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RELAP5 simulations of total loss of feedwater in a 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 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.

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 the loss of all feedwater are multiple failures in which, besides the loss of main feedwater also auxiliary feedwater is lost. This system consists of two motor-driven and a turbine driven pump. The operator's action is also assumed to trip the reactor coolant pump in accordance with Emergency Operating Procedures (EOP). Initially the reactor is assumed to operate at 100 % power. It is assumed that both High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) trains and batteries are available. 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.

1 INTRODUCTION

The latest International Atomic Energy Agency (IAEA) SSR-2/1 (Rev. 1) standard was reviewed in the light of the Fukushima Dai-ichi accident, as well as the lessons identified in the IAEA report on the Fukushima Dai-ichi accident. The design extension conditions (DEC) definition was modified, DEC are being postulated accident conditions, which comprise conditions in events without significant fuel degradation and conditions in events with core melting. Western European Nuclear Regulators Association (WENRA) reference levels 2020 [2] did not follow the latest IAEA definition of DEC, but is rather use definition of IAEA SSR-2/1 [3].

In the present paper total loss of feedwater (TLOFW) was studied, which is also on the list of DEC, provided by IAEA and WENRA. First, the RELAP5 input model of a two-loop pressurized water reactor (PWR) and scenarios simulated by six different RELAP5 versions are described. Besides total loss of feedwater scenario also scenario with DEC safety feature has been considered. Then the results are presented. Finally, conclusions are given.

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 [4]:

- 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 in Section 2.1.

2.1 RELAP5 input model

The RELAP5 input model of two-loop pressurized water reactor (PWR) is shown in Figure 1. A two loop PWR reactor power is 1994 MW, while the nuclear steam supply system (NSSS) power is 2000 MW. The base model consists of 304 hydraulic components and 108 heat structures. For more detailed input model description refer to our previous studies [5], [6].

