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TRACE Analysis of Total Loss of Feedwater in PWR

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ABSTRACT

After introduction of design extension conditions (DEC) by Western European Association of Nuclear Regulators (WENRA) and the International Atomic Energy Agency (IAEA) several European countries implement requirement to consider DEC. The purpose of this paper is to study the total loss of feedwater (TLOFW) in a two-loop pressurized water reactor (PWR), which presents DEC. For the analyses, the U.S. Nuclear Regulatory Commission TRAC/RELAP Advanced Computational Engine (TRACE) best-estimate reactor systems computer code has been used. The TRACE input deck has been developed by converting the verified and validated RELAP5 standard input deck for a two loop PWR. The initiating event was total loss of feedwater, which is multiple failure (loss of all main and auxiliary feedwater pumps). A loss of non-emergency alternate current (AC) was assumed, which would cause the main feedwater loss, a loss of reactor coolant system flow (RCS) and a reactor trip on RCS flow trip or loss of power to control rods in the base case scenario. Another more conservative scenario has been studied assuming that these events are not occurring simultaneously. It was assumed that the reactor is tripped on low-low steam generator narrow level and after reactor trip the RCS flow is lost due to loss of AC power. A parametric study varying reactor trip time delay with respect to base case scenario has been also performed to study the impact on the time available before reactor uncovers below criterion 60.96 cm above the top of the core. In all scenarios these scenarios by assumption the auxiliary feedwater system is not available, which would normally start on low-low steam generator narrow level.

The results showed that delayed reactor trip reduced the time available before the reactor vessel margin is lost. As expected, the TLOFW scenarios showed the need for a DEC safety feature. Finally, it has been demonstrated that DEC safety feature starting at 60 min successfully prevents the core overheating for base case scenario. For conservative scenario results suggest that the DEC safety feature should be started 30 min after TLOFW event initiation.

1 INTRODUCTION

Slovenia implemented Western European Association of Nuclear Regulators (WENRA) reference levels, which require that the design extension conditions (DEC) should be considered. For both, WENRA and International Atomic Energy Agency (IAEA), the total loss of feedwater (TLOFW) event can be considered as a DEC event without significant fuel degradation (WENRA classifies it as a category DEC A). According to study [1] the frequency evaluation for various scenarios can give insights when to select DEC scenarios. The frequency for TLOFW with assumed main and auxiliary feedwater loss was estimated to be above 1e-7/yr.

In 2022 total loss of feedwater (TLOFW) scenario has been studied by RELAP5 computer code [2] based on the scenario, which has been initially used for Krško full scope simulator validation [3]. The objective of RELAP5 study from 2022 was to determine available elapsed time before core degradation and needed design extension conditions (DEC) safety features. For calculations 6 different RELAP5/MOD3.3 versions were used to determine the code version impact. In addition, base case scenario was simulated with assumed DEC safety feature available after 2800 s of accident start. Assumed DEC safety feature has been pump, injecting into both steam generators. Follow-up study [4] showed that the scenario assuming operation of normal safety systems is more conservative than scenario crediting safety systems mainly due to the fact that reactor is operating longer with nominal power due to later reactor trip. Namely, in case of loss of alternate current (AC) assumption reactor coolant pumps stop and the reactor trip occurred early on low reactor coolant flow after then.

In this study the U.S. Nuclear Regulatory Commission TRAC/RELAP Advanced Computational Engine (TRACE) V5.0 Patch 5 computer code has been used as an independent computer code to the RELAP5 version 3.3lj. The studied TLOFW scenarios credited only safety systems as recommended in IAEA specific safety guide SSG-2 on deterministic safety analysis [5].

2 TRACE INPUT MODEL AND SCENARIOS DESCRIPTION

2.1 TRACE Input Model Description

A two loop pressurized water reactor (PWR), Westinghouse type, with thermal power 2000 MW has been used for calculations. The conversion of the verified RELAP5 input model of two-loop PWR to the TRACE input model was performed using Symbolic Nuclear Analysis Package (SNAP) [6] and following the Jožef Stefan Institute (JSI) RELAP5 to TRACE conversion method. More information regarding the conversion procedure can be found in Ref. [7]. Several modifications were manually made in the TRACE input model during the conversion process, mostly related to Heat Structures boundary conditions, Accumulator model option and Hydraulic connections of Pipe components that originated from RELAP5 Branch components. Several Control Block Data have been modified too. For more details refer to [8]. TRACE input model is shown in Figure 1. The TRACE input model consists of 461 SNAP hydraulic components and 115 heat structures.

