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RELAP5 Analysis of Mitigation Strategy for Extended Blackout Power Condition in PWR

Prepared by:
Andrej Prošek, Andrija Volkanovski

Jožef Stefan Institute
Jamova cesta 39
SI-1000 Ljubljana, Slovenia

K. Tien, NRC Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
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ABSTRACT

A set of measures have been proposed and currently implemented in response of the accident at the Fukushima Daiichi nuclear power plant. Those measures include diverse and flexible mitigation strategies that increase the defense-in-depth for beyond-design-basis scenarios. Mitigation strategies are based on the utilization of the portable equipment to provide power and water to the nuclear power plants in order to maintain or restore key safety functions. The verification of the proposed measures with the plant specific safety analyses is endorsed in the mitigation strategies. The purpose of the study was to investigate the application of the deterministic safety analysis for mitigation strategy of the extended station blackout (SBO). A methodology for the assessment of flowrates for steam generator makeup using portable pump is proposed. The aim is to fill steam generator without available information on level in such a way that makeup is sufficient and that at the same time the steam generators are not overfilled. The RELAP5/MOD3.3 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. The calculated results show effectiveness of the proposed extended SBO mitigation strategy. The applicability of the developed method on operational power plant is validated.

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The events at the Fukushima Daiichi nuclear power plant and stress tests 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.

The strategies proposed for coping with these events include utilization of portable equipment, permanent equipment or combinations of portable and permanent equipment. The Diverse and Flexible Coping Strategies (FLEX) are focused on maintaining or restoring key plant safety functions. The FLEX strategies propose development and application of thermal hydraulic analyses to support plant specific decision-making.

In this study 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 European Union stress tests. Methodology is developed for assessment of the required pump flowrates within analyzed time period. The assessed flow rates should prevent core damage without overfilling the steam generators in the analyzed scenarios.

The RELAP/MOD3.3 Patch 04 input model of NPP with two-loop PWR is used in this study. The major difference between boiling water reactor and PWR is PWR has coolant under pressure in its primary cooling/heat transfer circuit, and generates steam in a secondary circuit while boiling water reactor (BWR) makes steam in the primary circuit above the reactor core. This PWR feature gives possibility to cool the reactor through the secondary side also during extended SBO providing that makeup water for secondary side is established.

Different scenario types are investigated including plant design improvements which may be done to better cope with extended SBO. Three different reactor coolant system (RCS) coolant loss pathways, with corresponding leakage rate, can be expected in the PWR plant during the extended SBO: normal system leakage, reactor coolant pump seal leakage and loss of RCS coolant through letdown relief valve unless automatically isolated or until isolation is procedurally directed.

The required injection rates to the steam generators in the first 24 h and from 24 h to 72 h are calculated from the cumulative water mass injected by the turbine driven auxiliary feedwater pump in the analyzed scenarios, when desired normal level is maintained automatically. It means that flow has to be adjusted just two times in 72 h. After determining the required flowrates the verification analyses using portable pump with required flows are performed.

The effective prevention strategy of core heatup during the extended blackout condition can be developed with the utilization of the presented method. The method can be extended from the analyzed 72 h and can be modified for different type of pumps and their characteristics.

ACKNOWLEDGMENTS

The authors acknowledge the financial support from Krško Nuclear Power Plant and Slovenian Nuclear Safety Administration within CAMP program (project no. POG-U3-KE-R4/104/12 – also NEK no.: 3120118) and from the state budget by the Slovenian Research Agency (program no. P2-0026).

ABBREVIATIONS

AFW	auxiliary feedwater
ASCII	American Standard Code for Information Interchange
BWR	boiling water reactor
CVCS	chemical and volume control system
ECCS	emergency core cooling system
EDG	emergency diesel generator
FLEX	Diverse and Flexible Coping Strategies
HPSI	high pressure safety injection
LOOP	loss of electrical power
LPSI	low pressure safety injection
MFW	main feedwater
MSIV	main steam isolation valve
NPP	nuclear power plant
PORV	power operated relief valve
PRZ	pressurizer
PWR	Pressurized Water Reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RELAP	Reactor Excursion and Leak Analysis Program
RPV	reactor pressure vessel
SBO	station blackout
SG	steam generator
SL	surge line
SNAP	Symbolic Nuclear Analysis Package
TD-AFW	turbine driven auxiliary feedwater
UHS	ultimate heat sink
WR	wide range

1. INTRODUCTION

The events at the Fukushima Daiichi nuclear power plant (Ref. 1) and stress tests (Refs. 2 and 3) 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 (Ref. 4).

The strategies proposed for coping with these events include utilization of portable equipment, permanent equipment or combinations of portable and permanent equipment. The Diverse and Flexible Coping Strategies (FLEX) (Ref. 5) are focused on maintaining or restoring key plant safety functions. The FLEX strategies propose development and application of thermal hydraulic analyses to support plant specific decision-making (Ref. 5).

For the LOOP, SBO and the loss of the UHS scenarios, maintaining water injection to reactors and/or steam generators (SGs) offers an ultimate means of cooling the core. 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 (Ref. 3). The availability of the pre-arranged connections and drills for actual establishment of the connection for feeding of steam generators is identified as prerequisite for the utilization of the fire trucks.

In this study the 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. Methodology is developed for assessment of the necessary pump flowrates within analyzed time period. The assessed flow rates should prevent core damage without overfilling the steam generators in the analyzed scenarios, therefore verification analyses are performed. 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. Obtained results from the deterministic safety analyses are given in Section 3 both for base calculations to assess the necessary flow and the verification calculations using the assessed pump constant flows.

2. METHOD FOR ASSESSMENT OF STEAM GENERATOR MAKEUP FLOW

The operation of pump, utilized for SG injection during extended SBO, is assumed to be divided into several intervals. The flow rate of the pump is constant in each interval. The pump can operate in continuous or discrete mode. The methodology is applicable for both cases. The operation with constant flow and long time intervals minimizes the required number of flow changes and operators errors during required manipulations. The method for assessing the necessary makeup flow into SG during extended SBO is shown in Figure 1.

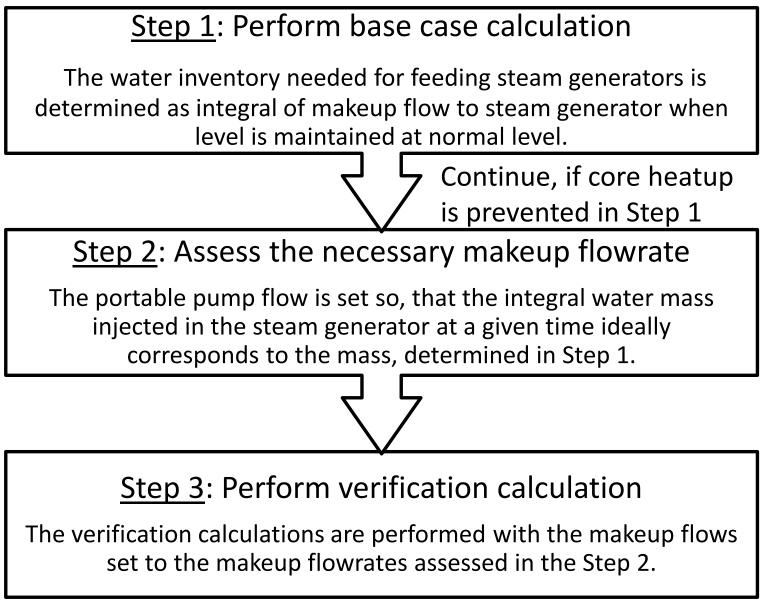


Figure 1 Method for Necessary Makeup Flow Into SGs During Extended SBO

It consists of three steps. In the Step 1 the base case calculation is performed for selected SBO scenario, but it is assumed that SG levels are maintained at normal level using artificial level control system. In this way, the required water mass, which has to be injected in the steam generator to restore and maintain the desired water level, is obtained. The analysis and assessment of the required 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 are not analyzed further. If core heatup is prevented in Step 1 then the mass flow rate of the portable pump has to be set so, that the integral water mass injected in the steam generator 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 generator (between 8% and 96% for selected PWR). In the simplest case the mass flow rate of the portable pump is not adjusted regularly, but is set constant for a longer time period. Care should be taken that the mass flow rate is high enough that in the initial period, when the residual power is higher, the steam generator is not emptied. It may turn out that for longer time periods this may not be achieved with a constant mass flow rate and that more adjustments of the mass flow rate are required. Finally, in Step 3 the RELAP5 verification calculations are performed with the pump flows set to the necessary flow rates assessed in the Step 2. These calculations verify if calculated water injection in the SGs, for given scenario, prevents core heat up and SGs overfill.

