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LOCAs With Loss of One Active Emergency Cooling System Simulated by RELAP5

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ABSTRACT

The second generation reactors were designed and built to withstand without loss to the structures, systems, and components necessary to ensure public health and safety during design basis accidents. There are also accident sequences that are possible but were judged to be too unlikely and therefore were not fully considered in the design process of second generation reactors. In that sense, they were considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand. They were called beyond design basis accidents. After Fukushima-Daiichi in the Europe the design extension conditions were introduced as preferred method for giving due consideration to the complex sequences and severe accidents without including them in the design basis conditions. In the study, the analysis was performed to see if the plant design can prevent spectrum of loss of coolant accidents together with the complete loss of one emergency core cooling function (e.g. high pressure injection or low pressure injection). The analyzed break spectrum ranged from 1.27 cm to 30.48 cm. For calculations the latest RELAP5/MOD3.3 Patch 5 has been used and the RELAP5 input model of Krško nuclear power plant.

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The control of DECs is expected to be achieved primarily by features implemented in the design (safety features for DECs) and not only by accident management measures that are using equipment designed for other purposes. This means that in principle a DEC is such if its consideration in the design leads to the need of additional equipment or to an upgraded classification of lower class equipment to mitigate the DEC.

In the analysis presented the loss of coolant accidents (LOCA) break spectrum ranging from 1.27 cm to 30.48 cm has been analysed using the latest RELAP5/MOD3.3 Patch 5 and RELAP5 input model of Krško nuclear power plant, which is a two loop pressurized water reactor (PWR). The analysis was performed to see if the plant design can prevent spectrum of LOCAs together with the complete loss of one emergency core cooling function (e.g. high pressure injection or low pressure injection). Besides this for smaller breaks ranging from 1.27 cm to 5.08 cm leading to core heatup (being design extension conditions as new equipment is needed) accident management strategies to depressurize the primary system in order to enable injection of accumulators and low pressure safety injection system have been also studied.

ACKNOWLEDGMENTS

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ABBREVIATIONS AND ACRONYMS

AFW	auxiliary feedwater
AM	accident management
ATWS	anticipated transients without scram
BDBA	beyond design basis accidents
CVCS	chemical and volume control system
DBA	design basis accident
DEC	design extension conditions
ECCS	emergency core cooling system
IAEA	International Atomic Energy Agency
LOCA	loss of coolant accident
HPI	high pressure injection
HPIS	high pressure injection sytem
HPSI	high pressure safety injection
LSTF	Large Scale Test Facility
LPI	low pressure injection
LPIS	low pressure injection system
LPSI	low pressure safety injection
MD	motor driven
MFW	main feedwater
MSIV	main steam isolation valves
NPP	Nuclear Power Plant
NR	narrow range
PORV	power operated relief valve
PRZ	pressurizer
PWR	pressurized water reactor
RCP	reactor coolant pump
RL	reference level
RPV	reactor pressure vessel
SBO	station blackout
SD	steam dump
SI	safety injection
SL	surge line
SG	steam generator

TD	turbine driven
UOP	Ultimate Operating Procedure
WENRA	Western European Nuclear Regulators Association

1 INTRODUCTION

The second generation reactors were designed and built to withstand without loss to the structures, systems, and components necessary to ensure public health and safety during design basis accidents (DBAs). In the transient and accident analysis the effects of single active failures and operator errors were considered. There are also accident sequences that are possible but were judged to be too unlikely and therefore were not fully considered in the design process of second generation reactors. In that sense, they were considered beyond the scope of design basis accidents that a nuclear facility must be designed and built to withstand. They were called beyond design basis accidents (BDBA).

The term “design extension conditions” has been introduced to define some selected accident sequences due to multiple failures by the European Utility Requirements document [1]. The design extension conditions were introduced as preferred method for giving due consideration to the complex sequences and severe accidents at the design stage of the reactors of third generation without including them in the design basis conditions. The Western European Nuclear Regulators Association (WENRA) recommended a “design extension” analysis in 2007 [2] and they proposed a list of events to be analysed at minimum. By its meaning this list corresponds to DEC without core melt. After Fukushima Dai-ichi accident also the International Atomic Energy Agency (IAEA) adopted the term design extension conditions (DEC) [3]. WENRA reference levels (RLs) from 2014 [4] introduced DEC term. The WENRA guidance document for issue F for existing reactors (second generation) explains [5] explains that DEC in WENRA RLs are consistent with the definition of DEC in IAEA SSR-2/1 [3], published in 2012: "*Accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions.*" DEC are more complex and/or more severe than conditions postulated as design basis accidents [5].

Finally, the new revision 1 of IAEA SSR-2/1 [6] has been released in 2016, redefining the DEC term, which definition is now as follows: “*Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions comprise conditions in events without significant fuel degradation and conditions in events with core melting*”. WENRA did not follow the new IAEA definition of DEC in spite of the fact that modification is significant, as DEC are now defined as postulated accident conditions.

The DEC concept by IAEA and WENRA (DEC with prevention of core melt is called DEC A) is not completely new, since in some countries selected multiple failures have already been considered in the design (through back fitting process), for example anticipated transients without scram (ATWS) and station blackout (SBO). Also, the research for beyond design basis accidents with non-degraded core (i.e. DEC A) for existing reactors has been already done in 80’s and 90’ of the previous century.

Slovenia implemented WENRA reference levels issue F requirements into its Rules on radiation and nuclear safety factors [12]. It is prescribed that the selection process for design extension conditions A shall start by considering those events and combinations of events, which cannot be considered with a high degree of confidence to be extremely unlikely to occur and which may lead to severe fuel damage in the core or in the spent fuel storage. It shall cover: a) events occurring during the defined operational states of the plant; b) events resulting from internal or external hazards; and c) common cause failures. In the presented analysis two such DEC A scenarios were simulated, loss of coolant accident (LOCA) in cold leg together with complete loss of high pressure injection (HPI) system and LOCA with complete loss of low pressure injection (LPI) system. This is in line with IAEA and WENRA, as can be seen in the following.

WENRA guidance document [5] for issue F also provides this scenario as an example of DEC A:

- Loss of coolant accident (LOCA) together with the complete loss of one emergency core cooling function (e.g. high pressure injection (HPI) or low pressure injection (LPI)).

Also IAEA TECDOC-1791 document [13] also provides this scenario as an example of DEC derived from probabilistic safety assessment (PSA):

- LOCA plus loss of one emergency core cooling system (either the high pressure or the low pressure emergency cooling system).

The analysis presented will show if the plant design can prevent above LOCA scenarios with existing safety systems or not (in this case additional safety features are needed). Following document [13], the control of DECs is expected to be achieved primarily by features implemented in the design (safety features for DECs) and not only by accident management measures that are using equipment designed for other purposes. This means that in principle a DEC is such if its consideration in the design leads to the need of additional equipment or to an upgraded classification of lower class equipment to mitigate the DEC.

Similar experiments has already been investigated in the past dealing with accident management. Therefore, the results of these experiments have been first reviewed. Table 1-1 shows selected tests performed on BETHSY and LOFT test facilities for accident management in pressurized water reactors (PWRs) for loss of coolant accidents (LOCAs), in which operator actions were studied for BDBA with non-degraded core (in terms of WENRA such accidents are called DEC A). Experiments were mainly selected from cross-reference matrix for accident management for non-degraded core, which has been created in the frame of OECD/NEA [7].

Table 1-1 Accident Management in PWRs For BDBA (With LOCAs) With Non-Degraded Core

Test No.	Test type	Brief description
BETHSY 6.2TC	6” cold leg break without HPIS and LPIS	BETHSY 6.2TC test was a 15.24 cm (6 inch) cold leg break in the loop one without available high pressure and low pressure safety injection system. Accumulators were available in the intact loops. The main aims of this test were to compare the counterpart test data from BETHSY and Large Scale Test Facility (LSTF) facilities and qualification of CATHARE 2 computer code. Further information on BETHSY 6.2TC could be find in [8].
BETHSY 9.1b	2" cold leg break without HPIS and with delayed ultimate procedure	BDBA involves two failures: a break on the cold leg together with a complete failure of the HPIS, combined with a human error regarding the conditions in which the operators start the Ultimate Operating Procedure (UOP). The UOP then consists in depressurizing the primary circuit by means of a full opening of the 3 SG atmospheric steam dumps. Further information on BETHSY 6.2TC could be find in [9].
LSTF	PWR Cold-Leg small-break LOCA with total HPI failure	Cold-leg break tests were conducted at the LSTF for five break areas 0.5%, 1%, 2.5%, 5 and 10% of the scaled cold-leg flow area, with totally failed HPI [10].
LSTF	0.5% cold leg small-break LOCA total failure of the HPI and auxiliary feedwater (AFW) systems	The depressurization procedure was simulated in a 0.5% cold-leg break LOCA experiment [11].

In the scenario LOCA in cold leg together with complete loss of HPI system, the results of BETHSY 9.1b, which is 5.08 cm (2 inch, 0.5%) break on the cold leg together with a complete failure of the HPI system (HPIS), showed that core heatup is obtained without performing depressurization of the primary side through the secondary side. Test data of LSTF and model calculations showed that intentional primary system depressurization with use of the pressurizer power operated relief valves (PORVs) is effective for break areas of approximately 0.5% or less, is unnecessary for breaks of approximately 5% or more, and might be insufficient for intermediate break areas to maintain adequate core cooling. It was also shown that there might be possibility of core dryout after accumulator injection and before LPI system (LPIS) injection for break areas less than approximately 2.5%.

In the scenario LOCA in cold leg together with complete loss of LPI system, the results of BETHSY 6.2TC, which is 15.24 cm (6 inch, ≈5%) break on the cold leg together with a complete failure of the HPIS and LPIS, showed that core heatup is obtained before accumulator injection is started and after accumulator injection is terminated.

The report is organized as follows. In Section 2 the methods used are described. First the LOCA scenarios typical sequence is described. Then the RELAP5 thermal-hydraulic system computer code and input model description are briefly described, followed by the initial and boundary conditions, resulting from steady state calculations. Last, the simulated scenarios include eight break sizes, ranging from 1.27 cm to 30.48 cm equivalent diameter break size in cold leg. Both, LOCA with complete loss of HPIS and LOCA with complete loss of LPIS scenarios were simulated. The simulations include also some scenarios using accident management measures. In Section 3 the results of the LOCA calculations are presented, including discussion of the result. Finally, main conclusions are drawn.

2 METHODS USED

2.1 LOCA Scenario Description

In the LOCAs simulated at the beginning of transient from the emergency core cooling systems (ECCSs) two accumulators were assumed available and either the high pressure or the low pressure emergency core cooling system. The initiating event is opening of the valve simulating the break in the cold leg with reactor operating at 100% power. The reactor trip on (compensated) low pressurizer pressure (12.99 MPa) further causes the turbine trip. The safety injection (SI) signal is generated on the low-low pressurizer pressure signal at 12.27 MPa. On SI signal active safety systems either high pressure safety injection (HPSI) pumps or low pressure safety injection (LPSI) pumps and motor driven (MD) auxiliary feedwater (AFW) pumps start. The HPSI pumps started to inject, when pressure is lower than 15.14 MPa. When pressure drops below 49.55 bars, both accumulators start to inject. Larger is the break size, faster is the accumulator discharge. When pressure drops below 1.13 MPa, the LPSI pumps start to inject. In the case of smaller breaks, the high primary pressure can prevent accumulator and LPSI pumps injection.

2.2 RELAP5 Input Model Description

For calculations the latest RELAP5/MOD3.3 Patch 5 [14] has been used and the RELAP5 input model of Krško Nuclear Power Plant (NPP), used also in study [15]. Krško NPP is a two loop pressurized water reactor (PWR), Westinghouse type, with reactor power uprated to 1994 MW and thermal power 2000 MW. The base model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points. 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 also counted as one component and not as many heat structures as pipe has volumes like it is counted in the RELAP5 output file. This explains the difference in the number of heat structures in Figure 2-1 and that reported in RELAP5 output file.

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 1 and 2) seal flow. Emergency core cooling systems (ECCSs) piping includes two high pressure safety injection (HPSI) pumps, two accumulators (ACC 1 and 2), and two low pressure safety injection (LPSI) pumps. The secondary side consists of the two steam generators (SGs) - secondary side, main steam line, main steam isolation valves (MSIV 1 and 2), SG1 and SG2 relief and safety valves, and main feedwater (MFW1 and MFW2) piping. The turbine valve is modeled by the corresponding logic. Steam dump (SD) is also modeled by the corresponding logic. The turbine is represented by time dependent volume. The MFW and AFW (auxiliary feedwater) motor driven (MD) and turbine driven (TD) pumps are modeled as time dependent junctions.

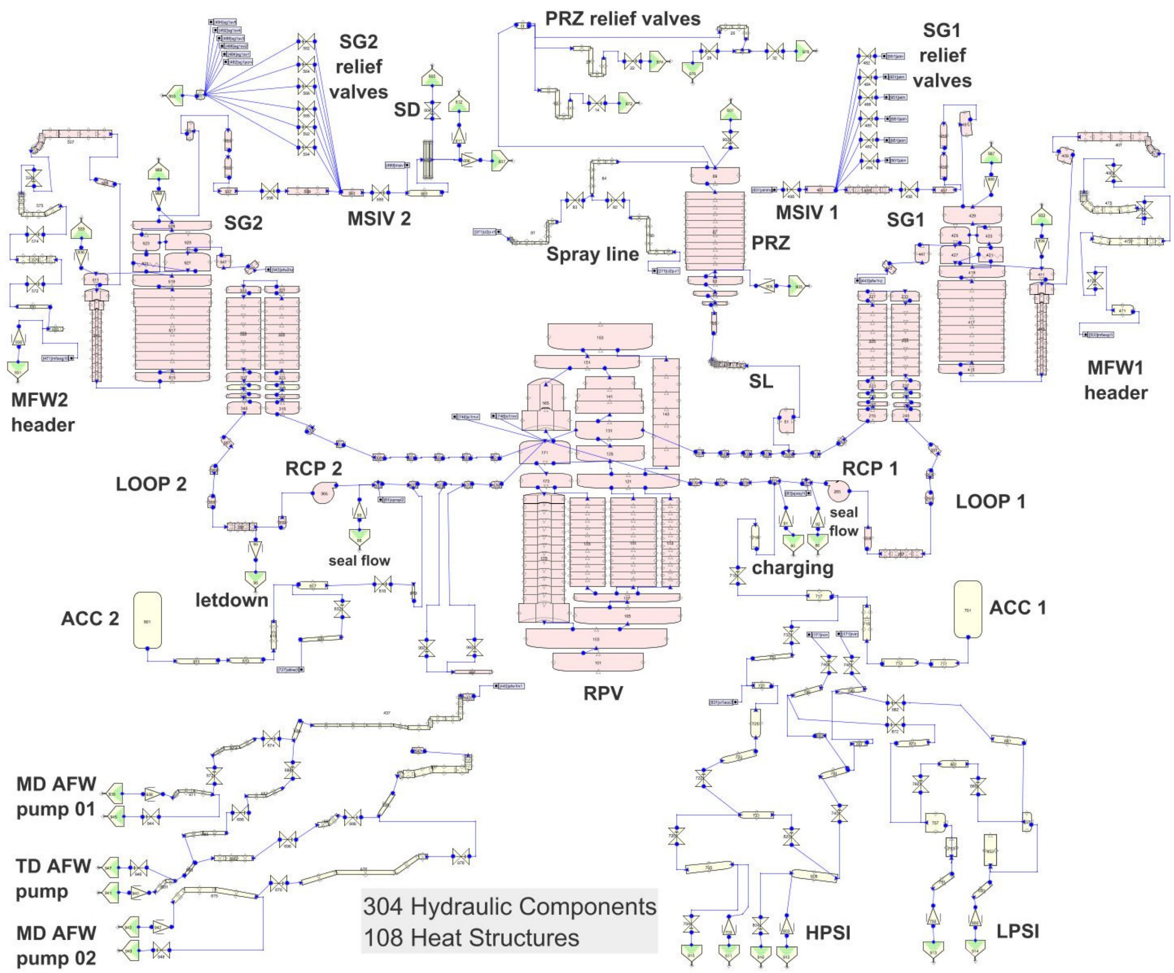

Figure 2-1 RELAP5 Krško NPP Hydraulic Components View

2.3 Initial and Boundary Conditions

Table 2-2 shows initial and boundary conditions at the beginning of simulation. Initial values and boundary conditions are given for both loops (where applicable). It can be seen that RELAP5 initial and boundary conditions are close to reference PWR values with the exception of steam generator pressures, where deviation is about 2.5%. Nevertheless, in LOCA calculations the influence of the secondary side is typically smaller than in the secondary side initiated transients because due to break the primary side empties and the natural circulation is terminated.

Table 2-2 Initial and Boundary Conditions

Parameter (unit)	Reference PWR	RELAP5/MOD3.3
Core power (MW)	1994	1994
Pressurizer pressure (MPa)	15.513	15.513
Pressurizer level (%)	55.7	55.8
Average RCS temperature no. 1 (K)	578.15	578.15
Average RCS temperature no. 2 (K)	578.15	578.06
Cold leg temperature no. 1 (K)	558.75	559.51
Cold leg temperature no. 2 (K)	558.75	559.32
Hot leg temperature no. 1 (K)	597.55	596.79
Hot leg temperature no. 2 (K)	597.55	596.79
Cold leg flow no. 1 (kg/s)	4694.7	4721.2
Cold leg flow no. 2 (kg/s)	4694.7	4719.6
Steam generator pressure no. 1 (MPa)	6.281	6.438
Steam generator pressure no. 2 (MPa)	6.281	6.415
Steam generator NR level no. 1 (%)	69.3	69.3
Steam generator NR level no. 2 (%)	69.3	69.3
Steam flow no. 1 (kg/s)	544.5	541.3
Steam flow no. 2 (kg/s)	544.5	544.5
Main feedwater temperature (K)	492.6	492.8

2.4 Simulated LOCA Break Cases

The breaks simulated were 1.27 cm (0.5 inch), 2.54 cm (1 inch), 5.08 cm (2 inch), 7.62 cm (3 inch), 10.16 cm (4 inch), 15.24 cm (6 inch), 20.32 cm (8 inch) and 30.48 cm (12 inch) equivalent diameter cold leg breaks. For each break size two simulations were performed, one without HPSI pumps and one without LPSI pumps, with other systems available as can be seen from Table 2-3. In all simulations default values for break flows were used for Ransom Trapp critical flow model.

Table 2-3 LOCA Scenario Cases Without One ECCS (HPIS or LPIS)

Break size diameter	case without HPSI pumps	case without LPSI pumps
1.27 cm (0.5 inch)	sb0.5_noHP	sb0.5_noLP
2.54 cm (1 inch)	sb1_noHP	sb1_noLP
5.08 cm (2 inch)	sb2_noHP	sb2_noLP
7.62 cm (3 inch)	sb3_noHP	sb3_noLP
10.16 cm (4 inch)	sb4_noHP	sb4_noLP
15.24 cm (6 inch)	sb6_noHP	sb6_noLP
20.32 cm (8 inch)	sb8_noHP	sb8_noLP
30.48 cm (12 inch)	sb12_noHP	sb12_noLP

In addition to scenarios (i.e. DEC), which could not be mitigated, scenarios with accident management (AM) measures were also simulated, shown in Table 2-4. The first accident management strategy AM1 was the use of pressurizer (PRZ) power operated relief valve (PORV) to depressurize reactor coolant system (RCS) to 1.1 MPa. Second AM strategy (AM2) was to depressurize the RCS through the secondary side, using SG PORVs to depressurize

SGs to 1.2 MPa. Third AM strategy (AM3) was to depressurize the RCS through the secondary side, using SG PORVs to depressurize SGs to 0.7 MPa.

