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United States Nuclear Regulatory Commission
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NUREG-2236
Volume 3

Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models-Duane Arnold

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NUREG-2236
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Appendices F to H

Manuscript Completed: May 2019
Date Published: October 2020

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ABSTRACT

This report extends the work documented in NUREG-2187, “Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron Unit 1,” issued January 2016, to the Duane Arnold Energy Center. Its purpose is to produce an additional set of best estimate thermal-hydraulic calculations that can confirm or enhance specific success criteria for system performance and operator timing found in the agency’s probabilistic risk assessment 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.

This report first describes major assumptions used in this study. It then discusses the major plant characteristics for the Duane Arnold Energy Center, 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 success criteria, the licensee’s success criteria, or other generic studies.

The study results provide additional timing information for several probabilistic risk assessment sequences, confirm many of the existing SPAR modeling assumptions, and give a technical basis for a few specific SPAR modeling changes, including the following potential changes:

- Degraded high-pressure injection and relief valve Criteria (non-anticipated transient without scram): A single control rod drive pump injecting at the postscram increased injection rate is sufficient for reactor pressure vessel (RPV) water inventory makeup. Additionally, two control rod drive pumps injecting at the postscram injection rateprovide enough makeup to the RPV to facilitate a cooldown of the RPV to cold shutdown conditions. This increased injection is currently not queried in the SPAR models but could be added.

- Mitigating strategies usage: If diverse and flexible coping strategies (FLEX) are not available, success of long-term cooling for these scenarios is only possible with both anticipatory venting and condensate storage tank (CST) availability. Currently, CST availability is not queried in the SPAR models. This could be added for scenarios for which no alternate injection is available. For loss-of-offsite-power scenarios, FLEX injection led to success in all scenarios that gave FLEX credit. Given the ability of FLEX to prevent core damage, this confirms that the SPAR models should have FLEX equipment added.

- Emergency core cooling system injection following containment failure or venting: Depending upon the size of containment failure, wetwell and drywell pressure will fall, potentially to the point of allowing high-pressure injection restart following its loss. This action could be added to the SPARmodels.

- Safe and stable end-state considerations: If the CST is unavailable, the long-term availability of high-pressure injection is questionable at best. CST should be queried when high-pressure injection systems are the source of long-term makeup. Additionally, increased postscram control rod drive hydraulic system injection is adequate for makeup. This increased injection is a candidate for inclusion in the SPAR model. Depressurizing when reaching the heat capacity limit curve is important, since the rate of

seal leakage, as well as the rate of injection, is pressure dependent. This depressurization is a candidate for consideration in the SPAR models.

FOREWORD

The U.S. Nuclear Regulatory Commission (NRC) uses its standardized plant analysis risk (SPAR) models to support many risk-informed initiatives. A number of processes ensure the fidelity and realism of these models, including cross-comparison with industry models, review and use by a wide range of technical experts, and confirmatory analysis. This report—prepared by the staff of the Office of Nuclear Regulatory Research, in consultation with the staff of the Office of Nuclear Reactor Regulation; experts from Energy Research, Inc. 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 and sequence timing assumptions. These criteria and assumptions determine what combination of system and componentavailabilities will lead to postulated core damage, 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 success criteria 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 success criteria in the SPAR model for the Duane Arnold Energy Center. Based on further evaluation, these results could apply to similar plants, while future analyses could apply to other design classes, as occurred in the past (see NUREG-2187, "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron Unit 1," issued January 2016). The staff expects to continue its focus on confirming success criteria and other aspects of PRA modeling using its state-of-the-art tools (e.g., the MELCOR computer code) as it develops and improves its risk tools.

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ACKNOWLEDGMENTS

This report is the result of the efforts of many members of the U.S. Nuclear Regulatory Commission staff and the staff of its contractors, Energy Research, Inc., and Idaho National Laboratory. Furthermore, the report benefited greatly from the contributions of senior risk analysts and other selected agency staff members in specific areas of the project. The authors would especially like to recognize the support from Mahesh Chawla, James Corson, Ron Harrington, Tarek Zaki, and Joseph Staudenmeier. The authors also appreciate the assistance provided by the licensee's staff at the Duane Arnold Energy Center.

ABBREVIATIONS AND ACRONYMS

ac	alternating current
ADAMS	Agencywide Documents Access andManagement System
ADS	automatic depressurization system
AIP	alternate injection procedure (EOP)
ANS	American Nuclear Society
AOP	abnormal operating procedure
ASP	accident sequence precursor
ATWS	anticipated transient without scram
BWR	boiling-water reactor
C	Celsius
CD	core damage
CDF	core damage frequency
CDS	condensate system
CFR	*Code of Federal Regulations*
CRDHS	control rod drive hydraulic system
CST	condensate storage tank
DAEC	Duane Arnold Energy Center
dc	direct current
ECCS	emergency core cooling system
ED	emergency depressurization
EDG	emergency diesel generator
ELAP	extended loss of ac power
EOP	emergency operating procedure
ESF	engineered safety feature
ESFAS	engineered safety featuresactuation system
F	Fahrenheit
FLEX	diverse and flexible copingstrategies
FSG	FLEX support guideline
HCL	heat capacity limit
HCV	hardened containment vent
HCVS	hardened containment vent system
HPCI	high-pressure coolant injection
HPI	high-pressure injection
IORV	inadvertent open relief valve
IPE	individual plant examination
ISG	interim staff guidance
LCO	limiting condition for operation
LOCA	loss-of-coolant accident
LOCHS	loss of condenser heat sink

LODCA	loss of vital dc bus A
LODCB	loss of vital dc bus B
LOIAS	loss of instrument air system
LOMFW	loss of main feedwater
LOOP	loss of offsite power
LOOPGR	loss of offsite power grid related
LOOPPPC	loss of offsite power plant centered
LOOPWR	loss of offsite power weather related
LORWS	loss of river water system
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LPI	low-pressure injection
MAAP	modular accident analysis program
MFW	main feedwater
MLOCA	medium loss-of-coolant accident
MSIV	main steam isolation valve
MSL	main steamline
NCV	noncited violation
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
PB	Peach Bottom
PCPL	primary containment pressure limit
PCS	power conversion system
PCT	peak clad temperature
PID	proportional-integral-derivative
PRA	probabilistic risk assessment
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RPV	reactor pressure vessel
RWCU	reactor water cleanup
SAMP	severe accident management procedure
SBO	station blackout
SC	success criterion/criteria
SDP	significance determination process
SEP	site emergency plan
SFP	spent fuel pool
SLC	standby liquid control
SLOCA	small loss-of-coolant accident

SNL	Sandia National Laboratories
SP	suppression pool
SPAR	standardized plant analysis risk
SRV	safety/relief valve
TAF	top of active fuel
TRANS	transient
UFSAR	updated final safety analysis report
UHS	ultimate heat sink
WW	wetwell

APPENDIX F
GRADUAL OVERPRESSURE FAILURE MODES AND LOCATIONS IN MARKI, MARKII, AND MARKIII CONTAINMENT SYSTEMS

GRADUAL OVERPRESSURE FAILURE MODES ANDLOCATIONS IN MARK I, MARK II, AND MARK III CONTAINMENT SYSTEMS

F.1 INTRODUCTION

This appendix is an extensive review of several documents, such as individual plant examination (IPE) reports, NUREGs, and final safety analysis reports (UFSARs) prepared for boiling-water reactors with Mark I, II, and III containment types, to determine how these types of containments will respond to gradual internal, long-term pressure.

The evaluation of these responses to such pressure includes determining the types of failures that will occur and the location of such failures, as well as the leakage rate associated with each failure type.

This review of existing studies, reports, and other documents excludes the effects of rapid pressurization and depressurization of the primary containment and assumes core melt has not occurred.

F.2 REFERENCES

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F.3 <u>DISCUSSION</u>

F.3.1 Mark I Containments

The Mark I containments can be divided into two groups, as summarized in Table 3 of NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories—An Overview," issued July 2006 (Reference 1); namely, a freestanding steel primary containment or a reinforced concrete primary containment with a steel liner. As described in References 2 through 7, both of these Mark I containments are pressure-suppression containment systems, consisting of a primary and a secondary containment. These two containment structures house the reactor pressure vessel, the reactor recirculation loops, and other branch connections of the reactor coolant system; a pressure suppression chamber; a vent system connecting the drywell and the pressure suppression chamber; isolation valves; containment cooling systems; and other service equipment.

The primary containment structure is a low-leakage, pressure-suppression containment system; in the event of a primary system pipe failure within the drywell, a mixture of drywell atmosphere and steam would be forced through the vents into the suppression pool, resulting in steam condensation and pressure reduction. The two major structural components of the primary containment system are the drywell and the suppression chamber (wetwell). Both the drywell and pressure suppression chamber are designed for an internal pressure range of 0.39–0.43 MPa (56–62 pounds per square inch, gauge (psig)), coincident with a temperature of 138 degrees Celsius (C) (281 degrees Fahrenheit (F)).

The secondary Mark I containment has several configurations. In some Mark I containments, the secondary containment building is used to house support systems, whereas in others, the secondary containment system consists of four subsystems: the reactor building, the reactor building isolation and control system, the standby gas treatment system, and the off-gas stack. In all cases, the secondary containment building is designed to provide secondary containment when the primary containment is operable and when the primary containment is open. The secondary containment system is designed to be sufficiently leak-tight to allow the standby gas treatment system to maintain the reactor building at a subatmospheric pressure of 0.64 centimeters (0.25 inches) of water when the standby gas treatment system is exhausting reactor building atmosphere.

F.3.1.1 Primary Containment Ultimate Capability

The following discusses containment ultimate capabilities under sustained overpressure and various temperature levels. The work presented in this section is based on the analyses performed by Chicago Bridge and Iron (CB&I) and detailed in Reference 2 for Duane Arnold and in Reference 3 for Peach Bottom, a similar Mark I containment. References 4 through 7 represent other reviewed documents.

Before the publication of the IPE reports for Duane Arnold and Peach Bottom, Mark I containment performance was evaluated for high-pressure and low-temperature conditions. References 2 and 3 reviewed a significant body of studies, including the work done by CB&I, and grouped their findings in the following three categories related to internal pressure:

(1) containment capability at low temperature (below 260 degrees C [500 degrees F])

(2) containment capability at intermediate temperature (between 260 degrees C [500 degrees F] and 427 degrees C [800 degrees F])

(3) containment capability at high temperatures (above 482 degrees C [900 degrees F])

F.3.1.2 Containment Failure Sizes

The referenced IPE reports define primary containment failure sizes as belonging to three ranges:

(1) negligible failure (intact)—a failure resulting in a small leak rate that equates to an equivalent cross-sectional area leak of slightly less than 3.14 square inches (in^2)

(2) small failure—a failure causing a leak emanating from an opening that is estimated to have a cross-sectional area between 3.14 in^2 and 1.0 square feet (ft^2)

(3) large failure—a failure creating a hole that is greater than 1.0 ft^2 in size

F.3.1.3 Containment Capability at Low Temperature (below 260 degrees C [500 degrees F])

References 2 and 3 both show that extensive review of Mark I containment evaluations concludes that, for temperatures below 171 degrees C (340 degrees F), a reasonable assessment of the mean containment pressure capability is 0.97 MPa (140 psig). The following are some of the general conclusions about low-temperature (i.e., less than 171 degrees C [340 degrees F]) containment performance that could be drawn from the many studies reviewed:

- The ultimate pressure capability of the Mark I containment is two to three times greater than the design pressure.

- The most probable containment failure modes as a result of pressurization (as pressure exceeds 0.97 MPa (140 psig) but temperature is still less than 171 degrees C [340 degrees F]) are (1) a rupture (ductile tear) in the wetwell airspace, (2) a rupture (ductile tear) in the wetwell water space, or (3) a break in the drywell head.

- These failure sizes are considered negligible (containment integrity is maintained), defined in References 2 and 8 as an opening with a diameter less than 5.1 centimeters (2 inches) (or an area of 3.14 in^2).

Review of the IPE reports listed in References 2 through 7 leads to a general conclusion that, for temperatures below 171 degrees C (340 degrees F), a reasonable assessment of the mean Mark I containment pressure capability is 0.97 MPa (140 psig). As stated above, as the pressure exceeds this limit, material yielding in the wetwell airspace begins, resulting in tearing in the wetwell shell (wetwell airspace). Leakage through the drywell closure head could begin about the same time or a bit later than the time that the tearing in the wetwell shell starts. Leakage through the vent line bellows could also happen as the pressure increased beyond the 0.97-MPa (140-psig) limit. As these failure sizes increase, depressurization of the containment will begin.

F.3.1.4 Containment Capability at Intermediate Temperature (between 260 degrees C [500 degrees F] and 427 degrees C [800 degrees F])

In this intermediate temperature range, the reduction in material strength and seal properties degrades the containment pressure capability. Seals are used around containment penetrations and around the drywell closure head. Based on the review of several studies, analysts concluded that primary containment failure in the 260-degree-C (500-degree F) to 427-degree-C (800-degree-F) range would occur at a pressure range of *0.43 to 0.61 MPa (63 to 88 psig)*, dominated by the drywell head seal failure and opening of the drywell flange. This failure is described in Reference 2, as a "*leakage dominated failure mode,*" which means that it is a small failure of approximately *18 in^2 to 19.5 in^2*.

The probabilities of other failure modes at these drywell temperatures and pressures were substantially less. For example, other containment penetrations were also examined as part of the review of the Pilgrim study detailed in Reference 5. Penetrations located in the torus (wetwell) were excluded since the temperature there was expected to be significantly lower than that in the drywell during severe accident conditions. Other penetrations, such as personnel airlock, equipment hatch, and electrical penetration, were shown to have very durable seals that could withstand temperatures of at least 371 degrees C (700 degrees F) and would not deteriorate before significant degradation of the drywell closure head seal.

F.3.1.5 Containment Capability at High Temperature (above 482 degrees C [900 degrees F])

Based on computer analyses for accidents in which core melt has occurred, there is very little confidence that the containment can withstand such high temperatures without significant material degradation. Based on the results of the studies performed by the Industry Degraded Core Rulemaking Program (as described in Section 4.4 of References 2 and 3), and by CB&I for Duane Arnold, the general conclusion is that containment strength becomes suspect at temperatures that *exceed 482 degrees C (900 degrees F), and the drywell would fail under any appreciable pressure load at this temperature*.

F.3.1.6 Summary and Conclusions

A detailed review of the IPE reports in References 2 through 7 leads to the following three conclusions about the Mark I containment failure characterization:

(1) for drywell temperatures below 260 degrees C (500 degrees F):

- The mean ultimate pressure capability of the Mark I containment is about 0.97 MPa (140 psig), and it is two to three times greater than the design pressure of 0.39 MPa (56 psig).

- The most probable containment failure modes caused by pressurization (as pressure exceeds the design pressure but temperature is still less than 171 degrees C [340 degrees F]) are (1) a rupture (ductile tear) in the wetwell airspace, (2) a rupture (ductile tear) in the wetwell water space, or (3) a break in the drywell closure head. The leakage rate is considered negligible (an opening with a cross-sectional area less than 3.14 in^2).

(2) for drywell temperatures between 260 degrees C (500 degrees F) and 427 degrees C (800 degrees F):

- Primary containment failure in the drywell temperature range of 260 degrees C (500 degrees F) to 427 degrees C (800 degrees F) was estimated to occur at a pressure range of 0.43 to 0.61 MPa (63 to 88 psig), dominated by the drywell head seal failure and opening of the drywell flange. This failure is described as a "leakage dominated failure mode," which means that it is a small failure of approximately 18 in^2 to 19.5 in^2.

(3) at high drywell temperature (e.g., above 482 degrees C [900 degrees F]):

- Computer analyses for accidents in which core melt has occurred demonstrate that there is very little confidence that the containment can withstand such high temperatures without significant material degradation. The general conclusions are that the strength of the containment becomes suspect at temperatures that exceed 482 degrees C (900 degrees F), and the primary containment would fail under any appreciable pressure load at this temperature.

F.3.2 Mark II Containments

The Mark II primary containment is also a pressure-suppression containment system, as described in References 8 through 11, which houses the reactor pressure vessel, the reactor recirculation loops, and other branch connections of the reactor coolant system. The reactor enclosure exterior walls, roof, floor, and penetrations form the secondary containment. The primary containment of the Columbia plant (Reference 9) consists of a freestanding steel structure housing a steel drywell and wet well, whereas the primary containments of the Limerick, Susquehanna, and Nine Mile Point Unit 2 plants (References 8, 10, and 11) are made from reinforced concrete with steel liner plate and both the drywell and wetwell are reinforced concrete structures. Like the Mark I containment, the two major structural components of the Mark II primary containment system are the drywell and the suppression chamber (wetwell). Both the drywell and pressure suppression chamber are designed for an internal pressure range of 0.31 to 0.38 MPa (45 to 55 psig), coincident with a temperature of 171 degrees C (340 degrees F) for the drywell and a range of 104 degrees C (220 degrees F) to 132 degrees C (270 degrees F) for the suppression chamber.

The Mark II secondary containment has several configurations. Although some Mark II secondary containments are very compartmentalized, in general, the secondary containment includes both the reactor enclosure and the refueling area. Blowout panels in the side of the reactor enclosure are designed to relieve excessive pressure in the building.

F.3.2.1 Primary Containment UltimateCapability

Based on the information in References 8 through 11, the following discusses the three categories of the effect of temperature and pressure on the ultimate capabilities of the three categories of Mark II primary containments:

(1) containment capability at low temperature (below 204 degrees C [400 degrees F])

(2) containment capability at intermediate or moderate temperature(between 204 degrees C [400 degrees F] and 482 degrees C [900 degrees F])

(3) containment capability at high temperatures (above 482 degrees C [900 degreesF])

As stated in NUREG/CR-6906 (Reference 1), failure size is one of the critical factors in the characterization of containment performance. The referenced IPE studies identified three possible failure sizes:

(1) a leak—a containment breach (typically a cross-sectional area of 0.1 ft^2) that would arrest a gradual pressure buildup but would not result in containment depressurizationin less than 2 hours

(2) a rupture—a containment breach (corresponding to a hole size in excess of approximately 1.0 ft^2) that would arrest a gradual pressure buildup and would result in containment depressurization within 2 hours

(3) a catastrophic rupture—a containment breach that results in the loss of a substantial portion of the containment boundary, including major penetrations attached to the containment wall

F.3.2.2 Containment Capability at Low Temperature (below 204 degrees C [400 degrees F])

An extensive review of the IPEs (References 8 through 11) showed that the structural behavior of the typical Mark II containments under low temperatures tends to be consistent. For temperatures below 204 degrees C (400 degrees F), a reasonable assessment of the structural capability of the containment at low temperature exceeded pressures of 0.83 MPa (120 psig) and probably 0.97 MPa (140 psig).

The most probable containment failure modes caused by pressurization (as pressure exceeds 0.97 MPa (140 psig) but temperature is still less than or equal to 171 degrees C [340 degrees F]) are (1) a rupture (ductile tear) in the wetwell airspace, (2) a rupture (ductile tear) in the equipment hatch, or (3) a break in the drywell head. These failures are defined as “leak-before-break” failures, where failure will first occur as tearing of limited size that progressively increases to gross failure as pressure increases towards the 171-degree C (340-degree F) level. The dominant failure modes at these temperatures (about 171 degrees C [340 degrees F]) and pressure (about 0.97 MPa [140 psig]) were leakage caused by rupture in the wetwell below the waterline and rupture in the drywell head seal.

F.3.2.3 Containment Capability at Intermediate or Moderate Temperature (between 204 degrees C [400 degrees F] and 482 degrees C [900 degrees F])

In this intermediate temperature range, the containment pressure capability is degraded by the reduction in material strength and properties of the seal used around containment penetrations and around the drywell closure head. Based on the review of several studies, analysts concluded that primary containment failure in the drywell temperature range of 204 degrees C (400 degrees F) to 482 degrees C (900 degrees F) was estimated to occur at a pressure range of *0.41 to 1.26 MPa (60 to 138 psig)*, dominated by rupture in the wetwell below the water line and the drywell head seal failure and opening of the drywell flange. This failure of the drywell closure head seal is described in Reference 8 as a "*leakage dominated failure mode,*" which is equivalent to an opening of *14 in^2 to 87 in^2*.

F.3.2.4 Containment Capability at High Temperature (above 482 degrees C [90 degrees F])

Computer analyses have calculated drywell temperatures above 482 degrees C (900 degrees F) for accidents in which core melt has occurred, the core has slumped into the drywell floor, and core debris cooling is unavailable. At such temperatures, other studies have postulated that the material (steel and concrete) properties may have changed sufficiently to cause containment structural degradation such that the integrity of the primary containment could not be maintained under any internal pressure. The general conclusion is that the strength of the containment becomes suspect at temperatures that *exceed 482 degrees* C (*900 degrees F), and the primary containment will experience catastrophic rupture under any appreciable pressure load at this temperature*.

F.3.2.5 Summary and Conclusions

A detailed review of the IPE reports, listed in References 8 through 11, leads to the following three conclusions about the Mark II containment failure characterization:

(1) for temperatures below 204 degrees C (400 degrees F):

- A review of the IPEs presented in References 8 through 11 concluded that the structural behavior of the typical Mark II containments under low temperatures tends to be consistent, and a reasonable assessment of the structural capability of the containment at low temperature exceeded pressures of 0.83 MPa (120 psig) and probably 0.97 MPa (140 psig).

- The dominant failure modes at these temperatures (about 171 degrees C [340 degrees F]) and pressure (about 0.97 MPa [140 psig]) were leakage caused by rupture in the wetwell below the waterline and rupture in the drywell head seal, with the leakage area size as defined in the previous section.

(2) for temperatures between 204 degrees C (400 degrees F) and 482 degrees C (900 degrees F):

- In this intermediate temperature range, the primary containment pressure capability is degraded by the reduction in material strength and properties of the seal used around containment penetrations and around the drywell closure head. Based on the review of several studies, analysts concluded that primary containment failure in the drywell temperature range of 204 degrees C (400 degrees F) to 482 degrees C (900 degrees F) would occur at a pressure range of *0.41 to 0.95 MPa (60 to 138 psig)*, dominated by rupture in the wetwell below the water line and the drywell head seal failure and opening of the drywell flange, equivalent to an opening of *14 in^2 to 87 in^2*.

(3) at temperatures above 482 degrees C (900 degrees F):

- Studies have suggested that, at drywell temperatures above 482 degrees C (900 degrees F), the material (steel and concrete) properties may have changed sufficiently to cause containment structural degradation such that the integrity of the primary containment could not be maintained under any internal pressure, *and the primary containment will experience catastrophic rupture under any appreciable pressure load at this temperature*.

F.3.3 Mark III Containments

The IPEs listed in References 12 through 15 describe two types of Mark III primary containment. Both are pressure-suppression containment systems, which house the reactor pressure vessel, the drywell, and the suppression pool (wetwell).

The primary containment of both Clinton (Reference 12) and Grand Gulf (Reference 13) consists of a right circular cylinder with a hemispherical domed roof and a flat base slab. The primary containment wall is constructed from reinforced concrete, completely lined with steel plate. The drywell is also a right circular cylinder located within and concentric to the steel-lined, reinforced-concrete containment. The drywell is designed for an internal pressure of 0.21 MPa (30 psig) coincident with an internal temperature of 166 degrees C (330 degrees F). The pressure suppression pool is located between the drywell weir wall and the containment wall. Like the containment wall, it is designed for an internal pressure of 0.10 MPa (15 psig), coincident with a temperature of 85 degrees C (185 degrees F) for the drywell and a temperature range of 104 degrees C (220 degrees F) to 132 degrees C (270 degrees F) for the suppression chamber.

The primary containment of both Perry (Reference 14) and River Bend (Reference 15) is a freestanding cylindrical steel containment vessel that encloses the drywell and the suppression pool (wetwell). The freestanding steel containment for both plants is designed for an internal pressure of 0.10 MPa (15 psig, coincident with an accident internal temperature of 85 degrees C (185 degrees F). The drywell of both of these plants is designed for an internal pressure range of 0.17 to 0.20 MPa (25 psig to 30 psig) with a coincident temperature of 166 degrees C (330 degrees F). Part of their pressure suppression pools is located inside the drywell, and the major parts are located outside the drywell between the outer drywell wall and the inner steel containment wall.

F.3.3.1 Primary Containment Ultimate Capability

References 12 through 15 did not consistently account for the effects of temperature and pressure on the ultimate capabilities of the Mark III containment, as was done in References 2 through 11 for the Mark I and II primary containments.

The Perry IPE (Reference 14) did not explicitly consider the effects of temperature in determining the ultimate capacities of the primary containment boundary nor the drywell boundary. It noted the following, however:

- Most applicable design codes required the design accident concrete surface temperatures to be below 177 degrees C (350 degrees F), after which the concrete starts to lose some of its strength. Section 8.0 of the Perry IPE (Reference 14) discusses the effects of high temperature on concrete, with Table 17, presented in the samereport, showing the impact on the compressive strength of concrete for temperatures ranging from 93 degrees C (200 degrees F) to 871 degrees C (1,600 degrees F) over a 48-hour exposure.

- The strength of steel decreases linearly from its value at 21 degrees C (70 degrees F)to about 80 percent of this value at 427 degrees C (800 degrees F). In general, the strength of the reinforcing steel, part of the reinforced concrete, is more significant and thus the capacity of reinforced concrete will be impacted by the effects of high temperatures on steel.

- The seals around the major penetrations in the containment and drywell, such as the personnel airlock, equipment hatches, and drywell head, are fabricated from ethylene propylene rubber, with a failure temperature of about 316 degrees C (600 degrees F), but an inflated ethylene propylene seal has been known to fail atapproximately 241 degrees C (465 degrees F).

However, the River Bend IPE (Reference 15) evaluated the median pressure capacities for three temperature conditions: 21 degrees C (70 degrees F), 149 degrees C (300 degrees F), and 427 degrees C (800 degrees F).

Containment Failure Definition and Sizes

The failure characterization for the two types of Mark III containment is described (as expected) differently for the two cases.

<u>Mark III Containments with a Reinforced Concrete Primary Containment with Steel Liner</u>

Unlike Mark II containments, the failure sizes associated with overpressurization of the Mark III containment with a reinforced concrete primary containment with steel liner were not well defined in IPE reports for Clinton (Reference 12) and Grand Gulf (Reference 13). To attribute a generalized size to the containment breach, the IPEs assumed failures of the containment shell or equipment hatch were gross failures or rupture (i.e., failures that were large enough to cause rapid depressurization of the containment). On the other hand, failure of the containment liner was considered a leakage—failure that would prevent further pressurization of the containment but allow for slow and gradual depressurization.

<u>Mark III Containments with a Freestanding Steel Primary Containment</u>

For the Mark III containments with a freestanding steel primary containment, the failure pressures are defined as the pressure associated with mean yield stress in the structural components (even though higher pressures are possible, based on the ultimate strength of the materials). The Perry IPE (Reference 14) defined the failure sizes as belonging to three ranges:

(1) leakage—a containment breach, typically with a cross-sectional area of 0.1 ft^2, that would result in containmentdepressurization

(2) rupture—a containment breach corresponding to a hole with an area from 0.1 ft^2 to 7.0 ft^2 that would result in containment depressurization

(3) gross rupture—a containment breach corresponding to a hole with an area greater than 7.0 ft^2

River Bend (Reference 15) also has a freestanding steel primary containment and, asdescribed in its IPE report, failure modes associated with gross structural failure would result in rapid depressurization of the primary containment, and thus no leakage areas are estimated for these types of failures. For nonstructural failure modes, leakage associated with these modes could be described as follows:

- insignificant leakage area
- leakage area less than or equal to 12 in^2
- leakage area between 12 in^2 and 32 in^2

Containment Capability at Various Temperatures

<u>Mark III Containment with a Reinforced Concrete Primary Containment with Steel Liner</u>

Clinton and Grand Gulf are the two power stations that have the Mark III containment with a reinforced concrete primary containment with steel liner.

As stated in Section 4.4.9 of the Clinton IPE (Reference 12), the primary containment would have a 50-percent probability of failure at 0.647 MPa (93.8 psig), with the likely failure mode being tearing of the steel liner in the vicinity of one of the major containment penetrations (such as the personnel airlocks or the equipment hatches). The personnel airlocks and equipment hatches have higher capacities that the steel liner surrounding them. As the pressure increases, failure of the containment shell (hoop rebar) will occur around midheight, at significantly higher pressure. Finally, approximately 14 percent of the failures in the liner will occur below the Water level in the suppression chamber and the rest will occur above the Water level.

The Grand Gulf IPE report (Reference 13) provides, in Tables 4.4.1-1 and 4.4.1-2 (reproduced herein), ultimate static pressure capacities for both the primary containment and the drywell boundaries, for various failure mechanisms. These capacities are based on specified material strengths and do not account for thermal loads. The failure capacities given in Table 4.4.1 of Reference 13 for the various locations around the major containment penetrations are significantly less than the 0.647 MPa (93.8 psig) capacity (at 50 percent probability of failure) stated earlier for Clinton, a similar Mark III primary containment. *The present data on the ultimate pressure capacity of a Mark III reinforced concrete primary containment with steel liner are limited and as such, failure pressure ranges may have to be developed for various failure modes and incident temperatures.*

TABLE 4.4.1-1
(PRIMARY CONTAINMENT ULTIMATE CAPACITIES)

PRESSURE (PSI) [1]	DESCRIPTION OF FAILURE AND LOCATION
-15	Failure of the Containment Liner due to negative pressure.
56 [2]	Cylinder failure due to shear, at Upper Personnel Air Lock.
56	Cylinder failure due to hoop membrane forces, at Els. 136' --> 215'.
57	Base mat failure due to radial moment, at radius = 59'. (approx.)
60	Failure of the joint between the Cylinder and the Base Mat.
64	Cylinder failure at edges of Upper Personnel Air Lock penetrations.
64	Dome failure due to membrane forces, above 45 degrees from the Spring Line.
65	Failure of the Upper Personnel Air Lock.
72	Failure of Piping penetrations begins.
77	Failure of the Lower Personnel Air Lock.
79	Cylinder failure due to shear, at Base Mat.
81	Cylinder failure due to meridional moment, at Base Mat.
82	Cylinder failure due to meridional membrane forces, at El. 220. (approx.)
83	Cylinder failure due to moment, at Upper Personnel Air Lock.
108	Cylinder failure due to moment, at Equipment Hatch.
129	Cylinder failure due to meridional moment, at Spring Line.
135	Cylinder failure at edges of Equipment Hatch penetrations.
136	Dome failure due to meridional moment, at Spring Line.
206	Failure of the Containment Equipment Hatch.
218 [2]	Cylinder failure due to shear, at Equipment Hatch.
2136	Dome failure due to shear, at Spring Line.
2535	Cylinder failure due to shear, at Spring Line.

[1] Pressure capacity based on stated material strength Unless Noted Otherwise.

[2] Capacity independent of stated material strength of Reinforcement.

Source: Grand Gulf Nuclear Station—Individual Plant Examination Summary Report (Reference 13)

TABLE 4.4.1-2
(DRYWELL ULTIMATE CAPACITIES)

PRESSURE (PSI) [1]	DESCRIPTION OF FAILURE AND LOCATION (Due to Internal Pressure U.N.O.)
<600>	Failure of the Drywell Shell (External Pressure)
<89>	Failure of the Drywell Head (External Pressure)
<55>	Failure of Drywell roof. (External Pressure)
56	Drywell failure at edge of Personnel Air Lock
60	Failure of Drywell 300 and 400 series penetrations.
67	Failure of Drywell walls at approximate Elevation 122'
72	Failure of the Drywell Personnel Air Lock.
84	Failure of the joint between the Drywell and the Base Mat.
102	Failure of Drywell roof. (Internal Pressure)
159	Failure of Drywell Equipment Hatch.
169	Failure of the Drywell Head (Internal Pressure)

[1]Pressure capacity based on stated material strength Unless Noted Otherwise.

Source: Grand Gulf Nuclear Station—Individual Plant Examination Summary Report (Reference 13)

Mark III Containment with a Freestanding Steel Primary Containment

Perry and River Bend are the two power stations that have the Mark III containment with a freestanding steel primary containment. References 14 and 15 evaluated both the failure modes for the containment and the drywell for both plants, and the ultimate pressure capacities varied for both. The failure capacities evaluated for Perry make no mention of the temperature levels associated with these failure pressure capacities. Therefore, the assumption should be made that these pressure capacities are at room temperature (approximately 21 degrees C [70 degrees F]).

STEEL CONTAINMENT

Containment Vessel Wall

The study described in Reference 14 determined that the failure pressure capacity of the Perry containment cylinder wall would occur at a pressure level of 0.597 MPa (86.6 psig) with a 5-percent probability of failure (with no mention of the temperature level), whereas the study described in Reference 15 determined the failure of the River Bend containment at the same location would occur at 0.861 MPa (125 psig), 0.765 MPa (111 psig), and 0.58 MPa (84 psig) for 21 degrees C (70 degrees F), 149 degrees C (300 degrees F), and 427 degrees C (800 degrees F), respectively.

Containment Vessel Dome

Similarly, the analysts determined the failure of the containment dome would occur at a pressure level of 0.5983 MPa (86.77 psig) with a 5-percent probability of failure (with no mention of the temperature level), whereas the failure of the River Bend containment dome would occur at 0.827 MPa (120 psig), 0.738 MPa (107 psig), and 0.55 MPa (80 psig) for 21 degrees C (70 degrees F), 149 degrees C (300 degrees F), and 427 degrees C (800 degrees F), respectively.

The failure modes for these locations of the steel containment vessel (wall and dome) for both plants are considered structural failures, with no leakage area assigned to them, as defined earlier.

Equipment Hatch

The failure pressure capacity of the equipment hatch at Perry is 0.419 MPa (60.8 psig), equal to a 5-percent probability of failure. For the River Bend plant, the controlling failure mode was determined to be leakage through the seals. The median pressure at which leakage could be expected would occur at pressure levels 0.34 MPa (49 psig) and 0.27 MPa (39 psig) for 21 degrees C (70 degrees F), and 149 degrees C (300 degrees F), respectively. At temperatures higher than 149 degrees C (300 degrees F), leakage through the seals would begin at an internal pressure level of 0.23 MPa (33 psig).

Personnel Airlock

The Perry IPE determined the maximum pressure the airlock could resist would be 0.648 MPa (94.0 psig). For the River Bend plant, the controlling failure mode would be leakage through the inflatable seals. The median pressure at which leakage could be expected was 0.12 MPa (17 psig) and 0.19 MPa (28 psig) for 21 degrees C (70 degrees F) and 149 degrees C (300 degrees F), respectively. At 204 degrees C (400 degrees F), leakage pressure through the seals would be 0.23 MPa (34 psig). Estimates for leakage around the personnel doorswere 12 in^2 at 149 degrees C (300 degrees F) and the leakage area increasedtoapproximately 32 in^2 as the temperature rose higher than 316 degrees C (600 degrees F).

DRYWELL

Drywell Wall

As stated in Reference 15, the failure pressure capacity of the River Bend drywell was controlled by yielding of the hoop reinforcing steel at pressure levels 0.861 MPa (125 psig), 0.765 MPa (111 psig), and 0.58 MPa (84 psig), coincident with temperature levels 21 degrees C (70 degrees F), 149 degrees C (300 degrees F), and 427 degrees C (800 degrees F). The mean yield failure pressure for the drywell wall, based on hoop capacity, would be 0.679 MPa (98.5 psig) for Perry (Reference 14).

Drywell Head

The governing failure mode of the River Bend drywell head was shear failure caused by the pin tearing through the tongue-and-groove joint of the upper cylinder. The failure pressure capacities associated with this failure mode are 1.19 MPa (173 psig), 0.896 MPa (130 psig), and 0.67 MPa (97 psig), coincident with temperature levels 21 degrees C (70 degrees F), 149 degrees C (300 degrees F), and 427 degrees C (800 degrees F). The mean yield failure pressure for the Perry drywell head was 0.536 MPa (77.8 psig), controlled by the flange capability.

Equipment Hatch and Personnel Airlock Combination

The drywell for both of these plants has one personnel airlock and one combination personnel airlock and equipment hatch. As stated in Section 4.4.3.5 of the River Bend IPE, the critical failure mode for this combination of airlock and hatch is leakage through the personnel doors. Incipient leakage through the door seals was expected at internal pressure levels of 29 psig and 42 psig, coincident with temperatures of 21 degrees C (70 degrees F) and 149 degrees C (300 degrees F), respectively. Severe degradation of the seals around the door was expected at temperatures higher than 316 degrees C (600 degrees F). Estimated leakage area around the personnel doors was 12 in^2 at 149 degrees C (300 degrees F), and the leakage area increased to approximately 32 in^2 as the temperature rose higher than 316 degrees C (600 degrees F).

The personnel airlock for Perry is similar to the containment airlock, with a mean yield pressure of 0.7391 MPa (107.2 psig), whereas the pressure was 0.583 MPa (84.7 psig) for the equipment hatch.

F.3.3.2 Summary and Conclusions

A detailed review of the IPE reports listed in References 12 through 15 resulted in the following conclusions about the Mark III containment failure characterization:

Mark III Containment with a Reinforced Concrete Primary Containment with Steel Liner (Clinton and Grand Gulf)

The primary containment for both of these plants would have a 50-percent probability of failure at a pressure range of 0.39 to 0.646 MPa (56 psig to 93.8 psig), with the likely failure mode being tearing of the steel liner in the vicinity of one of the major containment penetrations (such as the personnel airlocks or the equipment hatches). The personnel airlocks and equipment hatches have higher capacities that the steel liner surrounding them. As the pressure increases, failure of the containment shell (hoop rebar) would occur around midheight, at significantly higher pressure. Such failures would be considered limited in size such that containment pressurization is prevented, and a gradual depressurization would occur. Failure around the airlocks and hatches would be structural failures resulting in rapid depressurization of the containment.

Mark III Containment with a Freestanding Steel Primary Containment (Perry and River Bend)

A review of the IPE reports for both of these plants concluded that the pressure capacity of their freestanding steel primary containment was in the range of 0.27 and 0.46 MPa (39 psig and 67 psig), coincident with a temperature range of 21 degrees C (70 degrees F) to 149 degrees C

(300 degrees F). Failure for both of these containments occurred at around penetrations in the steel containment.

APPENDIX G
DETAILED CHAPTER 6 ANALYSIS RESULTS

DETAILED CHAPTER 6 ANALYSIS RESULTS

G.1 LOOP Scenarios

G.1.1 Case 1: LOOPGR-38-9, CST Available, Initial CST Level Decreased to 24 ft., Nominal Recirculation Pump Seal Leakage