3. RELAP5 INPUT MODEL AND SCENARIOS DESCRIPTION

3.1 RELAP5 Input Model

To perform the analyses, the base RELAP5 input model of Krško NPP has been used. Krško NPP is a two loop PWR, Westinghouse type, with reactor power uprated to 1994 MW. The input model has been validated by plant transients (e.g. Ref. 6). It has been used for several safety analyses including reference calculations for Krško full scope simulator verification (Refs. 7 and 8). The base model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points. When imported ASCI file into SNAP, the following hydraulic components view has been generated semi automatically (hydraulic components with connections generated automatically, annotations and layout manually). In terms of SNAP 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 counted as one component in SNAP and not as many heat structures as pipe volumes like counted in RELAP5 output file. This explains the difference in numbers of heat structures in Figure 2 and that reported in RELAP5 output file.

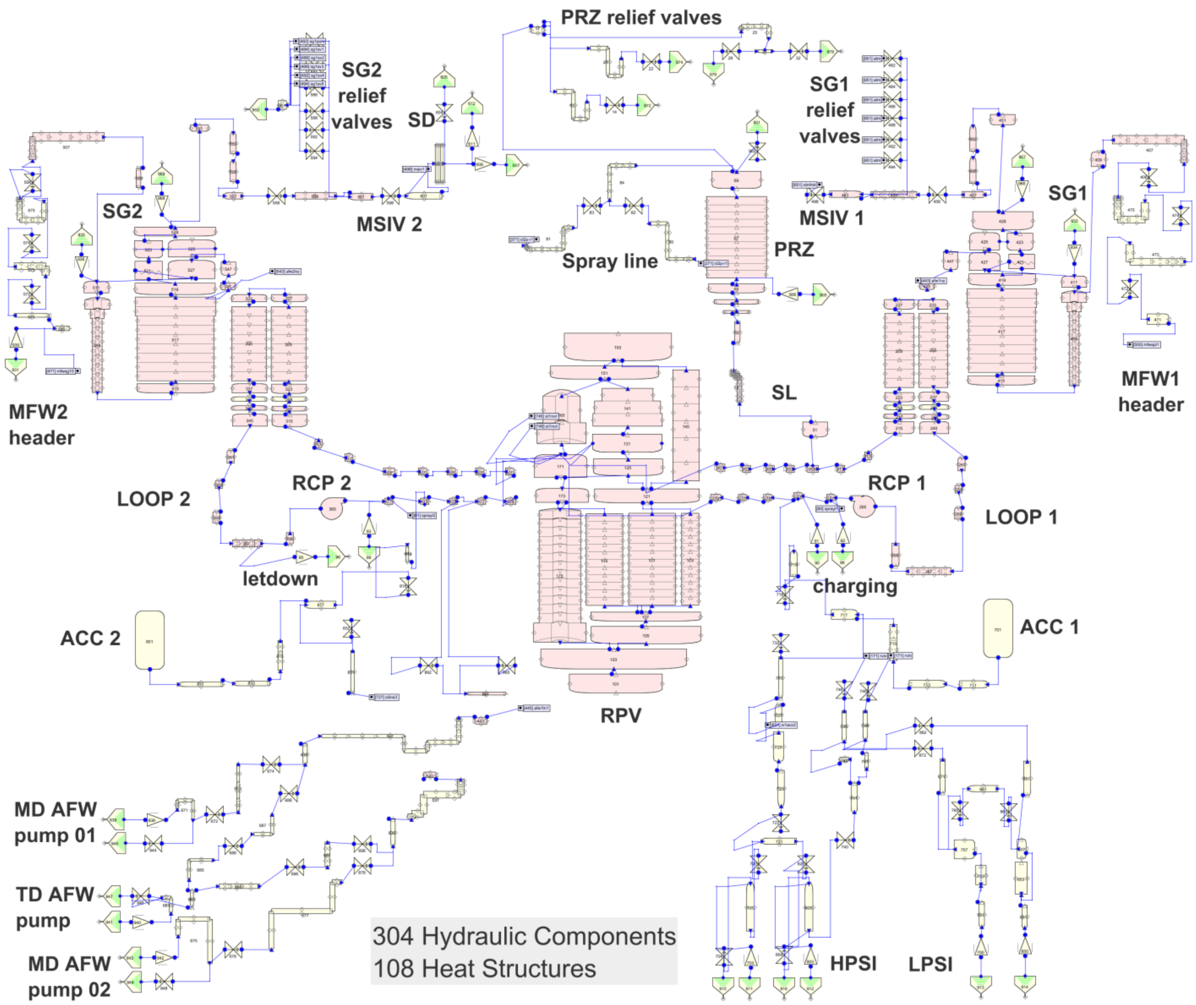


Figure 2 RELAP5 Krško NPP Hydraulic Components View

Modeling 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 (CVCS) charging and letdown flow, and reactor coolant pump (RCP) seal flow. Emergency core cooling system (ECCS) piping includes high pressure safety injection (HPSI) pumps, accumulators (ACCs), and low pressure safety injection (LPSI) pumps. The secondary side consists of the SG secondary side, main steam line, main steam isolation valves (MSIVs), SG relief and safety valves, and main feedwater (MFW) piping. The turbine valve is modeled by the corresponding logic. The turbine is represented by time dependent volume. The MFW and AFW (auxiliary feedwater) pumps are modeled as time dependent junctions.

3.2 Main Assumptions for Extended SBO

The main assumptions considered in the input model for base calculations of extended SBO are the following:

- A1) The reactor automatically trips and all rods are inserted following the LOOP;
- A2) The systems that are available for cooling of the reactor coolant system (RCS) during LOOP and SBO are assumed unavailable in this study. In depressurization scenarios the steam generators pressure indication is assumed to be available.
- A3) The cooling on the primary side is through the pressurizer safety valves (SV) and/or leaks causing RCS inventory loss.
- A4) The integrity of the RCS is maintained with the exception of assumed loss of coolant in the analysed scenario.
- A5) Connection point to the outlet of the TD-AFW is assumed to be available for portable pump and included in the model;
- A6) Availability of water for operation of the water pump is assumed in the model.
- A7) Pressurizer and steam generator safety valves are assumed available.
- A8) The nitrogen gas required for the operation of the steam generator power operated relief valves (SG PORVs) is assumed available in the depressurization scenarios. The alternative compressed air supply from the portable diesel compressor is providing required gas.
- A9) 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%.
- A10) The criterion used for core heatup start is significant core uncovery causing core heatup.

In addition to above assumptions the following two assumptions were used for verification calculations of extended SBO:

- A11) Loss of all electric power in the plant, including batteries, during SBO condition is assumed. This plant condition is referred to in this paper as extended SBO. This results in loss of all instrumentation and control in the plant.
- A12) The pump, TD-AFW or portable, 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.

In base calculations normally operable TD-AFW is assumed, with all instrumentation and control. The regulation of the TD-AFW is 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 A9 criterion than SGs are not overfilled. Namely, in SBO event the permanent steam generator level indication is not available and when developing mitigation strategies it is helpful to know how much of mass need to be injected to SGs. The SGs are overfilled if feed flow to the steam generators is larger than required. The SG overfill can result in different deteriorating effects (Ref. 11). The potential effects of steam generator overfill are hydraulic forces, excessive dead weight loads, failure of valves to reseat, loss of emergency (auxiliary) feedwater pump turbine, steam generator tube rupture and acceleration of accumulated water.

