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Loss of Coolant Accident with Total Failure of High Pressure Injection System in Two-loop PWR

Andrej Prošek
Jožef Stefan Institute
Jamova cesta 39
SI-1000, Ljubljana, Slovenia
andrej.prosek@ijs.si

ABSTRACT

After Fukushima Dai-ichi accident Western European Association of Nuclear Regulators (WENRA) and the International Atomic Energy Agency (IAEA) require due consideration of design extension conditions (DEC). The purpose of this paper is to study the loss of coolant accident (LOCA) together with the complete loss of high pressure injection system (HPIS). Such multiple failure has been recognized by WENRA and IAEA documents as possible DEC.

For analysis the U.S. Nuclear Regulatory Commission TRAC/RELAP Advanced Computational Engine (TRACE) computer is used. The TRACE input deck has been developed based on the conversion of verified and validated RELAP5 standard input deck for a two-loop pressurized water reactor (PWR). For automatic conversion the Symbolic Nuclear Analysis Package (SNAP) has been used, which was not perfect and it required several manual corrections. The initiating event LOCA together with total failure of HPIS is multiple failure in which both high pressure safety injection pumps are lost. Other safety systems are assumed available. The LOCA calculations have been performed for a spectrum of break sizes in the cold leg. The results showed that DEC safety feature is needed for smaller breaks, because such breaks could not remove all decay heat through the break. When the breaks are larger, the decay heat removal through the break is sufficient and the pressure drops sufficiently to allow accumulators and low pressure injection system to inject. Finally, simulations demonstrated that in the scenarios assuming DEC safety feature the significant core heatup has been prevented.

1 INTRODUCTION

The second-generation reactors were designed and built to withstand without loss to the structures, systems, and components necessary to ensure public health and safety during design basis accidents (DBAs). In the transient and accident analysis the effects of single active failures and operator errors were considered. There are also accident sequences that are possible but were judged to be too unlikely and therefore were not fully considered in the design process of second-generation reactors. In that sense, they were considered beyond the scope of design basis accidents that a nuclear facility must be designed and built to withstand. They were called beyond design basis accidents (BDBA). Nowadays, according to Western European Association of Nuclear Regulators (WENRA) and the International Atomic Energy Agency (IAEA) the design extension conditions (DEC) is used to describe those BDBA, for which additional prevention and mitigation provisions are required.

The analysis for a two-loop pressurized water reactor (PWR) has been performed to show if the plant design can prevent loss of coolant accident LOCA scenarios with existing safety systems or not (in this case additional DEC safety features are needed). Following document [1], the control of DEC is expected to be achieved primarily by features implemented in the design (safety features for DEC) and not only by accident management measures that are using equipment designed for other purposes. This means that in principle a DEC is such if its consideration in the design leads to the need of additional equipment or to an upgraded classification of lower-class equipment to mitigate the DEC.

The paper is organised as follows. First, LOCA scenario together with high pressure injection system (HPIS) failure scenario is described, followed by TRACE input model and simulated cases description. Then results of LOCA spectrum simulations are presented and discussed. Finally, some conclusions are drawn.

2 LOCA SCENARIO, INPUT MODEL AND SIMULATED CASES DESCRIPTION

2.1 LOCA Scenario Description

In the LOCAs simulated, at the beginning of accident the assumed emergency core cooling systems (ECCSs) available were two accumulators and the low pressure injection system (LPIS). The initiating event is opening of the valve simulating the break in the cold leg with reactor operating at 100 % power. The reactor trip on (compensated) low pressurizer pressure (12.99 MPa) further causes the turbine trip. The safety injection (SI) signal is generated on the low-low pressurizer pressure signal at 12.27 MPa. On SI signal active safety systems, i.e. low pressure safety injection (LPSI) pumps and motor driven (MD) auxiliary feedwater (AFW) pumps start. When primary pressure drops below 4.96 MPa, both accumulators start to inject. Larger is the break size, faster is the accumulator discharge. When primary pressure drops below 1.13 MPa, two LPSI pumps start to inject. In the case of smaller breaks, the high primary pressure can prevent accumulators and LPSI pumps injection, if no primary side depressurization is performed.

