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International Agreement Report

RELAP5 Extended Station Blackout Analyses

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ABSTRACT

Following the accident at the nuclear power plant Fukushima in Japan the stress test were performed in European countries. Considering the stress tests specifications it was necessary to evaluate the consequences of loss of safety functions from any initiating event (earthquake or flooding) for loss of electrical power, including Station Blackout (SBO), loss of the ultimate heat sink or combination of both. In this report long term station blackout sequences for Krško two-loop pressurized water reactor with loss of normal or all secondary side heat sinks were performed. For calculations the latest RELAP5/MOD3.3 Patch 04 computer code was used. The verified standard RELAP5/MOD3.3 input model delivered by Krško nuclear power plant was used.

SBO scenario involves a loss of offsite power, failure of the redundant emergency diesel generators, failure of alternate current power restoration and the eventual degradation of the reactor coolant pump seals resulting in a long term loss of coolant. In the study different reactor coolant pump seal leaks were studied due to SBO. Besides, scenarios were performed for different primary side depressurizations performed by operator through the secondary side power operated relief valves, providing that turbine driven auxiliary feedwater pump is available. Finally, the effect of having some injection into the reactor coolant system was also evaluated. It can be concluded that calculated results obtained by RELAP5 give good indication about time available before core degradation started. The results suggest that RELAP5 can be used for extended SBO studies until core damage started. It is especially useful in studying maintaining core cooling function and time available before core uncovers as part of severe accident management. The benefit of using RELAP5 is in the fact that best estimate system codes are more accurate than severe accident codes in phases before core degradation started.

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SBO scenario involves a loss of offsite power, failure of the redundant emergency diesel generators, failure of alternate current (AC) power restoration and the eventual degradation of the reactor coolant pump seals resulting in a long term loss of coolant. It is assumed that AC power exists only on the AC buses powered by inverters connected to the station batteries. Loss of all AC power results in unavailability of all normal electrical equipment and most of the safety electrical equipment. The only possible corrective actions are reactor trip and residual heat removal using steam generator safety and relief valves and turbine (steam) driven auxiliary feedwater pump.

In the study six different seal leaks per reactor coolant pump were studied due to SBO, ranging from 9.4 cm^2 to 148.8 cm^2. These breaks represent leaks from 1.32 l/s (21 gpm) to 18.93 l/s (300 gpm) at nominal conditions. The same sizes of breaks on both reactor coolant pumps were assumed. Besides, scenarios were performed for four different primary side depressurizations performed by operator through the secondary side power operated relief valves, providing that turbine driven auxiliary feedwater is available. Finally, the effect of having some injection into the reactor coolant system was also evaluated (e.g. if positive displacement charging pump would be powered from mobile diesel generator). It can be concluded that calculated results obtained by RELAP5 give good indication about time available before core degradation started.

The results suggest that best estimate system codes like RELAP5 can be used for extended SBO studies until core damage started. It is especially useful in studying maintaining core cooling function and time available before core uncovers as part of severe accident management. The benefit of using RELAP5 is in the fact that best estimate system codes are more accurate than severe accident codes in phases before core degradation started.

ACKNOWLEDGEMENTS

The authors acknowledge the financial support from Krško Nuclear Power Plant and Slovenian Nuclear Safety Administration within CAMP program (project no. POG-3473) and from the state budget by the Slovenian Research Agency (program no. P2-0026).

ABBREVIATIONS

ACC	accumulator
AFW	auxiliary feedwater
CPU	central processing unit
CVCS	chemical and volume control system
ECCS	emergency core cooling system
HPSI	high-pressure safety injection
LD	letdown
LOCA	loss of coolant accident
LPSI	low-pressure safety injection
MAAP	Modular Accident Analysis Program
MD	motor driven
MFW	main feedwater
MSIV	main steam isolation valve
NEK	Krško nuclear power plant (in Slovene: Nuklearna elektrarna Krško)
NPP	nuclear power plant
PORV	power operated relief valve
PRZ	pressurizer
PWR	pressurized water reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RPV	reactor protection vessel
RELAP	Reactor Excursion and Leak Analysis Program
SG	steam generator
SL	surge line
TD	turbine driven
USAR	updated safety analysis report

1. INTRODUCTION

Following the accident at the nuclear power plant Fukushima in Japan the “stress tests” were performed in European countries (Ref. 1). Considering the stress tests specifications it was necessary to evaluate the consequences of loss of safety functions from any initiating event (earthquake or flooding) for loss of electrical power, including Station Blackout (SBO), loss of the ultimate heat sink or combination of both. SBO scenario involves a loss of offsite power, failure of the redundant emergency diesel generators, failure of alternate current (AC) power restoration and the eventual degradation of the reactor coolant pump seals resulting in a long term loss of coolant. It is assumed that AC power exists only on the AC buses powered by inverters connected to the station batteries. Loss of all AC power results in unavailability of all normal electrical equipment and most of the safety electrical equipment. The only possible corrective actions are reactor trip and residual heat removal using steam generator safety and relief valves and turbine (steam) driven auxiliary feedwater pump.

“Stress tests” had to be performed also for Krško nuclear power plant (NPP), which is a two-loop pressurized water reactor. Normally such long scenarios are simulated with severe accident codes. For example, in the case of Krško NPP the Modular Accident Analysis Program (MAAP) Version 4.0.5 was used to analyze long term Station Blackout (SBO) accident sequences in 2011 (Ref. 2). The study presented analyses performed by MAAP in which the operator action was used to rapidly depressurize the secondary side to 2.1 MPa (20 kp/cm^2 gauge) and then maintain this pressure. Secondary side depressurization leads to primary side cooldown and depressurization. The calculations performed by MAAP were performed for different break sizes of reactor coolant pump (RCP) seals. Namely, following the loss of all AC power the RCP seals would lose their cooling support systems (the RCP seal injection flow and component cooling water to the RCP thermal barrier heat exchanger would be unavailable) and would undergo a severe thermal transient. The MAAP long term SBO accident sequences were analyzed with the focus on the containment response after the core damage.

On the opposite, the focus in this study is to evaluate the plant response before the core degradation. Please note that base case MAAP scenario demonstrated that with turbine driven auxiliary feedwater (TD AFW) pump available and by operator depressurization of primary system to 2.1 MPa the core damage can be prevented for the first seven days. Same scenario could be performed by RELAP5, which is best estimate system code for design basis transient and accidents, while the MAAP code is intended for simulations of severe accidents. It is expected that the calculations by RELAP5 could therefore be performed with smaller uncertainties than those with MAAP code. Performing RELAP5 calculations direct code to code comparison could be done.

For example, in the study (Ref. 3) describing the comparison of the SBO results obtained by MAAP4 and CENTS computer codes indicates that: (1) the overall trends of key parameters are similar, and (2) there are differences in the timing of significant occurrences (e.g., SG dryout, core uncovery). Nevertheless it is stated that although the timings and durations of key occurrences and actuations vary, MAAP4 predictions of core uncovery tend to be conservatively biased. They concluded that the simplified single phase natural circulation model utilized by MAAP4 drives differences in the thermal hydraulic response of the SGs as well as the RCS. They also pointed out that older generations of MAAP4, such as MAAP 4.0.5, have been known to skew the RCS pressure responses for feed and bleed and SBO transients. MAAP versions 4.0.6 and beyond include a number of enhancements that yield more consistent results for the pressure traces of the SBO and the feed and bleed transients. It is recommended that the user be cautious in the selection of the code version that is employed for SBO event analysis. Further it was stated (Ref. Ref. 3) that a new version of the MAAP code, MAAP5, incorporates a momentum equation to model the primary side natural circulation flowrate and a more detailed SG model to more accurately predict secondary side behavior. These code modifications are expected to minimize the impact of the uncertainties seen with the MAAP4 version of the code. The study for Krško NPP (Ref. 2) was performed by MAAP 4.0.5.

In RELAP5 study six different reactor coolant pump seal leaks were studied due to SBO, ranging from 9.4 cm^2 to 148.8 cm^2. These breaks represent leak from 1.32 l/s (21 gpm) to 18.93 l/s (300 gpm) at nominal conditions. Besides, scenarios were performed for four different primary side depressurizations performed by operator through the secondary side power operated relief valves, providing that turbine driven auxiliary feedwater is available. One of the aims of the stress tests was also to indicate time before water level reaches the top of the core, and time before fuel degradation (fast cladding oxidation with hydrogen production). Therefore the purpose of our study was also to estimate the effect of depressurization on the time before water level reaches the top of the core, and time before time before fuel degradation. For example, in Ref. 4 it is stated that at temperatures above 1200 °C the rapid oxidation of Zircaloy and of stainless steel by steam is present.

Finally, it was investigated the effect of having some small injection into the reactor coolant system by having positive displacement charging pump powered from mobile diesel generator.

The organization of the report is as follows. In the Section 2 the plant analyzed is briefly described, while RELAP5 input model is described in Section 3. The description of scenarios analyzed is given in Section 4. The analysis results for the selected scenarios are described in Section 5. Finally, the conclusions are given in Section 6.

2. PLANT DESCRIPTION

Krško NPP is a Westinghouse two-loop pressurized-water reactor (PWR) plant with a large dry containment. The plant has been in commercial operation since 1983. After modernization in 2000, the plant's fuel cycle was gradually prolonged from 12 (cycle 17) to 18 months (cycle 21). The power rating of the Krško NPP nuclear steam supply system is 2,000 megawatt thermal (MWt) (1,882 MWt before the plant modernization and power uprate), comprising 1,994 MWt (1,876 MWT before the plant modernization and power uprate) of core power output plus 6 MWt of reactor coolant pumps (RCPs) heat input. The reactor coolant system (RCS) is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing an RCP and a steam generator (SG). An electrically heated pressurizer is connected to one of the loops.

The reactor core is composed of 121 fuel assemblies. The RCPs, one per coolant loop, are Westinghouse vertical, single-stage, centrifugal pumps of the shaft-seal type. The SGs, one per loop, are vertical U-tube, Siemens-Framatome type SG 72 W/D4-2 units, installed during the plant modernization in 2000.

For more detailed description of the plant the reader is referred to Ref. 5.

3. RELAP5 INPUT MODEL DESCRIPTION

To perform the analysis, Krško NPP has provided the base RELAP5 input model, so called "Master input deck", which have been used for several analyses, including reference calculations for Krško full scope simulator verification (Refs. 6, 7, 8). The analysis was performed for uprated conditions (2000 MWt) with new steam generators (SGs) and Cycle 23 settings, corresponding to the expected plant state after outage and refueling in October 2007. The base model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points and is shown in Figure 1. The analyses were performed with direct injection of TD AFW into steam generators (AFW piping was removed from the RELAP5 model) as it was shown that the influence on results is negligible, while calculation performs about ten times faster. The model without TD AFW piping consisted of 432 control volumes, 459 junctions while the number of heat structures remained unchanged as can be seen from Figure 2.

Modeling of the primary side without the reactor vessel and both loops includes 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 RCP seal flow. The reactor vessel (RPV) consists of the lower downcomer, lower head, lower plenum, core inlet, reactor core, core baffle bypass, core outlet, upper plenum, upper head, upper downcomer, and guide tubes. The primary loop is represented by the hot leg, primary side of the steam generator (SG), intermediate leg with cold leg loop seal, and cold leg, separately for loop 1 and loop 2. Loops are symmetrical except for the pressurizer surge line and the chemical and volume control system connections layout. The primary side of the SG consists of the inlet and outlet plenum, tubesheet, and the U-tube bundle represented by a single pipe. 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 (riser, separator and separator pool, downcomer, steam dome), main steamline, main steam isolation valves (MSIVs), SG relief and safety valves, and main feedwater (MFW) piping. Auxiliary feedwater (AFW) piping was removed and only TD AFW injects above the SG riser. The main steam no. 1 has same volumes as main steam no. 2, but the geometry data differ depending on pipeline. Turbine valve is modeled by the corresponding logic, while turbine is represented by time dependent volume. MFW and AFW pumps are modeled as time dependent junctions, pumping water from time dependent volumes, representing the condensate storage tank.

In order to accurately represent the Krško NPP behavior, a considerable number of control variables and general tables are part of the model. They represent protection, monitoring and simplified control systems used only during steady state initialization, as well as the following main plant control systems: (a) rod control system, (b) PRZ pressure control system, (c) PRZ level control system, (d) SG level control system, and (e) steam dump. It must be noted that rod control system has been modeled for point kinetics. The reactor protection system was based on trip logic. It includes reactor trip signal, safety injection signal, turbine trip signal, steam line isolation signal, MFW isolation signal, and AFW start signal.

For further details of the above mentioned plant systems and components, plant signals and control systems schemes the reader can refer to Reference 5.

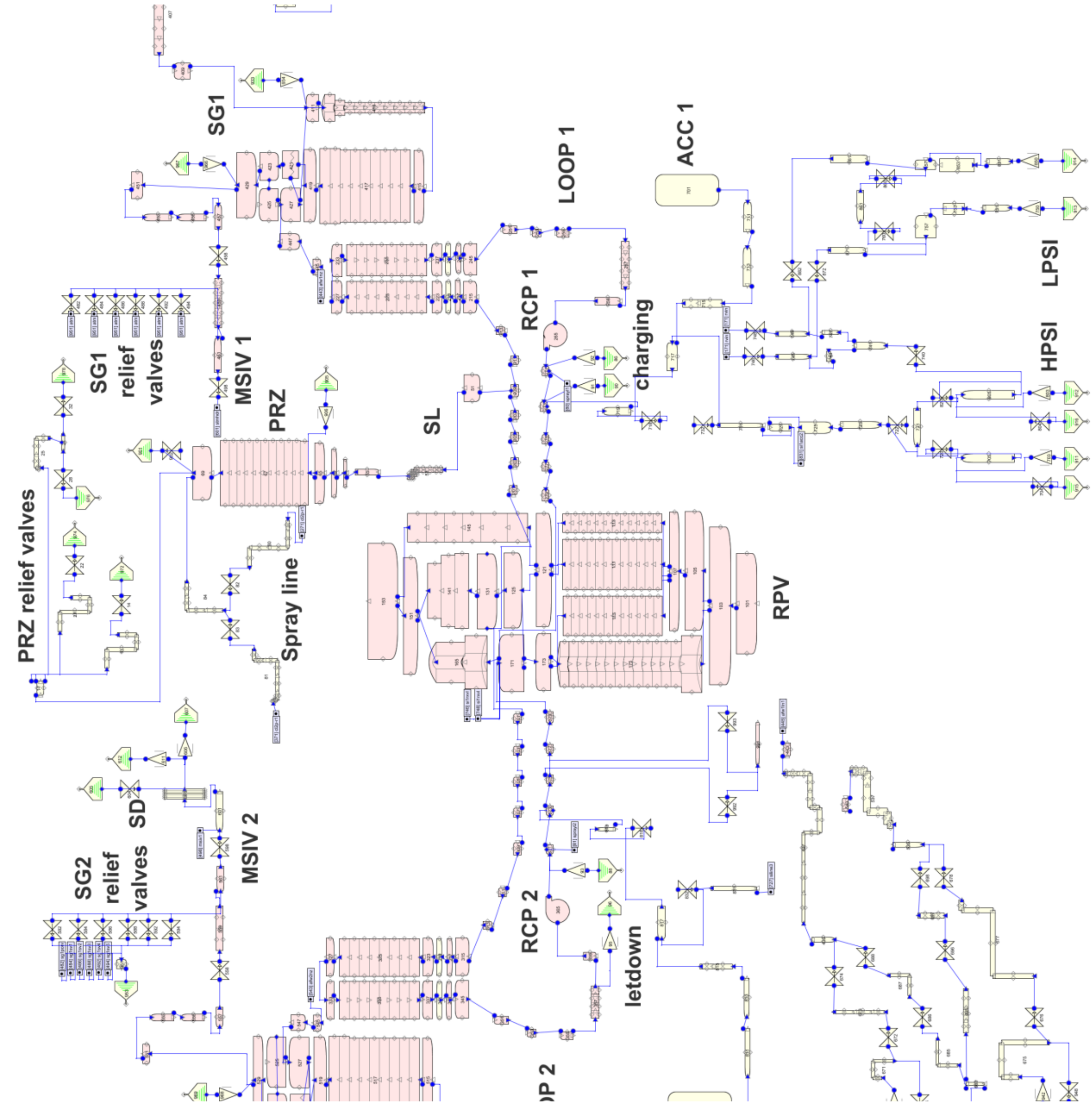

Figure 1 Krško NPP base nodalization scheme – SNAP hydraulics component view

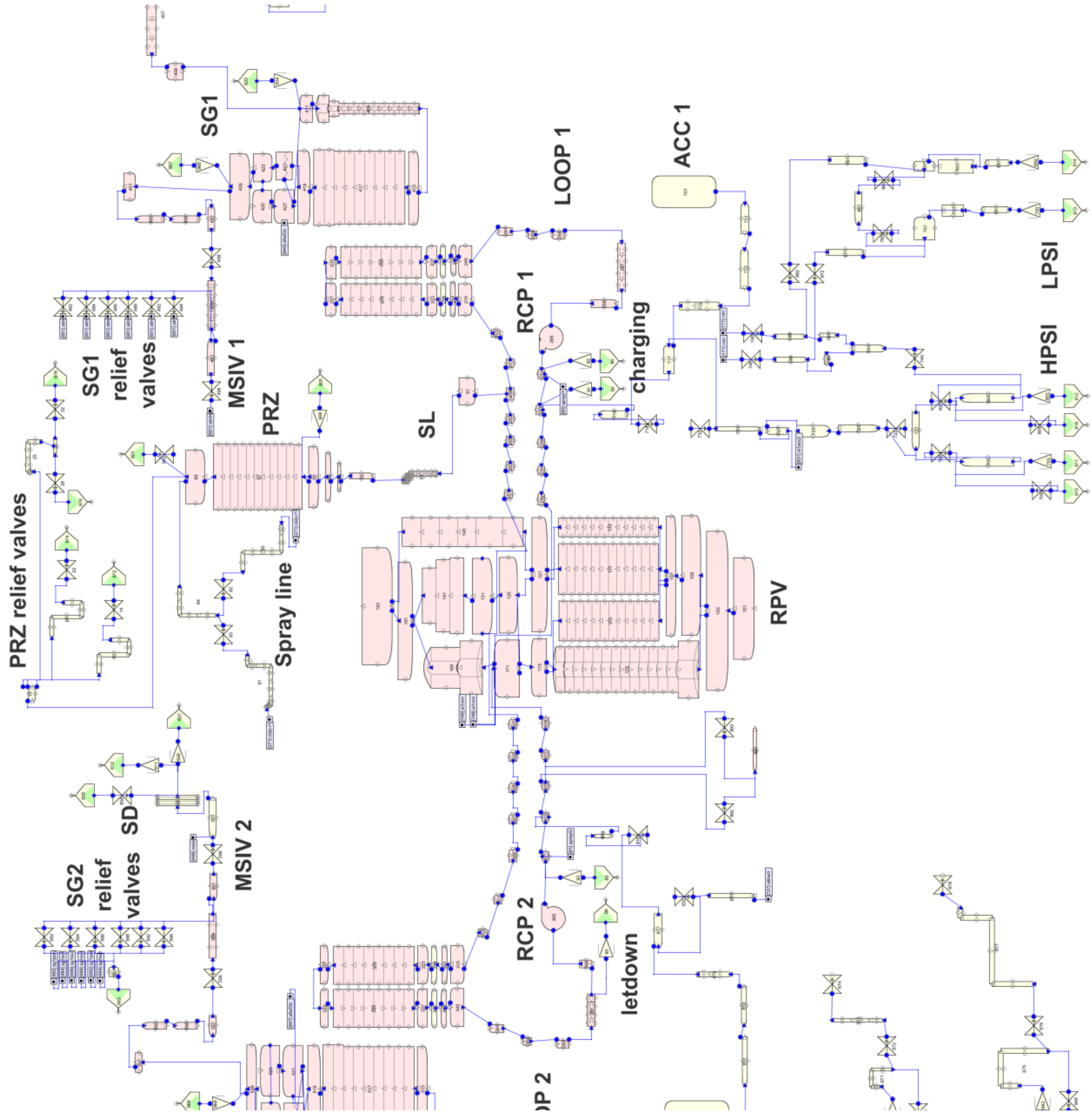

Figure 2 Krško NPP modified nodalization scheme – SNAP hydraulics component view

4. DESCRIPTION OF SCENARIOS ANALYZED

In this study the RELAP5 analyses were performed until significant reactor heatup occurred (up to 1500 K). The set of scenarios analyzed by RELAP5/MOD3.3 Patch 04 is shown in Table 1. Each set of scenarios was analyzed at for different depressurization pressures (fast depressurization to the specified SG pressure and maintaining the specified SG pressure): 2.55 MPa (25 kp/cm^2 gauge), 2.06 MPa (20 kp/cm^2 gauge), 1.57 MPa (15 kp/cm^2 gauge) and 1.33 MPa (12.5 kp/cm^2 gauge).

The scenarios (in majority) and the initial and boundary conditions used were based on Ref. 2. To better understand the transient progression, the assumptions used in the RELAP5 scenarios are listed below.

At time 0 s the following trips were actuated:

- reactor trip,
- turbine trip,
- MFW1 and MFW2 isolation,
- SI signal generation,
- RCP 1 and RCP 2 trip,
- MSIV1 and MSIV2 isolation.

The following safety systems were assumed unavailable at time 0 s:

- HPSI pump 1 and 2 unavailable,
- LPSI pump 1and 2 unavailable,
- AFW MD pump 1 and 2 unavailable.

Other systems unavailable or disabled at time 0 s were:

- pressurizer proportional and backup heaters disabled,
- pressurizer spray disabled,
- CVCS (charging and letdown) flow not available,
- condenser (steam dump) unavailable.

The following assumptions were also used:

- opening of letdown relief valve 8120 to pressurizer relief tank, if RCS pressure greater than 4.23 MPa,
- TD AFW available if SG pressure greater than 0.79 MPa,
- TD AFW control valves available.

Also it was assumed, when TD AFW was available, that condensate storage tanks are refilled, thus providing unlimited source of water.

The following operator actions were modeled:

- control of the TD AFW flow to maintain SG NR level around 60%,
- fast depressurization of the SGs to selected pressure (2.55 MPa, 2.06 MPa, 1.57 MPa or 1.33 MPa) by opening the SG PORVs,
- maintaining SGs pressure at selected pressure (2.55 MPa, 2.06 MPa, 1.57 MPa or 1.33 MPa) by a manual control of SG PORVs.

Besides main assumptions listed above, for each scenario additional assumptions were used as shown in Table 1. Two different groups of scenarios were simulated: S0 scenarios with TD AFW pump assumed available all the time and S1 scenarios with TD AFW pump assumed available first four hours. The calculations were performed up to 604800 s (7 days). Turbine Driven AF pump does not require electric power and can operate if SG pressure is above 0.79 MPa (7 kp/cm^2 gauge) so it can provide AFW injection. Namely, the AFW control valves are air-operated and provided with a 4-hour supply of nitrogen gas to control the TD AFW pump and the power-operated relief valves for releasing steam from SGs. When taking into account the assumption that the AFW regulator valves are operable (nitrogen or alternative compressed air supply is assumed available) and that condensate storage tanks can be refilled, the secondary side heat sink is available towards the whole transient. For the transient analysis duration seven days the SG pressure is above 0.79 MPa.

Different seal leaks per reactor coolant pump were assumed (1.32 l/s, 3.15 l/s, 4.73 l/s, 6.31 l/s, 9.46 l/s and 18.93 l/s) for scenarios with TD AFW pump assumed available all the time (S0 scenarios). For convenience the break flows are specified also in gallons per minute (gpm). Volumetric break flow was first converted to mass flow (density was considered 753.5 kg/m^3). The mass flows were modeled by equivalent break area giving specified mass flow at nominal pressure and temperature conditions (15.51 MPa, 578 K). If RCS pressure was greater than 4.23 MPa, also the letdown (LD) leak was considered (5.68 l/s).

Besides simulating scenarios with TD AFW available all the time also scenarios with TD AFW available (S1 scenarios) first four hours were performed for the case with 1.32 l/s volumetric break flow. Besides base case (S1-21) also cases with primary side depressurization were analyzed using one (S1-21v1) and two PRZ PORVs (S1-21v2). In the last case (S1-21p) injection by PDP charging pump was assumed.

Table 1 Set of scenarios analyzed for each of four depressurization cases

Scenario	Seal Leak l/s (gpm)	LD Leak l/s (gpm)	TD AFW Pump	PRZ PORV	PDP
S0-21	1.32 (21)	90	ON	No	No
S0-50	3.15 (50)	90	ON	No	No
S0-75	4.73 (75)	90	ON	No	No
S0-100	6.31 (100)	90	ON	No	No
S0-150	9.46 (150)	90	ON	No	No
S0-150p	9.46 (150)	90	ON	No	Yes
S0-300	18.93 (300)	90	ON	No	No
S1-21	1.32 (21)	90	OFF at 4hr	No	No
S1-21v1	1.32 (21)	90	OFF at 4hr	Yes (one PORV)	No
S1-21v2	1.32 (21)	90	OFF at 4hr	Yes (two PORV)	No
S1-21p	1.32 (21)	90	OFF at 4hr	No	Yes

5. RESULTS

In total 44 calculations were performed for eleven scenarios at four different SG depressurization pressure setpoints. The calculations were performed up to 604800 s (7 days) or heatup of the core (calculations were aborted due to high clad temperature or due to reactor kinetics error), whatever occurred first. First set of plots shows the dependence of each calculated scenario on the depressurization of SGs to selected values and maintaining that pressure. Second set of plots shows the RCP seal leak dependence of S0 scenarios with TD AFW pump available all the time and the third set shows S1 scenarios with TD AFW available four hours after SBO event start, in which the equipment used was varied.

5.1 Dependence on depressurization of selected SBO scenarios

For each scenario, the following six variables are shown: RCS pressure, core exit temperature, core collapsed liquid level, average fuel cladding temperature, total mass discharged from RCS (through letdown isolation valve when RCS pressure greater than 4.23 MPa, both RCP seal leaks and PRZ PORVs – only scenarios S1-21p1 and S1-21p2) and mass injected into RCS (accumulators (opening below 4.96 MPa) and PDP pump in cases S0150p and S1-21p).

5.1.1 Scenario S0-21

The results for scenario S0-21 are shown in Figures 3 through 8. It can be seen that primary pressure (Figure 3) follows the secondary side depressurization. At the end of transient analysis the pressure start to drop below the depressurization pressure setpoint, what means that cooling through the break and by steam assumed to be consumed by TD AFW pump is sufficient. Core exit temperature shown in Figure 4 has similar trend as primary pressure. Figure 5 shows the core collapsed liquid level. It can be seen, that initially the level dropped due to stopped injection from accumulators and later remains around 80% (please note that this is collapsed liquid level, denoting voids in the core without real core uncovery). Figure 6 shows the average fuel cladding temperature, which slowly decrease during transient. The mass discharged from the primary system is shown in Figure 7. Initially more mass is discharged in the cases with larger depressurization (the larger injection from the accumulators the larger break flow). From Figure 8 it can be seen that the accumulators were not completely emptied (further depressurization would be needed to enable discharge of all 72 tons of water from both accumulators). The results showed that one TD AFW pump is sufficient to cool the primary system. Also it can be concluded, that depressurization is beneficial, especially below 4.23 MPa, by eliminating letdown break flow and enabling accumulator injection. Nevertheless, after one week the plant is in similar state for all selected depressurizations.

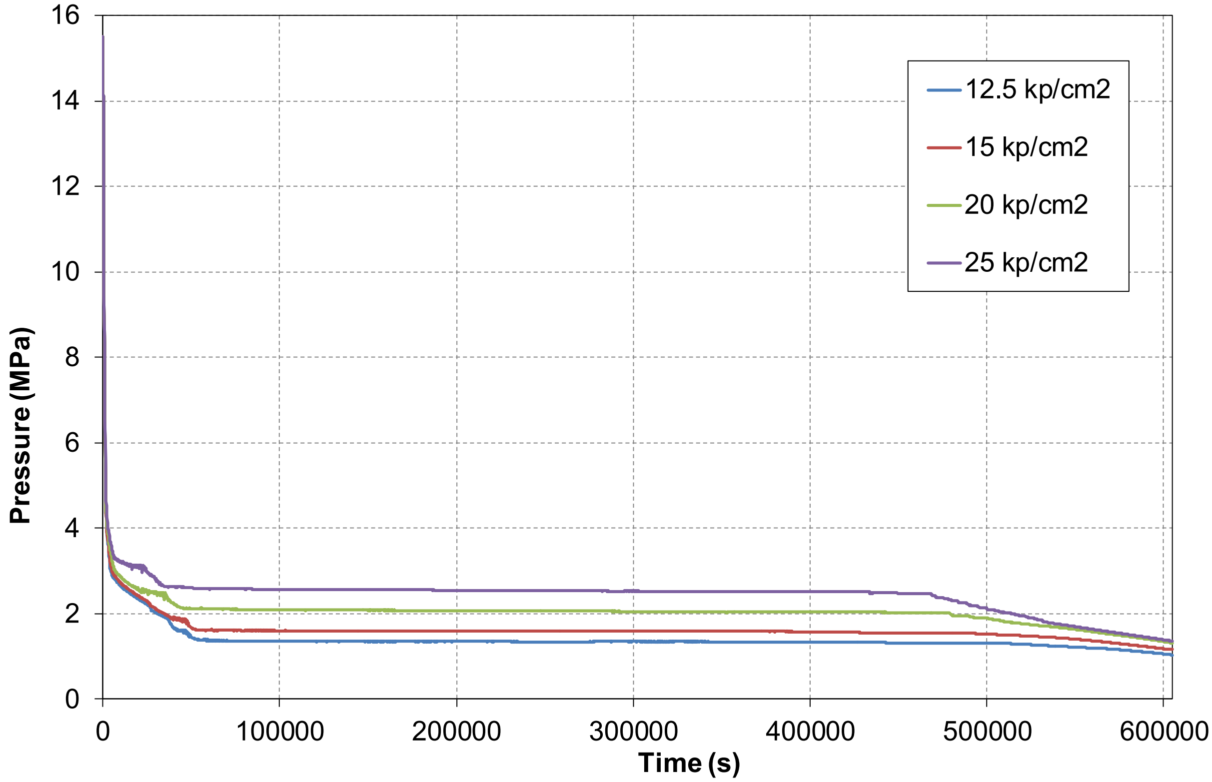

Figure 3 RCS pressure – scenario S0-21

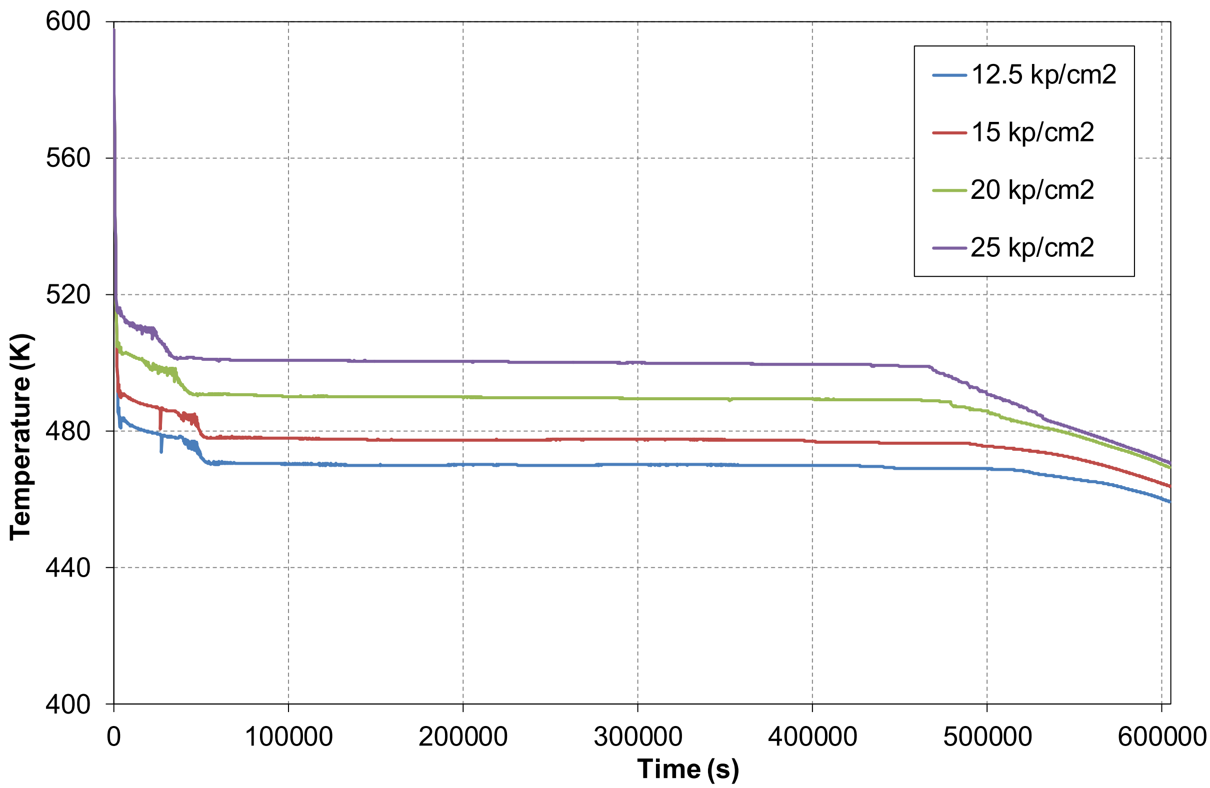

Figure 4 Core exit temperature – scenario S0-21

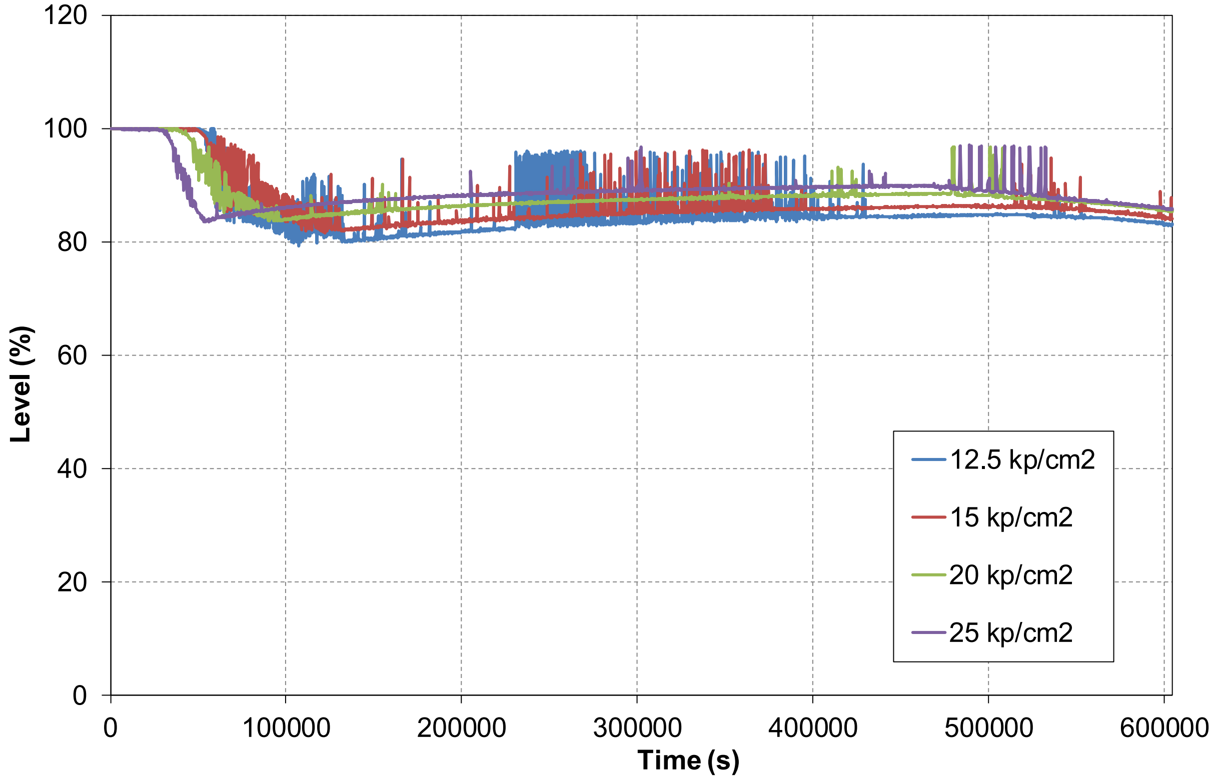

Figure 5 Core collapsed liquid level – scenario S0-21

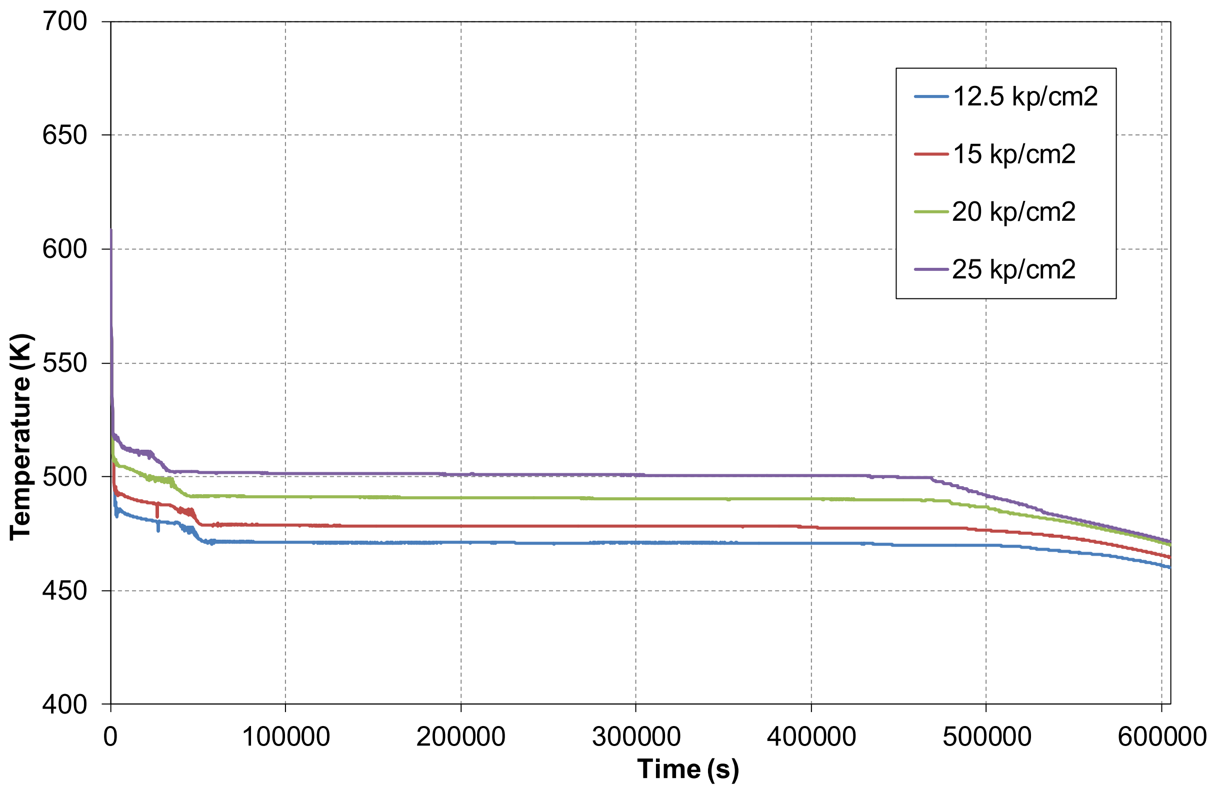

Figure 6 Average fuel cladding temperature – scenario S0-21

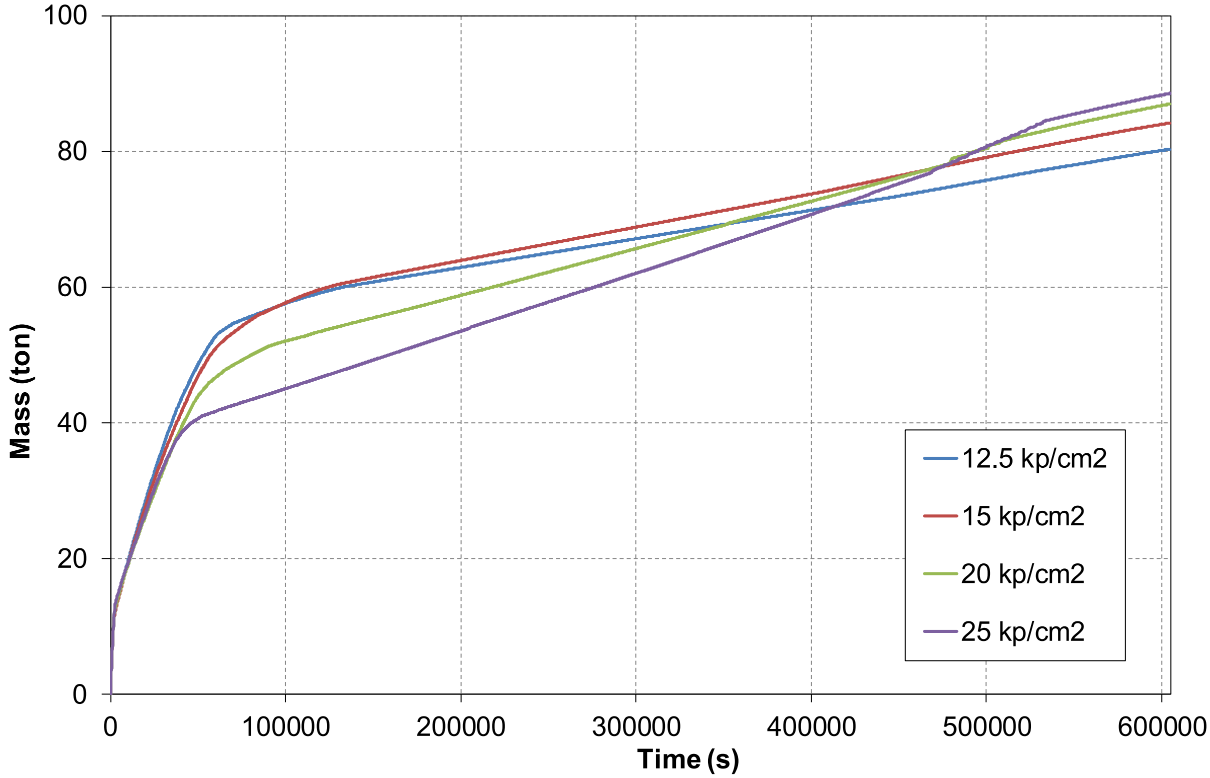

Figure 7 Total mass discharged from RCS – scenario S0-21

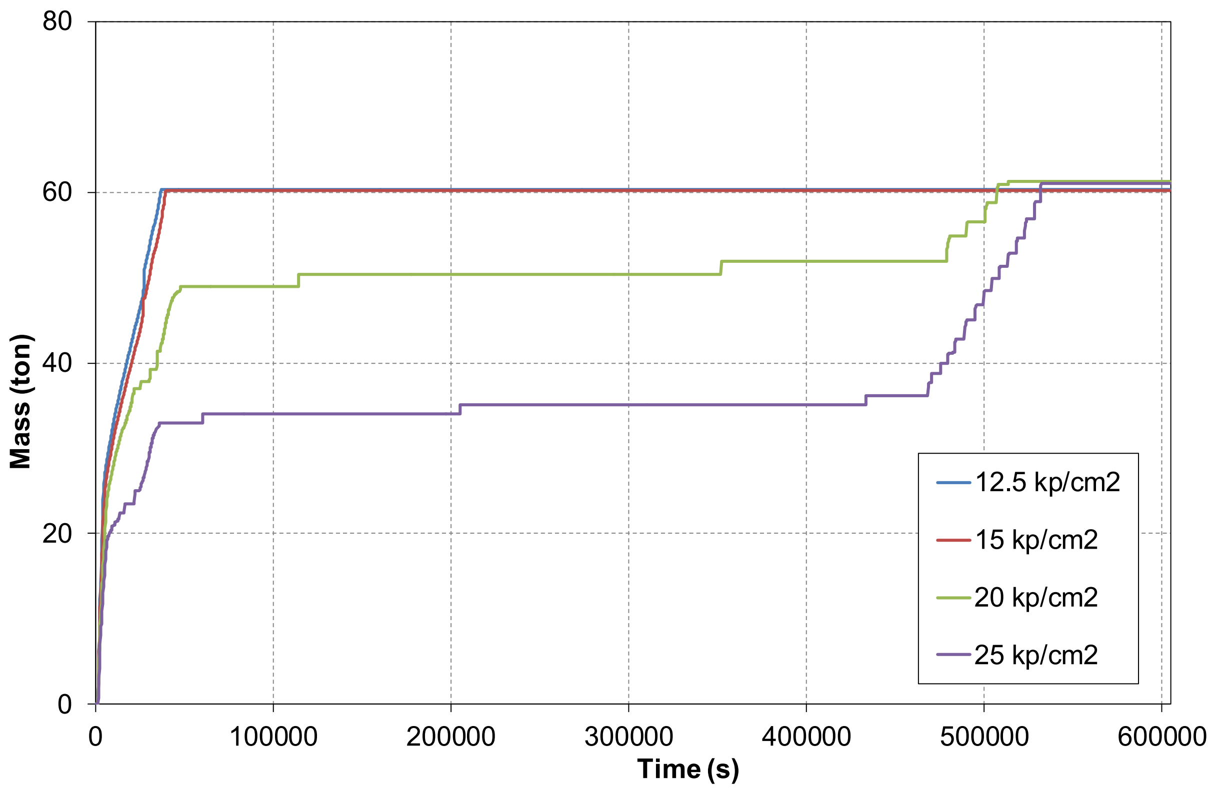

Figure 8 Mass injected into RCS – scenario S0-21

5.1.2 Scenario S0-50

The results for scenario S0-50 are shown in Figures 9 through 14. It can be seen that primary pressure (Figure 9) follows the secondary side depressurization. As break in scenario S0-50 is larger than in scenario S0-21, the importance of RCS depressurization increases. Core exit temperature (Figure 10) shows that only the case with largest depressurization does not lead to core heatup in the first seven days. Figure 11 shows the core uncovery for cases with lower depressurization. It can be seen, that initially the level dropped due to stopped injection from accumulators and then remains around 80% until core uncovery. Figure 12 shows the average fuel cladding temperature. At larger depressurization initially more mass is discharged from the primary system as shown in Figure 13 because also more mass is injected from the accumulators (Figure 14). Later larger depressurization means smaller discharge of RCS inventory.