3.3 Scenarios Description

The method for assessment of necessary makeup flowrate to SGs is demonstrated on the input model of the two-loop PWR design described in Section 3.1. Different scenario types are investigated including plant design improvements which may be done to better cope with extended SBO. Three different RCS coolant loss pathways, with corresponding leakage rate, can be expected in the PWR plant during the extended SBO:

1. Normal system leakage. It is assumed that this kind of leakage is equal to plant technical specifications identified leakage (0.63 l/s at nominal RCS conditions).
2. RCPs seal leakage. During the extended SBO the cooling of the seals is not available resulting in their heat up and loss of integrity. 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, as in reference NPP (Ref. 9). 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 (Ref. 10)
3. Loss of RCS coolant through letdown relief valve unless automatically isolated or until isolation is procedurally directed. During extended SBO and nominal RCS pressure the letdown loss of 5.68 l/s is expected. Some NPPs have option of manual isolation of the letdown. The depressurization of the primary side through the secondary side results in decrease of the RCS pressure. Decreased RCS pressure stops letdown loss of coolant and allows RCS coolant makeup from accumulators. In case of heat sink loss, the decrease of the RCS pressure also delays opening of the pressurizer safety valves and loss of the RCS coolant over those valves.

The pressurizer safety valves open when RCS increases to their set points. The RCS pressure depends on the plant parameters and is specific for each analysis. Therefore the loss of coolant over pressurizer safety valves is not included in Table 1.

Six types of RCS coolant loss scenarios, given in Table 1, are 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 (SLD case with additionally assumed 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

Two cases are developed and simulated for the RCS loss scenarios given in Table 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. Case 2 (C2) assumes that the extended SBO occurrence is concurrent with LOOP. For Case 1 availability of all safety systems is assumed one hour, before the extended SBO is assumed. This time is assumed because in Fukushima event the EDGs were running for 45 minutes until tsunami arrived. LOOP of offsite power causes reactor trip and after one hour the decay heat level already significantly drops.

For each type of scenarios different time delays between the extended SBO start and start of the pump injections to SG were considered and analysed in Step 2. For case C1 in all six type scenarios the start of pump injection was delayed 0, 0.5 h, 1 h, 2 h, 3 h, 4 h and 5 h (42 scenarios in total). For case C2 the time delays 0, 0.5 h, 1 h, 2 h and 3 h were considered (30 scenarios in total). The scenario name is composed from scenario type name and the delay time (e.g. SLD_LOSS type scenario with 4 h delay of steam generator feeding is labelled as SLD_LOSS_4). It should be noted that in addition to assumed delays the no delay scenario was analysed for base case to verify if for selected scenario type the core heatup could be prevented through cooling the RCS by secondary side (e.g. RCS makeup would be needed in case of large leaks).

The scenarios were simulated until 72 h after SBO occurrence or core heatup, whichever is earlier. In the simple case having two time intervals with constant flow the first pump operational interval selected is 24 hours reduced for start delay and the second time interval selected is from 24 h to 72 h. The constant flows for these two intervals were then used for verification calculations.

4. RESULTS

Two kinds of SBO calculations for two-loop PWR are performed, base and verification. Base calculations are needed to determine necessary flowrate for SGs feeding in such a way that it is not overfilled or emptied. The verification calculations are then performed to verify if the determined minimum flowrates are sufficient to prevent core heatup.

4.1 Base Calculations

All six types of scenarios shown in Table 1 were simulated. For SL_LOSS type scenarios, as shown on Figure 3, core heatup starts well before 24 hours.

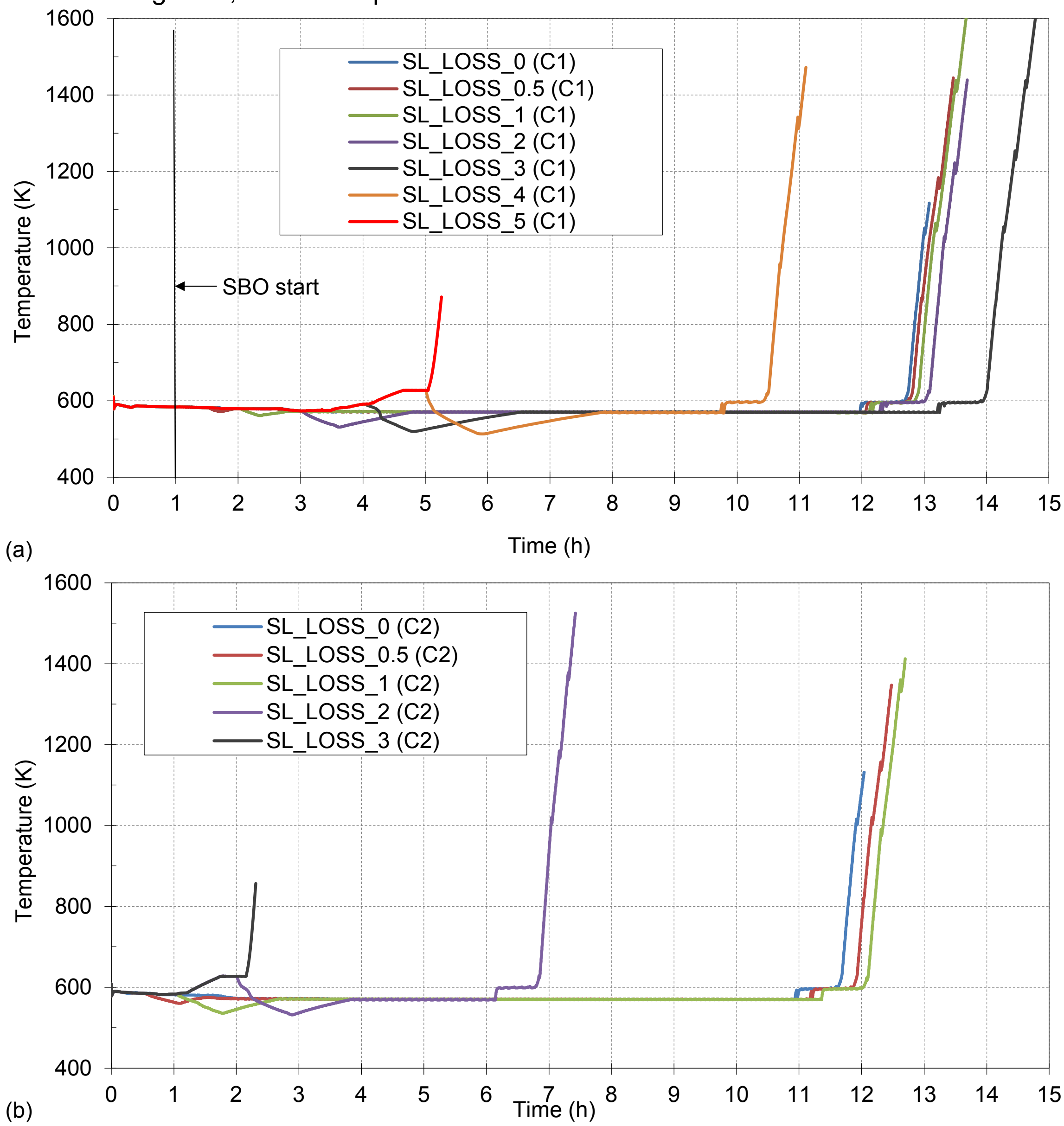
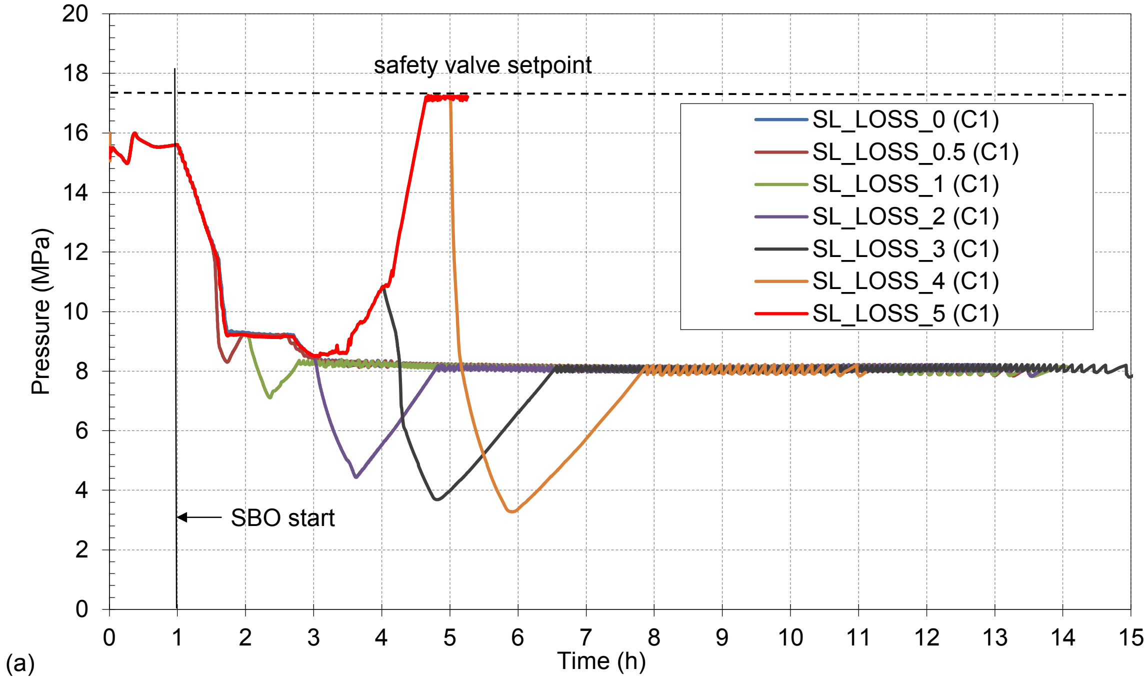


