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International Agreement Report

RELAP5 Simulation of Total Loss of Feedwater in Two-Loop PWR

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ABSTRACT

In Europe the design extension conditions (DEC) were introduced after the Fukushima Dai-ichi accident as preferred method for giving due consideration to the complex sequences and severe accidents without including them in the design basis conditions. The purpose of the study is to determine available elapsed time before core degradation and needed DEC safety features to prevent total loss of all feedwater (TLOFW) in a two-loop pressurized water reactor. In its documents, both WENRA (Western European Association of Nuclear Regulators) and the International Atomic Energy Agency (IAEA) present TLOFW initiating event as a possible DEC for existing nuclear power plants.

For simulations six U.S. Nuclear Regulatory Commission RELAP5 computer code versions were used to study the possible impact of code version on the results. The initiating event for TLOFW are multiple failures in which, besides the loss of main feedwater also auxiliary feedwater is lost. It is assumed that both high pressure and low pressure safety injection trains and batteries are available. Four different scenarios of TLOFW have been studied. The results section shows the comparison of calculated results obtained by several RELAP5 versions. Finally, the simulated results of total loss of feedwater with DEC safety feature available are shown.

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EXECUTIVE SUMMARY

In Europe the design extension conditions (DEC) were introduced after the Fukushima Dai-ichi accident as preferred method for giving due consideration to the complex sequences and severe accidents without including them in the design basis conditions. The purpose of the study is to determine available elapsed time before core degradation and needed DEC safety features to prevent total loss of all feedwater in a two-loop pressurized water reactor (PWR). In its documents, both WENRA (Western European Association of Nuclear Regulators) and the International Atomic Energy Agency (IAEA) present total loss of all feed water initiating event as a possible DEC for existing power plants.

The purpose of this study is to determine whether the total loss of feedwater should be considered as DEC A in a specific two-loop pressurized water reactor (PWR). Namely, the control of DEC is expected to be achieved primarily by the features implemented in the design (safety features for DEC) and not only by accident management measures that are using equipment designed for other purposes. The selected total loss of feedwater DEC has been derived from probabilistic safety assessment (PSA) according to IAEA recommendations. The initiating event is the loss of all feedwater. This means that besides the loss of main feedwater also both motor driven auxiliary feedwater (AFW) pumps and the turbine driven AFW pump are assumed to be unavailable.

Four different scenarios of total loss of feedwater have been studied. The plant is at the nuclear steam supply system (NSSS) power (2000 MW). Both trains of high pressure safety injection (HPSI), low pressure safety injection (LPSI) and accumulators are assumed to be available. The standard input deck for selected two-lop PWR has been used for RELAP5 calculations.

For simulations six U.S. Nuclear Regulatory Commission RELAP5/MOD3.3 computer code versions were used to study the possible impact of code version on the results. These versions range from RELAP5/MOD3.3 Release in February 2002 to developmental version 3.3lj from May 2022. It should be noted that in the RELAP5/MOD3.3 version 3.3lj several uncertainty parameters have been added (the parameters relate to interfacial heat transfer and wall heat transfer) and some of them have been tested.

The simulation results of total loss of feedwater showed that operator action to depressurize the primary system or a new DEC safety feature would be needed. Namely, the capacity of existing HPSI pumps is insufficient to refill the primary pressure and to maintain the long term cooldown of the core. Without considering the operator's action or the new DEC safety feature, the total loss of feedwater leads to overheating of the core in about one hour. The increasing heatup trend suggests later core damage, provided that no injection to the steam generators is established. The scenario with DEC safety feature available after 1800 s shows that further core uncovery and heatup is prevented.

ACKNOWLEDGMENTS

The financial support provided by the Slovenian Research Agency through the grant L2-1827 and the core research program P2-0026 is gratefully acknowledged. Research is also part of the CAMP project no. POG-U3-KE-R4/104/12 (NEK no. 3180019) co-funded by Krško nuclear power plant (NPP Krško) and supported by Slovenian Nuclear Safety Administration (SNSA).

ABBREVIATIONS AND ACRONYMS

AFW	auxuliary feedwater
DEC	design extension conditions
HPSI	high pressure safety injection
HTC	heat transfer coefficient
IAEA	International Atomic Energy Agency
LPSI	low pressure safety injection
MD	motor driven
MFW	main feedwater
NPP	nuclear power plant
PORV	power operated relief valve
PRZ	pressurizer
PWR	pressurized water reactor
RCP	reactor coolant pump
RL	reference level
RPV	reactor pressure vessel
SG	steam generator
SI	safety injection
TD	turbine driven
TLOFW	total loss of feedwater
U.S.NRC	U.S. Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators Association

1 INTRODUCTION

After the Fukushima Dai-ichi accident the International Atomic Energy Agency (IAEA) published standard introducing the term design extension conditions (DEC) [1], which was actually prepared prior to the Fukushima Dai-ichi accident. Specific safety standard IAEA SSR-2/1 [1] was intended to ensure higher level of safety of nuclear power plants (NPPs) taking into account the achieved state of the art in science and technology, and to reflect large consensus among the Member States. WENRA reference levels (RLs) from 2014 [2] also introduced the DEC term. The WENRA guidance document for issue F for second generation reactor [3] explains that DEC in WENRA RLs are consistent with the definition of DEC in IAEA SSR-2/1 [1], published in 2012: "*Accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions*." DEC are more complex and/or more severe than conditions postulated as design basis accidents [3].

The lessons learned from the Fukushima Dai-ichi accident have led to the reinforcement of some requirements in IAEA SSR-2/1 and revision 1 of IAEA SSR-2/1 has been released in 2016 [4]. One the main areas revised was prevention of severe accidents by strengthening the design basis for the plant. Also, the DEC term was redefined as follows: "*Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions comprise conditions in events without significant fuel degradation and conditions in events with core melting*". WENRA did not follow the newest IAEA definition of DEC in spite of the fact that modification is significant, as DEC are now defined as postulated accident conditions.

The DEC concept by IAEA and WENRA (DEC with prevention of core melting is called DEC A by WENRA) is not completely new, since in some countries selected multiple failures have already been considered in the design (through back fitting process), for example anticipated transients without scram (ATWS) and station blackout (SBO). Also, the research for beyond design basis accidents with non-degraded core (i.e. DEC A) for existing reactors has been already done in 80's and 90' of the previous century.

Slovenia implemented WENRA reference levels issue F requirements into its Rules on radiation and nuclear safety factors. In the presented analysis the loss of all feedwater has been considered as DEC A in a specific two-loop pressurized water reactor (PWR). The control of DECs is expected to be achieved primarily by features implemented in the design (safety features for DECs) and not only by accident management measures that are using equipment designed for other purposes.

The report is organized as follows. In Section 2 the RELAP5 input model and scenarios are described. Section 3 describes the results of scenarios simulated and results of sensitivity study, while conclusions are given in Section 4.

2 RELAP5 INPUT MODEL AND SCENARIOS DESCRIPTION

For calculations the following six RELAP5/MOD3.3 versions have been used, where latest official release is RELAP5/MOD3.3 Patch 5 [5]:

- RELAP5/MOD3.3 Release – version 3.3bf from February 2002
- RELAP5/MOD3.3 Patch 2 - version 3.3ef from August 2004
- RELAP5/MOD3.3 Patch 3 - version 3.3gl from March 2006
- RELAP5/MOD3.3 Patch 4 – version 3.3iy from October 2010
- RELAP5/MOD3.3 Patch 5 – version 3.3km from July 2016
- RELAP5/MOD3.3 developmental – version 3.3lj from May 2022

In calculations performed by above code versions the same RELAP5 input model has been used, presented below.

2.1 RELAP5 Input Model

For calculations six RELAP5/MOD3.3 code versions were used applying the RELAP5 input model of two-loop pressurized water reactor (PWR) that has already been used in other studies [6], [7]. A two loop PWR reactor power is 1994 MW and the nuclear steam supply system (NSSS) power is 2000 MW. The base model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points. In terms of SNAP (Symbolic Nuclear Analysis Package) this gives 304 hydraulic components and 108 heat structures. Hydraulic components in SNAP consist of both volumes and junctions, where pipe with more volumes is counted as one component. Each heat structure in SNAP connected to pipe is also counted as one component and not as many heat structures as pipe has volumes like it is counted in the RELAP5 output file. This explains the difference in the number of heat structures in Figure 2-1 and that reported in the RELAP5 output file. Besides, control variables and logical conditions (trips) represent the instrumentation, regulation (rod control, pressurizer (PRZ) level and pressure, steam dump, steam generator (SG) pressure, etc.) and protection systems (reactor protection, main feedwater (MFW) isolation, safety injection (SI) and auxiliary feedwater (AFW) triggering logic, steam line isolation, etc.).

Initial and boundary conditions for design extension conditions category A (DEC A) total loss of feedwater (TLOFW) are given in Table 2-1. Primary and secondary side parameters are input for the RELAP5/MOD3.3 transient simulation.

To analyze this scenario neither additional hydrodynamic components nor nodalization changes were introduced. Some additional logic was added to trigger the loss of main feedwater (main feedwater isolation) and to prevent the start of auxiliary feedwater motor driven (MD) and/or turbine driven (TD) pumps after main feedwater isolation initiating event.

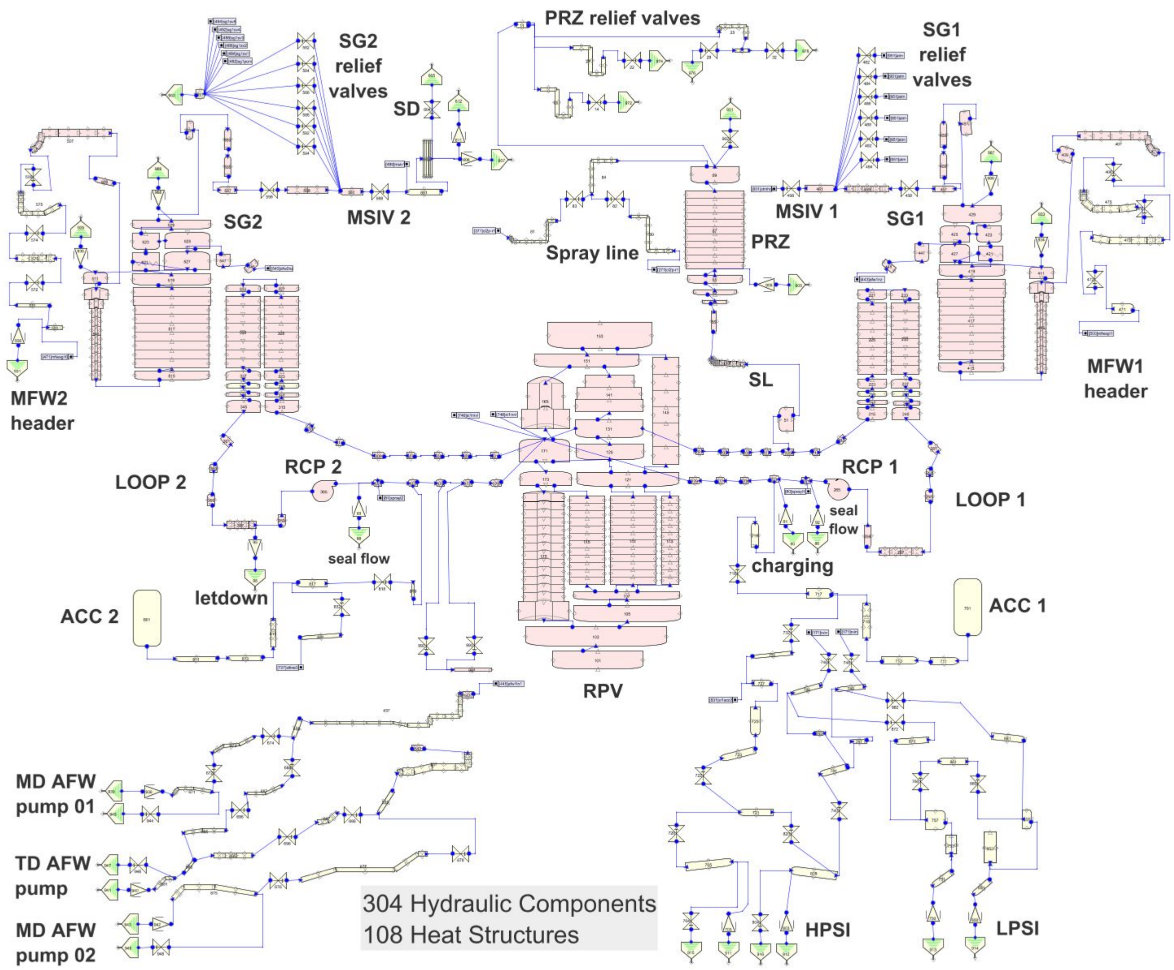

Figure 2-1 RELAP5 Two-loop PWR Hydraulic Components View

Table 2-1 Initial Conditions for DEC A TLOFW Analysis

Parameter	Two-loop PWR	RELAP5
Reactor power (MW)	1994	1994
Steam generator power (MW)	1000	996.5 / 1002.5
Pressurizer pressure (MPa)	15.51	15.51
Steam generator pressure (MPa)	6.28	6.44 / 6.44
Cold leg temperature (K)	559.2	559.51 / 559.32
Hot leg temperature (K)	596.9	596.79 / 596.79
Feedwater temperature (K)	492.6	492.5
Pressurizer level (%)	55.7	55.8
Steam generator narrow range level (%)	69.3	69.3 / 69.3
Steam mass flow (kg/s)	544.5	541.3 / 544.5

2.2 Scenarios Description

In this study the DEC A total loss of all feedwater accident starts at time 0. The initial and boundary conditions are:

- 100 % NSSS power – 2000 MW
- availability of both trains of high pressure safety injection, low pressure safety injection and accumulators (HPSI, LPSI and ACC, respectively),
- both motor driven auxiliary feedwater (AFW) pumps as well as turbine driven AFW pump are unavailable.

