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International Agreement Report

TRACE VVER-1000/V-320 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-1000/V-320 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. Additionally, the plant incident with inadvertent opening of steam generator safety relief valve in hot shutdown is simulated.

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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-1000 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 the turbine;
- trip of 1 out of 4 operating reactor coolant pumps;
- uncontrolled withdrawal of control rods group;
- break of the pressurizer surge line.

Additionally, the plant incident with inadvertent opening of steam generator safety relief valve in hot shutdown was simulated.

Comparison of the results obtained with TRACE and RELAP5 models indicates some differences in calculated parameters. 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, and quantitatively the values are in a good agreement.

The results of cross-code validation calculations and simulation of plant incidents demonstrate that developed VVER-1000/V-320 thermal-hydraulic model for TRACE code is able to reproduce adequately the plant response to transients and accidents without core melt. For the majority of plant parameters good correspondence between TRACE and RELAP5 results or plant incident data 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
CR	Control Rod
DBA	Design Basis Accident
DG	Emergency Power Supply Diesel-Generator
ECCS	Emergency Core Cooling System
EFW	Emergency Feedwater System
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
PORV	Pilot-operated Relief Valve
PZ	Preventive Reactor Protection
PRZ	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
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
UPZ	Fast Power Reduction
USNRC	United States Nuclear Regulatory Commission
VVER	Pressurized Water Reactor, Russian design
ZNPP	Zaporizhzhya Nuclear Power Plant

1 INTRODUCTION

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

In the framework of the Agreement SNRIU and SSTC NRS obtained the state-of the-art TRACE code [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-1000 was converted to TRACE code format.

In order to justify capabilities of VVER-1000 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-0490 [3]. Since these events do not cover all the phenomena expected to occur during accidents, the VVER-1000 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 VVER-1000/V-320 design which are important for development of thermal-hydraulic model, as well as to a description of TRACE model for this type of NPP unit.

The results of comparative TRACE and RELAP calculations for selected design basis accident (DBA) scenarios are provided in Section 3 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 VVER-1000/V-320 model for TRACE code and adjustment of steady state calculation several transient calculations were performed simulating the actual incidents that had actually occurred at NPPs in Ukraine and the results of calculations were compared with the plant measurement data. In particular the following incidents were simulated:

- main feedwater pump trip at Zaporizhzhya NPP (ZNPP) Unit 5;
- inadvertent closure of main steam isolation valve (MSIV) at ZNPP Unit 6;
- pressurizer (PRZ) pilot-operated relief valve (PORV) stuck open during testing at Rivne NPP Unit 3.

The results of these validation calculations, as well as a brief description of the model and of the main VVER-1000/V-320 design features are presented in NUREG/IA-0490 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 one reactor coolant pump (RCP);
- uncontrolled withdrawal of control rods (CR) group;
- 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.

Additionally, the plant incident with inadvertent opening of steam generator (SG) safety relief valve (SRV) in hot shutdown was simulated.

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 530×28 mm, that leads to a loss of secondary circuit coolant from all SG, decrease of SG pressure and 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	3000±60	3104
Reactor outlet pressure	kgf/cm^2	158…162	159.8
Reactor inlet temperature	°C	289	289.6
Coolant temperature at reactor outlet	°C	320	320
Reactor coolant flow rate	m^3/h	84800^{+4000}_{-4800}	84350
PRZ level	m	8.77±0.150	8.82
SG pressure	kgf/cm^2	60…64	60.8
SG level (narrow range gauge)	m	0.270±0.05	0.305...0.315
SG steam production	t/h	1470±103	1394.1
SG feedwater temperature	°C	220±5	220
Water temperature in emergency core cooling system (ECCS) tanks	°C	20…60	60.0
Water temperature in ECCS hydroaccumulators (HA)	°C	55±5	60.0

3.1.3 Boundary Conditions

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

As an initiating event the guillotine break of MSH with a diameter of 530×28 mm with double-ended discharge of steam is modeled.

No operator actions are taken into account.

Operation of reactor power controller (ARM), reactor power limiter (RPL), preventive reactor protection (PZ) and fast power reduction (UPZ) to decrease power is not considered. Reactor scram operates according to the design with a delay of 1.0 s from formation of actuation signal to a start of control rods drop. Maximal allowed time of control rods drop to the core (4 s) is assumed.

As a single failure a failure to close of MSIV at the main steam line (MSL) from SG-1 is postulated.

Main feedwater (MFW) pumps trip is modelled at the time of turbine stop valves closure. It is postulated that MFW control valve to SG-1 remains in the same position as at the beginning of accident, and correspondent cut-off valve is failed to close (i.e., passive feedwater flow from MFW pipelines to SG-1 is possible). Cut-off valves to other SGs are closed after formation of signal "Pressure decrease in correspondent MSL below 45 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C and temperature in correspondent hot leg >200°C". Auxiliary feedwater (AFW) pumps start after SG level decrease below 0.22 m by narrow range level meter.

RCP at the loop no.1 remains in operation that increases primary circuit cooling by SG-1 with a potential to reach core recriticality temperature.

Operation of PRZ heaters and PRZ spray is not taken into account.

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	MSH break	Start of double-ended steam discharge from MSH
0.4	0.3	Closure of turbine stop valves	MSH pressure decrease below 52 kgf/cm^2
0.4	0.3	Trip of MFW pumps	Postulated after closure of turbine stop valves
8.0	10.2	Scram signal	Pressure decrease in any MSL below 50 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C
8.0	10.2	Formation of ECCS safeguard	Pressure decrease in MSL-1 below 50 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C and temperature in hot leg >200°C
8.0–15.0	10.2–27.0	Closure of MSIV at MSL-2, 3, 4	Pressure decrease in MSL below 50 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C and temperature in correspondent hot leg >200°C. Failure of MSIV at MSL-1 is postulated
10.0–16.0	12.0–28.0	RCP-2, 3, 4 trip	Pressure decrease in MSL below 45 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C and temperature in correspondent hot leg >200°C. Continuous operation of RCP-1 is postulated
35.0	23.0	Start of AFW pump	Due to SG level decrease below 0.22 m (by narrow range level meter)
220.0	290.0	Start of injection to the primary circuit from high pressure injection system (HPIS) pumps	Decrease of RCS pressure below HPIS pump shut-off head
520	575	Termination of HPIS injection	Increase of RCS pressure above HPIS pump shut-off head
3600.0	3600.0	End of calculation	Stabilization of parameters

MSH break with double-ended discharge of steam causes rapid decrease of SG pressure (Figure 3-19 – Figure 3-22). Due to decrease of MSH pressure below 52 kgf/cm^2 the turbine stop valves are closed (Figure 3-27) at 0.3/0.4 s of the accident (in RELAP and TRACE calculations, respectively). At the same time the MFW pumps trip is assumed due to closure of turbine stop valves and low MSH pressure.

At 8.0/10.2 s after IE occurrence the scram signal "Pressure decrease in any MSL below 50 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C" is formed. With 1 s delay the control rods drop to the reactor core starts and the reactor power decreases to the decay heat (Figure 3-1).

Actuation of ECCS "MSL break" safeguards "Pressure decrease in MSL below 50 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C and temperature in correspondent hot leg >200°C" (within 10.2–27 s in RELAP calculation and 8–15 s in TRACE calculation) results in formation of signals to close MSIV at all main steam lines and steam loss from SG-2, 3, 4 is terminated. This causes temporary restoration of pressure in these SGs (Figure 3-20 – Figure 3-22), however before MSIVs are fully closed the steam lines pressure decreases below 45 kgf/cm^2 that corresponds to the setpoints for trip of RCPs at the affected loops (Figure 3-13 – Figure 3-15).

Steam loss from SG-1 continues since failure of MSIV at MSL-1 to close is postulated (Figure 3-28). RCP of correspondent loop also remains in operation as postulated (Figure 3-12).

Upon ECCS safeguard actuation all HPIS, low pressure injection system (LPIS) and containment spray pumps start operation through correspondent minimal bypass flow (recirculation) lines. Increased (compared to a decay power) heat removal from the primary circuit by SG-1 causes decrease of primary coolant temperature (Figure 3-4 – Figure 3-11), shrinkage of coolant volume and decrease of PRZ level (Figure 3-3) and RCS pressure (Figure 3-2). As RCS pressure decreases below HPIS pump shut-off head (~110 kgf/cm^2) boric acid injection to the primary circuit begins at 290/220 s of accident (Figure 3-32 – Figure 3-34) and continues till RCS pressure increases at ~575/520 s.

Continuous operation of AFW (Figure 3-30) and decrease of steam loss through a break (Figure 3-35) cause gradual restoration of SG-1 level (Figure 3-23) starting from ~1200 s.

At ~1700/1400 s PRZ level starts to recover gradually (Figure 3-3) due to operation of make-up pump (Figure 3-16).

The calculation is ended at 3600 s of accident close to the stabilization of main primary and secondary circuit parameters. Core recriticality conditions are not reached (Figure 3-18) due to timely injection by HPIS. Cladding temperature (Figure 3-36) decreases the whole duration of accident and does not exceed the initial full power operation values.

The plots of the main parameters of calculation are presented below on Figure 3-1 – Figure 3-37.

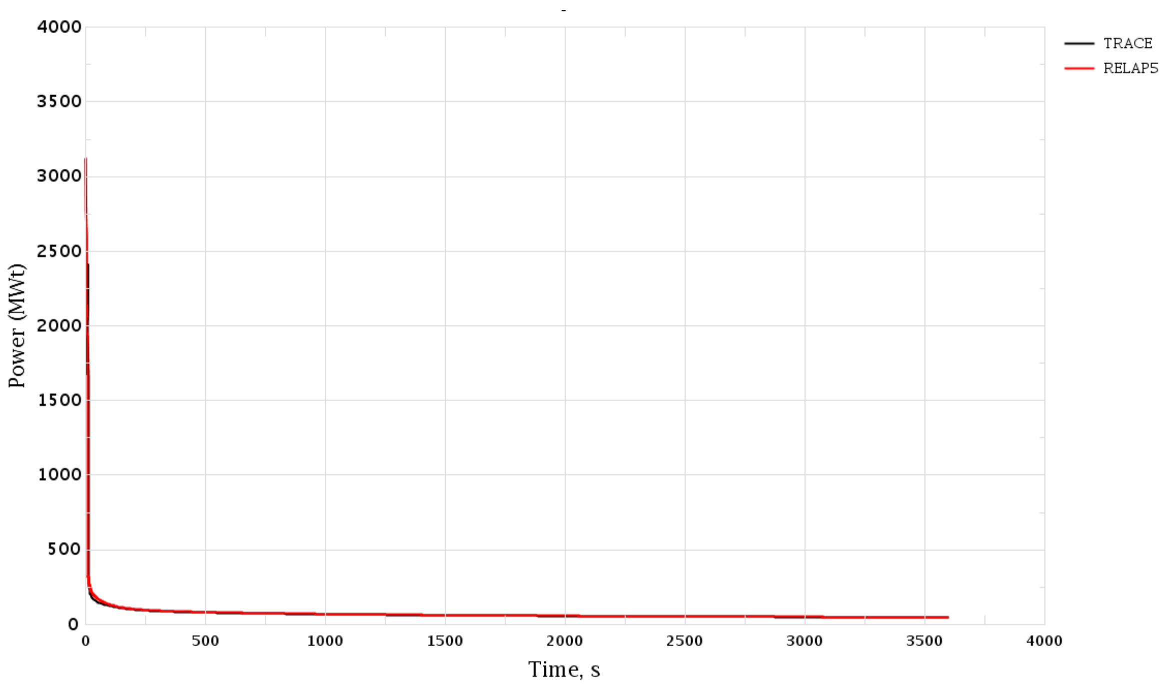

Figure 3-1 MSH Break. Core Thermal Power

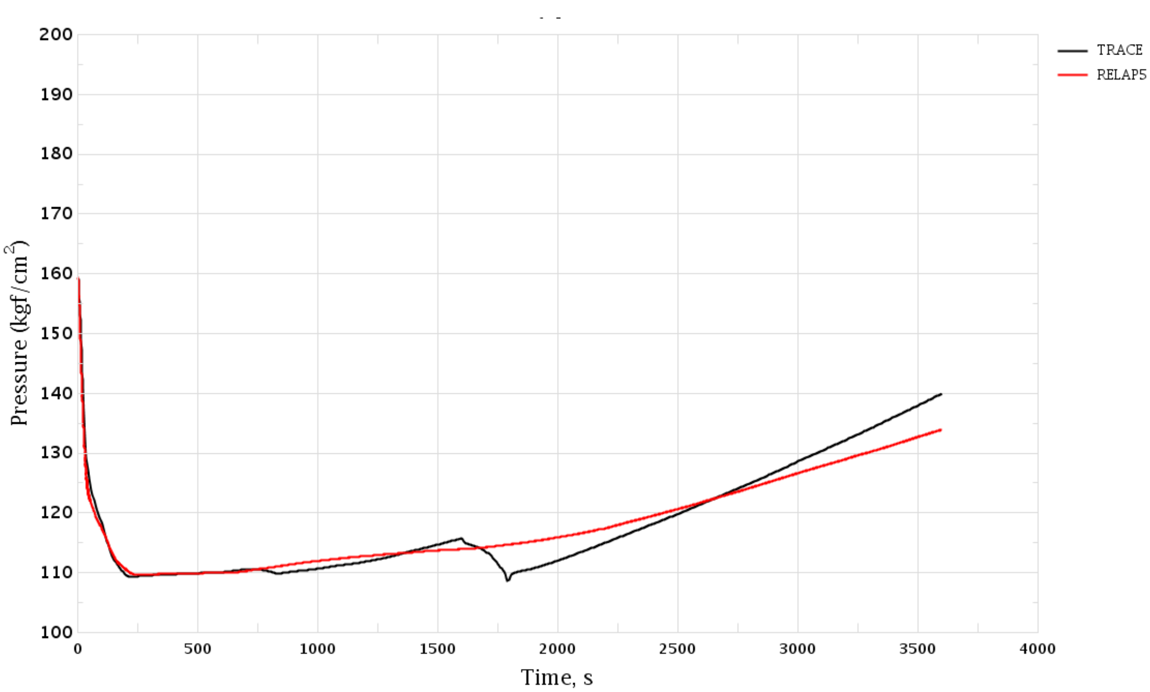

Figure 3-2 MSH Break. RCS Pressure

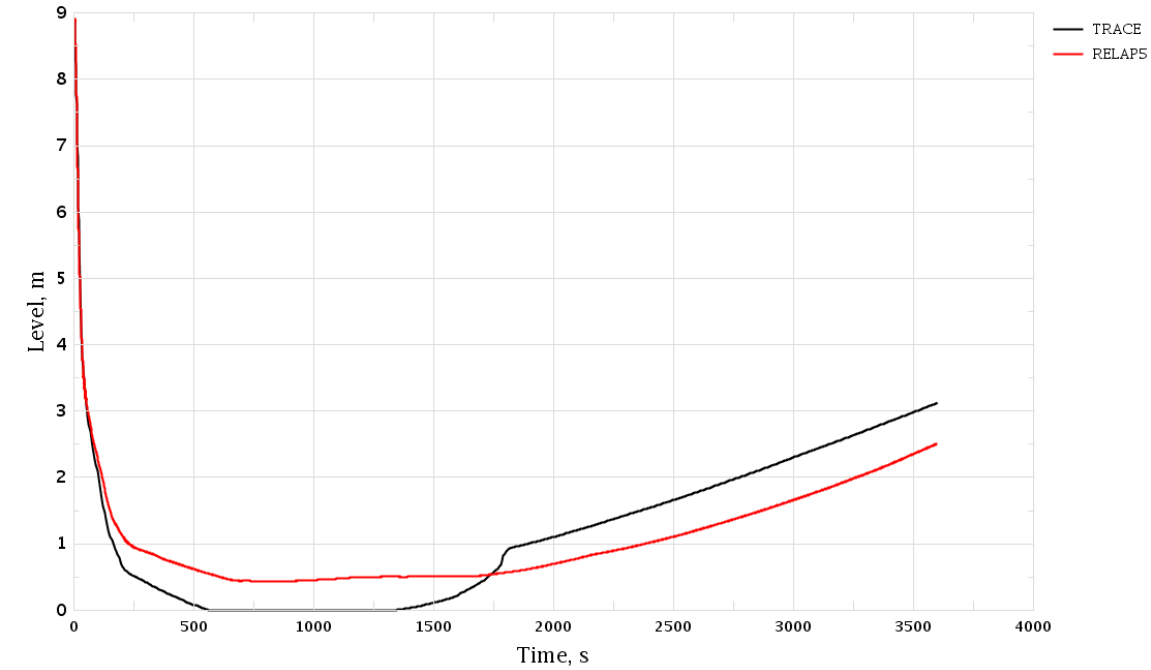

Figure 3-3 MSH Break. Pressurizer Level

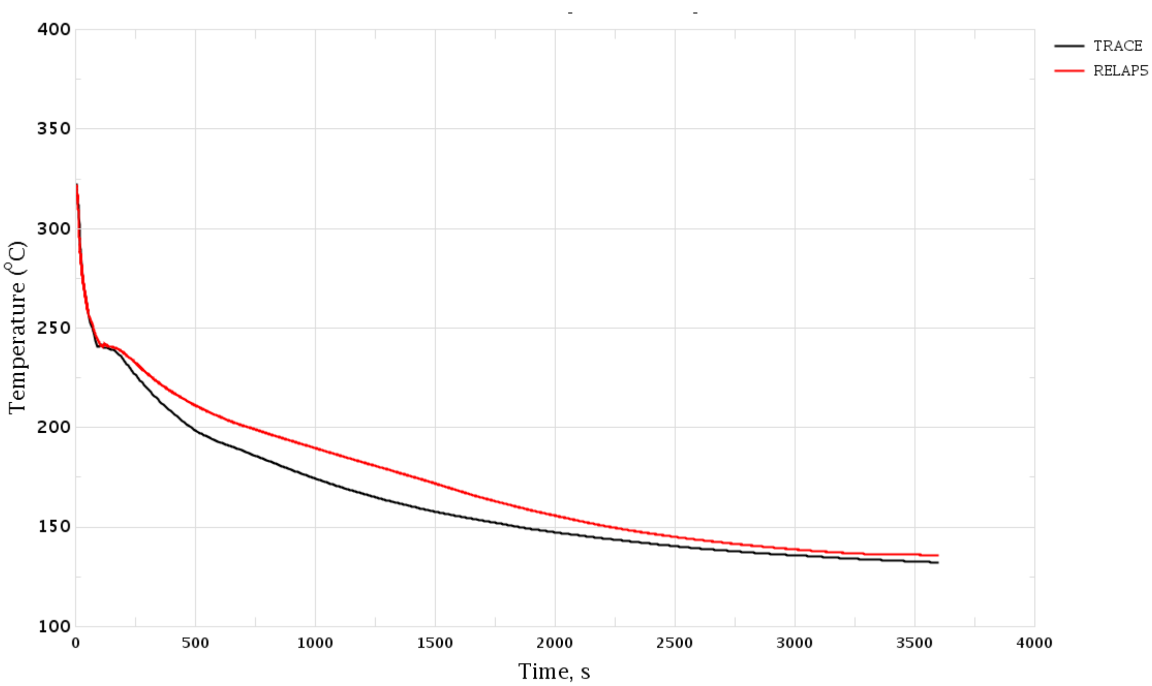

Figure 3-4 MSH Break. Coolant Temperature in Hot Leg, Loop 1

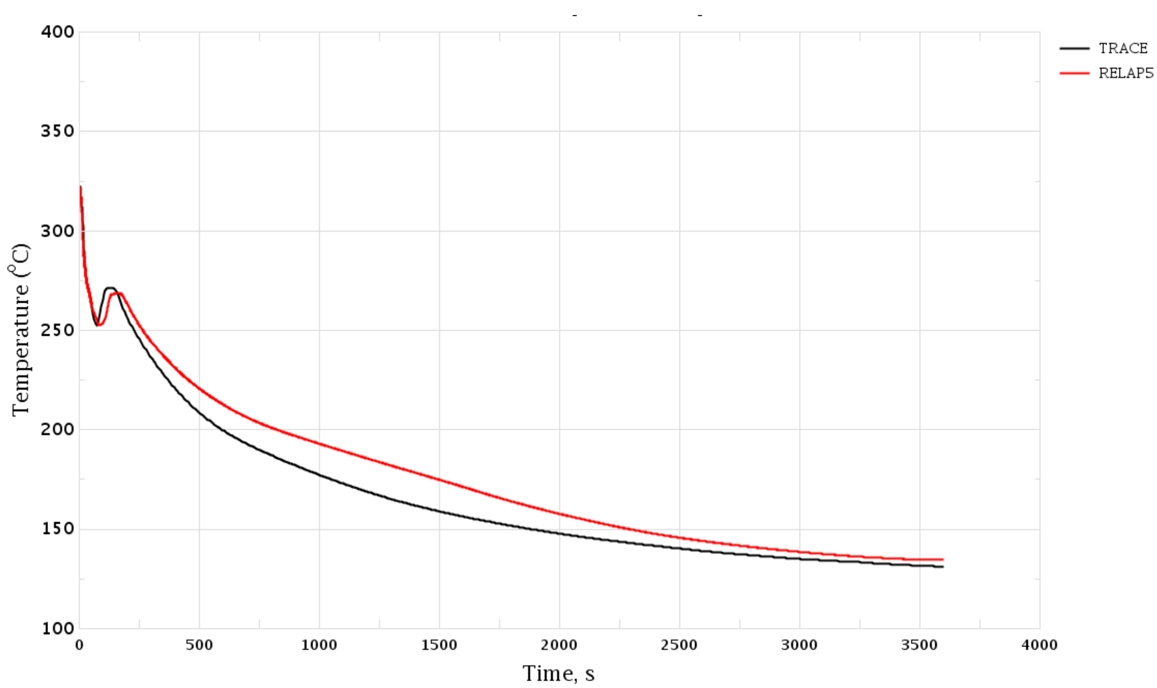

Figure 3-5 MSH Break. Coolant Temperature in Hot Leg, Loop 2

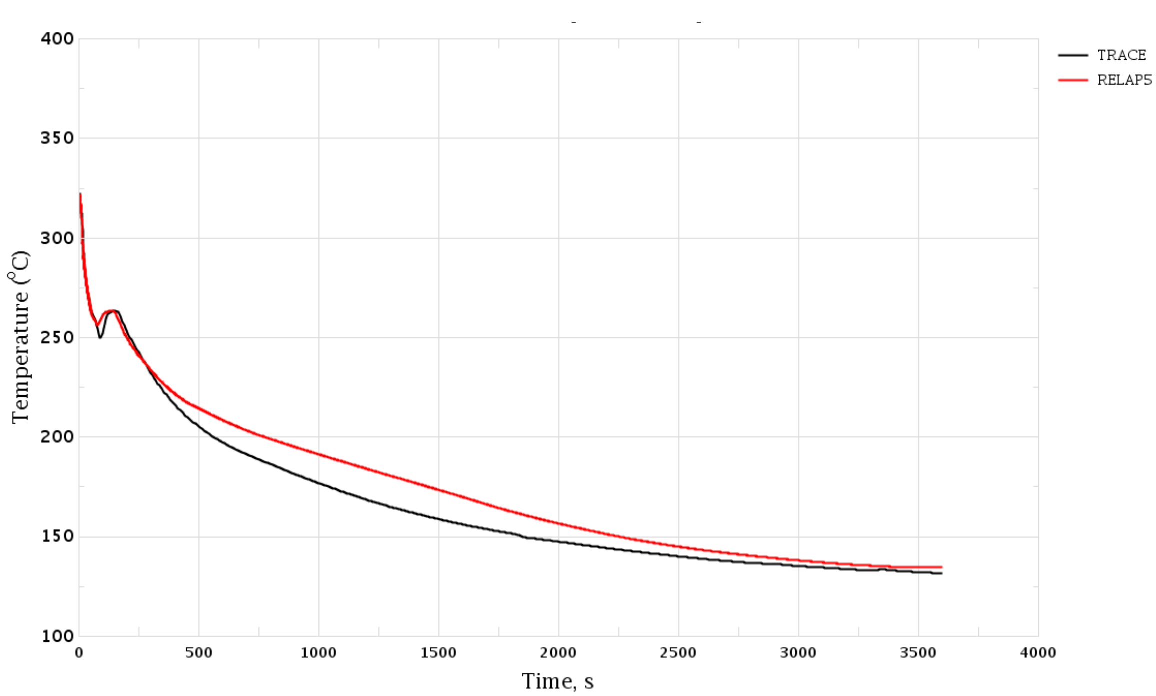

Figure 3-6 MSH Break. Coolant Temperature in Hot Leg, Loop 3

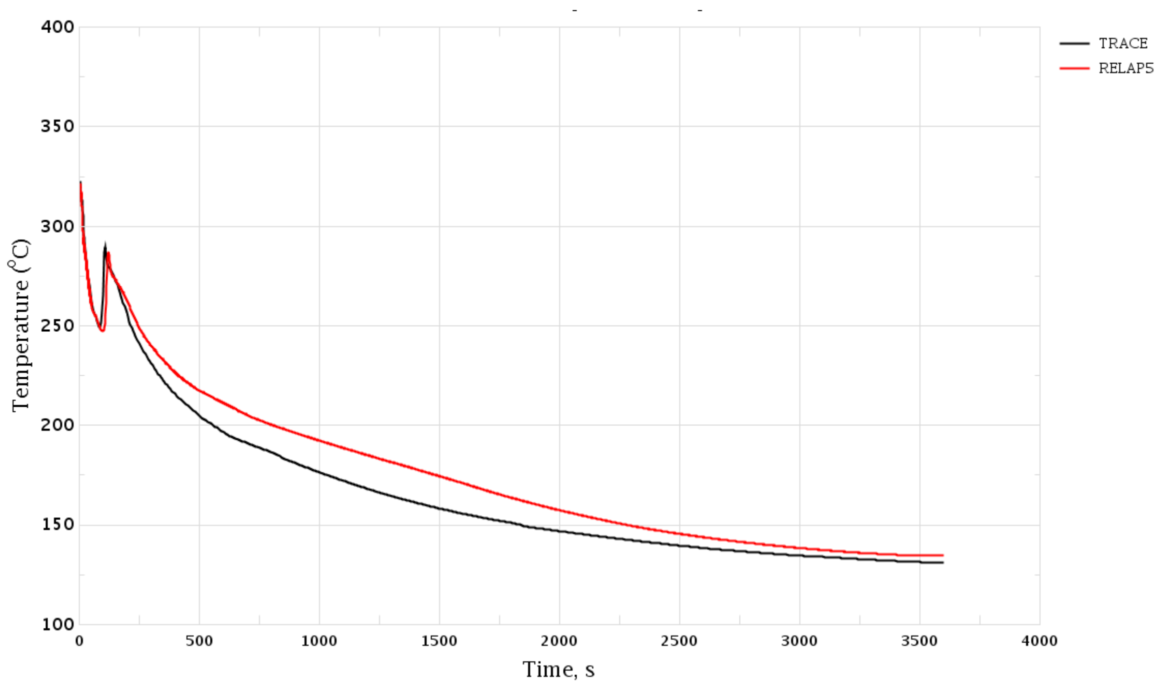

Figure 3-7 MSH Break. Coolant Temperature in Hot Leg, Loop 4

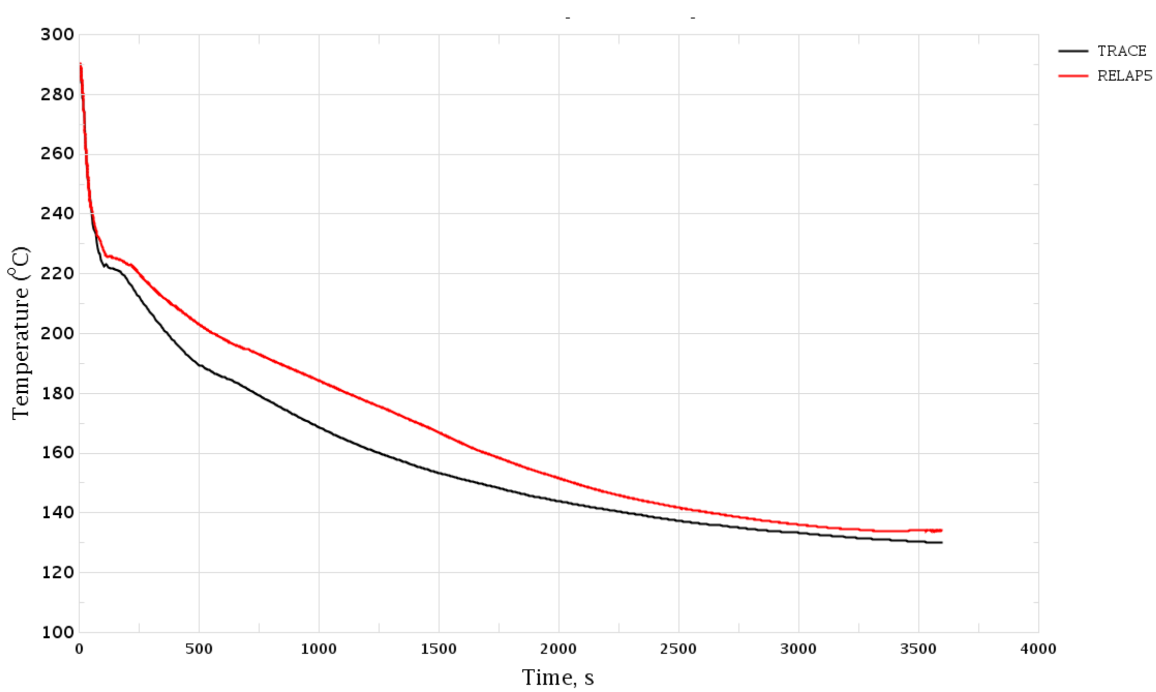

Figure 3-8 MSH Break. Coolant Temperature in Cold Leg, Loop 1

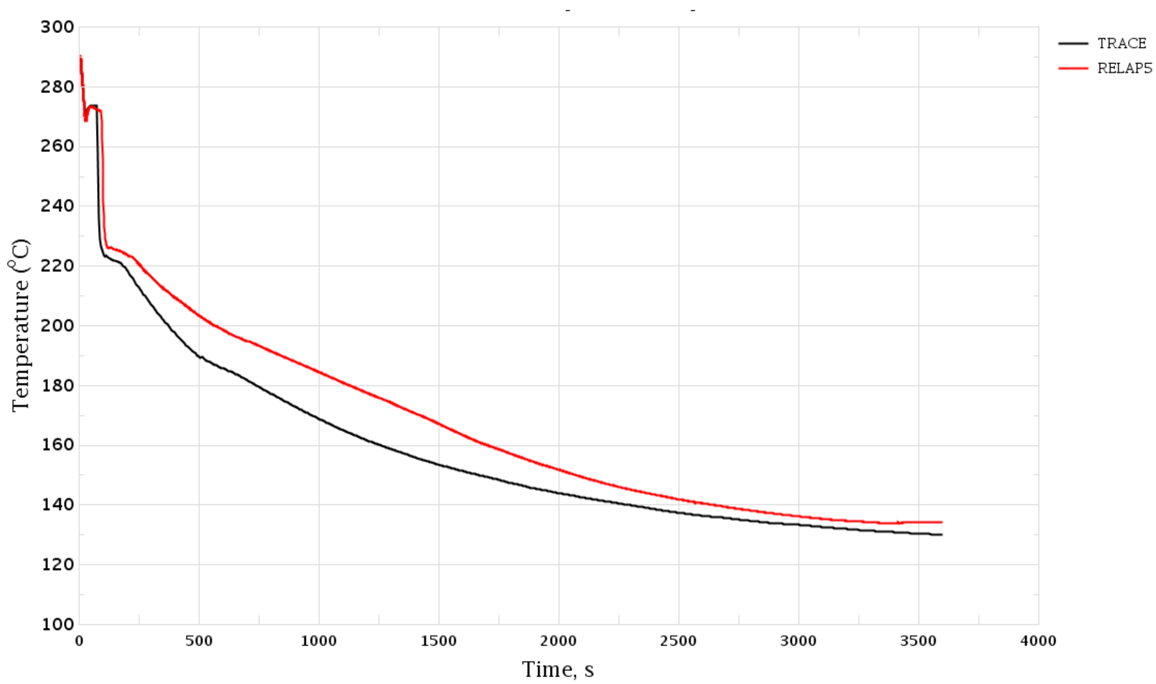

Figure 3-9 MSH Break. Coolant Temperature in Cold Leg, Loop 2

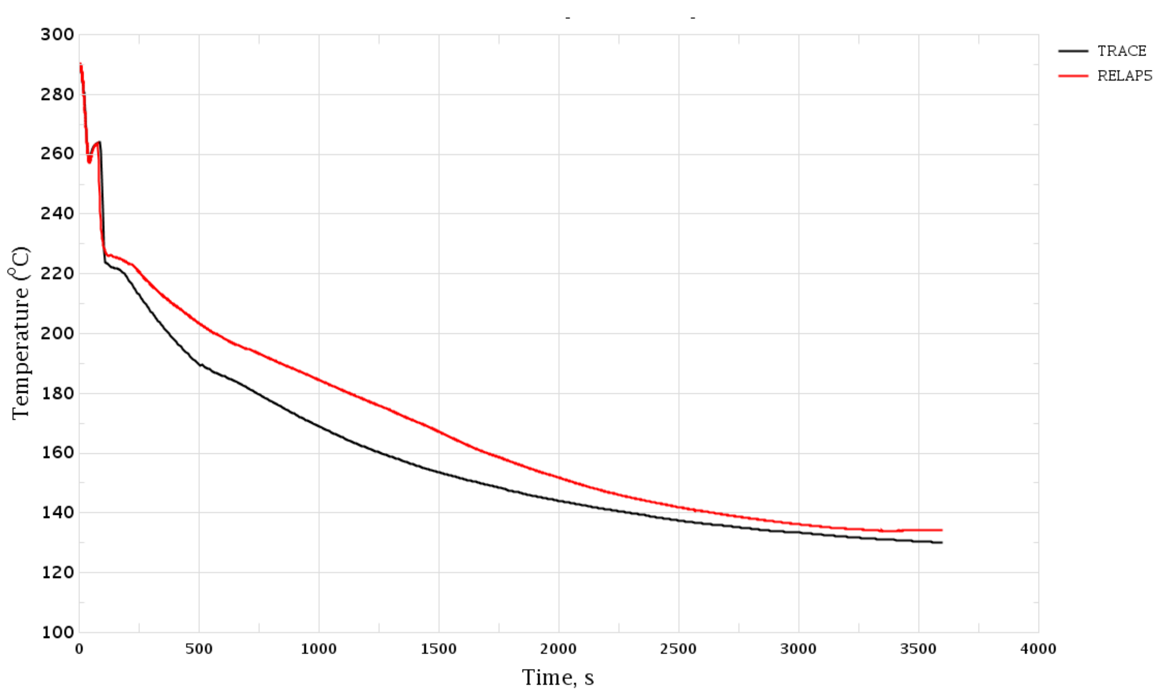

Figure 3-10 MSH Break. Coolant Temperature in Cold Leg, Loop 3

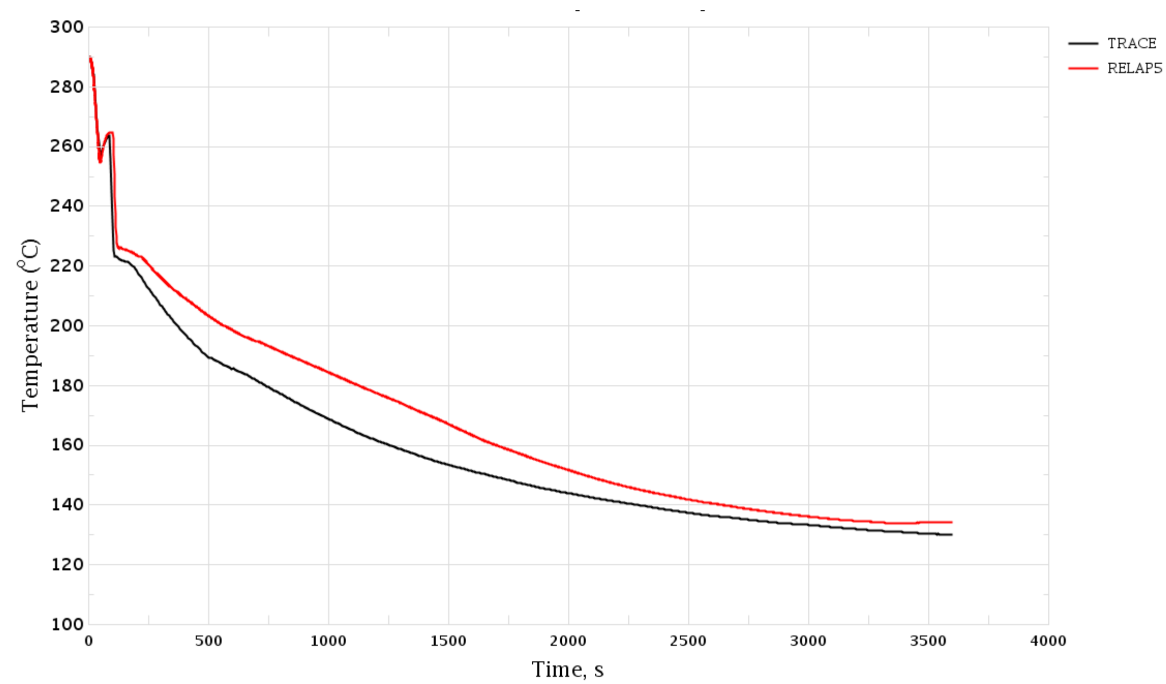

Figure 3-11 MSH Break. Coolant Temperature in Cold Leg, Loop 4

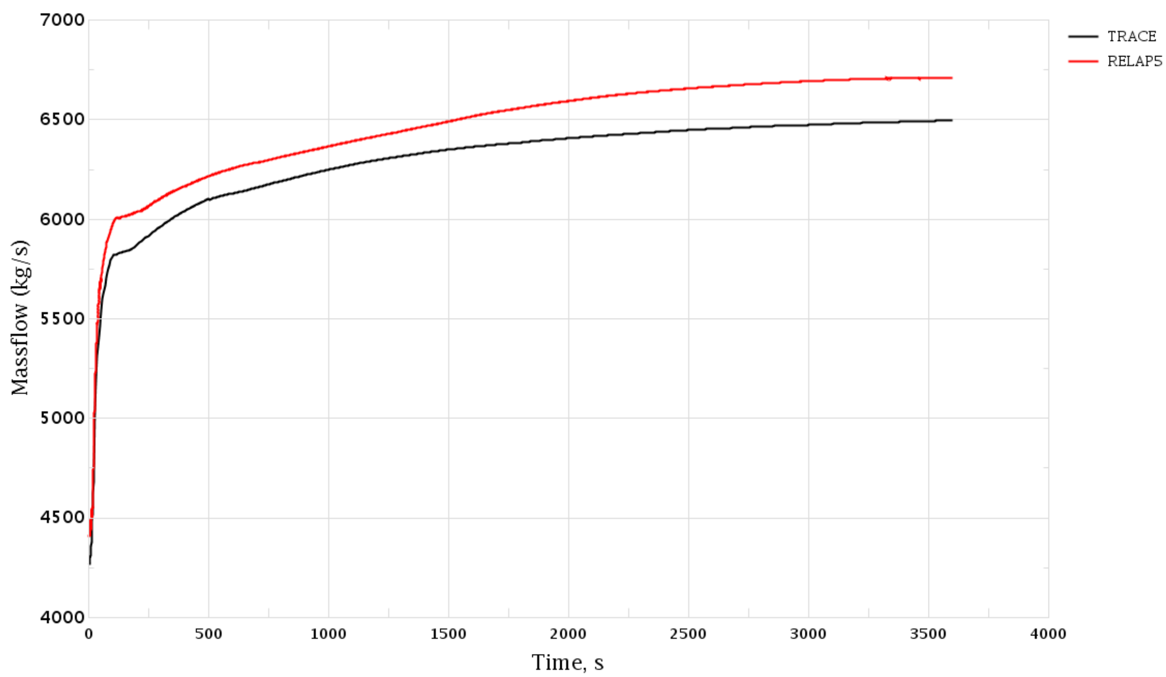

Figure 3-12 MSH Break. RCS Loop 1 Mass Flow Rate

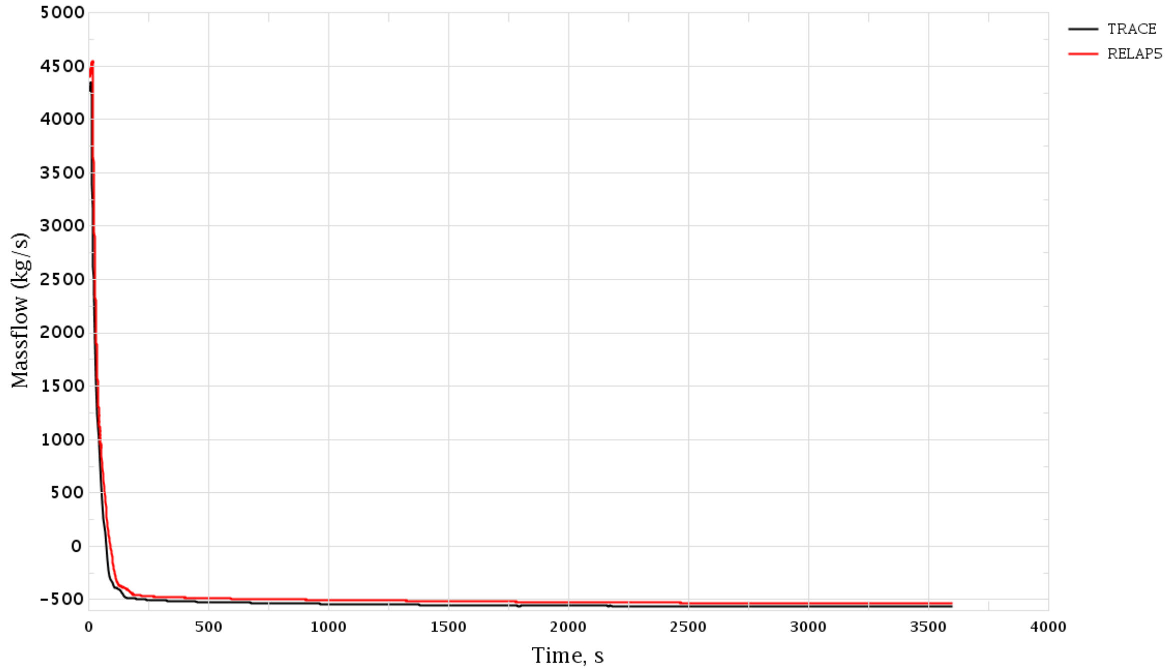

Figure 3-13 MSH Break. RCS Loop 2 Mass Flow Rate

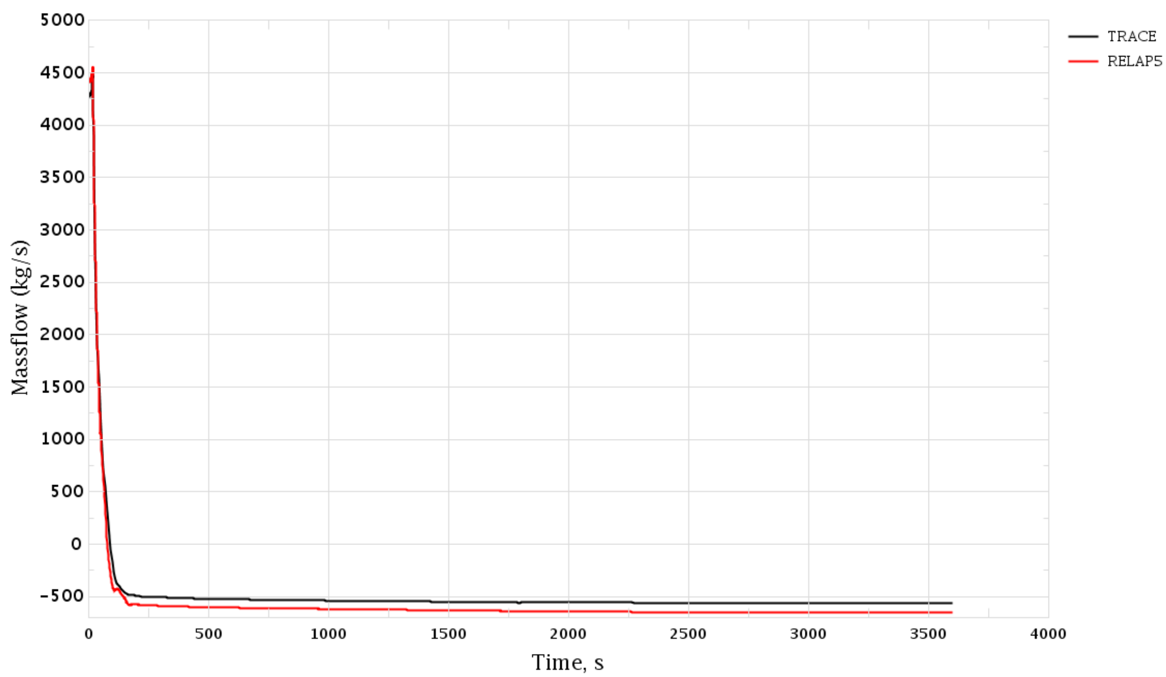

Figure 3-14 MSH Break. RCS Loop 3 Mass Flow Rate

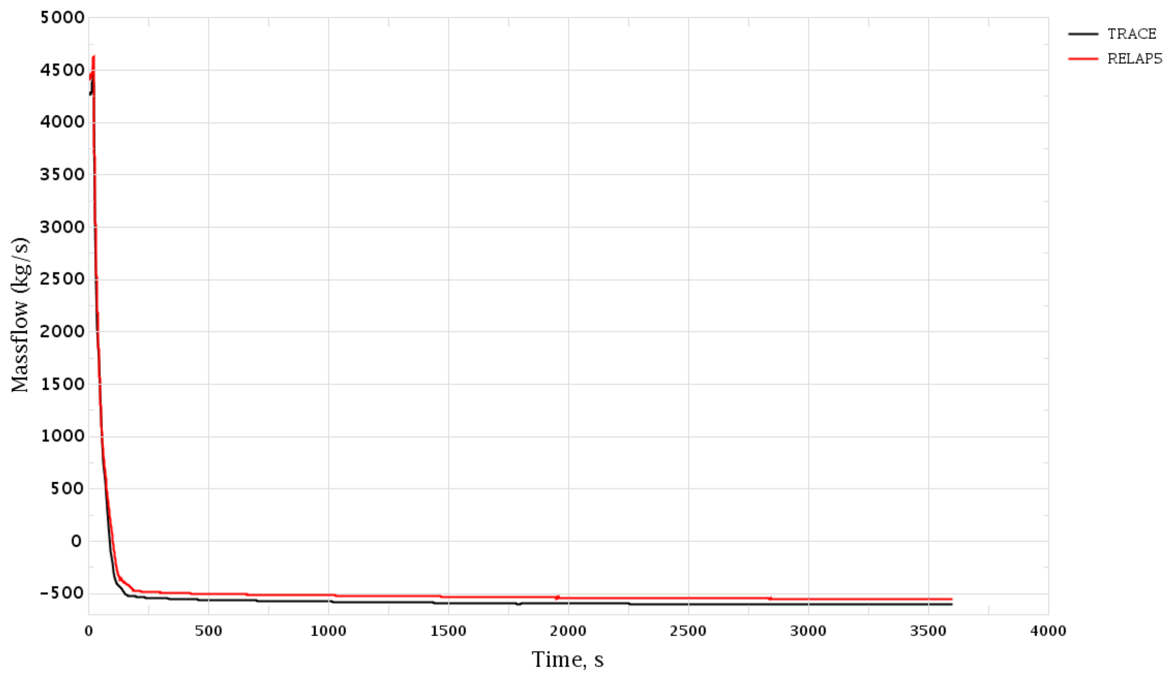

Figure 3-15 MSH Break. RCS Loop 4 Mass Flow Rate

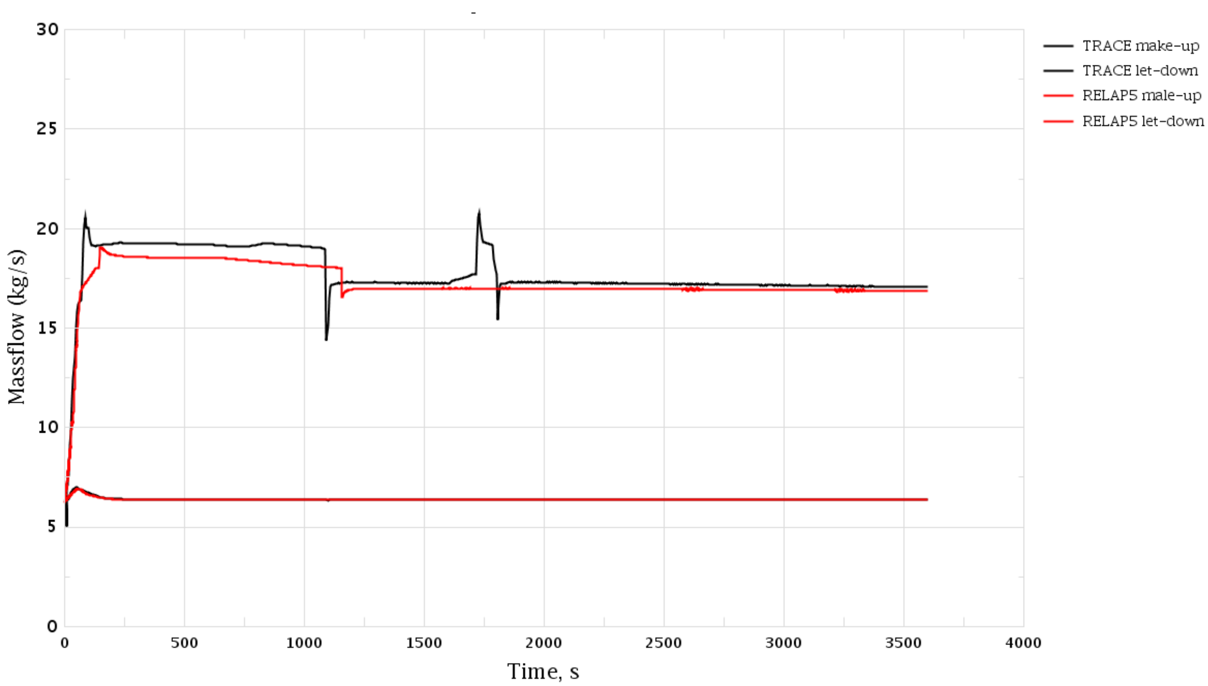

Figure 3-16 MSH Break. Make-Up and Let-Down Mass Flow Rate

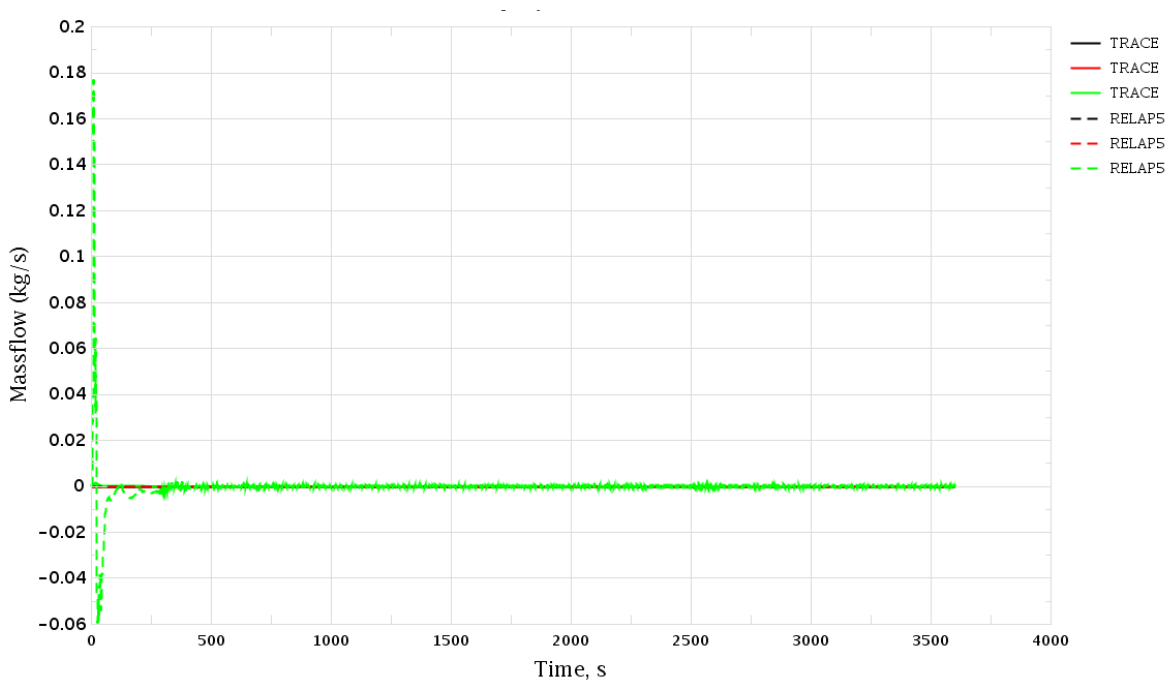

Figure 3-17 MSH Break. PRZ Spray Mass Flow Rate

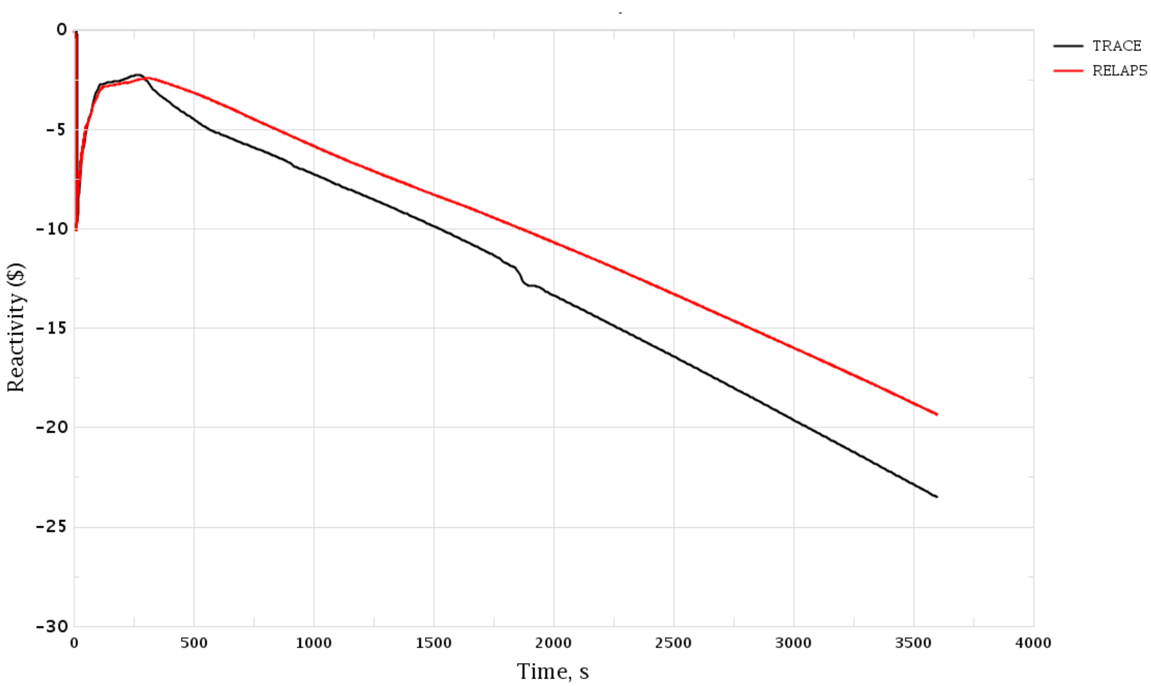

Figure 3-18 MSH Break. Core Reactivity

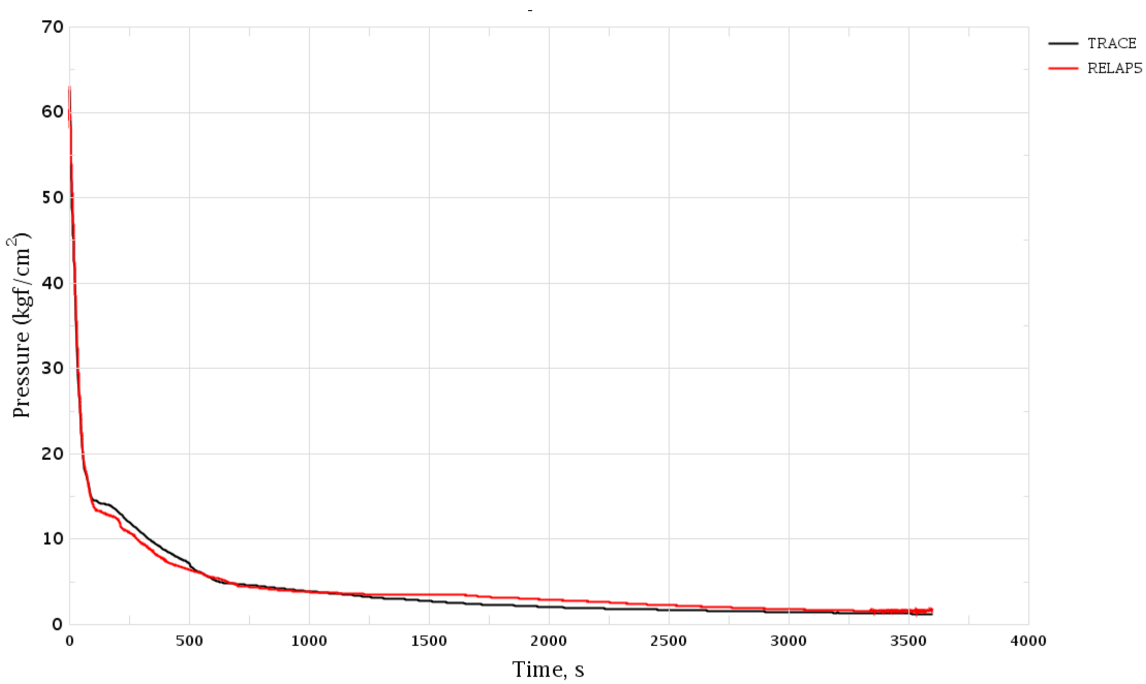

Figure 3-19 MSH Break. SG-1 Pressure

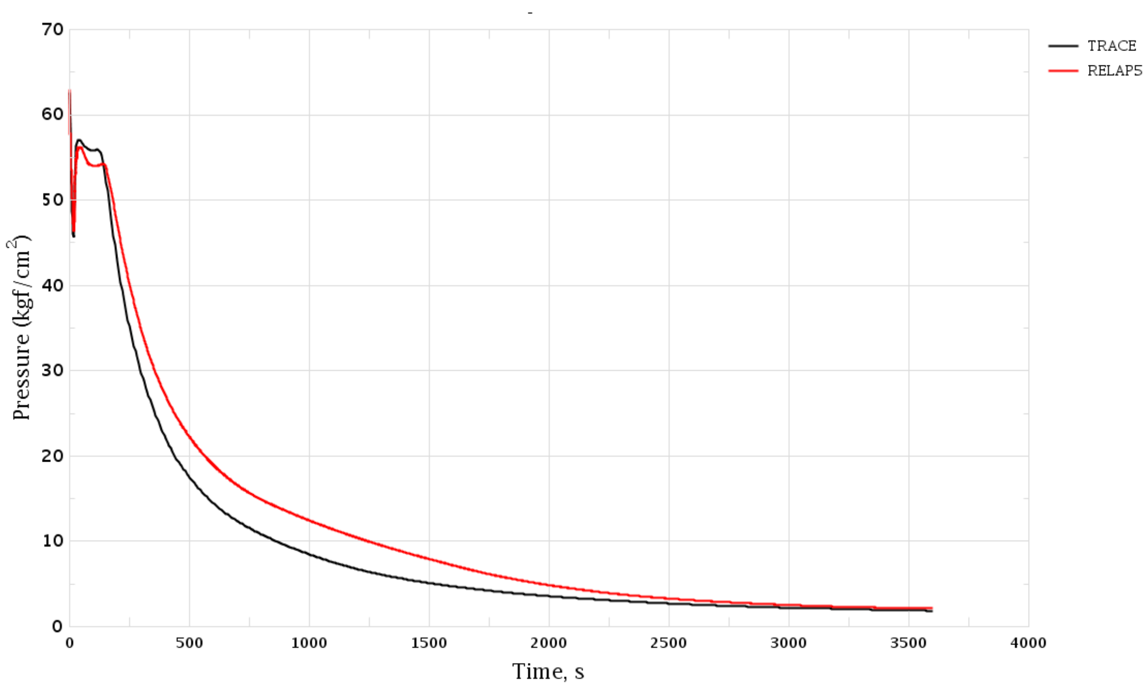

Figure 3-20 MSH Break. SG-2 Pressure

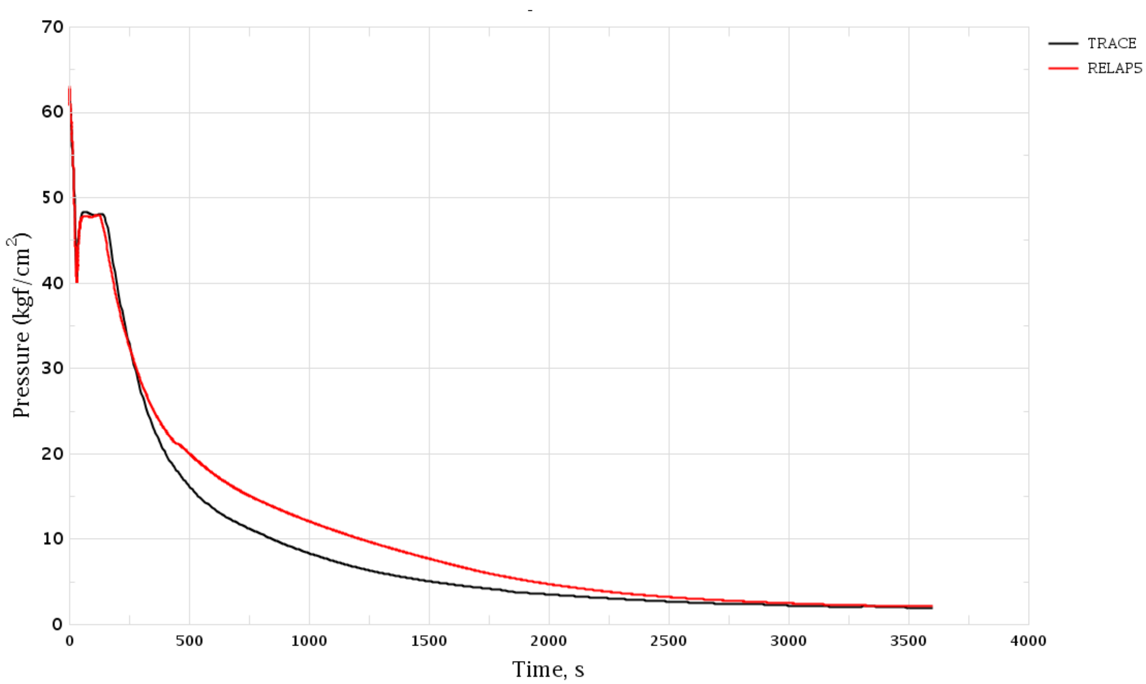

Figure 3-21 MSH Break. SG-3 Pressure

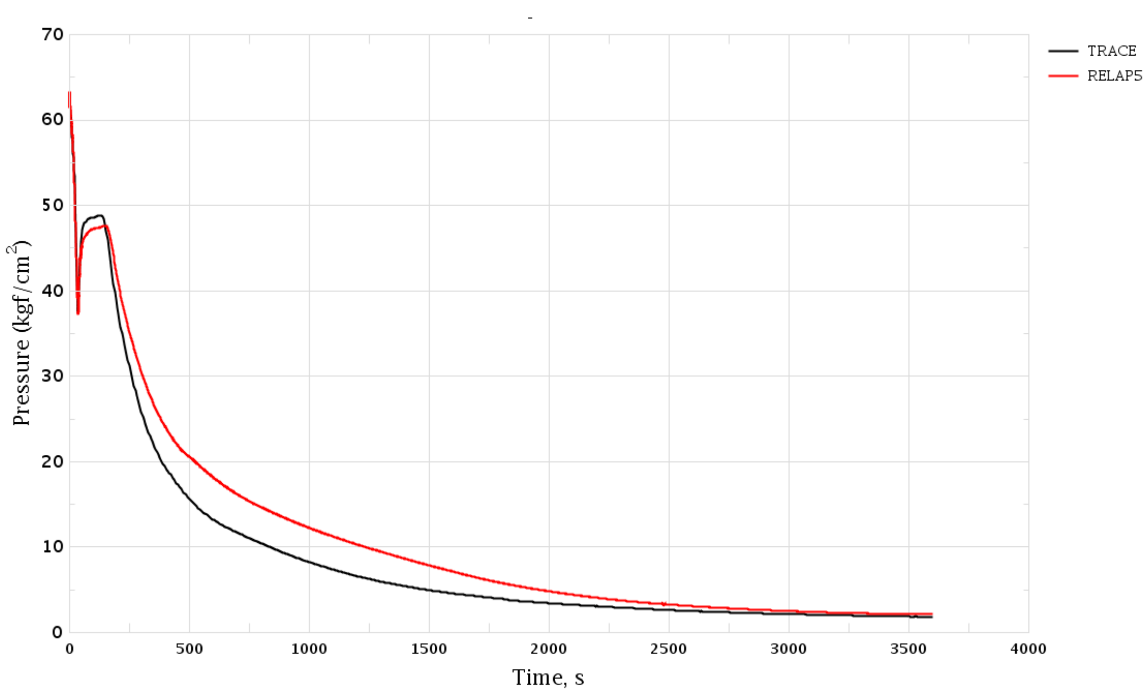

Figure 3-22 MSH Break. SG-4 Pressure

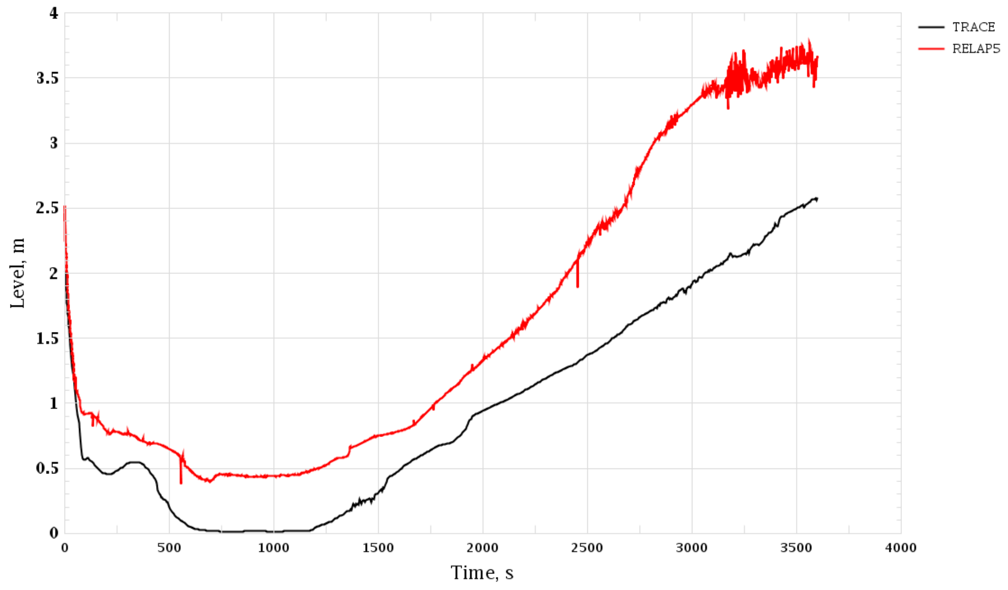

Figure 3-23 MSH Break. SG-1 Level (Wide Range)