Figure 3 Influence of SG Injection Delay on Average Rod Cladding Temperature for SL_LOSS Scenario Type: (a) Case C1, (b) Case C2

Obtained results are expected considering large letdown loss of RCS coolant, and consequential core uncovery. This means that for these type scenarios cooling of RCS by secondary side is not sufficient and that RCS injection is also needed. Figure 4 shows the RCS pressure. In SL_LOSS type scenarios the pressure initially drops due to RCS loss. Later there is plateau due to coupling of primary and secondary pressure. When injection to SGs is started, the RCS pressures drop but later start to increase again due to natural circulation lost. Therefore the core boil-off started and when the core is uncovered it starts to heatup.



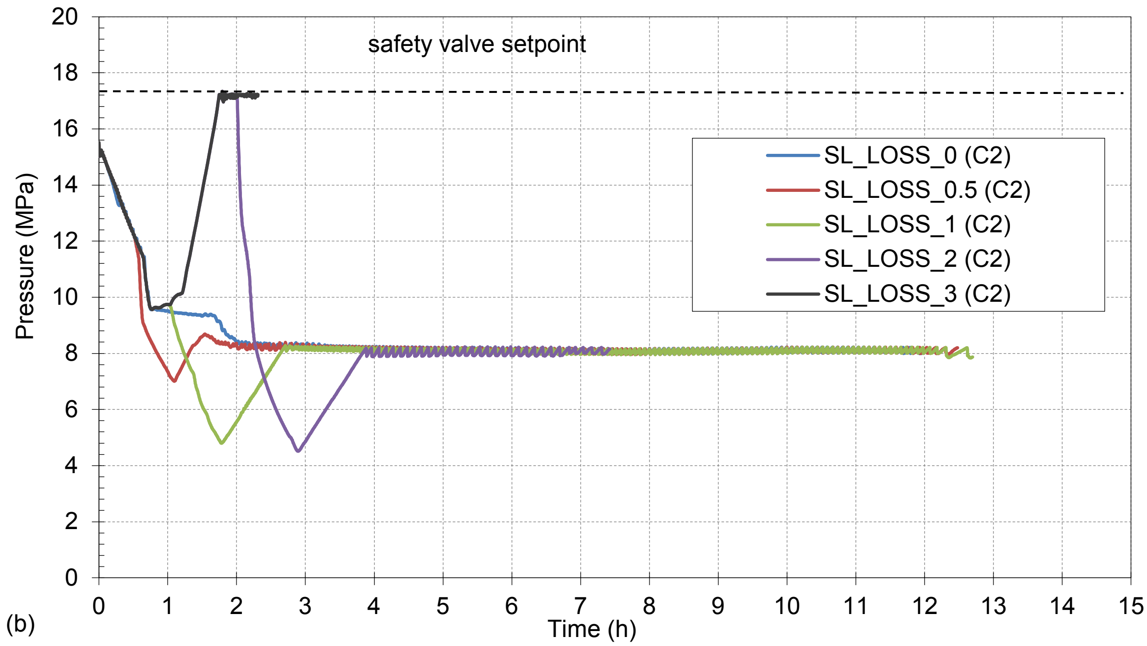


Figure 4 Influence of SG Injection Delay on RCS Pressure for SL_LOSS Scenario Type: (a) Case C1, (b) Case C2

For other than SL_LOSS type scenarios there is in general no core heatup if injection to SGs is not delayed too much. For the case C1 with EDG running one hour after reactor trip on LOOP, the time 5 h time delay in injection causes core heatup, while in C2 case with reactor trip on SBO without running EDG the 3h time delay in injection causes core heatup (see Figure 6). If delays are one hour less, only for C1 and C2 S_LOSS_4 scenarios the core heatup started after 61 h and 48 h, as shown in Figure 5.

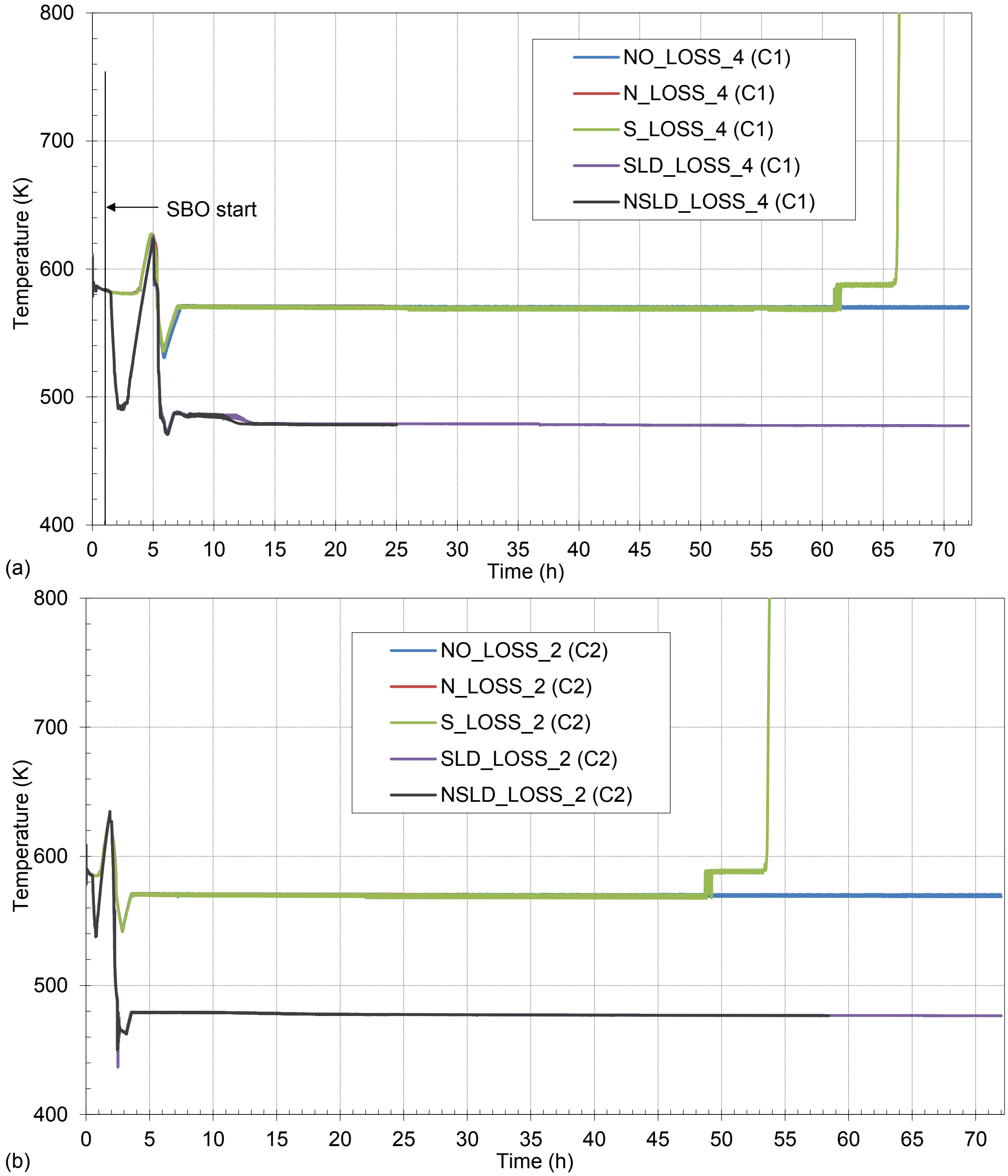


Figure 5 Comparison of Scenario Types for Average Rod Cladding Temperature: (a) Case C1 With 4 H Delay, (b) Case C2 With 2 H Delay