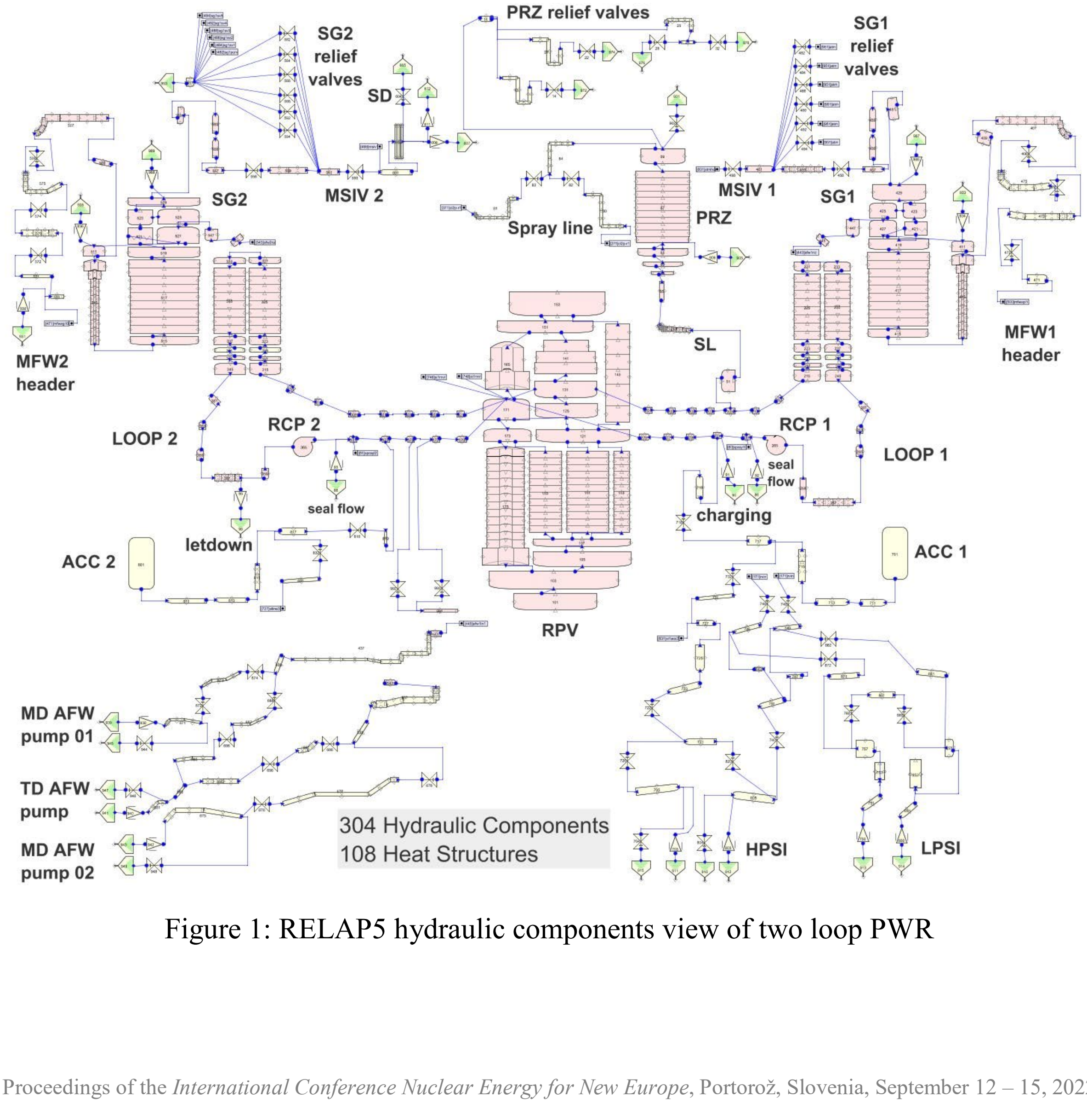

Figure 1: RELAP5 hydraulic components view of two loop PWR

2.2 Scenarios description

In this study the DEC A total loss of all feedwater accident starts after the 1000 seconds of steady state plant operation. 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.

In addition, a scenario was simulated with assumed DEC safety feature available after 2800 s of accident start. Assumed DEC safety feature is pump, injecting into both steam generators. All scenarios simulated are shown in Table 1.

Table 1: Scenarios simulated with six different RELAP5 versions

Scenario name	**RELAP5 code version used**	**Calculation label used**
total loss of feedwater	RELAP5/MOD3.3 Release	TLOFW
	RELAP5/MOD3.3 Patch 2	TLOFW_P2
	RELAP5/MOD3.3 Patch 3	TLOFW_P3
	RELAP5/MOD3.3 Patch 4	TLOFW_P4
	RELAP5/MOD3.3 Patch 5	TLOFW_P5
	RELAP5/MOD3.3 version lj	TLOFW_lj
loss of feedwater (LOFW) with DEC safety feature available	RELAP5/MOD3.3 Release	LOFW
	RELAP5/MOD3.3 Patch 2	LOFW_P2
	RELAP5/MOD3.3 Patch 3	LOFW_P3
	RELAP5/MOD3.3 Patch 4	LOFW_P4
	RELAP5/MOD3.3 Patch 5	LOFW_P5
	RELAP5/MOD3.3 version lj	LOFW_lj

3 RESULTS

The results are shown in Figures 2 through 5. 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 10^{th} 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 (RCP) no. 1 and no. 2 are tripped. In LOFW with DEC safety feature available the injection to both steam generator is started after 2800 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 2(d)) caused that both the secondary and primary pressure started to increase (see Figure 2(a) and Figure 3(a), respectively).

