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Simulation of Total Loss of Feedwater LOFT LP-FW-1 Test using RELAP5/MOD3.3

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ABSTRACT

After Fukushima-Daiichi in the Europe the design extension conditions (DEC) were introduced as preferred method for giving due consideration to the complex sequences and severe accidents without including them in the design basis conditions. The purpose of this study was to assess the latest RELAP5/MOD3.3 Patch 05 computer code for the simulation of such DEC. The LP-FW-01 test performed in 1983 on the Loss of Fluid Test Facility (LOFT) has been used for simulation. The LP-FW-01 test represents a fault sequence in which a total loss of feedwater to the steam generator is followed by recovery by primary system feed-and-bleed. The RELAP5/MOD3 steady state input deck available from literature has been adapted to RELAP5/MOD3.3 Patch 05, while transient input deck to simulate LP-FW-01 test has been newly developed. The simulation results for short term response (0-300 s) and long term response (0-7000 s) are presented in the report. The results suggest that in the short term simulation of LP-FW-1 test the simulated results matches the major events very good. In the long term the simulation results suggest that the entrainment to the surge line is important for the correct results.

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EXECUTIVE SUMMARY

After Fukushima-Daiichi both Western European Nuclear Regulators Association (WENRA) guidance document for issue F and International Atomic Energy Agency (IAEA) document on design requirements provides the total loss of feed water (LOFW) as an example of design extension condition (DEC), which should be taken into account for selection of DEC. The LP-FW-01 test performed in 1983 on the Loss of Fluid Test Facility (LOFT) has been used for simulation to assess the RELAP5/MOD3.3 Patch 05 computer code capability to simulate such DEC.

The RELAP5/MOD3 steady state input deck available from literature has been adapted to RELAP5/MOD3.3 Patch 05, while transient input deck to simulate LP-FW-01 test has been newly developed. The LP-FW-01 test represents a fault sequence in which a total loss of feedwater to the steam generator is followed by recovery by primary system feed-and-bleed. The coolant is simultaneously injected by the high pressure injection system (HPIS) and vented via the primary side power operated relief valve (PORV).

The LOFT facility was a 50 MWth two-loop pressurized water reactor (PWR). It was designed to study the thermo-hydraulic response of the system to a variety of simulated loss of coolant accident (LOCAs) scenarios. The LOFT facility conducted 38 experiments, before it was shutdown in 1985. In October 2006 its decontamination, decommissioning and demolition was completed. The LP-FW-1 test measured data have been obtained through Organisation for Economic Co-operation and Development (OECD)/Nuclear Energy Agency (NEA) data bank. The objectives of LP-FW-1 test were to provide data for code assessment, to allow assessment of the effectiveness of using PORVs and HPIS injection to remove reactor decay heat until residual heat removal (RHR) conditions are approached during a total LOFW and to provide information on transient characteristics to aid operators in the identification of, and recovery from, a LOFW transient. This can support DEC analyses for existing nuclear power plants on the need of DEC safety features for total LOFW. Namely, 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.

The purpose of the present study was to assess the latest RELAP5/MOD3.3 Patch 05 computer code using the off-take model option, which is available for the Henry-Fauske (HF) choke flow model besides simulations using Ransom-Trapp and Henry-Fauske choke flow model only. Finding from the literature also suggested to make sensitivity analysis on heat losses in the primary system.

The simulation results for steady-state, short term response (0-300 s) and long term response (0-7000 s) are shown. The results suggest that in the short term simulation of LP-FW-1 test the simulated results matches the major events very well. In the long term the simulation results suggest that besides code models also input modelling of steam flow may be important for correct predictions.

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ABBREVIATIONS AND ACRONYMS

ATWS	anticipated transient without scram
BETHSY	Boucle d'Etudes Thermohydrauliques de Systemes
BDBA	beyond design basis accident
CLIV	cold leg isolation valve
DEC	design extension conditions
ECC	emergency core coolant
ECCS	emergency core cooling system
EOP	emergency operating procedure
IAEA	International Atomic Energy Agency
HF	Henry-Fauske
HLIV	hot leg isolation valve
HPIS	high pressure injection sytem
LOCA	loss of coolant accident
LOFW	loss of feedwater
LOFT	Loss of Fluid Test Facility
LPIS	low pressure injection system
MSCV	main steam control valve
NEA	Nuclear Energy Agency
OECD	Economic Co-operation and Development
PCS	primary coolant system
PORV	power operated relief valve
PSA	probabilistic safety assessment
PWR	pressurized water reactor
QOBV	quick-opening blowdown valve
RABS	reflood assist bypass system
RHR	residual heat removal
RT	Ransom-Trapp
SI	safety injection
SG	steam generator
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. After Fukushima-Daiichi both International Atomic Energy Agency (IAEA) document on design requirements [1] and Western European Nuclear Regulators Association (WENRA) reference levels [2] require that design extension condition (DEC) should be taken into account.

Slovenia implemented WENRA reference levels issue F requirements into its Rules on radiation and nuclear safety factors [3]. 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 one DEC A scenario was simulated, total loss of feedwater. This is in line with IAEA and WENRA documents regarding DEC, as can be seen in the following.

IAEA TECDOC-1791 document [4] provides the following scenario as an example of DEC derived from probabilistic safety assessment (PSA):

- total loss of feed water.

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

- total loss of feed water.

The LP-FW-01 test performed in 1983 on the Loss of Fluid Test Facility (LOFT) has been used for simulation to assess the RELAP5/MOD3.3 Patch 05 computer code capability to simulate such DEC. For example, the analyses performed in the past by RELAP5/MOD2 demonstrated that improved modelling of entrainment in the hot-leg/surge line connection is needed [6]. A modified version of RELAP5/MOD2 containing an improvement to the horizontal stratification entrainment model was found to give an improved prediction of the long term pressure history [6]. Recently, the results on the applicability and accuracy of the SPACE computer code against LOFT LP-FW-1 experiment have been published [7]. The SPACE base calculation showed that the primary pressure and temperature during a long-term transient were over-estimated, similarly as in the study by RELAP5/MOD2 [6]. In the SPACE simulation, the off-take model available in the SPACE computer code using Ransom-Trapp choke flow model has been used. The sensitivity analysis has been also performed by SPACE computer code for the major parameters including the discharge coefficients in the primary side power operated relief valve (PORV) and heat losses in the system. The data assimilation showed a better scale prediction with the off-take model used, increase of two-phase discharge coefficient and higher heat loss in the primary system. Namely, in in the steady state the total heat losses predicted by SPACE computer code were 185 kW, while in Croxfod RELAP5/MOD2 study [6] it was predicted to be 272 kW. Croxfod [6] also reported that the experimental total heat losses were 250 ± 100 kW. The purpose of the present study was to assess the latest RELAP5/MOD3.3 Patch 05 computer code using the findings described in [7]. In the latest RELAP5 the off-take model option is available for the Henry-Fauske (HF) choke flow model. It should be noted that RELAP5/MOD2 code has only Ransom-Trapp (RT) choke flow model without the off-take option model and that in 1998 the Henry-Fauske choke flow model has been added as a user option into the RELAP5/MOD3.2.2Beta code version [8].

Similar DEC experiments with total loss of feedwater have 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, LOFT, PKL and PMK test facilities for accident management in pressurized water reactors (PWRs) for total loss of feedwater accidents, in which operator actions were studied for BDBA with non-degraded core (in terms of WENRA such accidents are called DEC A). Experiments were selected from cross-reference matrix for accident management for non-degraded core, which has been created in the frame of OECD/NEA [9] and from published NUREG/IA reports.

Table 1-1 Accident Management in PWRs for BDBA with Non-Degraded Core (Total Loss of Feedwater)

Test No.	Test type	Brief description
BETHSY 5.2c2	Total loss of feedwater	During BETHSY (Boucle d'Etudes Thermohydrauliques de Systemes) test 5.2c2 [9], the emergency operating procedure (EOP) was conducted in accordance with the rules presently implemented in plant control rooms, which allow operators more time for the recovery of feedwater systems: it consisted in manually starting the high pressure injection system (HPIS) as soon as 2 SG liquid levels reached 3 m; as a consequence, primary pressure slowly increases up to 16.3 MPa, and is then maintained at this value through pressurizer power operated relief valves (PORVs) automatic operation. 30 minutes after EOP initiation, or earlier if the core outlet fluid temperature reaches 603 K, the pressurizer PORVs are actuated at full discharge capacity.
LOFT L9-1 / L3-3	Total loss of feedwater (LOFW) accident followed by small break LOCA	Experiment L9-1 was the first anticipated transient with multiple failures performed at Loss-of-Fluid-Test (LOFT), and consisted of a simulated LOFW accident with delayed reactor scram and no auxiliary feedwater injection [13]. Experiment L3-3 simulated two independent recovery procedures from the LOFW accident L9-1, without engaging the emergency core coolant (ECC).
LOFT L9-3	Loss of feedwater without reactor trip	Experiment L9-3 conducted in the LOFT facility was a unique one simulating an ATWS event in pressurized water reactor. The experiment simulated a loss of feedwater induced ATWS in a commercial plant. The experiment consisted of two parts: the ATWS itself, which lasted about 600 s, and the plant recovery [10].
PKL III B1.2	Total loss of feedwater with secondary side feed and bleed	Total loss of feedwater (loss of main and auxiliary feedwater) with no core cooling systems (high and low pressure injection pumps and accumulators) was studied. Secondary side bleed and feed was performed. Injection of water was due to flashing in feedwater line and subsequent injection by a mobile pump [9].
PMK-2	Total loss of feed water with primary side feed and bleed	The total loss of feedwater test simulated a beyond design-basis accident scenario with unavailability of the hydro-accumulators. For prevention of core damage, accident management strategies were applied, including a primary side bleed-and-feed procedure with intentional depressurization of the secondary side [11].

The report is organized as follows. In Section 2 the methods used are described. First the LOFT facility and LP-FW-01 test sequence are described. Then the RELAP5 thermal-hydraulic system computer code input model description is briefly described. Then the simulated scenarios are described. Three are base case simulations using Ransom-Trapp (RT) and Henry-Fauske (HF) choke flow model with/without off-take model, respectively and the other 9 cases are sensitivity calculations varying the primary system heat losses in the base calculations. At the end of Section 2 the initial and boundary conditions, resulting from steady state calculations, are described. In Section 3 the results of the total LOFW simulations are presented, including discussion of the result. Finally, main conclusions are drawn.

2 METHODS USED

2.1 LOFT Facility Description

The LOFT facility was a 50 MW_{th} two-loop pressurized water reactor (PWR). It was designed to study the thermo-hydraulic response of the system to a variety of simulated loss of coolant accident (LOCAs) scenarios. The LOFT facility conducted 38 experiments, before it was shutdown in 1985. In October 2006, its decontamination, decommissioning and demolition was completed. The LP-FW-1 test measured data have been obtained through Organisation for Economic Co-operation and Development (OECD)/Nuclear Energy Agency (NEA) data bank. A detailed description of the LOFT system is given in reference [12]. The facility was volumetrically scaled by a ratio of 1/60 in comparison to a full-scale commercial PWR with a power of 3000 MW_{th}. Inherent in the scaling are some compromises in the geometric similarity (for example, the 1.7 m long LOFT reactor core was around half the length of that of a commercial PWR). The main systems of the LOFT facility comprise of reactor system, primary coolant system, blowdown suppression system, secondary coolant system and the emergency core coolant (ECC) system. The primary coolant system of the LOFT facility was comprised of an intact loop, which consisted of active hydraulic components (pressurizer, primary and secondary side of steam generator, coolant pumps and ECC system) and provided the main coolant flow to the reactor, and a broken loop, which simulated a fractured loop in the system.

2.2 LP-FW-1 Test Description

The LP-FW-01 test represents a fault sequence in which a total loss of feedwater to the steam generator is followed by recovery by primary system feed-and-bleed. The coolant is simultaneously injected by the high pressure injection system (HPIS) and vented via the primary side power operated relief valve (PORV). The objectives of LP-FW-1 test were also to provide data for code assessment. This can support DEC analyses for existing nuclear power plants on the need of DEC safety features for total LOWF. Namely, 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.

The main sequence of events is shown in Table 1-1. The initiating event is main feedwater trip resulting in the primary pressure increase. This activated pressurizer spray, but the reactor tripped automatically on high pressurizer pressure due to pressure increase, the main steam control valve (MSCV) starts to shut automatically and the PORV was latched open by operator. At 8.72 MPa, the primary coolant pumps were tripped by operator and HPIS injection was initiated. The primary coolant pumps coastdown was terminated at 235 s. Due to open PORV the pressurizer liquid level reached top of indicating range, the discharged flow changed from liquid to two-phase flow. At 2370 s, the HPIS flow exceeds PORV discharge flow. The experiment was terminated when the primary pressure reached 4.69 MPa at 6820 s into the transient.

Table 2-2 Main Sequence of Events for LOFT LP-FW-1 Test

Event	Time (s)
Main feed tripped	0.0
Pressurizer spray initiated	33.2 ± 0.3
Reactor tripped on high pressure	48.8 ± 0.01
Main steam control valve (MSCV) starts to shut	48.8 ± 0.2
Power operated relief valve (PORV) latched open (by operator)	50.8 ± 0.2
Steam generator liquid level reached bottom of indicating range	85 ± 15
MSCV fully shut	61.0 ± 0.2
Primary coolant pump coastdown initiated	219.2 ± 0.1
High pressure injection system (HPIS) initiated	221.6 ± 0.2
Primary coolant pump coastdown complete	235.4 ± 2.0
First void indication of formation in primary system	245 ± 10
Pressurizer liquid level reached top of indicating range	333.2 ± 0.4
PORV transitioned from steam flow to two phase flow	339.0 ± 2.0
HPIS flow greater than PORV discharge flow	2370 ± 100
Experiment terminated	6820 ± 110

2.3 RELAP5 Input Model Description

The RELAP5/MOD3 steady state ASCII input deck available from the literature [13] has been adapted to the latest RELAP5/MOD3.3 Patch 05 in the frame of studying the L9-1/L3-3 experiment with multiple failures on LOFT facility [13]. I should be noted that also input deck in [13] was adapted from original input deck for RELAP5/MOD1 code received from Idaho National Engineering Laboratory (INEL) RELAP5/MOD3 code developer at January 1991. The transient input deck to simulate LP-FW-01 has been newly developed. The heat transfer area of steam generator tubing was returned from 110% used in the study [13] back to original value (used in the original INEL input deck) for RT choke flow model, while for HF choke flow model it was set to 101% to better match initial conditions. The total heat losses to the containment in the original ASCII input deck [7] were 160 kW and this value was used for base case calculations. The RELAP5 nodalization for LOFT facility in shown in Figure 2-1, where HLIV is hot leg isolation valve, CLIV is cold leg isolation valve, QOBV is quick-opening blowdown valve and RABS reflood assist bypass system.

Based on the component number, the 100, 200, 300, 400, 500, 600, 800 and 900 orders indicate the intact loop, reactor vessel including the core, broken loop, pressurizer, steam generator secondary side, emergency core cooling system (ECCS), containment and intact hot leg loop boundary, respectively. The intact loop contains pumps, pressurizer, steam generator, ECCS, and connecting pipes. The pressurizer contains a cylindrical-type pressure vessel, immersion-type heater, spray line, surge line, PORV line, and PORV. The ECCS is modeled with the accumulator, the high pressure injection system (HPIS) and the low pressure injection system (LPIS). The steam generator is a vertical shell and U-tube recirculation-type heat exchanger. On the secondary side separator, steam dome and main steam control valve (MSCV) were modelled.

The heat losses at the outer wall of components were modelled by heat structures. The heat loss in the LOFT LP-FW-1 calculation is considered for the primary system (including the reactor vessel with the pipes of the hot-leg and cold-leg), the pressurizer, the broken hot leg loop and the steam generator. Ambient temperature of 311 K was assumed and different heat transfer coefficients.

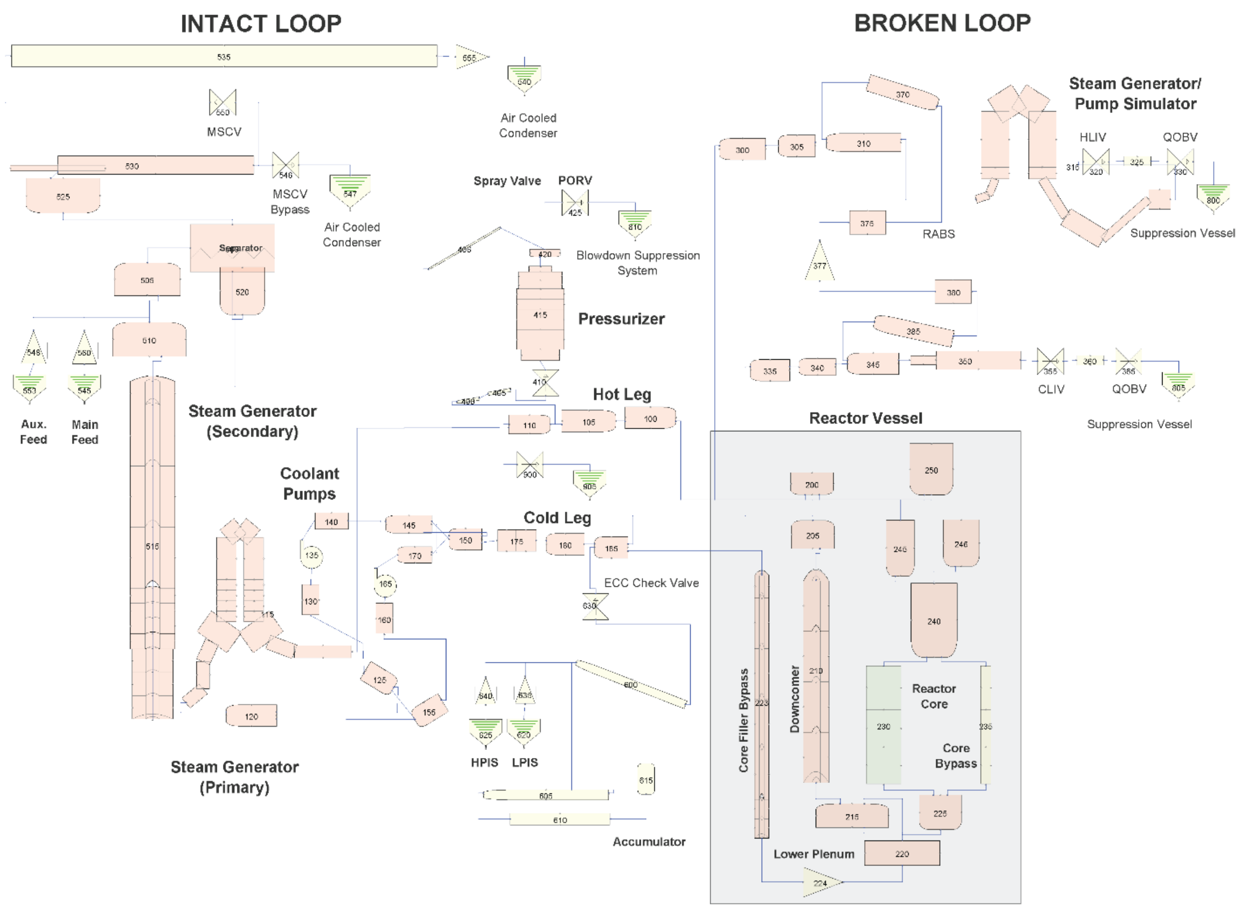

Figure 2-1 RELAP5 Nodalization of the LOFT Facility for LP-FW1 Test

2.4 <u>Simulated Scenarios</u>

Table 2-3 shows scenarios, for which simulations have been performed. Three are base case simulations and 9 are sensitivity calculations varying heat losses in the primary system. For primary system the base heat transfer coefficient of 18.83 W/m^2/K (100%) was varied to 150%, 200% and 250% of the value. Namely, in the calculation by RELA5/MOD2 [6] a value of 272 kW was used for the total system heat loss to the containment based on the value of 250 ± 100 kW, given in the experimental data report. In the received ASCII input deck this value was 160 kW. By changing the primary system heat transfer coefficient to 150% (28.25 W/m^2/K), 200% (i.e. 37.66 W/m^2/K) and 250% (47.0866 W/m^2/K), the total system heat loss to the containment change to 222 kW, 285 kW and 347 kW, respectively. All values of assumed total system heat loss to the containment varying from 160 kW to 347 kW, being inside the uncertainty range.

Table 2-3 Simulated Scenarios for LOFT LP-FW-1 Test

Simulated scenario cases	Label
Base cases	
RELAP5/MOD3.3 simulation using Ransom-Trapp (RT) choke flow model	RT
RELAP5/MOD3.3 simulation using Henry-Fauske (HF) choke flow model	HF
RELAP5/MOD3.3 simulation using HF choke flow model and off-take model option	HF-off
Sensitivity cases	
RT case with increased primary system heat transfer coefficient to 200%	RT-h
RT case with increased primary system heat transfer coefficient to 150%	RT-h1
RT case with increased primary system heat transfer coefficient to 250%	RT-h2
HF case with increased primary system heat transfer coefficient to 200%	HF-h
HF case with increased primary system heat transfer coefficient to 150%	HF-h1
HF case with increased primary system heat transfer coefficient to 250%	HF-h2
HF-off case with increased primary system heat transfer coefficient to 200%	FH-off-h
HF-off case with increased primary system heat transfer coefficient to 150%	FH-off-h1
HF-off case with increased primary system heat transfer coefficient to 250%	FH-off-h1

2.5 Initial and Boundary Conditions

First, the steady-state calculation was performed to set the LP-FW-1 test conditions. The major calculated conditions at end of steady-state (initial conditions) are shown in Table 2-4, where are compared to measured data. All values are inside the uncertainty band except the secondary pressure was slightly overestimated in case of using RT choke model (we decided to keep original heat transfer area for SG tubing).

Table 2-4 Initial and Boundary Conditions

Core temperature (K)	Measured	Calculated - RT	Calculated - HF
Primary coolant system			
Hot leg pressure (MPa)	14.80 ± 0.06	14.83	14.83
Hot leg temperature (K)	581.3 ± 1.3	580.3	580.4
Cold leg temperature (K)	554.3 ± 1.0	554.0	553.3
Mass flow rate (kg/s)	346.13 ± 2.59	346.13	346.16
Power Level (MW)	49.2 ± 0.5	49.2	49.2
Steam Generator, secondary side			
Pressure (MPa)	5.30 ± 0.06	5.37	5.31
Feed flow rate (kg/s)	26.36 ± 0.79	25.91	26.27
Water temperature (K)	537.7 ± 2.6	535.6	535.2
Liquid level (m)	2.78 ± 0.04	2.78	2.78

3 RESULTS

The primary coolant system (PCS) pressure for all calculated cases is shown in Figure 3-1. It is one of the most important parameters, because HPIS sequence of events, including injection depends on it. It can be seen that both RELAP5 base calculations using Ransom-Trap (see 'RT' calculation) and Henry-Fauske (see 'HF' calculation) choke flow model overestimated the primary coolant pressure and that in this respect they are similar like the study by RELAP5/MOD2 [6] or recent SPACE base calculation [7]. The pressure increase during the initial period of two-phase discharge from the PORV was overestimated, leading to an overprediction of primary system pressure for the remainder of the transient. As already indicated, one of the reasons is that entrainment in the hot-leg/surge line is not properly modelled. To take into account this effect, the off-take model option was used (see 'HF-off' calculation), which is available only when HF choke flow model is used. The PCS pressure drops, but it is still overestimated after 1000 s. Following the study in [7] the sensitivity to primary coolant system heat losses was also studied. It can be seen that calculations with increased heat loses in the primary coolant system by doubling the heat transfer coefficient (cases 'RT-h' and 'HF-h') resulted in the pressure decrease, which is still above the experimental data in the first half of the calculation (later dropped below the experimental value), while calculation with increased heat losses and off-take model option selected (see 'HF-off-h' calculation) resulted in PCS pressure close to experimental values. These results confirmed findings obtained for SPACE computer code [7], where heat losses were also increased. Here it should be noted that the value of 160 kW used in the original ASCII input deck is close to the lower value of uncertain total heat losses to the containment, having uncertain range from 150 kW to 350 kW. Doubling the primary system heat transfer coefficient gives the value of 285 kW for total heat losses, being closer to the mean value of 250 kW than the value of 160 kW. In the case of 'RT-h', 'HF-h' and 'HF-h-off' calculations the total heat losses were closer to the mean value reported for the experiment than in our base case simulations. Nevertheless, in spite of heat losses closer to reported experimental value only the calculation using off-take model was able to match good with experimental curve. Here it should be also noted that the code developers [9], when comparing Ransom-Trapp vs. Henry-Fauske found out that Ransom-Trapp does not do as good of a job of calculating the critical flow rate for the two-phase blowdown case, therefore Henry-Fauske (default) choke flow model for two-phase blowdown has been recently recommended [15]. At the time of RELAP5/MOD2 study [6] off-take model was not available. In the following Figures 3-2 through 3-72 other parameters were compared to experimental data for short term (period from 0 to 300 s) and long term (period from 0 to 7000 s) response. It should be noted that the focus of the study was on the correct primary coolant pressure prediction. Nevertheless, the agreement for other plant variables was satisfactory for the short and long term simulation. For each of the three base calculations the sensitivity study on heat losses variation is shown in Sections 3.3 to 3.5. Finally, in Section 3.6 the results of base cases with increased primary system heat transfer coefficient to 200% are shown, which showed to be best adjusted from the point of primary coolant pressure agreement.

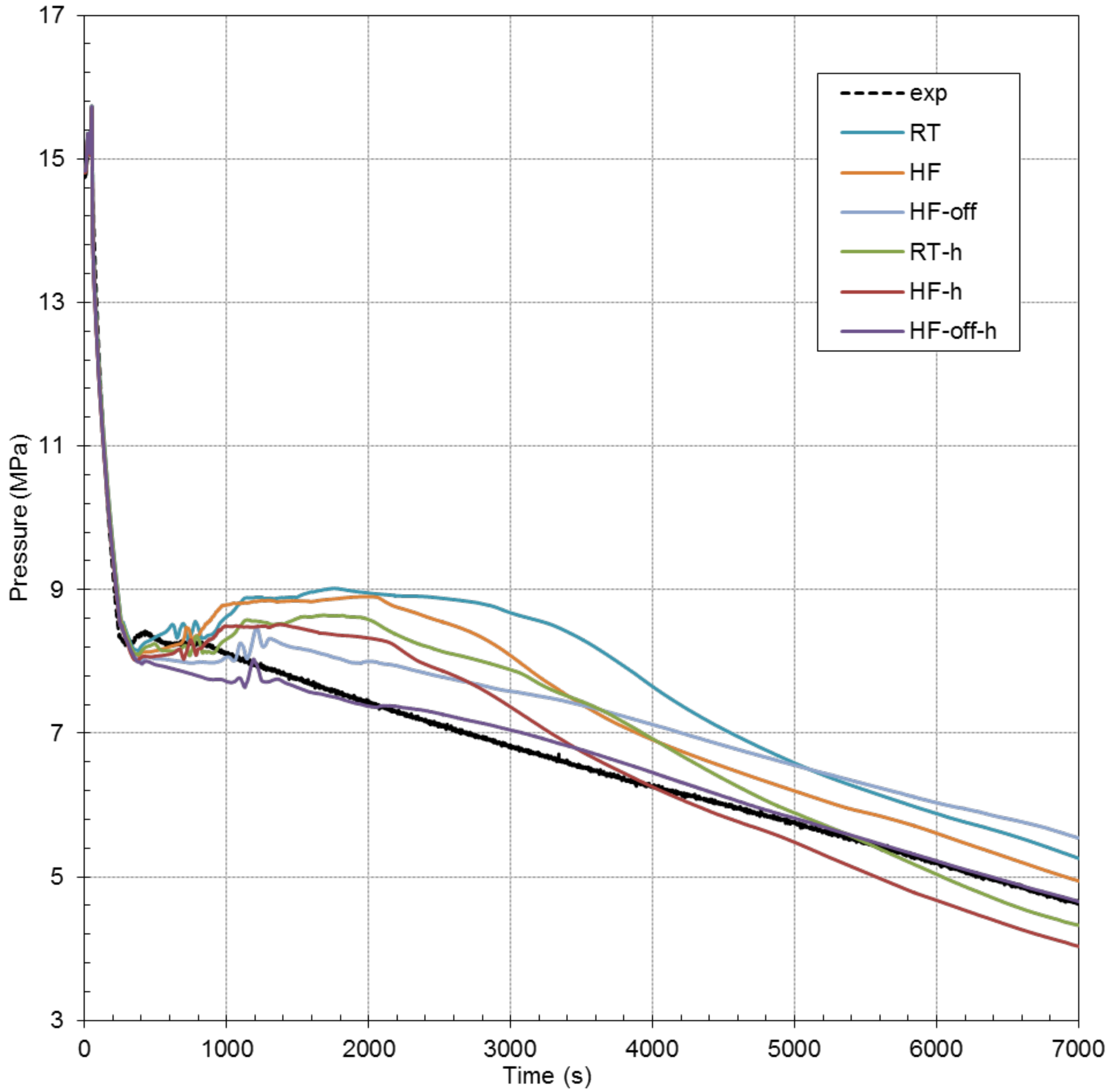

Figure 3-1 Primary Coolant System Pressure Comparison

3.1 Results Comparison of Base Cases ('RT', 'HF', 'HF-off') – Short Term Response (0 – 300 s)

Figures 3-2 through 3-17 shows short term response (0-300 s), in which all RELAP5 performed calculations are comparable and reasonably agree with the experimental data. The initiating event is loss of main feedwater (see Figure 3-17), resulting in the heatup causing pressure increase. In the experiment the primary coolant inventory increased (see Figure 3-9), while in calculation the value is constant. Here it should be noted that measured uncertainty in the primary coolant is large and calculated value is inside the measurement uncertainty band (± 500 kg). The rate of increase of primary pressure (see Figure 3-2) in the period up to reactor trip is well predicted by RELAP5, while secondary pressure increase is faster (see Figure 3-3) and trend is similar as liquid temperature in secondary coolant system downcomer (see reactor power drop in Figure 3-16). After reactor trip (see Figure 3-8) the MSCV started to close, and the pressurizer PORV was latched open (see PORV discharge flow in Figure 3-11). RELAP5 gave a good prediction of the rate of decrease of primary pressure during the period in which there was single phase steam discharge through the PORV. The depressurization halted when saturation conditions (boiling) occurred in the hot leg around 250 s. The calculated hot leg temperature increases (see Figure 3-6) and the cold leg temperature decreases (see Figure 3-23). The pressurizer and steam generator levels are shown in Figure 3-4 and Figure 3-5, respectively. The pressurizer level initially increases due to heatup, while secondary coolant system level decreases due to loss of main feedwater flow (see Figure 3-17). The primary coolant pumps started coastdown at 219 s (see Figure 3-10). The timing of coastdown start is pretty well predicted, but the calculated coastdown is faster. Finally, the primary pressure behavior very much depend on the break flow through PORV. The PORV discharge flow is shown in Figure 3-11 and the mass discharged through PORV in Figure 3-12. The PORV flow density measured before reactor trip is high compared to calculations (see Figure 3-14), however the RELAP5 calculations agree with the one in [6]. Finally, at 221 s, the HPIS pump started its injection (see Figure 3-13). The initial spike in the measured flow has not been predicted in the calculations.