2.2 TRACE Input Model Description

The TRACE input model shown in Figure 1 represents a two-loop PWR, Westinghouse type, with thermal power 2000 MW. The TRACE input model was obtained by the conversion of the verified RELAP5 input model into TRACE input model, using Symbolic Nuclear Analysis Package (SNAP) [2] and using the steps of Jožef Stefan Institute (JSI) RELAP5 to TRACE conversion method [3]. Several modifications had to be made manually in the TRACE input model during the conversion process, mostly related to heat structures boundary conditions, accumulator model options and hydraulic connections of pipe components that originated from RELAP5 branch components etc. Several control block data had to be modified too. The TRACE input model consists of 461 SNAP hydraulic components and 115 heat structures and represents the primary and secondary side of PWR. All important systems and components were modelled, including control systems, reactor protection system logic and safety systems.

2.3 Simulated LOCA Break Cases and Scenarios Description

TRACE simulations of scenarios shown in Table 1 have been performed, representing a spectrum of break sizes in the cold leg, ranging from 5.08 cm through 30.48 cm. The simulations have been performed for two groups of scenarios, scenarios without DEC safety feature assumed and scenarios with DEC safety feature assumed with the start after accident initiation (in this first study start time is assumed like for safety systems). For comparison purposes, the results of previous RELAP5 calculations performed were used [4].

For scenarios without DEC safety feature assumed, six different break sizes were considered, ranging from 5.08 cm through 30.48 cm. For scenarios with DEC safety feature assumed (i.e. alternative safety injection pump), the smallest two break sizes from the break spectrum were simulated.

