

# **Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron Unit 1**

Appendices D to G

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# **Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron Unit 1**

## **Appendices D to G**

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## ABSTRACT

This report extends the work documented in NUREG-1953, “Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom” to the Byron Station, Unit 1. Its purpose is to produce an additional set of best-estimate thermal-hydraulic calculations that can be used to confirm or enhance specific success criteria (SC) for system performance and operator timing found in the agency’s probabilistic risk assessment (PRA) tools. Along with enhancing the technical basis for the Agency’s independent standardized plant analysis risk (SPAR) models, these calculations are expected to be a useful reference to model end-users for specific regulatory applications (e.g., the Significance Determination Process). The U.S. Nuclear Regulatory Commission selected Unit 1 of the Byron Station for this study because it is generally representative of a group of four-loop Westinghouse plants with large, dry containment designs.

This report first describes major assumptions used in this study, including the basis for using a core damage (CD) surrogate of 2,200 degrees Fahrenheit (1,204 degrees Celsius) peak cladding temperature (PCT). The justification for this PCT is documented in NUREG/CR-7177, “Compendium Of Analyses To Investigate Select Level 1 Probabilistic Risk Assessment End-State Definition And Success Criteria Modeling Issues.” The major plant characteristics for Byron Unit 1 are then described, in addition to the MELCOR model used to represent the plant. Finally, the report presents the results of MELCOR calculations for selected initiators and compares these results to SPAR SC, the licensee’s PRA sequence timing and SC, or other generic studies.

The study results provide additional timing information for several PRA sequences, confirm many of the existing SPAR model modeling assumptions, and provide a technical basis for a few specific SPAR modeling changes. Potential SPAR model changes supported by this study include:

- Small-Break Loss-of-Coolant Accident (SLOCA) Sequence Timing for Alignment of Sump Recirculation—For sequences where operator cooldown is credited as an alternative to high-pressure recirculation (HPR), the SPAR success criteria related to containment cooling could be enhanced by requiring one containment fan cooler to prevent containment spray actuation. Avoiding spray actuation extends the time available prior to refueling water storage tank depletion and allows the operators to successfully depressurize the plant using the post-LOCA procedures for cases when HPR is not available.
- SLOCA Success Criteria for Steam Generator (SG) Depressurization and Condensate Feed—Action to depressurize the SGs early and align condensate feed is a candidate for inclusion in the SPAR model. This would provide an additional success path for a loss of auxiliary feedwater event. If this is done, hotwell refill or alignment of alternate feedwater later in the scenario would also need to be modeled. Early depressurization to achieve condensate feed was not found to require primary-side depressurization actions (e.g., opening a power-operated relief valve (PORV)).
- SLOCA Success Criteria for Primary Side Bleed and Feed (B&F)—These calculations have demonstrated a potential conservatism that can be removed from the applicable SPAR models. It is proposed that the SC for SLOCA B&F be changed from (one safety

injection (SI) or centrifugal charging pump (CCP) and two PORVs) to (one SI pump and two PORVs) or (one CCP and one PORV). In other words, for SLOCAs the requirement for availability of a second PORV can be removed when a CCP is available.

- Loss of DC Bus-111 – Unavailable Diesel-Driven Auxiliary Feedwater, and Subsequent Primary Side B&F—These calculations are generally representative of non-loss-of-coolant accident (non-LOCA) B&F situations and have demonstrated a potential improvement that can be implemented in the Byron SPAR model. It is proposed that the SC for non-LOCA B&F be changed from (one SI or CCP and two PORVs) to (one CCP and one PORV). In other words, the same one CCP and one PORV enhancement as above is credited, but credit is eliminated for cases with no CCP available. This initiator was chosen because it was qualitatively felt to be more restrictive than those scenarios categorized as general transients in the PRA, and thus the conclusions are believed to be applicable to those initiators as well. Note that the applicability of the loss of DC bus SC may vary, (e.g., due to the unique reactor coolant pump trip situation that this initiator creates) and should be evaluated on a case-by-case basis before implementation for other plant models.
- SGTR – Spontaneous Steam Generator Tube Rupture with No Operator Action—For sequences with successful high-pressure injection (HPI) and auxiliary feedwater, but with steam generator isolation having failed, an additional success path or additional recovery credit may be justifiable pending additional consideration of closely-related accident sequence and human reliability modeling assumptions.
- Medium-Break Loss-of-Coolant Accident (MLOCA) – Injection SC— For breaks in the lower half of the MLOCA range, it was found that an early operator-induced depressurization based on the Functional Restoration Procedure (FRP) for inadequate core cooling would be needed to avoid core damage if HPI fails. The time available to implement these actions following the FRP entry criterion being met could be short. The accident sequence modeling and human reliability analysis associated with secondary-side cooldown for these situations (MLOCA with HPI failed) should be reviewed.

## FOREWORD

The U.S. Nuclear Regulatory Commission's (NRC's) standardized plant analysis risk (SPAR) models are used to support a number of risk-informed initiatives. The fidelity and realism of these models is ensured through a number of processes, including cross-comparison with industry models, review and use by a wide range of technical experts, and confirmatory analysis. The following report—prepared by staff in the Office of Nuclear Regulatory Research in consultation with staff from the Office of Nuclear Reactor Regulation, experts from Energy Research Incorporated and Idaho National Laboratory, and the agency's senior reactor analysts—represents a major confirmatory analysis activity.

Probabilistic risk assessment (PRA) models for nuclear power plants rely on underlying modeling assumptions known as success criteria (SC) and sequence timing assumptions. These criteria and assumptions determine what combination of system and component availabilities will lead to postulated core damage (CD), as well as the timeframes during which components must operate or operators must take particular actions. This report investigates certain thermal-hydraulic aspects of a particular SPAR model (which is generally representative of other models within the same class of plant design), with the goal of further strengthening the technical basis for decisionmaking that relies on the SPAR models. This report augments the existing collection of contemporary Level 1 PRA SC analyses, and as such, supports (1) maintaining and enhancing the SPAR models that the NRC develops, (2) supporting the NRC's risk analysts when addressing specific issues in the accident sequence precursor program and the significance determination process, and (3) informing other ongoing and planned initiatives. This analysis employs the MELCOR computer code and uses a plant model developed for this project.

The analyses summarized in this report provide the basis for confirming or changing SC in the SPAR model for the Byron Station Unit 1. Further evaluation of these results will be performed to extend the results to similar plants. In addition, future work is planned to perform similar analysis for other design classes, and past work has already considered other design classes (see NUREG-1953, "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models – Surry and Peach Bottom"). In addition, work has been recently completed to scope other aspects of this topical area, including the degree of variation typical in common PRA sequences and the quantification of conservatisms associated with CD surrogates (see NUREG/CR-7177, "Compendium of Analyses to Investigate Select Level 1 Probabilistic Risk Assessment End-State Definition and Success Criteria Modeling Issues"). Where applicable, insights from that work are referenced in this report. The confirmation of SC and other aspects of PRA modeling using the agency's state-of-the-art tools (e.g., the MELCOR computer code) is expected to receive continued focus as the agency continues to develop and improve its risk tools.



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## ABBREVIATIONS AND ACRONYMS

°C	degree(s) Celsius
°C/hr	degree(s) Celsius per hour
°F	degree(s) Fahrenheit
°F/hr	degree(s) Fahrenheit per hour
ΔT	temperature difference
ACC	accumulator
ADAMS	Agencywide Documents Access and Management System
AFW	auxiliary feedwater
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor
B&F	bleed and feed
BAF	bottom of active fuel
BEP	Byron Emergency Procedure
BWR	boiling-water reactor
CCP	centrifugal charging pump
CCW	component cooling water
CD	core damage
CDF	core damage frequency
CET	core exit temperature
CFR	<i>Code of Federal Regulations</i>
cm	centimeter(s)
CNMT	containment
COR	MELCOR core package
CS	containment spray
CST	condensate storage tank
CVH	control volume hydrodynamics (MELCOR package)
CVTR	Carlinas Virginia Tube Reactor
DC	direct current
DD-AFW	diesel-driven auxiliary feedwater
ECA	emergency contingency action
ECCS	emergency core cooling system
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
ESF	Engineered Safety Features
FCL	fan cooler
FRP	functional restoration procedure
FSAR	Final Safety Analysis Report
ft	foot/feet
ft <sup>3</sup>	cubic foot/feet
FW	feedwater
gal	gallon(s)
gpm	gallon(s) per minute
HEM	homogeneous equilibrium model
HEP	human error probability
HFM	homogeneous frozen model
HPI	high-pressure [ECCS] injection

HPR	high-pressure [ECCS] recirculation
hr	hour(s)
HS	heat structure
in.	inch(es)
iPWR	integral pressurized-water reactor
K	Kelvin
kg	kilogram(s)
kg/s	kilogram(s) per second
kPa	kilopascal(s)
lb/s	pound(s) per second
LBLOCA	large-break loss-of-coolant accident
lbm/hr	pound(s) mass per hour
LOCA	loss-of-coolant accident
LoDCB-111	loss of DC bus 111
LOFT	loss-of-fluid test
LPI	low pressure [ECCS] injection
LPR	low pressure [ECCS] recirculation
LTOP	low temperature overpressure protection
m	meter(s)
m <sup>3</sup>	cubic meter(s)
m <sup>3</sup> /min	cubic meter(s) per minute
m <sup>3</sup> /s	cubic meter(s) per second
MAAP4	Modular Accident Analysis Program version 4
MD-AFW	motor-driven auxiliary feedwater
MELCOR	Not an acronym
MFW	main feedwater
min	minute(s)
MLOCA	medium-break loss-of-coolant accident
MPa	megapascal(s)
MPa abs	megapascal(s) absolute
MSIV	main steam isolation valve
MUR	measurement uncertainty recapture
MW	megawatt(s)
MWt	megawatt(s) thermal
NPSH	net positive suction head
NR	narrow range [water level]
NRC	U.S. Nuclear Regulatory Commission
PCT	peak cladding temperature
PORV	power- (or pilot-) operated relief valve
PRA	probabilistic risk assessment
PRT	pressurizer relief tank
PSA	Probabilistic Safety Assessment
psi	pound(s) per square inch
psia	pound(s) per square inch absolute
psid	pound(s) per square inch differential
psig	pound(s) per square inch gage
PWR	pressurized-water reactor
PZR	pressurizer
RCFC	reactor containment fan cooler
RCP	reactor coolant pump
RCS	reactor coolant system

recirc	recirculation
RHR	residual heat removal
RHR HX	residual heat removal heat exchanger
RPS	reactor protection system
RPV	reactor pressure vessel
RWST	refueling water storage tank
s	second(s)
SC	success criterion/criteria
SDP	significance determination process
scfm	standard cubic foot/feet per minute
SG	steam generator
SG-x	steam generator in loop x
SGTR	steam generator tube rupture
SI	safety injection
SLOCA	small-break loss-of-coolant accident
SOARCA	State-of-the-Art Reactor Consequence Analyses
SPAR	standardized plant analysis risk
SRV	safety relief valve
TAF	top of active fuel
$T_{avg}$	loop average temperature
TBV	turbine bypass valve
TCL	cladding temperature
TRACE	TRAC/RELAP5 Advanced Computational Engine
VCT	volume control tank
WR	wide range [water level]



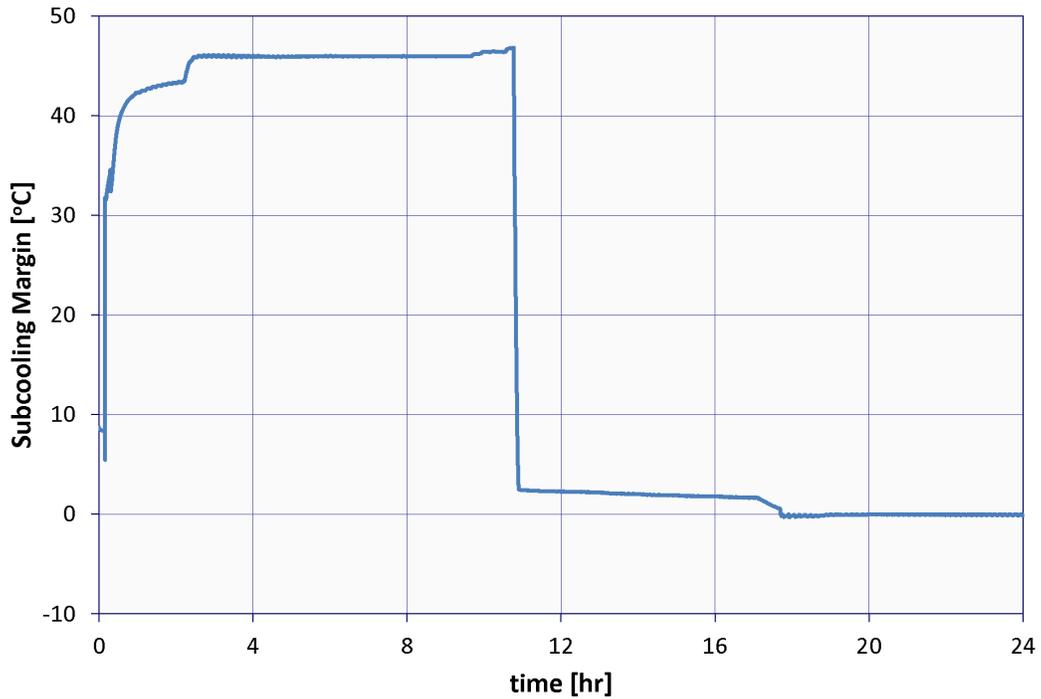
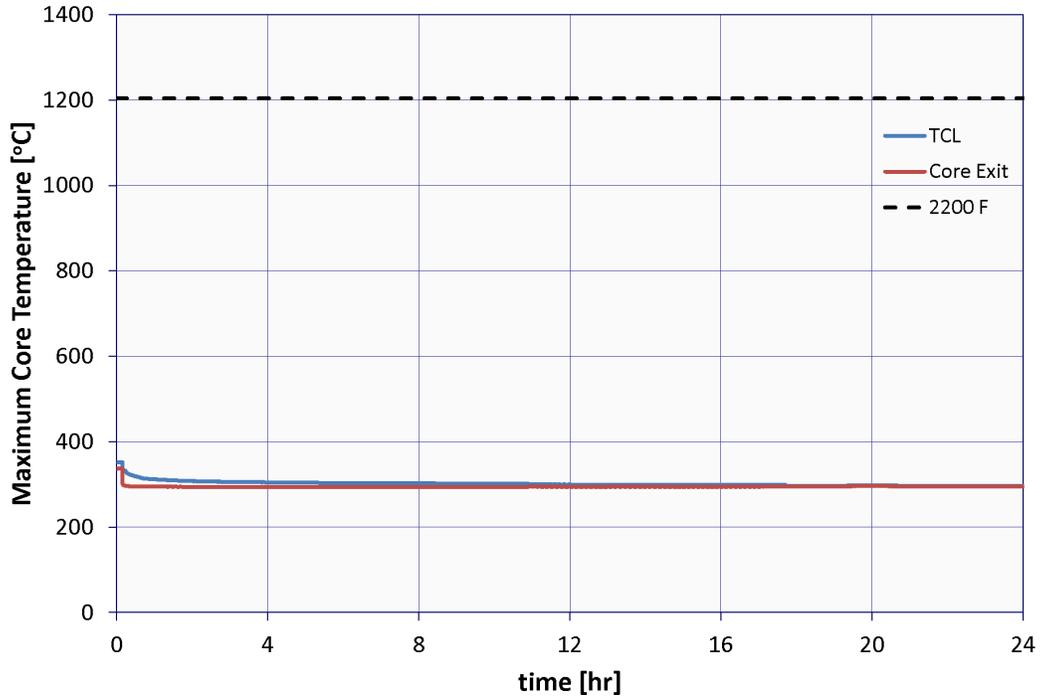
## **APPENDIX D**

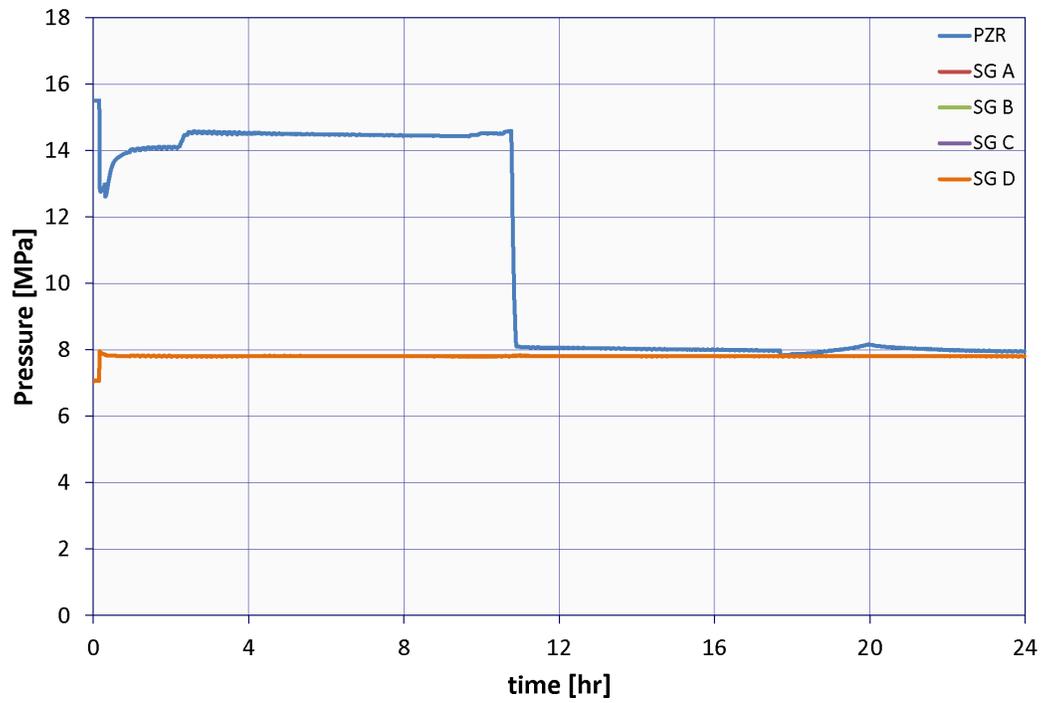
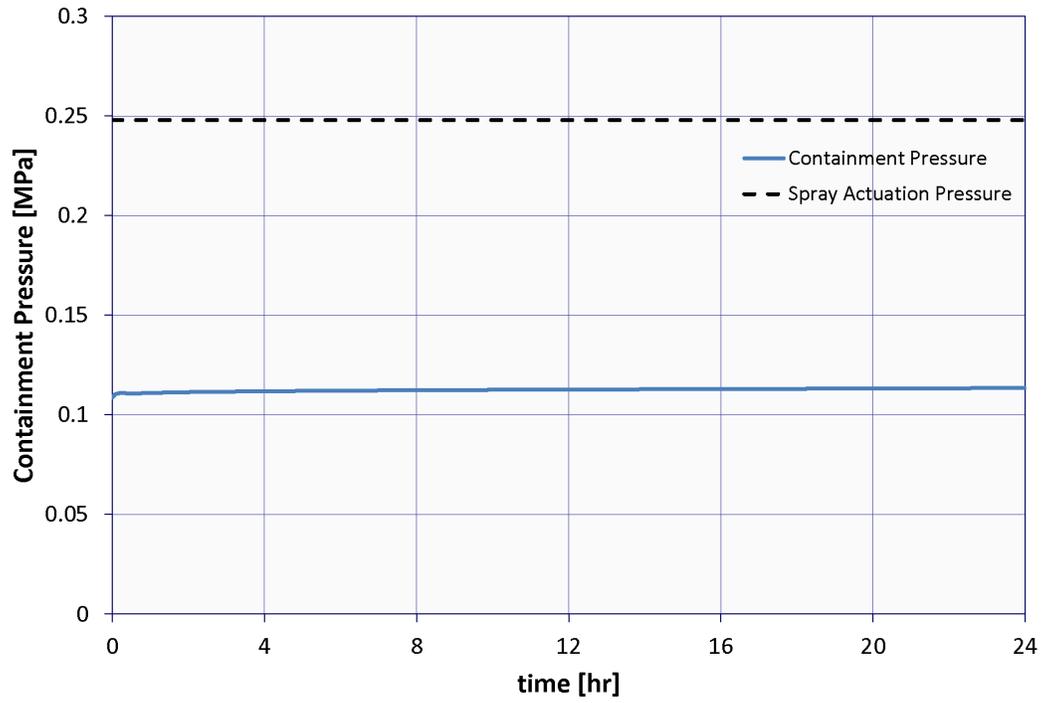
# **DETAILED STEAM GENERATOR TUBE RUPTURE ANALYSIS RESULTS**

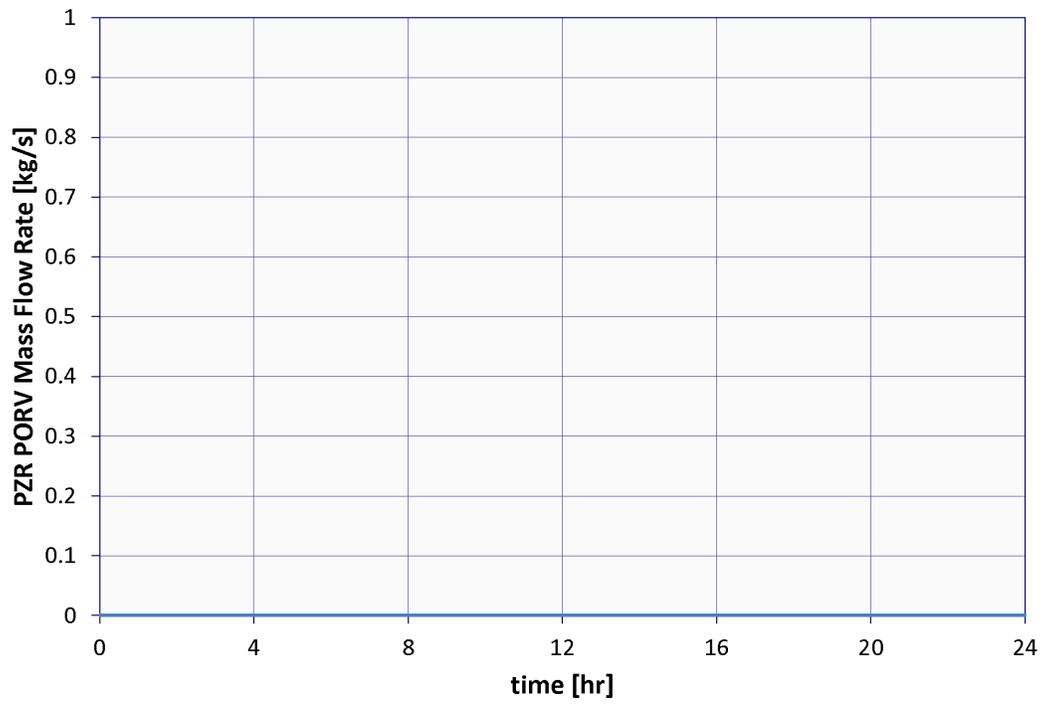
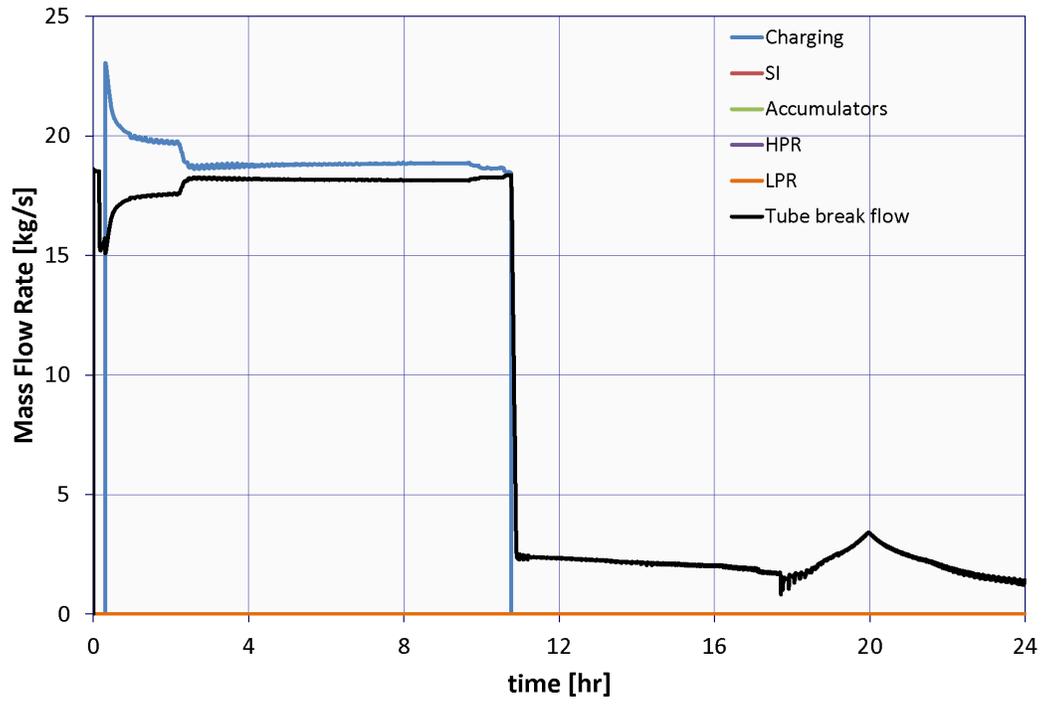


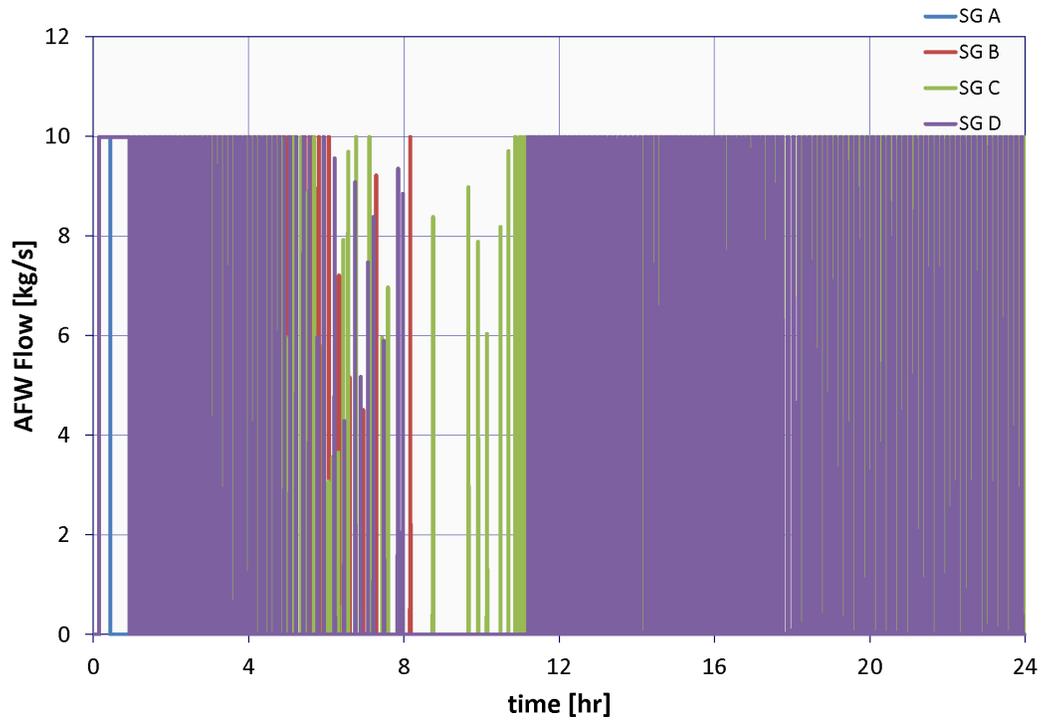
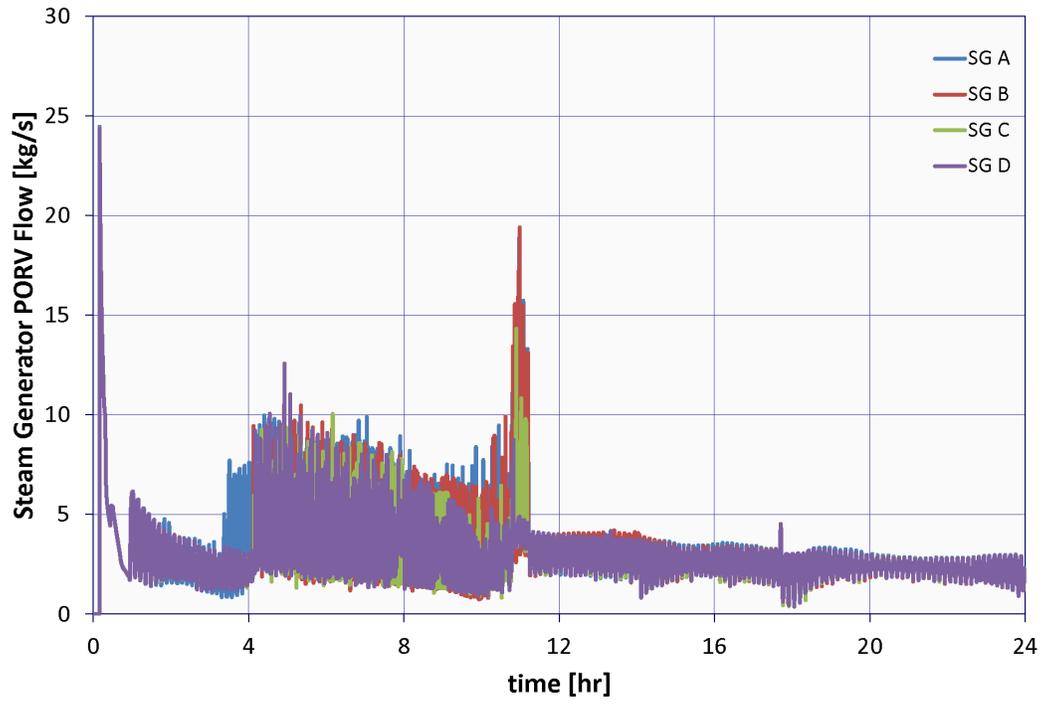
## D.1 Spontaneous SG Tube Rupture with No Operator Action

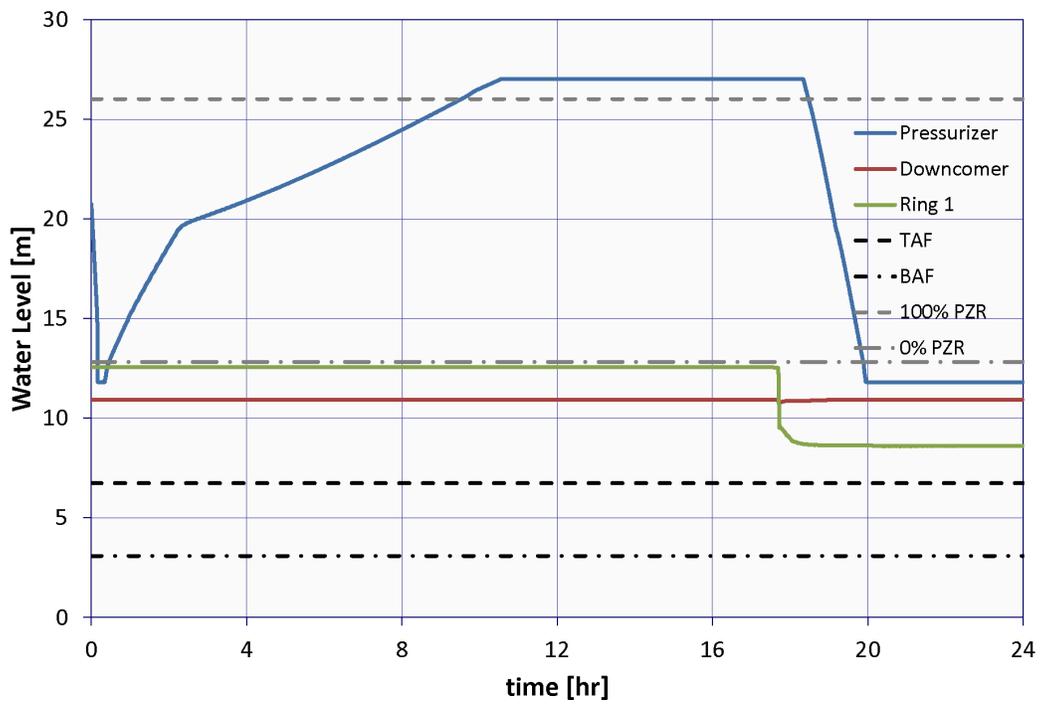
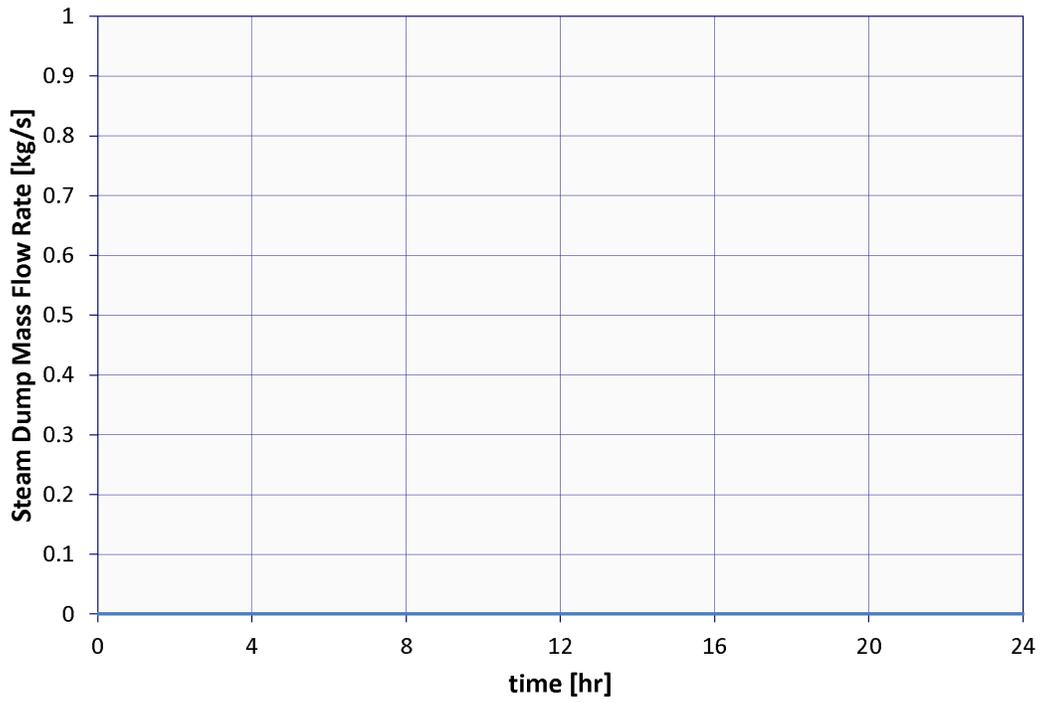
### D.1.1 Case 1: 0.5 Tube, Min ECCS, No Steam Dumps

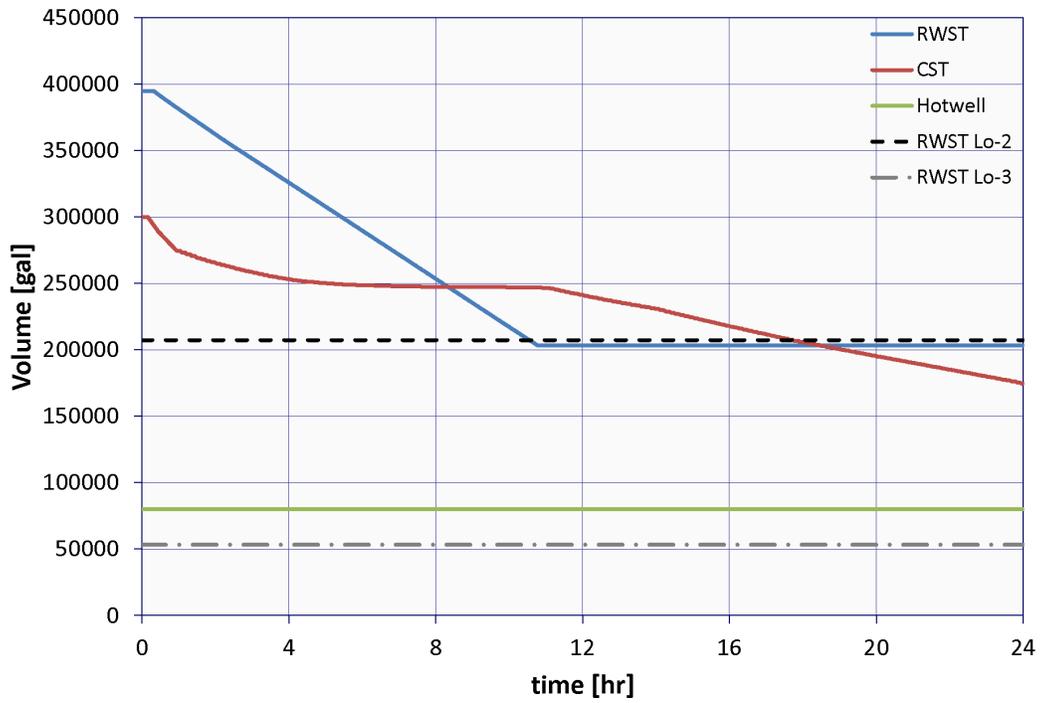
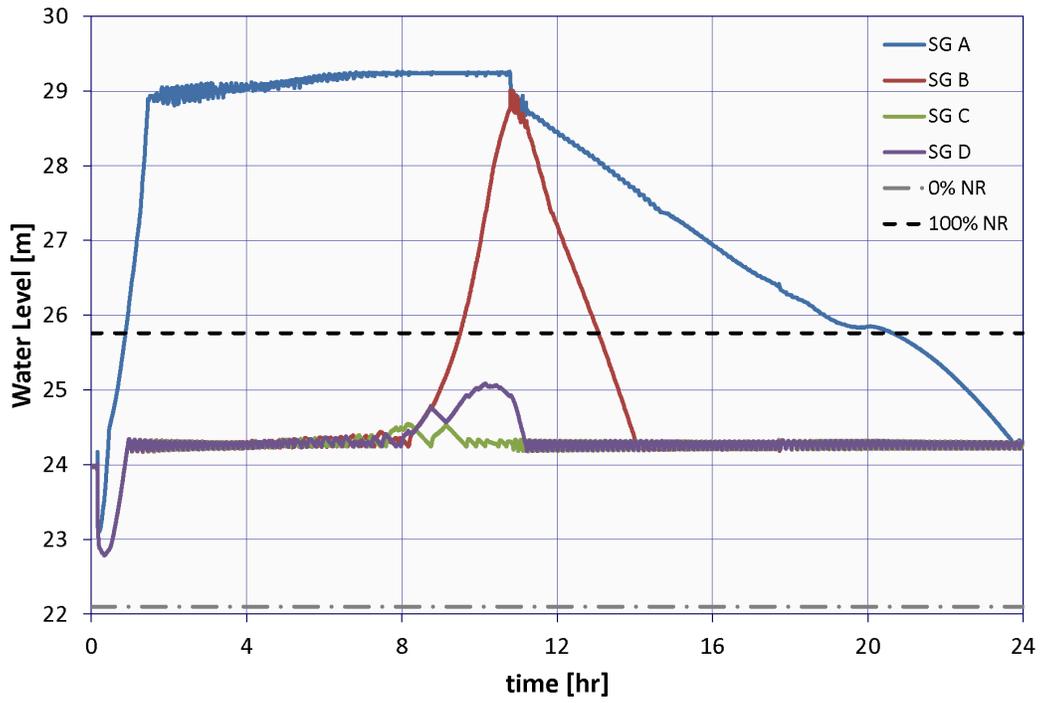




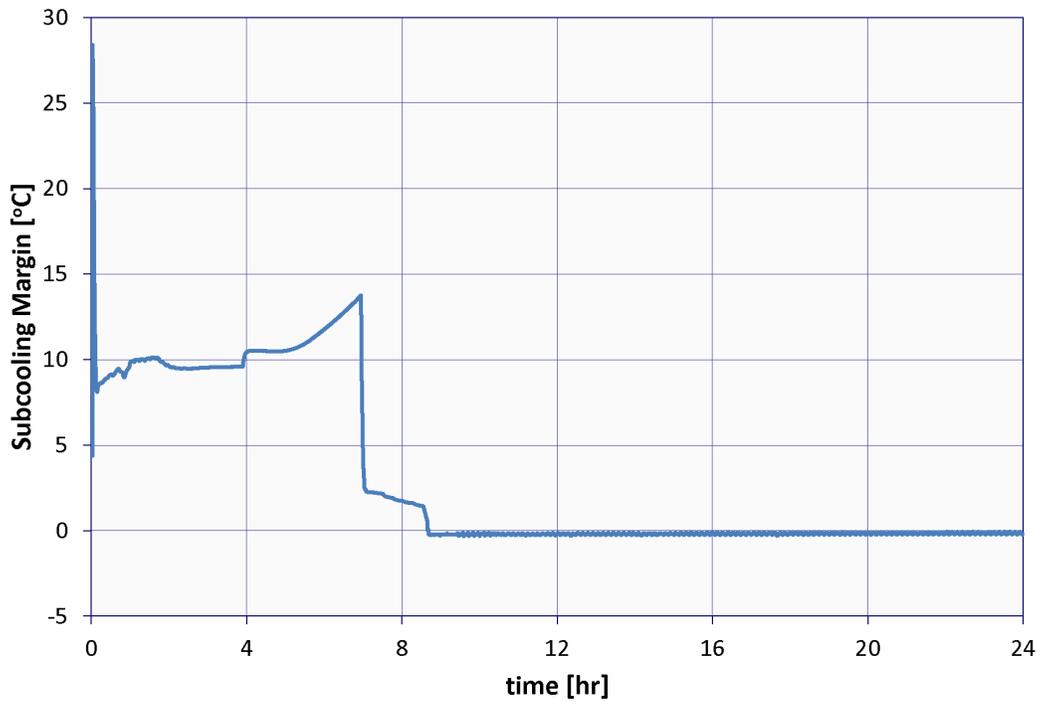
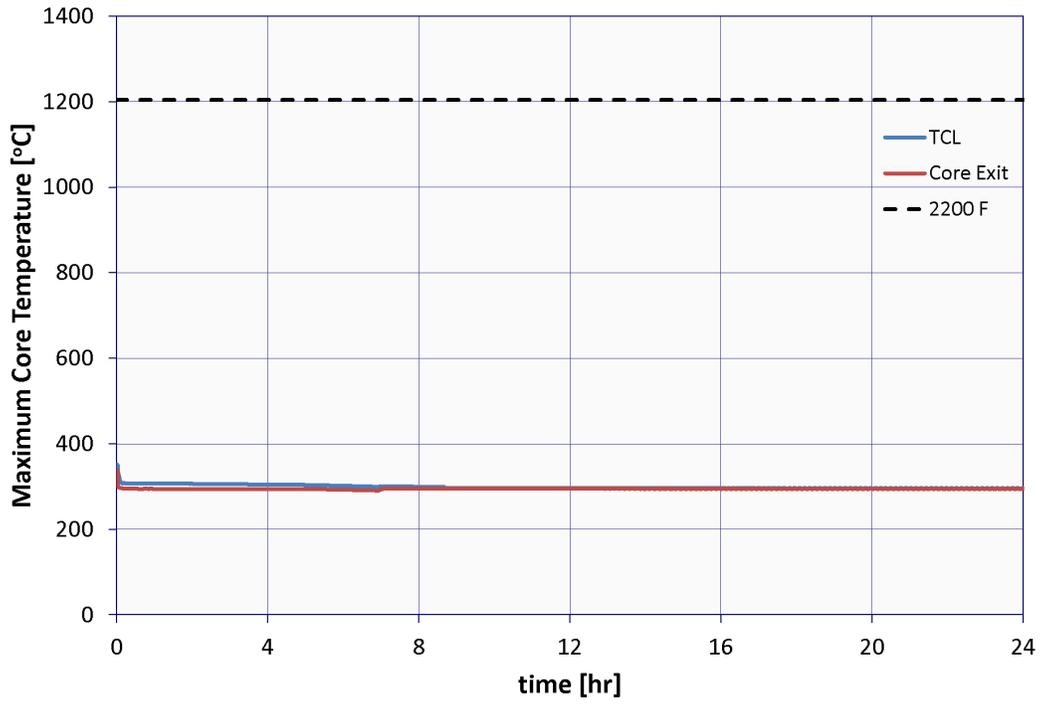


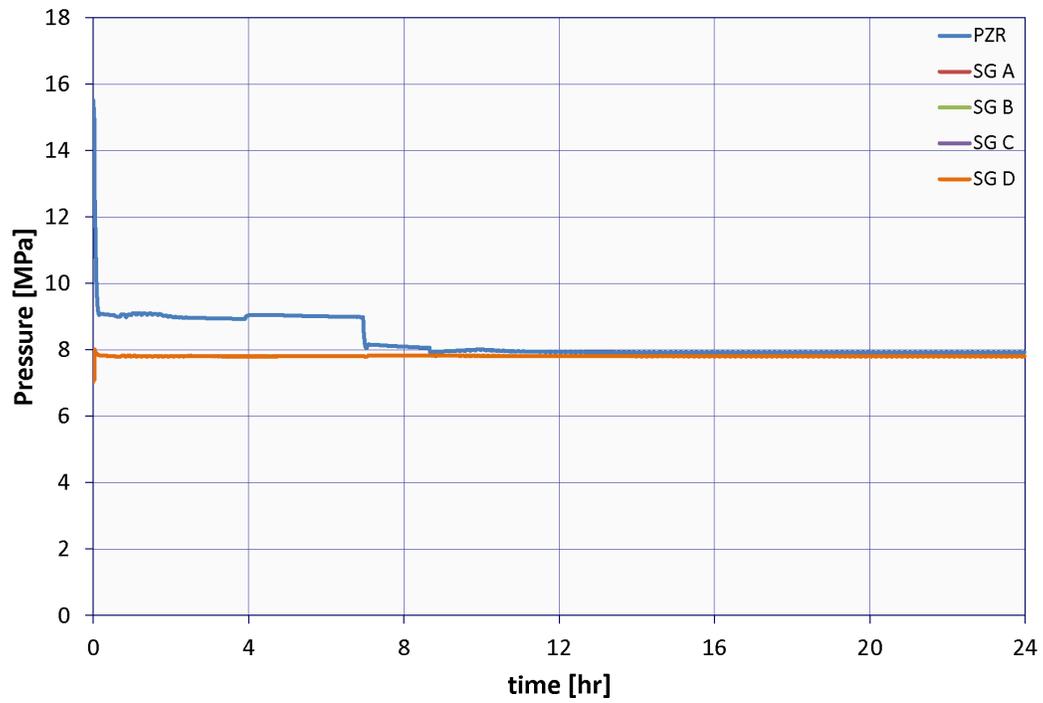
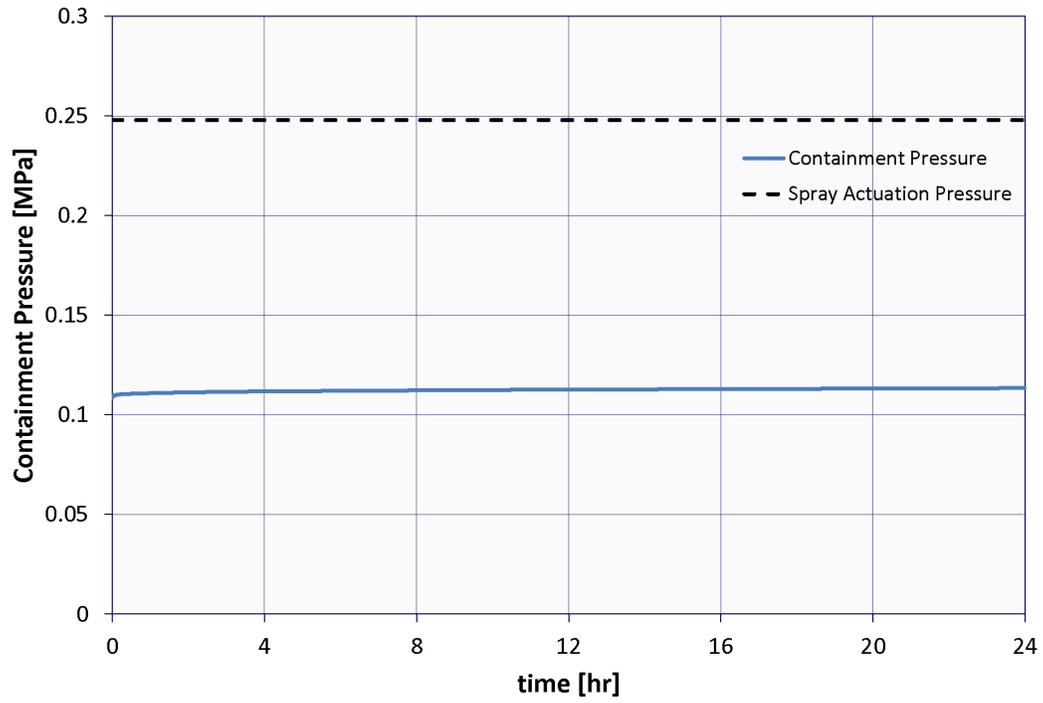


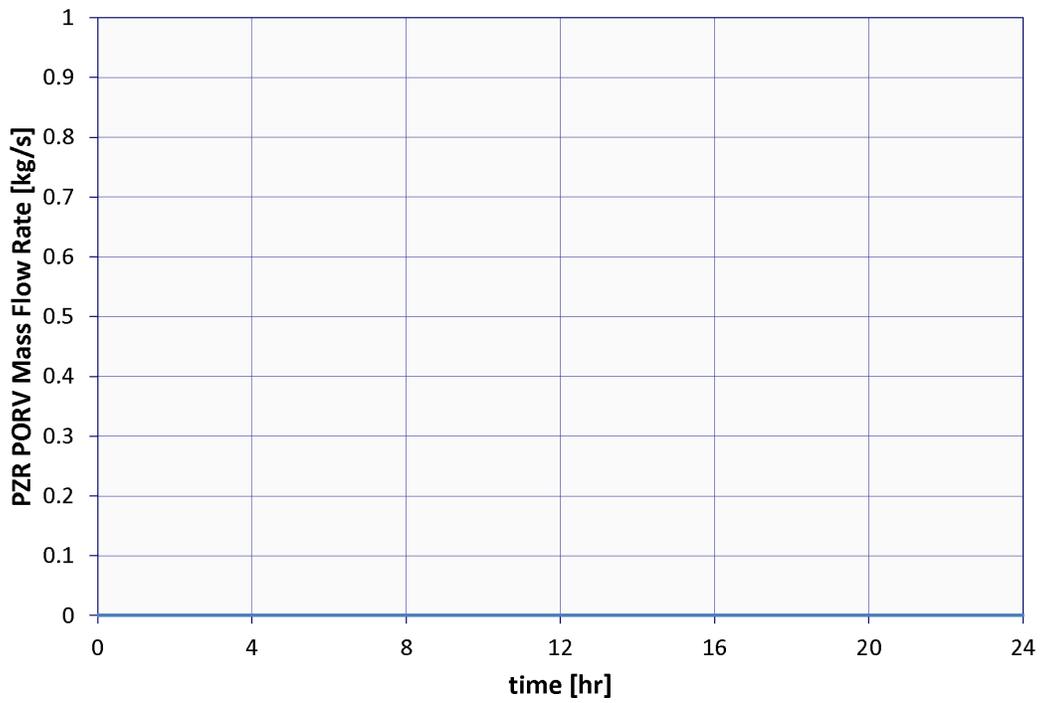
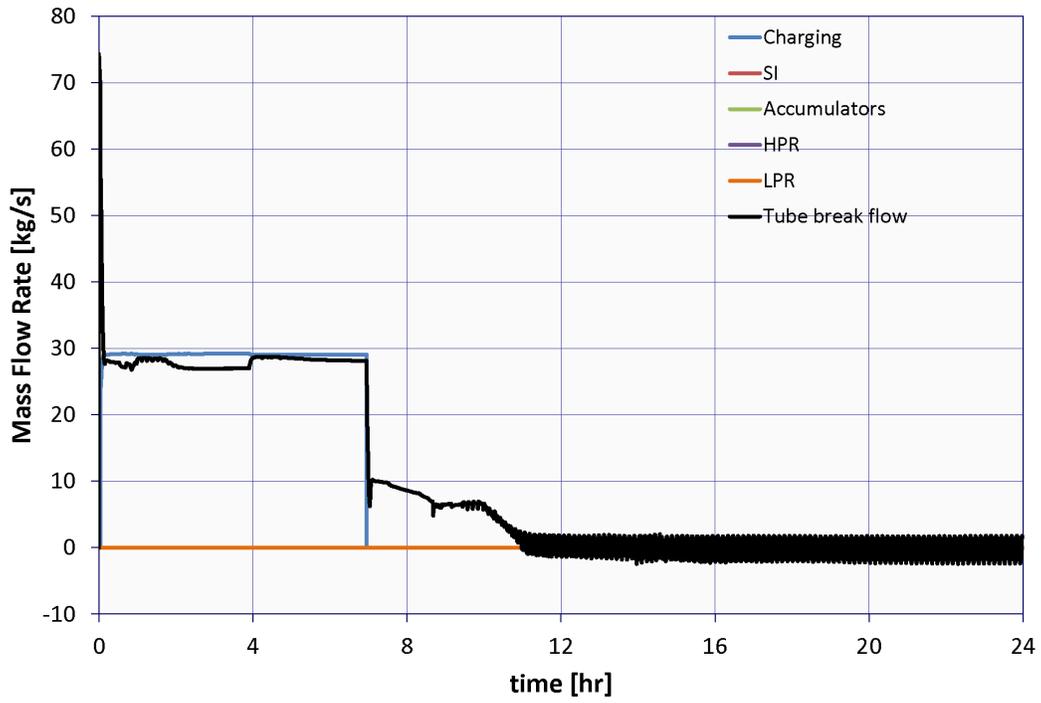


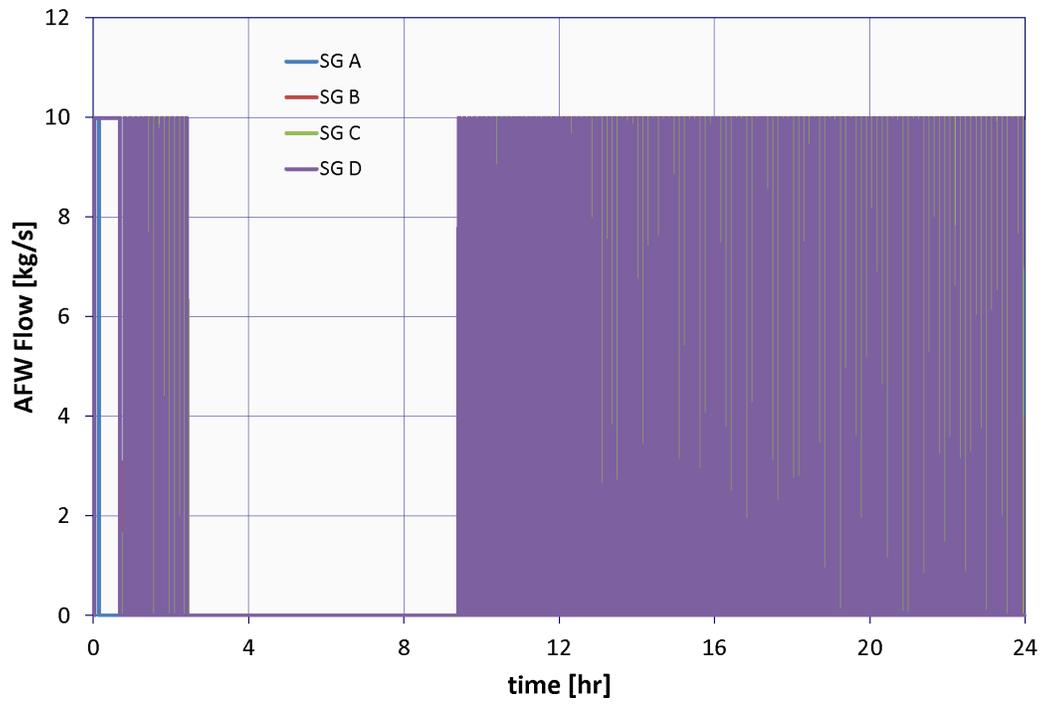
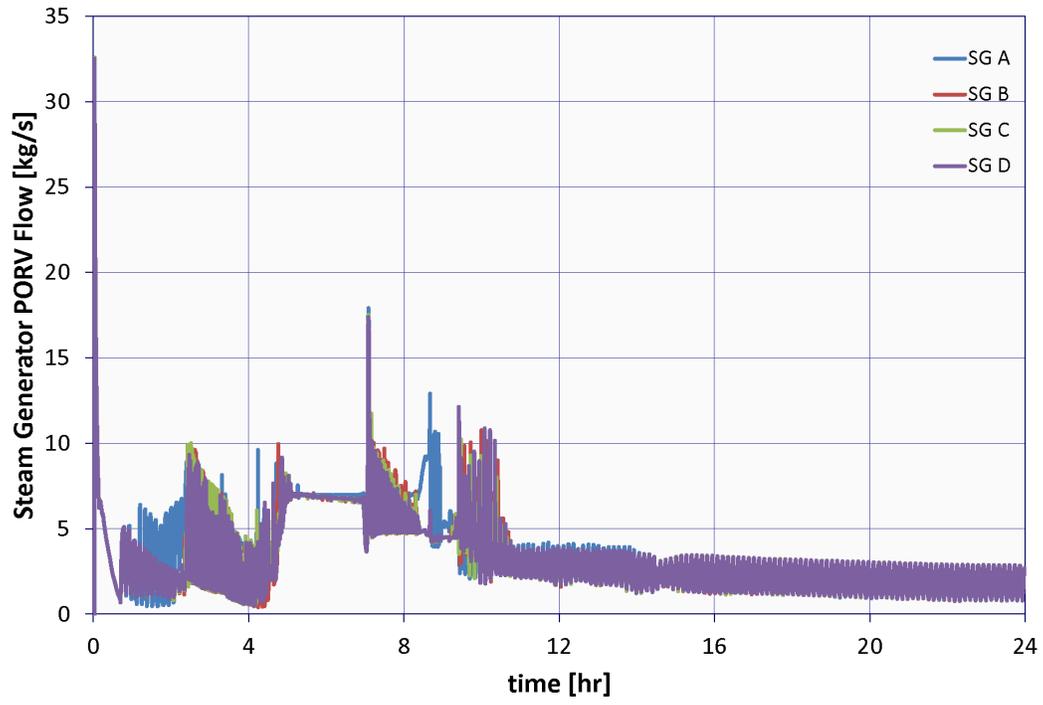


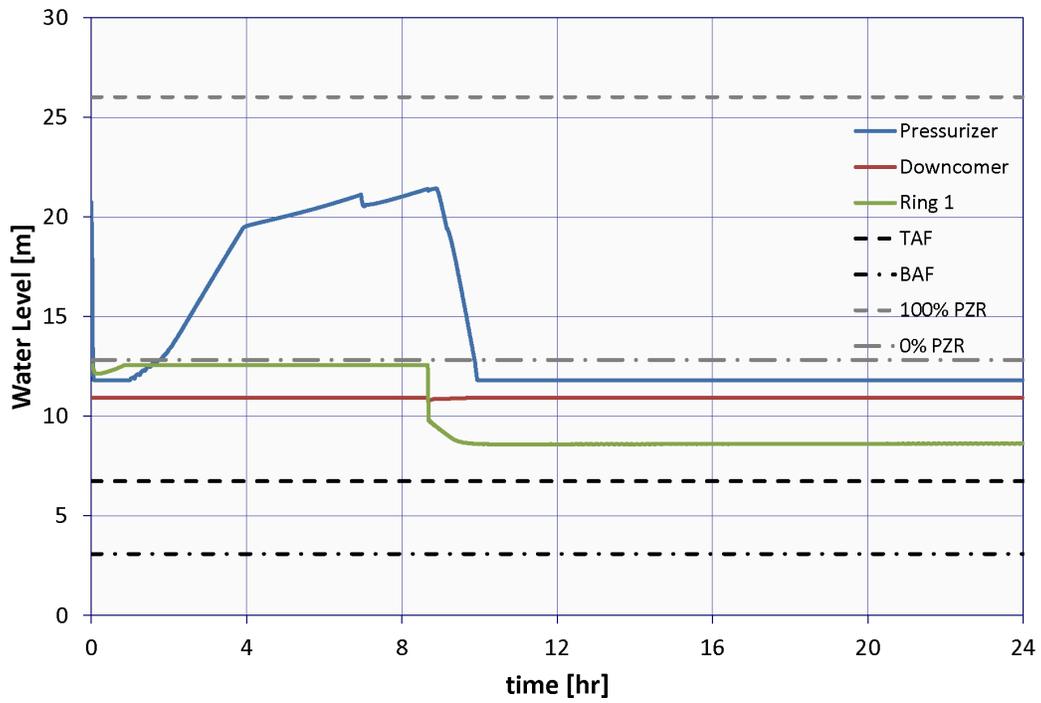
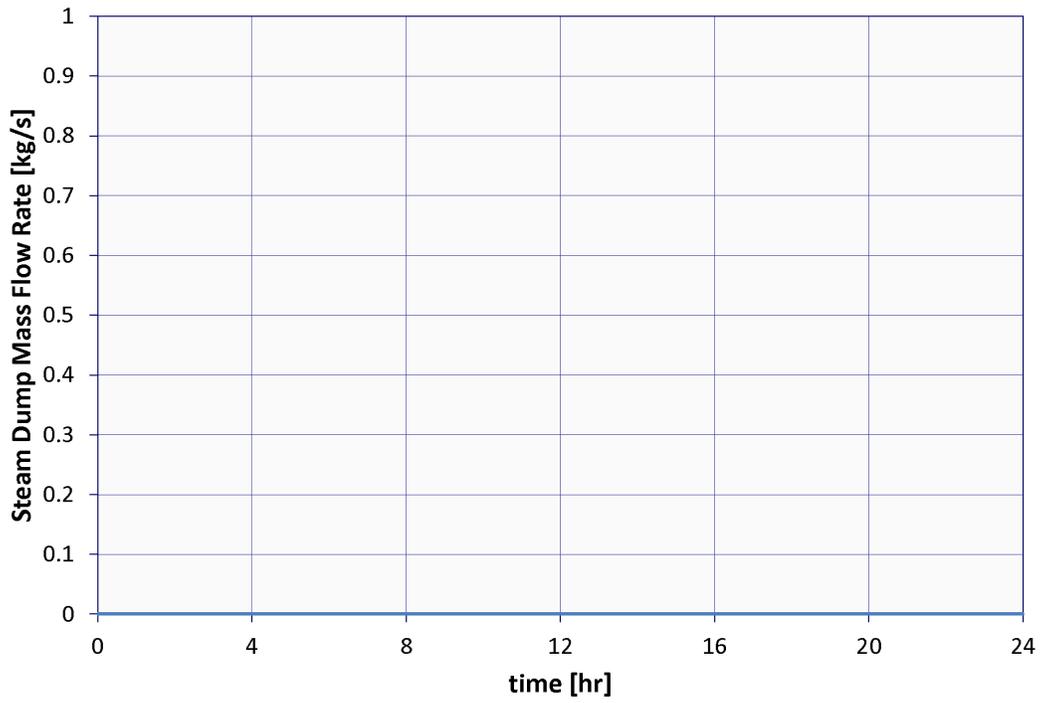
D.1.2 Case 2: 2 Tubes, Max ECCS, No Steam Dumps

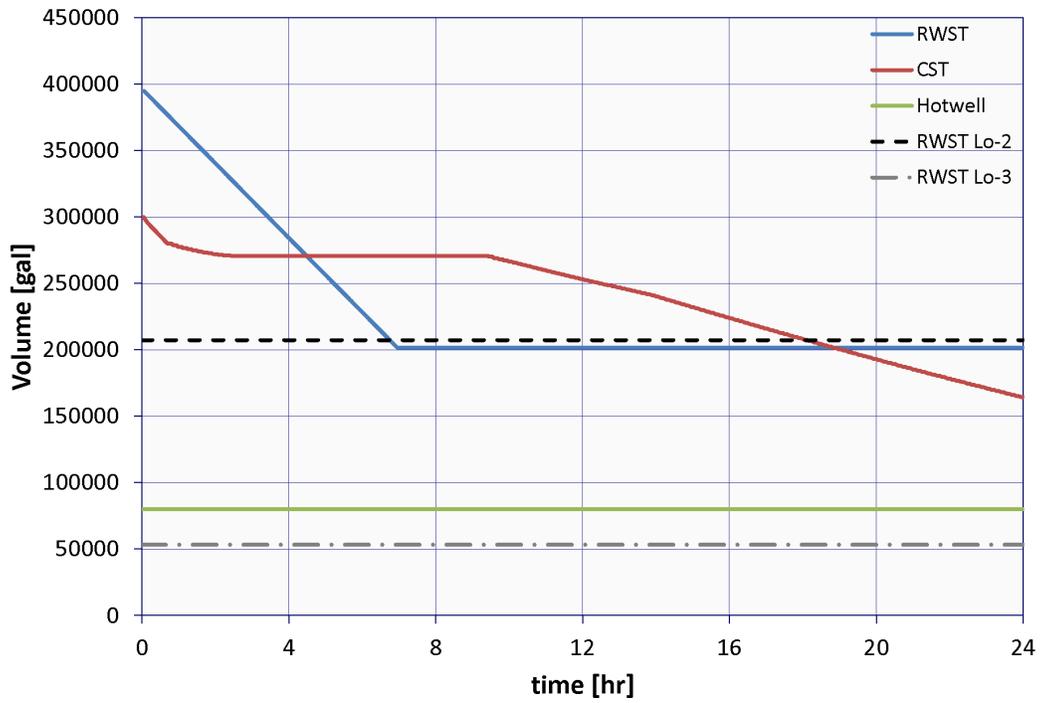
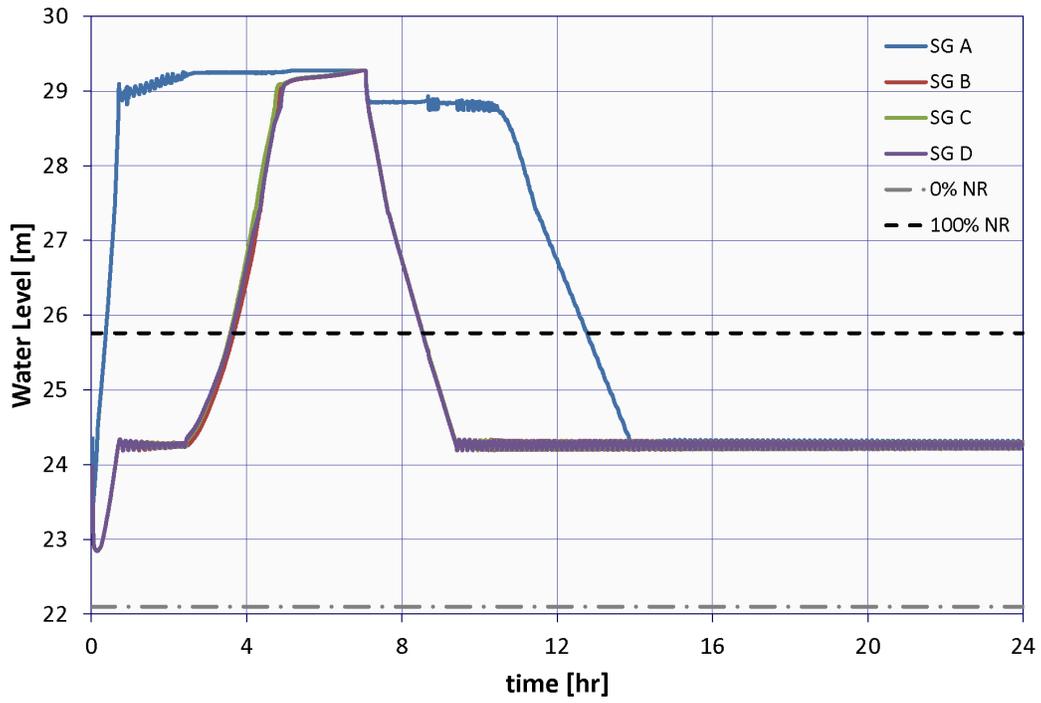




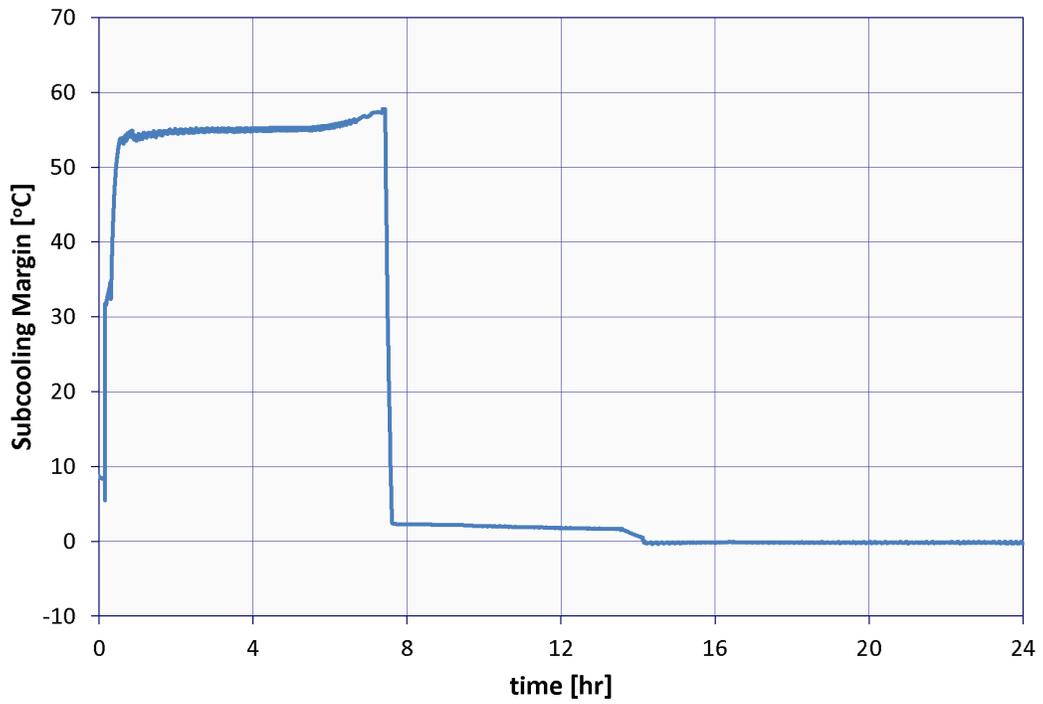
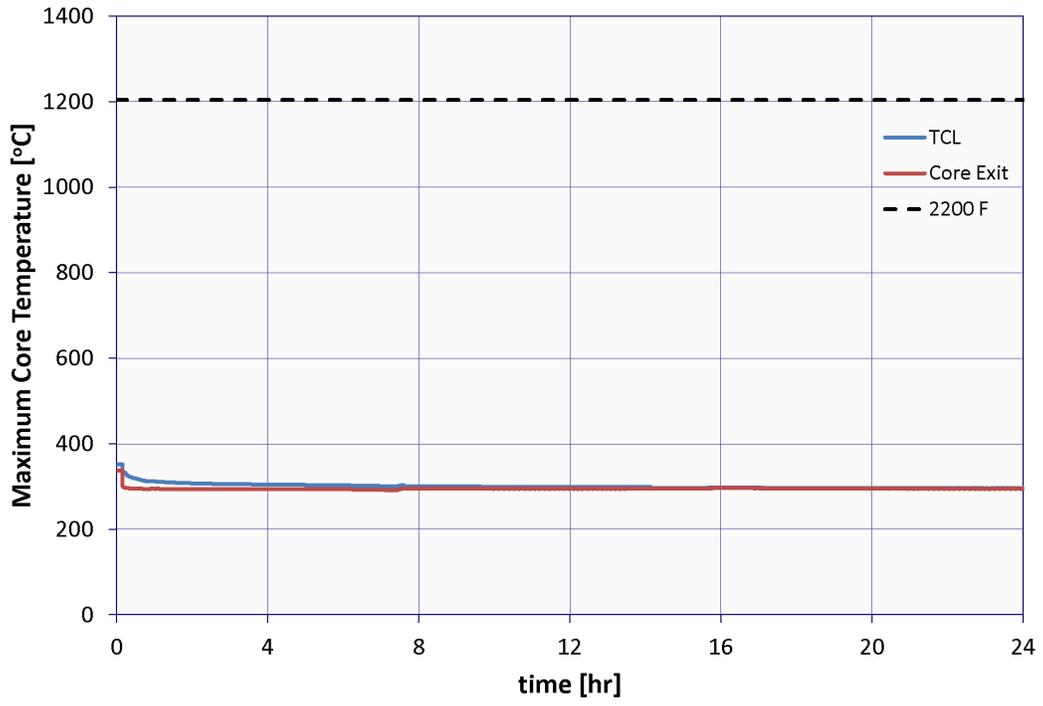


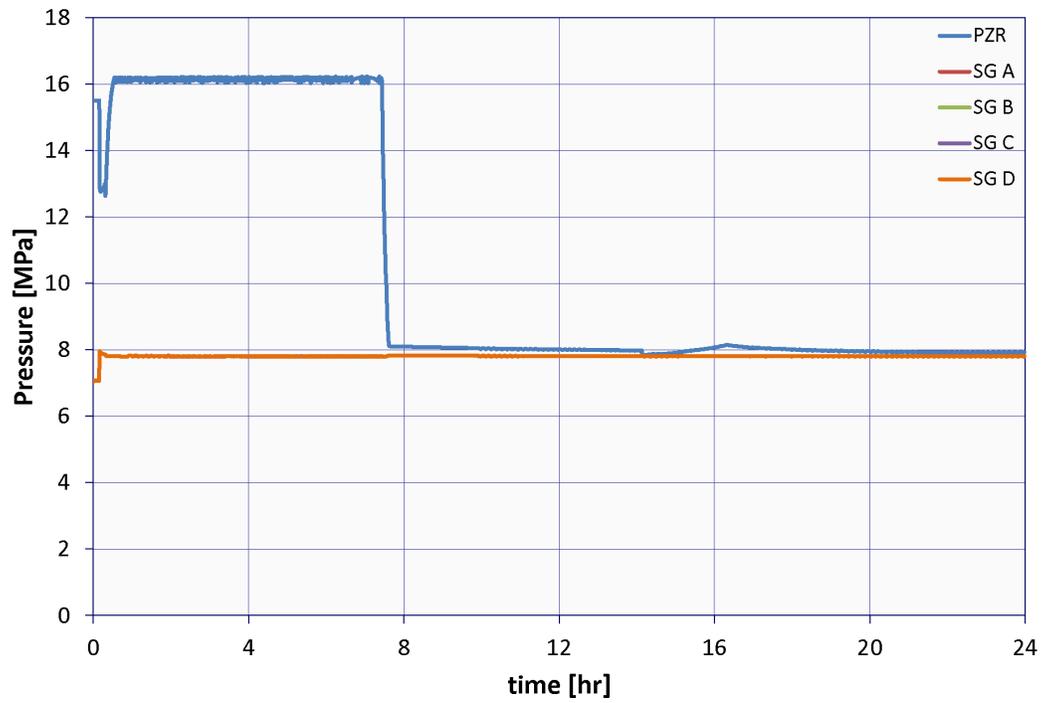
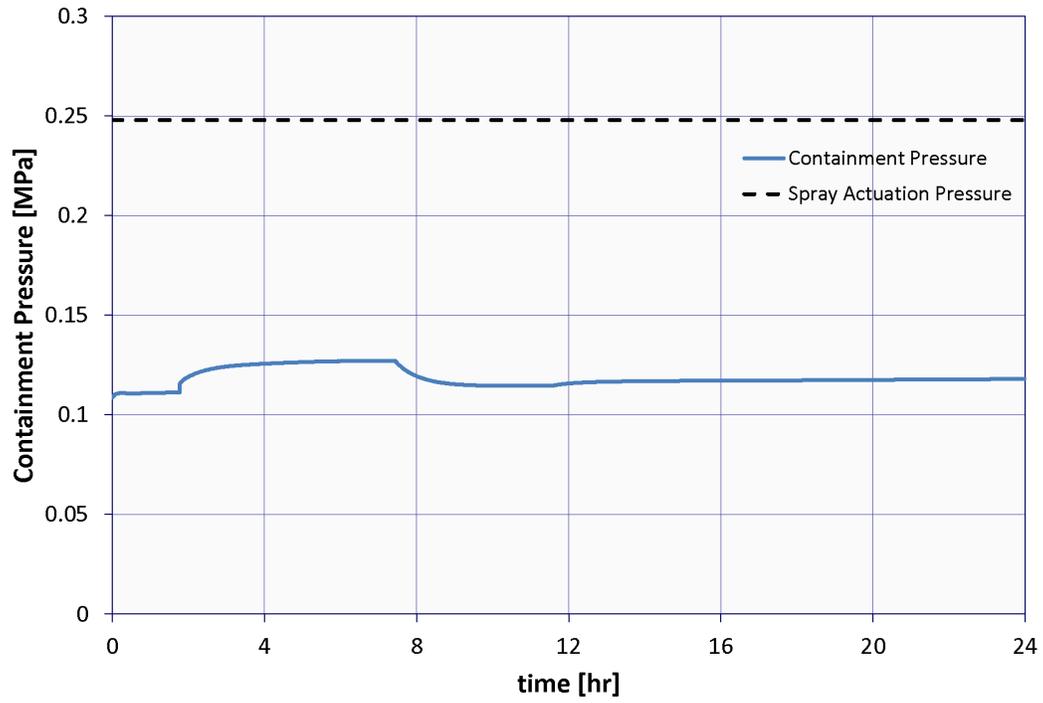


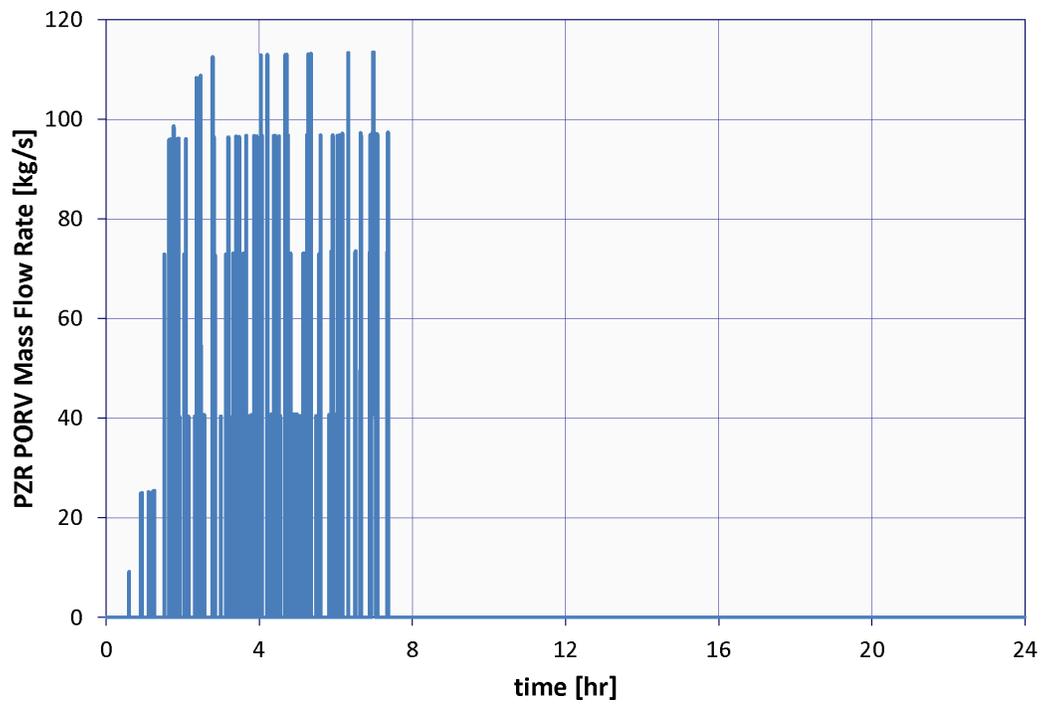
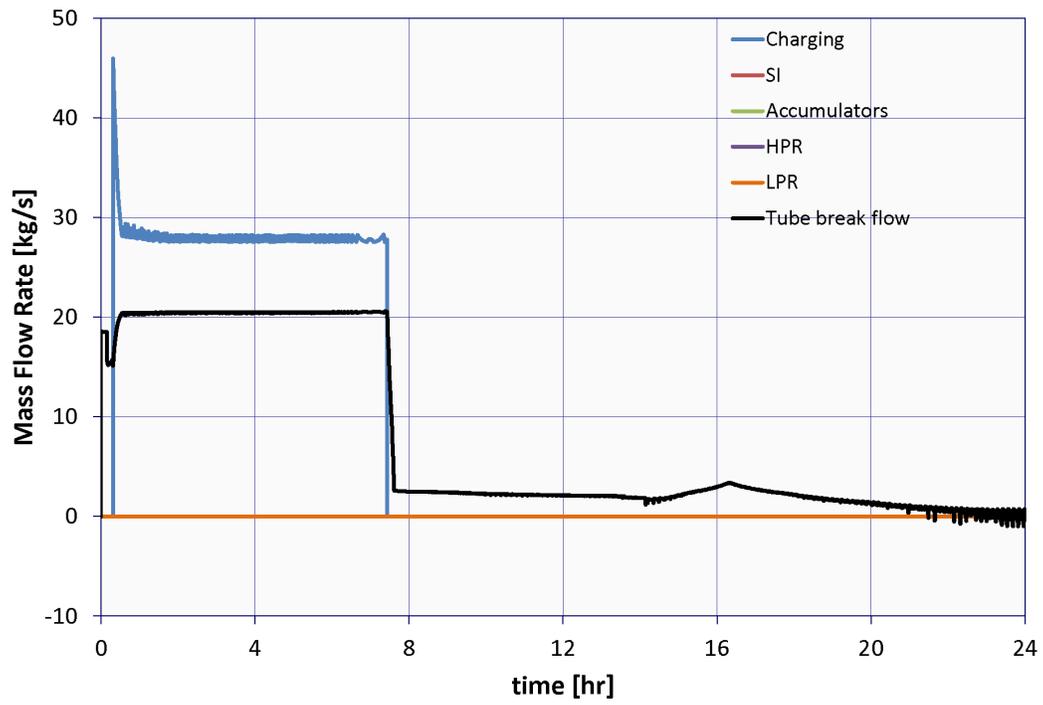


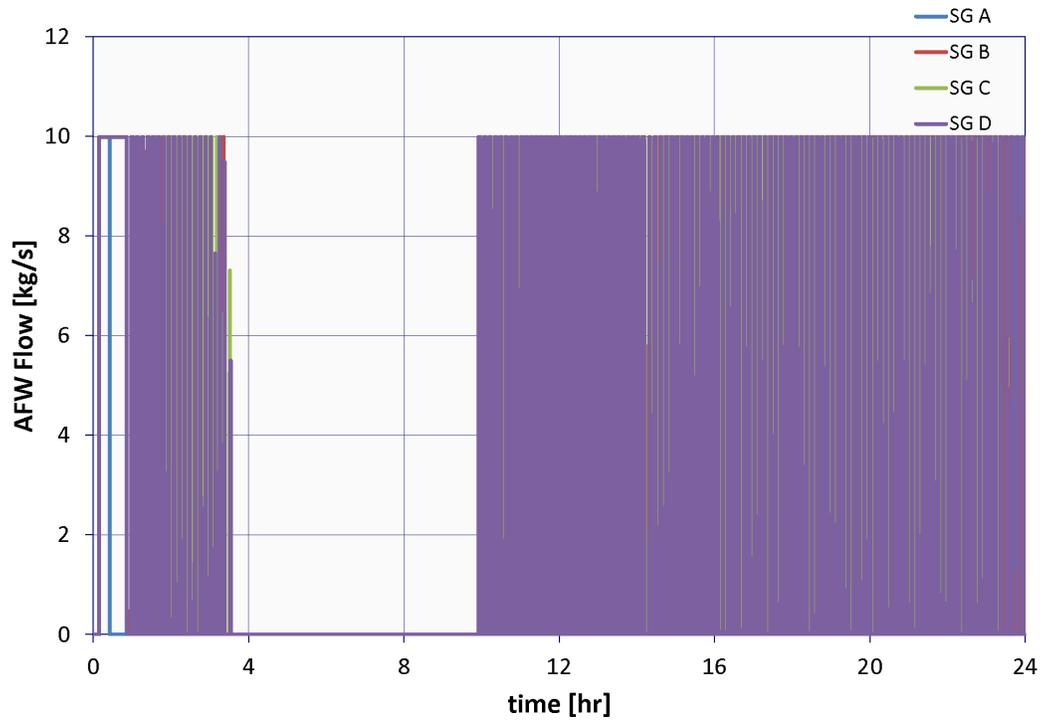
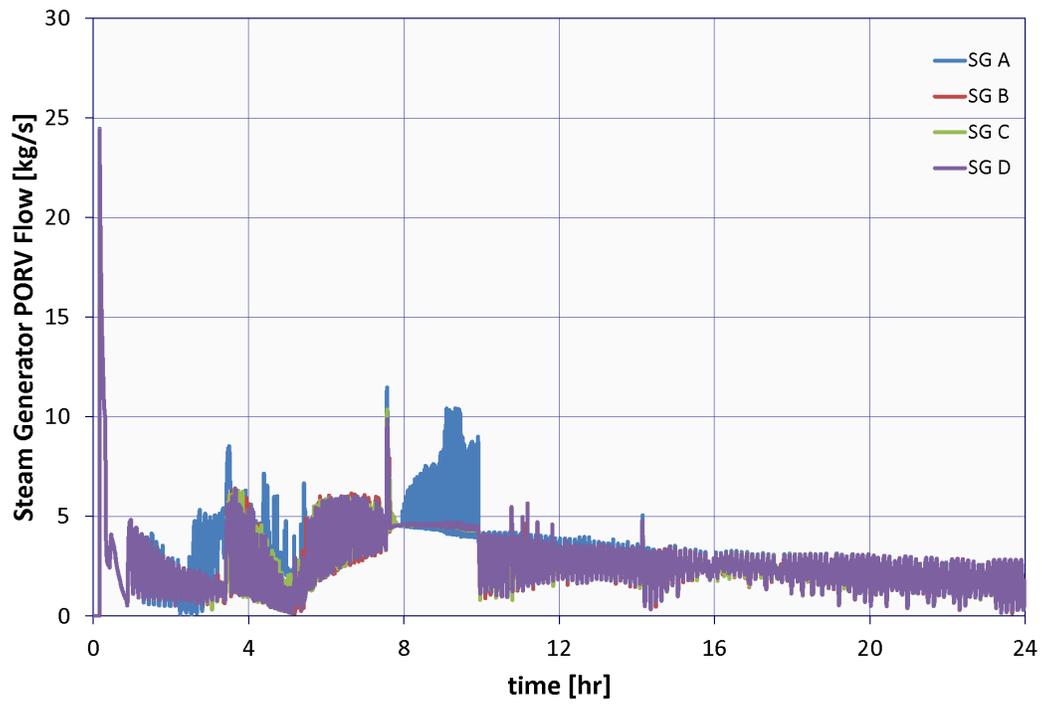


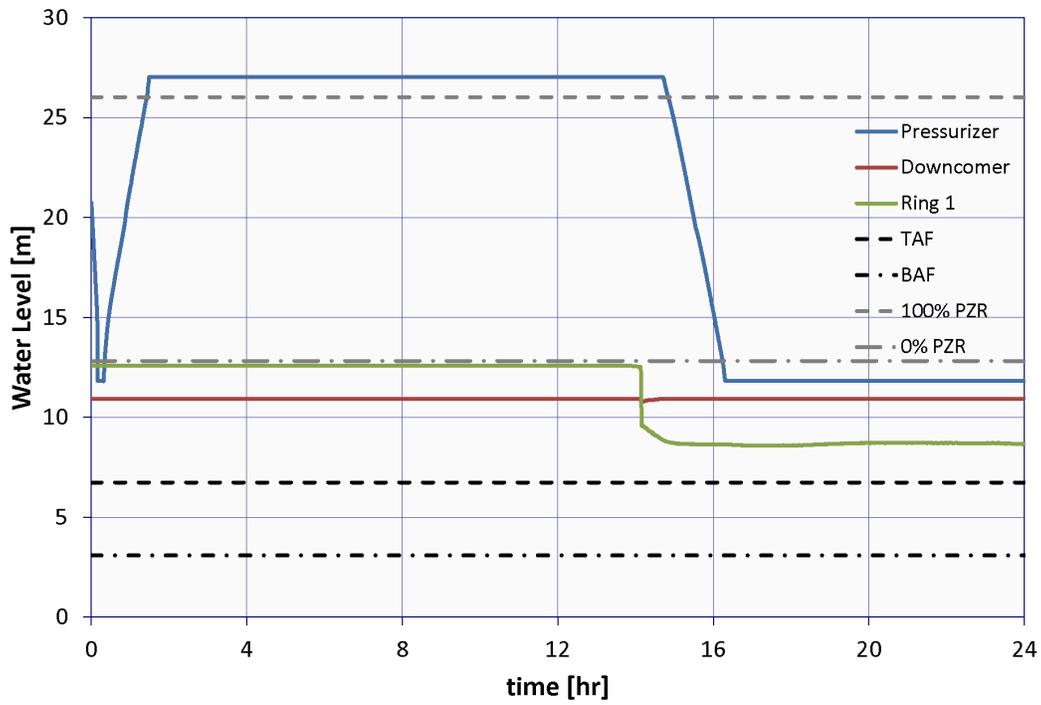
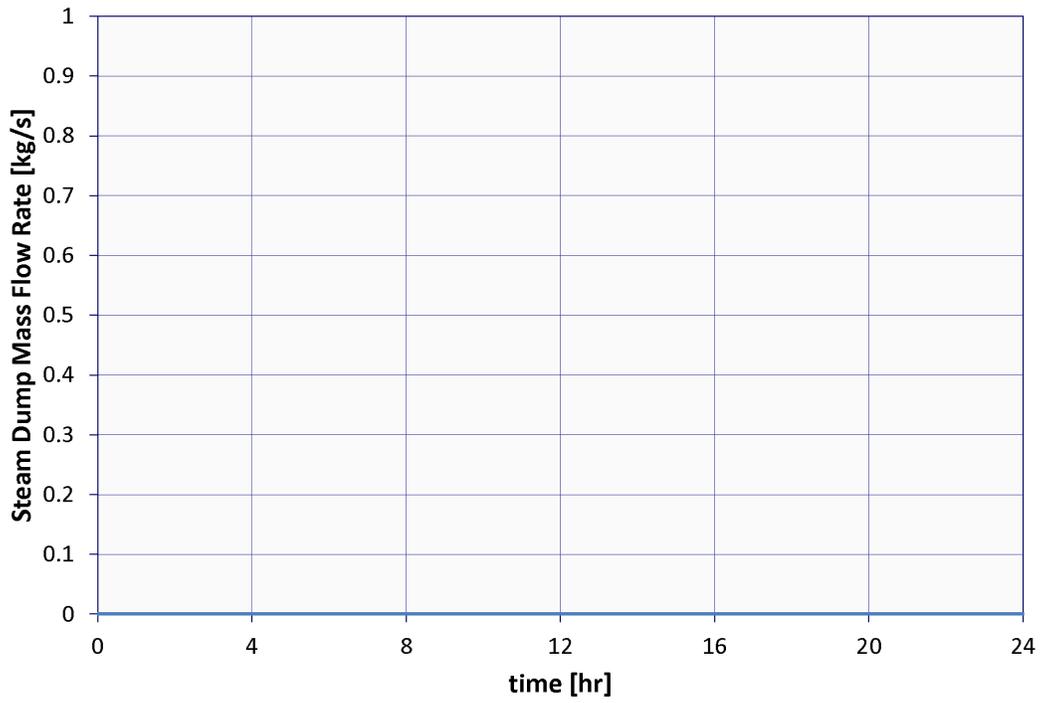
**D.1.3 Case 3: 0.5 Tube, Min ECCS, No Steam Dumps**

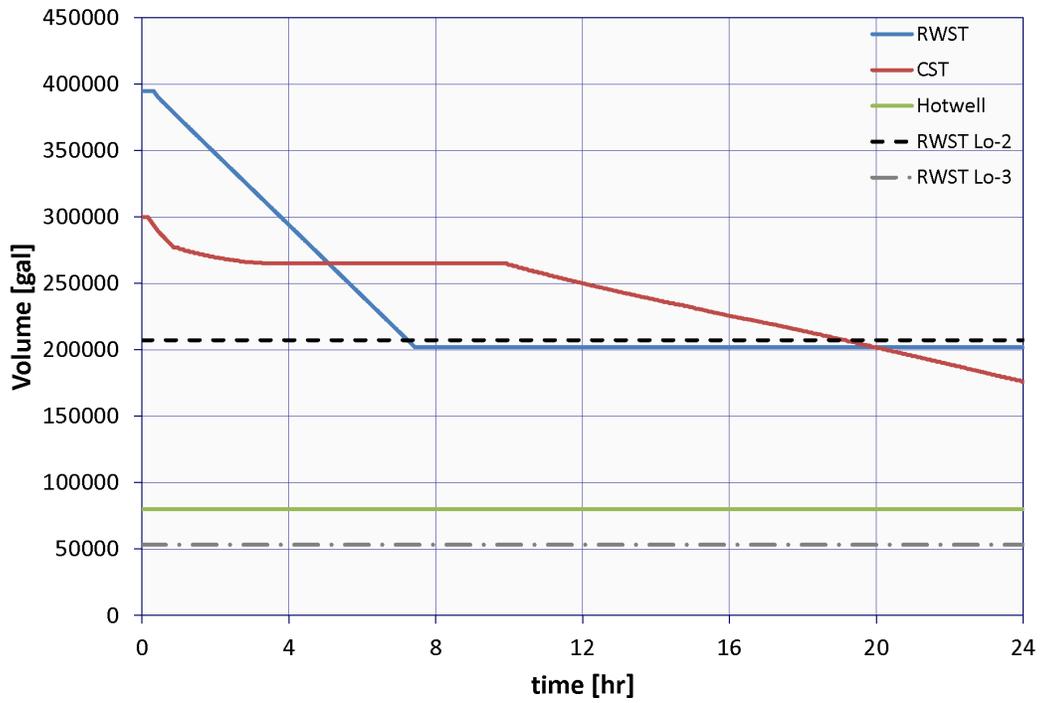
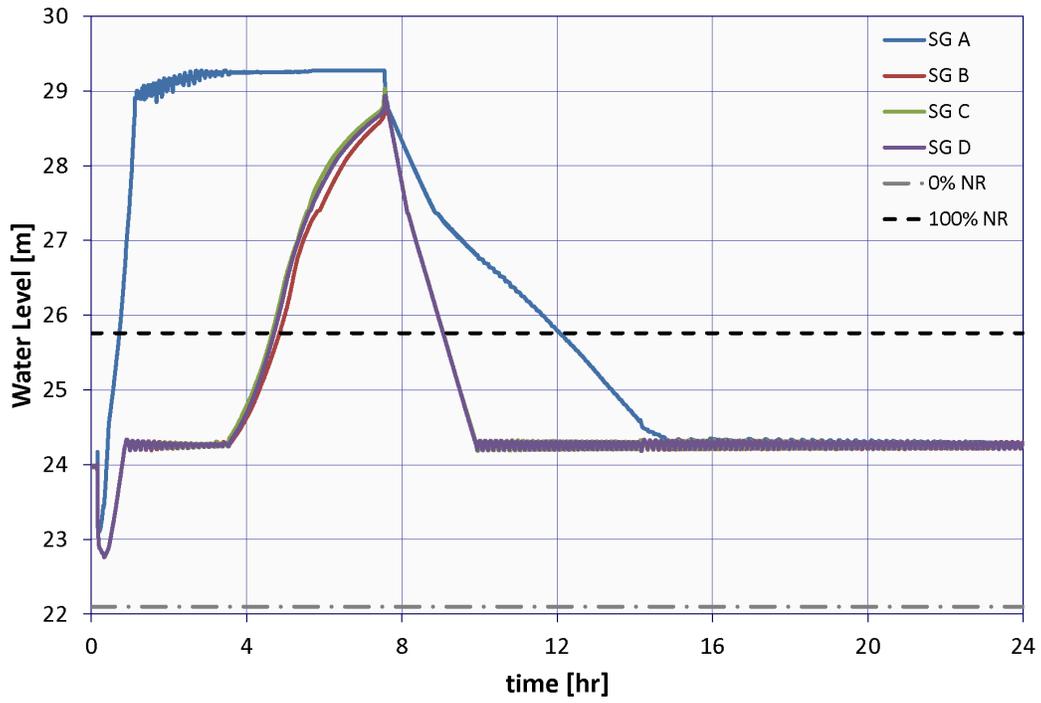




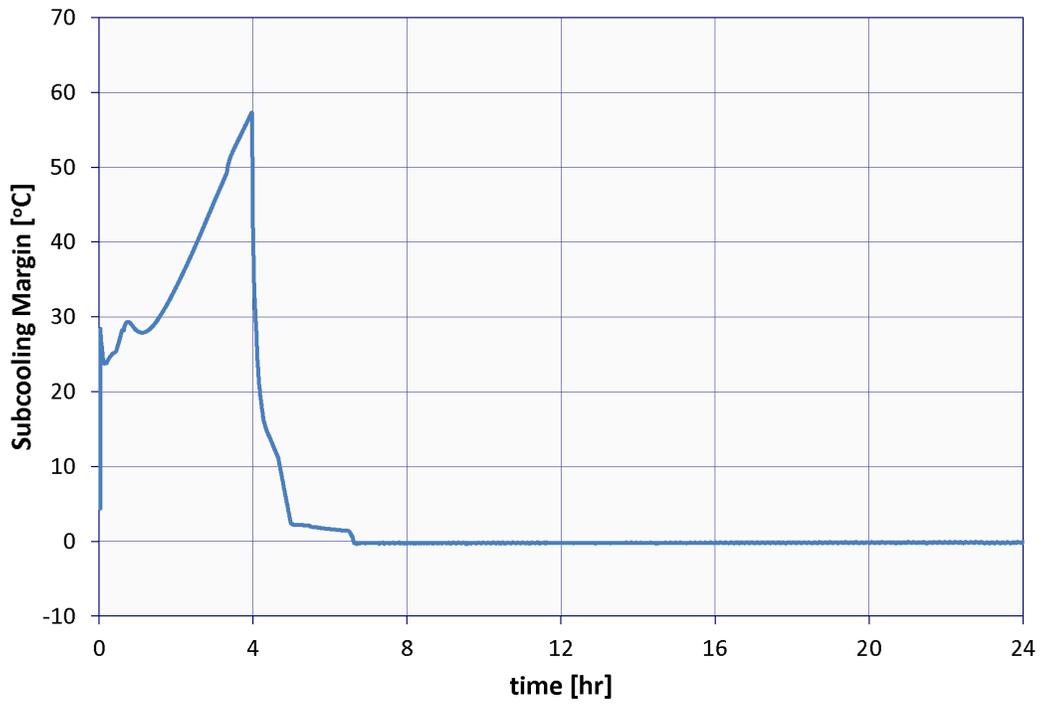
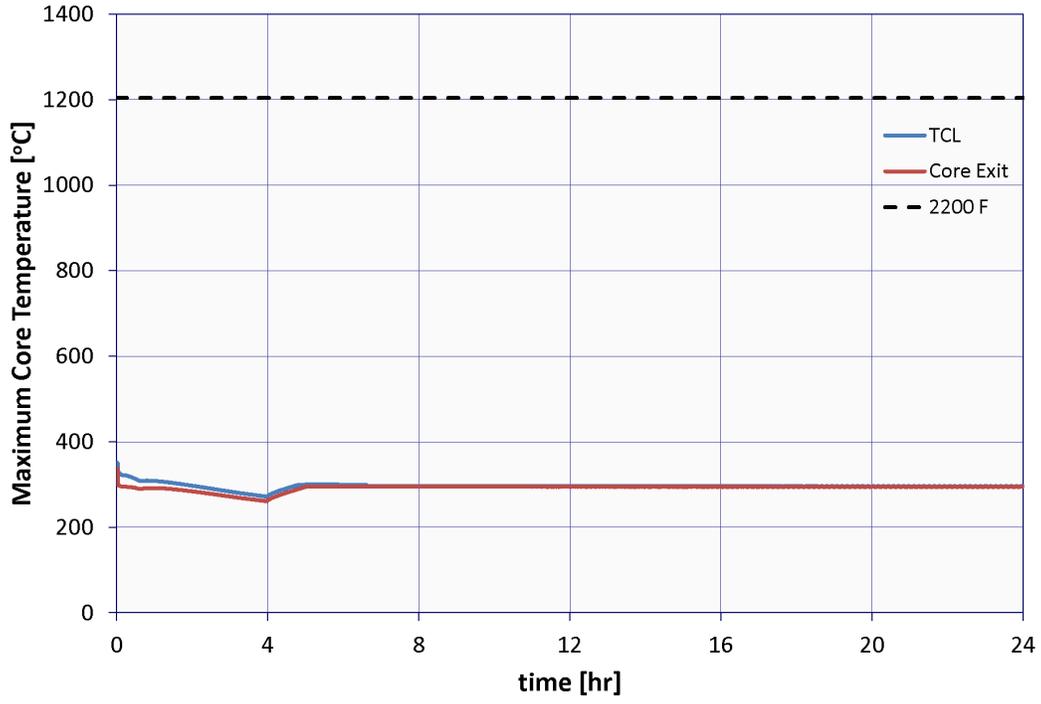


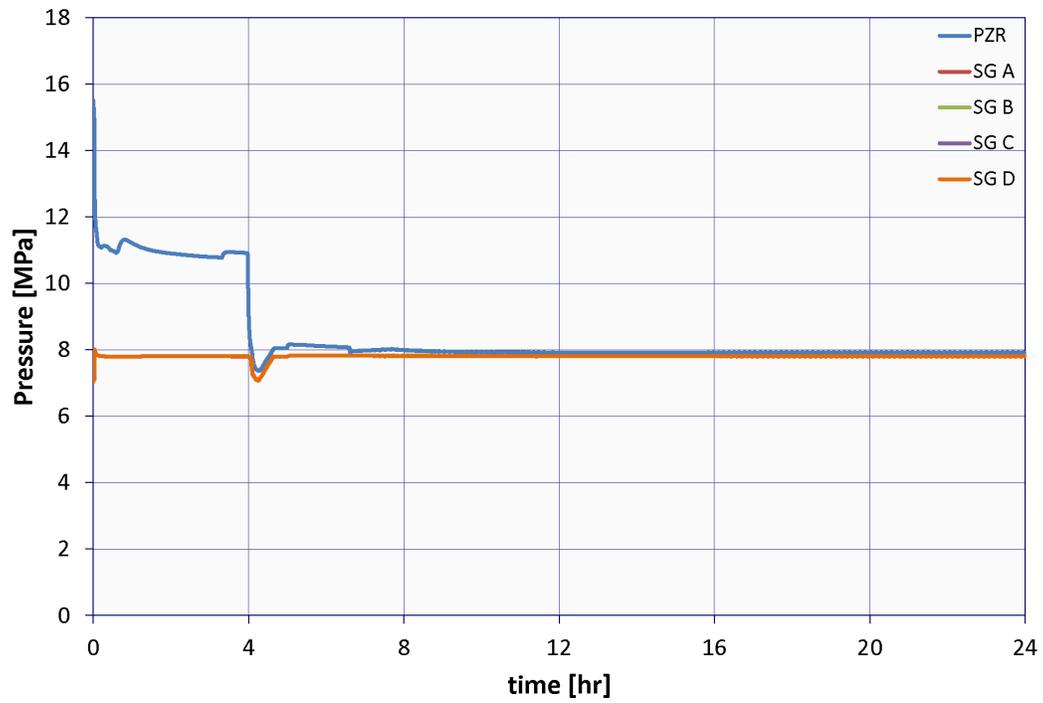
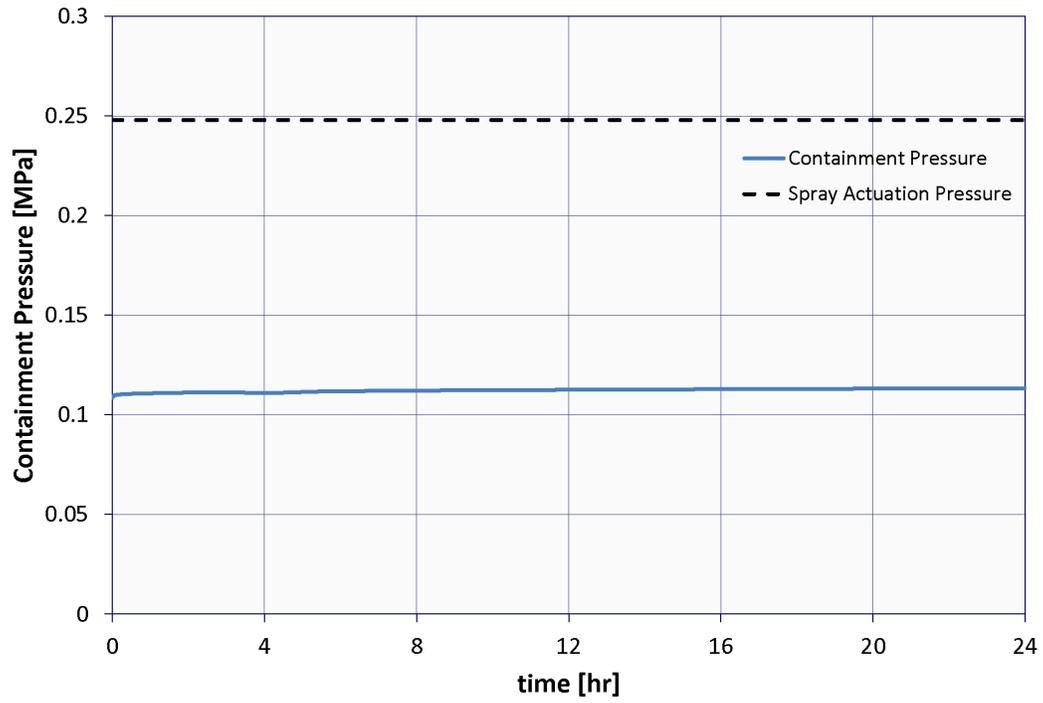


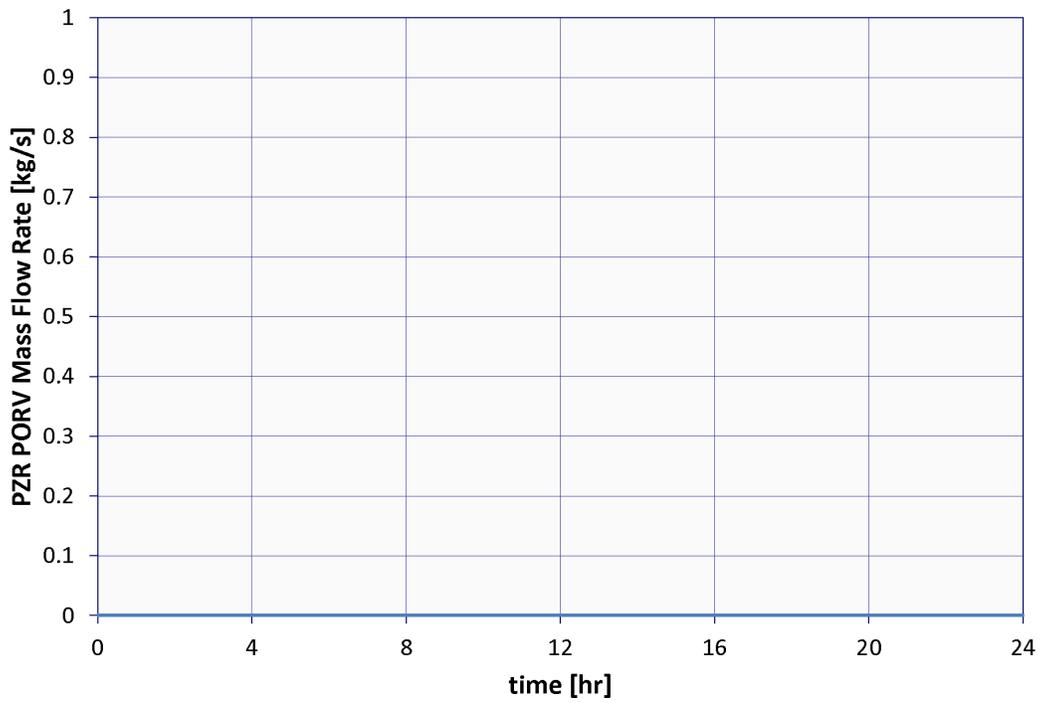
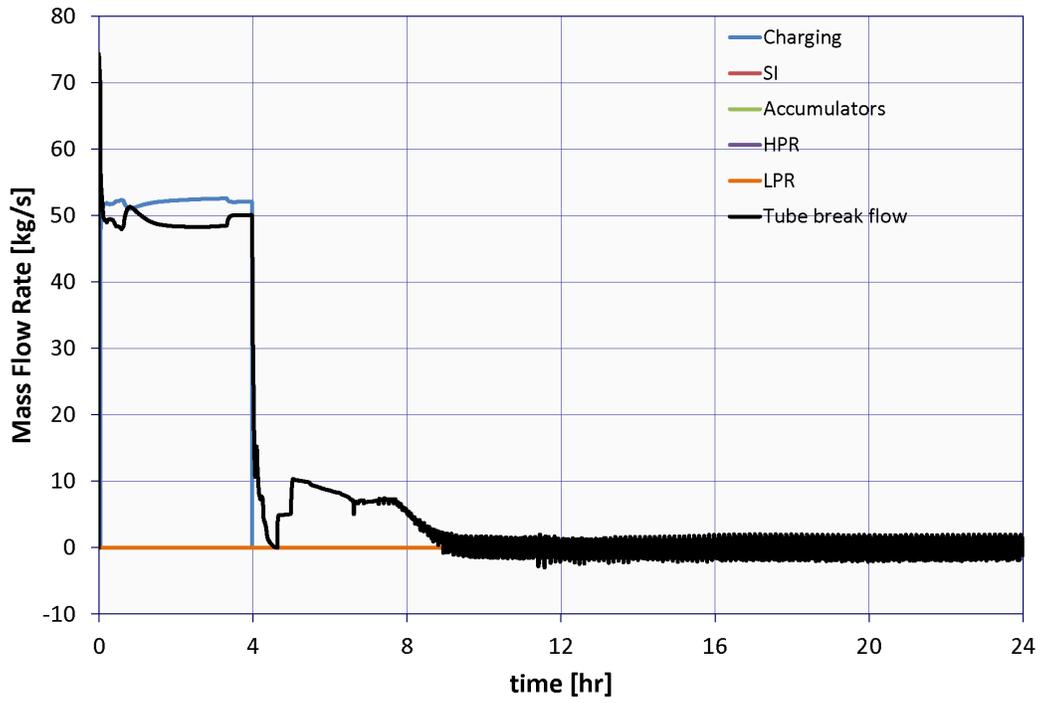


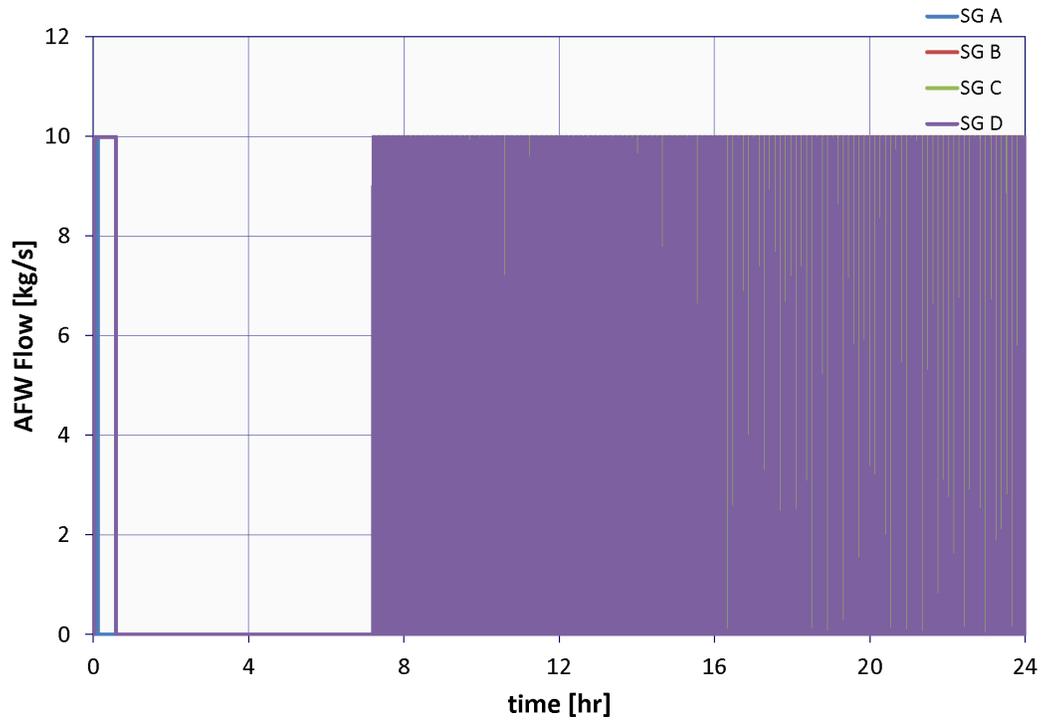
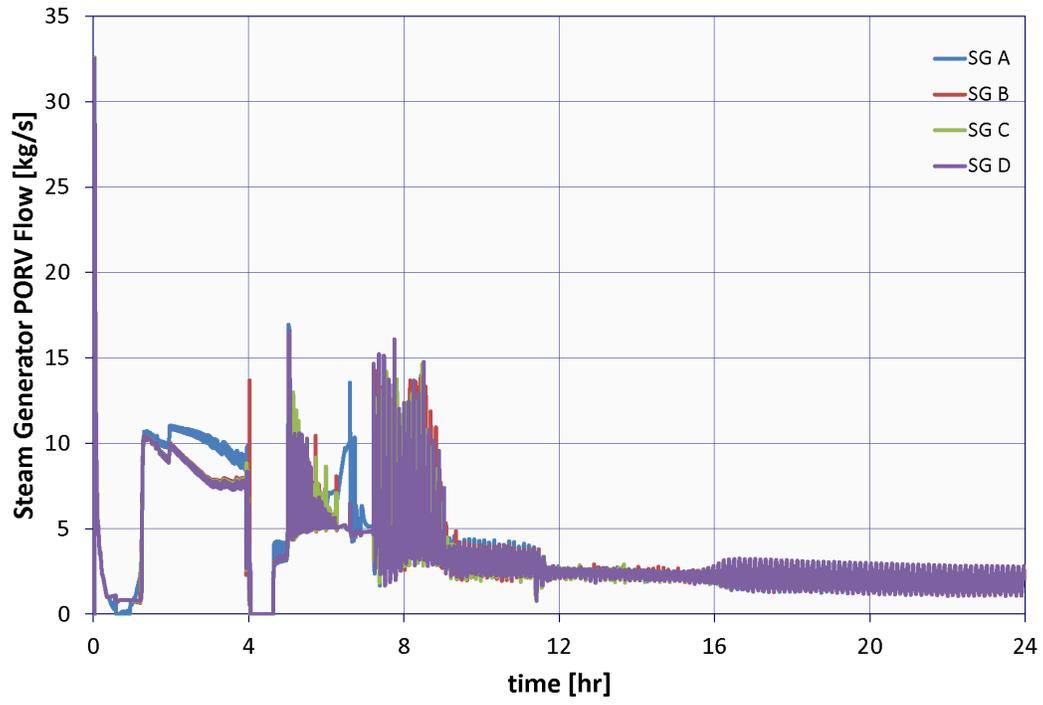


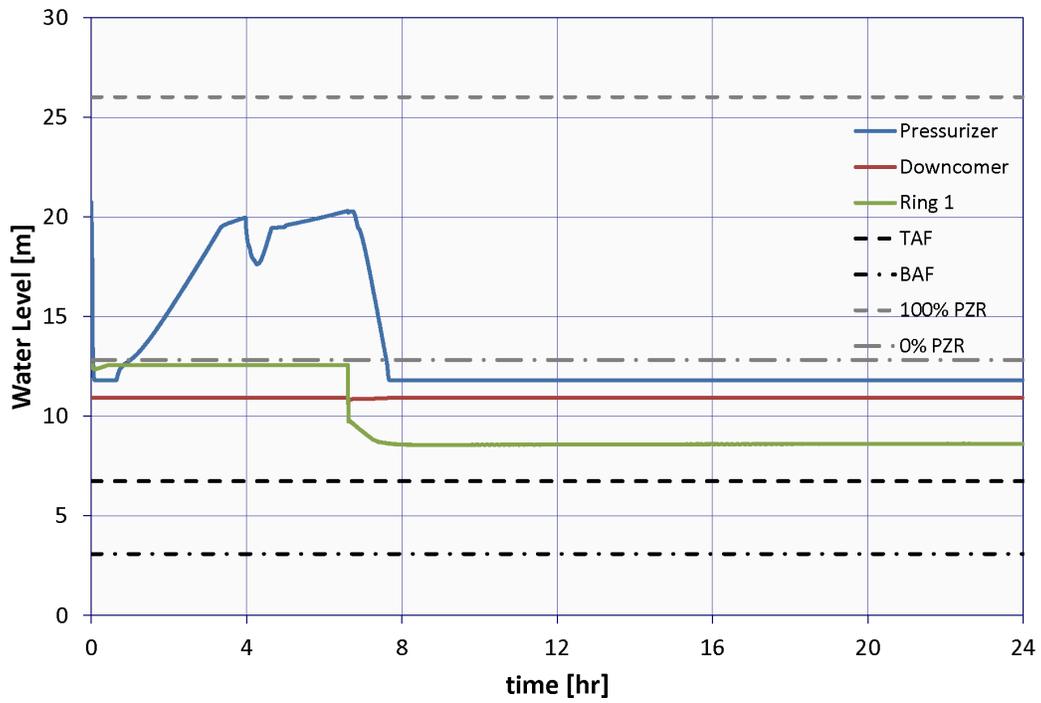
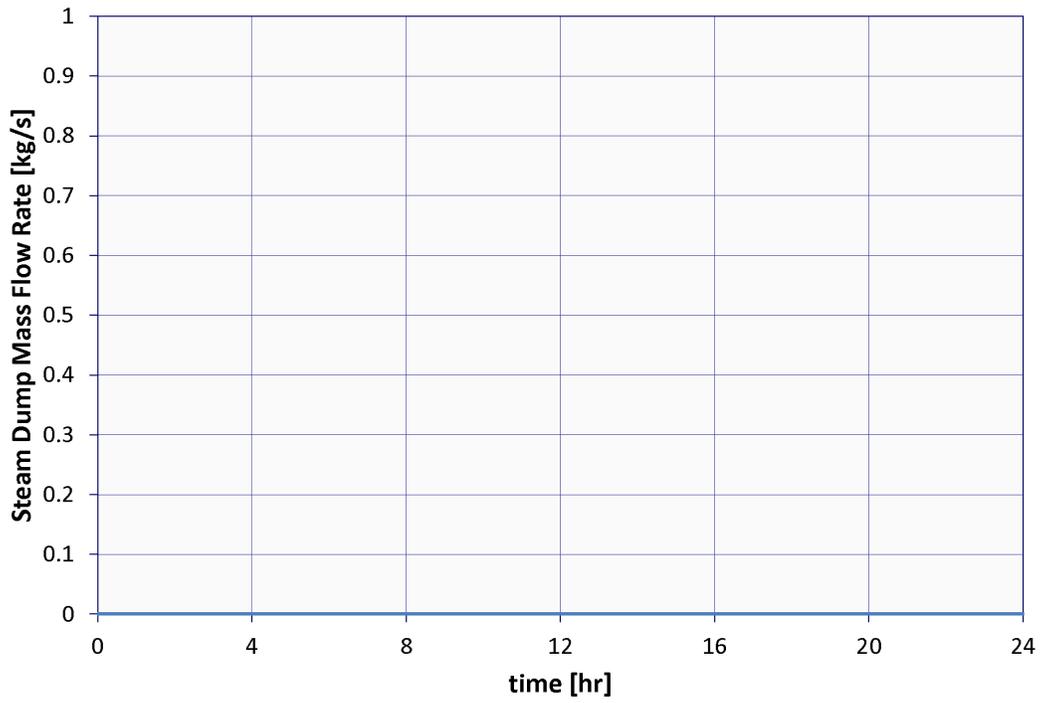
D.1.4 Case 4: 2 Tubes, Max ECCS, No Steam Dumps

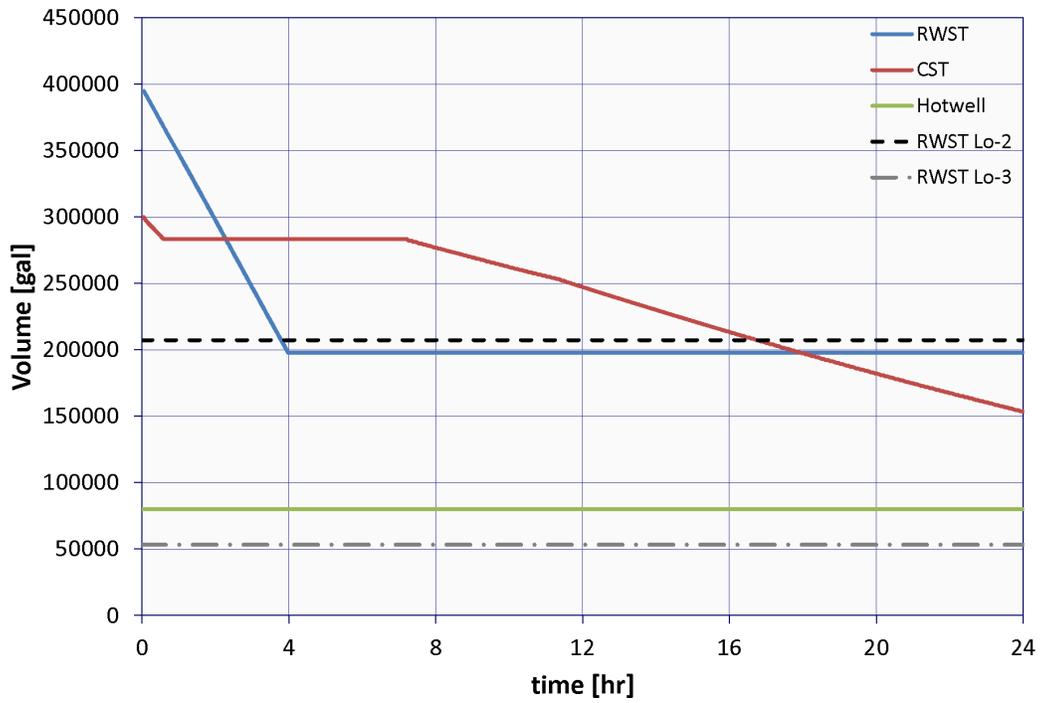
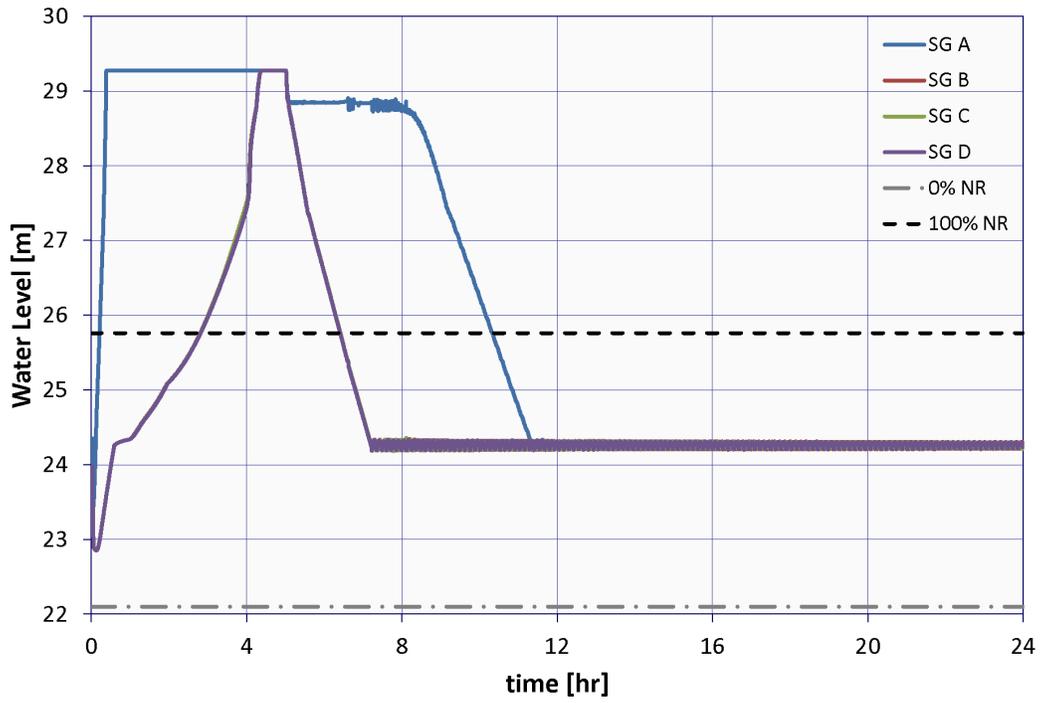




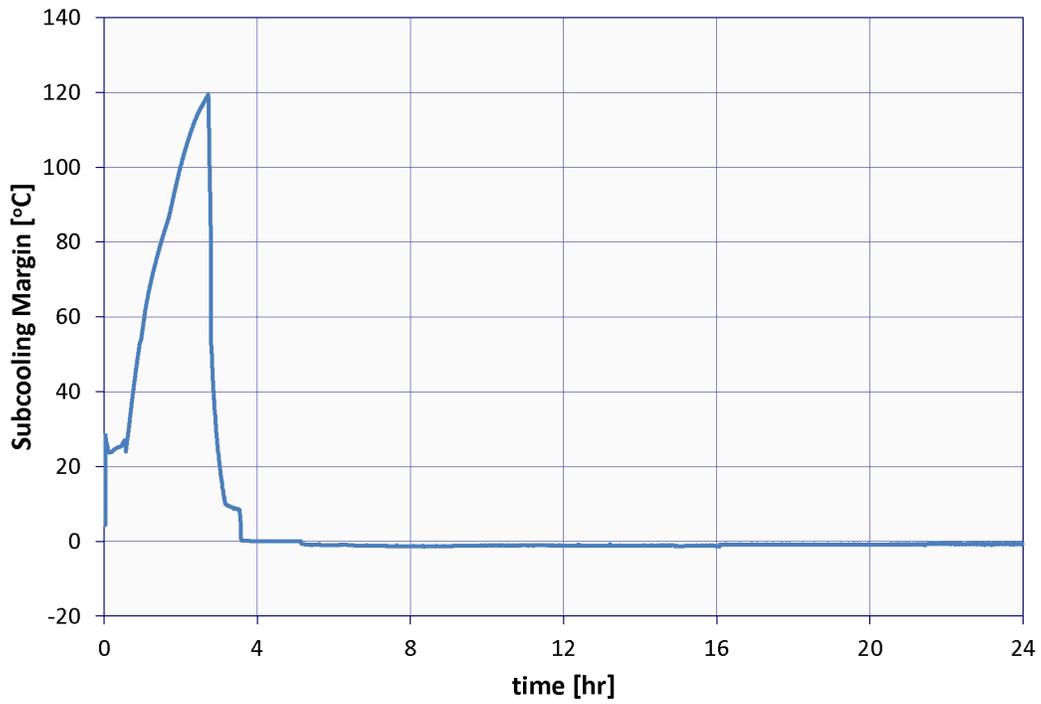
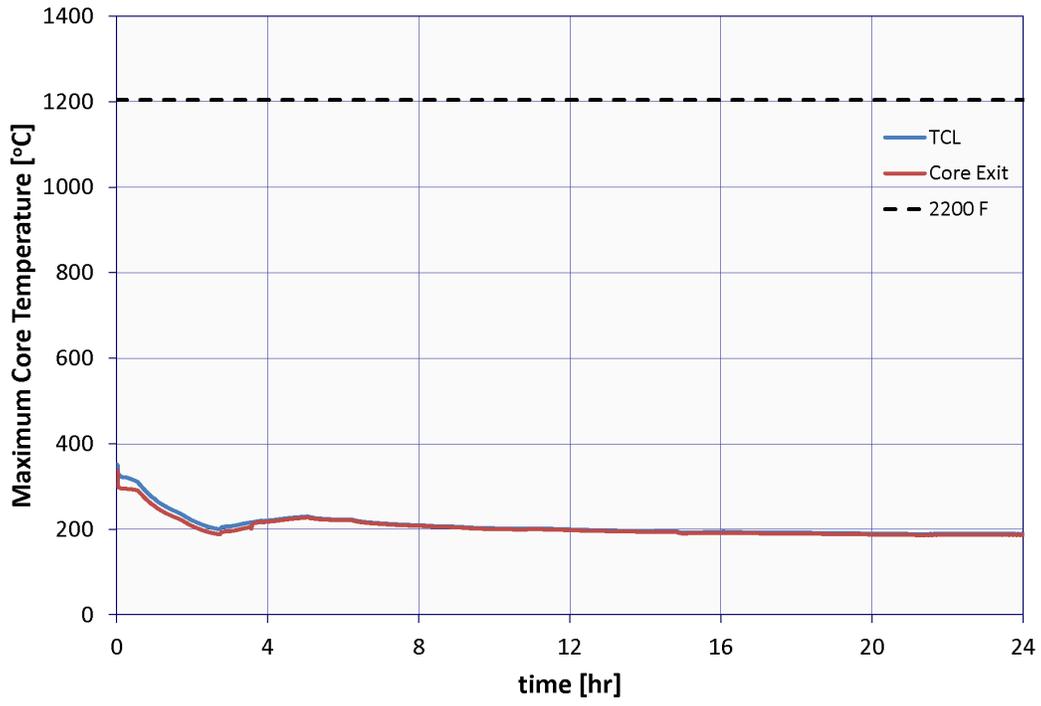


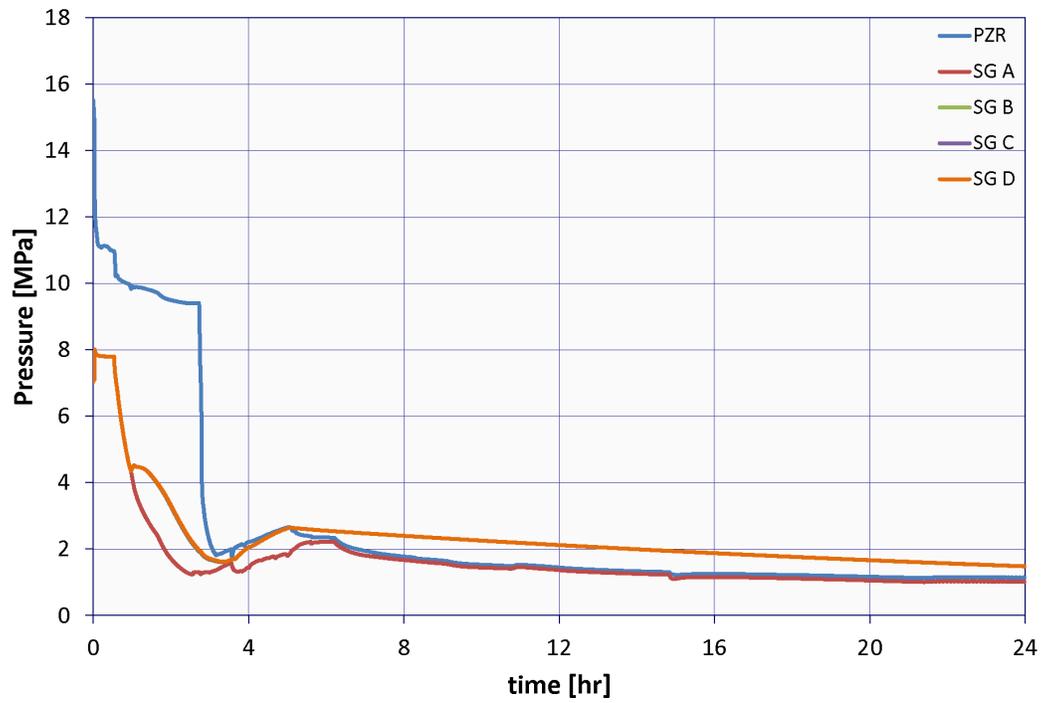
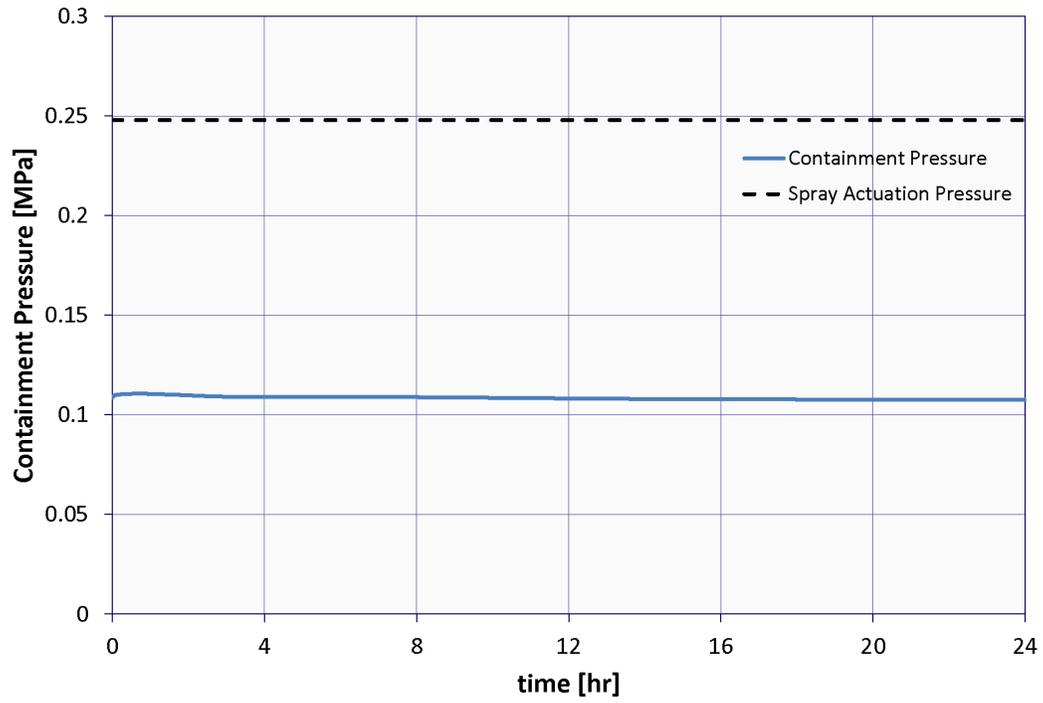


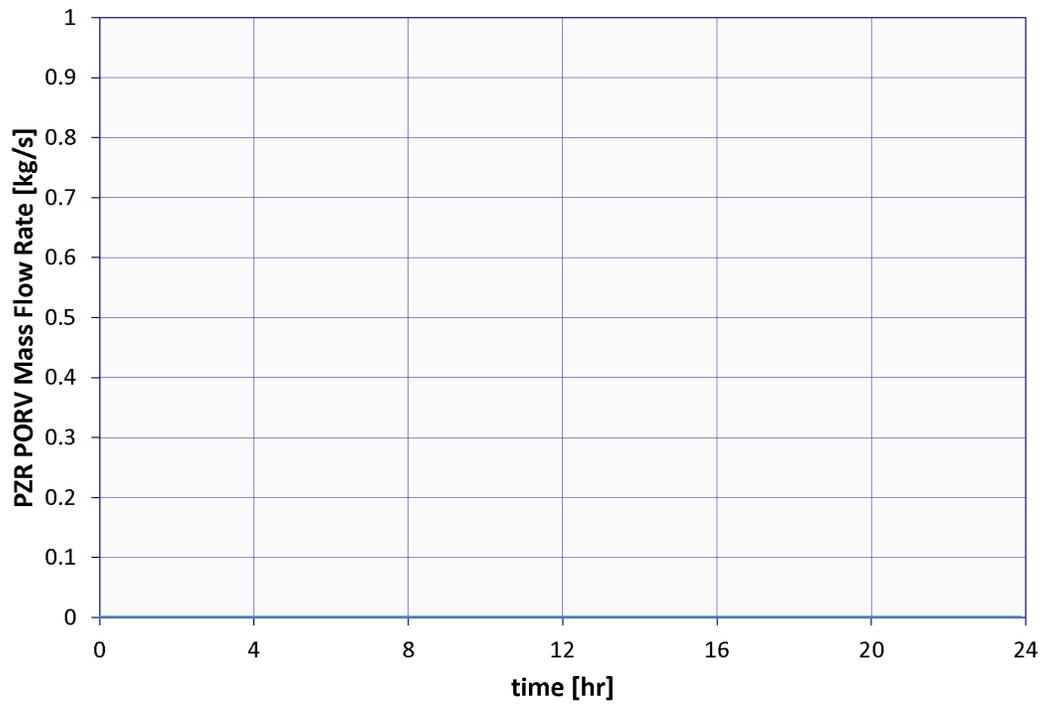
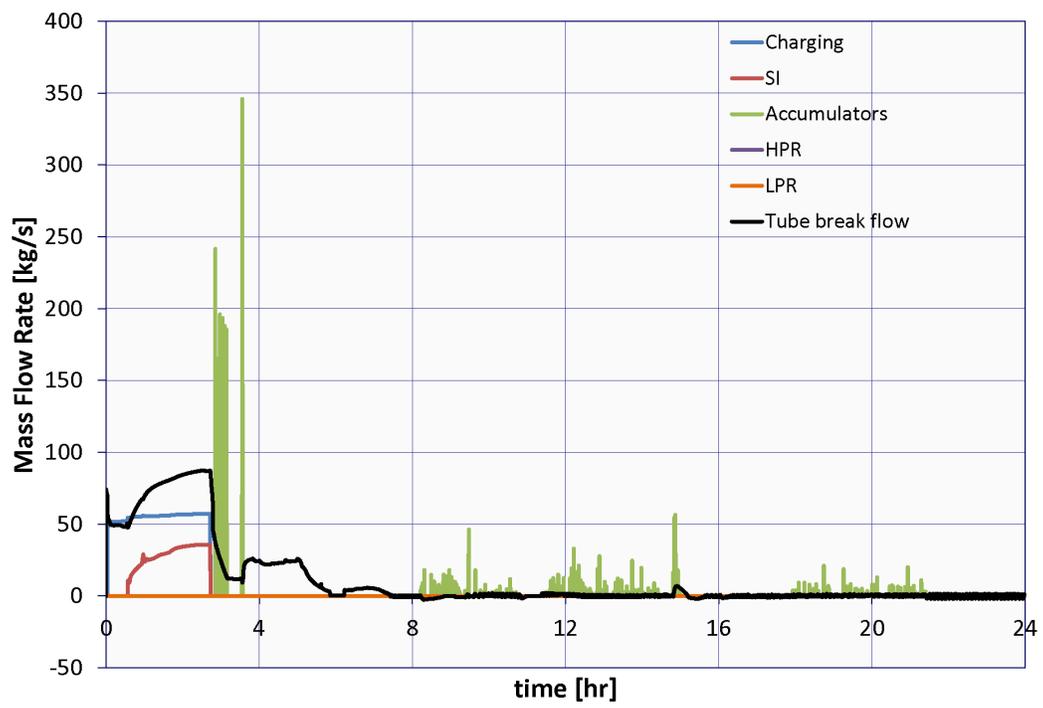


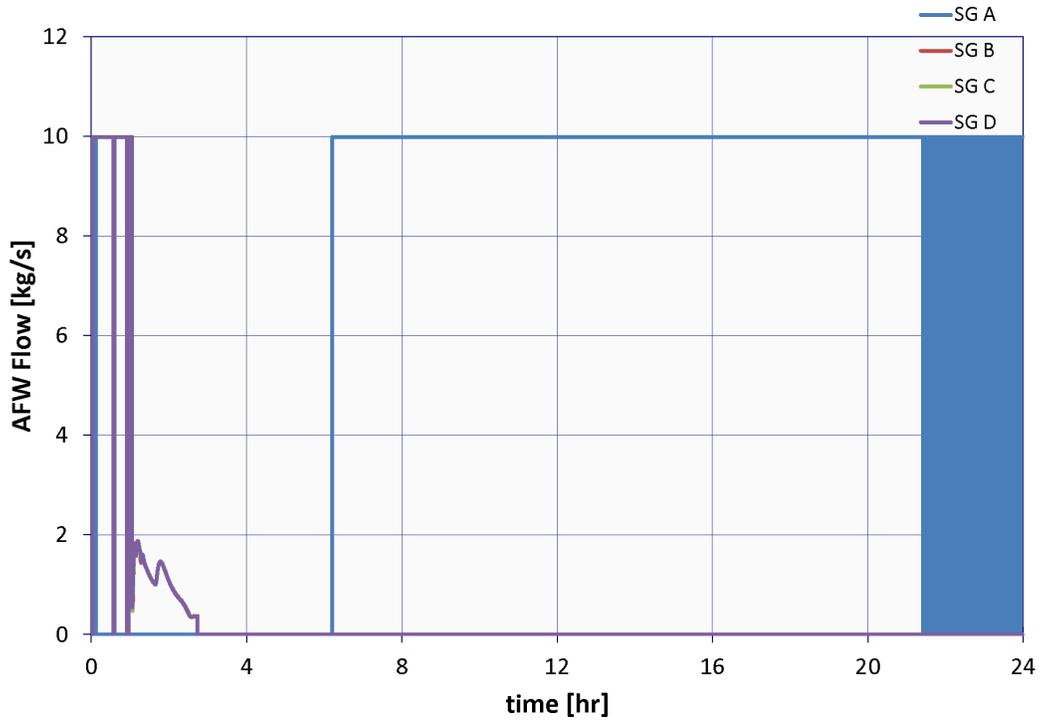
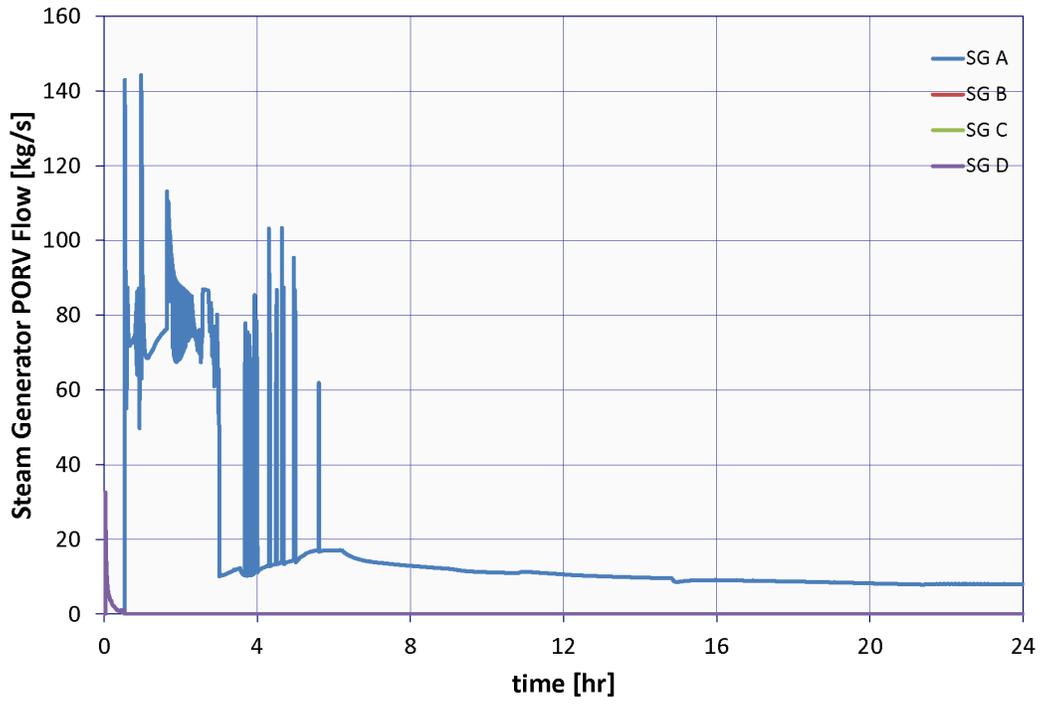


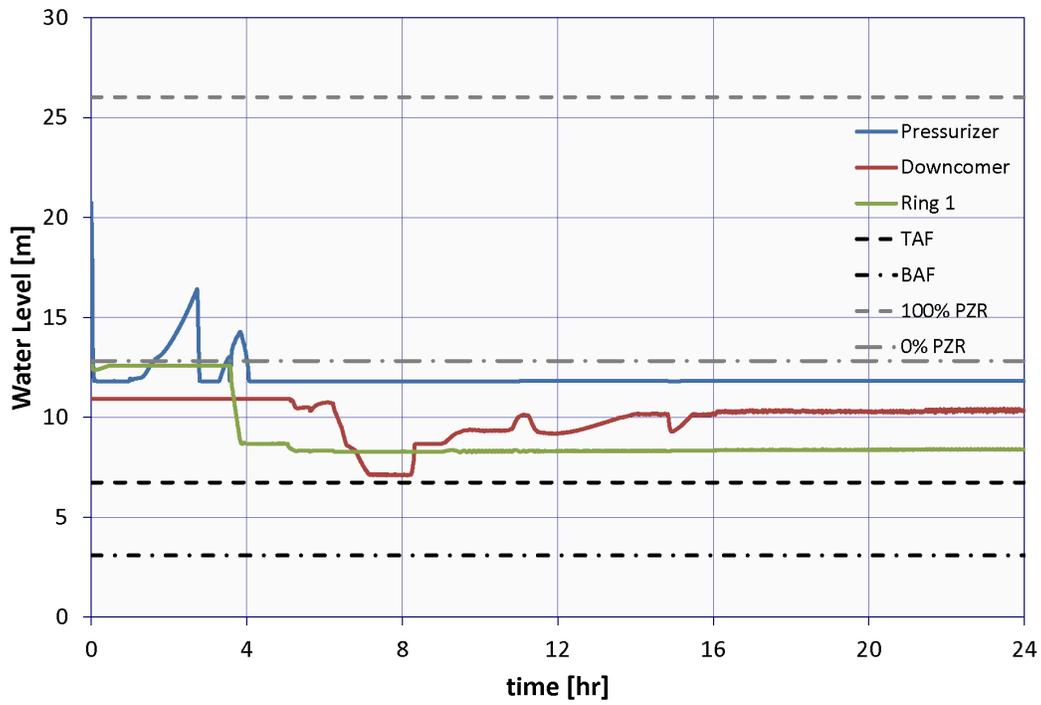
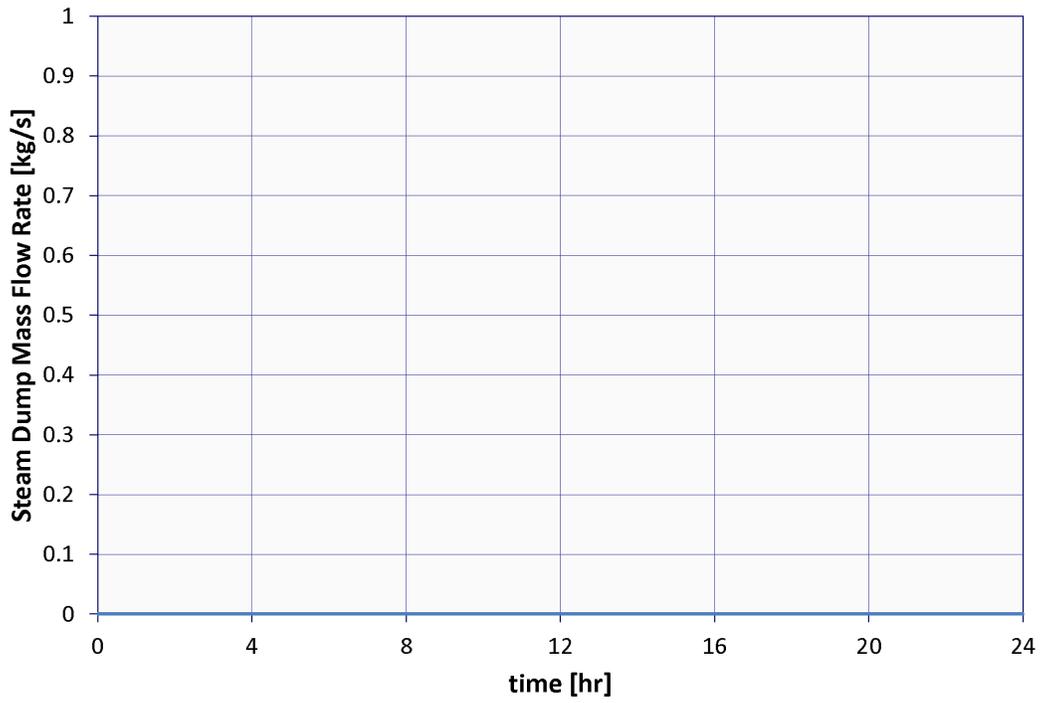
D.1.4.1 Case 4a: 2 Tubes, Max ECCS, SG PORV Sticks Open

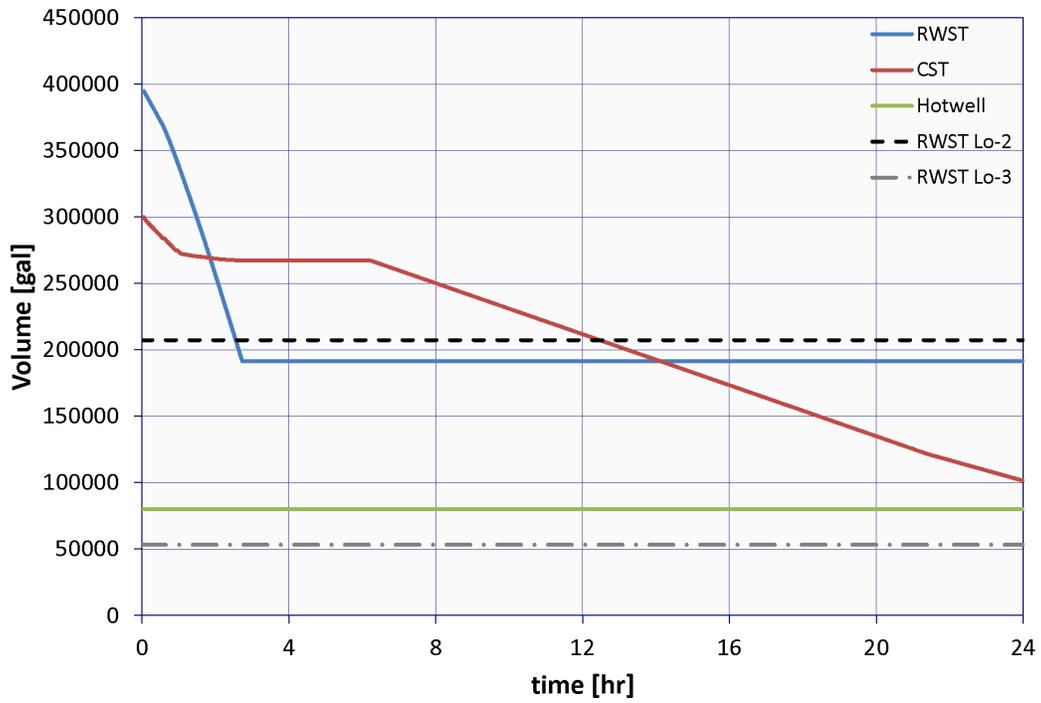
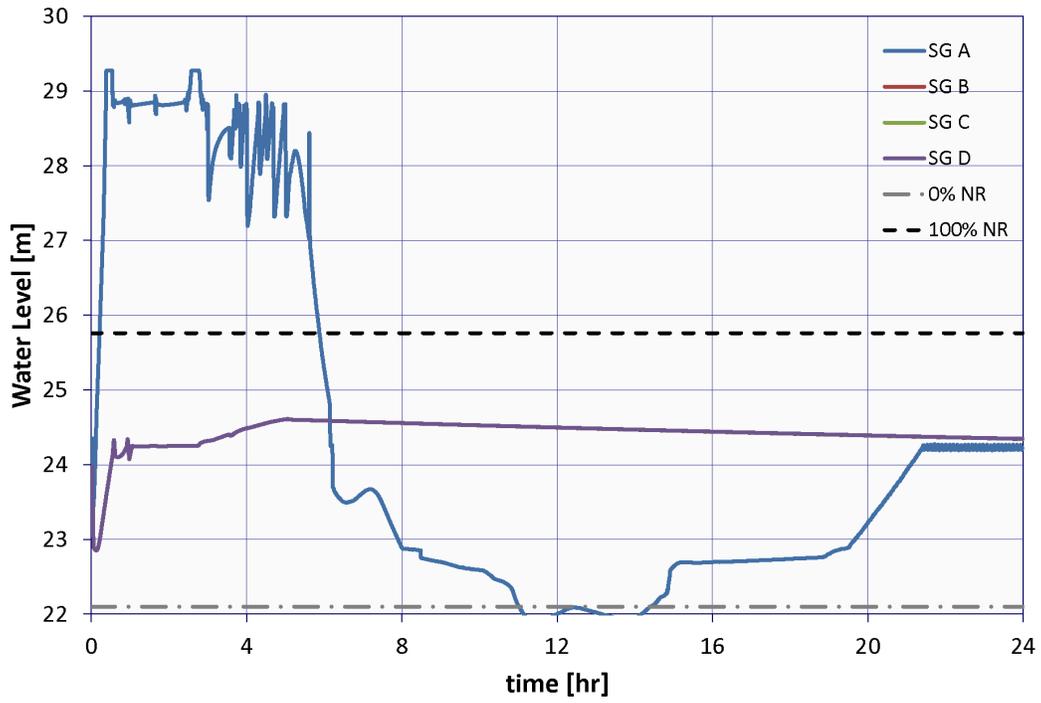




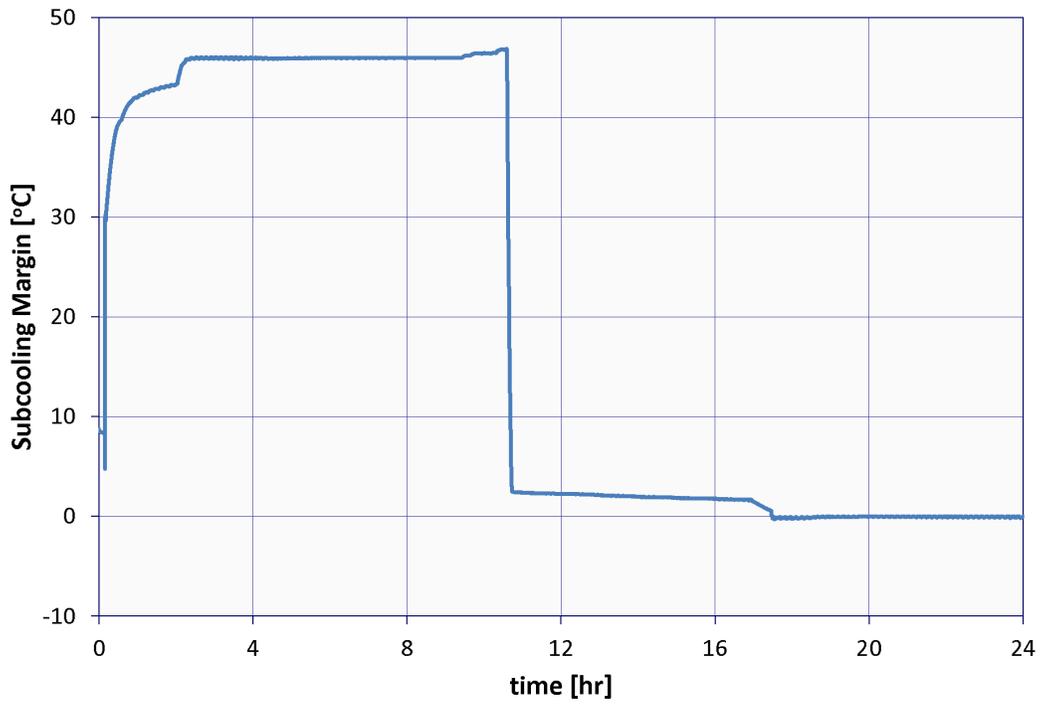
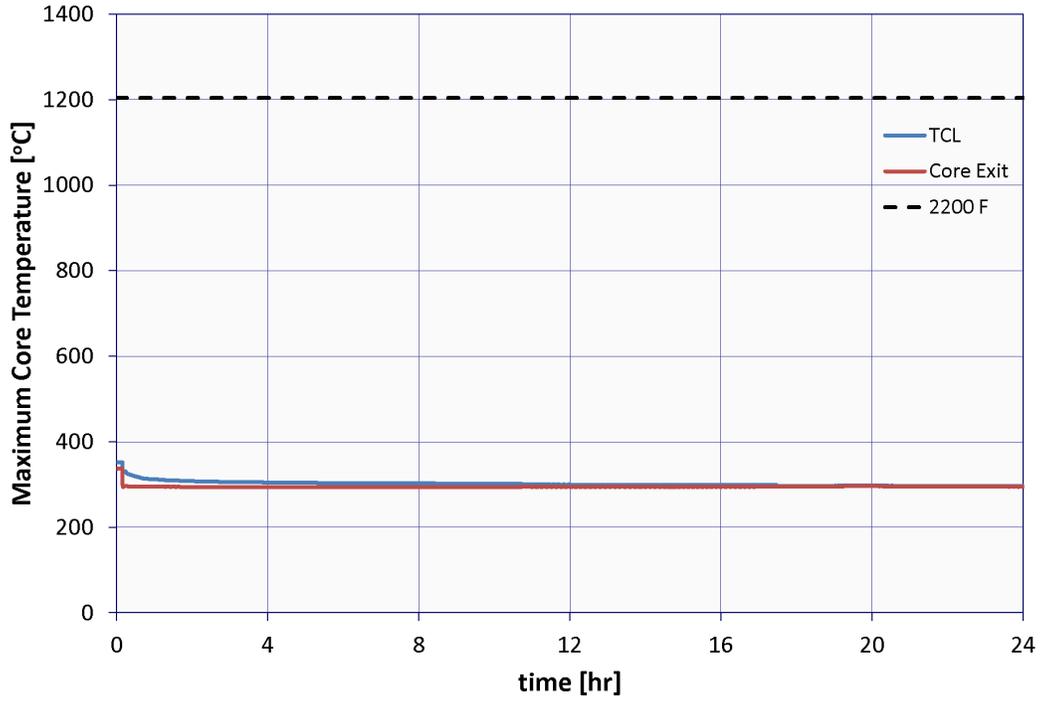


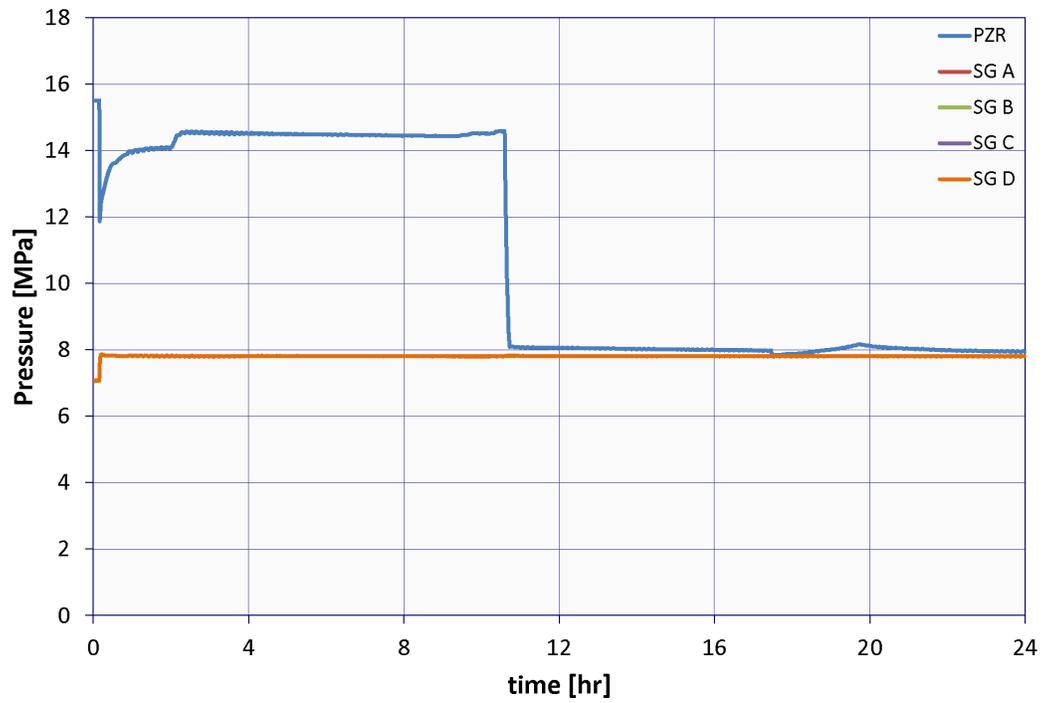
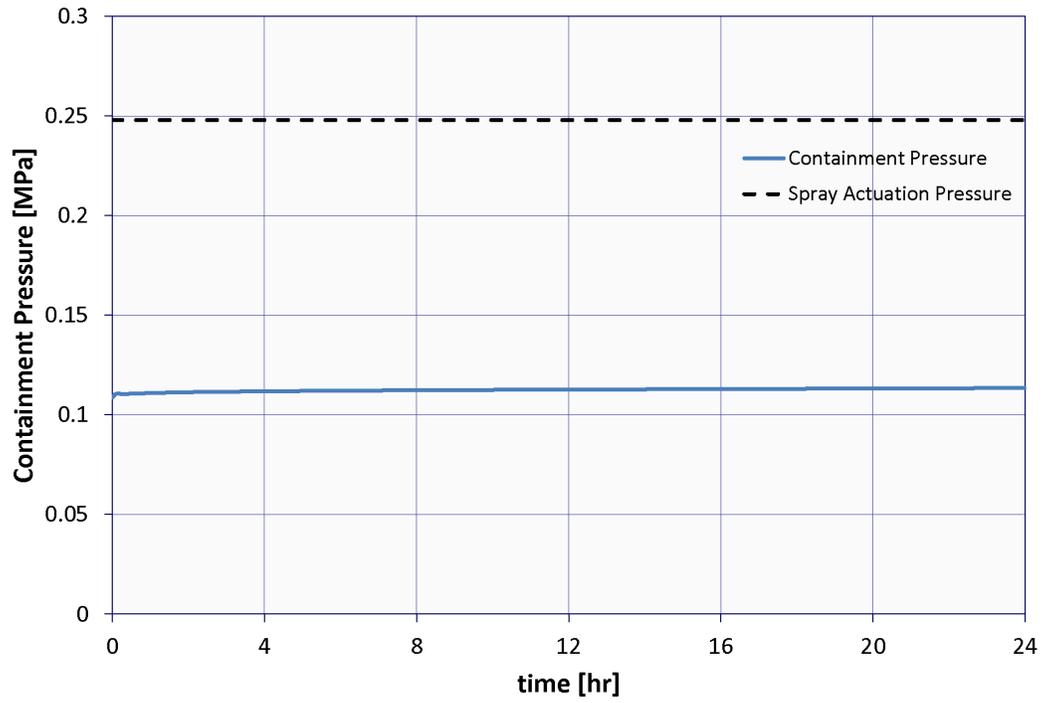


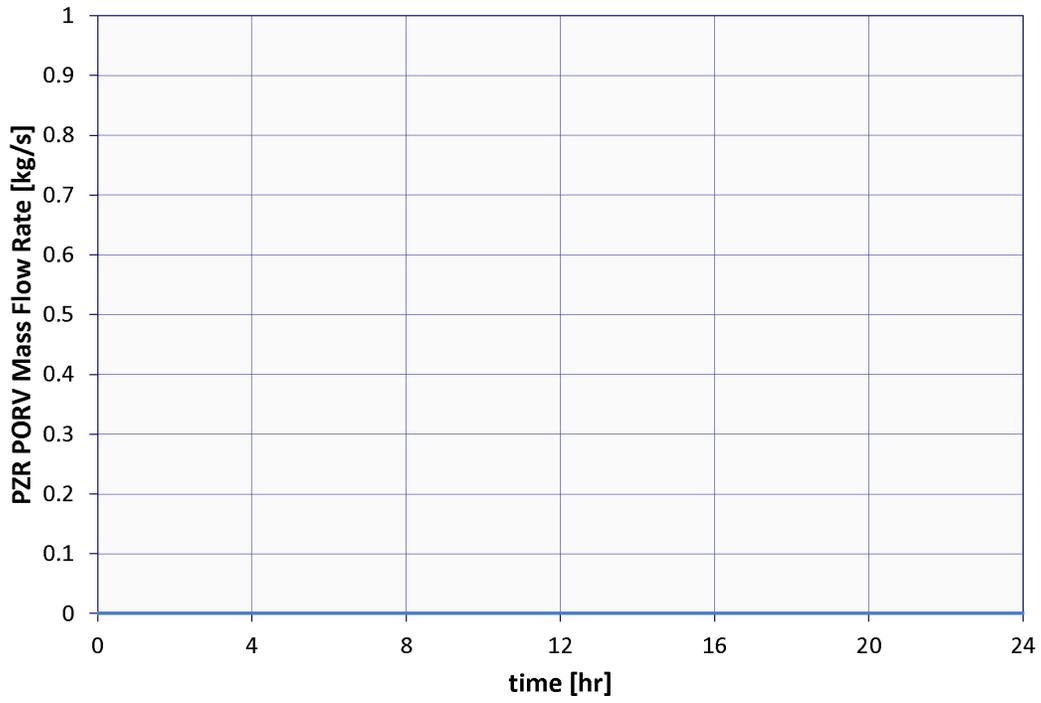
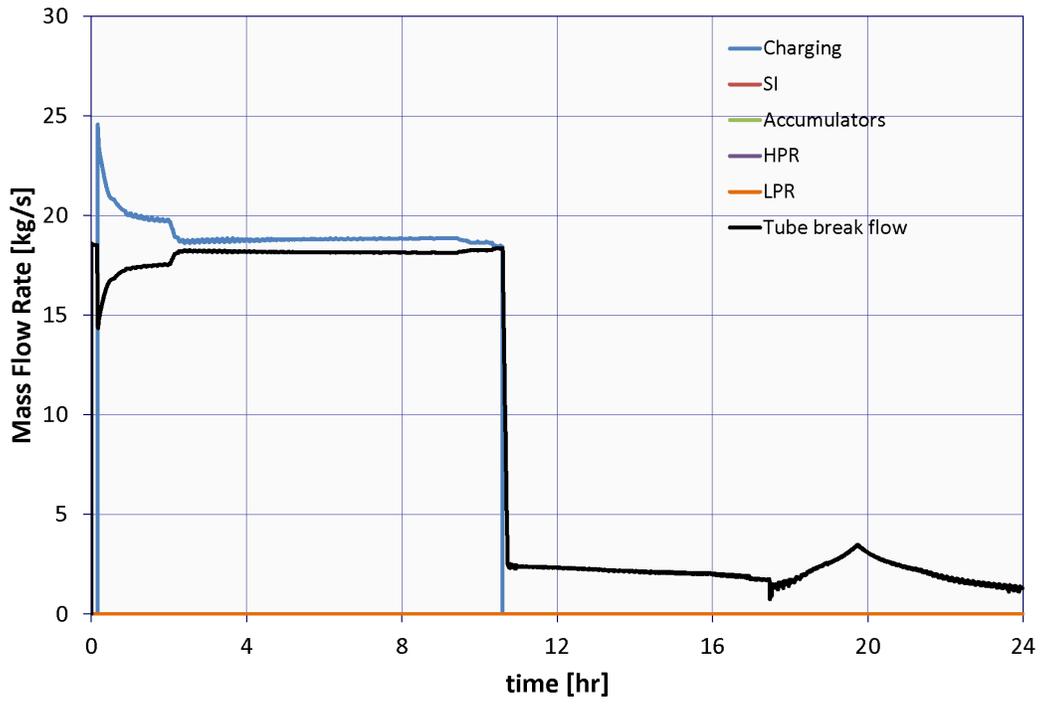


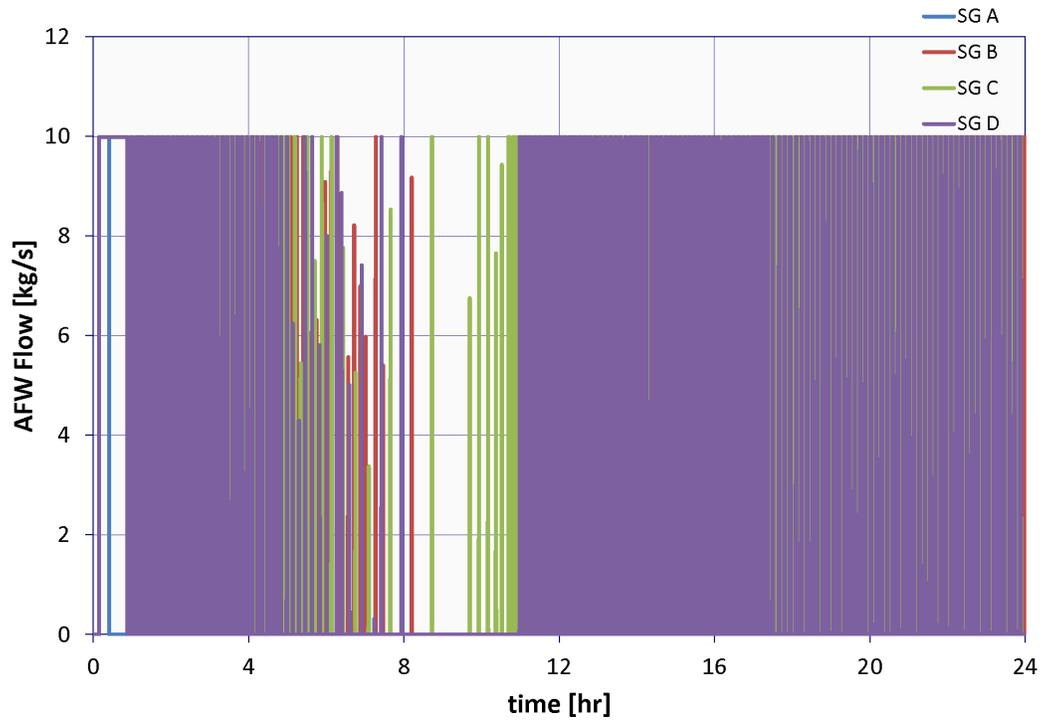
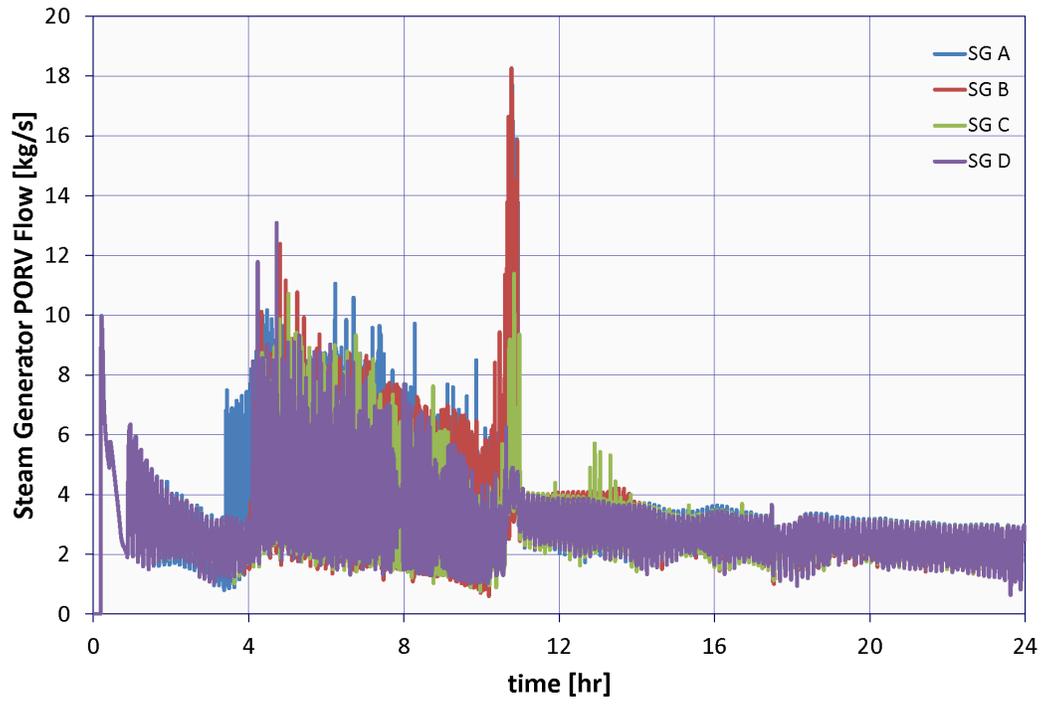


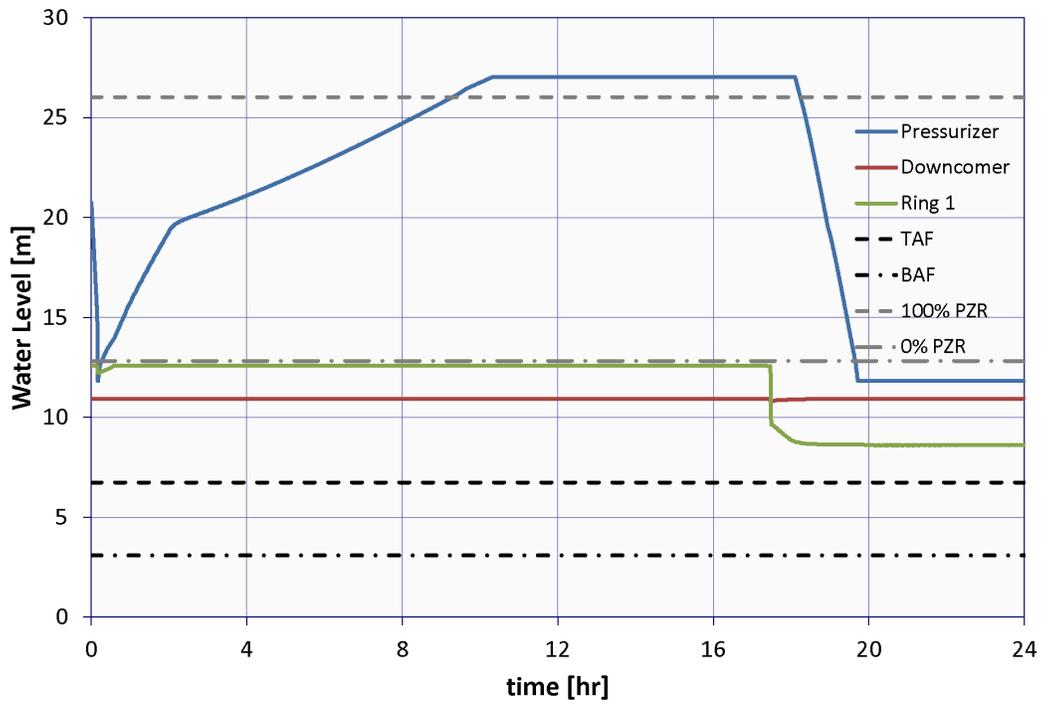
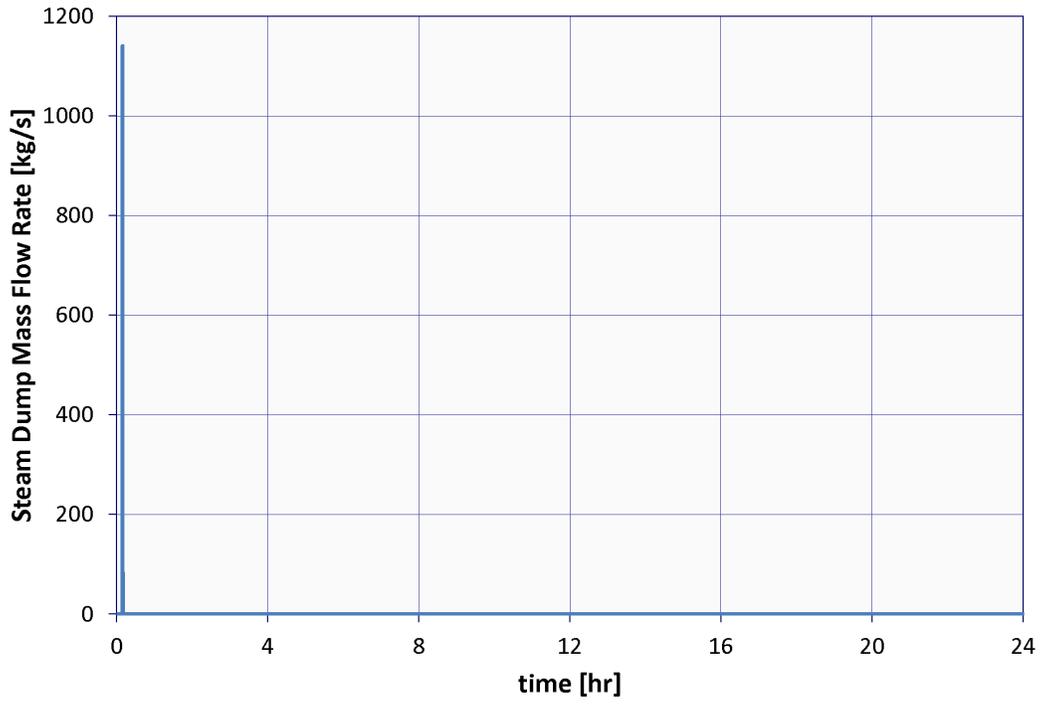
**D.1.5 Case 5: 0.5 Tube, Min ECCS, Steam Dumps Available**