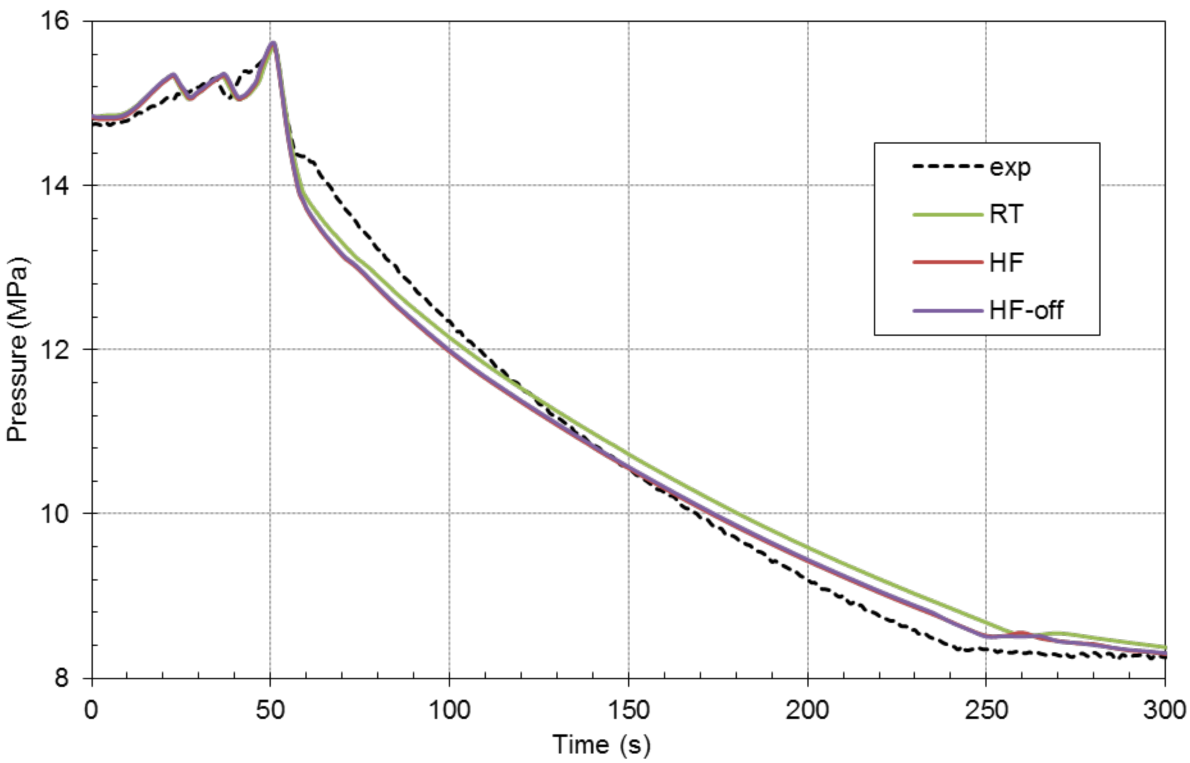

Figure 3-2 Primary Coolant System Pressure - Base Cases (0 – 300 s)

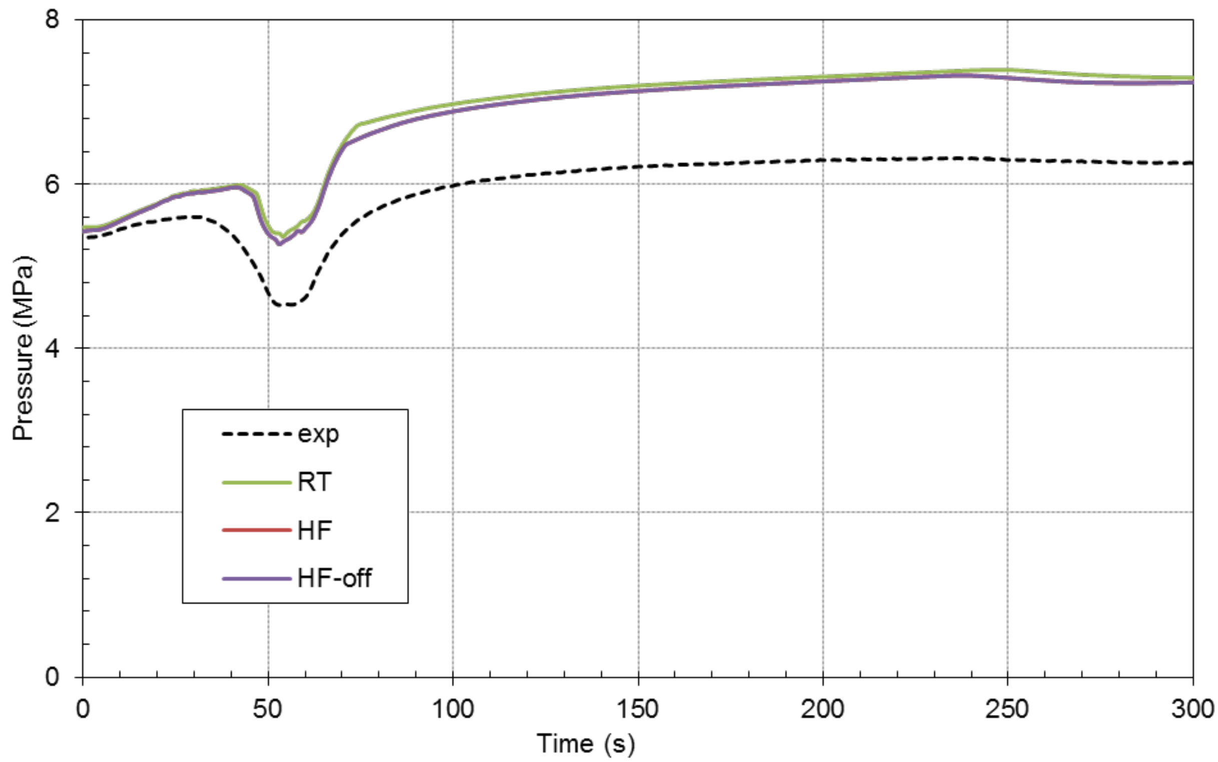

Figure 3-3 Secondary Coolant System Pressure - Base Cases (0 – 300 s)

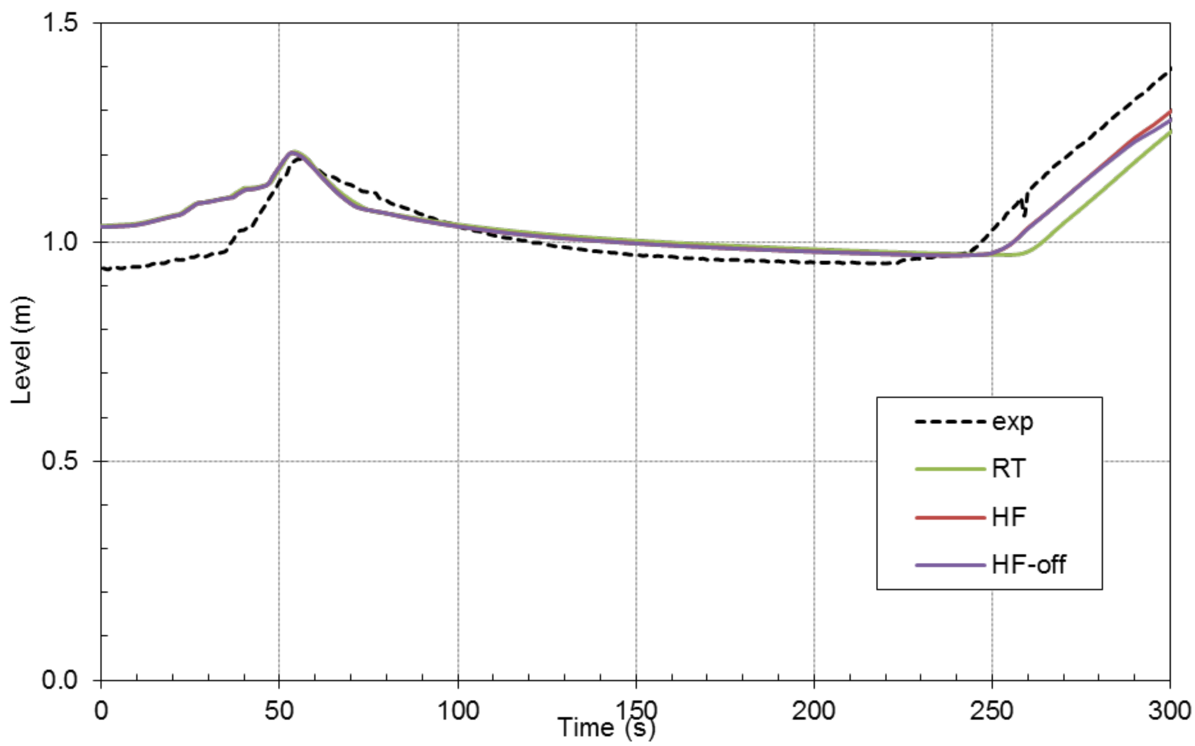

Figure 3-4 Pressurizer Level - Base Cases (0 – 300 s)

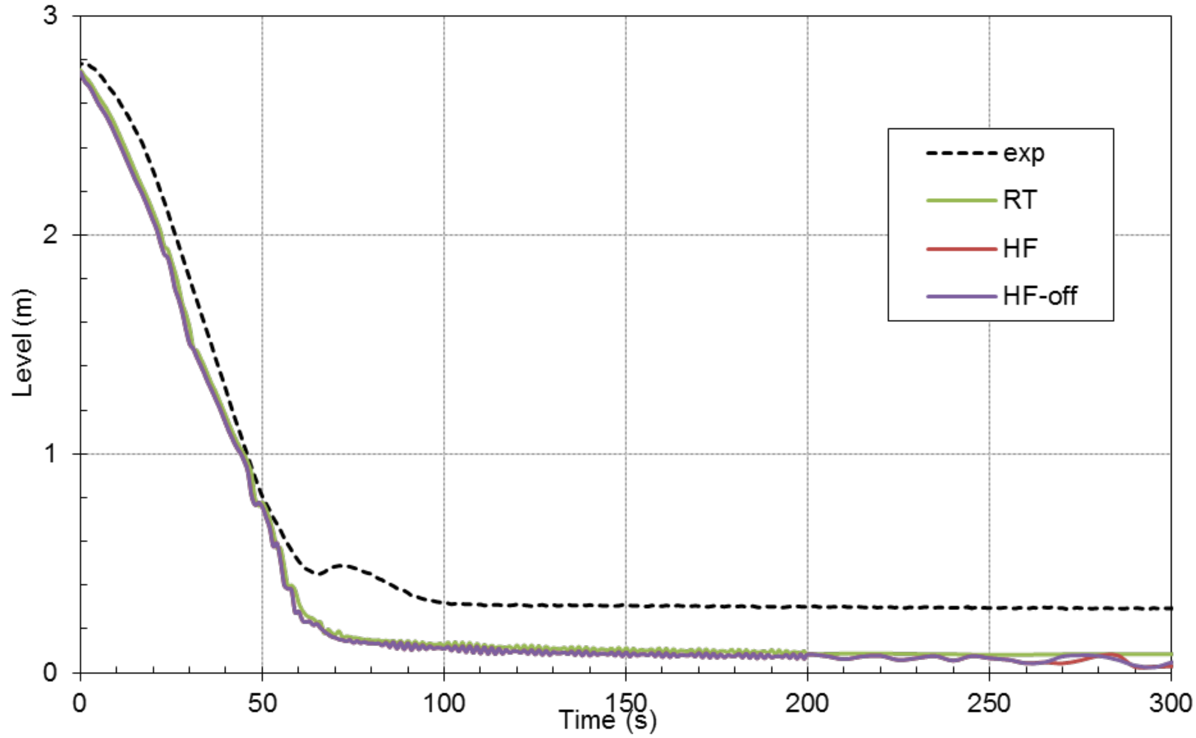

Figure 3-5 Steam Generator Level - Base Cases (0 – 300 s)

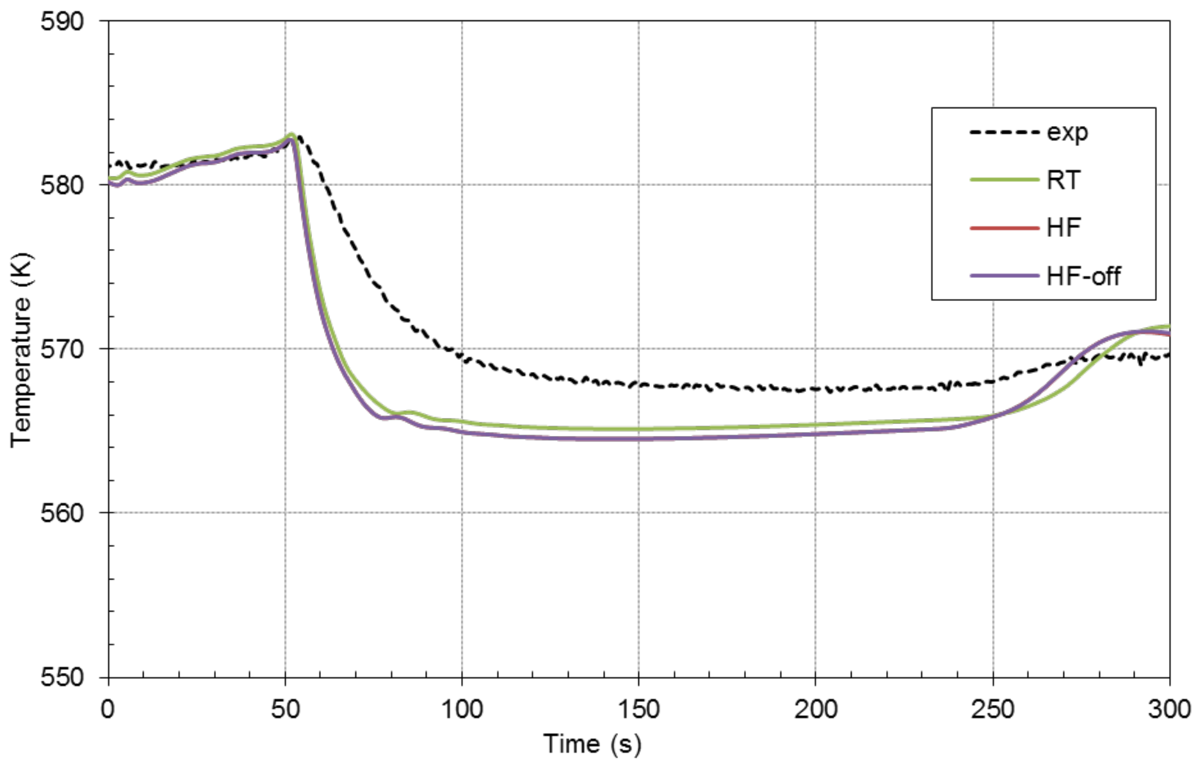

Figure 3-6 Hot Leg Temperature - Base Cases (0 – 300 s)

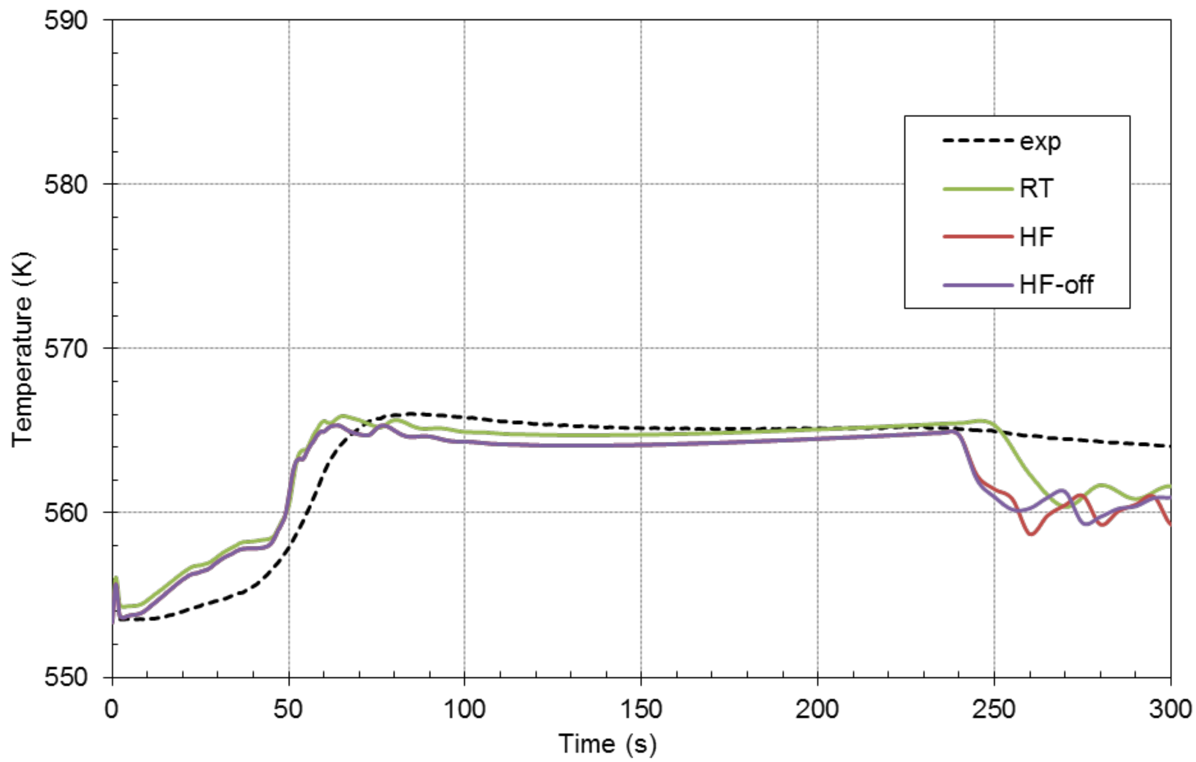

Figure 3-7 Cold Leg Temperature - Base Cases (0 – 300 s)

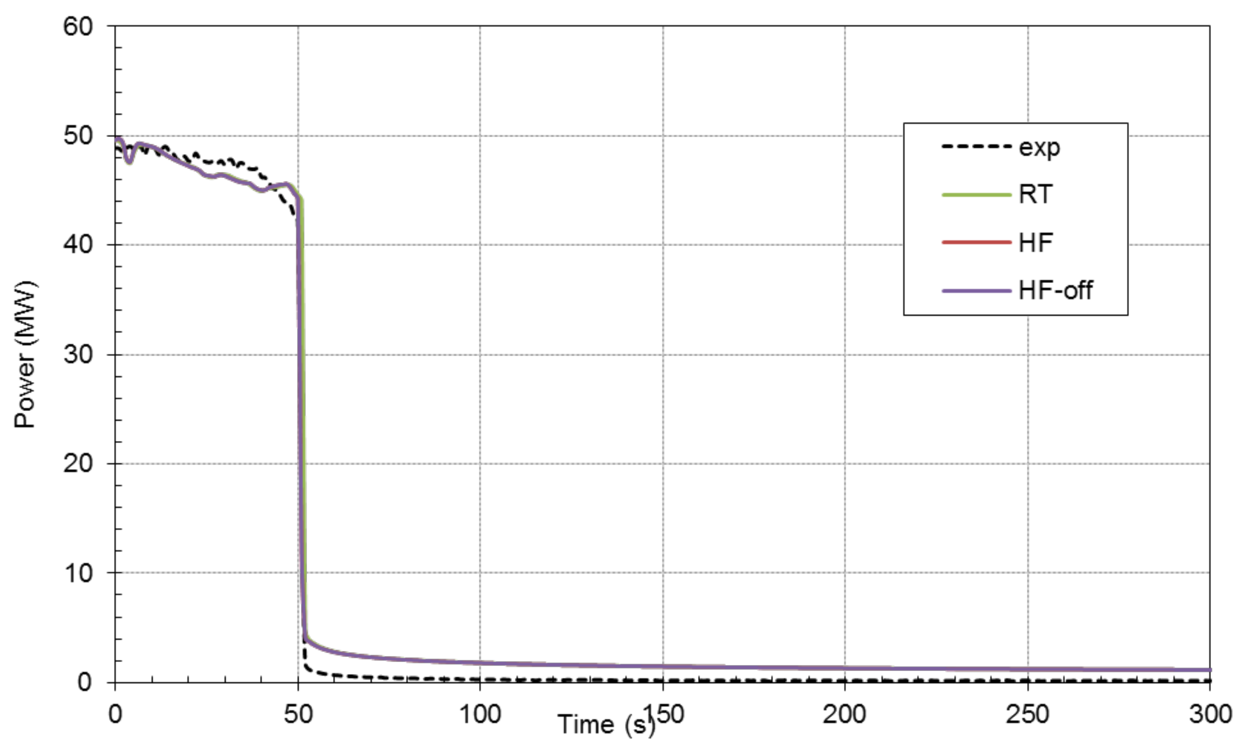

Figure 3-8 Reactor Power - Base Cases (0 – 300 s)

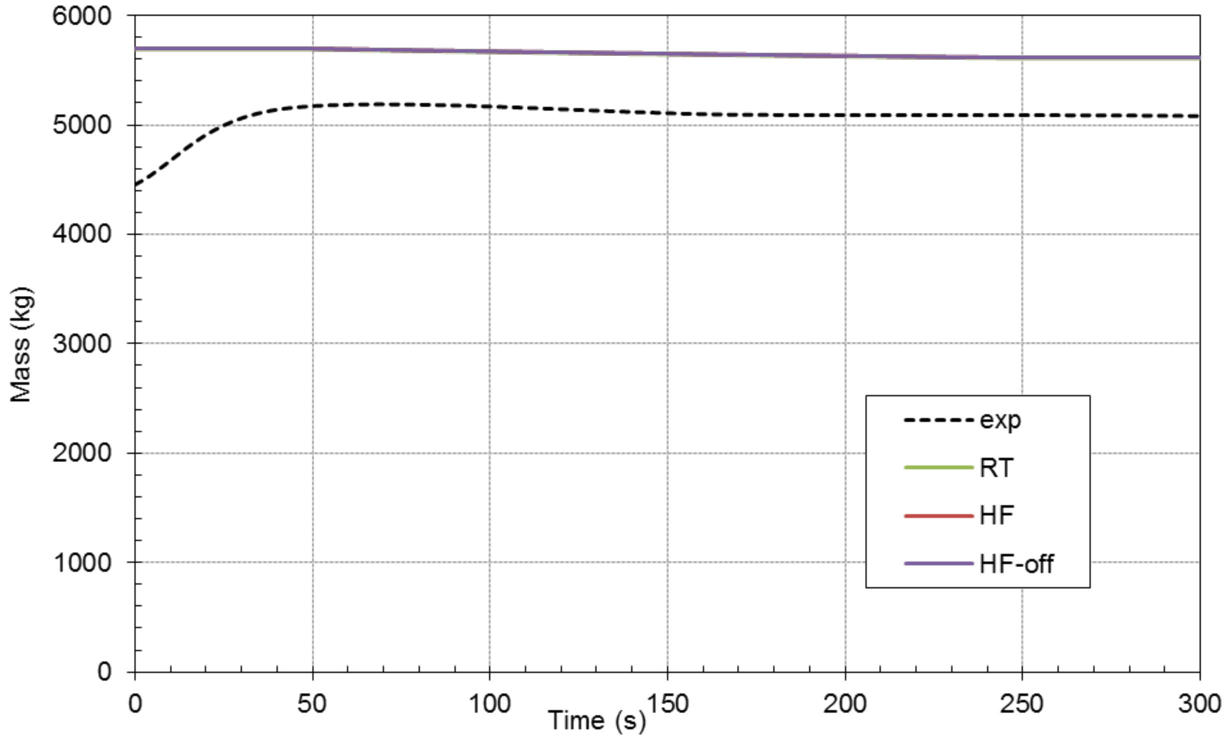

Figure 3-9 Primary Coolant System Inventory - Base Cases (0 – 300 s)

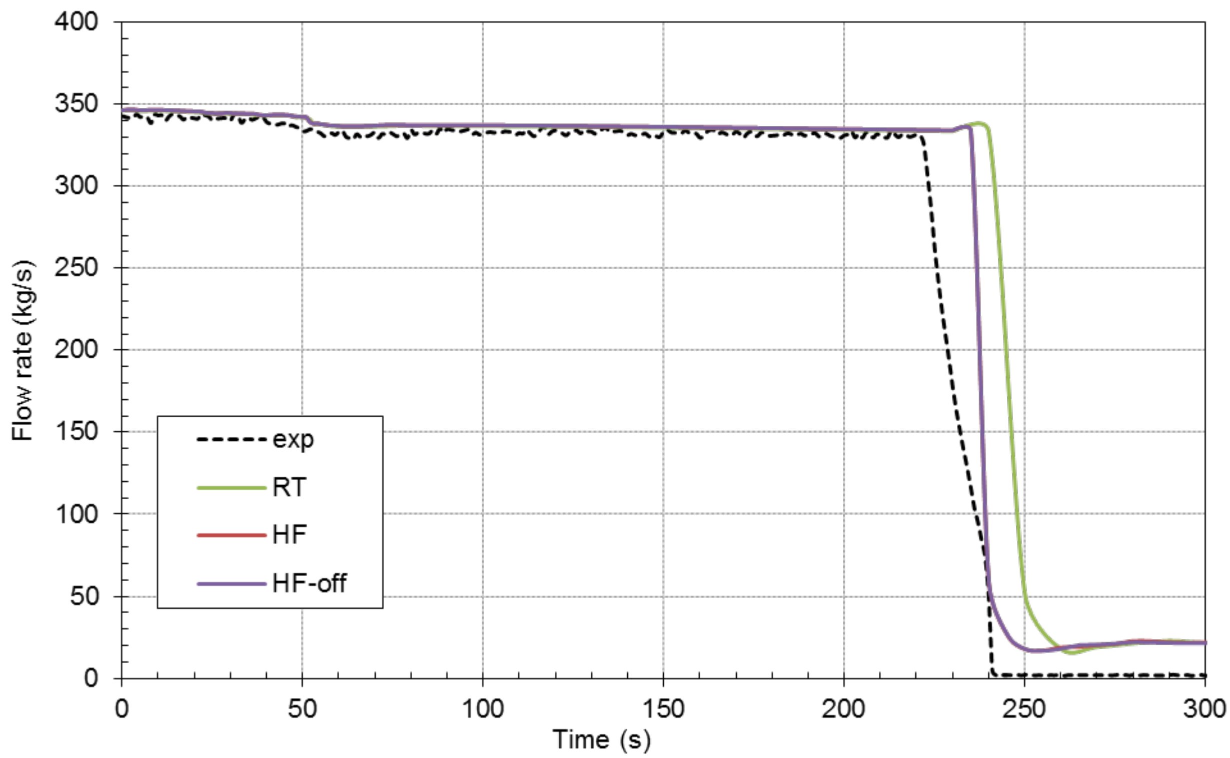

Figure 3-10 Intact Loop Mass Flow - Base Cases (0 – 300 s)

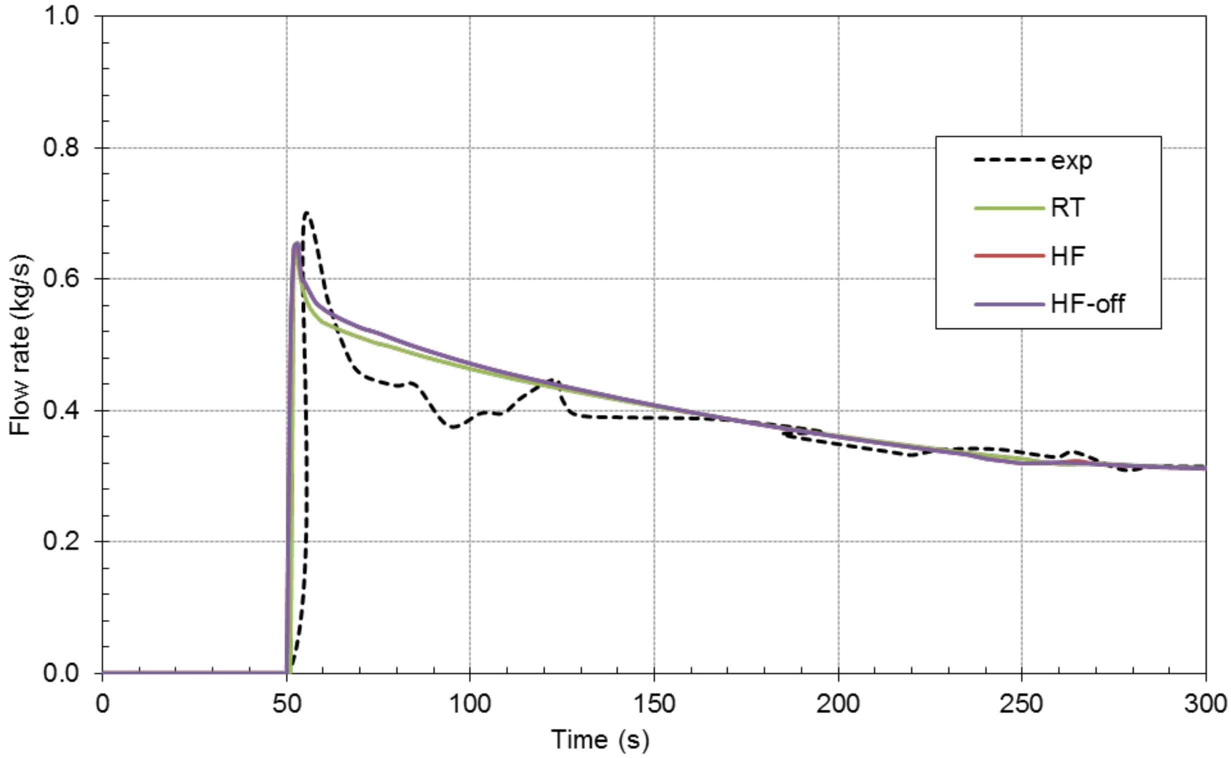

Figure 3-11 Pressurizer PORV Mass Flow - Base Cases (0 – 300 s)

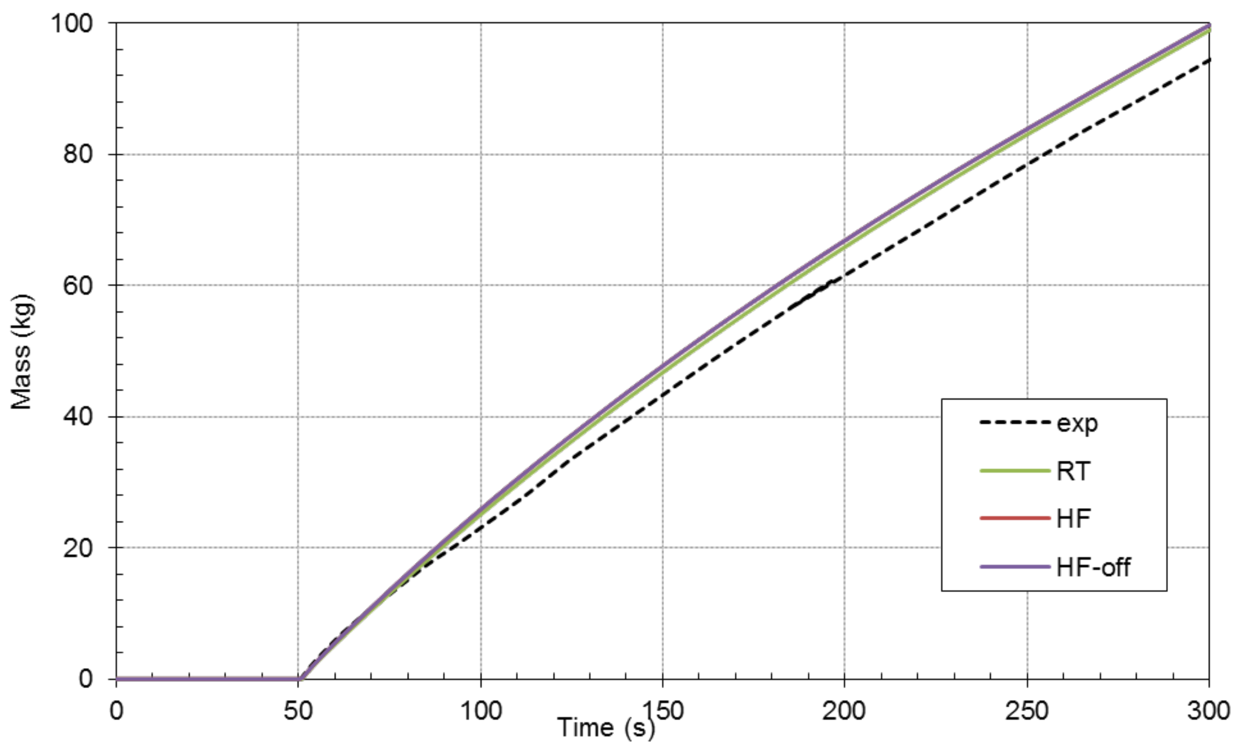

Figure 3-12 Pressurizer PORV Integrated Mass - Base Cases (0 – 300 s)