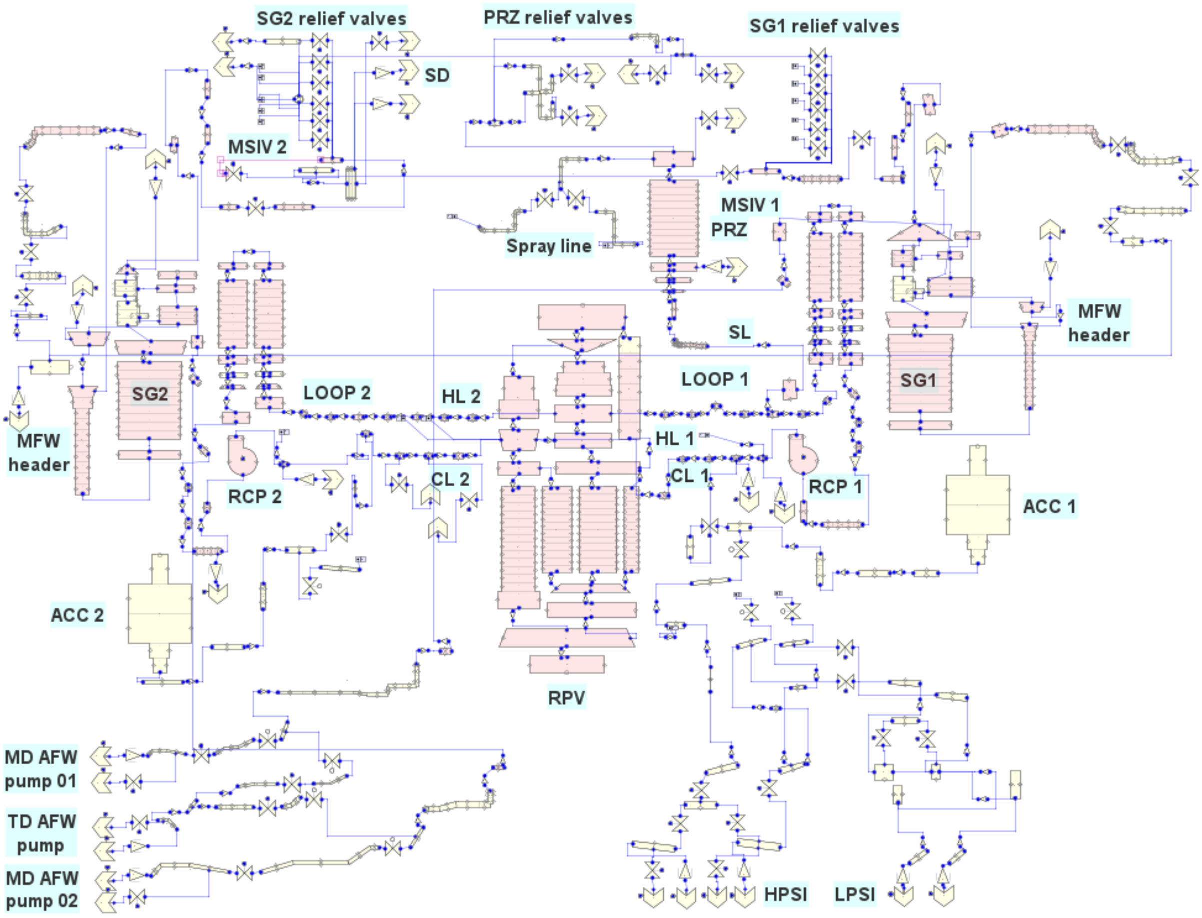

Figure 1: TRACE two-loop PWR hydraulic components view

Table 1: LOCA scenarios simulated by TRACE and compared to RELAP5 results from 2019 [4]

Break size diameter	TRACE V5.0 Patch 5	RELAP5/MOD3.3 Patch 5 simulations from 2019 [4]
without DEC safety feature assumed		
5.08 cm (2 inch)	TR_sb2	R5_sb2
7.62 cm (3 inch)	TR_sb3	R5_sb3
10.16 cm (4 inch)	TR_sb4	R5_sb4
15.24 cm (6 inch)	TR_sb6	R5_sb6
20.32 cm (8 inch)	TR_sb8	R5_sb8
30.48 cm (12 inch)	TR_sb12	R5_sb12
with DEC safety feature assumed		
5.08 cm (2 inch)	TR_sb2_DEC	N.A.
7.62 cm (3 inch)	TR_sb3_DEC	N.A.

3 RESULTS

Results of simulated LOCAs are shown in Figures 2 through 4. Figure 2 shows comparison between the TRACE and RELAP5 for 5.08 cm and 7.62 cm equivalent diameter break size LOCAs together with complete loss of HPIS. In addition, TRACE calculations with DEC safety feature available are also shown.