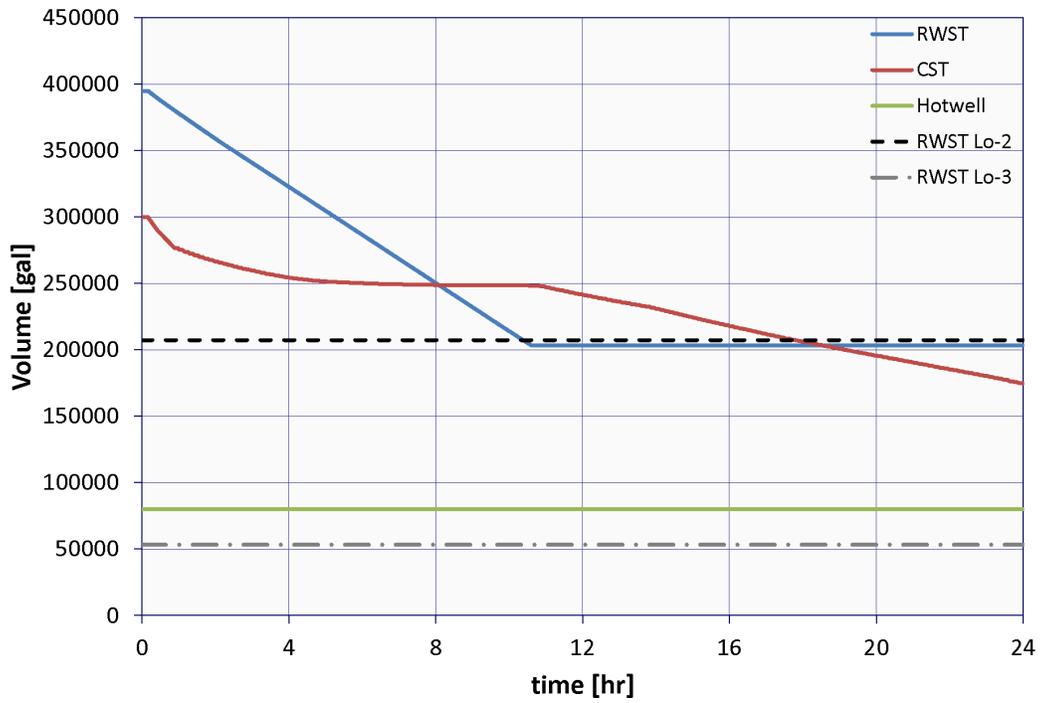
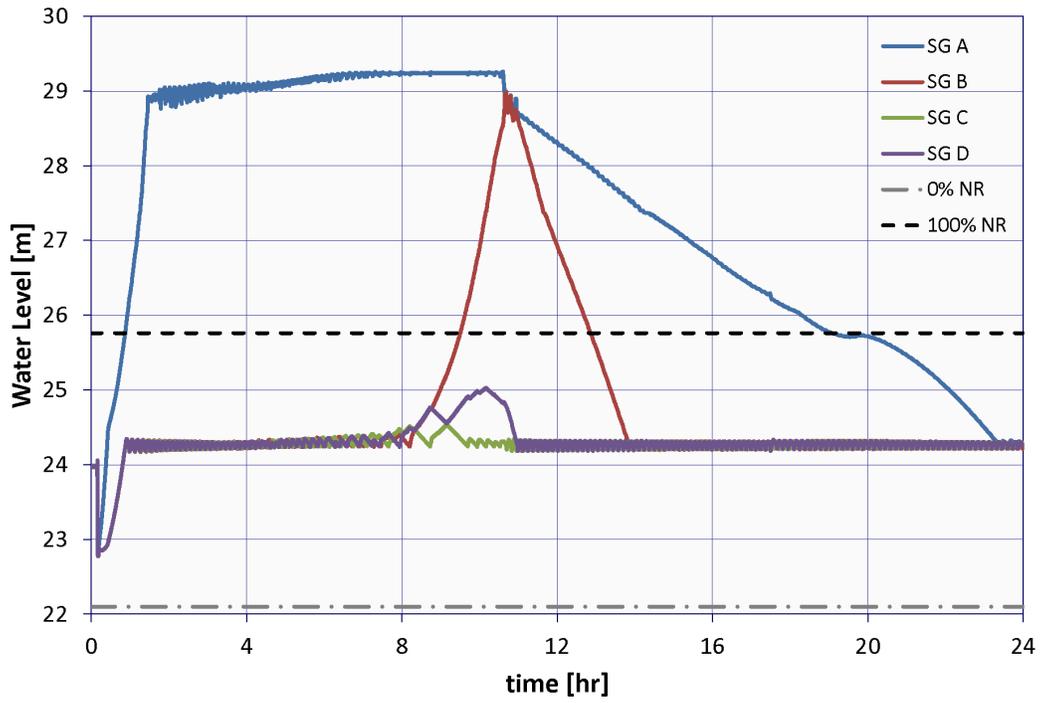




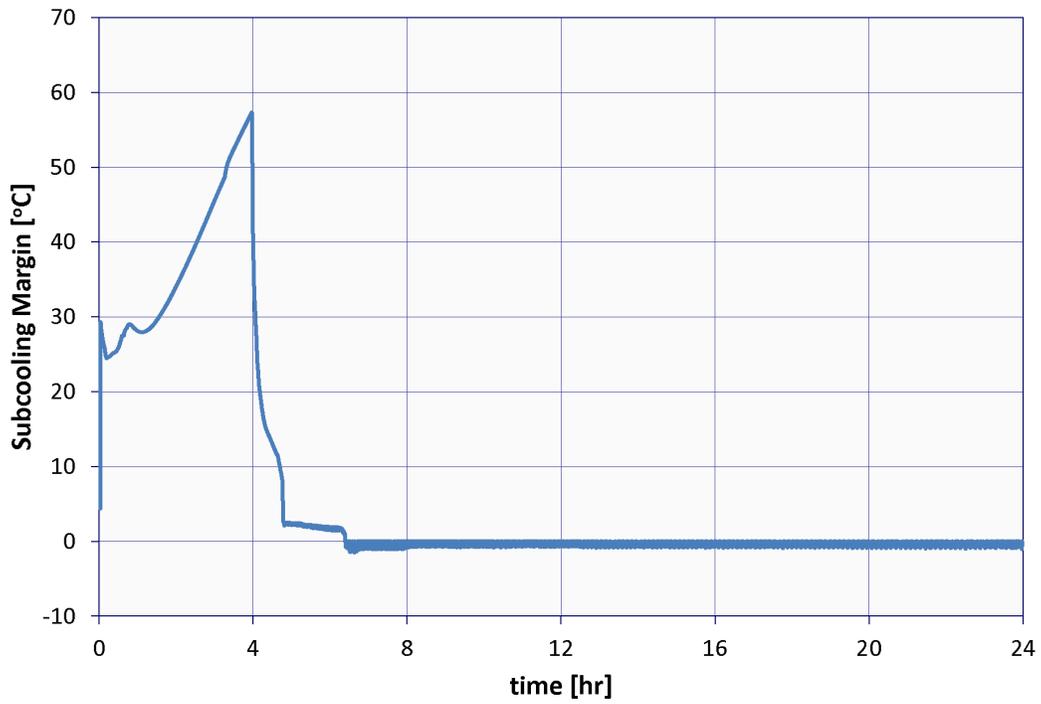
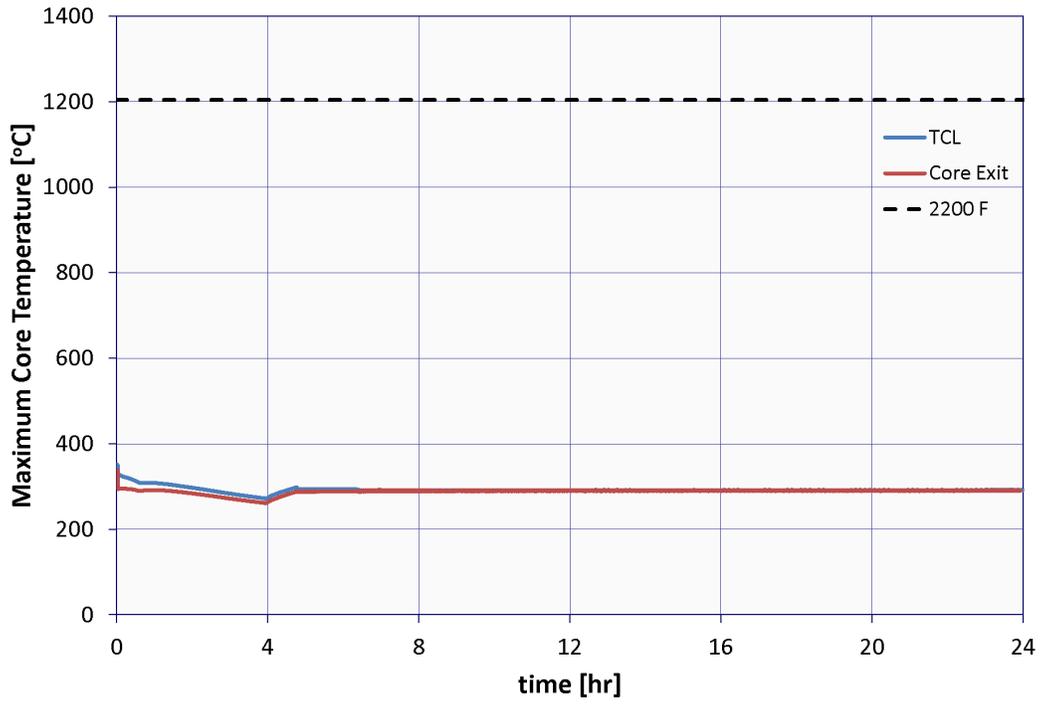


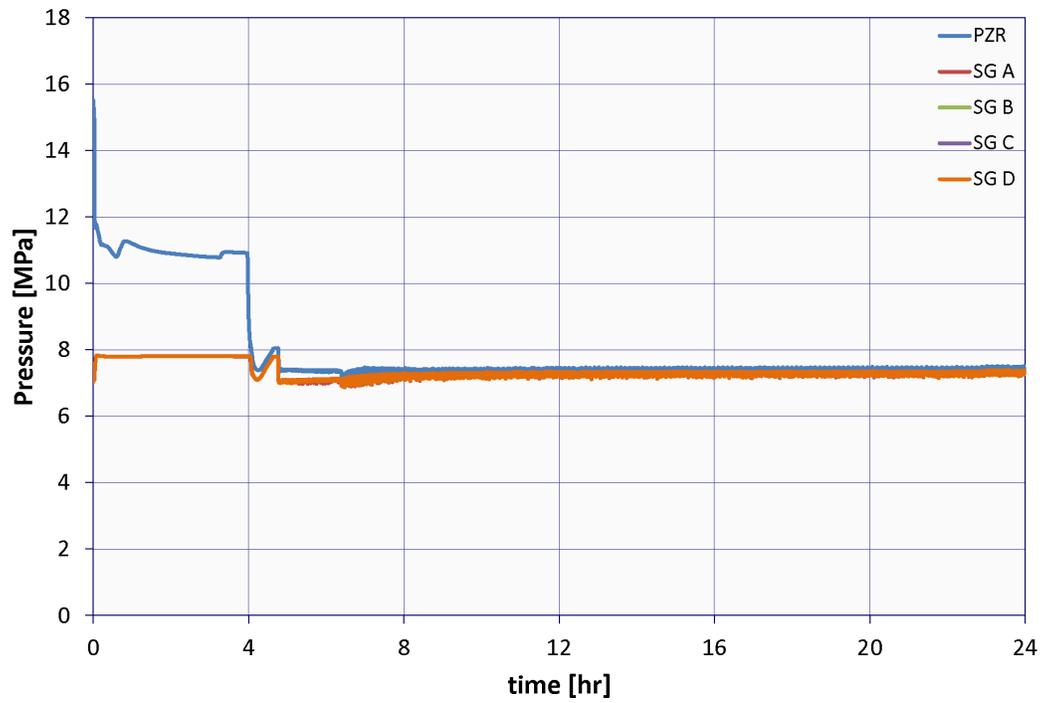
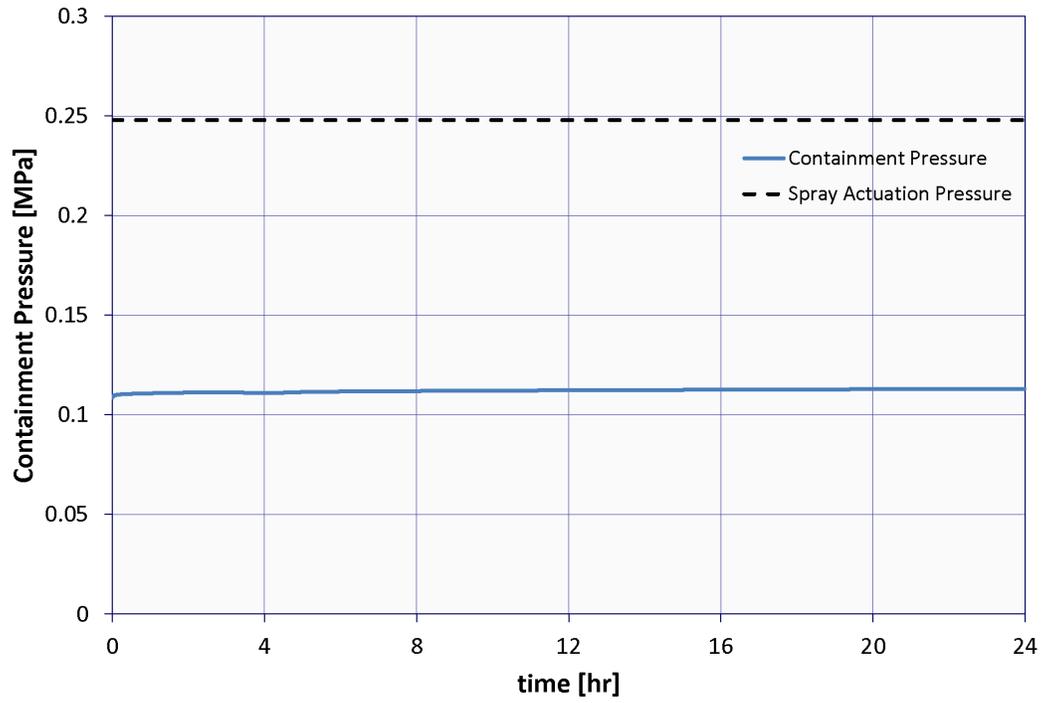


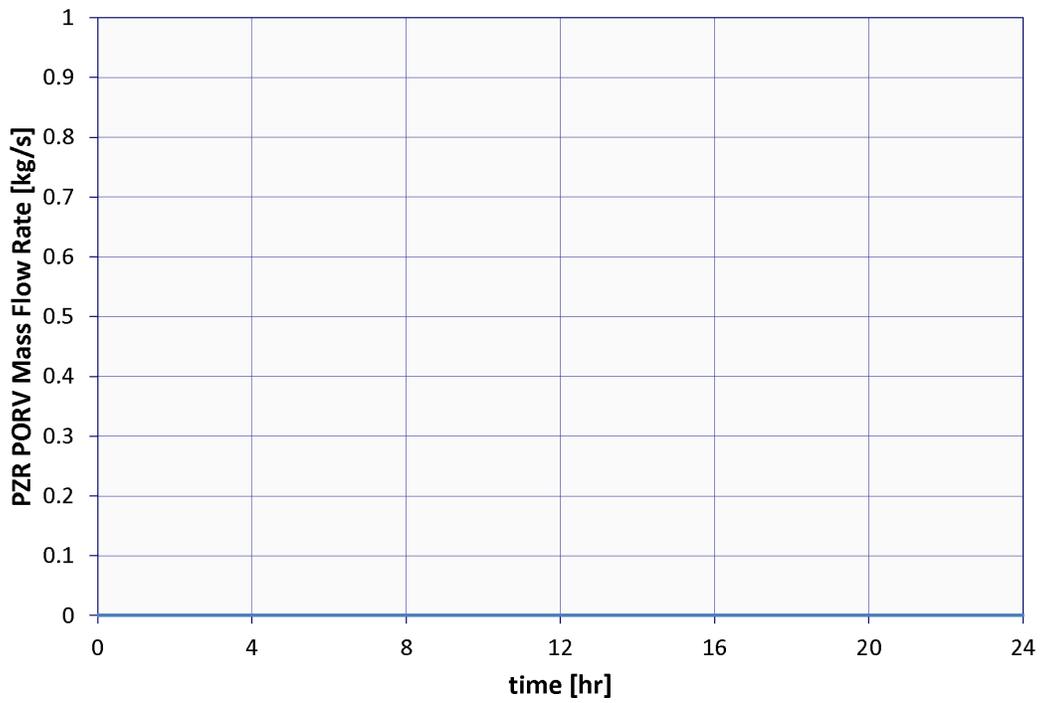
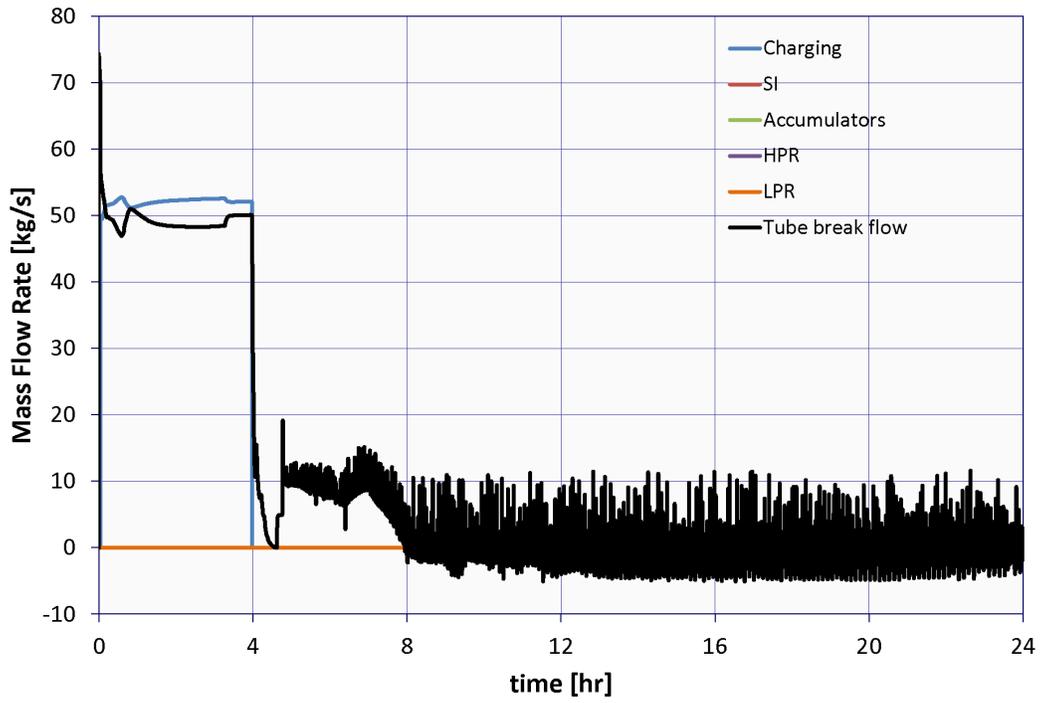


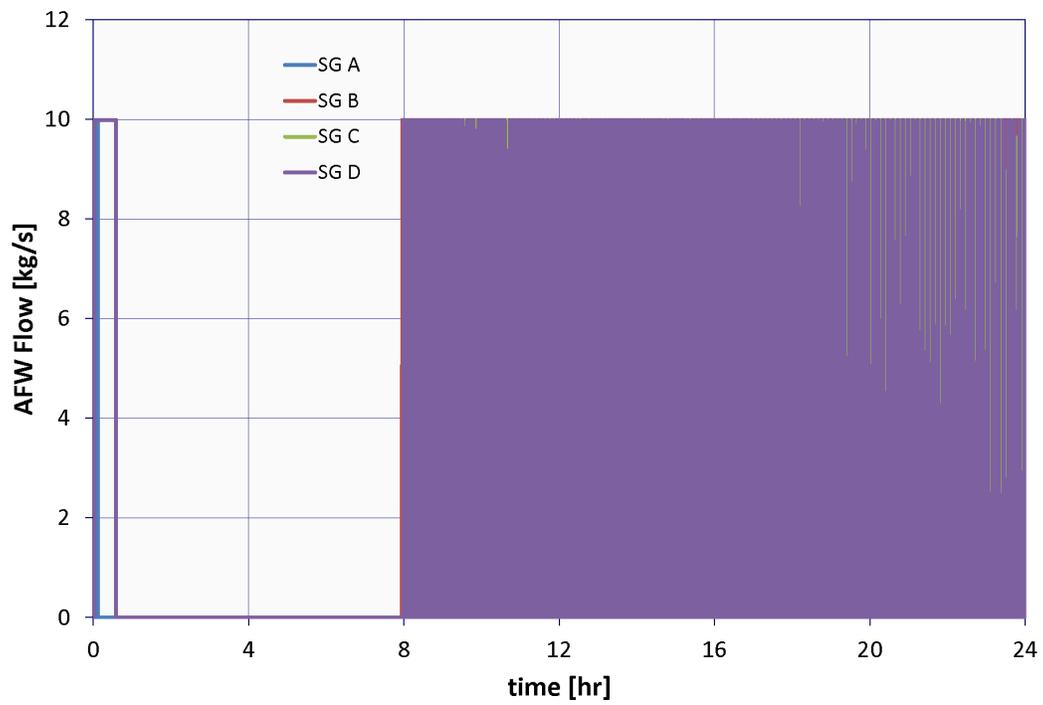
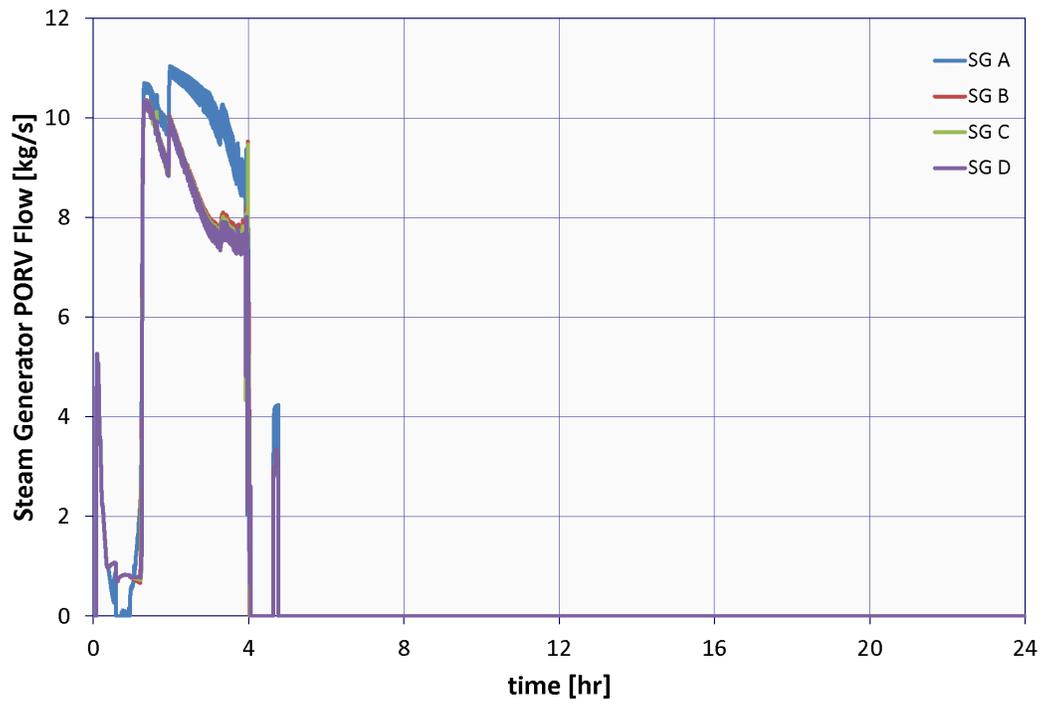


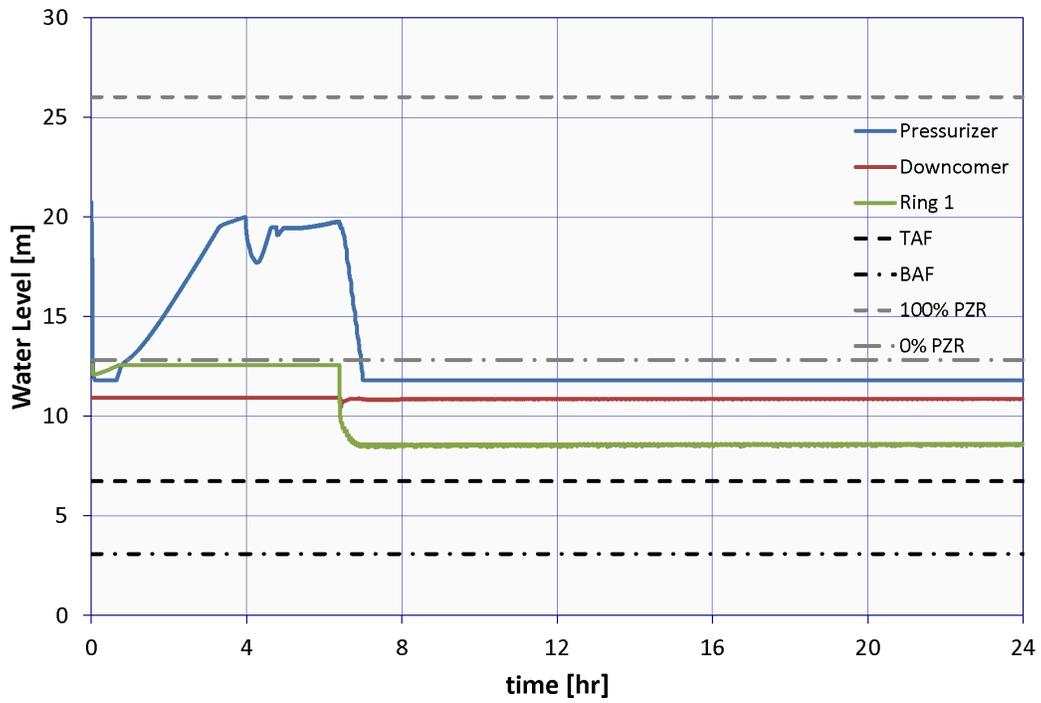
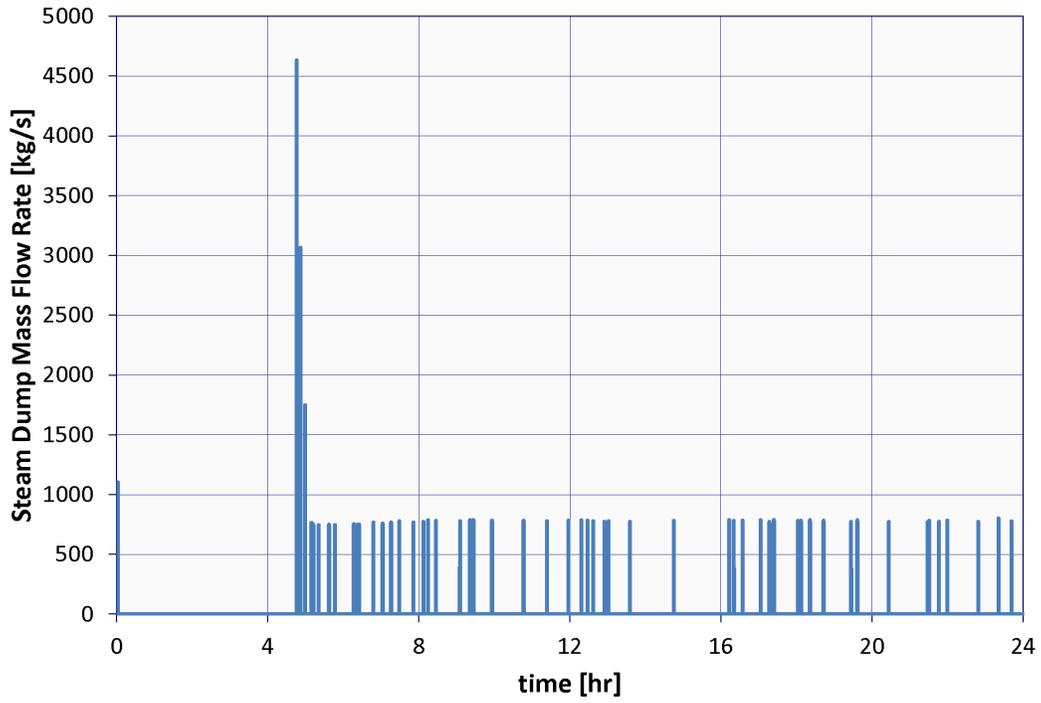
D.1.6 Case 6: 2 Tubes, Max ECCS, Steam Dumps Available

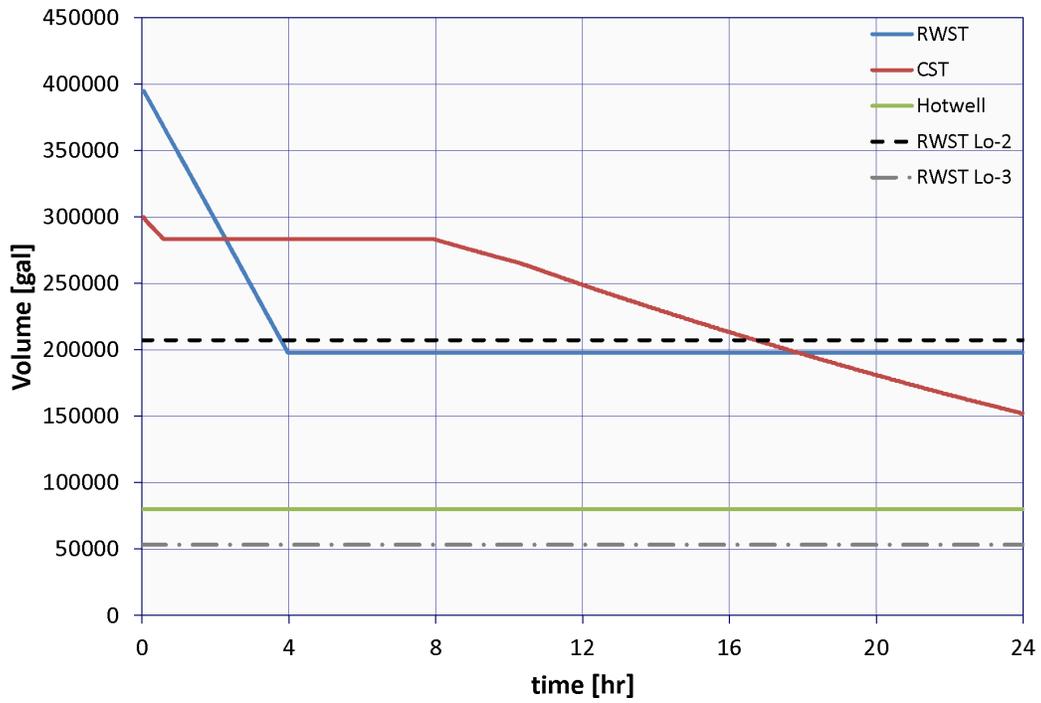
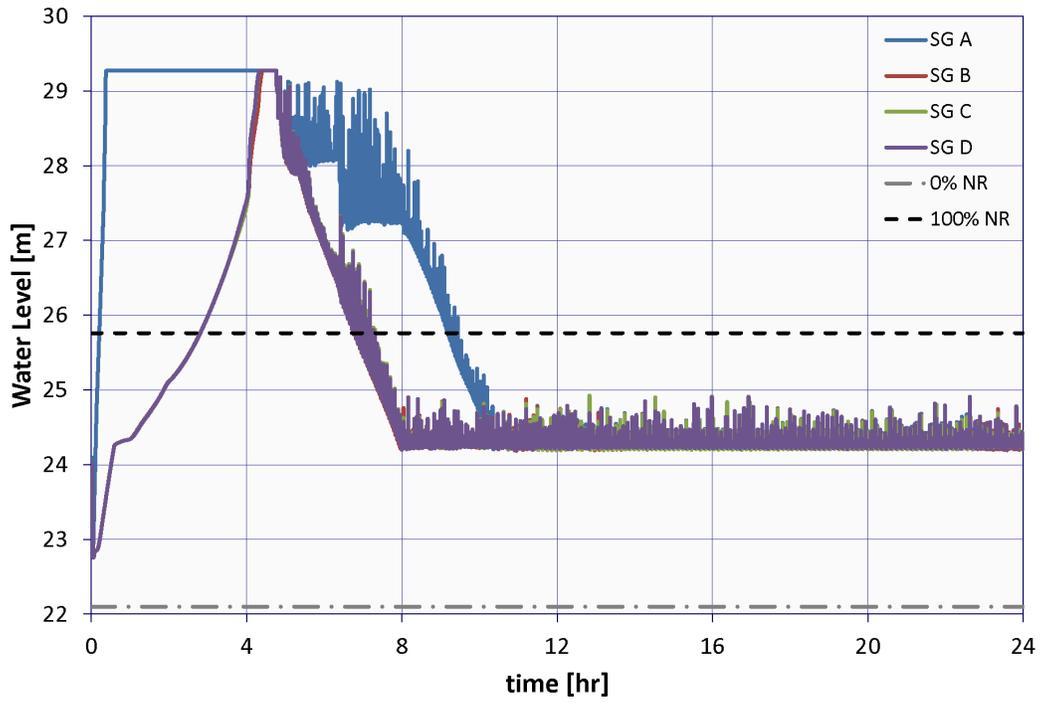




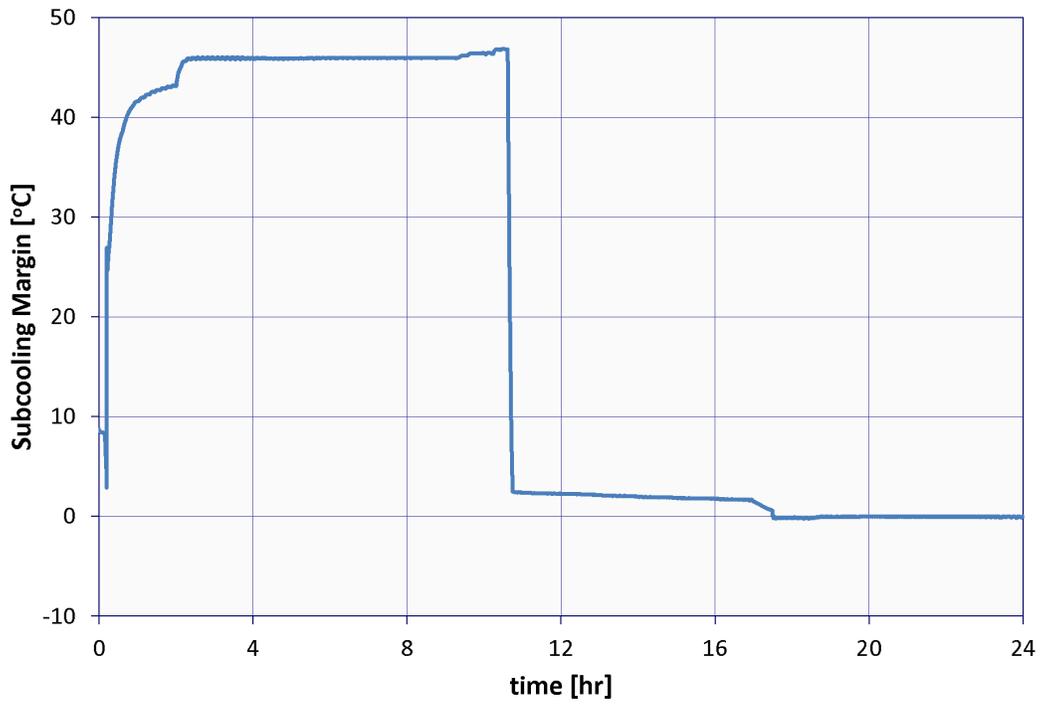
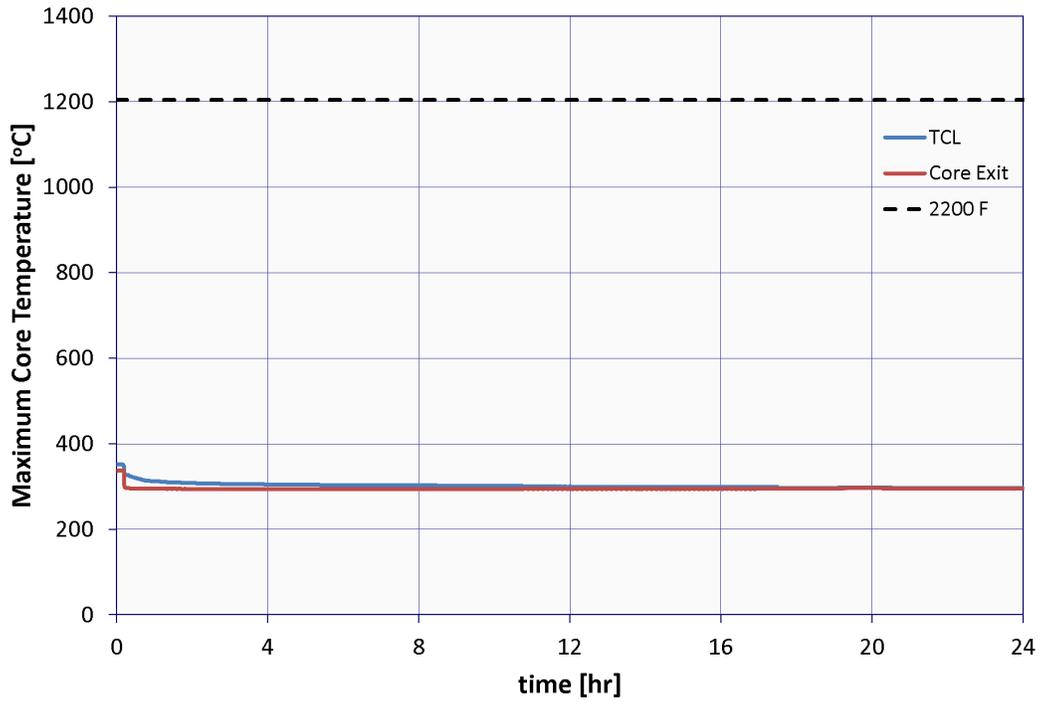


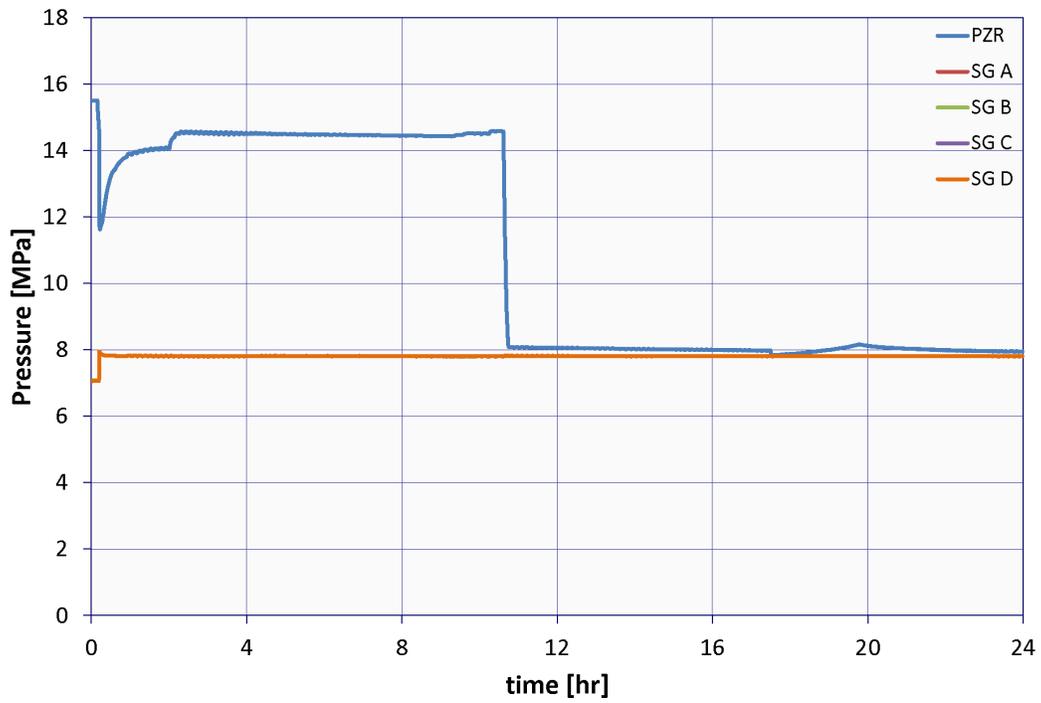
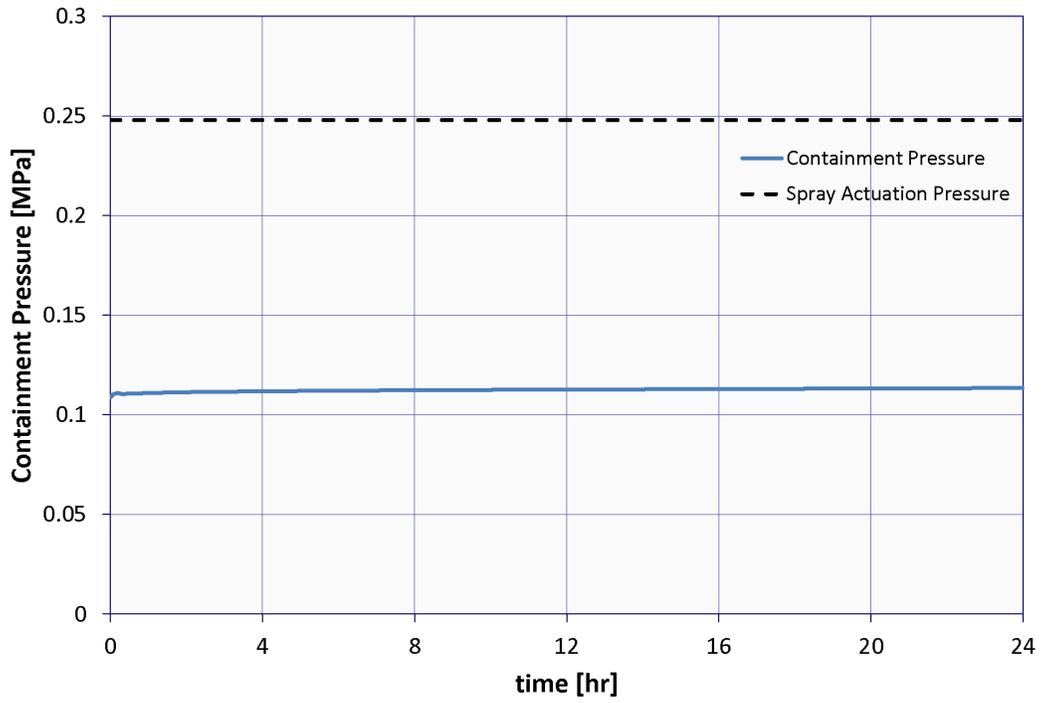


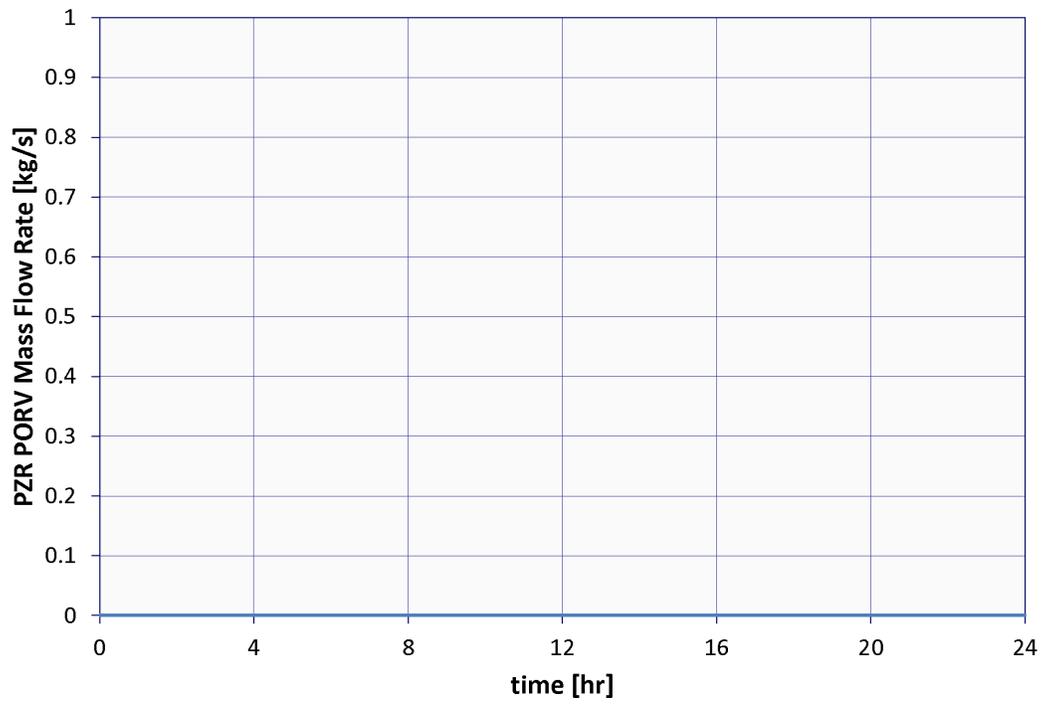
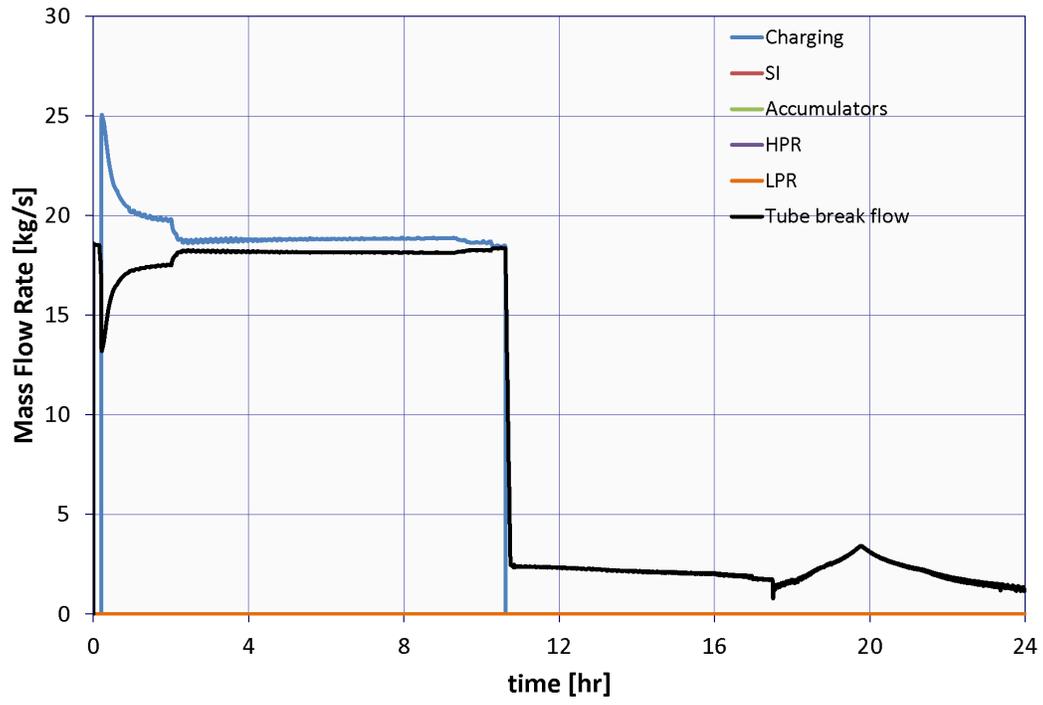


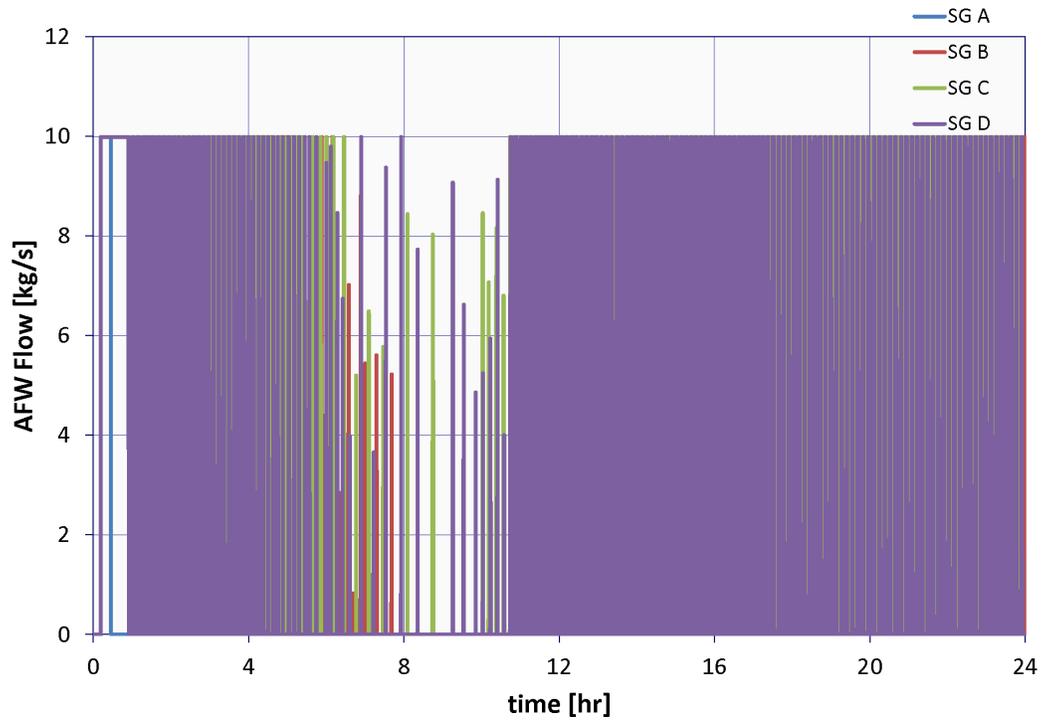
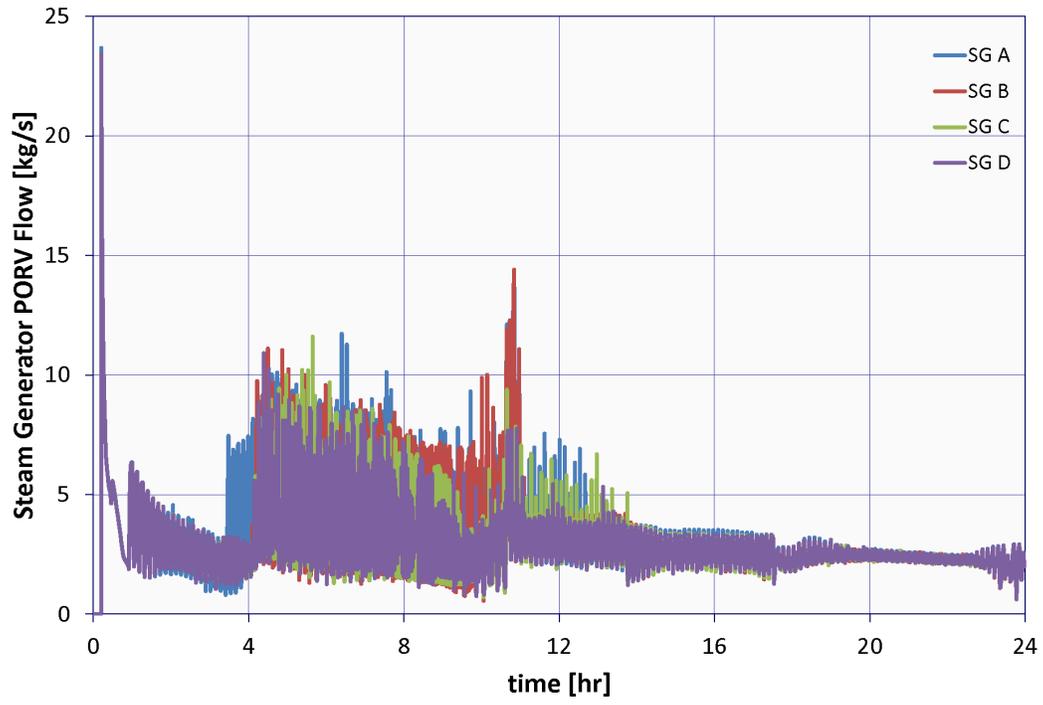


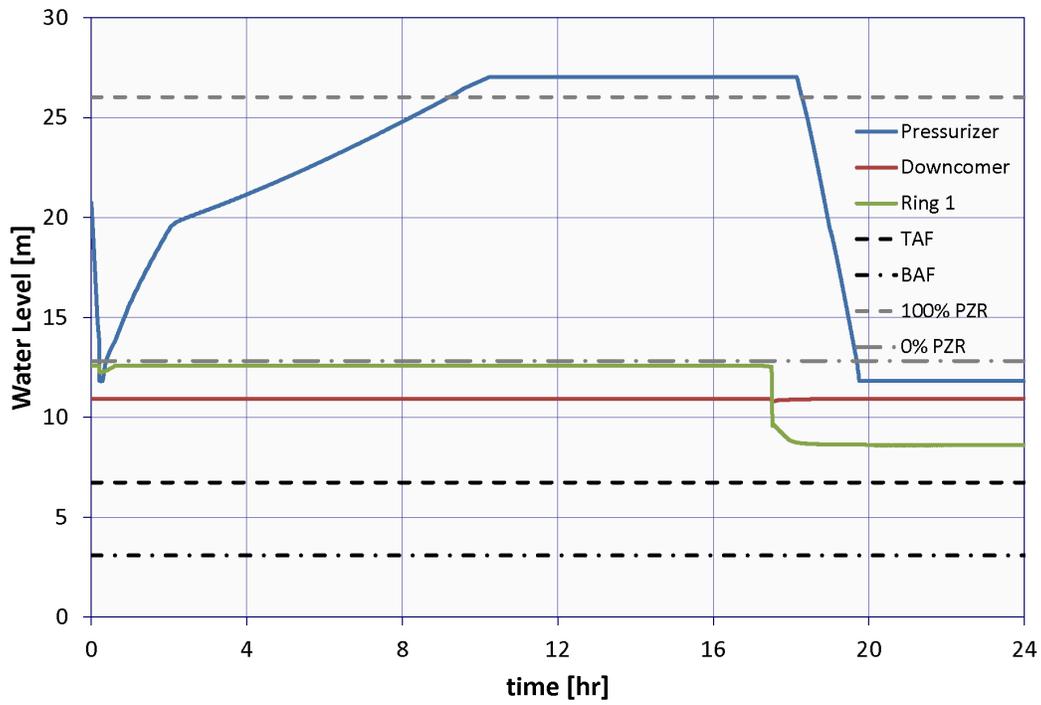
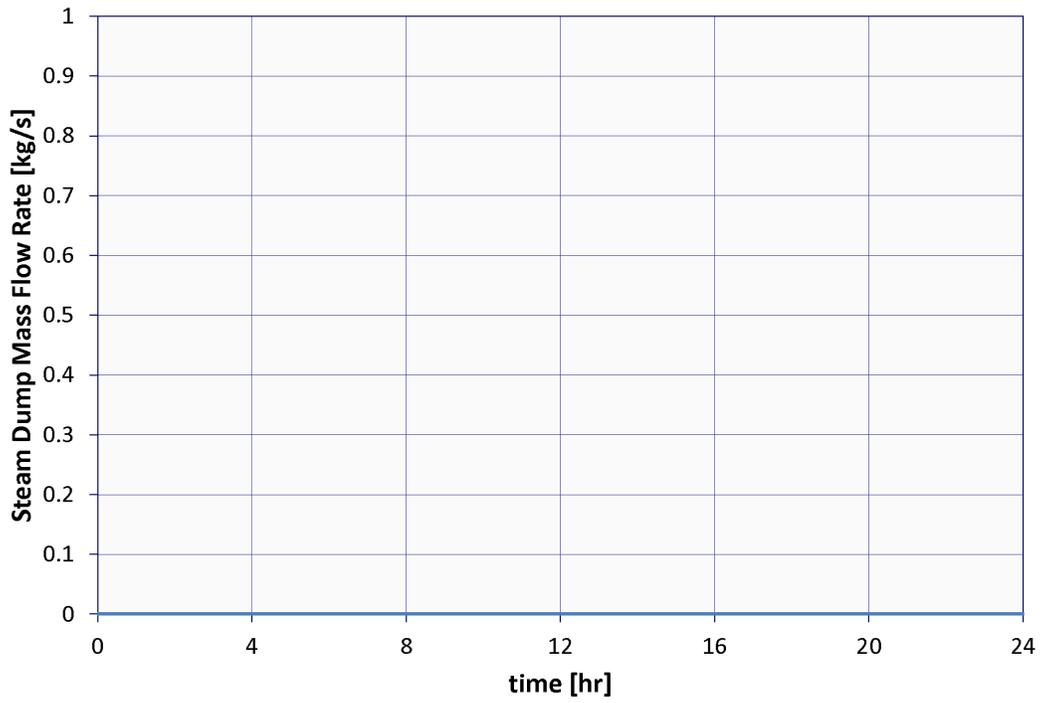
**D.1.7 Case 7: 0.5 Tube, Min ECCS, Automatic Scram, No Steam Dumps**

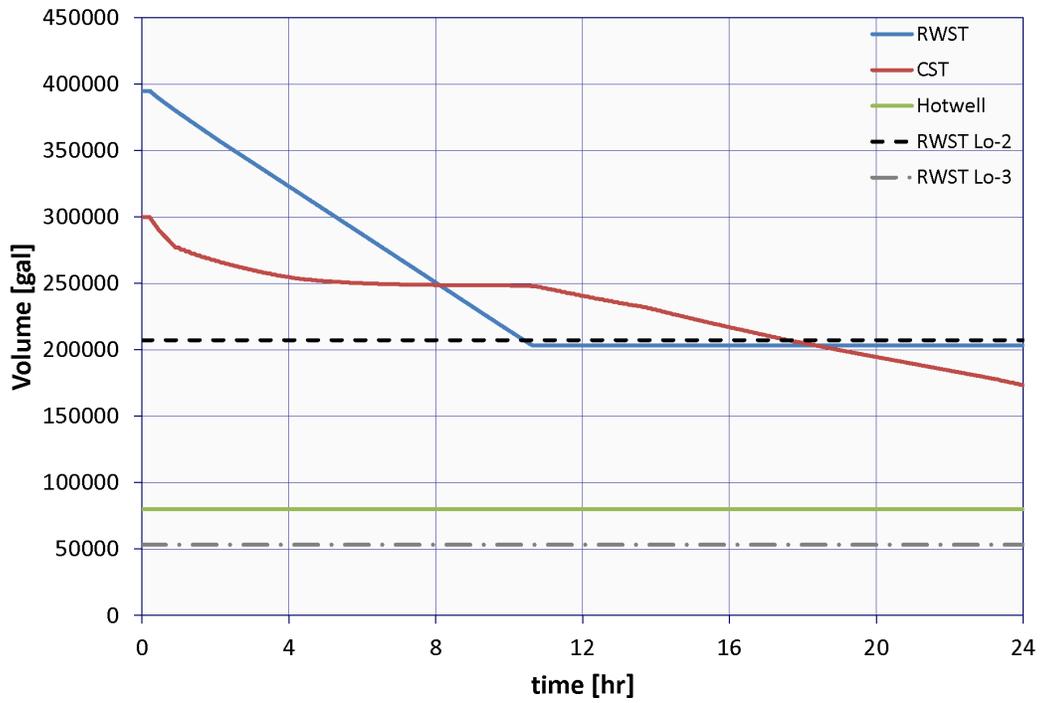
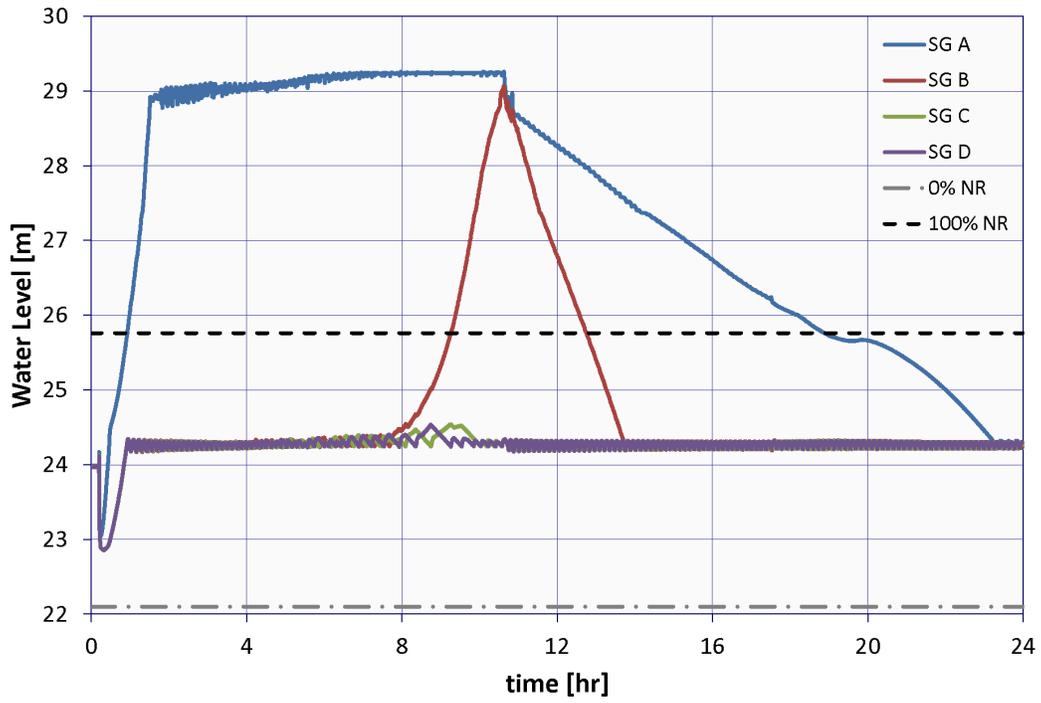




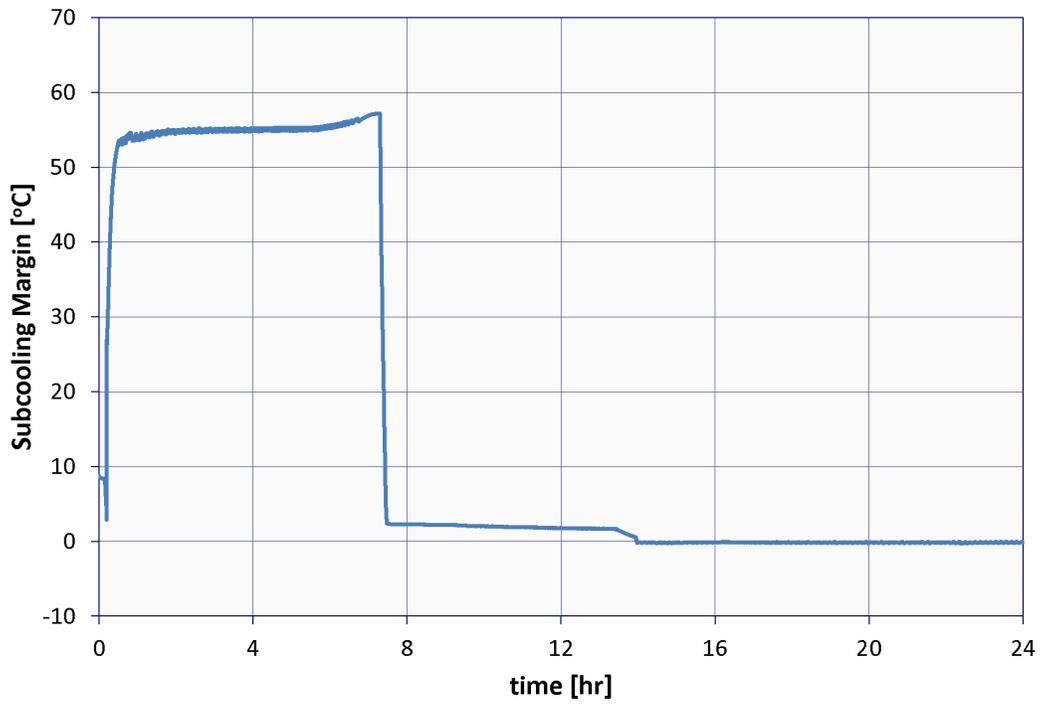
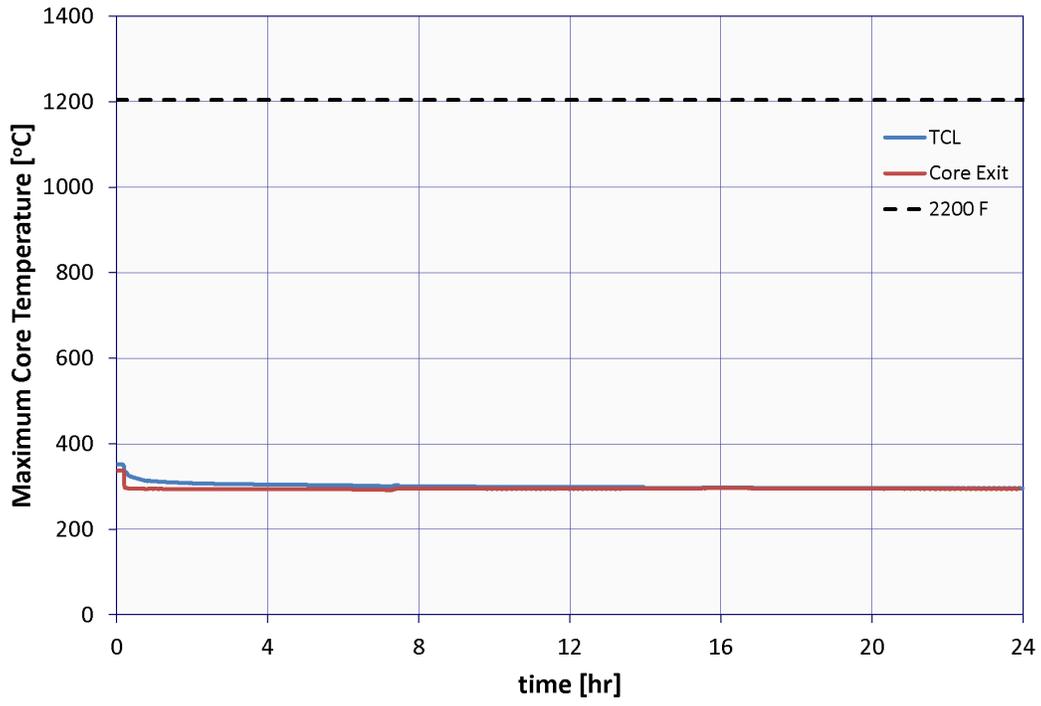


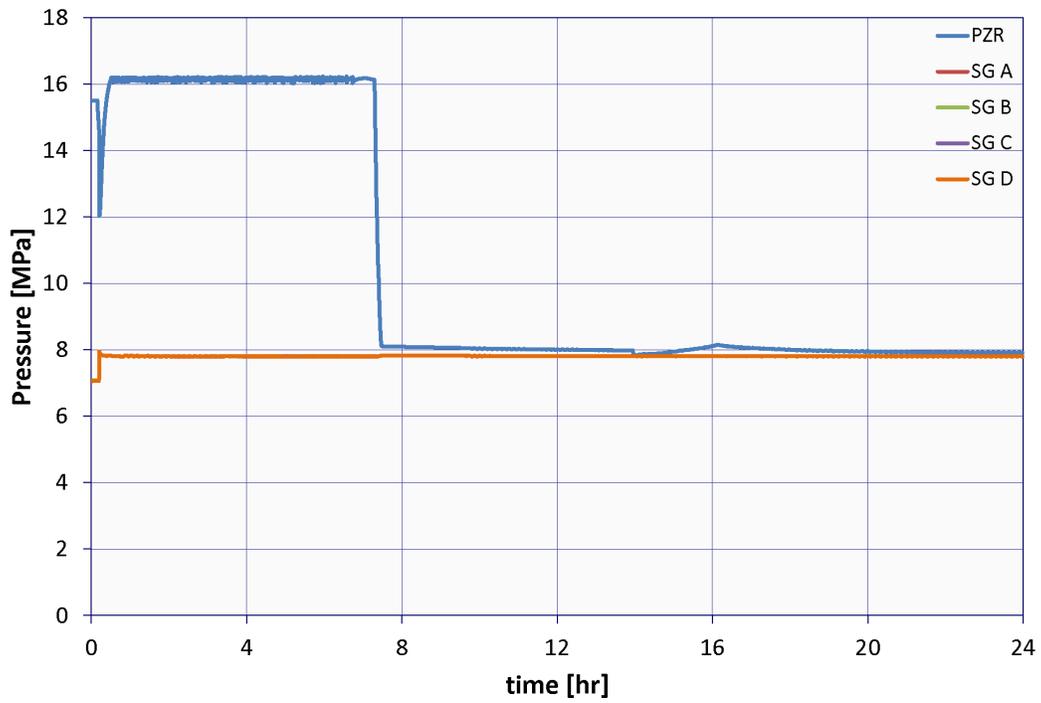
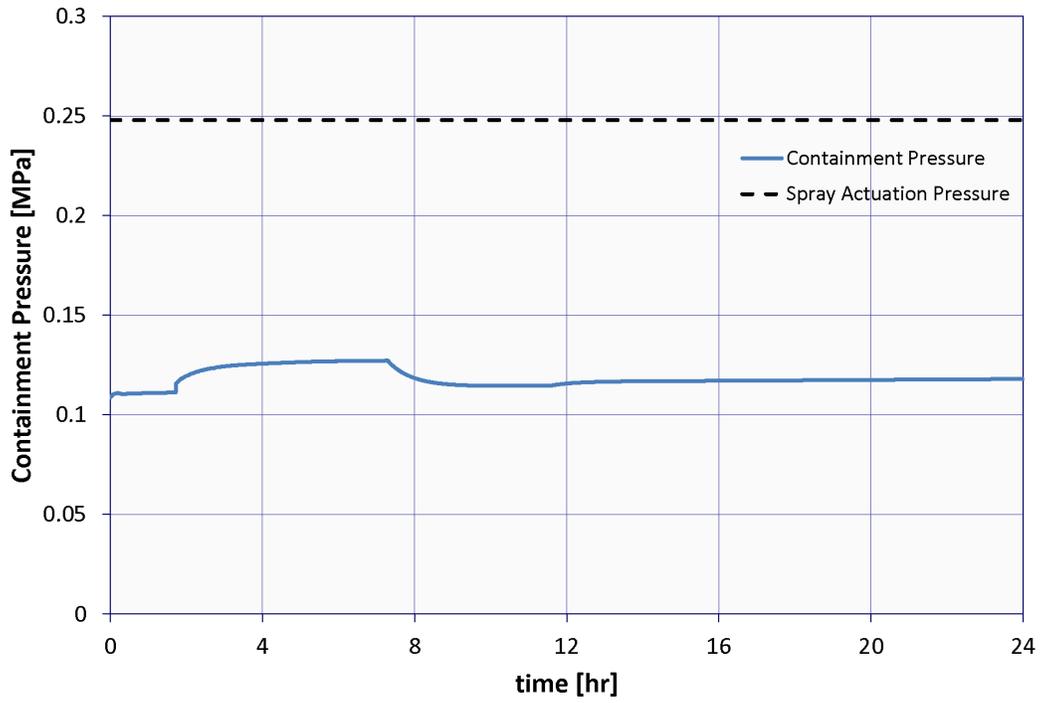


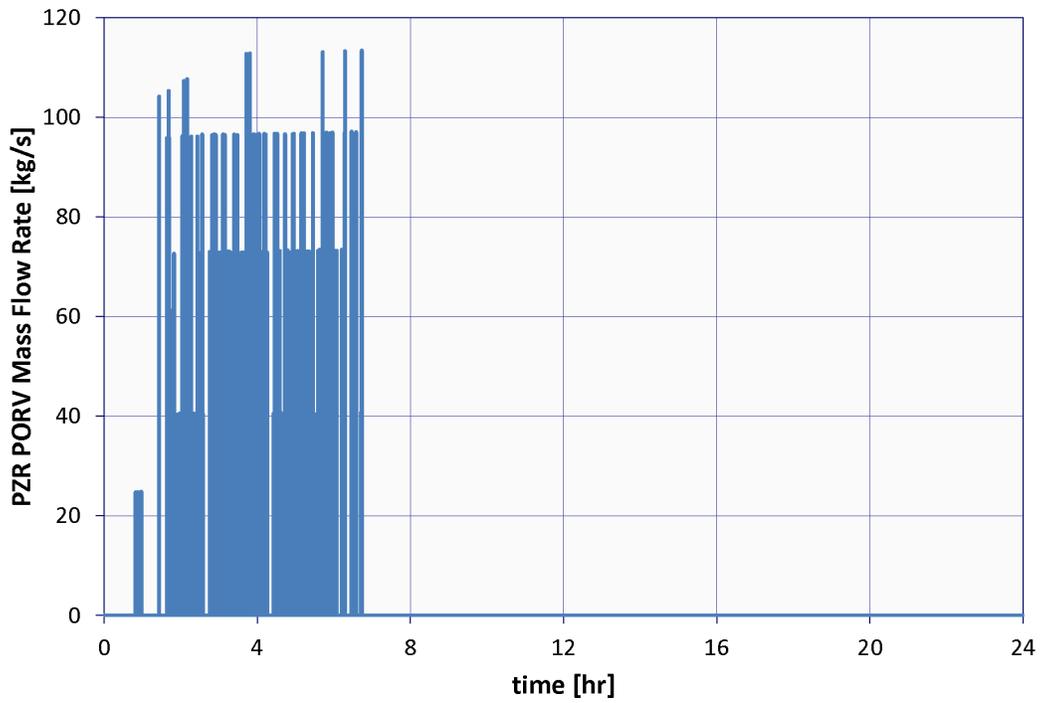
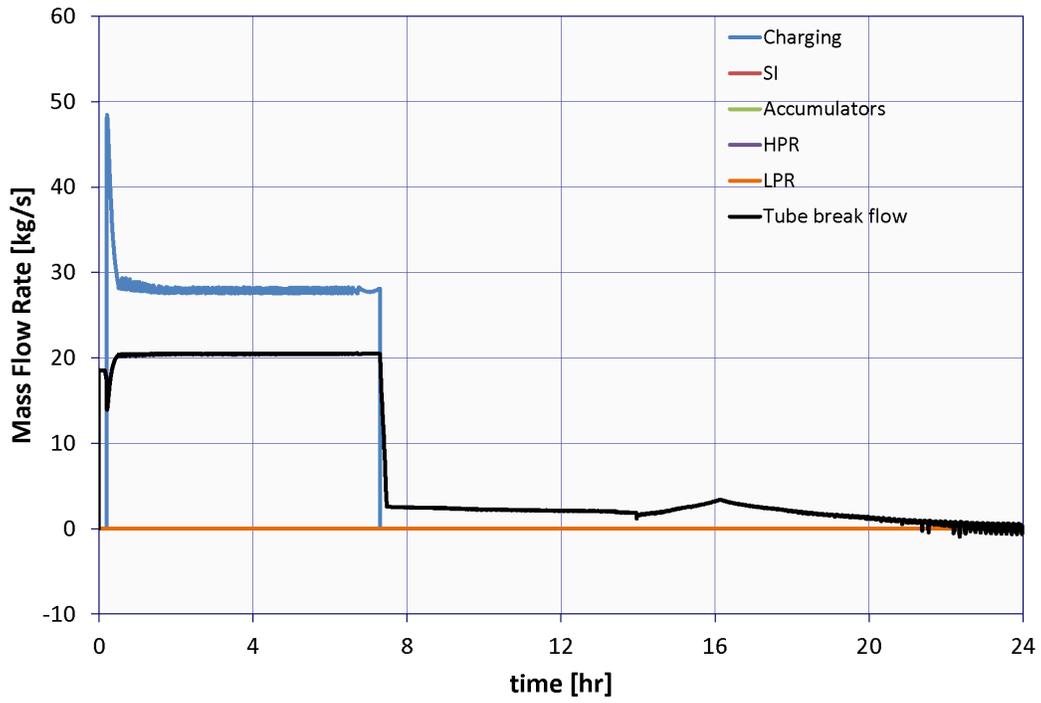


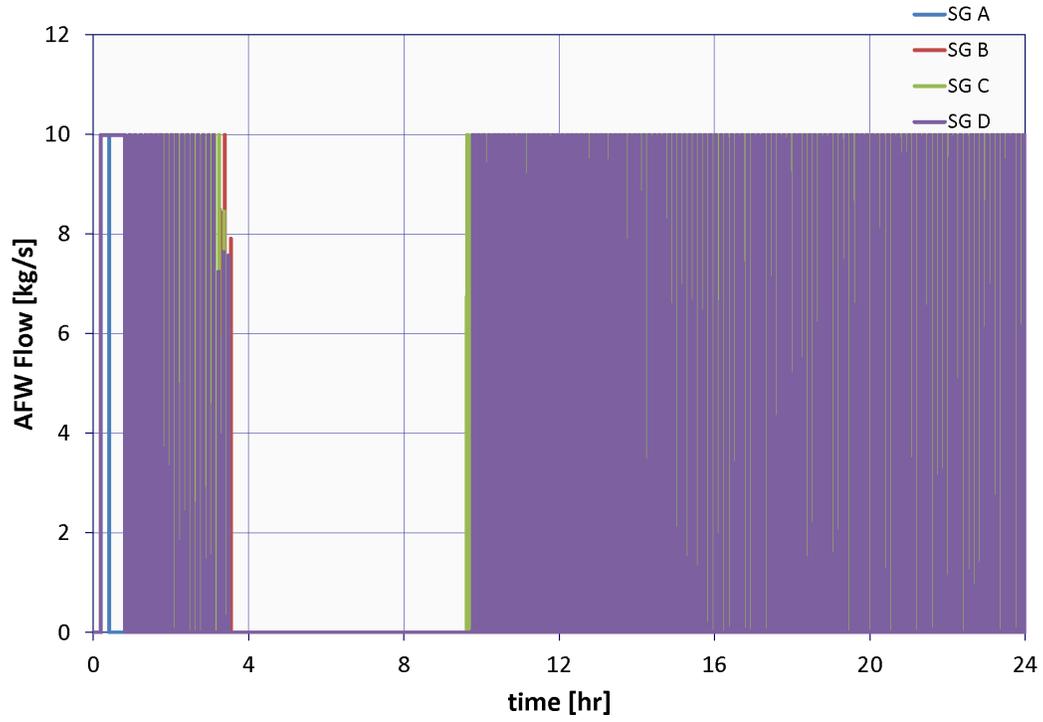
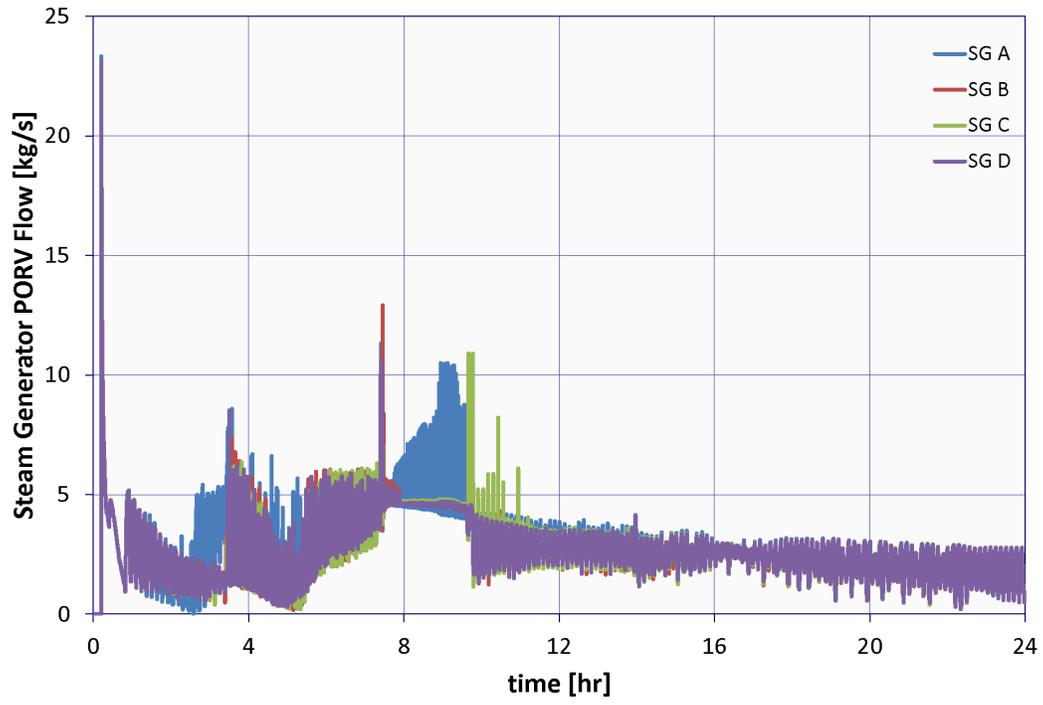


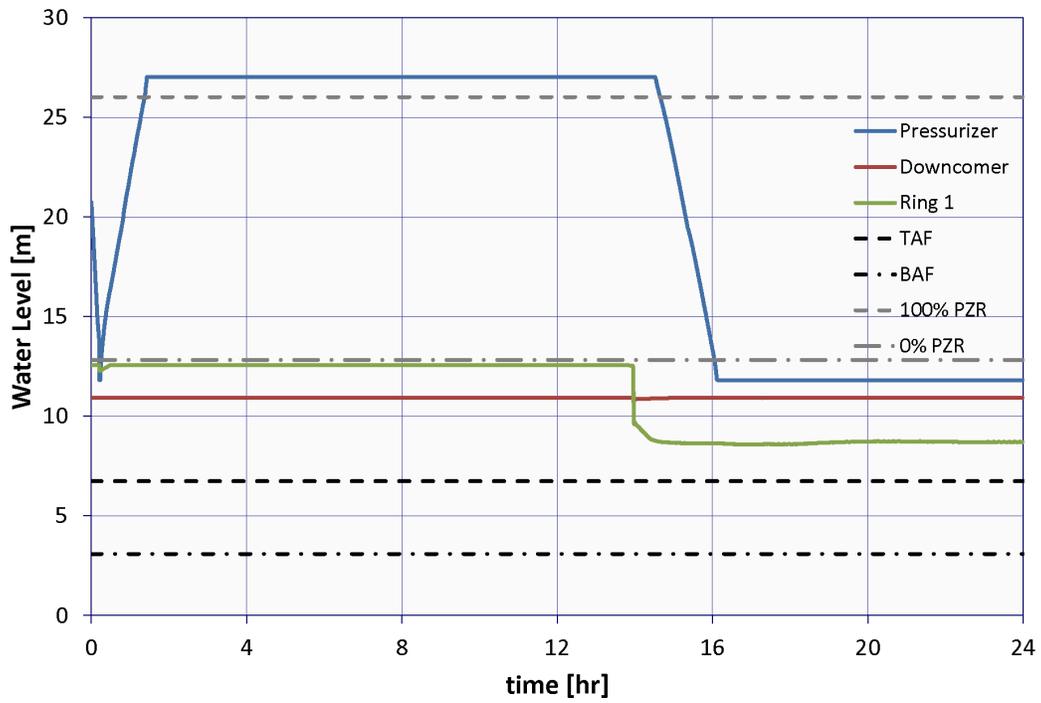
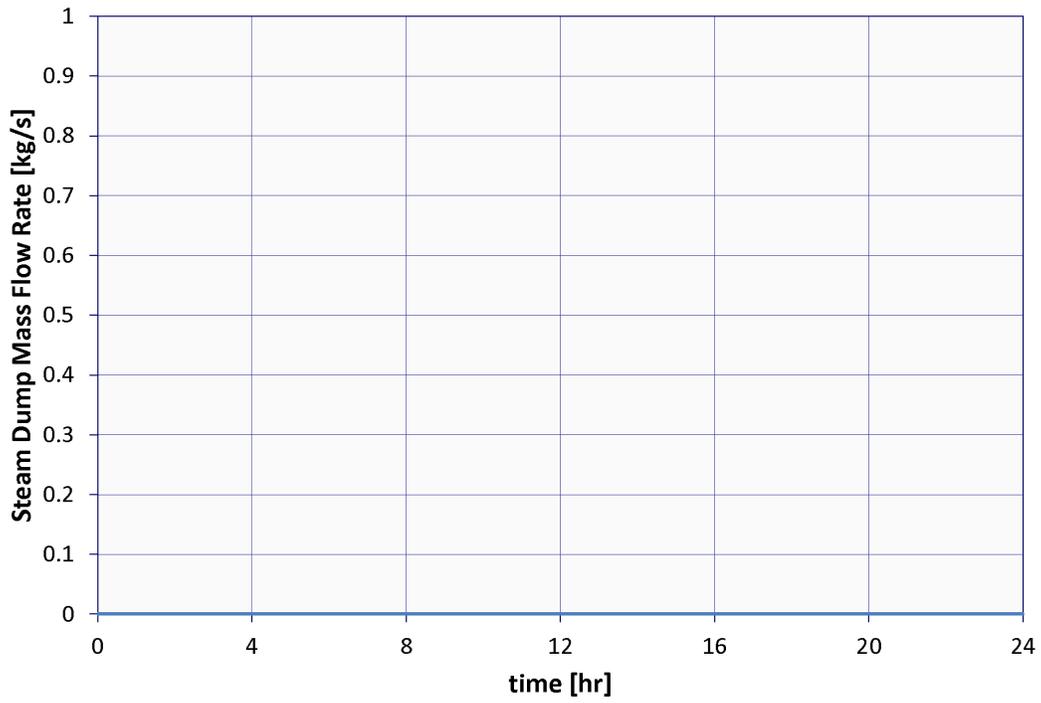
**D.1.8 Case 8: 0.5 Tube, Max ECCS, Automatic Scram, No Steam Dumps**

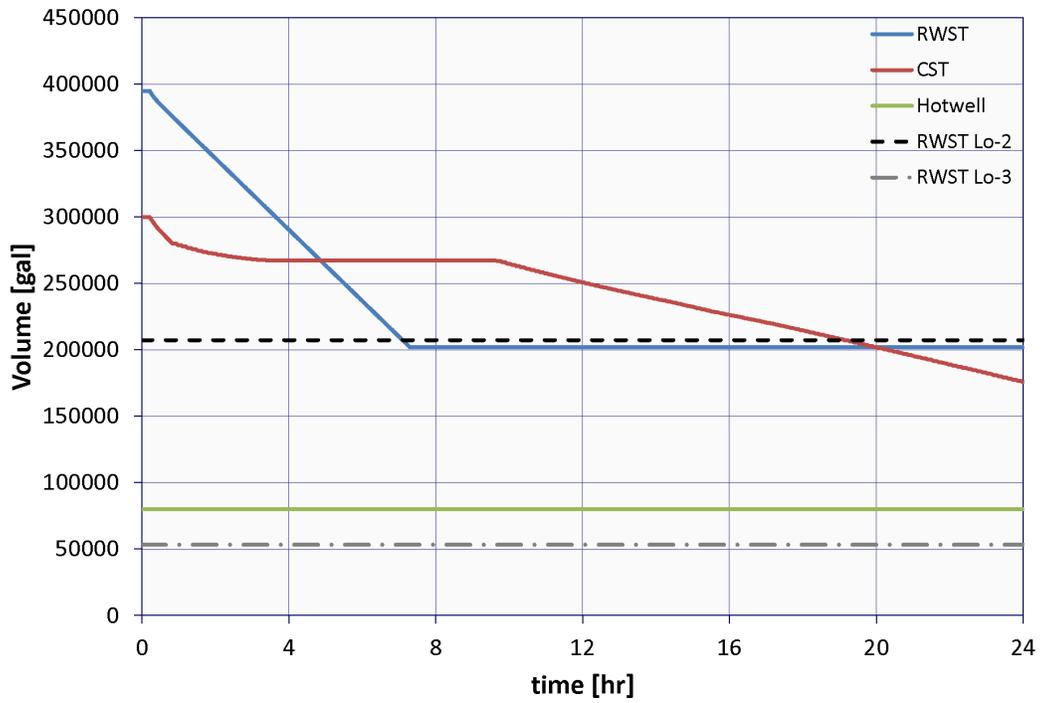
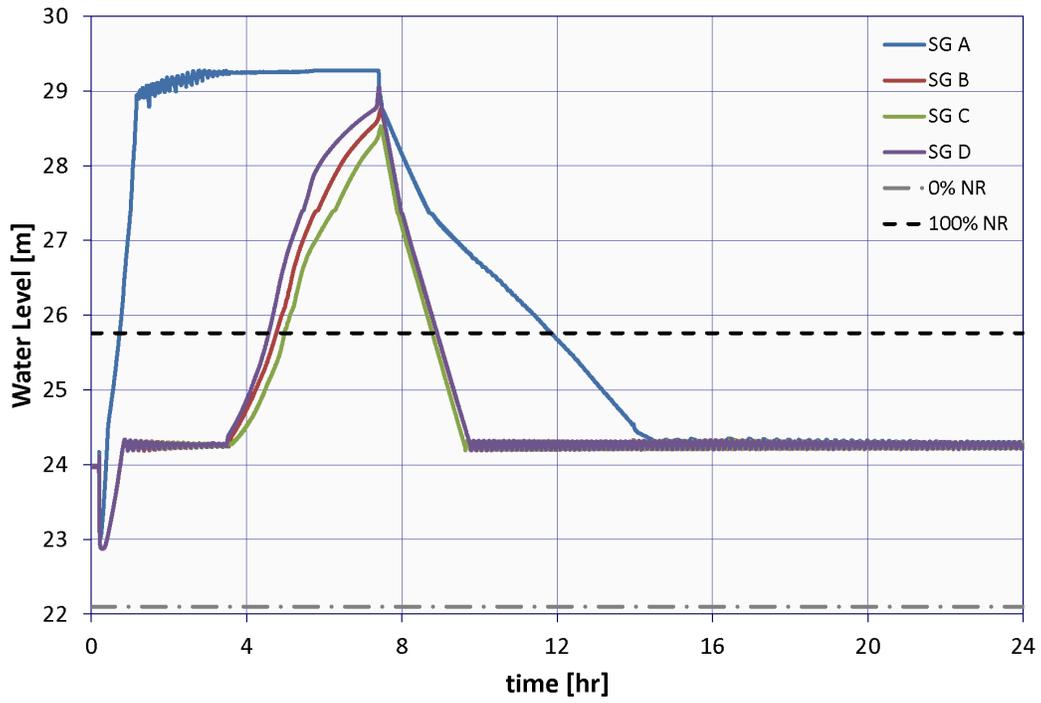












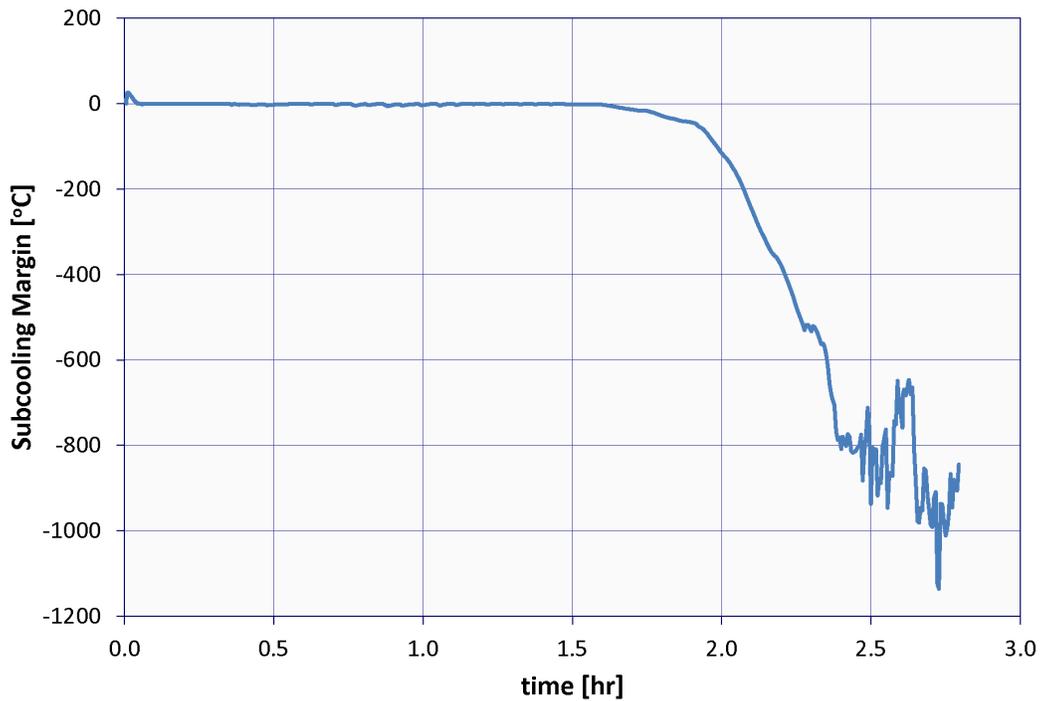
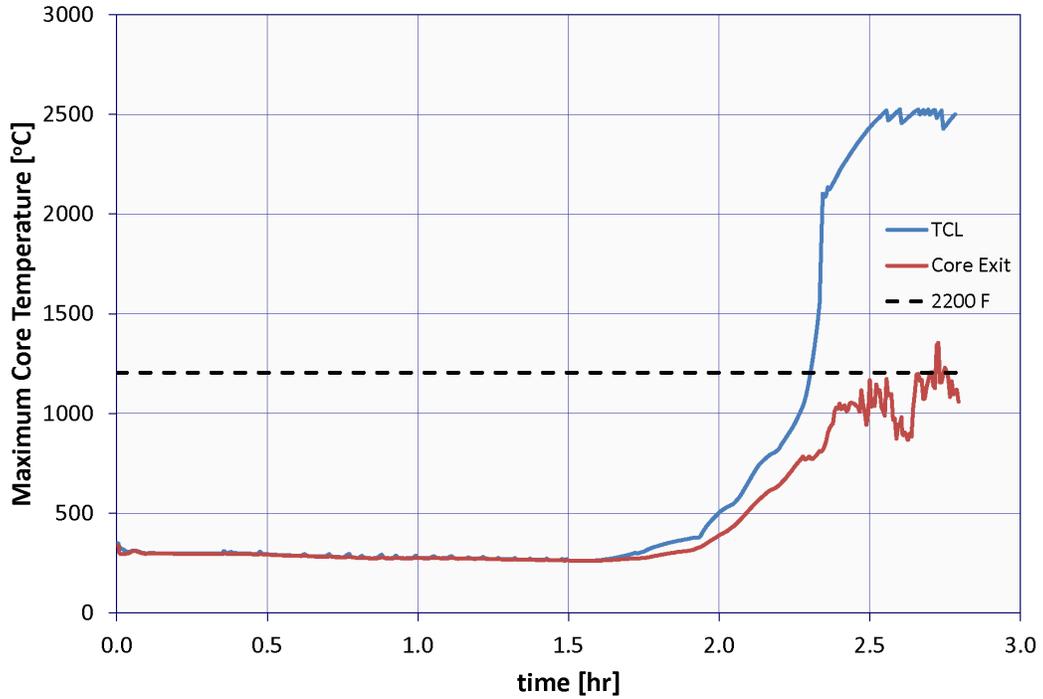
## **APPENDIX E**

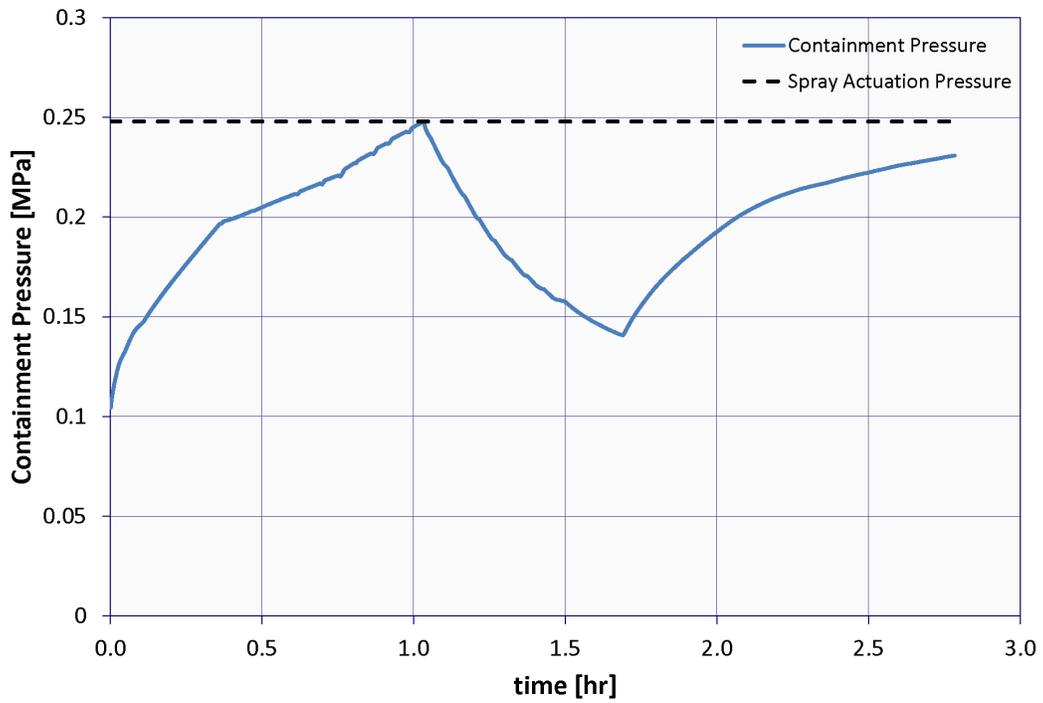
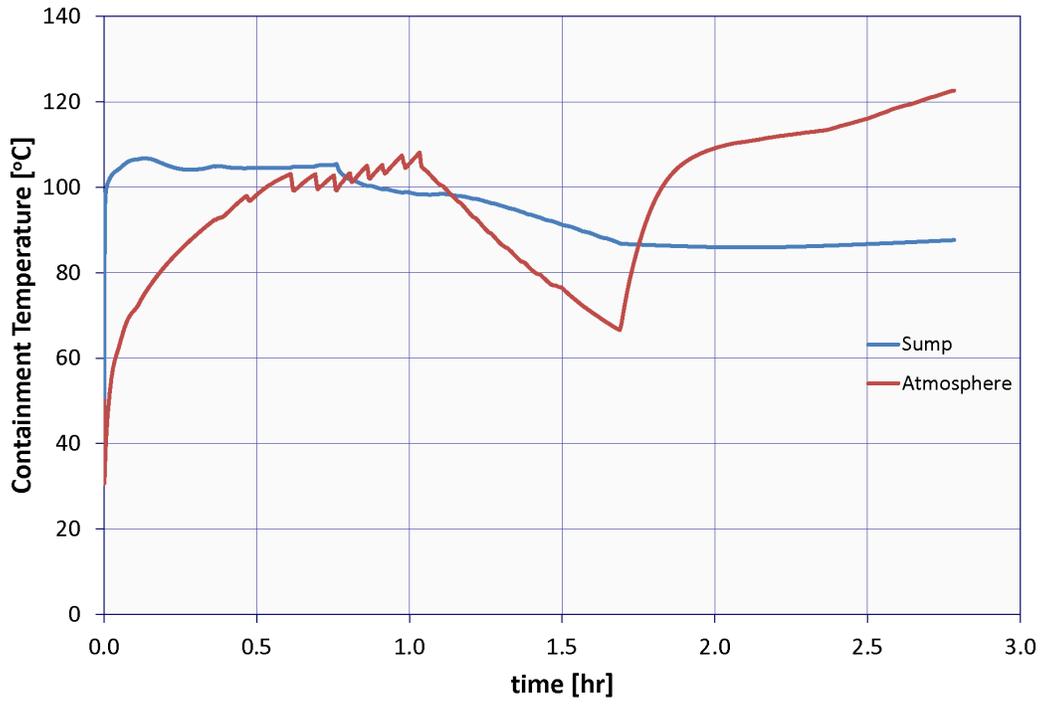
# **DETAILED MEDIUM-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS RESULTS**

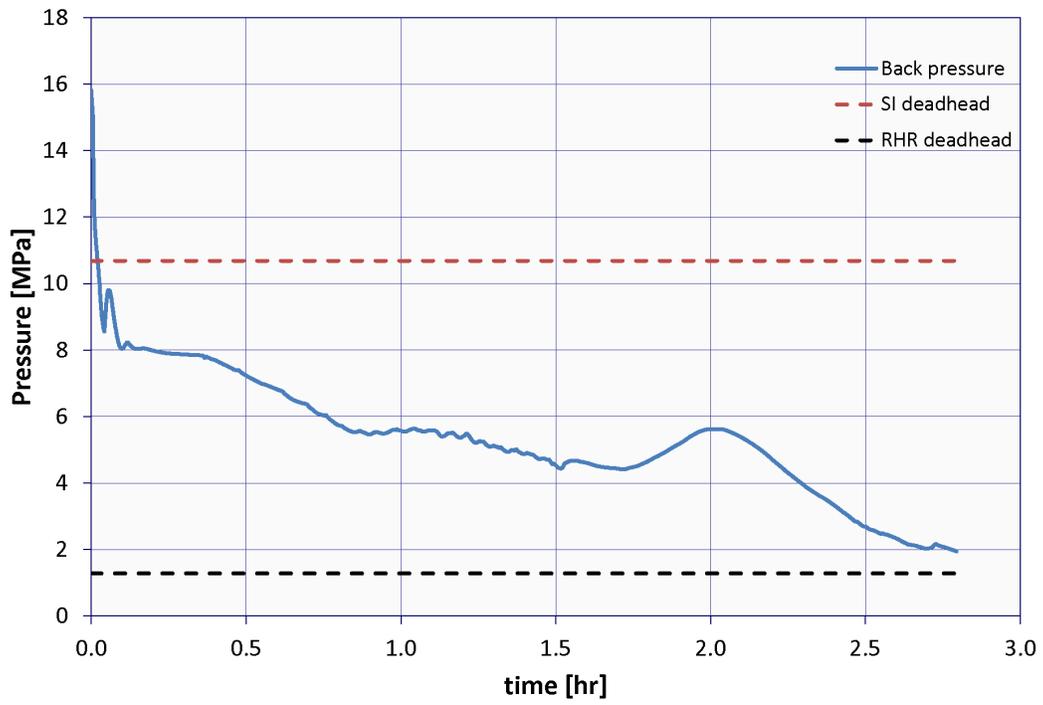
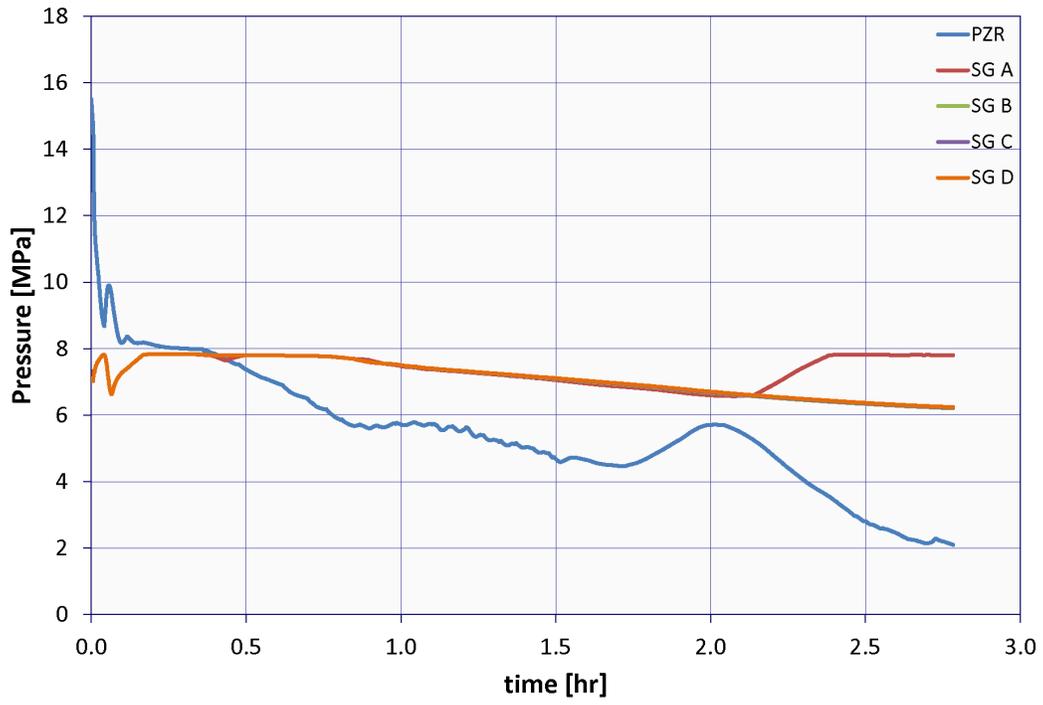


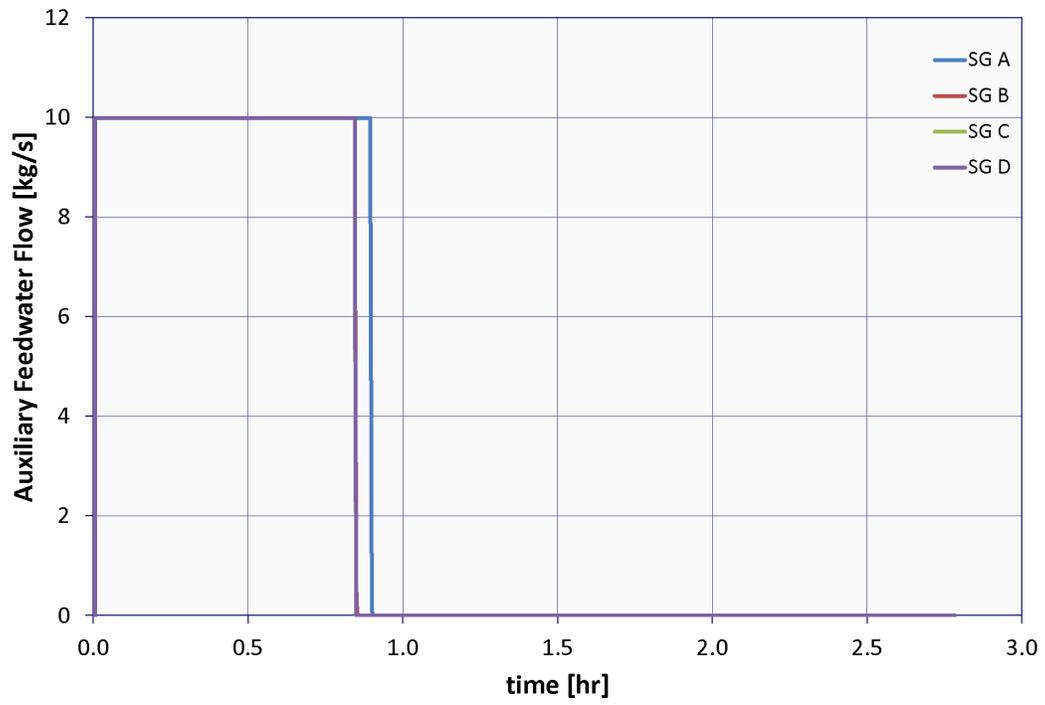
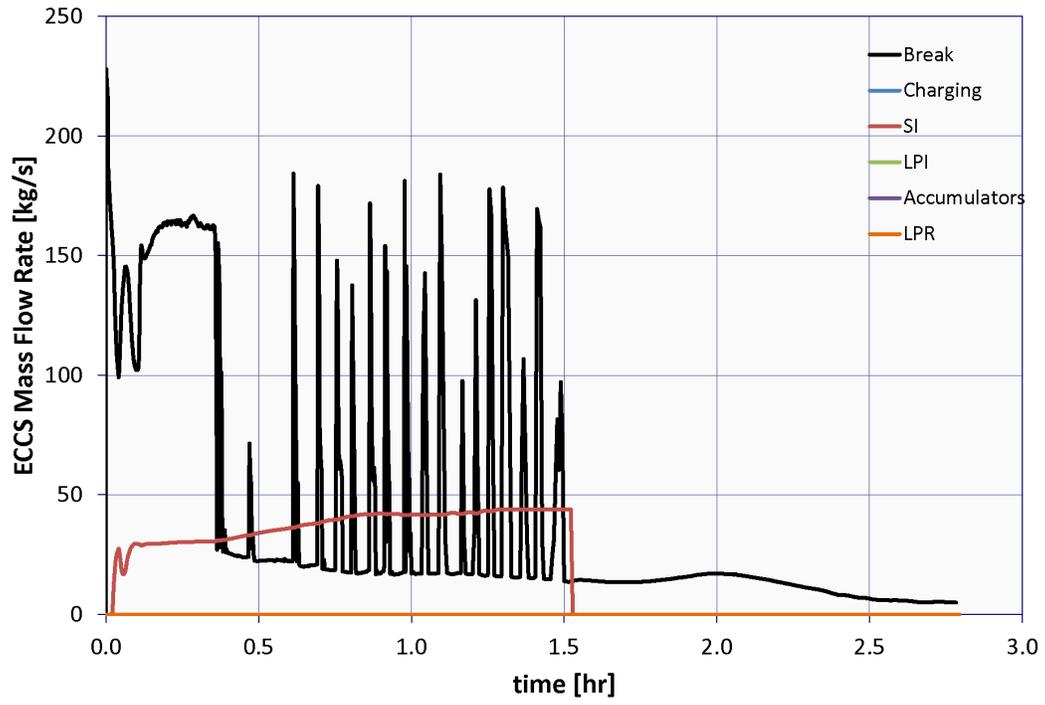
## E.1 Medium-Break Loss-of-Coolant Accident Injection Success Criteria

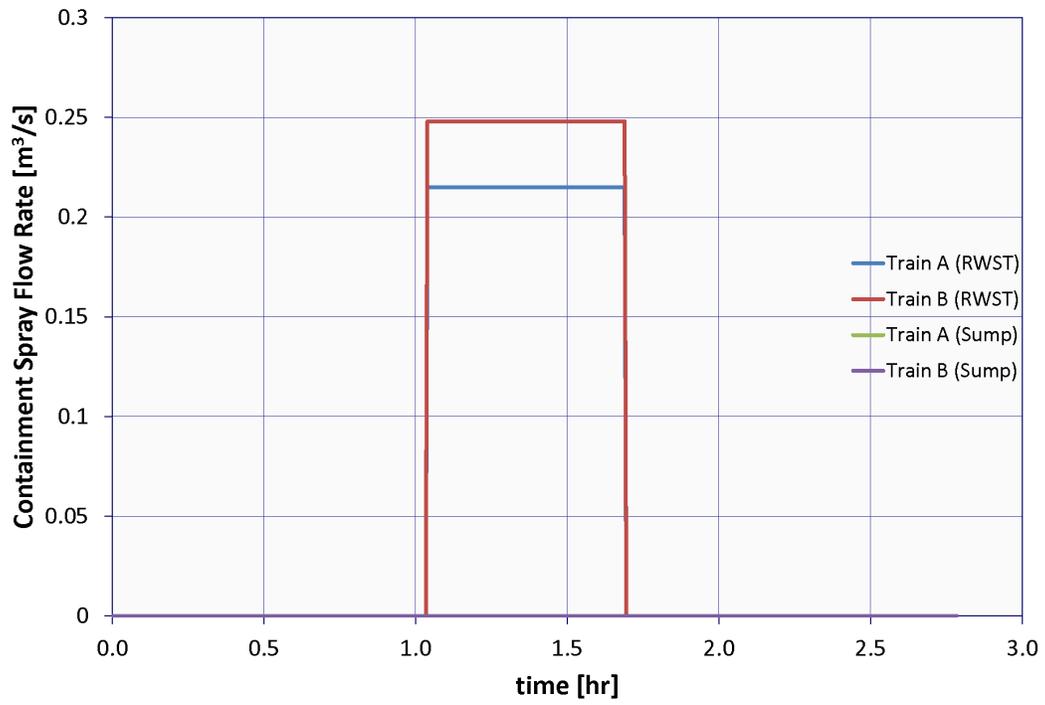
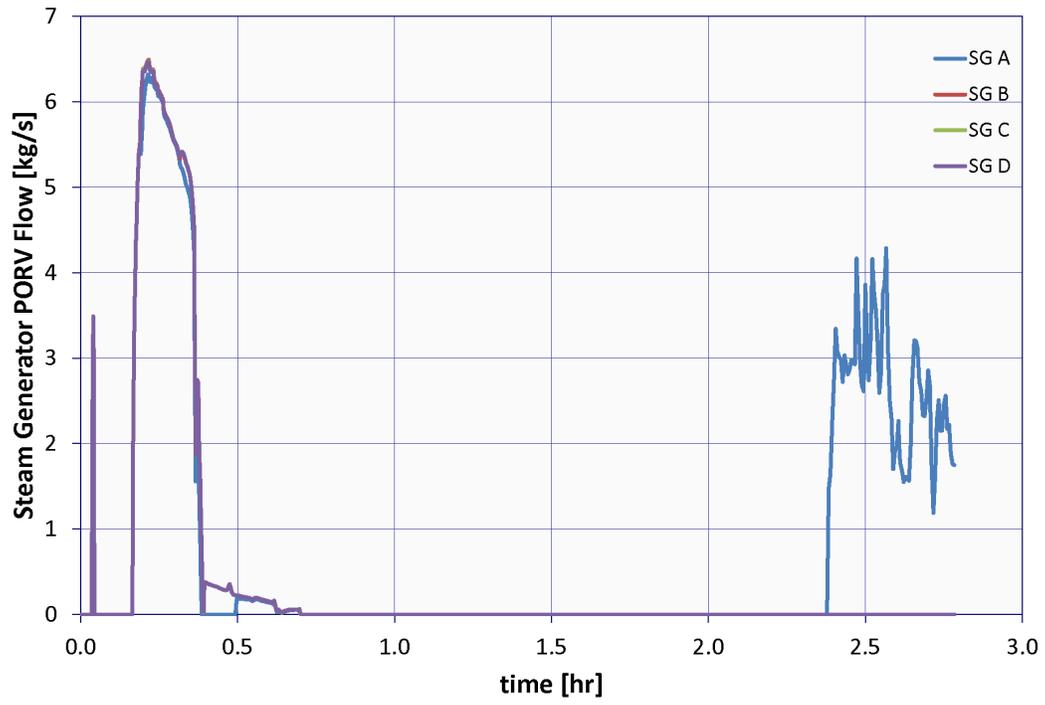
### E.1.1 Case 1: 2-in. Break with 1/2 SI, 1/2 RHR, 2/2 CS Pumps

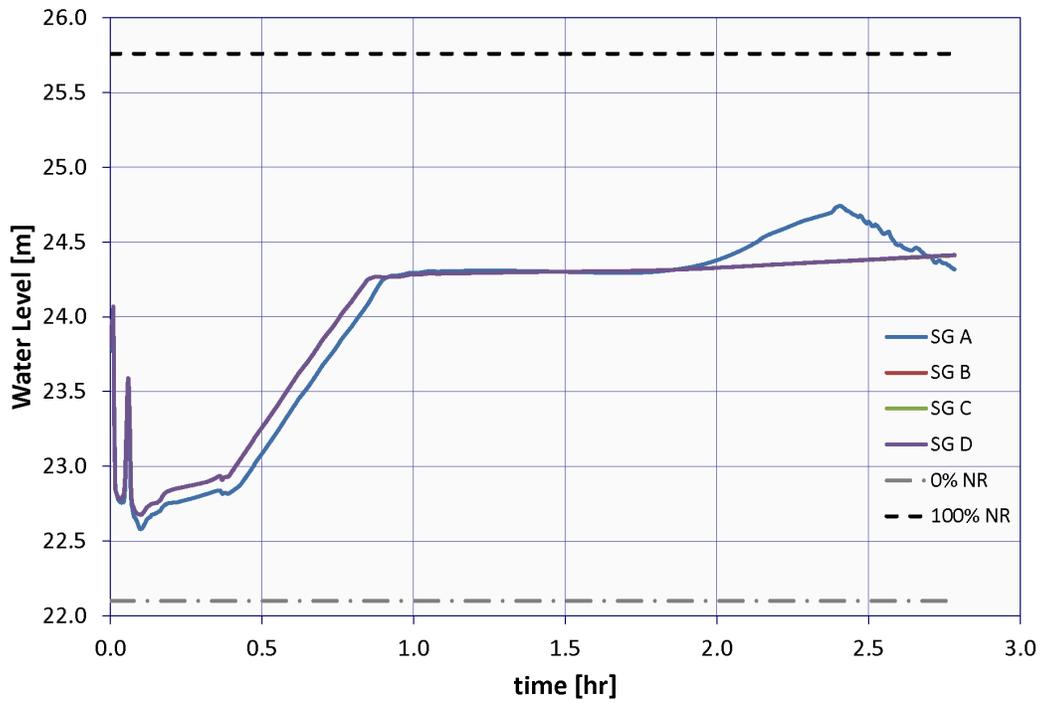
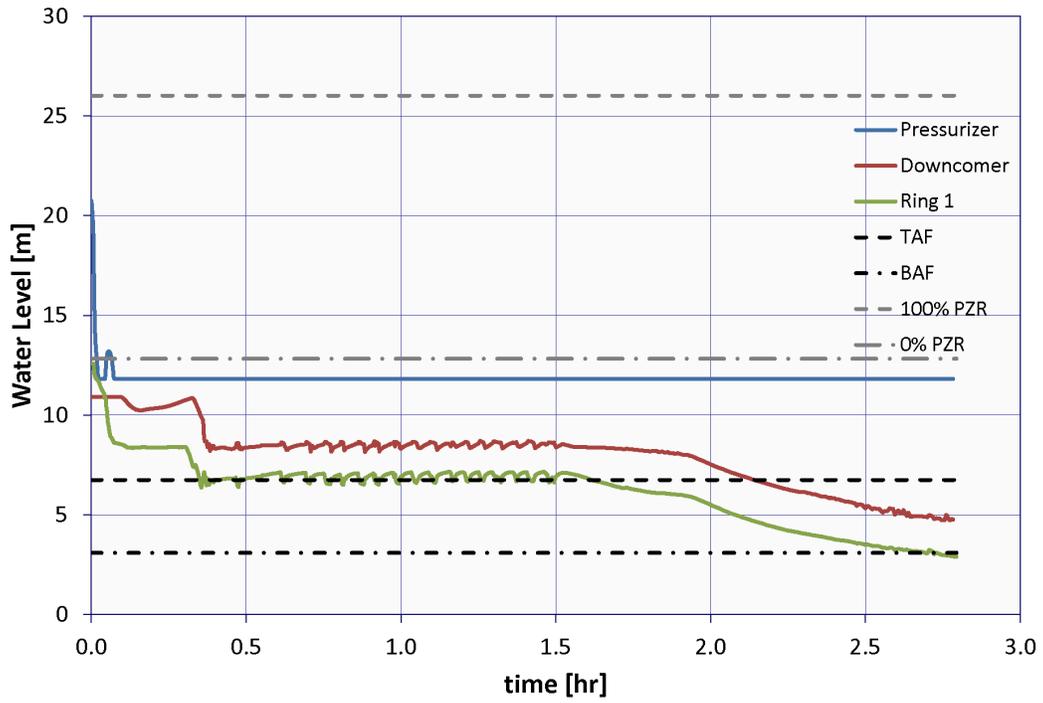


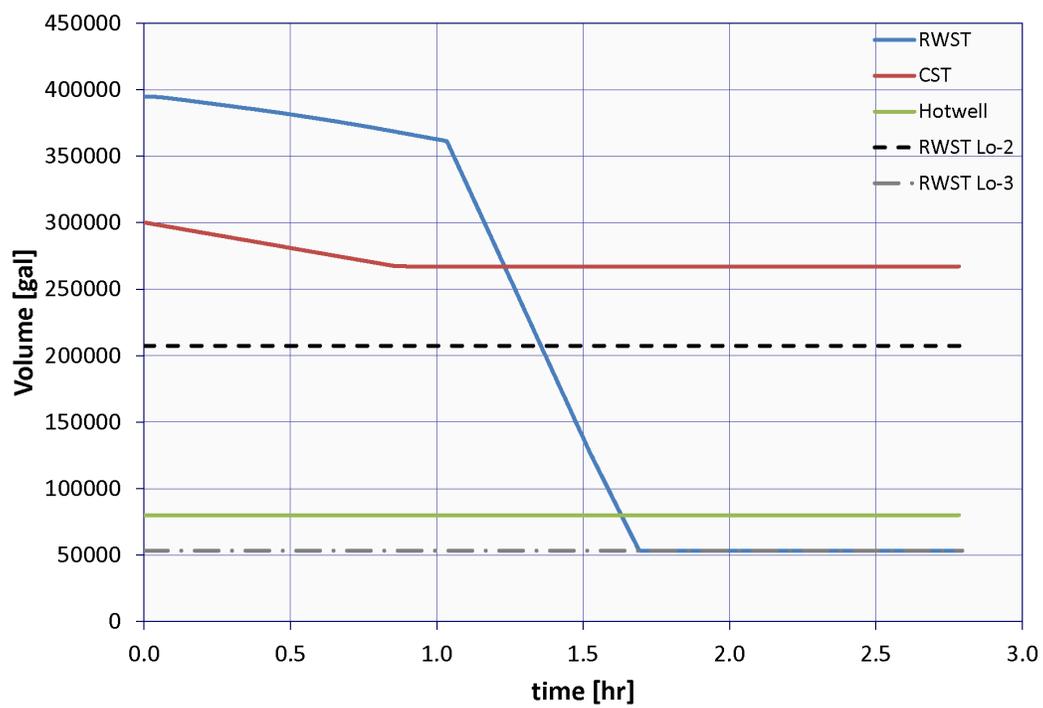




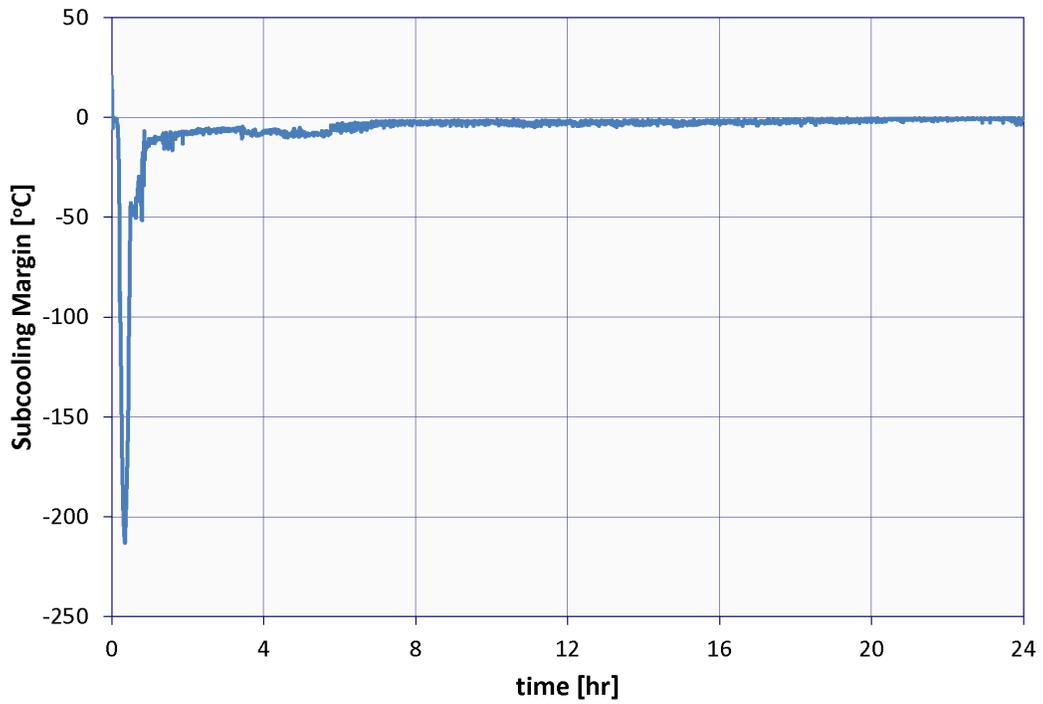
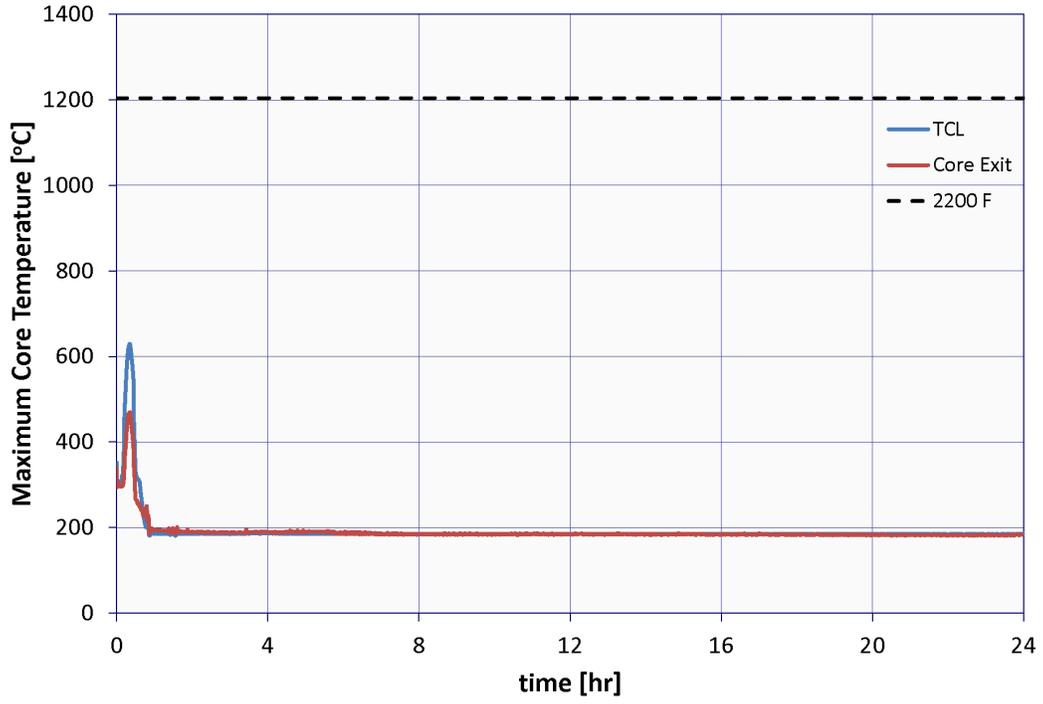


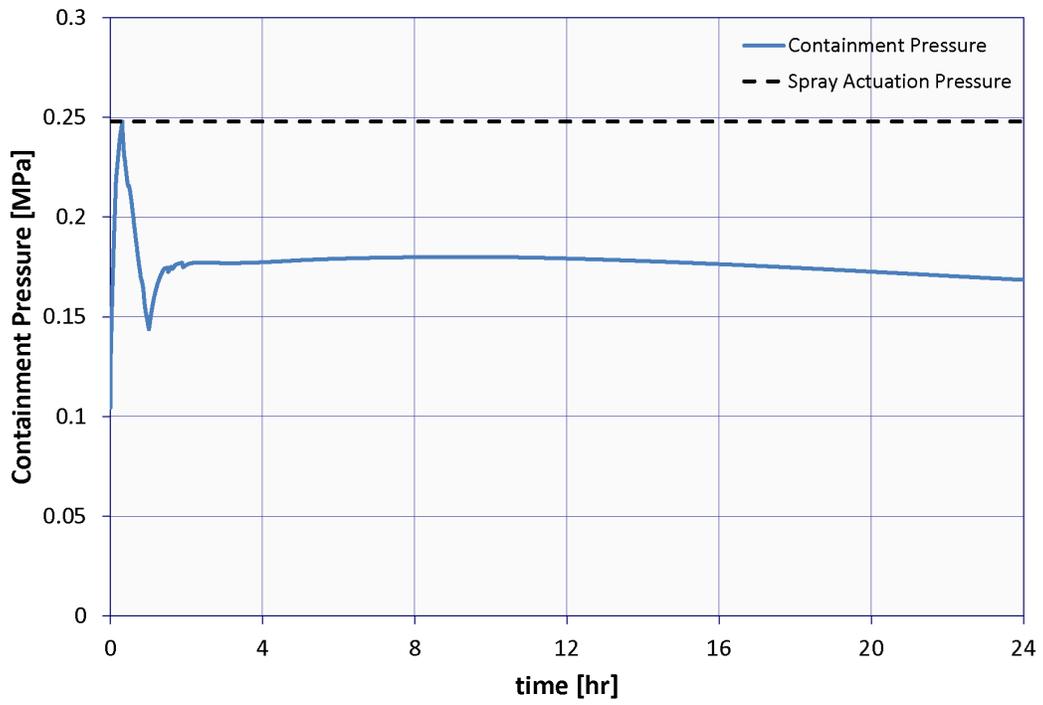
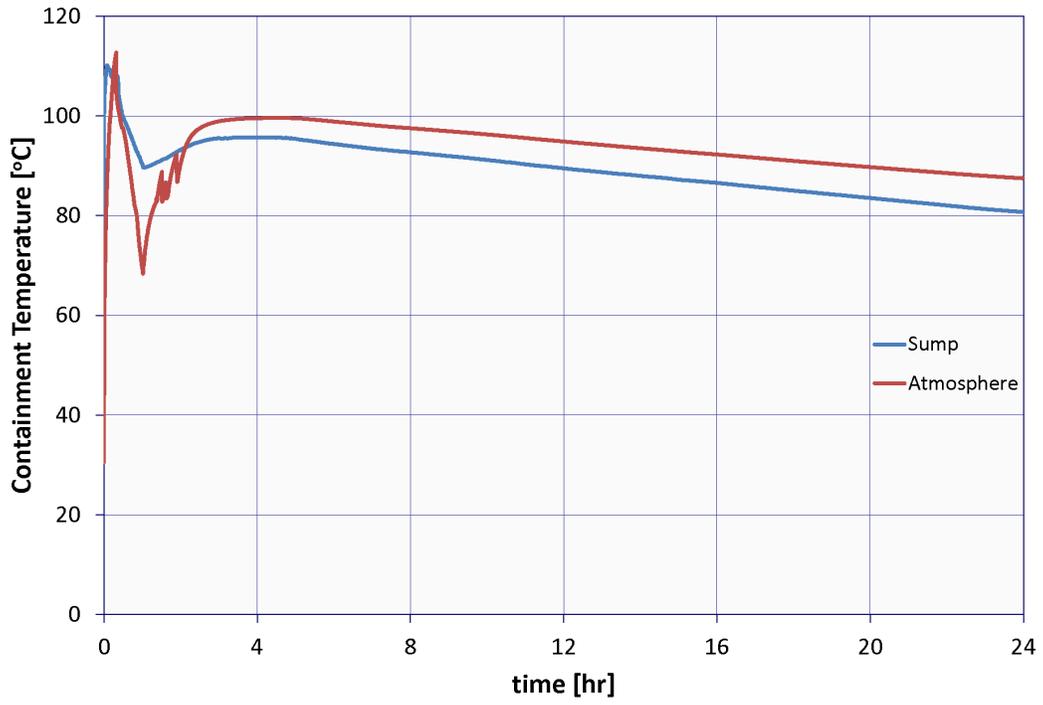


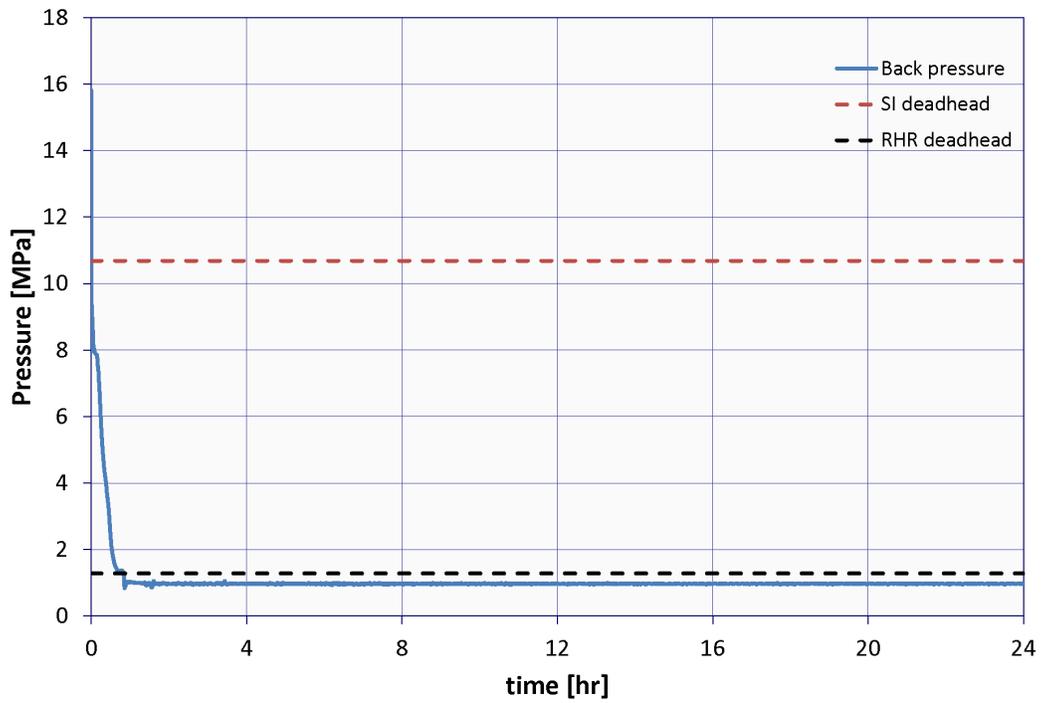
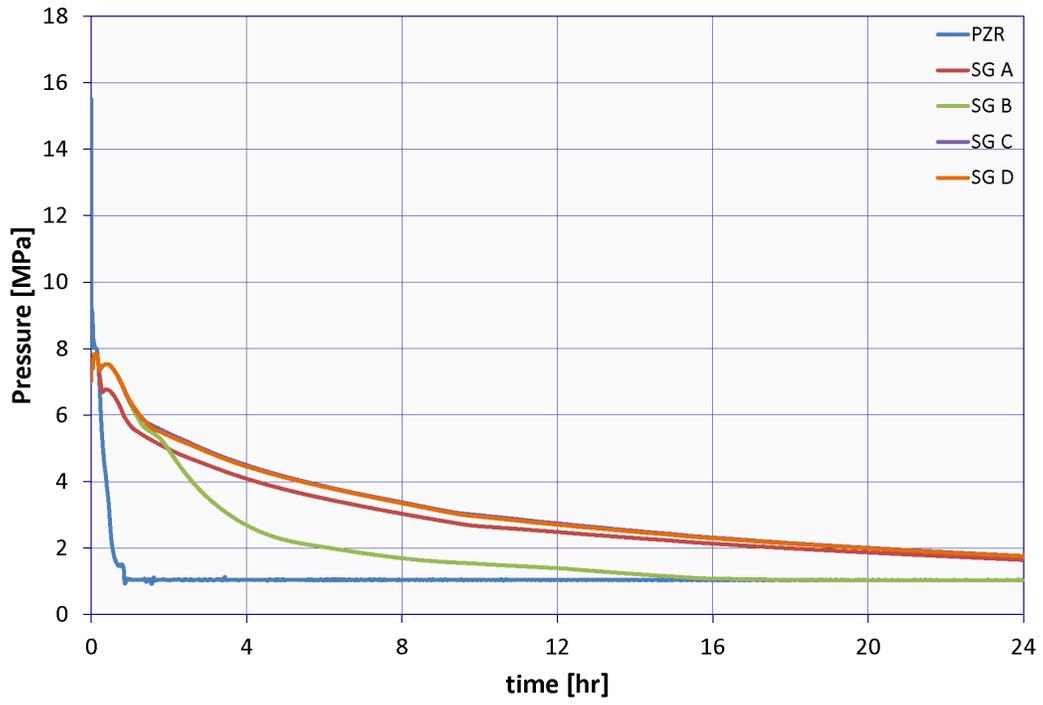


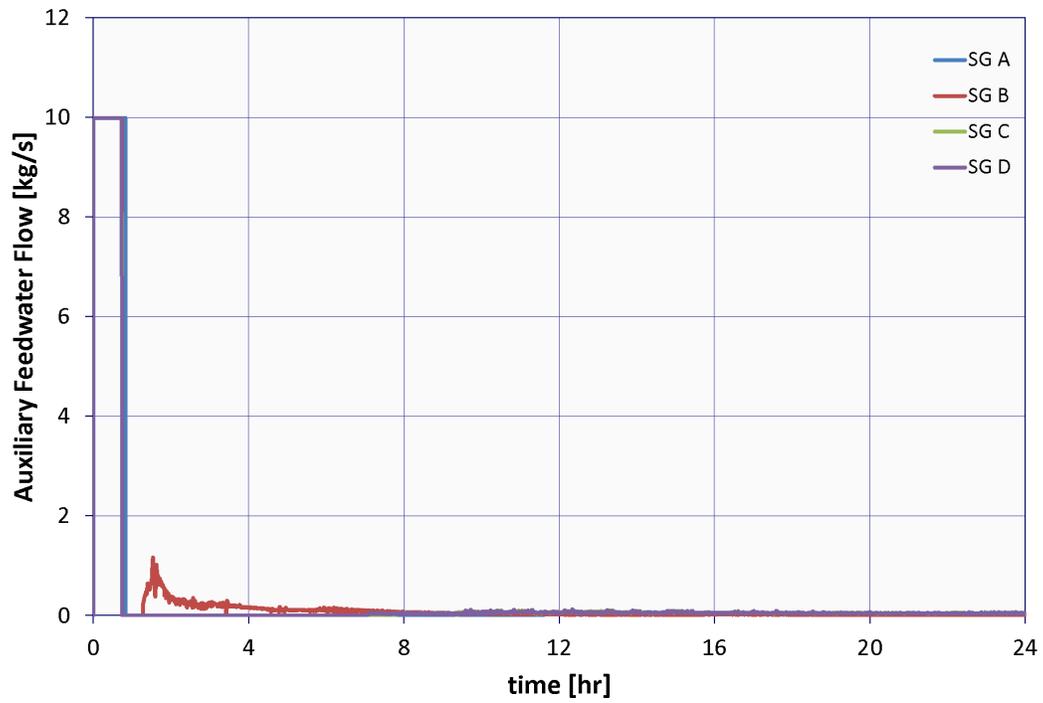
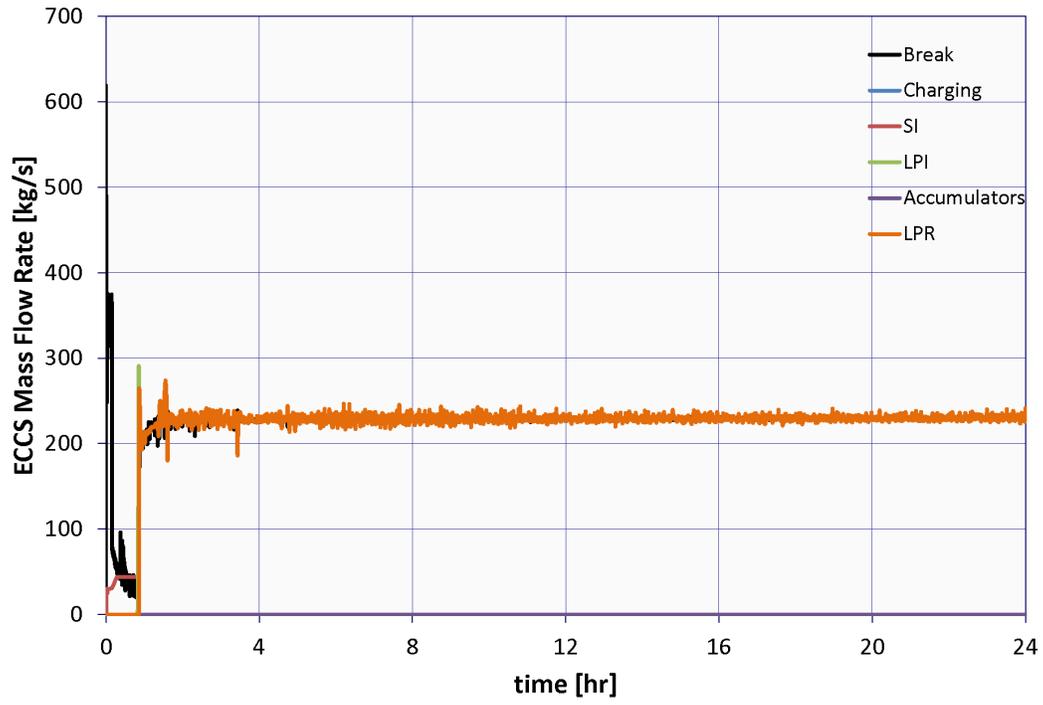


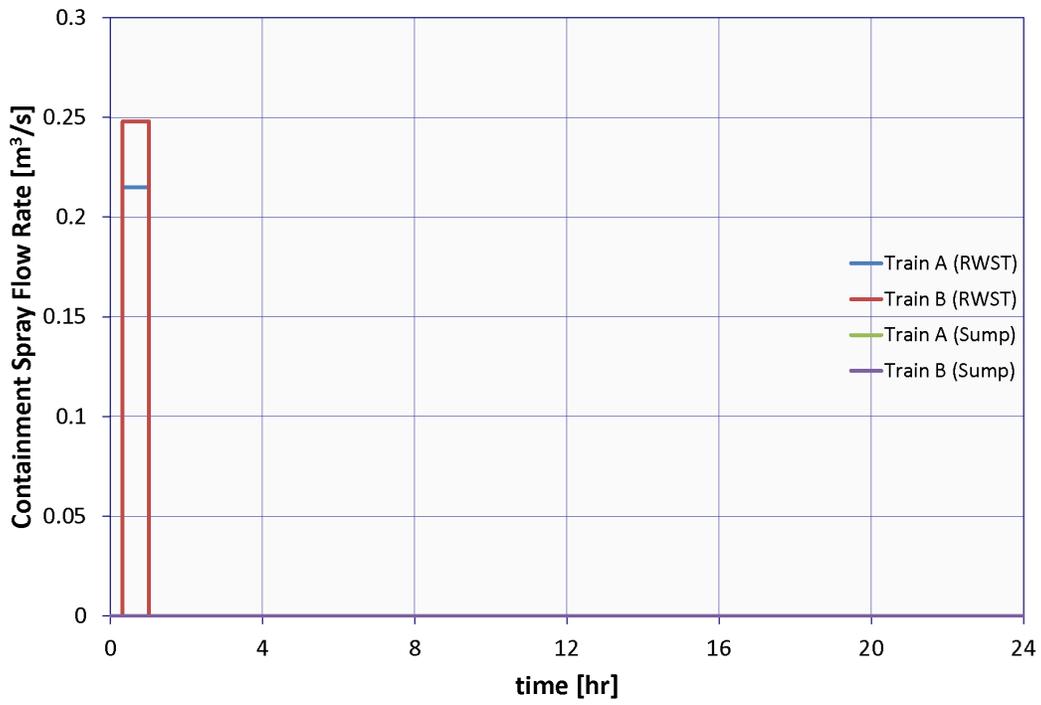
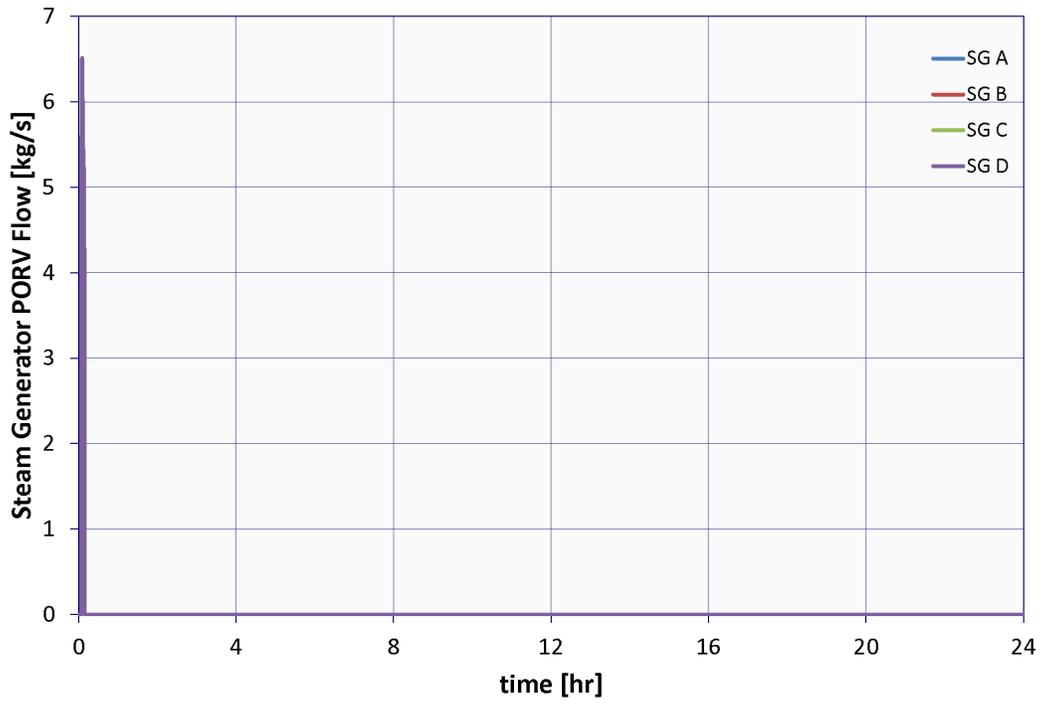
E.1.2 Case 2: 3.33-in. Break with 1/2 SI, 1/2 RHR, 2/2 CS Pumps

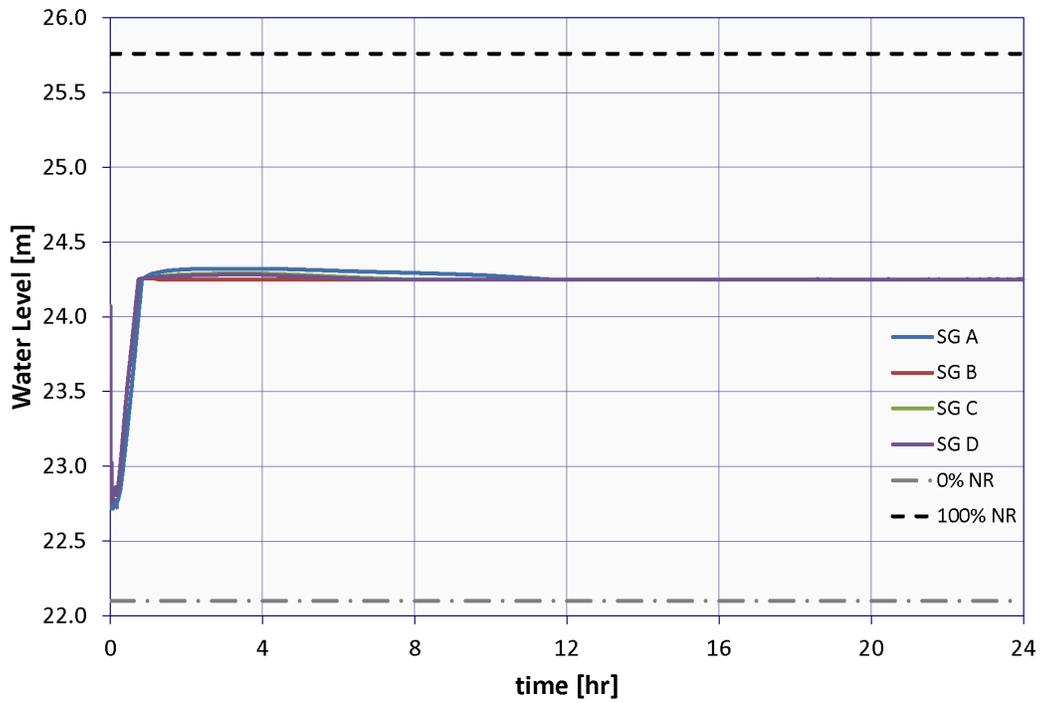
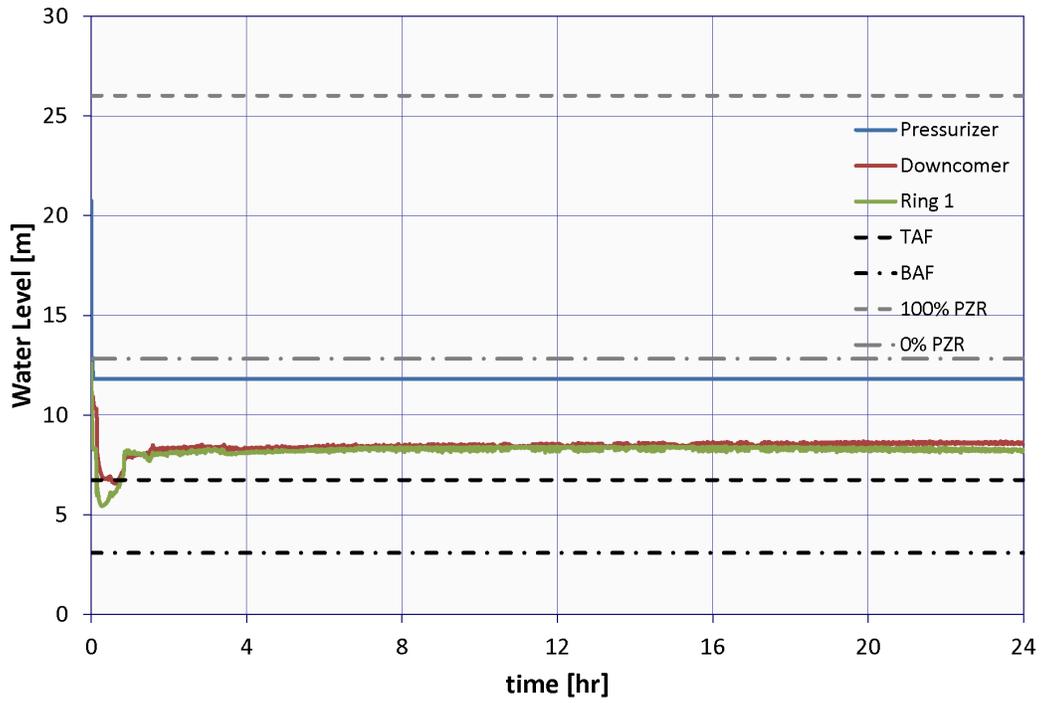


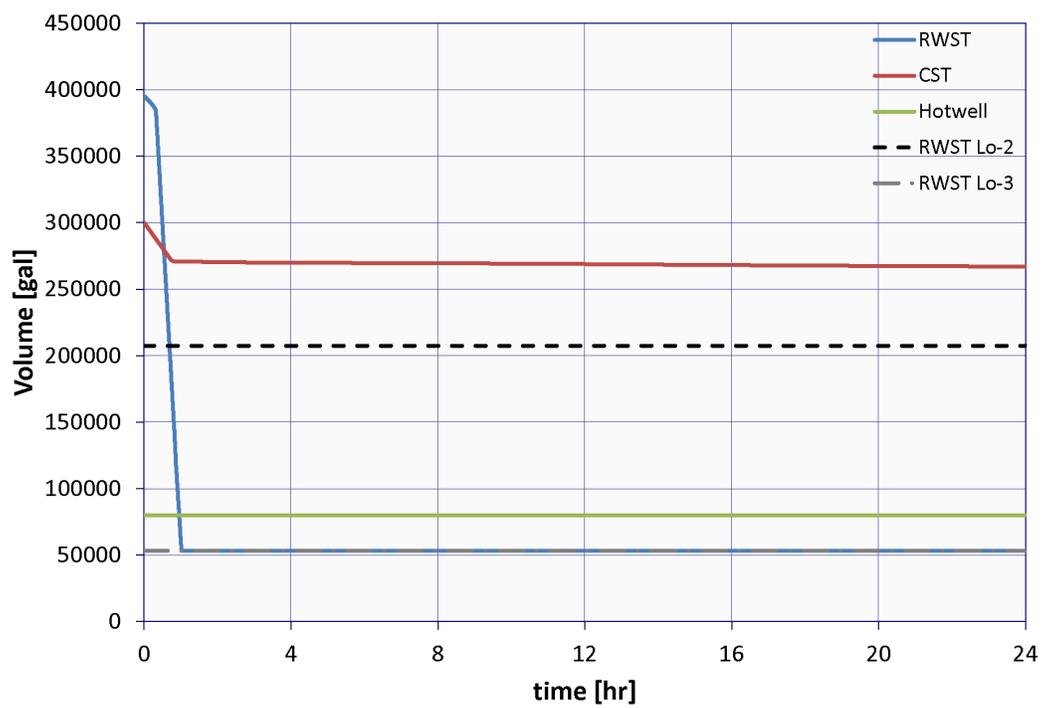




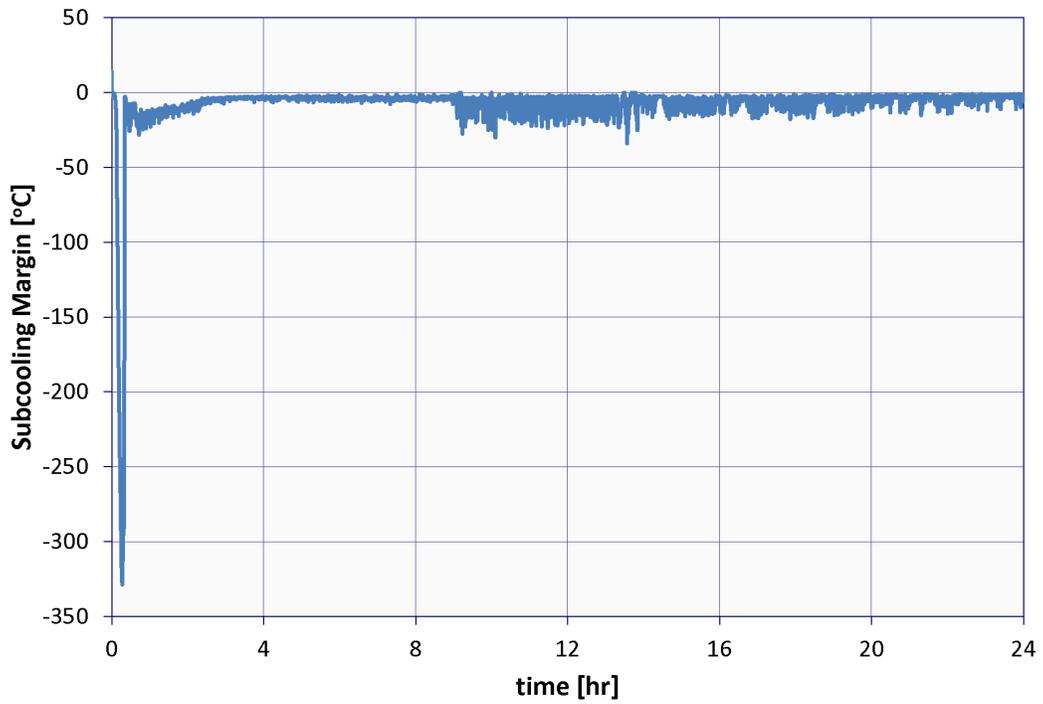
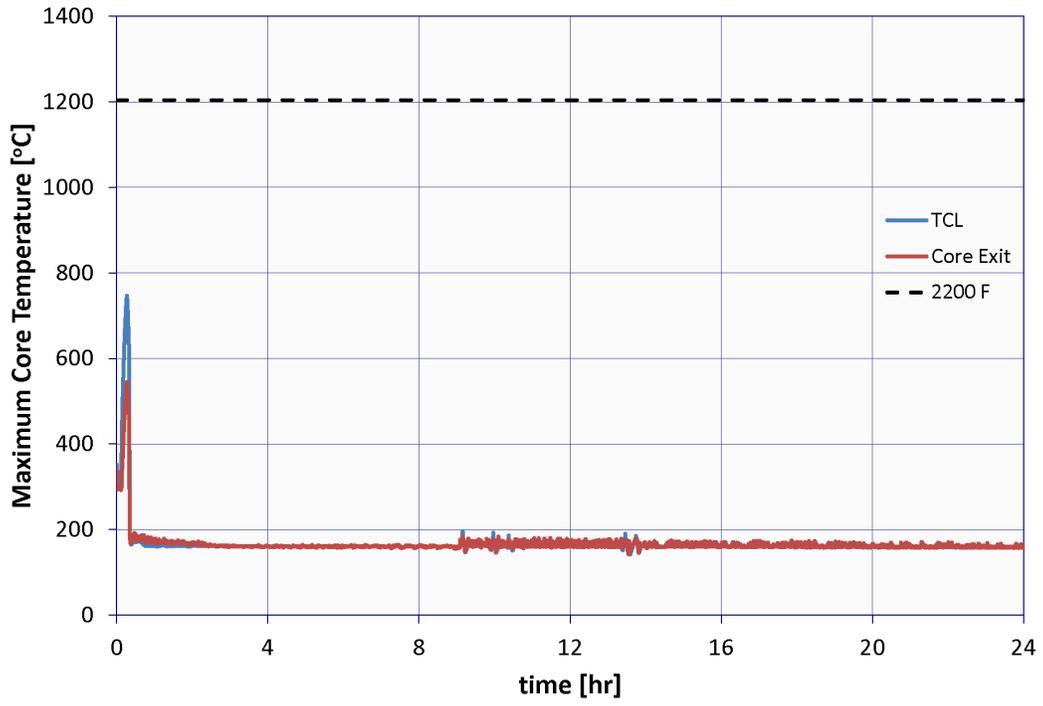


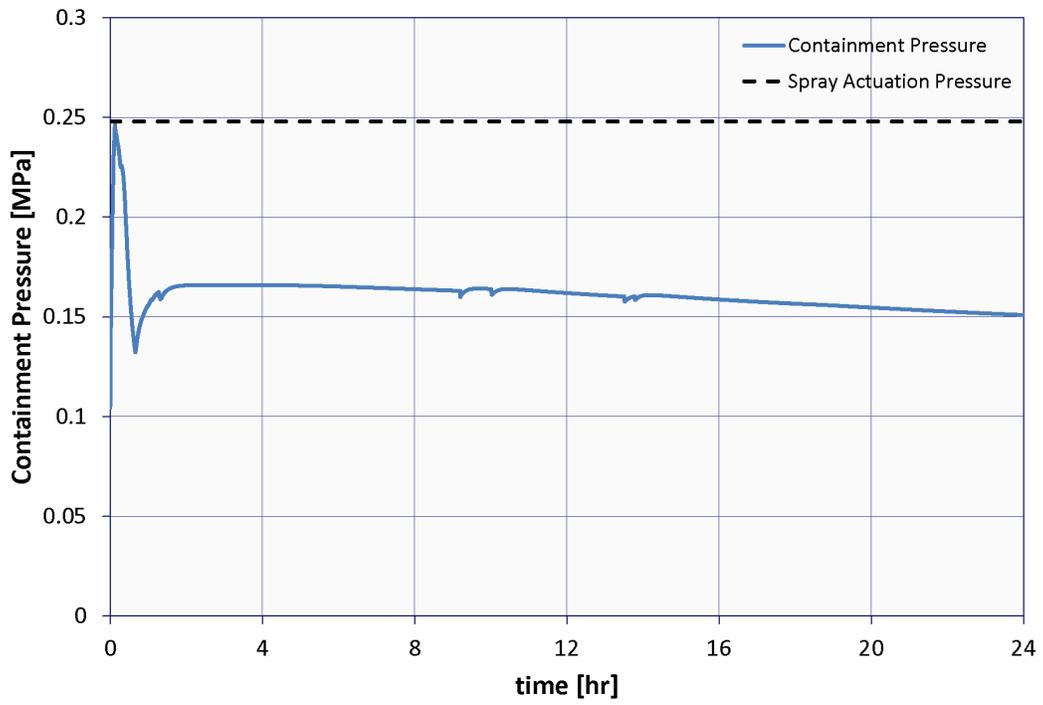
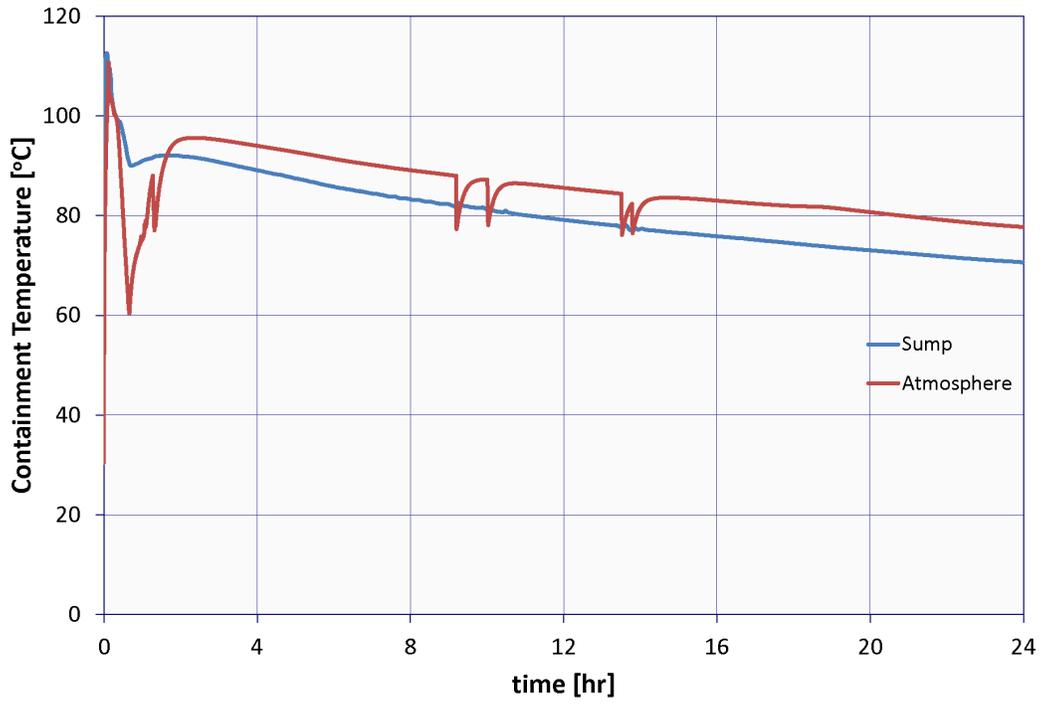


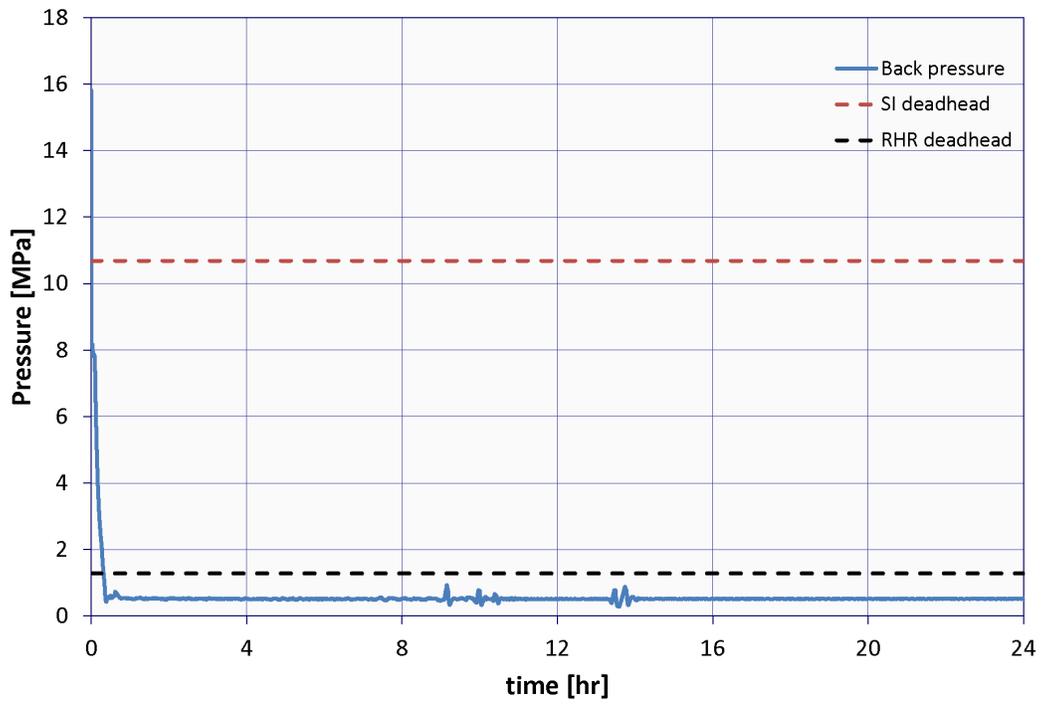
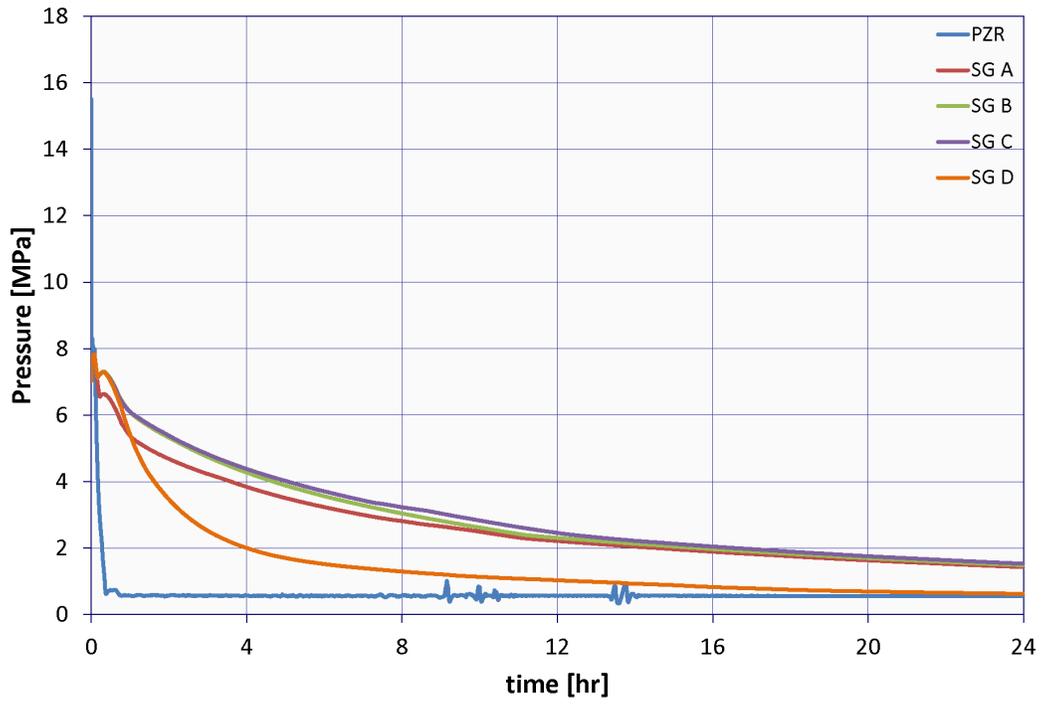


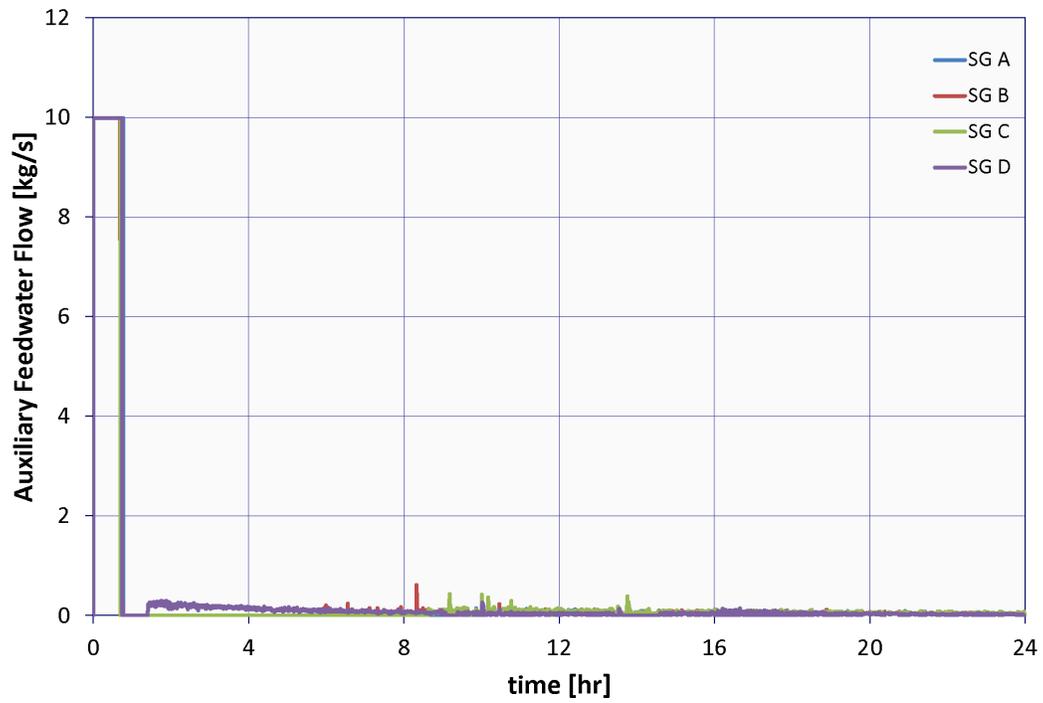
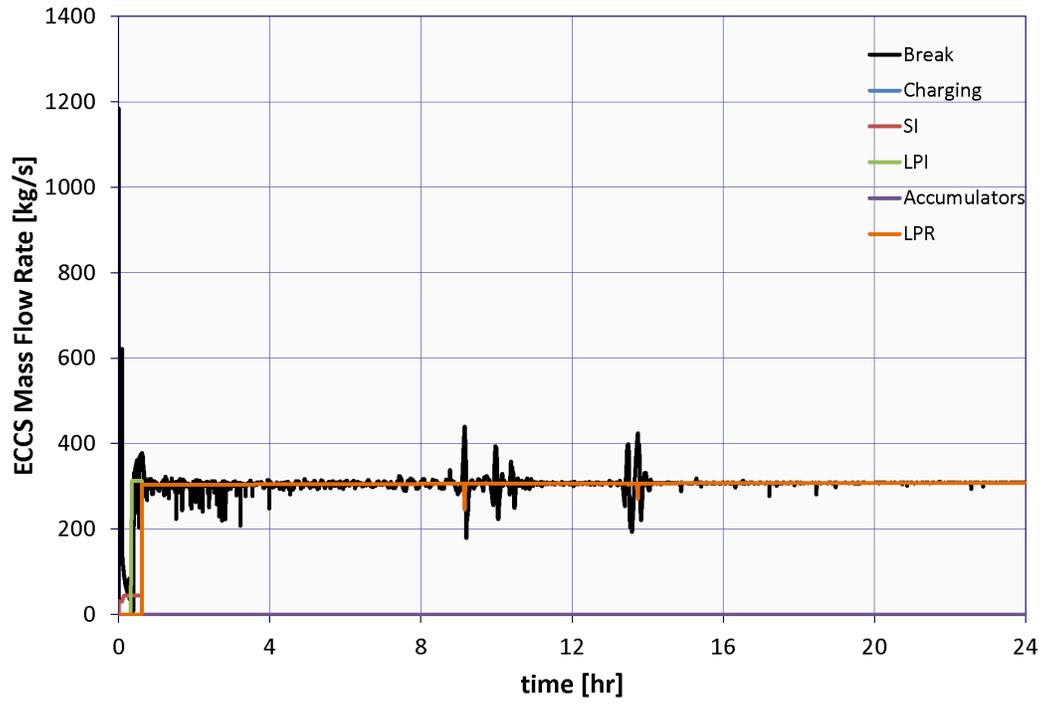


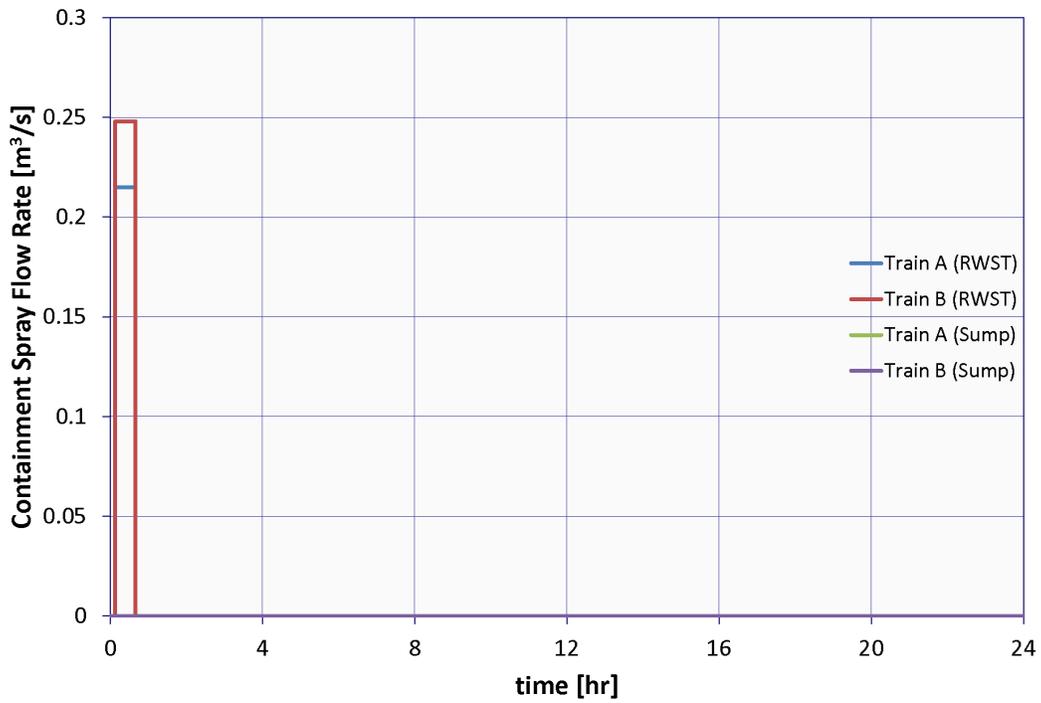
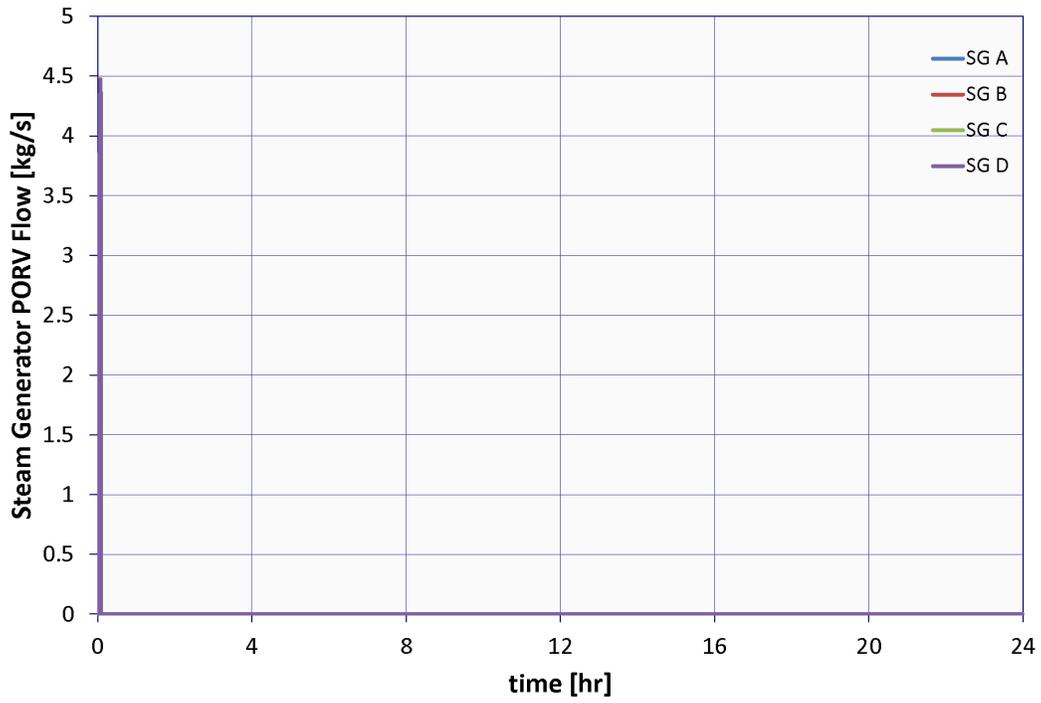
E.1.3 Case 3: 4.67-in. Break with 1/2 SI, 1/2 RHR, 2/2 CS Pumps

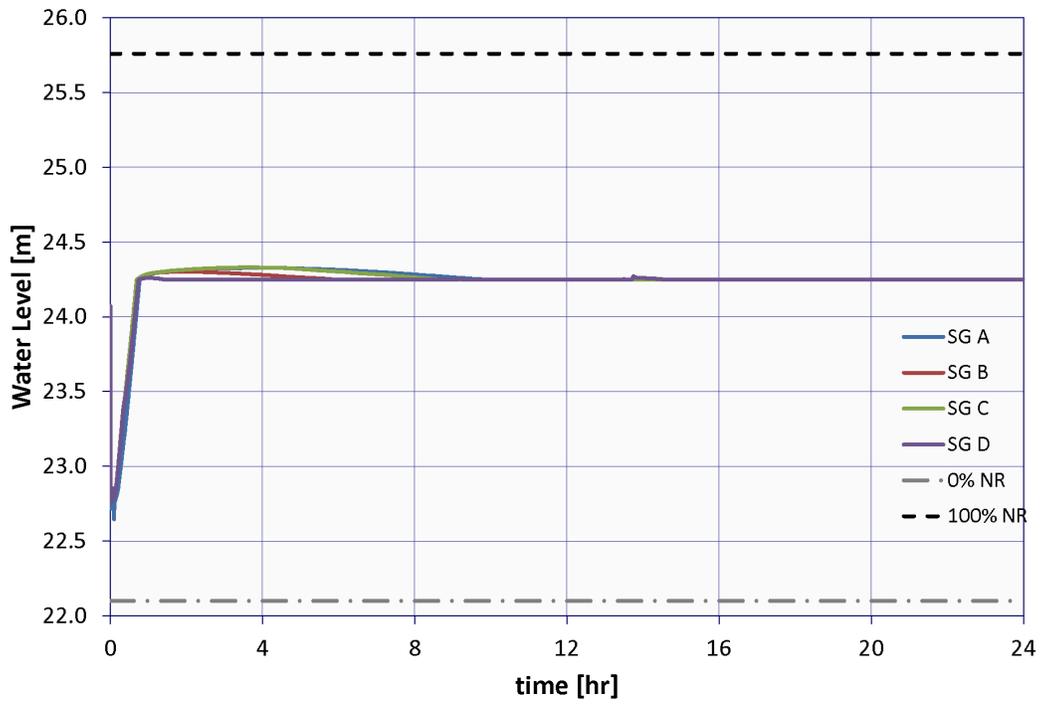
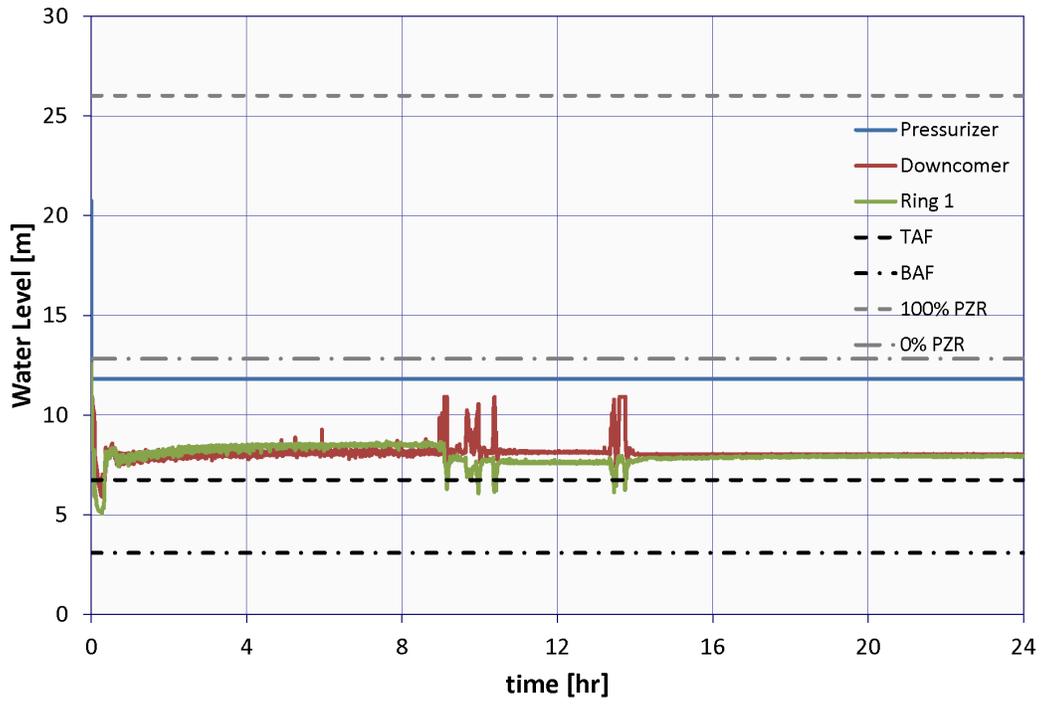