Total loss of all feedwater accident presents design extension condition where complete loss of feedwater flow into both steam generators occurs. For above conditions, four scenarios shown in Table 2-4 have been studied. In first scenario (labelled TLOFW-N) the reactor coolant pump was assumed initially and later tripped according to the emergency operating procedures (EOP) and control systems available have been assumed, as it turns out that these normal operation systems have a negative impact on the course of the accident. This is in conformance with IAEA specific safety guide SSG-2 (Rev. 1) [8]. In the second scenario no operator actions and no normal operation systems, including control systems have been assumed available (labelled 'TLOFW-WN'). Therefore, the primary and secondary inventory is discharged through pressurizer and SG safety valves, respectively. In the third scenario labeled 'TLOFW-RCP', safety systems were available with reactor coolant pumps assumed available. Finally, in the fourth scenario, DEC safety feature has been assumed available after 1800 s in addition to the TLOFW-N assumptions.

Table 2-2 Scenarios Simulated

Scenario name	Label	Description
Total loss of feedwater with normal systems available	TLOFW-N	In addition to safety systems available, control systems are assumed to be operable, while reactor coolant pumps (RCP) are tripped in accordance with plant emergency operating procedures (this scenario was used for plant full scope simulator verification [9]).
Total loss of feedwater without normal systems available	TLOFW-WN	Only safety systems are available.
Total loss of feedwater without normal systems (except RCP) available	TLOFW-RCP	Same as TLOFW-WN except that reactor coolant pumps are running.
Total loss of feedwater with normal systems available plus DEC safety feature	TLOFW-DEC	Same as TLOFW-N, with DEC safety feature (alternate auxiliary feedwater pump) available after 1800 s.

Each of the four scenarios shown in Table 2-2 has been simulated by six different RELAP5 versions (see Table 2-3) developed in the last 20 years. This give 24 simulations, which are described in Table 2-4. The calculation labels are used on the plots in the results section.

Table 2-3 RELAP5 Versions Used for Simulations

RELAP5 code used	version and year of release	Calculation label used
RELAP5/MOD3.3 Release	version 3.3bf from February 2002	XXXX_rel
RELAP5/MOD3.3 Patch 2	version 3.3ef from August 2004	XXXX_P2
RELAP5/MOD3.3 Patch 3	version 3.3gl from March 2006	XXXX_P3
RELAP5/MOD3.3 Patch 4	version 3.3iy from October 2010	XXXX_P4
RELAP5/MOD3.3 Patch 5	version 3.3km from July 2016	XXXX_P5
RELAP5/MOD3.3 developmental	version 3.3lj from May 2022	XXXX_lj

where XXXX is label of the scenario (TLOFW-N, TLOFW-WN, TLOFW-RCP or TLOFW-DEC

Table 2-4 Performed Simulations

Scenario name	RELAP5 code version used	Calculation label used
Total loss of feedwater with normal systems available (TLOFW-N)	RELAP5/MOD3.3 Release	TLOFW-N_rel
	RELAP5/MOD3.3 Patch 2	TLOFW-N_P2
	RELAP5/MOD3.3 Patch 3	TLOFW-N_P3
	RELAP5/MOD3.3 Patch 4	TLOFW-N_P4
	RELAP5/MOD3.3 Patch 5	TLOFW-N_P5
	RELAP5/MOD3.3 version lj	TLOFW-N_lj
Total loss of feedwater without normal systems available (TLOFW-WN)	RELAP5/MOD3.3 Release	TLOFW-WN_rel
	RELAP5/MOD3.3 Patch 2	TLOFW-WN_P2
	RELAP5/MOD3.3 Patch 3	TLOFW-WN_P3
	RELAP5/MOD3.3 Patch 4	TLOFW-WN_P4
	RELAP5/MOD3.3 Patch 5	TLOFW-WN_P5
	RELAP5/MOD3.3 version lj	TLOFW-WN_lj
Total loss of feedwater without normal systems (except reactor coolant pump (RCP) available) (TLOFW-RCP)	RELAP5/MOD3.3 Release	TLOFW-RCP_rel
	RELAP5/MOD3.3 Patch 2	TLOFW-RCP_P2
	RELAP5/MOD3.3 Patch 3	TLOFW-RCP_P3
	RELAP5/MOD3.3 Patch 4	TLOFW-RCP_P4
	RELAP5/MOD3.3 Patch 5	TLOFW-RCP_P5
	RELAP5/MOD3.3 version lj	TLOFW-RCP_lj
Total loss of feedwater with normal systems available plus DEC safety feature (TLOFW-DEC)	RELAP5/MOD3.3 Release	TLOFW-DEC_rel
	RELAP5/MOD3.3 Patch 2	TLOFW-DEC_P2
	RELAP5/MOD3.3 Patch 3	TLOFW-DEC_P3
	RELAP5/MOD3.3 Patch 4	TLOFW-DEC_P4
	RELAP5/MOD3.3 Patch 5	TLOFW-DEC_P5
	RELAP5/MOD3.3 version lj	TLOFW-DEC_lj

For the 'RELAP5/MOD3.3 version lj' developmental version sensitivity study varying one uncertainty parameter at a time has been performed for the TLOFW-RCP scenario. The uncertainty parameters built into the 'RELAP5/MOD3.3 version lj' developmental version' have been varied (minimum value 0.8, maximum value 1.2):

- Liquid heat transfer (HTC and heat flux) – label 'HT-L';
- Gas heat transfer (HTC and heat flux) – label 'HT-G';
- Volumetric wetted wall liquid interface heat transfer coefficient – label 'HIF';
- Volumetric wetted wall gas interface heat transfer coefficient – label 'HIG';
- Volumetric wetted wall direct liquid-gas heat transfer coefficient – label 'HGF';
- Junction form loss value – label 'FLOSS';
- Two-phase friction multiplier – label 'WDRAG'.

3 RESULTS

Results of simulations are shown in Figures 3-1 through 3-20 for scenarios TLOFW-N, TLOFW-WN, TLOFW-RCP, TLOFW-DEC, respectively and sensitivity study.

3.1 Results for TLOFW-N

This scenario has been selected because it was used for Krško full scope simulator verification [9]. The scenario in the [9] was considered as beyond design basis accident.

The results are shown in Figures 3-1 through 3-3 for the secondary side, the pressurizer and the primary side related parameters, respectively. The secondary side related parameters are steam generator (SG) no. 1 pressure, SG no. 1 wide range level, SG no. 1 steam flow, SG no. 1 main feedwater flow, steam dump flow and SG no. 1 power operated relief valve (PORV) flow. The pressurizer related parameters are pressurizer pressure, pressurizer level, pressurizer spray no. 1 mass flow rate, pressurizer spray no. 2 mass flow rate, pressurizer PORV no. 1 discharge mass flow rate and pressurizer PORV no. 2 discharge mass flow rate. Finally, the primary side related parameters are core collapsed liquid level, cladding temperature at 2.29 m in the core, cold leg no. 1 liquid temperature, hot leg no. 1 liquid temperature, core power and cold leg no. 1 mass flow rate.

It can be seen that different RELAP5 versions have very small influence on the calculated results. The TLOFW accident is started by manual main feedwater (MFW) isolation, causing MFW pump no. 1 and no. 2 trips, pressure increase resulting in the steam dump discharge and reactor trip, followed by turbine trip. All this happened in the first minute of the accident. In the 10th minute the safety injection (SI) signal is generated, causing steam line isolation, charging isolation, HPSI pump no. 1 and no. 2 injection, while LPSI pump no. 1 and no. 2 are operating, but not injecting because of high primary pressure. After around 24 minutes the reactor coolant pump no. 1 and no. 2 are tripped. In TLOFW with DEC safety feature available the injection to both steam generator is started after 1800 s (at this time the level in the core started to drop significantly).

Initially, after the MFW pumps trip on manual MFW isolation, drop in feedwater flow (see Figure 3-1(d)) caused that both the secondary and primary pressure started to increase (see Figure 3-1(a) and Figure 3-2(a), respectively).

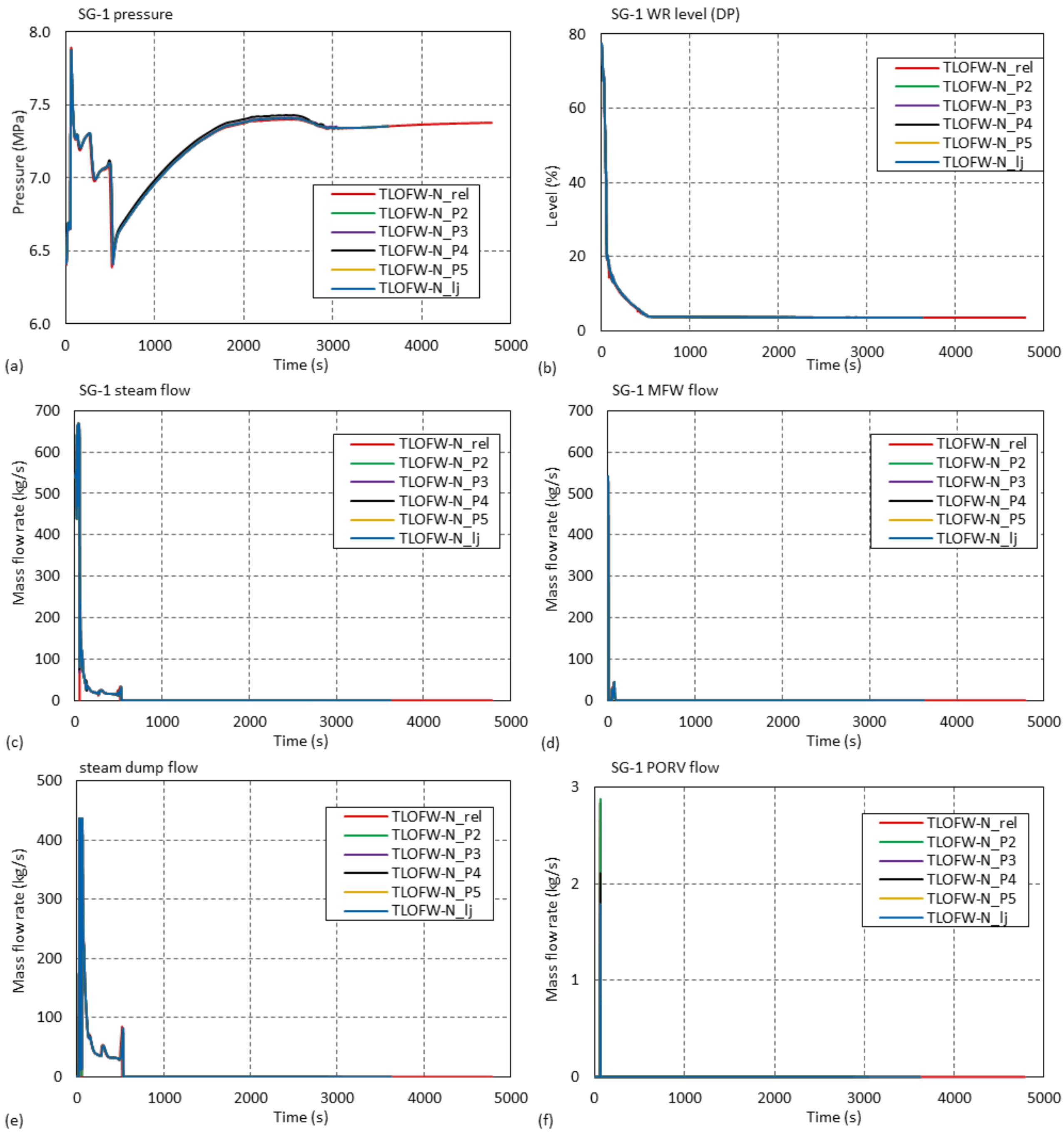

Figure 3-1 TLOFW-N Secondary Side Related Parameters - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Steam Generator No. 1 Steam Flow, (d) Steam Generator No. 1 Main Feedwater Flow, (e) Steam Dump Flow, (f) Steam Generator No. 1 PORV Flow

Both lines of pressurizer spray lines were activated to reduce the primary pressure (see Figure 3-2(c) and 3-2(d)). On the secondary side, the steam is dumped to steam dump system (see Figure 3-1(e)), causing loss of steam generator inventory, resulting in steam generator (SG) level decrease (see Figure 3-1(b)). When low-low level setpoint in the steam generator is reached (set to 13 % narrow range (NR) span), the reactor trip occurred, causing turbine trip.

The primary pressure sharply decreased (see Figure 3-2(a)) due to the reactor trip, while the secondary pressure sharply increased due to the turbine trip (see Figure 3-1(a)).