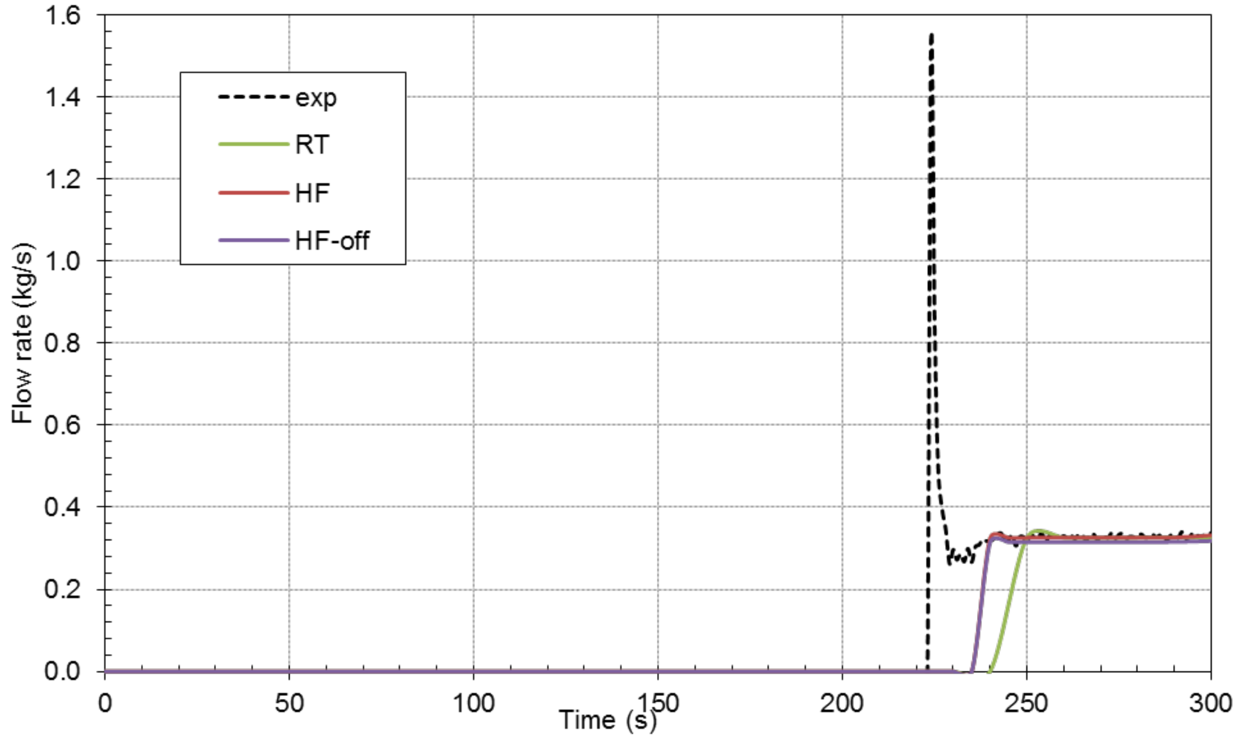

Figure 3-13 HPIS Mass Flow - Base Cases (0 – 300 s)

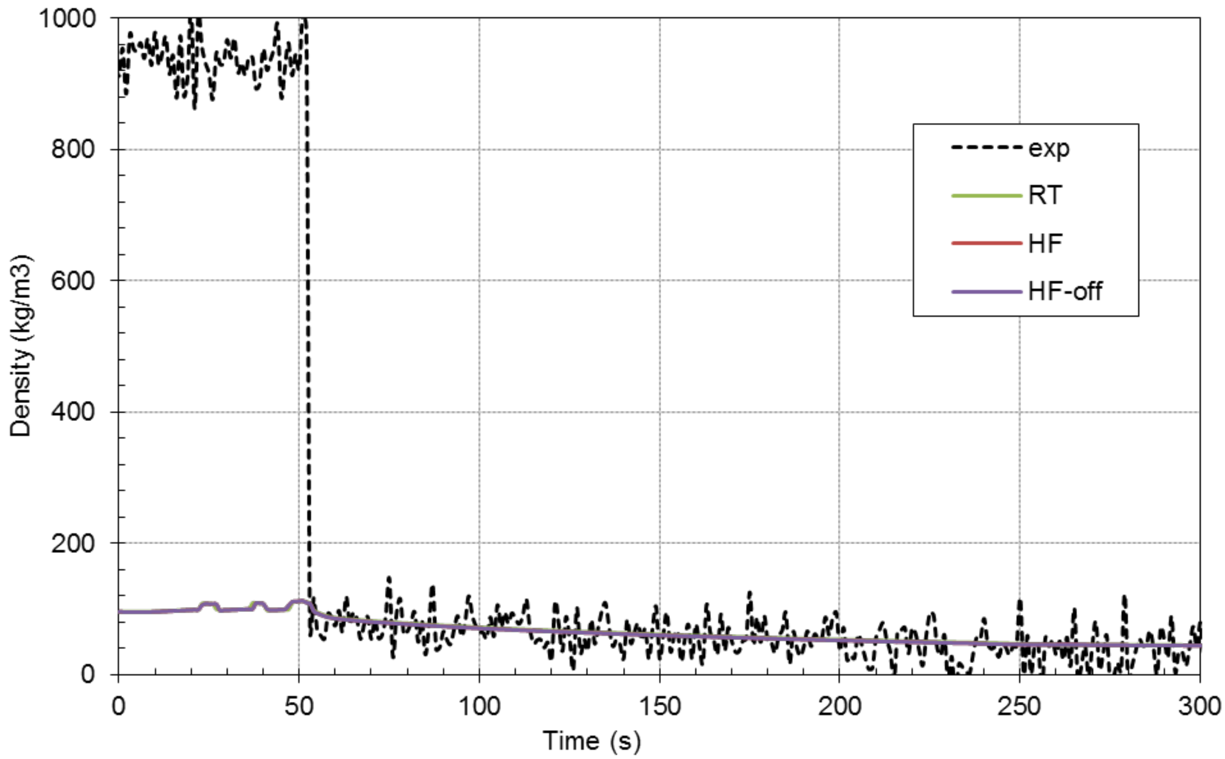

Figure 3-14 PORV Flow Density - Base Cases (0 – 300 s)

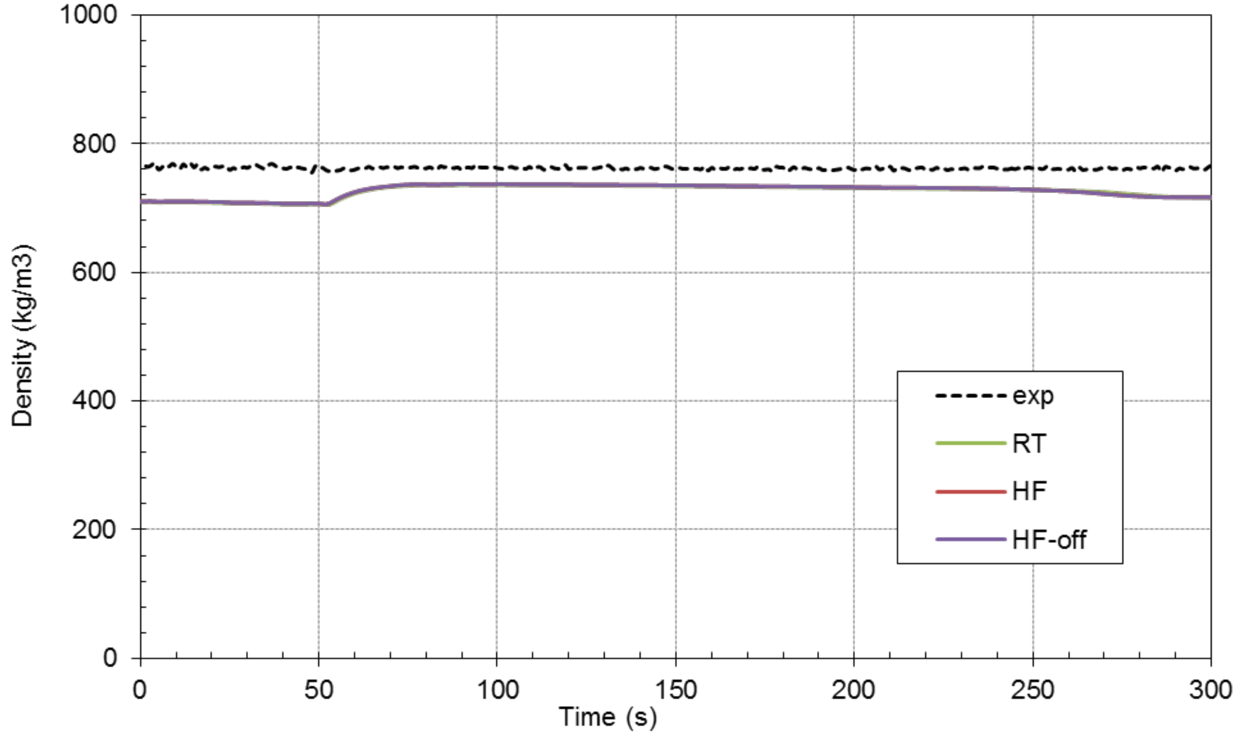

Figure 3-15 Hot Leg Density - Base Cases (0 – 300 s)

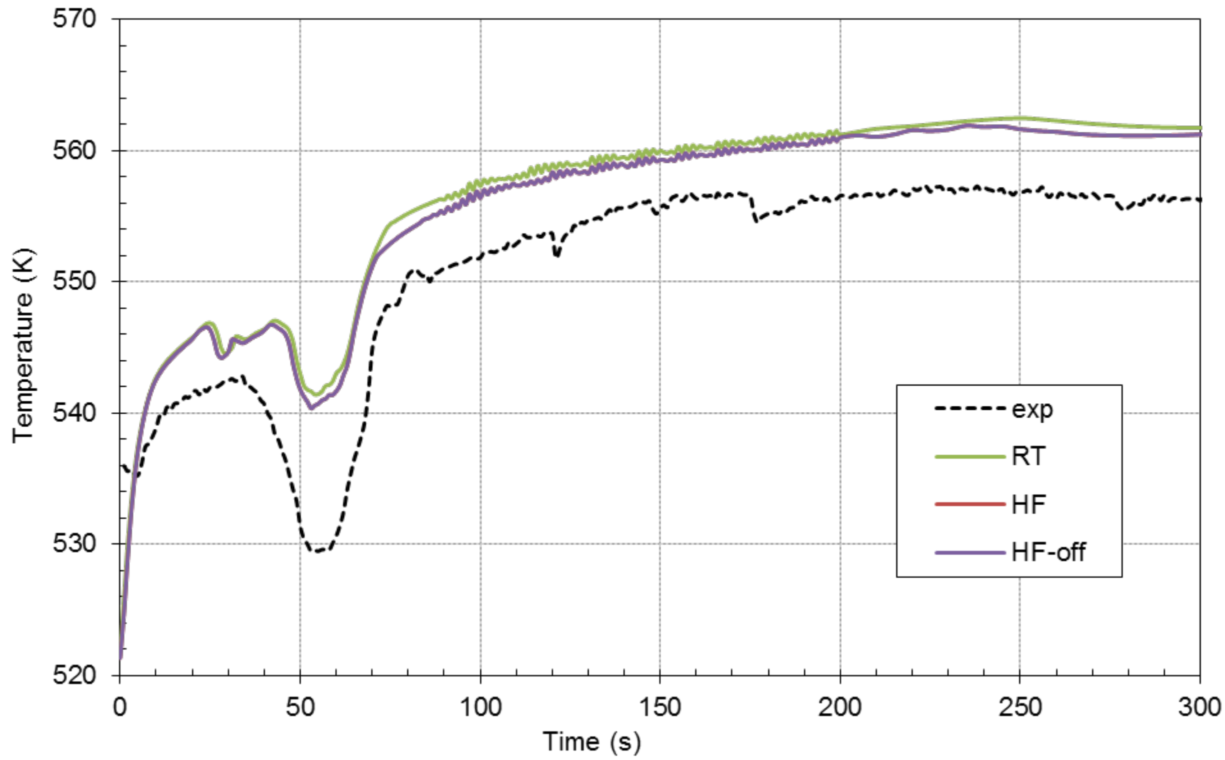

Figure 3-16 Steam Generator Downcomer Liquid Temperature - Base Cases (0 – 300 s)

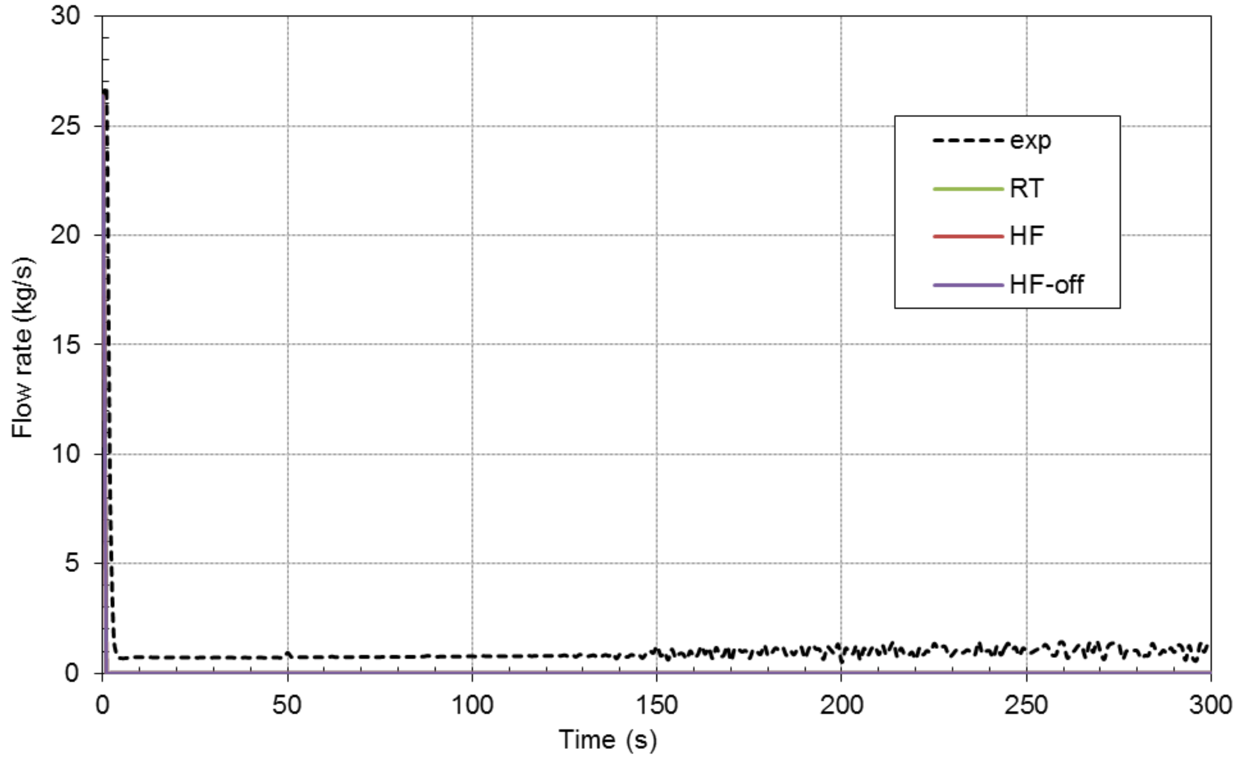

Figure 3-17 Feedwater Mass Flow - Base Cases (0 – 300 s)

3.2 Results Comparison of Base Cases (‘RT’, ‘HF’, ‘HF-off’) – Long Term Response (0 – 7000 s)

Figures 3-18 and 3-19 show the primary and secondary pressure predictions in the time period 0-7000 s, respectively. The primary and secondary pressures are accurately calculated during the steam discharge phase (50-380 s). After 380 s the liquid level in the pressurizer reaches the top of the pressurizer and two-phase mixture begins to be discharged through the PORV, giving a reduced energy discharge rate. This results in a repressurization of the primary system, the repressurization rate being somewhat overpredicted by RELAP5. Figure 3-24 shows that the PORV discharge density is overpredicted in the period of few thousand seconds when off-take model is not used. As a consequence there is an overprediction of the rate of increase in primary system pressure. For measured secondary pressure there was a gradual decrease as the result of MSCV leakage, as reported in [7]. This partly explains larger increase in the calculated secondary coolant system pressure after the reactor trip (see Figure 3-19). The PORV flow density shown in Figure 3-24 trend is similar to PORV mass flow (see Figure 3-25), which has not be digitalized from plot in [6] due to oscillatory behavior (see the measured PORV density in Figure 3-24). The injected HPIS flow is shown in see Figure 3-28, while the discharged PORV mass and HPIS injected mass are shown in Figure 3-26 and Figure 3-27, respectively. In all calculations HPIS flow is greater than the PORV discharged flow after 4000 s, while in experiment this occurred at 2370 s. Also it should be noted that in the study [7] using SPACE computer code the HPIS flow was boundary condition, while in the presented calculations the HPIS injection depends on the primary coolant pressure.

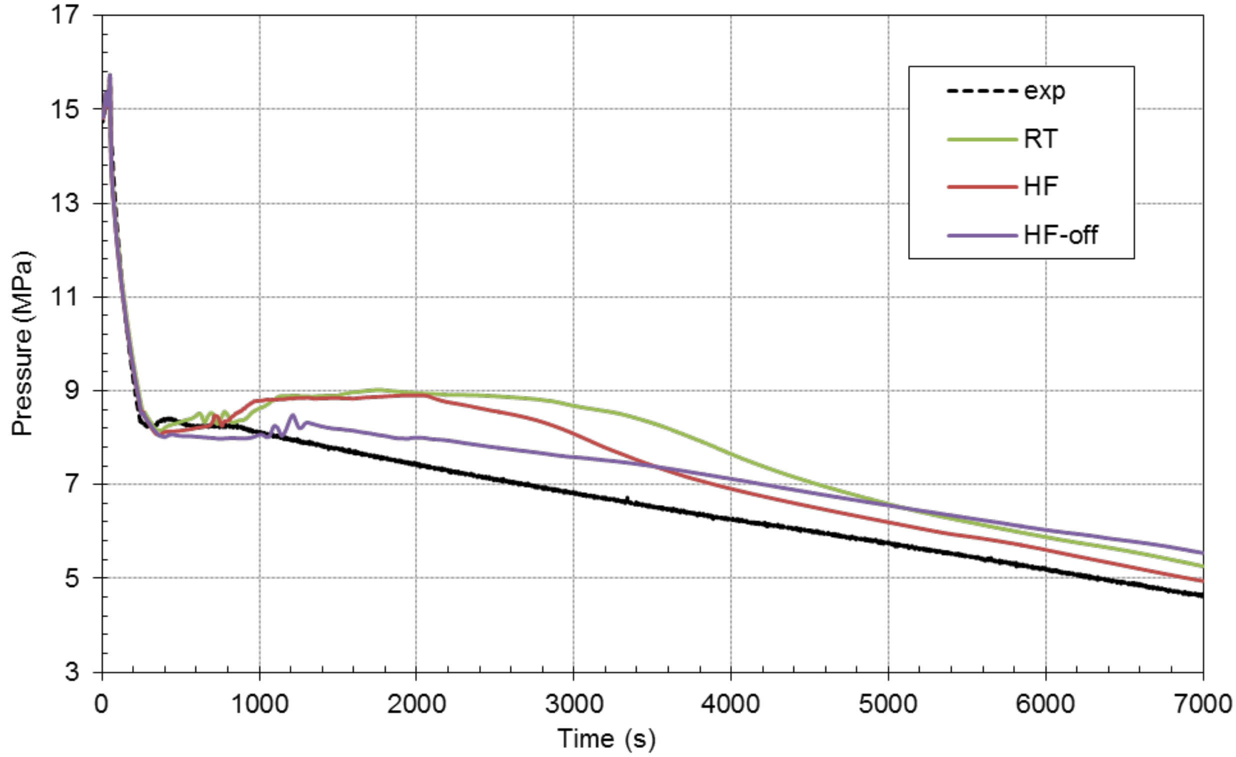

Figure 3-18 Primary Coolant System Pressure - Base Cases (0 – 7000 s)

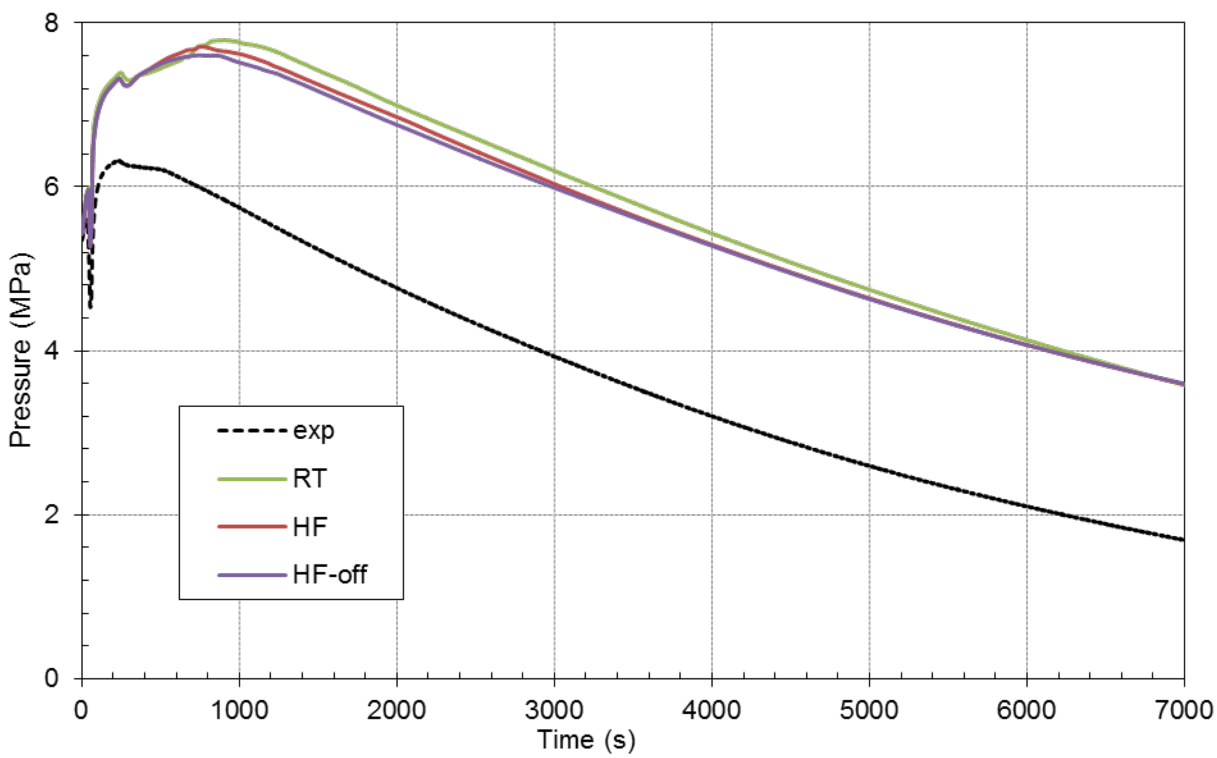

Figure 3-19 Secondary Coolant System Pressure - Base Cases (0 – 7000 s)

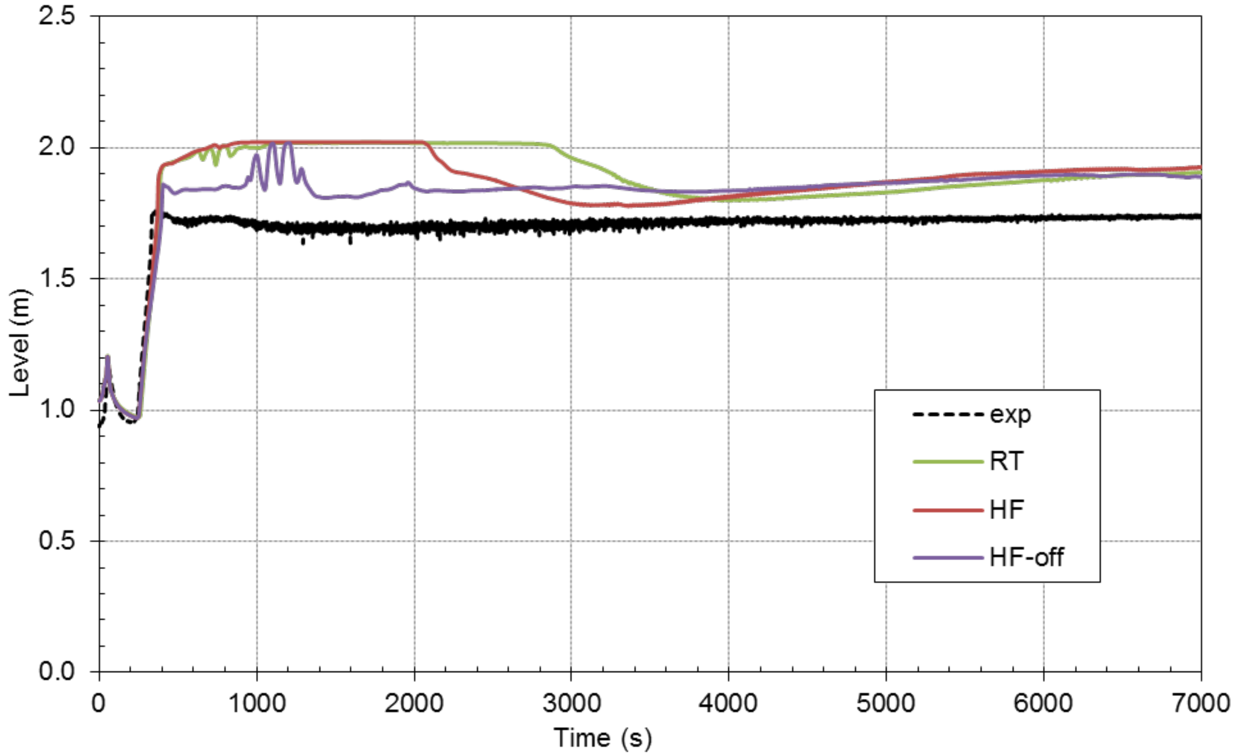

Figure 3-20 Pressurizer Level - Base Cases (0 – 7000 s)

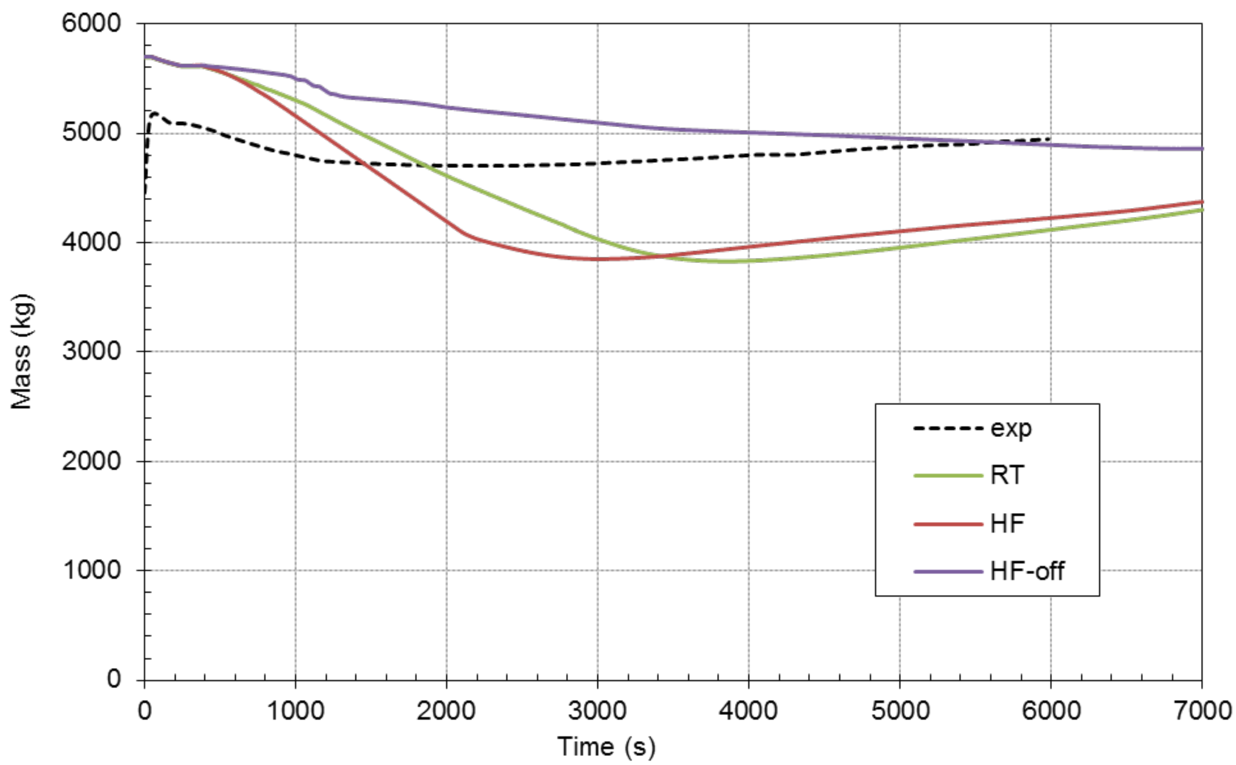

Figure 3-21 Primary Coolant System Inventory - Base Cases (0 – 7000 s)

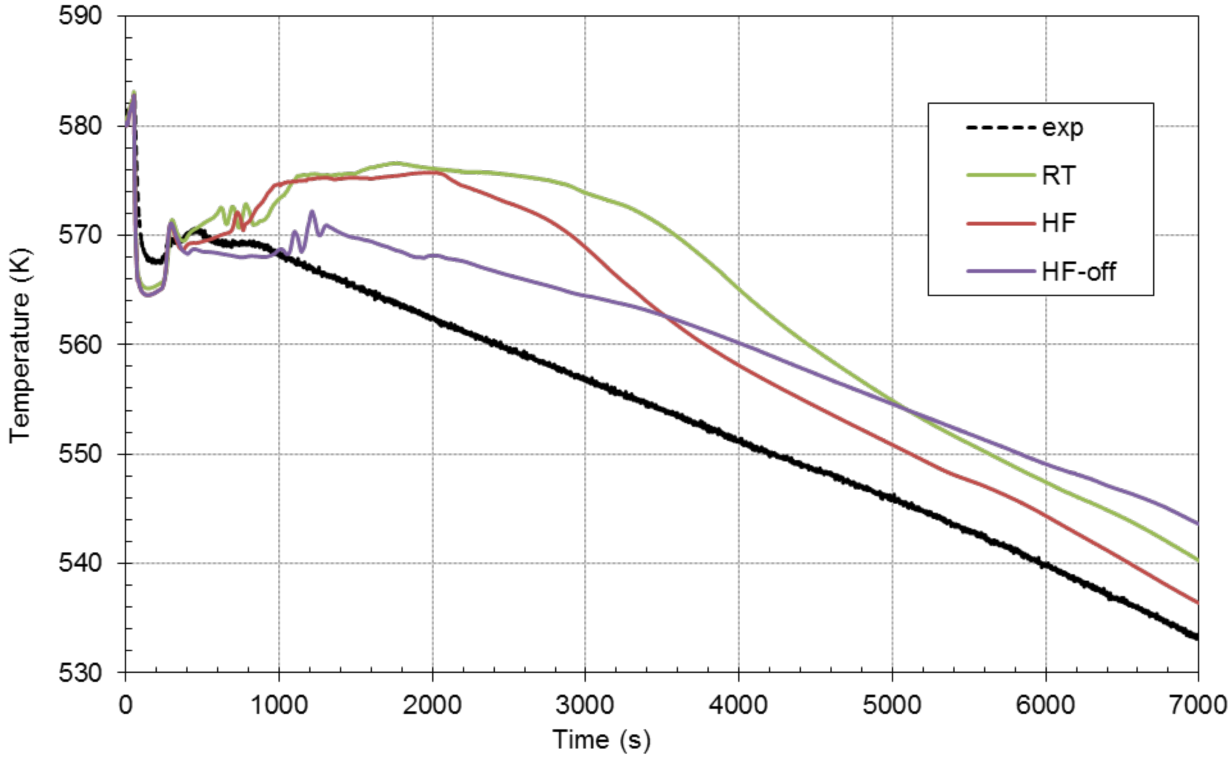

Figure 3-22 Hot Leg Temperature - Base Cases (0 – 7000 s)