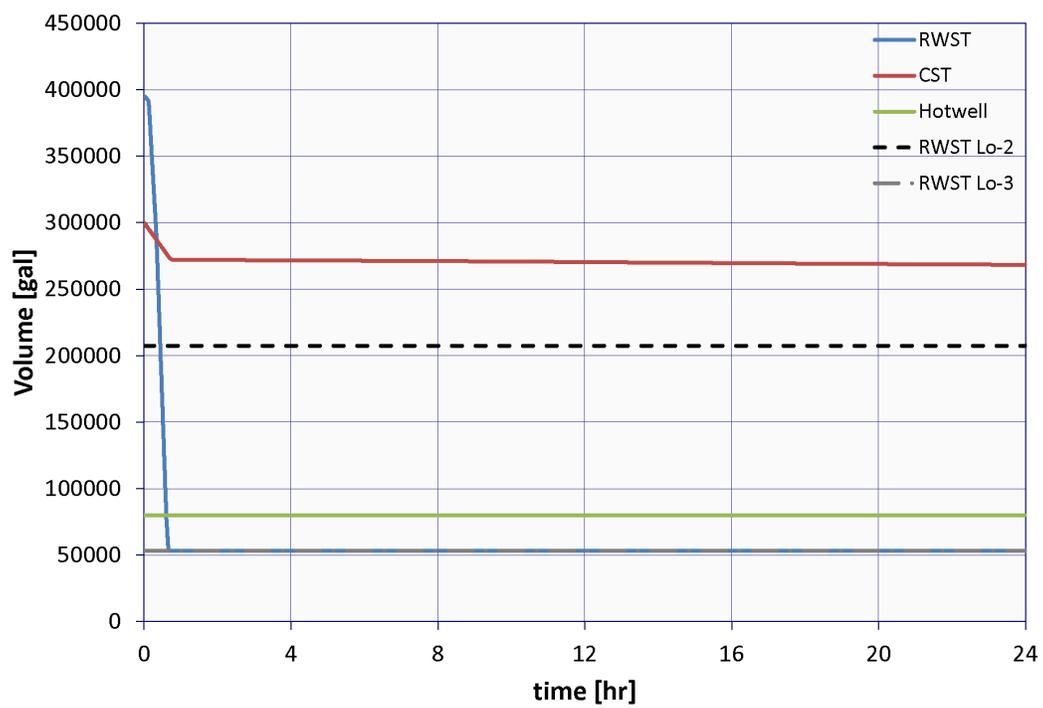




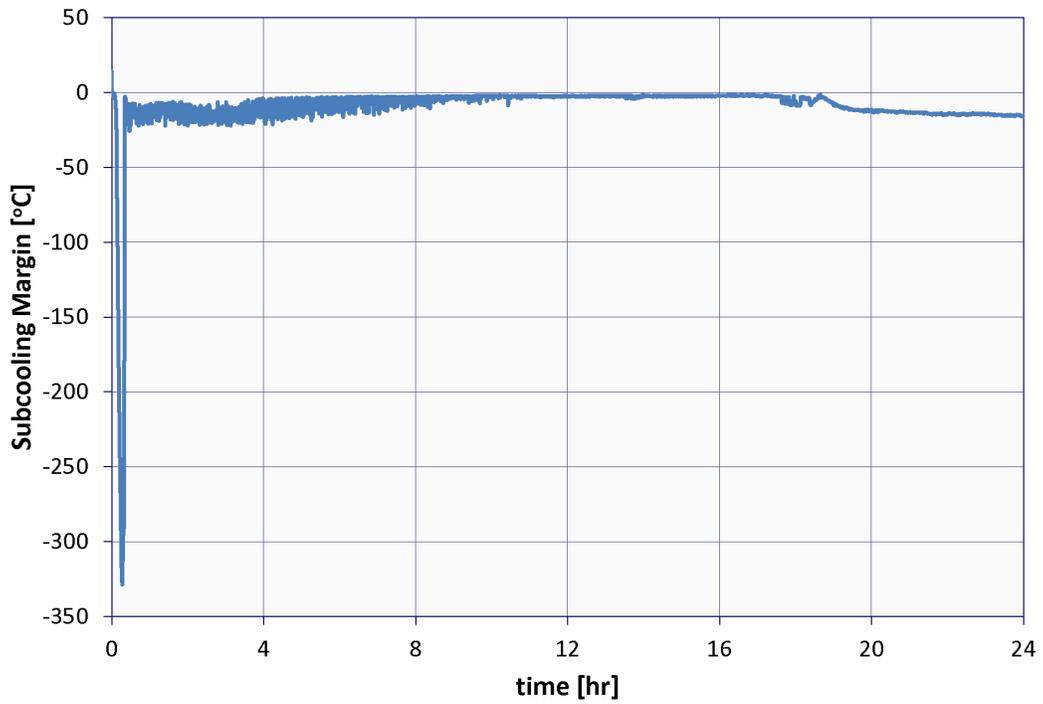
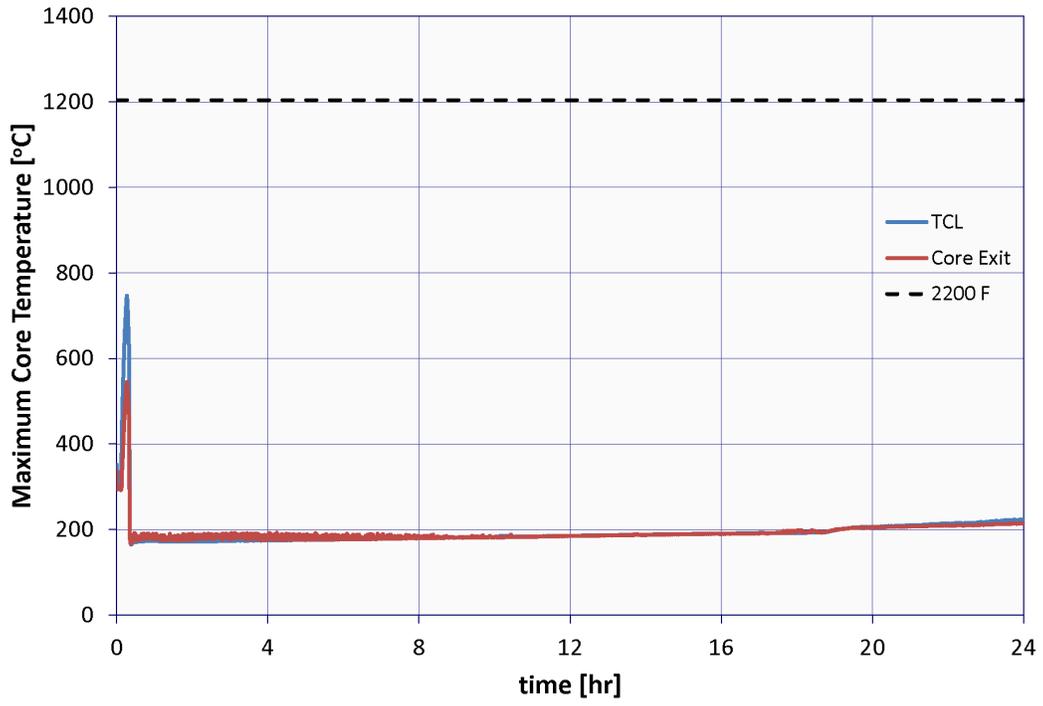


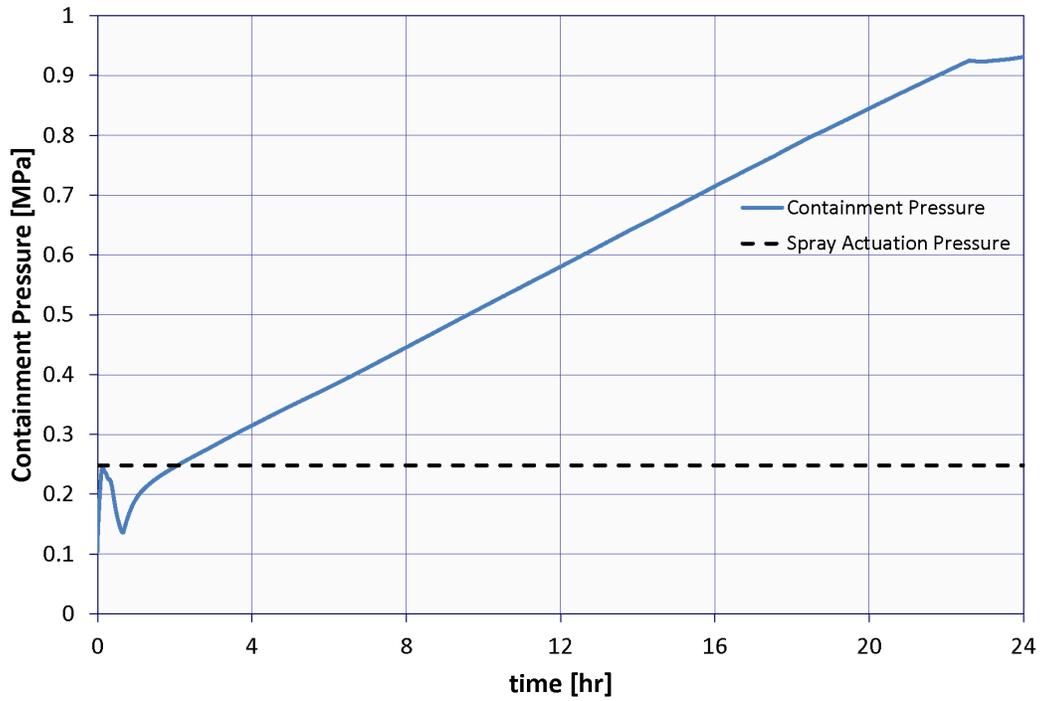
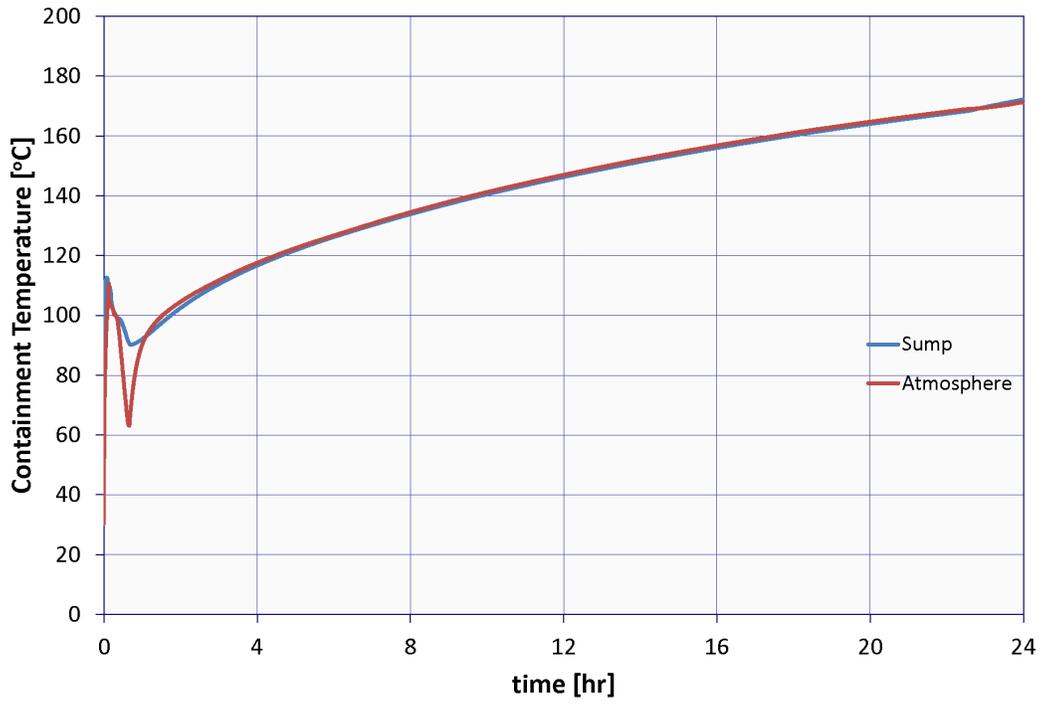


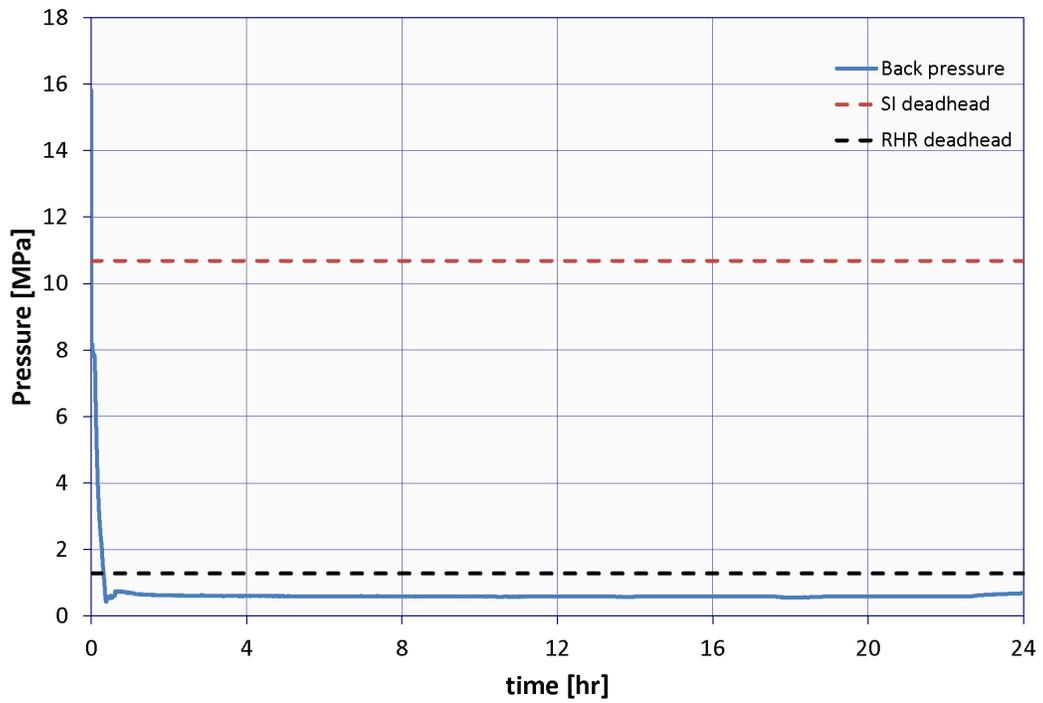
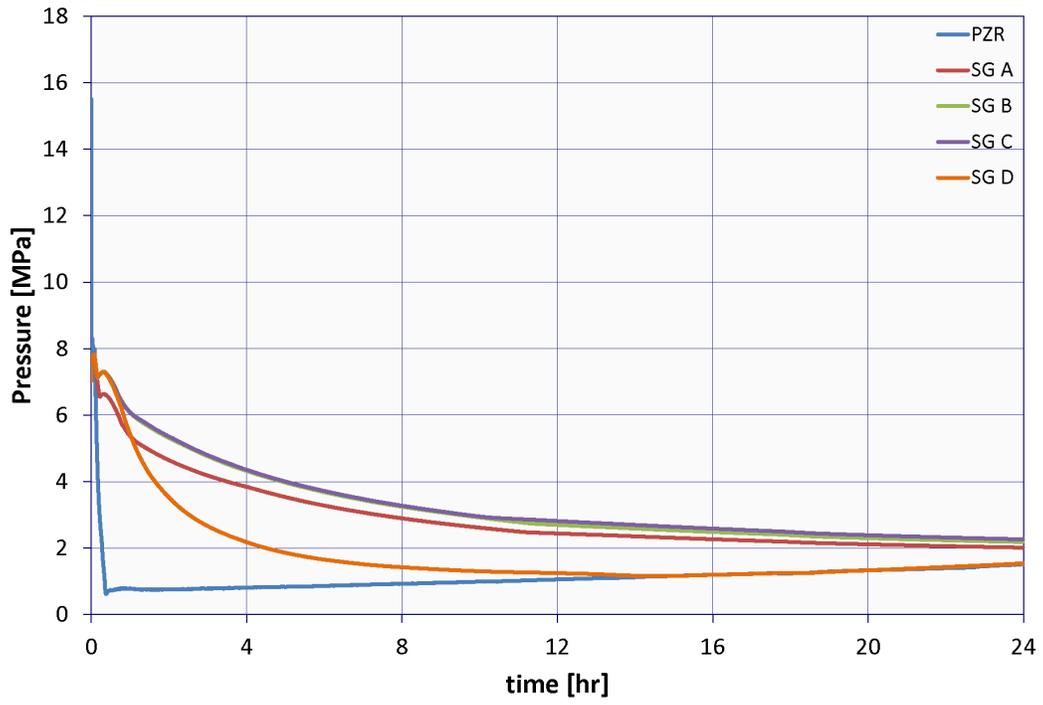


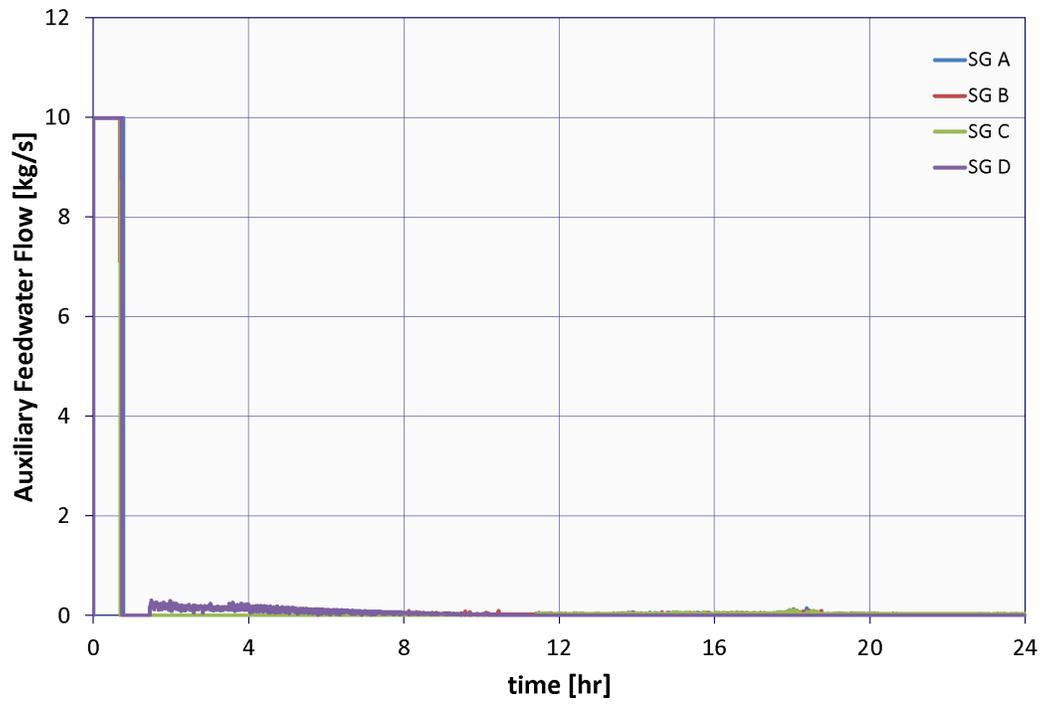
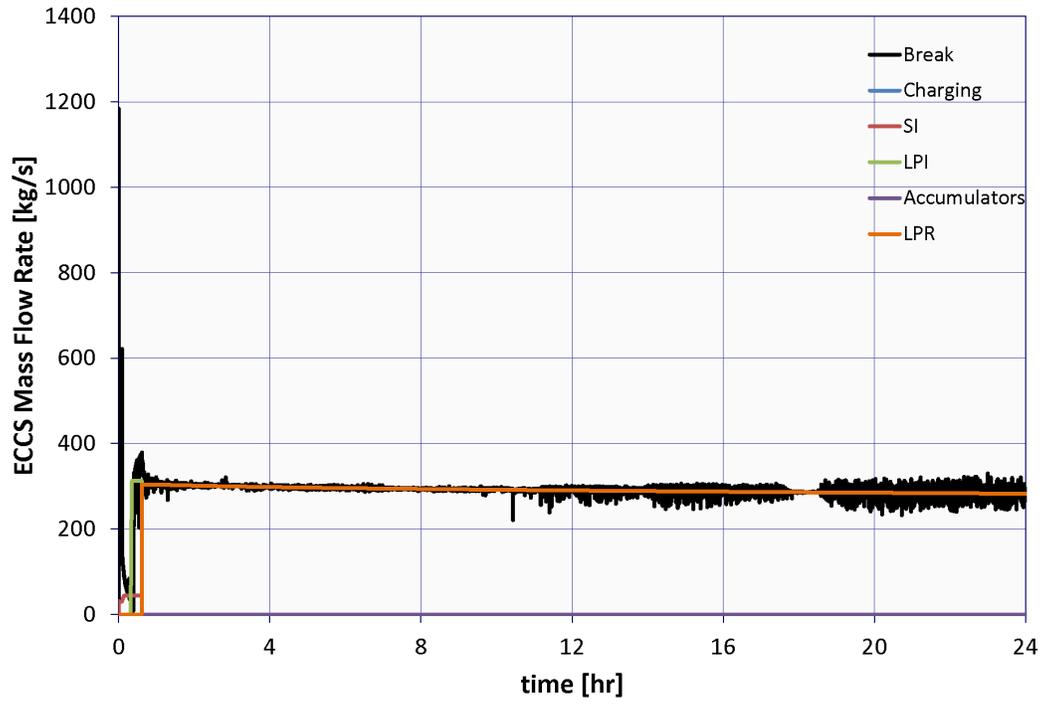


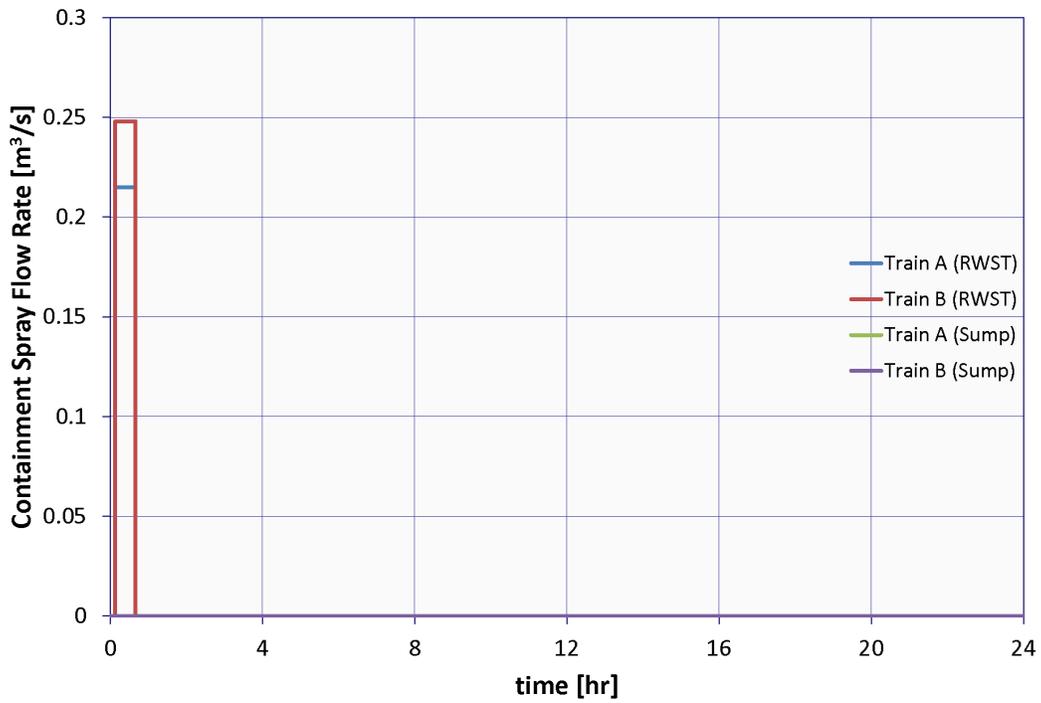
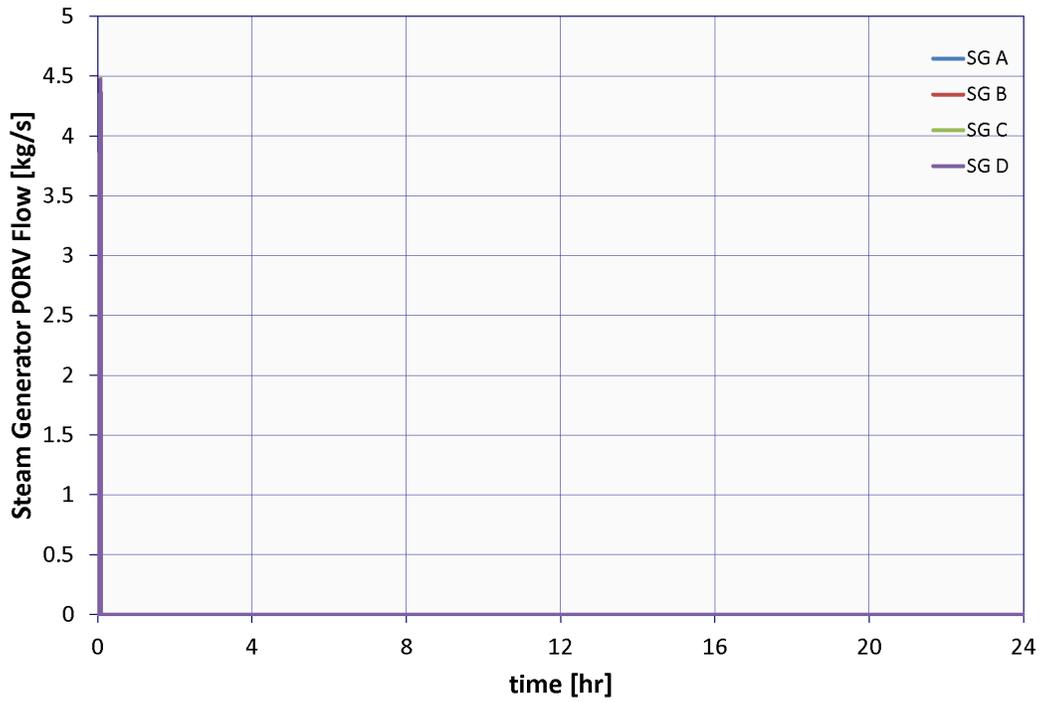
E.1.3.1 Case 3a: 4.67-in. Break with 1/2 SI, 1/2 RHR, 2/2 CS Pumps, No RHRHX

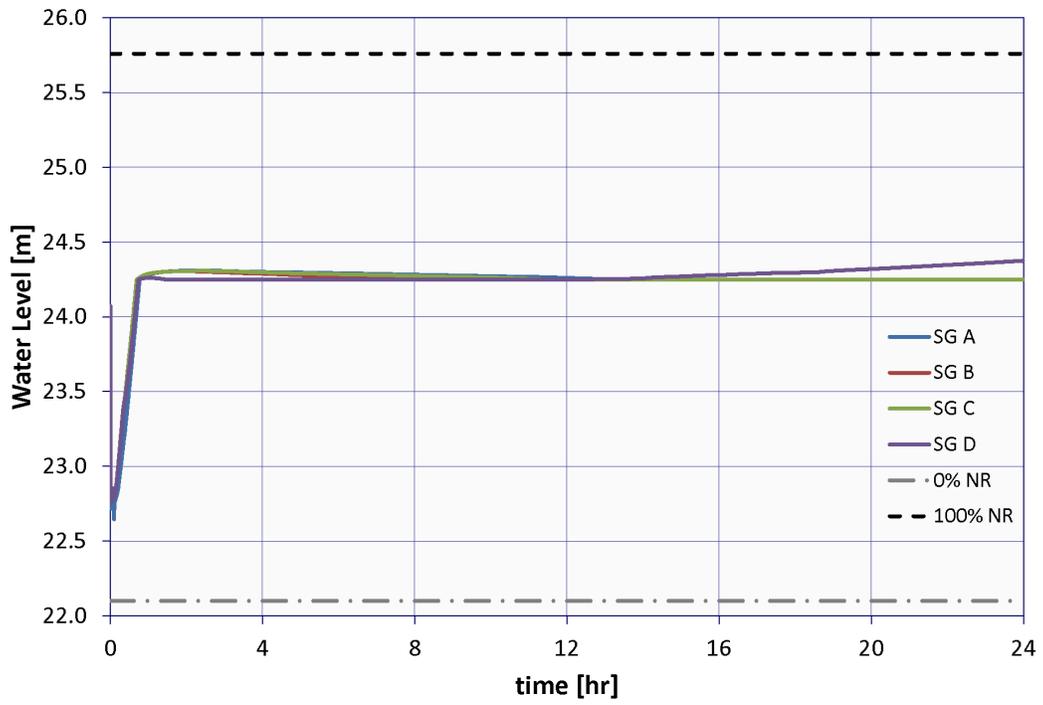
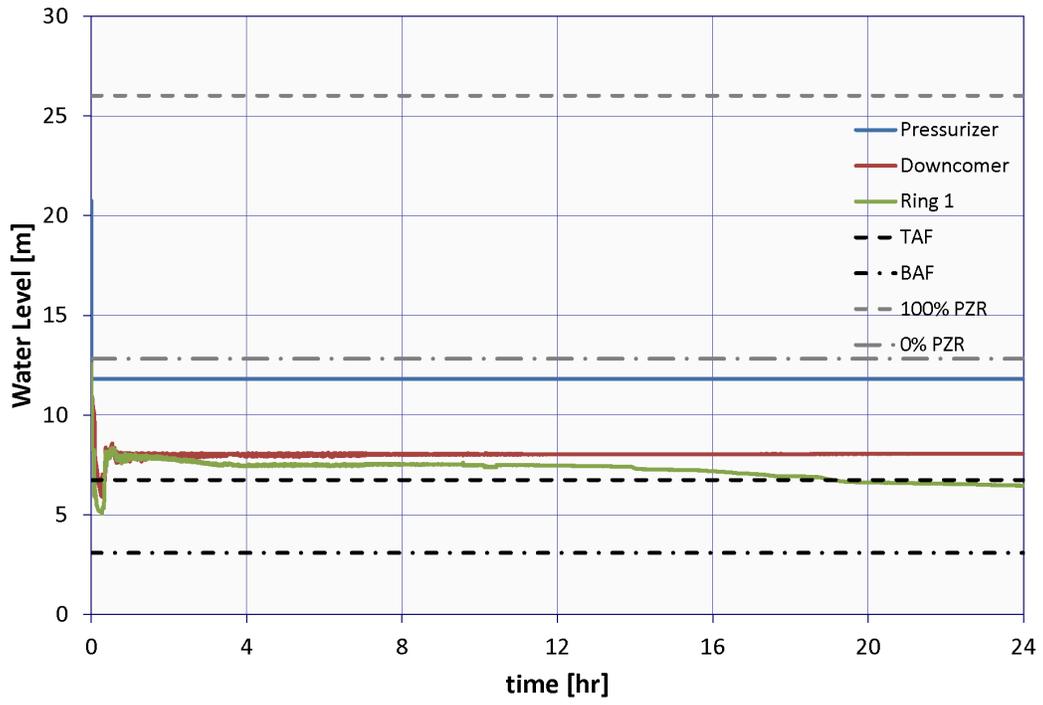


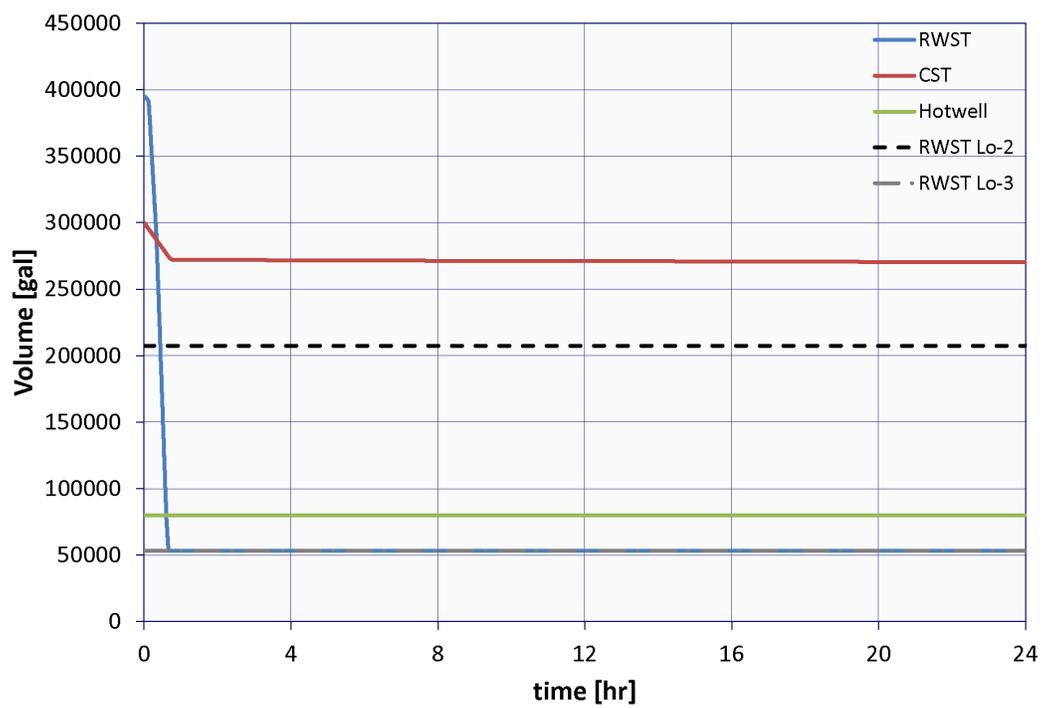




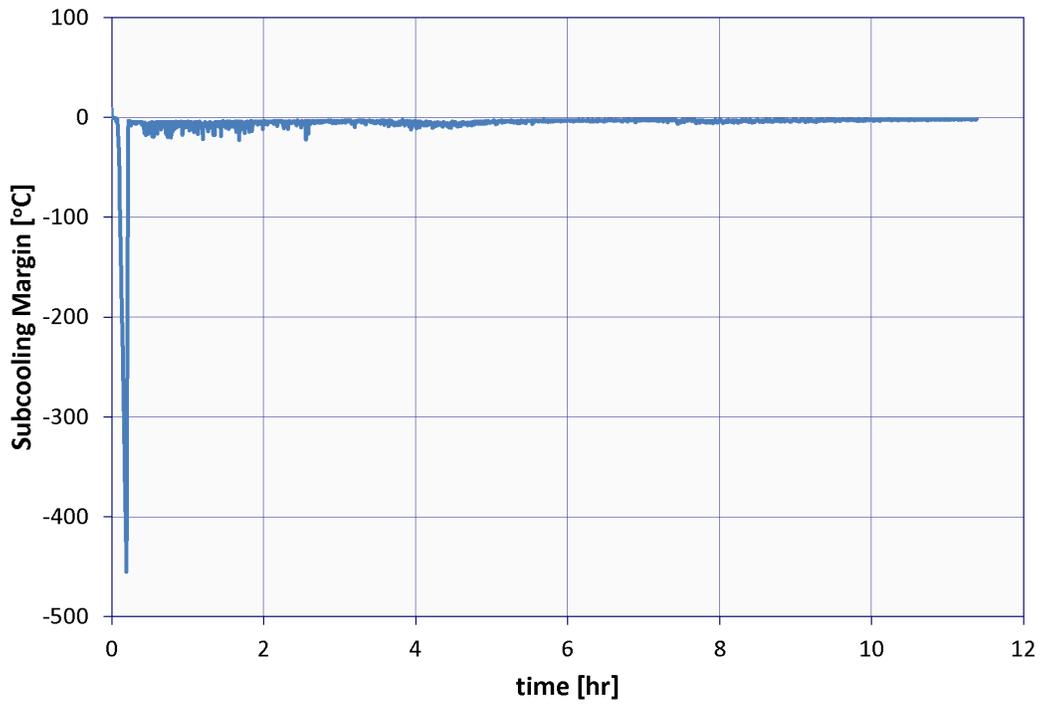
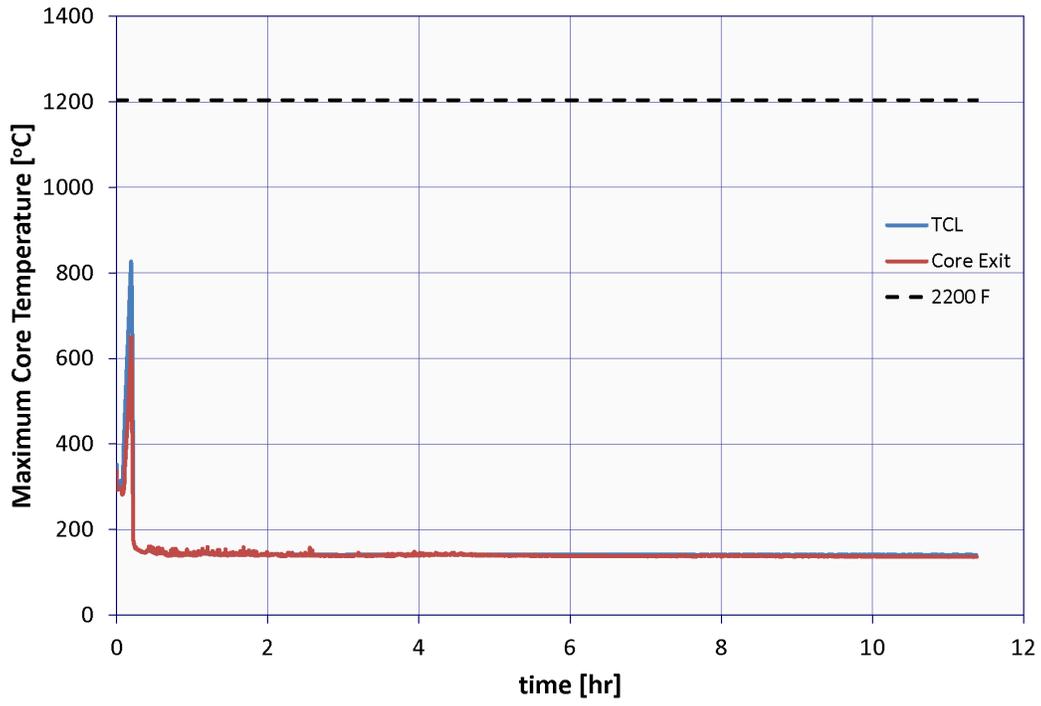


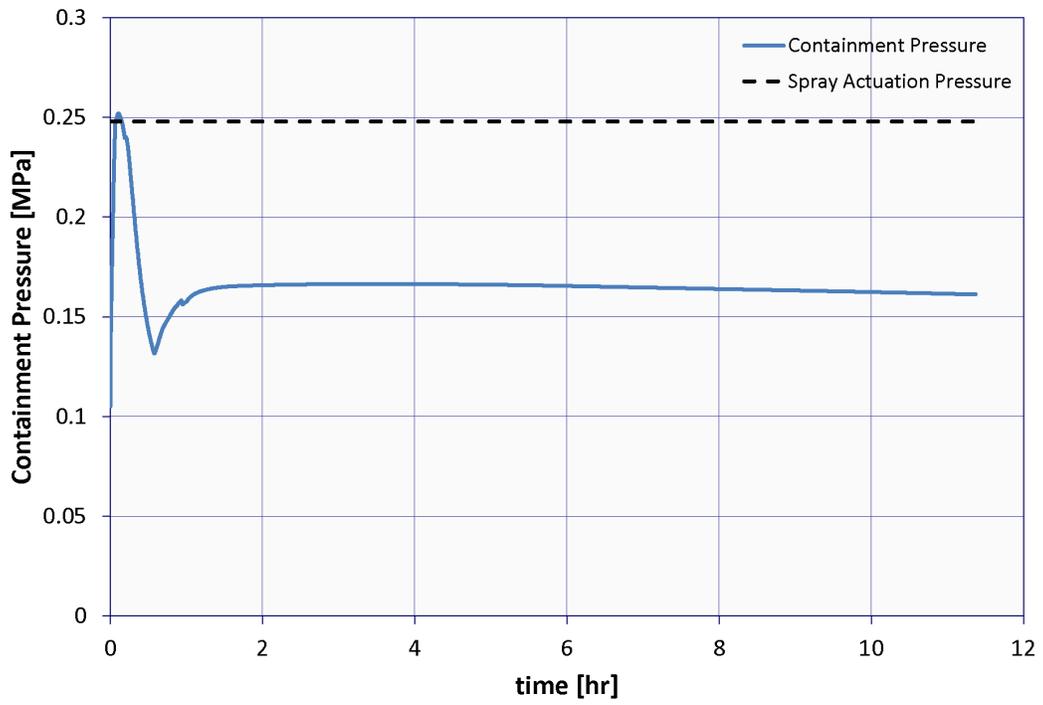
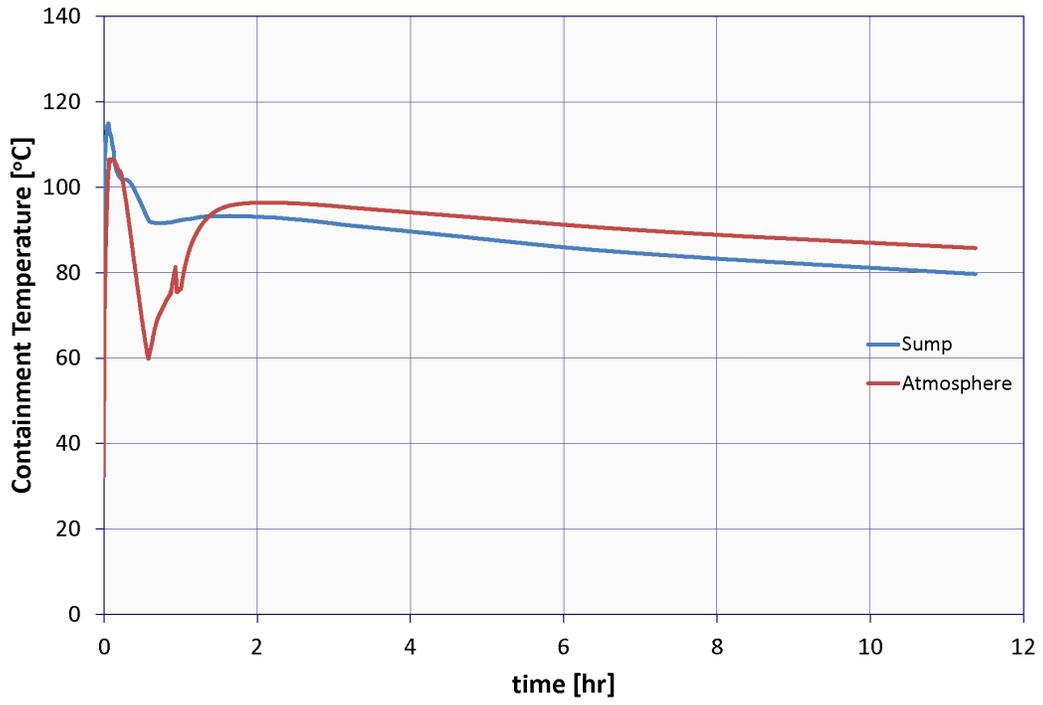


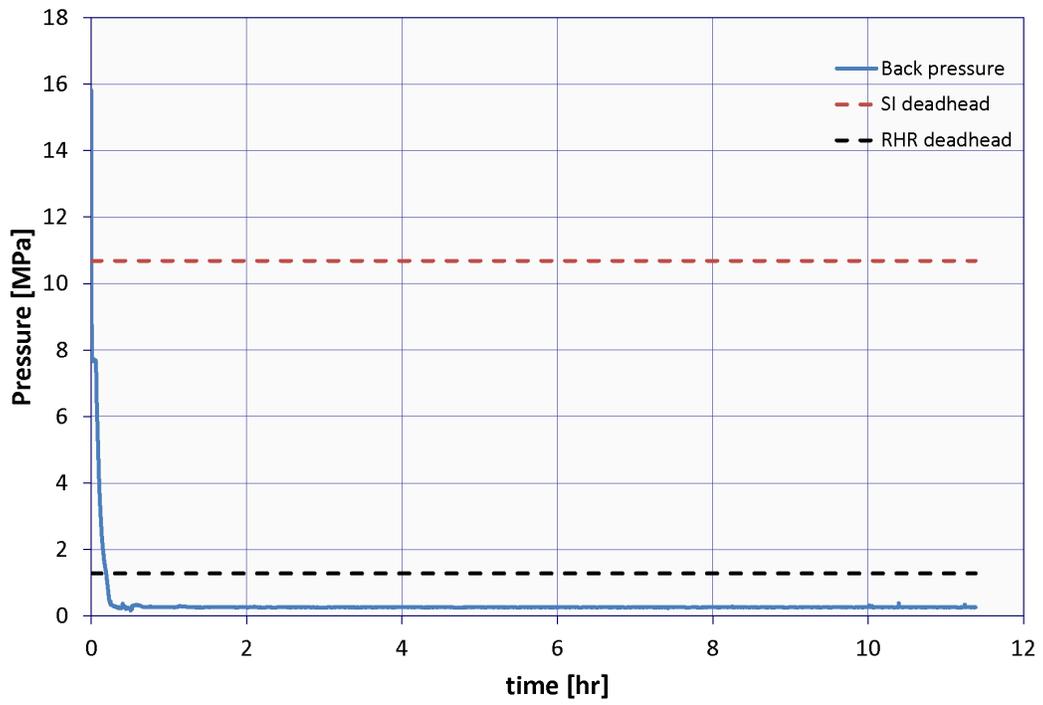
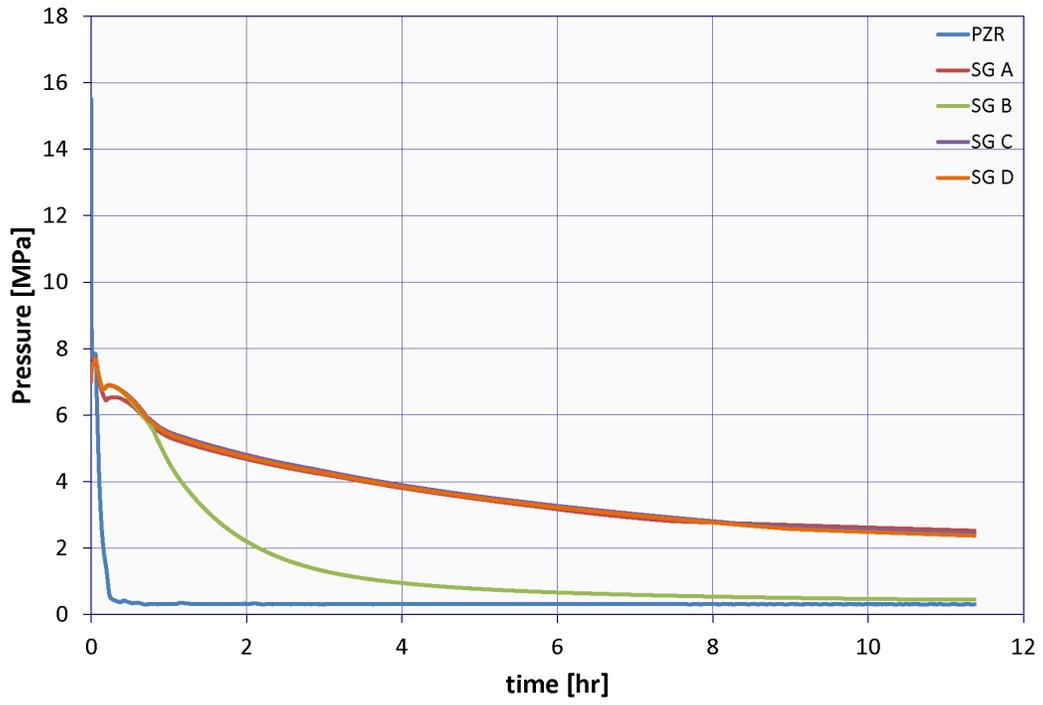


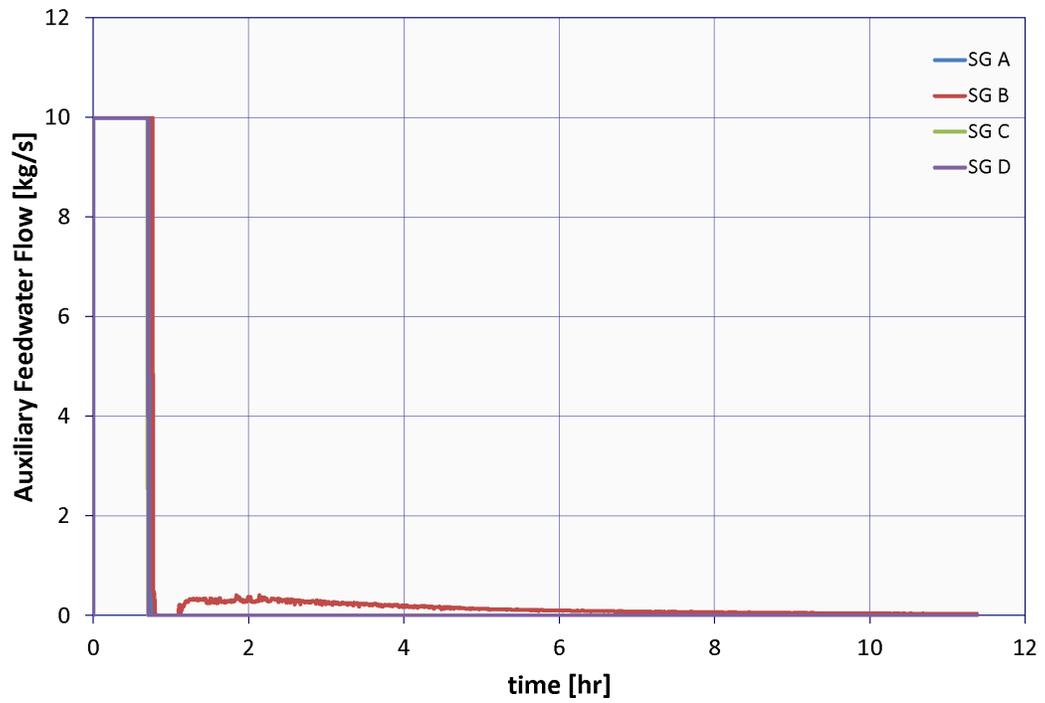
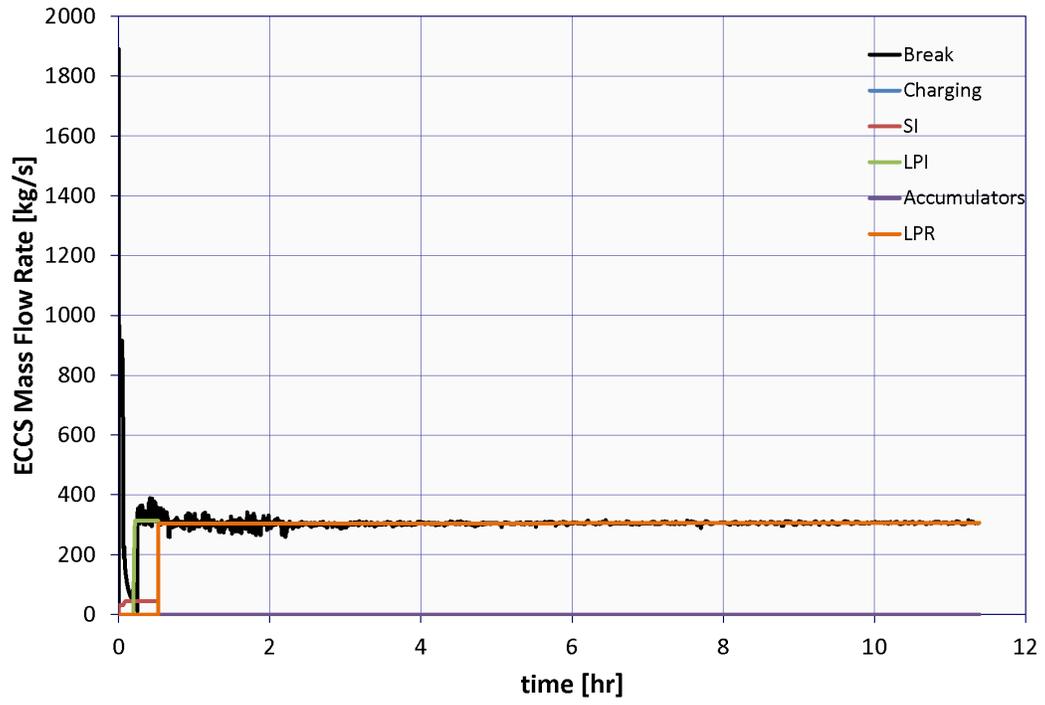


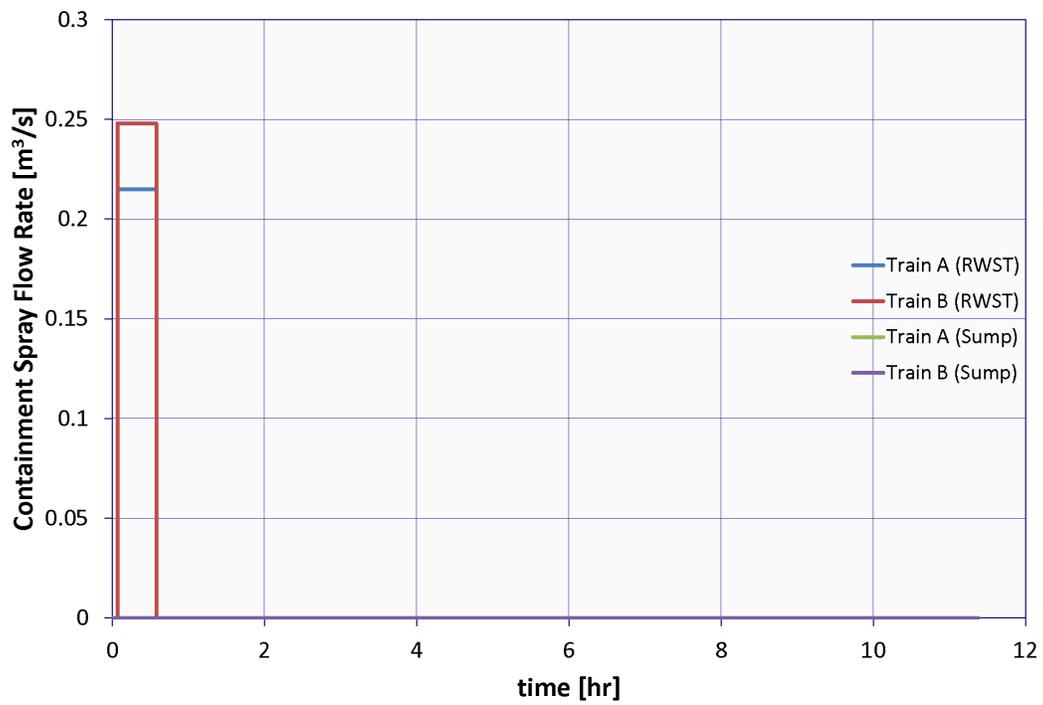
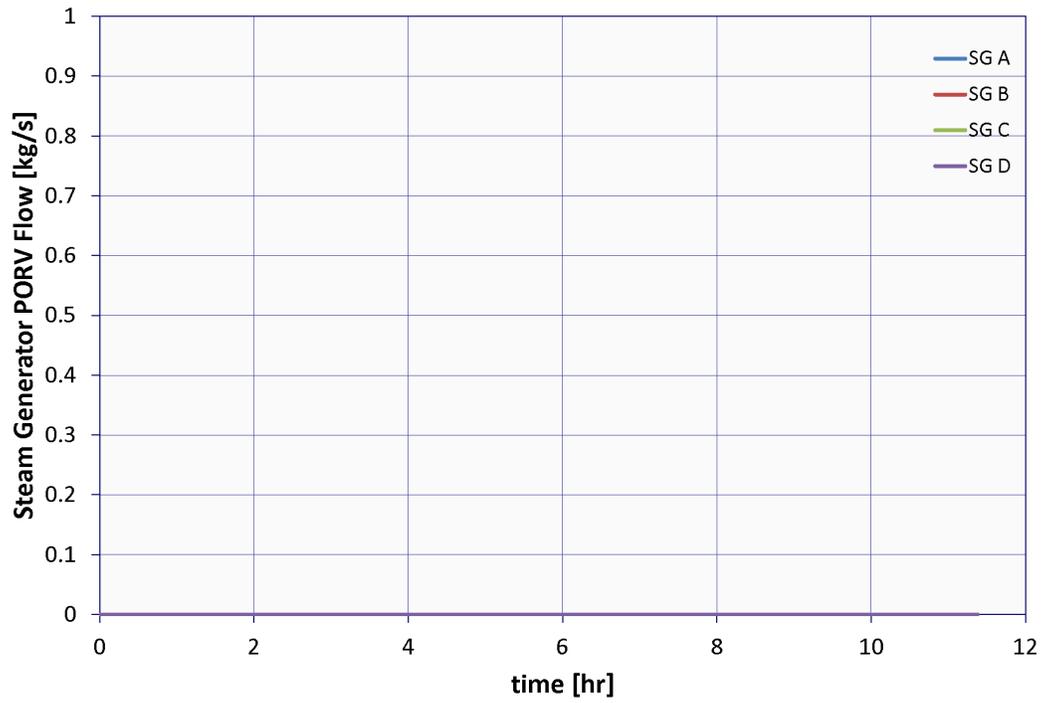
E.1.4 Case 4: 6-in. Break with 1/2 SI, 1/2 RHR, 2/2 CS Pumps

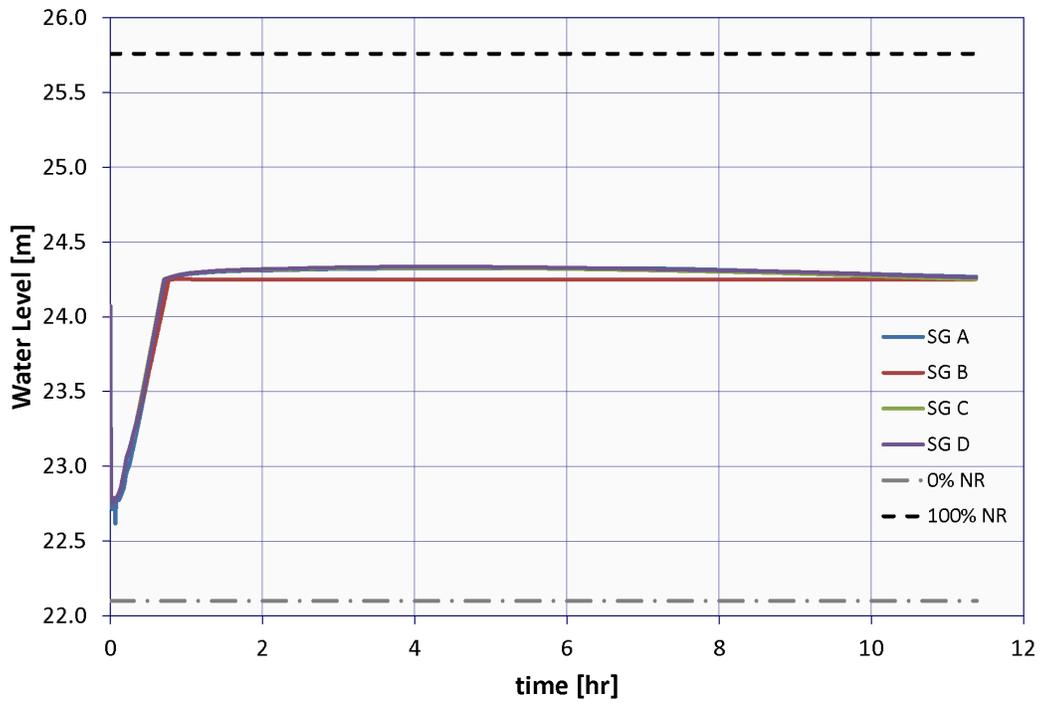
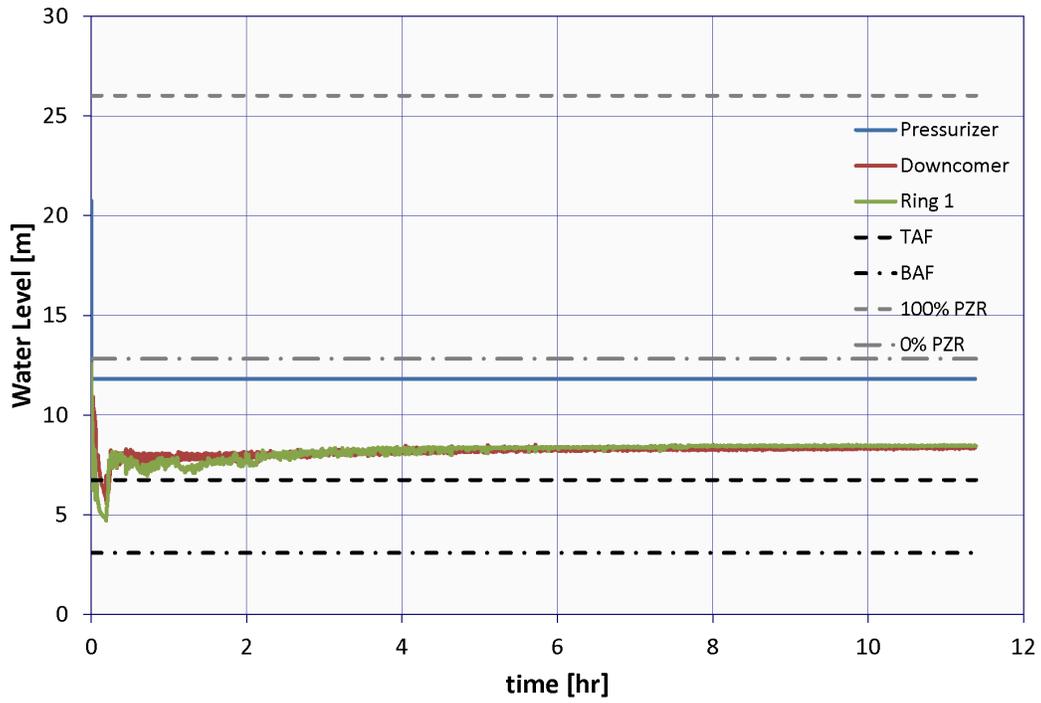


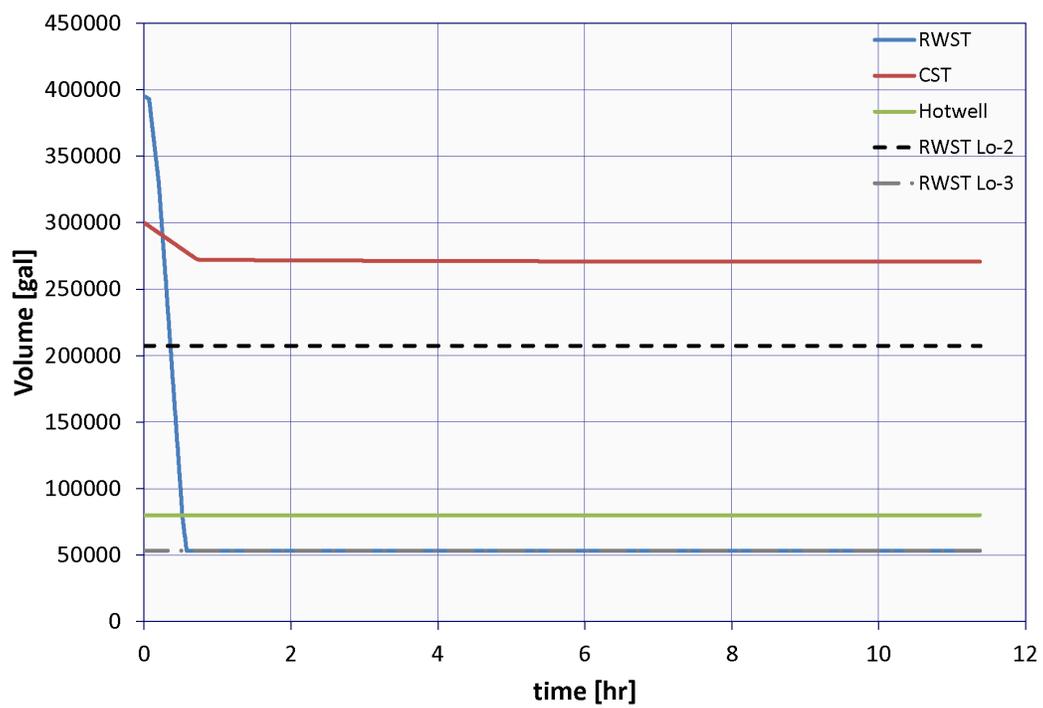




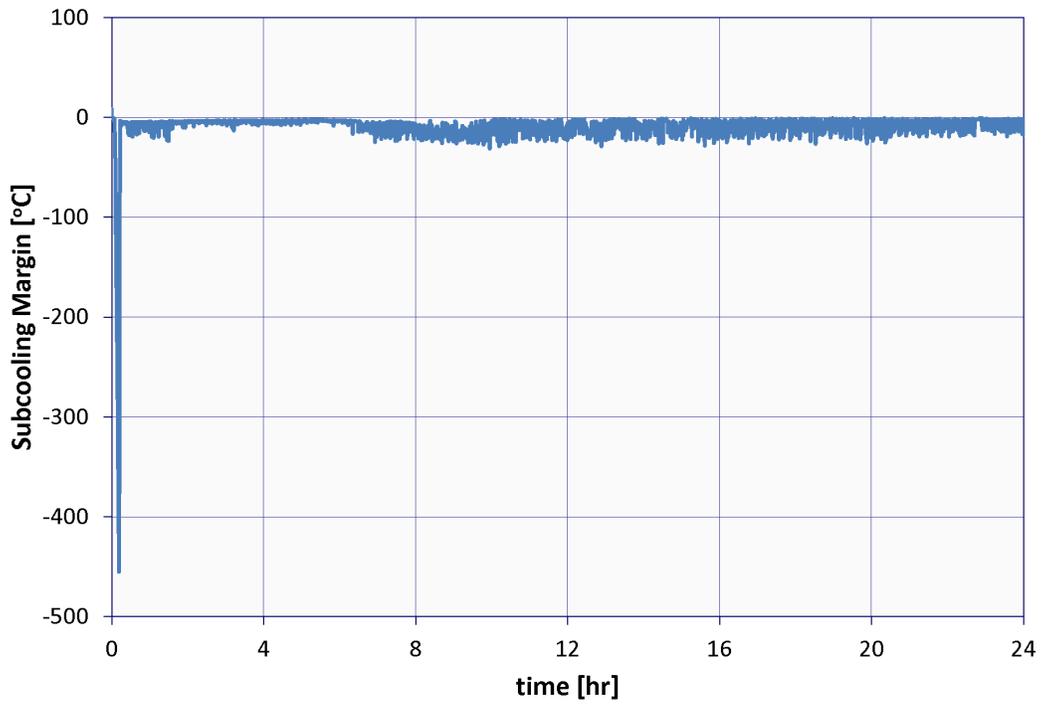
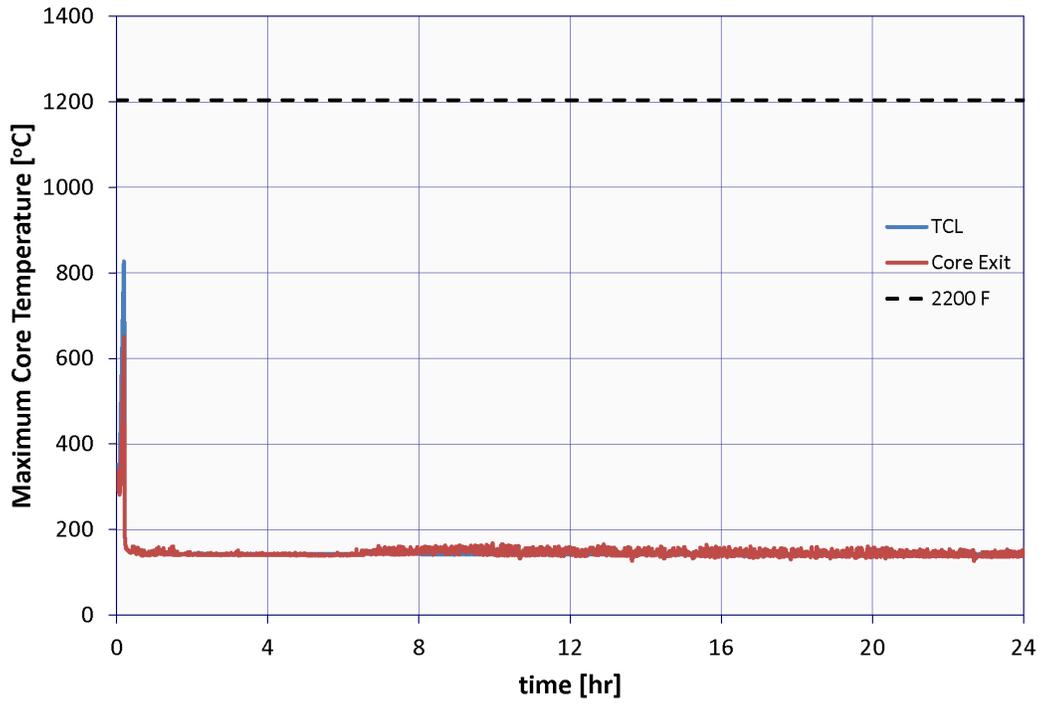


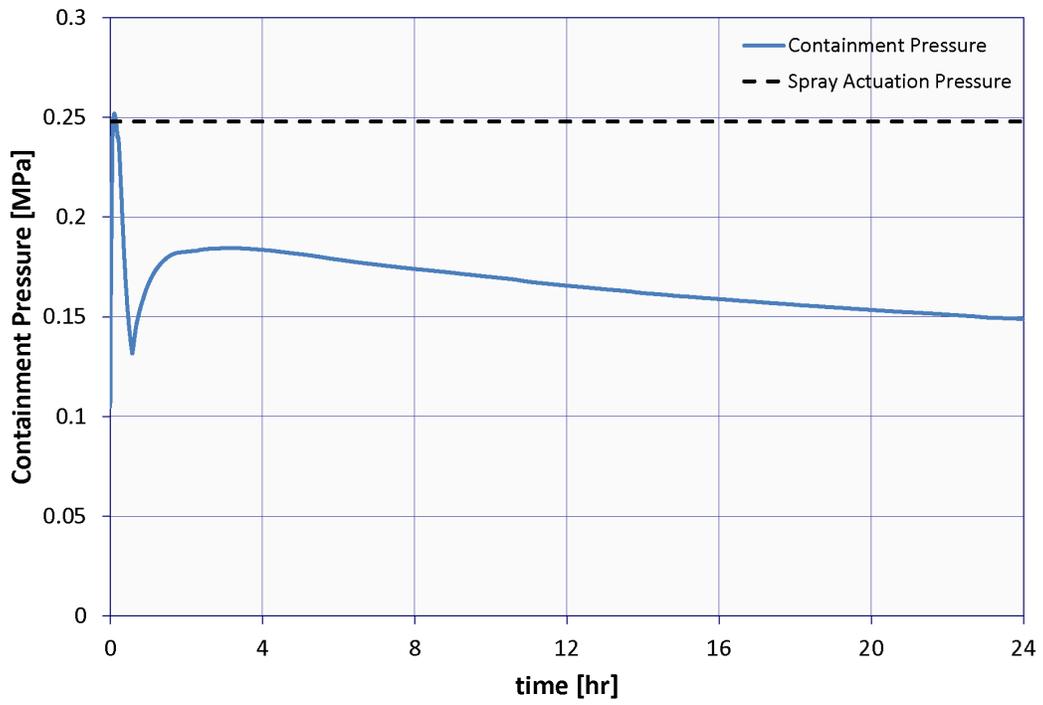
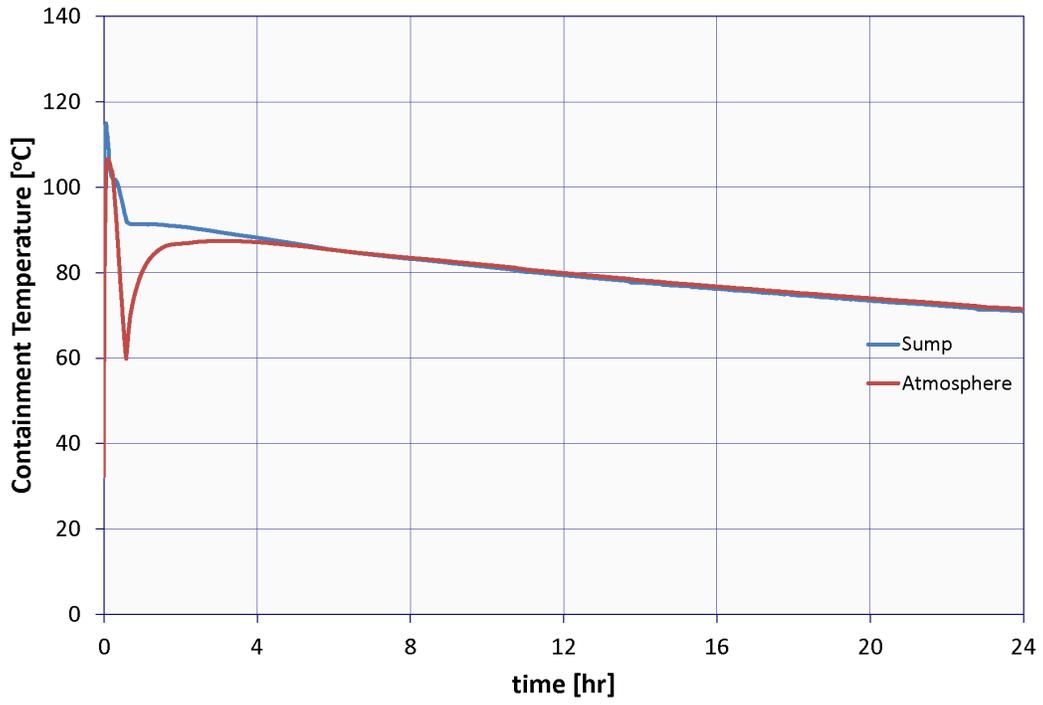


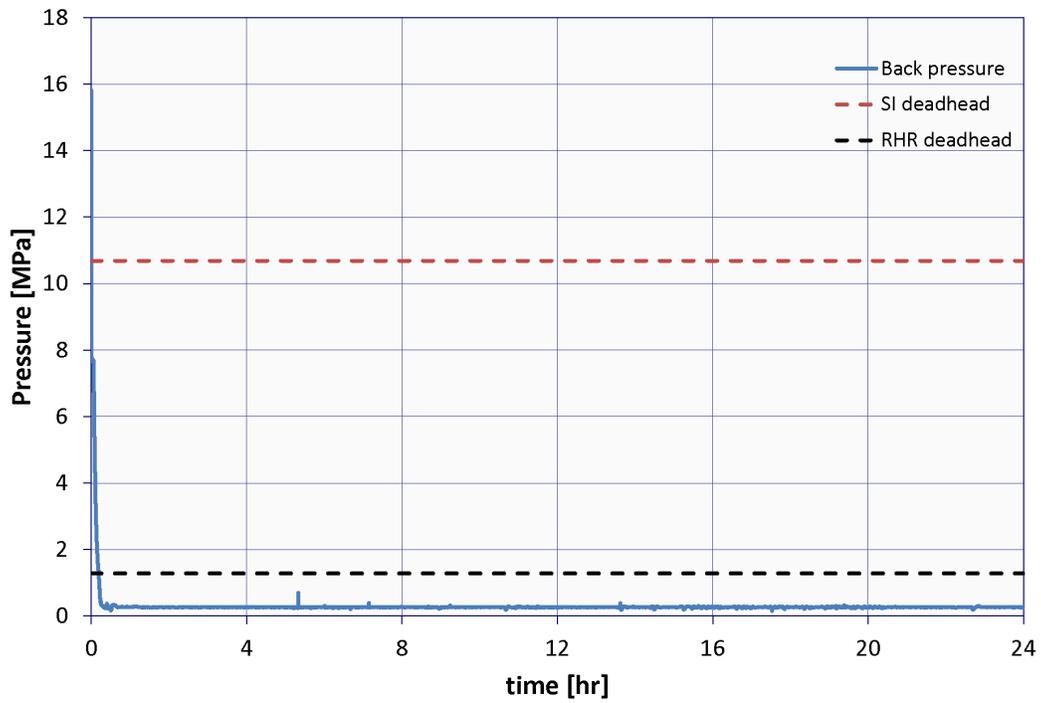
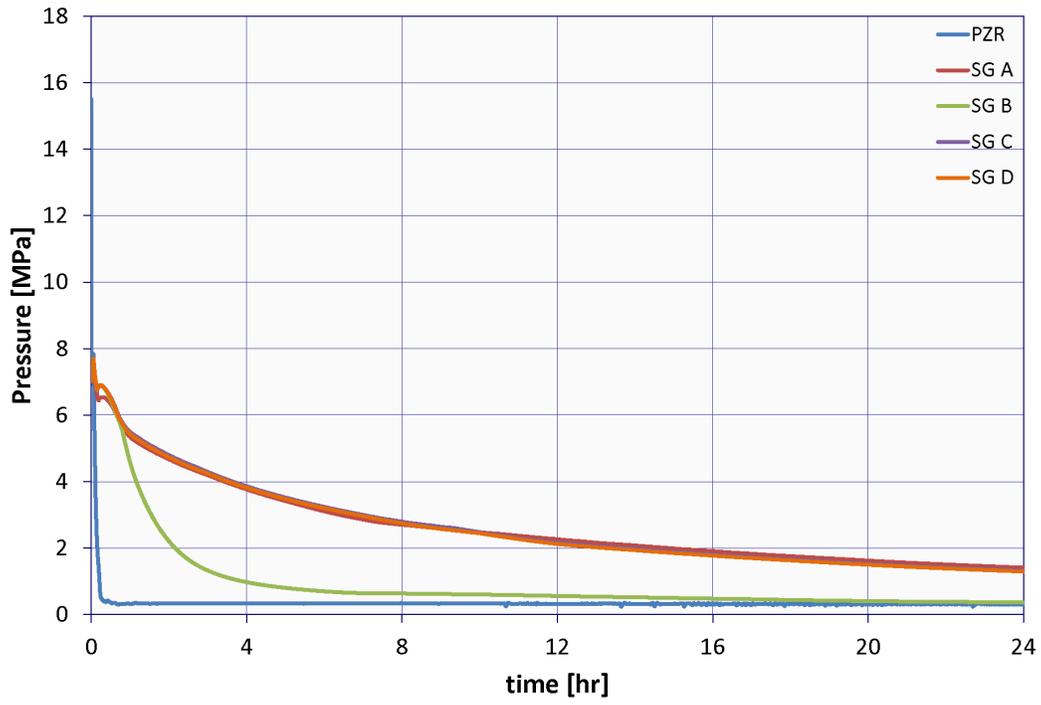


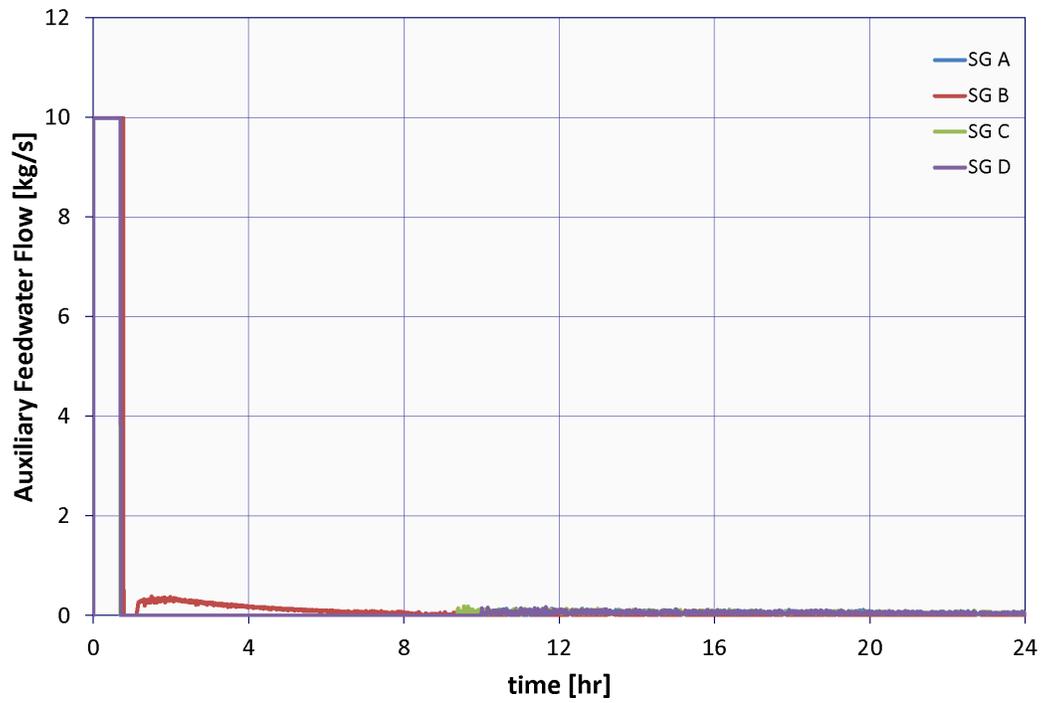
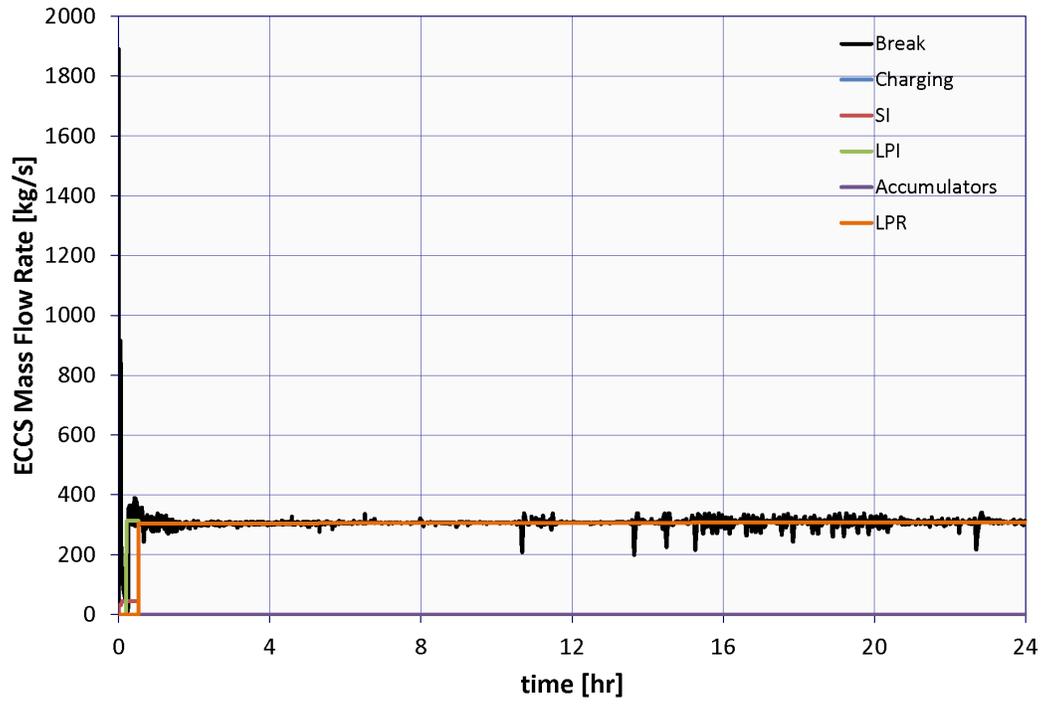


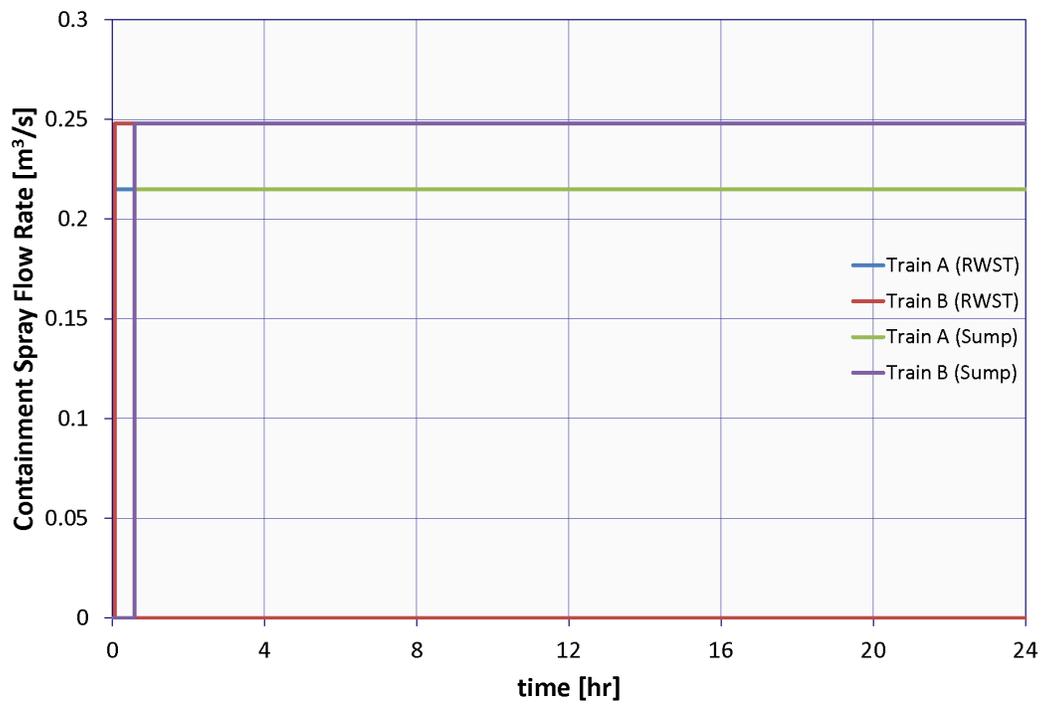
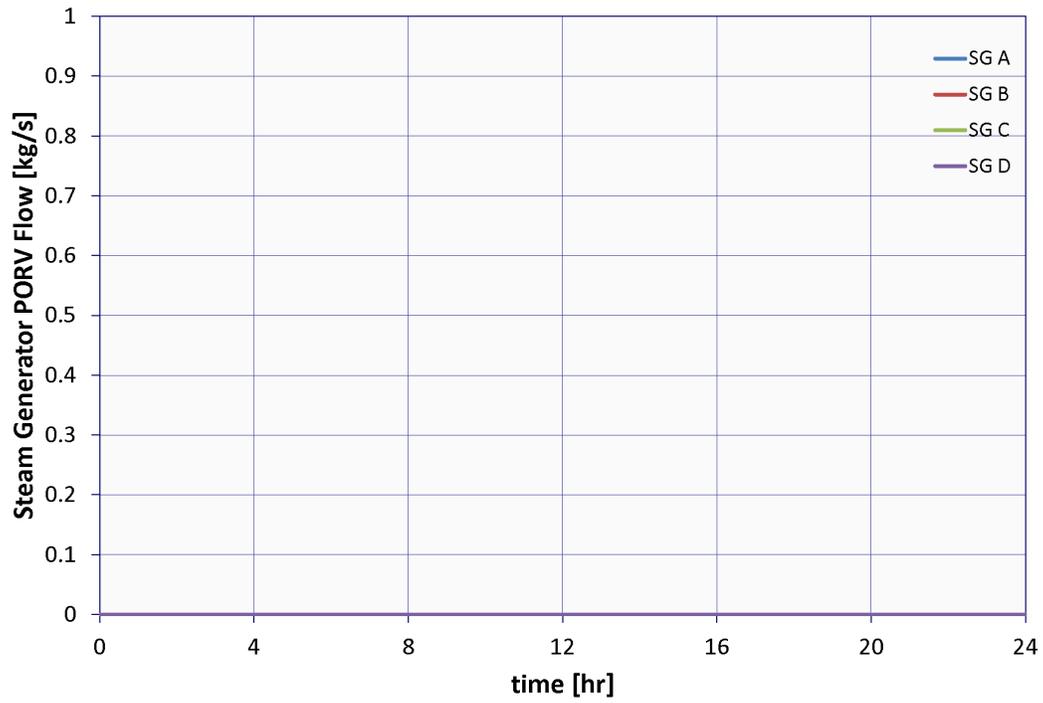
E.1.4.1 Case 4a: 6-in. Break with 1/2 SI Pumps, 1/2 RHR Pump, 2/2 CS Pumps, CS Recirc

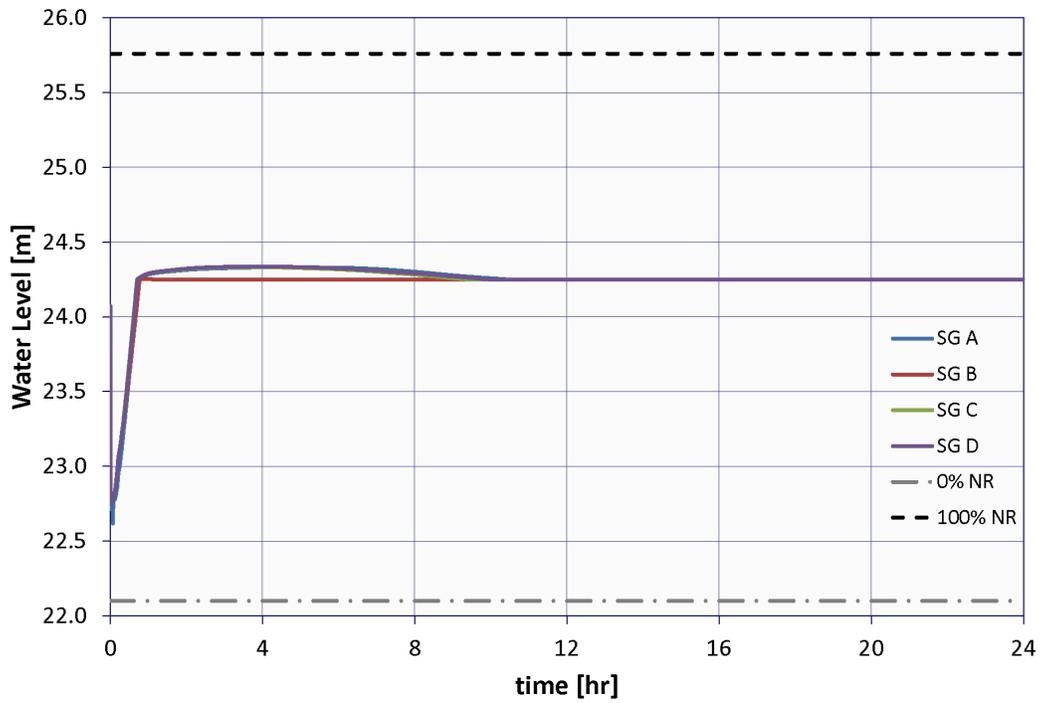
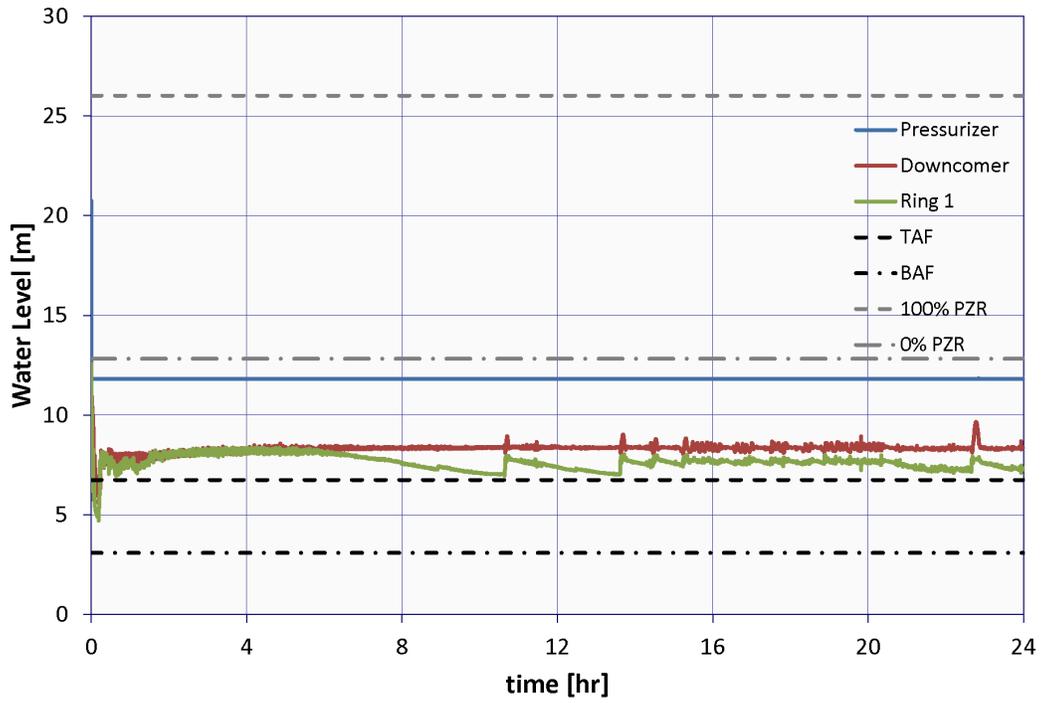


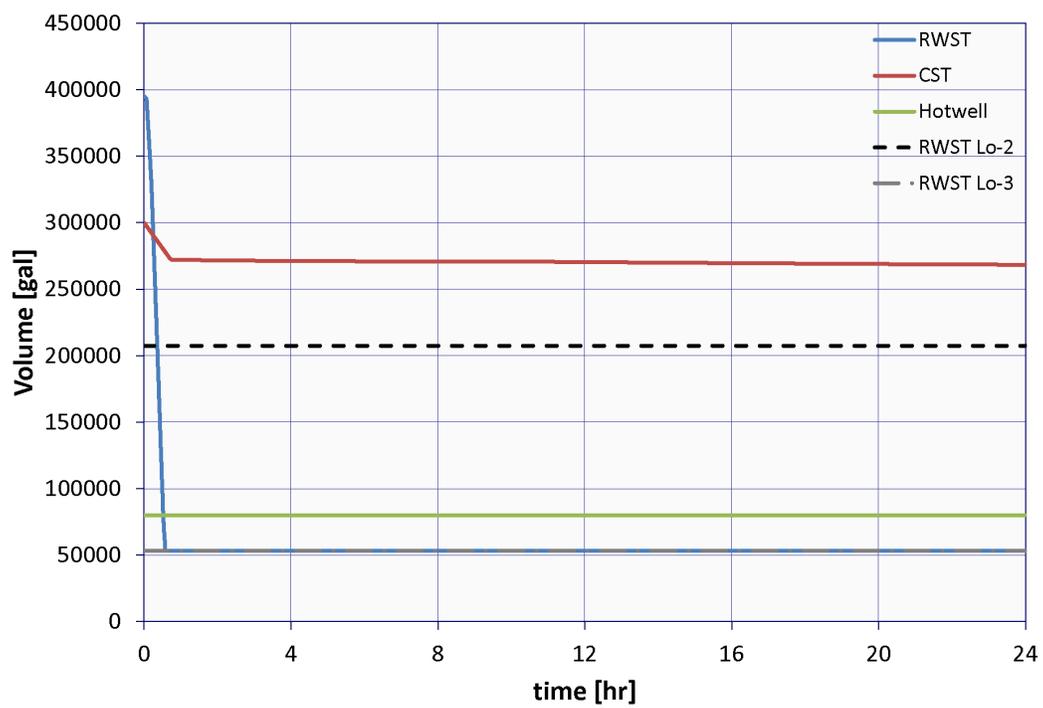




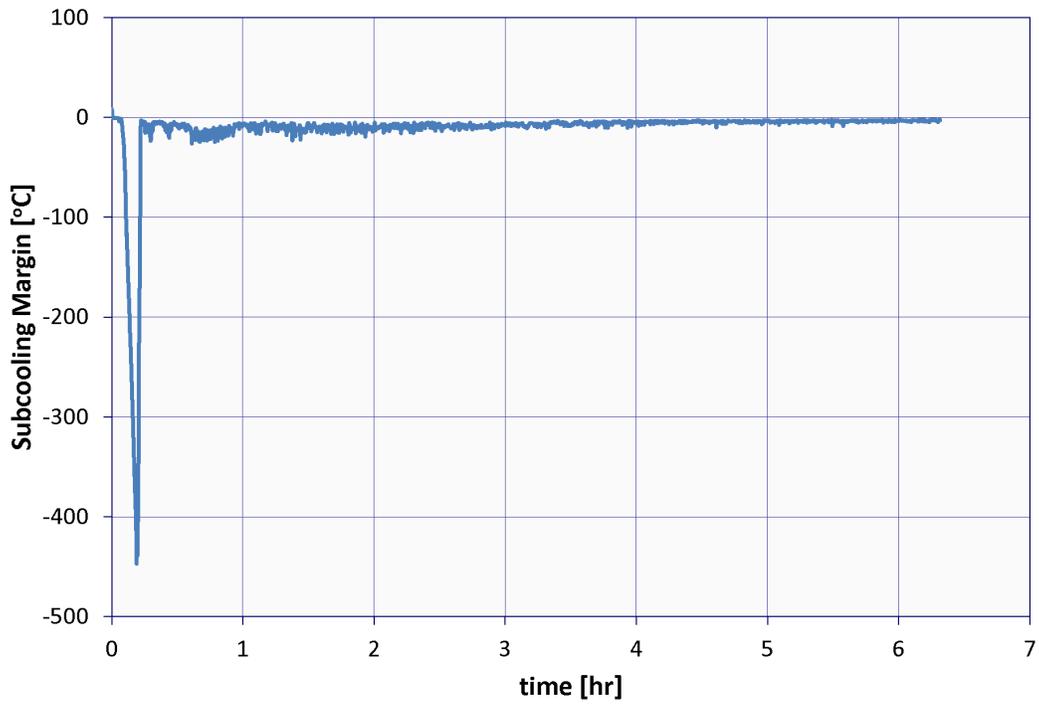
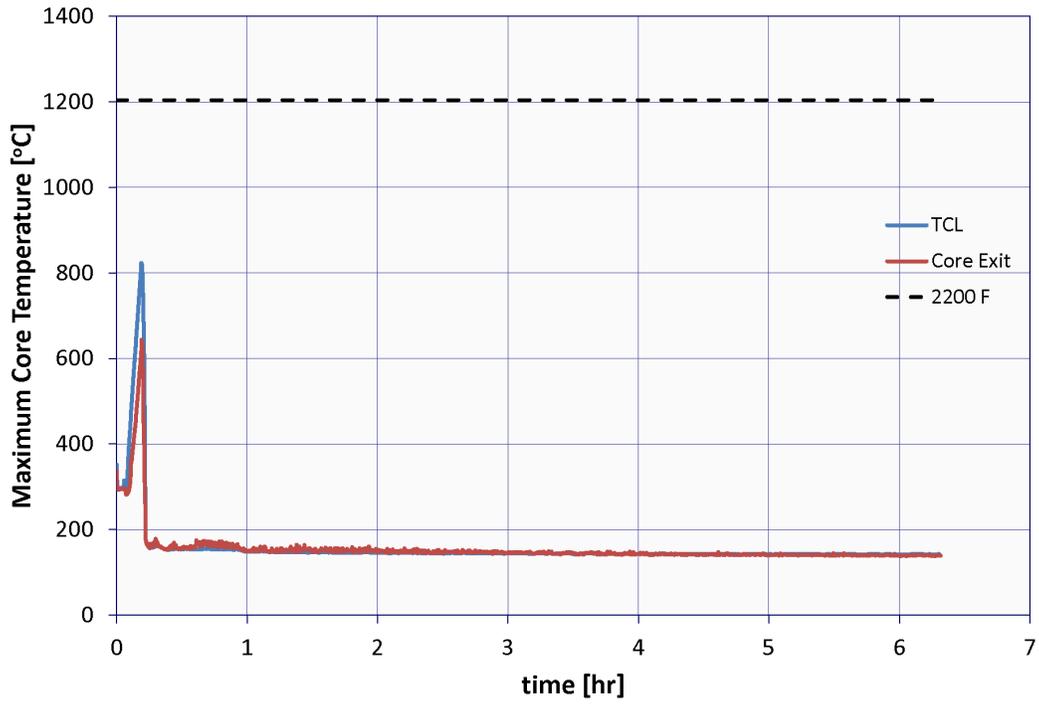


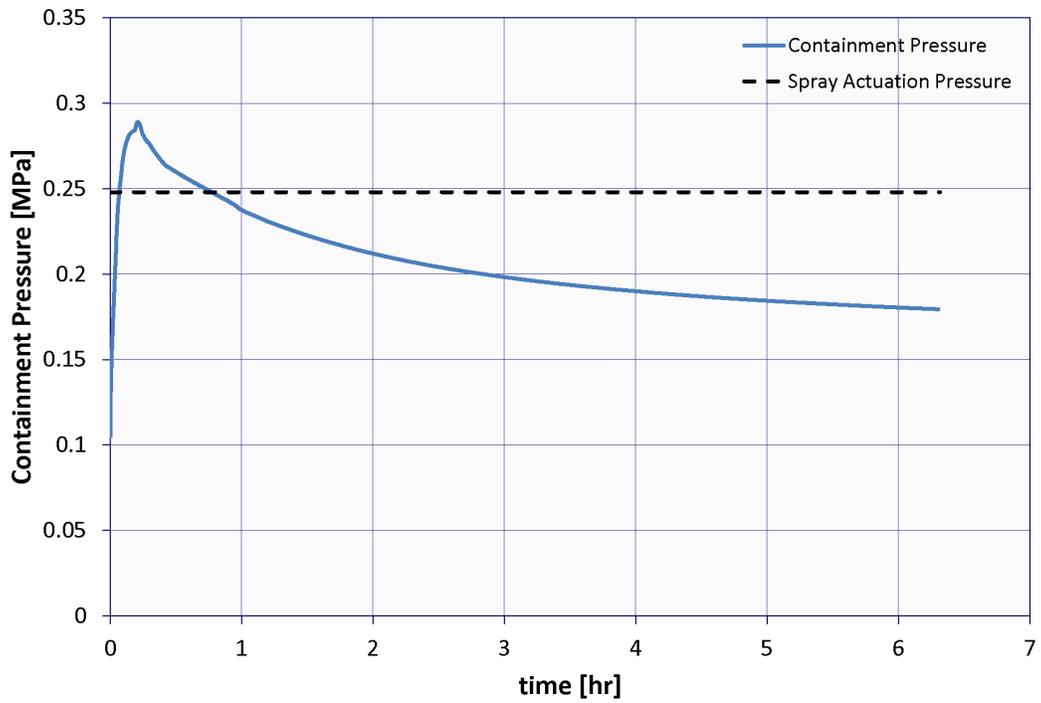
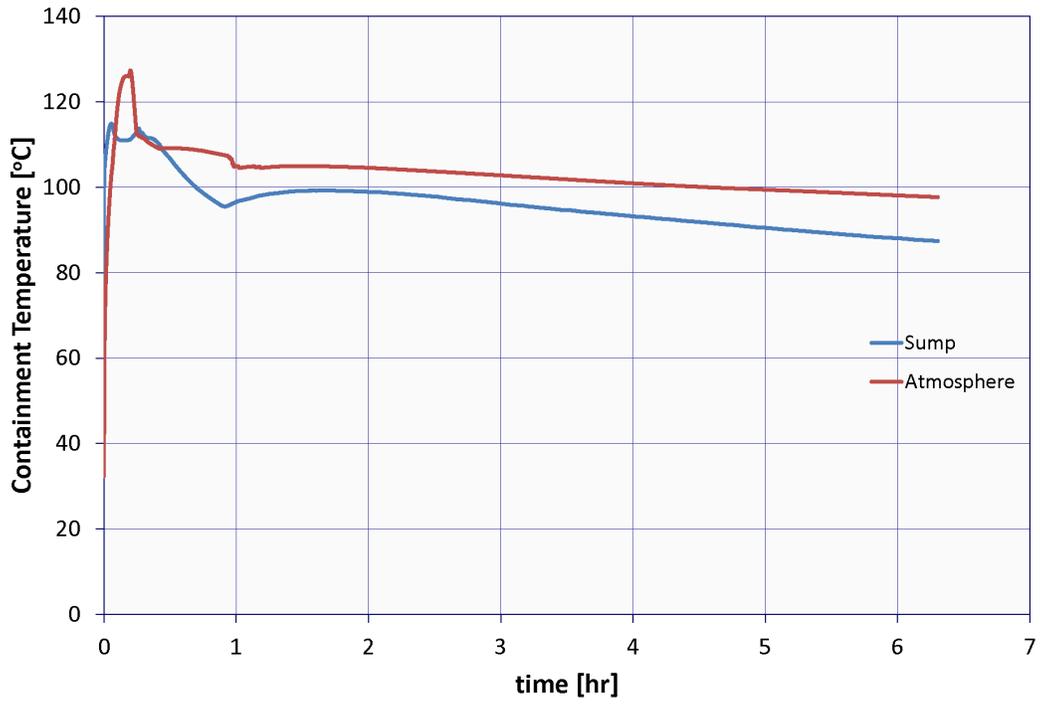


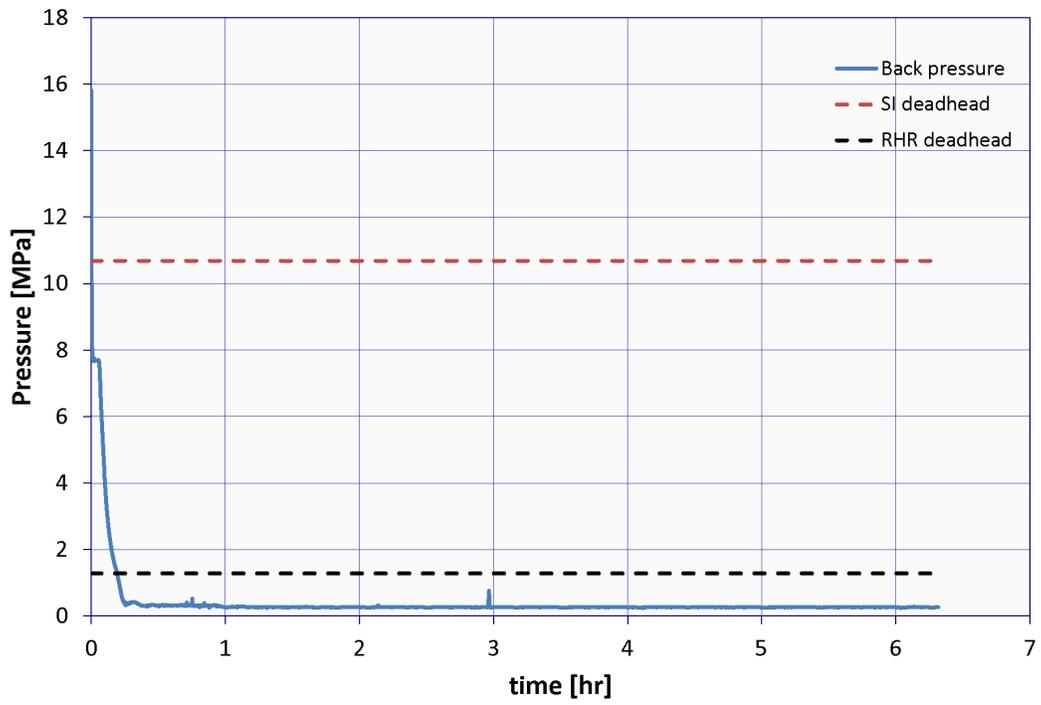
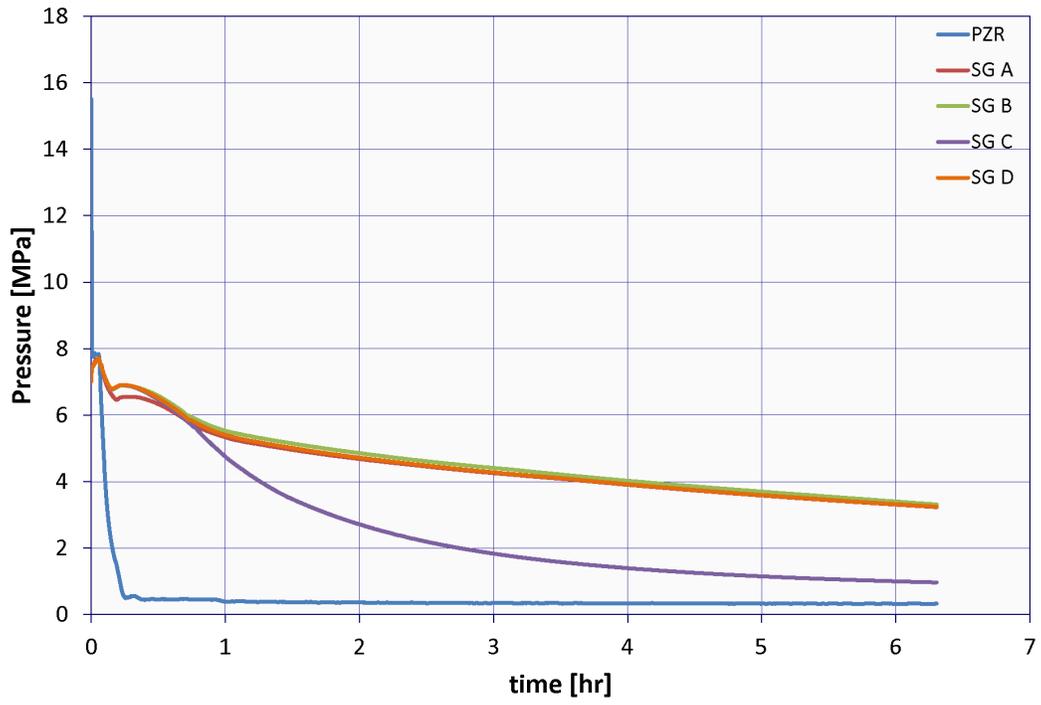


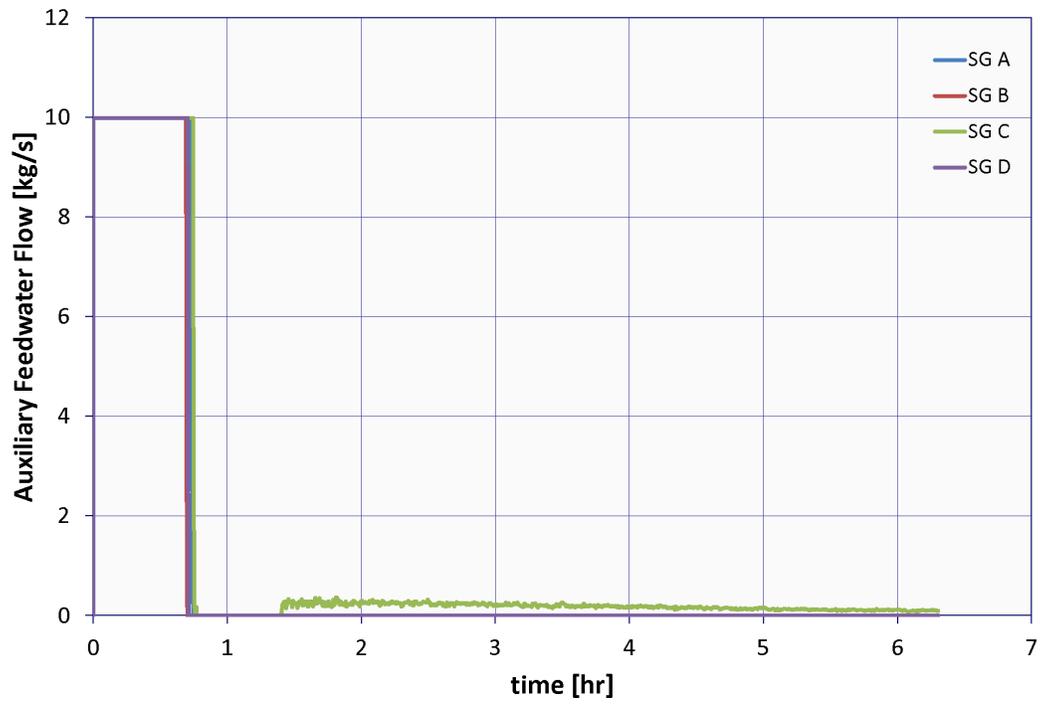
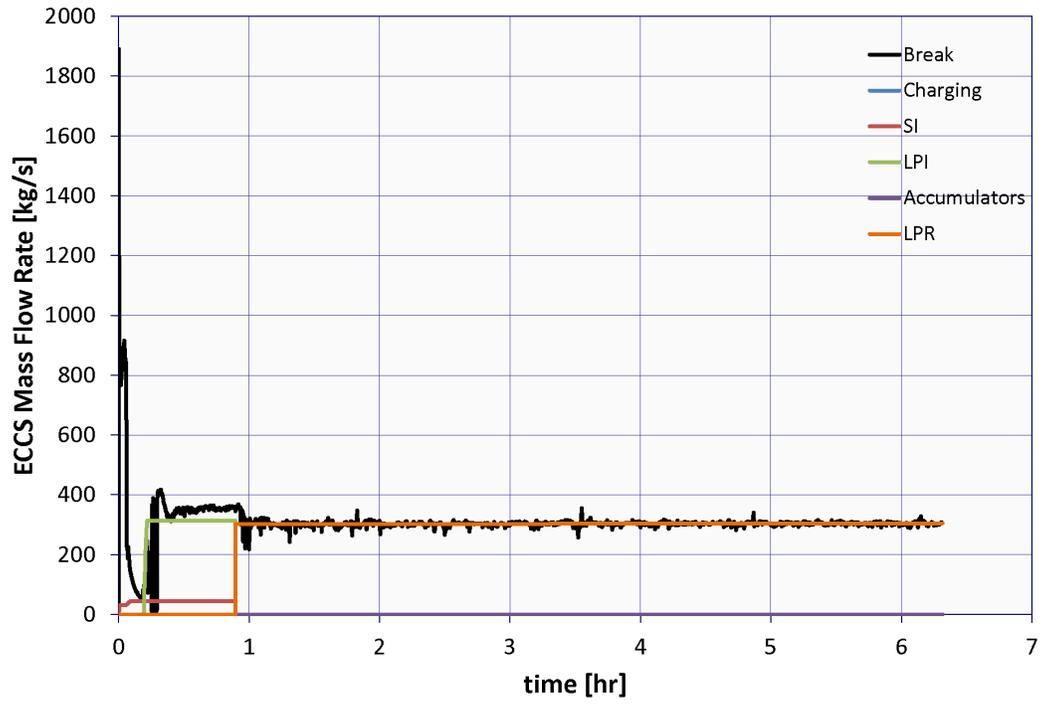


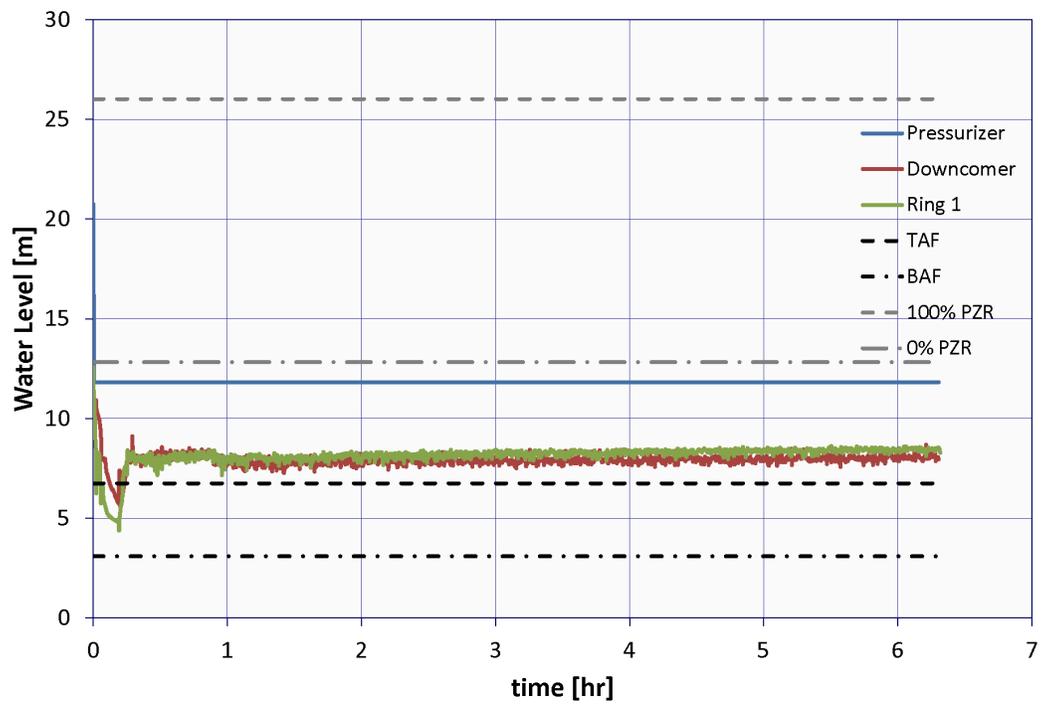
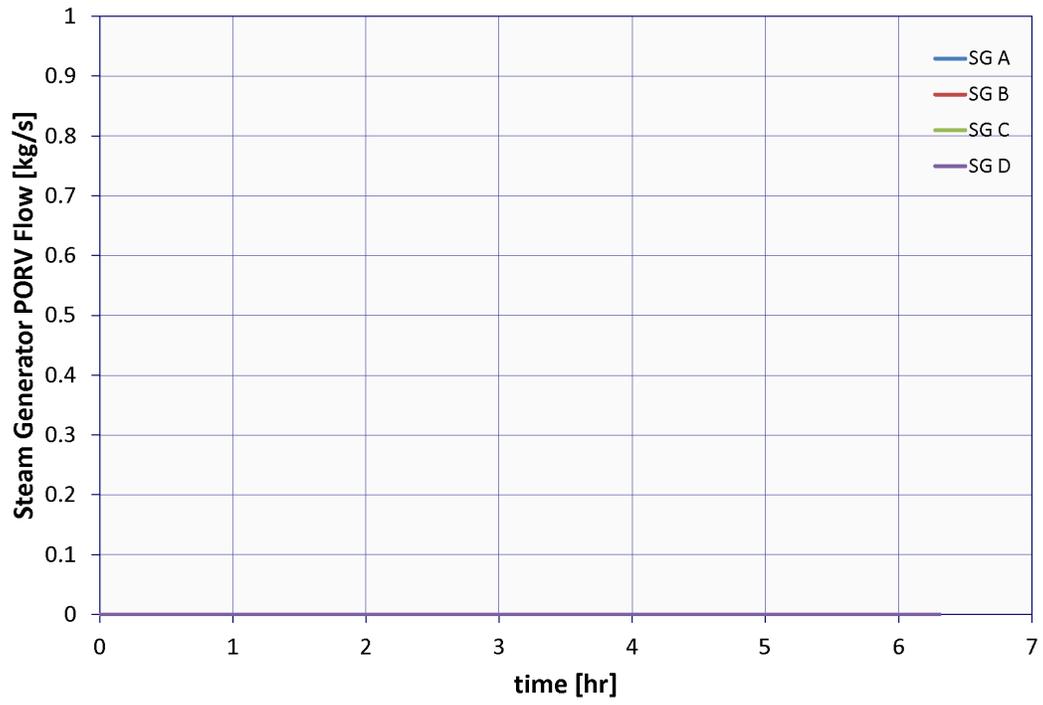
E.1.4.2 Case 4b: 6-in. Break with 1/2 SI Pumps, 1/2 RHR Pump, 0/2 CS Pumps

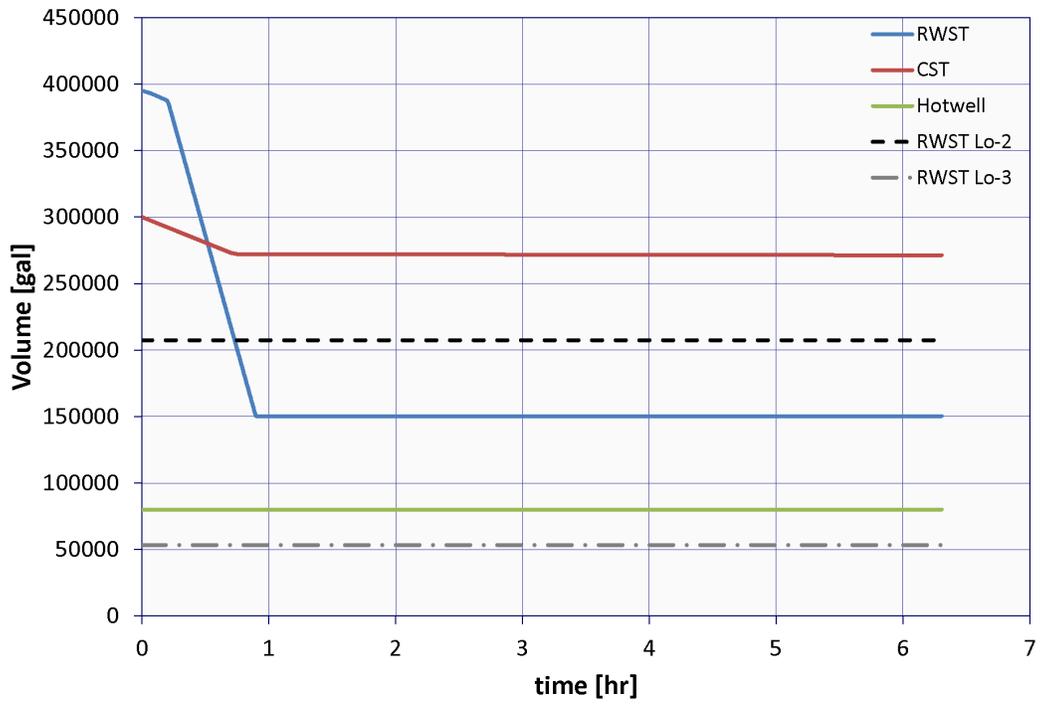
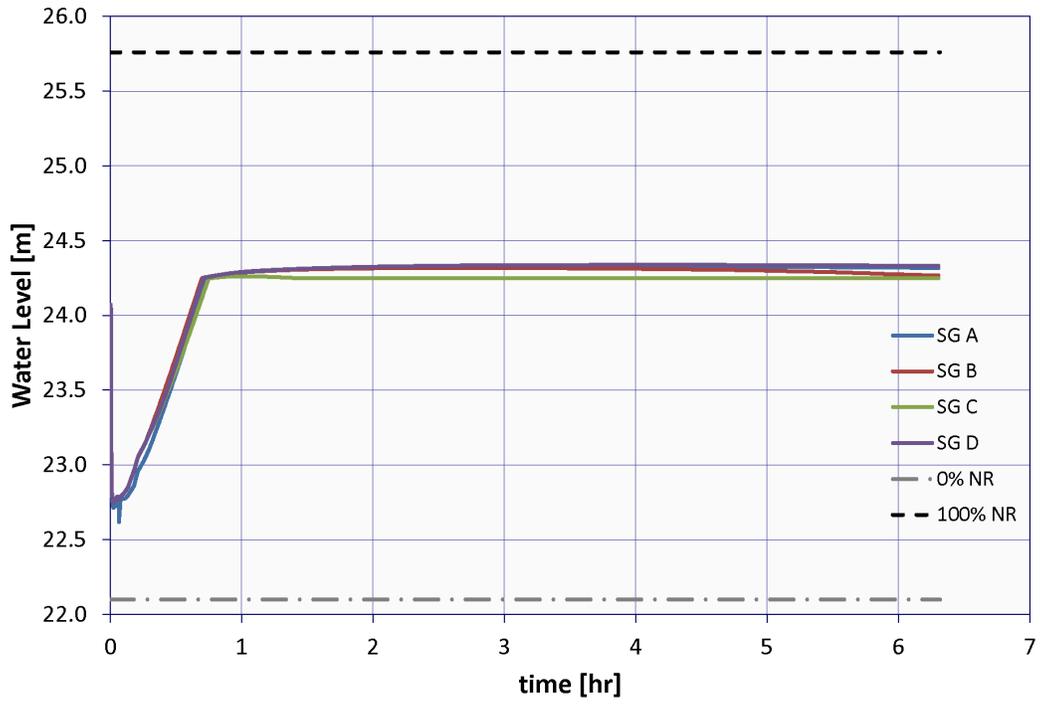




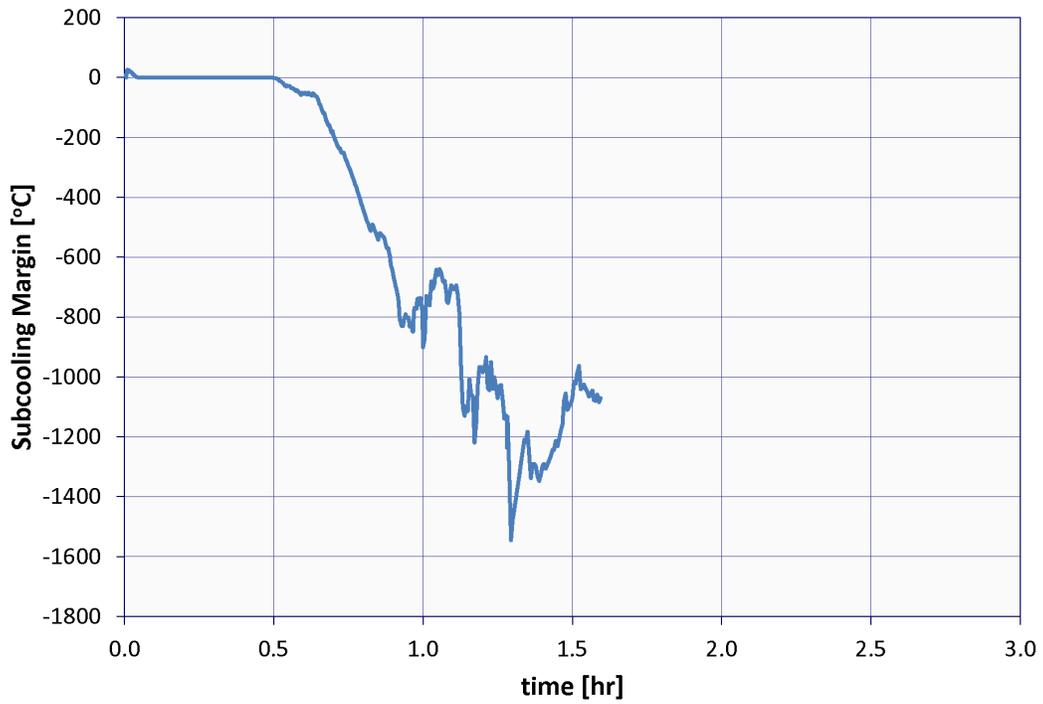
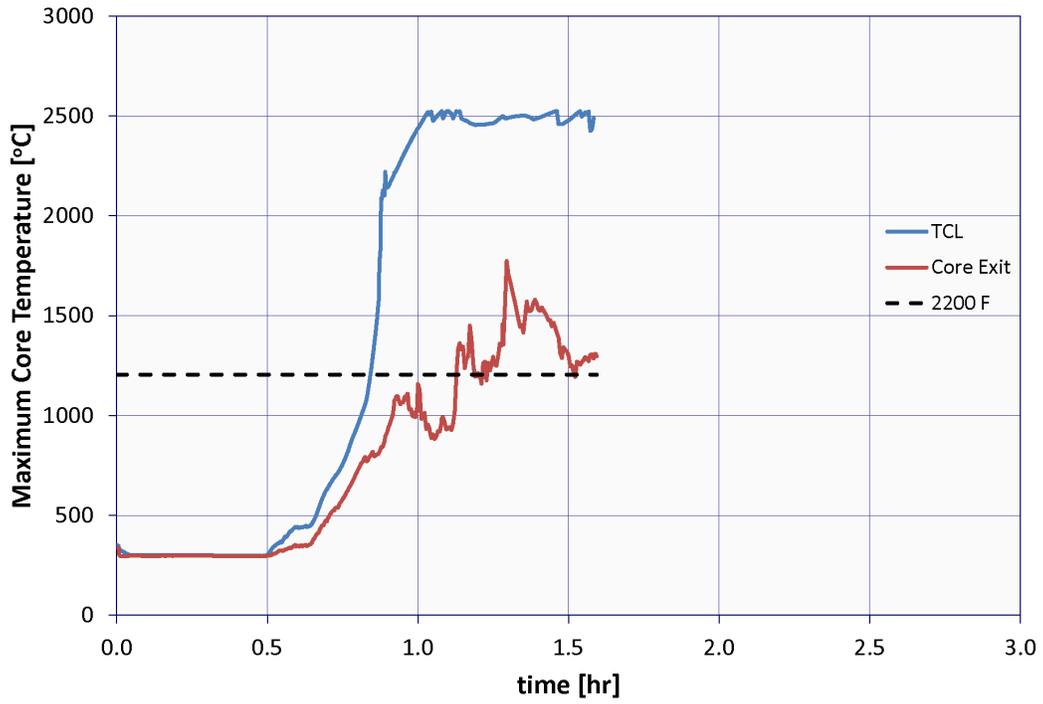


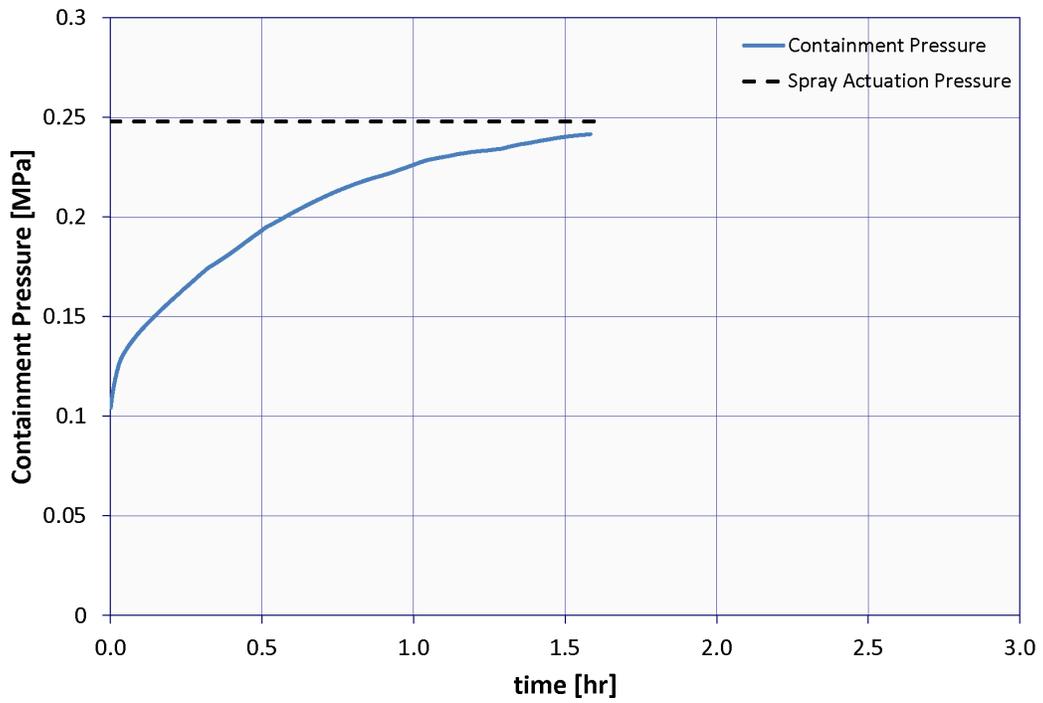
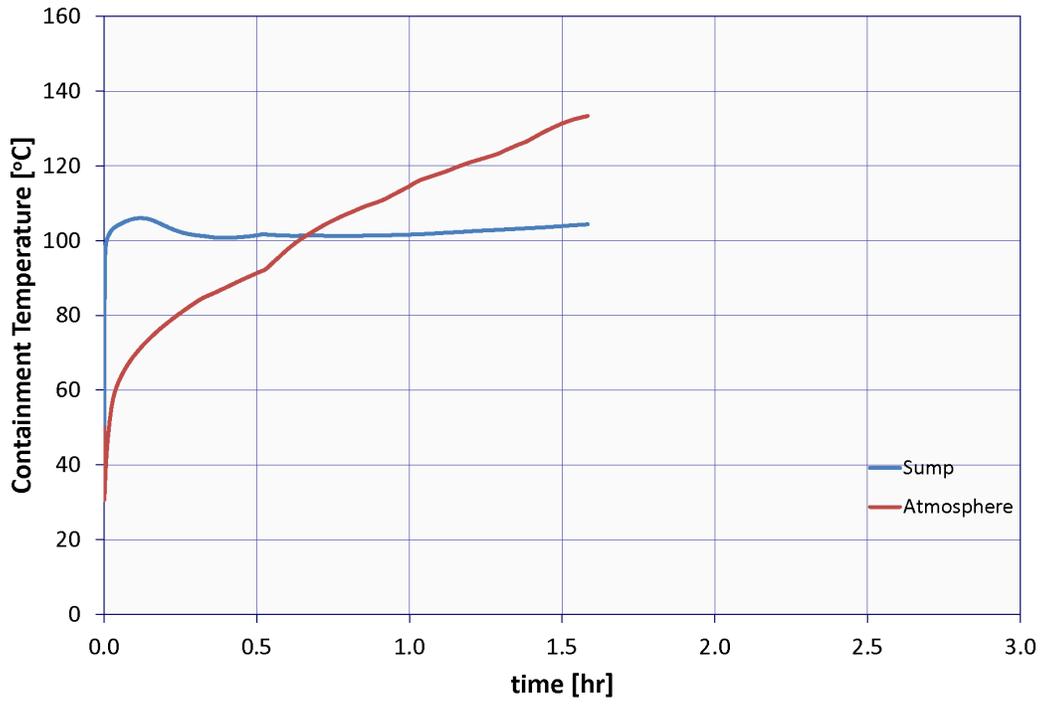


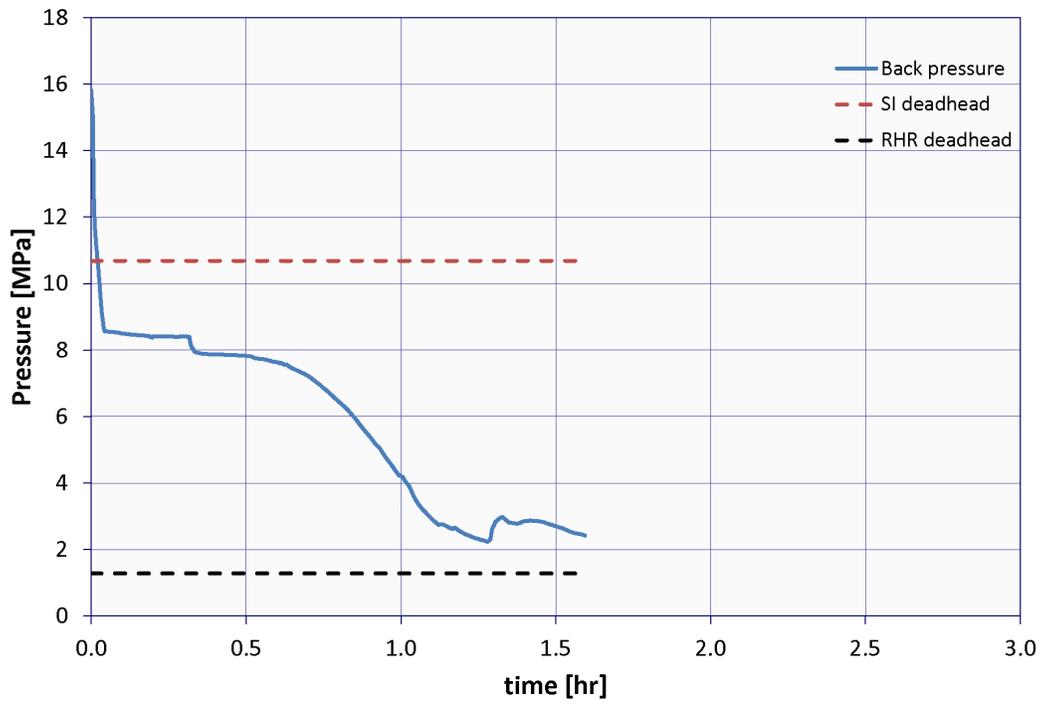
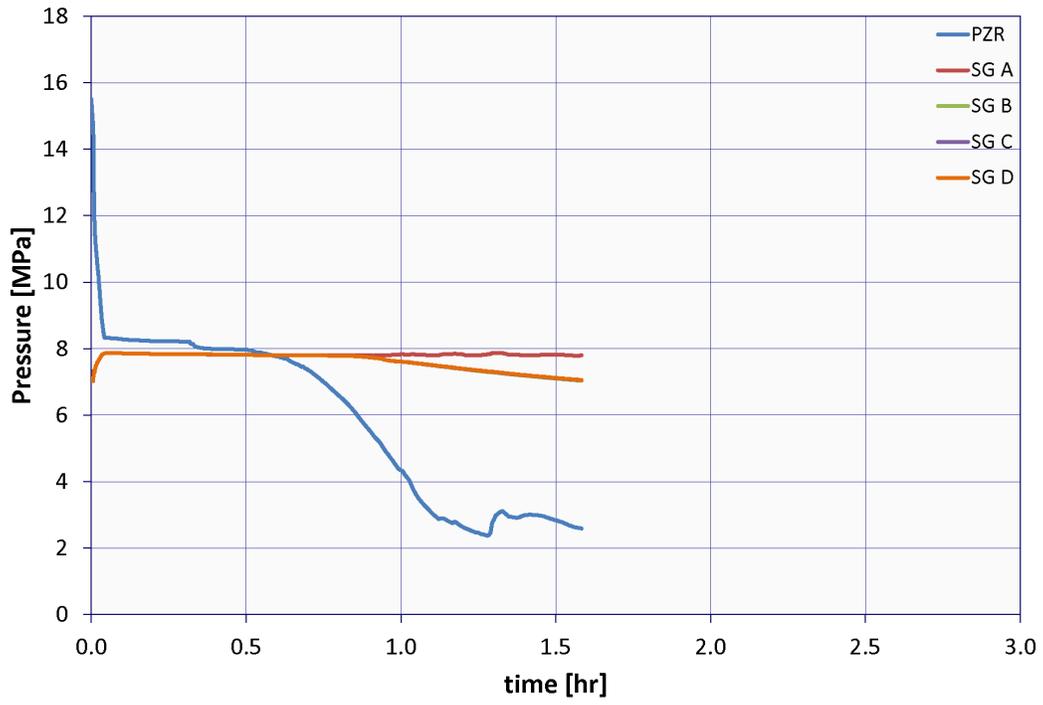


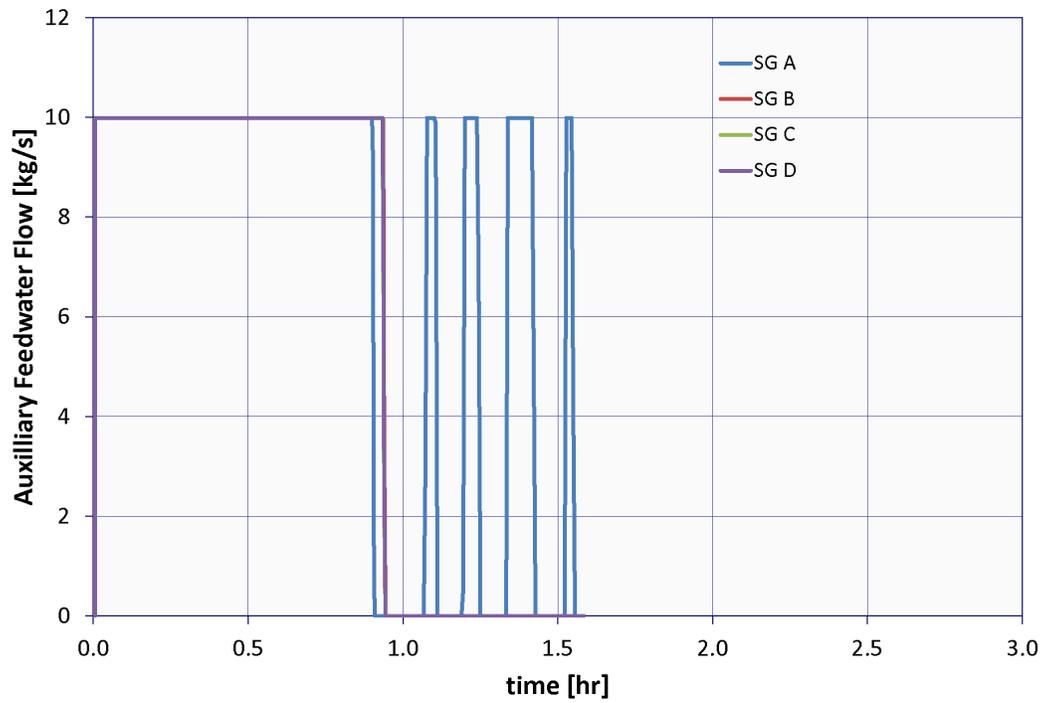
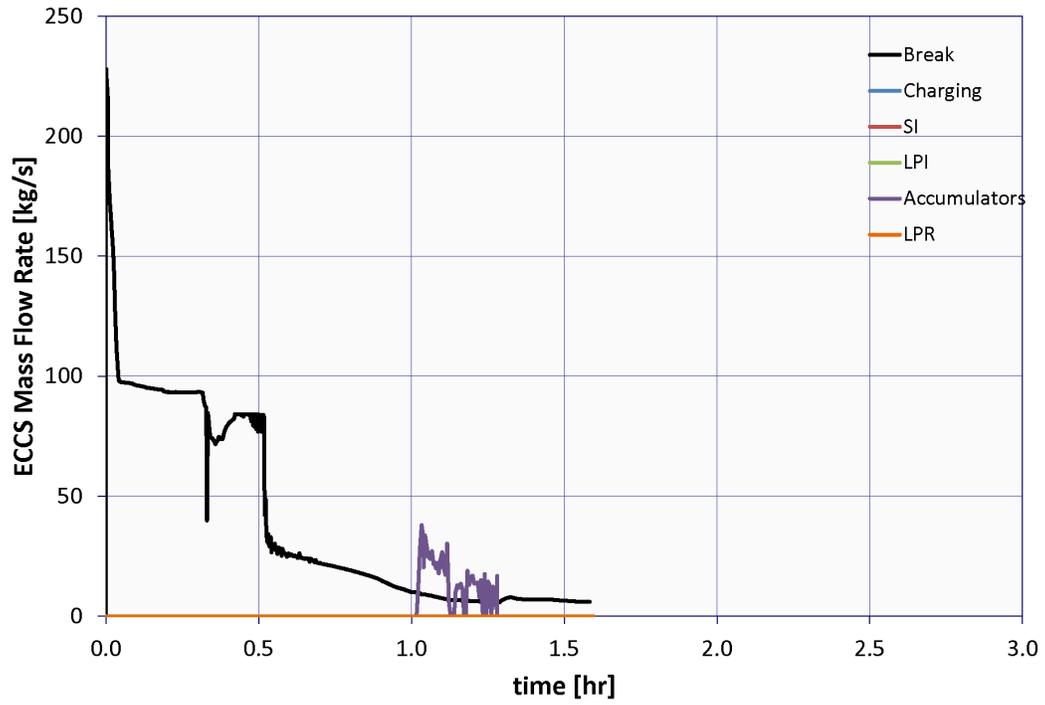


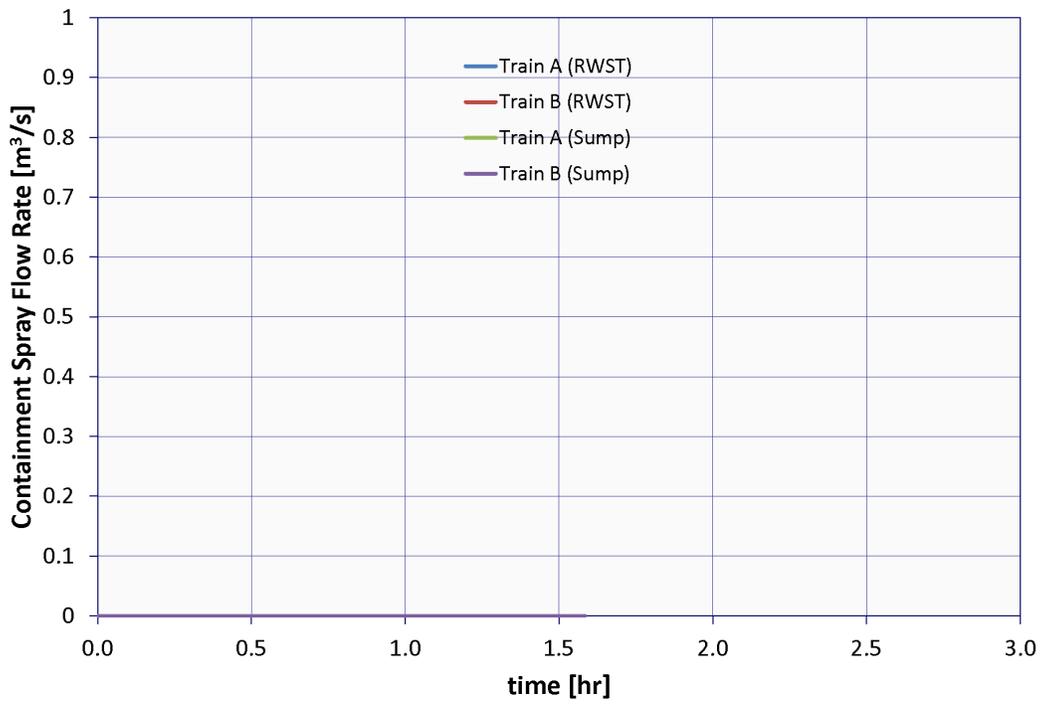
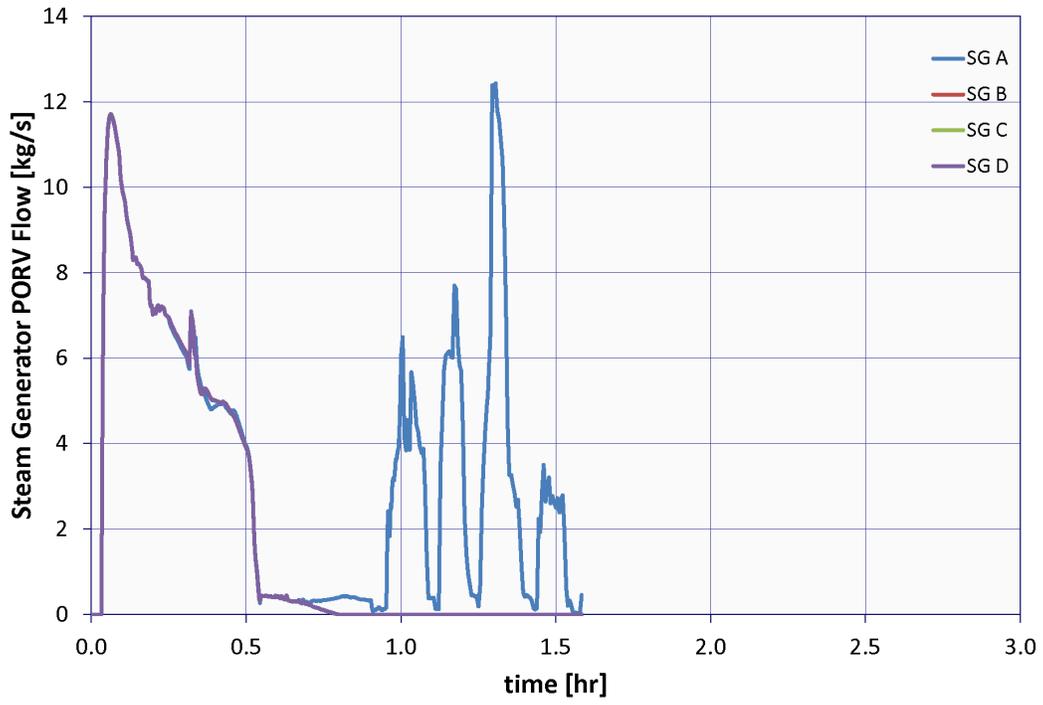
E.1.5 Case 5: 2-in. Break with 2 Accumulators, 1 RHR, 2/2 CS Pumps

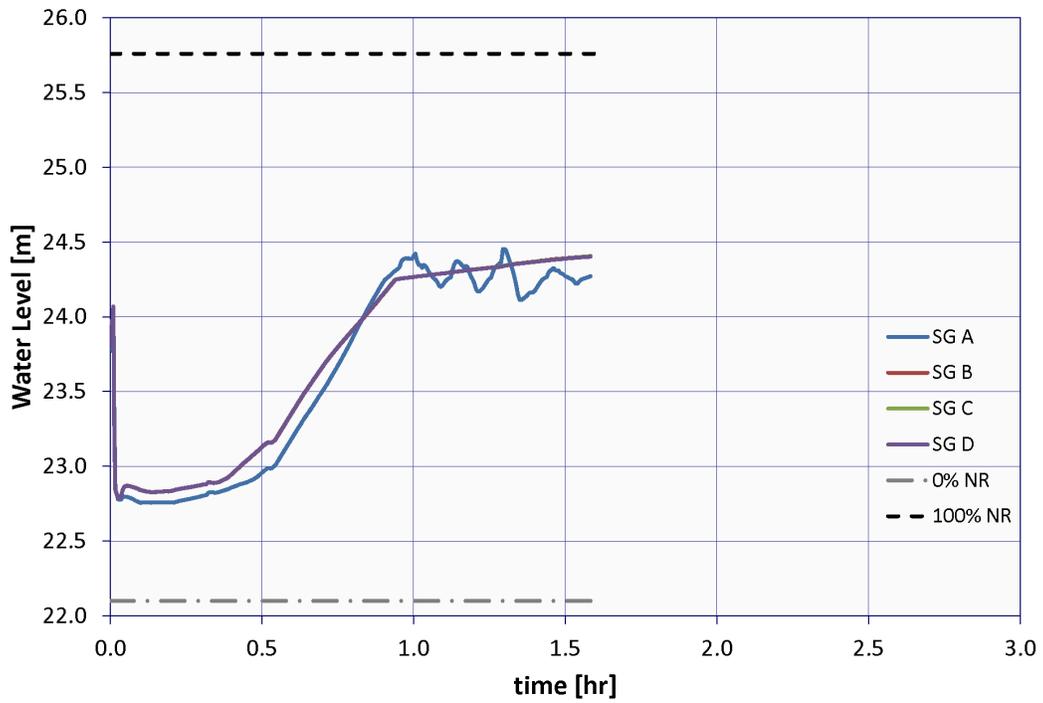
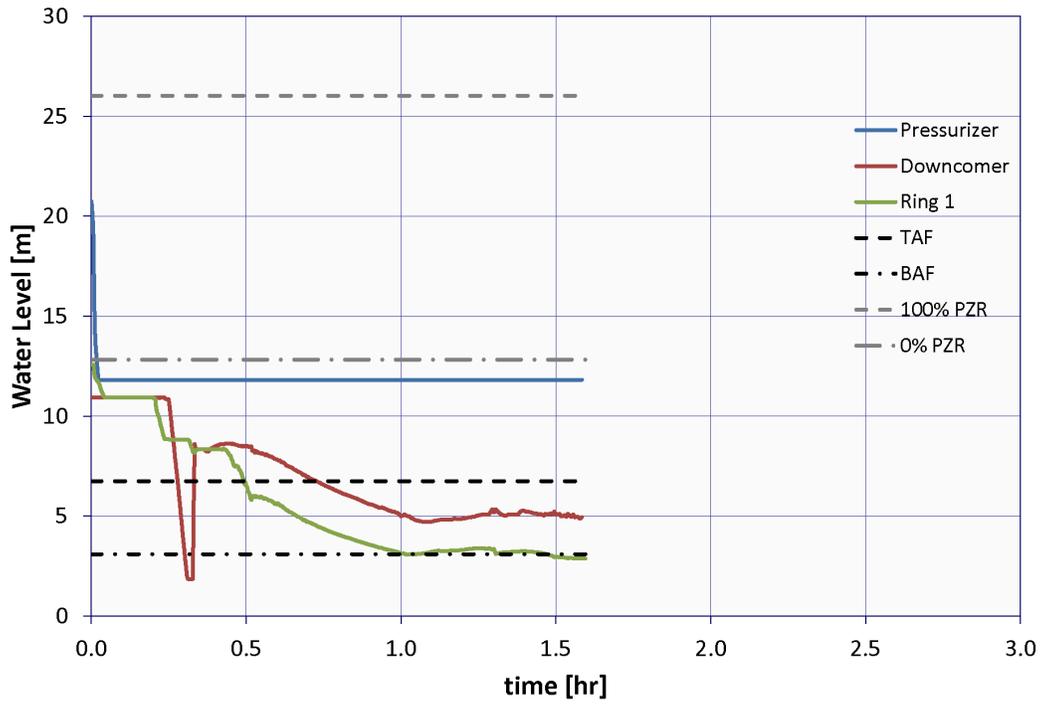


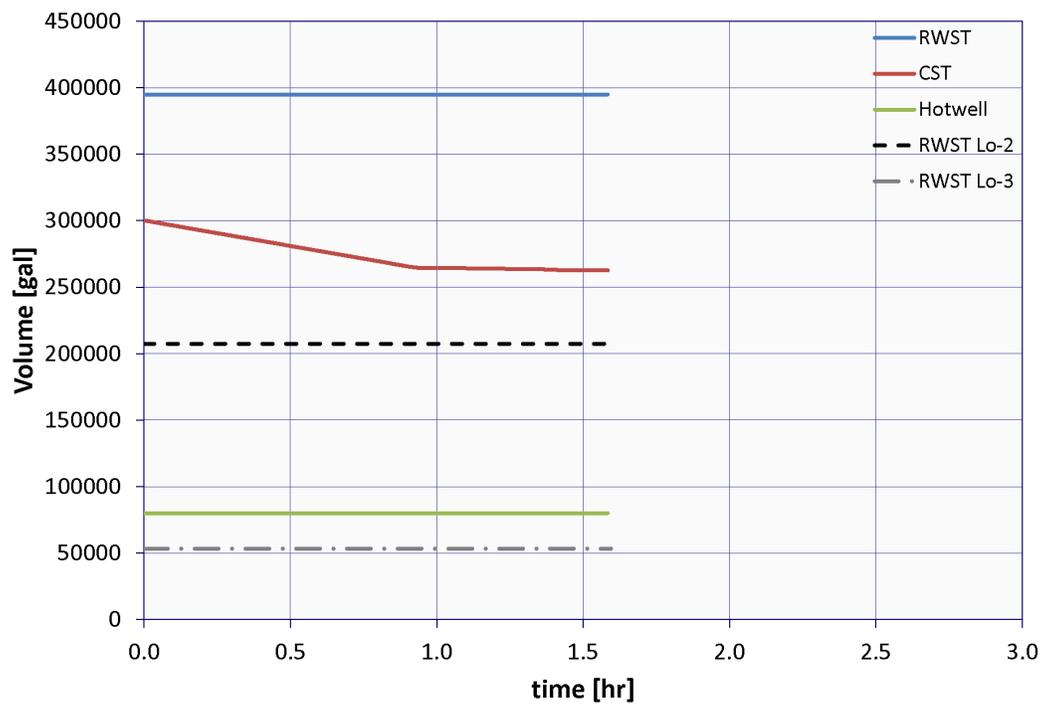




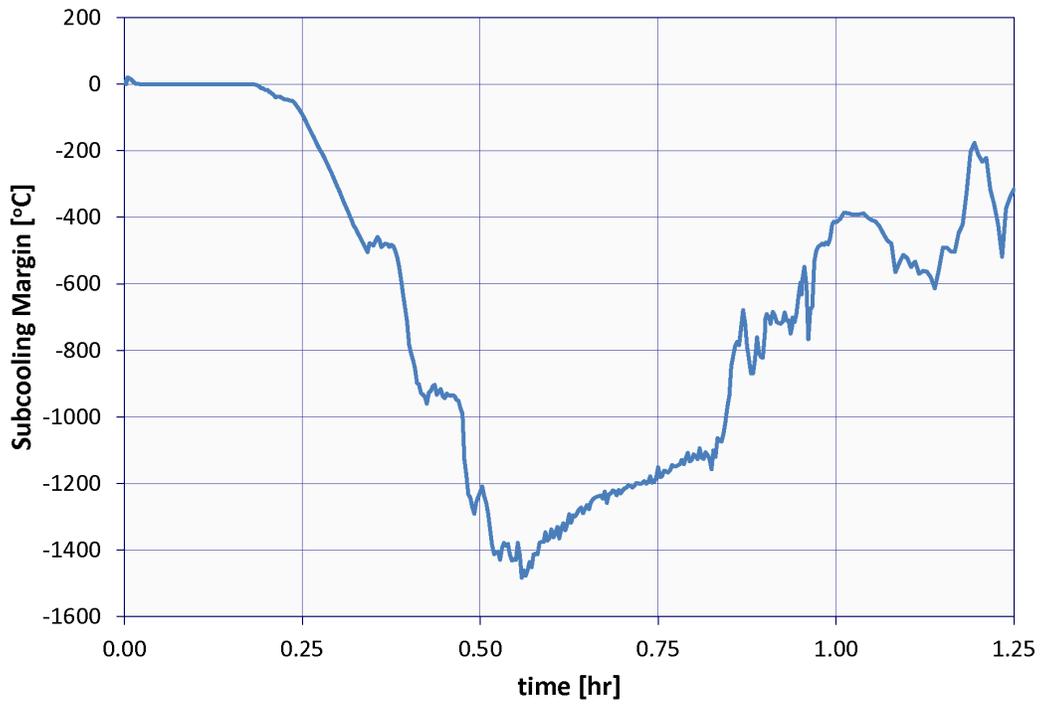
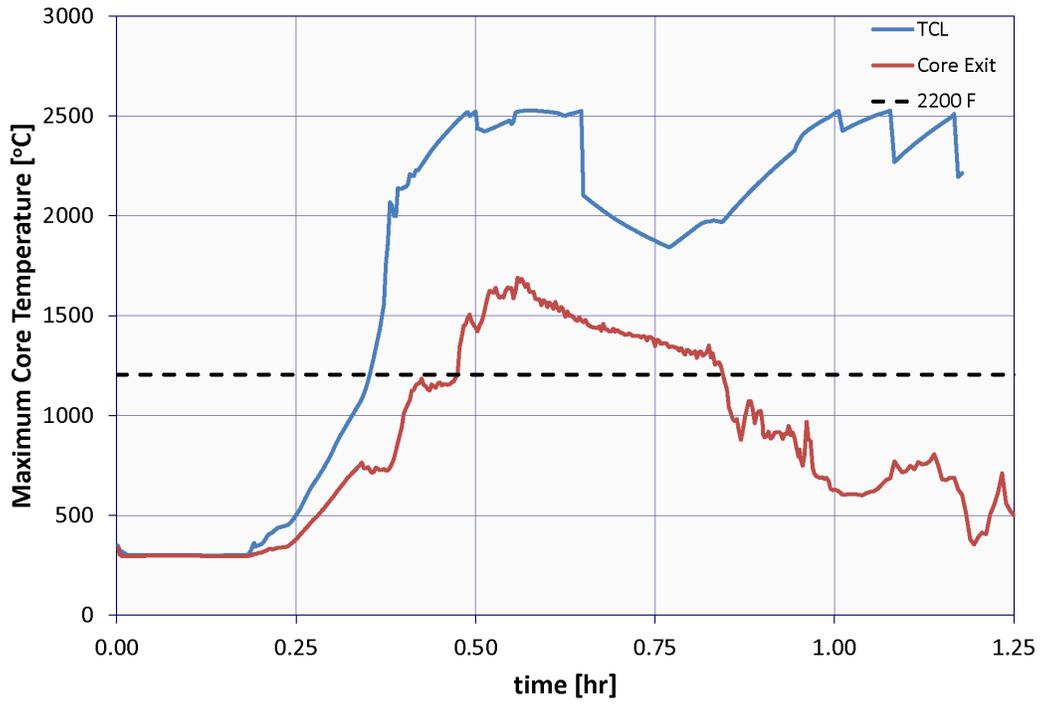


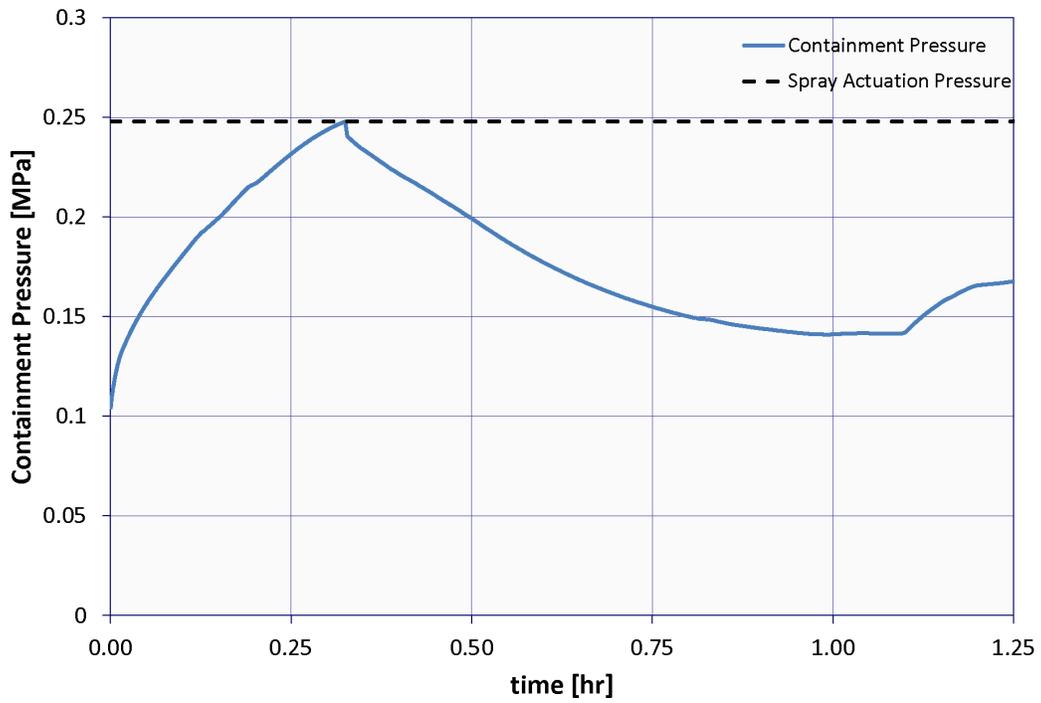
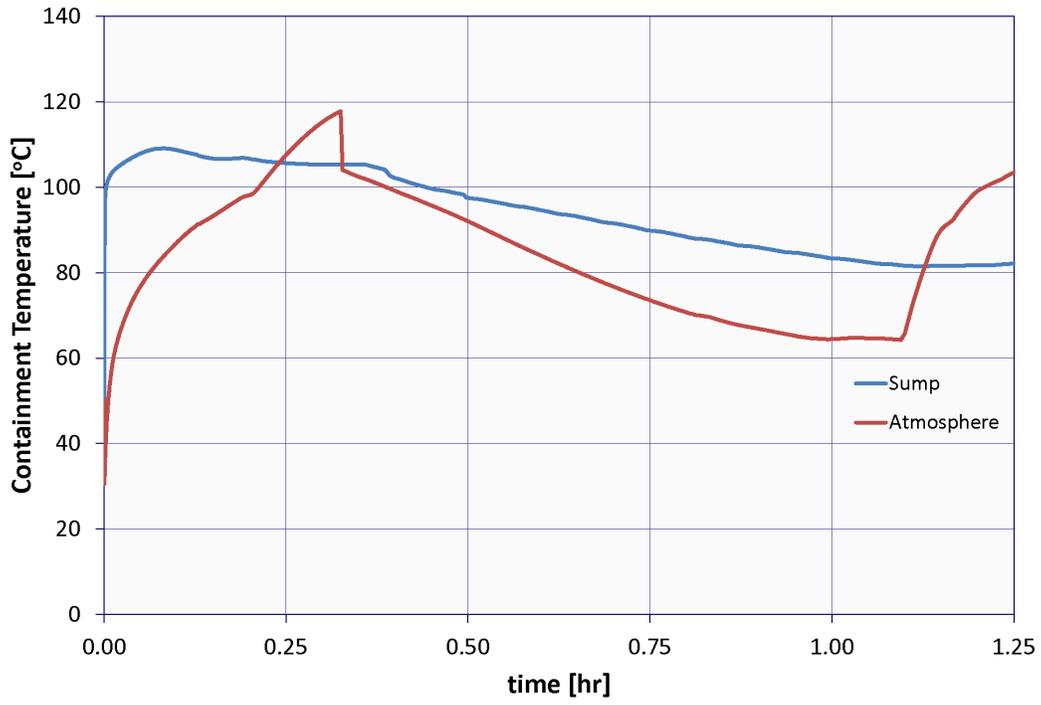


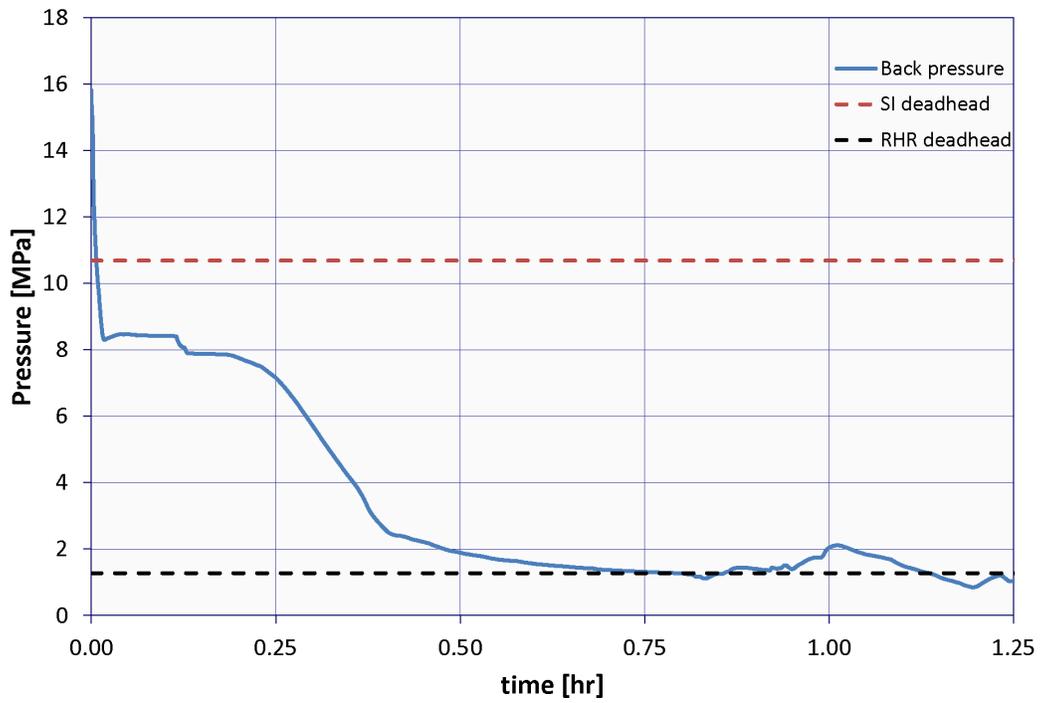
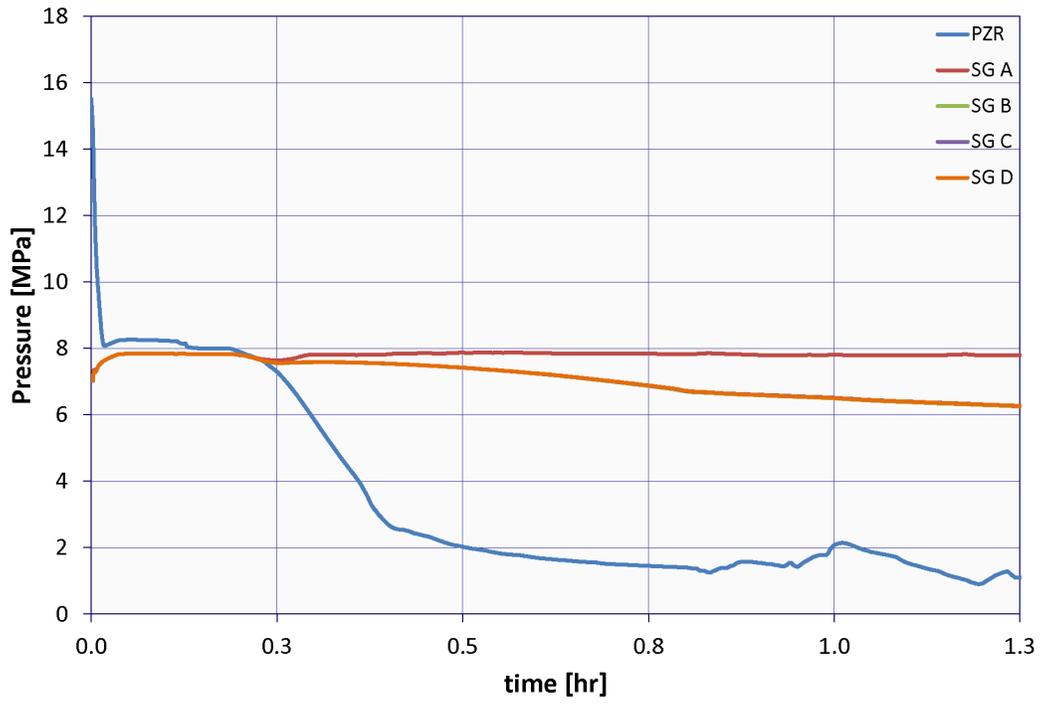


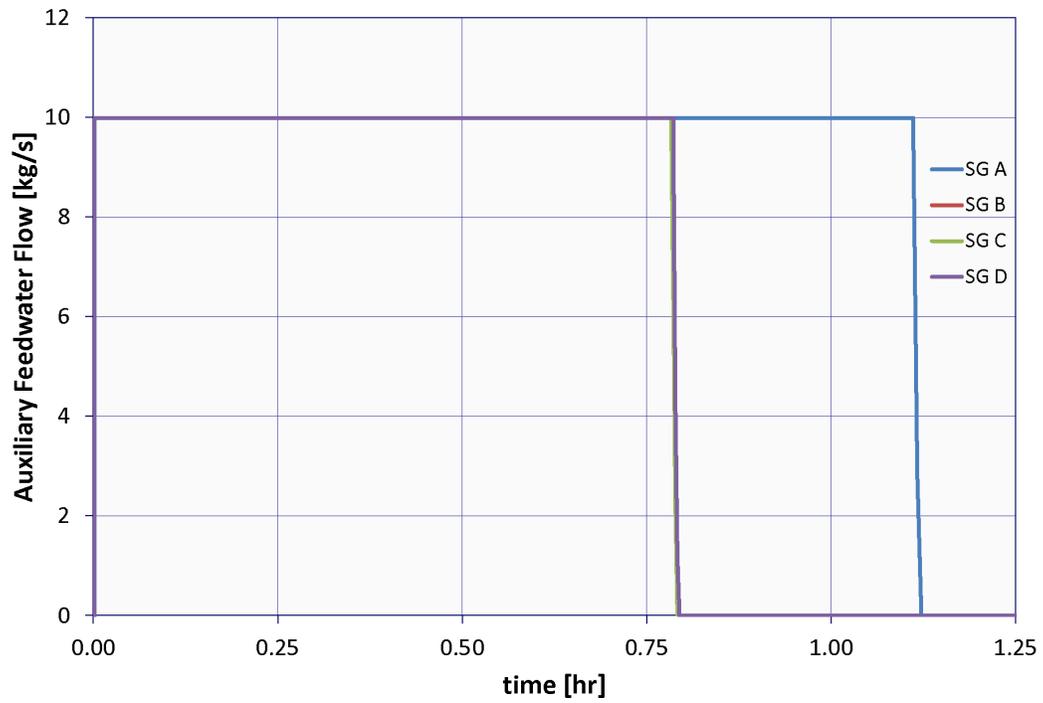
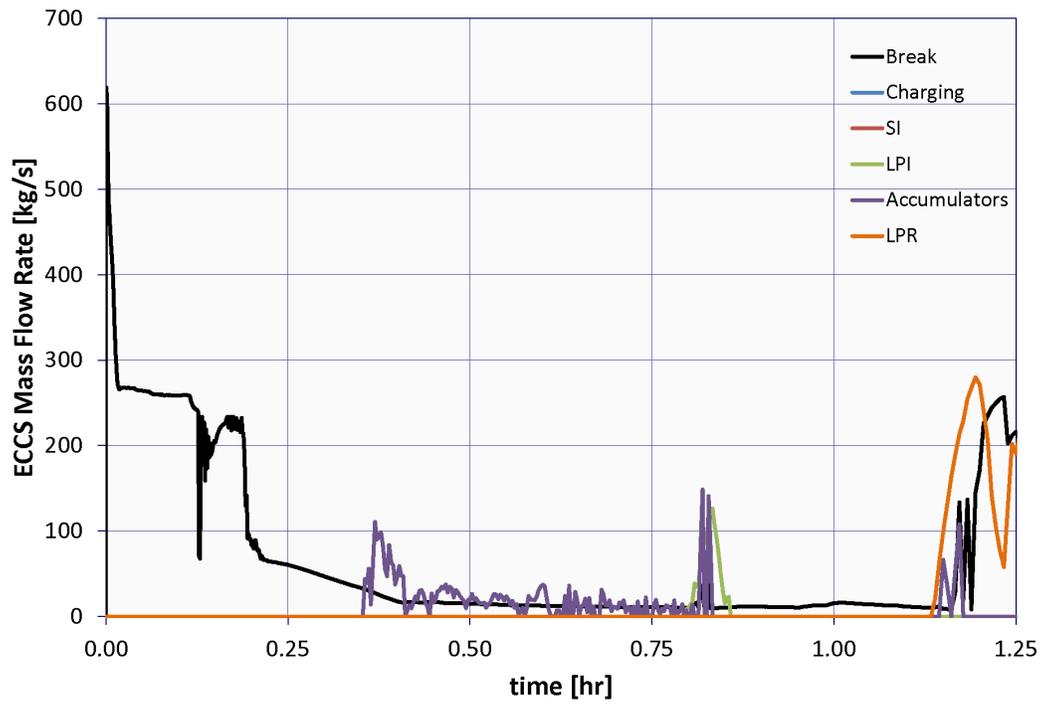


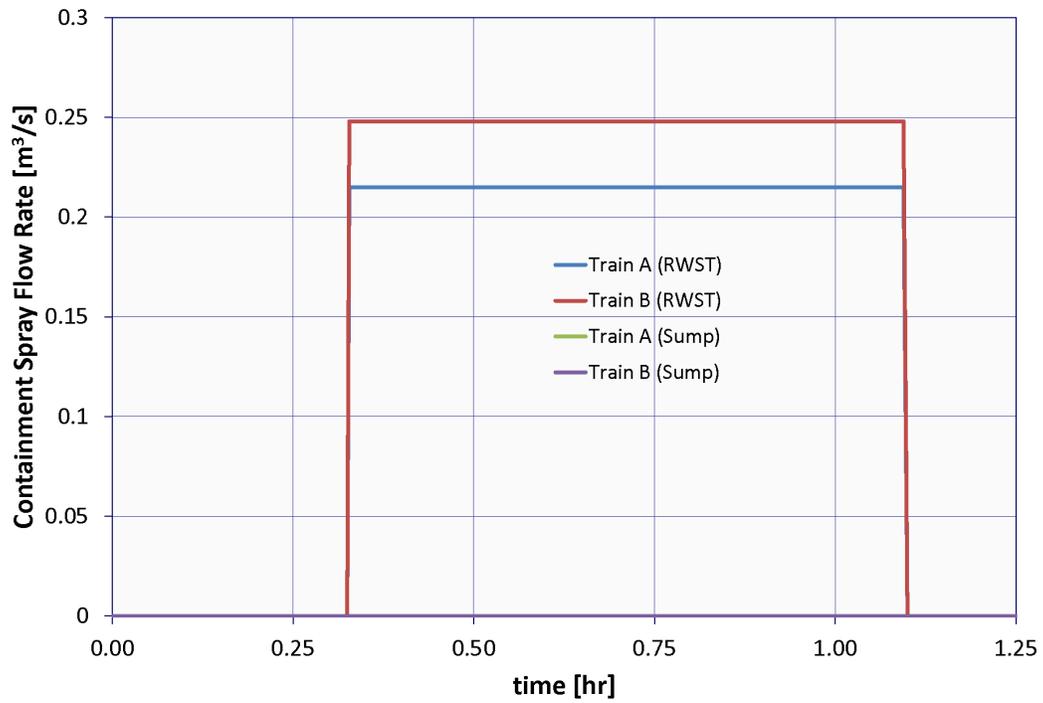
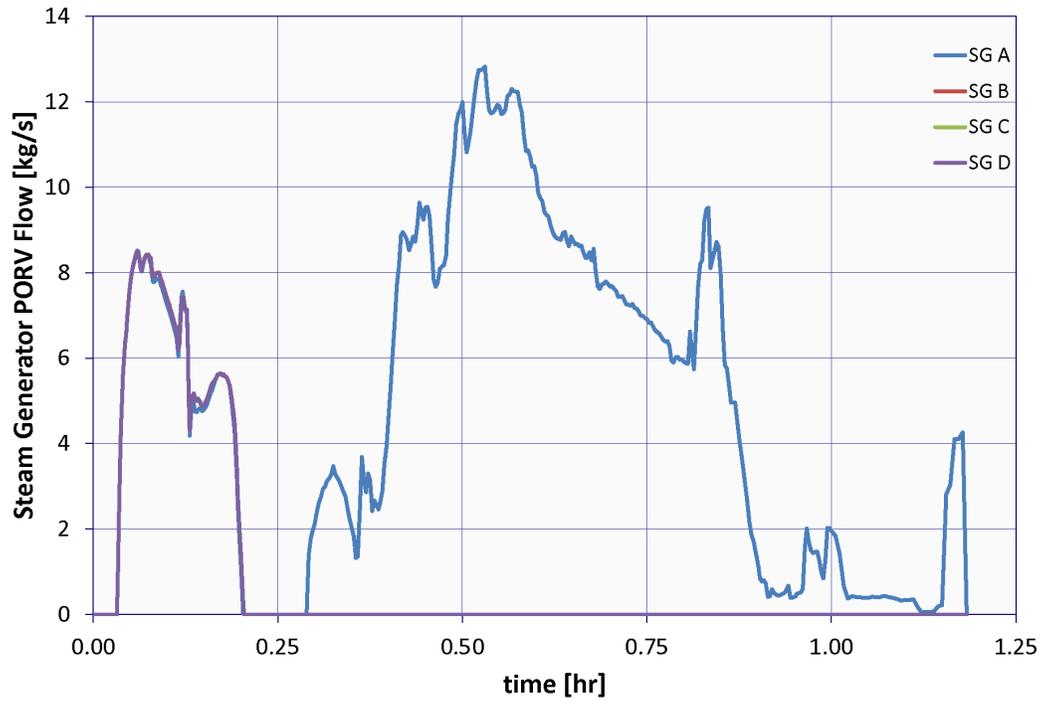
E.1.6 Case 6: 3.33-in. Break with 2 Accumulators, 1 RHR, 2/2 CS Pumps

