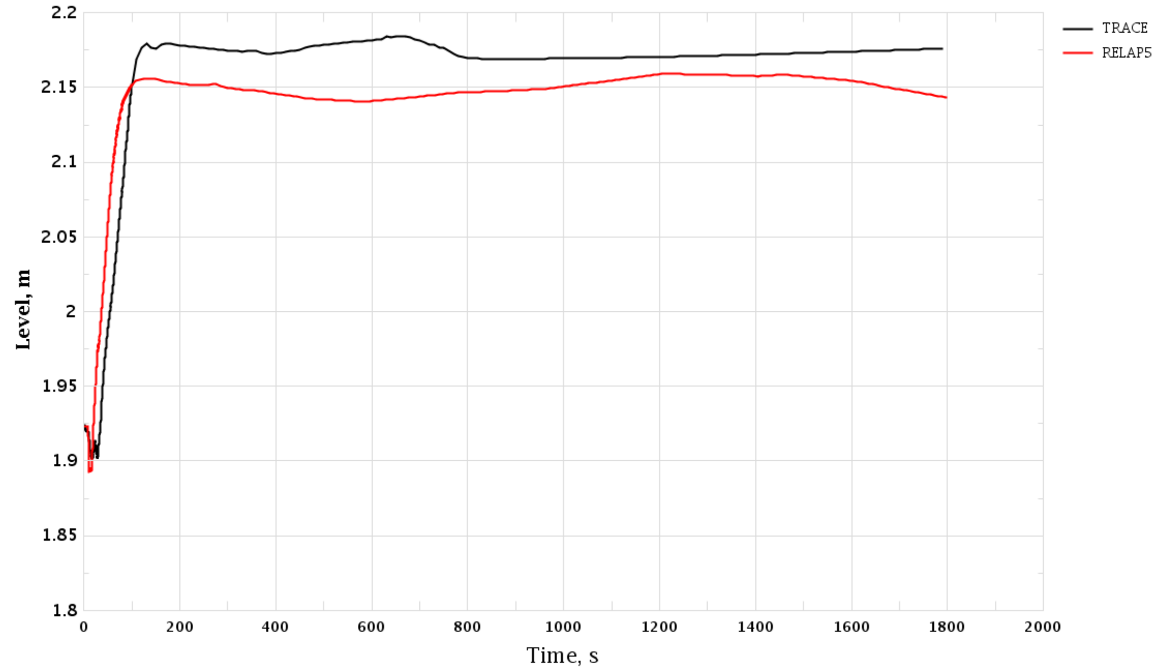

Figure 3-183 Uncontrolled Withdrawal of Control Assemblies Group. SG-6 Level (Wide Range)

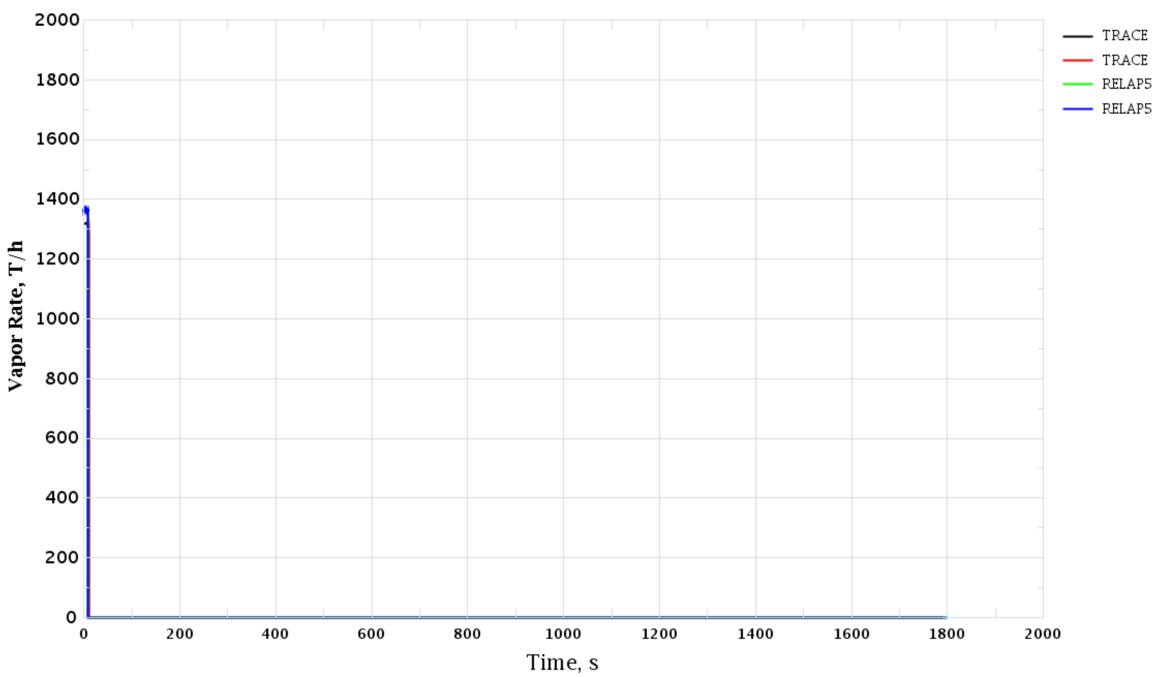

Figure 3-184 Uncontrolled Withdrawal of Control Assemblies Group. Turbines Mass Flow Rate

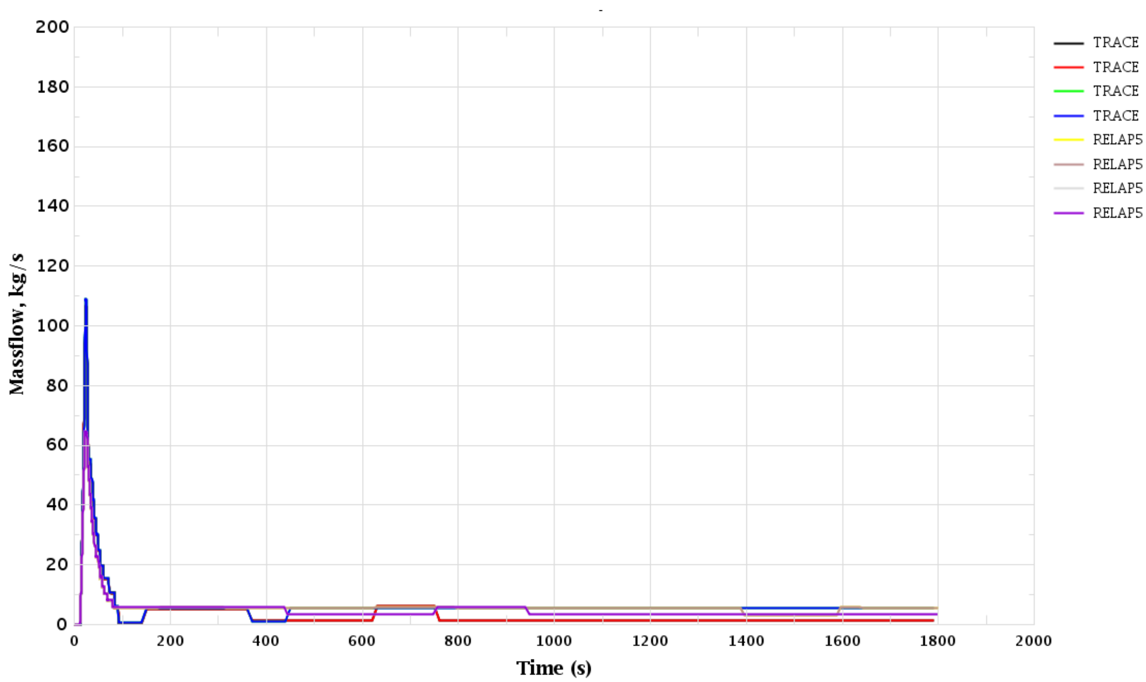

Figure 3-185 Uncontrolled Withdrawal of Control Assemblies Group. BRU-K Steam Mass Flow Rate

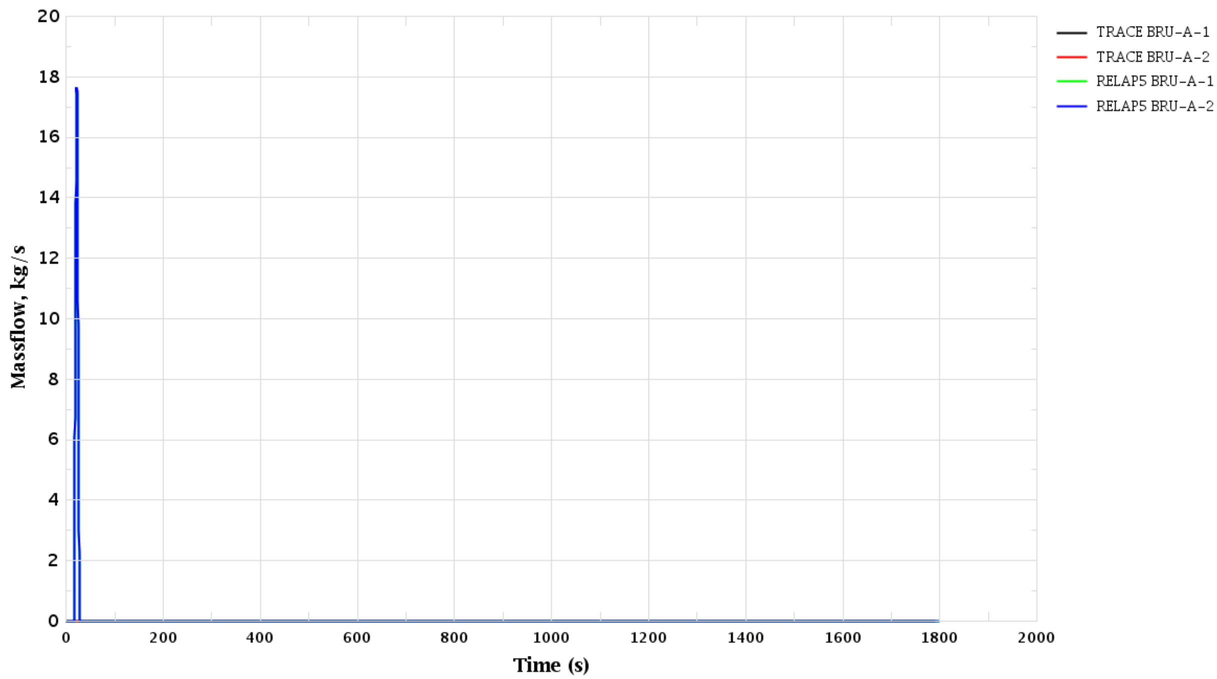

Figure 3-186 Uncontrolled Withdrawal of Control Assemblies Group. BRU-A Steam Mass Flow Rate

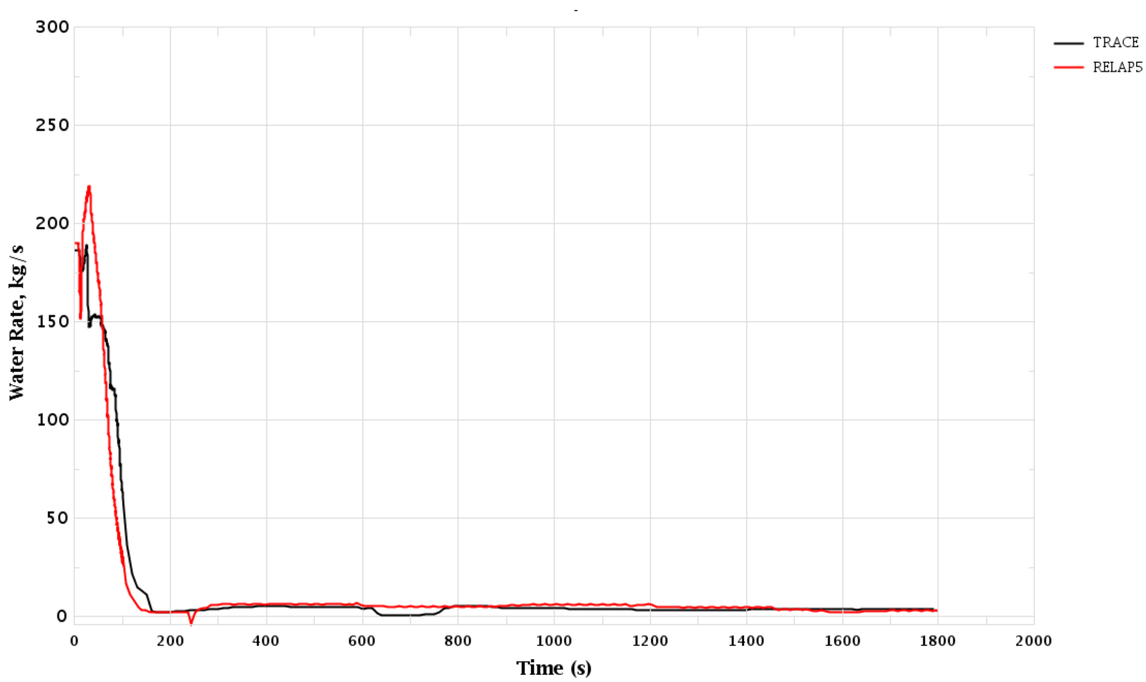

Figure 3-187 Uncontrolled Withdrawal of Control Assemblies Group. MFW Pump No.1 Mass Flow Rate

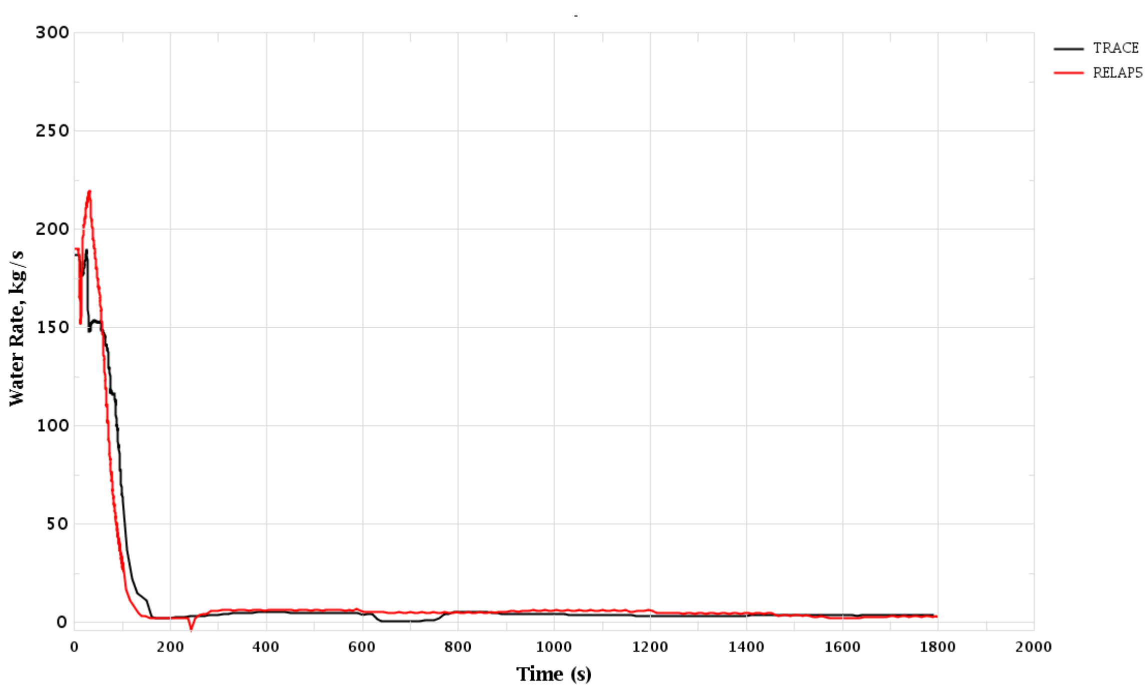

Figure 3-188 Uncontrolled Withdrawal of Control Assemblies Group. MFW Pump No.2 Mass Flow Rate

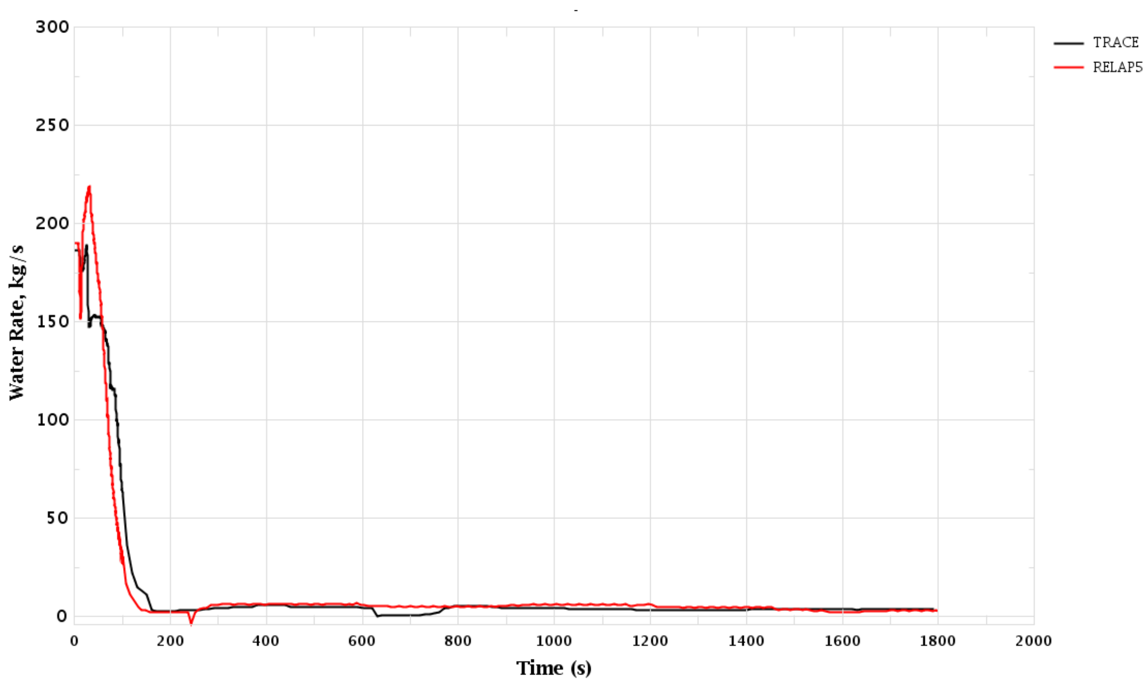

Figure 3-189 Uncontrolled Withdrawal of Control Assemblies Group. MFW Pump No.3 Mass Flow Rate

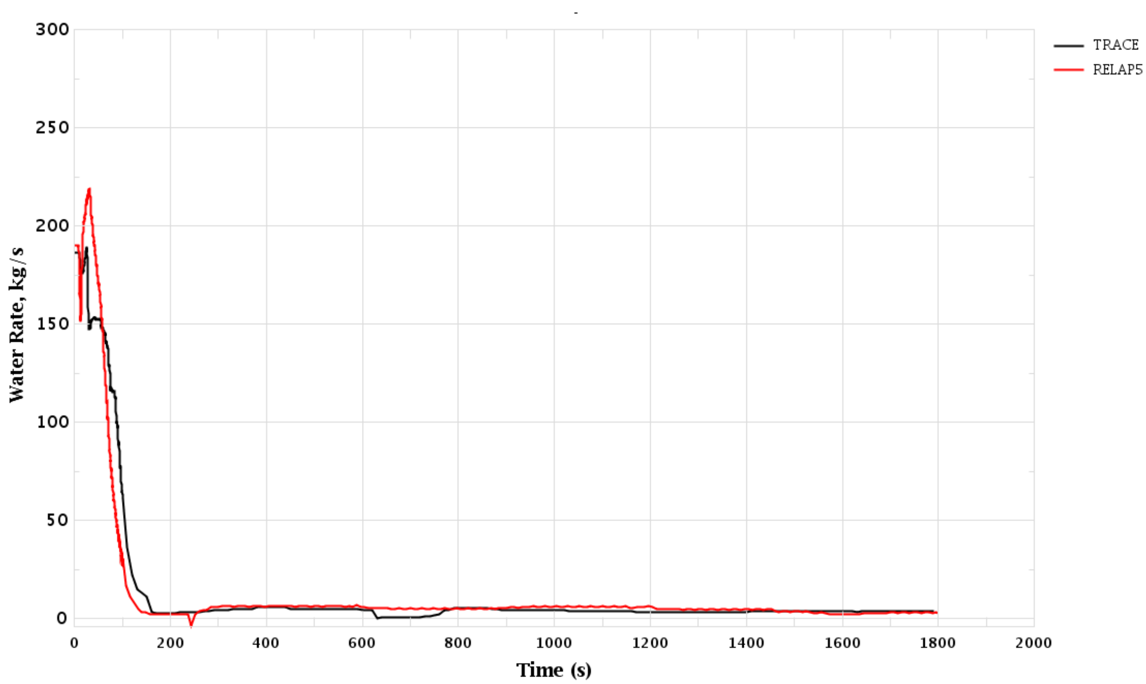

Figure 3-190 Uncontrolled Withdrawal of Control Assemblies Group. MFW Pump No.4 Mass Flow Rate

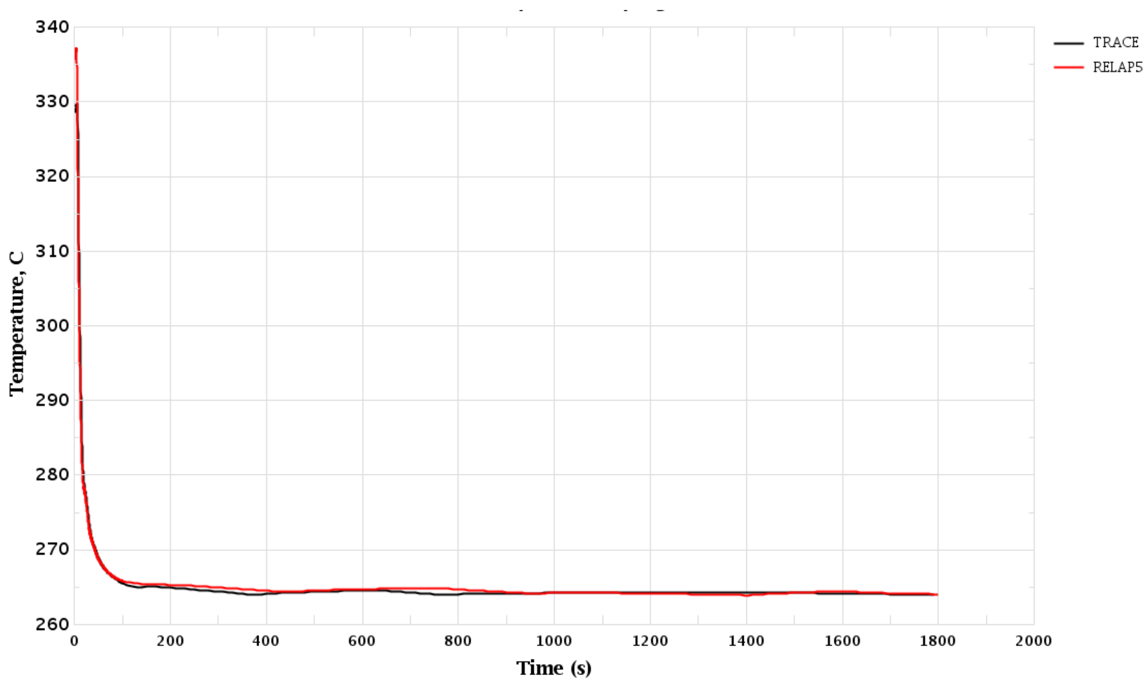

Figure 3-191 Uncontrolled Withdrawal of Control Assemblies Group. Maximal Cladding Temperature

3.5 PRZ Surge Line Break

3.5.1 Brief Description of Initiating Event

This initiating event assumes break of surge line with equivalent diameter of 277 mm connecting hot leg of RCS loop no.1 with PRZ. According to the expected frequency of occurrence the initiating events is categorized as a design basis accident and pertains to loss of coolant accidents (LOCA) IE group.

