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RELAP5/MOD3.3 Analyses of Core Heatup Prevention Strategy during Extended Station Blackout in PWR

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ABSTRACT

The accident at the Fukushima Dai-ichi nuclear power plant demonstrated the vulnerability of the plants on the loss of electrical power for several days, so called extended station blackout (SBO). A set of measures have been proposed and implemented in response of the accident at the Fukushima Dai-ichi nuclear power plant. The purpose of the study was to investigate the application of the deterministic safety analysis for core heatup prevention strategy of the extended SBO in pressurized water reactor, lasting 72 h. The prevention strategy selected was water injection into steam generators using turbine driven auxiliary feedwater pump (TD-AFW) or portable water injection pump.

Method for assessment of the necessary pump injection flowrate is developed and presented. The necessary injection flowrate to the steam generators is determined from the calculated cumulative water mass injected by the turbine driven auxiliary feedwater pump in the analysed scenarios, when desired normal level is maintained automatically. The developed method allows assessment of the necessary injection flowrates of pump, TD-AFW or portable, for different plant configurations and number of flowrate changes.

The RELAP5/MOD3.3 Patch04 computer code and input model of a two-loop pressurized water reactor is used for analyses, assuming different injection start times, flowrates and reactor coolant system losses. Three different reactor coolant system (RCS) coolant loss pathways, with corresponding leakage rate, can be expected in the pressurized water reactor (PWR) during the extended SBO: normal system leakage, reactor coolant pump seal leakage, and RCS coolant loss through letdown relief valve unless automatically isolated or until isolation is procedurally directed. Depressurization of RCS was also considered. In total, six types of RCS coolant loss scenarios were considered. Two cases were defined regarding the operation of the emergency diesel generators. Different delays of the pump injection start following the station blackout were assumed and analysed. For each scenario, two kinds of SBO calculations for two-loop PWR were performed, base and verification. Base calculations were needed to determine necessary minimum flowrate for steam generators feeding in such a way that they are not overfilled or emptied. Namely, it was assumed that instrumentation is not available during extended SBO. The verification calculations have been then performed to verify if the determined minimum flowrates are sufficient to prevent the core heatup.

The calculated results show effectiveness of the proposed extended SBO prevention strategy provided that the water injection is available in the first two hours after SBO occurring at full power. If diesel generator is running after loss of offsite power for some time, e.g. one hour, the available time for steam generator water injection is significantly longer. The obtained results demonstrate the need for assessment of the pump injection flowrates before the utilization of the pump for mitigation of the event. The applicability of the developed method for assessment of the required pump injection flowrate has been validated on PWR.

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1 INTRODUCTION

The events at the Fukushima Dai-ichi nuclear power plant [1] and stress tests [2] showed that the loss of electrical power (LOOP) followed by station blackout event (SBO) and loss of the ultimate heat sink (UHS) can have large impact on the safety of the nuclear power plant (NPP). The SBO event when power from all emergency power sources, including batteries, is lost is named extended SBO and leads ultimately to core heatup and core damage [3]. After Fukushima Dai-ichi accident the strategies were proposed for coping with such events. They include utilization of portable equipment, permanent equipment or combinations of portable and permanent equipment. Severe accident management guidelines (SAMGs) have been also updated [5]. In the United States of America (USA) the Diverse and Flexible Coping Strategies (FLEX) [4] have been developed which are focused on maintaining or restoring key plant safety functions. Further, in FLEX guidelines [4] it is stated that while FLEX strategies are focused on the prevention of fuel damage, these strategies would be available to support accident mitigation efforts following fuel damage. However, coordination of the FLEX equipment with Severe Accident Management Guidelines (SAMGs) is not within the scope of the guideline.

For the LOOP, SBO and the loss of the UHS scenarios, cooling of the core can be established by means of water injection to reactor coolant system (RCS) and/or steam generators (SGs). For RCS injection the source of borated water is needed. On the other hand, for SG makep sustained source of water is needed. Different systems for performing this function have been identified in the stress test reports, including electric power-independent turbine driven pumps, arrangements for gravity feed, dedicated diesel driven pumps and pre-installed connections for external feed, such as from the on-site fire trucks [6].

The FLEX strategies also suggest development of thermal hydraulic analyses to support plant specific decision-making. Therefore in this paper utilization of the pump, either turbine driven auxiliary feedwater or portable pump, for mitigation of the extended SBO event and prevention of core damage in pressurized water reactor (PWR) is investigated. It presents a follow-up study to stress tests [7]. Methodology is developed for assessment of the necessary fix/portable pump flowrates within analysed time period. It should be noted that the FLEX strategy to inject water into steam generator [4] suggests that in certain circumstances, auxiliary feedwater (AFW) system operation may be extended by throttling flow to a constant rate, rather than by stroking valves in open-shut cycles. Further, the assessed flowrates should also prevent the core damage without overfilling the steam generators in the analysed scenarios.

The paper is organised as follows. The description of the deterministic safety analysis input model, the methodology used for the assessment of the necessary injection flow of the pump to SGs and developed case scenarios is given in Section 2, while in Section 3 the obtained results from the deterministic safety analyses are given.

2 METHODS DESCRIPTION

For calculations the RELAP/MOD3.3 Patch 04 thermal-hydraulic computer has been used [8]. The RELAP5 input model of NPP with two-loop PWR, described in details in refs. [7] and [9] is briefly described first. The method for calculation of necessary pump flowrate during extended SBO event is described. Finally, the scenarios are described.

2.1 RELAP5 Input Model

The noding scheme of PWR, used in the study, is shown in Figure 1. It was created using Symbolic Nuclear Analysis Package (SNAP) [10]. The base plant model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points. However, since in SNAP the pipes are represented as one component (not by volumes) and since heat structures connected to pipe volumes are represented as one heat structure, the number of SNAP hydraulic components is 304 and the number of heat structures is 108. The connection point for the portable pump is the same as for turbine driven auxiliary feedwater pump.