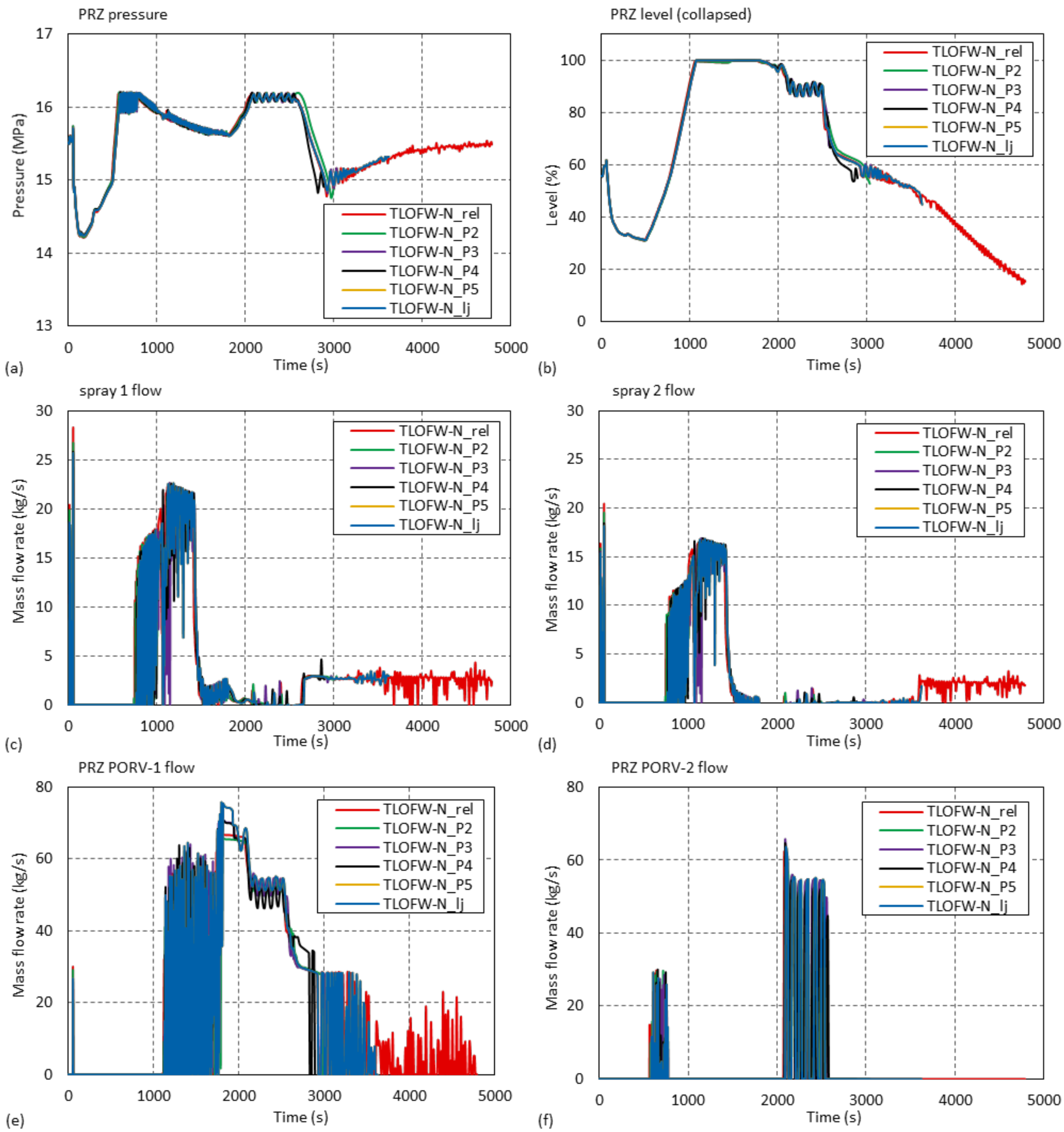

Figure 3-2 TLOFW-N Pressurizer Related Parameters – (a) Pressurizer Pressure, (b) Pressurizer Level, (c) Pressurizer Spray No. 1 Mass Flow Rate, (d) Pressurizer Spray No. 2 Mass Flow Rate, (e) Pressurizer PORV No. 1 Discharge Mass Flow Rate, (f) Pressurizer PORV No. 2 Discharge Mass Flow Rate

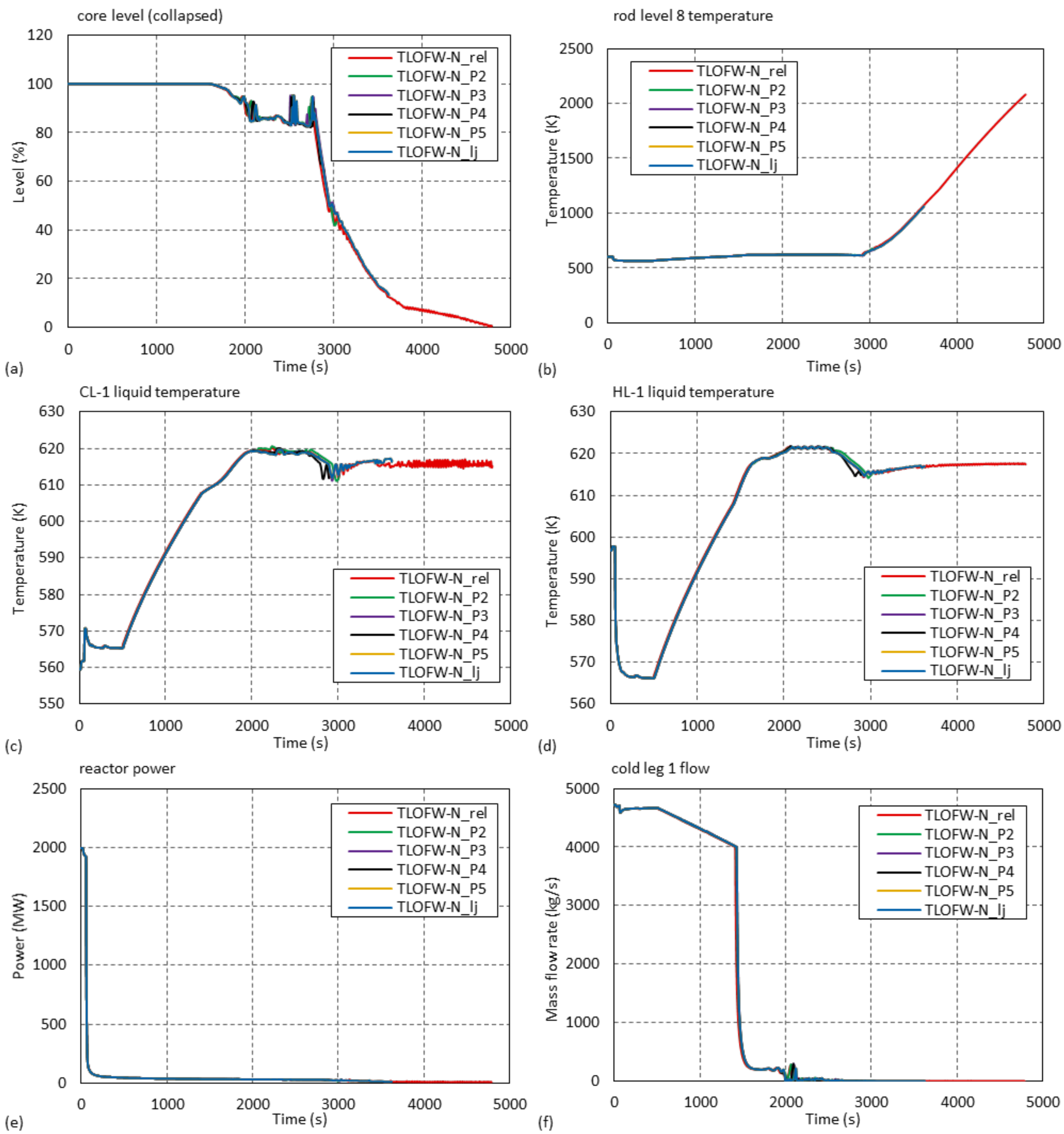

Figure 3-3 TLOFW-N Primary Side Related Parameters – (a) Core Collapsed Liquid Level, (b) Cladding Temperature at 2.29 m in the Core, (c) Cold Leg No. 1 Liquid Temperature, (d) Hot Leg No. 1 Liquid Temperature, (e) Core Power, (f) Cold Leg No. 1 Mass Flow Rate

Since all feedwater is lost, there is no steam generators filling (see Figure 3-1(b)). The pressurizer (PRZ) pressure rate sensitive PORV no. 1 initially discharged briefly (Figure 3-2(e)) and the steam generators PORV (see SG no. 1 PORV on Figure 3-2(f)). During the initial transient stage and later, after the reactor trip until the safety injection signal isolation causing the steam line isolation, the steam dump provided the continuous heat sink (Figure 3-1(e)). The second PRZ PORV first opened around 600 s (see Figure 3-2(f)), after the steam dump flow termination, and again later approximately between 2100 s and 2500 s.

The second PRZ PORV opened on the set pressure 16.2 MPa, while first PRZ rate sensitive PORV reopened after 1100 seconds and remained open till the end of calculation (see Figure 3-2(e)). When both, first and second PRZ PORVs are discharging, the PRZ rate sensitive PORV discharge flow is oscillatory, while in case where only the PRZ rate sensitive PORV opened, the discharged flow is continuous (see Figures 3-2(e) and 3-2(f)). After the saturation of the primary coolant at 1730 s, the primary pressure increased (see Figure 3-2(a)) because relief valves could no longer compensate the volume swell of the primary coolant.

After the emptying of steam generators, the primary temperature (see Figures 3-3(c) and (d)) and pressure (see Figure 3-2(a)) started again to increase, resulting in core uncovery start as shown in Figure 3-3(a). In spite of HPSI pumps and charging pumps operating the core uncovering could not be prevented as the primary pressure (see Figure 3-2(a)) became higher than the shutoff head of HPSI and charging (chemical and volume control system) pumps. As can be seen from Figure 3-3b, core heat-up started after 2930 s and lasted until the calculation was terminated due to a code failure ("Reactor kinetics time step reduced below minimum value of 1.0E-07, problem terminated" – reactor kinetics time step is inherent to computer code).

3.2 Results for TLOFW-WN

The results are shown in Figures 3-4 through 3-6 for the secondary side, the pressurizer and the primary side related parameters, respectively. Same set of parameters is shown as described in Section 3.1. As can be seen from Figures 3-4 through 3-6, the big difference to the previous scenario TLOFW-N is that only safety systems are used what resulted also in the reactor coolant pump unavailability. Reactor coolant pump not available caused reactor coolant flow low-low signal generation resulting in reactor trip. This is beneficial for the transient progression after loss of all feedwater, because less heat is generated in the reactor after the accident start. The steam generator initial inventory loss is not so step and the boiling off process is a bit slower comparing to the case with RCP pumps operable (TLOFW-N), as heat addition with the two pumps is not negligible (app. 6 MW versus 51.4 MW of decay heat at 300 s). For this reason the boil-off time extends for around one hour, resulting in later core heatup.

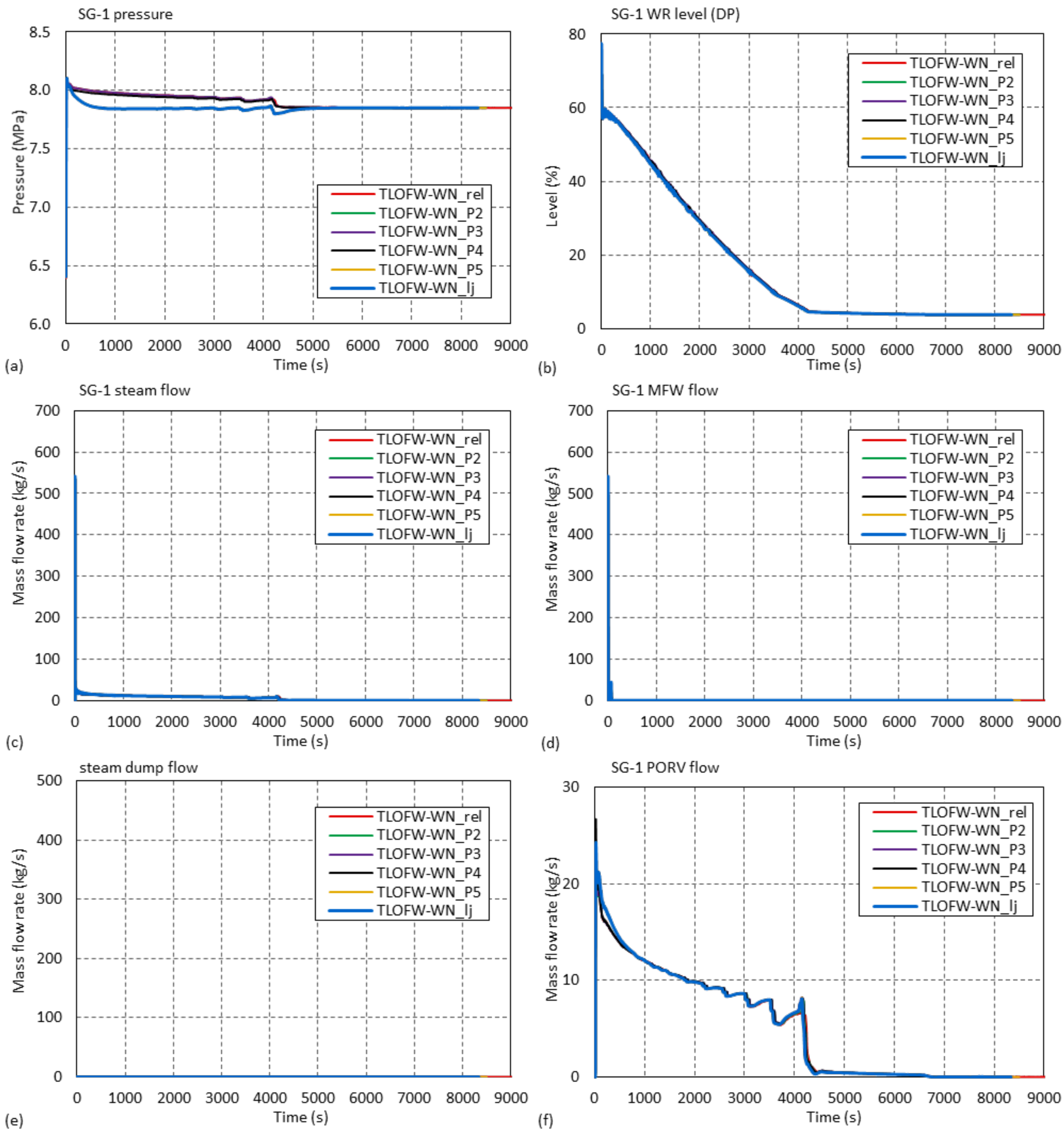

Figure 3-4 TLOFW-WN Secondary Side Related Parameters - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Steam Generator No. 1 Steam Flow, (d) Steam Generator No. 1 Main Feedwater Flow, (e) Steam Dump Flow, (f) Steam Generator No. 1 PORV flow