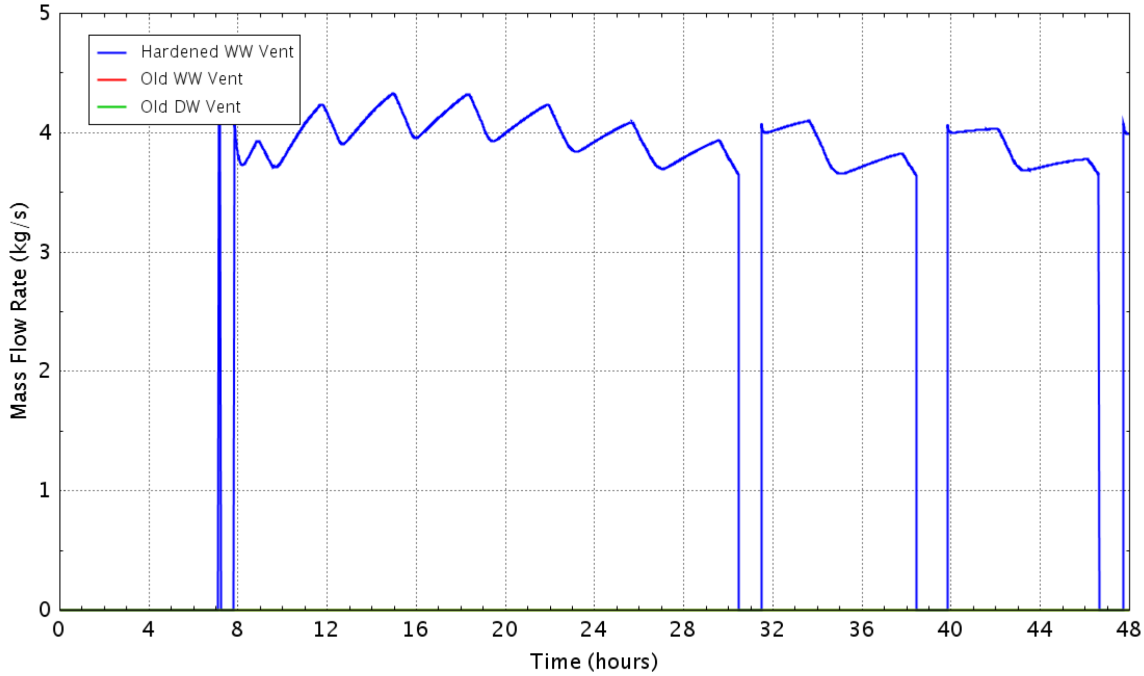

Figure G – 1 Flow rate of the containment vents

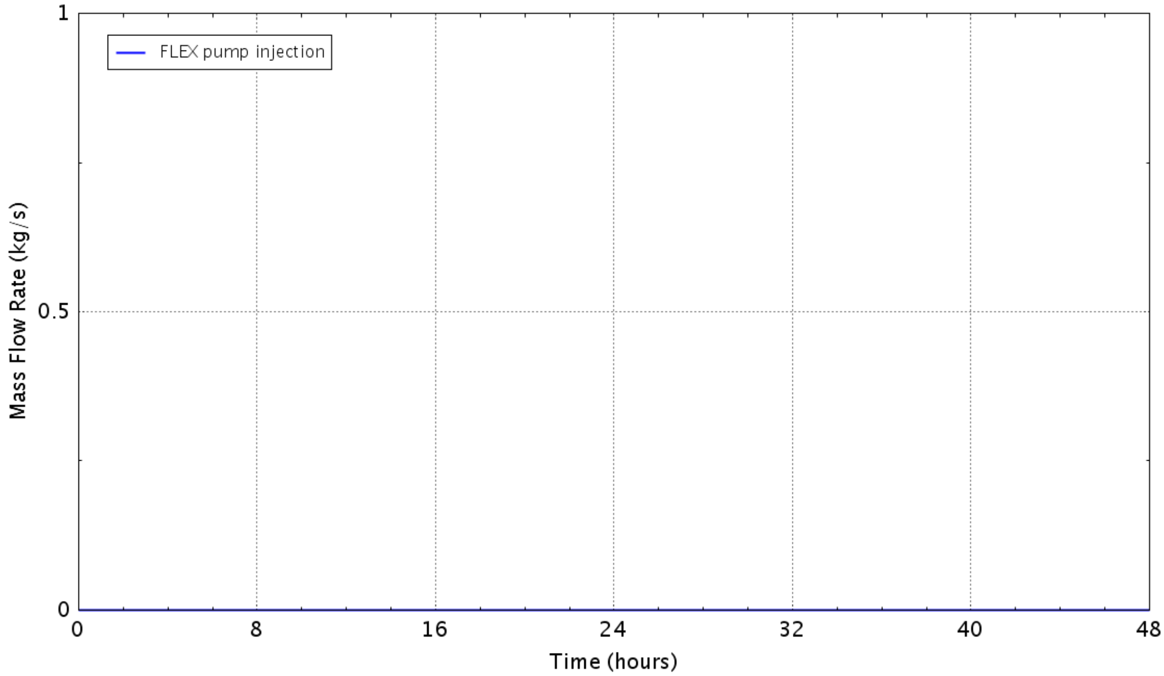

Figure G – 2 Flow rate of the FLEX pump

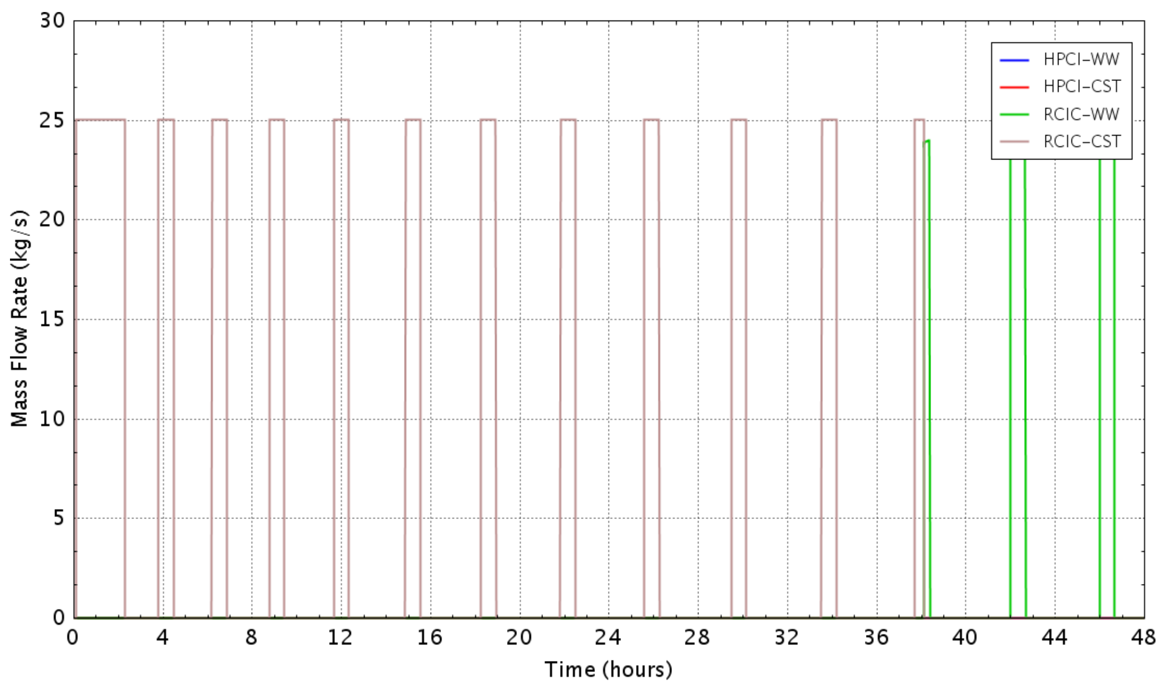

Figure G – 3 Flow rate of the HPCI/RCIC pumps

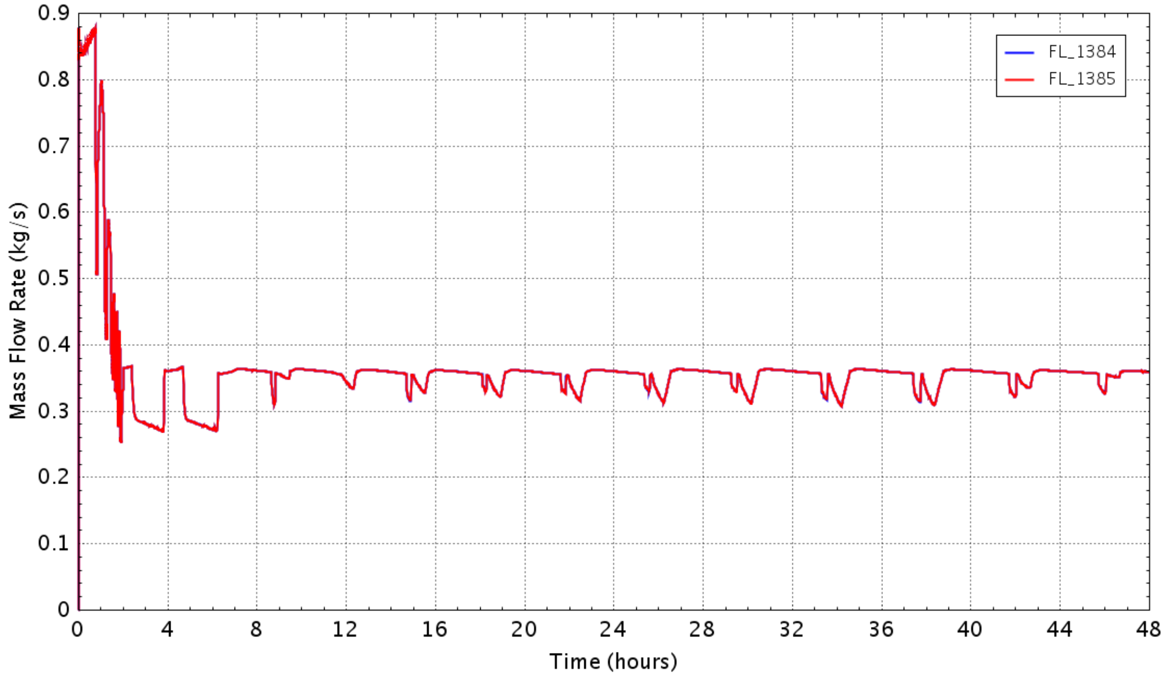

Figure G – 4 Flow rate of the recirculating pump seal leakage

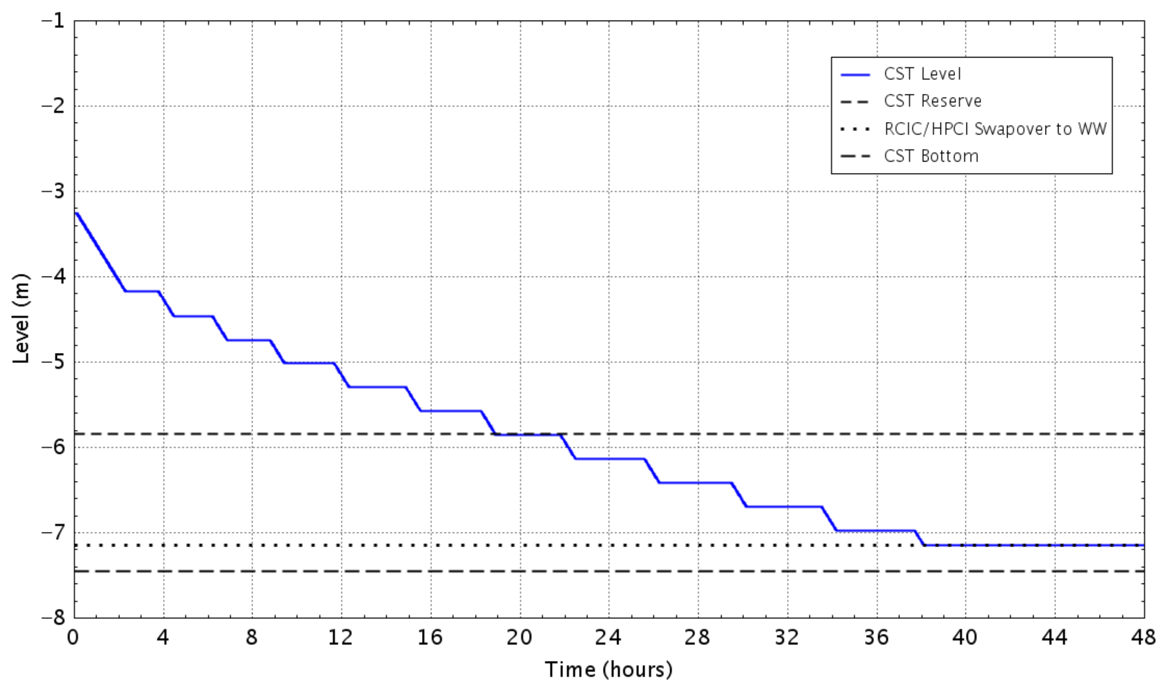

Figure G – 5 Water level in the CST

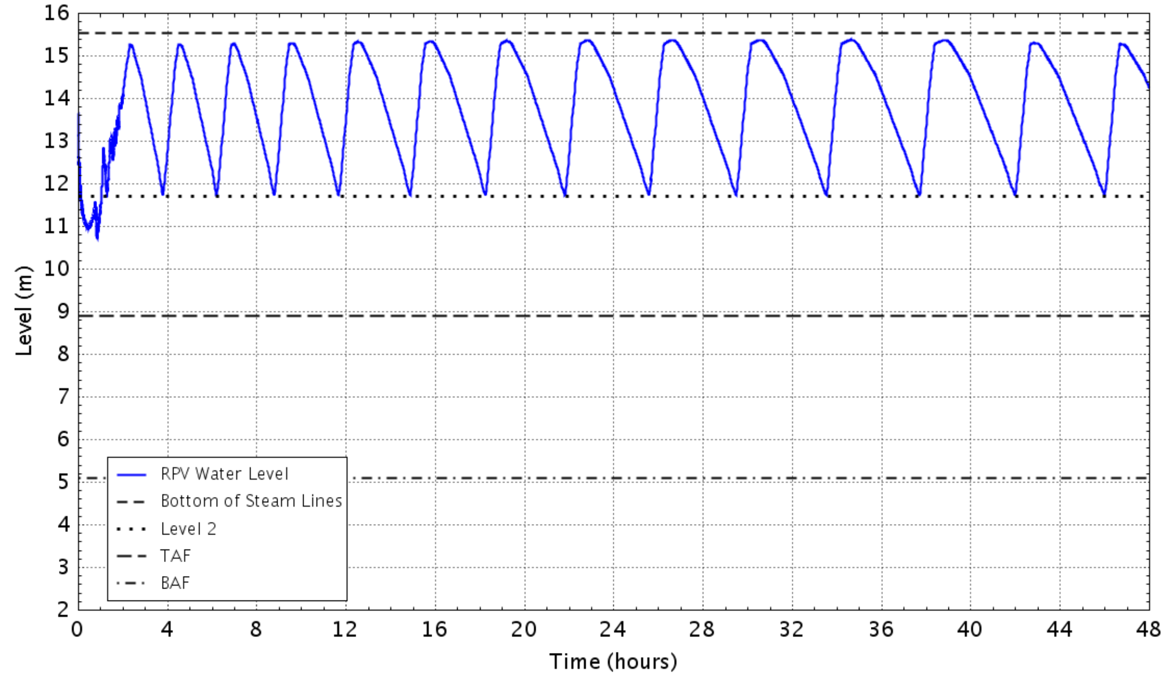

Figure G - 6 RPV Downcomer water level

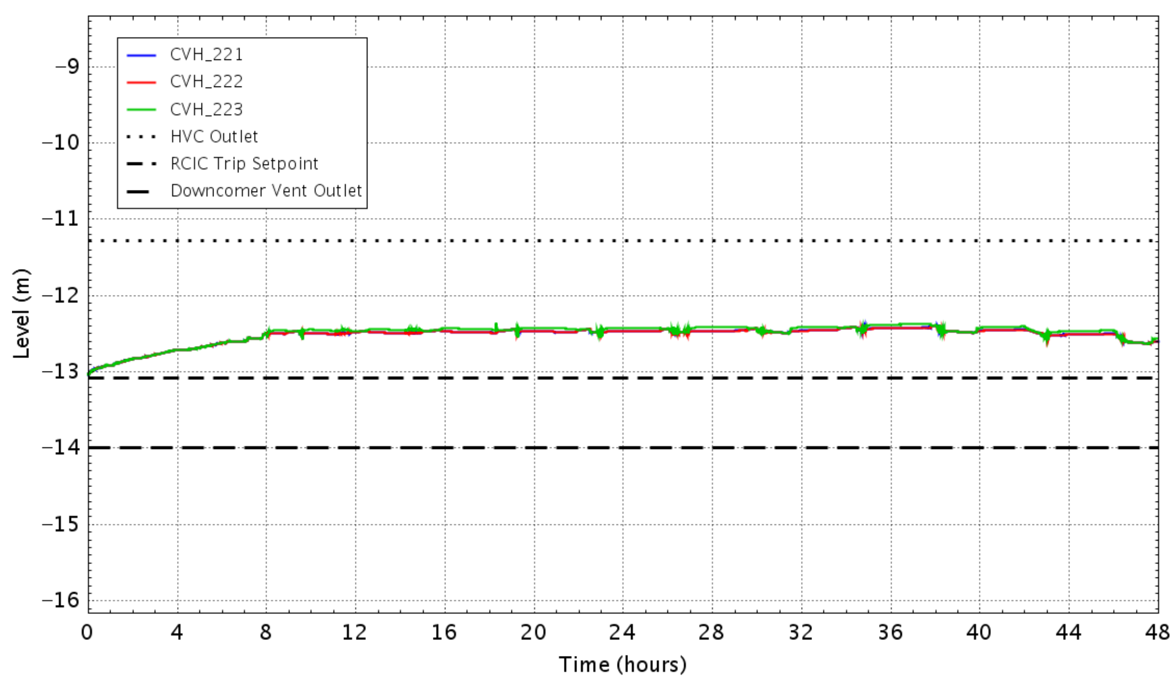

Figure G – 7 Water level in the wetwell

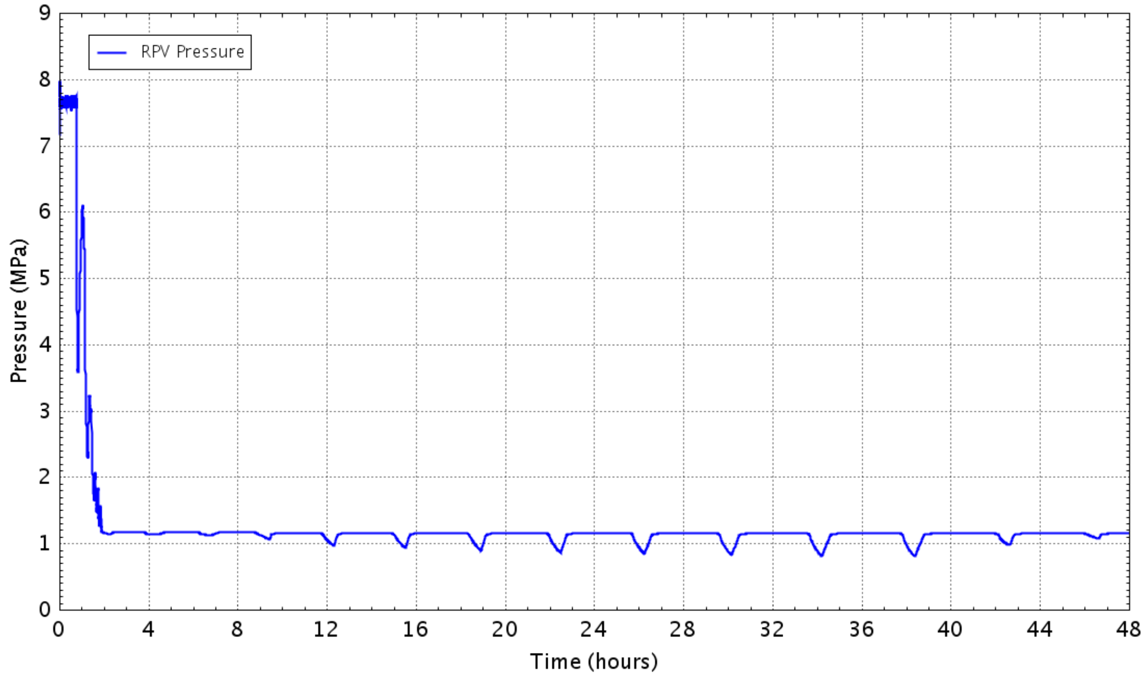

Figure G – 8 Pressure in the RPV

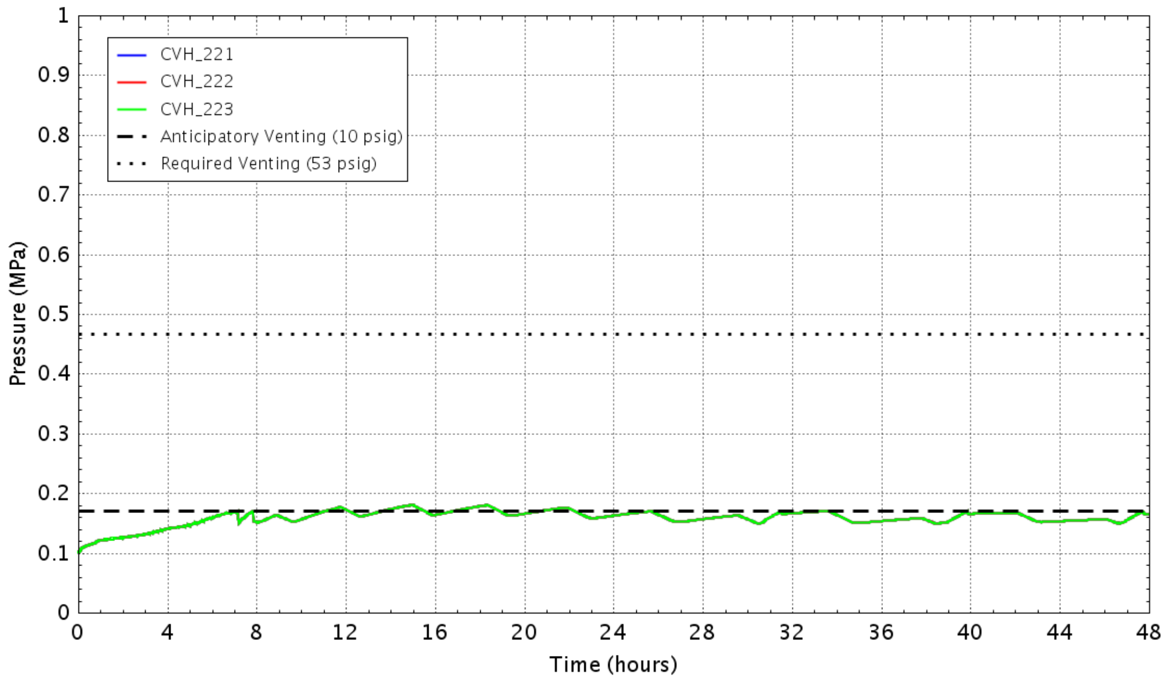

Figure G – 9 Pressure in the wetwell

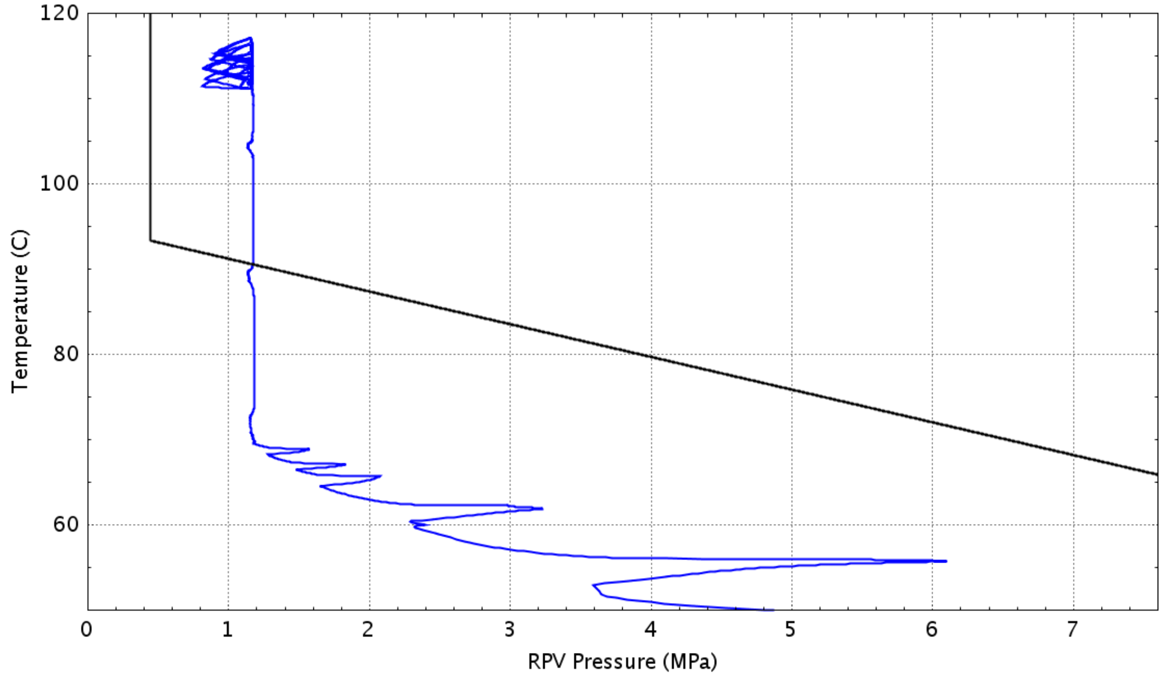

Figure G – 10 Plant status relative to the HCL curve (Graph 4 of the EOPs)

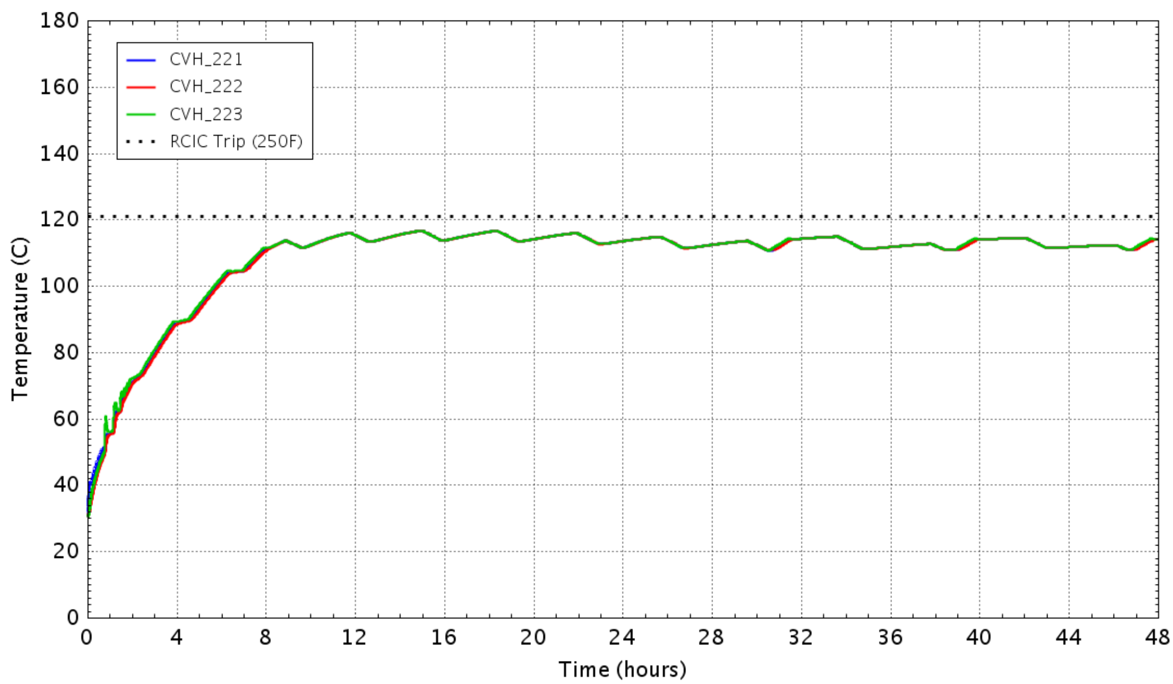

Figure G – 11 Water temperature in the wetwell

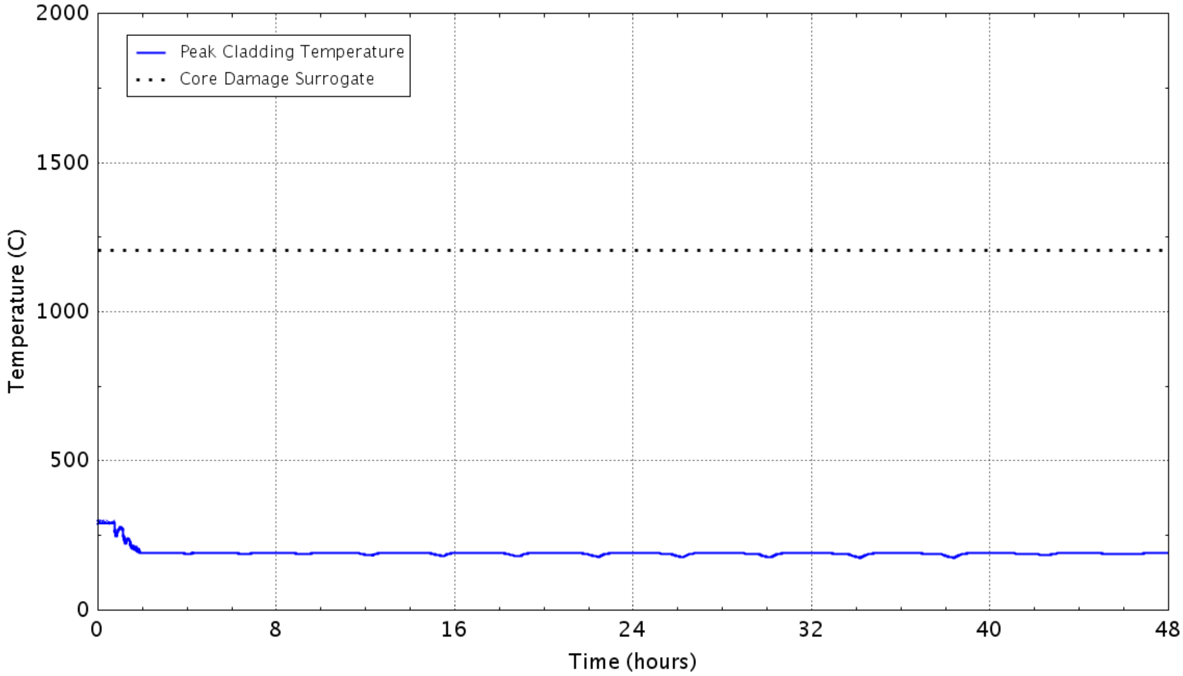

Figure G – 12 Peak temperature of the fuel cladding as a function of time

G.1.2 Case 2: LOOPGR-38-9, CST Available, Initial CST Level Decreased to 24 ft., 200 gpm Recirculation Pump Seal Leakage

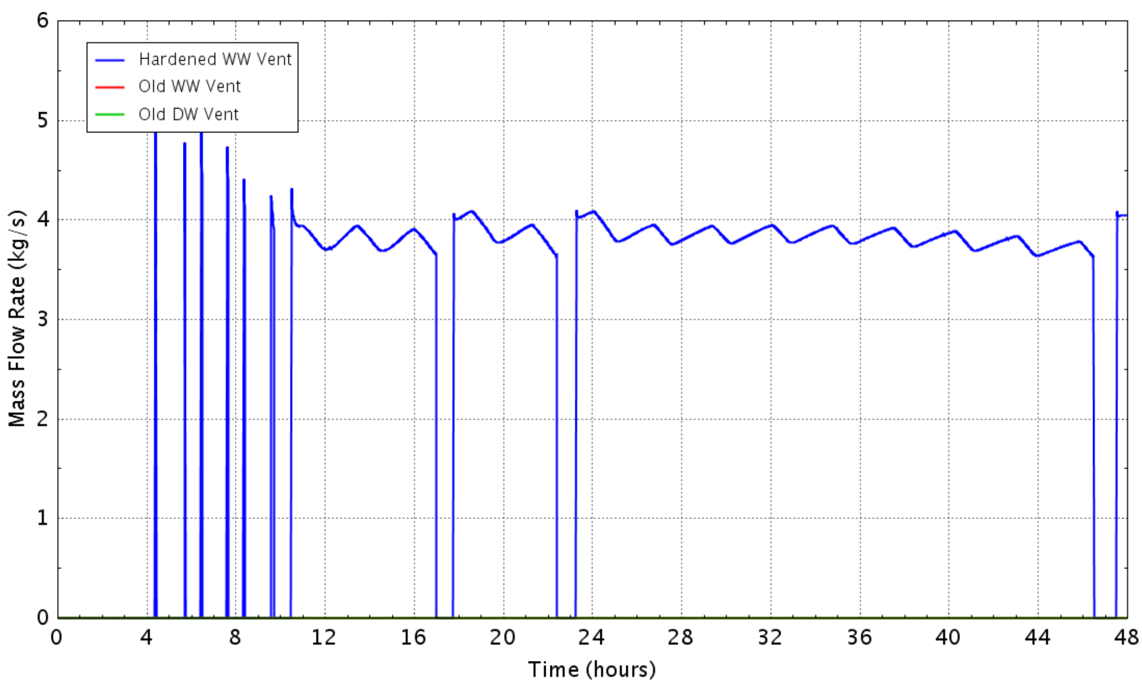

Figure G – 13 Flow rate of the containment vents

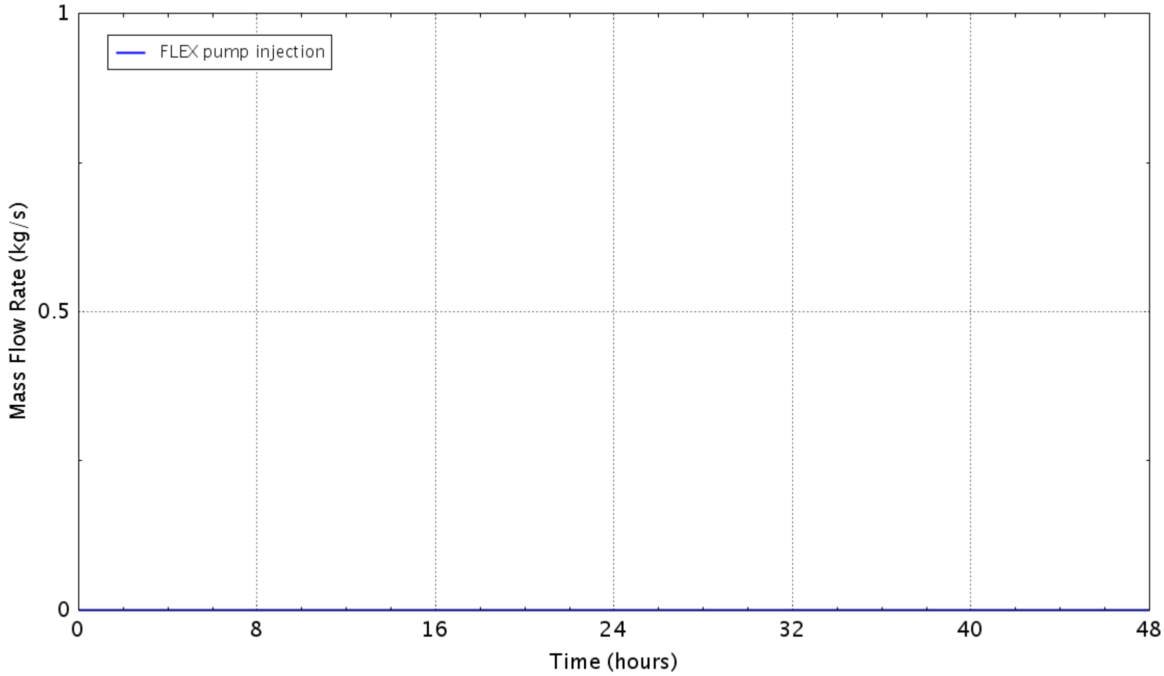

Figure G – 14 Flow rate of the FLEX pump

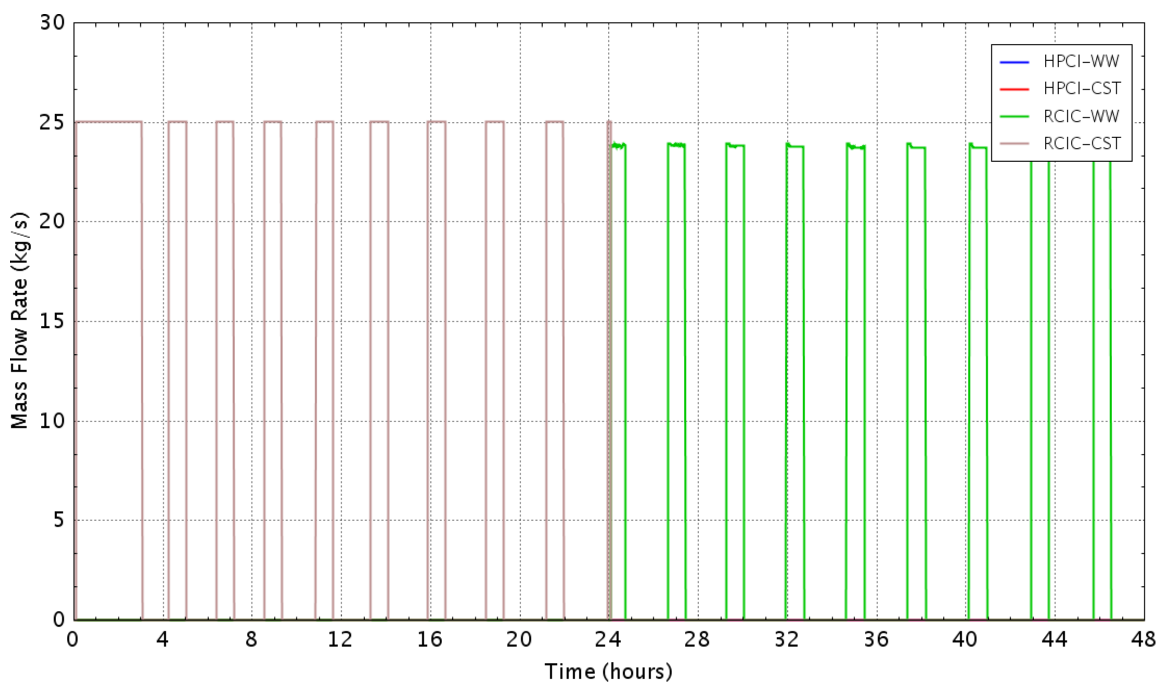

Figure G – 15 Flow rate of the HPCI/RCIC pumps

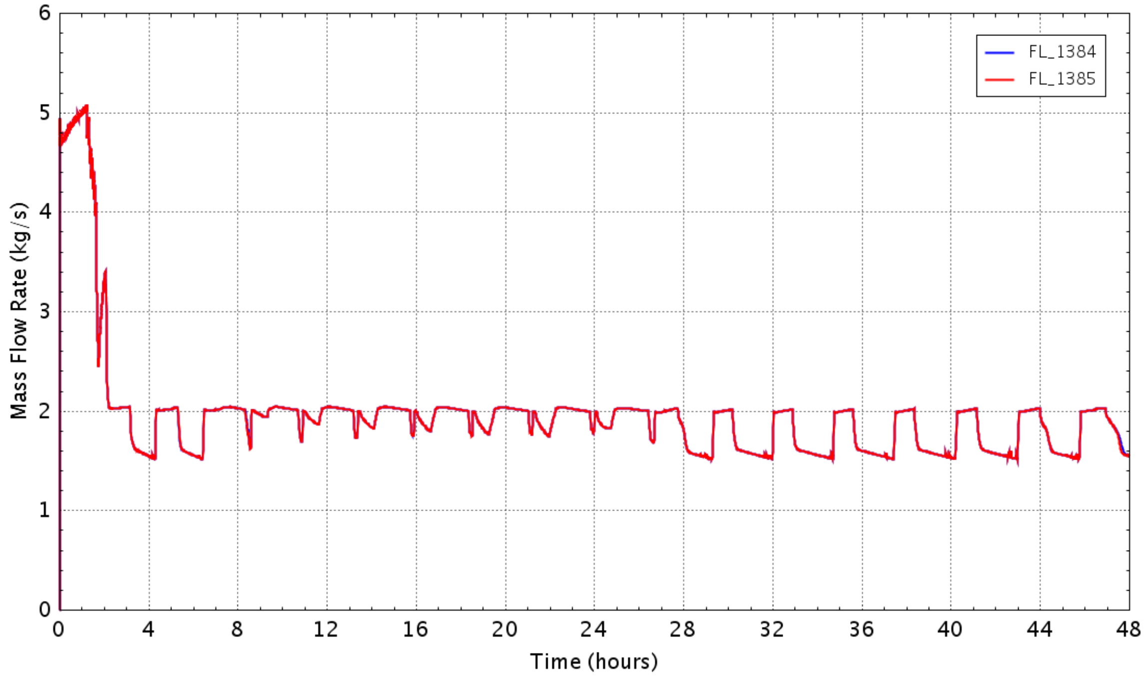

Figure G – 16 Flow rate of the recirculating pump seal leakage

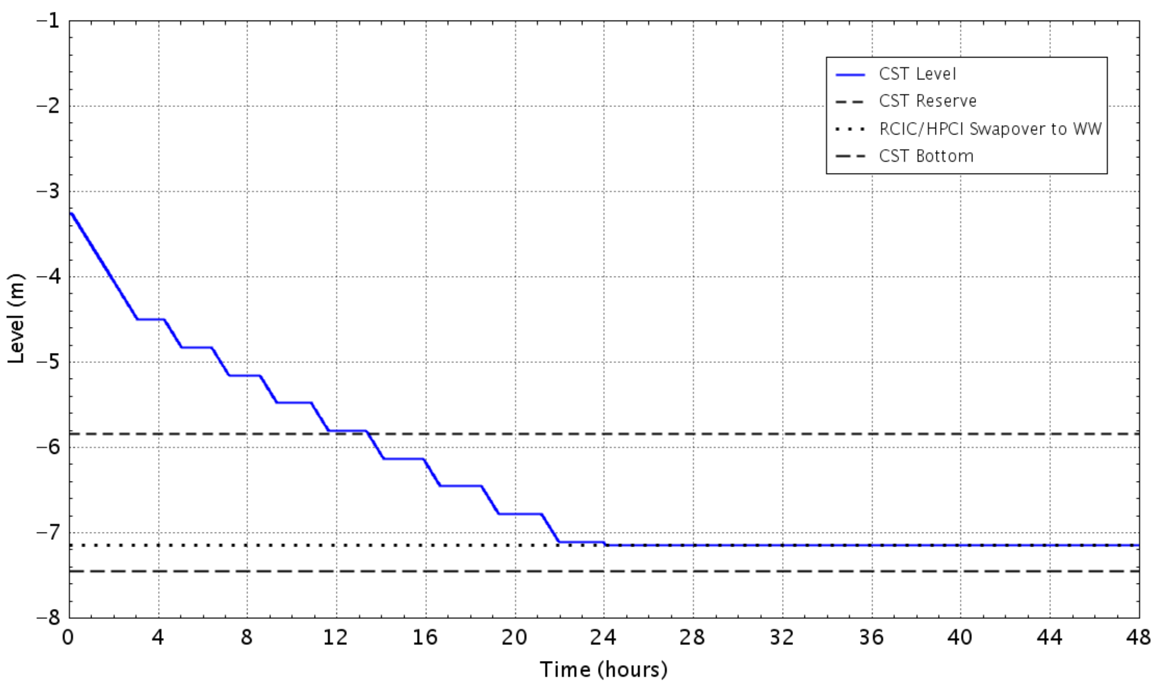

Figure G – 17 Water level in the CST

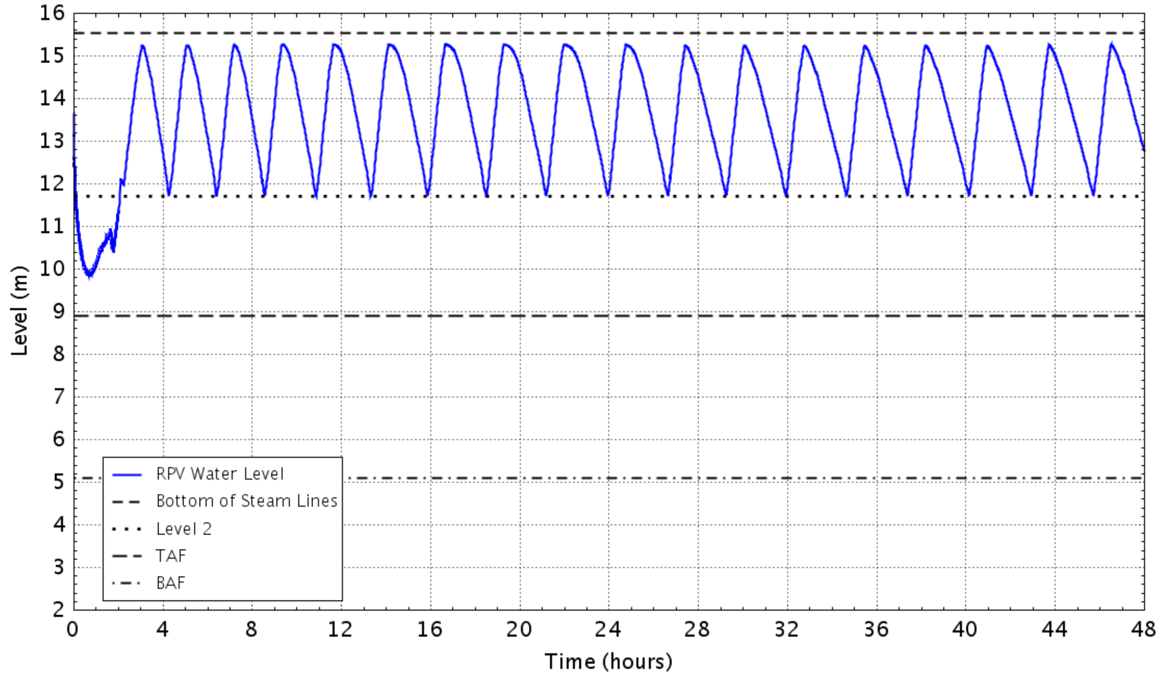

Figure G – 18 RPV Downcomer water level

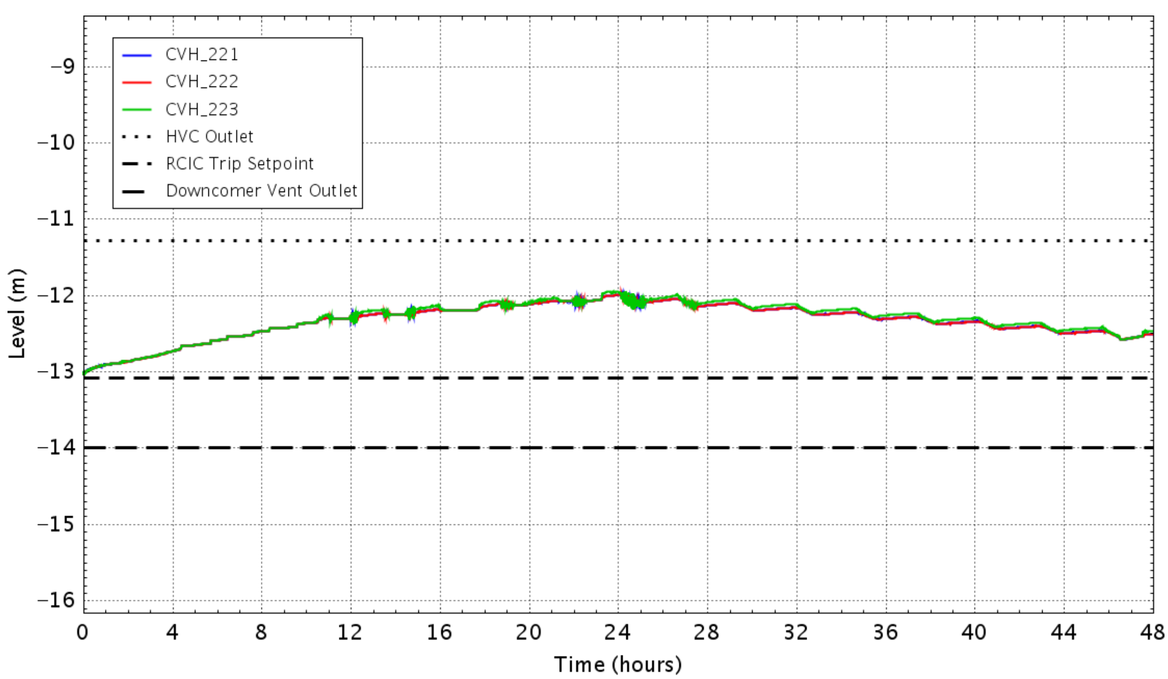

Figure G – 19 Water level in the wetwell

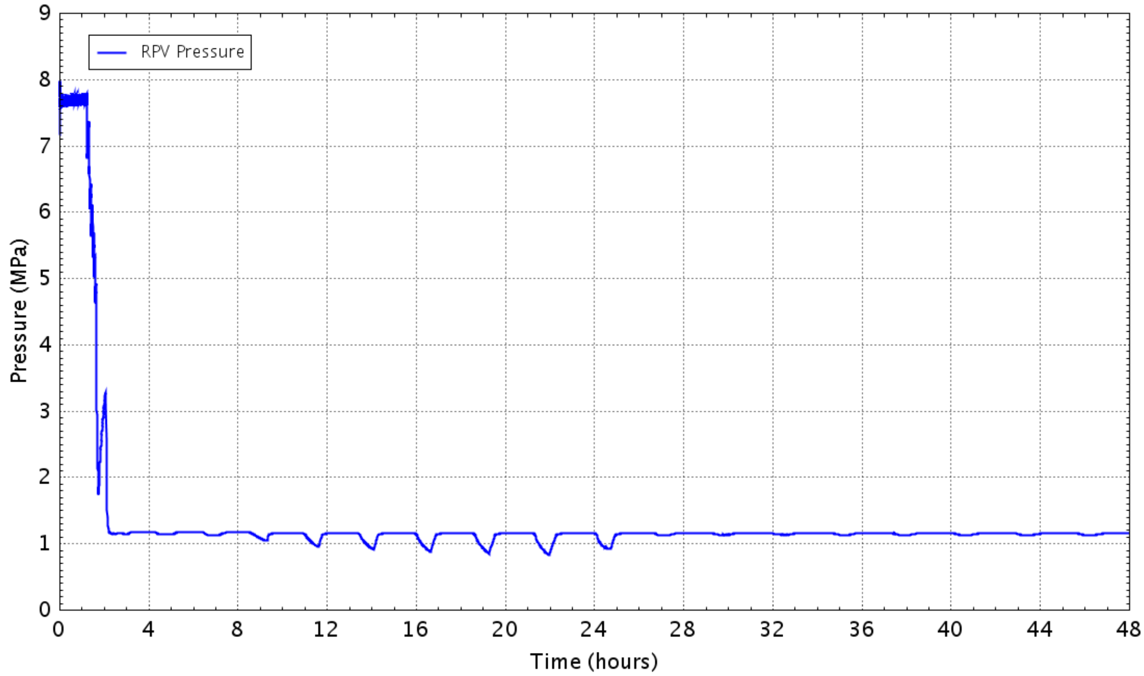

Figure G – 20 Pressure in the RPV

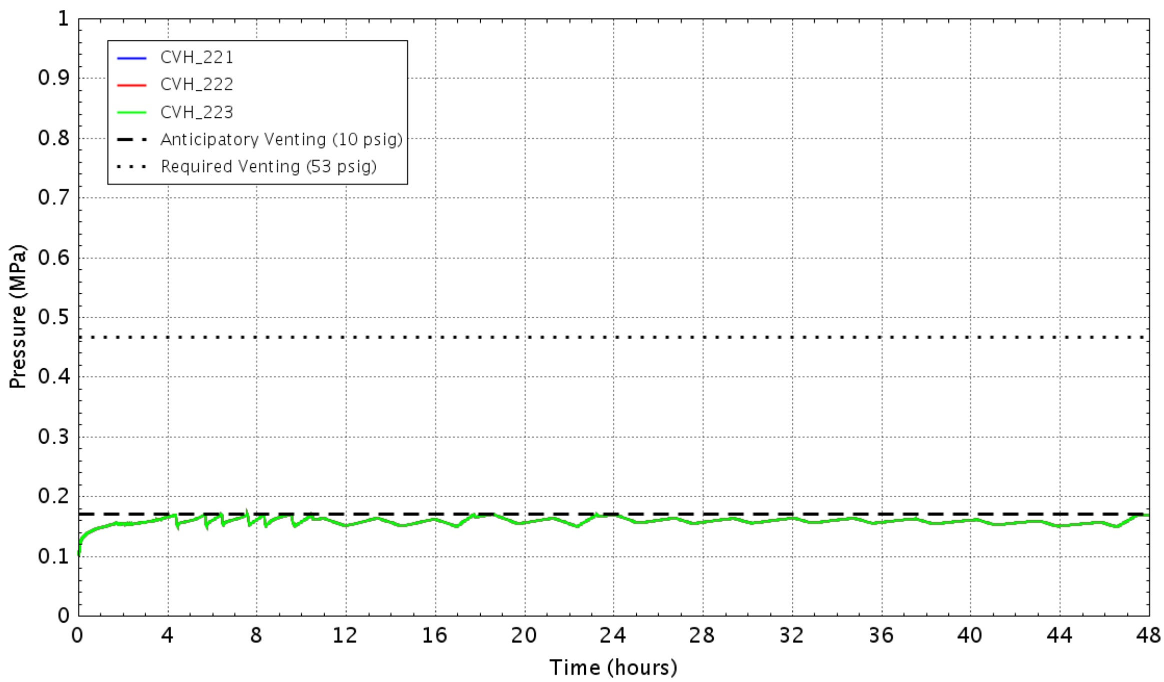

Figure G – 21 Pressure in the wetwell

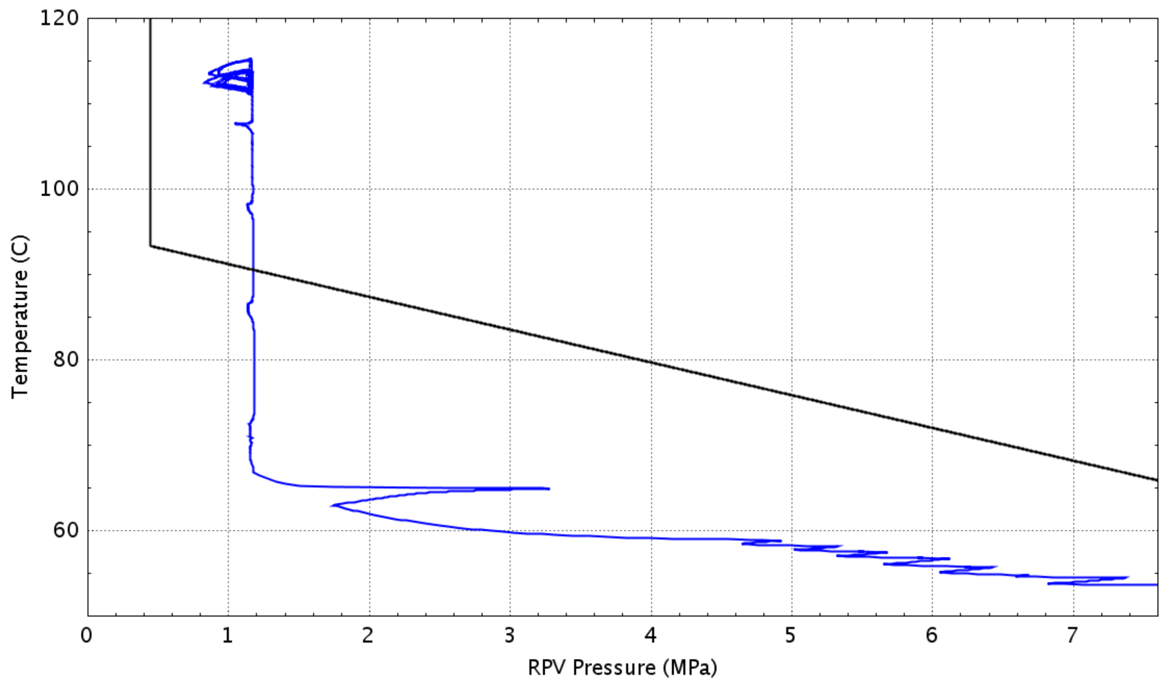

Figure G - 22 Plant status relative to the HCL curve (Graph 4 of the EOPs)