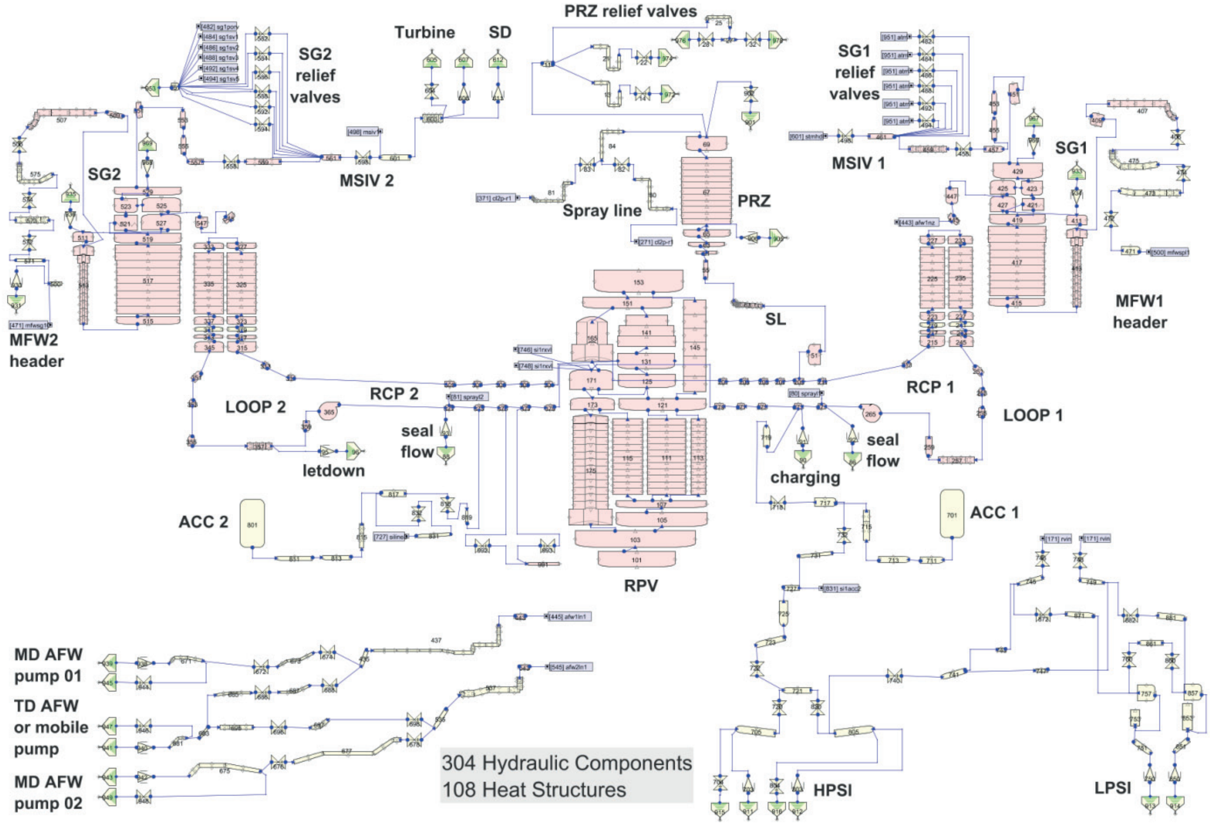

Figure 1: Noding scheme of two-loop PWR represented by SNAP

2.2 Method for Necessary Flowrate Determination

Base calculations of extended SBO with assuming available TD-AFW pump and steam generator level control are needed to determine the necessary minimum flowrate for steam generators feeding in such a way that they are not overfilled or emptied. The later verification calculations are performed to verify if the determined minimum flowrates are sufficient to prevent the core heatup. The method for necessary flowrate consists of the following steps:

- Step 1: From the base calculation for given scenario with given boundary conditions the integral of necessary water mass to be injected into steam generators is obtained to restore and maintain the desired water level. The analysis and evaluation of the cumulative water mass is done for time a window equal to the operational time of the pump during extended SBO. The scenarios that in the analysed time window result in the core heatup and damage need not to be considered in verification calculations.
- Step 2: From cumulative water mass the necessary mass flowrate of the pump for steam generators makeup has to be set so, that the integral water mass injected in the steam generators at a given time ideally corresponds to the mass, determined in step 1, but in any case remains in the operable range of the steam generators (between 8% and 96%). In the simplest case the necessary mass flowrate is set constant for a longer time period. Care should be taken that the mass flowrate is high enough because of the initial period, in which the residual power is higher than in later times. It may turn out that for longer time periods this may not be achieved with a constant mass flowrate (i.e. the constant flow to

be too small in initial period) and that more adjustments of the mass flowrate are required as we progress with time.

- Step 3: The verification calculations are performed with the determined necessary flowrate in the Step 2. These calculations verify if injected water into the SGs, for given scenario, prevents core heat up and SGs overfill.

2.3 Scenarios description

Different scenario types have been investigated to demonstrate proposed method for necessary flow, including plant design improvements which may be done to better cope with extended SBO. During extended SBO three different RCS coolant loss pathways, with corresponding leakage rate, can be expected in the PWR: (a) normal system leakage, (b) RCPs seal leakage, and (c) RCS coolant loss through letdown relief valve unless automatically isolated or until isolation is procedurally directed. Normal system leakage is assumed to be equal to plant technical specifications identified leakage (0.63 l/s at nominal RCS conditions) for selected PWR. The seal leak rate of 1.32 l/s per RCP (at nominal RCS conditions) due to loss of seal cooling and RCP pump stop is assumed as representative for the plants using a high temperature O-ring RCP seal package [11]. The seal leakage can be practically prevented (negligible loss of the order of 0.06 l/s) with the installation of special passive RCP thermal shutdown seals [12]. At nominal RCS pressure the letdown loss of 5.68 l/s is expected and this value was assumed in the study.

To limit the RCS loss and enable passive accumulator injection the RCS depressurization (through the depressurization of the secondary side) scenarios have been also investigated.

The main assumptions considered in the model for verification calculations are the following:

- A1) Loss of all electric power in the plant, including batteries, what results in loss of active safety systems depending on AC power and loss of all instrumentation and control in the plant.

- A2) The pump, TD-AFW or portable (through available connection point), is available for the whole analysed period for injection of water in the steam generators. The portable pump flow measurement and regulation is also assumed to be available in the study. The TD-AFW pump speed is manually controlled. Hand wheels are provided for local manual operation of TD-AFW control valves. The TD-AFW pump flowrate local indicators, not relying on electric power, are also available locally.

- A3) Availability of water for operation of the water pump is assumed in the model.

- A4) Pressurizer and steam generator safety valves are assumed available.

- A5) The nitrogen gas required for the operation of the steam generator power operated relief valves (SG PORVs) is assumed available in the RCS depressurization scenarios. The alternative compressed air supply from the portable diesel compressor is providing required gas.

- A6) The criterion used as indication for the steam generator overfill is 96% of wide range (WR) steam generator level, while for the loss of heat sink is 8%.

- A7) The criterion used for core heatup is significant core uncovery causing core heatup start.

In base calculations it was in addition assumed the operable TD-AFW with all instrumentation and control. The regulation of the TD-AFW was set to restore and maintain narrow range level at 69% (plant normal level, at normal power condition this means 77% wide range level). Using such assumptions the integrated mass injected to steam generators is obtained, which satisfy A6 criterion that SGs are not overfilled.

Six types of RCS coolant loss scenarios, given in Table 1, have been developed and analysed: the NO_LOSS (no RCS loss), N_LOSS with normal system leakage, the S_LOSS with RCP seal loss starting one hour after the start of extended SBO, the SL_LOSS with RCP seal loss and loss of coolant through the letdown relief valve when RCS pressure is greater than 4.24 MPa, SLD_LOSS with RCP seal and letdown loss (if RCS pressure is greater than 4.24 MPa) and depressurization of the primary side through the secondary side to 1.57 MPa, started 30 minutes after SBO occurrence, and NSLD_LOSS (SLD_LOSS case with additionally assumed normal system leakage).