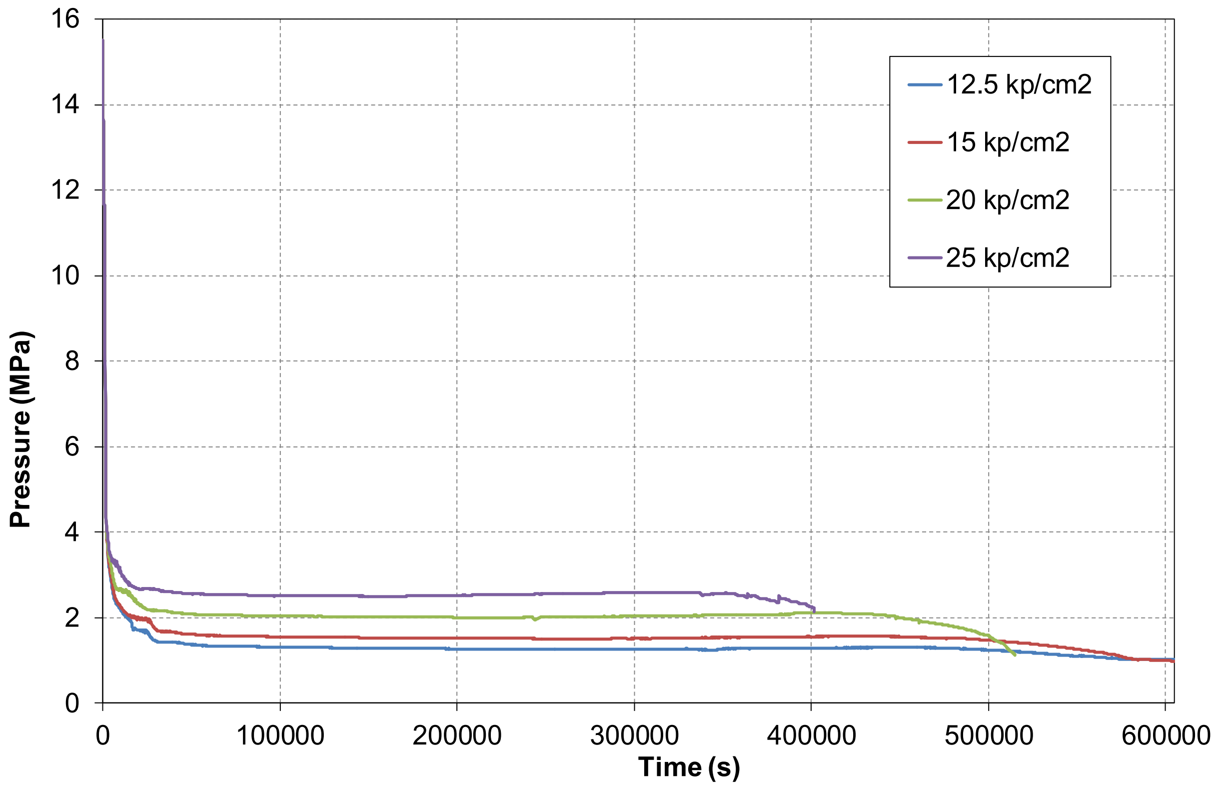

Figure 9 RCS pressure – scenario S0-50

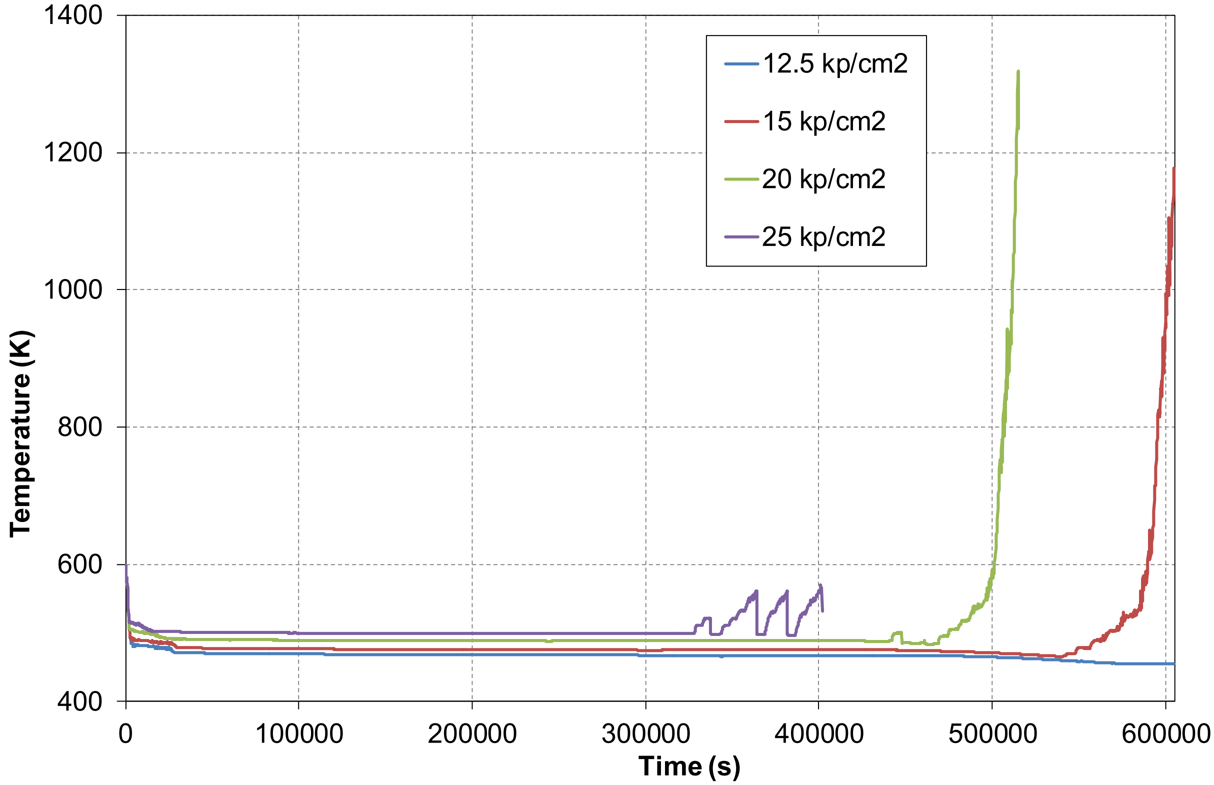

Figure 10 Core exit temperature – scenario S0-50

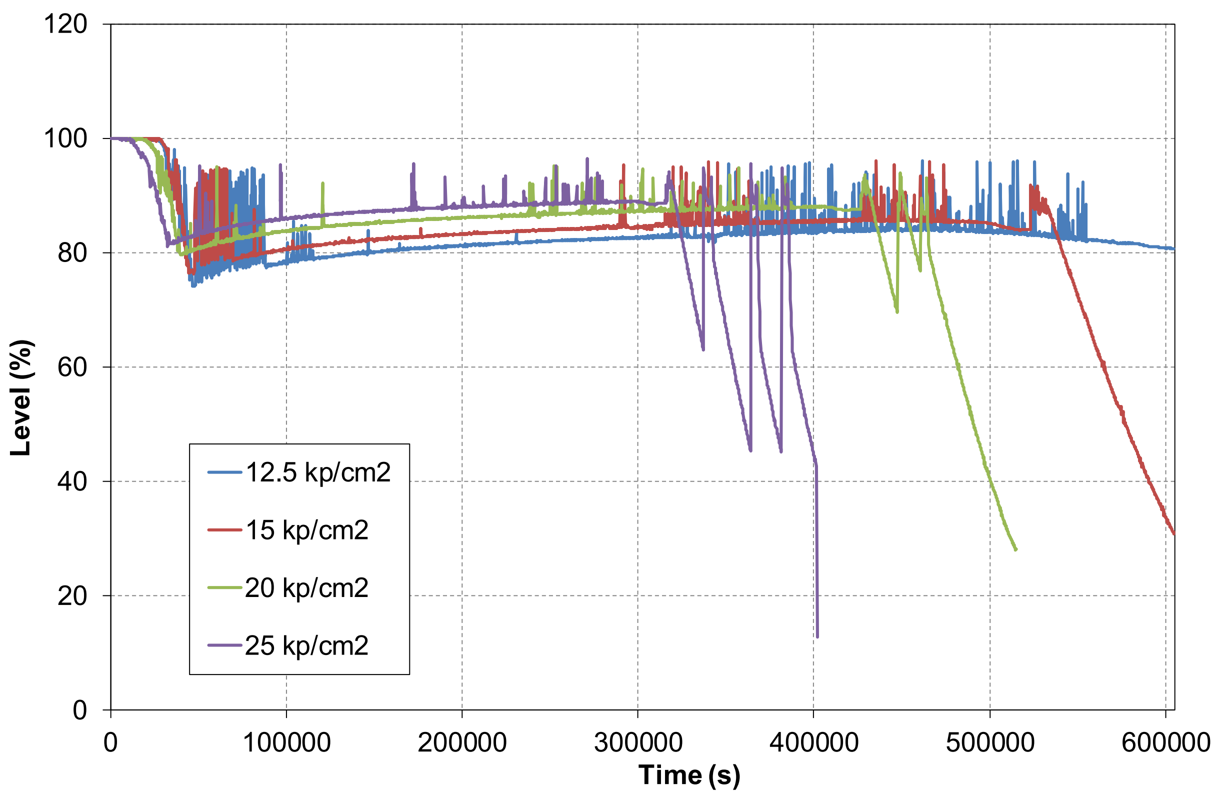

Figure 11 Core collapsed liquid level – scenario S0-50

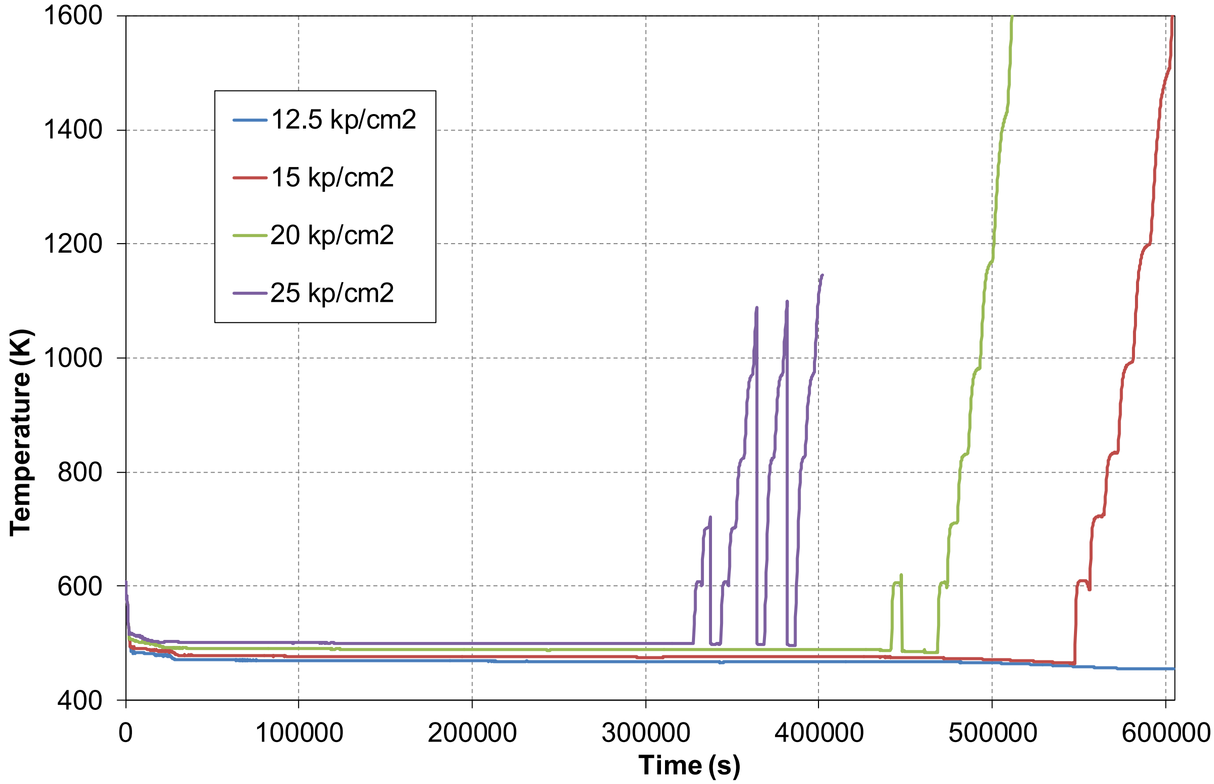

Figure 12 Average fuel cladding temperature – scenario S0-50

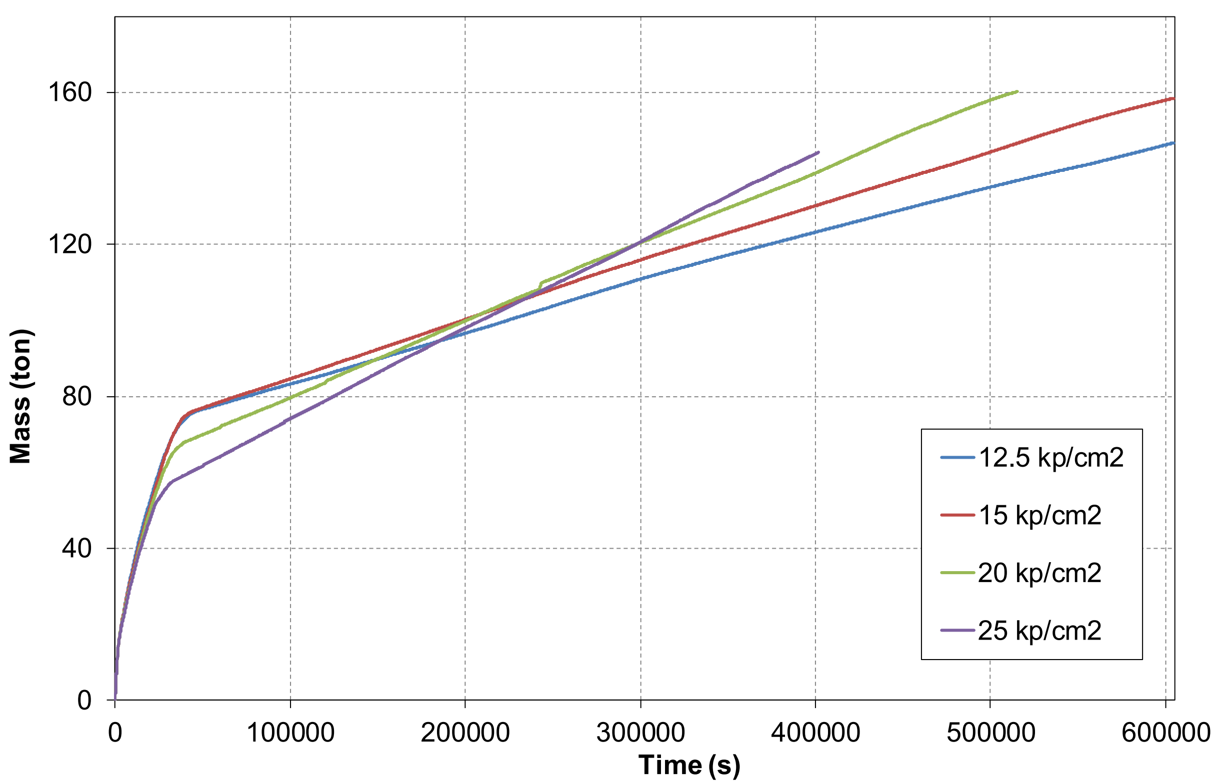

Figure 13 Total mass discharged from RCS – scenario S0-50

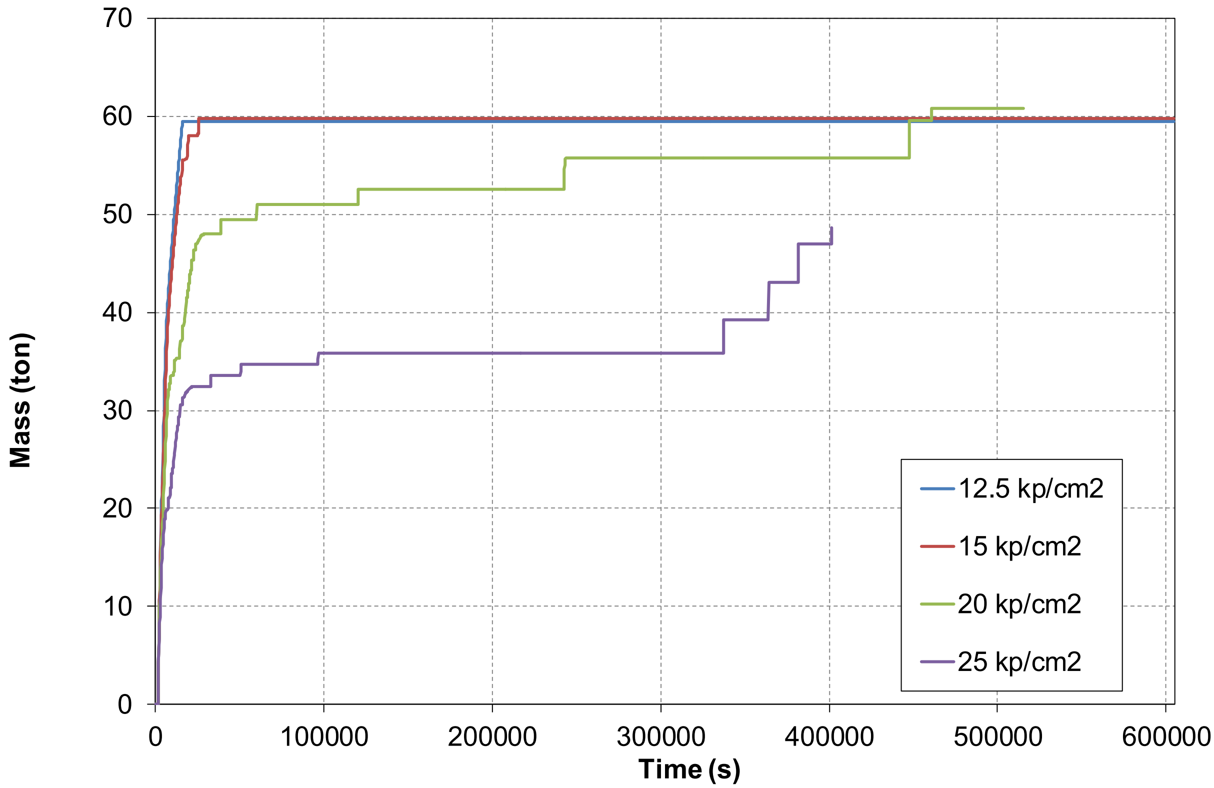

Figure 14 Mass injected into RCS – scenario S0-50

5.1.3 Scenario S0-75

The results for scenario S0-75 are shown in Figures 15 through 20. It can be seen that primary pressure (Figure 15) follows the secondary side depressurization. Core exit temperature (Figure 16) shows that core heatup occurred for all cases in the first five days. The core collapsed liquid level as a function of depressurization is shown in Figure 17. It can be seen, that initially the level dropped due to stopped injection from accumulators and then remains around 80% until core uncovery. Figure 18 shows the average fuel cladding temperature. Again, at larger depressurization initially more mass is discharged from the primary system than at lower depressurization (see Figure 19) because more mass is injected from the accumulators (see Figure 20). Later larger depressurization means smaller discharge of RCS inventory. With high depressurization (case 12.5 kp/cm2) the RCS could be sufficiently cooled two days more than by low depressurization case (25 kp/cm2). Nevertheless, with transient progression the core heatup happened in all cases before five days.

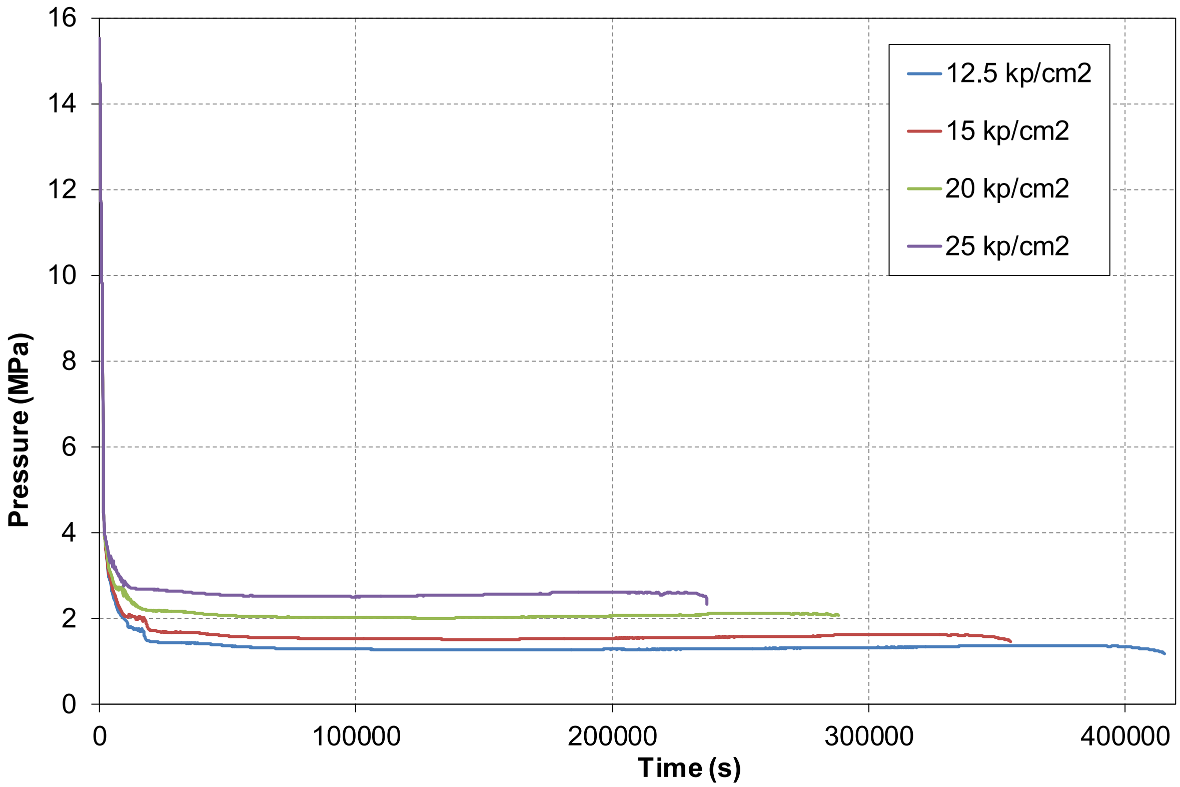

Figure 15 RCS pressure – scenario S0-75

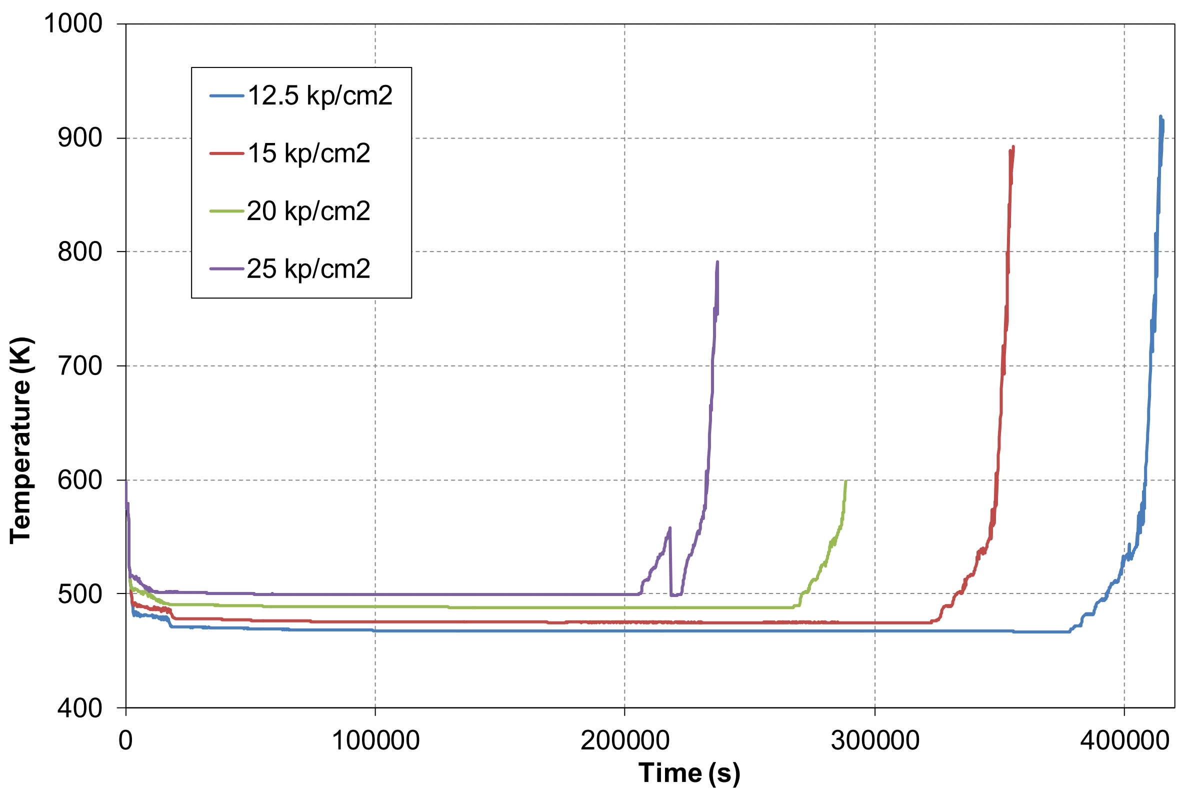

Figure 16 Core exit temperature – scenario S0-75

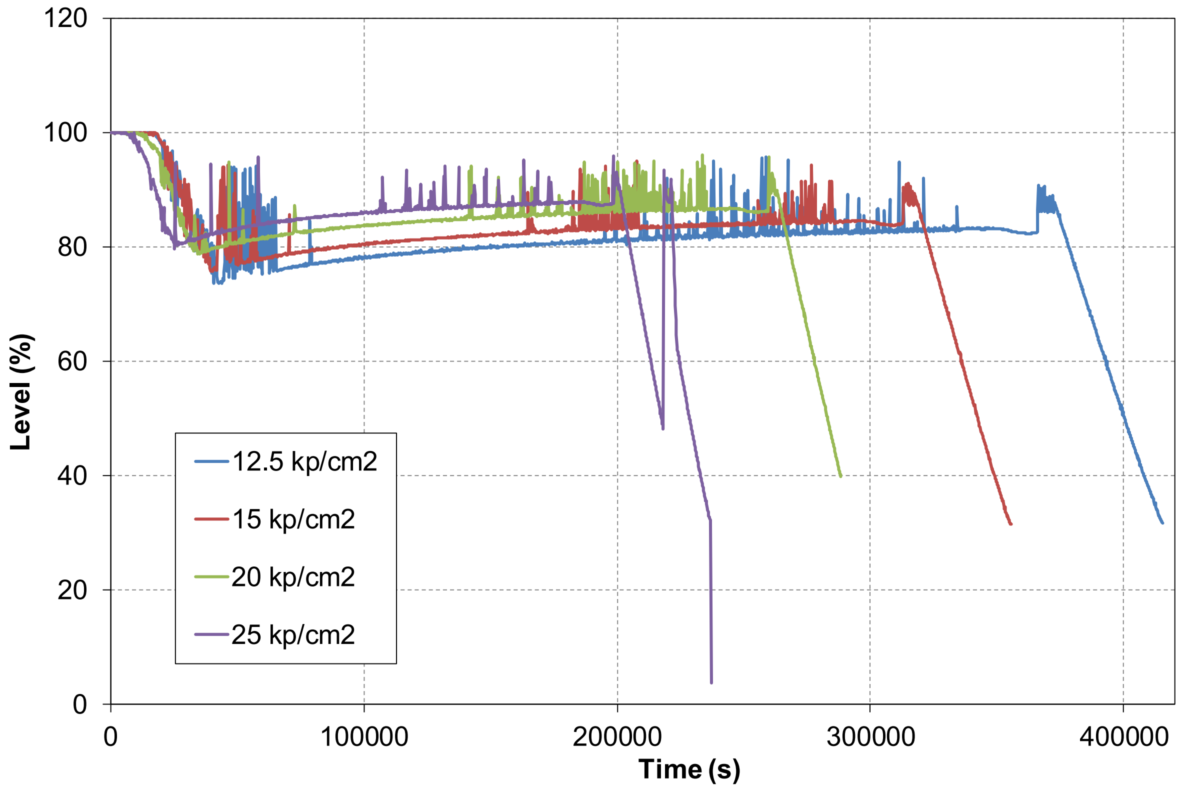

Figure 17 Core collapsed liquid level – scenario S0-75

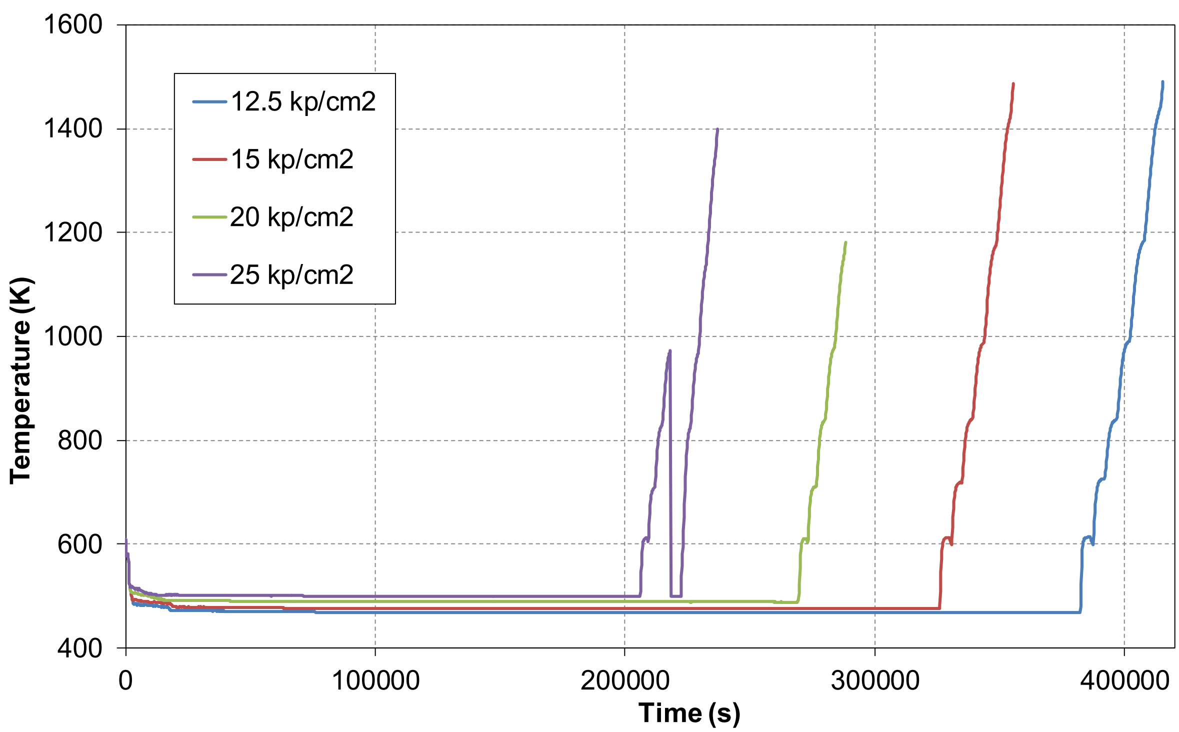

Figure 18 Average fuel cladding temperature – scenario S0-75

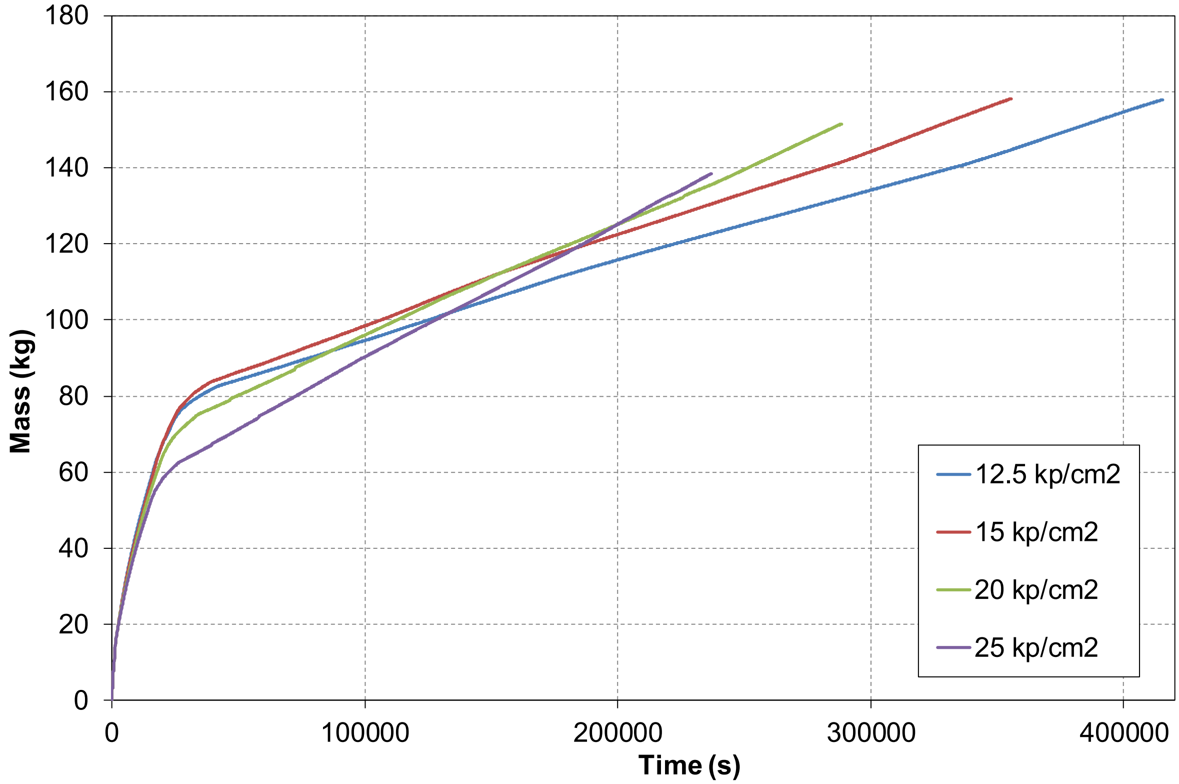

Figure 19 Total mass discharged from RCS – scenario S0-75

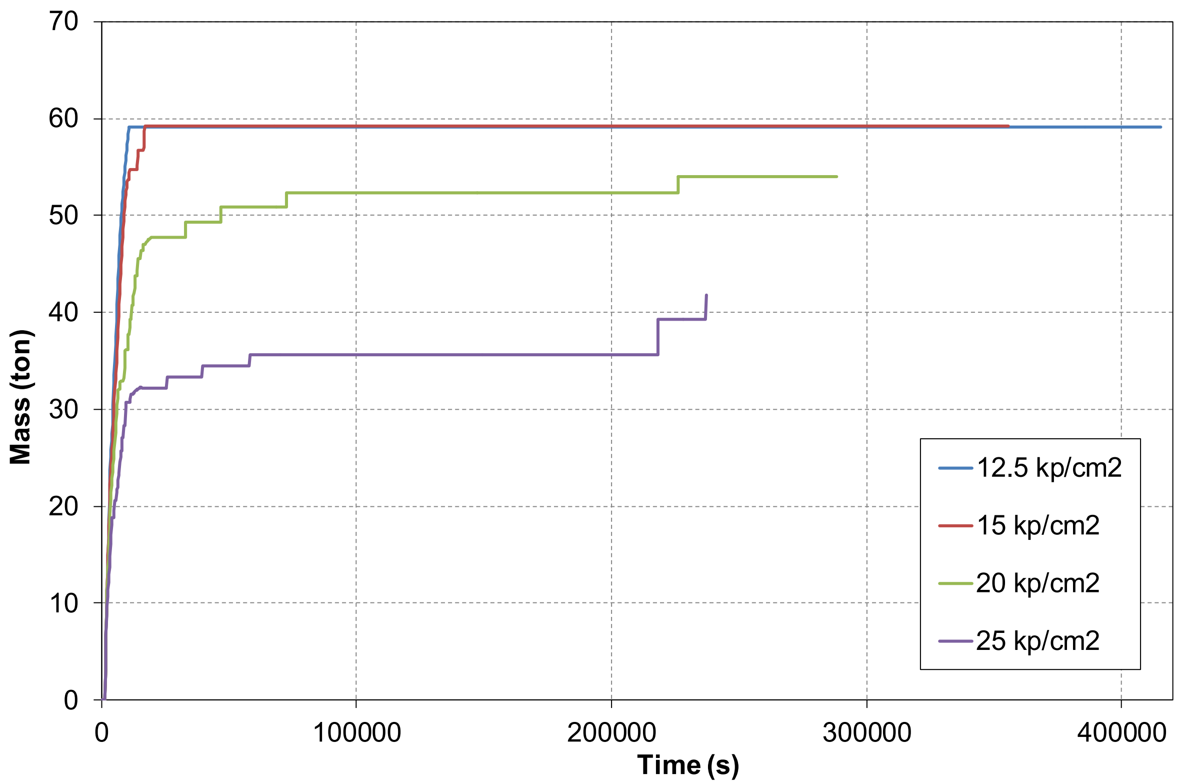

Figure 20 Mass injected into RCS – scenario S0-75

5.1.4 Scenario S0-100

The results for scenario S0-100 are shown in Figures 21 through 26. It can be seen that primary pressure (Figure 21) follows the secondary side depressurization. Core exit temperature (Figure 22) shows that core heatup occurred for all cases again faster than in the previous case. The core collapsed liquid level as a function of depressurization is shown in Figure 23. It can be seen, that initially the level dropped due to stopped injection from accumulators and then remains around 80% until core uncovery. Figure 24 shows the average fuel cladding temperature. Again, at larger depressurization initially more mass is discharged from the primary system than at lower depressurization (see Figure 25) because more mass is injected from the accumulators (see Figure 26). Later larger depressurization means smaller discharge of RCS inventory. With high depressurization (case 12.5 kp/cm2) the RCS could be sufficiently cooled one and half day longer than by low depressurization case (25 kp/cm2). Nevertheless, with transient progression the core heatup happened in all cases before four days.