Figure 6 shows that core heatup is not prevented for C1 scenarios with 5 hours delay and for C2 scenarios with 3 hours delay. The reason is that the steam generators are too long dry without makeup water and so core cooling through secondary side is lost. Due to insufficient cooling, the pressurizer safety valves start to discharge the RCS mass. The large loss of RCS mass results in the core uncovery and heatup after few hours.

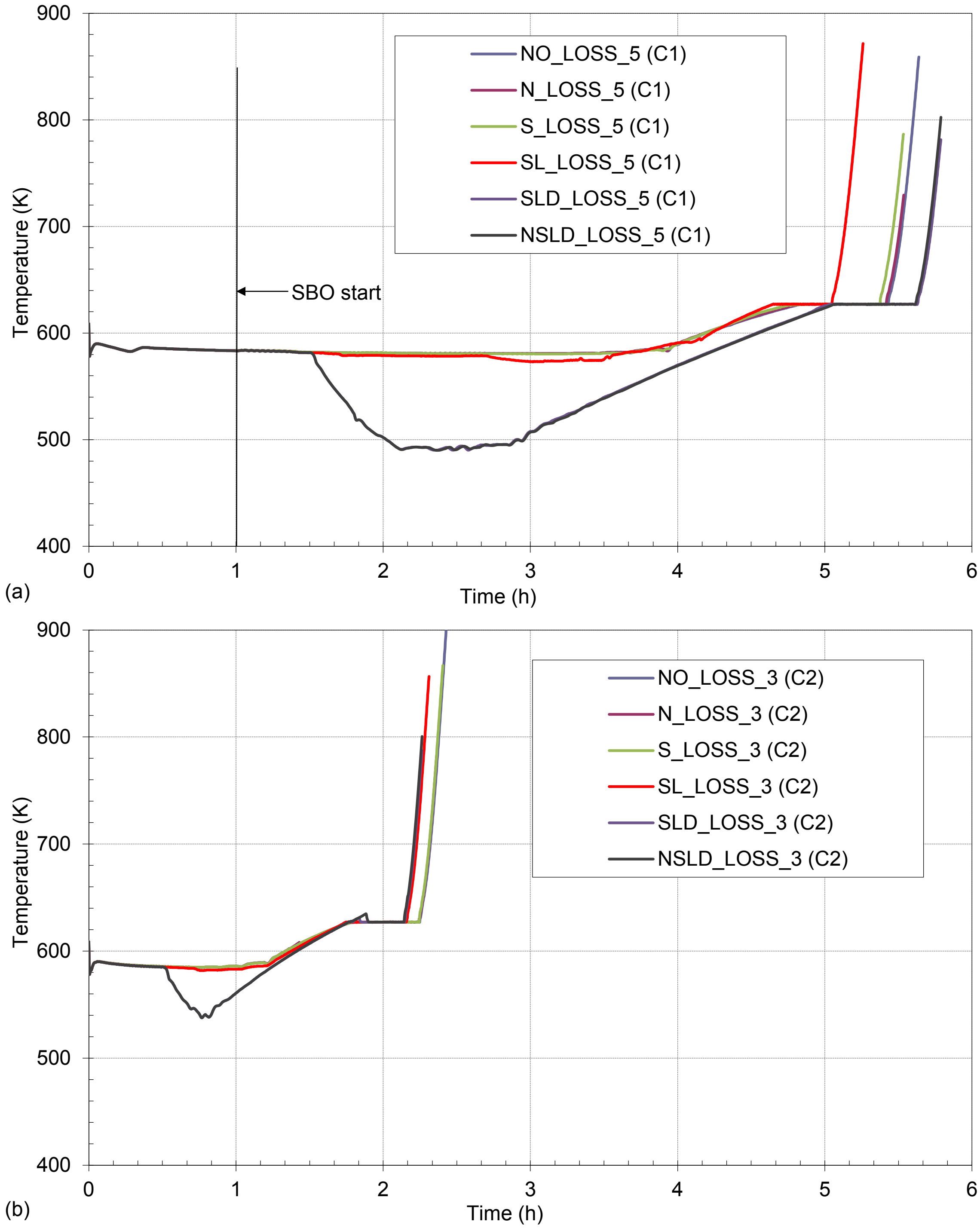


Figure 6 Comparison of Scenario Types for Average Rod Cladding Temperature: (a) Case C1 with 5 h Delay, (b) Case C2 with 3 h Delay

4.2 Assessment of Necessary Flowrates

The assessed necessary flowrates in the first 24 h for the scenarios not leading to core heatup (for both Case 1 and 2) are given in Table 2. The first row in Table 2 specifies the RCS coolant loss scenario defined in Section 3.3. The first column contains the delay time, in hours, of the pump start following the extended SBO. The pump constant flows in first 24 hours are given in the remaining columns for Case 1 and Case 2 scenarios. The largest minimum required flows are obtained for SLD_LOSS type scenarios and the smallest for S_LOSS type scenarios. For all SL_LOSS scenarios, as shown on Figure 3, core heatup starts before 24 hours. Obtained results are expected considering large letdown loss of RCS coolant, which consequently cause core uncovery and heatup.

Table 2 Minimum Necessary Constant Flows within the 24 h After SBO Start

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

For the time interval from 24 h to 72 h the minimum necessary flowrates assessed were 3.0 kg/s for NO_LOSS, SLD_LOSS and NSLD_LOSS cases and 2.8 kg/s for S_LOSS and N_LOSS.

4.3 Verification Calculations

The third step in the methodology presented in Section 2 is to verify the calculated minimum required pump flows. The verification calculations are performed for all scenarios shown in Table 2 in which core heat up is prevented in the first 24 h, except for zero delay. Namely, it is assumed that at least half an hour will be needed to start the portable pump.

The main results are shown in Figures 7 through 11. The obtained results show that if pump, TD-AFW or mobile, injects the assessed minimum necessary flows then effective cooling of reactor core is provided and SGs are not overfilled.

Results of the verification runs for SLD_LOSS type scenarios for C1 and C2 cases are given in Figure 7. Figure 7(b2) shows that in SLD_LOSS_2, Case 2 scenario with minimum required pump flow of 6.54 kg/s will result in core heat up. The TD-AFW is operating at maximum flow in the initial period of base calculation presented in previous section resulting in fast recovery of the SG inventory. The calculated constant flow of 6.54 kg/s is substantially smaller comparing to mass flow of 45 kg/s in base calculation in this initial period, therefore verification calculation resulted in insufficient cooling and core heat up. When the constant flow of 10 kg/s is assumed (verification calculation SLD_LOSS_2f1 (C2)) in the first 4 h and 5.77 kg/s in the remaining 18 h, core heatup is prevented as shown in Figure 7(b2). Figures 7(a1) and (a2) show that the RCS pressure drops very quickly after the start of the secondary side depressurization but then increase again, if secondary side injection is not started in due time. The RCS pressure in SLD_LOSS_4 scenario, Case 1 and SLD_LOSS_2, Case 2 reached the pressurizer safety valves setpoint 17.2 MPa resulting in RCS mass loss through the pressurizer safety valve. This RCS mass loss causes core uncovery resulting in heatup in SLD_LOSS_2, Case 2 as shown on Figure 7(b2). The steam generator wide range levels on Figures 7(c1), (c2), (d1) and (d2) show the steam generator levels start to increase gradually after injection with no overfill in the first 24 h.