Table 2-4 LOCA Scenario Cases Without One ECCS (HPIS or LPIS) and Using Accident Management Strategies

Break size diameter	accident management (AM) measures	case without HPIS pumps
1.27 cm (0.5 inch)	AM1: using PRZ PORV to depressurize RCS to 1.1 MPa	sb0.5_noHPd
2.54 cm (1 inch)	AM1: using PRZ PORV to depressurize RCS to 1.1 MPa	sb1_noHPd
5.08 cm (2 inch)	AM1: using PRZ PORV to depressurize RCS to 1.1 MPa	sb2_noHPd
1.27 cm (0.5 inch)	AM2: using SG PORVs to depressurize SGs to 1.2 MPa	sb0.5_noHPdsg
2.54 cm (1 inch)	AM2: using SG PORVs to depressurize SGs to 1.2 MPa	sb1_noHPdsg
5.08 cm (2 inch)	AM2: using SG PORVs to depressurize SGs to 1.2 MPa	sb2_noHPdsg
1.27 cm (0.5 inch)	AM3: using SG PORVs to depressurize SGs to 0.7 MPa	sb0.5_noHPdsg7
2.54 cm (1 inch)	AM3: using SG PORVs to depressurize SGs to 0.7 MPa	sb1_noHPdsg7
5.08 cm (2 inch)	AM3: using SG PORVs to depressurize SGs to 0.7 MPa	sb2_noHPdsg7

3 RESULTS

The results of LOCA break spectrum calculations are shown in Figures 3-1 through 3-63. For each break size the following nine parameters are shown: pressurizer pressure, SG1 pressure, core collapsed liquid level, cladding temperature, RCS mass, integrated break flow, accumulators integrated break flow, HPIS integrated break flow and LPIS integrated break flow. The pressurizer pressure is shown to know when we are below the setpoint for ECCSs injection. Secondary pressure is important when accident management strategies are performed. Core collapsed liquid level gives information on core uncovery and cladding temperature about core heatup. Finally, masses are shown for RCS inventory, mass inventory discharged through the break and mass injected by accumulators, HPIS and LPIS.

3.1 Results of LOCA Break Size Spectrum Without HPIS (noHP)

The results for simulations of LOCA spectrum ranging from 1.27 cm to 30.48 cm with assumed loss of HPI system are shown in Figures 3-1 to 3-18. The simulations were performed for 24 h (86400 s), if simulation was not terminated earlier due to significant heatup. It is shown that in case of loss of HPI system the breaks smaller than 7.62 cm lead to core heatup of average fuel rod cladding above 1100 K and/or low primary system mass with LPIS not injecting due to primary pressure being above the injection setpoint. This mean, that smaller break LOCAs are DEC for selected PWR. For breaks equal or larger 10.16 cm it was shown that cooling through the break is sufficient and LPI system has sufficient capacity in long term. This is in accordance with results obtained from BETHSY 9.1b experiment [9].

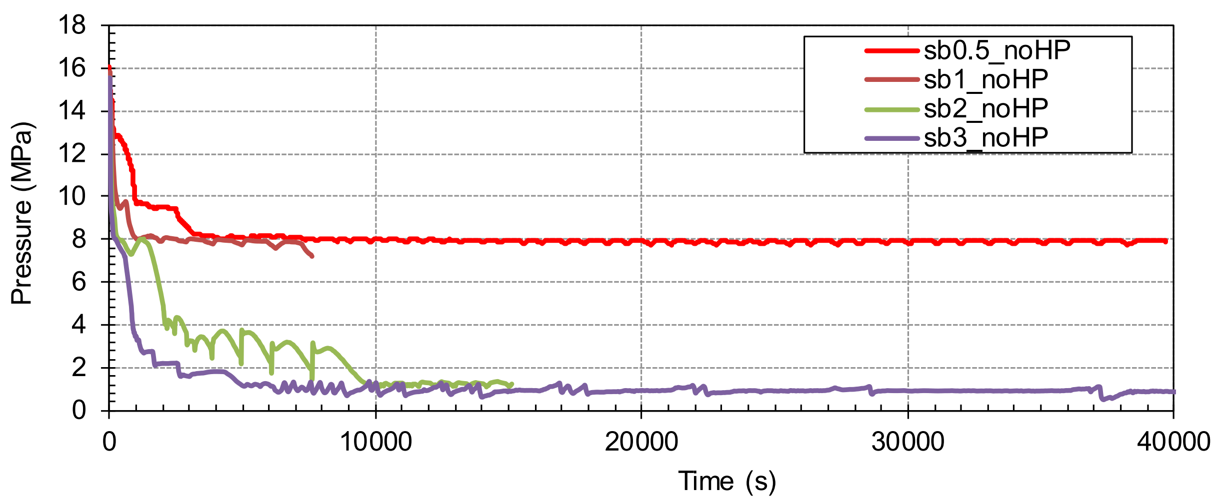

Figure 3-1 Pressurizer Pressure – noHP (1.27 cm to 7.62 cm)

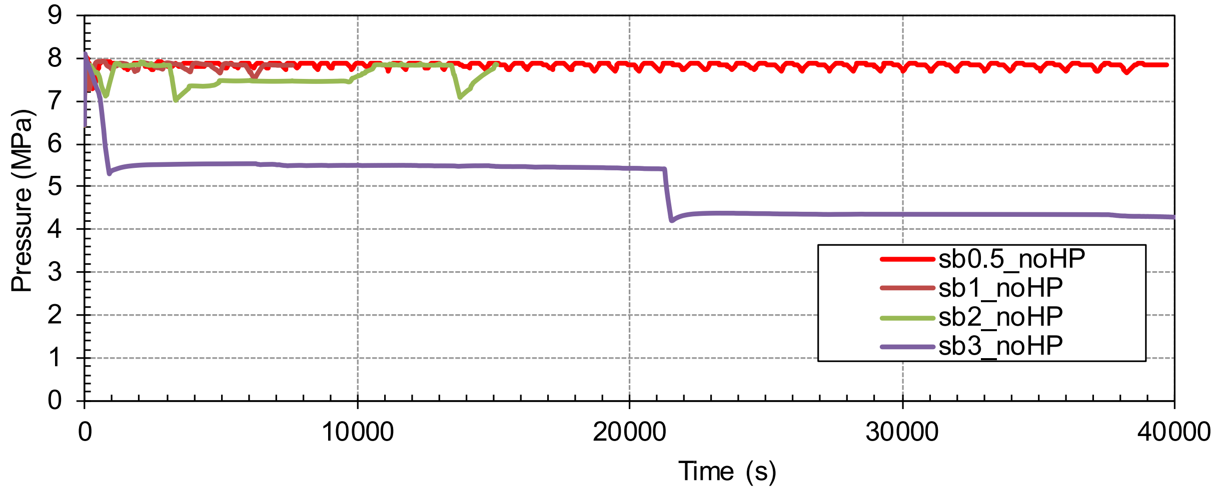

Figure 3-2 SG1 Pressure – noHP (1.27 cm to 7.62 cm)

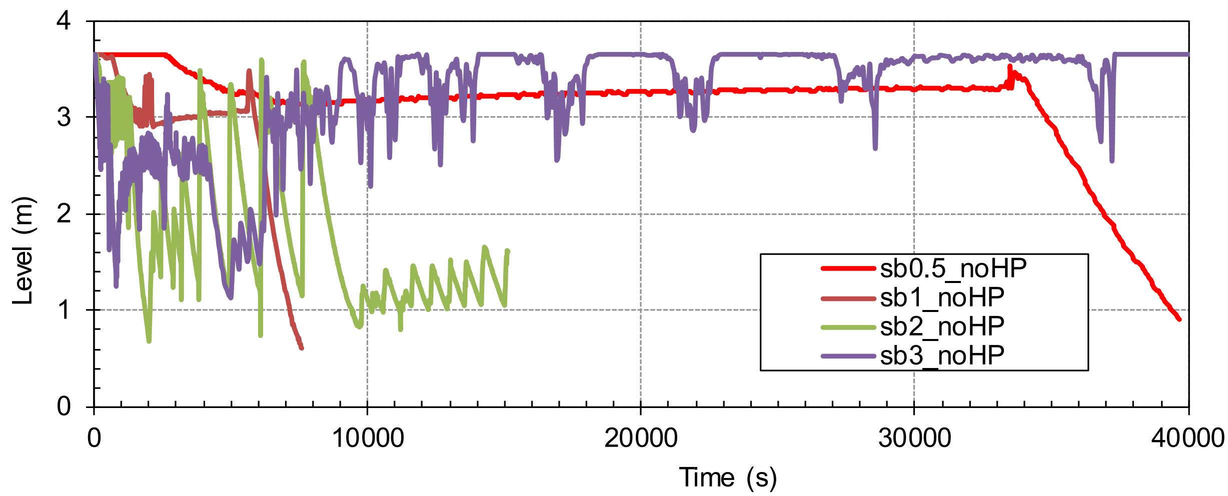

Figure 3-3 Core Collapsed Liquid Level – noHP (1.27 cm to 7.62 cm)

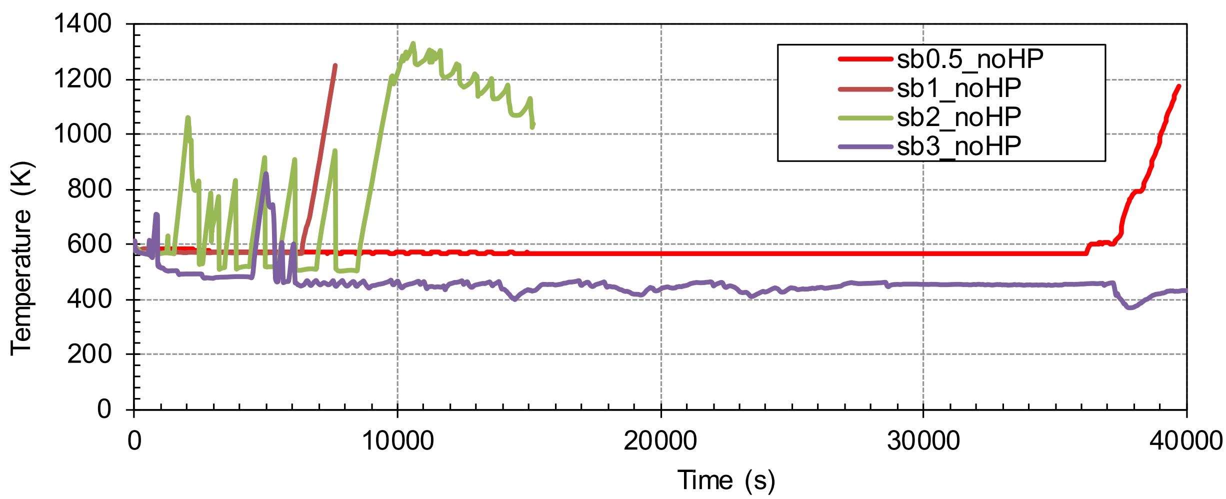

Figure 3-4 Cladding Temperature – noHP (1.27 cm to 7.62 cm)

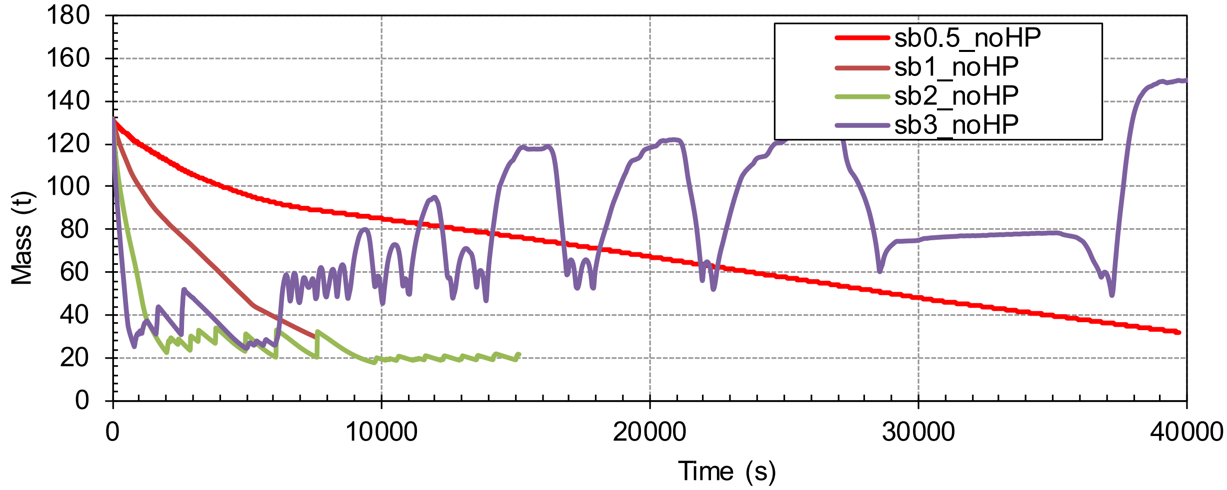

Figure 3-5 Primary System Mass – noHP (1.27 cm to 7.62 cm)

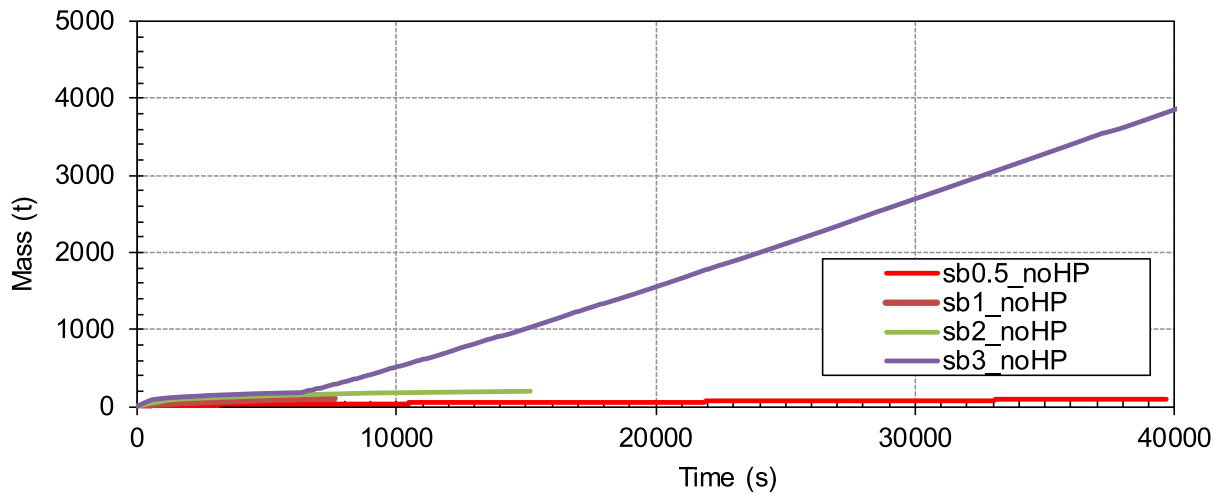

Figure 3-6 Total Integrated Break Flow – noHP (1.27 cm to 7.62 cm)

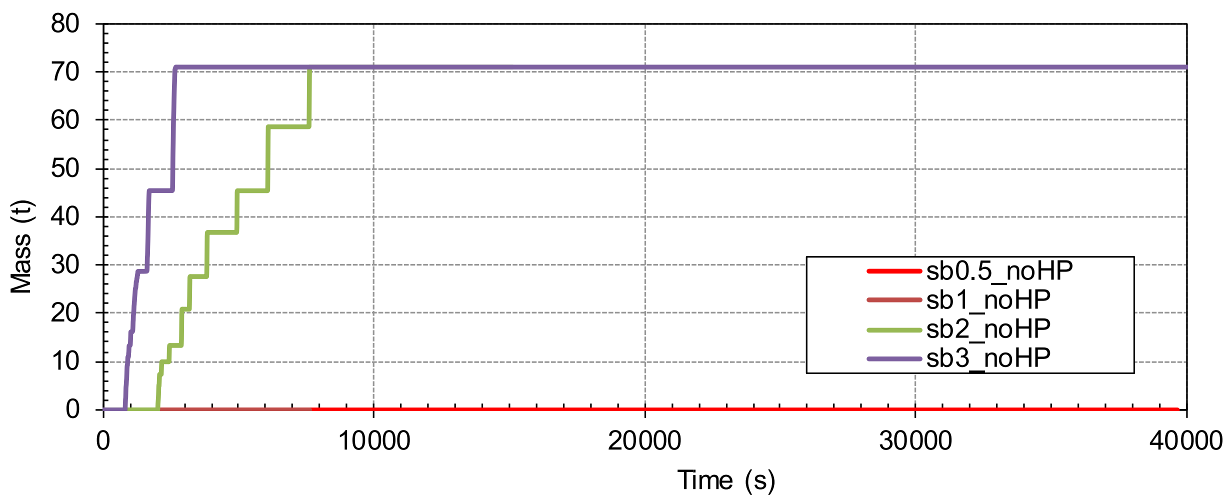

Figure 3-7 Total Integrated Accumulator Flow – noHP (1.27 cm to 7.62 cm)

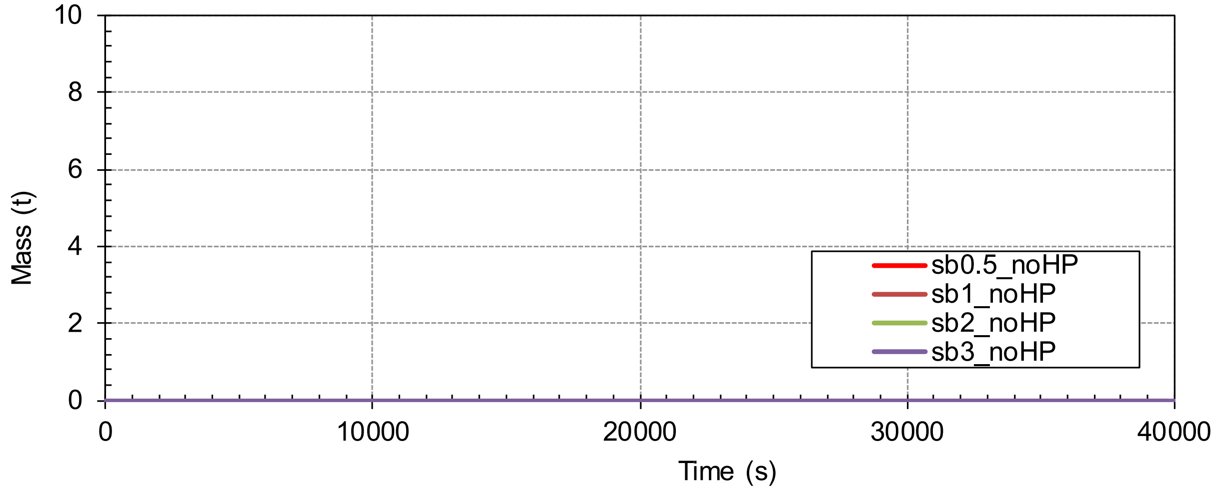

Figure 3-8 Total Integrated HPIS Flow – noHP (1.27 cm to 7.62 cm)