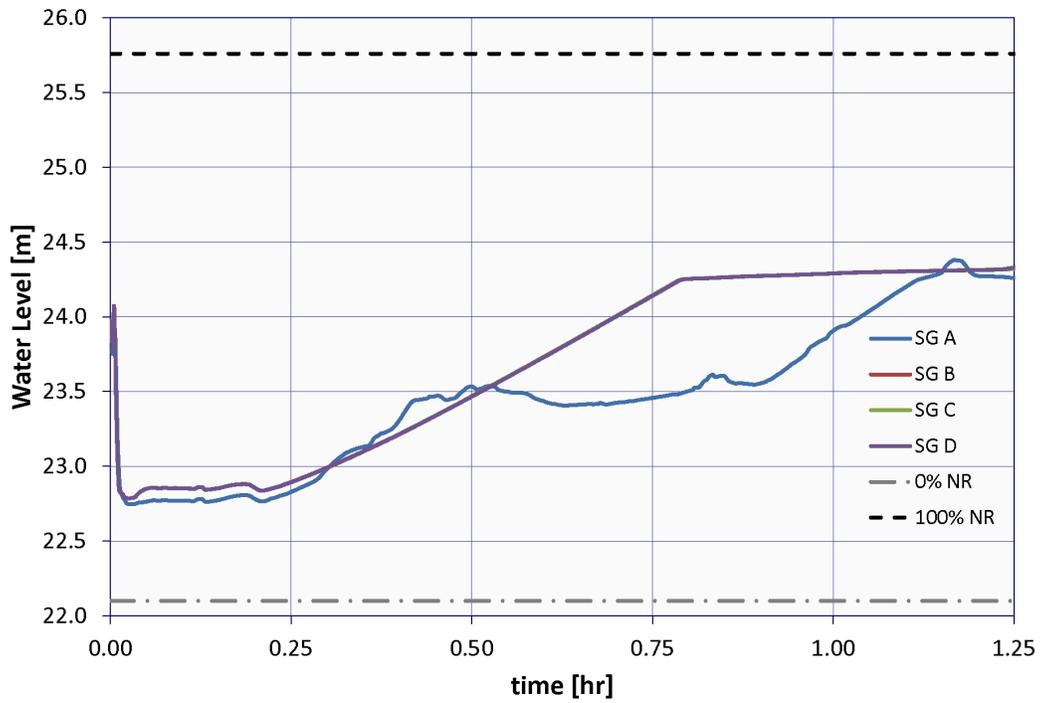
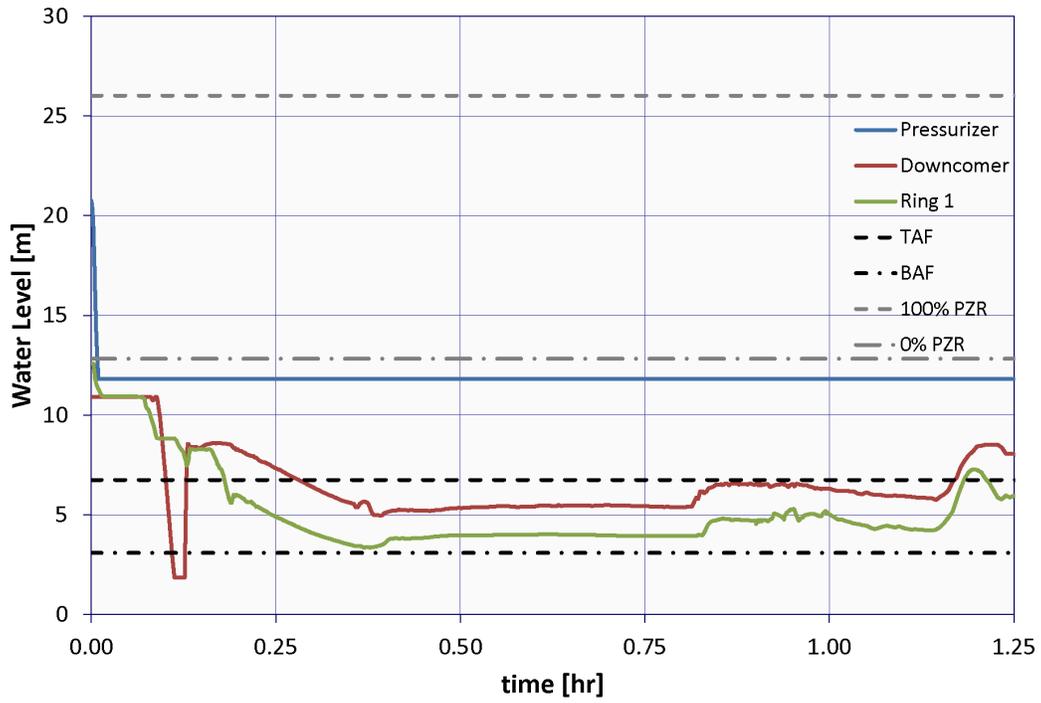


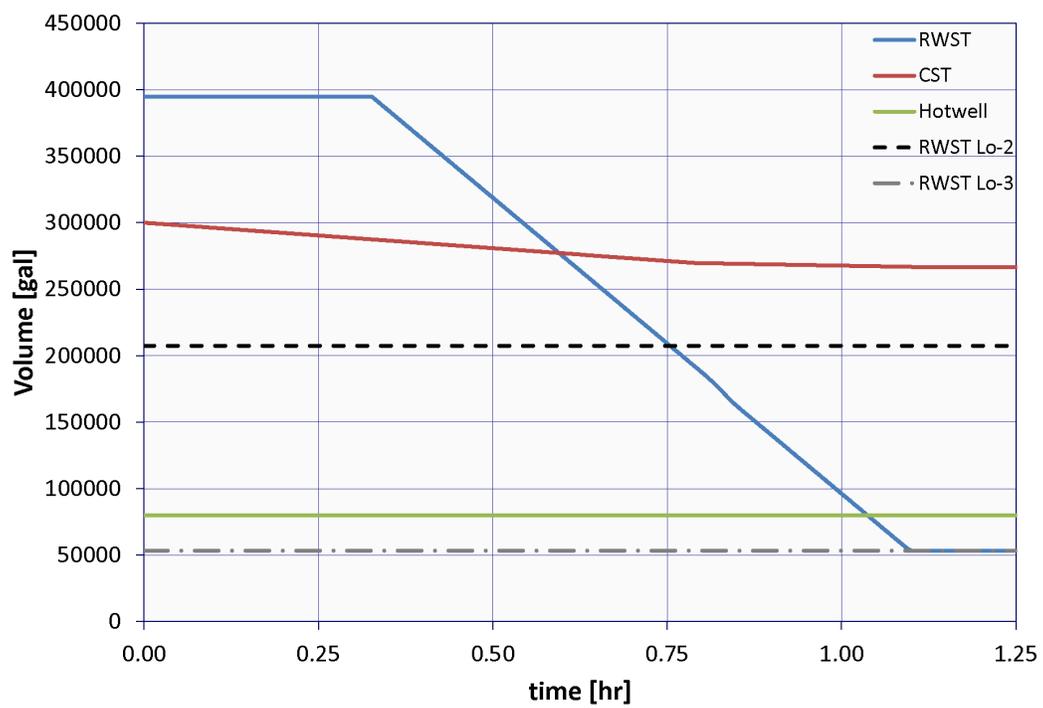




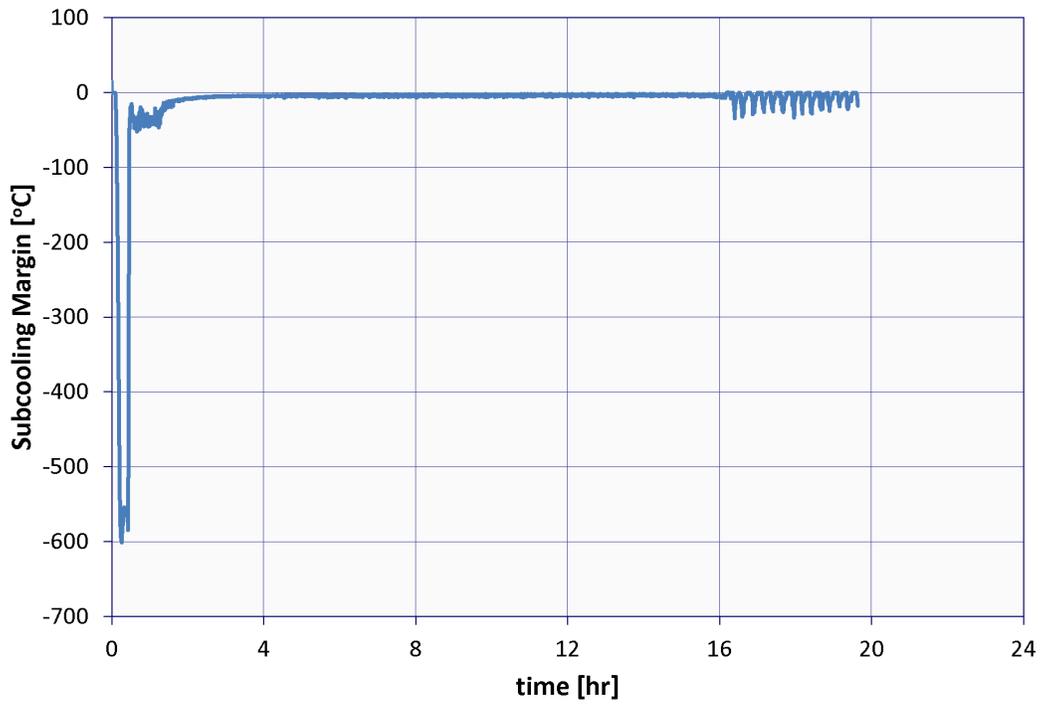
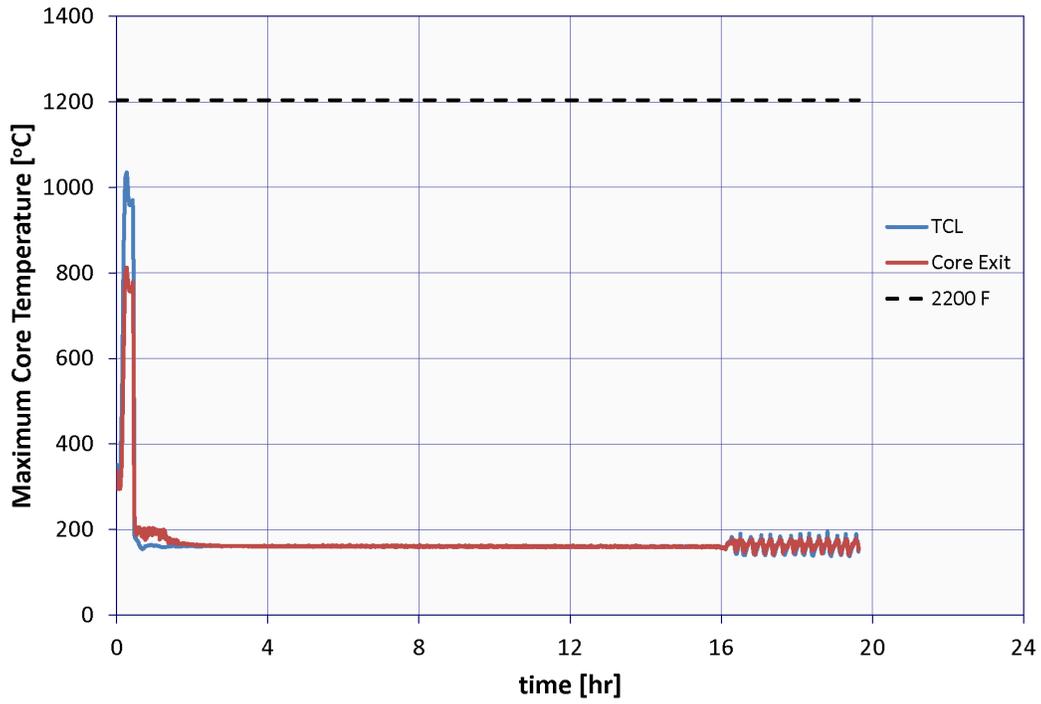


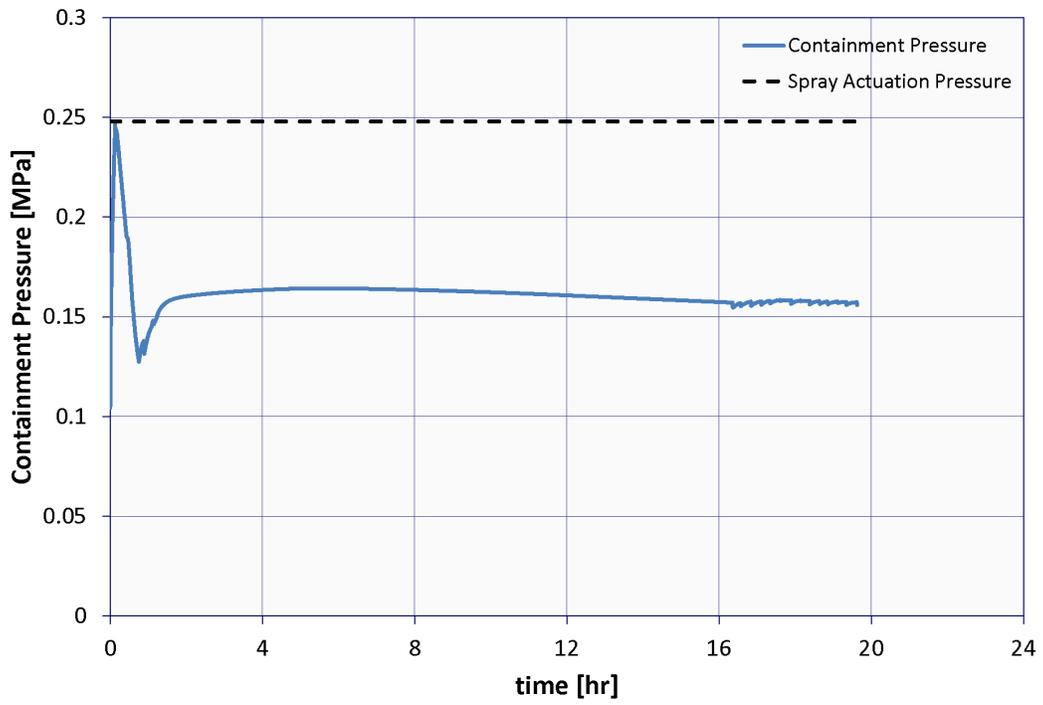
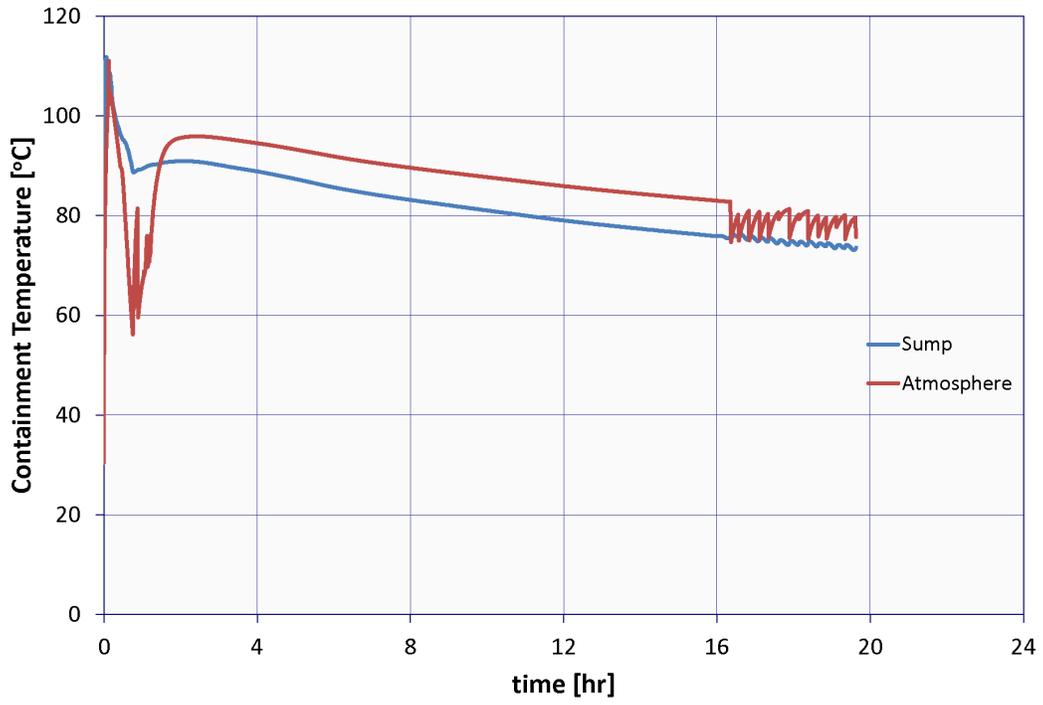


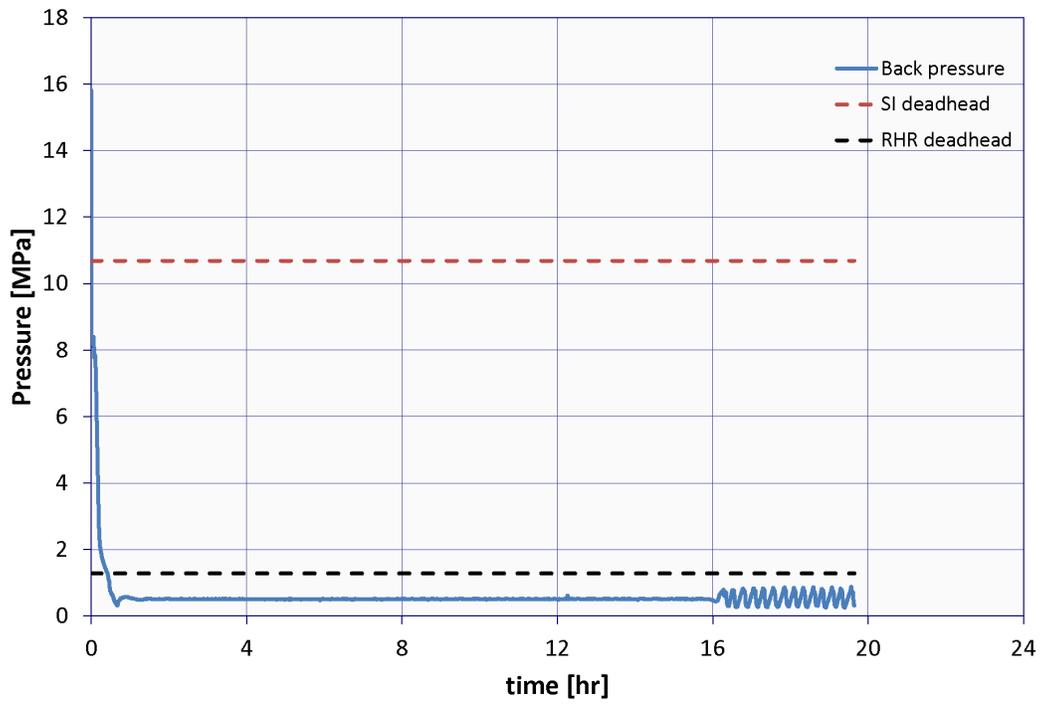
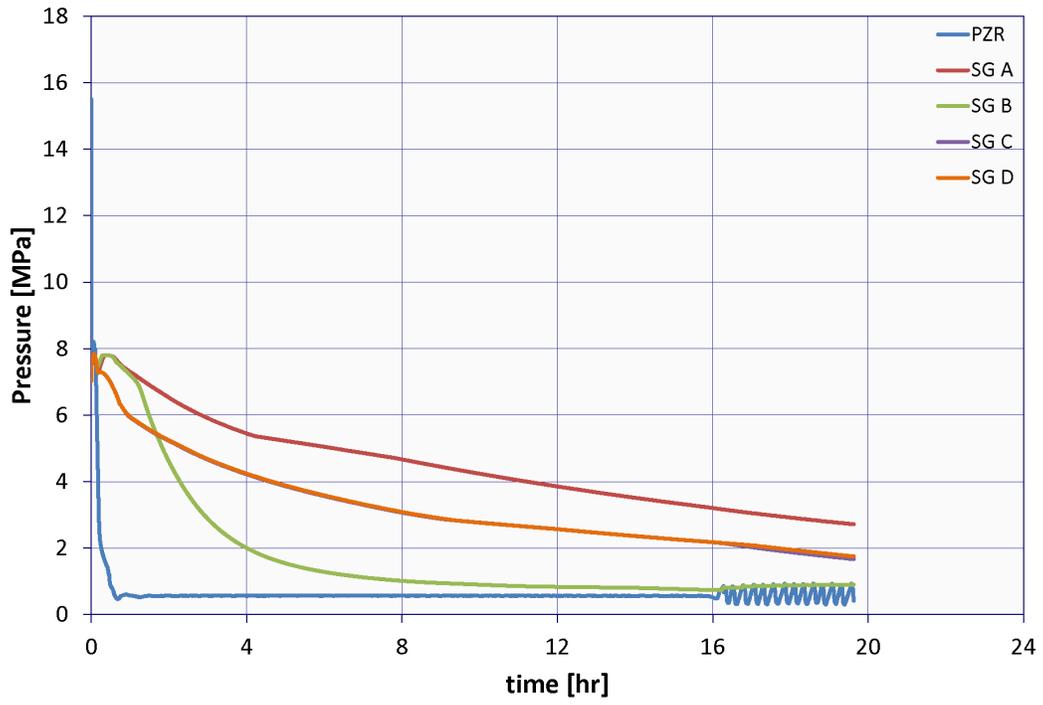


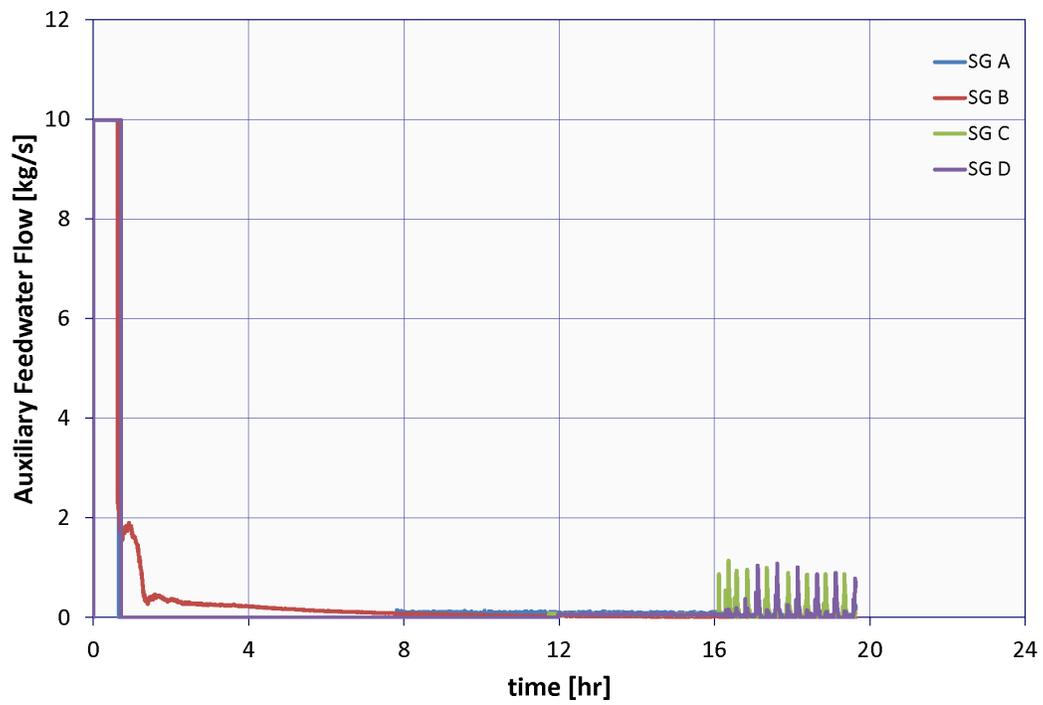
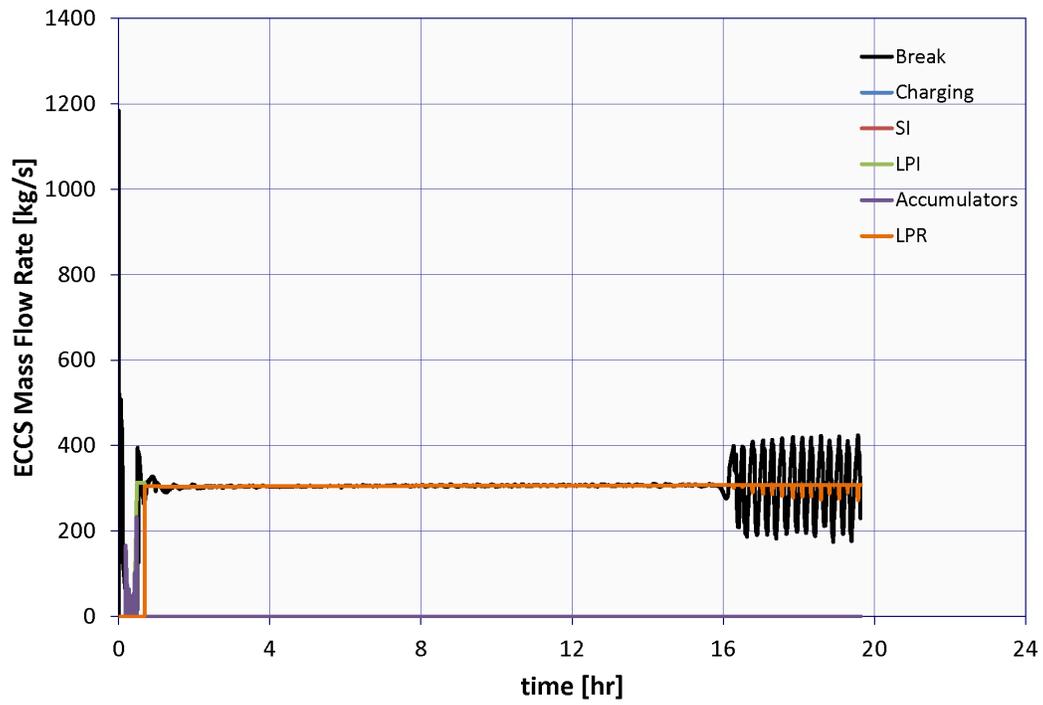


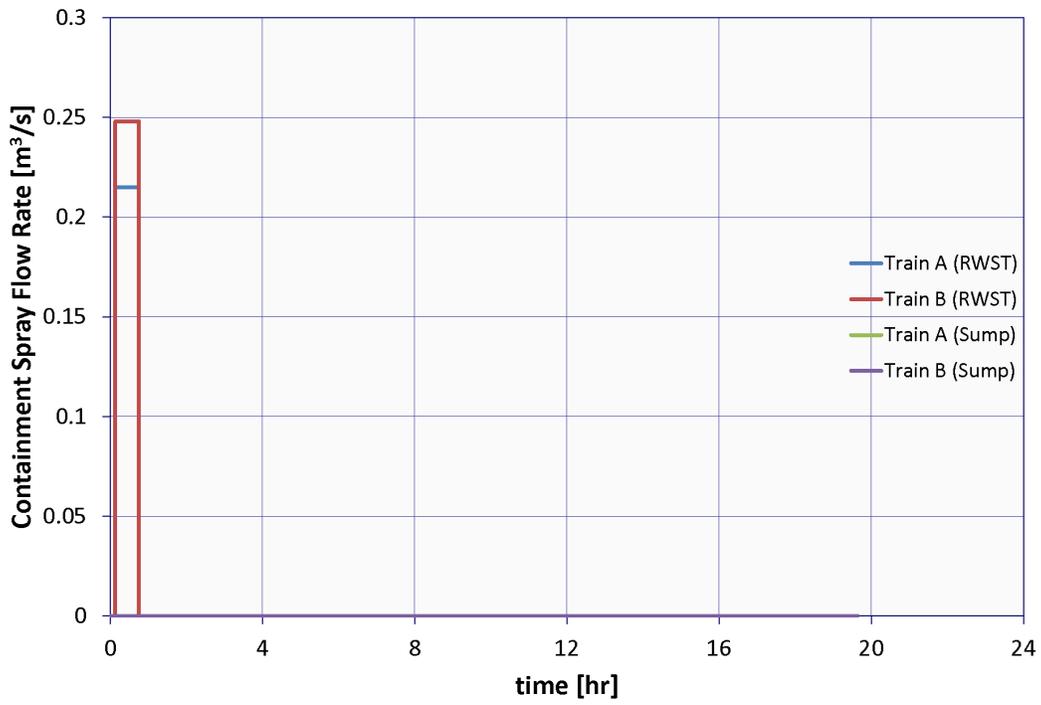
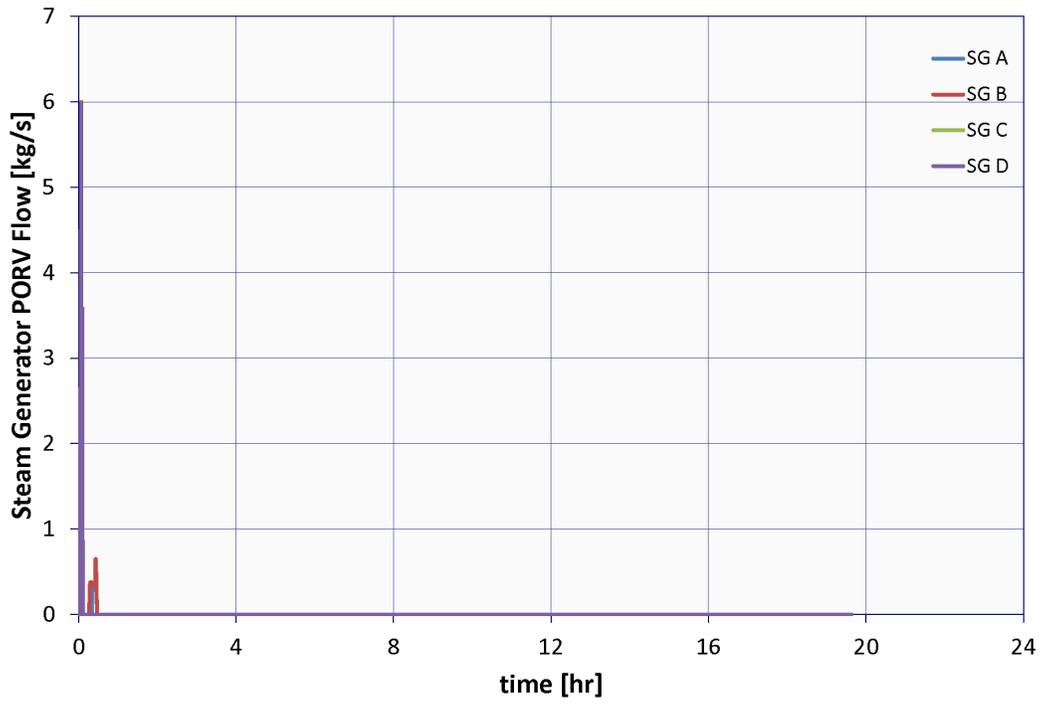
E.1.7 Case 7: 4.67-in. Break with 2 Accumulators, 1 RHR, 2/2 CS Pumps

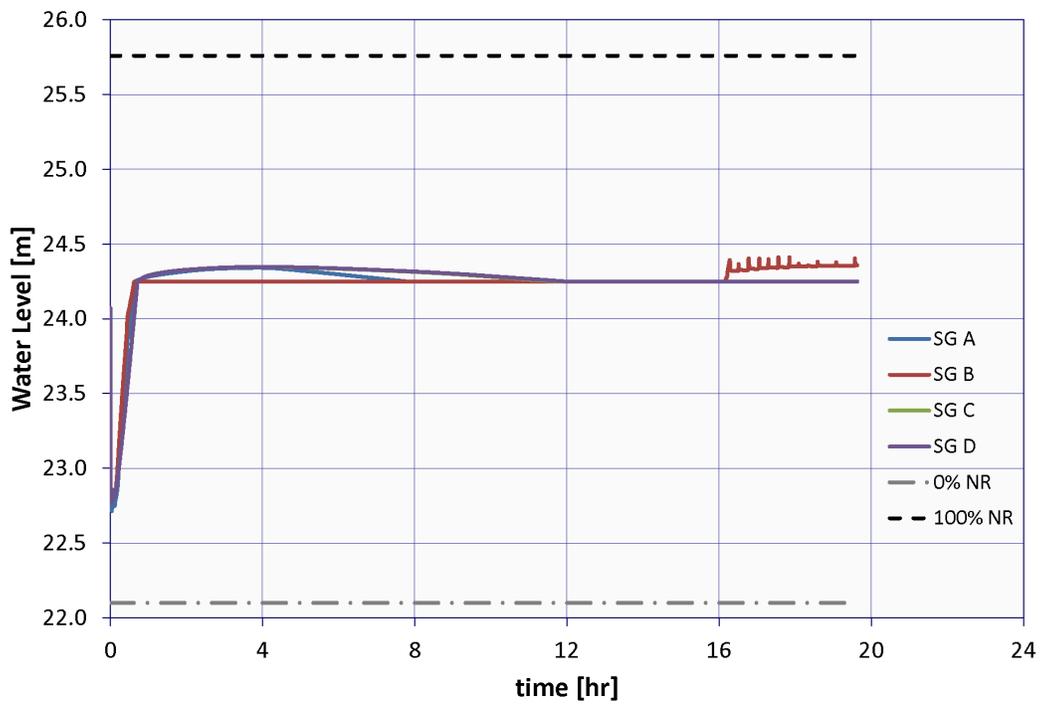
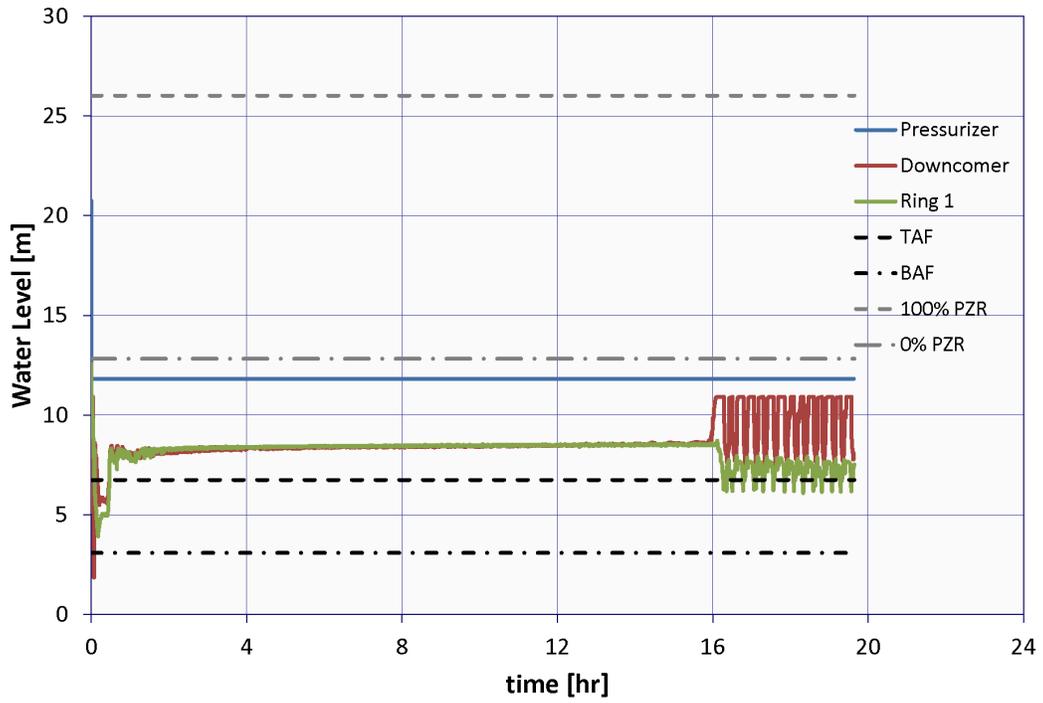


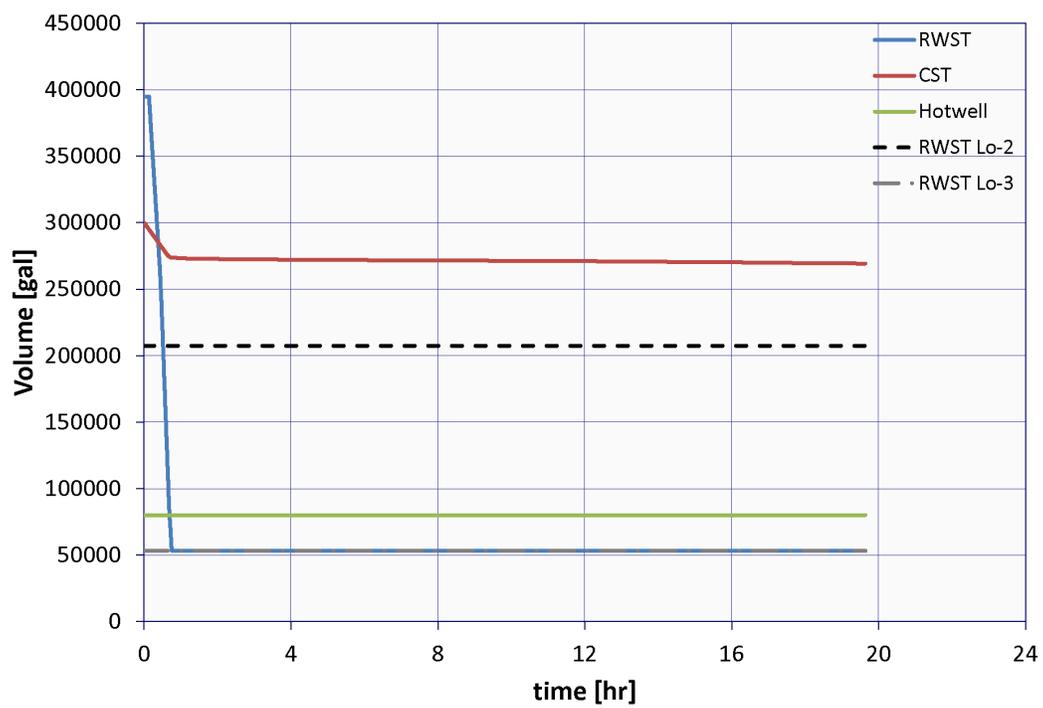




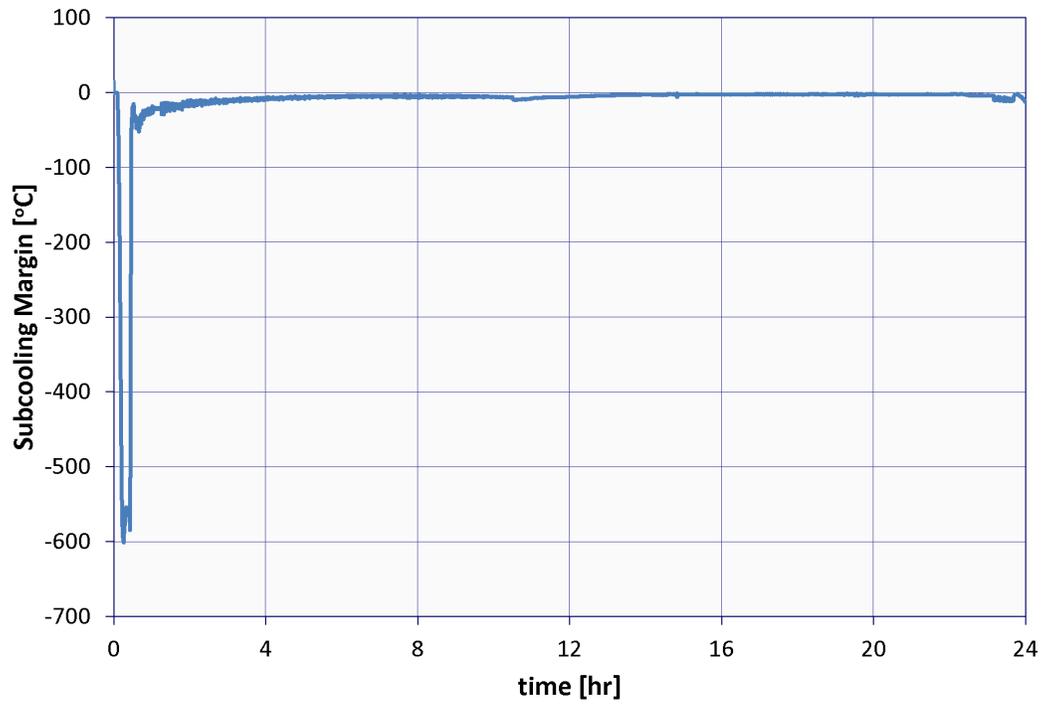
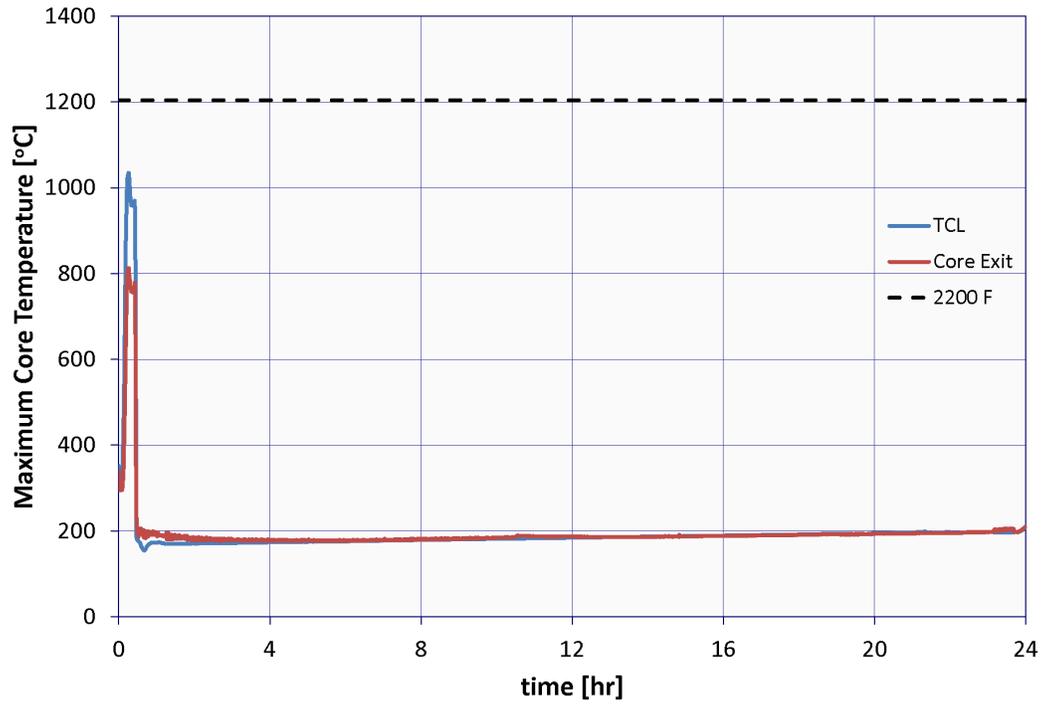


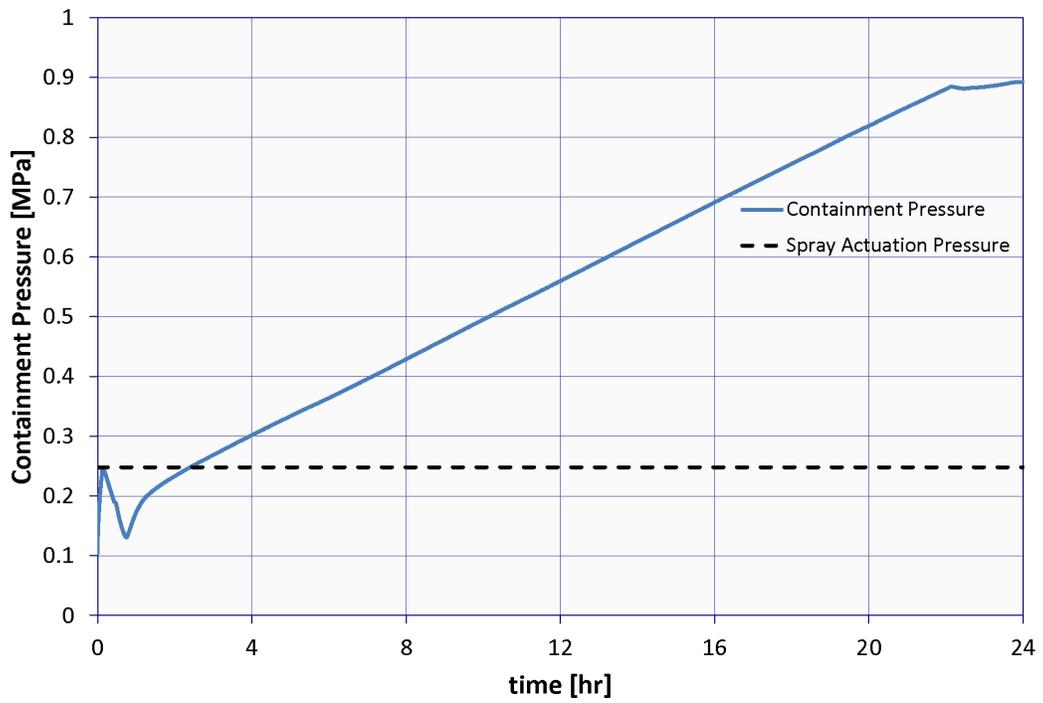
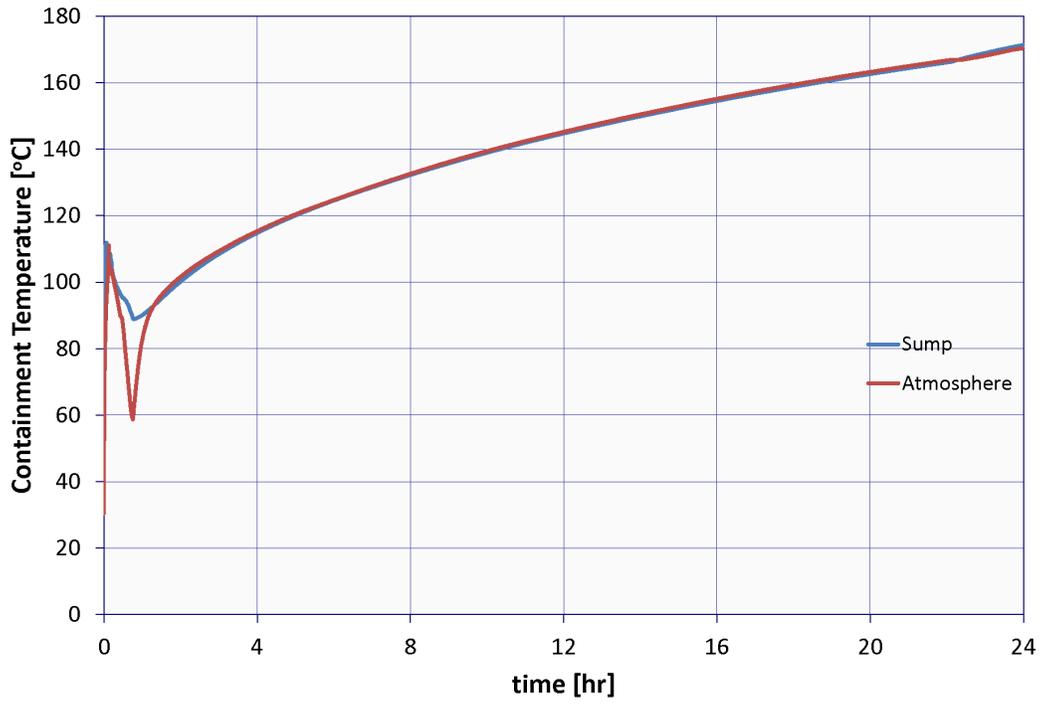


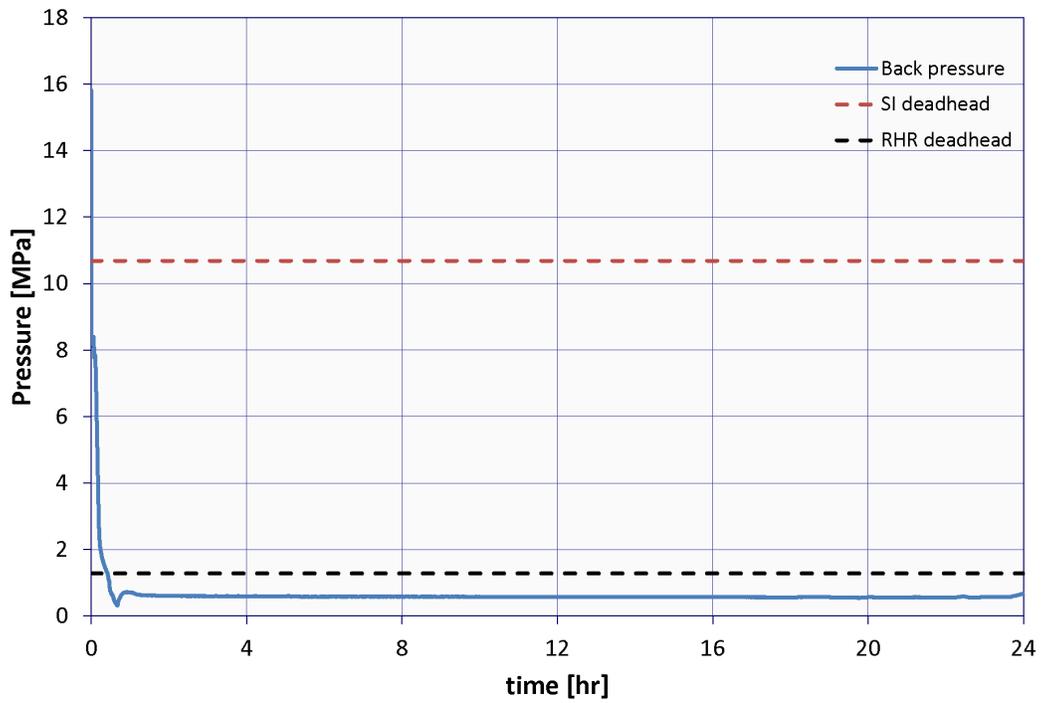
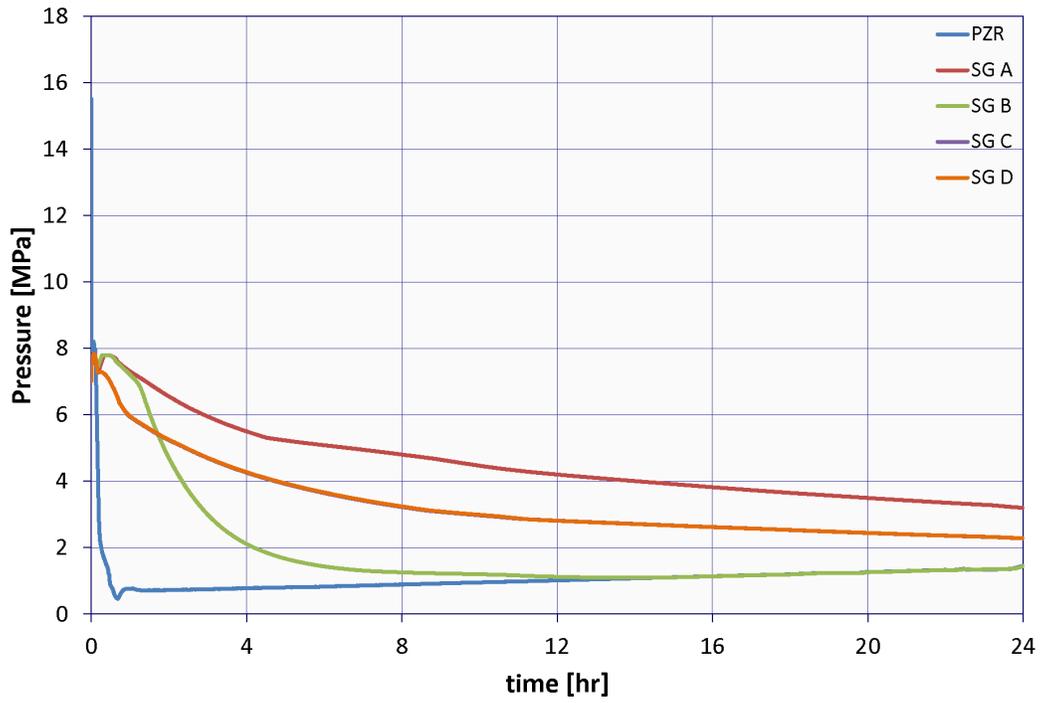


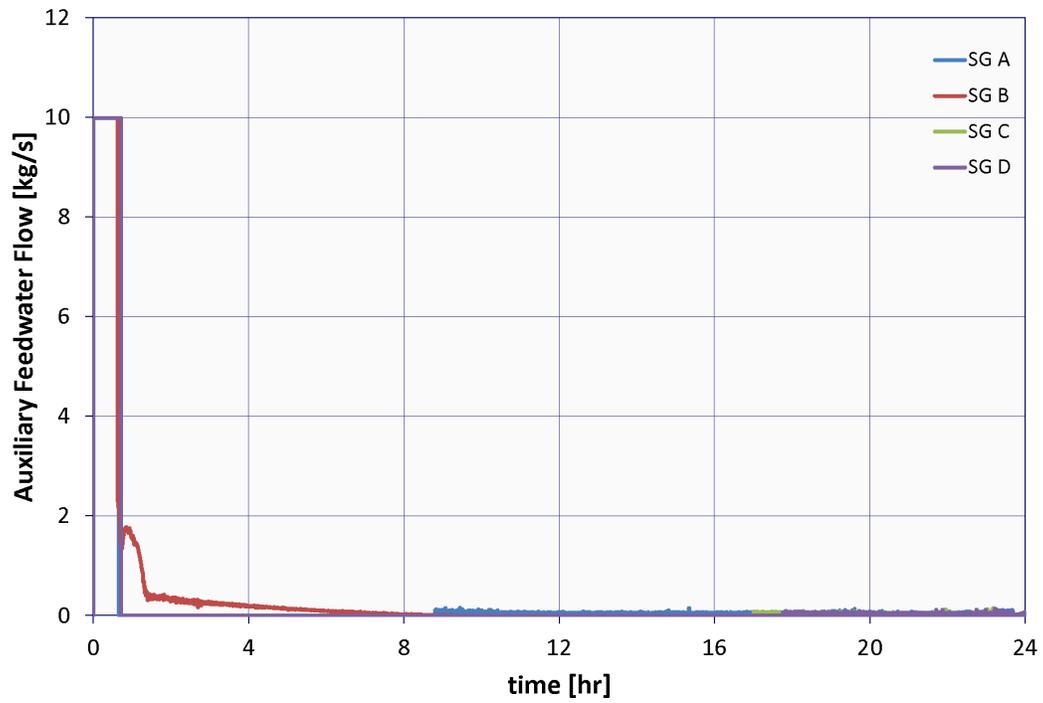
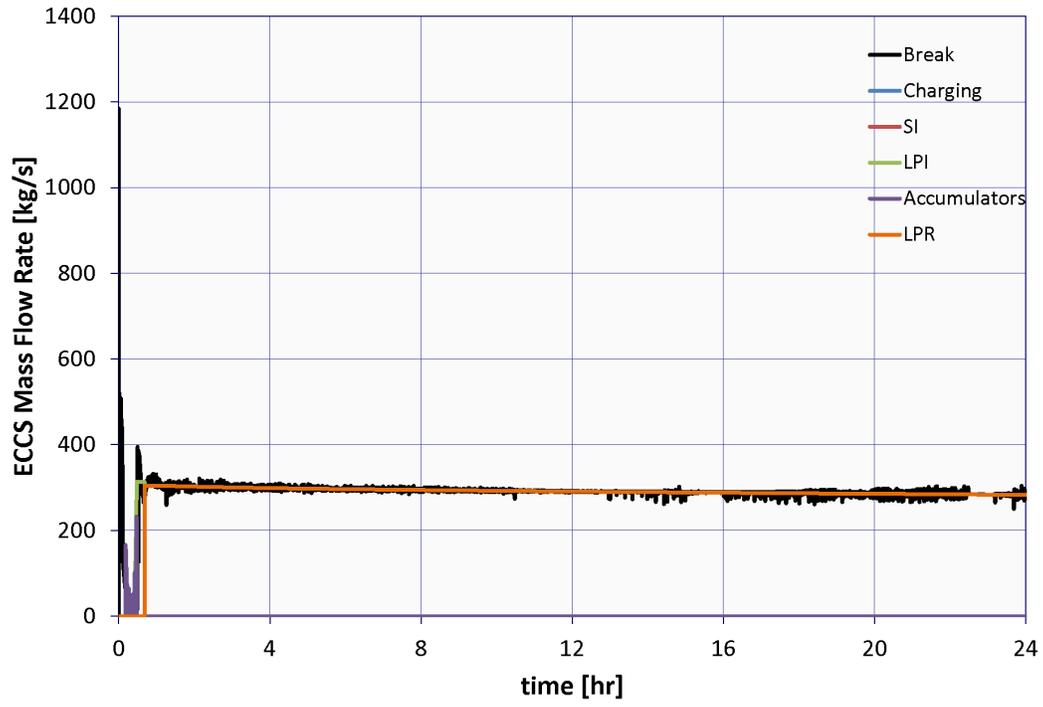


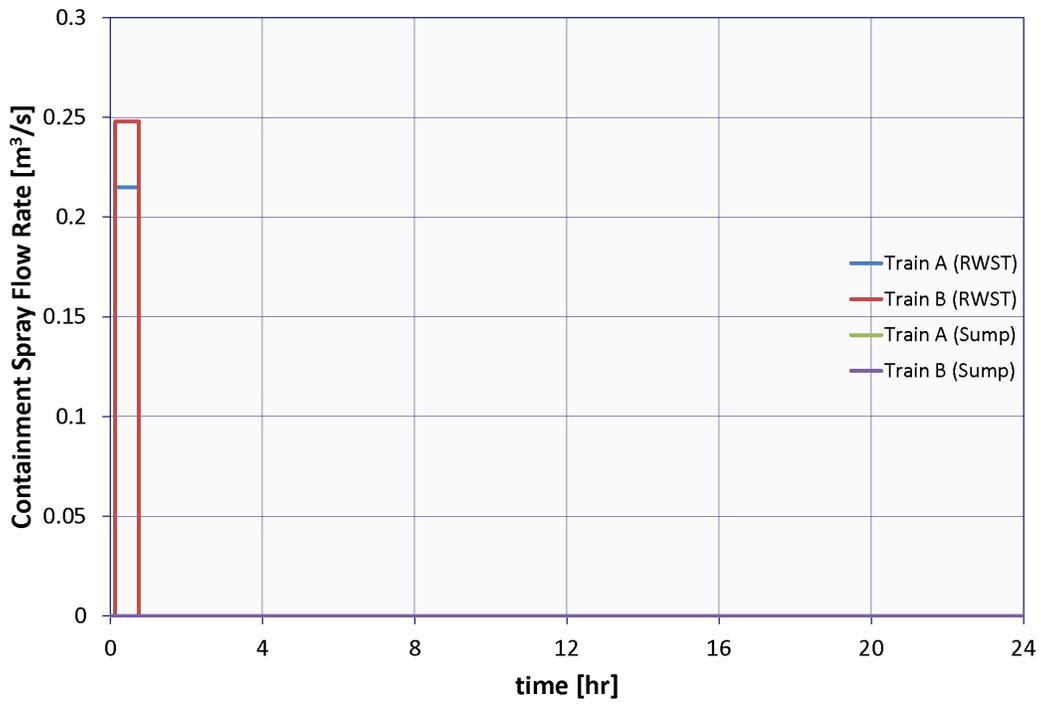
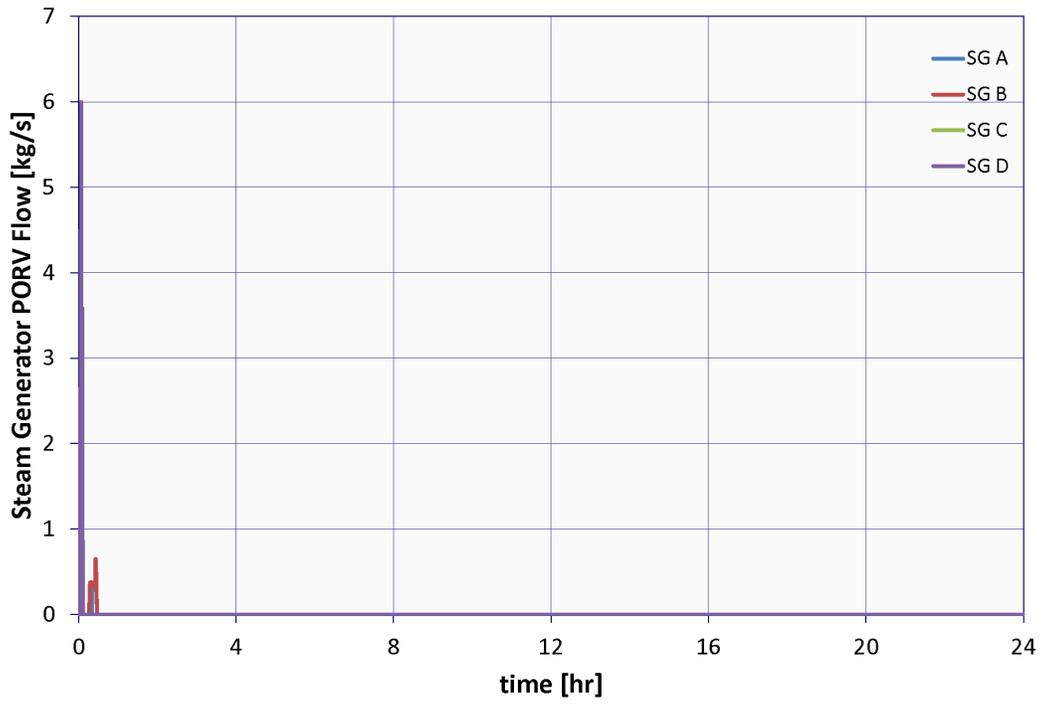
E.1.7.1 Case 7a: 4.67-in. Break with 2 Accumulators, 1 RHR, 2/2 CS Pumps, No RHRHX

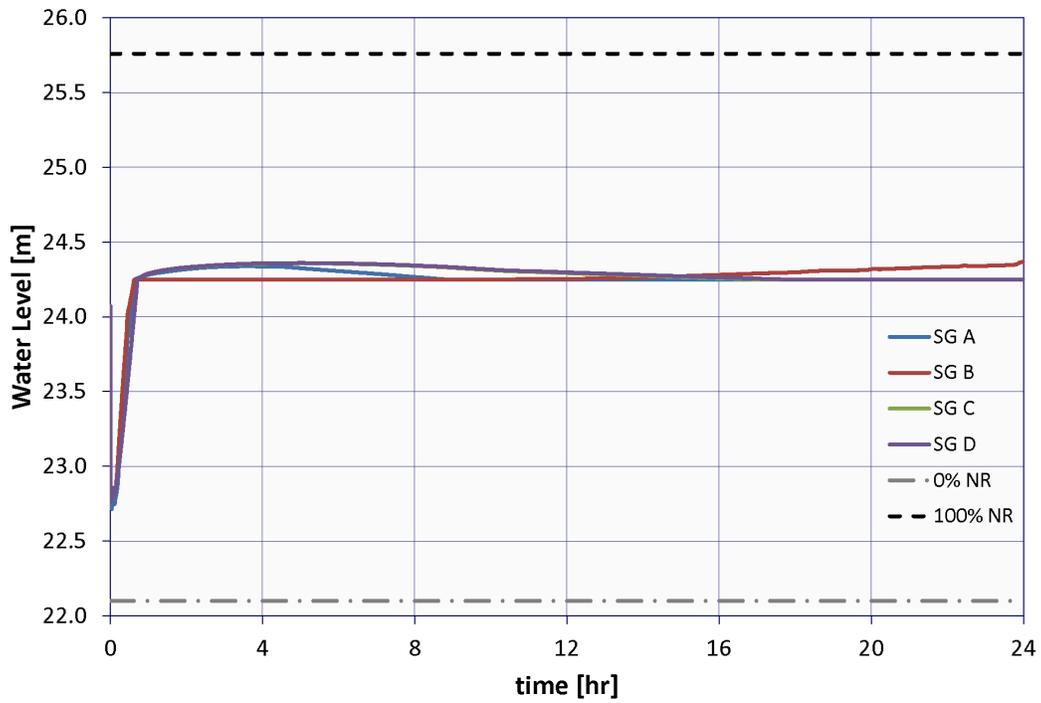
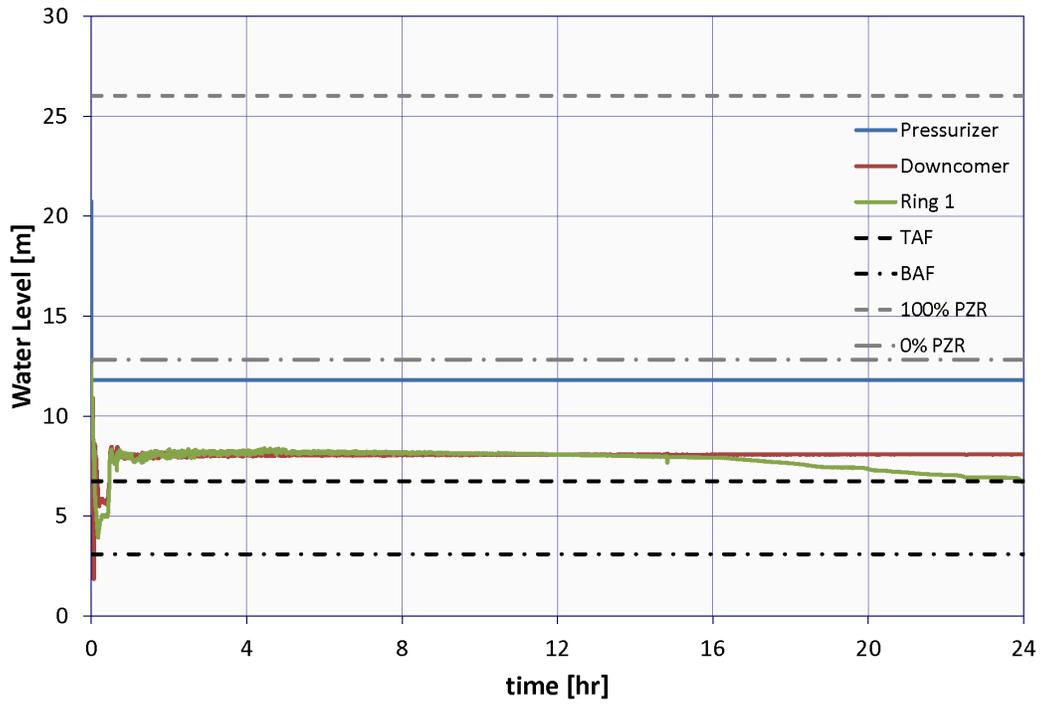


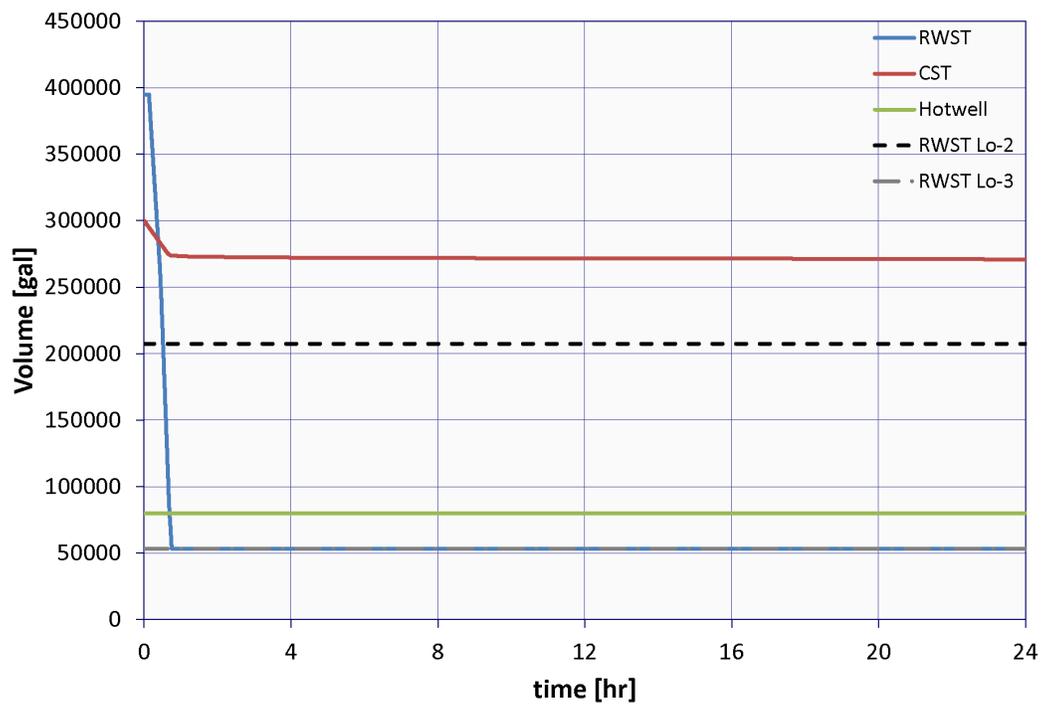




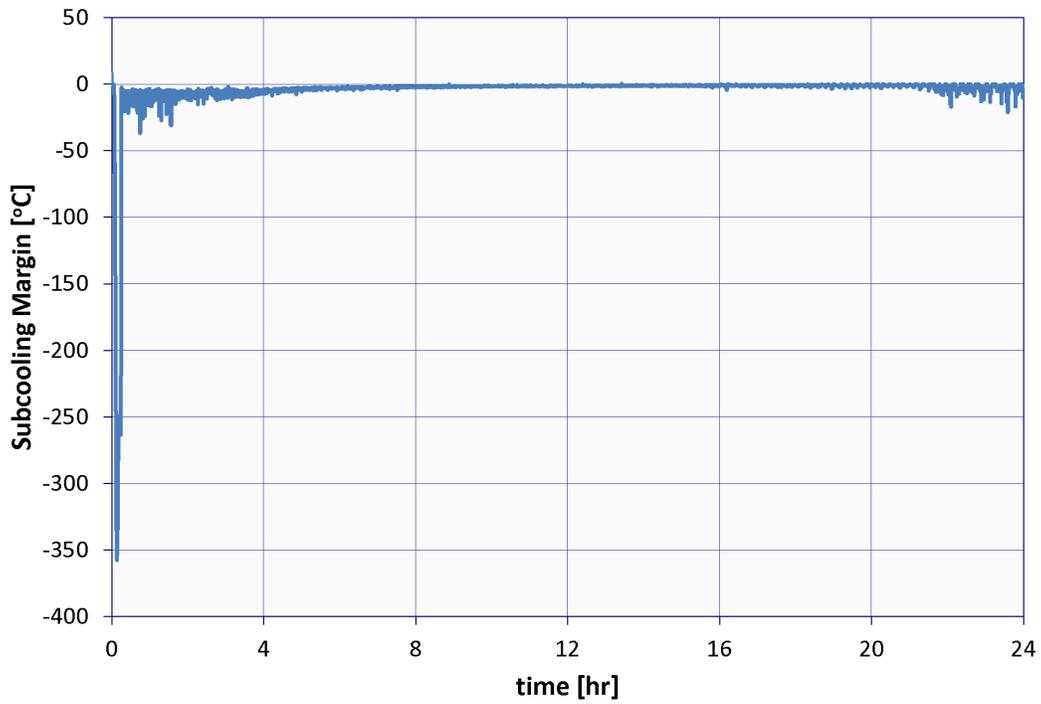
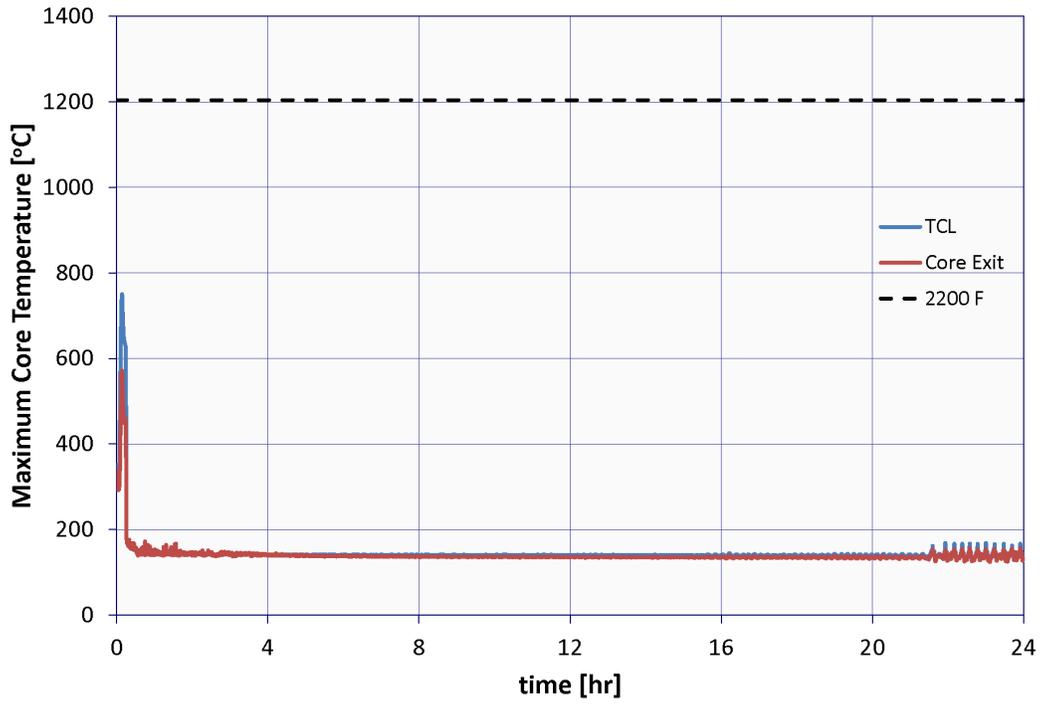


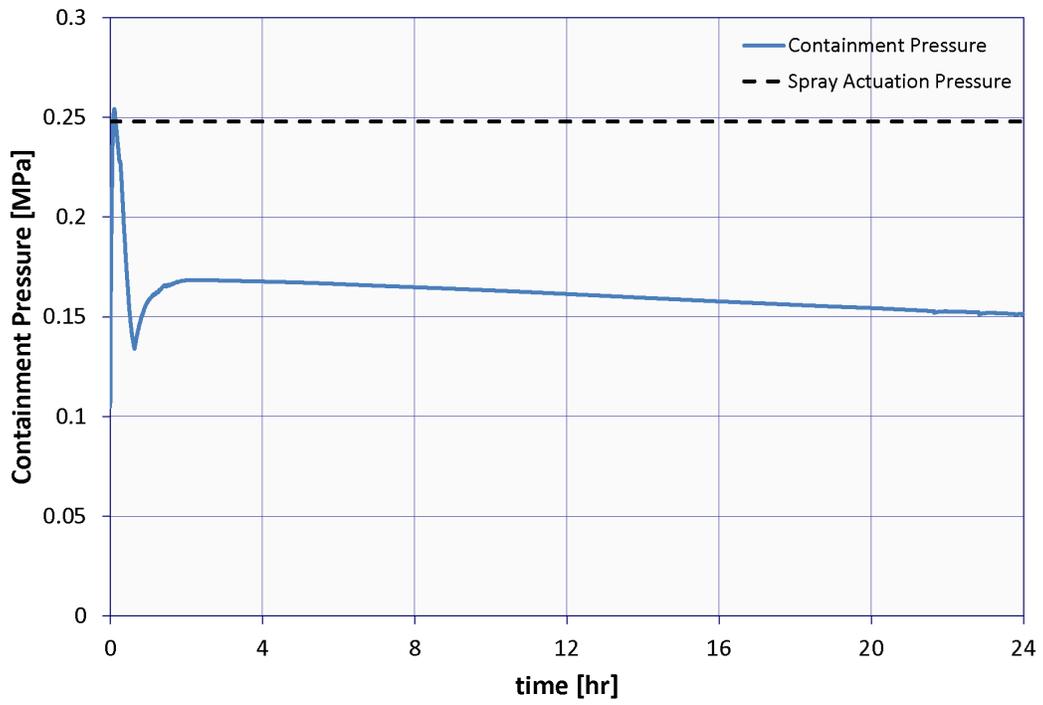
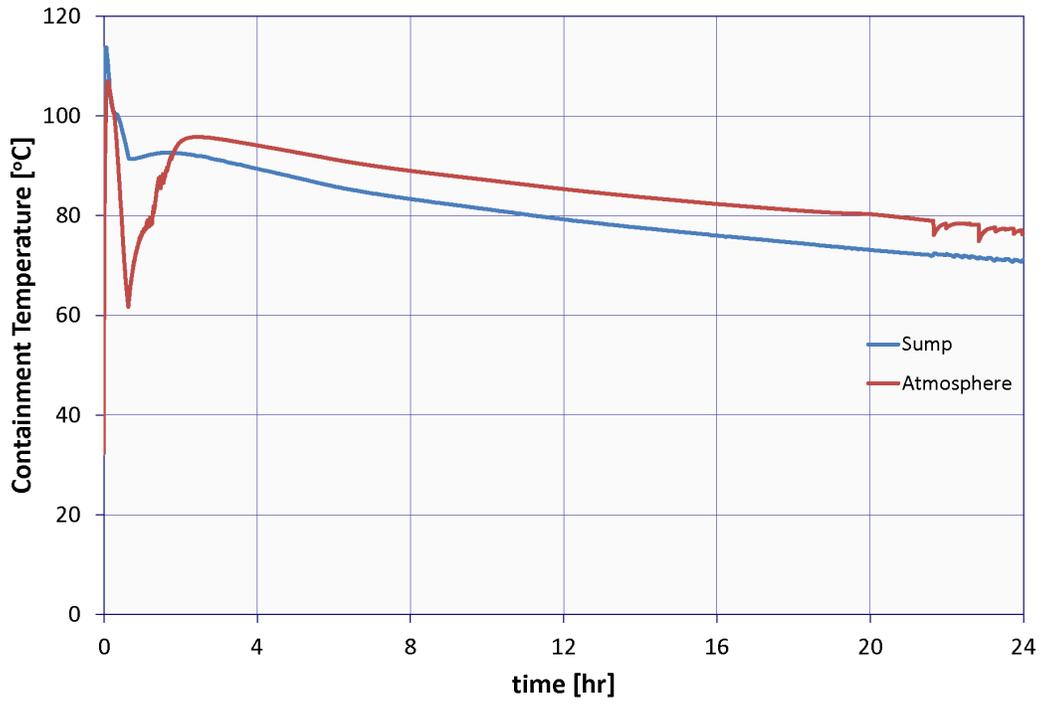


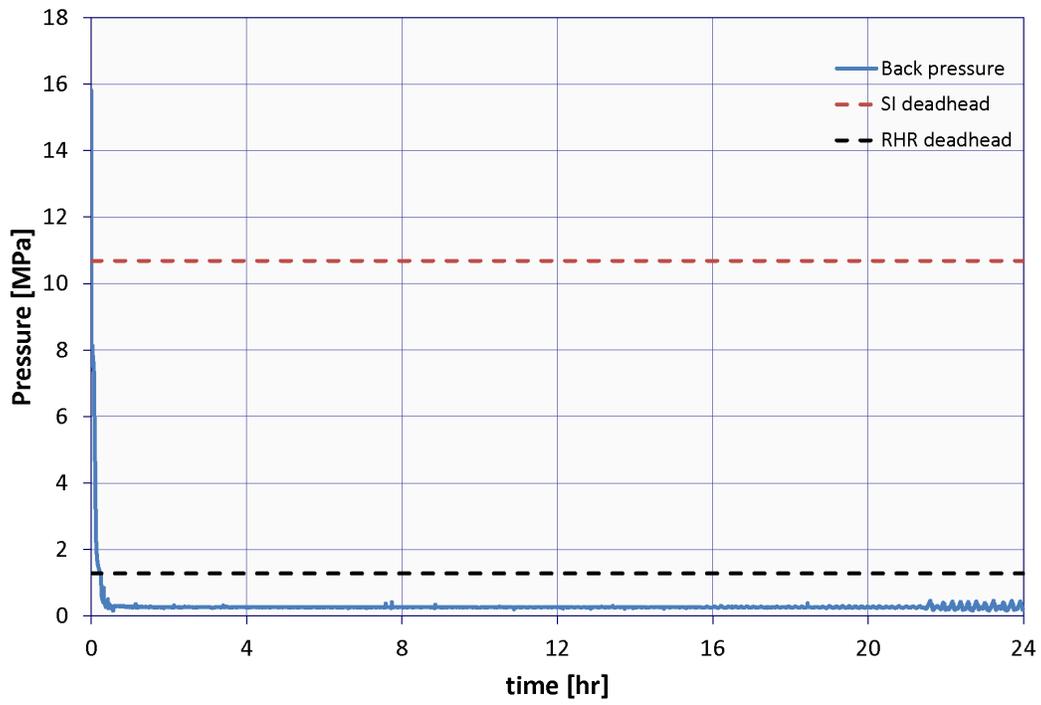
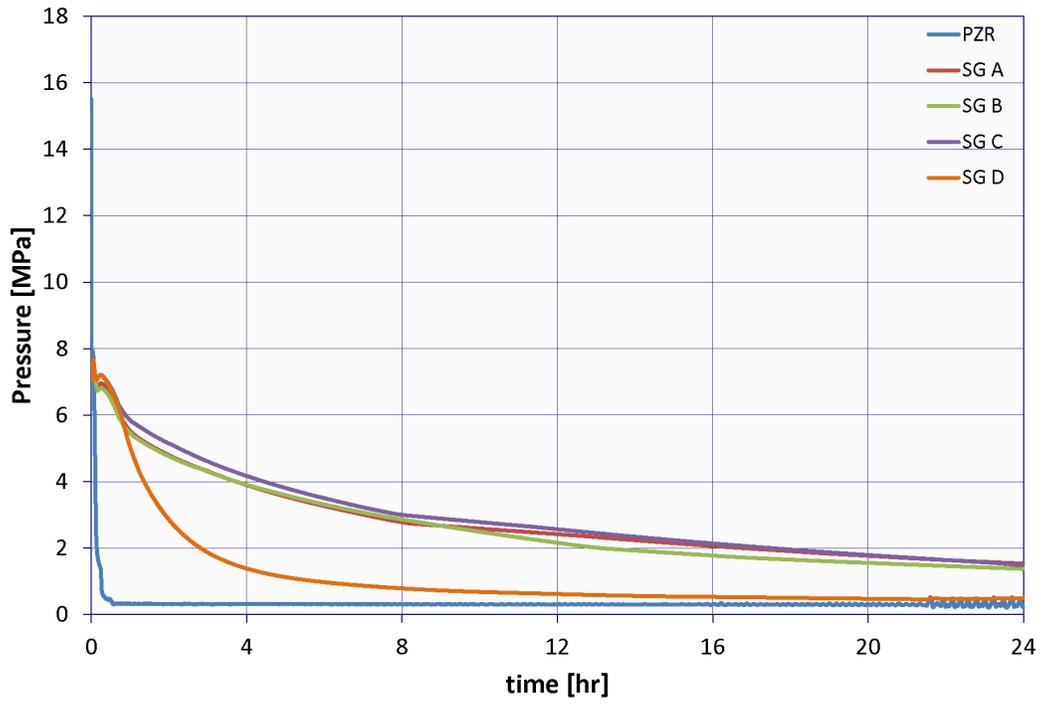


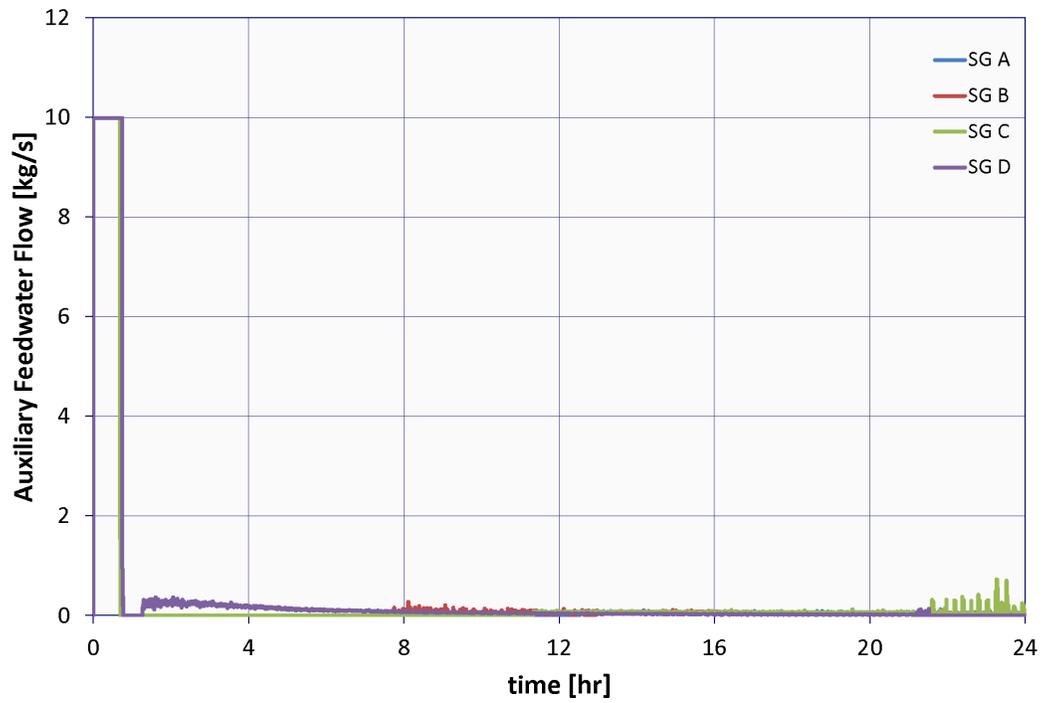
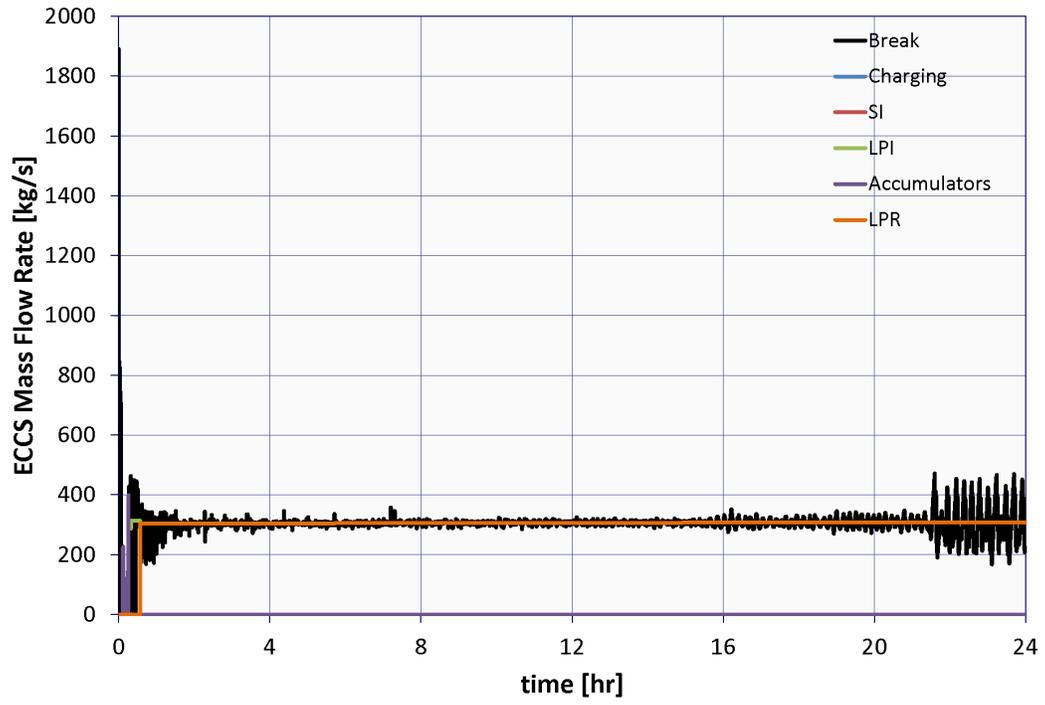


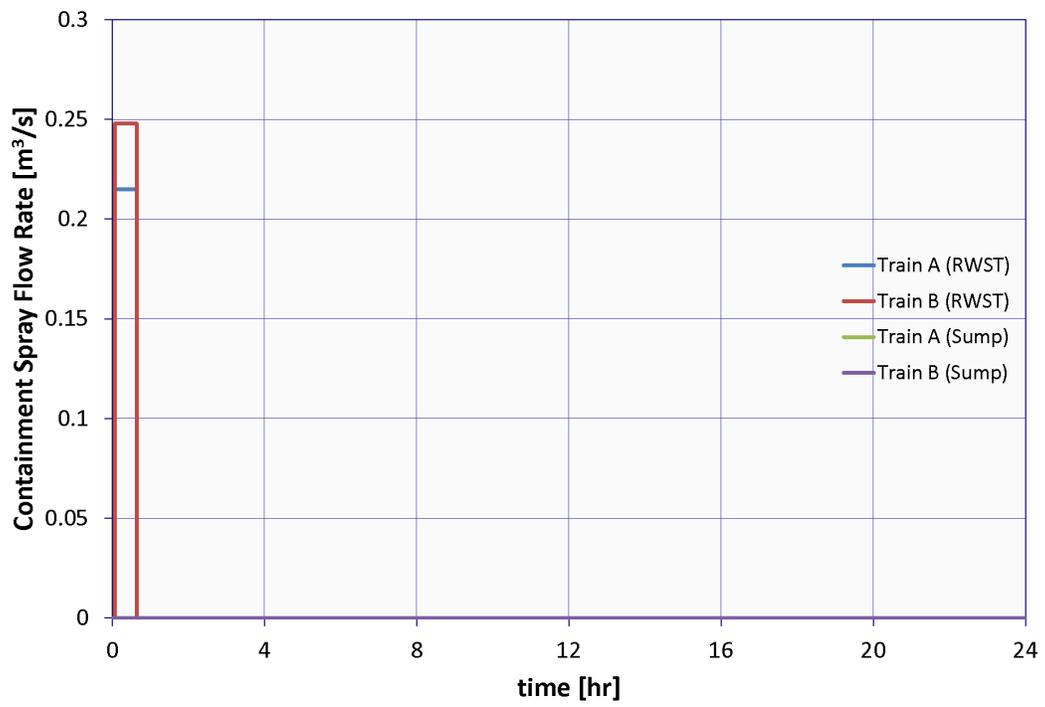
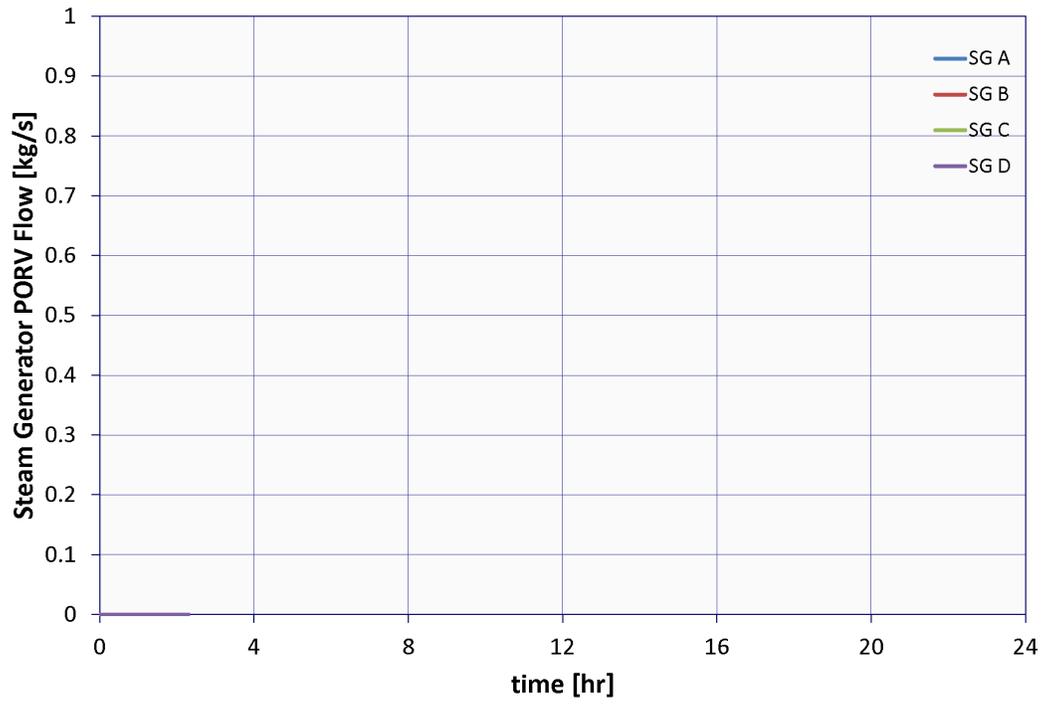
E.1.8 Case 8: 6-in. Break with 2 Accumulators, 1 RHR, 2/2 CS Pumps

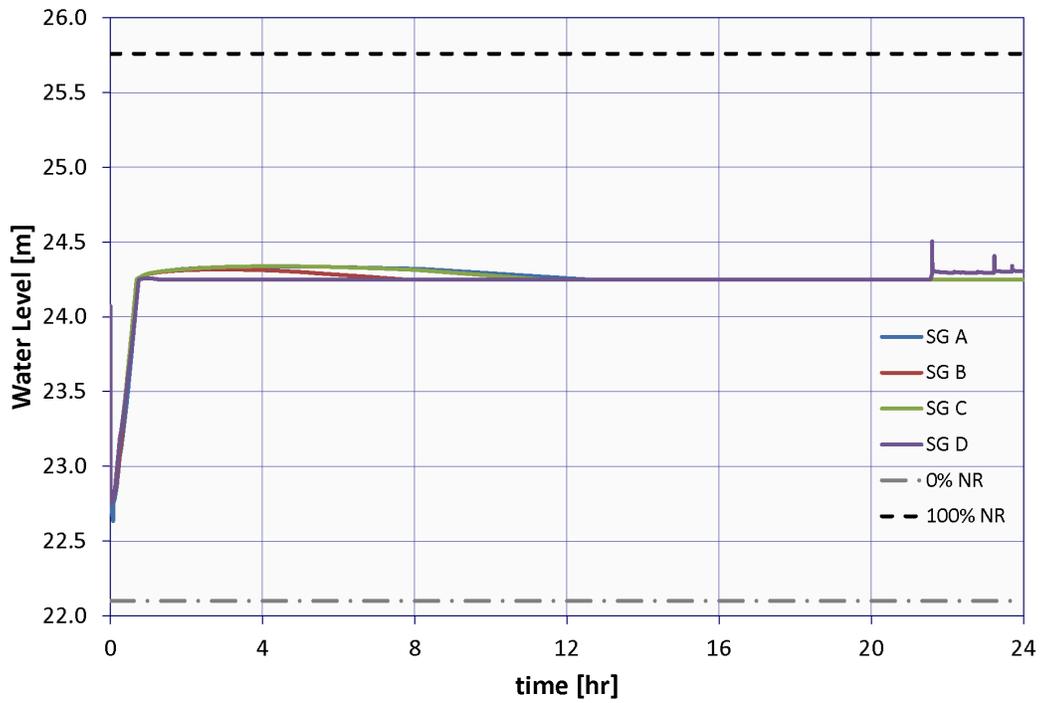
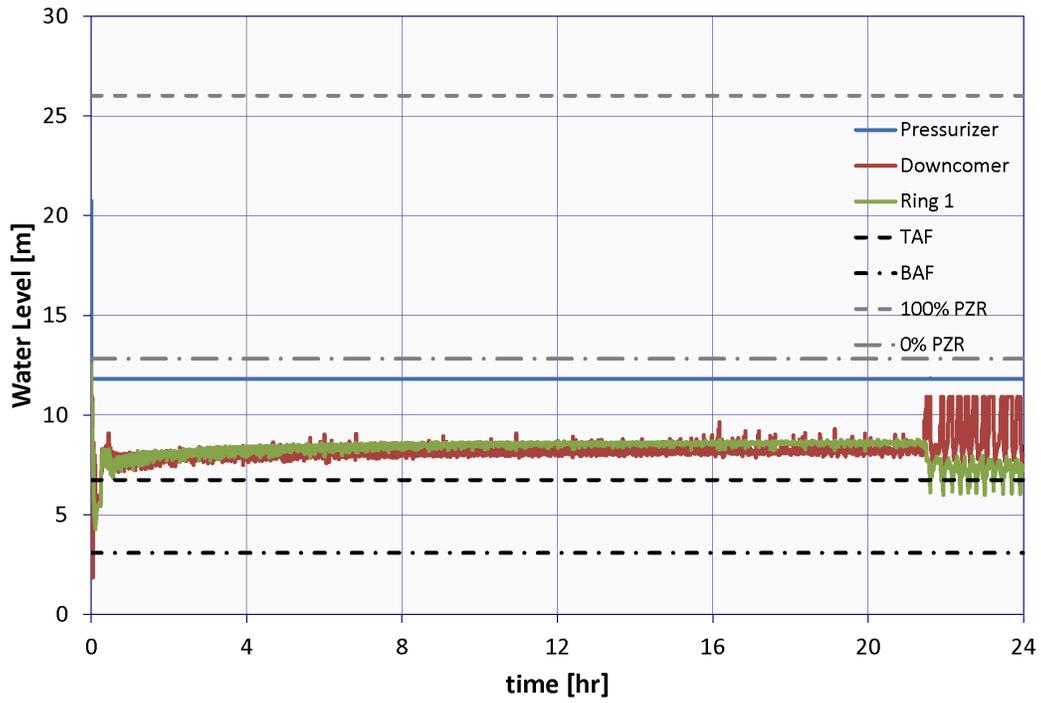


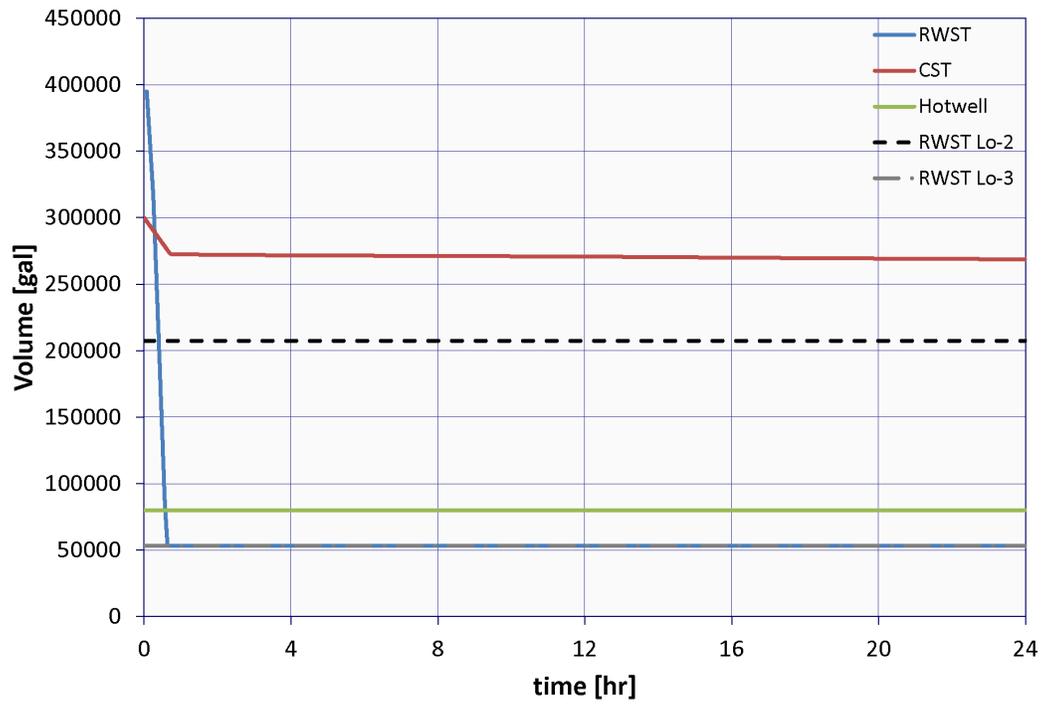




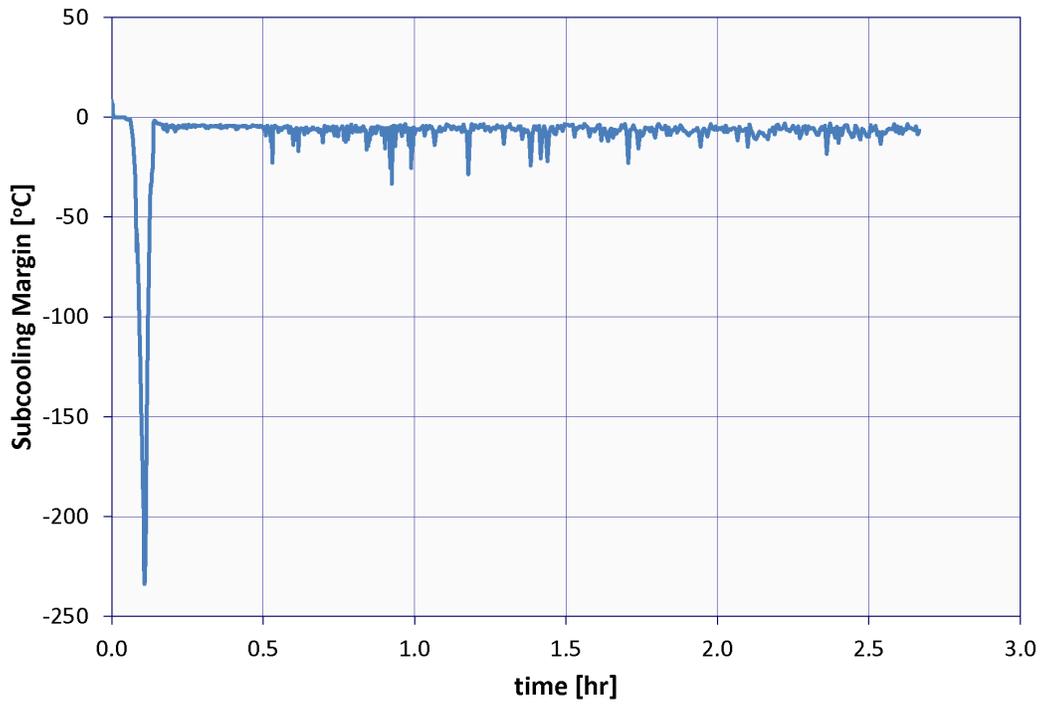
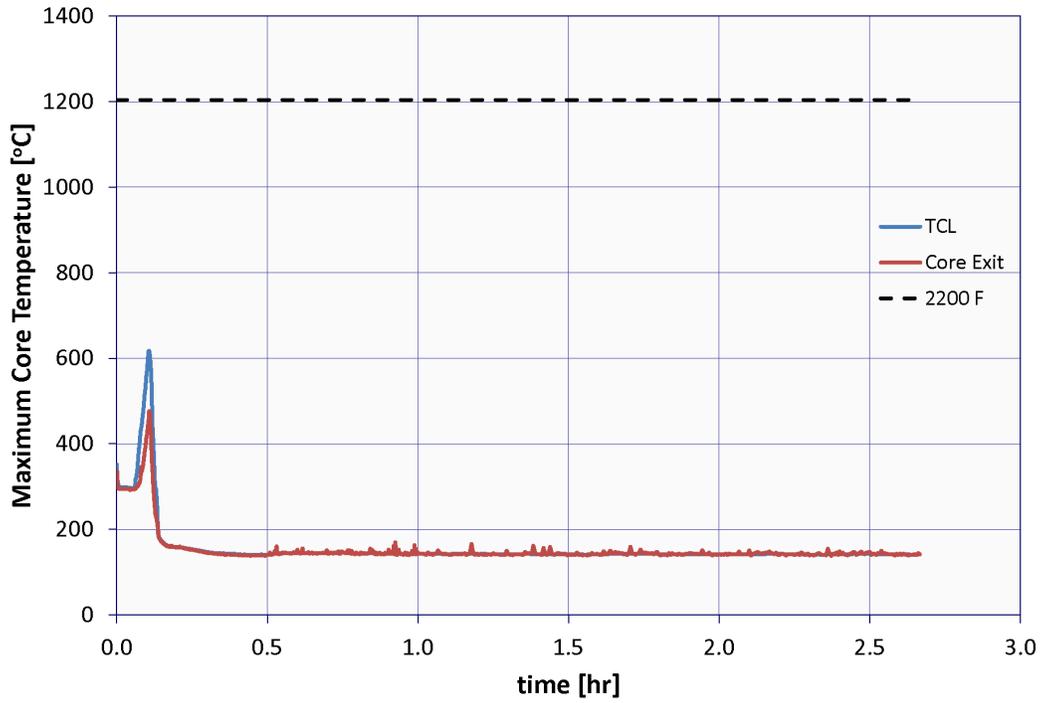


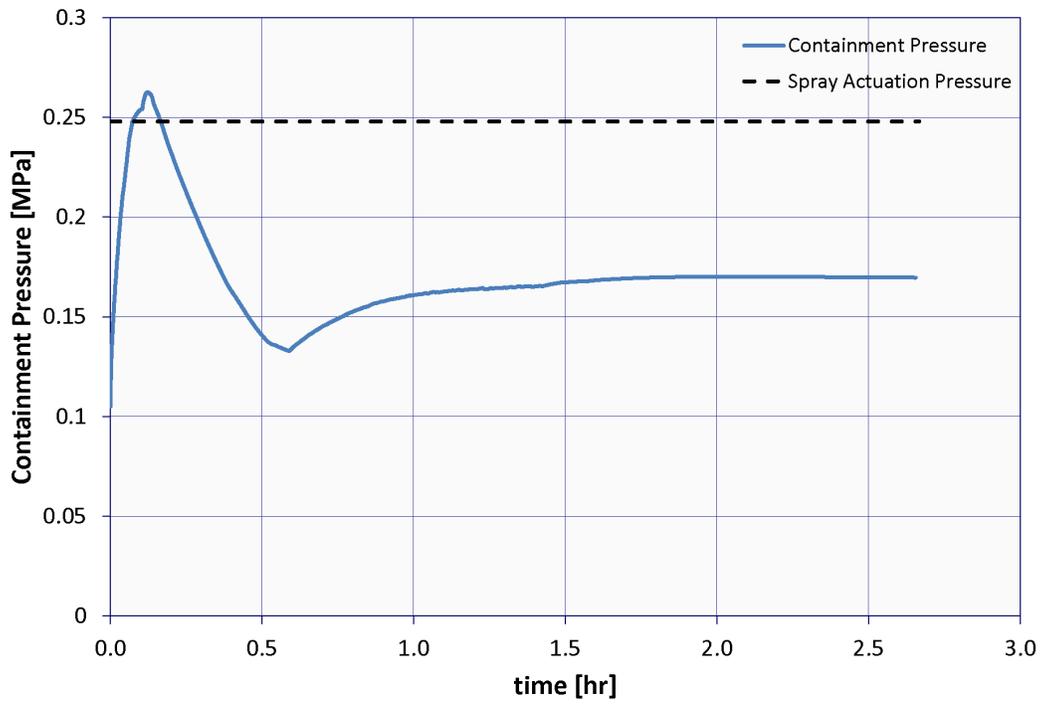
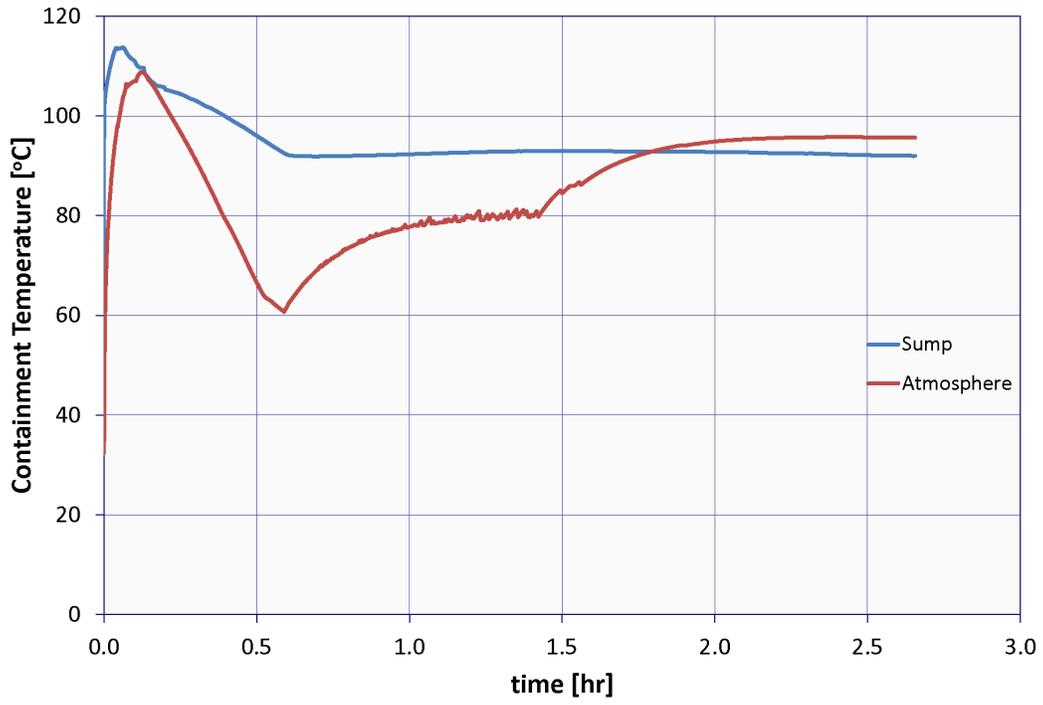


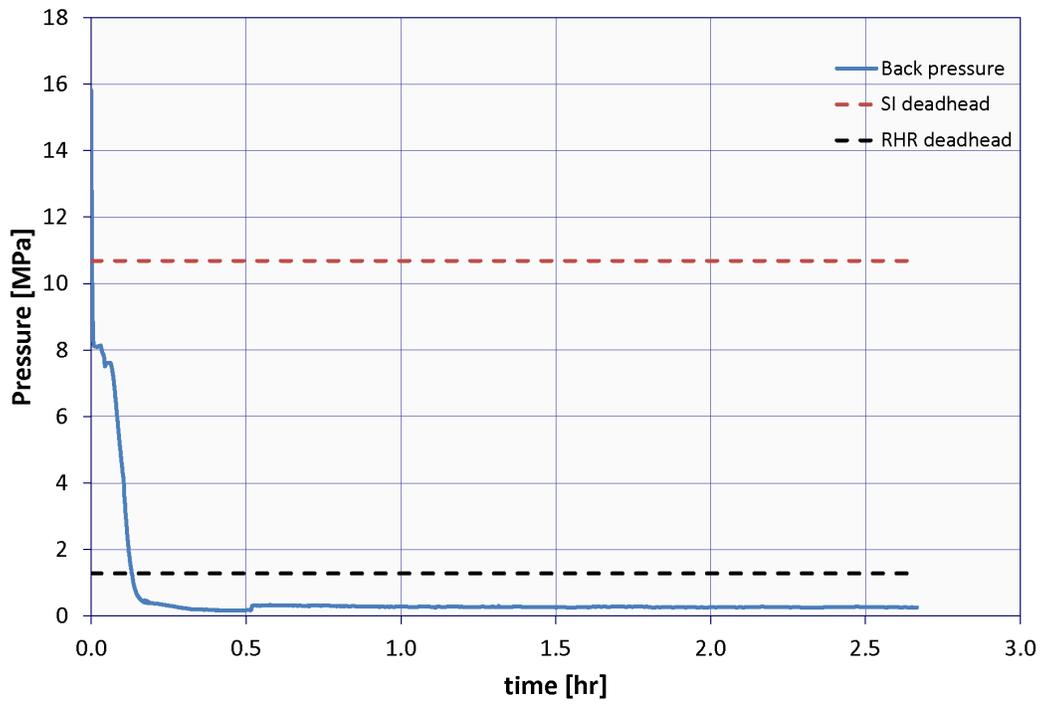
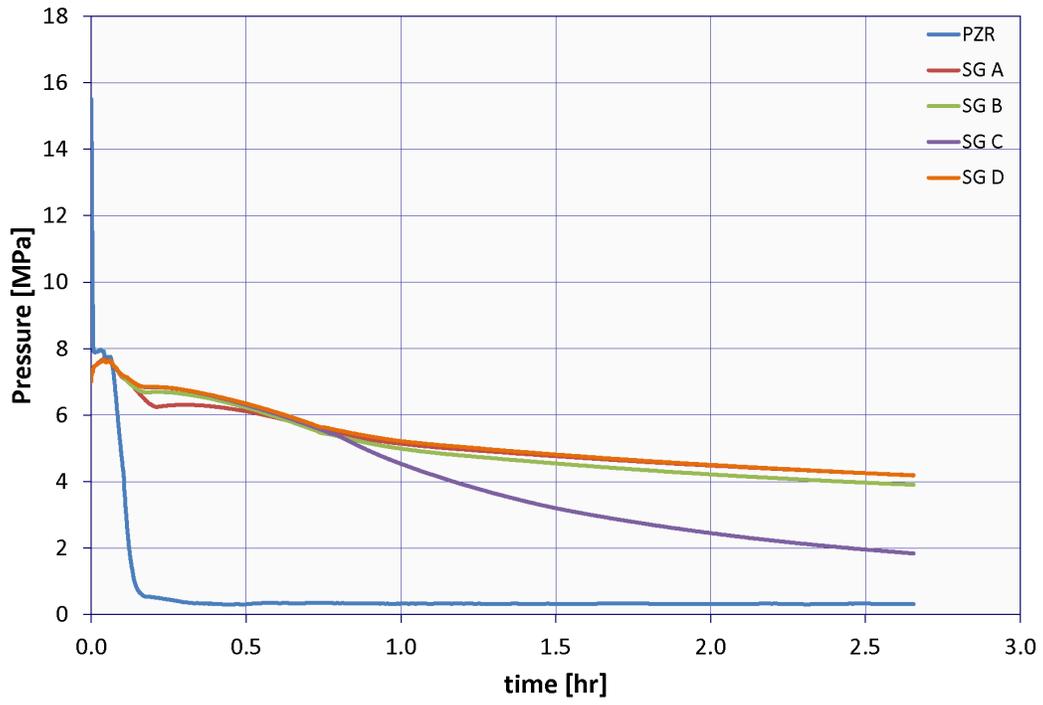


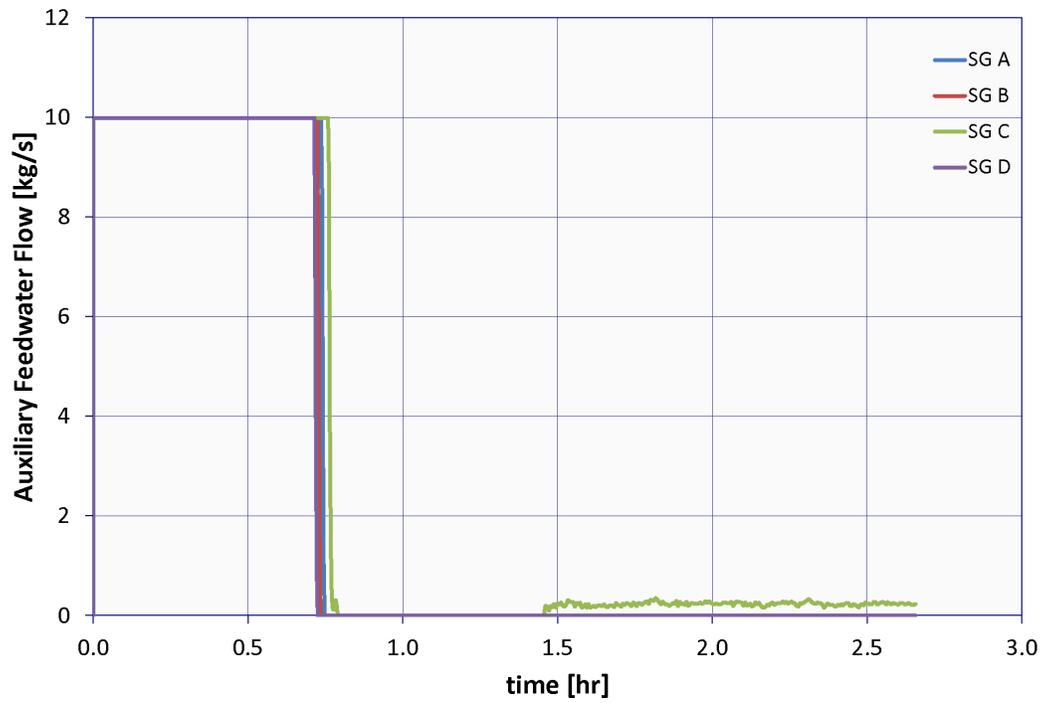
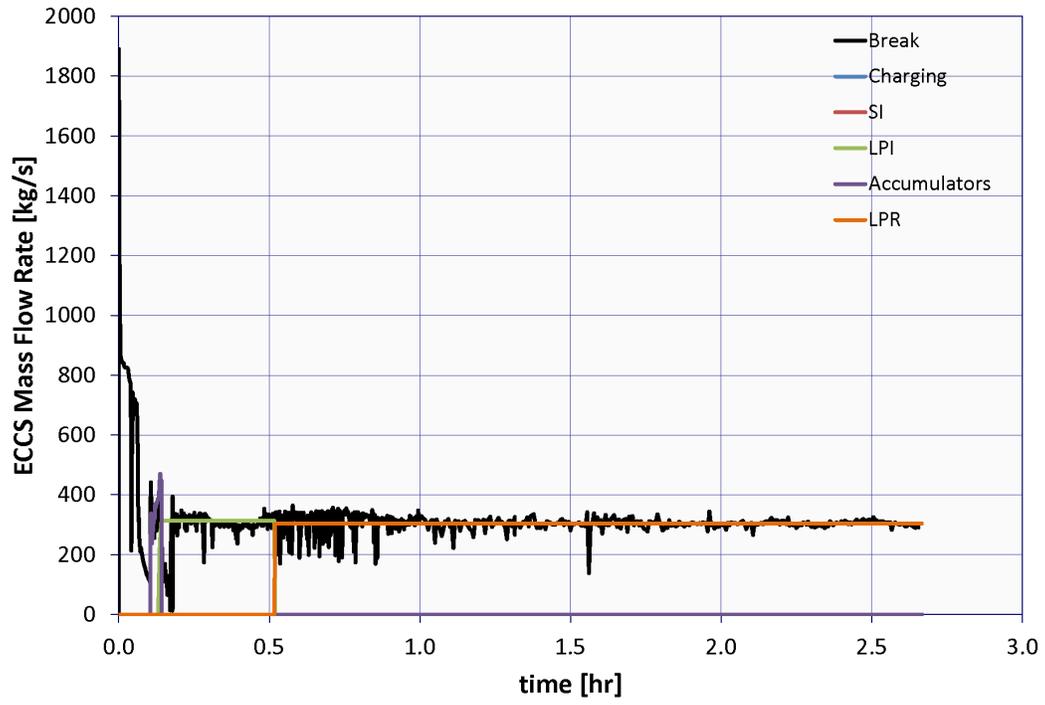


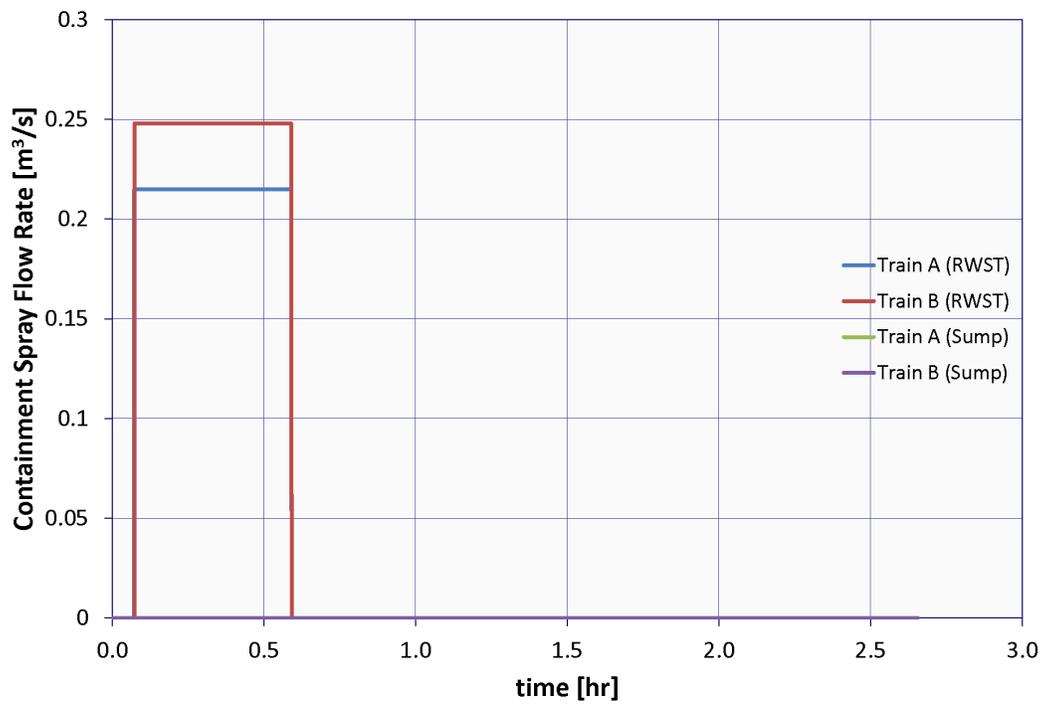
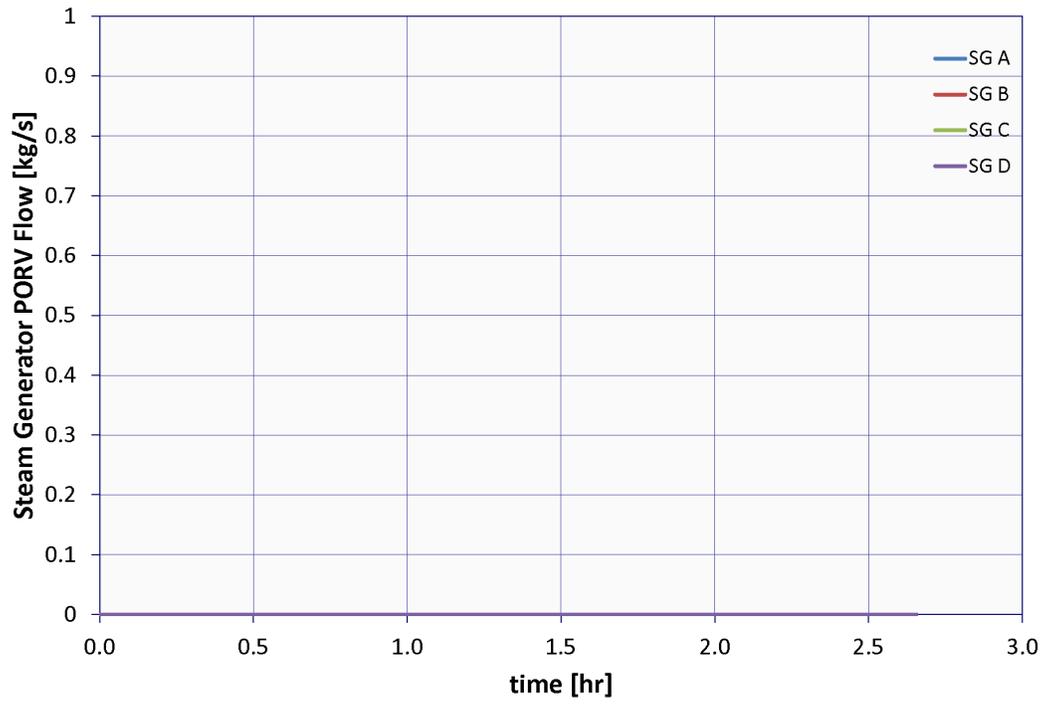
E.1.8.1 Case 8a: 6-in. Break with 2 Accumulators (1 in Broken Loop), 1 RHR, 2/2 CS Pumps

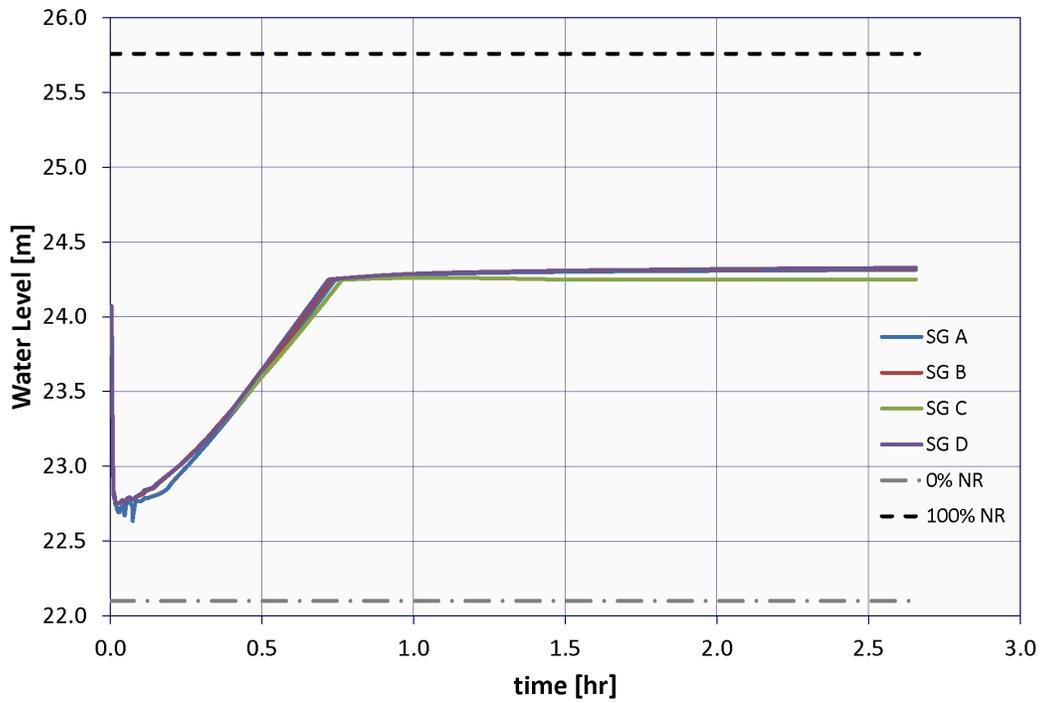
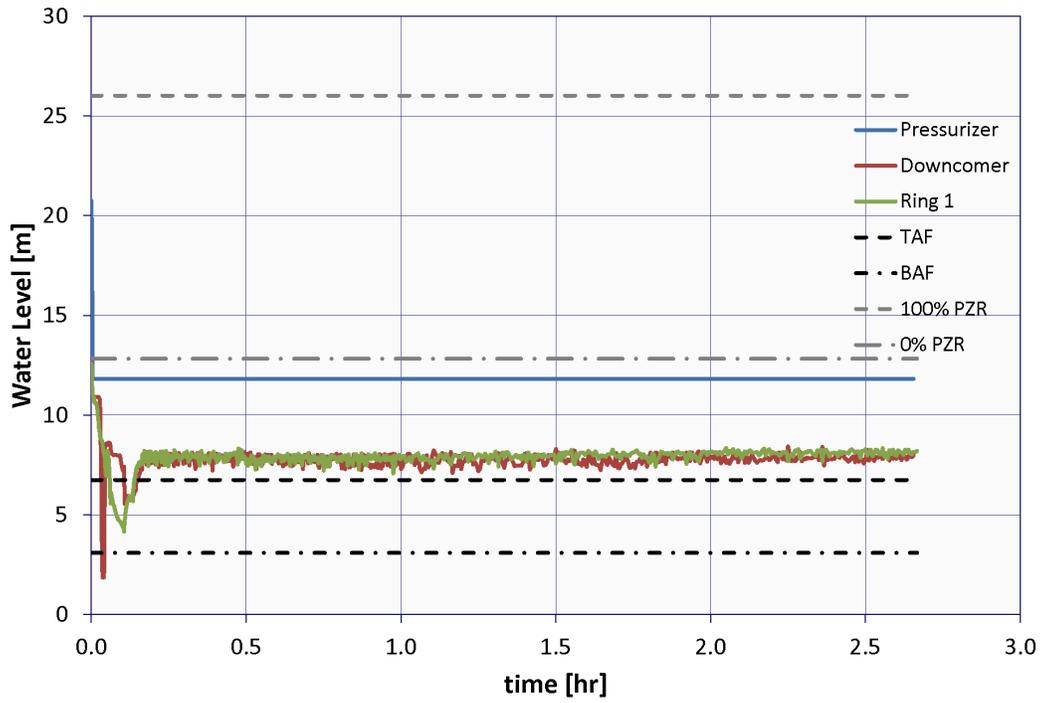


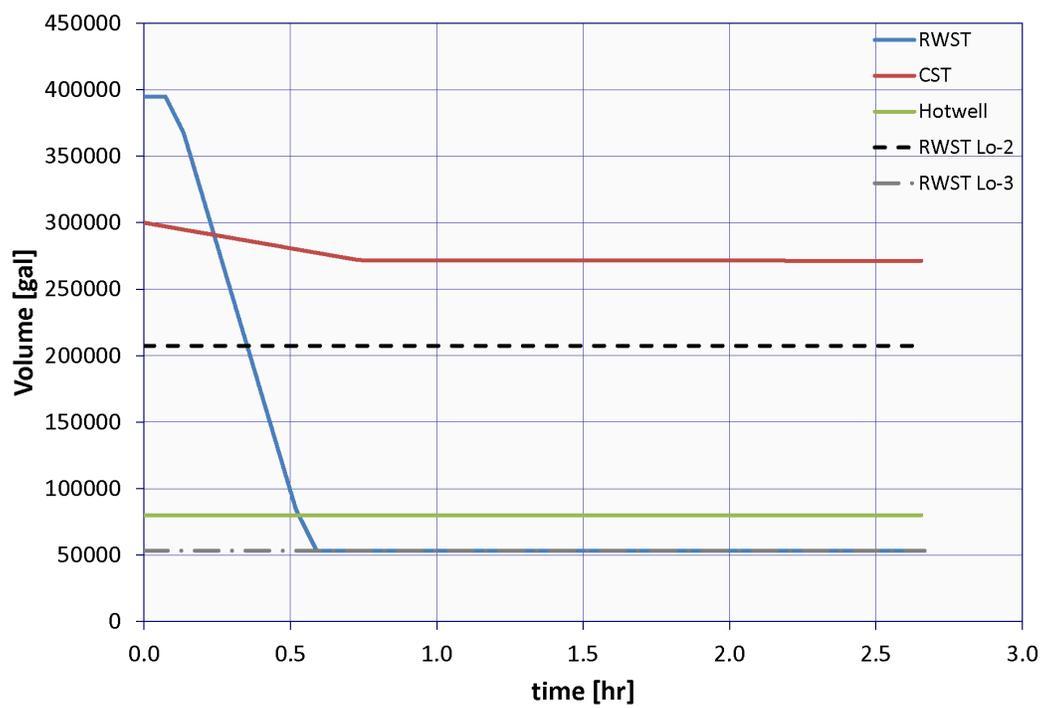






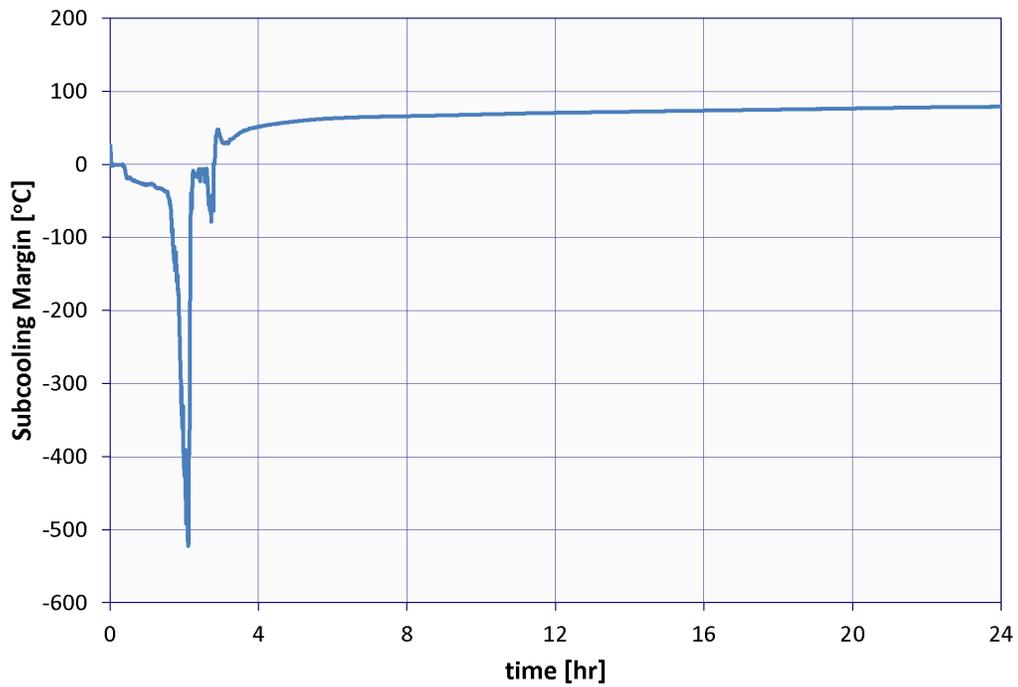
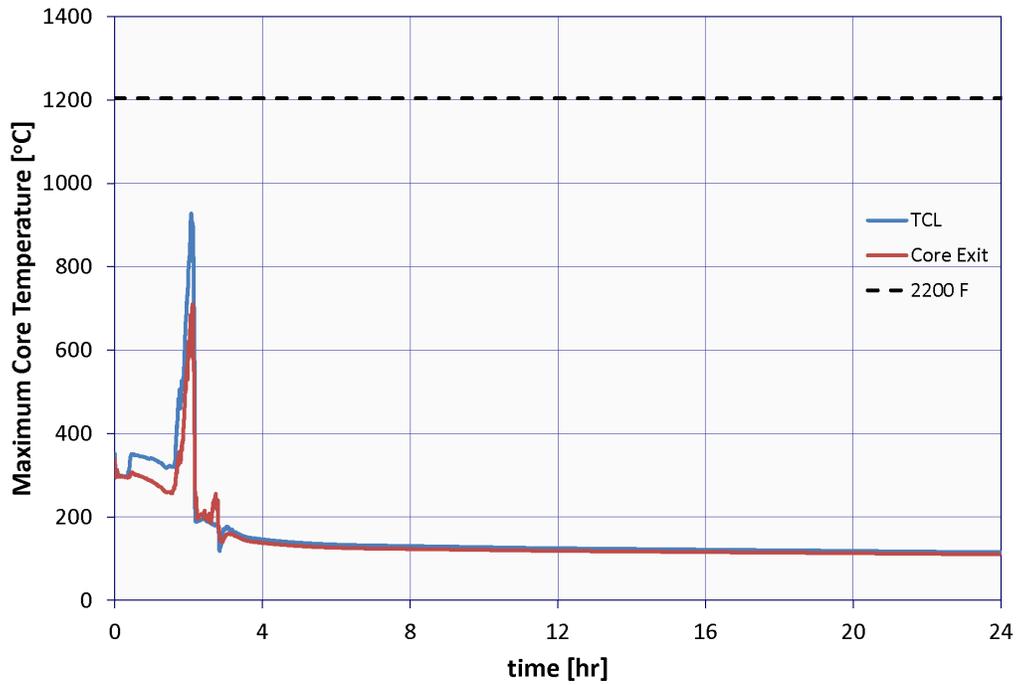


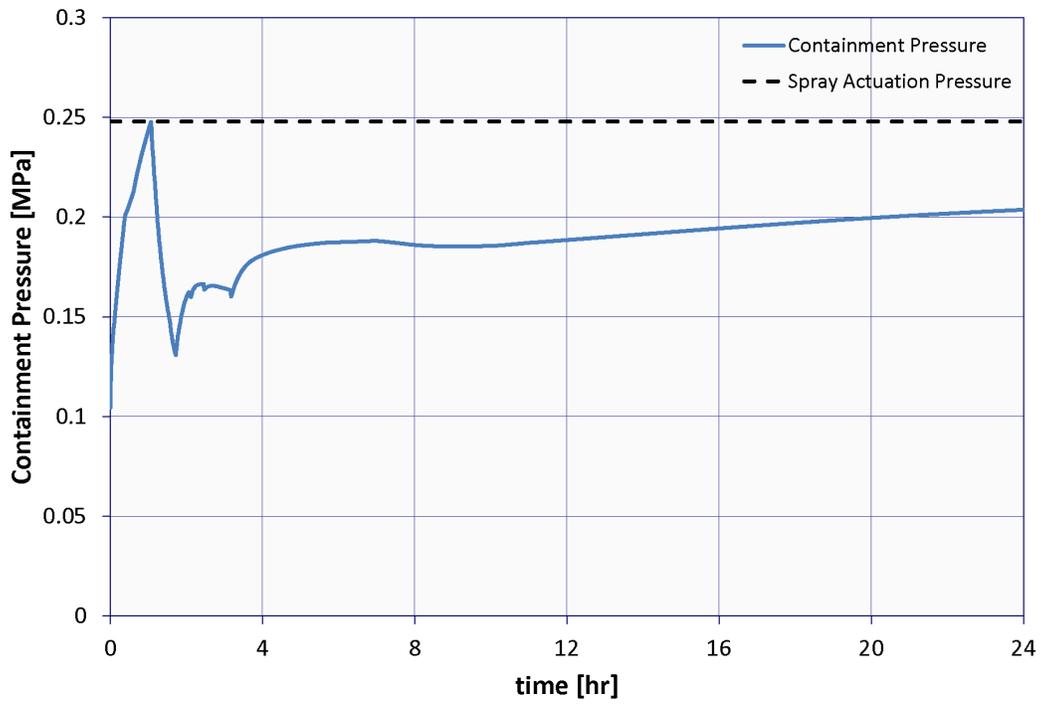
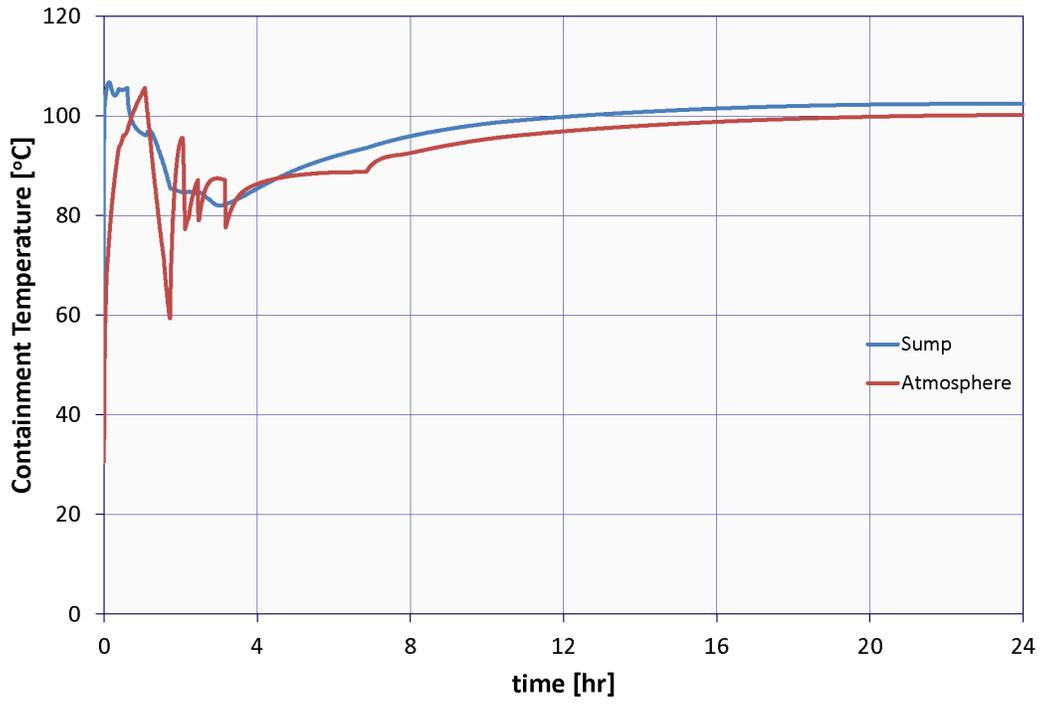


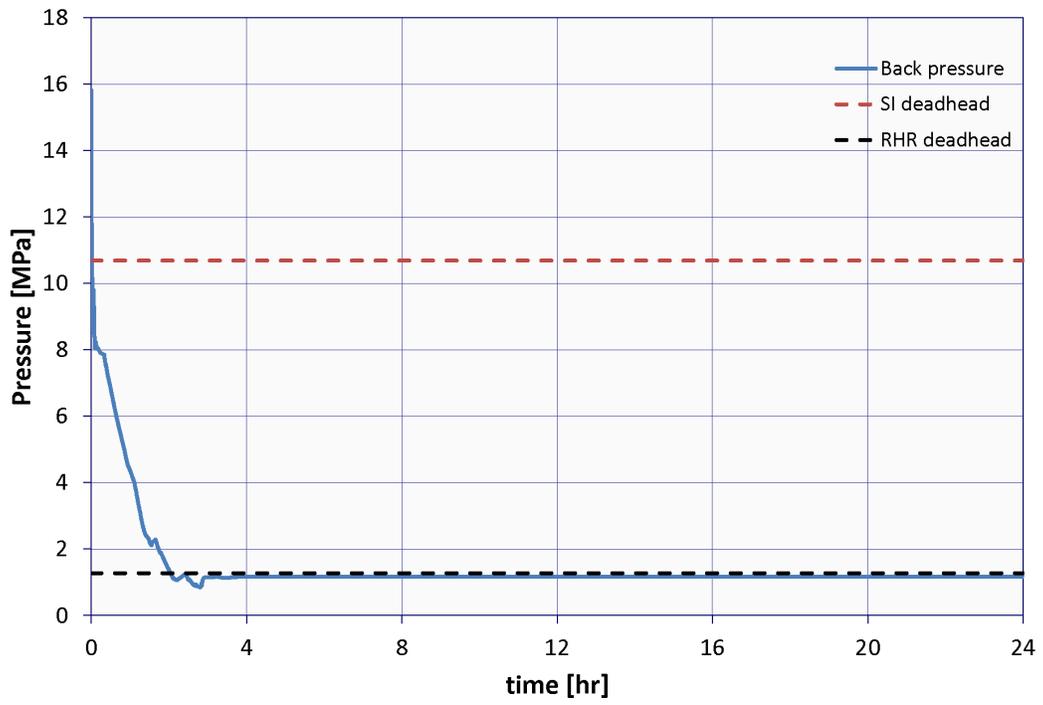
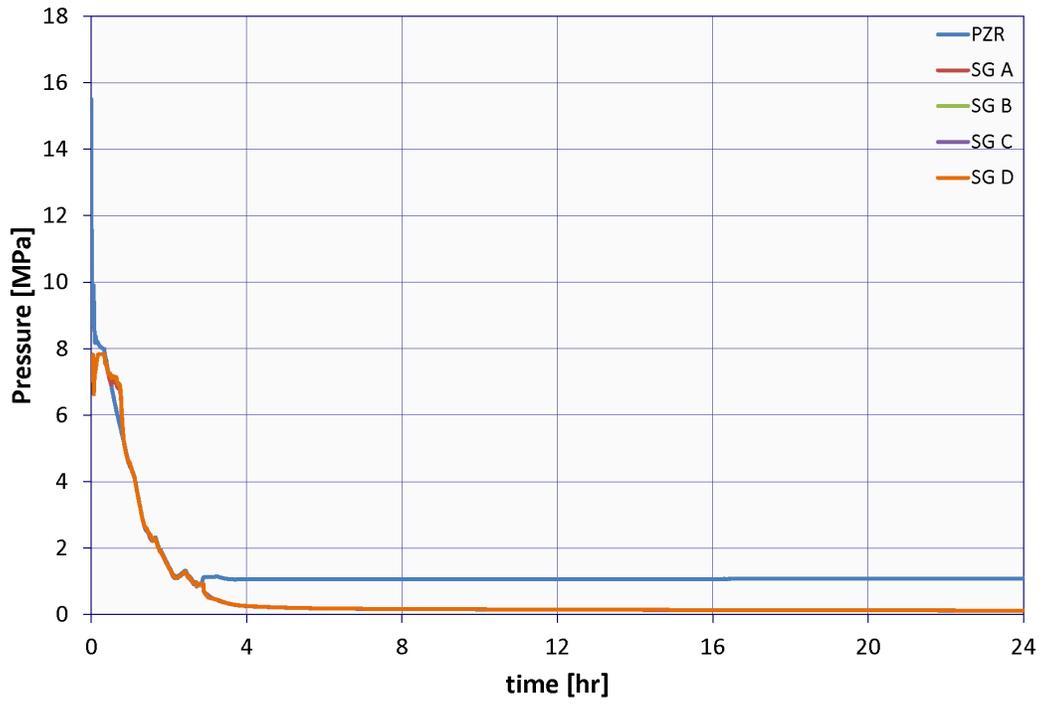


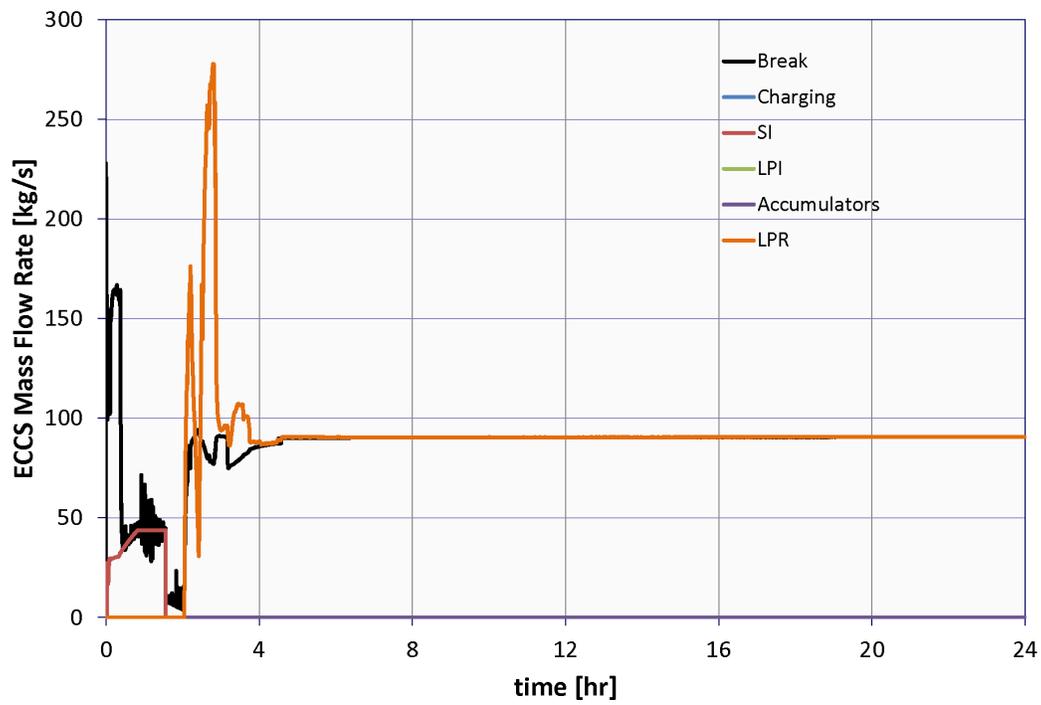
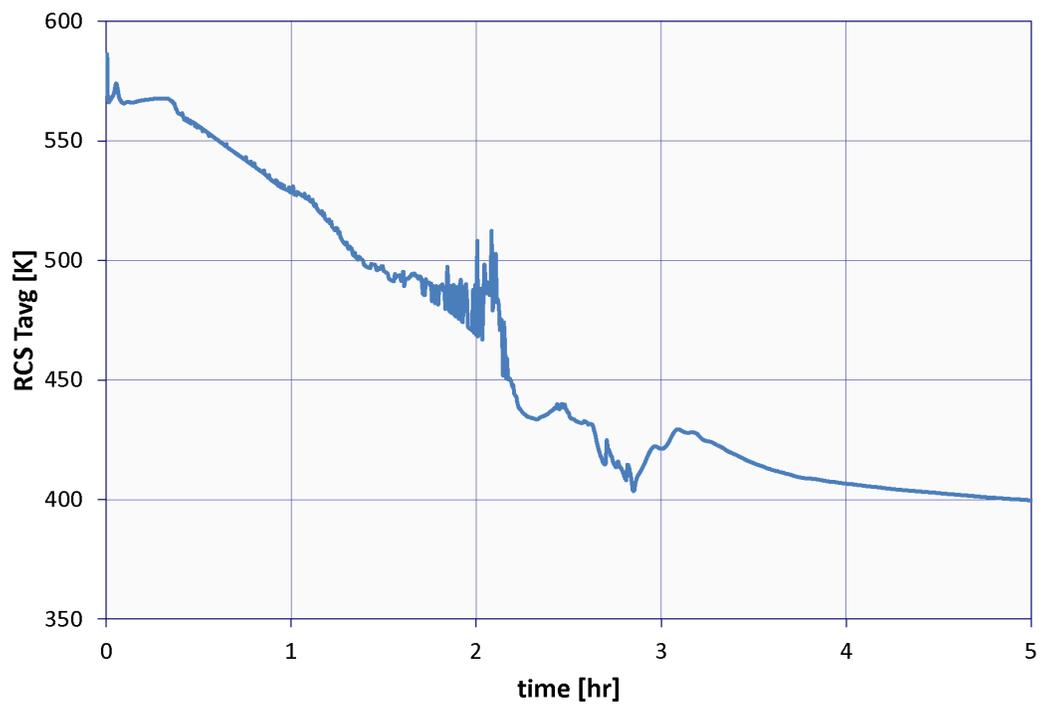
## E.2 Medium-Break Loss-of-Coolant Accident Cooldown Timing for Low-Pressure Recirculation

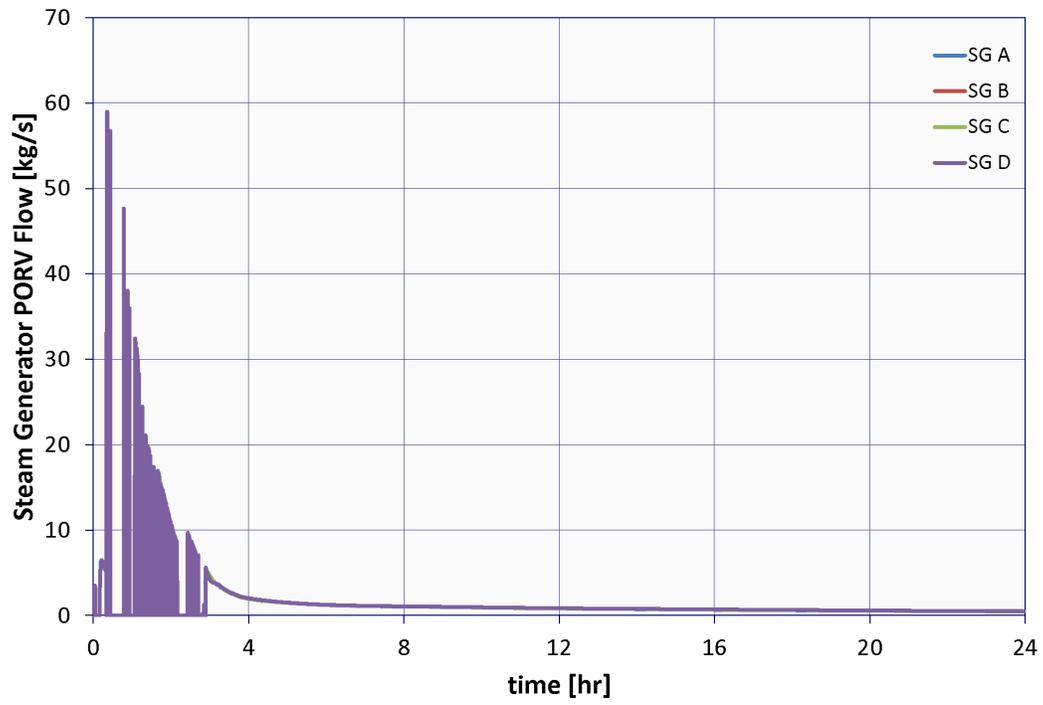
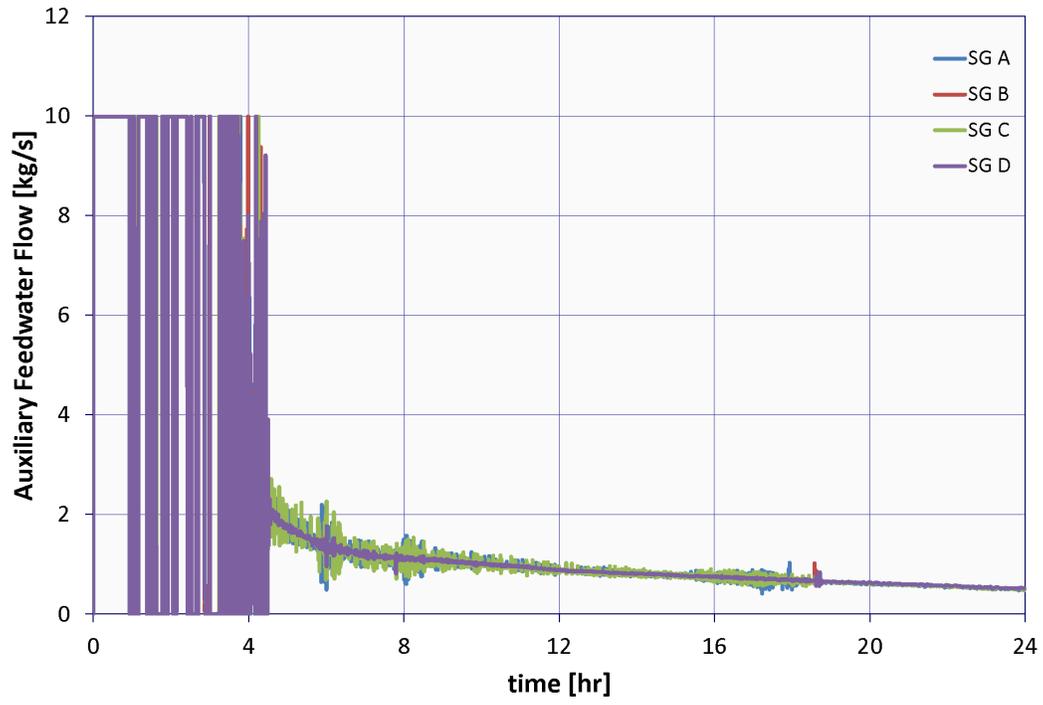
### E.2.1 Case 1: 2-in. Break, 100 °F/hr Cooldown at 20 min, 2/2 CS Pumps, 0/4 RCFC

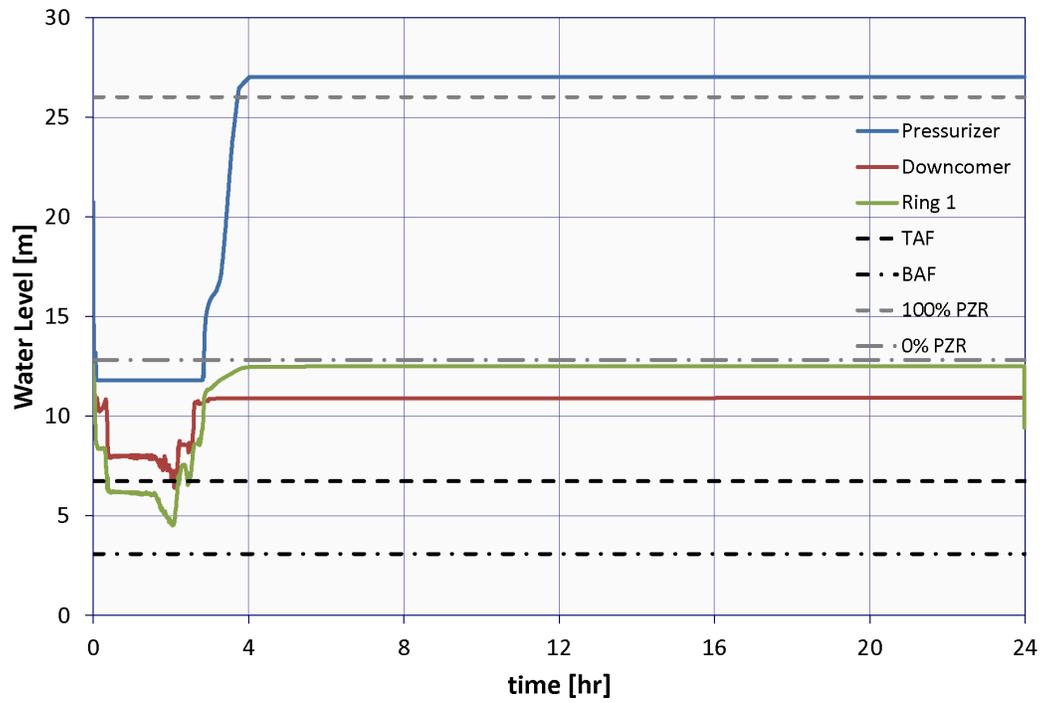
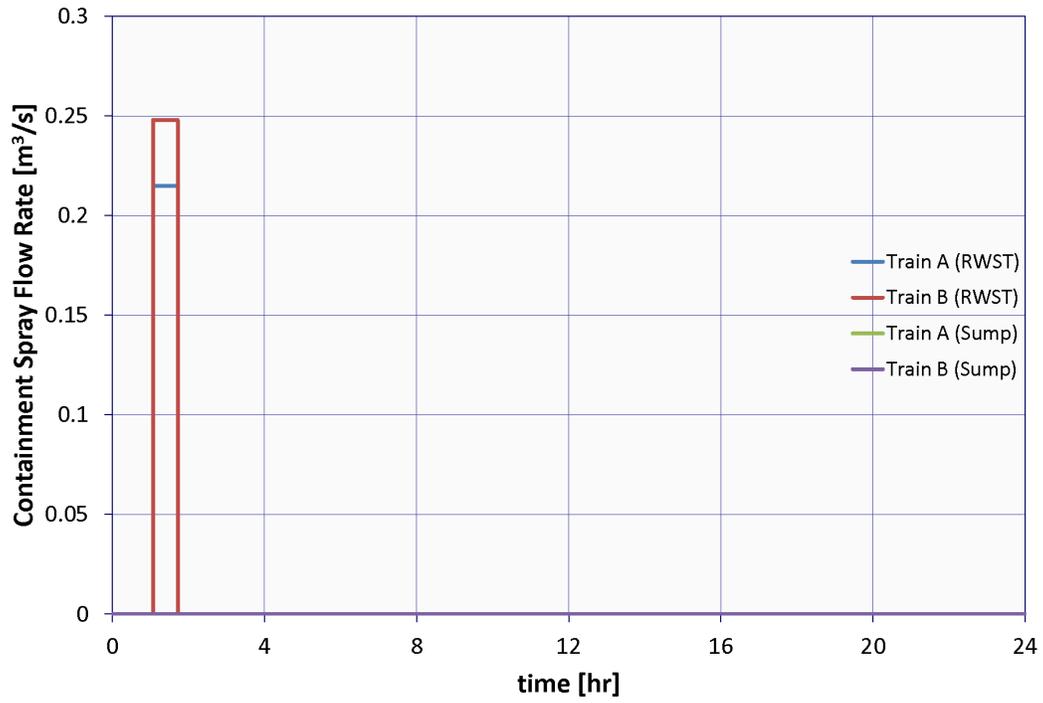


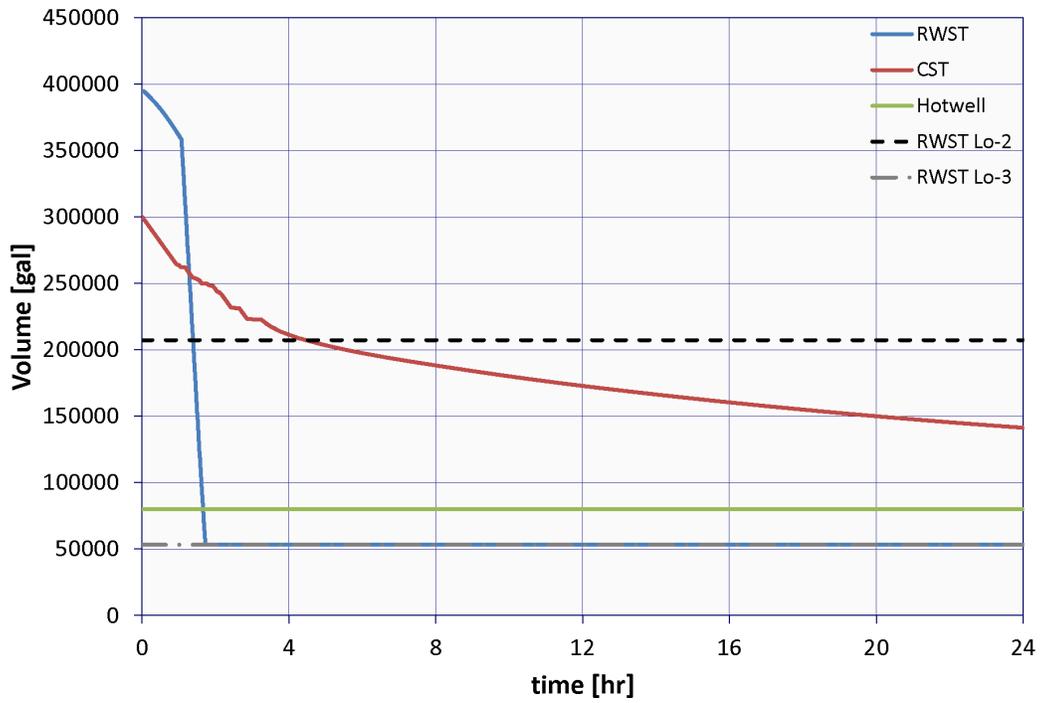
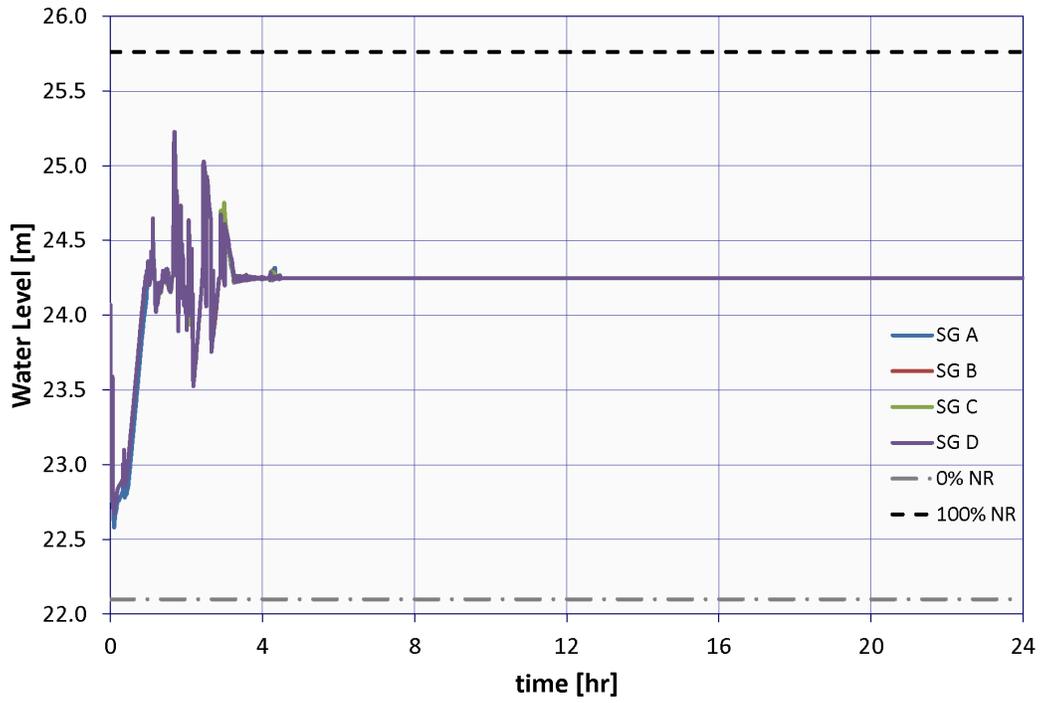




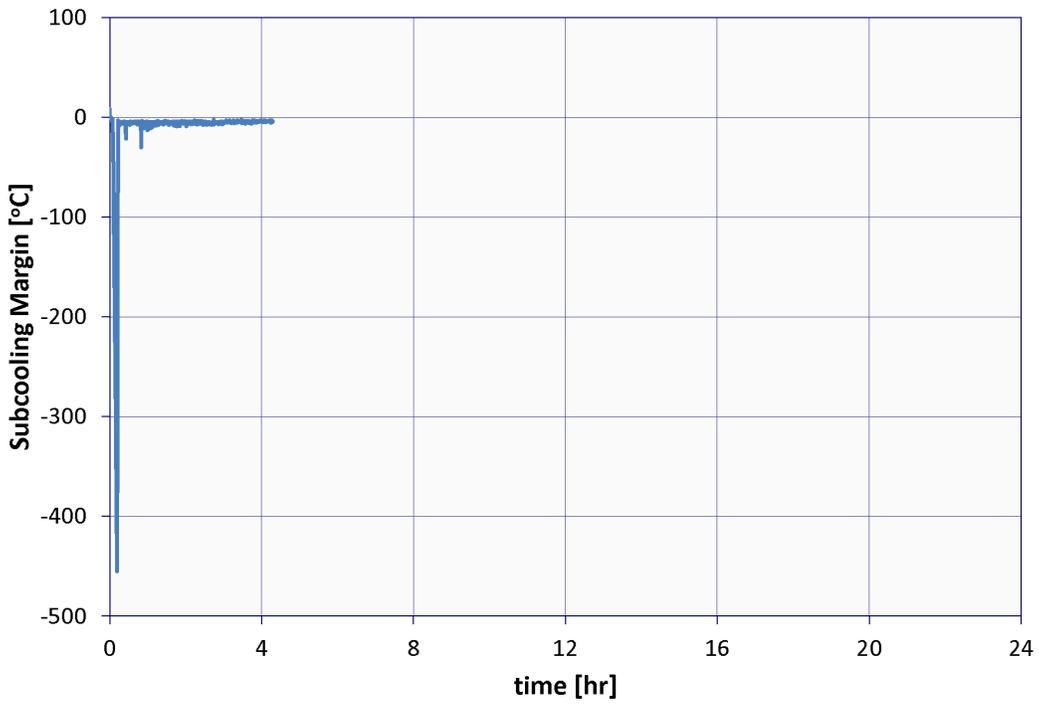
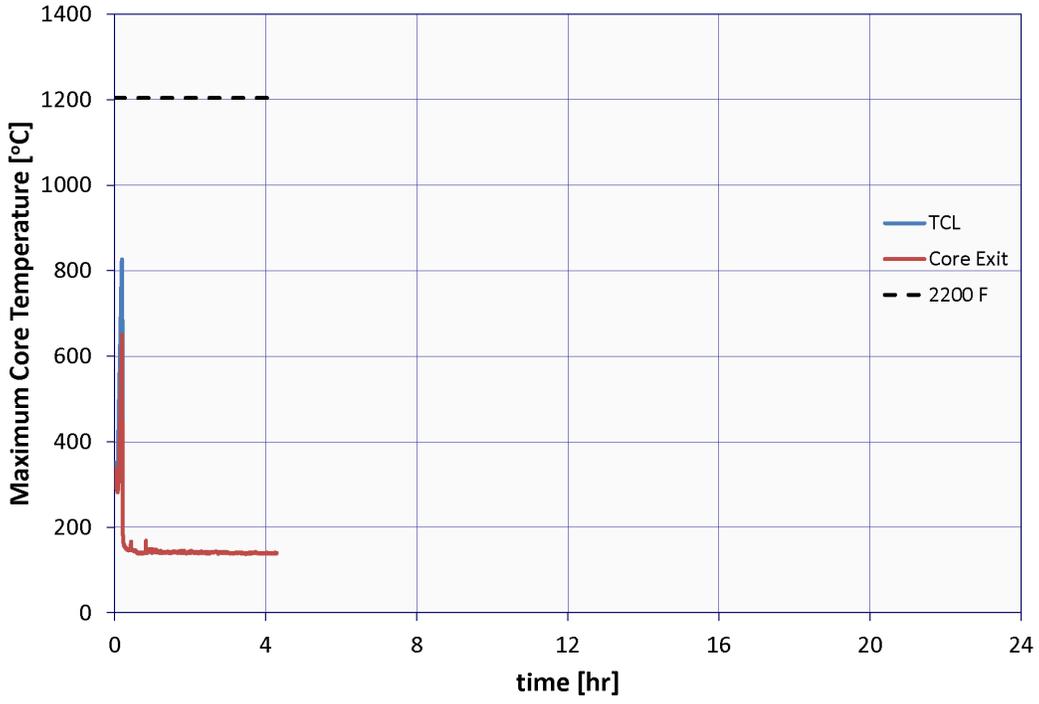


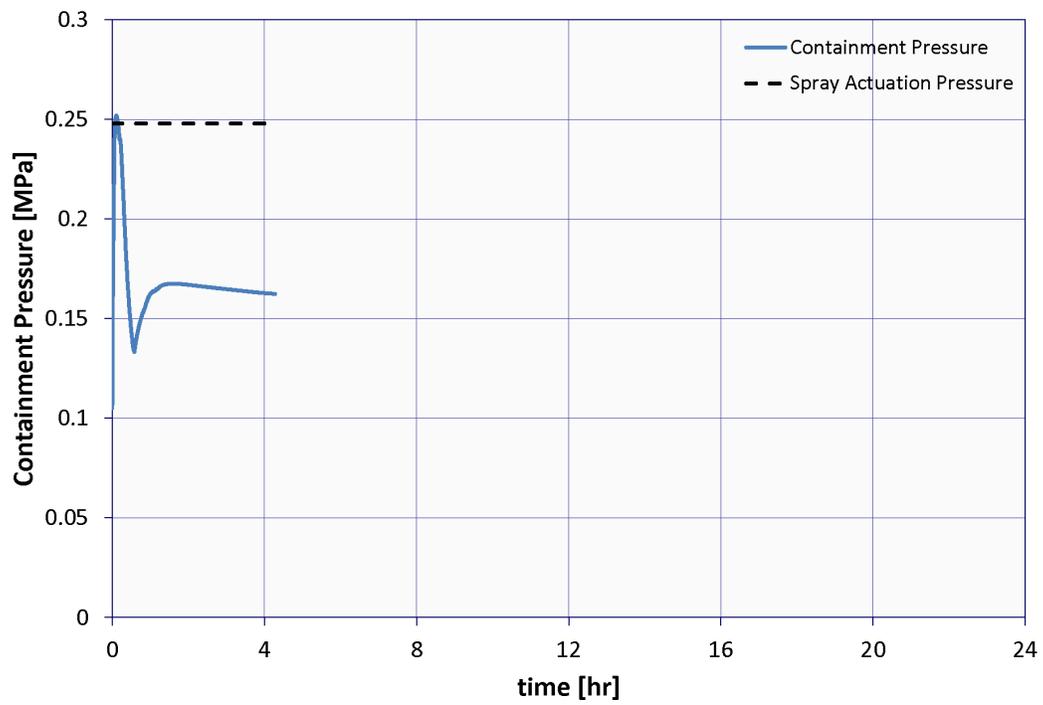
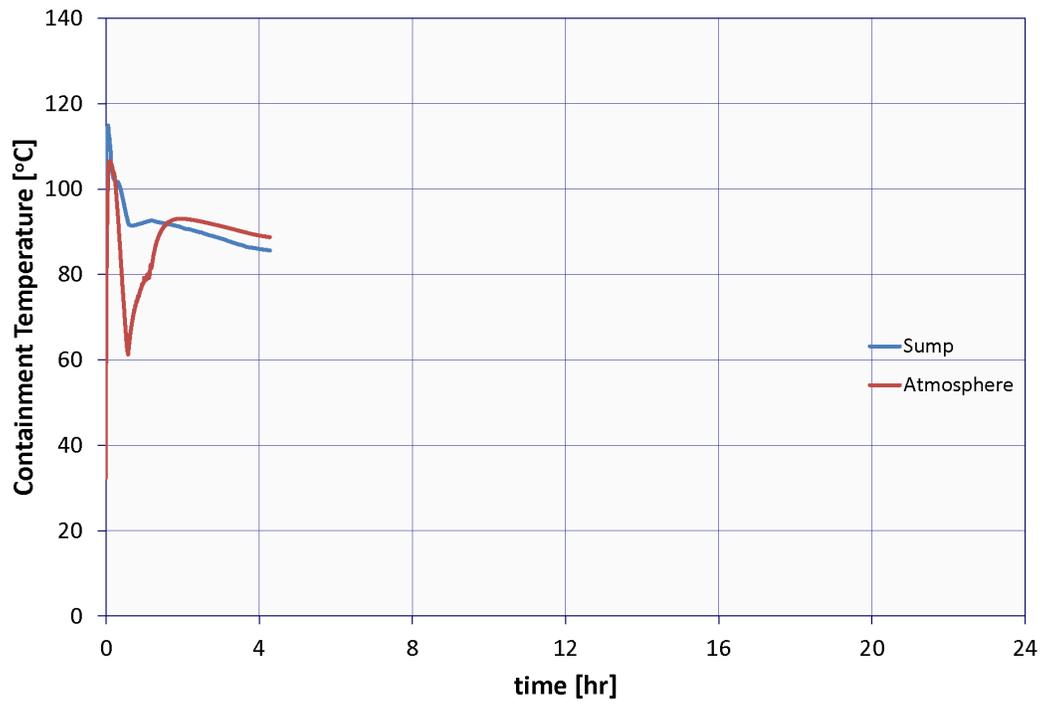


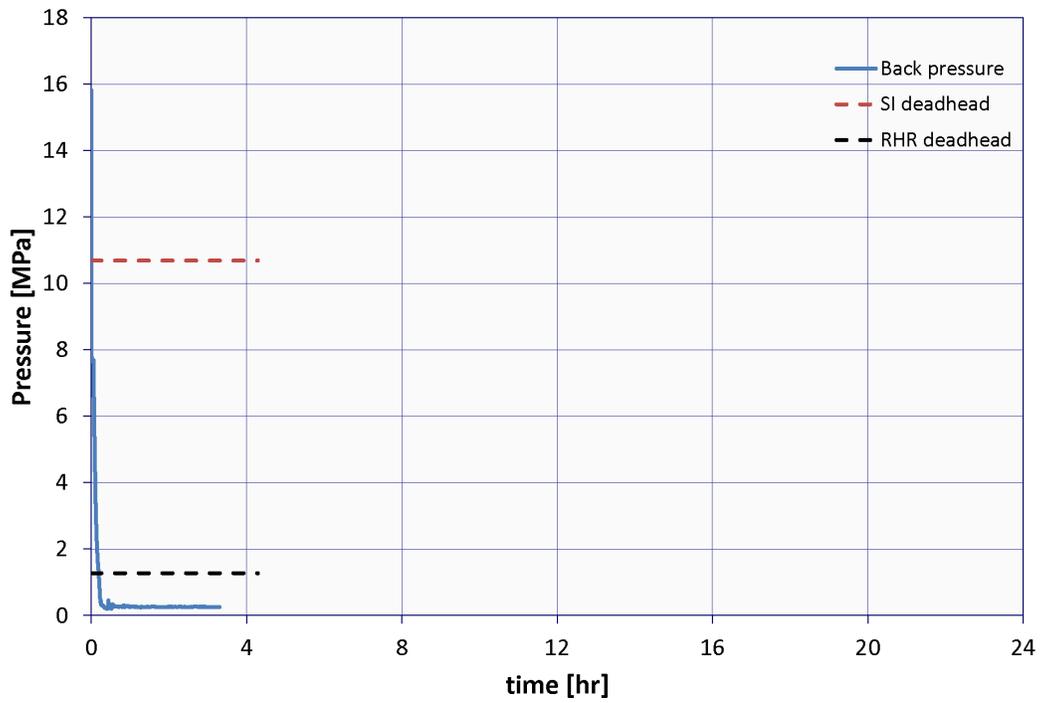
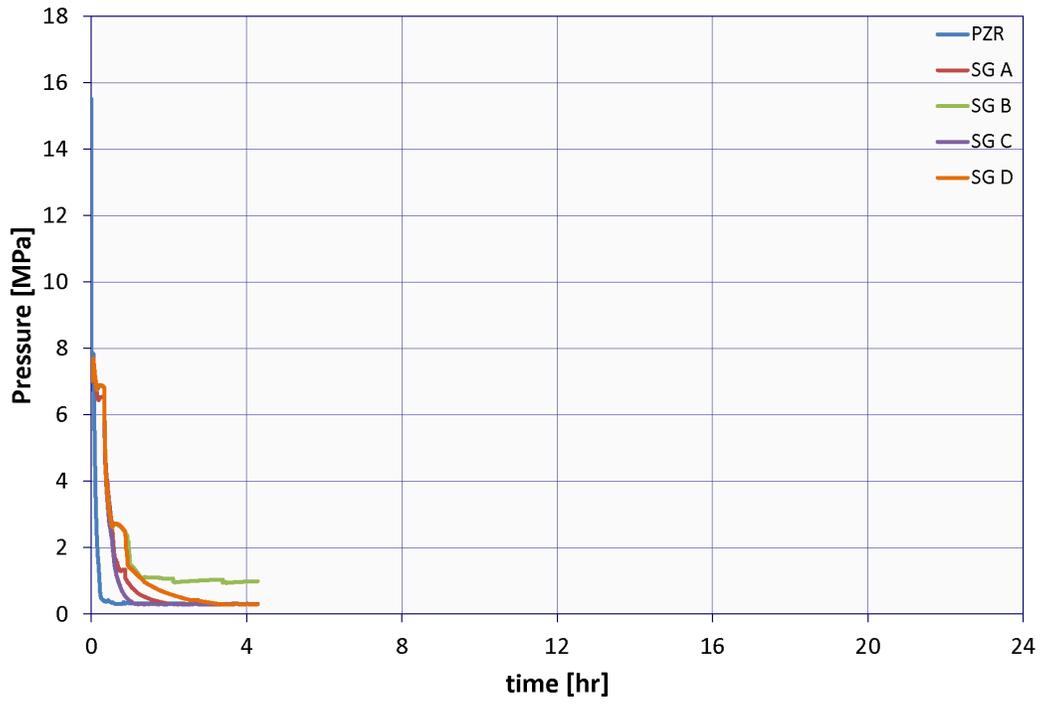