Results of the verification runs for NSLD_LOSS type scenarios for C1 and C2 cases are shown in Figure 8. There are very similar to SLD_LOSS type scenarios because due to depressurization the contribution of normal leakage loss is rather small (up to 5 tonnes RCS mass loss in 24 h). Slightly larger loss resulted in slightly earlier core heatup when comparing SLD_LOSS_2 (C2) and NSLD_LOSS_2 (C2) calculations.

Figure 9 shows the RCS pressure, core collapsed liquid level and both steam generator wide range levels for S_LOSS type scenarios. Figures 9(b1) and (b2) show the benefit of letdown isolation and limitation of RCS loss to the RCP seals. Figures 9(c1) to (d2) show that in certain periods only one out of two steam generators is filling and vice versa. The small oscillations present in the SG level trends are due to relief valves discharge at the discrete periods.

Figure 10 shows results for N_LOSS type scenarios for C1 and C2 cases. Results of N_LOSS type scenarios for first 72 h period are similar as for S_LOCA type, except that slightly larger (around 1.5%) feed flow to SGs is needed due to smaller RCS loss in N_LOSS type scenarios.

Figure 11 shows results for NO_LOSS type scenarios for C1 and C2 cases. In the 72 h period, as shown on Figures 11(b1) and (b2), core is covered as a result of the effective cooling. In NO_LOSS scenarios the loss of secondary side heat sink results in discharge of RCS coolant through pressurizer safety valves when safety valves setpoint is reached (see Figures 11(a1) and (a2)). The duration of the pressurizer safety valves opening and by this the mass of discharged RCS coolant depends on the delay of pump start and establishment of SG feed flow. The steam generator levels recover in 24 h and the level is not overfilled in the 72 h period as shown in Figures 11(c1) to (d2).

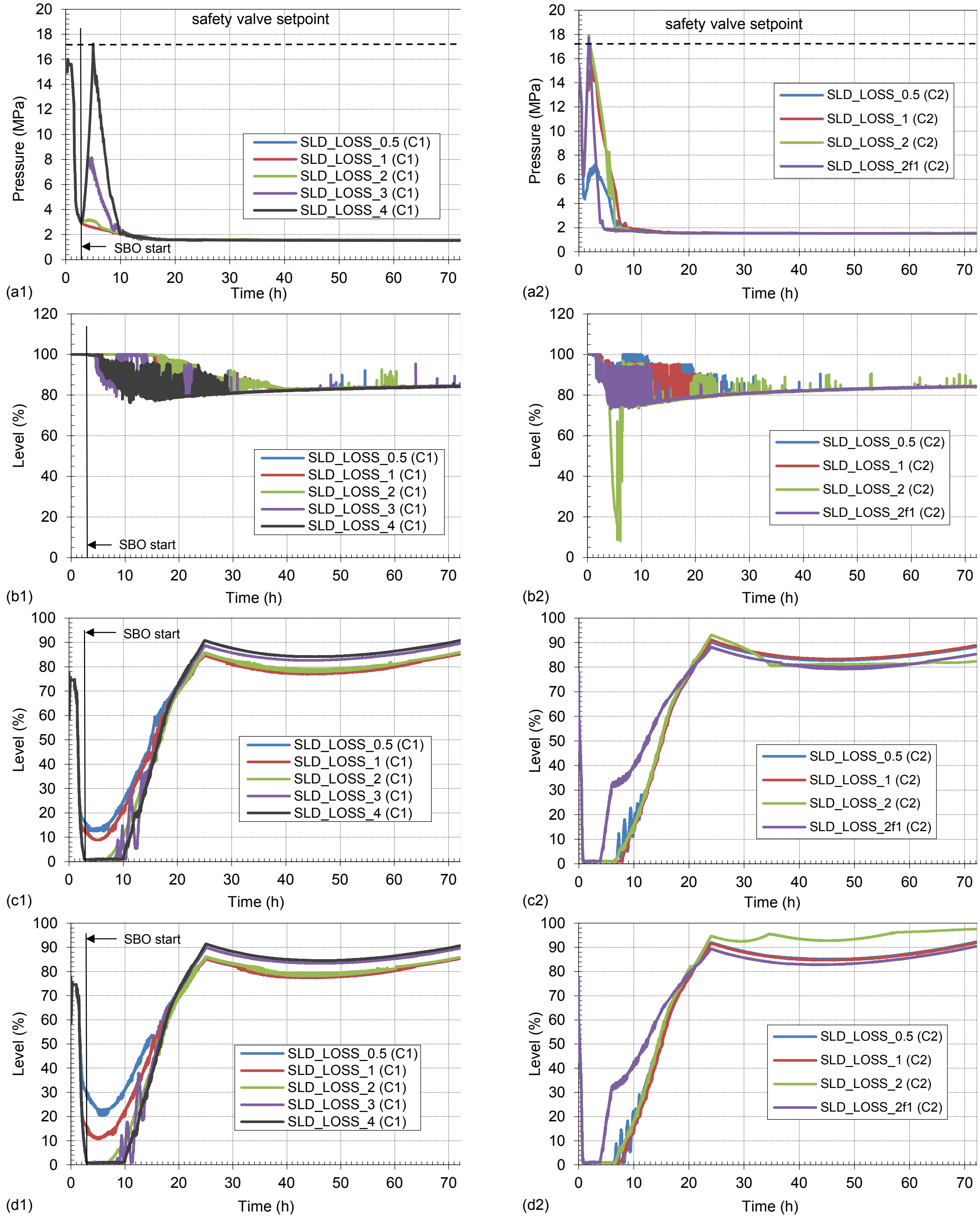


Figure 7 Main Parameters for SLD_LOSS Verification Calculations: (a) RCS Pressure, (b) Average Rod Cladding Temperature, (c) SG No. 1 WR Level, (d) SG No. 2 WR Level.