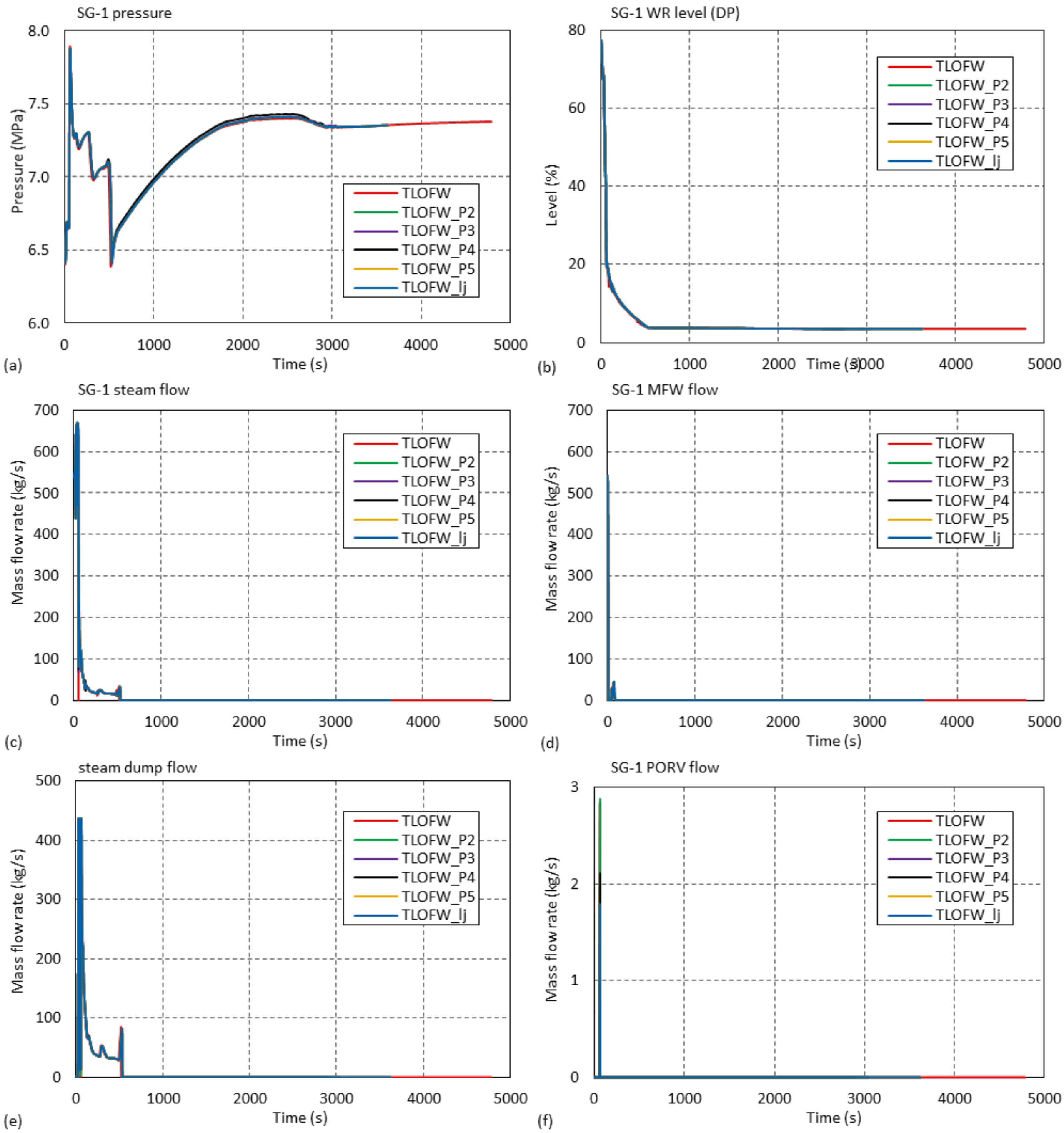

Figure 2: TLOFW Secondary Side Related Parameters - (a) Steam Generator no. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Steam Generator No. 1 Steam Flow, (d) Steam Generator No. 1 Main Feedwater Flow, (e) Steam Dump Flow, (f) Steam Generator No. 1 power operated relief valve (PORV) flow

Both lines of pressurizer spray lines were activated to reduce the primary pressure (see Figures 3(c) and 3(d)). On the secondary side, the steam is dumped to steam dump system (see Figure 2(e)), causing loss of steam generator inventory, resulting in steam generator (SG) level decrease (see Figure 2(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(a)) due to the reactor trip, while the secondary pressure sharply increased due to the turbine trip (see Figure 2(a)).