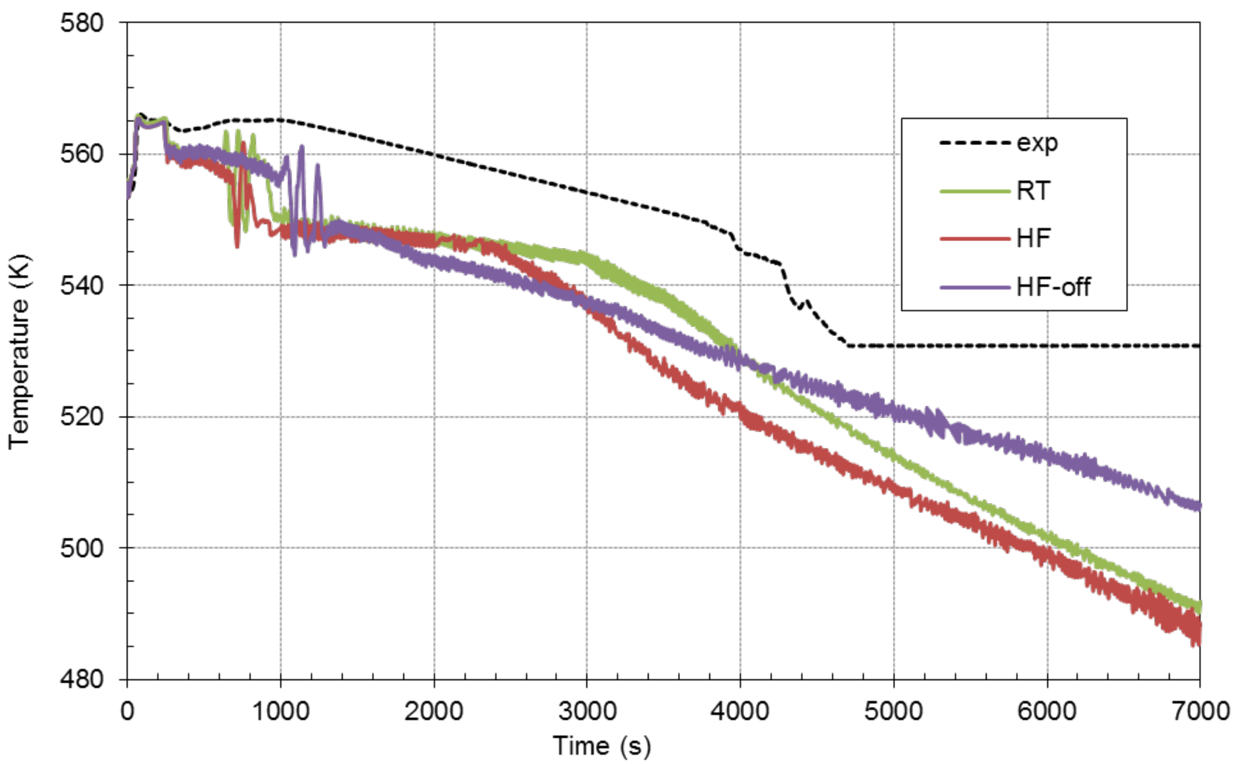

Figure 3-23 Cold Leg Temperature - Base Cases (0 – 7000 s)

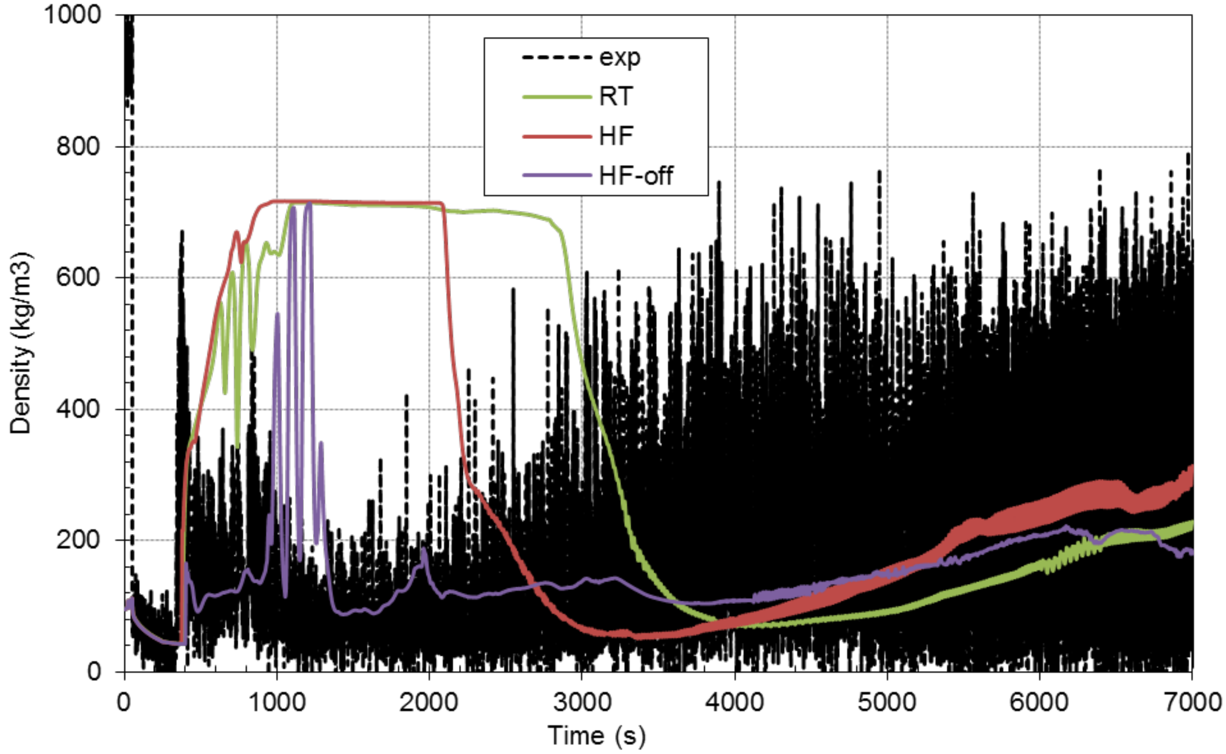

Figure 3-24 PORV Flow Density - Base Cases (0 – 7000 s)

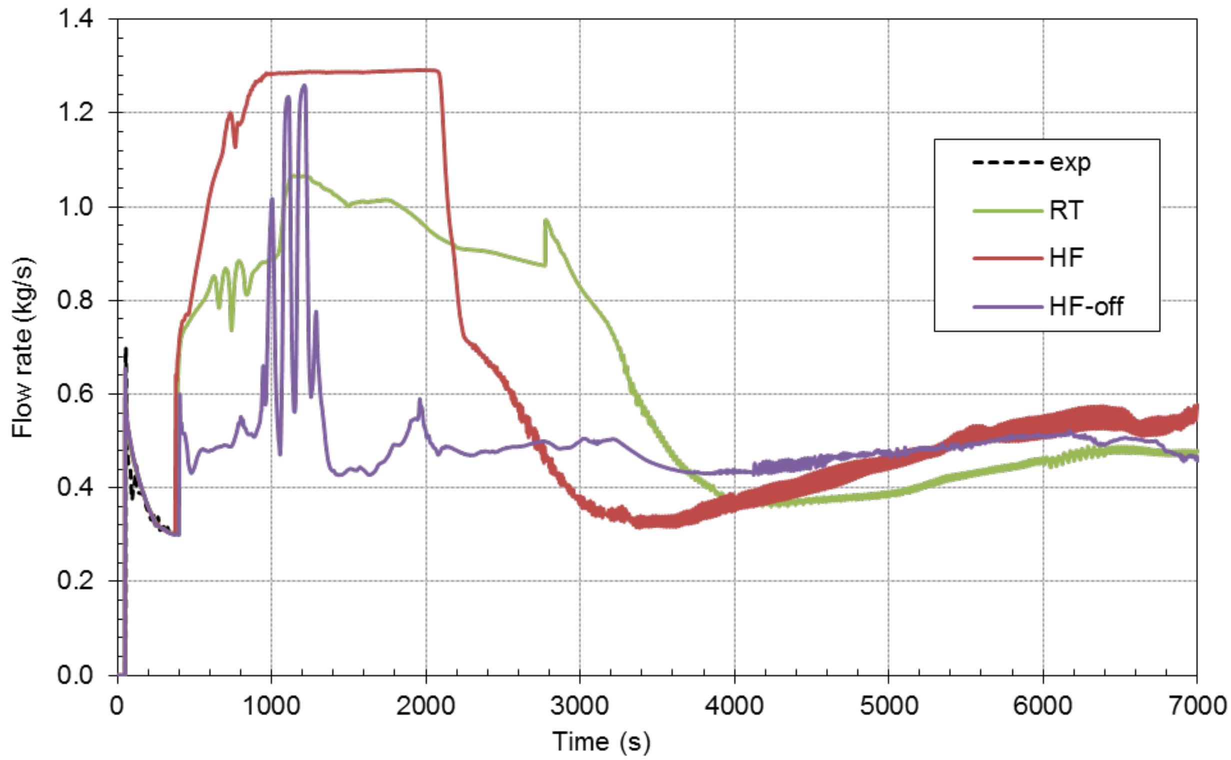

Figure 3-25 Pressurizer PORV Mass Flow - Base Cases (0 – 7000 s)

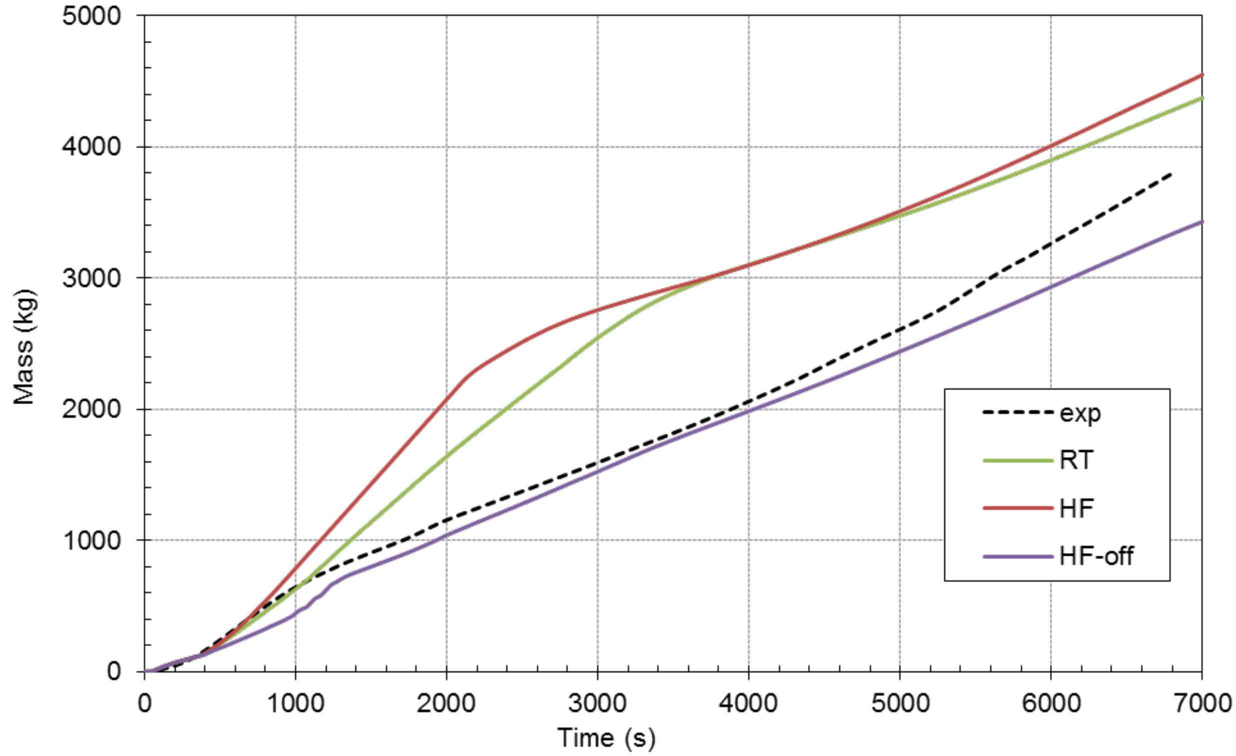

Figure 3-26 Pressurizer PORV Integrated Mass - Base Cases (0 – 7000 s)

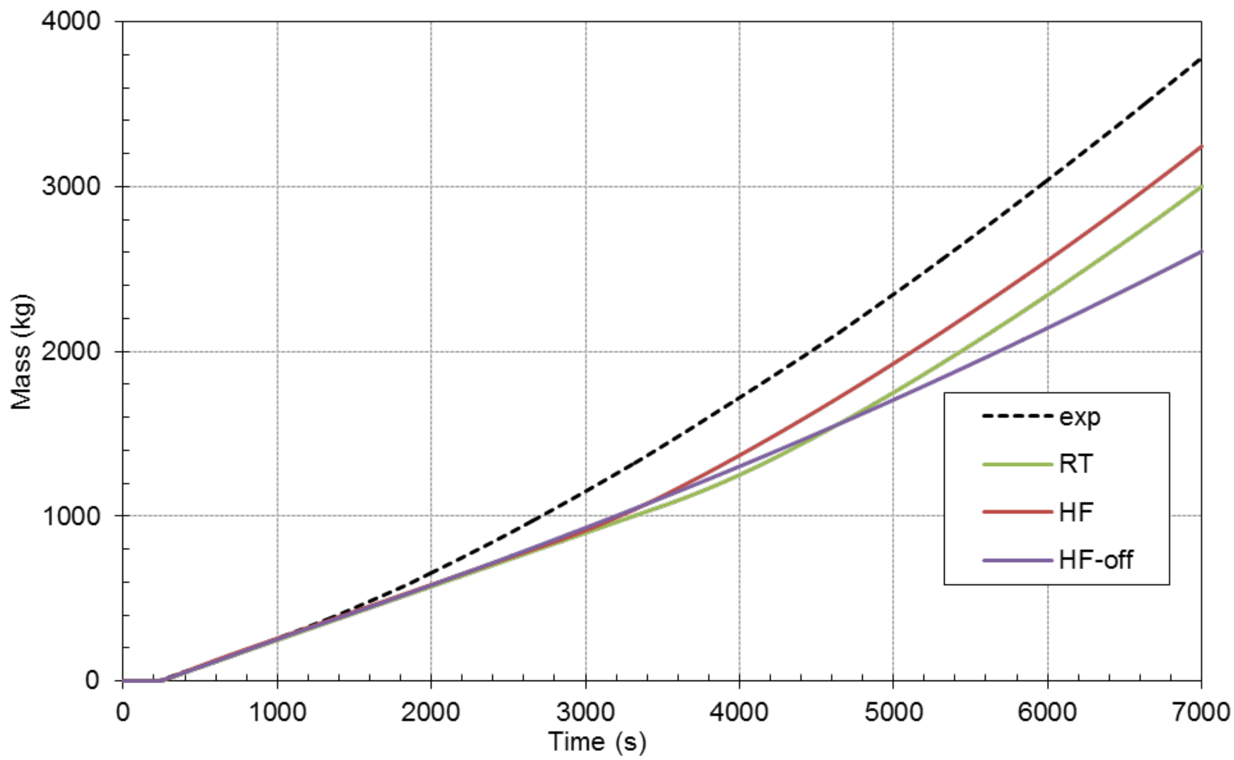

Figure 3-27 HPIS Integrated Mass - Base Cases (0 – 7000 s)

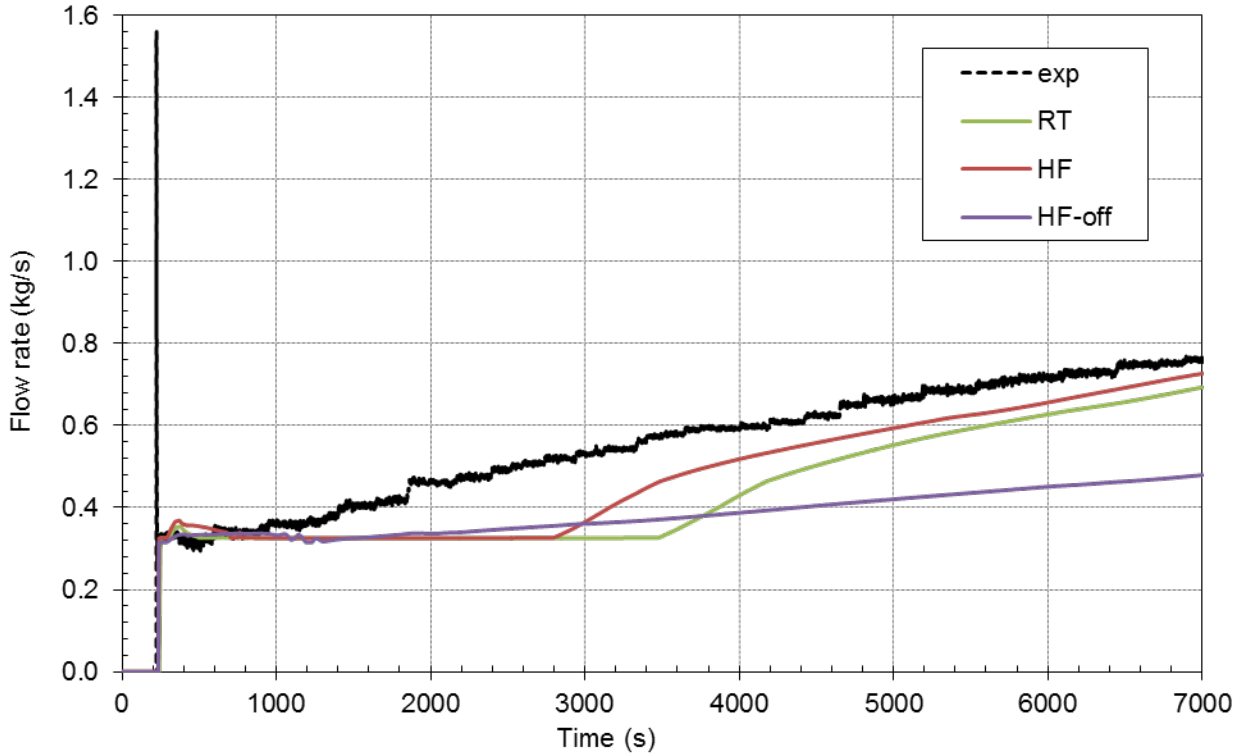

Figure 3-28 HPIS Mass Flow - Base Cases (0 – 7000 s)

3.3 Results Comparison of 'RT' Sensitivity Cases – Long Term Response (0 – 7000 s)

Figures 3-29 through 3-39 show same variables as shown in Section 3.2 for the time interval 0-7000 s. The dependence on the variation of primary coolant system heat losses to containment is shown. It can be seen in Figure 3-29, the higher are the heat losses, the faster is the pressure drop. The primary side heat losses have negligible effect on the secondary side pressure as expected (see Figure 3-30). In the long term the larger heat losses have small influence on PORV discharge (see Figure 3-35, Figure 3-36 and Figure 3-37), they resulted in decrease of hot (see Figure 3-33) and cold leg temperature (see Figure 3-34), and earlier HPIS flow start (see Figure 3-39) causes more mass to be injected (see Figure 3-38) resulting in PCS inventory increase (see Figure 3-32).

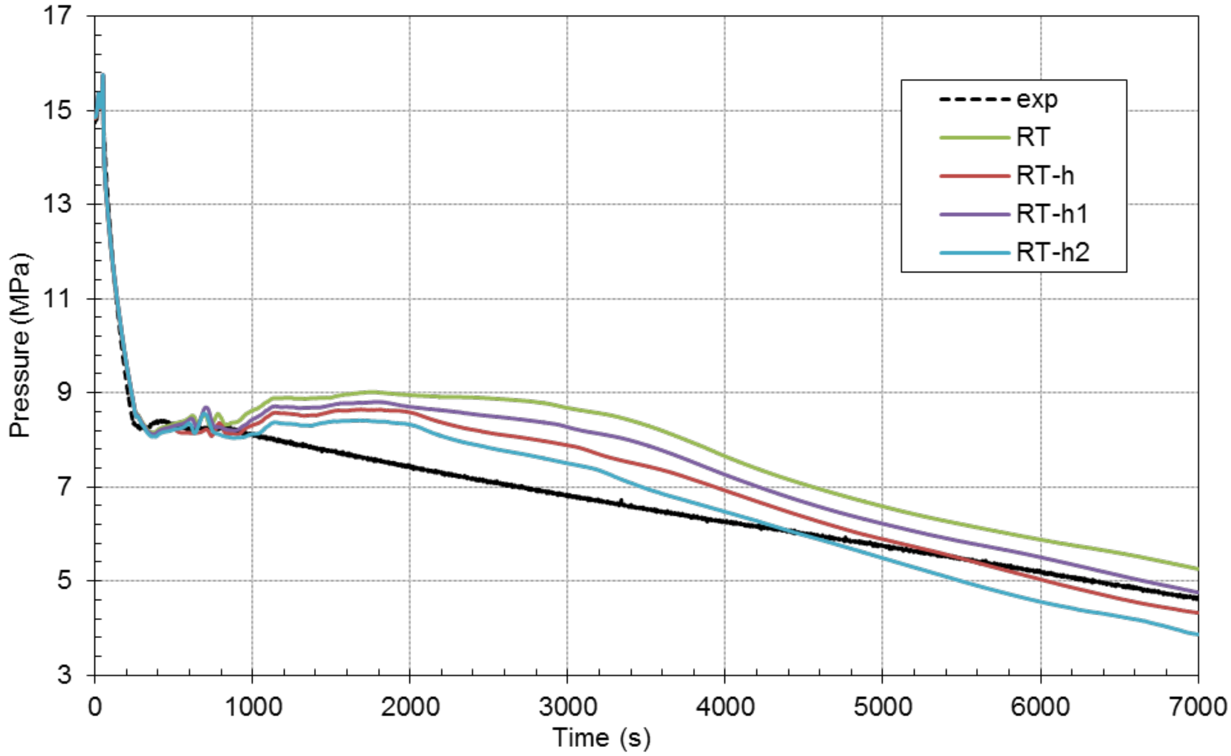

Figure 3-29 Primary Coolant System Pressure - 'RT' Sensitivity Cases (0 – 7000 s)

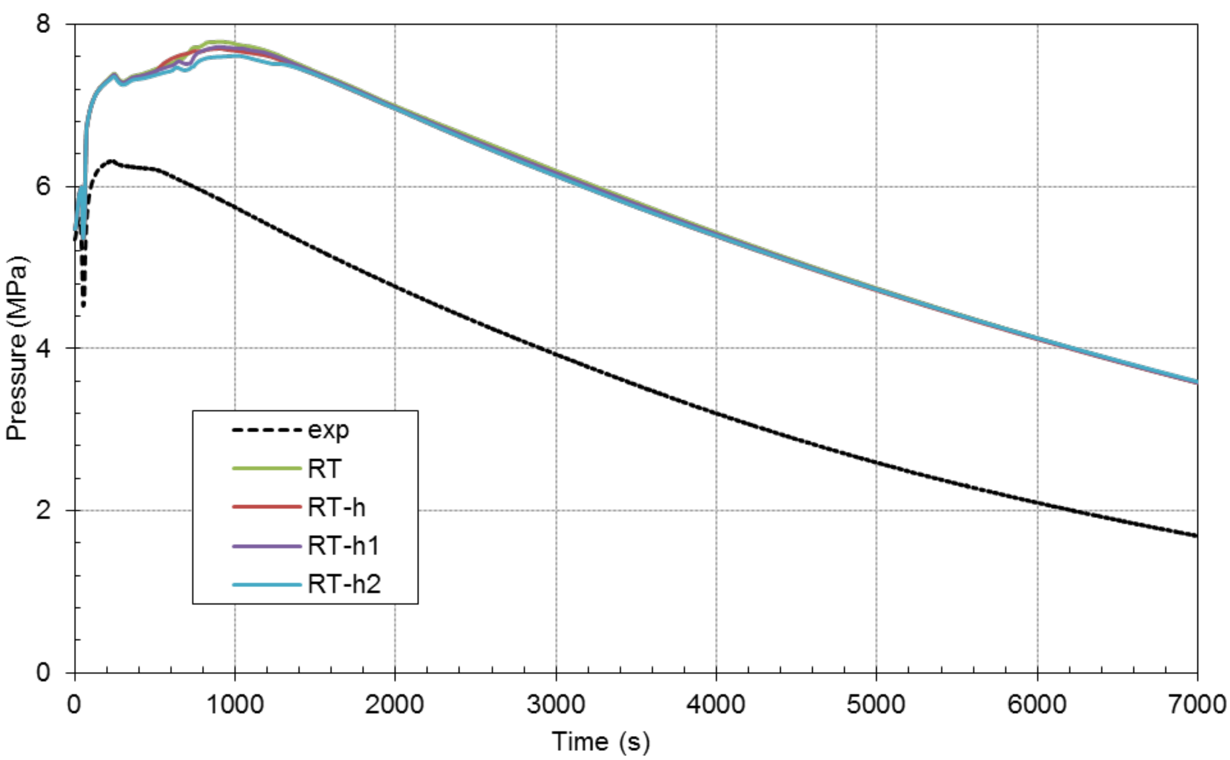

Figure 3-30 Secondary Coolant System Pressure - 'RT' Sensitivity Cases (0 – 7000 s)

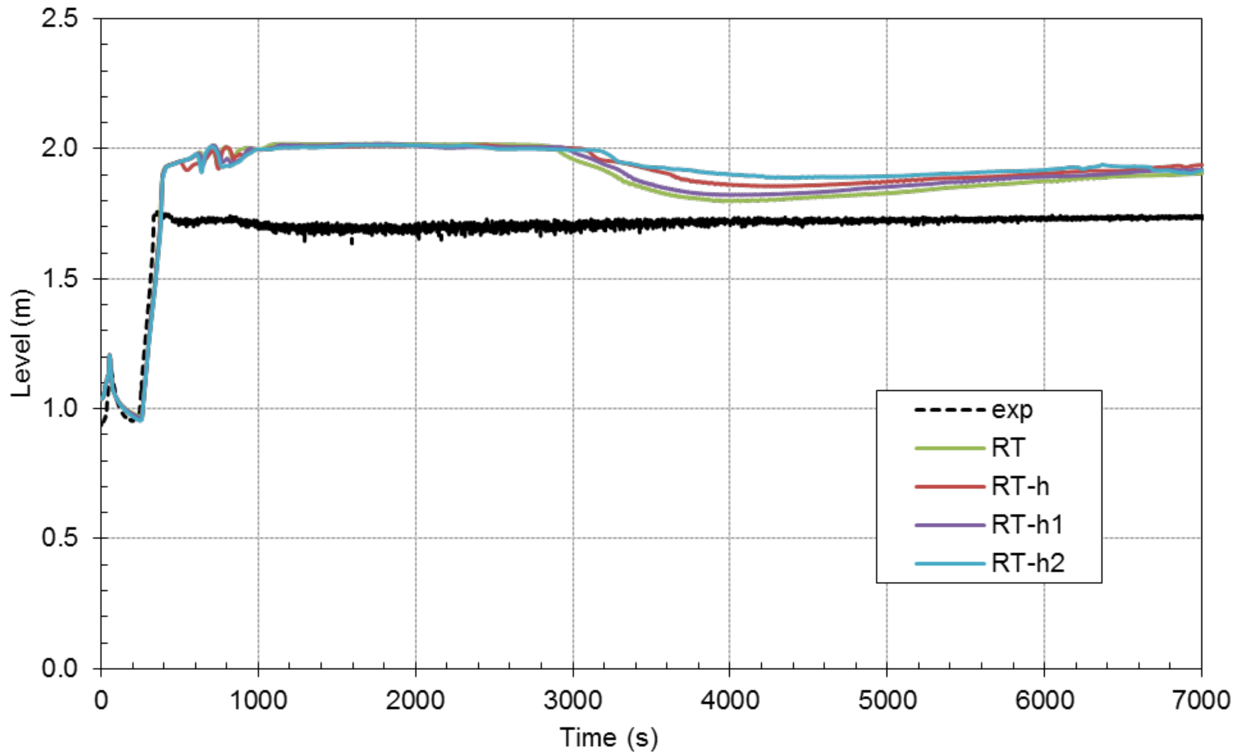

Figure 3-31 Pressurizer Level - 'RT' Sensitivity Cases (0 – 7000 s)

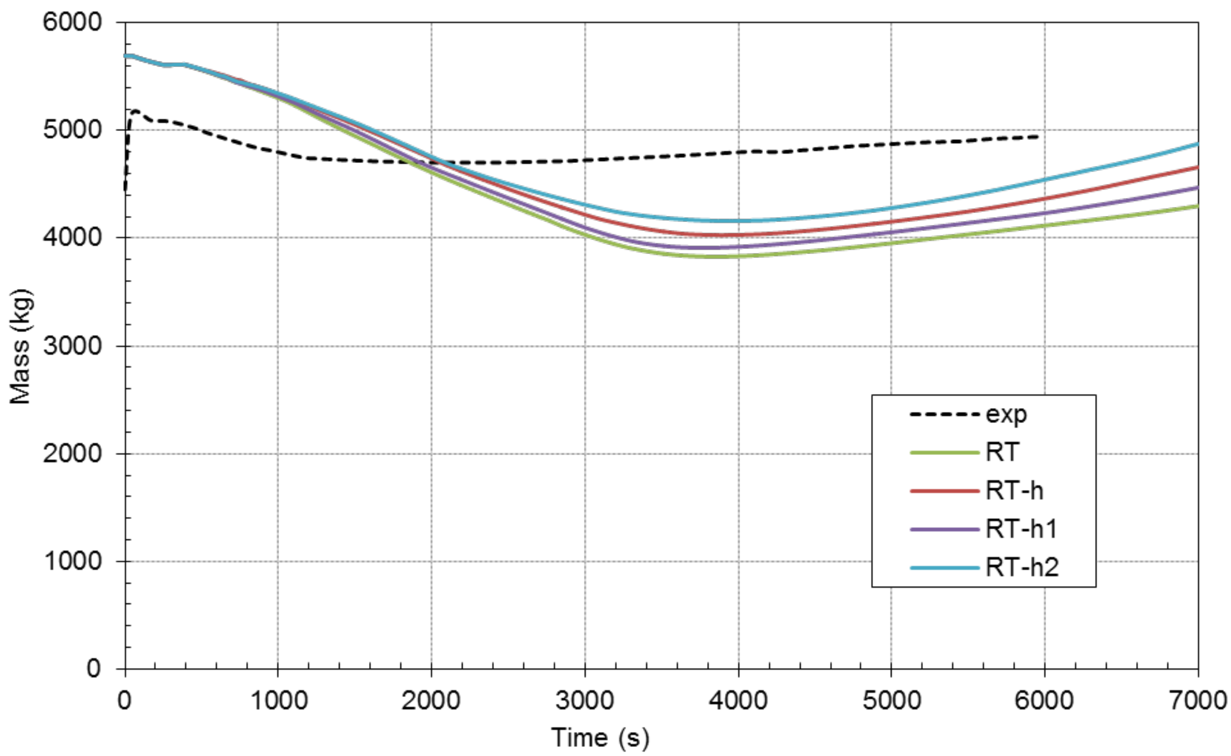

Figure 3-32 Primary Coolant System Inventory - 'RT' Sensitivity Cases (0 – 7000 s)