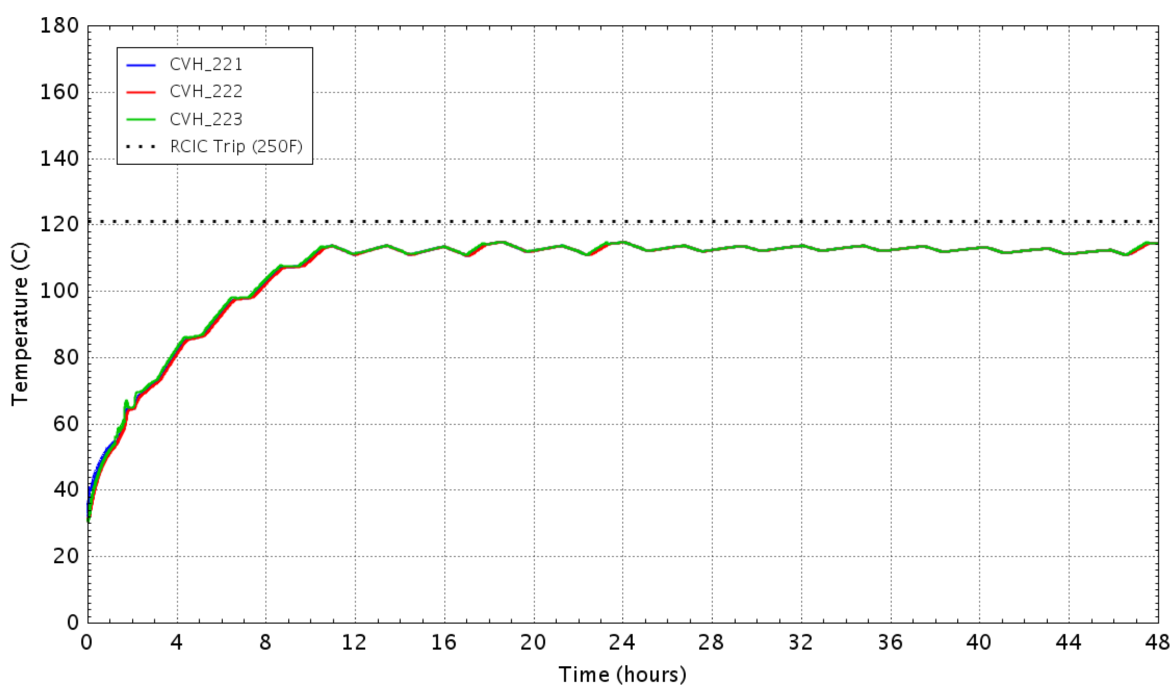

Figure G - 23 Water temperature in the wetwell

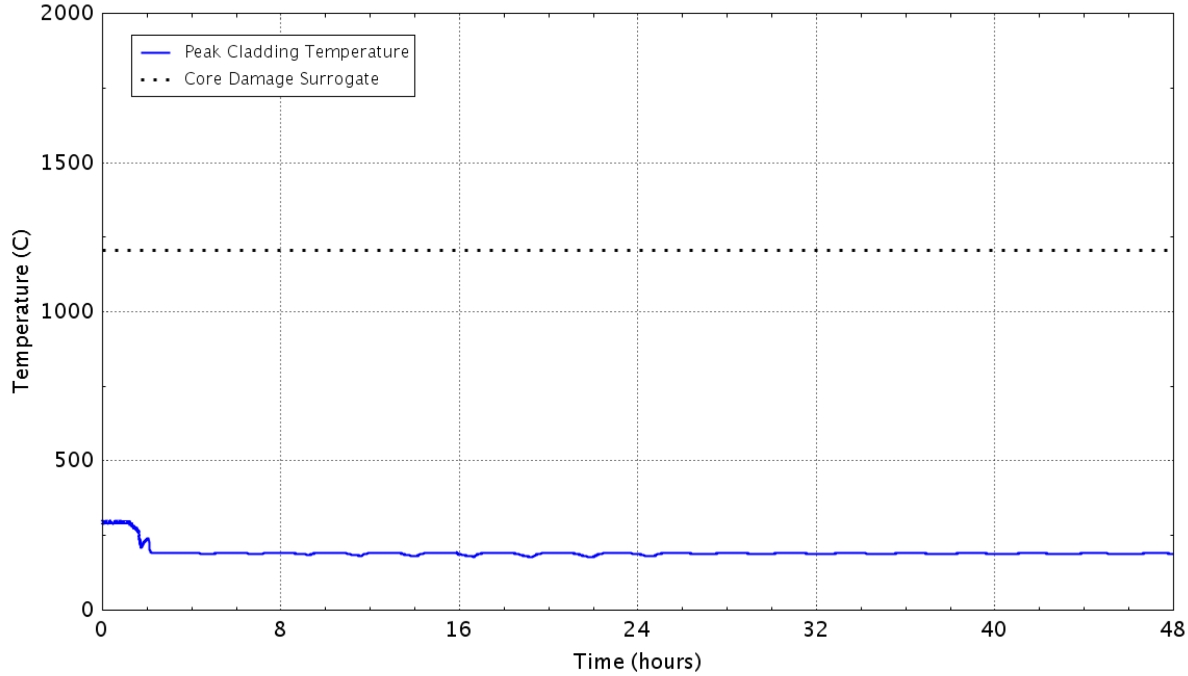

Figure G - 24 Peak temperature of the fuel cladding as a function of time

G.1.3 Case 3: LOOPGR-38-9, CST Available, Initial CST Level Increased to 36 ft., Nominal Recirculation Pump Seal Leakage

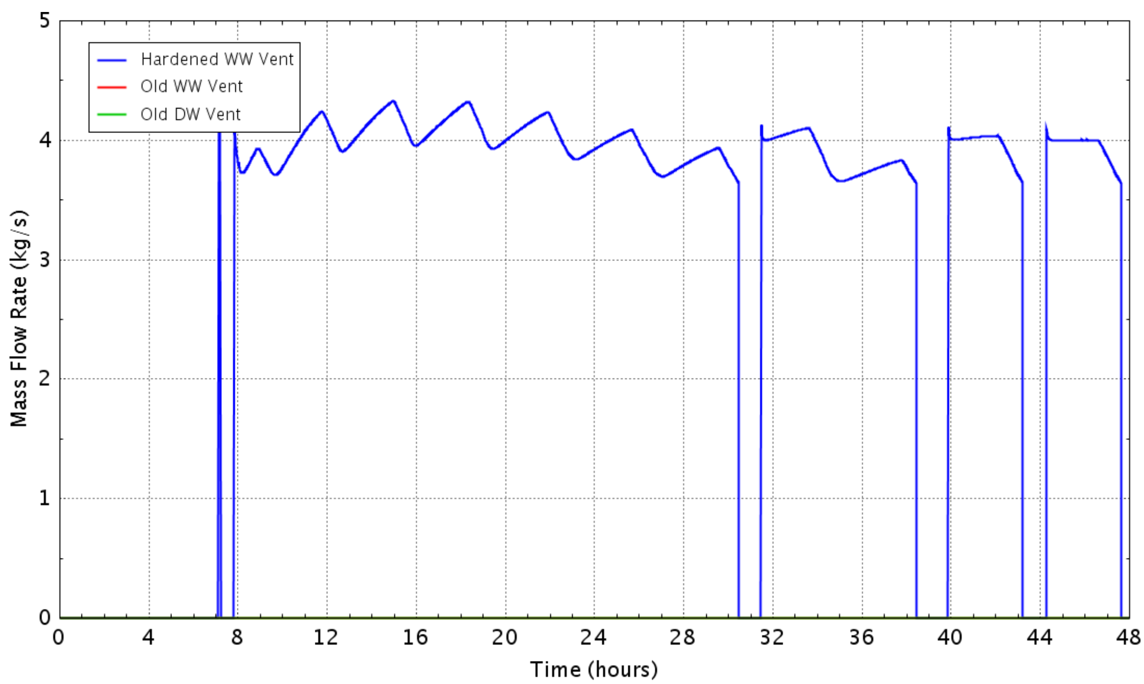

Figure G - 25 Flow rate of the containment vents

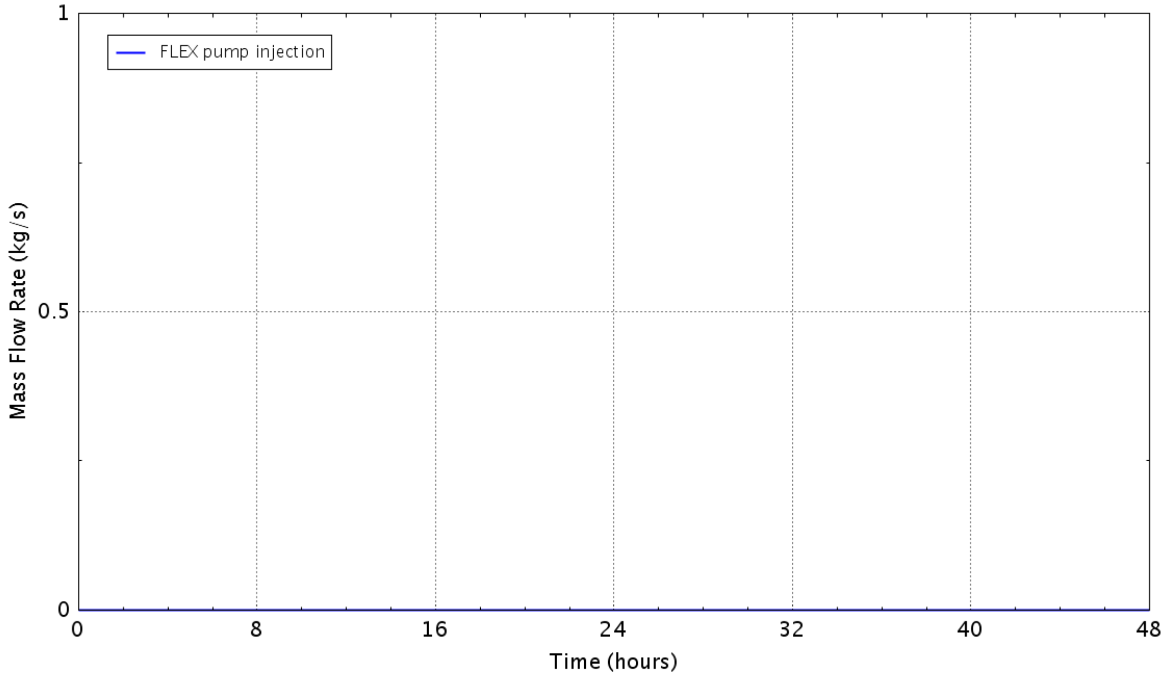

Figure G - 26 Flow rate of the FLEX pump

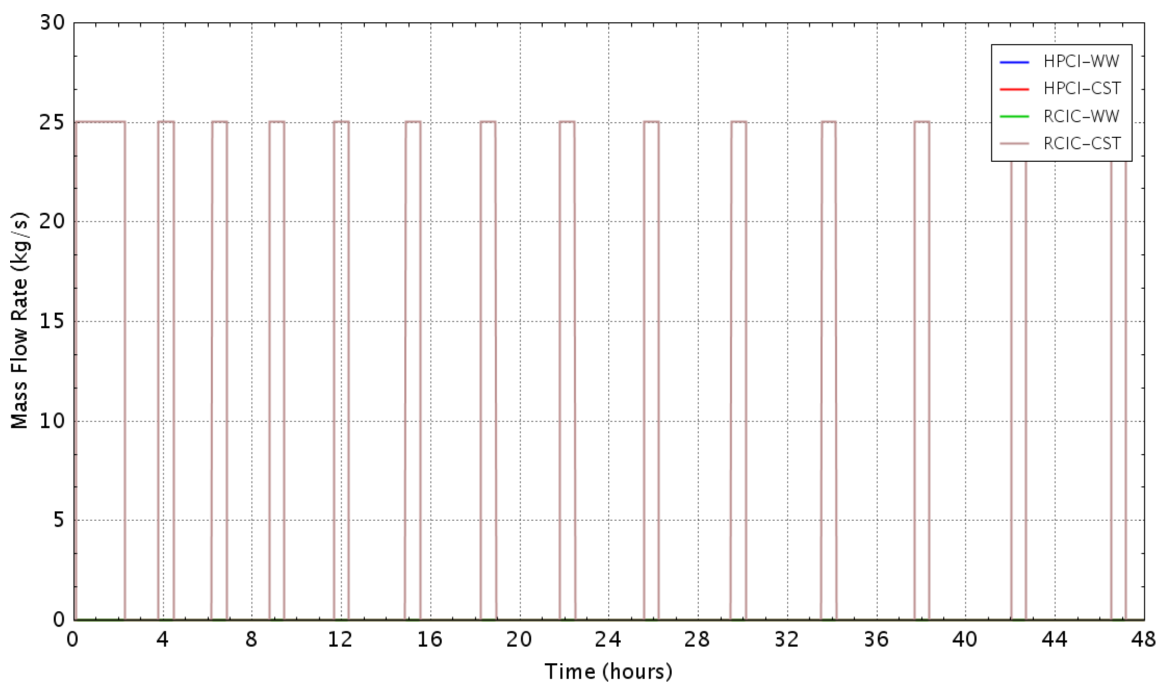

Figure G - 27 Flow rate of the HPCI/RCIC pumps

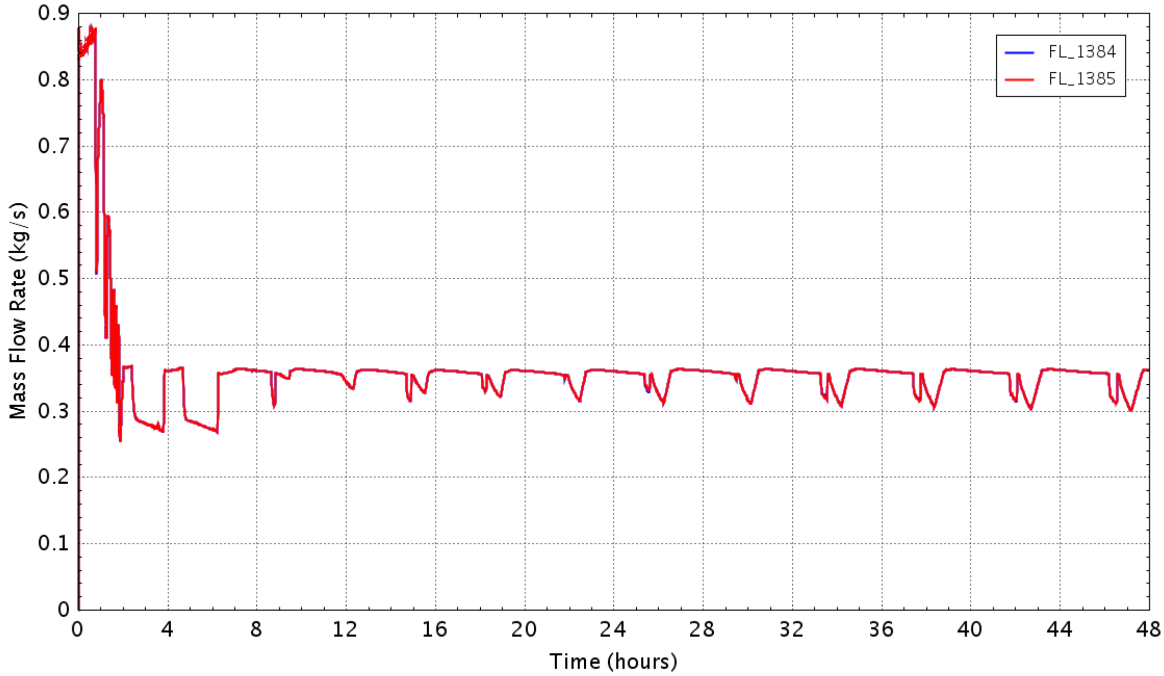

Figure G - 28 Flow rate of the recirculating pump seal leakage

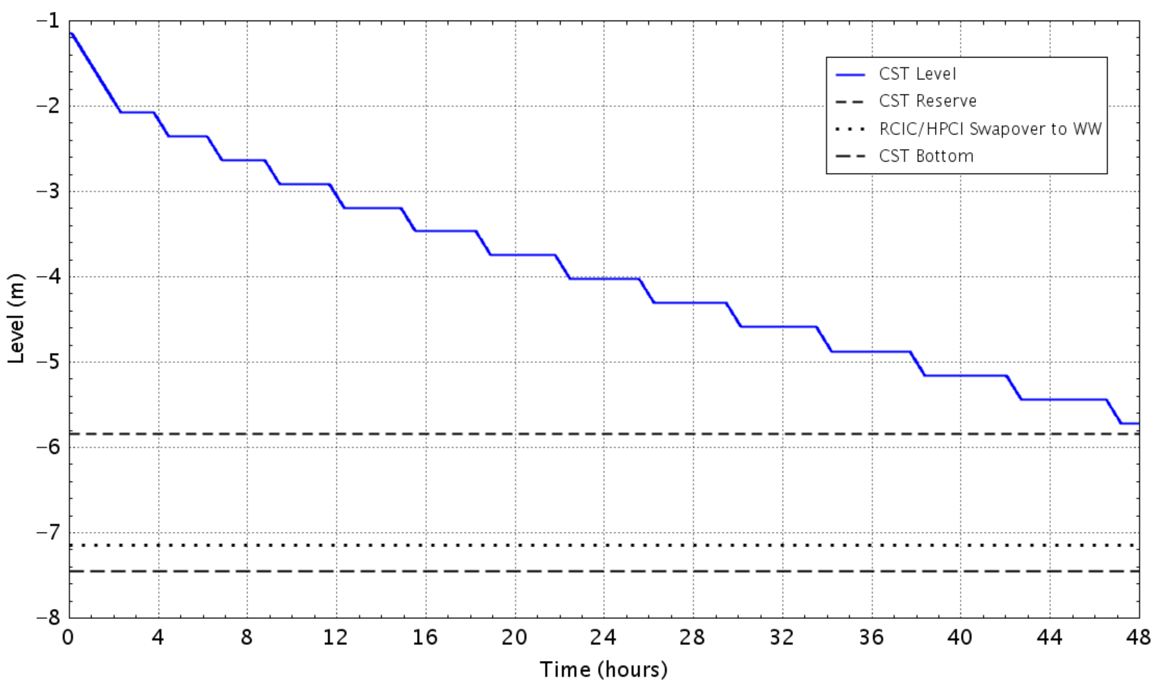

Figure G – 29 Water level in the CST

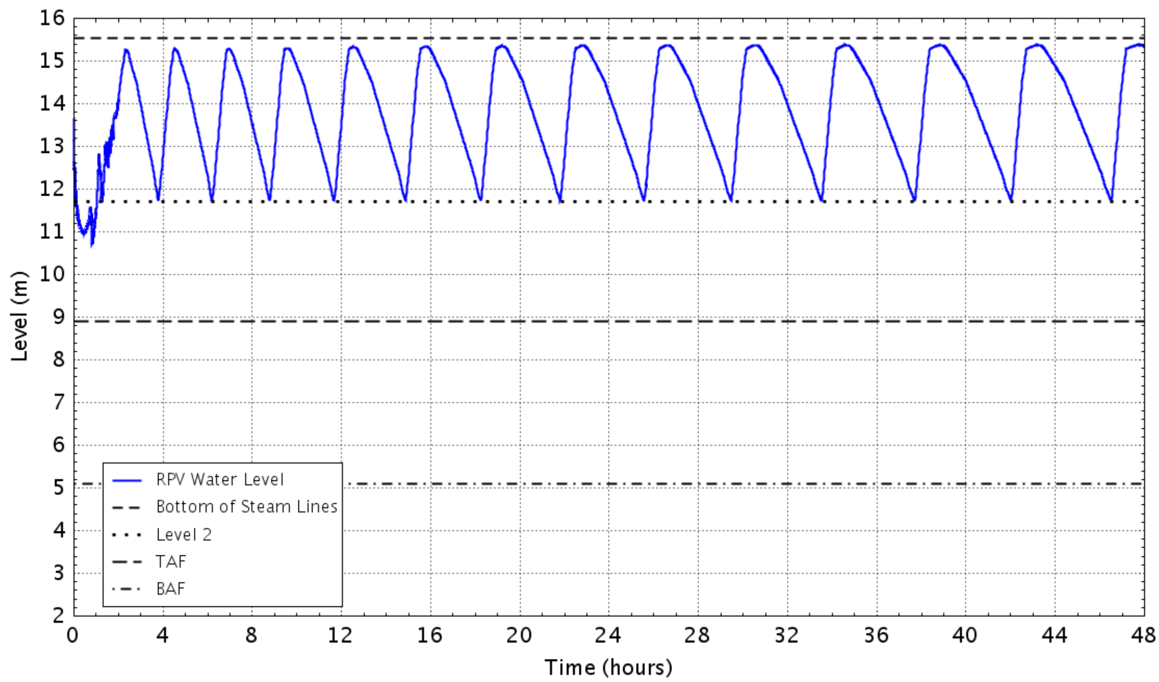

Figure G – 30 RPV Downcomer water level

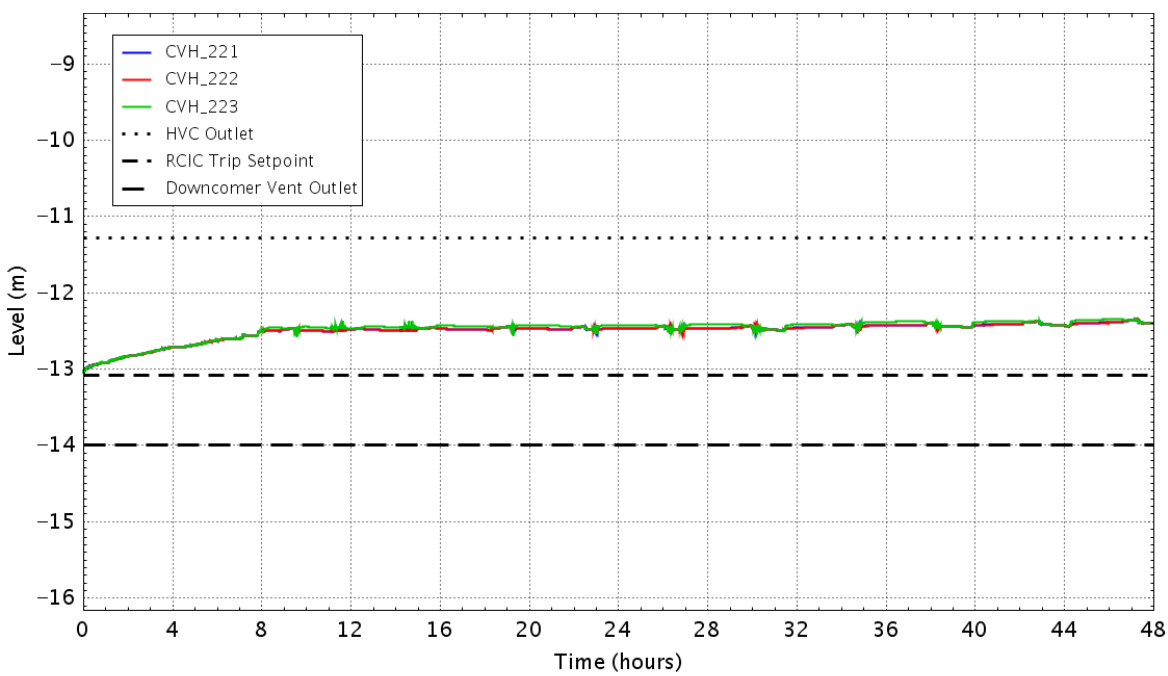

Figure G – 31 Water level in the wetwell

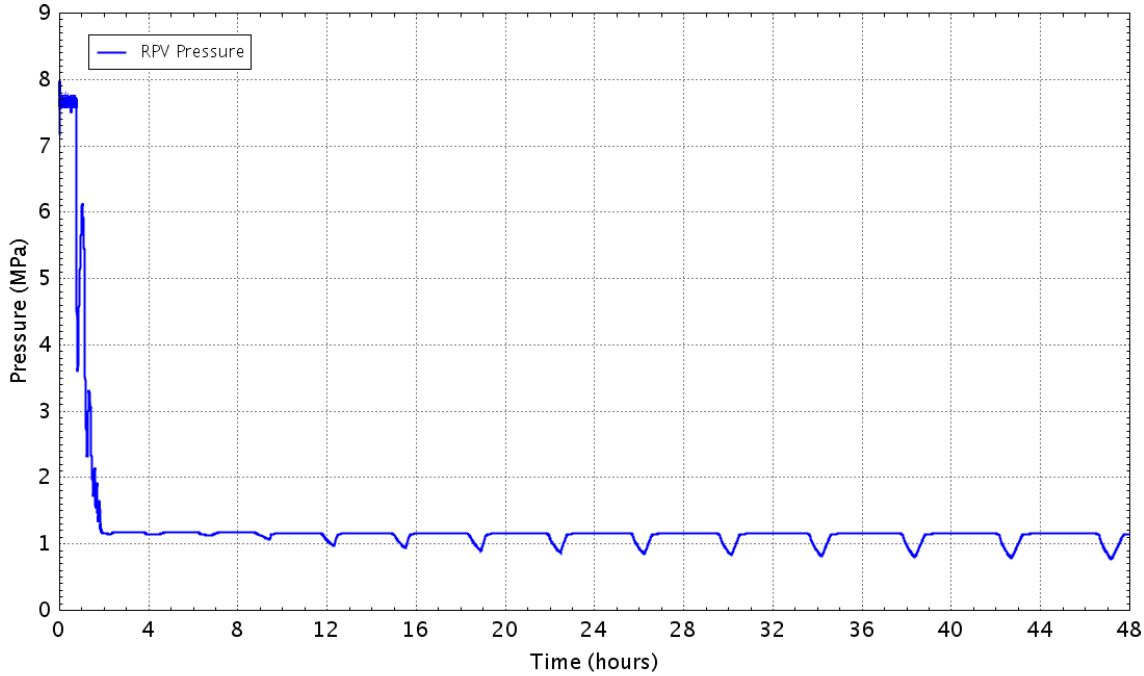

Figure G – 32 Pressure in the RPV

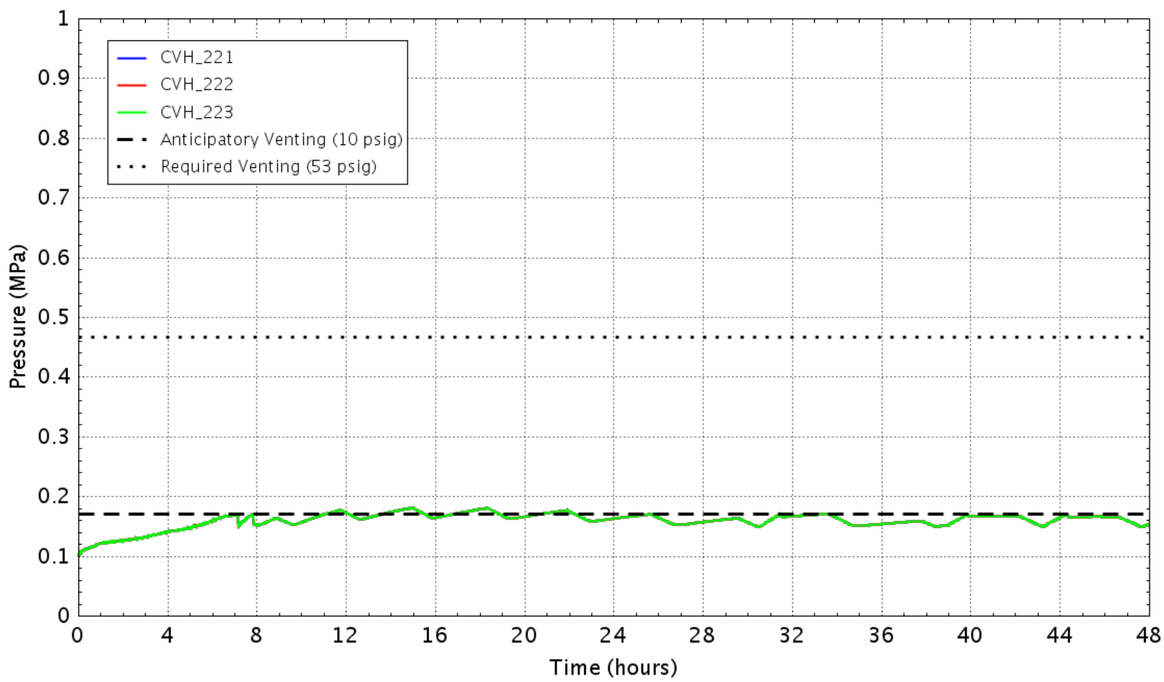

Figure G – 33 Pressure in the wetwell

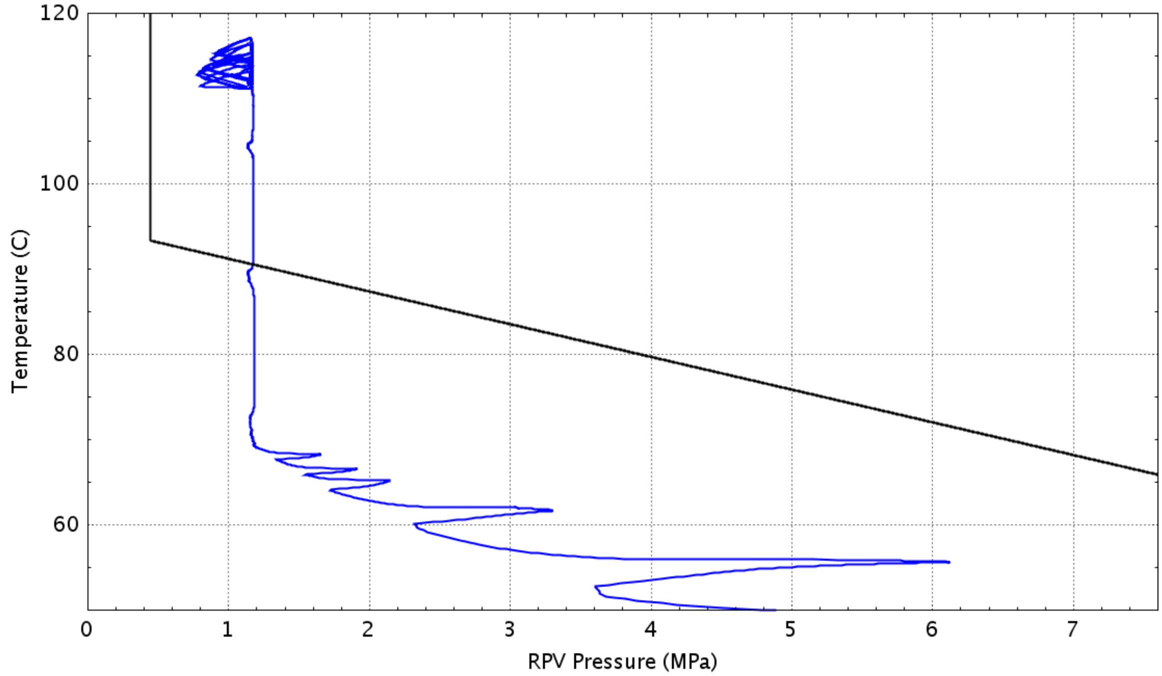

Figure G - 34 Plant status relative to the HCL curve (Graph 4 of the EOPs)

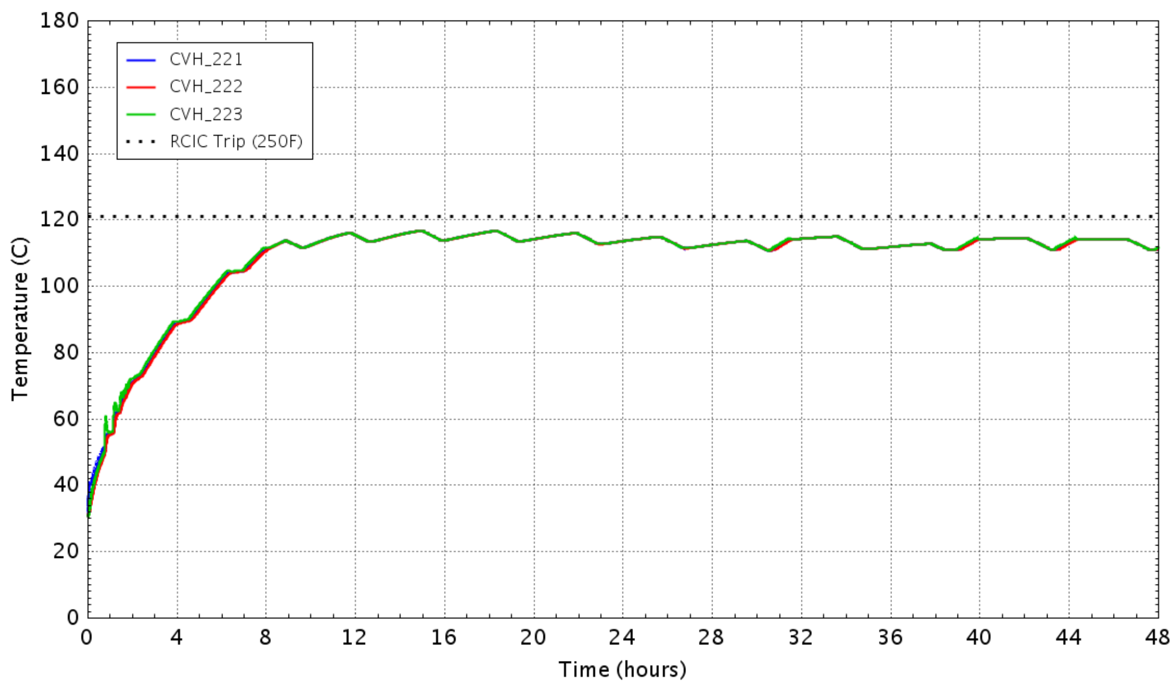

Figure G - 35 Water temperature in the wetwell

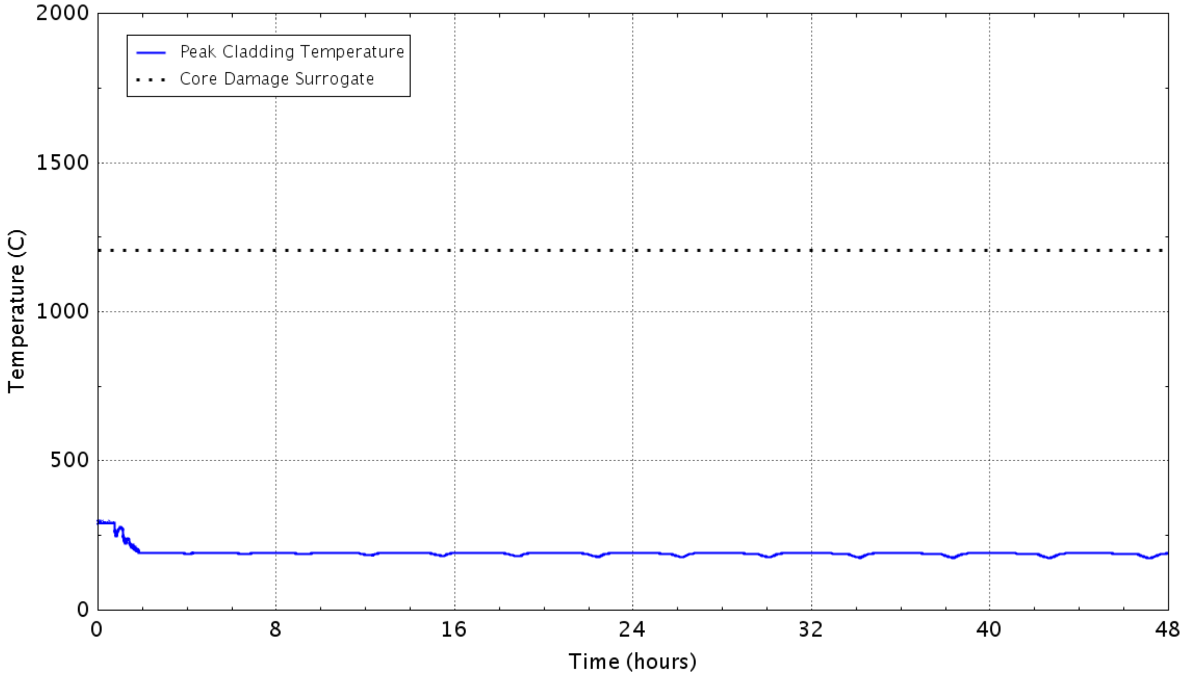

Figure G - 36 Peak temperature of the fuel cladding as a function of time

G.1.4 Case 4: LOOPGR-38-9, CST Available, Initial CST Level Increased to 36 ft., 200 gpm Recirculation Pump Seal Leakage