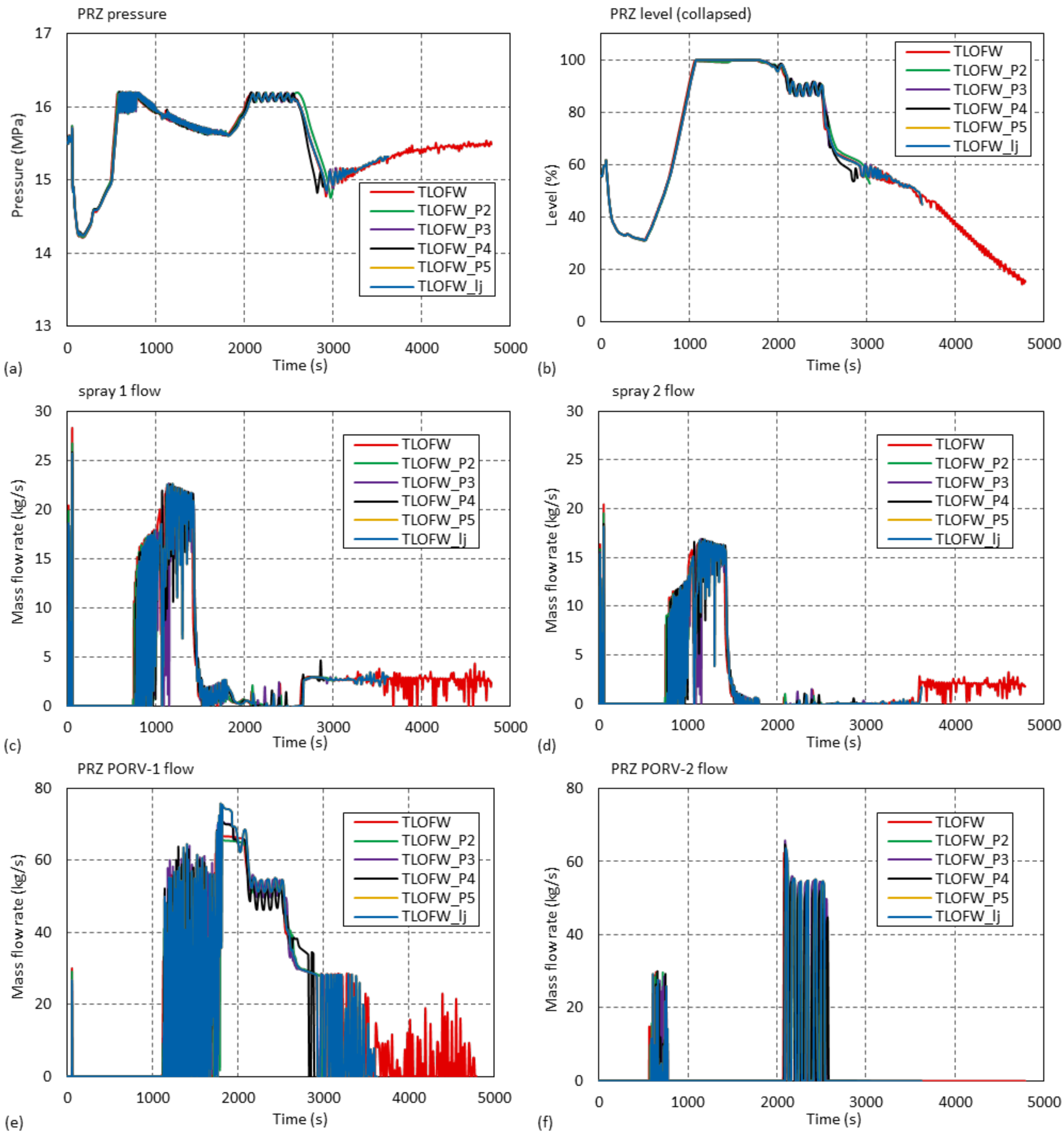

Figure 3: TLOFW Pressurizer Related Parameters – (a) Pressurizer Pressure, (b) Pressurizer Level, (c) Pressurizer Spray No. 1 Mass Flow Rate, (d) Pressurizer Spray No. 1 Mass Flow Rate, (e) Pressurizer PORV No. 1 Discharge Mass Flow Rate, (f) Pressurizer PORV No. 2 Discharge Mass Flow Rate

Since all feedwater is lost, there is no steam generators filling (see Figure 2). The pressurizer (PRZ) pressure rate sensitive PORV no. 1 initially discharged briefly (Figure 3(e)) and the steam generators PORV (see SG no. 1 PORV on Figure 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 2(e)). The second PRZ PORV first opened around 600 s (see Figure 3(f)), after the steam dump flow termination, and again later approximately between 2100 s and 2500 s.