Modelling of the primary side includes the reactor pressure vessel (RPV), both loops (LOOP 1 and 2), the pressurizer (PRZ) vessel, pressurizer surge line (SL), pressurizer spray lines and valves, two pressurizer power operated relief valves (PORVs) and two pressurizer safety valves, chemical and volume control system charging and letdown flow, and reactor coolant pump (RCP 1 and 2) seal flow. Emergency core cooling systems (ECCSs) piping includes two high pressure safety injection (HPSI) pumps, two accumulators (ACC 1 and 2), and two low pressure safety injection (LPSI) pumps. The secondary side consists of the two steam generators (SGs) - secondary side, main steam line, main steam isolation valves (MSIV 1 and 2), SG1 and SG2 relief and safety valves, and main feedwater (MFW1 and MFW2) piping. The turbine valve is modelled by the corresponding logic. Steam dump (SD) is also modelled by the corresponding logic. The turbine is represented by time dependent volume. The MFW and AFW (auxiliary feedwater) motor driven (MD) and turbine driven (TD) pumps are modelled as time dependent junctions.

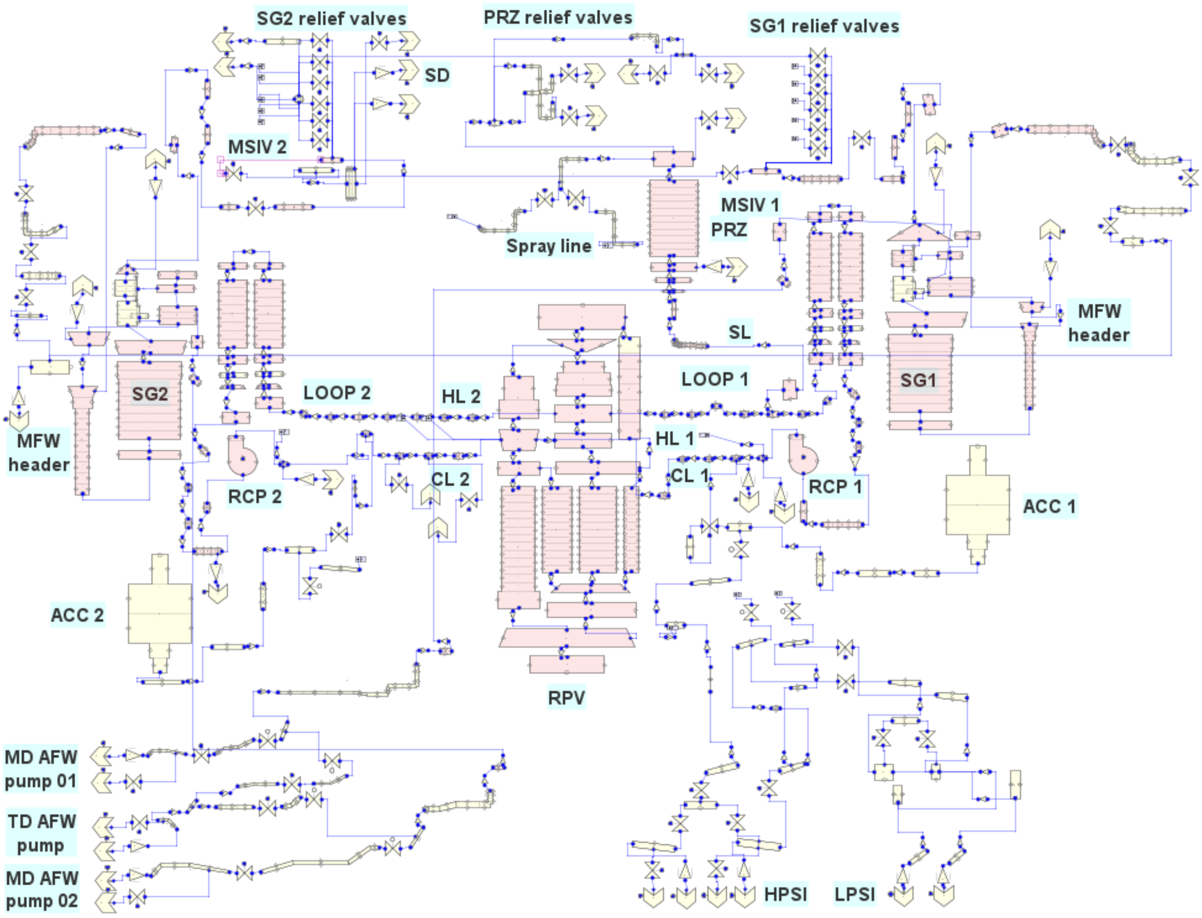

Figure 1: TRACE two-loop PWR hydraulic components view

2.2 Simulated Scenarios Description

The initiating event for TLOFW is multiple failure, in which all main and auxiliary feedwater pumps are assumed unavailable. Two groups of scenarios are studied by TRACE advanced, best-estimate reactor systems computer code as shown in Table 1. First group is represented by scenarios without assumed DEC safety features, while the second group comprise of scenarios with DEC safety feature (i.e. alternate auxiliary feedwater pump). In addition, only safety systems are credited in the analyses after reactor trip occurrence. Both trains of emergency core cooling system are assumed available, consisting of two high pressure and two low pressure safety injection pumps and two accumulators. DEC safety feature is started 3600 s after TLOFW event initiation. For the purpose of comparison between TRACE and RELAP5, two RELAP5 calculations of base case scenarios without and with DEC safety feature have been performed, labelled RELAP5-S and RELAP5-S_DEC respectively.