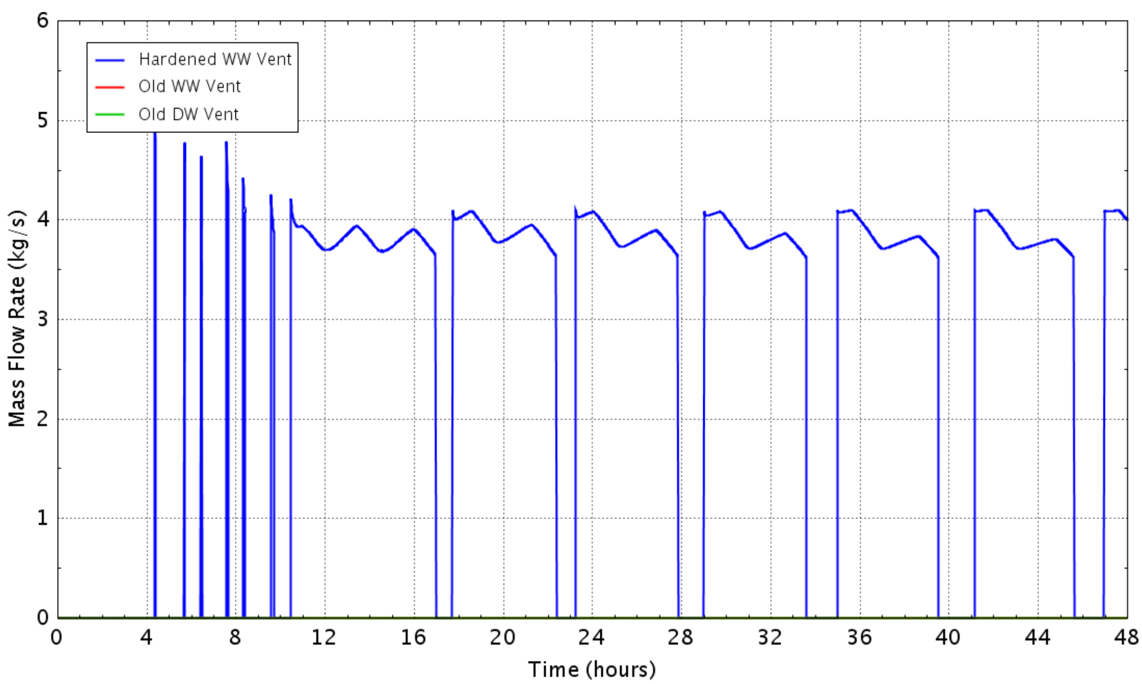

Figure G - 37 Flow rate of the containment vents

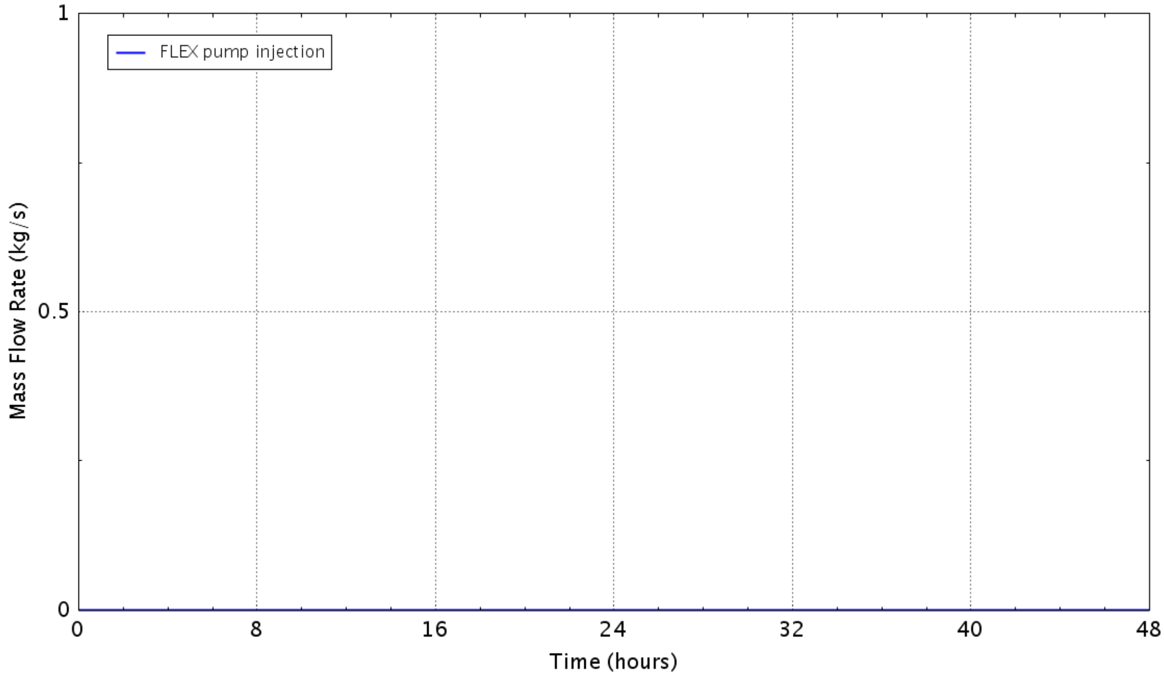

Figure G - 38 Flow rate of the FLEX pump

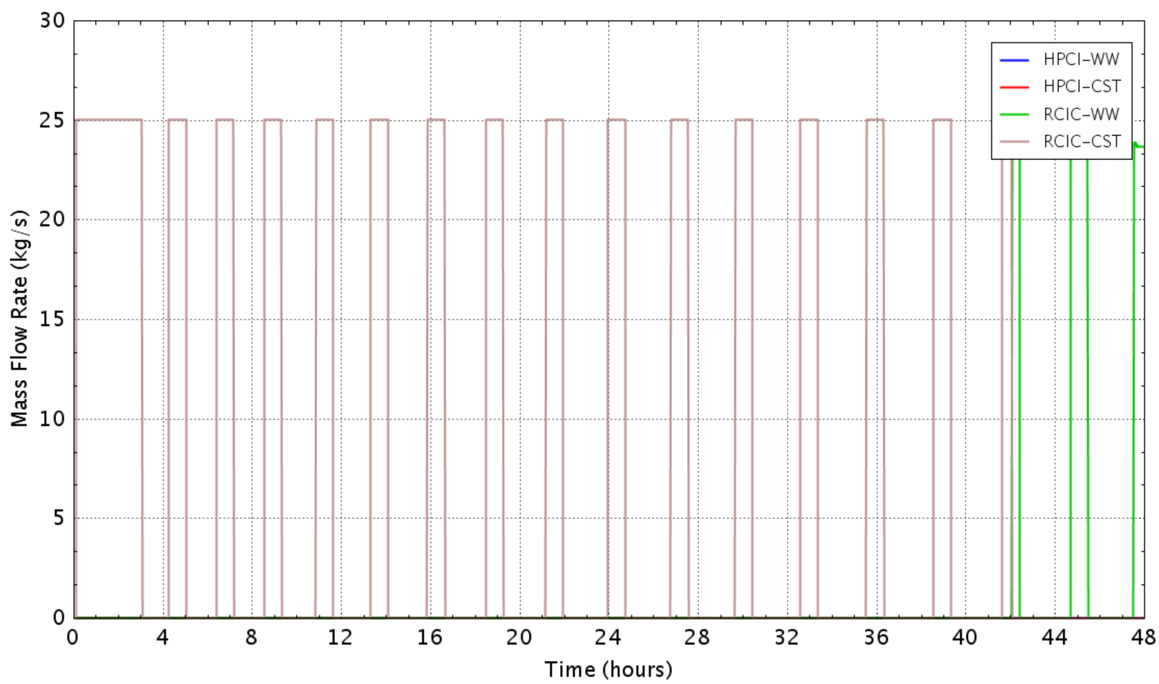

Figure G - 39 Flow rate of the HPCI/RCIC pumps

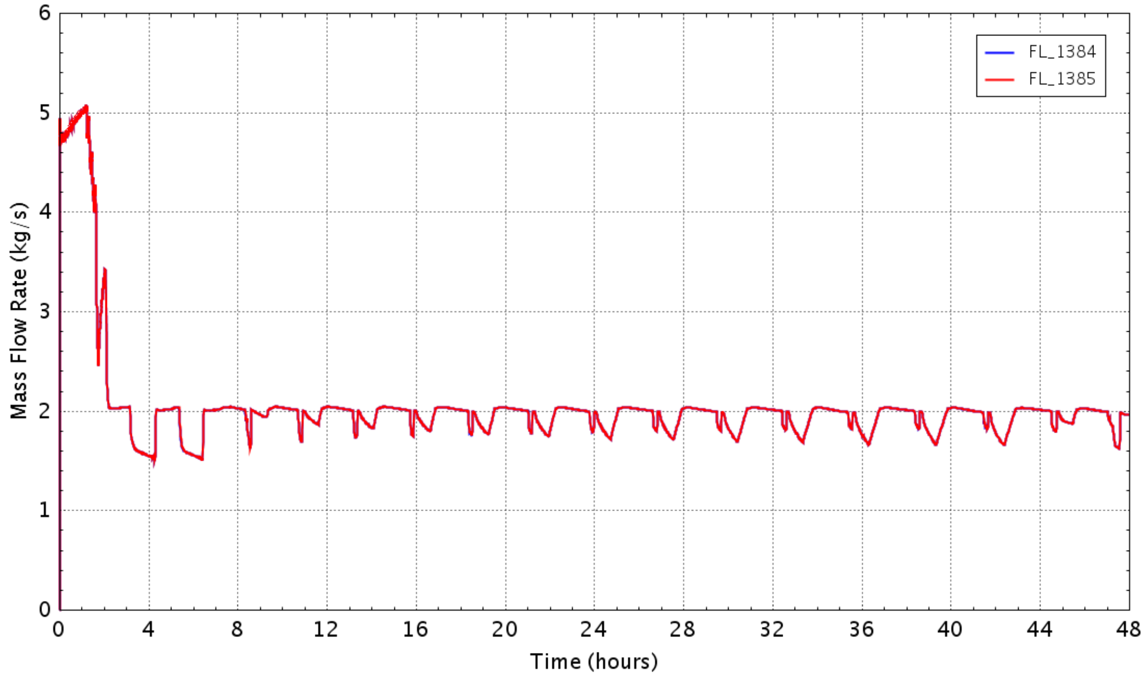

Figure G - 40 Flow rate of the recirculating pump seal leakage

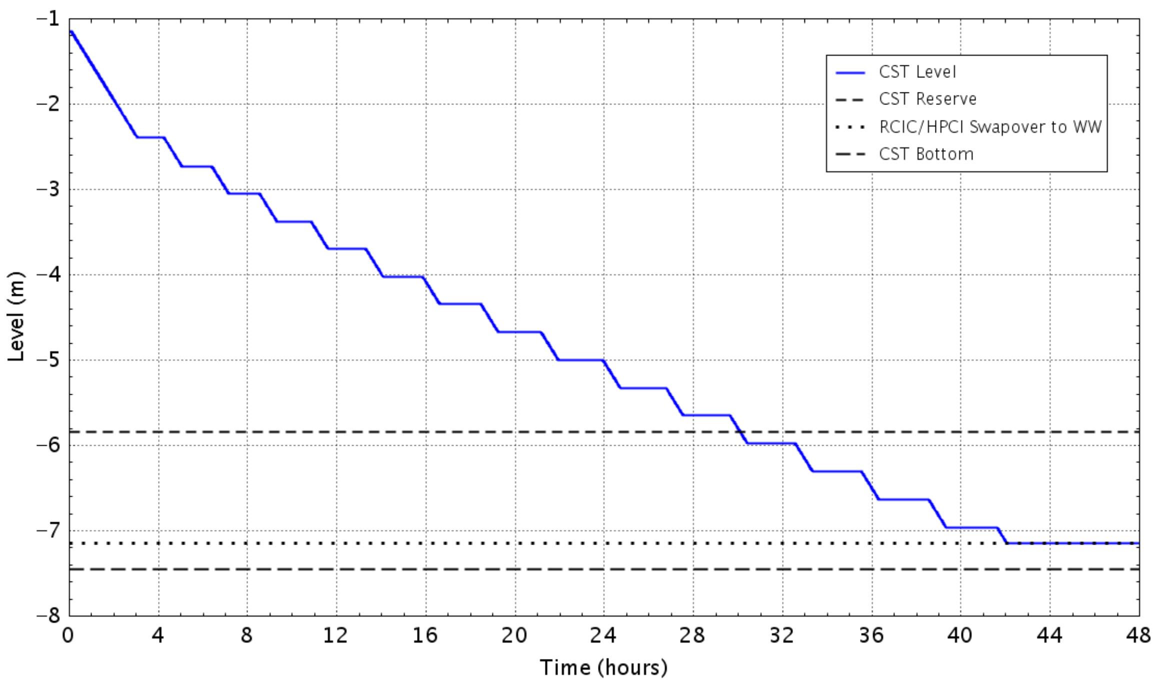

Figure G - 41 Water level in the CST

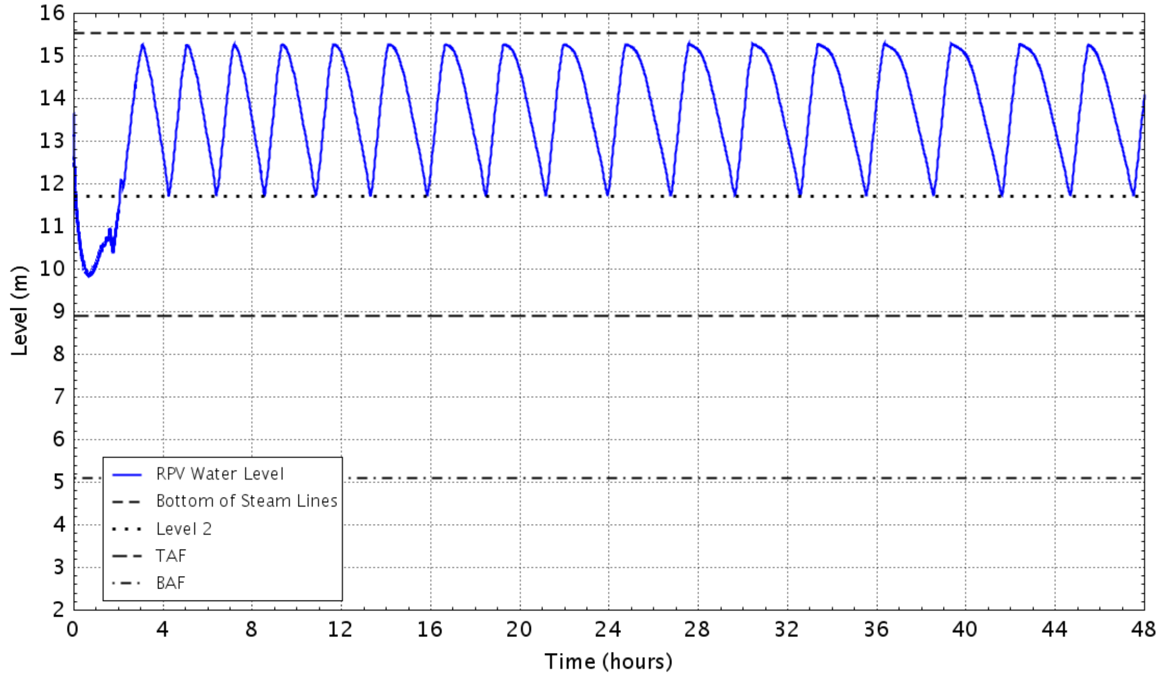

Figure G - 42 RPV Downcomer water level

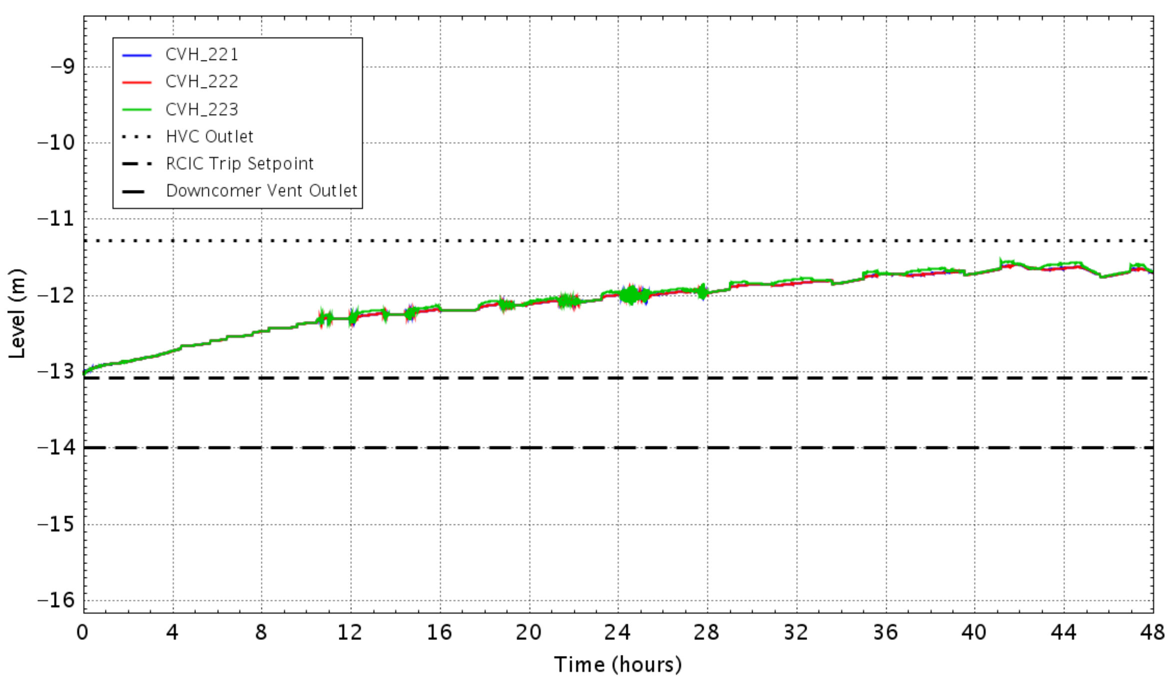

Figure G - 43 Water level in the wetwell

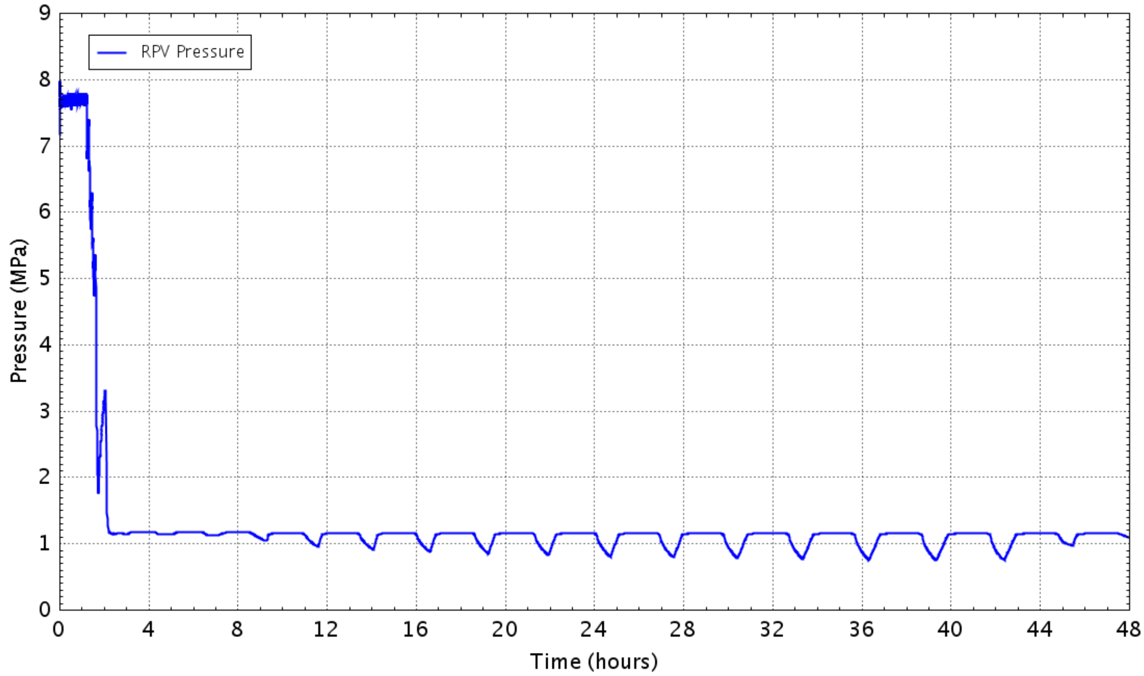

Figure G - 44 Pressure in the RPV

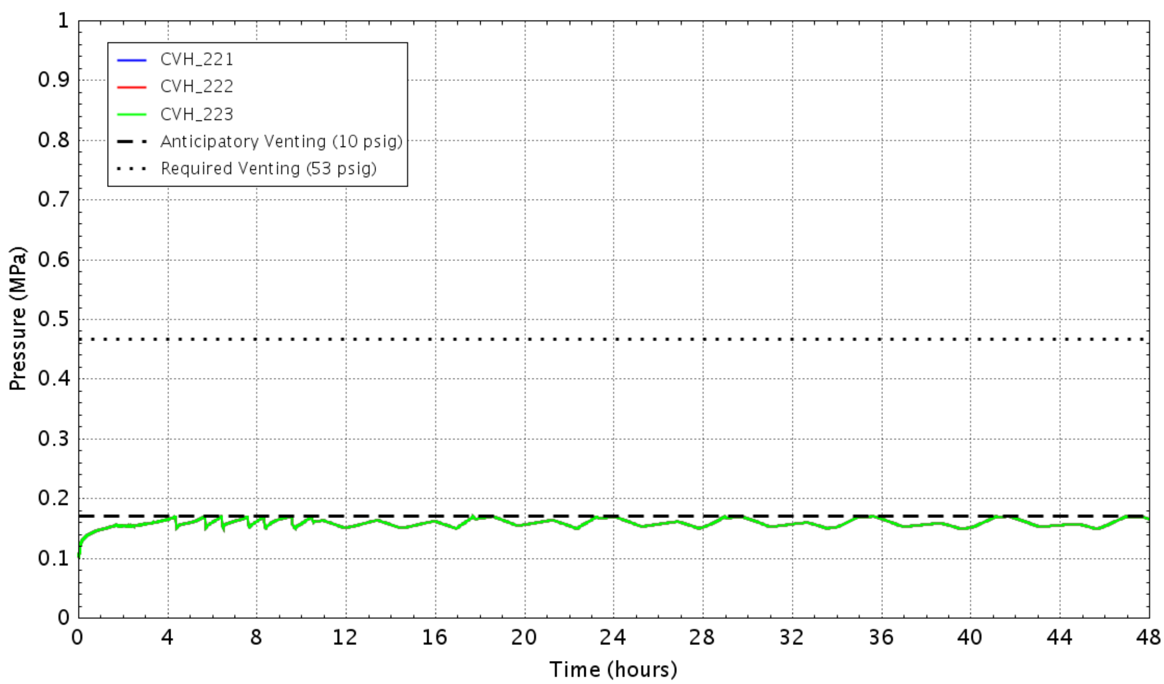

Figure G - 45 Pressure in the wetwell

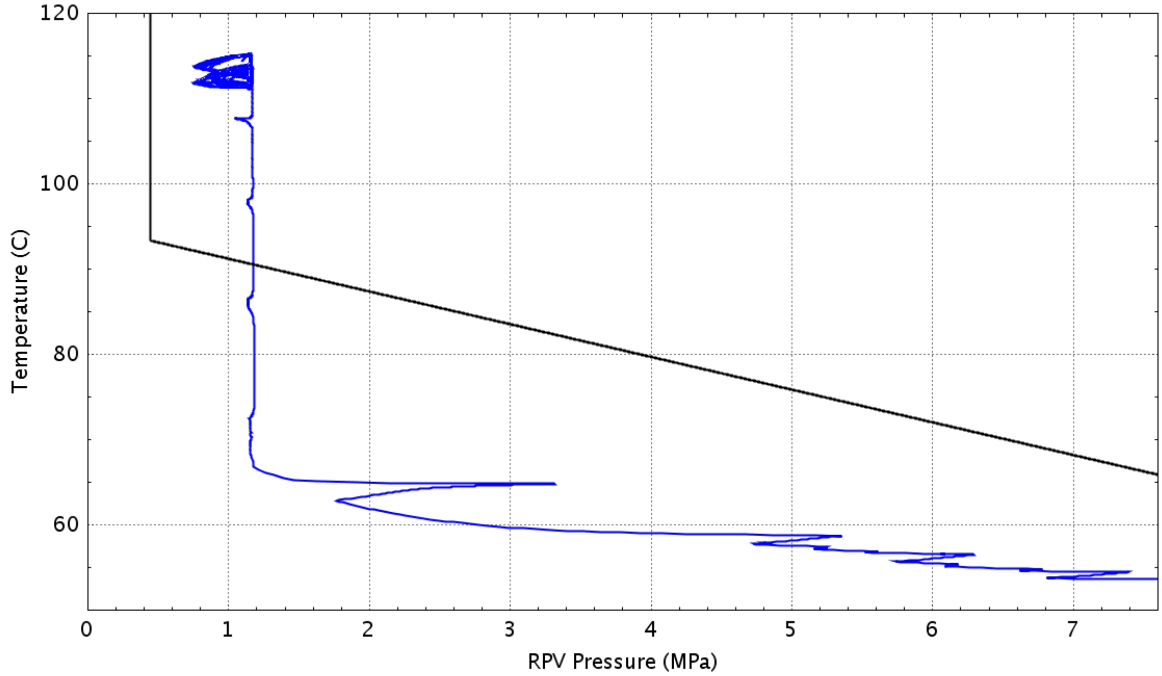

Figure G - 46 Plant status relative to the HCL curve (Graph 4 of the EOPs)

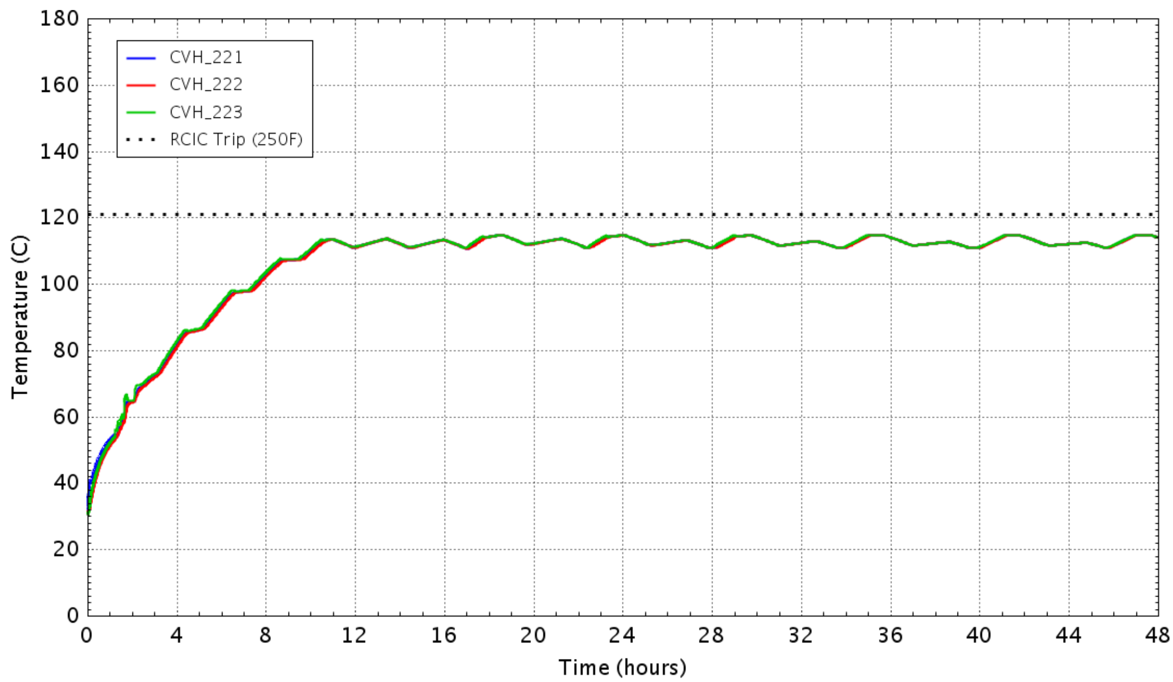

Figure G - 47 Water temperature in the wetwell

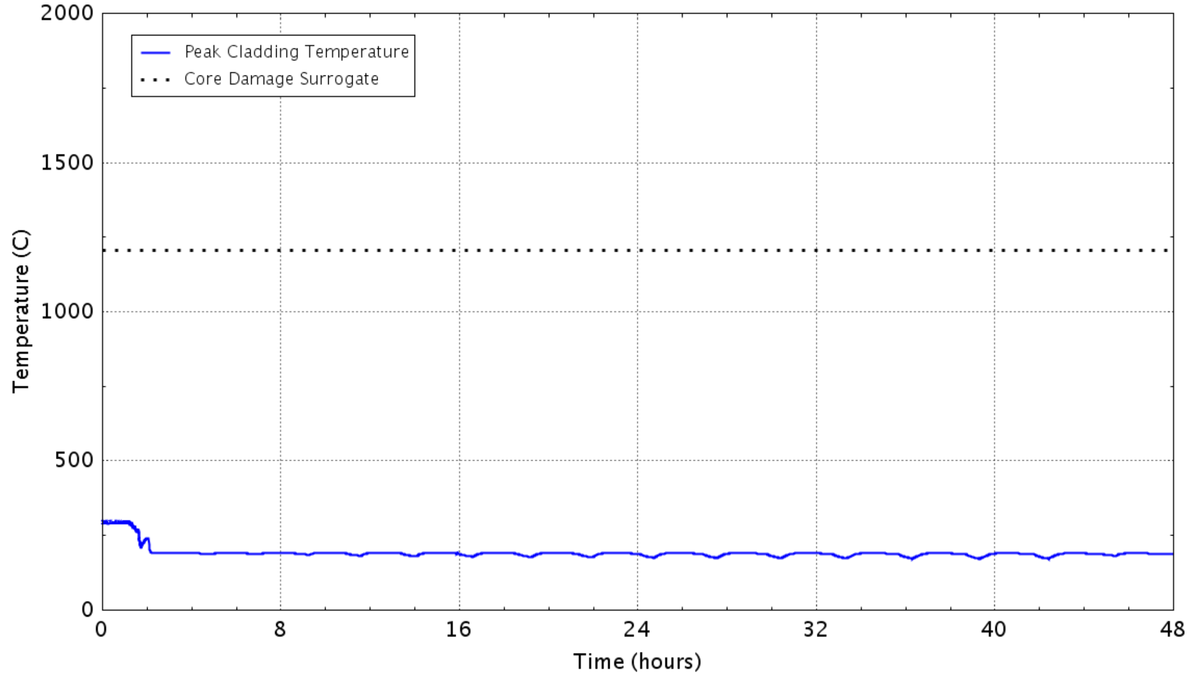

Figure G - 48 Peak temperature of the fuel cladding as a function of time

G.1.6 Case 5: LOOPGR-38-9, CST Unavailable, Initial Wetwell Level Decreased to 10.1 ft., Nominal Recirculation Pump Seal Leakage

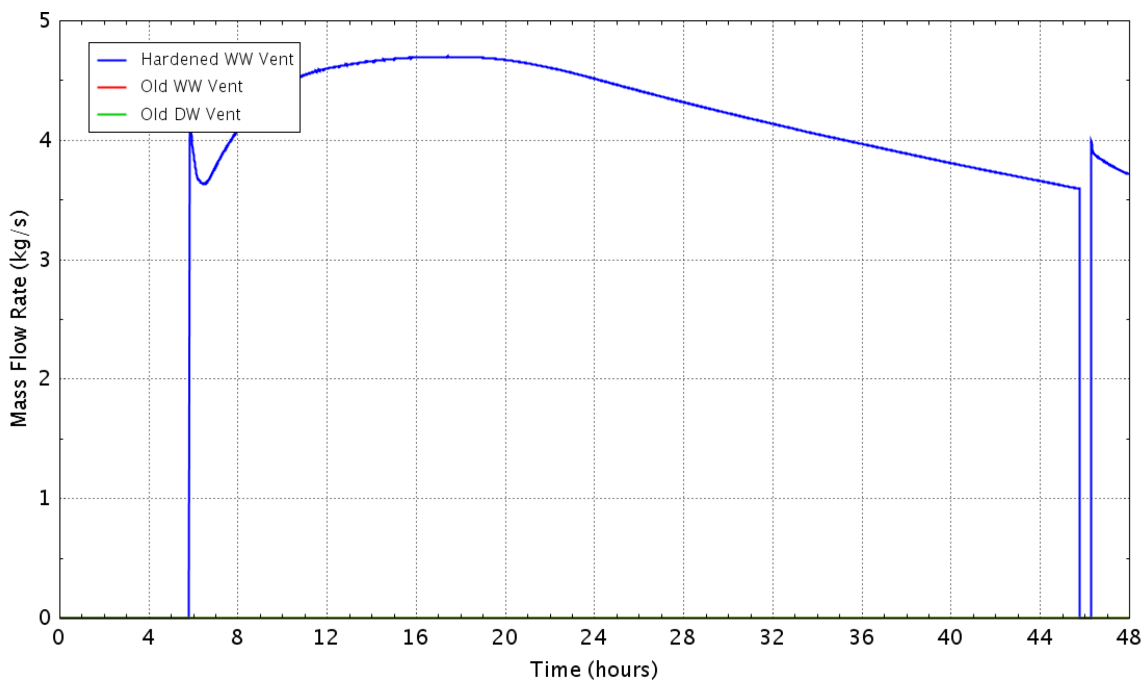

Figure G - 49 Flow rate of the containment vents

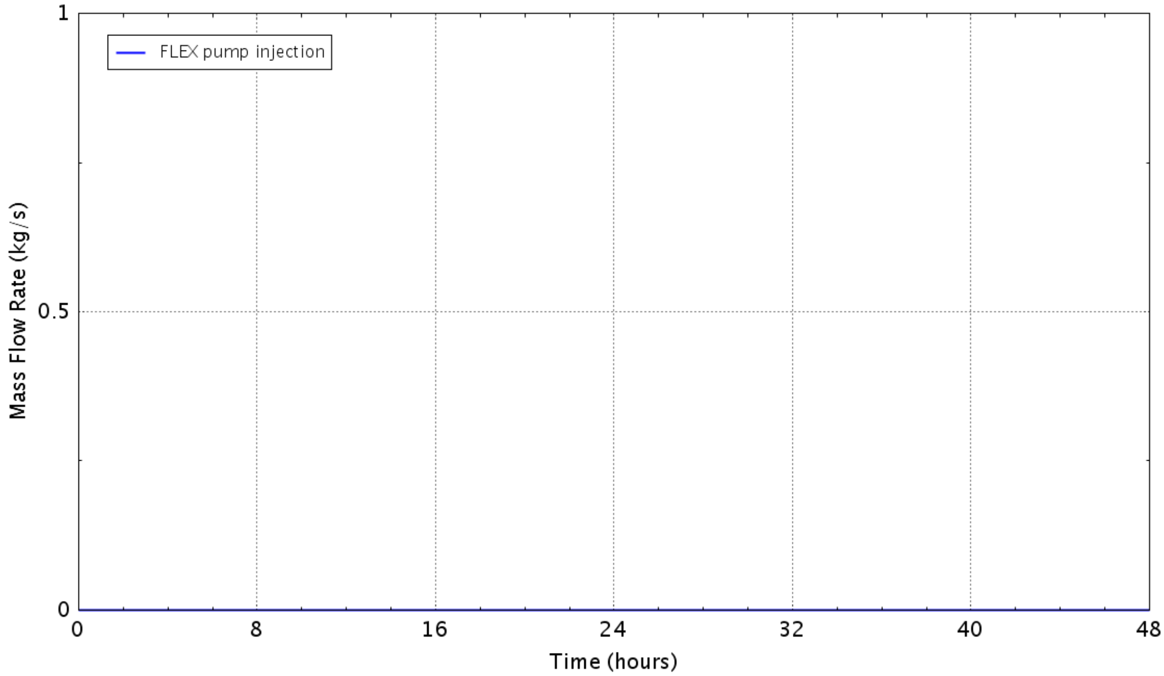

Figure G - 50 Flow rate of the FLEX pump

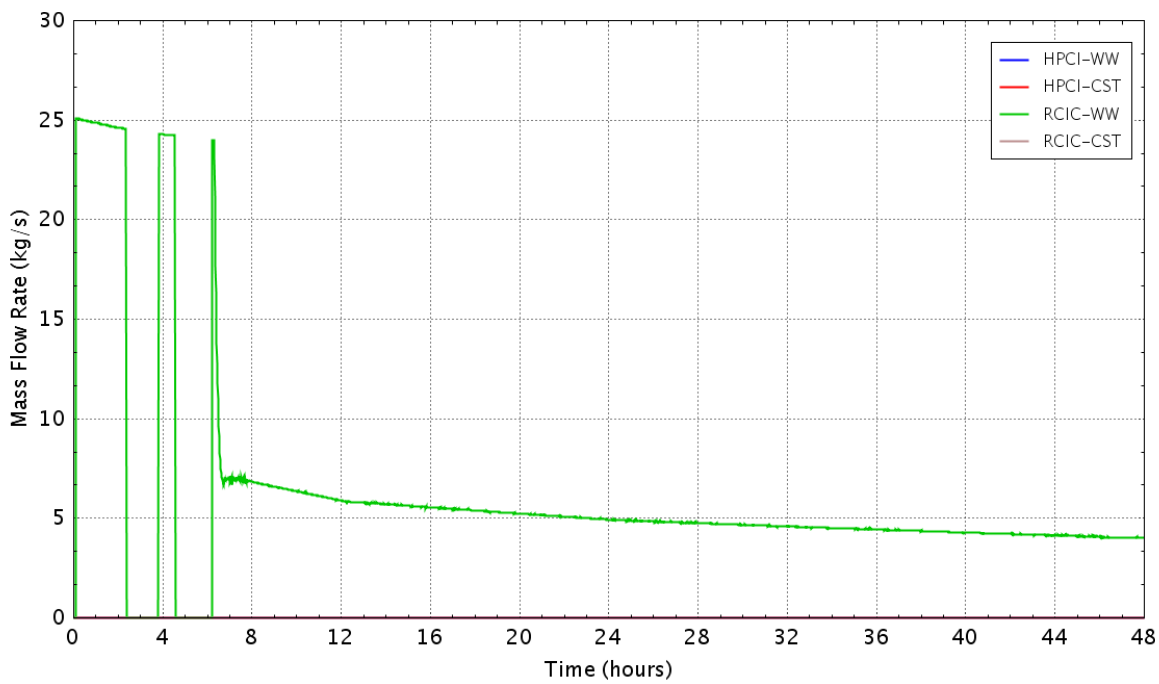

Figure G - 51 Flow rate of the HPCI/RCIC pumps

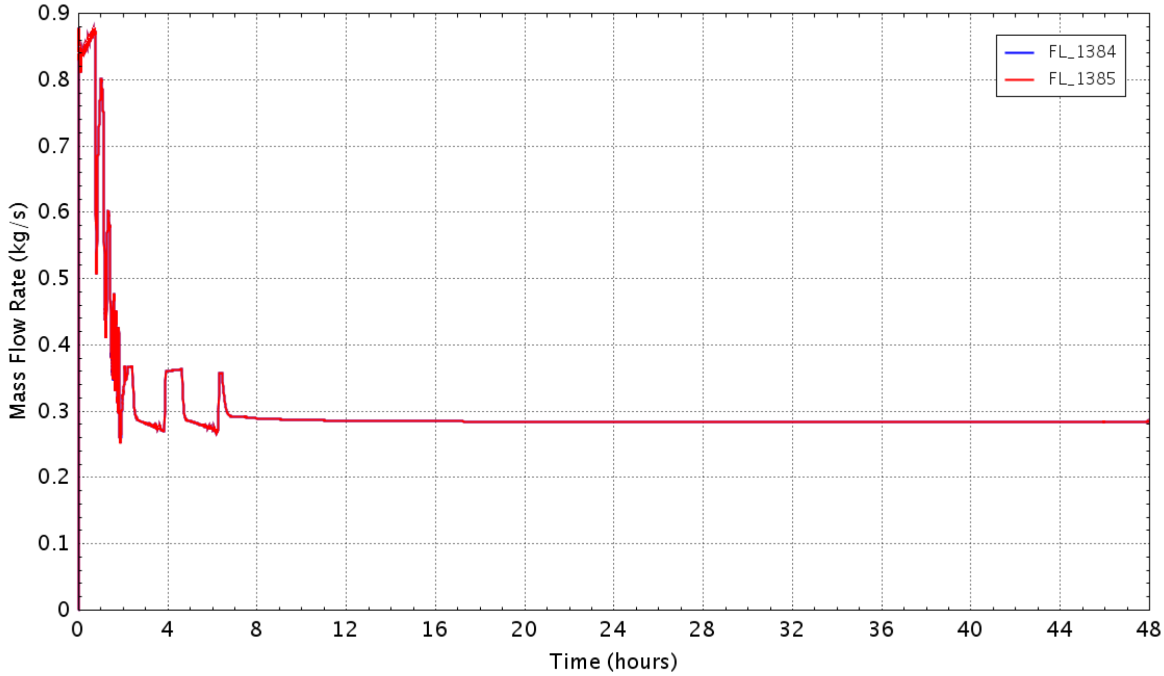

Figure G - 52 Flow rate of the recirculating pump seal leakage

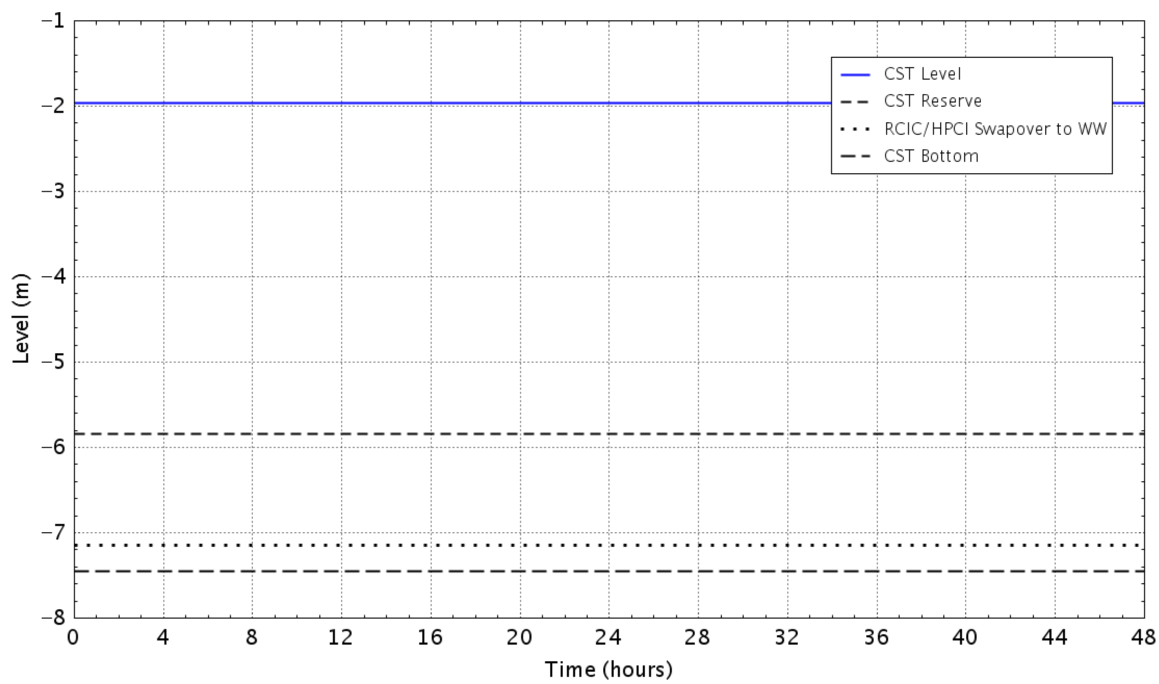

Figure G - 53 Water level in the CST

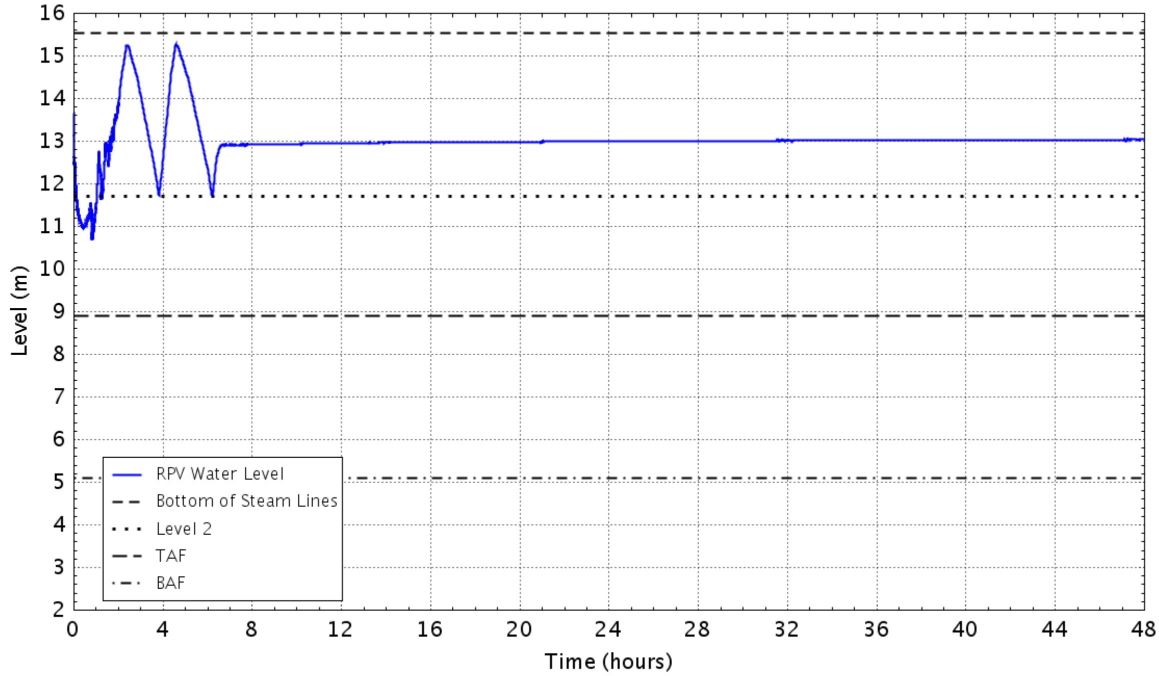

Figure G - 54 RPV Downcomer water level

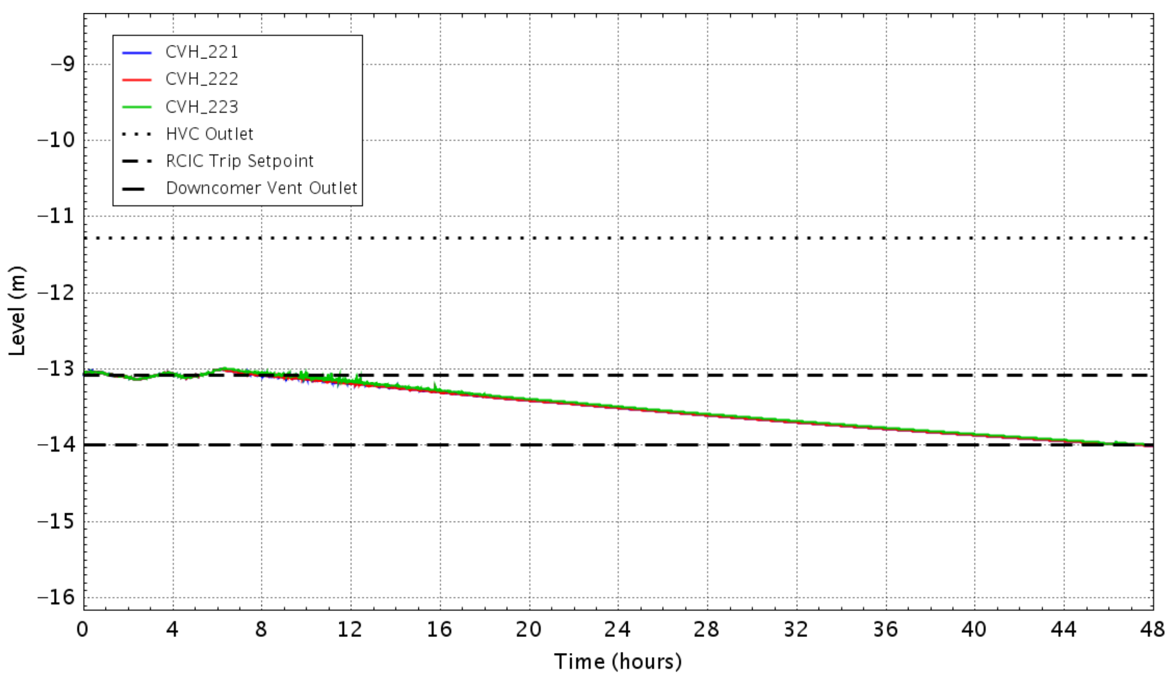

Figure G - 55 Water level in the wetwell

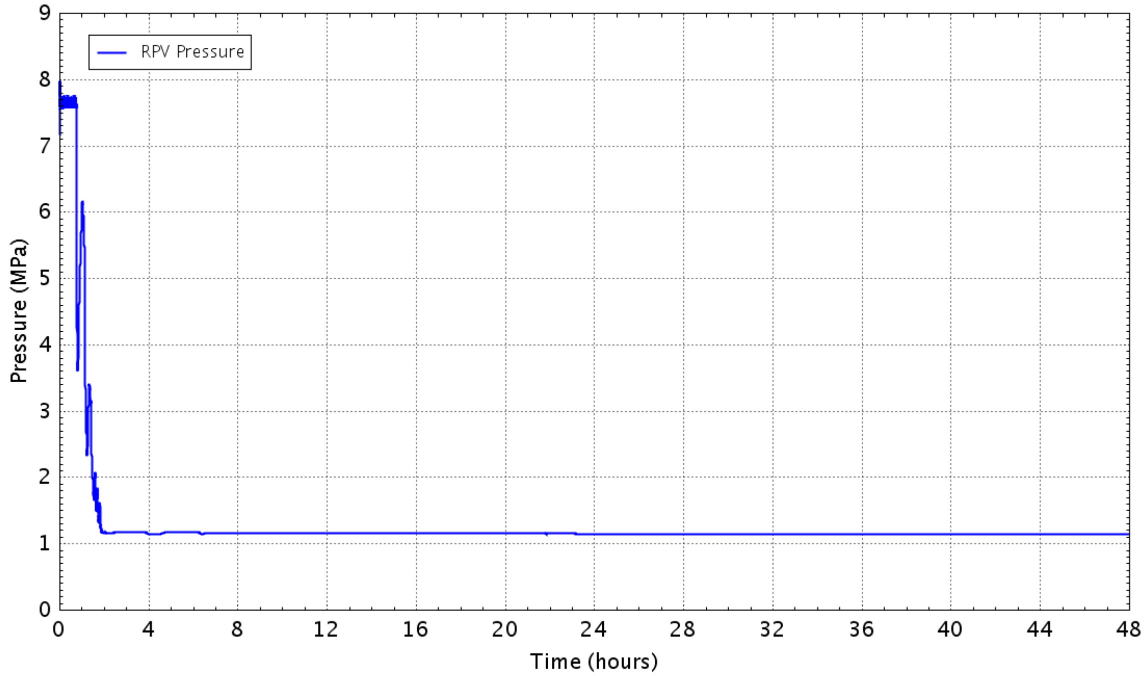

Figure G - 56 Pressure in the RPV

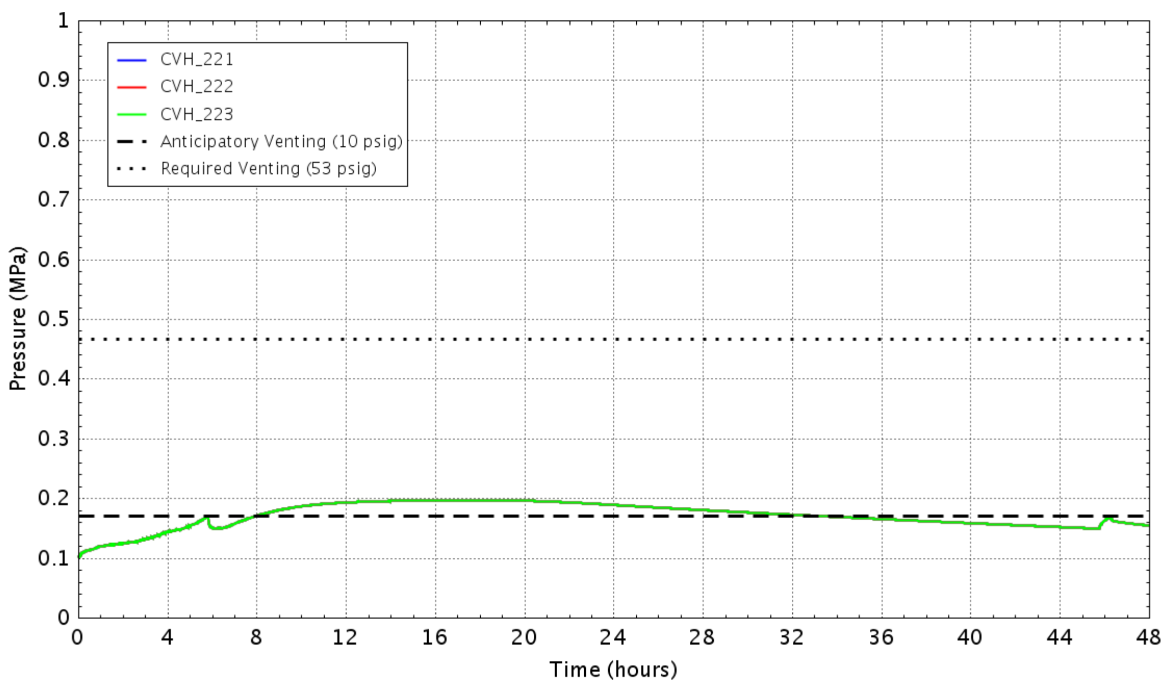

Figure G - 57 Pressure in the wetwell

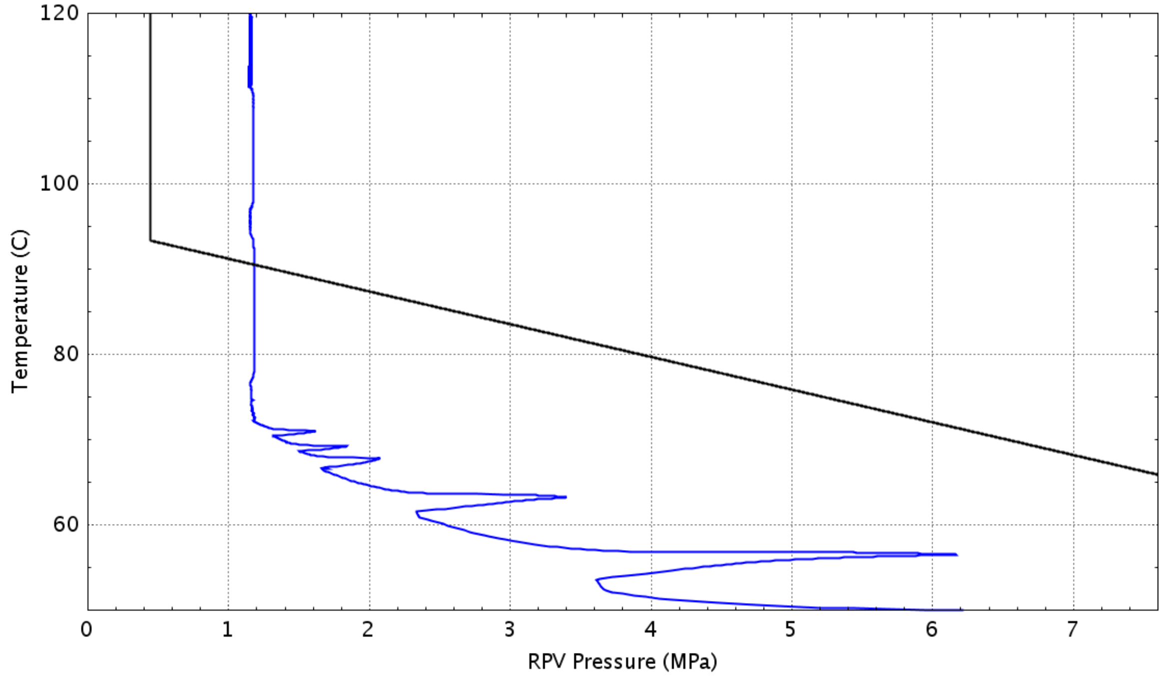

Figure G - 58 Plant status relative to the HCL curve (Graph 4 of the EOPs)