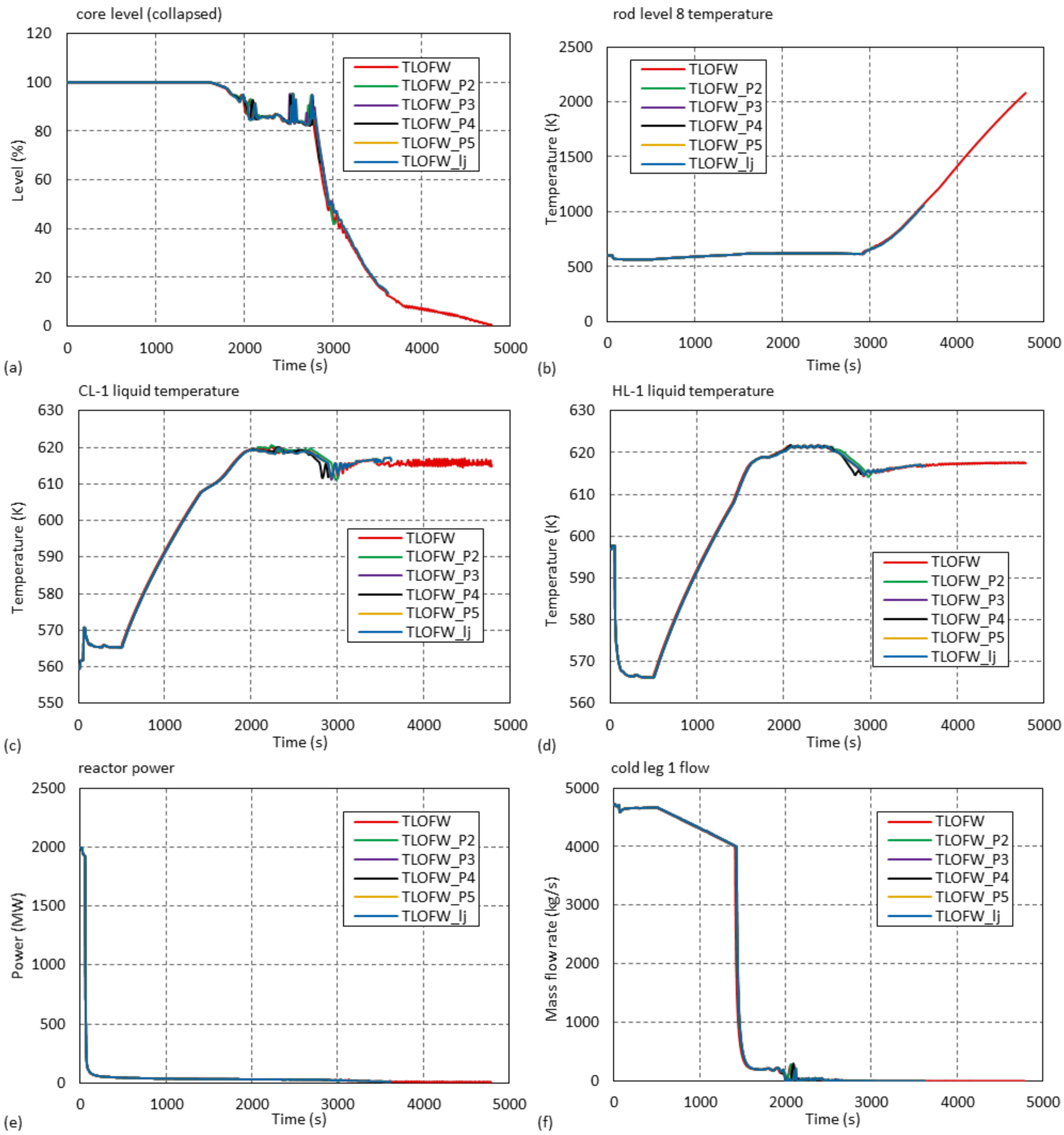

Figure 4: TLOFW Primary Side Related Parameters – (a) Core Collapsed Liquid Level, (b) Cladding Temperature at 2.29 m in the Core, (c) Cold Leg No. 1 Liquid Temperature, (d) Hot Leg No. 1 Liquid Temperature, (e) Core Power, (f) Cold Leg No. 1 Mass Flow Rate

The second PRZ PORV opened on the set pressure 16.2 MPa, while PRZ rate sensitive PORV reopened after 1100 seconds and remained open till the end of calculation (see Figure 3(e)). When both, first and second PRZ PORV 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(e) and 3(f)). After the saturation of the primary coolant at 1730 s, the primary pressure increased (see Figure 3(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 4(c) and (d)) and pressure (see Figure 3(a)) started again to increase, resulting in core uncovery start as shown in Figure 4(a). In spite of HPSI pumps and charging pumps operating the core uncovering could not be prevented as the primary pressure (see Figure 3(a)) became higher than the shutoff head of HPSI and charging (chemical and volume control system) pumps. As can be seen from Figure 4(b) 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).

In LOFW with DEC safety feature available scenario all RELAP5 versions gave comparable results as shown in Figure 5.