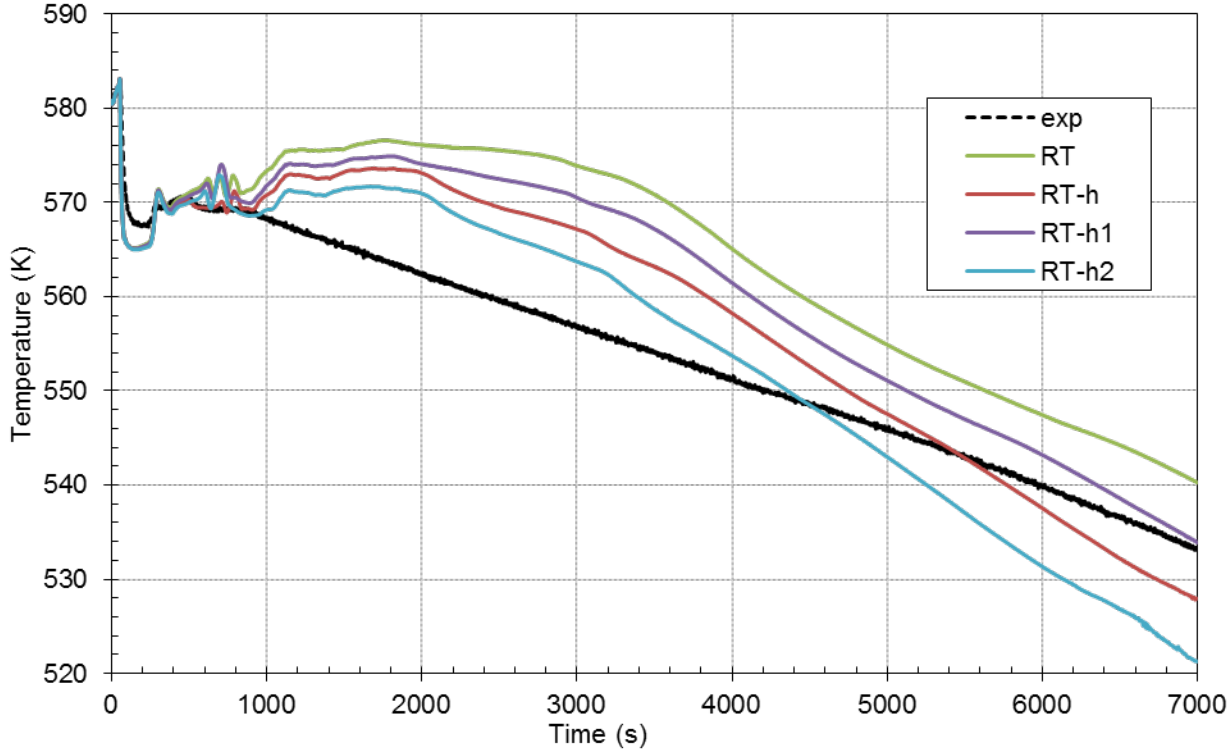

Figure 3-33 Hot Leg Temperature - 'RT' Sensitivity Cases (0 – 7000 s)

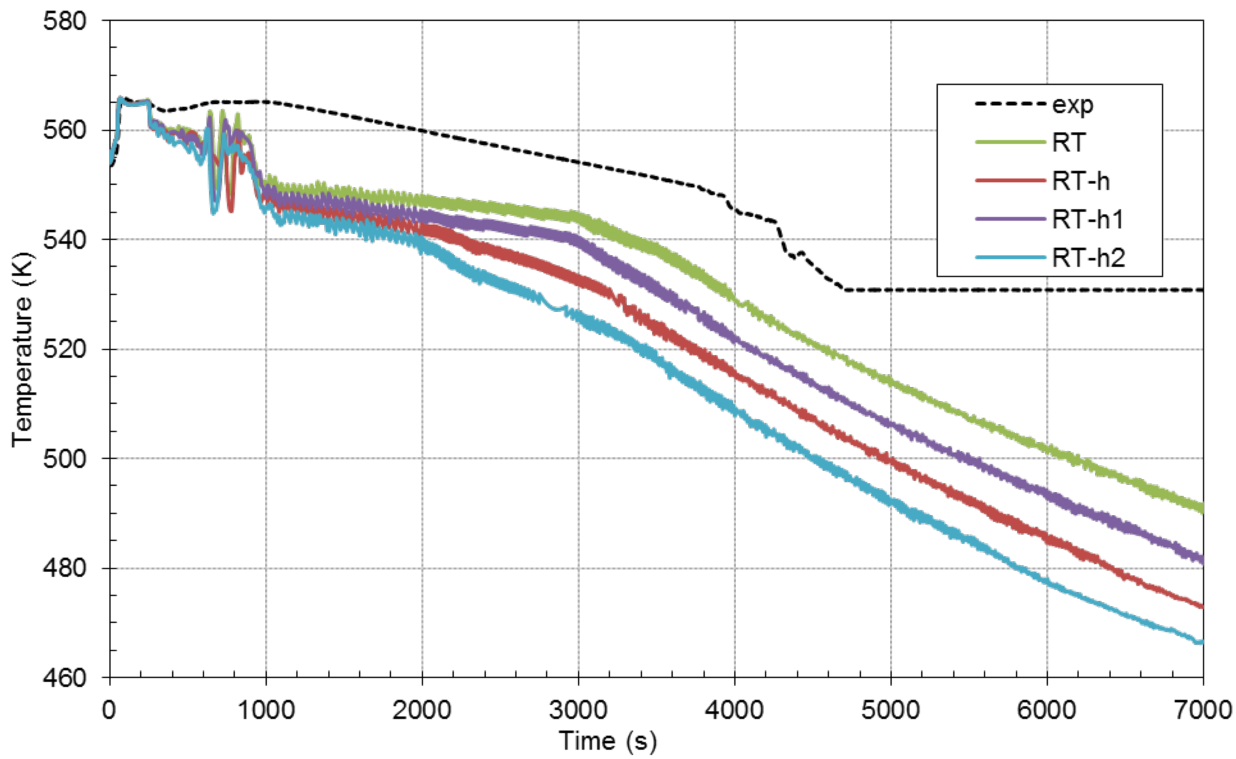

Figure 3-34 Cold Leg Temperature - 'RT' Sensitivity Cases (0 – 7000 s)

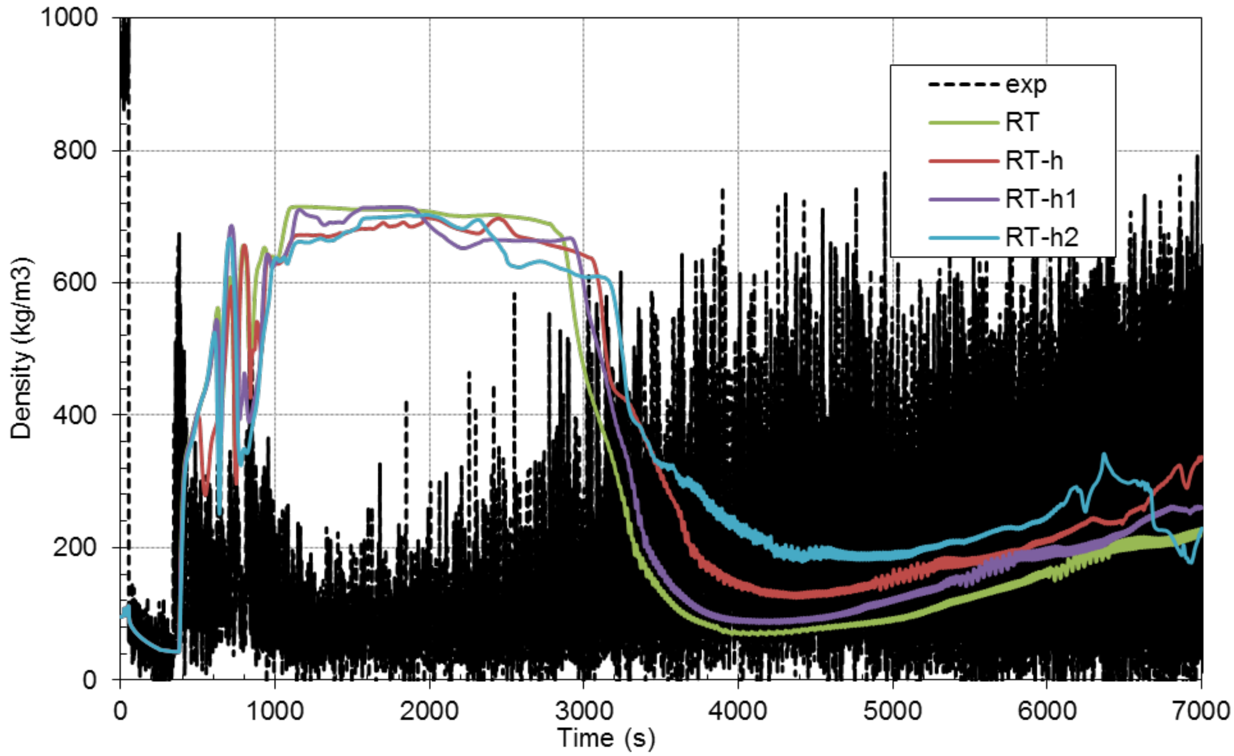

Figure 3-35 PORV Flow Density - 'RT' Sensitivity Cases (0 – 7000 s)

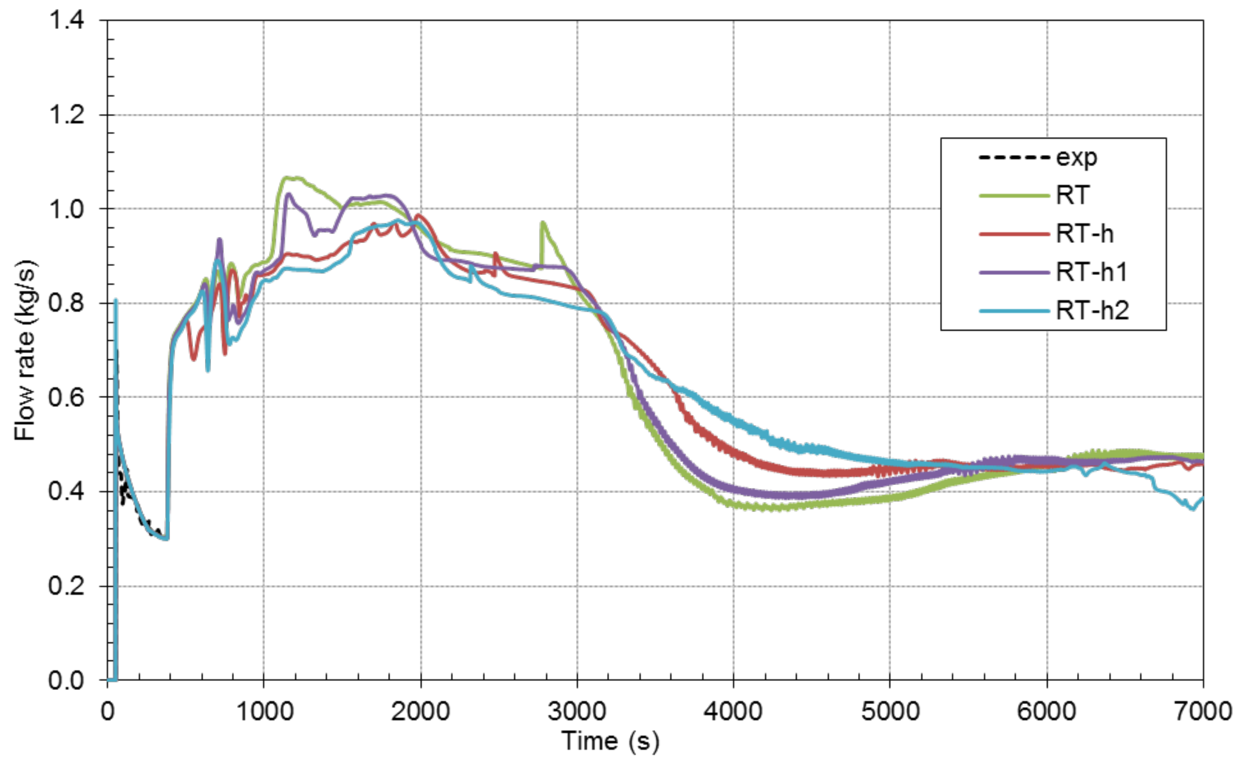

Figure 3-36 Pressurizer PORV Mass Flow - 'RT' Sensitivity Cases (0 – 7000 s)

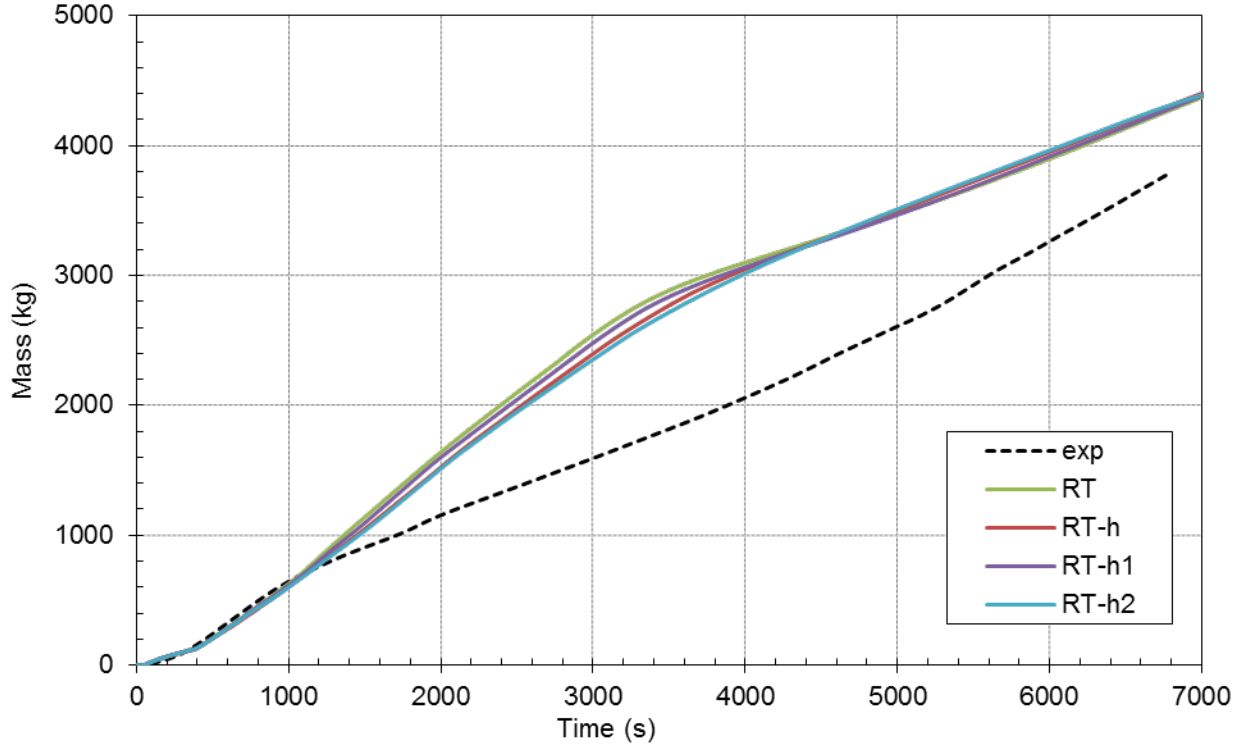

Figure 3-37 Pressurizer PORV Integrated Mass - ‘RT’ Sensitivity Cases (0 – 7000 s)

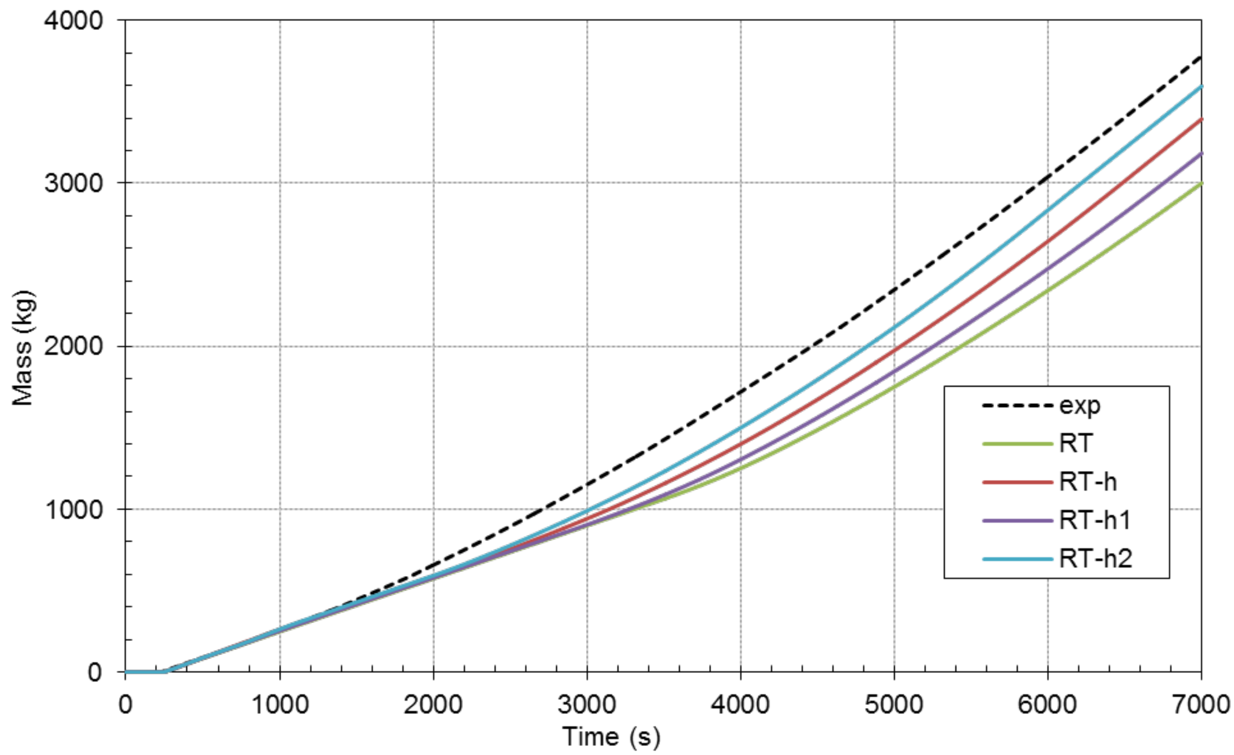

Figure 3-38 HPIS Integrated Mass - ‘RT’ Sensitivity Cases (0 – 7000 s)

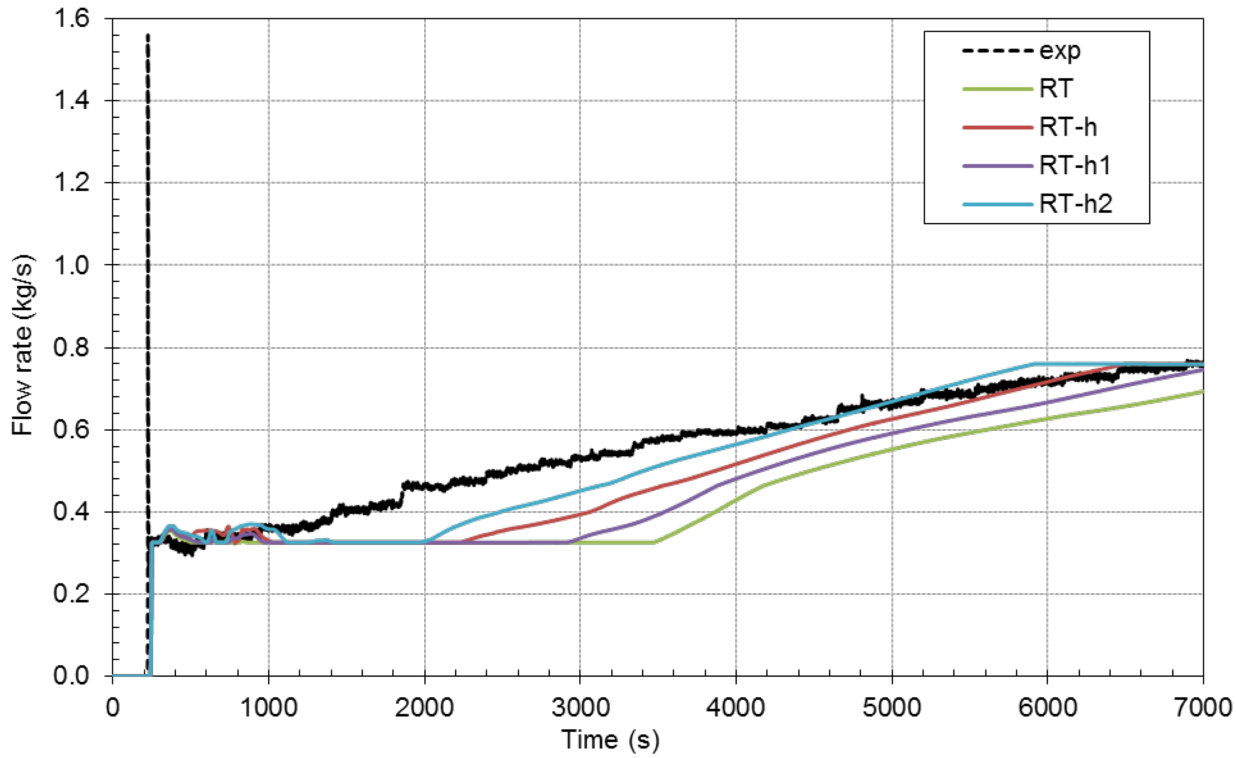

Figure 3-39 HPIS Mass Flow - 'RT' Sensitivity Cases (0 – 7000 s)

3.4 Results Comparison of 'HF' Sensitivity Cases – Long Term Response (0 – 7000 s)

In Figures 3-40 through 3-50 same variables are shown as in Section 3.2 for the time interval 0-7000 s. The dependence on the variation of primary coolant system heat losses to containment is shown. It can be seen in Figure 3-40, the higher are the heat losses, the faster is the pressure drop. The primary side heat losses have negligible effect on the secondary side pressure as expected (see Figure 3-41). In the long term the larger heat losses have small influence on PORV discharge (see Figure 3-46, Figure 3-47 and Figure 3-48), they resulted in decrease of hot (see Figure 3-44) and cold leg temperature (see Figure 3-45), and earlier HPIS flow start (see Figure 3-50) causes more mass to be injected (see Figure 3-49) resulting in PCS inventory increase (see Figure 3-43).

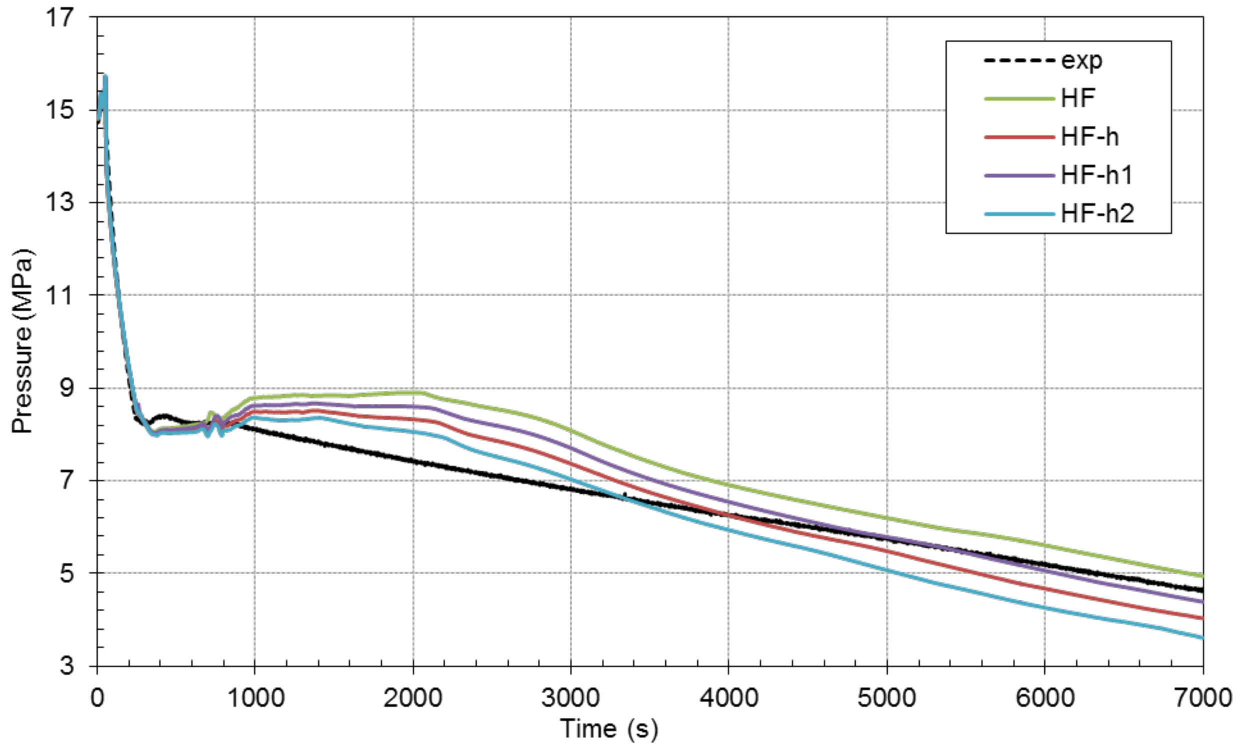

Figure 3-40 Primary Coolant System Pressure - ‘HF’ Sensitivity Cases (0 – 7000 s)

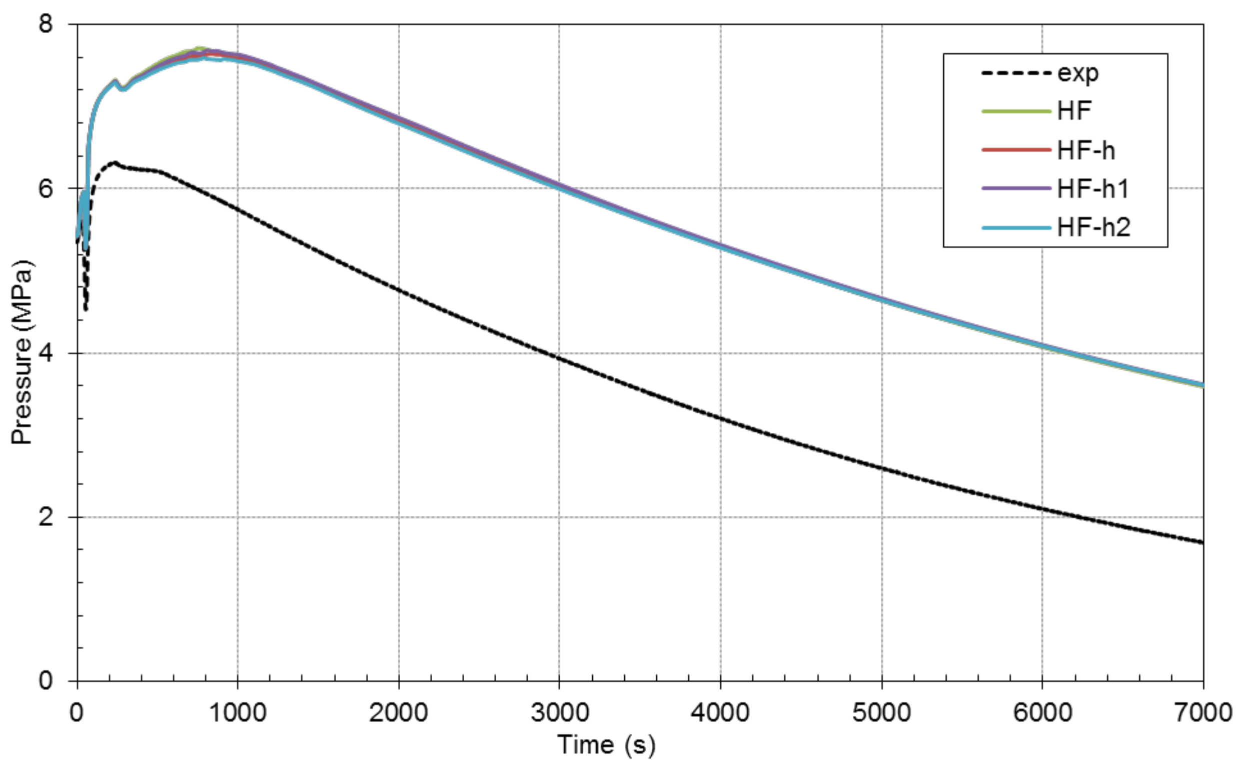

Figure 3-41 Secondary Coolant System Pressure - ‘HF’ Sensitivity Cases (0 – 7000 s)

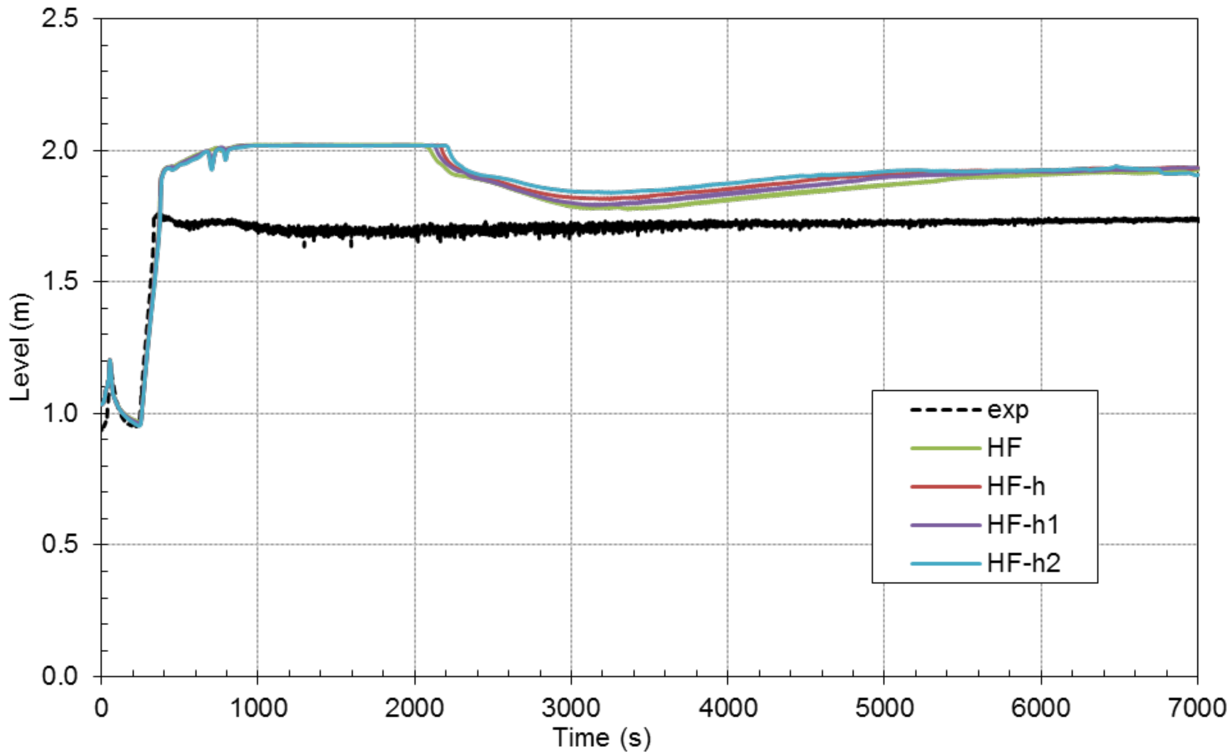

Figure 3-42 Pressurizer Level - 'HF' Sensitivity Cases (0 – 7000 s)