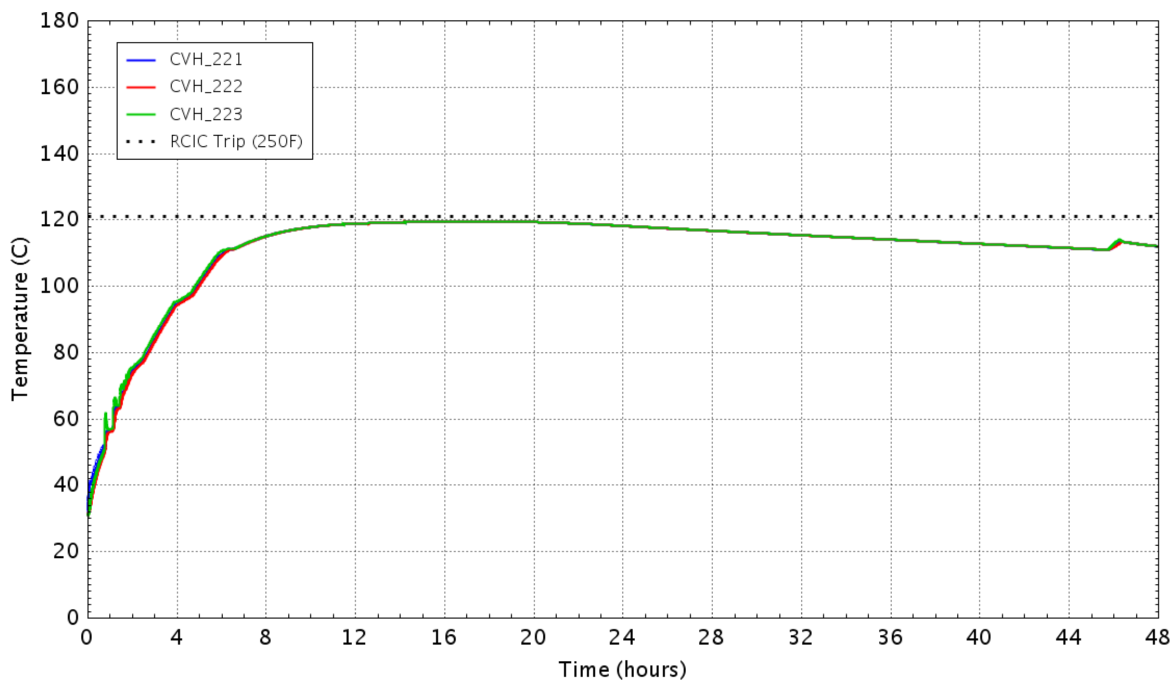

Figure G - 59 Water temperature in the wetwell

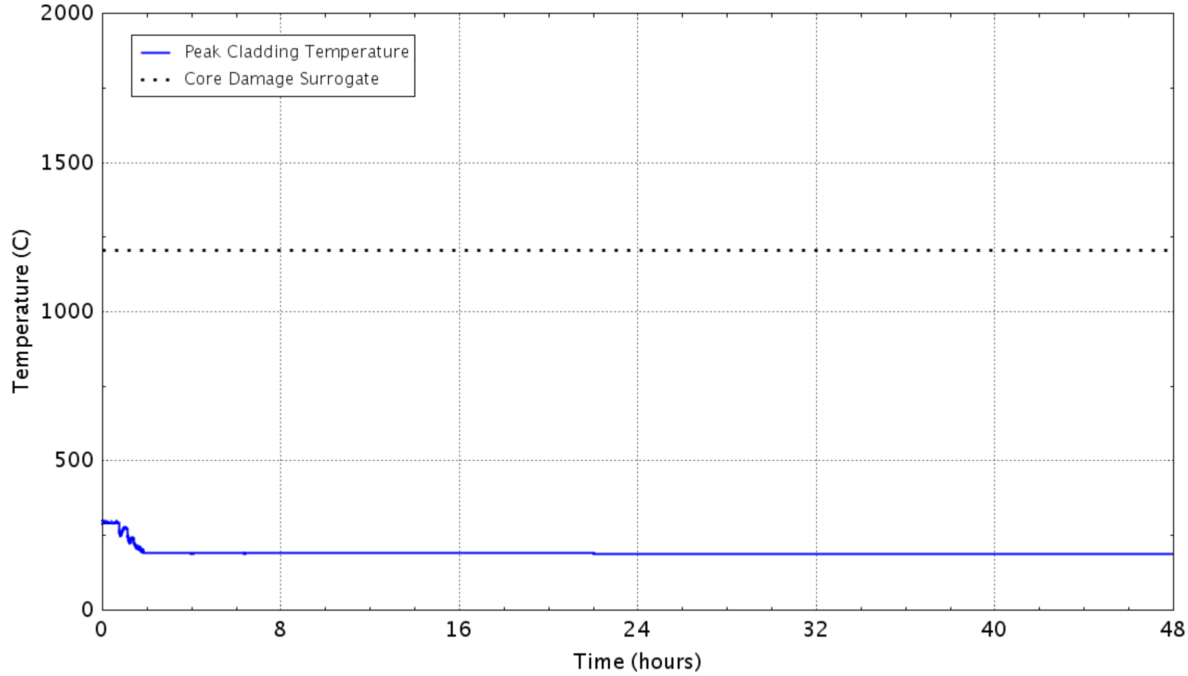

Figure G - 60 Peak temperature of the fuel cladding as a function of time

G.1.6 Case 6: LOOPGR-38-9, CST Unavailable, Initial Wetwell Level Decreased to10.1 ft., 200 gpm Recirculation Pump Seal Leakage

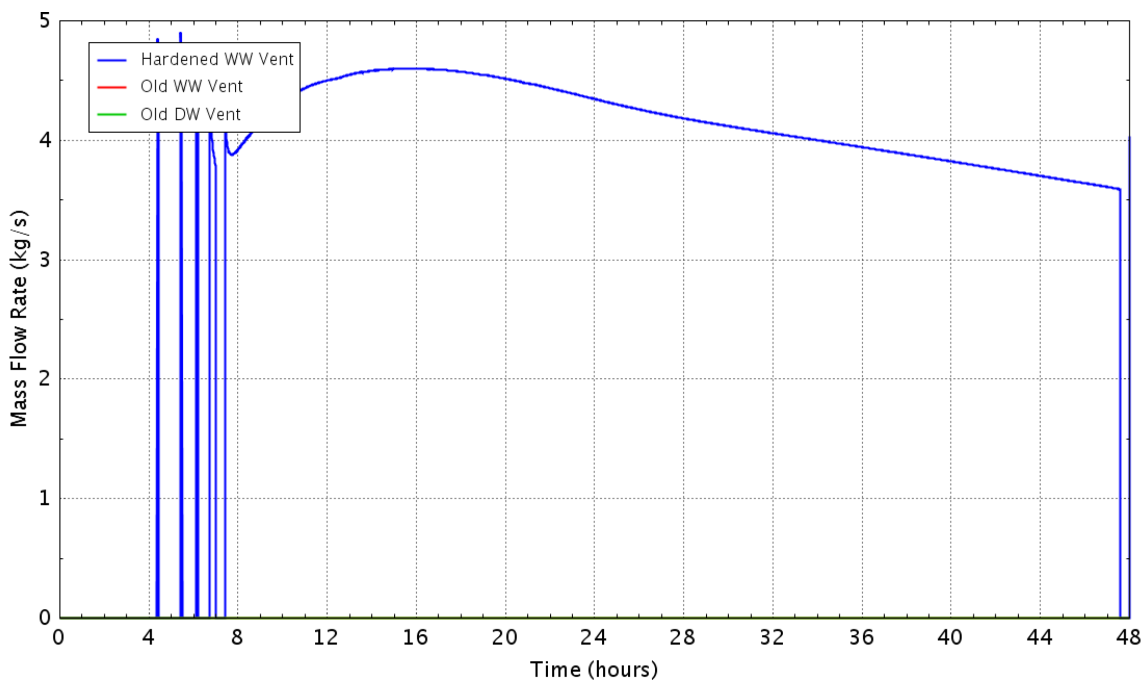

Figure G - 61 Flow rate of the containment vents

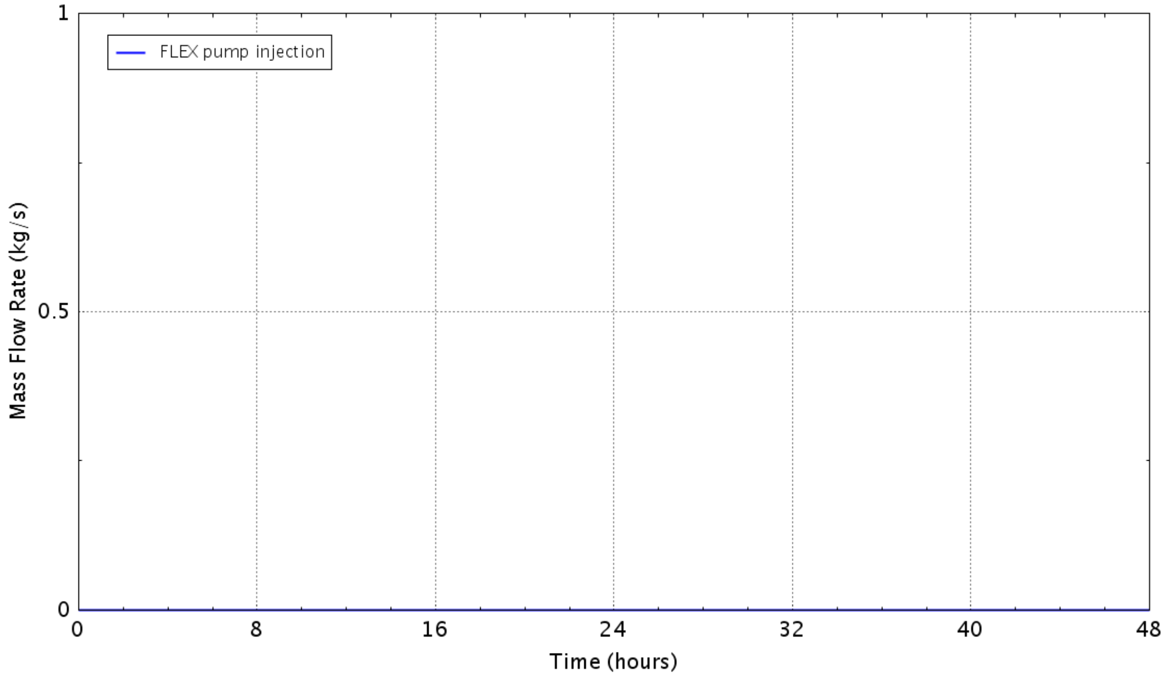

Figure G - 62 Flow rate of the FLEX pump

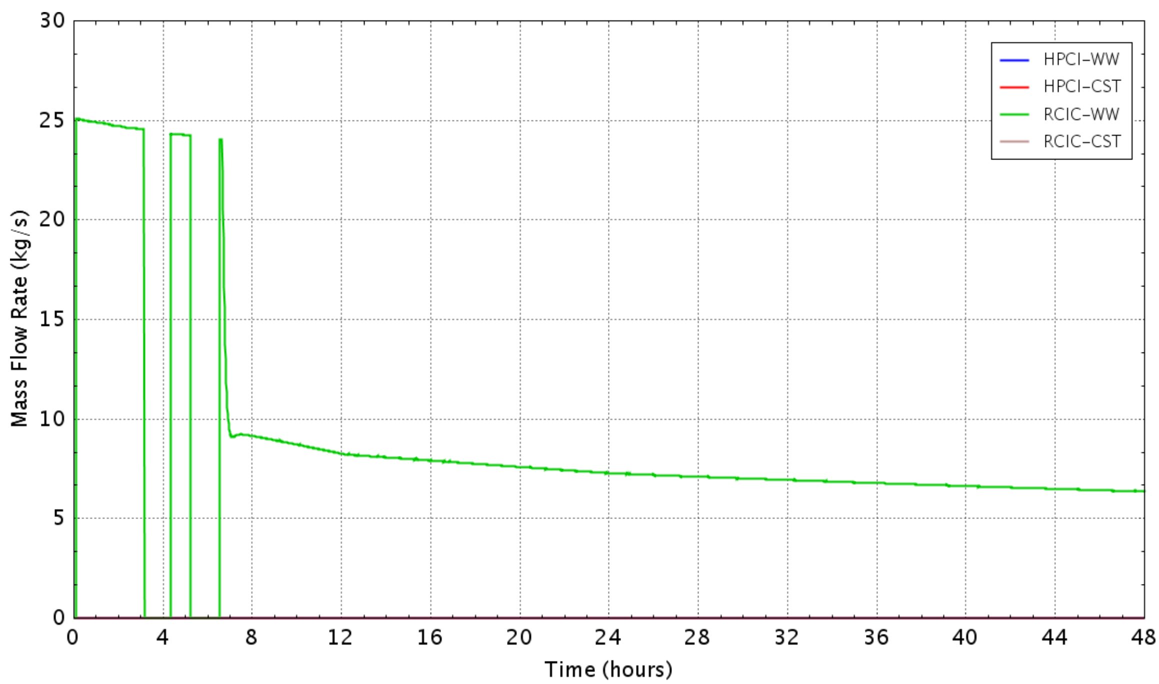

Figure G - 63 Flow rate of the HPCI/RCIC pumps

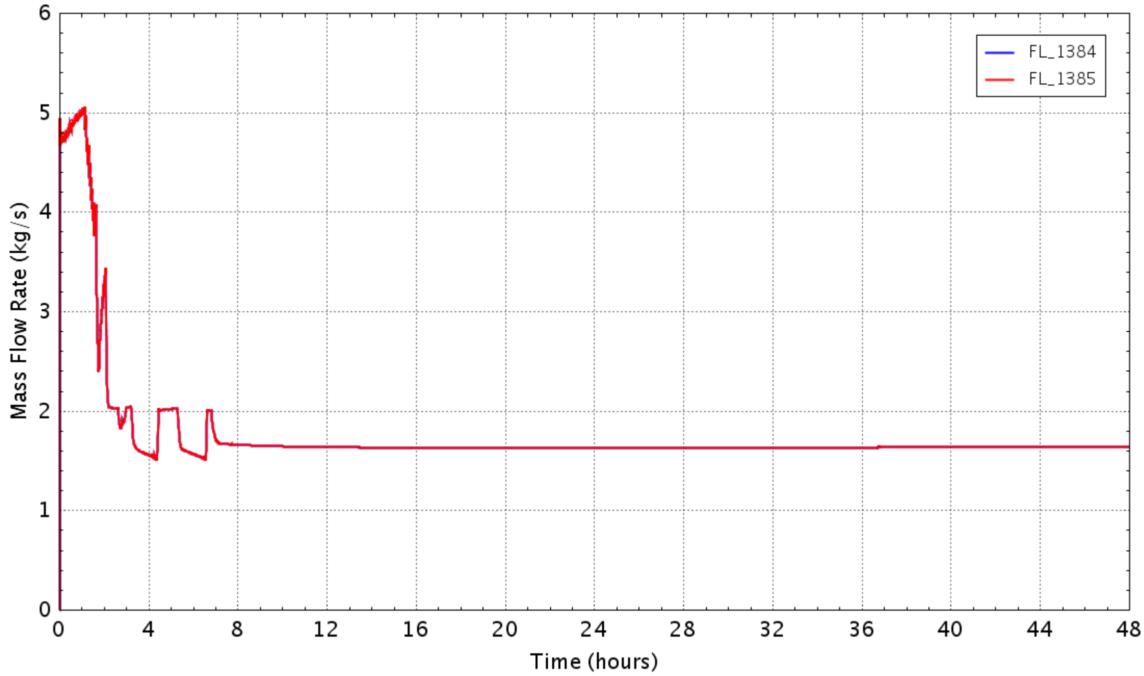

Figure G - 64 Flow rate of the recirculating pump seal leakage

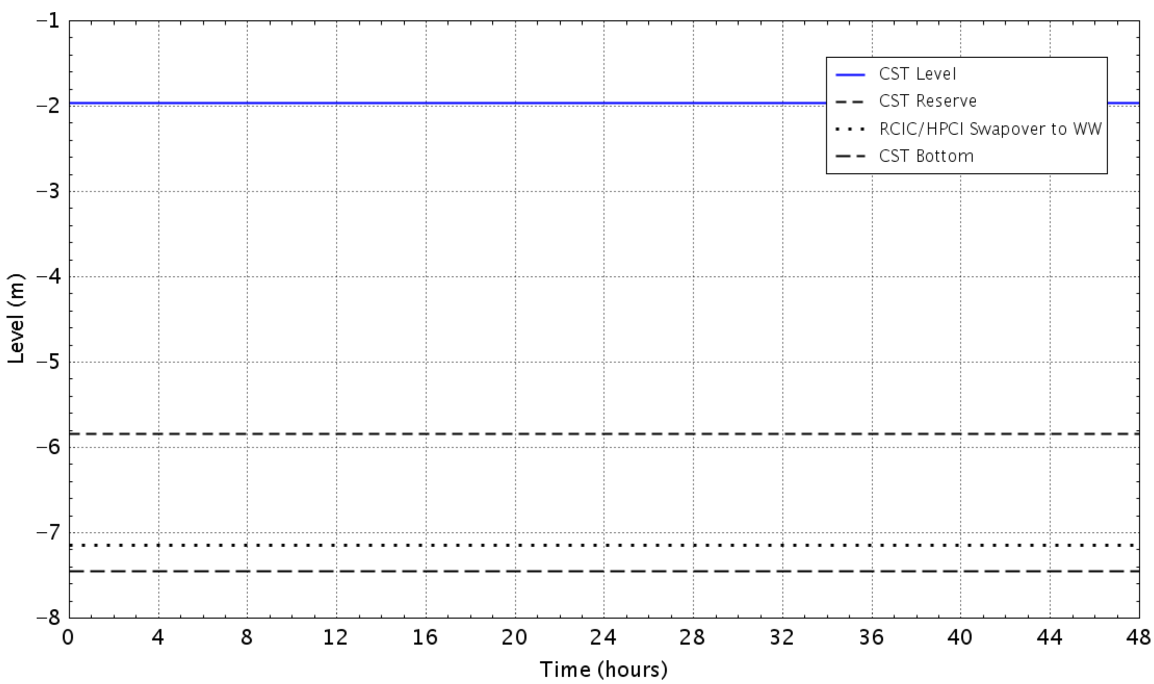

Figure G - 65 Water level in the CST

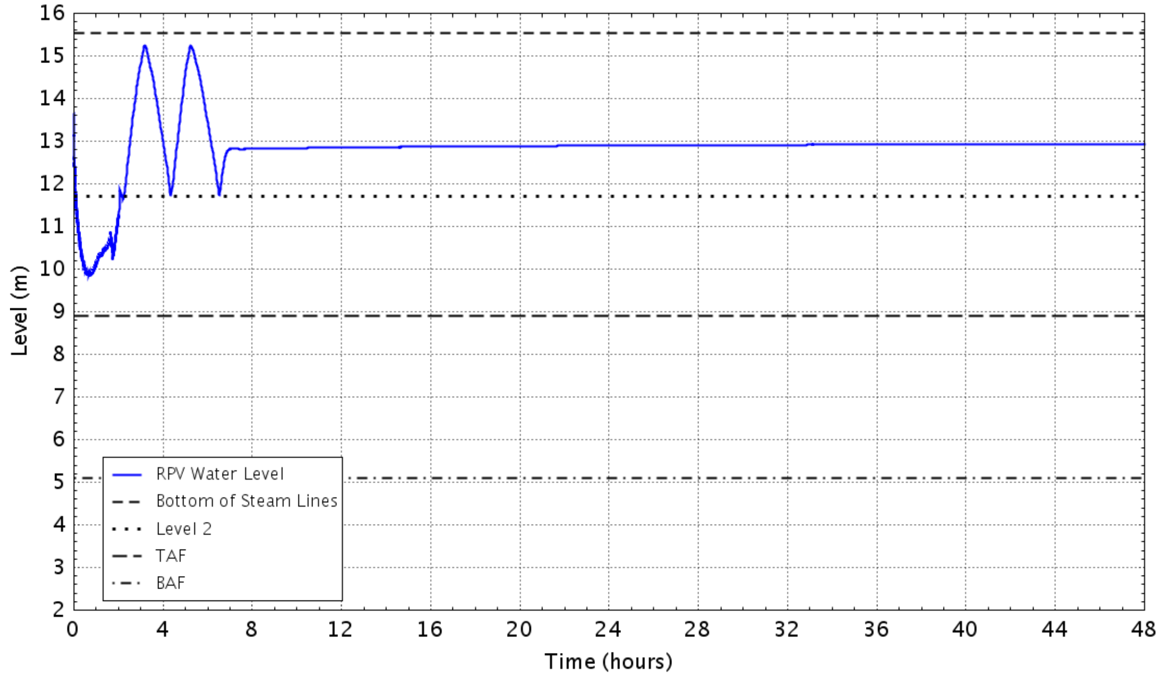

Figure G - 66 RPV Downcomer water level

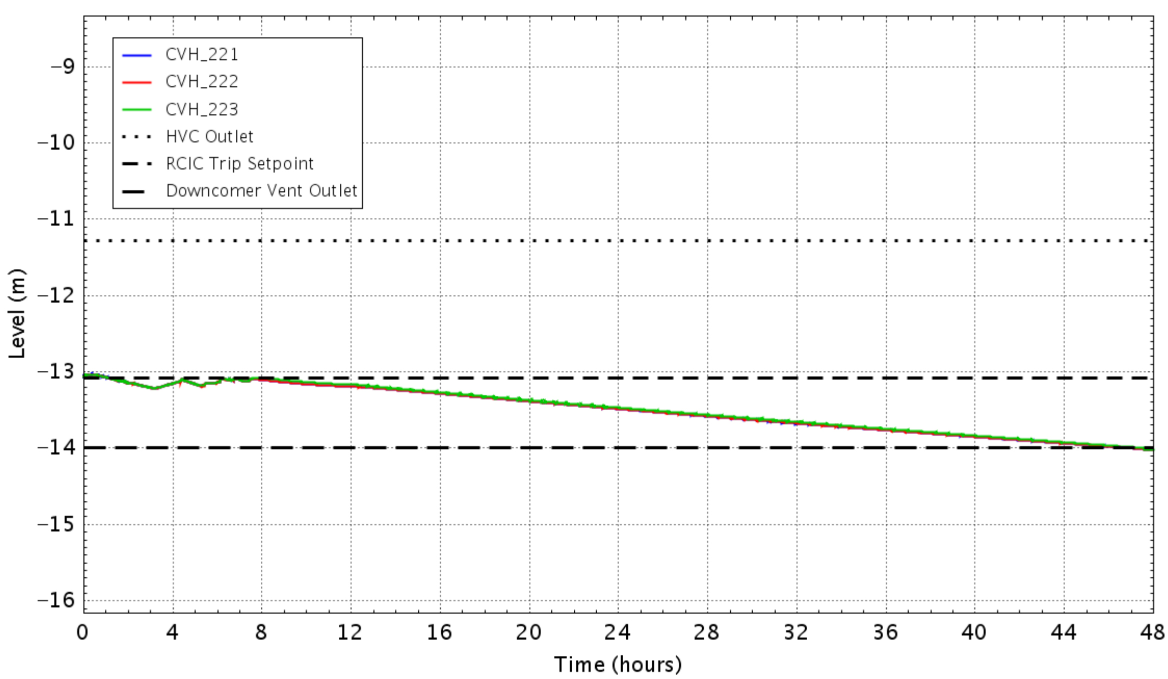

Figure G - 67 Water level in the wetwell

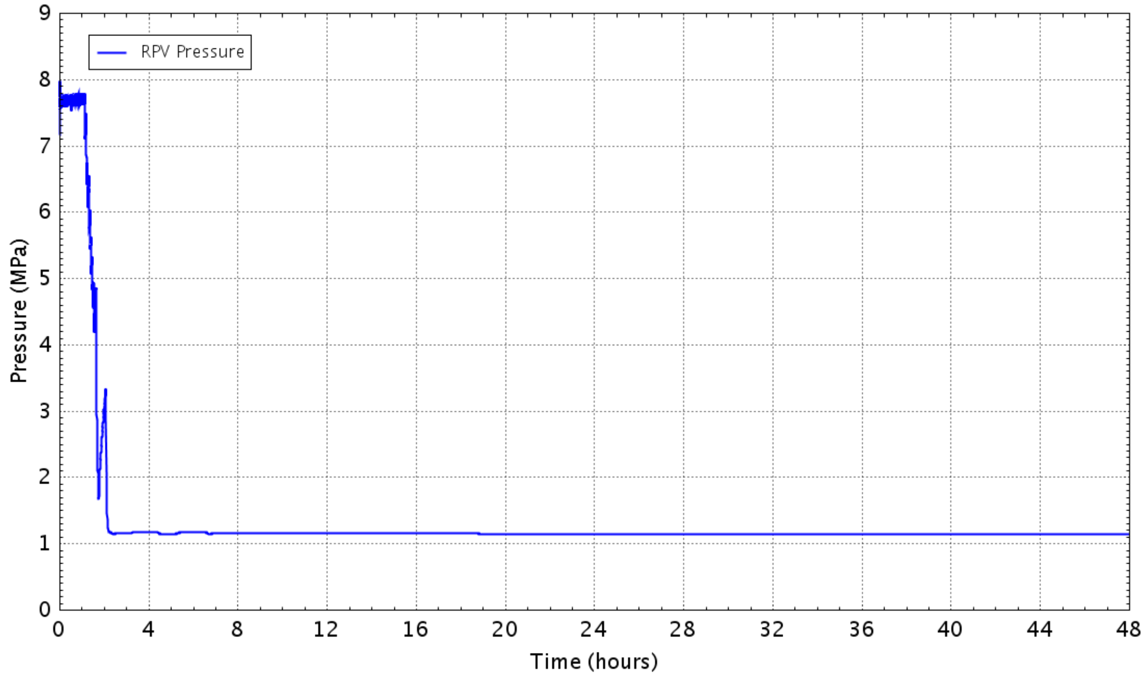

Figure G - 68 Pressure in the RPV

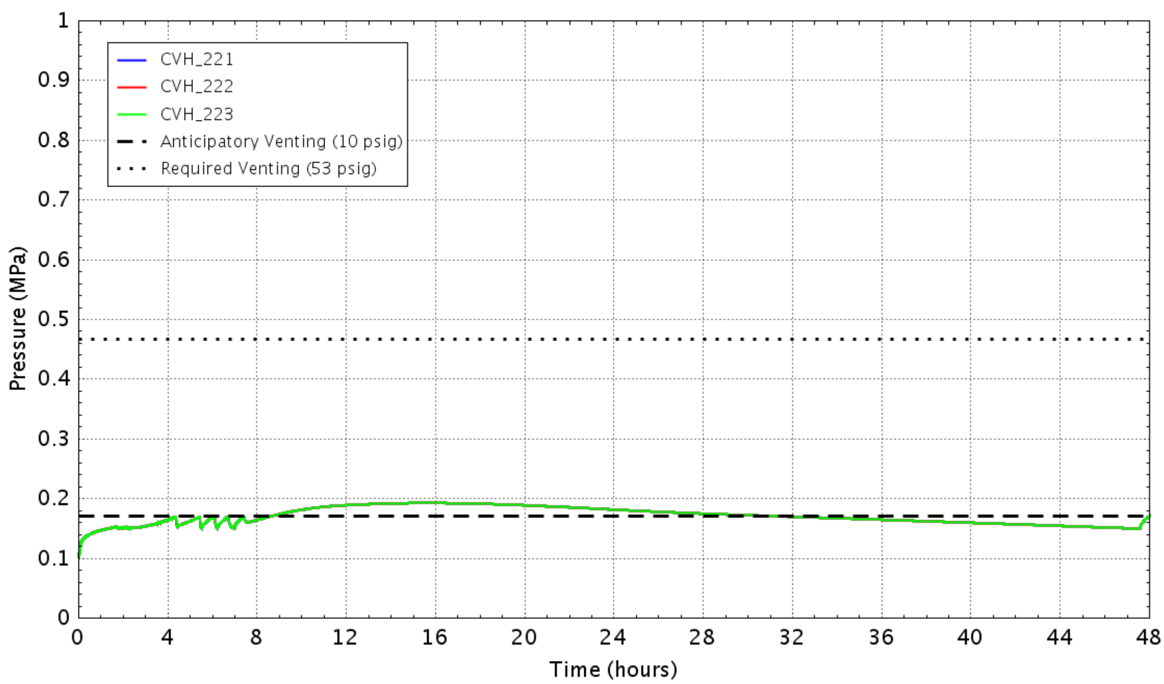

Figure G - 69 Pressure in the wetwell

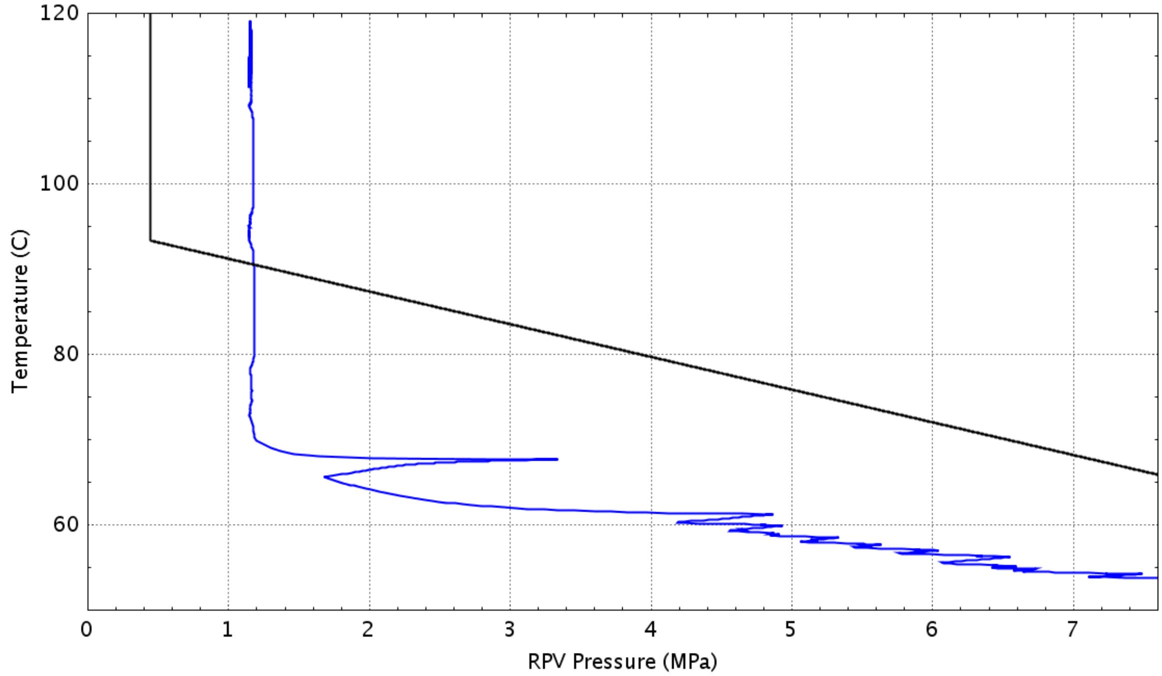

Figure G - 70 Plant status relative to the HCL curve (Graph 4 of the EOPs)

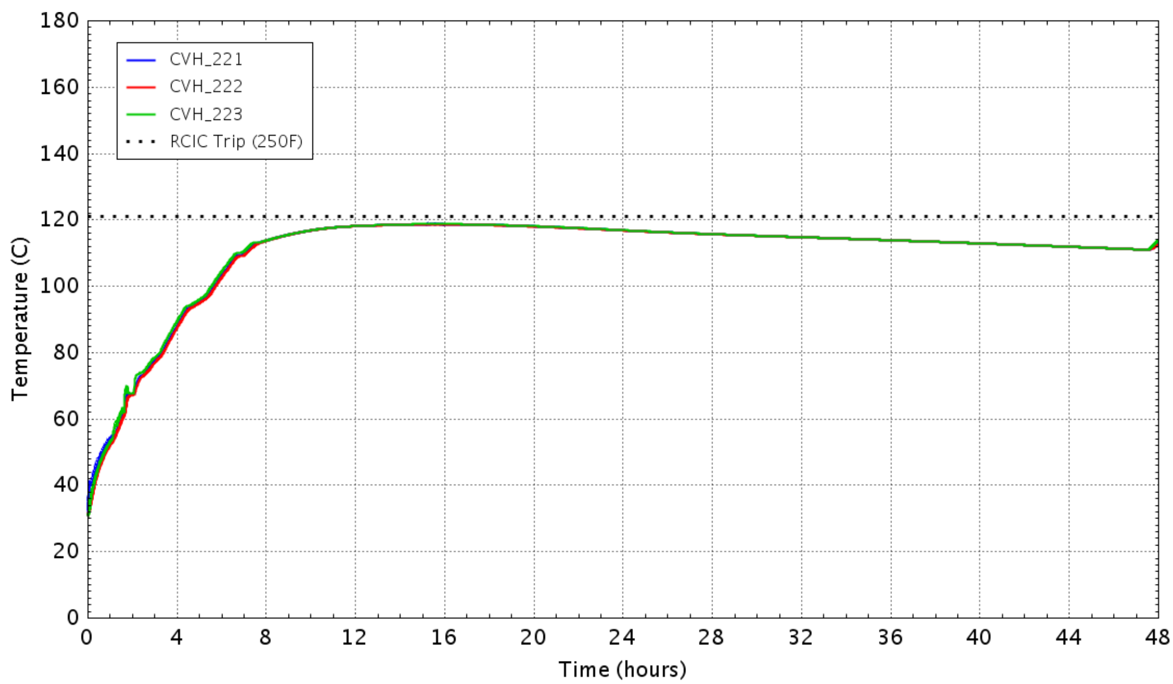

Figure G - 71 Water temperature in the wetwell

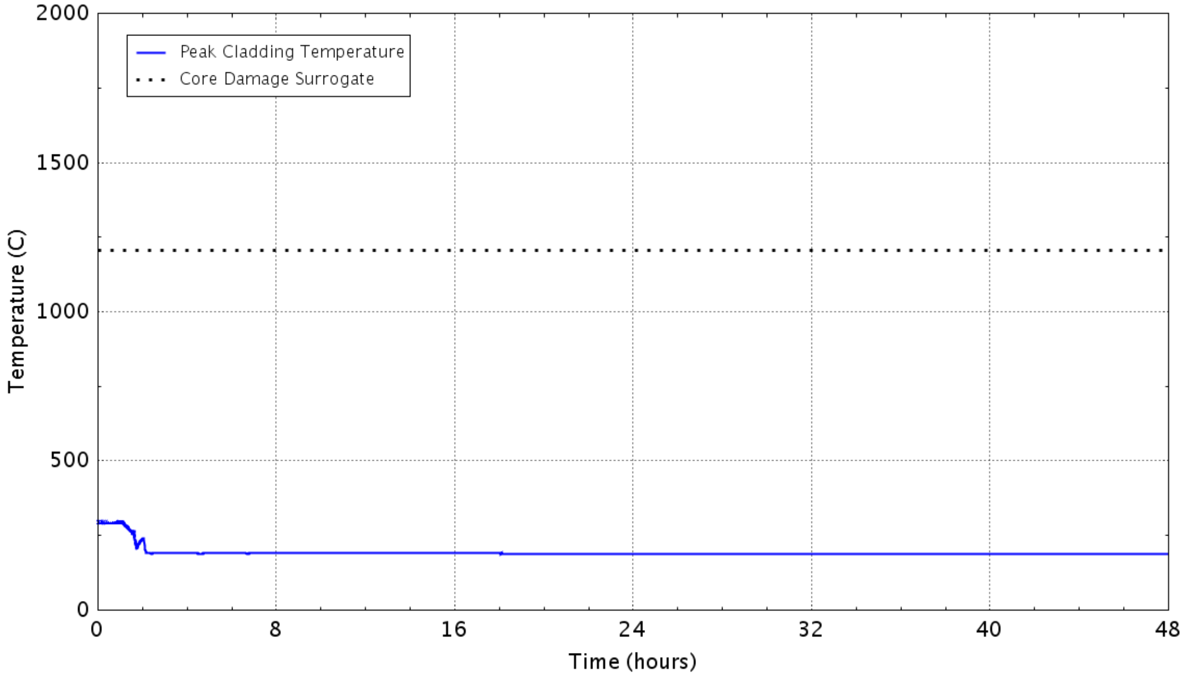

Figure G - 72 Peak temperature of the fuel cladding as a function of time

G.1.7 Case 7: LOOPGR-38-9, CST Unavailable, Initial Wetwell Level Decreased to 10.4 ft., Nominal Recirculation Pump Seal Leakage