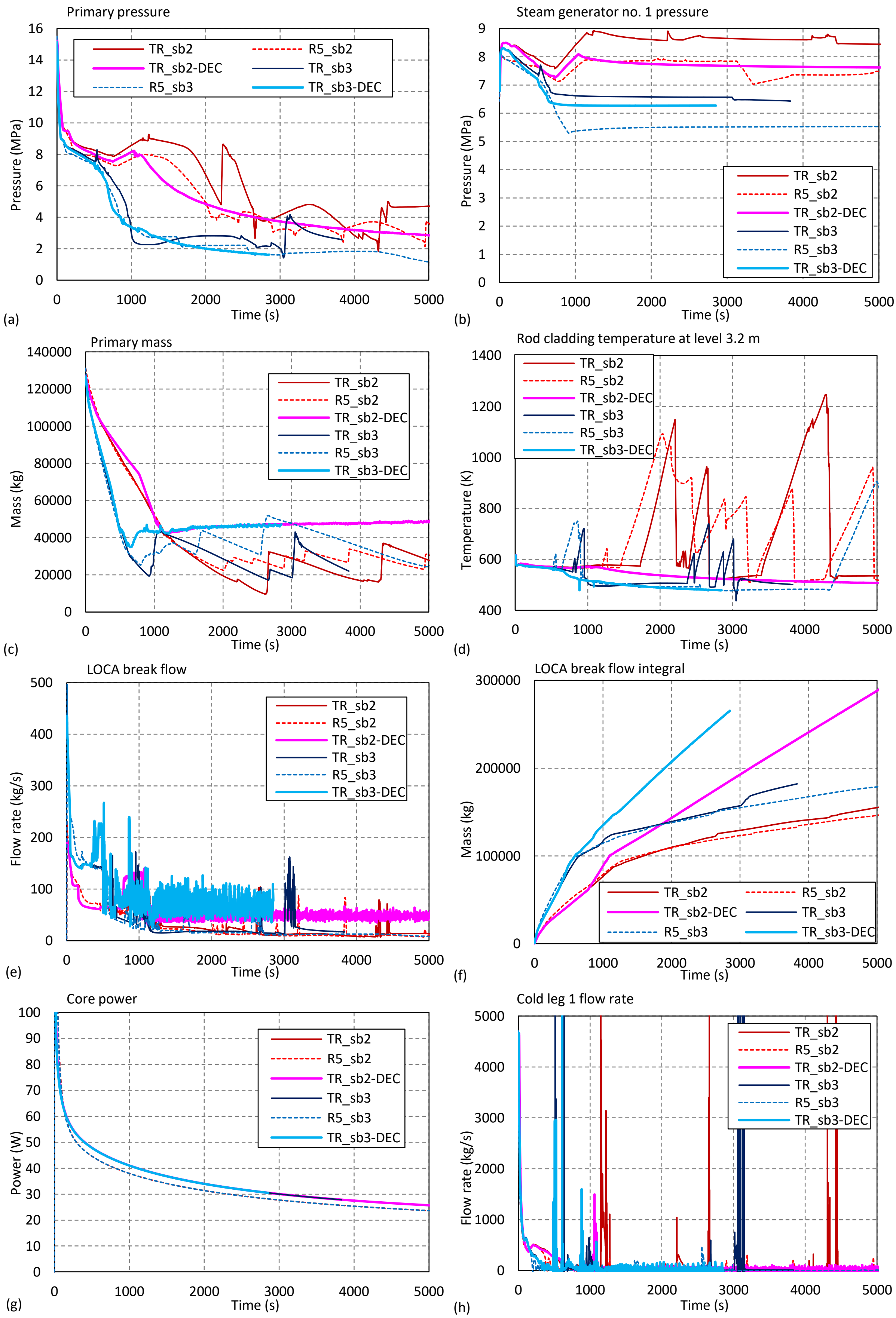

Figure 2: Comparison between TRACE and RELAP5 for 5.08 cm and 7.62 cm break size LOCAs

Figure 2 shows the following eight important variables: (a) primary pressure, (b) steam generator no. 1 pressure, (c) primary mass, (d) rod cladding temperature at level 3.2 m, (e) LOCA break flow, (f) LOCA break flow integral, (g) core power, and (h) cold leg 1 flow rate. Figure 2(d) shows that in scenarios without DEC safety feature there is core heatup both for 5.08 cm and 7.62 cm break sizes (see cases 'TR_sb2', 'R5_sb2', 'TR_sb3' and 'R5_sb3'). When DEC safety feature is used, core heatup is prevented. When looking variables shown in Figure 2, it can be seen that agreement between TRACE and RELAP5 results it is qualitatively good. Nevertheless, one should be aware that core power (see Figure 2(g)) resulting from decay heat it is few percent lower in the case of RELAP5 than TRACE after initial 120 s. Also, one can see from Figure 2(h) that TRACE calculation of the cold leg flow rate after pump coastdown is not physical and the reason for this needs to be identified in the future.