3.5.2 Initial Conditions

The initial conditions selected for calculation of this accident correspond to those specified in Table 3-1.

3.5.3 Boundary Conditions

Assumptions on the systems availability that are considered in calculation of the accident are specified below.

Operator actions are not modeled.

Primary makeup is terminated at the moment of accident occurrence.

To prolong reactor operation at the nominal power, a failure of Level 2, Level 3 and Level 4 emergency protection systems is postulated. ARM operates in the reactor power maintenance mode until scram signal is generated.

Simultaneously with reactor scram actuation the loss of normal power supply, as well as failure of BRU-A1 is postulated to reach higher cladding temperature.

3.5.4 Calculation Results

Sequence of events for this accident are presented in Table 3-6.

Table 3-6 Sequence of Events for PRZ Surge Line Break Accident

TRACE Time, s	RELAP5 Time, s	Event	Description
0.0	0.0	PRZ surge line break (LOCA with a diameter of 277 mm)	IE occurrence. Make-up failure (postulated)
2.0	3.0	Reactor scram actuation Loss of normal power supply	RCS pressure decrease below 95.0 kgf/cm^2
		RCP trip, start of RCS coast-down Trip of MFW pump	Postulated loss of normal power supply
7.0	8.0	Closure of turbines stop valves	Loss of vacuum in turbine condensers is assumed 5 s after loss of power supply
5.0	6.0	ECCS safeguard actuation, signal to start emergency DG	Postulated loss of normal power supply + 3 s delay
20.0	20.0	Start of ECCS HA injection to the reactor	Decrease of RCS pressure below 60.0 kgf/cm^2
20.0	21.0	Start of emergency DG sequential loading program, HPIS and LPIS pumps start	15 s delay for DG start-up after ECCS safeguard actuation
25.0	20.5	Maximal secondary circuit pressure 52.0/51.5 kgf/cm^2 is reached	
40.0	41.0	Start of HPIS-1,2,3 injection	Delay after sequential loading program actuation (for pumps start-up and transport delay)
73.0	73.0	Start of AFW pump no.1 injection to SG	35 s after sequential loading program + delay for pump start-up and transport delay. Only one AFW pump (no.1) is powered from emergency DG
160-170	130	End of ECCS HA injection	HA depletion
250.0	425.0	Start of stable LPIS injection to RCS	RCS pressure decrease below LPIS pump shut-off head
900.0	900.0	End of calculation	Stable core cooldown, stabilization of the main reactor parameters

Break of PRZ surge line with a diameter of 277 mm results in large break LOCA that causes fast decrease of RCS pressure (Figure 3-193) and PRZ level (Figure 3-194). After IE occurrence, due to RCS pressure decrease down to 95 kgf/cm^2, a scram signal is generated at 2.0/3.0 s.

Postulated loss of normal power supply results in a trip of the equipment powered from correspondent busbars, namely, RCP, MFW pumps, PRZ heaters, BRU-K, and causes actuation of ECCS safeguard with a start of emergency DG. Turbine stop valves are closed by the "Loss of vacuum in turbine condensers" signal after loss of normal power supply. This causes temporary increase of SG pressure (Figure 3-217 – Figure 3-222) with a maximal value reached at 25/20 s, however BRU-A actuation setpoints are not reached due to intensive energy loss via the break. Nearly at the same time RCS pressure decreases below 60 kgf/cm^2 and HA injection to the reactor begins (Figure 3-243 – Figure 3-246).

15 s after start of DG (delay for DG start-up) the sequential loading program is actuated with connection of equipment to emergency power supply busbars according to a design algorithm, and at 40.0/41.0 s after IE occurrence (with postulated delay for HPIS pump start-up and transport delay) HPIS starts to inject boric acid to the primary circuit (Figure 3-237 – Figure 3-239).

At 250.0/425.0 s of the accident RCS pressure decreases below LPIS pump shut-off head and LPIS injection to the primary circuit is started (Figure 3-240 – Figure 3-242). Combined operation of HPIS and LPIS pumps compensates coolant loss via the break and to 600 s of calculation the majority of the primary and secondary circuit parameters are stabilized. Decay heat removal is provided by heat-up of the cold coolant injected by LPIS and HPIS, and energy loss via a break. Secondary circuit pressure decreases due to SG cooling down by the primary circuit.

During the accident, a stable decrease of the maximal cladding temperature from the initial value at normal power operation to ~125 °C is observed (Figure 3-246).

The plots of the main parameters of calculation are presented below on Figure 3-192 – Figure 3-249.