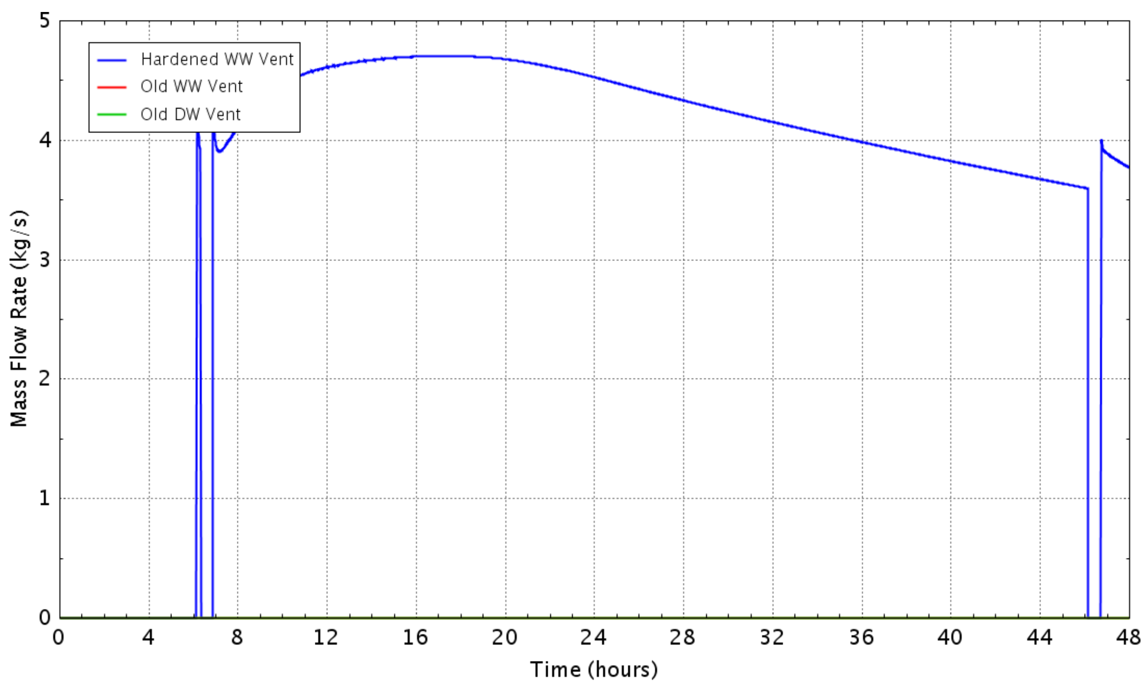

Figure G - 73 Flow rate of the containment vents

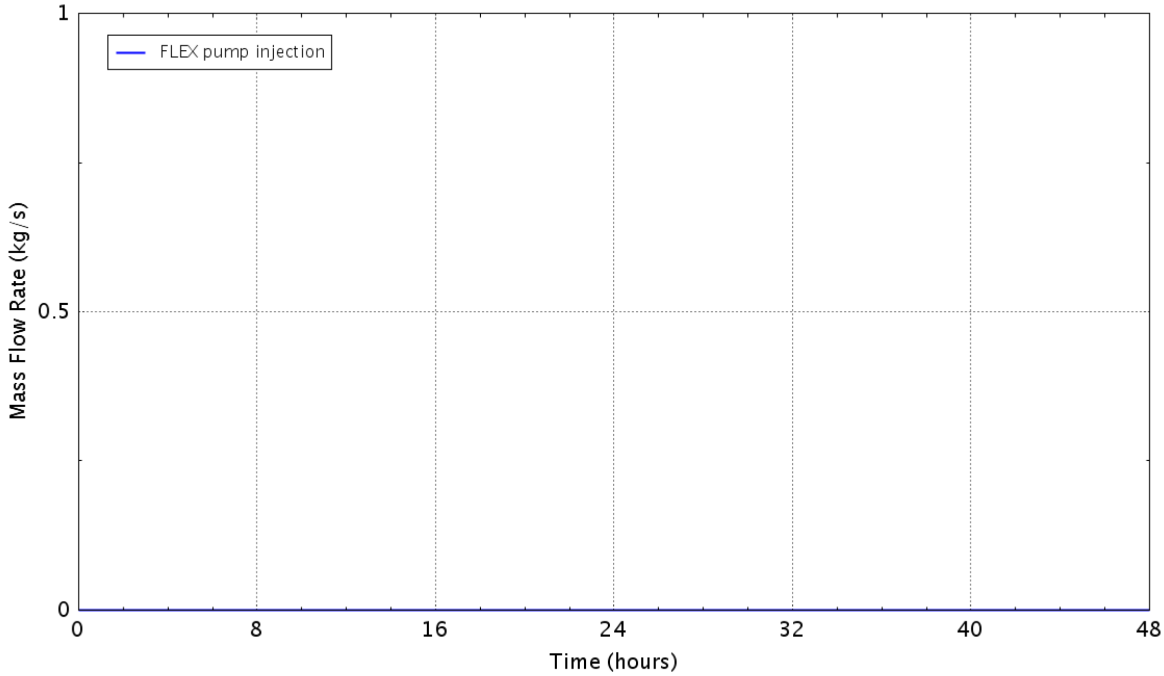

Figure G - 74 Flow rate of the FLEX pump

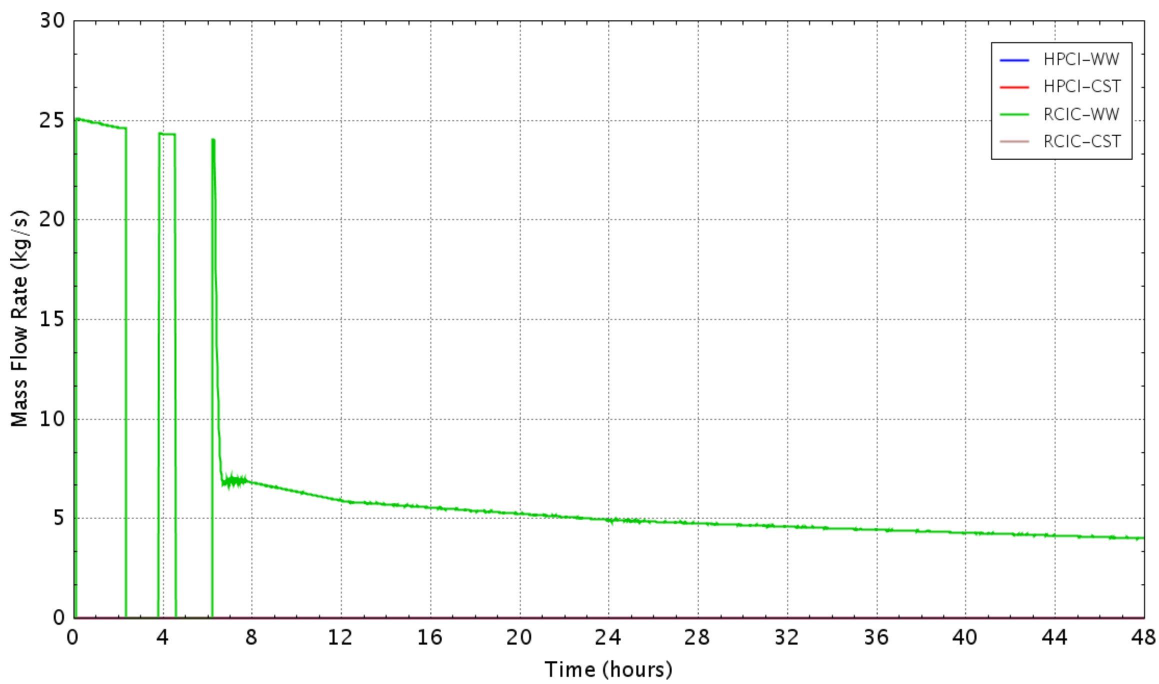

Figure G - 75 Flow rate of the HPCI/RCIC pumps

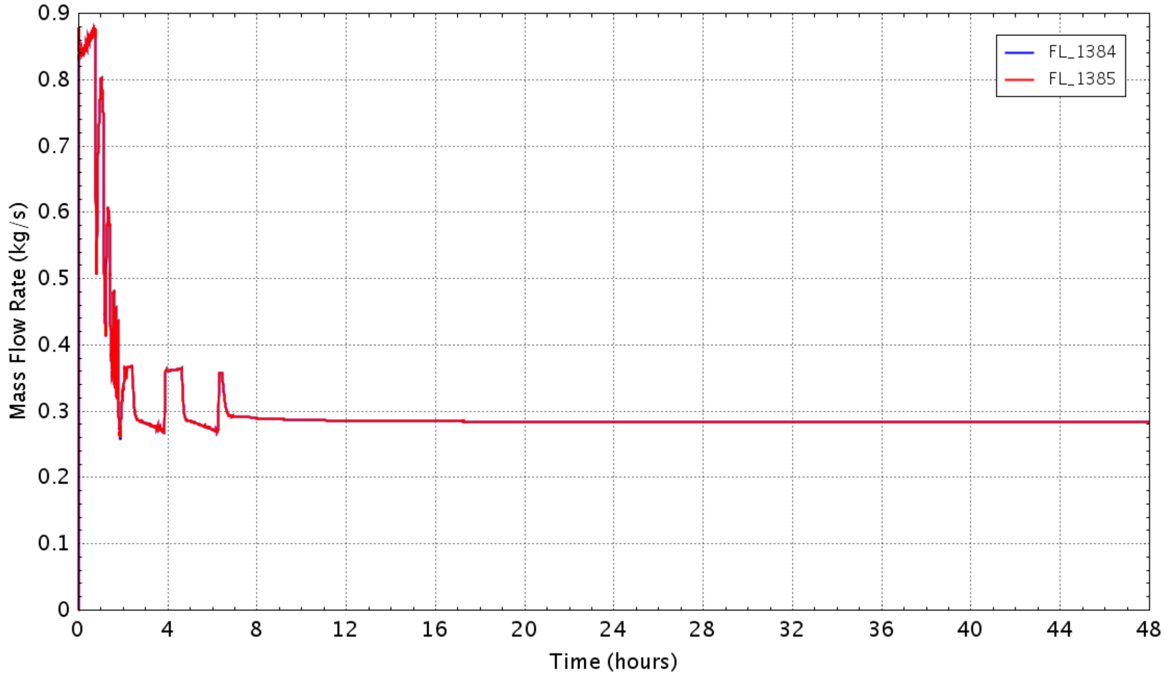

Figure G - 76 Flow rate of the recirculating pump seal leakage

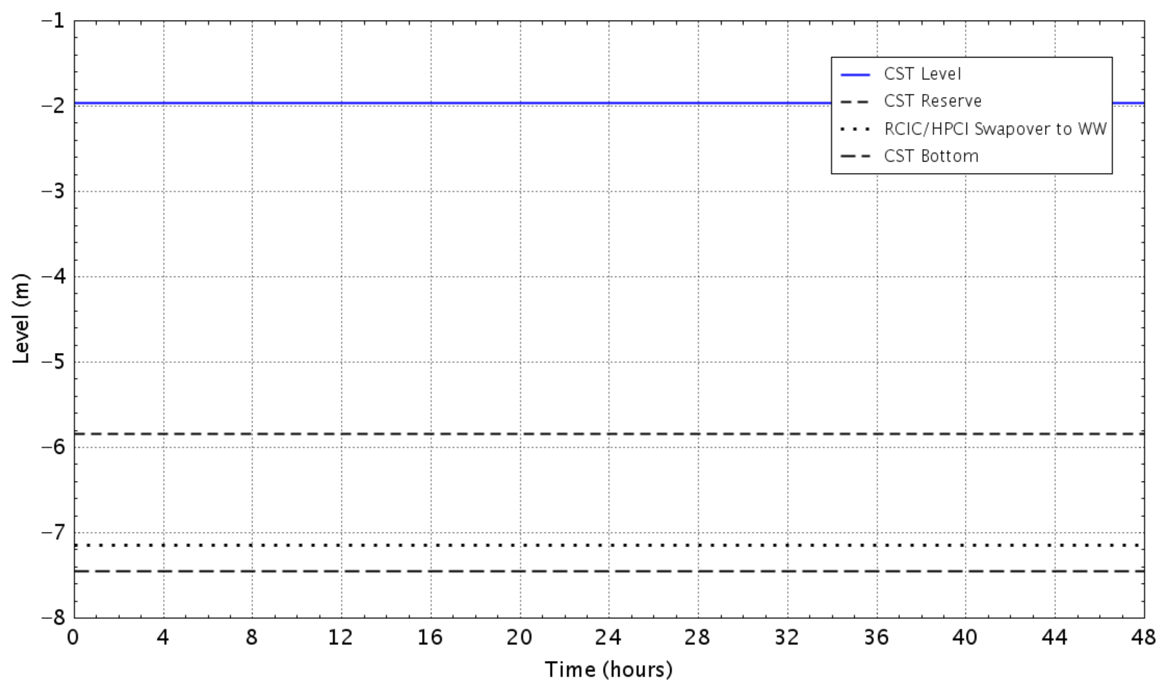

Figure G - 77 Water level in the CST

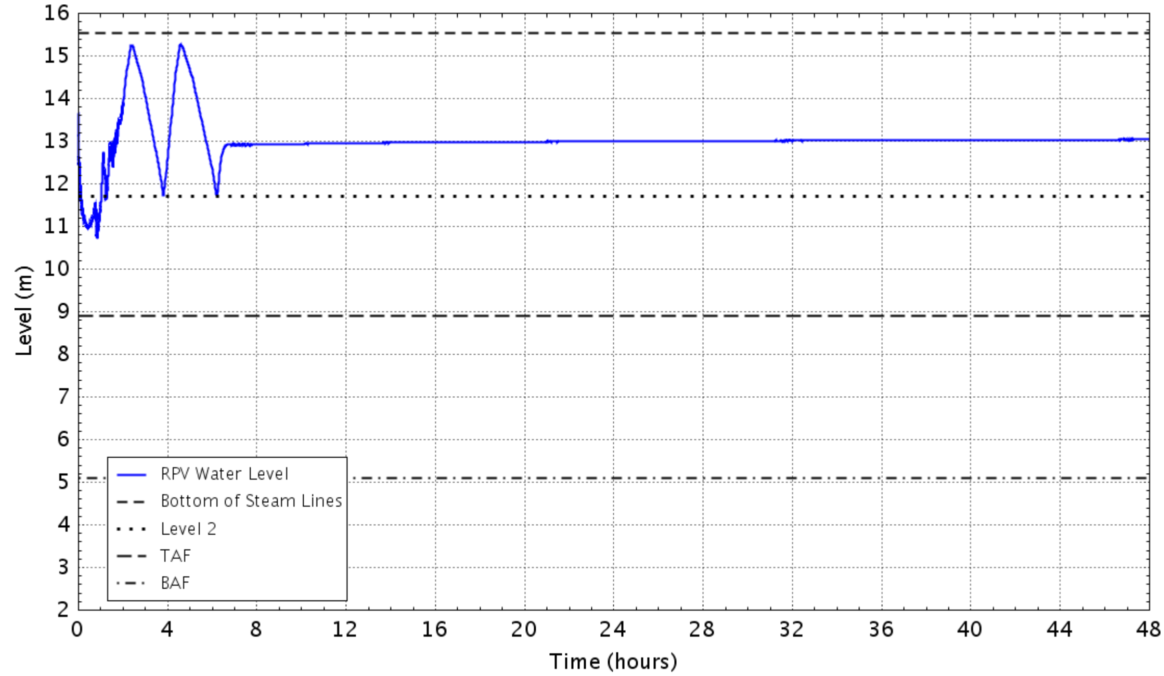

Figure G - 78 RPV Downcomer water level

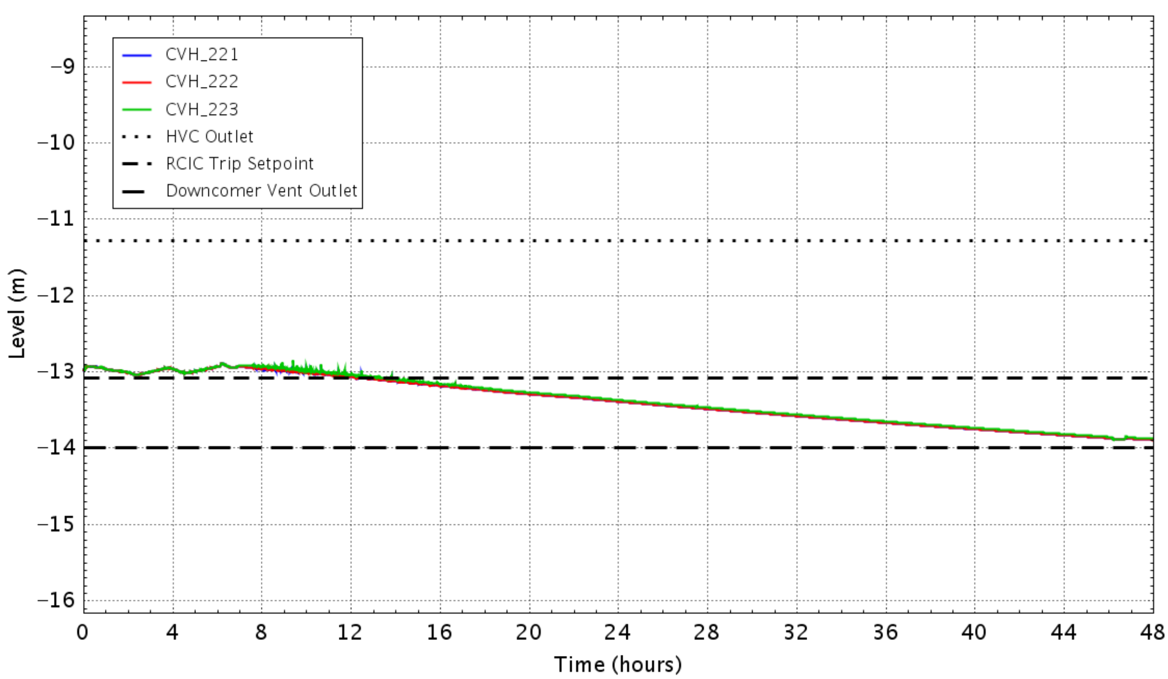

Figure G – 79 Water level in the wetwell

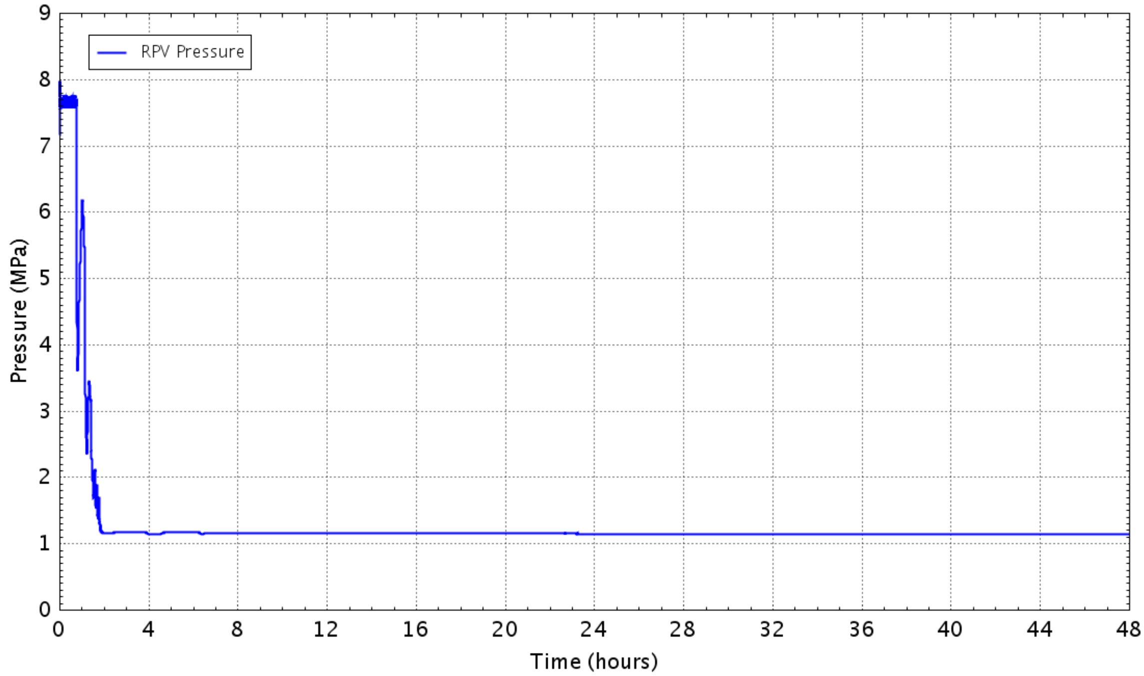

Figure G - 80 Pressure in the RPV

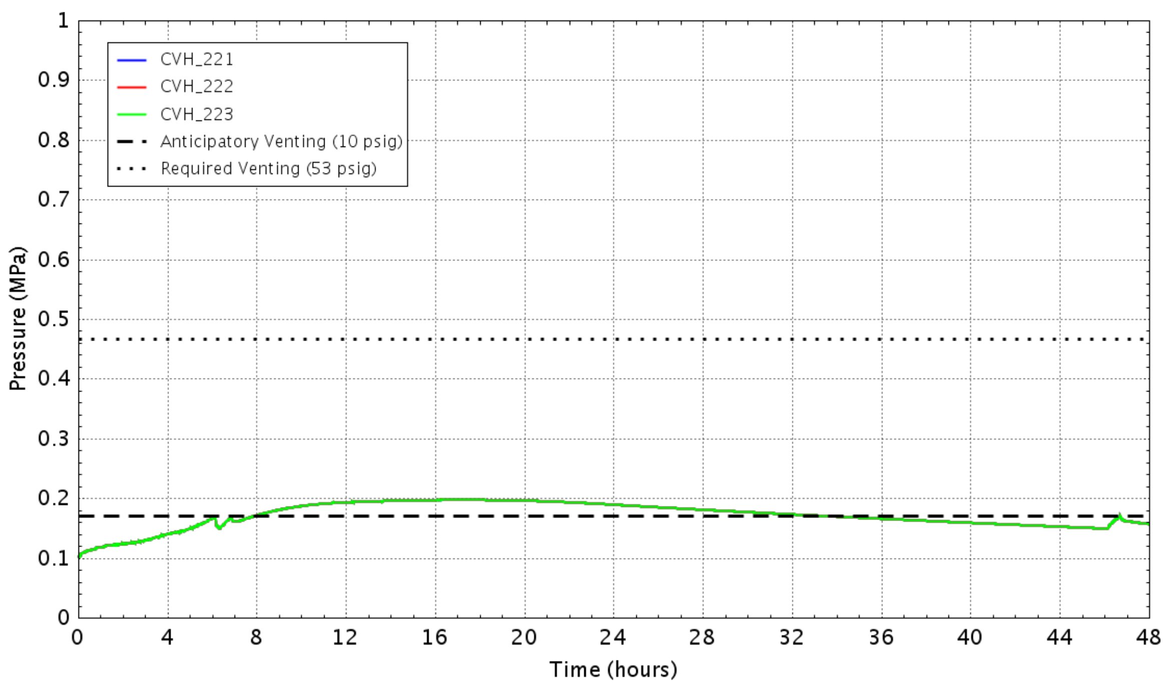

Figure G - 81 Pressure in the wetwell

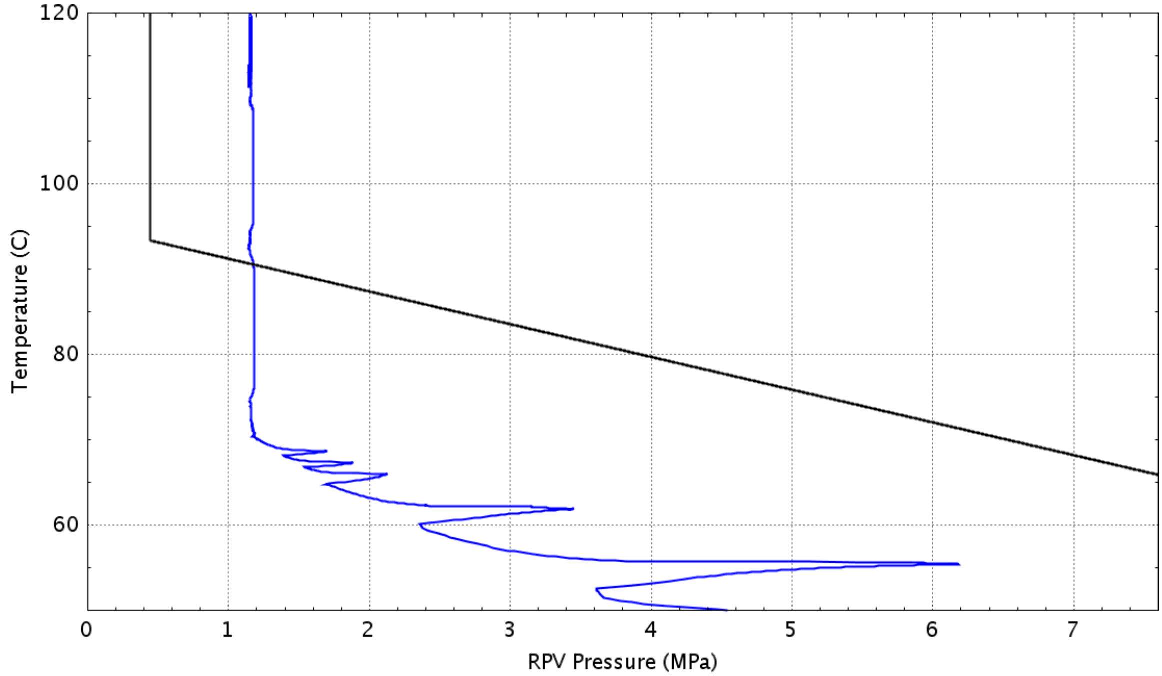

Figure G - 82 Plant status relative to the HCL curve (Graph 4 of the EOPs)

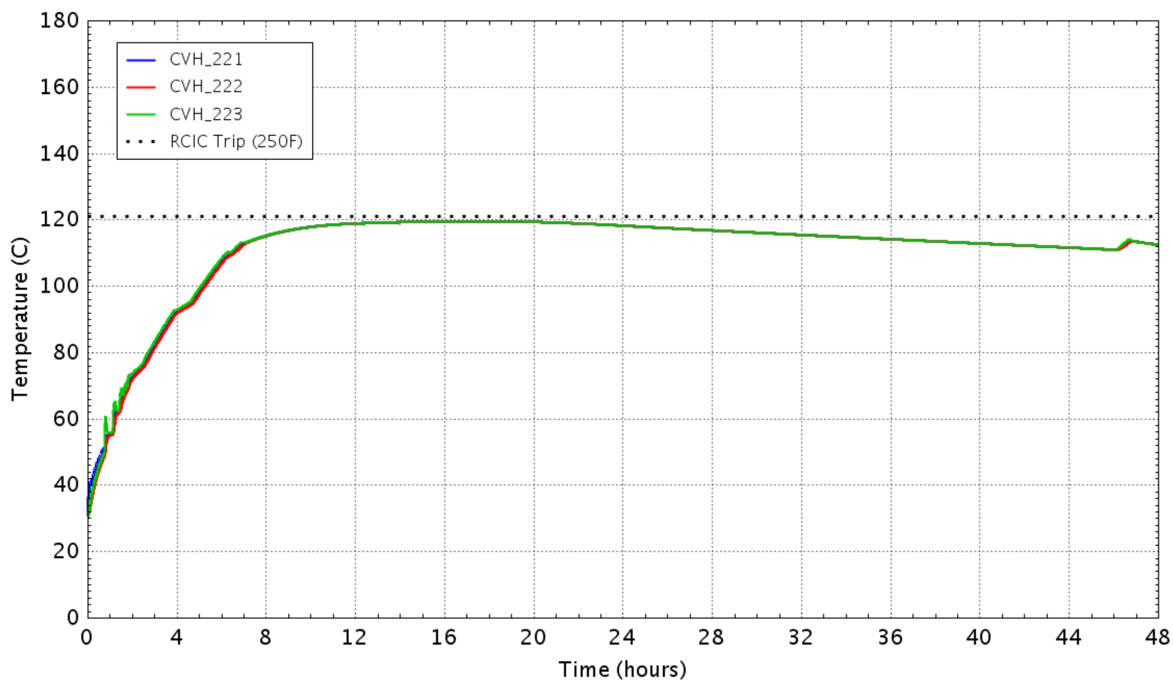

Figure G - 83 Water temperature in the wetwell

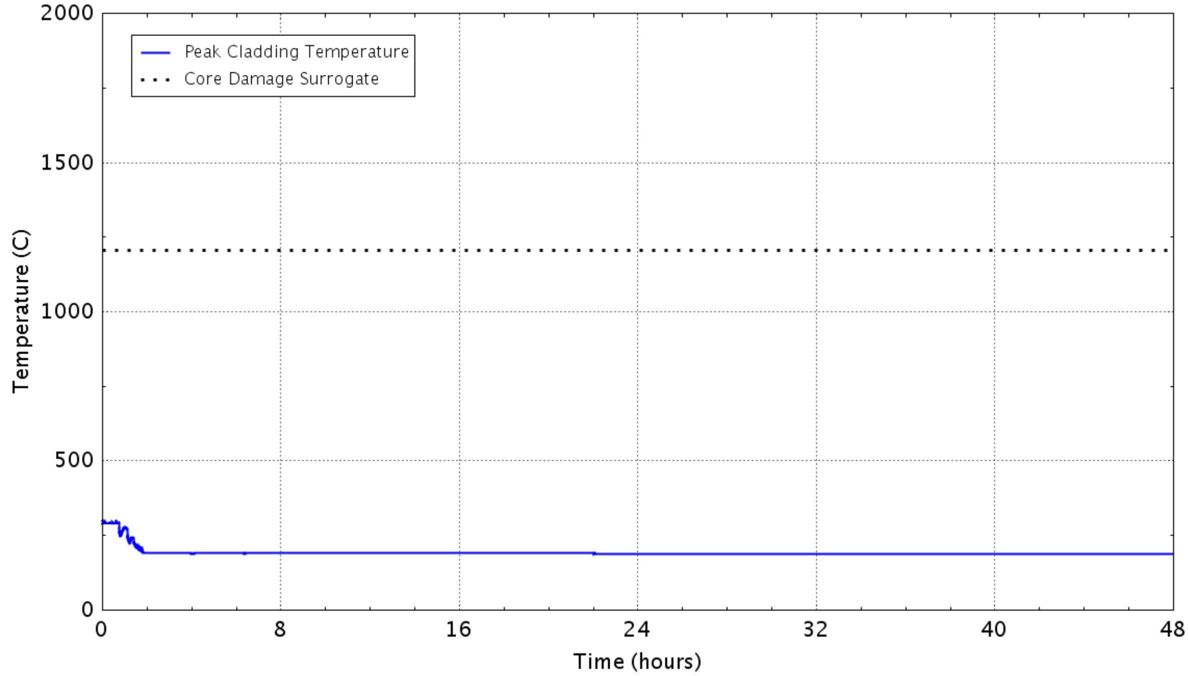

Figure G - 84 Peak temperature of the fuel cladding as a function of time

G.1.8 Case 8: LOOPGR-38-9, CST Unavailable, Initial Wetwell Level Decreased to 10.4 ft., 200 gpm Recirculation Pump Seal Leakage

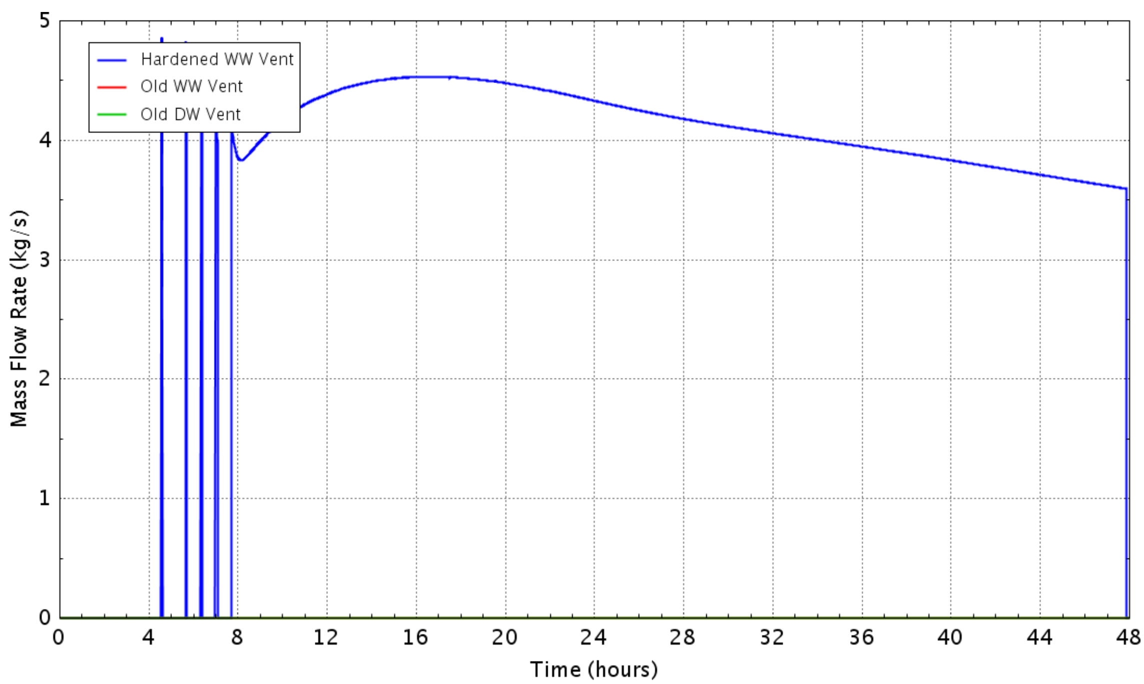

Figure G - 85 Flow rate of the containment vents

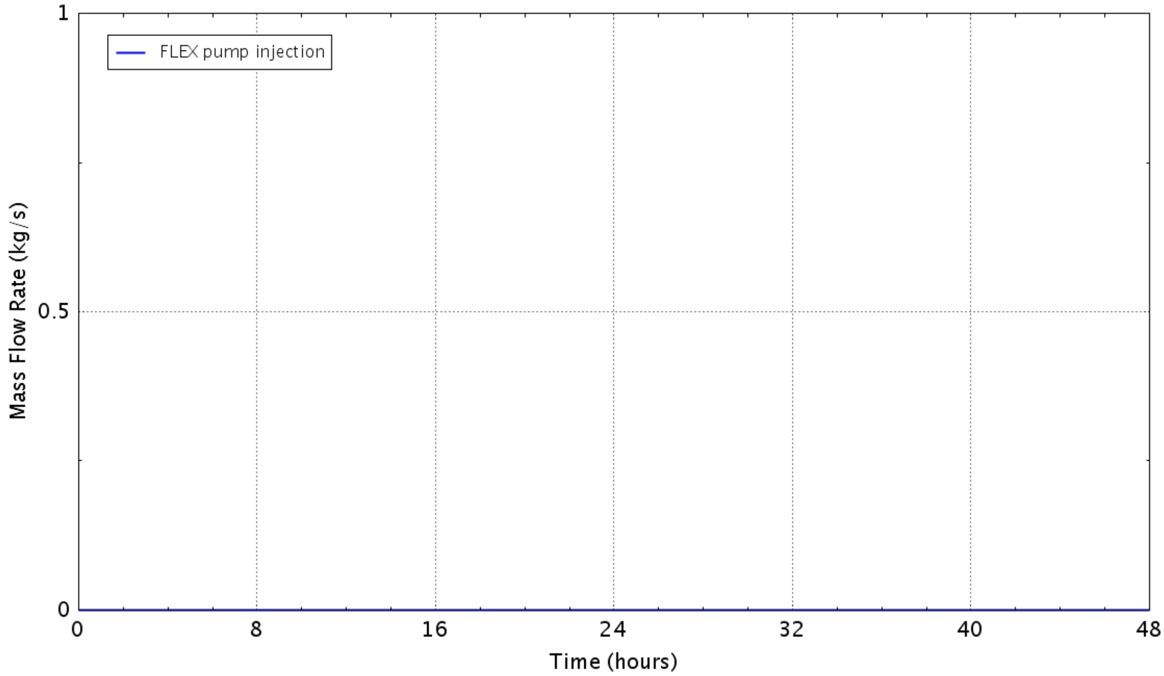

Figure G - 86 Flow rate of the FLEX pump

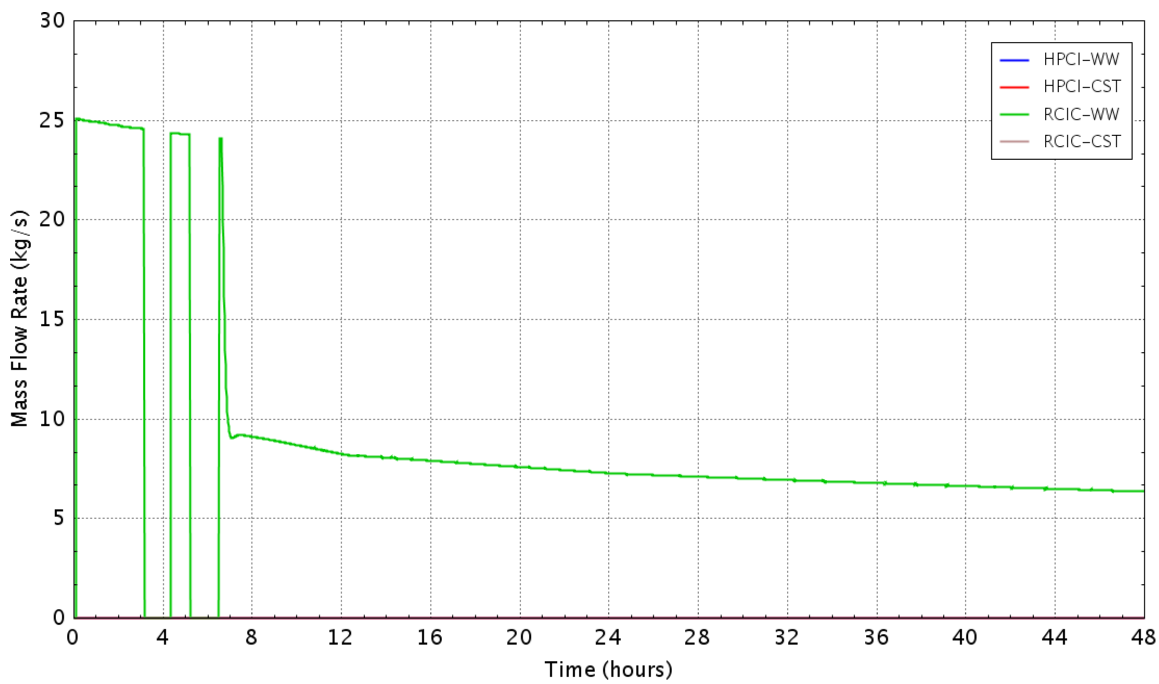

Figure G - 87 Flow rate of the HPCI/RCIC pumps

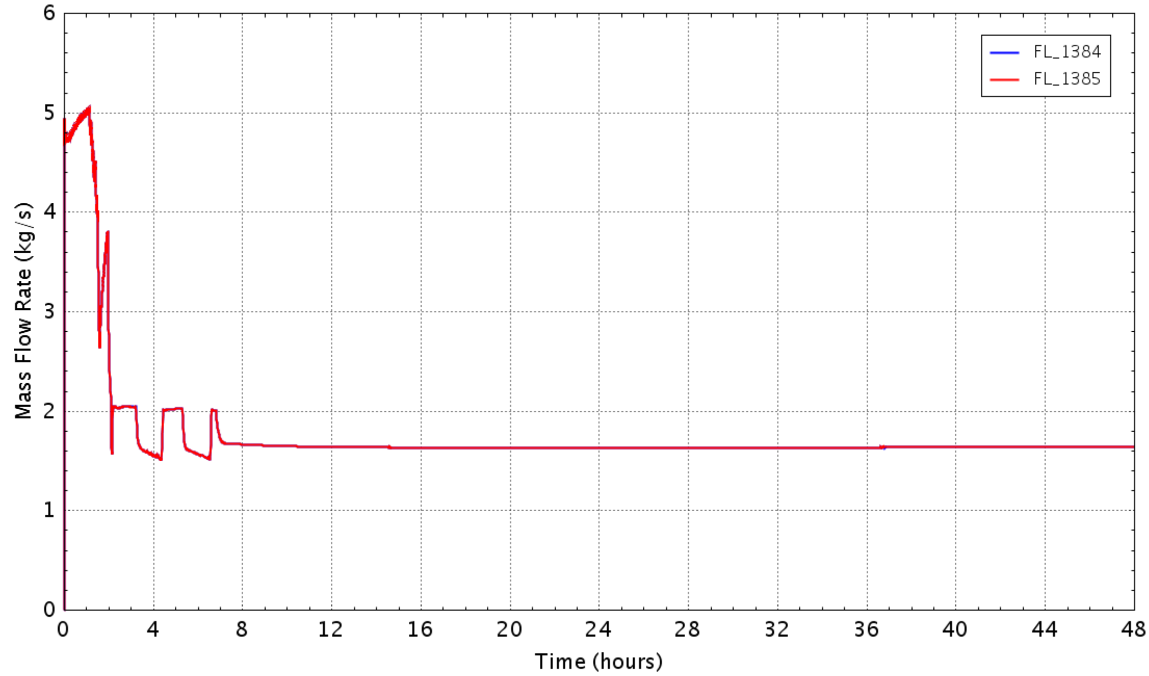

Figure G - 88 Flow rate of the recirculating pump seal leakage

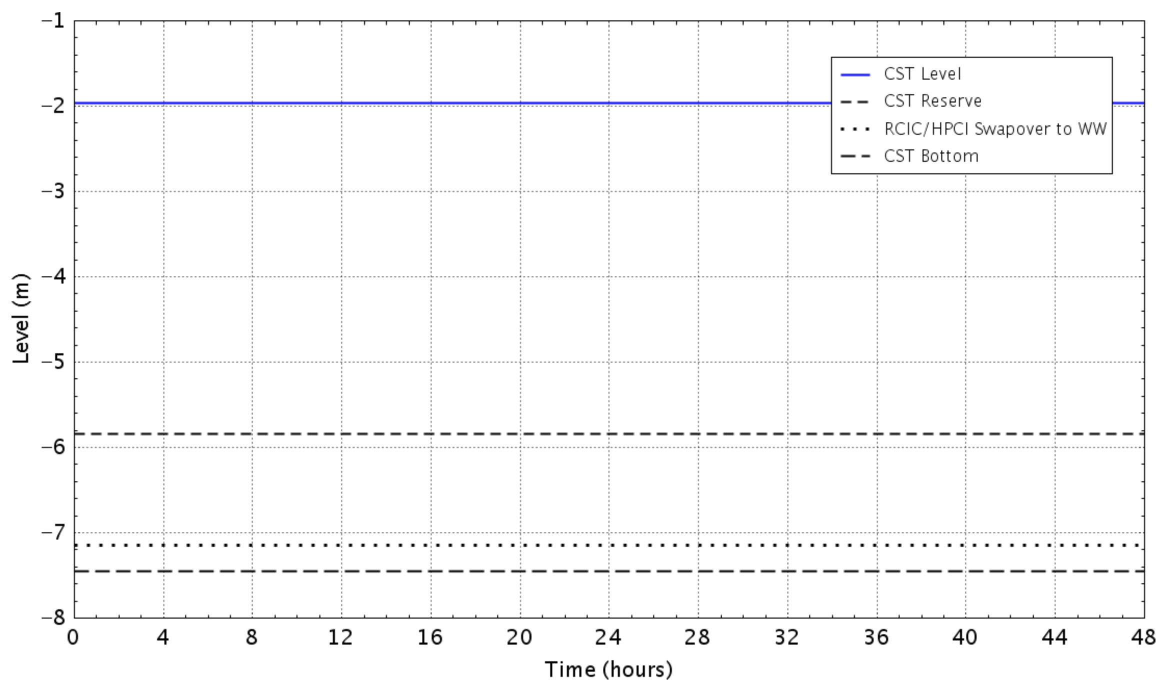

Figure G - 89 Water level in the CST

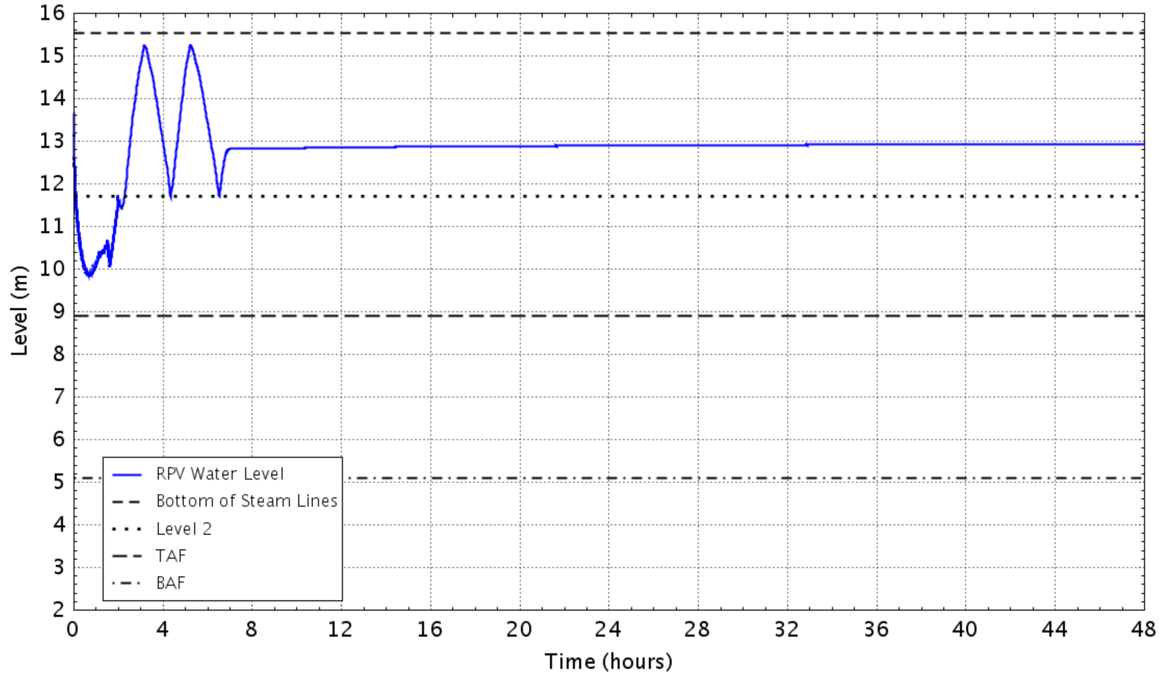

Figure G - 90 RPV Downcomer water level

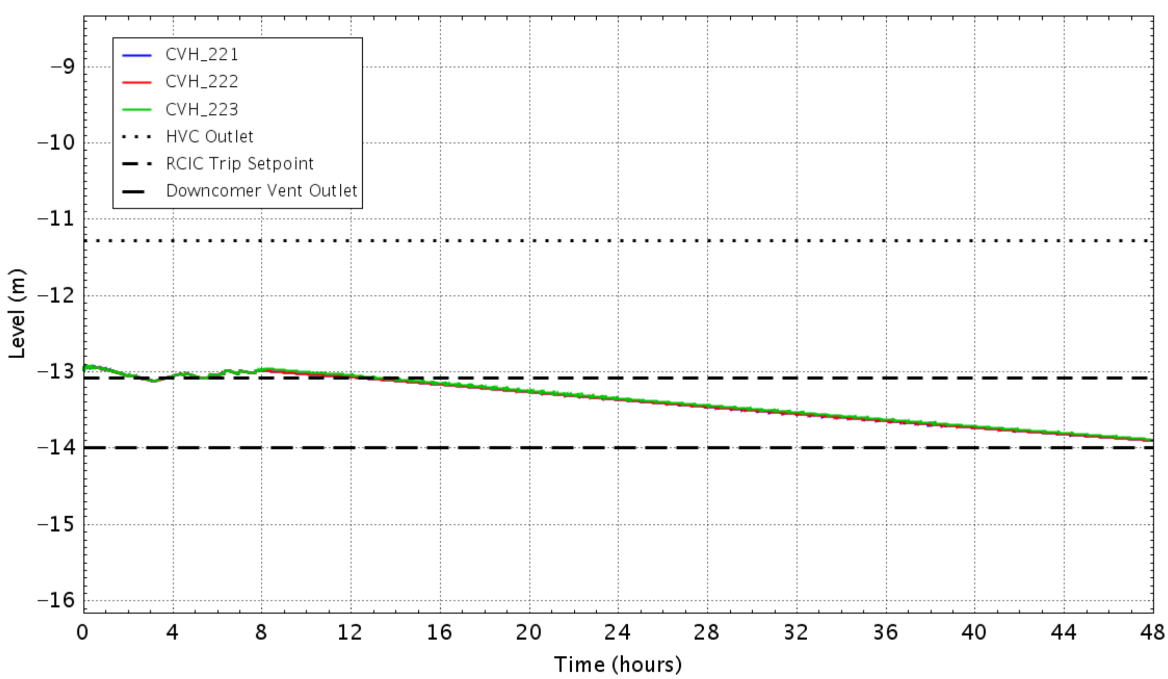

Figure G - 91 Water level in the wetwell

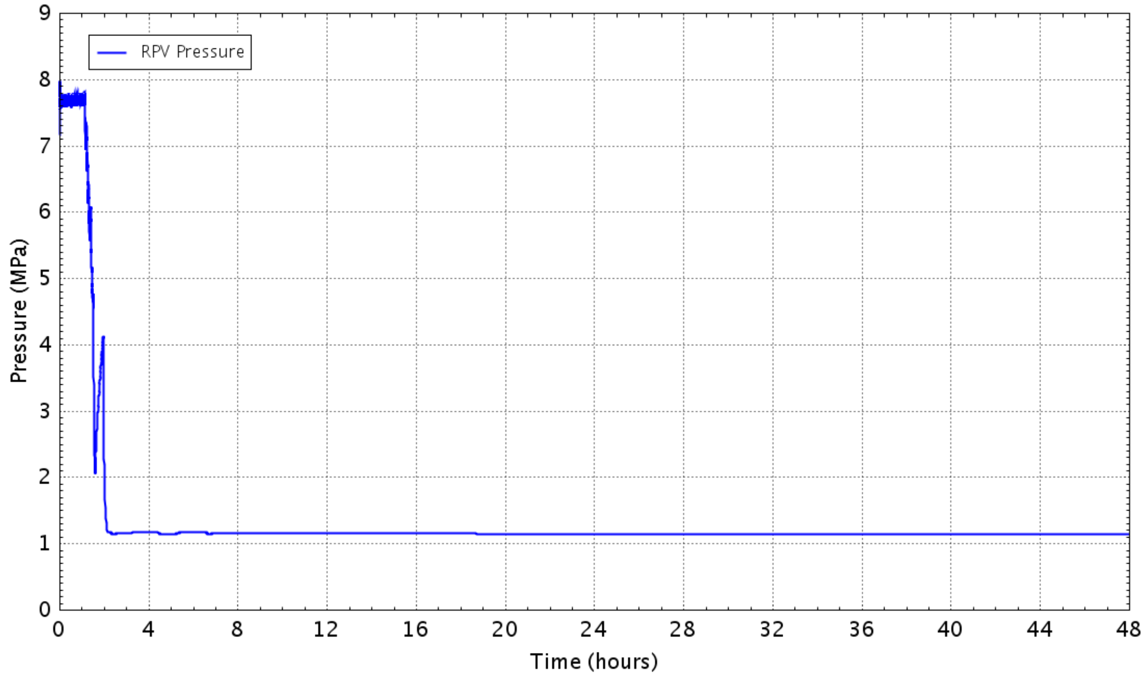

Figure G - 92 Pressure in the RPV

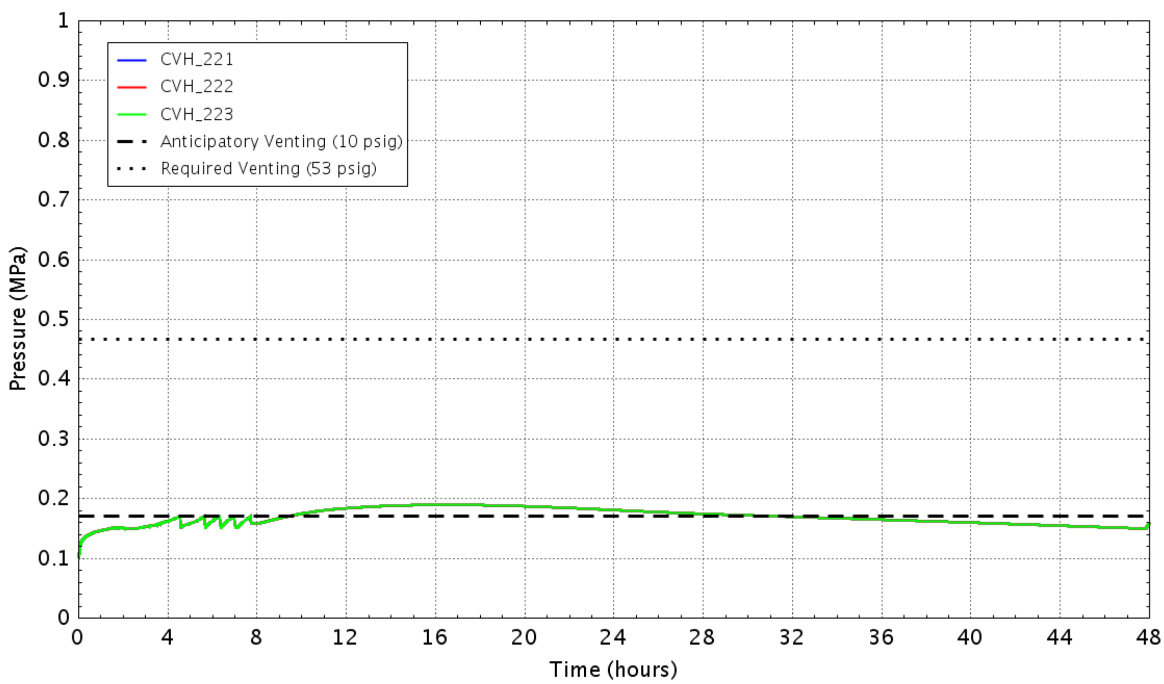

Figure G - 93 Pressure in the wetwell

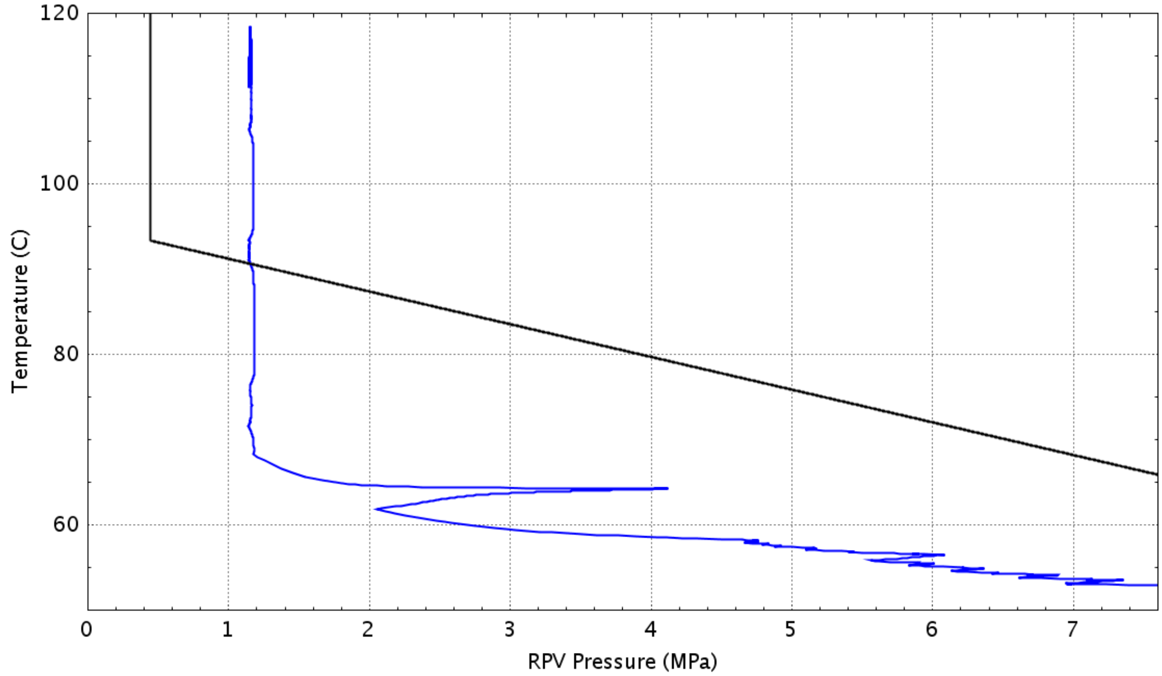

Figure G - 94 Plant status relative to the HCL curve (Graph 4 of the EOPs)