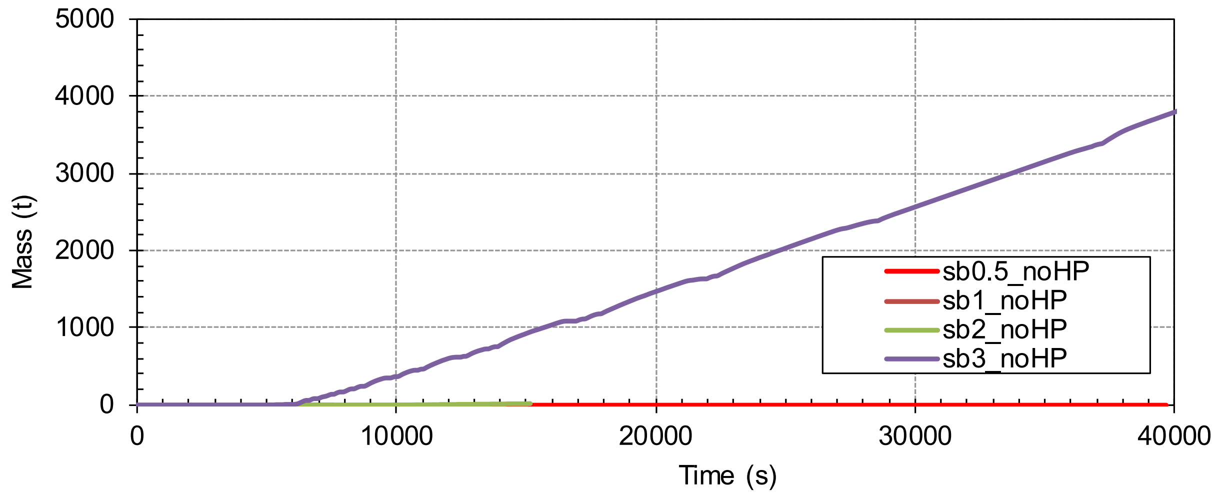

Figure 3-9 Total Integrated LPIS Flow – noHP (1.27 cm to 7.62 cm)

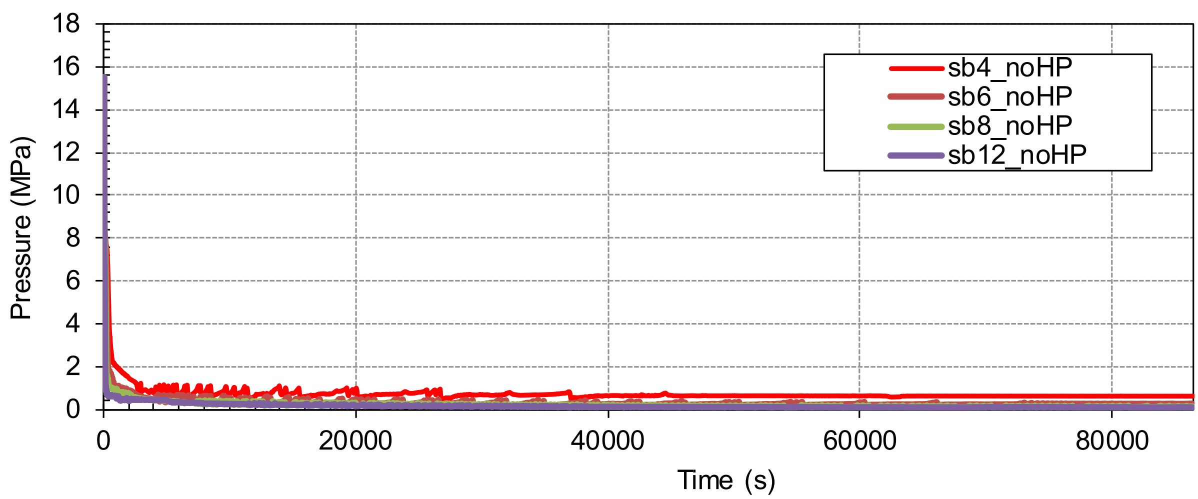

Figure 3-10 Pressurizer Pressure – noHP (10.16 cm to 30.48 cm)

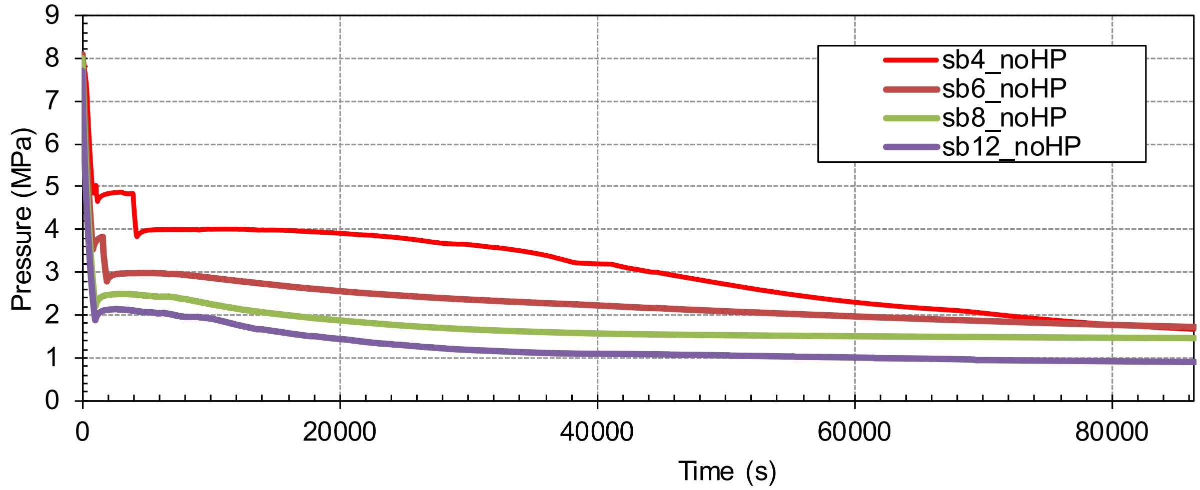

Figure 3-11 SG1 Pressure – noHP (10.16 cm to 30.48 cm)

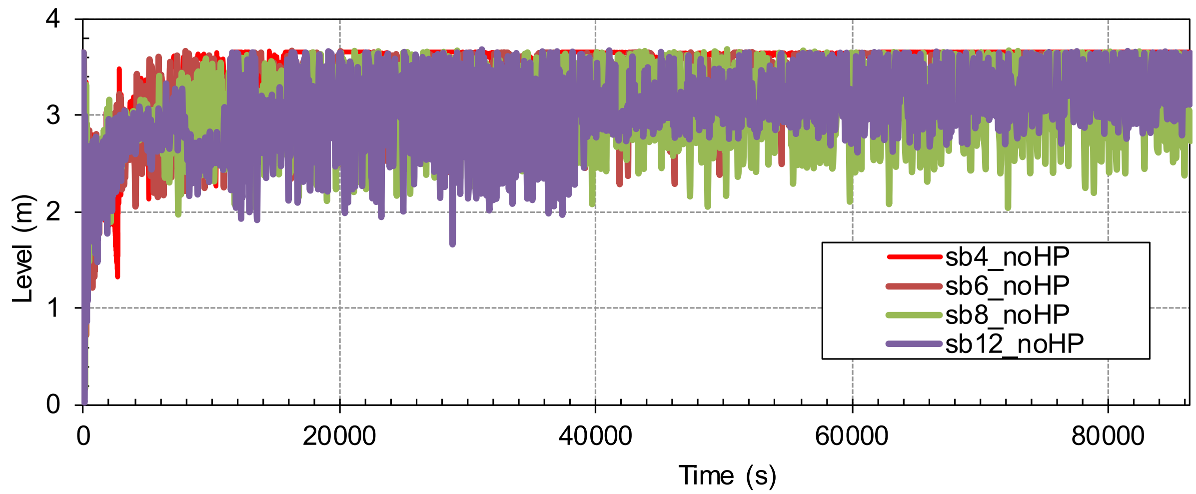

Figure 3-12 Core Collapsed Liquid Level – noHP (10.16 cm to 30.48 cm)

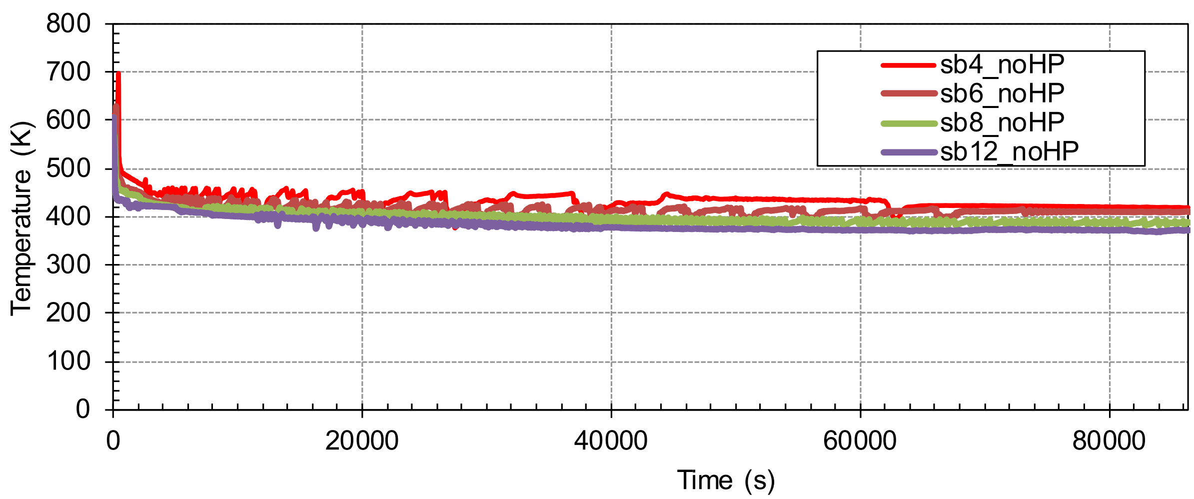

Figure 3-13 Cladding Temperature – noHP (10.16 cm to 30.48 cm)

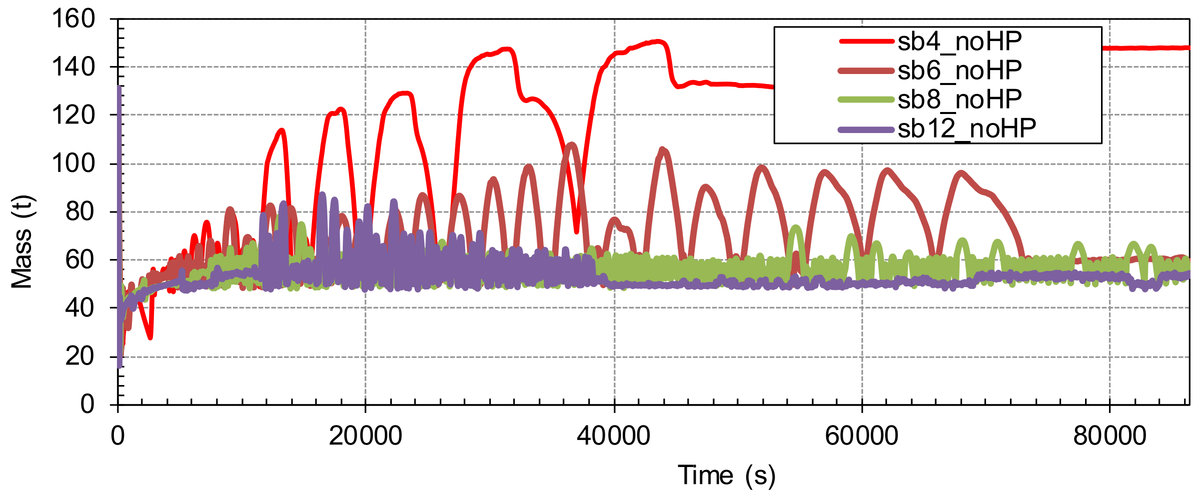

Figure 3-14 Primary System Mass – noHP (10.16 cm to 30.48 cm)

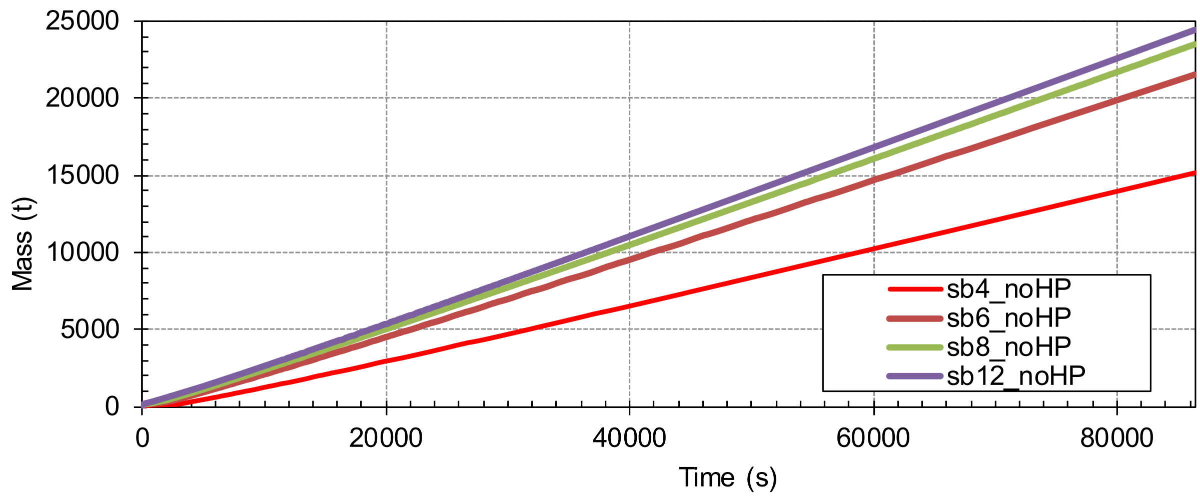

Figure 3-15 Total Integrated Break Flow – noHP (10.16 cm to 30.48 cm)

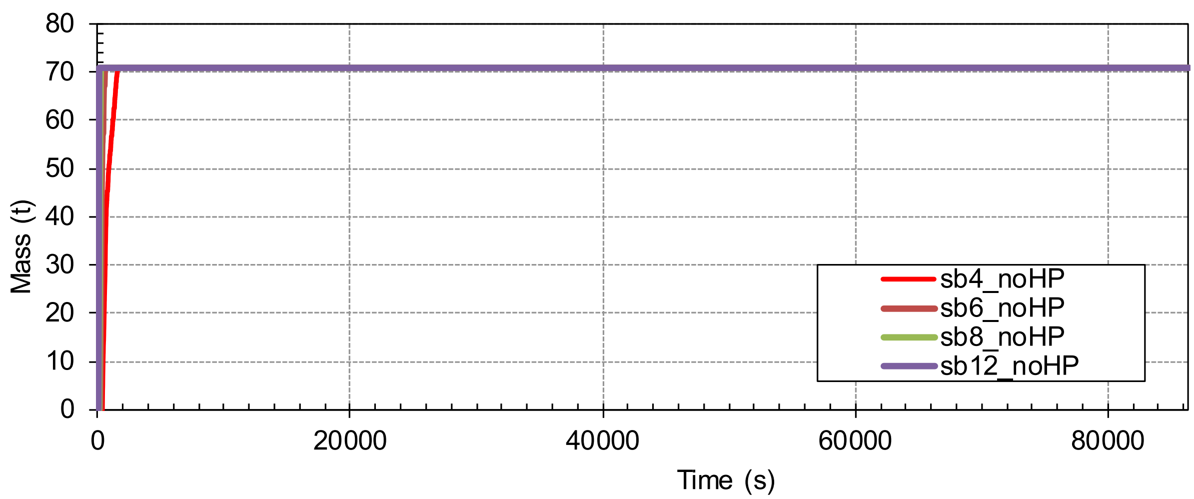

Figure 3-16 Total Integrated Accumulator Flow – noHP (10.16 cm to 30.48 cm)

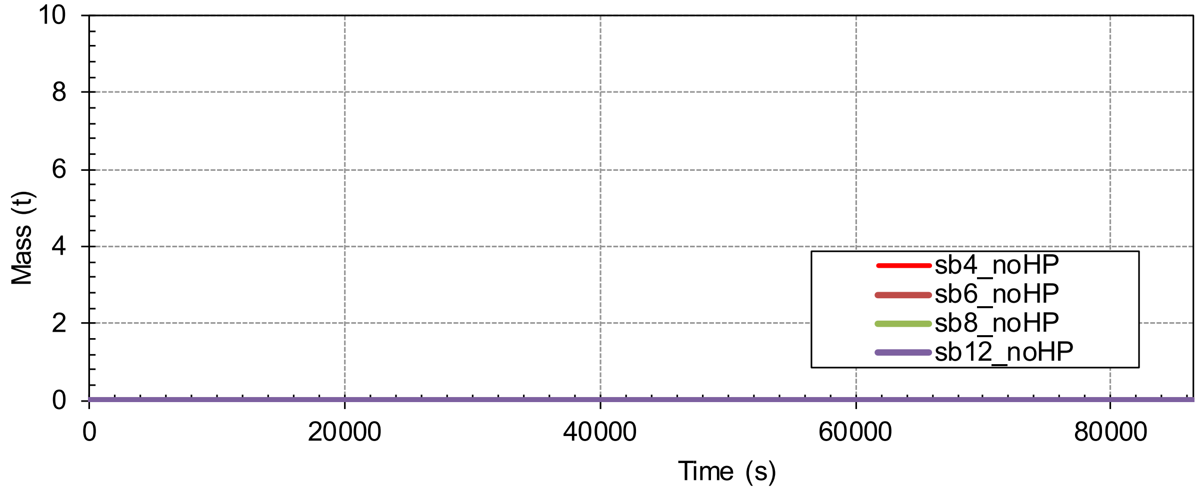

Figure 3-17 Total Integrated HPIS Flow – noHP (10.16 cm to 30.48 cm)

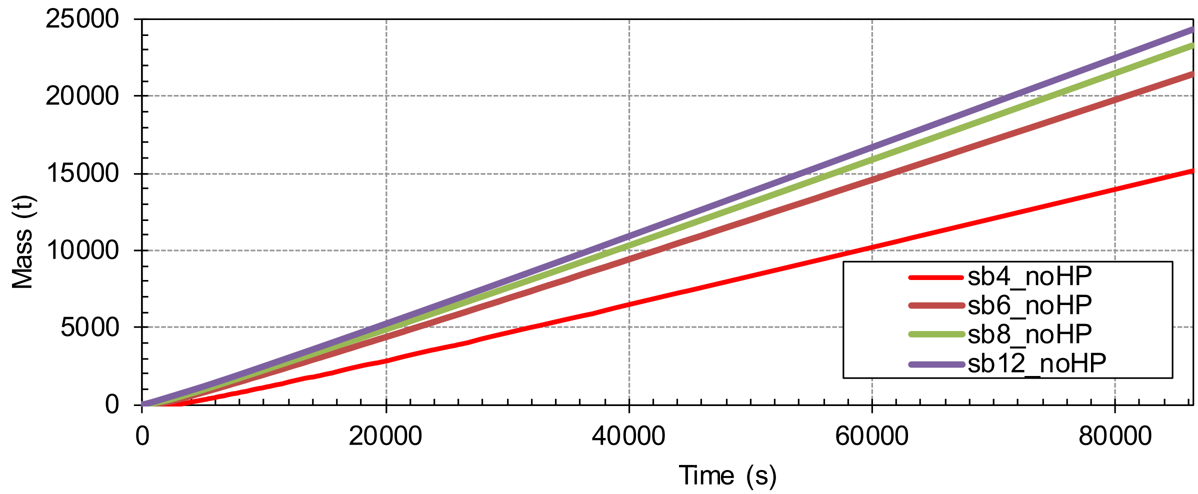

Figure 3-18 Total Integrated LPIS Flow – noHP (10.16 cm to 30.48 cm)

3.2 Results of LOCA Break Size Spectrum Without LPIS (noLP)

The results for simulations of LOCA spectrum ranging from 1.27 cm to 30.48 cm with assumed loss of LPI system are shown in Figures 3-19 to 3-36. The simulation time was 24 h (86400 s).