Table 1 Types of RCS Coolant Loss Scenarios

Scenario type	**N**ormal system leakage	**S**eal loss	**L**etdown loss	**D**epressurization
NO_LOSS	no	no	no	no
N_LOSS	yes	no	no	no
S_LOSS	no	yes	no	no
SL_LOSS	no	yes	yes	no
SLD_LOSS	no	yes	yes	yes
NSLD_LOSS	yes	yes	yes	yes

For each type of loss scenarios given in Table 1 two cases were simulated:

(1) In the Case 1 (C1) it is assumed that the emergency diesel generators (EDG) started and normally operated for one hour after the loss of offsite power. Availability of all safety systems is assumed during that hour. After one hour the extended SBO is assumed. It was similar in Fukushima Dai-ichi event when the EDGs were running for 45 minutes until tsunami arrived.

(2) In Case 2 (C2) it is assumed that the extended SBO occurrence is concurrent with LOOP, resulting in large decay heat in the initial period. This is more severe as larger time delays of SBO start (i.e. delays of EDG loss) would mean smaller decay heat levels at the time of SBO occurrence (i.e. less heat to be removed from the core).

Different time delays between the extended SBO start and start of the pump injections to steam generators have been considered and analysed in base calculations. In the Case 1 delays 0, 0.5 h, 1 h, 2 h, 3 h, 4 h and 5 h (42 scenarios in total) and in the Case 2 delays 0, 0.5 h, 1 h, 2 h and 3 h were considered (30 scenarios in total). The scenario name is compouned from scenario type name and the delay time (e.g. NO_LOSS type scenario with 4 hour delay of steam generator feeding is labelled as NO_LOSS_4). The no delay scenarios have been analysed for base calculations to verify if for selected scenario type the core heatup could be prevented by proposed strategy with SGs makeup.

The calculations have been performed for 72 h, consisting of two time intervals for injections. The first time interval is 24 h reduced for pump start delay. The second time interval lasts from 24 h to 72 h. In the simplest case, the flowrate in the interval is constant.

3 RESULTS

3.1 Base Calculations

The obtained necessary flowrates obtained from base calculations using method described in Section 2.2 are shown in Table 2. The first row in Table 2 specifies the RCS coolant loss scenario type defined in Section **Error! Reference source not found.**. The first column contains the delay time, in hours, of the pump start following the extended SBO. The pump constant flows in the first time interval (from pump start until 24 h), in kg/s, are given in the remaining columns for Case 1 and Case 2 scenarios. The largest minimum necessary flows are obtained for SLD_LOSS type scenarios and the smallest for S_LOSS type scenarios. For the second time interval starting 24 h after SBO start and lasting 48 h the constant mass flow of 3 kg/s was assumed. Finally, Table 2 shows that core heatup and damage is calculated for some of the analysed scenarios.

Table 2: Minimum Necessary Constant Flowrates in First Time Interval (from Pump Start until 24 h after SBO Start)

Scenario type	NO_LOSS	N_LOSS	S_LOSS	SL_LOSS	SLD_LOSS	NSLD_LOSS
Case 1, EDG running 1 h						
Delay (h)	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*
0	4.79	4.77	4.68	N.A.	5.51	5.50
0.5	4.90	4.87	4.78	N.A.	5.63	5.62
1	4.99	4.98	4.88	N.A.	5.75	5.74
2	5.22	5.20	5.11	N.A.	6.00	5.99
3	5.47	5.45	5.34	N.A.	6.25	6.25
4	5.63	5.62	5.51	N.A.	6.48	6.47
5	N.A.	N.A	N.A.	N.A.	N.A.	N.A.
Case 2, EDG not running						
Delay (h)	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*	Flow (kg/s)*
0	5.50	5.46	5.38	N.A.	6.21	6.19
0.5	5.60	5.58	5.49	N.A.	6.34	6.33
1	5.72	5.70	5.61	N.A.	6.46	6.45
2	5.86	5.83	5.72	N.A.	6.54	6.57
3	N.A.	N.A.	N.A.	N.A.	N.A.	N.A.

* - N.A. because the core heatup occurs due to the core uncovery before 24 h

For all SL_LOSS scenarios, as shown on Figure 2(a1) and Figure 2(a2), core uncover (results until calculation was aborted) before 24 h. Obtained results are expected considering large letdown loss of RCS coolant, causing core uncovery as shown on Figure 2(a1) and Figure 2(a2). When steam generator empties (see on Figure 2(b1) and Figure 2(b2)), the RCS starts to heat what resulted in the pressure increase and additional RCS inventory loss due to pressurizer safety valves opening. Even if SG injection started before the steam generators are emptied as shown on Figure 2(b1) and Figure 2(b2), continuous RCS mass loss due to RCP seal leakages and unisolated letdown line loss resulted in the core uncovery in approximately 12 h (in case of significant accumulator injection this time may be prolonged for some hour - see scenario SL_LOSS_3 (C1)).

The core heatup, as shown in Table 2, is obtained also for other type scenarios when pump operation is delayed for extended period. Figure 3 shows that the core uncovery is not prevented for C1 scenarios with delay 5 hours and for C2 scenarios with delay 3 hours. With no heat sink for the core decay heat due to steam generator boil-off, the RCS starts to heat up and the primary pressure increases until pressurizer safety valves open. This resulted in RCS mass discharge and core uncovery with heatup. As can be seen from Figure 3, the EDG operating one hour delays core uncovery for additional two hours. This demonstrated that operation of safety systems in the initial hour besides cooling the core gives additional time to operators and requires smaller injection flows (smaller demand for pump flow capacity). In the worst case scenario, with SBO occurring at full power, the core starts to uncover after 2 h in case no injection to steam generators is available. This means little available time to the operators, especially to use the portable pump.