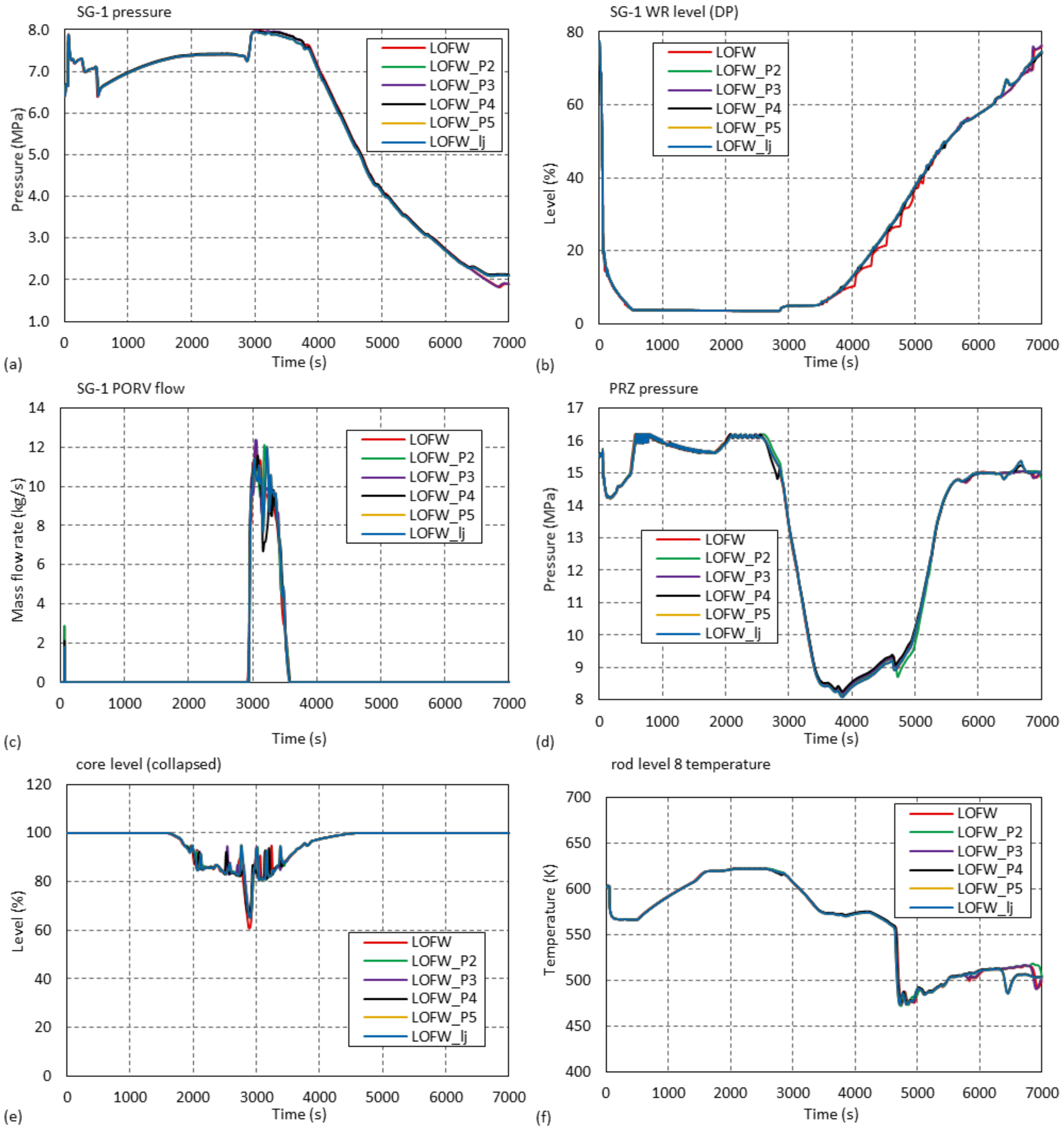

Figure 5: LOWF with DEC Safety Feature Parameters - (a) Steam Generator no. 1 Pressure, (b) Steam Generator No. 1 Wide Range Level, (c) Steam Generator No. 1 PORV flow, (d) Pressurizer Pressure, (e) Core Collapsed Liquid Level, (f) Cladding Temperature at 2.29 m in the Core

When after 2800 s injecting into steam generators is started, the SG level started to increase (see Figure 5(b)), the SG injected water started to evaporate, therefore the secondary pressure sharply increased (see Figure 5(a)), resulting in the SG-PORV opening (see Figure 5(c)). This also helps to further reduce the primary pressure drop (see Figure 5(d)), enabling HPSI pumps injection and core level recovery (see Figure 5(e)). After closure the SG-PORV, also the level in steam generator started to recover. When the core level is recovered, the core is quenched (see cladding temperature in Figure 5(f)). Continuous operation of DEC safety feature provides the continuous cooling of the primary side through the secondary side cooling and the simulation is therefore terminated at 7000 s, when steam generator are refilled.

4 CONCLUSIONS

The design extension condition (DEC) scenario with total loss of all feedwater has been studied for a specific two-loop pressurized water reactor (PWR) using six RELAP5/MOD3.3 computer code versions. The analysis of DEC scenario with total loss of all feedwater showed that without the operator action the capacity of high pressure safety injection (HPSI) and charging pumps are insufficient to refill the primary pressure and to maintain the long term cooldown of the core. Thus, this type of transient leads to the core overheating and damage, indicating that DEC safety feature is needed to mitigate the consequences of such DEC scenario. The scenario with DEC safety feature available after 2800 s shows that further core uncovery and heatup is prevented.

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