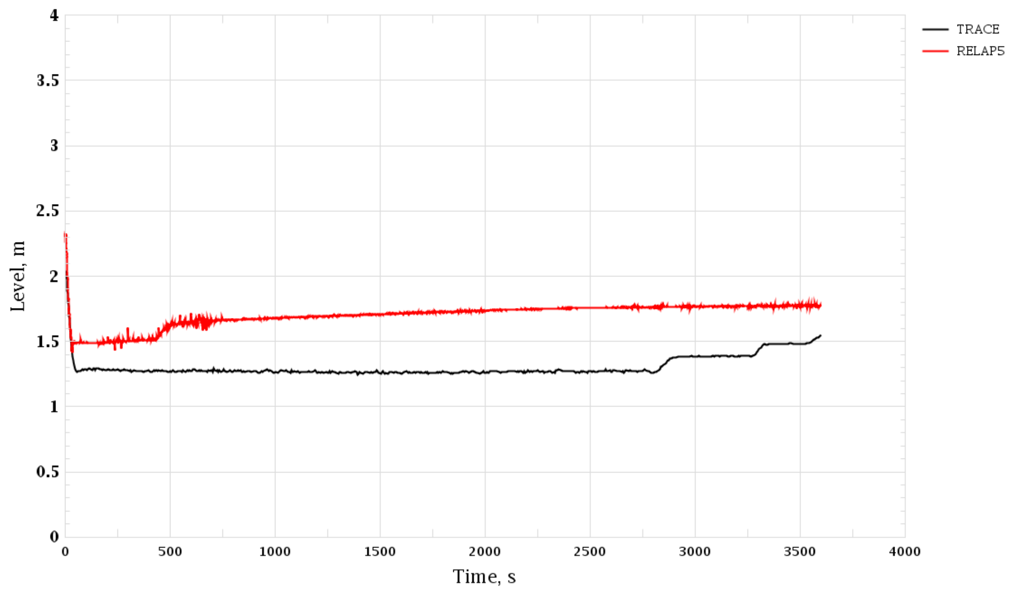

Figure 3-24 MSH Break. SG-2 Level (Wide Range)

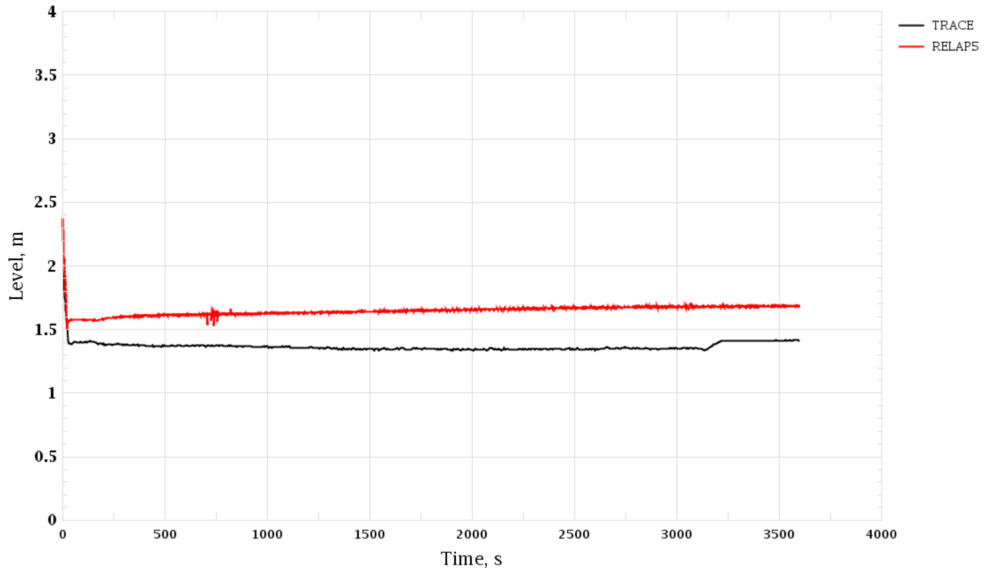

Figure 3-25 MSH Break. SG-3 Level (Wide Range)

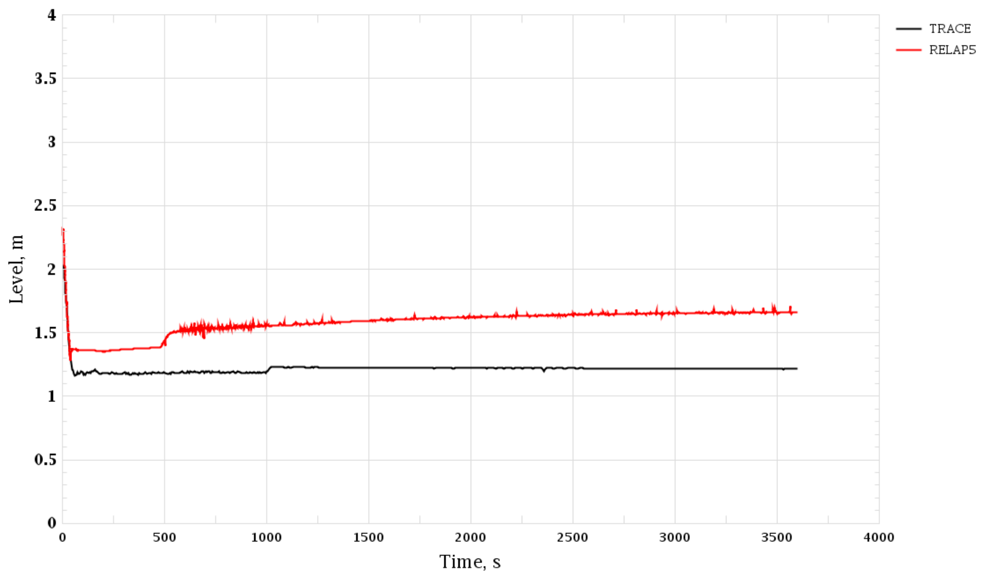

Figure 3-26 MSH Break. SG-4 Level (Wide Range)

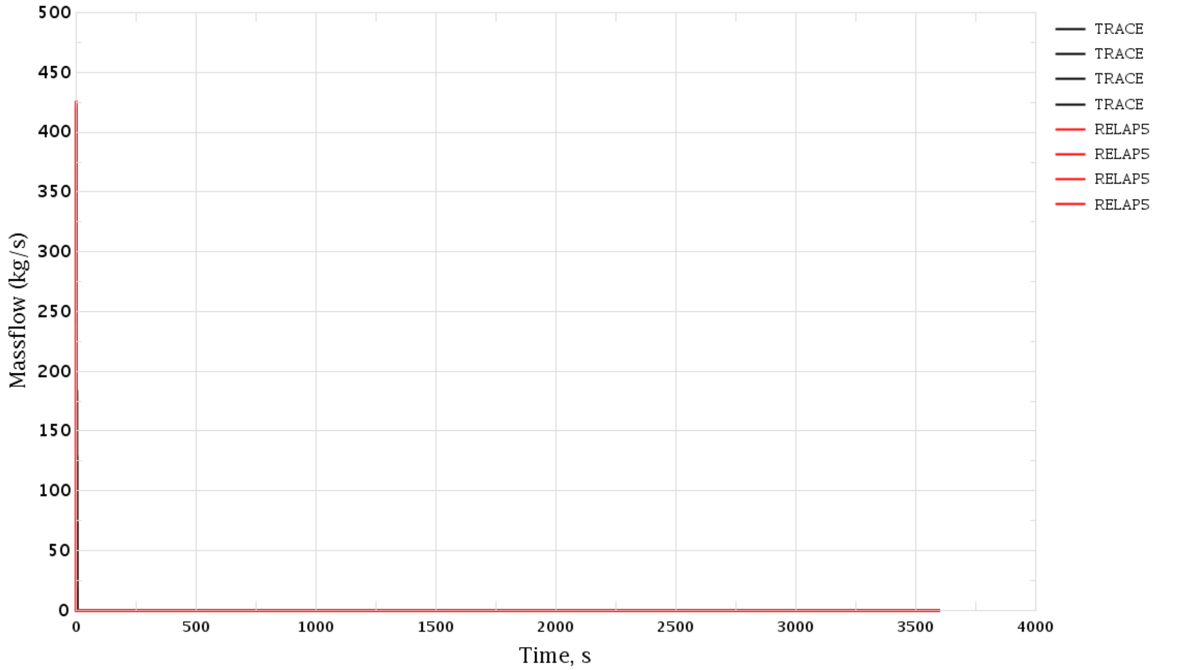

Figure 3-27 MSH Break. Turbine Mass Flow Rate

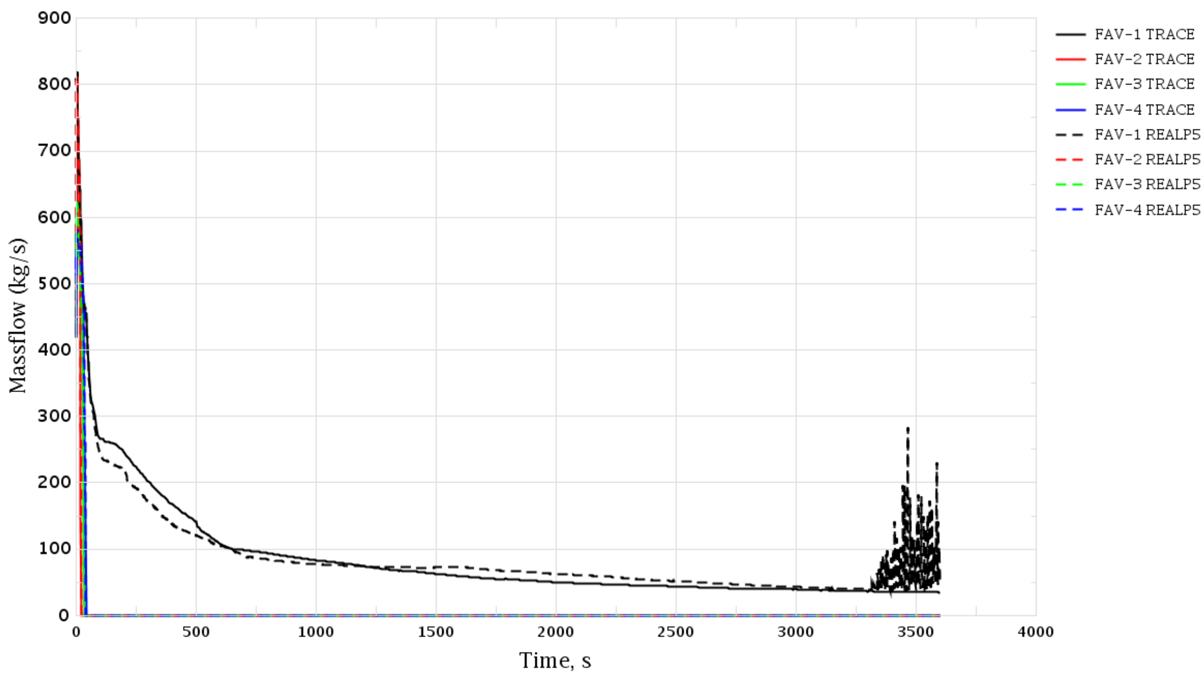

Figure 3-28 MSH Break. Mass Flow through MSIV

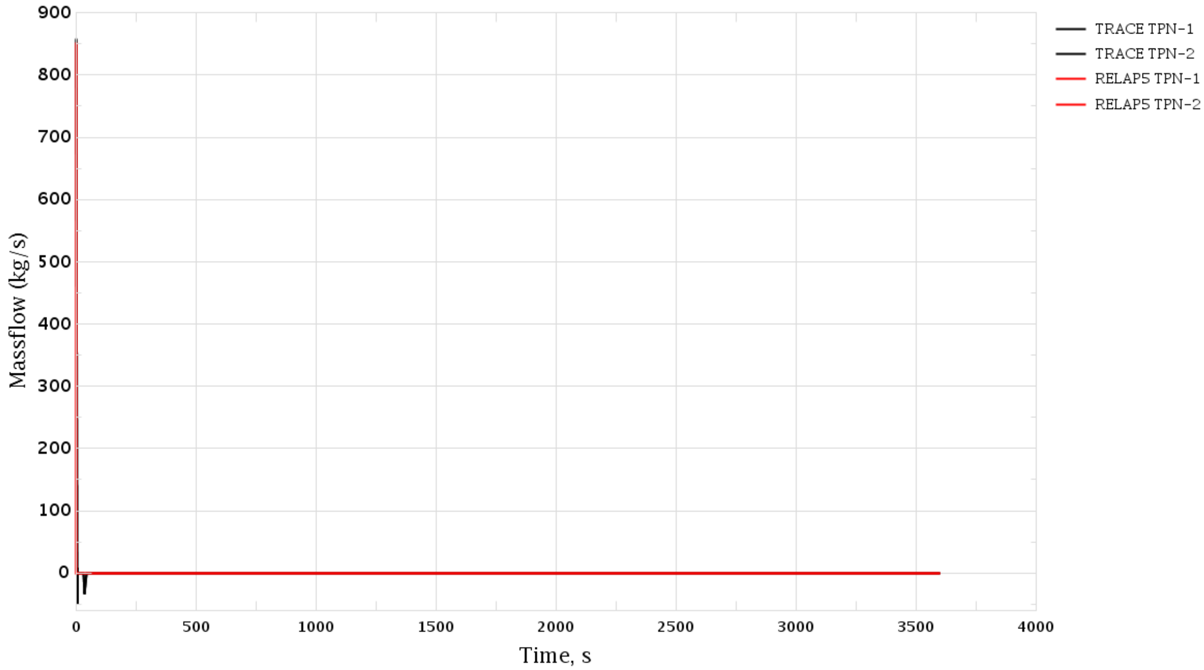

Figure 3-29 MSH Break. MFW Pumps Mass Flow Rate

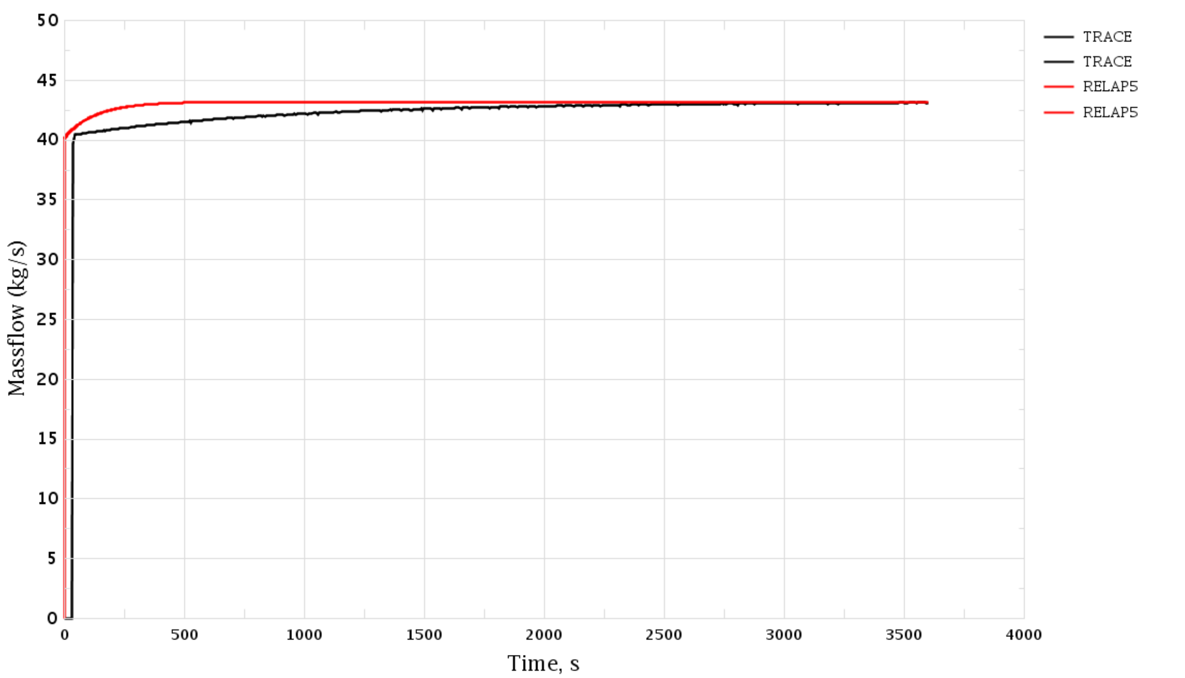

Figure 3-30 MSH Break. AFW Pumps Mass Flow Rate

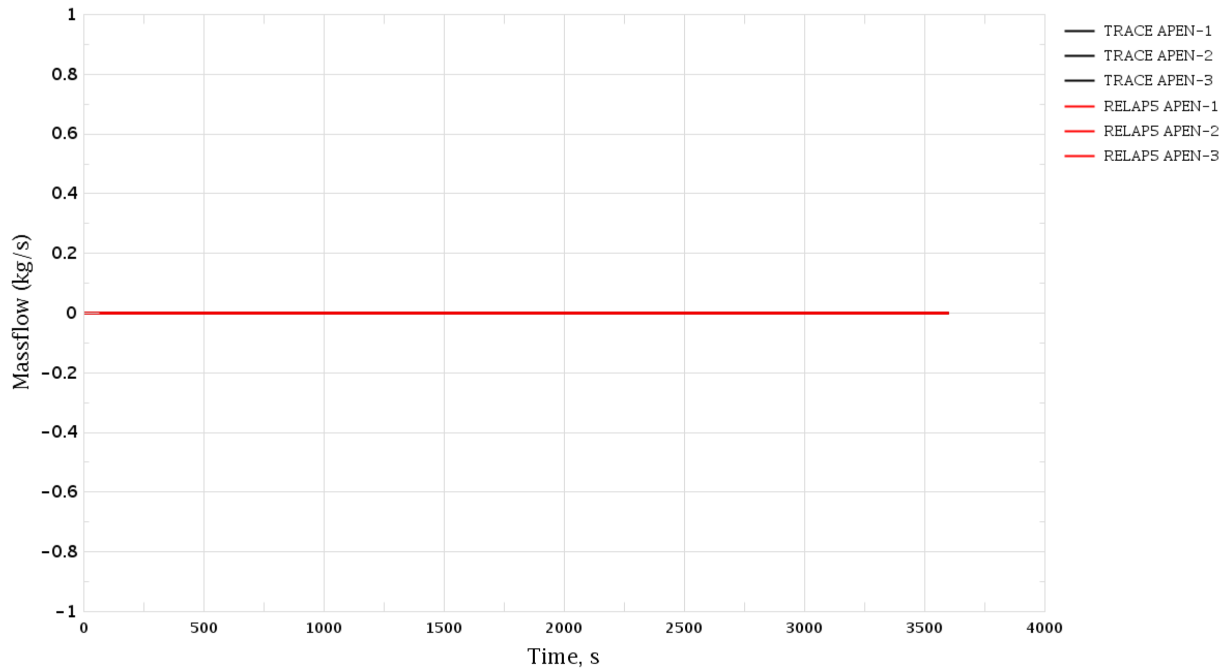

Figure 3-31 MSH Break. EFW Pumps Mass Flow Rate

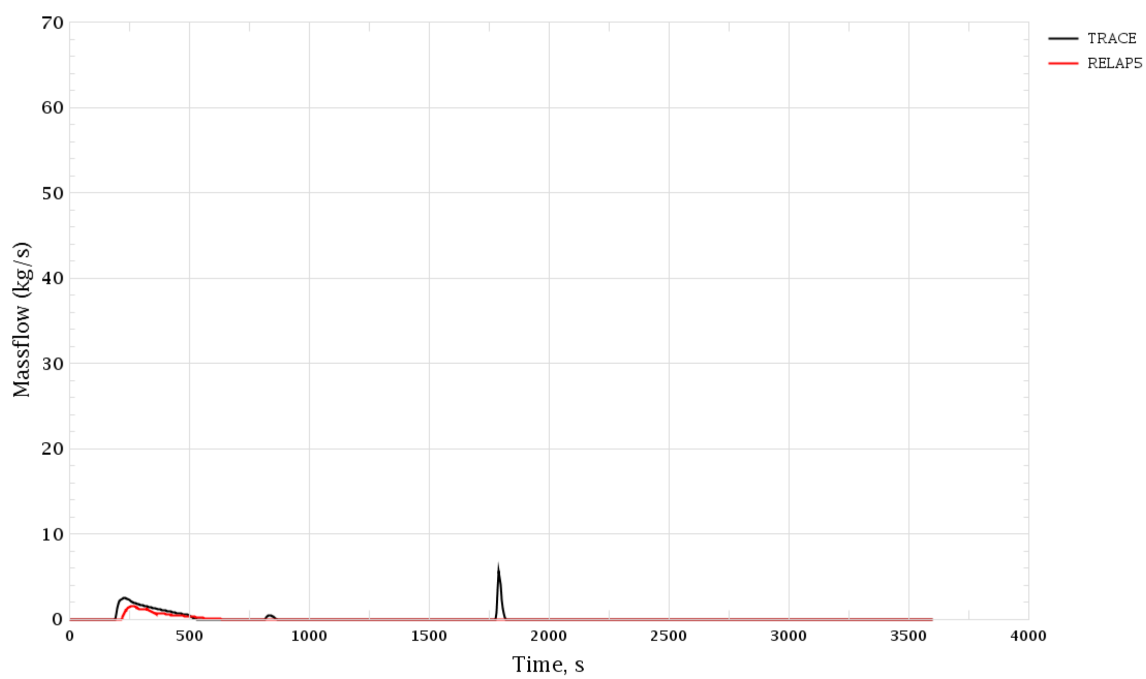

Figure 3-32 MSH Break. HPIS-1 Mass Flow Rate

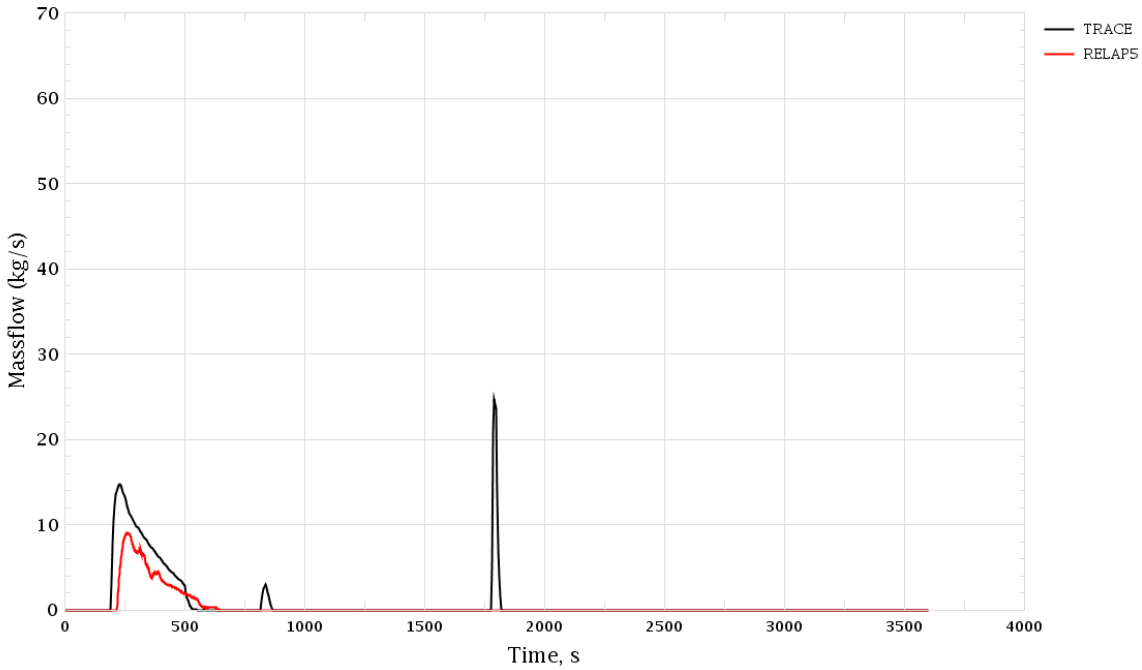

Figure 3-33 MSH Break. HPIS-2 Mass Flow Rate

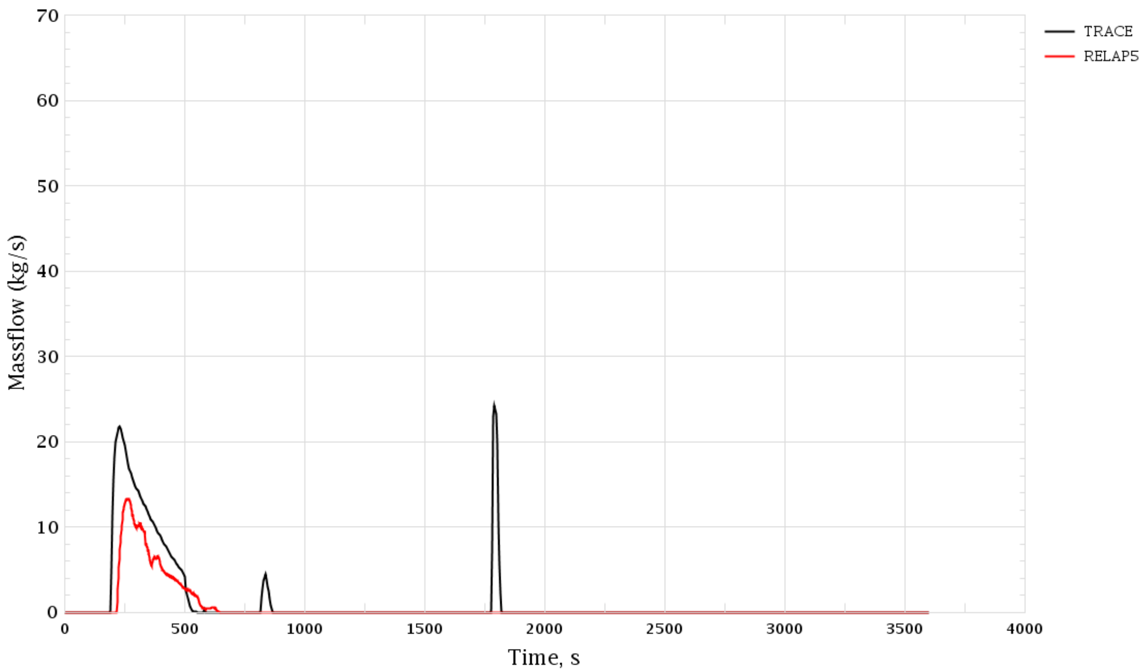

Figure 3-34 MSH Break. HPIS-3 Mass Flow Rate

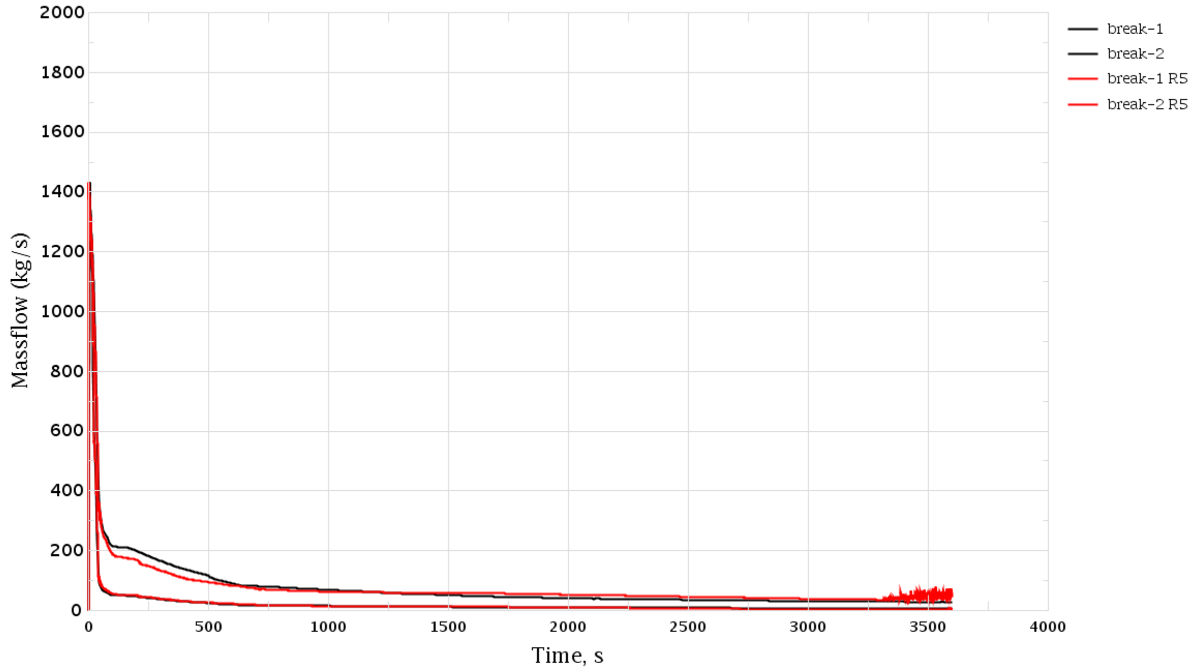

Figure 3-35 MSH Break. Break Mass Flow Rate

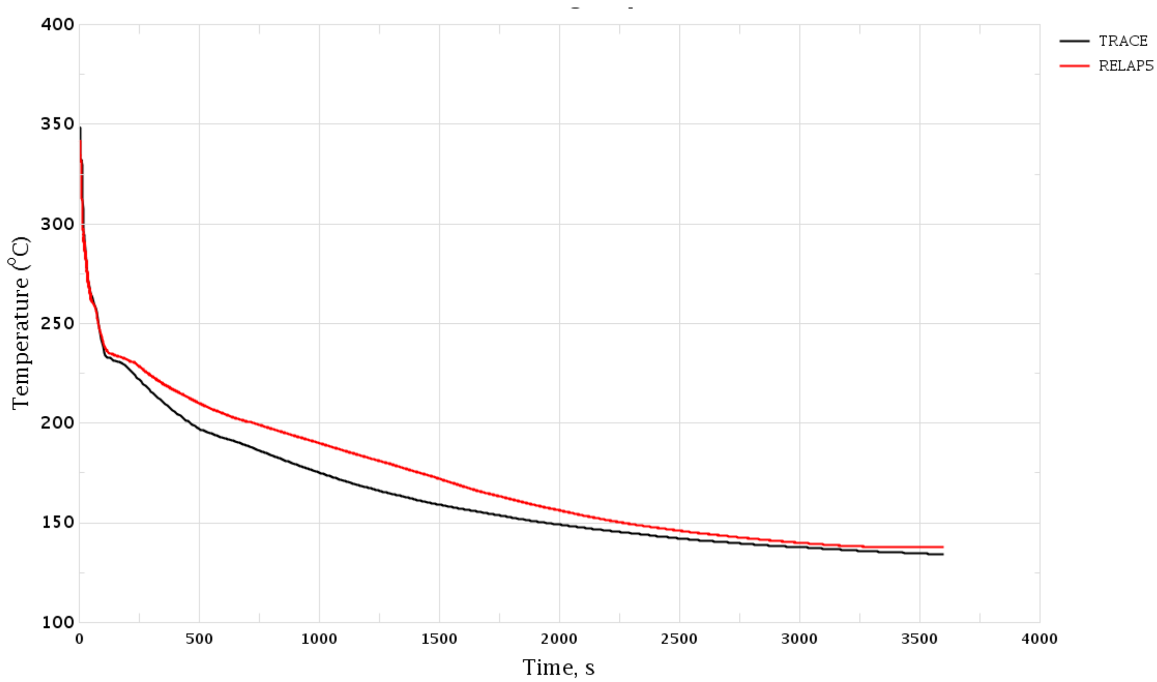

Figure 3-36 MSH Break. Maximal Cladding Temperature

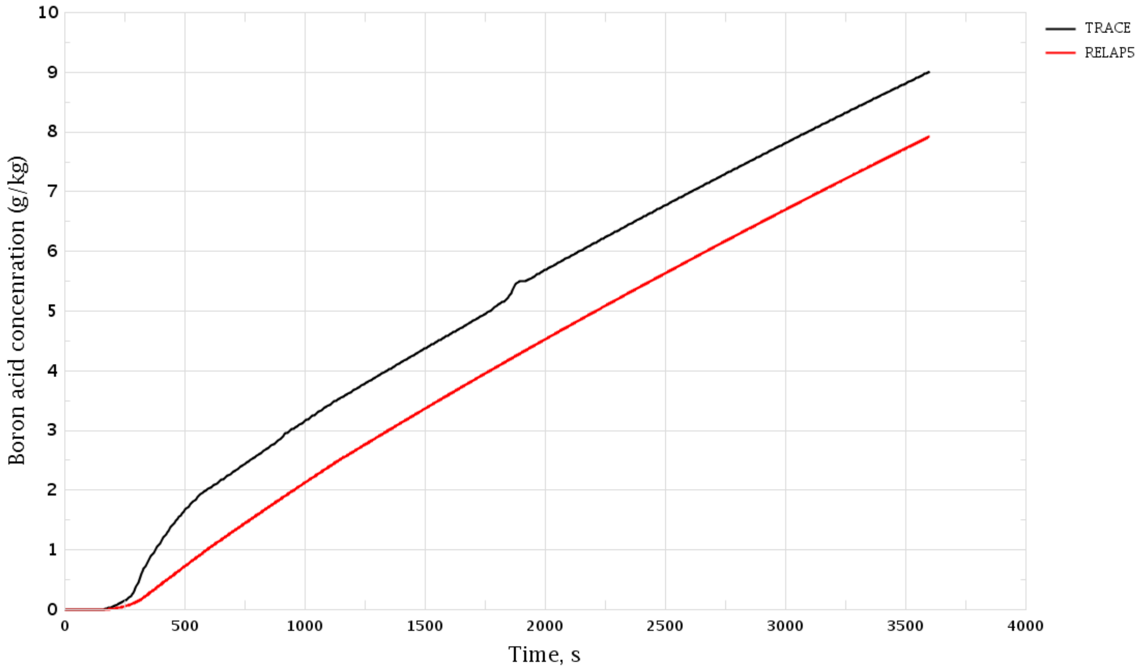

Figure 3-37 MSH Break. Boric Acid Concentration in the Core

3.2 Loss of Turbine Condenser Vacuum

3.2.1 Brief Description of Initiating Event

This initiating event postulates loss of vacuum in the turbine condenser that leads to a trip of the turbine with closure of the stop valves and prohibition of operation of turbine bypass to the turbine condenser (BRU-K). 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.

The initiating event is modeled as closure of turbine stop valves (closure time is 0.5 s) with dependent failure of BRU-K.

No operator actions are simulated in the scenario.

Operation of ARM, RPL, PZ and UPZ to decrease reactor power is not considered. Reactor scram operates according to the design with a delay of 1.0 s from formation of actuation signal to a start of control rods drop. Maximal allowed time of control rods drop to the core (4 s) is assumed.

MFW pumps trip is postulated at the moment of turbine stop valves closure, operation of AFW pumps is not taken into account to prolong decrease of heat removal by the secondary circuit.

Operation of PRZ spray, as well as operation of the make-up and let-down system is also 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	Start of turbine stop valves closure upon pressure increase in condenser. Prohibition for BRU-K operation	Due to loss of vacuum in a turbine condenser
0.5	0.5	Full closure of turbine stop valves. Trip of MFW pumps	Boundary conditions
2.2	2.6	Trip of PRZ heaters group no.1	RCS pressure increase above 161 kgf/cm^2
4.0	3.5	Opening of steam dump valves to atmosphere (BRU-A). Start of BRU-A operation in pressure maintenance mode	Increase of SG pressure up to 73 kgf/cm^2
6.0	5.8	Reactor scram signal	Increase of SG pressure above 80 kgf/cm^2 and time after RCP trip is less than 50 s
10.0	8.0	Opening of SG SRV Maximal secondary circuit pressure is reached (84.0 kgf/cm^2 in TRACE calculation and 84.2 kgf/cm^2 in RELAP5)	Increase of SG pressure above 84 kgf/cm^2
10.0	13.5	Maximal primary circuit pressure is reached (184.9 kgf/cm^2 in TRACE calculation and 175.0 kgf/cm^2 in RELAP5)	
22.0	23.0	Closure of SG SRV	Decrease of SG pressure below 70 kgf/cm^2
30.0	40.0	Start of PRZ heaters' groups operation	Switching on/off of PRZ heaters' groups according to design algorithm of RCS pressure maintenance
50.0-60.0	50.0-60.0	Trip of all RCPs Signal to close main control valves of MFW supply to individual SGs	Decrease of SG level for 500 mm below nominal (by wide range level meter) Due to trip of RCPs
50.0	80.0	Minimal RCS pressure is reached (143.0 kgf/cm^2 in TRACE and 126 kgf/cm^2 in RELAP5 calculation)	
755.0	900.0	Start of EFW supply to SG	Decrease of SG level below 1.35 m
1750.0	1650.0	Restoration of SG level	Due to EFW operation
4000.0	4000.0	End of calculation	Stabilization of main parameters

Due to vacuum loss in the turbine condenser, a signal to close the turbine's stop valves is formed with a prohibition of BRU-K operation. At the moment of stop valves closure the MFW pumps trip is simulated (Figure 3-70). Fast termination of steam flow from SG causes increase of SG pressure (Figure 3-57 – Figure 3-60) with start of BRU-As operation in pressure maintenance mode (Figure 3-68), and initial (within first seconds of transient) increase in the primary circuit temperature (Figure 3-41 – Figure 3-48) and pressure (Figure 3-39). When SG pressure reaches 80 kgf/cm^2 the scram signal is formed and reactor power sharply decreases down to a decay heat (Figure 3-38). At 8.0 s in RELAP calculation and at 10.0 s in TRACE calculation the secondary circuit pressure increases up to 84 kgf/cm^2 that causes short-term opening of SG SRV (Figure 3-69).

Decrease of reactor power and heat removal by dumping steam via BRU-As causes decrease of RCS pressure and temperature until balance between heat generated by the reactor core and heat removed by the secondary circuit is established. Primary circuit pressure is maintained by periodic operation of PRZ heaters (Figure 3-54).

Due to trip of MFW pumps at the beginning of transient and postulated failure of AFW pumps the SG level gradually decreases (Figure 3-61 – Figure 3-64). Within 50–60 s of transient SG level decreases for 500 mm from the nominal value that causes actuation of signal to trip RCPs and termination of forced circulation of primary circuit coolant (Figure 3-49 – Figure 3-52). Further decrease of SG level down to 1.35 m leads to beginning of EFW supply to SGs at 900 s in RELAP calculation and 755 s in TRACE calculation.

At ~1500–1600 s SG level is restored by operation of EFW and within next 500–1000 s stabilization of the main primary and secondary circuit parameters occurs. The heat generated in the reactor core is removed in the natural circulation mode by the secondary circuit via dumping steam via BRU-As and SG feeding by EFW pumps from correspondent tanks. Calculation is terminated at 4000 s.

The maximal primary circuit pressure reached in calculation is 184.9/175.0 kgf/cm^2 (Figure 3-39) in RELAP and TRACE calculations, respectively. Maximal secondary pressure is 84.2/84.0 kgf/cm^2 (Figure 3-57 – Figure 3-60).

The plots of the main parameters of calculation are presented below on Figure 3-38 – Figure 3-73.