It is shown that in case of loss of LPI system the HPI system has sufficient capacity in long term for all break sizes for the selected PWR.

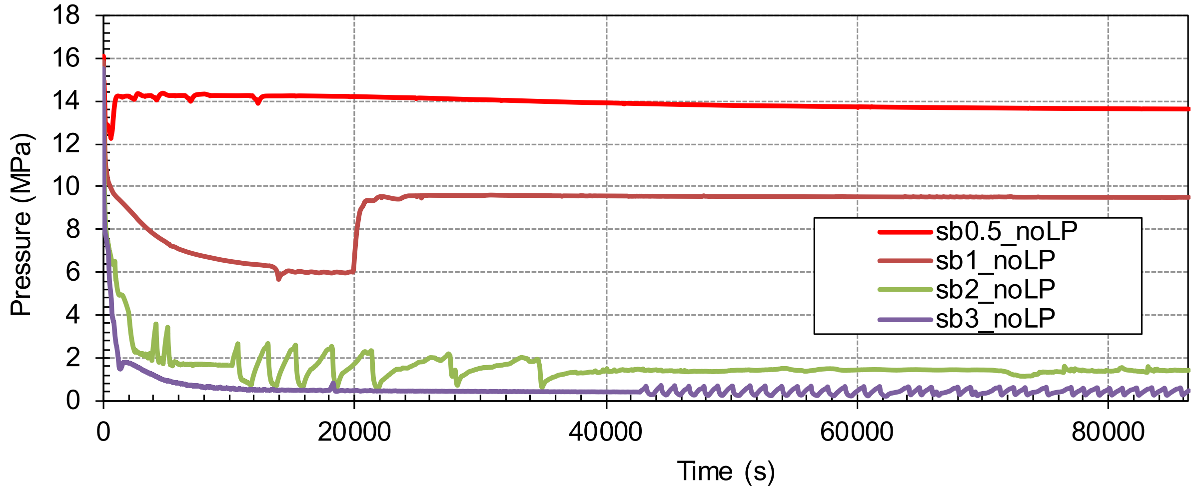

Figure 3-19 Pressurizer Pressure – noLP (1.27 cm to 7.62 cm)

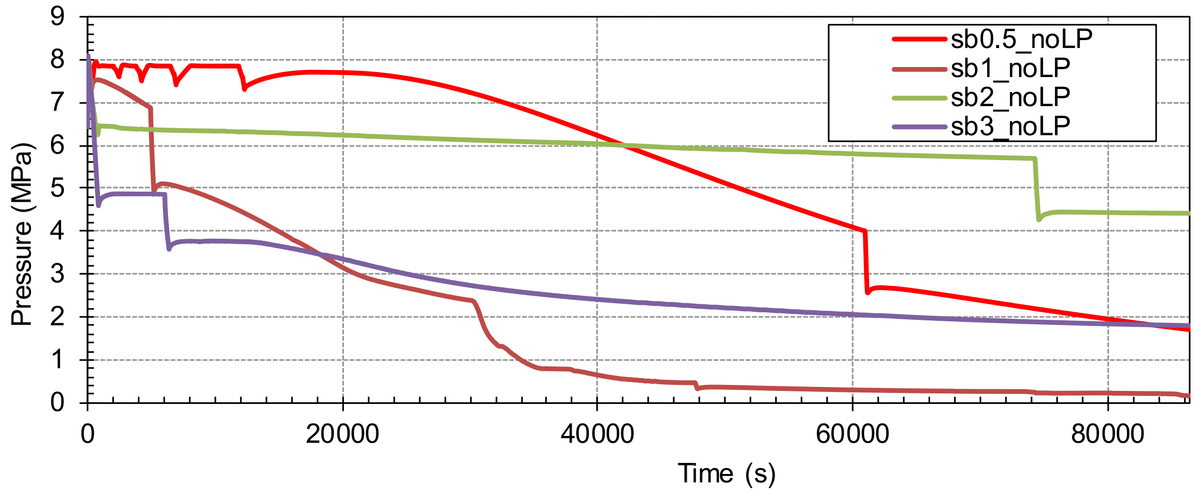

Figure 3-20 SG1 Pressure – noLP (1.27 cm to 7.62 cm)

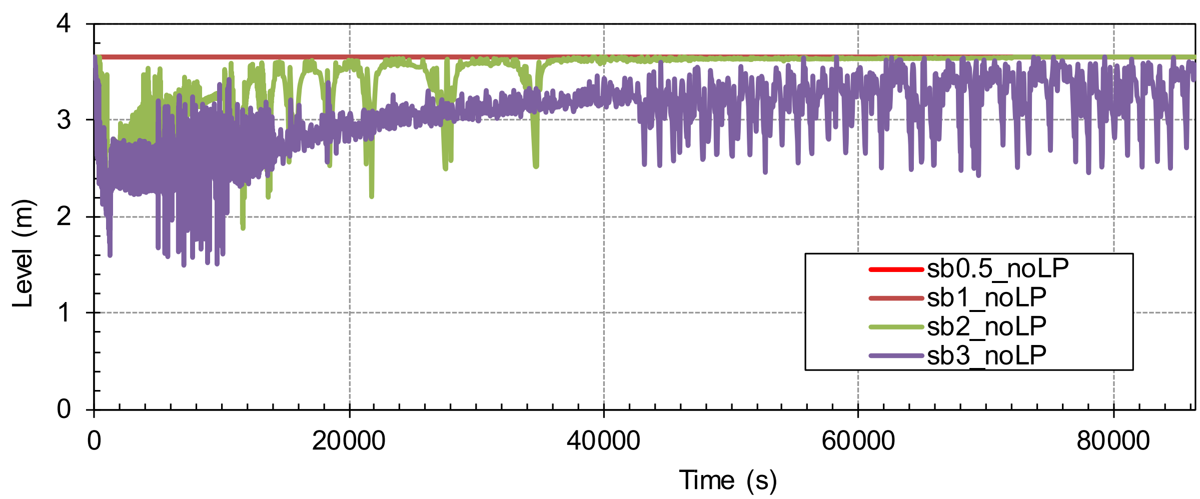

Figure 3-21 Core Collapsed Liquid Level – noLP (1.27 cm to 7.62 cm)

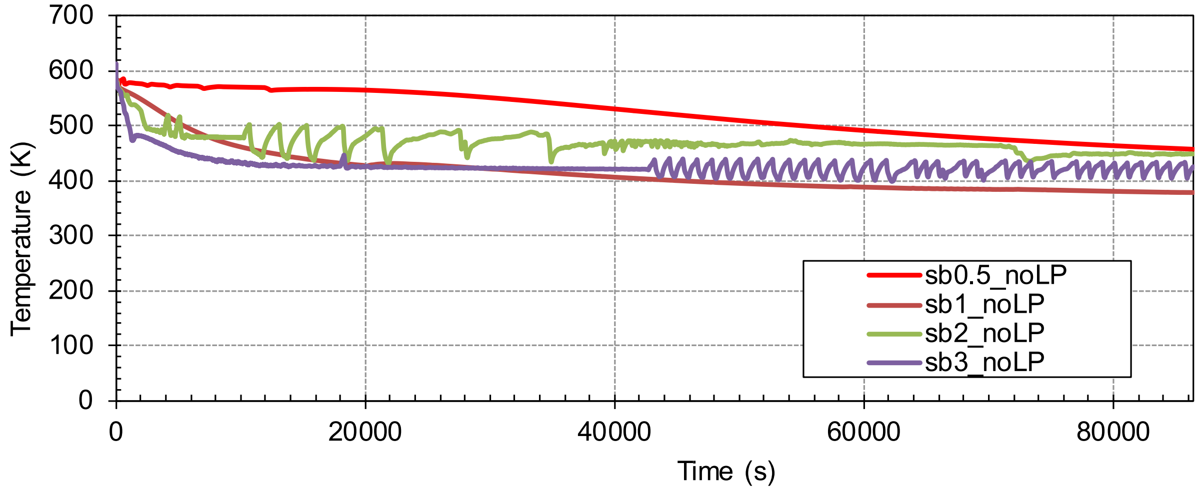

Figure 3-22 Cladding Temperature – noLP (1.27 cm to 7.62 cm)

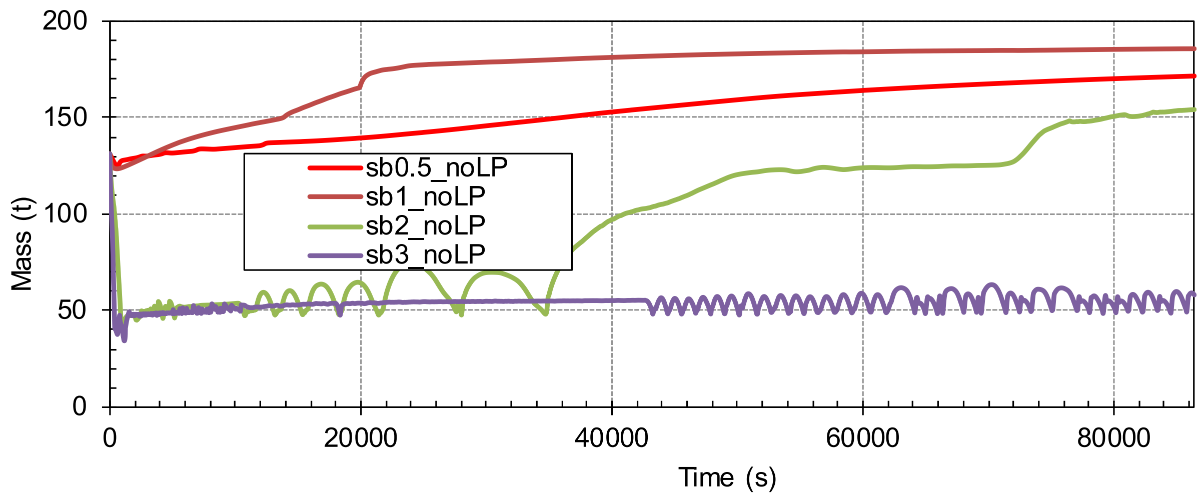

Figure 3-23 Primary System Mass – noLP (1.27 cm to 7.62 cm)

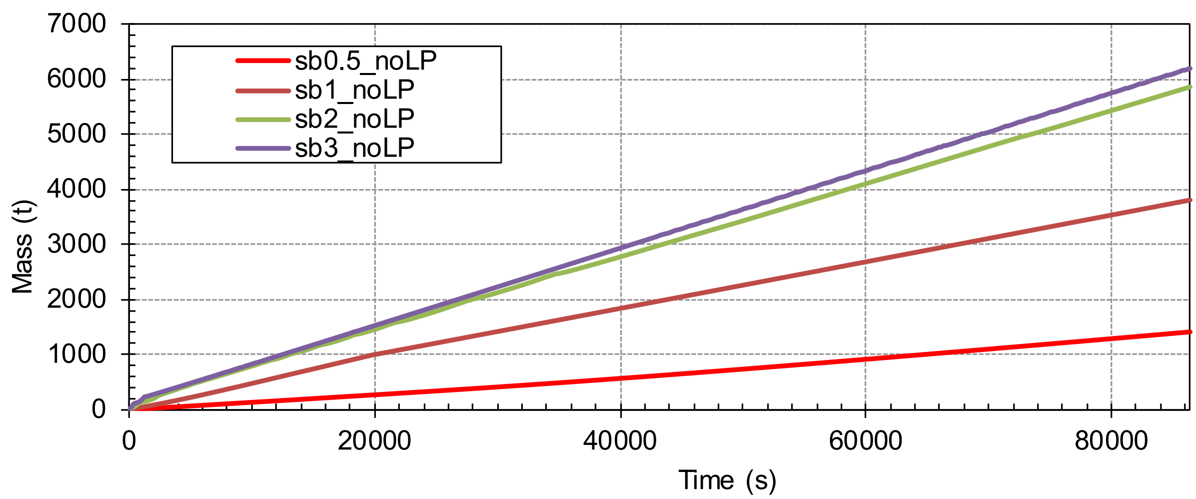

Figure 3-24 Total Integrated Break Flow – noLP (1.27 cm to 7.62 cm)

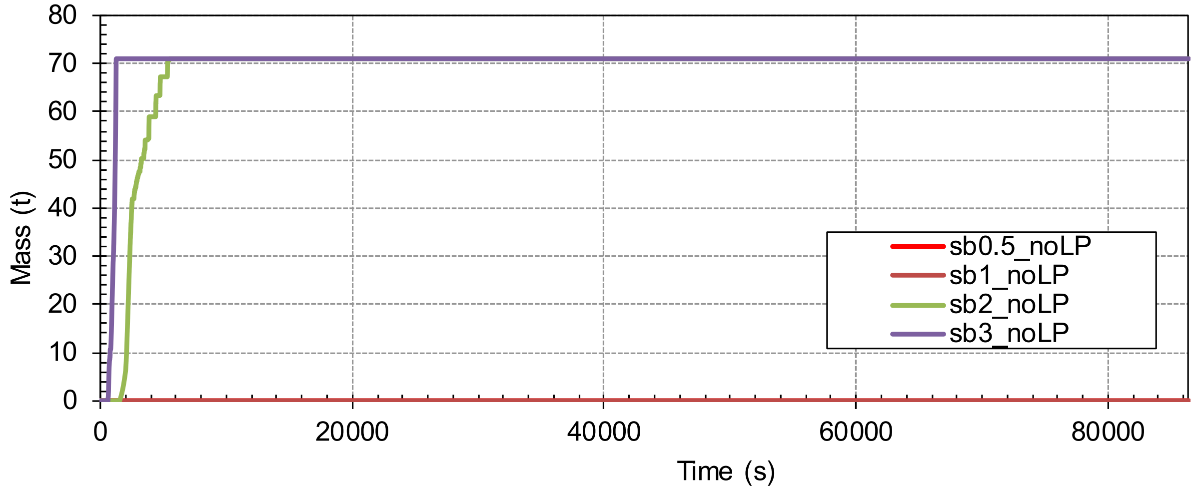

Figure 3-25 Total Integrated Accumulator Flow – noLP (1.27 cm to 7.62 cm)

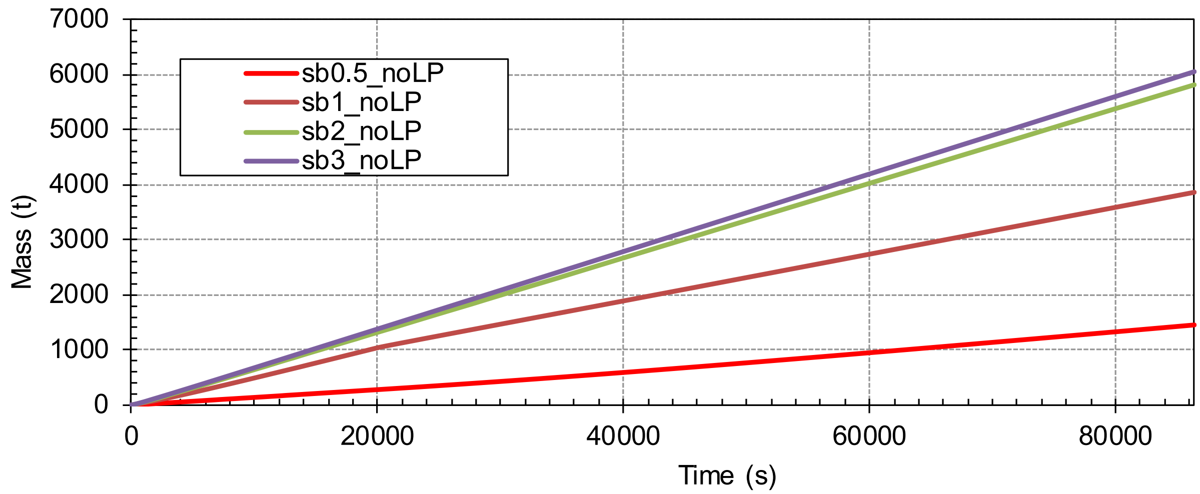

Figure 3-26 Total Integrated HPIS Flow – noLP (1.27 cm to 7.62 cm)

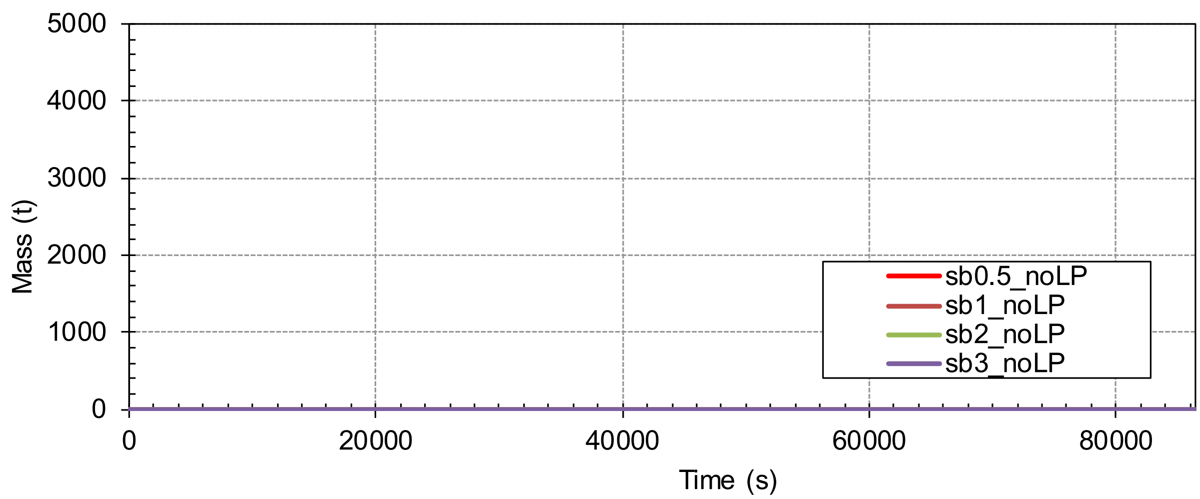

Figure 3-27 Total Integrated LPIS Flow – noLP (1.27 cm to 7.62 cm)