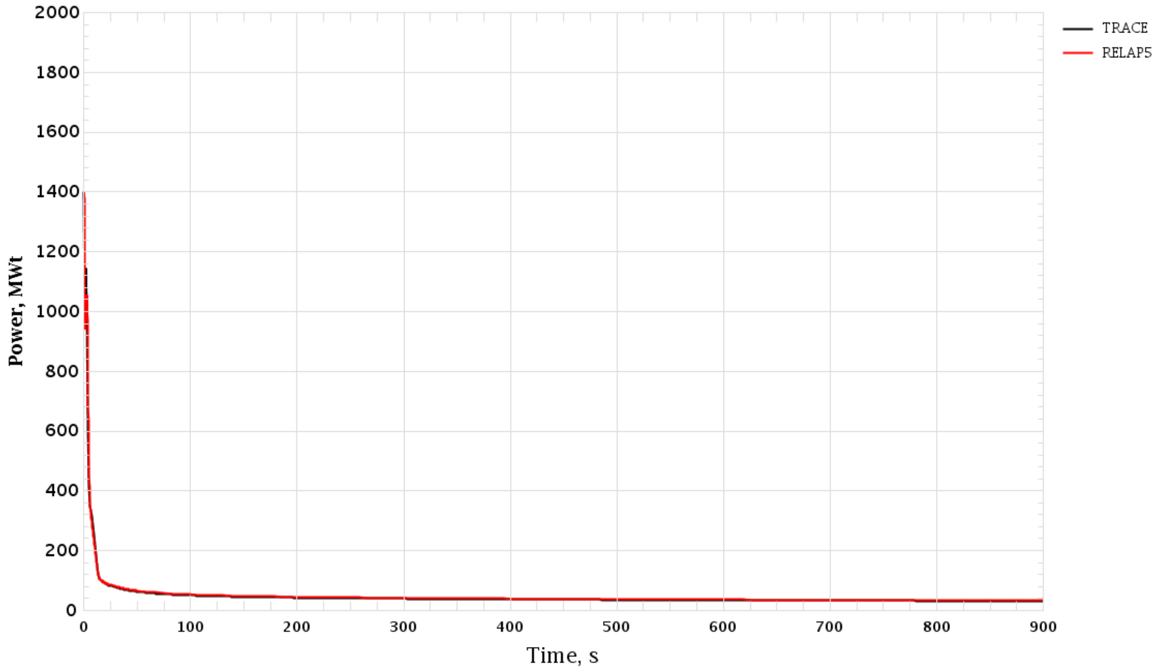

Figure 3-192 PRZ Surge Line Break. Core Thermal Power

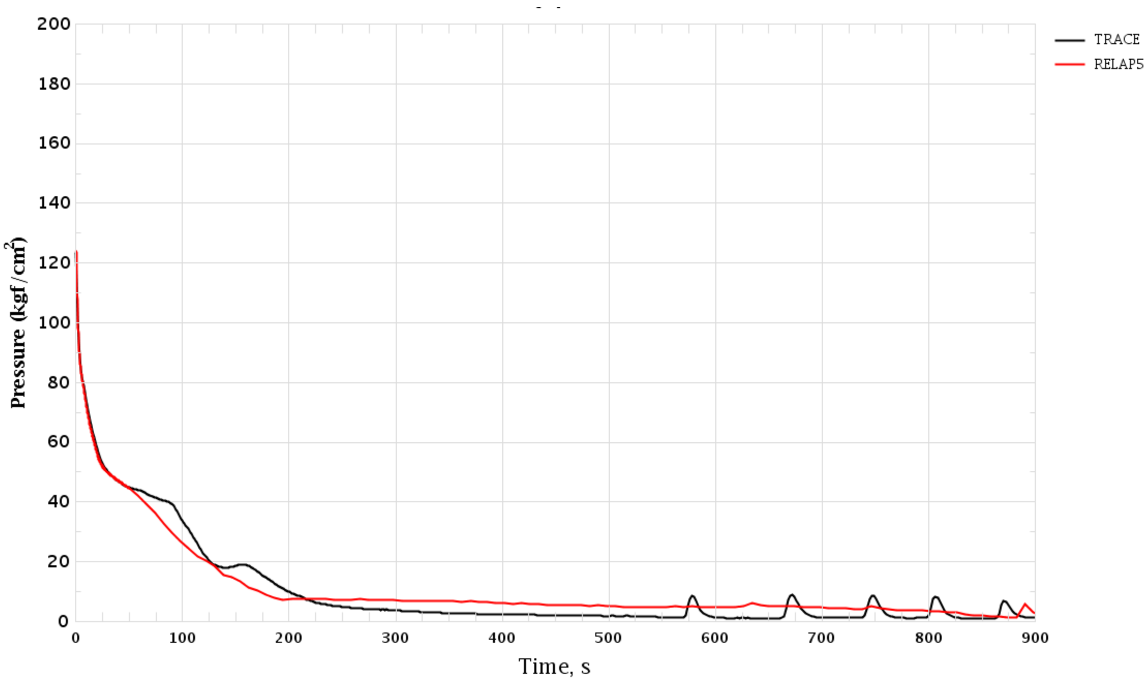

Figure 3-193 PRZ Surge Line Break. RCS Pressure

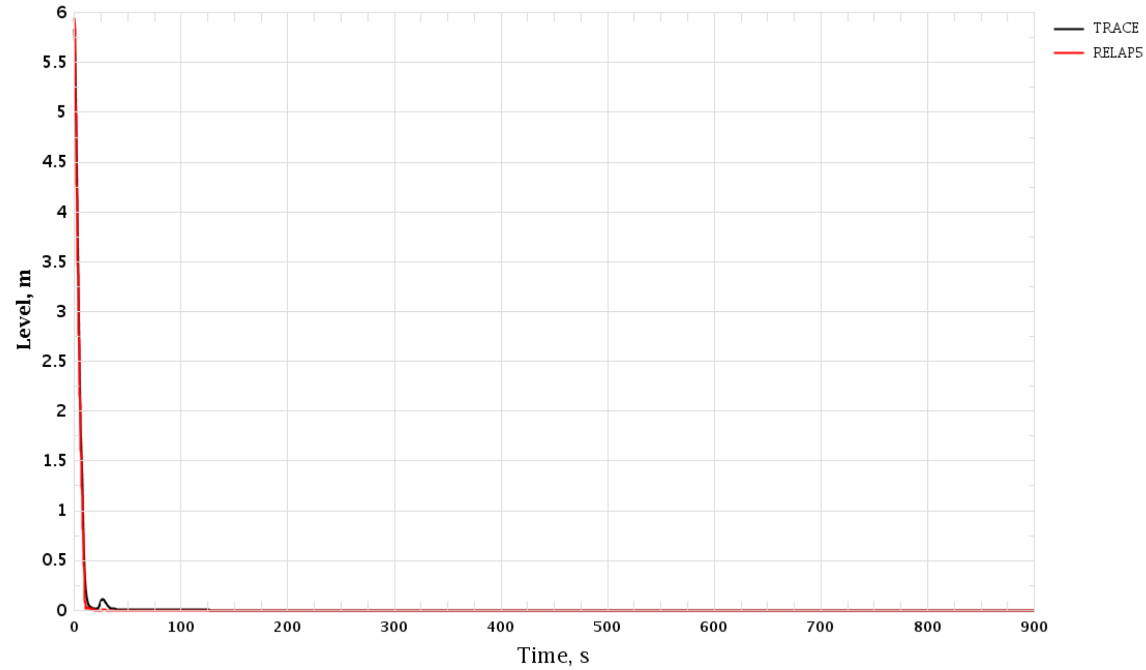

Figure 3-194 PRZ Surge Line Break. Pressurizer Level

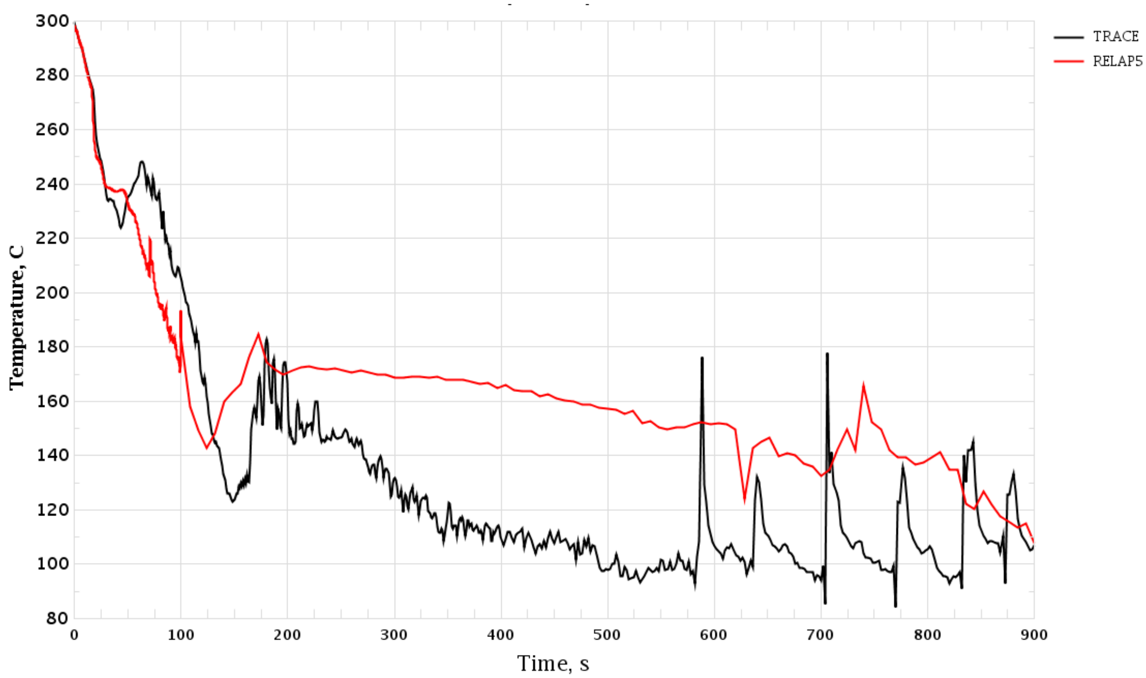

Figure 3-195 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 1

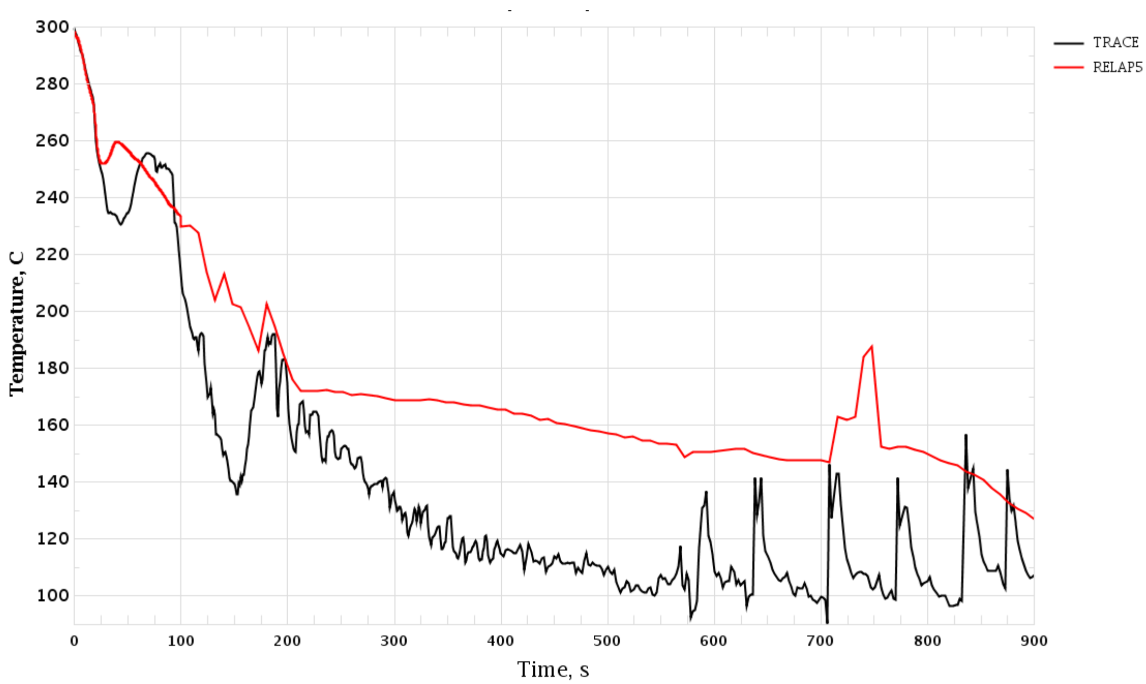

Figure 3-196 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 2

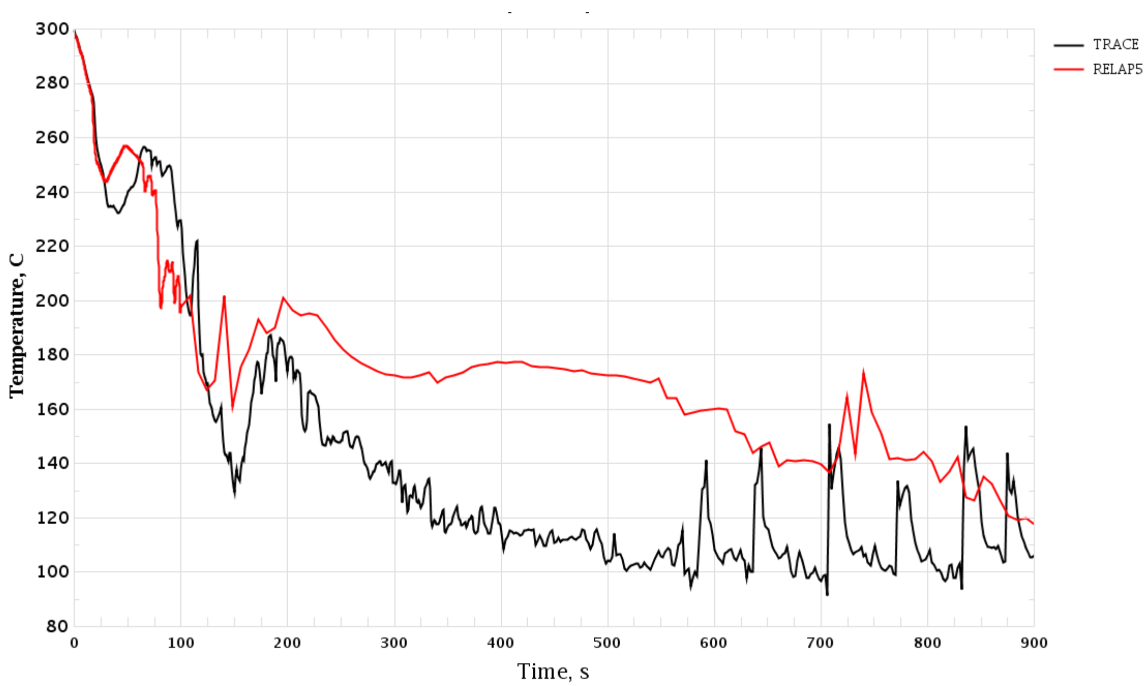

Figure 3-197 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 3

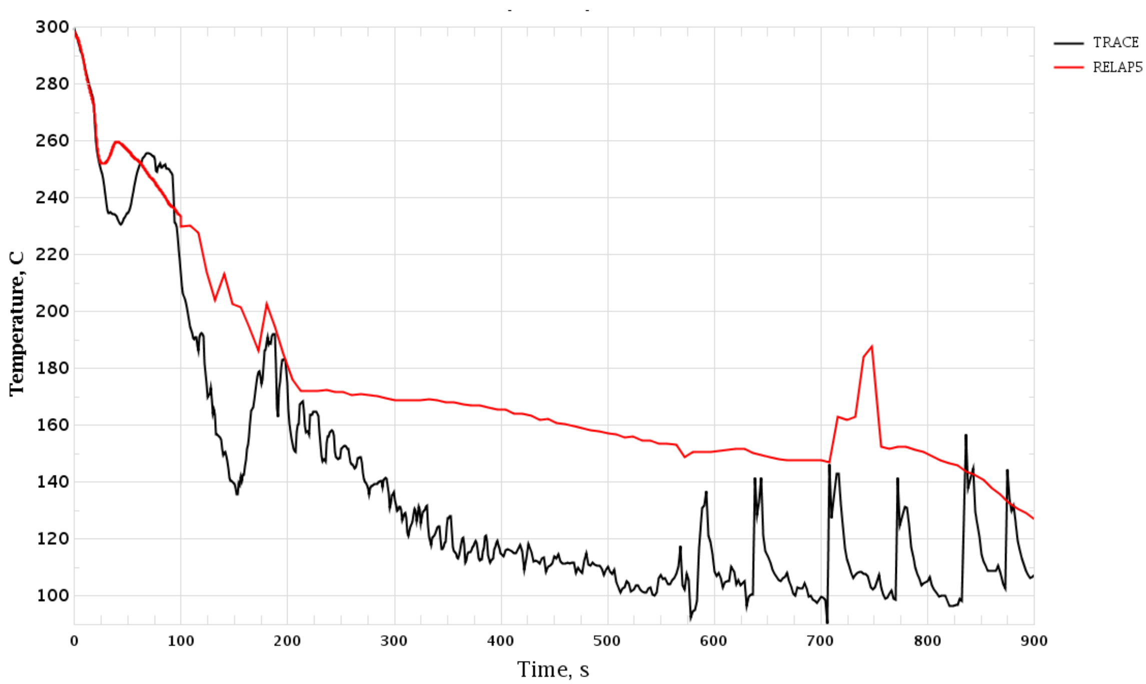

Figure 3-198 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 4

Figure 3-199 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 5

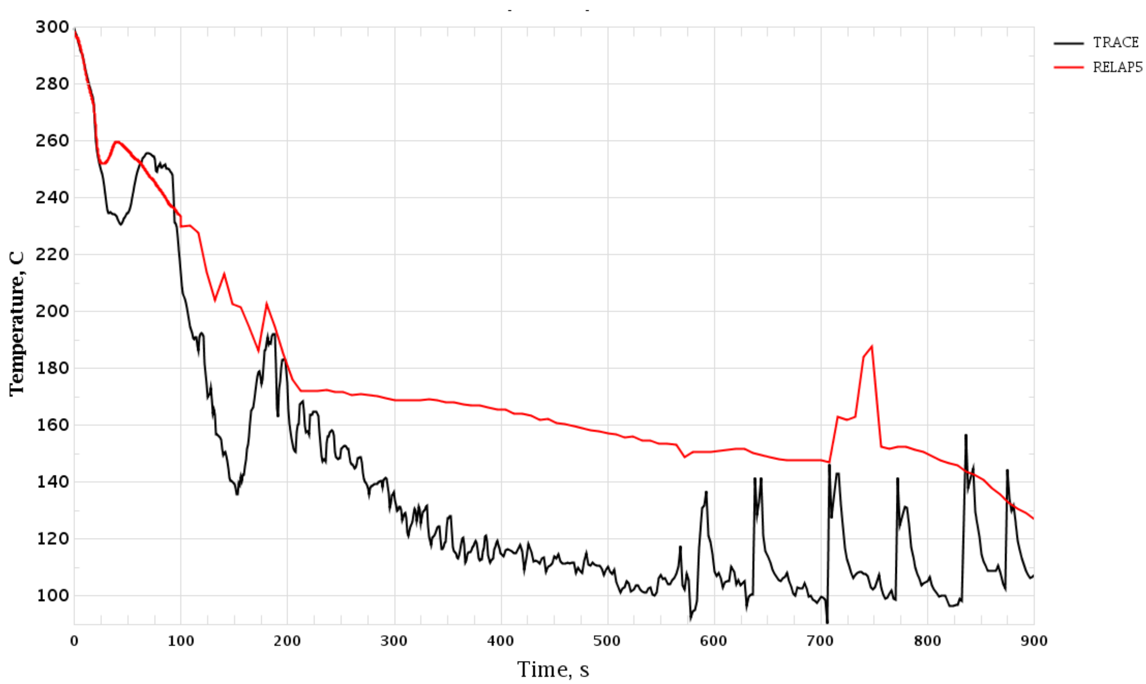

Figure 3-200 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 6

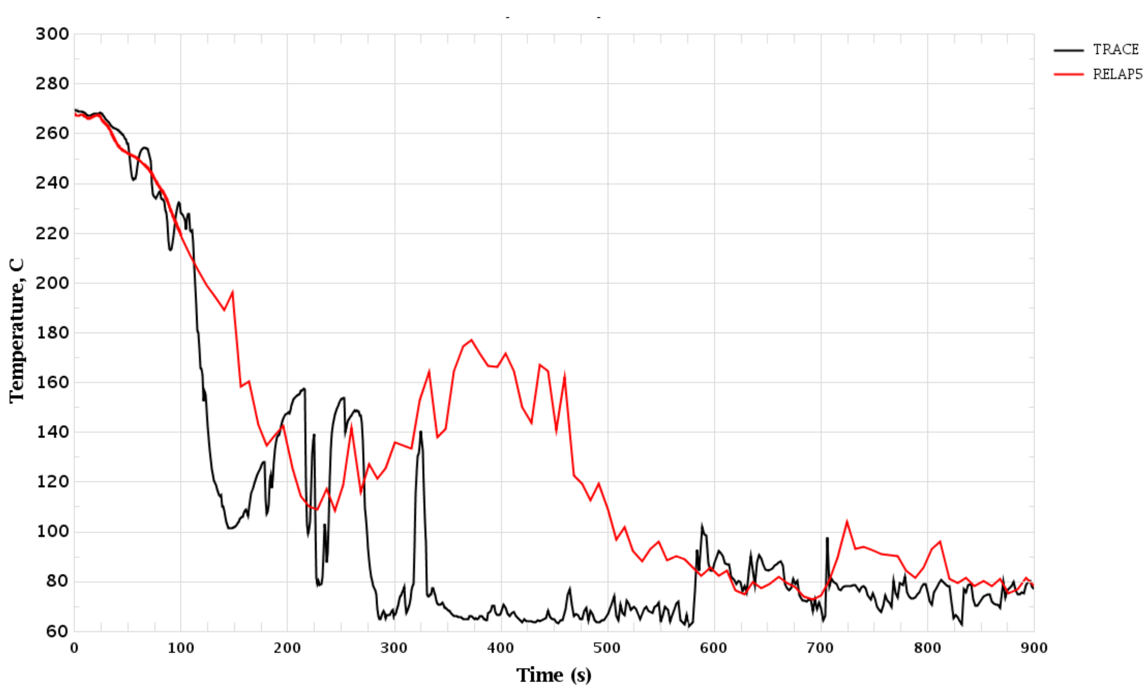

Figure 3-201 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 1

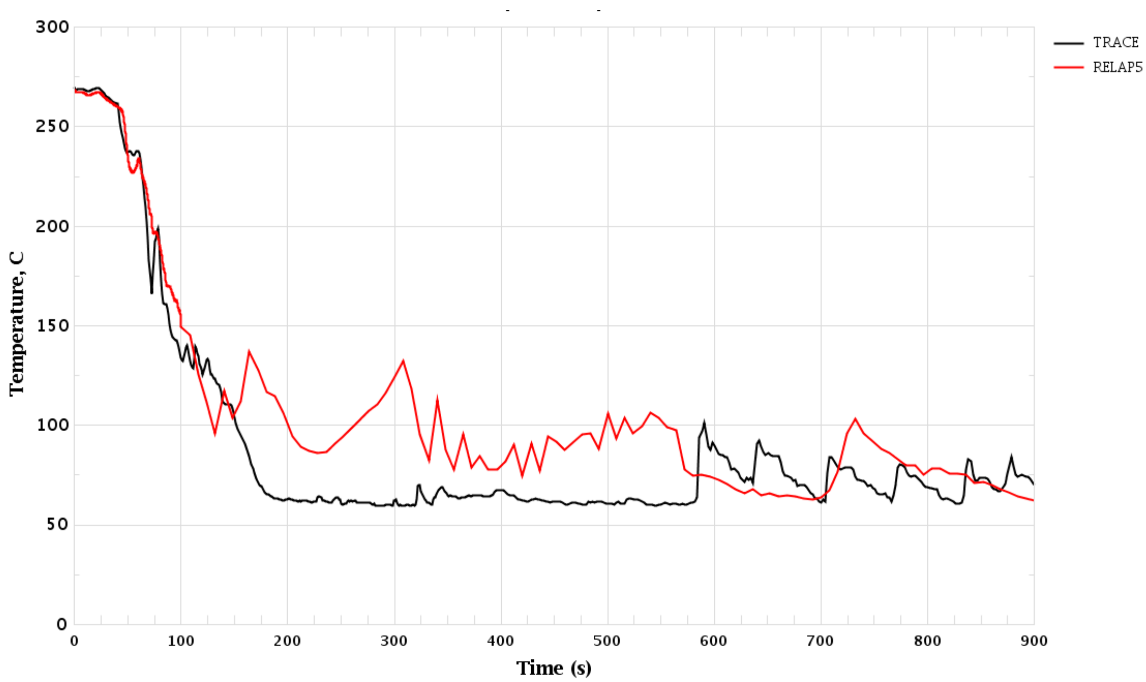

Figure 3-202 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 2