The main characteristic of the TLOFW accident is gradual emptying of steam generators resulting in the reduced capability of the secondary side to remove the heat generated in the core. Heat transfer is soon degraded during partial uncovering of the U-tubes and finally interrupted with complete emptying of steam generators. As no feedwater injection is available on the secondary side, the pressure on the primary side increases and the pressurizer (PRZ) safety valves open and close between their set and reset pressures to remove the decay heat from the primary system. The loss of RCS mass through PRZ safety valves leads to reactor vessel uncovering. Following study [9] that the core mixture level should be maintained with at least 60.96 cm (two feet) margin above the core top to ensure core coverage, in this study the level criterion was set to maintain the reactor vessel collapsed liquid level with at least

60.96 cm above the top of the core. If the reactor vessel water level meet the above level criterion, the maximum fuel cladding temperature would not exceed 1477.6 K.

Table 1: TLOFW scenarios description calculated by TRACE

Scenario	**Label**	**Description**
Scenarios without assumed DEC safety feature		
Base case	TRACE-S	This is base case scenario with a reactor trip on low reactor coolant system (RCS) flow occurring on loss of AC power. Safety systems available by design are assumed available.
Conservative case	TRACE-Sc	This is conservative case scenario with a reactor trip on low-low steam generator narrow level. After the reactor trip the RCS flow is assumed lost due to loss of AC power. Safety systems available by design are assumed available.
Conservative case with reactor trip time variation	Rx-aa, where aa={10, 20, …, 80}	This is conservative case scenario with a reactor trip on low-low steam generator narrow level at selected reactor trip (Rx) time. Rx trip times are delayed from event initiation in 10 sec increments. After the reactor trip the RCS flow is lost due to loss of AC power. Safety systems available by design are assumed available.
Scenarios with assumed DEC safety feature		
Base case with DEC safety feature	TRACE-S_DEC	Same as TRACE-S above with assumed DEC safety feature, with manual start 3600 s after TLOFW event initiation.
Conservative case with DEC safety feature	TRACE-Sc_DEC	
Conservative case with DEC safety feature and reactor trip time variation	RX-aa_DEC	Same as Rx-aa above with assumed DEC safety feature, with manual start 3600 s after TLOFW event initiation.