Figures 3 and 4 show comparison between the TRACE and RELAP5 for 10.16 cm, 15.24 cm, 20.32 cm and 30.48 cm equivalent diameter break size LOCAs together with complete loss of HPIS. On Figure 3 are shown (a) primary pressure, (b) steam generator no. 1 pressure, (c) cold leg no. 1 liquid temperature, (d) hot leg no. 1 liquid temperature, (e) primary mass, and (f) rod cladding temperature at level 3.2 m, while on Figure 4 are shown (a) LOCA break flow, (b) LOCA break flow integral, (c) refuelling water storage flow integral, (d) steam generator no. 1 mass, (e) core power, and (f) cold leg 1 flow rate. These exhaustive list of variables makes it easier to the reader visually judge the agreement between the TRACE and RELAP5 calculations. It should be noted that RELAP5 input deck has been thoroughly verified and validated, while TRACE simulation is one of the first simulations using in Section 2.2 presented input deck for a two-loop PWR. Such code to code comparison may help in identifying possible deficiencies in TRACE input deck. For primary pressure shown in Figure 3(a) the pressure drop is in good agreement, but it can be seen that after few hundred seconds there are some pressure increases, timing of which is concurrent with spikes in the cold leg flow rates shown in Figure 4(f). This deficiency needs to be explained the future. The hot leg loop temperature trend shown in Figure 3(d) is similar to primary pressure trend. The agreement between TRACE and RELAP5 for initial reactor coolant system (RCS) mass inventory trend shown in Figure 3(e) is very good, while later in the transient in the TRACE calculation the core uncovers more times, resulting in more cladding temperature peaks as shown Figure 3(f). Break flow integral shown in Figure 4(b) is comparable between TRACE and RELAP5 except for the largest 30.48 cm break size case. This difference is seen also in Figure 4(c) showing the RWST water injected by LPIS pumps. In case of 10.16 cm break size there is no LPIS injection in the presented time interval because the primary pressure is too high. For core power shown in Figure 4(e) it can be seen that after initial period the TRACE calculated core decay heat is higher than RELAP5 calculated core decay heat. Finally, Figure 4(f) shows unphysical spikes in TRACE calculated loop flow.

The results showed that breaks equal and larger than 15.24 cm are sufficient to depressurize the RCS and by this enabling injection by accumulators and LPIS after reaching their setpoints. Break 10.16 cm is limiting. RELAP5 results suggest that cooling through the break is sufficient, while in the case of TRACE results another core heatup occurred after 2000 s as shown in Figure 3(f). On the other hand, Figure 2 showing 5.08 cm and 7.62 cm breaks suggests that DEC safety feature is needed to prevent further rod cladding heatups after first rod cladding heatup. If DEC safety feature is starting at the beginning of the accident, the core heatup is prevented. Finally, both Figure 2(h) and Figure 4(f) show unphysical spikes in TRACE calculated cold leg loop flow rate. In this first simulation the reason has not been identified and should be identified in the future. Initially the reactor coolant pump (RCP) coastdown model has been suspected, because after tripping the reactor the RCP pump speed increased. Simulating the RCP coastdown curve did not resolve the problem of spikes in flow rate. In the future the new TRACE V5.0 Patch 8 is planned to be used first to see (release in autumn 2023), if this problem is code related. In spite of this issue it can be concluded that TRACE confirmed the results obtained by RELAP5 for LOCA together with loss of HPIS.