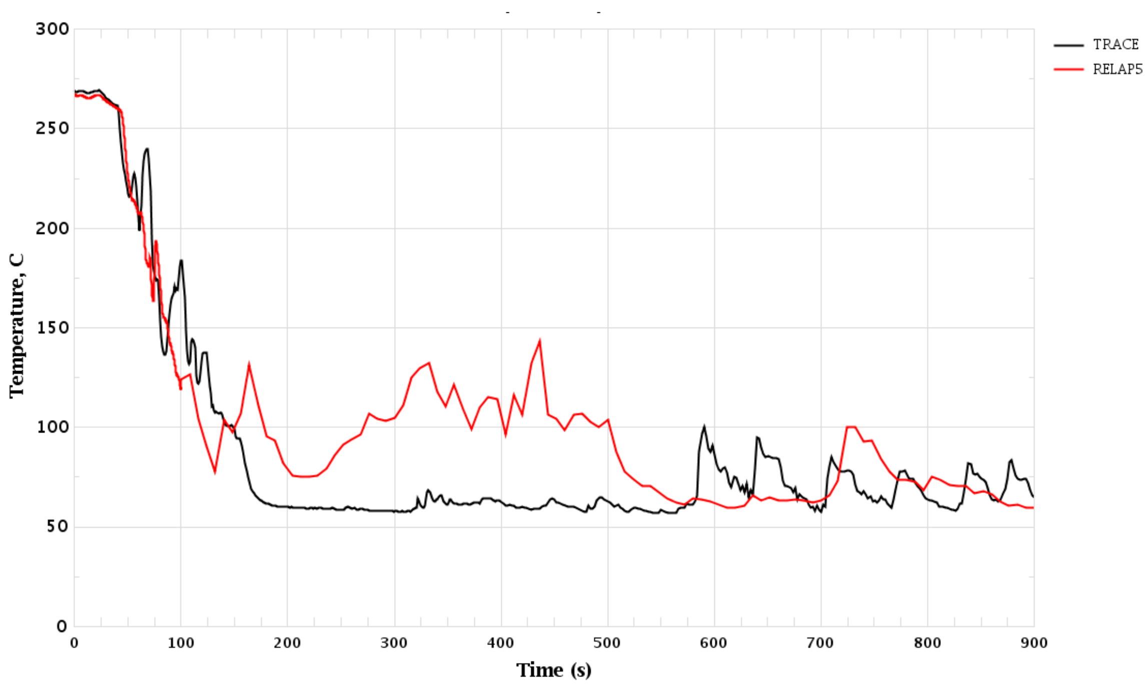

Figure 3-203 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 3

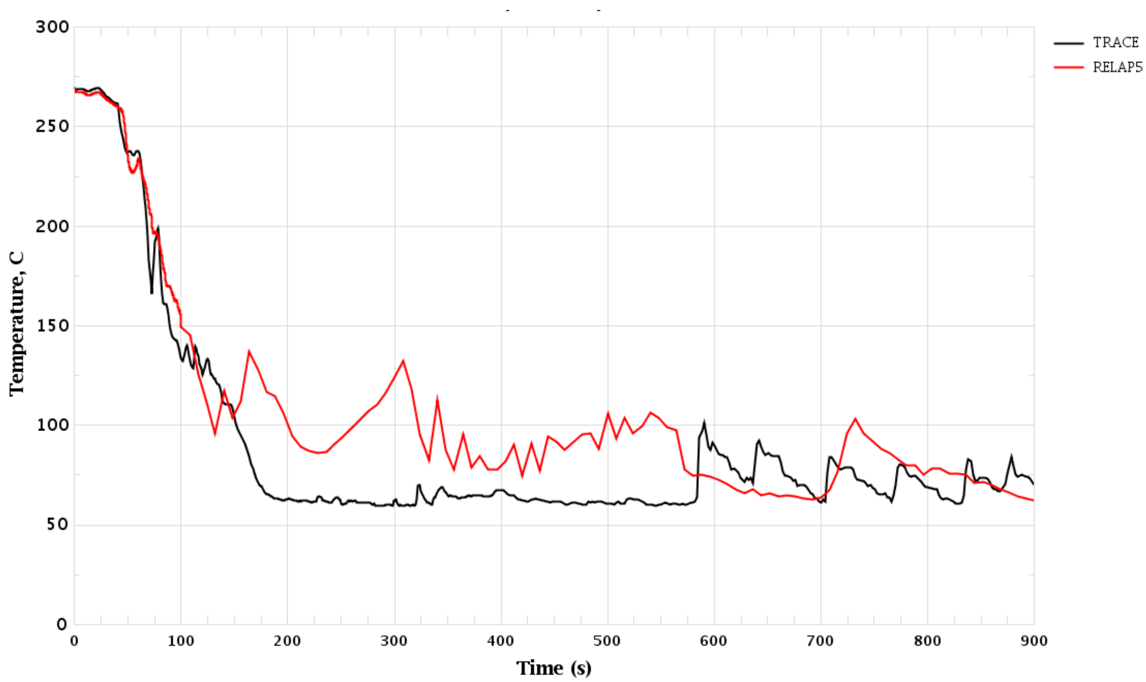

Figure 3-204 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 4

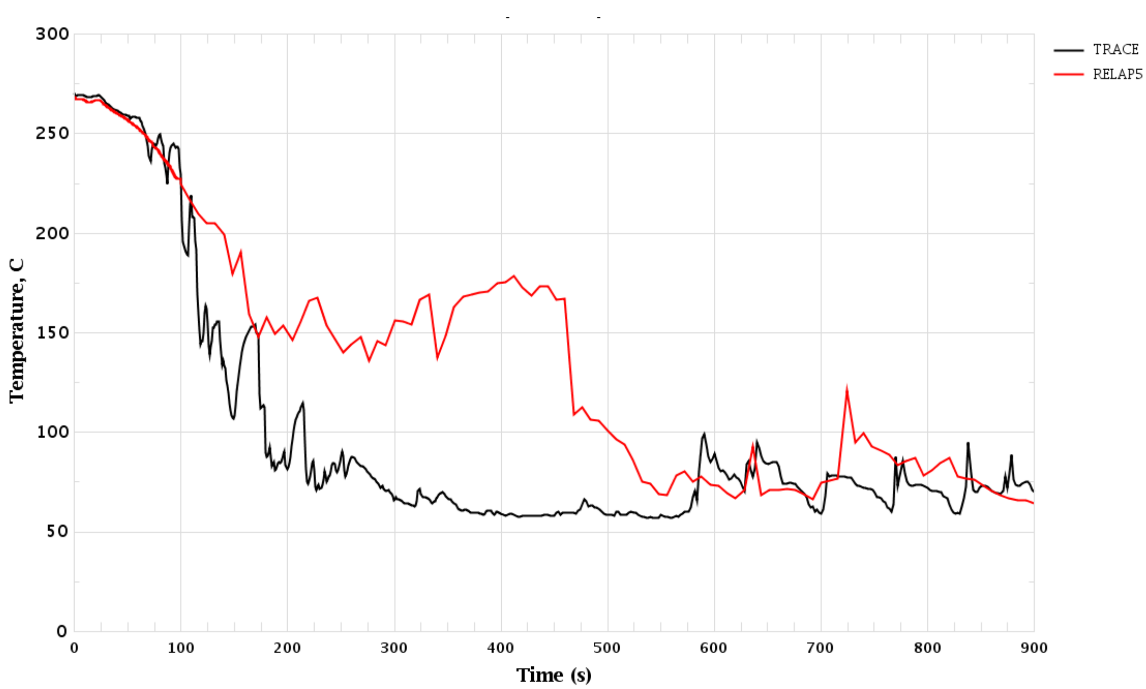

Figure 3-205 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 5

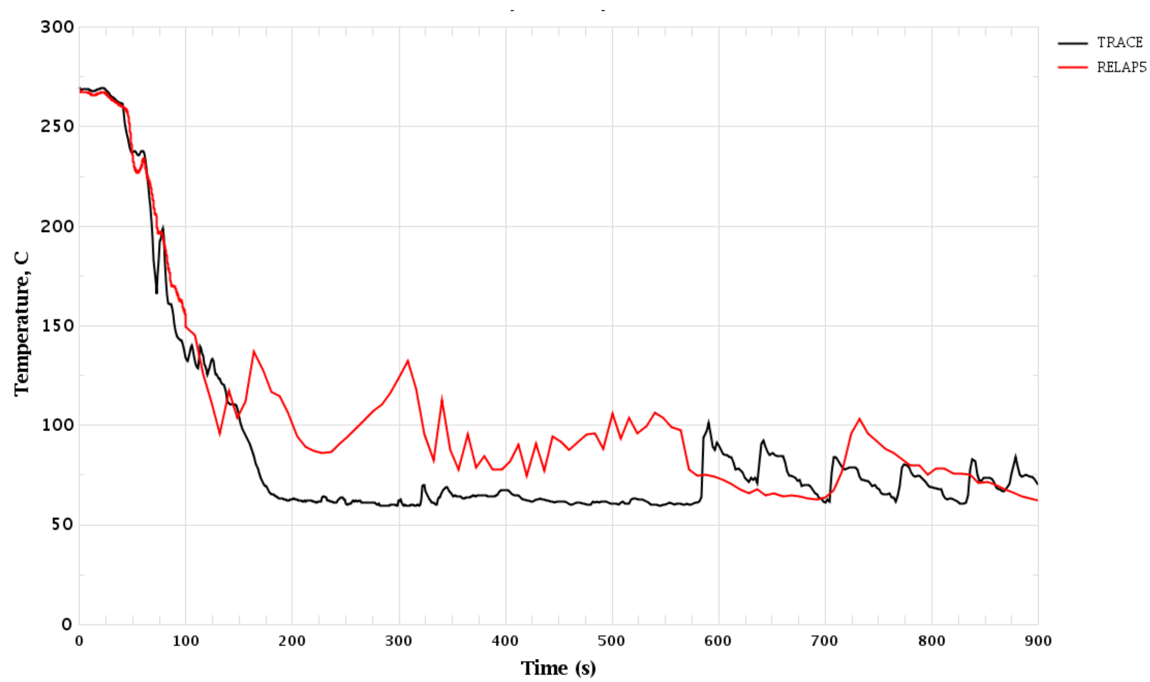

Figure 3-206 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 6

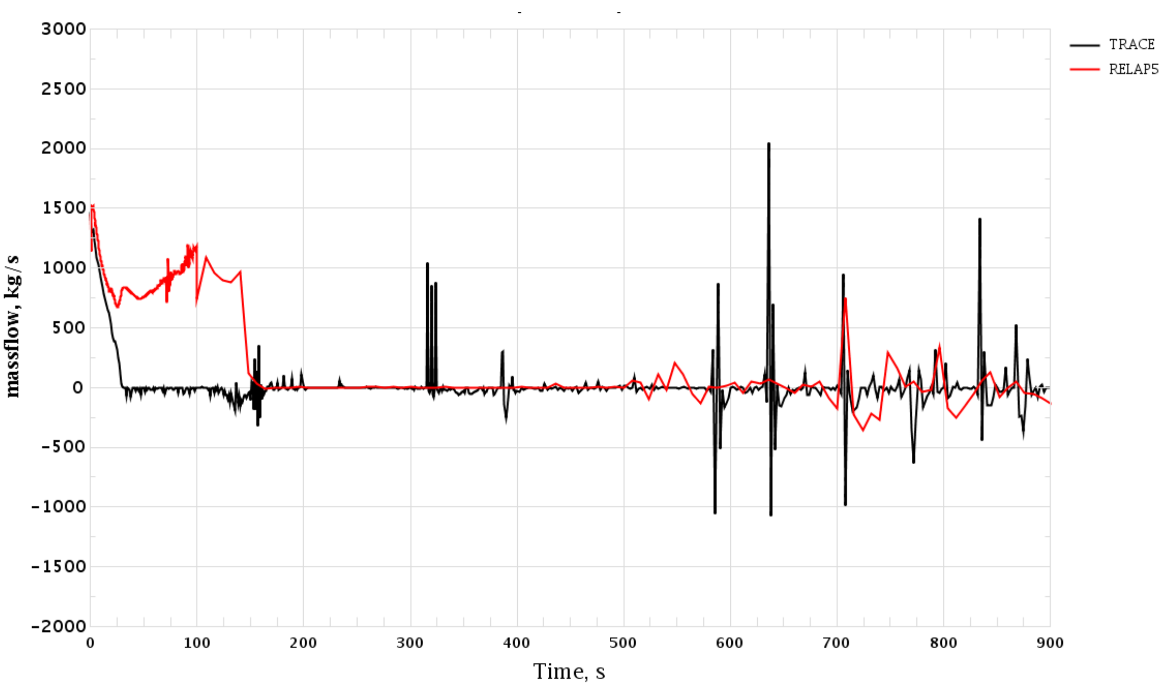

Figure 3-207 PRZ Surge Line Break. RCS Loop 1 Mass Flow Rate

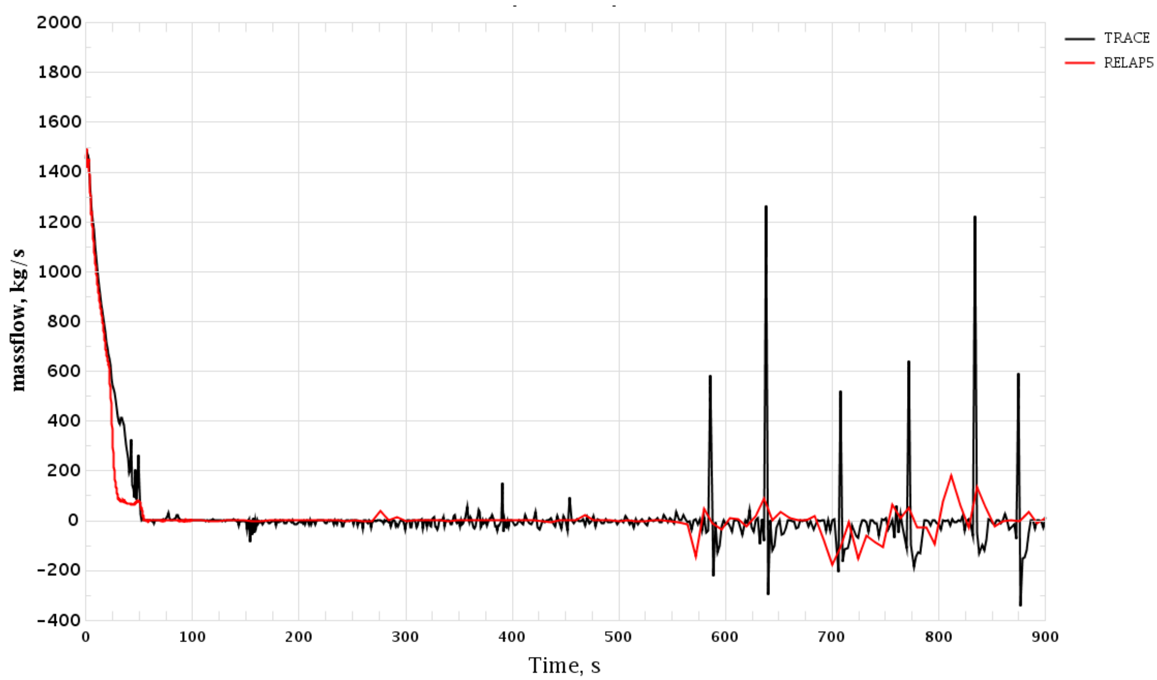

Figure 3-208 PRZ Surge Line Break. RCS Loop 2 Mass Flow Rate

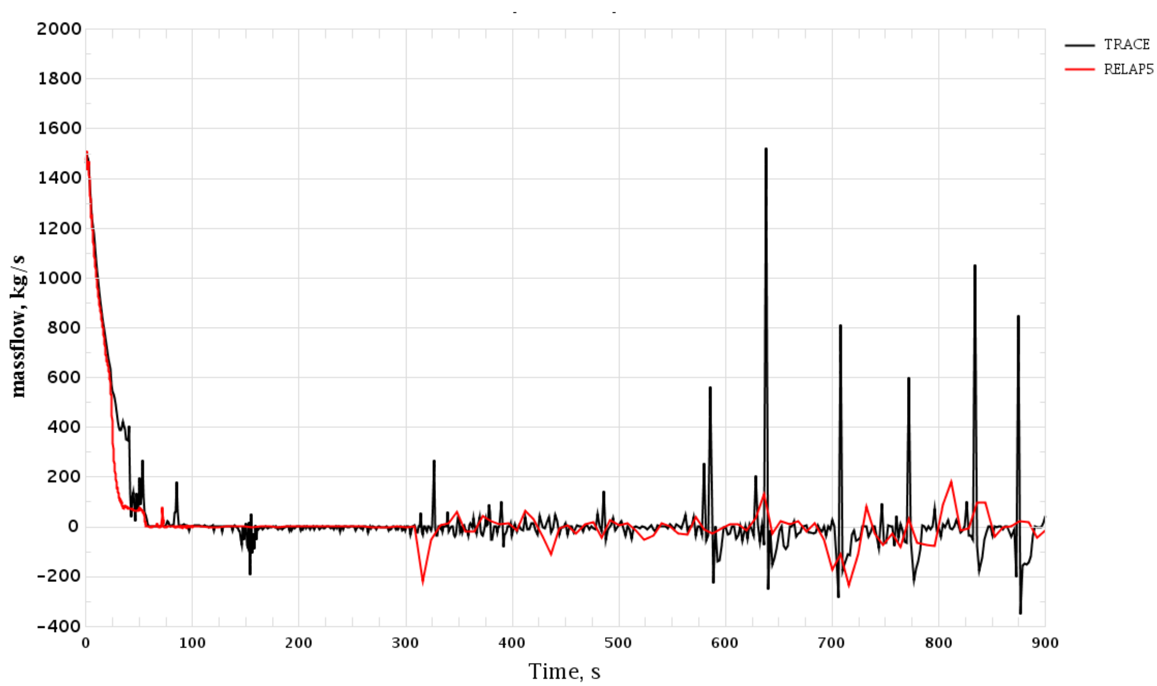

Figure 3-209 PRZ Surge Line Break. RCS Loop 3 Mass Flow Rate

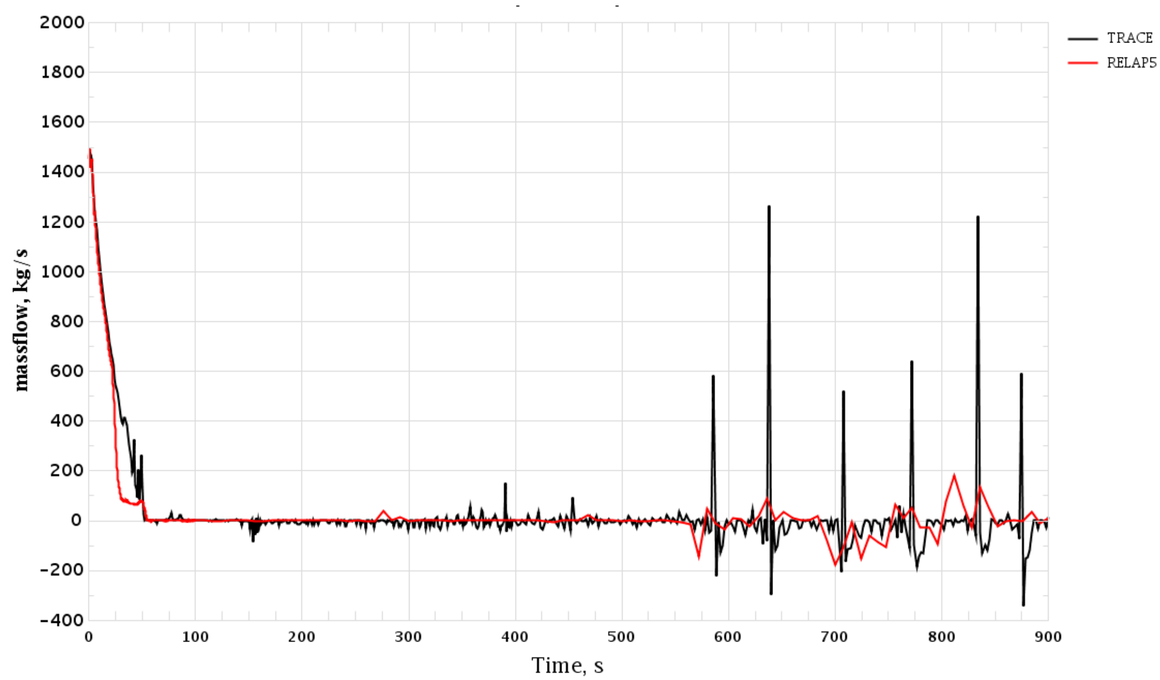

Figure 3-210 PRZ Surge Line Break. RCS Loop 4 Mass Flow Rate

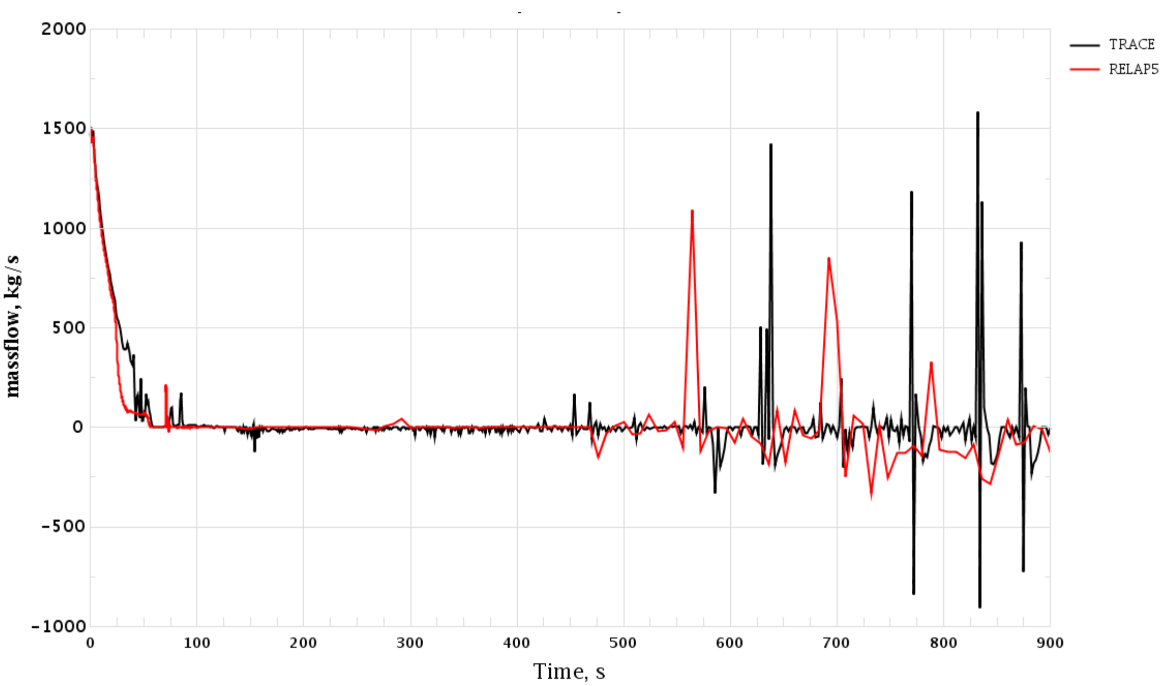

Figure 3-211 PRZ Surge Line Break. RCS Loop 5 Mass Flow Rate

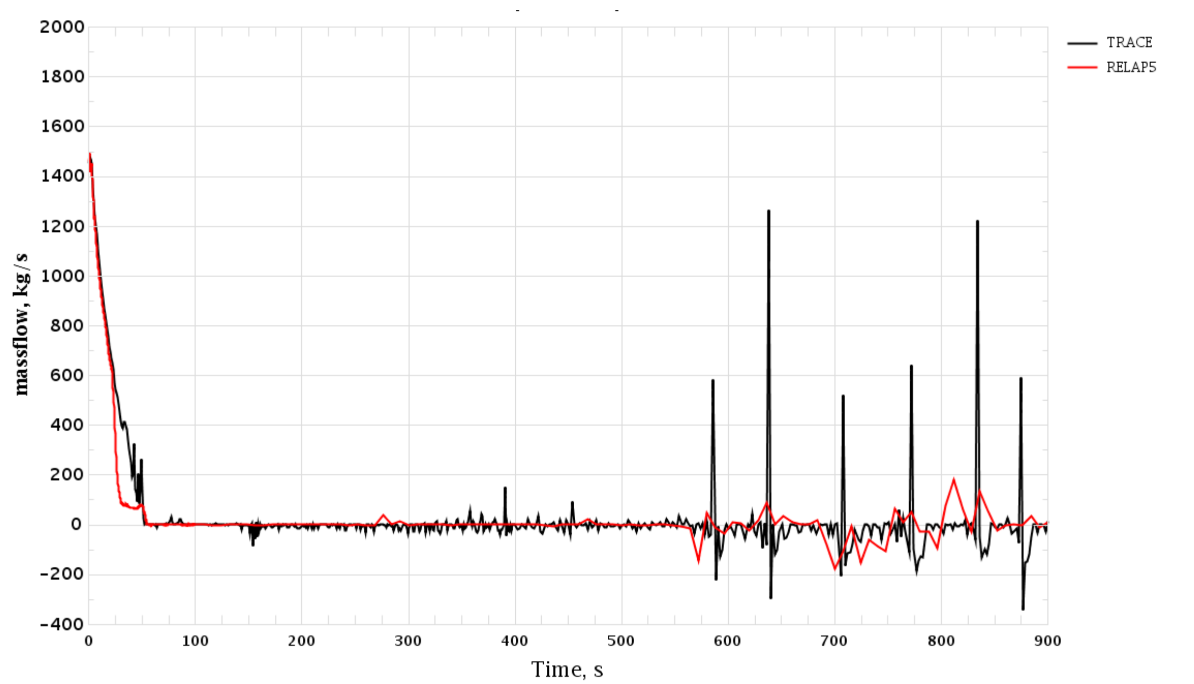

Figure 3-212 PRZ Surge Line Break. RCS Loop 6 Mass Flow Rate

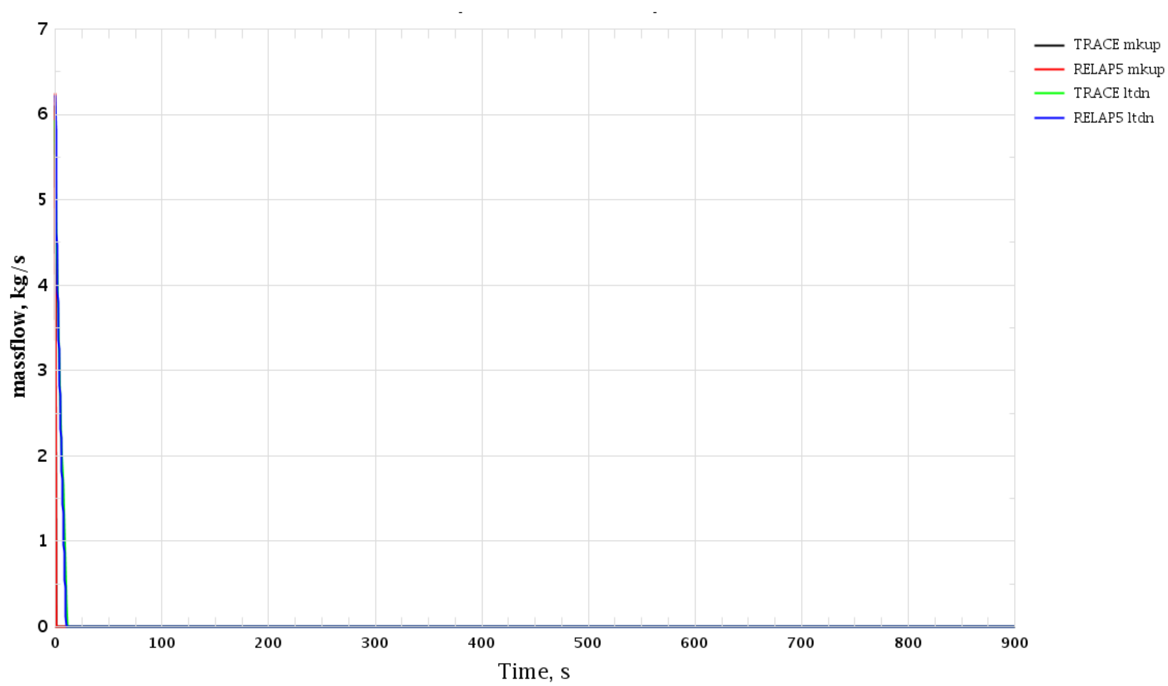