3 RESULTS

The results are shown in Figures 2 through 3. In Figure 2 the conservative cases with reactor trip variation labelled 'Rx-10' through 'Rx-80' (reactor trips at 10 s, 20 s, …,80 s, respectively) and 'TRACE-Sc' conservative case (reactor trips at 42 s) are compared to base case 'TLOFW-S' (reactor trips within less than 3 s). Eight important variables are shown: (a) reactor power, (b) pressurizer pressure, (c) cold leg temperature in loop 1, (d) rod cladding temperature at around two third of the core height (level 8 of 12), (e) reactor vessel level (collapsed), (f) integrated mass flow rate through PRZ safety valves, (g) SG-1 wide range (WR) level and (h) SG-1 valves flow integral. Comparison of all TRACE conservative case calculations with TRACE base case calculation shows the impact of delayed reactor trip time on selected variables. As can be seen from Figure 2(e), in conservative calculation labelled 'TRACE-Sc' (the reactor trip on low low SG level) the reactor vessel level drops below 61 % (this value represents level 60.96 cm above top of core) at around 3560 s, while in 'TRACE-S' base case calculation this happened at 6900 s. Consequently, core heatup for the above cases started at around 4440 s and 7900 s, respectively (see Figure 2(d)). Figure 2(g) shows the SG-1 WR level and it can be seen that the levels after 60 s of event initiation are 25 % and 60 % for 'TRACE-Sc' and 'TRACE-S' case, respectively. Higher SG-1 level (similar to SG-2 level) shown in Figure 2(g) means more SG-1 mass inventory available for cooling and therefore more mass was discharged through SG-1 valves (see Figure 2(h)), which also significantly delay the start of pressurizer safety valves discharge (see Figure 2(f)).

As shown in Figure 2, 'TRACE-Sc' case results with reactor trip on low-low steam generator narrow level around 42 s are very close to 'Rx-40' conservative case with assumed reactor trip at 40 s. The assumed reactor trips in the range from 10 s to 80 s in 10 s time increments give good picture, that early reactor trip is very beneficial and vice versa.