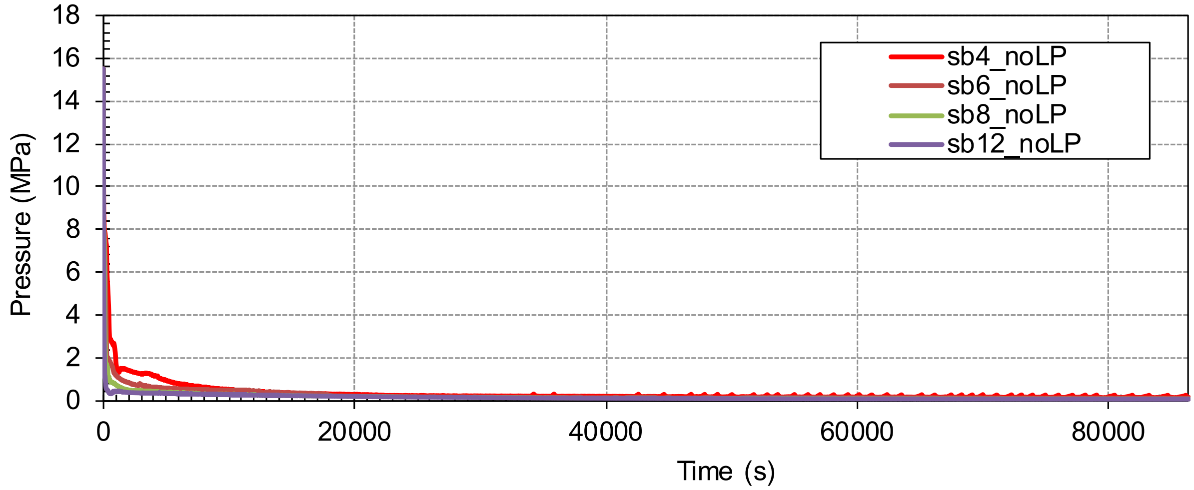

Figure 3-28 Pressurizer Pressure – noLP (10.16 cm to 30.48 cm)

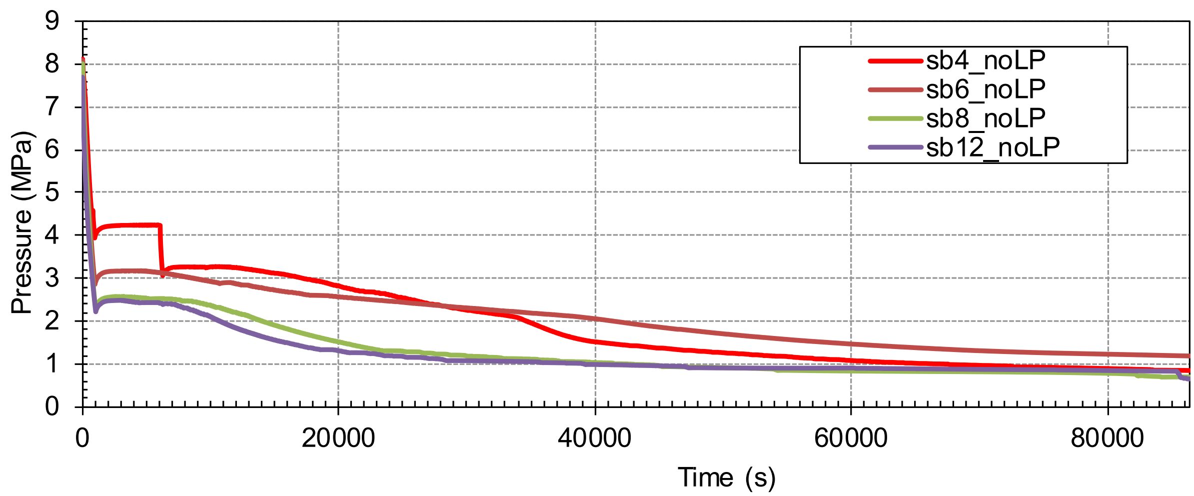

Figure 3-29 SG1 Pressure – noLP (10.16 cm to 30.48 cm)

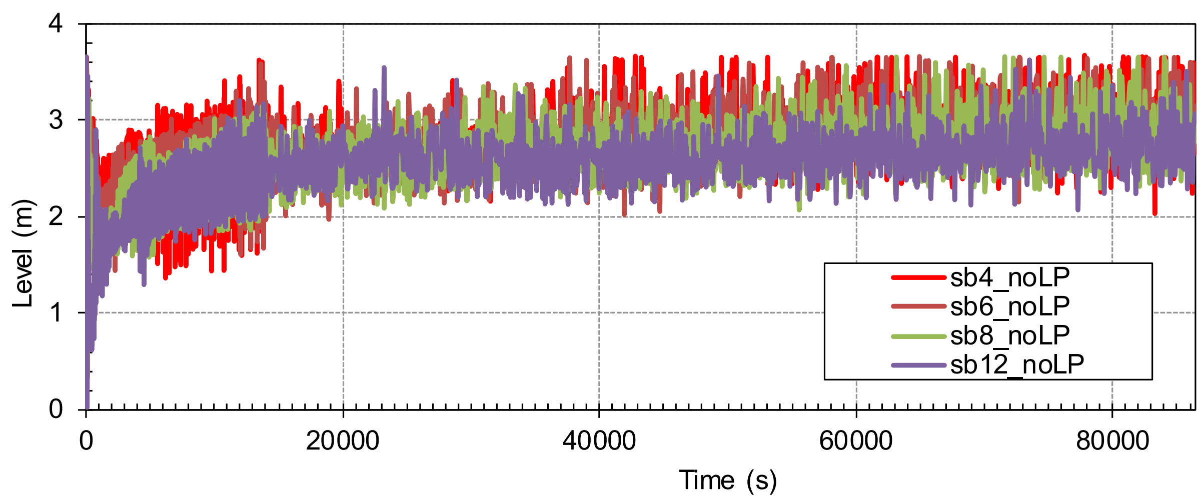

Figure 3-30 Core Collapsed Liquid Level – noLP (10.16 cm to 30.48 cm)

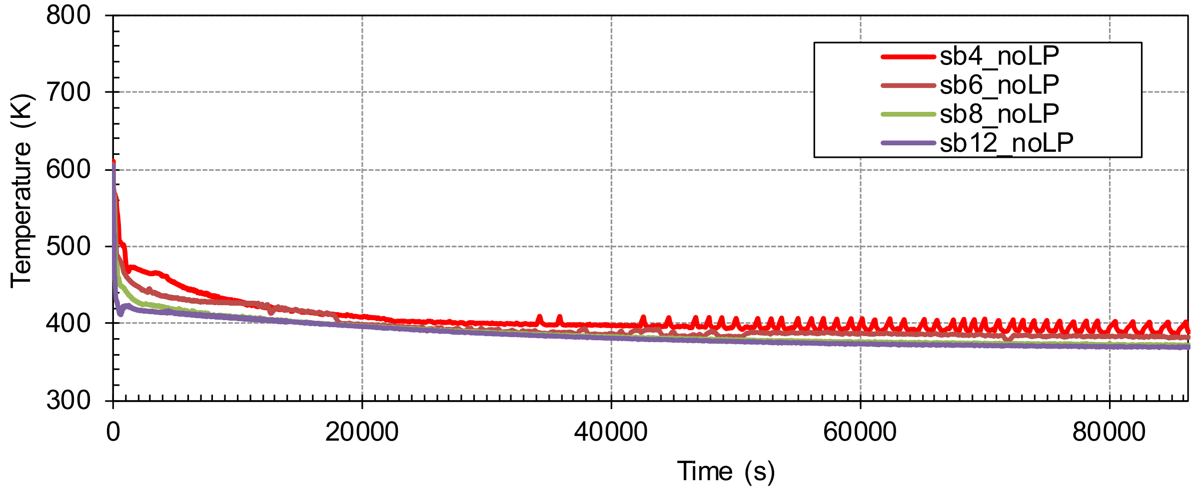

Figure 3-31 Cladding Temperature – noLP (10.16 cm to 30.48 cm)

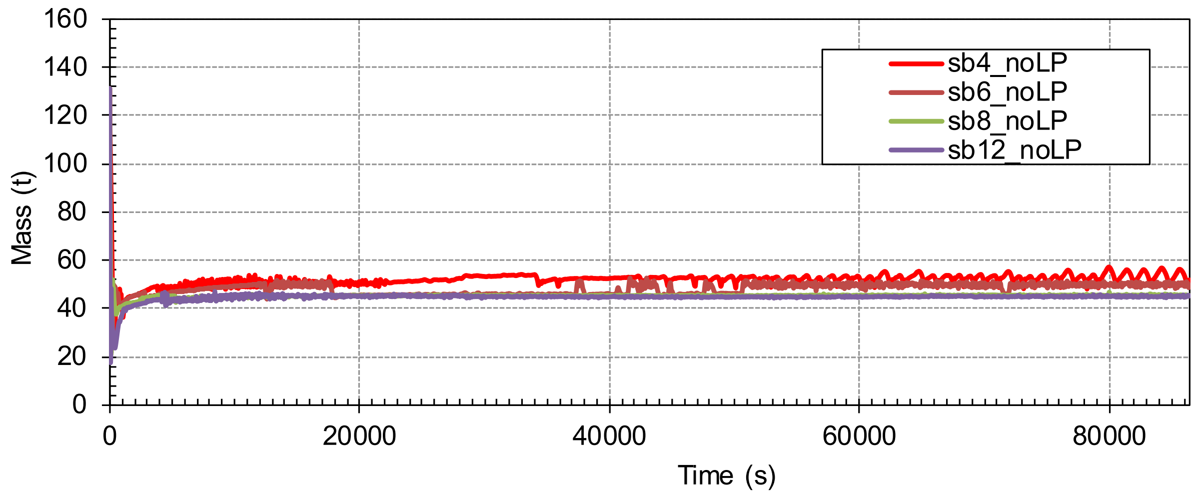

Figure 3-32 Primary System Mass – noLP (10.16 cm to 30.48 cm)

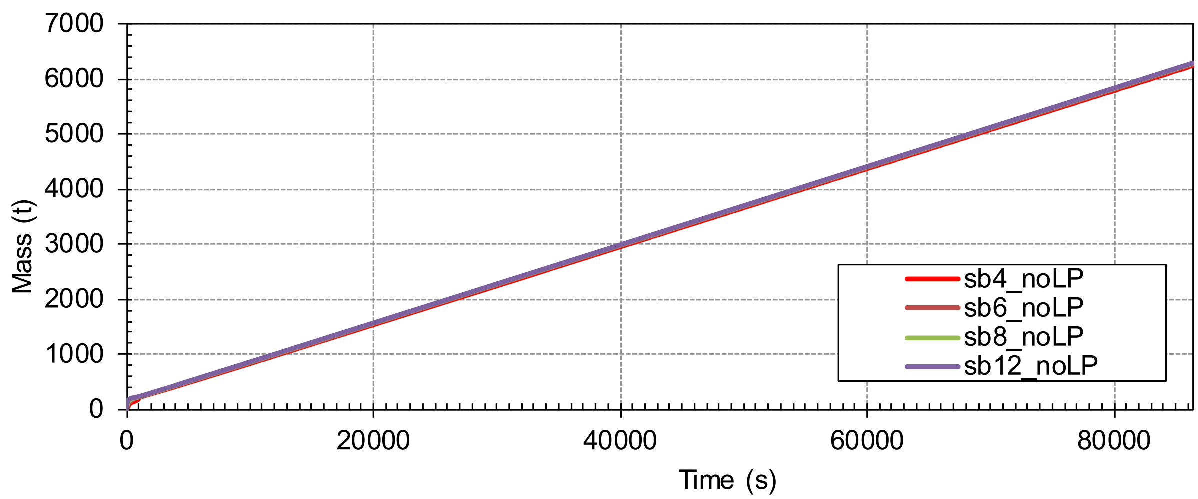

Figure 3-33 Total Integrated Break Flow – noLP (10.16 cm to 30.48 cm)

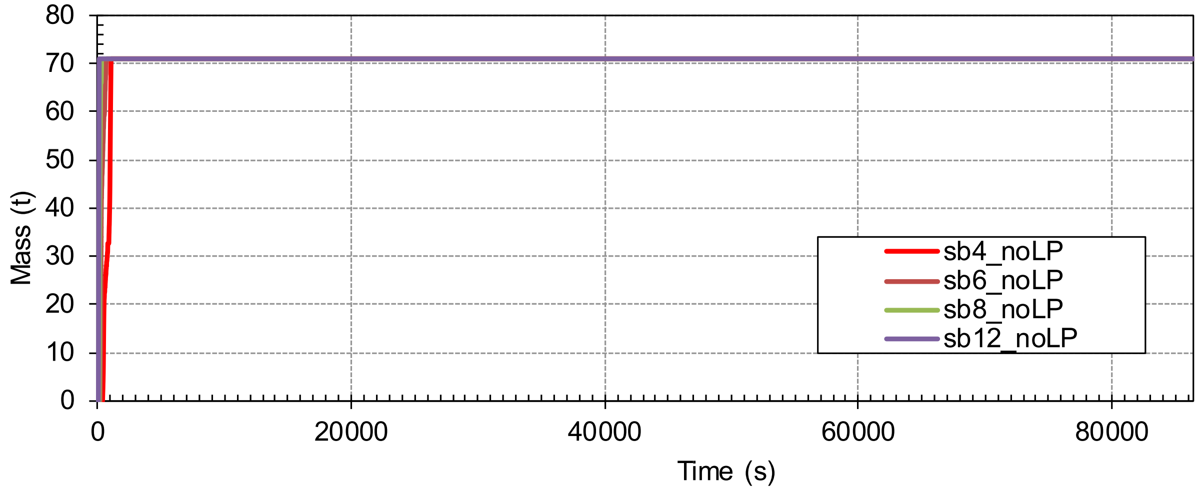

Figure 3-34 Total Integrated Accumulator Flow – noLP (10.16 cm to 30.48 cm)

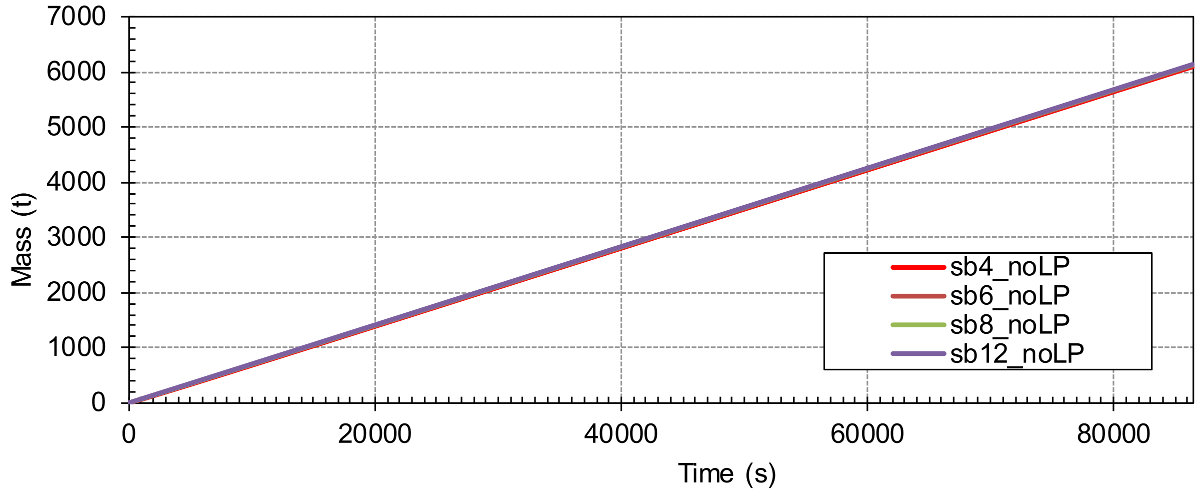

Figure 3-35 Total Integrated HPIS Flow – noLP (10.16 cm to 30.48 cm)

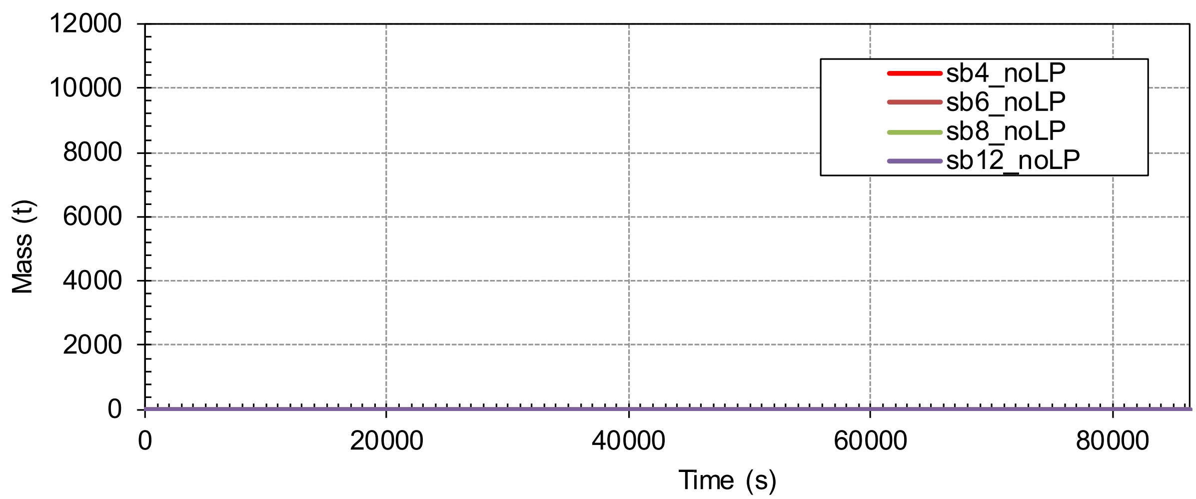

Figure 3-36 Total Integrated LPIS Flow – noHP (10.16 cm to 30.48 cm)

3.3 Results of LOCA Break Size Spectrum 1.27 cm to 5.08 cm Without HPIS and Using AM1 Strategy (noHP and AM1)

In section 3.1 it is shown that in case of loss of HPI system the breaks smaller than 7.62 cm lead to final core uncovery, because primaty pressure is above injection setpoint of the accumulators and LPI system. Therefore AM1 accident management strategy to reduce the primary pressure by one PRZ PORV has been assumed in the simulations. The results are shown in Figures 3-37 to 3-45. As can be seen the strategy AM1 using one PRZ PORV to depressurize the primary system has not been successful for break spectrum from 1.27 cm to 5.08 cm.