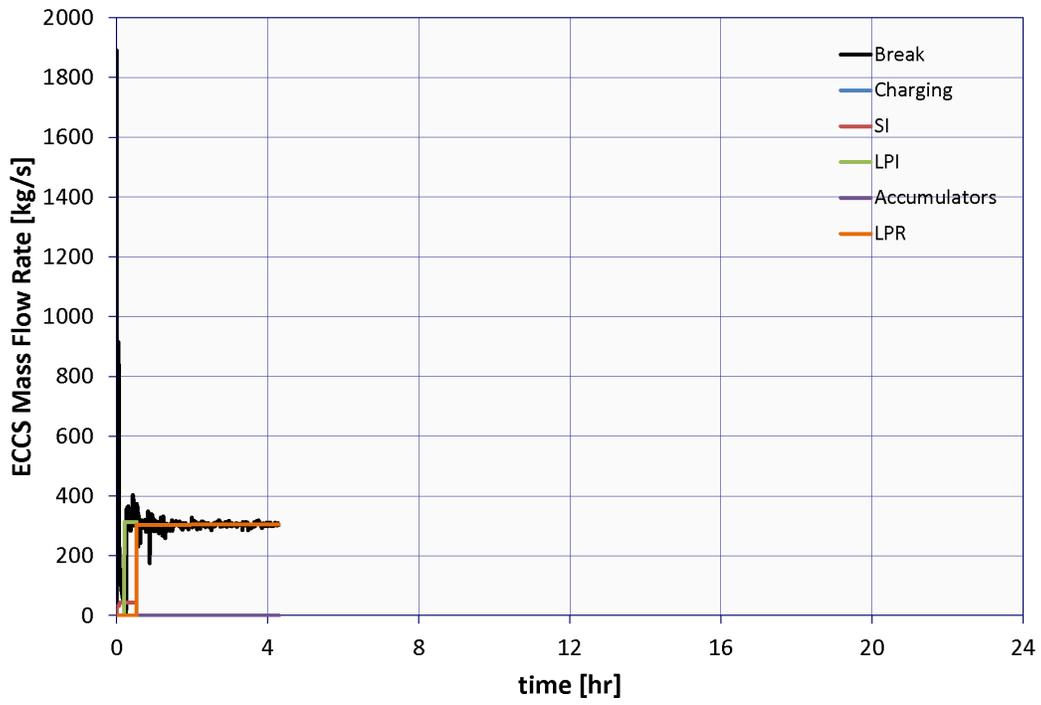
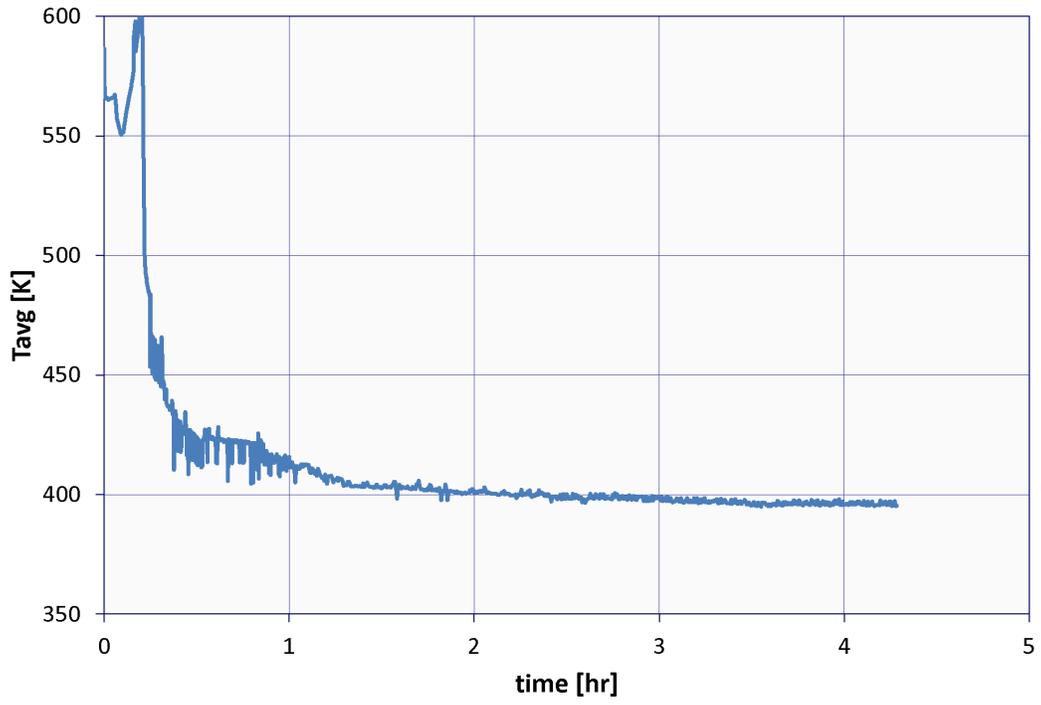


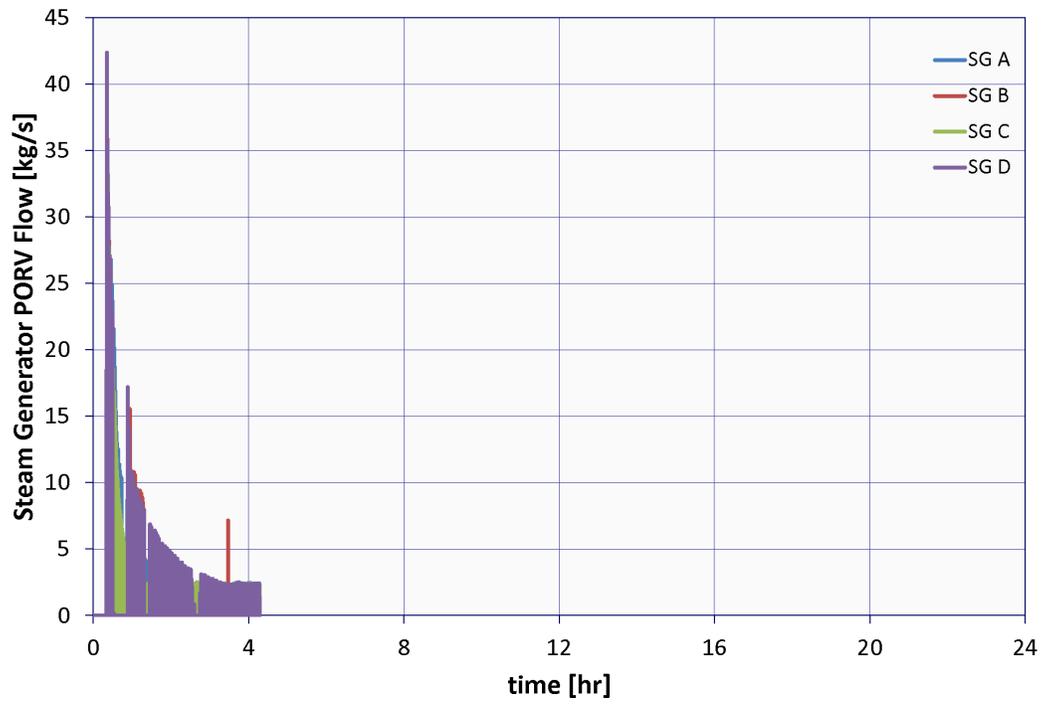
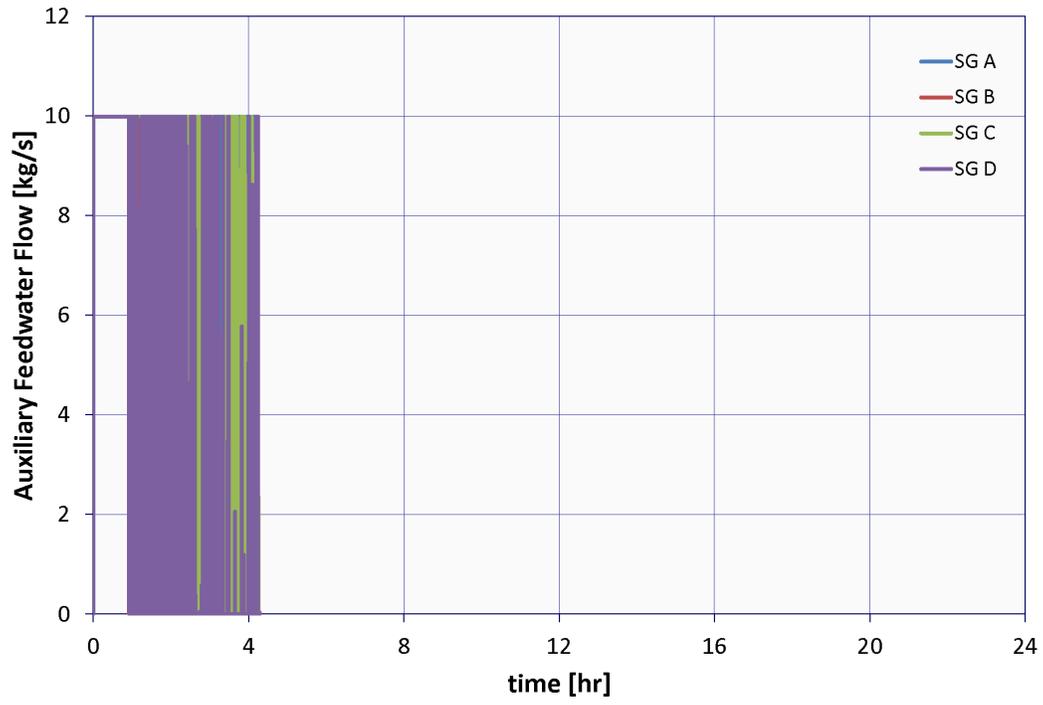
**E.2.2 Case 2: 6-in. Break, 100 °F/hr Cooldown at 20 min, 2/2 CS Pumps, 0/4 RCFC**

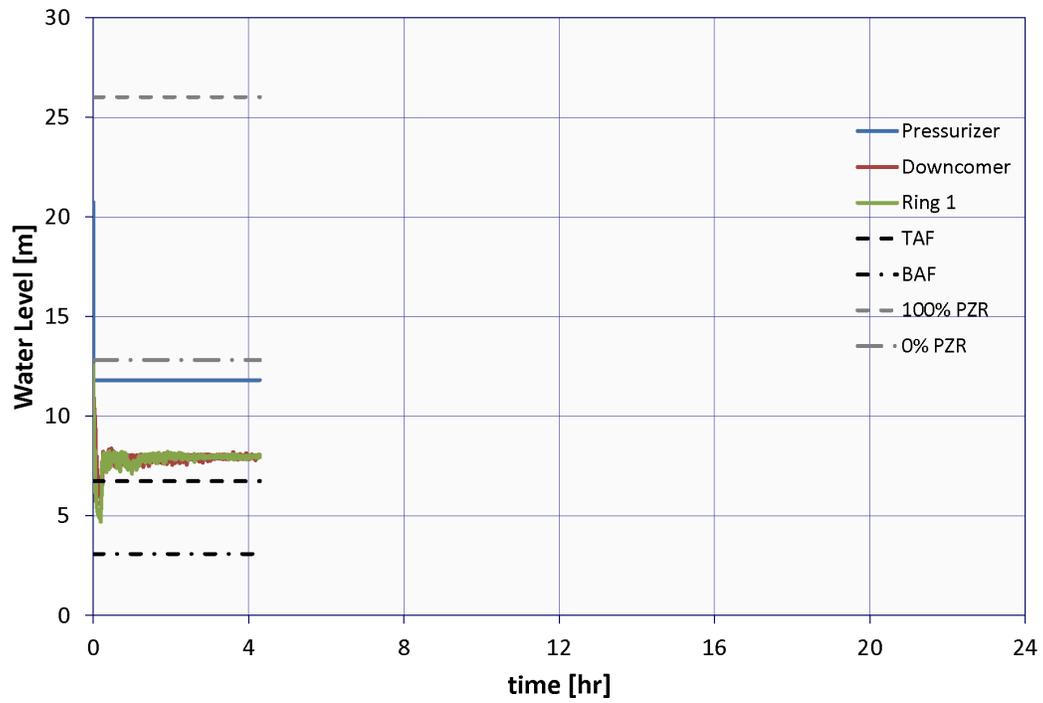
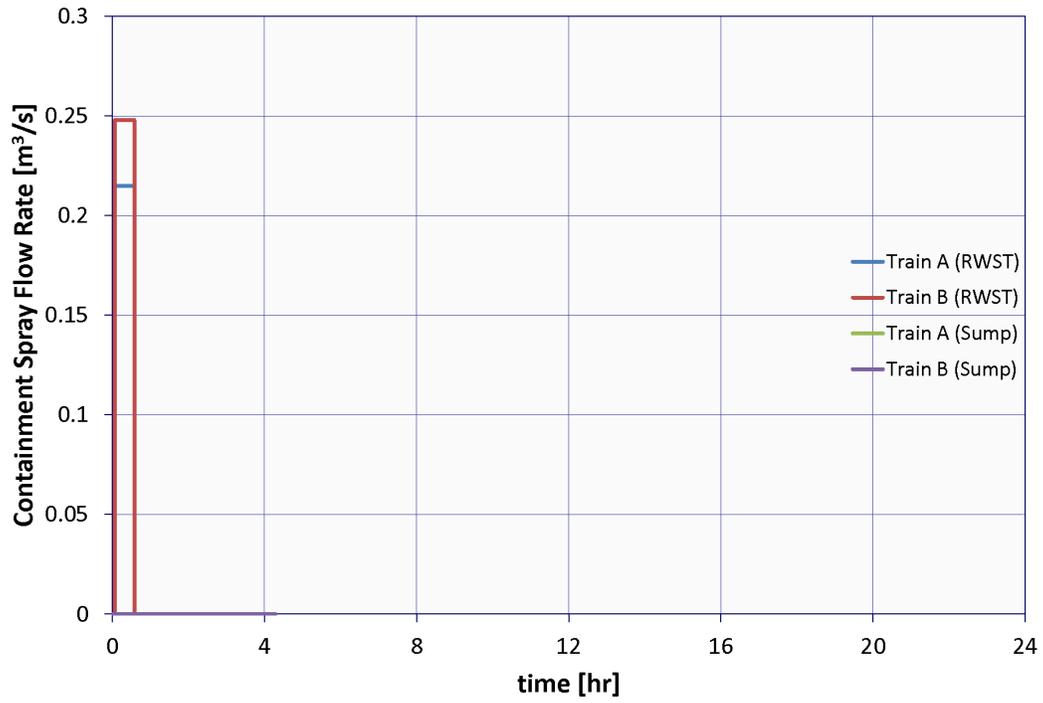


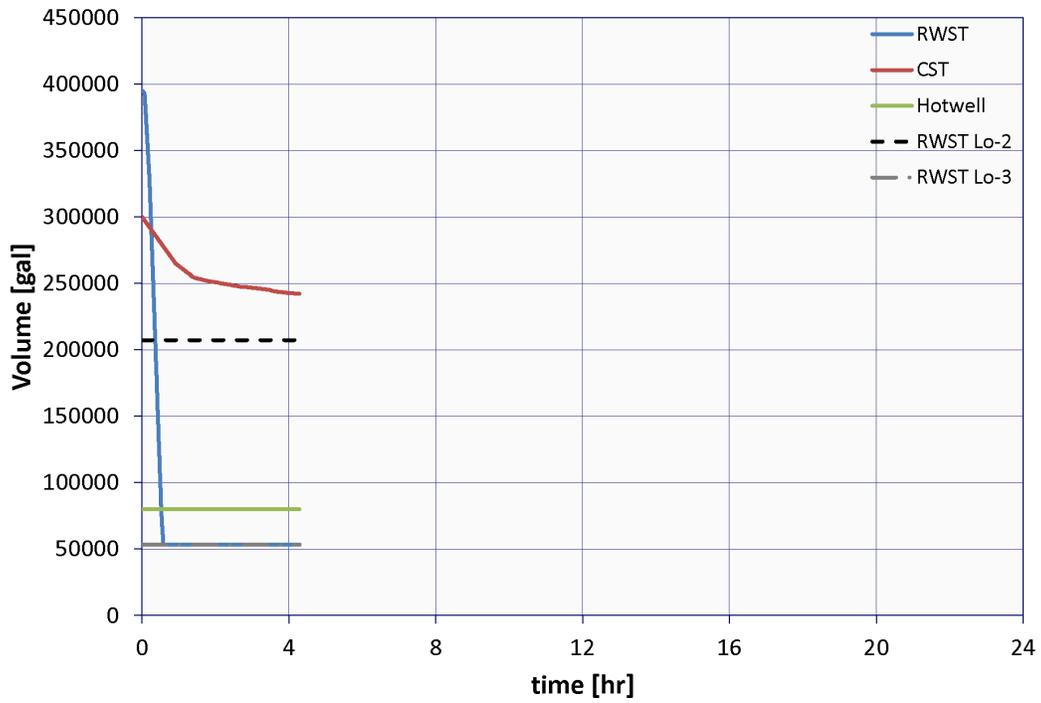
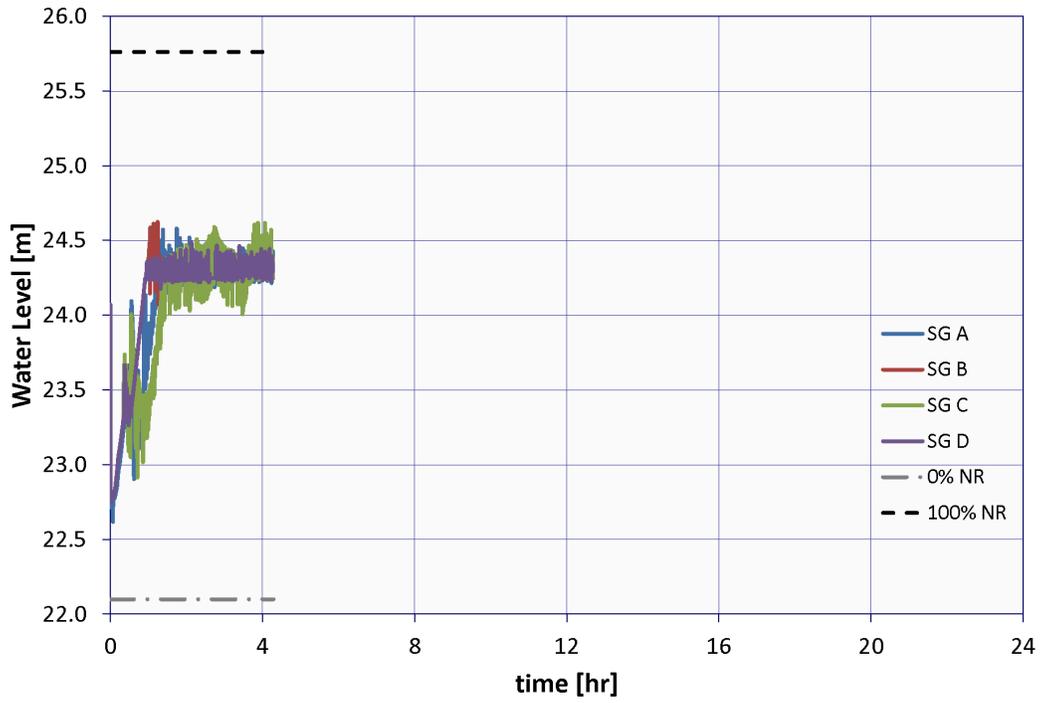




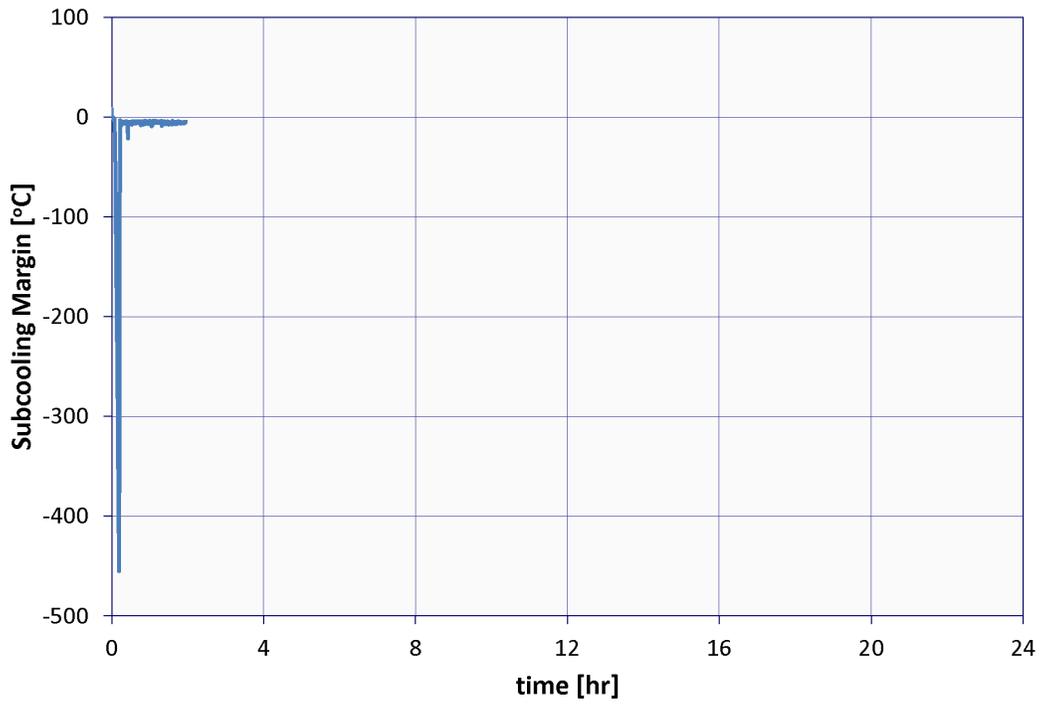
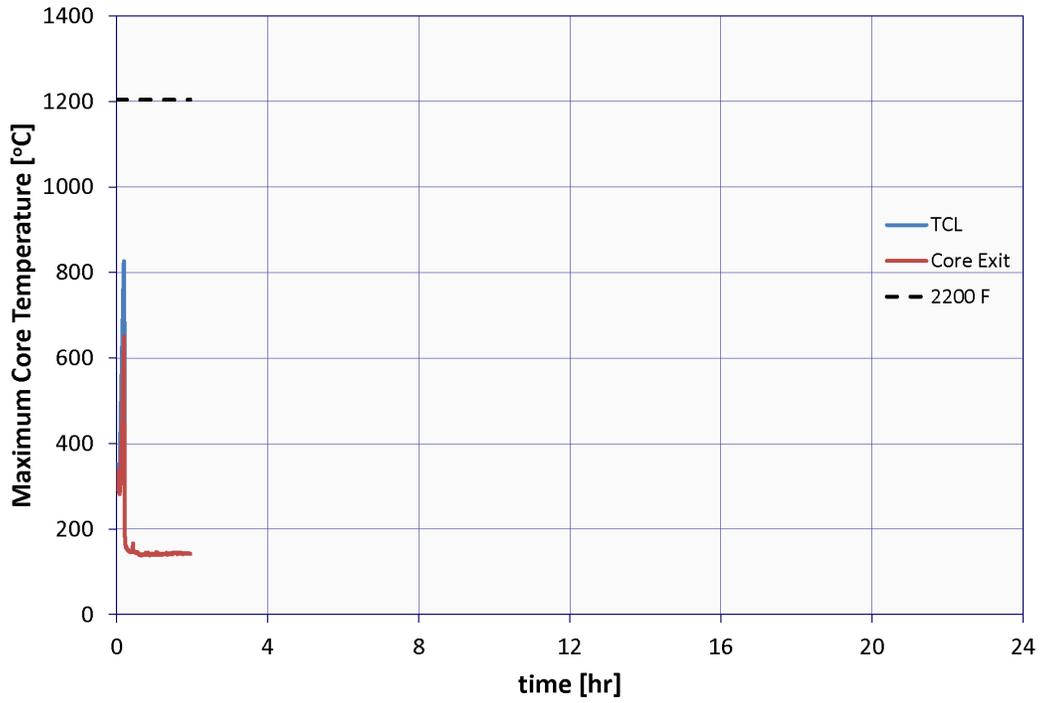


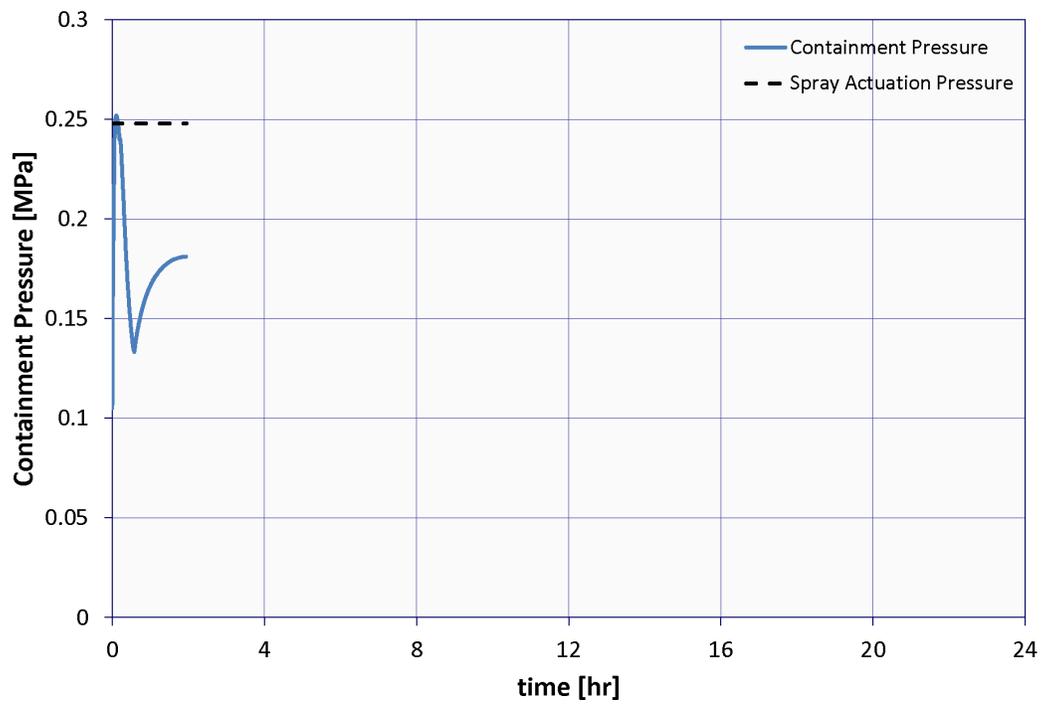
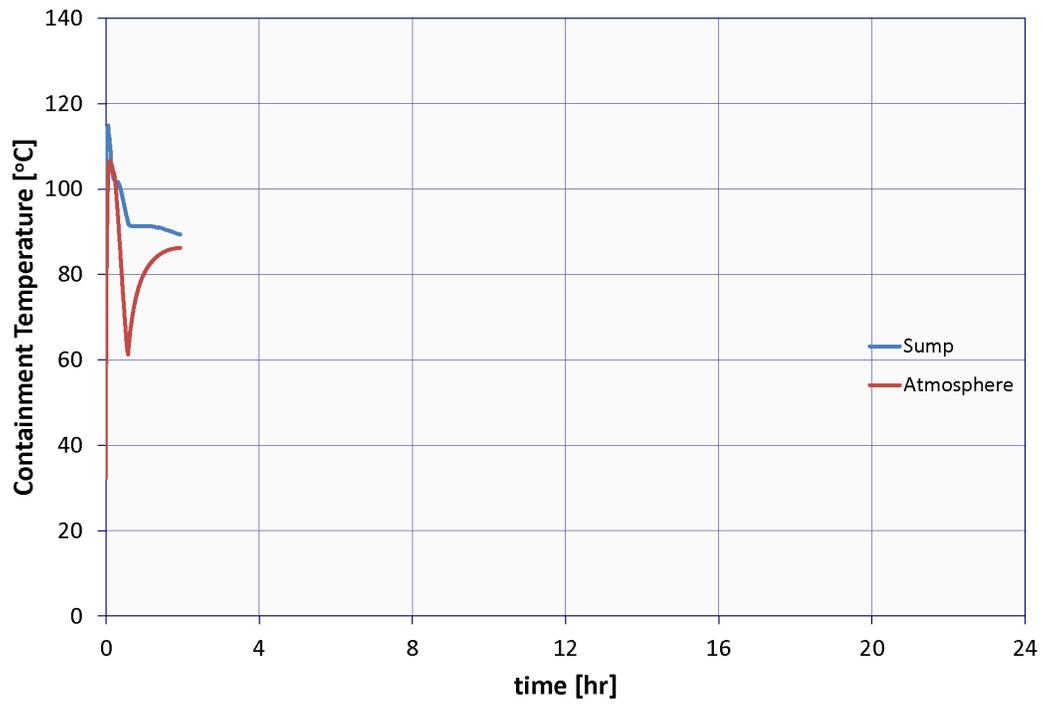


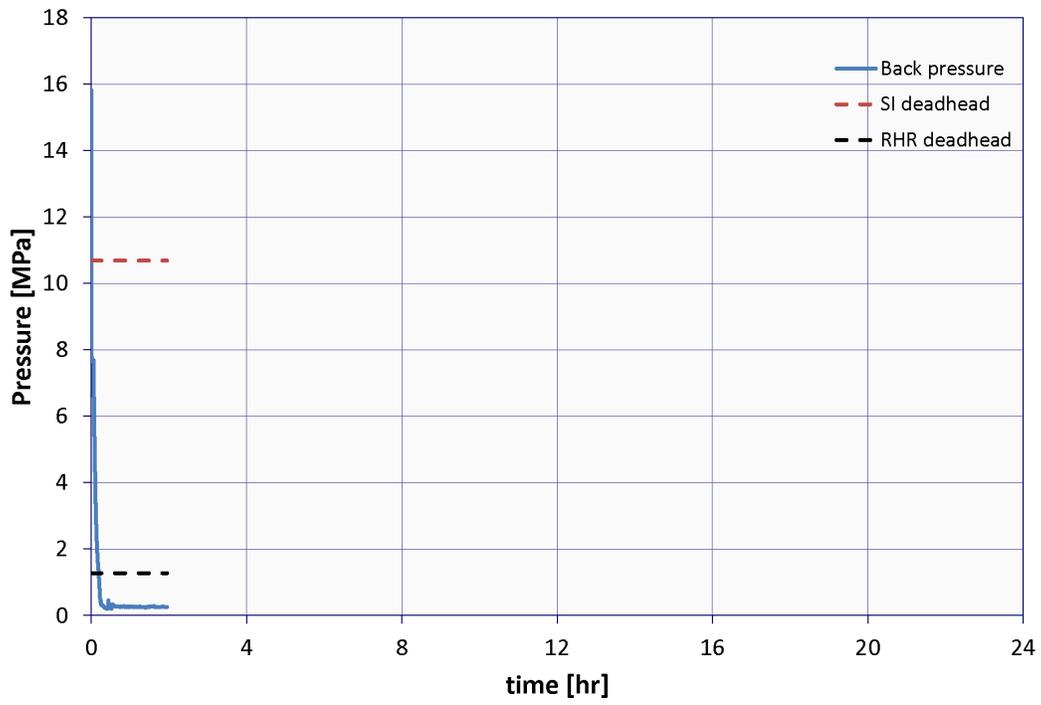
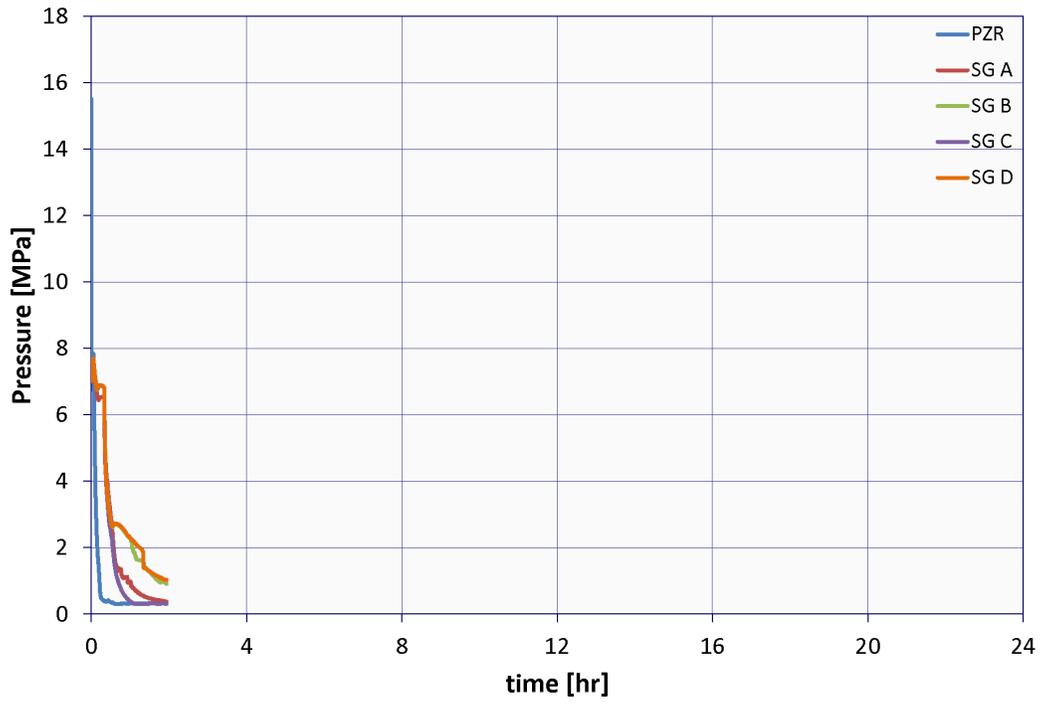


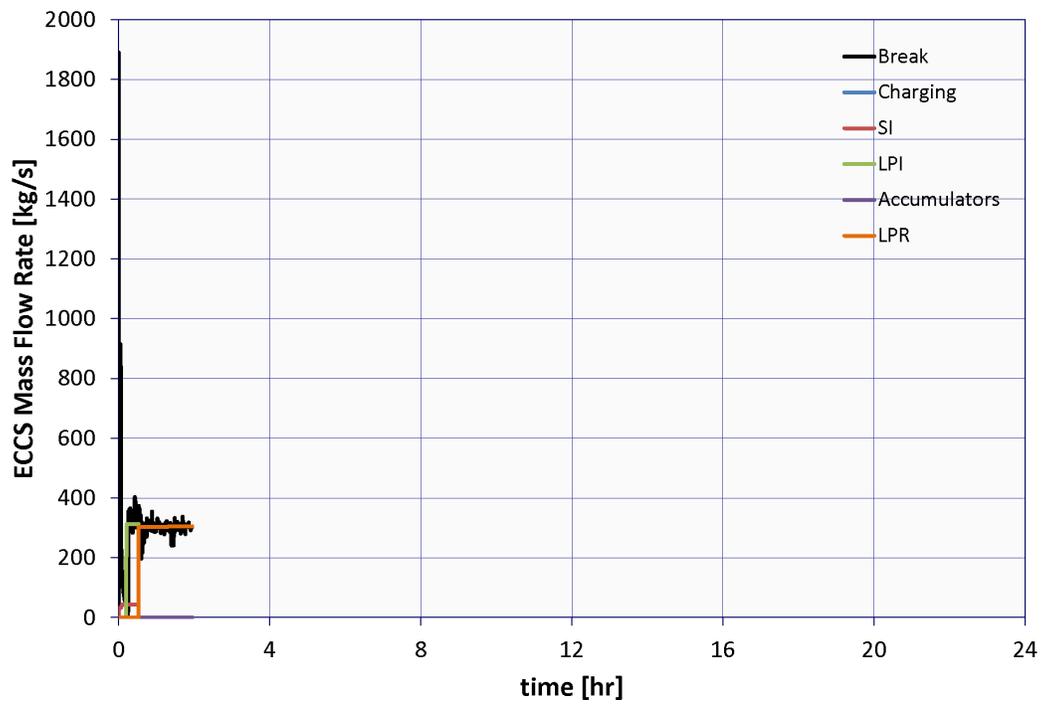
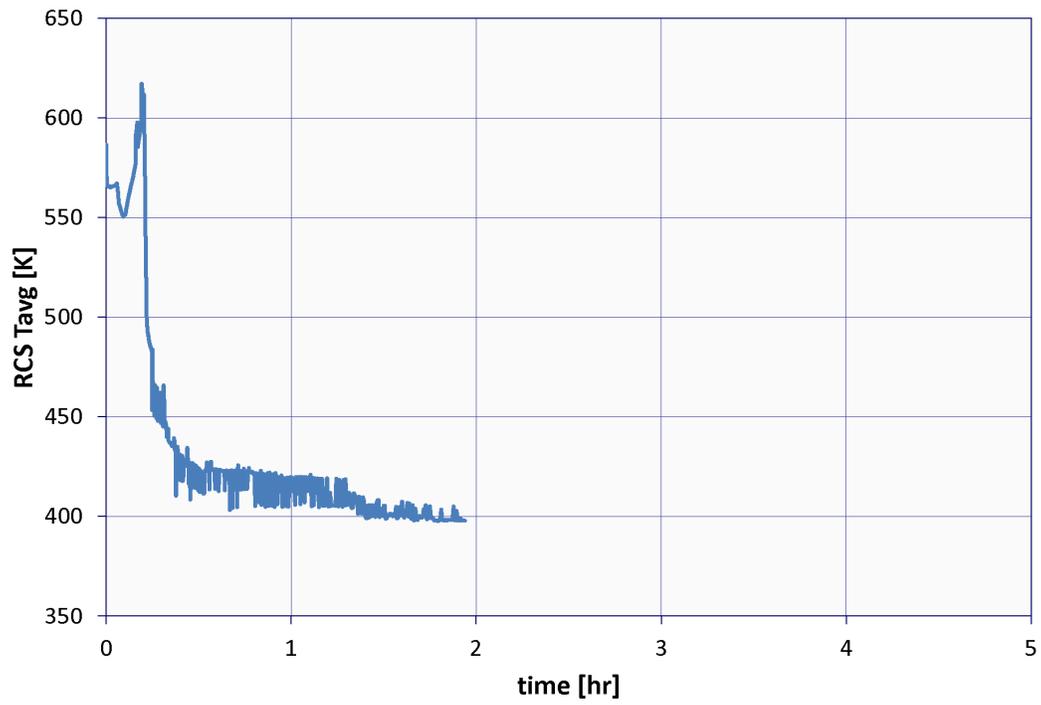


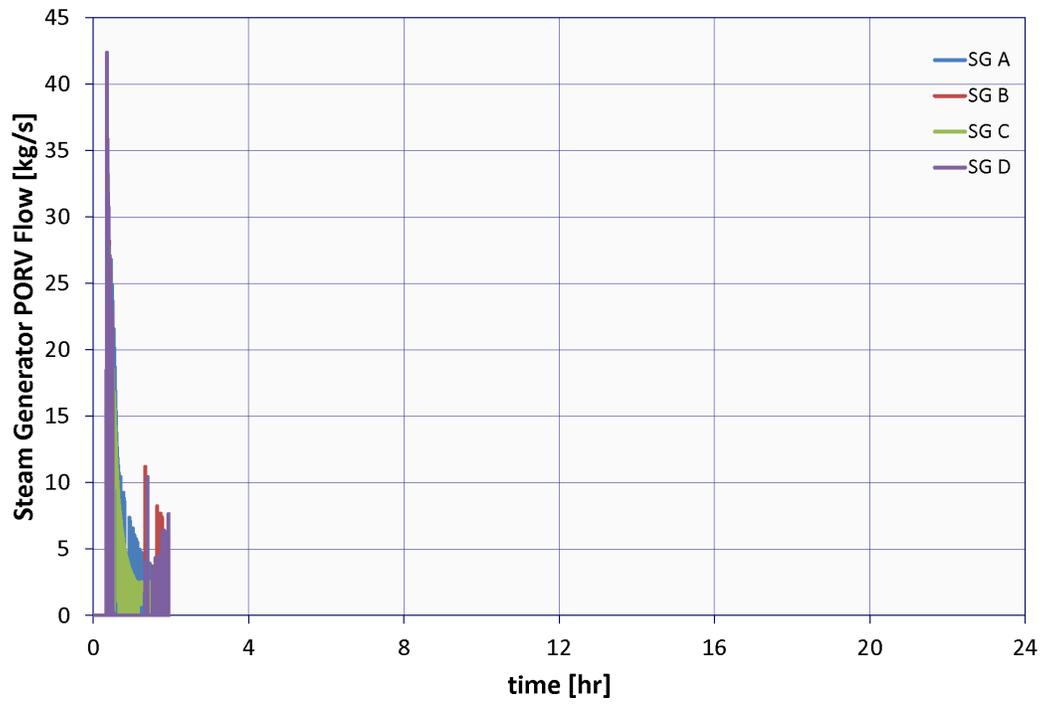
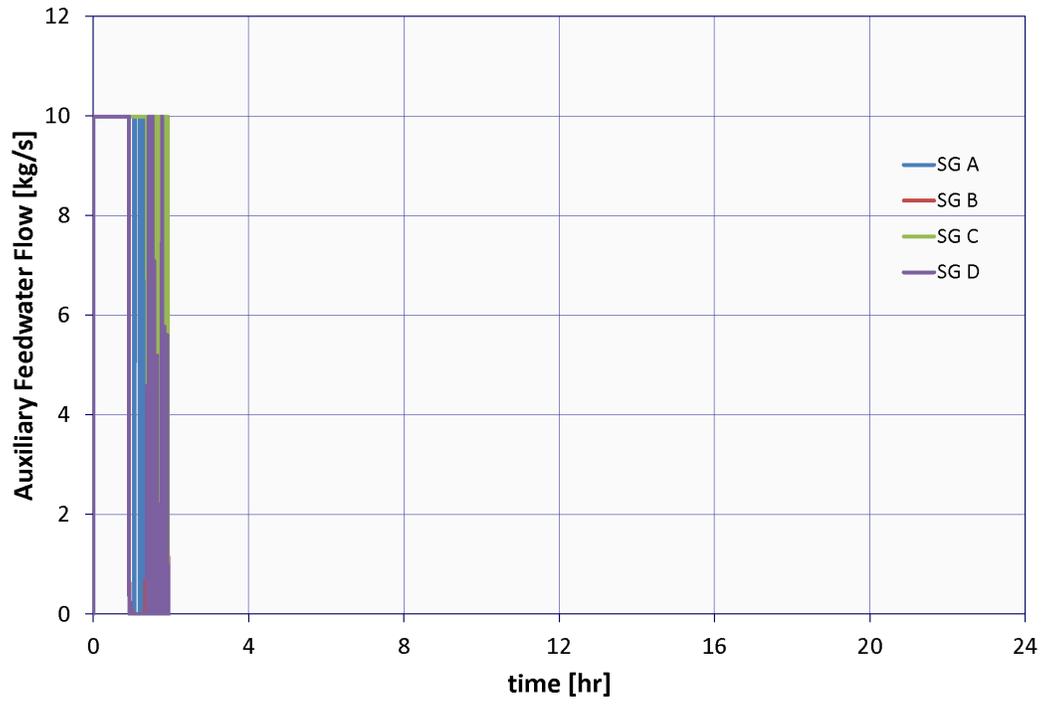
E.2.2.1 Case 2a: 6-in. Break, 100 °F/hr Cooldown at 20 min, 2/2 CS Pumps, 0/4 RCFC, CS Recirc

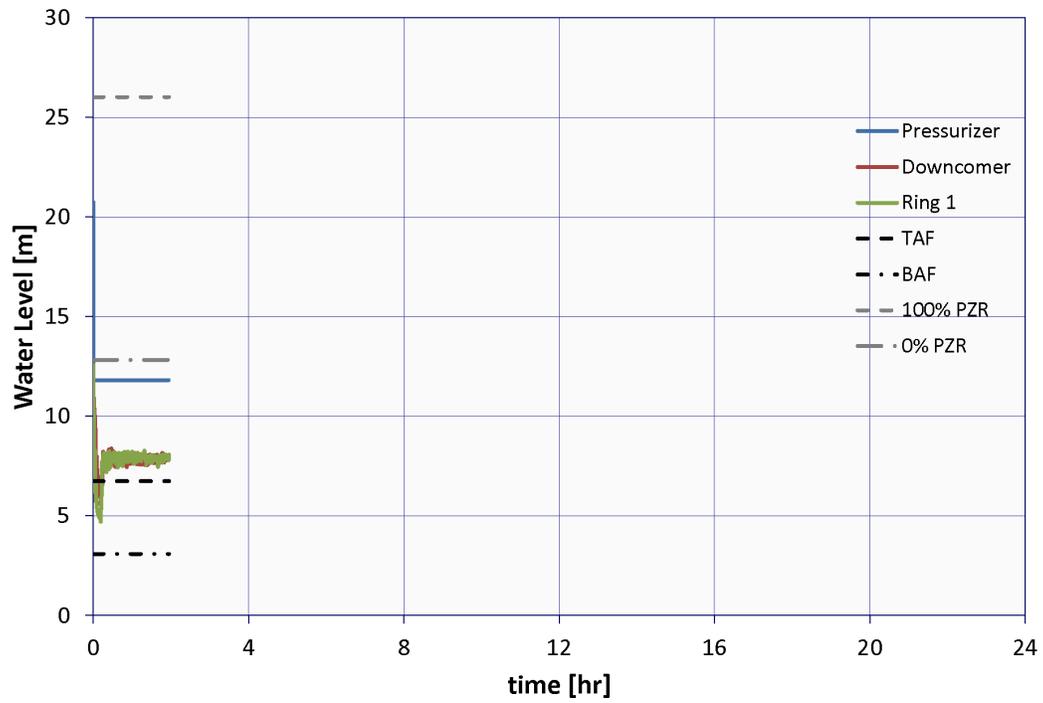
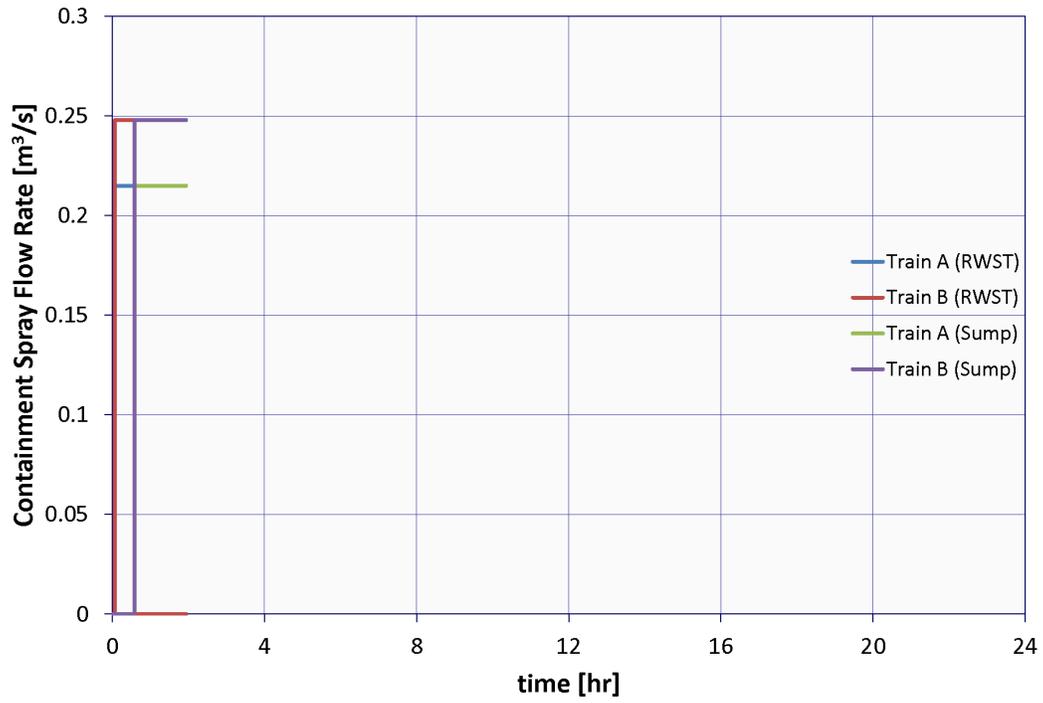


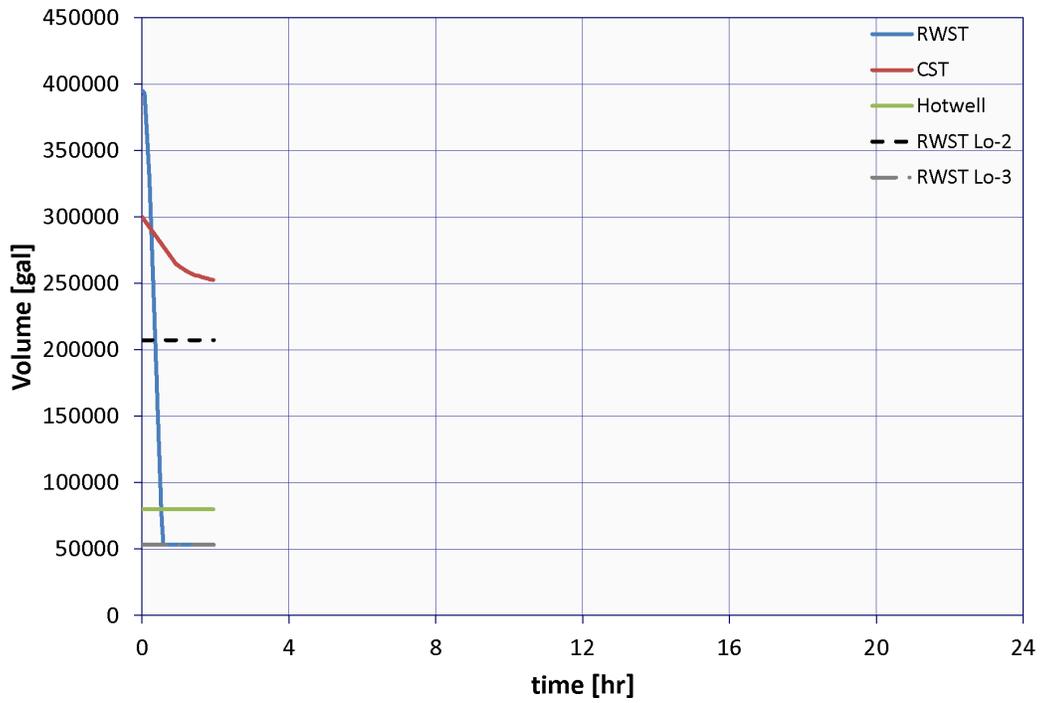
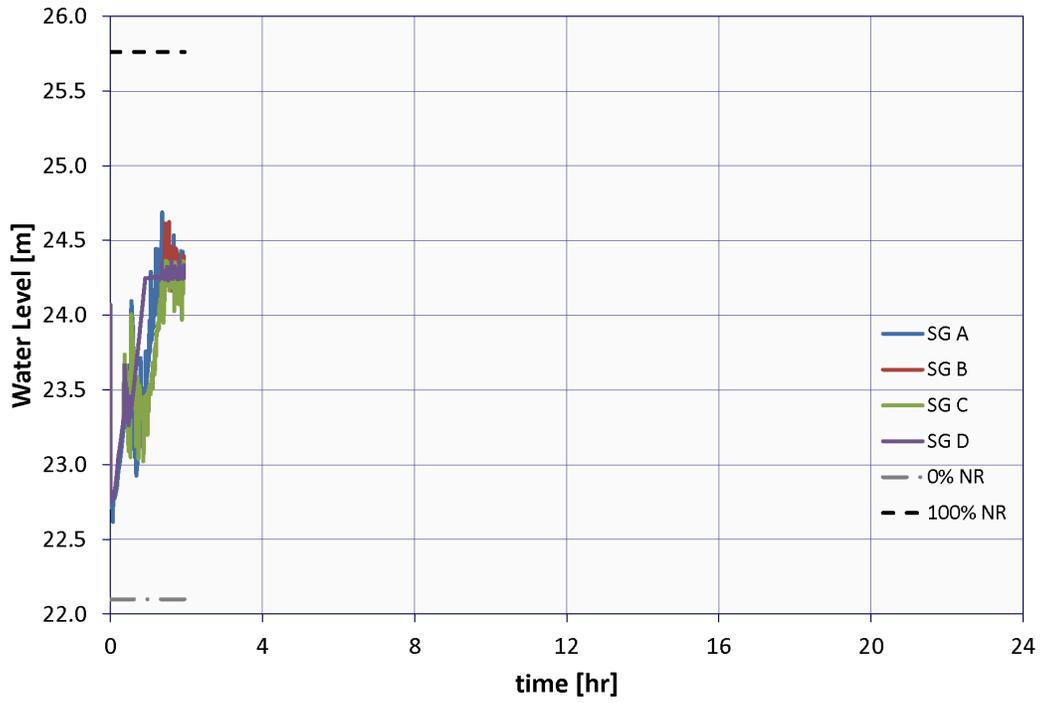




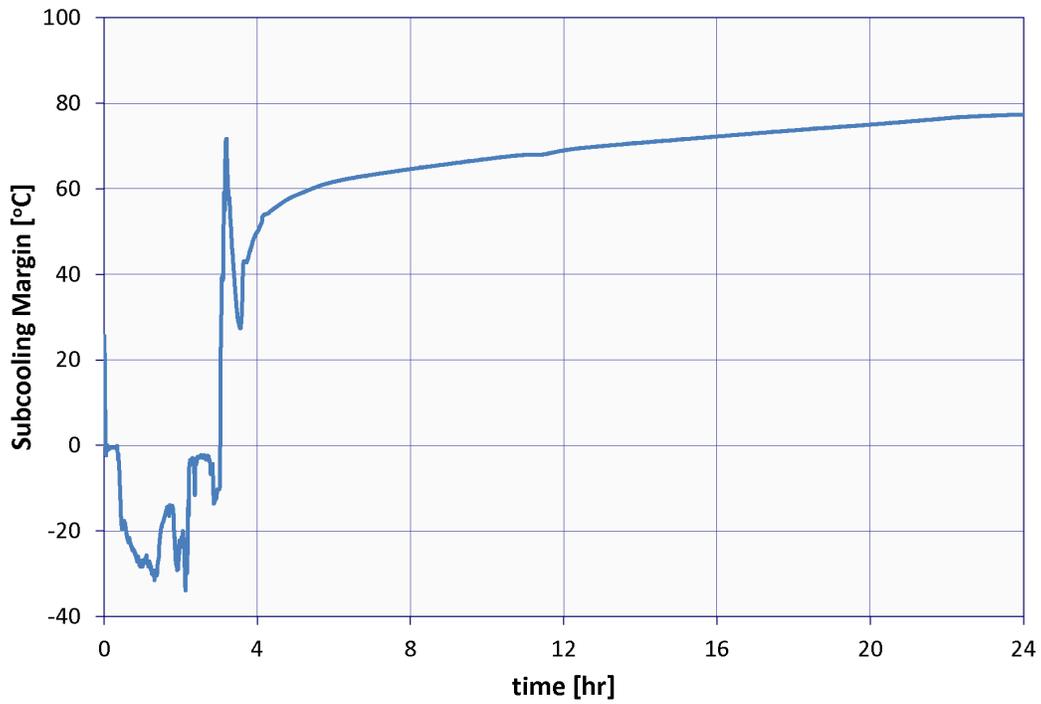
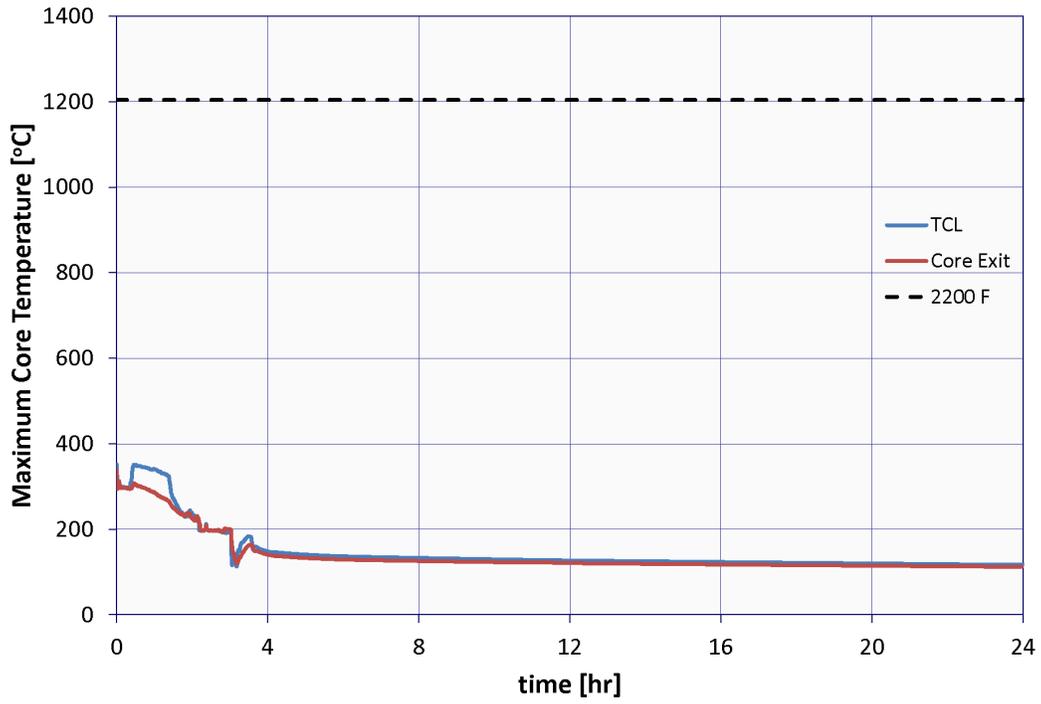


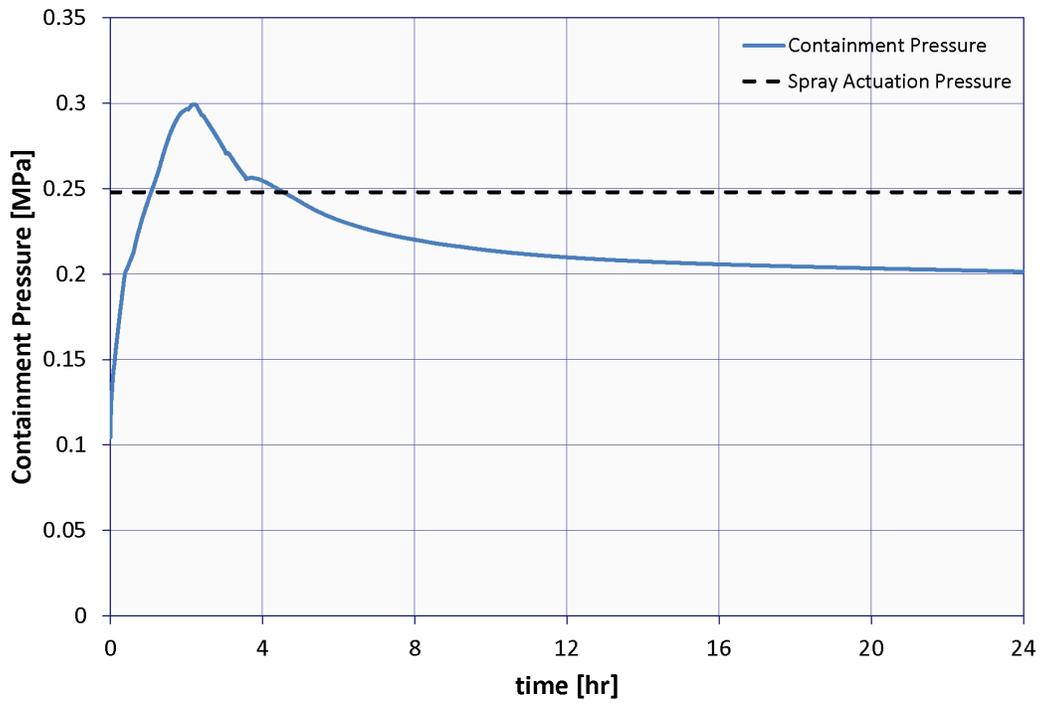
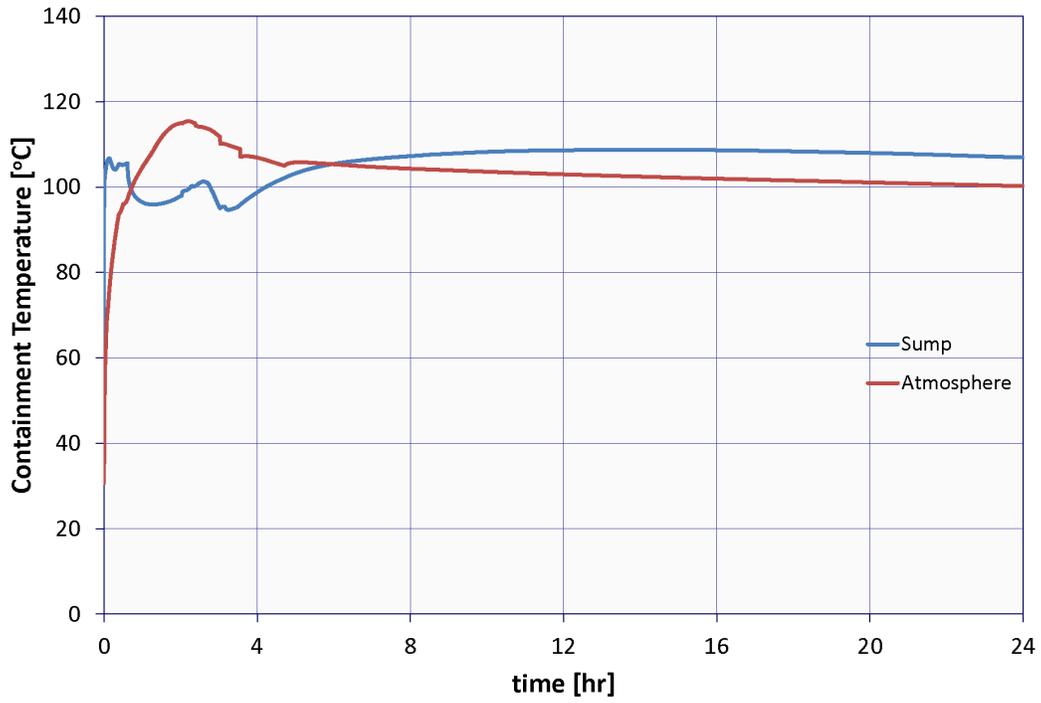


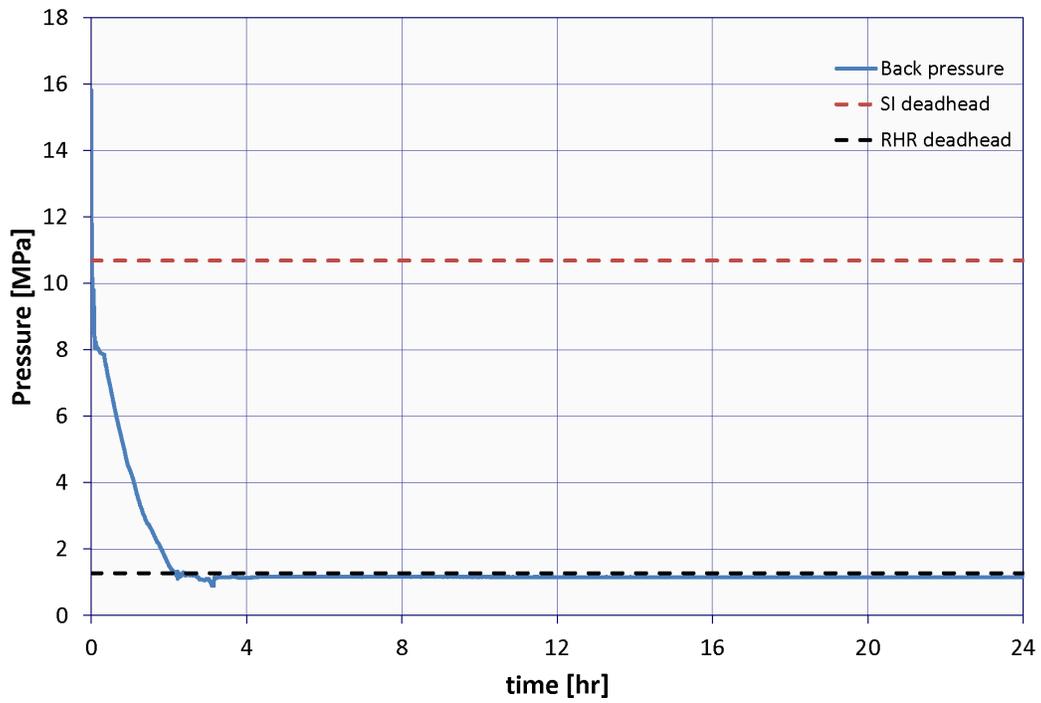
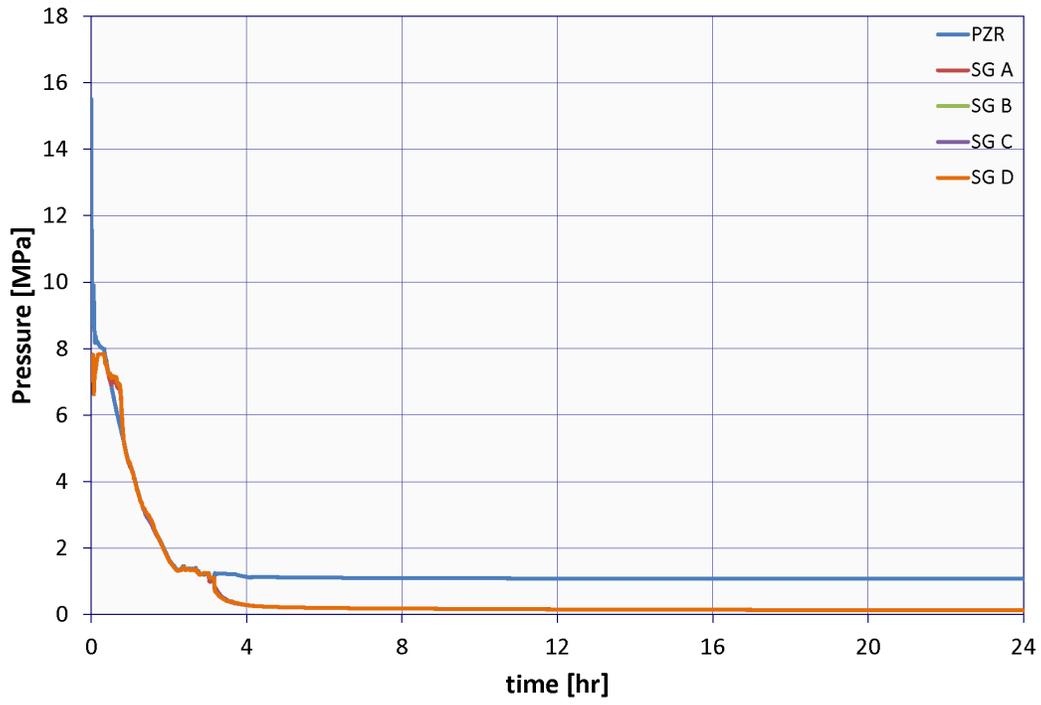


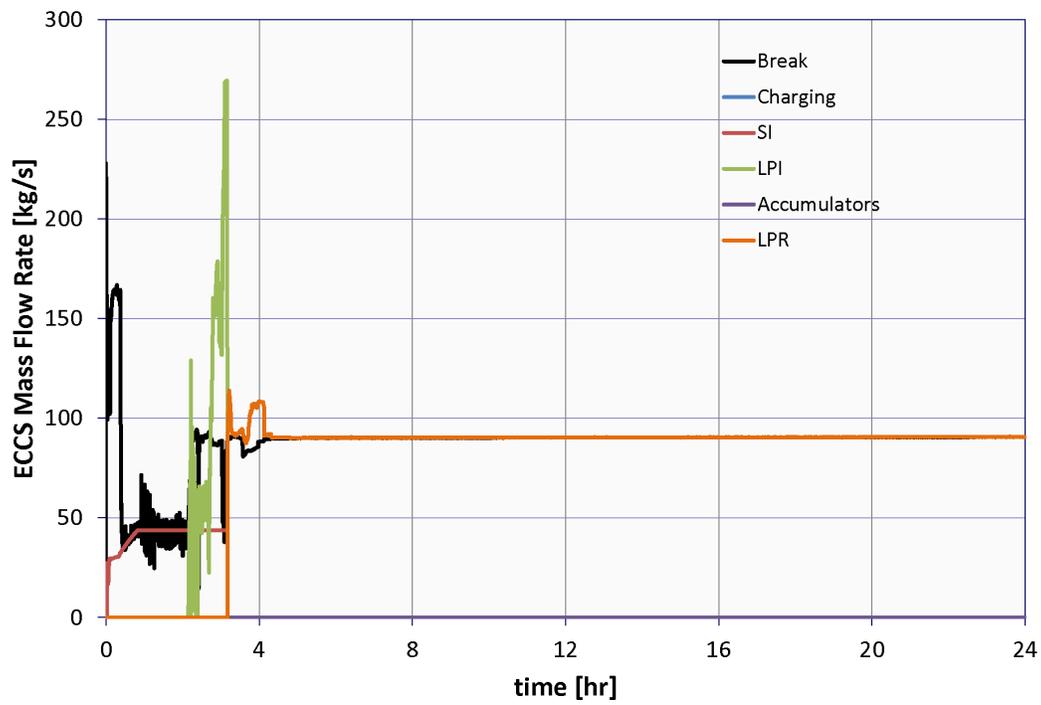
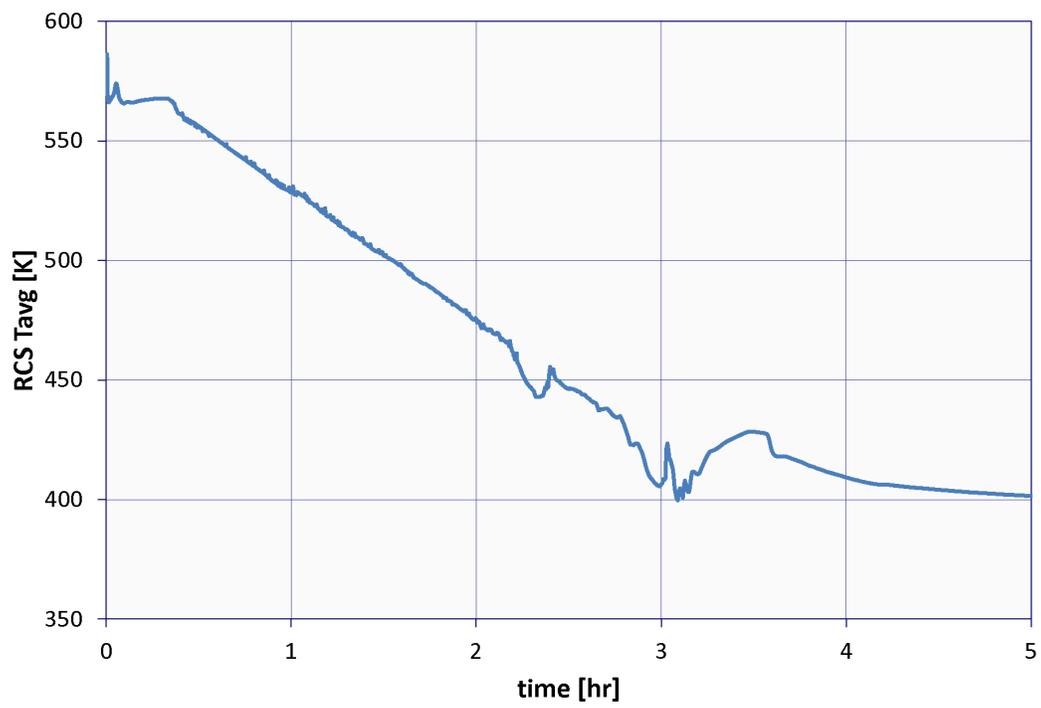


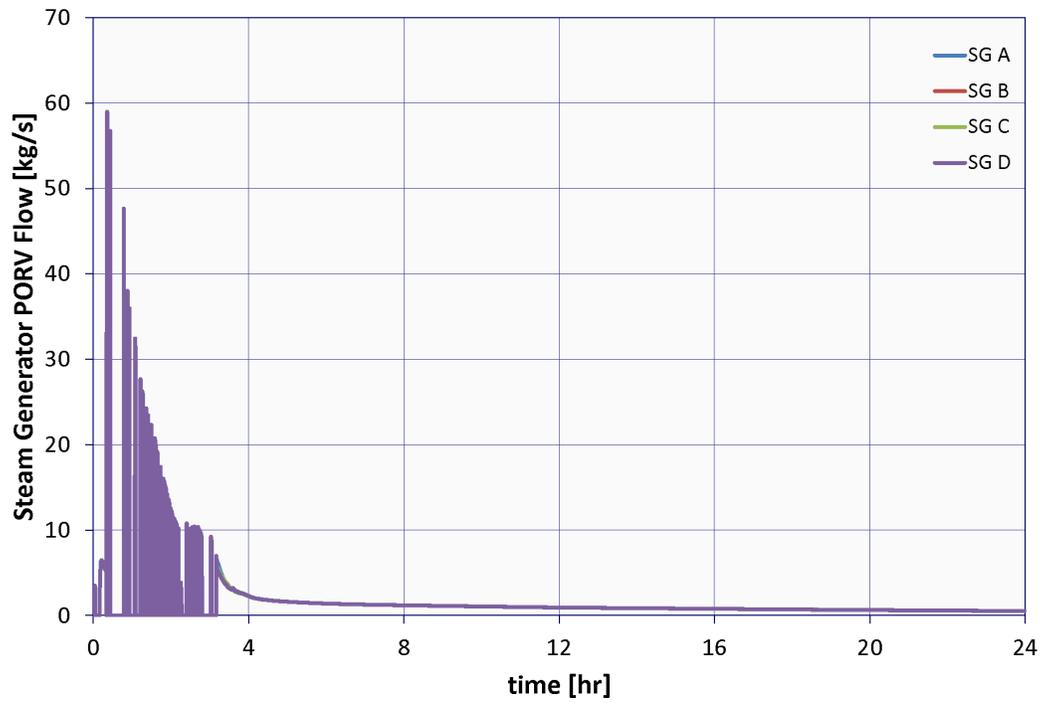
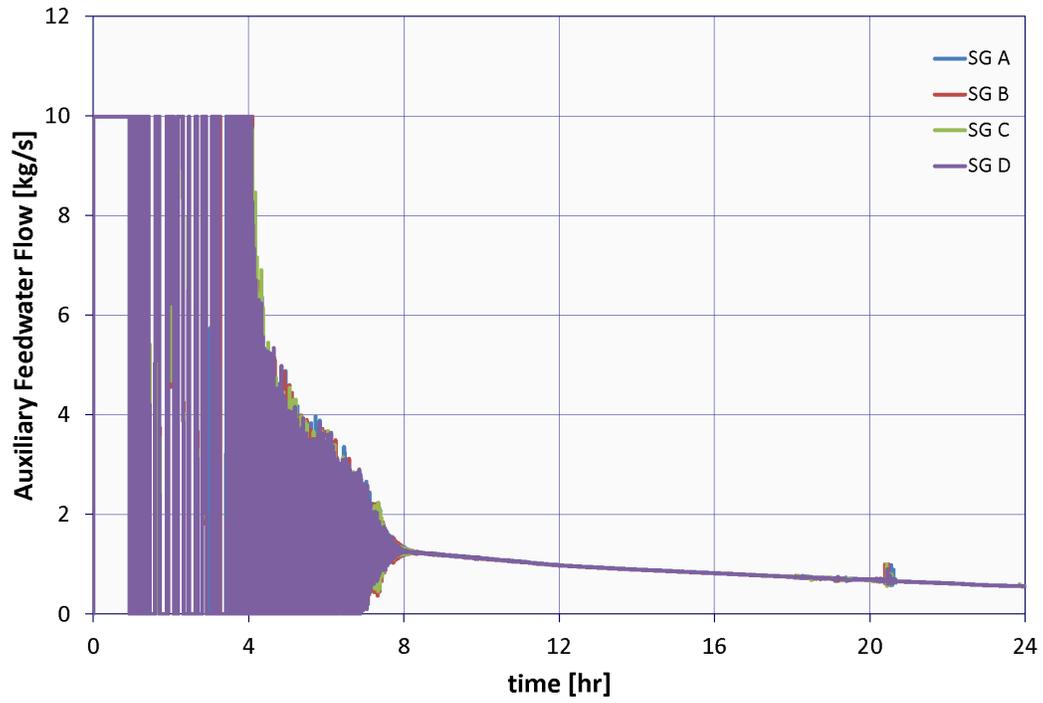
**E.2.3 Case 3: 2-in. Break, 100 °F/hr Cooldown at 20 min, 0/2 CS Pumps, 0/4 RCFC**

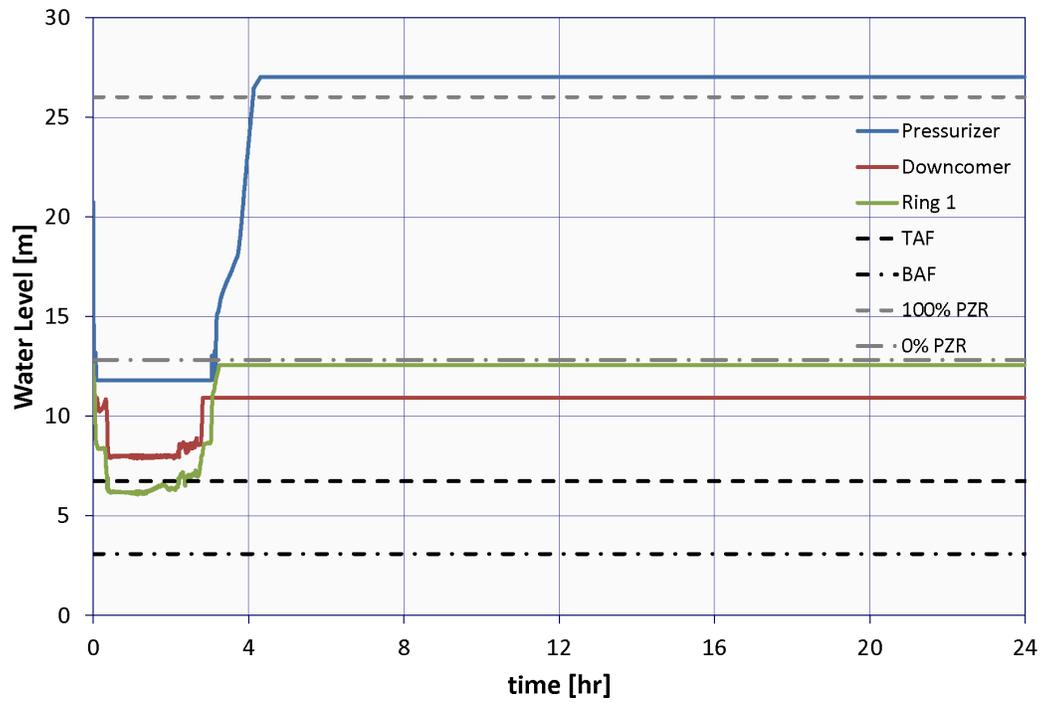
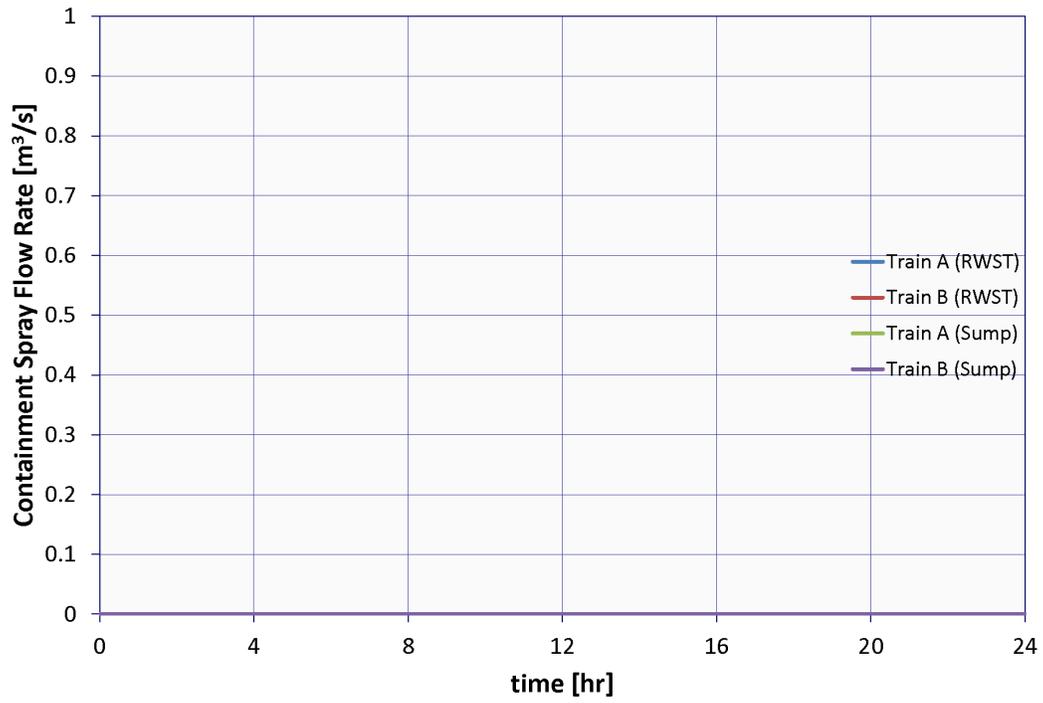


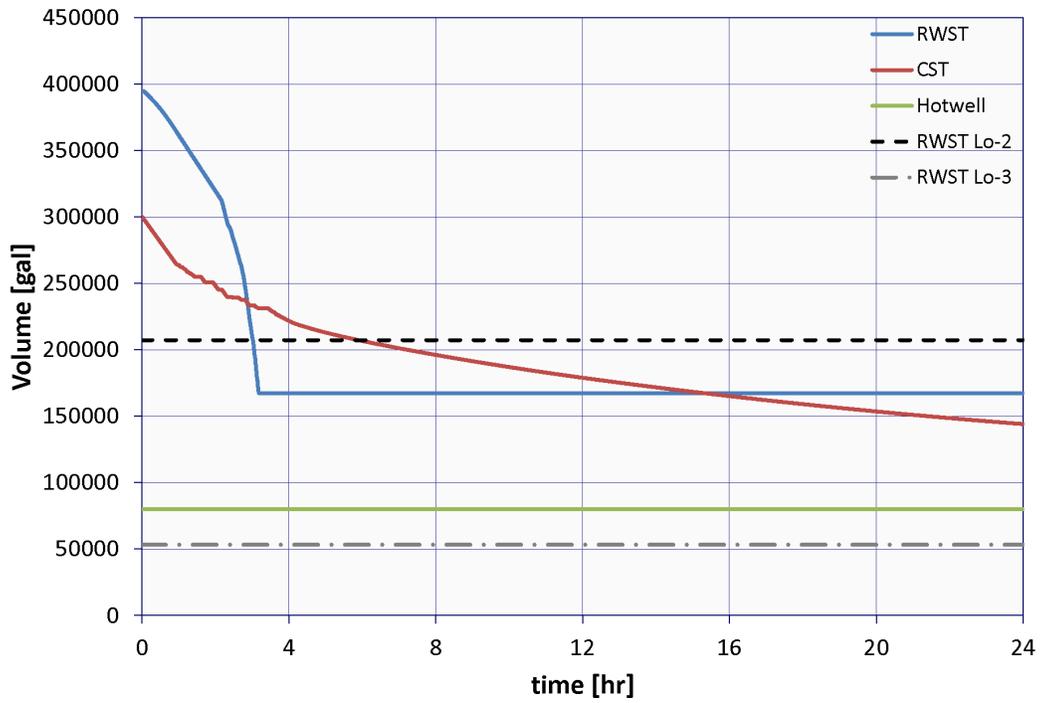
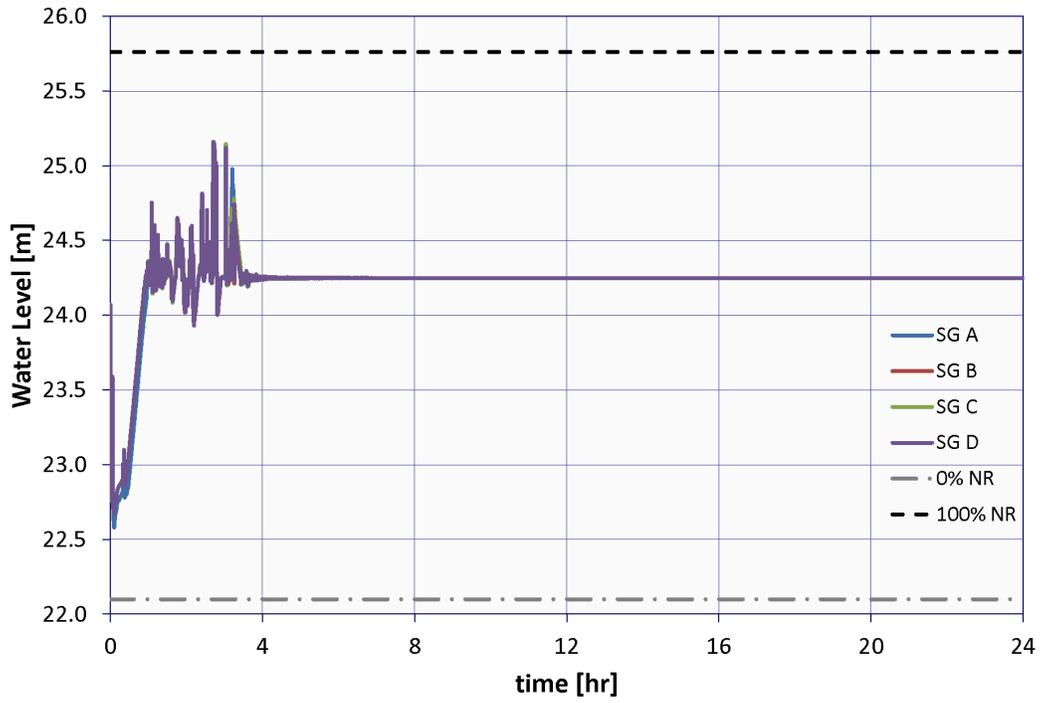




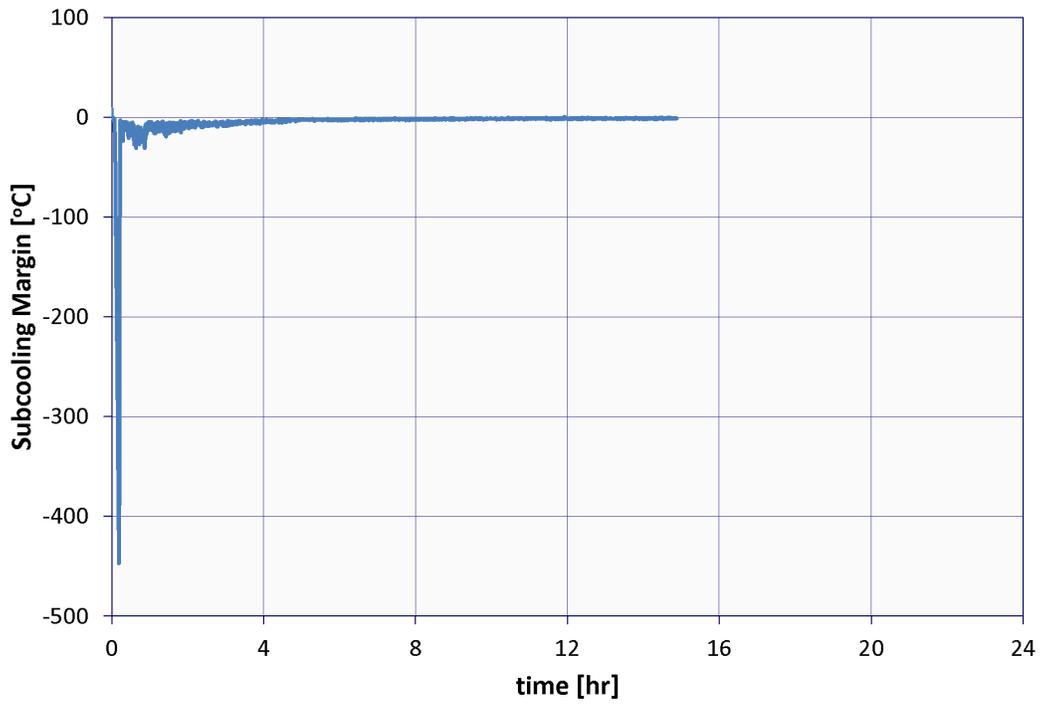
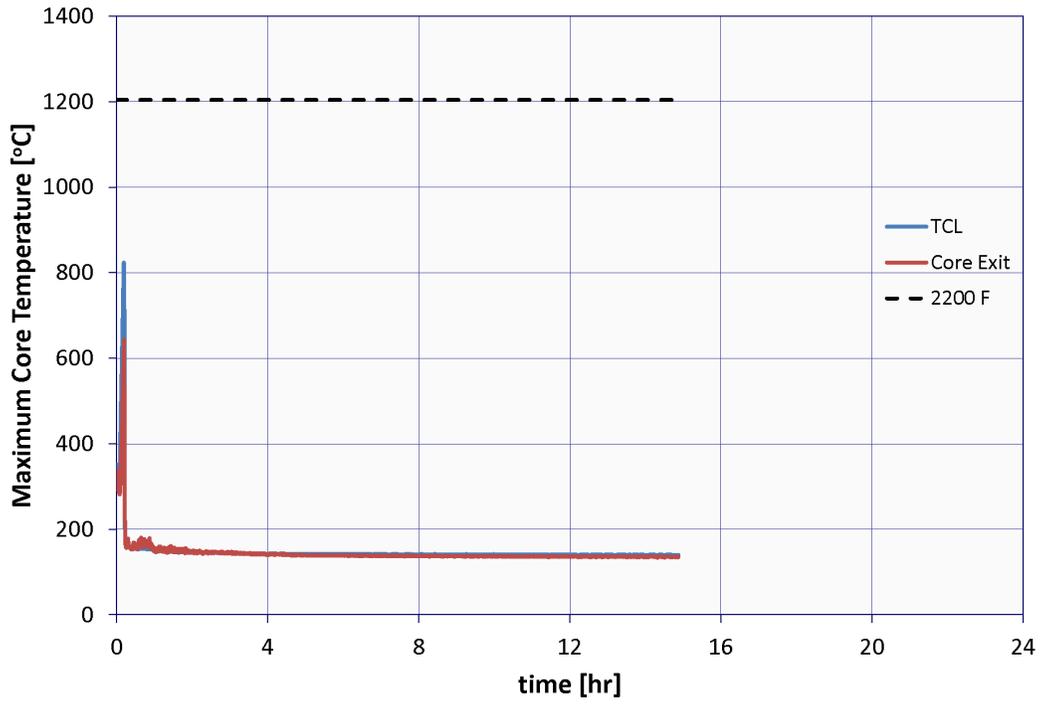


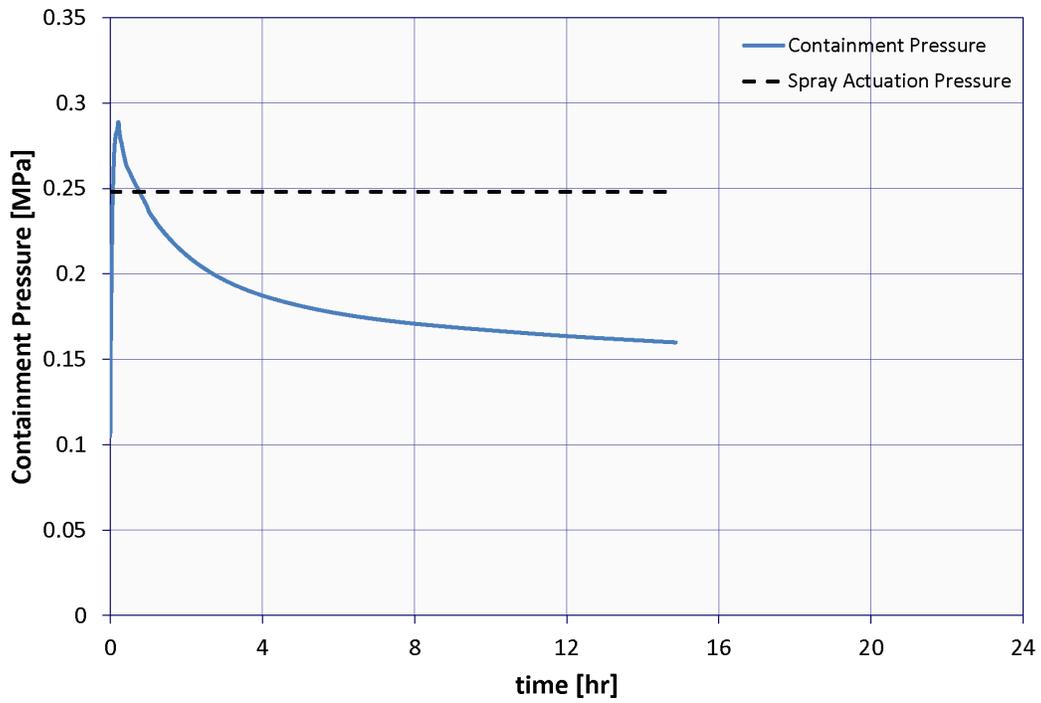
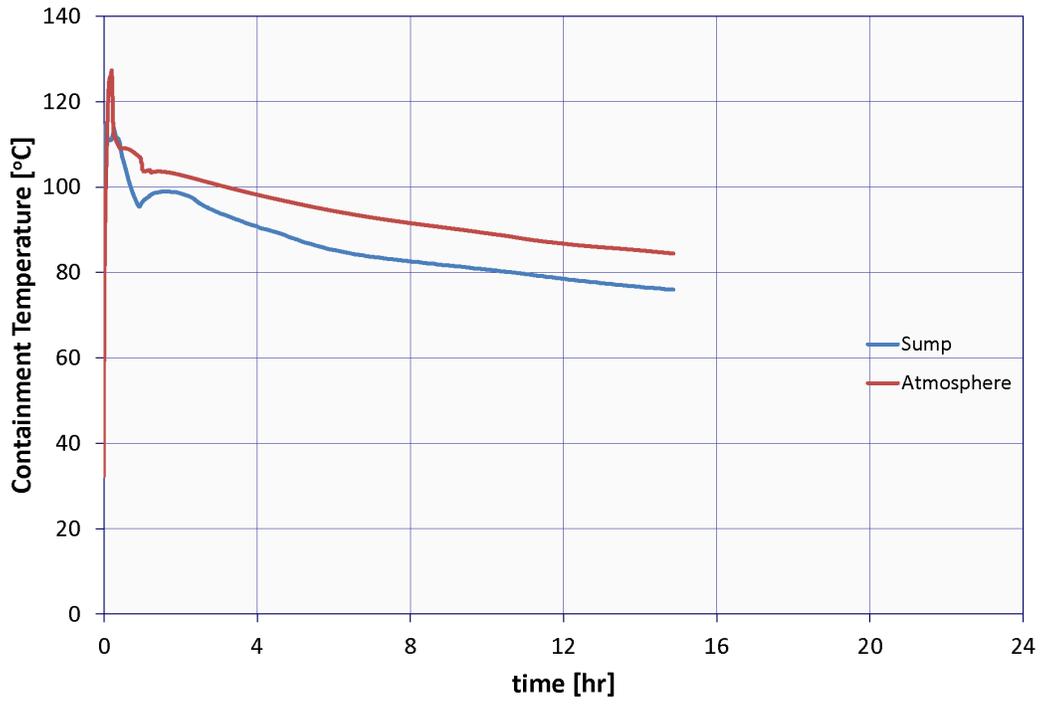


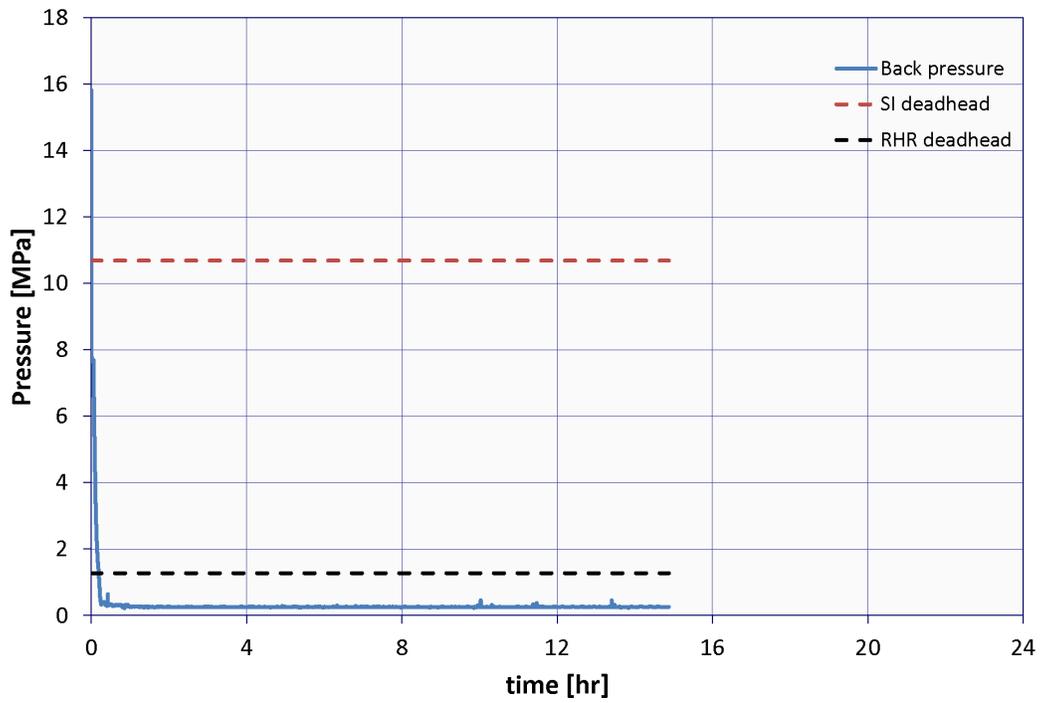
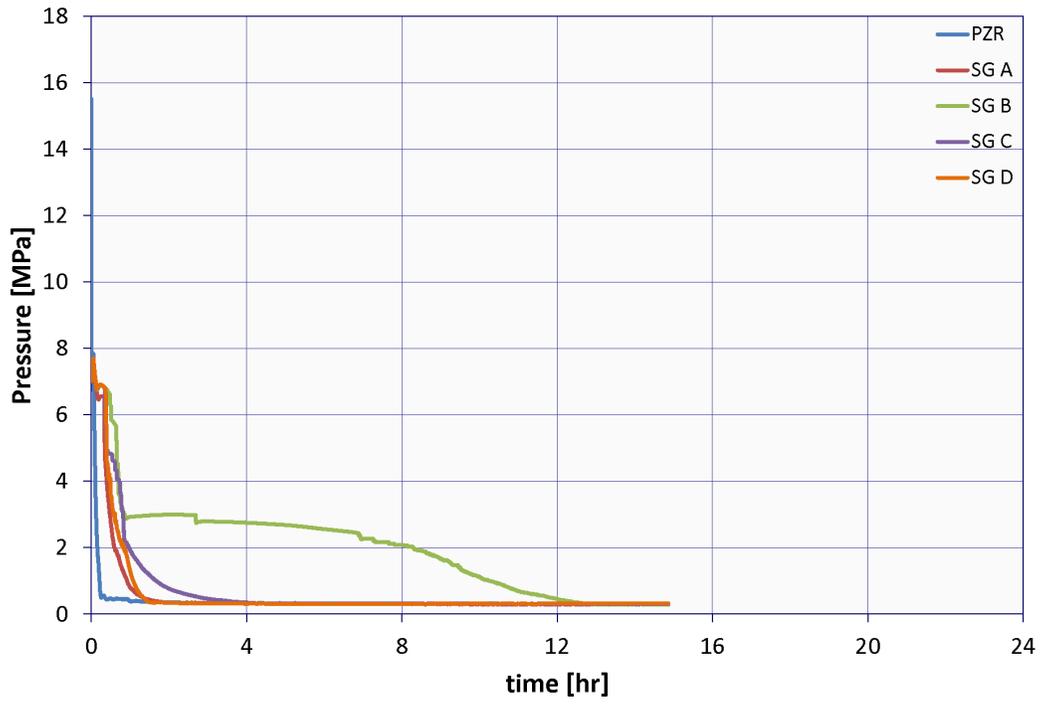


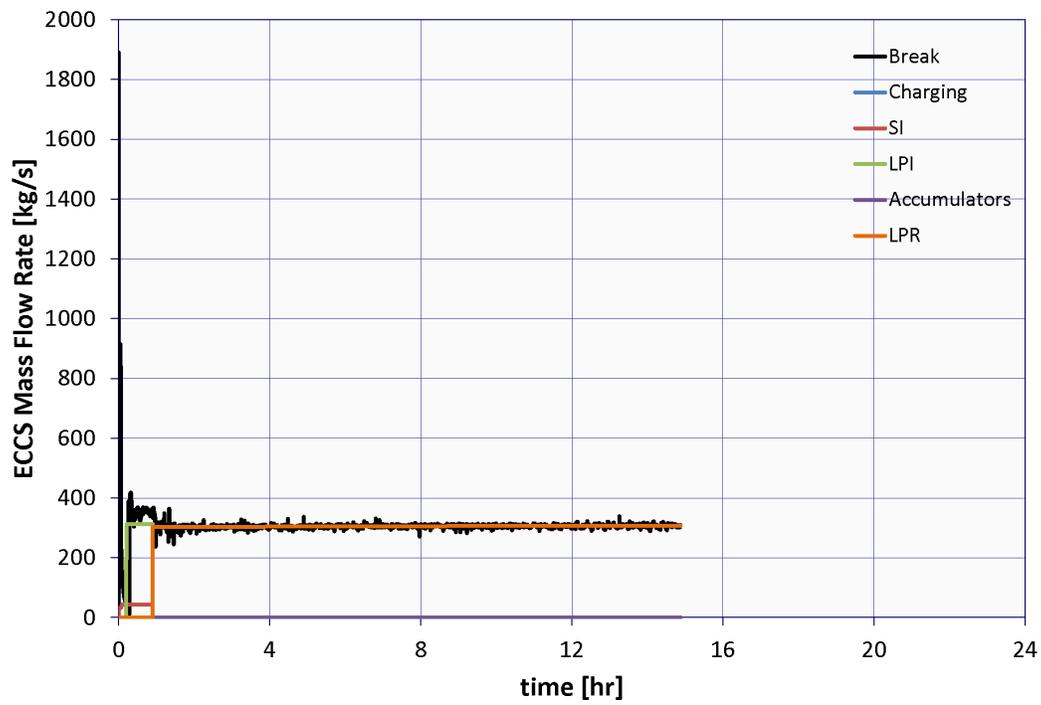
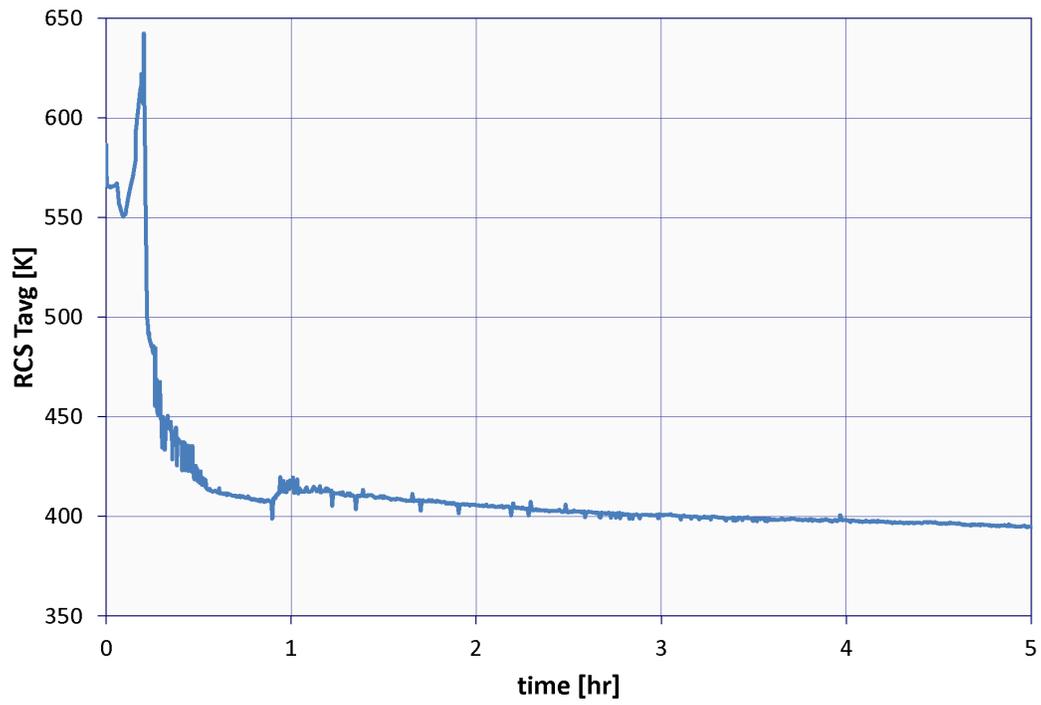


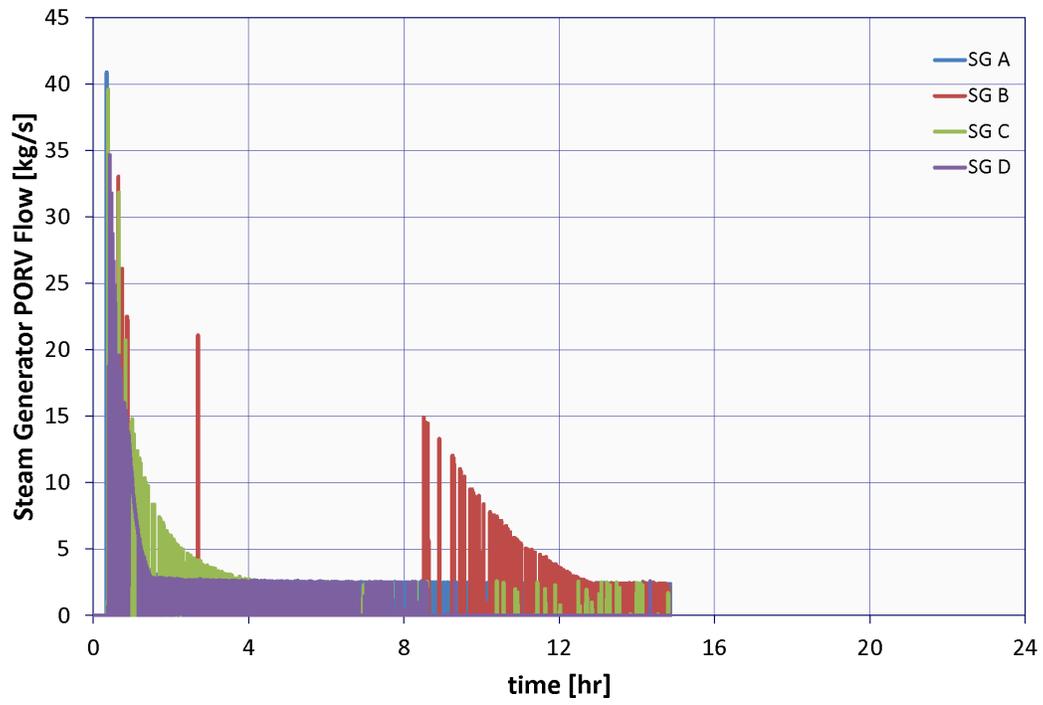
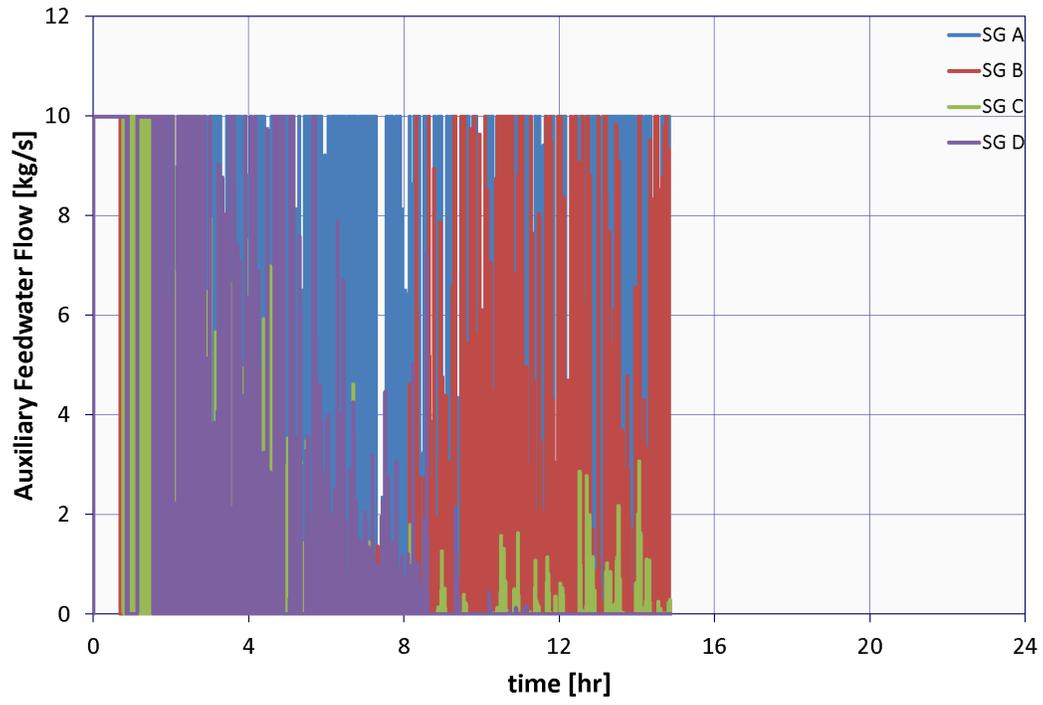
E.2.4 Case 4: 6-in. Break, 100 °F/hr Cooldown at 20 min, 0/2 CS Pumps, 0/4 RCFC

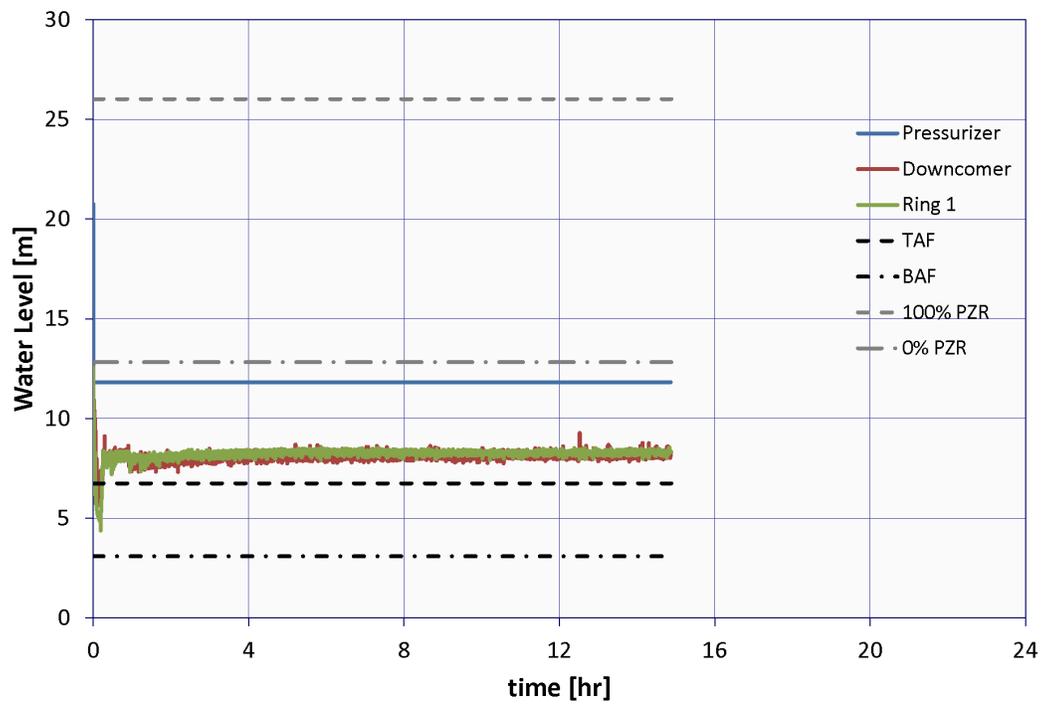
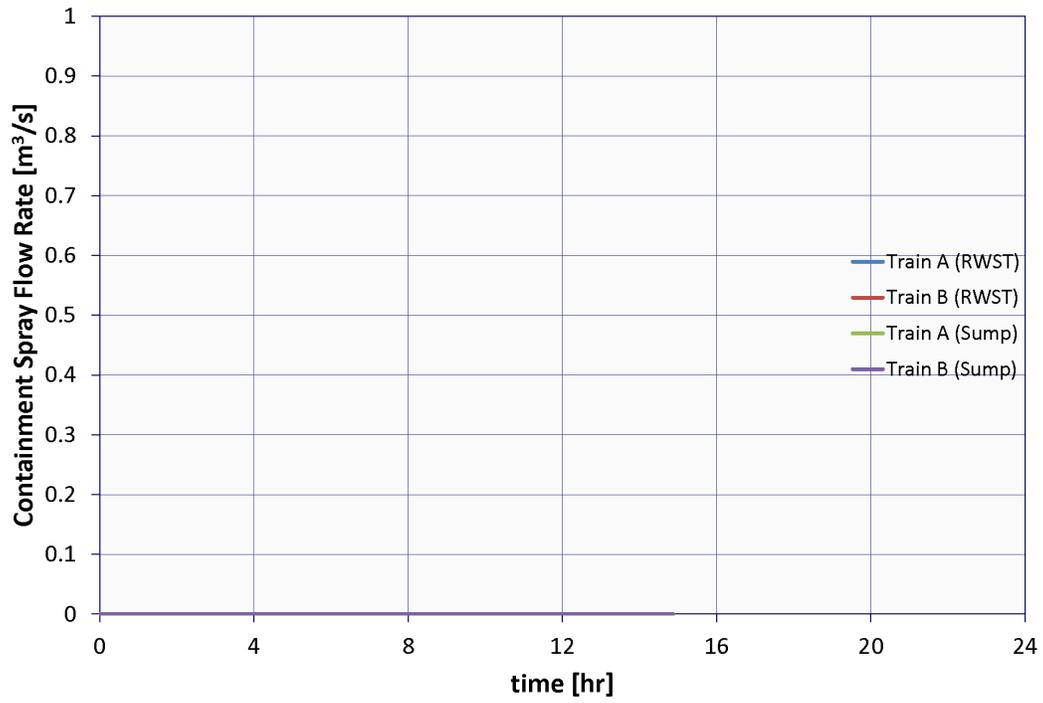


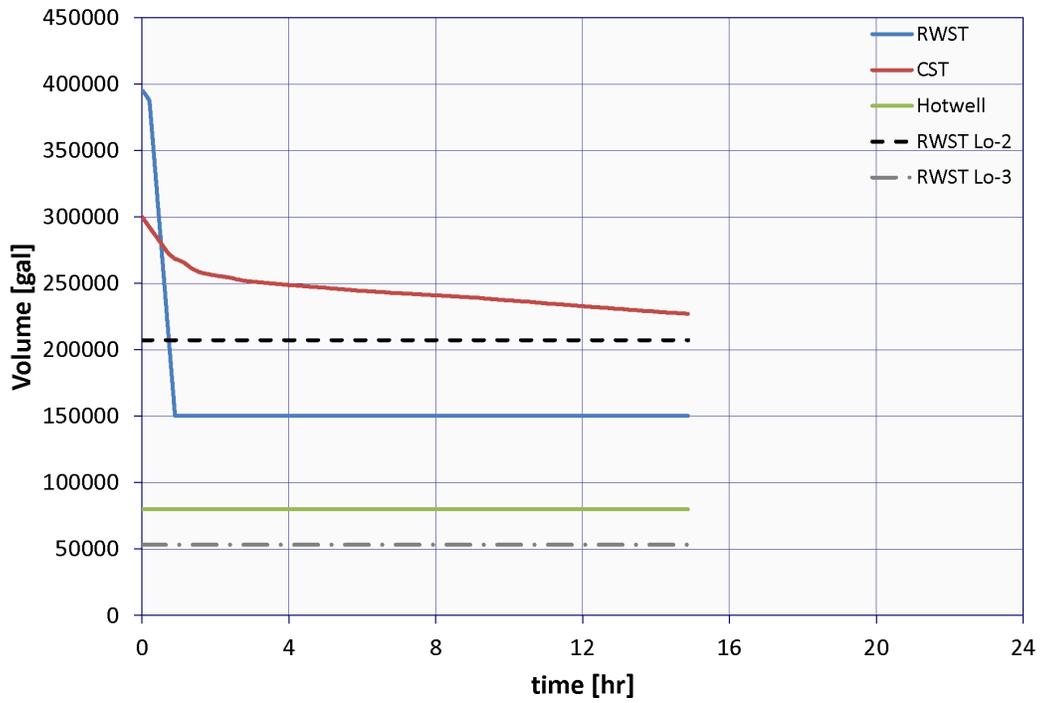
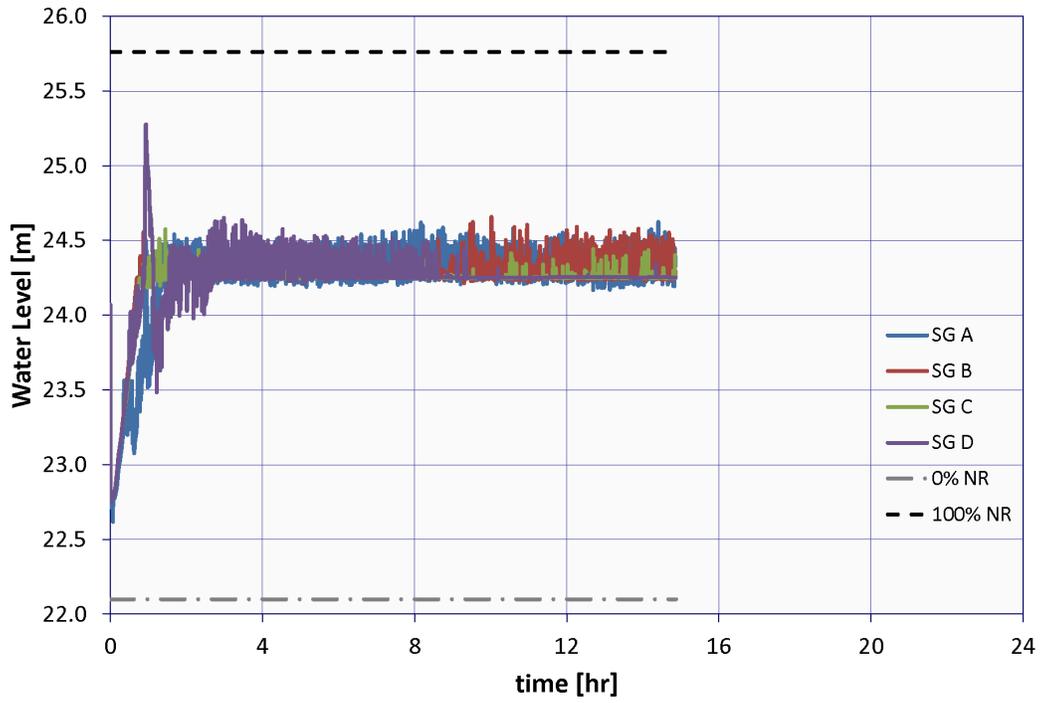




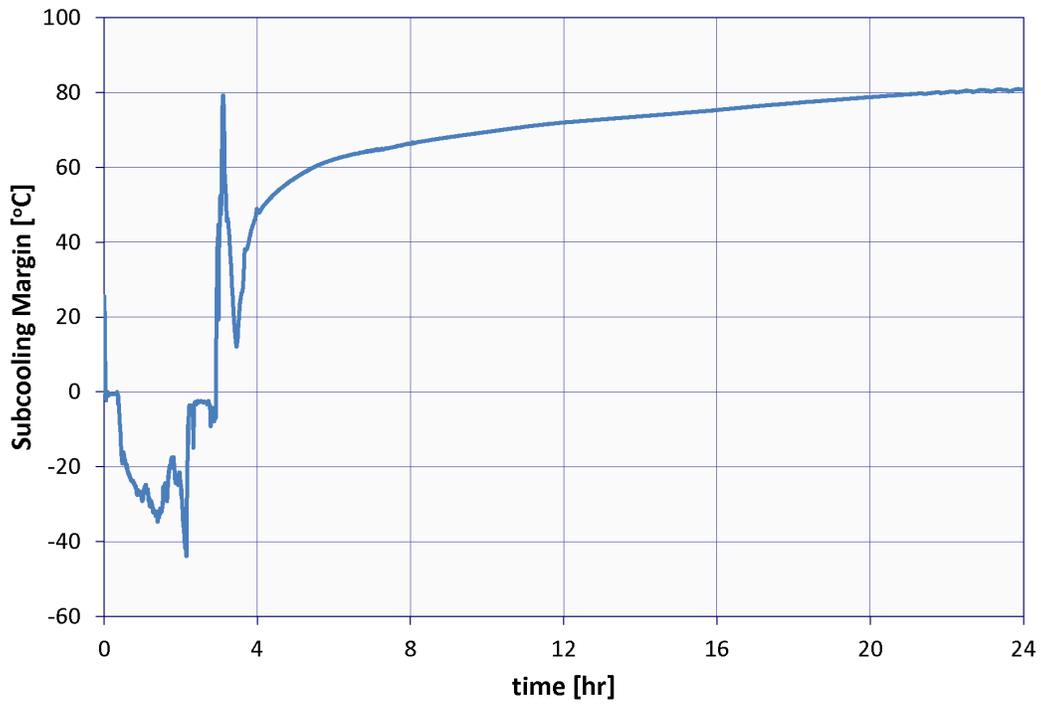
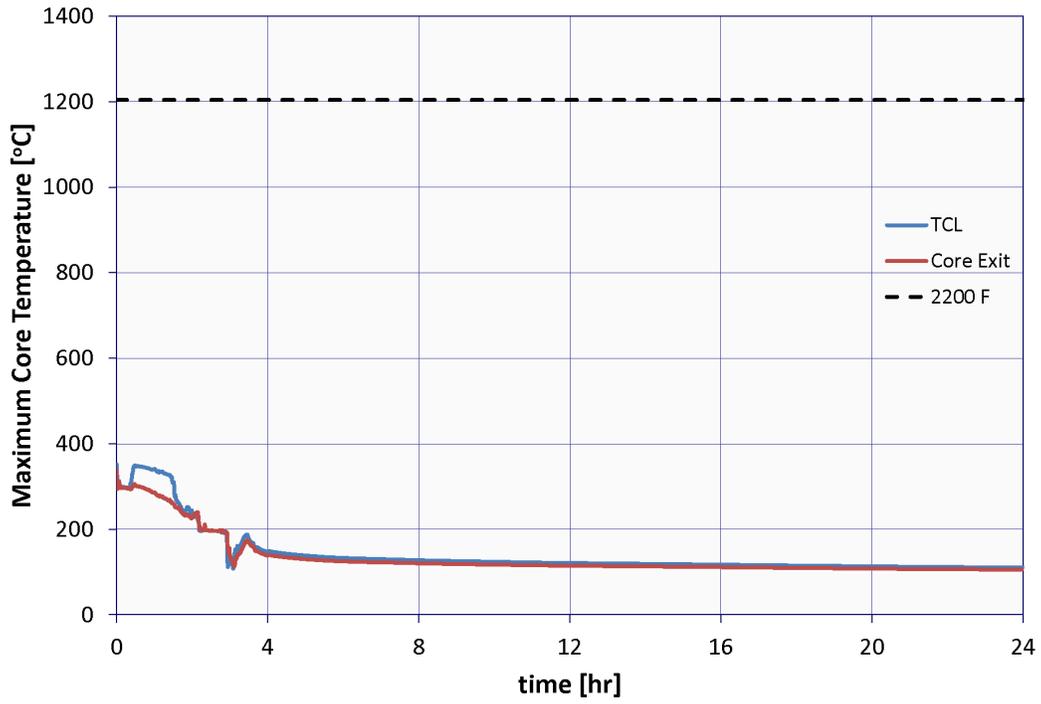


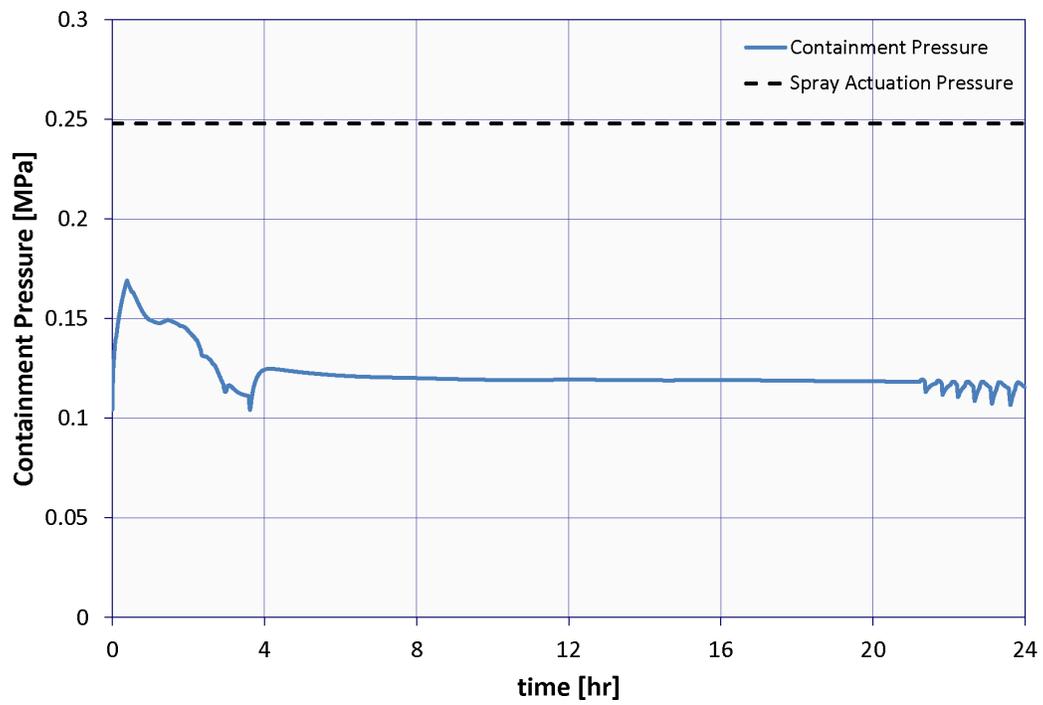
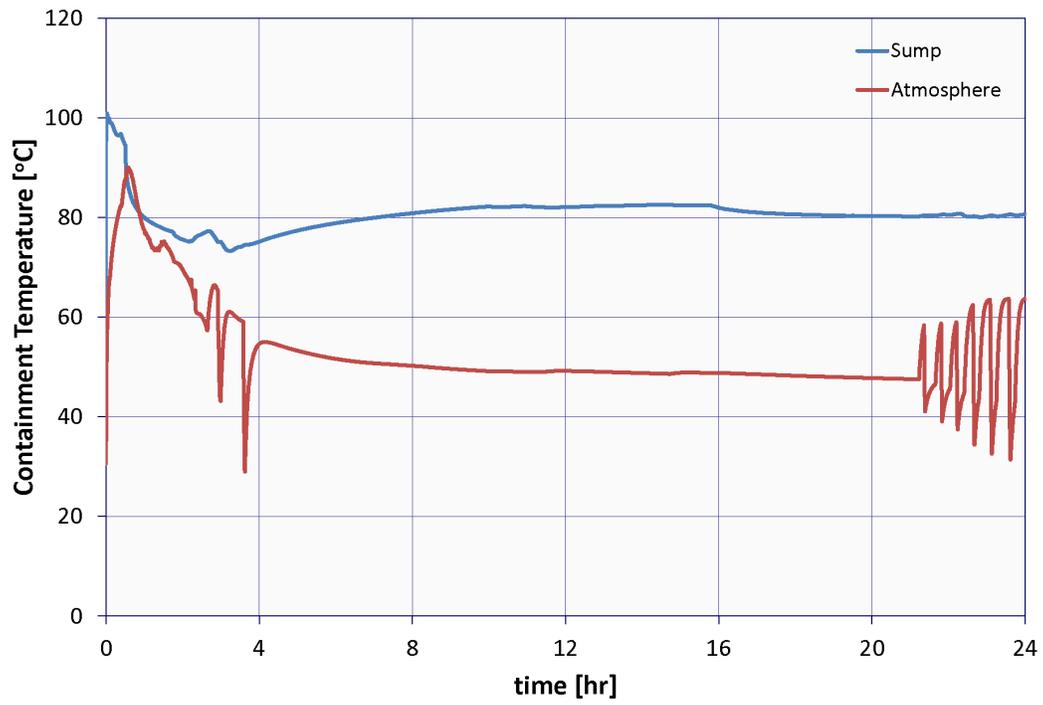


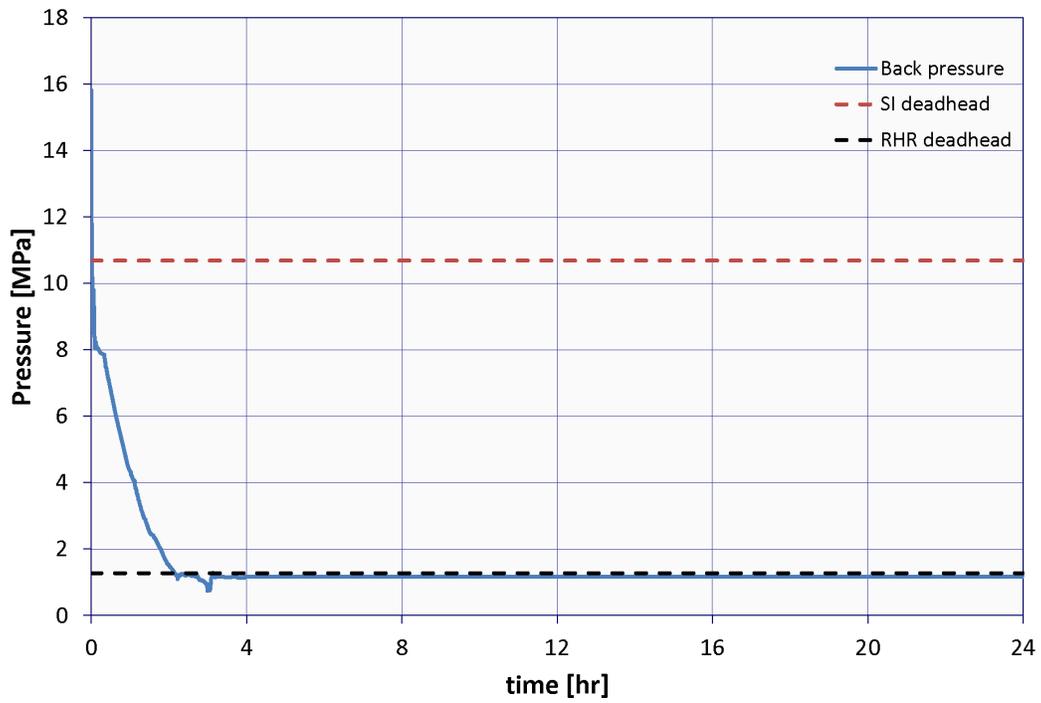
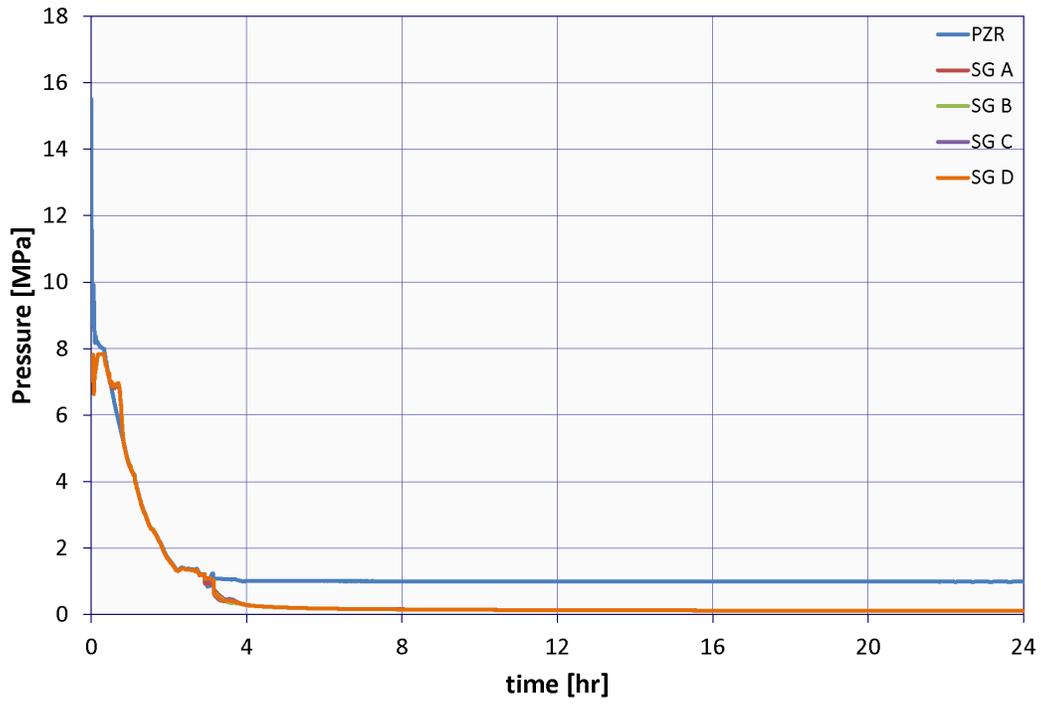


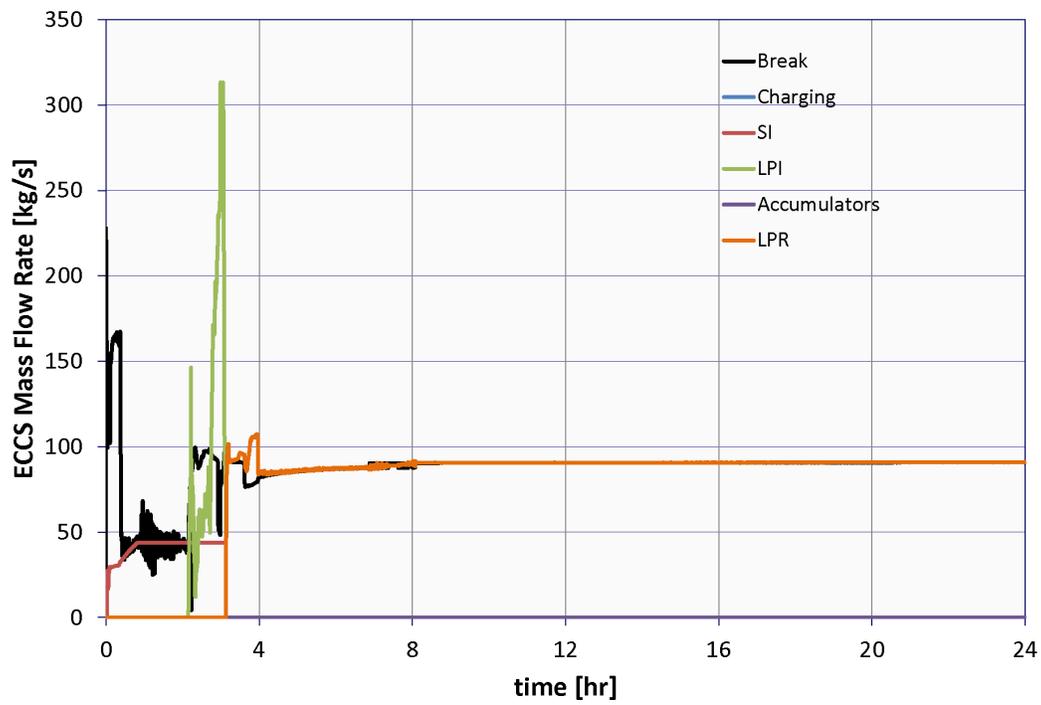
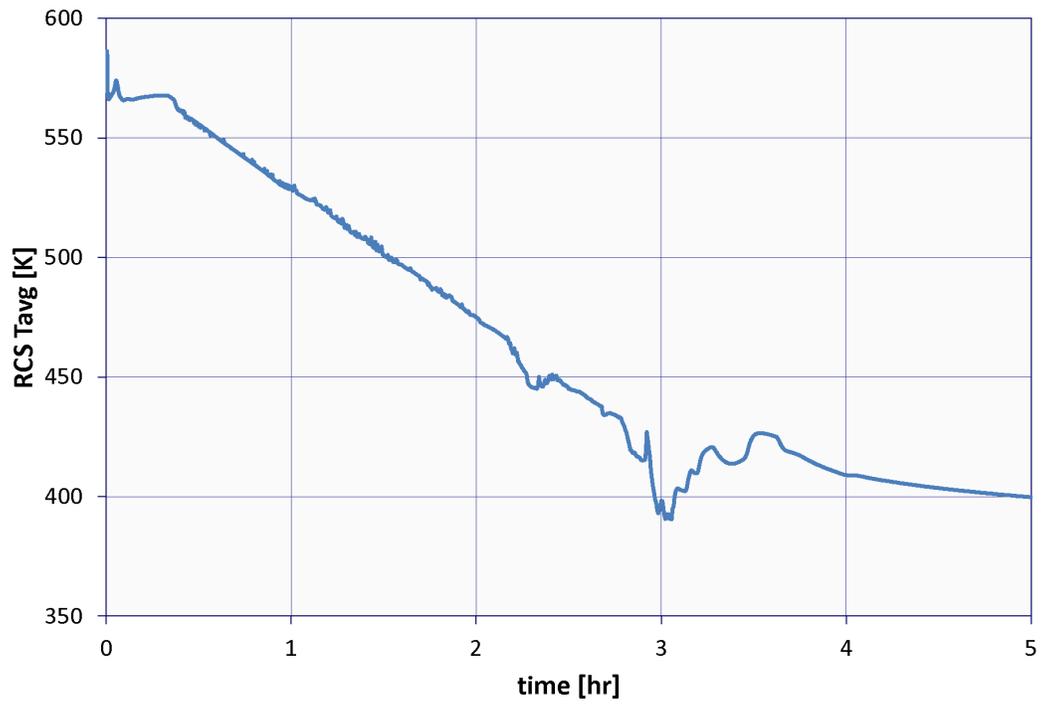


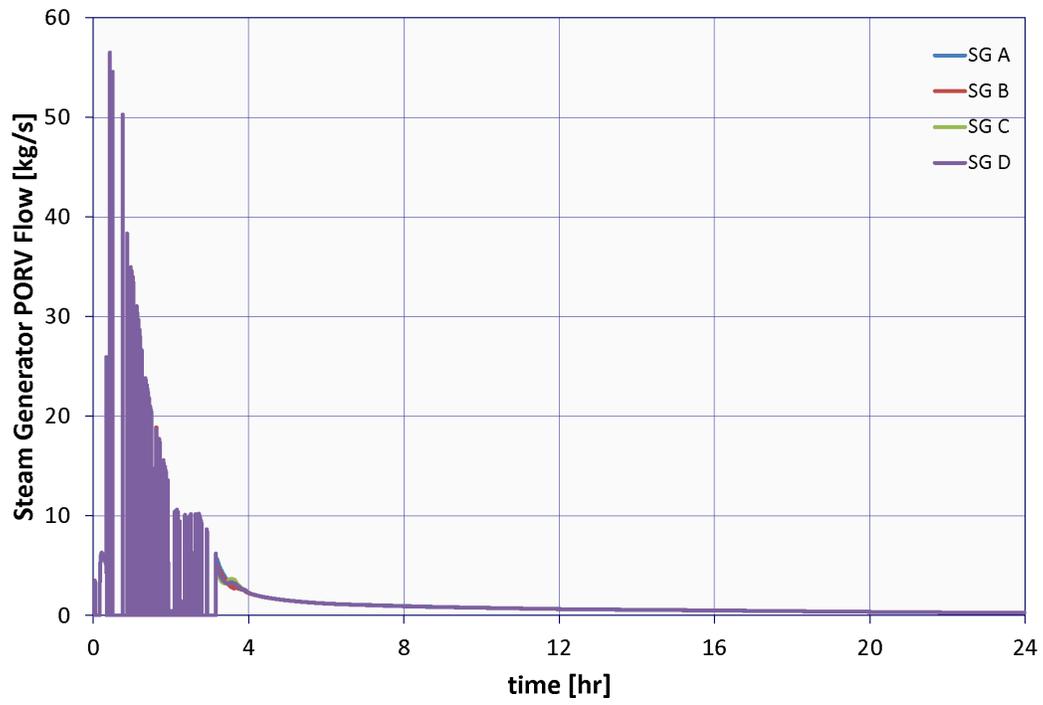
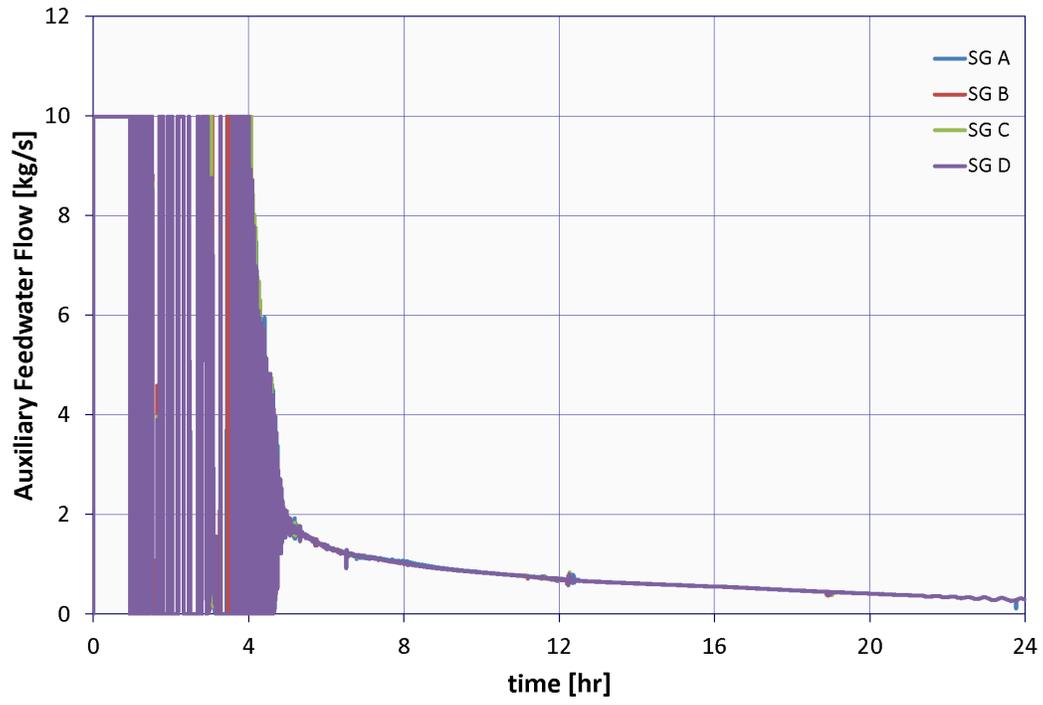
E.2.5 Case 5: 2-in. Break, 100 °F/hr Cooldown at 20 min, 0/2 CS Pumps, 4/4 RCFC

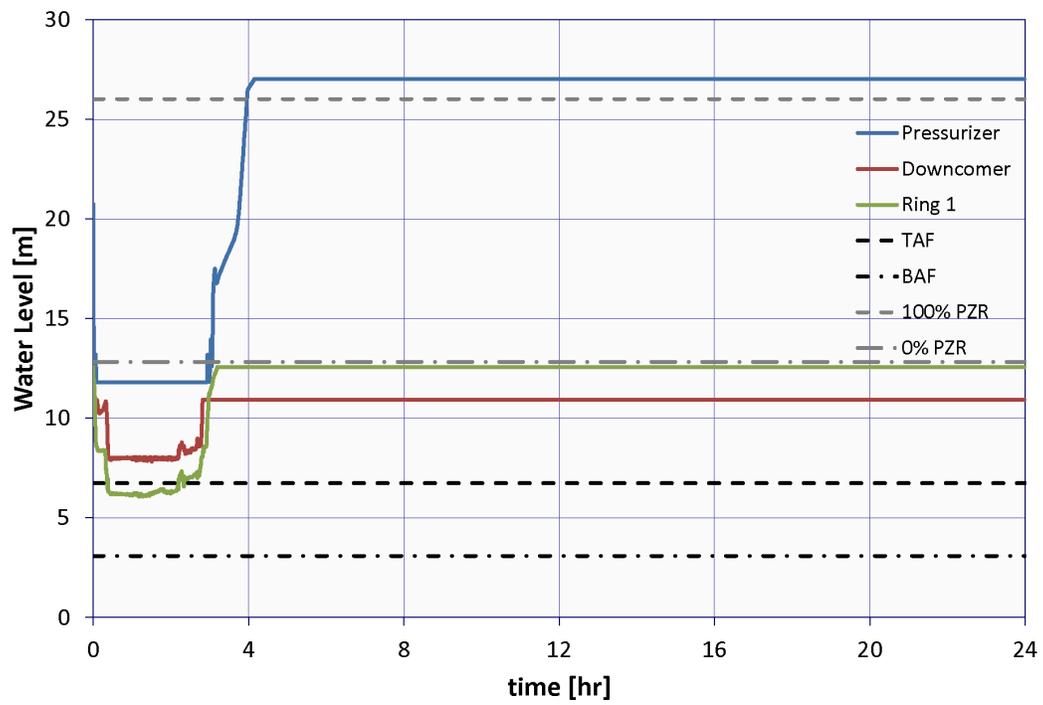
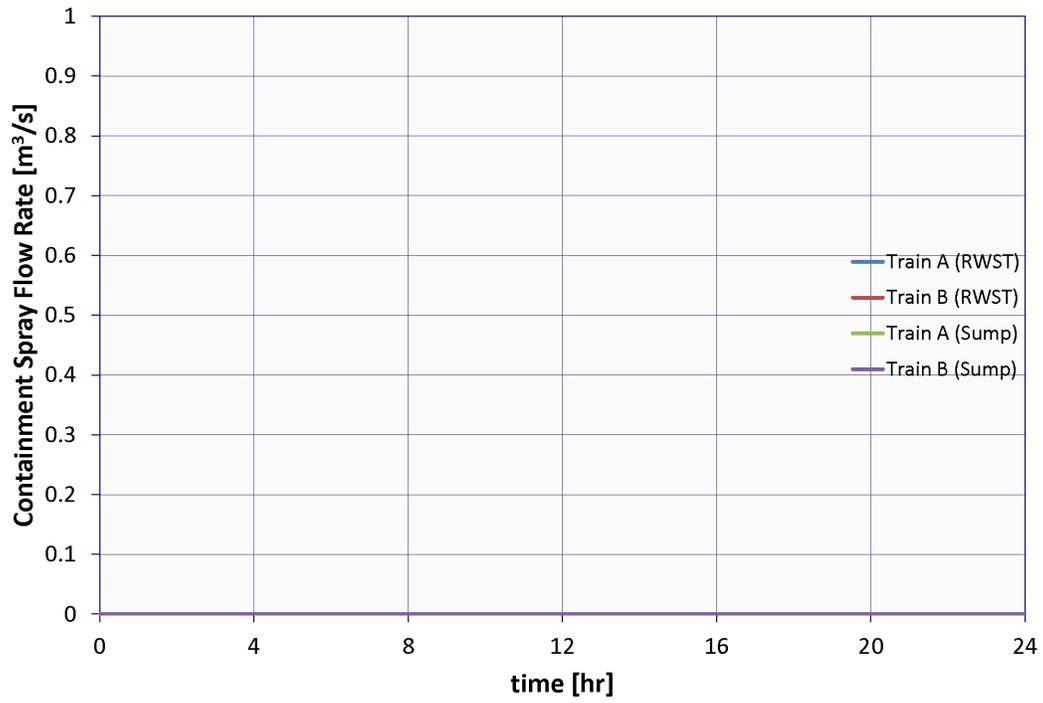


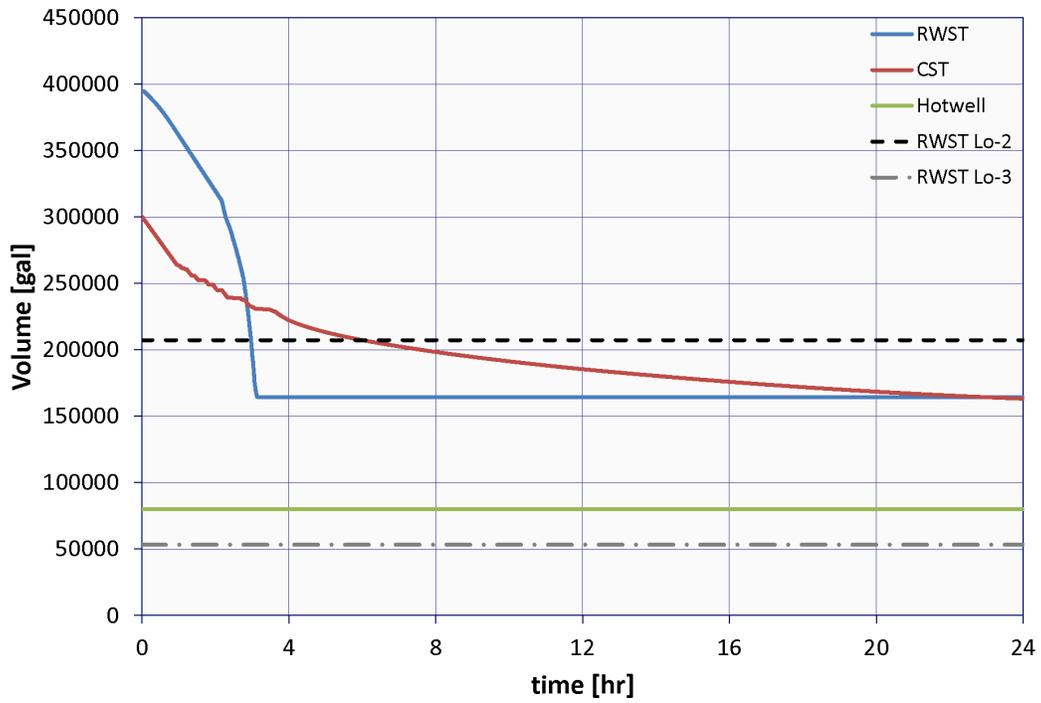
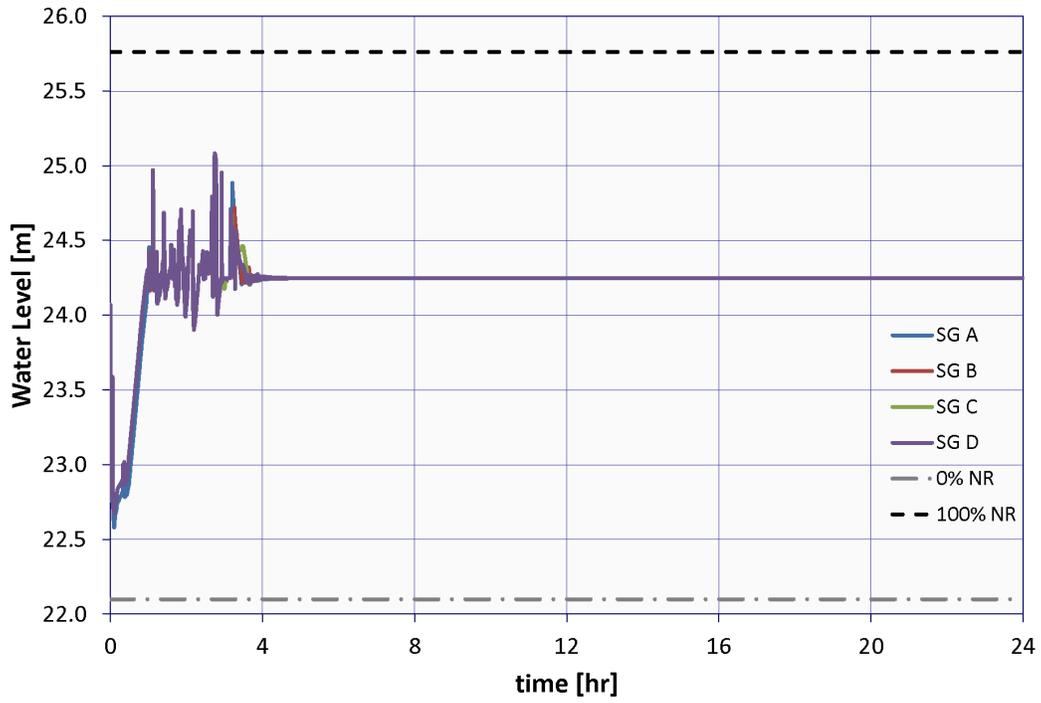




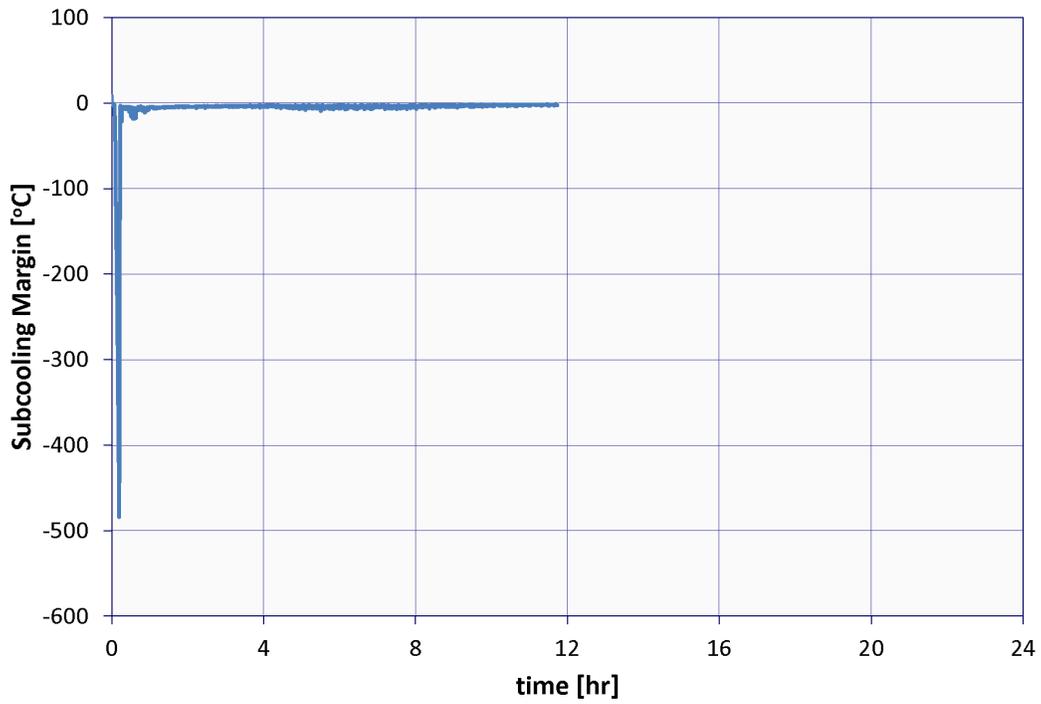
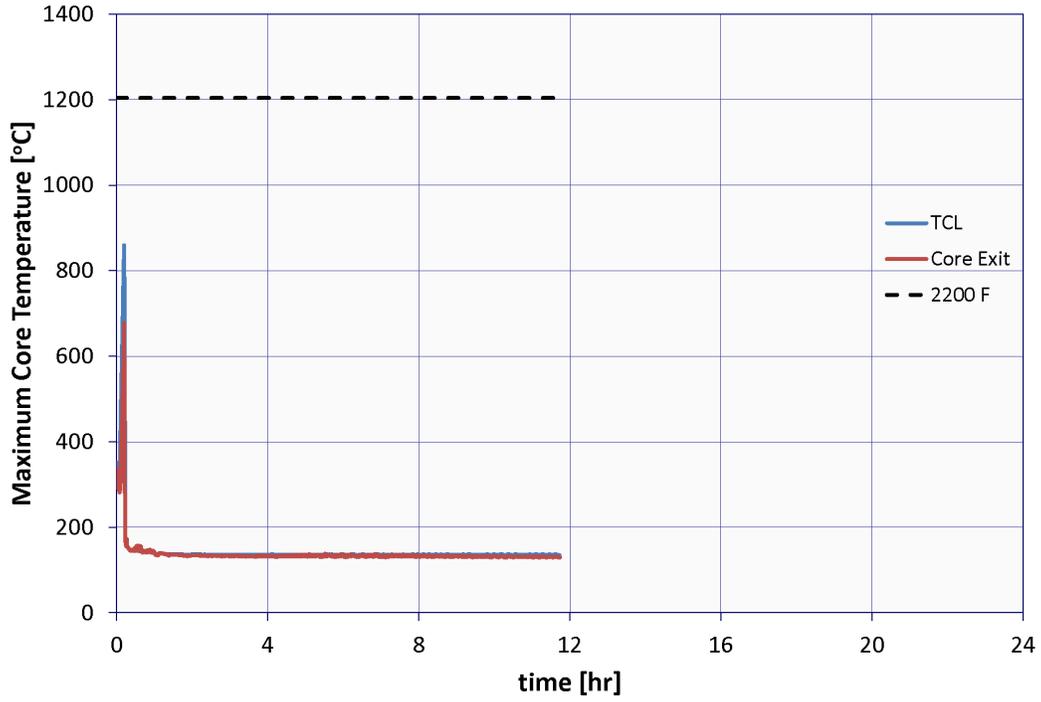


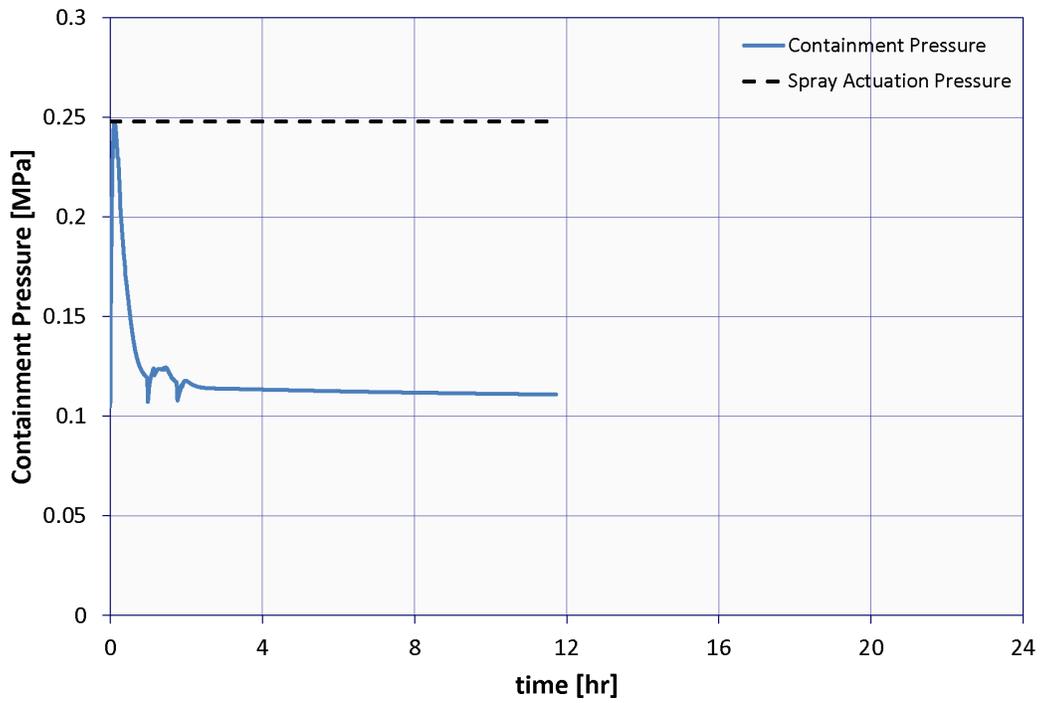
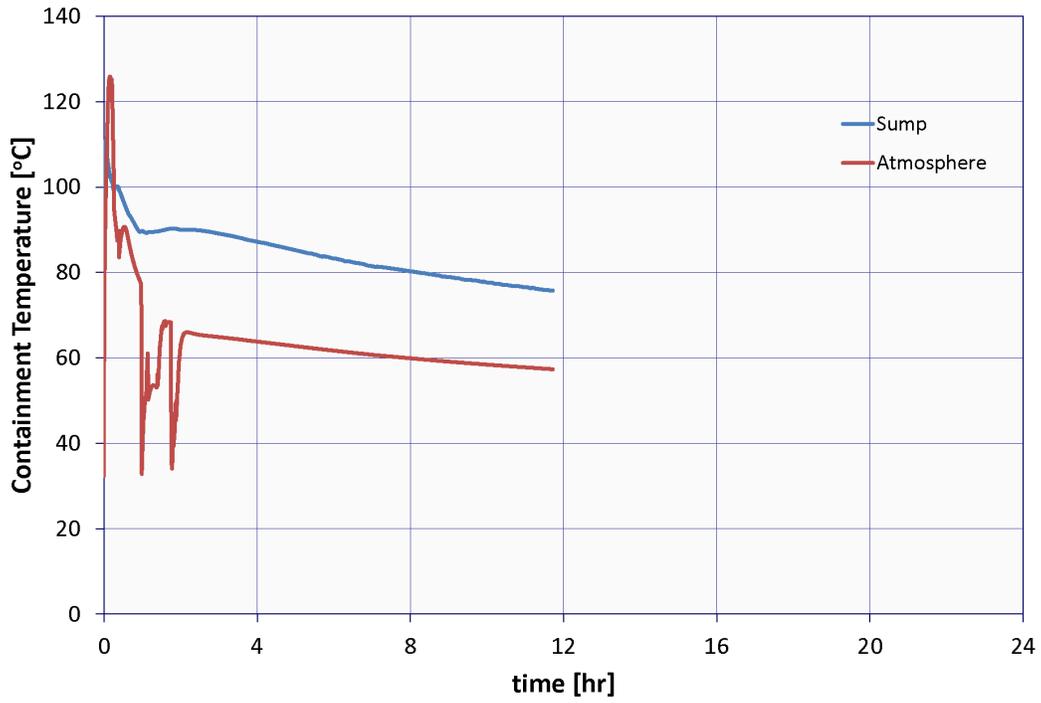


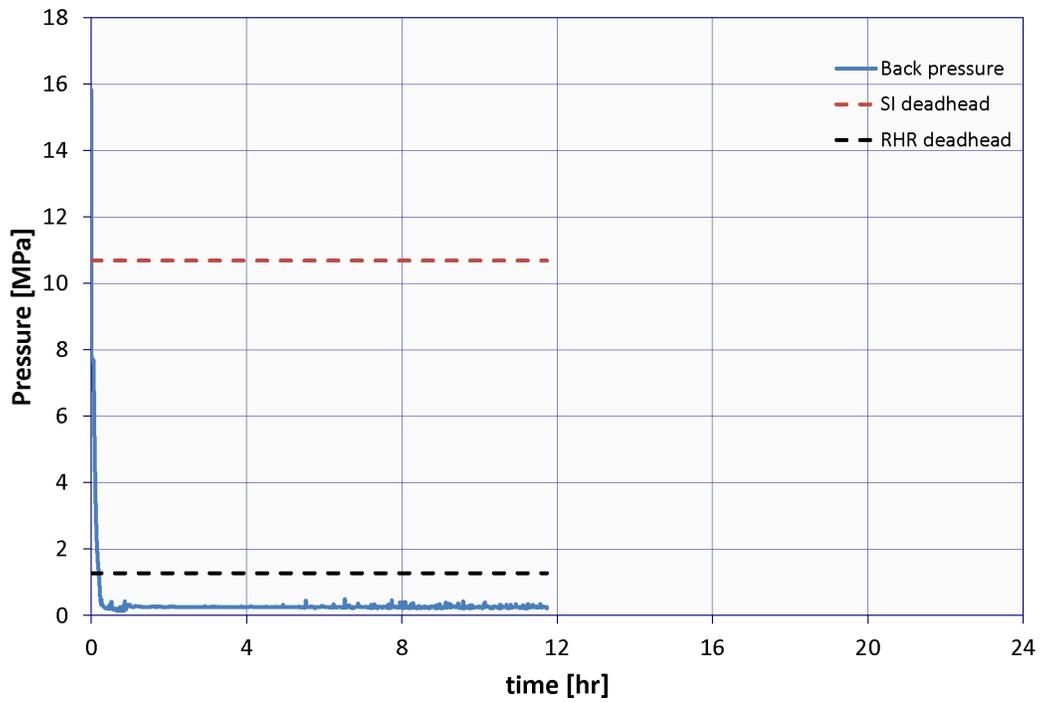
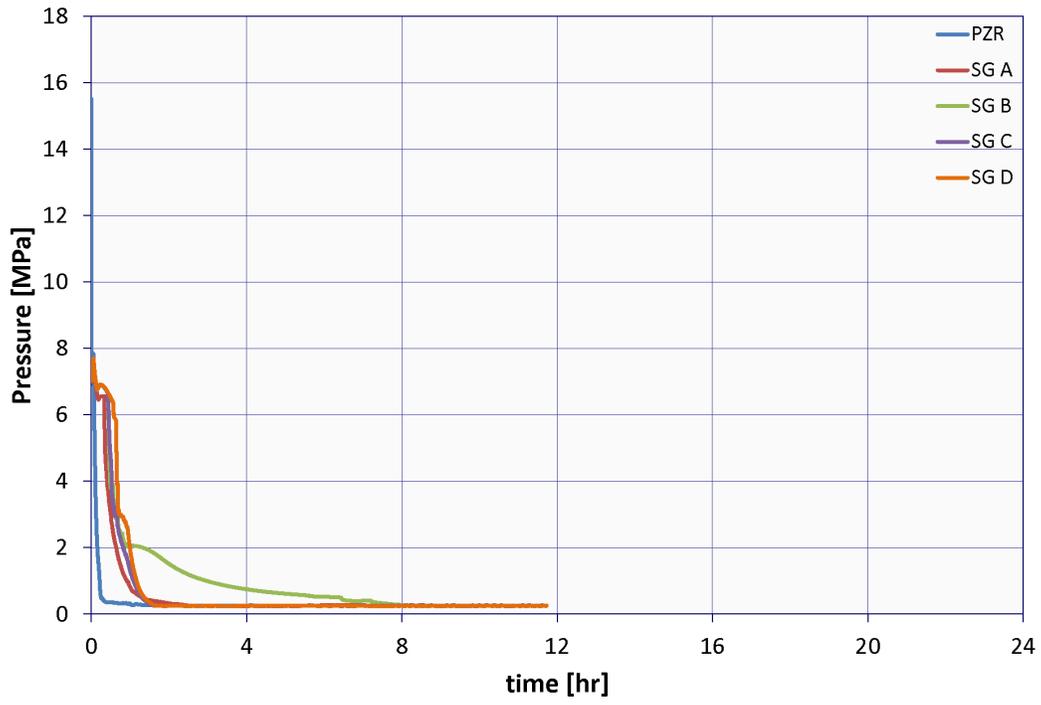


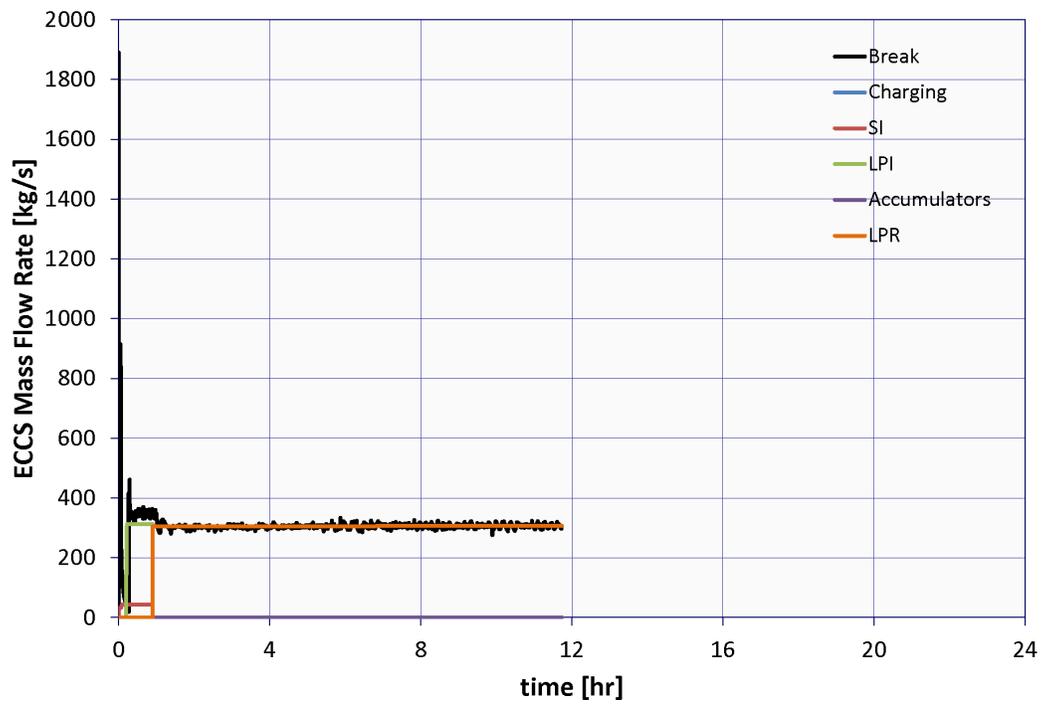
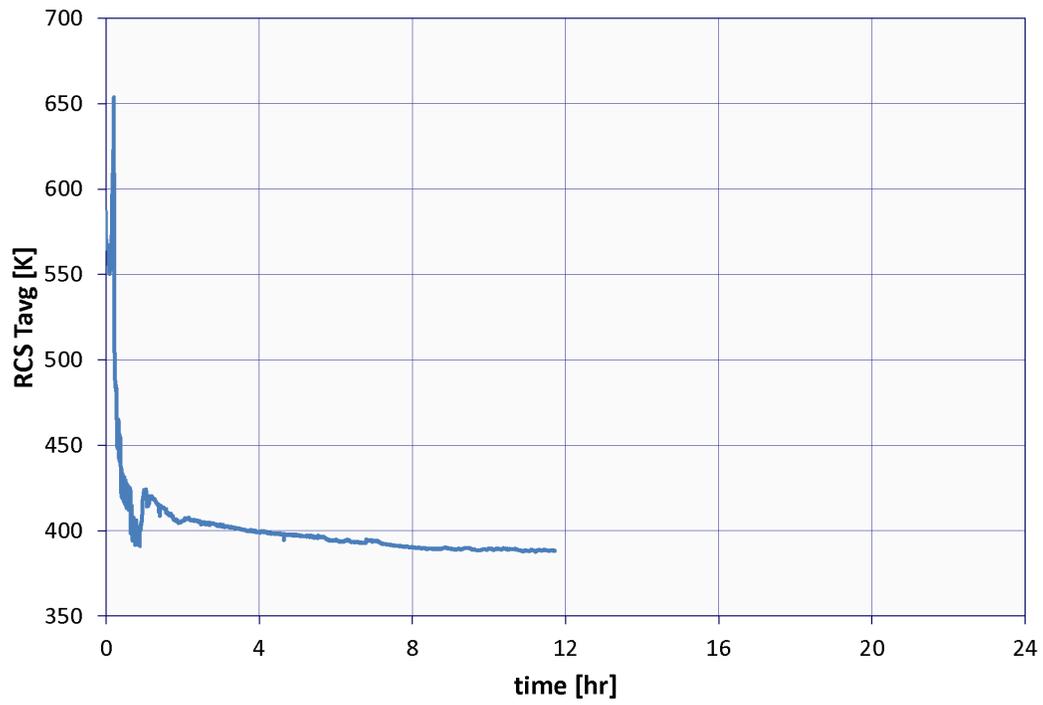


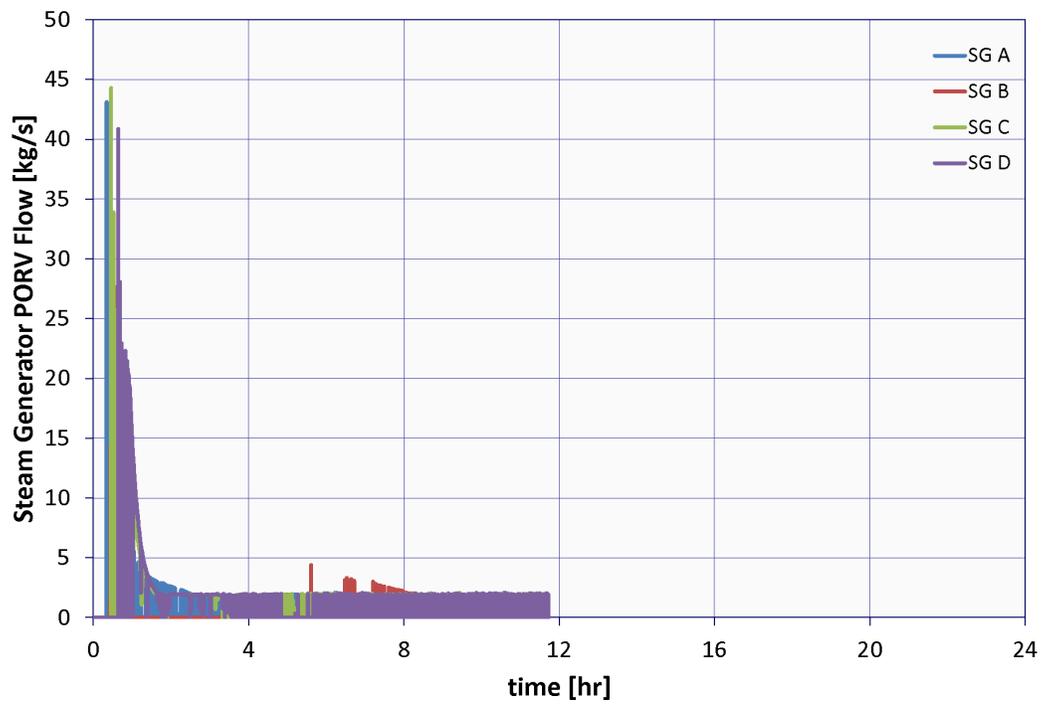
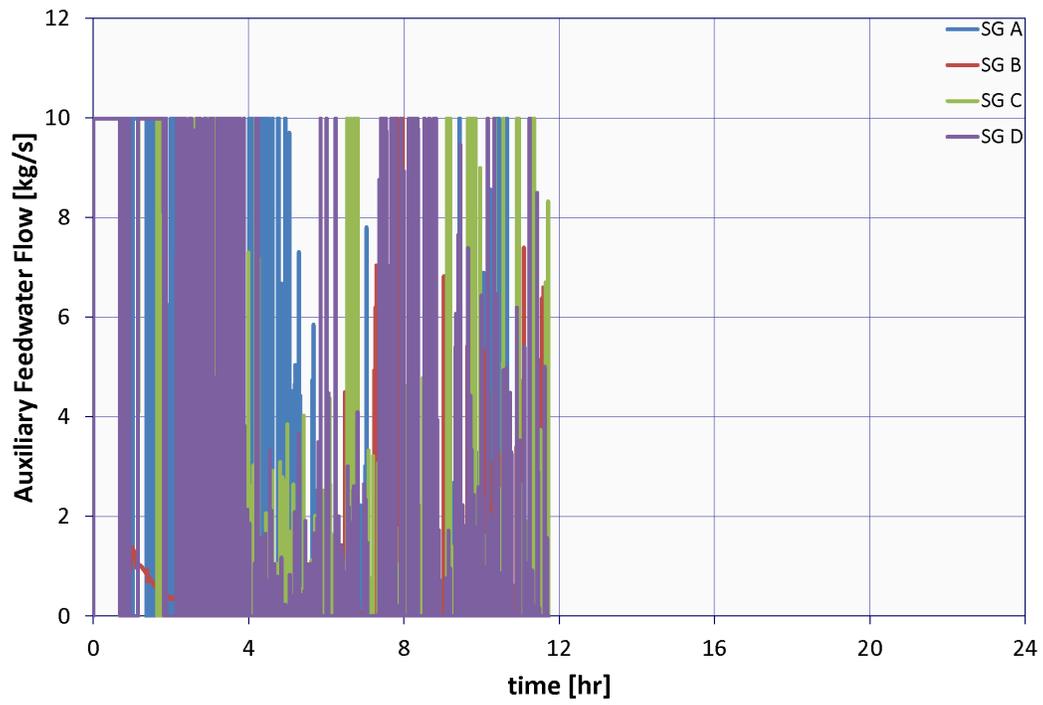
E.2.6 Case 6: 6-in. Break, 100 °F/hr Cooldown at 20 min, 0/2 CS Pumps, 4/4 RCFC

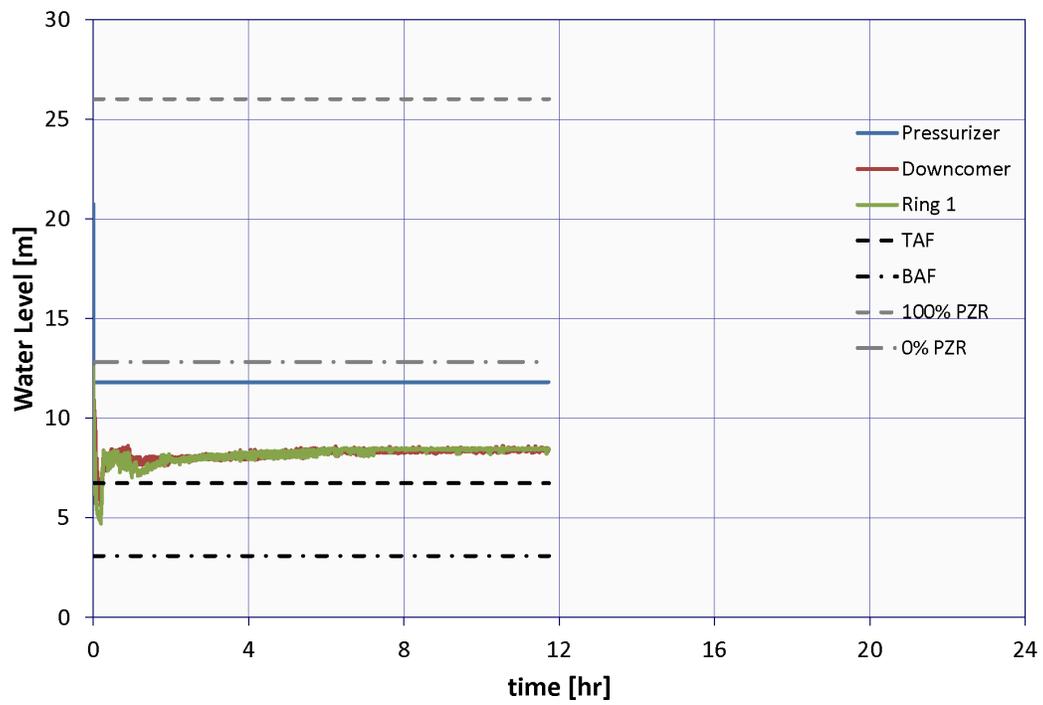
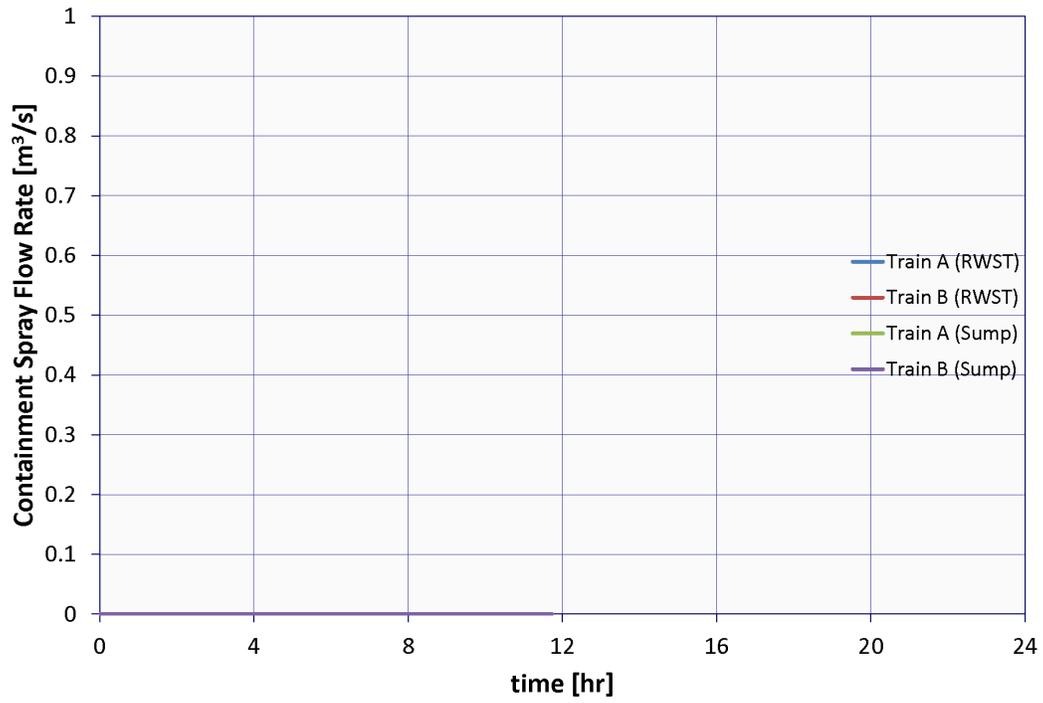