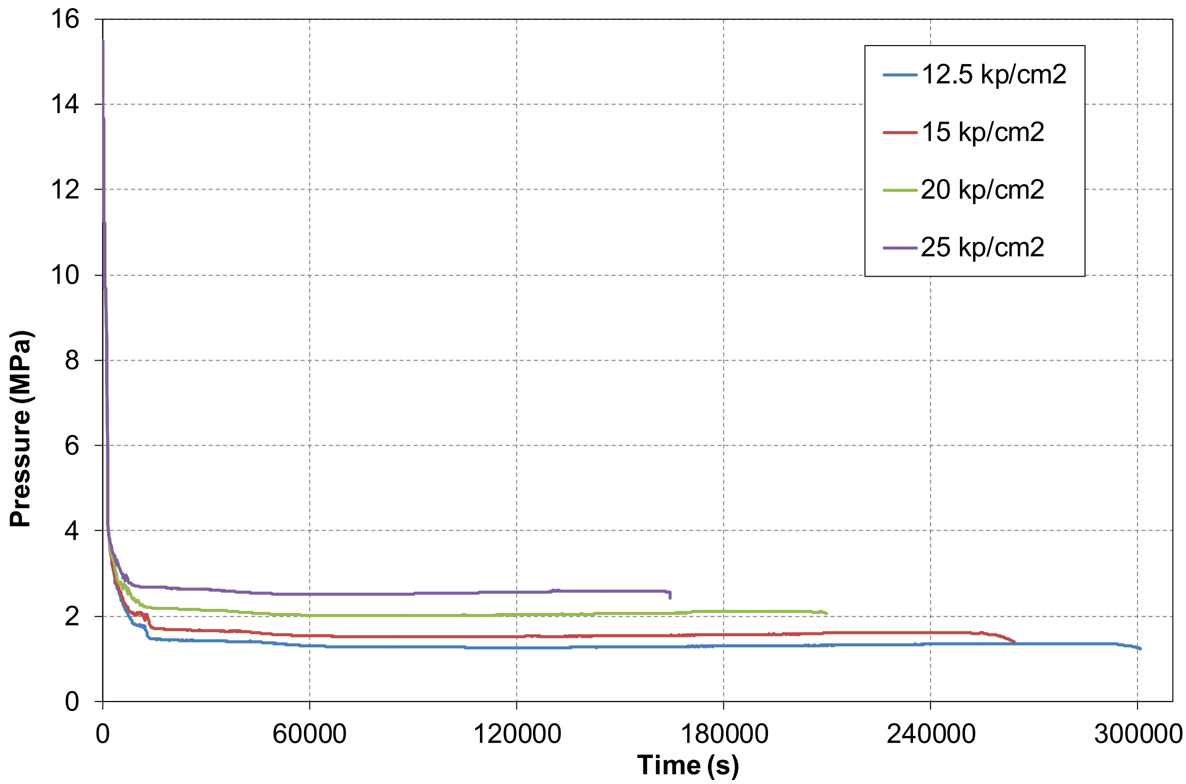

Figure 21 RCS pressure – scenario S0-100

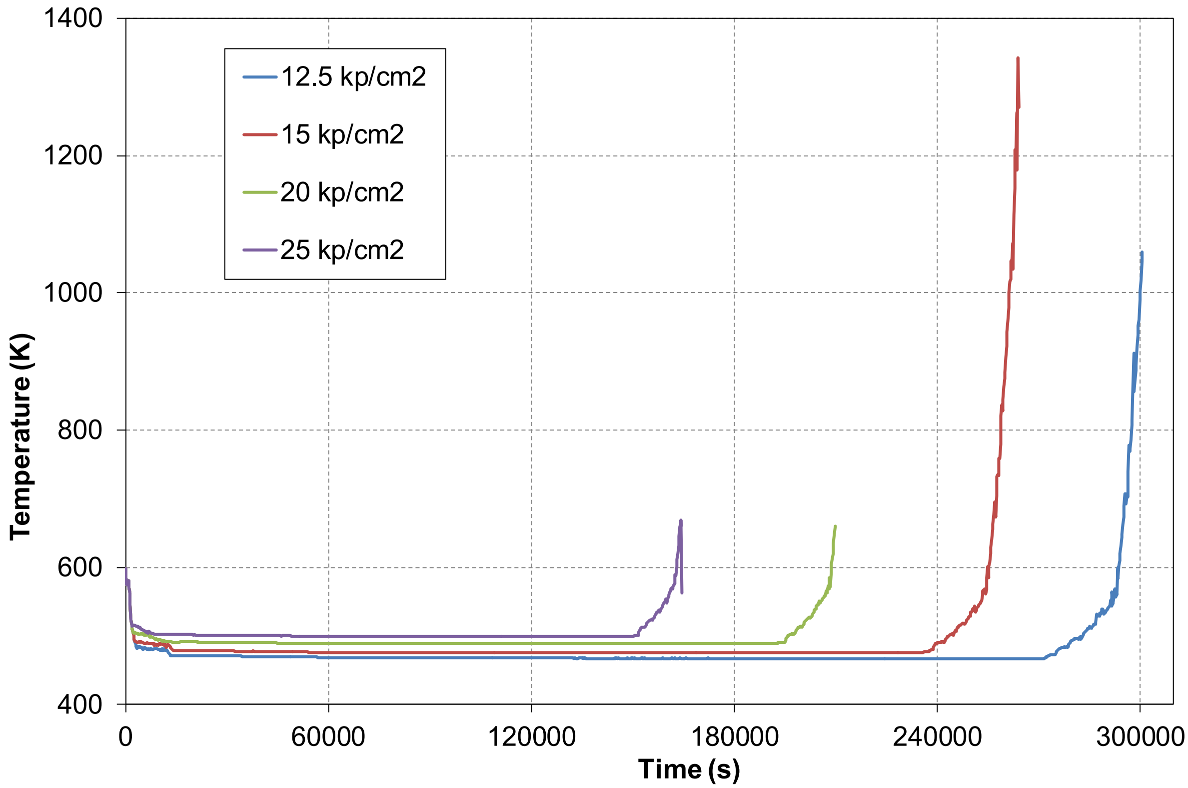

Figure 22 Core exit temperature – scenario S0-100

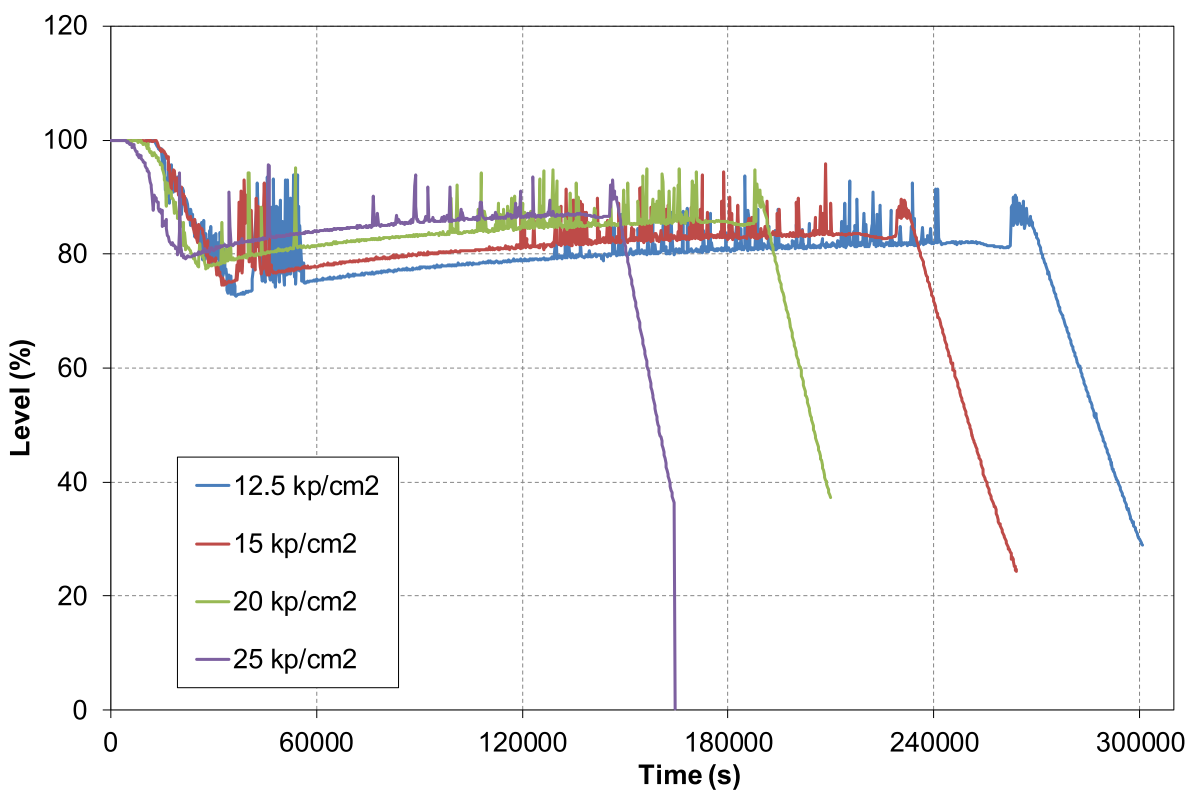

Figure 23 Core collapsed liquid level – scenario S0-100

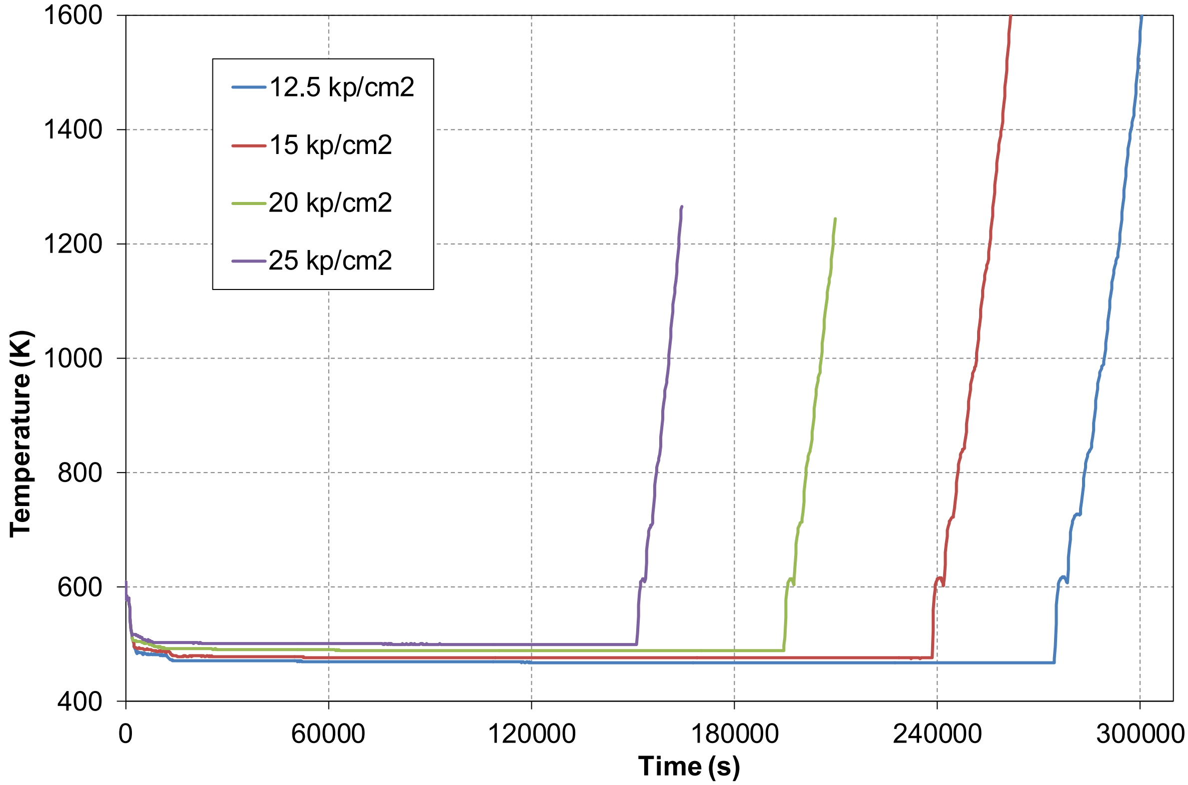

Figure 24 Average fuel cladding temperature – scenario S0-100

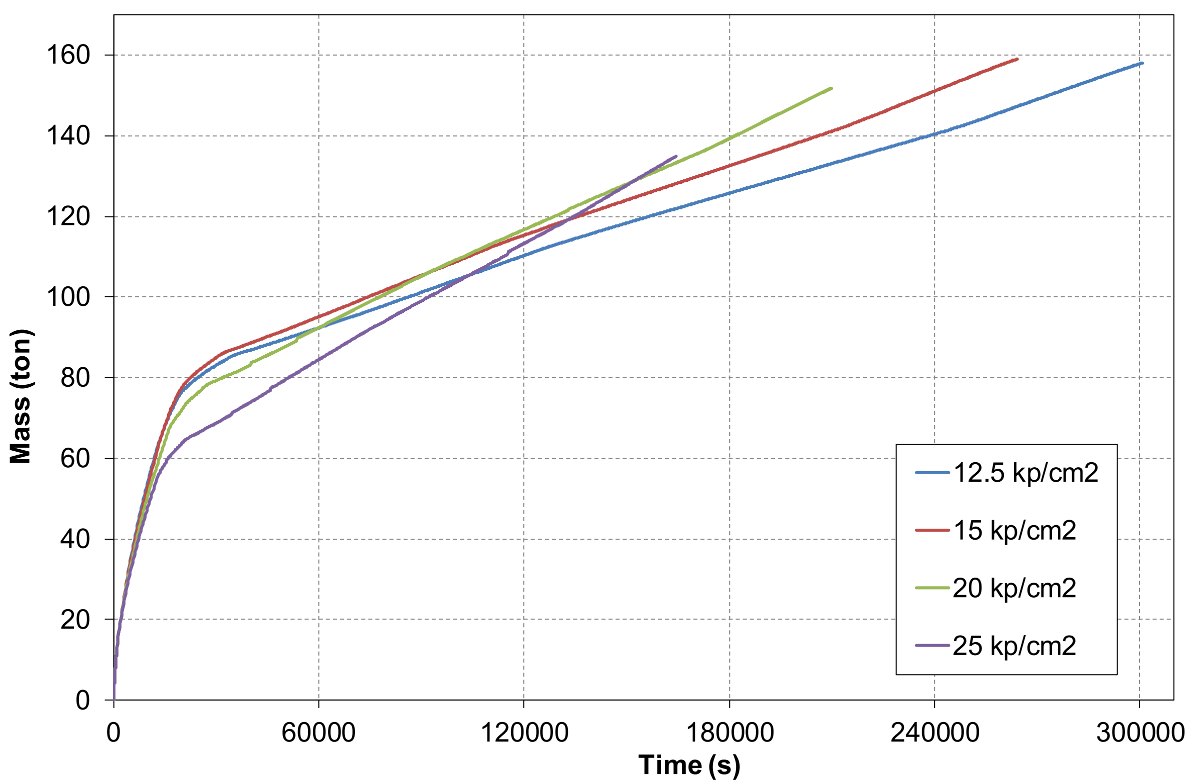

Figure 25 Total mass discharged from RCS – scenario S0-100

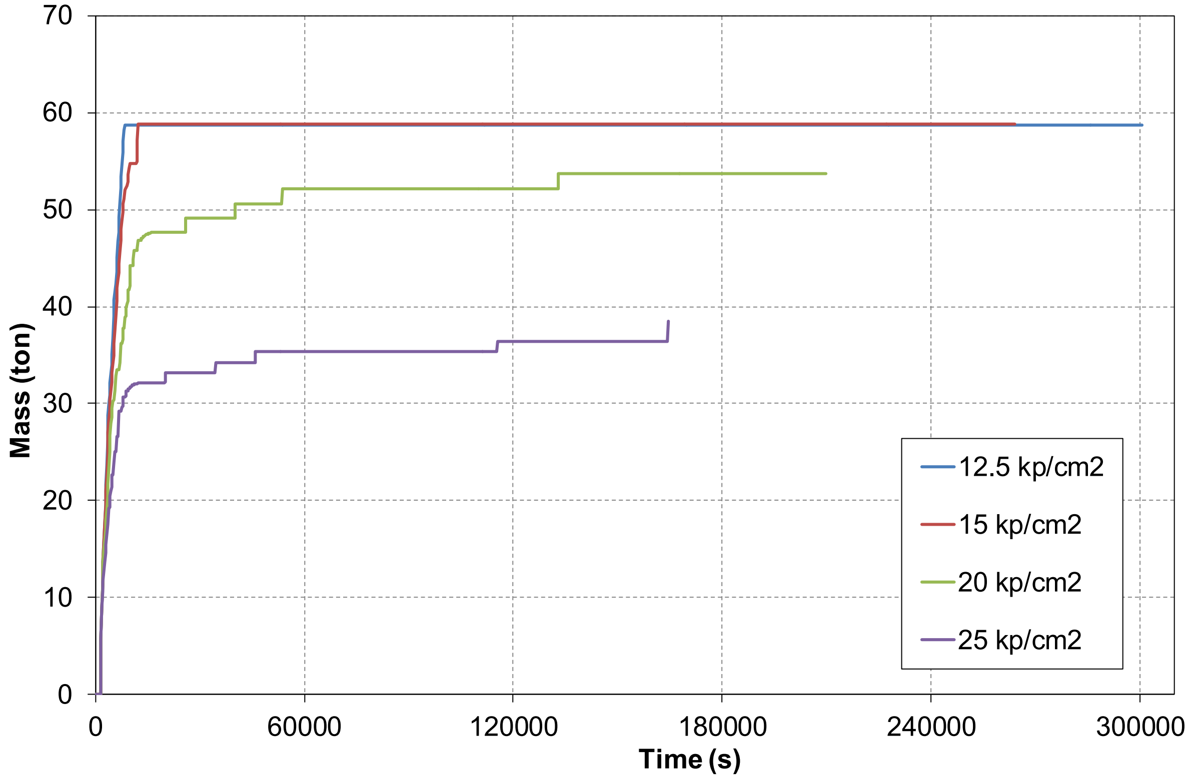

Figure 26 Mass injected into RCS – scenario S0-100

5.1.5 Scenario S0-150

The results for scenario S0-150 are shown in Figures 27 through 32. It can be seen that primary pressure (Figure 27) follows the secondary side depressurization. Core exit temperature (Figure 28) shows that core heatup occurred for all cases again faster than in the previous case. The core collapsed liquid level as a function of depressurization is shown in Figure 29. It can be seen, that initially the level dropped due to stopped injection from accumulators and then remains around 80% until core uncovery. Figure 30 shows the average fuel cladding temperature. Again, at larger depressurization initially more mass is discharged from the primary system than at lower depressurization (see Figure 31) because more mass is injected from the accumulators (see Figure 32). Later larger depressurization means smaller discharge of RCS inventory. With high depressurization (case 12.5 kp/cm2) the RCS could be sufficiently cooled one day more than by low depressurization case (25 kp/cm2). Nevertheless, with transient progression the core heatup happened in all cases before two and half days.

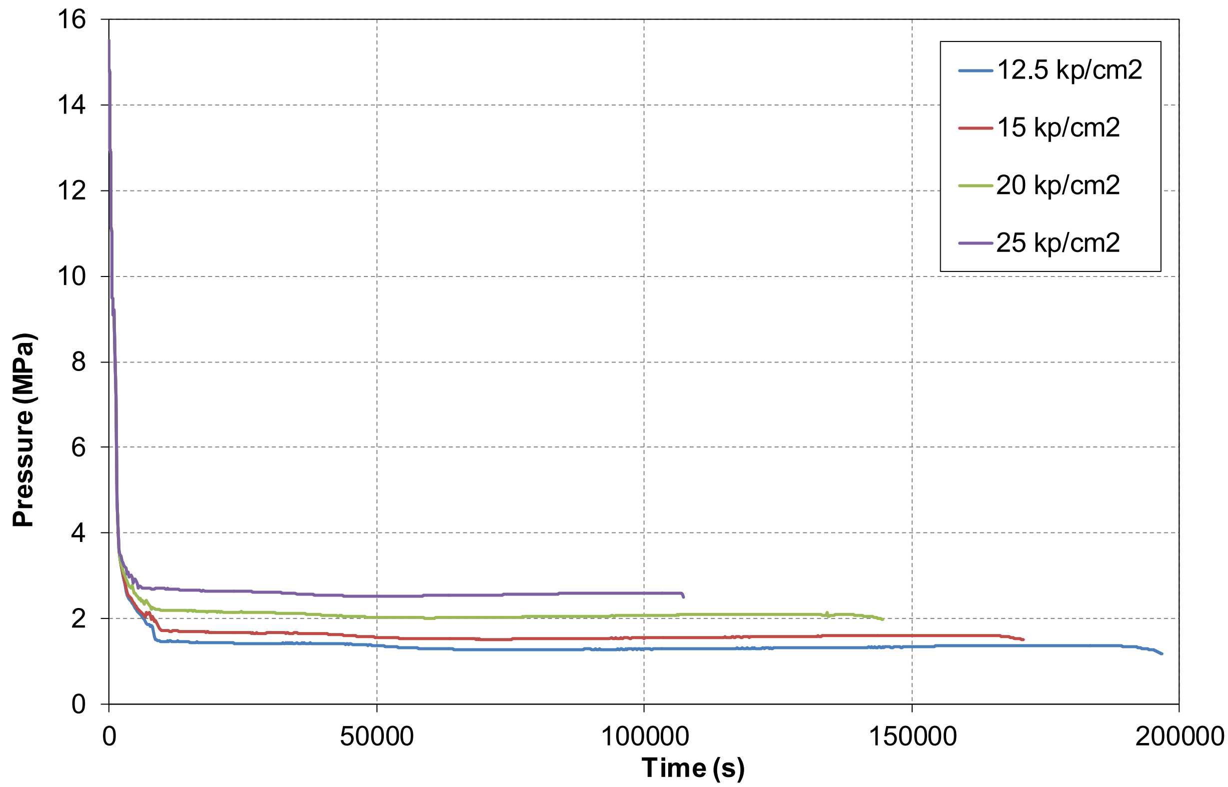

Figure 27 RCS pressure – scenario S0-150

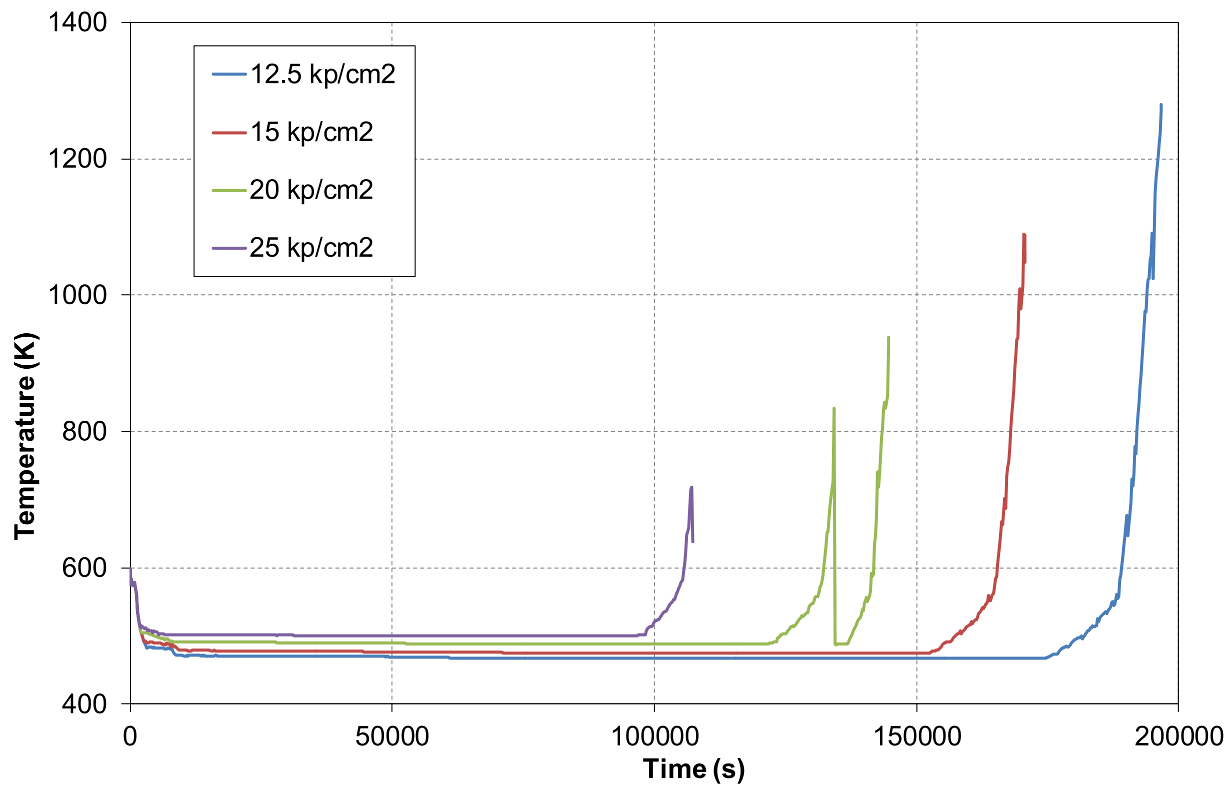

Figure 28 Core exit temperature – scenario S0-150

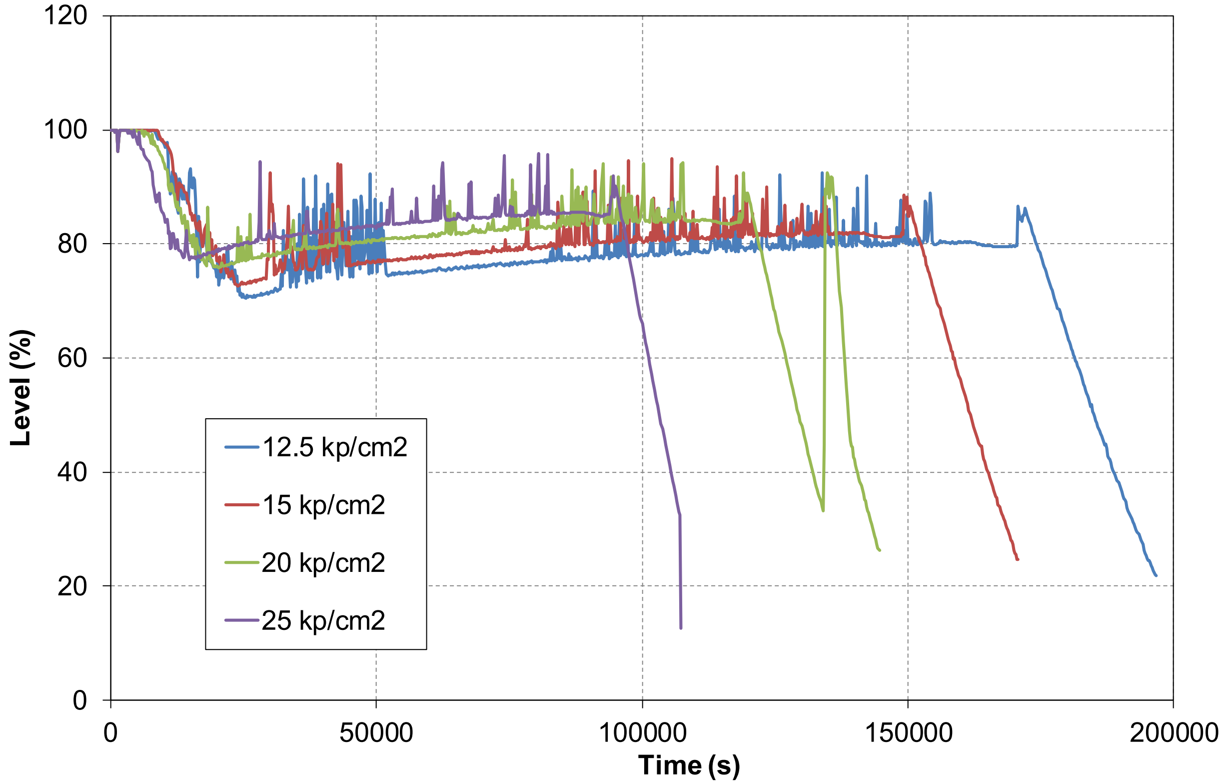

Figure 29 Core collapsed liquid level – scenario S0-150

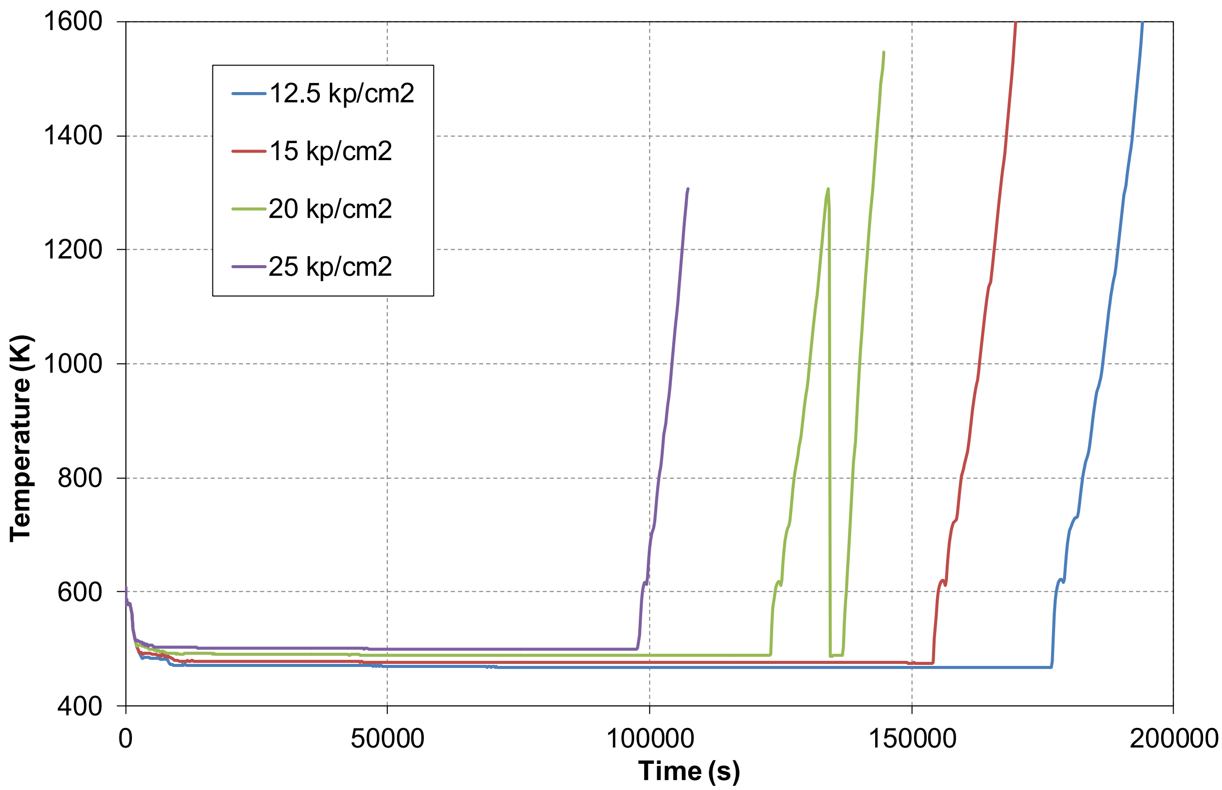

Figure 30 Average fuel cladding temperature – scenario S0-150

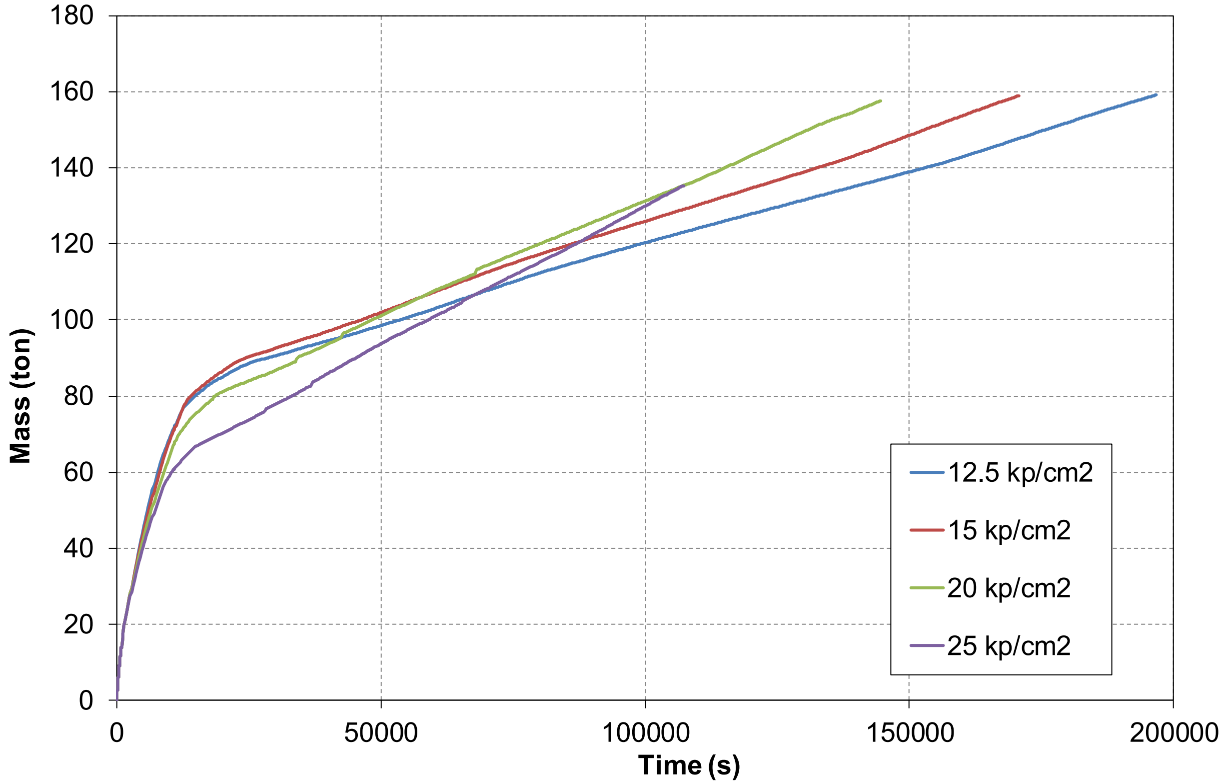

Figure 31 Total mass discharged from RCS – scenario S0-150

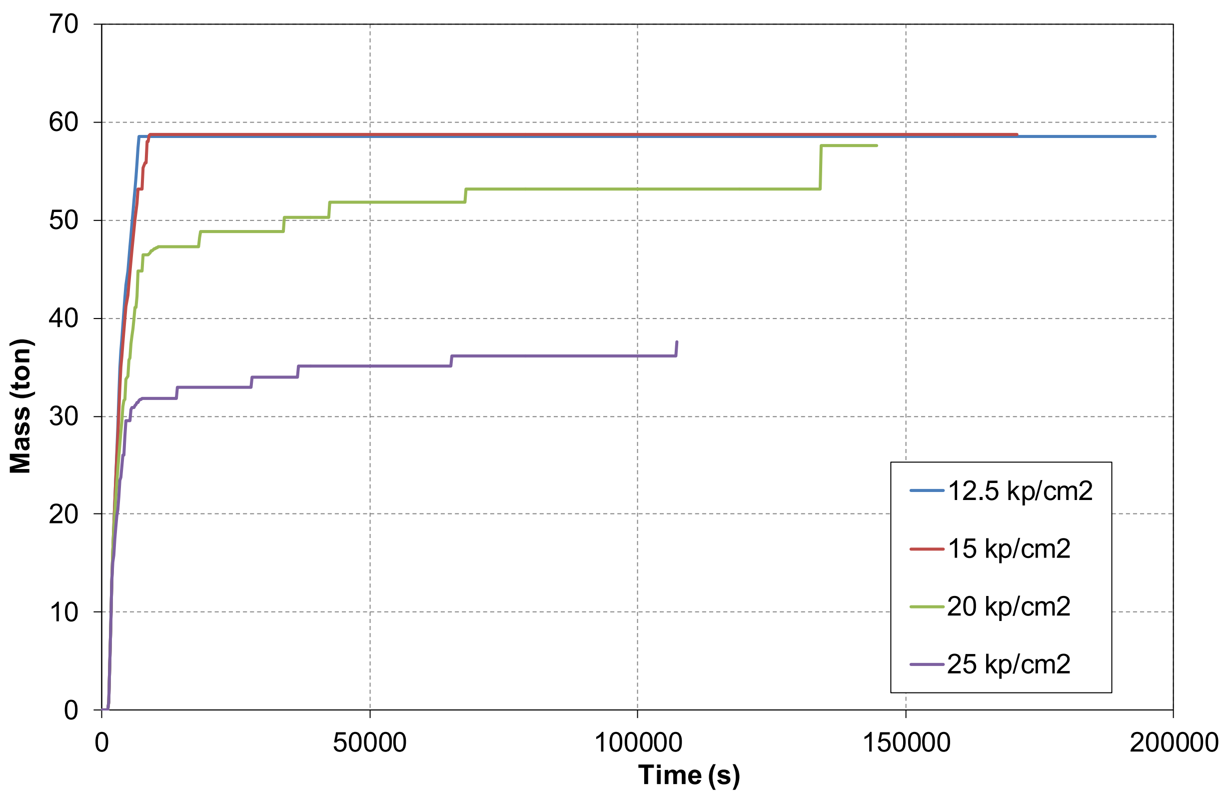

Figure 32 Mass injected into RCS – scenario S0-150

5.1.6 Scenario S0-150p

The results for scenario S0-150p are shown in Figures 33 through 38. The only difference comparing to S0-150 scenario is, that PDP charging pump with the capacity to inject 2.2 kg/s into RCS is used after 4 hours. Namely, it is not sufficient just to cool the primary system, as core heatup resulted from RCS inventory depletion. Therefore RCS inventory injection is needed. It can be seen that RCS pressure (Figure 33) follows the secondary side depressurization in the first part only. Later cooling through the breaks and steam consumption by TD AFW pumps is sufficient provided that RCS inventory makeup is provided. Core exit temperature (Figure 34) shows that core heatup does not occur. The core collapsed liquid level as a function of depressurization is shown in Figure 35. It can be seen, that initially the level dropped due to stopped injection from accumulators and then remains around 90% until core uncovery. Figure 36 shows the average fuel cladding temperature, which is decreasing. The mass discharged from RCS (see Figure 37) and injected mass (see Figure 38) is not dependent on the depressurization in the second part of transient. The injected mass to and discharged mass from RCS are practically balanced. The RCS system is efficiently cooled through breaks besides secondary side cooling. The selected case clearly showed that RCS injection is also very important for preventing core uncovery; especially in the cases with larger breaks this is the only way to prevent core heatup in the first seven days.