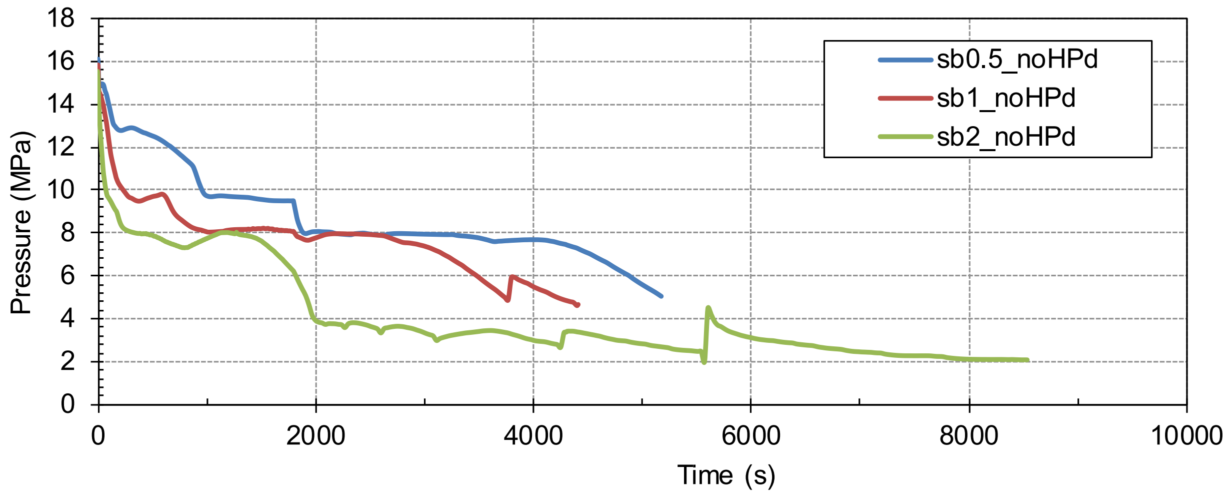

Figure 3-37 Pressurizer Pressure – noHP and AM1 (1.27 cm to 5.08 cm)

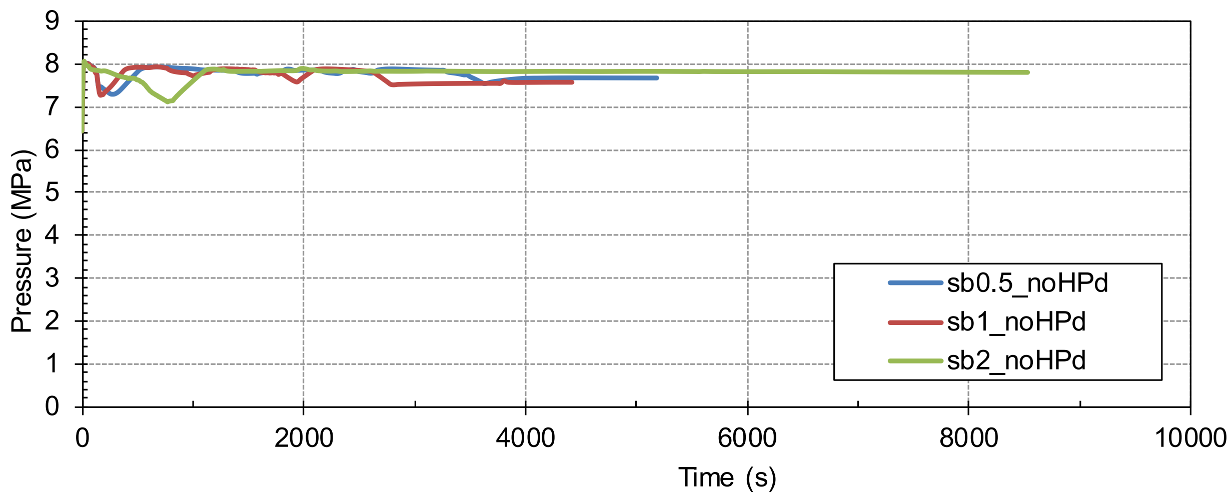

Figure 3-38 SG1 Pressure – noHP and AM1 (1.27 cm to 5.08 cm)

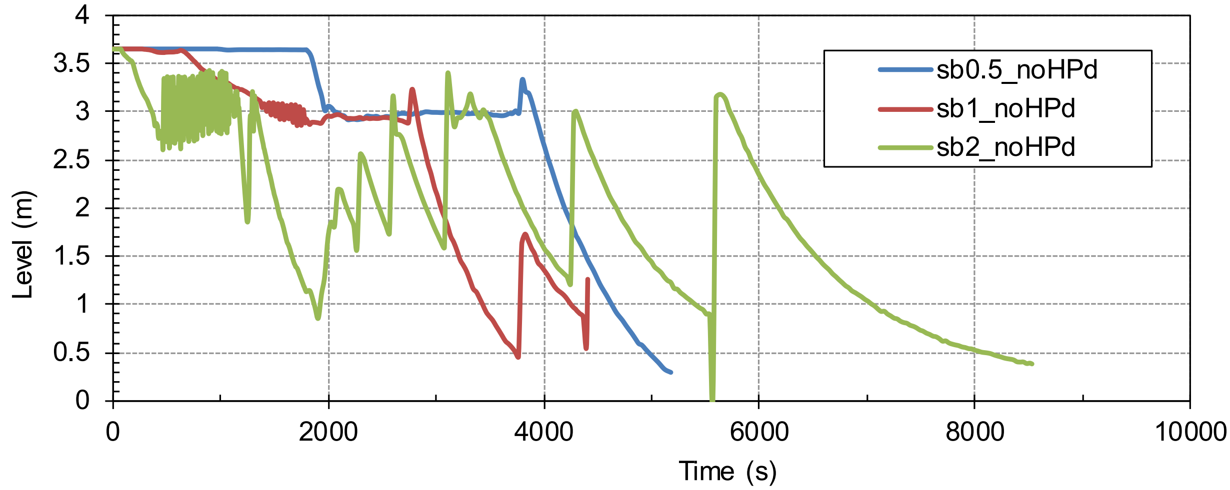

Figure 3-39 Core Collapsed Liquid Level – noHP and AM1 (1.27 cm to 5.08 cm)

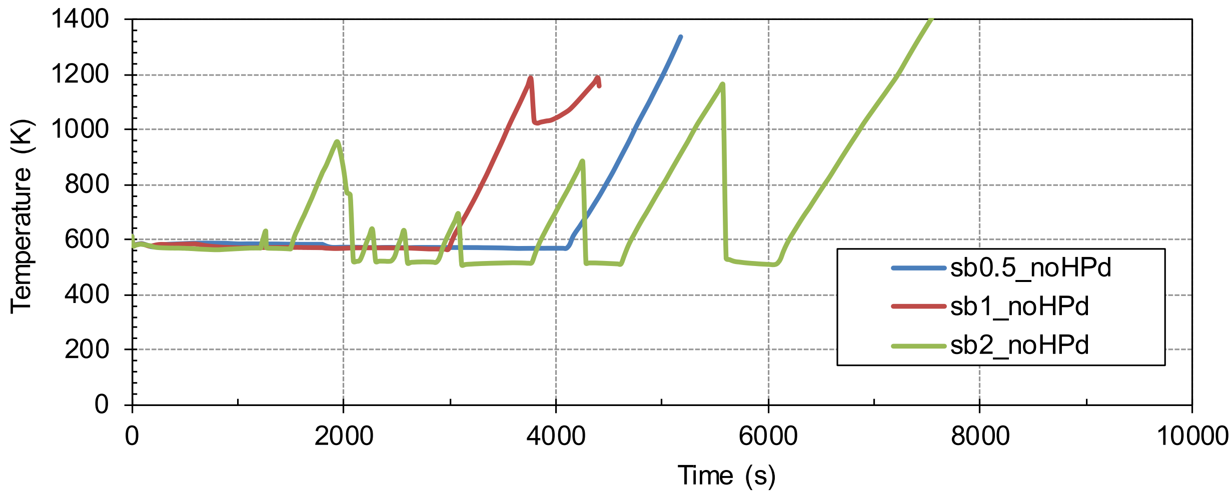

Figure 3-40 Cladding Temperature – noHP and AM1 (1.27 cm to 5.08 cm)

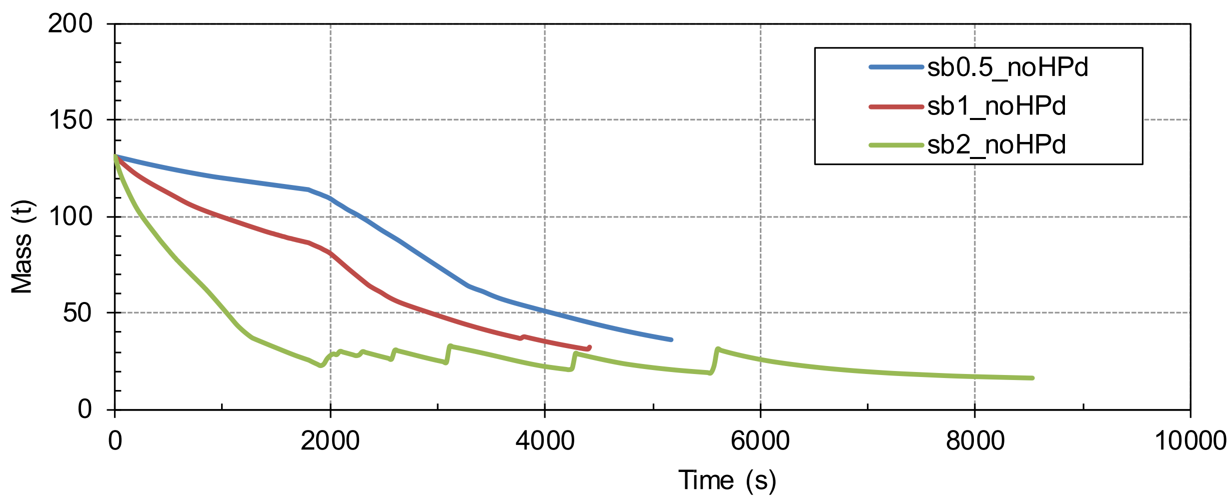

Figure 3-41 Primary System Mass – noHP and AM1 (1.27 cm to 5.08 cm)

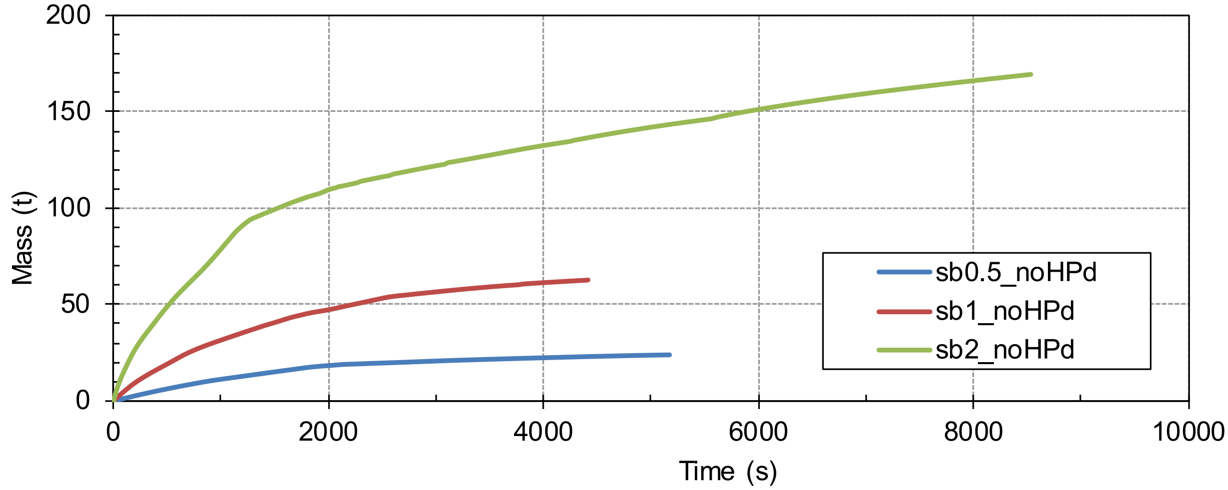

Figure 3-42 Total Integrated Break Flow – noHP and AM1 (1.27 cm to 5.08 cm)

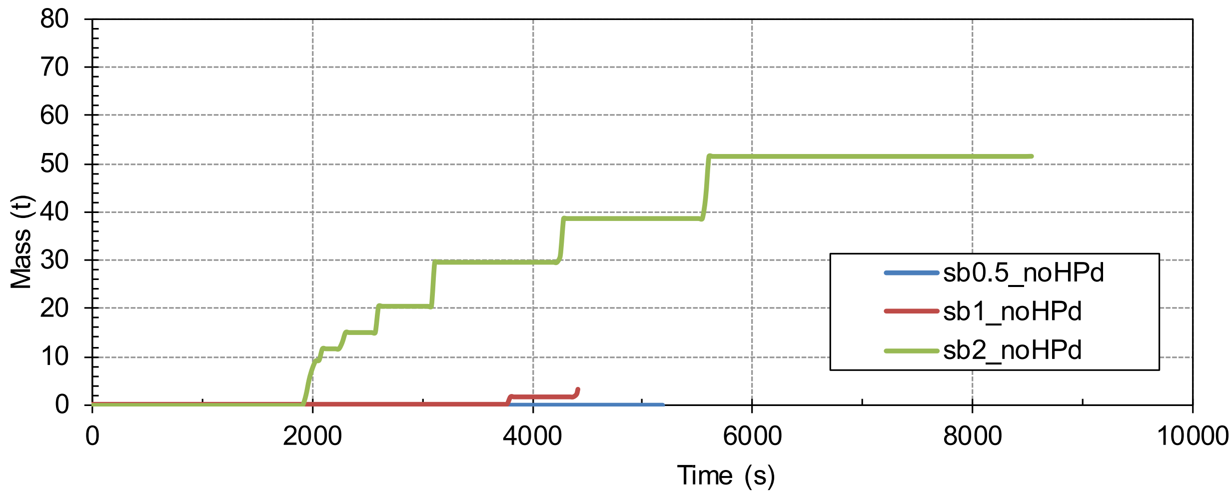

Figure 3-43 Total Integrated Accumulator Flow – noHP and AM1 (1.27 cm to 5.08 cm)

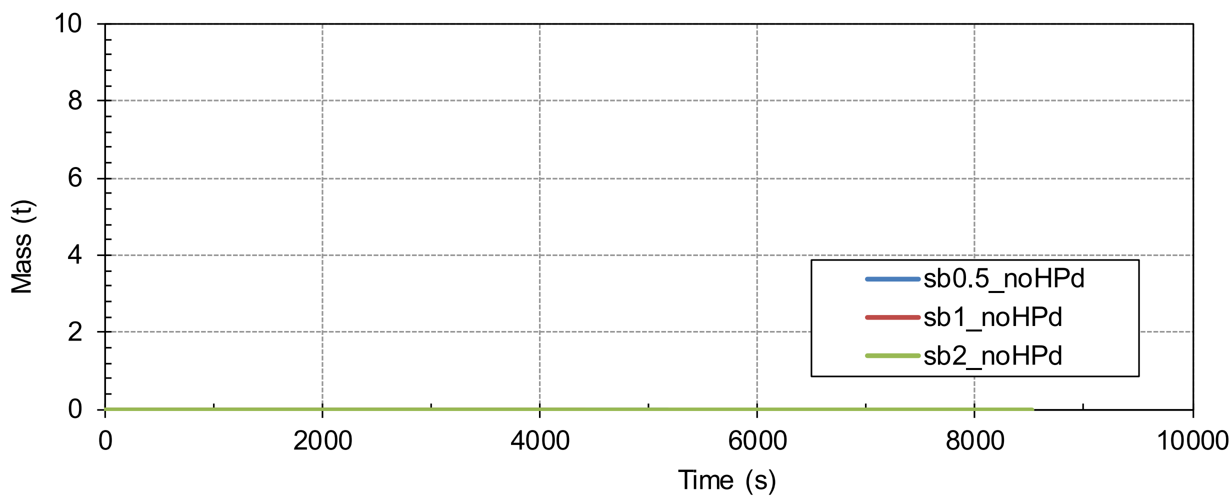

Figure 3-44 Total Integrated HPIS Flow – noHP and AM1 (1.27 cm to 5.08 cm)

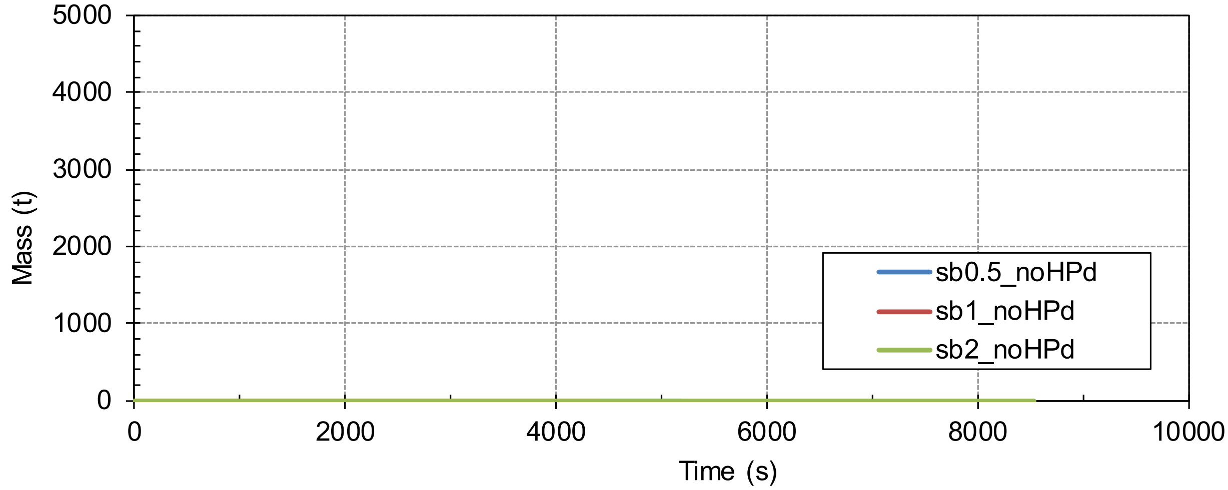

Figure 3-45 Total integrated LPIS flow – noHP and AM1 (1.27 cm to 5.08 cm)

3.4 Results of LOCA Break Size Spectrum 1.27 cm to 5.08 cm With One ECCS and Using AM2 Strategy (noHP and AM2)

In section 3.1 it is shown that in case of loss of HPI system the breaks smaller than 7.62 cm lead to final core uncovery, because the primary pressure is above the setpoints to inject with the accumulators and LPI system. Therefore AM2 accident management strategy reducing the primary pressure through the secondary side has been assumed in simulations. The simulation time was 24 h (86400 s). The steam generators were depressurized to 1.1 MPa. The results are shown in Figures 3-46 to 3-54. As can be seen the AM2 strategy has been successful for breaks 1.27 cm and 2.54 cm, while in the case of 5.08 cm break it was not successful. In the long term it can be seen that the primary system was not sufficiently depressurized to enable LPI system injection.