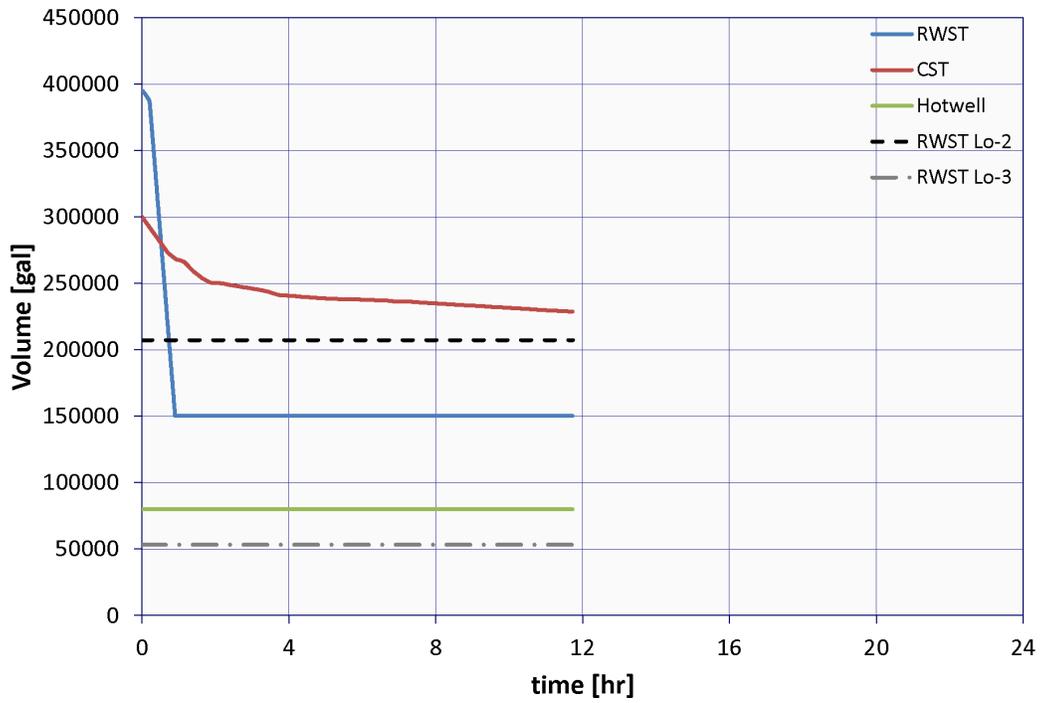
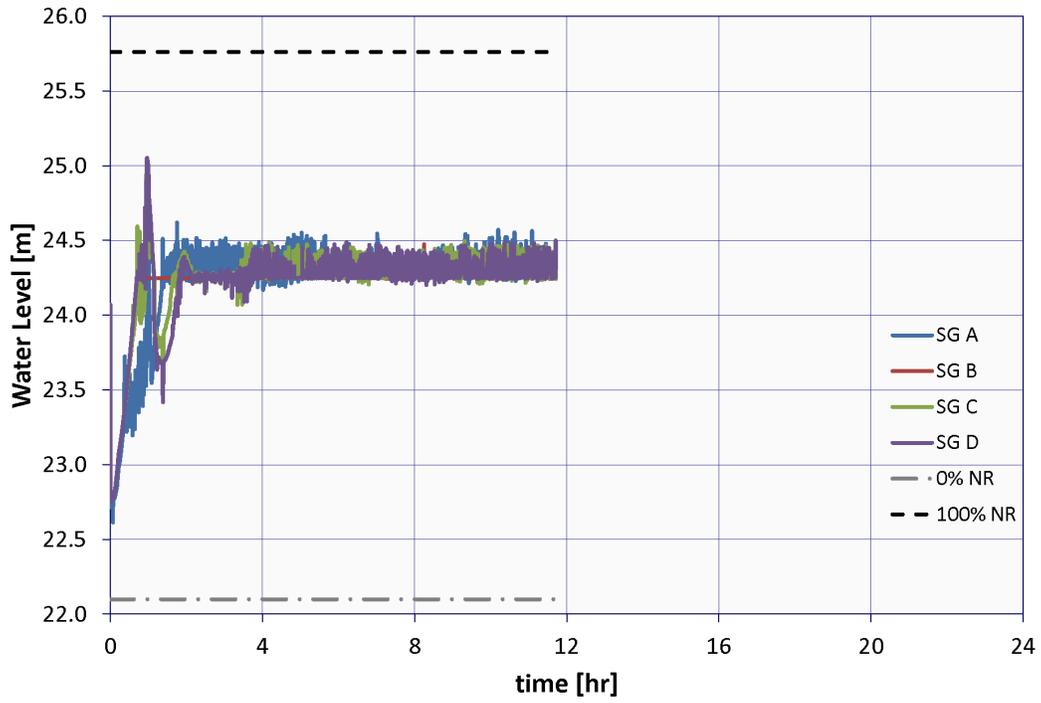




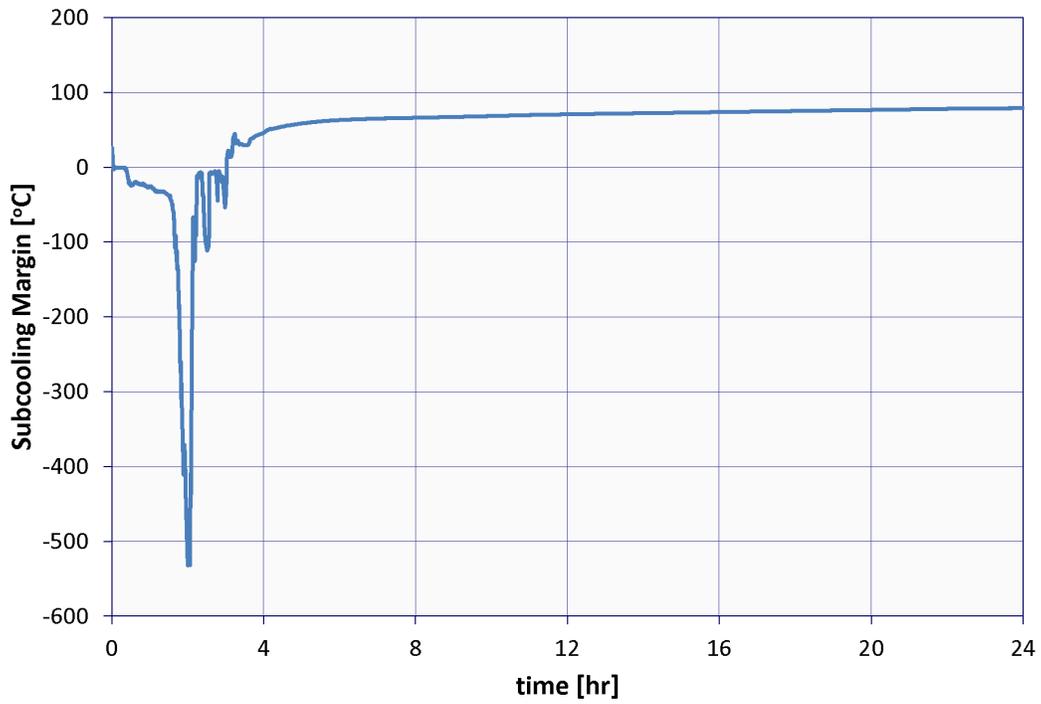
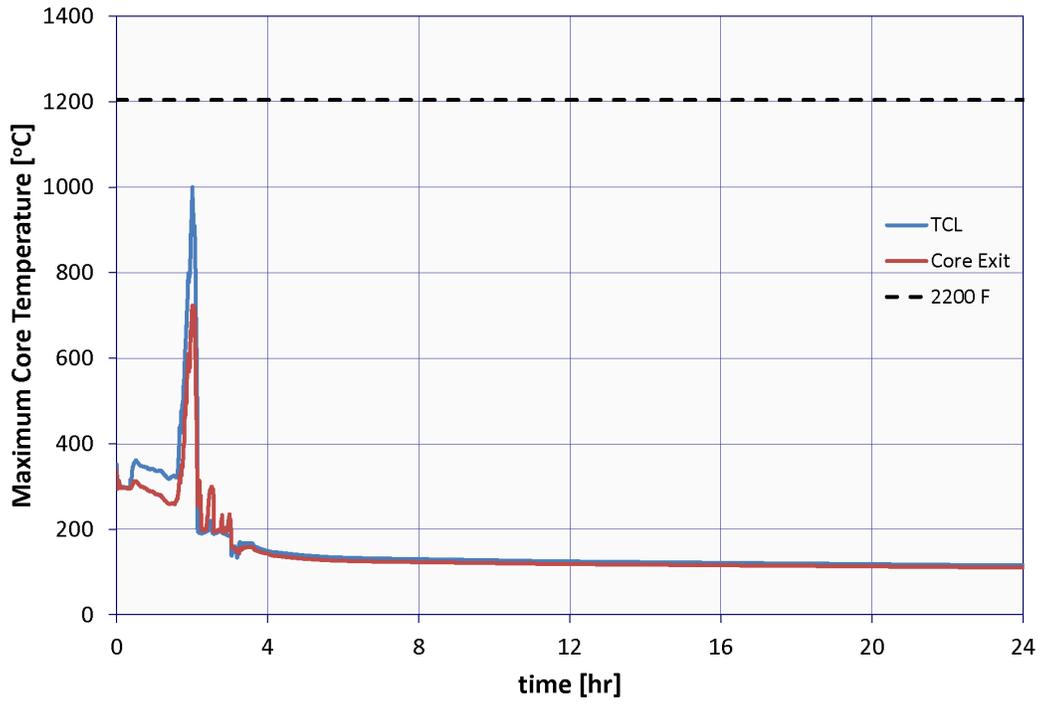


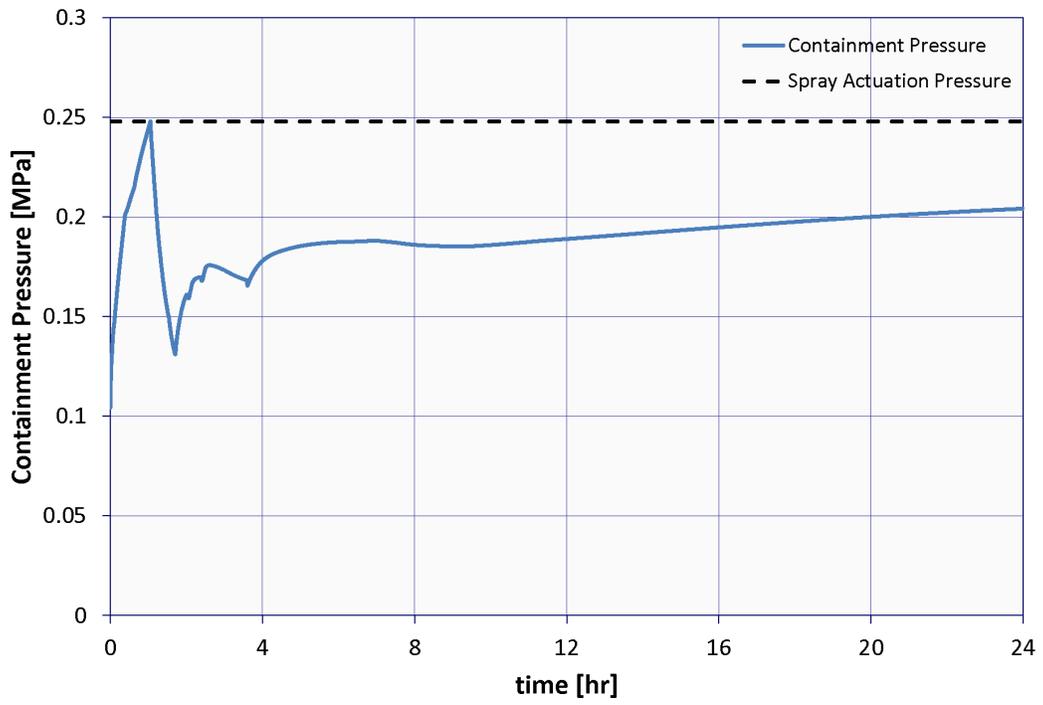
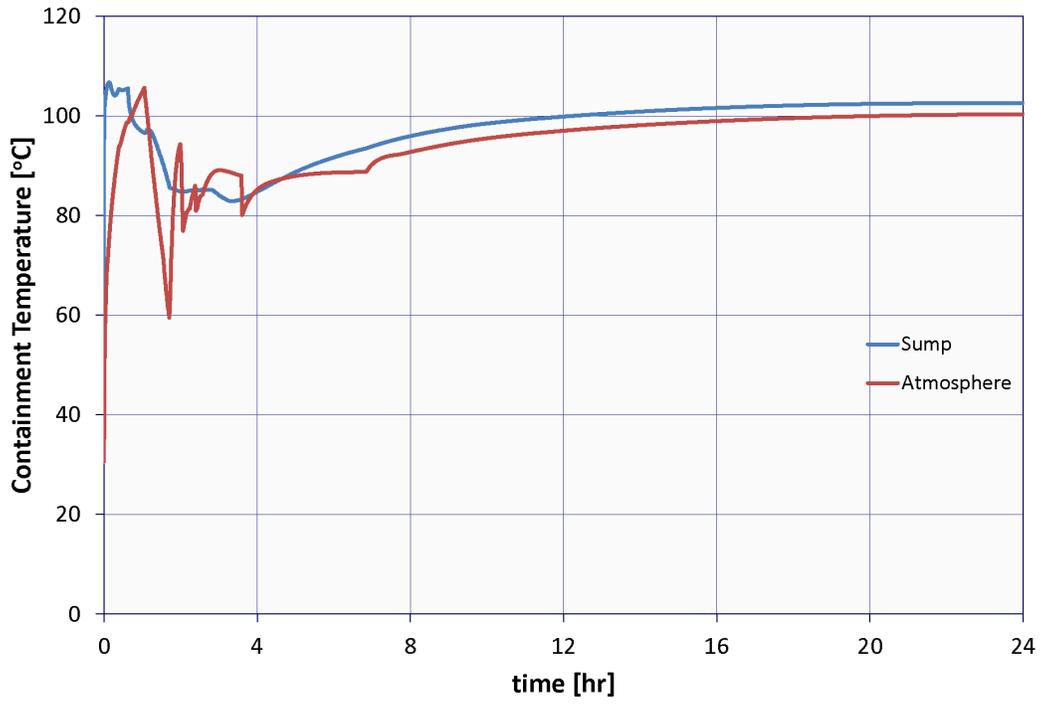


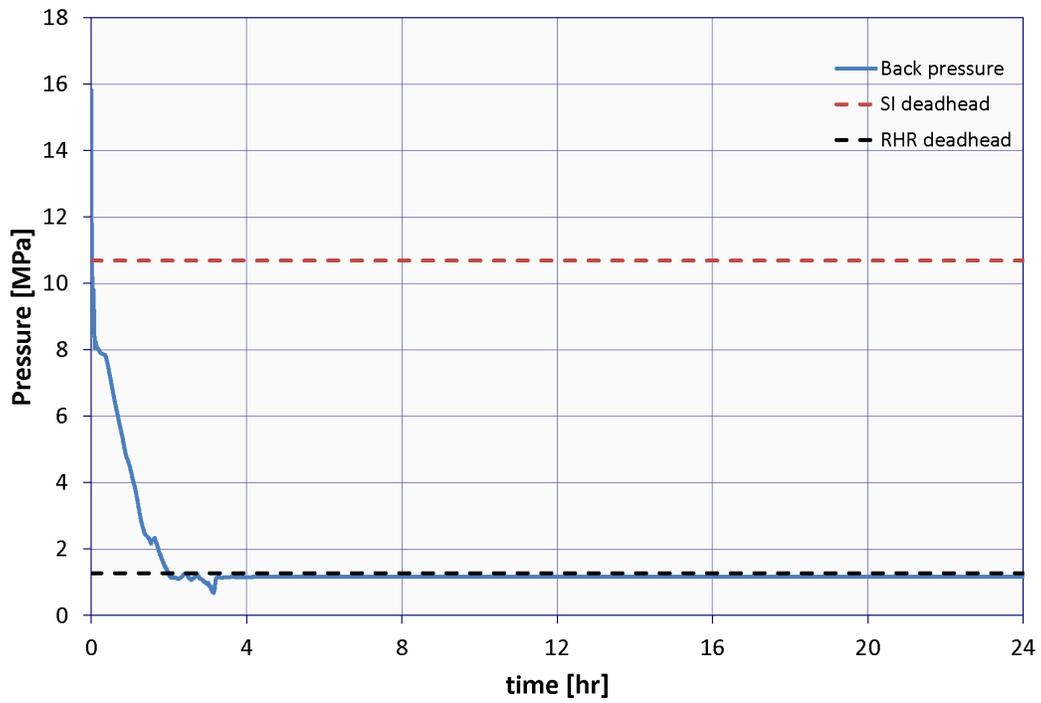
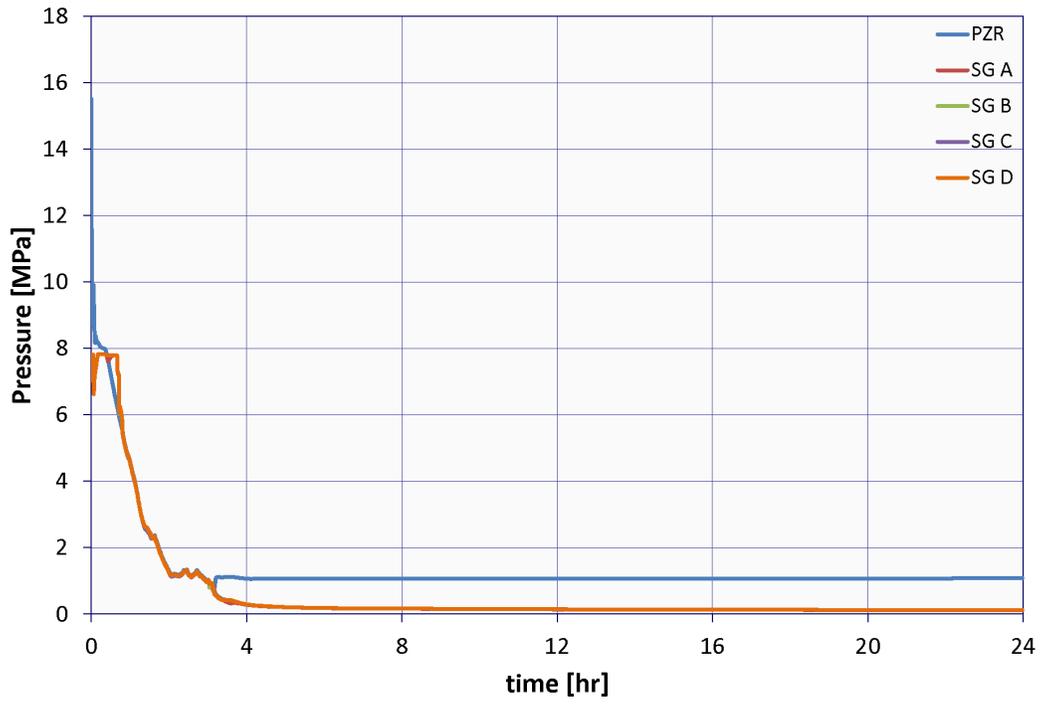


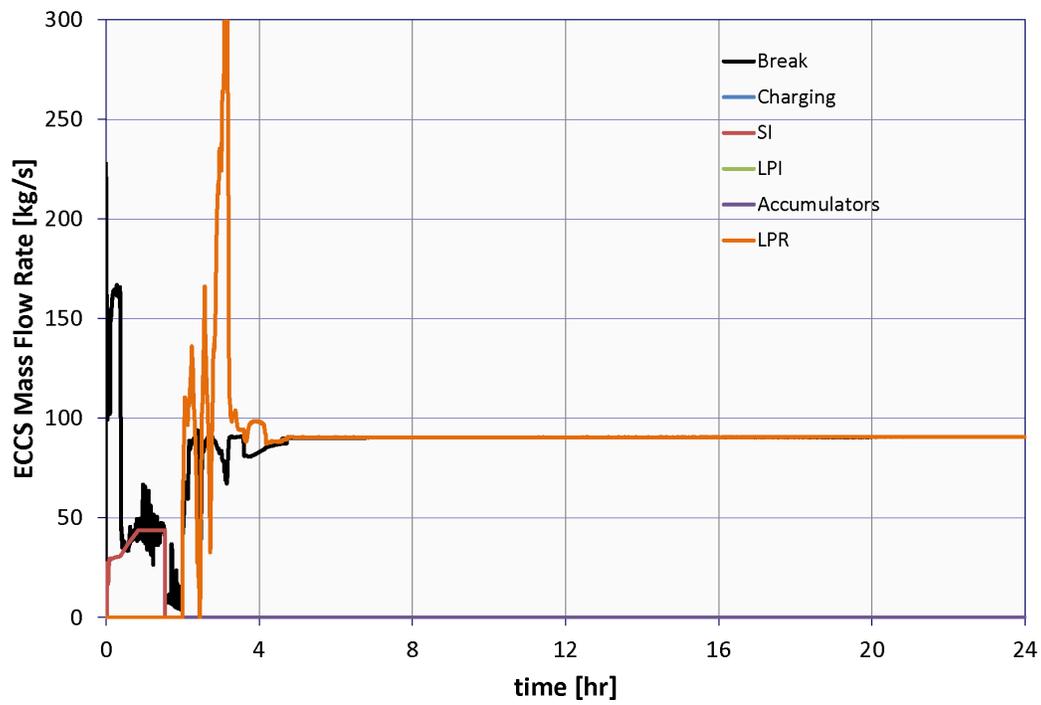
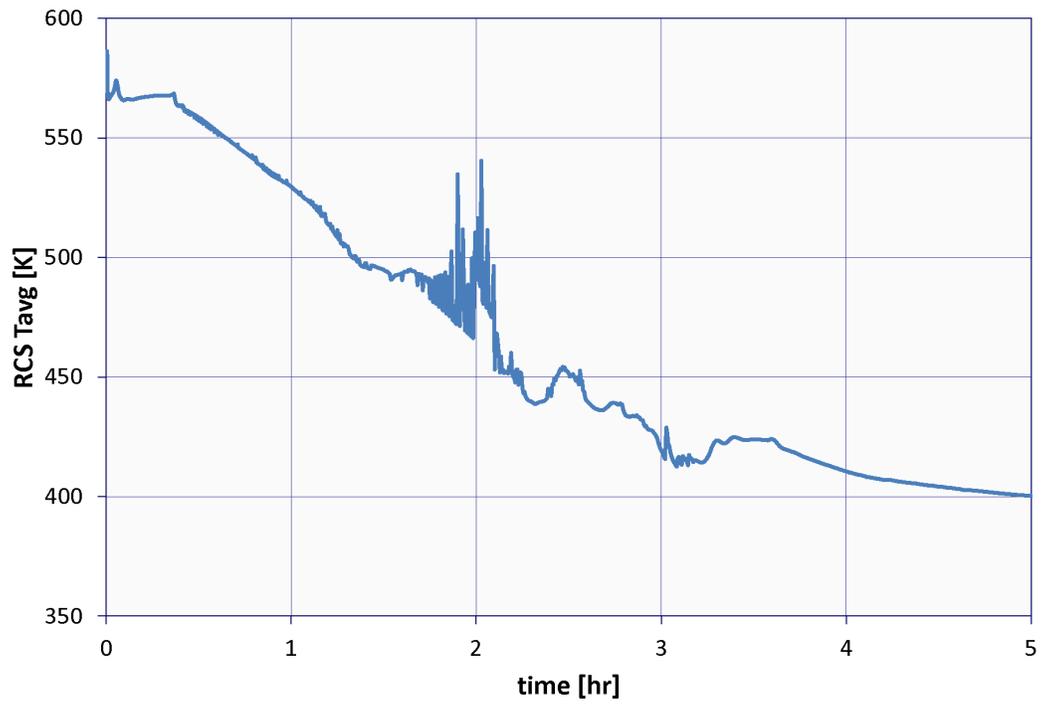


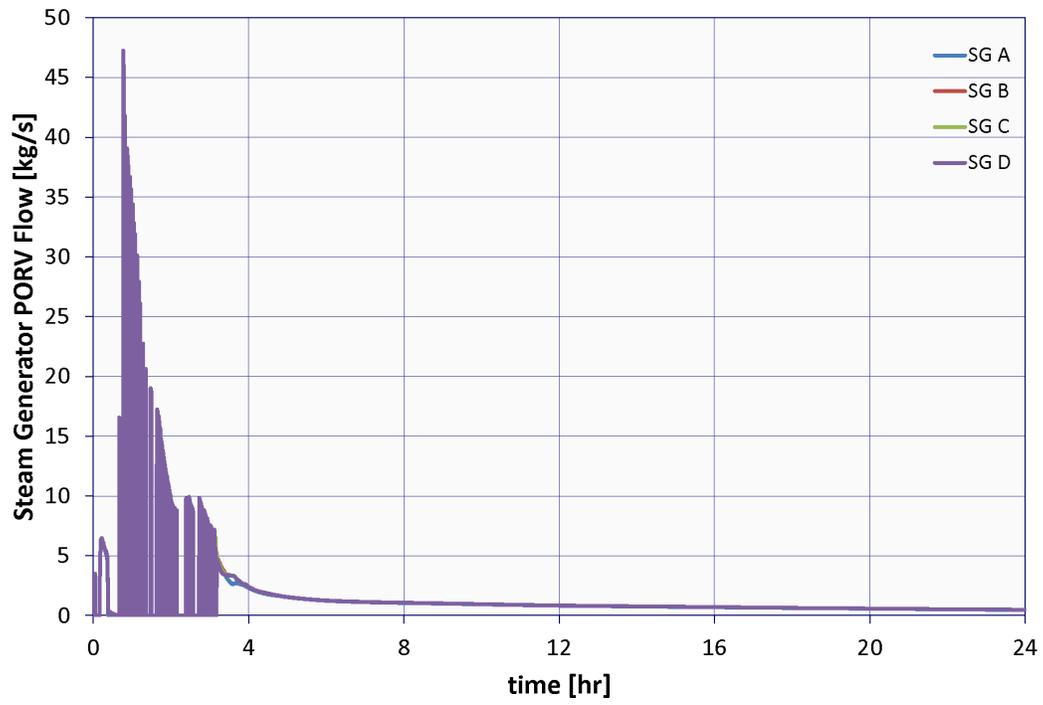
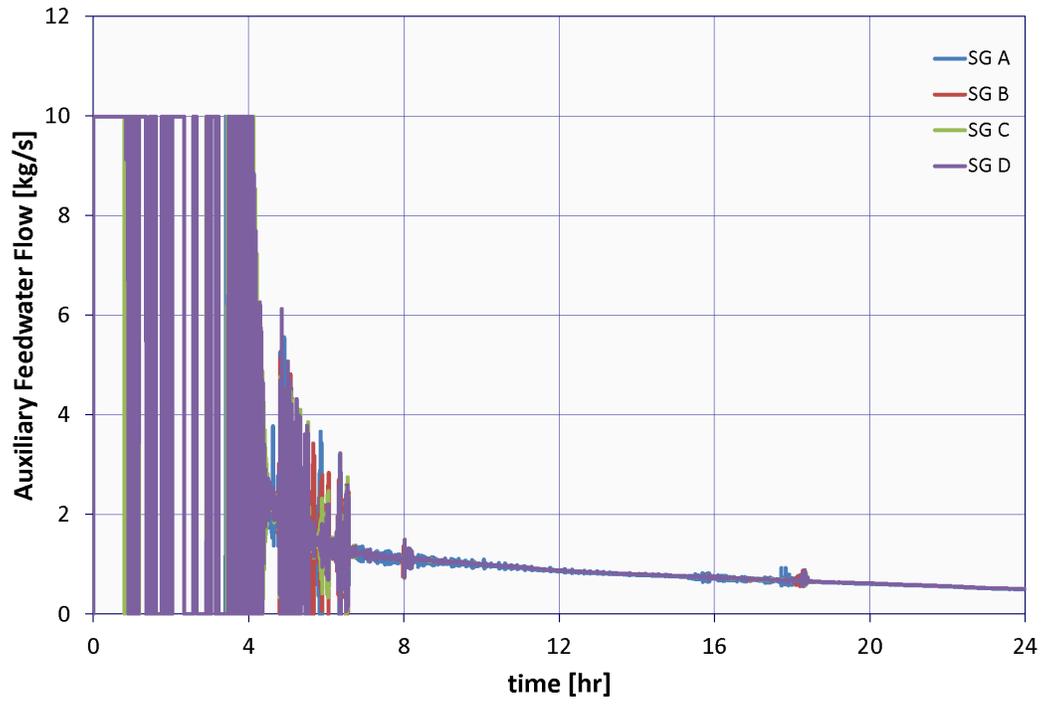
**E.2.7 Case 7: 2-in. Break, 100 °F/hr Cooldown at 40 min, 2/2 CS Pumps, 0/4 RCFC**

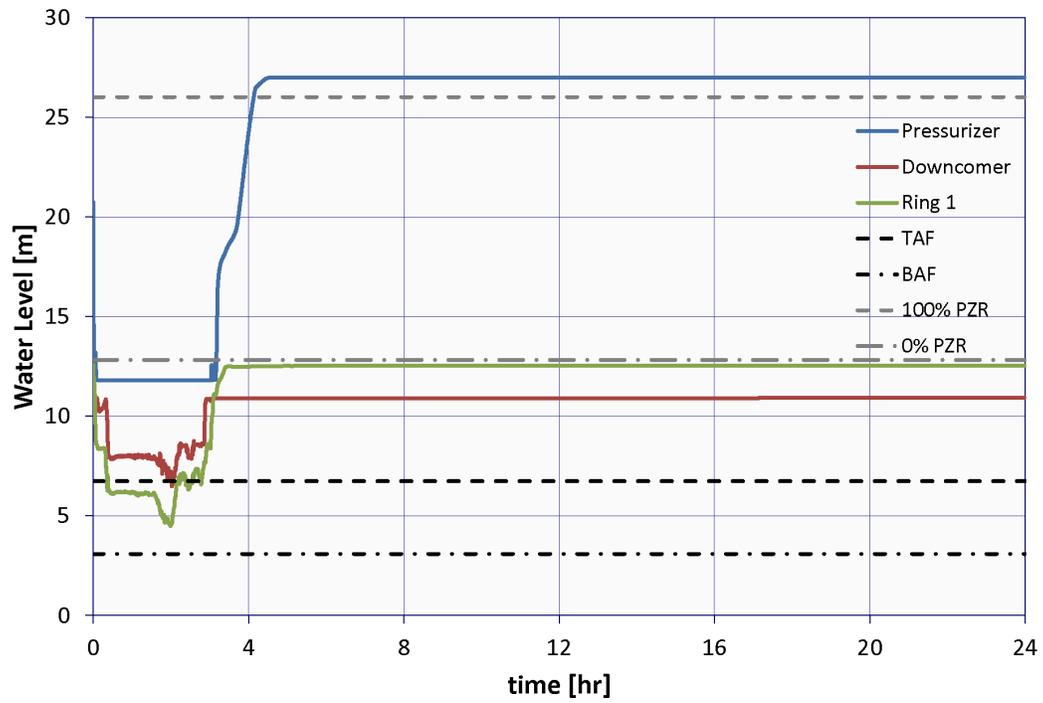
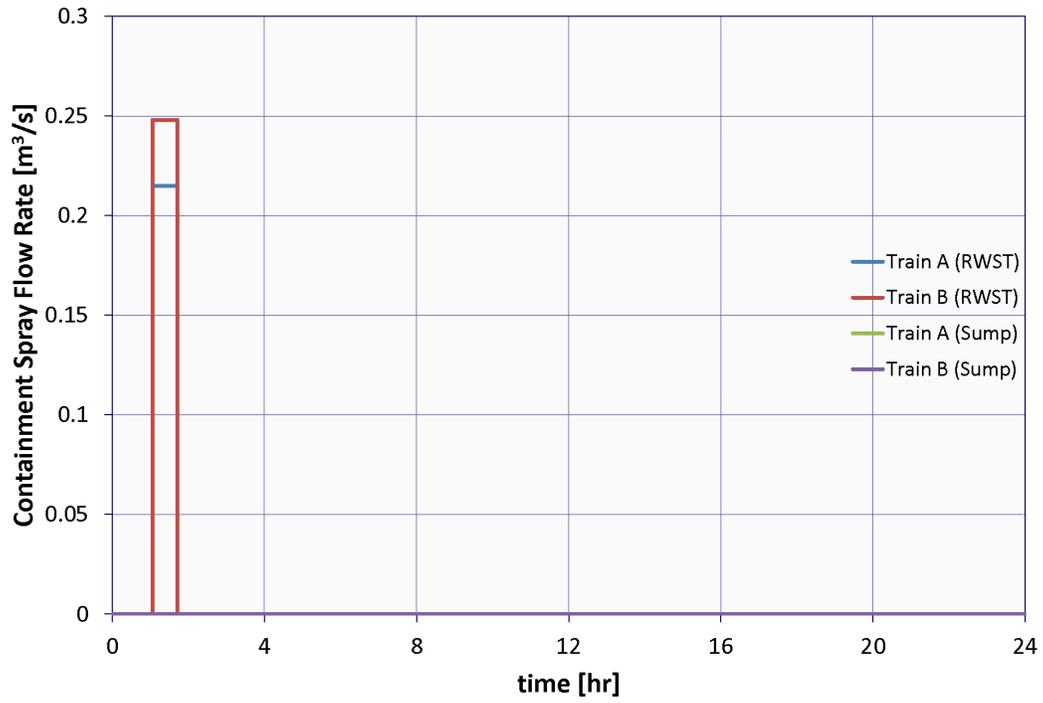


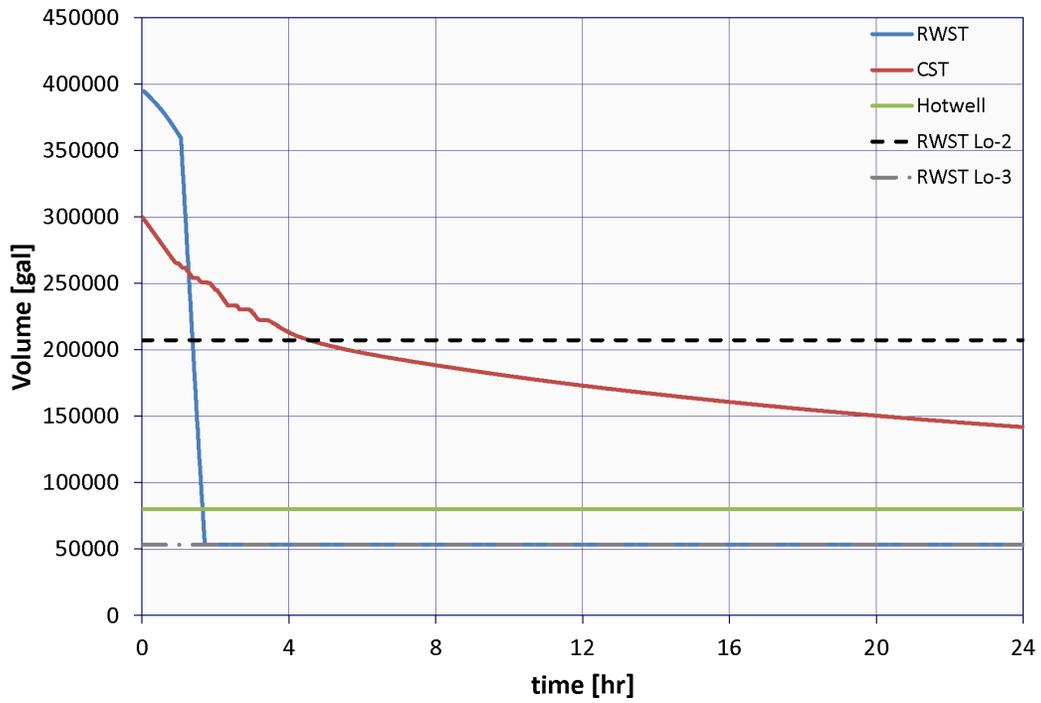
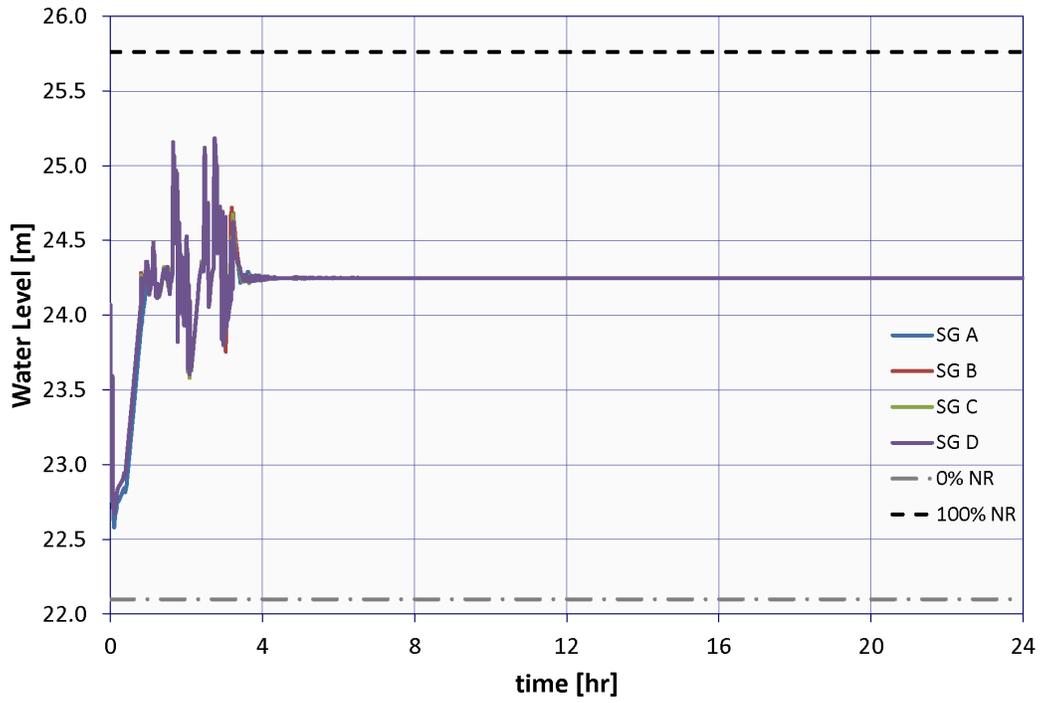




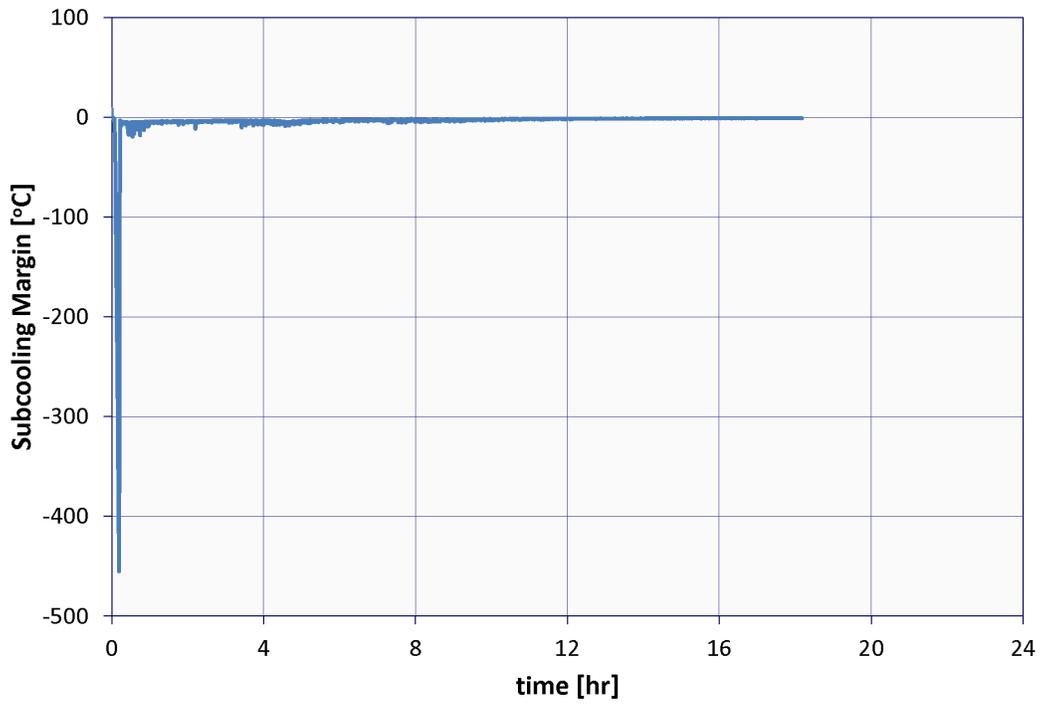
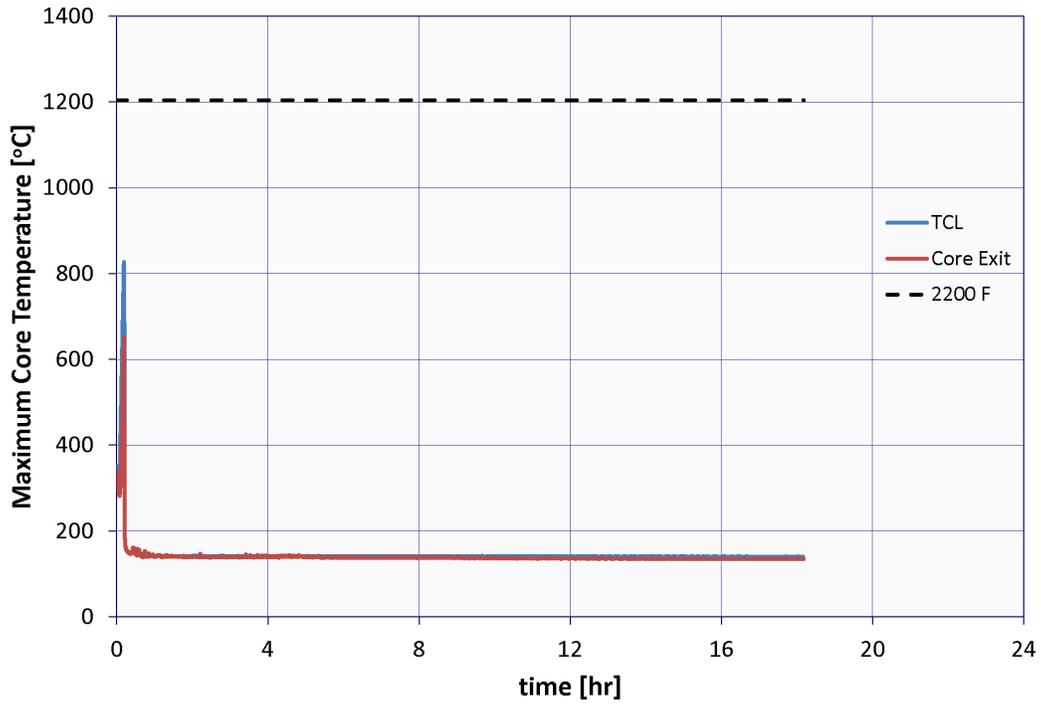


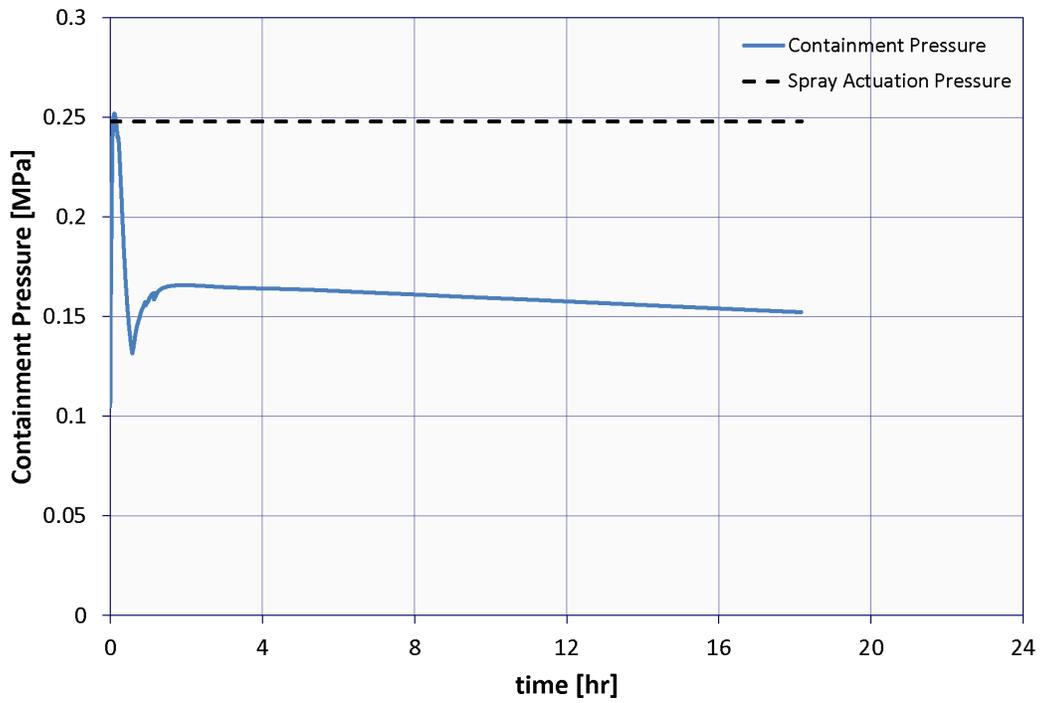
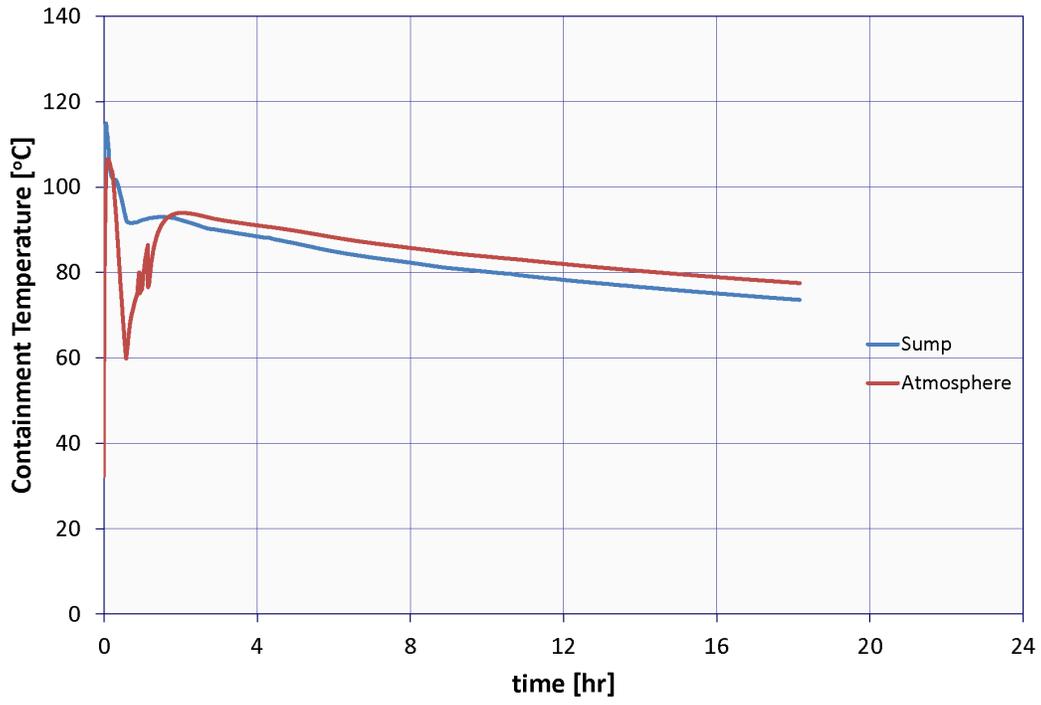


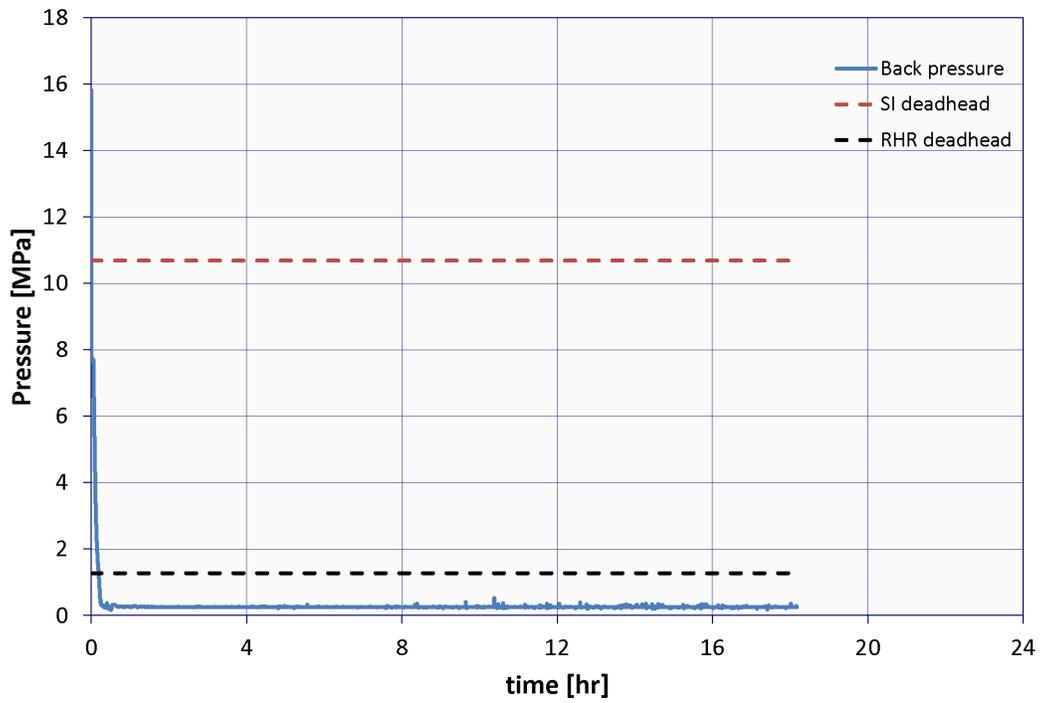
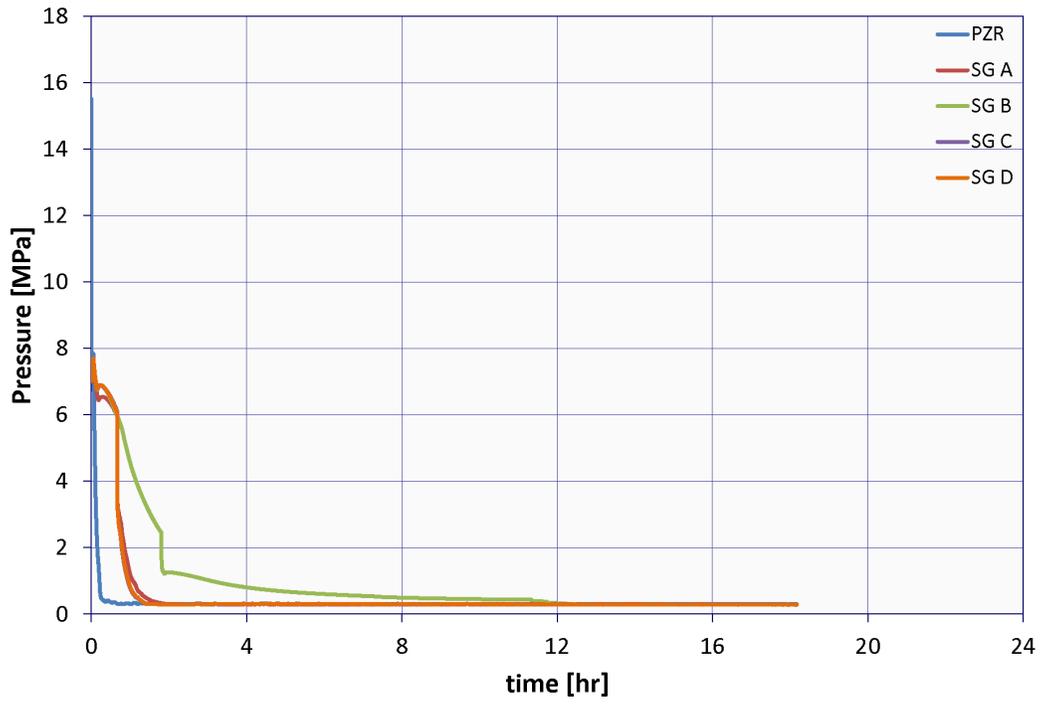


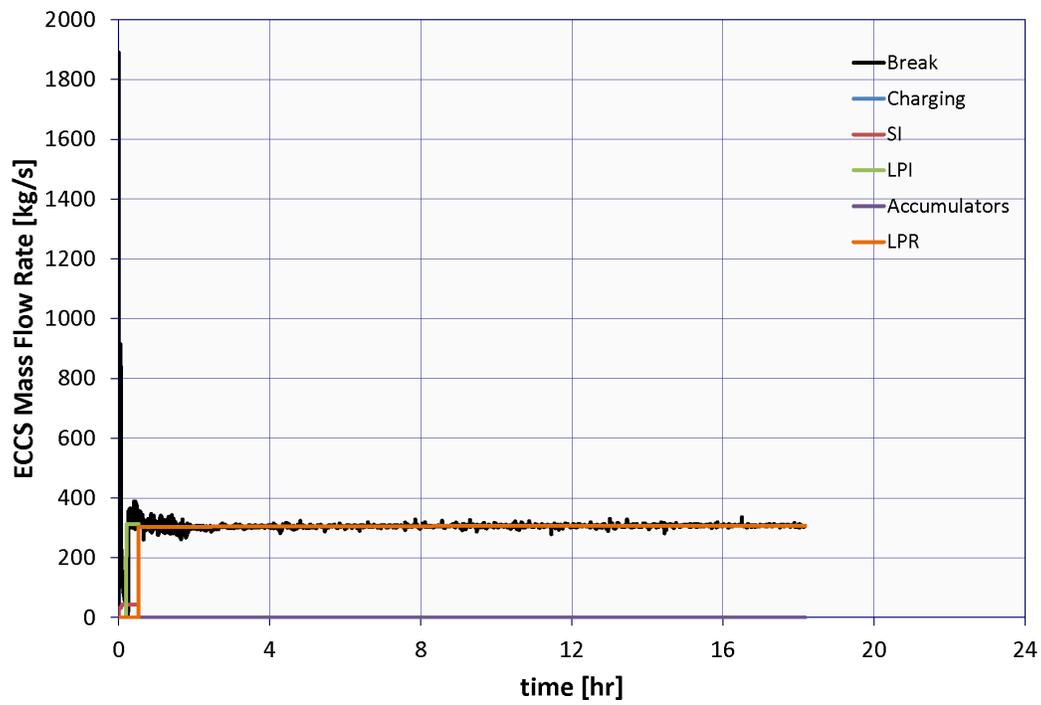
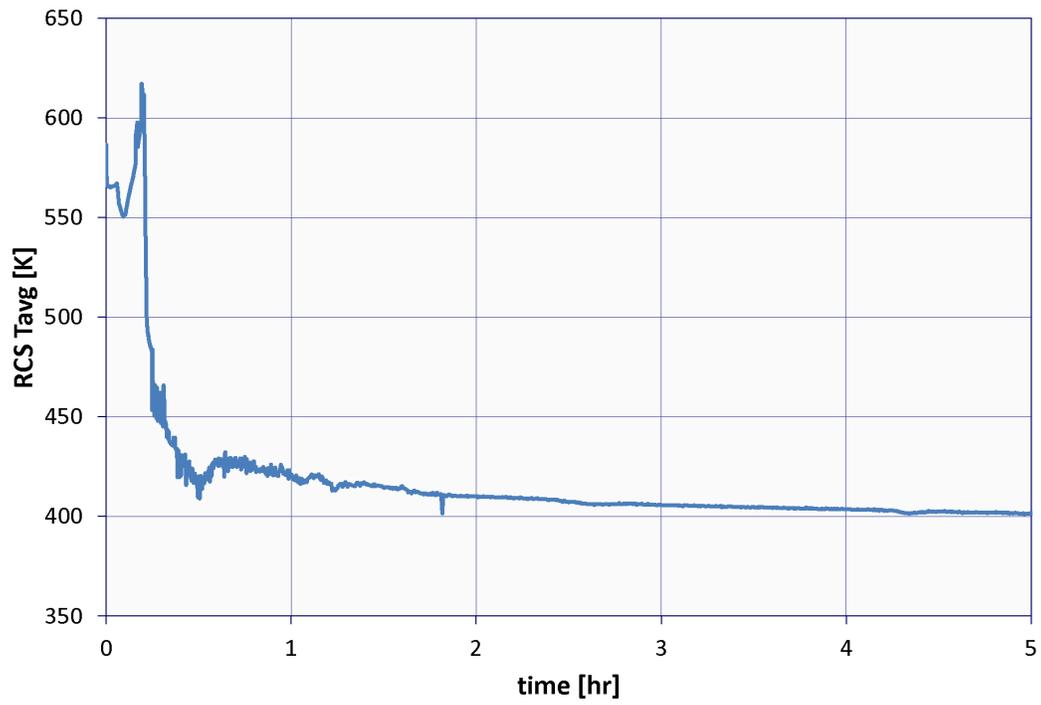


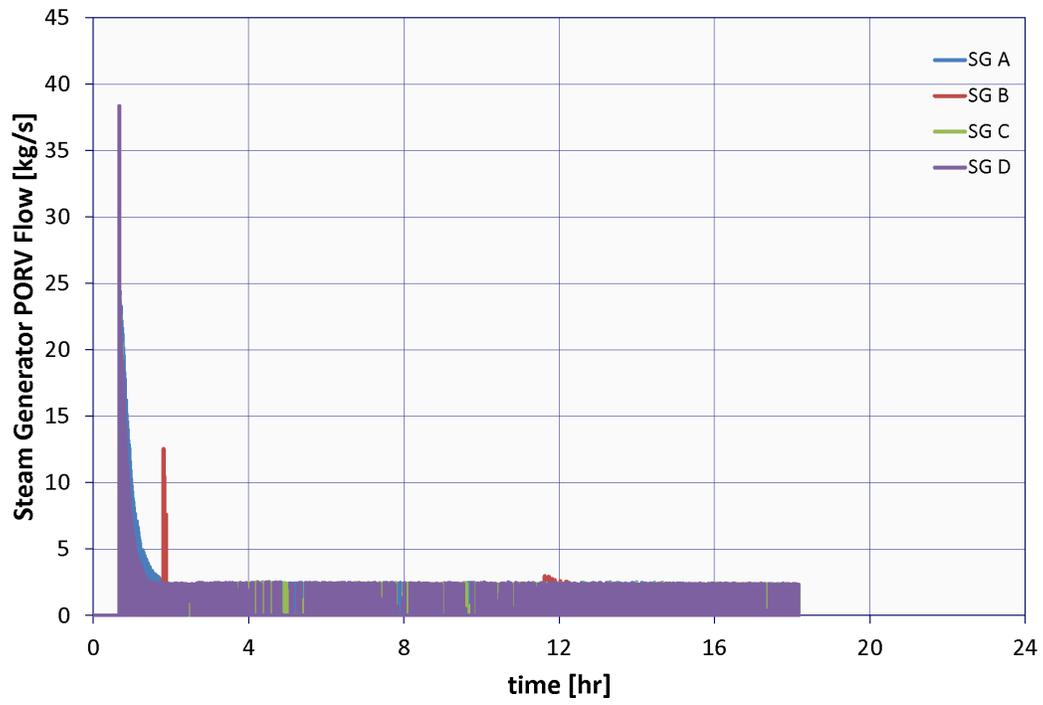
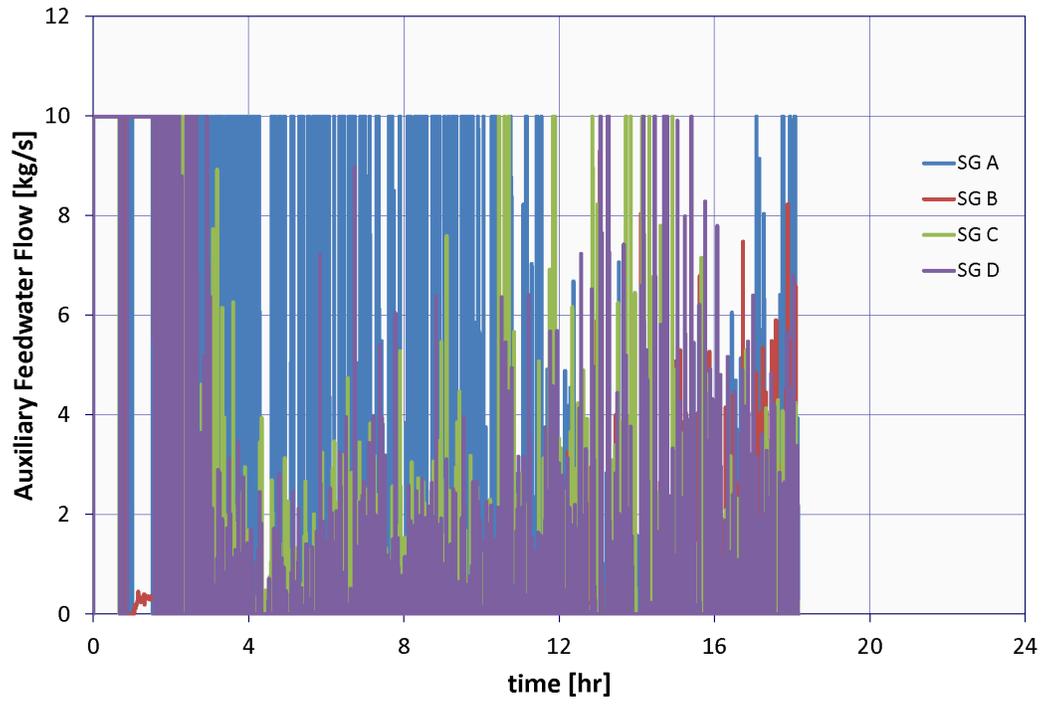
**E.2.8 Case 8: 6-in. Break, 100 °F/hr Cooldown at 40 min, 2/2 CS Pumps, 0/4 RCFC**

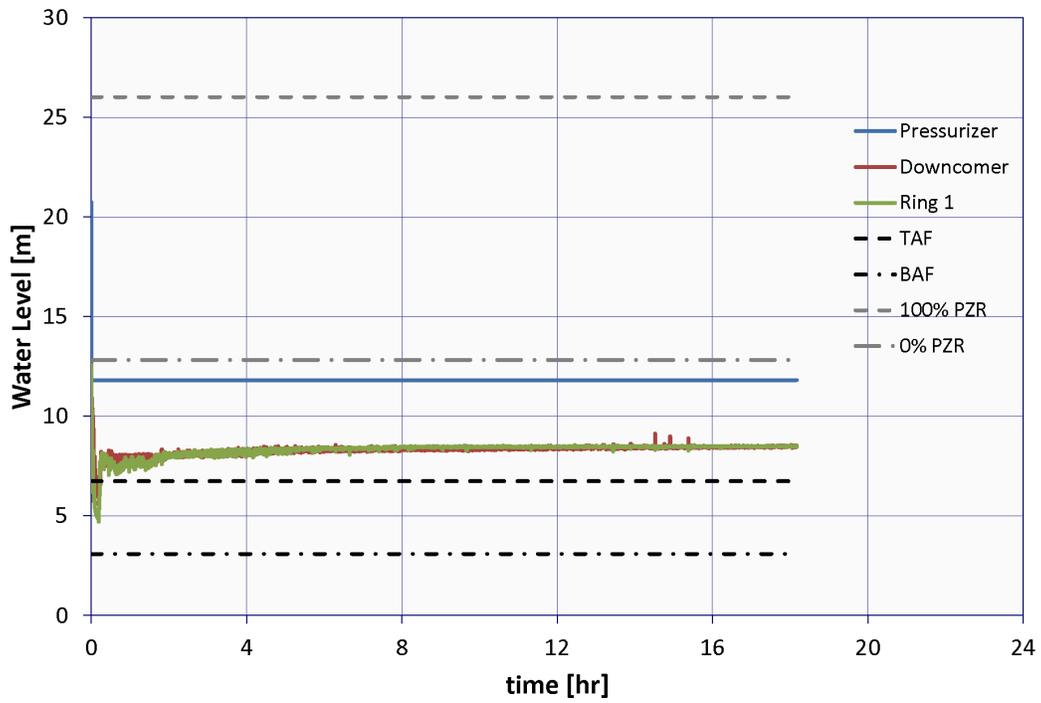
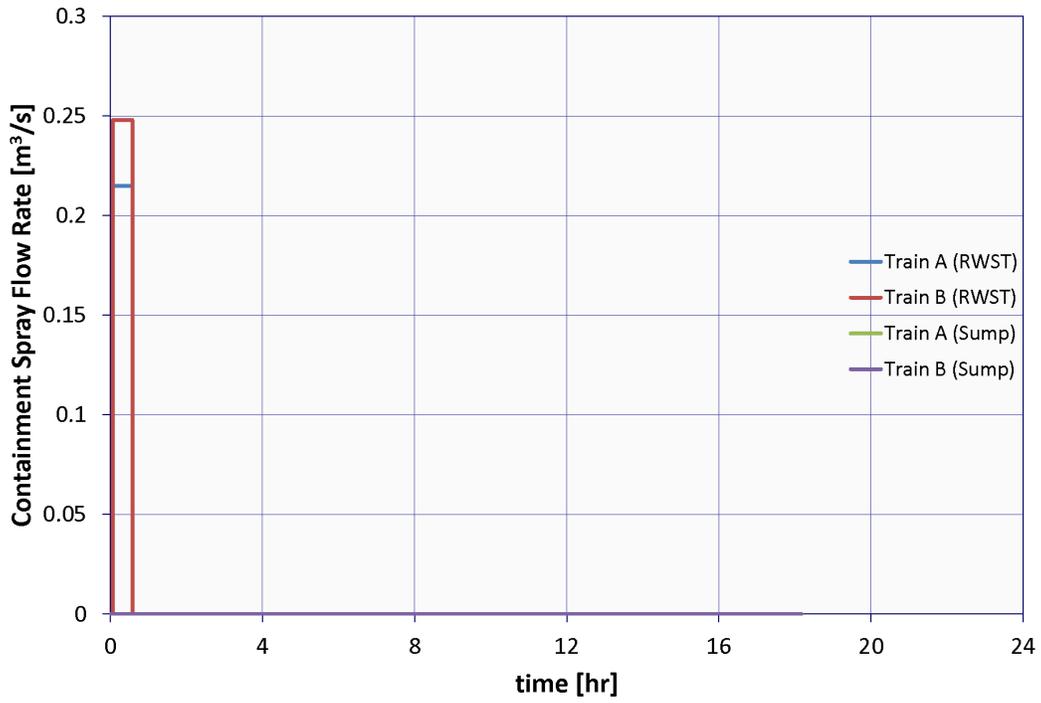


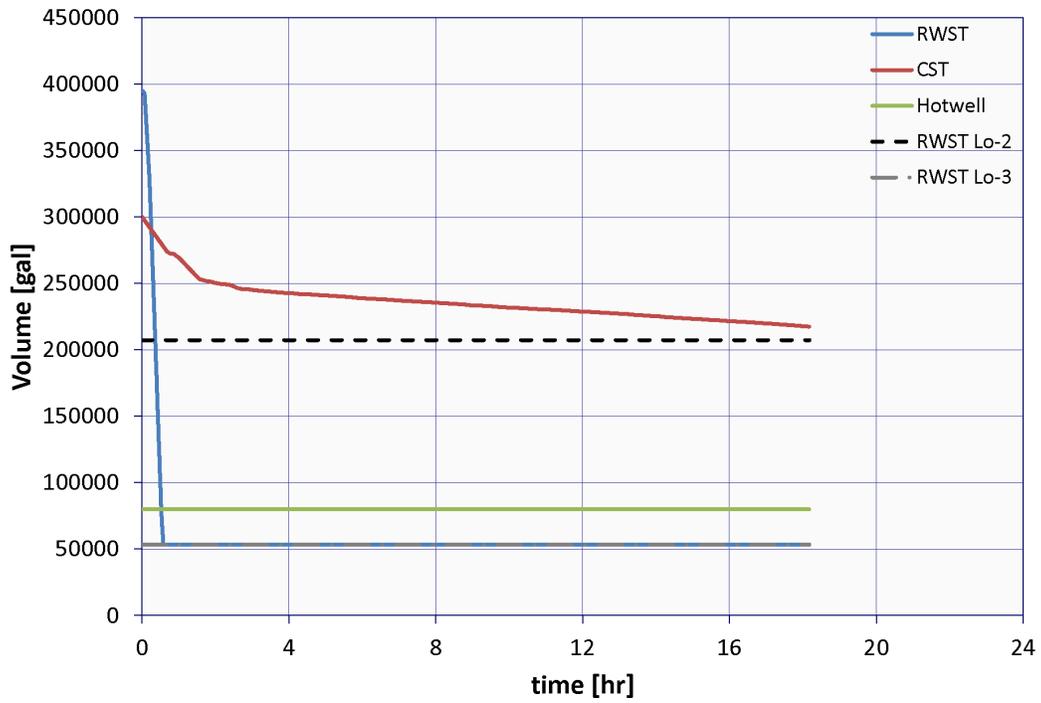
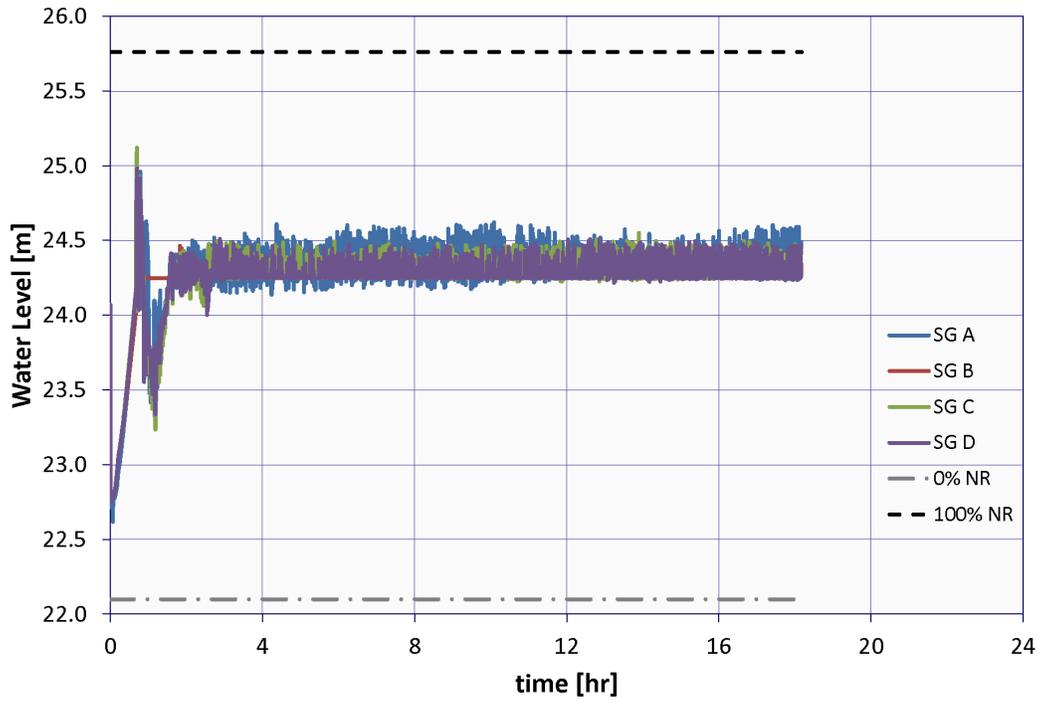




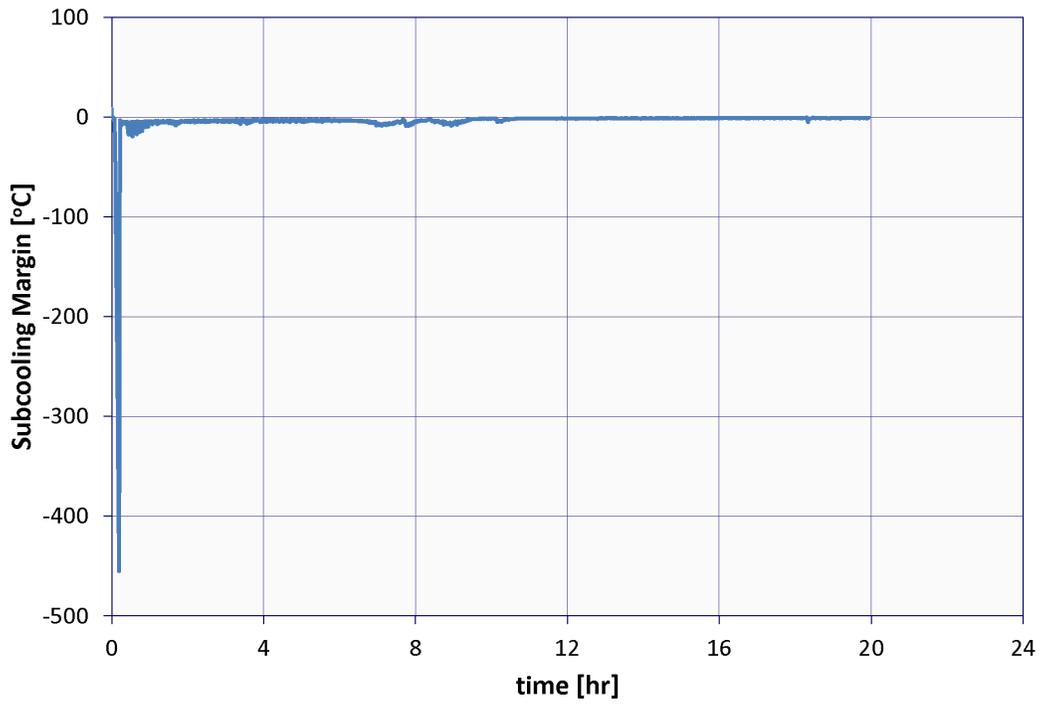
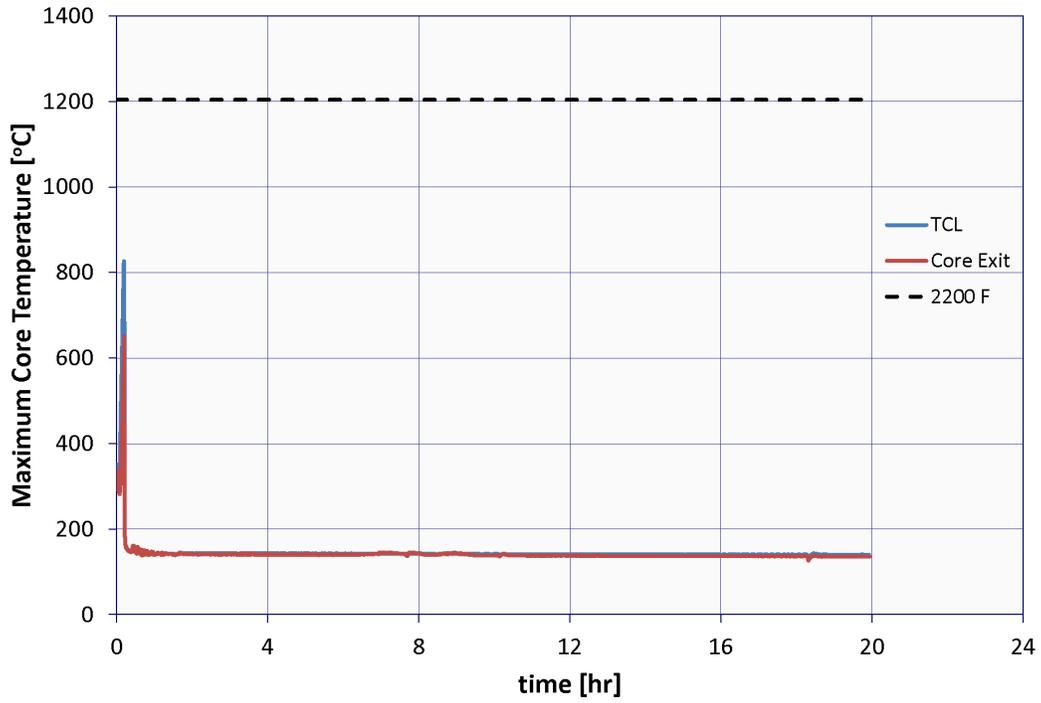


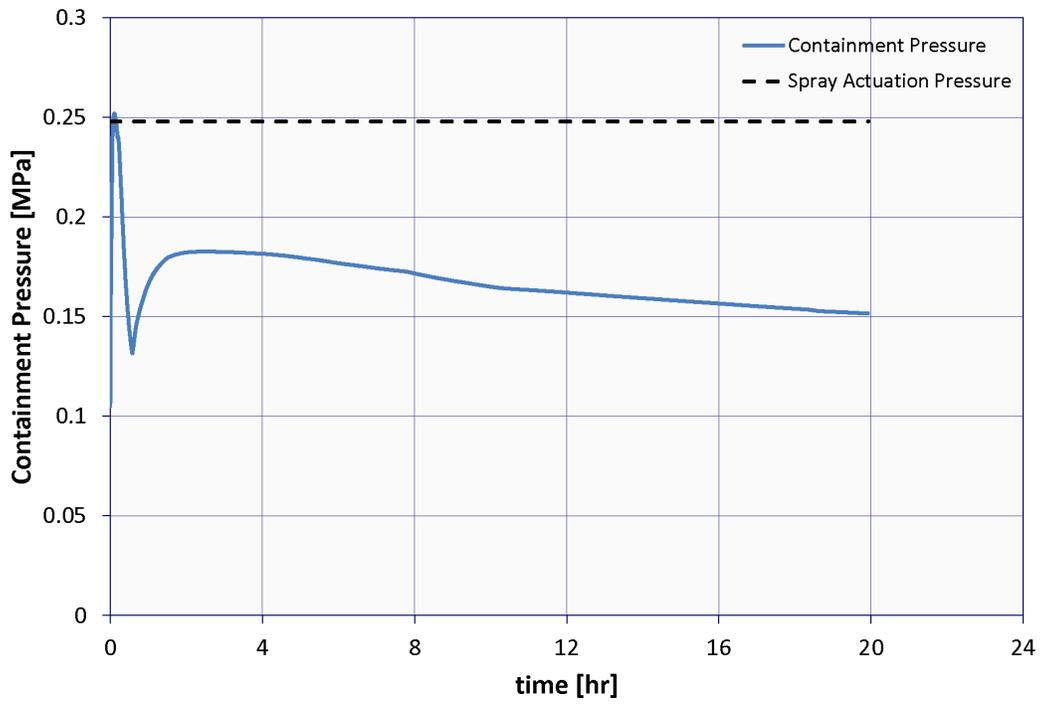
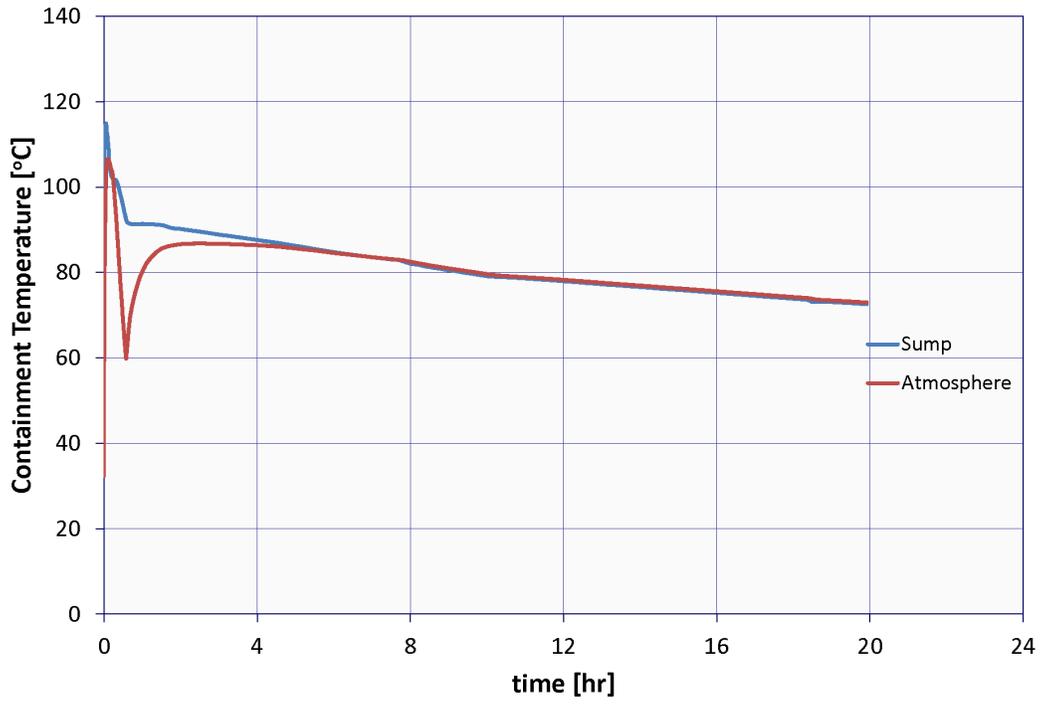


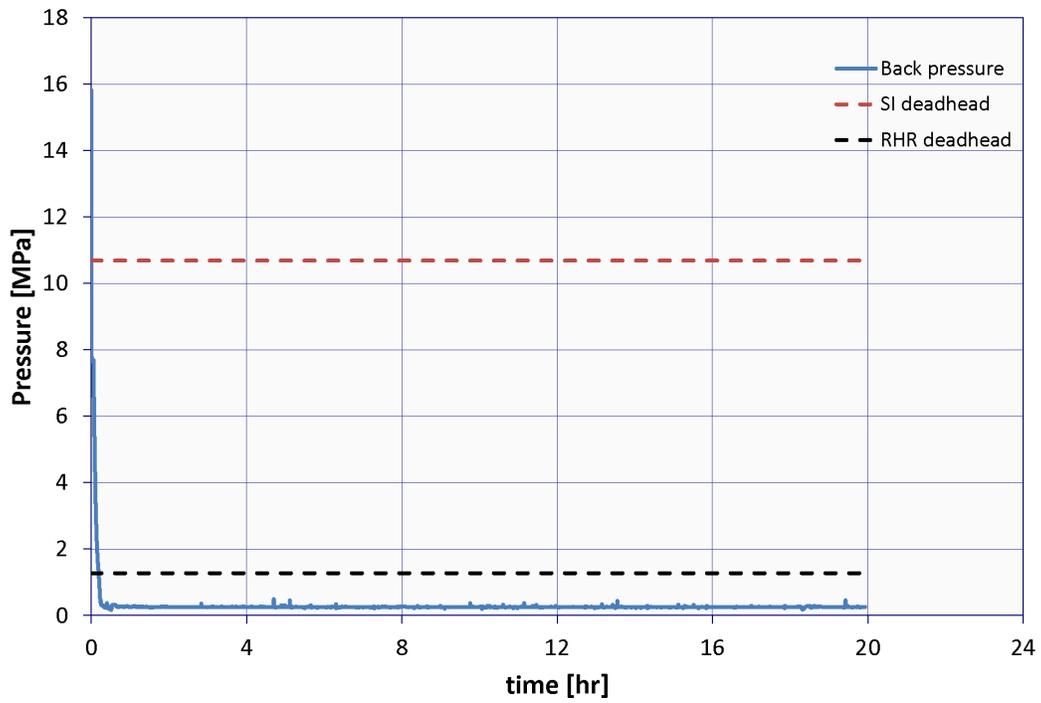
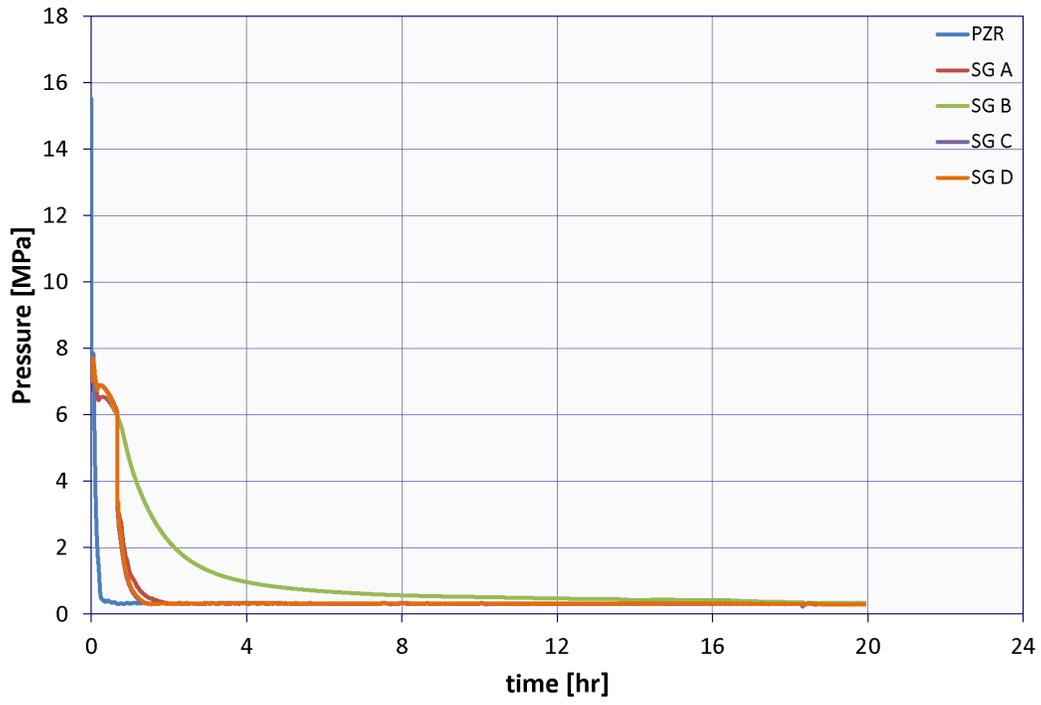


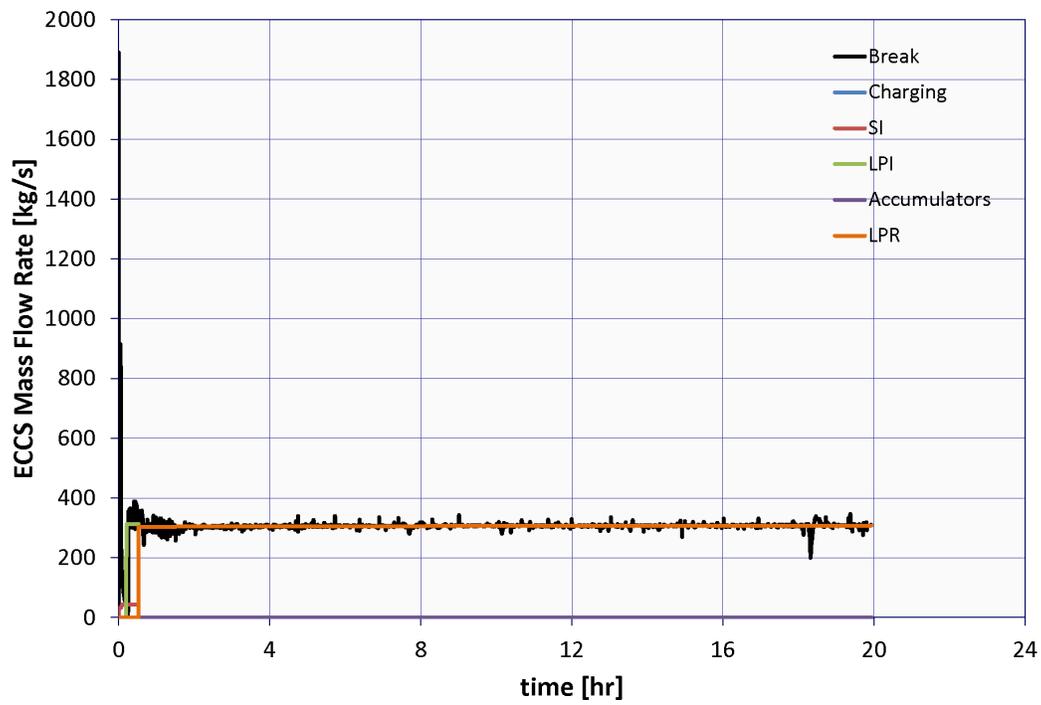
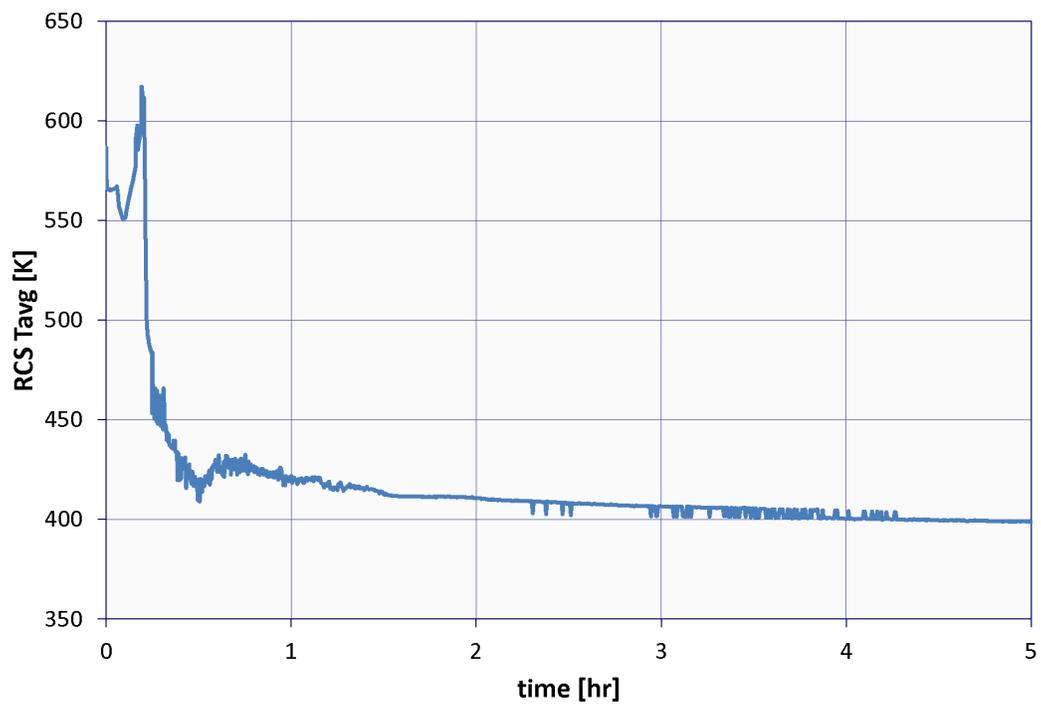


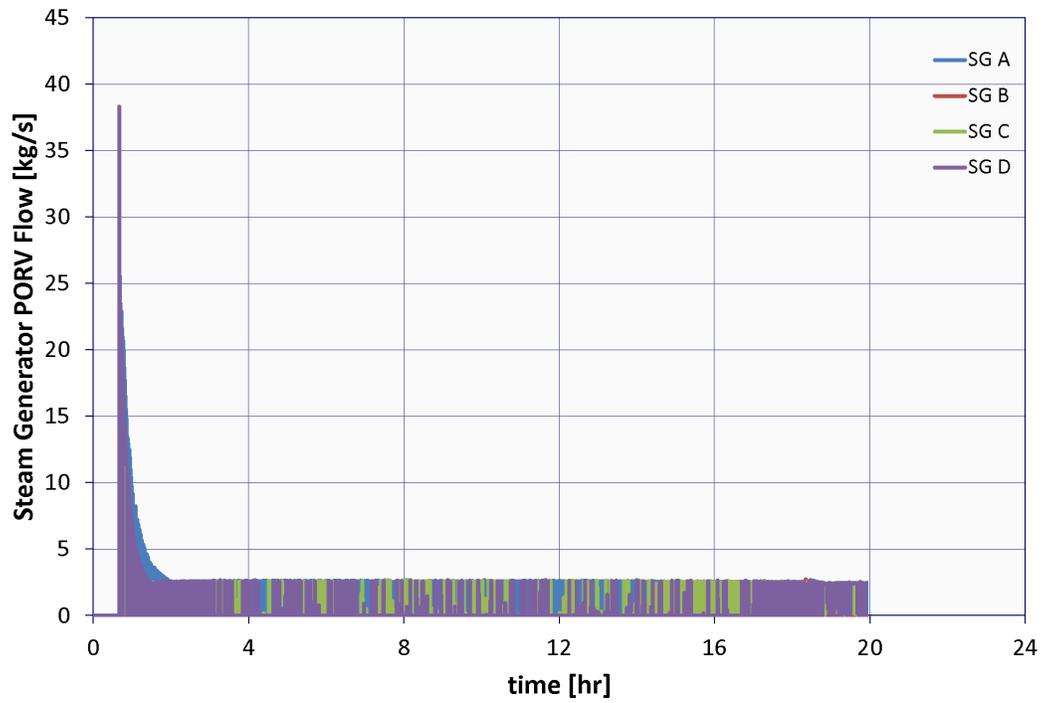
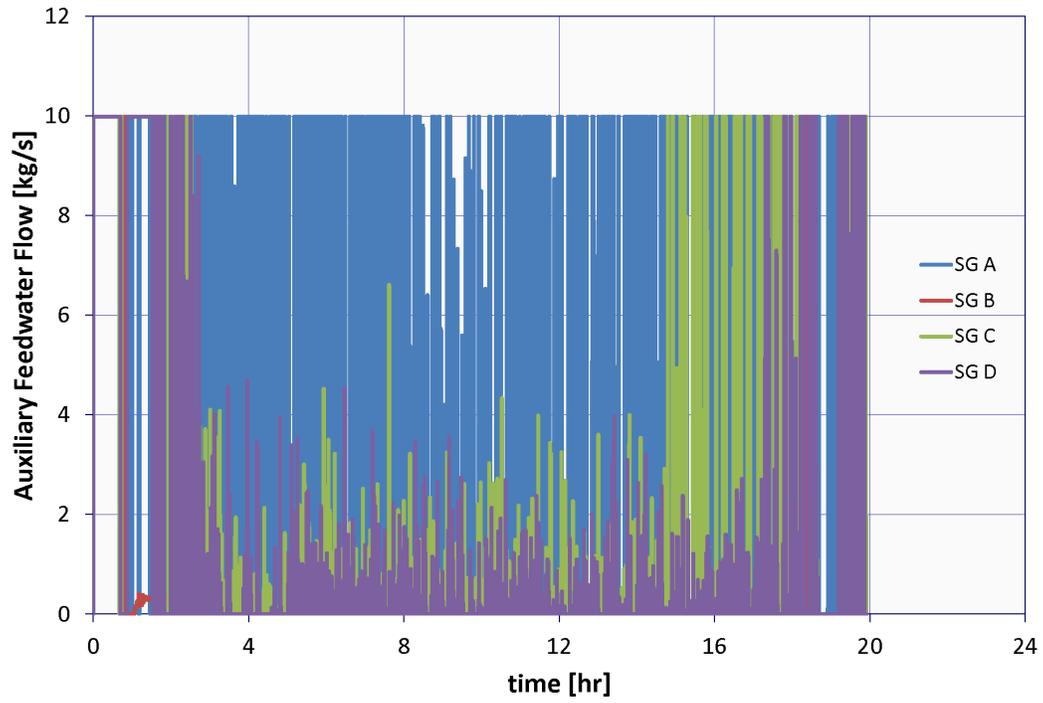
E.2.8.1 Case 8a: 6-in. Break, 100 °F/hr Cooldown at 40 min, 2/2 CS Pumps, 0/4 RCFC, CS Recirc

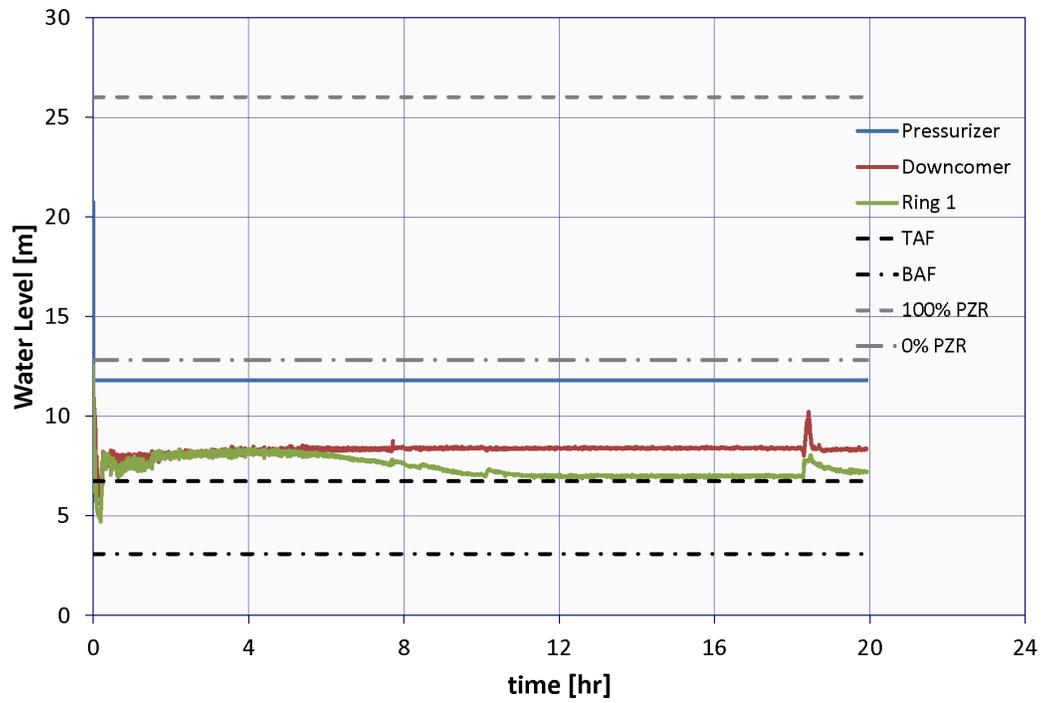
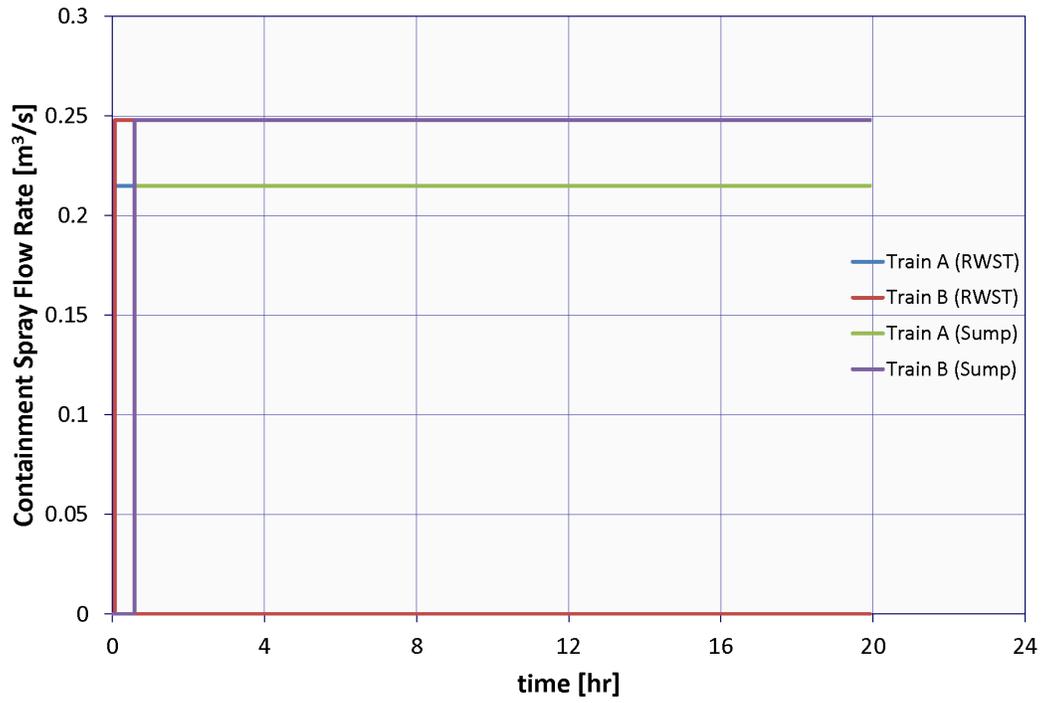


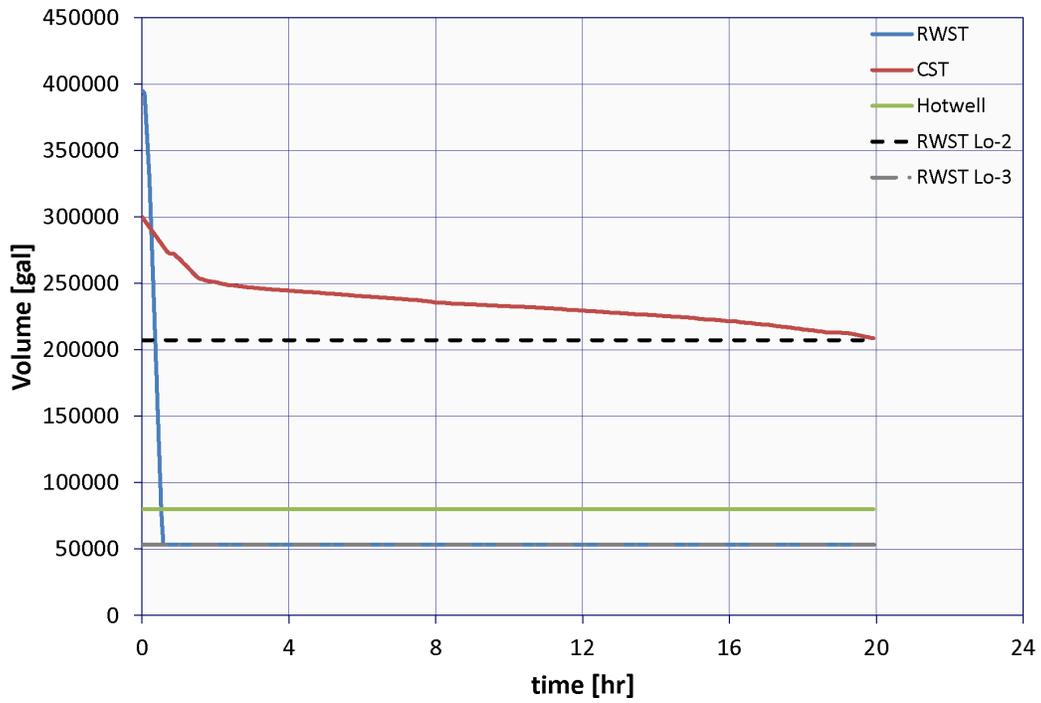
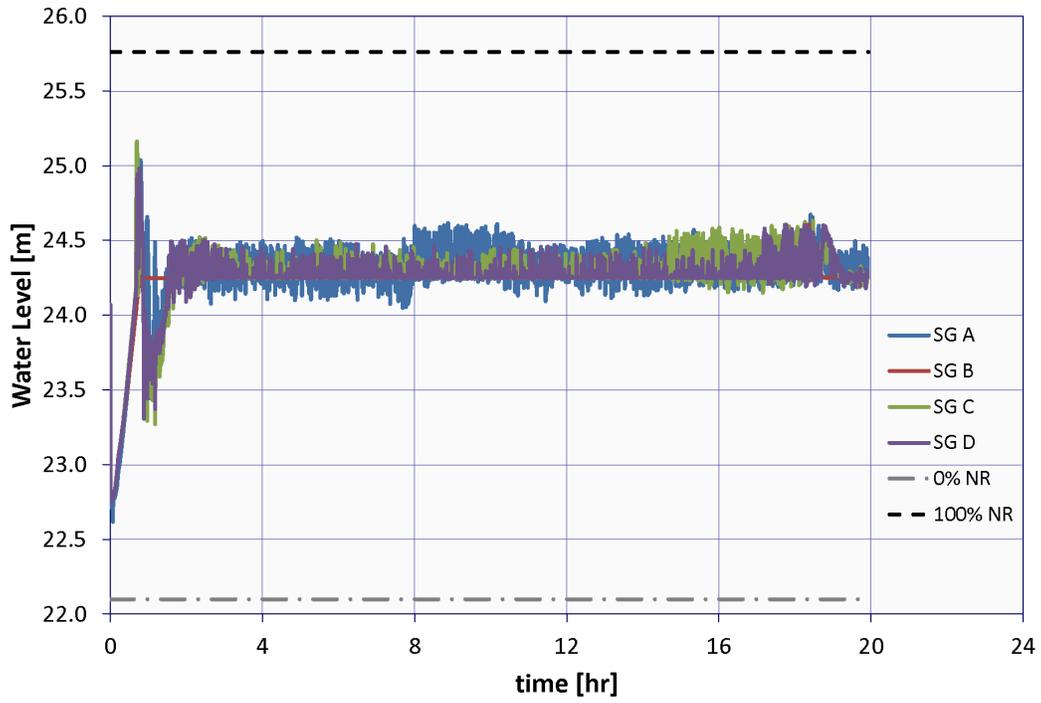




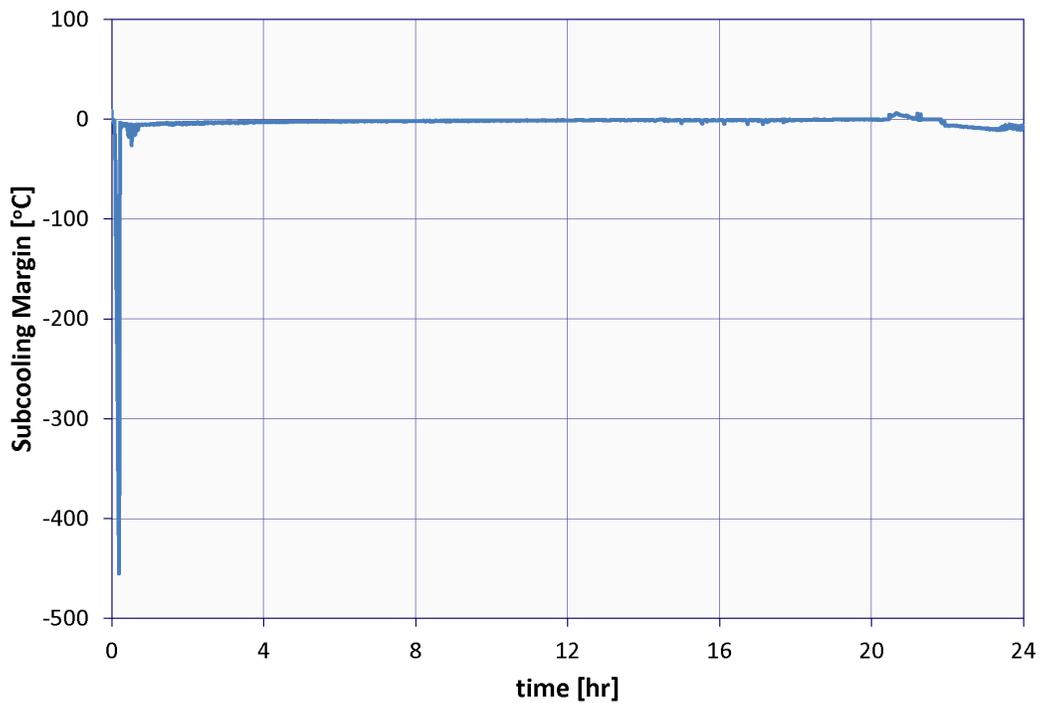
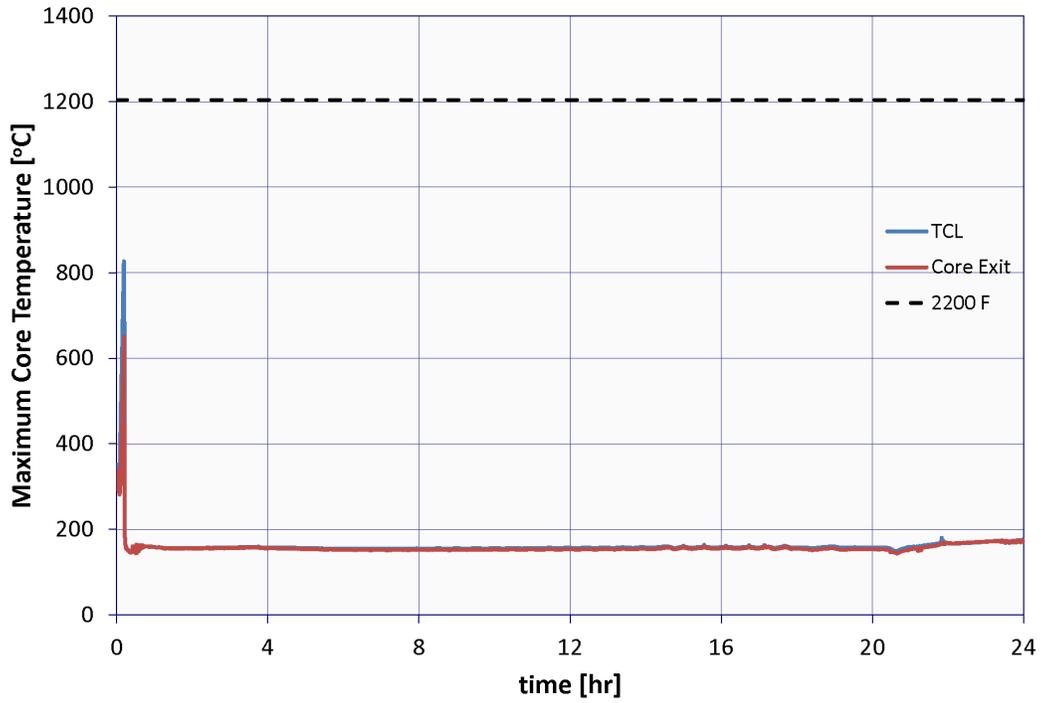


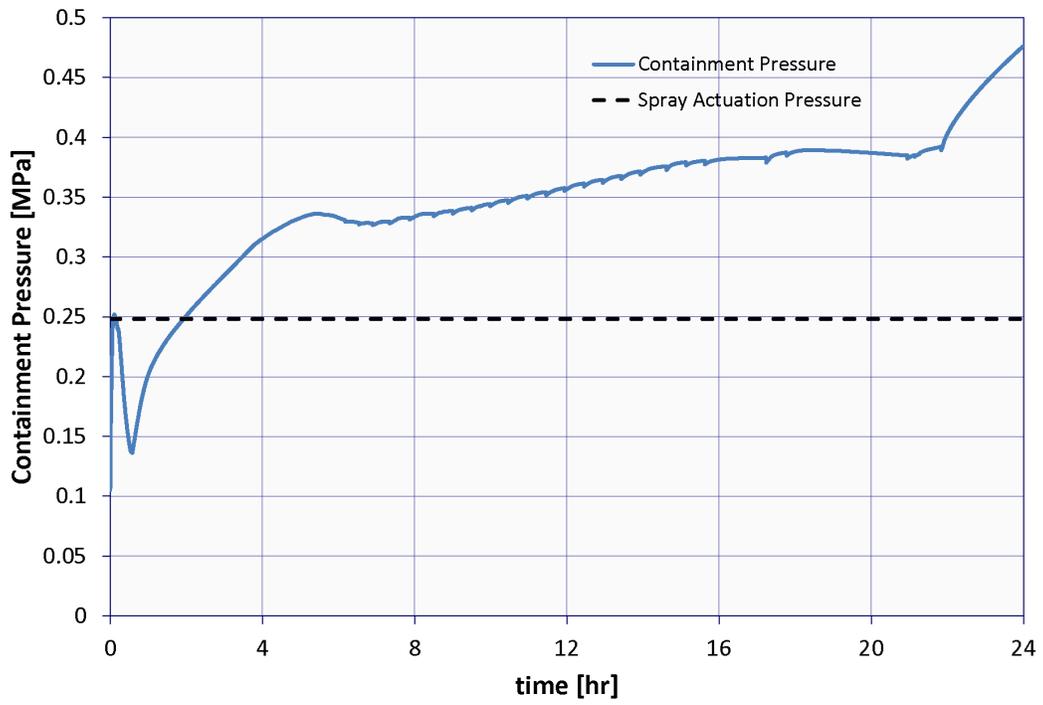
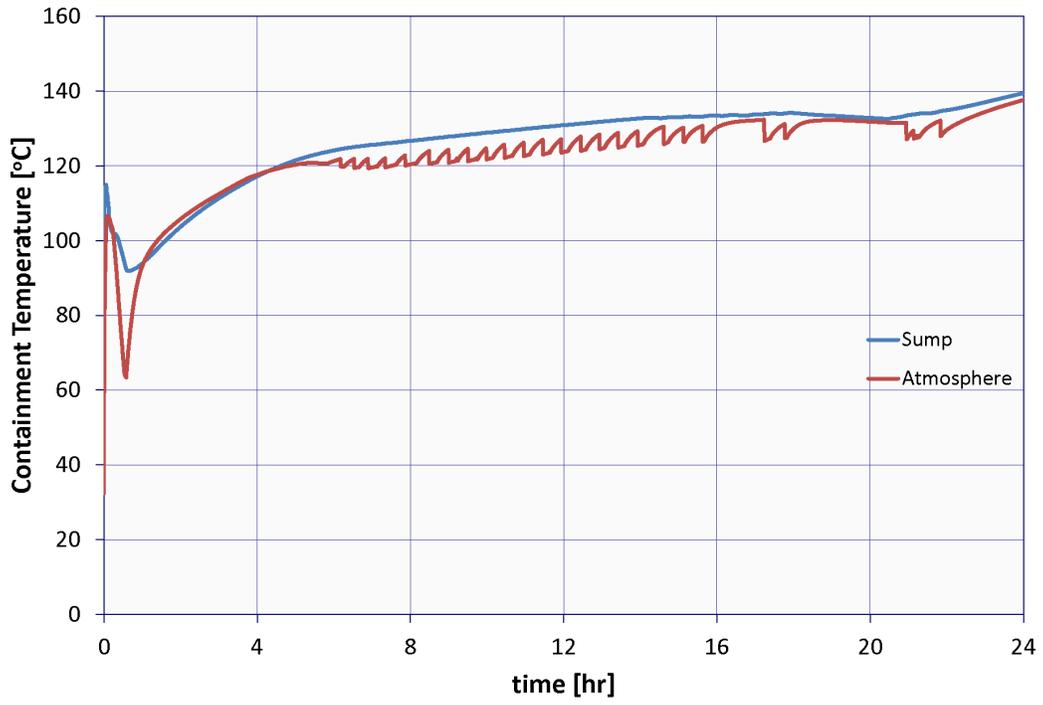


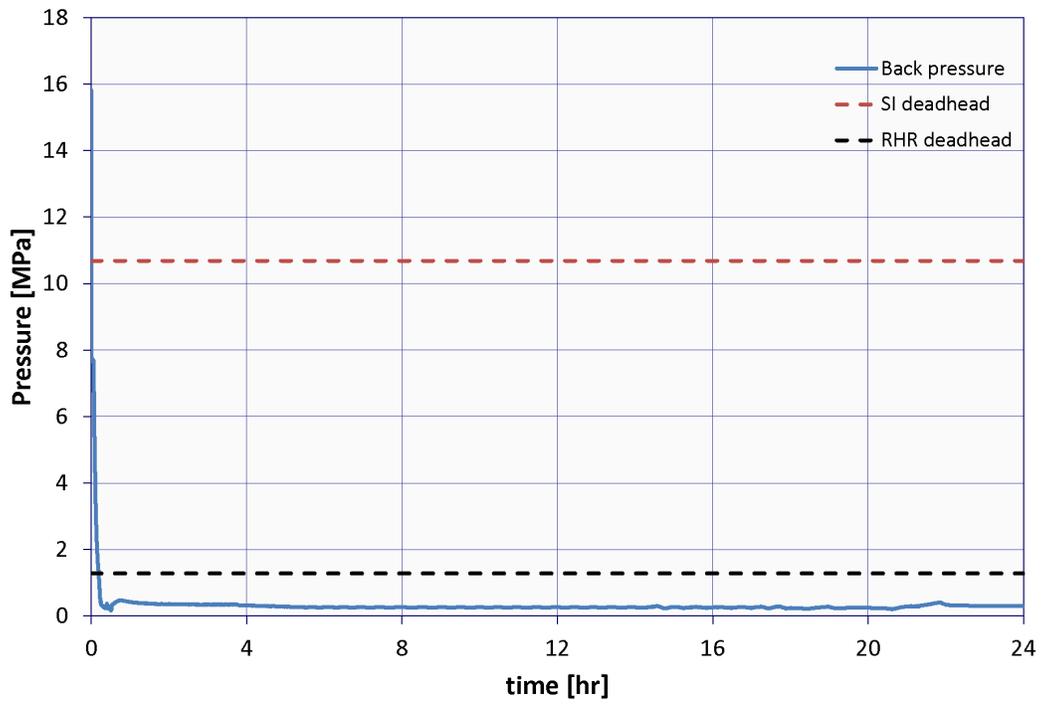
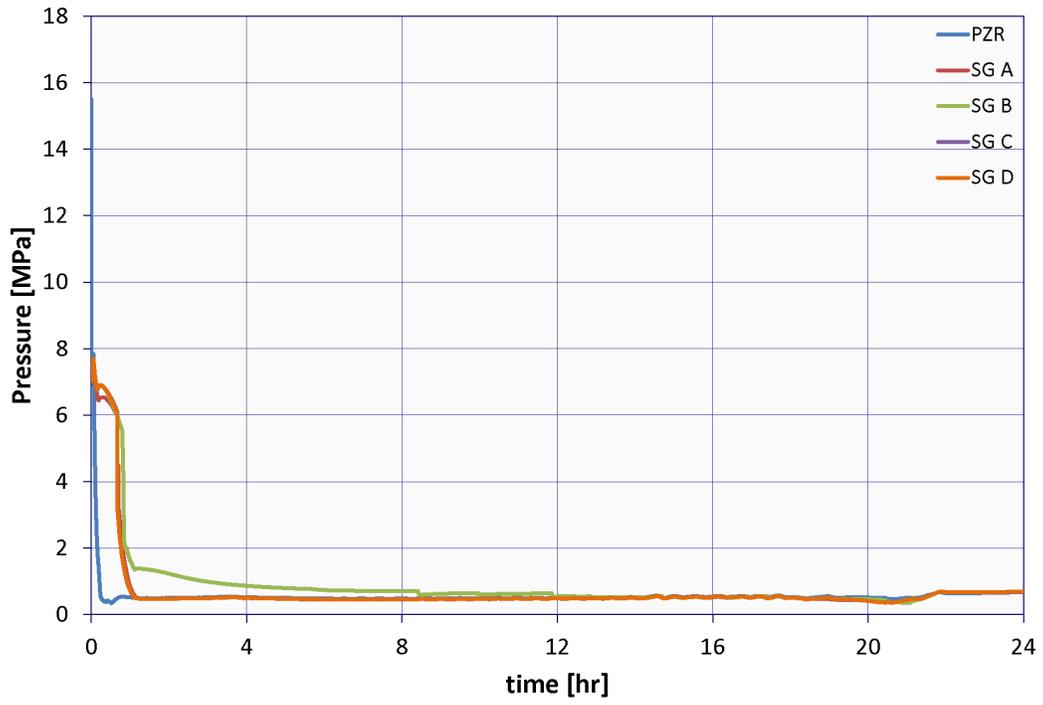


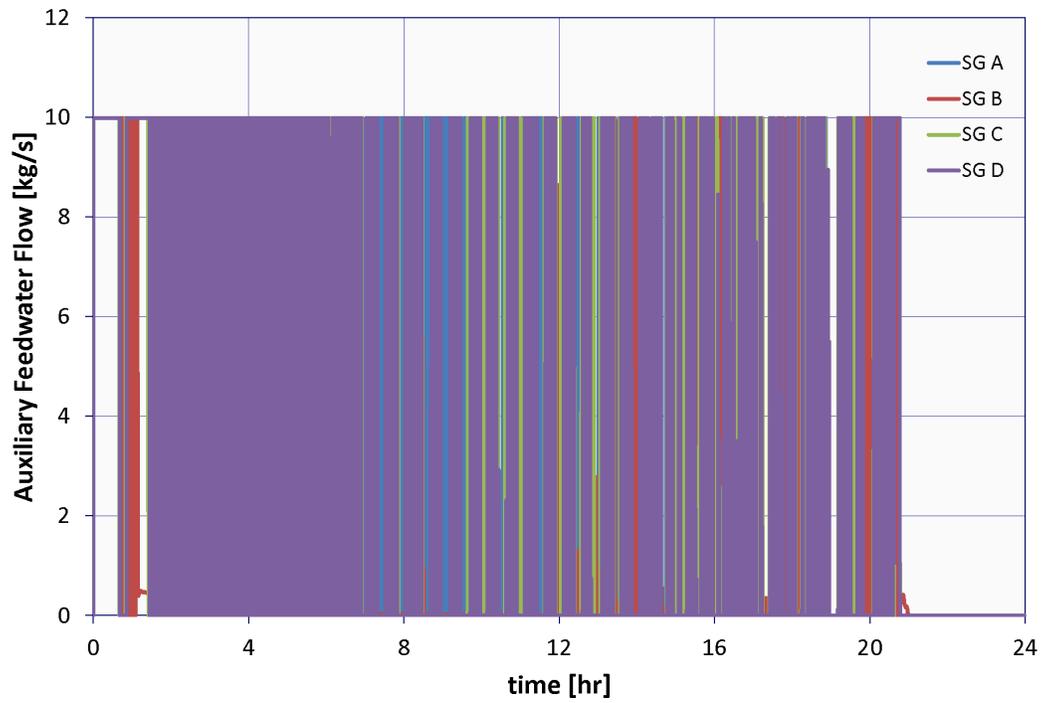
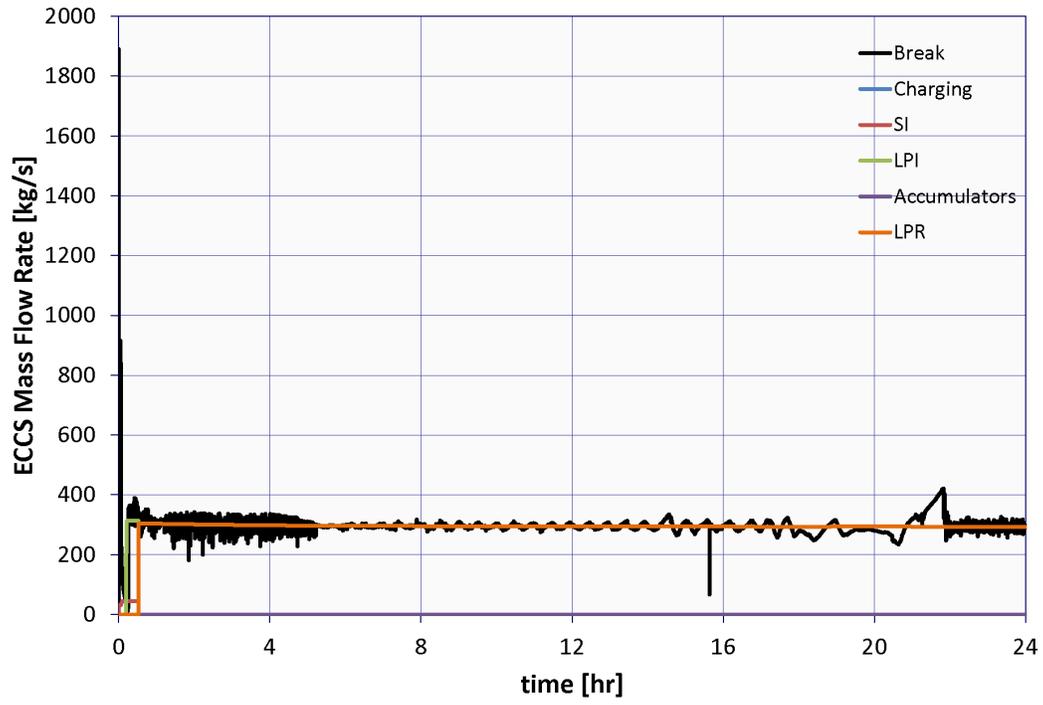


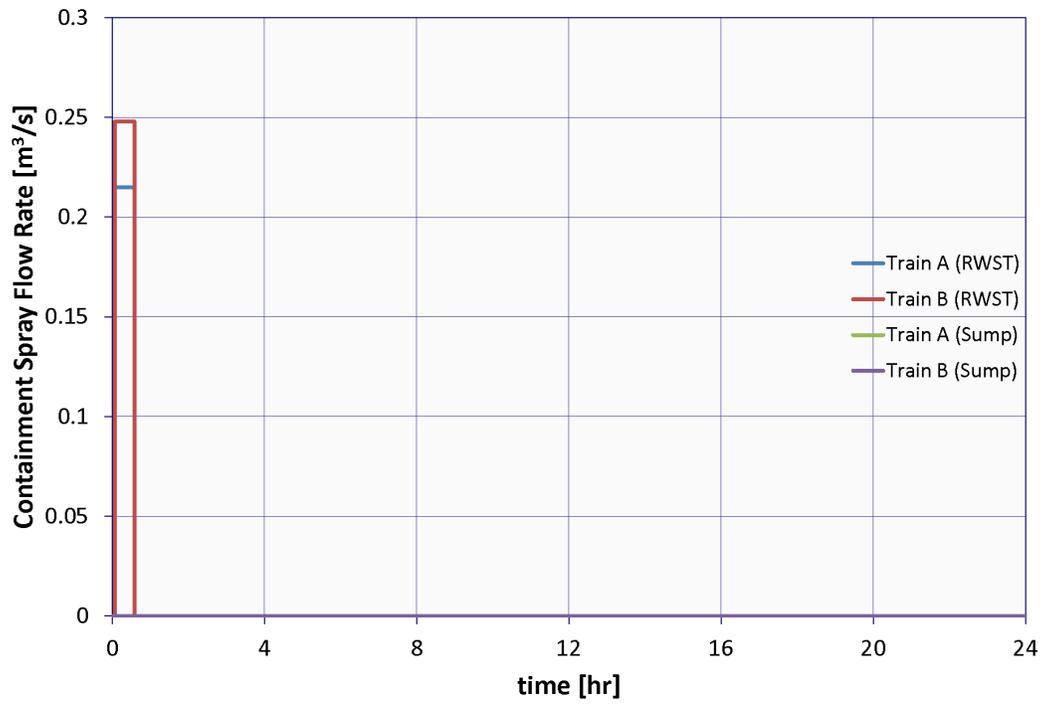
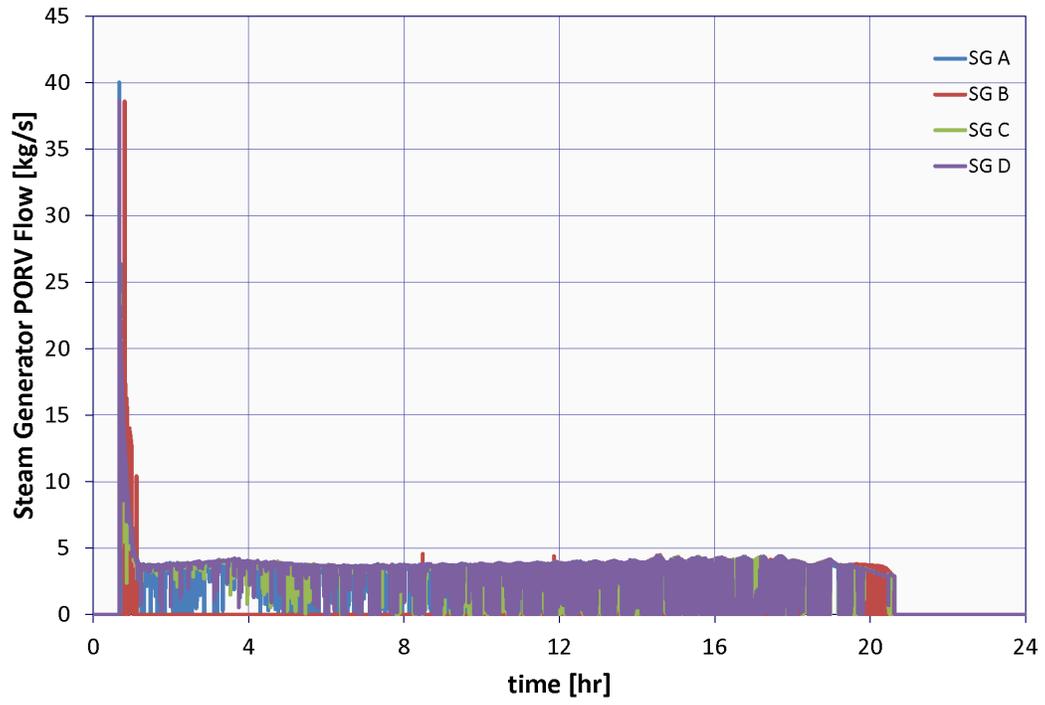
E.2.8.2 Case 8b: 6-in. Break, 100 °F/hr Cooldown at 40 min, 2/2 CS Pumps, 0/4 RCFC, No RHRHX

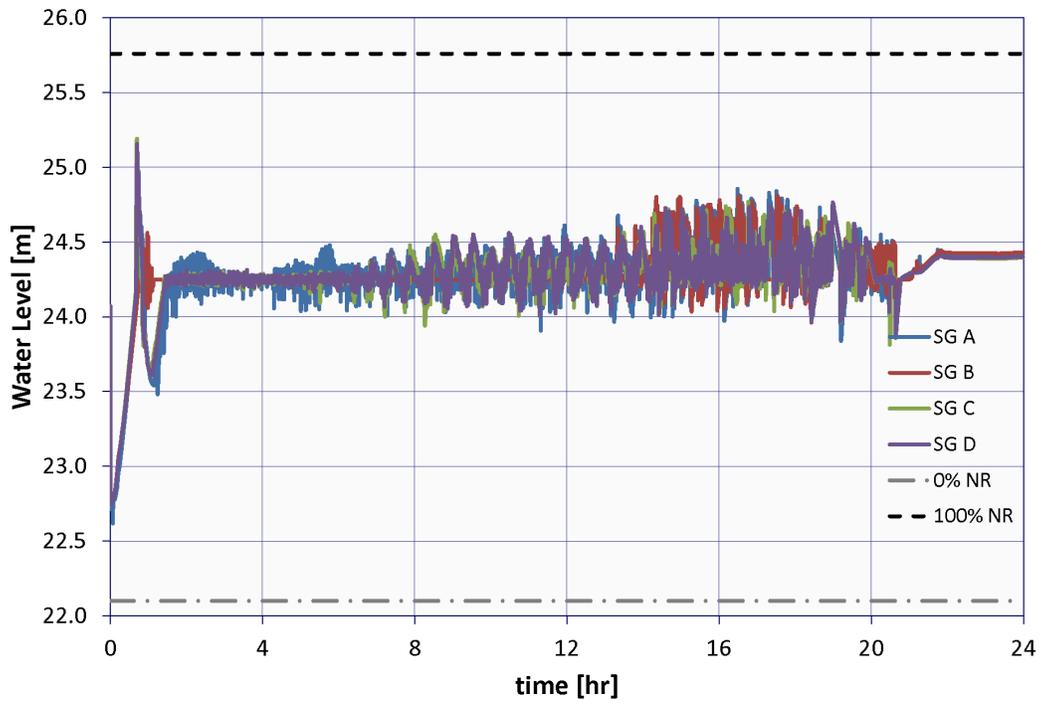
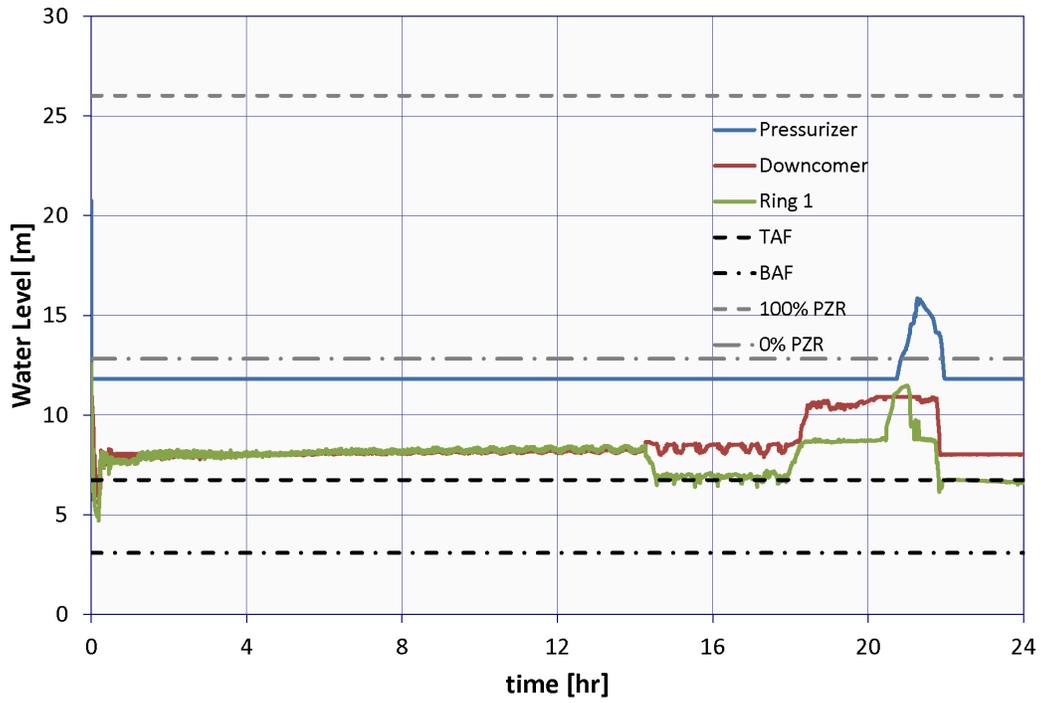


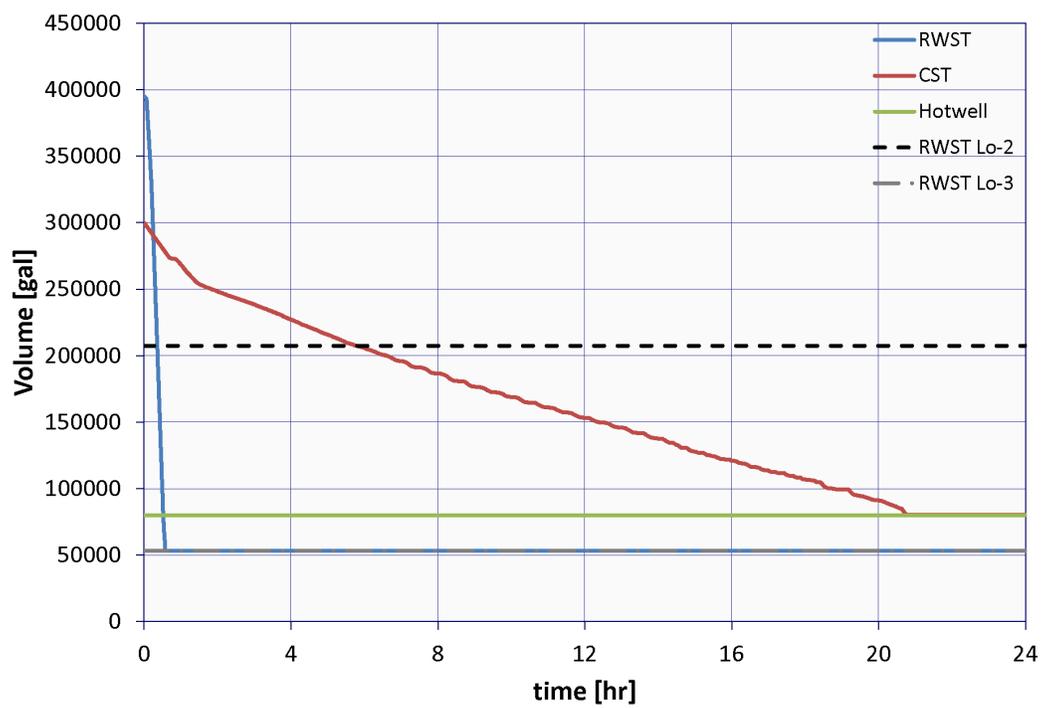




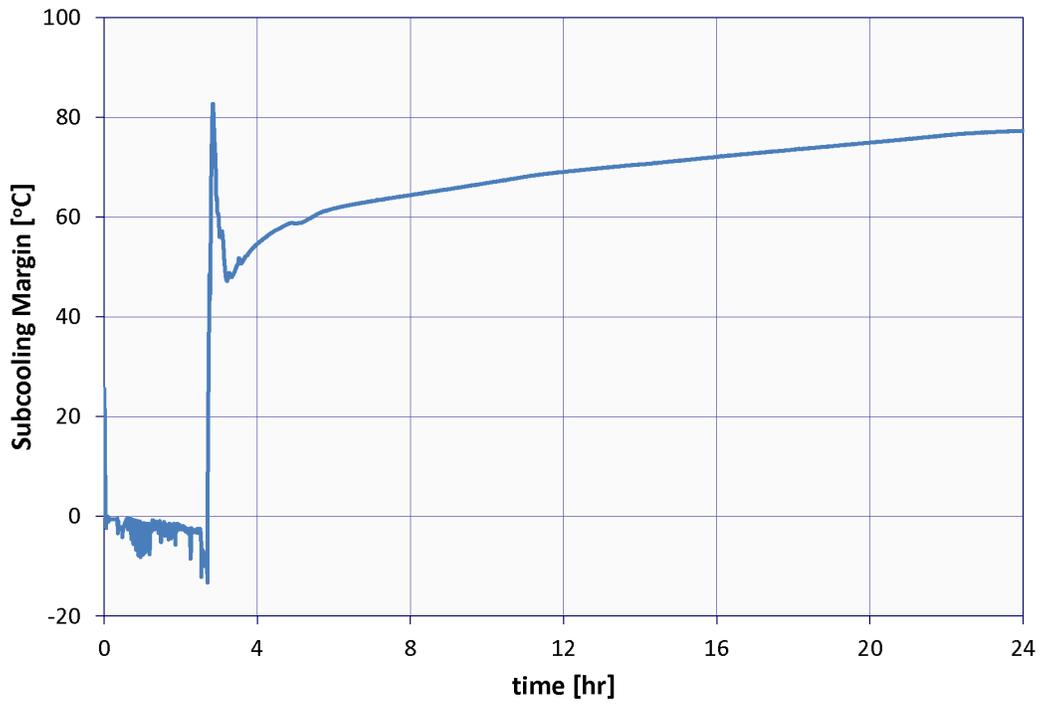
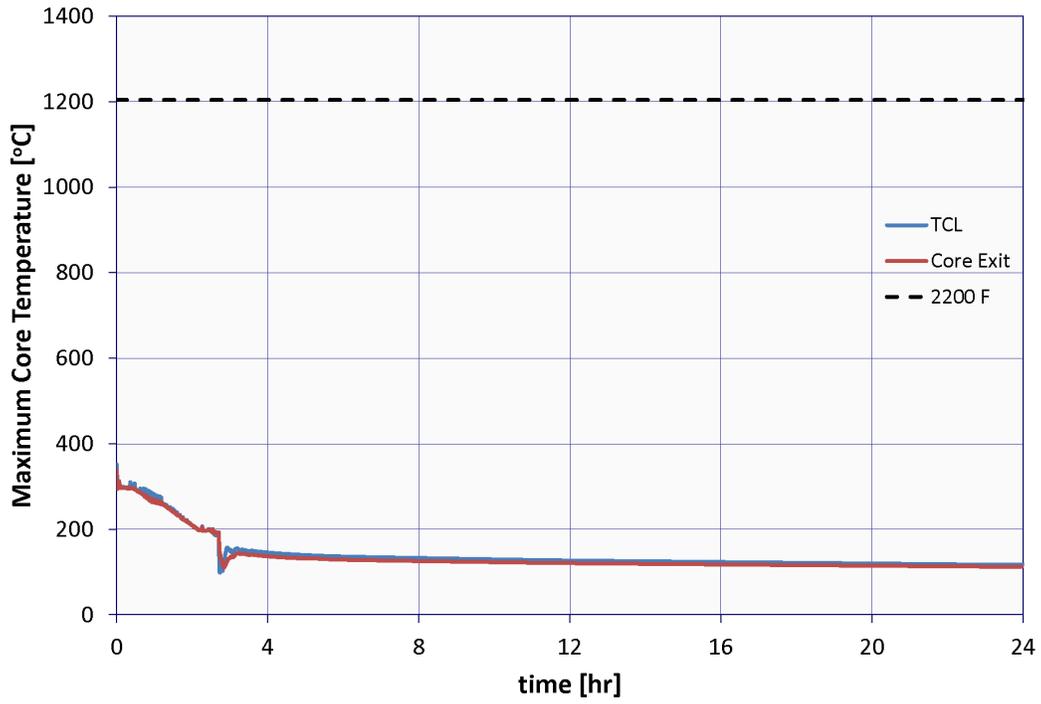


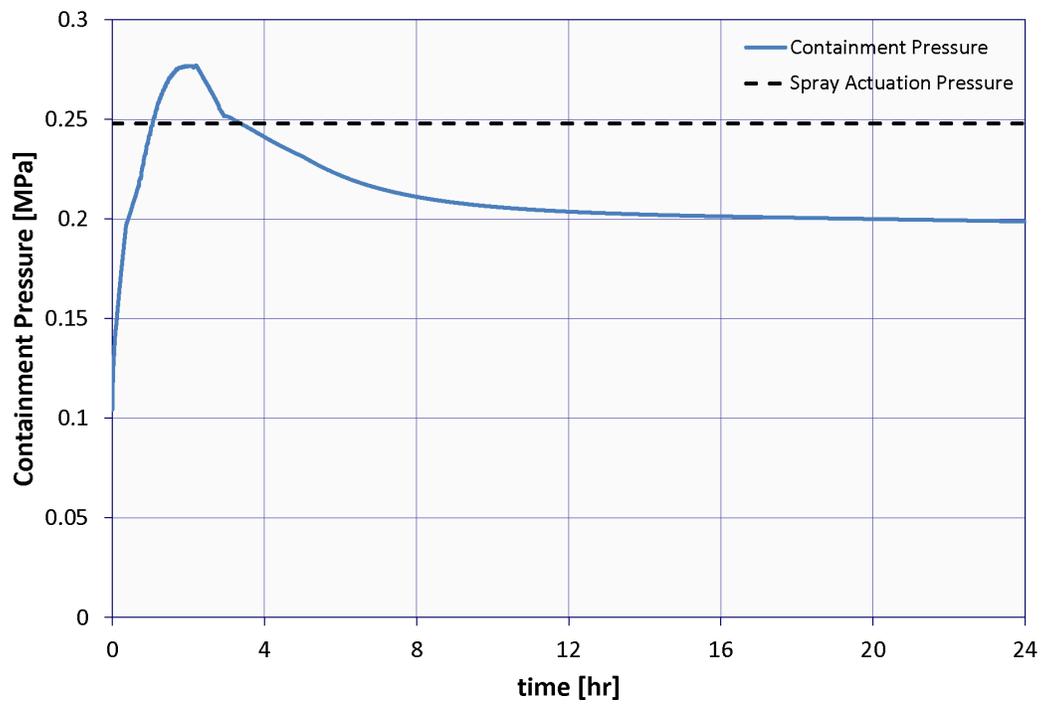
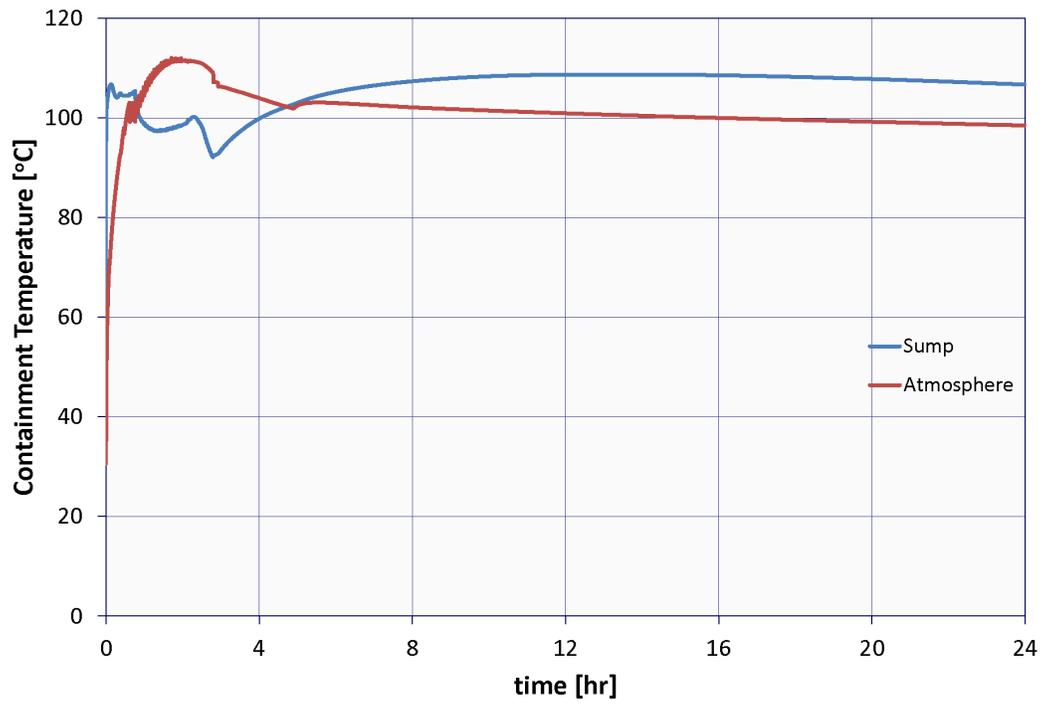


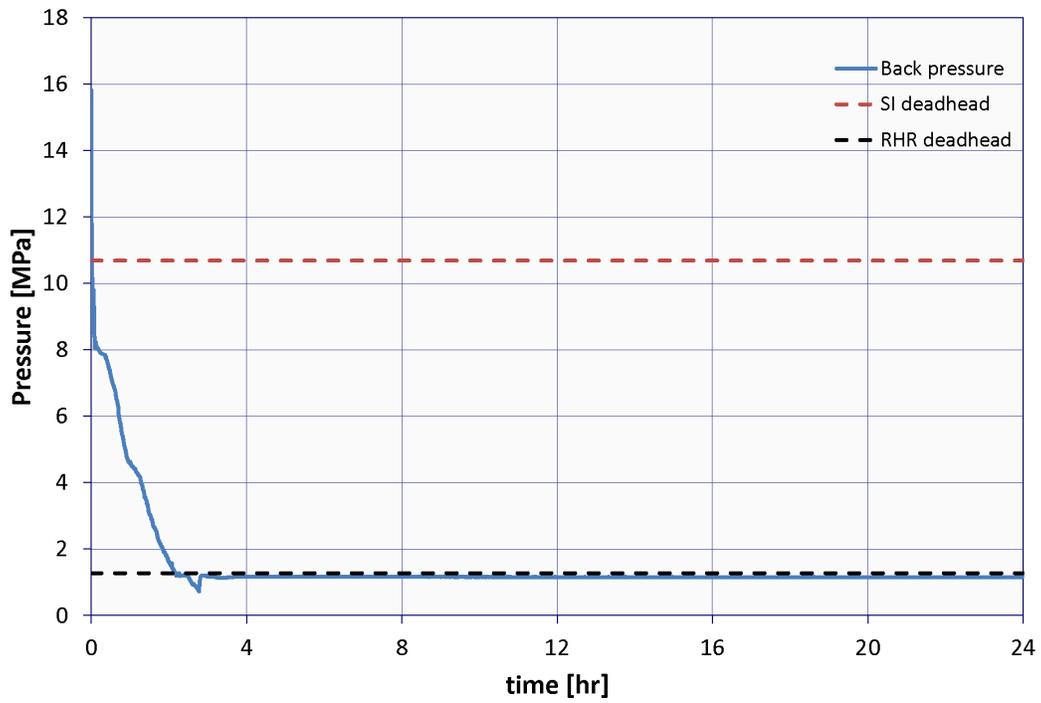
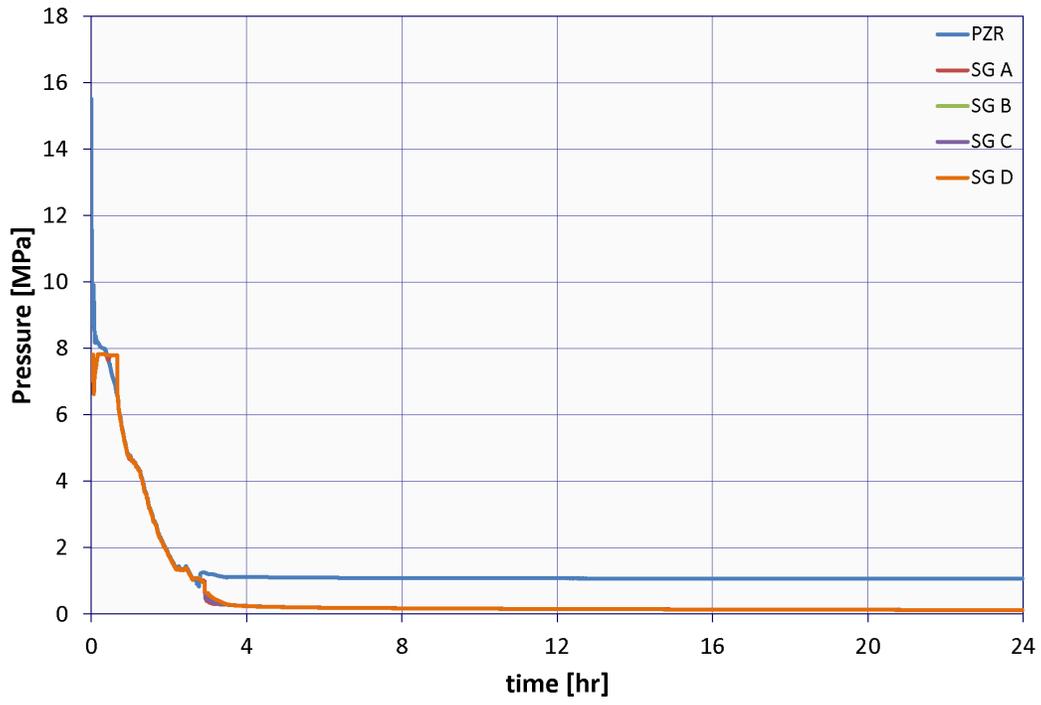


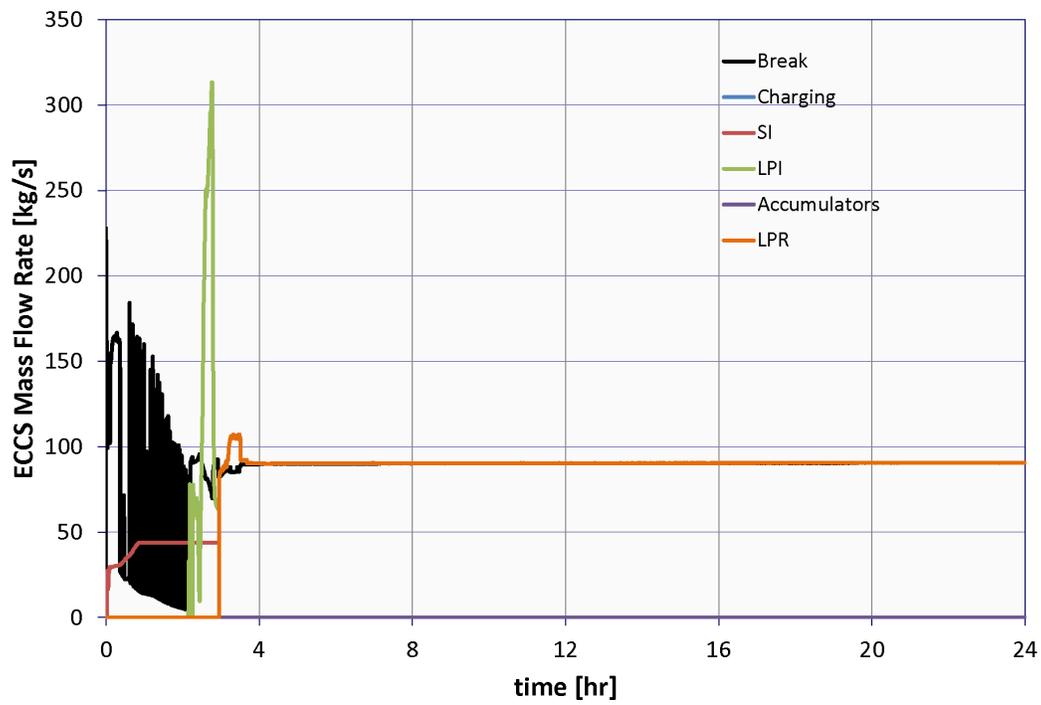
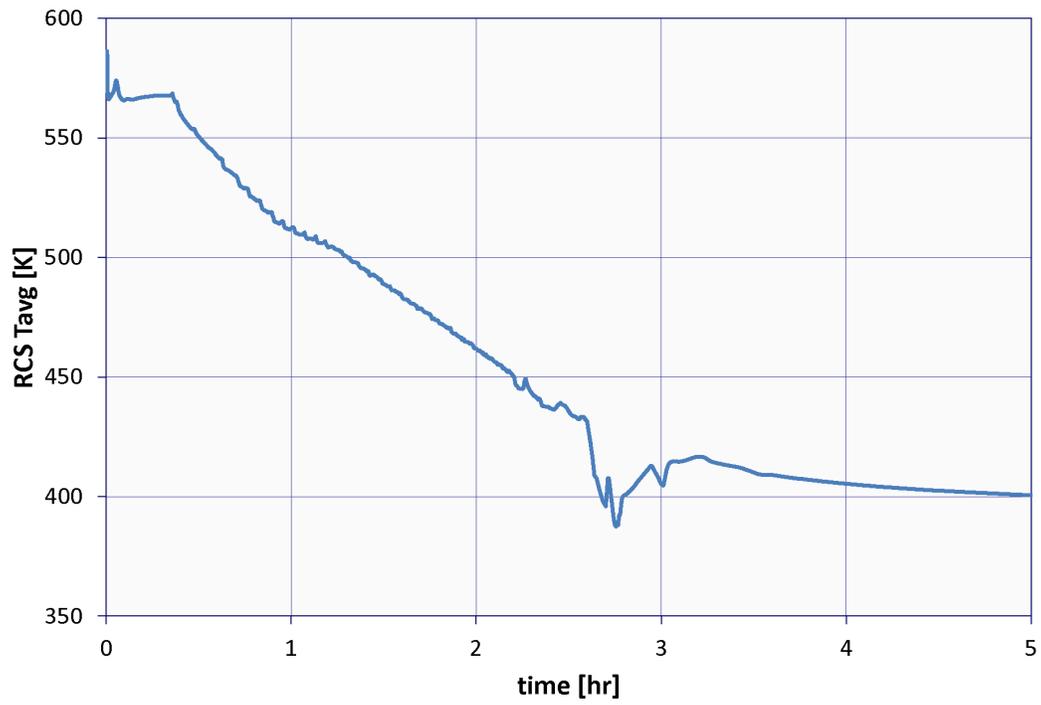


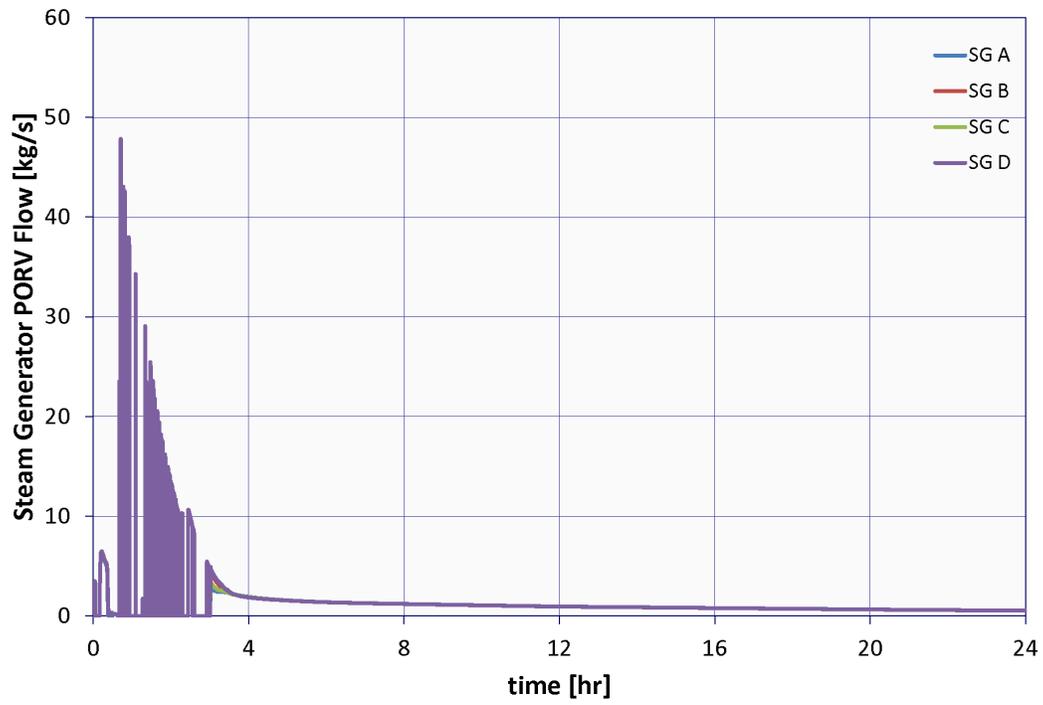
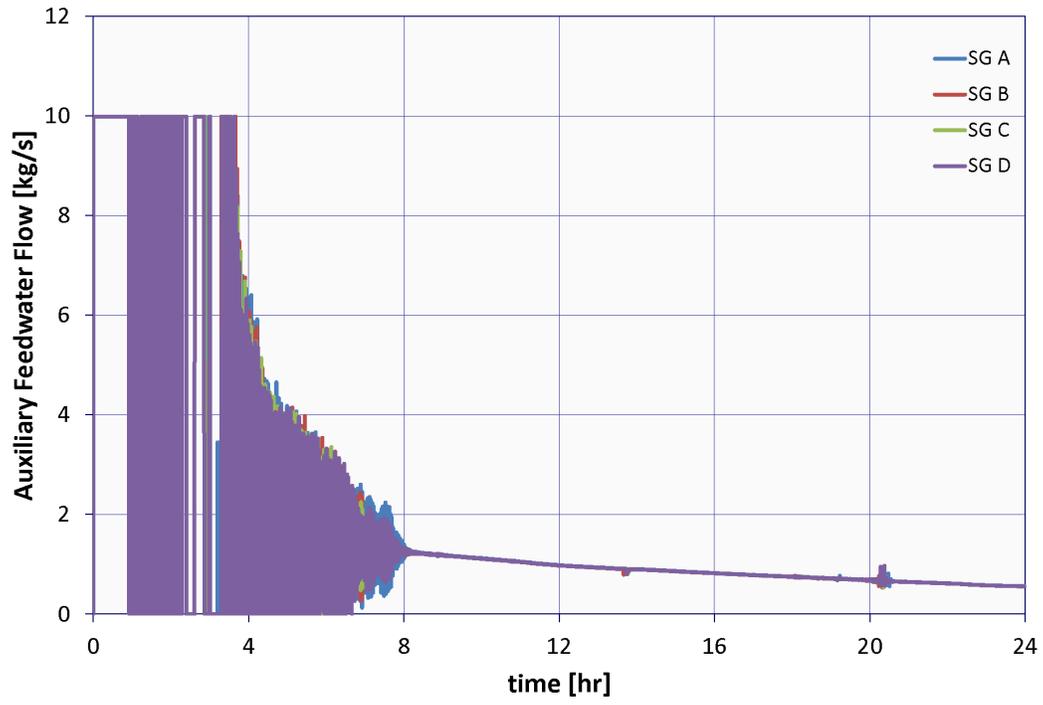
E.2.9 Case 9: 2-in. Break, 100 °F/hr Cooldown at 40 min, 0/2 CS Pumps, 0/4 RCFC

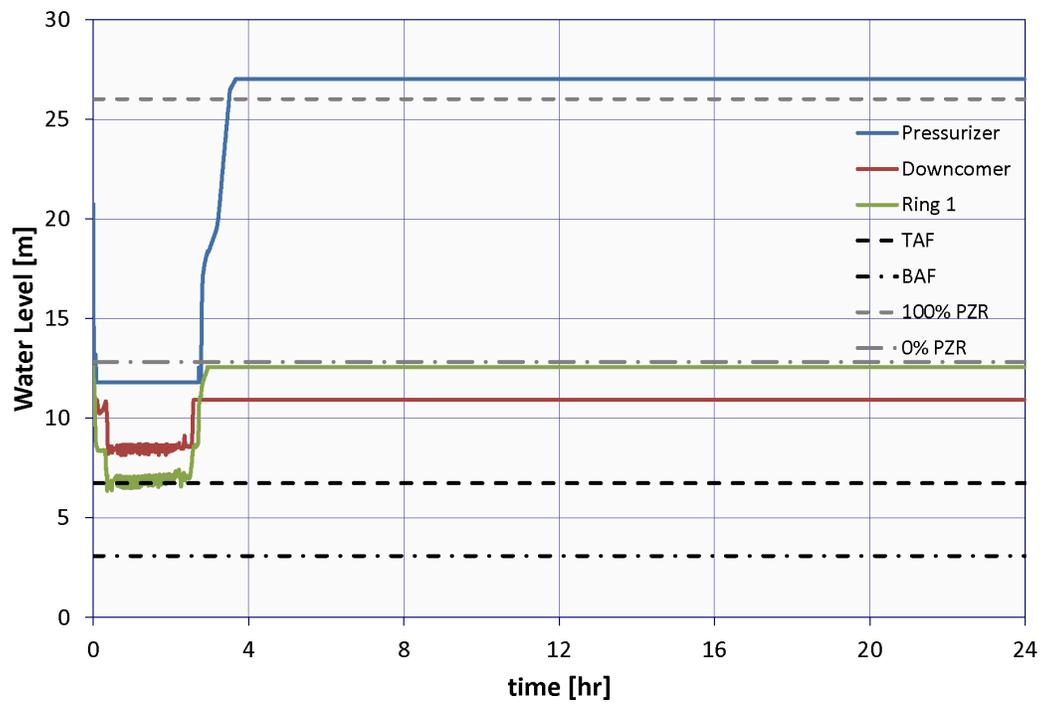
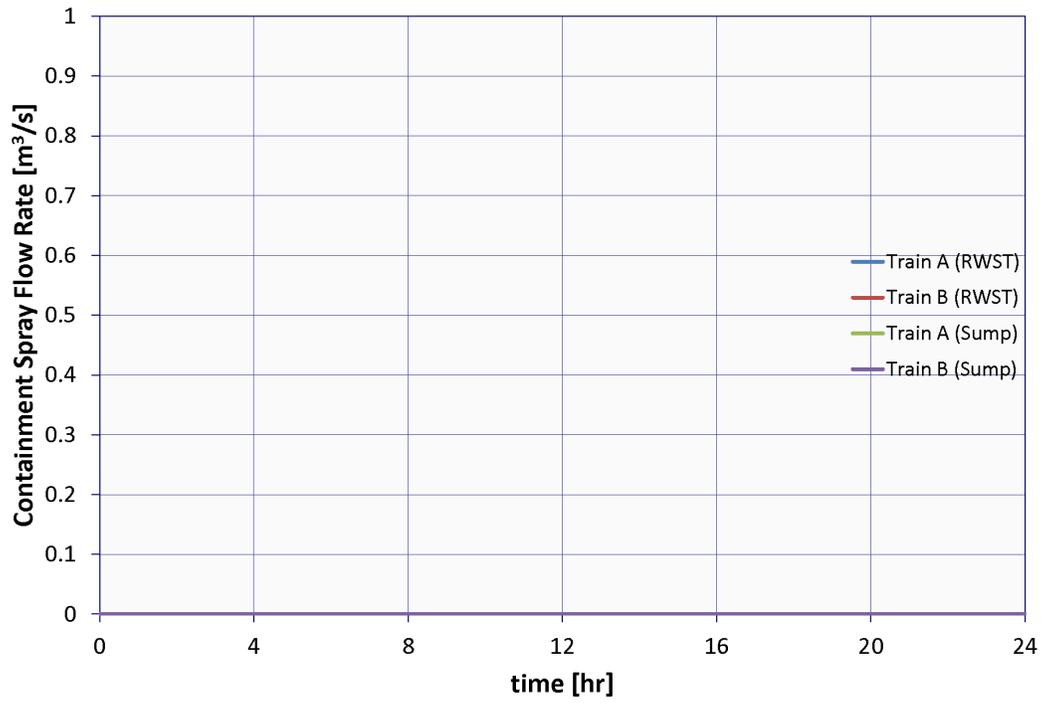


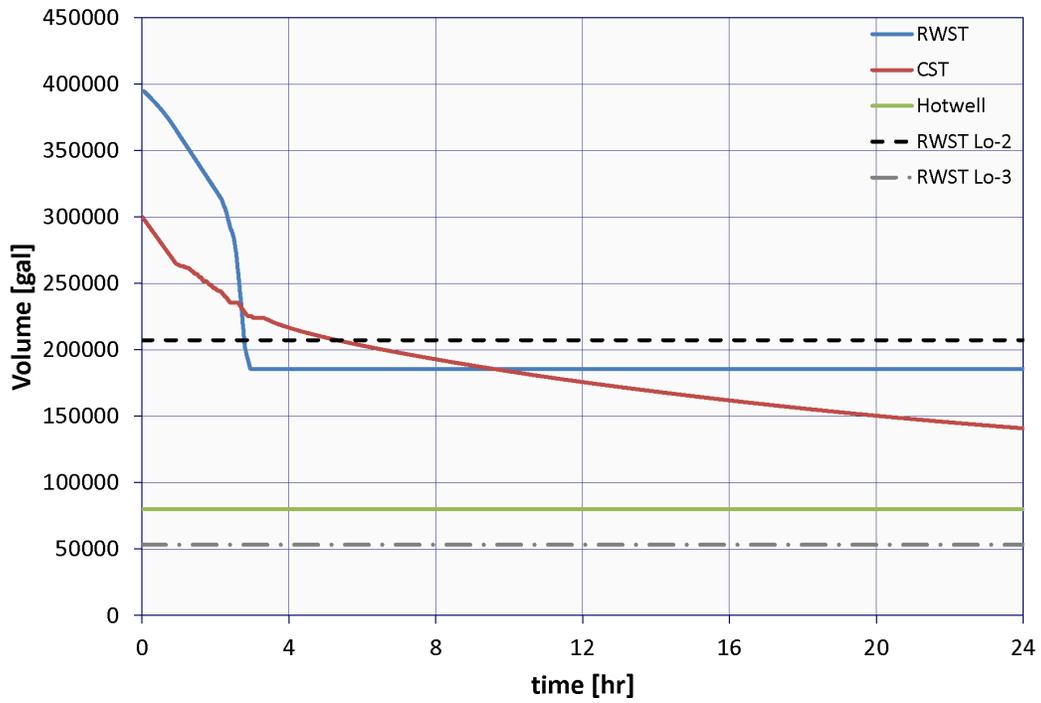
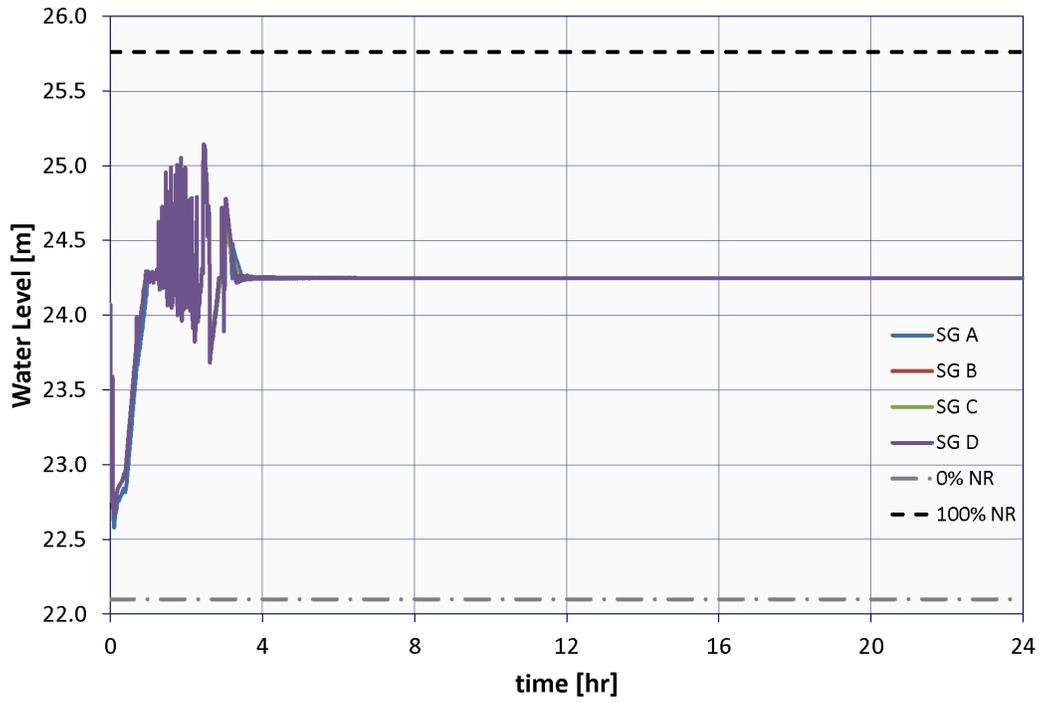




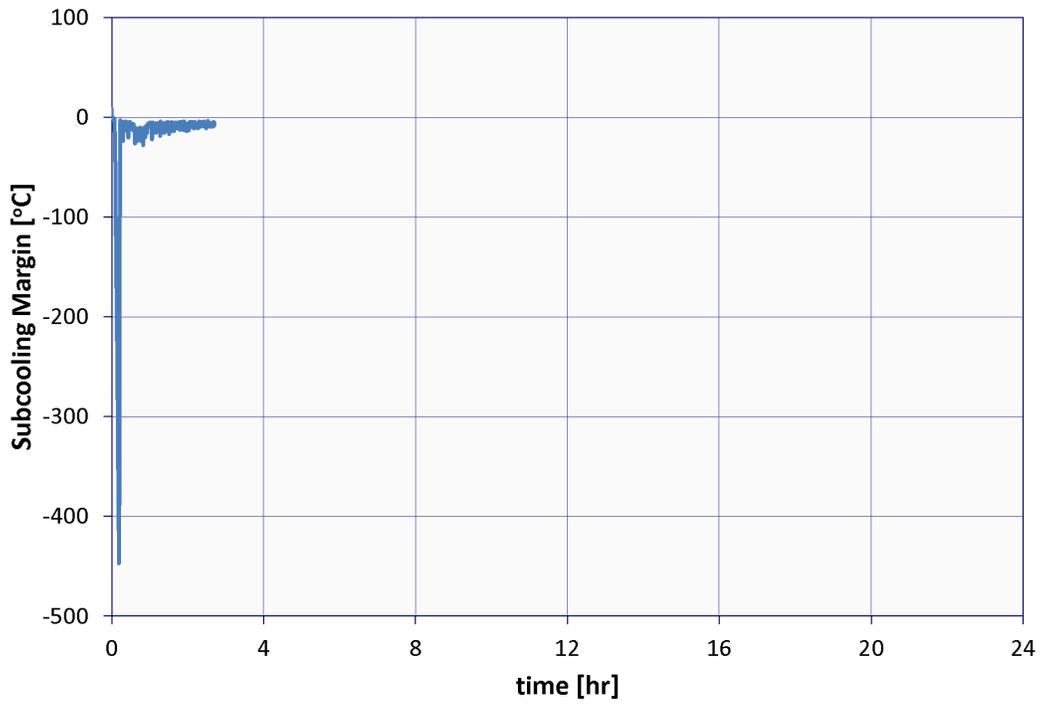
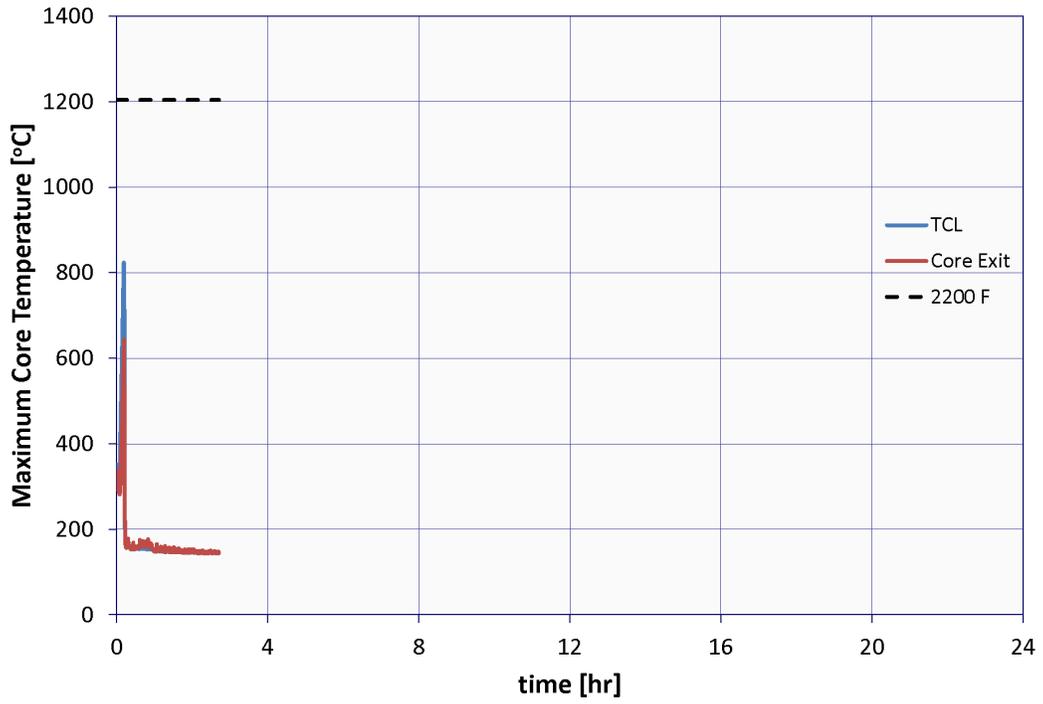


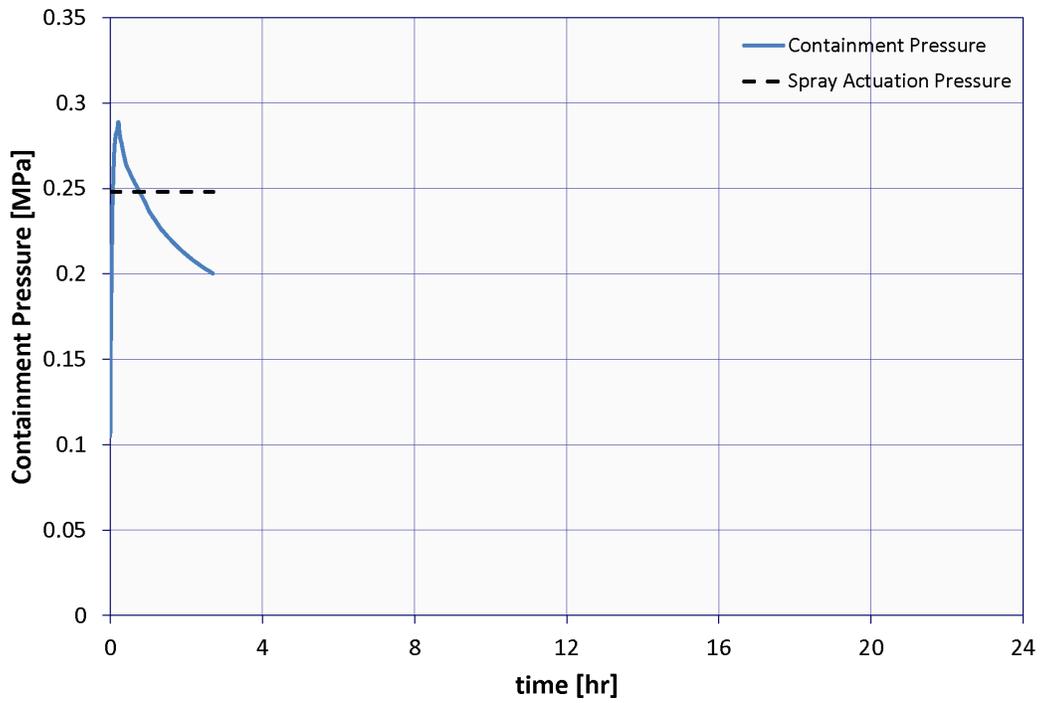
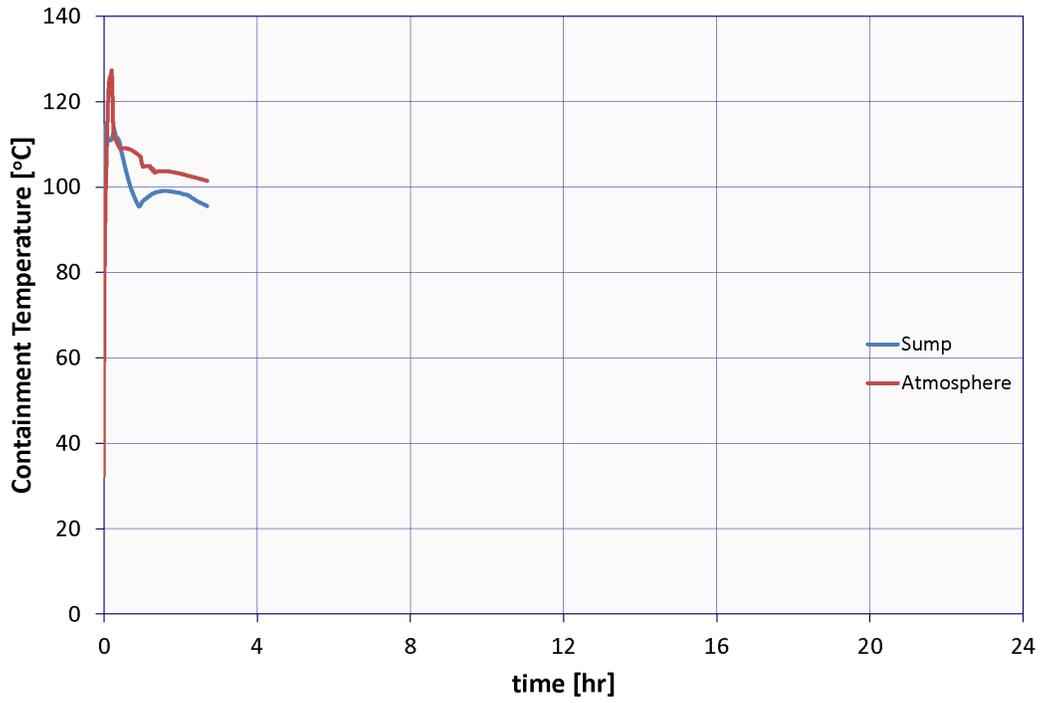