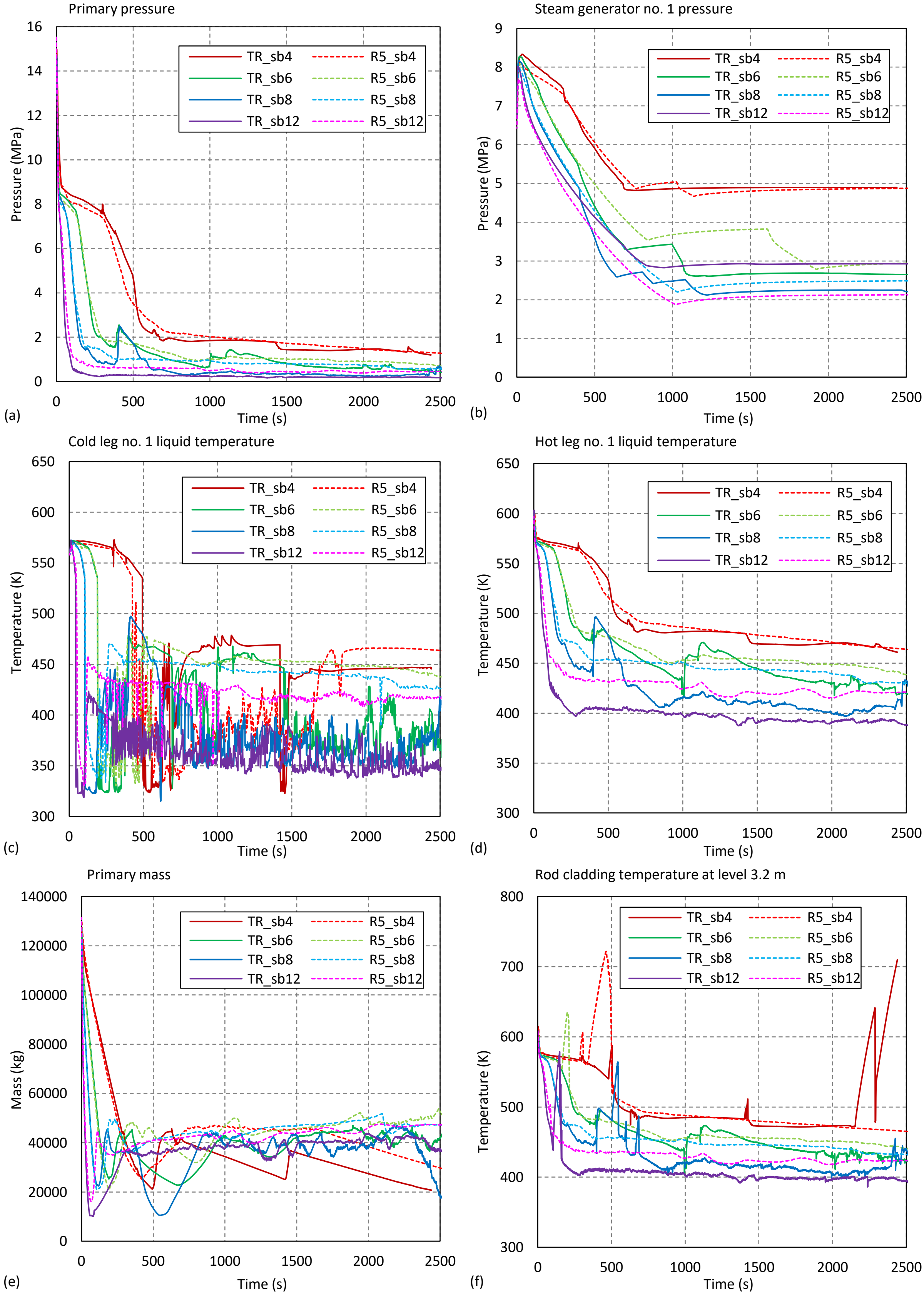

Figure 3: Comparison between TRACE and RELAP5 for 10.16 cm, 15.24 cm, 20.32 cm and 30.32 cm break size LOCAs (part 1)