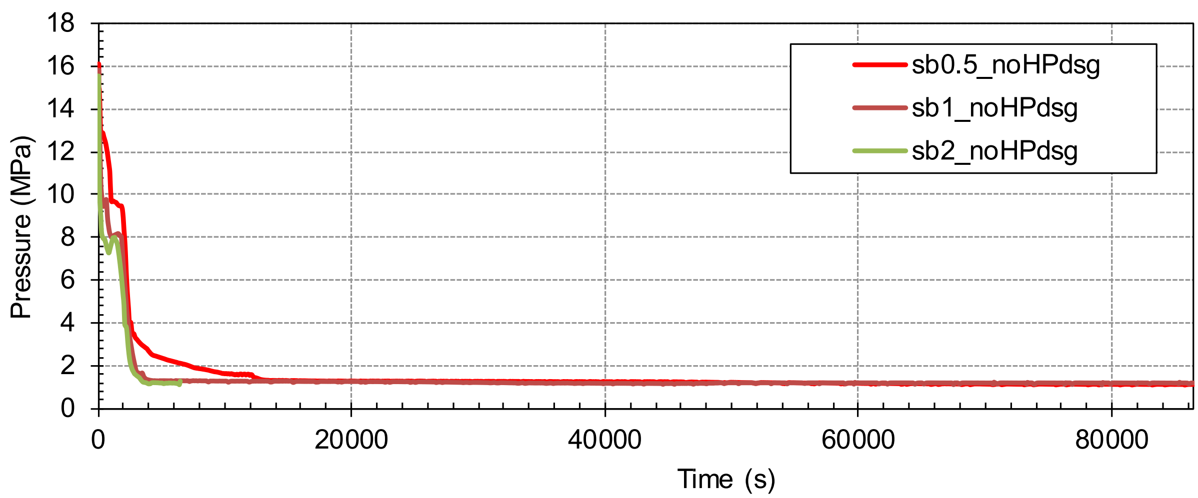

Figure 3-46 Pressurizer Pressure – noHP and AM2 (1.27 cm to 5.08 cm)

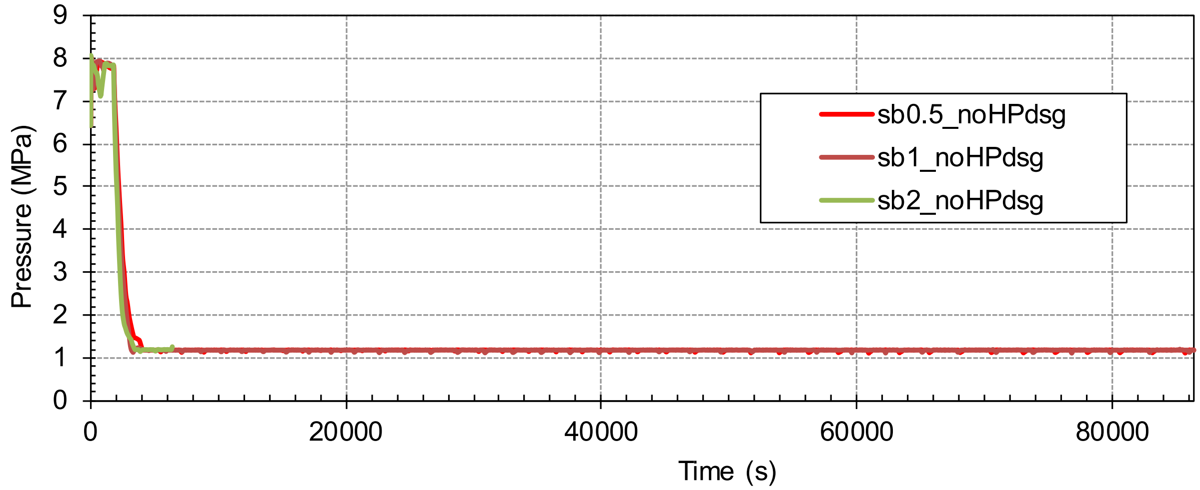

Figure 3-47 SG1 Pressure – noHP and AM2 (1.27 cm to 5.08 cm)

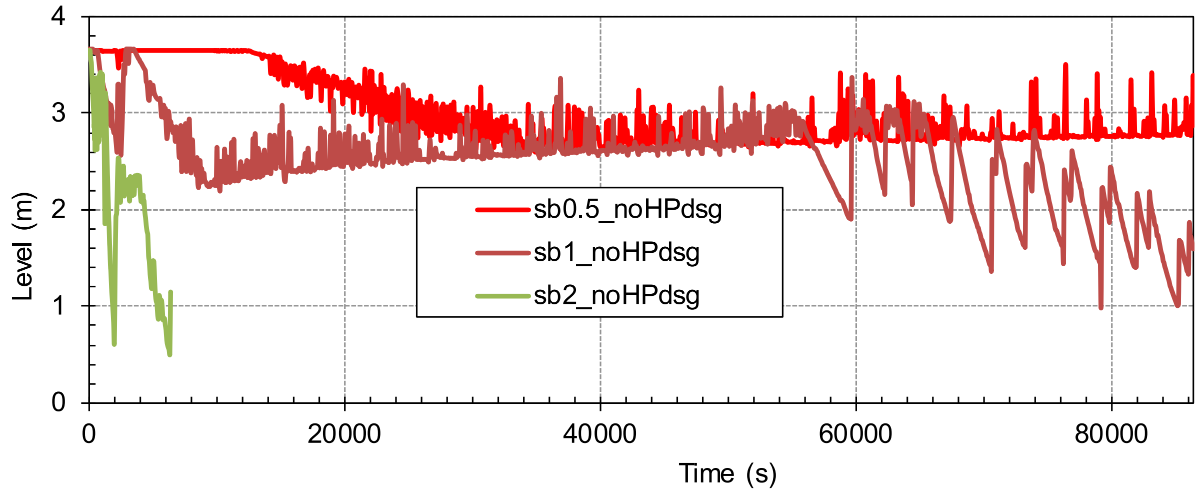

Figure 3-48 Core Collapsed Liquid Level – noHP and AM2 (1.27 cm to 5.08 cm)

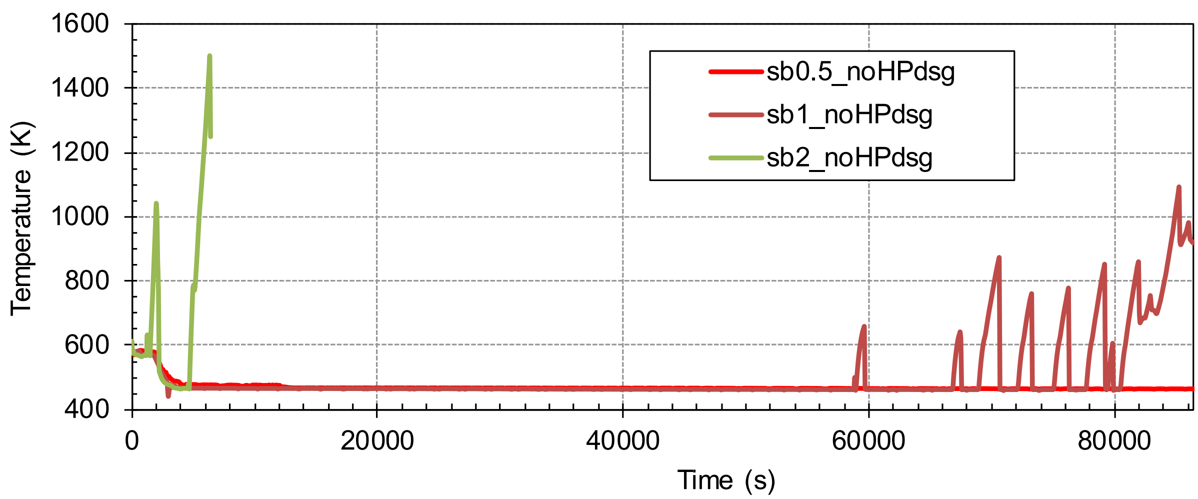

Figure 3-49 Cladding Temperature – noHP and AM2 (1.27 cm to 5.08 cm)

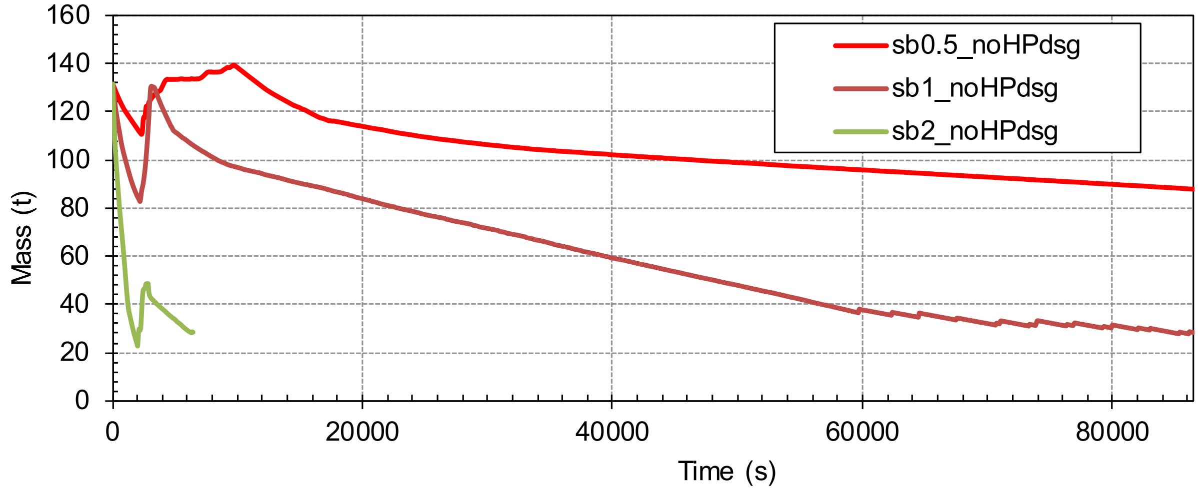

Figure 3-50 Primary System Mass – noHP and AM2 (1.27 cm to 5.08 cm)

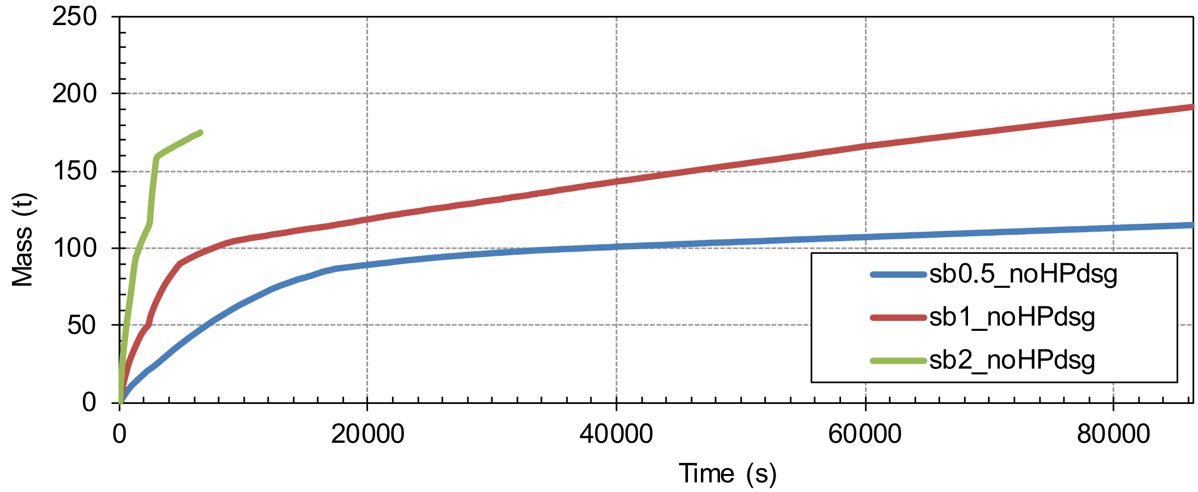

Figure 3-51 Total Integrated Break Flow – noHP and AM2 (1.27 cm to 5.08 cm)

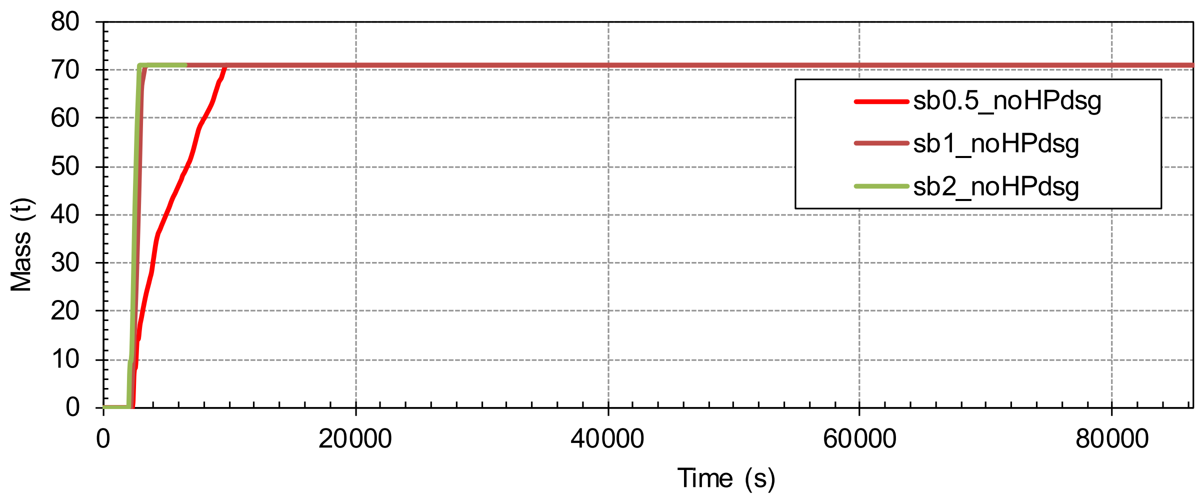

Figure 3-52 Total Integrated Accumulator Flow – noHP and AM2 (1.27 cm to 5.08 cm)

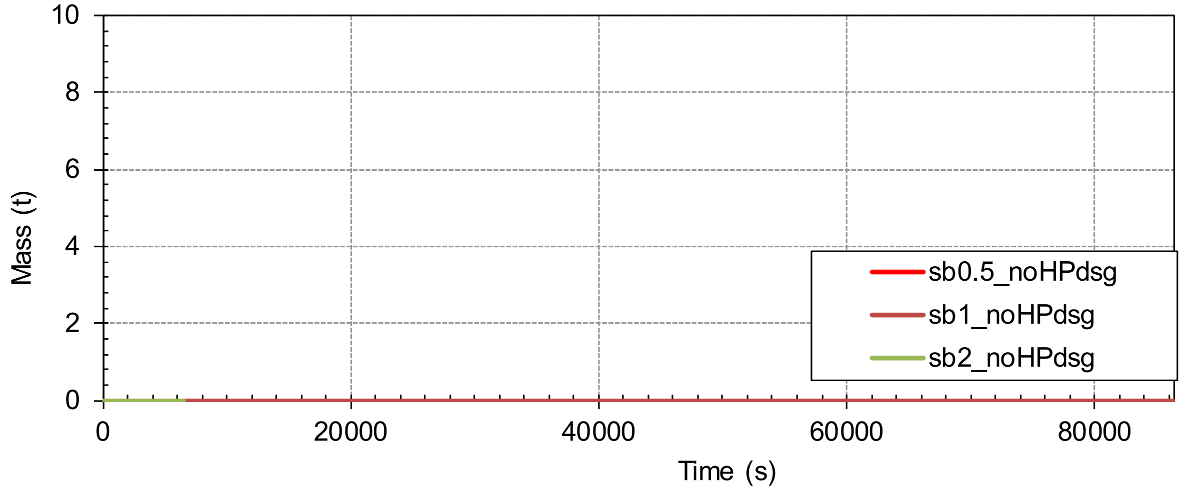

Figure 3-53 Total Integrated HPIS Flow – noHP and AM2 (1.27 cm to 5.08 cm)

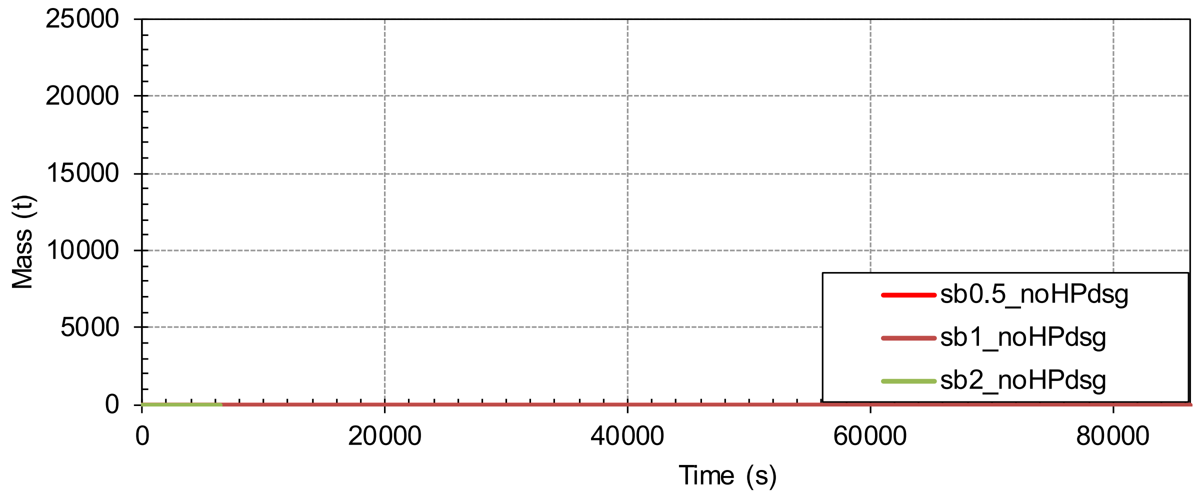

Figure 3-54 Total Integrated LPIS Flow – noHP and AM2 (1.27 cm to 5.08 cm)

3.5 Results of LOCA Break Size Spectrum 1.27 cm to 5.08 cm With One ECCS and Using AM3 Strategy (noHP and AM3)

In previous section 3.4 it was shown that the primary system was not sufficiently depressurized to enable LPI system injection. Therefore the steam generators were depressurized to 0.7 MPa. Again accident management strategy reducing the primary pressure through the secondary side has been assumed in simulations. The simulation time was 24 h (86400 s). The results are shown in Figures 3-55 to 3-63. As can be seen the strategy has been successful for breaks 1.27 cm and 2.54 cm, and also 5.08 cm break with initial heatup. Short term initial peak in cladding temperature occurred only in the case of 5.08 cm break size LOCA. The primary system was sufficiently depressurized to enable LPI system injection in long term.