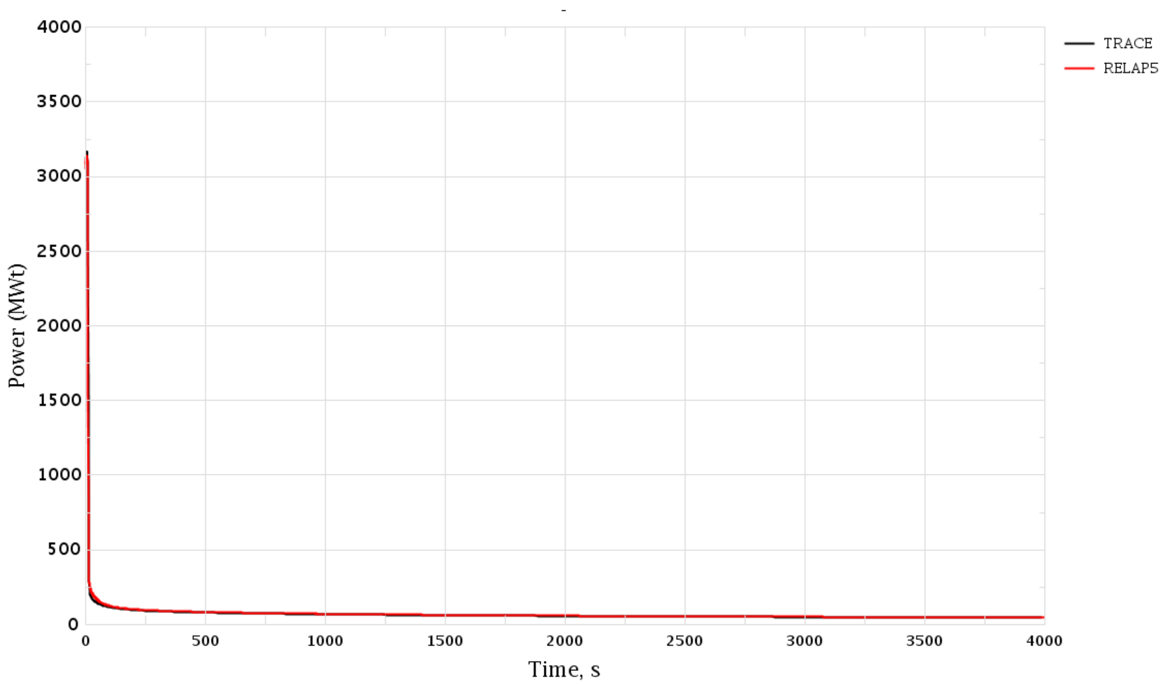

Figure 3-38 Loss of Turbine Condenser Vacuum. Core Thermal Power

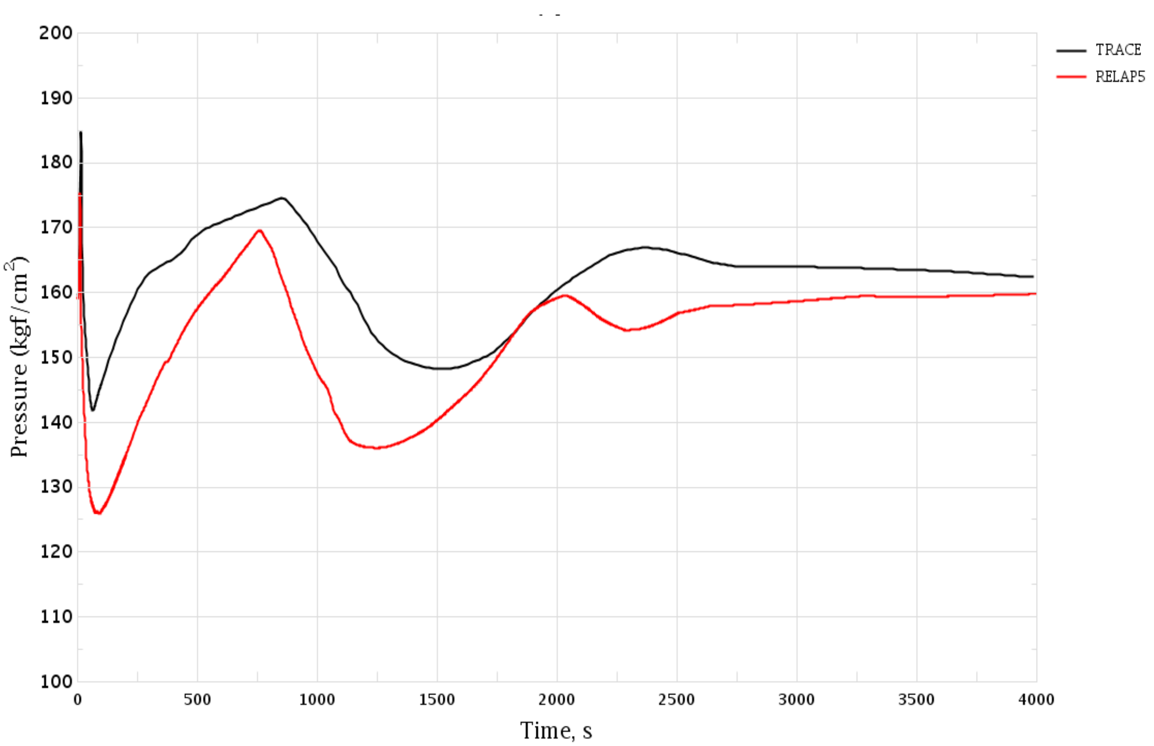

Figure 3-39 Loss of Turbine Condenser Vacuum. RCS Pressure

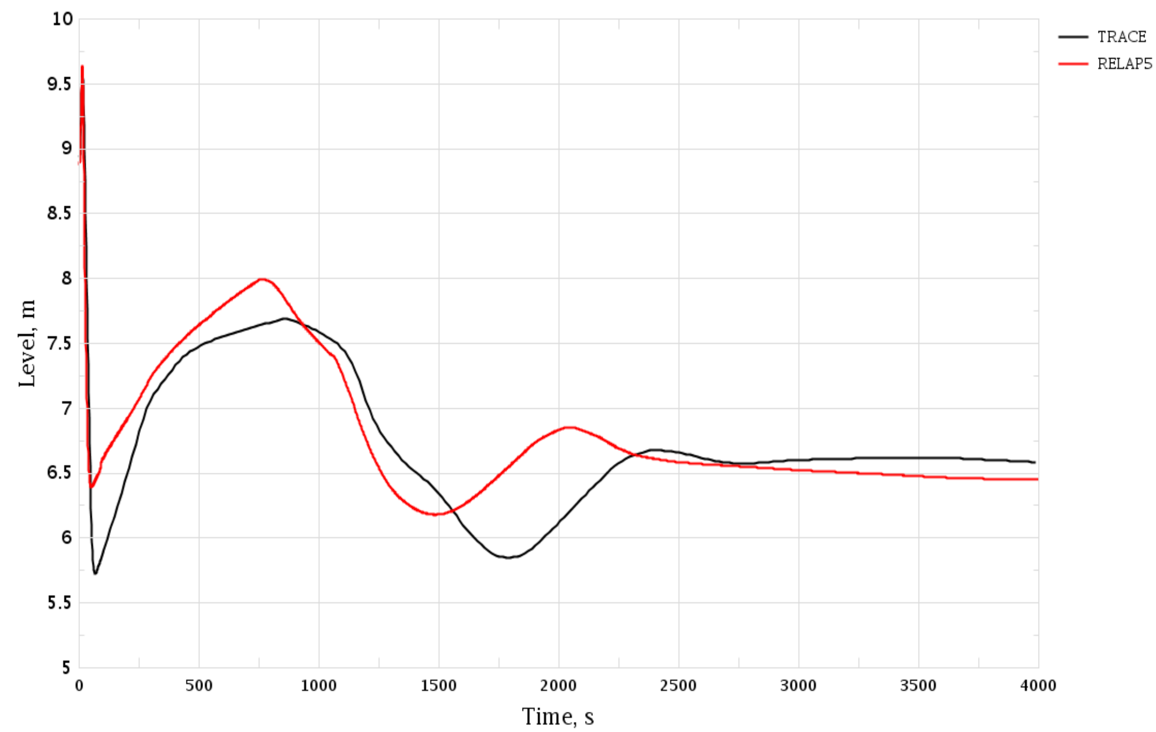

Figure 3-40 Loss of Turbine Condenser Vacuum. Pressurizer Level

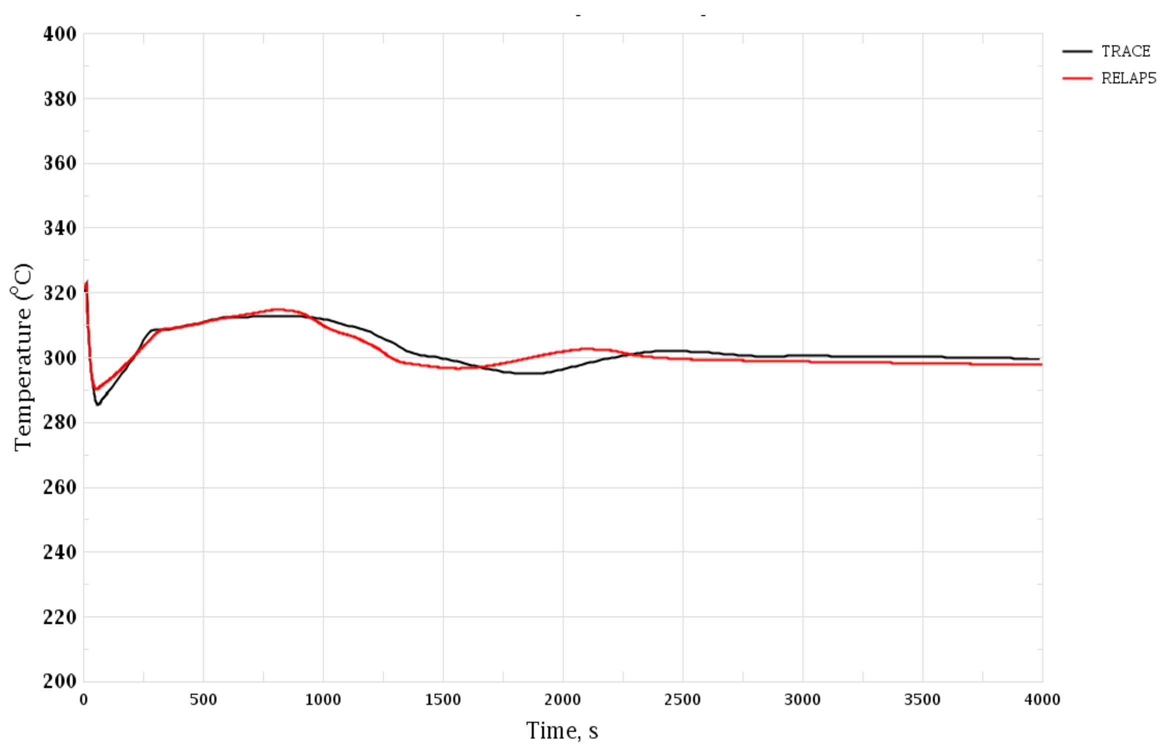

Figure 3-41 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 1

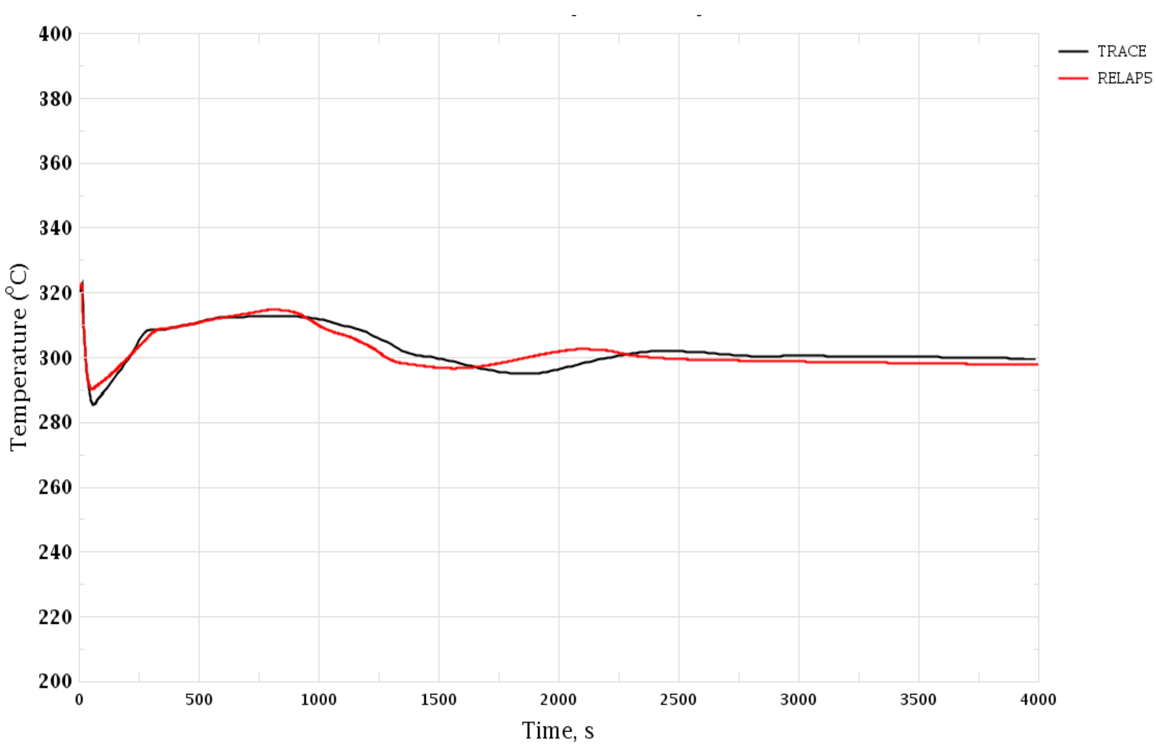

Figure 3-42 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 2

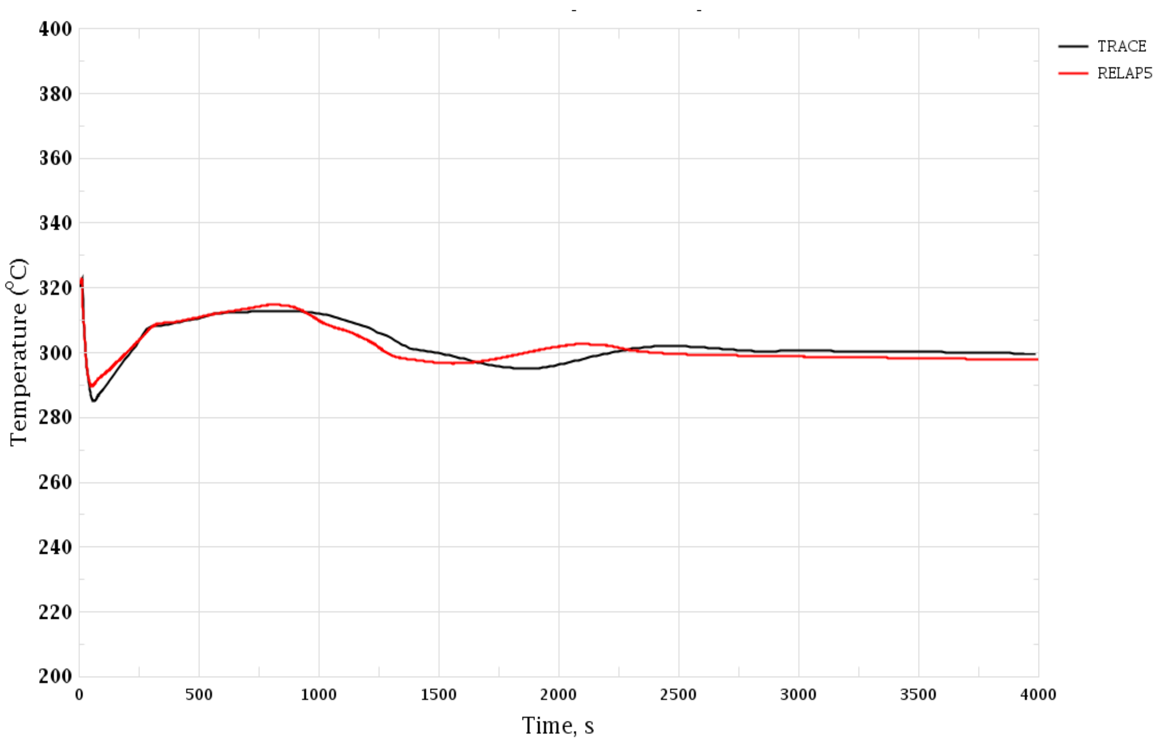

Figure 3-43 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 3

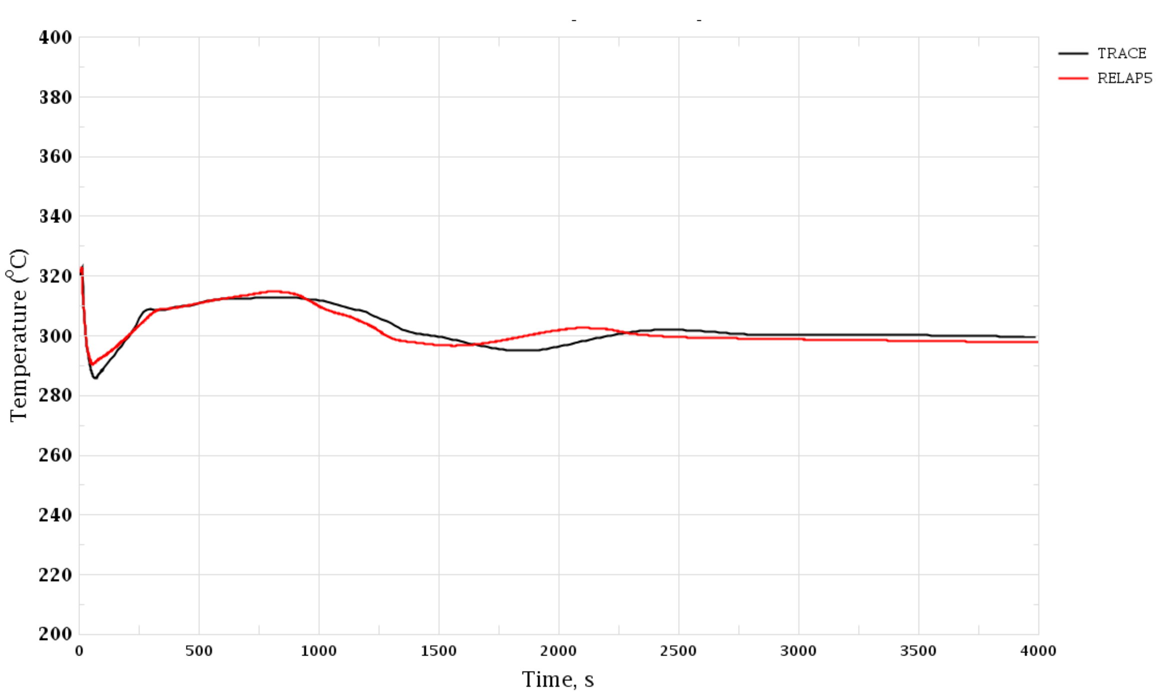

Figure 3-44 Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 4

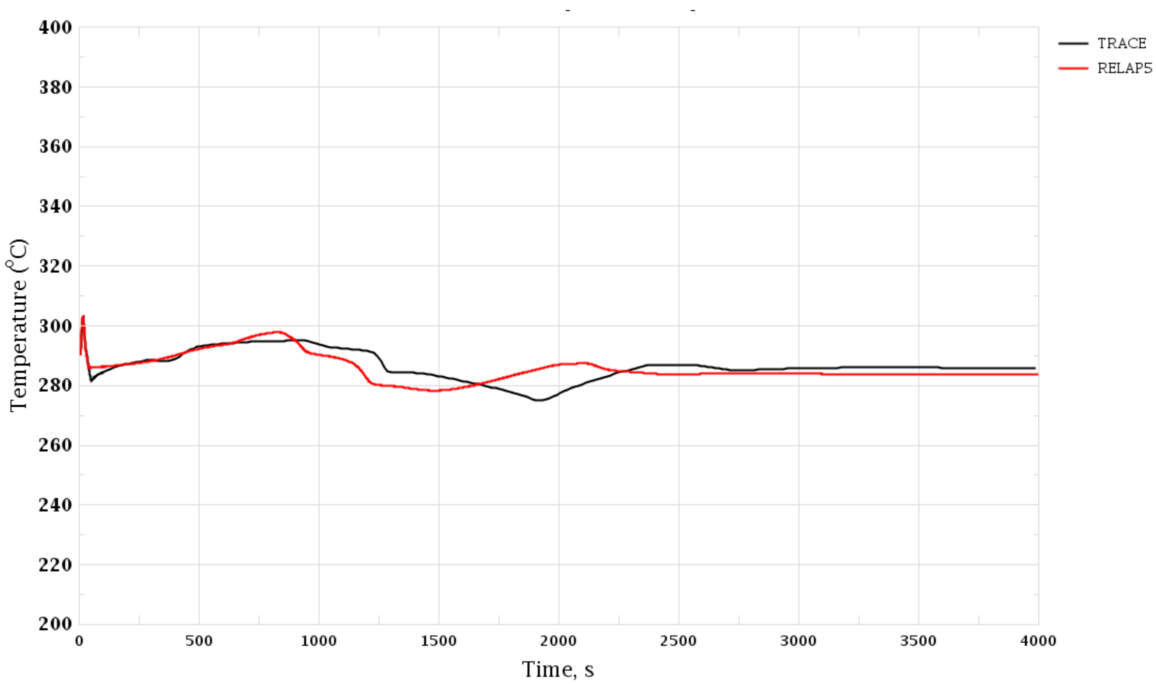

Figure 3-45 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 1

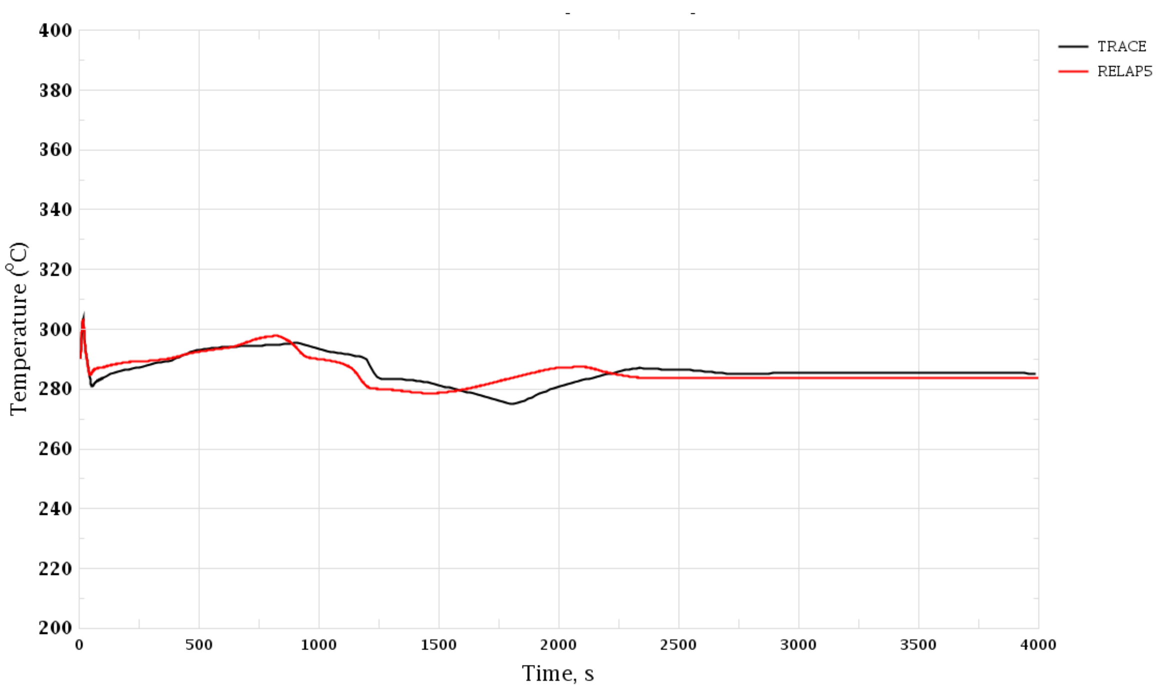

Figure 3-46 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 2

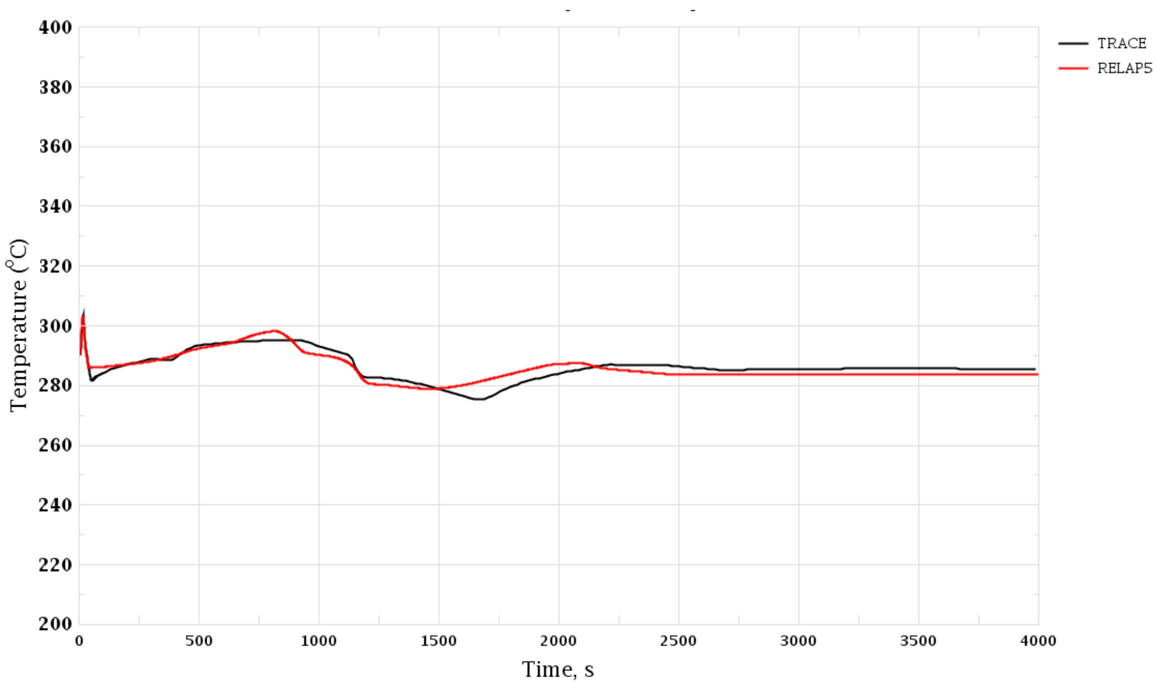

Figure 3-47 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 3

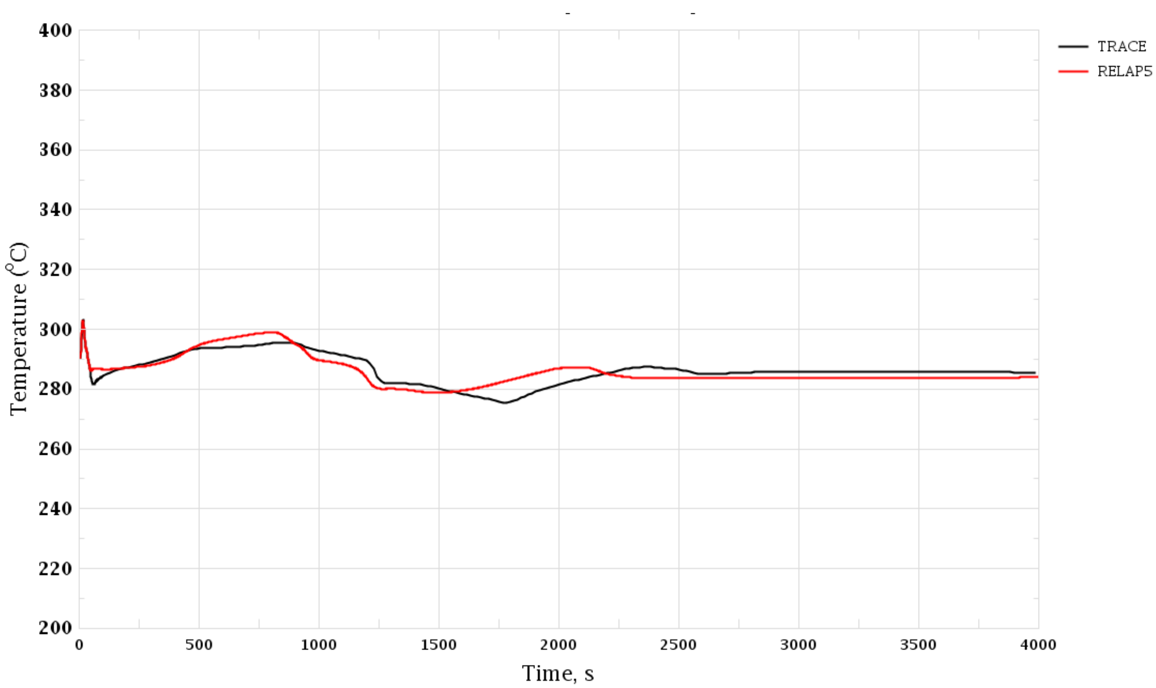

Figure 3-48 Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 4

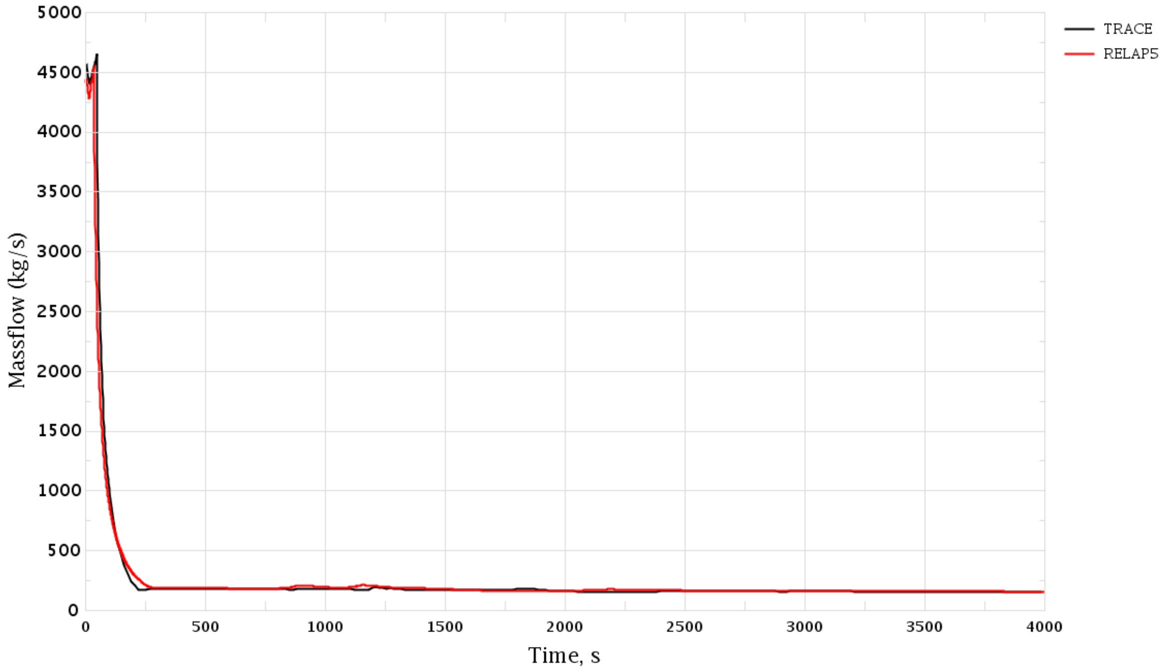

Figure 3-49 Loss of Turbine Condenser Vacuum. RCS Loop 1 Mass Flow Rate

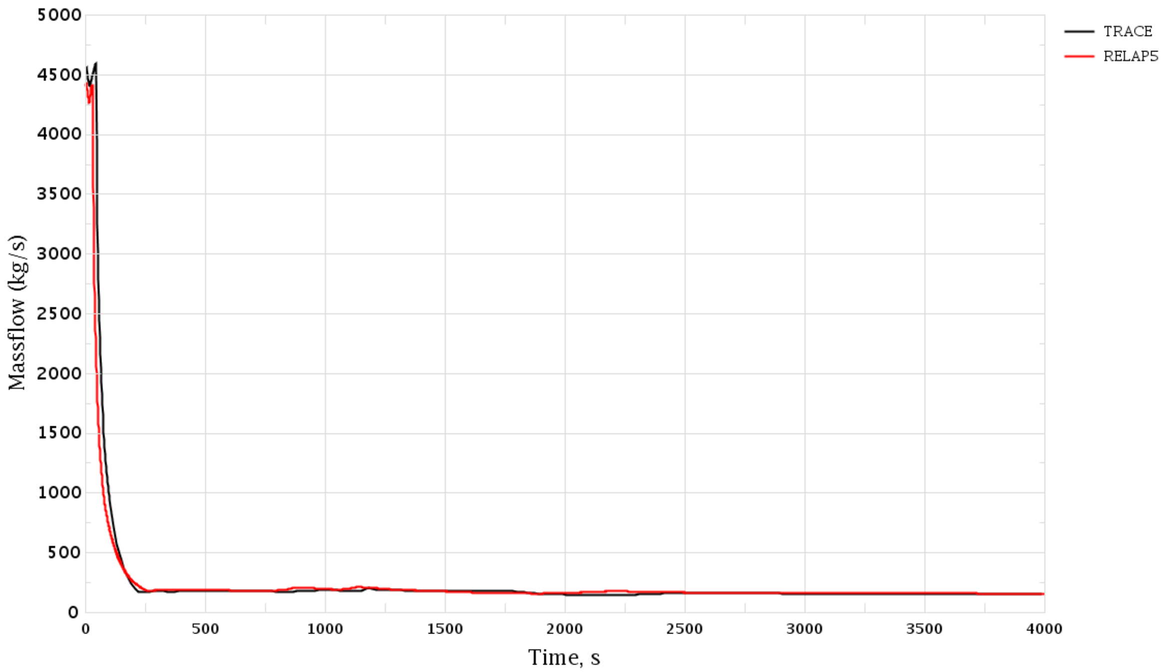

Figure 3-50 Loss of Turbine Condenser Vacuum. RCS Loop 2 Mass Flow Rate

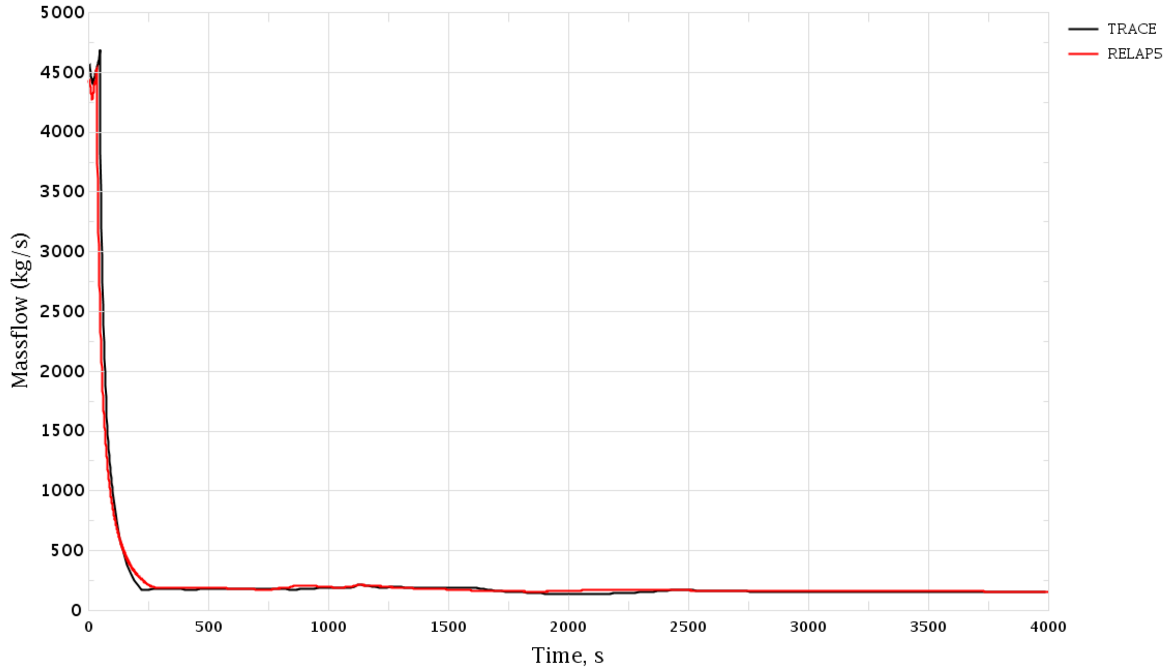

Figure 3-51 Loss of Turbine Condenser Vacuum. RCS Loop 3 Mass Flow Rate

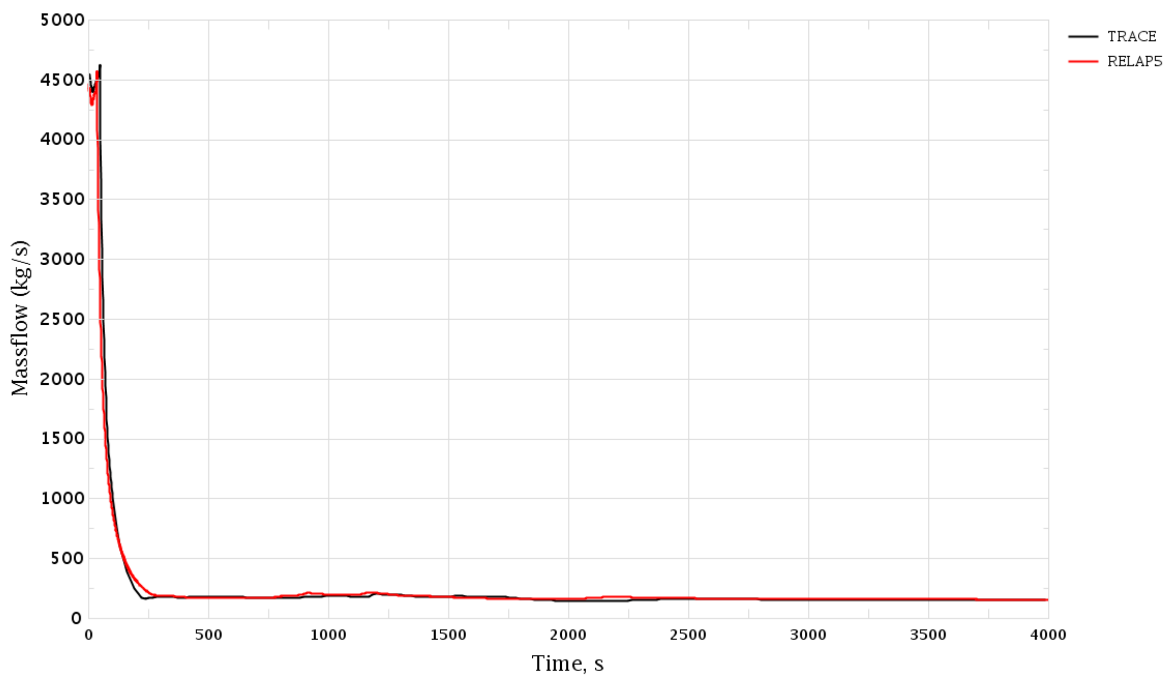

Figure 3-52 Loss of Turbine Condenser Vacuum. RCS Loop 4 Mass Flow Rate

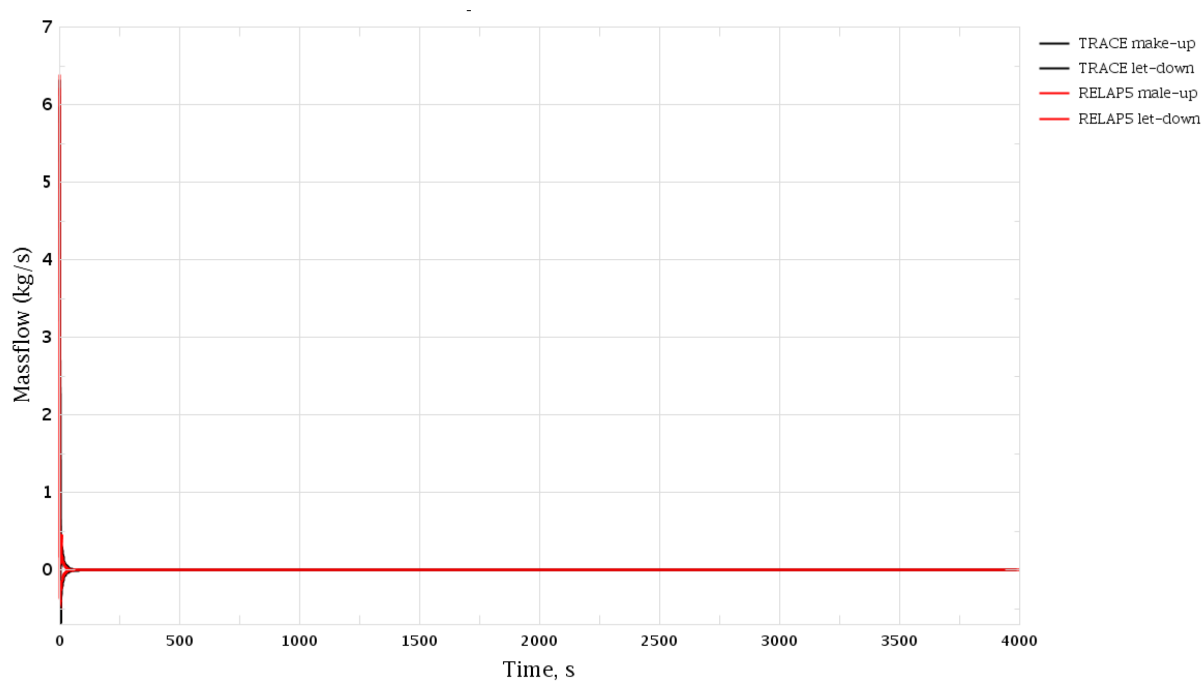

Figure 3-53 Loss of Turbine Condenser Vacuum. Make-Up and Let-Down Mass Flow

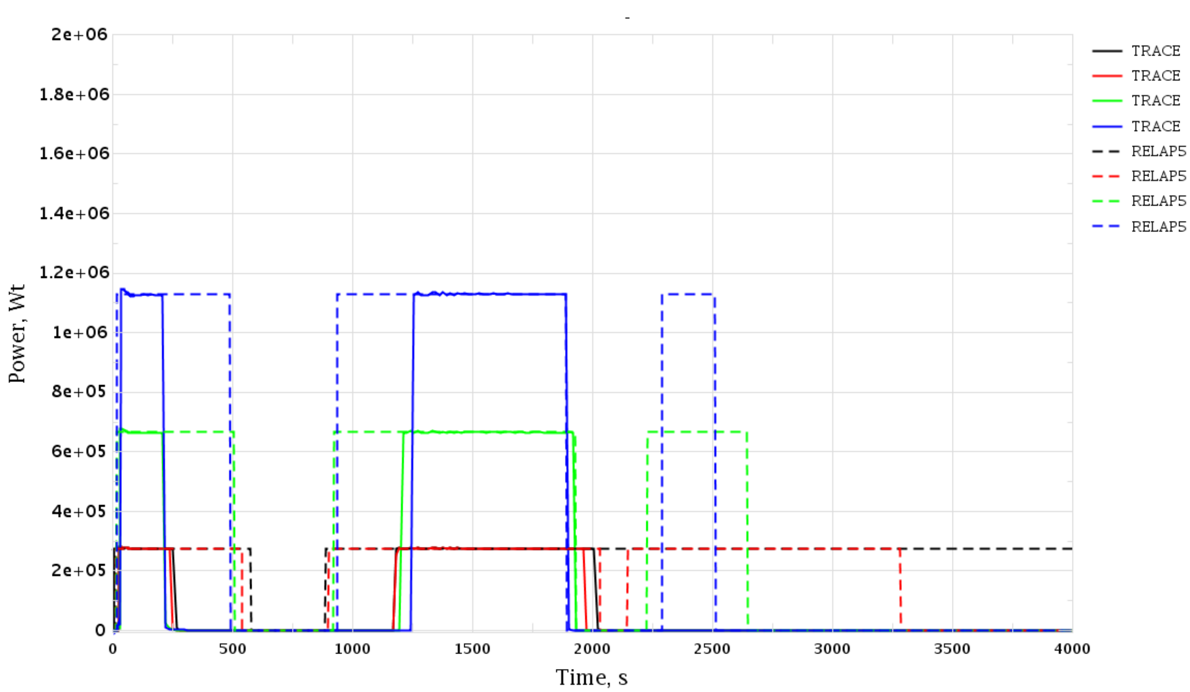

Figure 3-54 Loss of Turbine Condenser Vacuum. PRZ Heaters Power

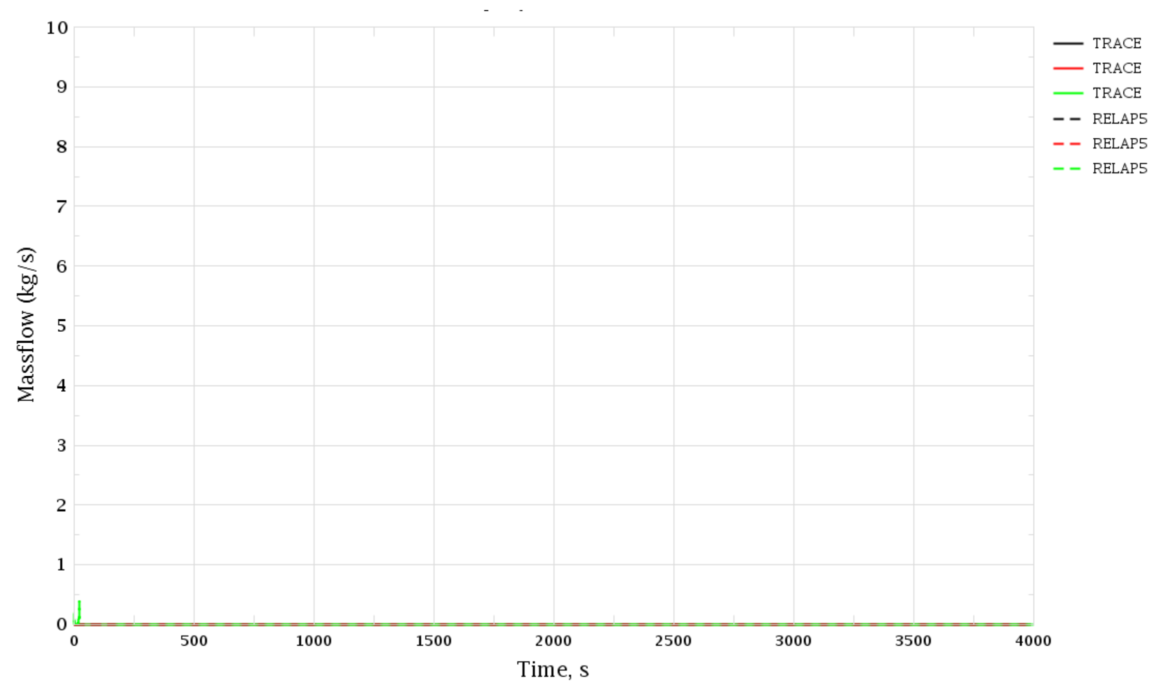

Figure 3-55 Loss of Turbine Condenser Vacuum. PRZ Spray Mass Flow Rate

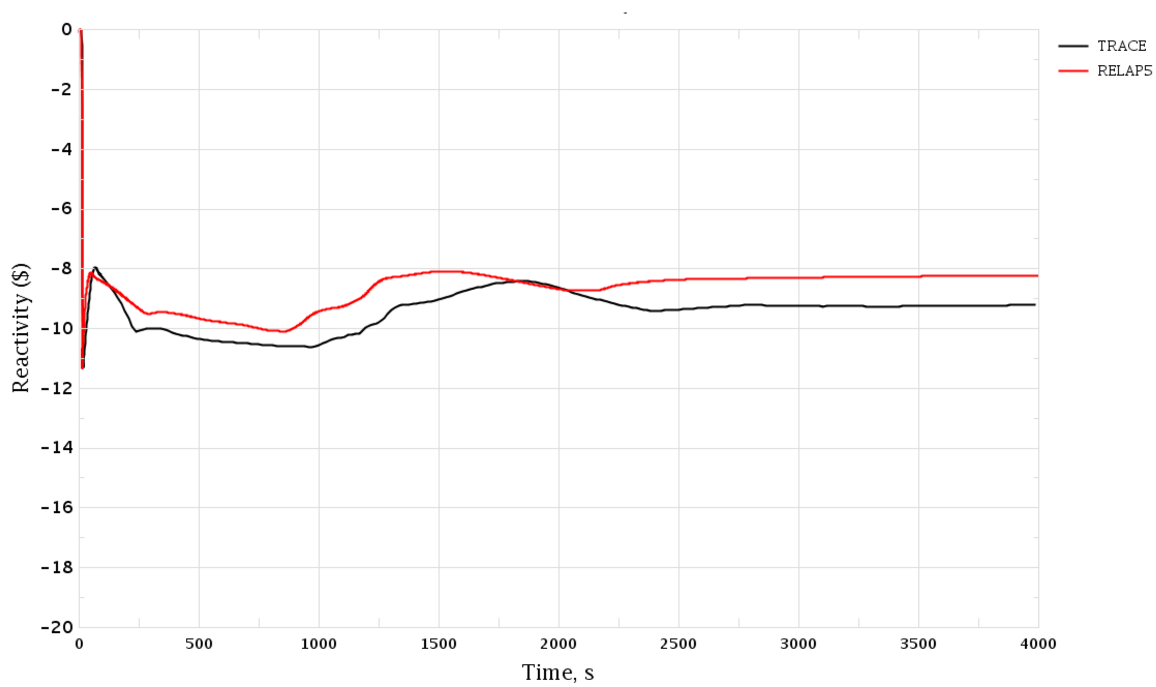

Figure 3-56 Loss of Turbine Condenser Vacuum. Core Reactivity

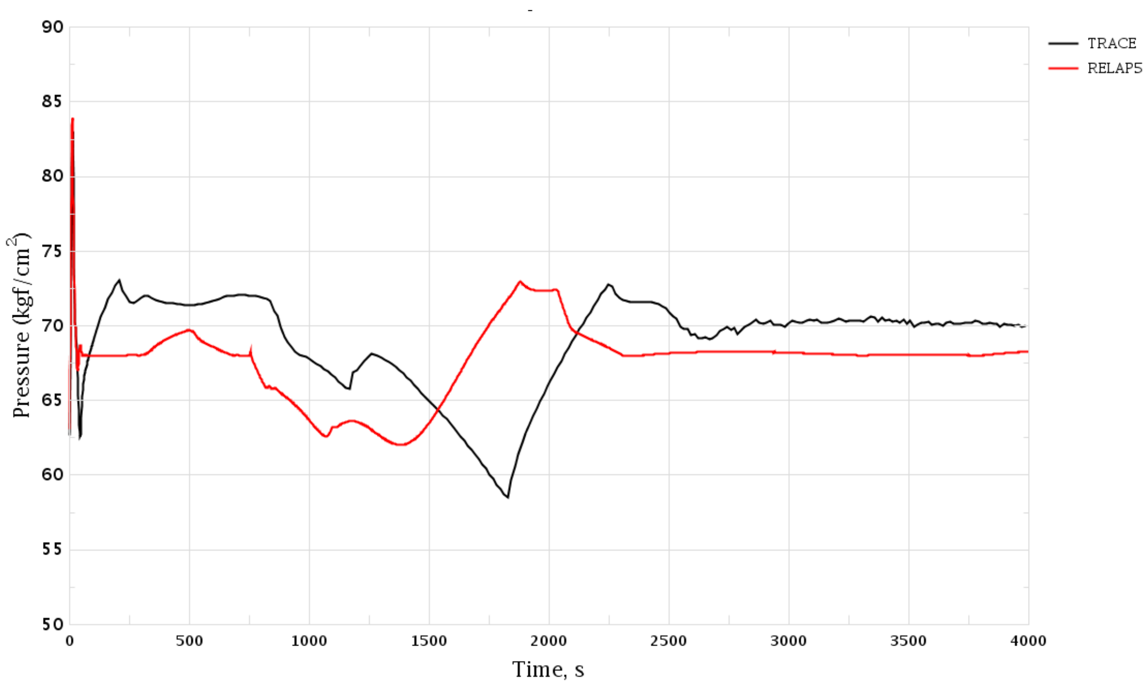

Figure 3-57 Loss of Turbine Condenser Vacuum. SG-1 Pressure

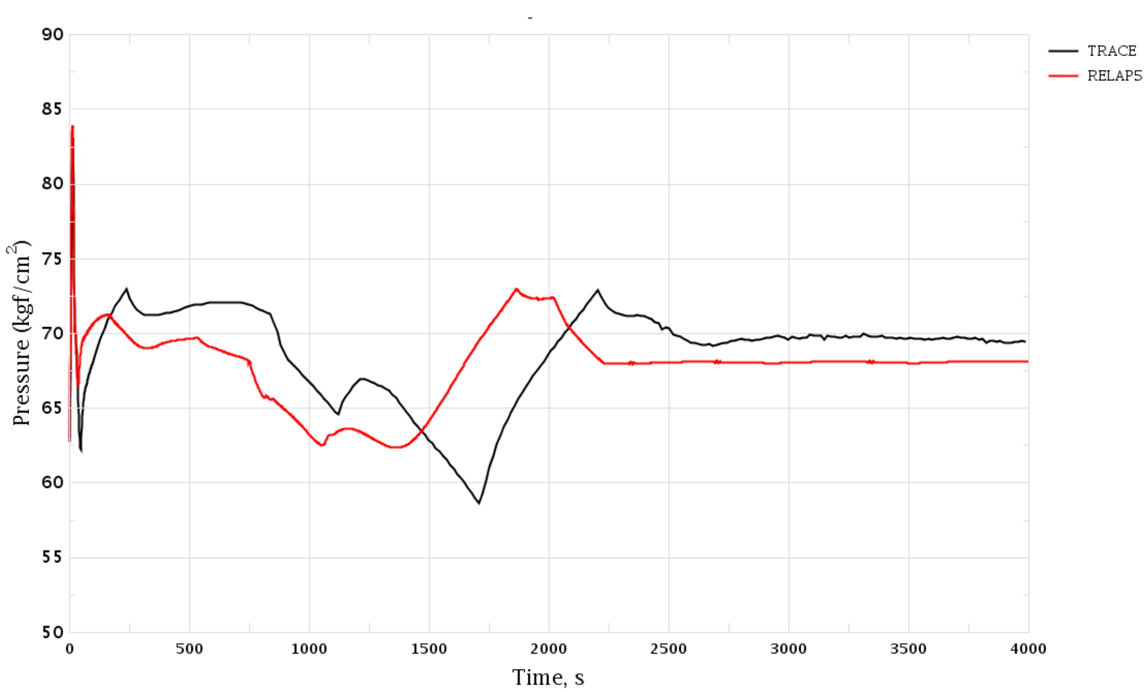

Figure 3-58 Loss of Turbine Condenser Vacuum. SG-2 Pressure

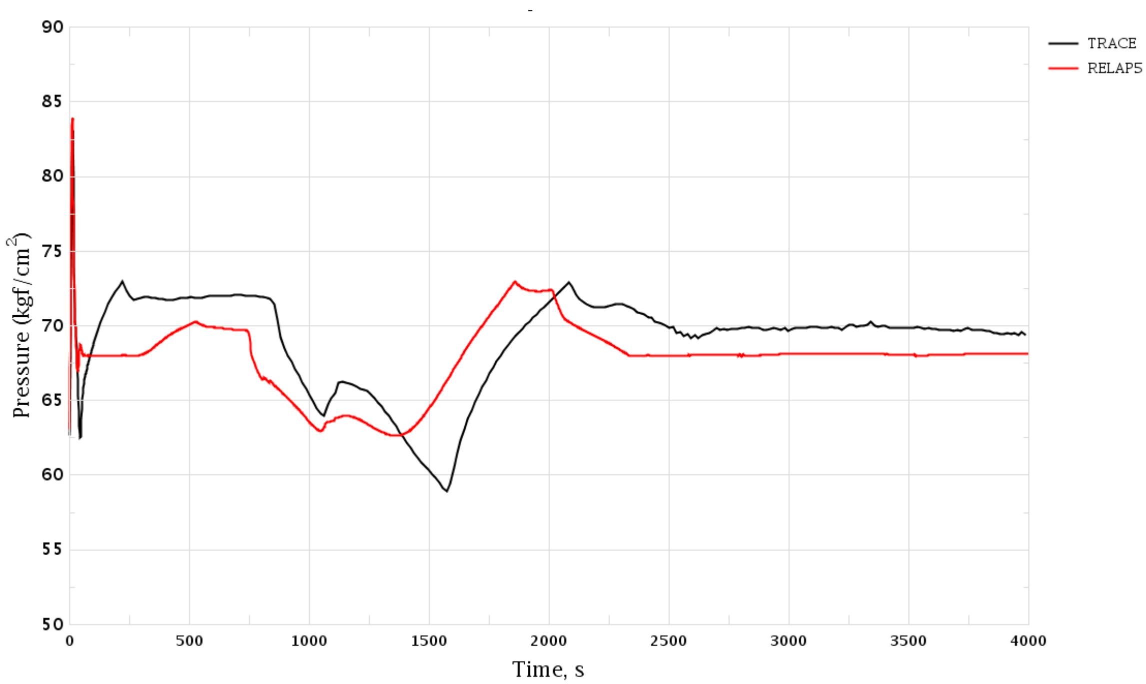

Figure 3-59 Loss of Turbine Condenser Vacuum. SG-3 Pressure

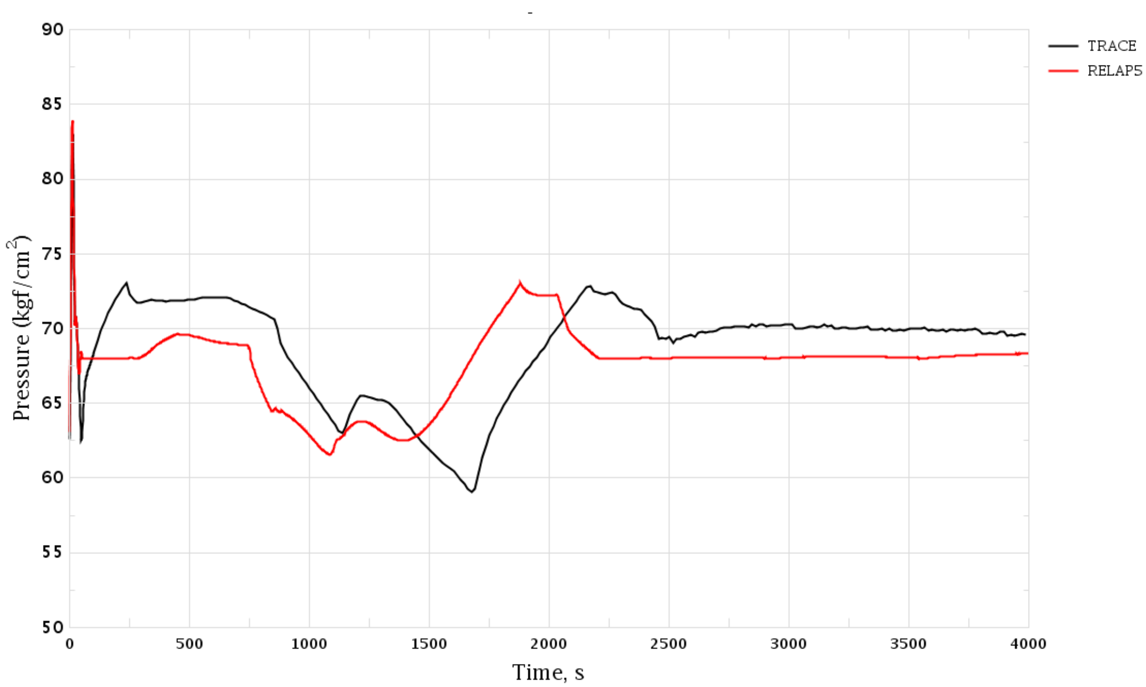

Figure 3-60 Loss of Turbine Condenser Vacuum. SG-4 Pressure

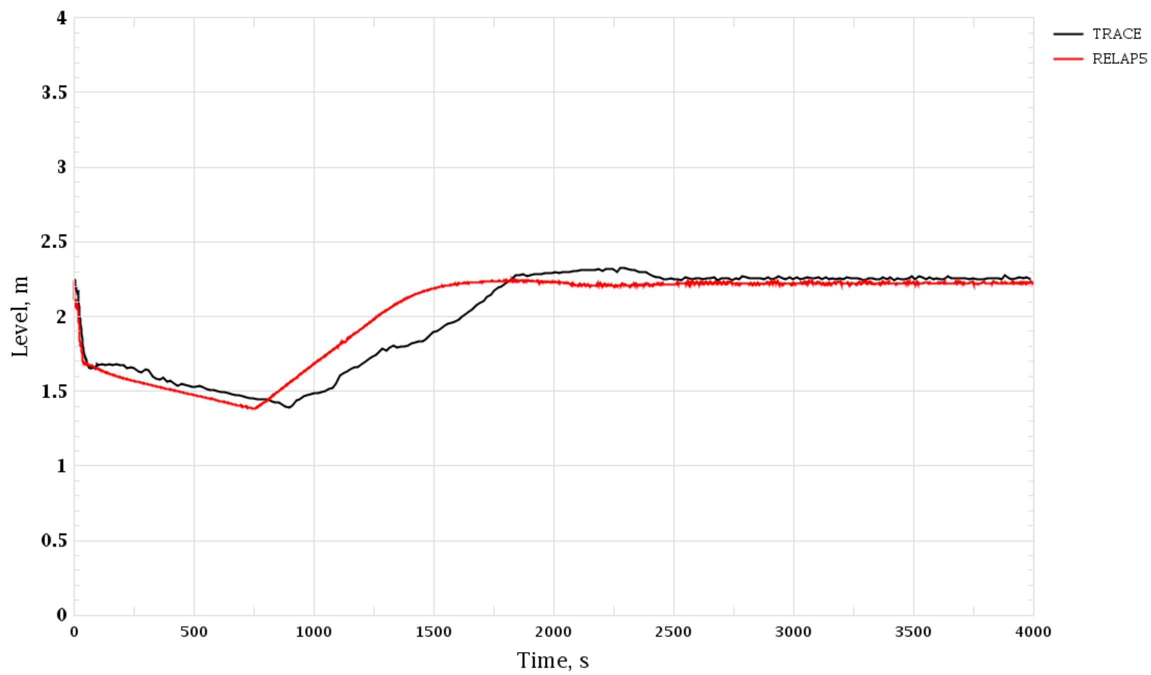

Figure 3-61 Loss of Turbine Condenser Vacuum. SG-1 Level (Wide Range)

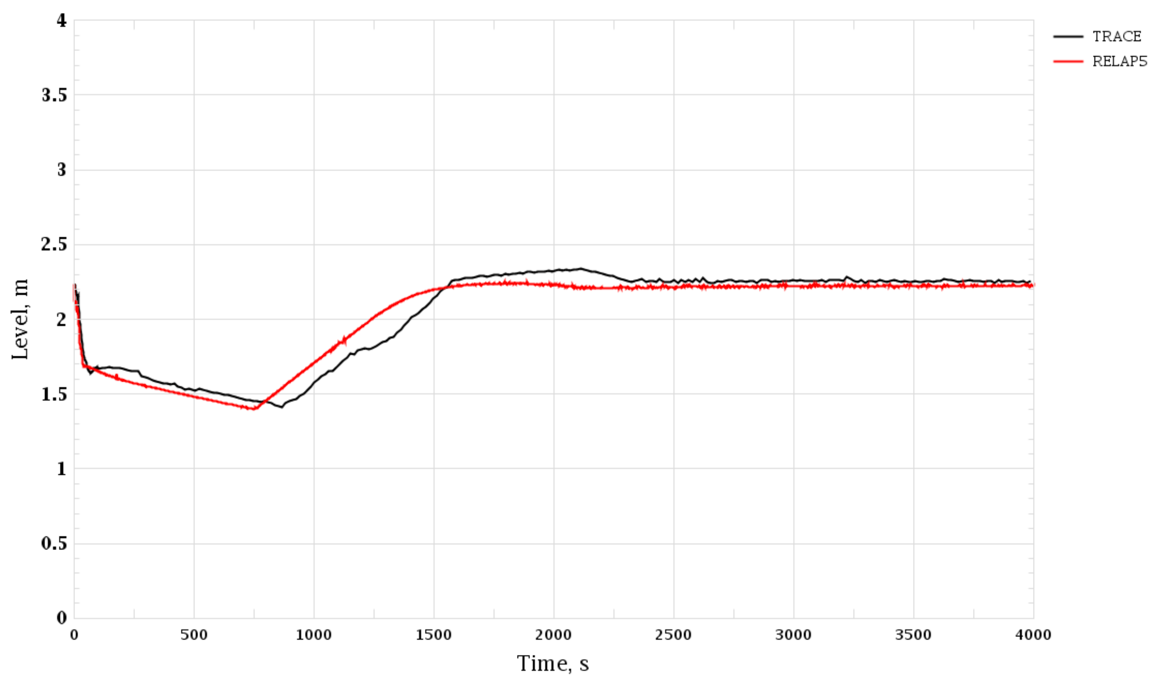

Figure 3-62 Loss of Turbine Condenser Vacuum. SG-2 Level (Wide Range)

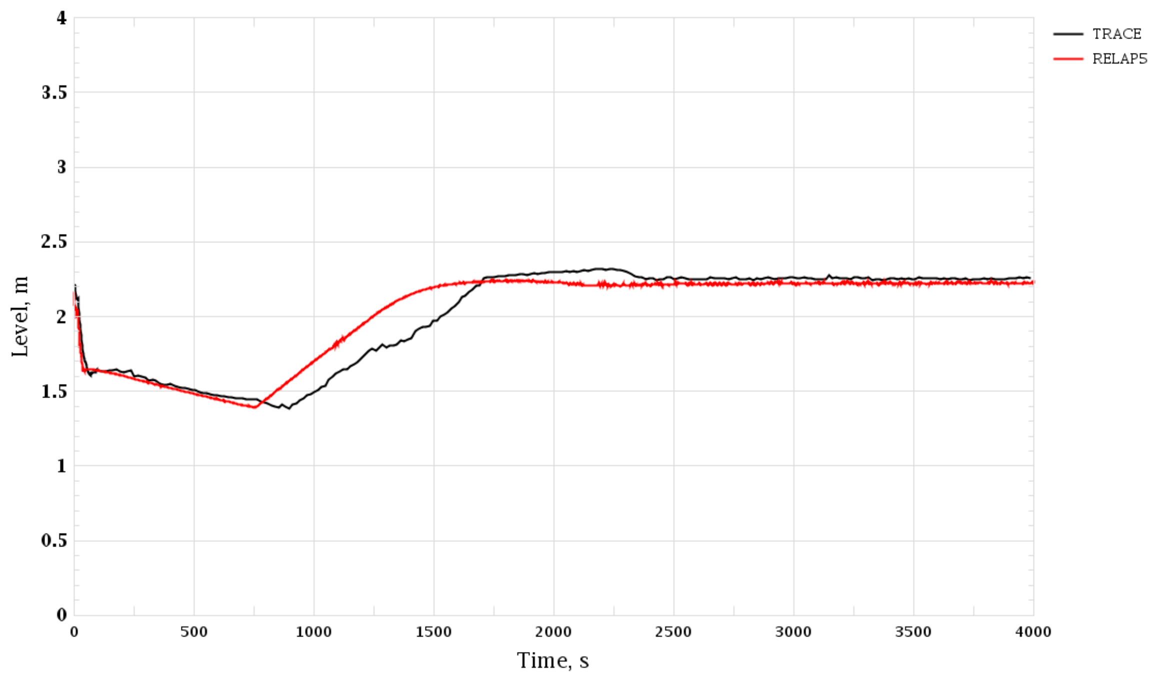

Figure 3-63 Loss of Turbine Condenser Vacuum. SG-3 Level (Wide Range)

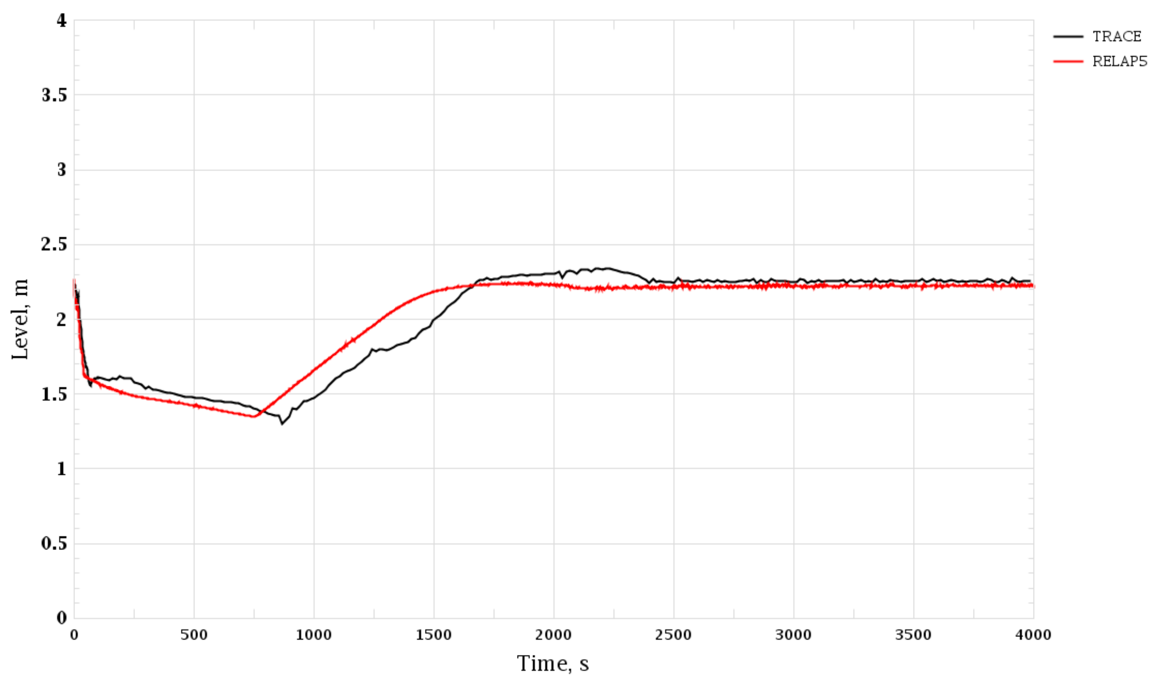

Figure 3-64 Loss of Turbine Condenser Vacuum. SG-4 Level (Wide Range)

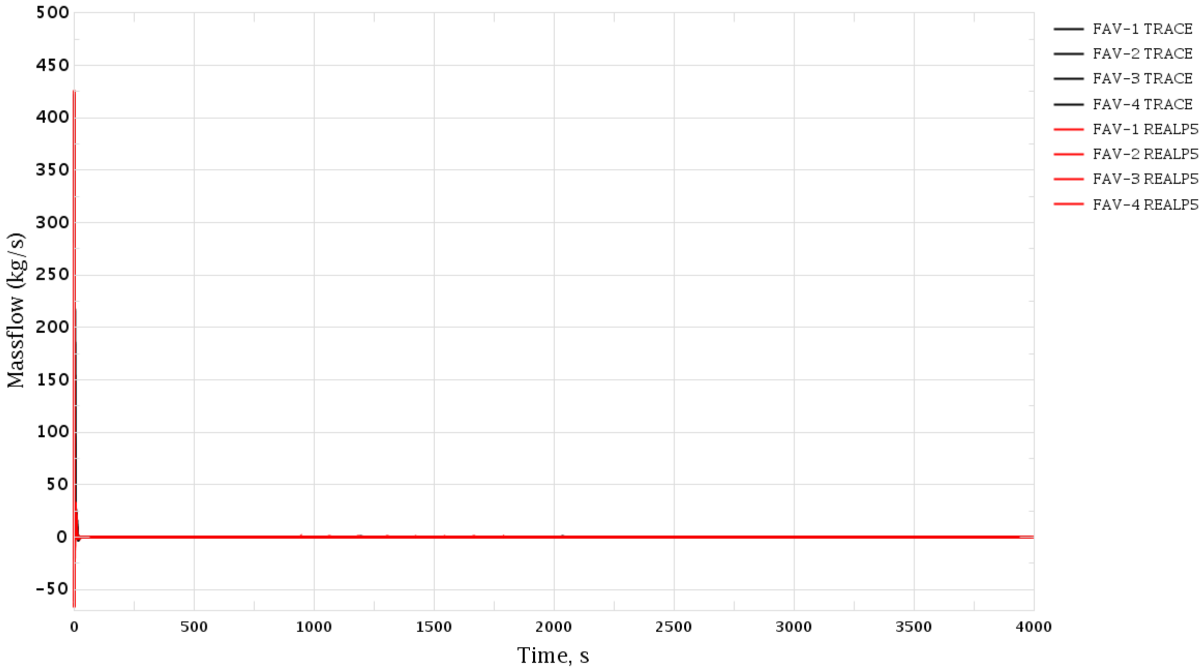

Figure 3-65 Loss of Turbine Condenser Vacuum. Mass Flow through MSIV

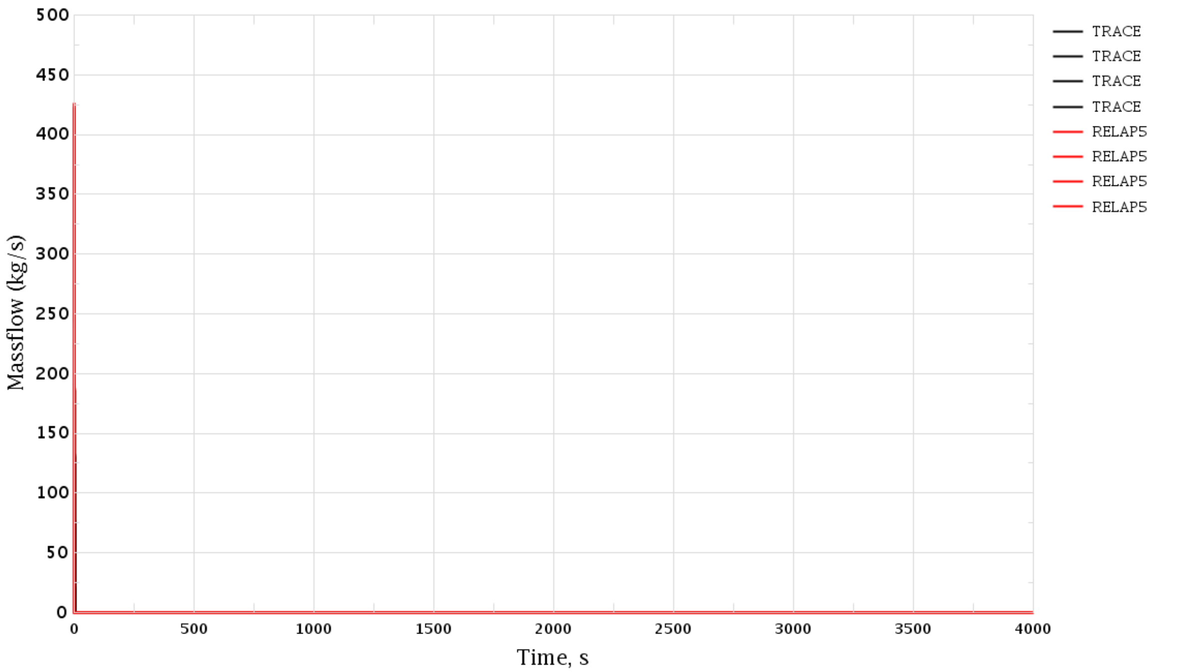

Figure 3-66 Loss of Turbine Condenser Vacuum. Turbine Mass Flow Rate

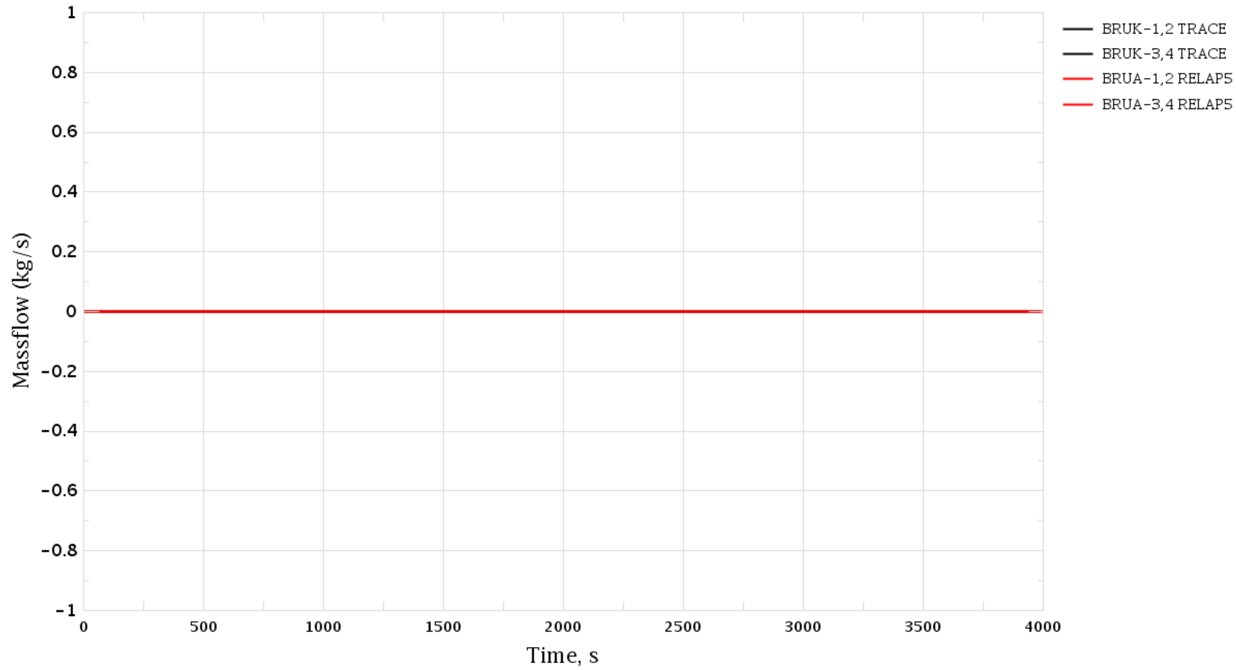

Figure 3-67 Loss of Turbine Condenser Vacuum. BRU-K Steam Mass Flow Rate

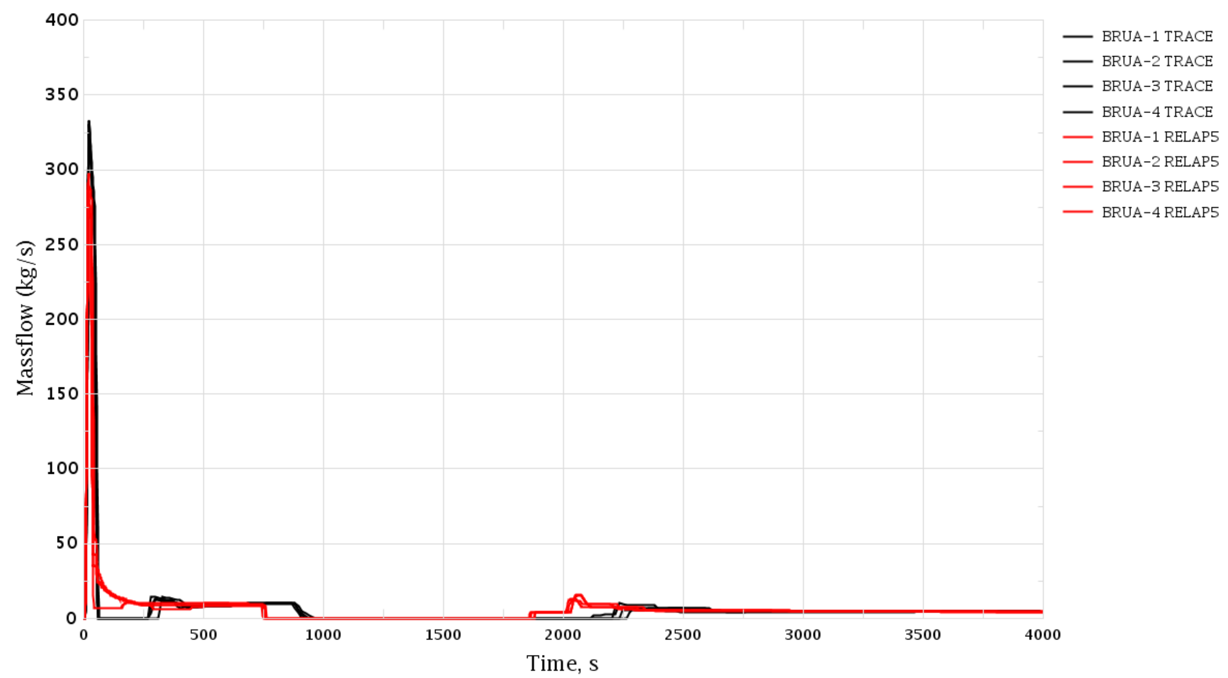

Figure 3-68 Loss of Turbine Condenser Vacuum. BRU-A Steam Mass Flow Rate

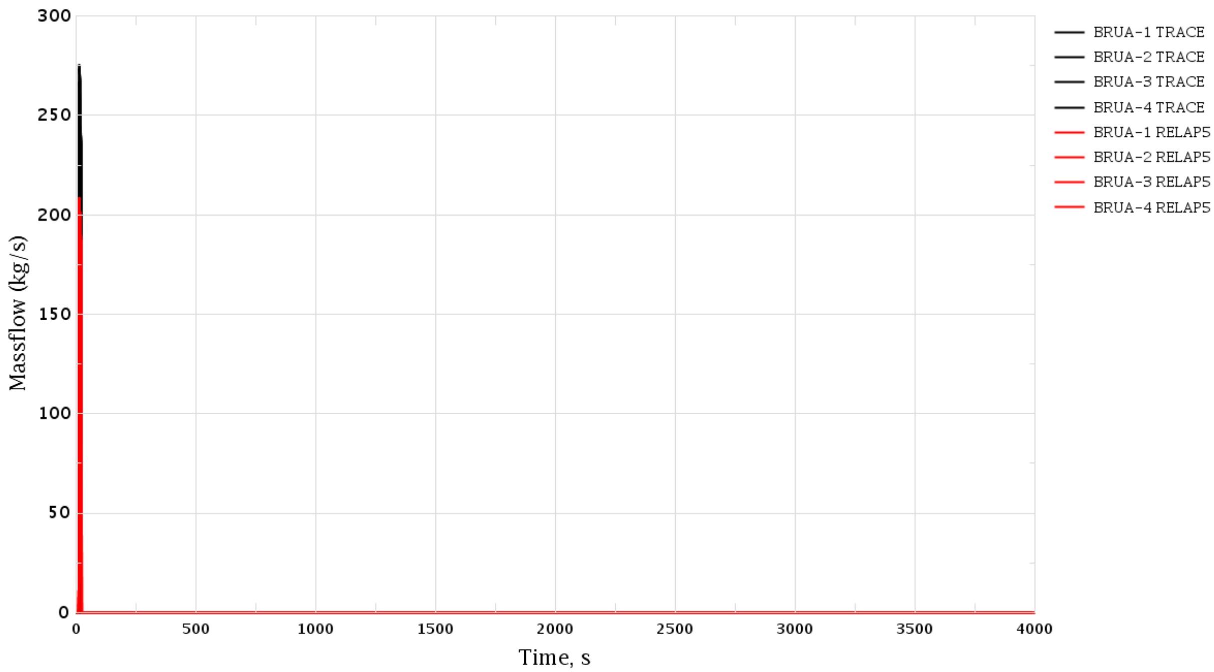

Figure 3-69 Loss of Turbine Condenser Vacuum. SG SRV Steam Mass Flow Rate

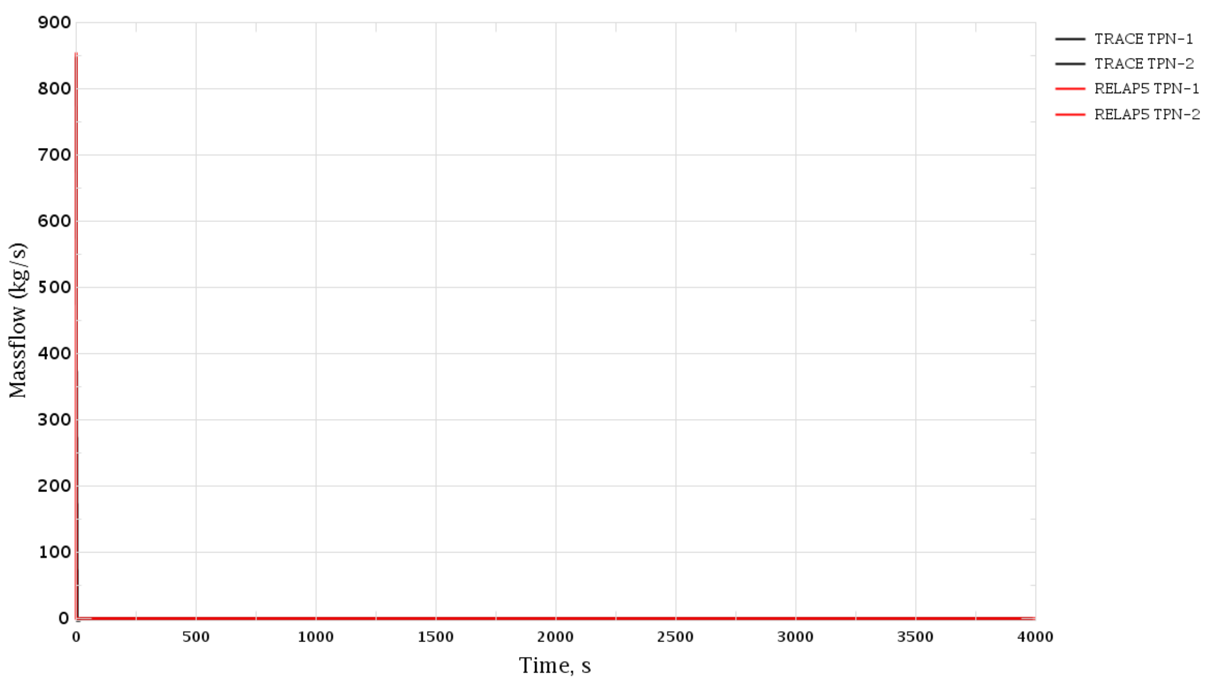

Figure 3-70 Loss of Turbine Condenser Vacuum. MFW Mass Flow Rate

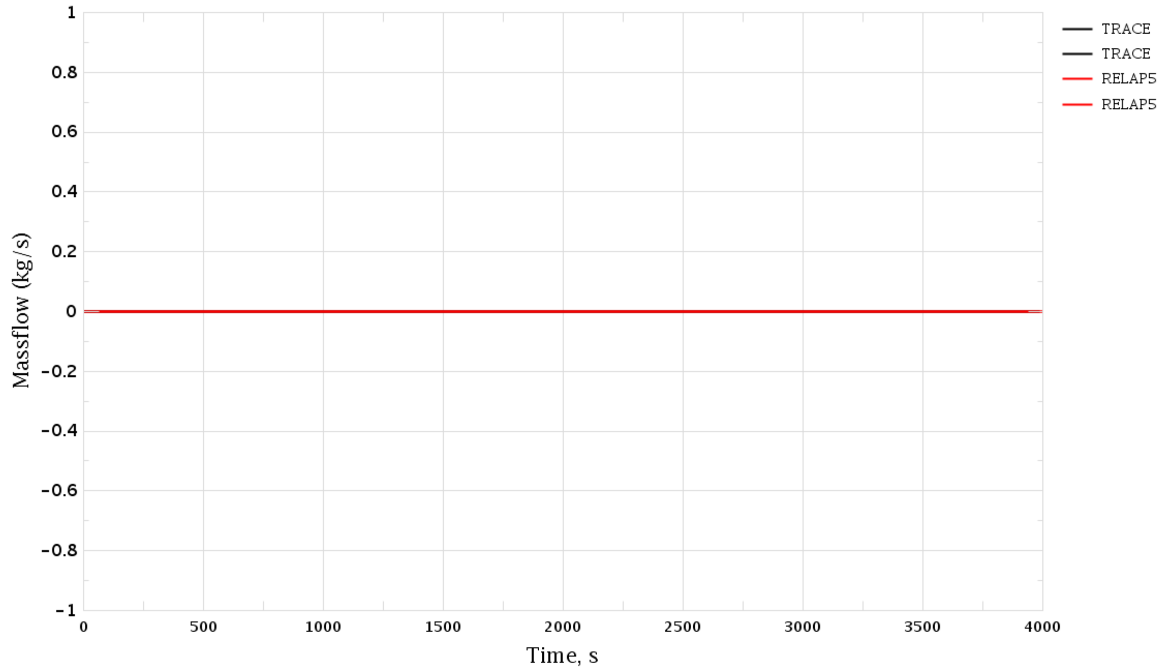

Figure 3-71 Loss of Turbine Condenser Vacuum. AFW Mass Flow Rate

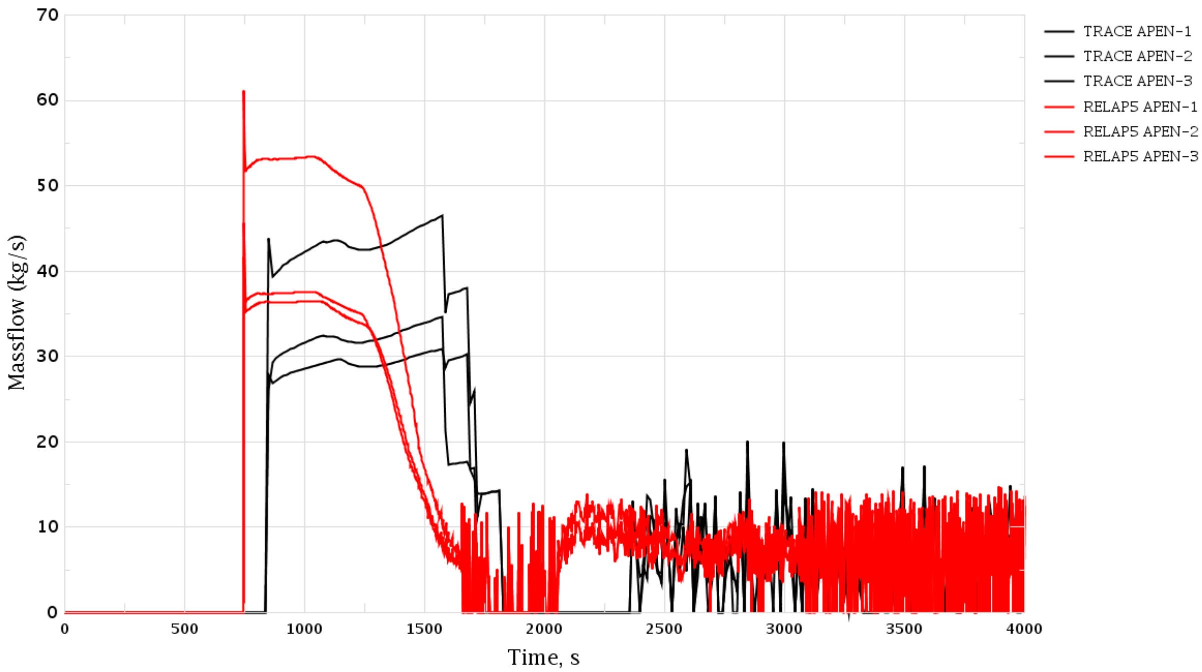

Figure 3-72 Loss of Turbine Condenser Vacuum. EFW Pumps Mass Flow Rate

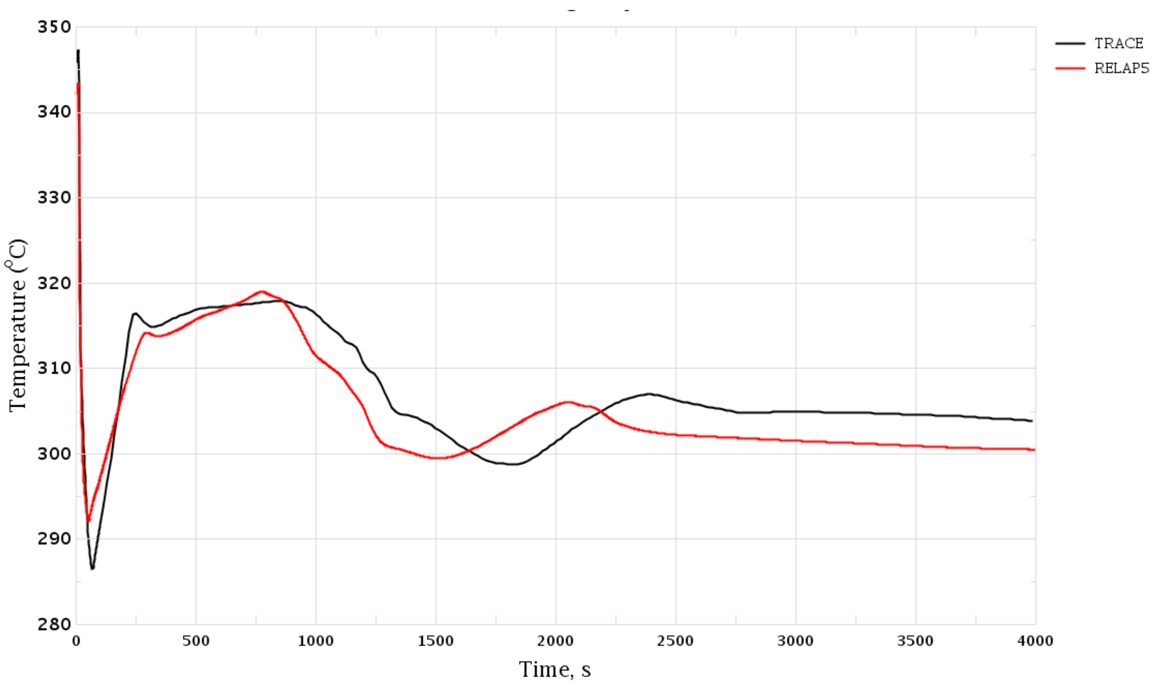

Figure 3-73 Loss of Turbine Condenser Vacuum. Maximal Cladding Temperature

3.3 Trip of One RCP

3.3.1 Brief Description of Initiating Event

This initiating event assumes trip of 1 out of 4 operating 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

Design operation of plant systems is taken into account.

RCP coast-down is performed according to the pump characteristics. Control of the reactor power is performed by ARM in neutron power maintenance mode and operation of RPL.

No operator actions are simulated in the scenario.

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 One RCP" Transient

TRACE Time, s	RELAP5 Time, s	Event	Description
0.0	0.0	Trip of RCP-1	IE occurrence
0.0–89.0	0.0–90.0	Formation of RPL signal to decrease reactor power until neutron power is lower than new power setpoint of 67% + 2%.	Due to trip of one out 4 RCPs
42.0	41.5	Flow reversal in RCS loop no.1	Due to stoppage of RCP-1
95.0	100.0	Start of ARM operation to maintain reactor power at 67%	
110.0	125.0	Minimal RCS pressure is155.7 kgf/cm^2 (TRACE) and 154.9 kgf/cm^2 (RELAP5)	
120.0	135.0	Minimal PRZ level is 7.95/7.9 m	
1000.0	1000.0	End of calculation	

Trip of one out of 4 operating RCPs results in a decrease of coolant flow through the reactor core and initial increase of RCS coolant temperature (Figure 3-77 – Figure 3-84). The allowed power setpoint is automatically changed by RPL from 102% to 69% and reactor power is decreased for 2% below the allowed value (Figure 3-74) by insertion of control rods to the reactor core with normal operation speed.

Due to decrease of coolant flow in loop no.1 (Figure 3-85) and decrease of energy transfer to the secondary circuit, SG-1 pressure decreases for the first 50 s of the transient (Figure 3-93). Until steam flow to the turbine is decreased (Figure 3-106) by partial closure of turbine's control valves an excessive steam flow (compared to new reactor power level) to the turbine causes decrease of RCS temperature (Figure 3-77 – Figure 3-84), RCS pressure (Figure 3-75) and PRZ level (Figure 3-76). The latter in turn leads to opening of make-up control valves (Figure 3-89) and switching on of the PRZ heaters groups (Figure 3-90) to restore PRZ level and primary circuit pressure.

Decrease of steam flow to the turbine results in a decrease of MFW flow (Figure 3-107). SG level (Figure 3-101 – Figure 3-104) is maintained at the nominal values by operation of MFW control valves.

The calculation is ended at 1000 s of transient close to the stabilization of all primary and secondary circuit parameters. At the end of calculation the heat generated in the reactor core at the reduced power level (~67%) is removed in forced circulation mode (with 3 RCPs in operation) by the secondary circuit with providing the steam produced in SGs to the turbine and SG feed by MFW pumps. In RCS loop no.1 the backward flow of coolant is established.

The plots of the main parameters of calculation are presented below on Figure 3-74 – Figure 3-109.