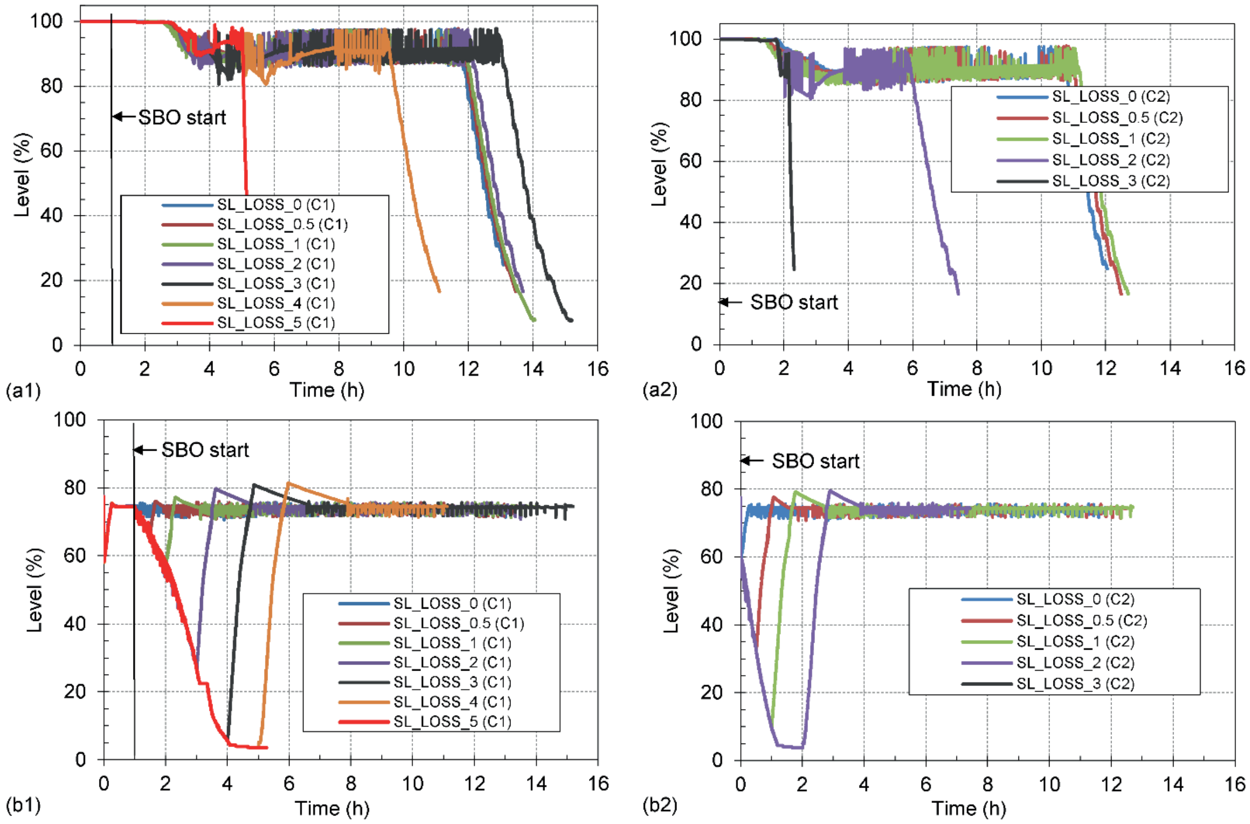

Figure 2 Comparison of C1 and C2 Cases for SL_LOSS Type Base Calculations with Different SG Injection Delays: (a) Core collapsed liquid level, (b) SG. no. 2 WR level.

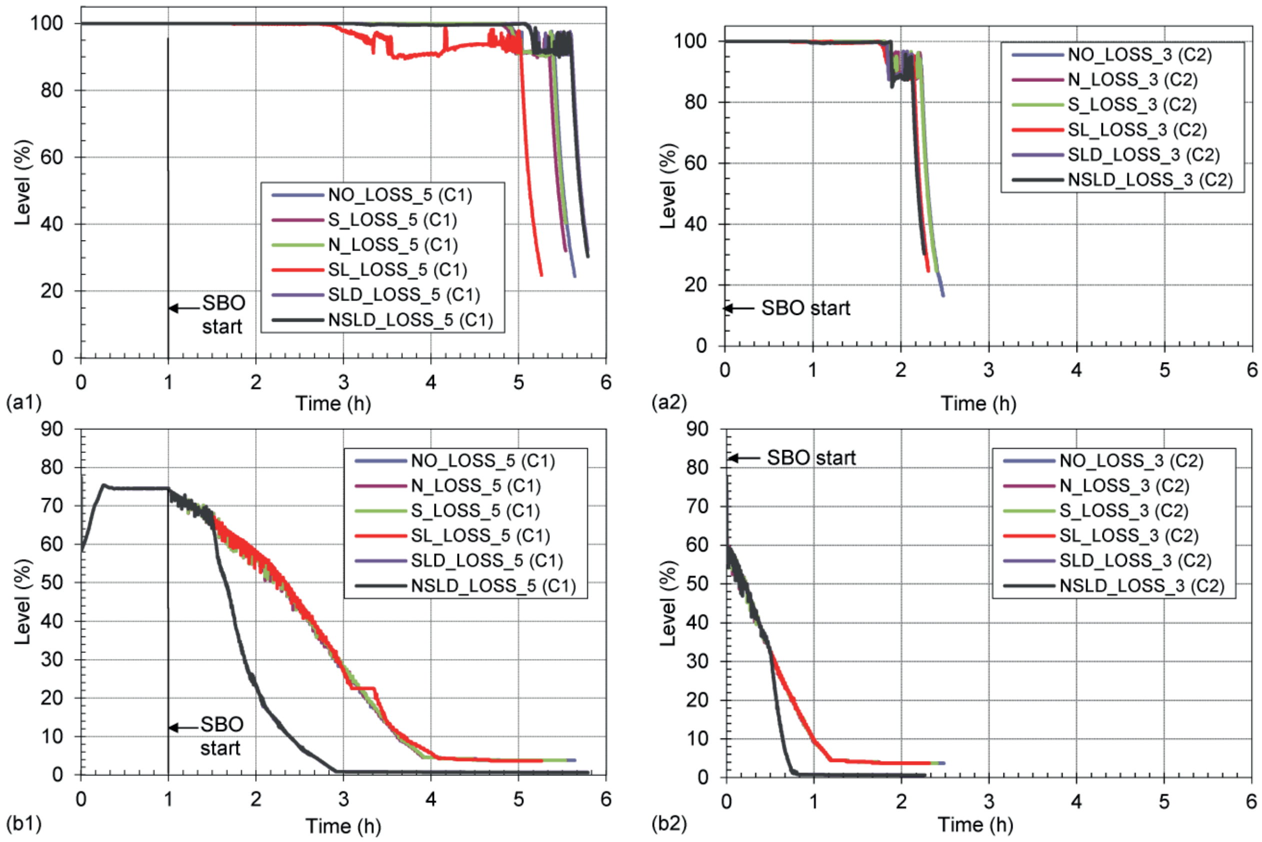

Figure 3: Comparison of Base Calculations for C1 (5 h Injection Delay) and C2 (3 h Injection Delay): (a) Core collapsed liquid level, (b) SG. no. 2 WR level.

3.2 Verification Calculations

The third step in the methodology presented in Section 2.2 is to verify the calculated minimum necessary pump flowrates given in Table 2. The verification calculations are performed for all scenarios listed in Table 2 in which core heat up is prevented. The results are shown in Figures 4 through 7 for RCS pressure, core collapsed liquid level, SG no. 1 and SG no. 2 wide range level, respectively. The results are shown for the 72 h except for RCS pressure the 24 h interval is used as the initial hours are the most important from the point of RCS discharge through safety valves. Later (after one day) the RCS pressure start to follow the secondary pressure or it is on this trend. Each of Figures 4 through 7 show five scenario types with different assumed injection delays (all types except SL_LOSS) for both cases (C1 and C2).