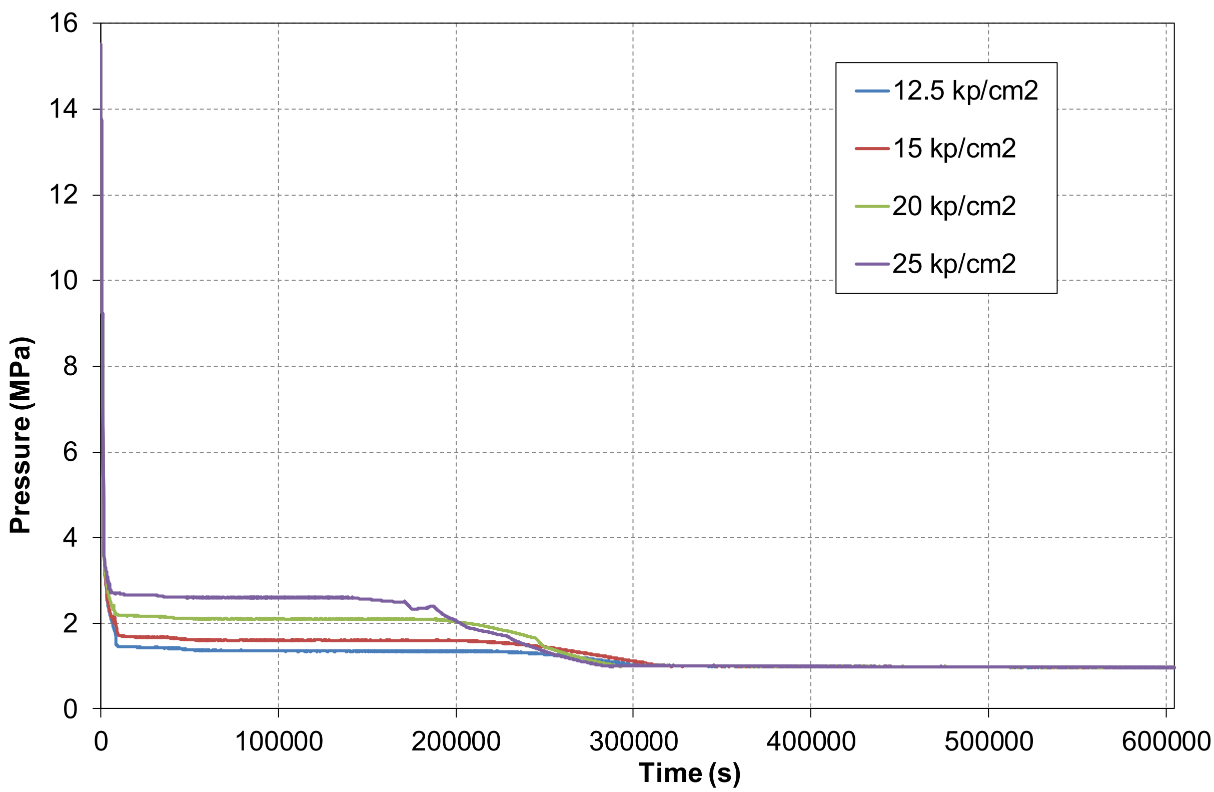

Figure 33 RCS pressure – scenario S0-150p

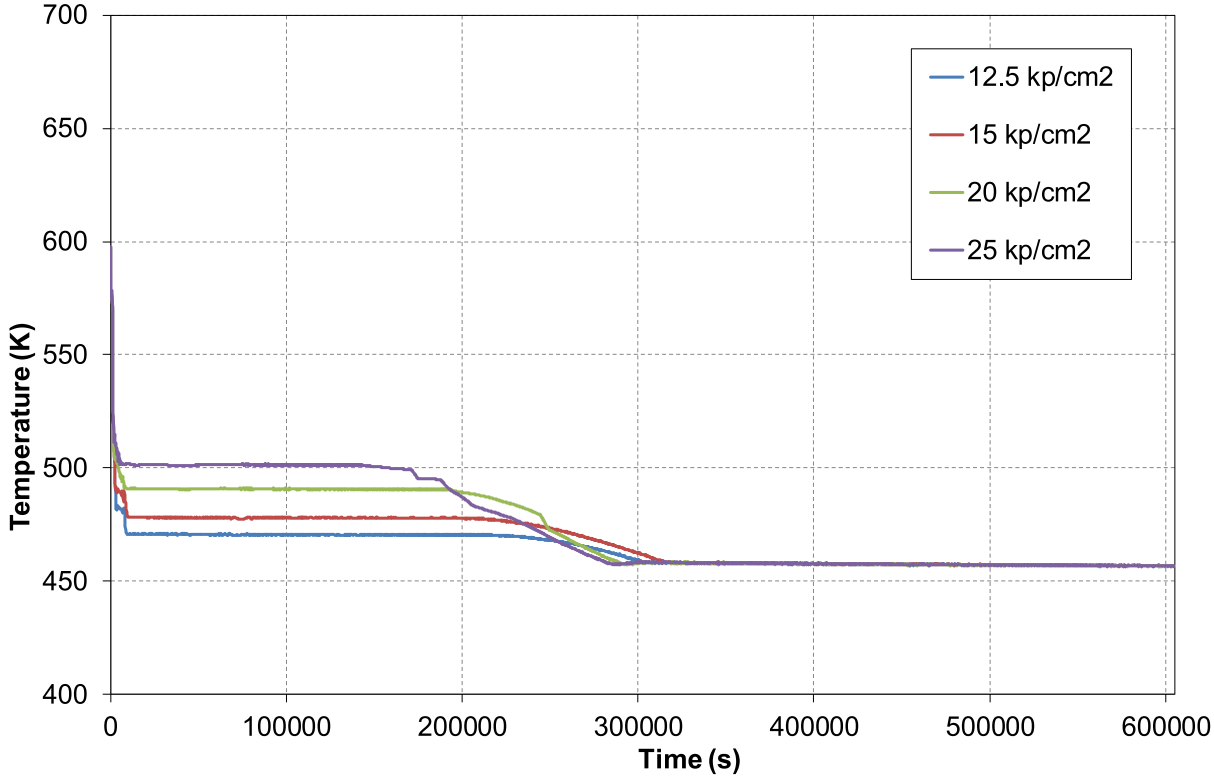

Figure 34 Core exit temperature – scenario S0-150p

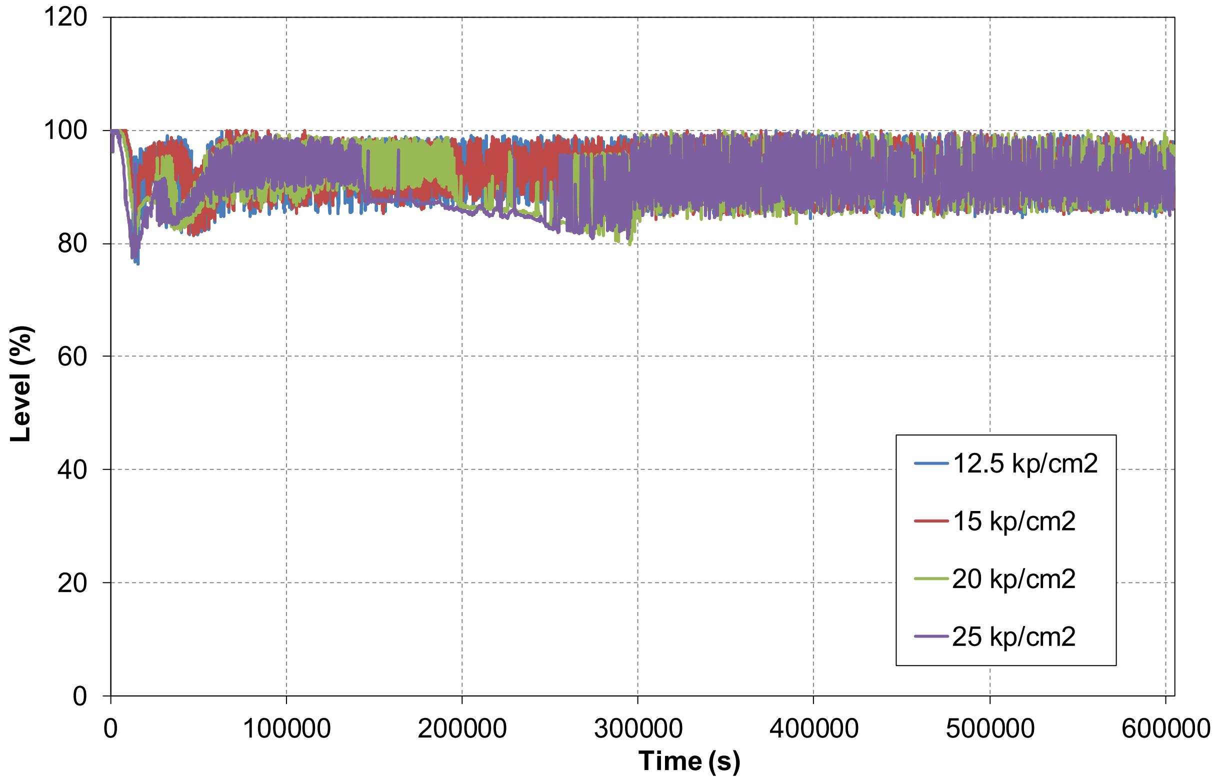

Figure 35 Core collapsed liquid level – scenario S0-150p

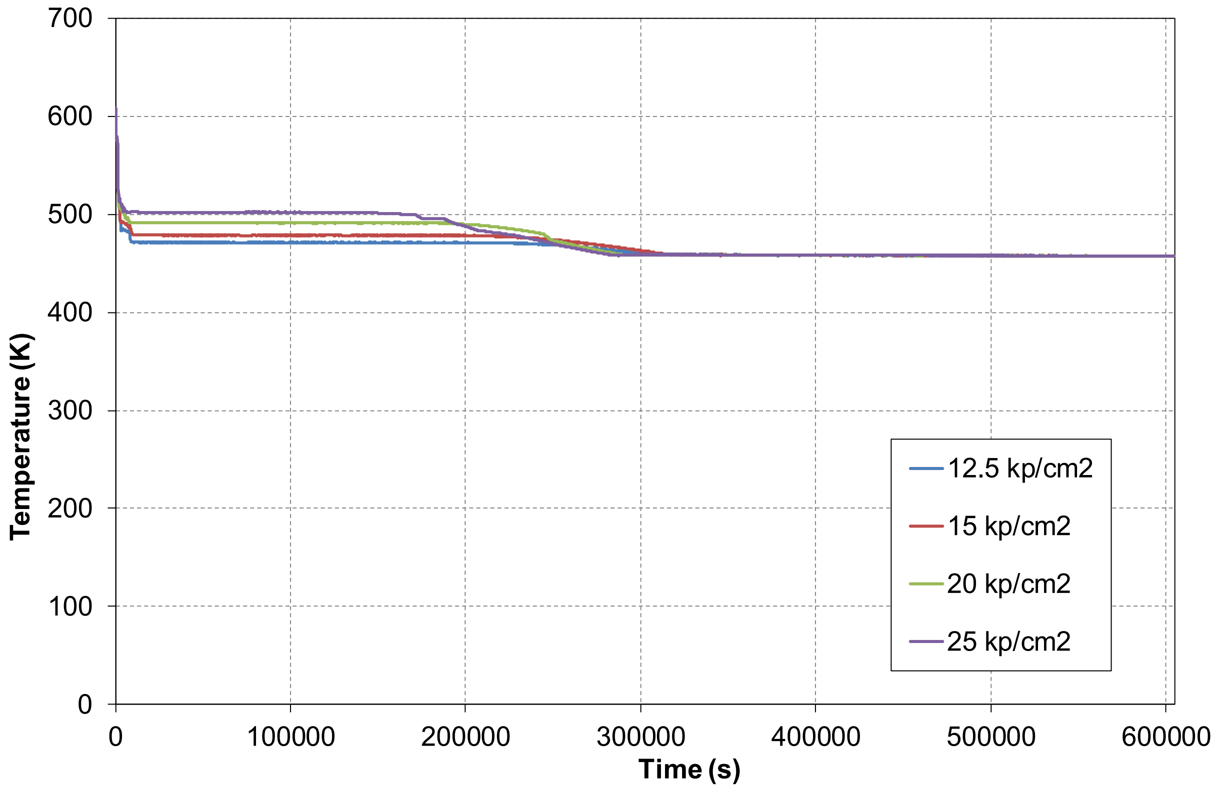

Figure 36 Average fuel cladding temperature – scenario S0-150p

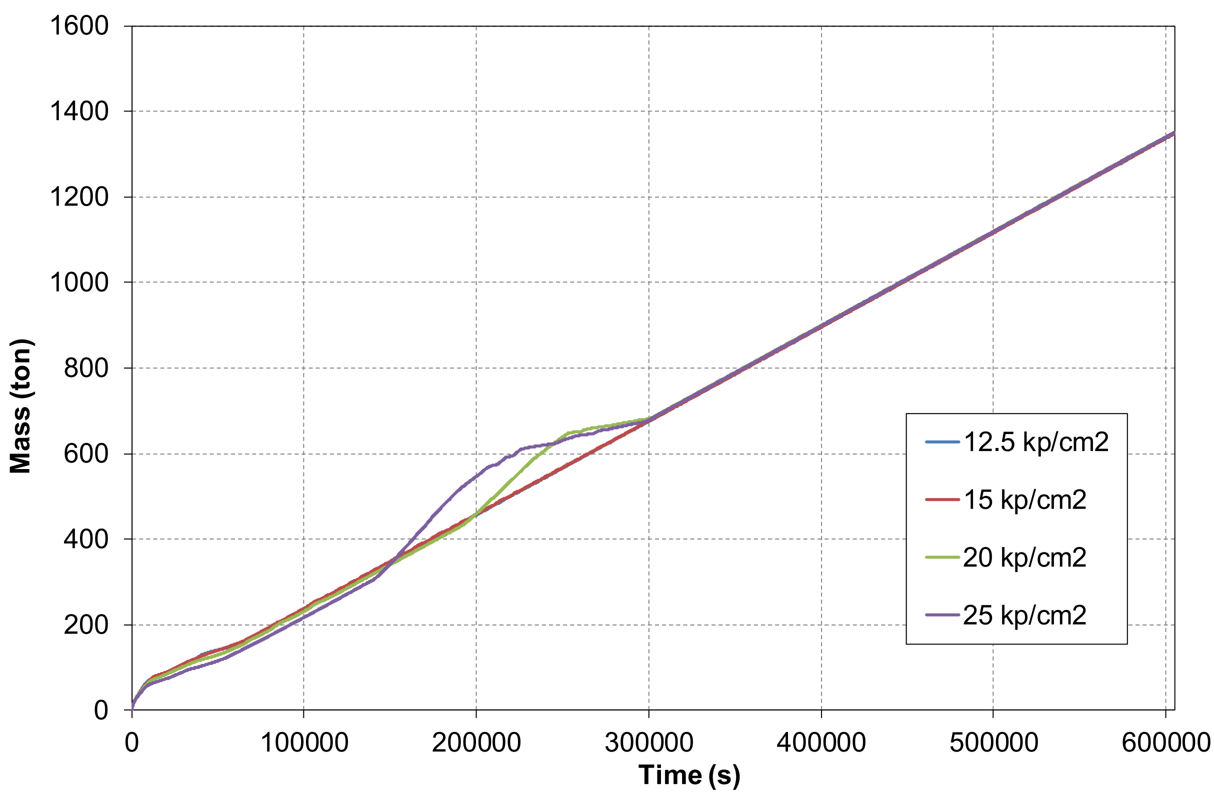

Figure 37 Total mass discharged from RCS – scenario S0-150p

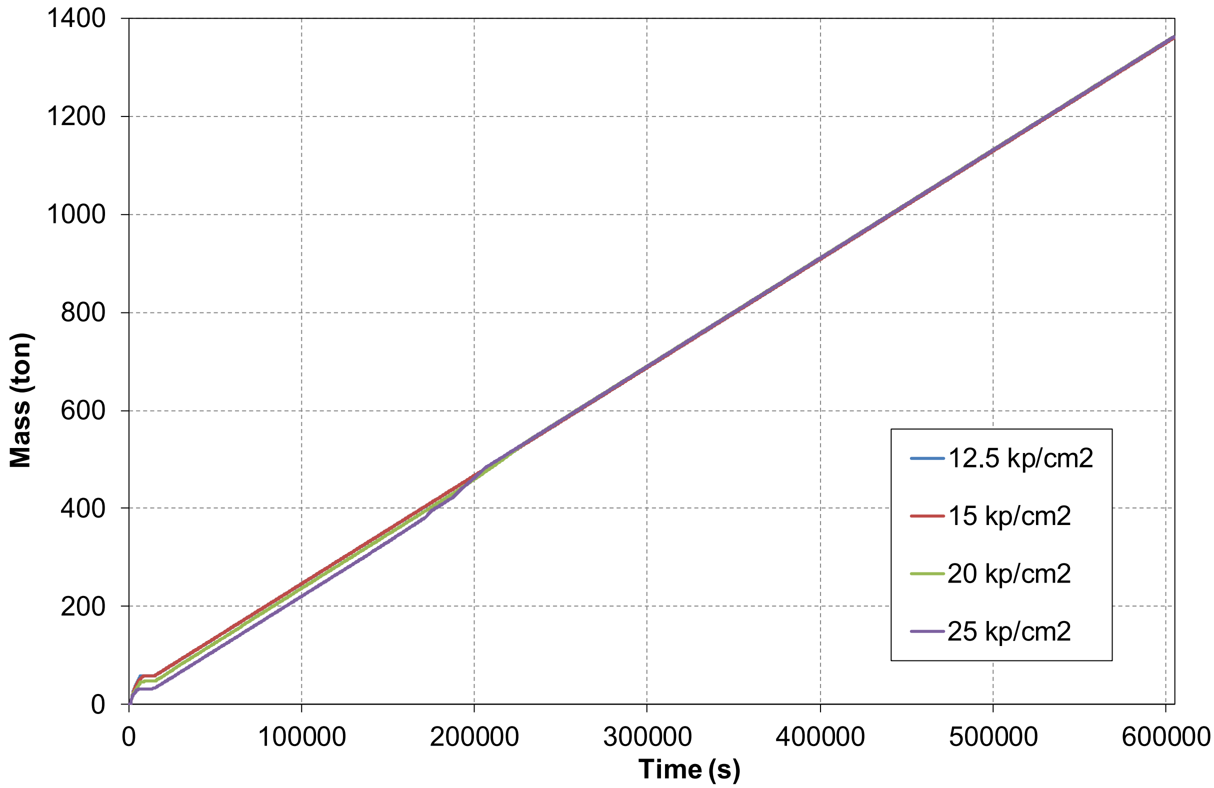

Figure 38 Mass injected into RCS – scenario S0-150p

5.1.7 Scenario S0-300

The results for scenario S0-300 are shown in Figures 39 through 44. It can be seen that RCS pressure (Figure 39) follows the secondary side depressurization. Core exit temperature (Figure 40) shows that core heatup occurred for all cases again faster than in the S0-150 case. The core collapsed liquid level as a function of depressurization is shown in Figure 41. It can be seen, that initially the level dropped due to stopped injection from accumulators and then remains around 80% until core uncovery. Figure 42 shows the average fuel cladding temperature. Again, at larger depressurization initially more mass is discharged from the primary system than at lower depressurization (see Figure 43) because more mass is injected from the accumulators (see Figure 44). Later larger depressurization means smaller discharge of RCS inventory. With high depressurization (case 12.5 kp/cm2) the RCS could be sufficiently cooled half day more than by low depressurization case (25 kp/cm2). Nevertheless, with transient progression the core heatup happened in all cases in one day. The only alternative to avoid core uncovery is to provide RCS inventory by PDP charging pump or other means.

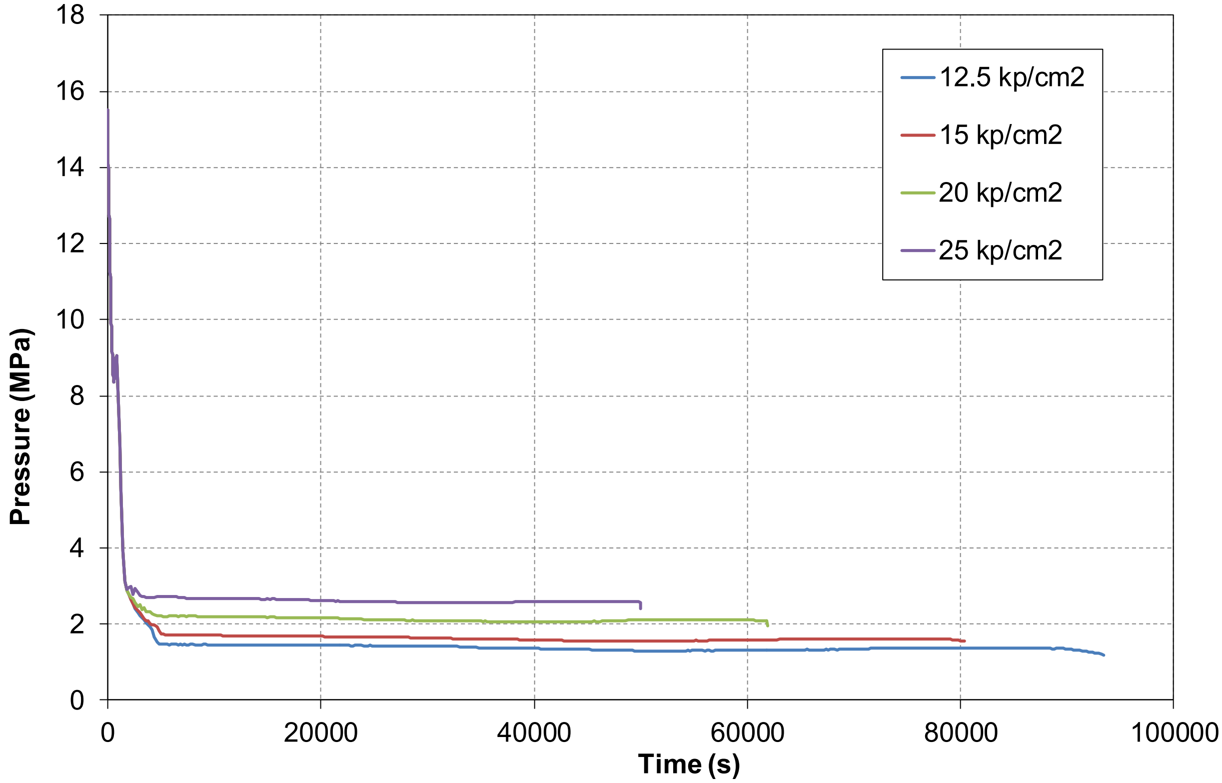

Figure 39 RCS pressure – scenario S0-300

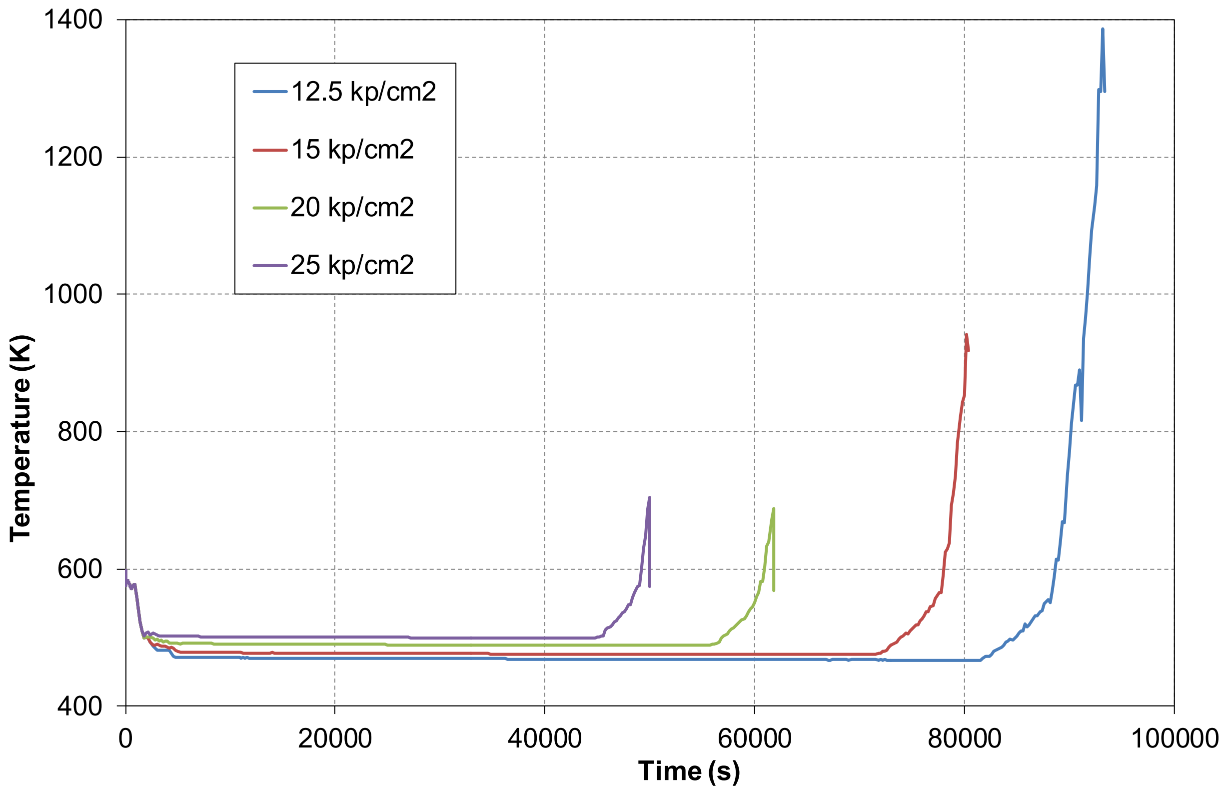

Figure 40 Core exit temperature – scenario S0-300

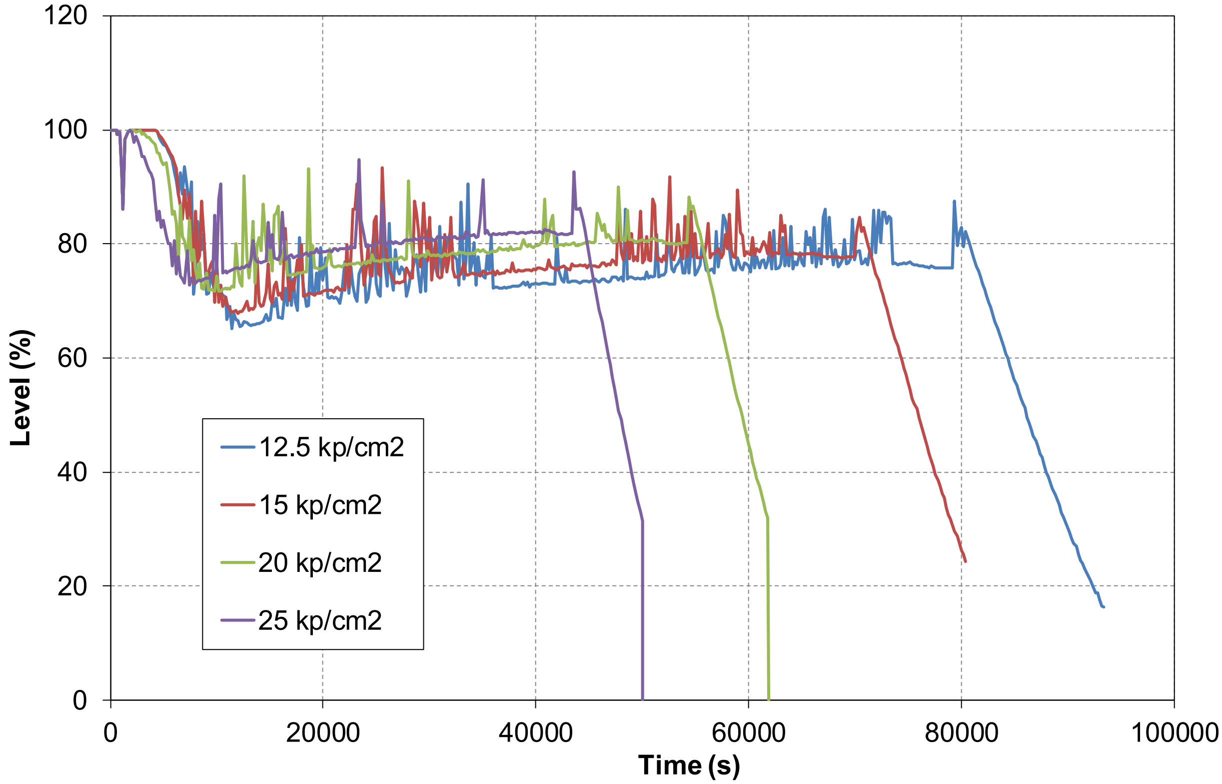

Figure 41 Core collapsed liquid level – scenario S0-300

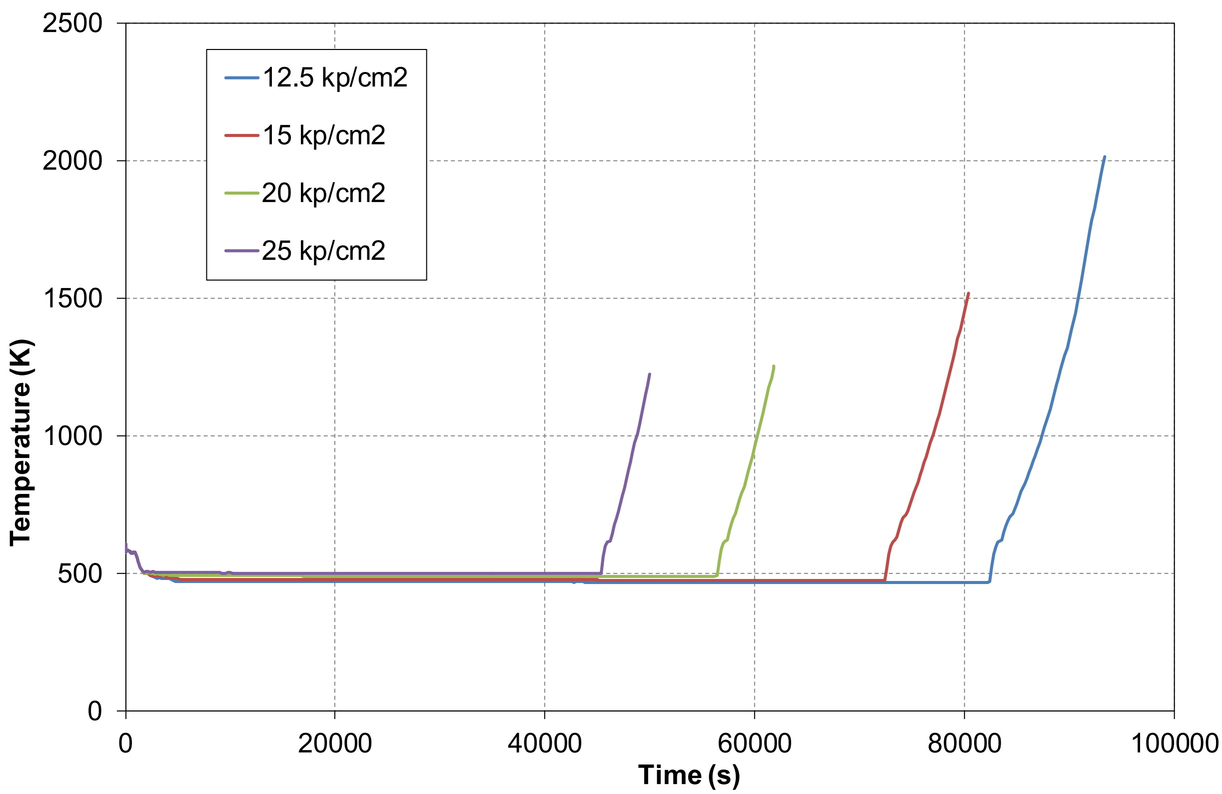

Figure 42 Average fuel cladding temperature – scenario S0-300

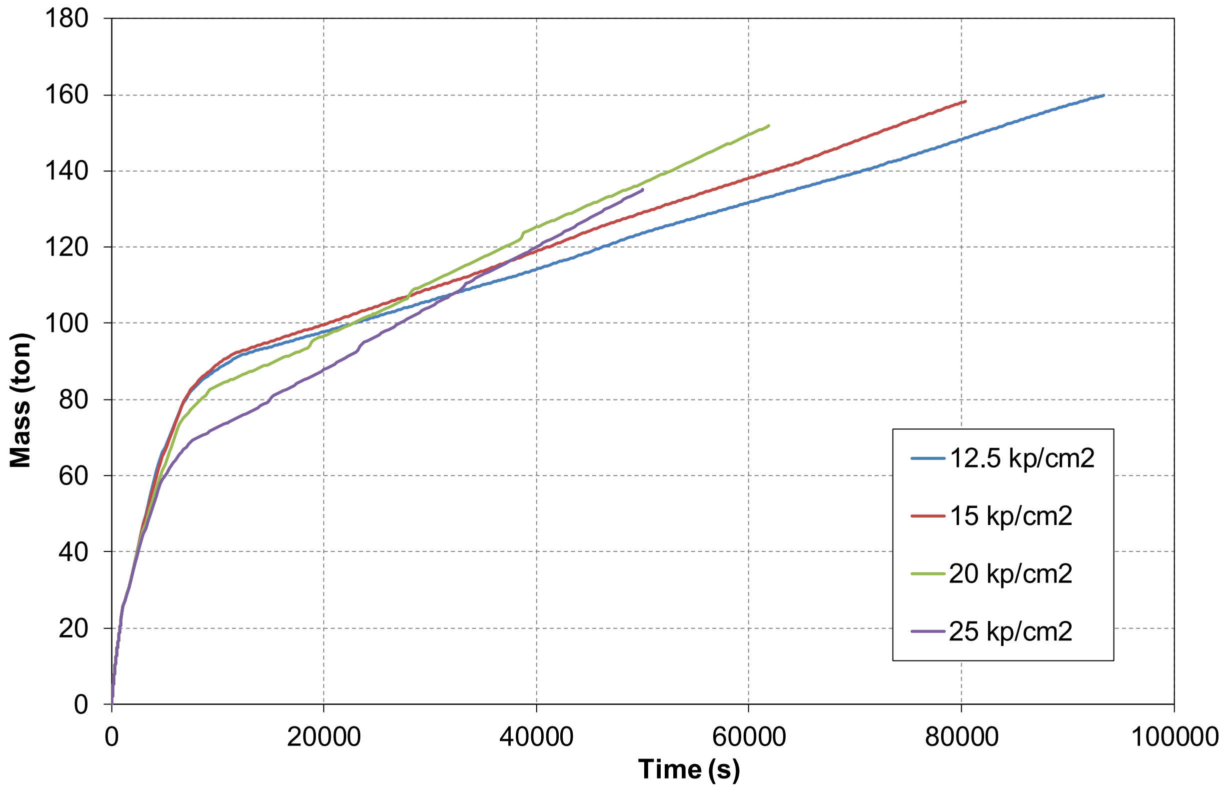

Figure 43 Total mass discharged from RCS – scenario S0-300

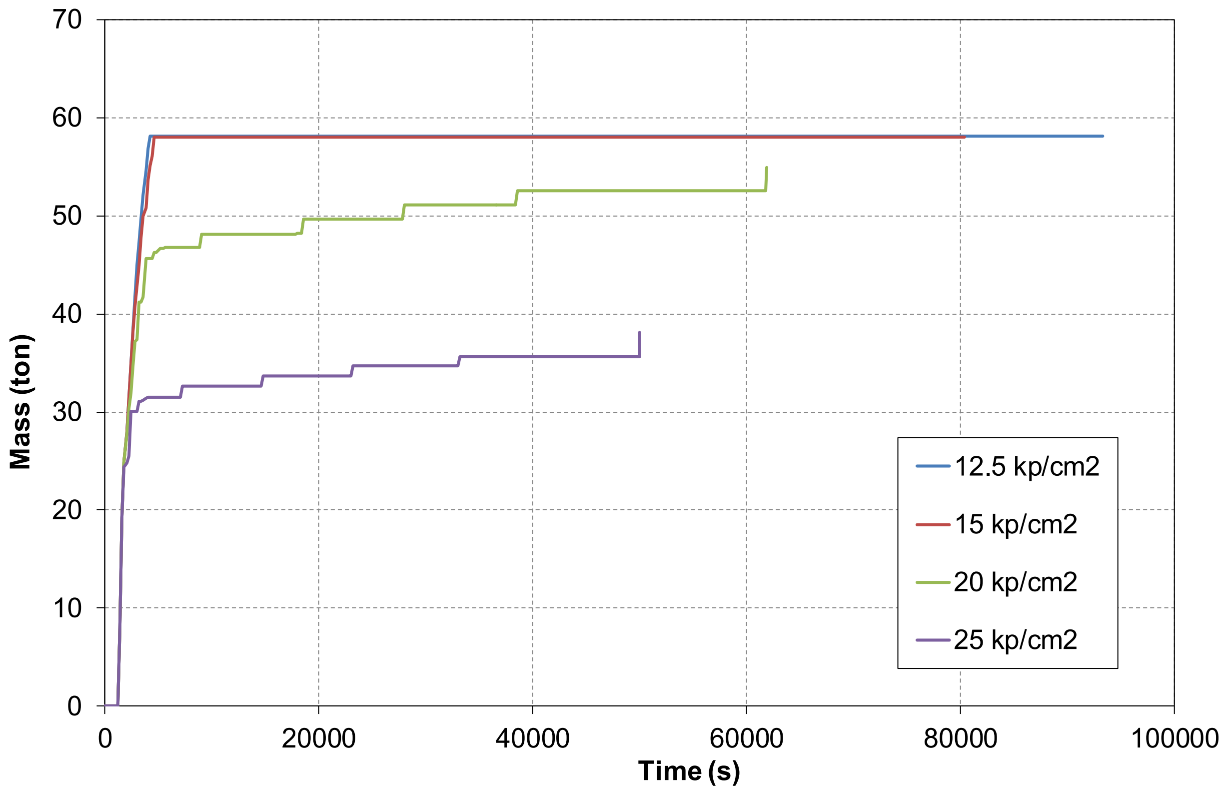

Figure 44 Mass injected into RCS – scenario S0-300

5.1.8 Scenario S1-21

The results for scenario S1-21 are shown in Figures 45 through 50. It can be seen that primary pressure (Figure 45) follows the secondary pressures until SGs as heat sink are lost. The steam generator pressures are maintained by operator until 14400 s. Later the TD AFW pump was assumed not to be available. Core exit temperature shown in Figure 46 increased after core heatup. Only small heatup is shown because calculations were stopped due to reactor kinetics time step reduced below minimum value. No attempt was made to restart calculations as from Figure 47 showing the core collapsed liquid level the core uncovery is evident at the time of calculation abortion and restarts were not very much successful when such error occured. It can also be seen, that after decreasing RCS inventory also the core collapsed liquid levels decrease. However, significant core uncovery happens after 40000 s for all depressurization cases. Figure 48 shows the average fuel cladding temperature, which starts to increase after the core is significantly uncovered. Figure 49 shows mass discharged from RCS, which after 4 hours increases again due to the lost heat sink. Finally, Figure 50 shows the mass injected by accumulators. After RCS system repressurization further injection by accumulators was prevented. The scenario clearly showed that after losing the heat sink after 4 hours on secondary side the plant can survive additional 8 to 10 hours (depending on depressurization) as some cooling is provided by RCP seal leaks.

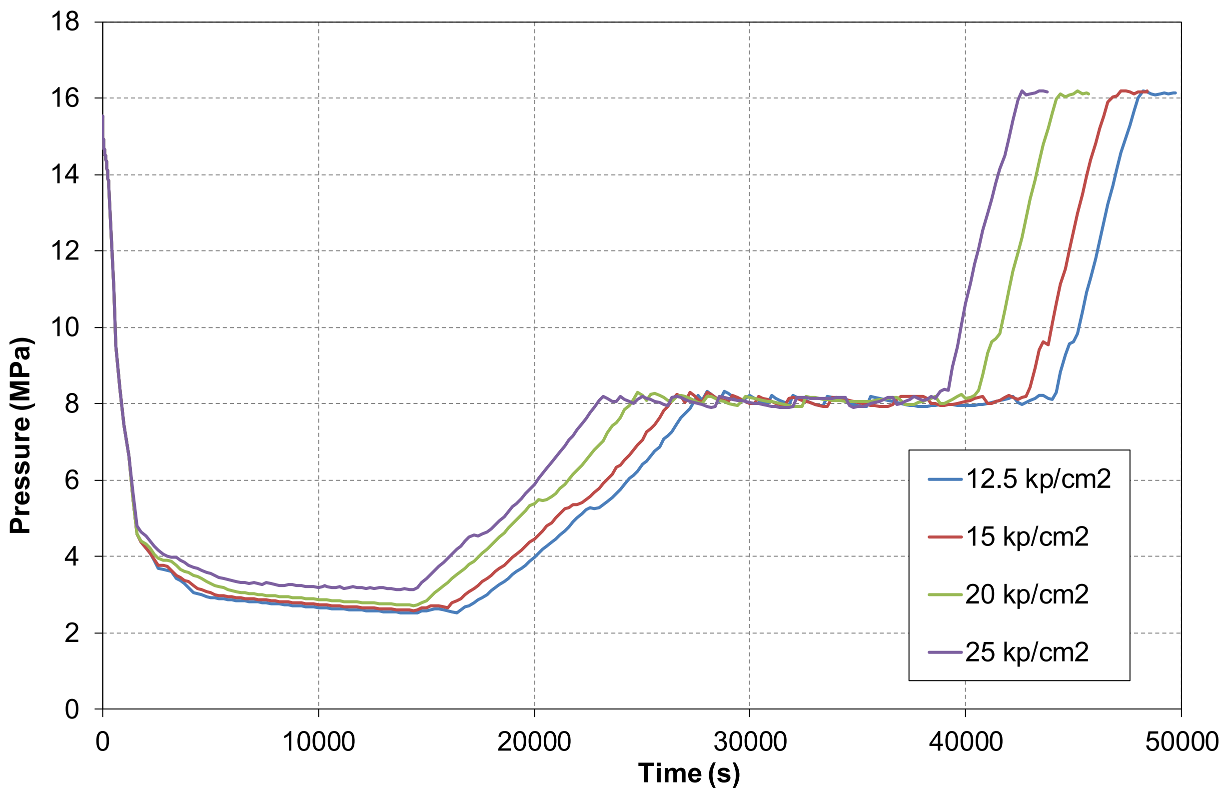

Figure 45 RCS pressure – scenario S1-21

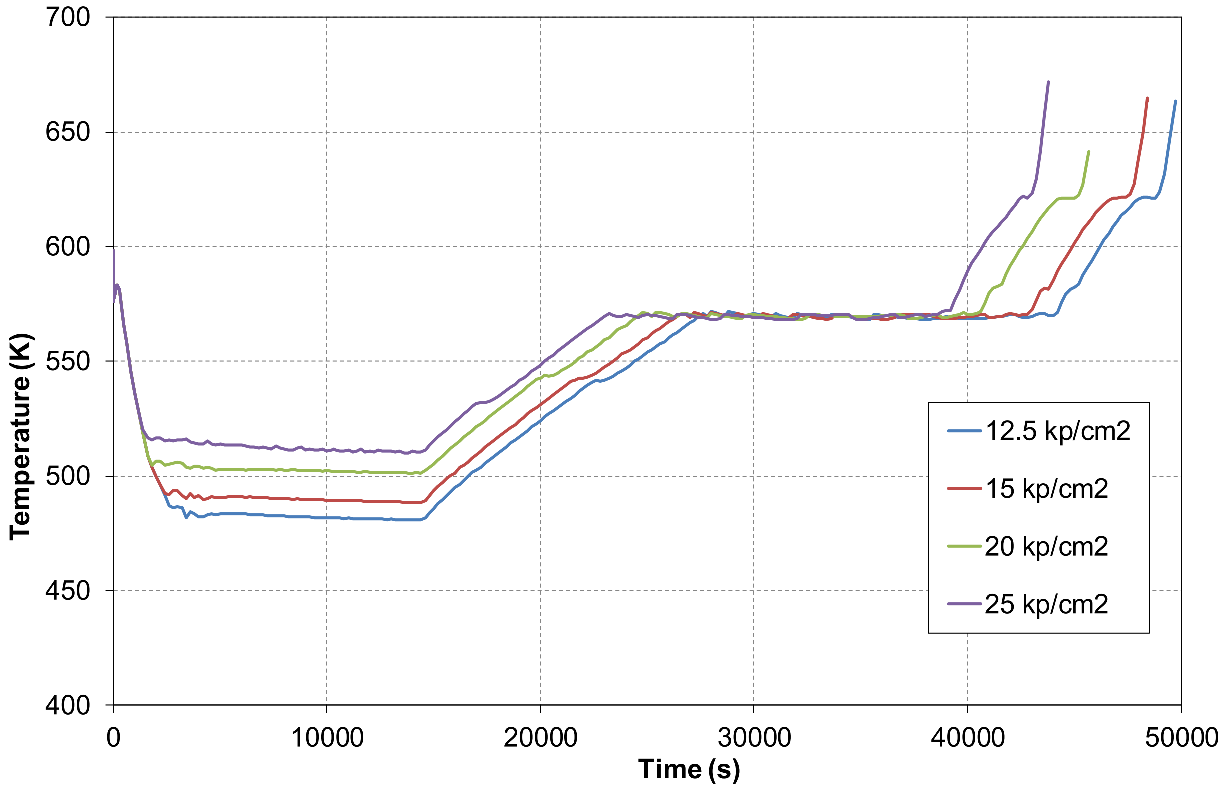

Figure 46 Core exit temperature – scenario S1-21

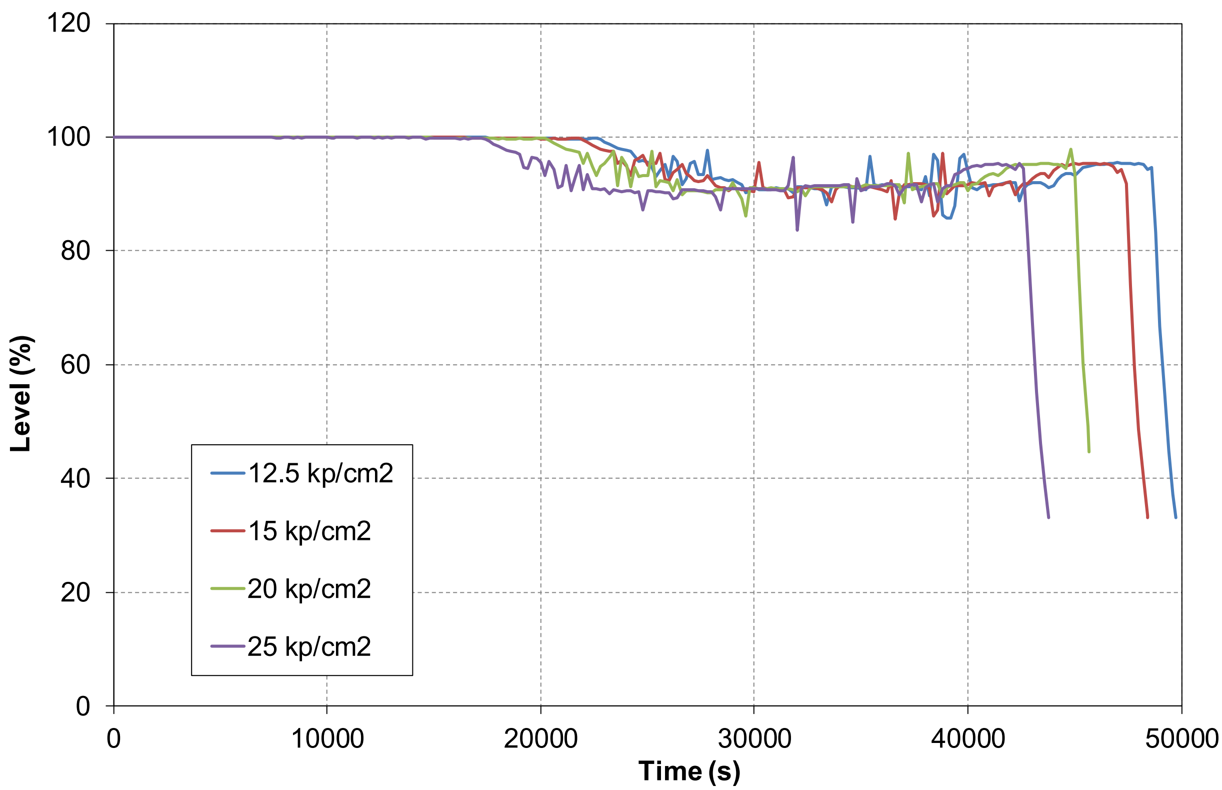

Figure 47 Core collapsed liquid level – scenario S1-21

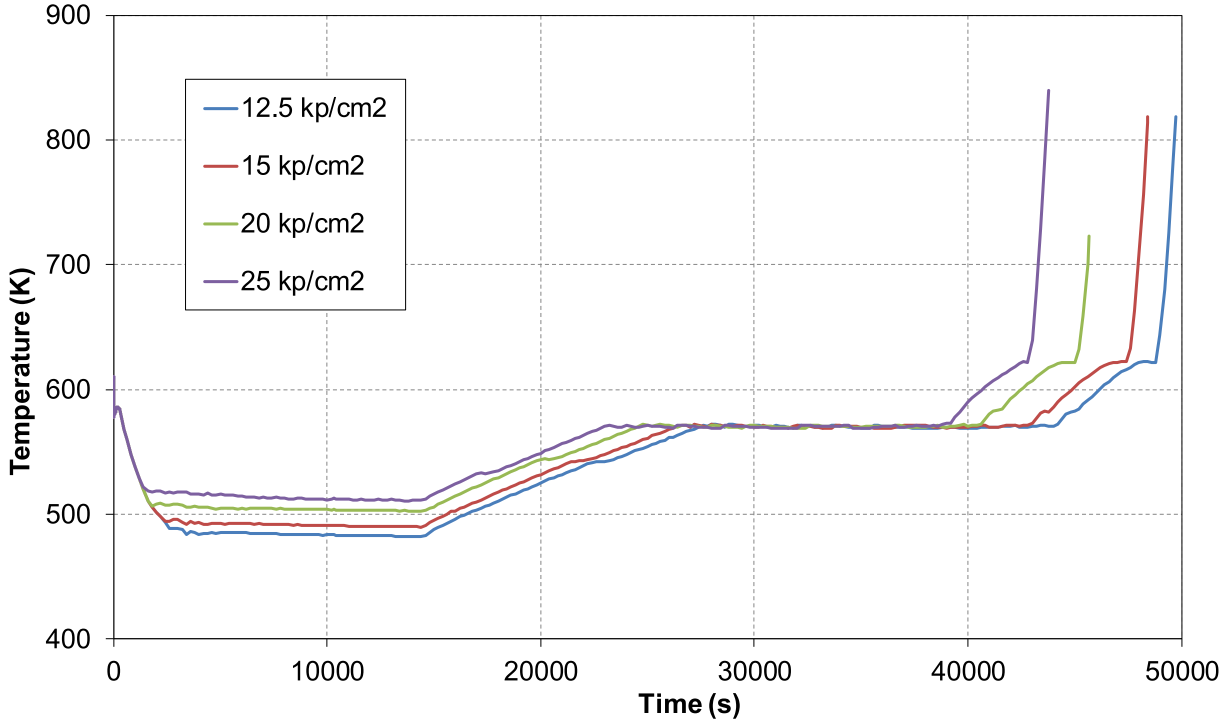

Figure 48 Average fuel cladding temperature – scenario S1-21

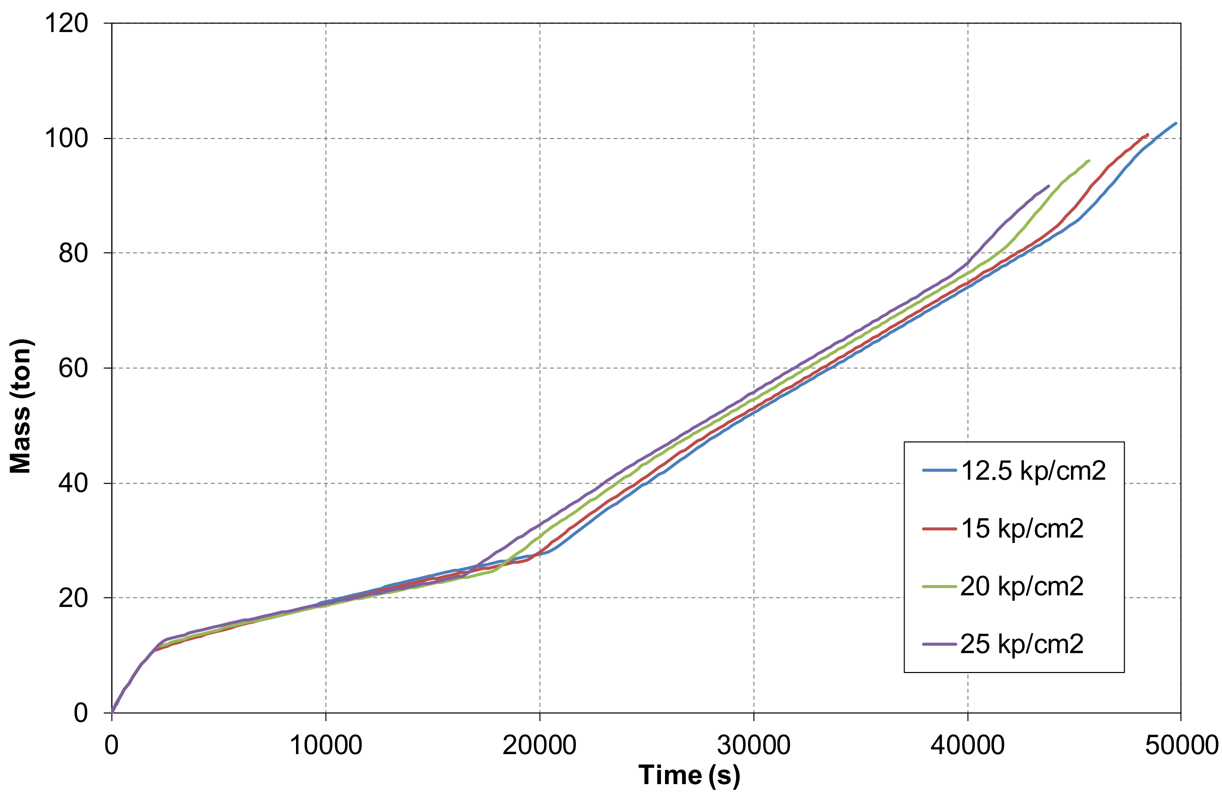

Figure 49 Total mass discharged from RCS – scenario S1-21

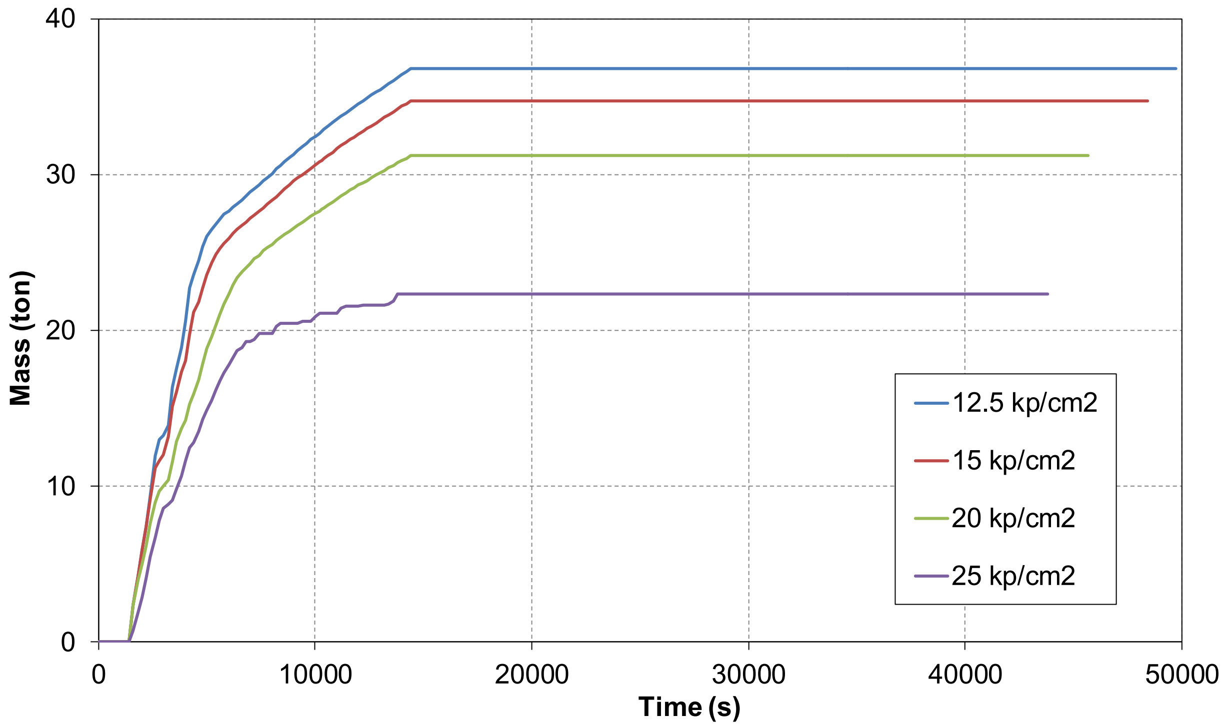

Figure 50 Mass injected into RCS – scenario S1-21

5.1.9 Scenario S1-21v1

The results for scenario S1-21v1, which differ from S1-21 scenario in that one pressurizer PORV is used arbitrarily 25000 s after transient start, are shown in Figures 51 through 56. It can be seen that RCS pressure (Figure 51) follows the secondary pressures until SGs as heat sink are lost. The steam generator pressures are maintained by operator until 14400 s. Later the TD AFW pump was assumed to be lost and SG PORVs were assumed not available. When setpoint was reached, the SG safety valves opened and provide cooling until SGs dry out. At that time RCS pressure increases again. PRZ PORV was used to limit the RCS pressure. Core exit temperature shown in Figure 52 increased after PRZ PORV opening causing core uncovery (Figure 53). From Figure 53 it can be seen, that due to the RCP leaks and lost TD AFW the core collapsed liquid levels decrease. However, significant core uncovery happens after PRZ PORV opening for all depressurization cases. Figure 54 shows the average fuel cladding temperature, which starts to increase when the core is significantly uncovered. Figure 55 shows mass release from RCS, which after 4 hours increases again due to lost heat sink. When PRZ PORV was opened, further large mass discharge happened. Finally, Figure 56 shows the mass injected by accumulators. After RCS system repressurization further injection by accumulators was prevented. The scenario clearly showed that after losing the heat sink after 4 hours on secondary side the plant heatup happened around one hour (independent on depressurization) after PRZ PORV is used, causing much discharge of RCS inventory.