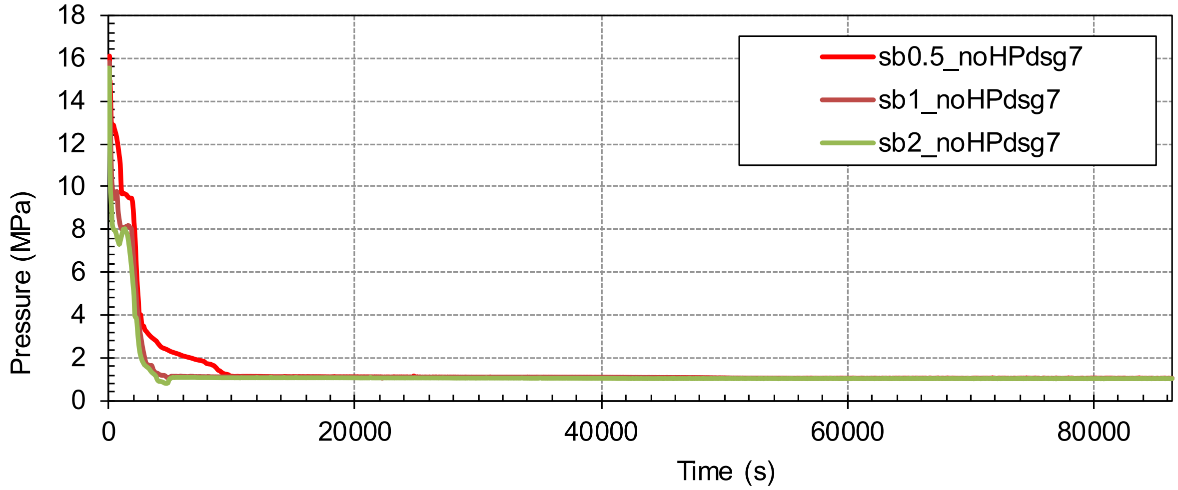

Figure 3-55 Pressurizer Pressure – noHP and AM3 (1.27 cm to 5.08 cm)

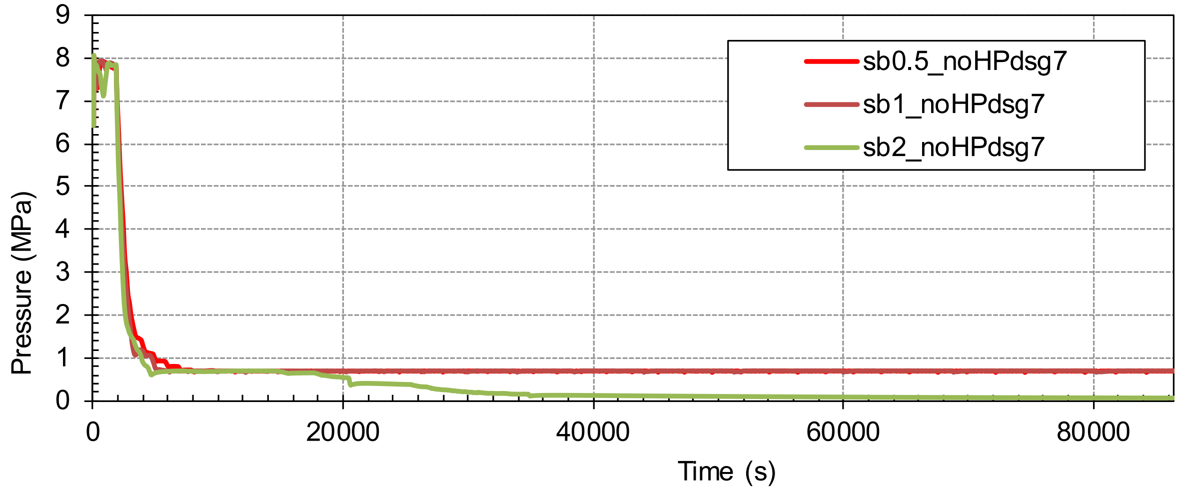

Figure 3-56 SG1 Pressure – noHP and AM3 (1.27 cm to 5.08 cm)

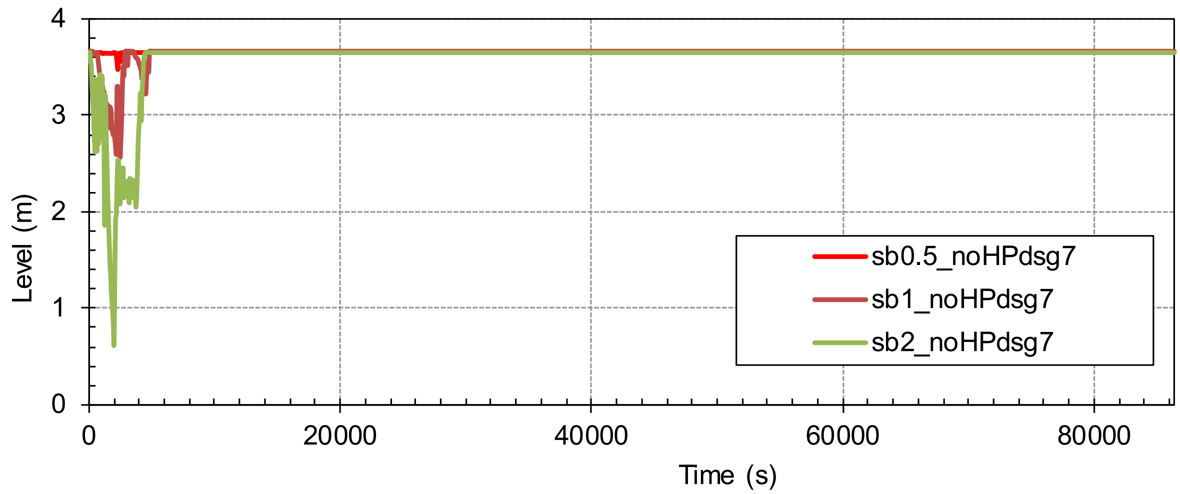

Figure 3-57 Core Collapsed Liquid Level – noHP and AM3 (1.27 cm to 5.08 cm)

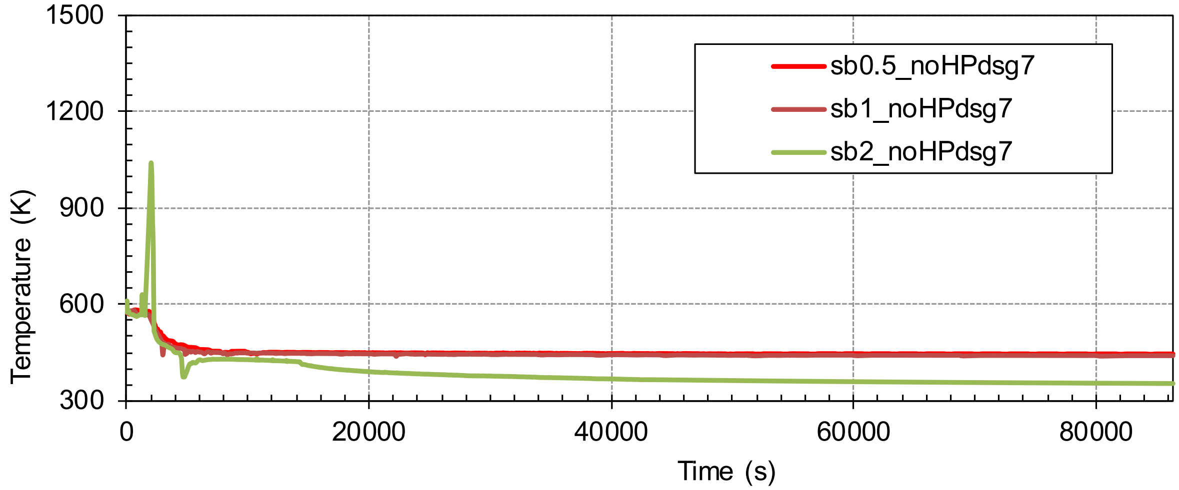

Figure 3-58 Cladding Temperature – noHP and AM3 (1.27 cm to 5.08 cm)

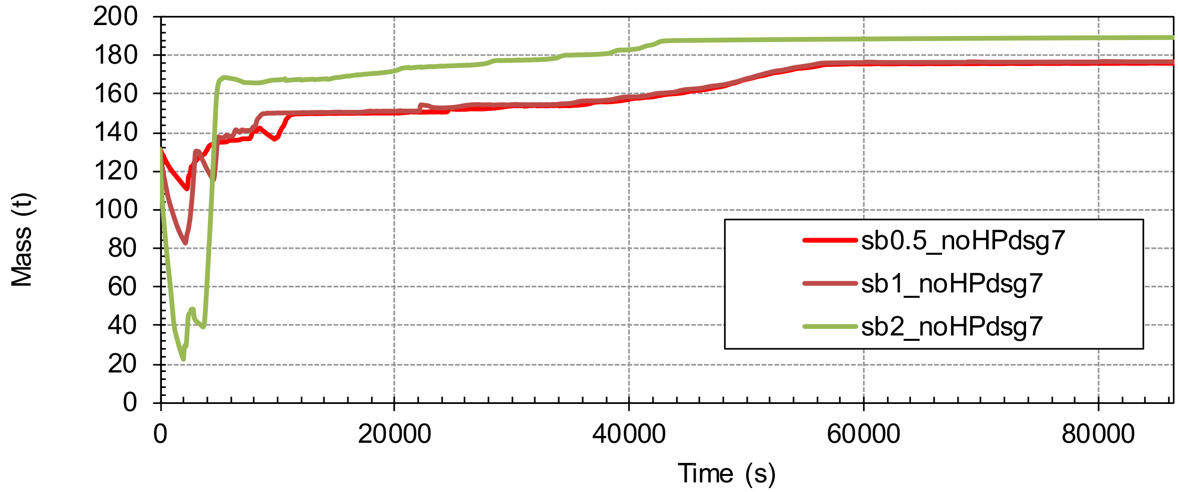

Figure 3-59 Primary System Mass – noHP and AM3 (1.27 cm to 5.08 cm)

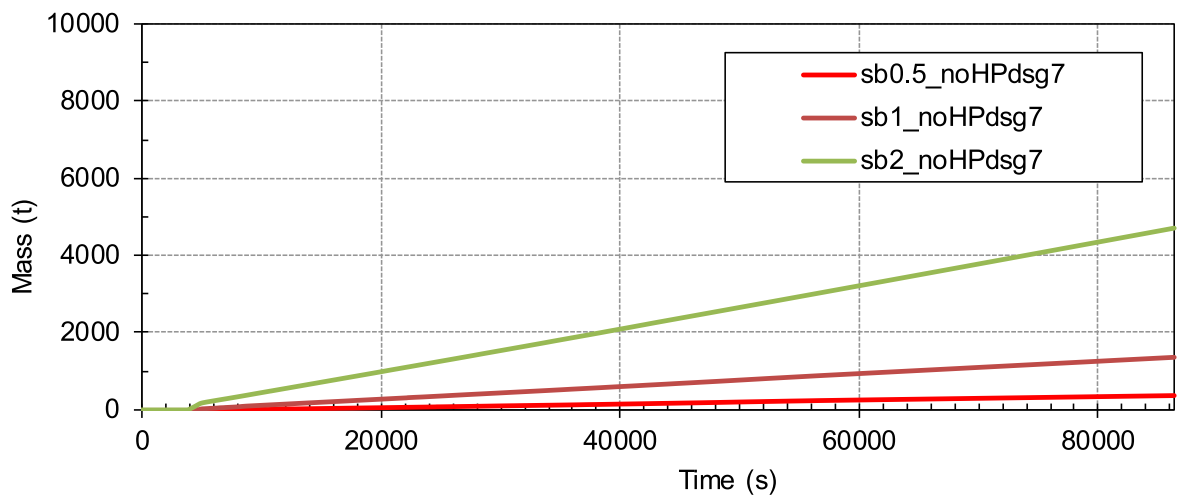

Figure 3-60 Total Integrated Break Flow – noHP and AM3 (1.27 cm to 5.08 cm)

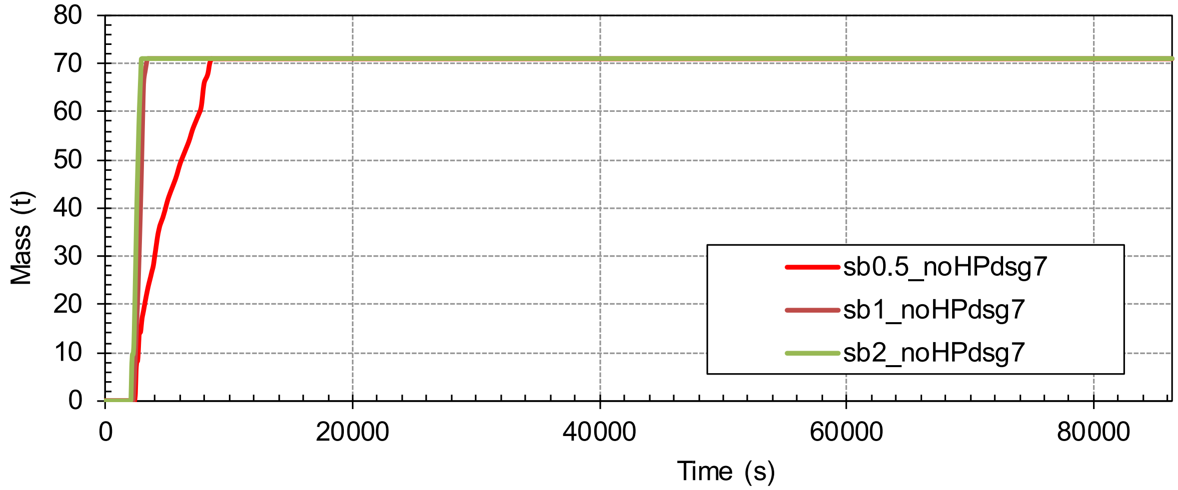

Figure 3-61 Total Integrated Accumulator Flow – noHP and AM3 (1.27 cm to 5.08 cm)

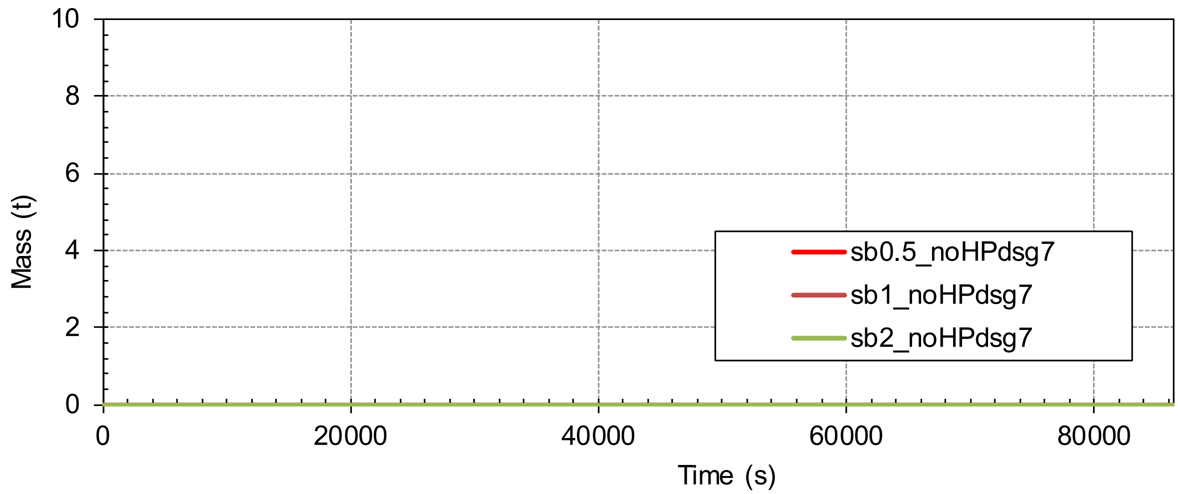

Figure 3-62 Total Integrated HPIS Flow – noHP and AM3 (1.27 cm to 5.08 cm)

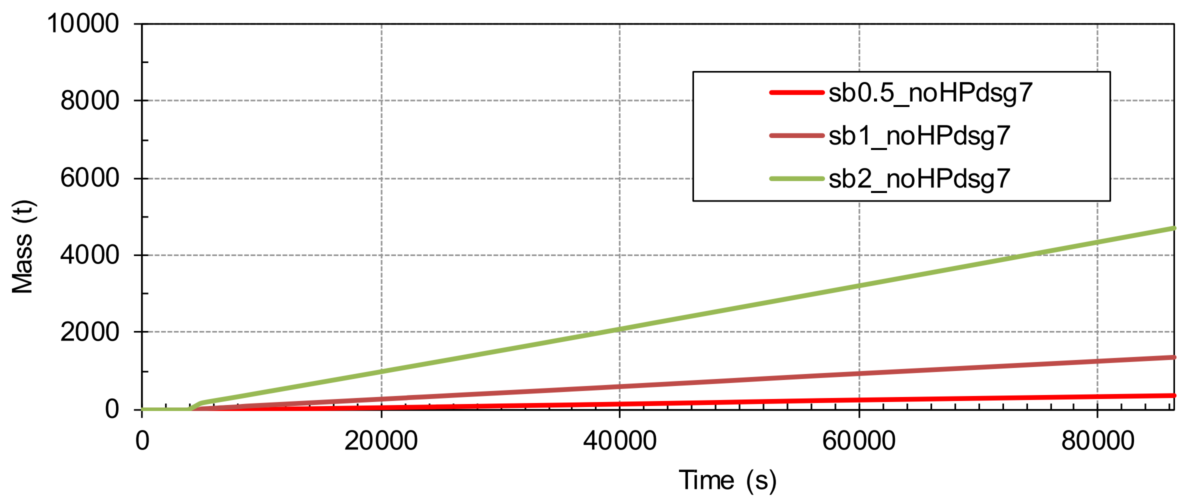

Figure 3-63 Total Integrated LPIS Flow – noHP and AM3 (1.27 cm to 5.08 cm)

4 CONCLUSIONS

The RELAP5/MOD3.3 Patch05 computer code calculations of loss of coolant accident (LOCA) without high pressure injection (HPI) system and LOCA without low pressure injection (LPI) system for a spectrum of break sizes from 1.27 cm (0.5 inch) to 30.48 cm (12 inch) have been performed. When HPI system is not available, depressurization of the primary side through the secondary side can be used in case of smaller breaks to mitigate consequences. For larger breaks, the pressure drop is faster and LPI systems maintained the core inventory. When LPI system is not available, the HPI system has sufficient capacity for safety injection in case of simulated LOCA break spectrum. It was shown that accident management strategy using steam generator PORVs to depressurize the primary system below LPIS injection setpoint are sufficient to mitigate smaller size LOCA without HPI systems operable. However, the control of design extension conditions (DEC) is expected to be achieved primarily by features implemented in the design (safety features for DECs) and not only by accident management measures that are using equipment designed for other purposes. This means that in principle a DEC is such if its consideration in the design leads to the need of additional equipment or to an upgraded classification of lower class equipment to mitigate the DEC. Therefore it can be concluded that safety injection pump with sufficient pressure capability should be introduced to prevent smaller break LOCAs in case HPI system is lost.

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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The second generation reactors were designed and built to withstand without loss to the structures, systems, and components necessary to ensure public health and safety during design basis accidents. There are also accident sequences that are possible but were judged to be too unlikely and therefore were not fully considered in the design process of second generation reactors. In that sense, they were considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand. They were called beyond design basis accidents. After Fukushima-Daiichi in the Europe the design extension conditions were introduced as preferred method for giving due consideration to the complex sequences and severe accidents without including them in the design basis conditions. In the study, the analysis was performed to see if the plant design can prevent spectrum of loss of coolant accidents together with the complete loss of one emergency core cooling function (e.g. high pressure injection or low pressure injection). The analyzed break spectrum ranged from 1.27 cm to 30.48 cm. For calculations the latest RELAP5/MOD3.3 Patch 5 has been used and the RELAP5 input model of Krško nuclear power plant.

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