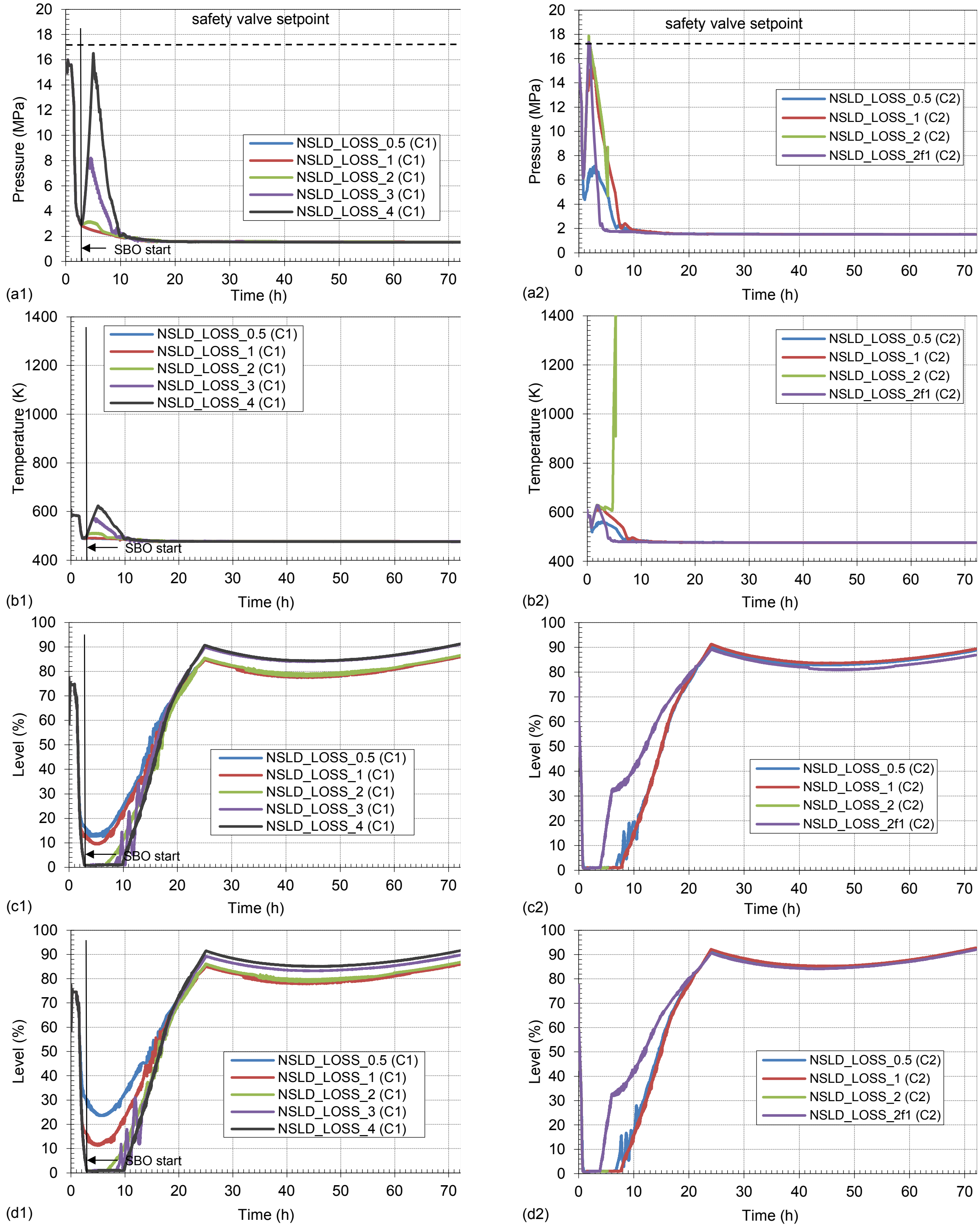


Figure 8 Main Parameters for NSLD_LOSS Verification Calculations: (a) RCS Pressure, (b) Average Rod Cladding Temperature, (c) SG No. 1 WR Level, (d) SG No. 2 WR Level.

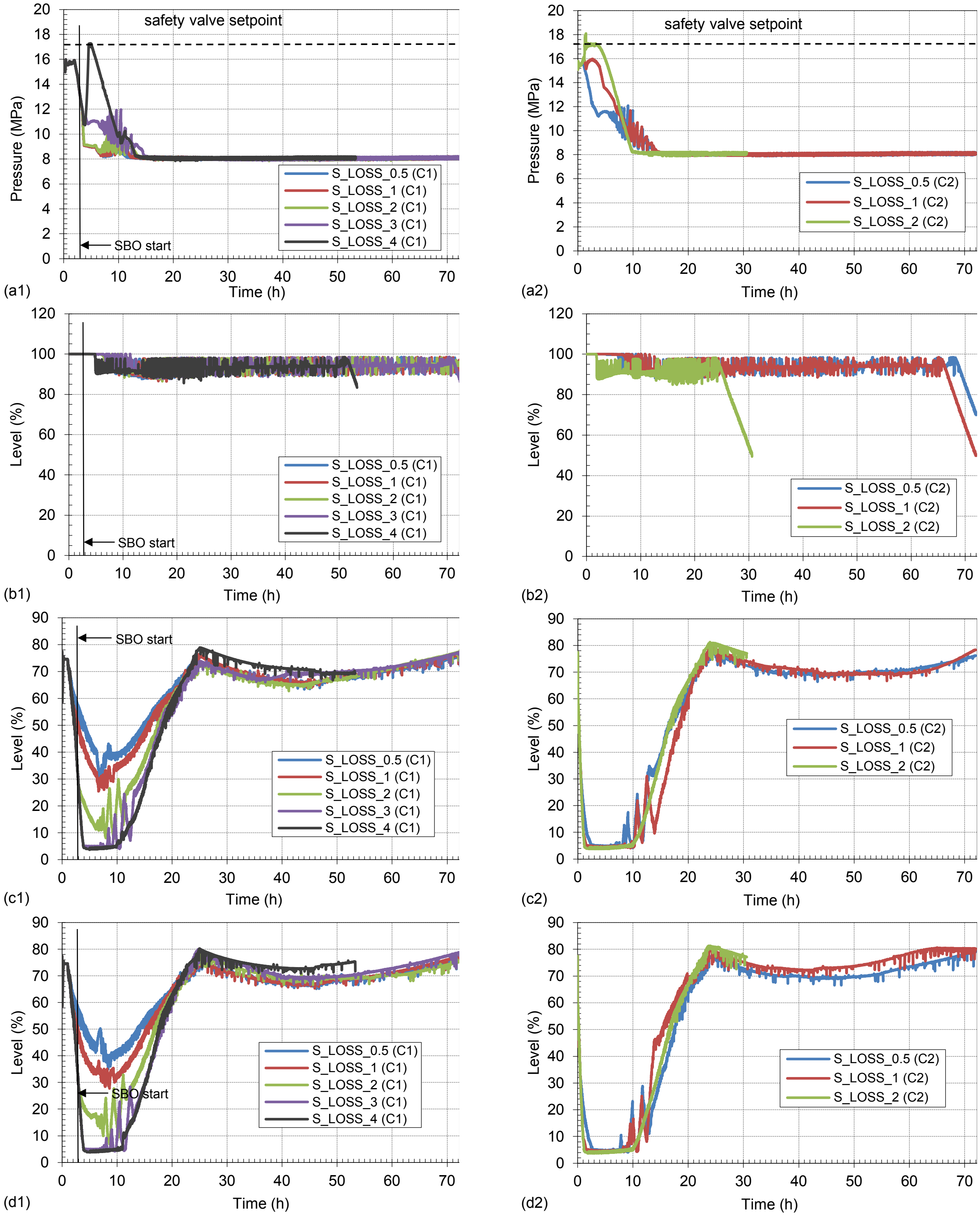


Figure 9 Main Parameters for S_LOSS Verification Calculations: (a) RCS Pressure, (b) Core Collapsed Liquid Level, (c) SG No.1 WR Level, (d) SG No.2 WR Level.