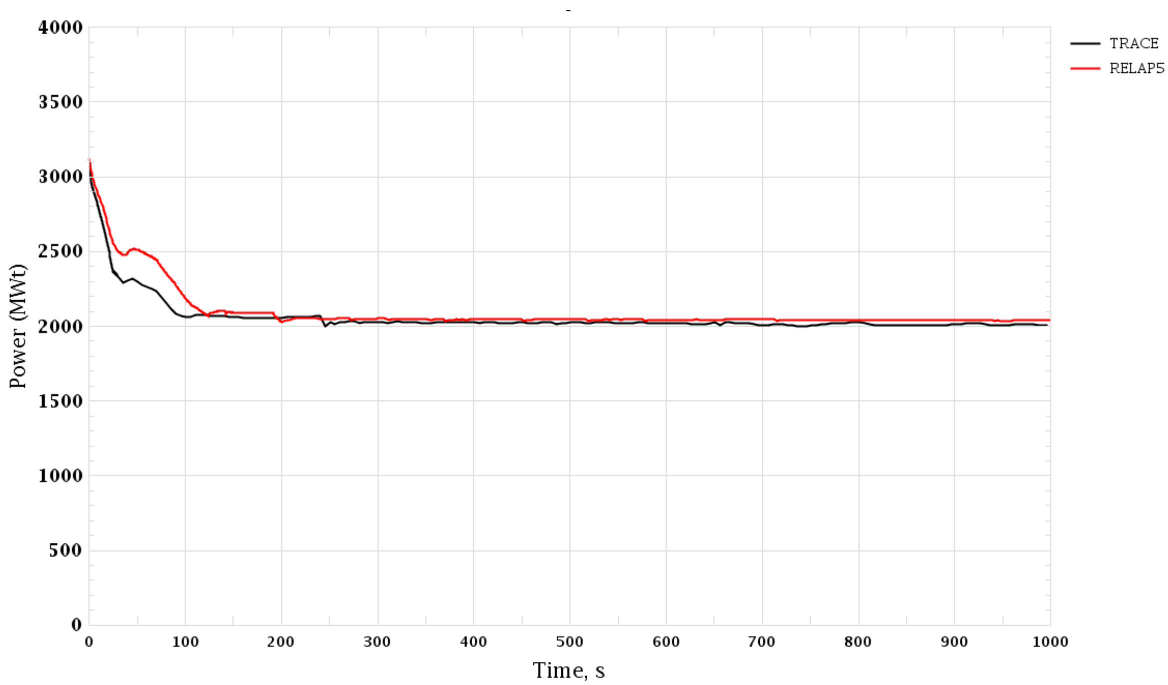

Figure 3-74 Trip of One RCP. Core Thermal Power

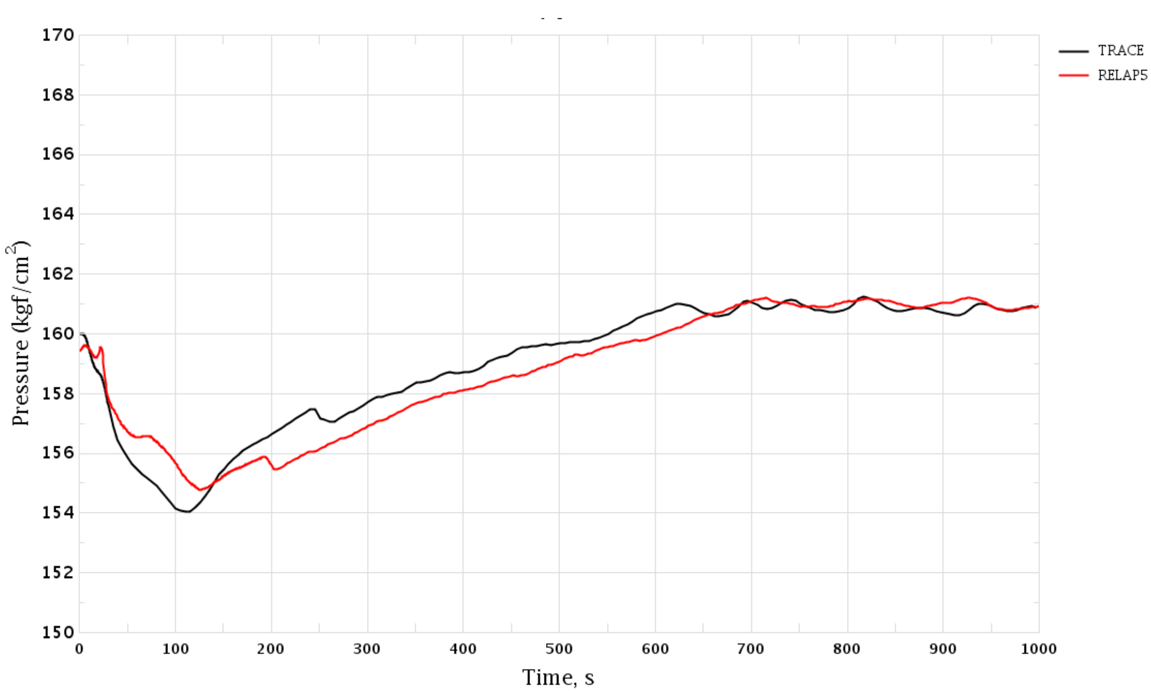

Figure 3-75 Trip of One RCP. RCS Pressure

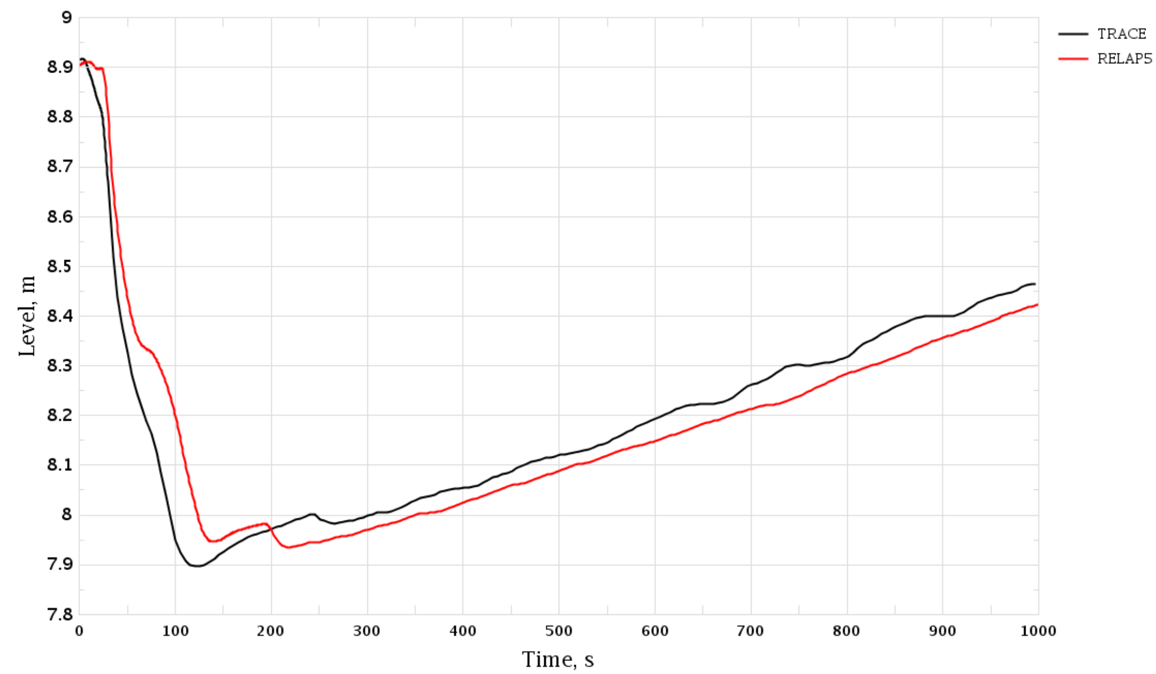

Figure 3-76 Trip of One RCP. Pressurizer Level

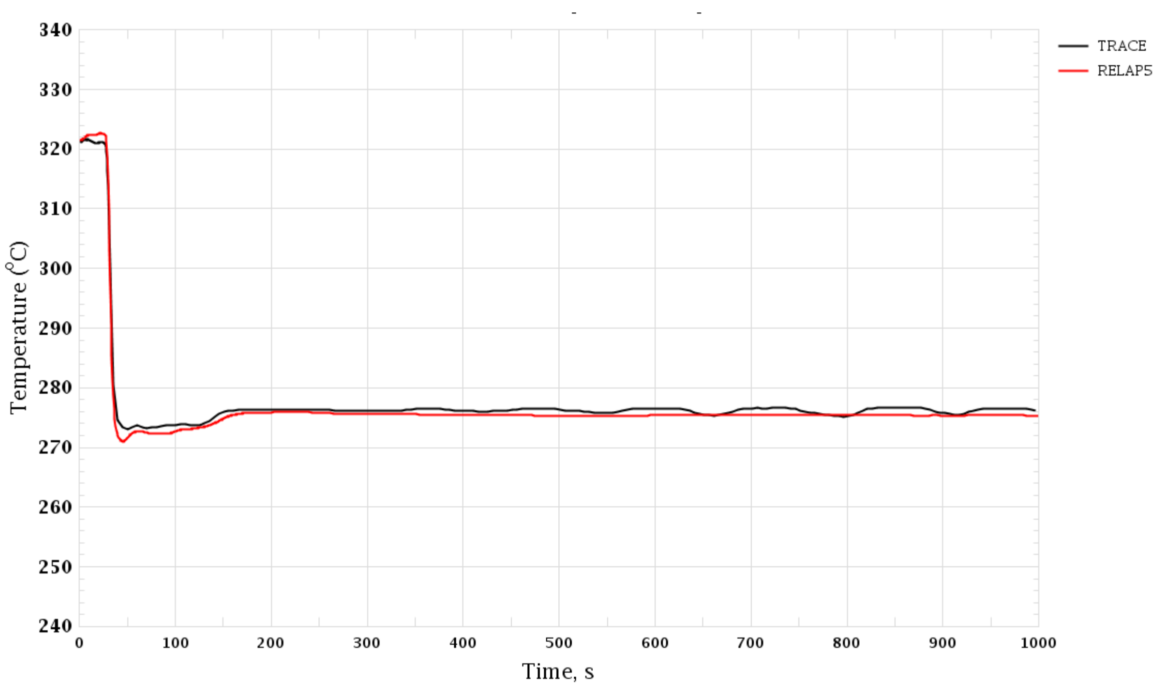

Figure 3-77 Trip of One RCP. Coolant Temperature in Hot Leg, Loop 1

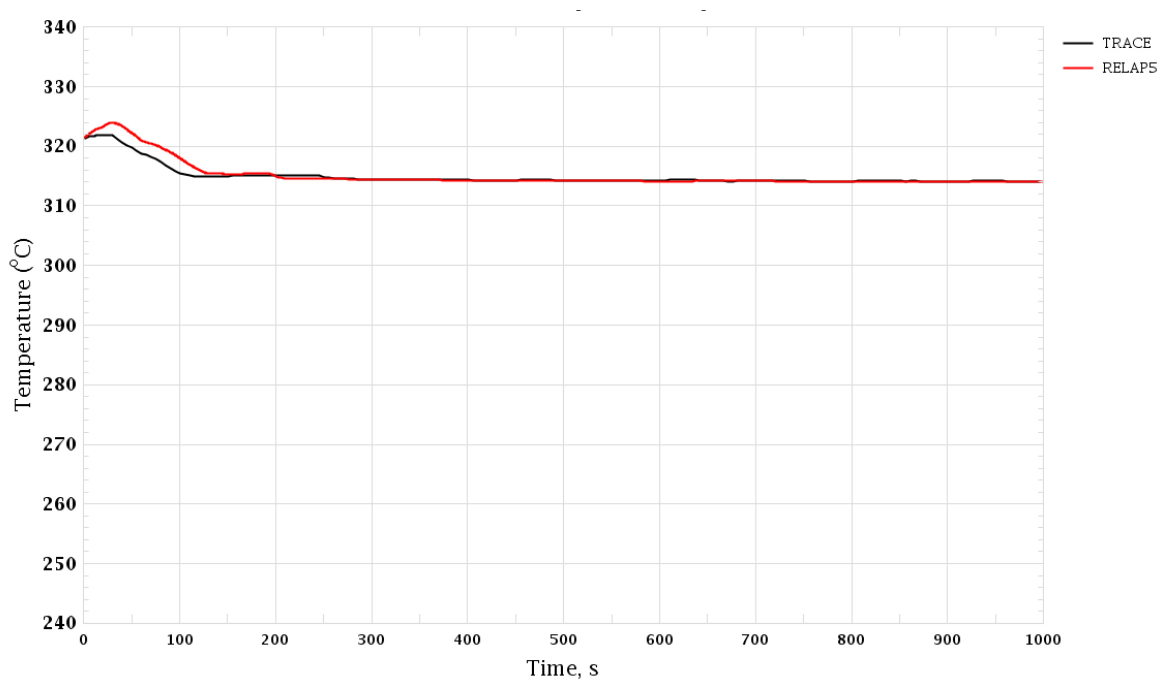

Figure 3-78 Trip of One RCP. Coolant Temperature in Hot Leg, Loop 2

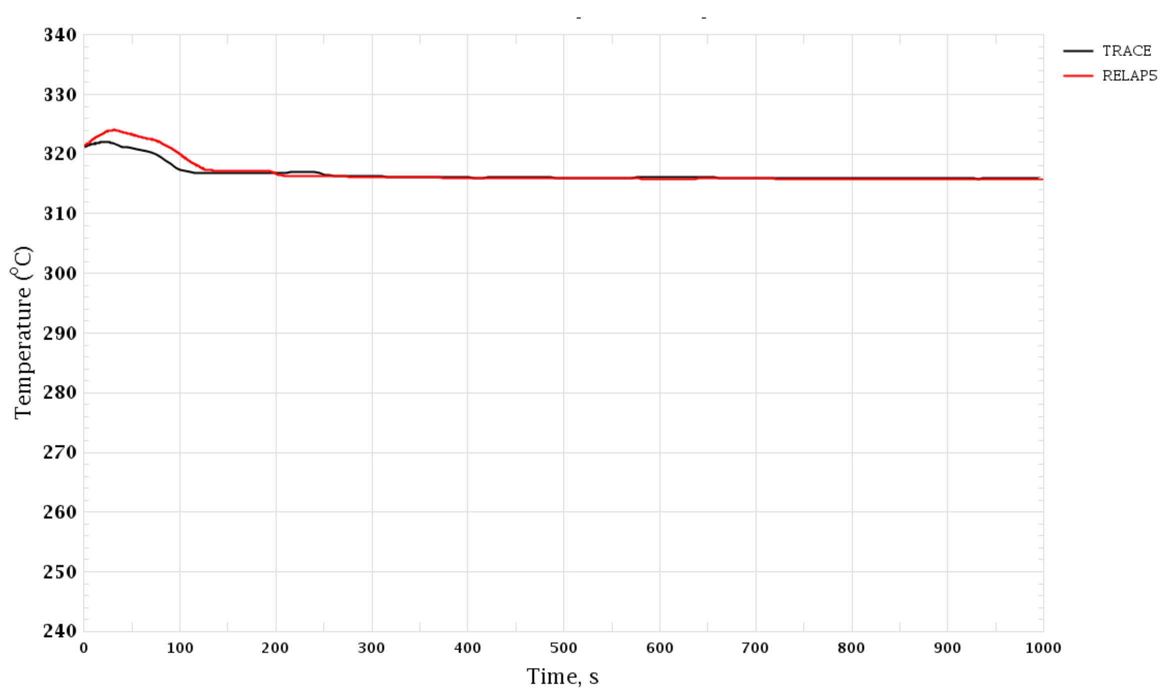

Figure 3-79 Trip of One RCP. Coolant Temperature in Hot Leg, Loop 3

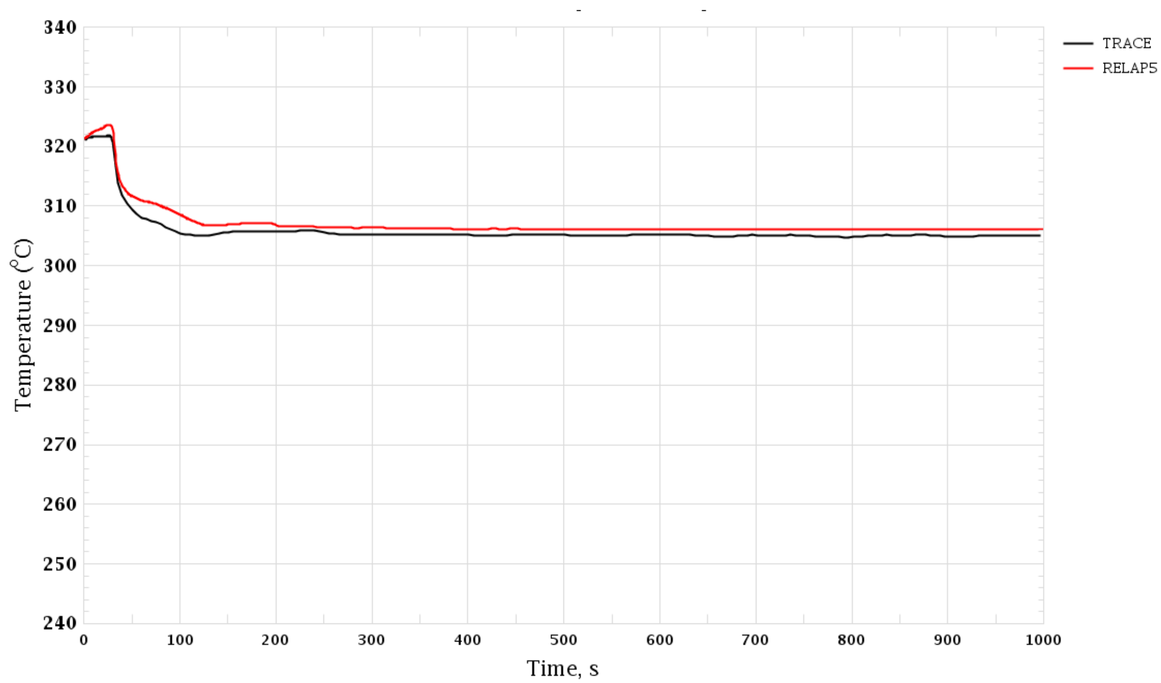

Figure 3-80 Trip of One RCP. Coolant Temperature in Hot Leg, Loop 4

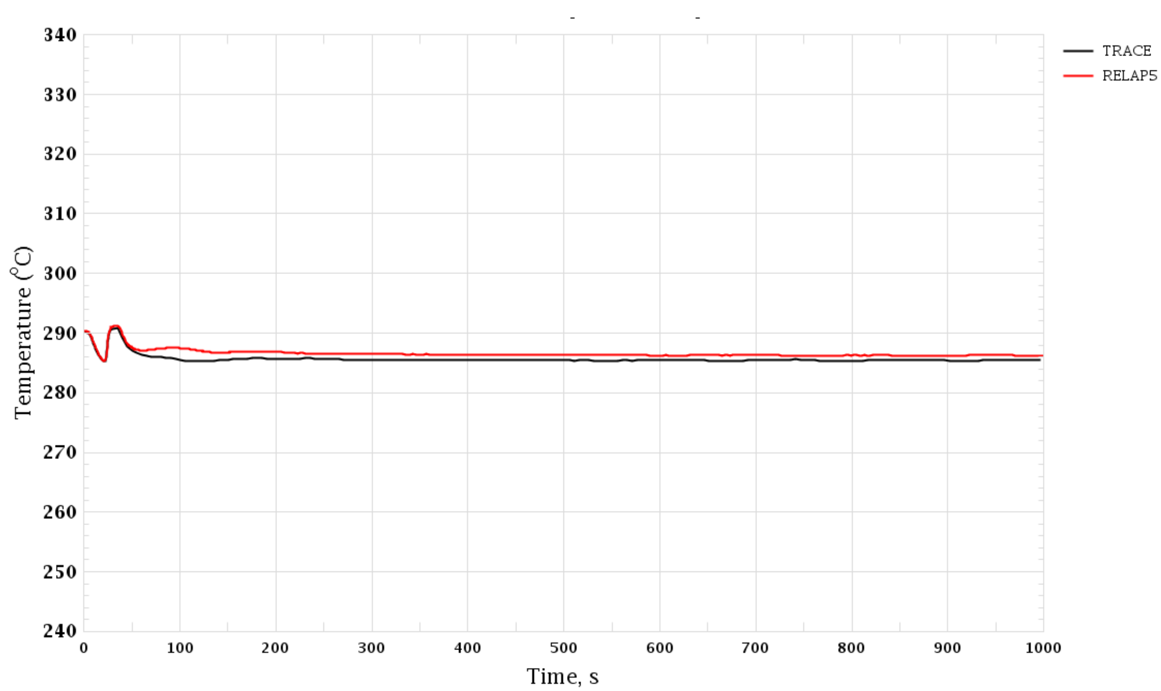

Figure 3-81 Trip of One RCP. Coolant Temperature in Cold Leg, Loop 1

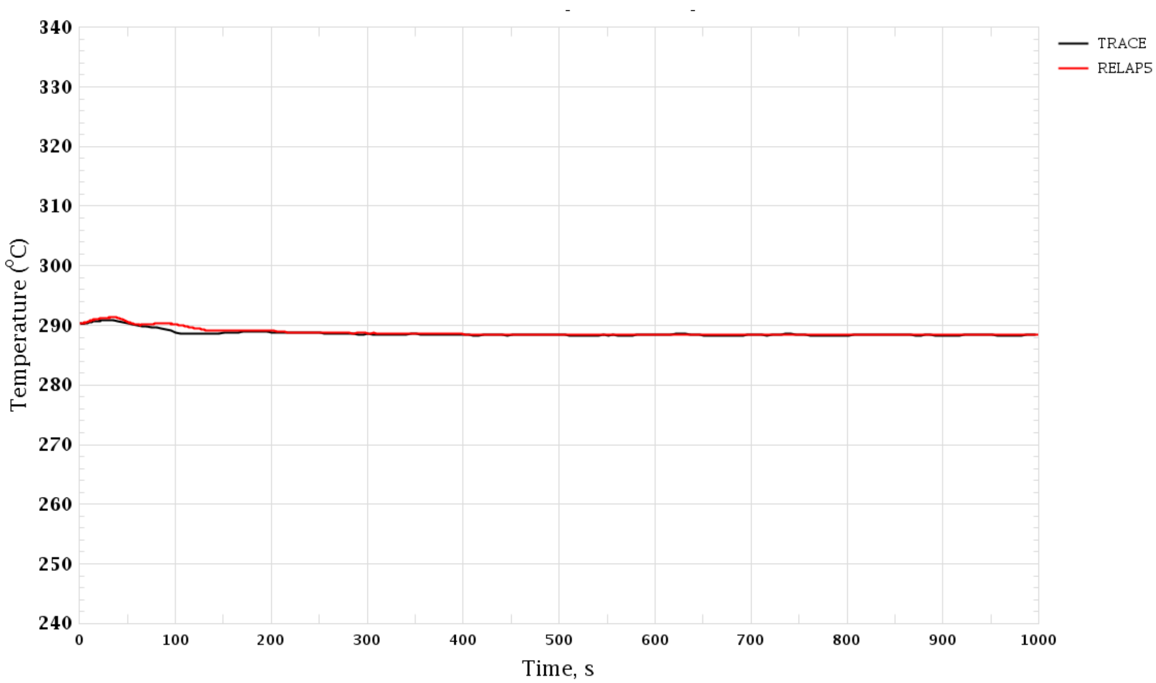

Figure 3-82 Trip of One RCP. Coolant Temperature in Cold Leg, Loop 2

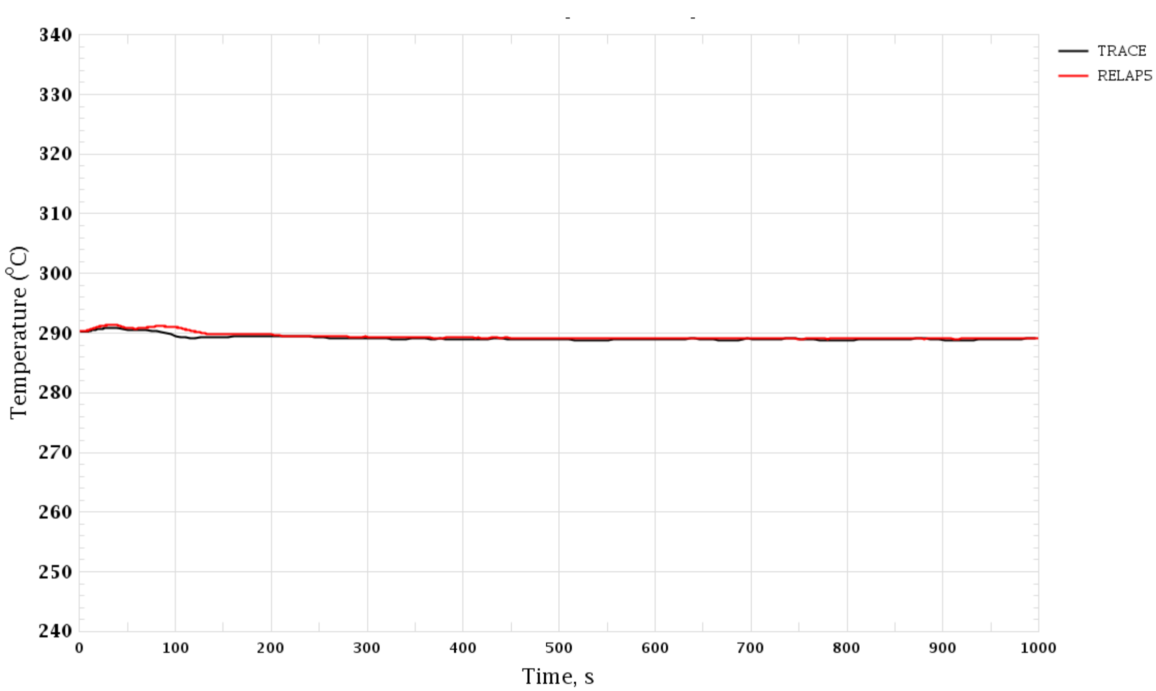

Figure 3-83 Trip of One RCP. Coolant Temperature in Cold Leg, Loop 3

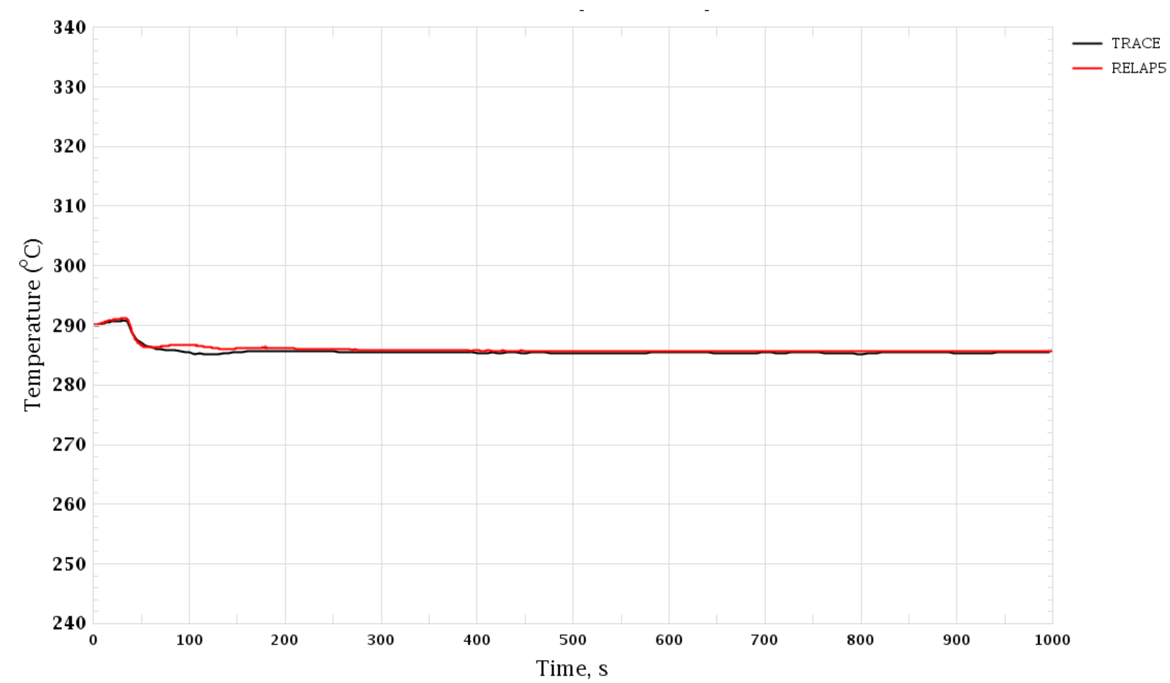

Figure 3-84 Trip of One RCP. Coolant Temperature in Cold Leg, Loop 4

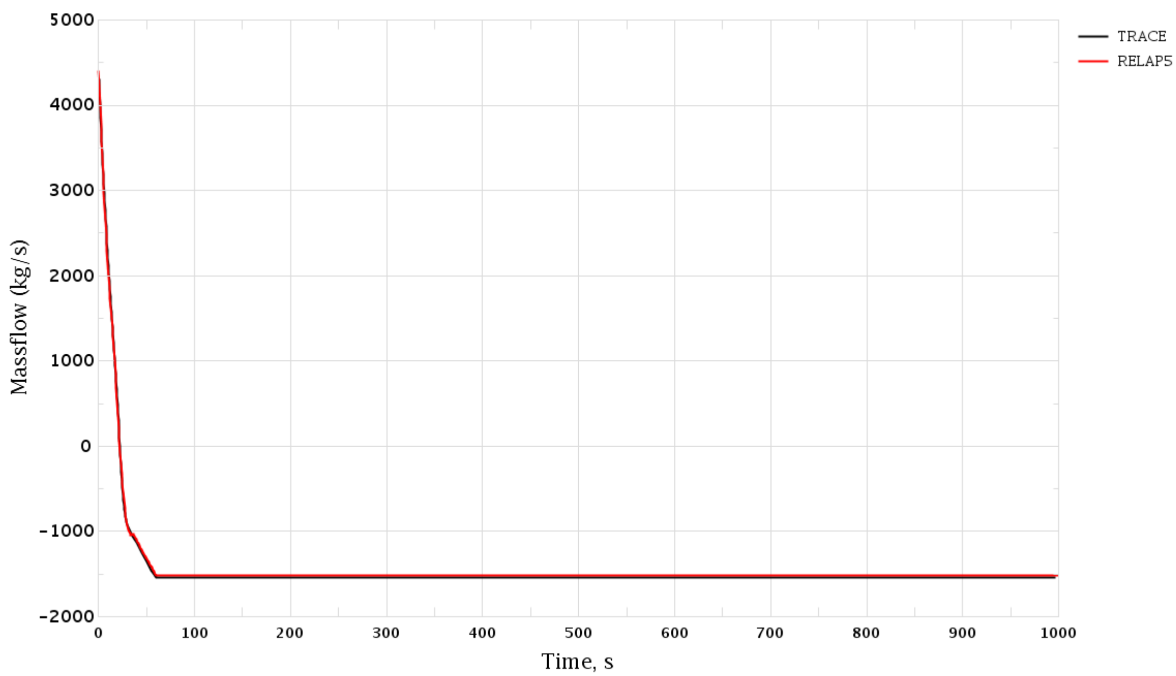

Figure 3-85 Trip of One RCP. RCS Loop 1 Mass Flow Rate

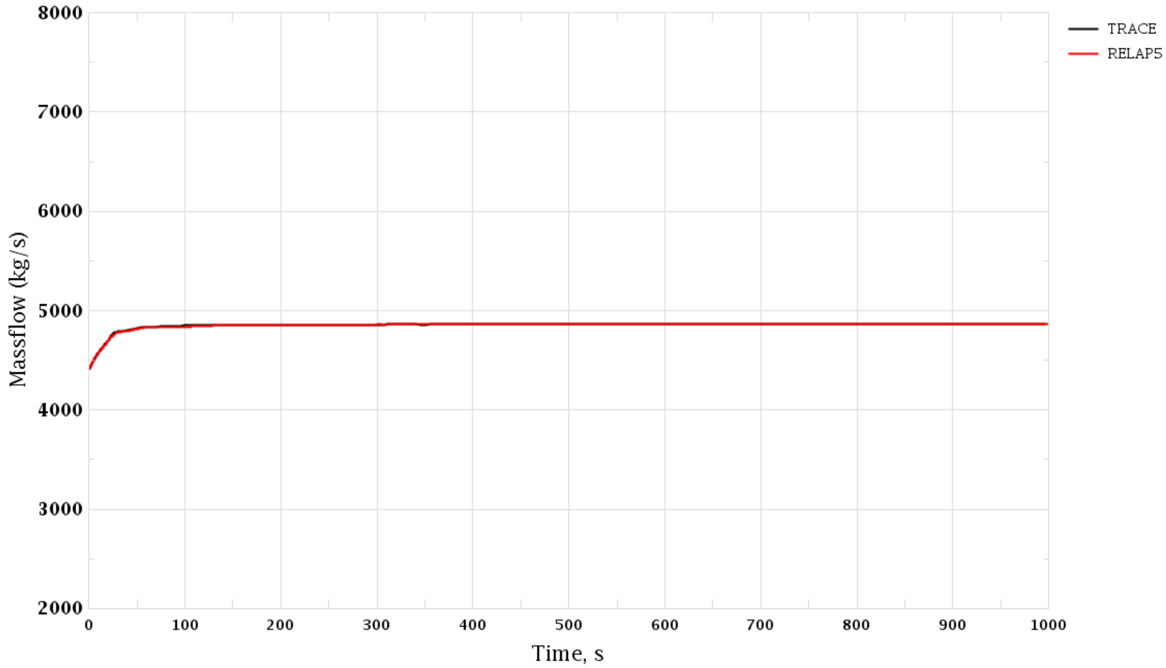

Figure 3-86 Trip of One RCP. RCS Loop 2 Mass Flow Rate

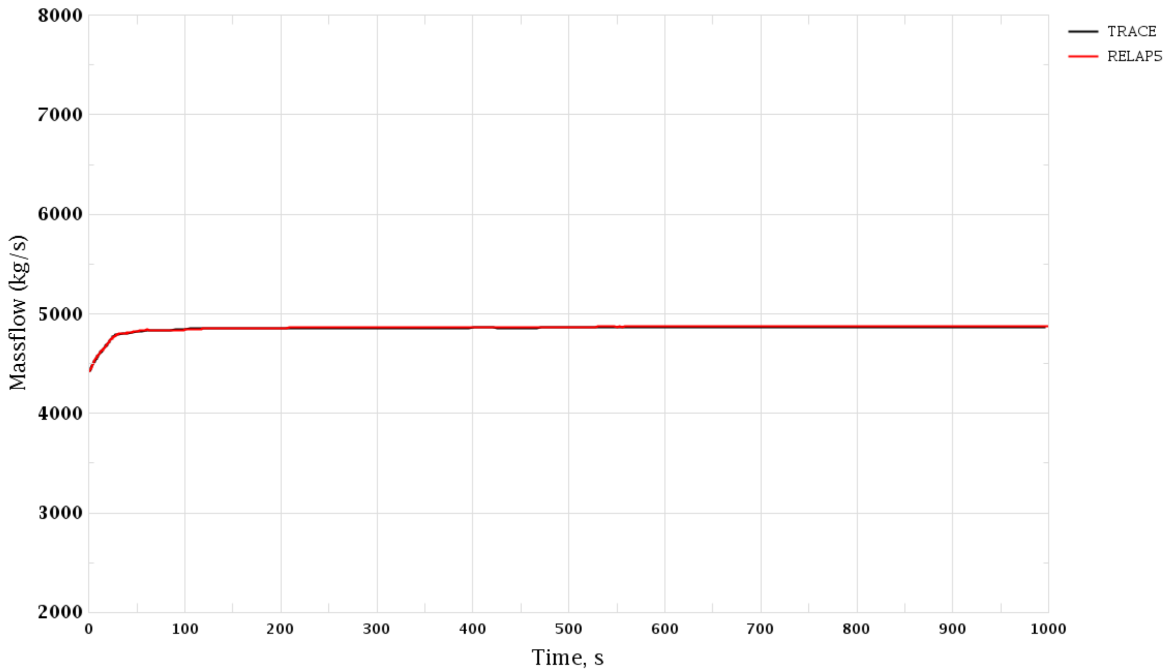

Figure 3-87 Trip of One RCP. RCS Loop 3 Mass Flow Rate

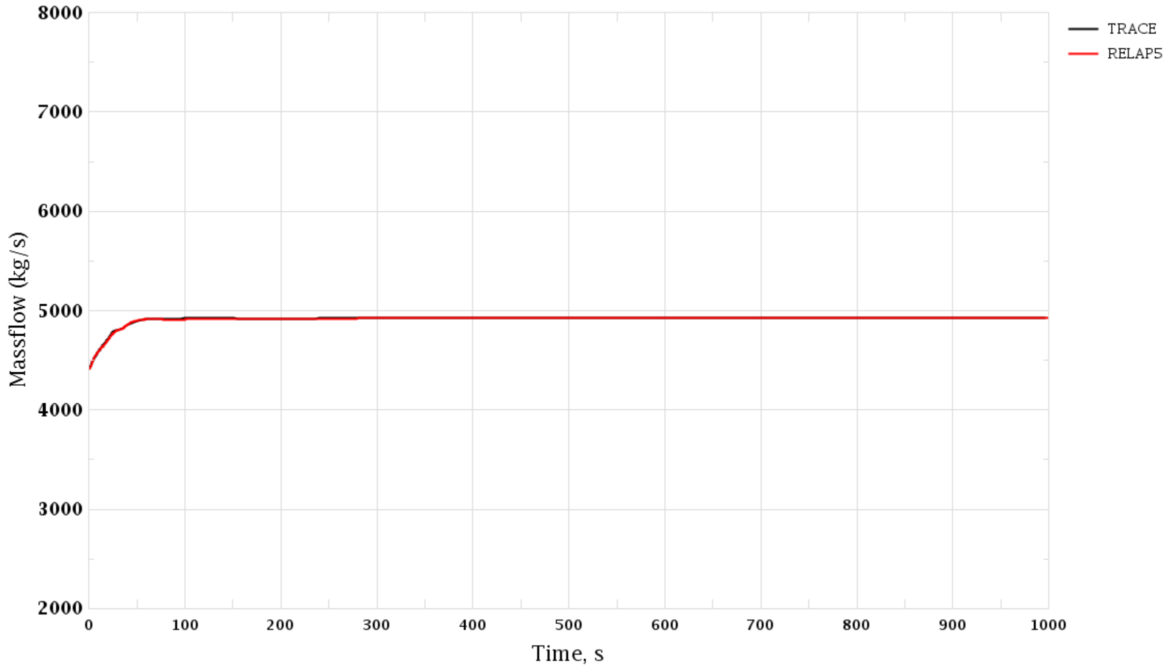

Figure 3-88 Trip of One RCP. RCS Loop 4 Mass Flow Rate

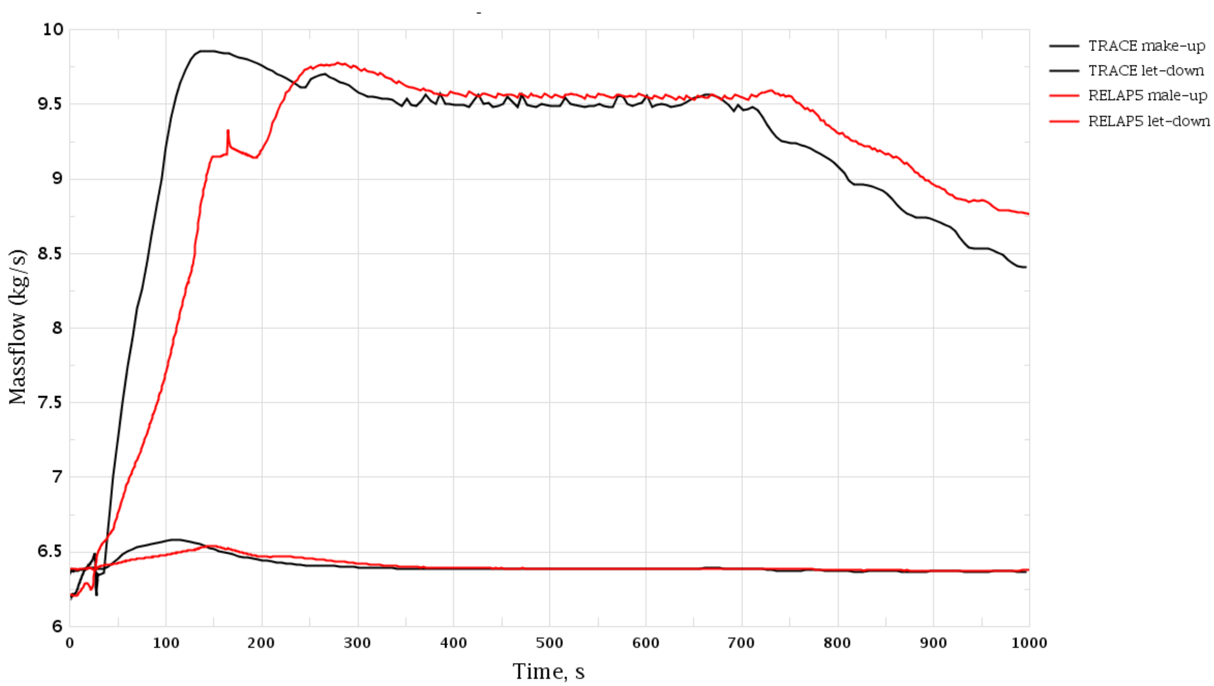

Figure 3-89 Trip of One RCP. Make-Up and Let-Down Mass Flow

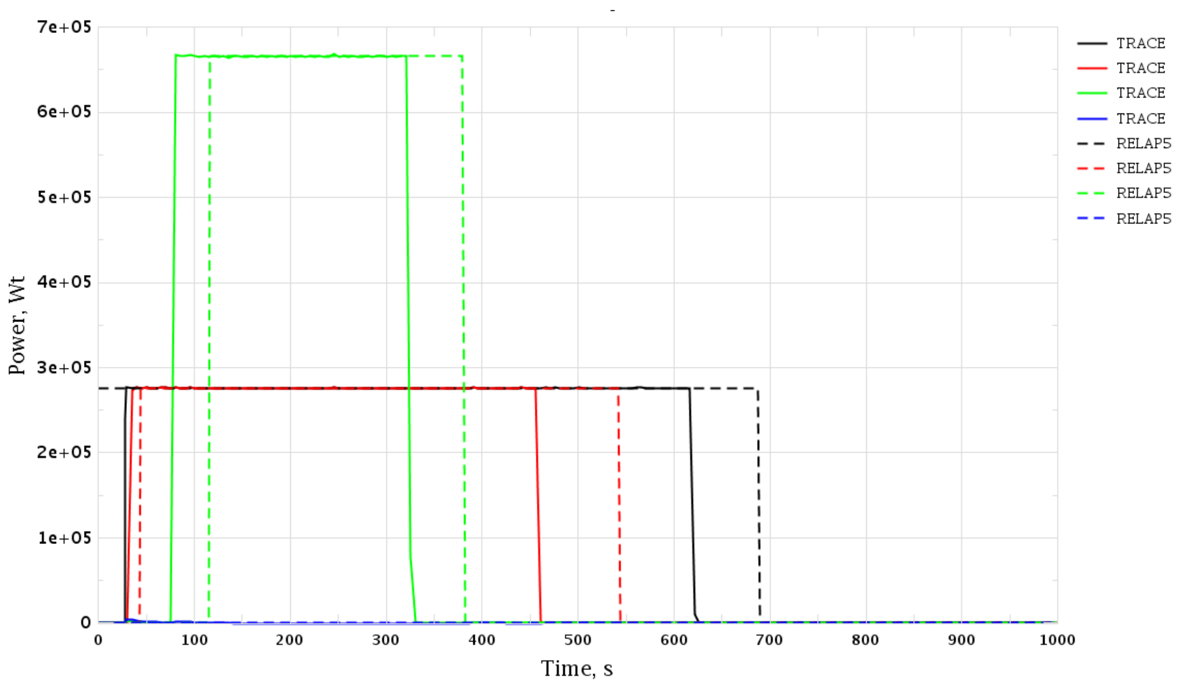

Figure 3-90 Trip of One RCP. PRZ Heaters Power

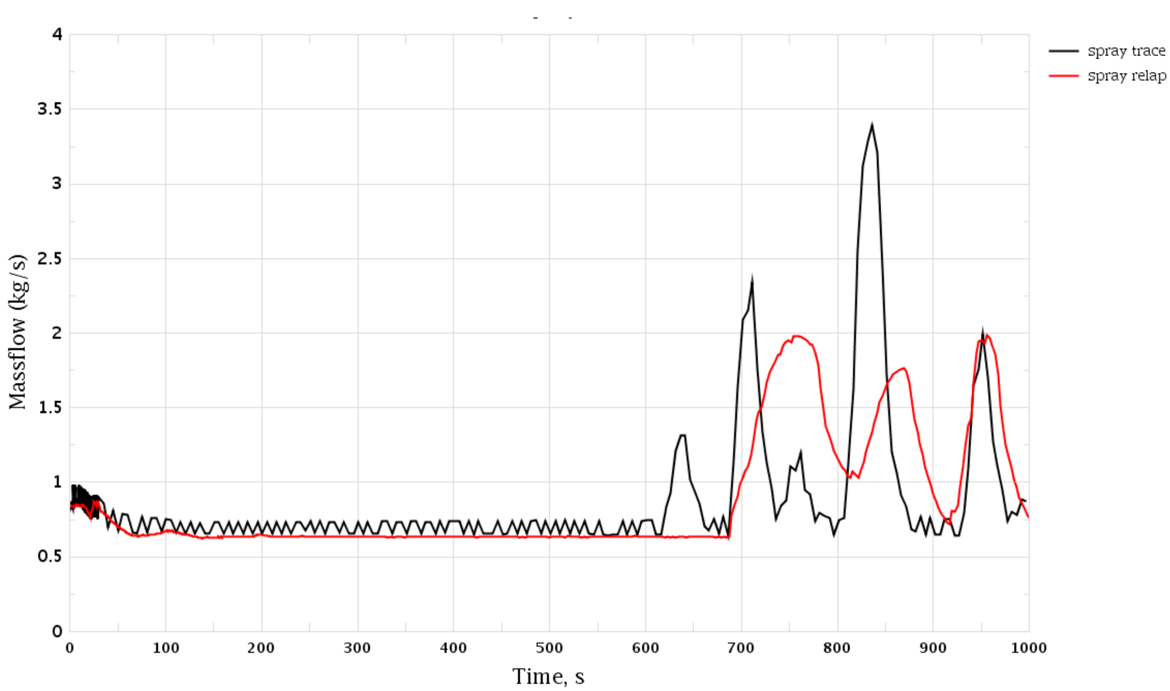

Figure 3-91 Trip of One RCP. PRZ Spray Mass Flow Rate

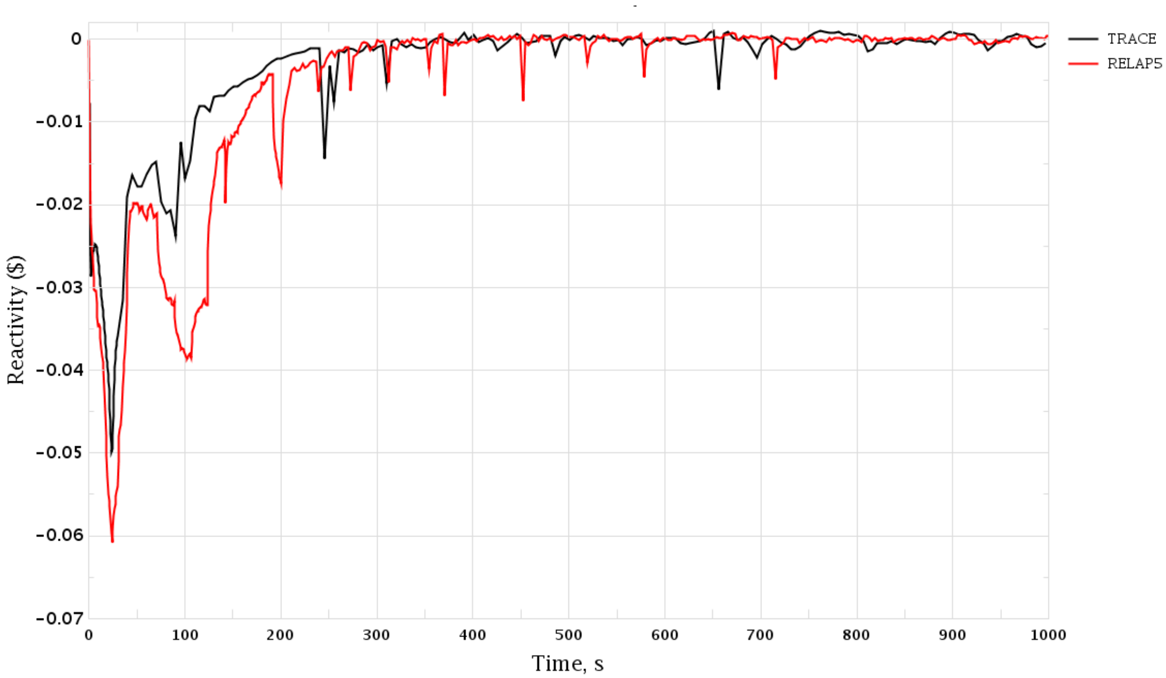

Figure 3-92 Trip of One RCP. Core Reactivity

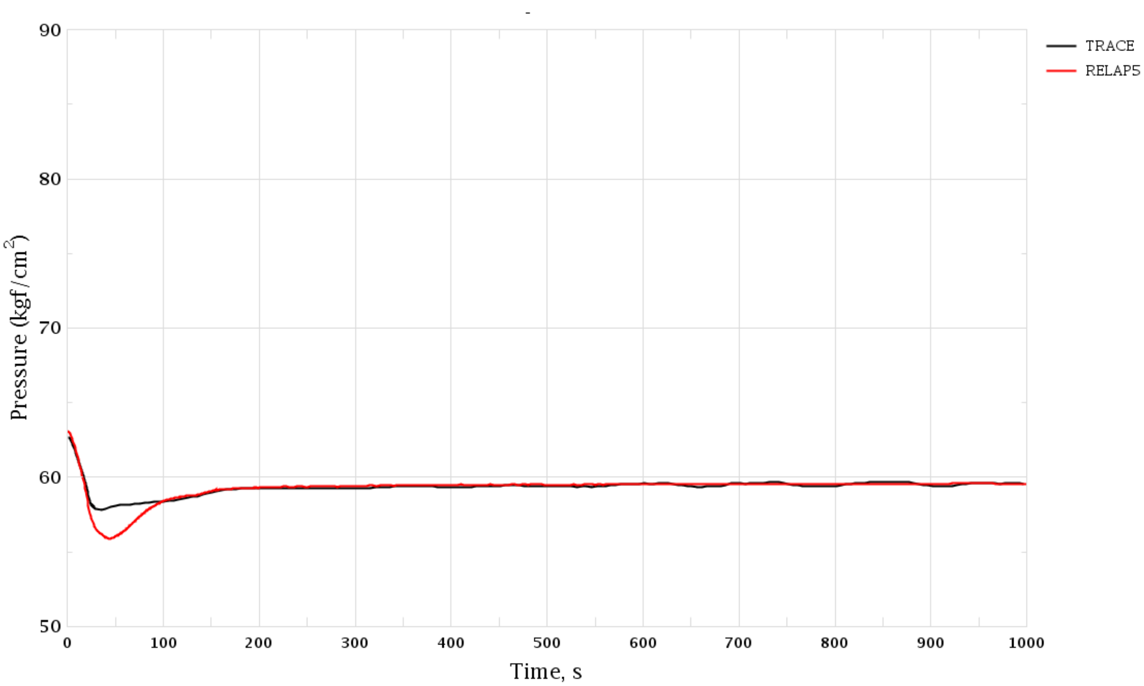

Figure 3-93 Trip of One RCP. SG-1 Pressure

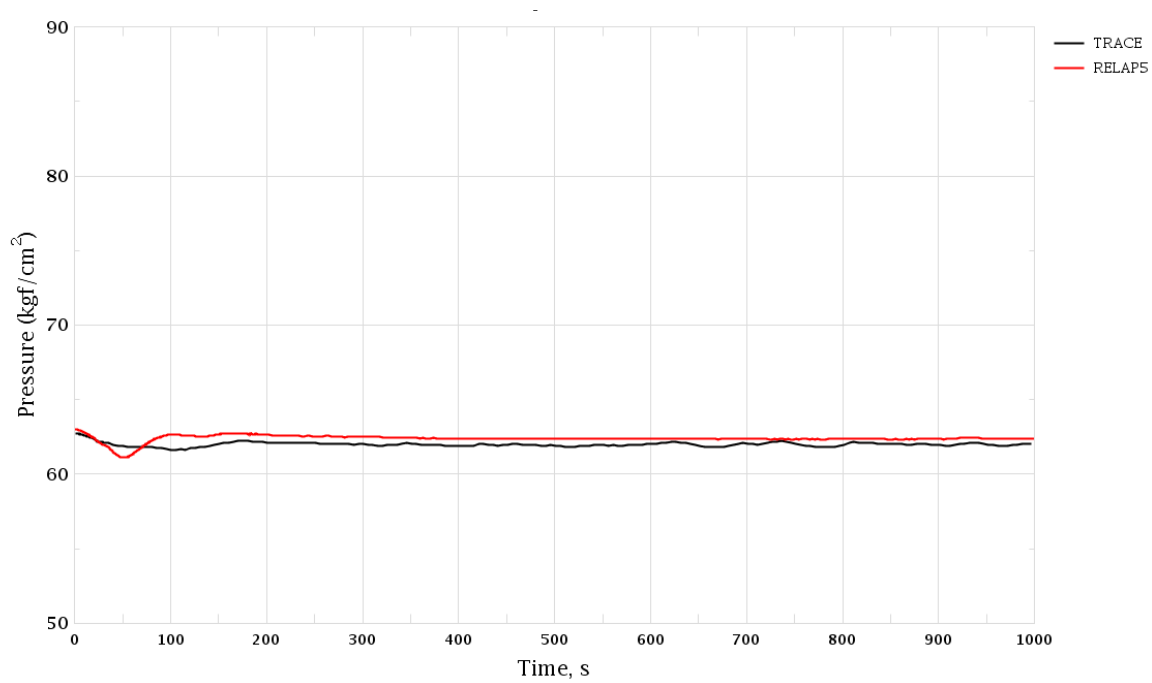

Figure 3-94 Trip of One RCP. SG-2 Pressure

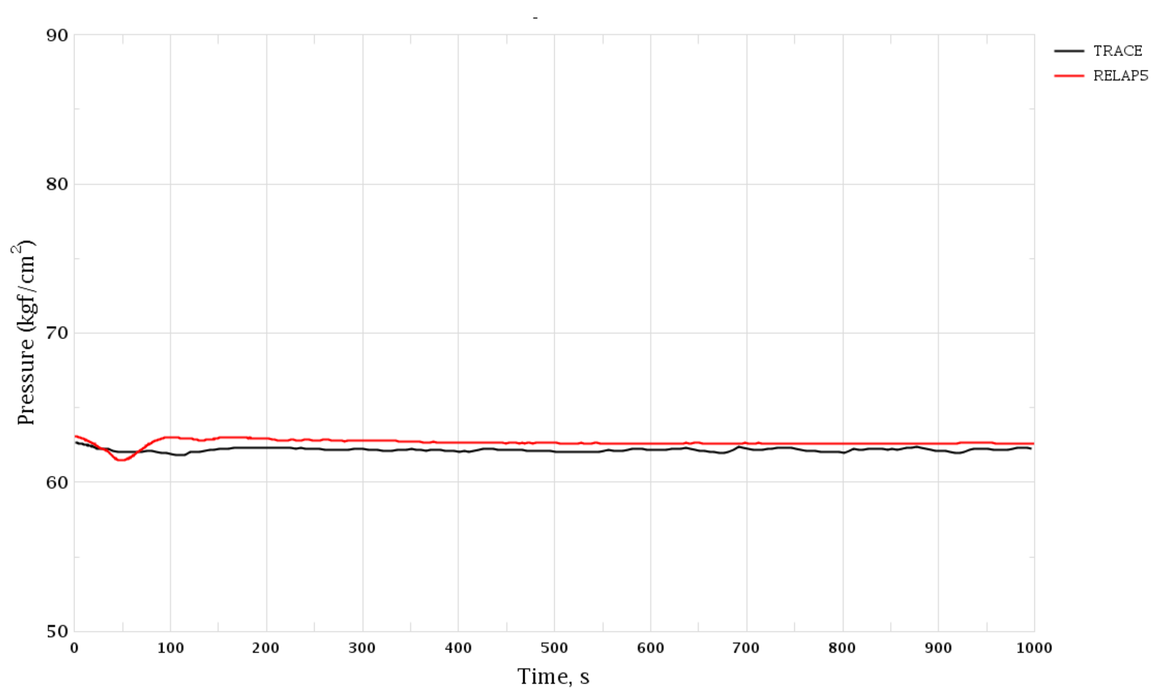

Figure 3-95 Trip of One RCP. SG-3 Pressure

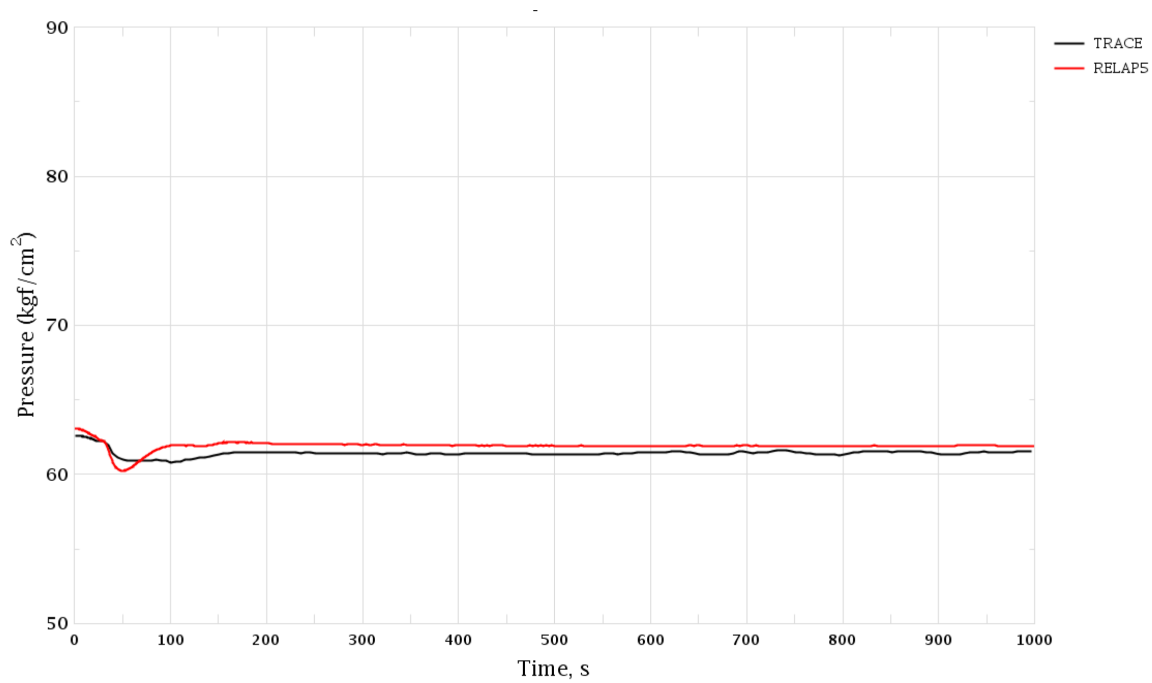

Figure 3-96 Trip of One RCP. SG-4 Pressure

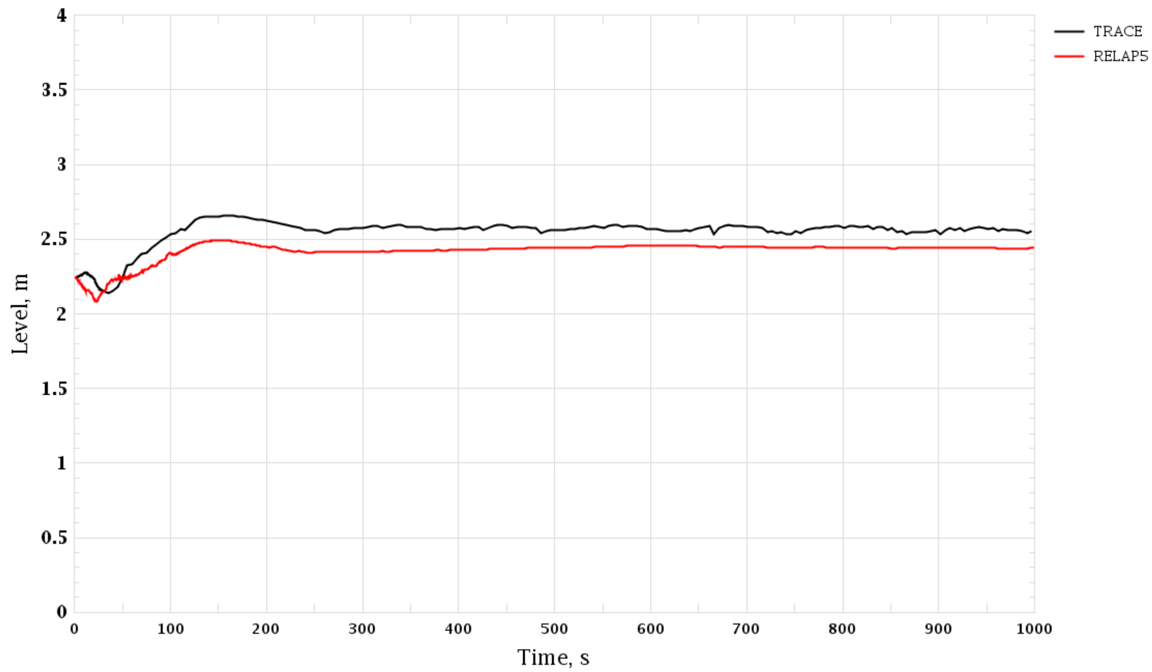

Figure 3-97 Trip of One RCP. SG-1 Level (Wide Range)

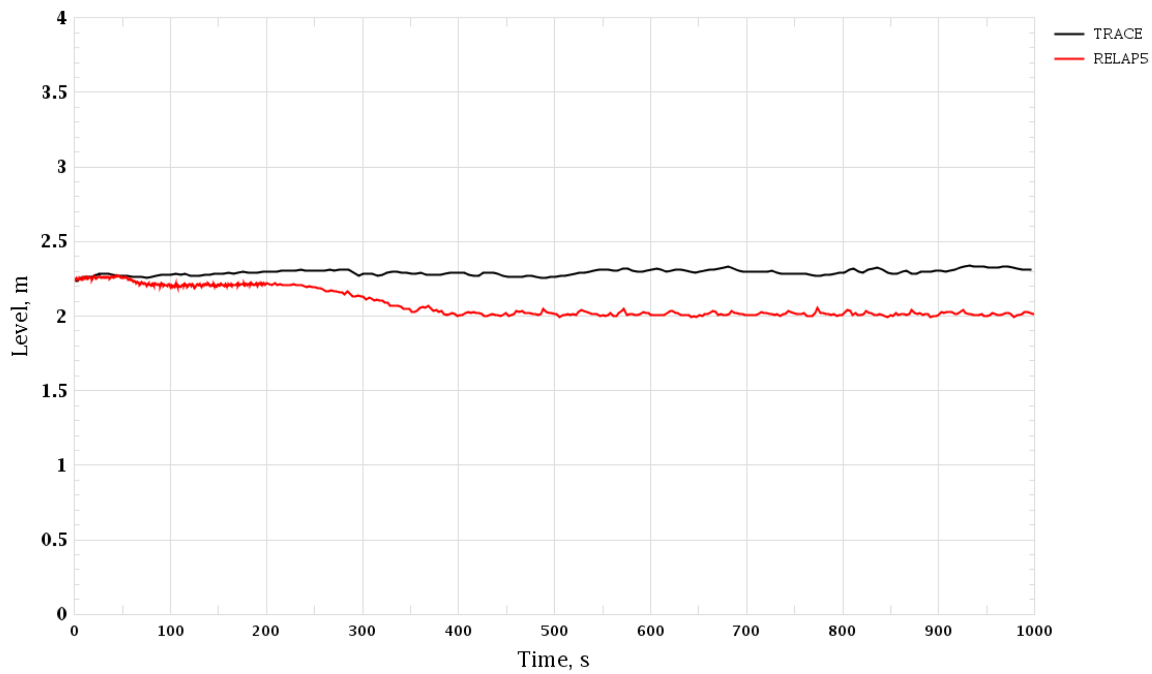

Figure 3-98 Trip of One RCP. SG-2 Level (Wide Range)

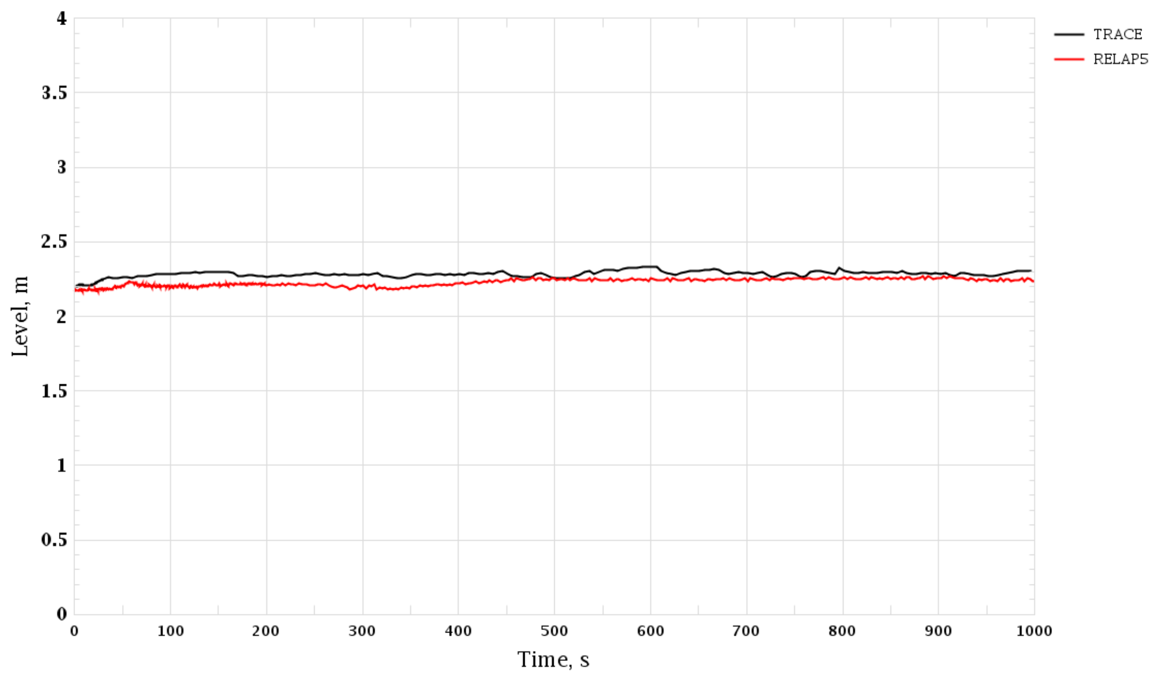

Figure 3-99 Trip of One RCP. SG-3 Level (Wide Range)

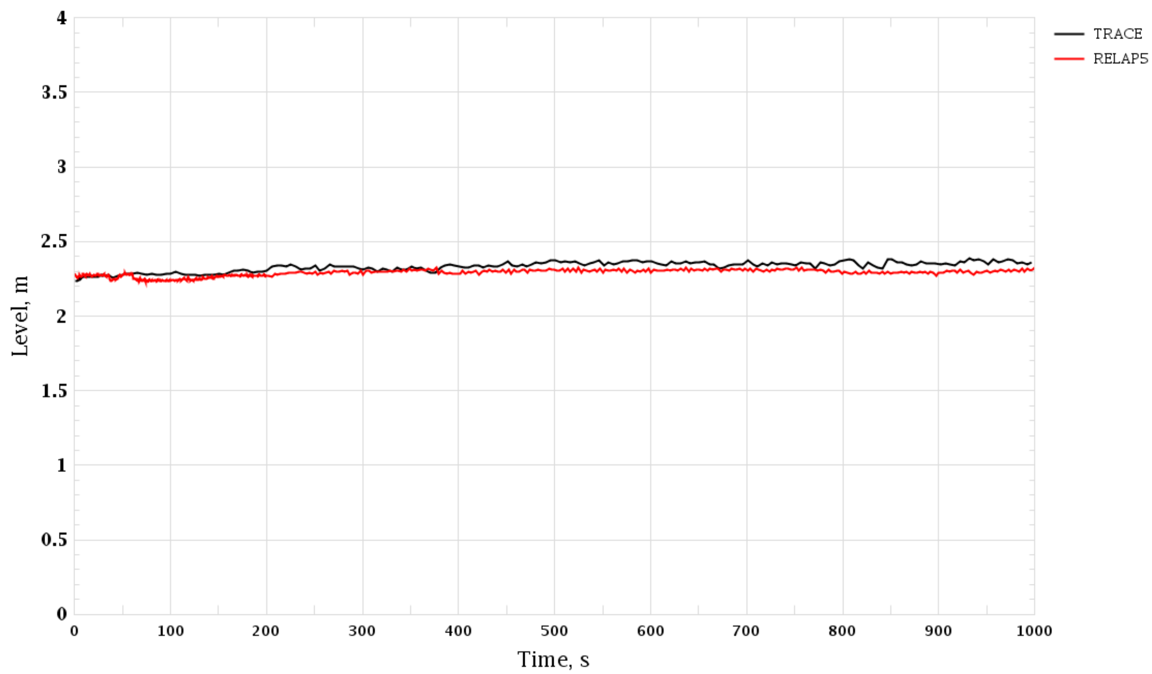

Figure 3-100 Trip of One RCP. SG-4 Level (Wide Range)