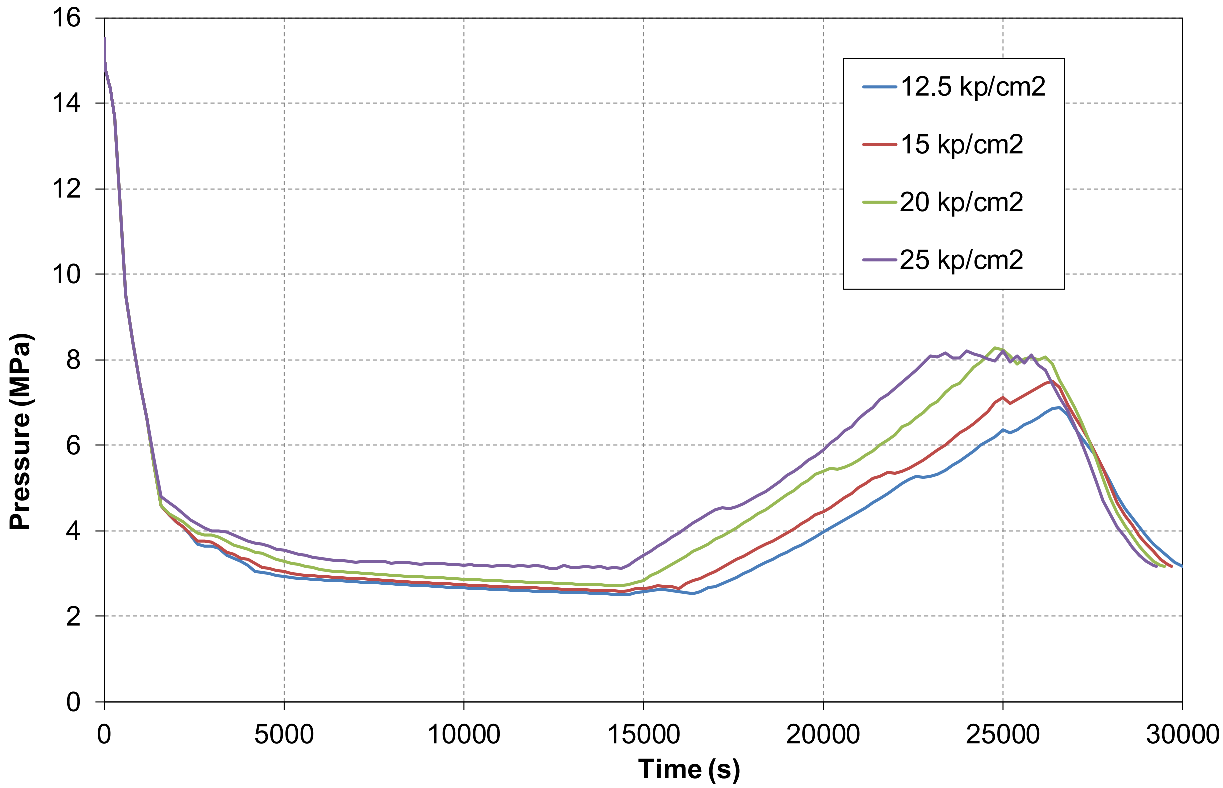

Figure 51 RCS pressure – scenario S1-21v1

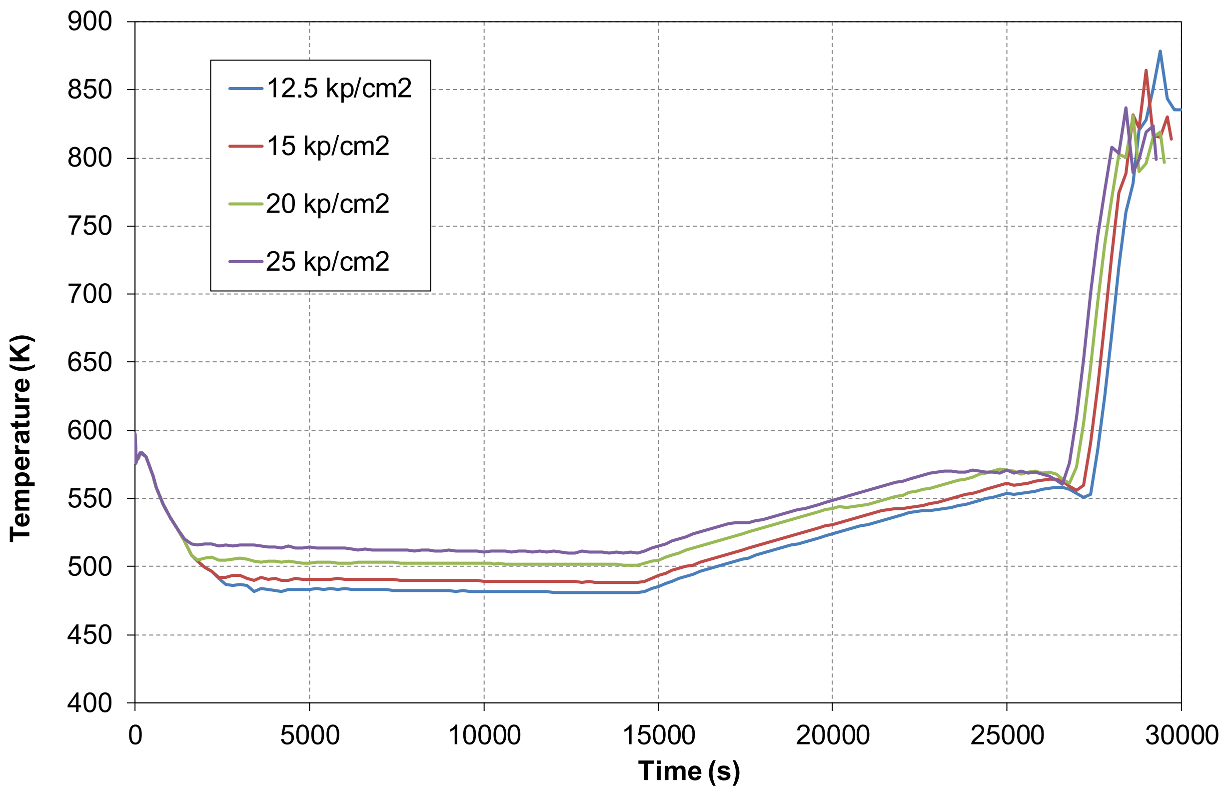

Figure 52 Core exit temperature – scenario S1-21v1

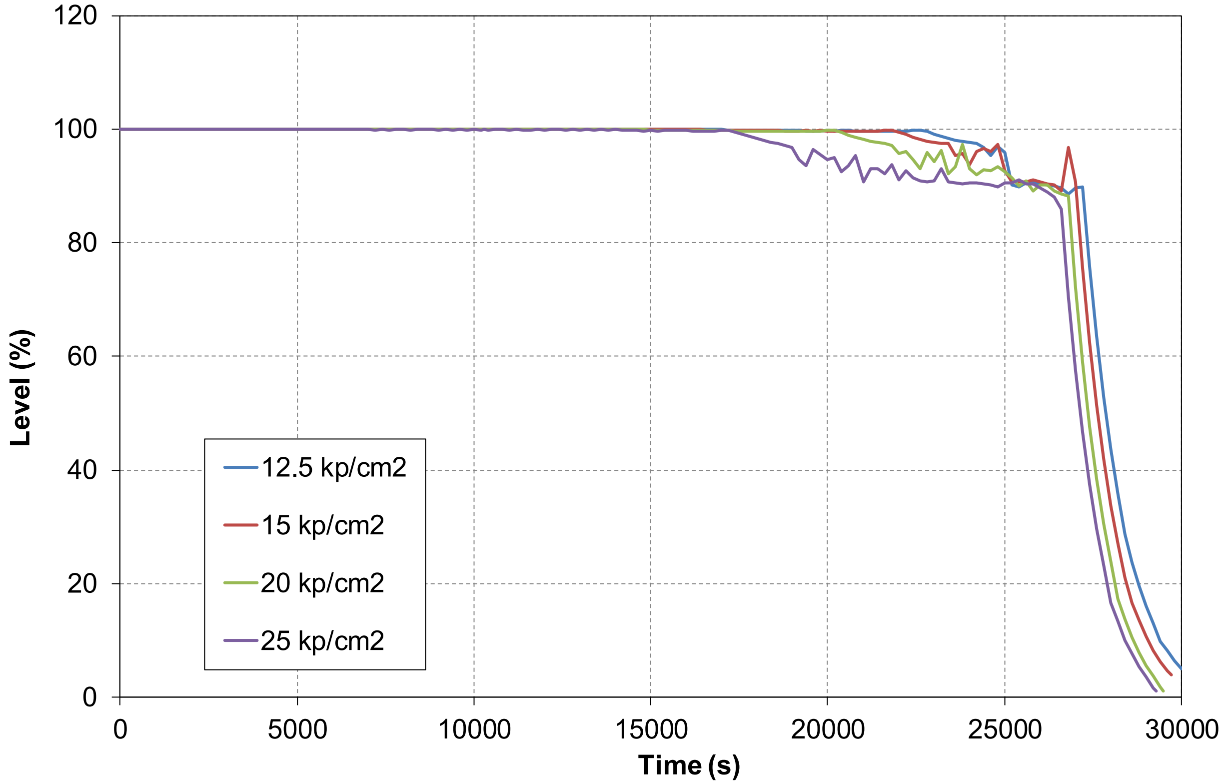

Figure 53 Core collapsed liquid level – scenario S1-21v1

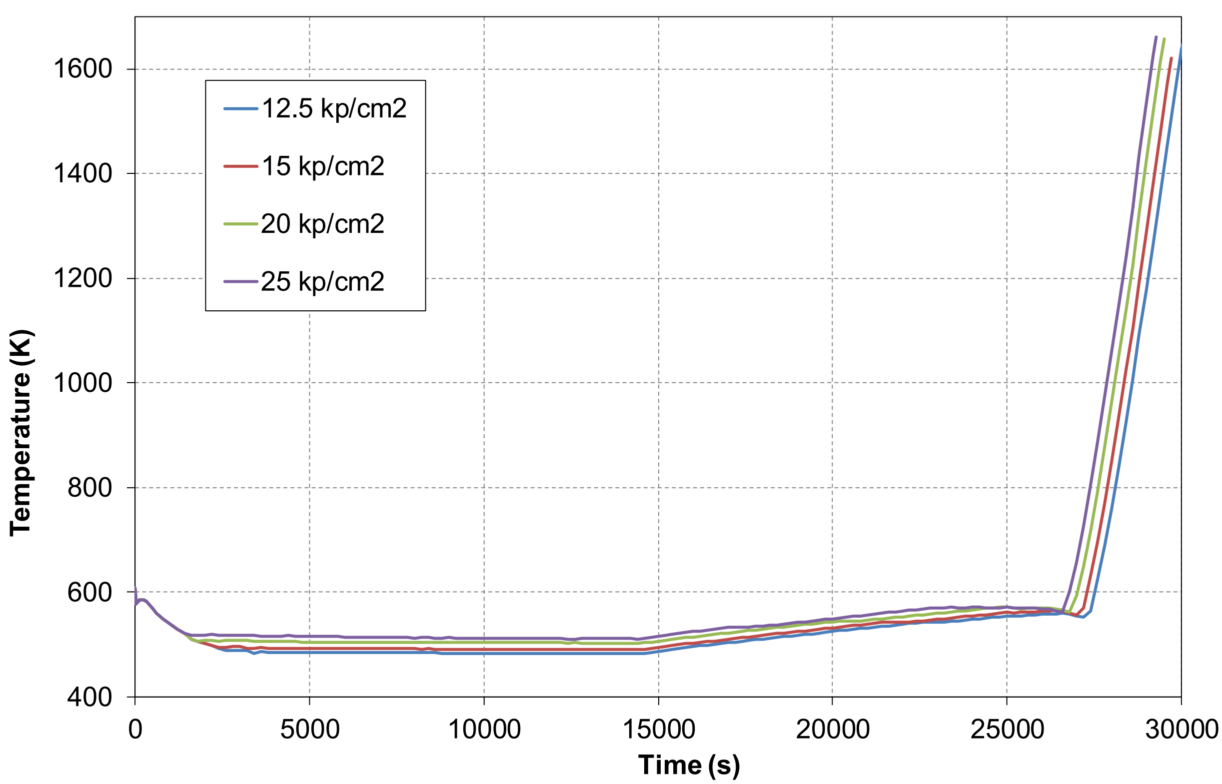

Figure 54 Average fuel cladding temperature – scenario S1-21v1

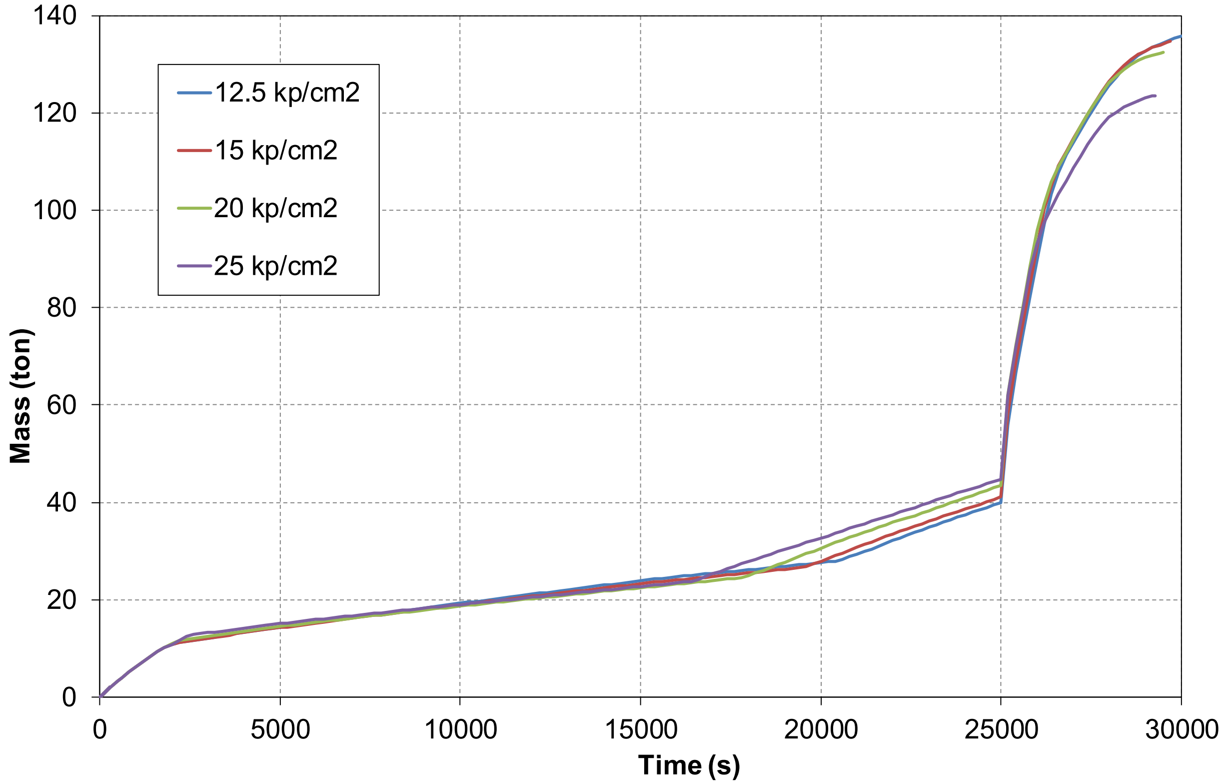

Figure 55 Total mass discharged from RCS – scenario S1-21v1

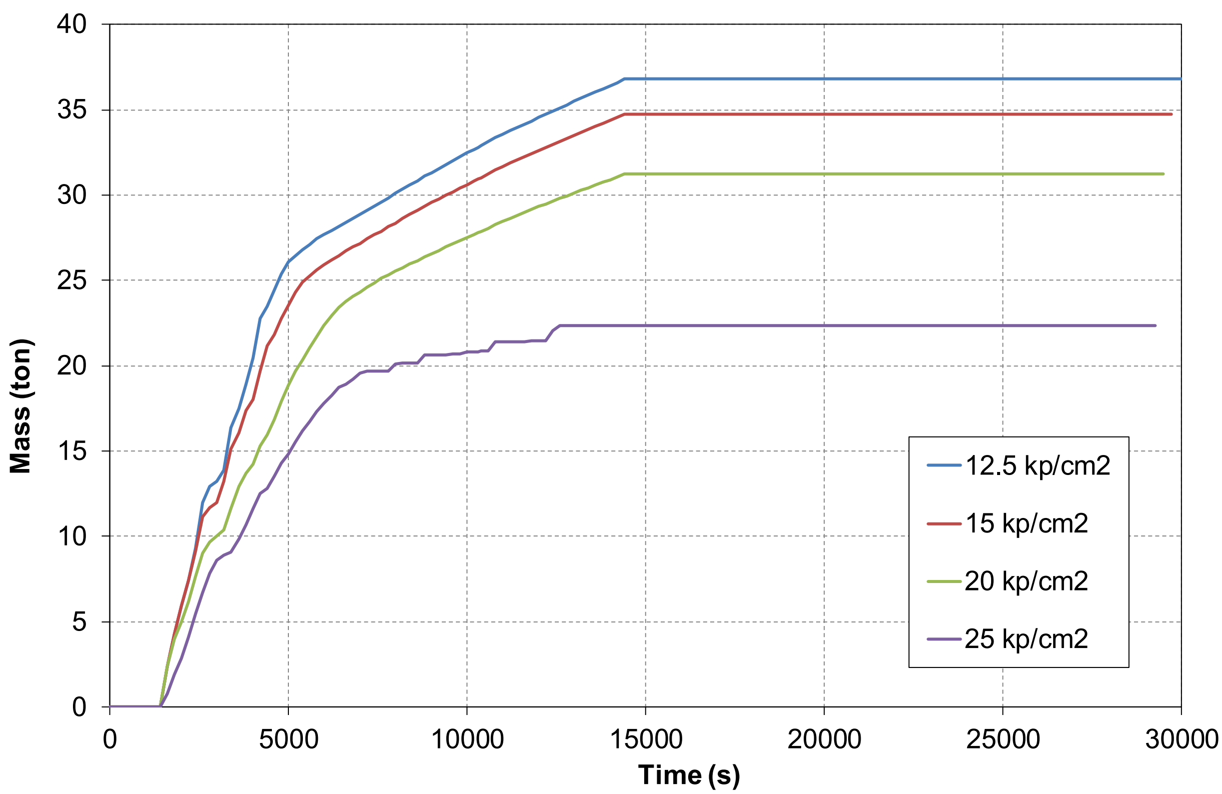

Figure 56 Mass injected into RCS – scenario S1-21v1

5.1.10 Scenario S1-21v2

The results for scenario S1-21v2, which differ from S1-21v1 scenario in that two pressurizer PORV are used instead of one in 25000 s after transient start, are shown in Figures 57 through 62. It can be seen that RCS pressure (Figure 57) follows the secondary pressures until SGs as heat sink are lost. The steam generator pressures are maintained by operator until 14400 s. Later the TD AFW pump was assumed to be lost and SG PORVs were assumed not available. When setpoint was reached, the SG safety valves opened and provide cooling until SGs dry out. At that time RCS pressure increases again. PRZ PORVs were used to limit the RCS pressure. Core exit temperature shown in Figure 58 increased after PRZ PORVs opening causing core uncovery (Figure 59). From Figure 59 it can be seen, that due to the RCP leaks and lost TD AFW the core collapsed liquid levels start to decrease. However, significant core uncovery happens after PRZ PORVs opening for all depressurization cases. Figure 60 shows the average fuel cladding temperature, which starts to increase when the core is significantly uncovered soon after PRZ PORVs opening. Figure 61 shows mass discharge from RCS, which after 4 hours increases again due to lost heat sink. When PRZ PORVs were opened, further large mass discharge happened. Finally, Figure 62 shows the mass injected by accumulators. After RCS system repressurization further injection by accumulators was prevented. The scenario clearly showed that after losing the heat sink after 4 hours on secondary side the plant heatup happened around one hour (independent on depressurization) after PRZ PORVs are used, causing much release of RCS inventory.

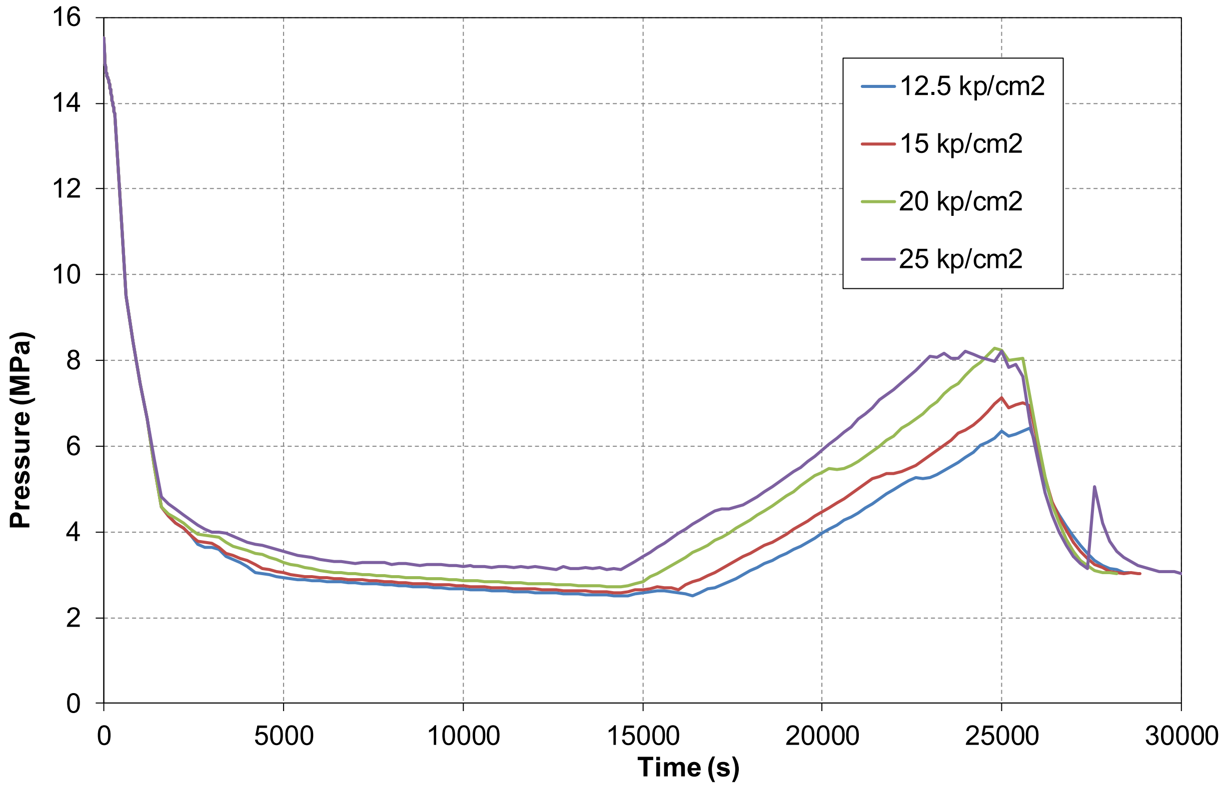

Figure 57 RCS pressure – scenario S1-21v2

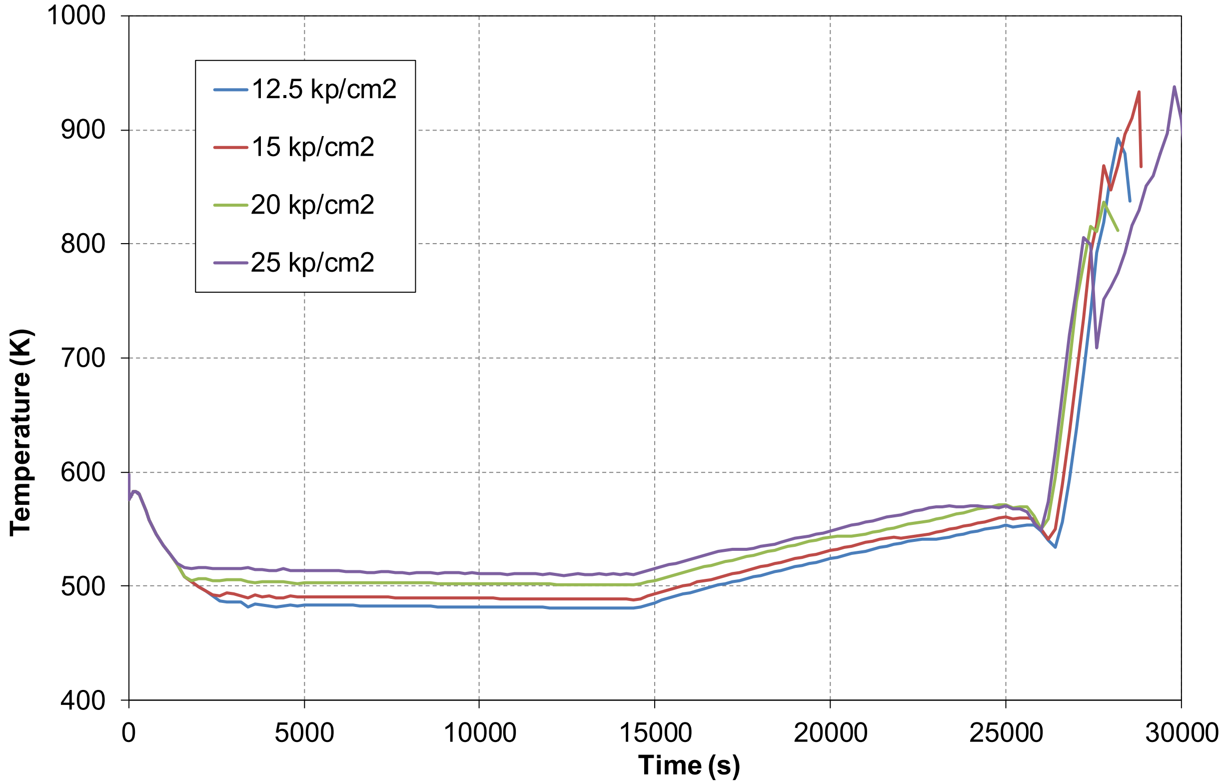

Figure 58 Core exit temperature – scenario S1-21v2

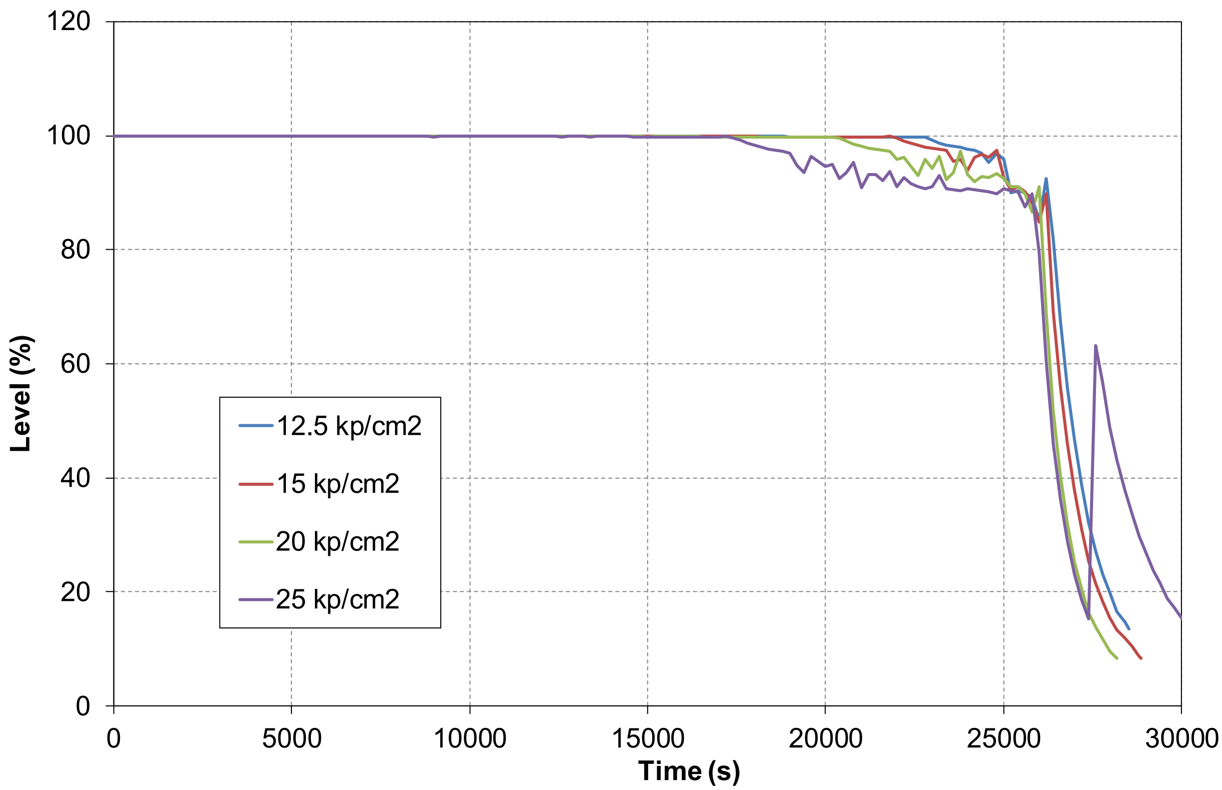

Figure 59 Core collapsed liquid level – scenario S1-21v2

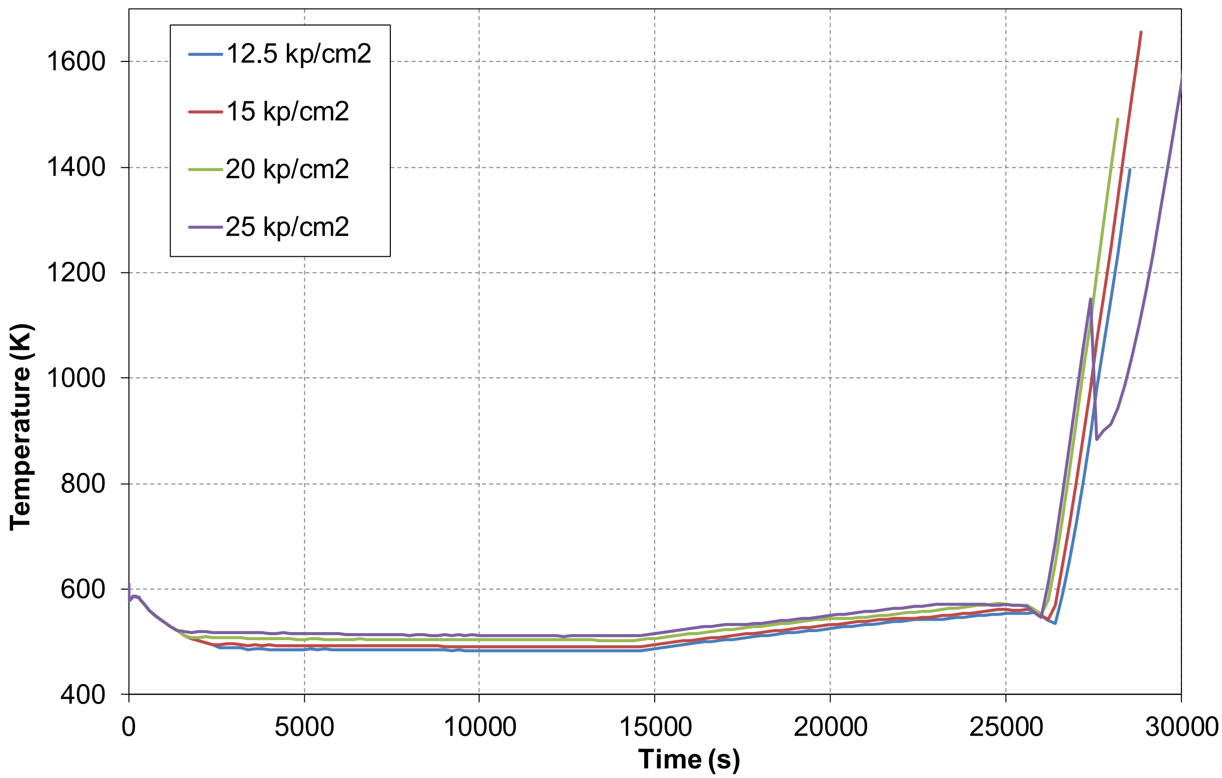

Figure 60 Average fuel cladding temperature – scenario S1-21v2

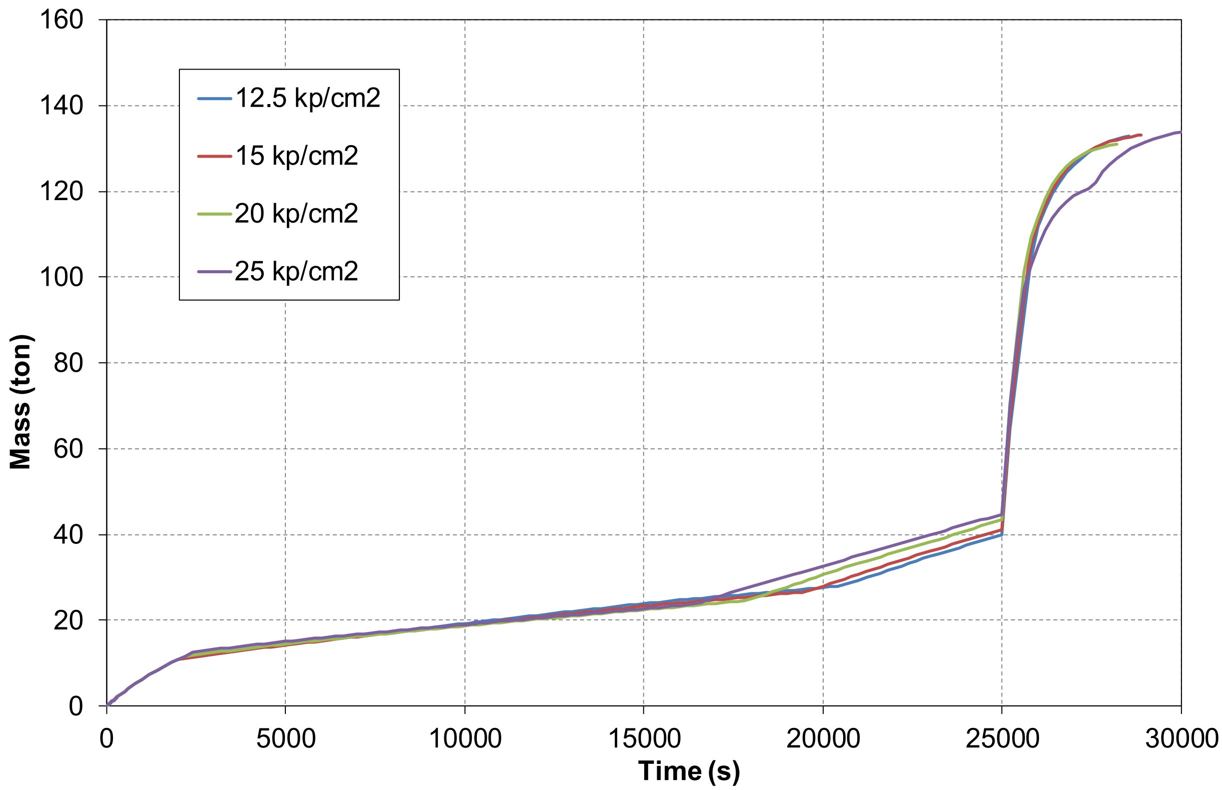

Figure 61 Total mass discharged from RCS – scenario S1-21v2

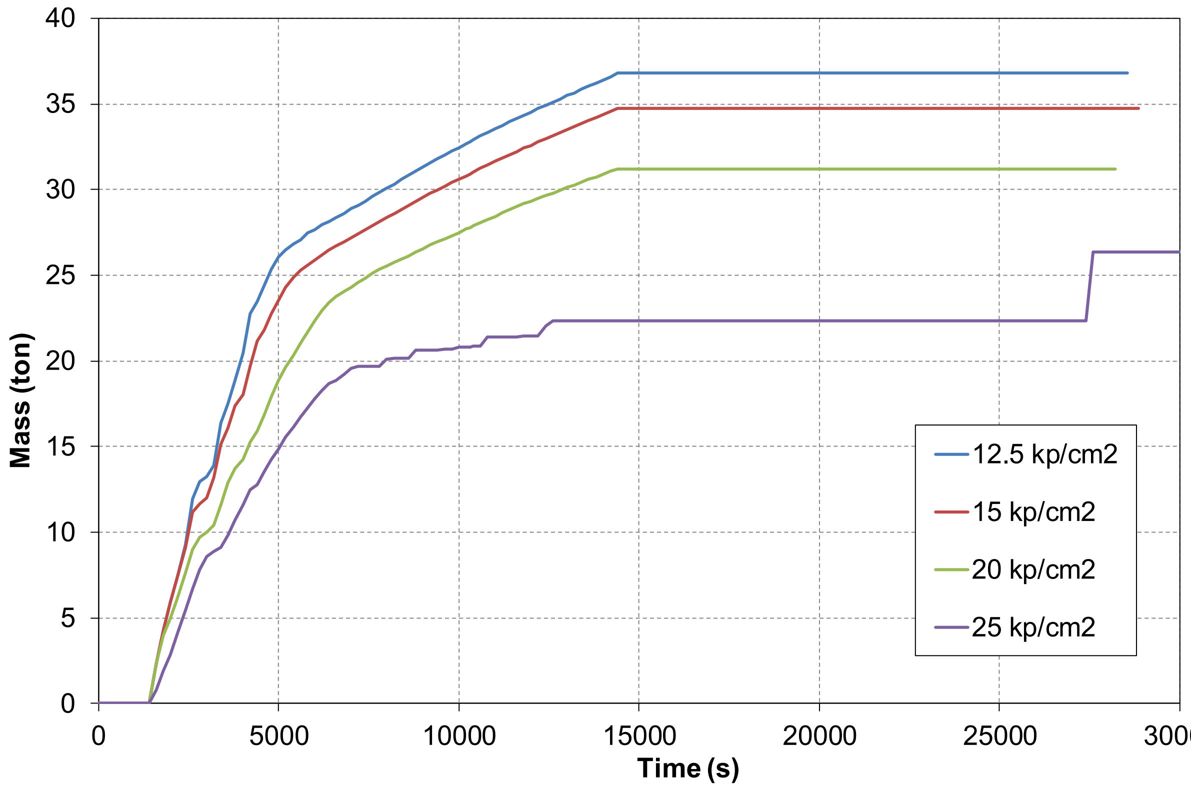

Figure 62 Mass injected into RCS – scenario S1-21v2

5.1.11 Scenario S1-21p

The results for scenario S1-21p, which differ from S1-21 scenario in that PDP charging pump is used 4 hours after transient start, when TD AFW is lost, are shown in Figures 63 through 68.

It can be seen that RCS pressure (Figure 63) follows the secondary pressures until SGs as heat sink are lost. The steam generator pressures are maintained by operator until 14400 s. Later the TD AFW pump was assumed to be lost and SG PORVs were assumed not available. Core exit temperature shown in Figure 64 increased after core heatup. Figure 65 shows the core collapsed liquid level. It can be seen, that after decreasing RCS inventory also the core collapsed liquid levels decrease. However, significant core uncovery happens after 50000 s for all depressurization cases due to primary pressure increase causing automatic PRZ relief valve opening. Figure 66 shows the average fuel cladding temperature, which starts to increase after the core is significantly uncovered as result of PRZ relief valve opening.
Figure 67 shows mass release from RCS, which after 4 hours increases again due to lost heat sink. When PRZ relief valve was opened, further large mass discharge happened. Finally, Figure 68 shows the mass injected by accumulators and later by PDP charging pump. After RCS system repressurization further injection by accumulators was prevented. The scenario clearly showed that after losing the heat sink after 4 hours on secondary side the plant can survive at least 10 hours (depending on depressurization) as some cooling is provided also by RCP seal leaks. Comparing to base case the operation of PDP prolongs the time with core uncovery for few hours. This example clearly showed that TD AFW operation is needed to provide long term cooling, if RCS makeup is too small as in our case.