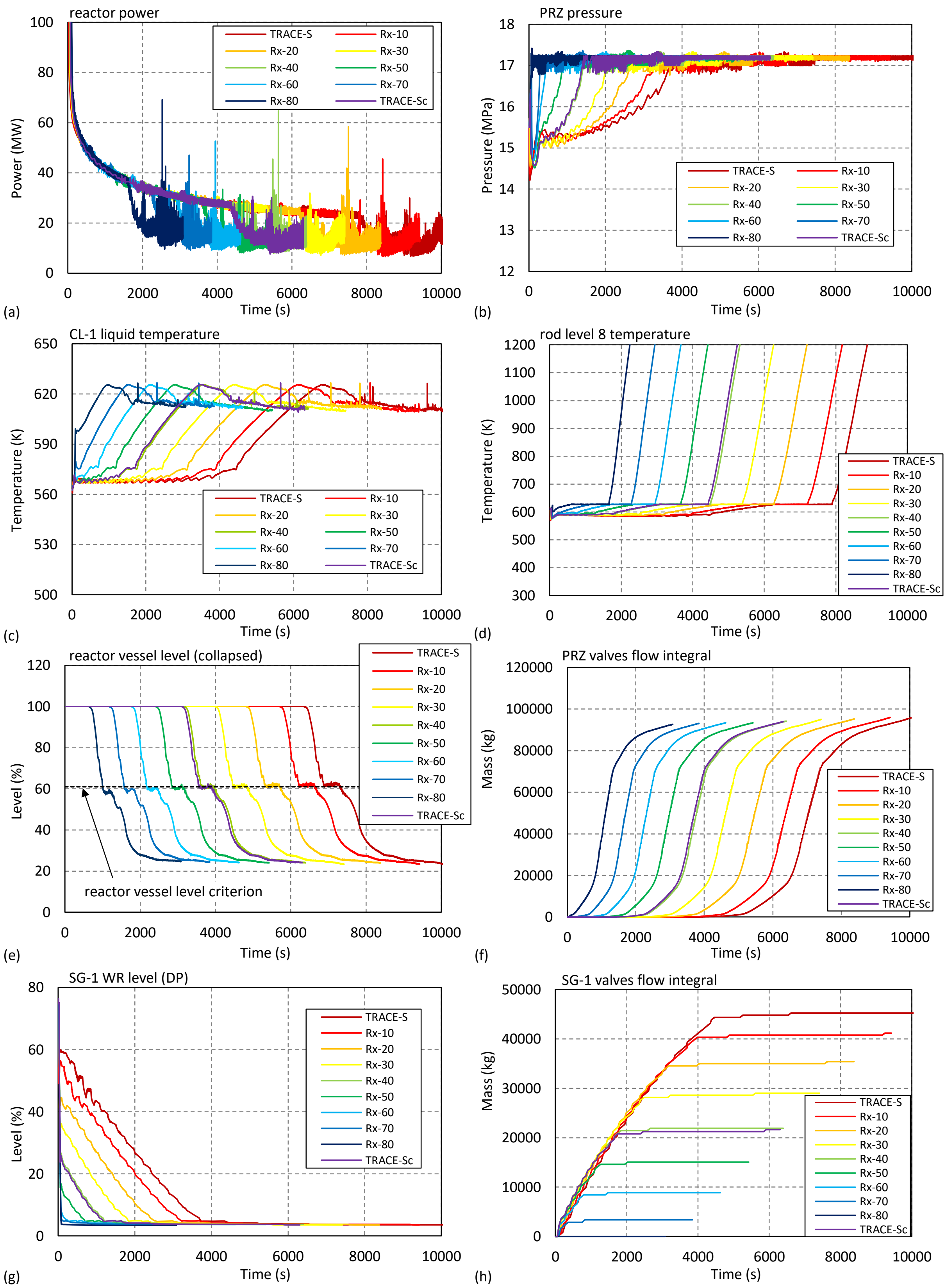

Figure 2: The impact of delayed reactor trip time in conservative cases compared to base case TLOFW calculated by TRACE

Comparison between TRACE base case with assumed DEC safety features and the conservative cases with reactor trip variation labelled 'Rx-10' through 'Rx-80' with assumed DEC safety features is shown in Figure 3. As can be seen, the DEC safety feature is started at 3600 s after event initiation. The reactor vessel level criterion shown in Figure 3(c) is satisfied for 'TRACE-S_DEC', 'Rx-10_DEC' and 'Rx-20_DEC' cases. This demonstrates that reactor vessel criterion provides sufficient margin to core heatup. As can be seen from Figure 3(d) also for 'Rx-40' and 'Rx-50' cases no heatup occur. Figure 3 also suggests that starting DEC safety feature after 1800 s the reactor vessel criterion would be satisfied for 'Rx-40' and 'Rx-50' cases, and core heatup would be prevented for 'Rx-60' and Rx-70' cases.