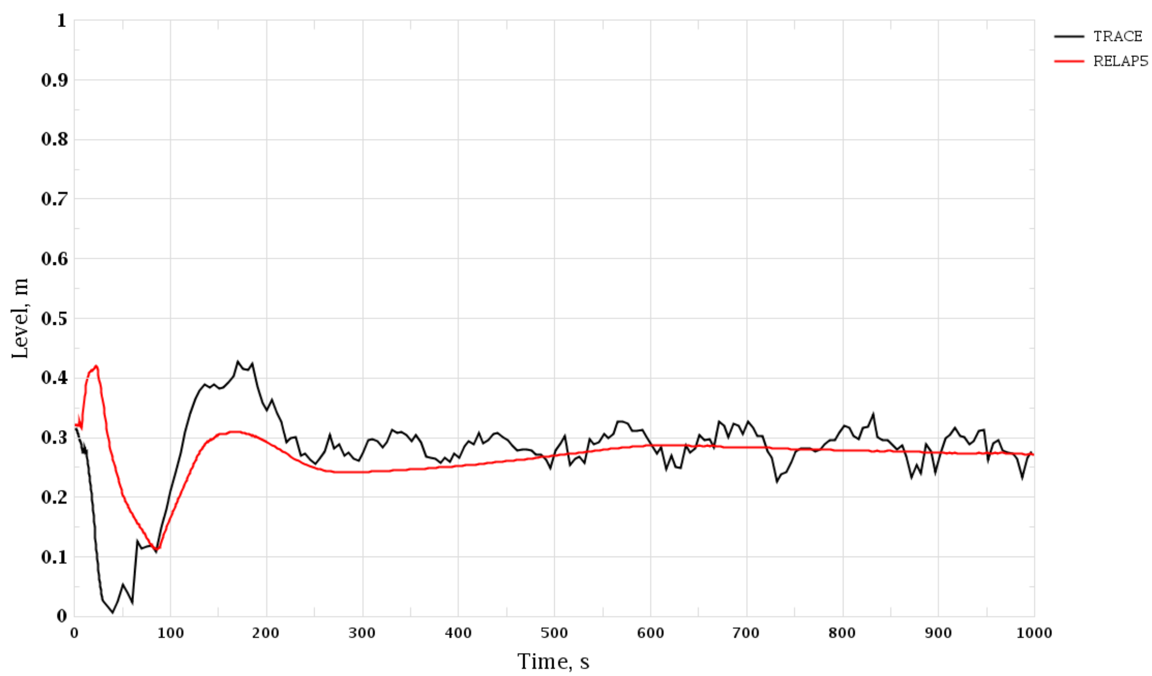

Figure 3-101 Trip of One RCP. SG-1 Level (Narrow Range)

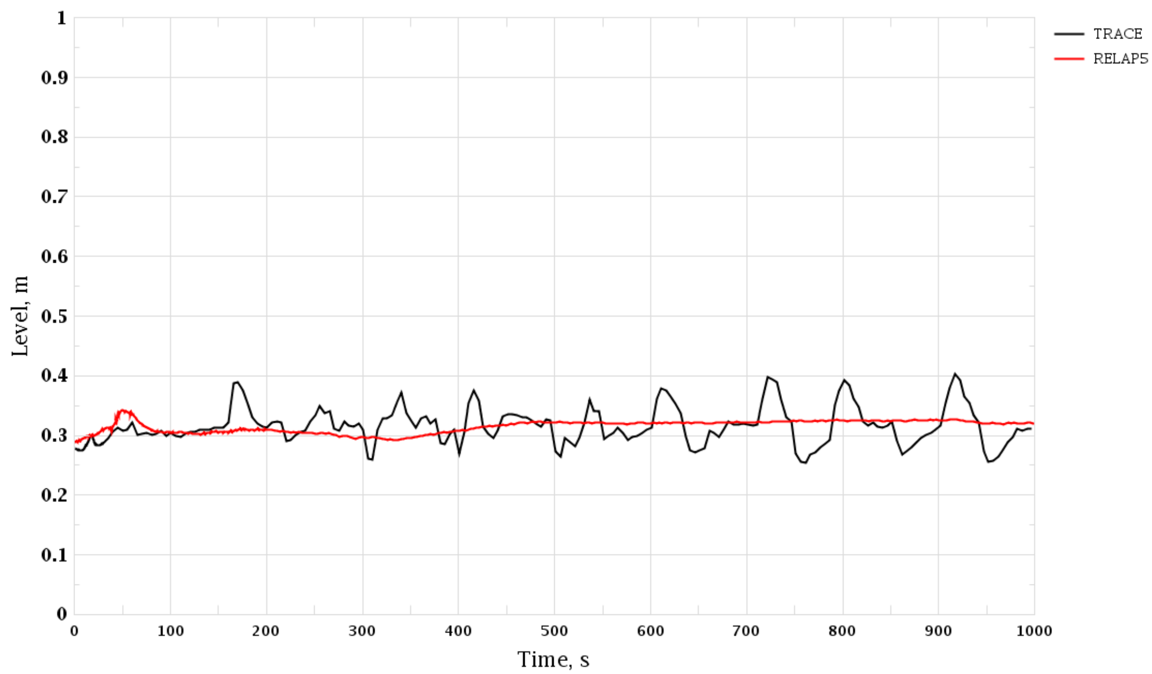

Figure 3-102 Trip of One RCP. SG-2 Level (Narrow Range)

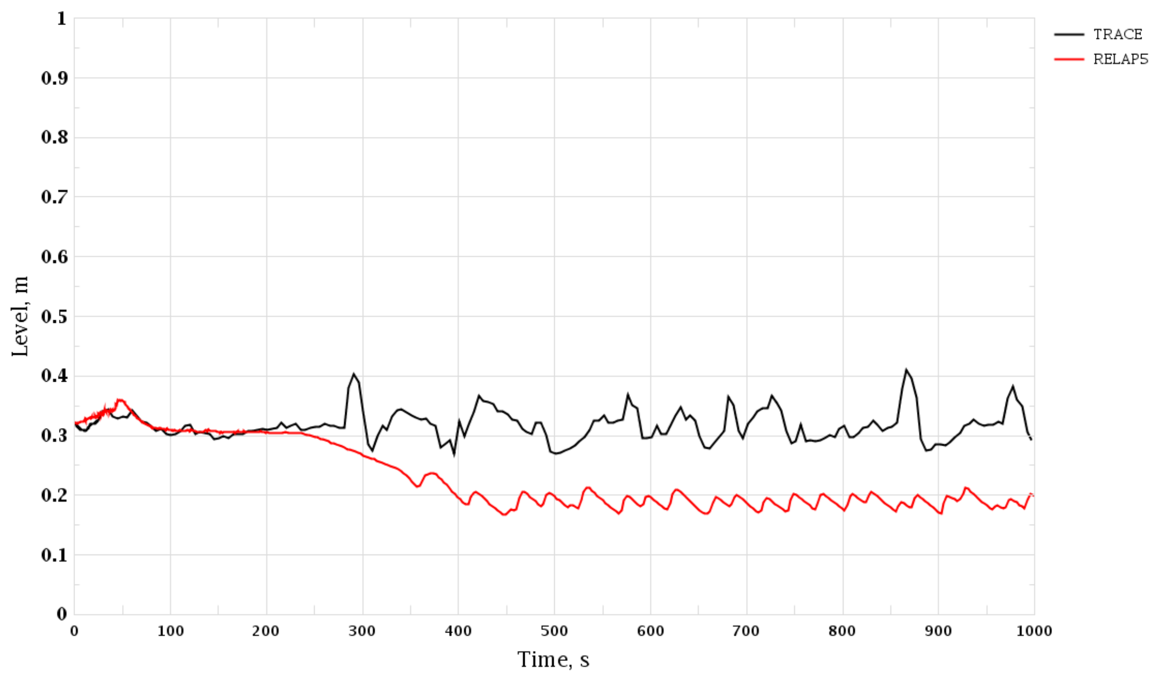

Figure 3-103 Trip of One RCP. SG-3 Level (Narrow Range)

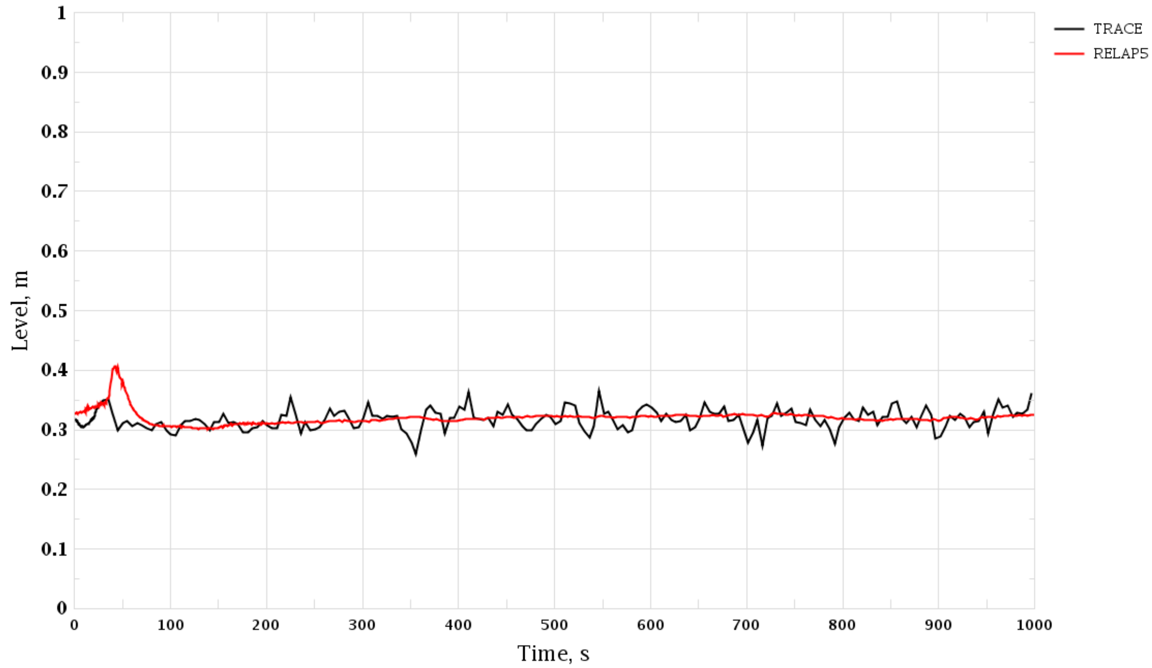

Figure 3-104 Trip of One RCP. SG-4 Level (Narrow Range)

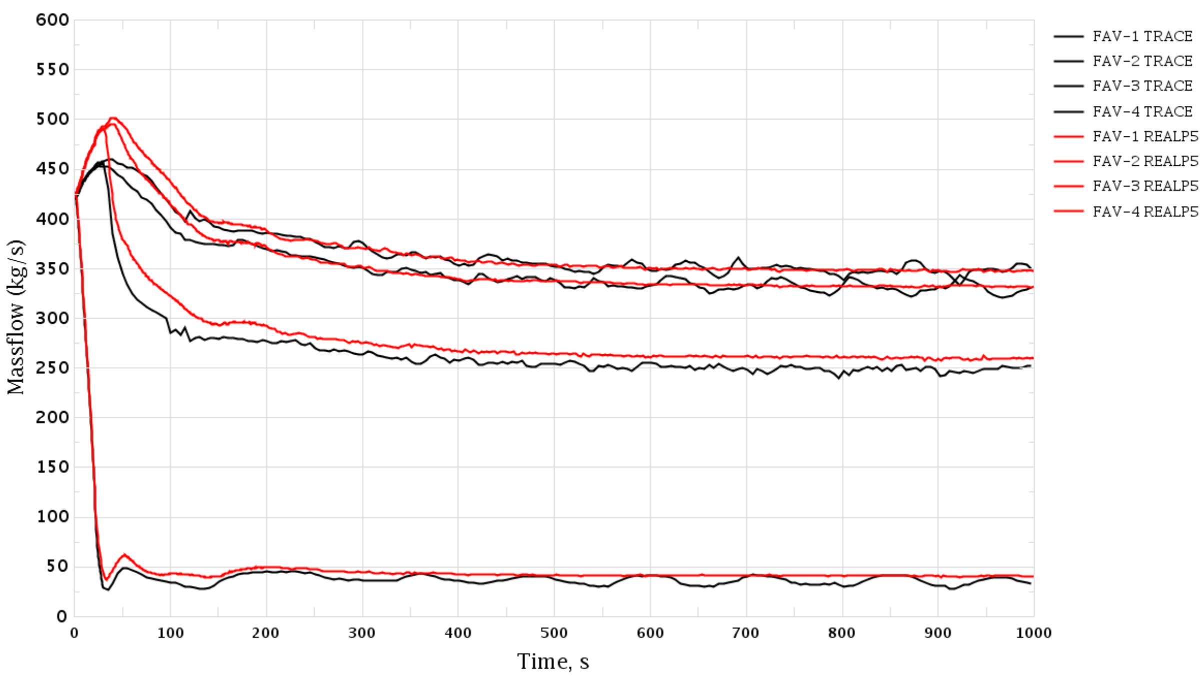

Figure 3-105 Trip of One RCP. Mass Flow through MSIV

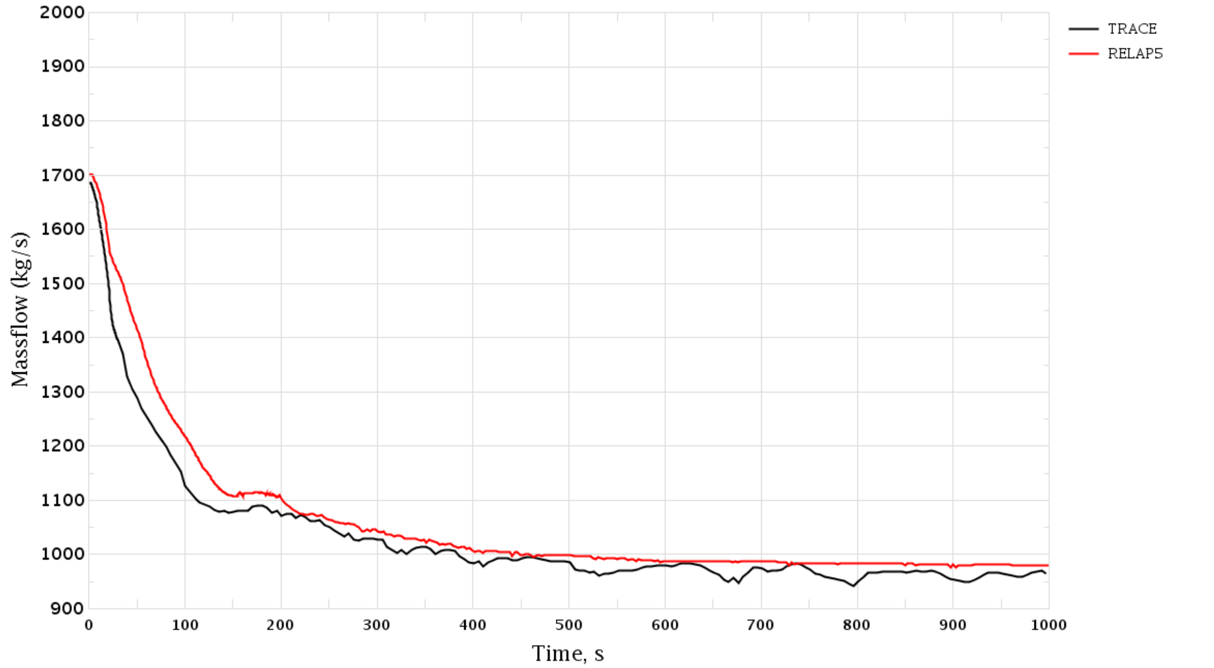

Figure 3-106 Trip of One RCP. Turbine Mass Flow Rate

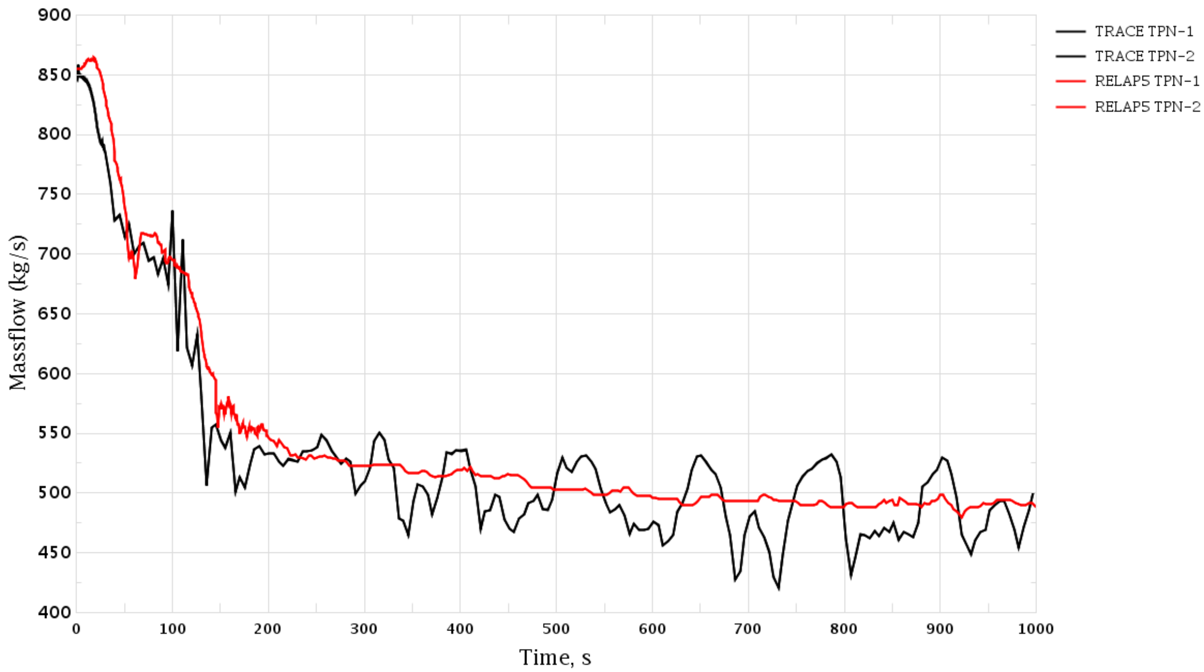

Figure 3-107 Trip of One RCP. MFW Pumps Mass Flow Rate

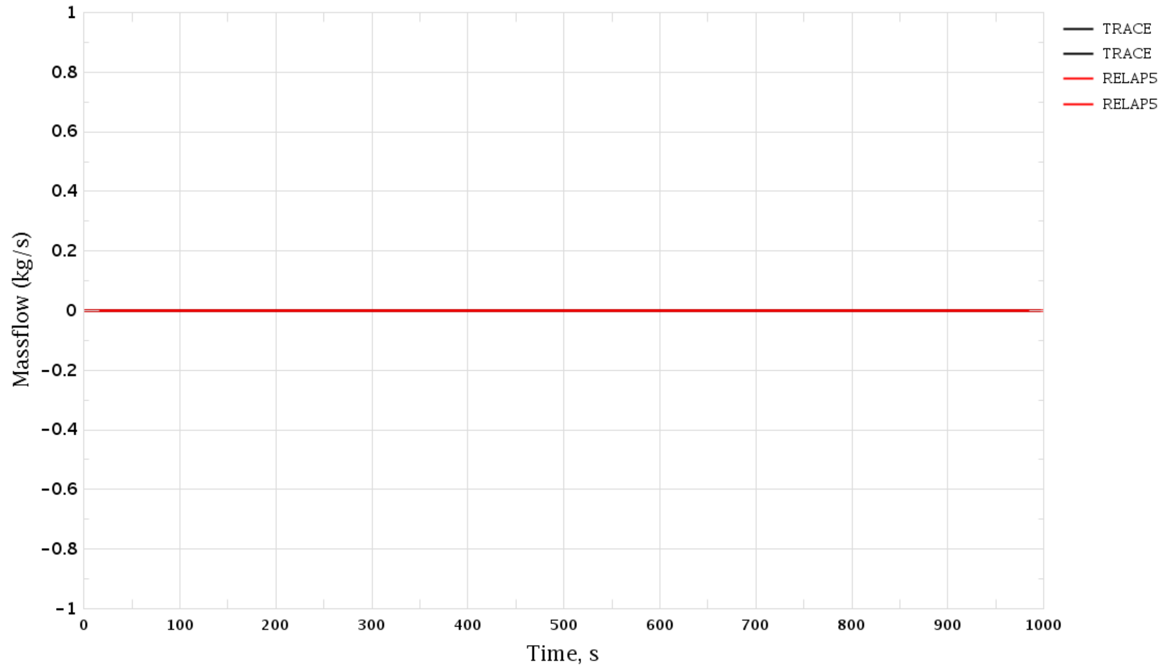

Figure 3-108 Trip of One RCP. AFW Mass Flow Rate

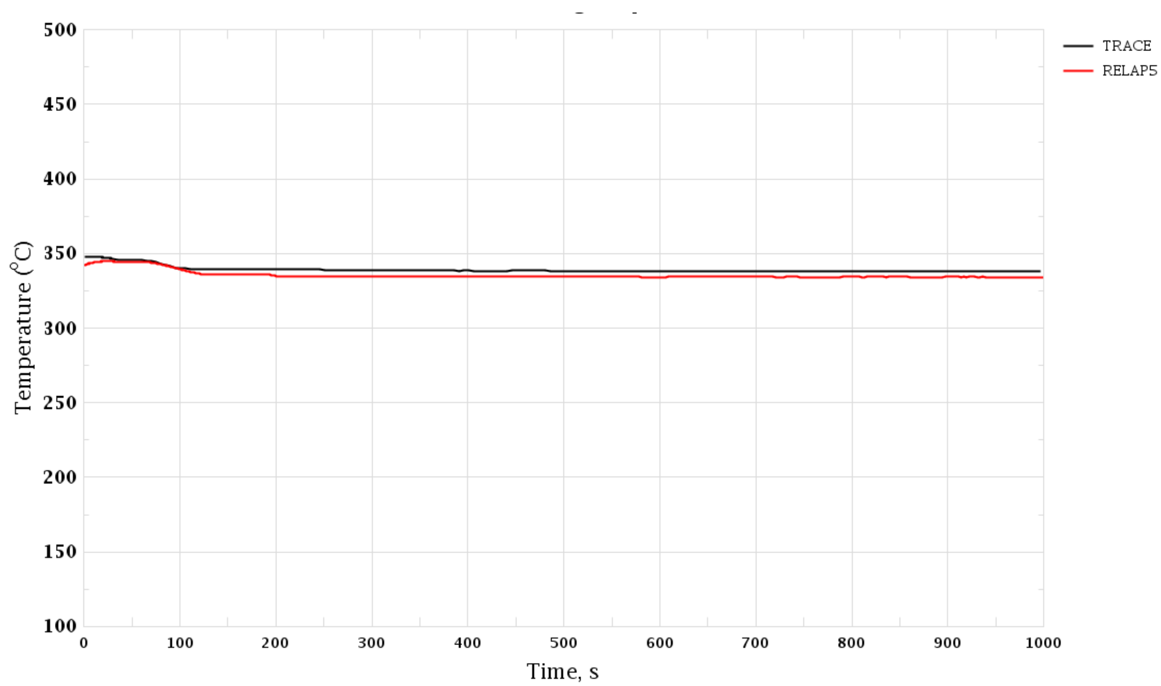

Figure 3-109 Trip of One RCP. Maximal Cladding Temperature

3.4 Uncontrolled Withdrawal of Control Rods Group

3.4.1 Brief Description of Initiating Event

This initiating event assumes uncontrolled withdrawal of control group of control rods 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.

Operation of ARM, RPL, PZ and UPZ to decrease power is not considered. Reactor scram occurs due to increase of neutron power to 107% taking into account +2% allowance for measurement error.

Loss of normal power supply is postulated at the time of reactor scram actuation. Postulated time delay for emergency diesel-generator (DG) start-up is 22 s.

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 Rods Group" Transient

TRACE Time, s	RELAP5 Time, s	Event	Description
0.0	0.0	Start of uncontrolled withdrawal of control group of control rods	
20.0	17.0	Start of PRZ spray valve YP13S02 opening	RCS pressure increase above 161 kgf/cm^2
34.0	27.0	Reactor scram	Reactor neutron power >107% +2% allowance for measurement error
		Loss of normal power supply	Postulated condition
		Turbine trip, trip of RCPs	Due to loss of normal power supply
		Trip of MFW pumps	Postulated (after turbine trip)
35.0	30.0	Maximal RCS pressure is 161.5 kgf/cm^2 (TRACE) and 165.5 kgf/cm^2 (RELAP5)	
35.0	30.0	First peak of maximal cladding temperature (348°C in TRACE calculation and 345.2°C in RELAP5 calculation)	
40.0	29.0	BRU-A opening	Increase of pressure in correspondent steam line over 73+1 kgf/cm^2
56.0	49.0	DG start-up Start of sequential DG loading program (SLP)	Time delay for DG start-up is 22 s (postulated)
74.0	75.0	End of signal for opening of PRZ spray valve	Decrease of RCS pressure
79.0	68.0	Start of AFW supply to SG	Decrease of narrow range SG level below 0.22 m (after start-up of site emergency diesel-generators)
220.0	251.0	Second peak of maximal cladding temperature (316.5°C in TRACE calculation and 312.5°C in RELAP5 calculation)	
1800.0	1800.0	End of calculation	

Uncontrolled withdrawal of the control group of control rods with a normal operating speed of 20 mm/s causes insertion of positive reactivity and thus, an initial increase of reactor power (Figure 3-110).

At 27.0/34.0 s (in RELAP and TRACE calculations, respectively), the reactor neutron power reaches 107.0+2% of the nominal value (Figure 3-110) that causes scram signal actuation. Increase of coolant temperature (Figure 3-113 – Figure 3-120) and RCS pressure (Figure 3-111) at the beginning of transient due to initial reactor power increase is quickly terminated after reactor scram, and RCS temperature and pressure decrease rapidly.

Simultaneously with reactor scram the loss of normal power supply is postulated that leads to a trip of RCPs (Figure 3-121 – Figure 3-124), closure of turbine stop valves (Figure 3-137), trip of MFW pumps (Figure 3-139) and formation of signal to start emergency DG (including additional site emergency DGs).

Closure of turbine's stop valves after the reactor scram causes sharp increase of the secondary circuit pressure (Figures 131–134). This causes actuation of BRU-A in pressure maintenance mode (Figure 3-138) that automatically decrease the secondary circuit pressure according to a design algorithm. After start-up of additional site emergency DGs, AFW pumps are started at 68/79 s of transient restoring FW supply to SGs (Figure 3-140).

Decrease of primary coolant temperature (Figure 3-113 – Figure 3-120) within first 100 s of transient and correspondent coolant shrinkage result in a decrease of PRZ level (Figure 3-112), which is gradually restored by operation of PRZ level controllers that adjust make-up and let-down flow (Figure 3-125).

Within 1800 s of transient all primary and secondary circuit parameters are close to stabilization. The decay heat is removed in natural circulation mode by the secondary circuit via BRU-A and AFW pumps operation.

The plots of the main parameters of calculation are presented below on Figure 3-110 – Figure 3-141.

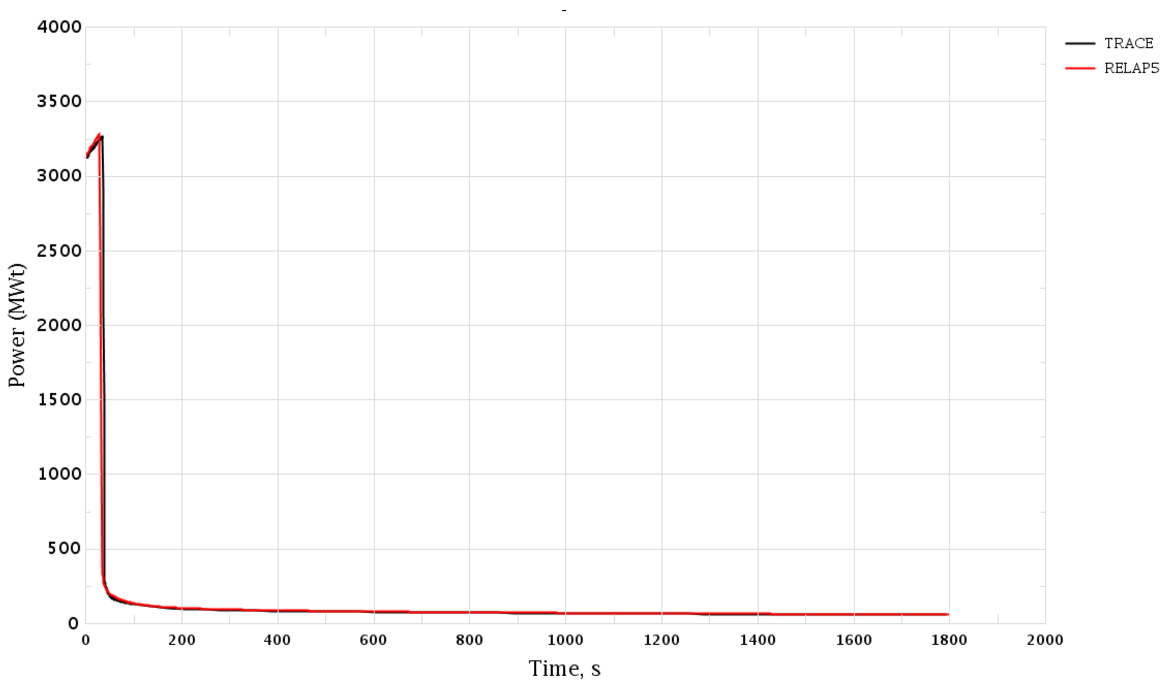

Figure 3-110 Uncontrolled Withdrawal of CR Group. Core Thermal Power

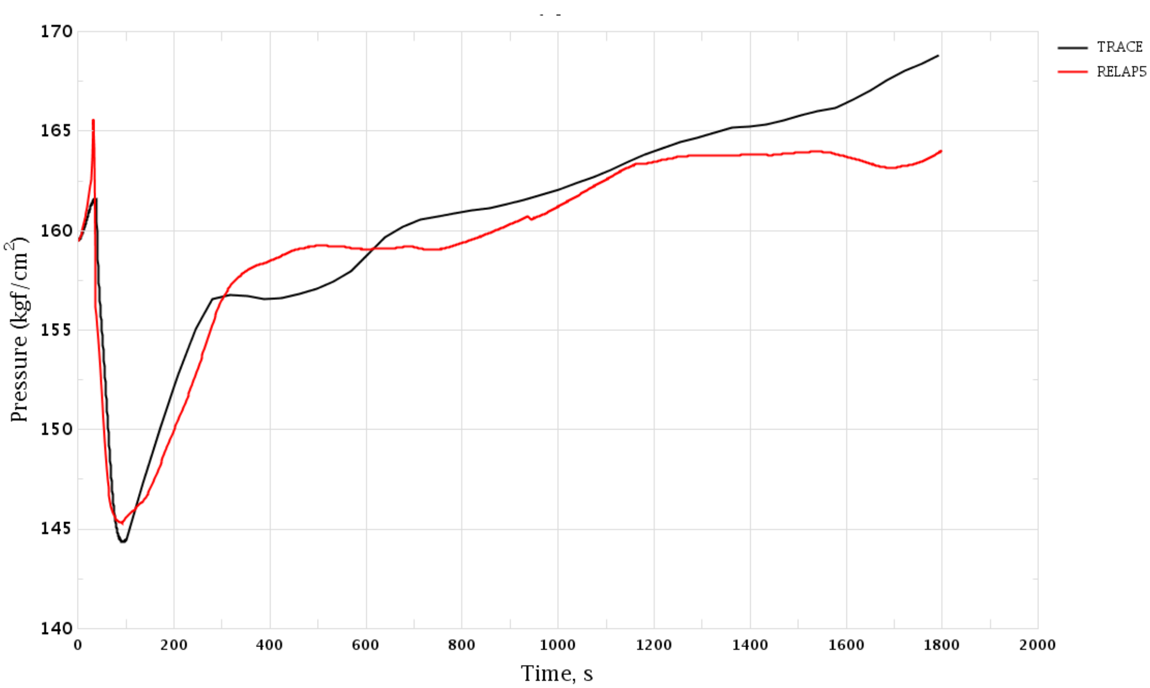

Figure 3-111 Uncontrolled Withdrawal of CR Group. RCS Pressure

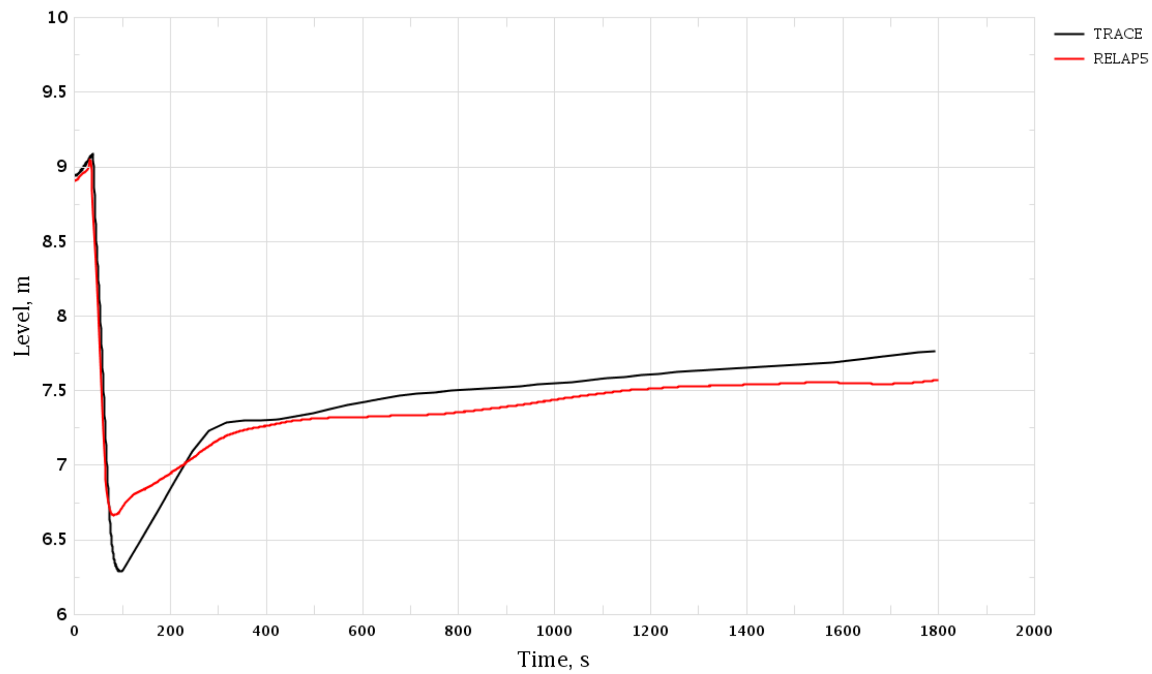

Figure 3-112 Uncontrolled Withdrawal of CR Group. Pressurizer Level

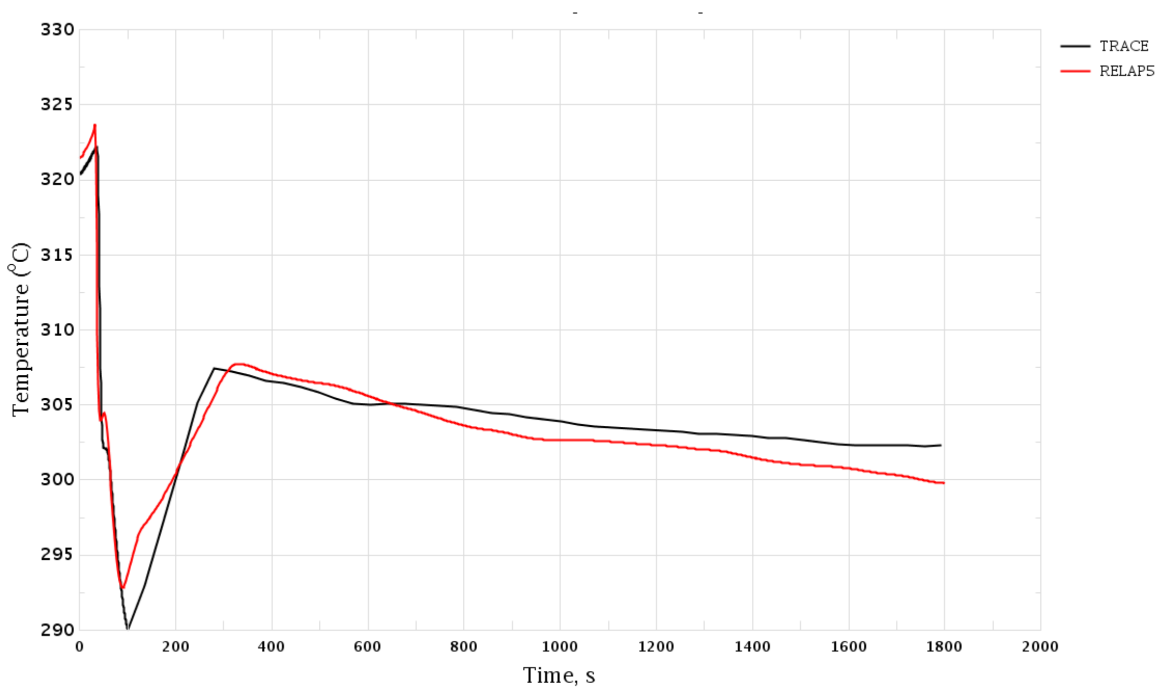

Figure 3-113 Uncontrolled Withdrawal of CR Group. Coolant Temperature in Hot Leg, Loop 1

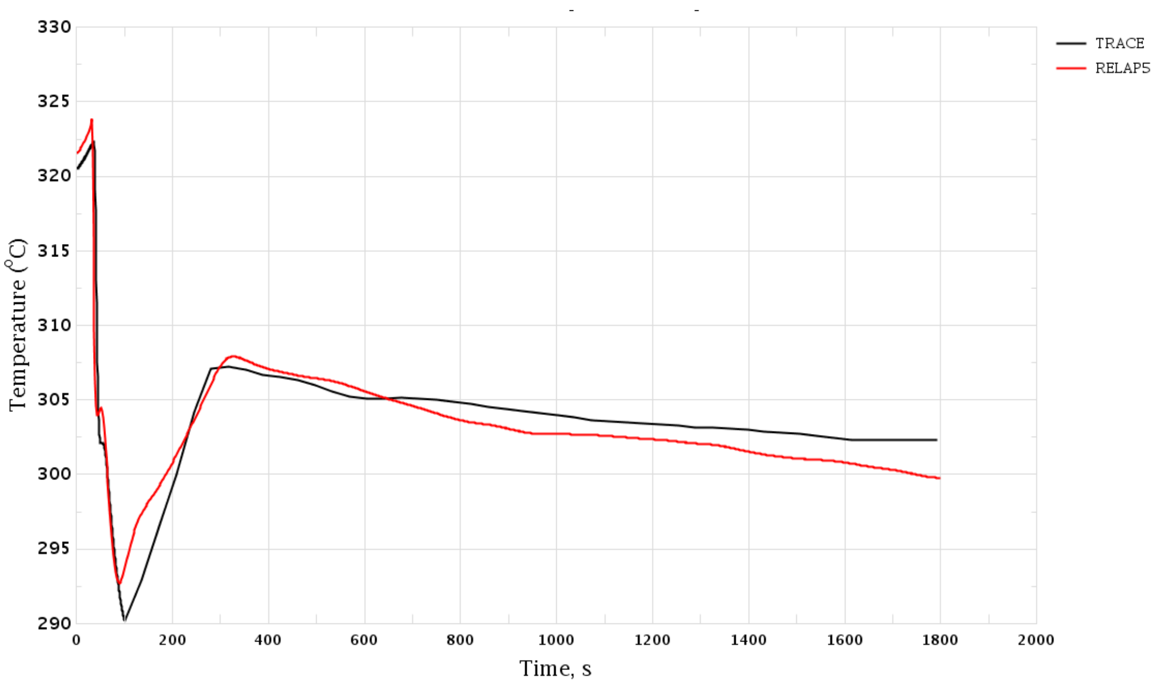

Figure 3-114 Uncontrolled Withdrawal of CR Group. Coolant Temperature in Hot Leg, Loop 2

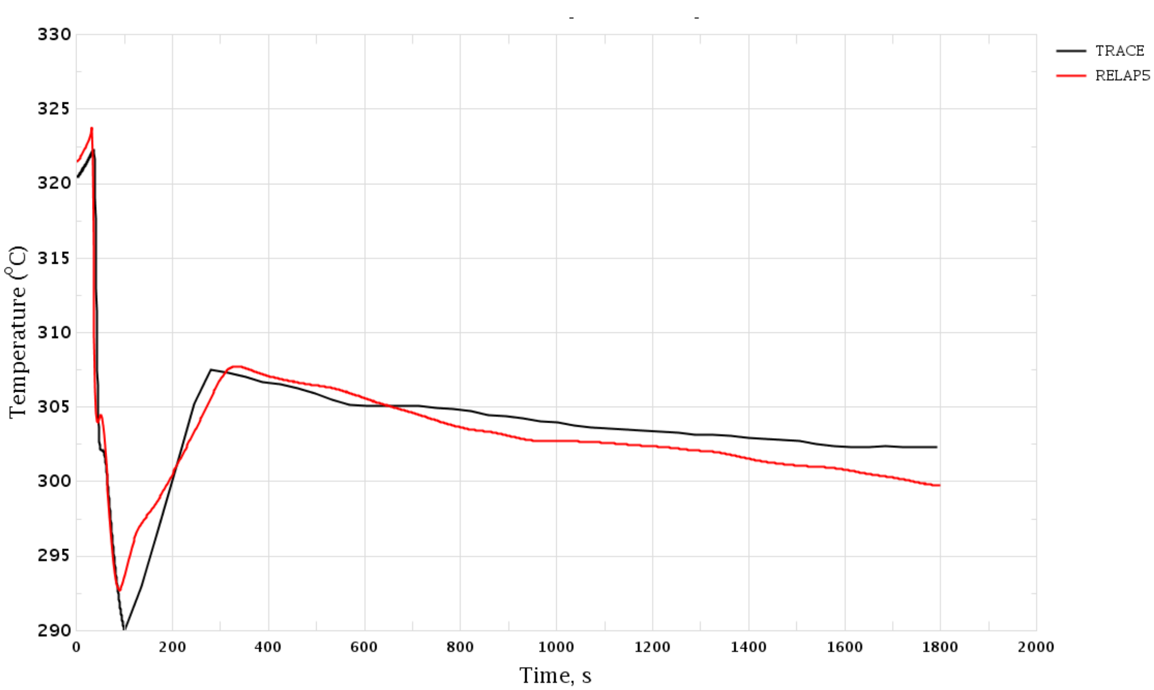

Figure 3-115 Uncontrolled Withdrawal of CR Group. Coolant Temperature in Hot Leg, Loop 3

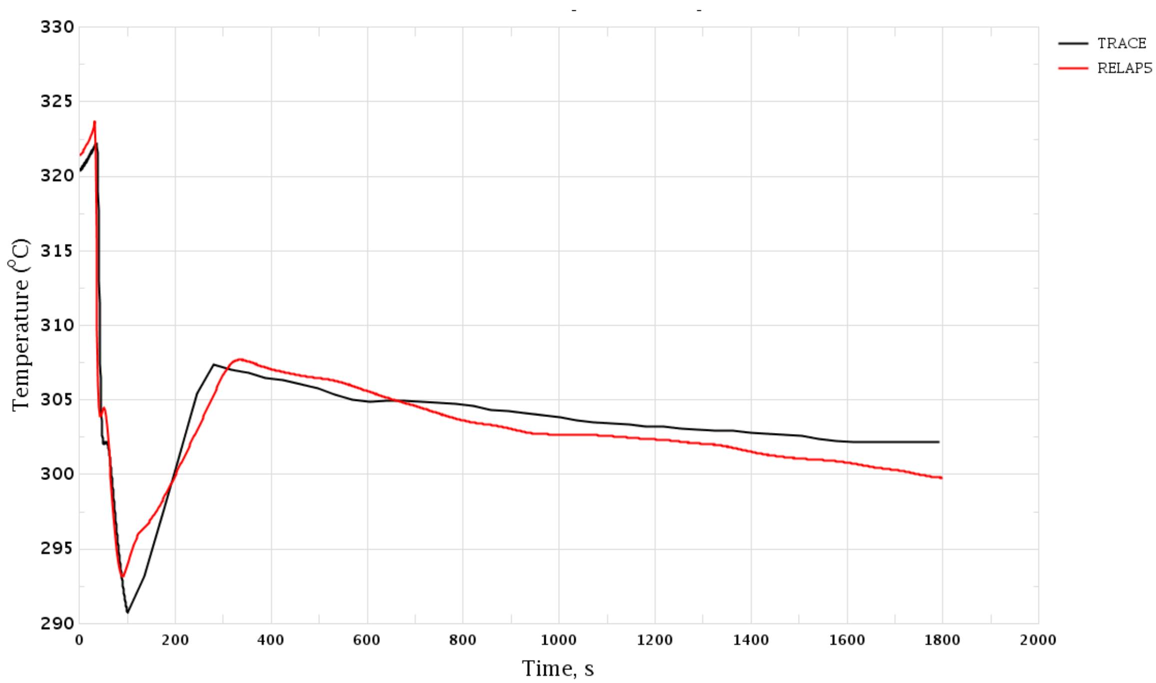

Figure 3-116 Uncontrolled Withdrawal of CR Group. Coolant Temperature in Hot Leg, Loop 4

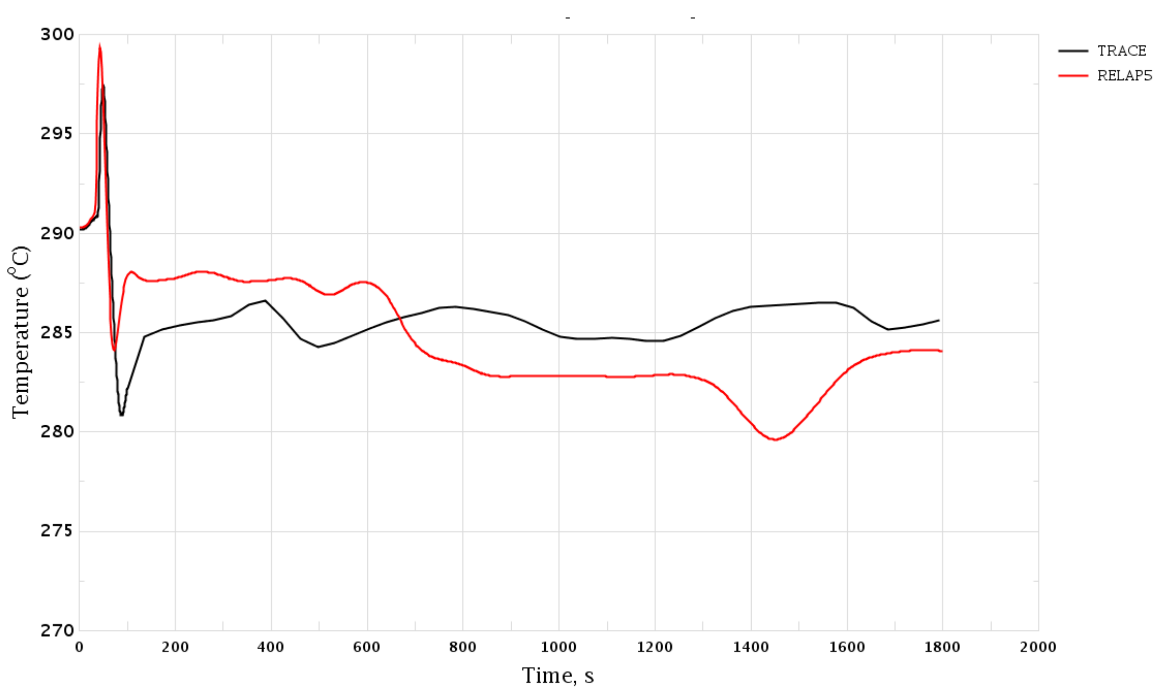

Figure 3-117 Uncontrolled Withdrawal of CR Group. Coolant Temperature in Cold Leg, Loop 1

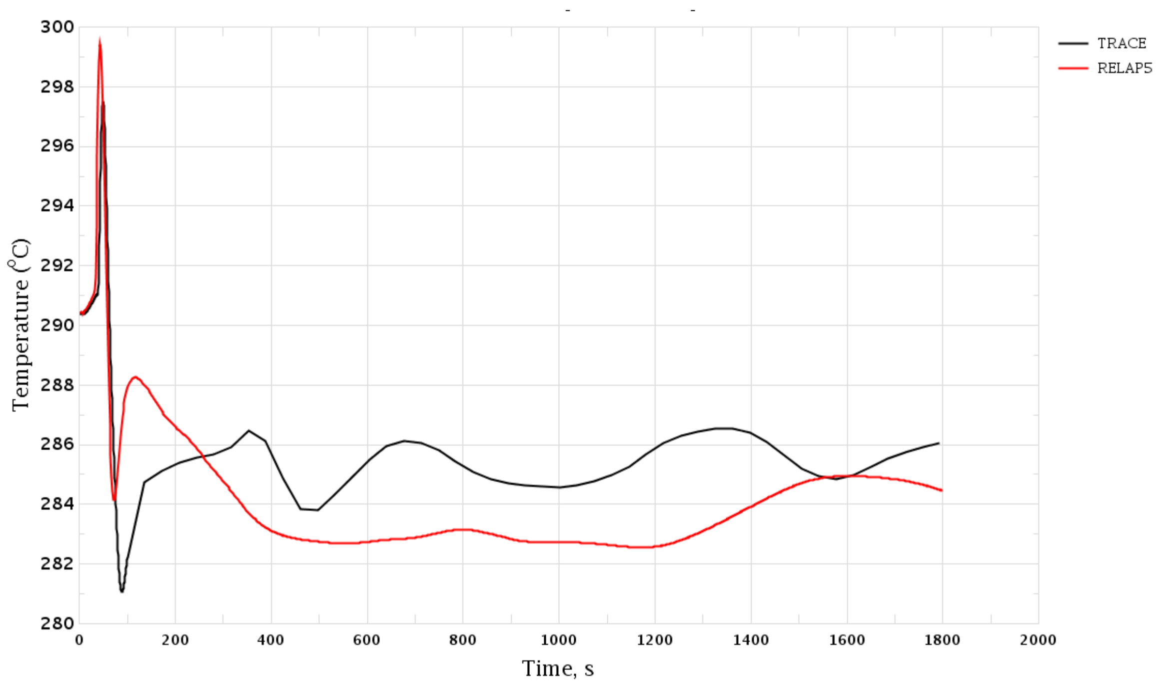

Figure 3-118 Uncontrolled Withdrawal of CR Group. Coolant Temperature in Cold Leg, Loop 2

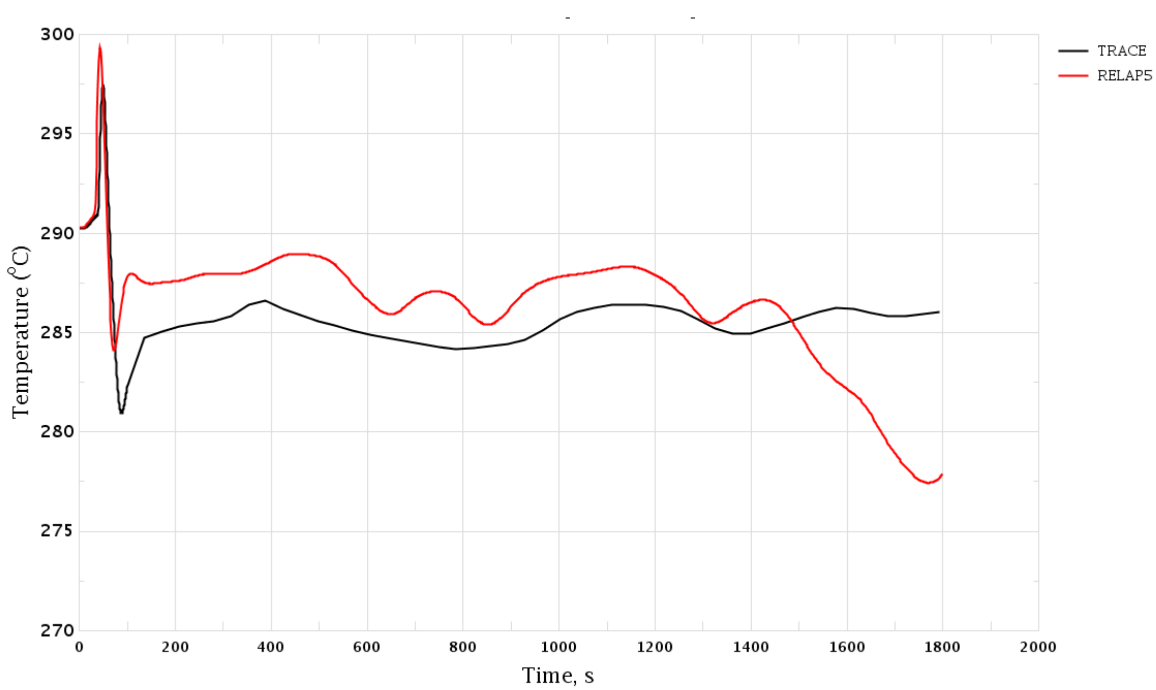

Figure 3-119 Uncontrolled Withdrawal of CR Group. Coolant Temperature in Cold Leg, Loop 3

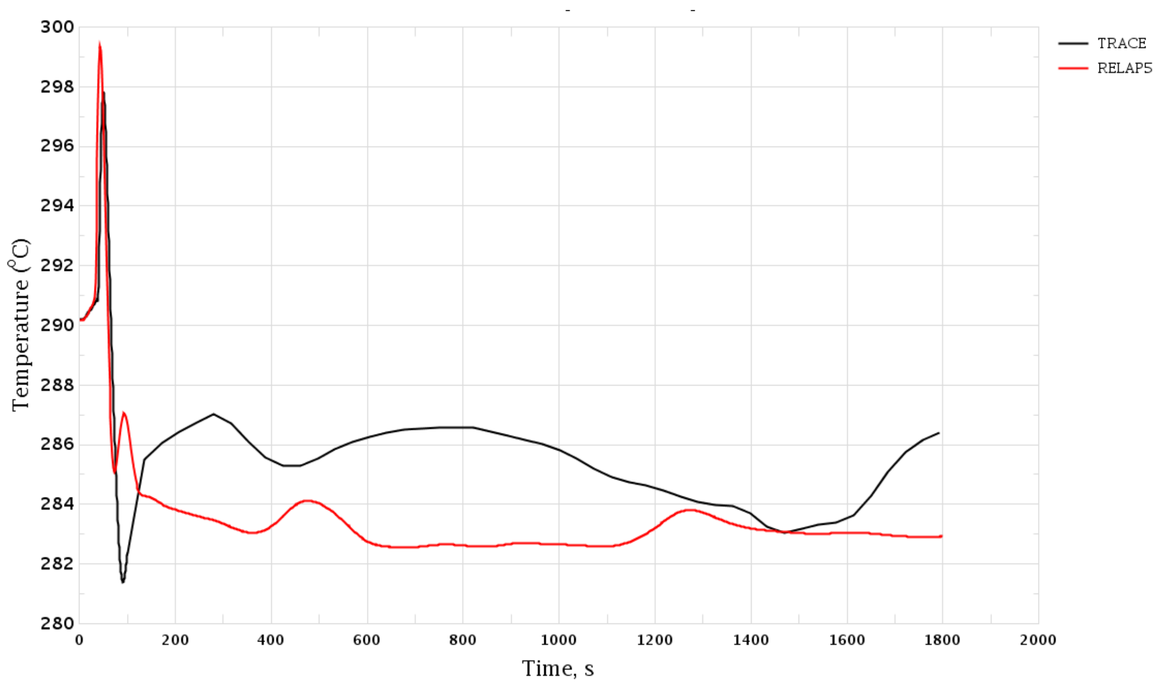

Figure 3-120 Uncontrolled Withdrawal of CR Group. Coolant Temperature in Cold Leg, Loop 4