Figure 3-213 PRZ Surge Line Break. Make-Up and Let-Down Mass Flow

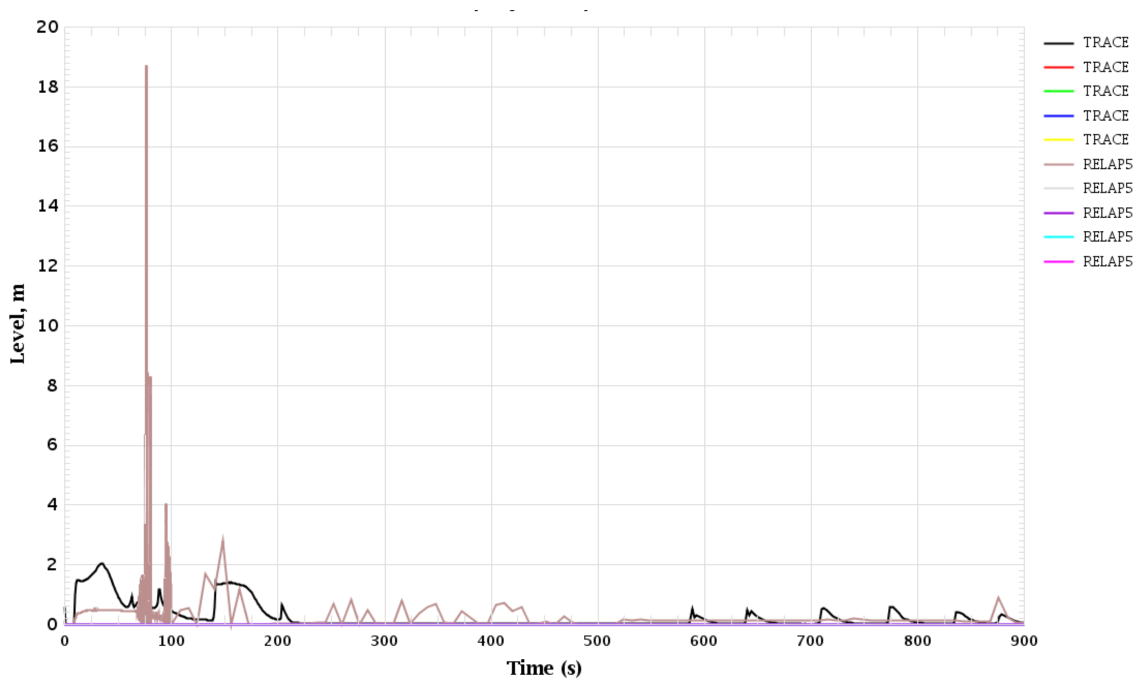

Figure 3-214 PRZ Surge Line Break. PRZ Spray Mass Flow Rate

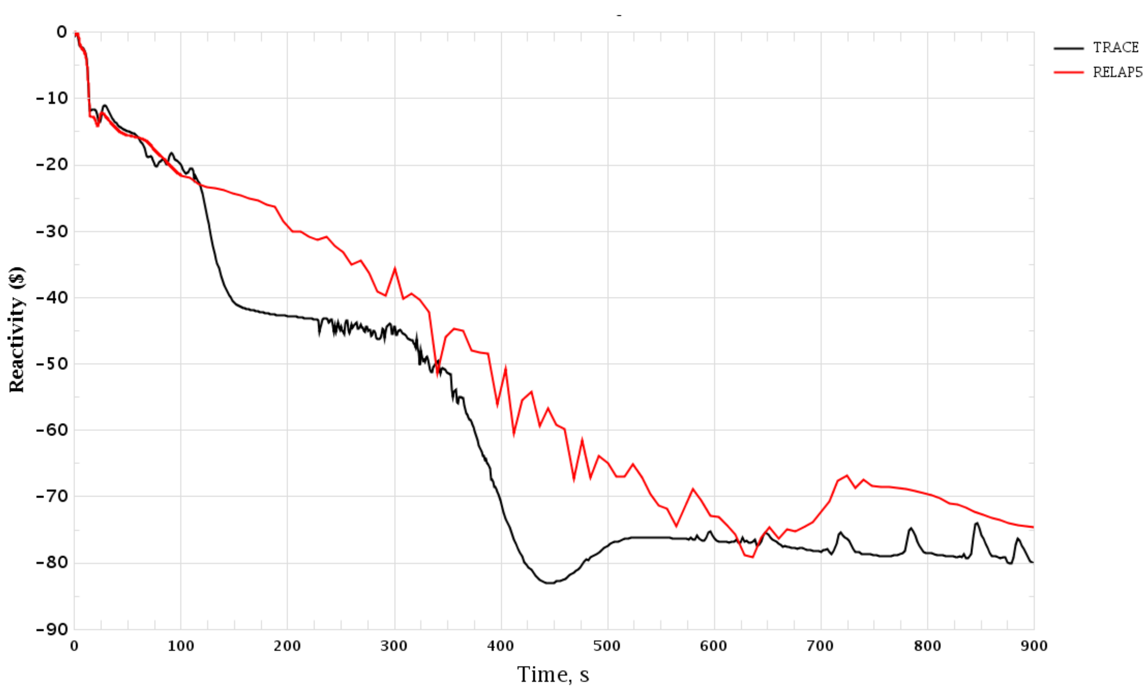

Figure 3-215 PRZ Surge Line Break. Core Reactivity

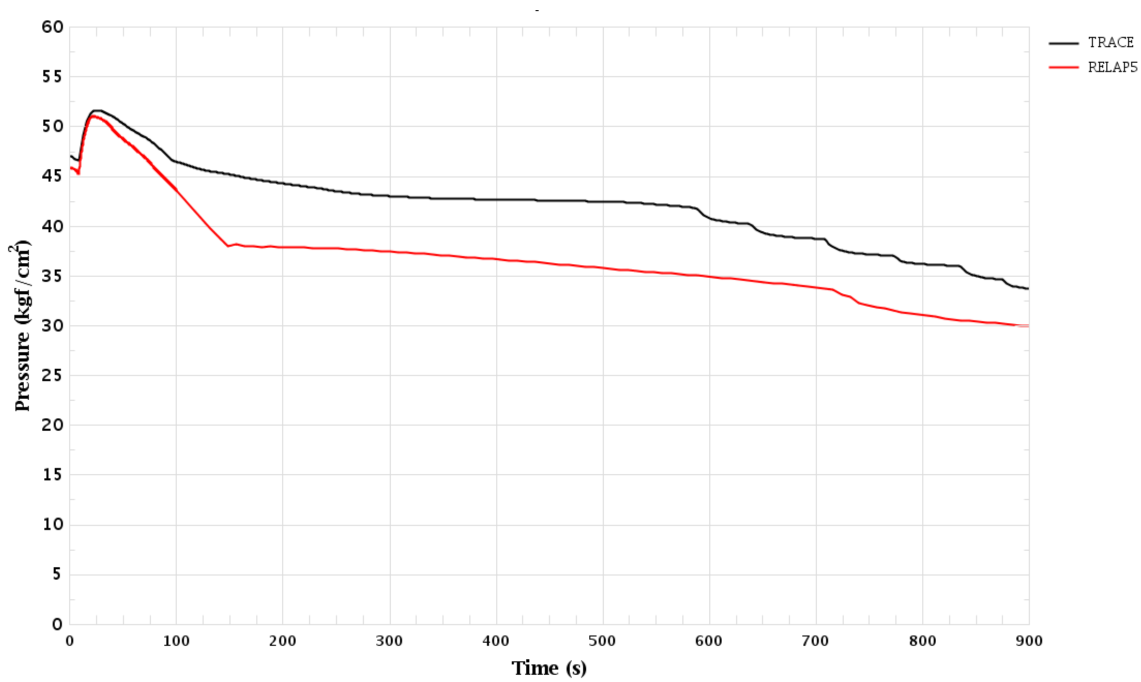

Figure 3-216 PRZ Surge Line Break. SG-1 Pressure

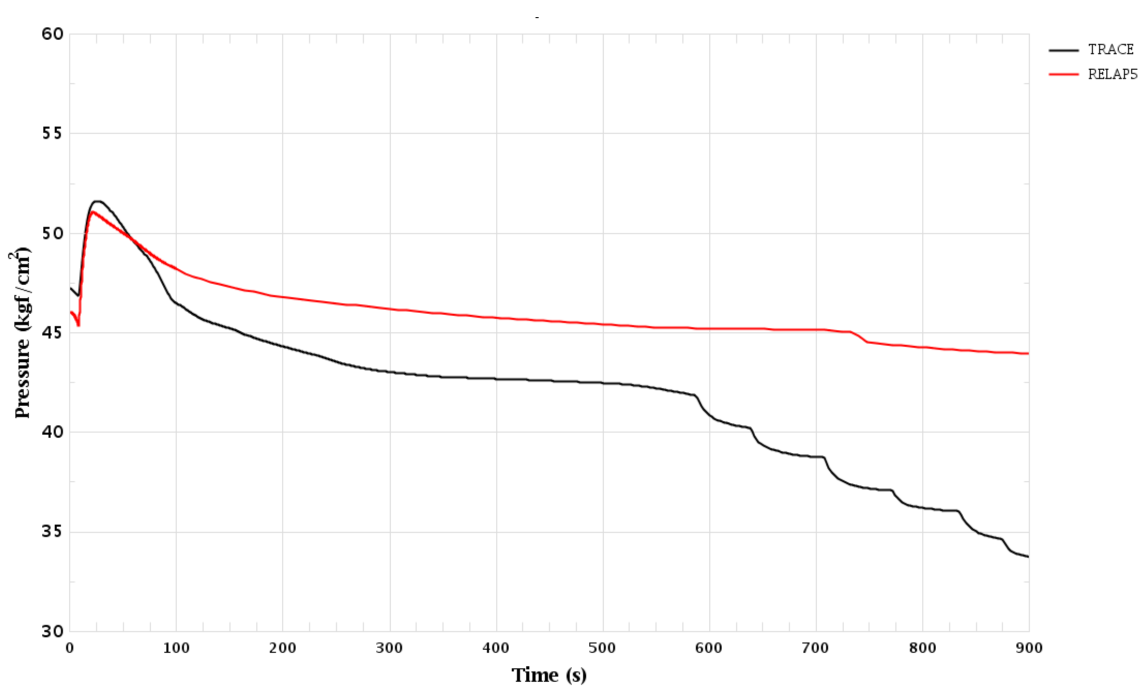

Figure 3-217 PRZ Surge Line Break. SG-2 Pressure

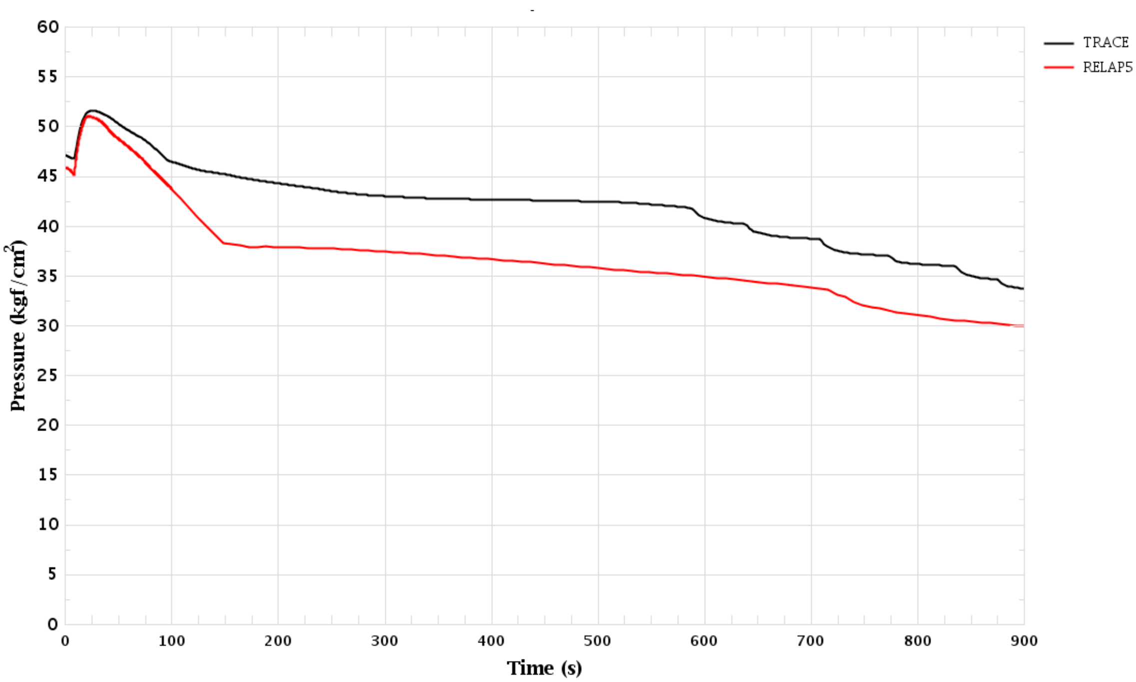

Figure 3-218 PRZ Surge Line Break. SG-3 Pressure

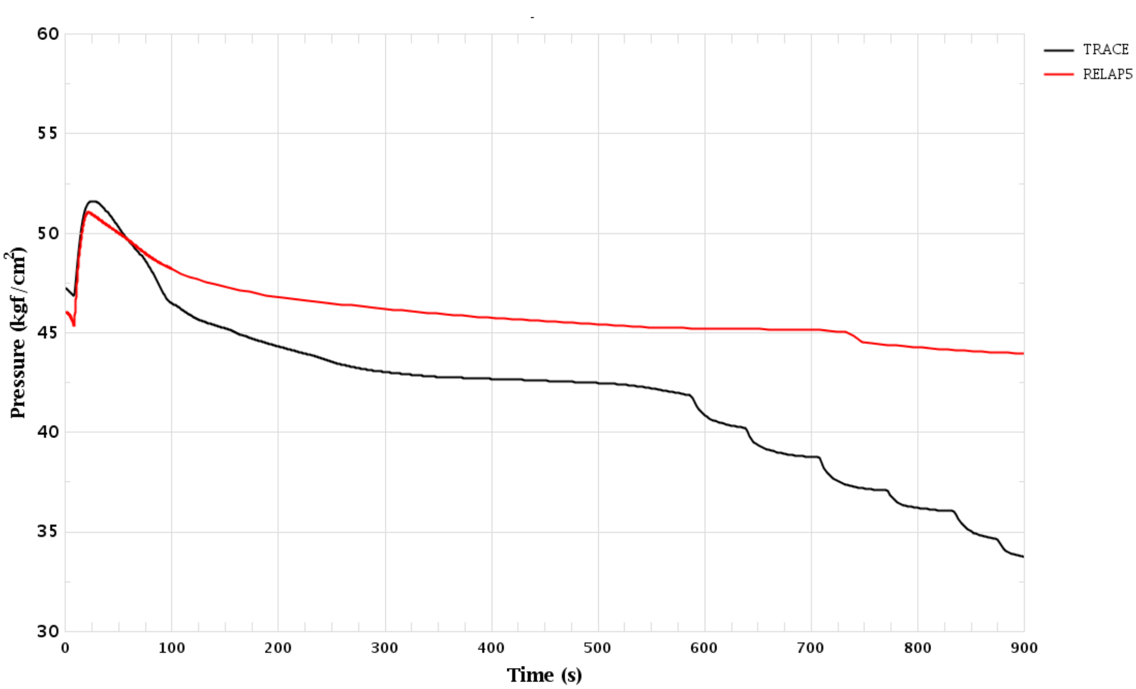

Figure 3-219 PRZ Surge Line Break. SG-4 Pressure

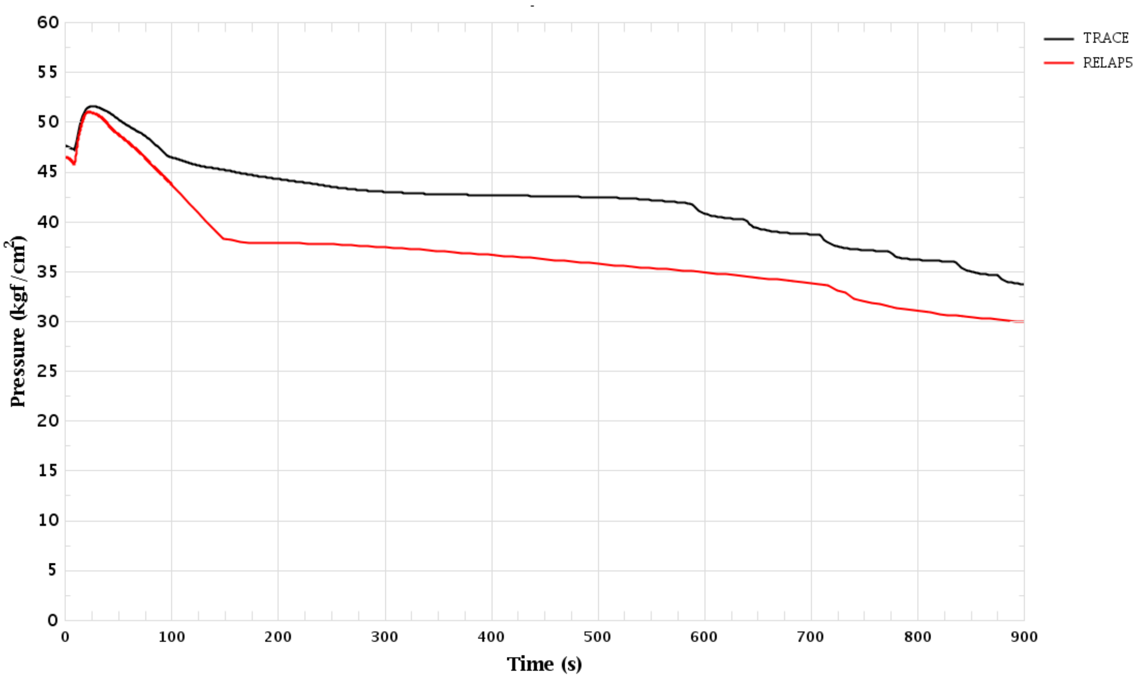

Figure 3-220 PRZ Surge Line Break. SG-5 Pressure

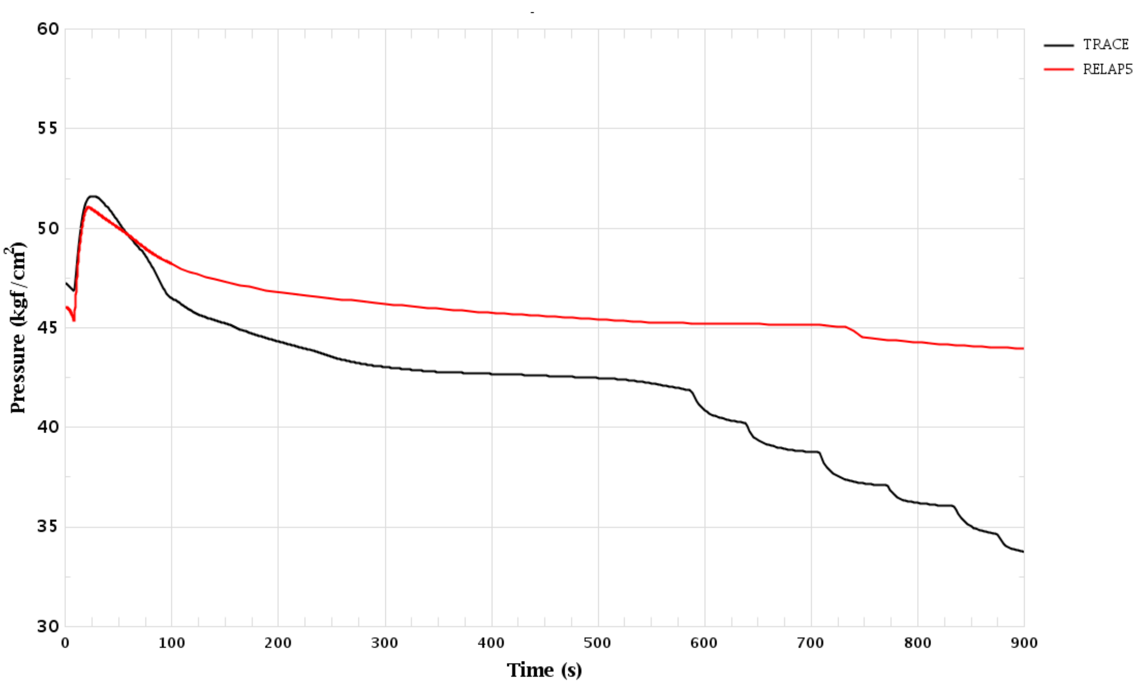

Figure 3-221 PRZ Surge Line Break. SG-6 Pressure

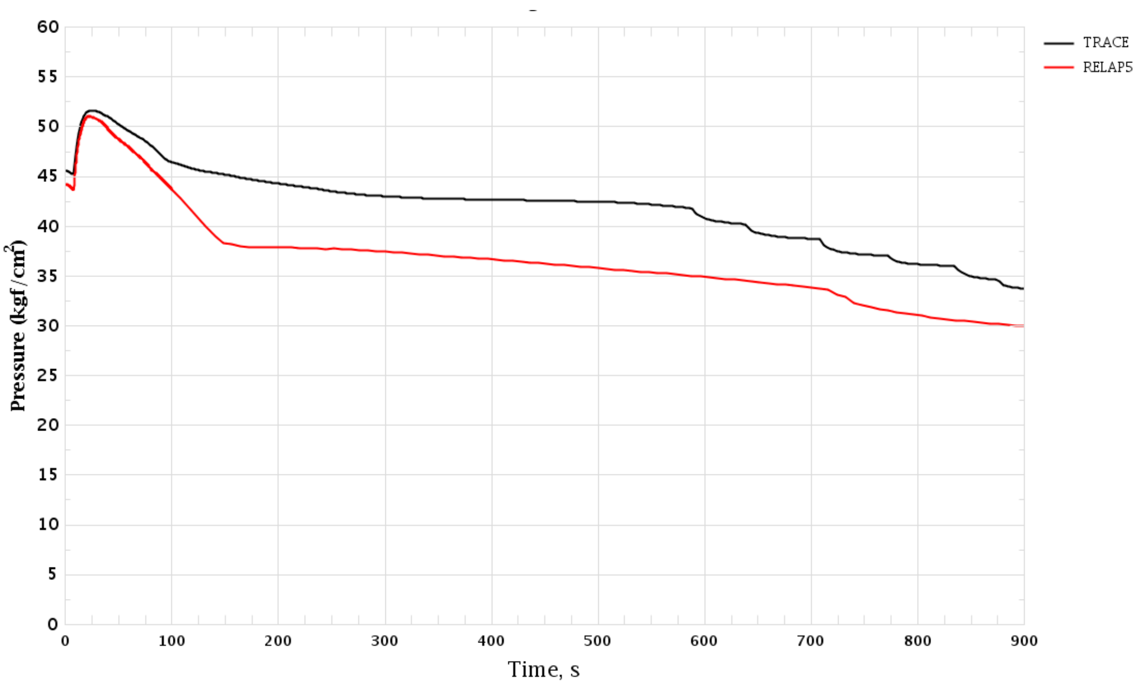

Figure 3-222 PRZ Surge Line Break. MSH-1 Pressure

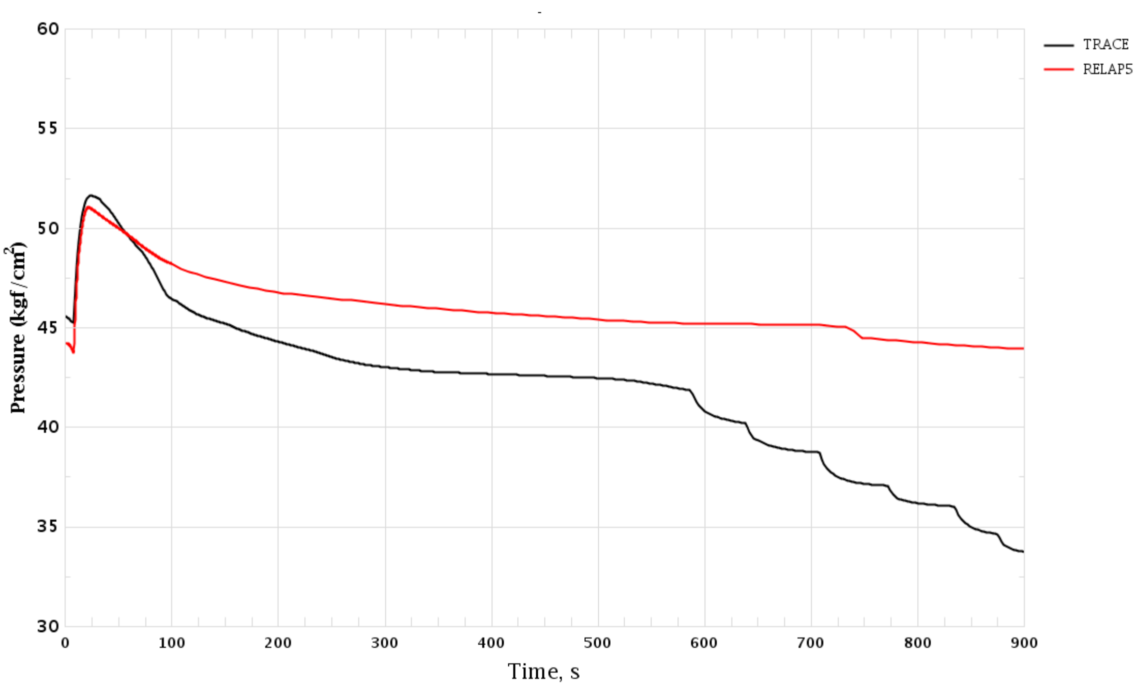

Figure 3-223 PRZ Surge Line Break. MSH 2 Pressure

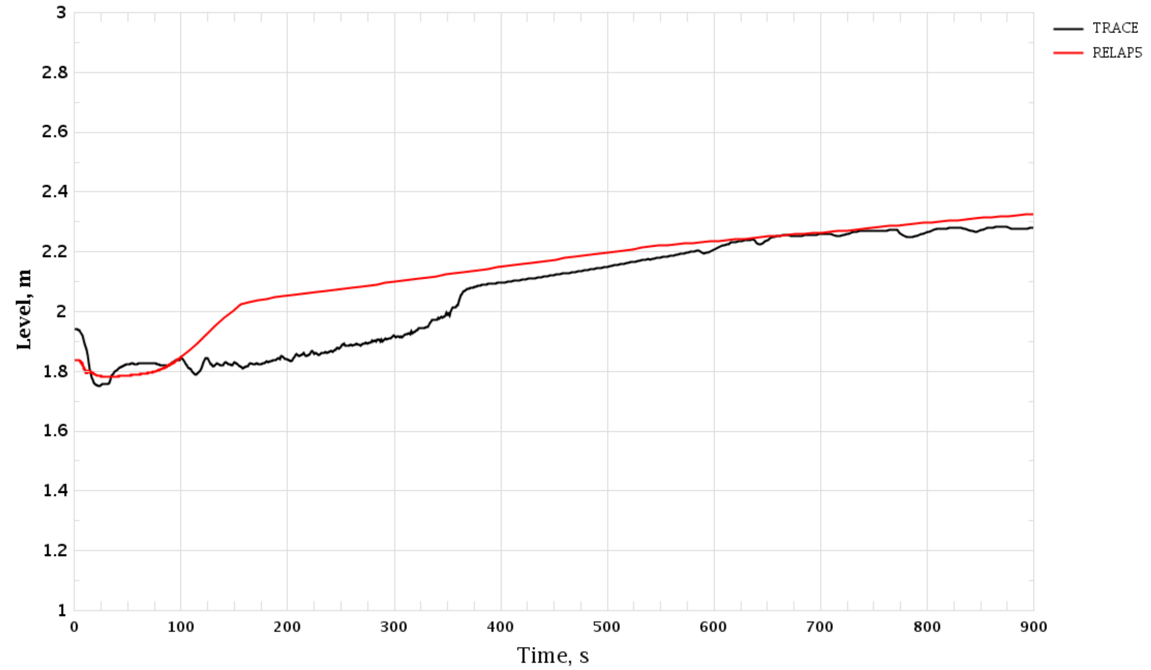

Figure 3-224 PRZ Surge Line Break. SG-1 Level (Wide Range)