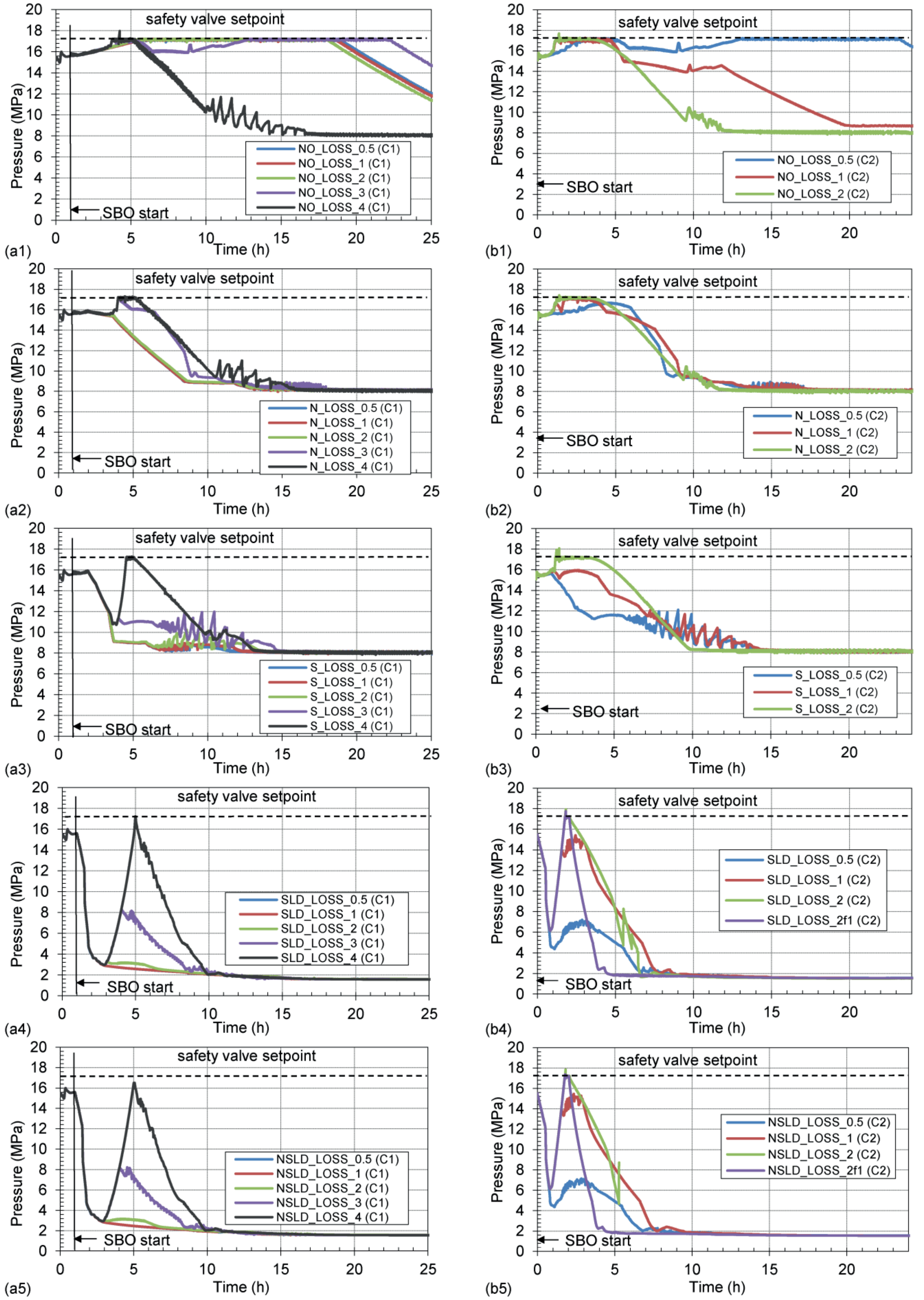

Figure 4 shows that it takes the longest time for NO_LOSS type scenarios that the RCS pressure drops below the safety valve setpoint (see Figures 4(a1) and (a2)). In all other type scenarios the one or more RCS losses are present (see Section 2.3) resulting in faster initial RCS pressure decrease. If secondary side cooling is not available, the RCS may repressurize until the injection into SGs starts. As normal RCS loss is smaller than RCS loss through RCPs, the pressure decrease is slower in N_LOSS type scenarios than in the S_LOSS type scenarios. However, if depressurization is performed, there is no qualitative difference between SLD_LOSS and NSLD_LOSS type scenarios.