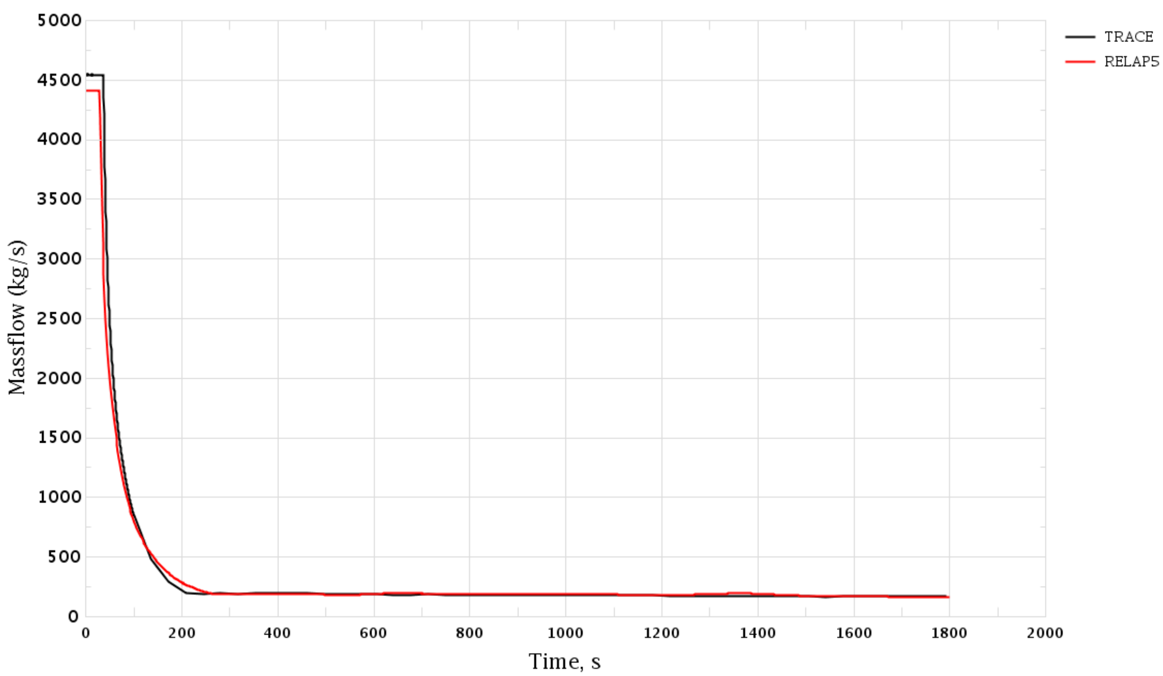

Figure 3-121 Uncontrolled Withdrawal of CR Group. RCS Loop 1 Mass Flow Rate

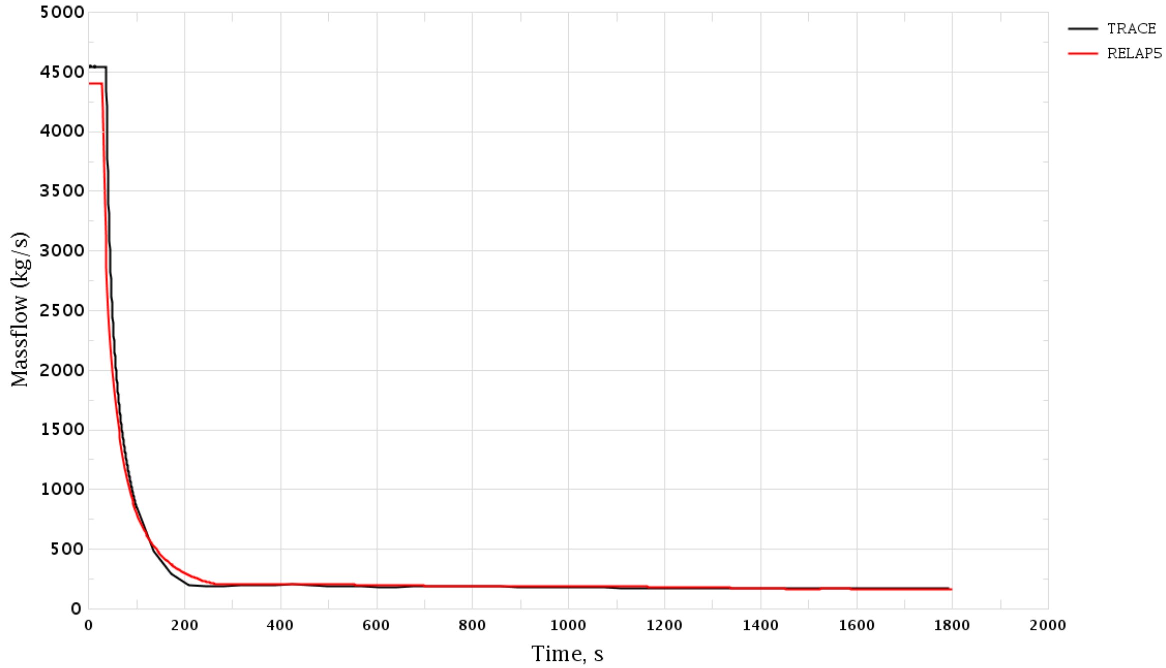

Figure 3-122 Uncontrolled Withdrawal of CR Group. RCS Loop 2 Mass Flow Rate

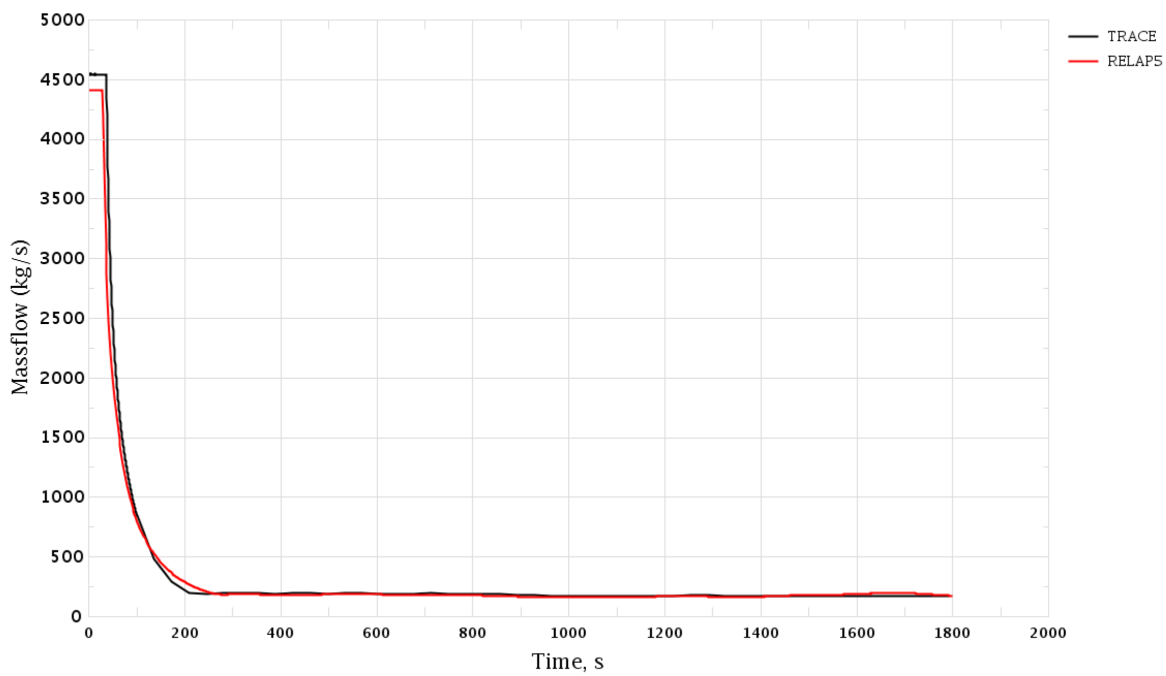

Figure 3-123 Uncontrolled Withdrawal of CR Group. RCS Loop 3 Mass Flow Rate

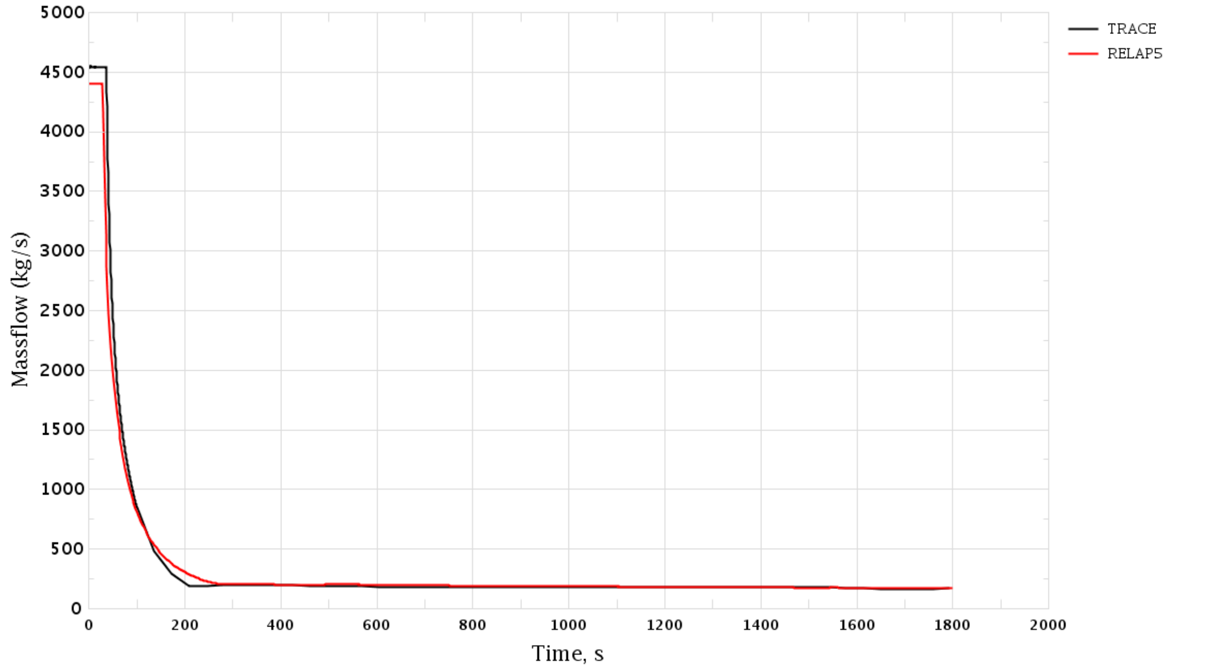

Figure 3-124 Uncontrolled Withdrawal of CR Group. RCS Loop 4 Mass Flow Rate

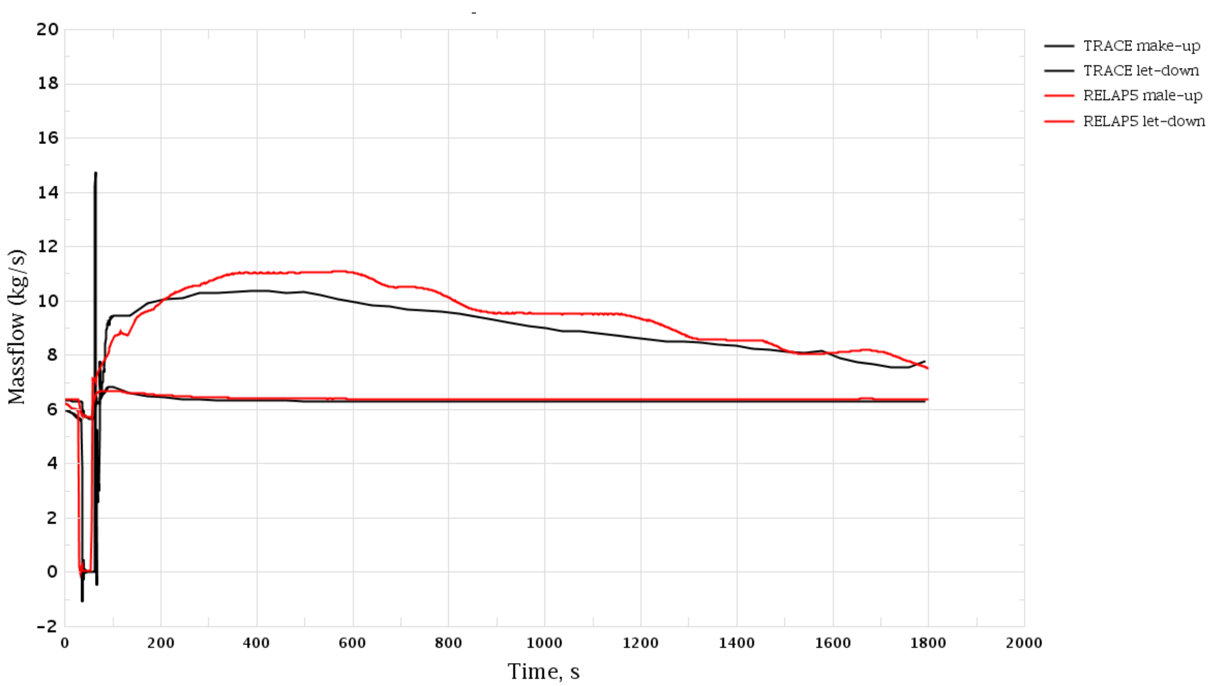

Figure 3-125 Uncontrolled Withdrawal of CR Group. Make-Up and Let-Down Mass Flow

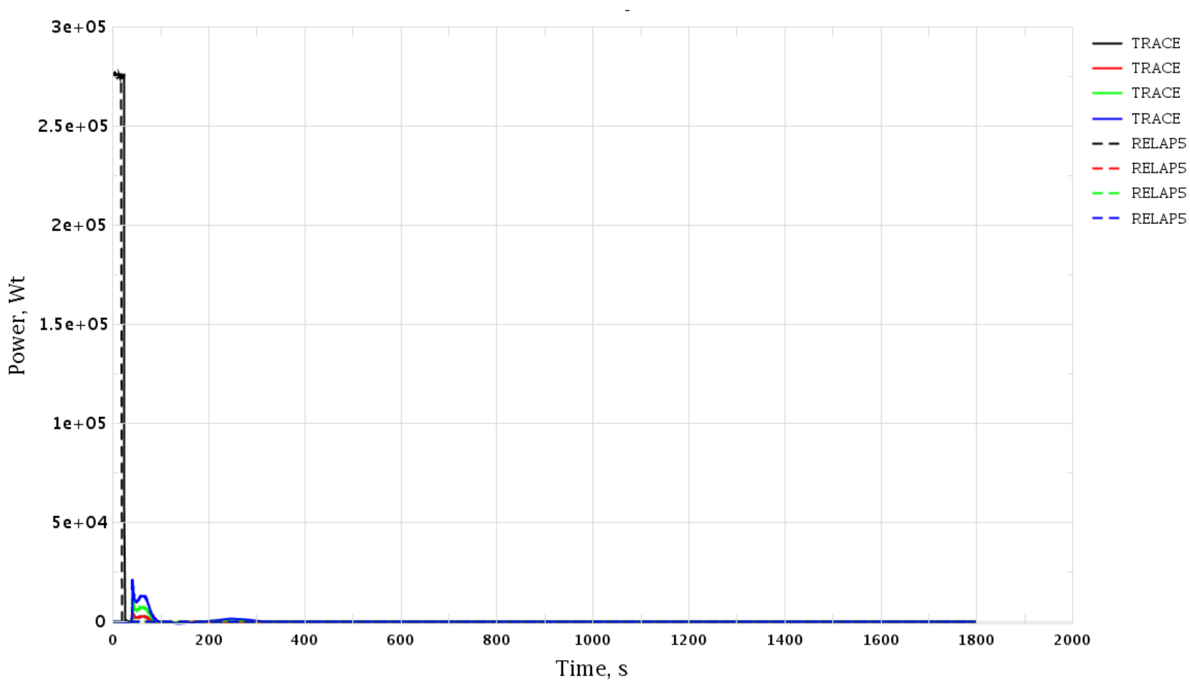

Figure 3-126 Uncontrolled Withdrawal of CR Group. PRZ Heaters Power

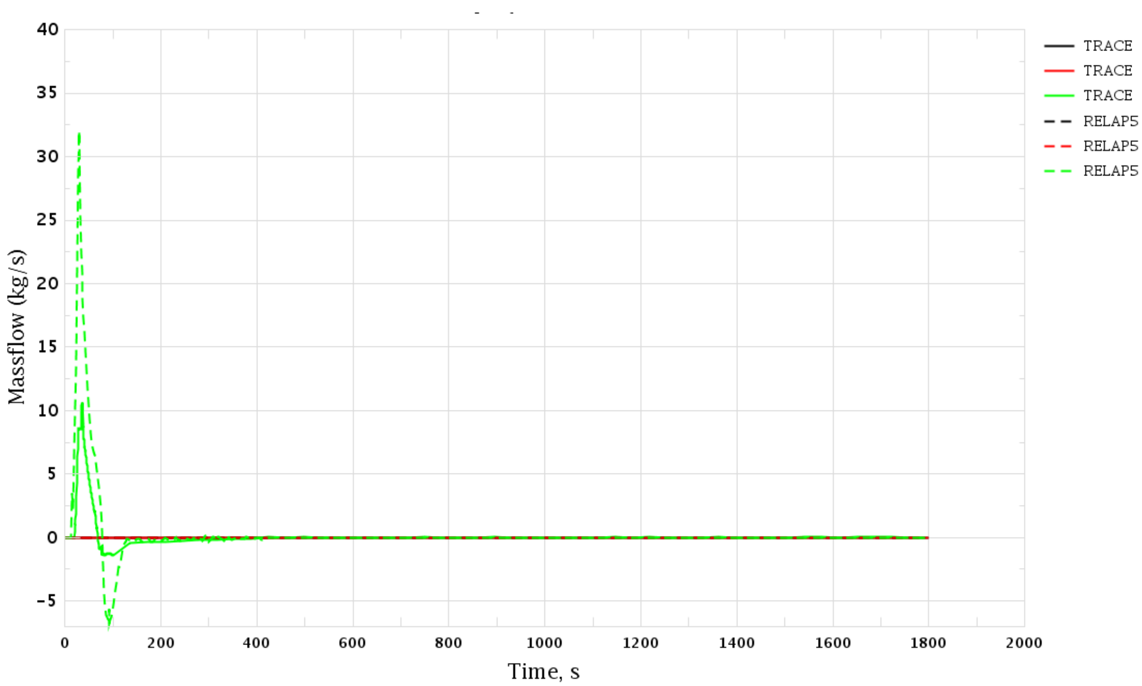

Figure 3-127 Uncontrolled Withdrawal of CR Group. PRZ Spray Mass Flow Rate

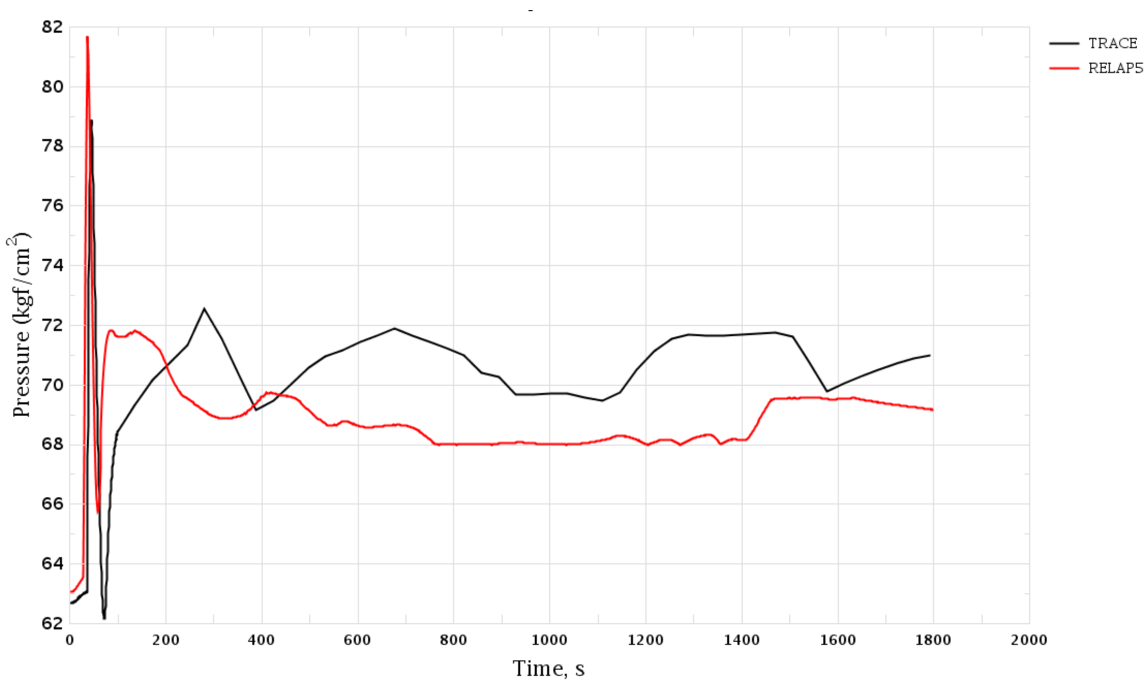

Figure 3-128 Uncontrolled Withdrawal of CR Group. SG-1 Pressure

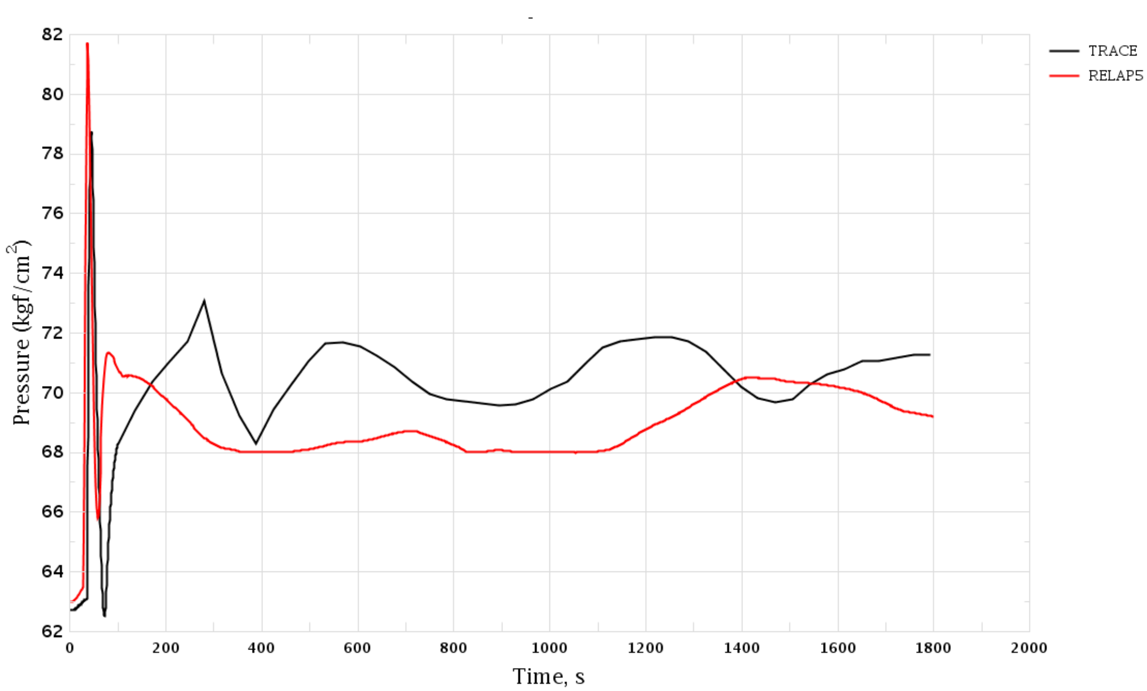

Figure 3-129 Uncontrolled Withdrawal of CR Group. SG-2 Pressure

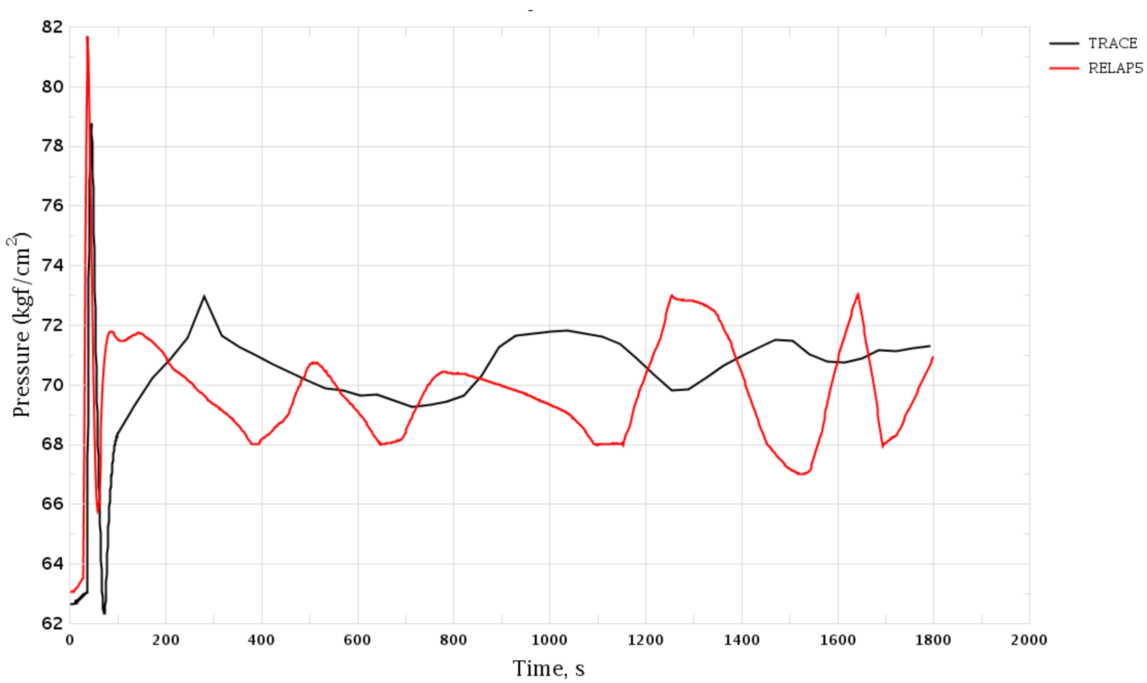

Figure 3-130 Uncontrolled Withdrawal of CR Group. SG-3 Pressure

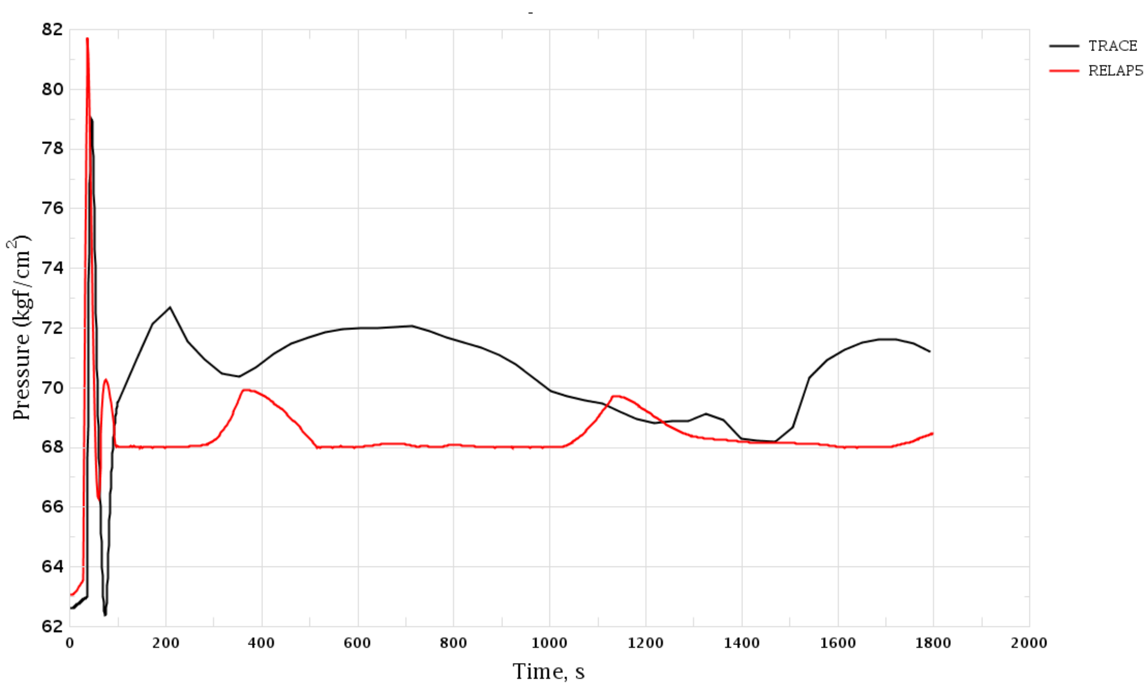

Figure 3-131 Uncontrolled Withdrawal of CR Group. SG-4 Pressure

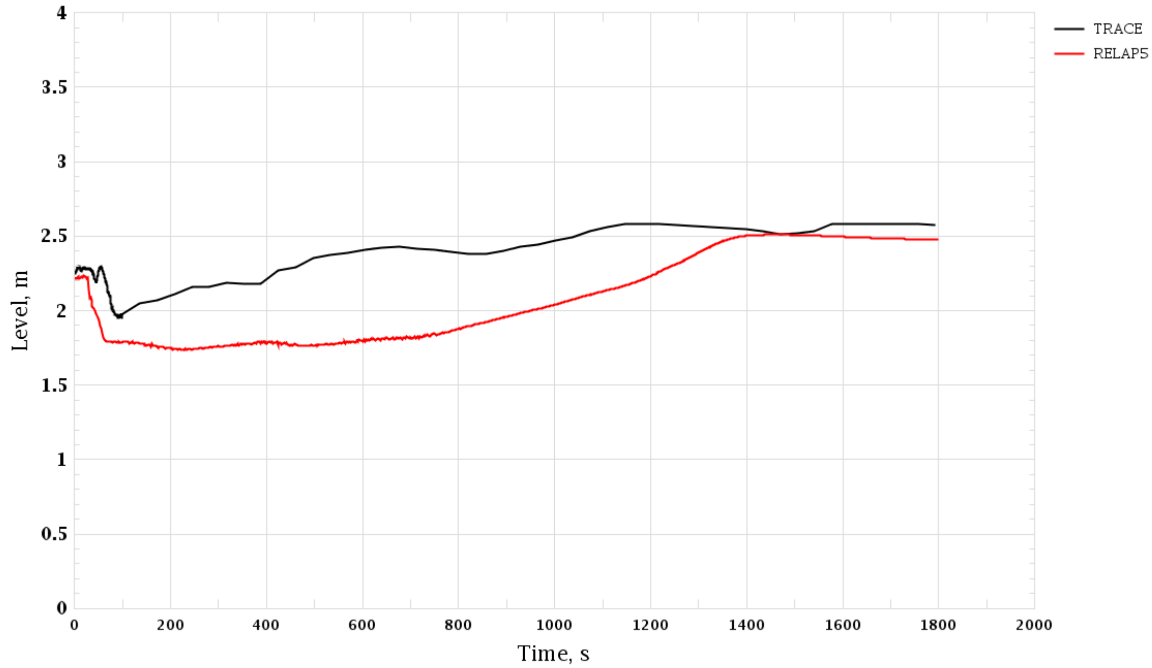

Figure 3-132 Uncontrolled Withdrawal of CR Group. SG-1 Level (Wide Range)

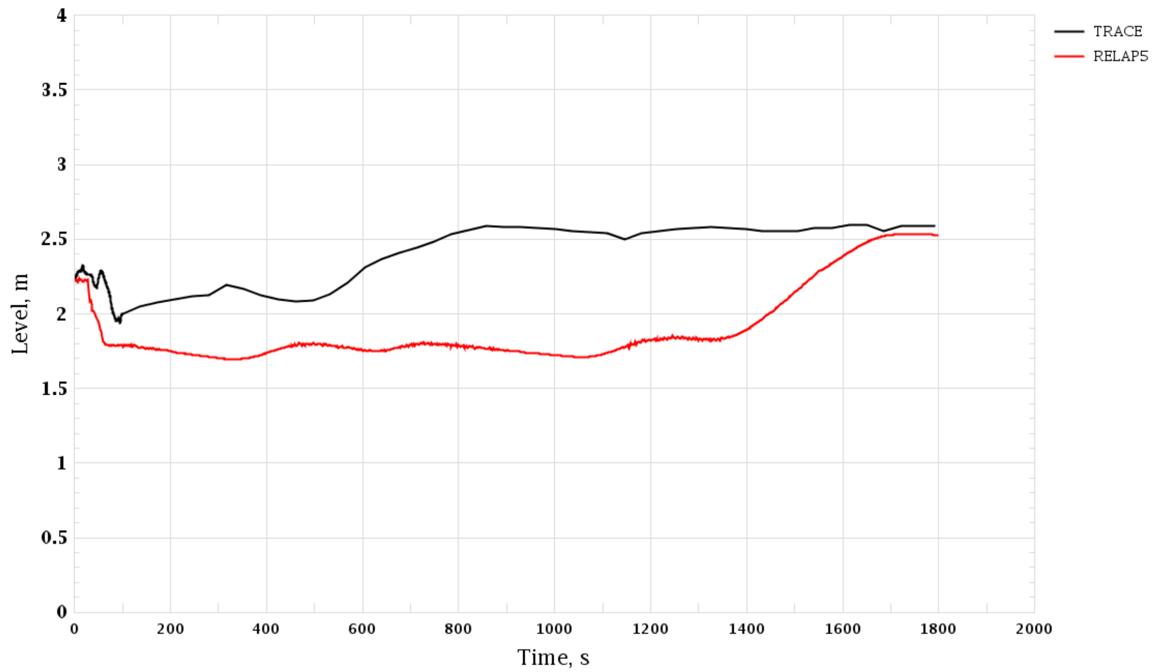

Figure 3-133 Uncontrolled Withdrawal of CR Group. SG-2 Level (Wide Range)

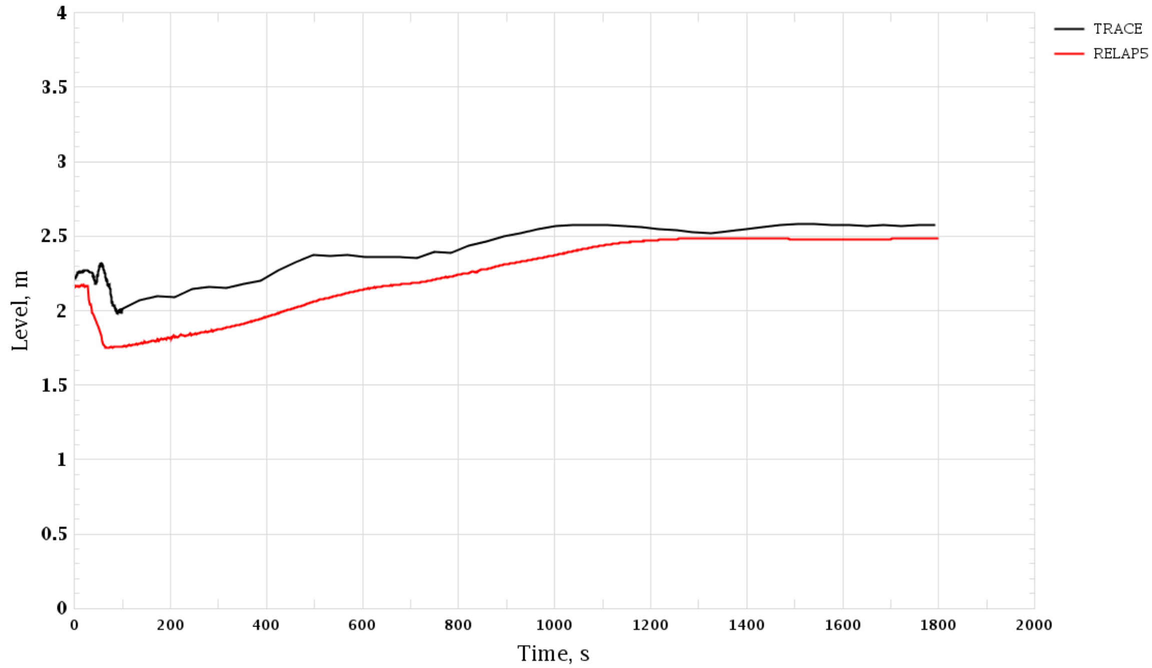

Figure 3-134 Uncontrolled Withdrawal of CR Group. SG-3 Level (Wide Range)

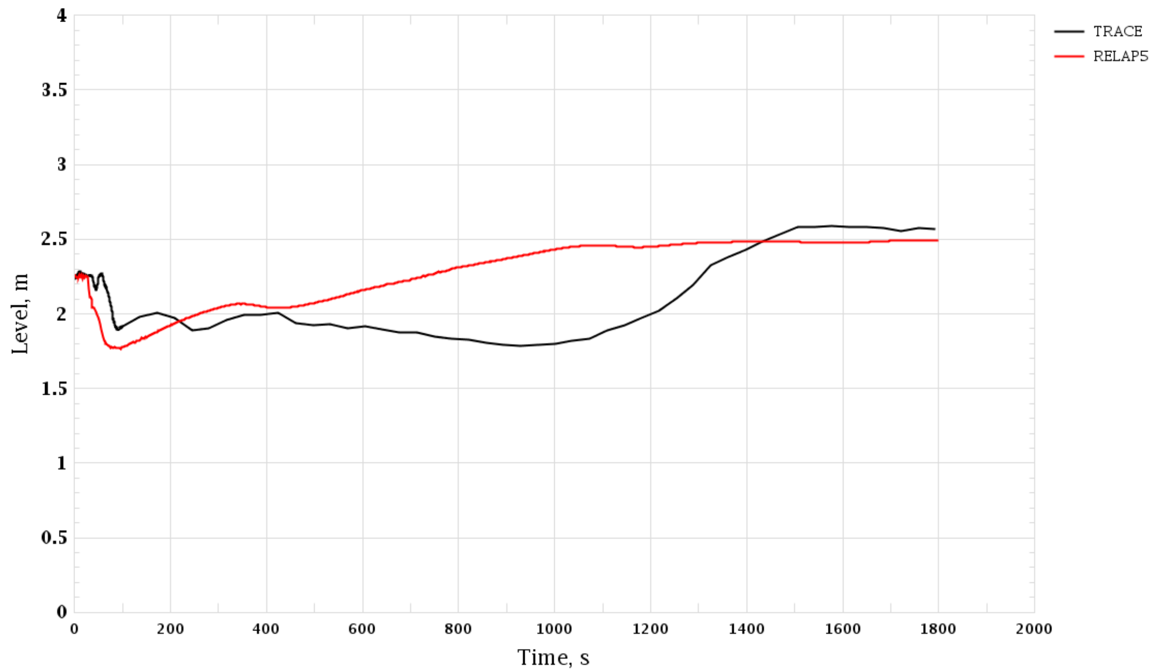

Figure 3-135 Uncontrolled Withdrawal of CR Group. SG-4 Level (Wide Range)

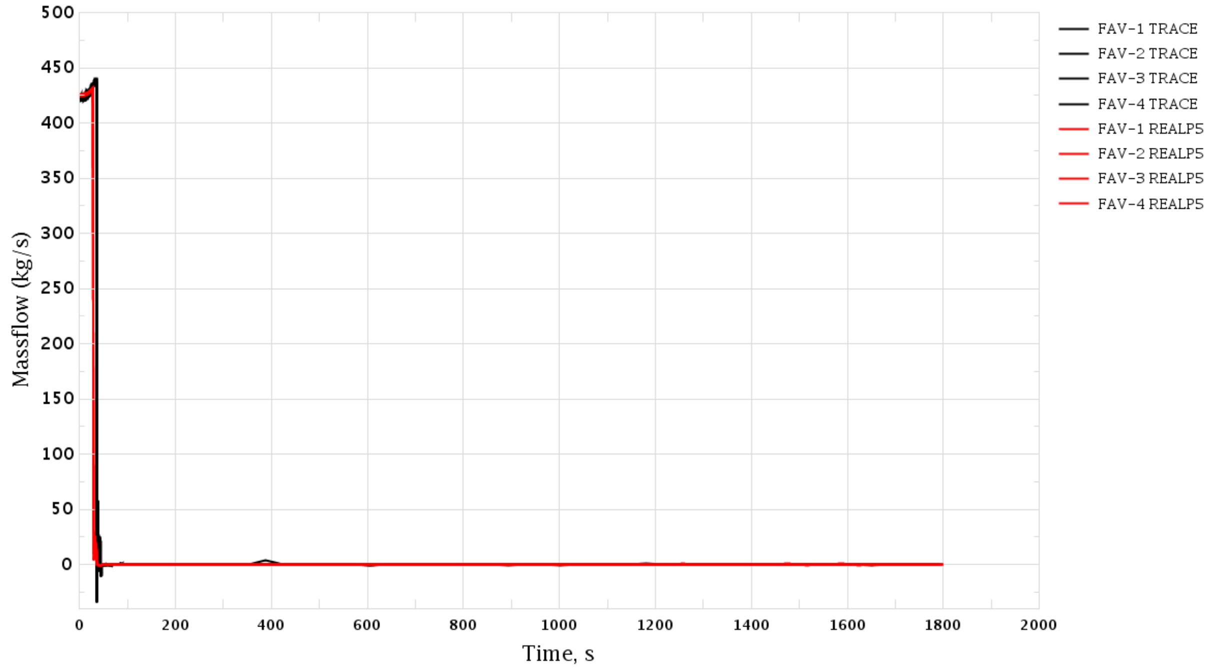

Figure 3-136 Uncontrolled Withdrawal of CR Group. Mass Flow through MSIV

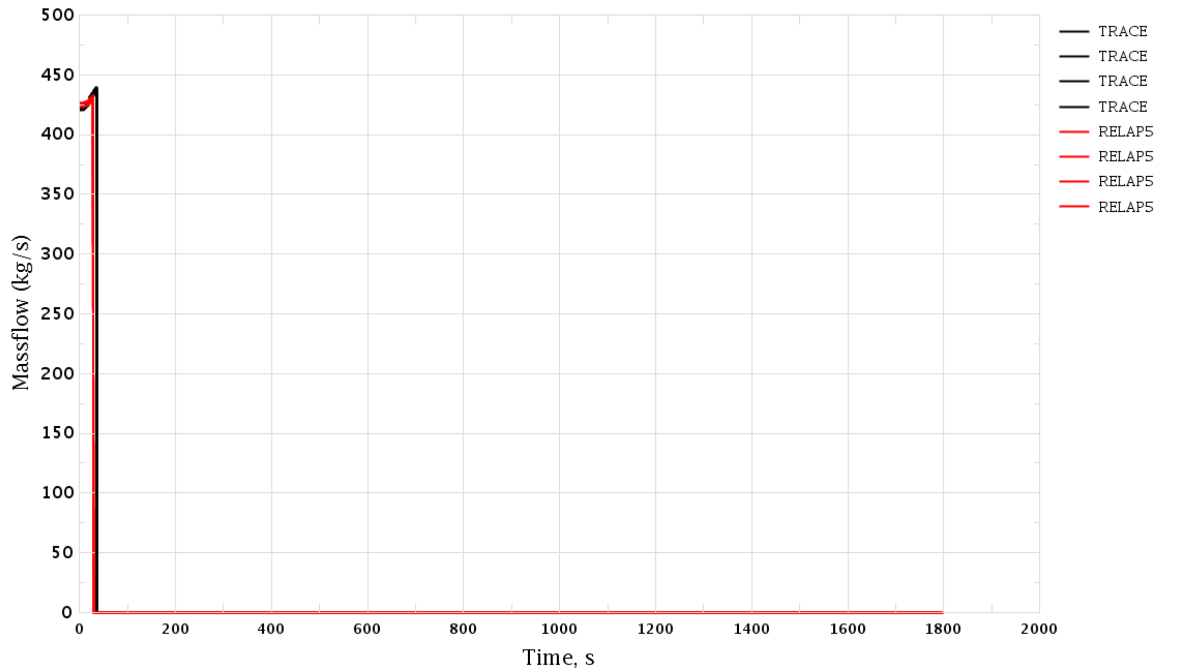

Figure 3-137 Uncontrolled Withdrawal of CR Group. Turbines Mass Flow Rate

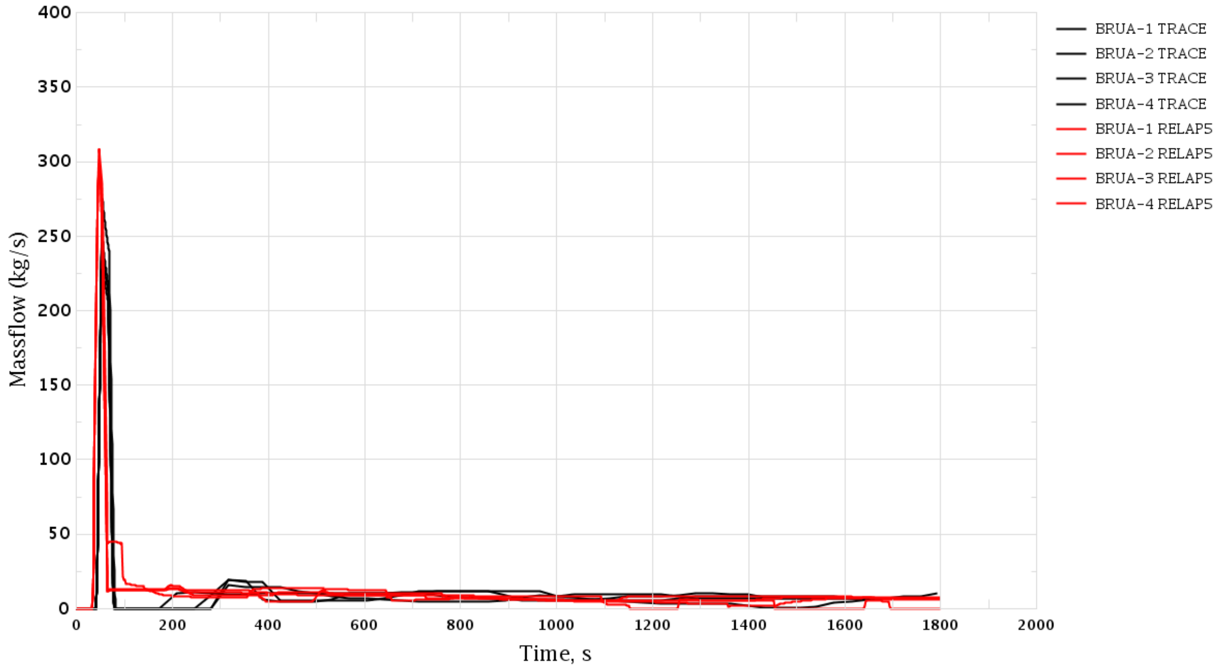

Figure 3-138 Uncontrolled Withdrawal of CR Group. BRU-A Steam Mass Flow Rate

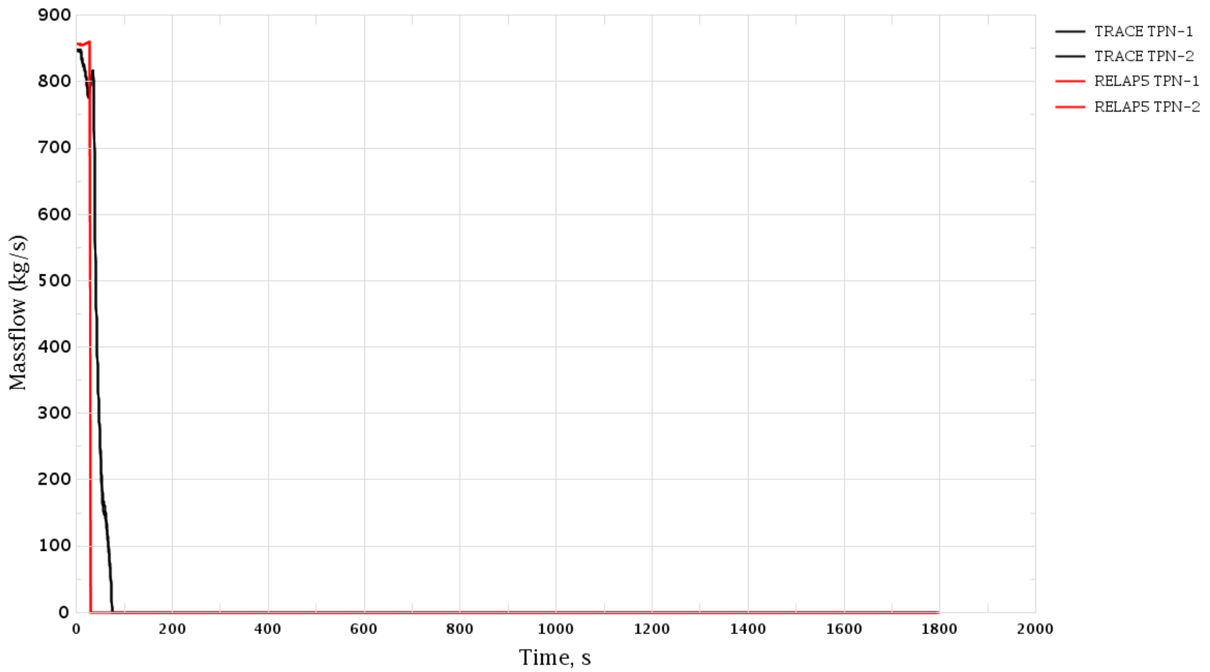

Figure 3-139 Uncontrolled Withdrawal of CR Group. MFW Pumps Mass Flow Rate

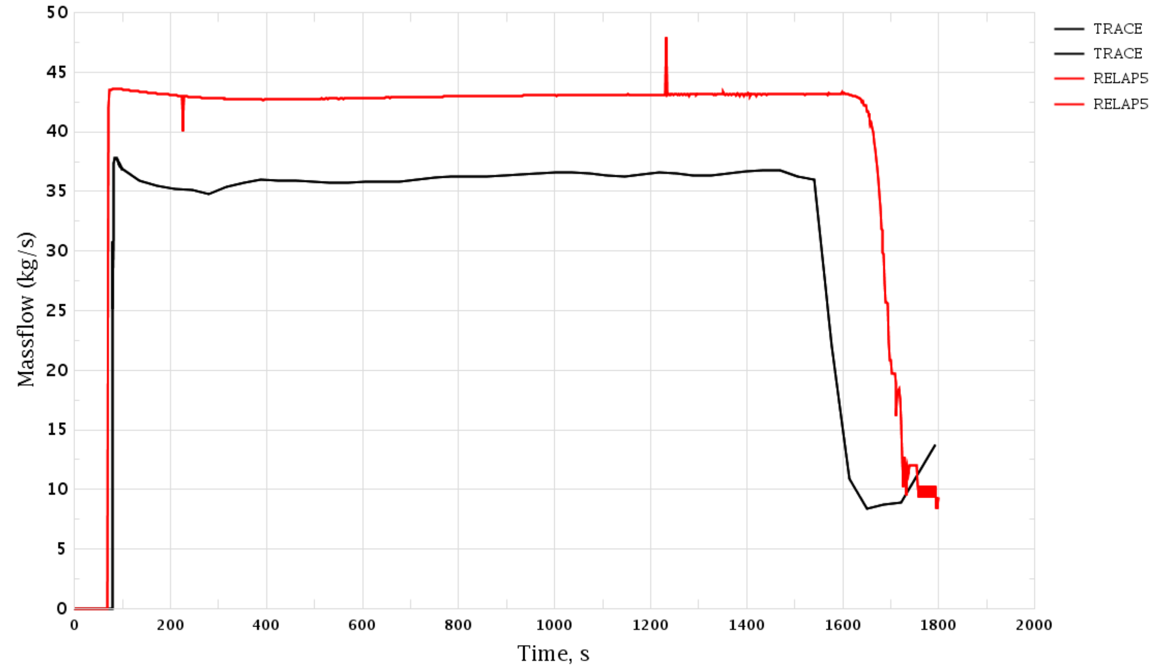

Figure 3-140 Uncontrolled Withdrawal of CR Group. AFW Mass Flow Rate

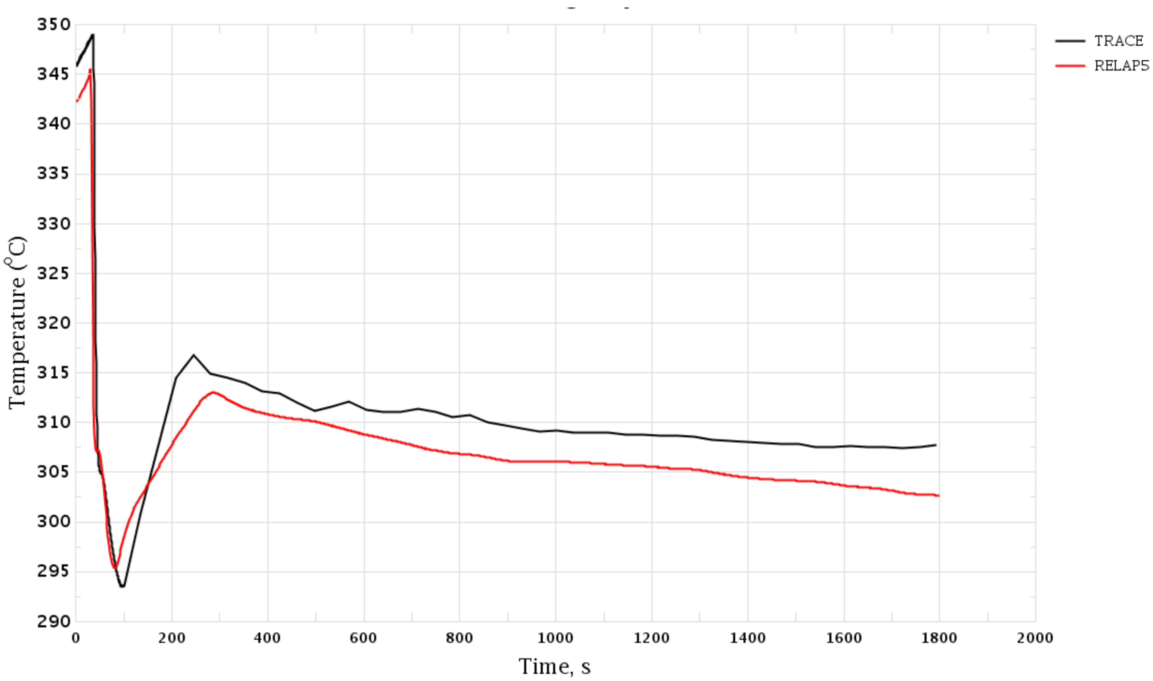

Figure 3-141 Uncontrolled Withdrawal of CR Group. Maximal Cladding Temperature

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 350 mm connecting hot leg of RCS loop no.4 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.

Operation of ARM, RPL, PZ-2, PZ-3, and UPZ is not taken into account. Reactor scram operates according to the design with a delay of 0.3 s from formation of actuation signal to a start of control rods drop. Maximal allowed time of control rods drop to the core (4 s) is assumed.

Simultaneously with formation of reactor scram signal the loss of normal power supply is postulated resulting in a trip of RCPs, turbine and MFW pumps.

Operation of primary circuit make-up and let-down is not taken into account.

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 equivalent diameter of 350 mm)	IE occurrence
0.4	0.8	Reactor scram actuation Loss of normal power supply	RCS pressure decrease below 148.0 kgf/cm^2 at the reactor power greater than 75%
		RCP trip, start of RCS coast-down Trip of turbine and MFW pumps	Postulated loss of normal power supply
2.4	2.8	ECCS safeguard actuation, signal to start emergency DG	Postulated loss of normal power supply + 2 s delay
6.0	5.9	Signal to open BRU-A	Pressure increase in relevant MSL over 73 kgf/cm^2
24.4	24.8	Start of emergency DG sequential loading program, HPIS and LPIS pumps start	22 s delay for DG start-up after ECCS safeguard actuation
42.4	42.8	Start of HPIS injection to the primary circuit	SLP (after DG start-up) + transportation delay of 18 s
45.0	43.0	BRU-A closure	Pressure decrease in relevant MSL below 68 kgf/cm^2
50.0	65.0	Start of ECCS HA injection to the reactor	Decrease of RCS pressure below 60.0 kgf/cm^2
95.0	145.0	Start of stable LPIS injection to RCS	RCS pressure decrease below LPIS pump shut-off head
350.0	260.0	End of ECCS HA injection	HA depletion
1200.0	1200.0	End of calculation	Stable core cooling, stabilization of the main reactor parameters

Break of PRZ surge line with equivalent diameter of 350 mm results in large break LOCA that causes fast decrease of RCS pressure (Figure 3-143) and PRZ level (Figure 3-144). After IE occurrence, due to RCS pressure decrease down to 148.0 kgf/cm^2 at the reactor power greater than 75%, a scram signal is generated at 0.22/0.26 s (in RELAP5 and TRACE calculations, respectively), and core thermal power is quickly reduced to the decay heat level (Figure 3-142).

Postulated loss of normal power supply results in a trip of the equipment powered from correspondent busbars, namely, RCP, PRZ heaters, BRU-K, and causes actuation of ECCS safeguard with a start of emergency DG. Turbine trip causes temporary increase of SG pressure (Figure 3-160 – Figure 3-163) with a maximal value of 77.5/74.2 kgf/cm^2 reached at

8/9 s. At 5.9/6.0 s after IE occurrence BRU-A actuation setpoints (increase of pressure in correspondent MSL over 73 kgf/cm^2) are reached (Figure 3-169). Their operation and intensive energy loss via the break (and, consequently, cooling of SG by the primary circuit) result in a decrease of SG pressure below 68 kgf/cm^2 and closure of all BRU-As at 43/45 s of accident.

Nearly at the same time (at 65/50 s of accident) RCS pressure decreases below 60 kgf/cm^2 and HA injection to the reactor begins and continues to their depletion at 260/350 s (Figure 3-177 – Figure 3-180).

22 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 42.8/42.4 s after IE occurrence (with postulated 18 s delay for HPIS pump start-up and transport) HPIS starts to inject boric acid to the primary circuit (Figure 3-171 – Figure 3-173).

At 145.0/95.0 s of the accident RCS pressure decreases below LPIS pump shut-off head and LPIS injection to the primary circuit is started (Figure 3-174 – Figure 3-176). Combined operation of HPIS and LPIS pumps compensates coolant loss via the break and to 1000 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 ~100 °C is observed (Figure 3-181).

The calculation is ended at 1200 s of accident.

The plots of the main parameters of calculation are presented below on Figure 3-142 – Figure 3-183.