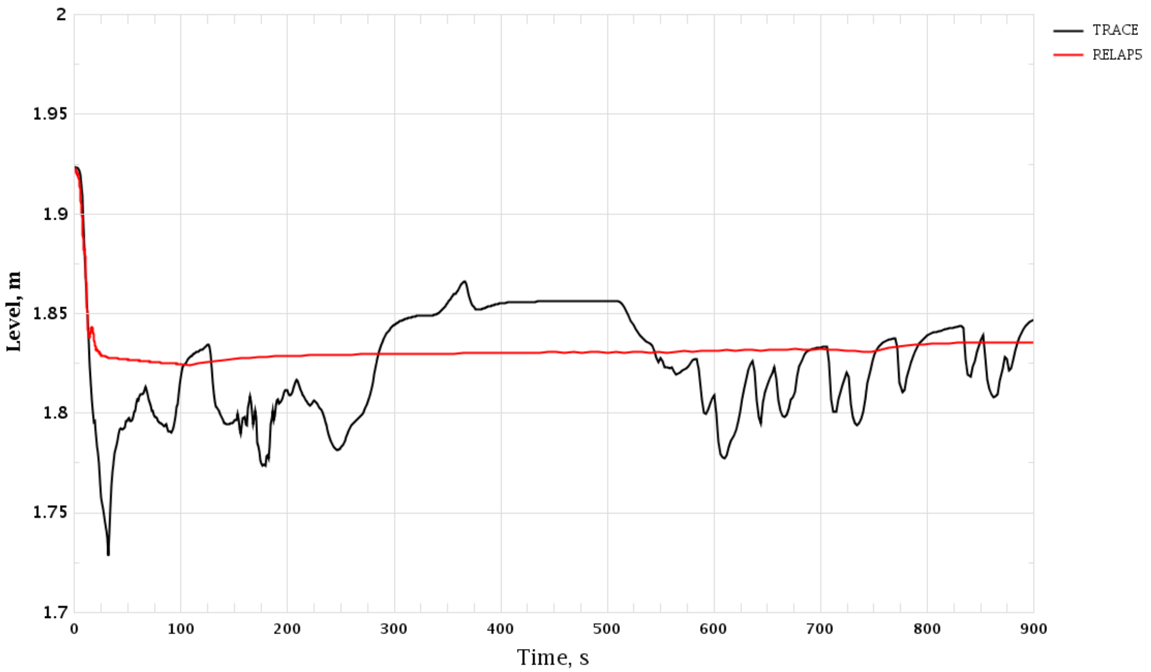

Figure 3-225 PRZ Surge Line Break. SG-2 Level (Wide Range)

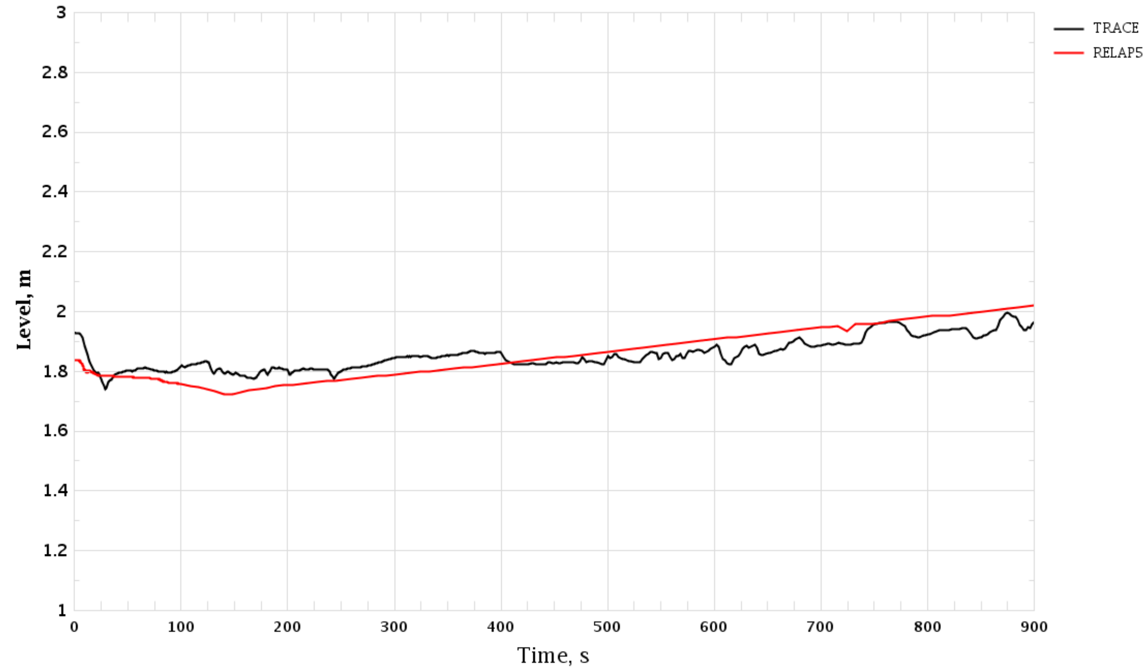

Figure 3-226 PRZ Surge Line Break. SG-3 Level (Wide Range)

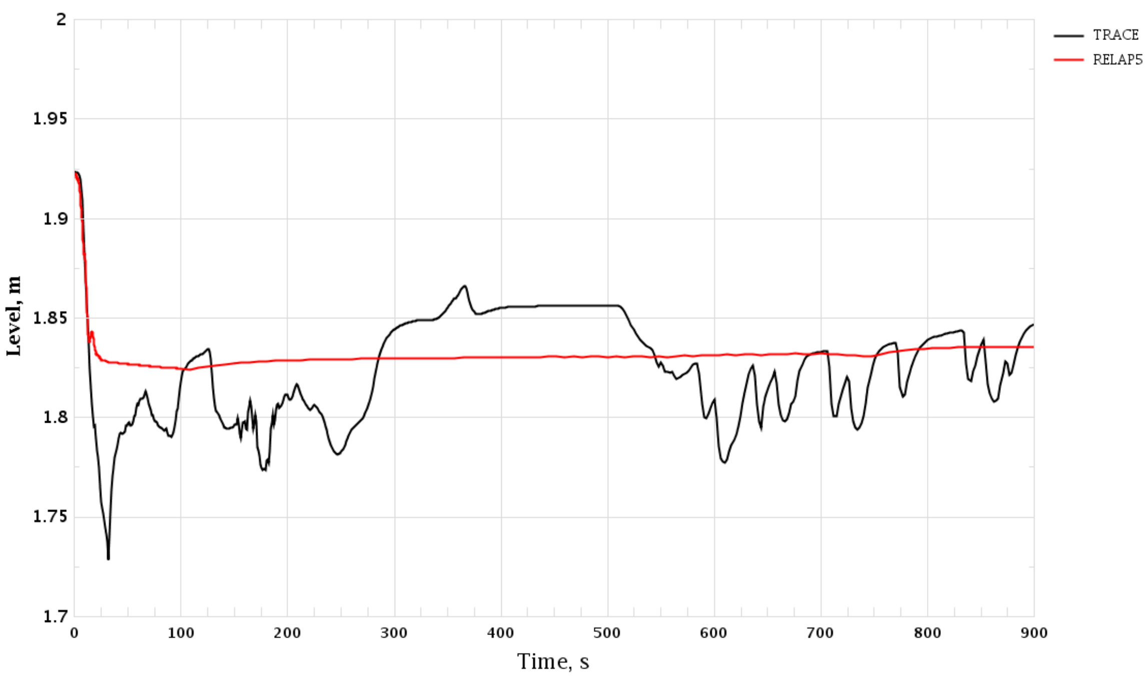

Figure 3-227 PRZ Surge Line Break. SG-4 Level (Wide Range)

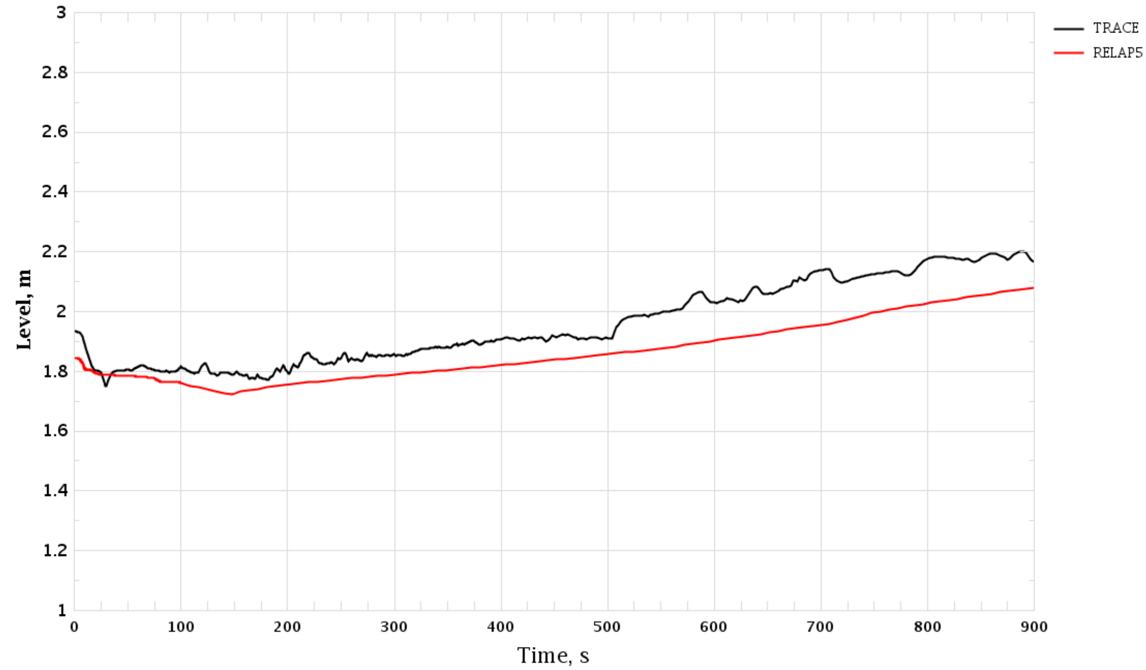

Figure 3-228 PRZ Surge Line Break. SG-5 Level (Wide Range)

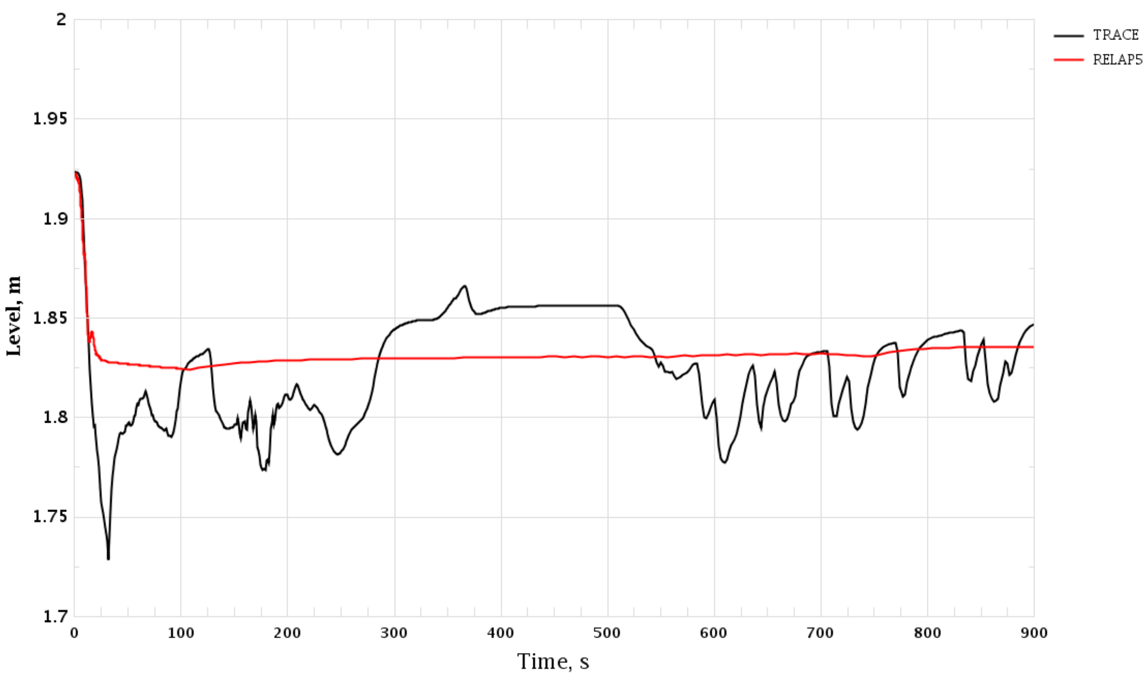

Figure 3-229 PRZ Surge Line Break. SG-6 Level (Wide Range)