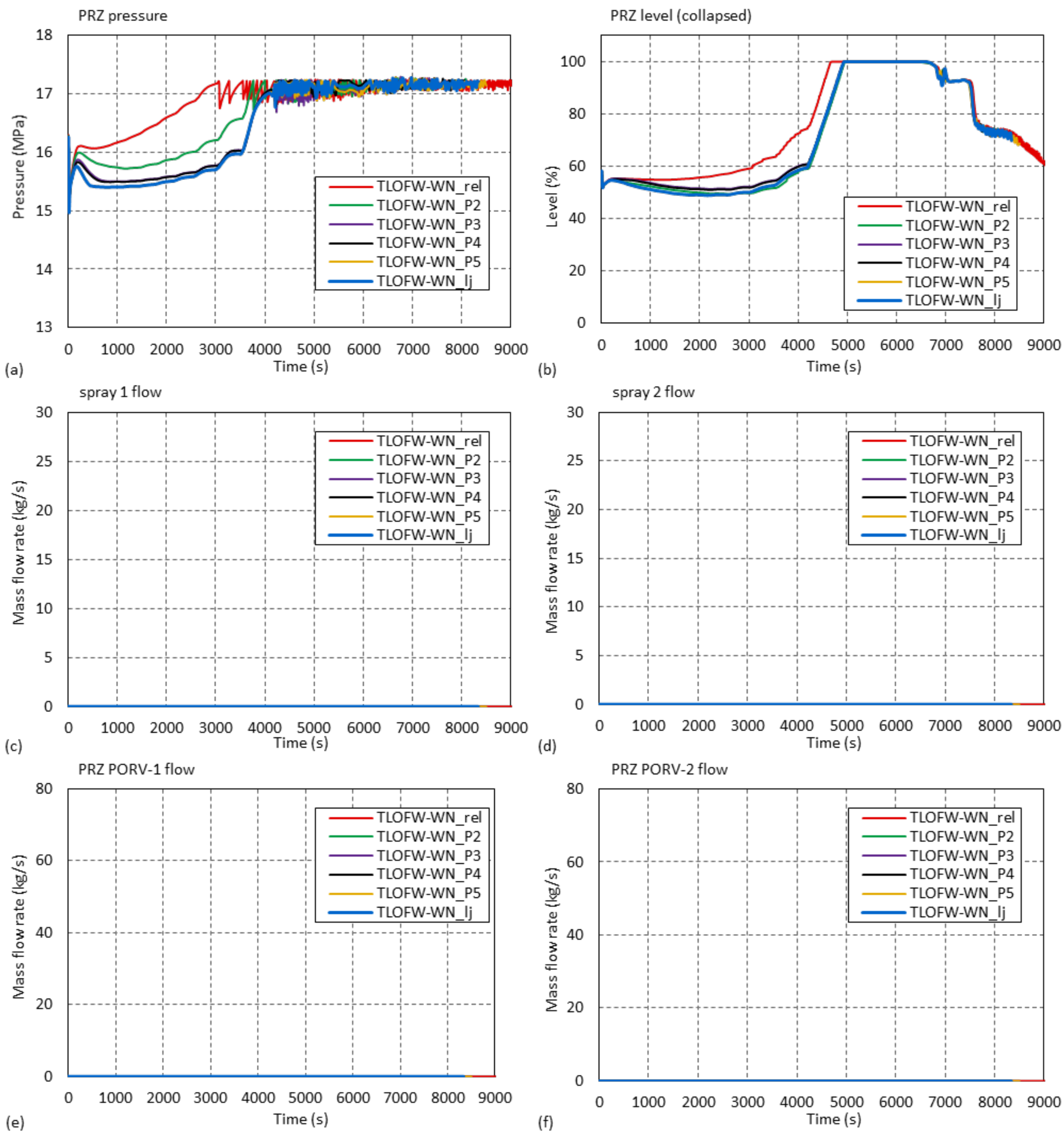

Figure 3-5 TLOFW-WN Pressurizer Related Parameters – (a) Pressurizer Pressure, (b) Pressurizer Level, (c) Pressurizer Spray No. 1 Mass Flow Rate, (d) Pressurizer Spray No. 1 Mass Flow Rate, (e) Pressurizer PORV No. 1 Discharge Mass Flow Rate, (f) Pressurizer PORV No. 2 Discharge Mass Flow Rate

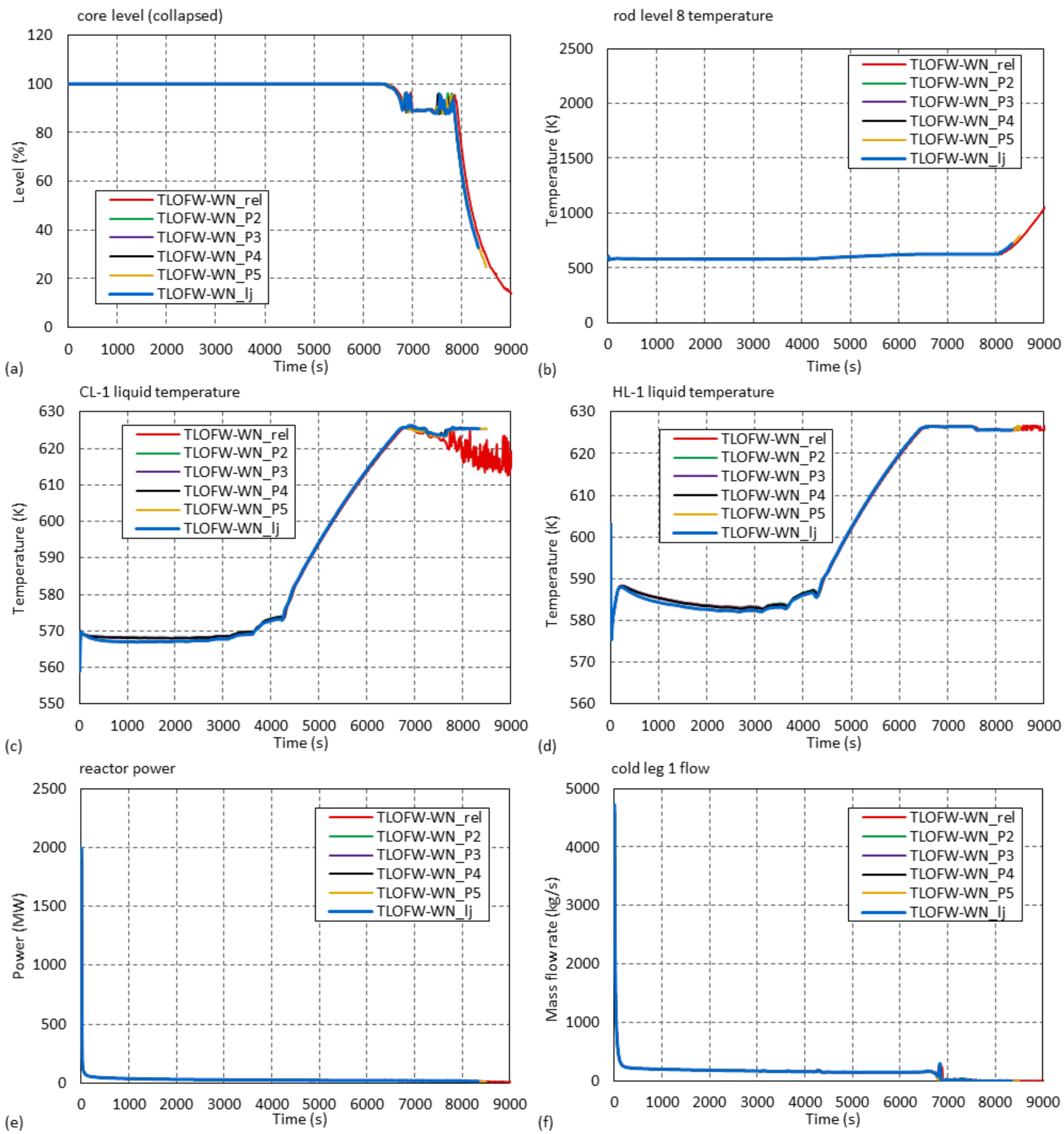

Figure 3-6 TLOFW-WN Primary Side Related Parameters – (a) Core Collapsed Liquid Level, (b) Cladding Temperature at 2.29 m in the Core, (c) Cold Leg No. 1 Liquid Temperature, (d) Hot Leg No. 1 Liquid Temperature, (e) Core Power, (f) Cold Leg No. 1 Mass Flow Rate

3.3 Results for TLOFW-RCP

To show that assuming RCP operation in addition to safety systems is conservative, the TLOFW-RCP scenario has been simulated. The results are shown in Figures 3-8 through 3-10 for the secondary side, the pressurizer and the primary side related parameters, respectively. Same set of parameters is shown as described in Section 3.1. As can be seen from Figures 3-8 through 3-10, the big difference to the previous scenario TLOFW-WN is that RCP pumps available do not result in reactor coolant flow low-low signal generation. The reactor is tripped on low-low steam generator level 36 s after transient start as shown in Figure 3-7(a), where reactor trip can be indicated from dropping reactor power. Figure 3-7(b) shows that initially for scenario TLOFW-RCP much more liquid boils off due to operating reactor and that level decrease is steeper for TLOFW-RCP scenario comparing to TLOFW-WN due to RCP pumps heat addition to reactor coolant system.

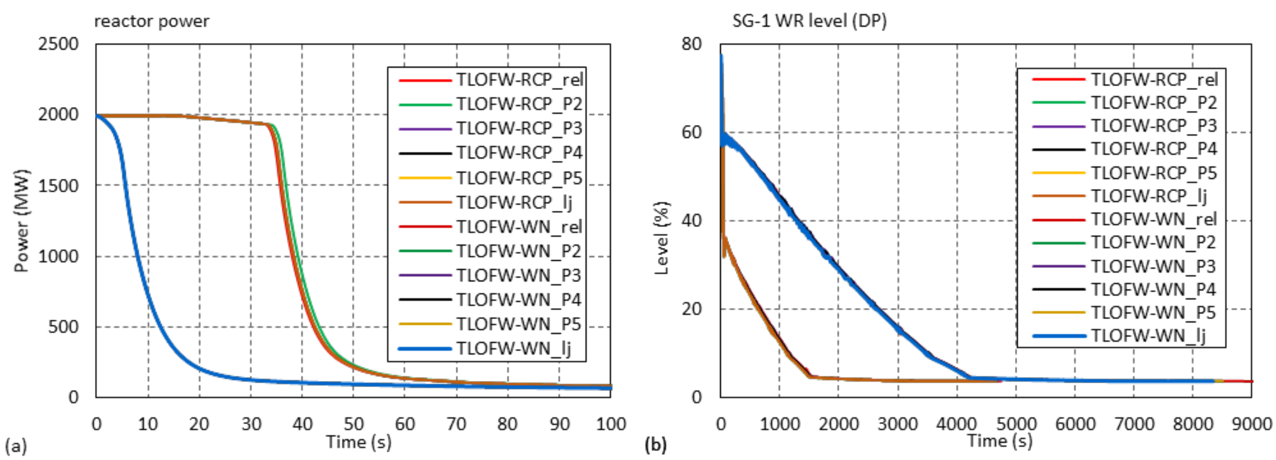

Figure 3-7 Comparing of Reactor Power and SG-1 Wide Range Level Between TLOFW-RCP and TLOFW-WN Scenarios

Due to pump operation there is faster steam generators boil-off (see Figure 3-8(b)) what causes primary pressure increase (see Figure 3-9(a)) and pressurizer safety valves open and start to discharge reactor coolant system liquid inventory. This leads to start of core uncover (see Figure 3-10(a) in less than one hour.

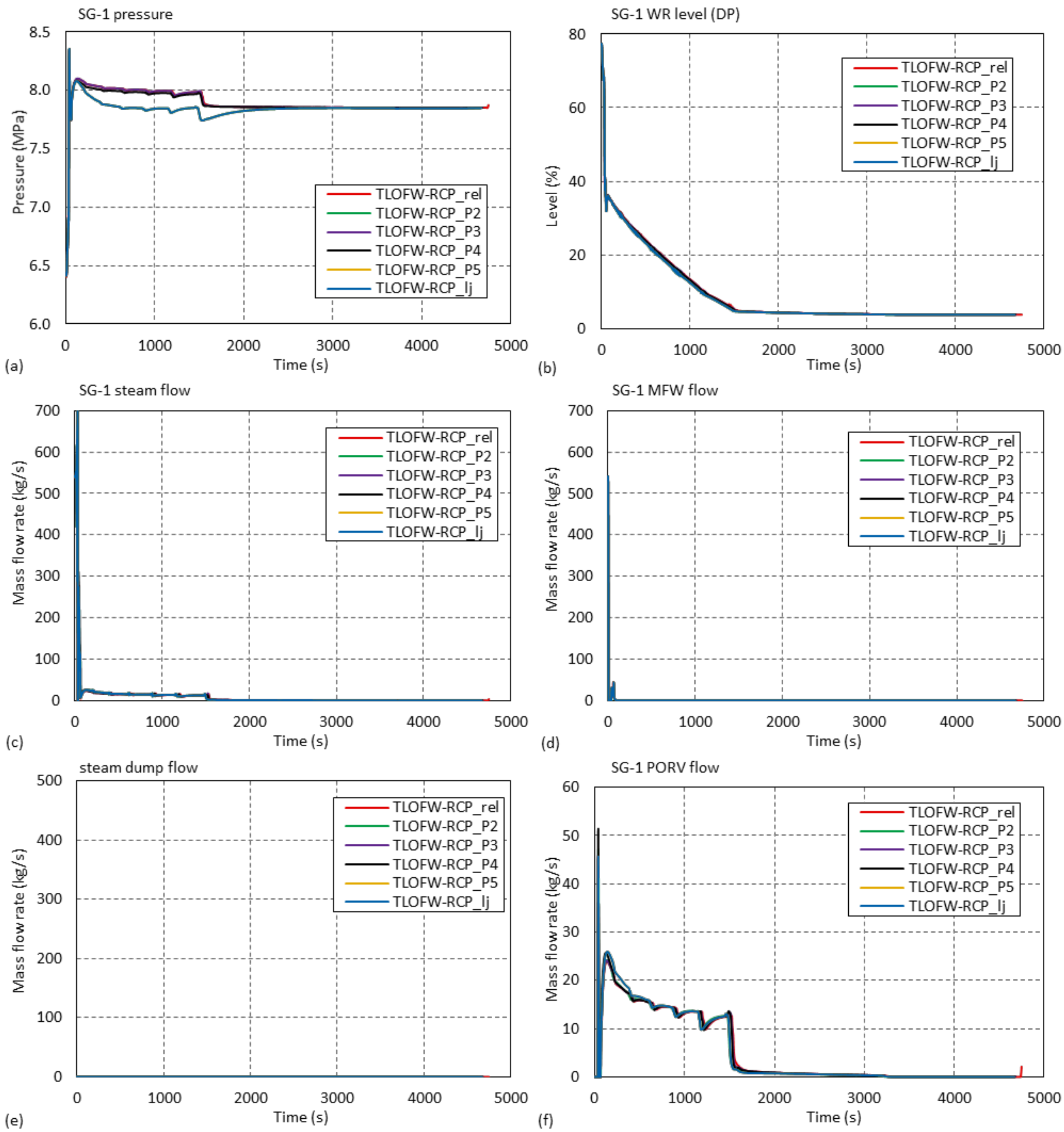

Figure 3-8 TLOFW-RCP Secondary Side Related Parameters - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Steam Generator No. 1 Steam Flow, (d) Steam Generator No. 1 Main Feedwater Flow, (e) Steam Dump Flow, (f) Steam Generator No. 1 PORV flow