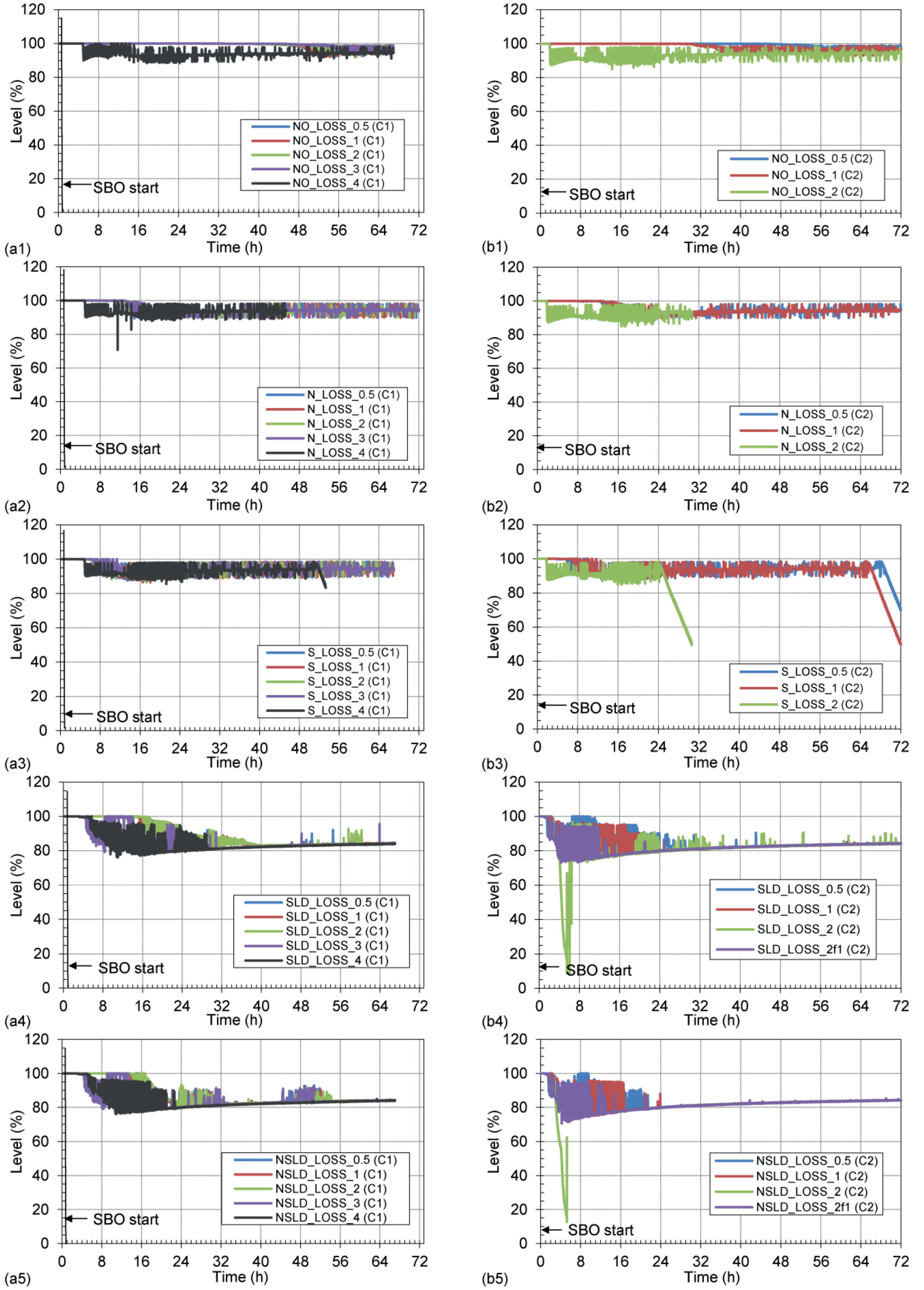

Figure 5 shows that the core in the first 24 h is not significantly uncovered except for SLD_LOSS_2 and NSLD_LOSS_2 scenarios. The reason is too small flowrate in the beginning of transient comparing to base calculations in which according to assumption the SG level was automatically maintained. When the first time interval was divided in two sub-intervals, i.e. assuming the constant flow of 10 kg/s (verification calculations SLD_LOSS_2f1 (C2) and NSLD_LOSS-2f1 (C2)) in the first four hours and 5.77 kg/s in the remaining 18 hours instead of 6.54 kg/s until 24 h, core uncovery was prevented as shown in Figures 5(b4) and 5(b5). However, when larger RCS loss is present (S_LOCA type scenarios, Case 2), after one day the core may start to uncover, even if there is sufficient water inventory in the steam generators. This time may be prolonged, if injection to SGs is started very early (in the first hour after SBO occurrence). On the other hand, if EDG is running one hour after LOOP with reactor trip (Case 1), the decay heat level is lower than after reactor trip and therefore the calculations showed that the core remains uncovered if injection into SGs starts in 3 h. Nevertheless, when the reactor is depressurized the core uncovery is prevented for assumed RCP seal loss and maximum normal RCS loss in the first 72 h with proposed flow injection to the steam generators.

Finally, Figures 6 and 7 show steam generator wide range levels for steam generator no. 1 and 2, respectively. It is clearly demonstrated the efficiency of the method for necessary flow determination and at the same time not to overfill or empty steam generators. It can be seen that steam generator levels in all shown scenarios are well maintained. As one pump is feeding both steam generators, in certain scenarios (e.g. SLD_LOSS_3 (C1)) the SG level oscillates. It should be noted, when level in one steam generator increases, in the other decreases, and vice versa. The small oscillations present in the SG level trends are due to SG relief valves discharge at the discrete periods. Nevertheless, later the filling of both steam generators is smooth and similar for both steam generators.

The assessed flowrates from the TD-AFW cumulative injected mass in the base calculations of analysed case scenarios minimize the required number of flow changes and potential operator's errors during required manipulations. The assessed flow for steam generator makeup assures effective core cooling without overfilling the steam generators.

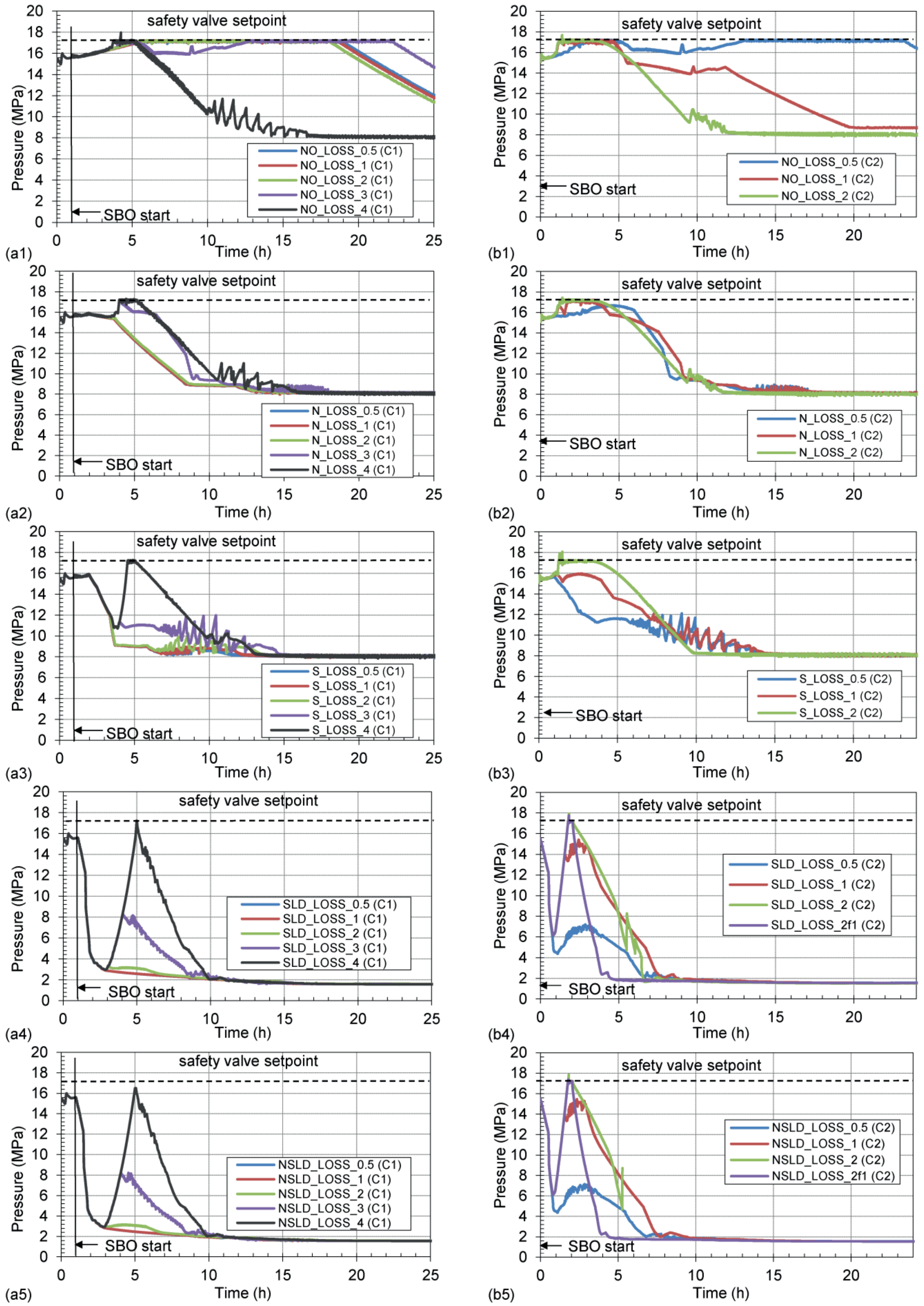

Figure 4: Dependence of RCS Pressure on Injection Time Delays during Different Scenario Types for: (a) C1 – SBO Start 1 h after LOOP, (b) C2 – SBO Concurrent with LOOP.