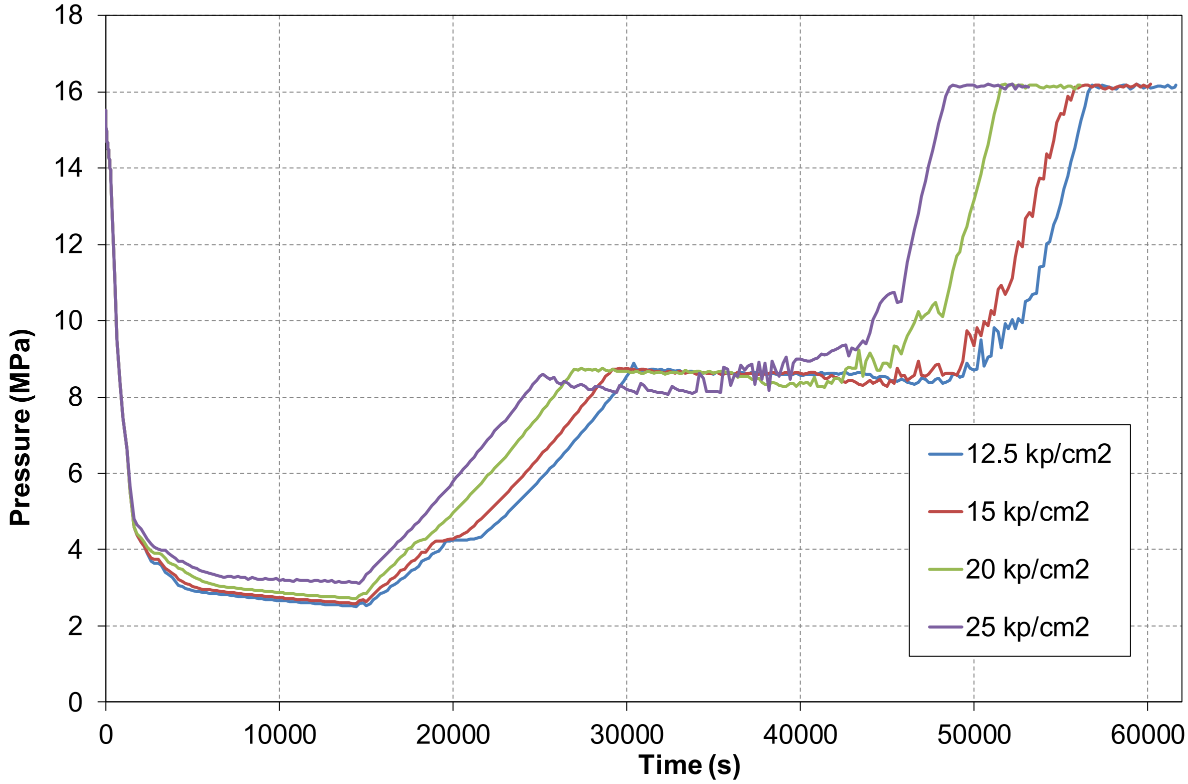

Figure 63 RCS pressure – scenario S1-21p

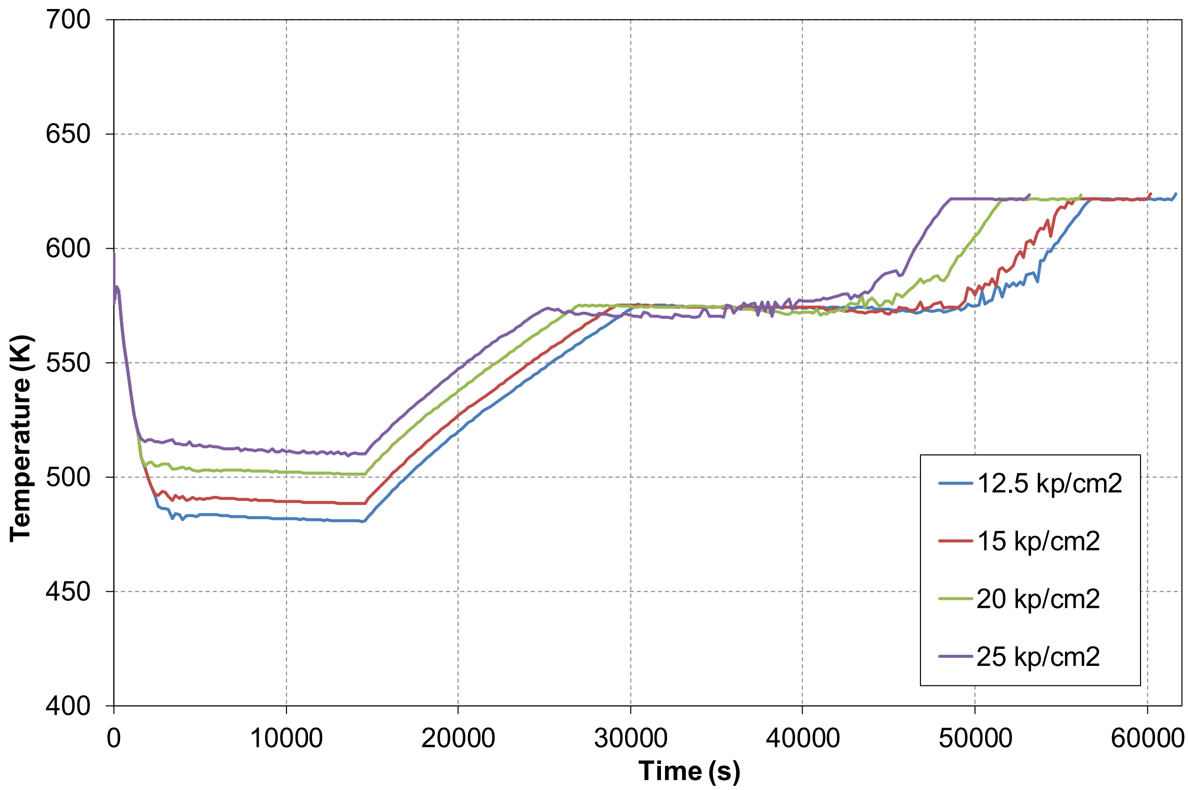

Figure 64 Core exit temperature – scenario S1-21p

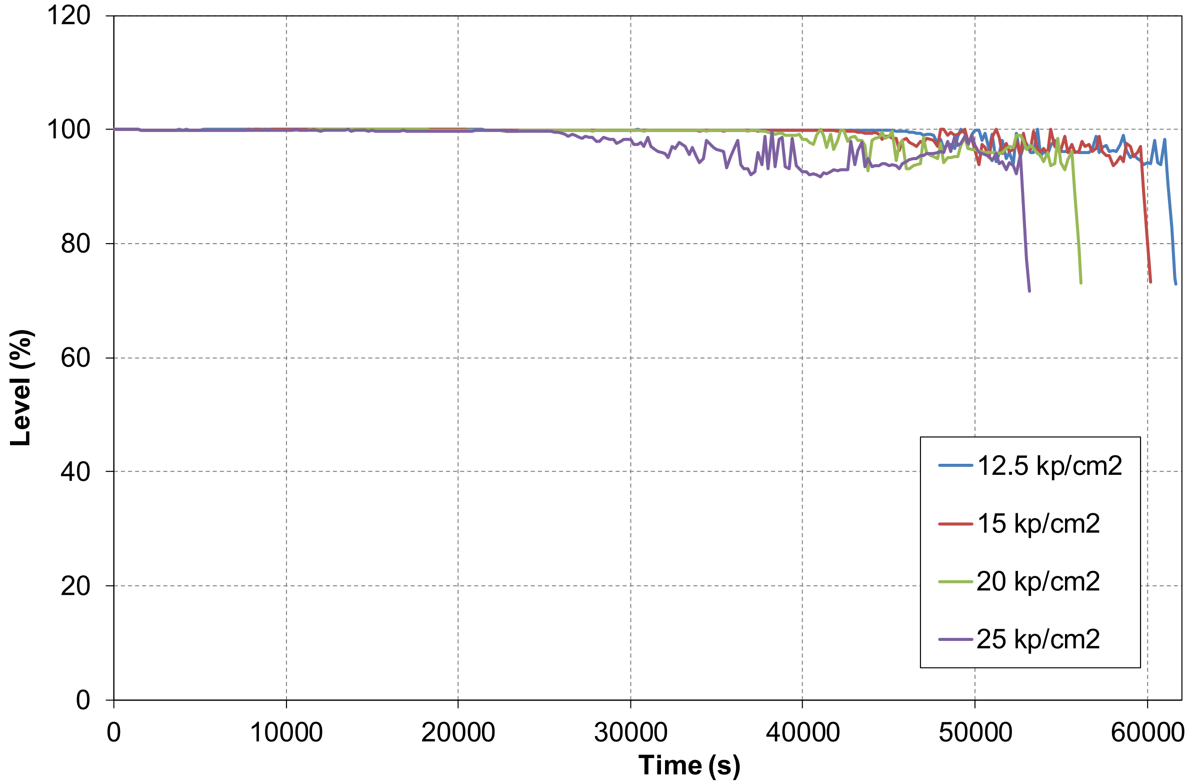

Figure 65 Core level – scenario S1-21p

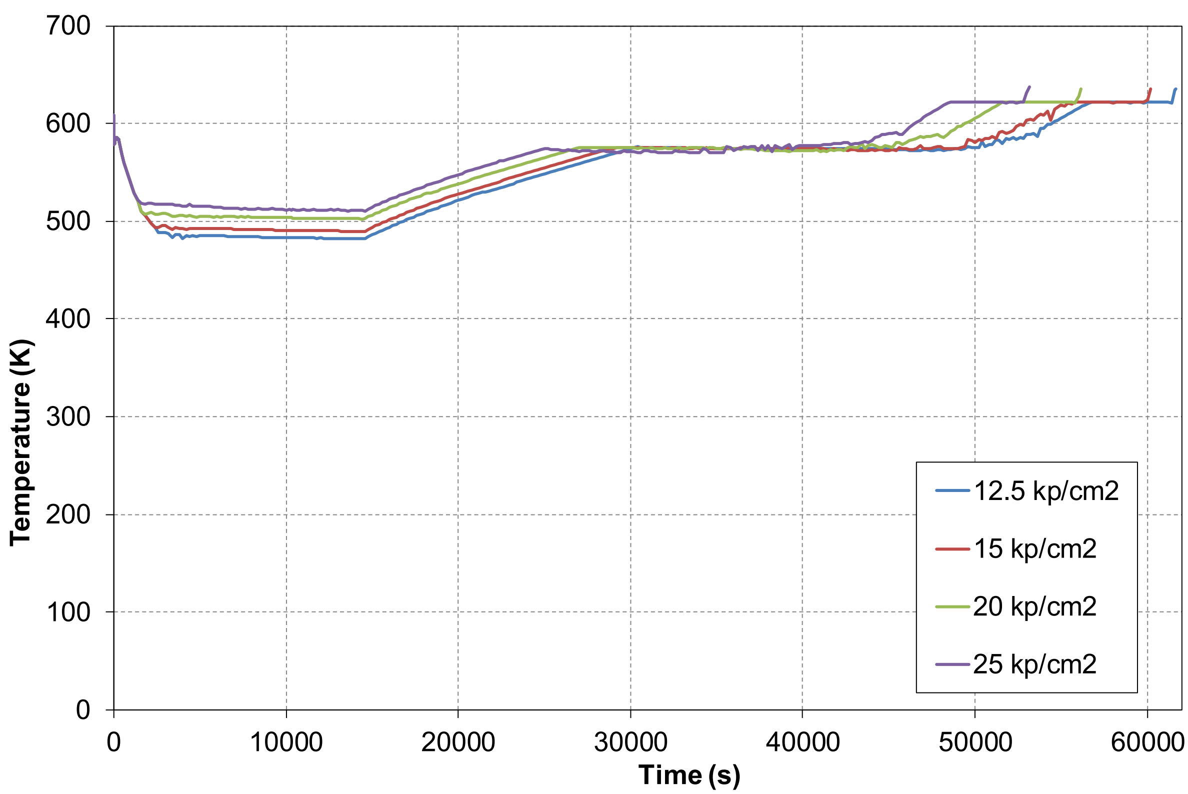

Figure 66 Average fuel cladding temperature – scenario S1-21p

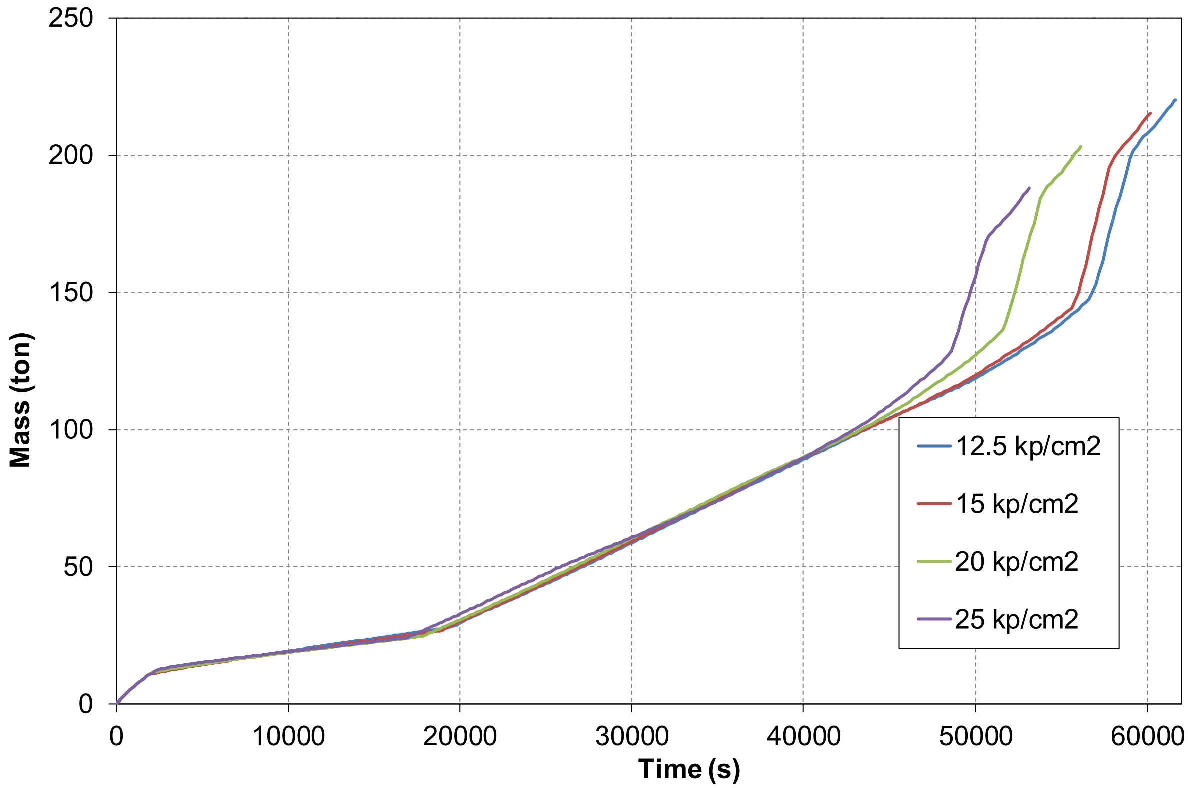

Figure 67 Total mass discharged from RCS – scenario S1-21p

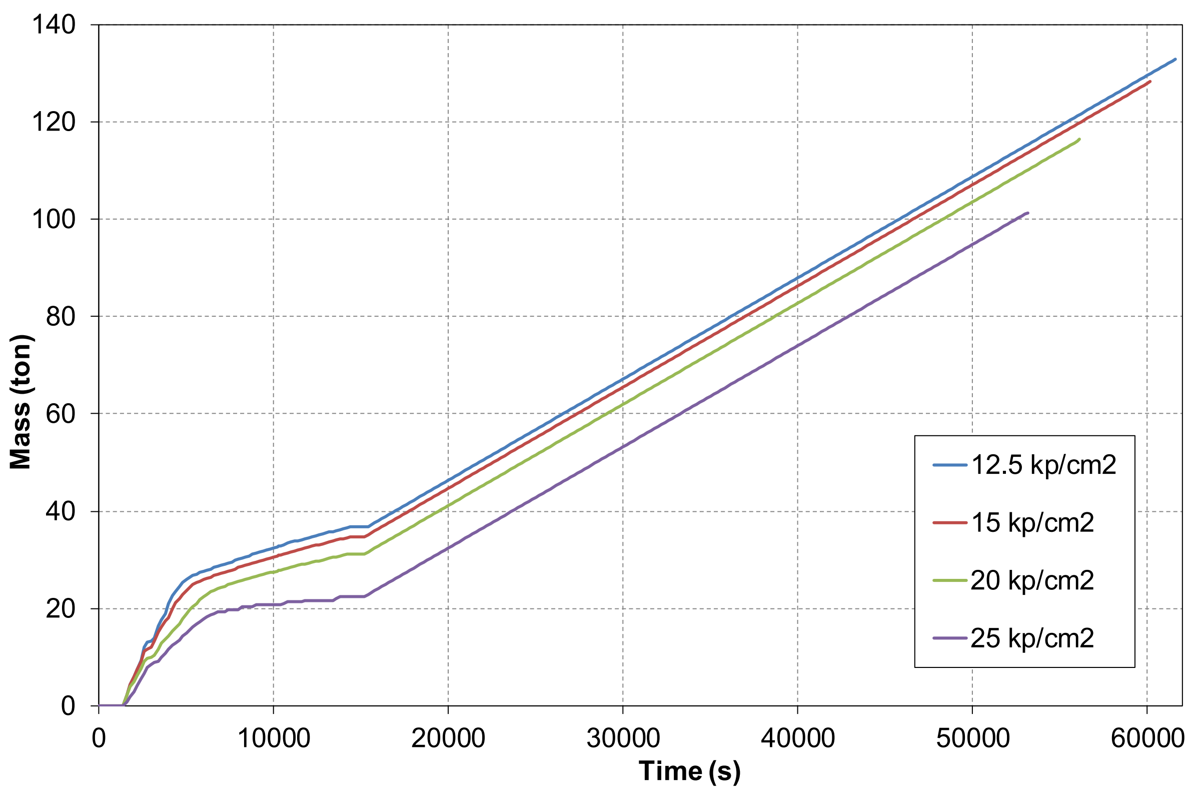

Figure 68 Mass injected into RCS – scenario S1-21p

5.2 Dependence on RCPs seal leak of SBO scenarios with TD AFW available

For each set of S0 scenarios (consisting of six different leaks) on depressurization pressure, the following five variables are shown: RCS pressure, SG no. 1 pressure, core collapsed liquid level, average fuel cladding temperature, and RCS inventory. As the variables have already been plotted for all cases, Figures 69 through 88 will not be described in detail. Rather, some remarks will be made. Please note, that SG no. 1 pressure and RCS mass inventory have not been plotted in Section 5.1.

5.2.1 S0 scenarios depressurized to 1.33 MPa (12.5 kp/cm^2 gauge)

The results of S0 scenarios, in which the operator depressurizes and maintains SG pressure at 1.33 MPa are shown in Figures 69 through 73. The larger the break is, the shorter the time available before core uncovery and core heatup is.

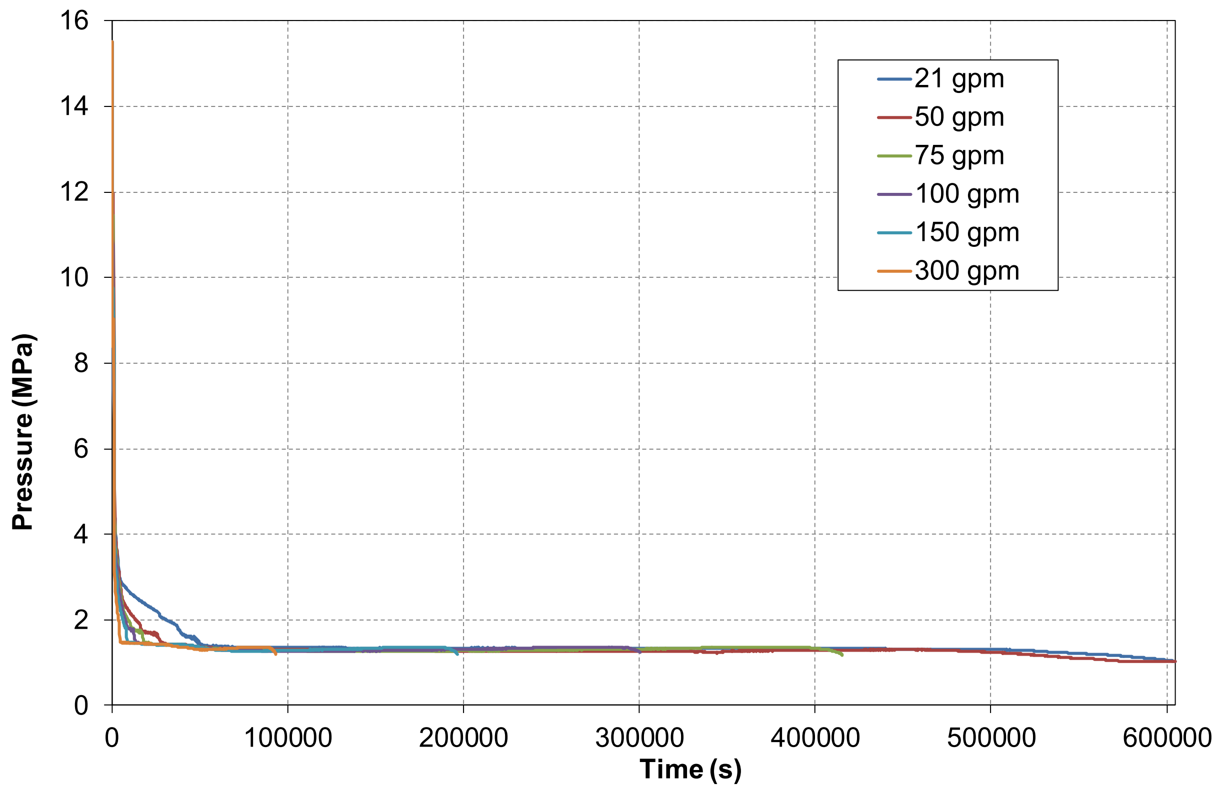

Figure 69 RCS pressure – S0 scenarios (1.33 MPa)

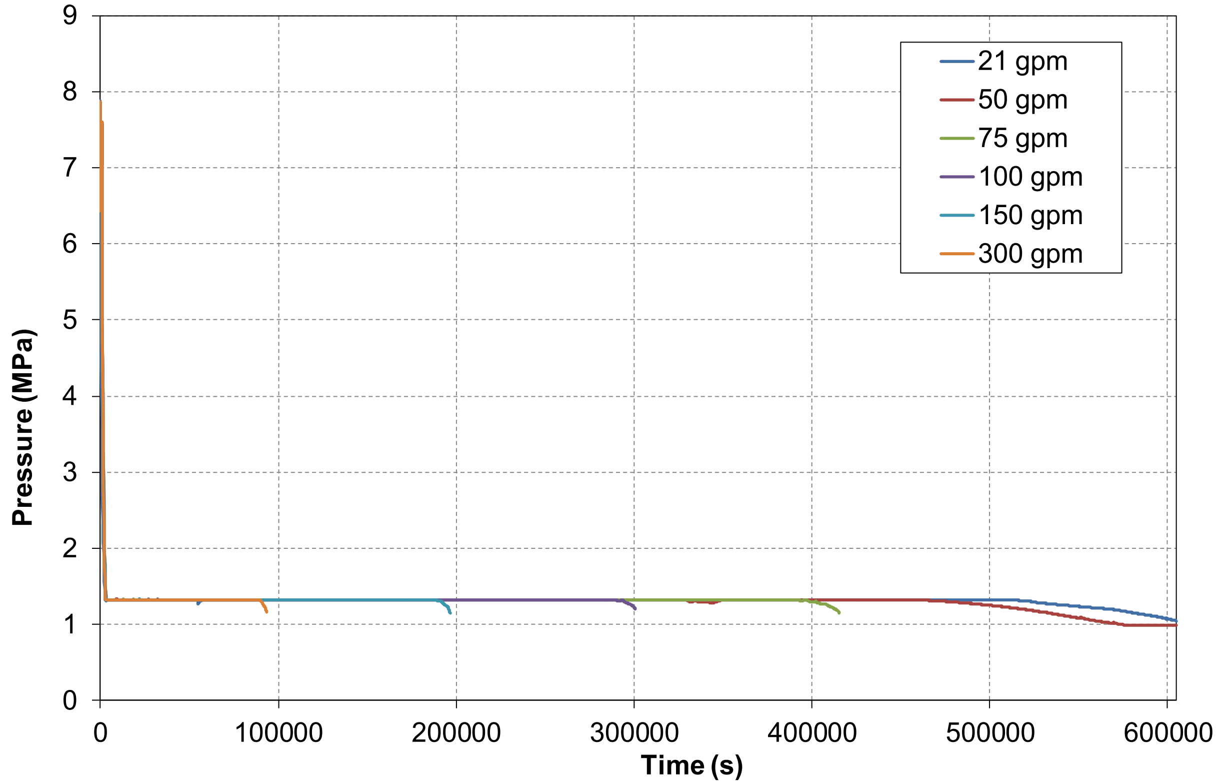

Figure 70 SG no.1 pressure – S0 scenarios (1.33 MPa)

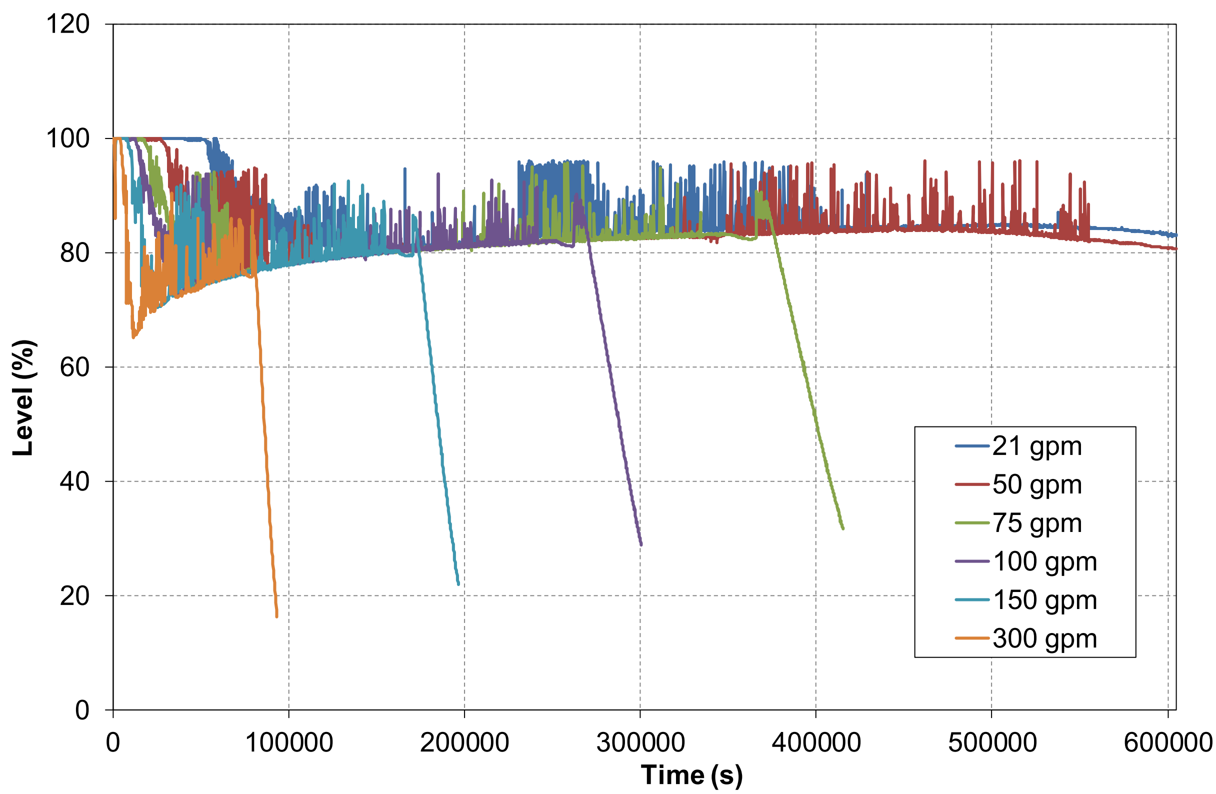

Figure 71 Core collapsed liquid level – S0 scenarios (1.33 MPa)

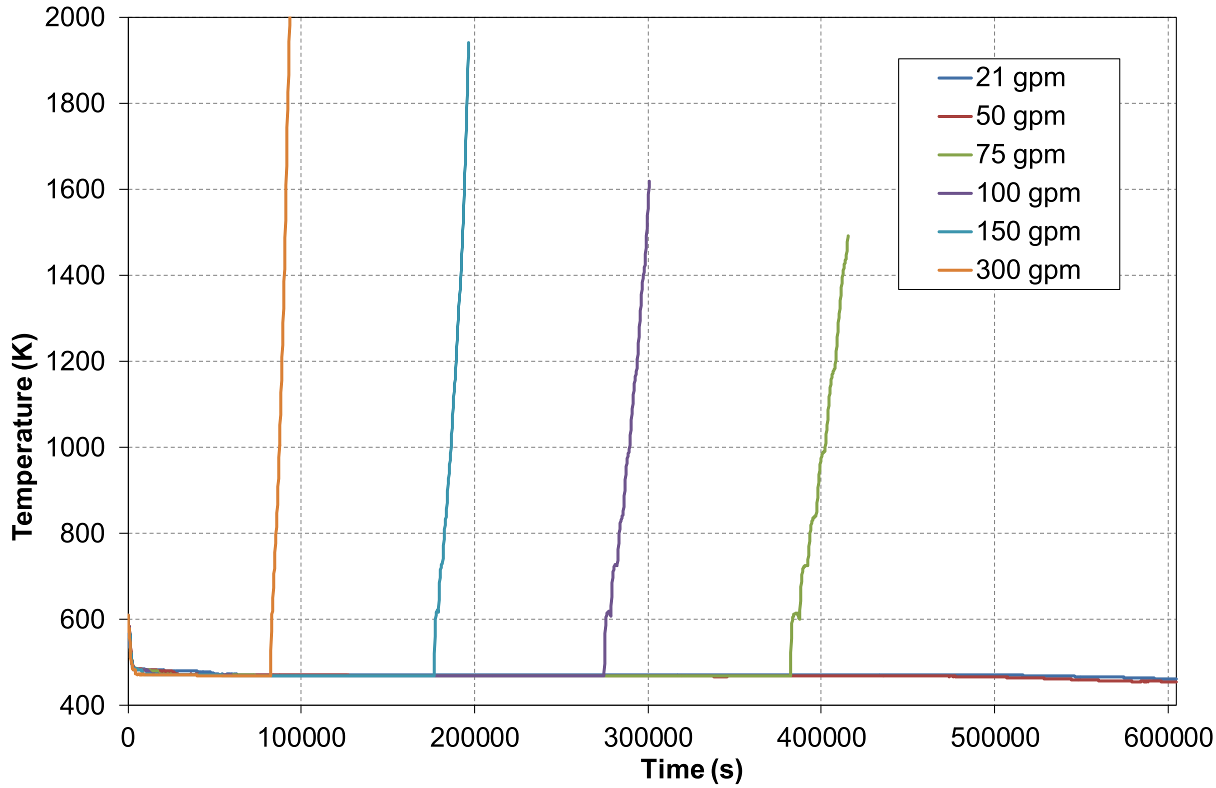

Figure 72 Average fuel cladding temperature – S0 scenarios (1.33 MPa)

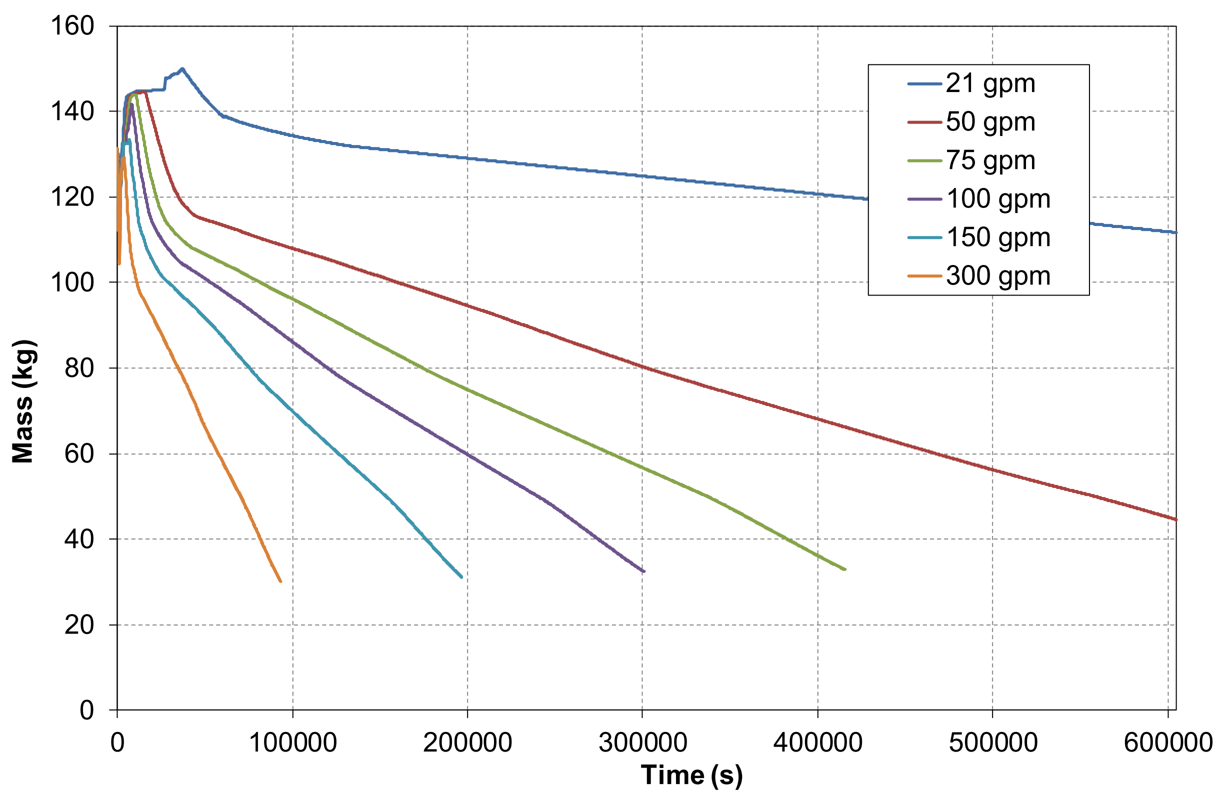

Figure 73 RCS mass inventory – S0 scenarios (1.33 MPa)

5.2.2 S0 scenarios depressurized to 1.57 MPa (15 kp/cm^2 gauge)

The results scenarios S0, in which the operator depressurizes and maintains SG pressure at 1.57 MPa are shown in Figures 74 through 78. The larger the break is, the shorter the time available before core uncovery and core heatup is.

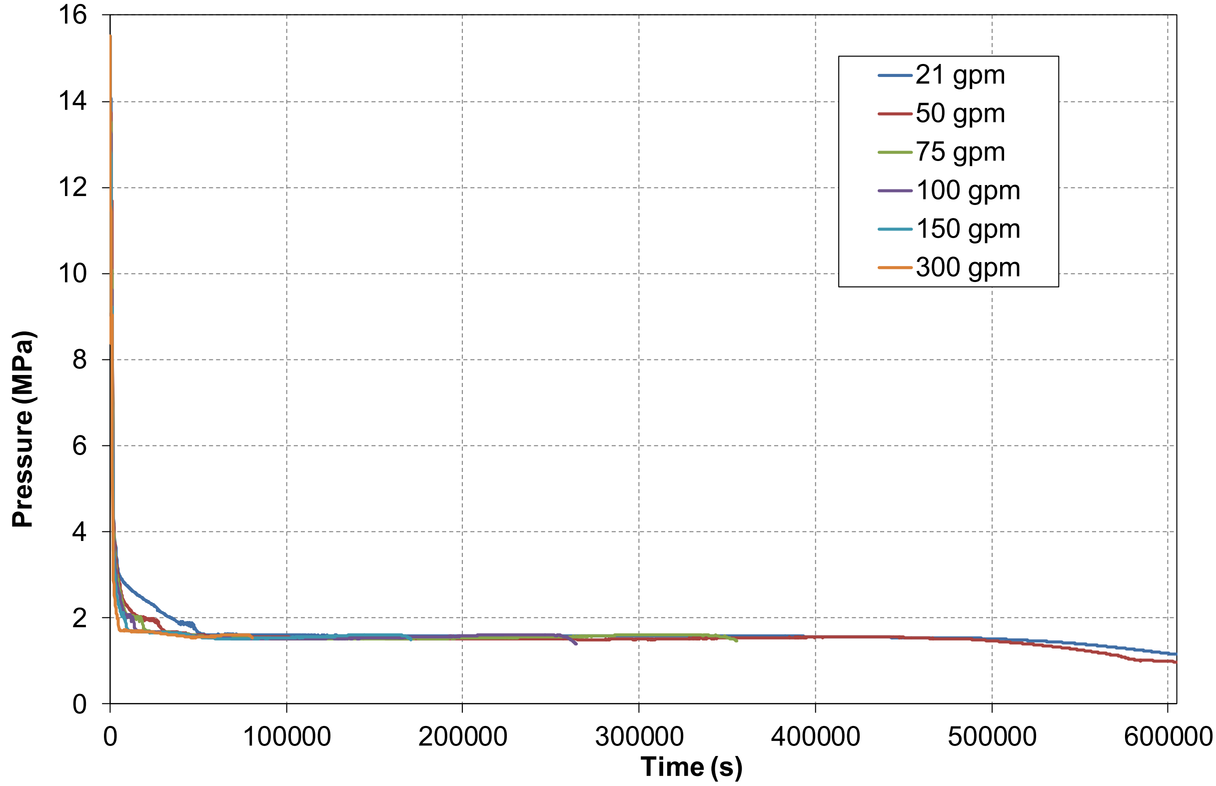

Figure 74 RCS pressure – S0 scenarios (1.57 MPa)

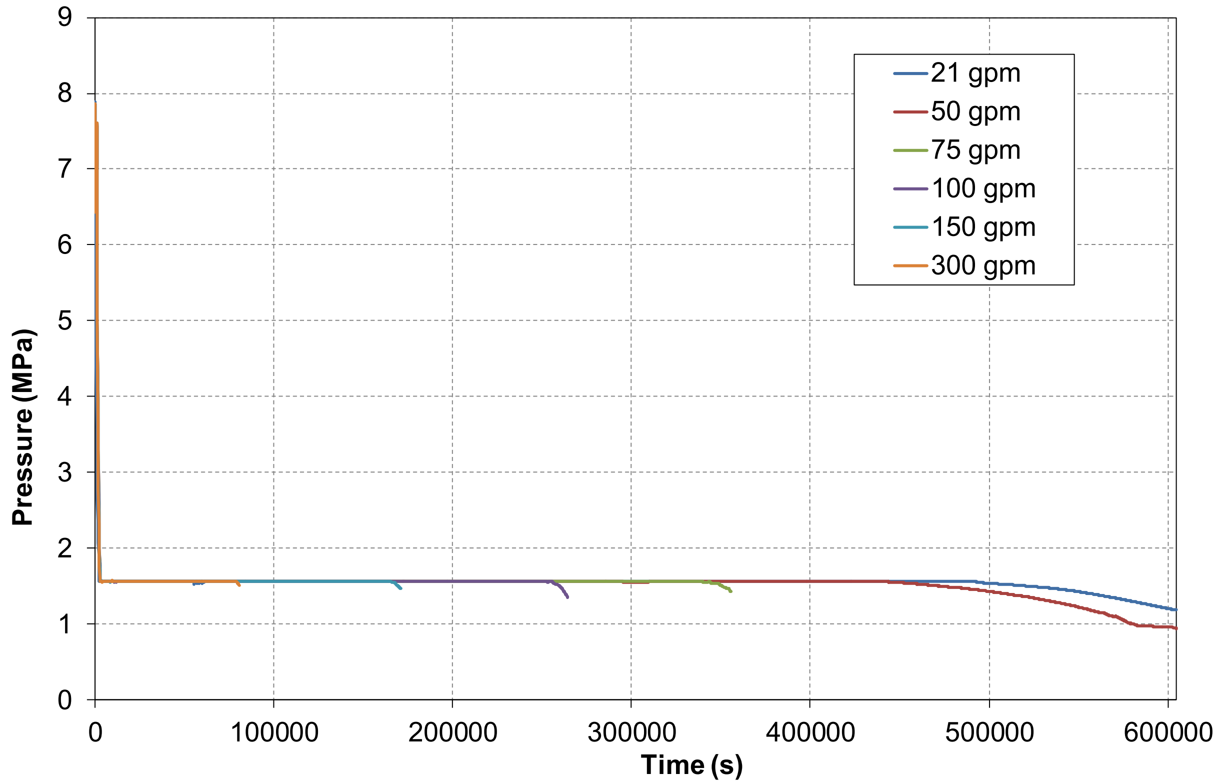

Figure 75 SG no.1 pressure – S0 scenarios (1.57 MPa)

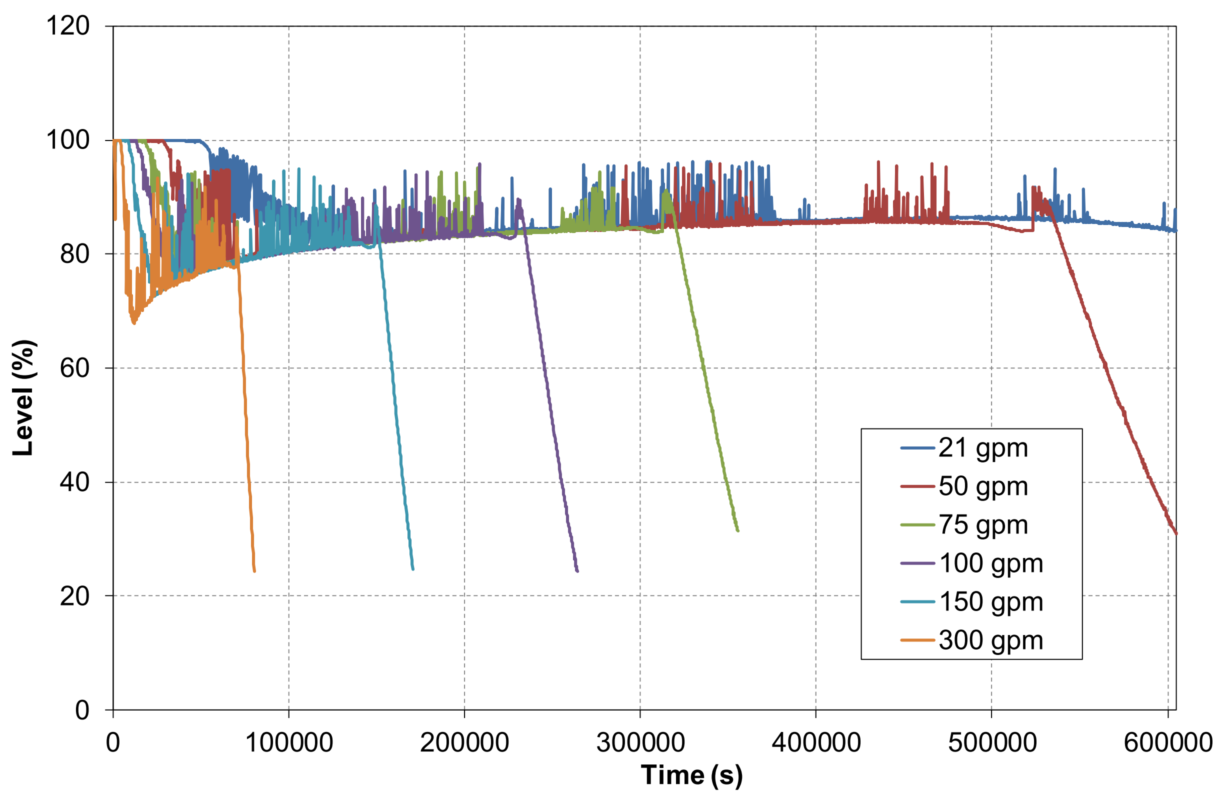

Figure 76 Core collapsed liquid level – S0 scenarios (1.57 MPa)

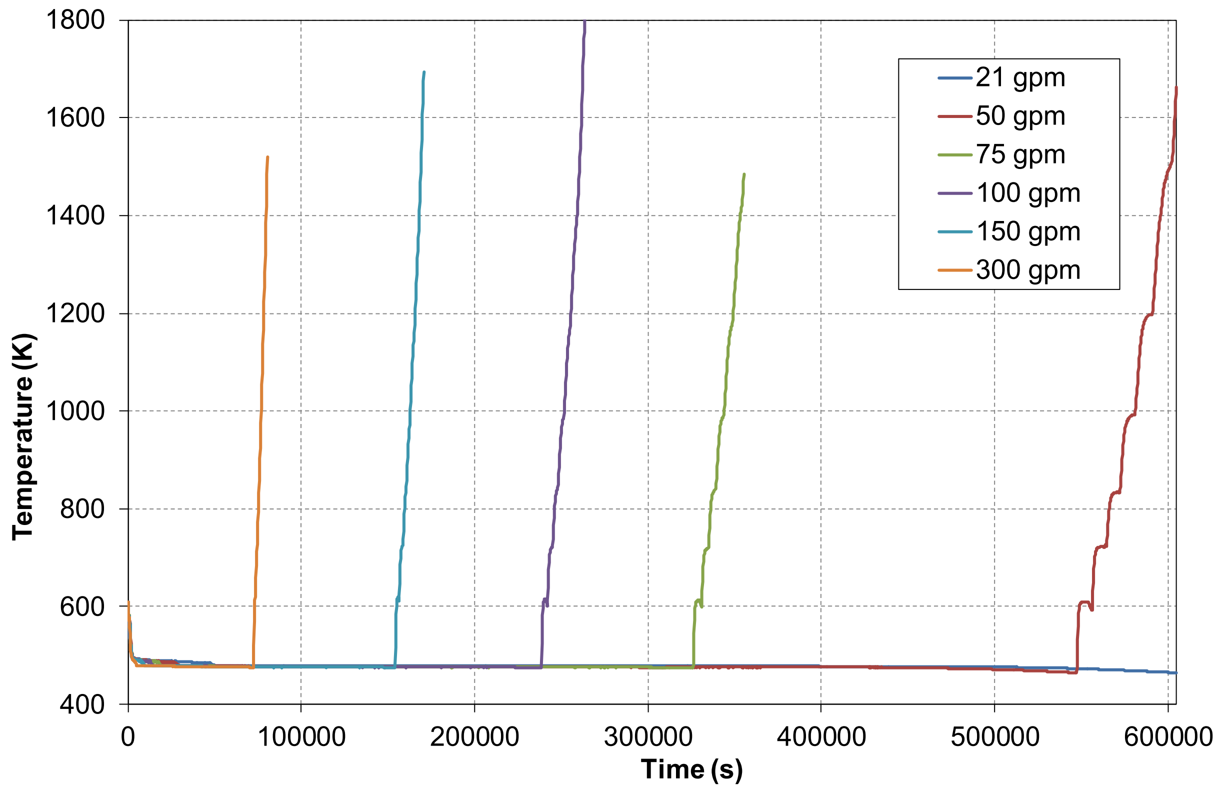

Figure 77 Average fuel cladding temperature – S0 scenarios (1.57 MPa)

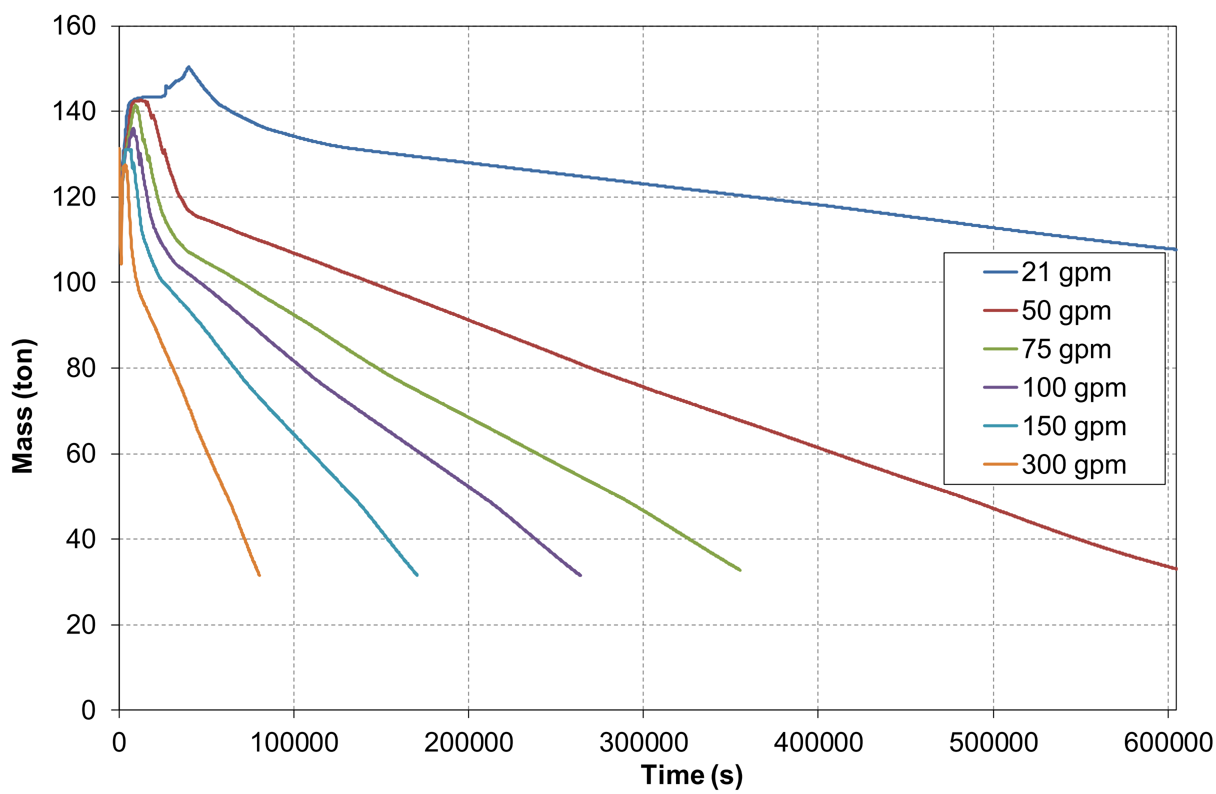

Figure 78 RCS mass inventory – S0 scenarios (1.57 MPa)

5.2.3 S0 scenarios depressurized to 2.06 MPa (20 kp/cm^2 gauge)

The results scenarios S0, in which the operator depressurizes and maintains SG pressure at 2.06 MPa are shown in Figures 79 through 83. The larger the break is, the shorter the time available before core uncovery and core heatup is.