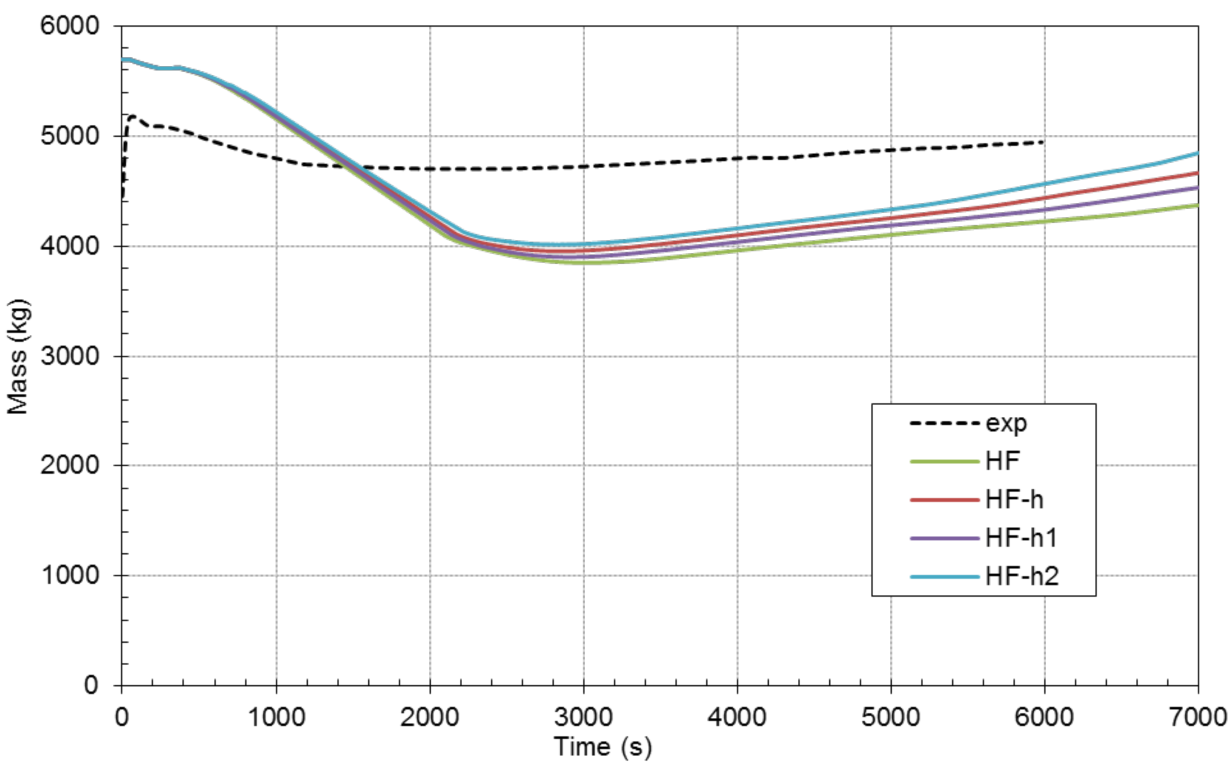

Figure 3-43 Primary Coolant System Inventory - 'HF' Sensitivity Cases (0 – 7000 s)

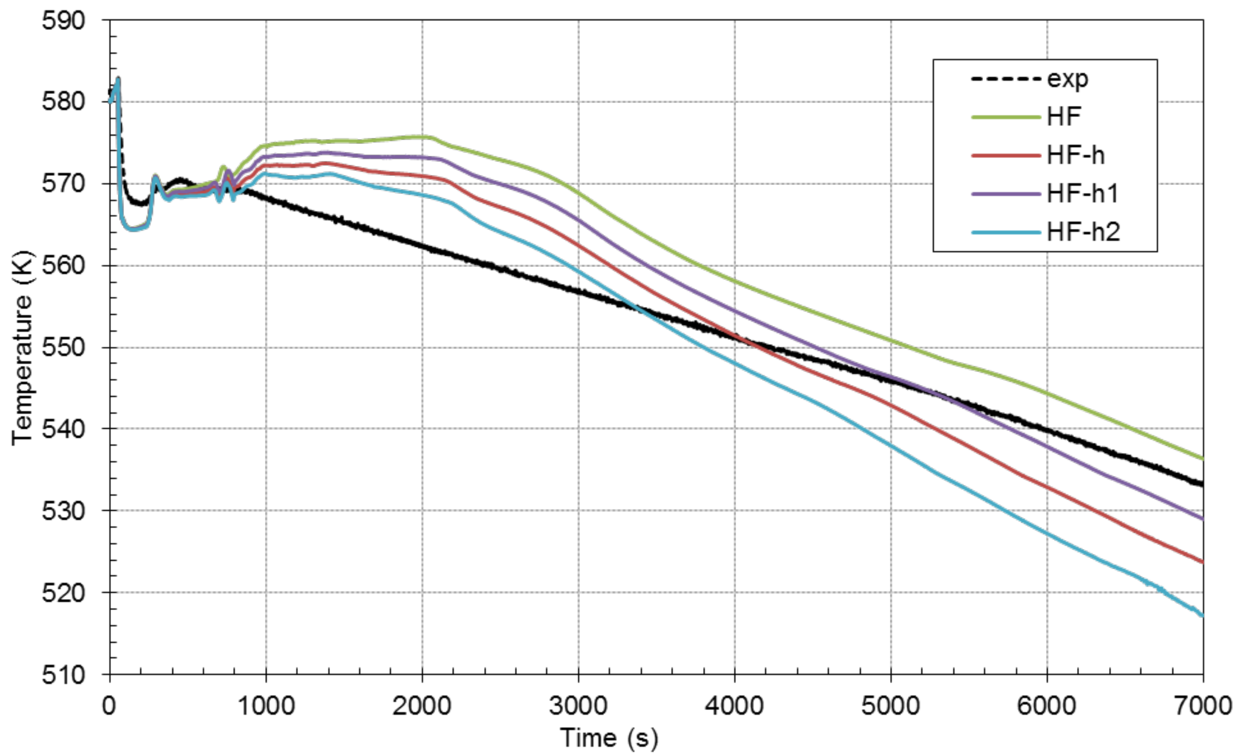

Figure 3-44 Hot Leg Temperature - 'HF' Sensitivity Cases (0 – 7000 s)

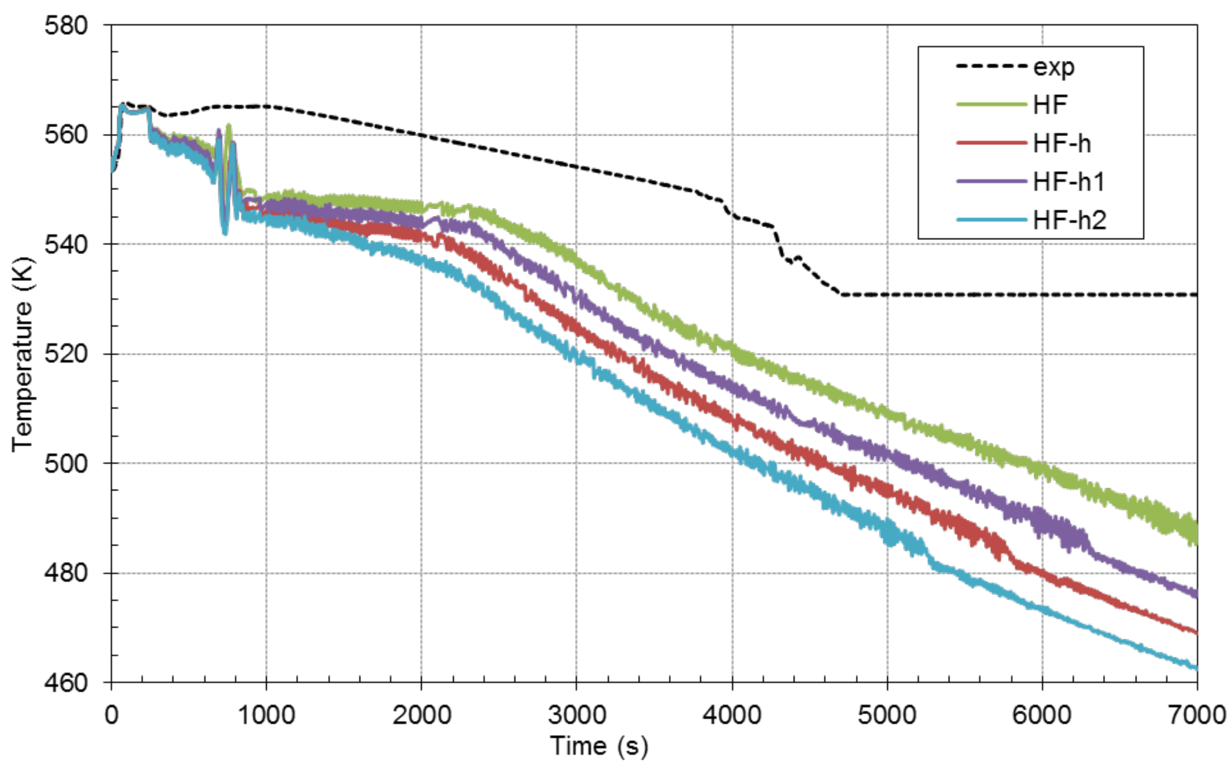

Figure 3-45 Cold Leg Temperature - 'HF' Sensitivity Cases (0 – 7000 s)

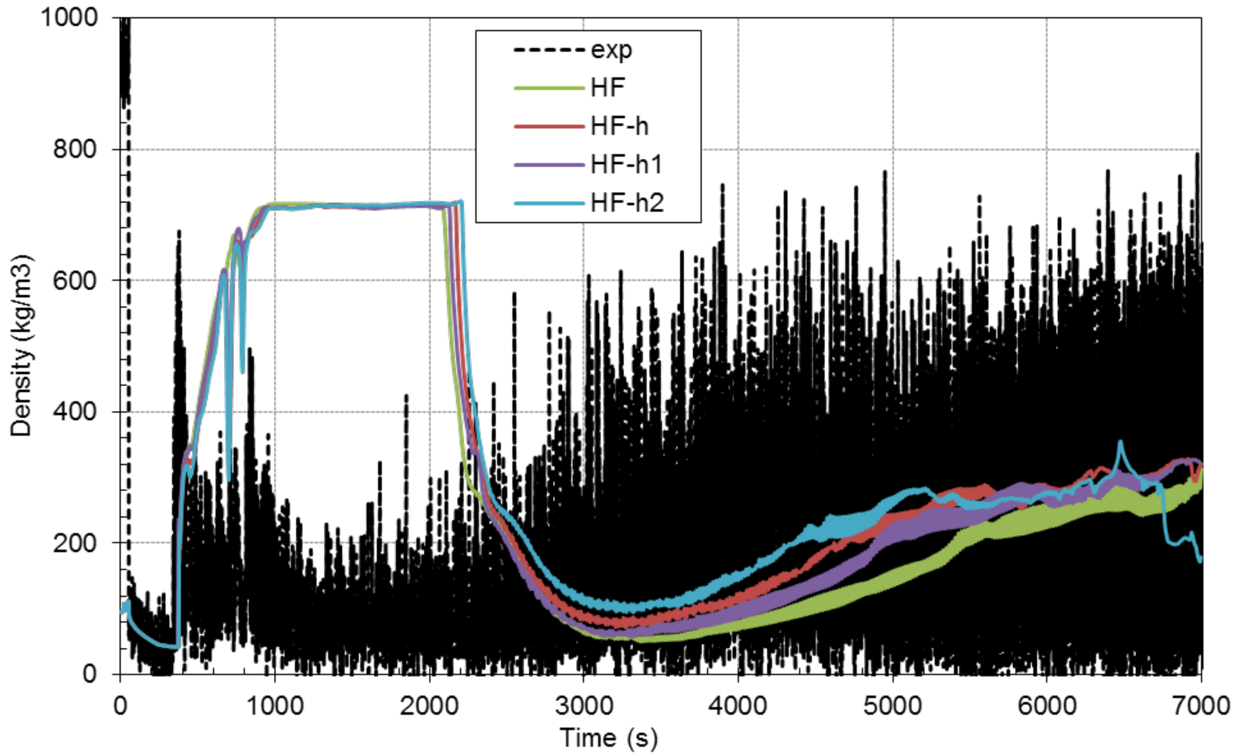

Figure 3-46 PORV Flow Density - 'HF' Sensitivity Cases (0 – 7000 s)

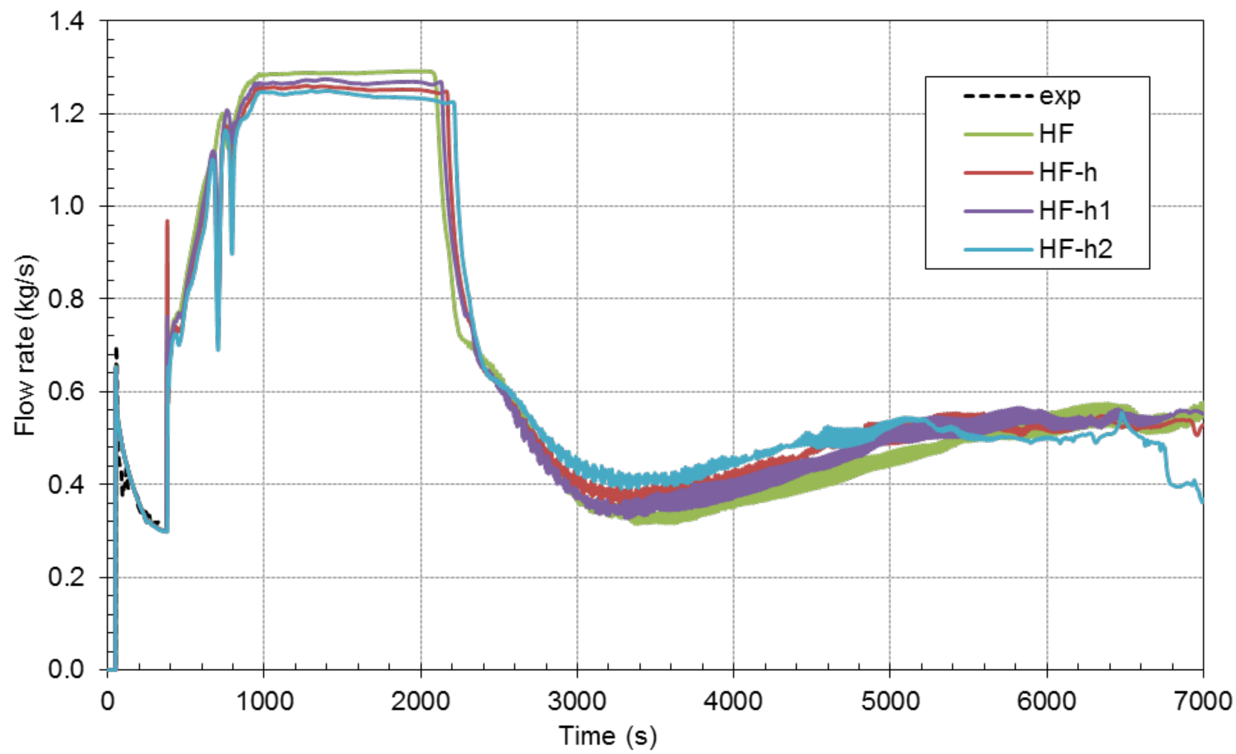

Figure 3-47 Pressurizer PORV Mass Flow - 'HF' Sensitivity Cases (0 – 7000 s)

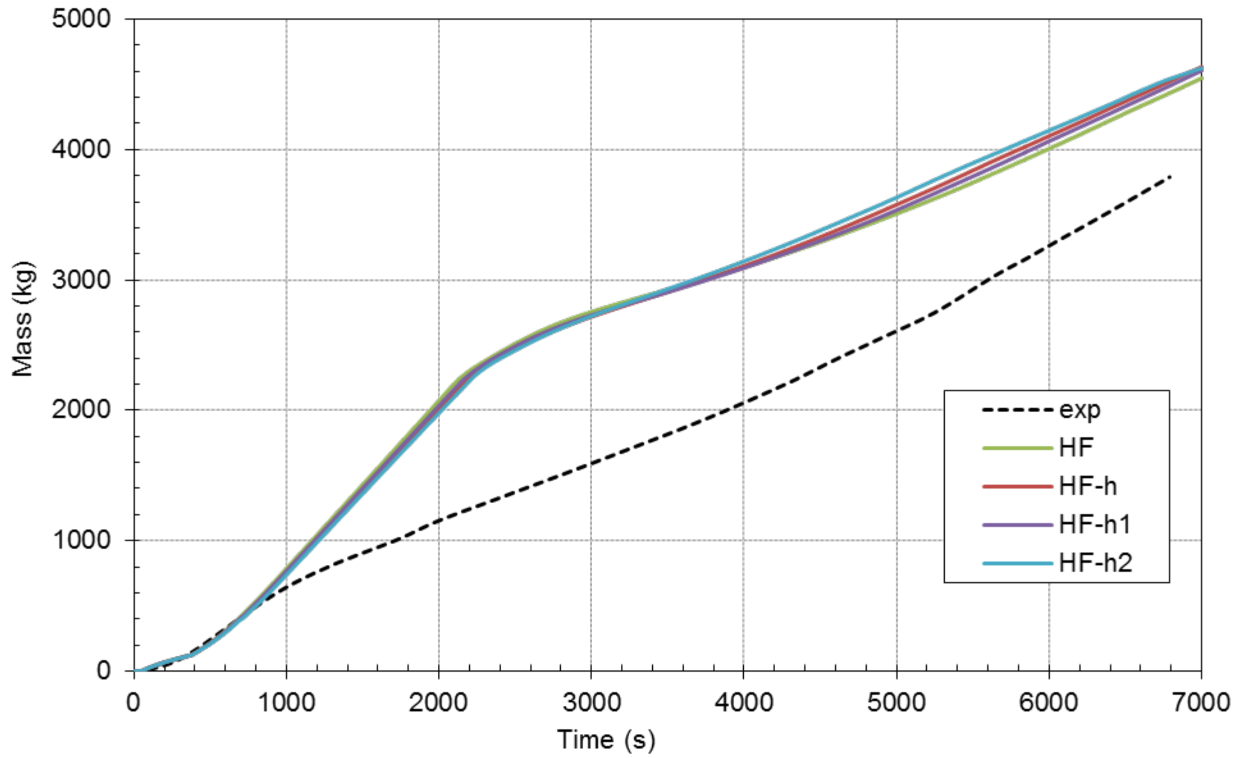

Figure 3-48 Pressurizer PORV Integrated Mass - ‘HF’ Sensitivity Cases (0 – 7000 s)

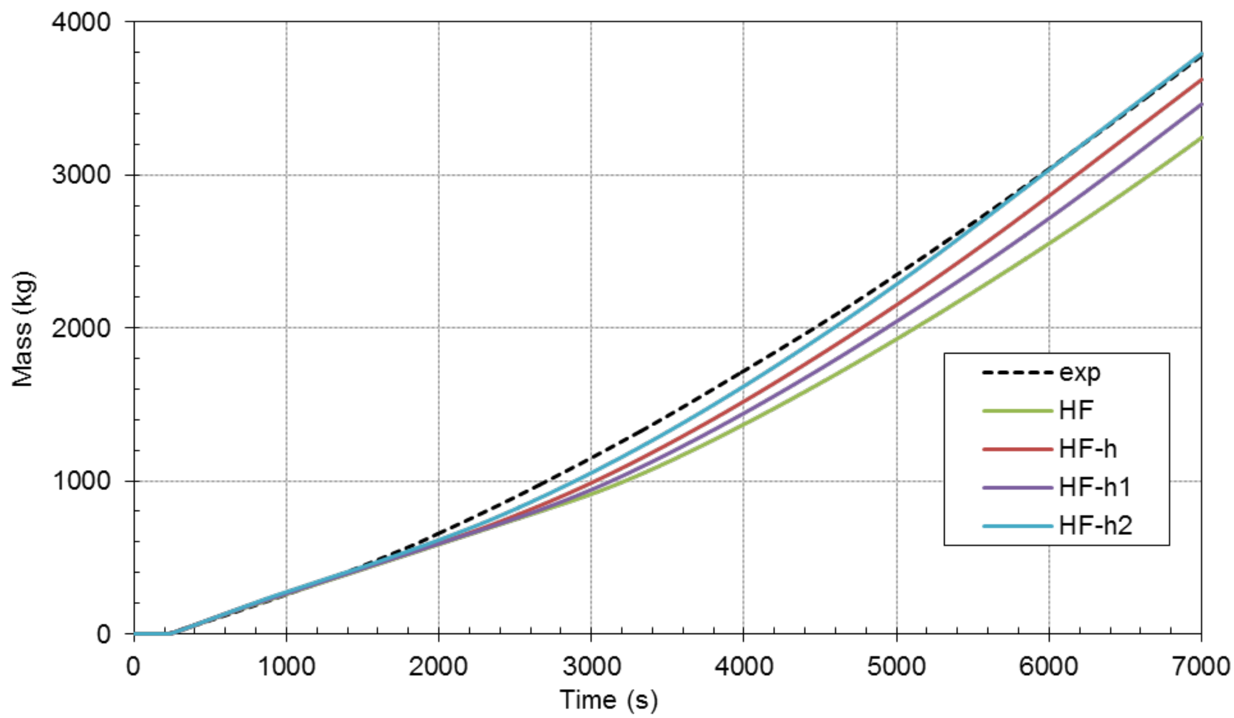

Figure 3-49 HPIS Integrated Mass - ‘HF’ Sensitivity Cases (0 – 7000 s)

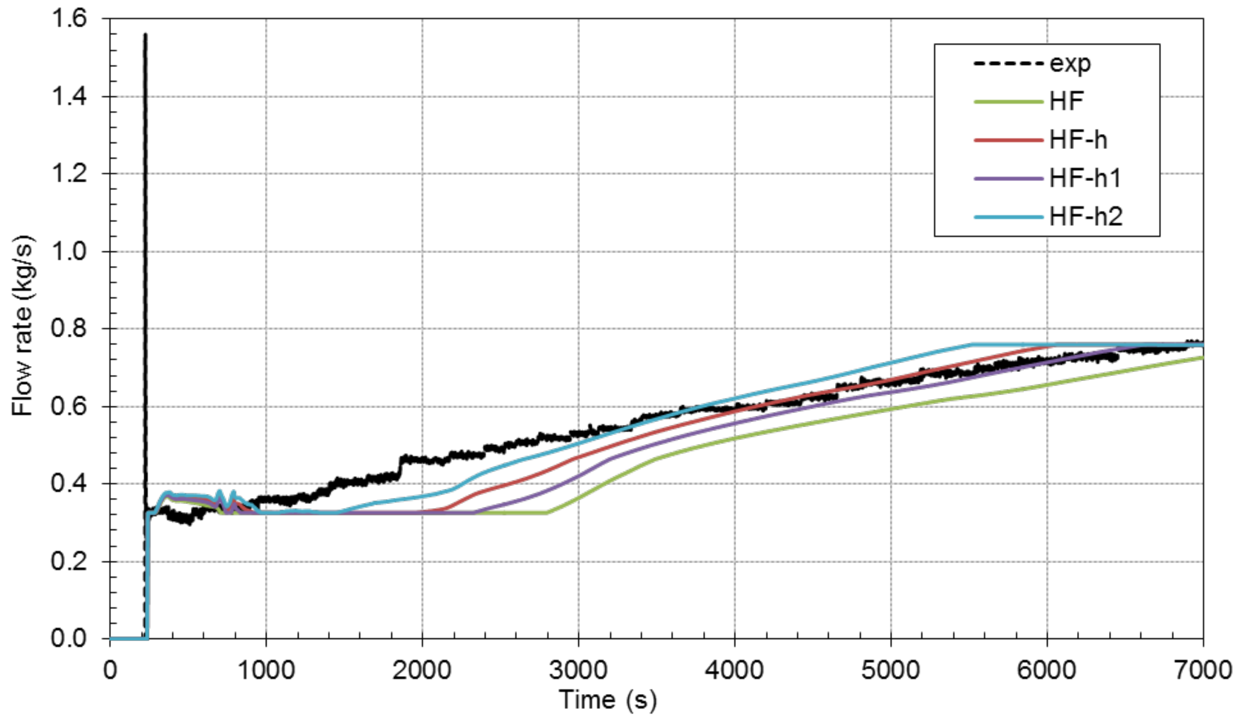

Figure 3-50 HPIS Mass Flow - ‘HF’ Sensitivity Cases (0 – 7000 s)

3.5 Results Comparison of ‘HF-off’ Sensitivity Cases – Long Term Response (0 – 7000 s)

Figures 3-51 through 3-61 show same variables as are in Section 3.2 for the time interval 0-7000 s. The dependence on the variation of primary coolant system heat losses to containment is shown. It can be seen in Figure 3-51, the higher are the heat losses, the faster is pressure drop. The primary side heat losses have negligible effect on the secondary side pressure as expected (see Figure 3-52). In the long term the larger heat losses have small influence on PORV discharge (see Figure 3-57, Figure 3-58 and Figure 3-59), they resulted in decrease of hot (see Figure 3-55) and cold leg temperature (see Figure 3-56), and earlier HPIS flow start (see Figure 3-61) causes more mass to be injected (see Figure 3-60) resulting in PCS inventory increase (see Figure 3-54).

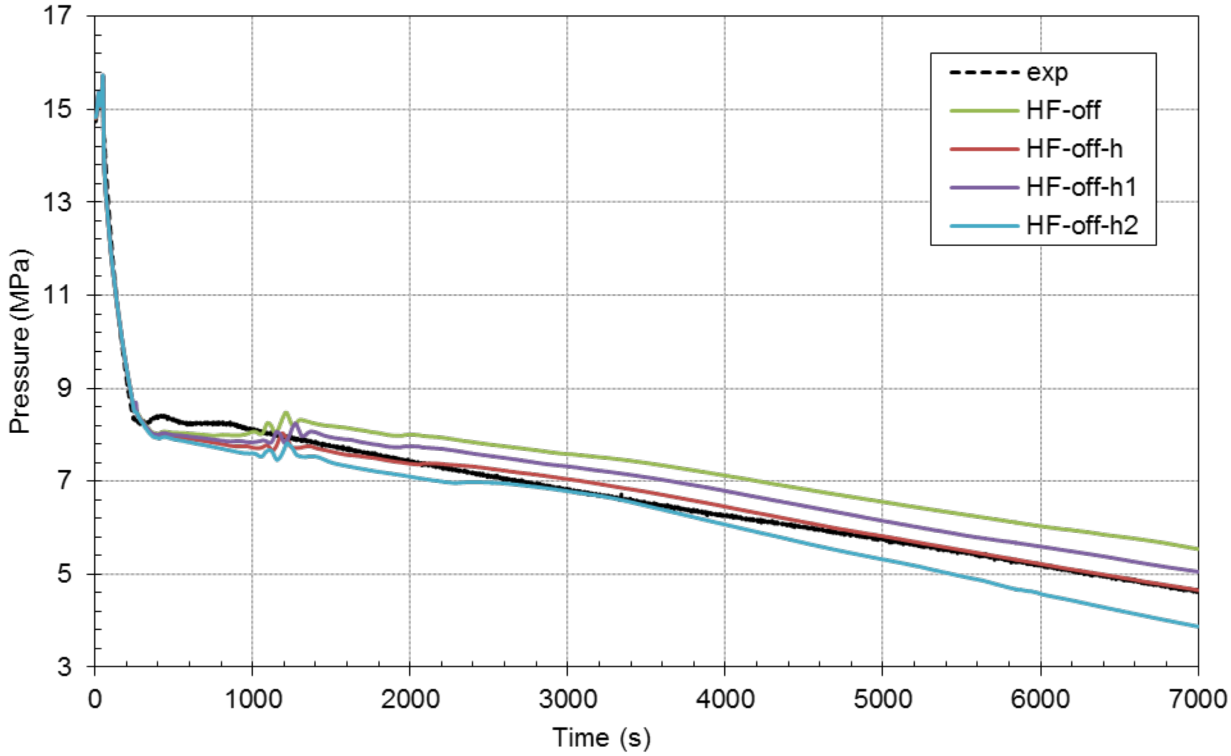

Figure 3-51 Primary Coolant System Pressure - 'HF-off' Sensitivity Cases (0 – 7000 s)

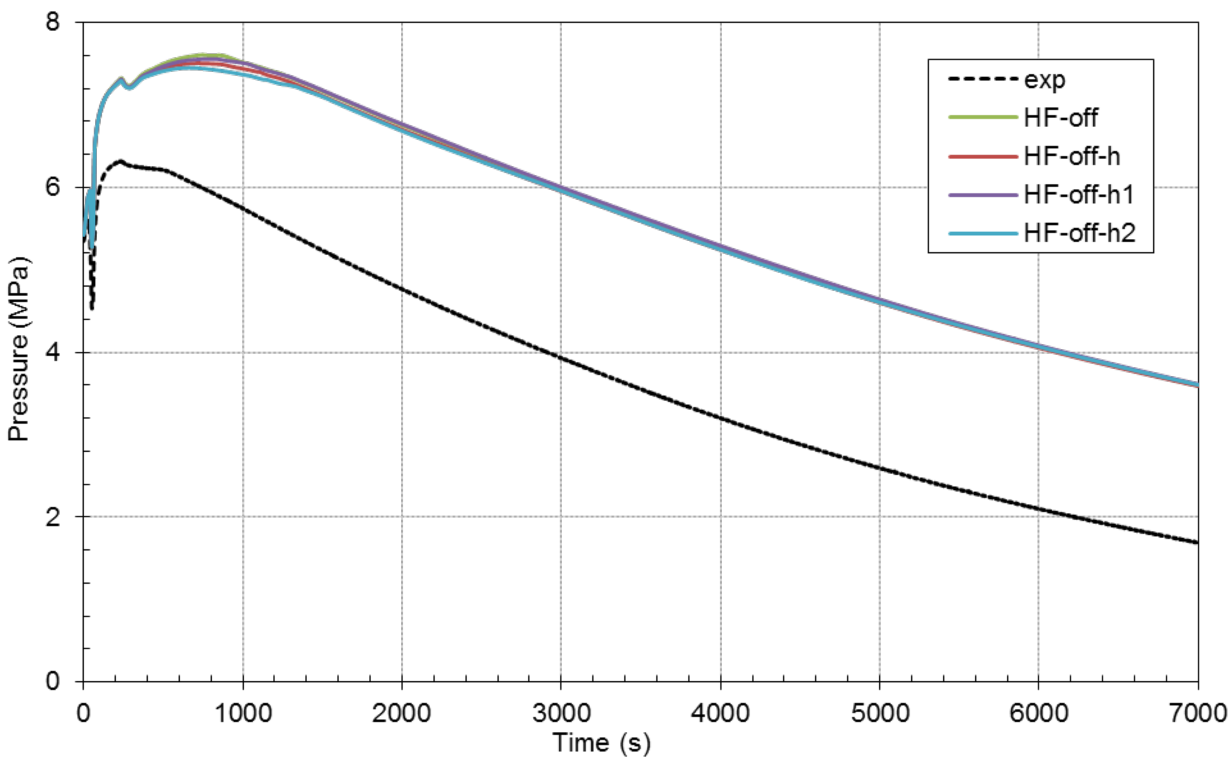

Figure 3-52 Secondary Coolant System Pressure - 'HF-off' Sensitivity Cases (0 – 7000 s)