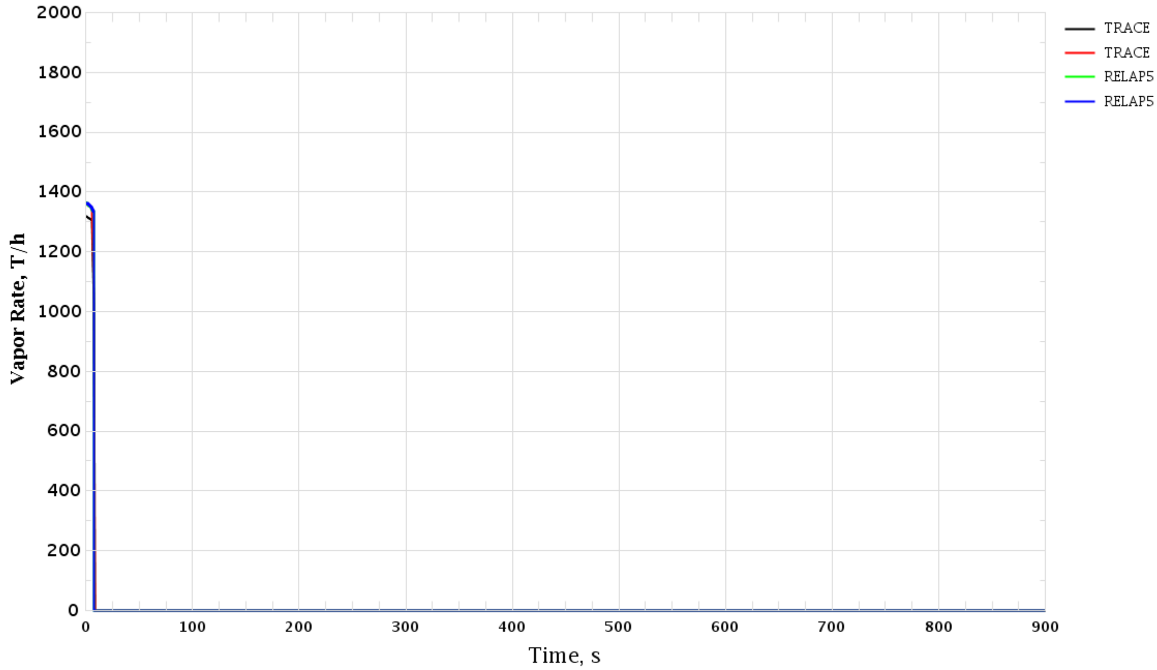

Figure 3-230 PRZ Surge Line Break. Turbines Mass Flow Rate

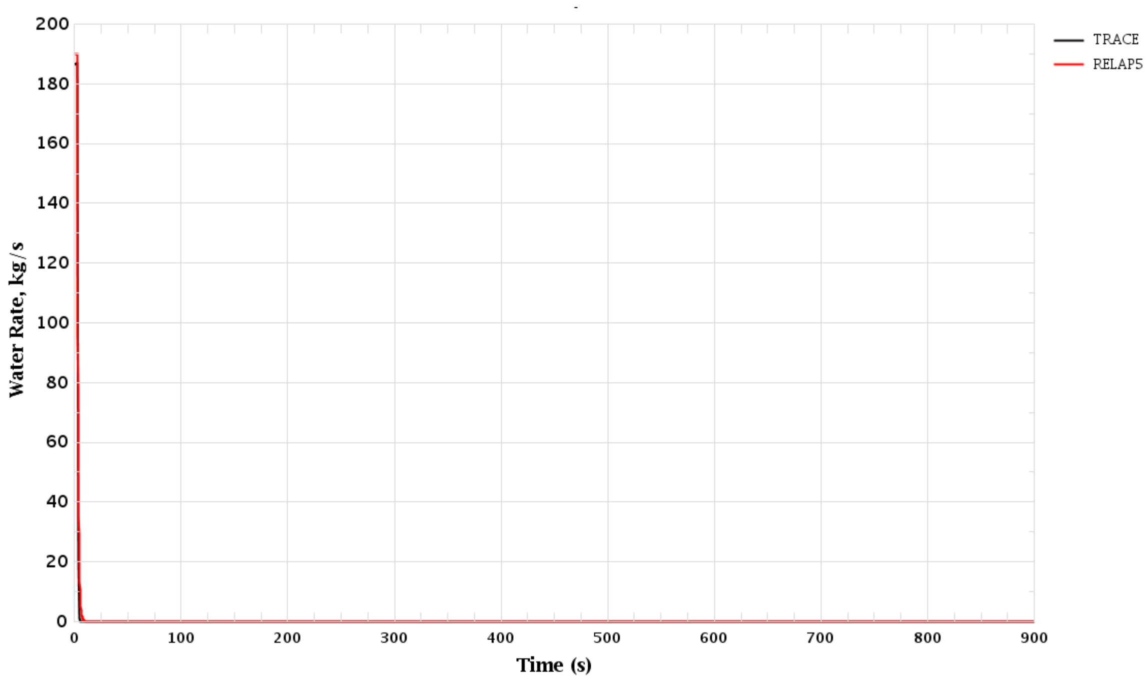

Figure 3-231 PRZ Surge Line Break. MFW Pump No.1 Mass Flow Rate

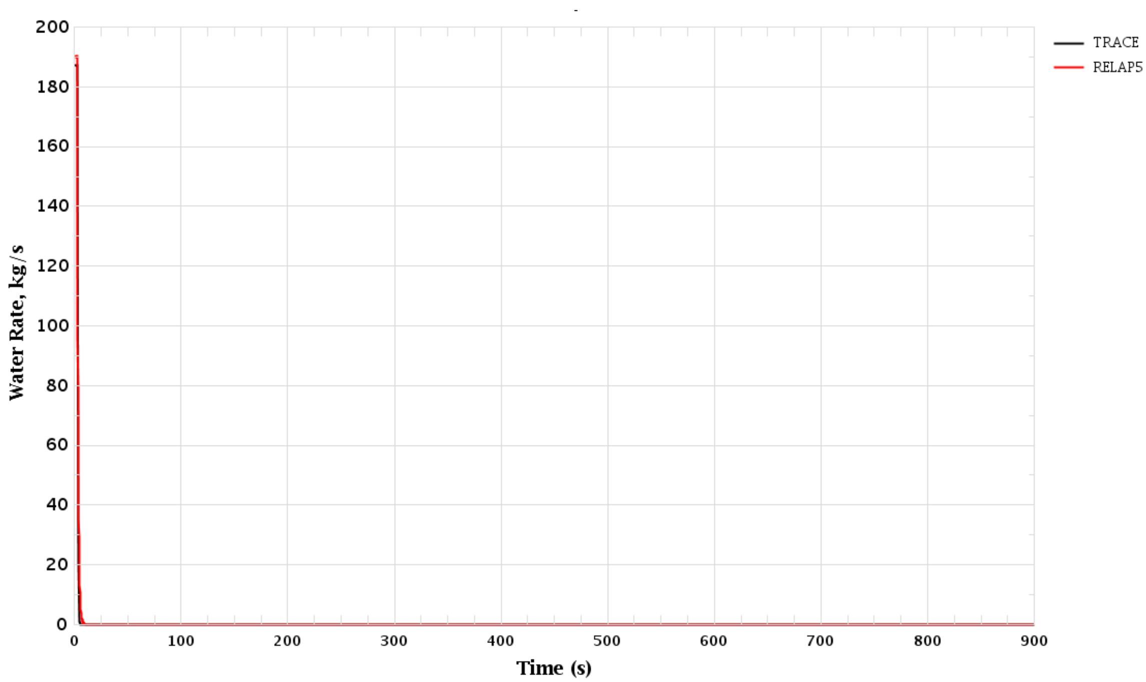

Figure 3-232 PRZ Surge Line Break. MFW Pump No.2 Mass Flow Rate

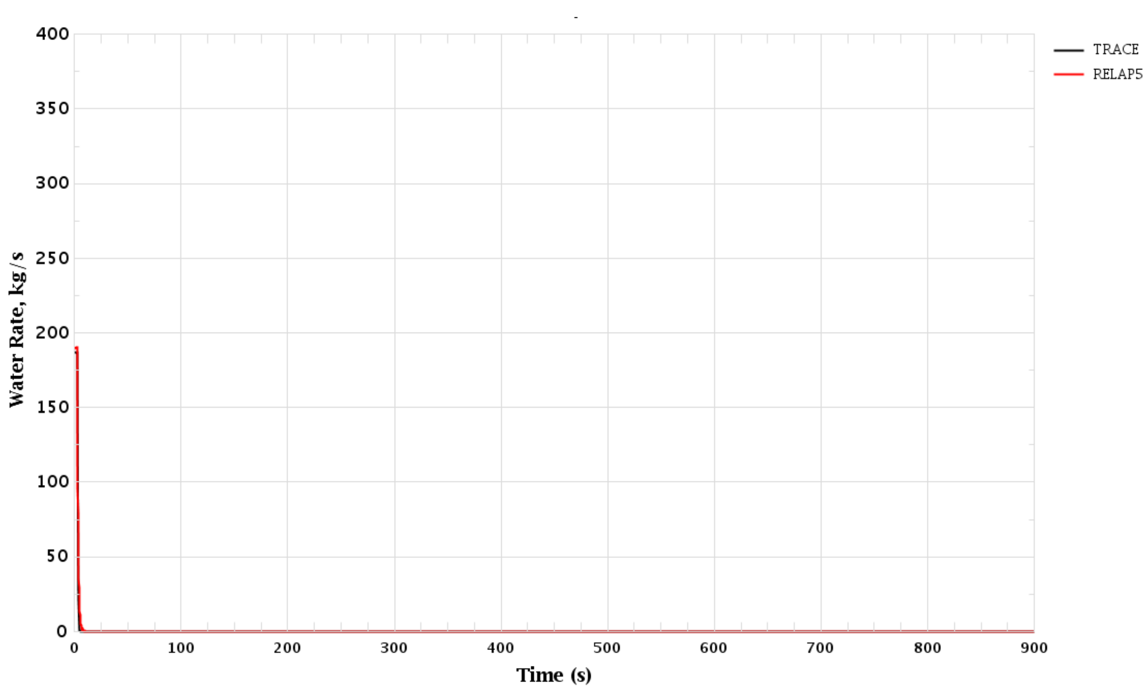

Figure 3-233 PRZ Surge Line Break. MFW Pump No.3 Mass Flow Rate

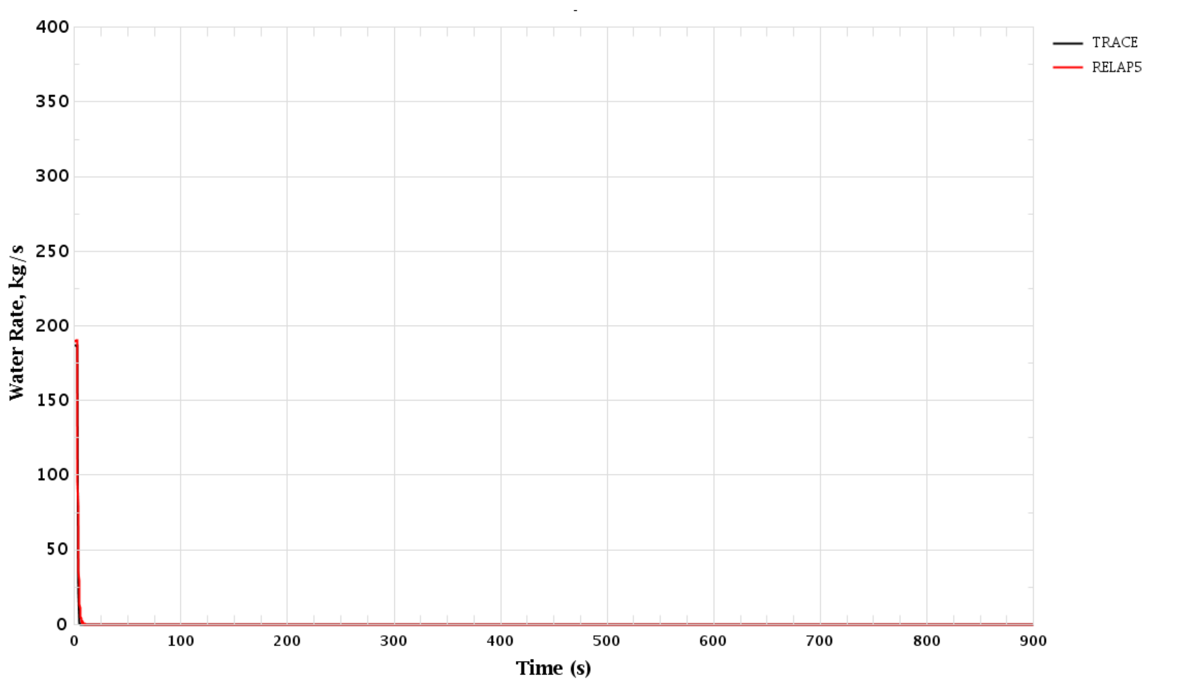

Figure 3-234 PRZ Surge Line Break. MFW Pump No.4 Mass Flow Rate

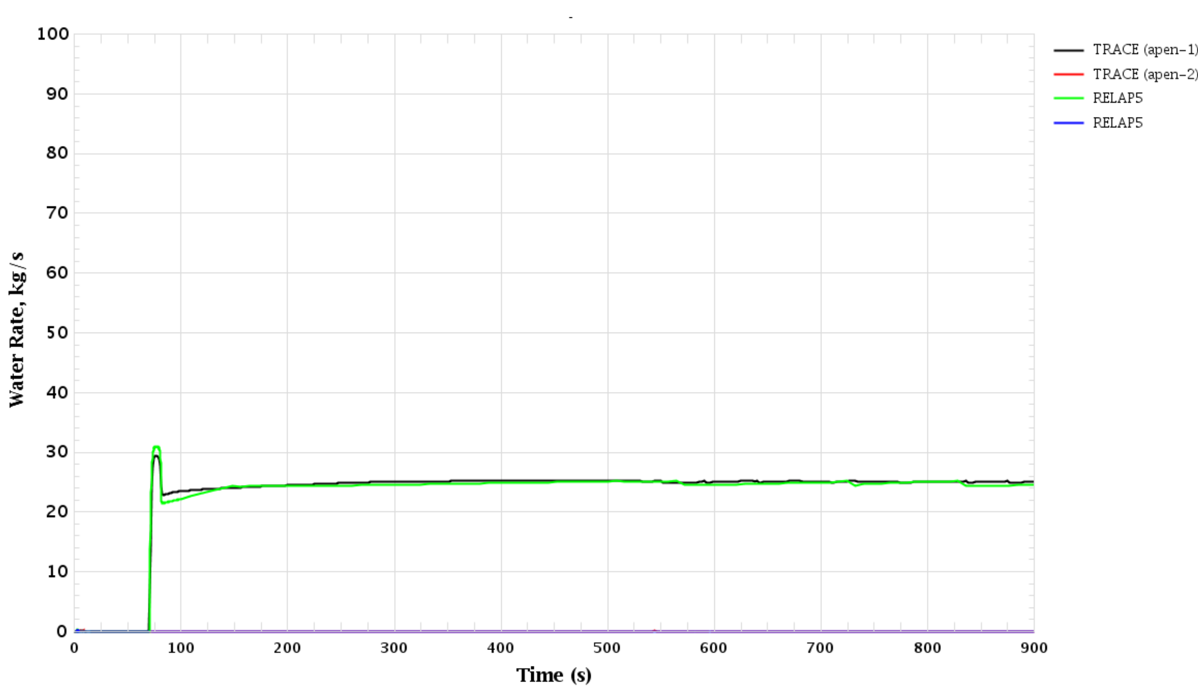

Figure 3-235 PRZ Surge Line Break. AFW Mass Flow

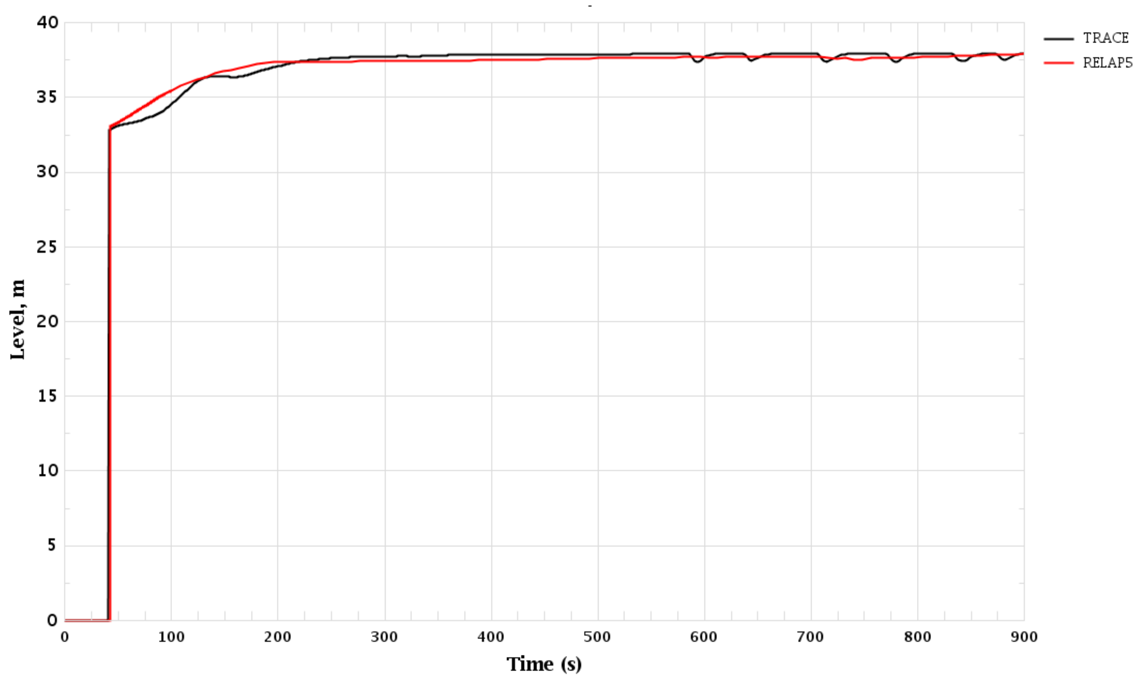

Figure 3-236 PRZ Surge Line Break. HPIS-1 Mass Flow Rate

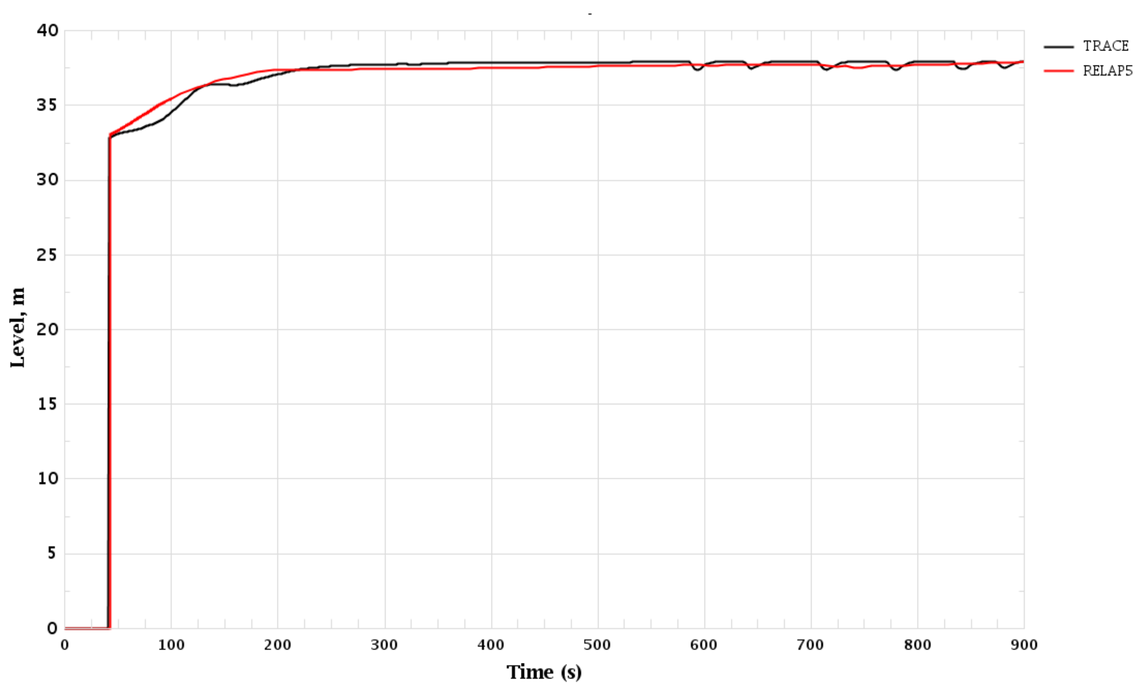

Figure 3-237 PRZ Surge Line Break. HPIS-2 Mass Flow Rate

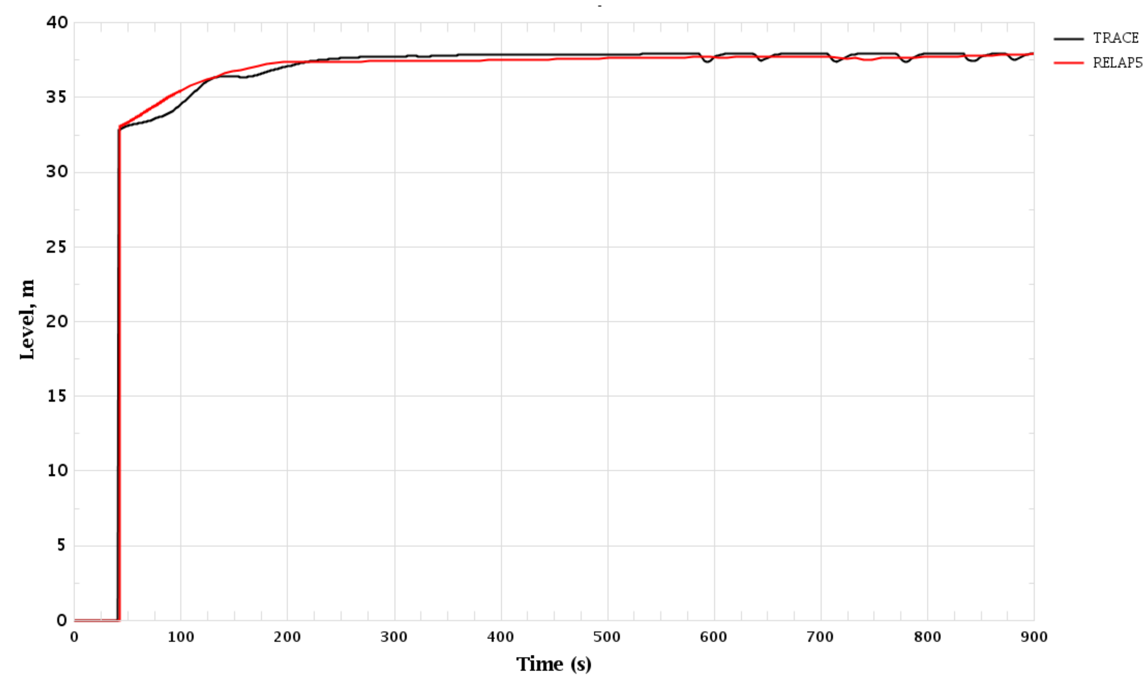

Figure 3-238 PRZ Surge Line Break. HPIS-3 Mass Flow Rate

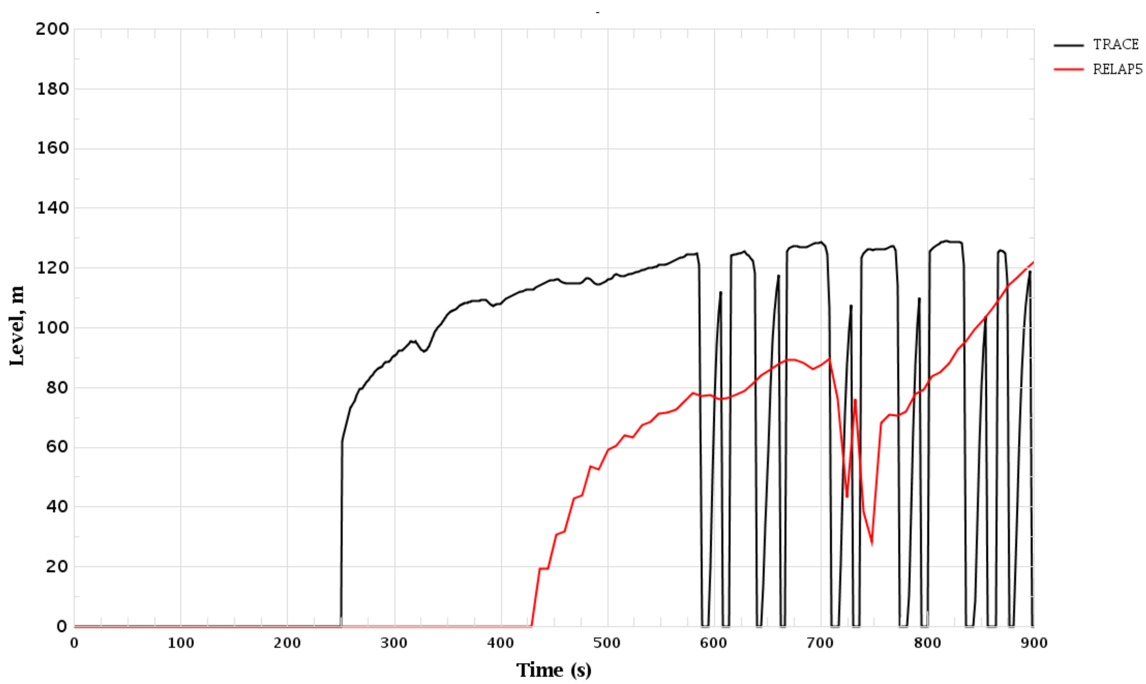

Figure 3-239 PRZ Surge Line Break. LPIS-1 Mass Flow Rate

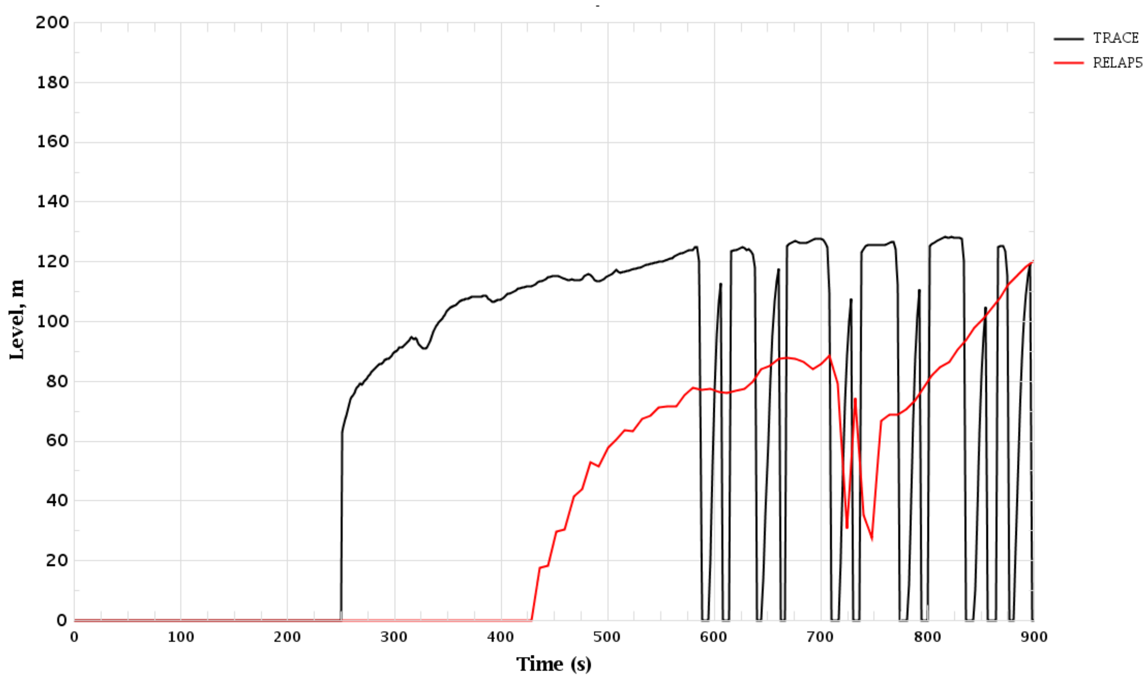

Figure 3-240 PRZ Surge Line Break. LPIS-2 Mass Flow Rate

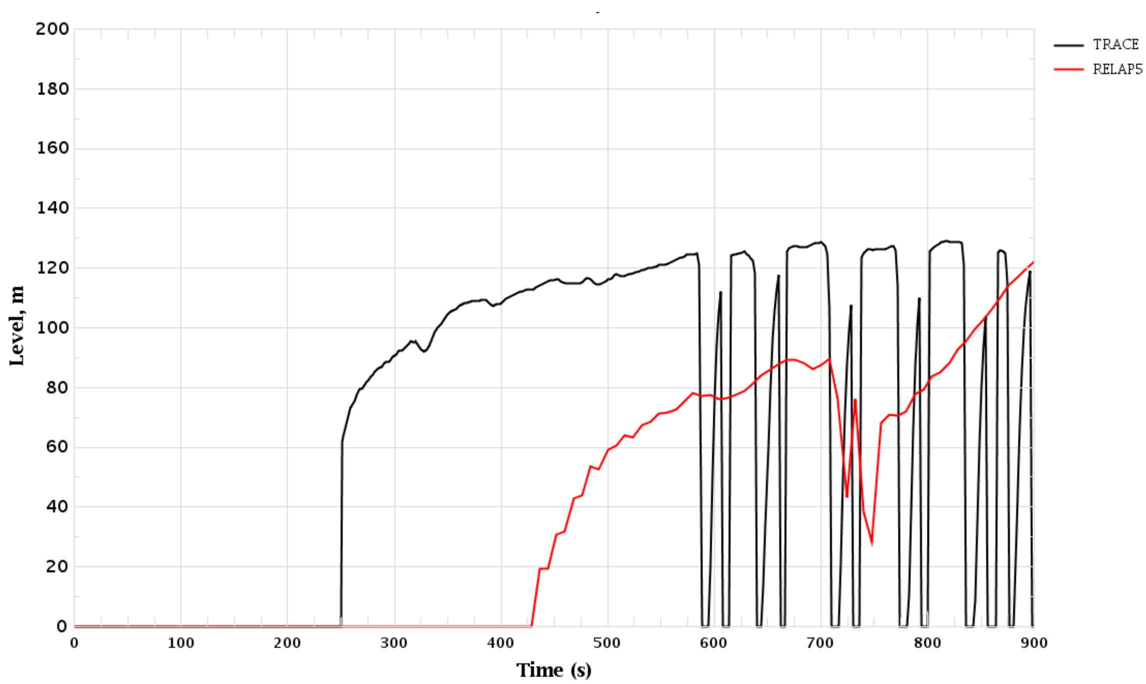

Figure 3-241 PRZ Surge Line Break. LPIS-3 Mass Flow Rate

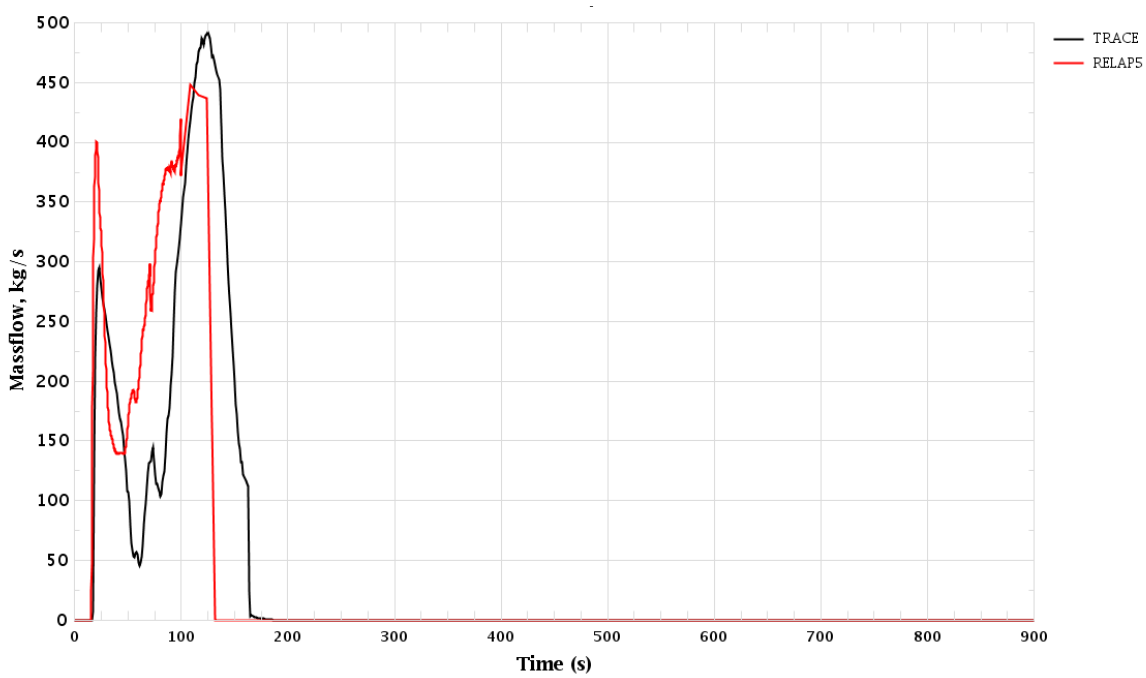

Figure 3-242 PRZ Surge Line Break. HA-1 Mass Flow Rate

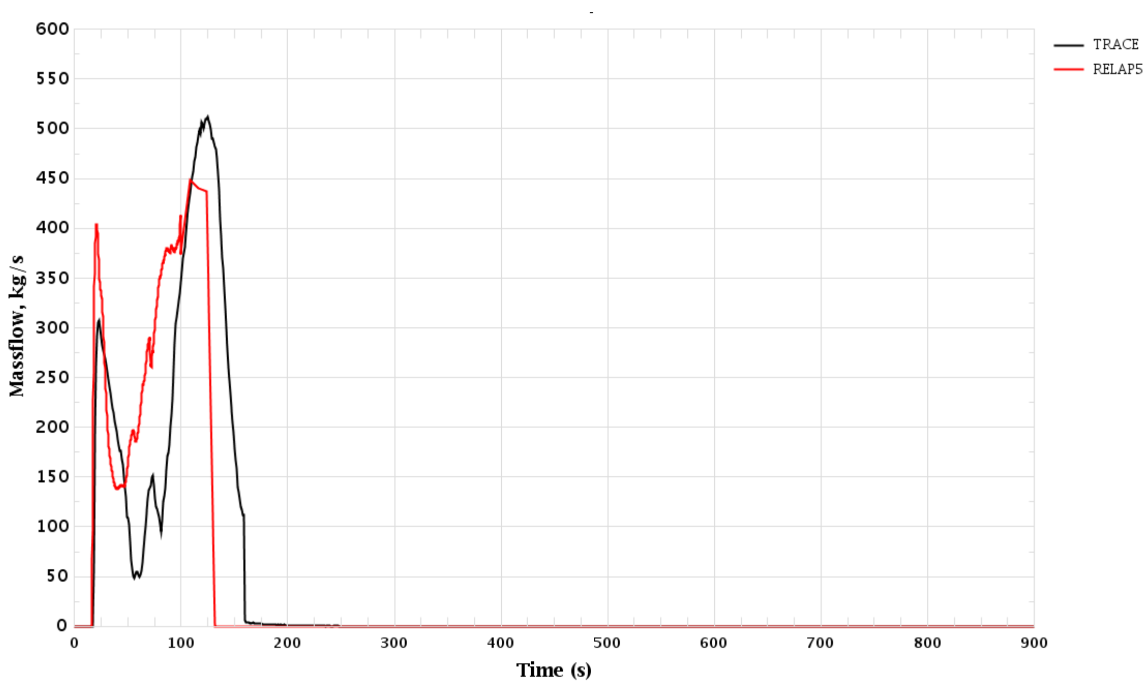

Figure 3-243 PRZ Surge Line Break. HA-2 Mass Flow Rate

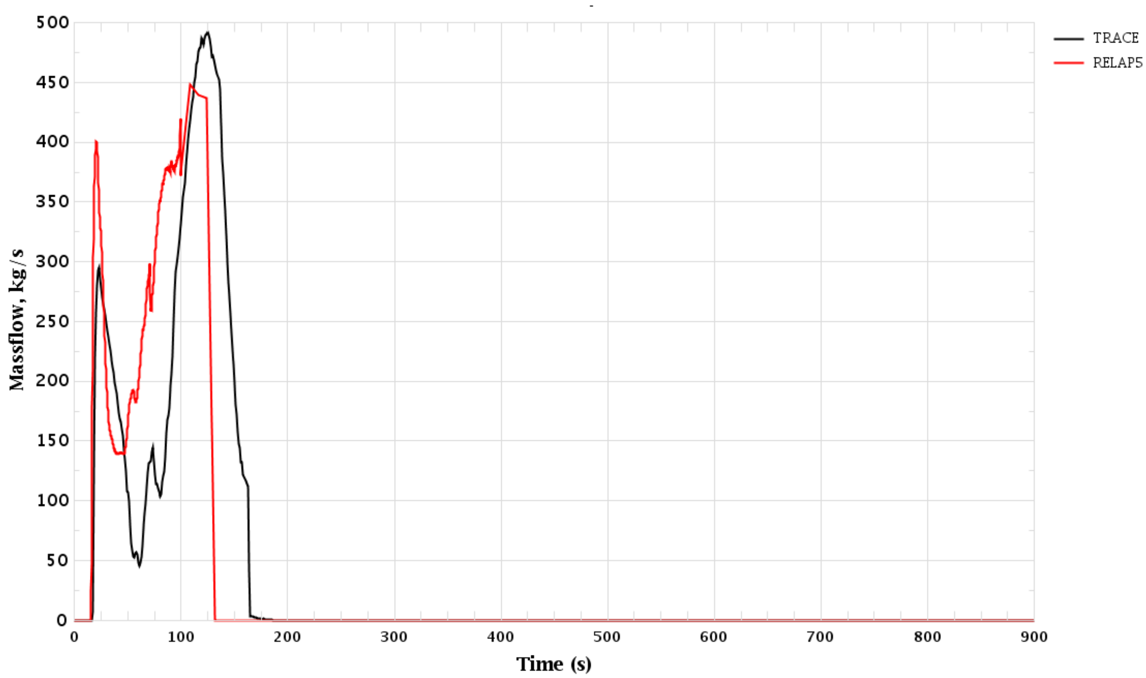

Figure 3-244 PRZ Surge Line Break. HA-3 Mass Flow Rate

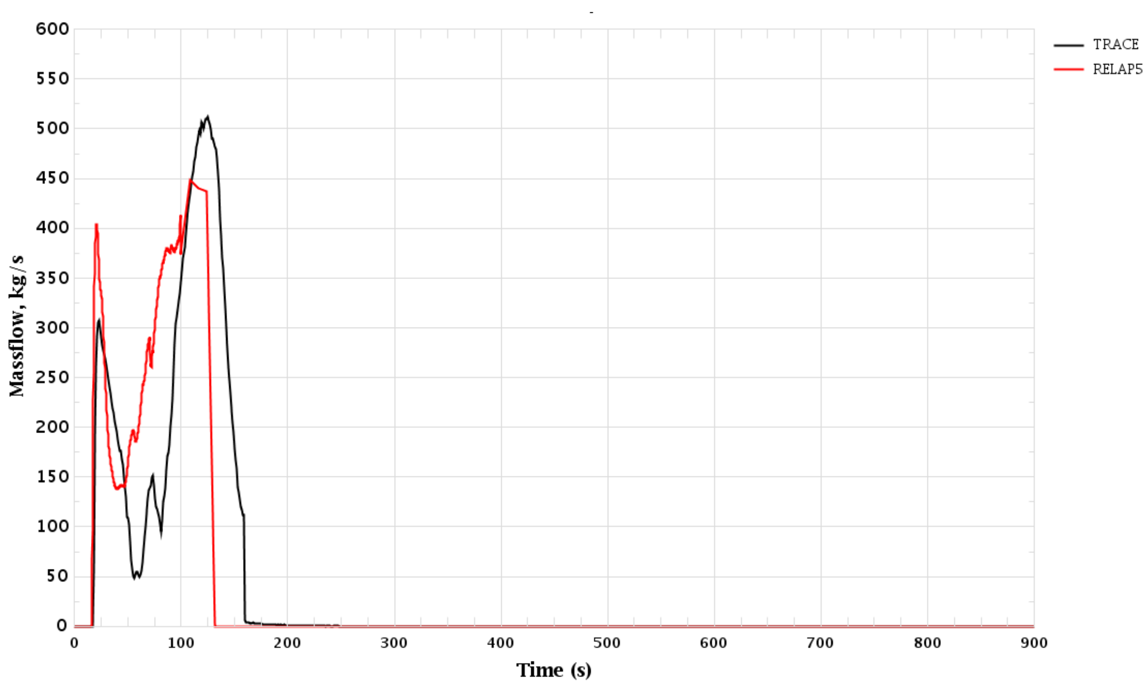

Figure 3-245 PRZ Surge Line Break. HA-4 Mass Flow Rate

Figure 3-246 PRZ Surge Line Break. Maximal Cladding Temperature

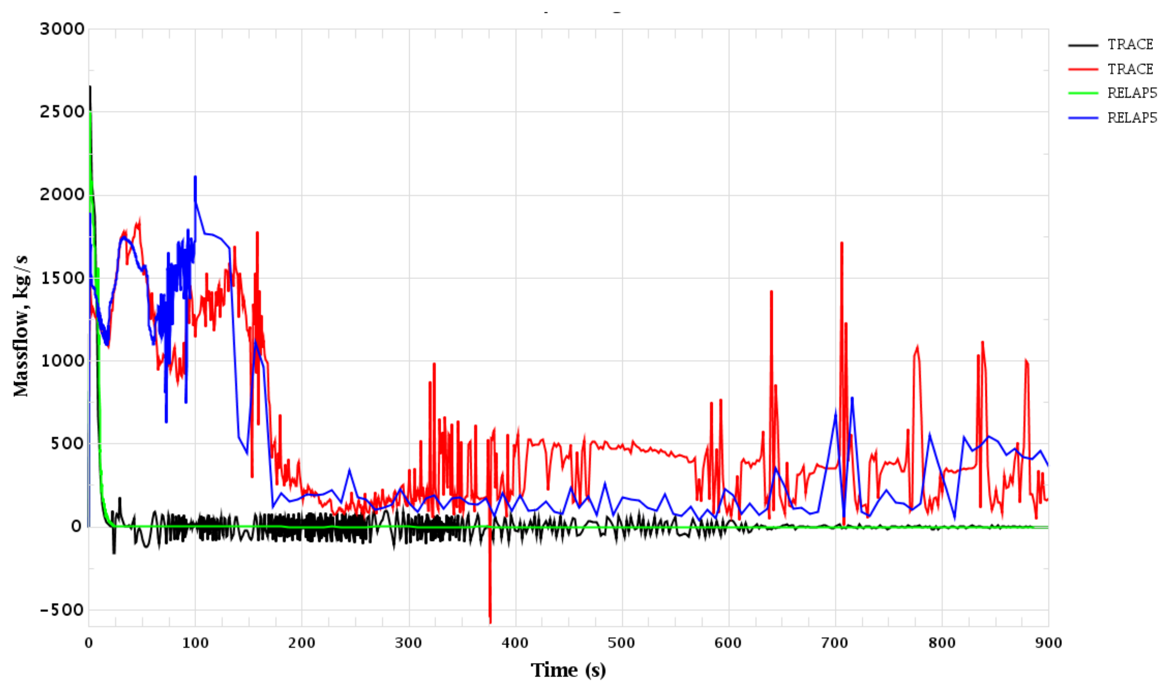

Figure 3-247 PRZ Surge Line Break. Break Mass Flow Rate

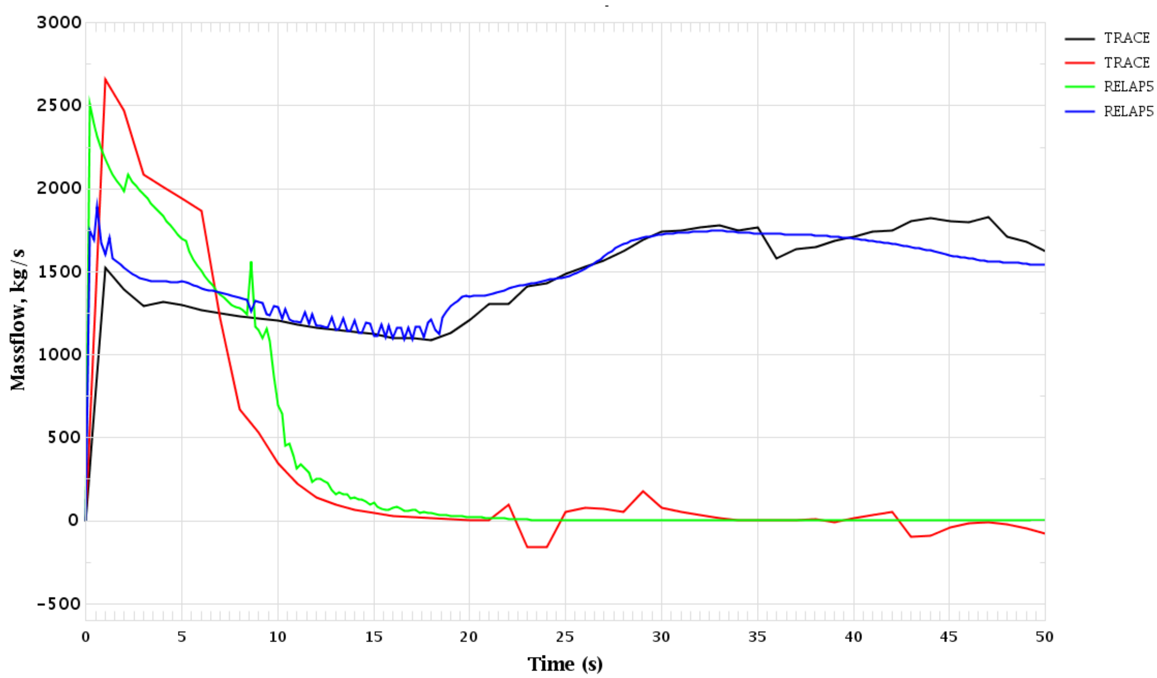

Figure 3-248 PRZ Surge Line Break. Break Mass Flow Rate (Fragment)

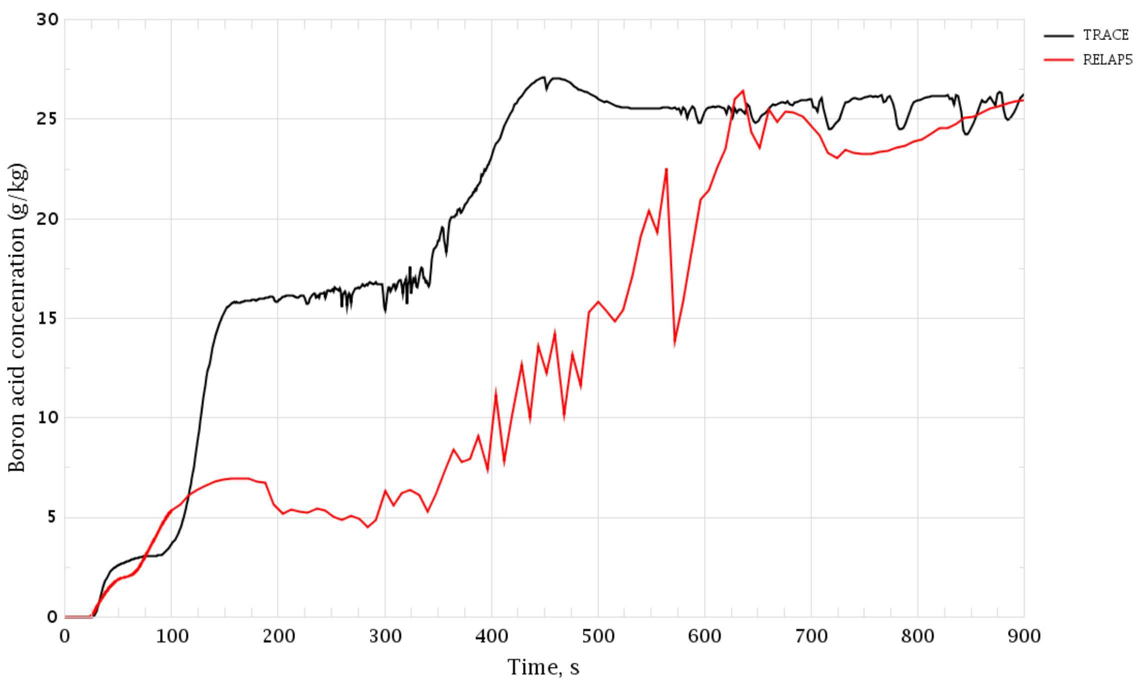

Figure 3-249 PRZ Surge Line Break. Boric Acid Concentration in the Core

4 CONCLUSIONS

After validation of VVER-440/V-213 thermal-hydraulic model for TRACE code by simulating several operational events that had occurred at Ukrainian NPPs, the validation effort was extended by calculations of 5 DBA scenarios and comparing the results obtained with this model and correspondent RELAP5/Mod3.2 model. Both TRACE and RELAP5 models applied for calculations are nearly identical with respect to model scope, nodalization, components geometry and equipment characteristics.