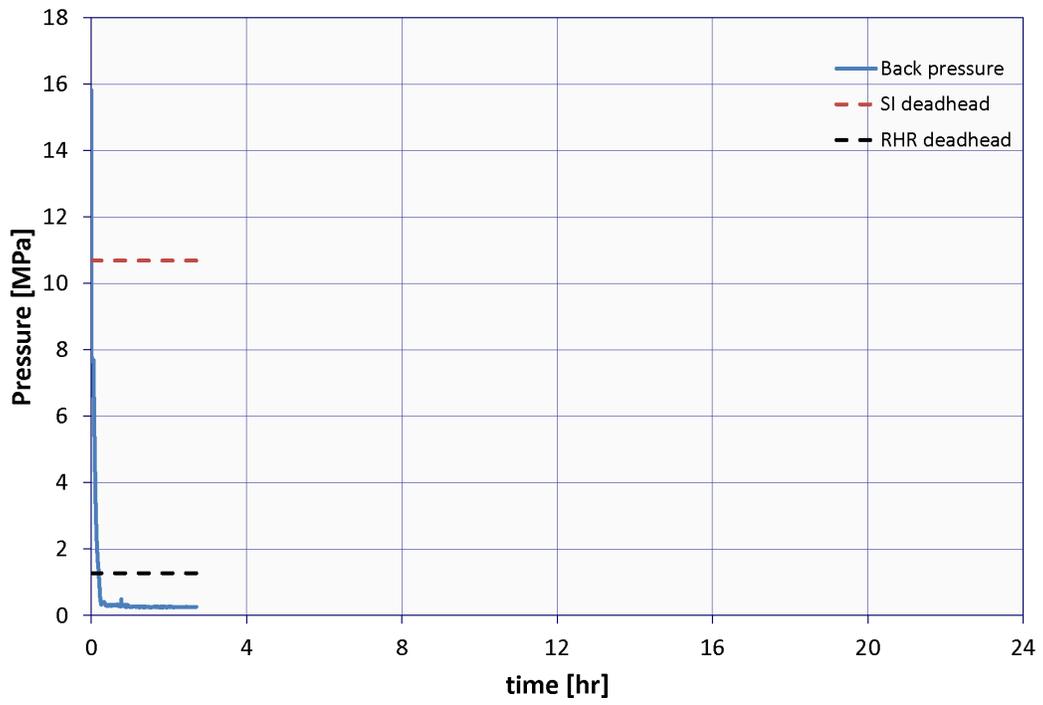
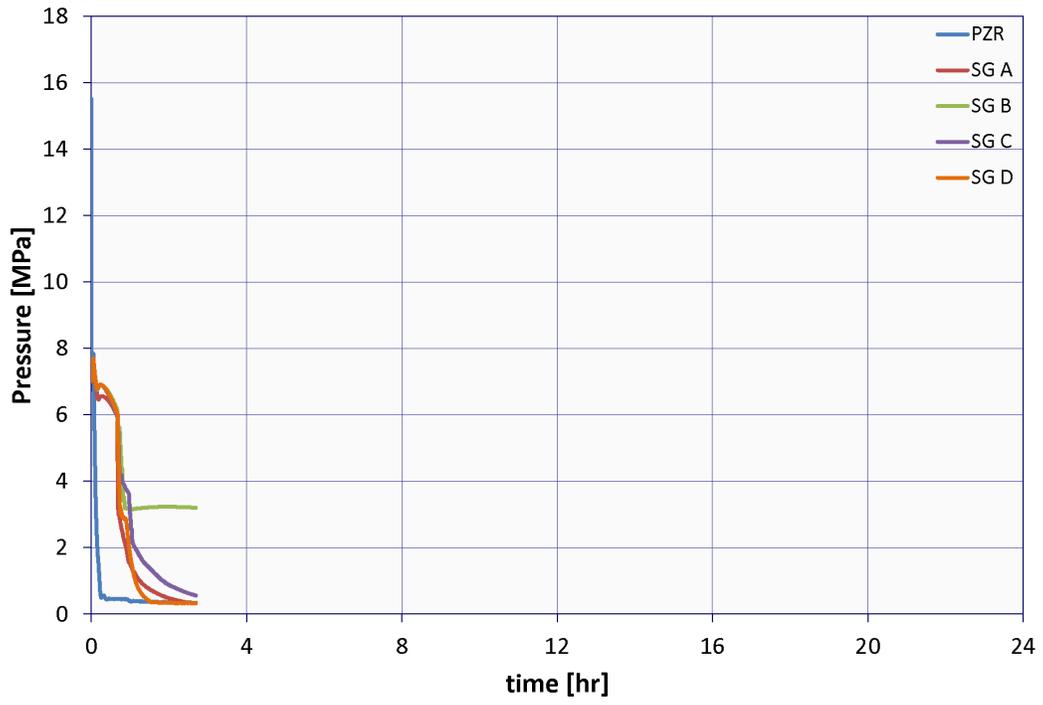


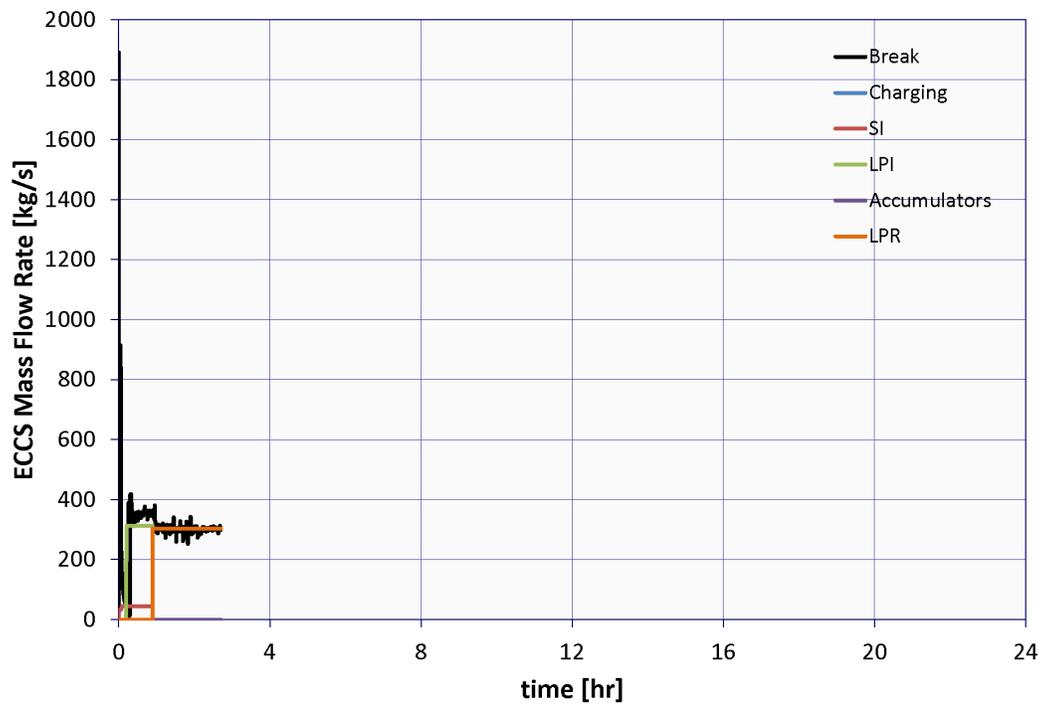
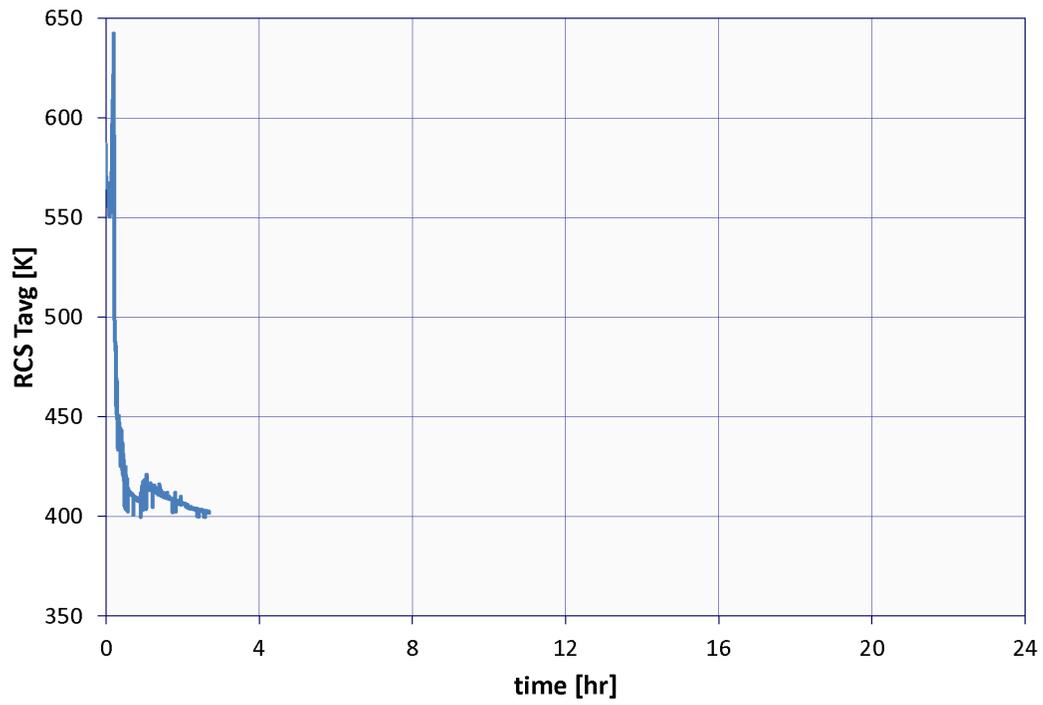


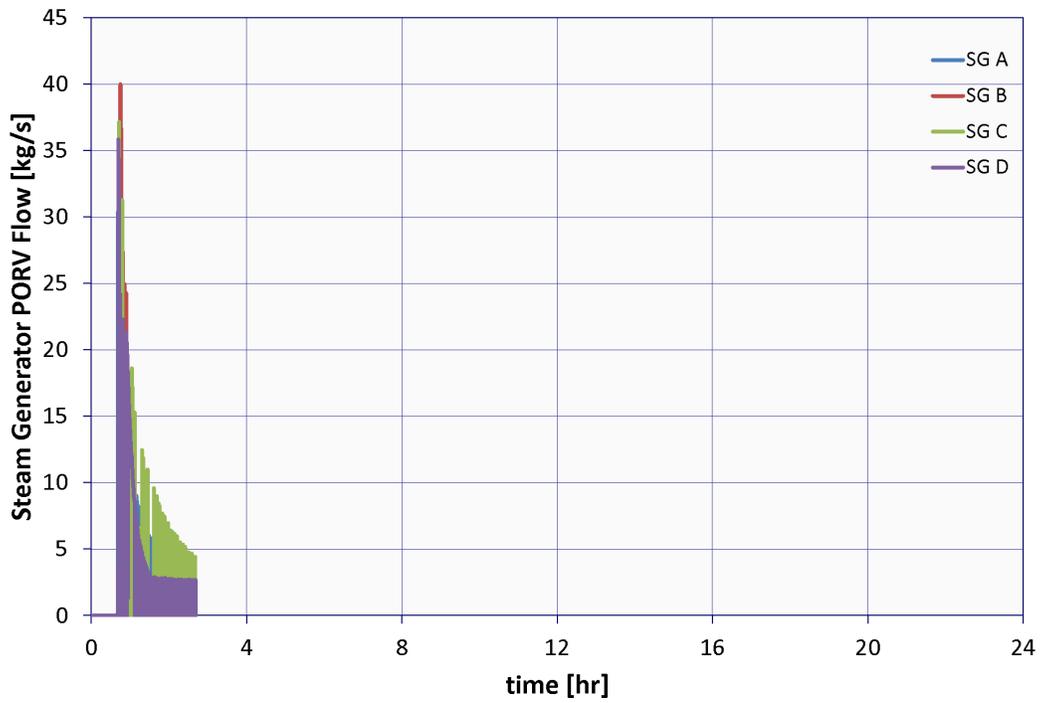
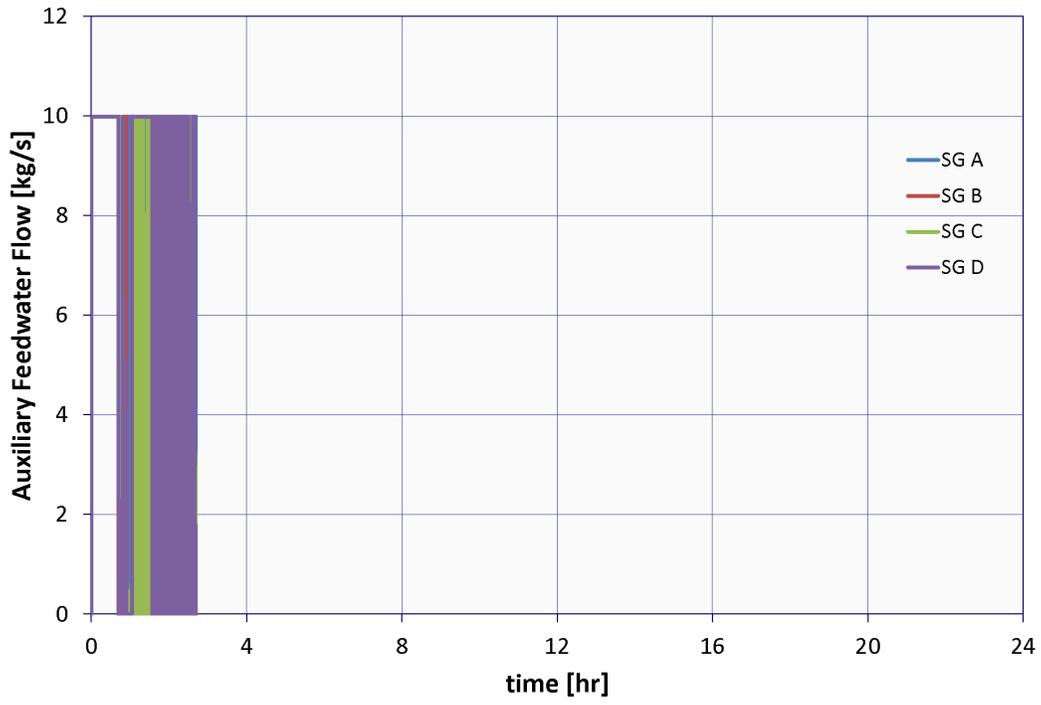
E.2.10 Case 10: 6-in. Break, 100 °F/hr Cooldown at 40 min, 0/2 CS Pumps, 0/4 RCFC

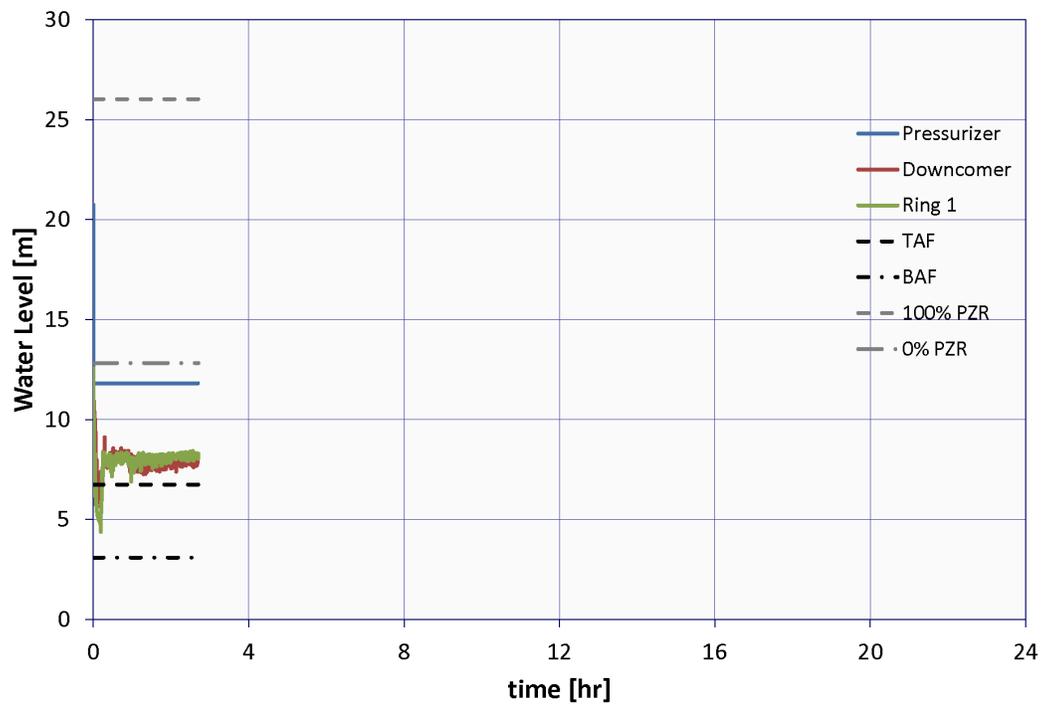
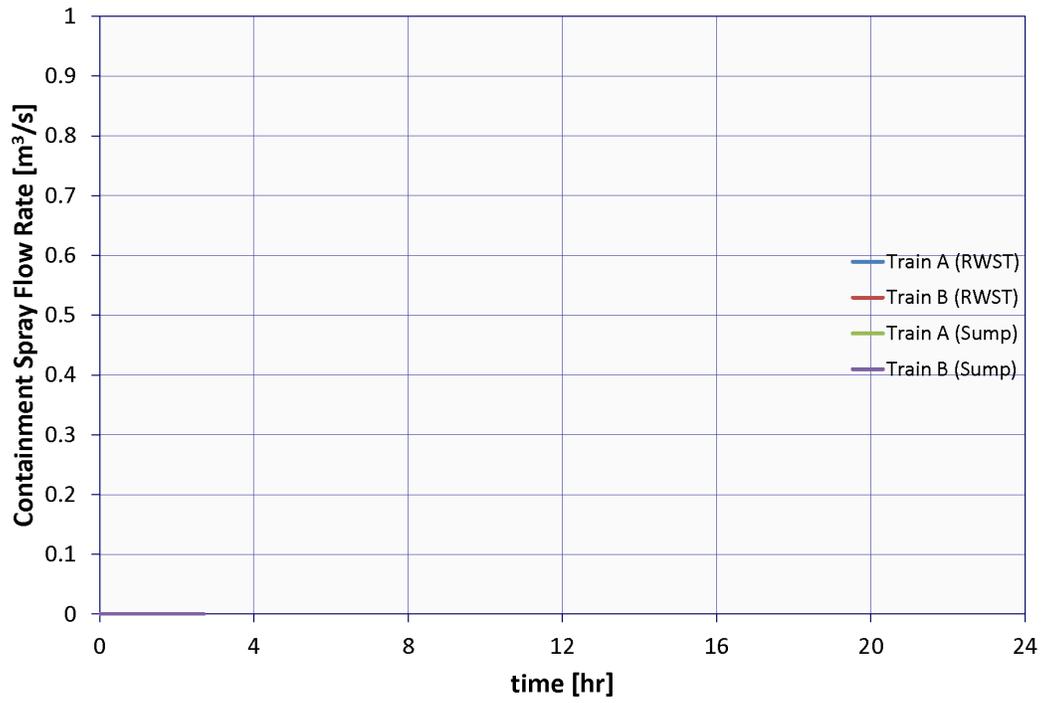


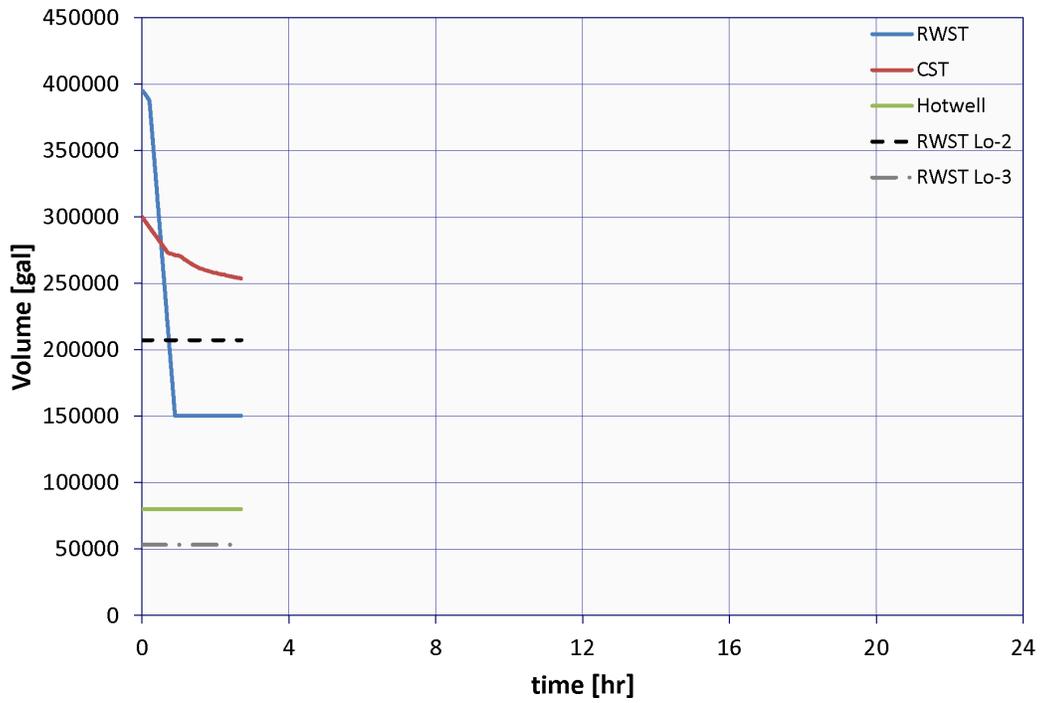
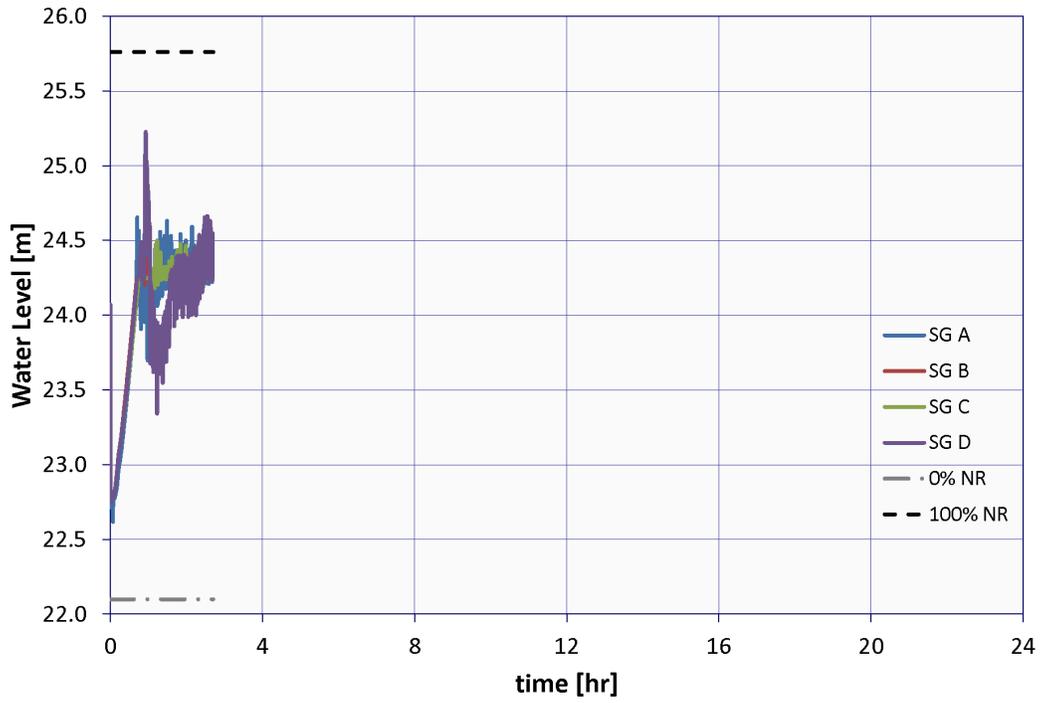














## **APPENDIX F**

### **DETAILED LOSS OF SHUTDOWN COOLING RESULTS**



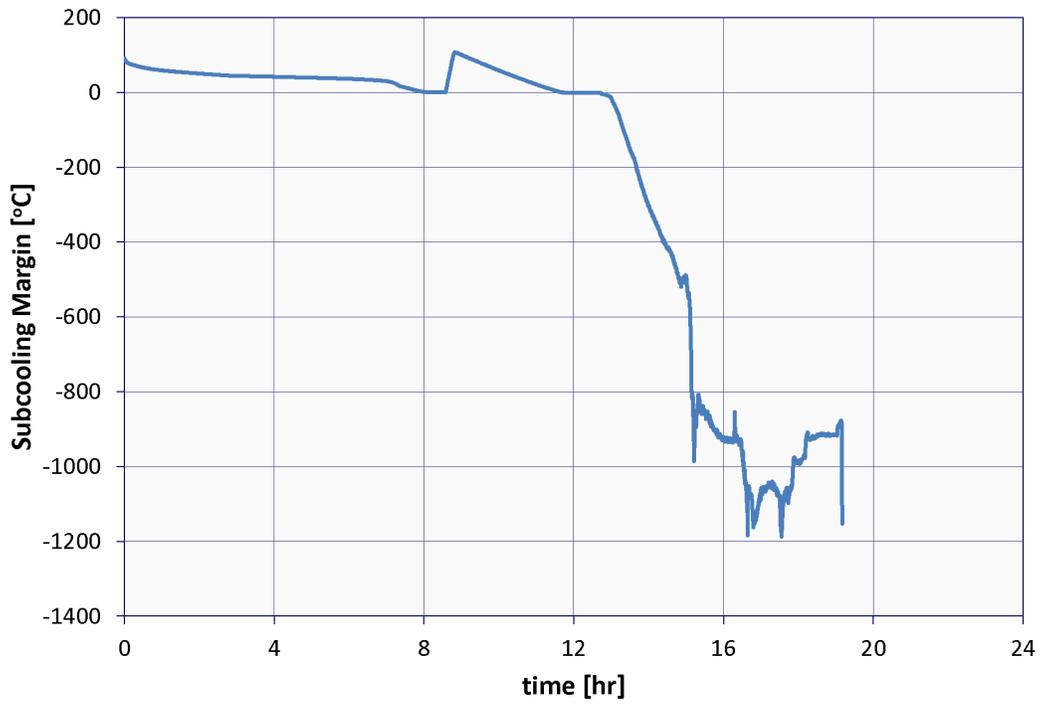
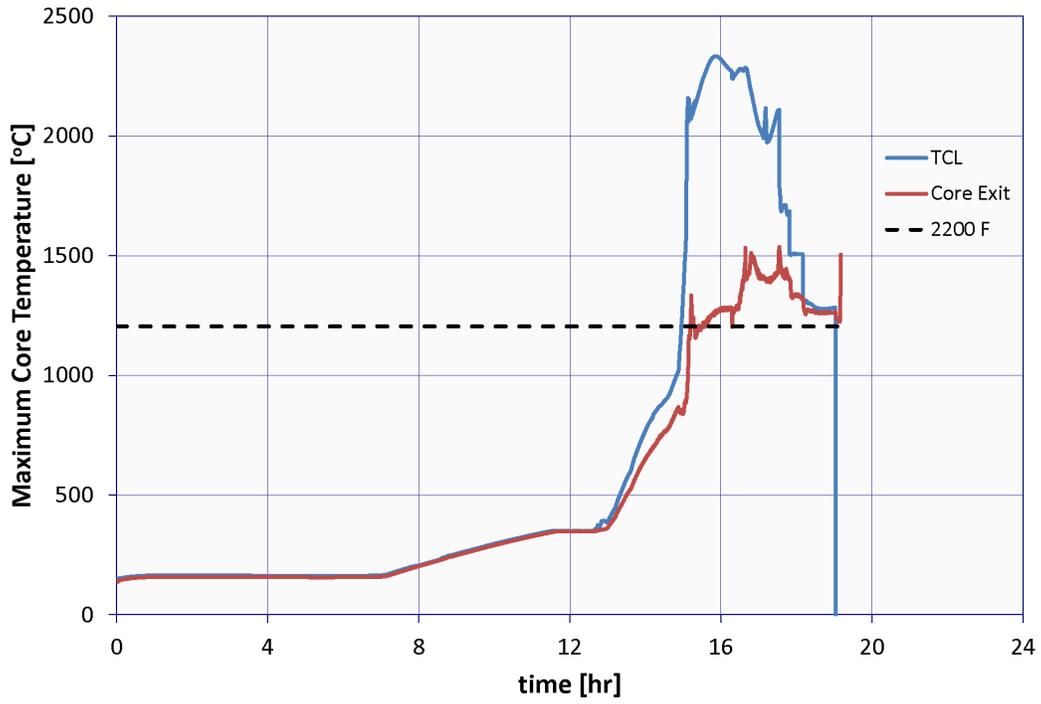
## F.1 Mode 4 Calculations

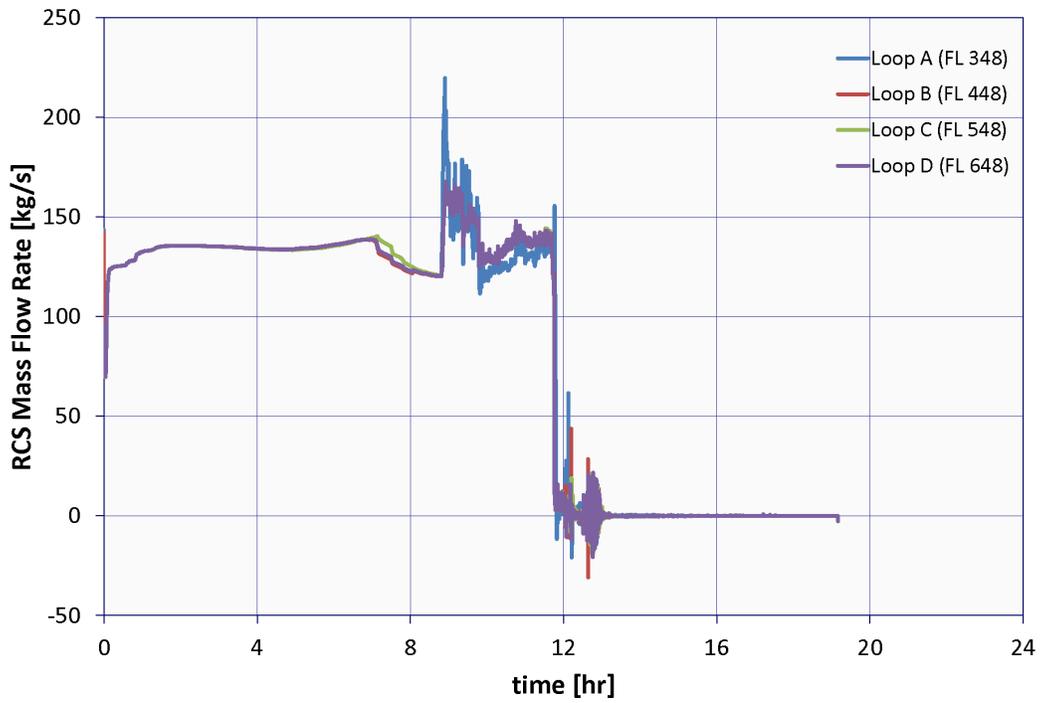
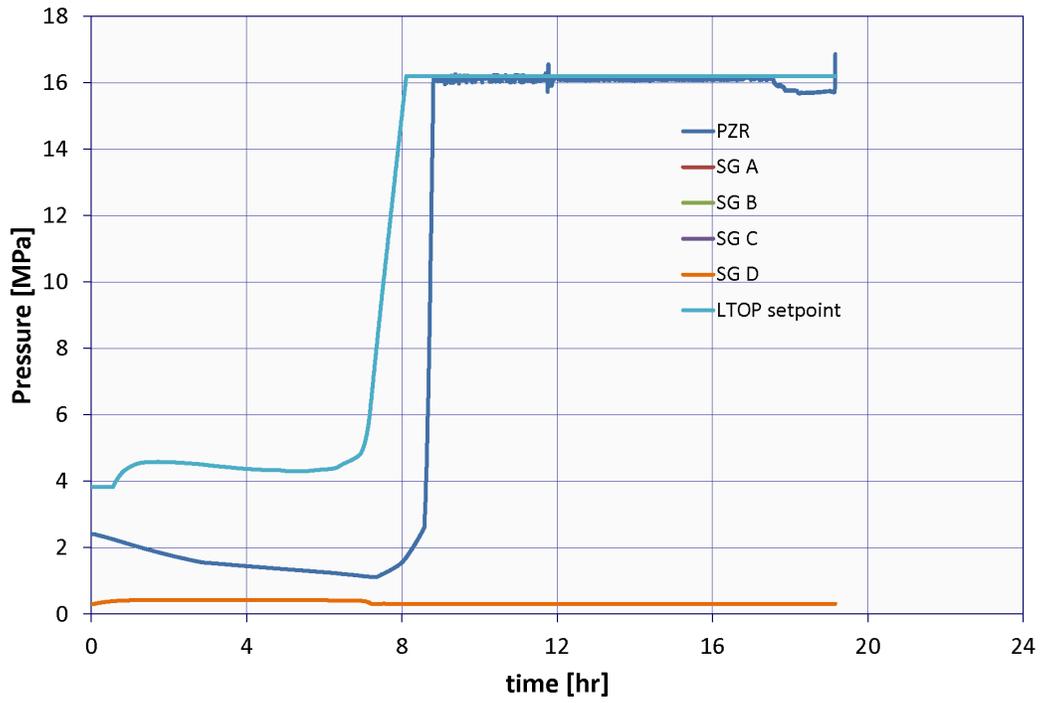
### Notes

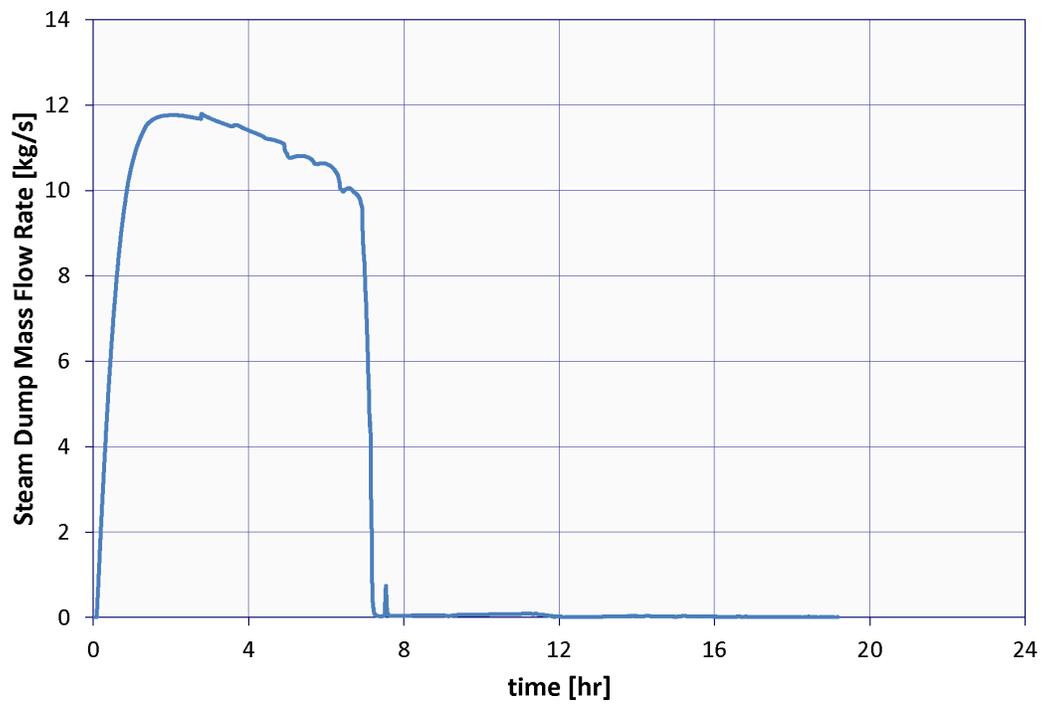
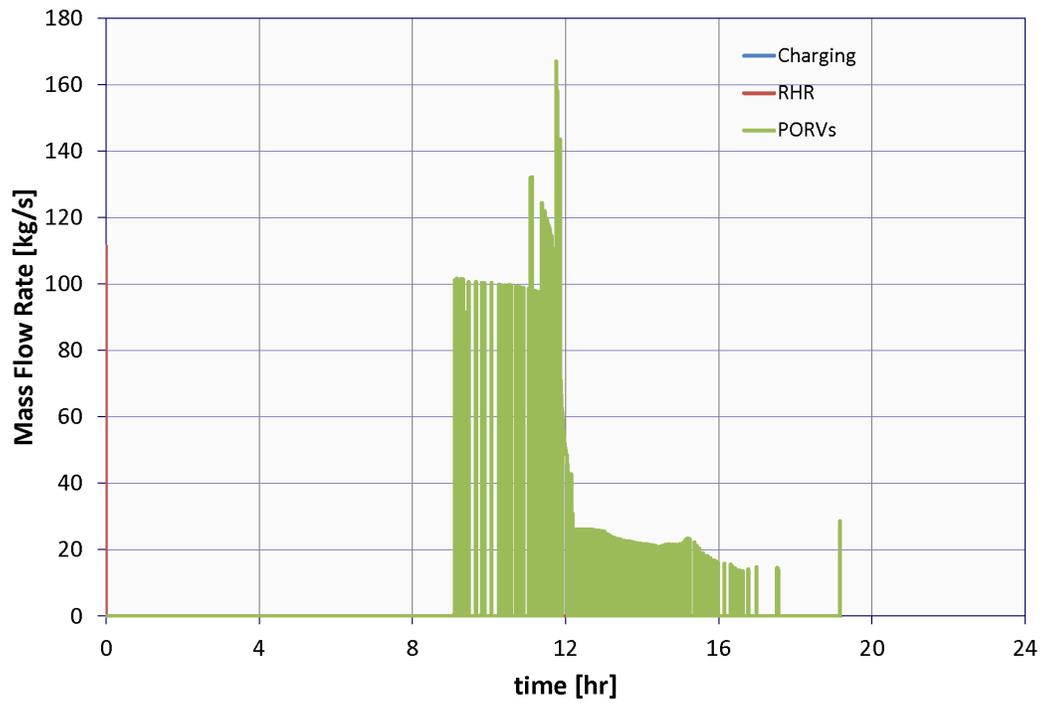
The following list identifies the major changes that were made to the MELCOR input deck in order to perform Mode 4 shutdown calculations.

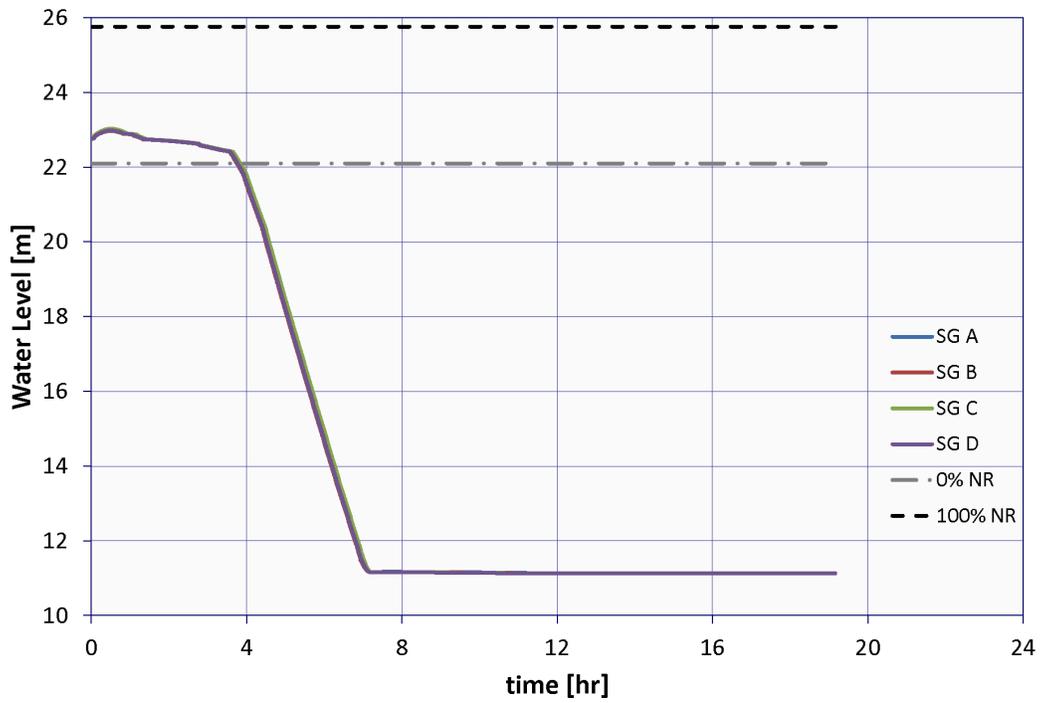
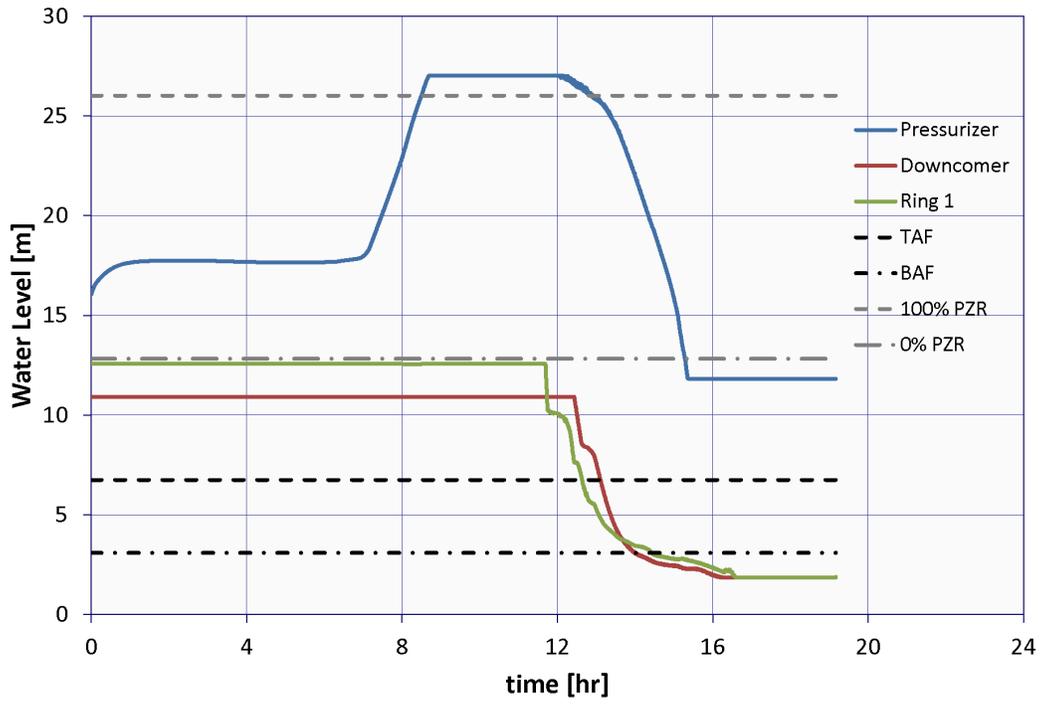
- Logic has been added to model the shutdown cooling function of the residual heat removal (RHR) system. This logic is set up such that RHR flow rate is adjusted in order to maintain a constant coolant temperature, up to the maximum flow rate of the system. The logic also includes provisions to achieve a target cooldown rate; however, this feature is not used in any of the shutdown calculations performed for this report.
- Pressurizer level control logic has been modified to control water level at the no-load setpoint (25 percent level) during the steady-state portion of the calculation.
- Similarly, pressurizer heater logic has been modified to achieve the desired pressure during the steady-state portion of the Mode 4 calculations.
- Logic that makes it possible to turn off emergency core cooling system (ECCS) flow to prevent overfilling the pressurizer has been modified in order to simulate recovery actions in which operators inject using a charging pump when reactor pressure vessel (RPV) level is low. This feature is exercised in Mode 4 Cases 2 and 5.
- The decay heat curves have been shifted in order to simulate the desired times after trip. For example, the decay heat curve is shifted by 12 hours for Mode 4 Cases 1–5. Note that during the steady-state portion of the calculation, the decay heat is assumed to be constant and to equal the decay power at 12 hours. The same is true for all other times since subcriticality that are analyzed in Section 5.8.2 of the report.
- Initial temperature and pressure of reactor coolant system (RCS) control volumes have been set to 275 degrees Fahrenheit (F) (408.15 Kelvin (K)) and 350 pounds per square inch absolute (psia) (2.413 megapascals (MPa)).
- Secondary-side temperatures (including feedwater temperature) have been set to 275 degrees F (408.15 K).
- Logic for the steam dump valves has been modified to maintain secondary-side pressure at 45 psia (0.313 MPa), which is the saturation pressure at 275 degrees F (408.15 K).
- Steam generator water level logic has been modified so that steady-state water level is controlled at 18 percent narrow range (NR) or 27 percent wide range (WR) level, depending on the case being analyzed.
- Cold volumes have been used in place of hot volumes for RCS control volumes. This decreases the RCS volume by approximately 1 percent.

**F.1.1 Case 1: SG at 18% NR Level, 12 hr after Shutdown, No Recovery Actions**



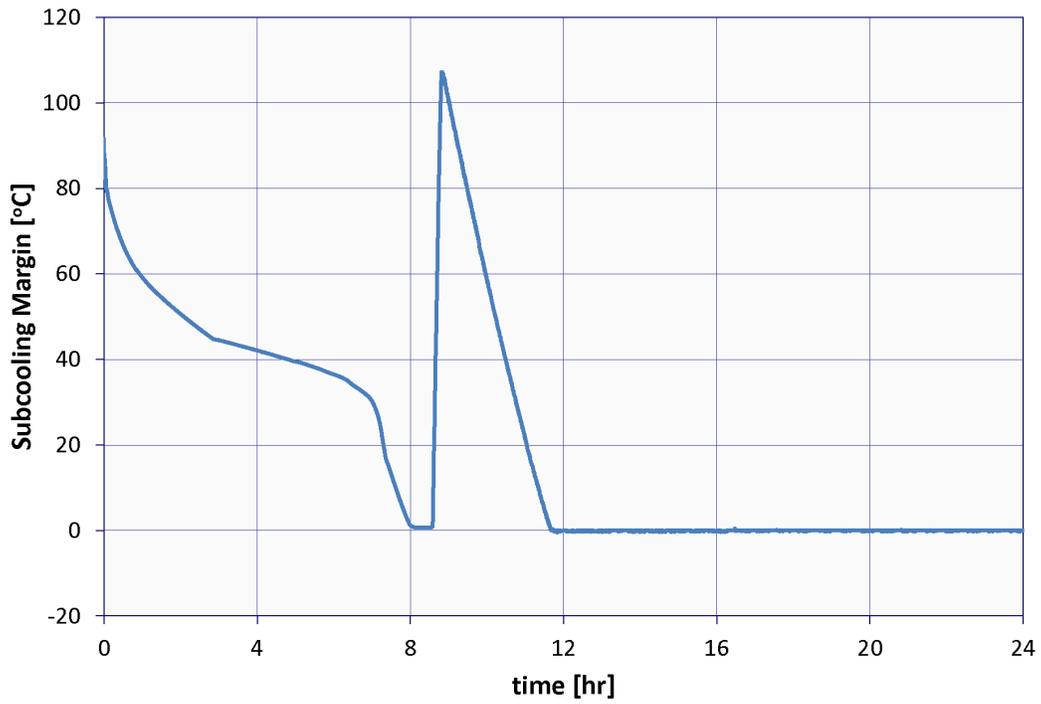
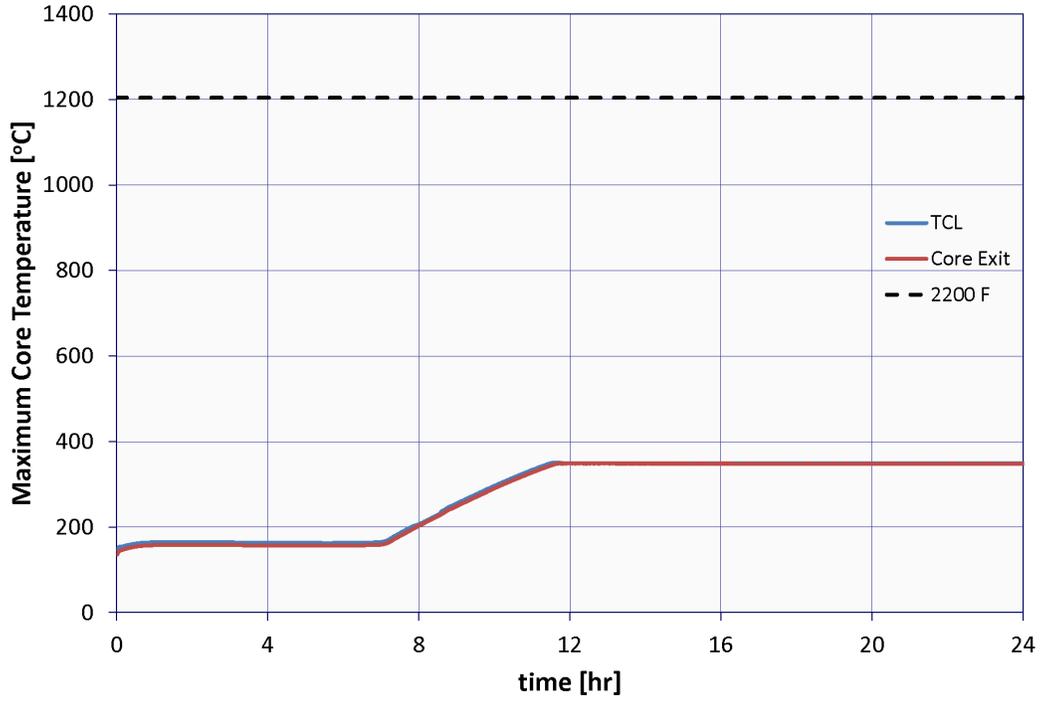


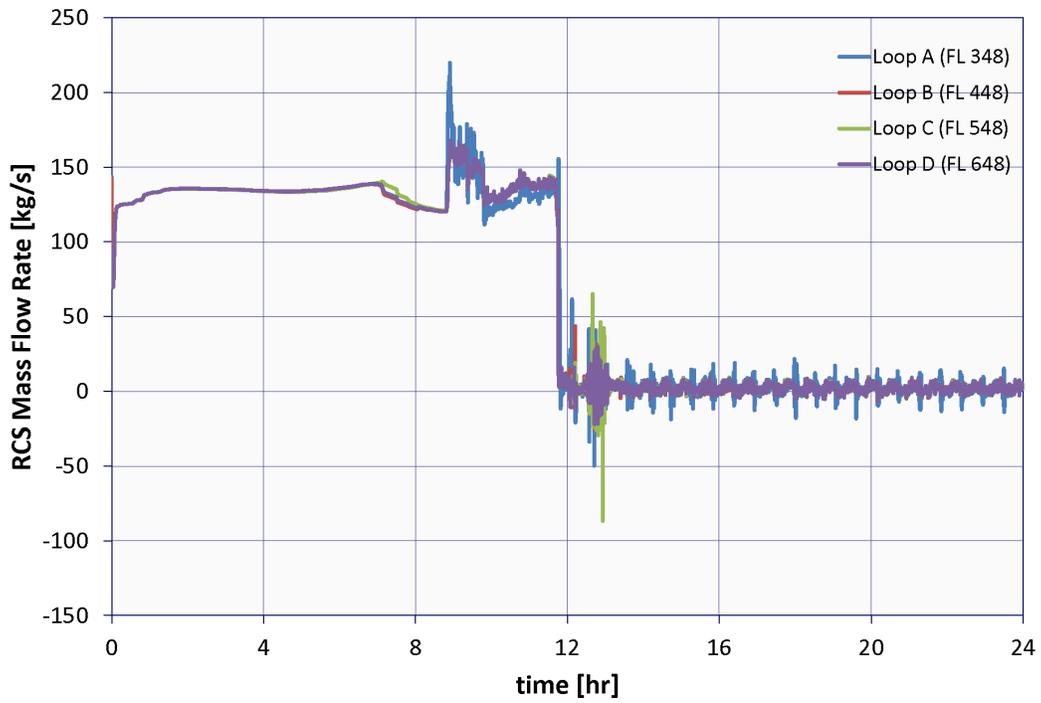
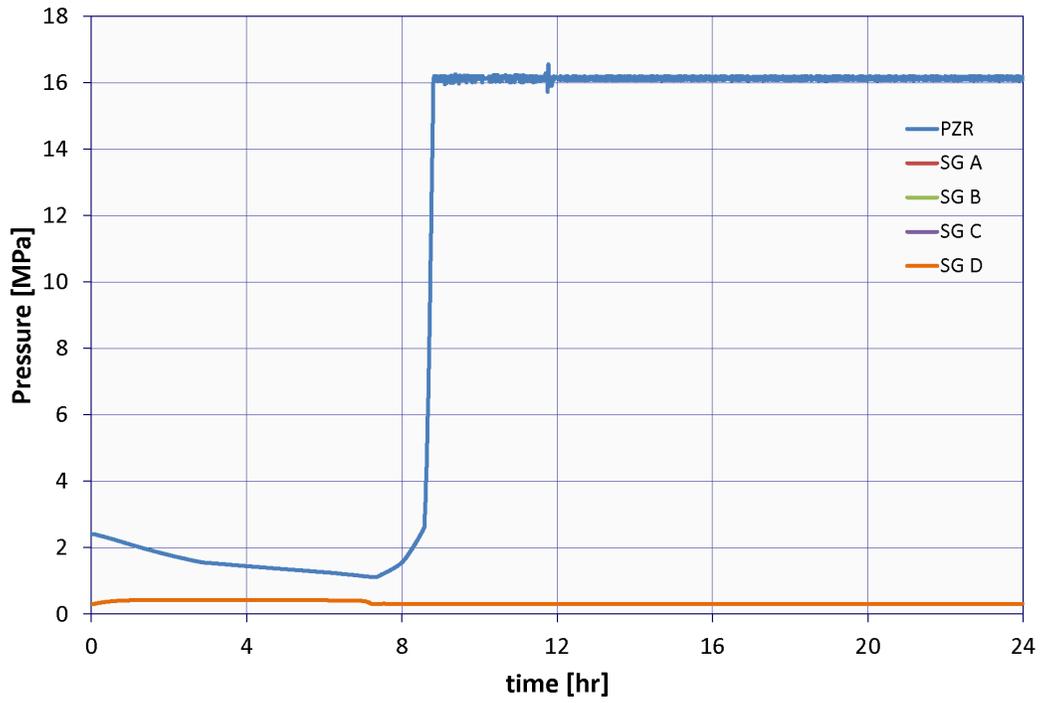


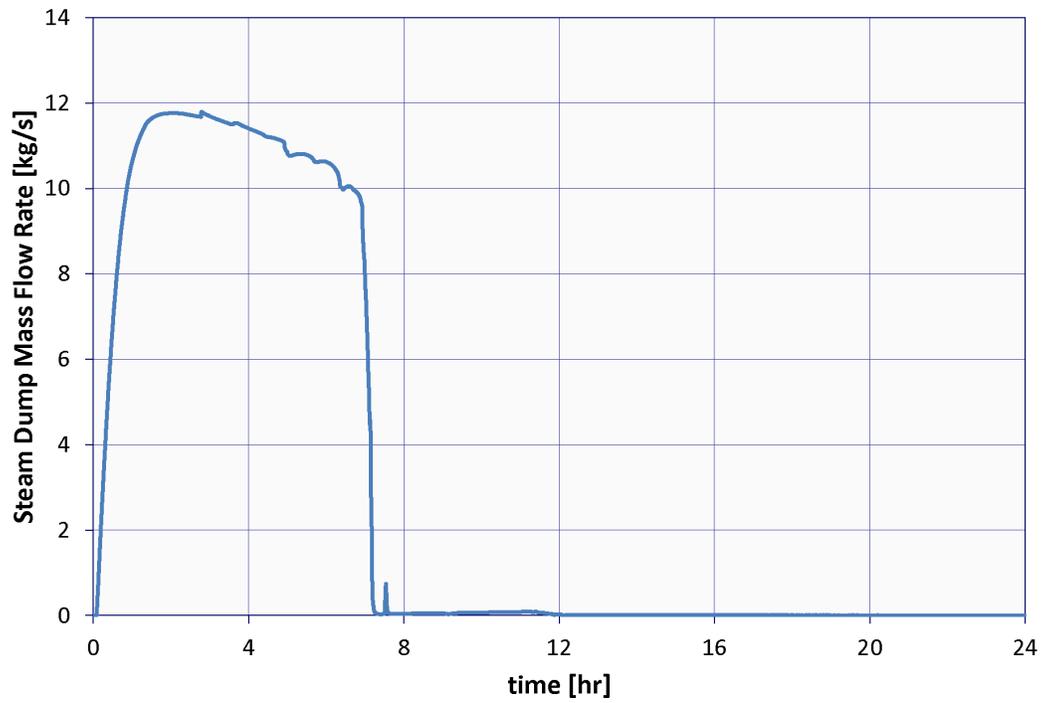
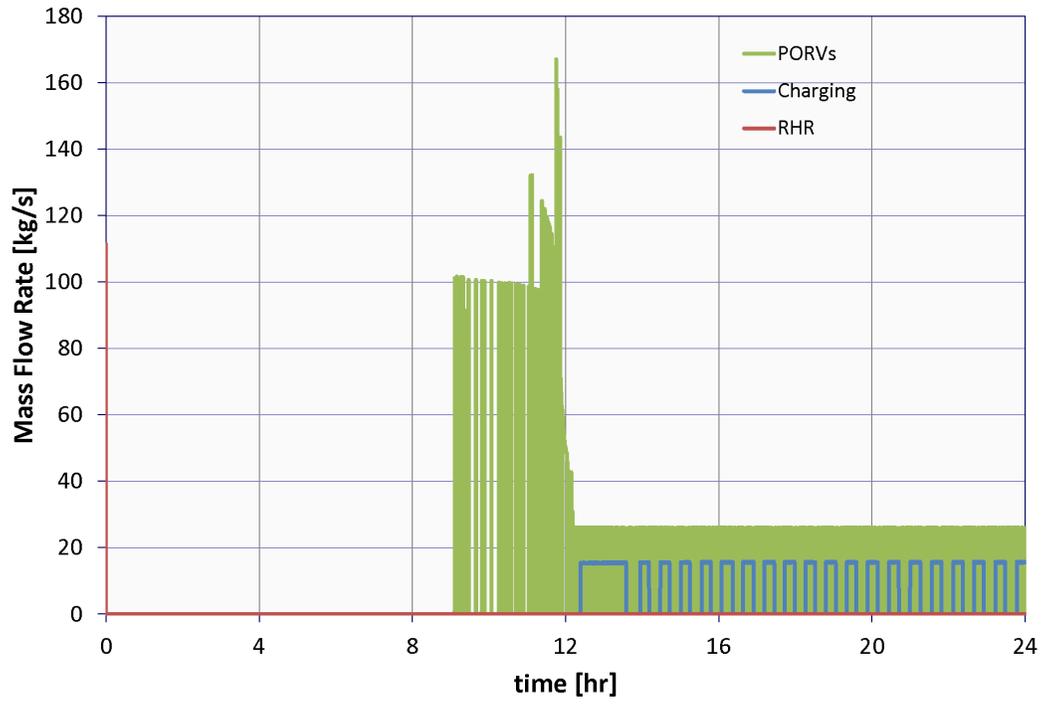


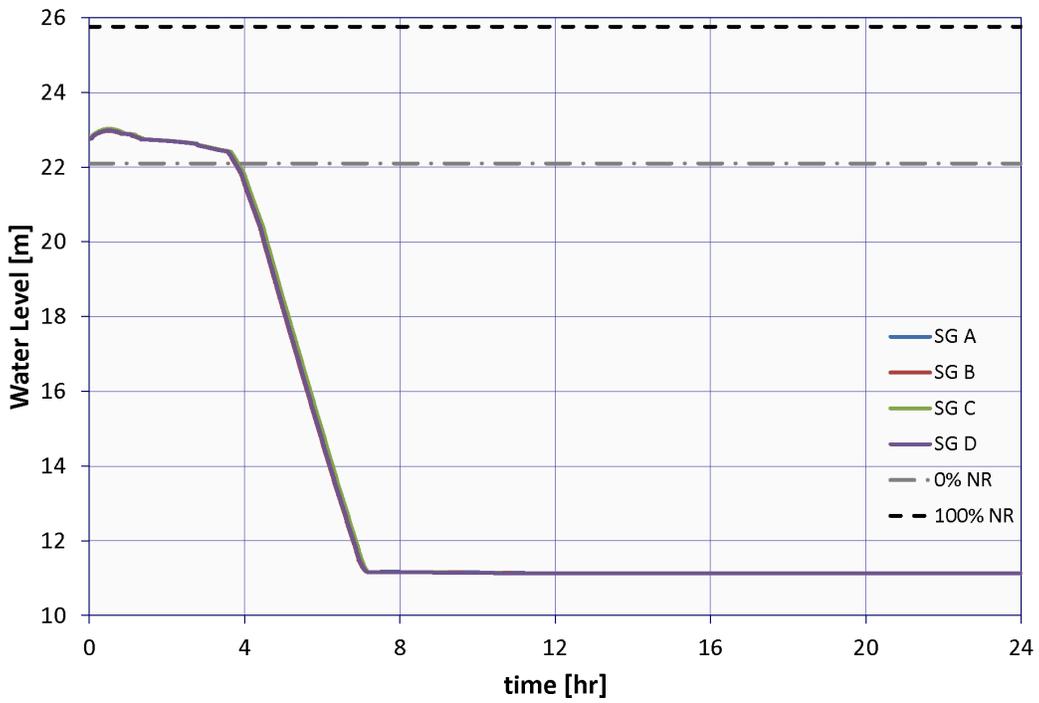
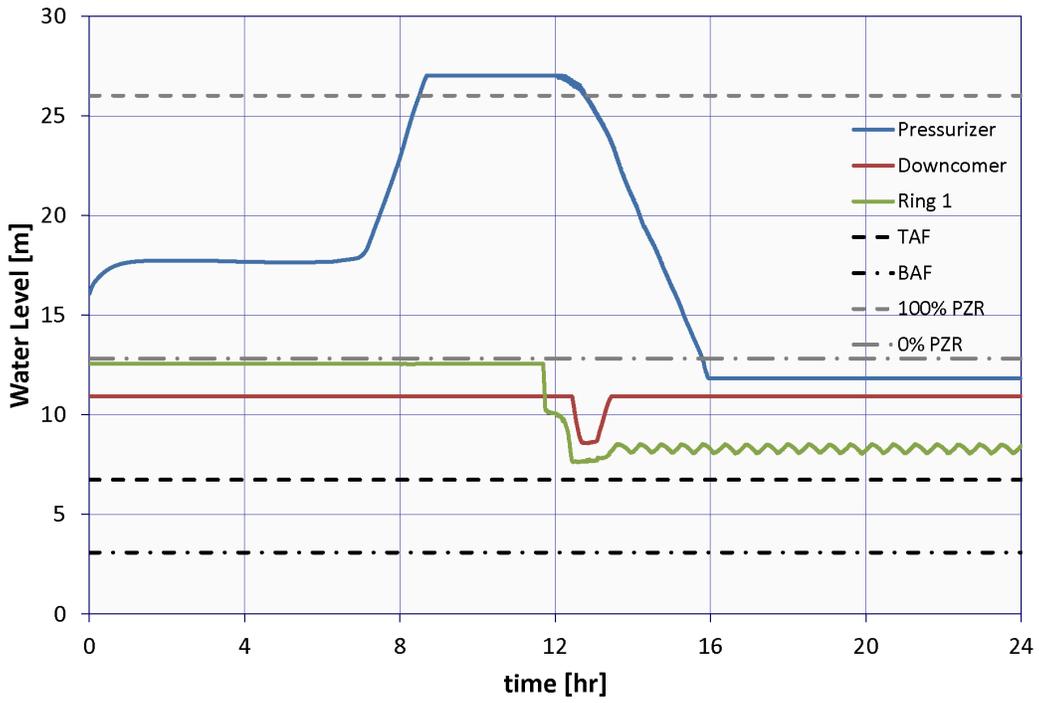
F.1.2

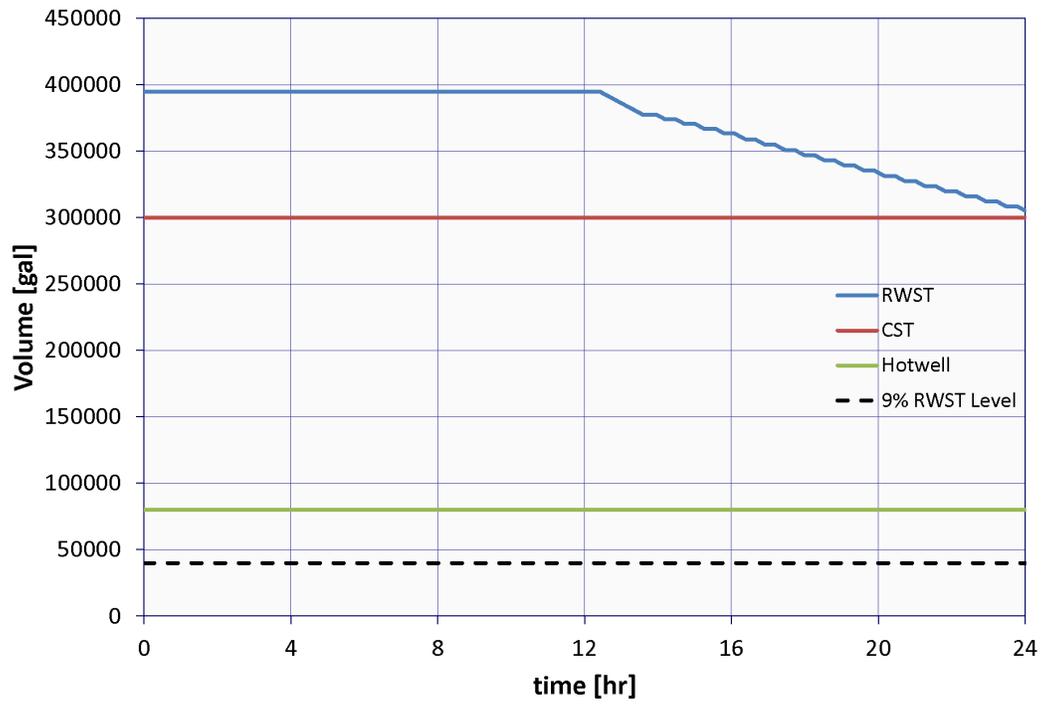
Case 2: SG at 18% NR Level, 12 hr after Shutdown, Start CCP on Low RPV Level



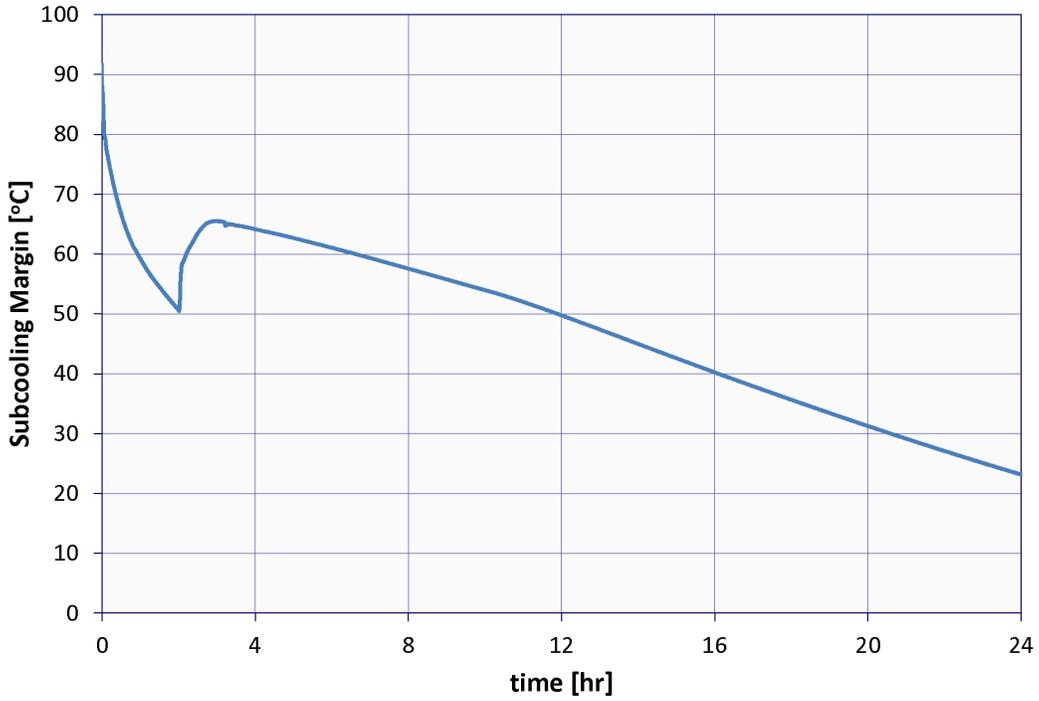
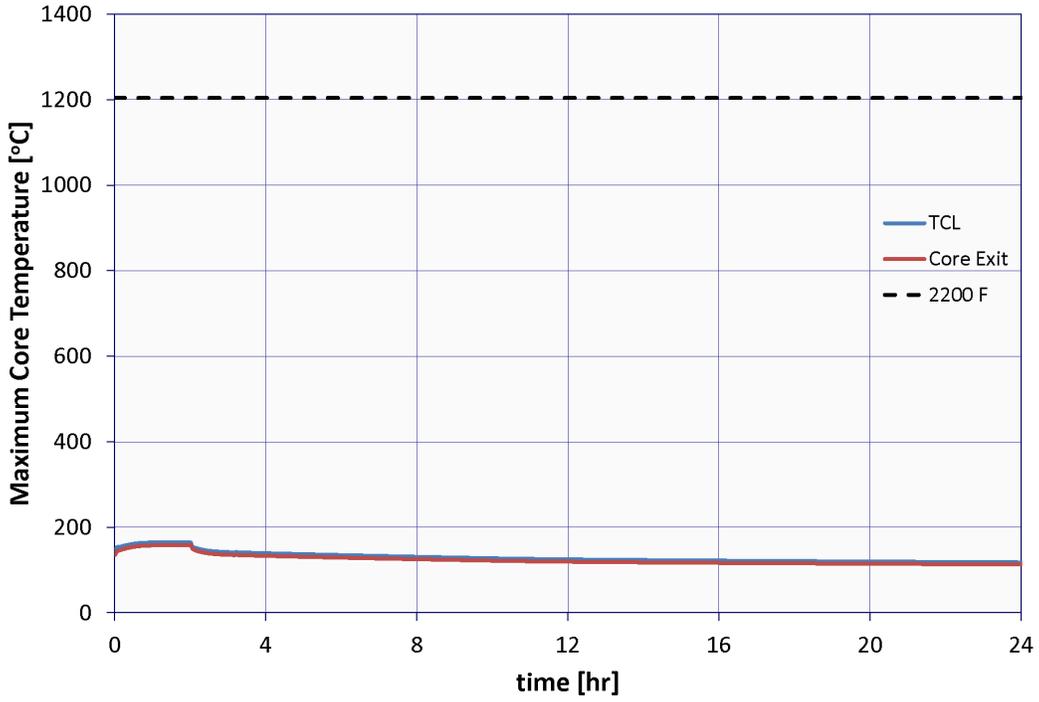


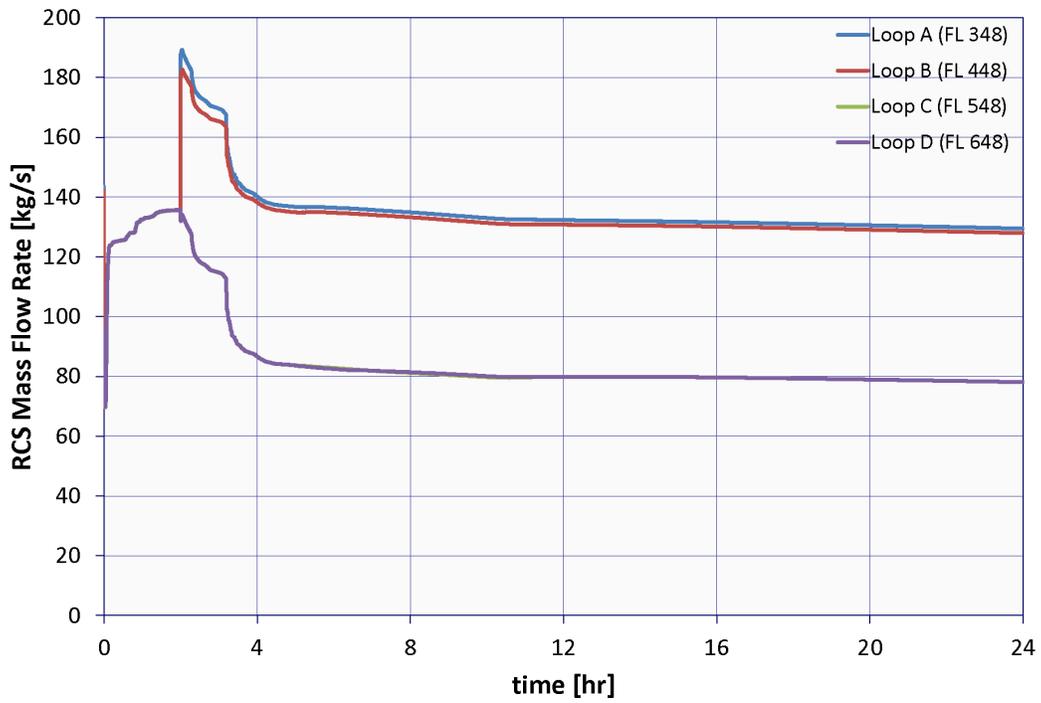
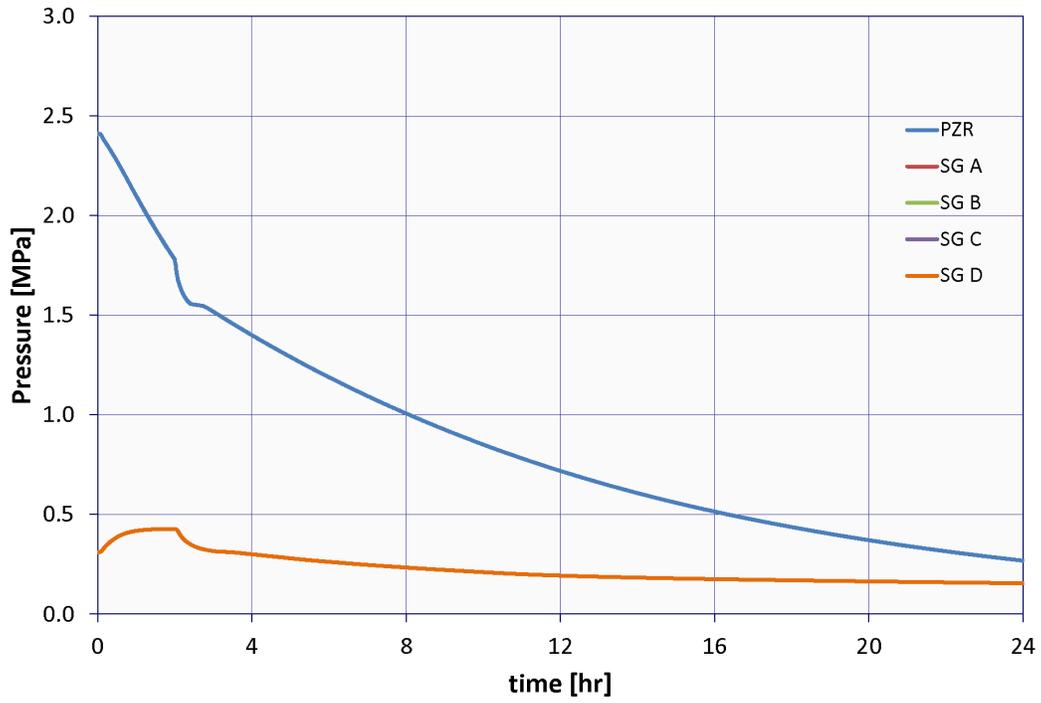


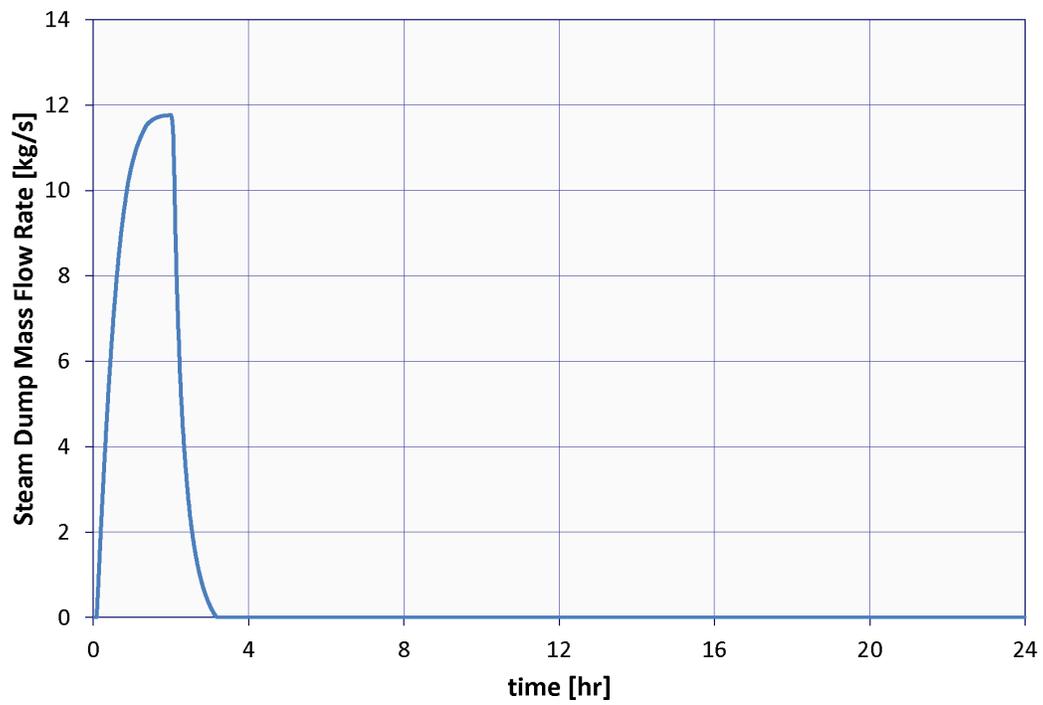
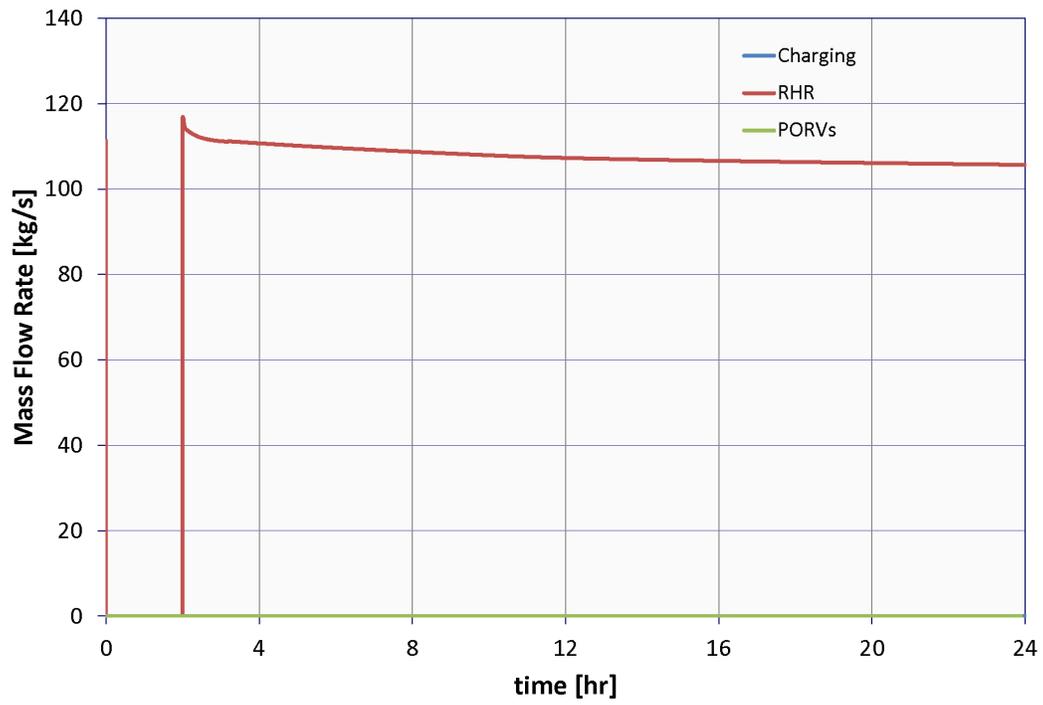


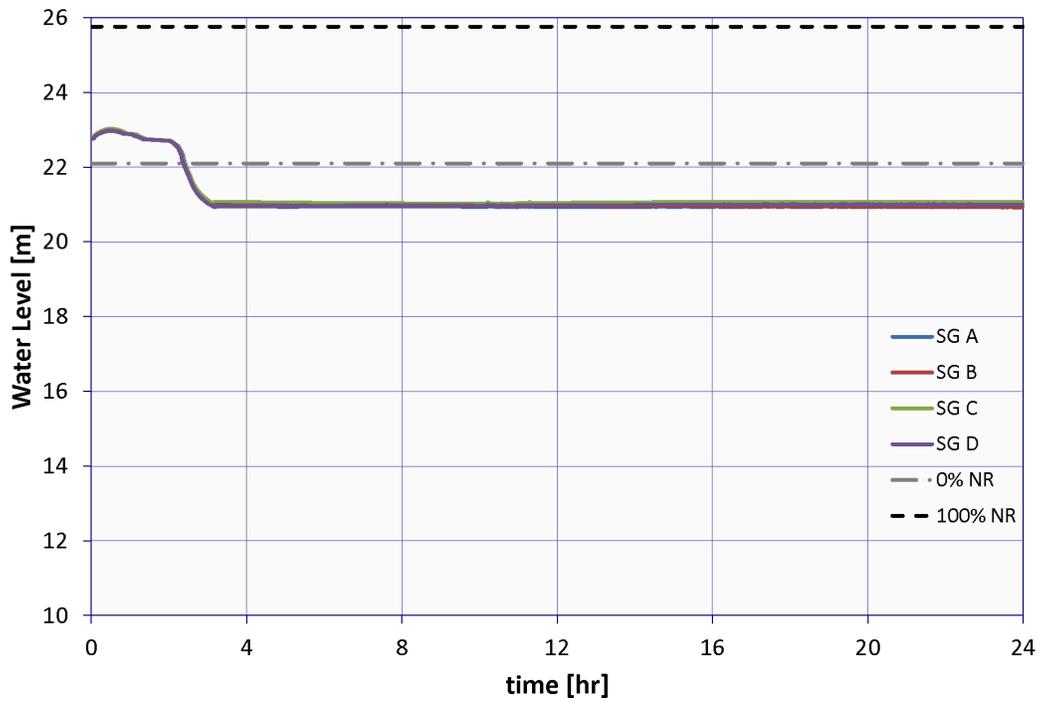
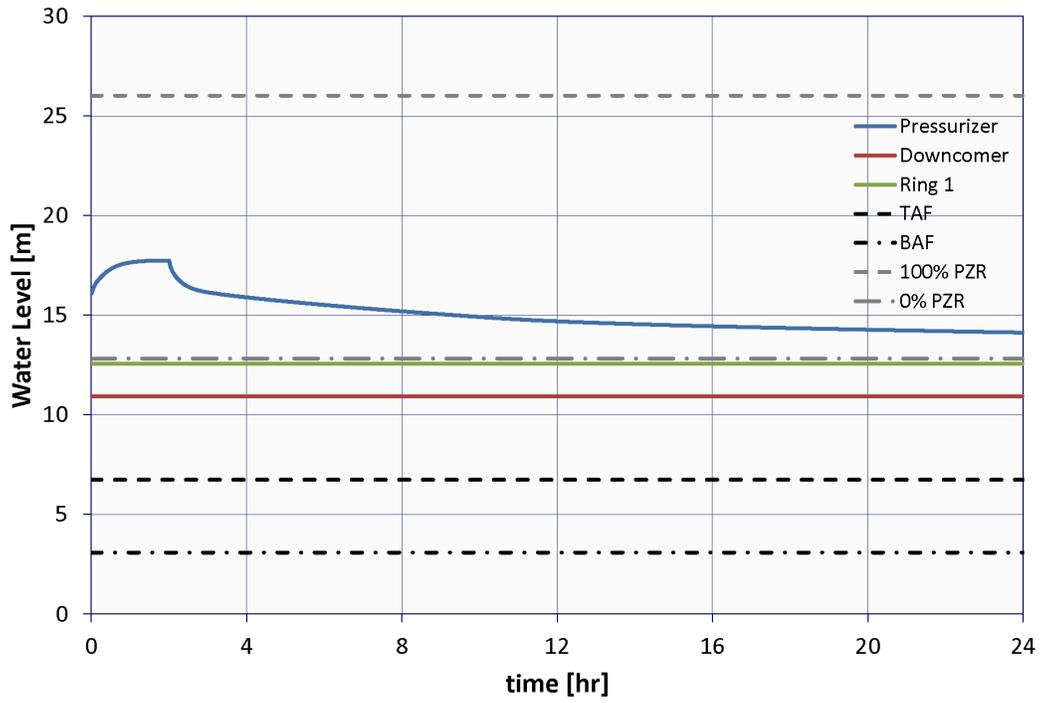


**F.1.3 Case 3: SG at 18% NR Level, 12 hr after Shutdown, Recover RHR at 2 hr**

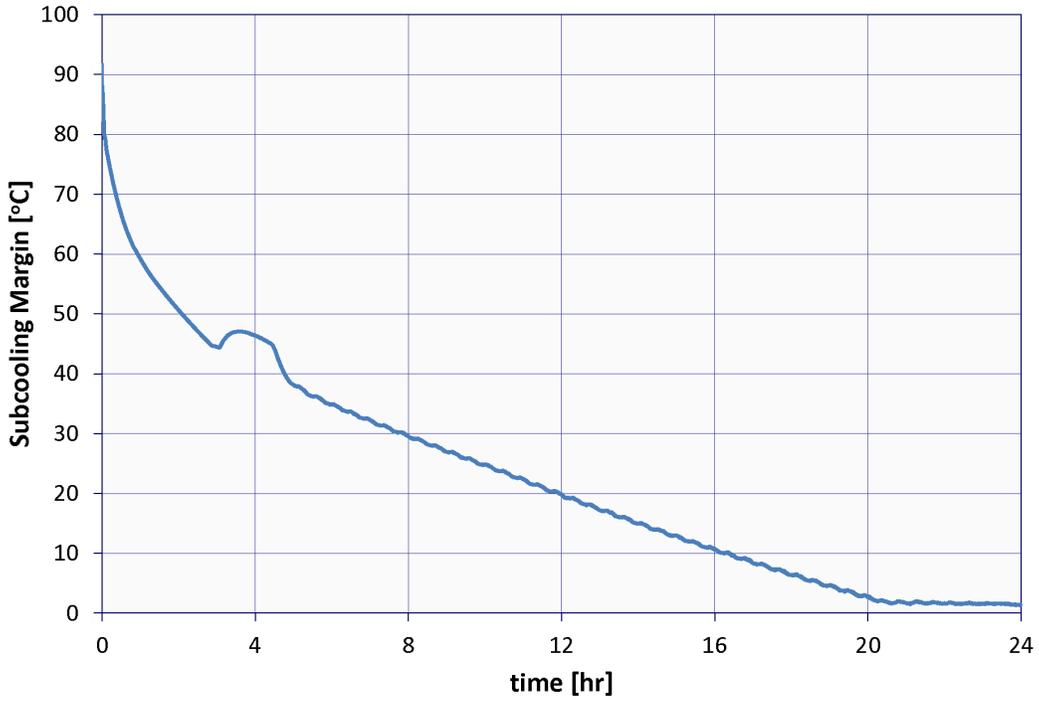
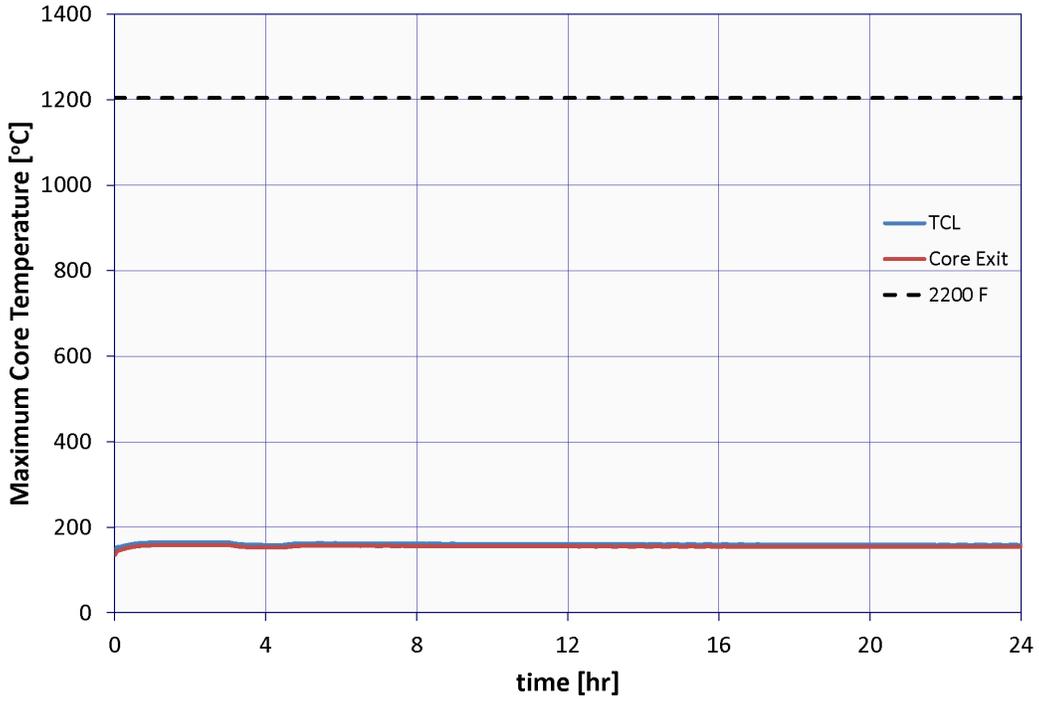


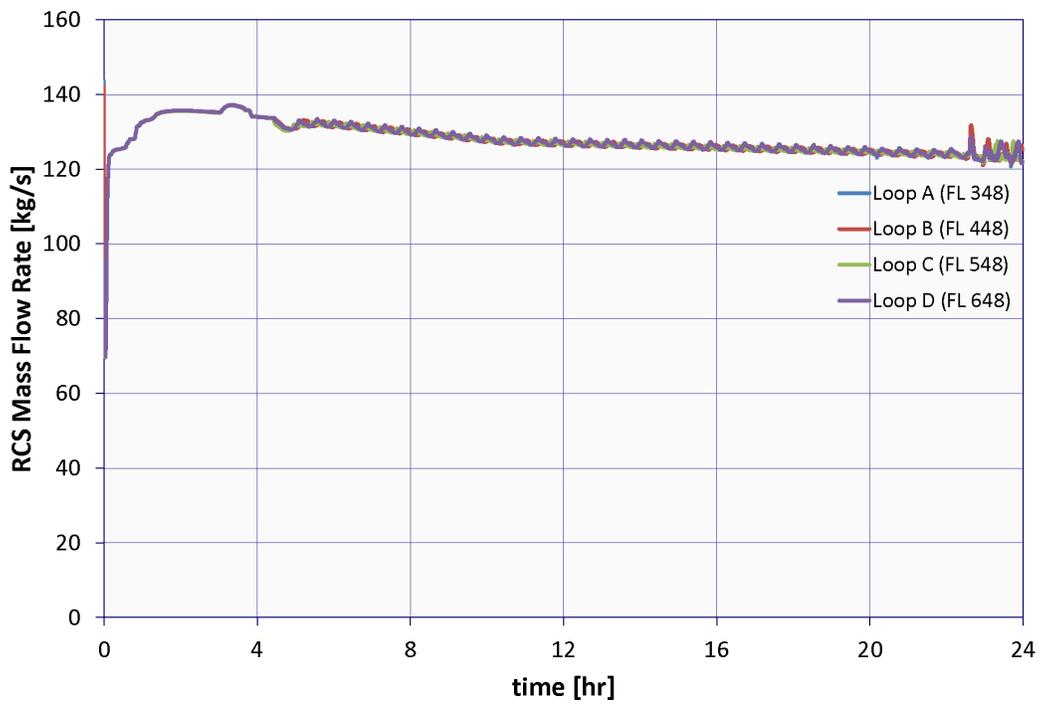
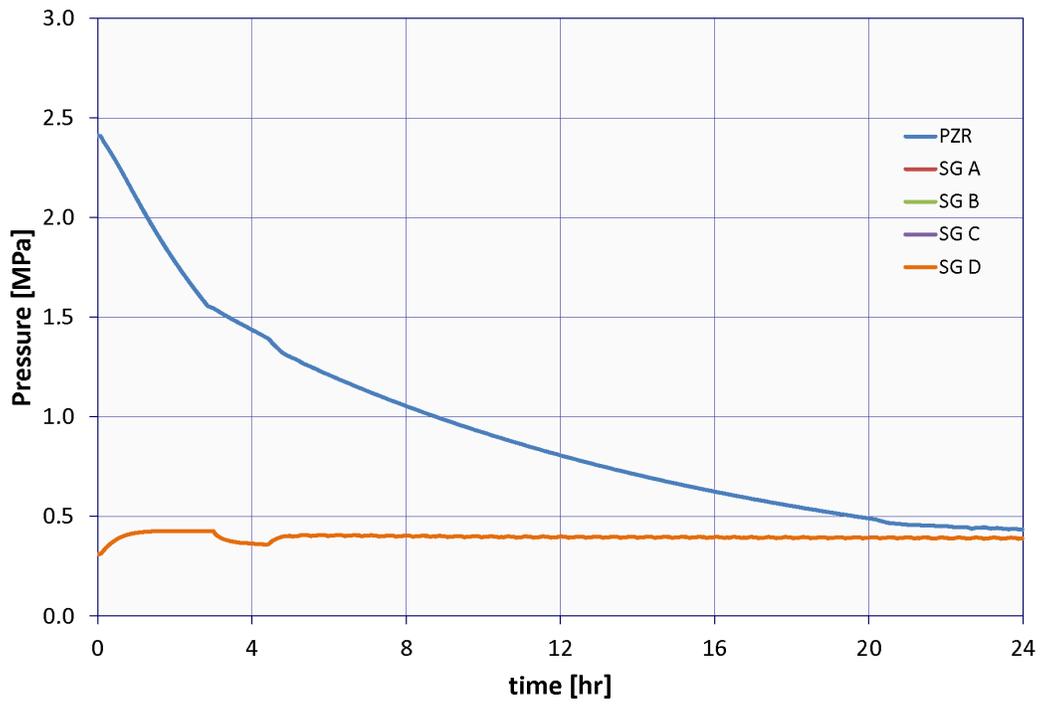


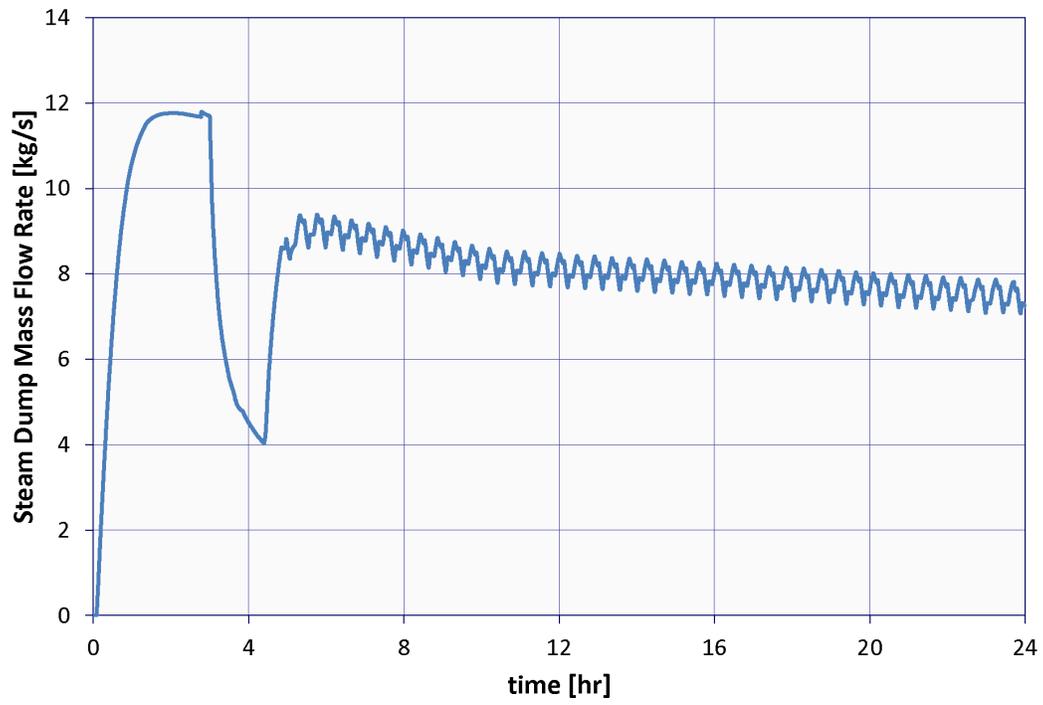
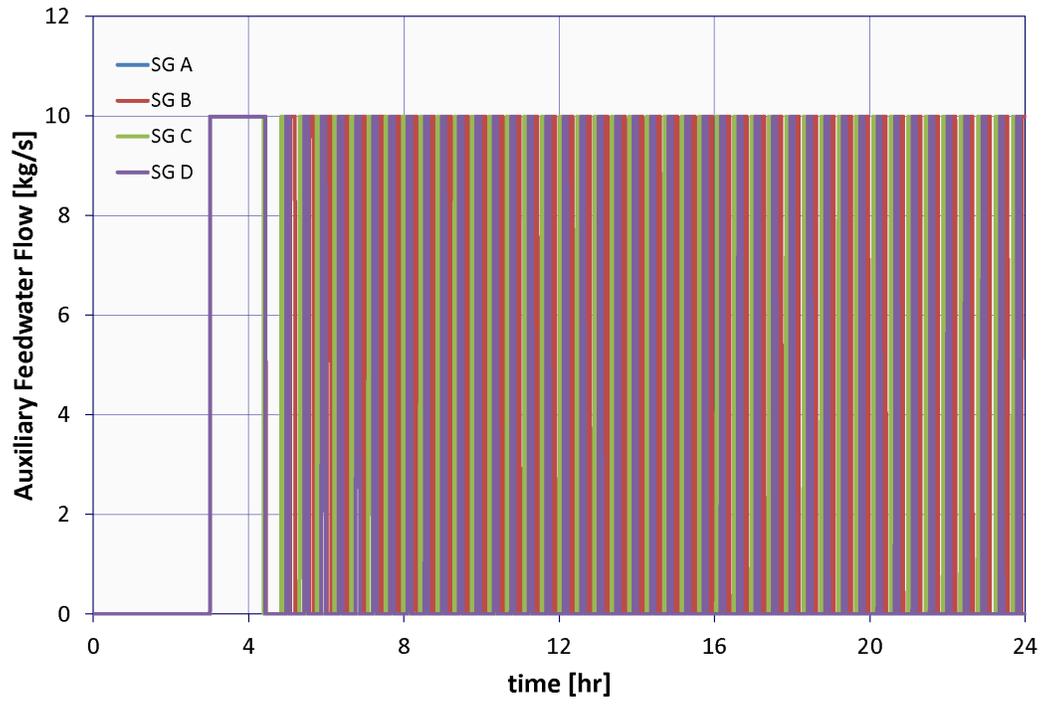


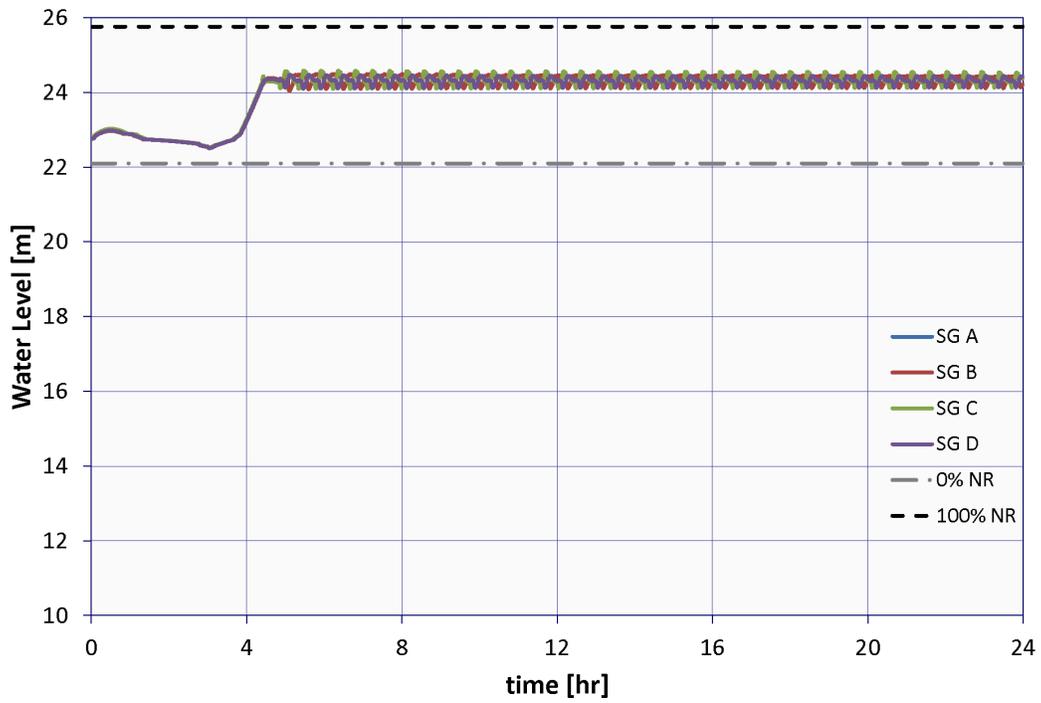
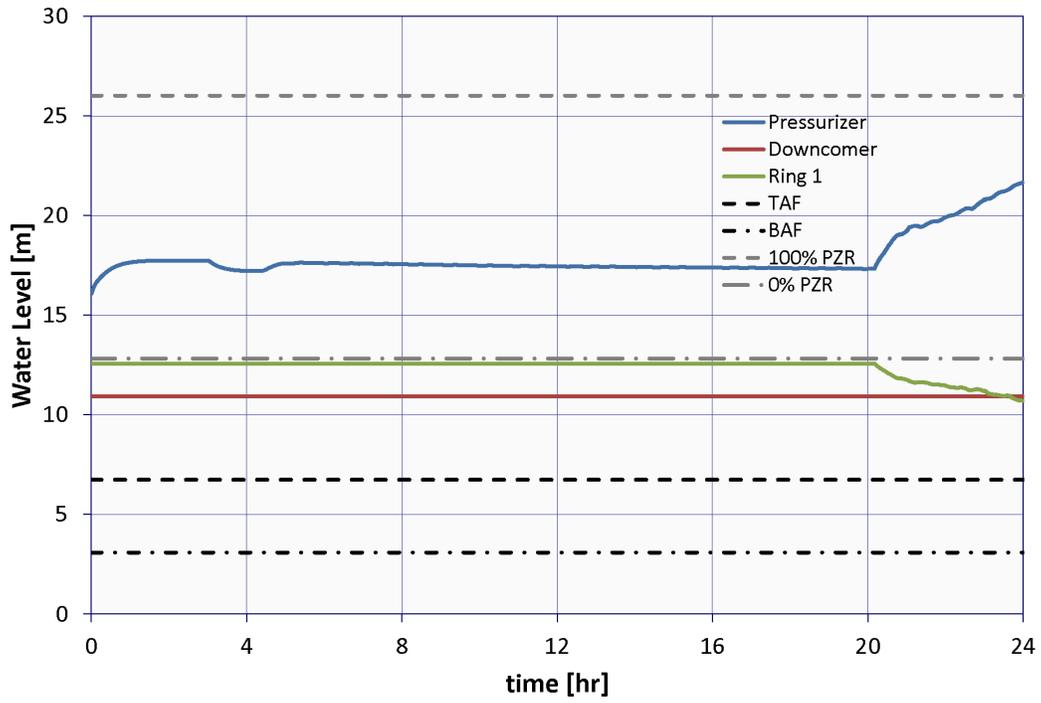


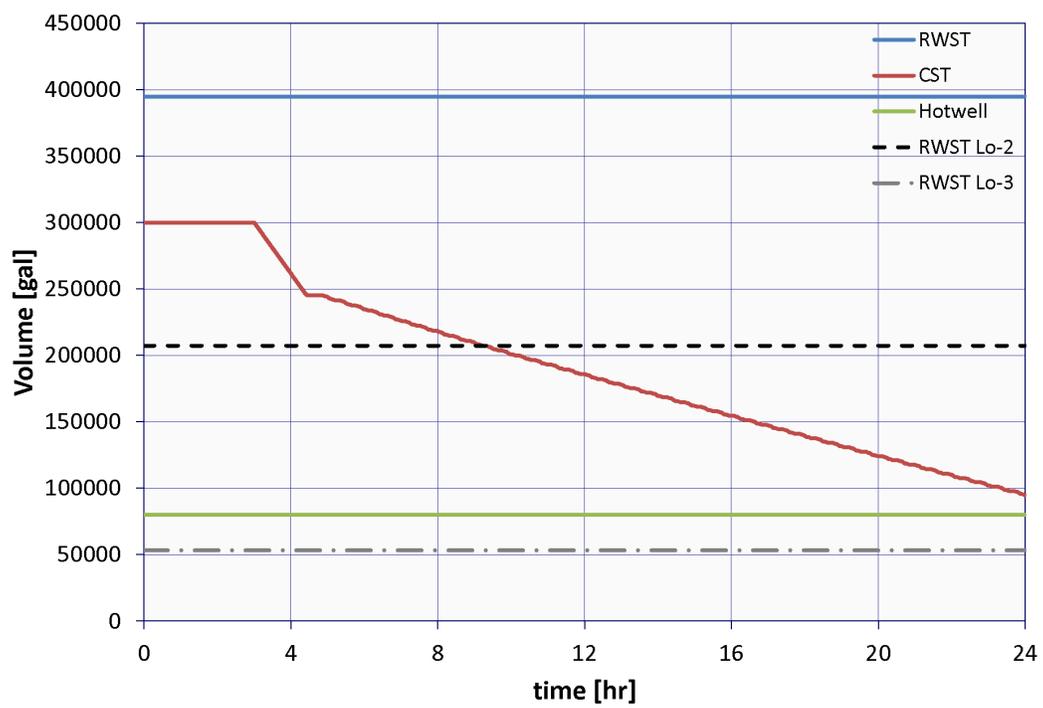
**F.1.4 Case 4: SG at 18% NR Level, 12 hr after Shutdown, Initiate AFW at 3 hr**



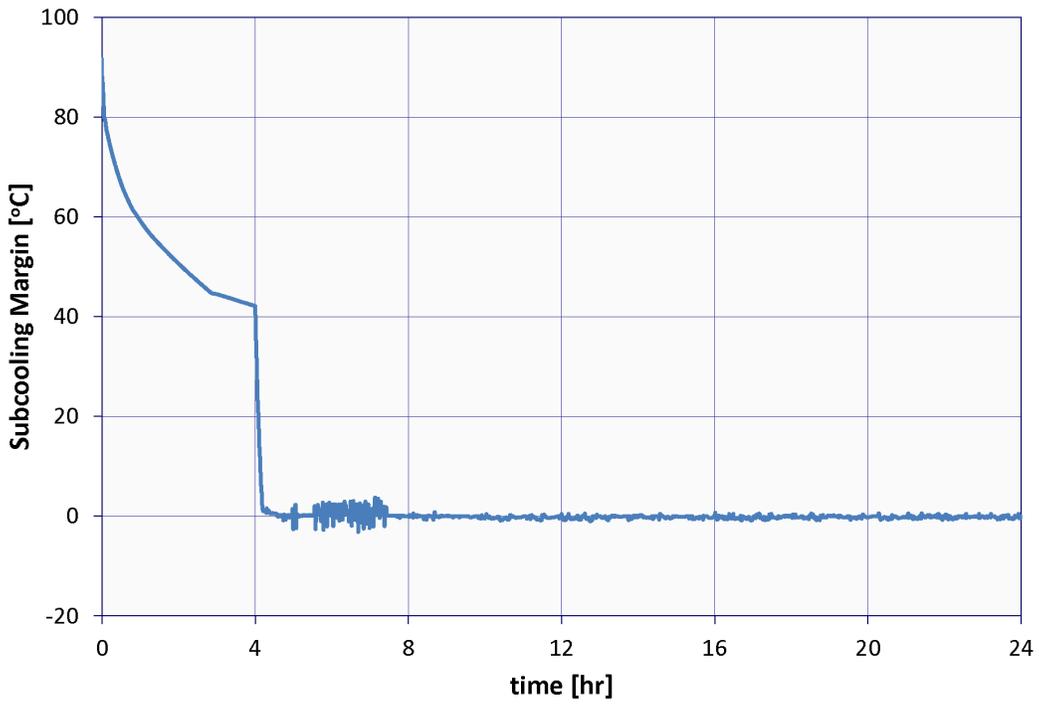
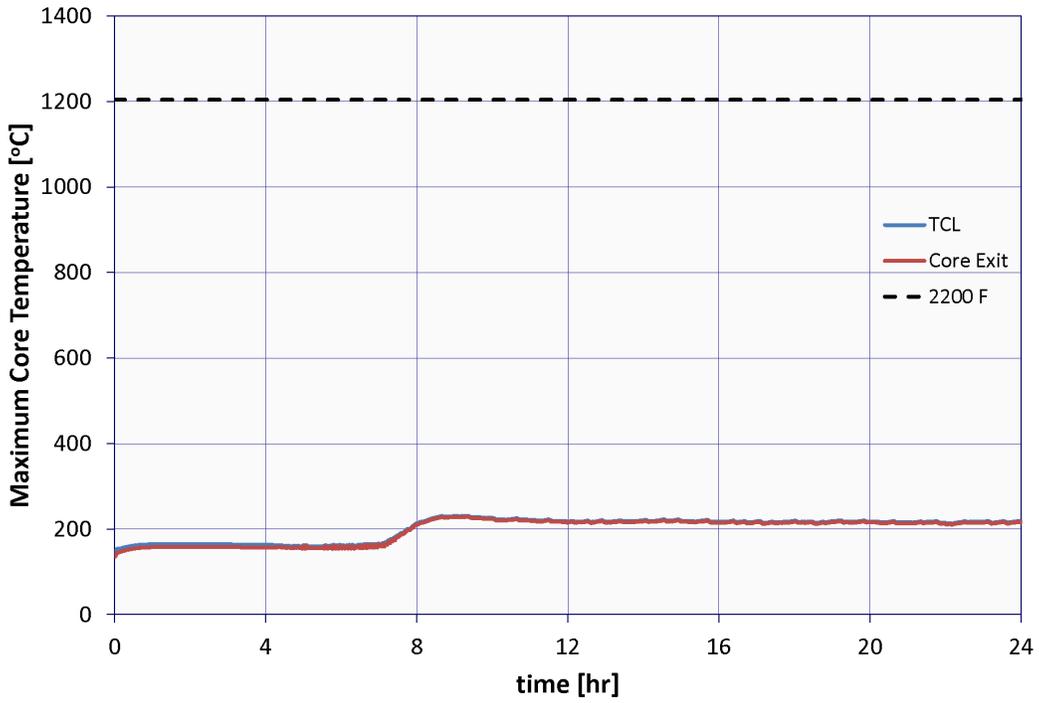


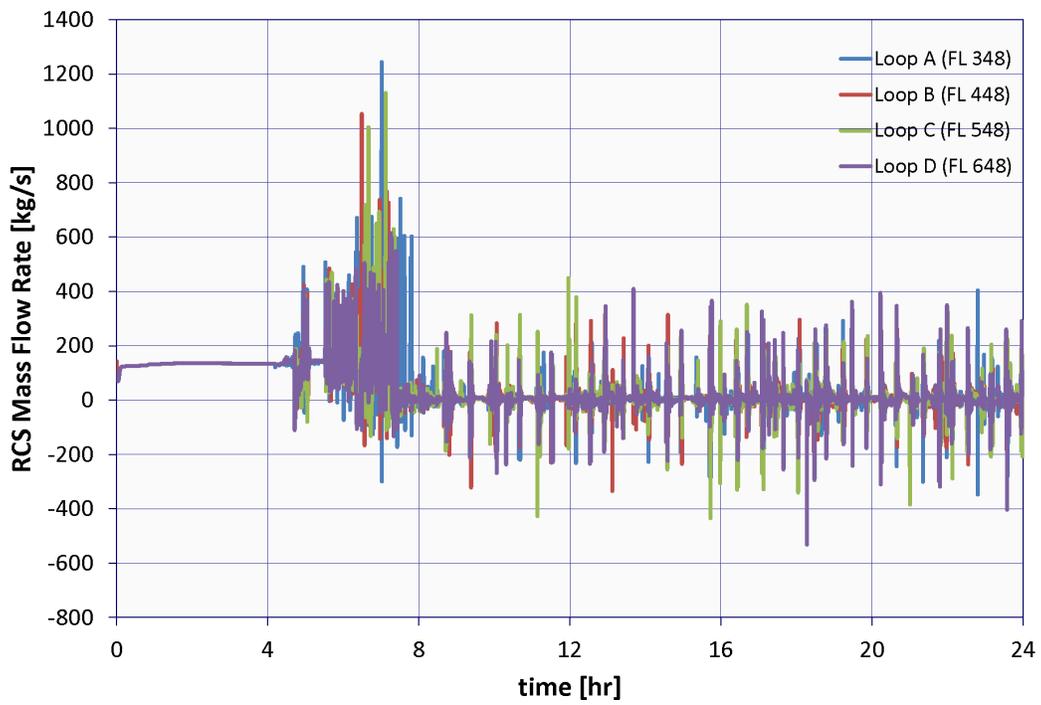
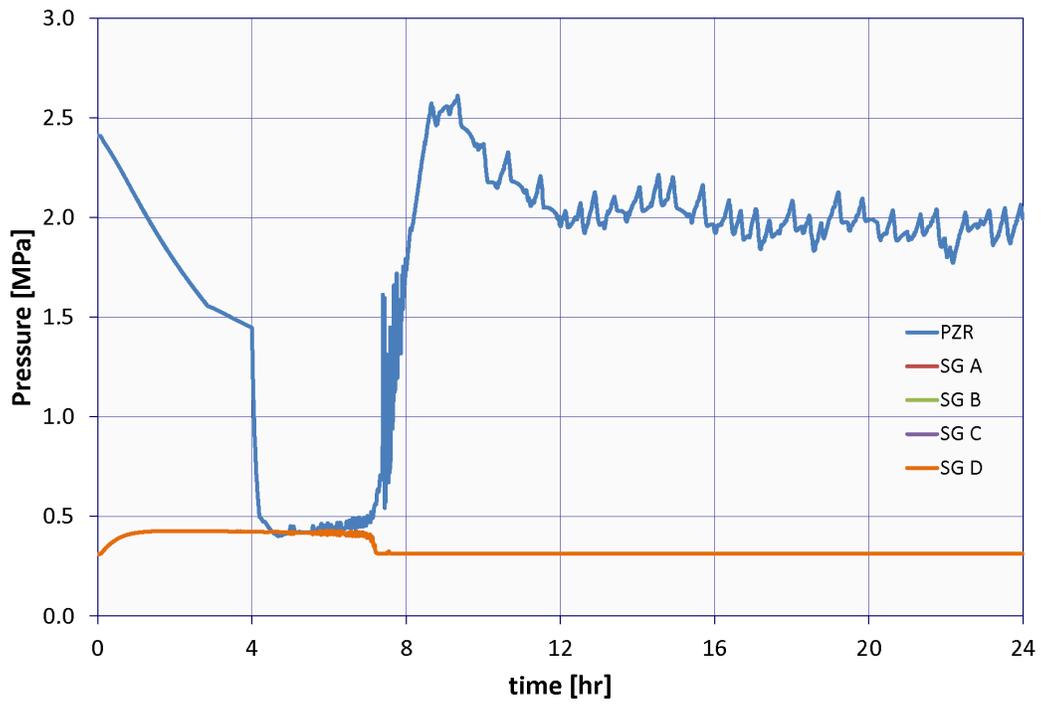


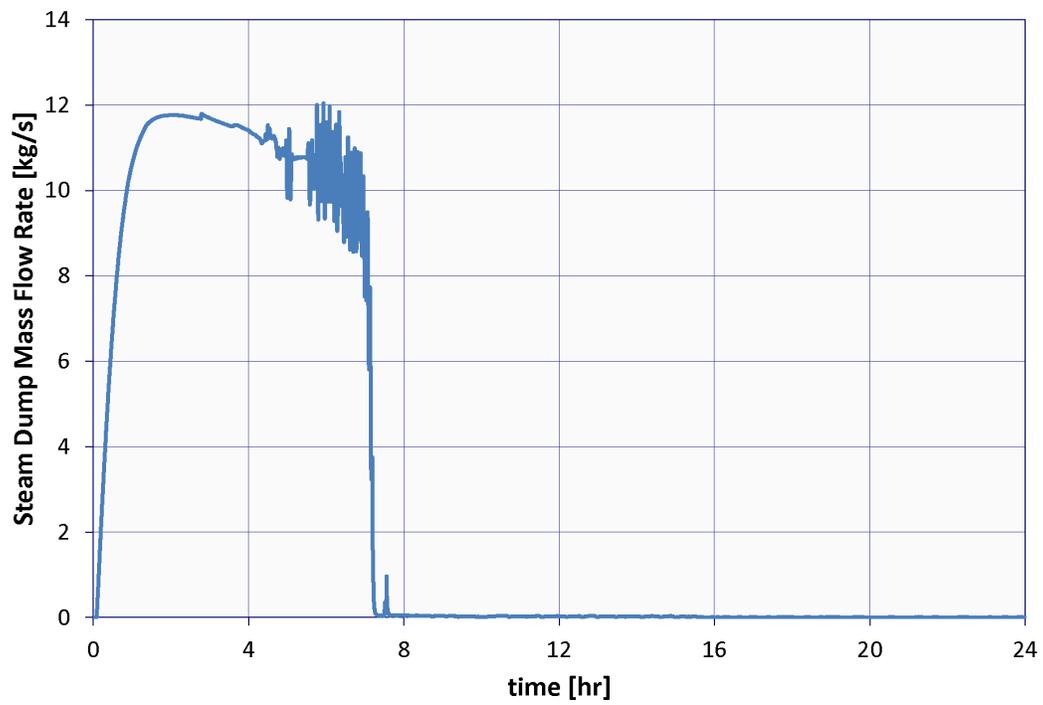
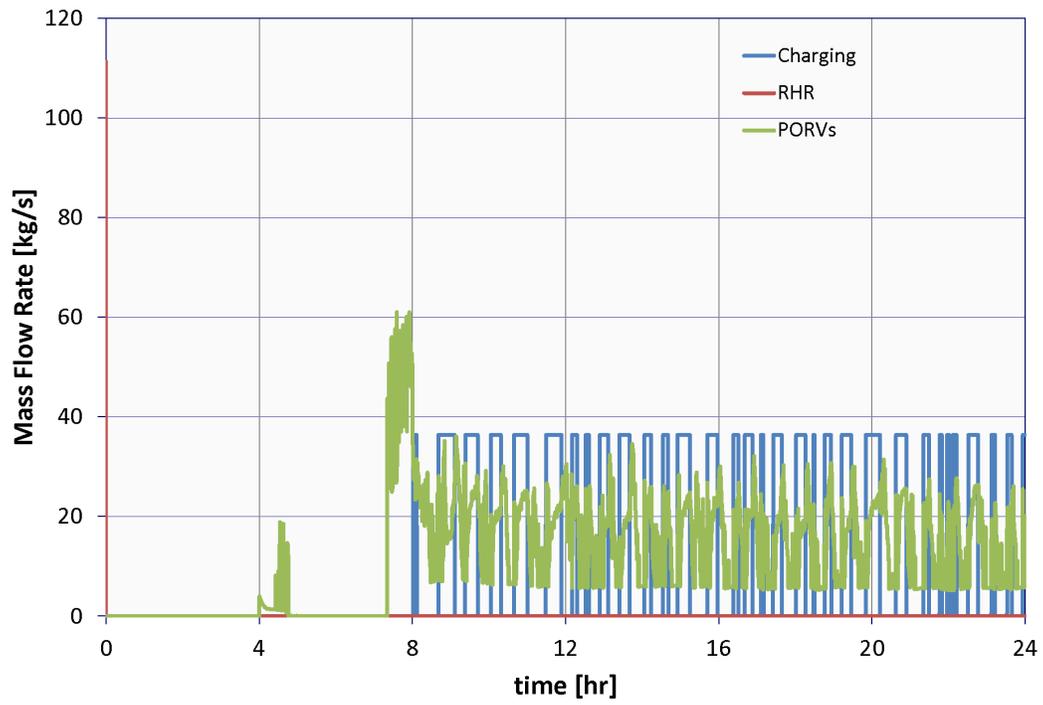


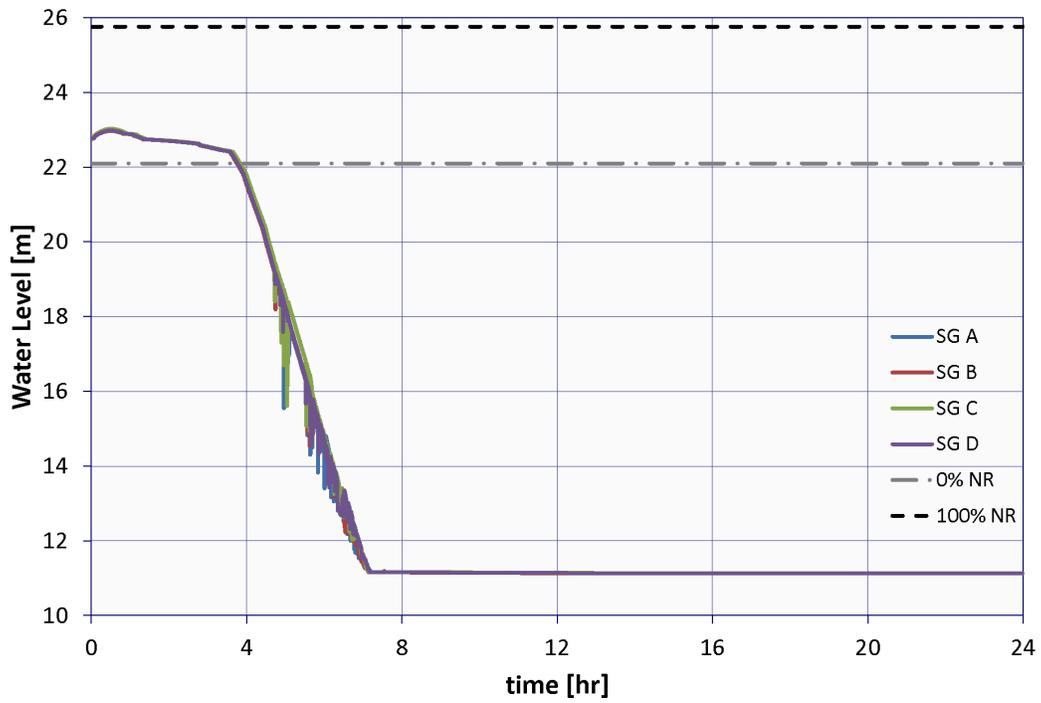
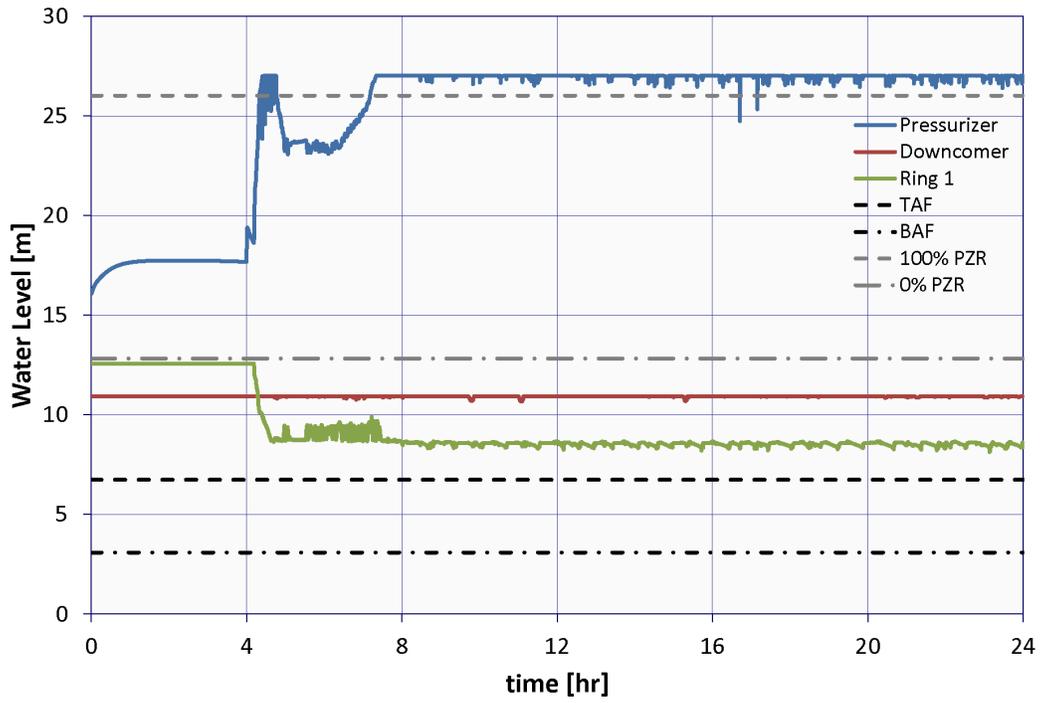


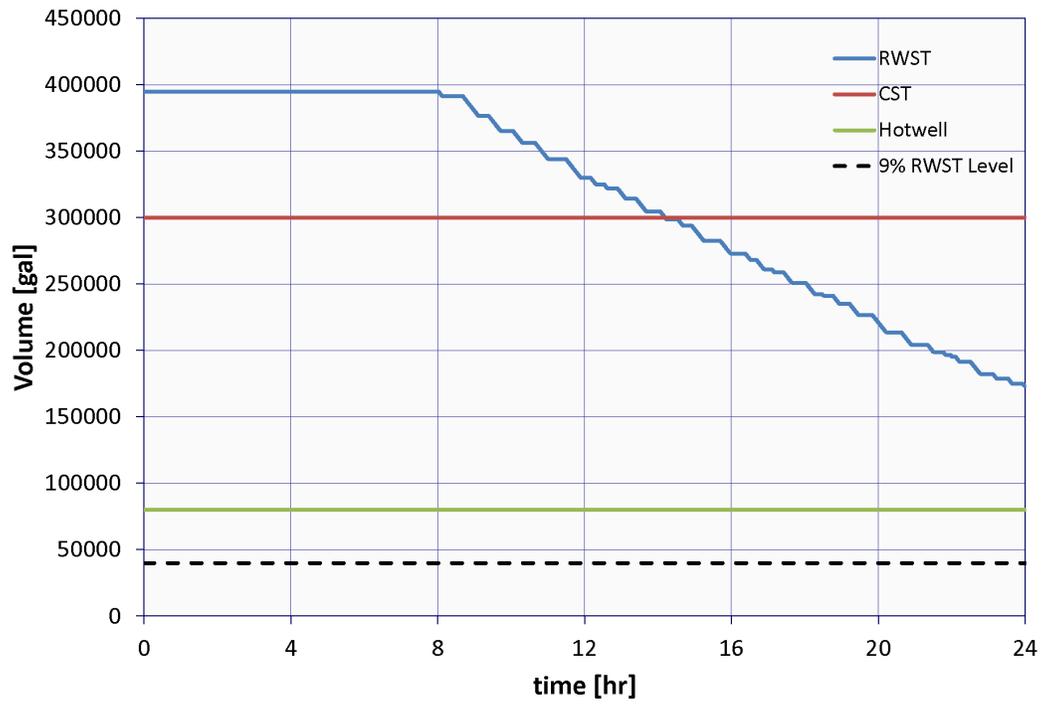
**F.1.5 Case 5: SG at 18% NR Level, 12 hr after Shutdown, Initiate Bleed & Feed at 5 hr**



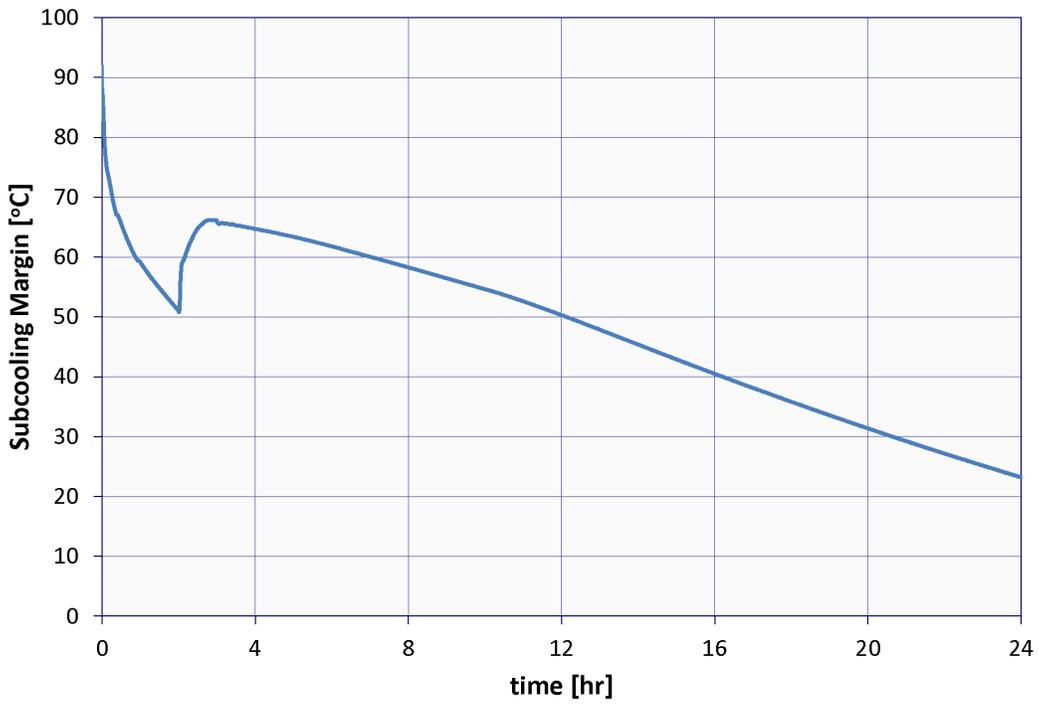
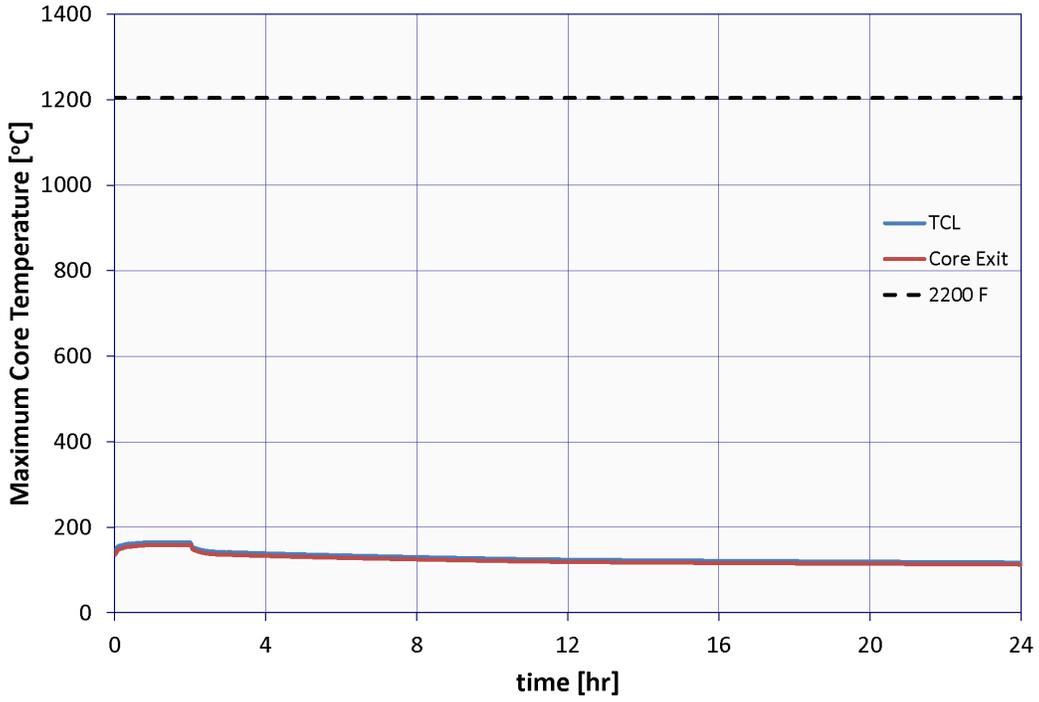


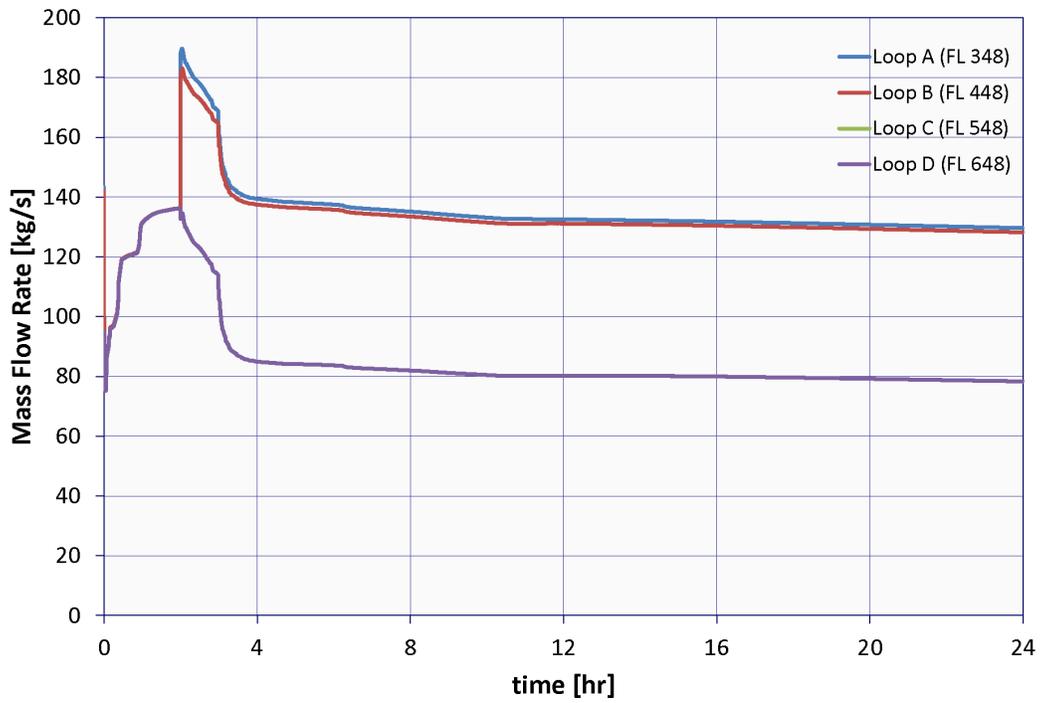
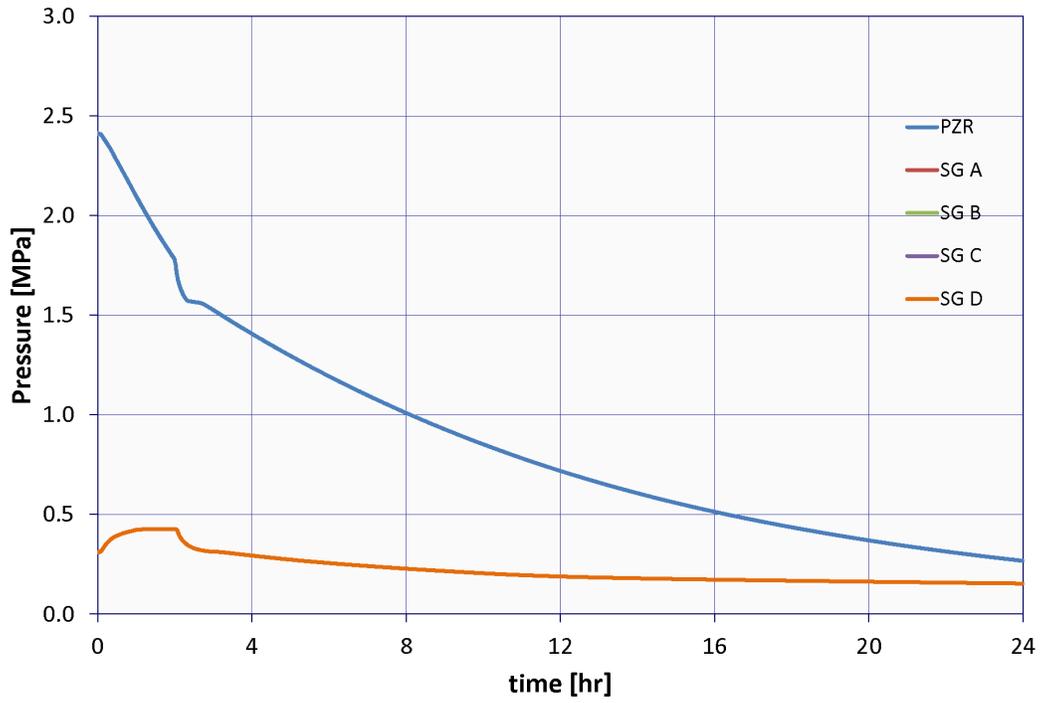


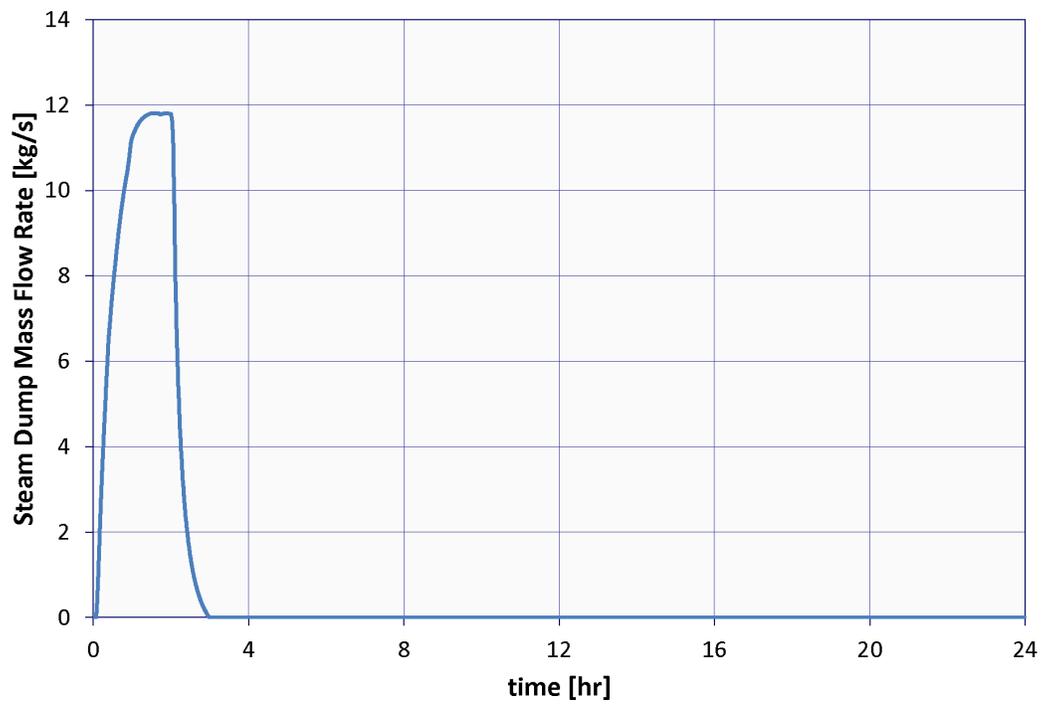
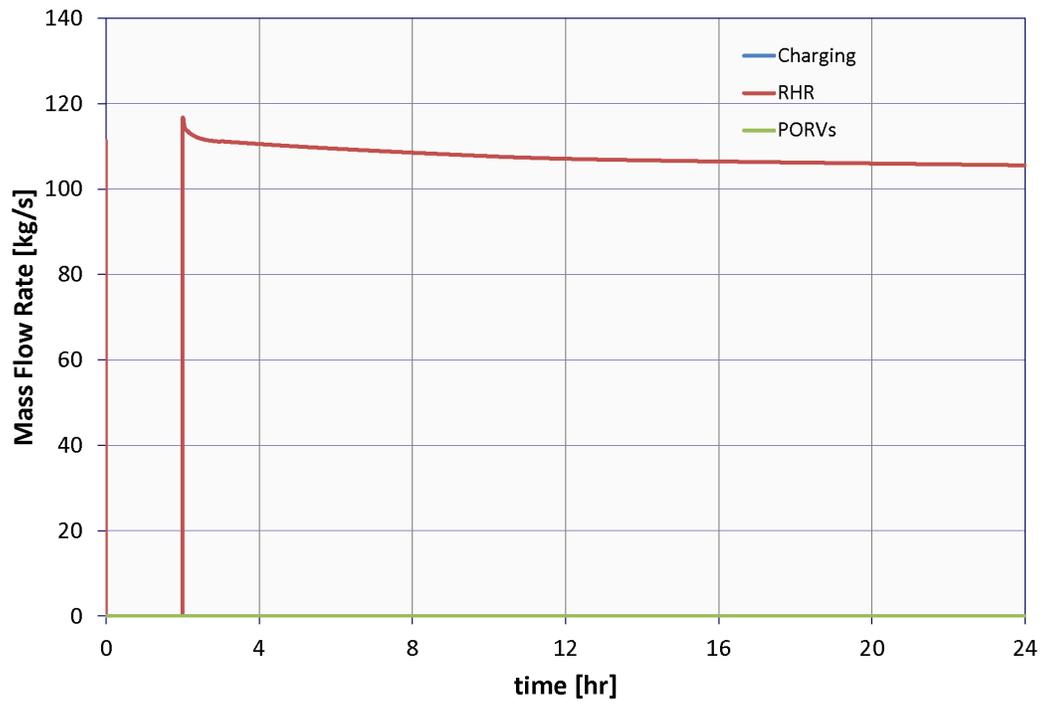


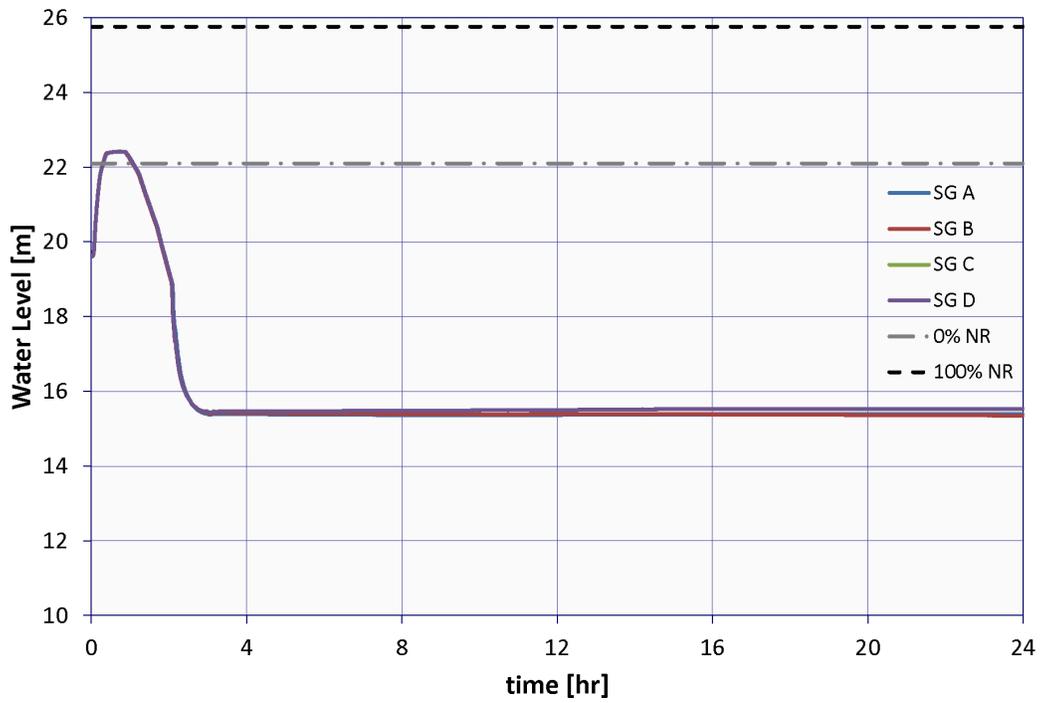
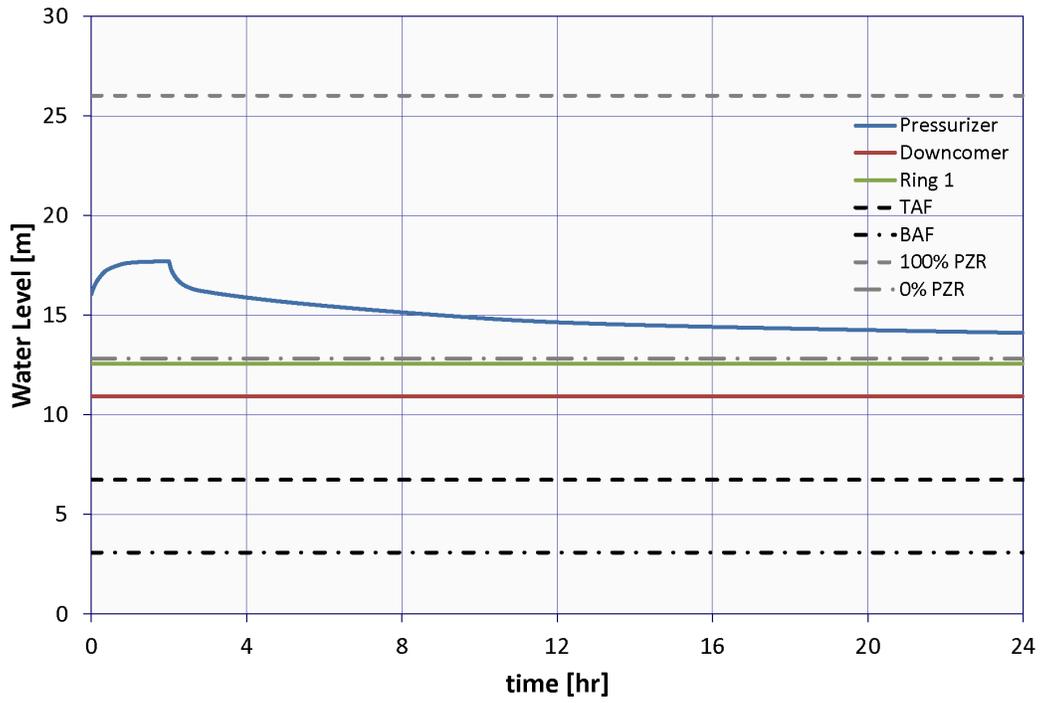


**F.1.6 Case 6: SG at 27% WR Level, 12 hr after Shutdown, Recover RHR at 2 hr**

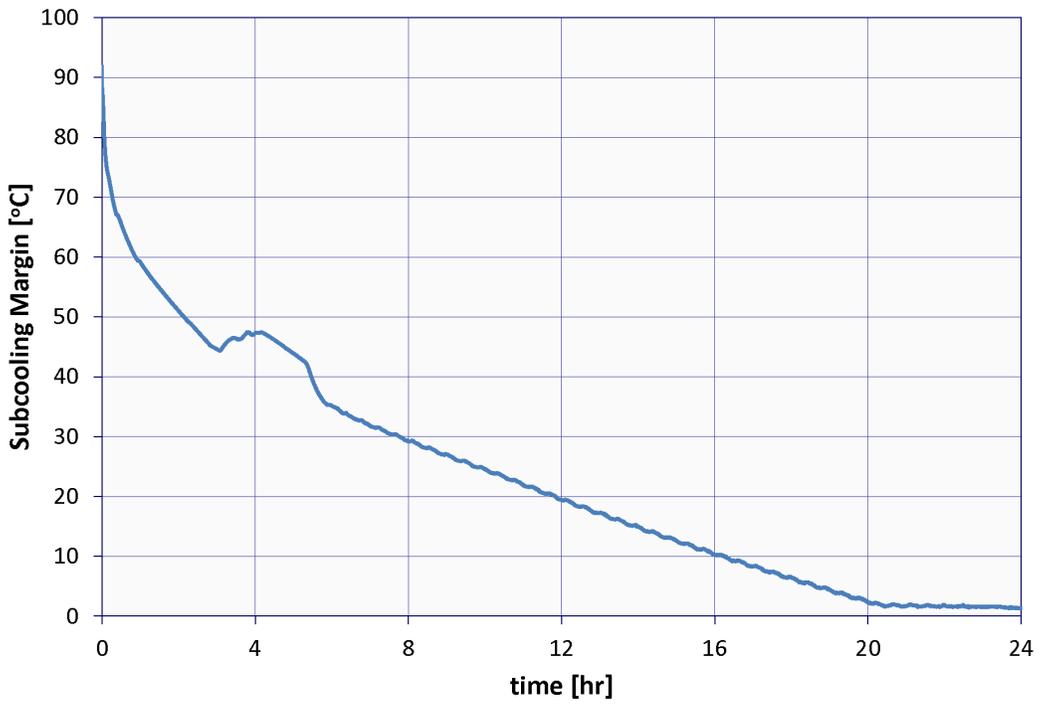
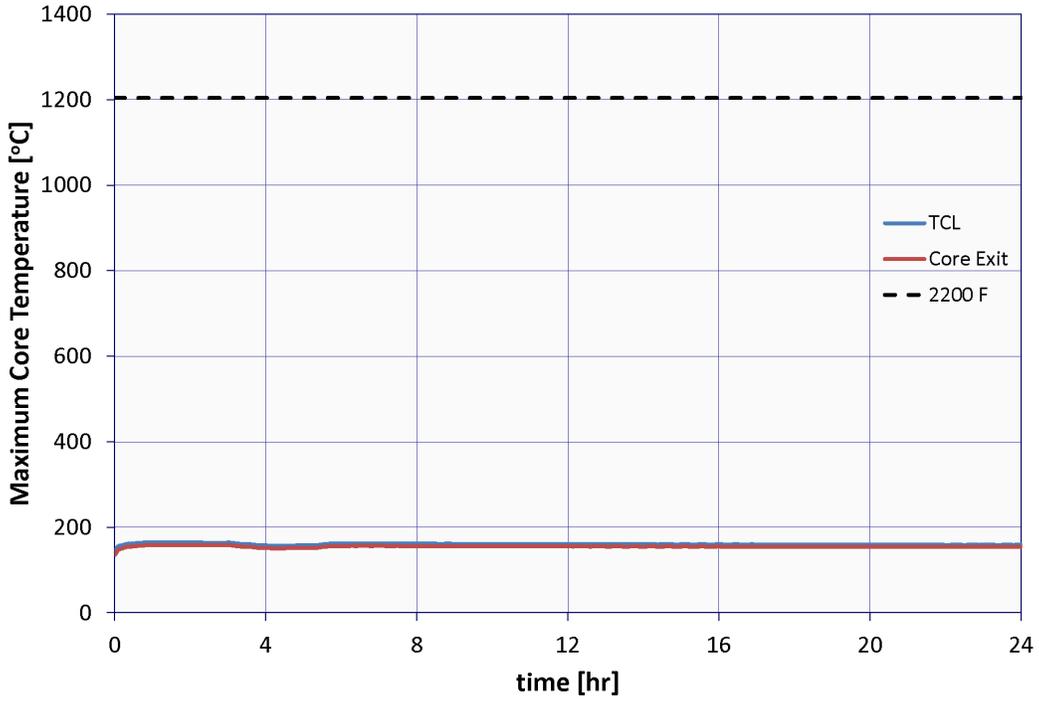


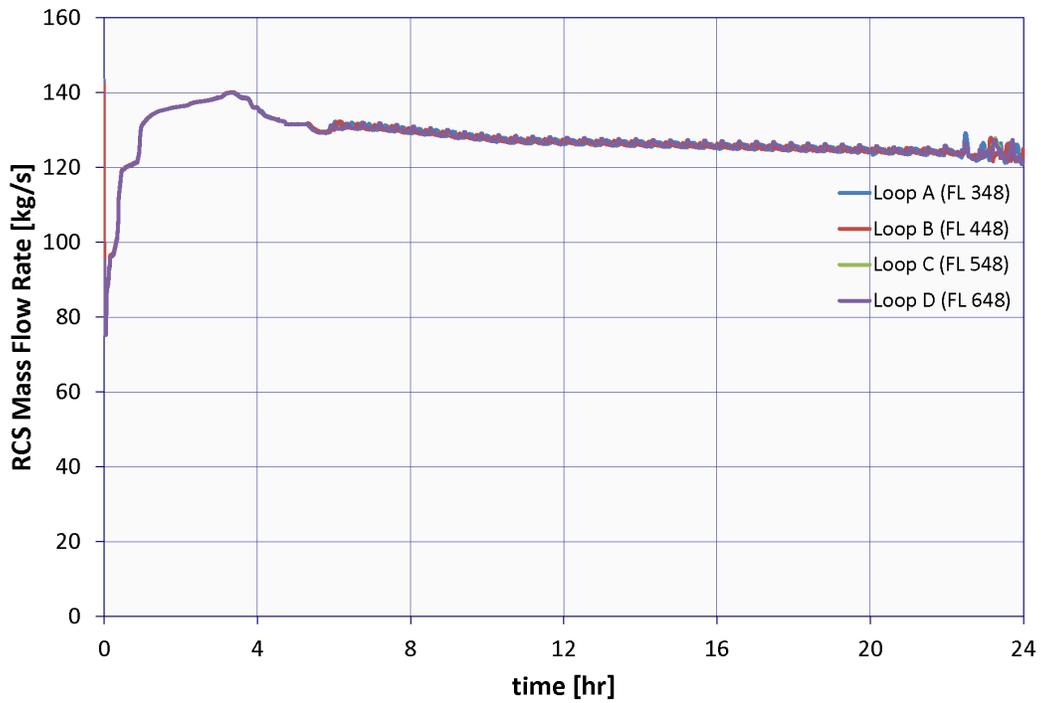
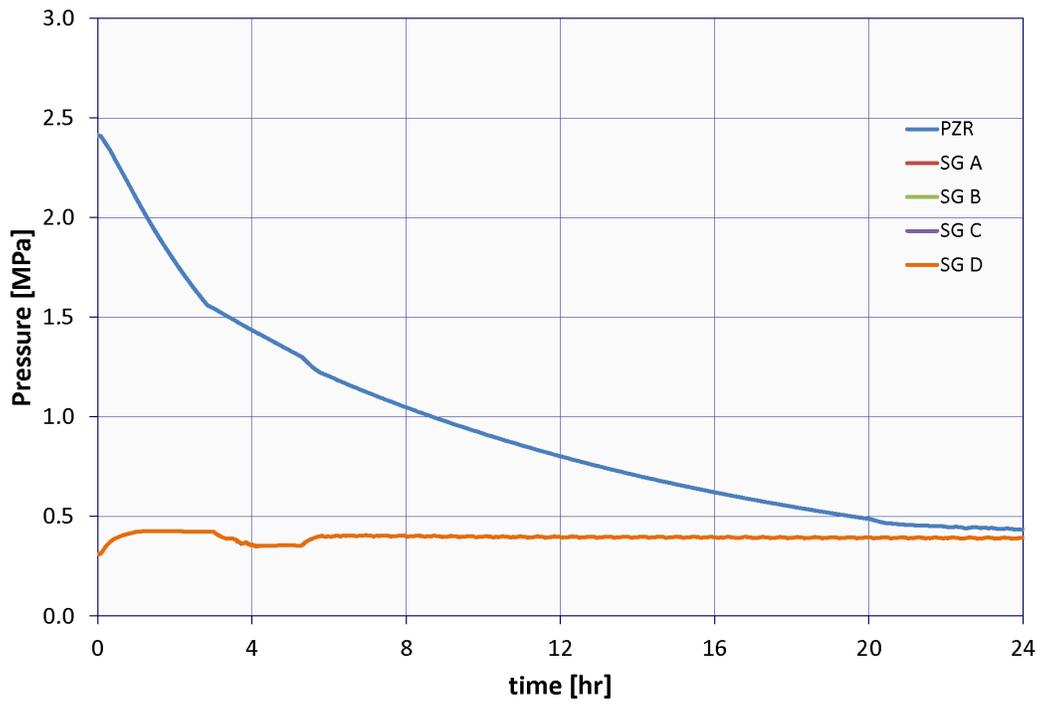


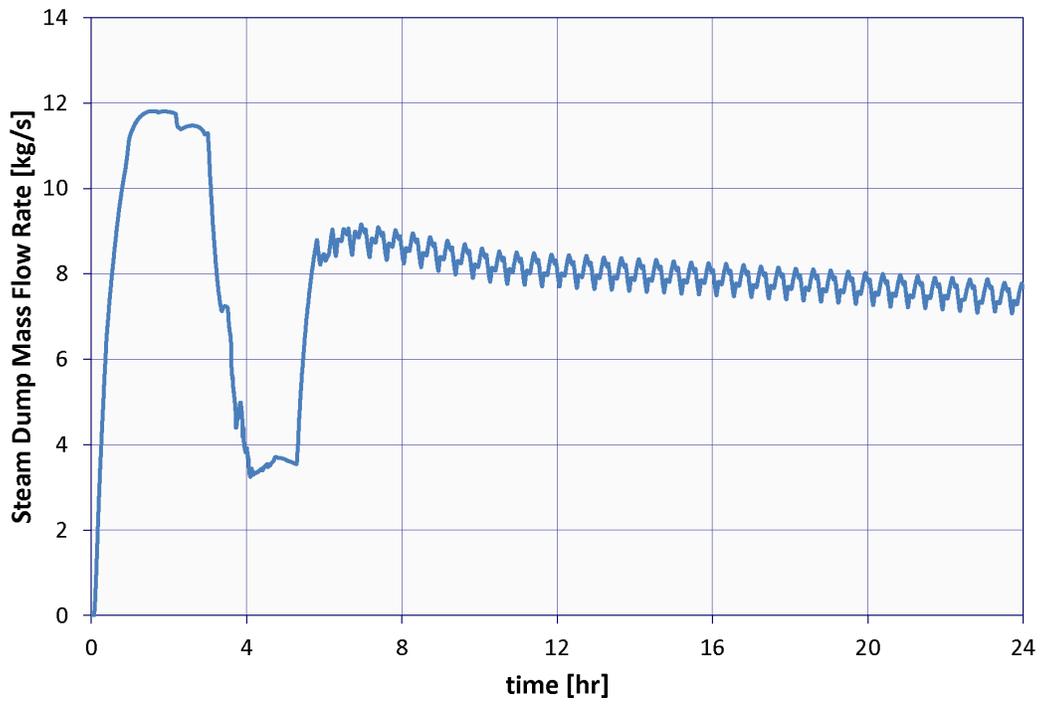
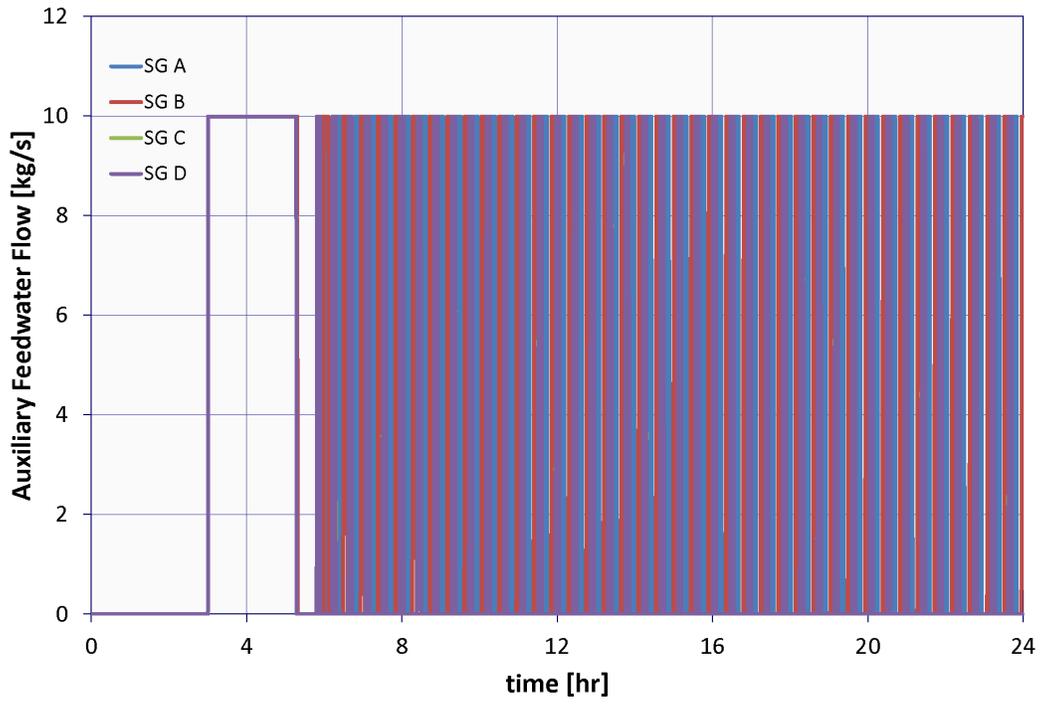


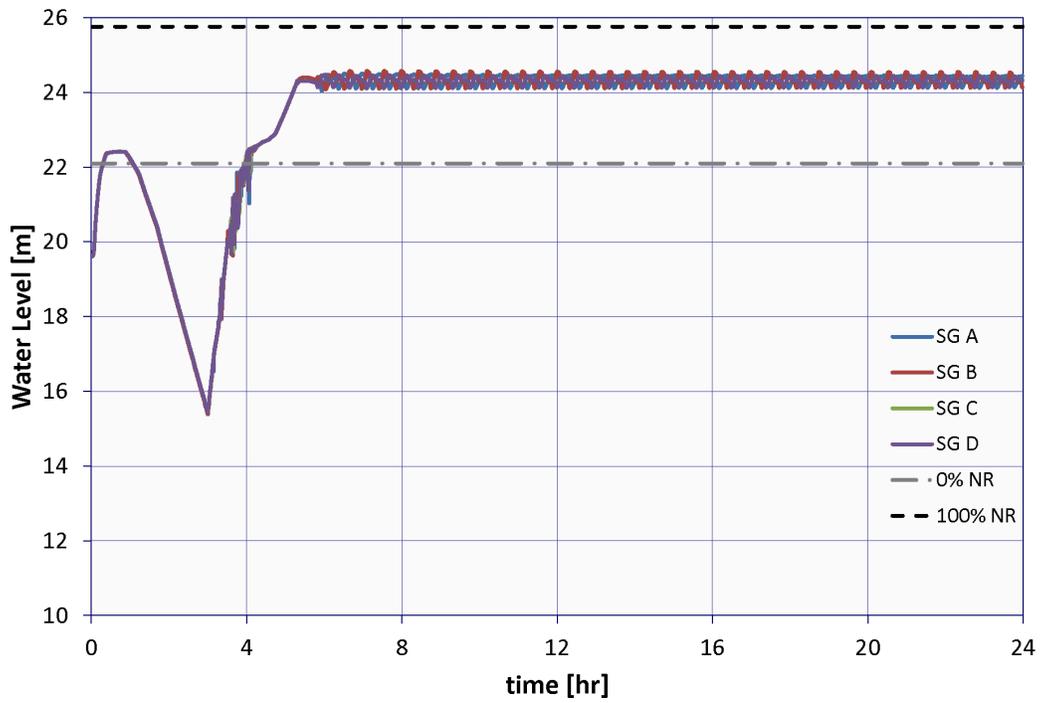
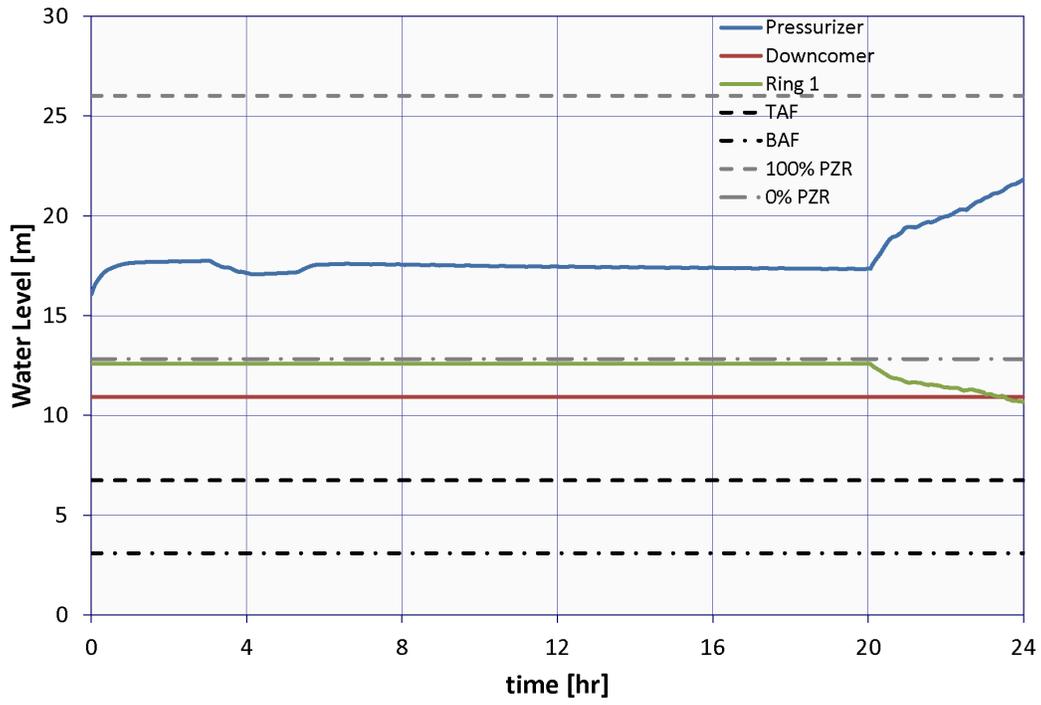


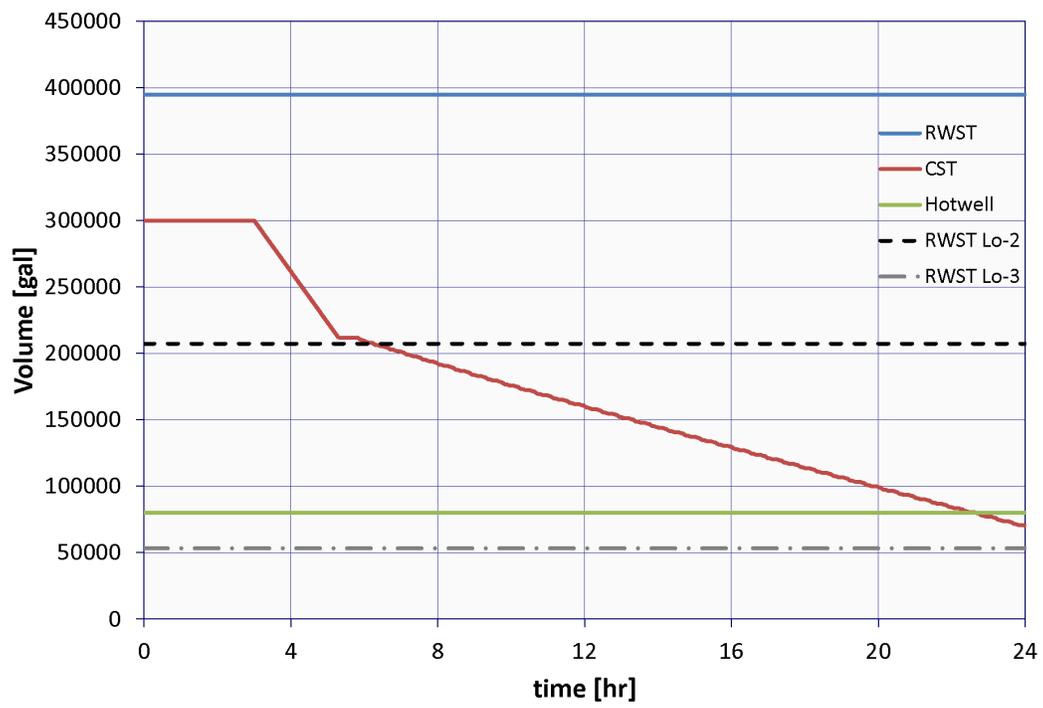
**F.1.7 Case 7: SG at 27% WR Level, 12 hr after Shutdown, Initiate AFW at 3 hr**



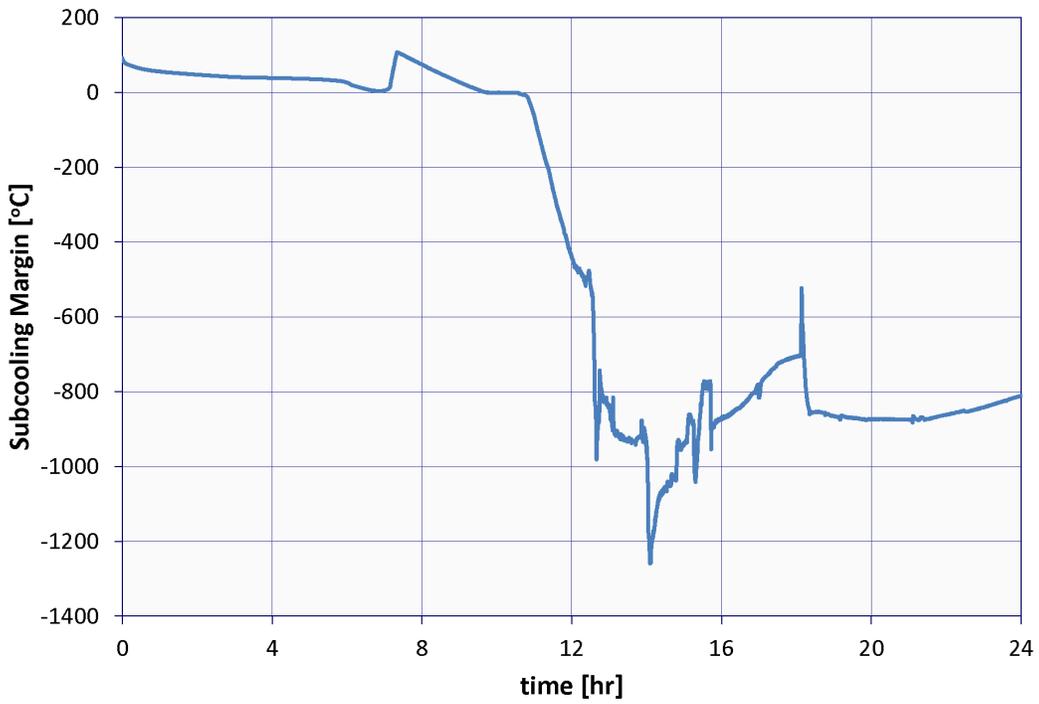
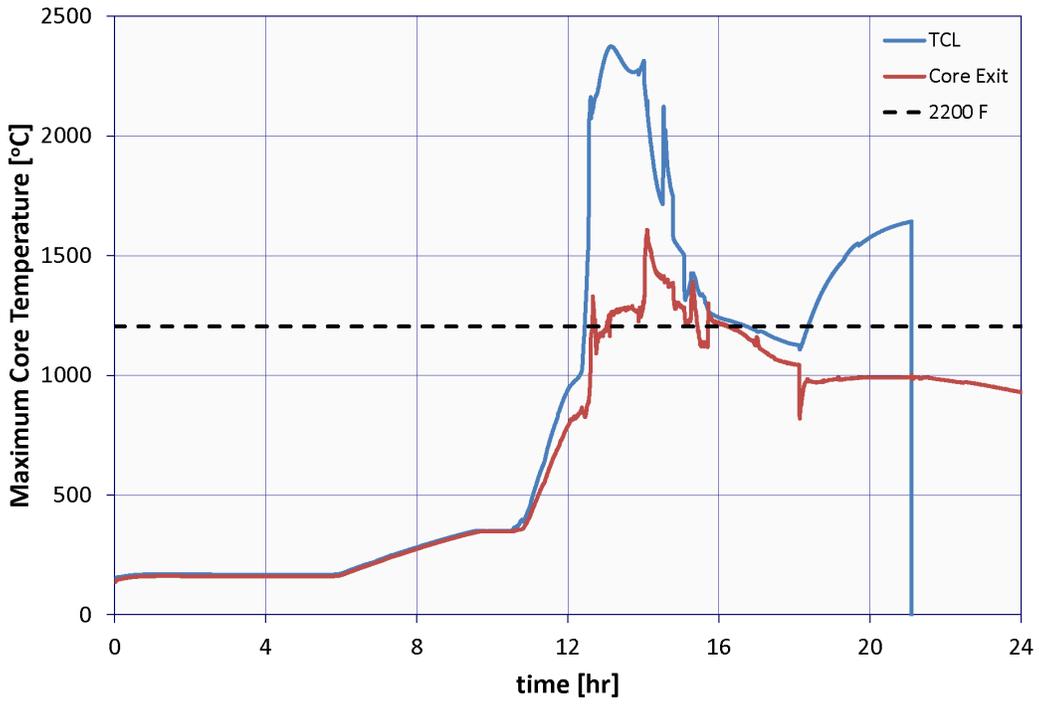


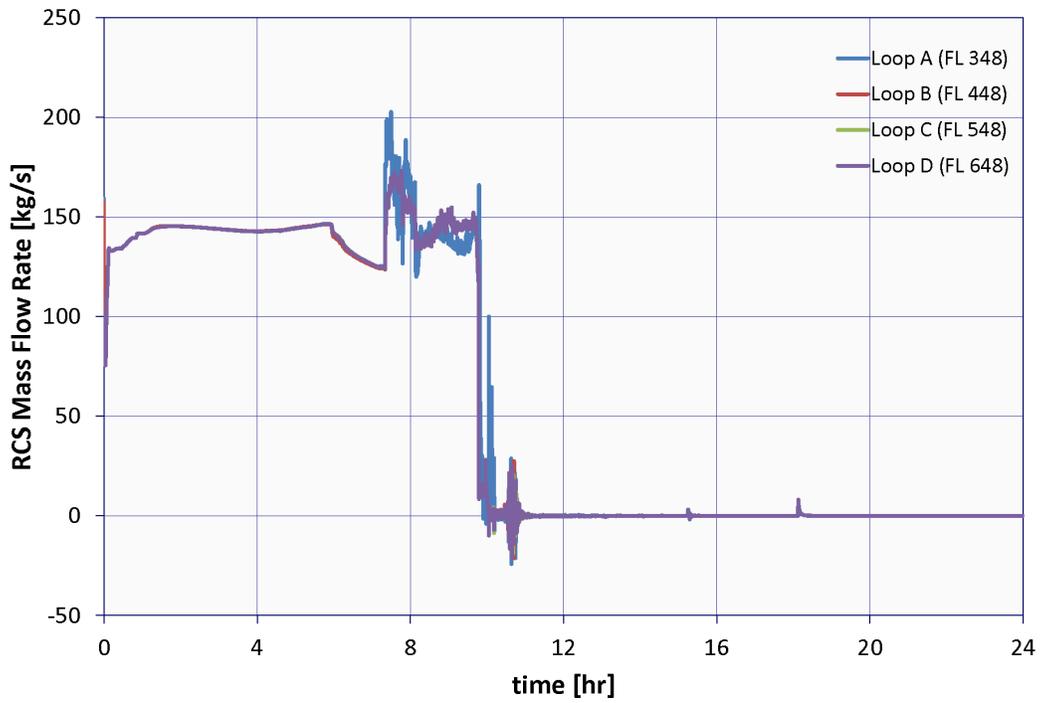
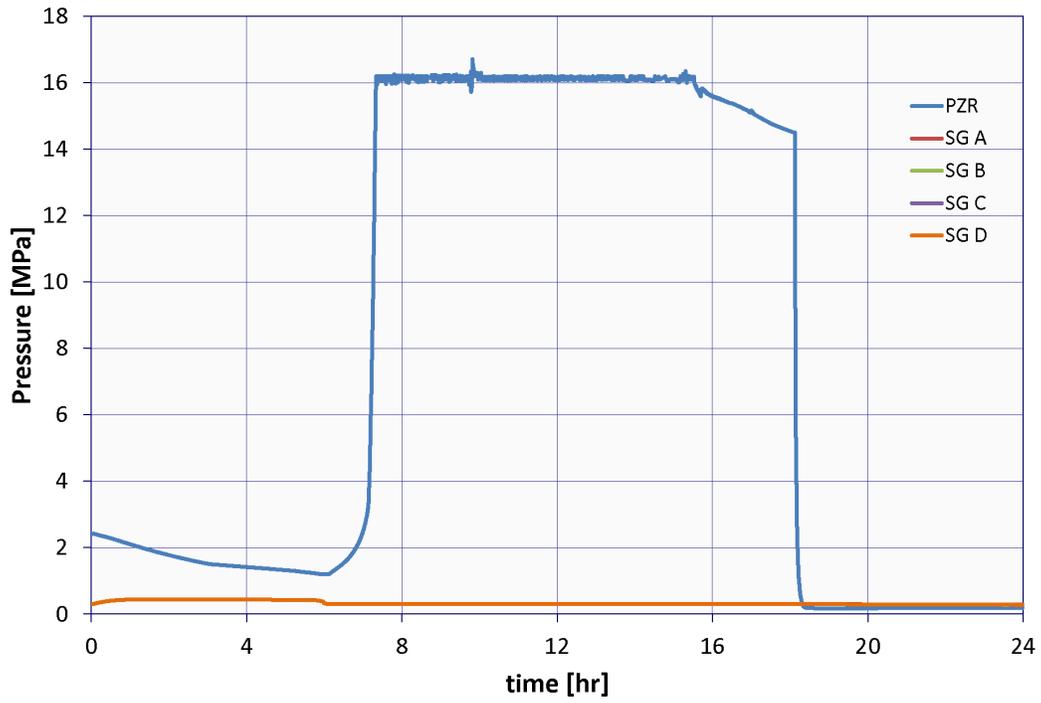