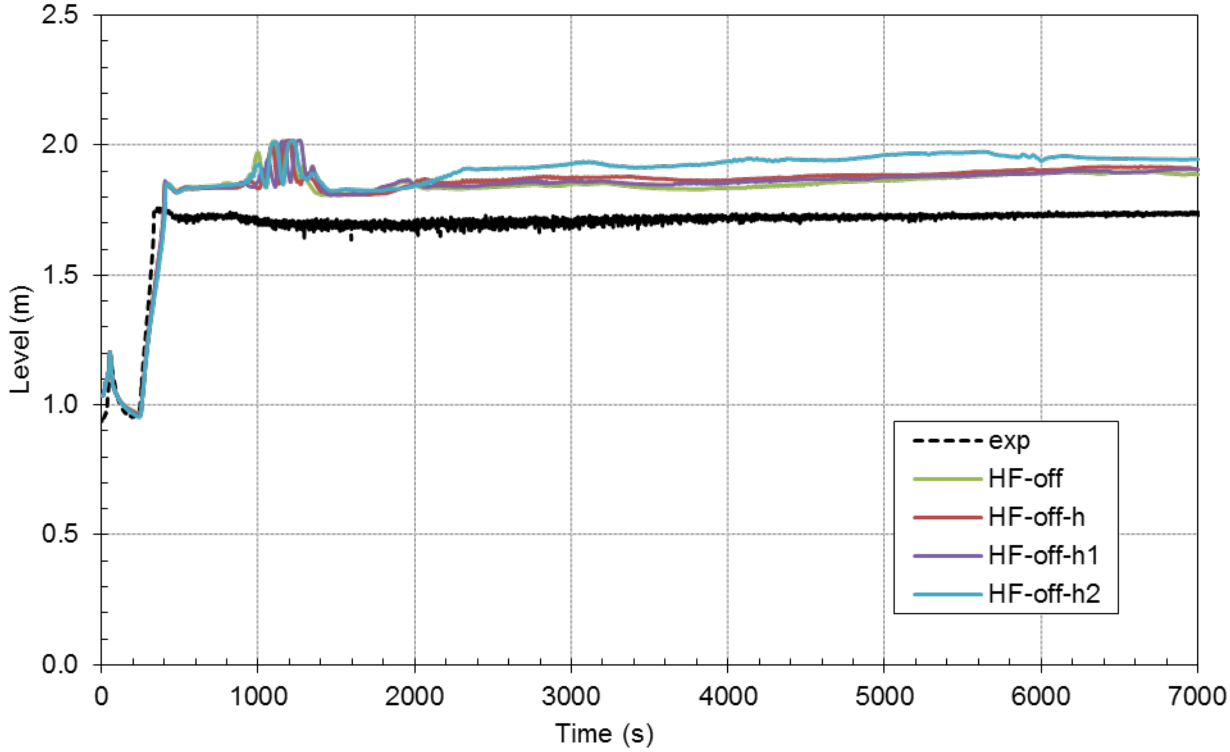

Figure 3-53 Pressurizer Level - ‘HF-off’ Sensitivity Cases (0 – 7000 s)

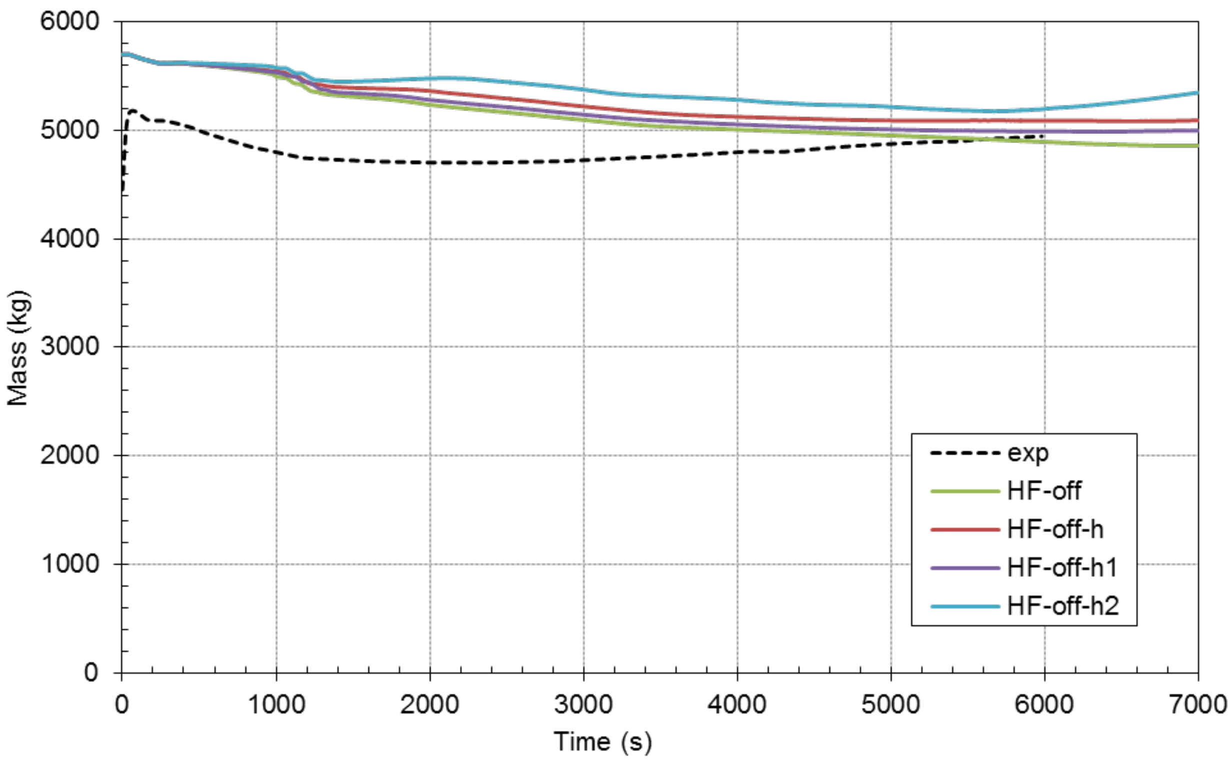

Figure 3-54 Primary Coolant System Inventory - ‘HF-off’ Sensitivity Cases (0 – 7000 s)

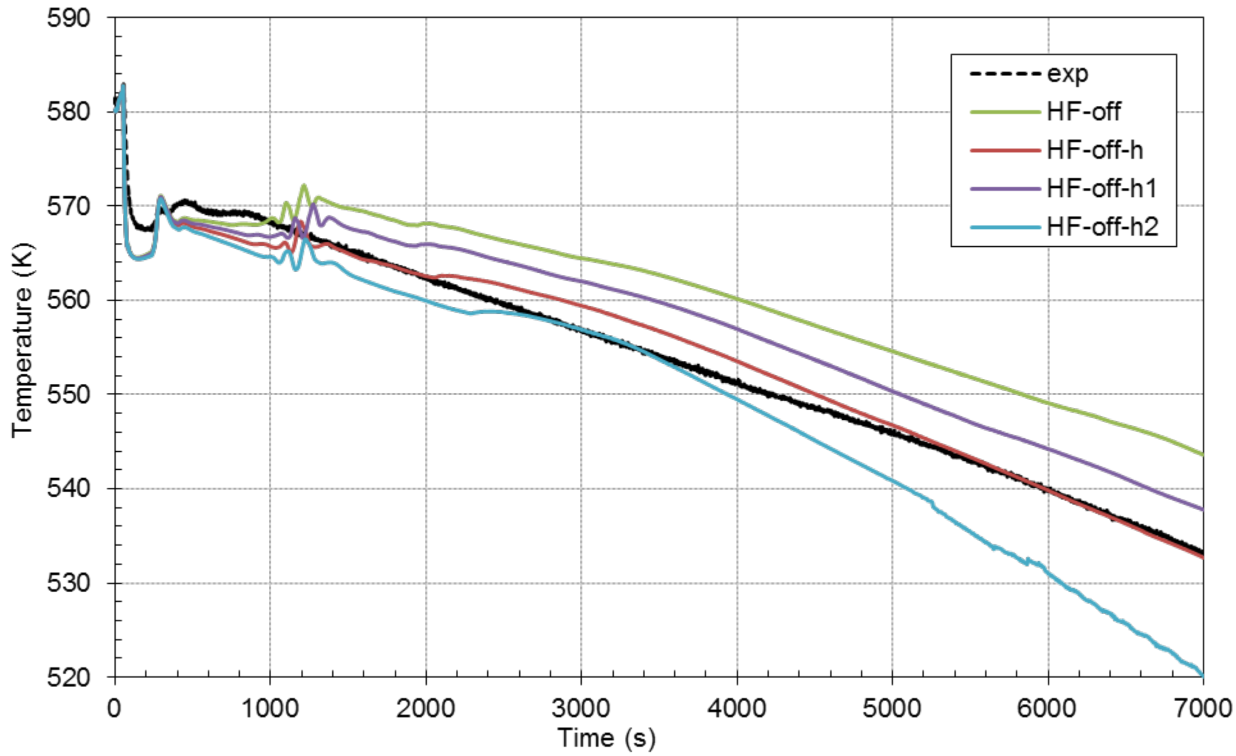

Figure 3-55 Hot Leg Temperature - 'HF-off' Sensitivity Cases (0 – 7000 s)

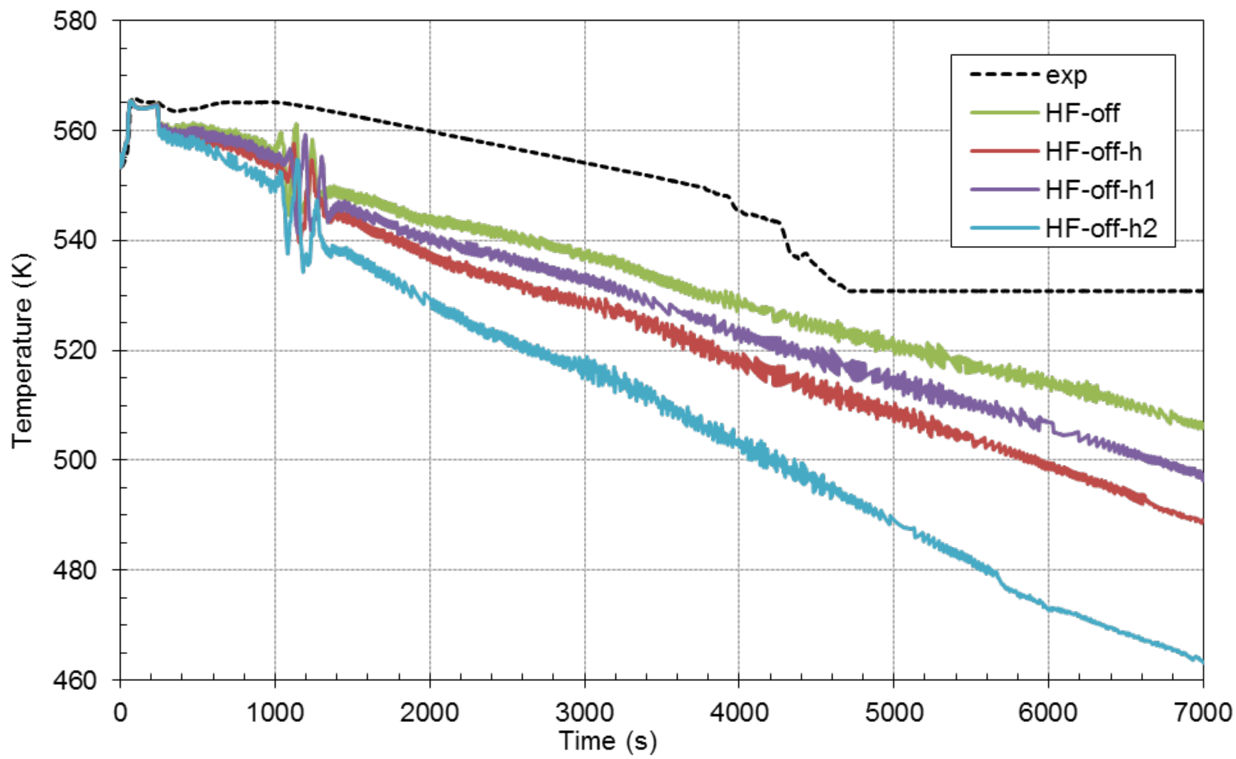

Figure 3-56 Cold Leg Temperature - 'HF-off' Sensitivity Cases (0 – 7000 s)

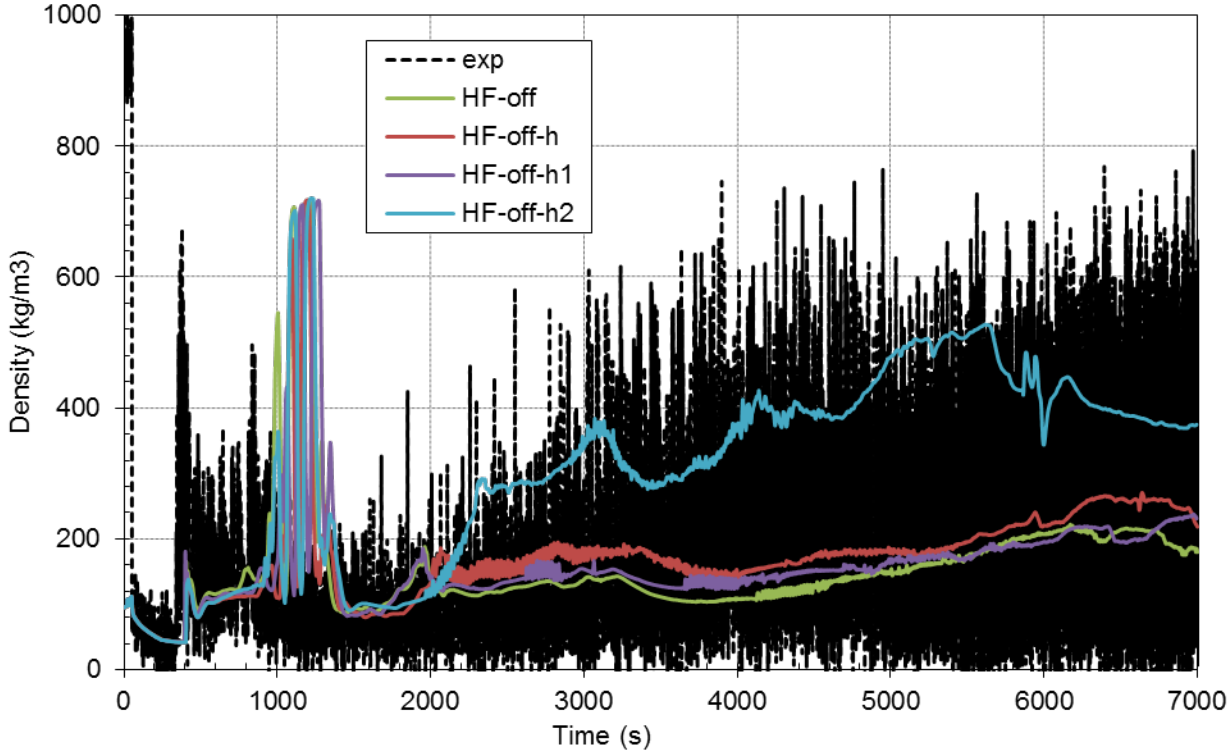

Figure 3-57 PORV Flow Density - ‘HF-off’ Sensitivity Cases (0 – 7000 s)

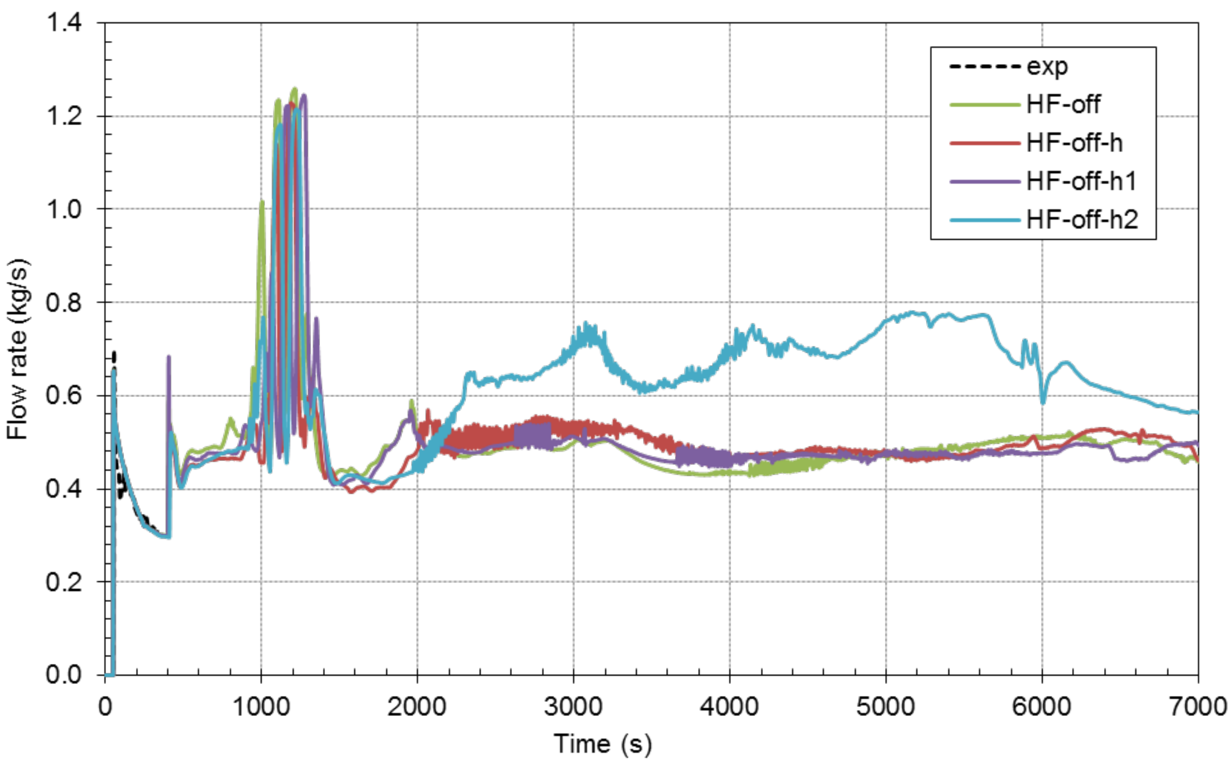

Figure 3-58 Pressurizer PORV Mass Flow - ‘HF-off’ Sensitivity Cases (0 – 7000 s)

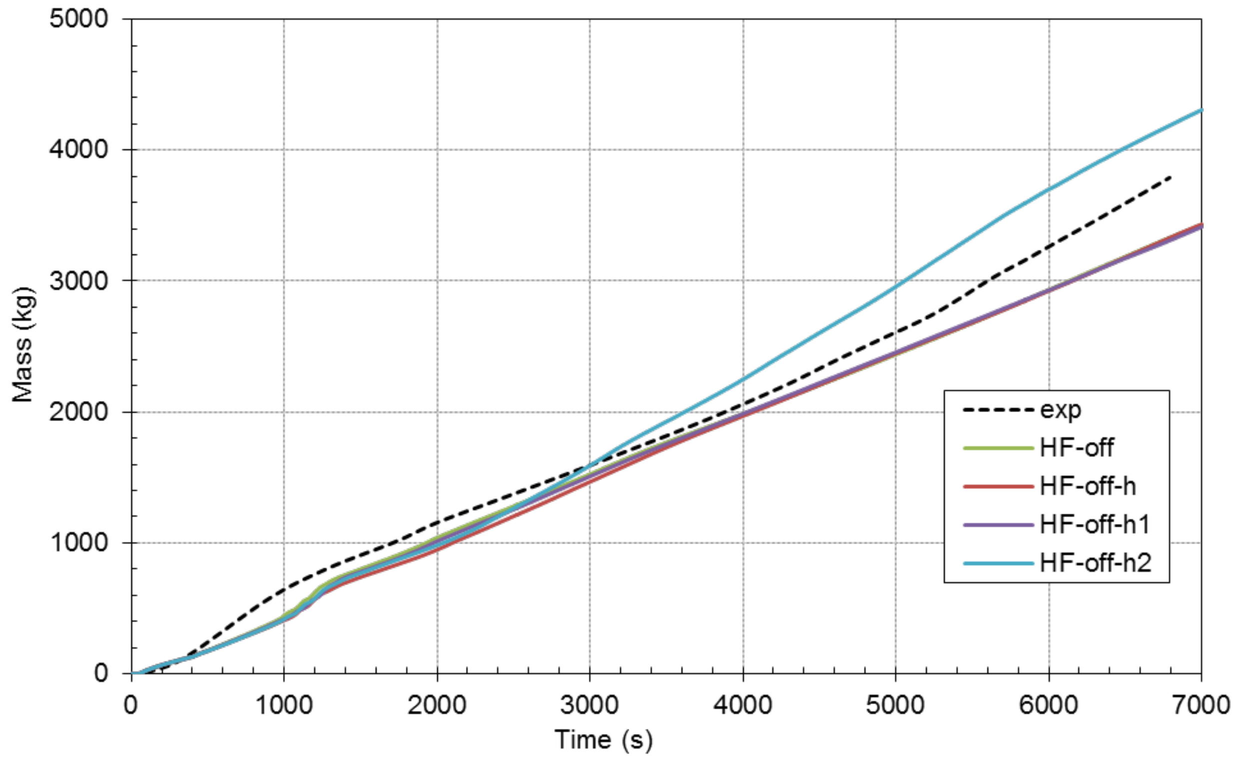

Figure 3-59 Pressurizer PORV Integrated Mass - ‘HF-off’ Sensitivity Cases (0 – 7000 s)

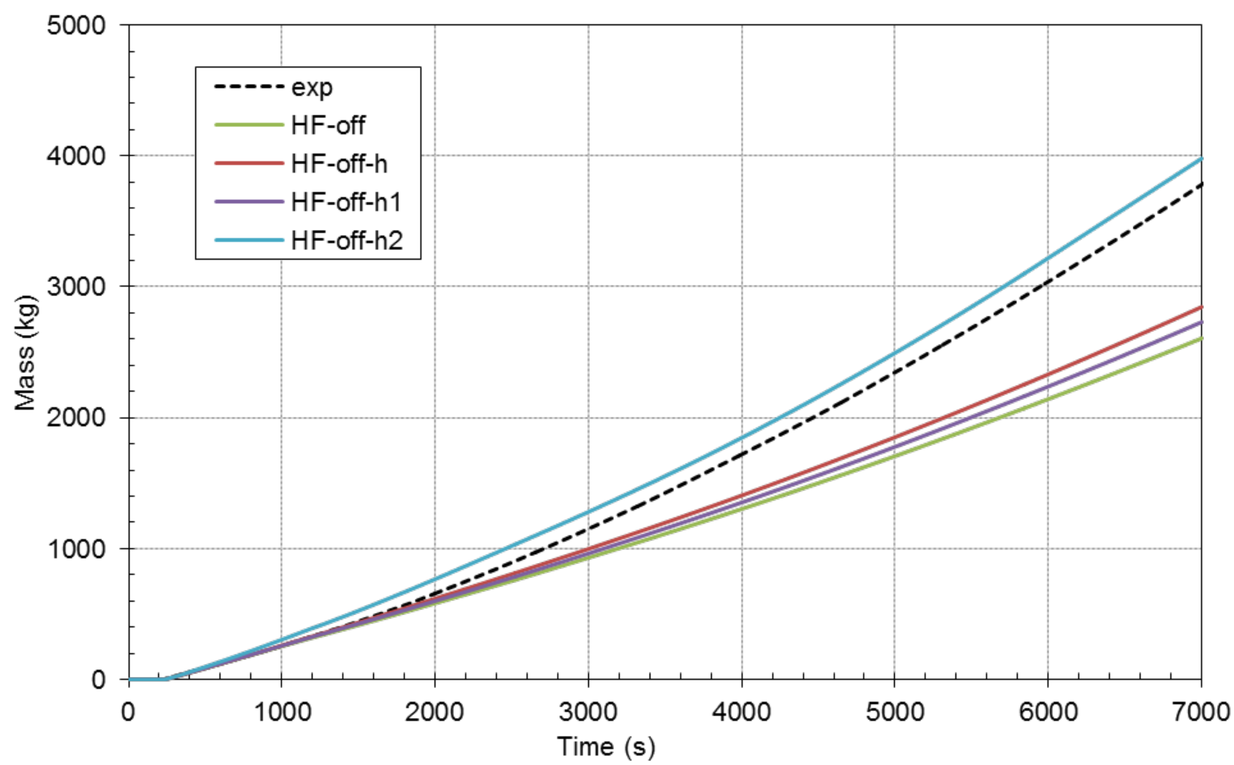

Figure 3-60 HPIS Integrated Mass - ‘HF-off’ Sensitivity Cases (0 – 7000 s)

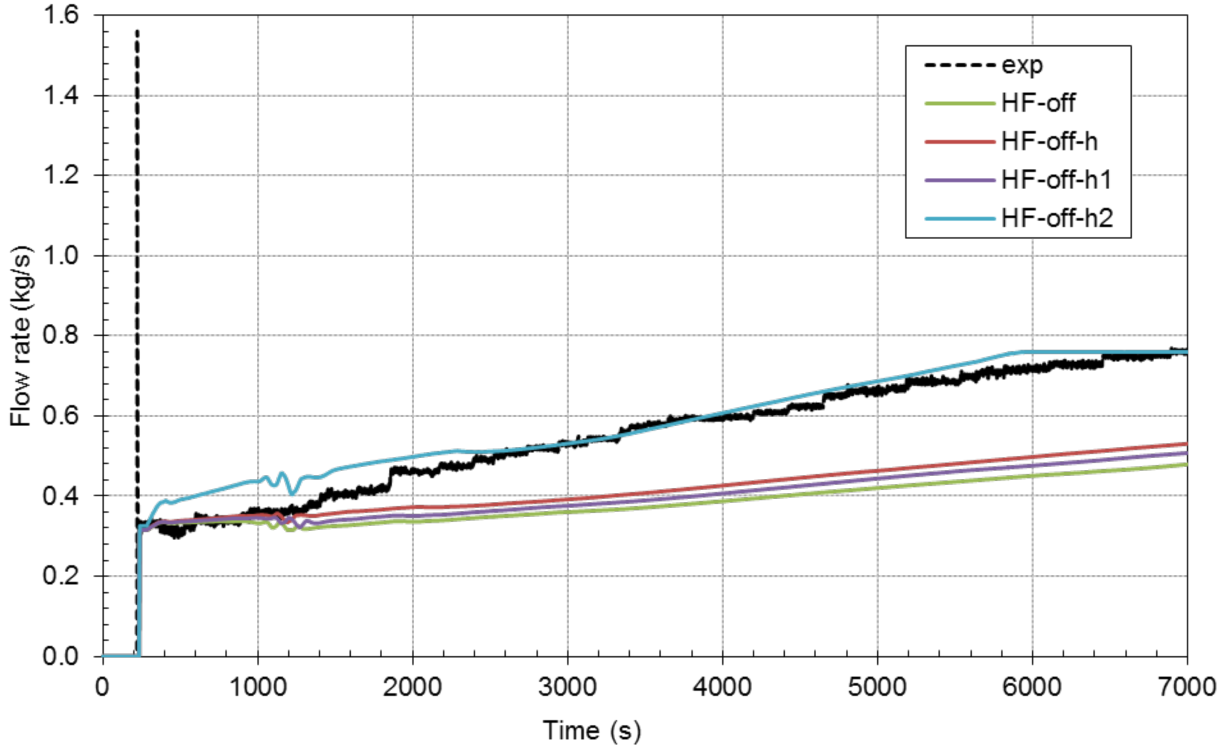

Figure 3-61 HPIS Mass Flow - 'HF-off' Sensitivity Cases (0 – 7000 s)

3.6 Results Comparison of Best Adjusted Cases ('RT-h', 'HF-h', 'HF-off-h') – Long Term Response (0 – 7000 s)

Figures 3-62 through 3-72 shown are the same variables as shown in Section 3.2 for the time interval 0-7000 s. The primary coolant system increased heat losses to containment caused the faster pressure drop (see Figure 3-62) and closer agreement with the measured data. The comparison between 'RT' and 'RT-h' variables, 'HF' and 'HF-h' and 'HF-off' and 'HF-off-h' is done in Sections 3.3, 3.4 and 3.5, respectively.

The 'HF-off-h' calculation the best agree for primary pressure (see Figure 3-62), secondary pressure (see Figure 3-63), pressurizer level (see Figure 3-64), primary coolant inventory (see Figure 3-65), hot leg temperature (see Figure 3-66), cold leg temperature (see Figure 3-67), PORV flow density (see Figure 3-68), pressurizer PORV mass flow (see Figure 3-69) and pressurizer PORV integrated mass (see Figure 3-70). Only for HPIS flow (see Figure 3-72) and HPIS integrated mass (see Figure 3-71) the 'HF-h' case is better due to larger primary pressure decrease.