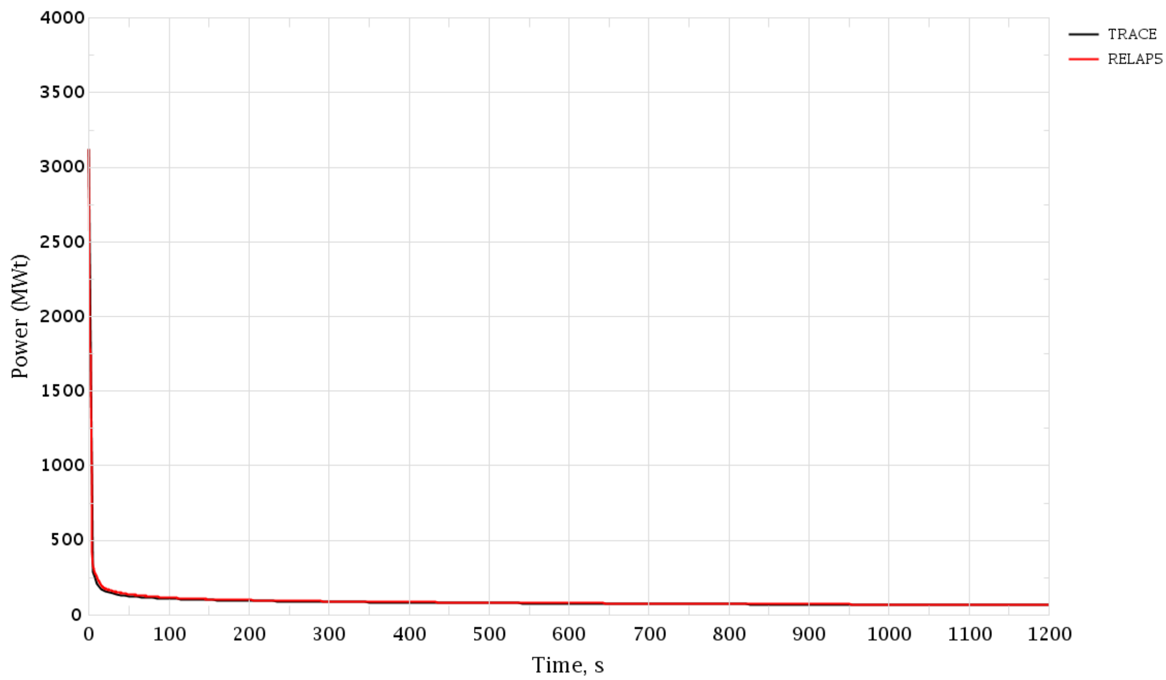

Figure 3-142 PRZ Surge Line Break. Core Thermal Power

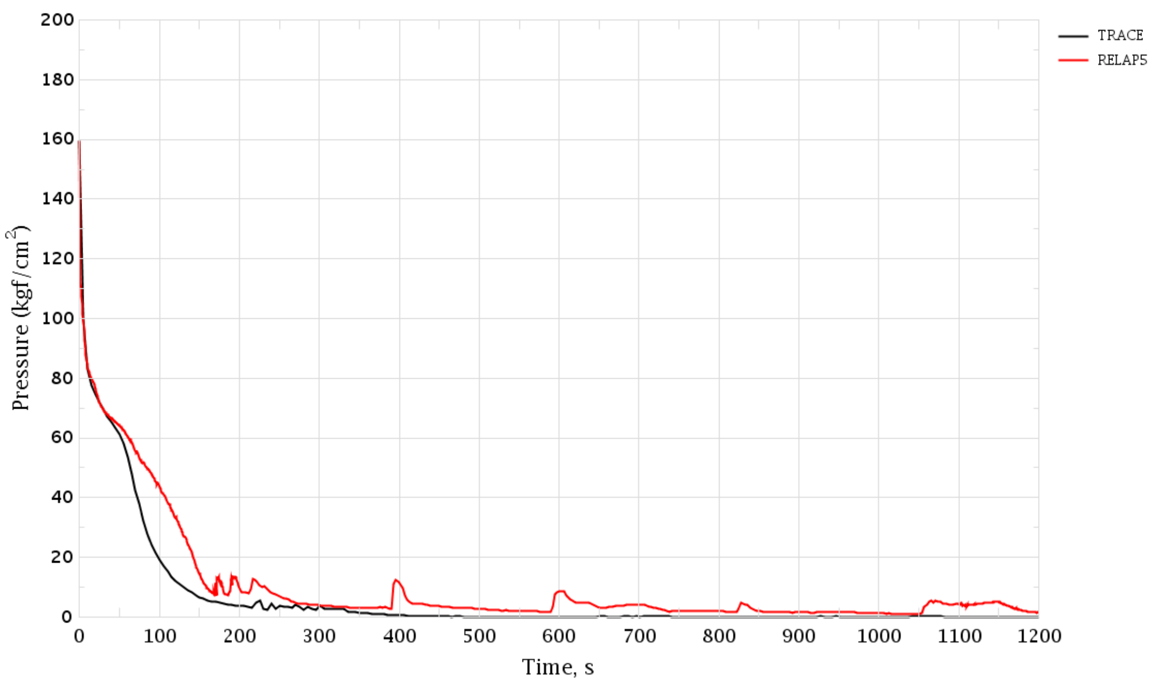

Figure 3-143 PRZ Surge Line Break. RCS Pressure

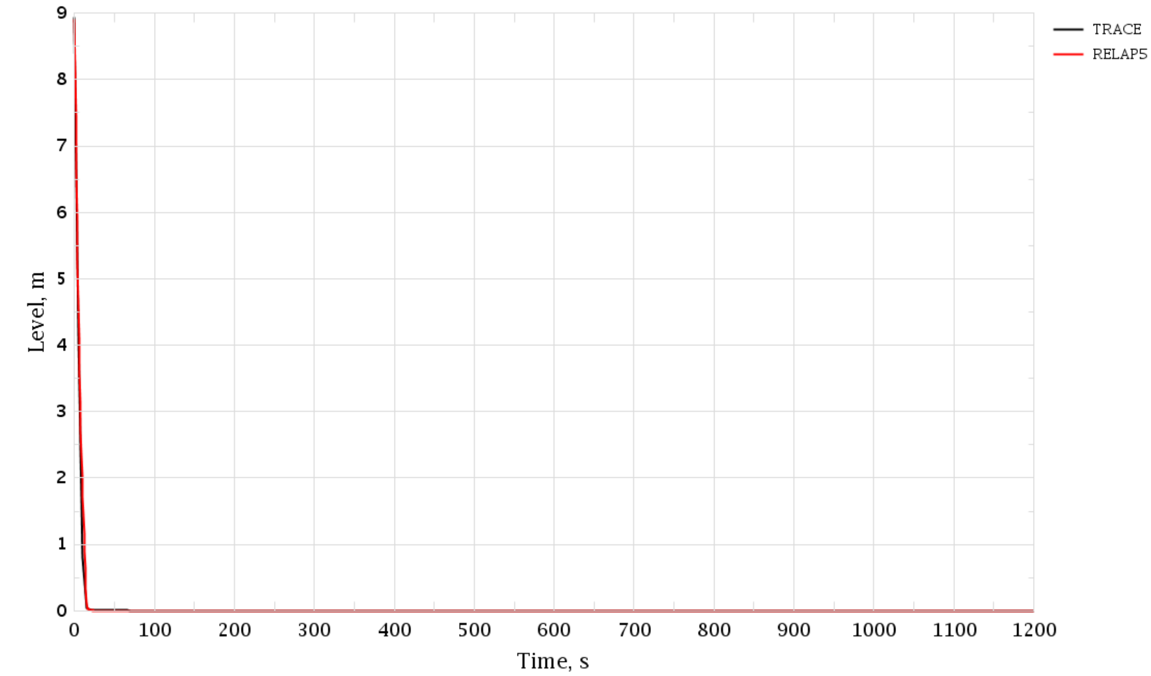

Figure 3-144 PRZ Surge Line Break. Pressurizer Level

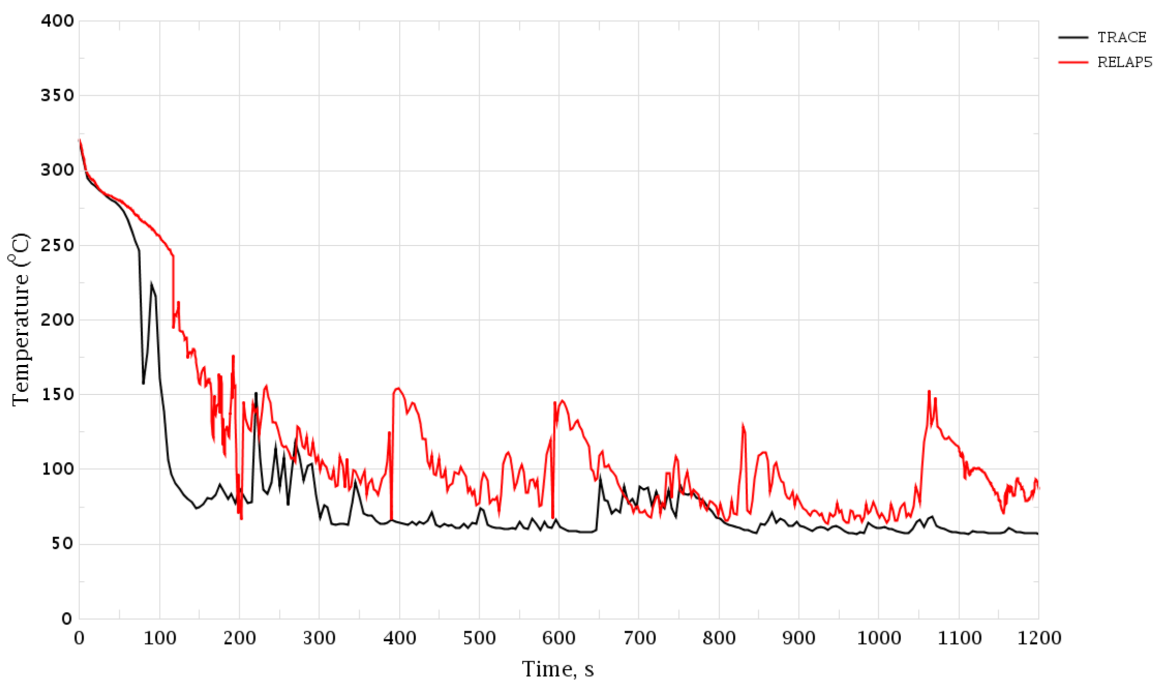

Figure 3-145 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 1

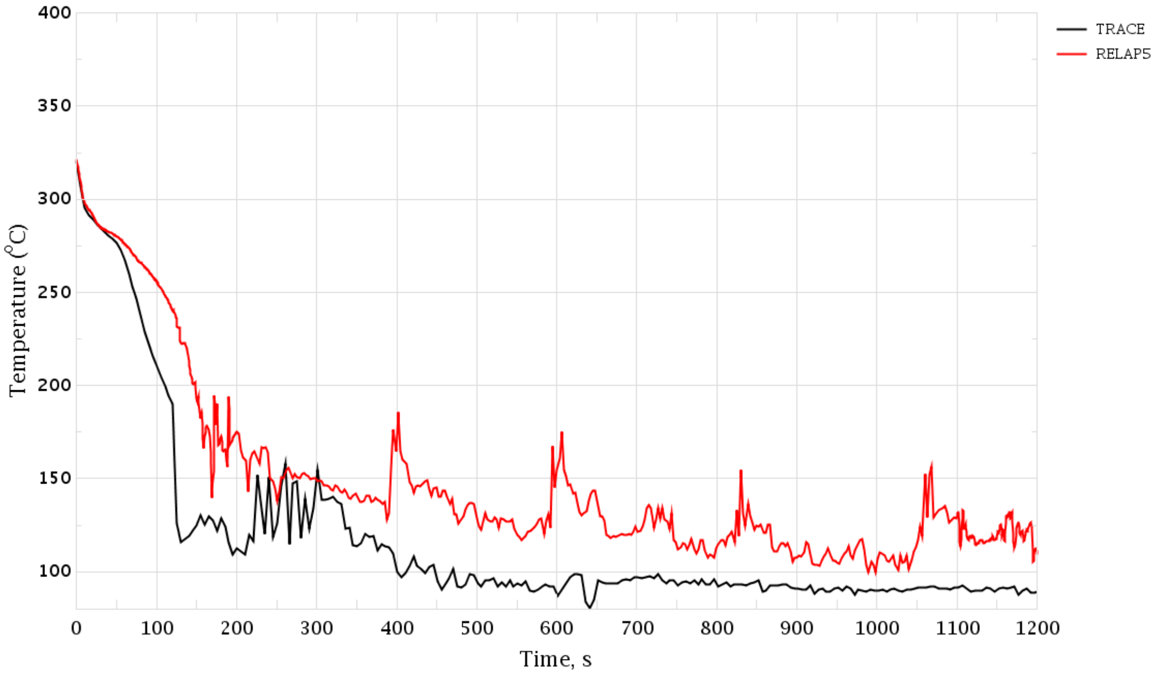

Figure 3-146 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 2

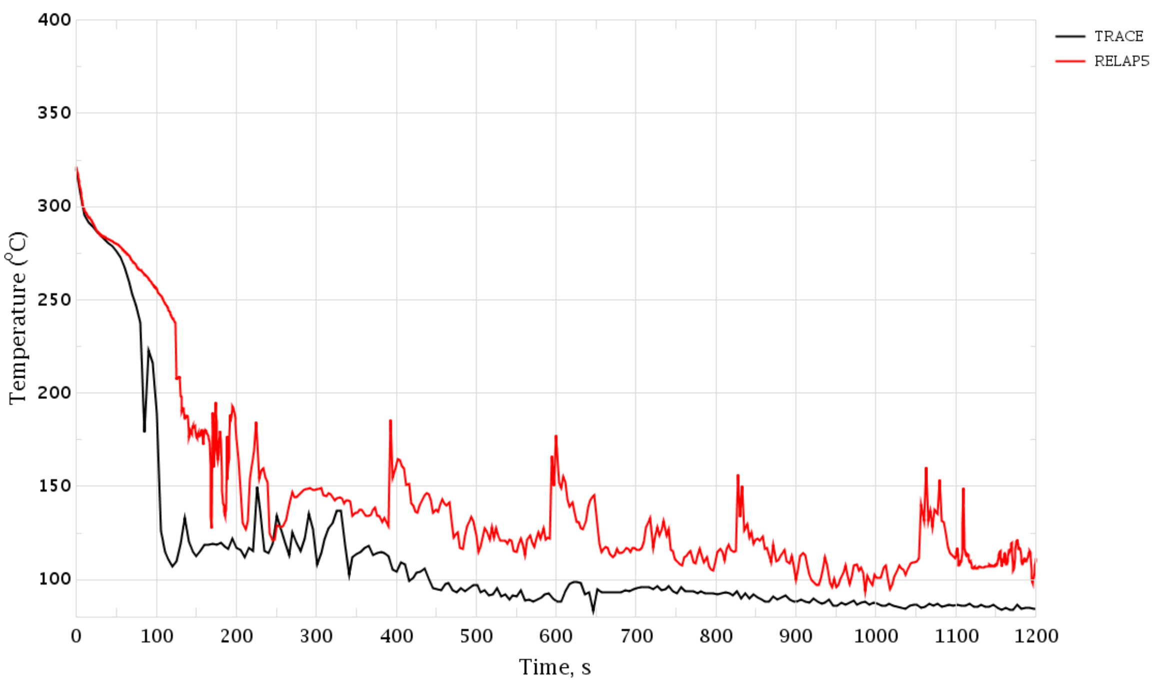

Figure 3-147 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 3

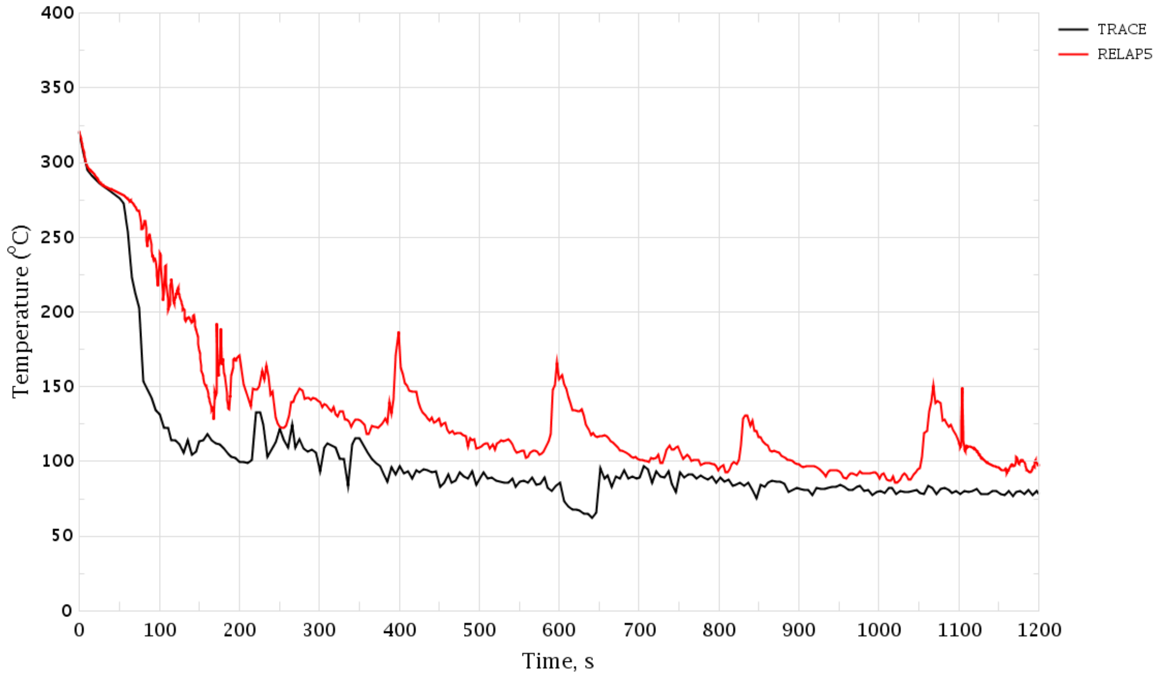

Figure 3-148 PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 4

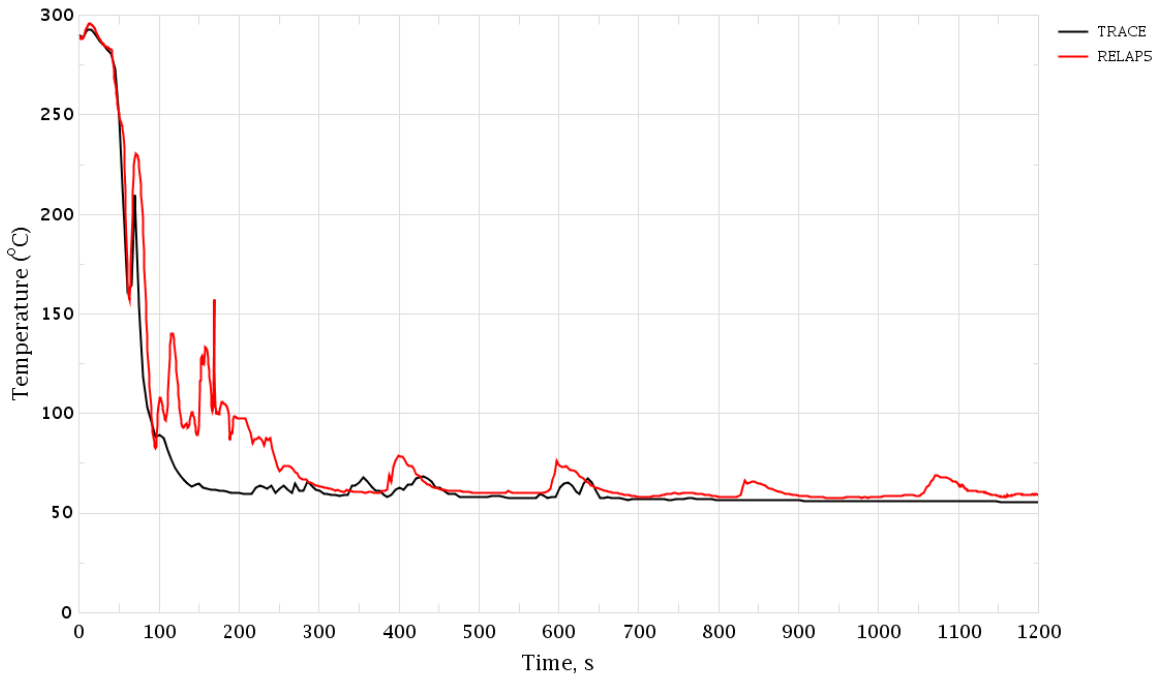

Figure 3-149 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop 1

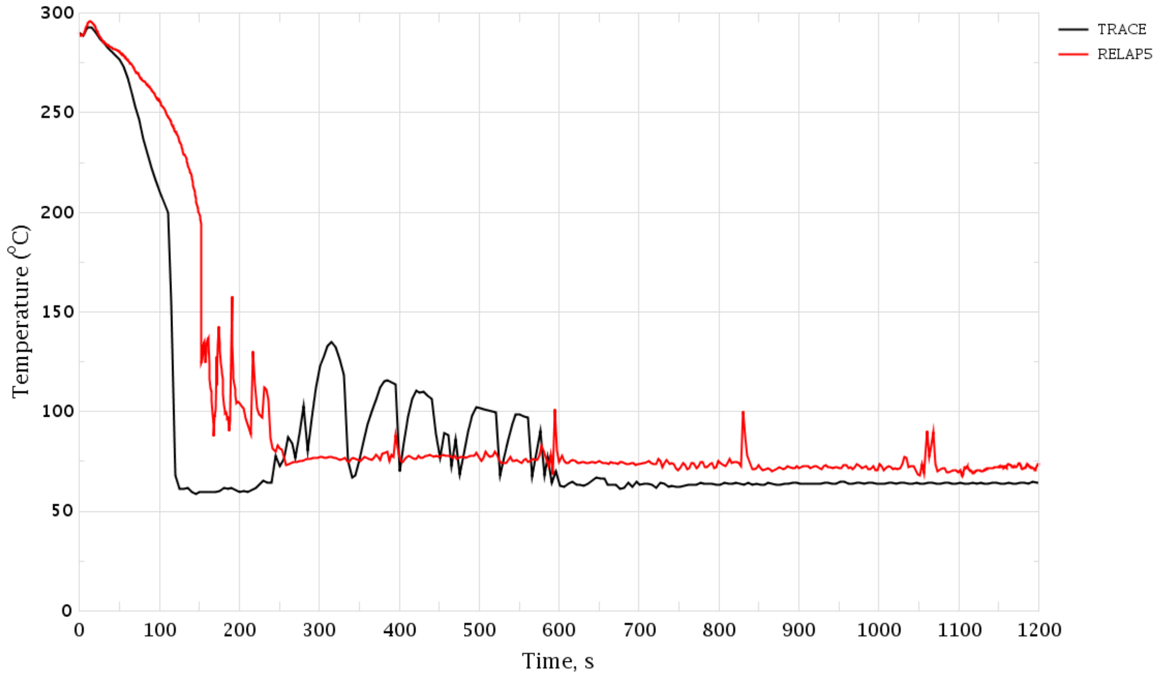

Figure 3-150 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop 2

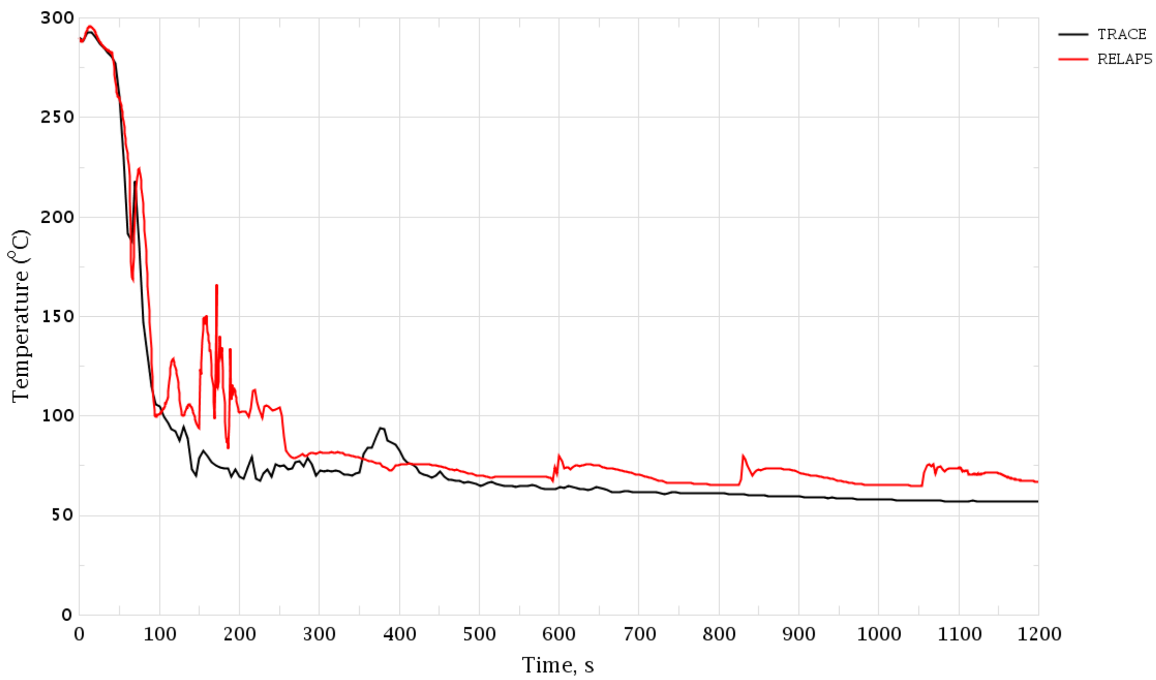

Figure 3-151 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop 3

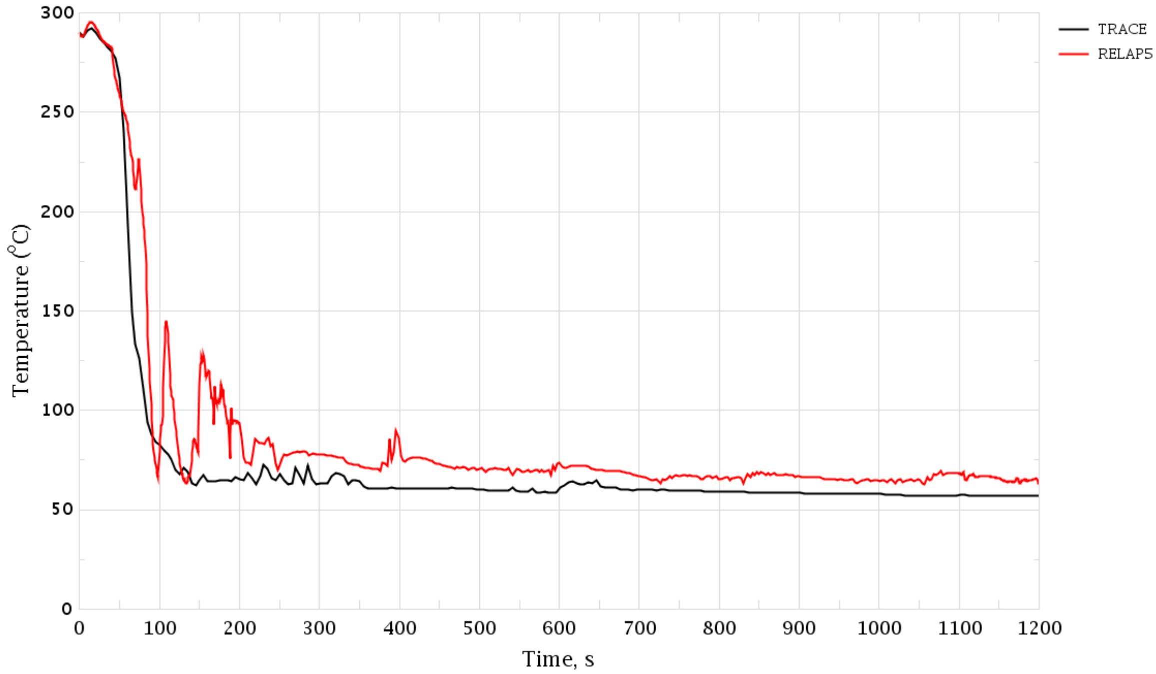

Figure 3-152 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop 4

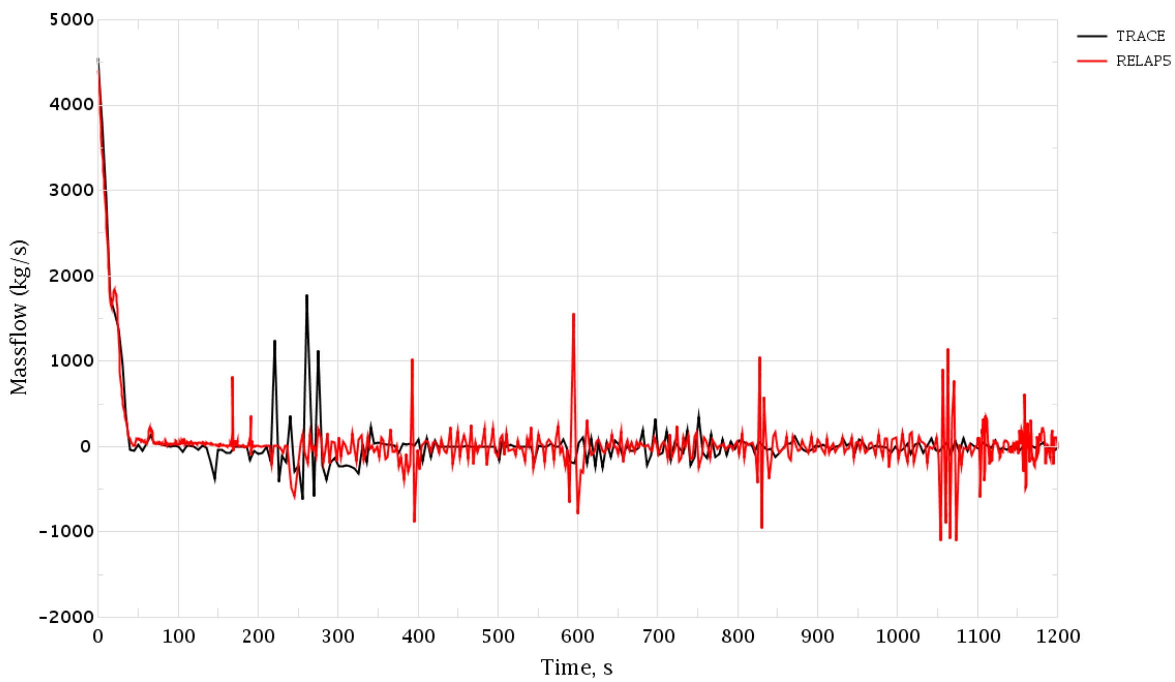

Figure 3-153 PRZ Surge Line Break. RCS Loop 1 Mass Flow Rate

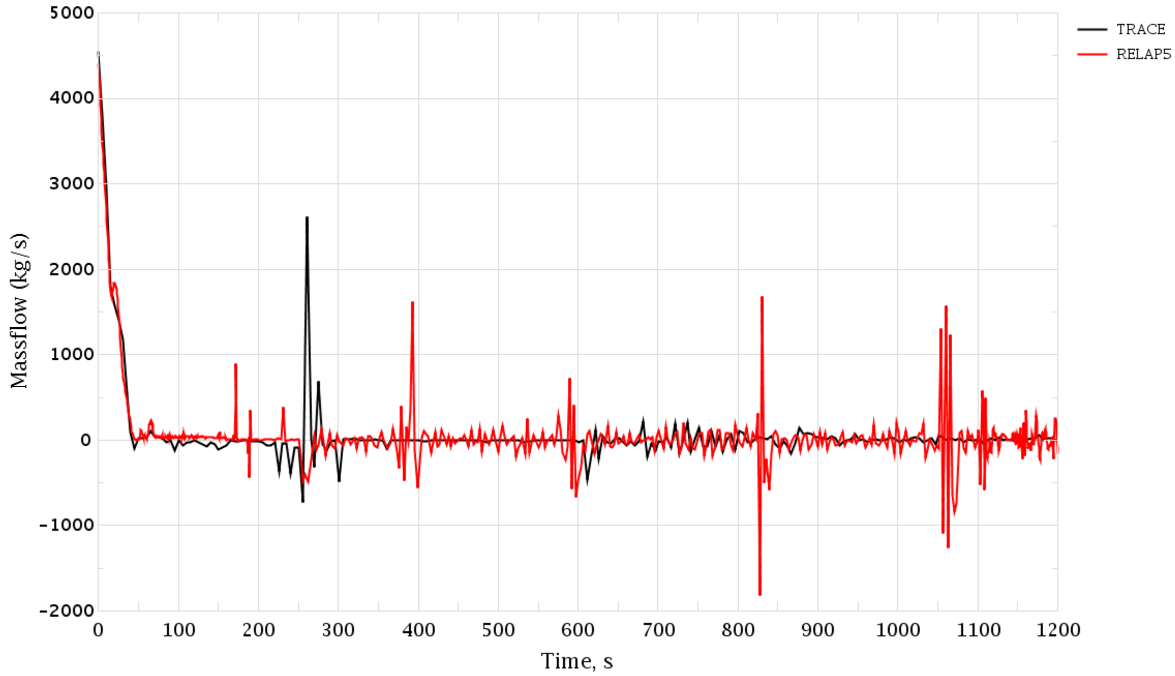

Figure 3-154 PRZ Surge Line Break. RCS Loop 2 Mass Flow Rate

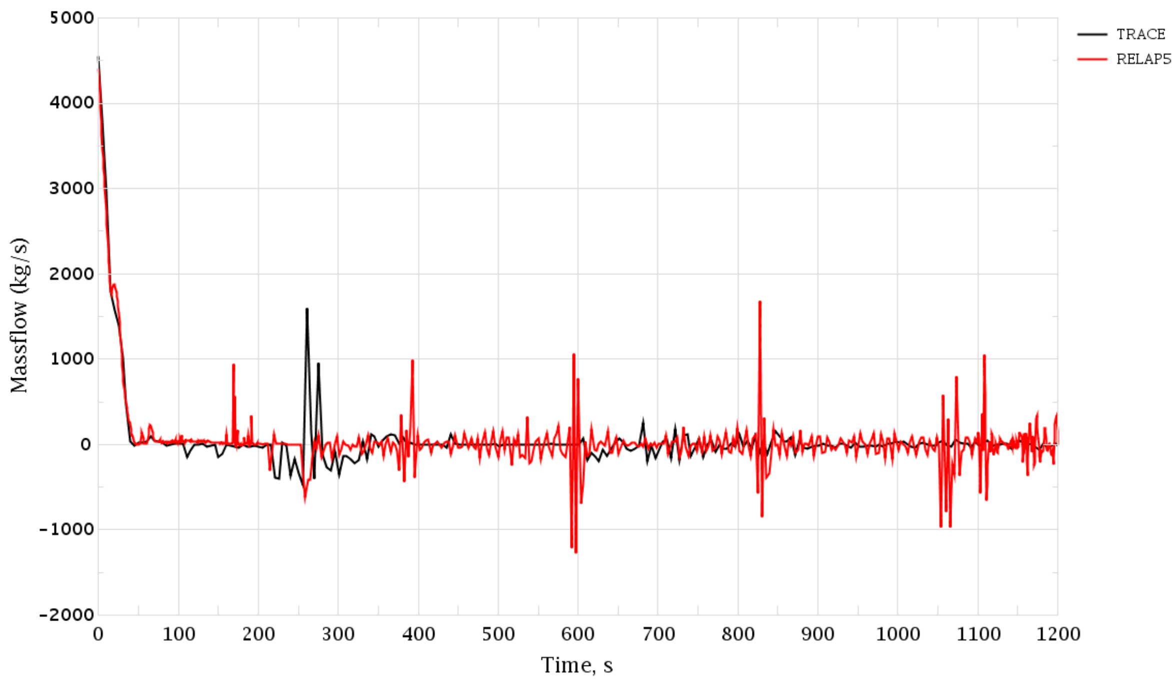

Figure 3-155 PRZ Surge Line Break. RCS Loop 3 Mass Flow Rate

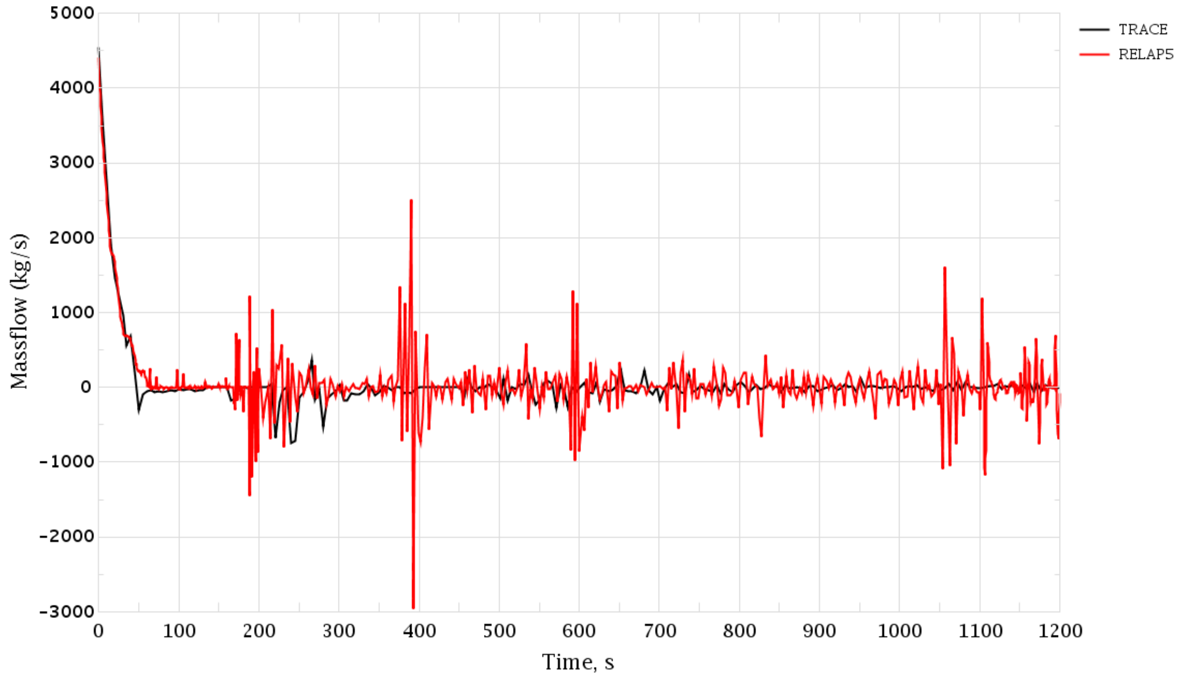

Figure 3-156 PRZ Surge Line Break. RCS Loop 4 Mass Flow Rate

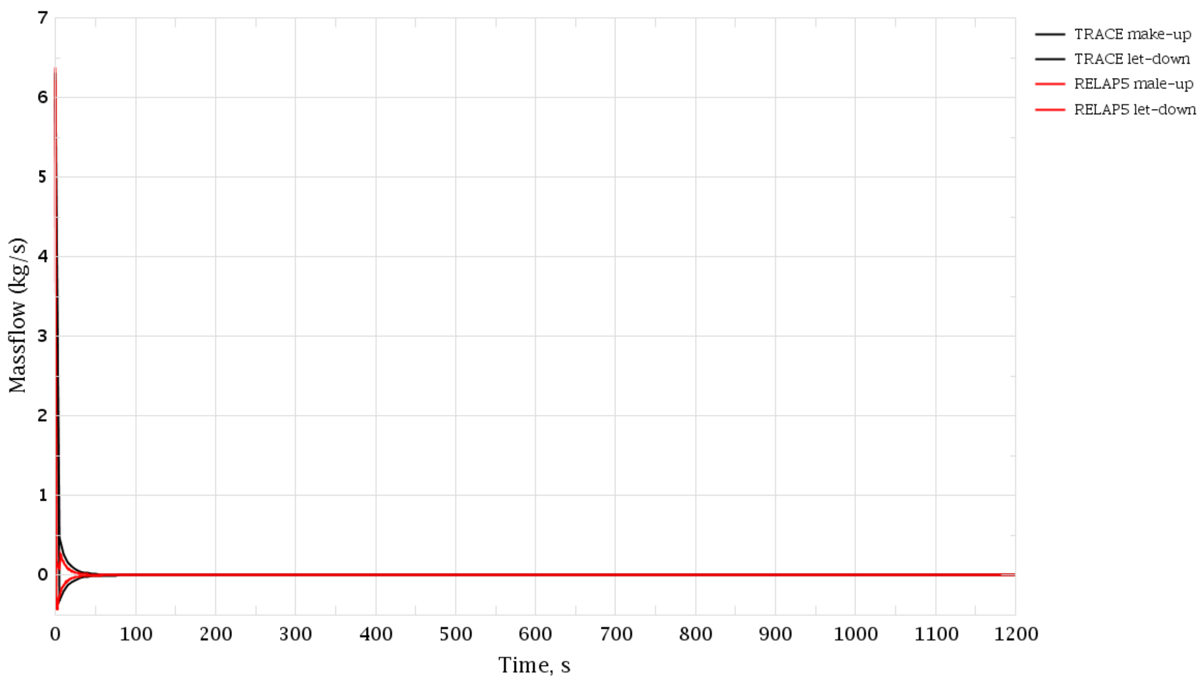

Figure 3-157 PRZ Surge Line Break. Make-Up and Let-Down Mass Flow

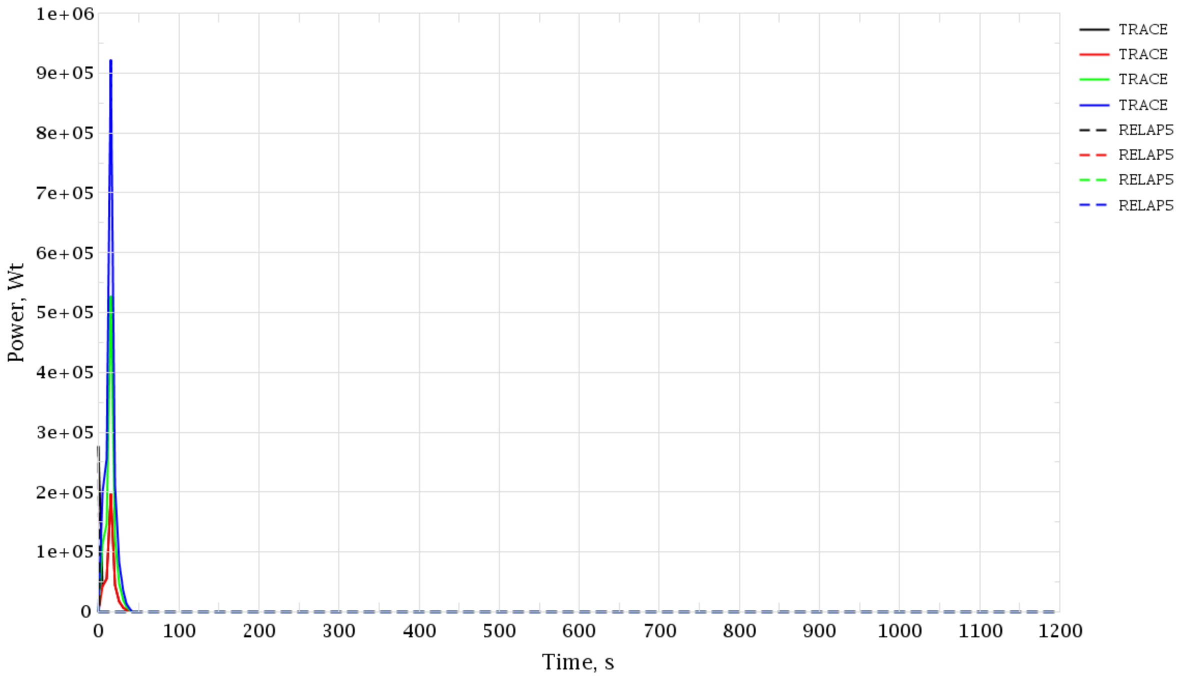

Figure 3-158 PRZ Surge Line Break. PRZ Heaters Power

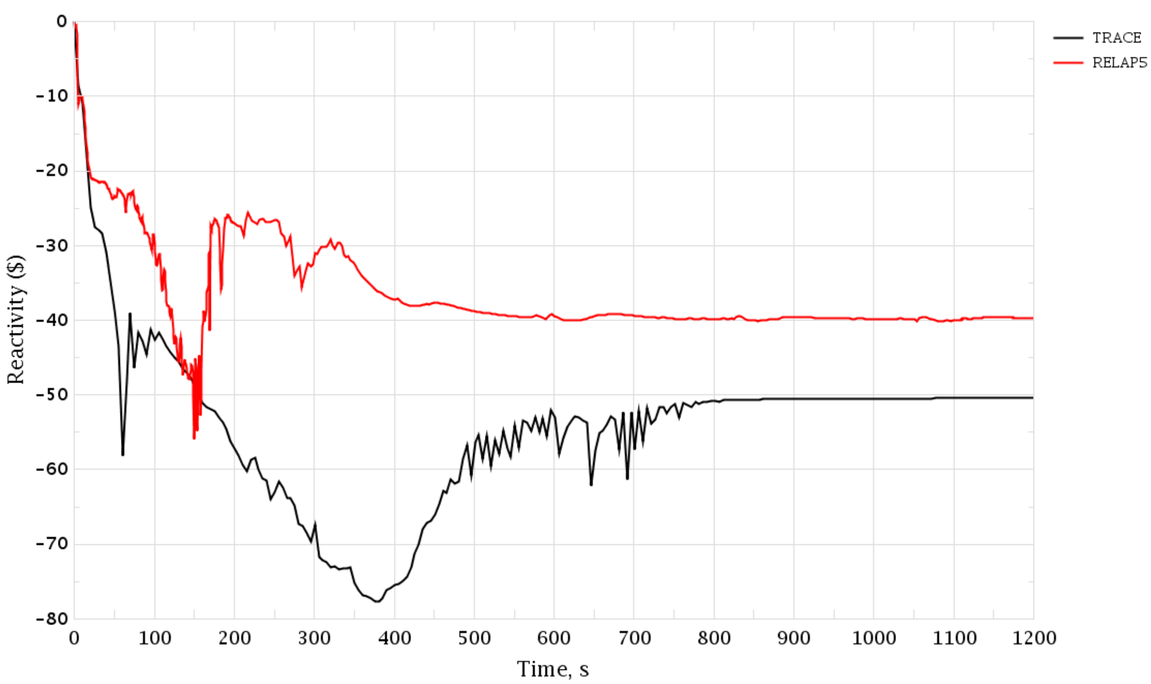

Figure 3-159 PRZ Surge Line Break. Core Reactivity

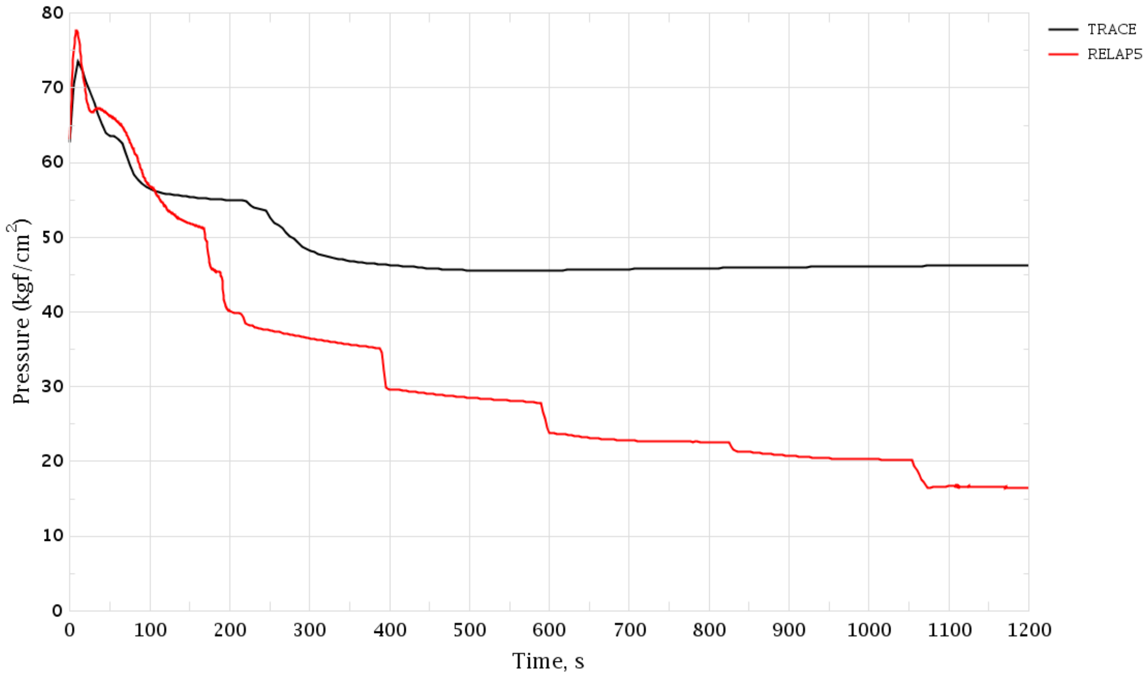

Figure 3-160 PRZ Surge Line Break. SG-1 Pressure

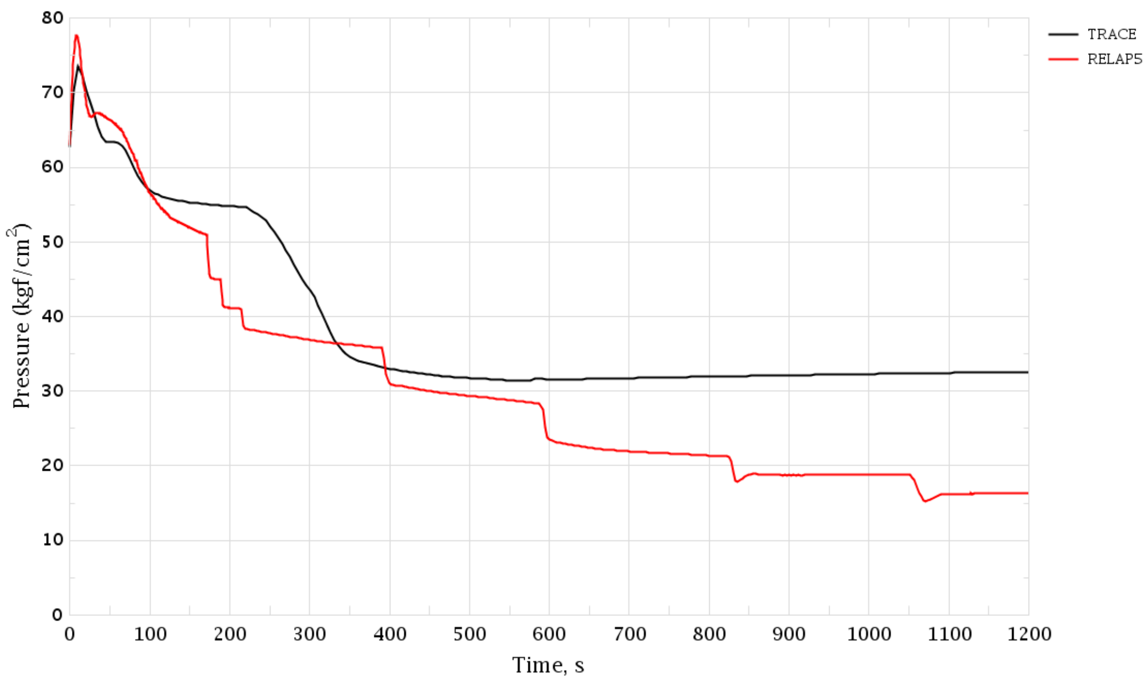

Figure 3-161 PRZ Surge Line Break. SG-2 Pressure

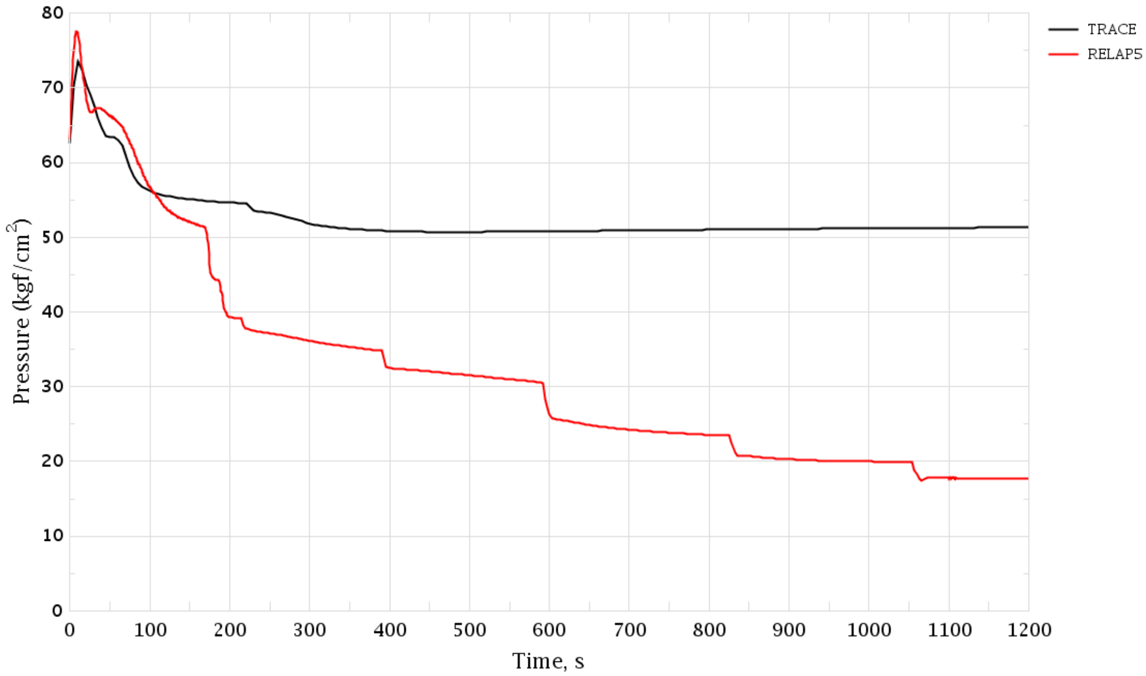

Figure 3-162 PRZ Surge Line Break. SG-3 Pressure

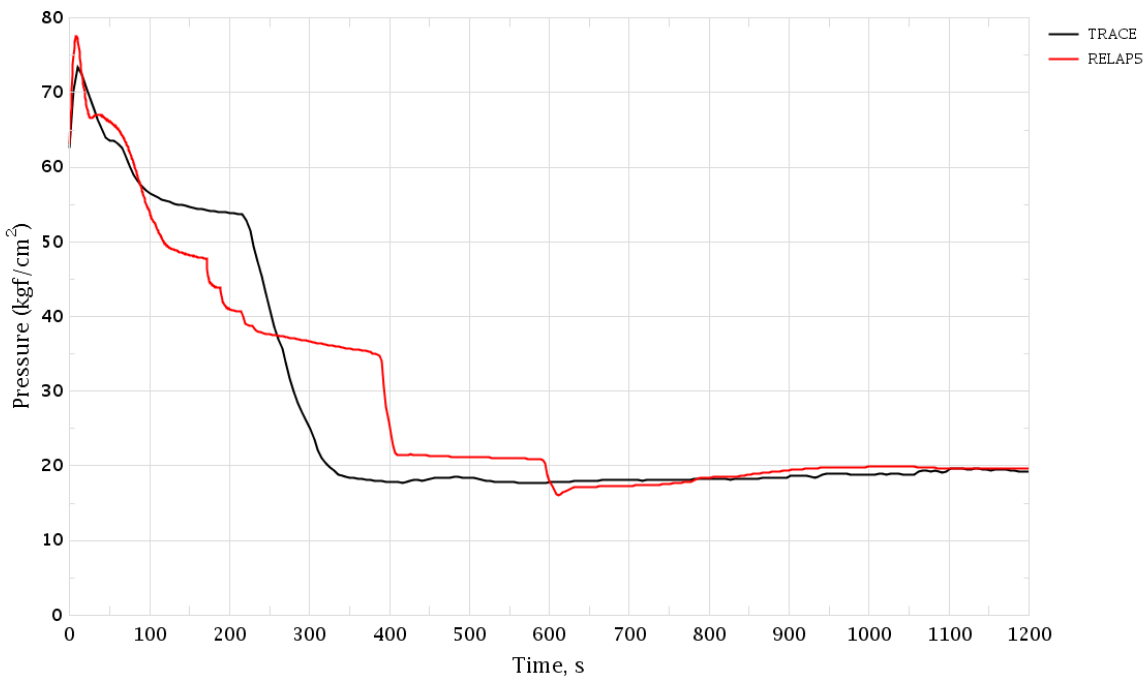

Figure 3-163 PRZ Surge Line Break. SG-4 Pressure

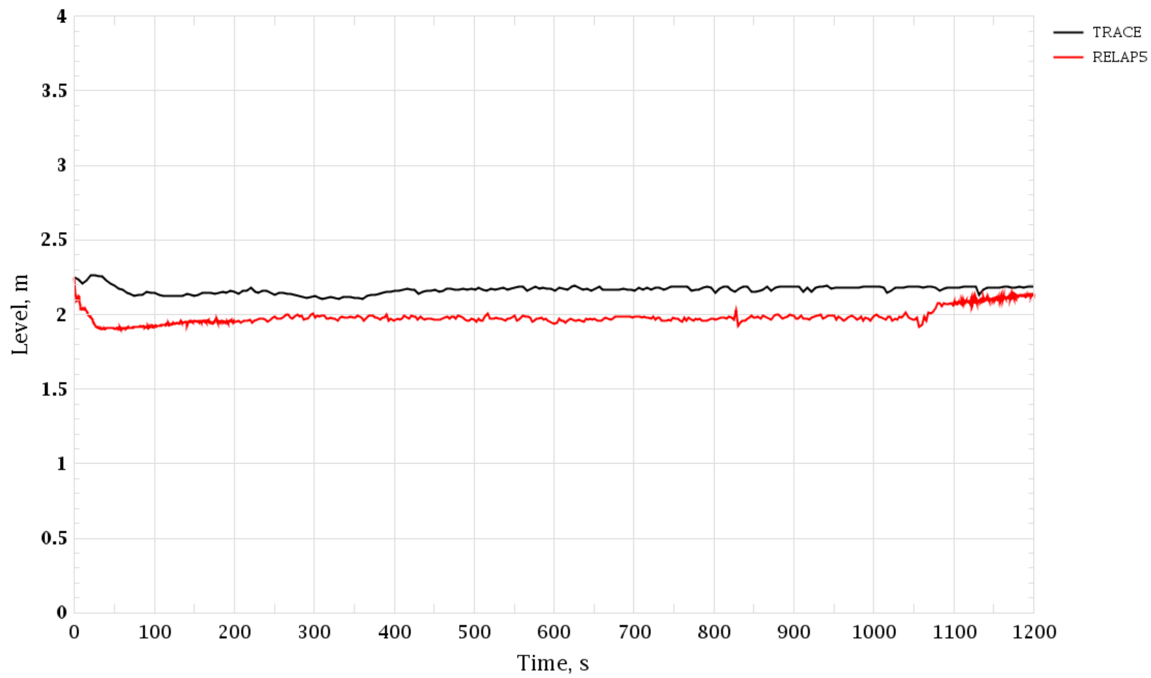

Figure 3-164 PRZ Surge Line Break. SG-1 Level (Wide Range)

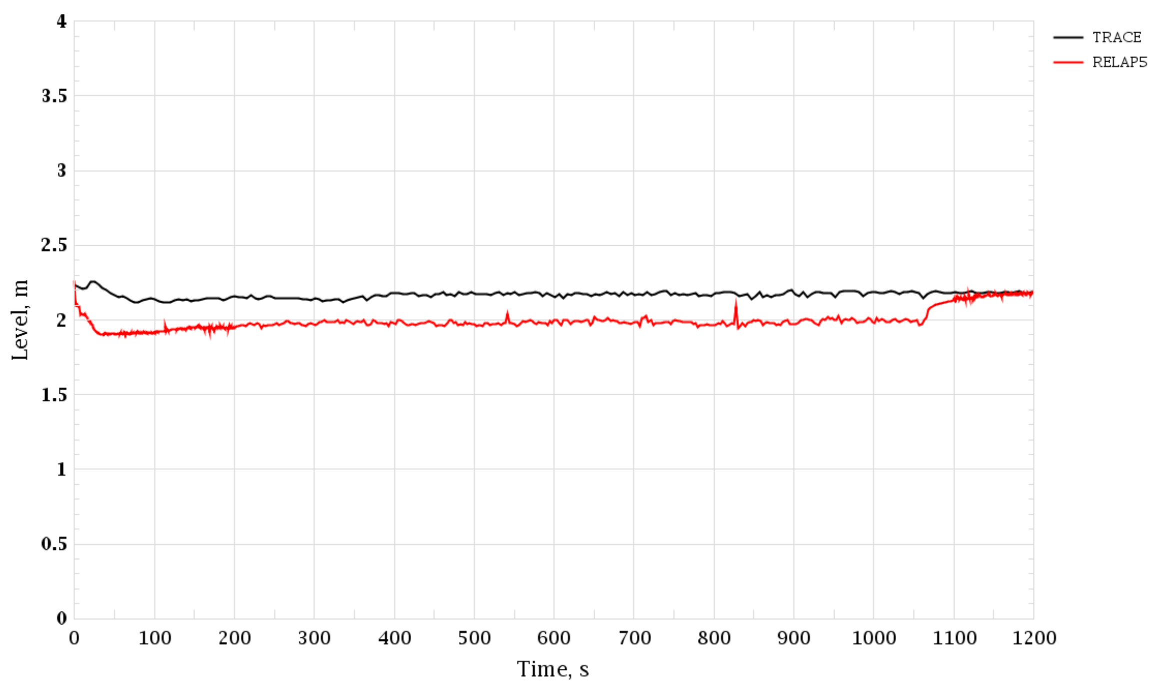

Figure 3-165 PRZ Surge Line Break. SG-2 Level (Wide Range)

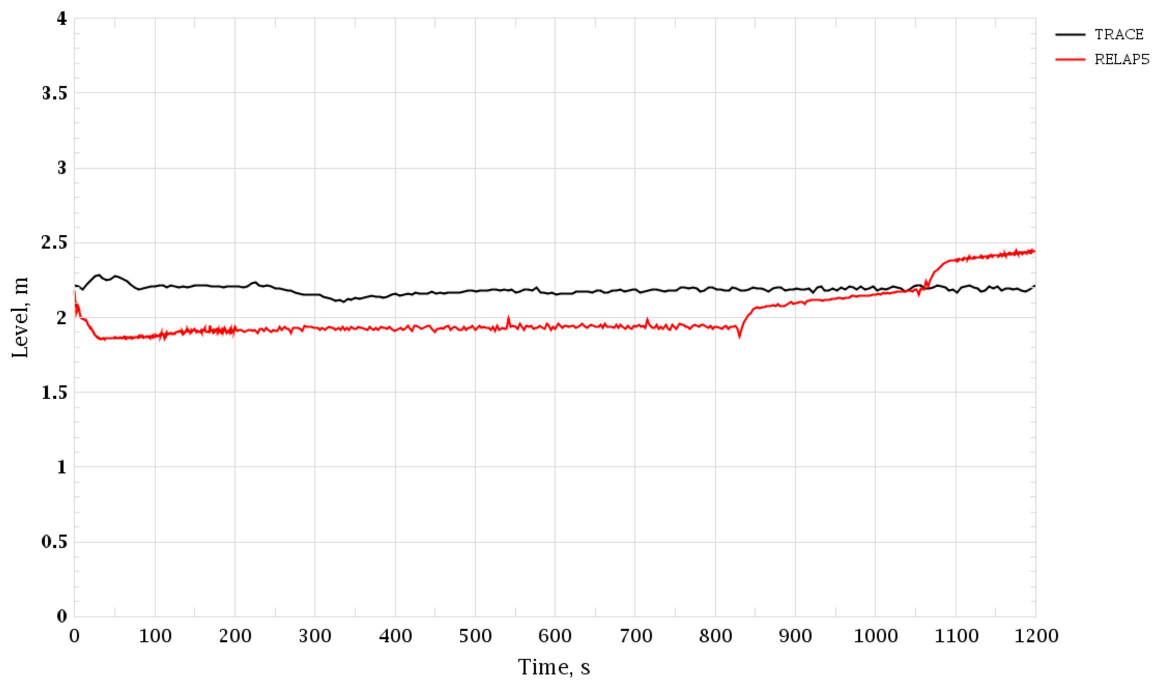

Figure 3-166 PRZ Surge Line Break. SG-3 Level (Wide Range)

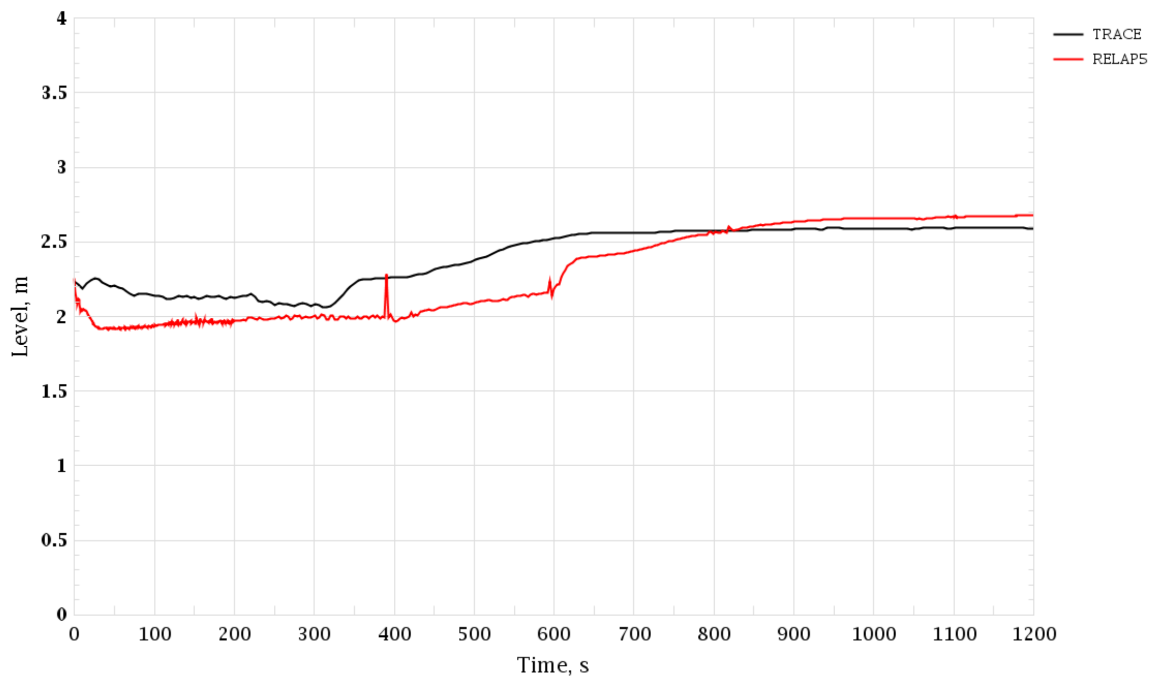

Figure 3-167 PRZ Surge Line Break. SG-4 Level (Wide Range)