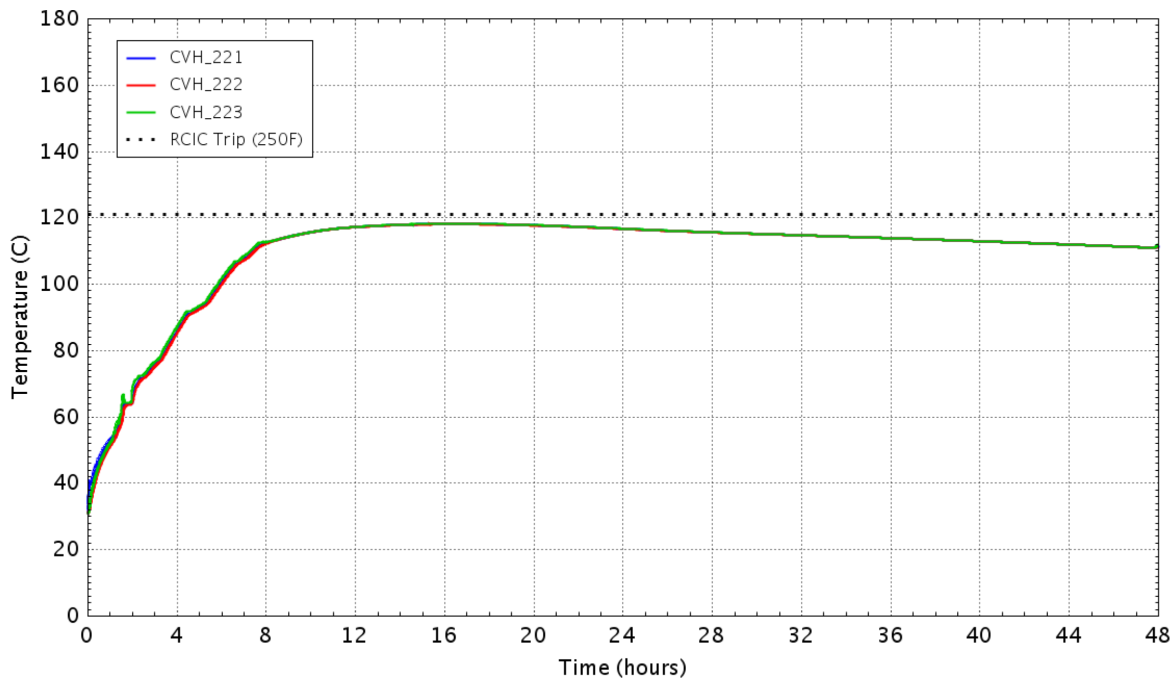

Figure G - 95 Water temperature in the wetwell

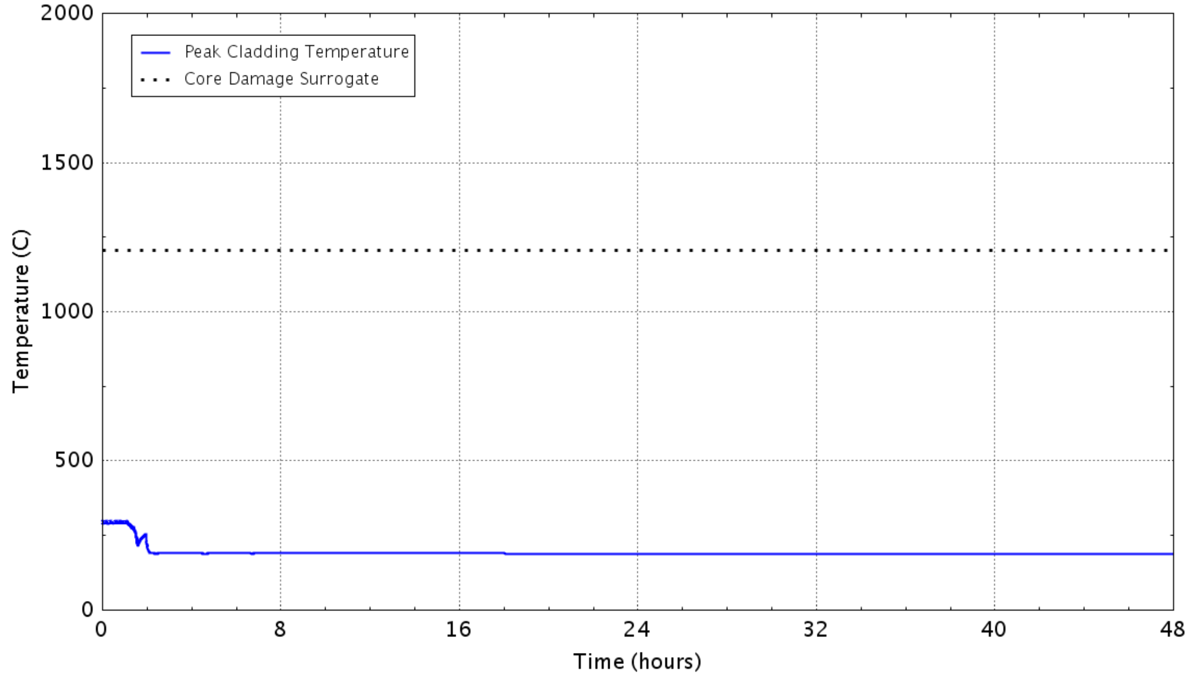

Figure G - 96 Peak temperature of the fuel cladding as a function of time

G.1.9 Case 9: LOOPGR-38-9, RCIC Injection from CST Initially with Swap to FLEX Injection at 7 hrs., Initial CST Level Decreased to 24 ft., Nominal Recirculation Pump Seal Leakage

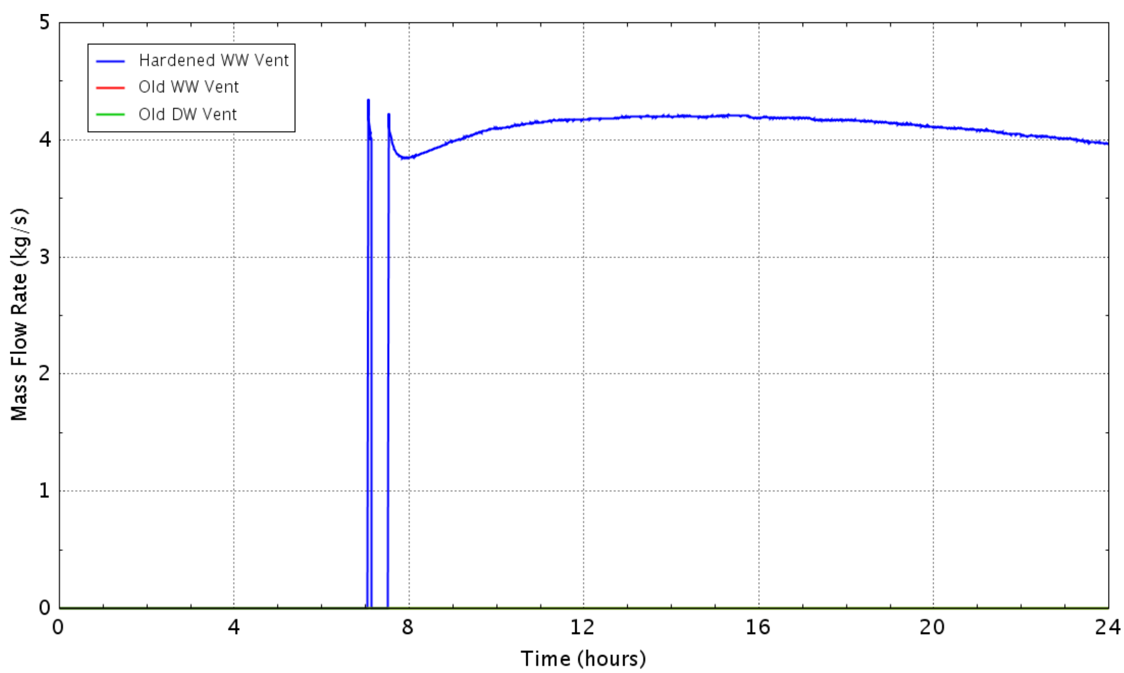

Figure G – 97 Flow rate of the containment vents

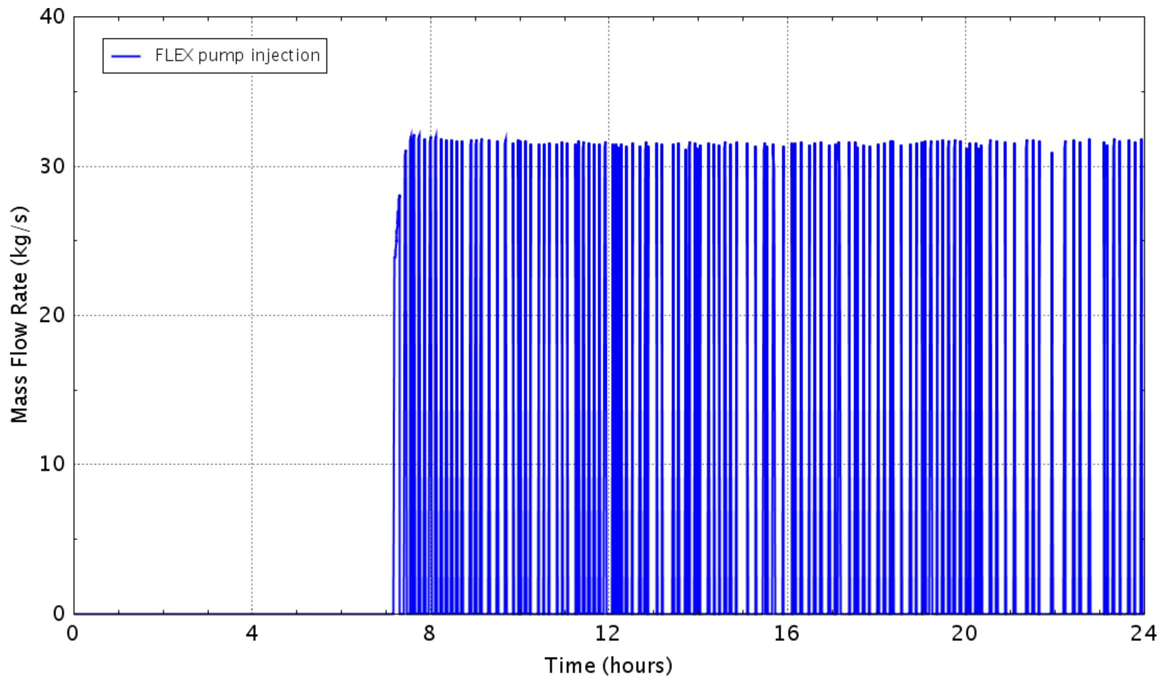

Figure G - 98 Flow rate of the FLEX pump

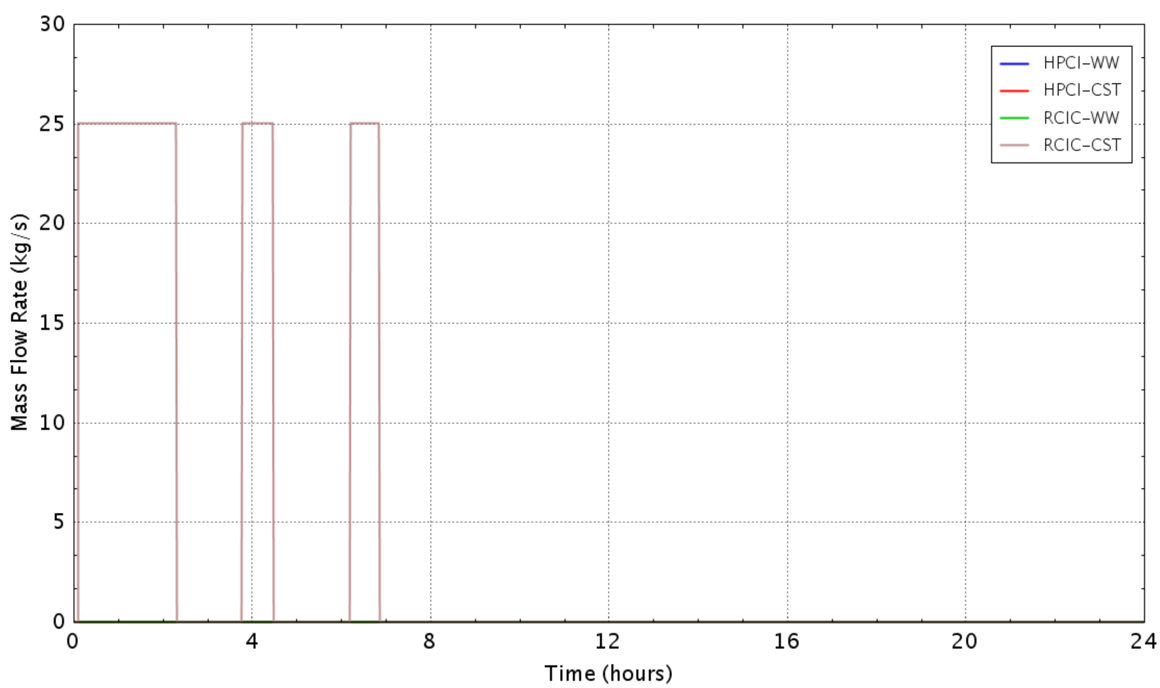

Figure G - 99 Flow rate of the HPCI/RCIC pumps

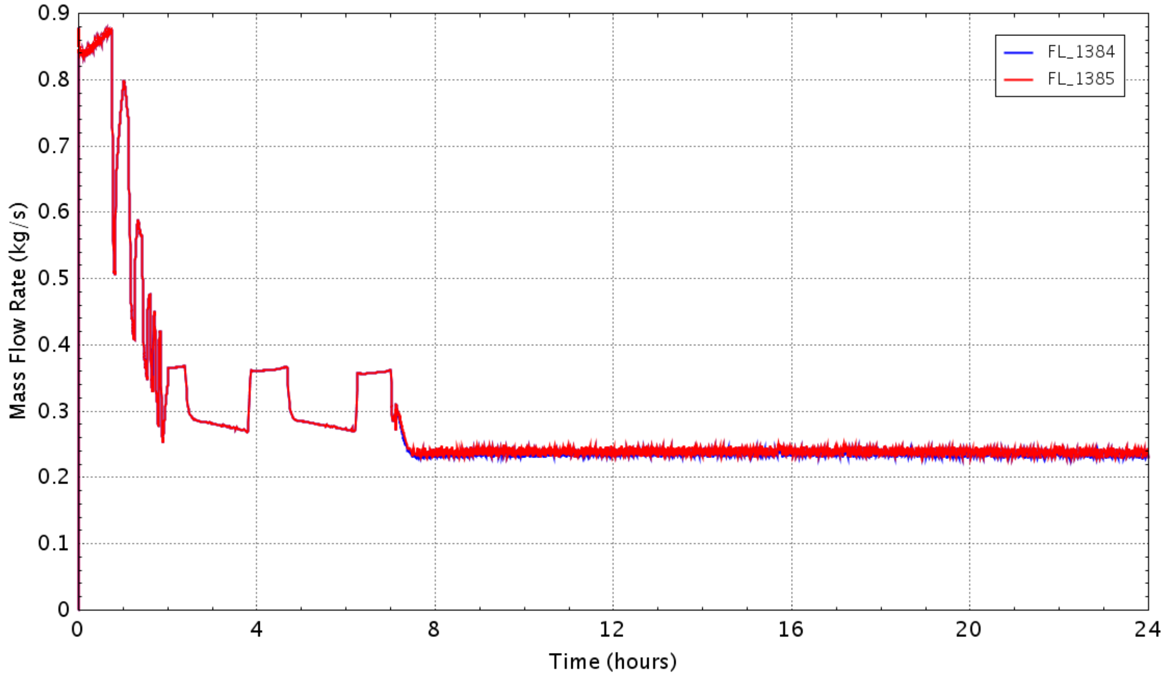

Figure G - 100 Flow rate of the recirculating pump seal leakage

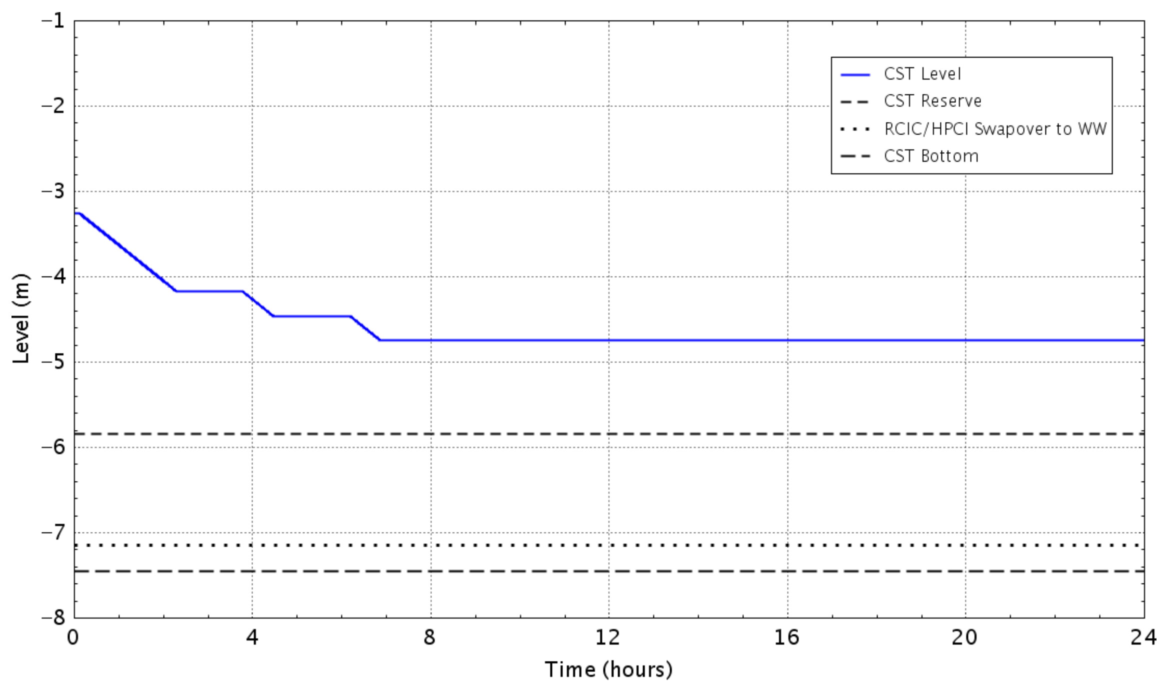

Figure G - 101 Water level in the CST

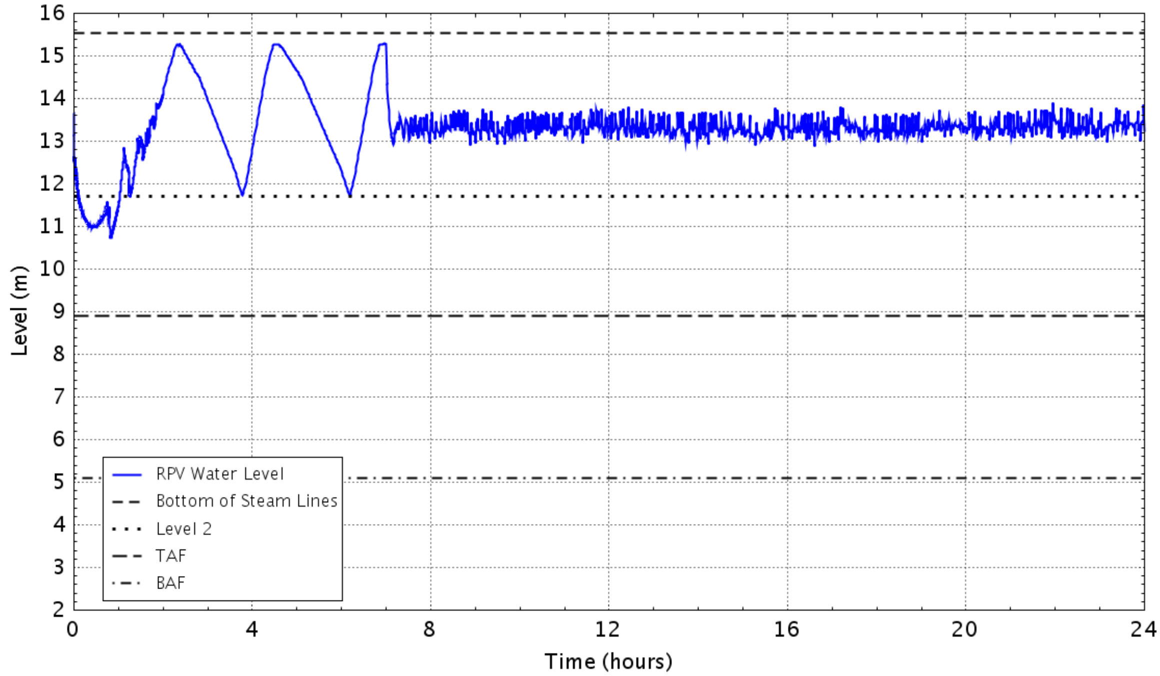

Figure G - 102 RPV Downcomer water level

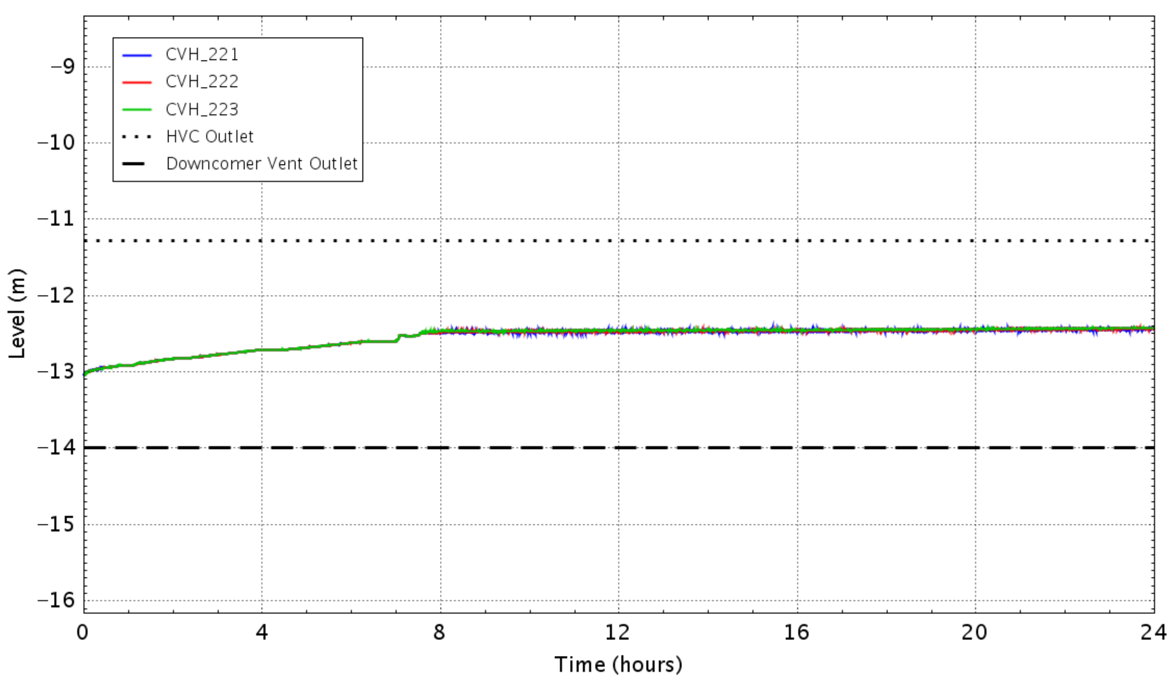

Figure G - 103 Water level in the wetwell

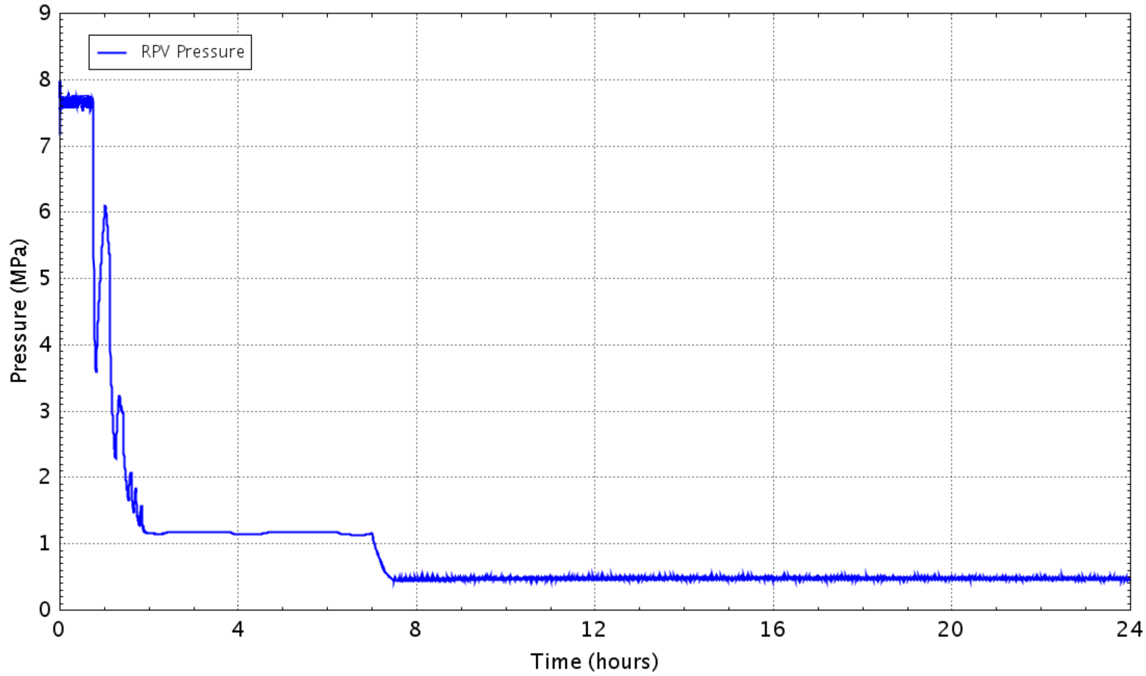

Figure G - 104 Pressure in the RPV

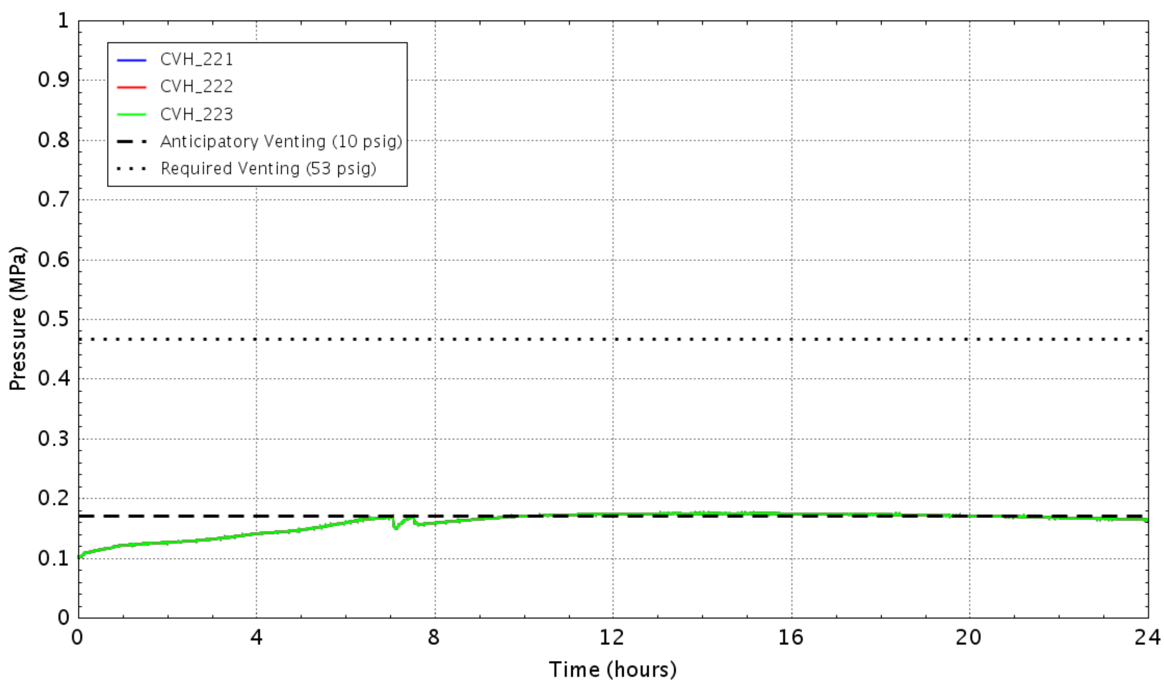

Figure G - 105 Pressure in the wetwell

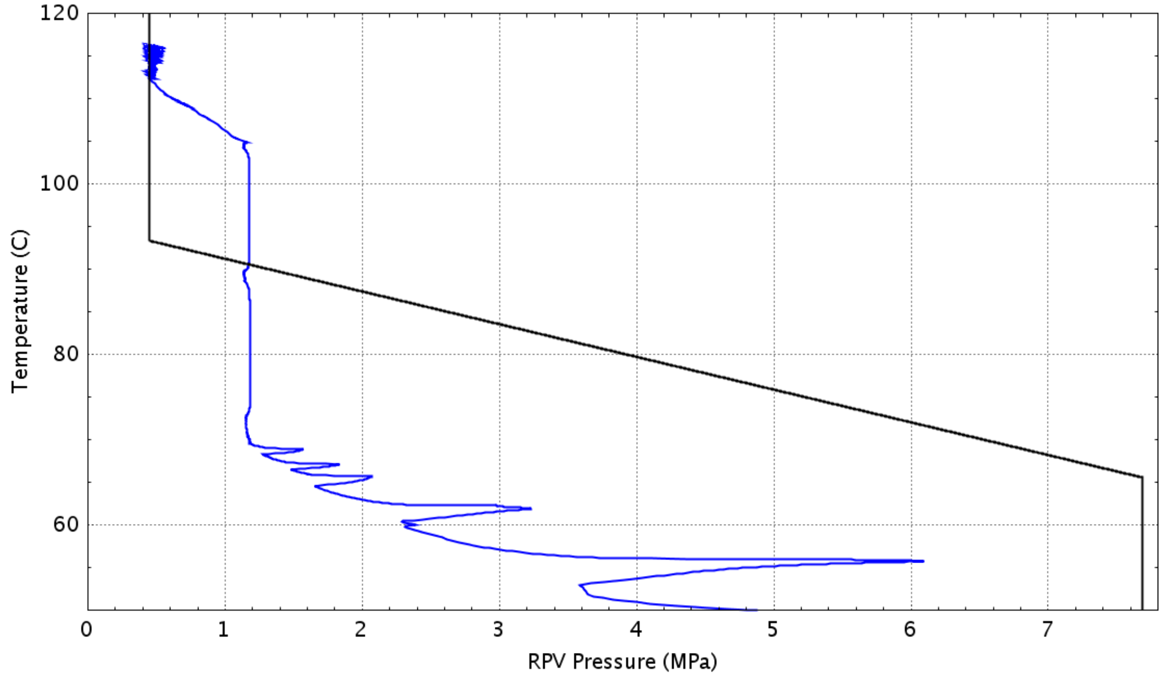

Figure G - 106 Plant status relative to the HCL curve (Graph 4 of the EOPs)

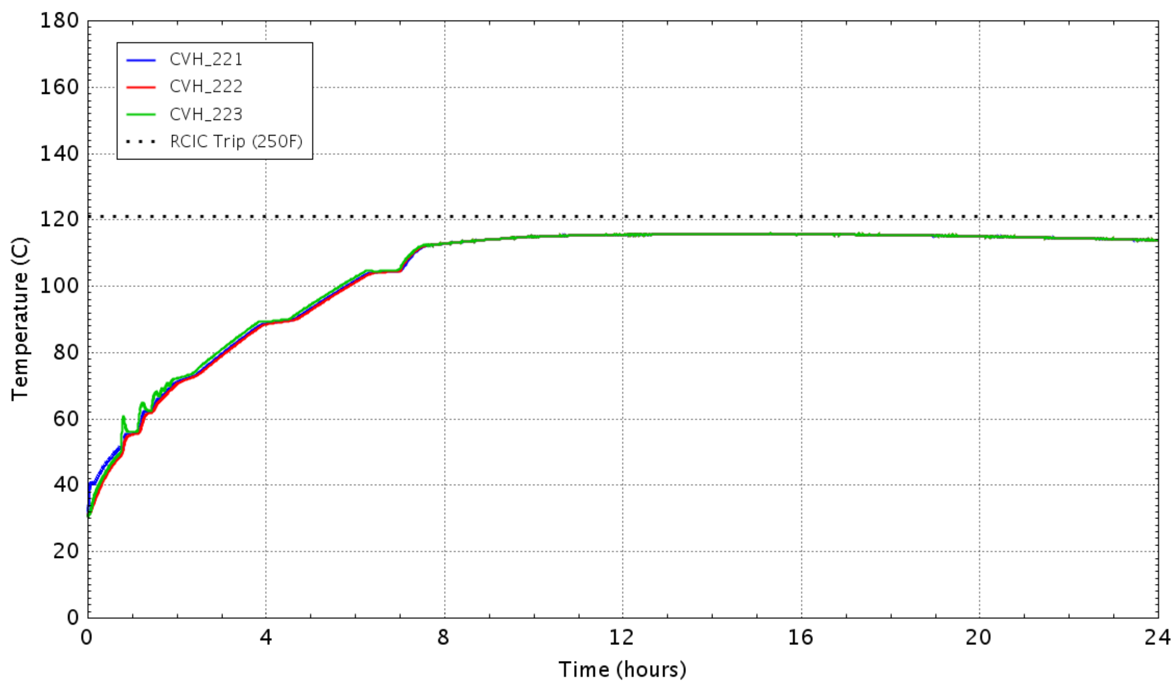

Figure G - 107 Water temperature in the wetwell

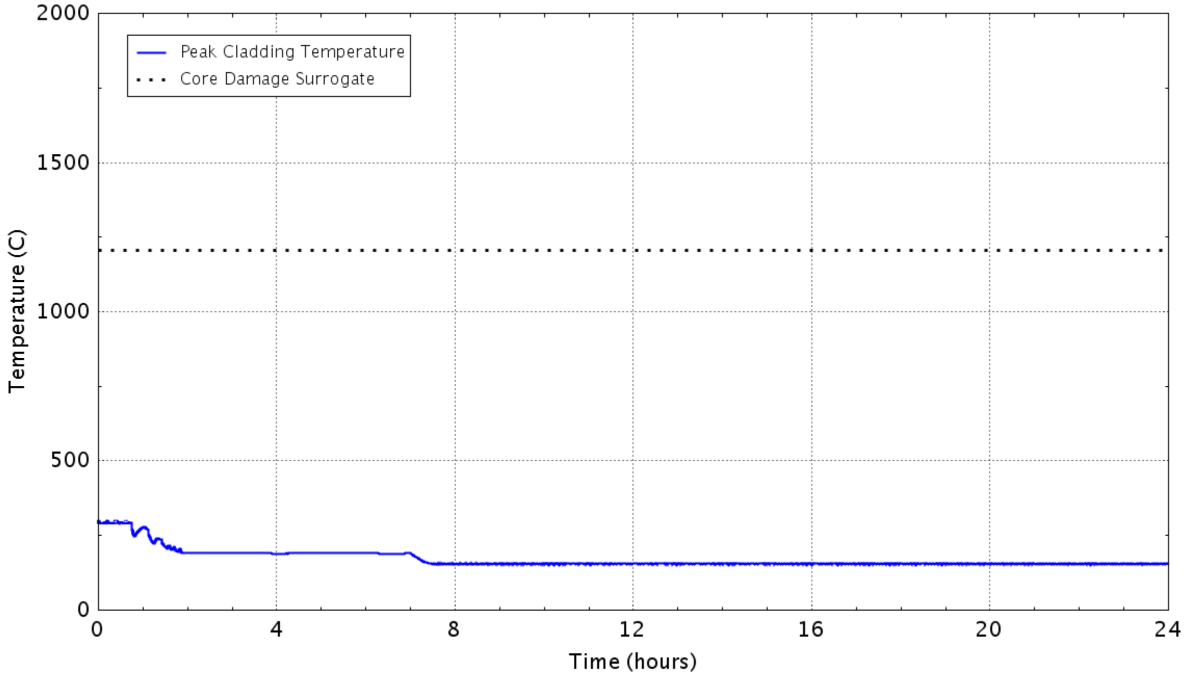

Figure G - 108 Peak temperature of the fuel cladding as a function of time

G.1.10 Case 10: LOOPGR-38-9, RCIC Injection from CST Initially with Swap to FLEX Injection at 7 hrs., Initial CST Level Decreased to 24 ft., 200 gpm Recirculation Pump Seal Leakage