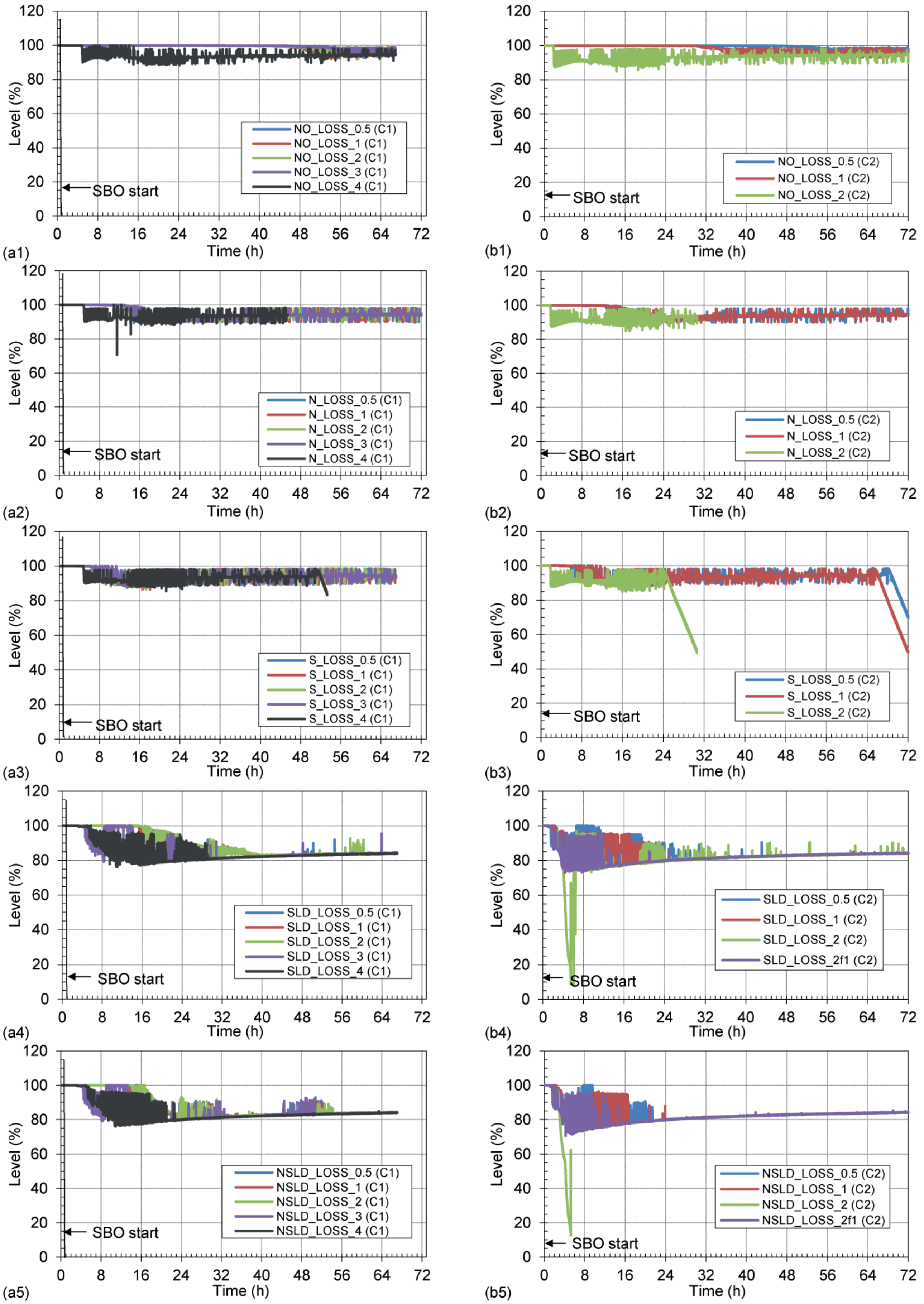

Figure 5: Dependence of Core Collapsed Liquid Level on Injection Time Delays during Different Scenario Types: (a) C1 – SBO Start 1 h after LOOP, (b) C2 – SBO Concurrent with LOOP.