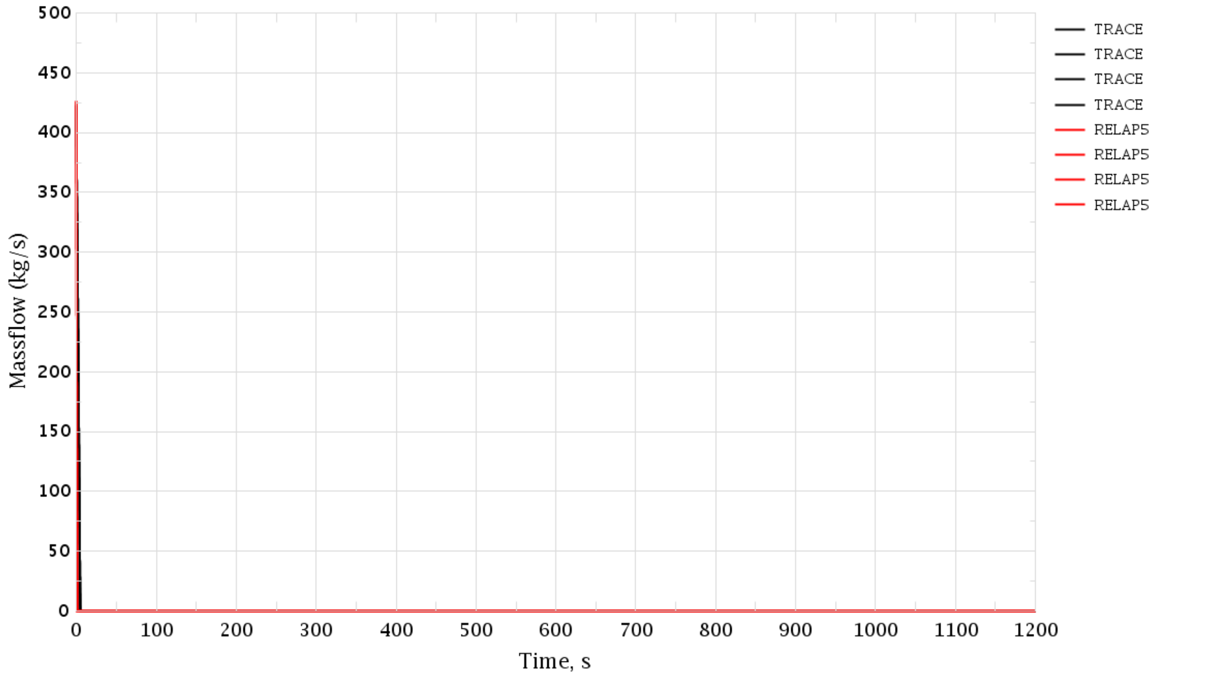

Figure 3-168 PRZ Surge Line Break. Turbines Mass Flow Rate

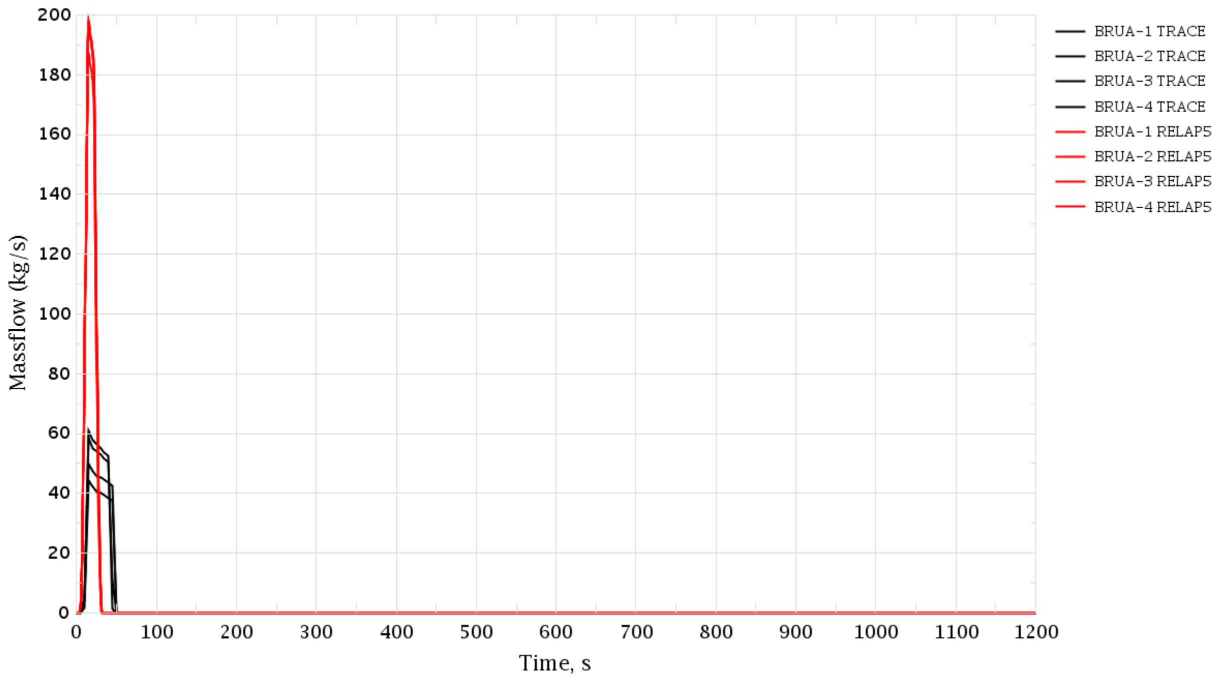

Figure 3-169 PRZ Surge Line Break. BRU-A Steam Mass Flow Rate

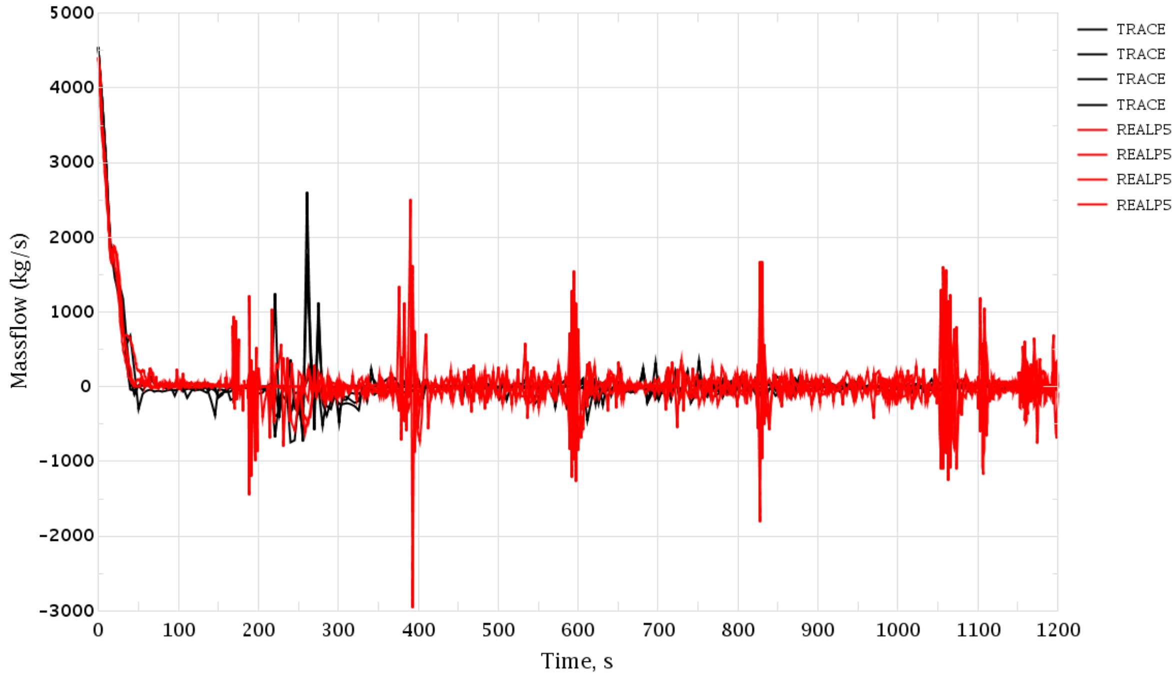

Figure 3-170 PRZ Surge Line Break. MFW Pumps Mass Flow Rate

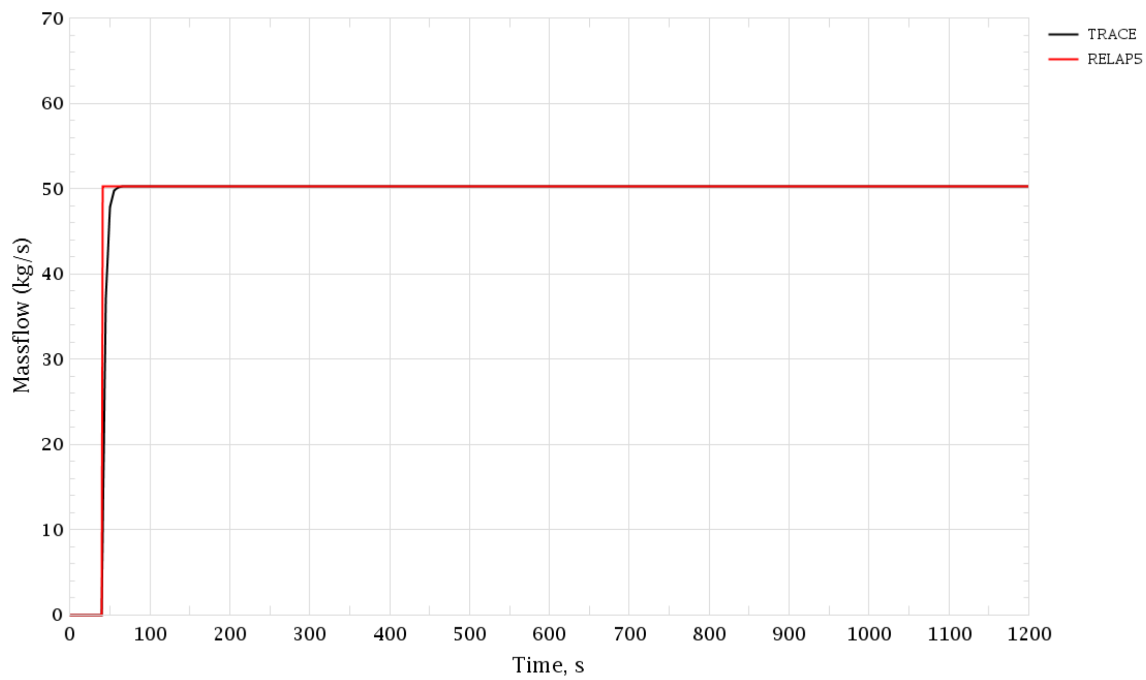

Figure 3-171 PRZ Surge Line Break. HPIS-1 Mass Flow Rate

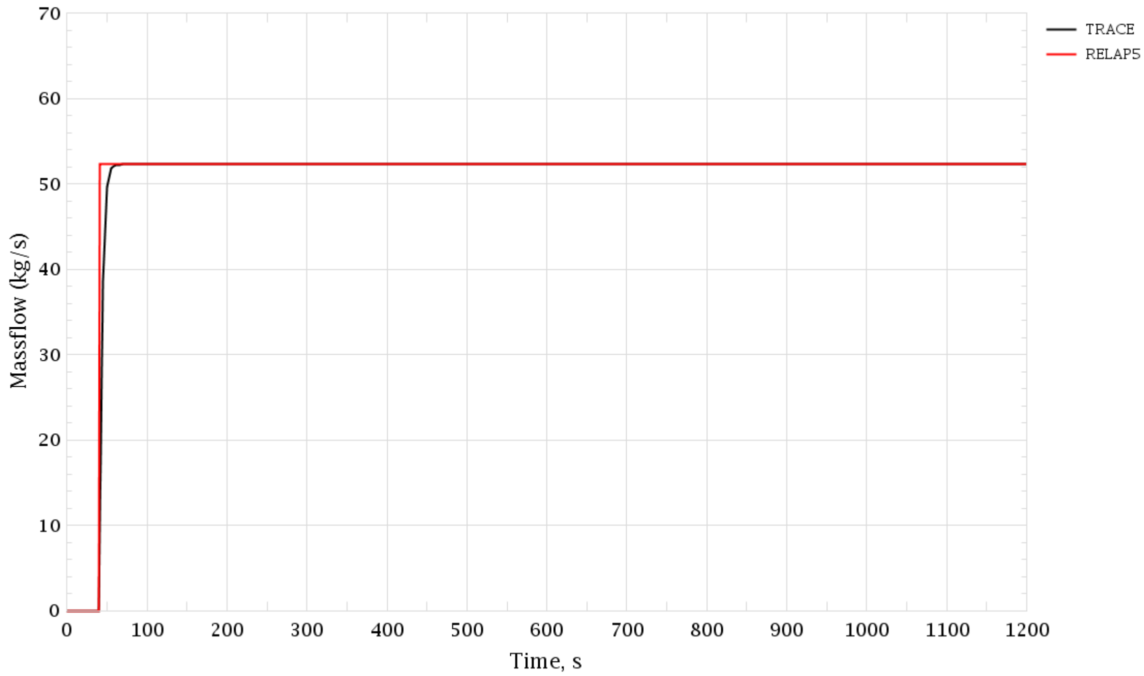

Figure 3-172 PRZ Surge Line Break. HPIS-2 Mass Flow Rate

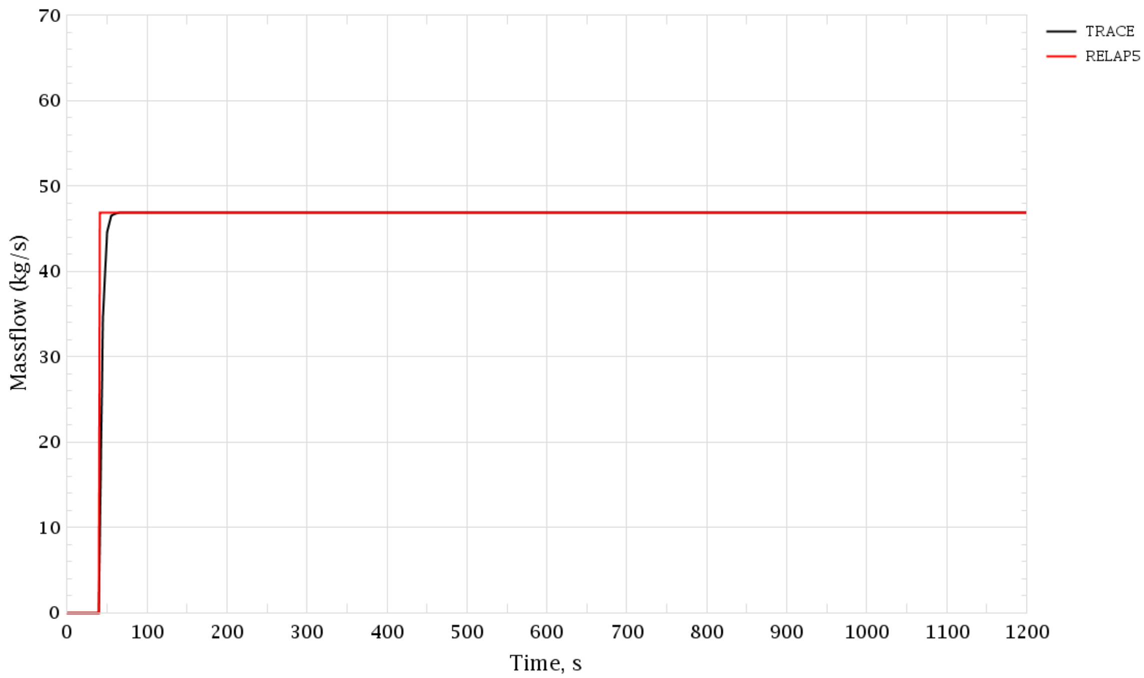

Figure 3-173 PRZ Surge Line Break. HPIS-3 Mass Flow Rate

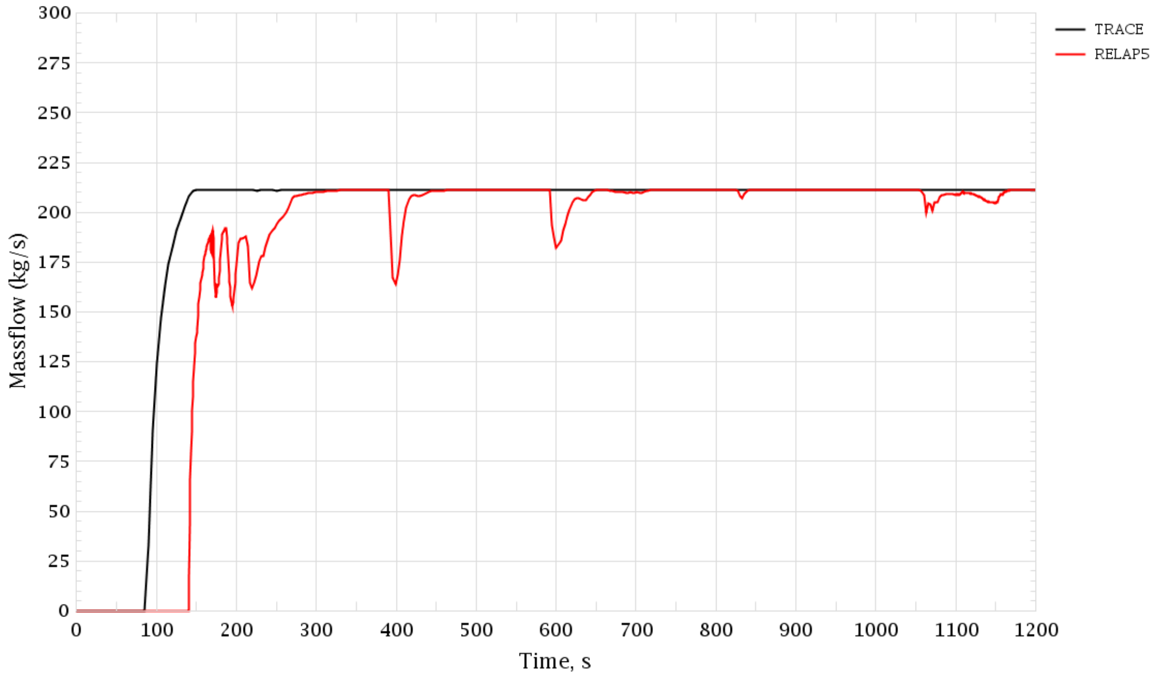

Figure 3-174 PRZ Surge Line Break. LPIS-1 Mass Flow Rate

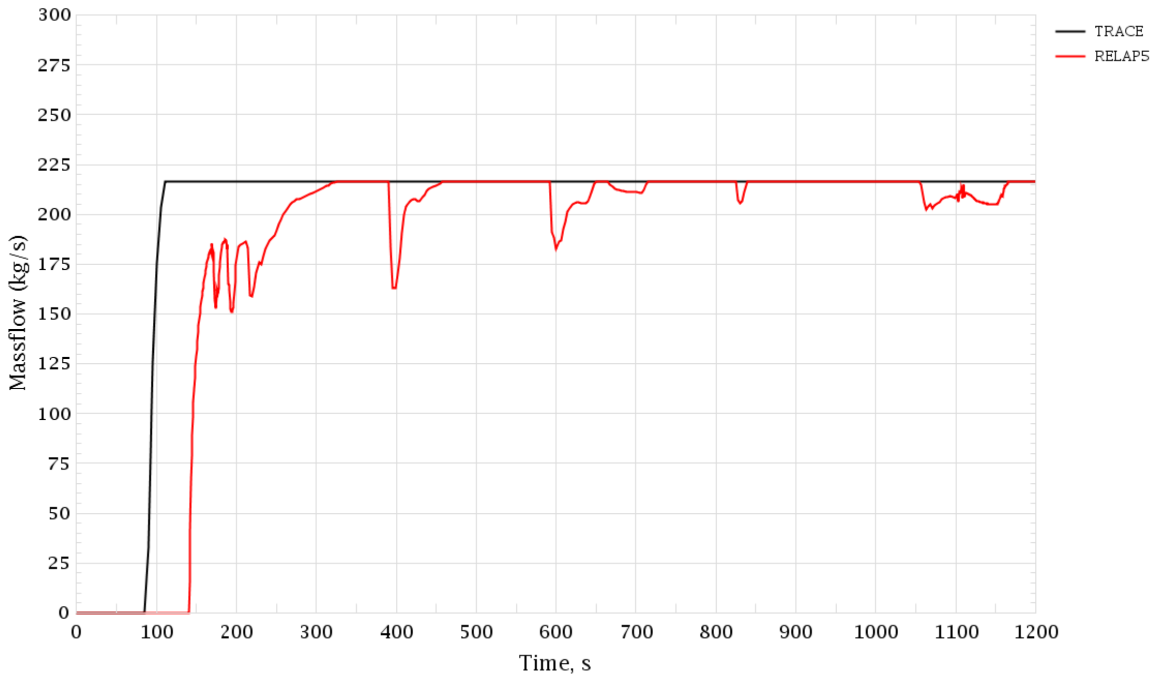

Figure 3-175 PRZ Surge Line Break. LPIS-2 Mass Flow Rate

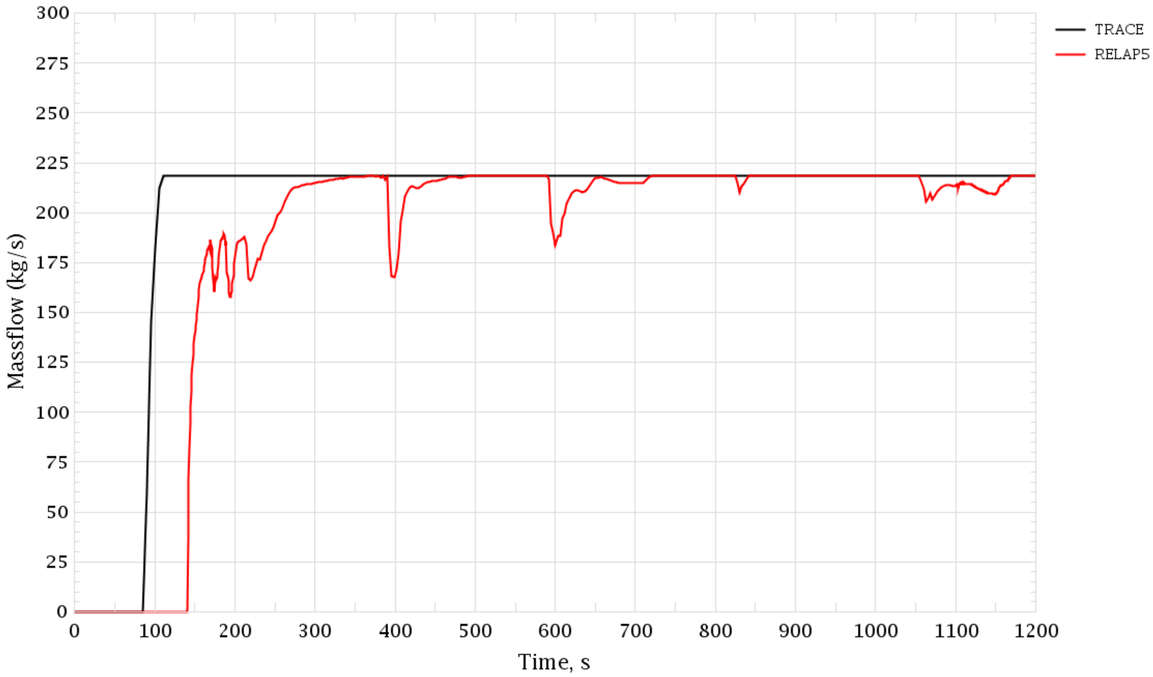

Figure 3-176 PRZ Surge Line Break. LPIS-3 Mass Flow Rate

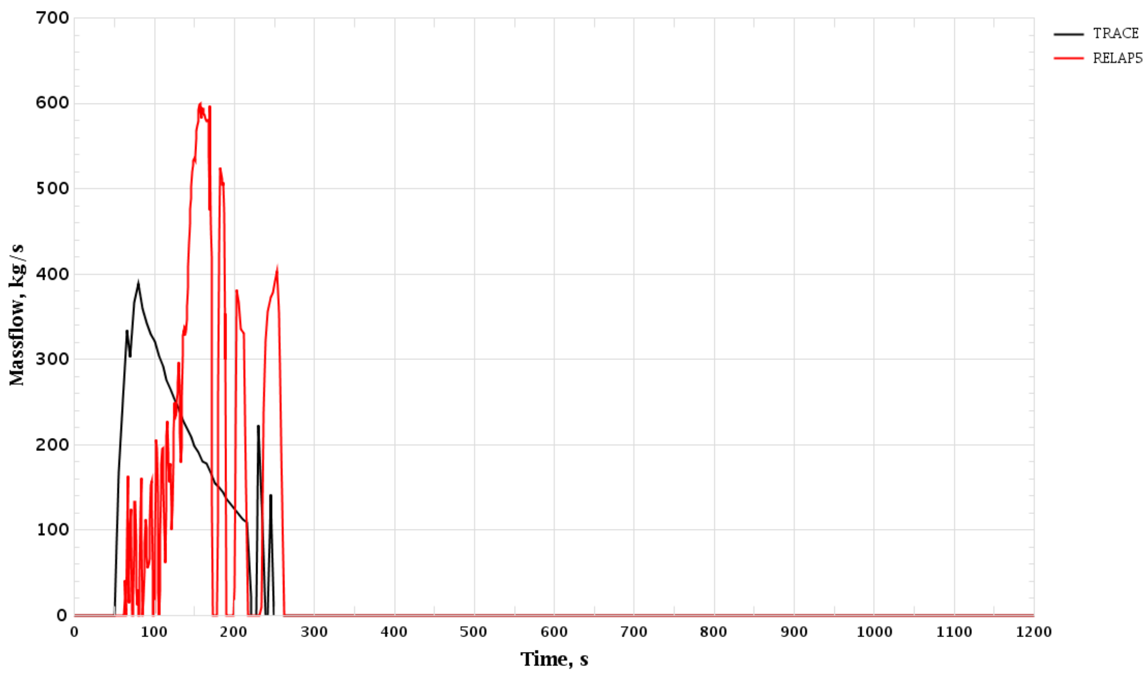

Figure 3-177 PRZ Surge Line Break. HA-1 Mass Flow Rate

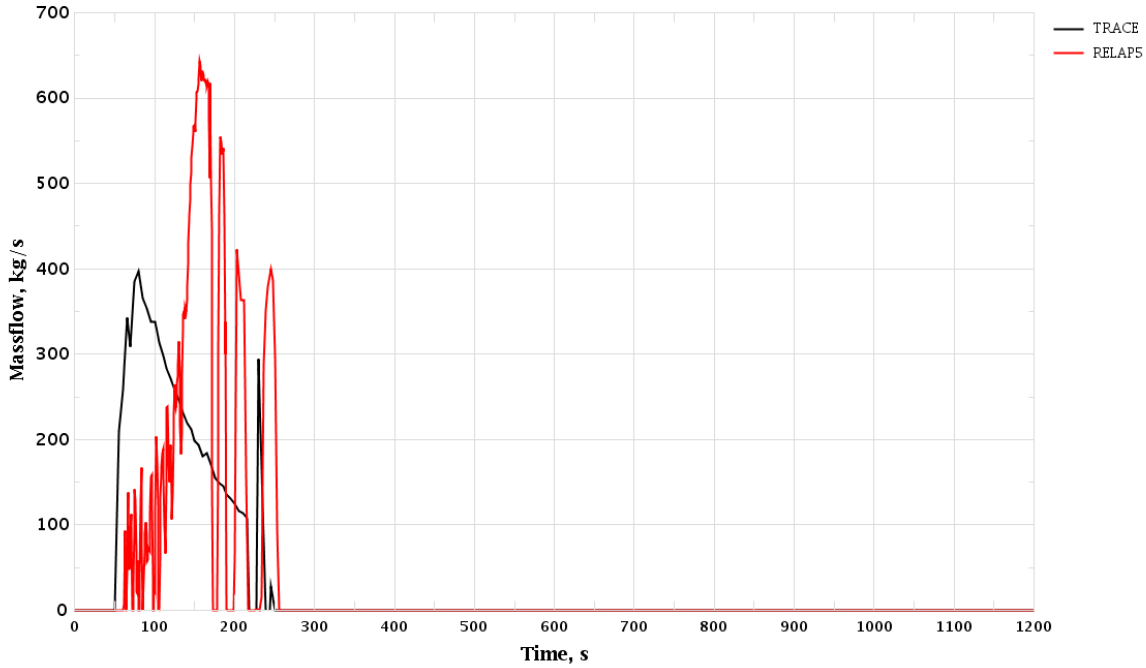

Figure 3-178 PRZ Surge Line Break. HA-2 Mass Flow Rate

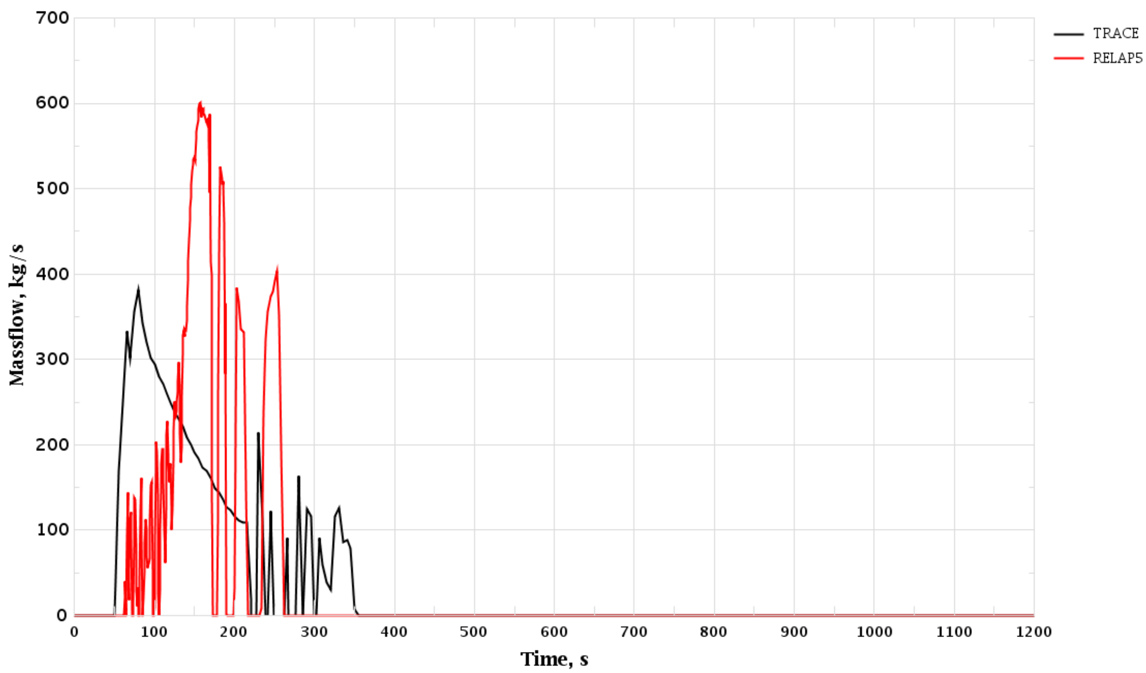

Figure 3-179 PRZ Surge Line Break. HA-3 Mass Flow Rate

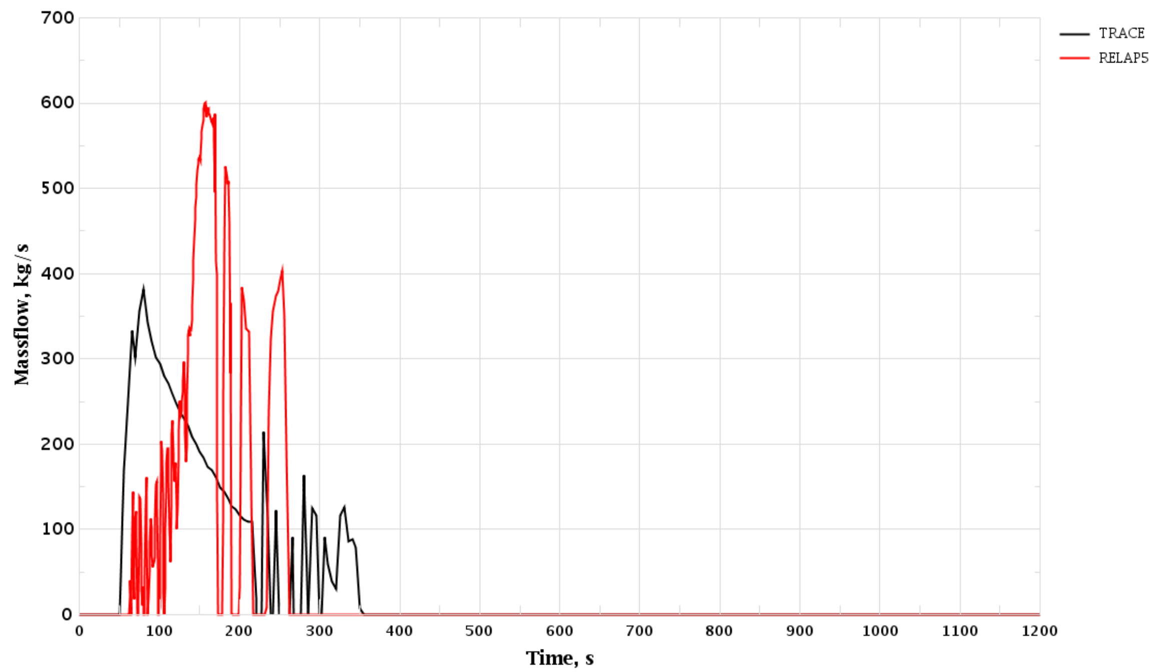

Figure 3-180 PRZ Surge Line Break. HA-4 Mass Flow Rate

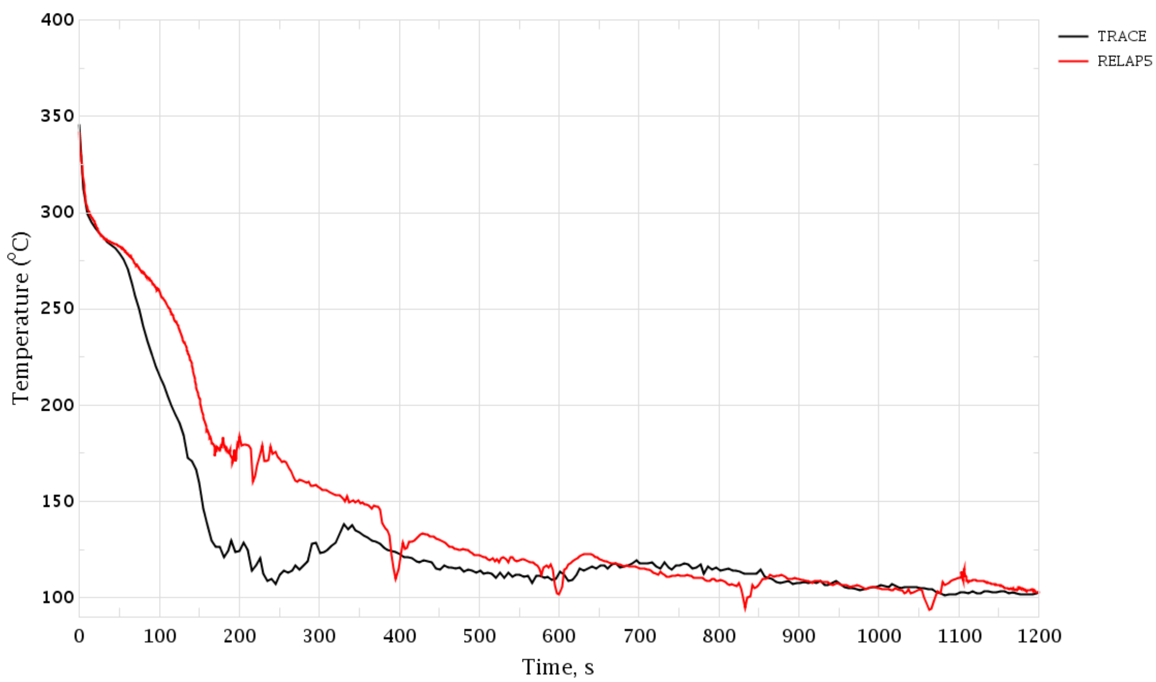

Figure 3-181 PRZ Surge Line Break. Maximal Cladding Temperature

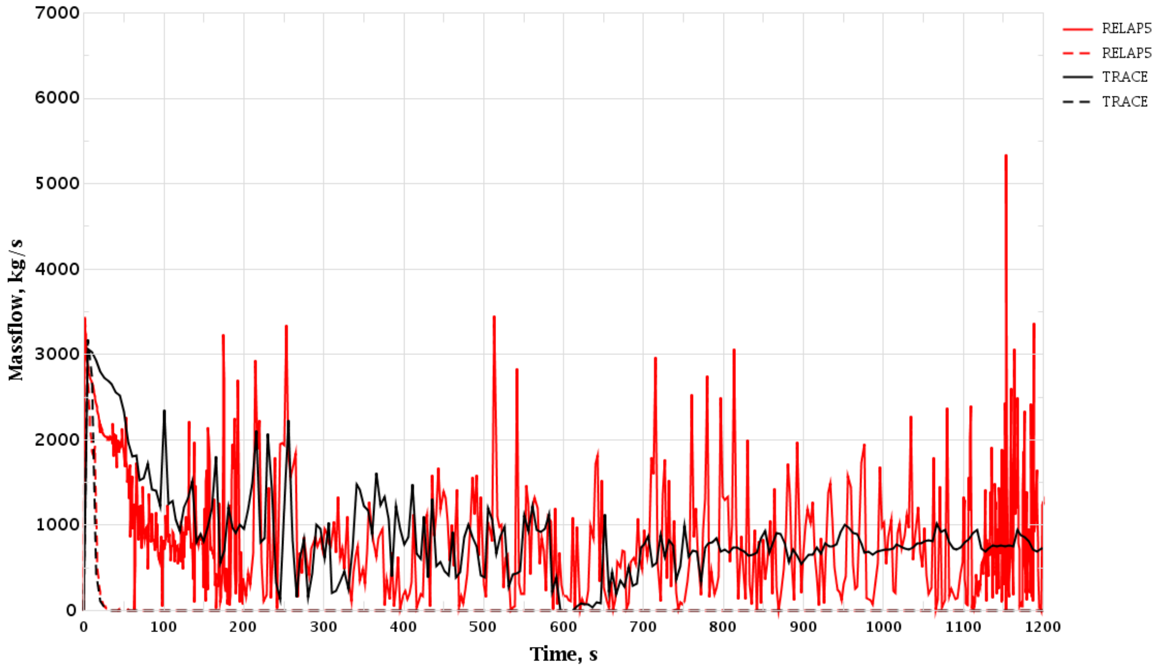

Figure 3-182 PRZ Surge Line Break. Break Mass Flow Rate

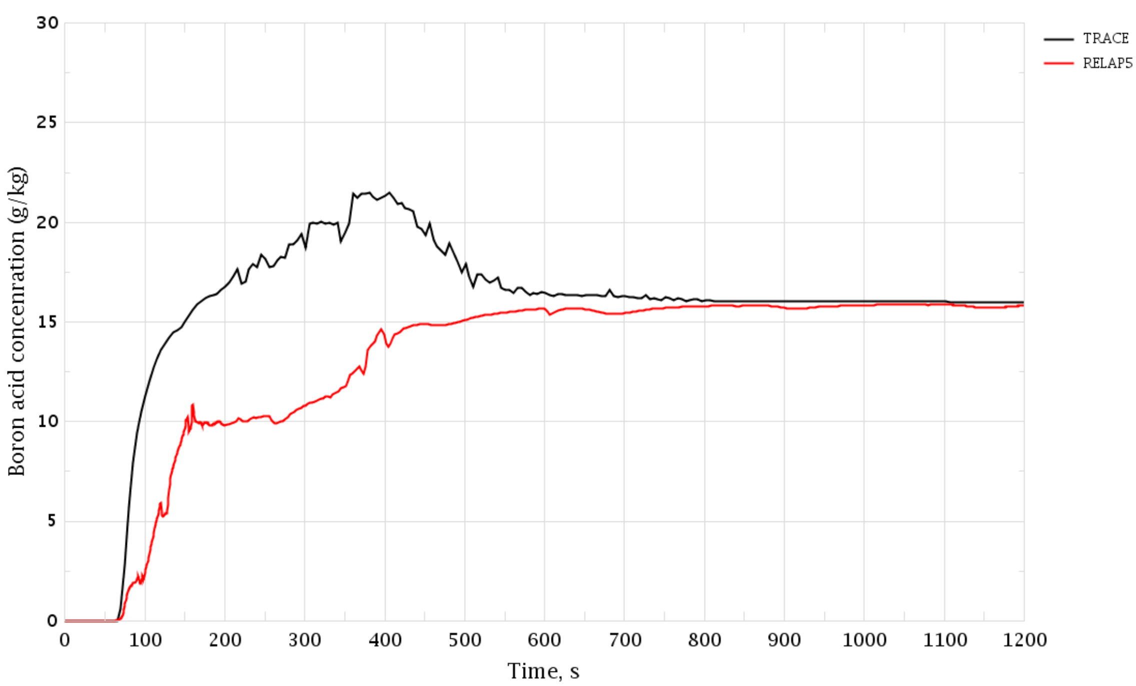

Figure 3-183 PRZ Surge Line Break. Boric Acid Concentration in the Core

3.6 Inadvertent Opening of SG SRV in Hot Shutdown

3.6.1 Brief Description of Incident

The incident occurred at Khmelnitsky NPP unit 1 on August 07, 1999, with the reactor in "hot shutdown" state during planned unit start-up after refueling. According to the incident evaluation report [4] at 11:46 the inadvertent opening of SG-1 SRV TX50S04 occurred that resulted in actuation of "MSL break" safeguard (Pressure decrease in any MSL below 50 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C at RCS temperature >200°C) with closure of MSIV at MSL-1 and automatic start of safety systems. Actuation of safeguards "Backward pressure difference across MSL check valve > 2.0 kgf/cm^2" in MSL-1,2,4 caused a trip of RCP-1,2,4.

Four minutes after the incident occurrence the SG-1 SRV was closed by operator and in next 5 minutes the automatically started safety systems were switched off. According to a description in [4], at 12:10 (at 12:04 according to plots in [4]) the operators restarted RCP-1,2,4 that caused restoration and stabilization of the primary and secondary circuit parameters.

It shall be noted that according to incident evaluation report [4] actuation of "MSL break" safeguard also caused isolation of SG-1 feedwater pipelines, however provided plots and simulation results do not allow to confirm this statement.

3.6.2 Initial Conditions

The initial conditions selected for the calculation of incident are presented in Table 3-7.

Table 3-7 Initial Conditions for Calculation of "Inadvertent Opening of SG SRV in Hot Shutdown" Incident

Parameter	Units	Reported value [4]	Calculated value
Core thermal power	MWt	–	1.65
Reactor outlet pressure	kgf/cm^2	160.0	159.8
Coolant temperature at reactor outlet	°C	278.0	278.0
Reactor coolant flow rate	m^3/h	84800^{+4000}_{-4800}	84350
PRZ level	m	5.78±0.150	5.78
SG pressure	kgf/cm^2	63…64	63.5
SG level (narrow range gauge)	m	0.300±0.05	0.305...0.315

3.6.3 Boundary Conditions

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

As an initiating event an inadvertent opening of SG-1 SRV is postulated at 0 s of calculation. Closure of the SRV is performed manually at 215 s. Actuation of the safeguards is simulated according to a design, except isolation of feedwater supply to SG-1, i.e. MFW control and cut-off valves to SG-1 remain open.

Trip of RCP-1, 2, 4 occurs after actuation of safeguard "Backward pressure difference across MSL check valve > 2.0 kgf/cm^2". RCP-3 remains in operation (failure of above mentioned safeguard for MSL-3 is postulated). At 1080 s of incident restart of RCP-1, 2, 4 by operator is simulated.

3.6.4 Calculation Results

Sequence of events for this incident are presented in Table 3-8.

Table 3-8 Sequence of Events for "Inadvertent Opening of SG SRV in Hot Shutdown" Incident

TRACE Time, s	NPP data Time, s	Event	Description
0	0	Opening of SG-1 SRV	IE occurrence
5	N/A	RCP-1 trip	Backward pressure difference across MSL-1 check valve > 2.0 kgf/cm^2
33	N/A	Automatic start of safety systems	Pressure decrease in any MSL below 50 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C and RCS temperature >200°C
		Closure of MSIV at MSL-1	MSL-1 break safeguard actuation
100	N/A	RCP-4 trip	Backward pressure difference across MSL-4 check valve > 2.0 kgf/cm^2
190	N/A	RCP-2 trip	Backward pressure difference across MSL-2 check valve > 2.0 kgf/cm^2
200	N/A	PRZ heaters are switched off	Decrease of PRZ level below 4.2 m
215	215	Closure of SG-1 SRV	Operator action
		Minimal SG-1 pressure is reached	27.8 kgf/cm^2 in TRACE calculation and 26.2 kgf/cm^2 according to NPP data
240	250	Minimal RCS pressure is reached	150.1 kgf/cm^2 in TRACE calculation and 151.9 kgf/cm^2 according to NPP data
435	N/A	PRZ heaters are switched on	Increase of PRZ level below 4.2 m and RCS pressure is below actuation setpoints
570	N/A	Closure of MSIV at MSL-2	MSL-2 break safeguard actuation
605	N/A	Closure of MSIV at MSL-4	MSL-4 break safeguard actuation
637	N/A	Closure of MSIV at MSL-3	MSL-3 break safeguard actuation
645	N/A	PRZ heaters are switched off	Increase of RCS pressure above the setpoints
1080	1080	Restart of RCP-1,2,4	Operator action
		Maximal RCS pressure is reached	168 kgf/cm^2 in TRACE calculation and 169 kgf/cm^2 according to NPP data
1100	N/A	PRZ heaters are switched on	Increase of PRZ level below 4.2 m and RCS pressure is below actuation setpoints
1320	N/A	PRZ heaters are switched off	Increase of RCS pressure above the setpoints
1800	1800	End of calculation	Stabilization of the main reactor parameters

Note: N/A means that exact timing of the event is not specified in the incident evaluation report [4]

Opening of SG-1 SRV causes loss of secondary circuit coolant and rapid decrease of SG-1 pressure (Figure 3-192). Due to increase of backward pressure across MSL-1 check valve greater than 2 kgf/cm^2 the signal to trip RCP-1 is formed (Figure 3-189). At 33 s of incident the "MSL-1 break" (Pressure decrease in any MSL below 50 kgf/cm^2 at increase of difference between saturation temperatures of primary and secondary circuit over 75°C and RCS temperature >200°C) is actuated with closure of MSIV at MSL-1 and automatic start of safety systems. Continued loss of secondary circuit coolant from SG-1 causes cool-down of the primary circuit and decrease of RCS pressure (Figure 3-185) and temperature (Figure 3-187, Figure 3-188). This in turn results in a decrease of PRZ level due to coolant shrinkage (Figure 3-186) and decrease of pressure in SG-2, 3, 4 (Figure 3-193 – Figure 3-195). The latter causes sequential (within first 190 s of accident) actuation of safeguards "Increase of backward pressure across MSL check valve greater than 2 kgf/cm^2' to trip RCPs at other RCS loops (Figure 3-189) with the exception of loop no.3, where failure of actuation of correspondent safeguard is postulated.

At 200 s the PRZ level decreases below 4.2 m (Figure 3-186) with prohibition for PRZ heaters operation and PRZ heaters are switched off till PRZ level is restored at 435 s (Figure 3-191).

At 215 s the loss of secondary circuit coolant is terminated by manual closure of failed SG-1 SRV and SG-1 pressure (Figure 3-192), as well as RCS temperature (Figure 3-187, Figure 3-188) and pressure (Figure 3-185) start to recover. The minimal RCS pressure is 150.1 kgf/cm^2 according to TRACE calculation and 151.9 kgf/cm^2 according to NPP data plots in [4], and minimal SG-1 pressure reached is 27.8 kgf/cm^2 and 26.2 kgf/cm^2, respectively.

Since RCS temperature remains lower than the secondary coolant temperature in SG-2,3,4, pressure in these SGs gradually decreases and after actuation of "MSL break" signals for these SGs, MSIVs at the correspondent steam lines are closed at 570–637 s of incident.

At 1080 s operator restarts RCP-1,2,4 (Figure 3-189). Operation of RCP-1 improves flow to PRZ spray and RCS pressure rapidly decreases (Figure 3-185). After temporary operation of PRZ heaters (Figure 3-191) at 1100–1320 s the RCS pressure is stabilized close to the nominal value.

The calculation is ended at 1800 s after stabilization of the main parameters of primary and secondary circuit. At the end of calculation the PRZ level slowly recovers (Figure 3-186) due to operation of primary circuit make-up (Figure 3-190).

The plots of the main parameters of calculation are presented below on Figure 3-184 – Figure 3-200.

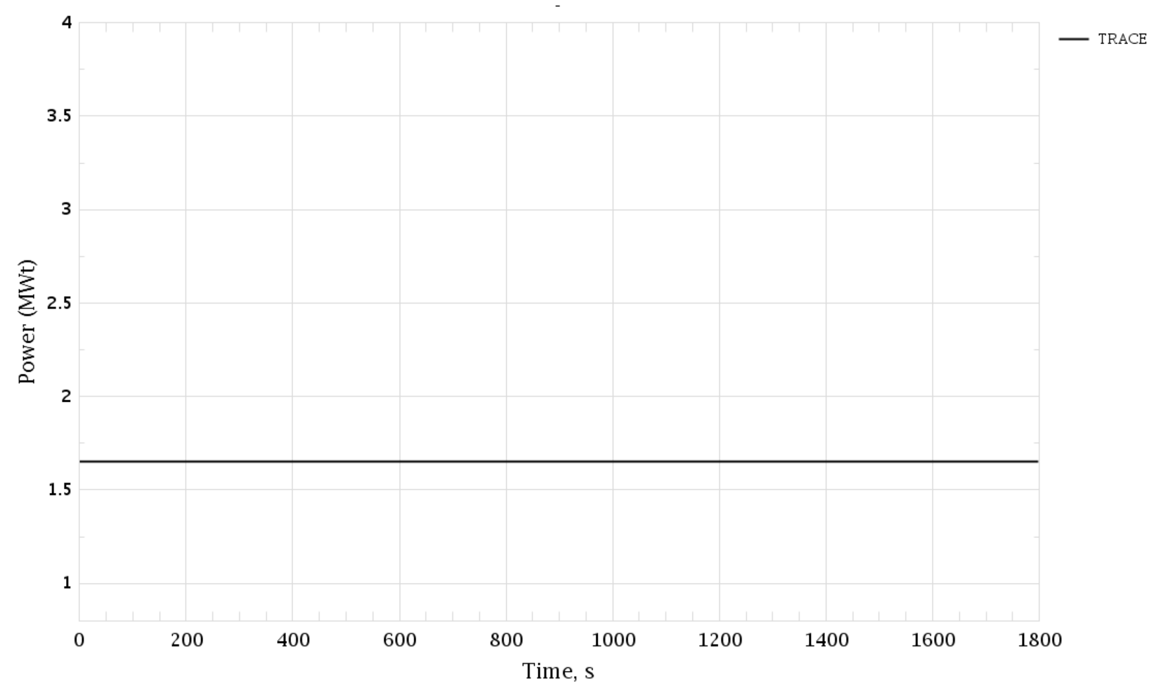

Figure 3-184 Inadvertent Opening of SG SRV in Hot Shutdown. Core Thermal Power

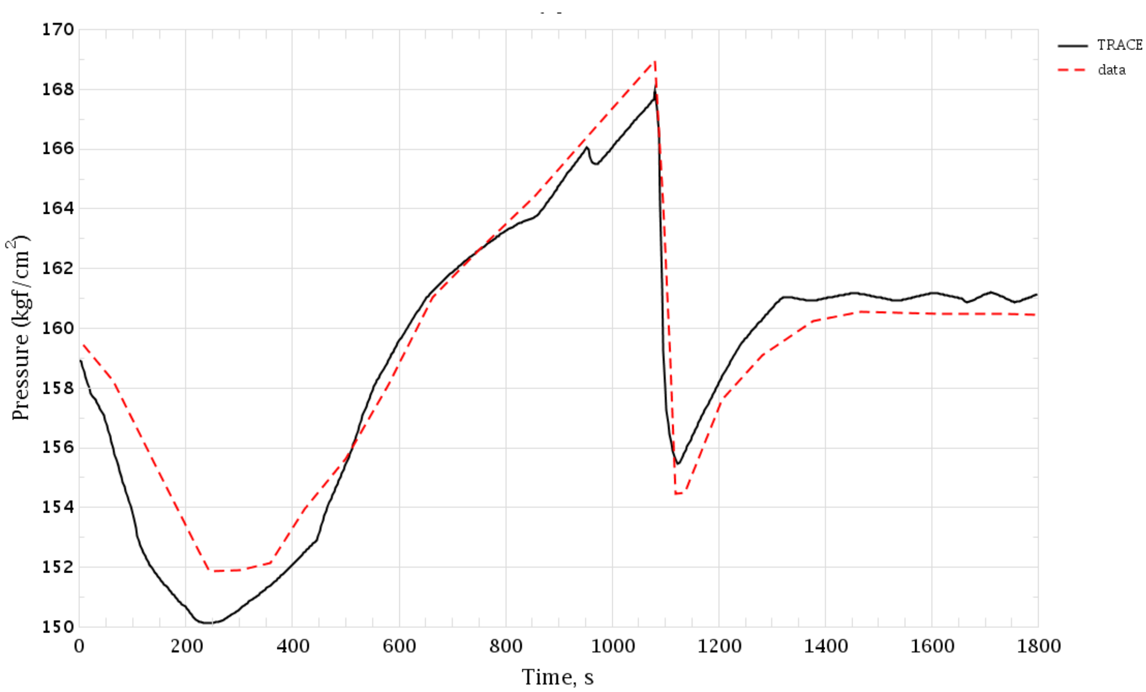

Figure 3-185 Inadvertent Opening of SG SRV in Hot Shutdown. RCS Pressure

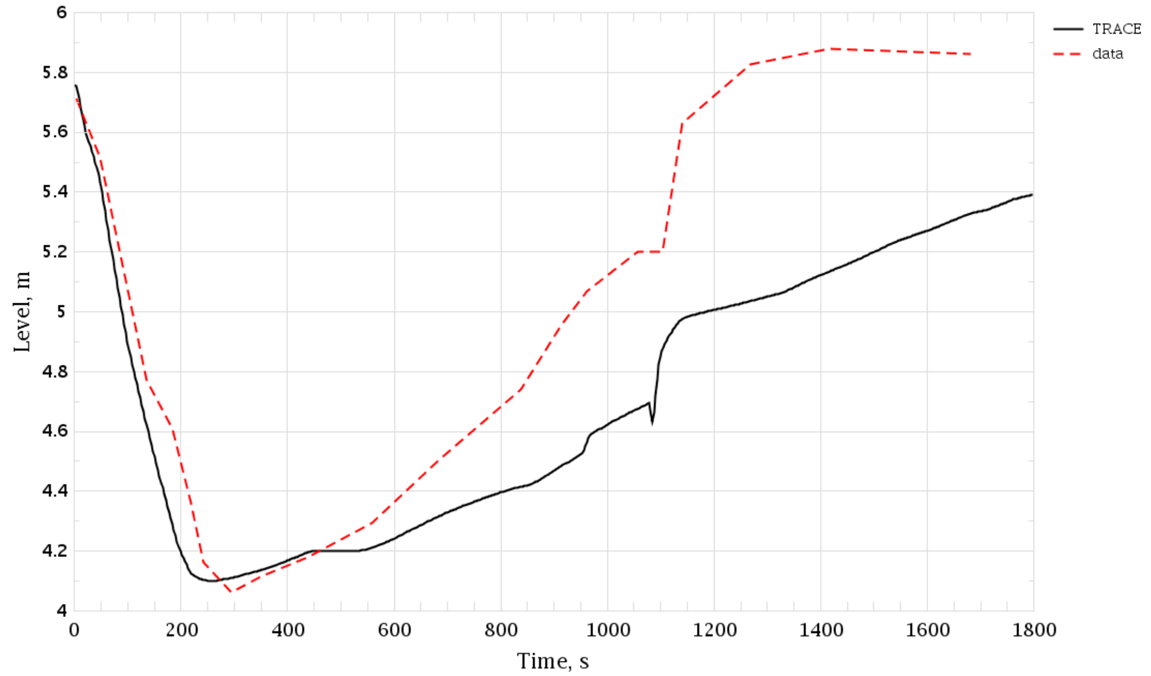

Figure 3-186 Inadvertent Opening of SG SRV in Hot Shutdown. Pressurizer Level

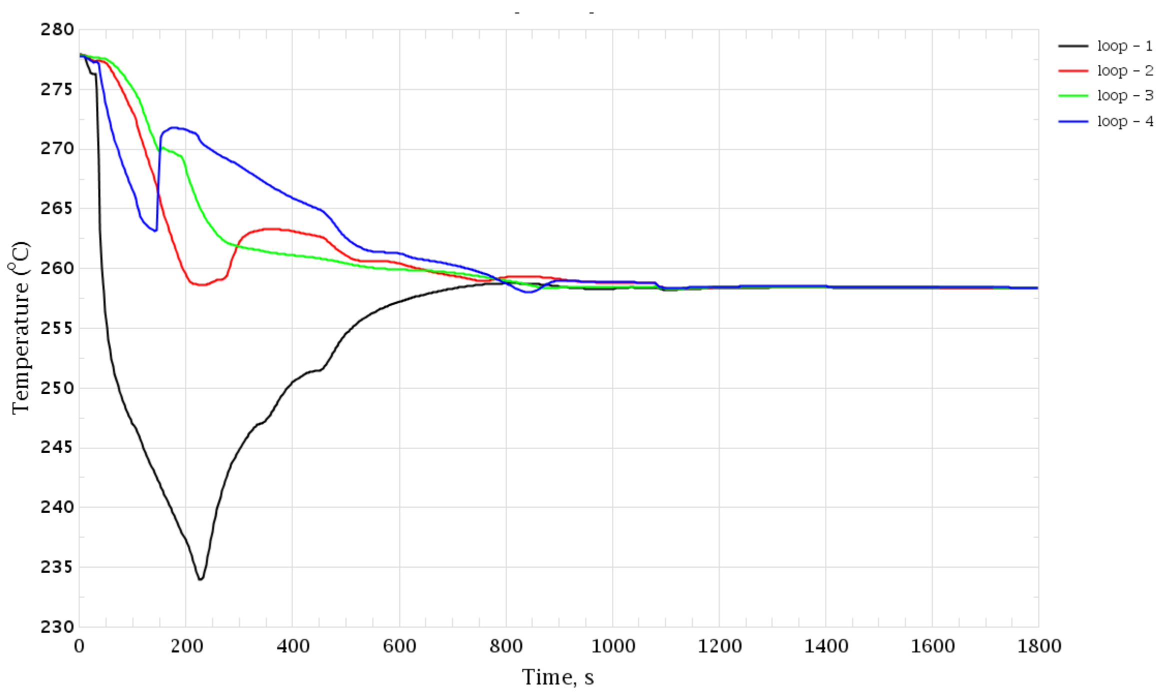

Figure 3-187 Inadvertent Opening of SG SRV in Hot Shutdown. Coolant Temperature in Hot Legs

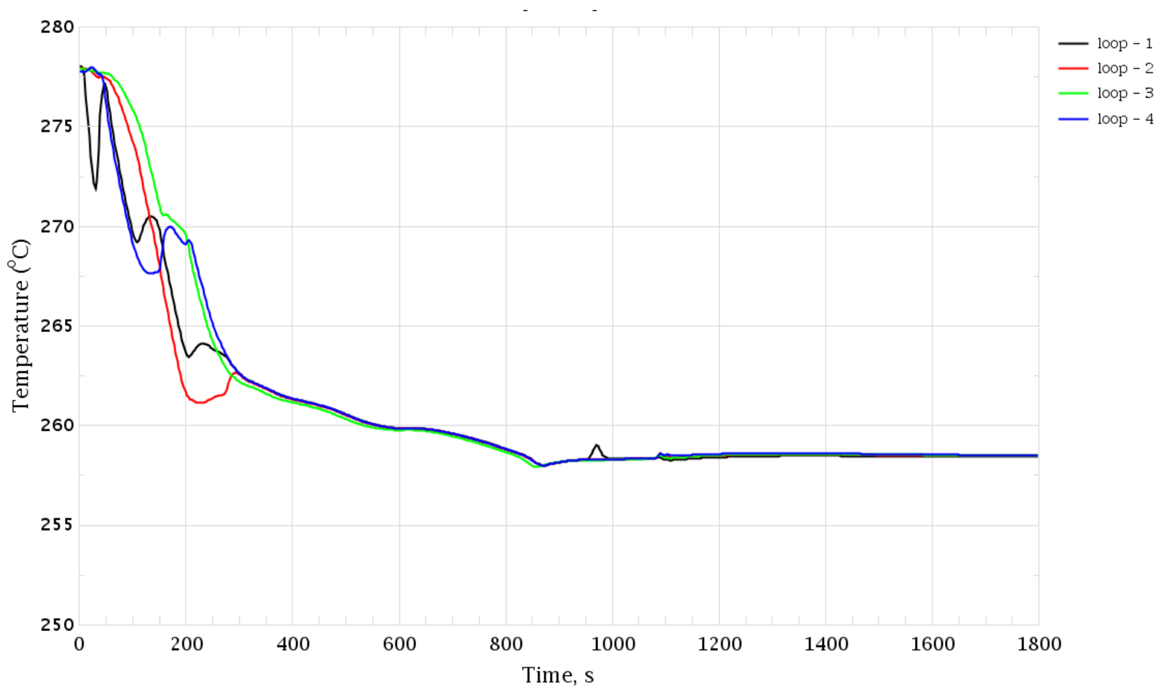

Figure 3-188 Inadvertent Opening of SG SRV in Hot Shutdown. Coolant Temperature in Cold Legs

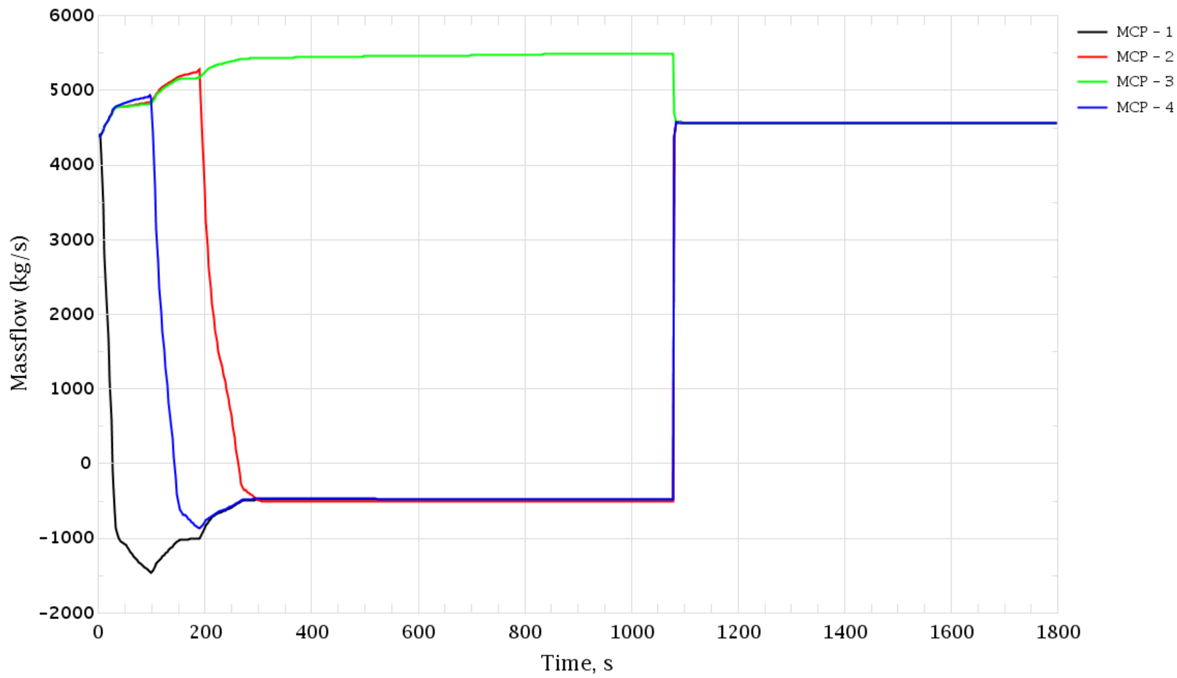

Figure 3-189 Inadvertent Opening of SG SRV in Hot Shutdown. RCS Loops Mass Flow Rate

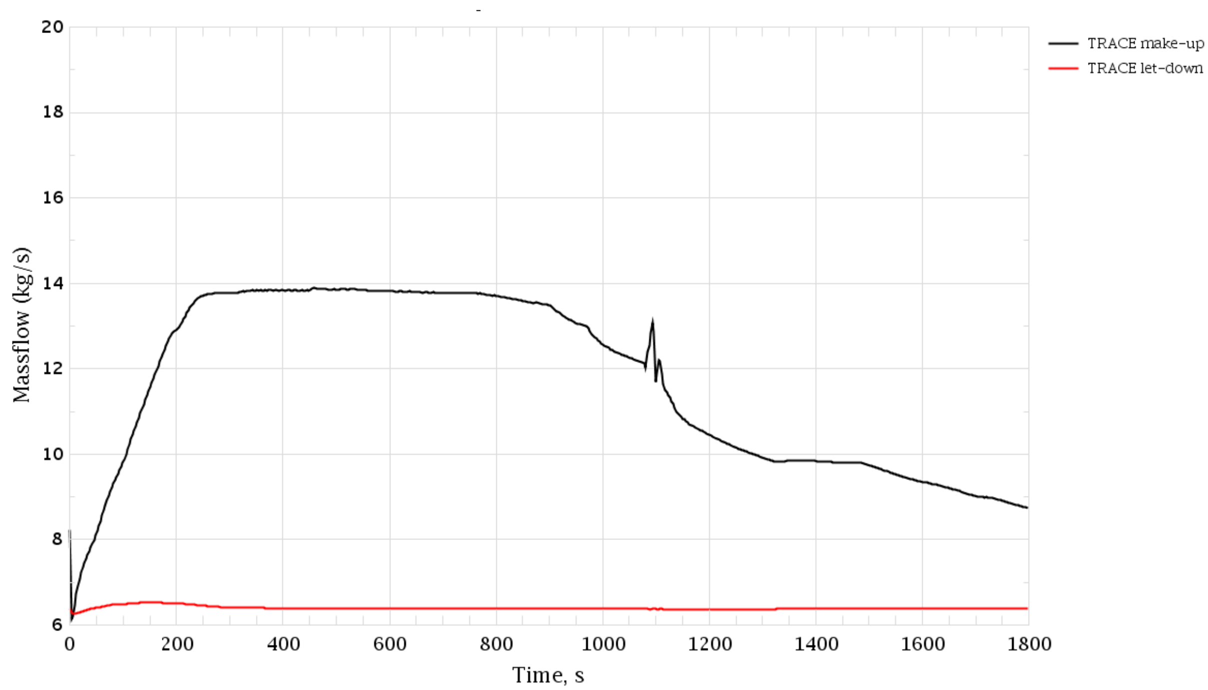

Figure 3-190 Inadvertent Opening of SG SRV in Hot Shutdown. Make-Up and Let-Down Mass Flow

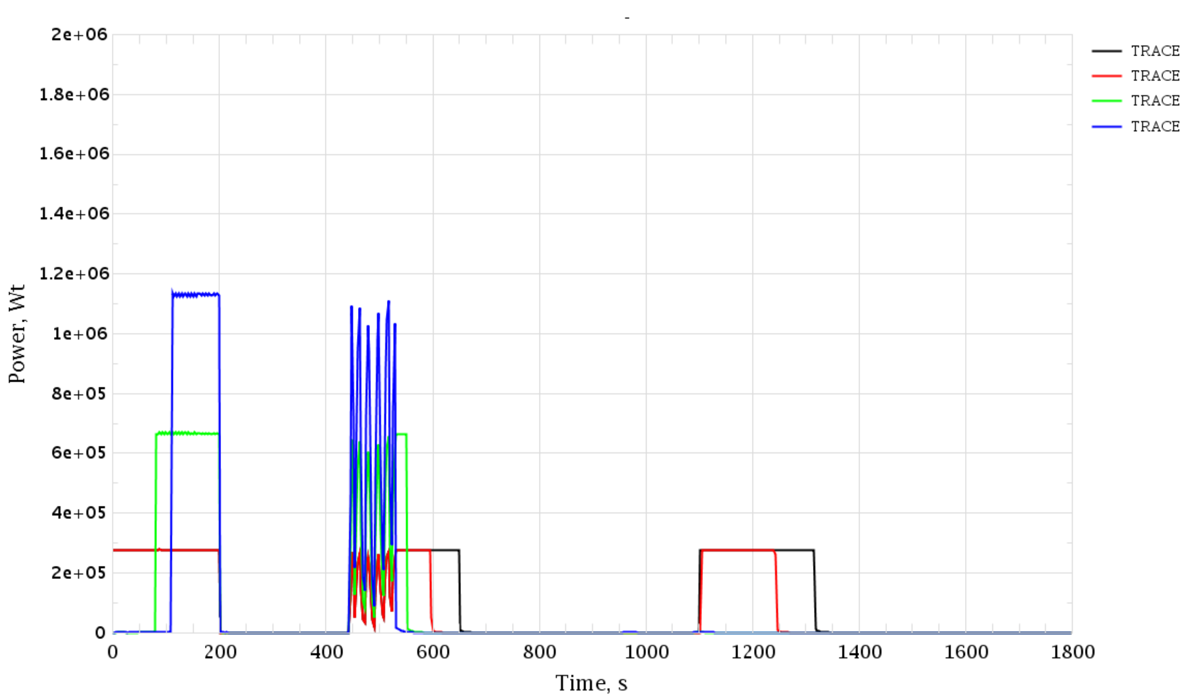

Figure 3-191 Inadvertent Opening of SG SRV in Hot Shutdown. PRZ Heaters Power

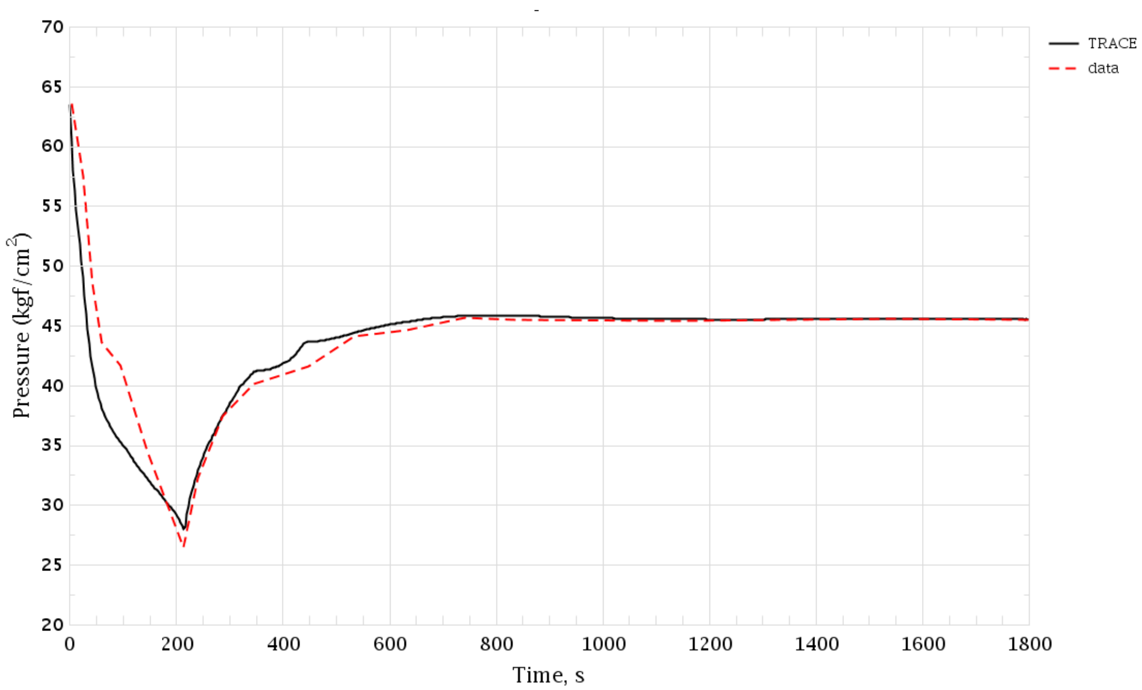

Figure 3-192 Inadvertent Opening of SG SRV in Hot Shutdown. SG-1 Pressure

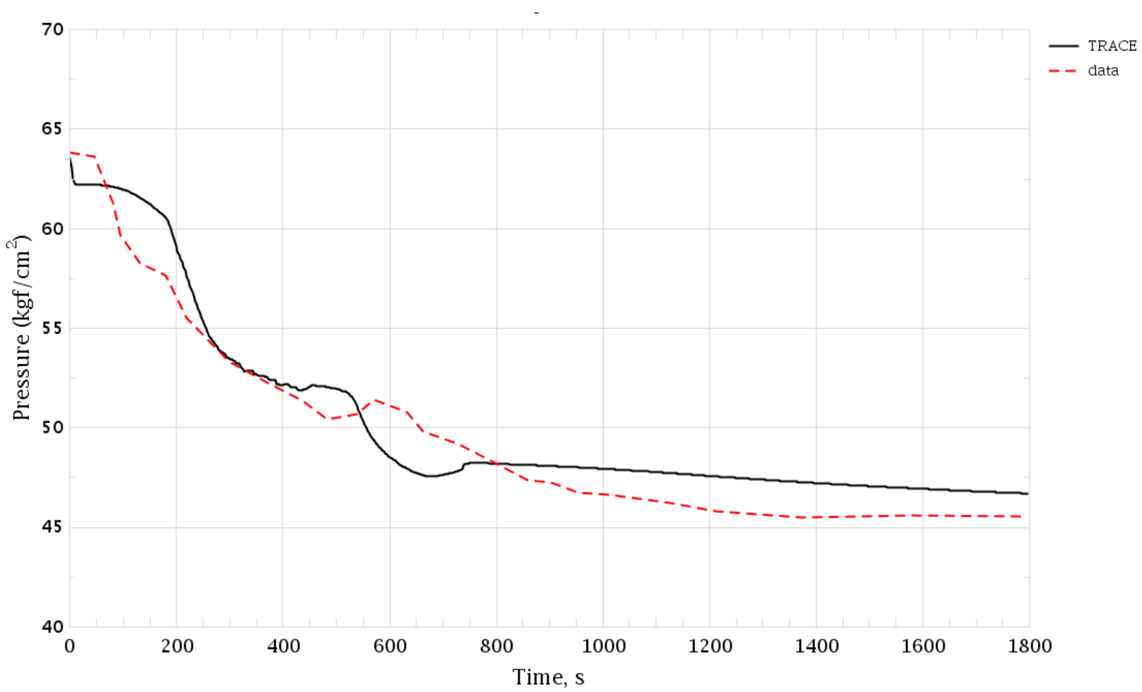

Figure 3-193 Inadvertent Opening of SG SRV in Hot Shutdown. SG-2 Pressure

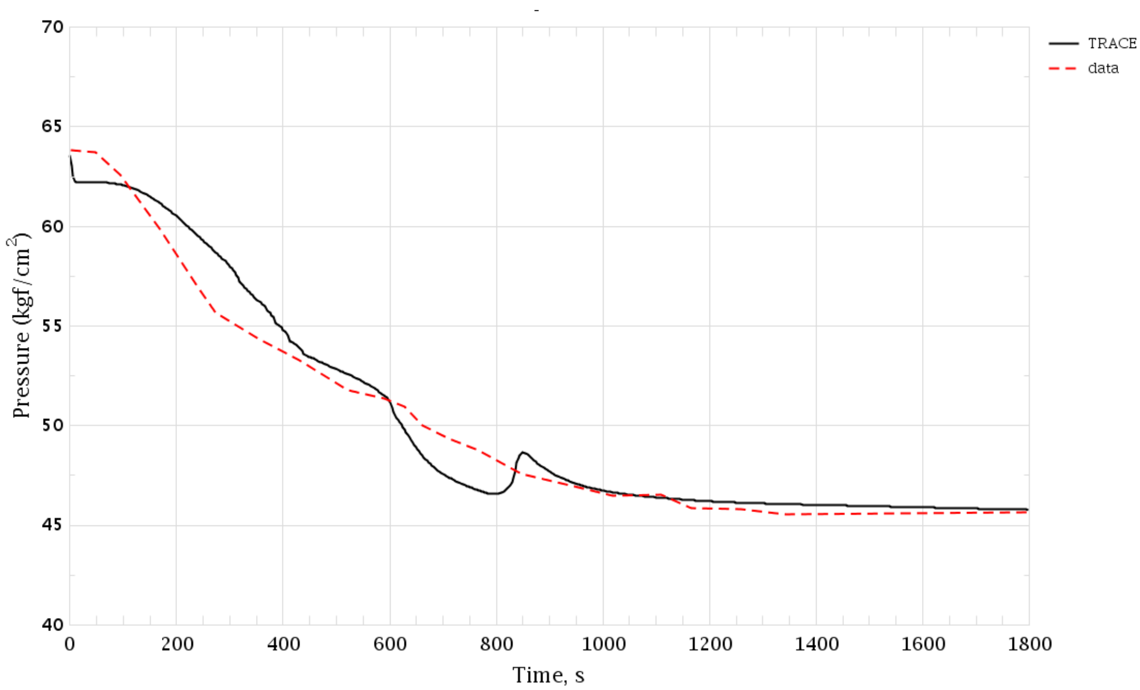

Figure 3-194 Inadvertent Opening of SG SRV in Hot Shutdown. SG-3 Pressure

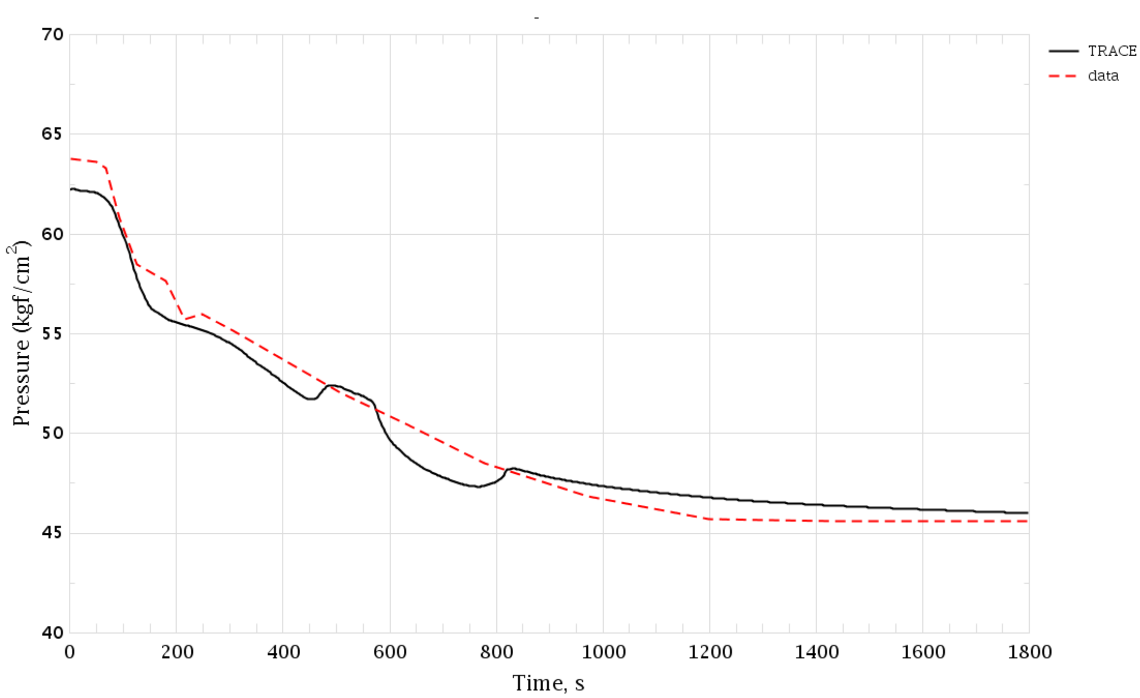

Figure 3-195 Inadvertent Opening of SG SRV in Hot Shutdown. SG-4 Pressure

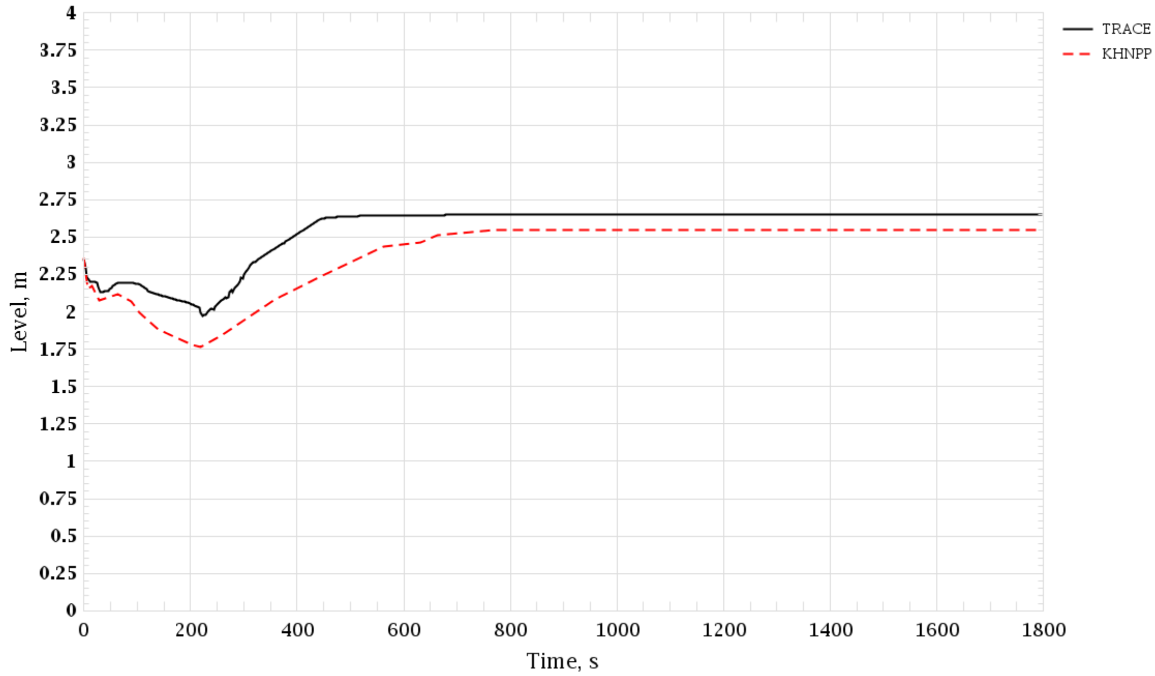

Figure 3-196 Inadvertent Opening of SG SRV in Hot Shutdown. SG-1 Level (Wide Range)

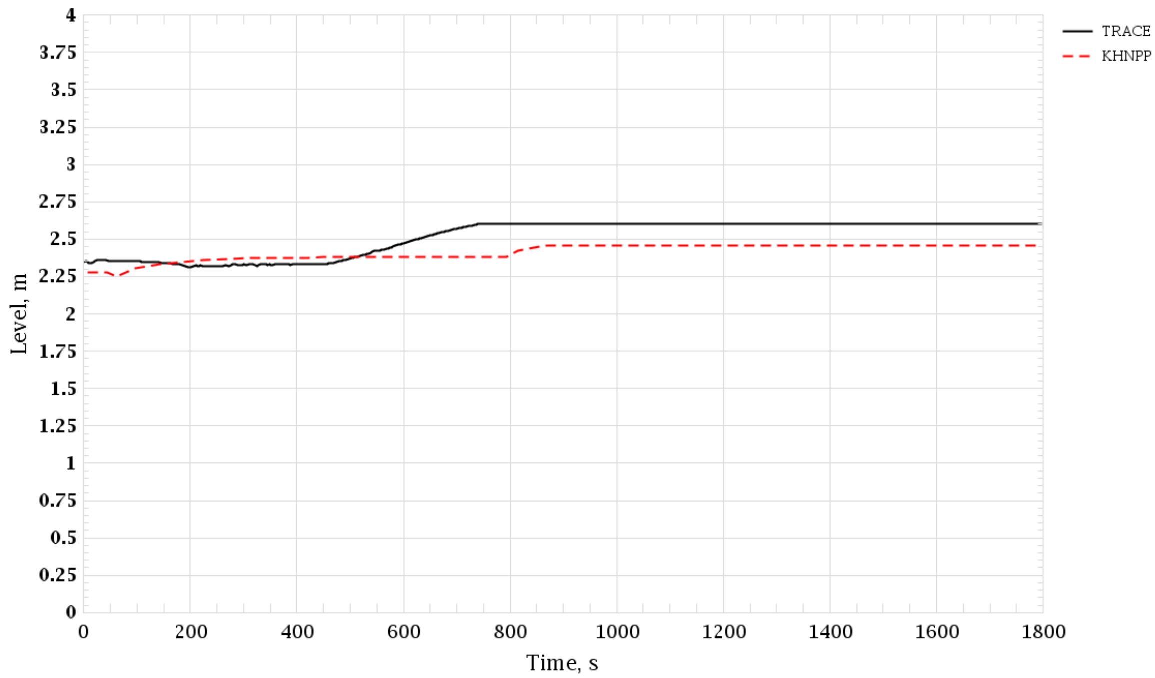

Figure 3-197 Inadvertent Opening of SG SRV in Hot Shutdown. SG-2 Level (Wide Range)

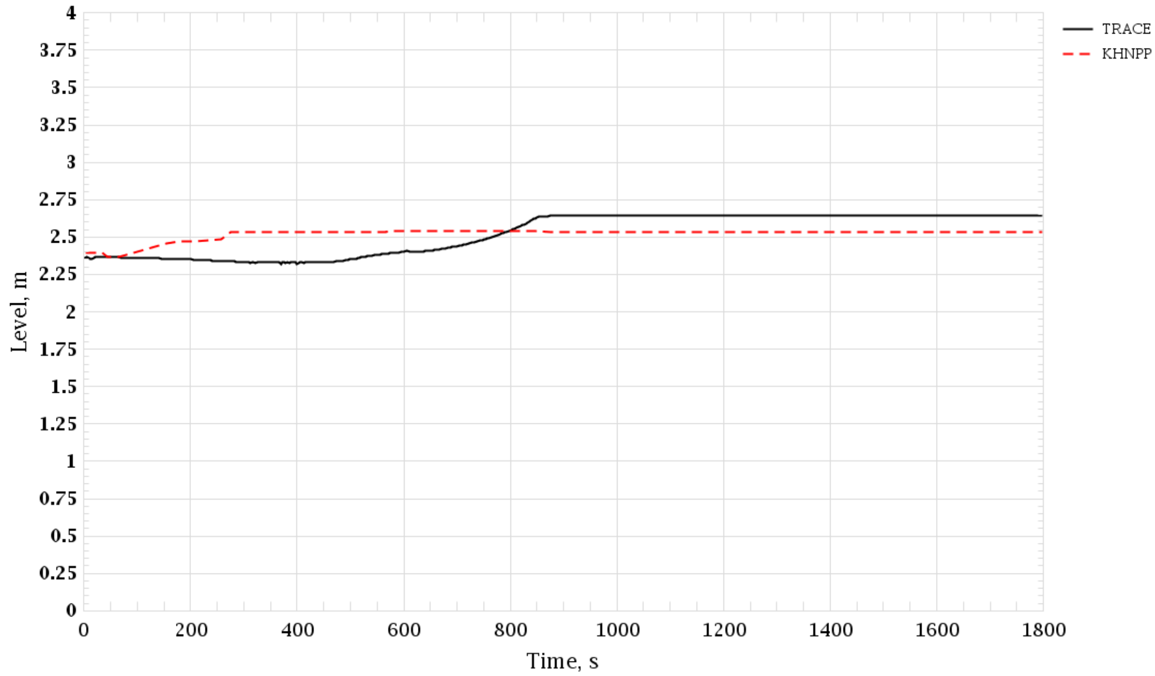

Figure 3-198 Inadvertent Opening of SG SRV in Hot Shutdown. SG-3 Level (Wide Range)

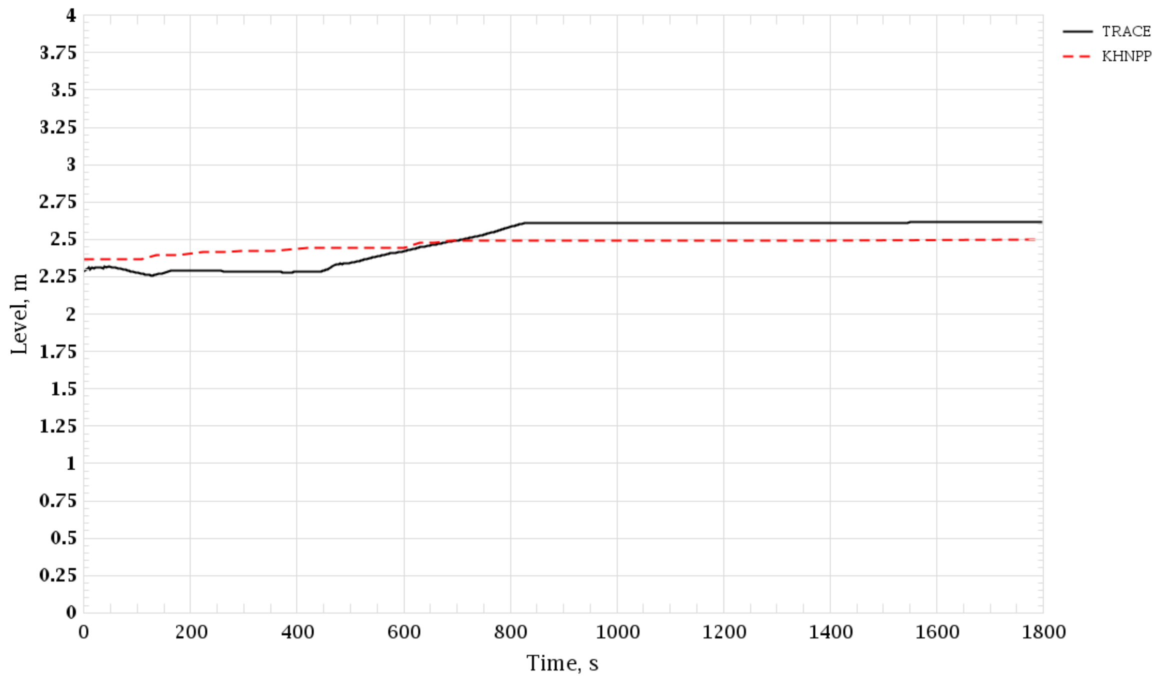

Figure 3-199 Inadvertent Opening of SG SRV in Hot Shutdown. SG-4 Level (Wide Range)

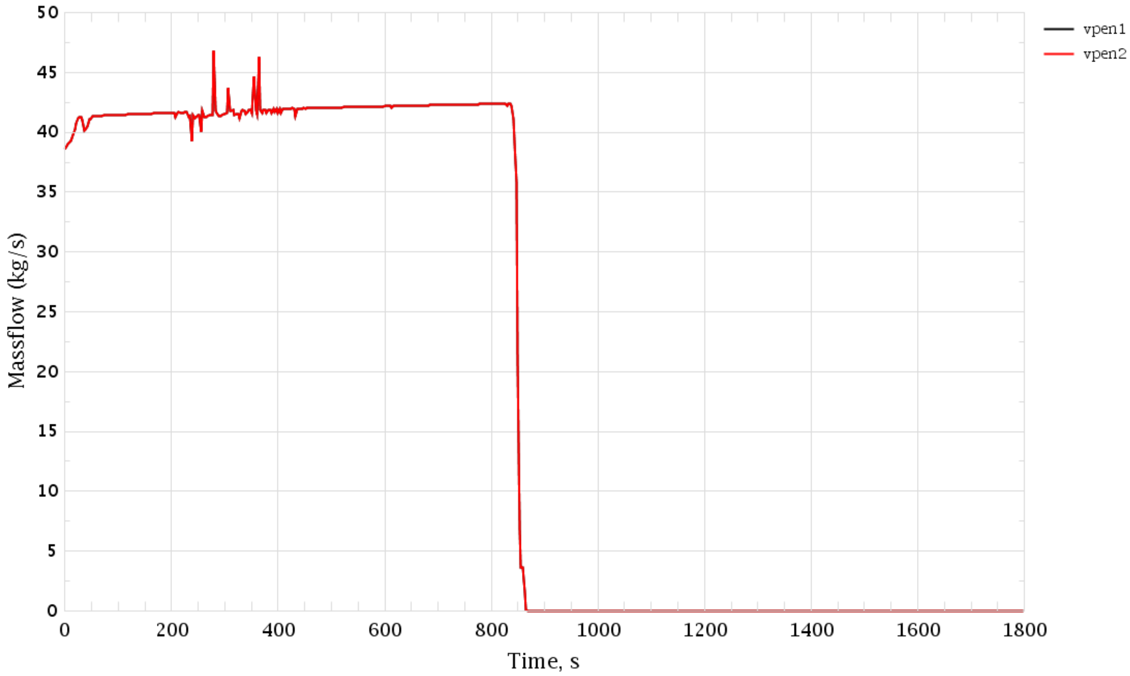

Figure 3-200 Inadvertent Opening of SG SRV in Hot Shutdown. AFW Mass Flow Rate

4 CONCLUSIONS

After validation of VVER-1000/V-320 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 these 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. 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, and quantitatively the values are in a good agreement.

Additionally, the plant incident with inadvertent opening of steam generator safety relief valve with the reactor in hot shutdown state was simulated with the TRACE code. The results obtained in this calculation are in a good agreement with plant measured data.

The results of cross-code validation calculations and simulation of plant incidents demonstrate that developed VVER-1000/V-320 thermal-hydraulic model for TRACE code is able to reproduce adequately the NPP response to transients and accidents without core melt. For the majority of plant parameters good correspondence between TRACE and RELAP5 results or plant incidents data is obtained.

Based on the results of model validation it can be concluded that developed WWER-1000/V-320 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.

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[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-0490. TRACE VVER-1000/V-320 Model Validation. International Agreement Report, 2018.

[4] 1KhME-P07-05-08-99. Khmelnitsky NPP. Incident evaluation report. Inadvertent opening of SG-1 SRV TX50S04, 1999.

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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-1000/V-320 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. Additionally, the plant incident with inadvertent opening of steam generator safety relief valve in hot shutdown is simulated.

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