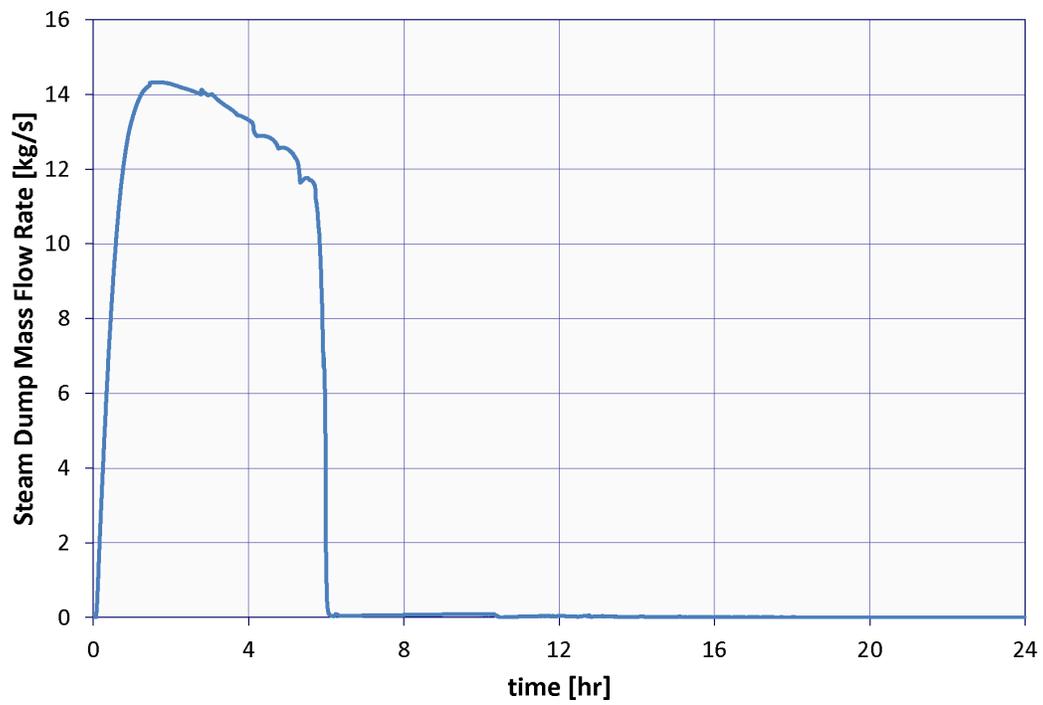
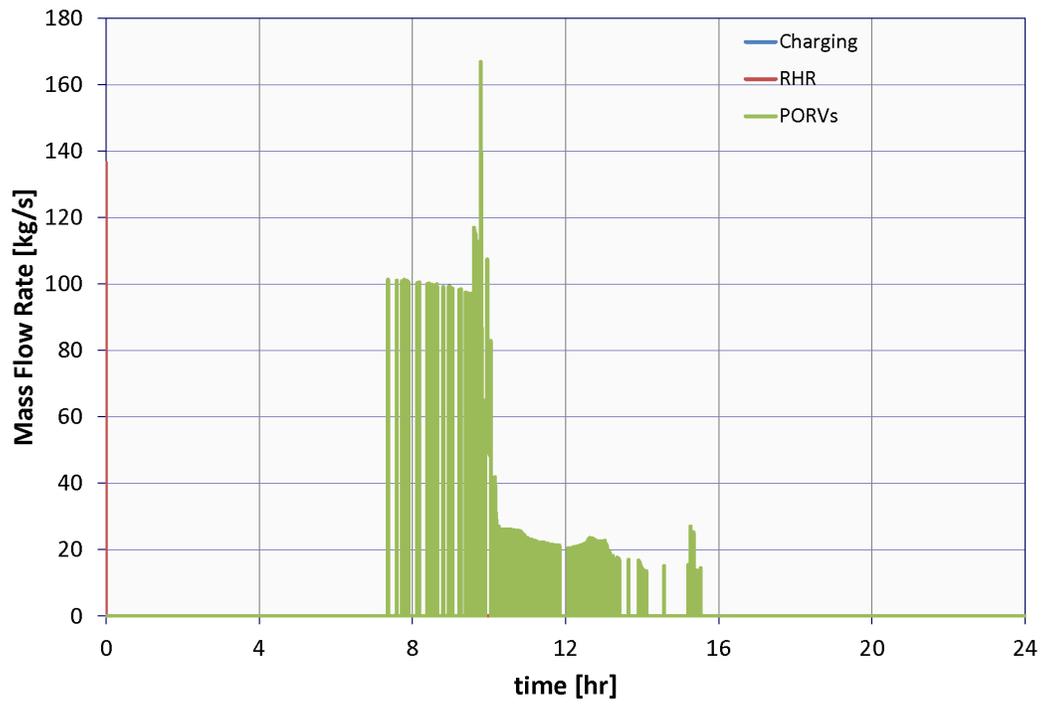


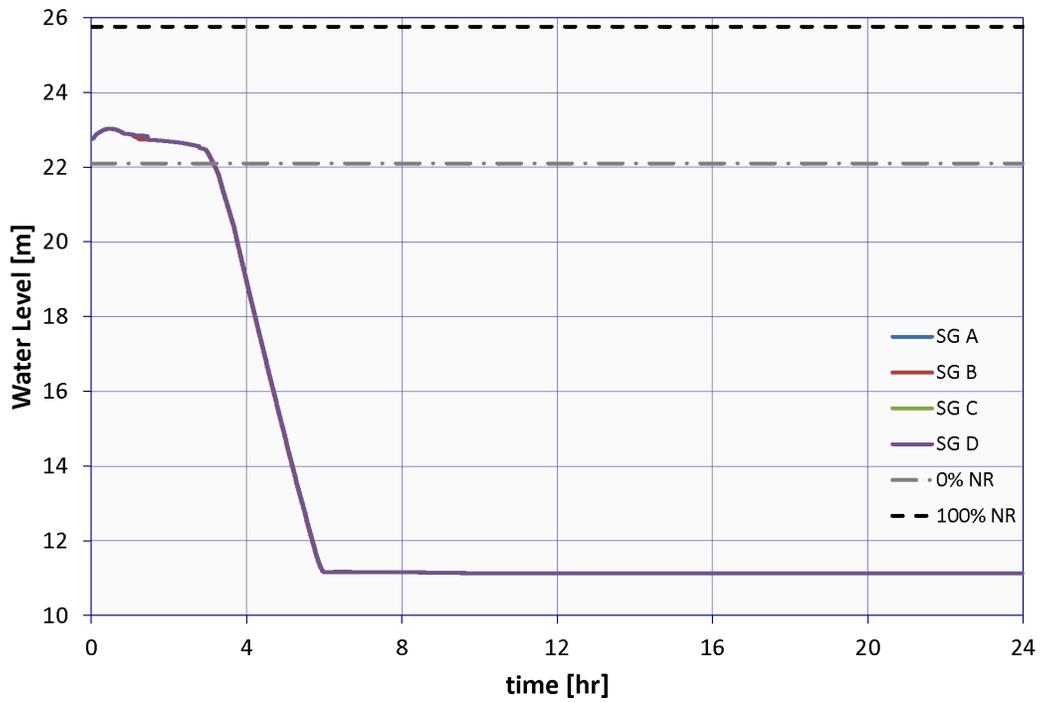
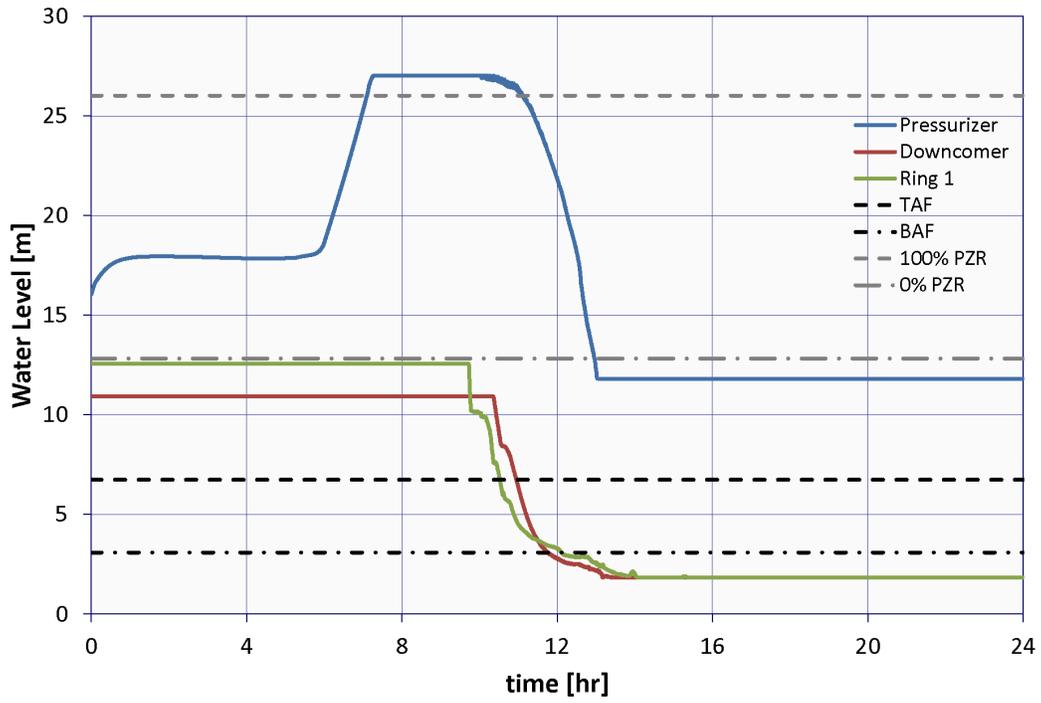


F.1.8 Case 8: SG at 18% NR Level, 6 hr after Shutdown, No Recovery Actions

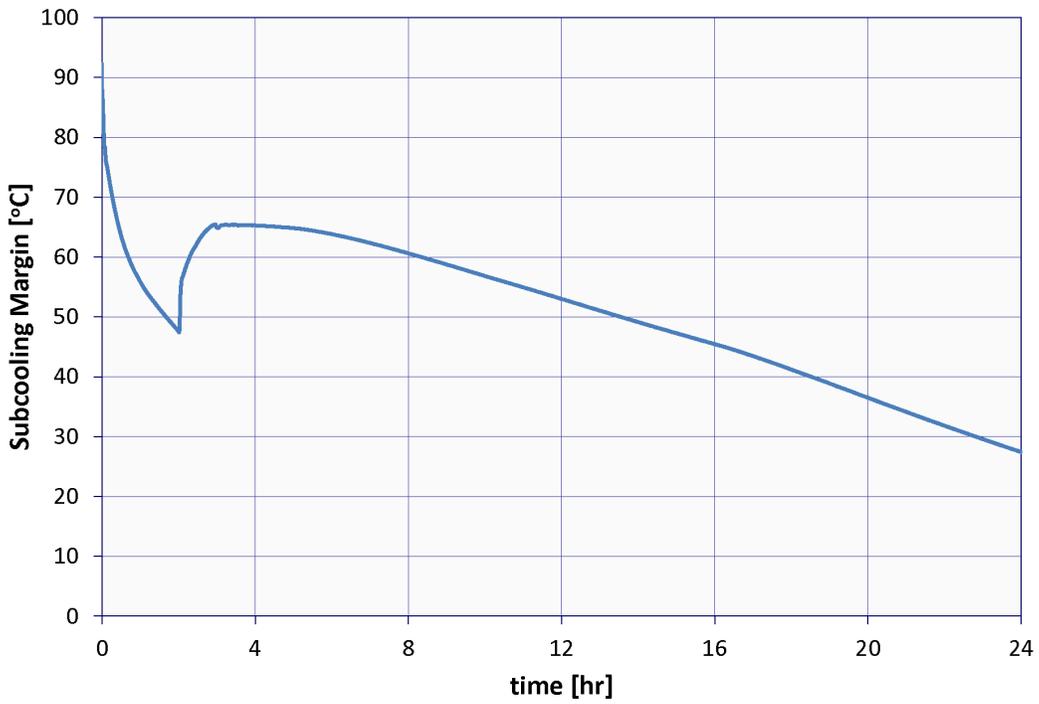
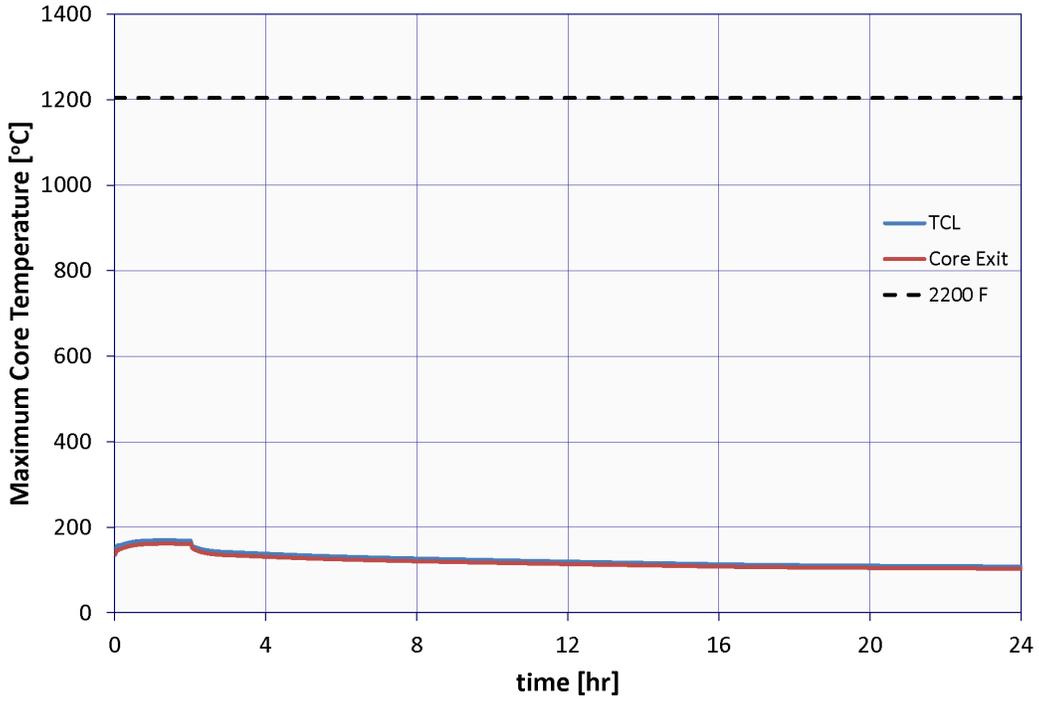


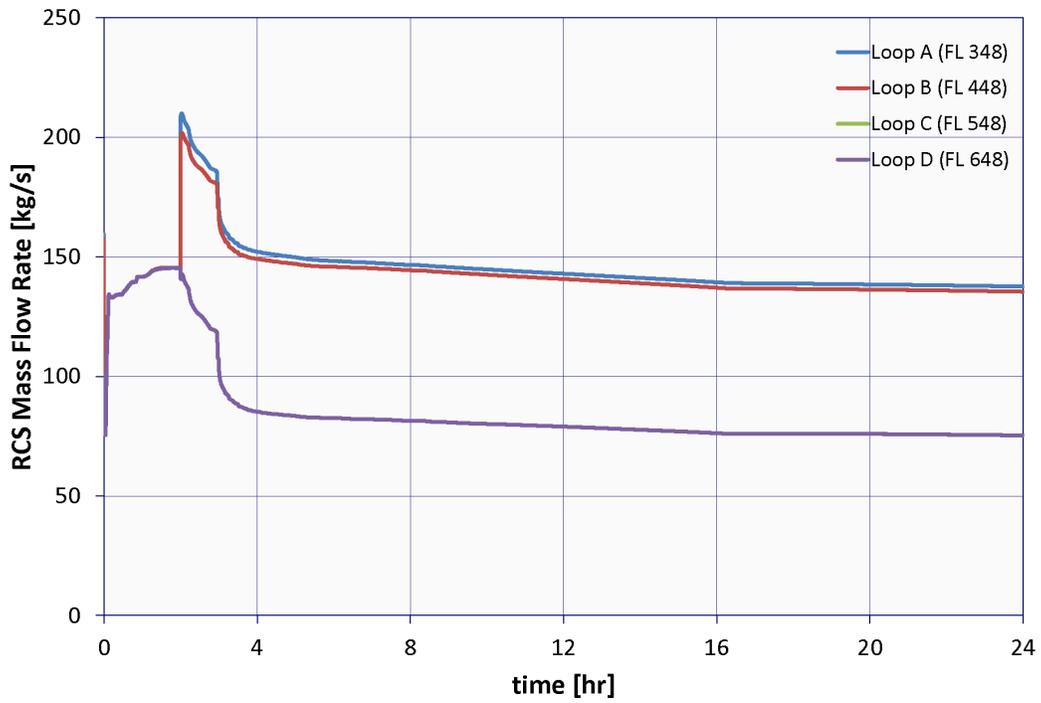
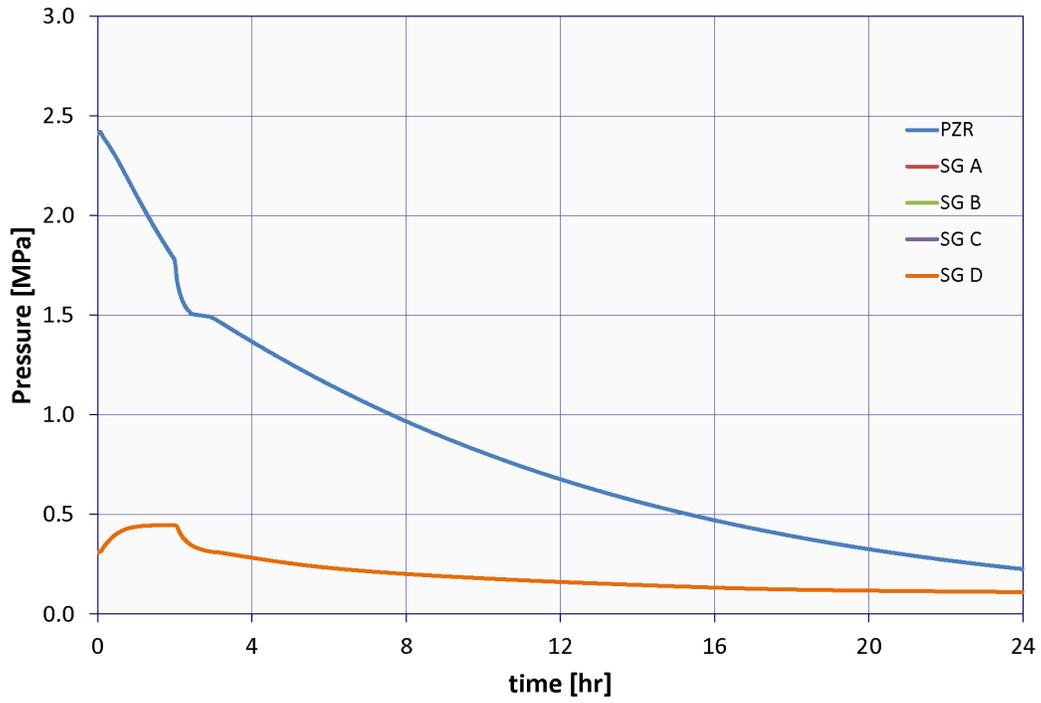


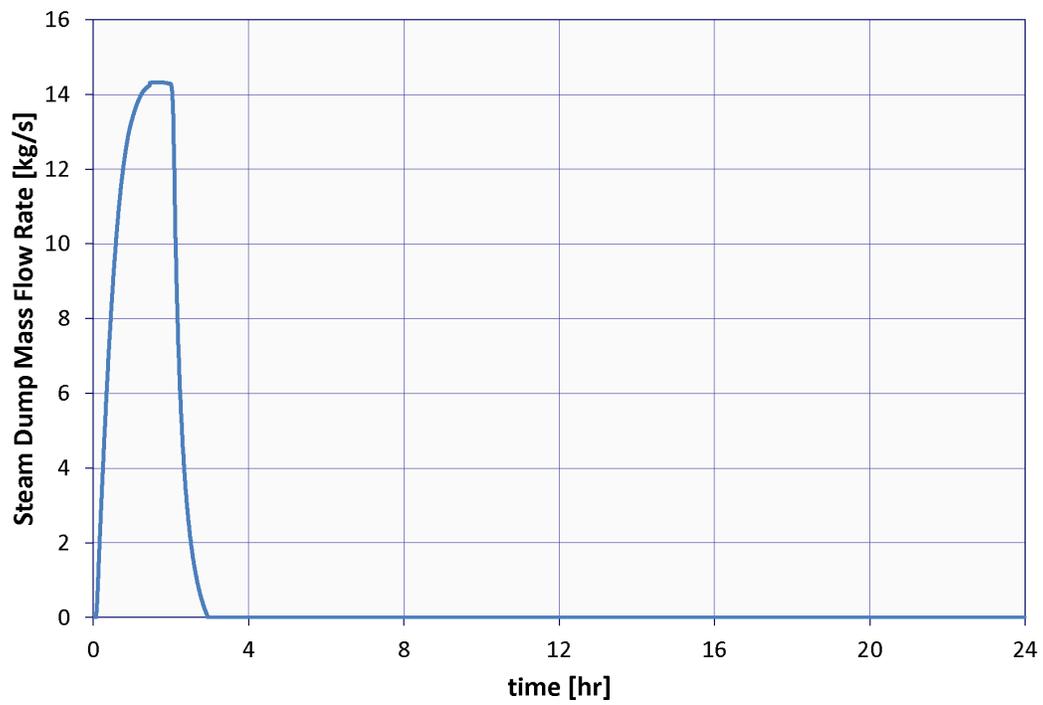
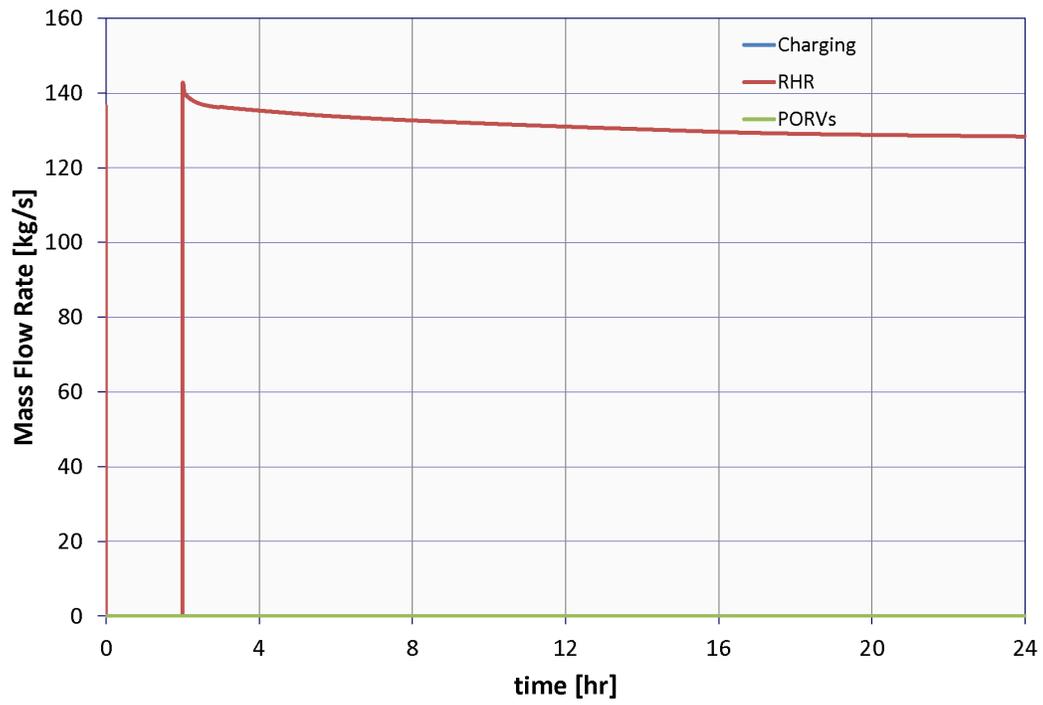


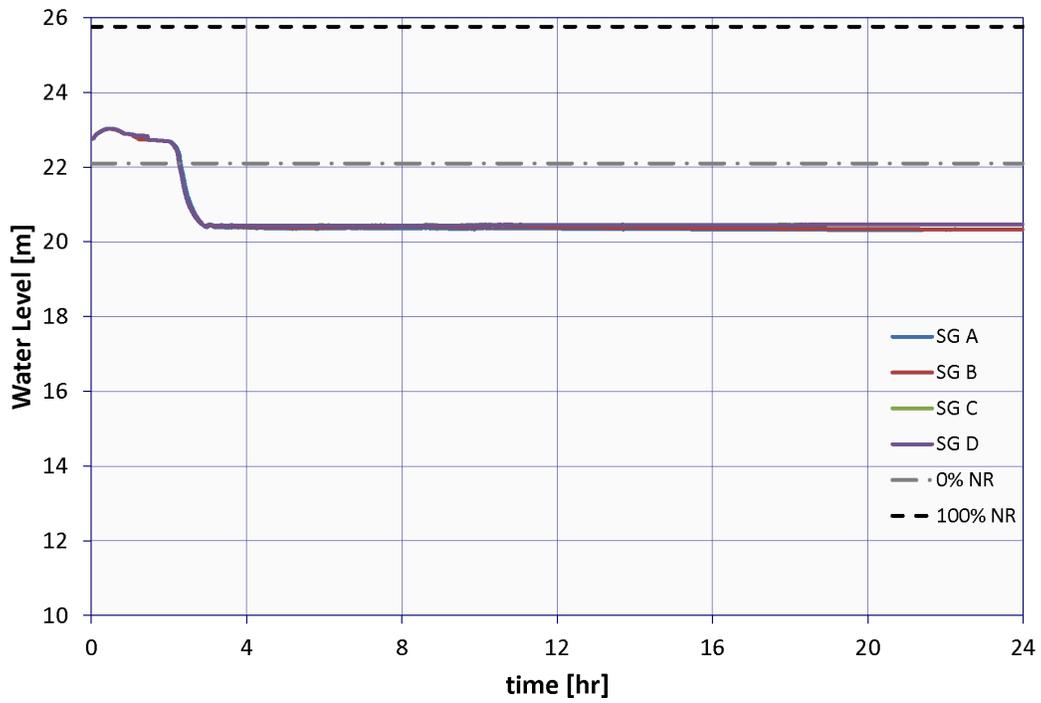
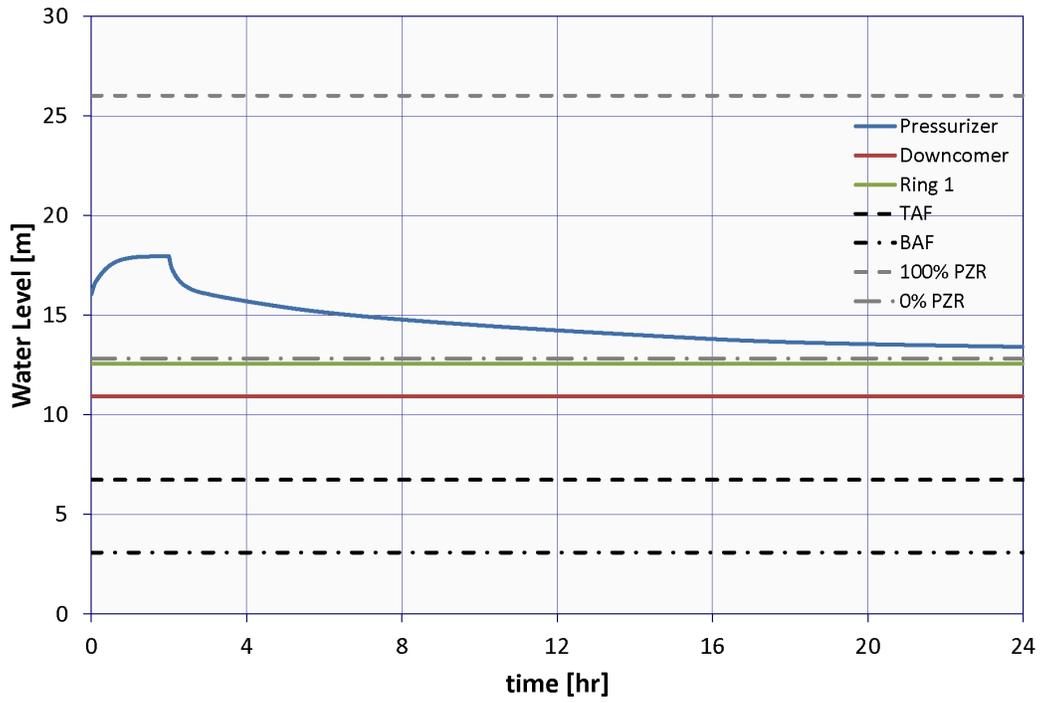


**F.1.9 Case 9: SG at 18% NR Level, 6 hr after Shutdown, Recover RHR at 2 hr**

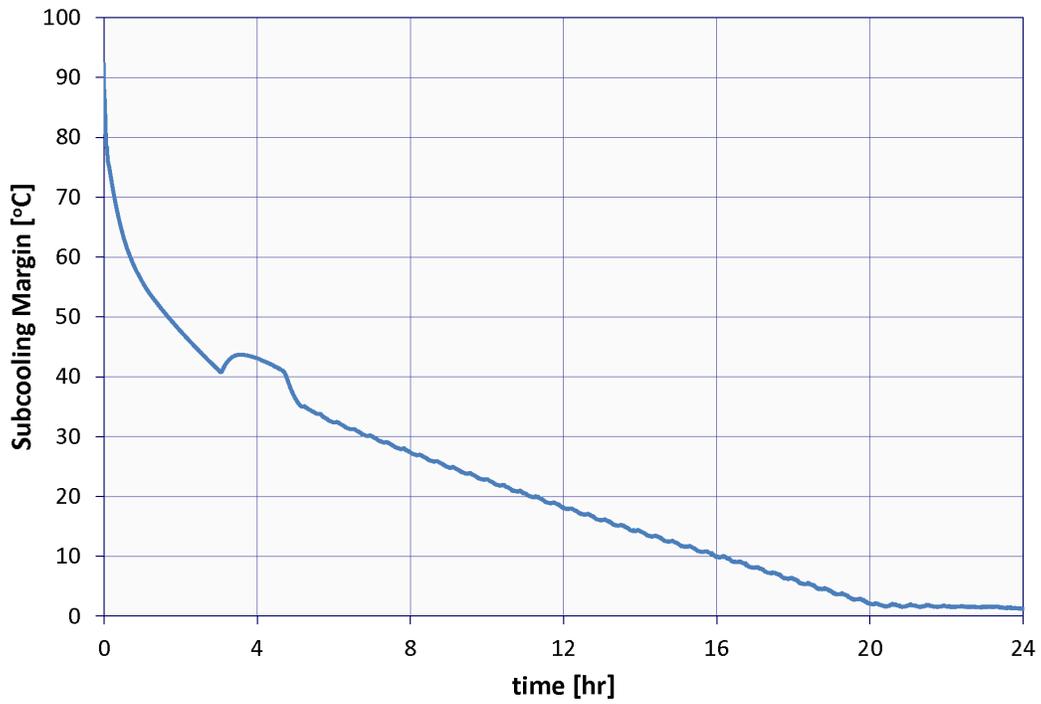
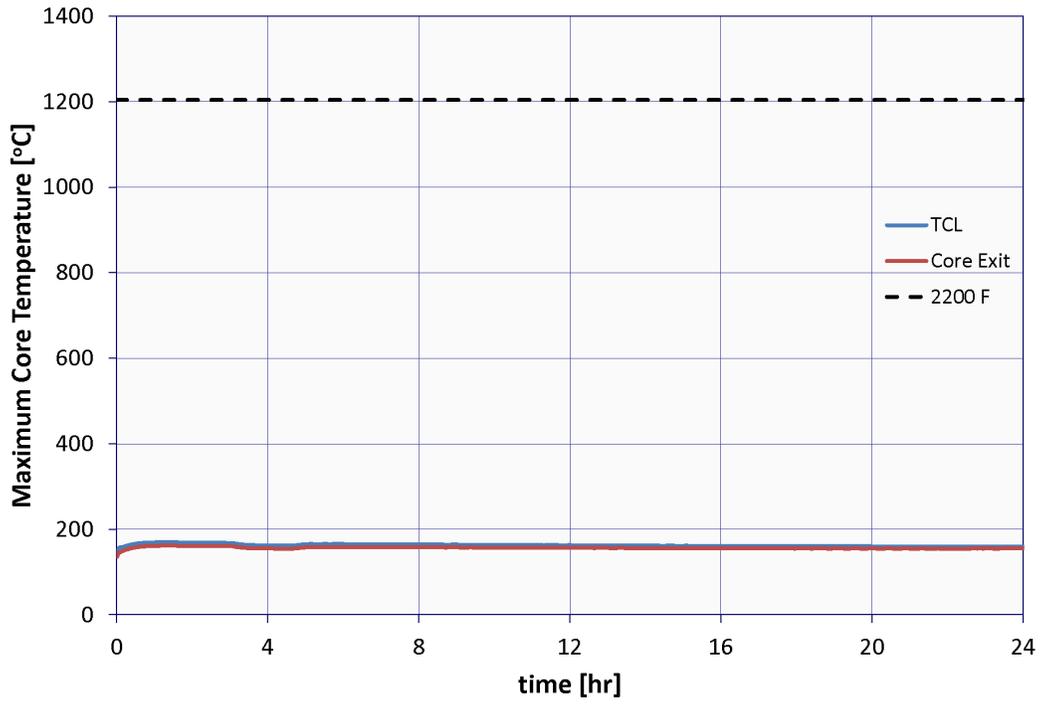


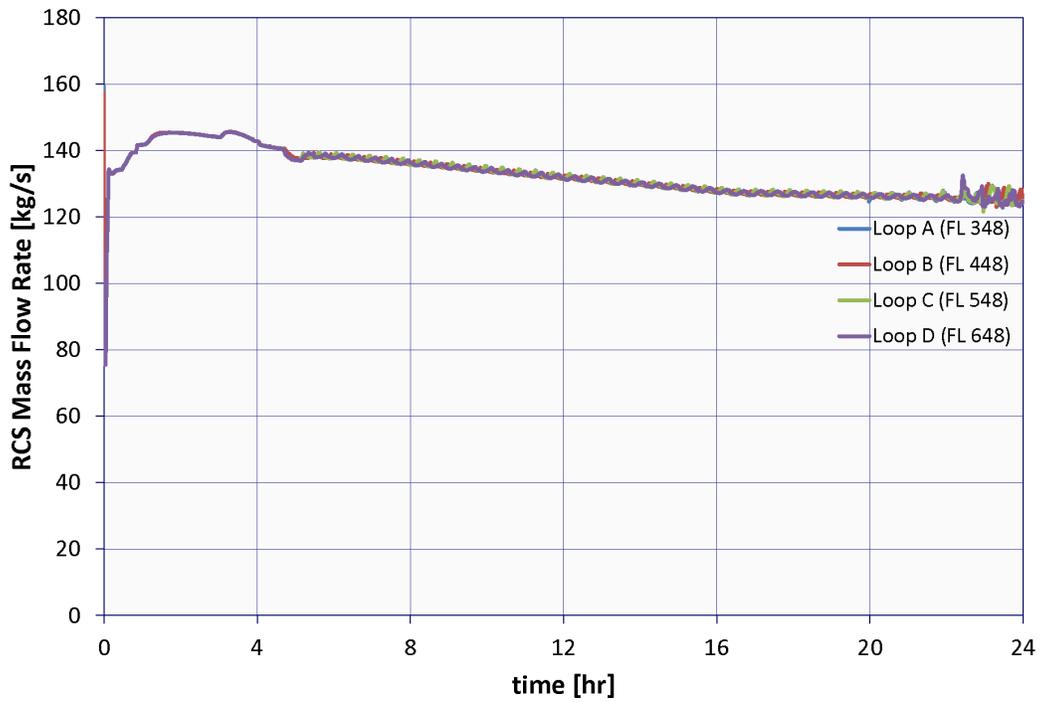
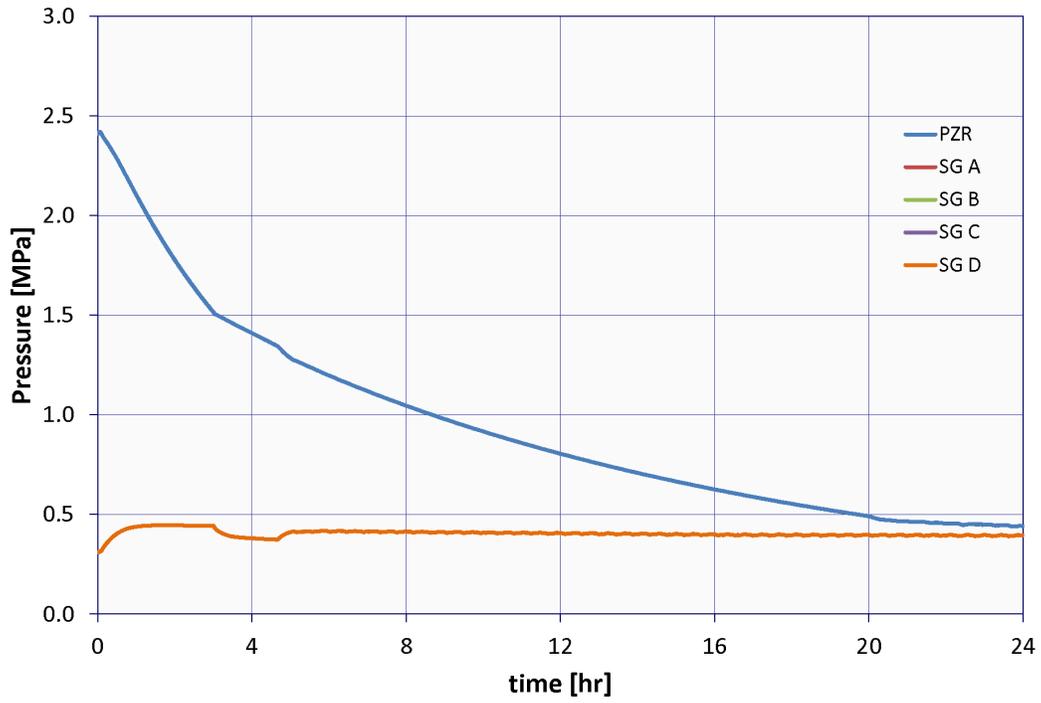


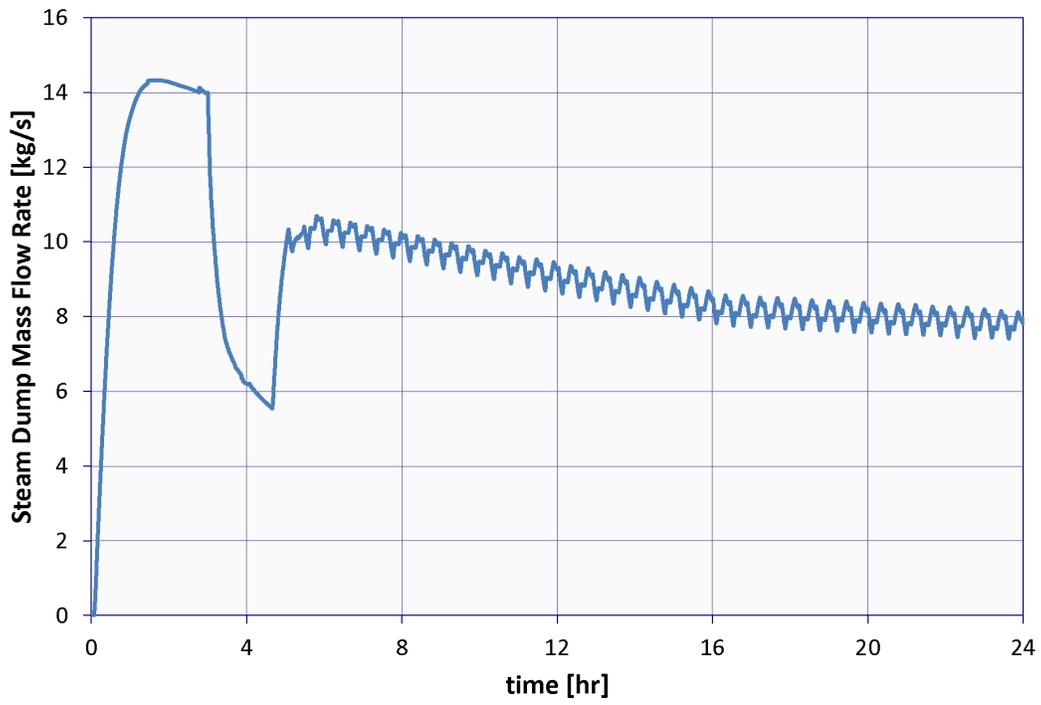
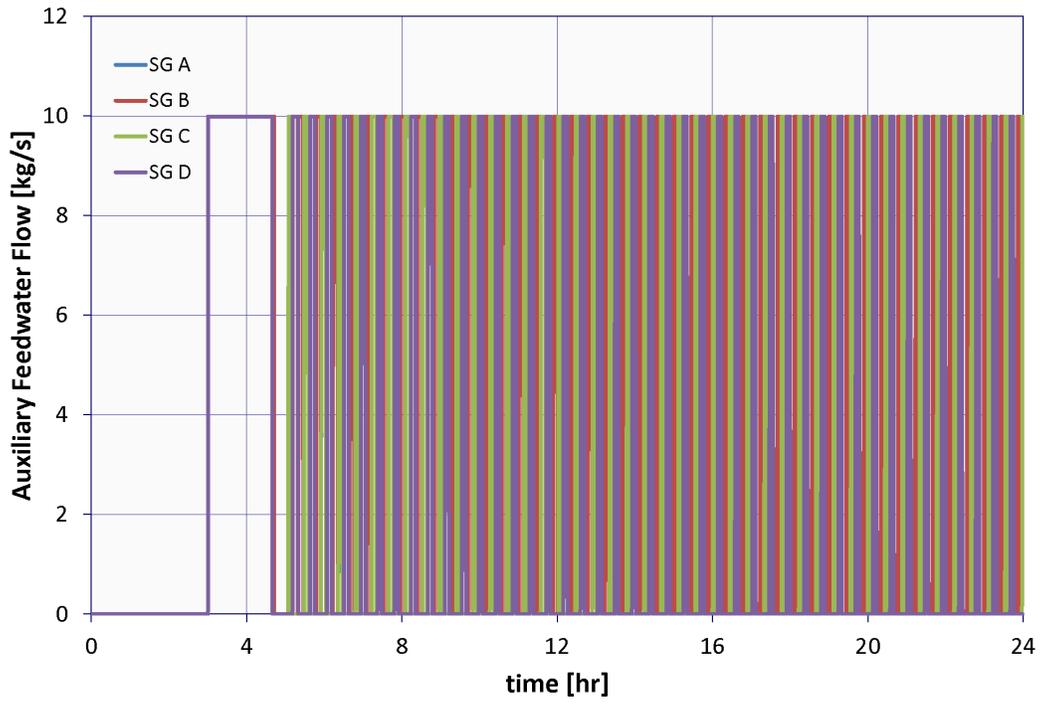


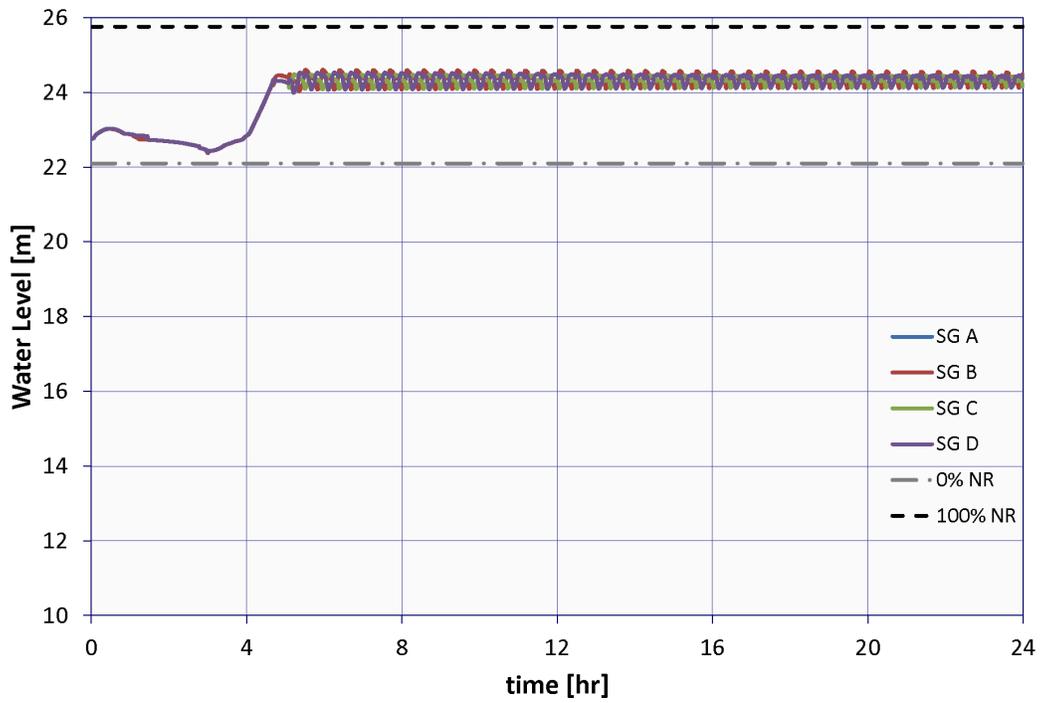
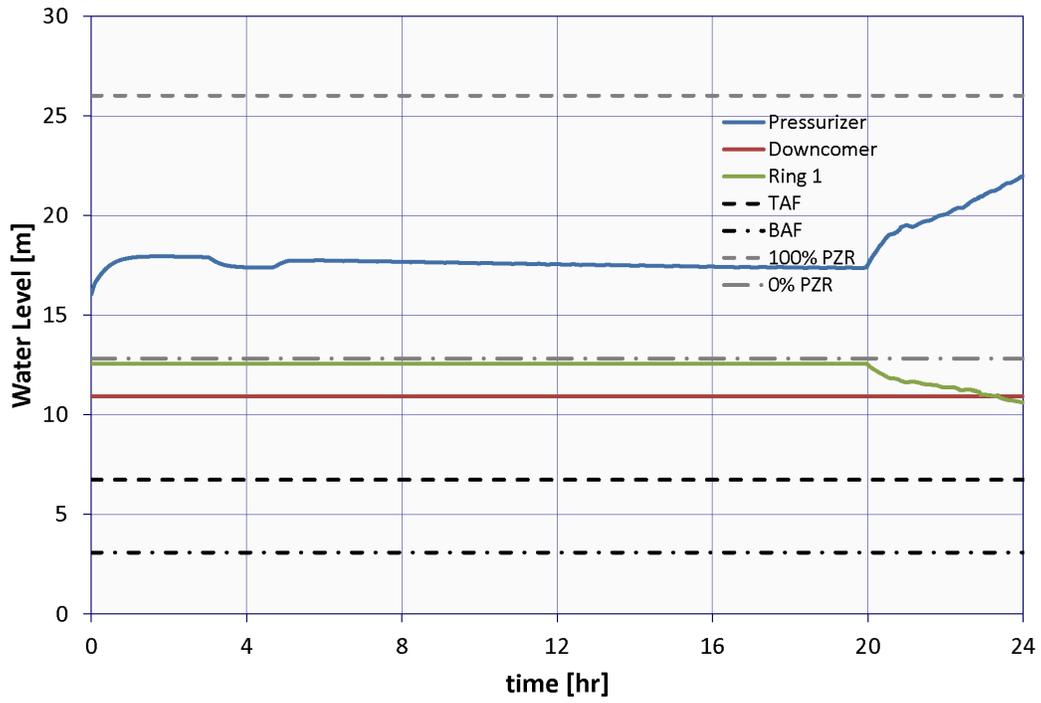


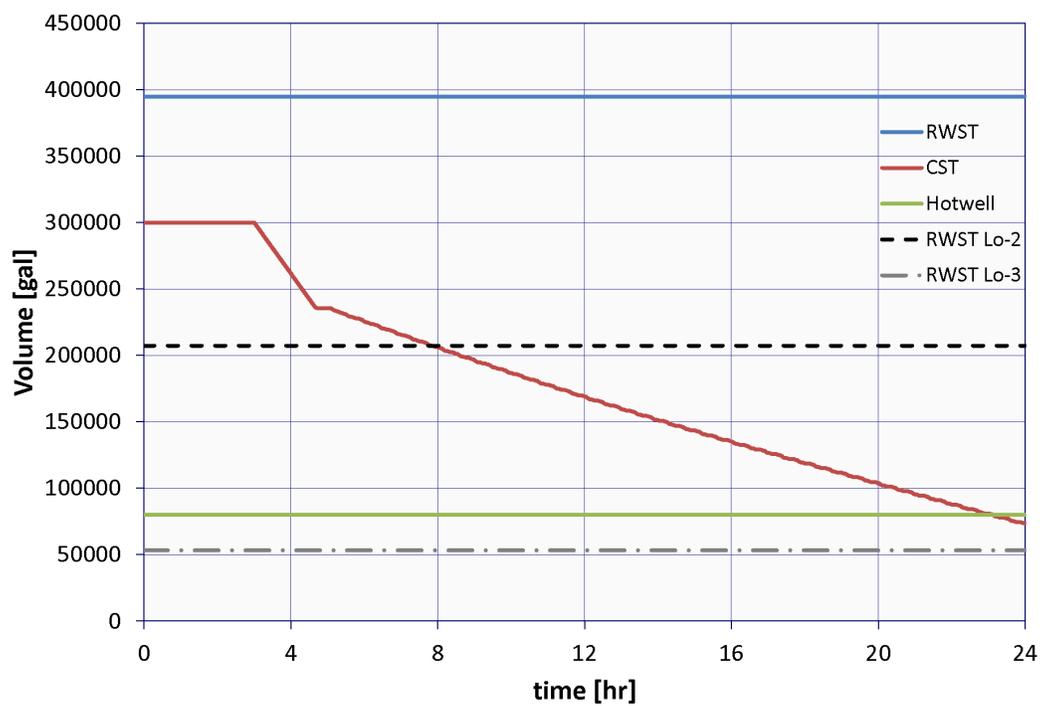
F.1.10 Case 10: SG at 18% NR Level, 6 hr after Shutdown, Initiate AFW at 3 hr











## F.2 Mode 5 Calculations

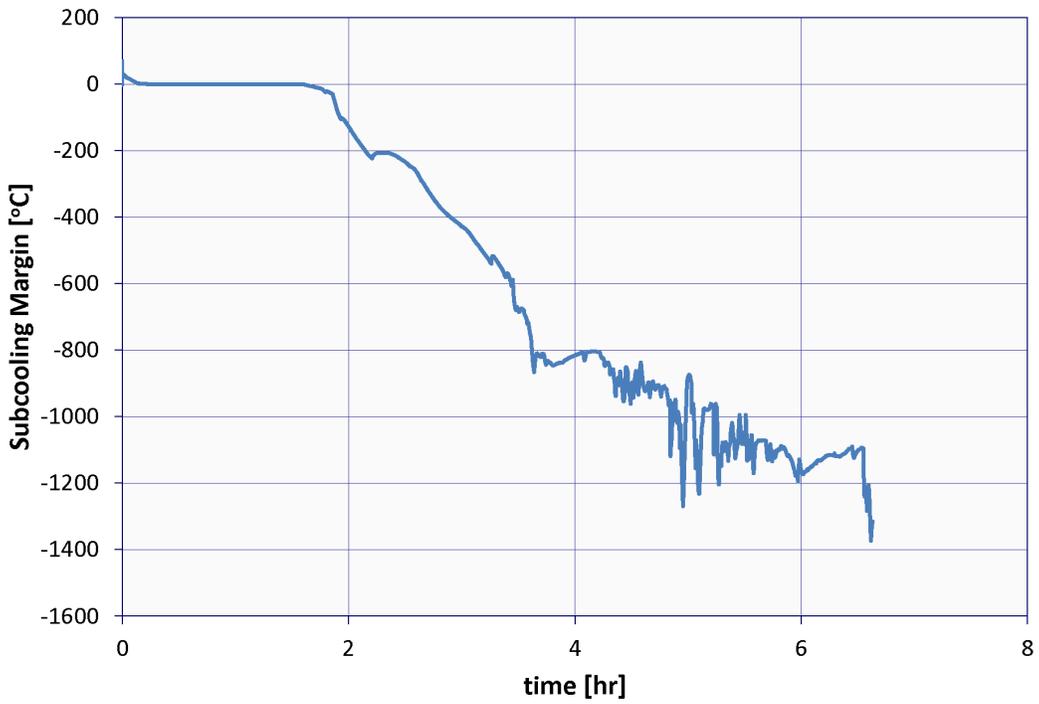
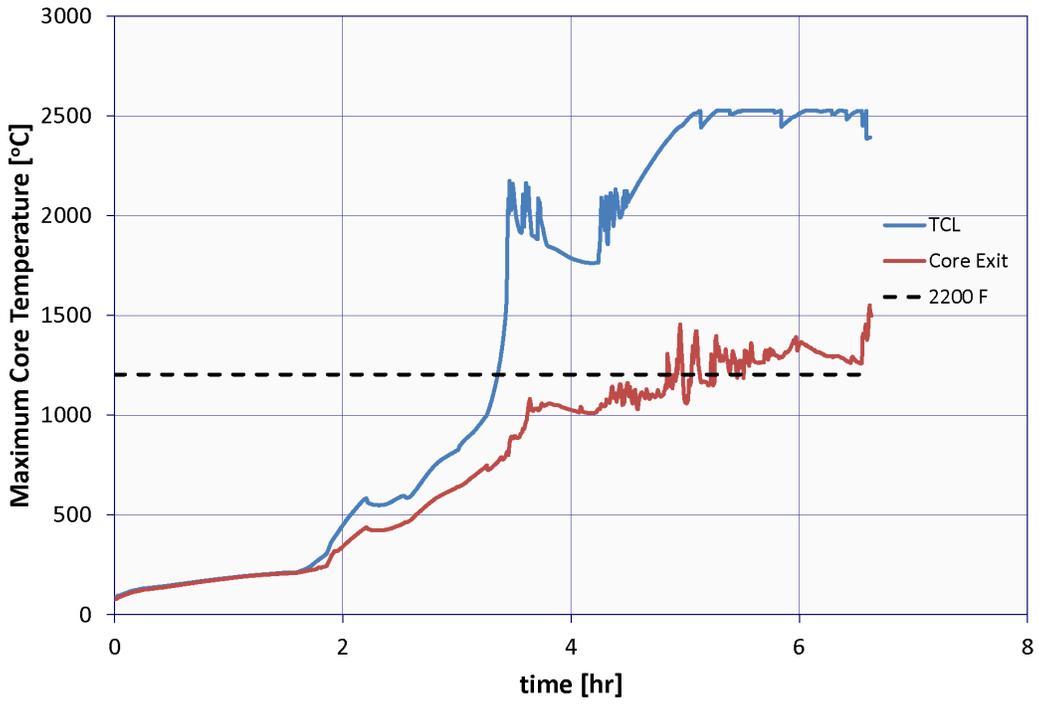
### Notes

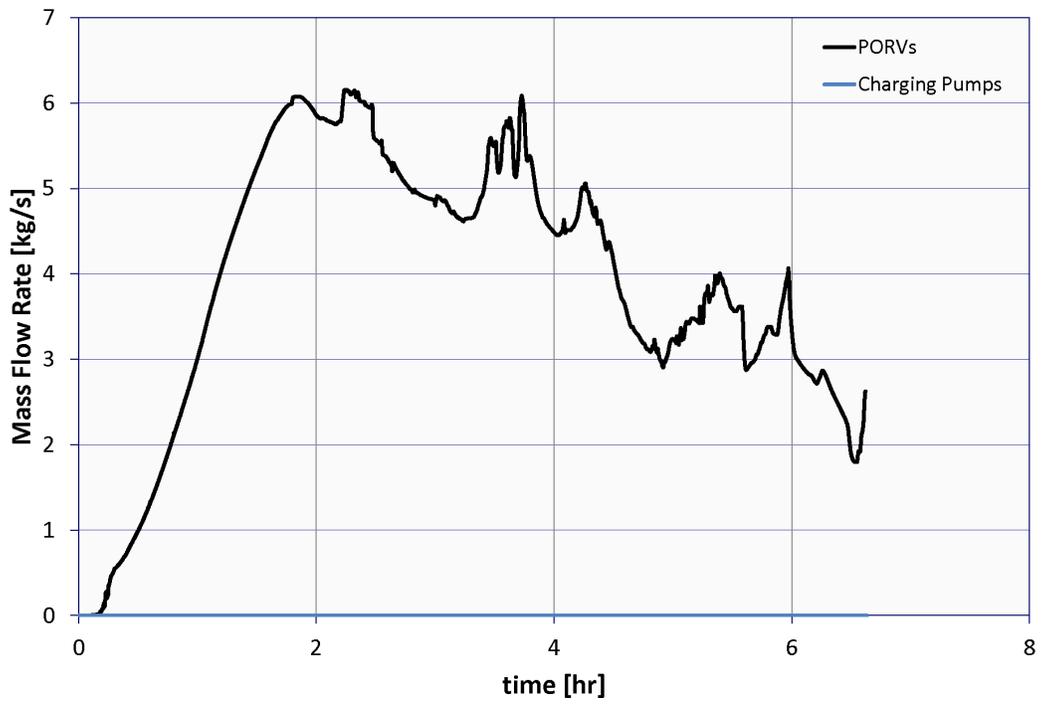
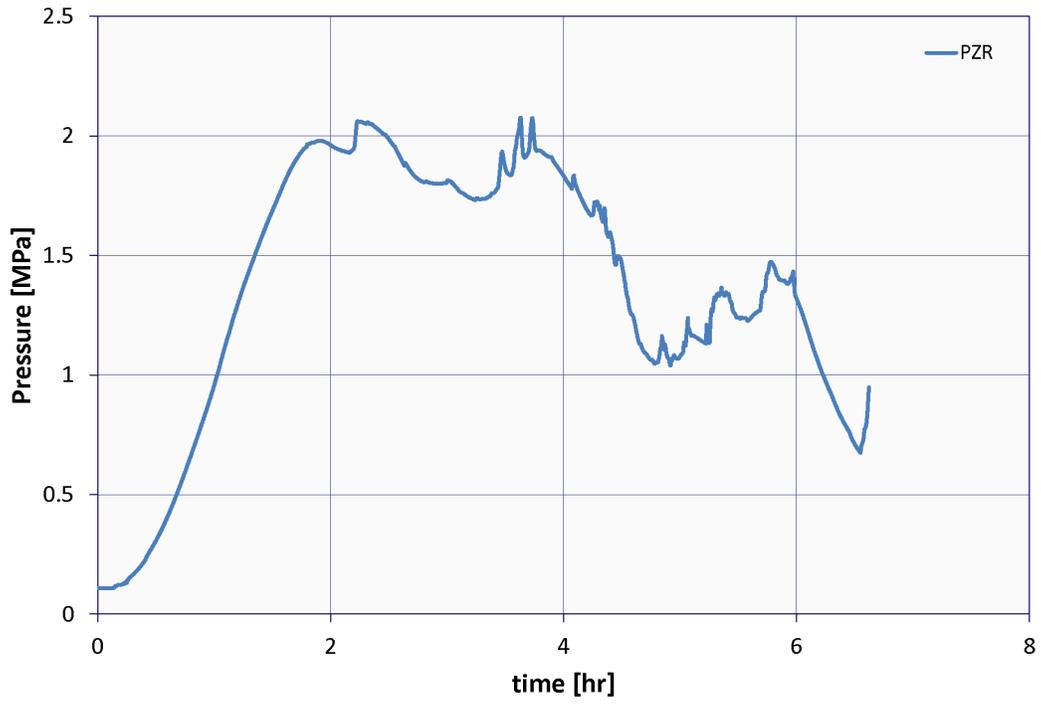
The following list identifies some of the changes that were made to the MELCOR input deck in order to perform Mode 5 shutdown calculations.

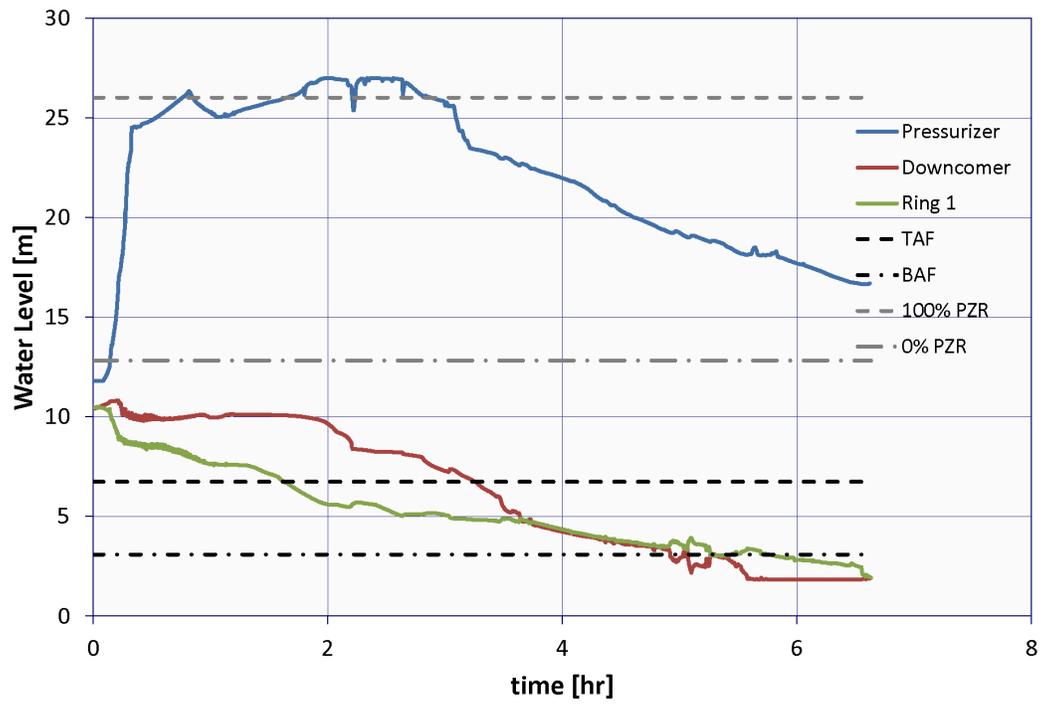
- Logic has been added to model the shutdown cooling function of the RHR system. This logic is set up such that RHR flow rate is adjusted in order to maintain a constant coolant temperature, up to the maximum flow rate of the system. The logic also includes provisions to achieve a target cooldown rate; however, this feature is not used in any of the shutdown calculations performed for this report.
- Pressurizer level control logic has been modified to control water level during the steady-state portion of the calculation. For the Mode 5 calculations, level control is based on RPV level because the level is assumed to be at the vessel flange, which is below the bottom of the pressurizer.
- Pressurizer heaters have been disabled because the pressurizer is empty.
- Logic that makes it possible to turn off ECCS flow to prevent overfilling the pressurizer has been modified in order to simulate recovery actions in which operators inject using a charging pump when RPV level is low. This feature is exercised in Mode 5 Cases 2, 5, and 8.
- The decay heat curves have been shifted in order to simulate the desired times after trip. For example, the decay heat curve is shifted by 40 hours for Mode 5 Cases 1–3. Note that during the steady-state portion of the calculation, the decay heat is assumed to be constant and to equal the decay power at 40 hours. The same is true for all other times since subcriticality that are analyzed in Section 5.8.3 of the report.
- Initial temperature and pressure of RCS control volumes have been set to 170 degrees F (349.8 K) and atmospheric pressure.
- Flow paths have been added to model the antisiphon hole in the line leading from the pressurizer power-operated relief valves to the pressurizer relief tank (PRT). It is necessary to include this flow path because, otherwise, the RCS will draw a vacuum when RHR is operating.
- The flow path representing the PRT rupture disk is held open throughout the Mode 5 calculations. It is expected that the PRT would be vented to containment during this operating stage; however, the characteristics of this vent path are unknown. In the absence of better information, the PRT rupture disk flow path is used as the vent path for this model.

- Flow paths from CV 310 and 311 to 320 and 321 have been deleted, or the valves in the flow paths have been closed, to simulate loop stop valve closure. The same is true for flow paths between CV 346 and 348 in the cold leg and between analogous control volumes in the other loops.
- Cold volumes have been used in place of hot volumes for RCS control volumes. This decreases the RCS volume by approximately 1 percent.

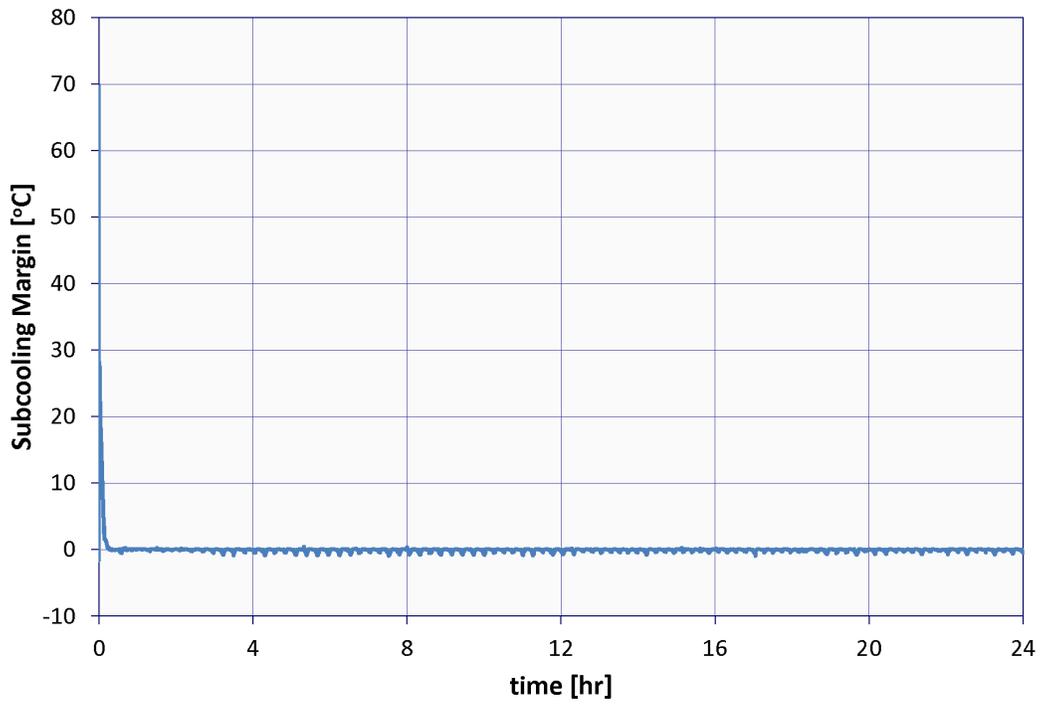
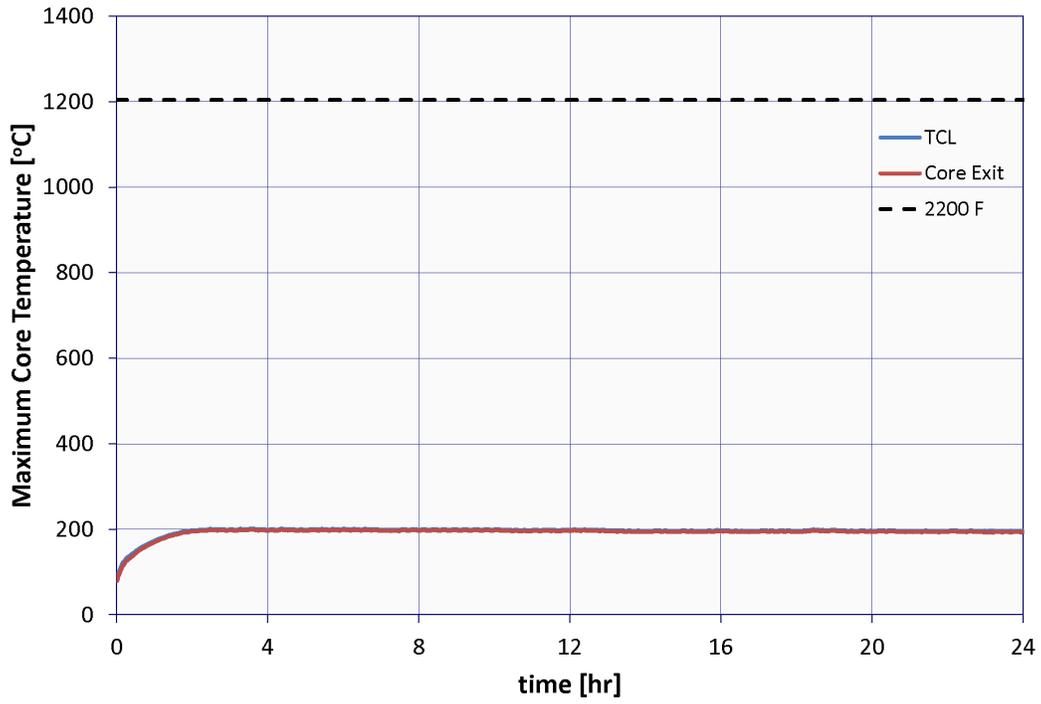
**F.2.1 Case 1: 40 hr after Shutdown, No Recovery Actions**

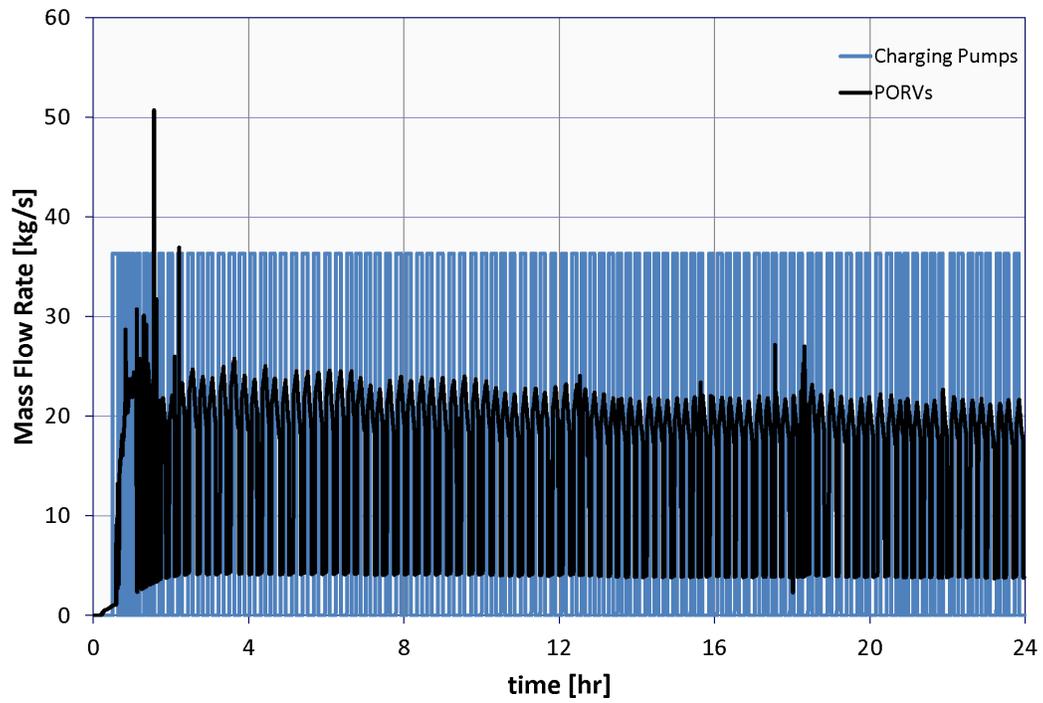
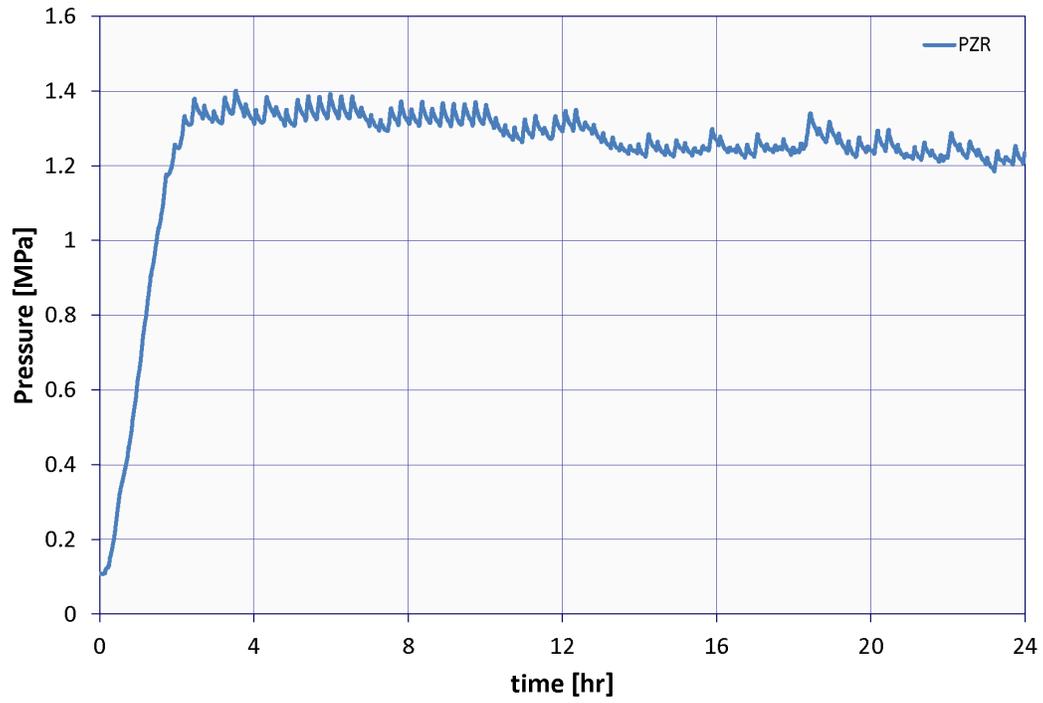


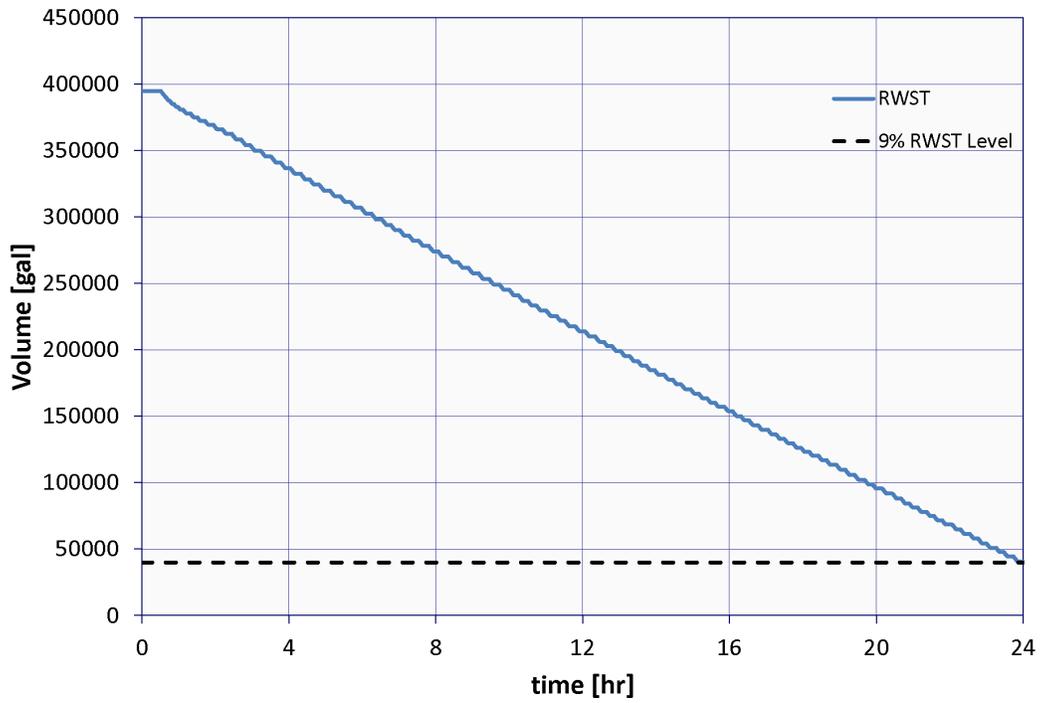
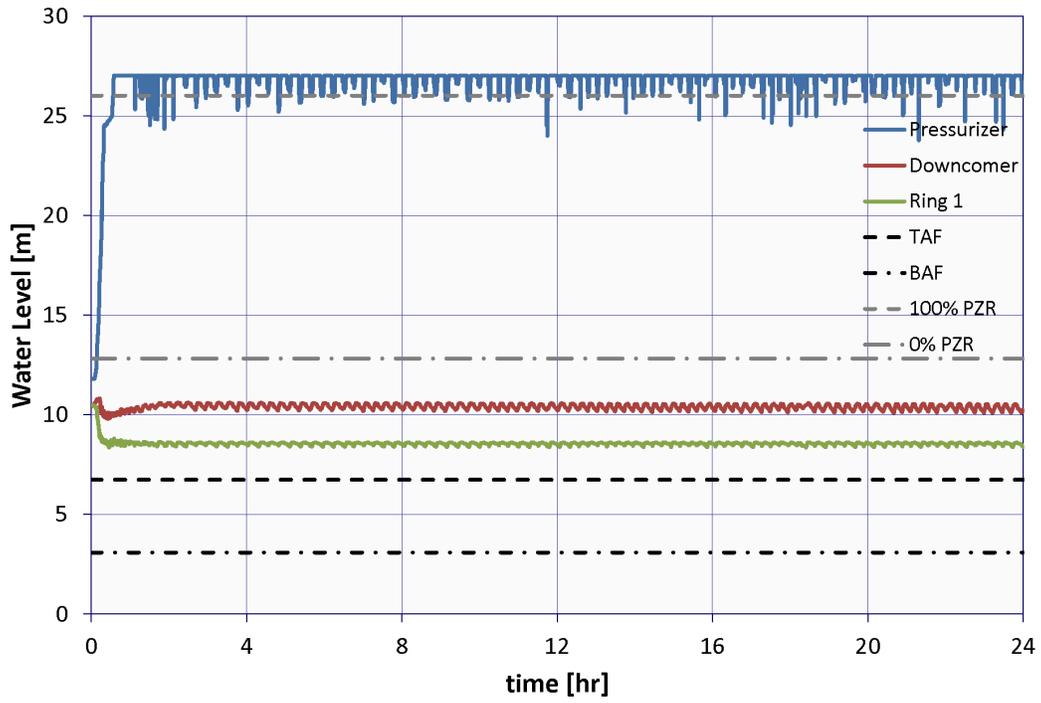




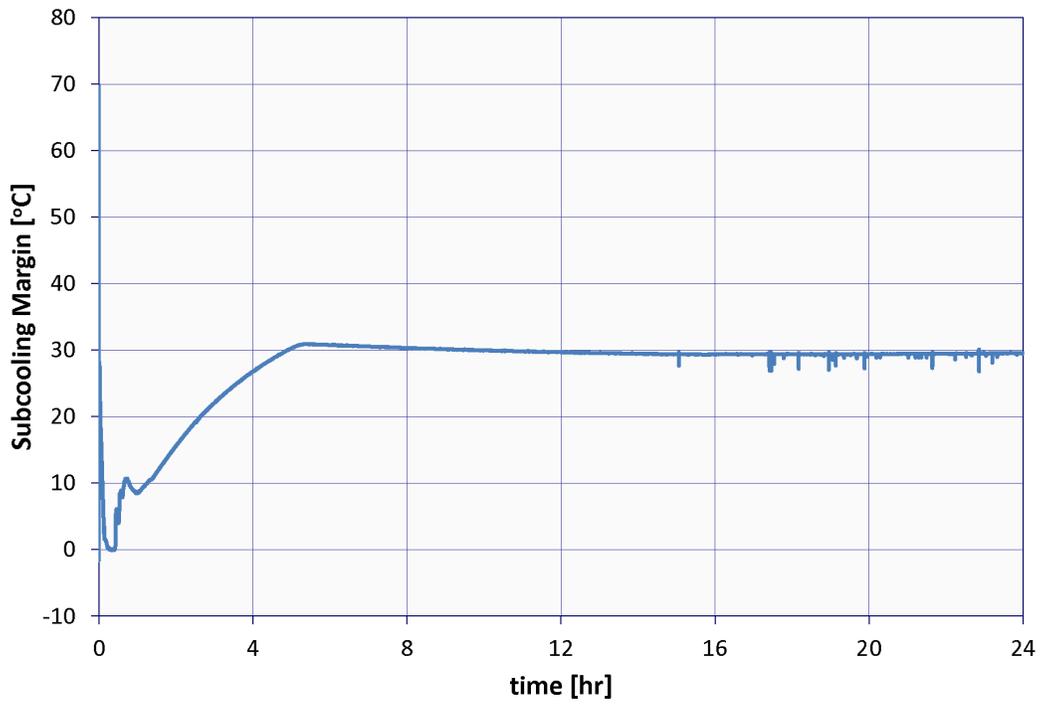
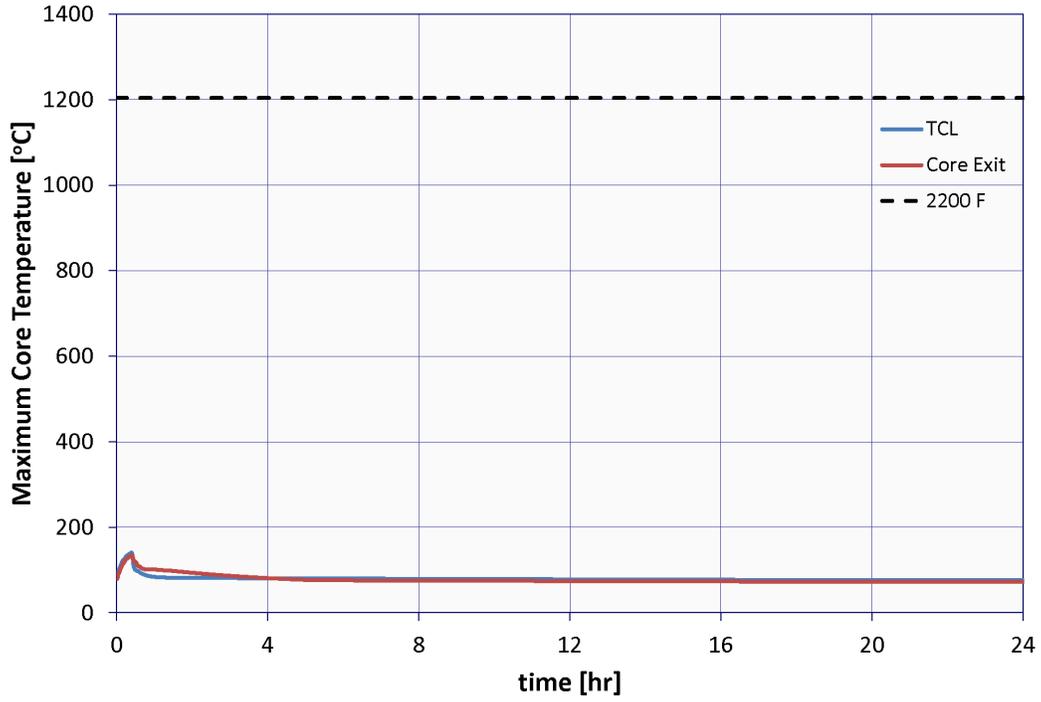
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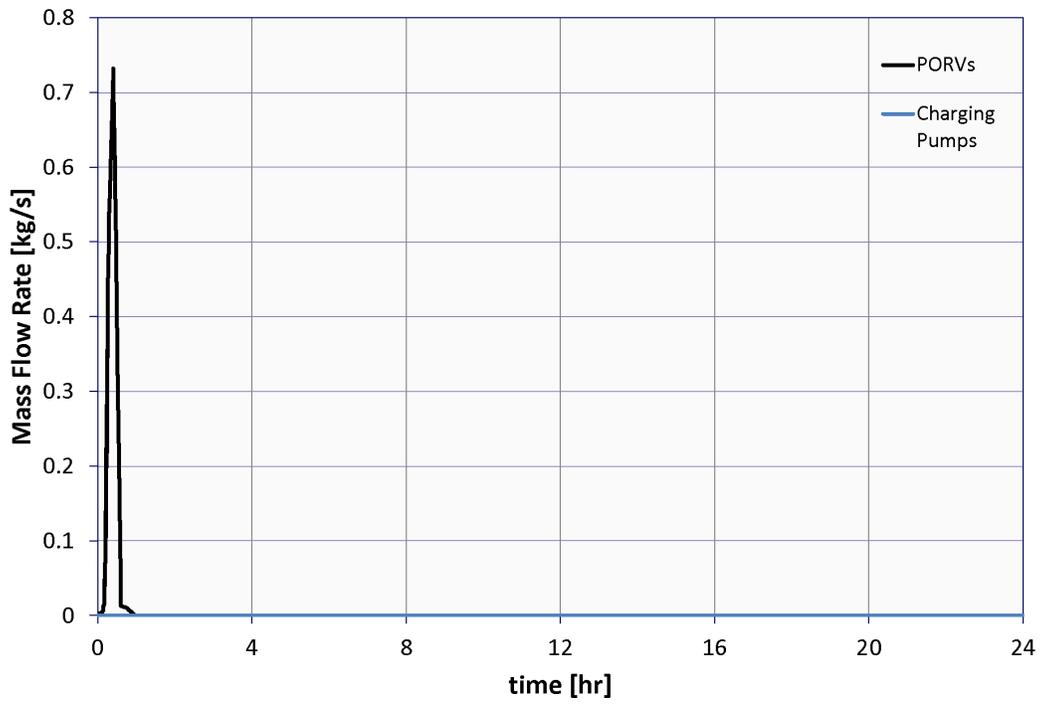
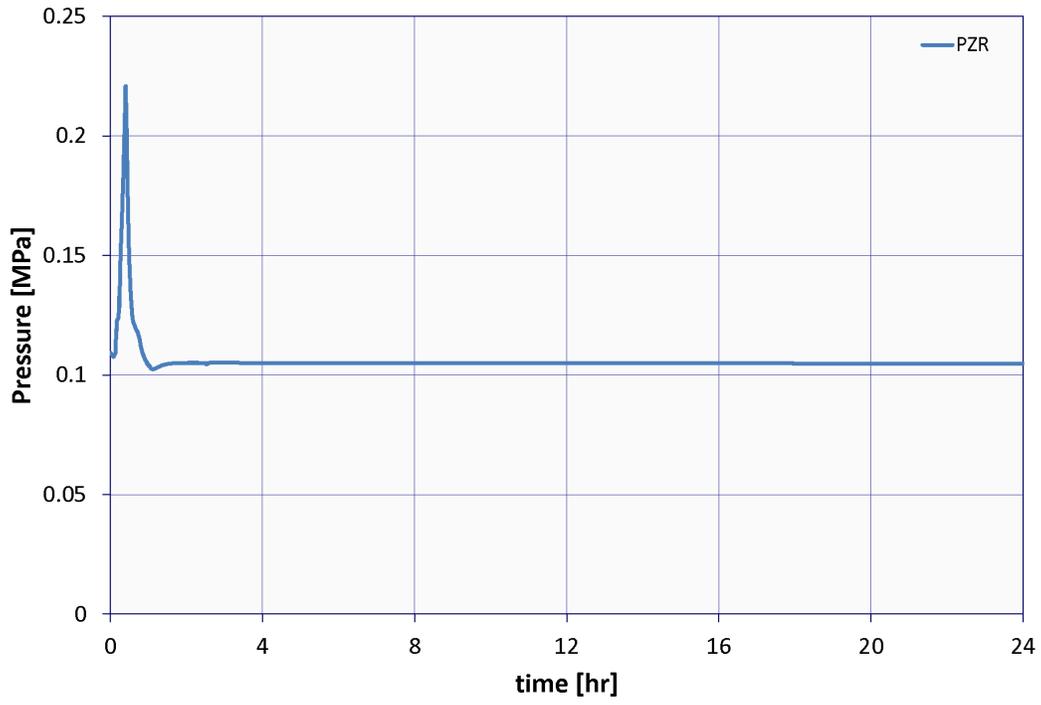


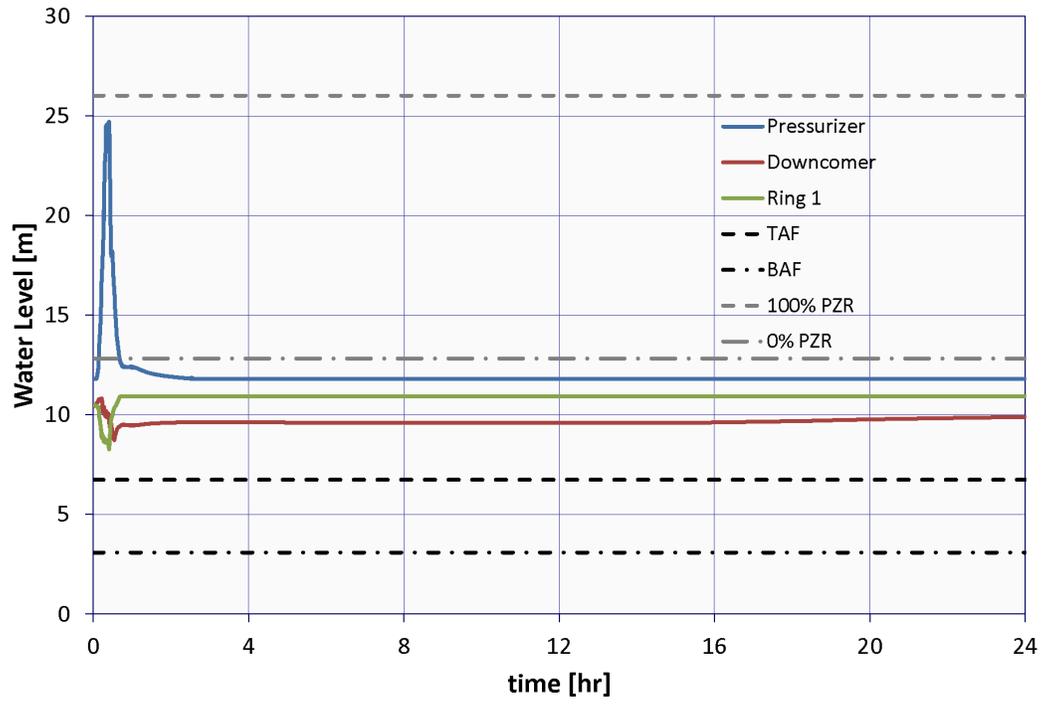




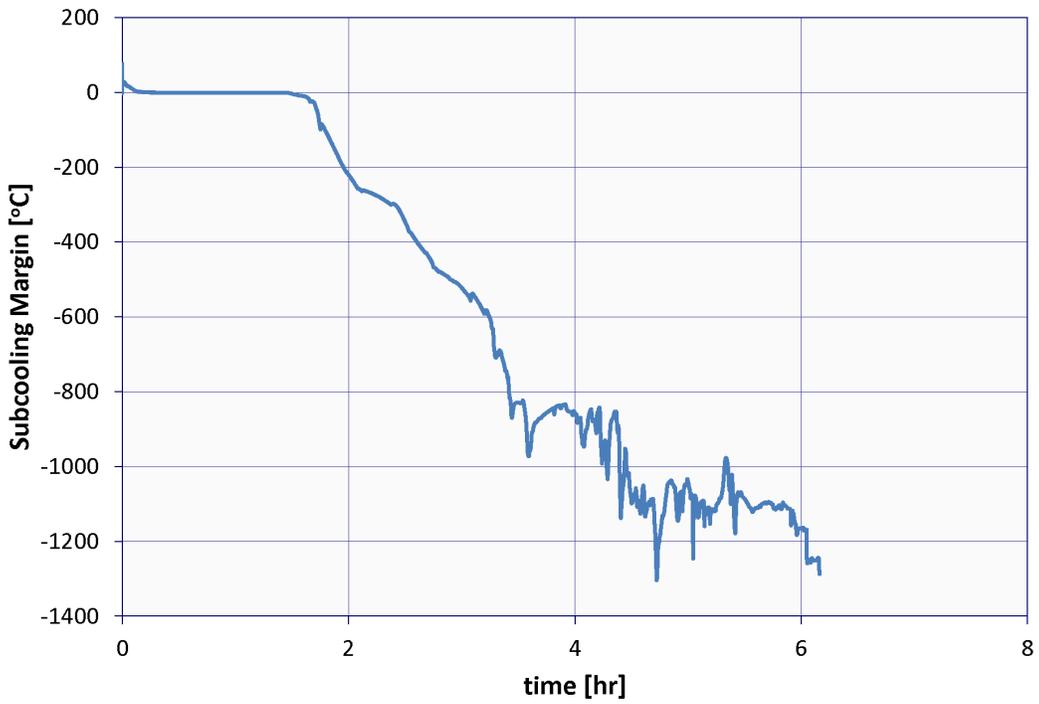
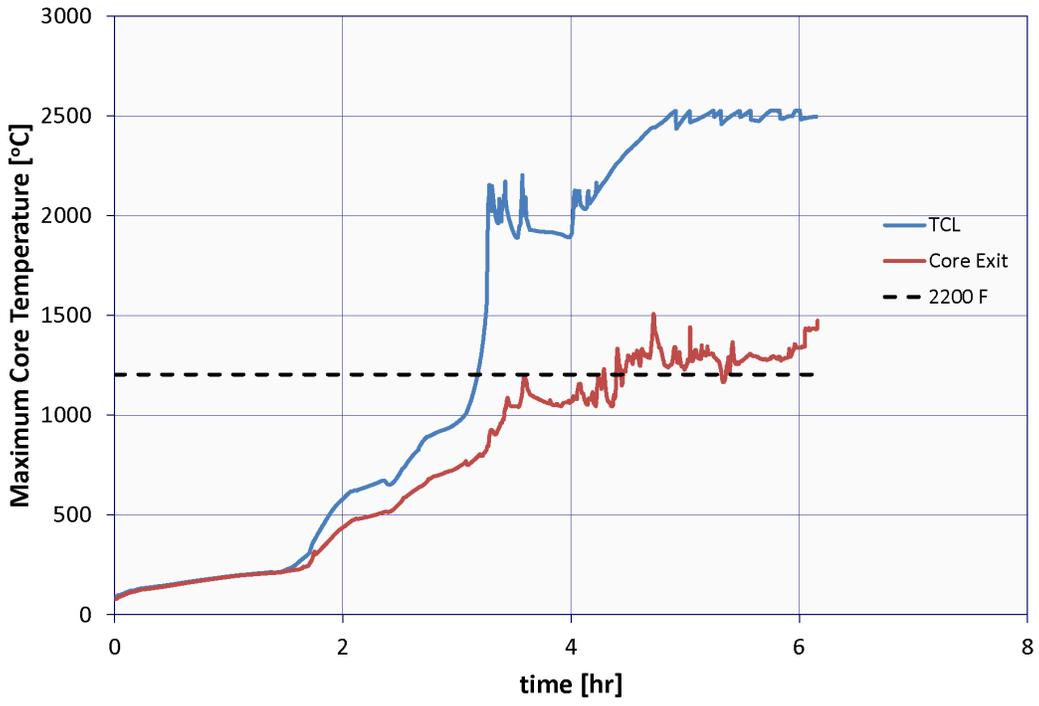
**F.2.3 Case 3: 40 hr after Shutdown, Recover RHR at 23 Minutes**

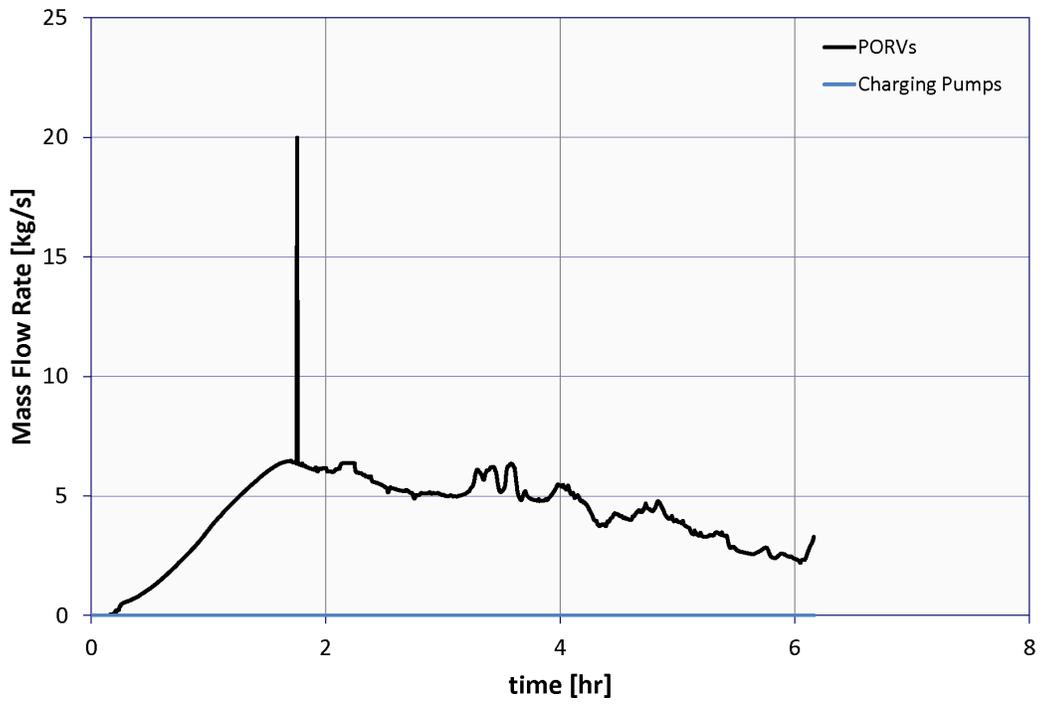
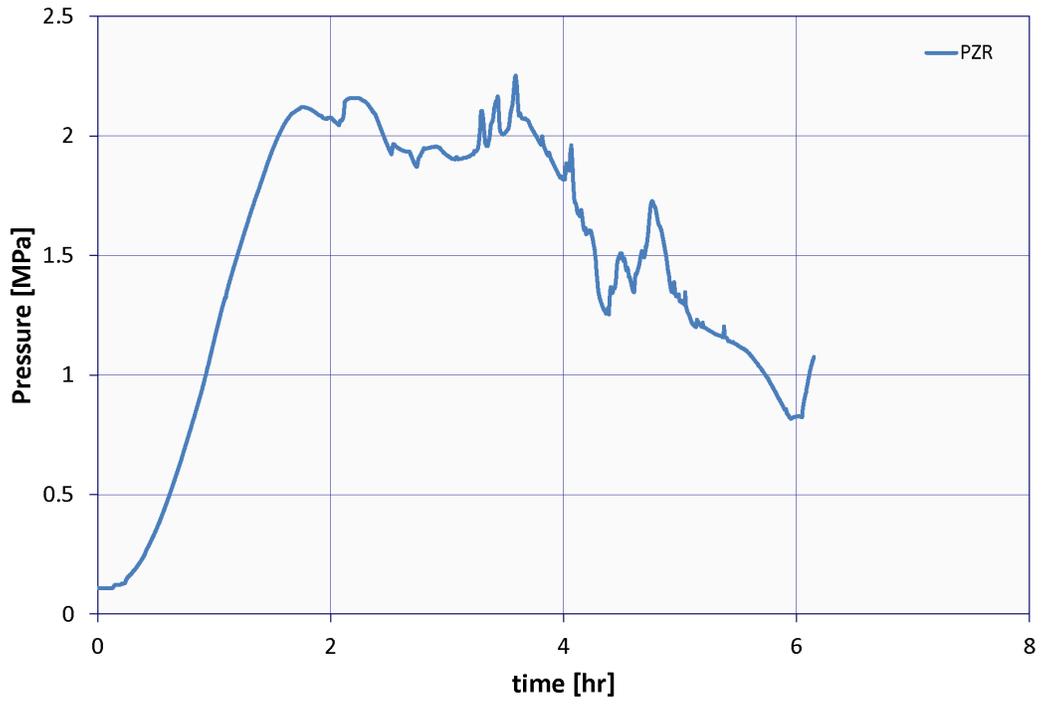


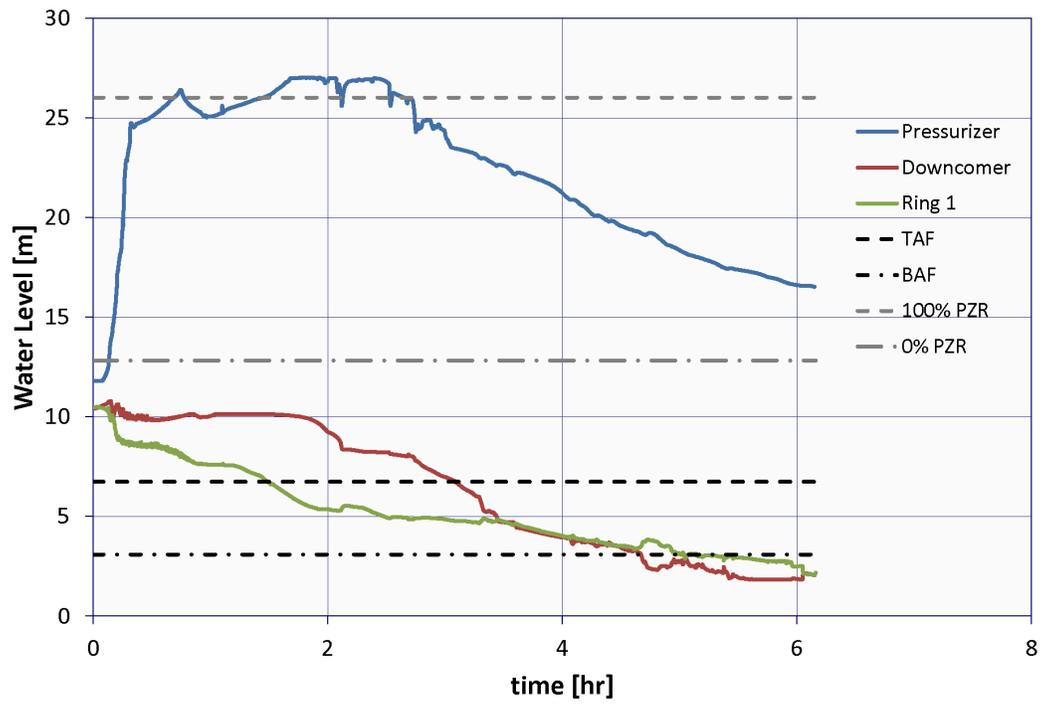




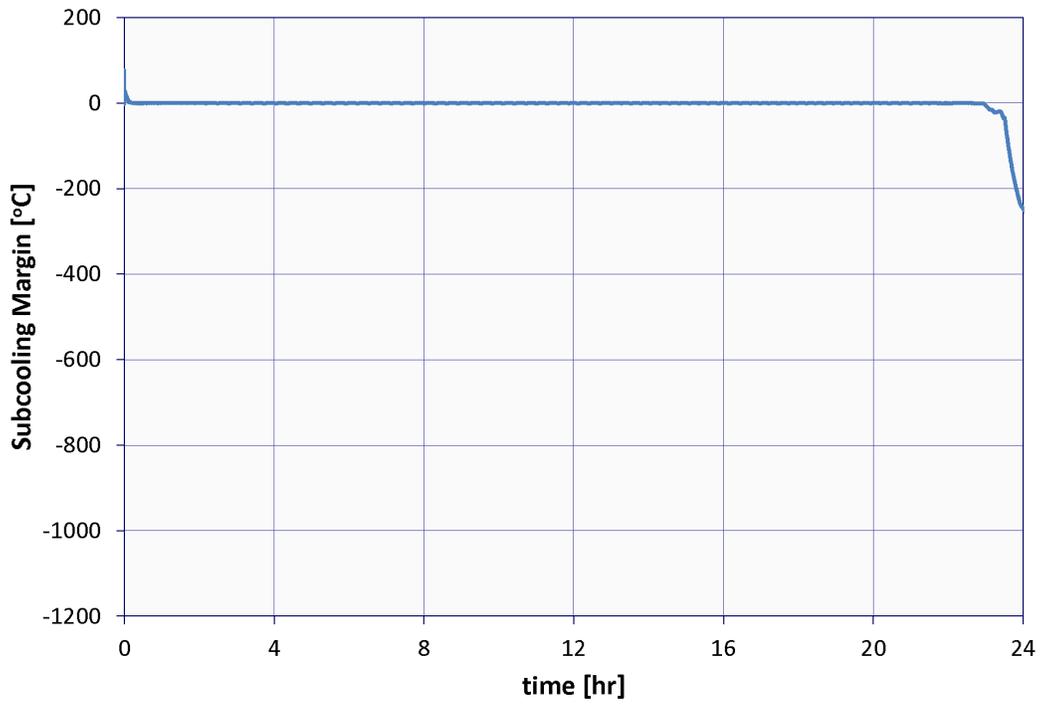
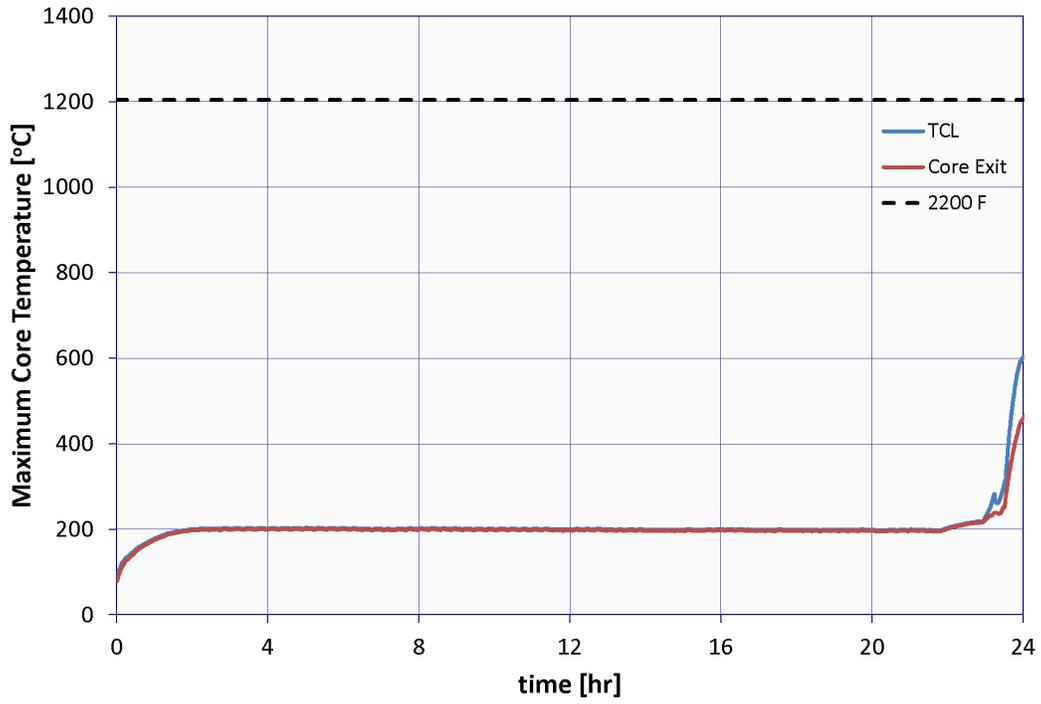
**F.2.4 Case 4: 30 hr after Shutdown, No Recovery Actions**

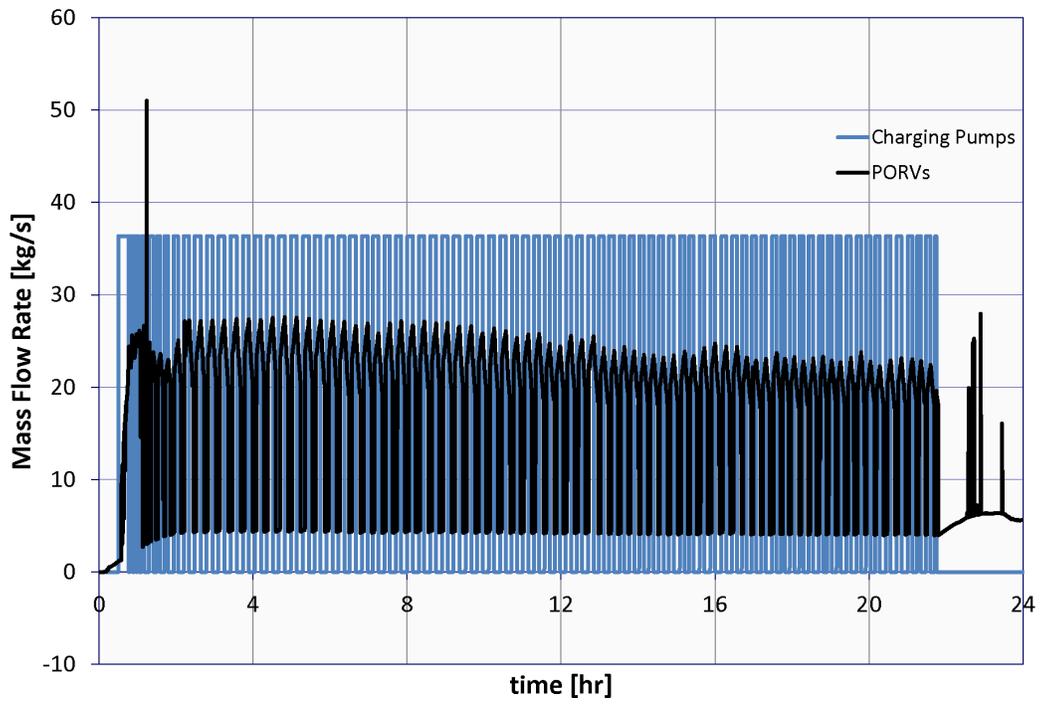
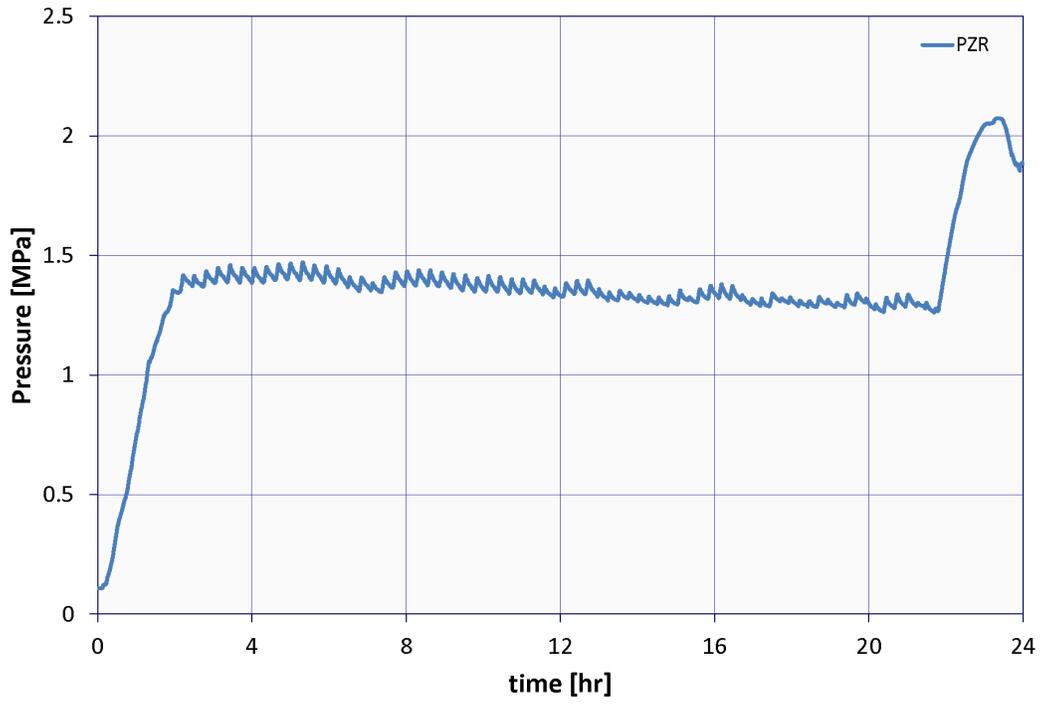


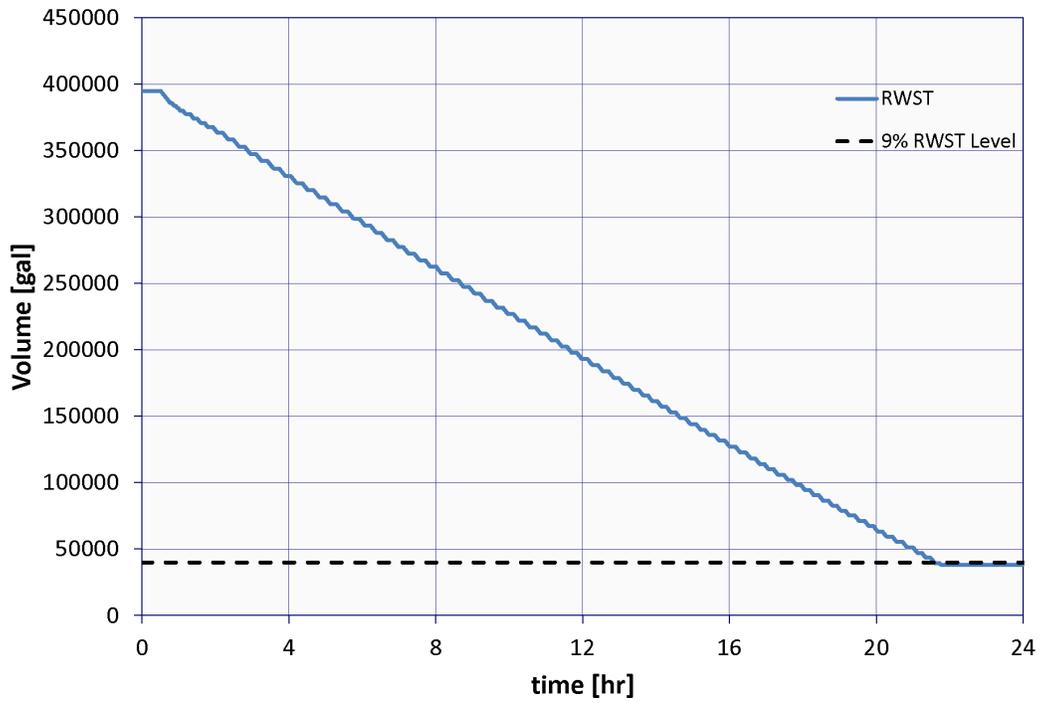
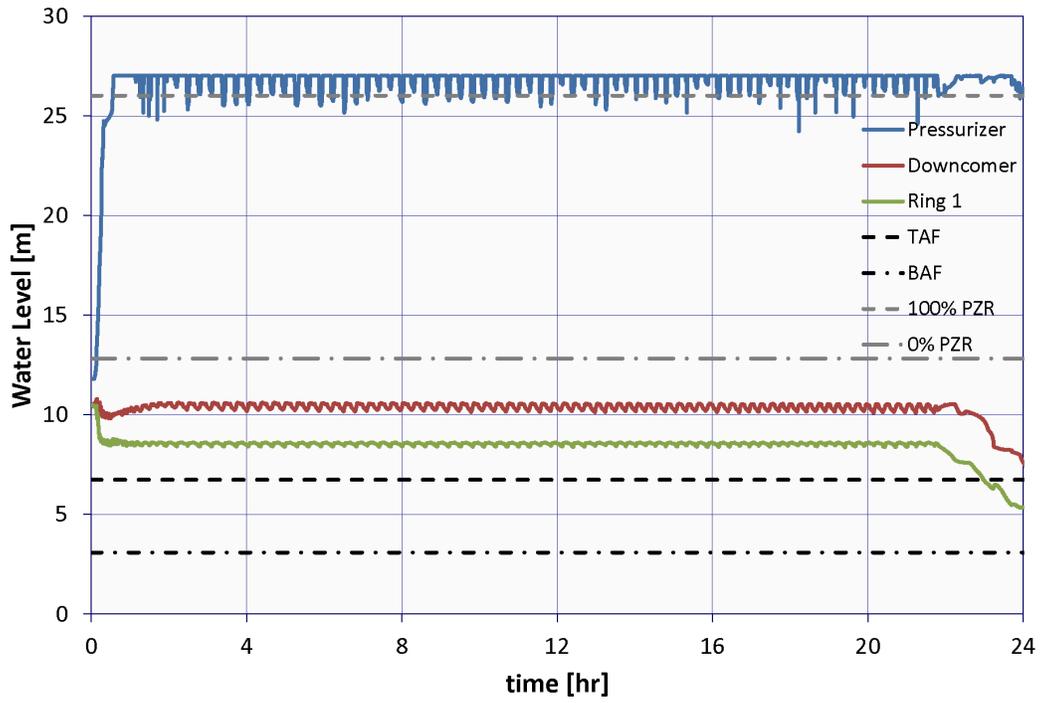




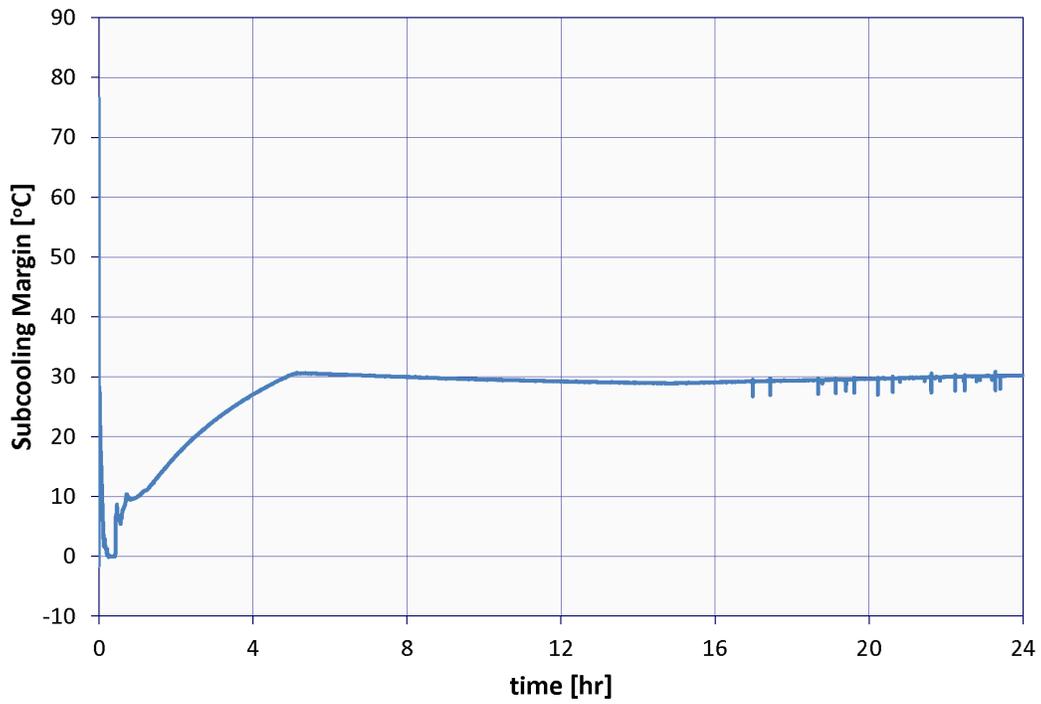
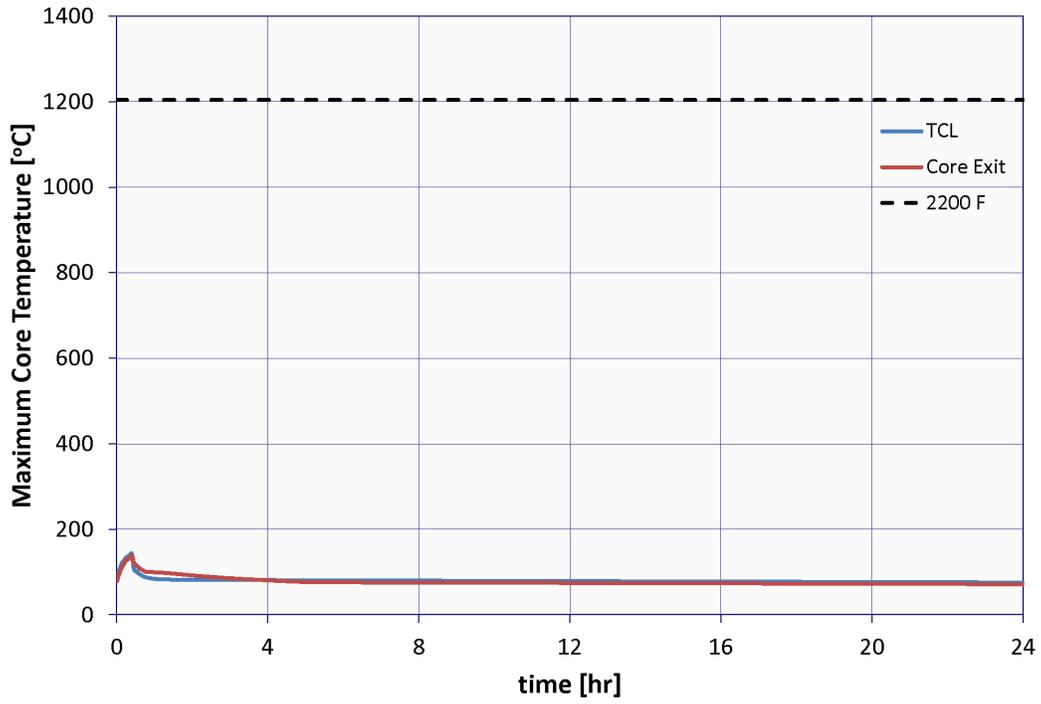
**F.2.5 Case 5: 30 hr after Shutdown, Start CCP on Low RPV Level**

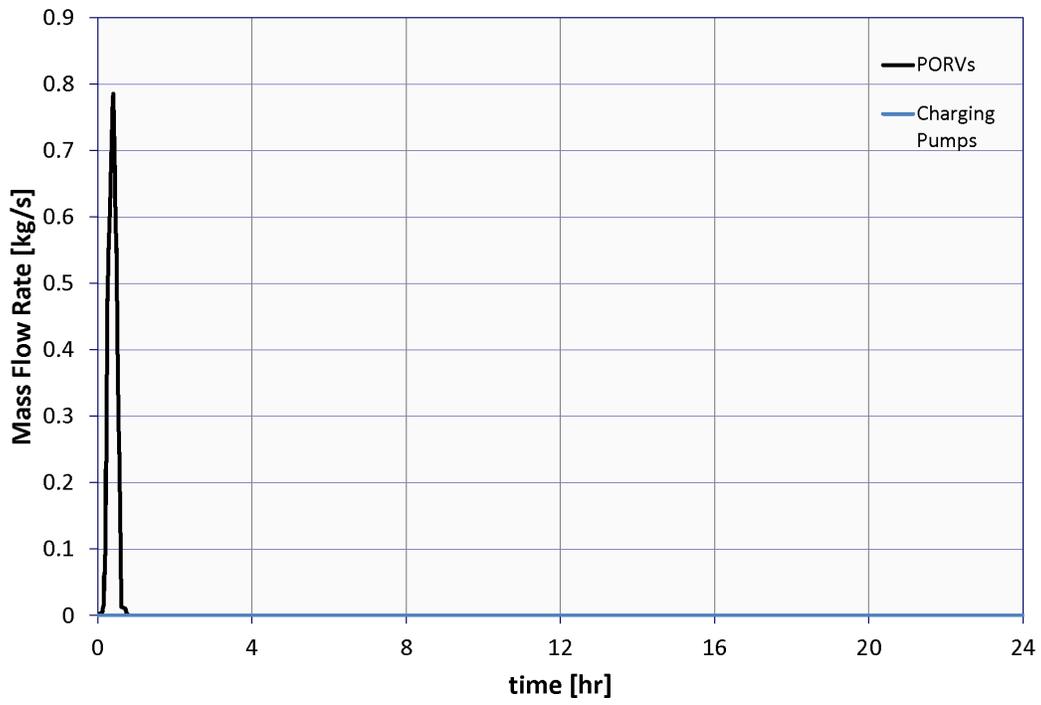
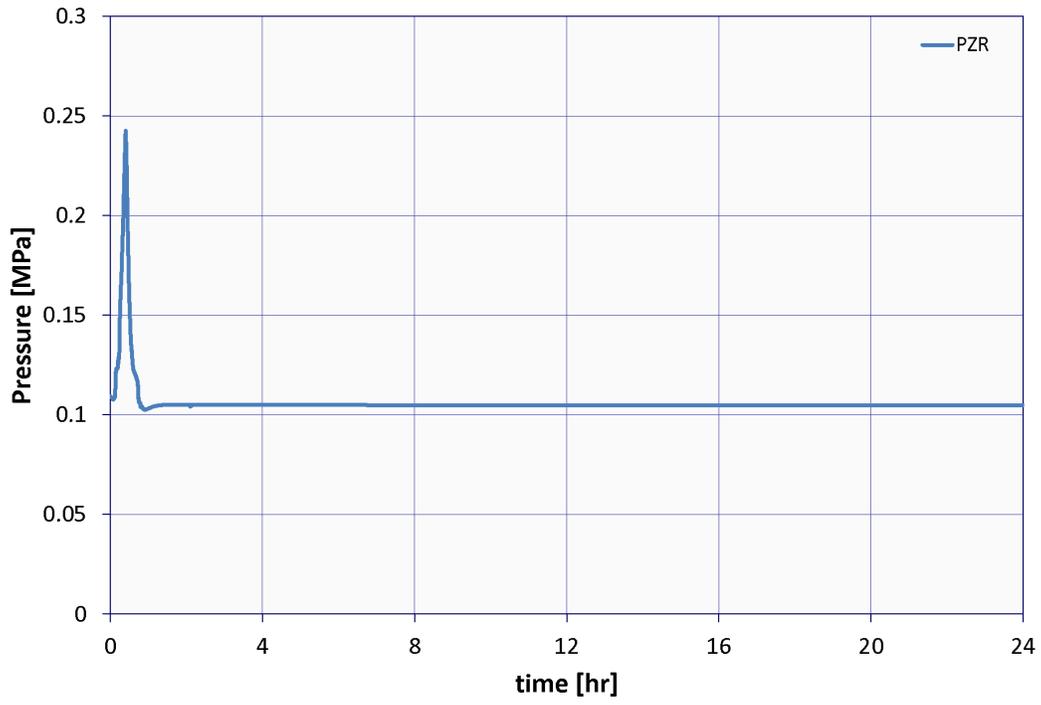


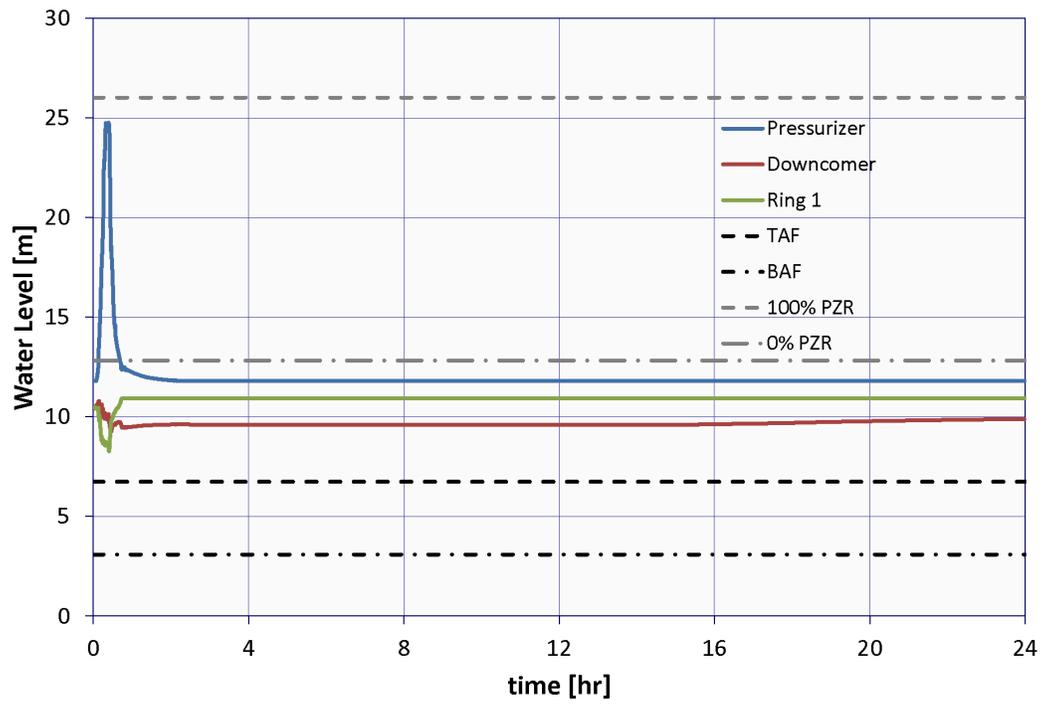




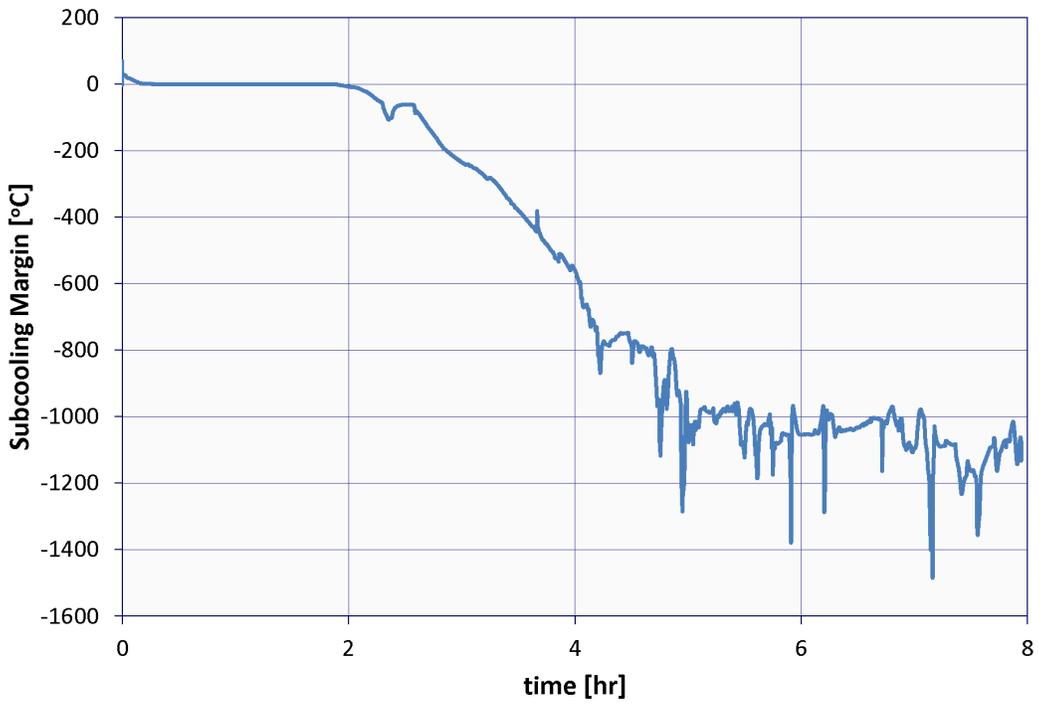
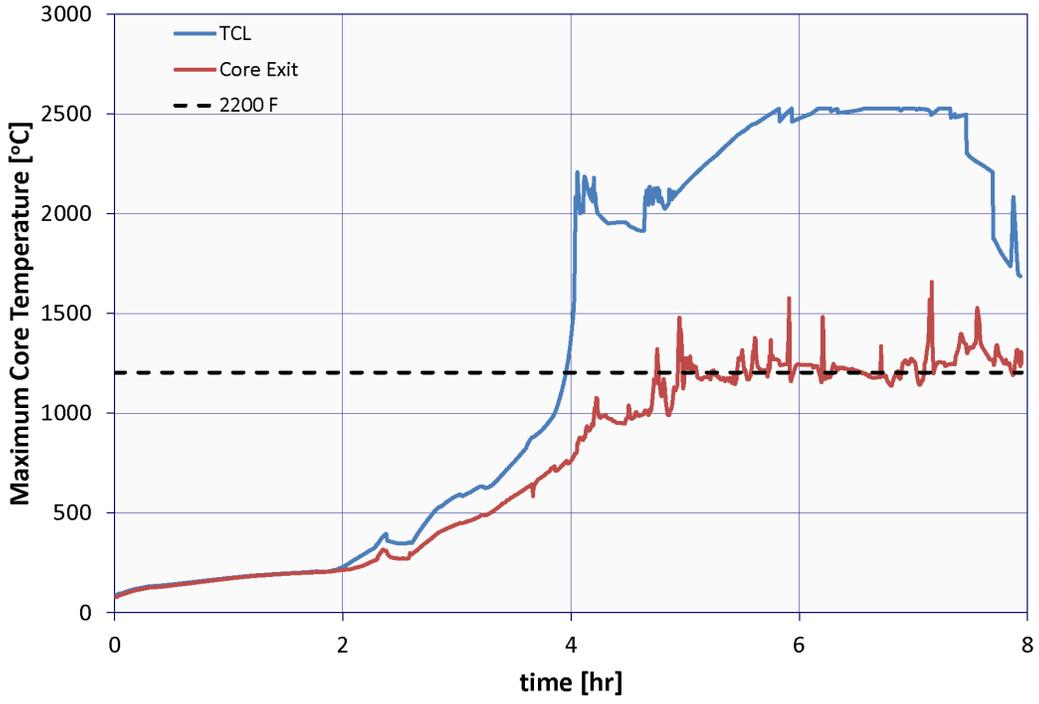
F.2.6 Case 6: 30 hr after Shutdown, Recover RHR at 23 Minutes

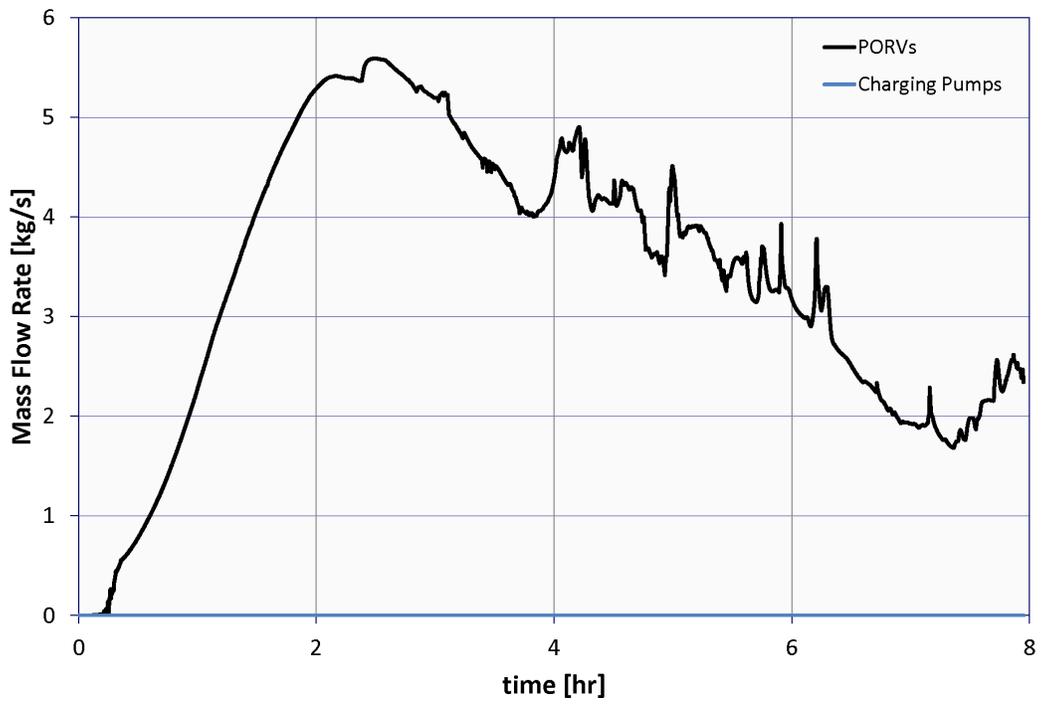
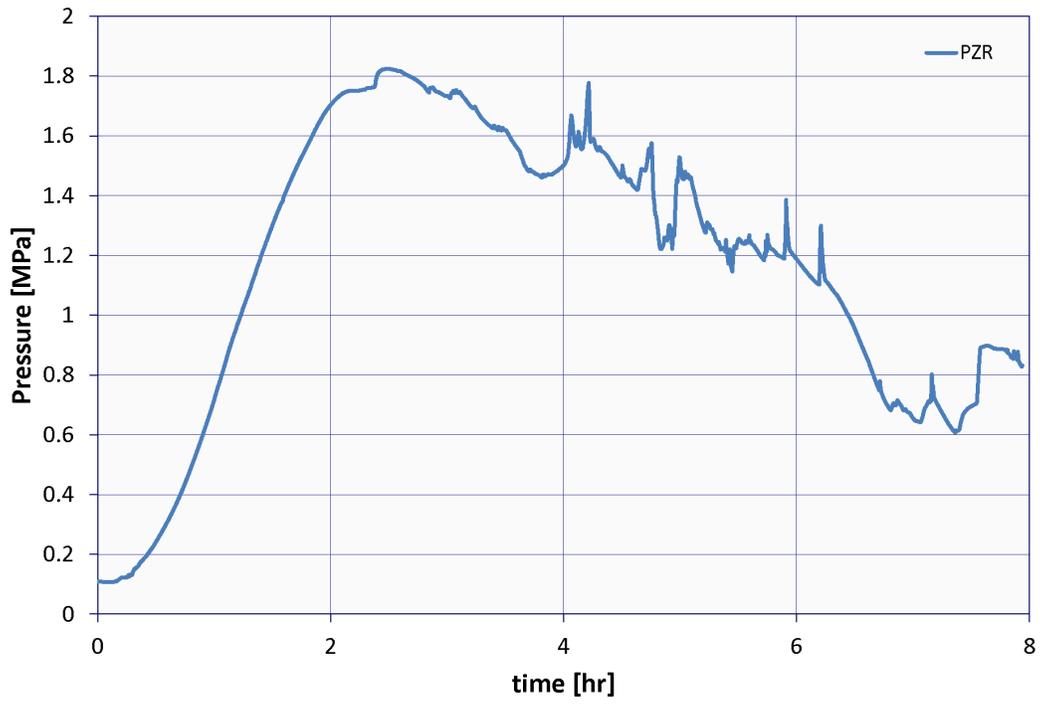


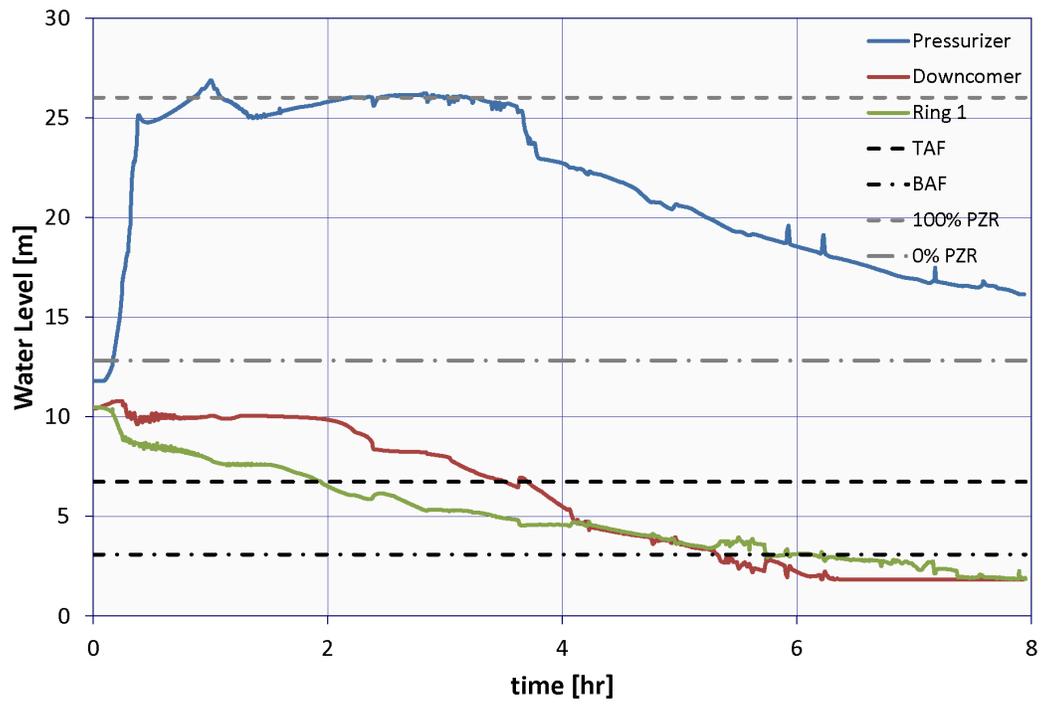




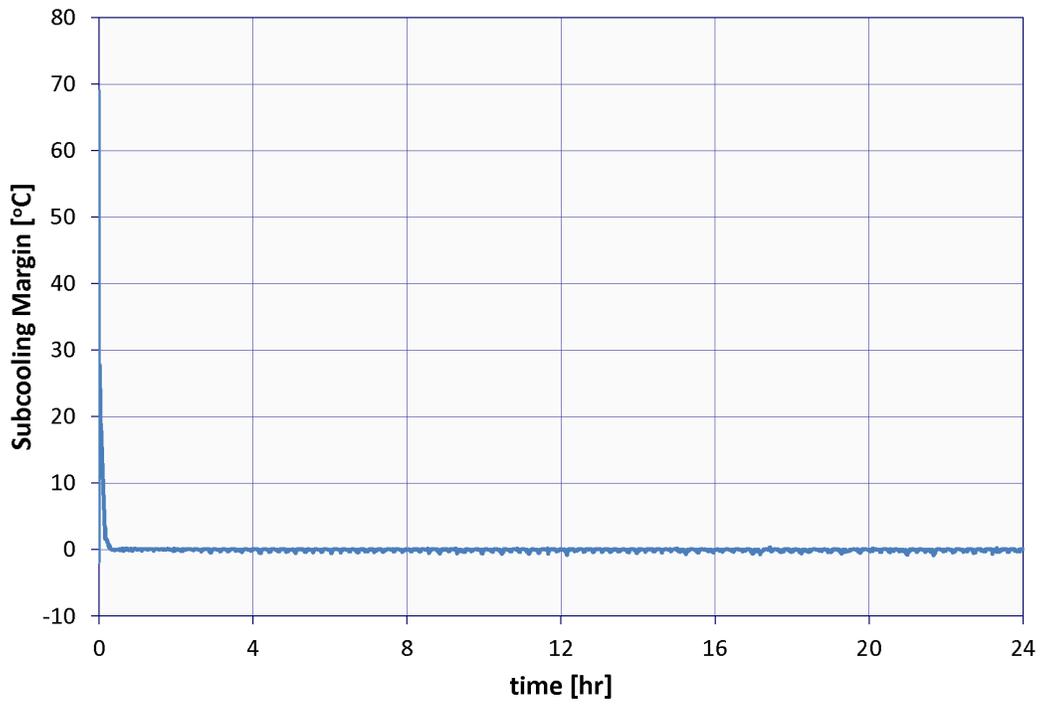
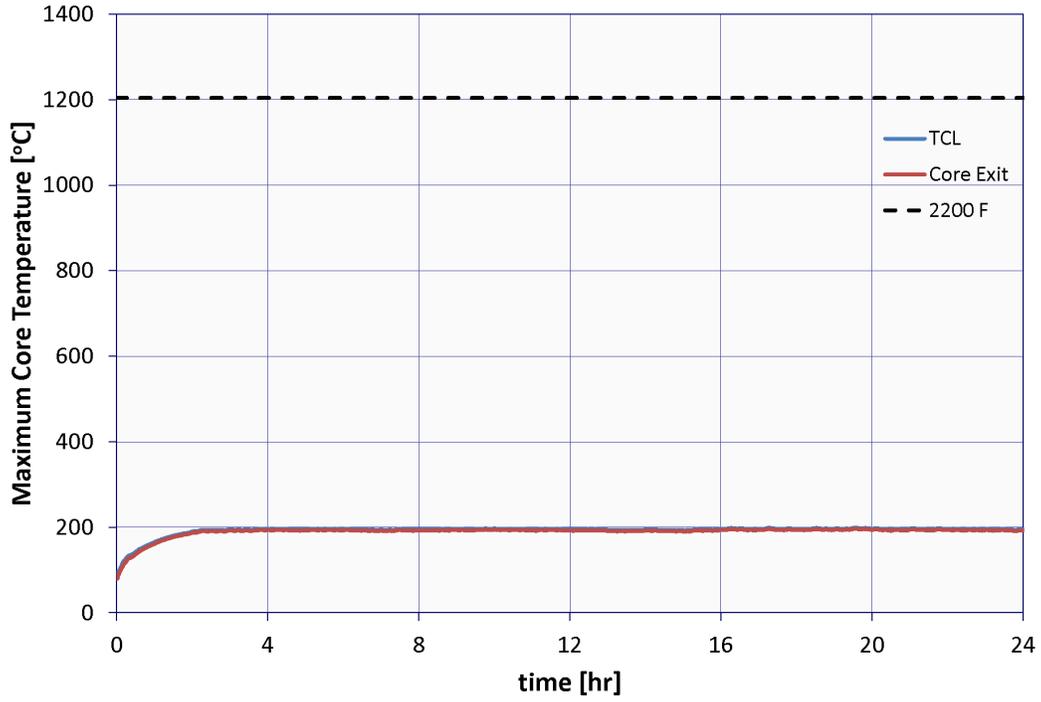
**F.2.7 Case 7: 60 hr after Shutdown, No Recovery Actions**

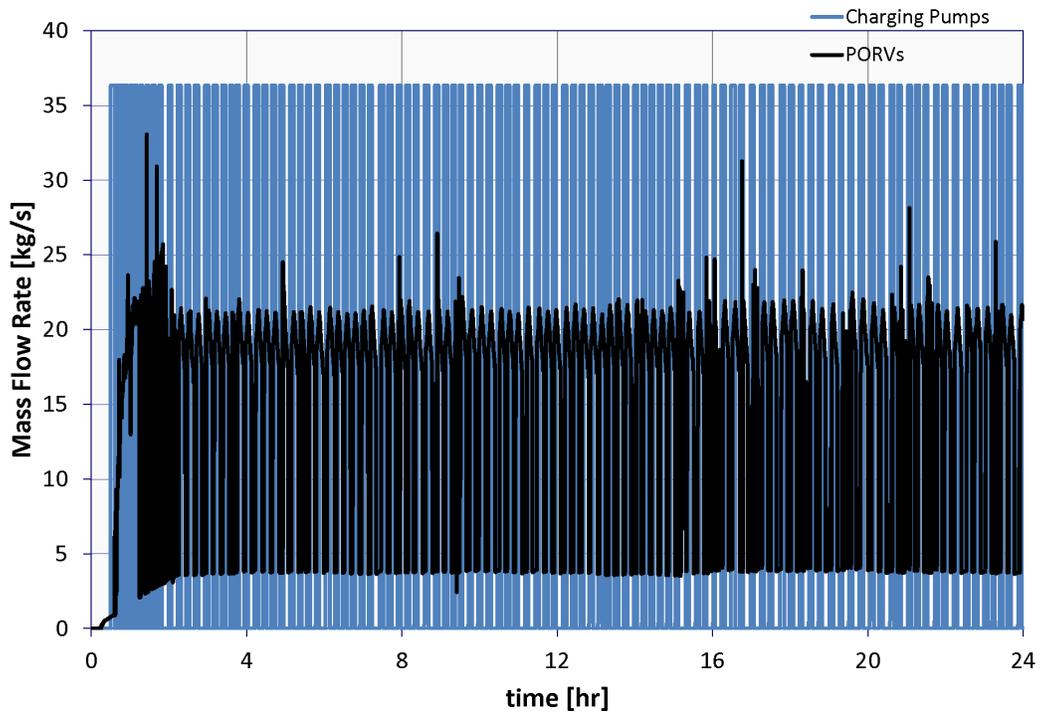
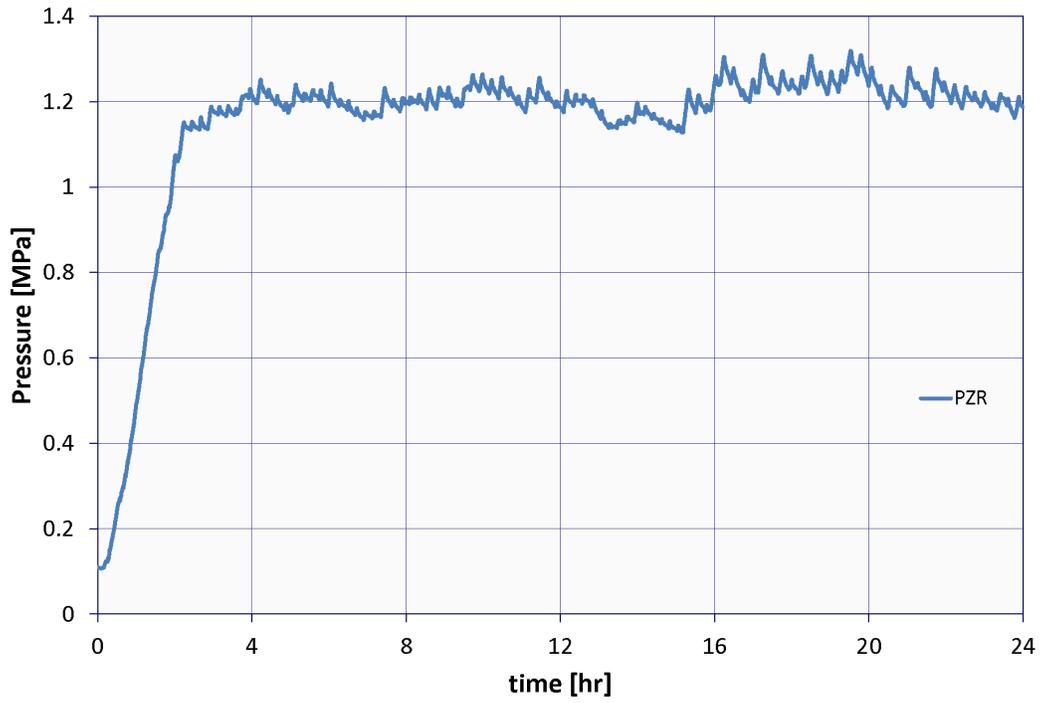


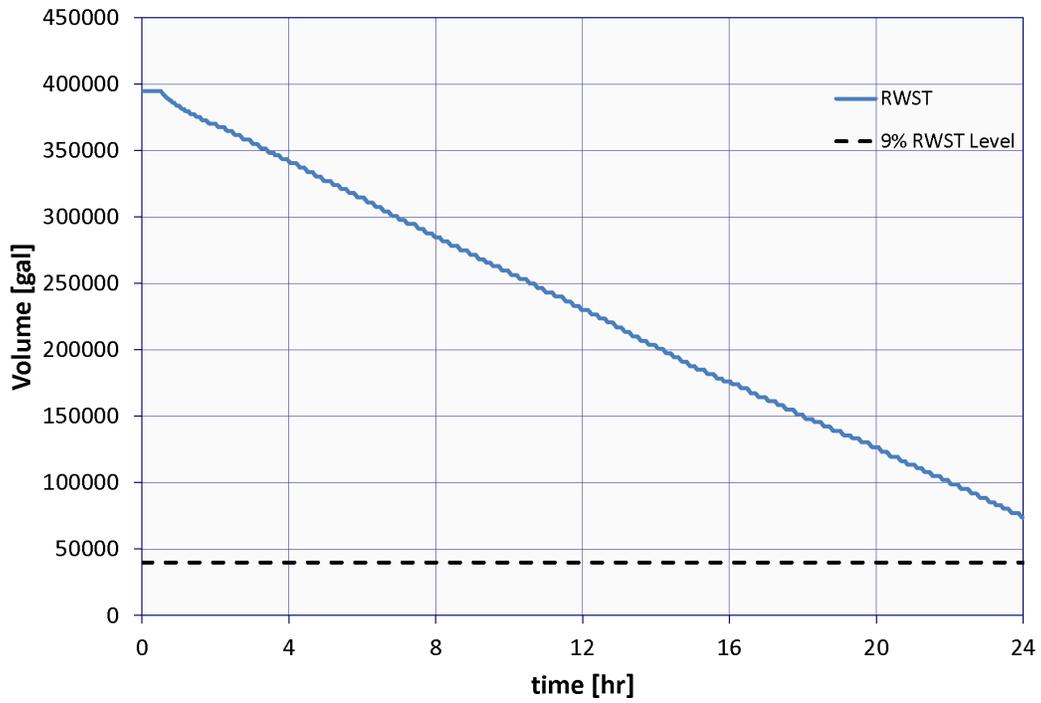
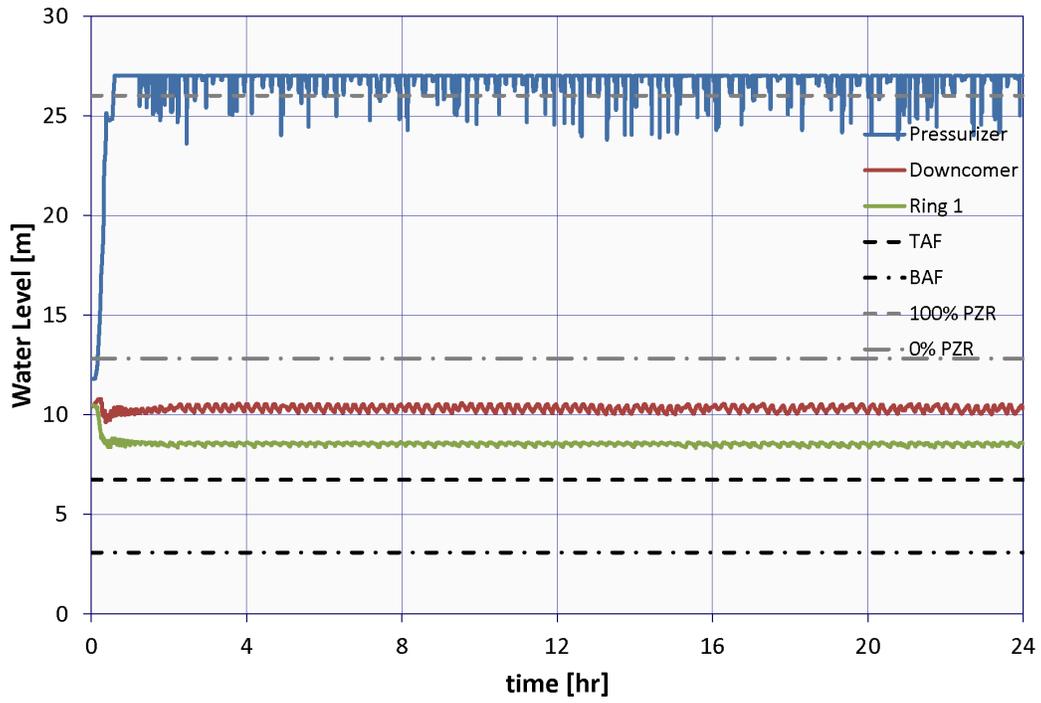




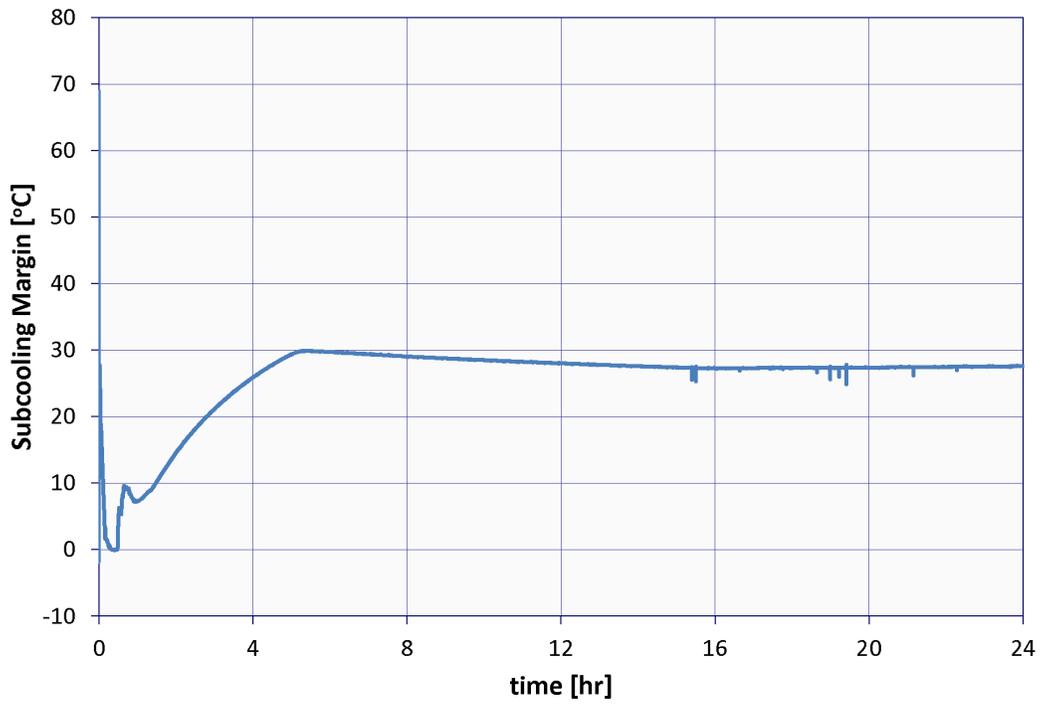
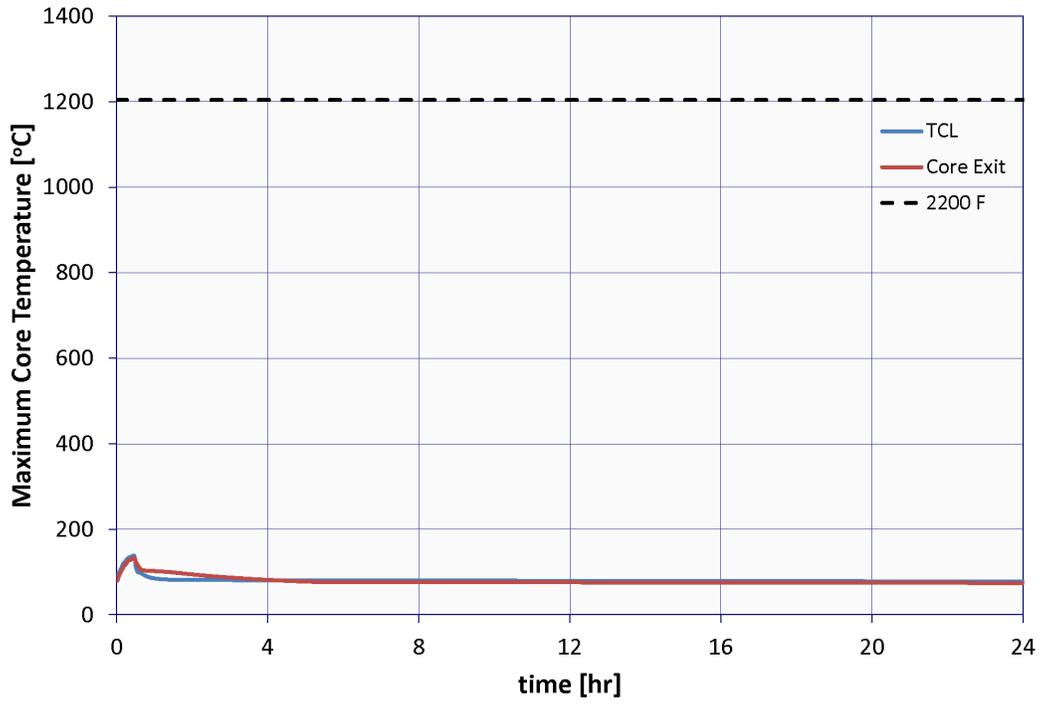
**F.2.8 Case 8: 60 hr after Shutdown, Start CCP on Low RPV Level**

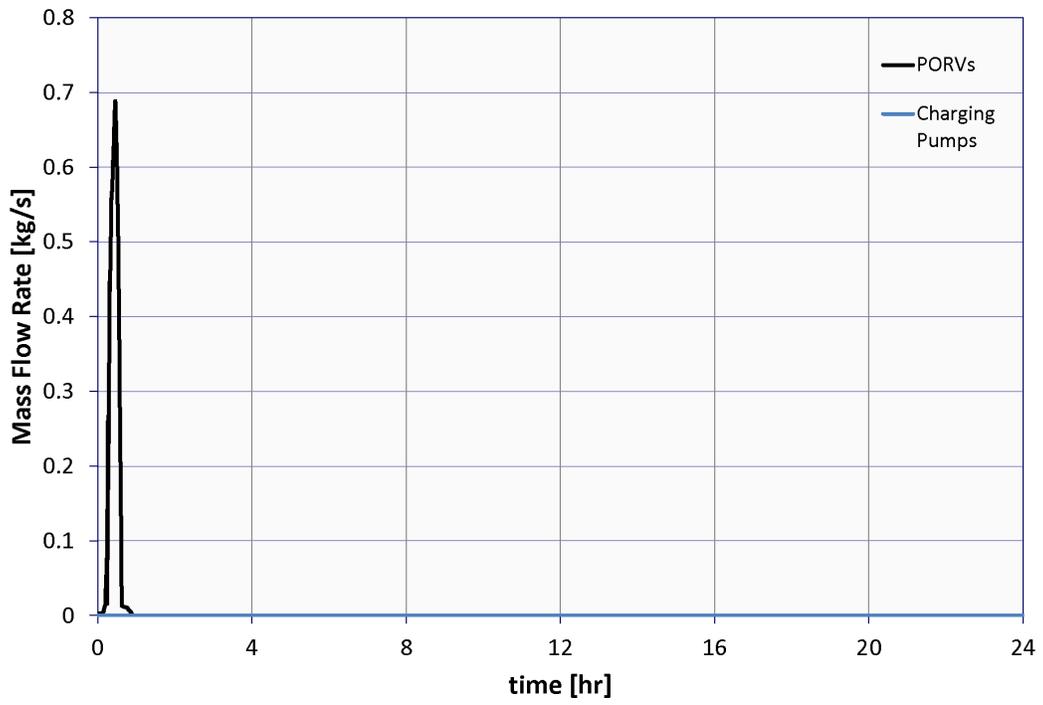
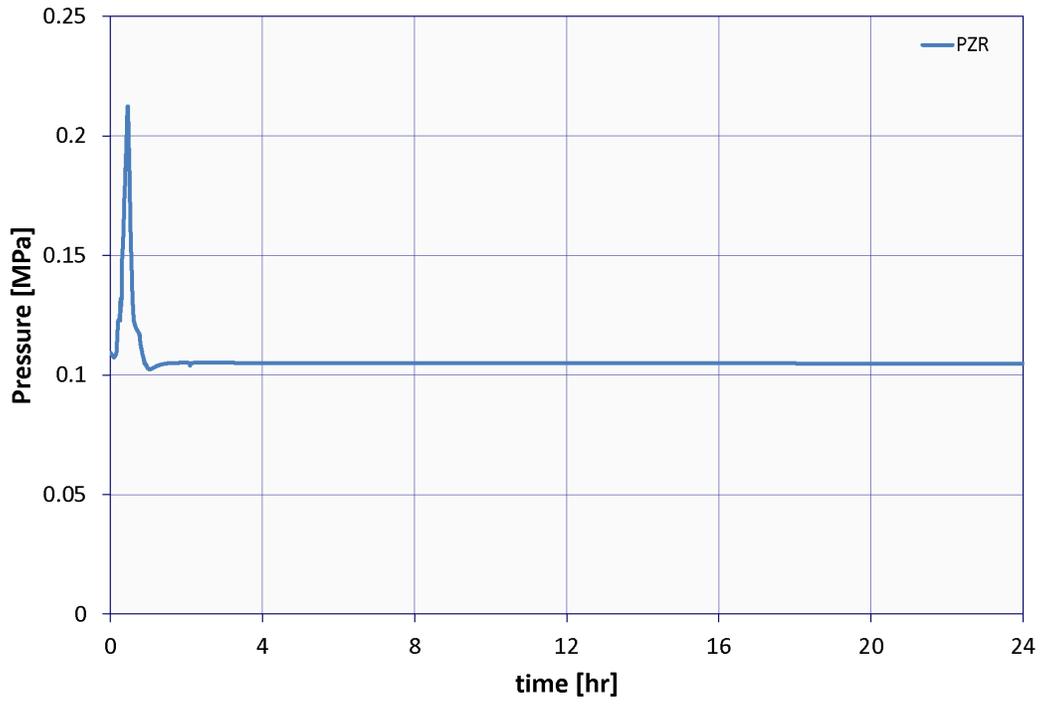


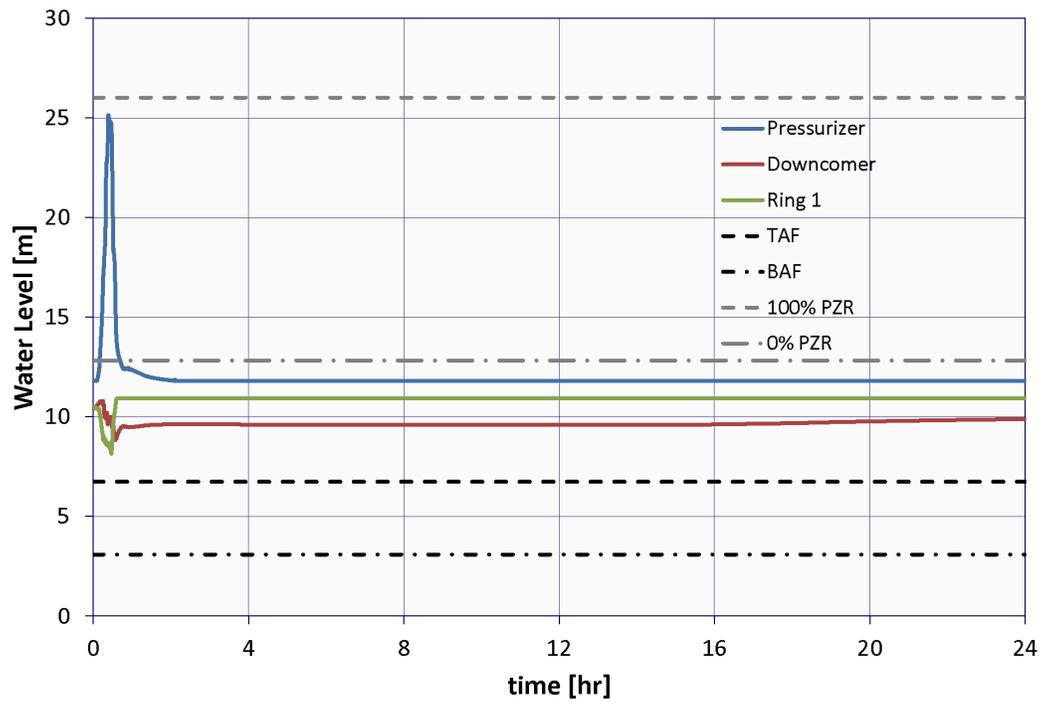




**F.2.9 Case 9: 60 hr after Shutdown, Recover RHR at 27 Minutes**









## **APPENDIX G**

### **EVENT TREE MODELS FOR STUDIED INITIATORS**



## **G.1 Byron SPAR Model Event Trees**

This section provides the relevant event trees from the Byron (v8.27) Standardized Plant Analysis Risk model dated April 2014. These event trees show the sequences described in the main report.

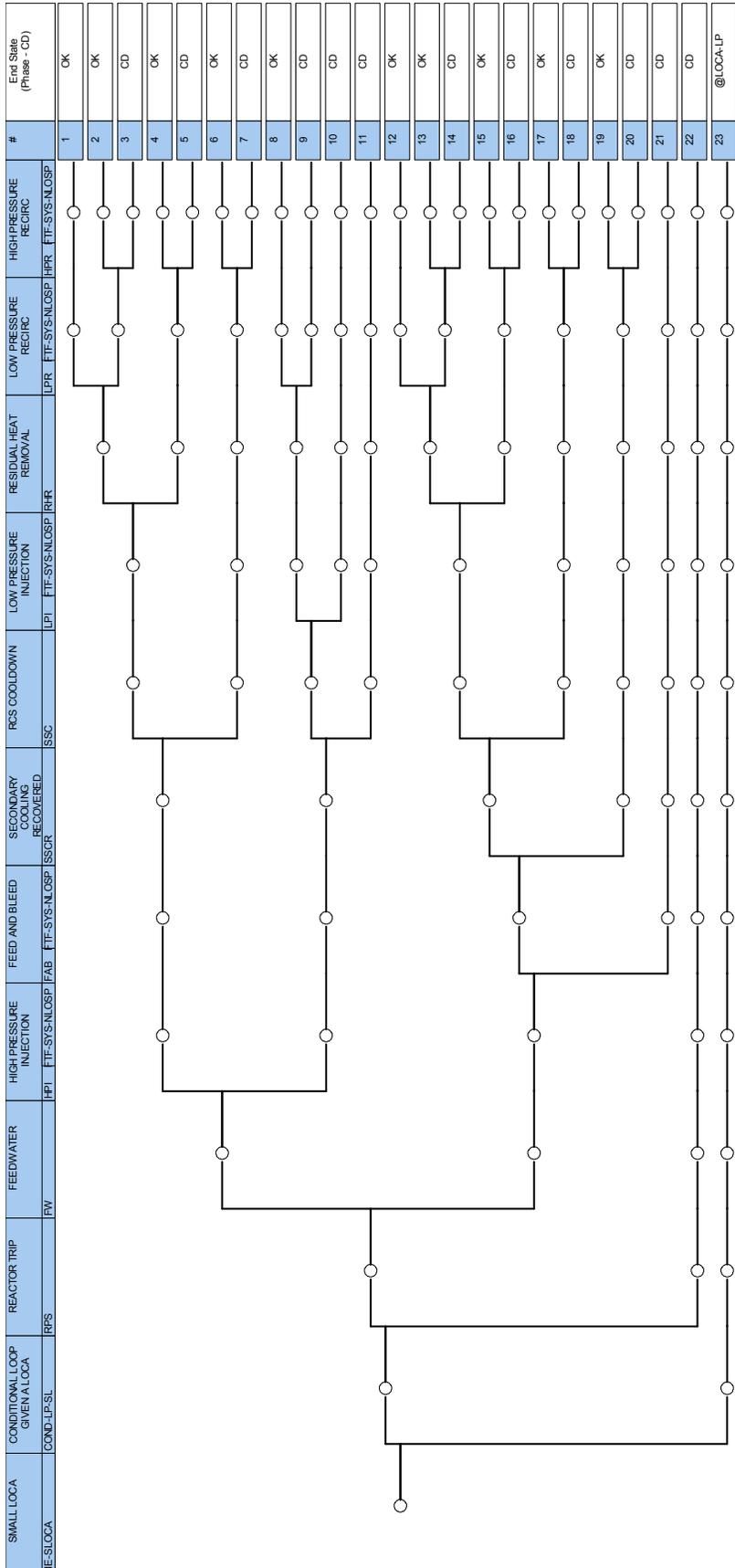


Figure G-1 Small-break loss-of-coolant accident (SLOCA) event tree

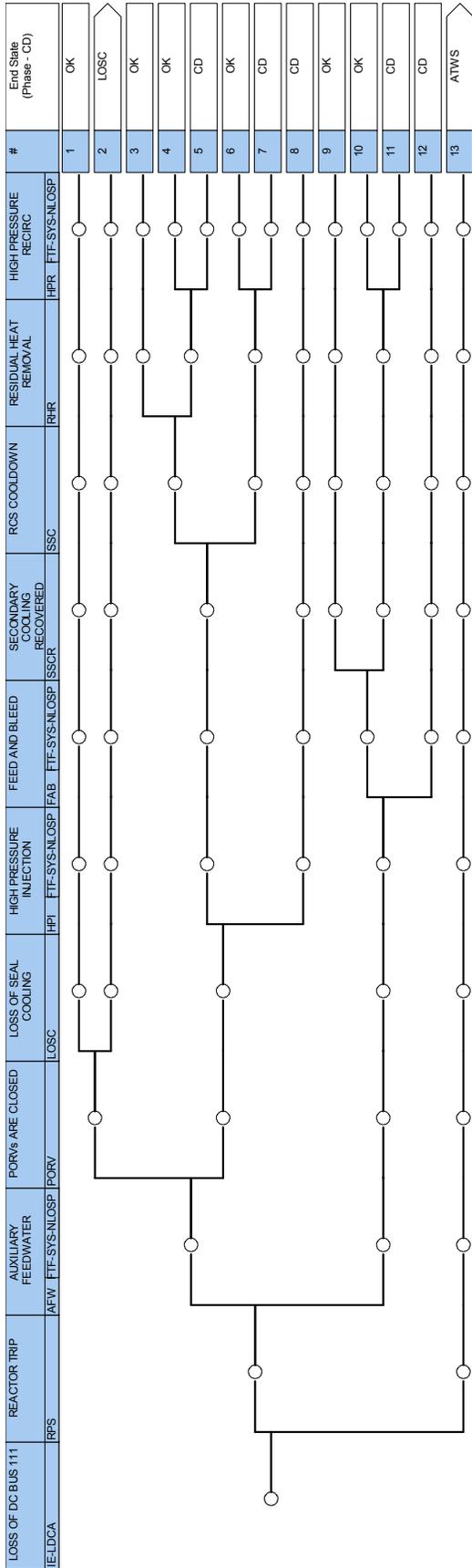


Figure G-2 Loss of 125V vital DC bus 111 event tree

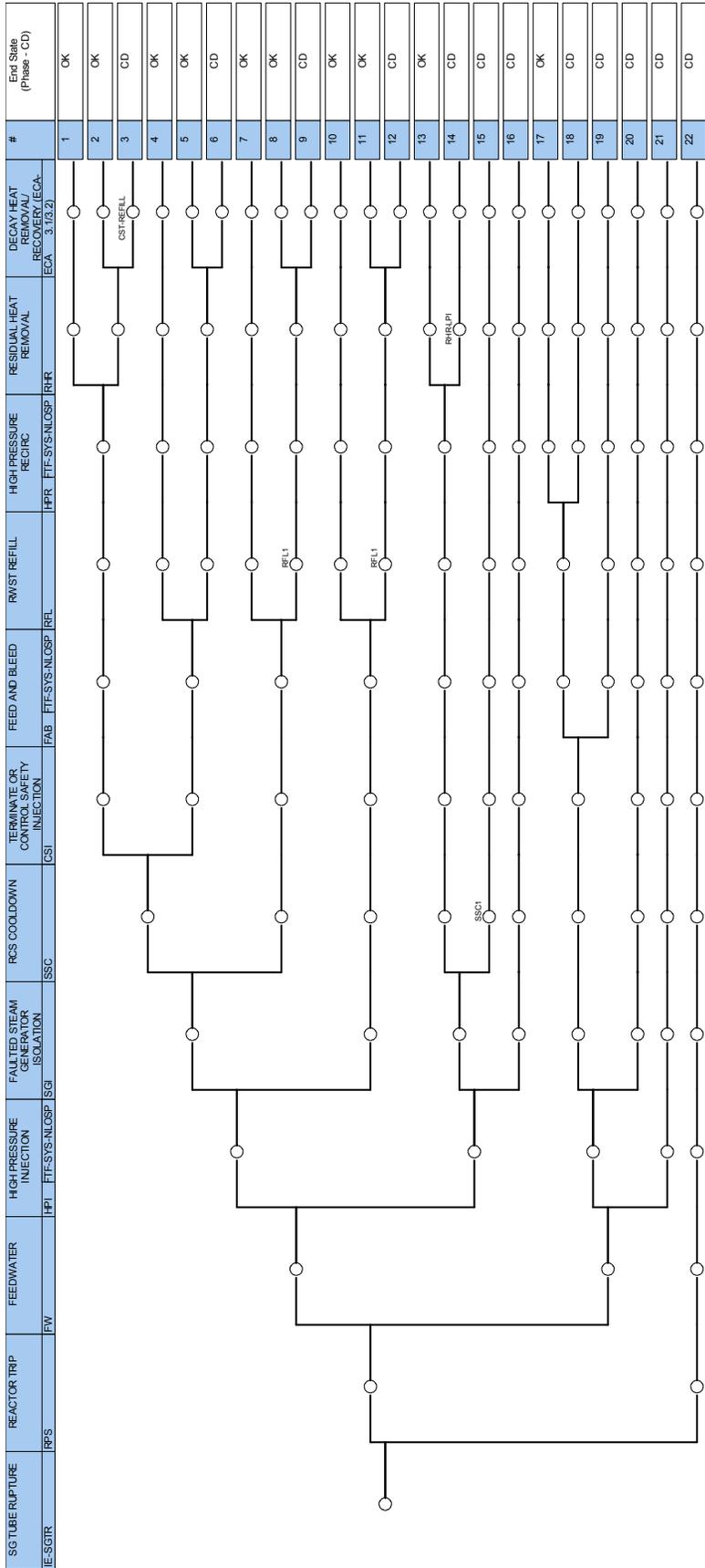


Figure G-3 Steam generator tube rupture (SGTR) event tree

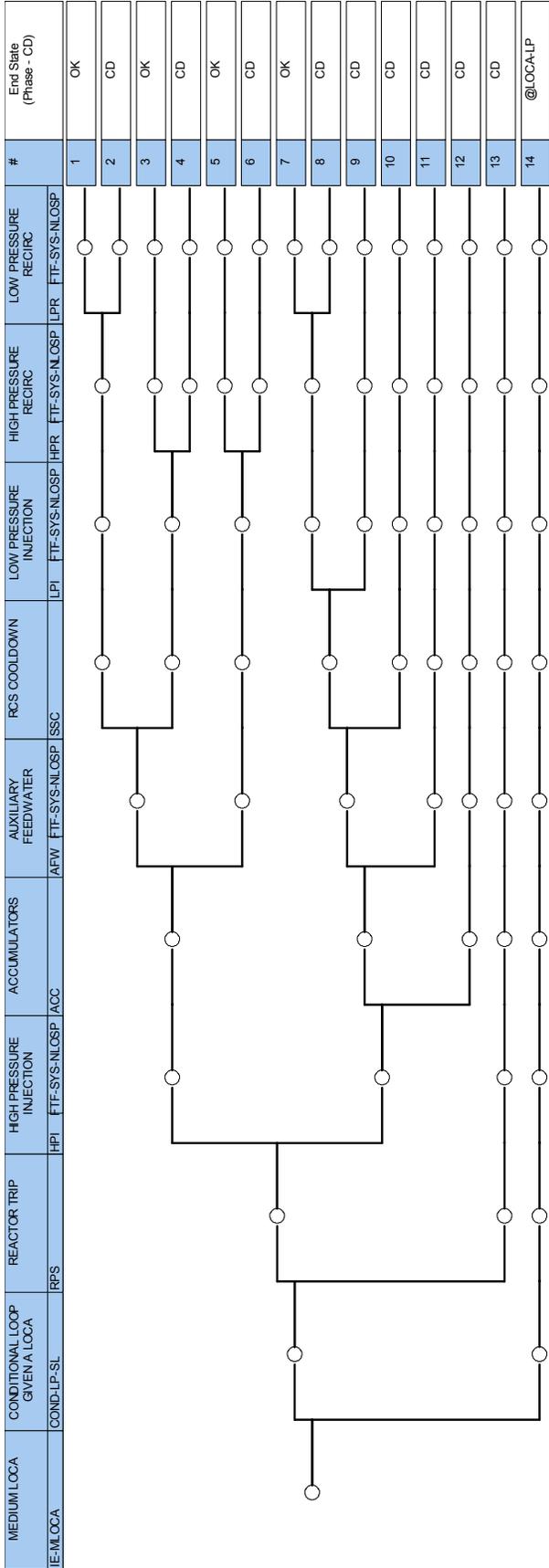


Figure G-4 Medium-break loss-of-coolant accident (MLOCA) event tree



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11. ABSTRACT (200 words or less)

This report extends the work documented in NUREG 1953, "Confirmatory Thermal Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom" to the Byron Station, Unit 1. Its purpose is to produce an additional set of best estimate thermal hydraulic calculations that can be used to confirm or enhance specific success criteria (SC) found in the agency's probabilistic risk assessment (PRA) tools. Along with promoting realism in the standardized plant analysis risk (SPAR) models, these calculations are expected to be a useful reference to model end users for specific regulatory applications (e.g., the Significance Determination Process). The U.S. Nuclear Regulatory Commission selected Unit 1 of the Byron Station for this study because it is generally representative of a group of four loop Westinghouse plants with large, dry containment designs.

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