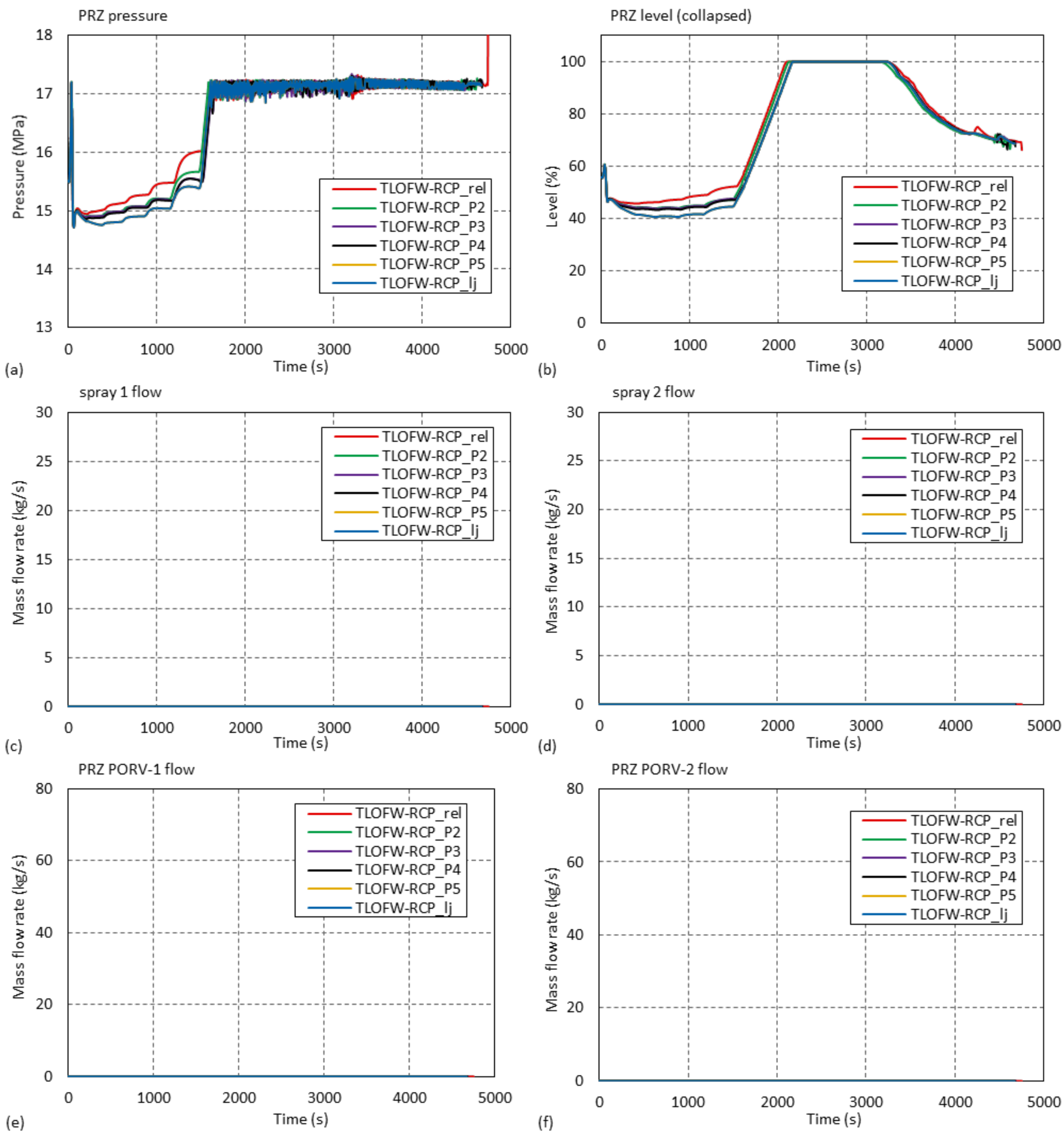

Figure 3-9 TLOFW-RCP Pressurizer Related Parameters – (a) Pressurizer Pressure, (b) Pressurizer Level, (c) Pressurizer Spray No. 1 Mass Flow Rate, (d) Pressurizer Spray No. 1 Mass Flow Rate, (e) Pressurizer PORV No. 1 Discharge Mass Flow Rate, (f) Pressurizer PORV No. 2 Discharge Mass Flow Rate

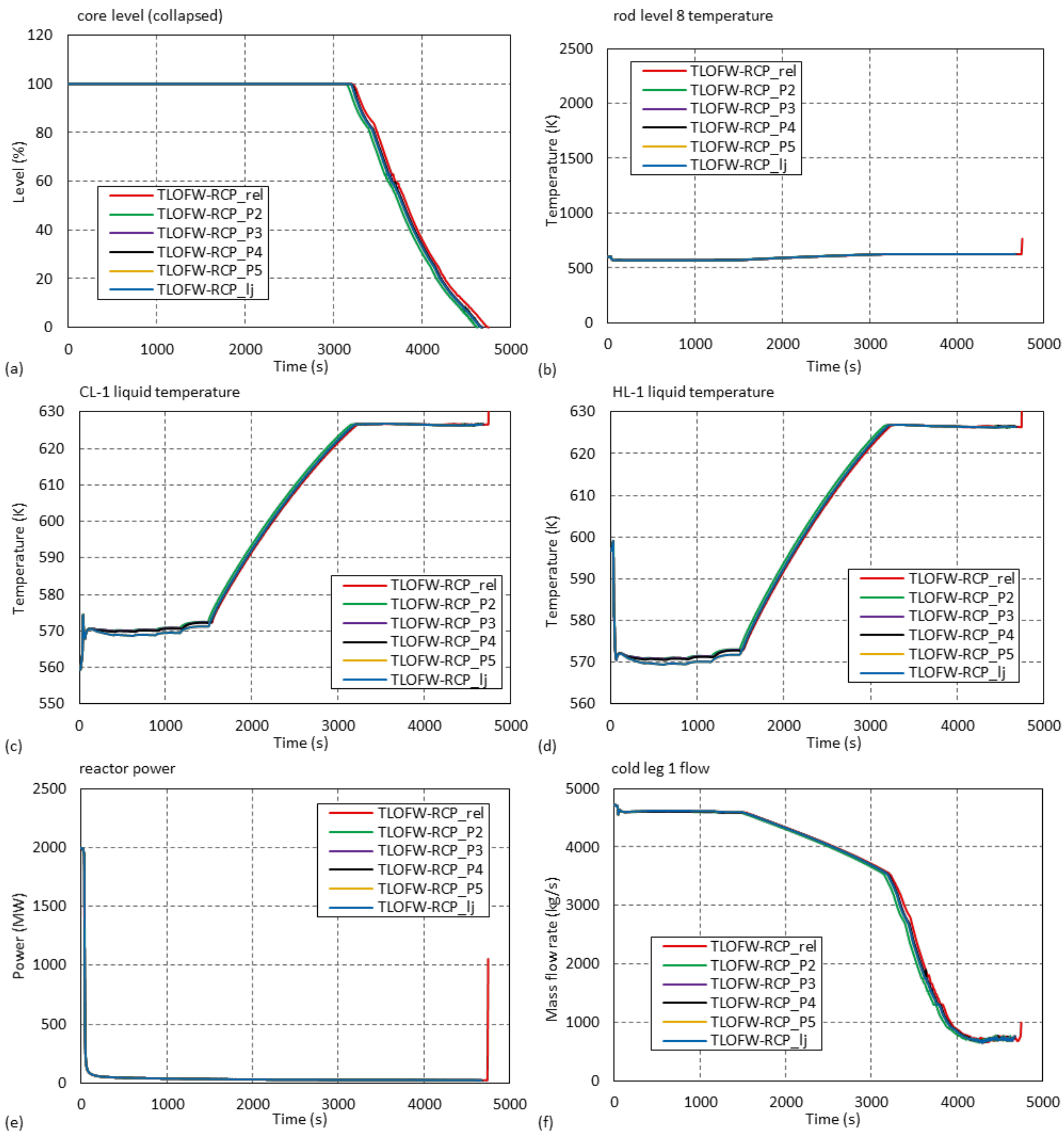

Figure 3-10 TLOFW-WN Primary Side Related Parameters – (a) Core Collapsed Liquid Level, (b) Cladding Temperature at 2.29 m in the Core, (c) Cold Leg No. 1 Liquid Temperature, (d) Hot Leg No. 1 Liquid Temperature, (e) Core Power, (f) Cold Leg No. 1 Mass Flow Rate

3.4 Results for TLOFW-DEC

The results are shown in Figures 3-11 through 3-13 for the secondary side, the pressurizer and the primary side related parameters, respectively. The assumptions for scenario TLOFW-DEC are the same as for TLOFW-N, except that in this case DEC safety feature (alternative auxiliary feedwater) is assumed available. When feeding of both steam generators started, the steam generators start to fill (see Figure 3-11(b)), heat transfer from primary side is established and therefore pressurizer pressure started to decrease (see Figure 3-12(a) and drop below safety valve closing setpoint before 3000 s. The core collapsed liquid level start to recover (see Figure 3-13(a)) and long term decay heat removal is established.

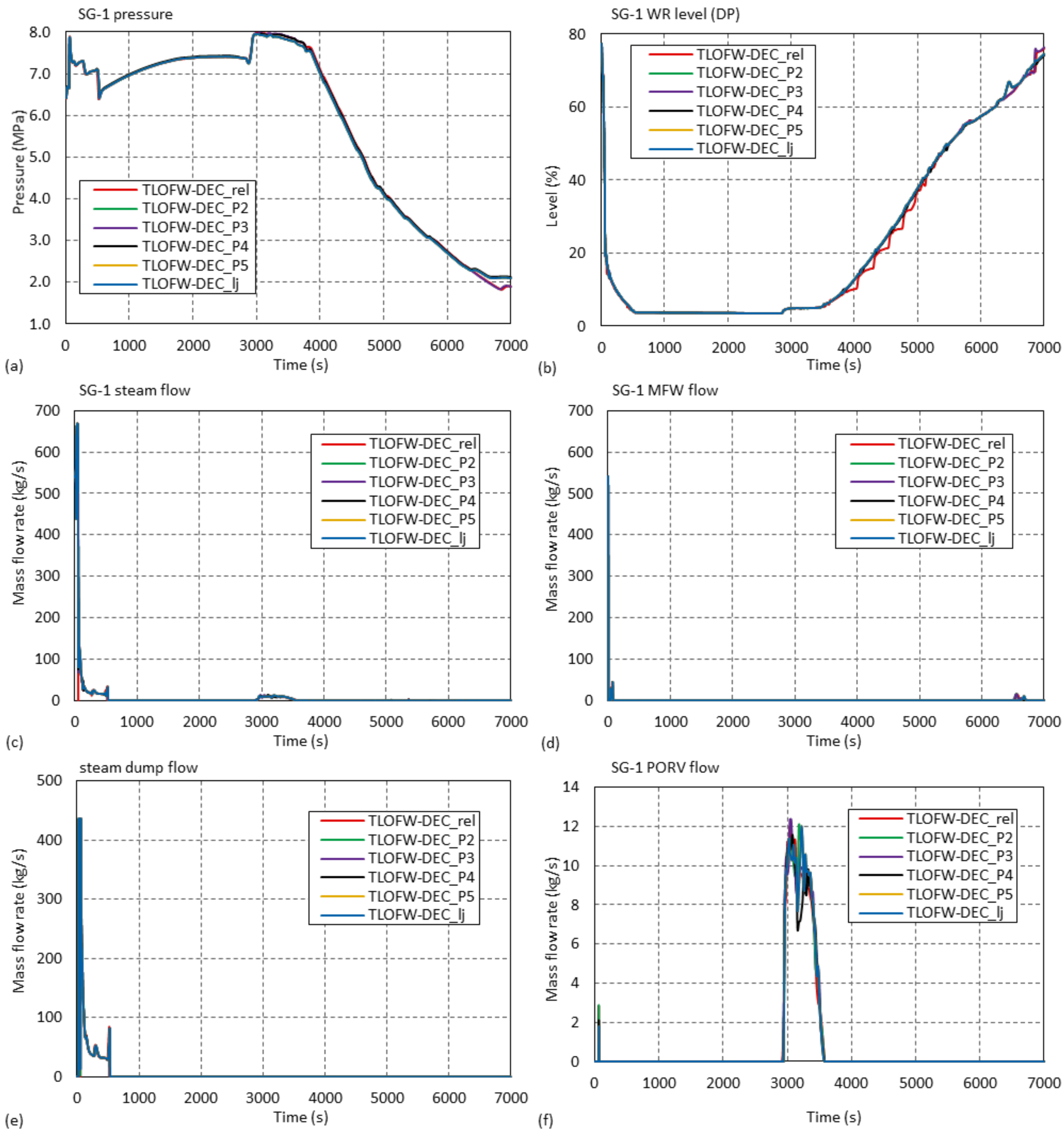

Figure 3-11 TLOFW-DEC Secondary Side Related Parameters - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Steam Generator No. 1 Steam Flow, (d) Steam Generator No. 1 Main Feedwater Flow, (e) Steam Dump Flow, (f) Steam Generator No. 1 PORV flow