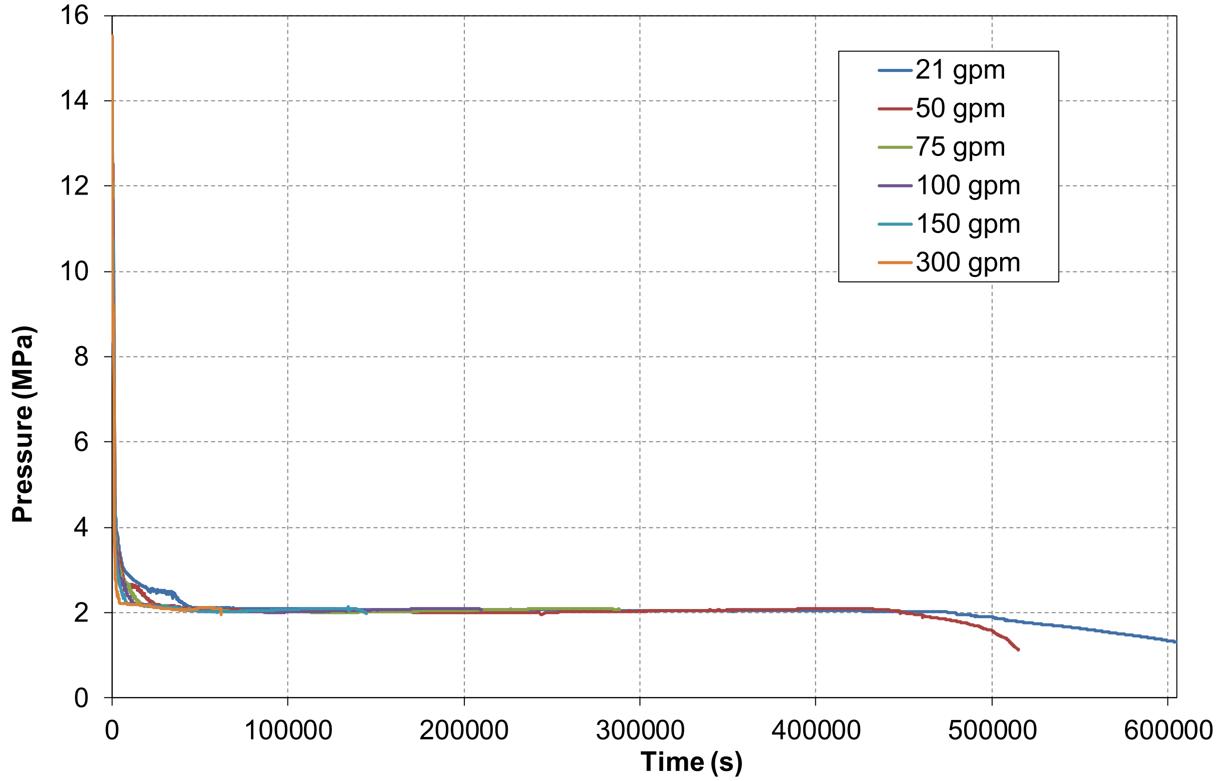

Figure 79 RCS pressure – S0 scenarios (2.06 MPa)

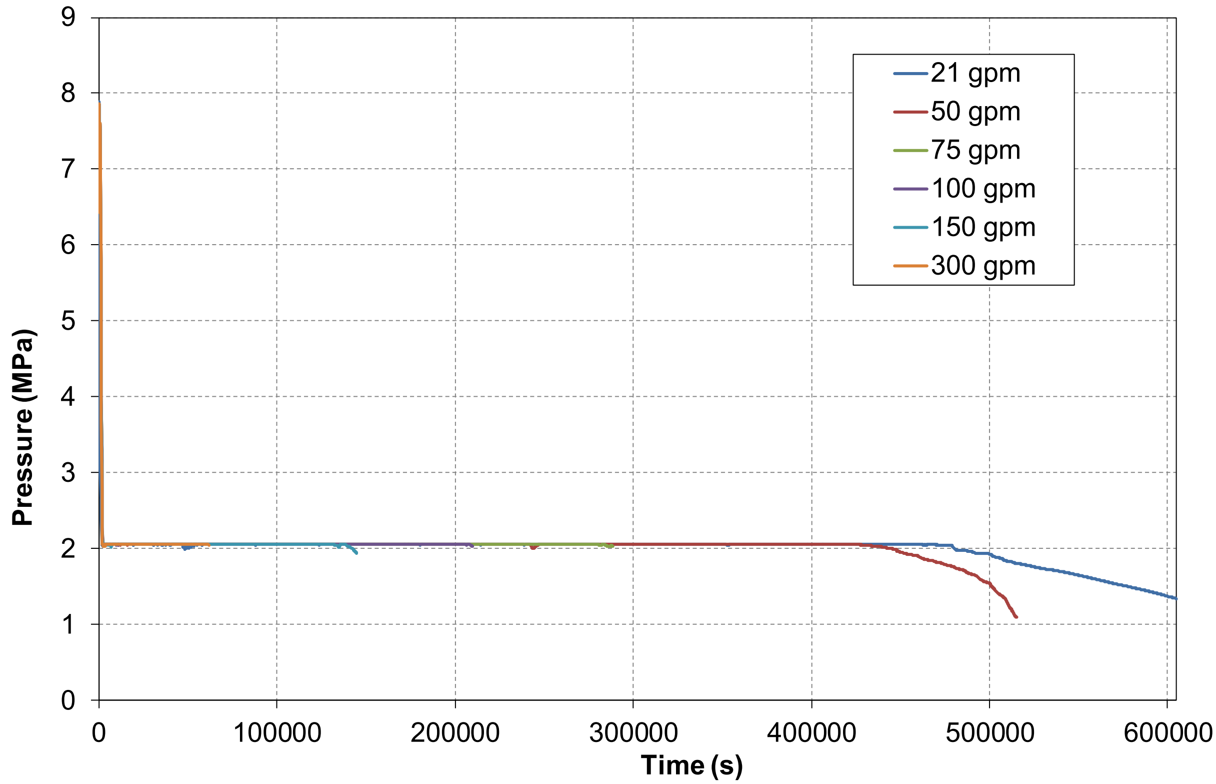

Figure 80 SG no.1 pressure – S0 scenarios (2.06 MPa)

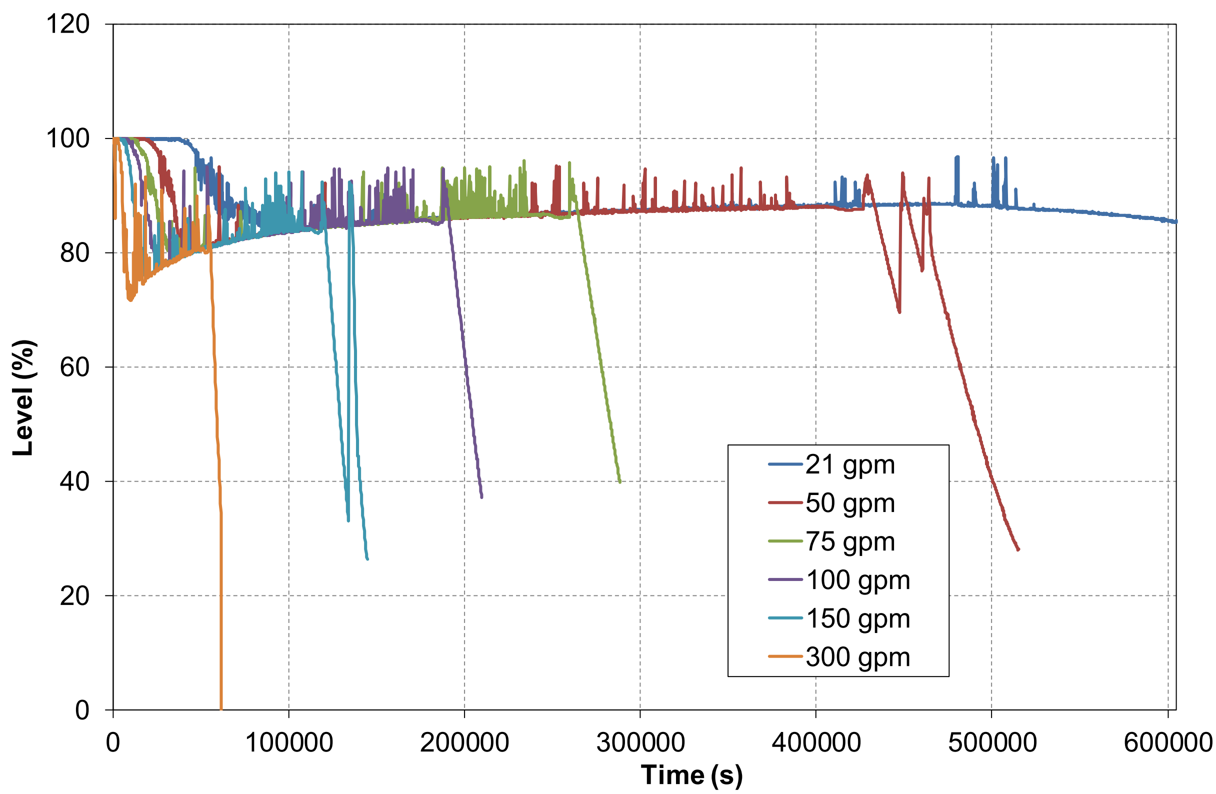

Figure 81 Core collapsed liquid level – S0 scenarios (2.06 MPa)

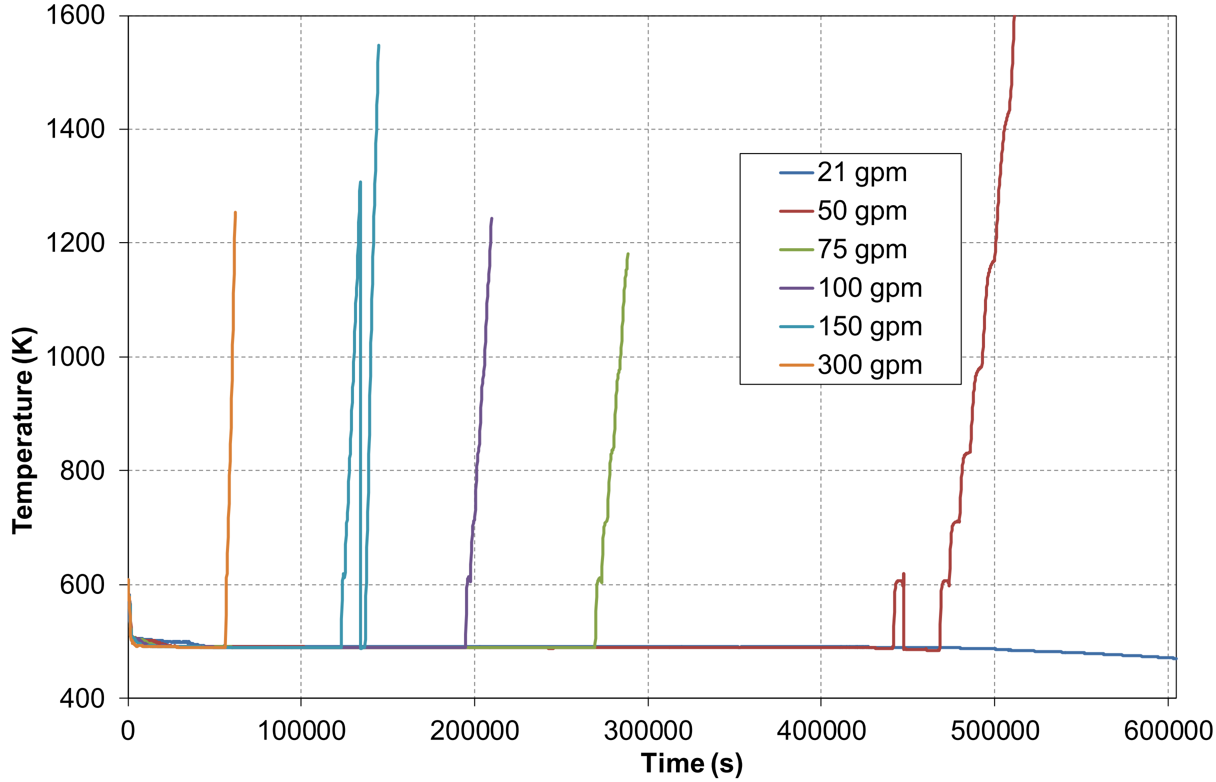

Figure 82 Average fuel cladding temperature – S0 scenarios (2.06 MPa)

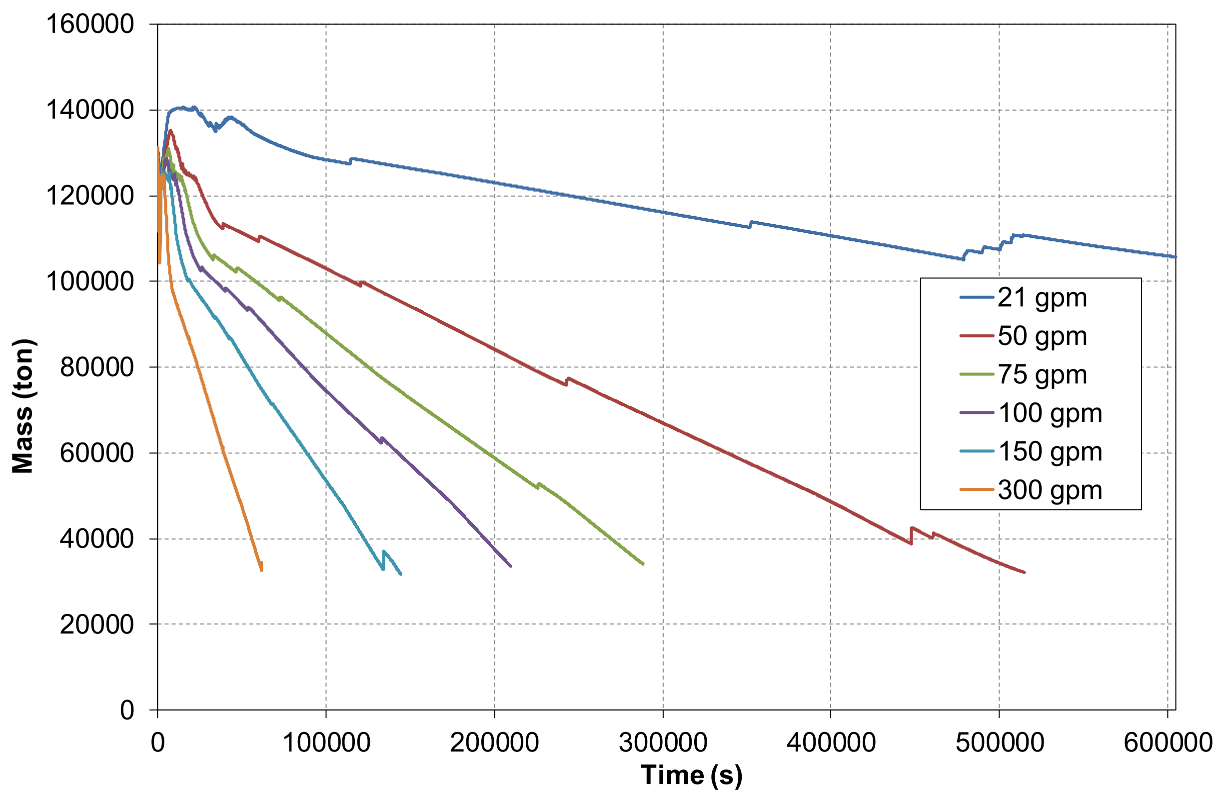

Figure 83 RCS mass inventory – S0 scenarios (2.06 MPa)

5.2.4 S0 scenarios depressurized to 2.55 MPa (25 kp/cm^2 gauge)

The results scenarios S0, in which the operator depressurizes and maintains SG pressure at 2.55 MPa are shown in Figures 84 through 88. The larger the break is, the shorter the time available before core uncovery and core heatup is.

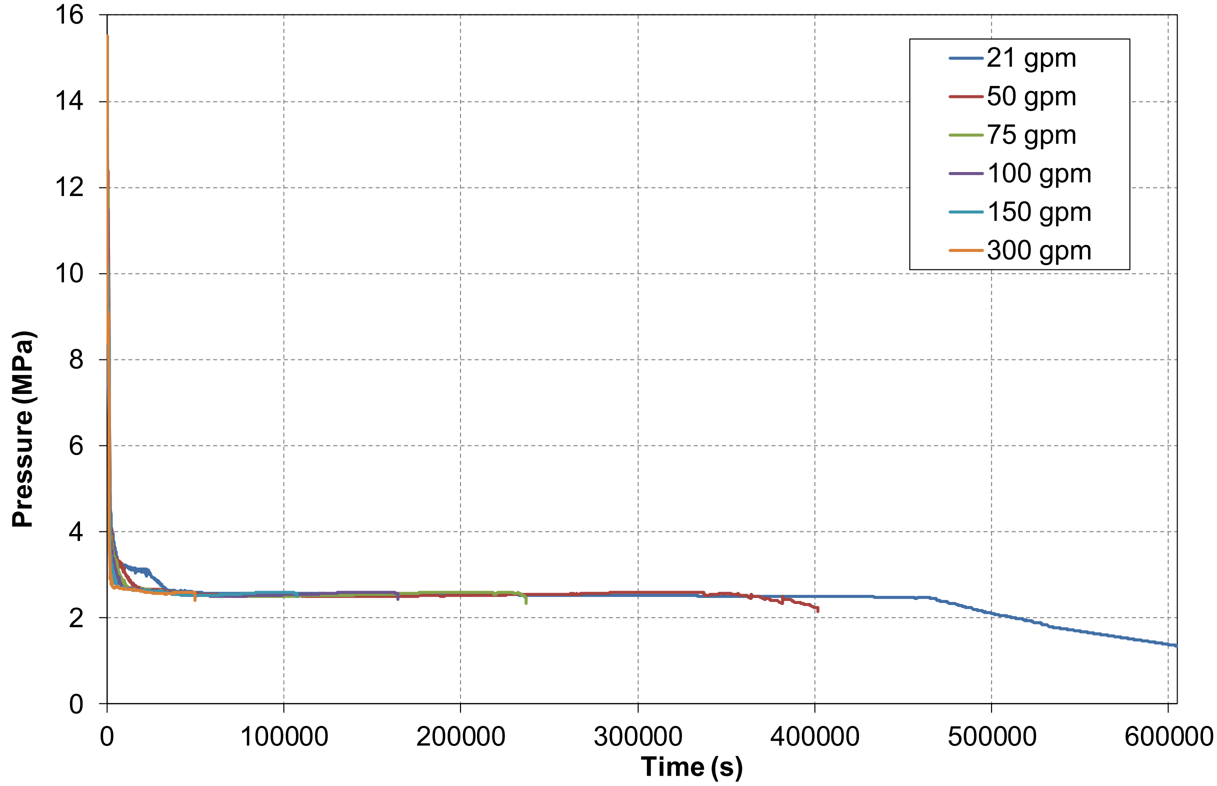

Figure 84 RCS pressure – S0 scenarios (2.55 MPa)

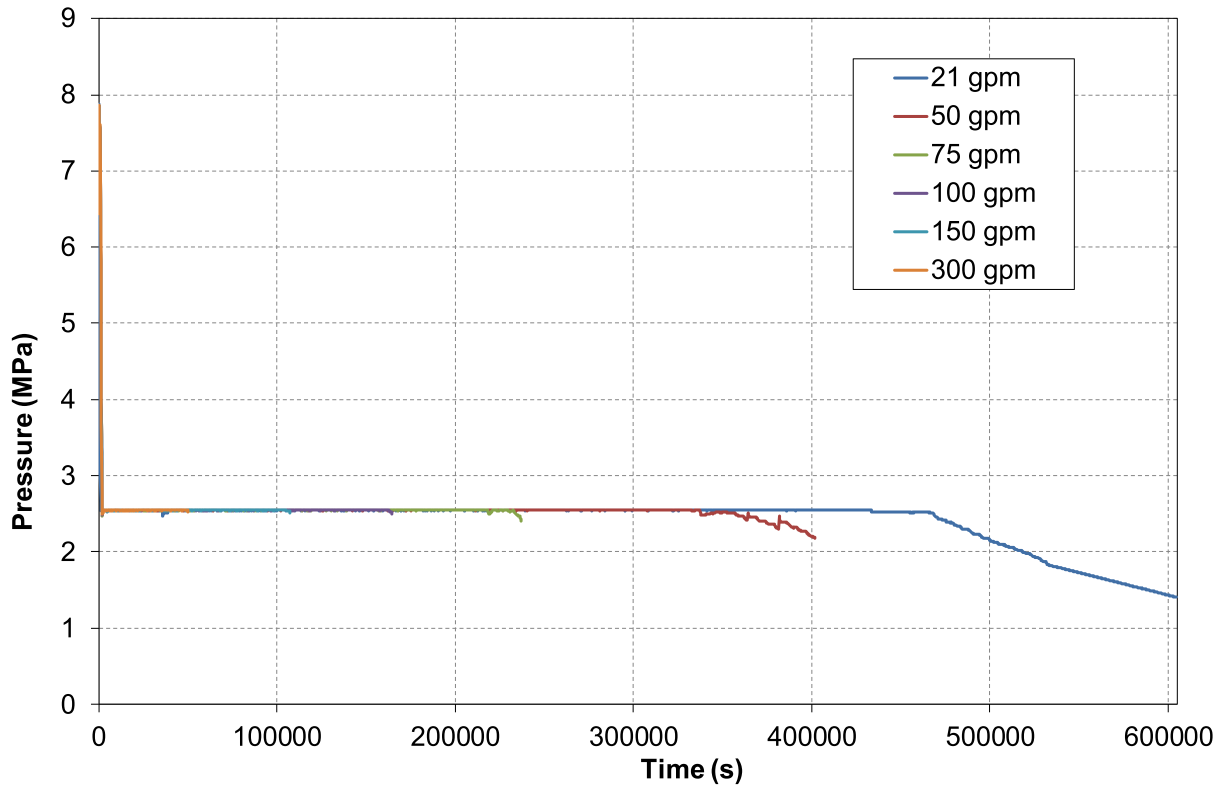

Figure 85 SG no.1 pressure – S0 scenarios (2.55 MPa)

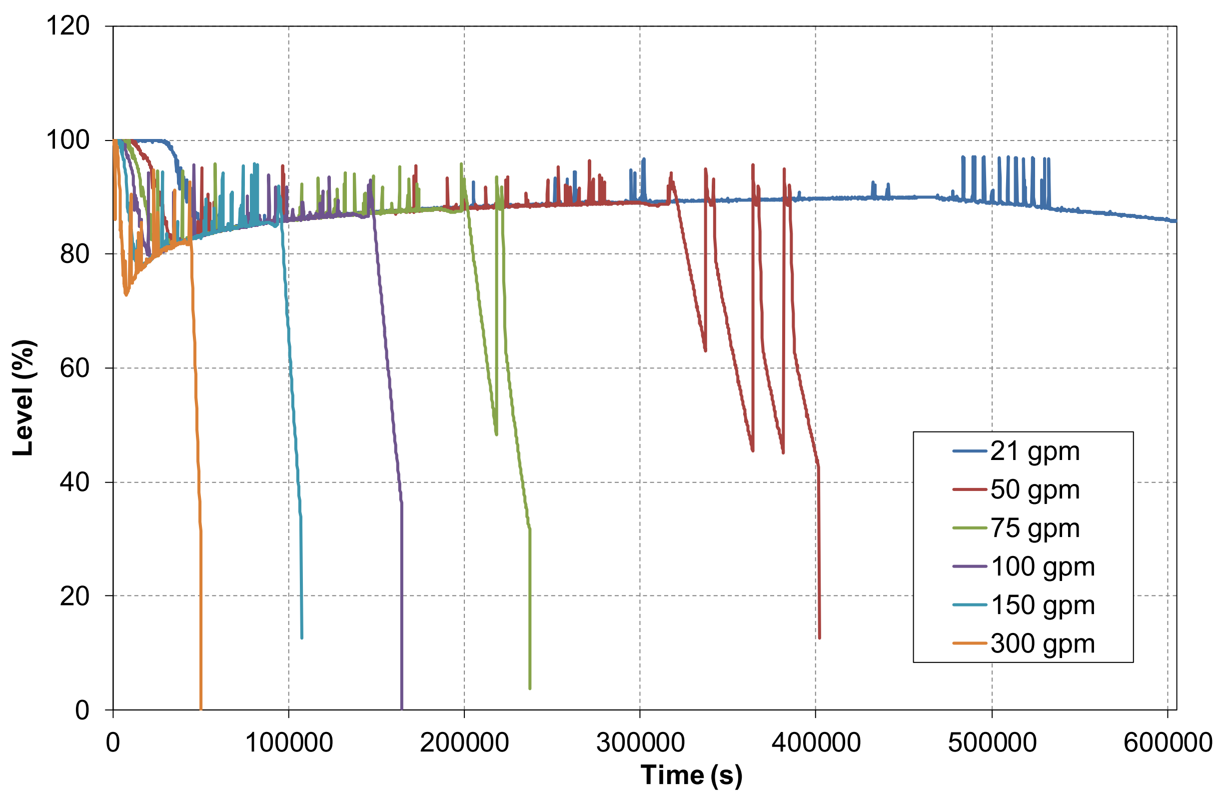

Figure 86 Core collapsed liquid level – S0 scenarios (2.55 MPa)

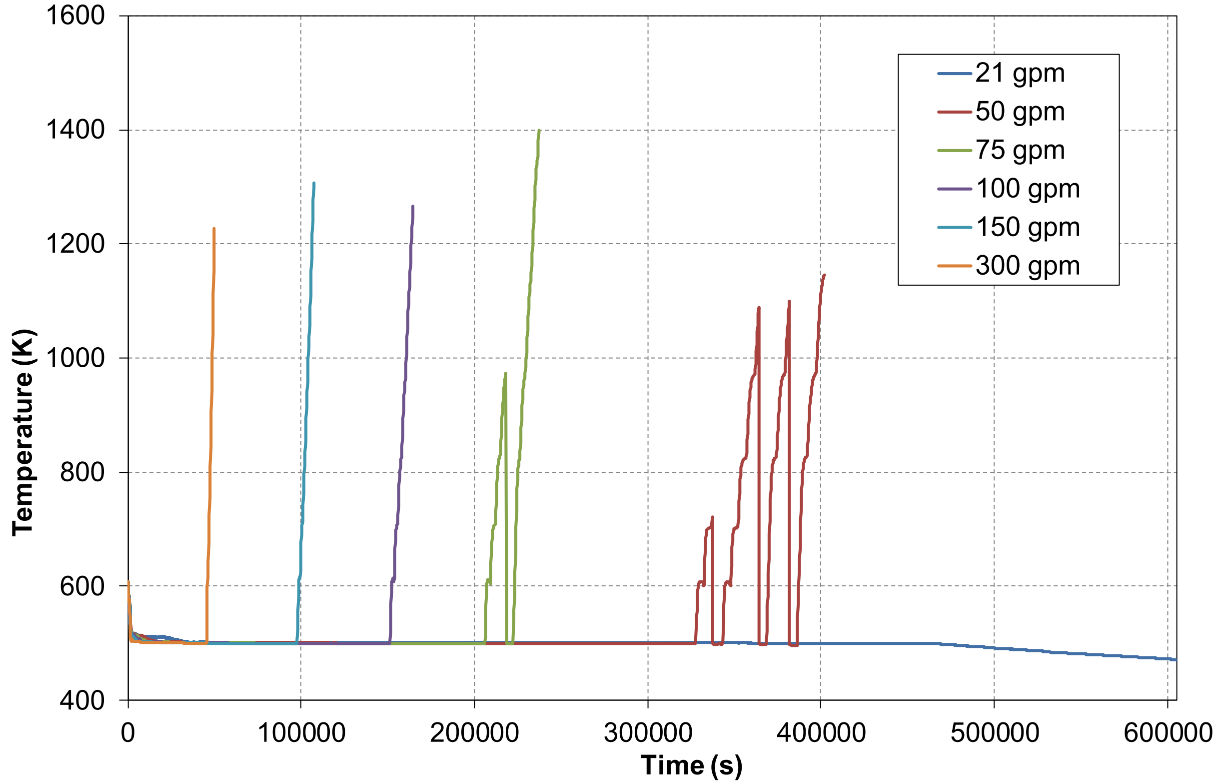

Figure 87 Average fuel cladding temperature – S0 scenarios (2.55 MPa)

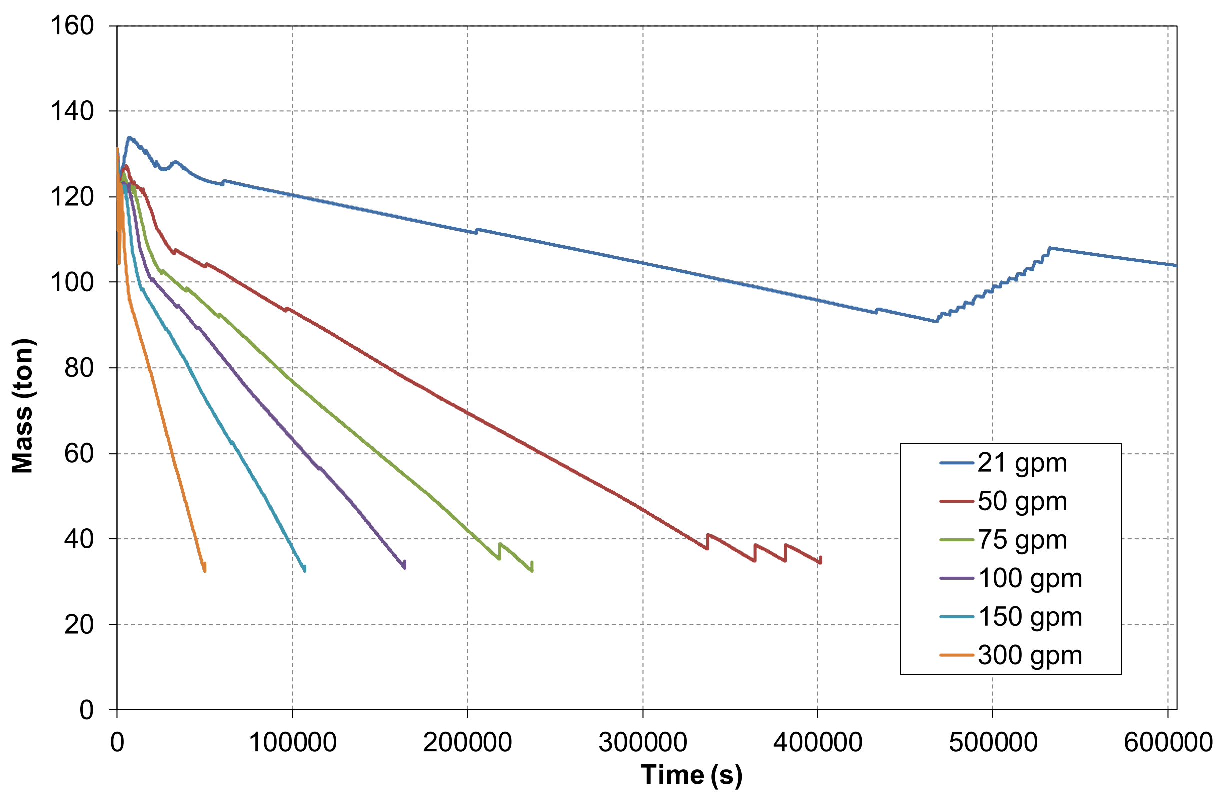

Figure 88 RCS mass inventory – S0 scenarios (2.55 MPa)

5.3 Dependence on equipment operation of SBO scenario with TD AFW lost 4 hours after transient start

For each set of S1 scenarios (consisting of four cases) on equipment operation, when TD AFW was lost after four hours into transient. The following four variables are shown: RCS pressure, RCS inventory, average fuel cladding temperature and total mass discharged from RCS (through letdown isolation valve when RCS pressure greater than 4.23 MPa, both RCP seal leaks and PRZ PORVs – only scenarios S1-21p1 and S1-21p2). As the variables have already been plotted in Section 5.1 for all cases except RCS mass inventory (it provides new information), the Figures 89 through 104 will not be described in detail. Rather, some remarks will be made.

It should be also noted that similar studies making depressurization by PRZ PORVs as in this section have been performed before Fukushima accident for scenarios in which also all heat sinks were lost besides loss of all AC power (see Ref. 5).

5.3.1 S1 scenarios depressurized to 1.33 MPa (12.5 kp/cm^2 gauge)

The results for scenarios S1, in which the operator depressurizes and maintains SG pressure at 1.33 MPa are shown in Figures 69 through 73. It is clearly shown that use of PRZ PORVs causes faster heatup when not using them. The results also emphasized the need for a heat sink (TD AFW) since it is essential to prevent core damage. Also, PDP charging pump prolongs the time when core uncovery and heatup occurred.

Regarding use of PRZ PORV the conclusions derived in Ref. 5 should be considered: “However it can be concluded that if the operators would be able to open both pressurizer relief valves after the core heatup starts, this would have positive effect on further progression of the severe accident. As it can be concluded from the presented analyses by performing this action within SAMG procedures, primary pressure will be at the time of the primary system vessel failure significantly lower than in the case that there will be no operator actions for primary system depressurization. If in such situation operators would be able to open only one pressurizer PORV would this be beneficial for later accident progression.”

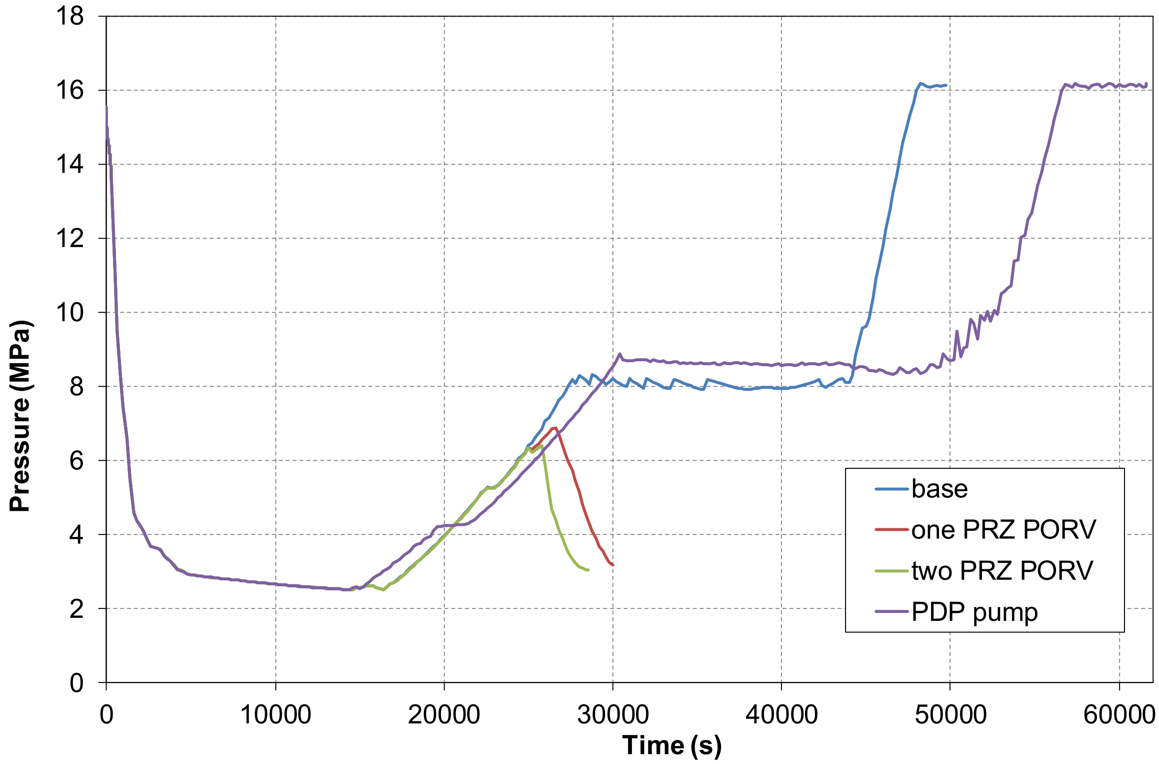

Figure 89 RCS pressure – S1 scenarios (1.33 MPa)

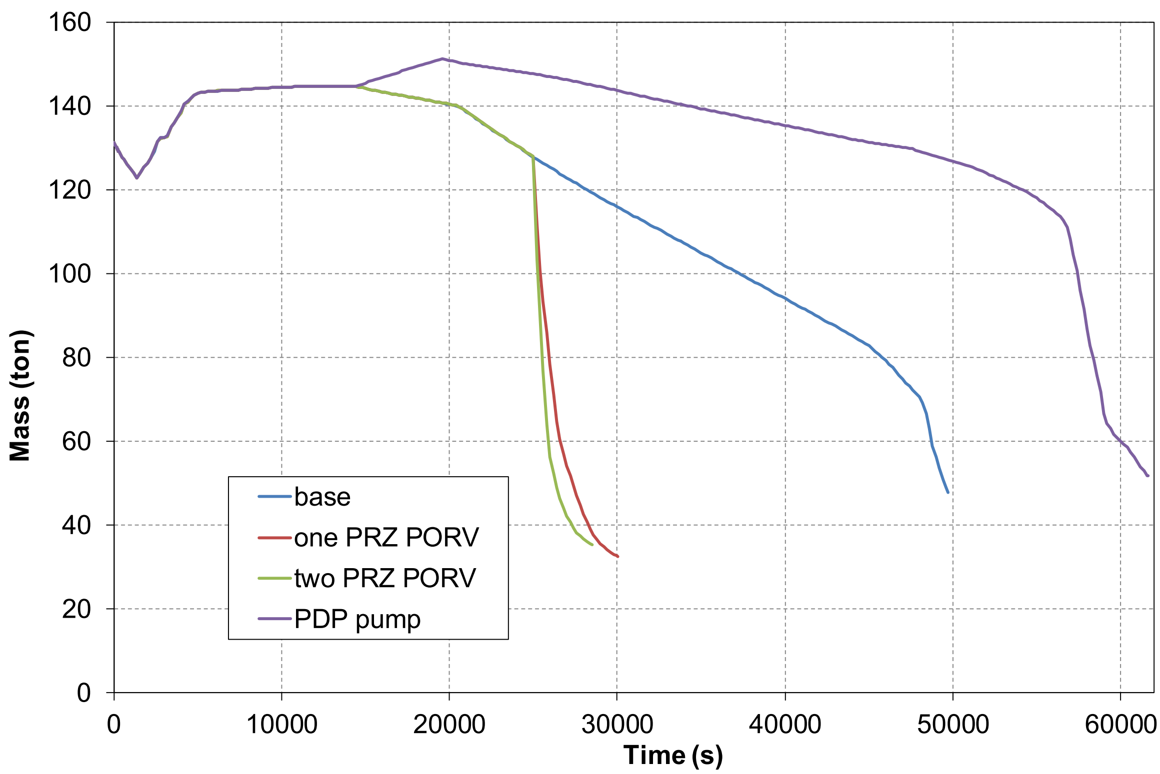

Figure 90 RCS mass inventory – S1 scenarios (1.33 MPa)

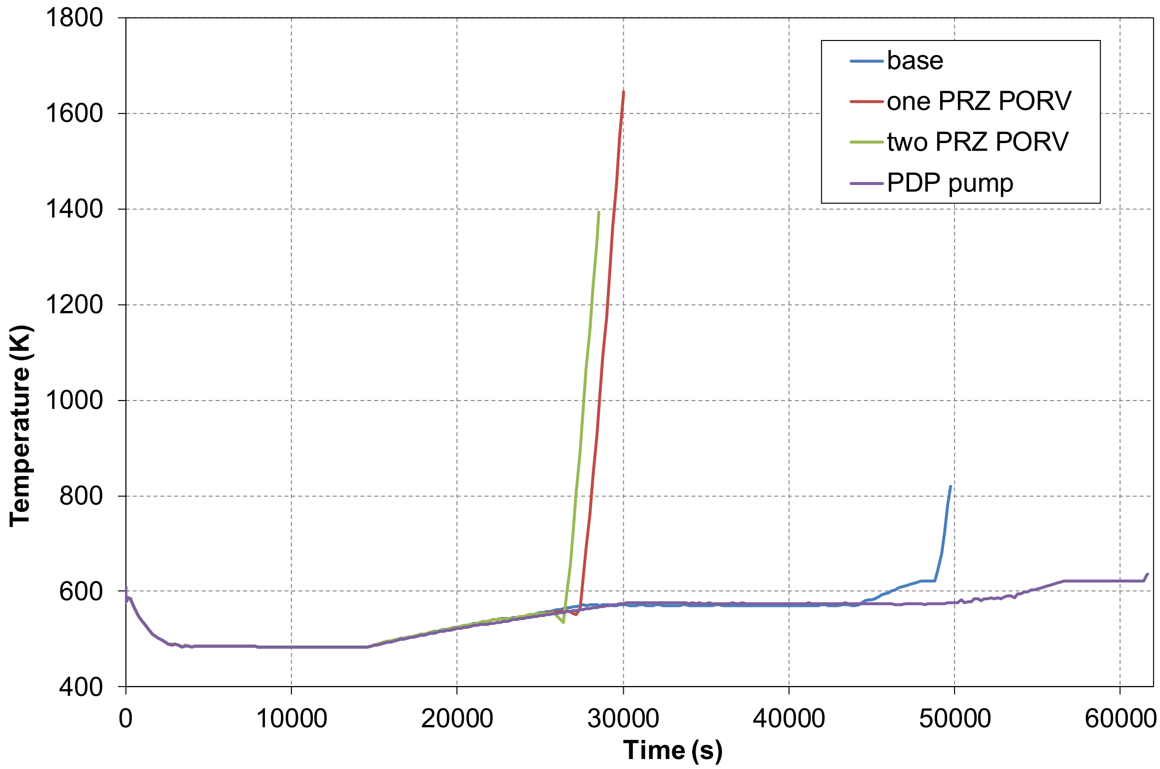

Figure 91 Average fuel cladding temperature – S1 scenarios (1.33 MPa)

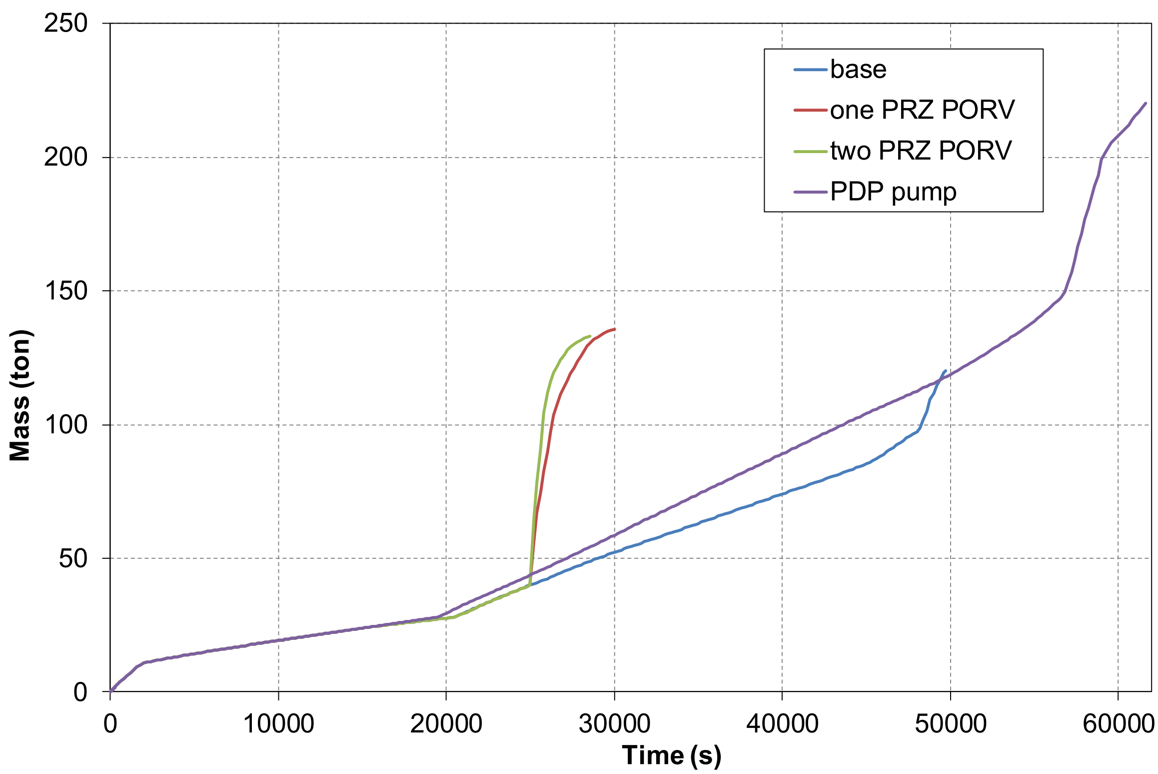

Figure 92 Total mass discharged from RCS – S1 scenarios (1.33 MPa)

5.3.2 S1 scenarios depressurized to 1.57 MPa (15 kp/cm^2 gauge)

The results for scenarios S1, in which the operator depressurizes and maintains SG pressure at 1.57 MPa are shown in Figures 93 through 96. It is clearly shown that use of PRZ PORVs causes faster heatup when not using them. The results also emphasized the need for a heat sink (TD AFW) since it is essential to prevent core damage. Also, PDP charging pump prolongs the time when core uncovery and heatup occurred.