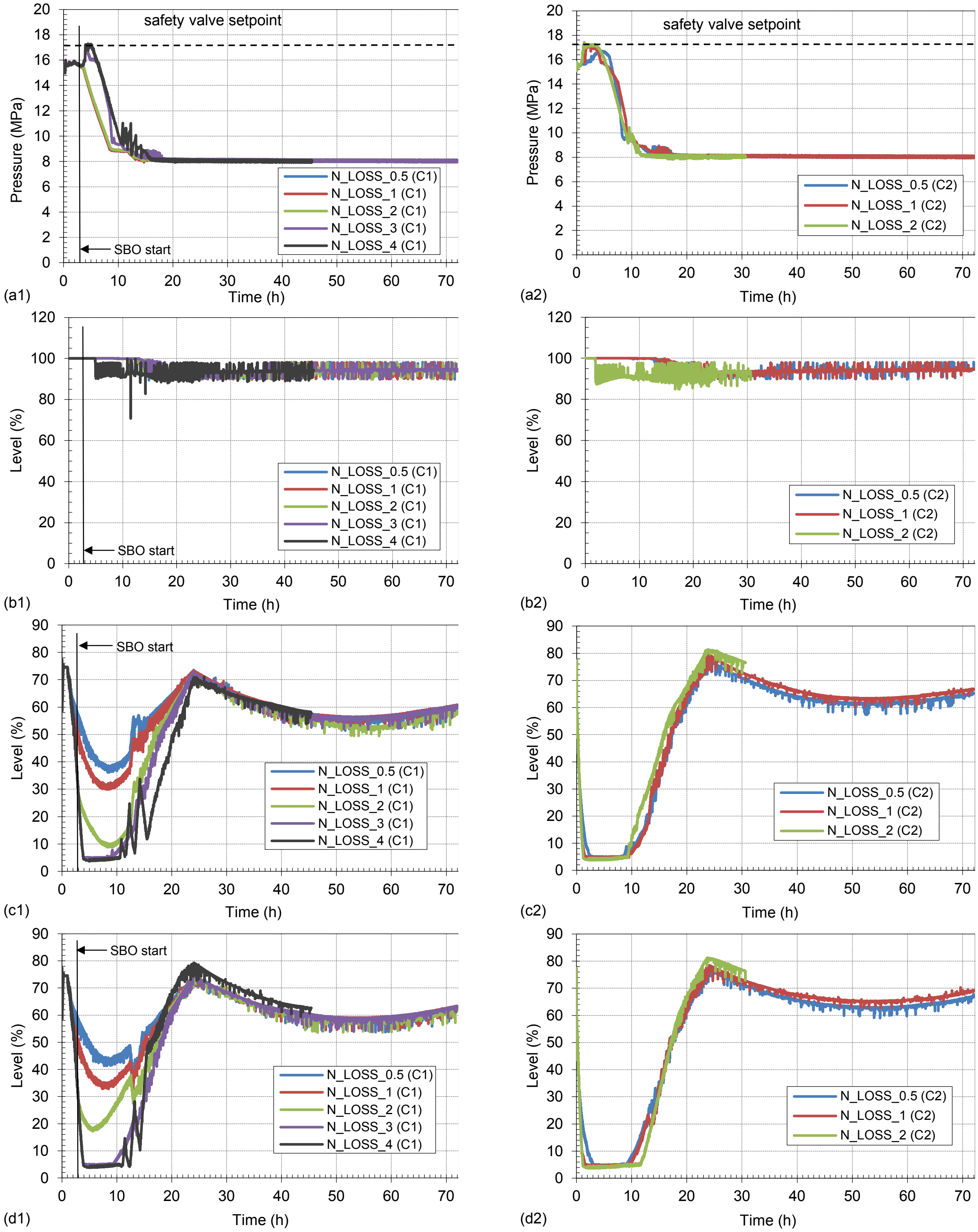


Figure 10 Main Parameters for N_LOSS Verification Calculations: (a) RCS Pressure, (b) Core Collapsed Liquid Level, (c) SG No. 1 WR Level, (d) SG No. 2 WR Level.

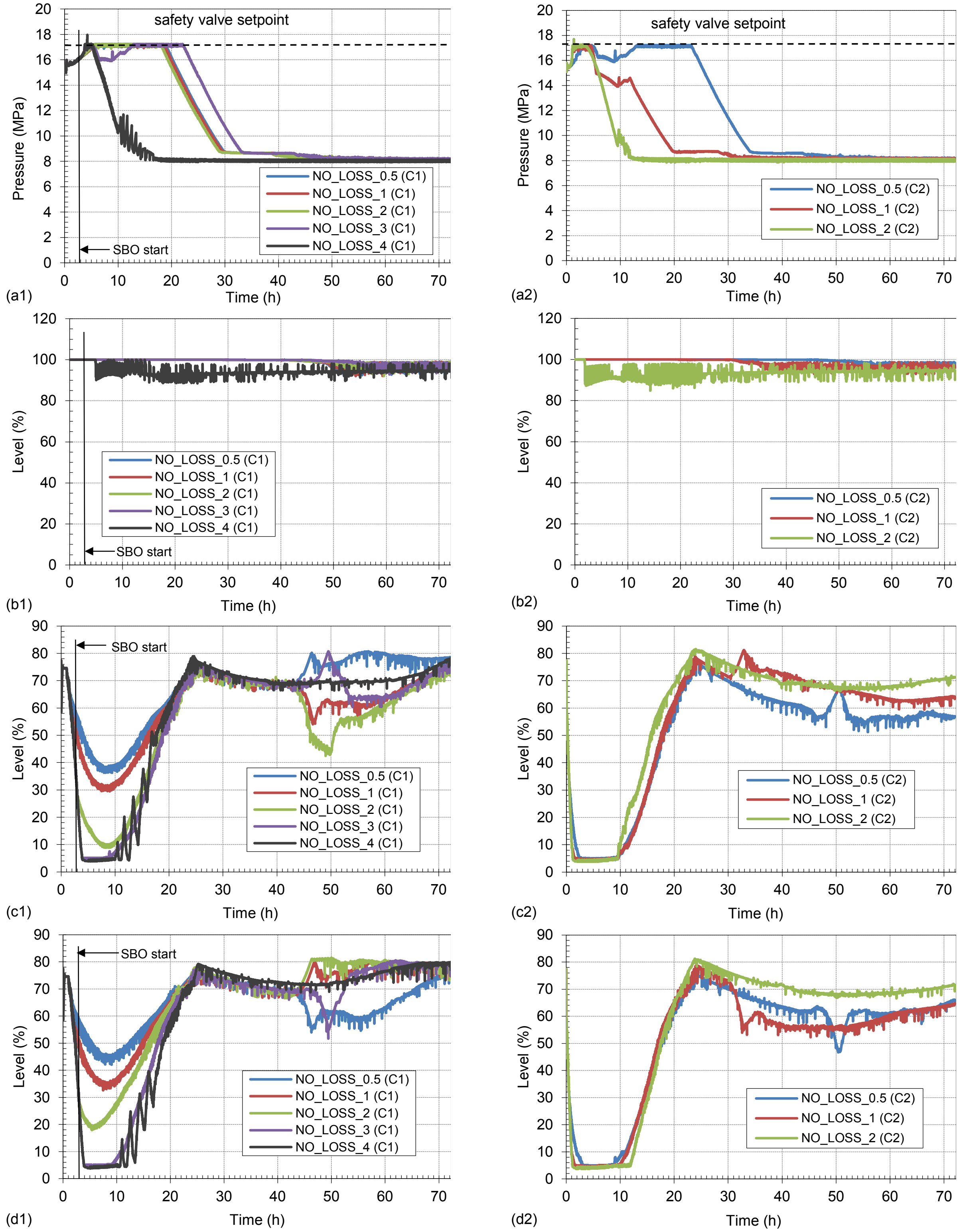


Figure 11 Main Parameters for NO_LOSS Verification Calculations: (a) RCS Pressure, (b) Core Collapsed Liquid Level, (c) SG No. 1 WR Level, (d) SG No. 2 WR Level.

5. CONCLUSIONS

Application of the RELAP5/MOD3.3 Patch 4 for study of the utilization of pump for mitigation of the extended blackout condition is investigated. The methodology for the assessment of the necessary flowrates to be injected into steam generators for mitigation of extended blackout event is proposed. The assessed flows minimize the required number of flow changes and operators errors during required manipulations. The verification calculations are performed with pump flowrate equal to assessed flowrate to assure effective core cooling without overfilling the steam generators.

Six different scenario types of reactor coolant loss are developed and analyzed. Two cases are defined considering the operation of the emergency diesel generators. Different delays of the pump injection start following the station blackout are assumed and analyzed.

Obtained results demonstrated the need for utilization of deterministic system computer code for assessment of the pump flows. 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 first 72 h of extended SBO if pump start to inject on secondary side within 2 h of the SBO start. Operation of the emergency diesel generator (EDG) for one hour extends the available time for the start of pump on 4 h. Failure to isolate letdown results in core damage before 24 h in all analyzed scenarios. Obtained results show that limitation of RCS loss prolongs the time before heat up.

One of the main conclusions in the study is that availability of equipment is prerequisite but not guarantee of successful mitigation, if it is not done in due time. The supporting analyses are necessary in order to verify the effectiveness of the proposed strategy.

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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

A set of measures have been proposed and currently implemented in response of the accident at the Fukushima Daiichi nuclear power plant. Those measures include diverse and flexible mitigation strategies that increase the defense-in-depth for beyond-design-basis scenarios. Mitigation strategies are based on the utilization of the portable equipment to provide power and water to the nuclear power plants in order to maintain or restore key safety functions. The verification of the proposed measures with the plant specific safety analyses is endorsed in the mitigation strategies. The purpose of the study was to investigate the application of the deterministic safety analysis for mitigation strategy of the extended station blackout (SBO). A methodology for the assessment of flowrates for steam generator makeup using portable pump is proposed. The aim is to fill steam generator without available information on level in such a way that makeup is sufficient and that at the same time the steam generators are not overfilled. The RELAP5/MOD3.3 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. The calculated results show effectiveness of the proposed extended SBO mitigation strategy. The applicability of the developed method on operational power plant is validated.

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