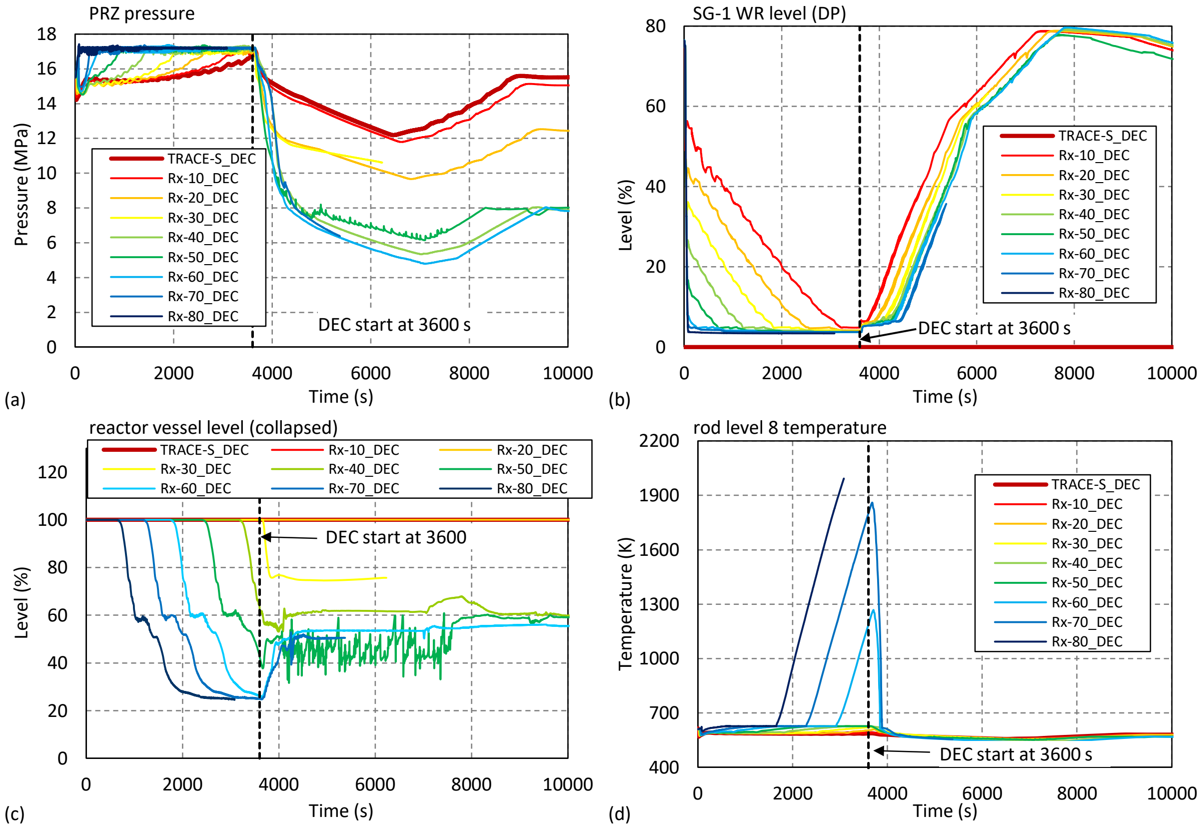

Figure 3: TRACE calculations of TLOFW scenarios with DEC safety feature

Figure 4 shows comparison between TRACE and RELAP5 (labelled 'RELAP5-S' and 'RELAP5-S_DEC' without and with DEC safety feature available, respectively) base case calculations. Four variables are shown: (a) reactor vessel level (collapsed), (b) integrated mass flow rate through PRZ safety valves, (c) SG-1 wide range (WR) level and (d) SG-1 valves flow integral. The TRACE and RELAP5 base case calculations without DEC safety feature agree well. The small difference is in the initial SG-1 WR level (see Figure 4(c)), which is slightly higher in the case of TRACE, what resulted in the later start time of reactor vessel uncovery (see Figure 4(a)), which occurred some 5 min later and also later pressurizer safety valves discharge (see Figure 4(b)).

Besides TRACE and RELAP5 comparison, Figure 4 also shows the comparison of calculations without DEC safety features (full line) and with DEC safety features (dashed lines). For base case calculations with DEC safety feature available after 3600 s the reactor vessel level uncovery is prevented (see Figure 4(a)) and decay heat is removed through the secondary side (see Figure 4(c)). On the other hand, TRACE calculation of conservative case with DEC safety features resulted in a significant core uncovery and the calculation was terminated when cladding temperature reached 2100 K. However, if DEC safety feature is available, it is true that the reactor vessel level criterion is not satisfied (see Figure 4(a)), but the level in the SG recovers (see Figure 4(c)) and the discharge trough pressurizer safety valve is terminated (see Figure 4(b)), while steam discharge is re-established through the SG relief valves (see Figure 4(c)). This is indication, that the core cooling is re-established through the secondary side.