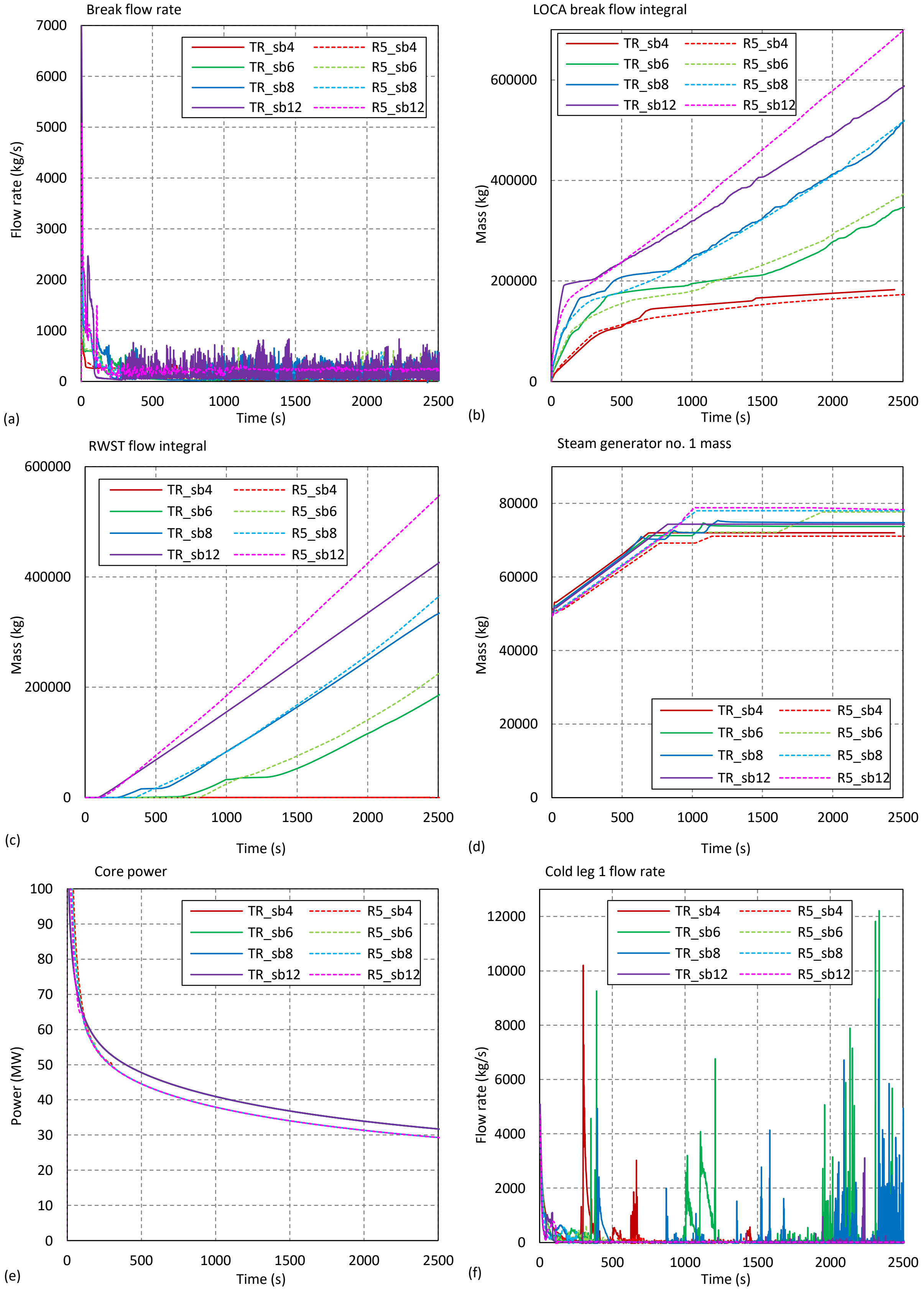

Figure 4: Comparison between TRACE and RELAP5 for 10.16 cm, 15.24 cm, 20.32 cm and 30.32 cm break size LOCAs (part 2)

CONCLUSIONS

The loss of coolant accident together with loss of high pressure injection system simulations have been performed for a spectrum of break sizes occurring in a two-loop pressurized water reactor (PWR). For simulations the TRACE advanced best-estimate system code has been used. Two groups of scenarios have been simulated, without design extension conditions (DEC) safety feature and with DEC safety feature. The results for scenarios without DEC safety feature showed that DEC safety features are needed for smaller breaks in selected PWR, because such breaks could not remove all the decay heat through the break. When the breaks are larger, the core decay heat removal through the break is sufficient as primary pressure drops sufficiently to allow accumulators and low pressure system injection. The comparison between TRACE and RELAP5 for scenarios without DEC safety feature assumed showed good qualitative agreement and revealed some deficiencies to be resolved in the TRACE simulations. Finally, TRACE simulations demonstrated that in the scenarios assuming DEC safety feature the significant core heatup has been prevented.

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