Comparison of the results obtained with TRACE and RELAP5 models indicates some differences in calculated parameters. In particular, the differences in the primary circuit pressure that were observed in some of the scenarios are caused by different mathematical models and correlations for steam condensation, which are used in the special PRESSURIZER model in TRACE, as compared to the pressurizer modelling in RELAP5. Cladding temperature differences are related to the specifics of the heat structure modeling approach and the absence of TRACE correlation options, which does not allow more precise model adjustment to ensure complete convergence with the relevant RELAP5 models.

Nevertheless, these differences do not affect significantly the overall behavior of the main parameters of the primary and secondary circuit and the sequence of events in the analyzed scenarios calculated with TRACE and RELAP5 models, and quantitatively the values are in a good agreement.

The results of cross-code validation calculations demonstrate that developed VVER-440/V213 thermal-hydraulic model for TRACE code is able to reproduce adequately the NPP response to transients and accidents without core melt that were calculated previously in safety analysis reports using the RELAP5 model. For the majority of plant parameters good correspondence between TRACE and RELAP5 results is obtained.

Based on the results of model validation it can be concluded that developed WWER-440/V-213 thermal hydraulic model for TRACE computer code can be used for calculations of transients and accidents in support of regulatory review of safety analyses documentation.

5 REFERENCES

[1] TRACE V5.840. User's Manual. Volume 1: Input Specification. Division of Safety Analysis, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC 20555-000.

[2] TRACE V5.840. User's Manual. Volume 2: Modeling Guidelines. Division of Safety Analysis, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC 20555-001.

[3] NUREG/IA-0485. TRACE VVER-440/V-213 Model Validation. International Agreement Report, 2018.

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K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

This report is developed by the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) and its technical support organization, the State Scientific and Technical Center for Nuclear and Radiation Safety of Ukraine (SSTC NRS), under Implementing Agreement On Thermal-Hydraulic Code Applications And Maintenance Between The United States Nuclear Regulatory Commission and State Nuclear Regulatory Inspectorate of Ukraine (signed in 2014) in accordance with Article III, Section C, of the Agreement.
The report provides results of the comparison calculations conducted with application of SSTC NRS model of VVER-440/V-213 for TRACE and RELAP5 computer codes. The calculation scenarios analyzed include design basis accidents and transients from several initiating event groups usually evaluated in safety analysis reports.

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