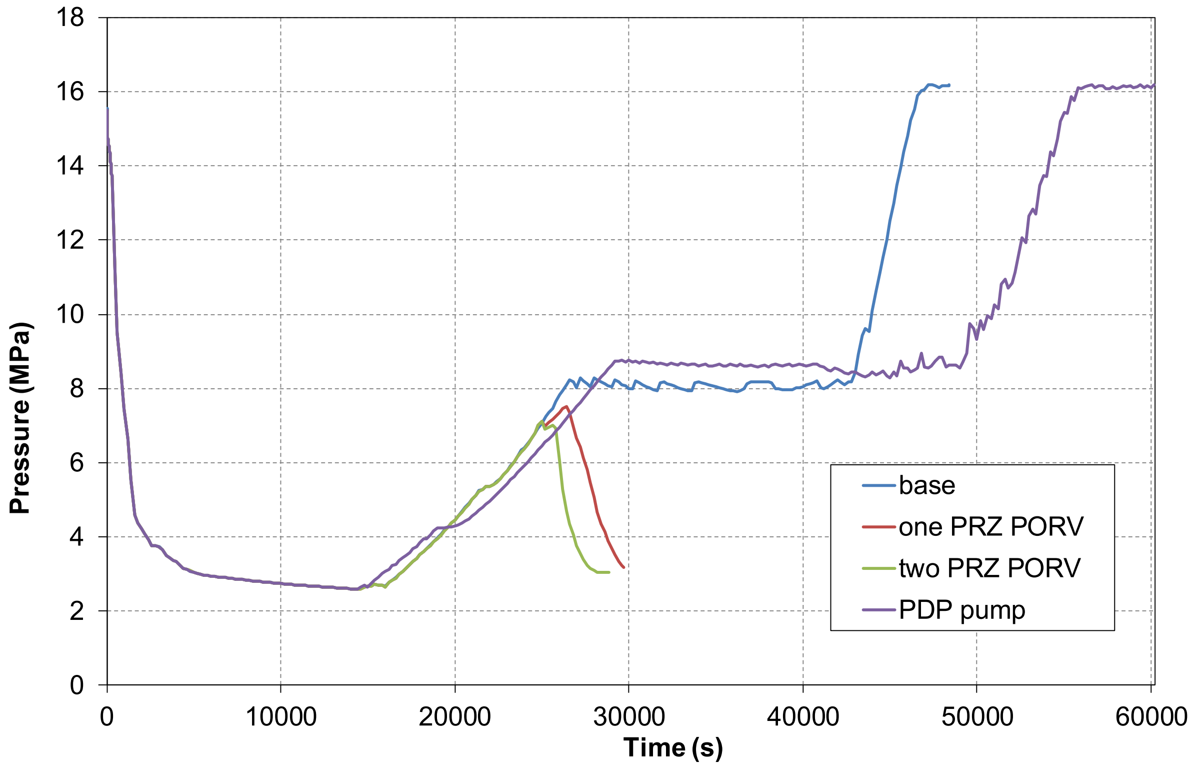

Figure 93 RCS pressure – S1 scenarios (1.57 MPa)

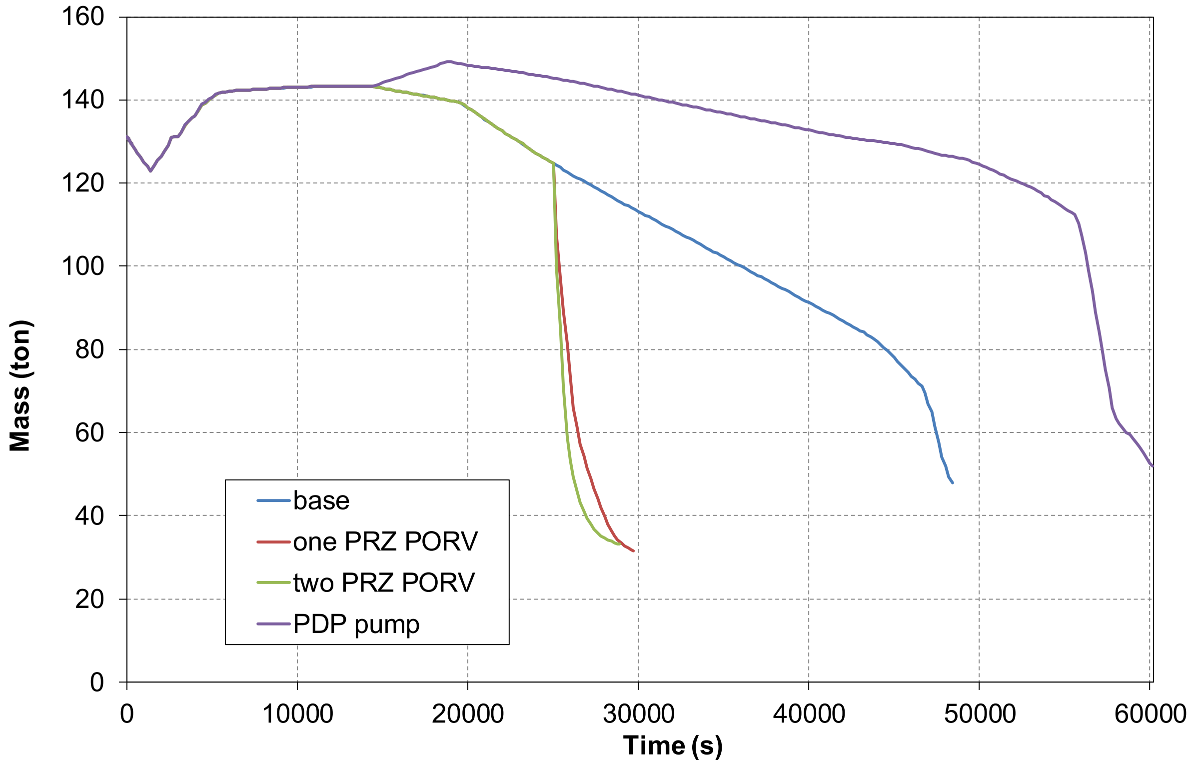

Figure 94 RCS mass inventory – S1 scenarios (1.57 MPa)

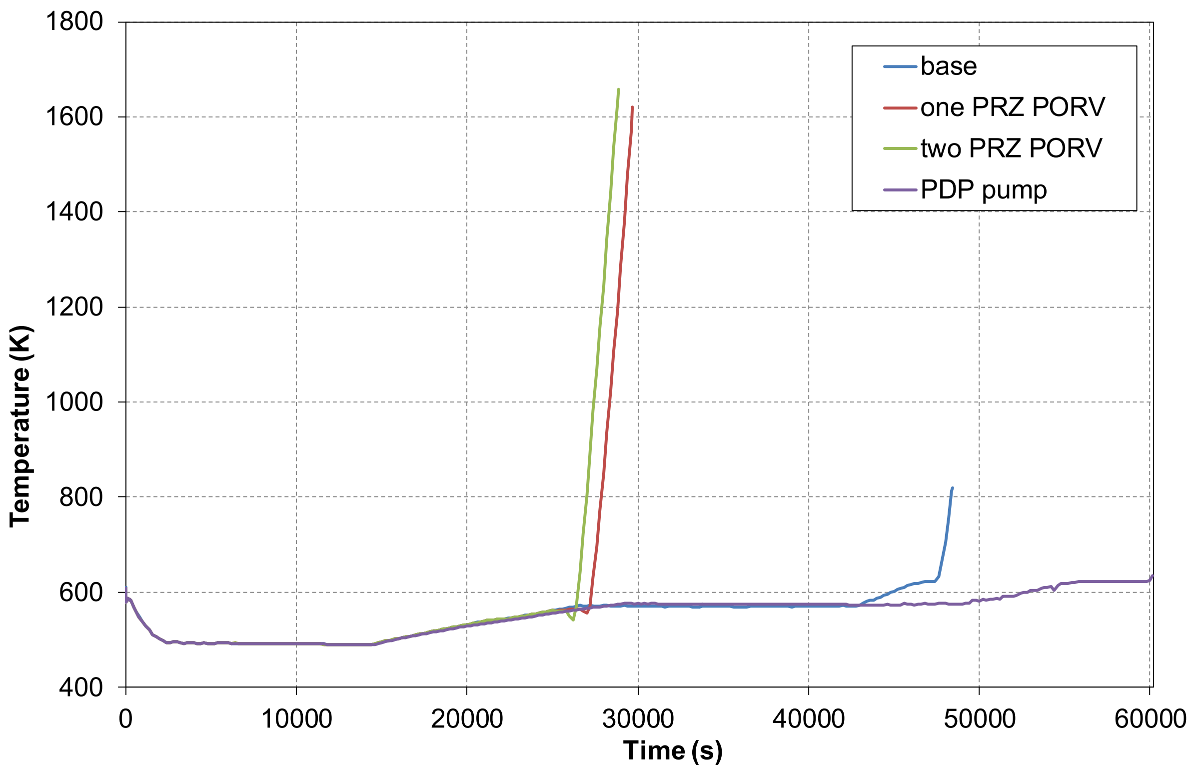

Figure 95 Average fuel cladding temperature – S1 scenarios (1.57 MPa)

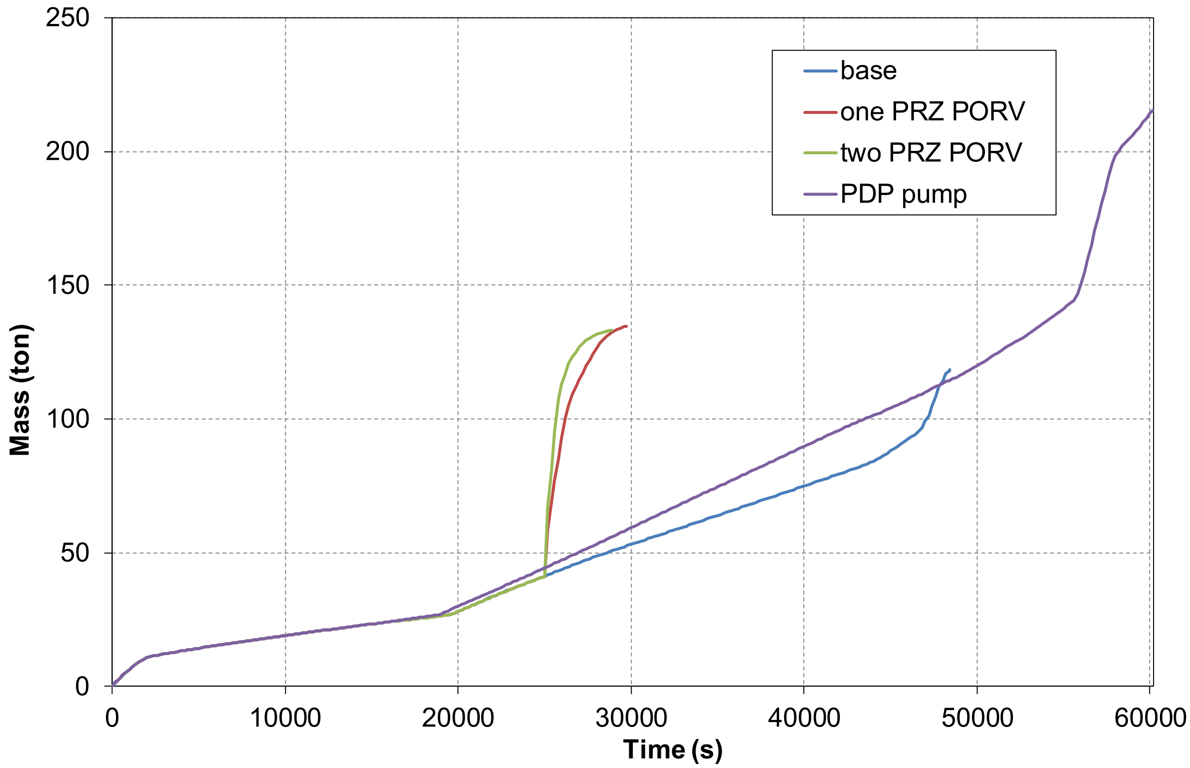

Figure 96 Total mass discharged from RCS – S1 scenarios (1.57 MPa)

5.3.3 S1 scenarios depressurized to 2.06 MPa (20 kp/cm^2 gauge)

The results for scenarios S1, in which the operator depressurizes and maintains SG pressure at 1.57 MPa are shown in Figures 97 through 100. It is clearly shown that use of PRZ PORVs causes faster heatup when not using them. The results also emphasized the need for a heat sink (TD AFW) since it is essential to prevent core damage. Also, PDP charging pump prolongs the time when core uncovery and heatup occurred.

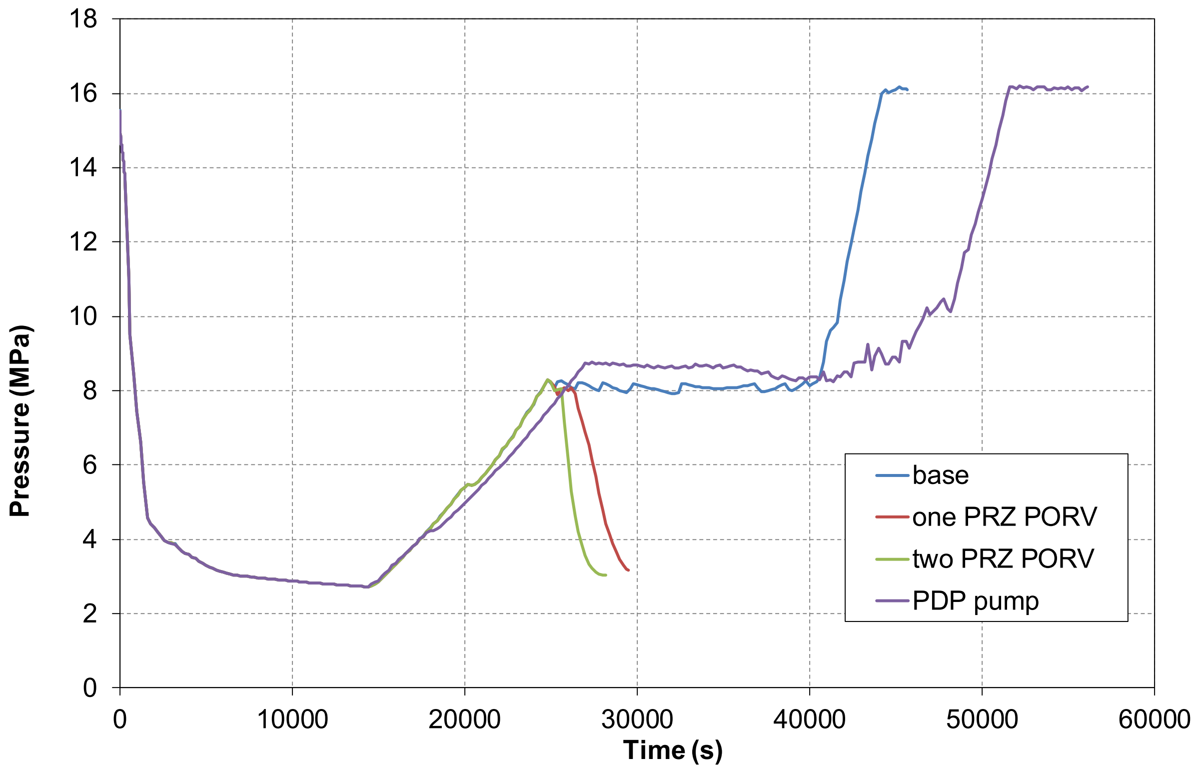

Figure 97 RCS pressure – S1 scenarios (2.06 MPa)

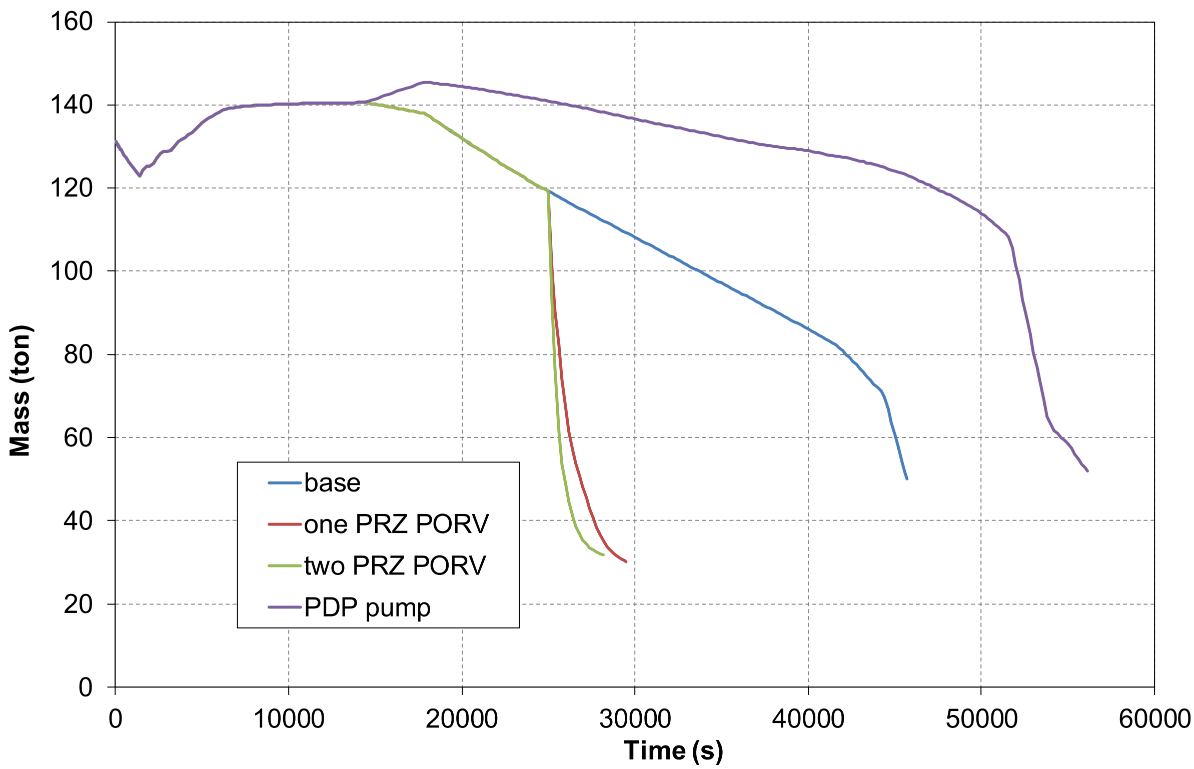

Figure 98 RCS mass inventory – S1 scenarios (2.06 MPa)

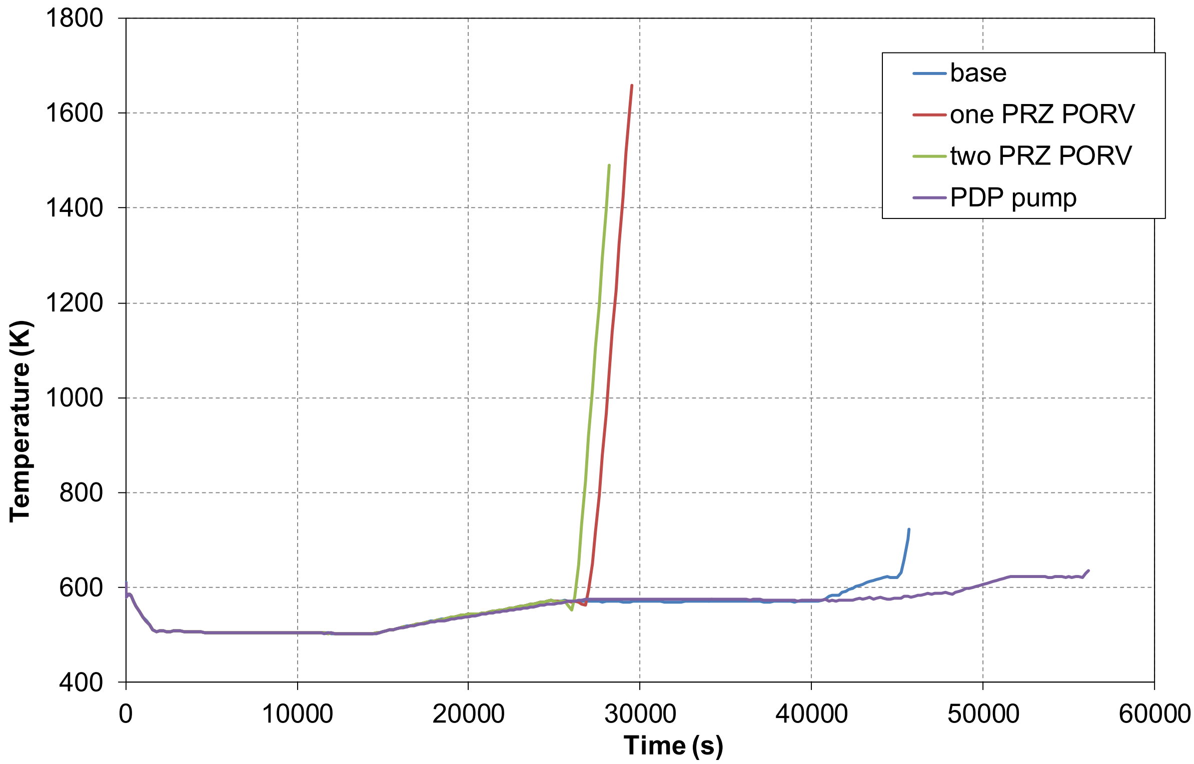

Figure 99 Average fuel cladding temperature – S1 scenarios (2.06 MPa)

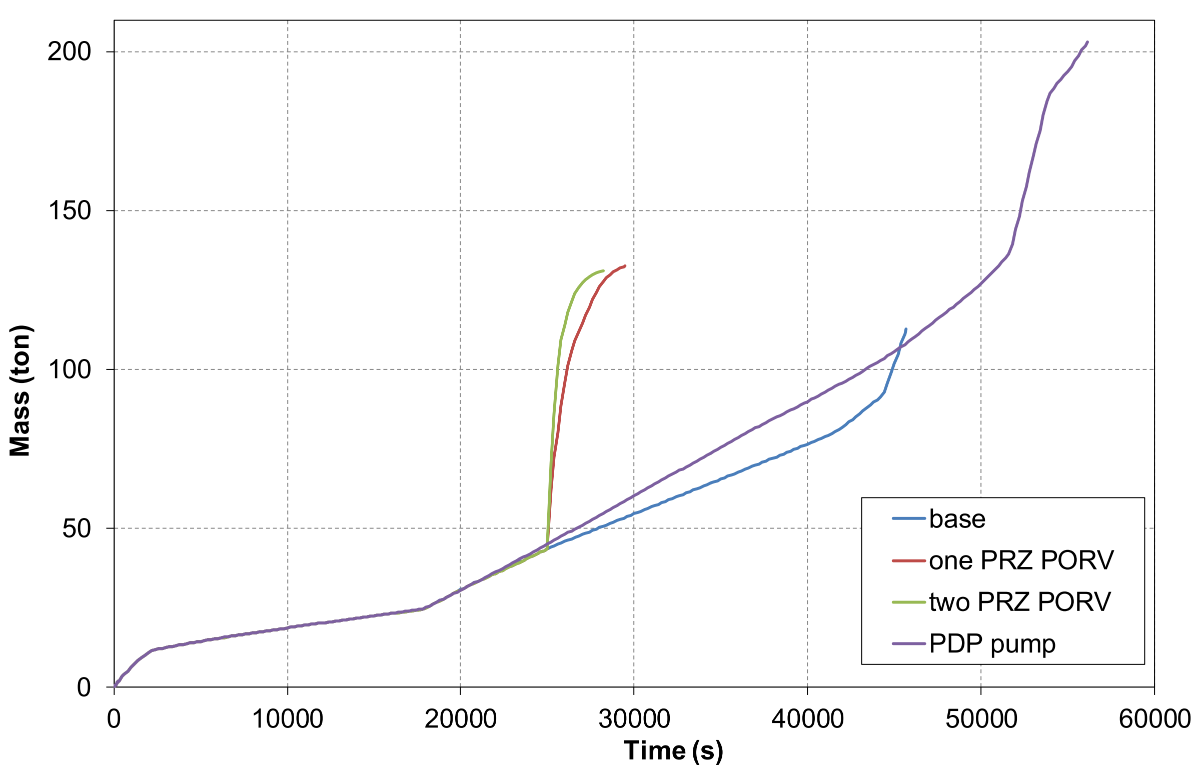

Figure 100 Total mass discharged from RCS – S1 scenarios (2.06 MPa)

5.3.4 S1 scenarios depressurized to 2.55 MPa (25 kp/cm^2 gauge)

The results for scenarios S1, in which the operator depressurizes and maintains SG pressure at 2.55 MPa are shown in Figures 101 through 104. It is clearly shown that use of PRZ PORVs causes faster heatup when not using them. The results also emphasized the need for a heat sink (TD AFW) since it is essential to prevent core damage. Also, PDP charging pump prolongs the time when core uncovery and heatup occurred.

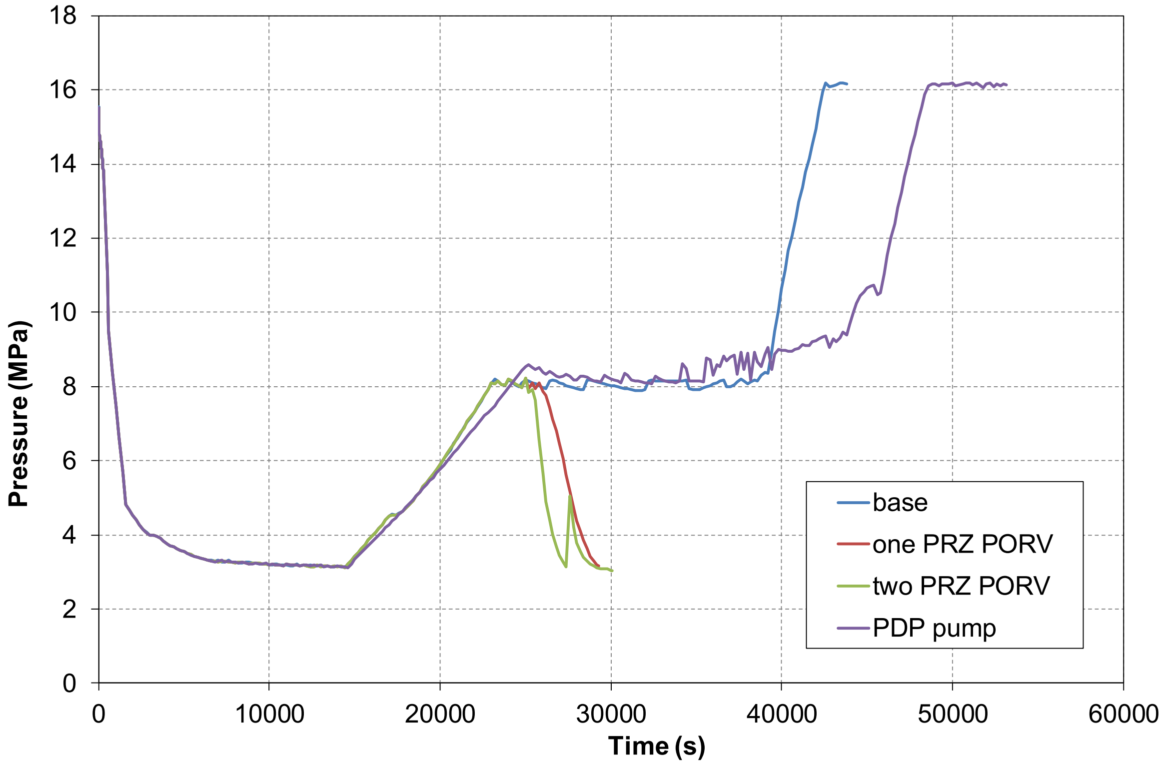

Figure 101 RCS pressure – S1 scenarios (2.55 MPa)

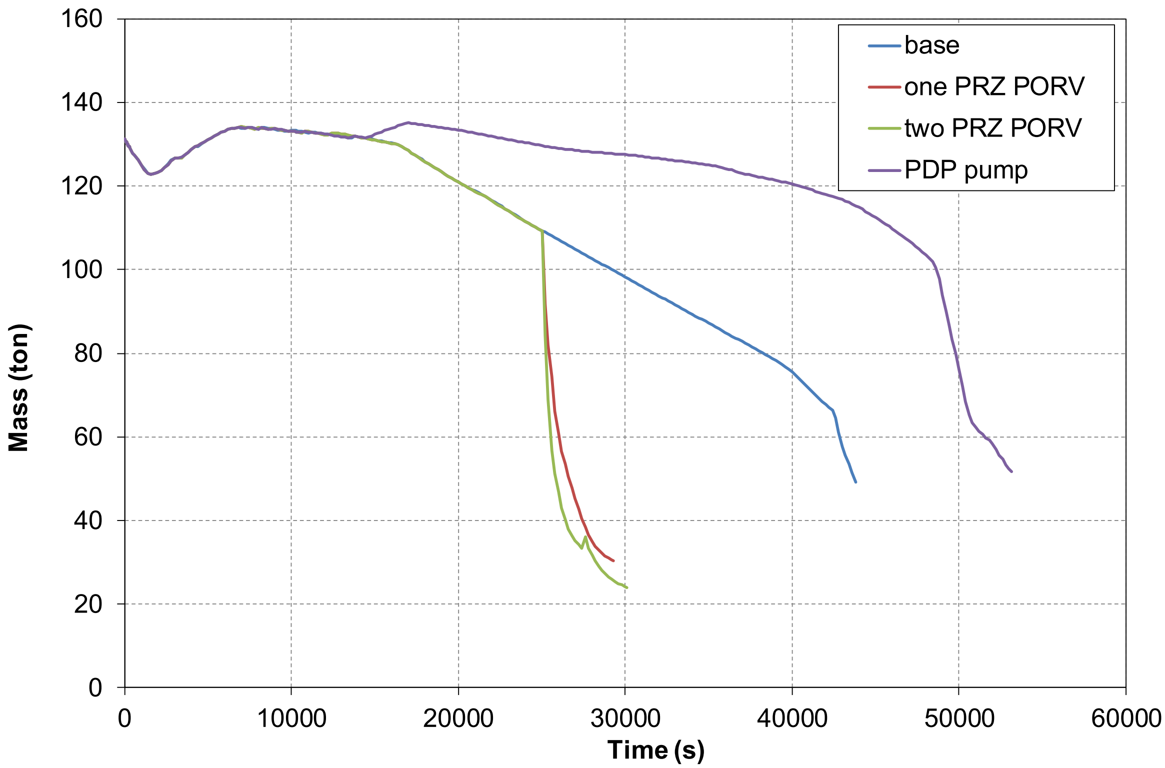

Figure 102 RCS mass inventory – S1 scenarios (2.55 MPa)

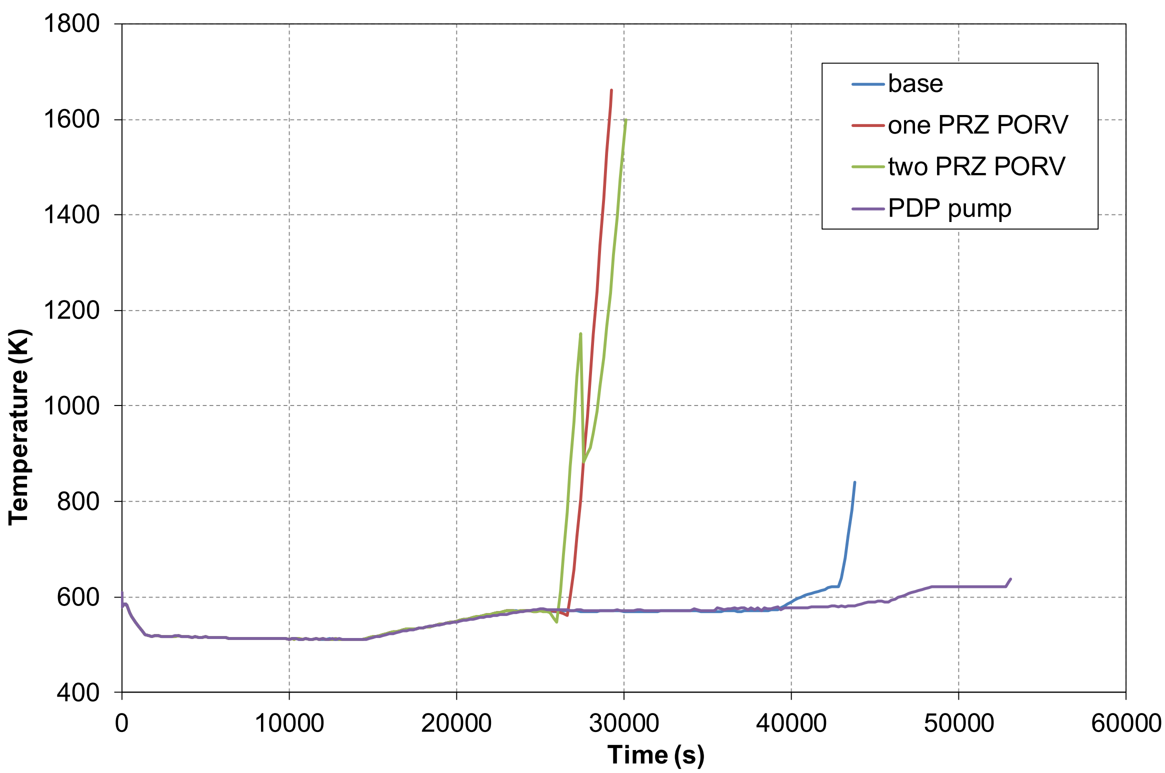

Figure 103 Average fuel cladding temperature – S1 scenarios (2.55 MPa)

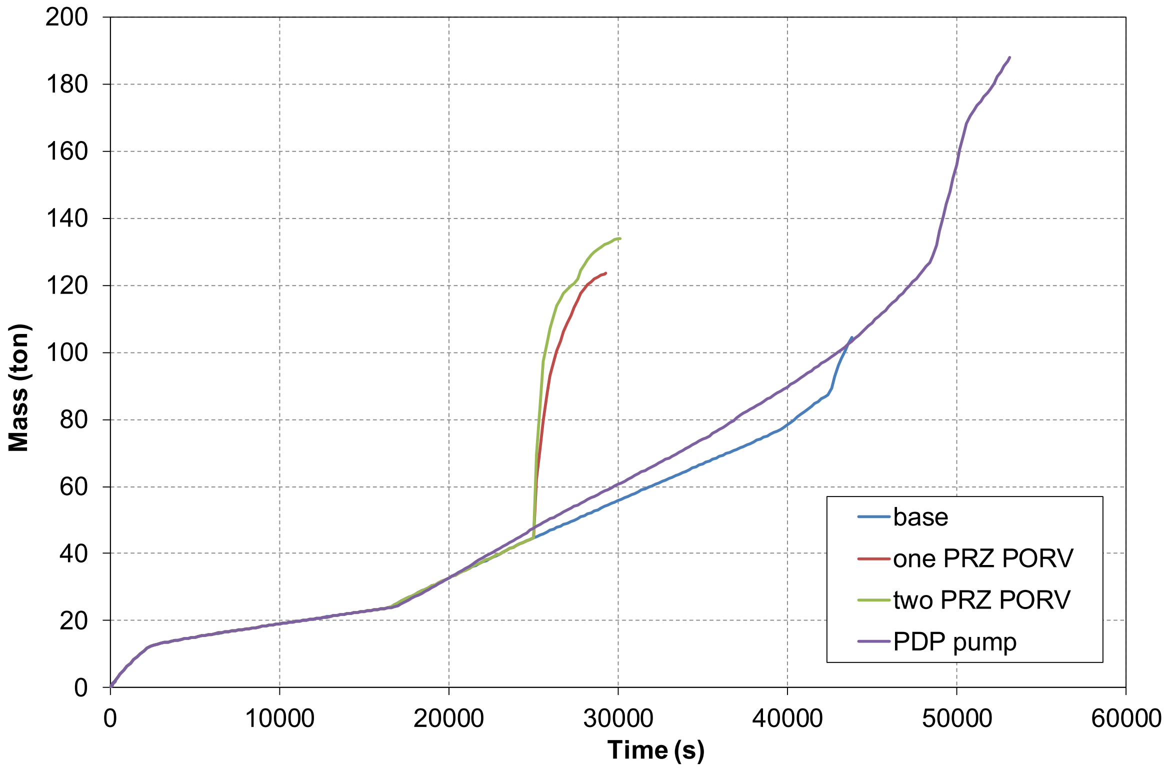

Figure 104 Total mass discharged from RCS – S1 scenarios (2.55 MPa)

6. RUN STATISTICS

The calculations with the RELAP5/MOD3.3 Patch 04 computer code (linux version relap5-33iy-linux-ifc-opt.x) were performed on cluster Krn with 50 nodes and 600 processor cores. Each node has two Intel Xeon 5670 @ 2.93 GHz processor, each having 6 cores and 6 threads. The operating system is SUSE Linux Enterprise Server 11 (x86_64) - service pack 1.

Table 2 shows the run statistics for base calculation (scenario S0-21, depressurization 2.55 MPa). For other calculations the statistic is similar for the same length of run. If runs are shorter, the CPU time is smaller accordingly. For all calculations, the number of volumes was 432. The calculations run five times faster than real time. Steady-state calculations for all runs lasted 1,000 seconds and required 231.2 seconds of central processing unit (CPU) and 25,948 steps.

Table 2 Run statistics

Transient Time (s)	CPU Time (s)	CPU/Transient Time	Number of Time Steps
604800	119206	0.197	15255188

7. CONCLUSIONS

In this report long term station blackout sequences for Krško two-loop pressurized water reactor with loss of normal or all secondary side heat sinks were studied. For calculations the latest RELAP5/MOD3.3 Patch 04 computer code was used. The verified standard RELAP5/MOD3.3 input model from 2008 (cycle 23) was delivered by Krško nuclear power plant.

SBO scenarios were analyzed for different RCP leak seals. Besides, scenarios were performed for different primary side depressurizations performed by operator through the secondary side power operated relief valves, providing that turbine driven auxiliary feedwater pump is available.

The results showed that when RCP seal leaks are small the core uncovery could be prevented in the first seven days by using TD AFW pump and manually depressurizing the RCS through SGs depressurization. When RCP seal leaks are larger, small capacity RCS makeup is needed in addition to TD AFW pump to prevent core uncovery and core heatup. It was also shown that with TD AFW not available after 4 hours after around 10 hours the core will start to heatup.

The results clearly showed that alternative RCS makeup (when emergency core cooling system is lost) is also very important for preventing core uncovery; especially in the cases with larger breaks this is the only way to prevent core heatup in the first seven days. It can be concluded, provided that TD AFW and some RCS injection of the order of 2 kg/s mass flowrate are available, RCP seal leaks are of no concern. If only TD AFW is available, it is very important to limit the RCP seal leaks. One of the ways to limit the RCP seal leaks is manual depressurization strategies, which is therefore very important in the absence of RCS makeup. Primary side depressurization is of very limited use in preventing core heatup. However, would have positive effect on further progression of the severe accident according to study in Ref. 5.

Finally, the results suggest that RELAP5 can be used for extended SBO studies until core damage started. It is especially useful in studying maintaining core cooling function and time available before core uncovers as part of severe accident management. The benefit of using RELAP5 is in the fact that best estimate system codes are more accurate than severe accident codes in phases before core degradation started.

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10. SUPPLEMENTARY NOTES
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11. ABSTRACT *(200 words or less)*
Following the accident at the nuclear power plant Fukushima in Japan the stress test were performed in European countries. Considering the stress tests specifications it was necessary to evaluate the consequences of loss of safety functions from any initiating event (earthquake or flooding) for loss of electrical power, including Station Blackout (SBO), loss of the ultimate heat sink or combination of both. In this report long term station blackout sequences for Krško two-loop pressurized water reactor with loss of normal or all secondary side heat sinks were performed. For calculations the latest RELAP5/MOD3.3 Patch 04 computer code was used. The verified standard RELAP5/MOD3.3 input model delivered by Krško nuclear power plant was used.

SBO scenario involves a loss of offsite power, failure of the redundant emergency diesel generators, failure of alternate current power restoration and the eventual degradation of the reactor coolant pump seals resulting in a long term loss of coolant. In the study different reactor coolant pump seal leaks were studied due to SBO. Besides, scenarios were performed for different primary side depressurizations performed by operator through the secondary side power operated relief valves, providing that turbine driven auxiliary feedwater pump is available. Finally, the effect of having some injection into the reactor coolant system was also evaluated. It can be concluded that calculated results obtained by RELAP5 give good indication about time available before core degradation started.

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