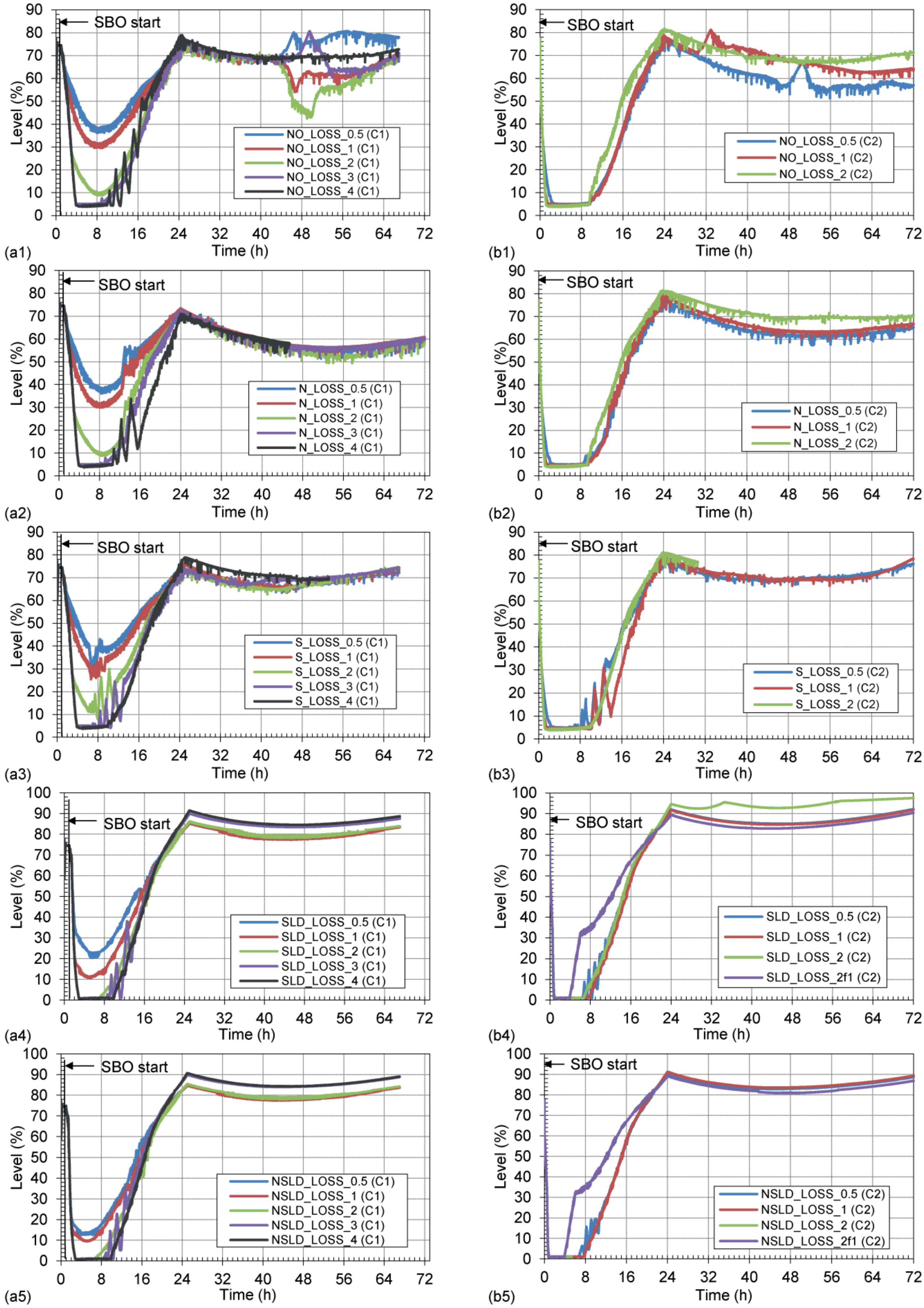

Figure 6: Dependence of SG No. 1 WR Level on Injection Time Delays during Different Scenario Types: (a) C1 – SBO Start 1 h after LOOP, (b) C2 – SBO Concurrent with LOOP.

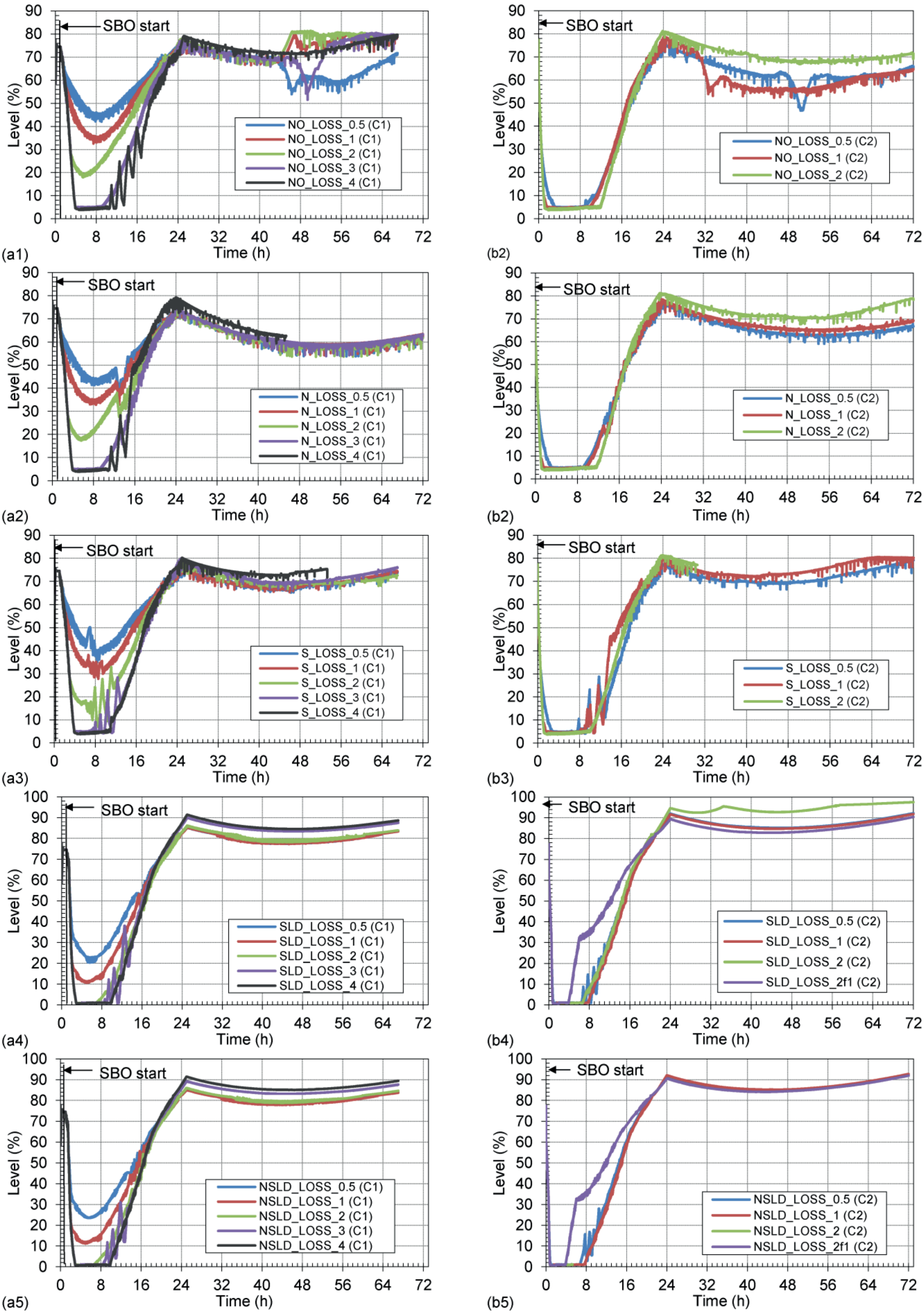

Figure 7: Dependence of SG No. 2 WR Level on Injection Time Delays during Different Scenario Types: (a) C1 – SBO Start 1 h after LOOP, (b) C2 – SBO Concurrent with LOOP.

4 CONCLUSION

To support the FLEX implementation to prevent damage to the fuel in the reactor the RELAP5/MOD3.3 Patch04 thermal-hydraulic system code has been used to study the utilization of pump for mitigation of the extended blackout condition. The method for the assessment of the necessary injection flowrate for steam generators makeup for mitigation of extended blackout event has been proposed.

Six different scenarios of reactor coolant loss have been developed and analysed. Two cases have been defined considering the operation of the emergency diesel generators. Different delays of the pump injection (into steam generators) start following the station blackout have been assumed and analysed.

Obtained results show that typical pressurized water reactor with the leak tight reactor coolant pump seals with small seal loss or depressurization of the reactor coolant system can cope the first 72 hours of extended SBO if pump is available to start inject into secondary side within two hours of the SBO start. Operation of the emergency diesel generator for one hour extends the available time for the start of pump on four hours. Failure to isolate the letdown results in the core damage before 24 h in all analysed scenarios.

The effective mitigation strategy for the extended blackout condition can be developed with the utilization of the presented method. The method can be modified for different type of pumps and their characteristics. One of the main conclusions in the study is that availability of equipment is prerequisite but not guarantee of successful mitigation. Namely, the verification analyses suggest that the time available requires quick response of operators in case of using portable equipment. On the other hand, base calculations show that there may be some delay in TD-AFW pump start and that it is urgent to isolate letdown line in order to prevent core heatup, when no RCS injection is available.

ACKNOWLEDGMENTS

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