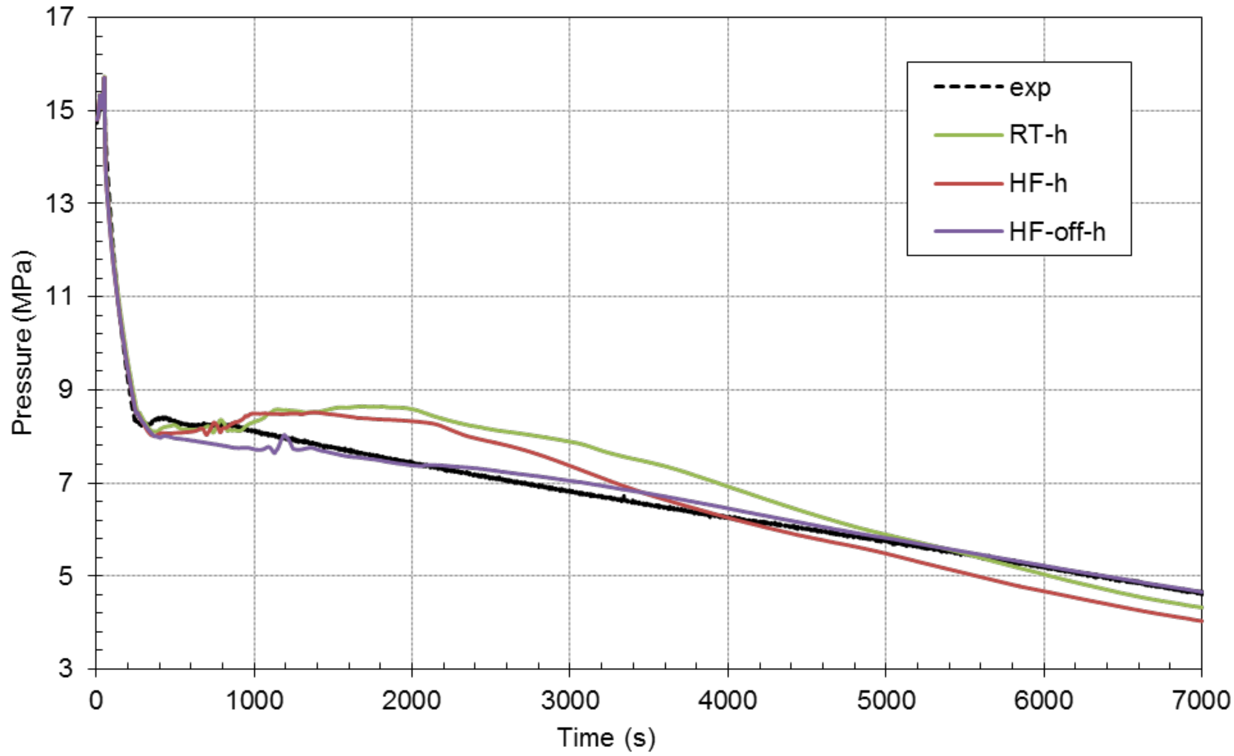

Figure 3-62 Primary Coolant System Pressure - Best Adjusted Cases (0 – 7000 s)

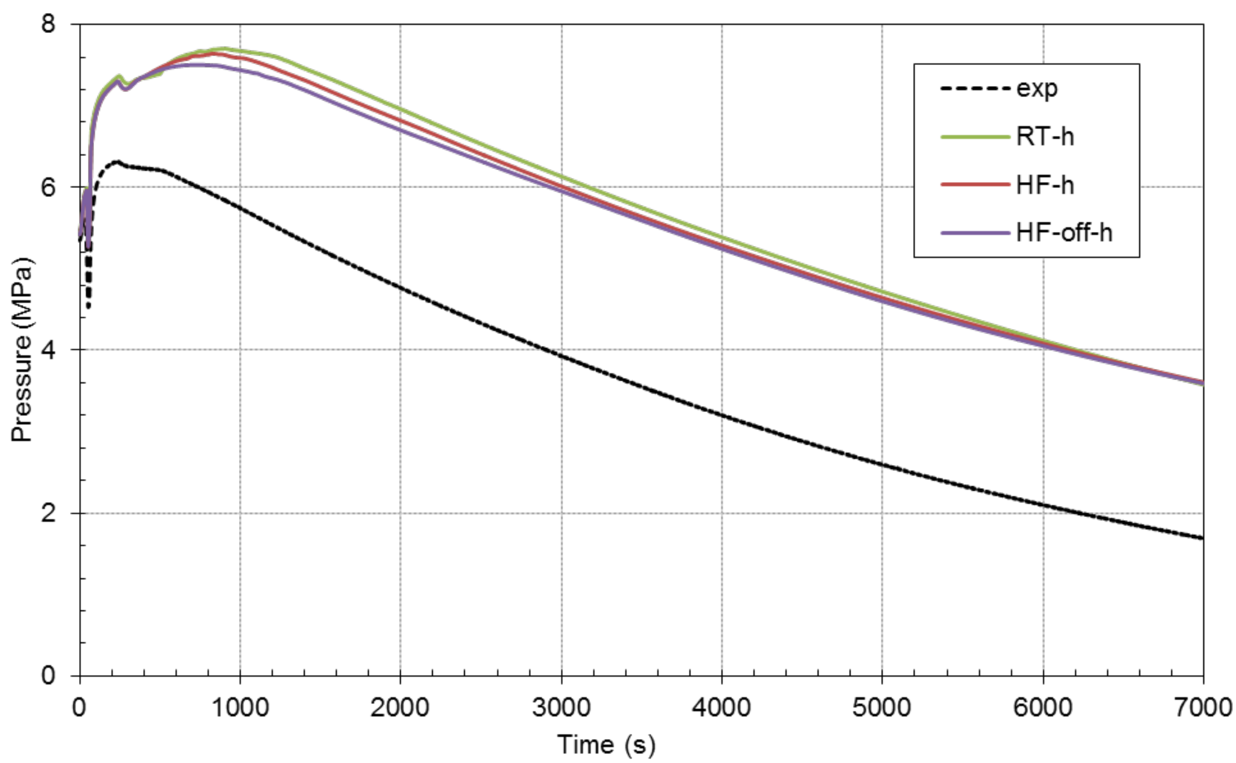

Figure 3-63 Secondary Coolant System Pressure - Best Adjusted Cases (0 – 7000 s)

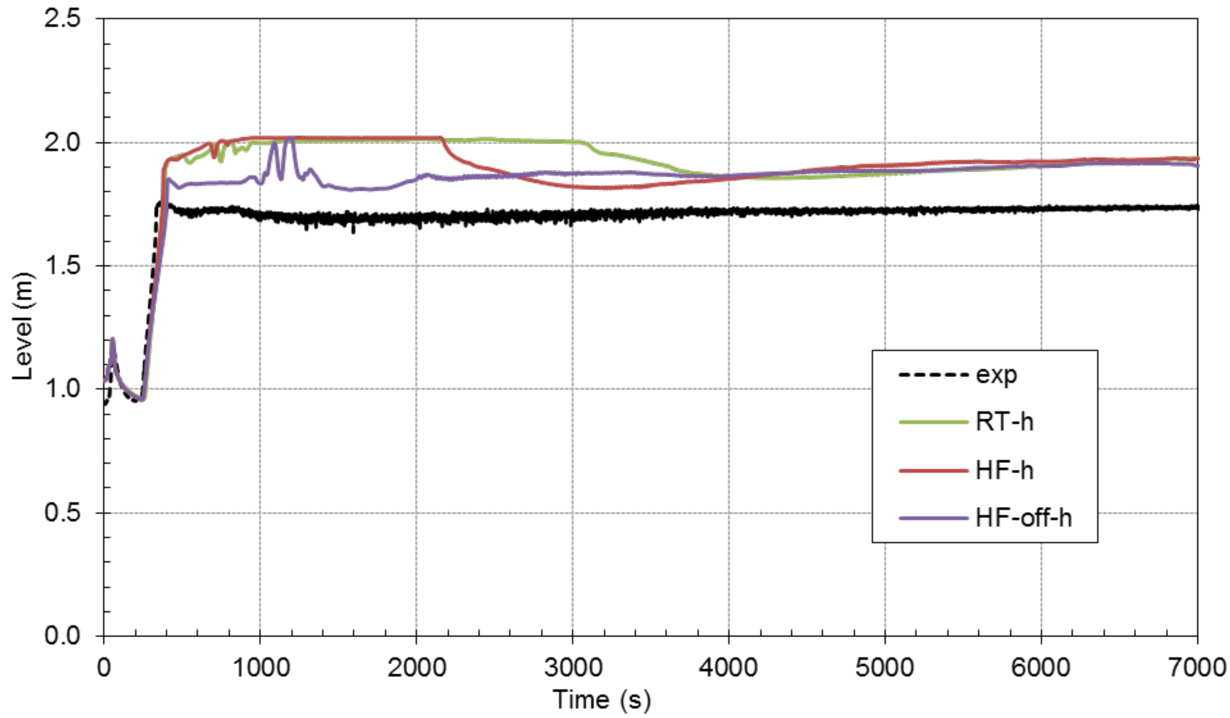

Figure 3-64 Pressurizer Level - Best Adjusted Cases (0 – 7000 s)

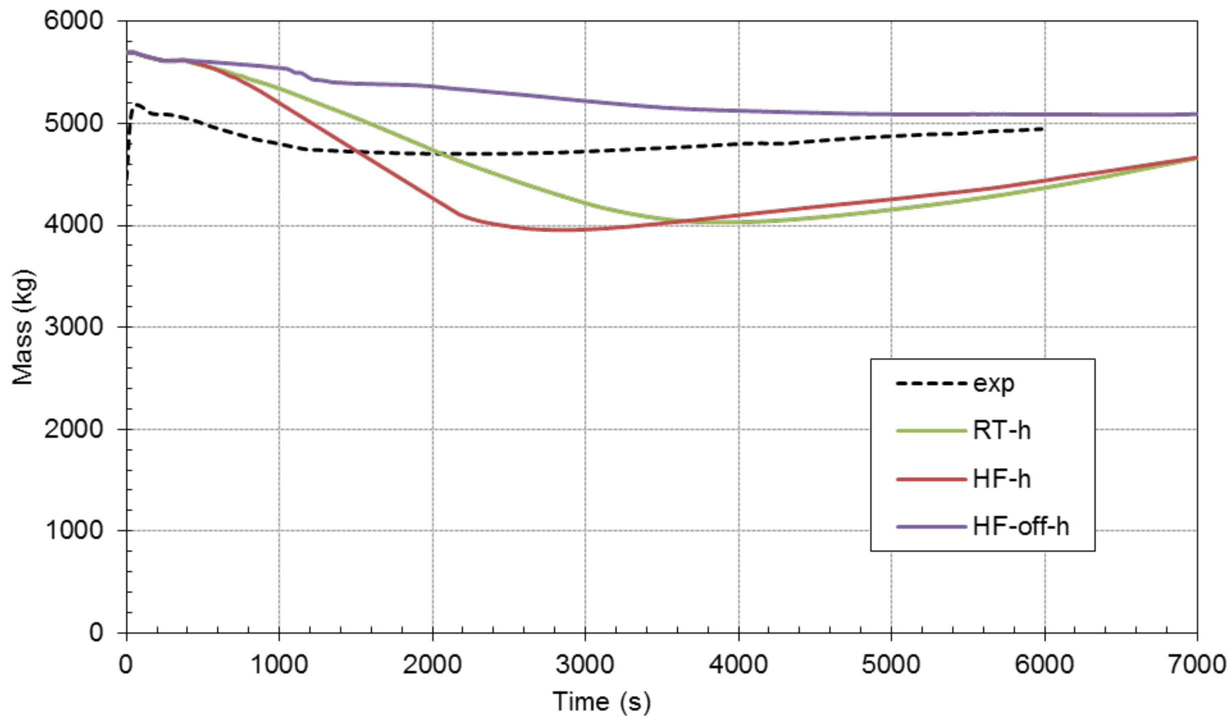

Figure 3-65 Primary Coolant System Inventory - Best Adjusted Cases (0 – 7000 s)

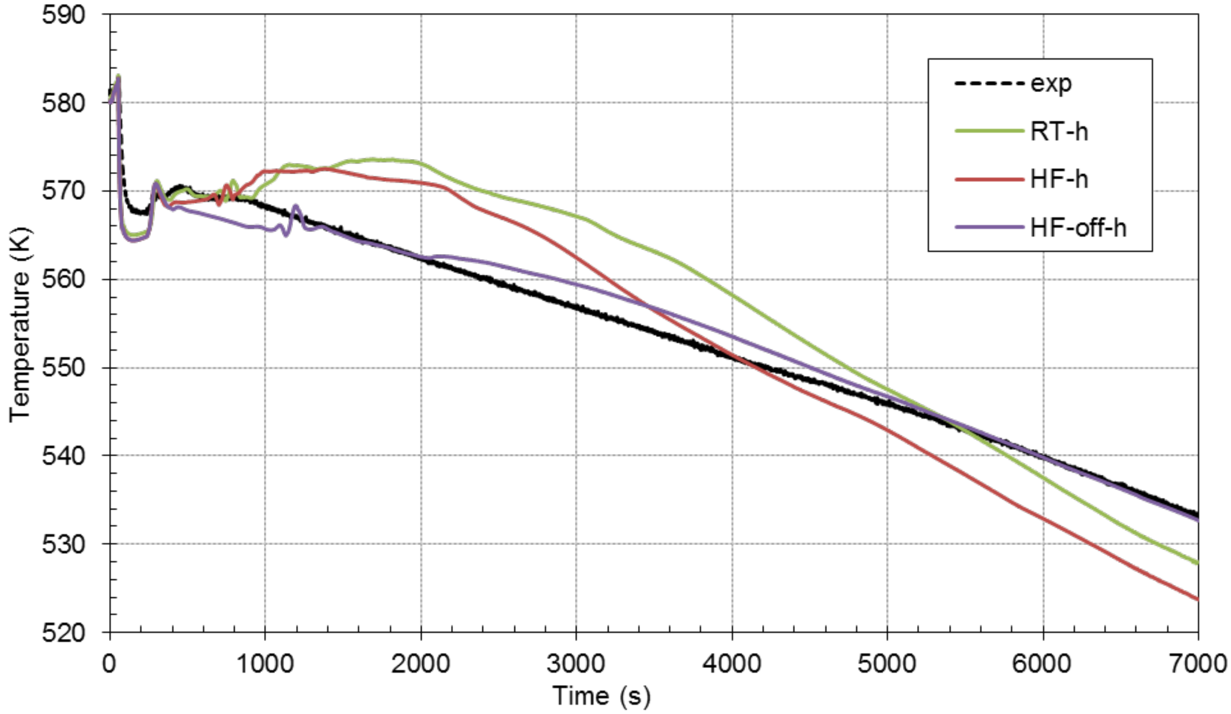

Figure 3-66 Hot Leg Temperature - Best Adjusted Cases (0 – 7000 s)

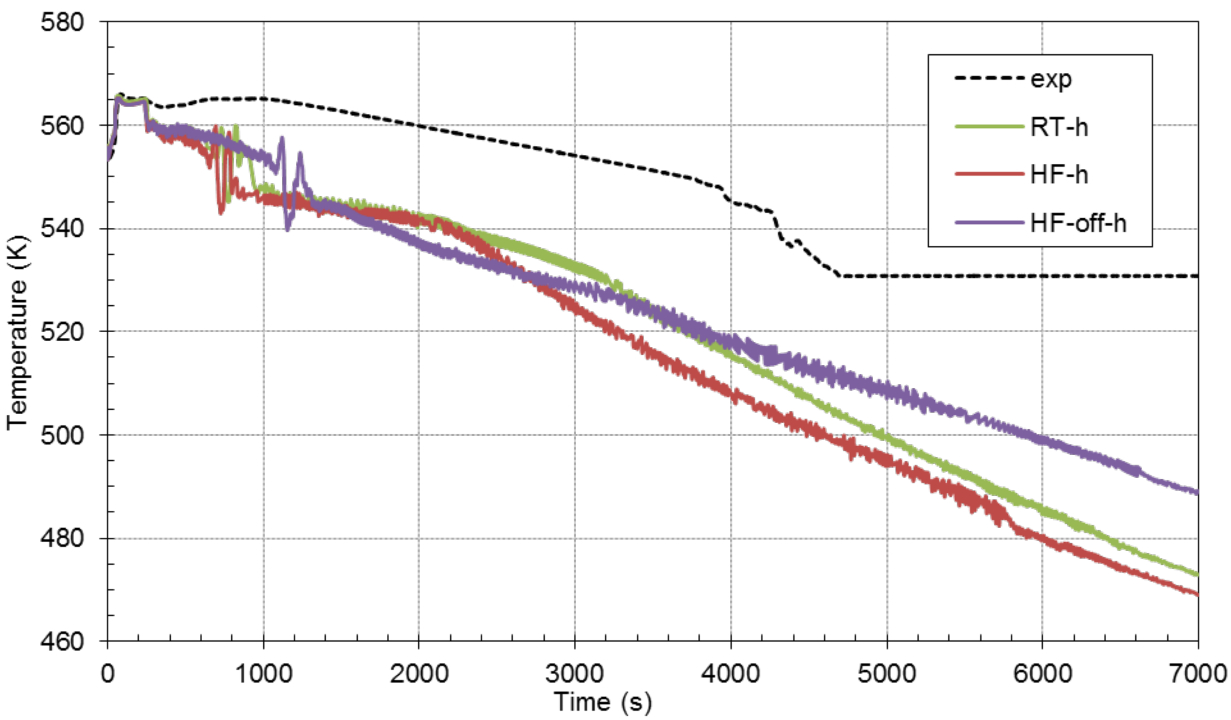

Figure 3-67 Cold Leg Temperature - Best Adjusted Cases (0 – 7000 s)

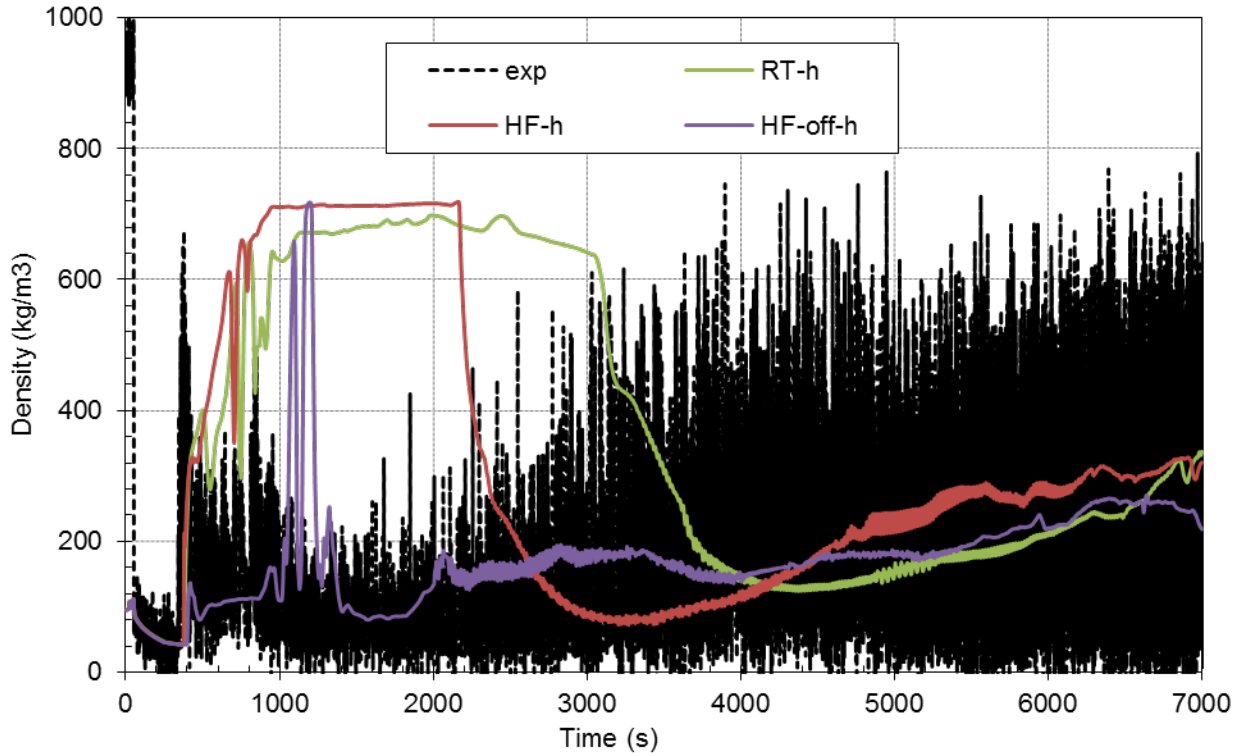

Figure 3-68 PORV Flow Density - Best Adjusted Cases (0 – 7000 s)

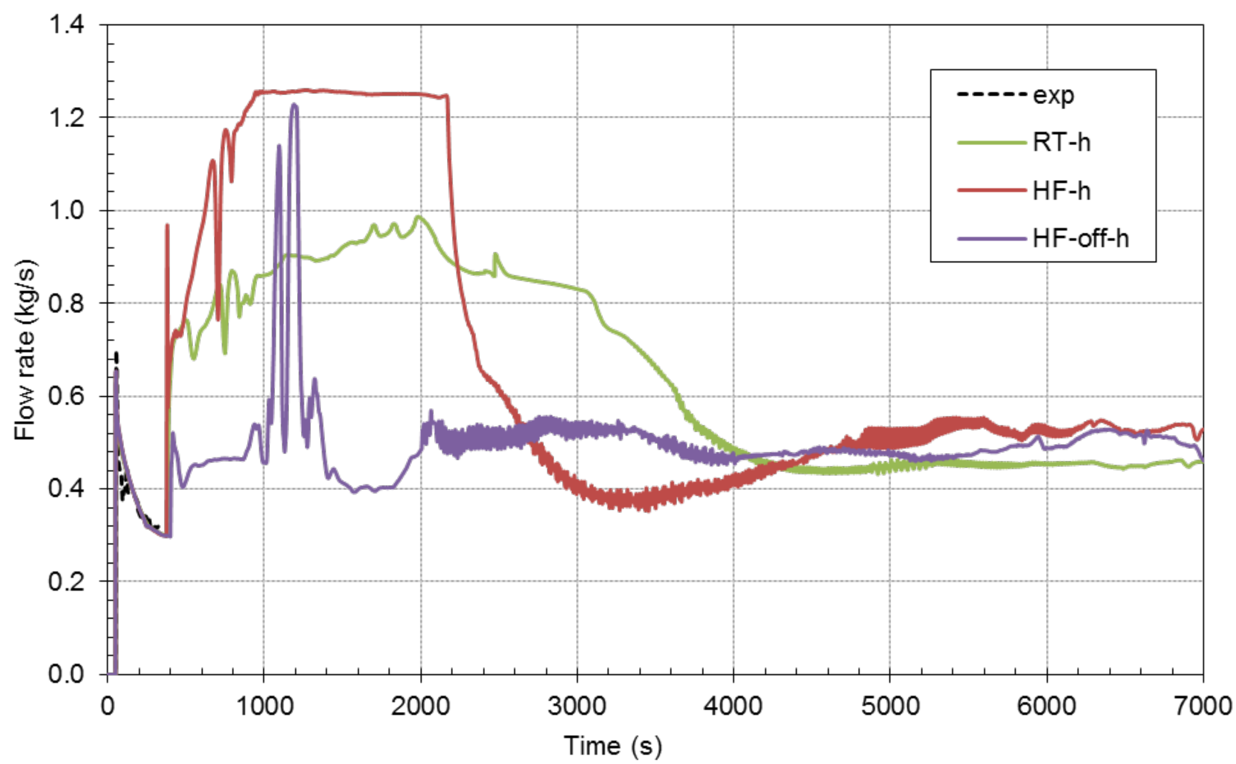

Figure 3-69 Pressurizer PORV Mass Flow - Best Adjusted Cases (0 – 7000 s)

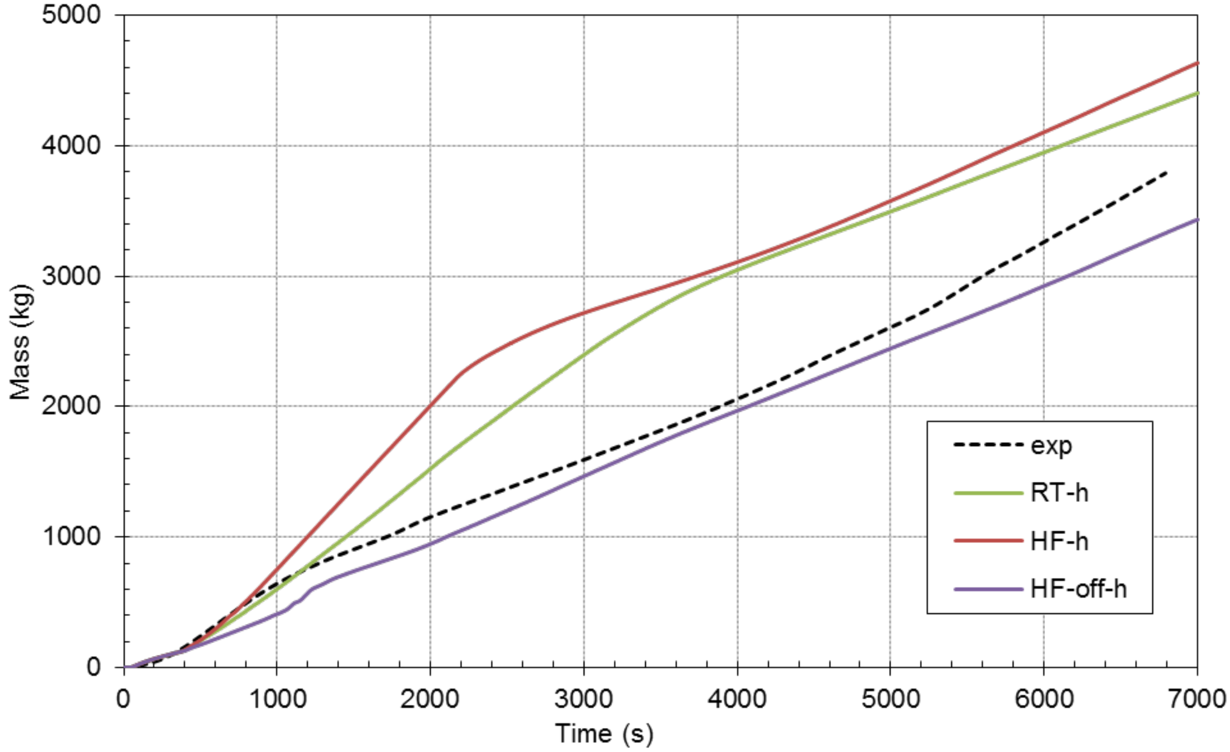

Figure 3-70 Pressurizer PORV Integrated Mass - Best Adjusted Cases (0 – 7000 s)

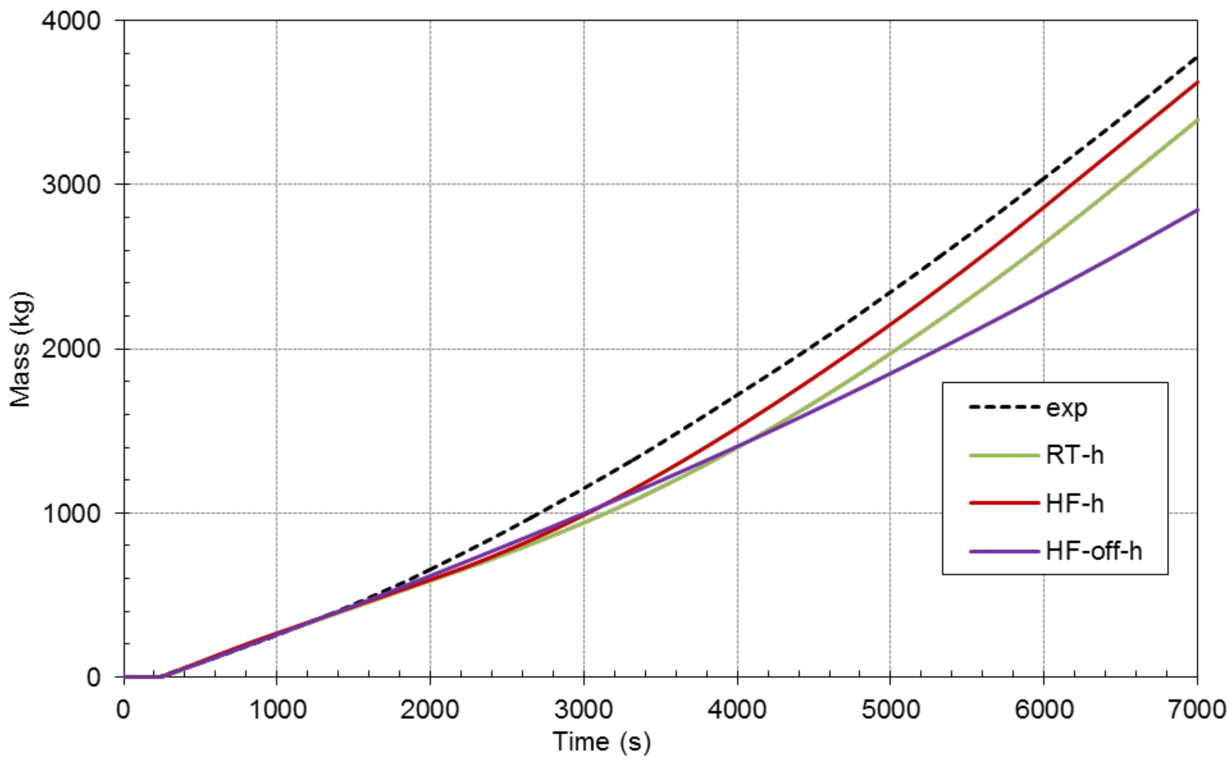

Figure 3-71 HPIS Integrated Mass - Best Adjusted Cases (0 – 7000 s)

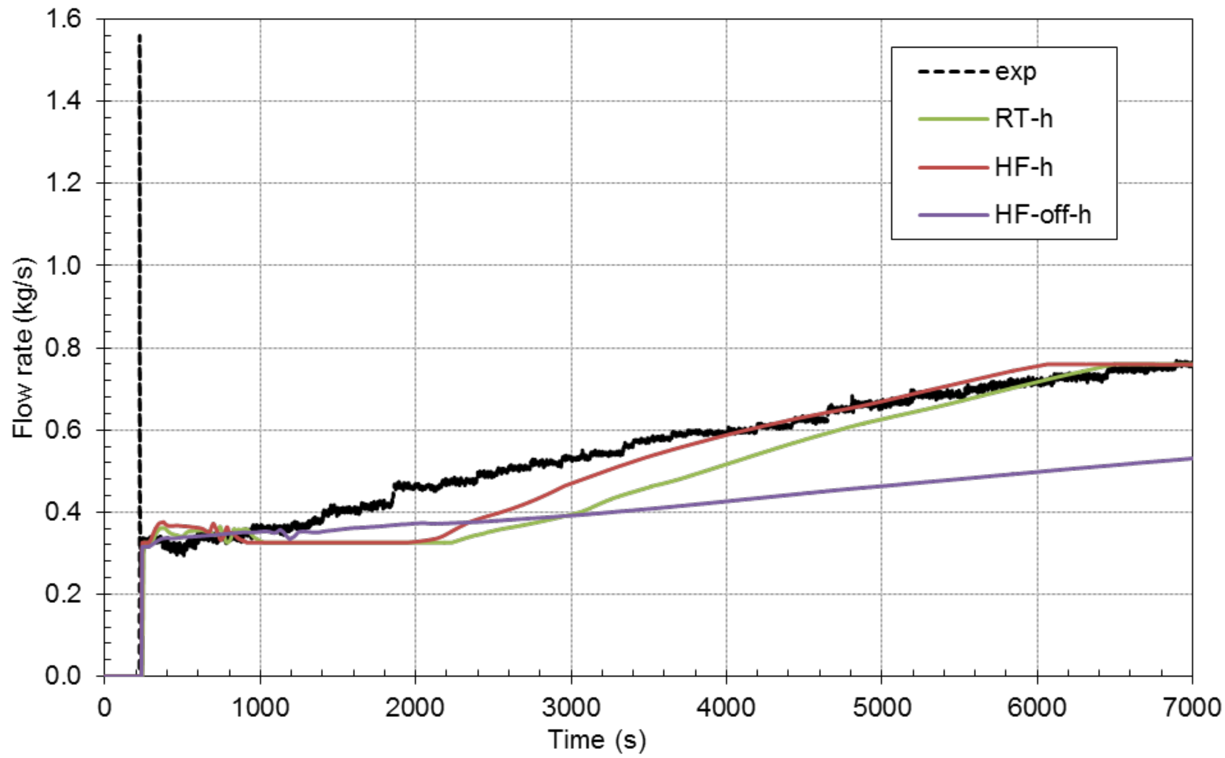

Figure 3-72 HPIS Mass Flow - Best Adjusted Cases (0 – 7000 s)

4 CONCLUSIONS

The RELAP5/MOD3.3 Patch 05 simulation results for short term response (0-300 s) and long term response (0-7000 s) are presented. The results suggest that in the short term simulation of LP-FW-1 test the simulated results match the major events quite well and that all calculations are very similar. In the long term simulation the results suggest that the entrainment to the surge line is important for the correct results. When using the Henry-Fauske choke flow model with the off-take model option for entrainment and increasing the heat losses of the primary system to the containment (changing heat transfer coefficient to 200% compared to base case), the code-predicted results were generally in good agreement with the measured data. Only when using the off-take model option the discharged flow through the primary power operated valves could be adequately simulated. Therefore, it is concluded that the RELAP5/MOD3.3 Patch 05 computer code has sufficient capability in predicting the thermal hydraulic phenomena during total loss of feedwater, which is the design extension condition.

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K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

After Fukushima-Daiichi in the Europe the design extension conditions (DEC) were introduced as preferred method for giving due consideration to the complex sequences and severe accidents without including them in the design basis conditions. The purpose of this study was to assess the latest RELAP5/ MOD3.3 Patch 05 computer code for the simulation of such DEC. The LP-FW-01 test performed in 1983 on the Loss of Fluid Test Facility (LOFT) has been used for simulation. The LP-FW-01 test represents a fault sequence in which a total loss of feedwater to the steam generator is followed by recovery by primary system feed-and-bleed. The RELAP5/MOD3 steady state input deck available from literature has been adapted to RELAP5/MOD3.3 Patch 05, while transient input deck to simulate LP-FW-01 test has been newly developed. The simulation results for short term response (0-300 s) and long term response (0-7000 s) are presented in the report. The results suggest that in the short term simulation of LP-FW-1 test the simulated results matches the major events very good. In the long term the simulation results suggest that the entrainment to the surge line is important for the correct results.

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