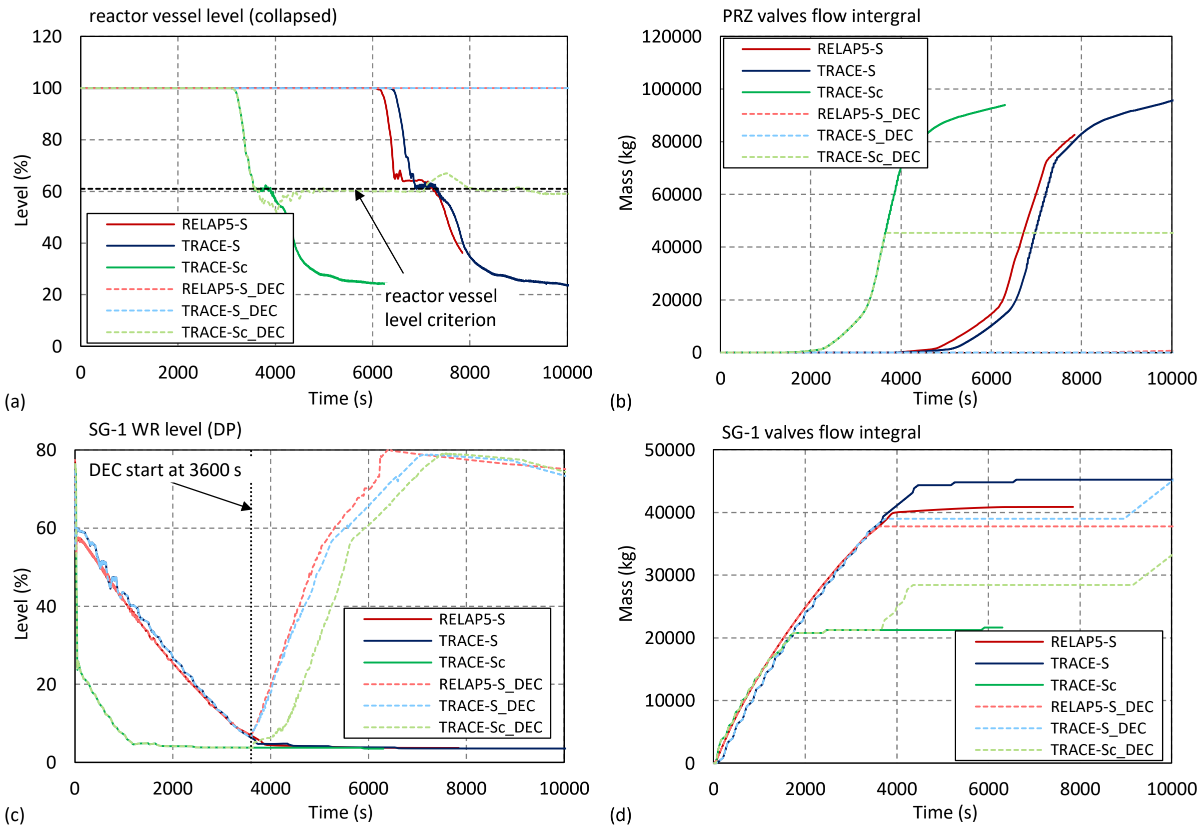

Figure 4: Comparison TRACE and RELAP5 results for scenarios without and with DEC safety feature

CONCLUSIONS

In the study several TRACE calculations of total loss of feedwater (TLOFW) have been performed, without and with design extension conditions (DEC) safety features. For the base case calculation, the comparison between TRACE and RELAP5 system codes has been also done.

The results showed that delayed reactor trip reduced the time available before the reactor vessel margin is lost. As expected, the TLOFW scenarios calculations showed the need for a DEC safety feature. Finally, it has been demonstrated that in the initial 60 min there is no need to start a DEC safety feature in a base case scenario. However, in the conservative cases with assuming reactor trip time it has been shown that reactor vessel level criterion would be satisfied for reactor trip times up to 30 s and core heatup would not occur for reactor trip times up to 50 s. The results also suggest that start of DEC safety feature after 30 minutes would satisfy reactor vessel criterion for conservative case with assumed reactor trip on low low steam generator level.

Finally, the comparison between TRACE and RELAP5 base case calculations showed that the results obtained by TRACE computer code agree well with the results obtained by RELAP5 computer code.

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