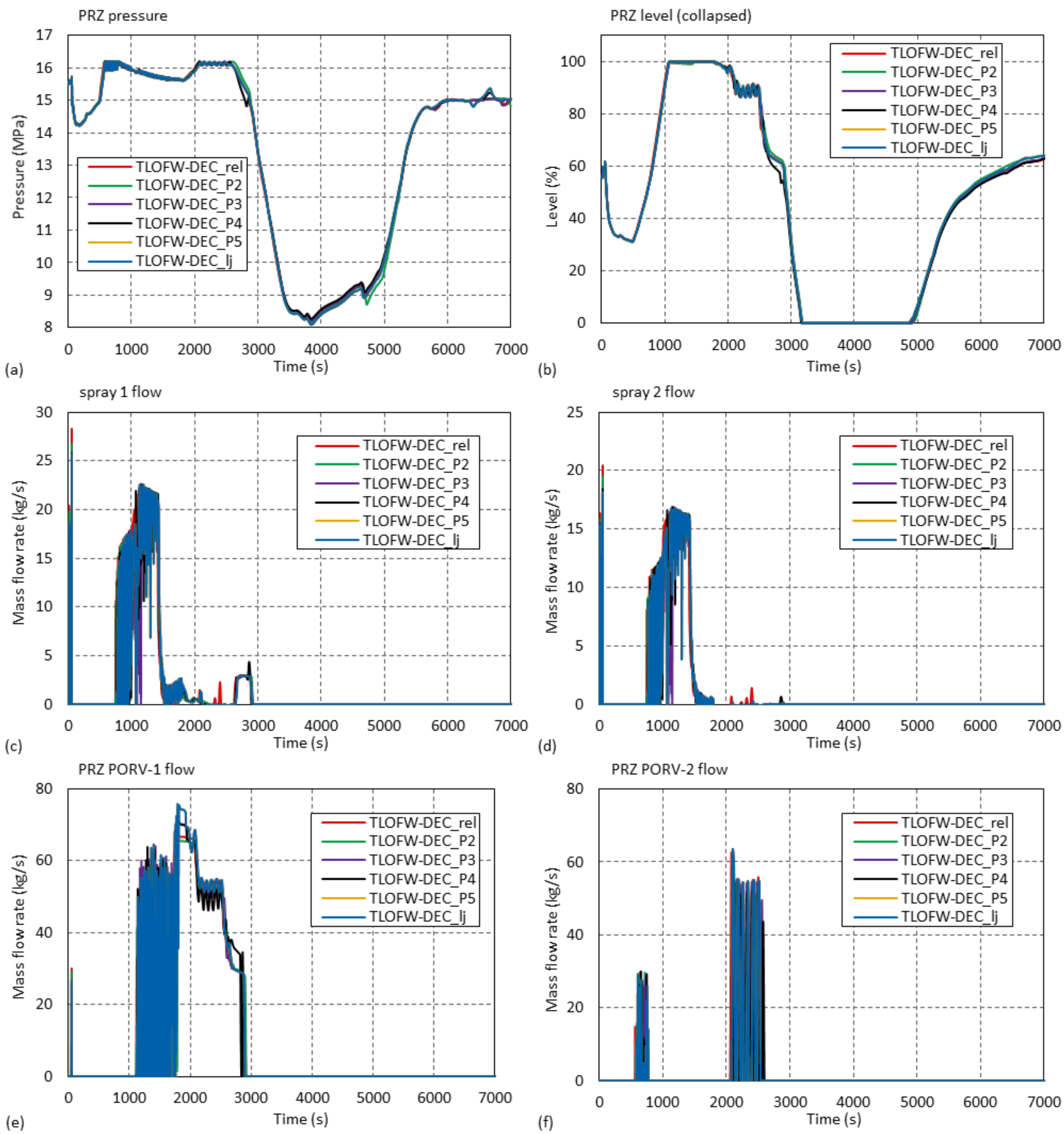

Figure 3-12 TLOFW-DEC Pressurizer Related Parameters – (a) Pressurizer Pressure, (b) Pressurizer Level, (c) Pressurizer Spray No. 1 Mass Flow Rate, (d) Pressurizer Spray No. 1 Mass Flow Rate, (e) Pressurizer PORV No. 1 Discharge Mass Flow Rate, (f) Pressurizer PORV No. 2 Discharge Mass Flow Rate

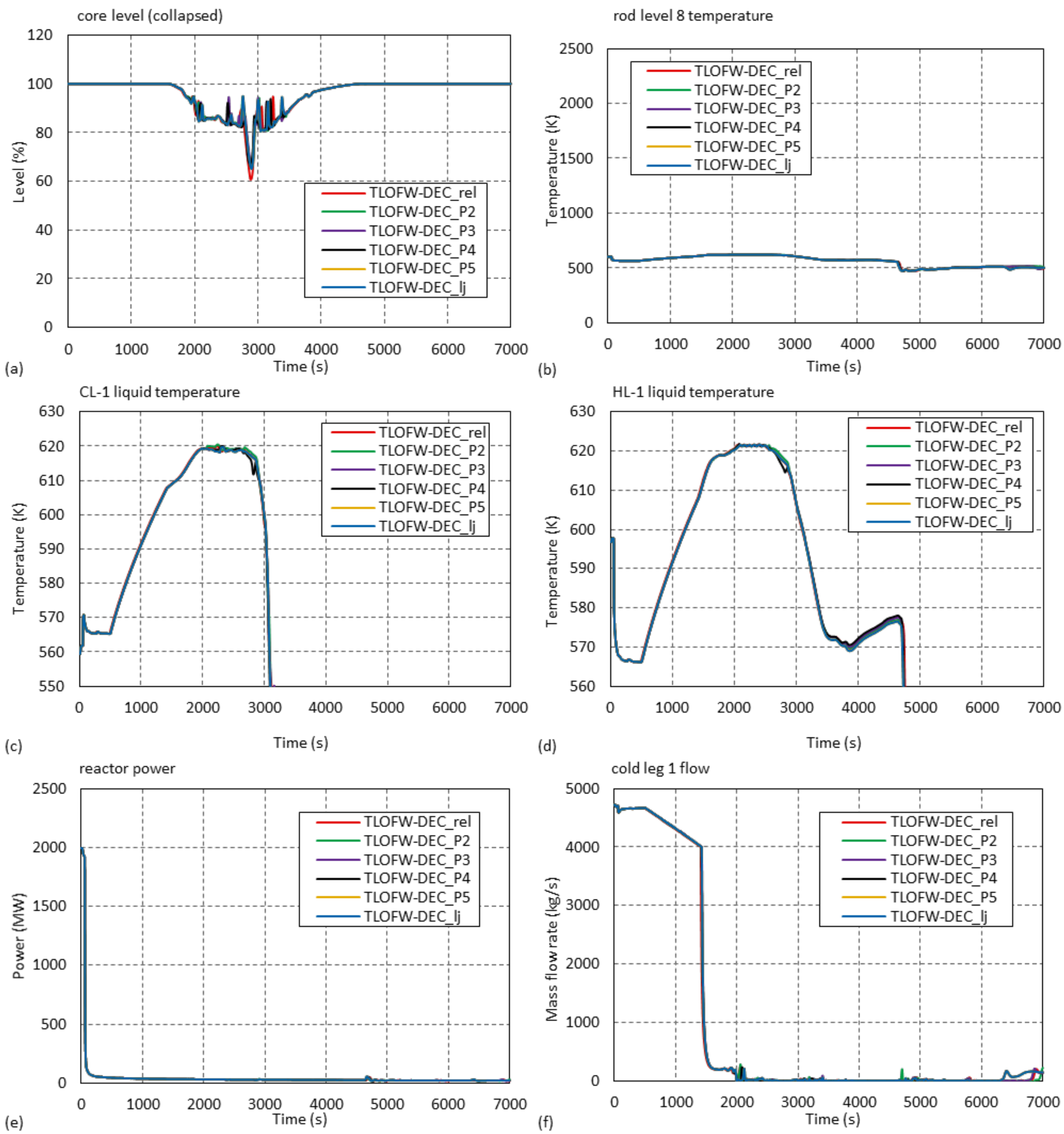

Figure 3-13 TLOFW-DEC Primary Side Related Parameters – (a) Core Collapsed Liquid Level, (b) Cladding Temperature at 2.29 m in the Core, (c) Cold Leg No. 1 Liquid Temperature, (d) Hot Leg No. 1 Liquid Temperature, (e) Core Power, (f) Cold Leg No. 1 Mass Flow Rate

3.5 Results for Sensitivity Study for TLOFW-RCP

Finally, sensitivity study is presented for the TLOFW-RCP scenario calculated by recent RELAP5/MOD3.3 version lj. As described in Section 2.2, seven uncertainty parameters are varied ±20 % of its nominal value. Figures 3-14 through 3-20 show the impact of liquid heat transfer (HTC and heat flux), gas heat transfer (HTC and heat flux), volumetric wetted wall liquid interface heat transfer coefficient, volumetric wetted wall gas interface heat transfer coefficient, volumetric wetted wall direct liquid-gas heat transfer coefficient, junction form loss value, and two-phase friction multiplier, respectively. The impact of uncertainty parameters on SG no. 1 pressure, SG no. 1 wide range level, pressurizer pressure, pressurizer level, core collapsed liquid level, and cold leg no. 1 mass flow rate is studied. It can be seen that the impact is such that the results change only quantitatively, while qualitatively nothing is changed. Uncertainty parameters gas heat transfer (HTC and heat flux) labeled 'HT-G', volumetric wetted wall gas interface heat transfer coefficient labelled 'HIG', volumetric wetted wall direct liquid-gas heat transfer coefficient labelled 'HGF', and two-phase friction multiplier labelled 'WDRAG' have very small impact on the results. On the other hand, liquid heat transfer (HTC and heat flux) labelled 'HT-L', volumetric wetted wall liquid interface heat transfer coefficient labelled 'HIF' and junction form loss value labelled 'FLOSS' have small but visible impact on the calculated results.

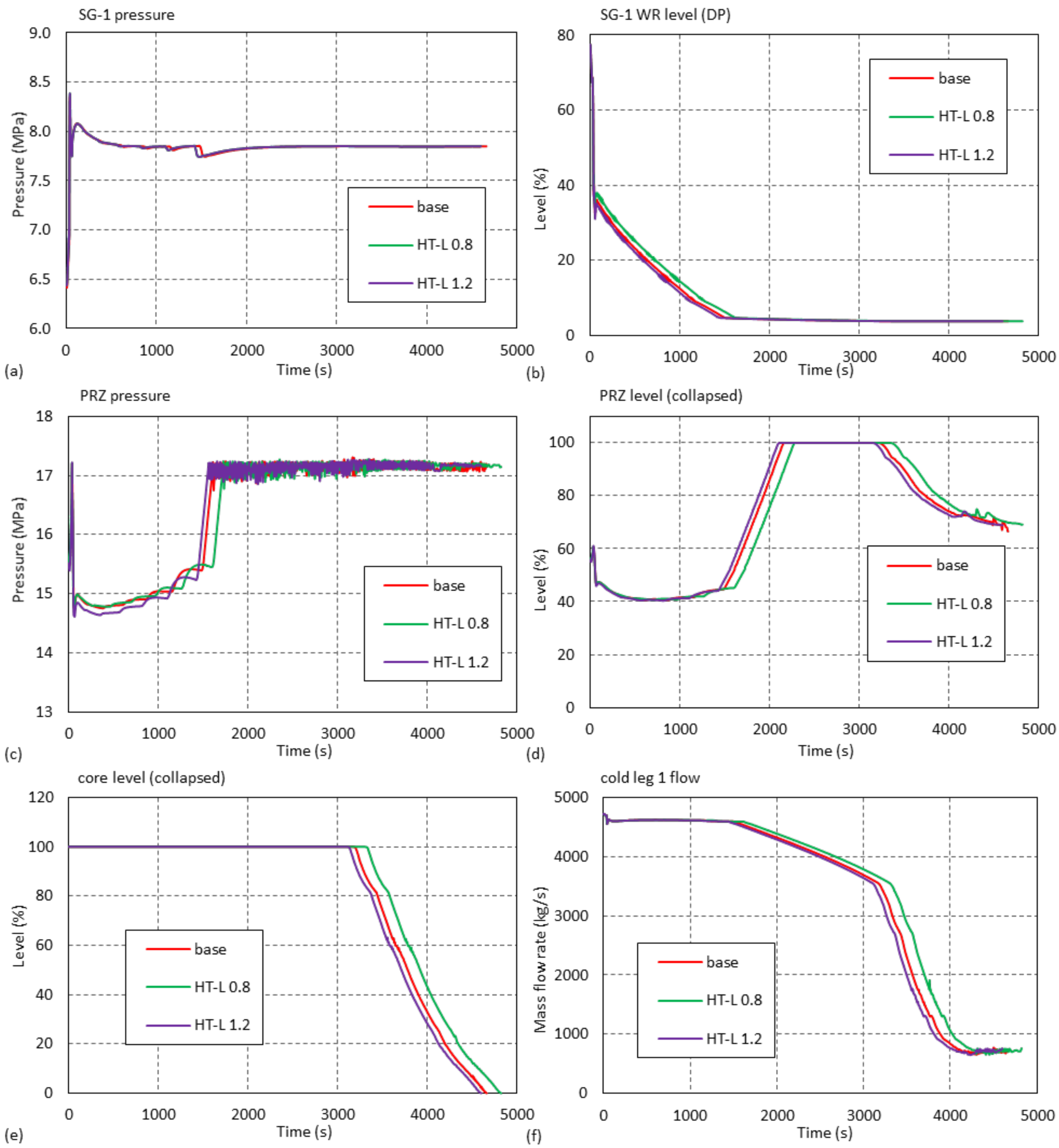

Figure 3-14 Sensitivity case HT-L - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Pressurizer Pressure, (d) Pressurizer Level, (e) Core Collapsed Liquid Level, (f) Cold Leg No. 1 Mass Flow Rate

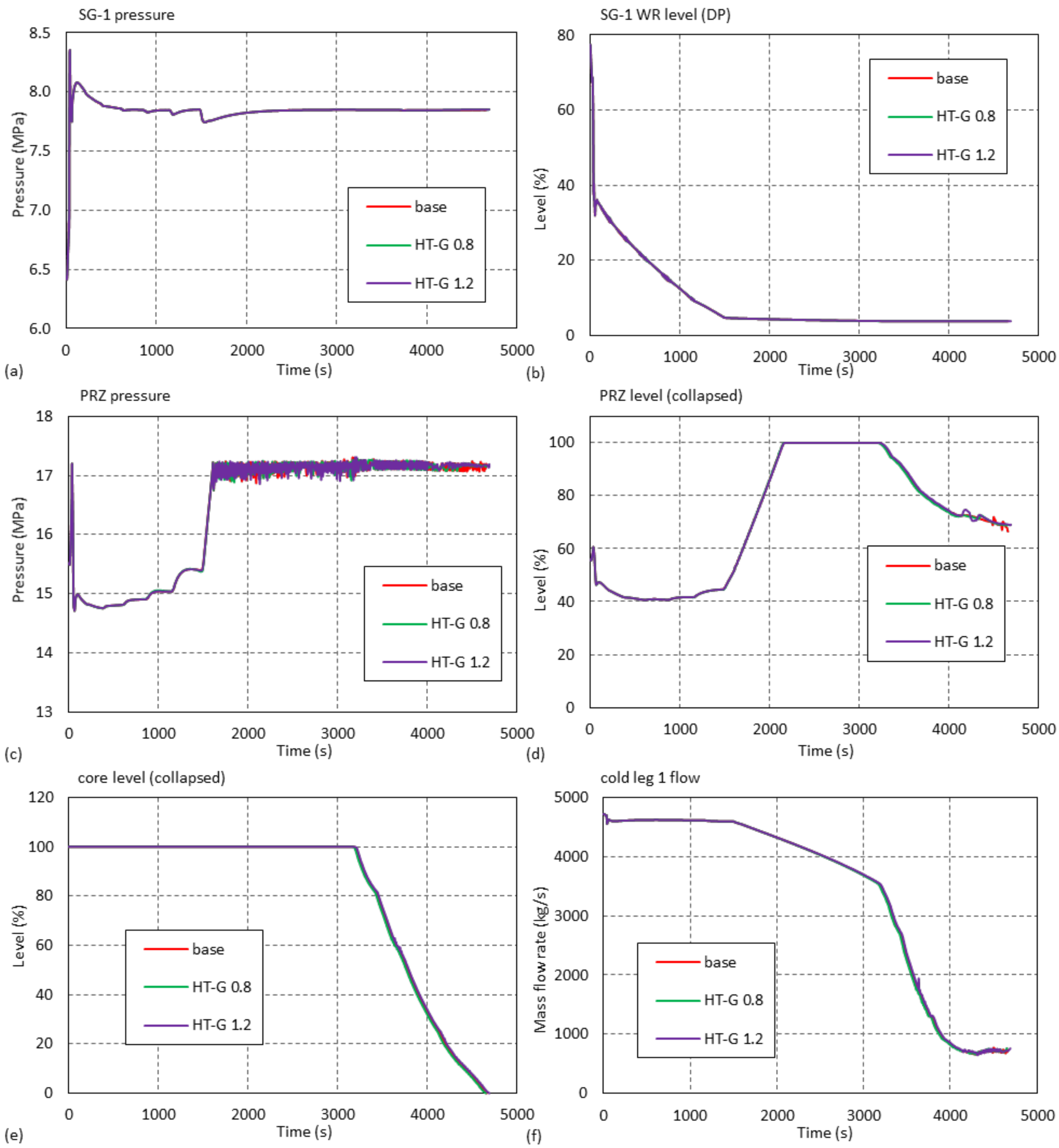

Figure 3-15 Sensitivity case HT-G - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Pressurizer Pressure, (d) Pressurizer Level, (e) Core Collapsed Liquid Level, (f) Cold Leg No. 1 Mass Flow Rate

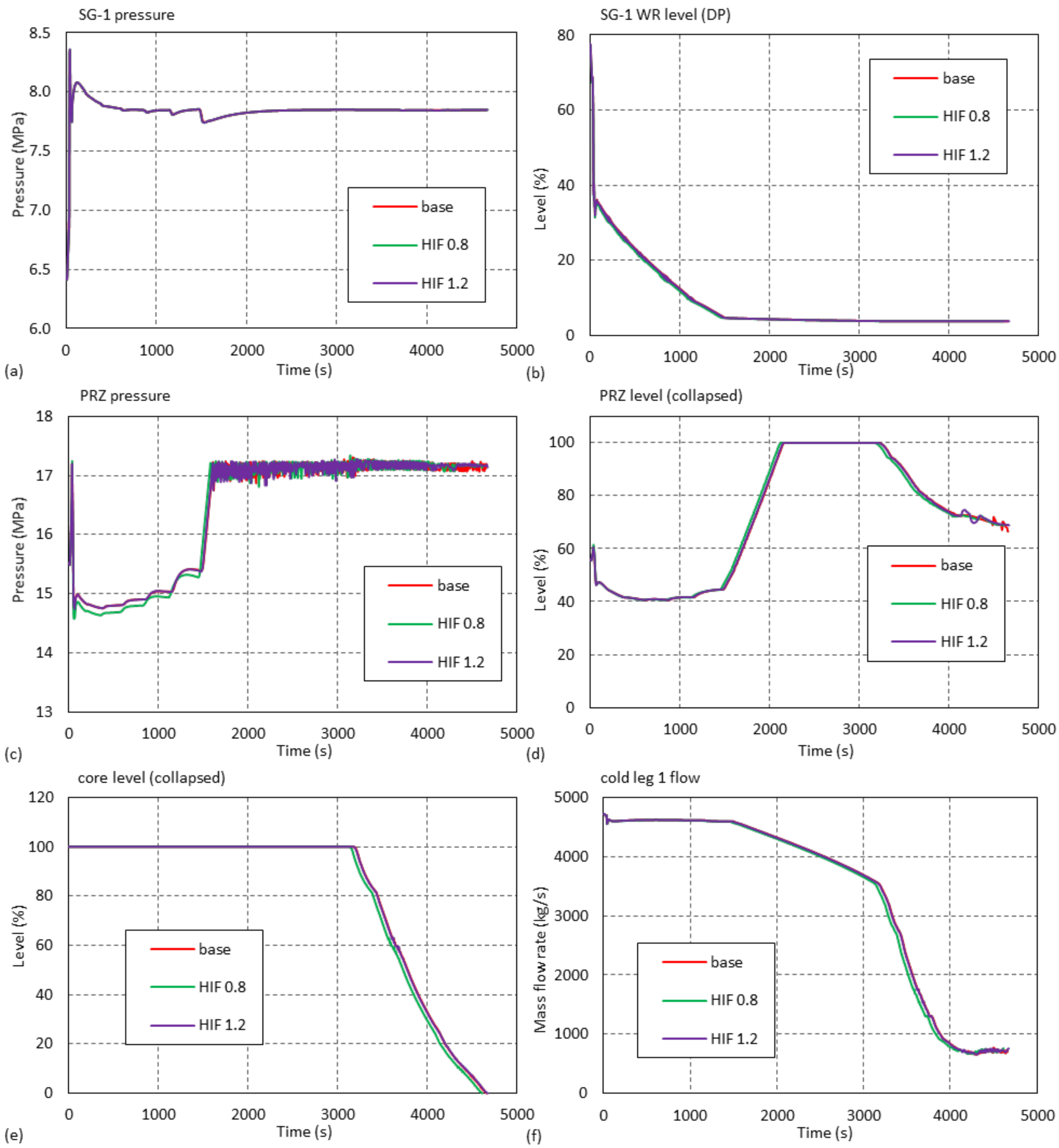

Figure 3-16 Sensitivity case HIF - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Pressurizer Pressure, (d) Pressurizer Level, (e) Core Collapsed Liquid Level, (f) Cold Leg No. 1 Mass Flow Rate

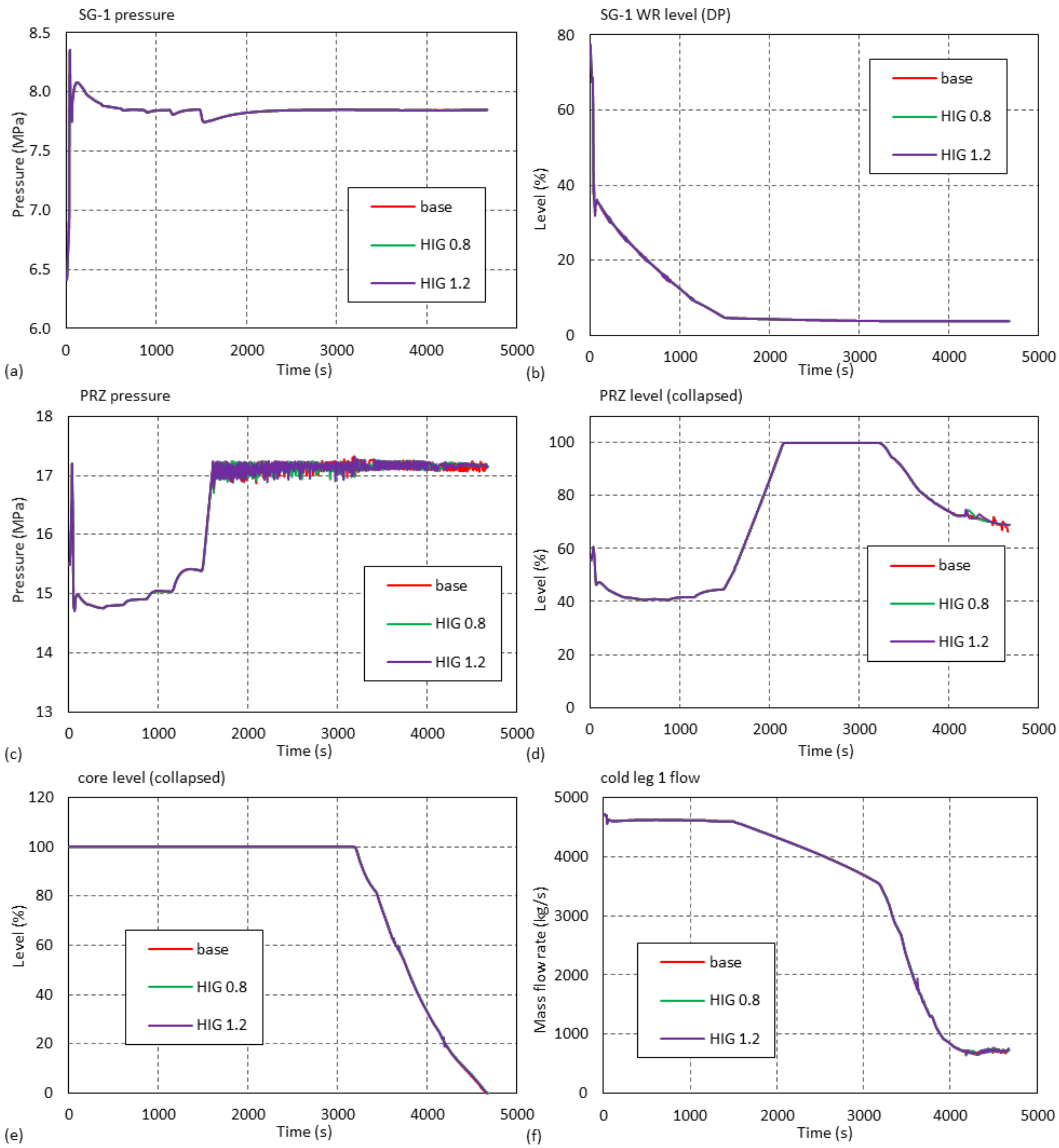

Figure 3-17 Sensitivity case HIG - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Pressurizer Pressure, (d) Pressurizer Level, (e) Core Collapsed Liquid Level, (f) Cold Leg No. 1 Mass Flow Rate

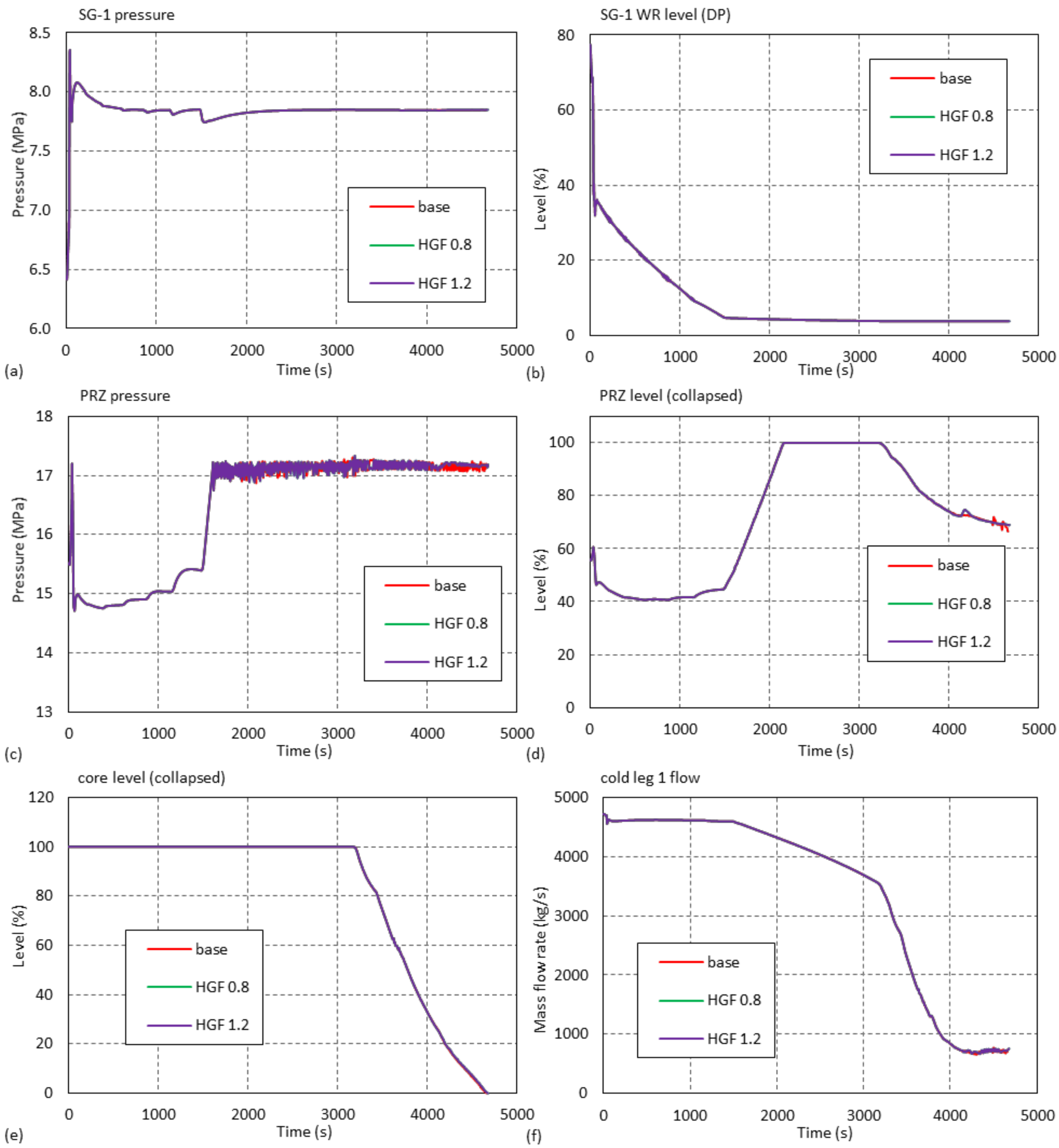

Figure 3-18 Sensitivity case HFG - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Pressurizer Pressure, (d) Pressurizer Level, (e) Core Collapsed Liquid Level, (f) Cold Leg No. 1 Mass Flow Rate

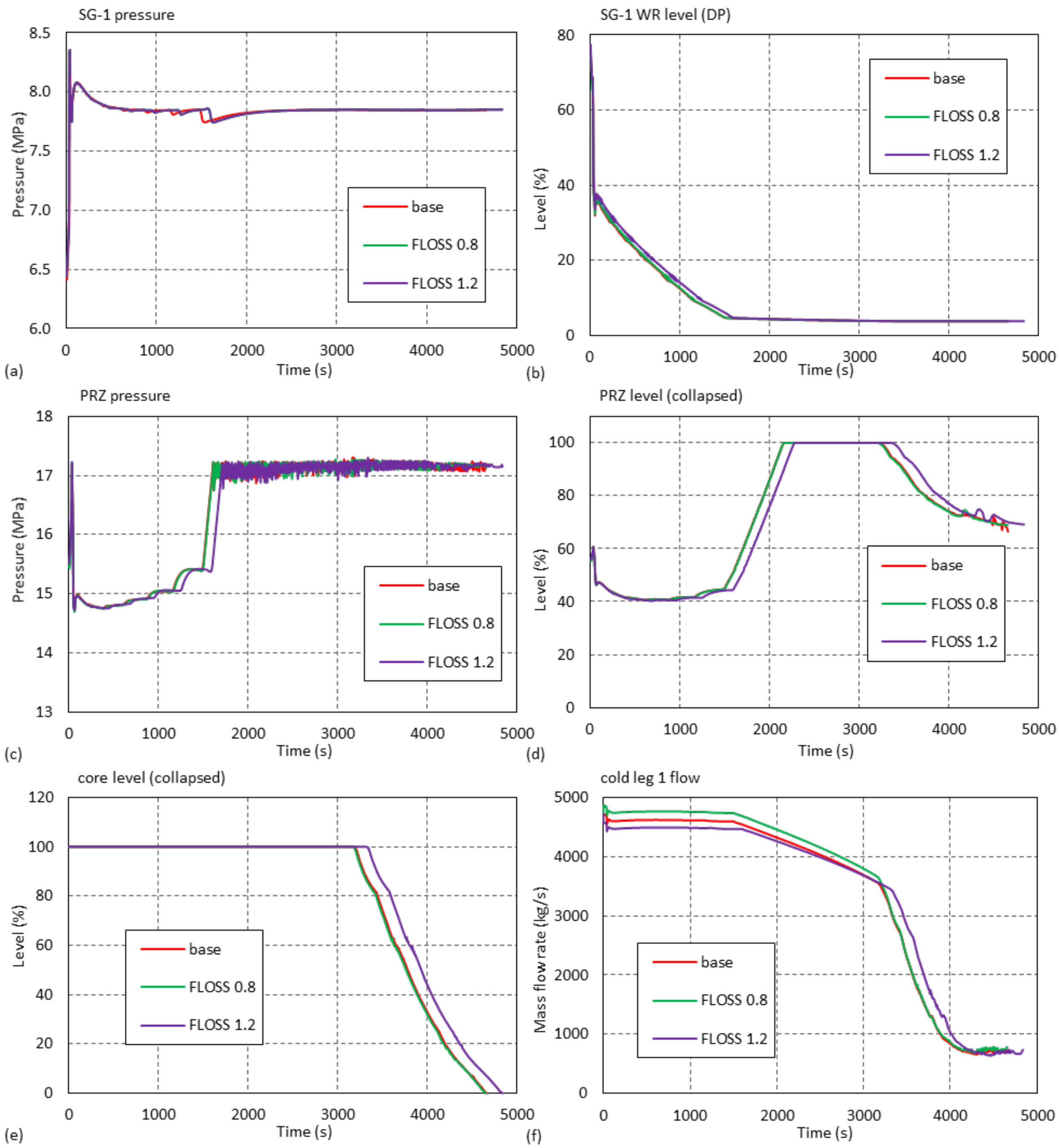

Figure 3-19 Sensitivity case FLOSS - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Pressurizer Pressure, (d) Pressurizer Level, (e) Core Collapsed Liquid Level, (f) Cold Leg No. 1 Mass Flow Rate

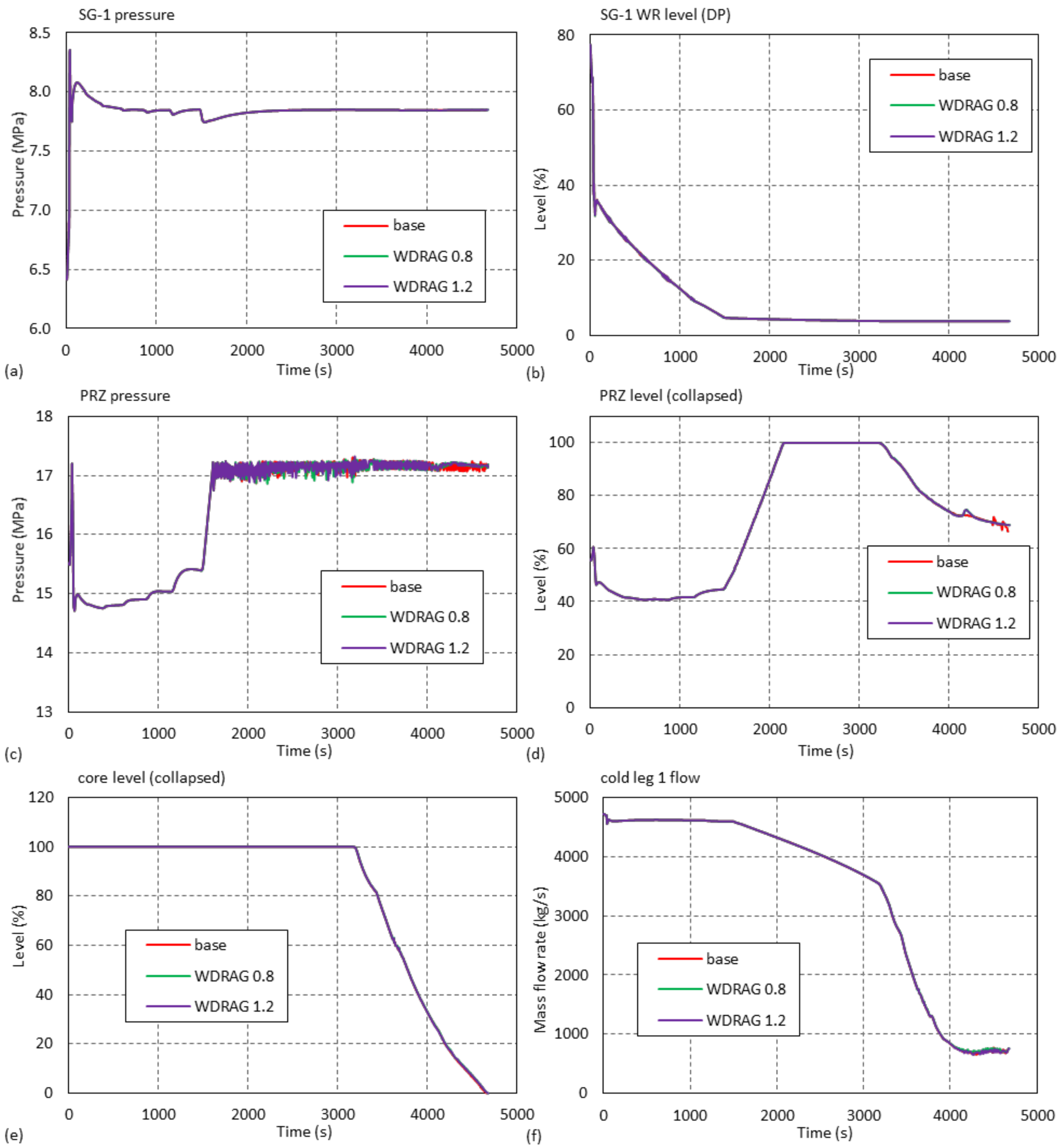

Figure 3-20 Sensitivity case WDRAG - (a) Steam Generator No. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Pressurizer Pressure, (d) Pressurizer Level, (e) Core Collapsed Liquid Level, (f) Cold Leg No. 1 Mass Flow Rate

4 CONCLUSIONS

Four different scenarios of total loss of feedwater have been studied. The scenarios are total loss of feedwater with normal systems available (TLOFW-N), total loss of feedwater without normal systems available (TLOFW-WN), total loss of feedwater without normal systems (except reactor coolant pumps) available (TLOFW-RCP) and total loss of feedwater with normal systems available plus design extension condition (DEC) safety feature (TLOFW-DEC).

The analysis using only safety systems is less conservative than analysis crediting normal systems – the main reason is early reactor trip due to reactor coolant flow low-low signal generation due to not assumed reactor coolant pumps. However, if DEC safety feature (i.e. alternate auxiliary feedwater) is started after 30 min, significant core uncovery is prevented. In longer term for all scenarios DEC safety feature is needed, if primary system is not depressurized.

The sensitivity study for TLOFW-RCP scenario showed that uncertainty parameters gas heat transfer (HTC and heat flux), volumetric wetted wall gas interface heat transfer coefficient, volumetric wetted wall direct liquid-gas heat transfer coefficient, and two-phase friction multiplier have very small impact on the results, while liquid heat transfer (HTC and heat flux), volumetric wetted wall liquid interface heat transfer coefficient and junction form loss value have small but visible impact on the calculated results.

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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

In Europe the design extension conditions (DEC) were introduced after the Fukushima Dai-ichi accident as preferred method for giving due consideration to the complex sequences and severe accidents without including them in the design basis conditions. The purpose of the study is to determine available elapsed time before core degradation and needed DEC safety features to prevent total loss of all feedwater (TLOFW) in a two-loop pressurized water reactor. In its documents, both WENRA (Western European Association of Nuclear Regulators) and the International Atomic Energy Agency (IAEA) present TLOFW initiating event as a possible DEC for existing nuclear power plants.

For simulations six U.S. Nuclear Regulatory Commission RELAP5 computer code versions were used to study the possible impact of code version on the results. The initiating event for TLOFW are multiple failures in which, besides the loss of main feedwater also auxiliary feedwater is lost. It is assumed that both high pressure and low pressure safety injection trains and batteries are available. Four different scenarios of TLOFW have been studied. The results section shows the comparison of calculated results obtained by several RELAP5 versions. Finally, the simulated results of total loss of feedwater with DEC safety feature available are shown.

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