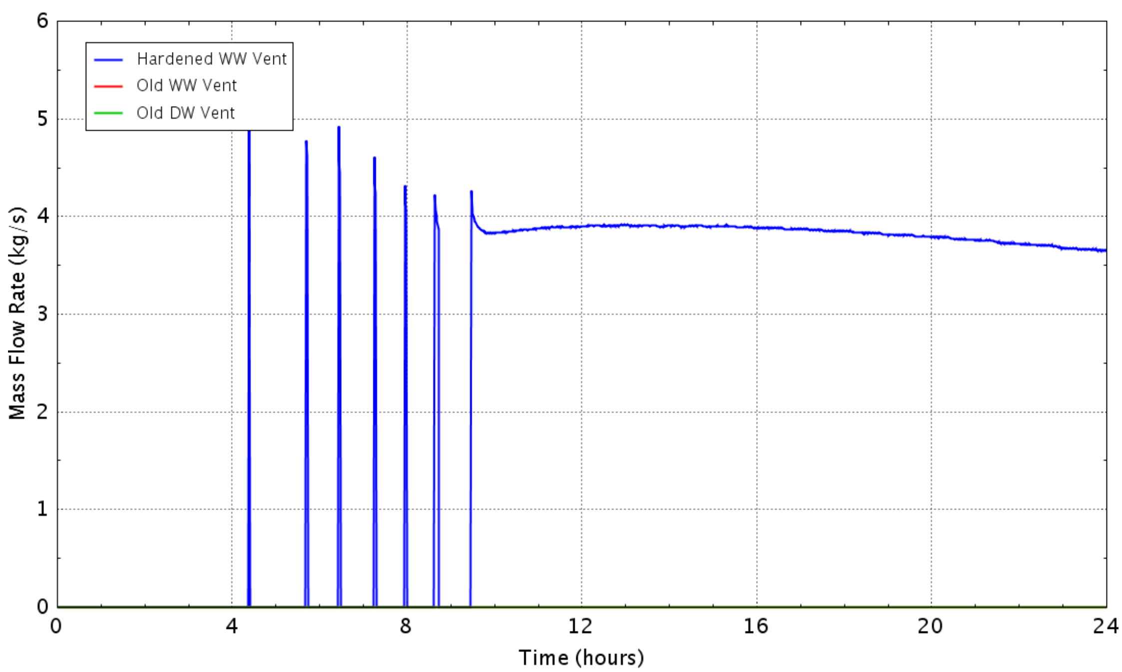

Figure G - 109 Flow rate of the containment vents

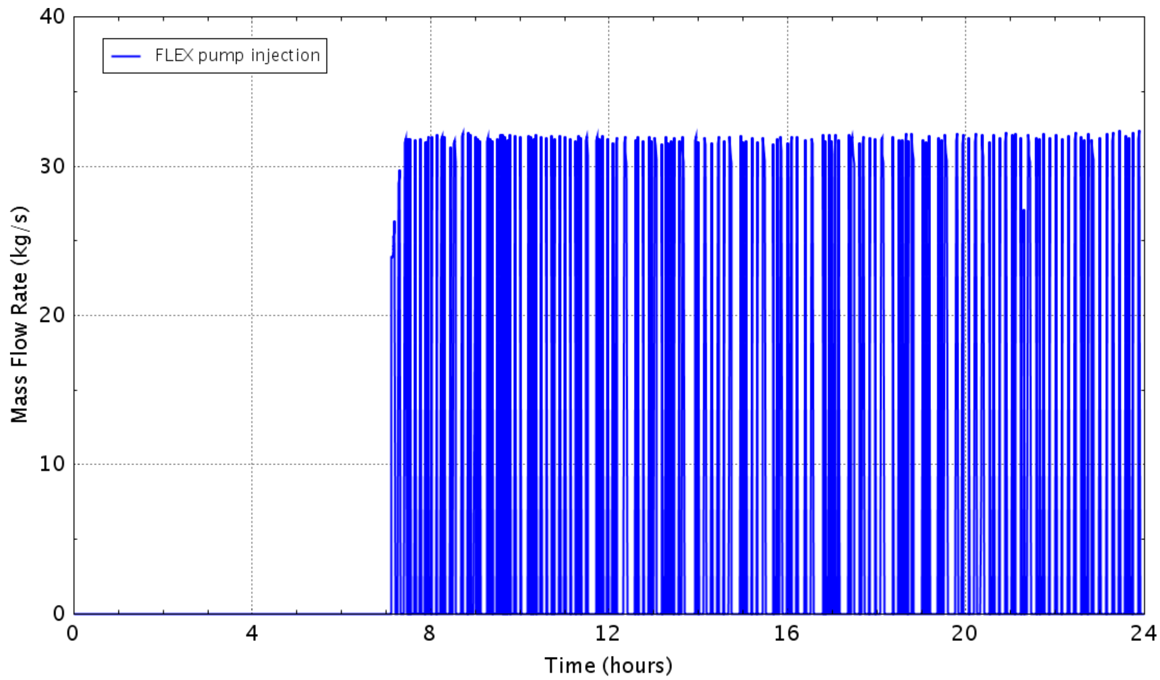

Figure G - 110 Flow rate of the FLEX pump

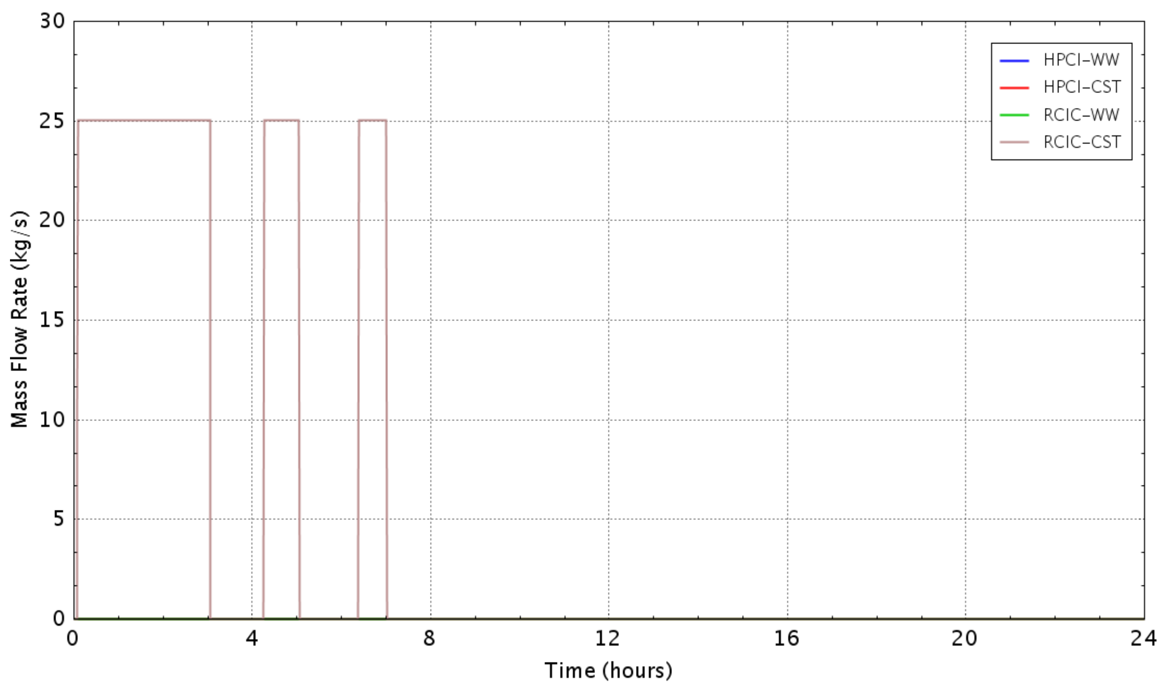

Figure G - 111 Flow rate of the HPCI/RCIC pumps

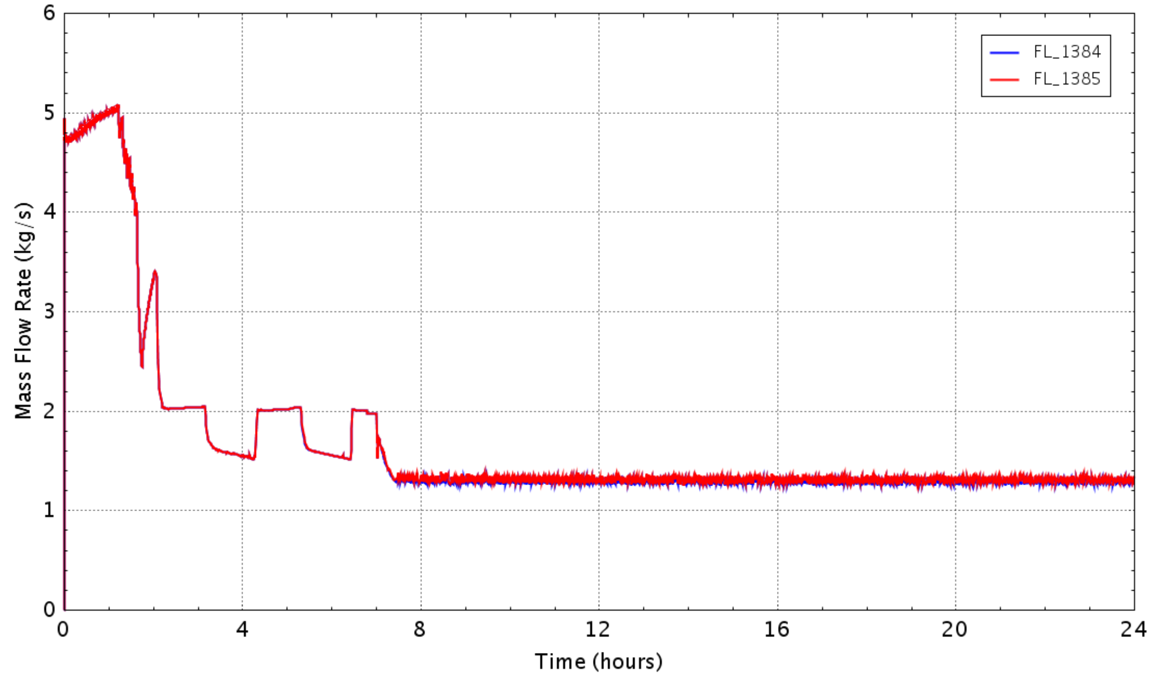

Figure G - 112 Flow rate of the recirculating pump seal leakage

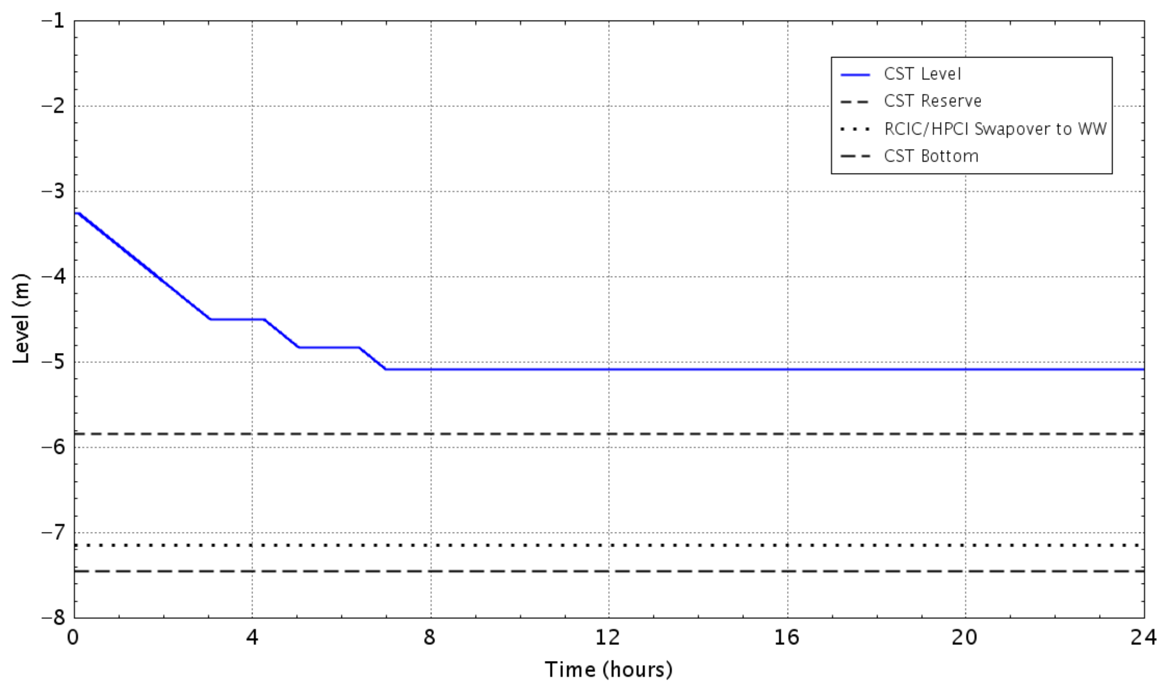

Figure G - 113 Water level in the CST

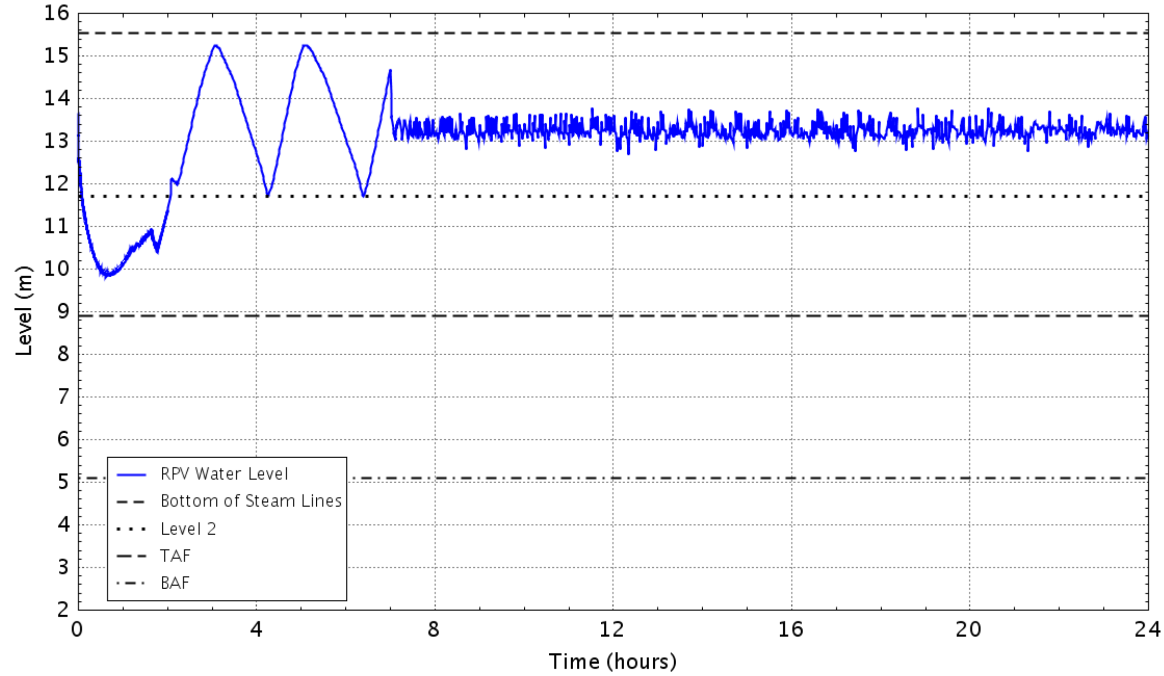

Figure G - 114 RPV Downcomer water level

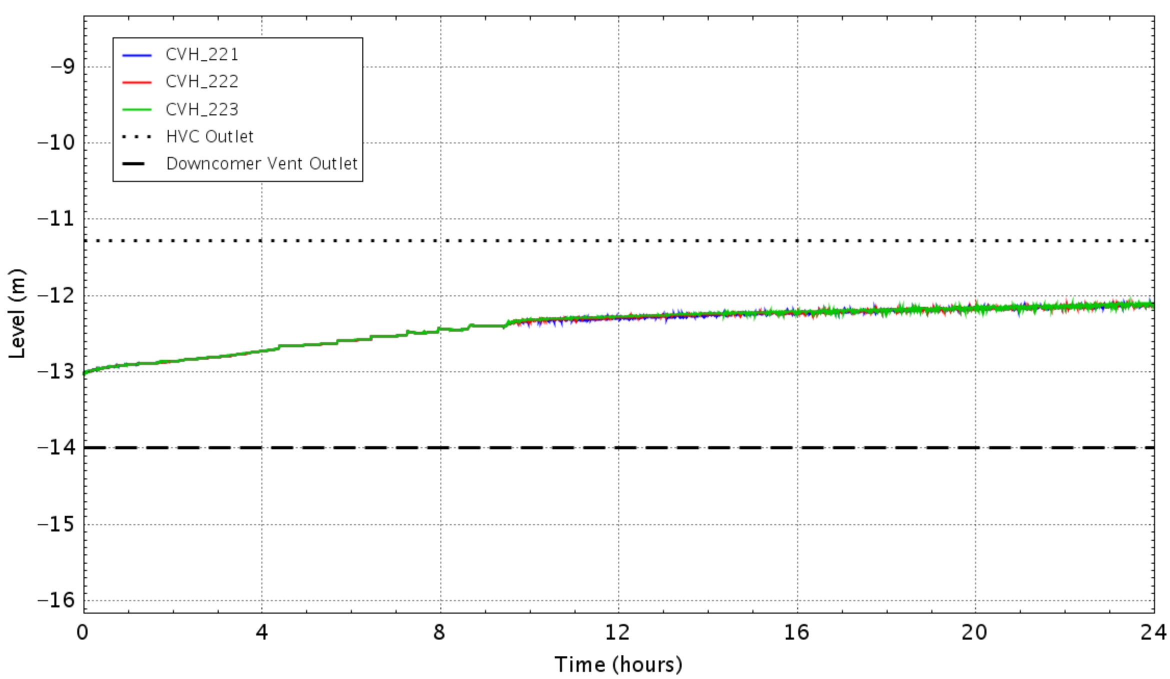

Figure G - 115 Water level in the wetwell

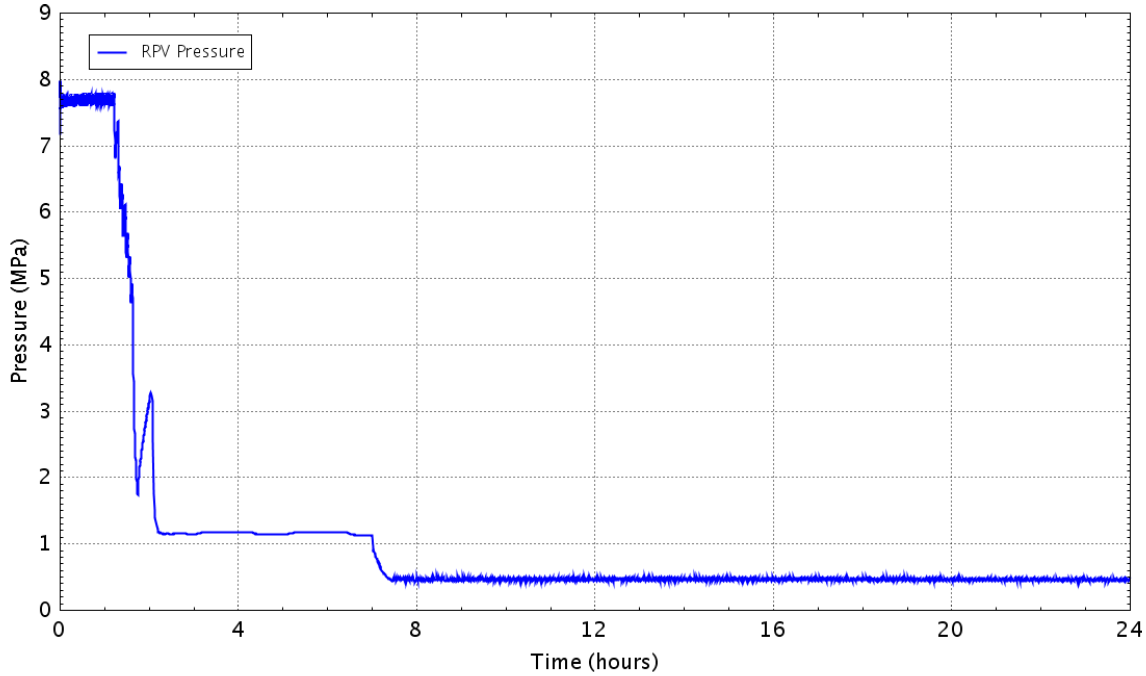

Figure G - 116 Pressure in the RPV

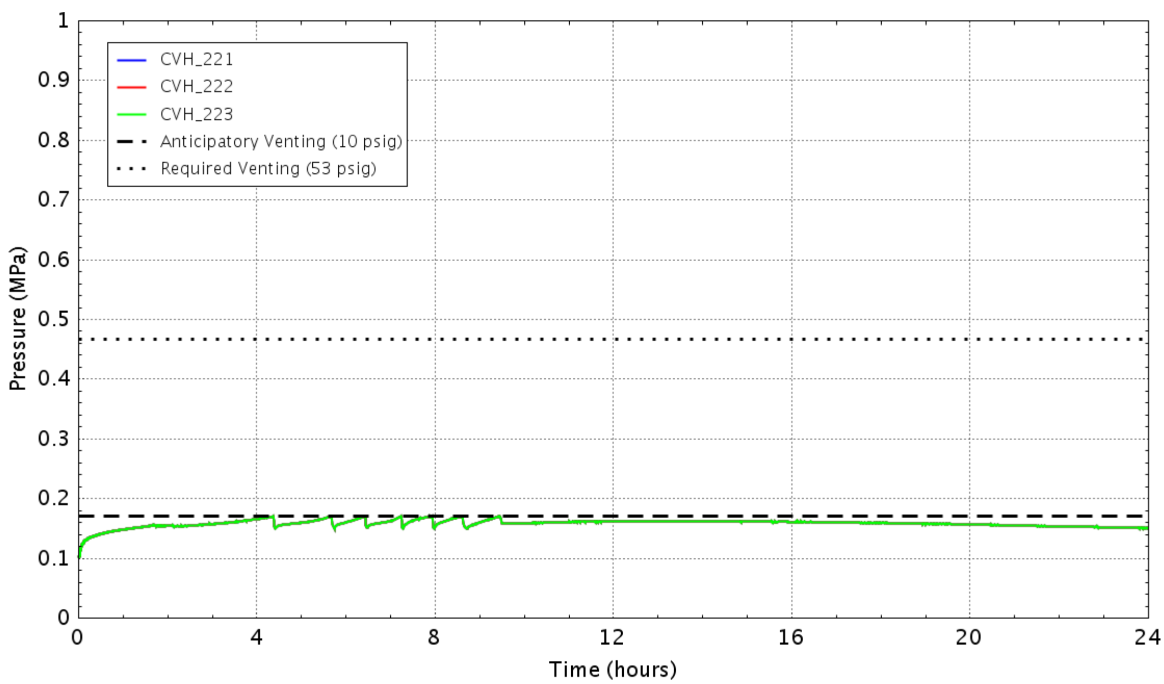

Figure G - 117 Pressure in the wetwell

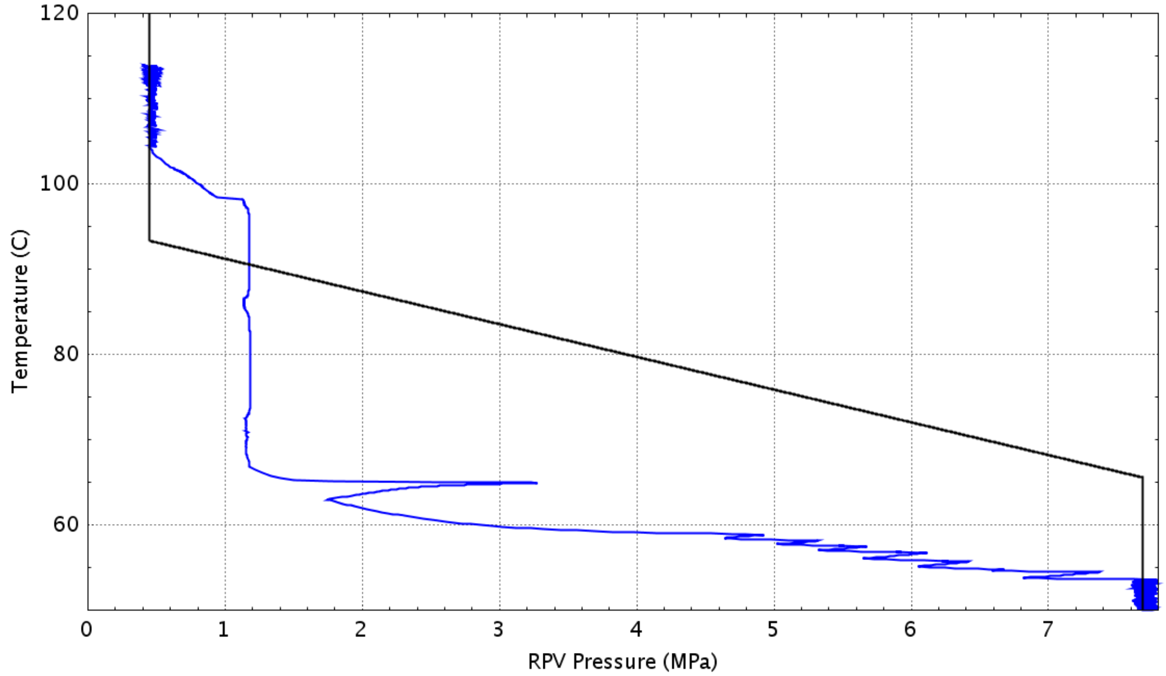

Figure G - 118 Plant status relative to the HCL curve (Graph 4 of the EOPs)

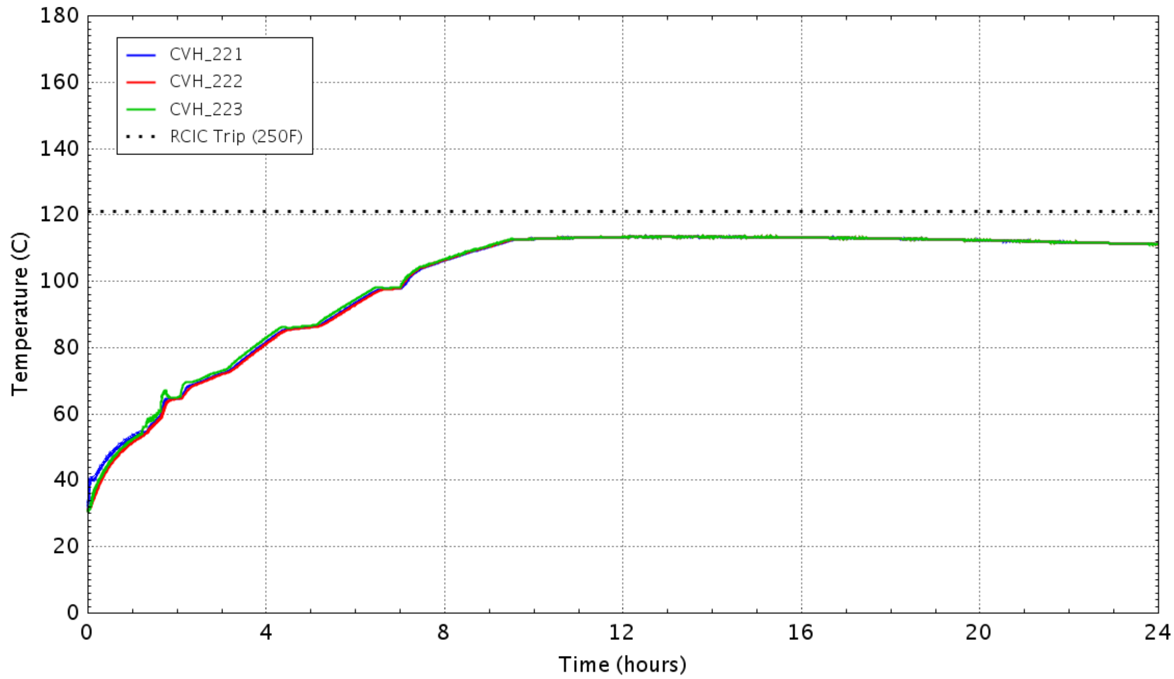

Figure G - 119 Water temperature in the wetwell

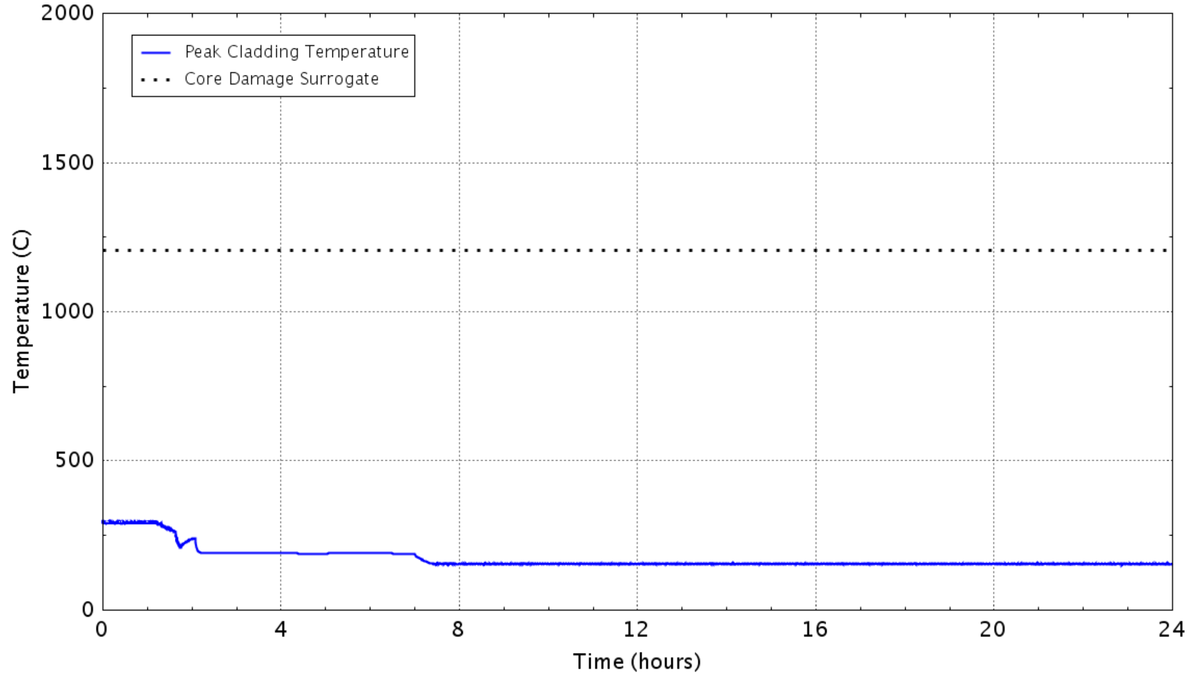

Figure G - 120 Peak temperature of the fuel cladding as a function of time

G.1.11 Case 11: LOOPGR-38-9, RCIC Injection from Wetwell Initially with Swap to FLEX Injection at 7 hrs., Initial Wetwell Level Decreased to 10.4 ft., Nominal Recirculation Pump Seal Leakage

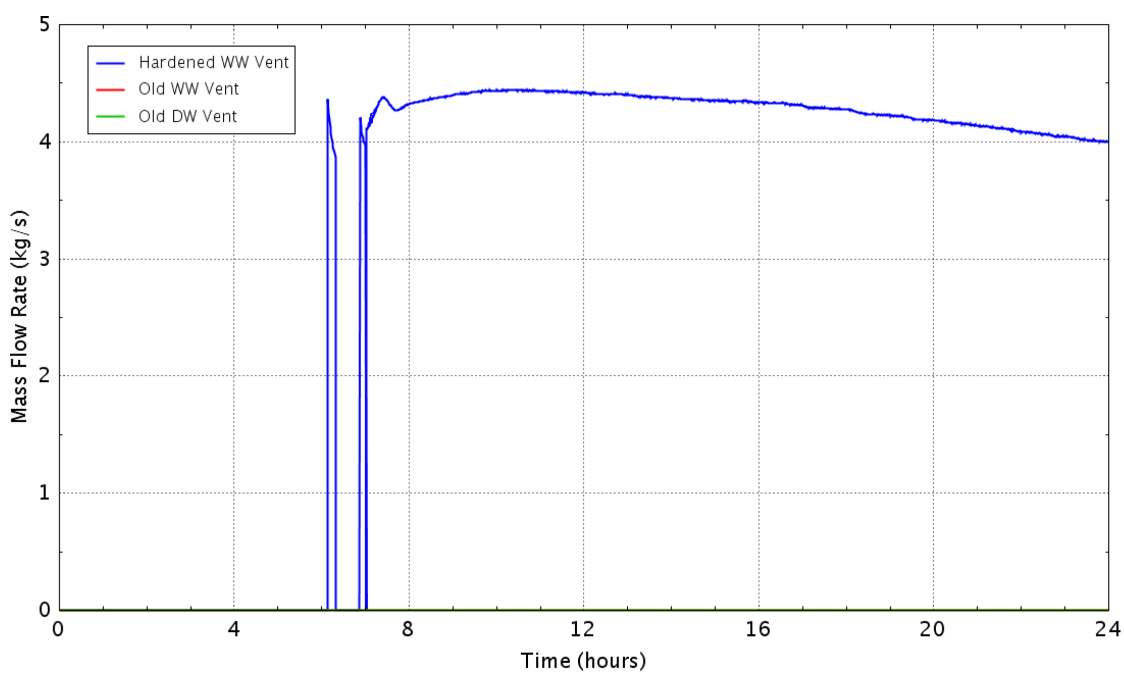

Figure G - 121 Flow rate of the containment vents

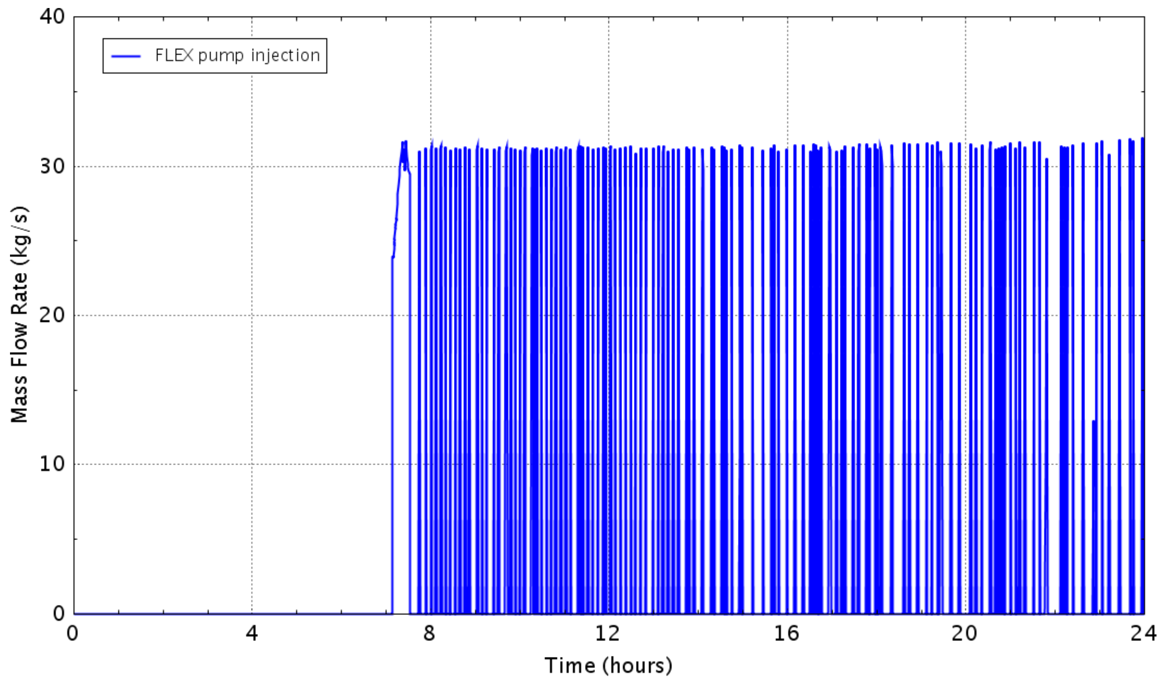

Figure G - 122 Flow rate of the FLEX pump

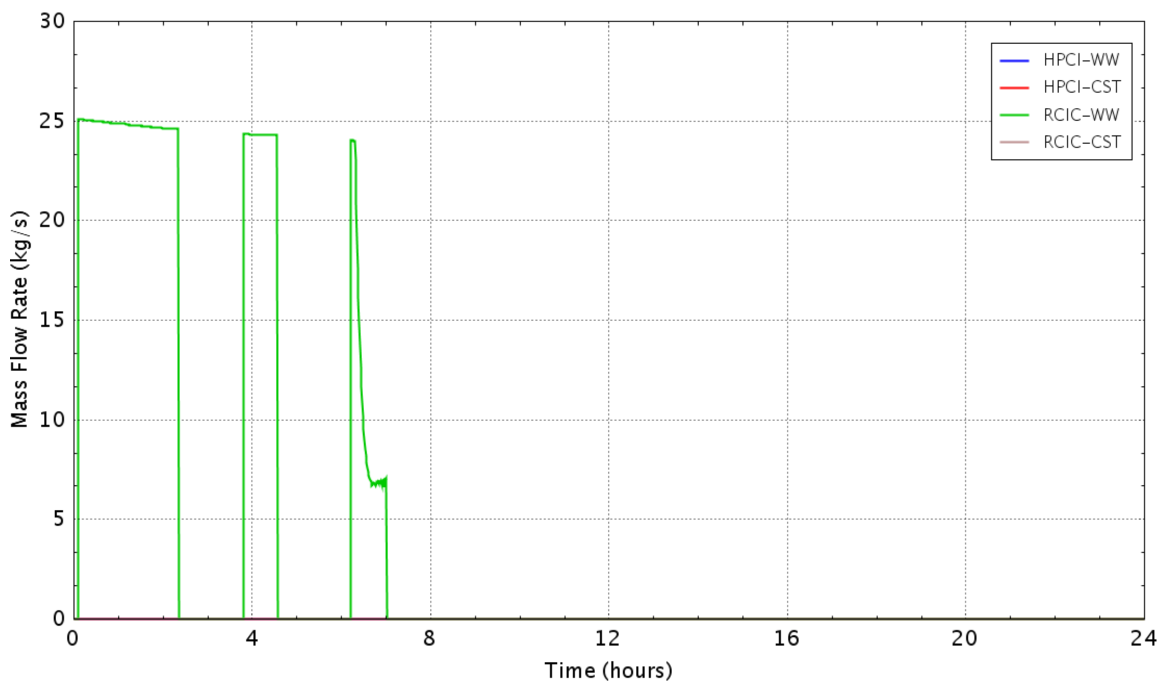

Figure G - 123 Flow rate of the HPCI/RCIC pumps

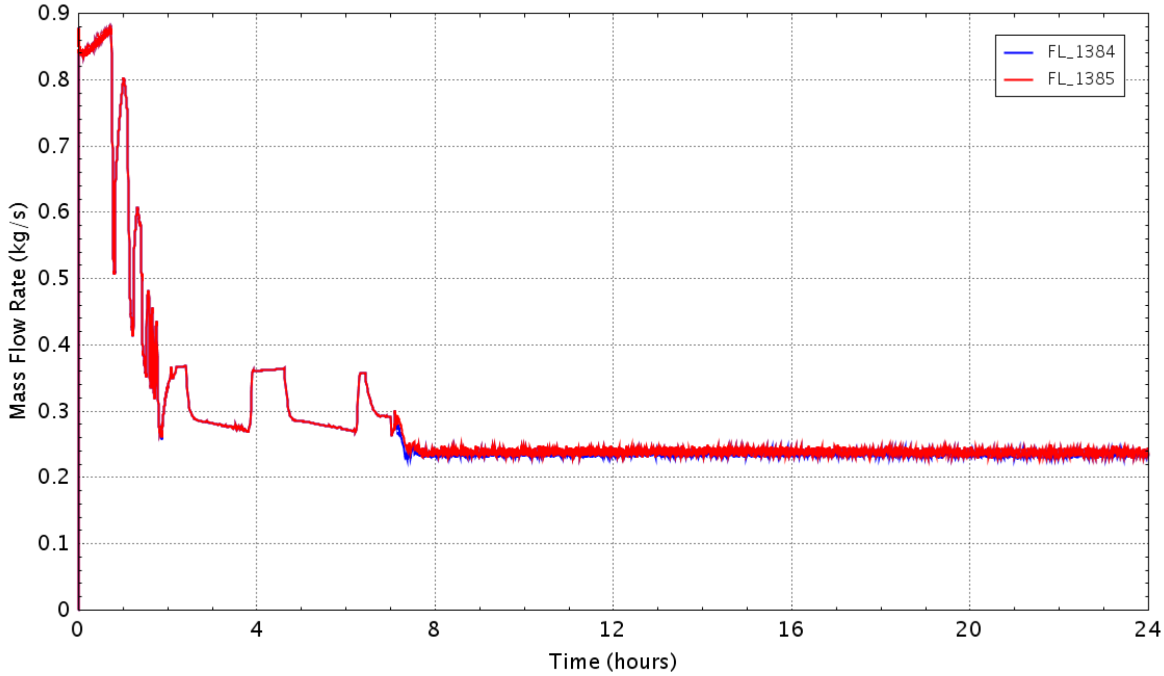

Figure G - 124 Flow rate of the recirculating pump seal leakage

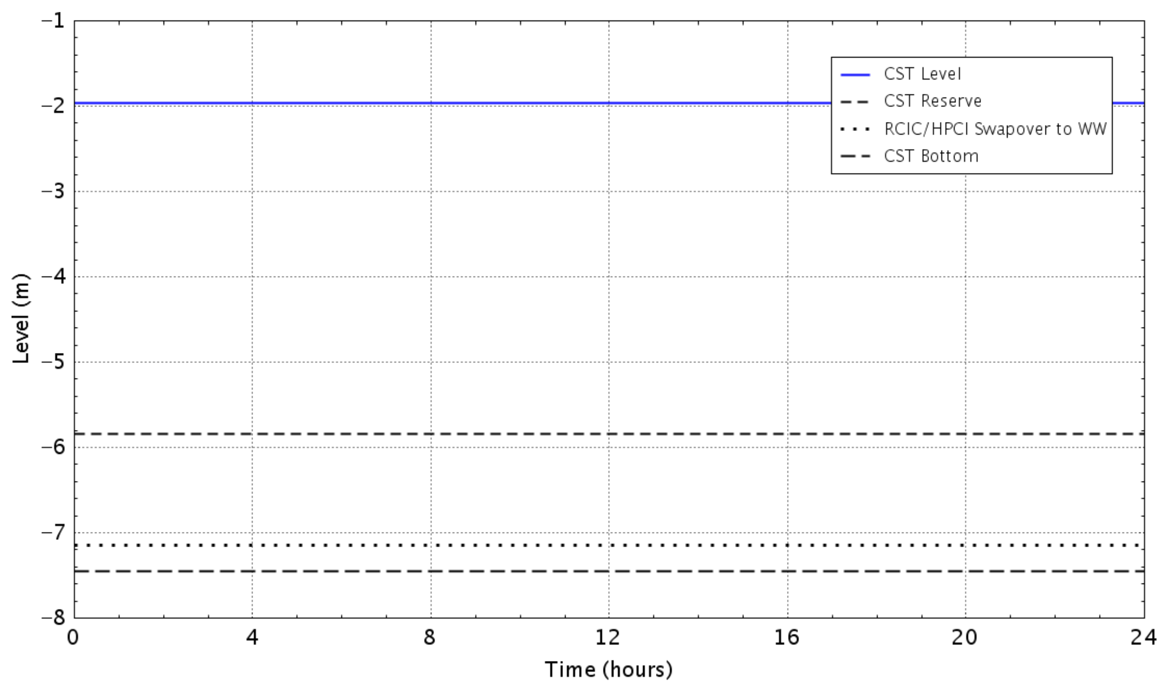

Figure G - 125 Water level in the CST

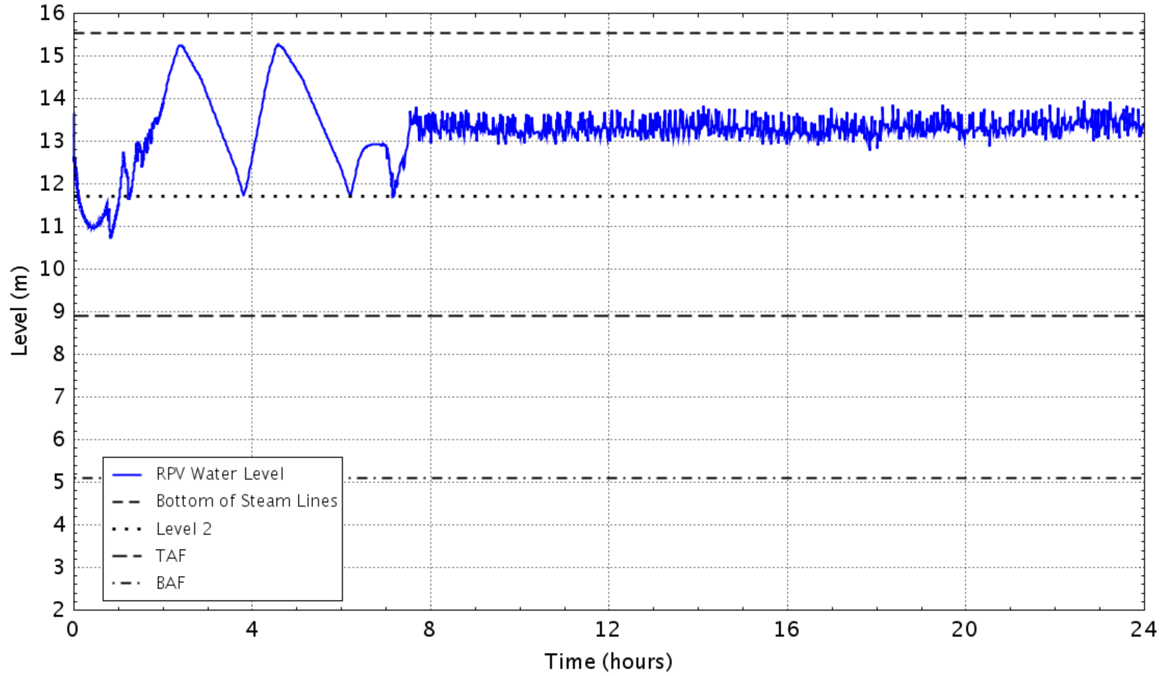

Figure G - 126 RPV Downcomer water level

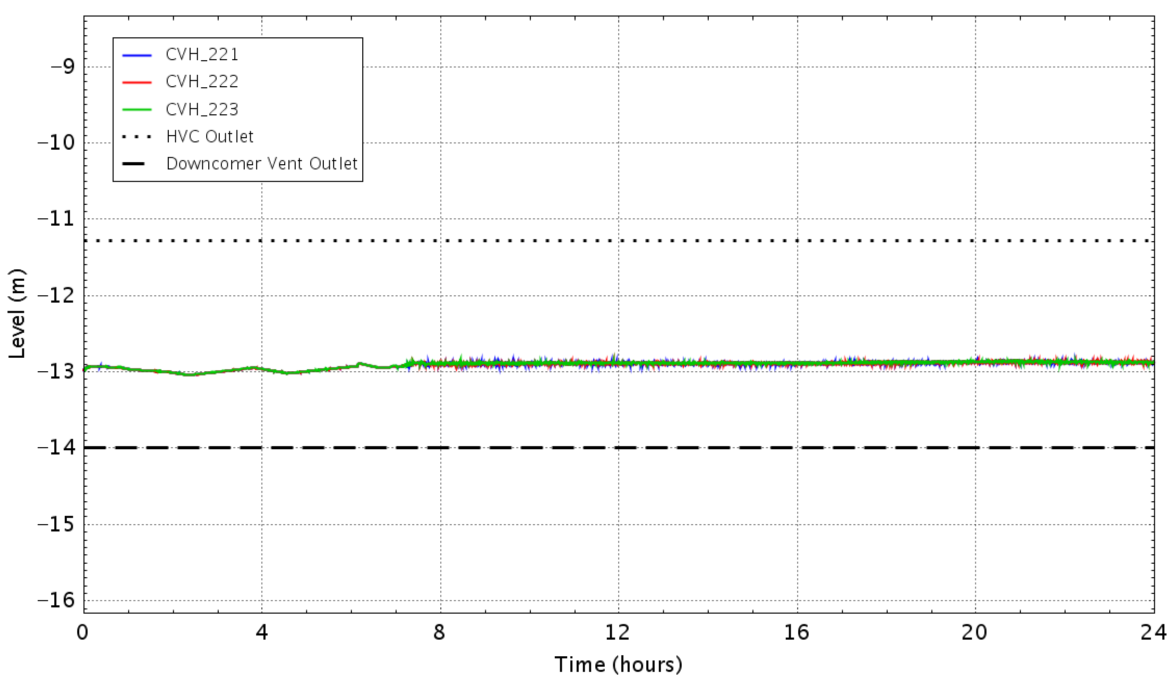

Figure G - 127 Water level in the wetwell

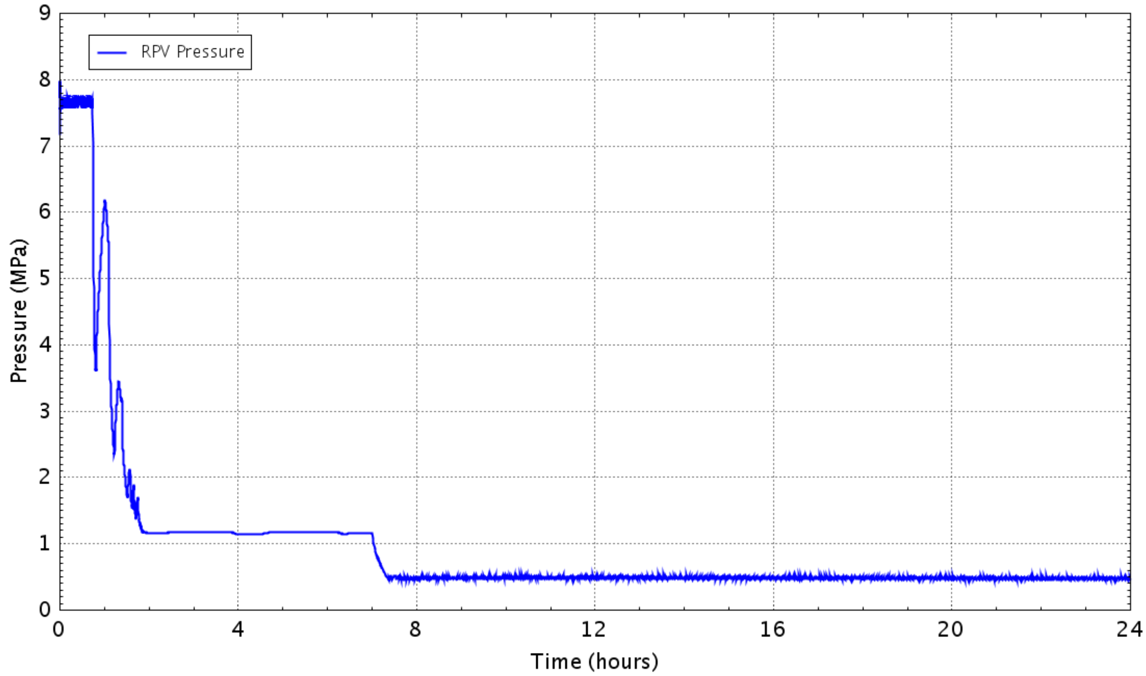

Figure G - 128 Pressure in the RPV

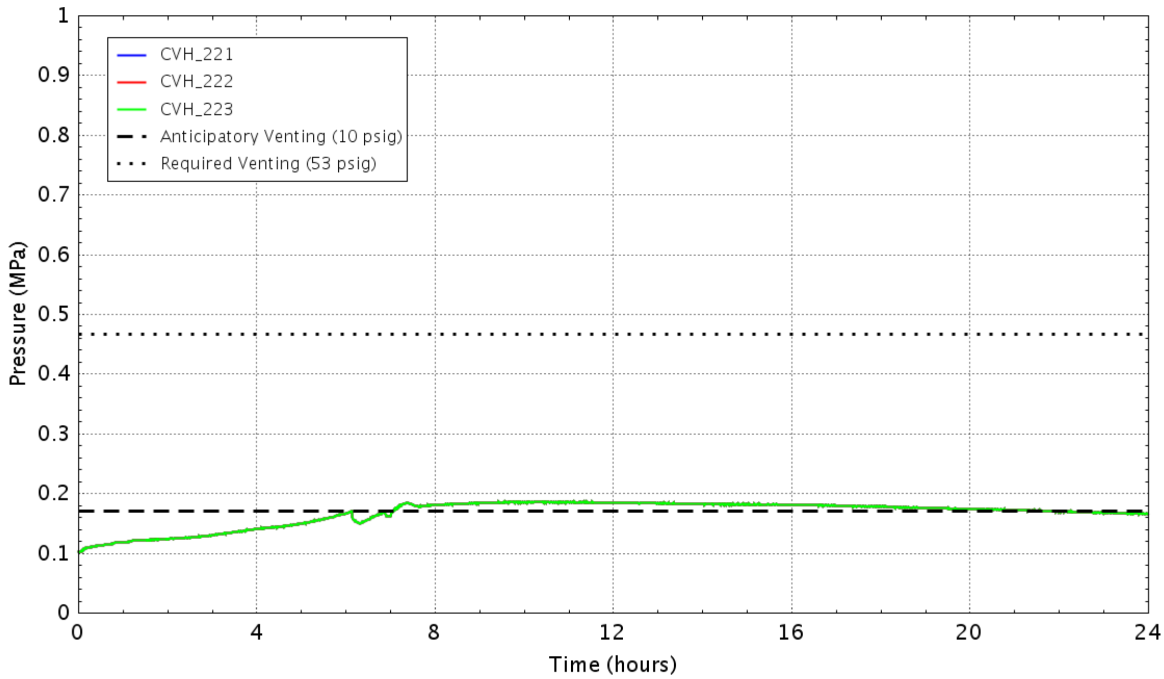

Figure G - 129 Pressure in the wetwell

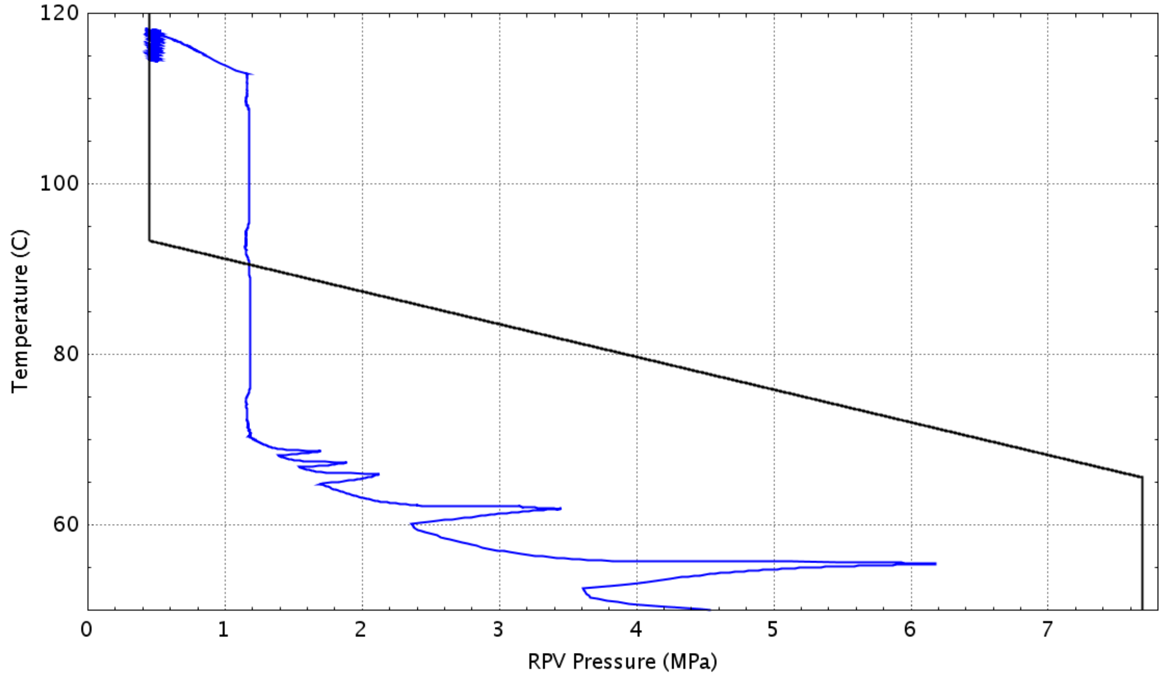

Figure G - 130 Plant status relative to the HCL curve (Graph 4 of the EOPs)

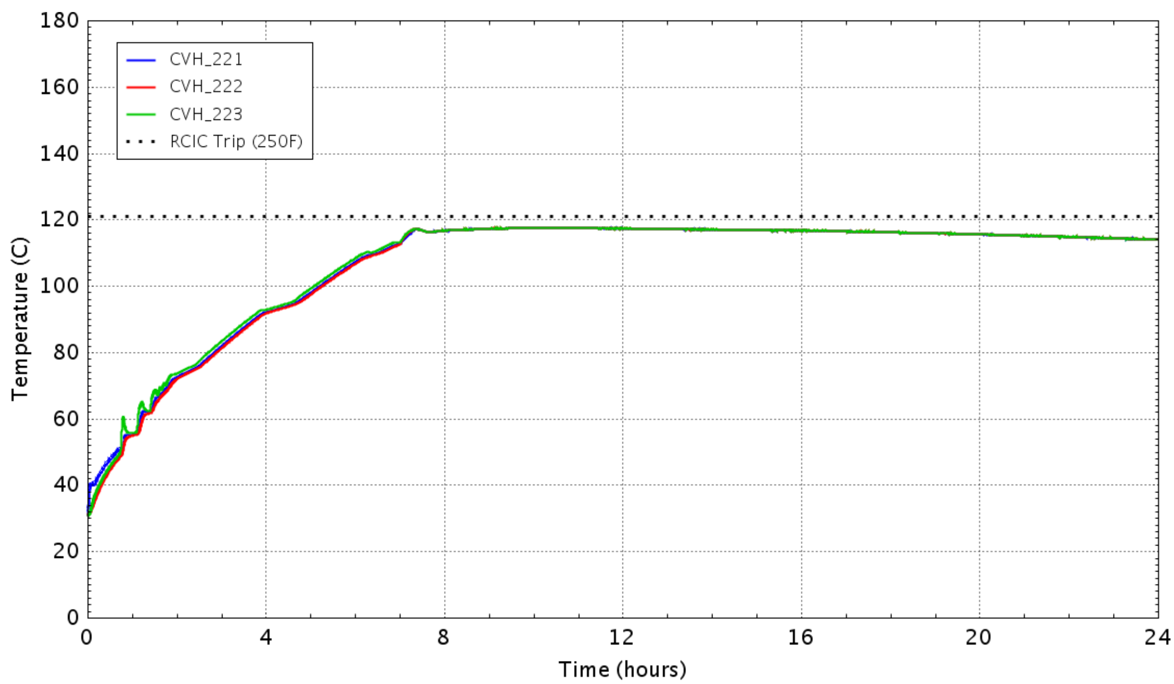

Figure G - 131 Water temperature in the wetwell

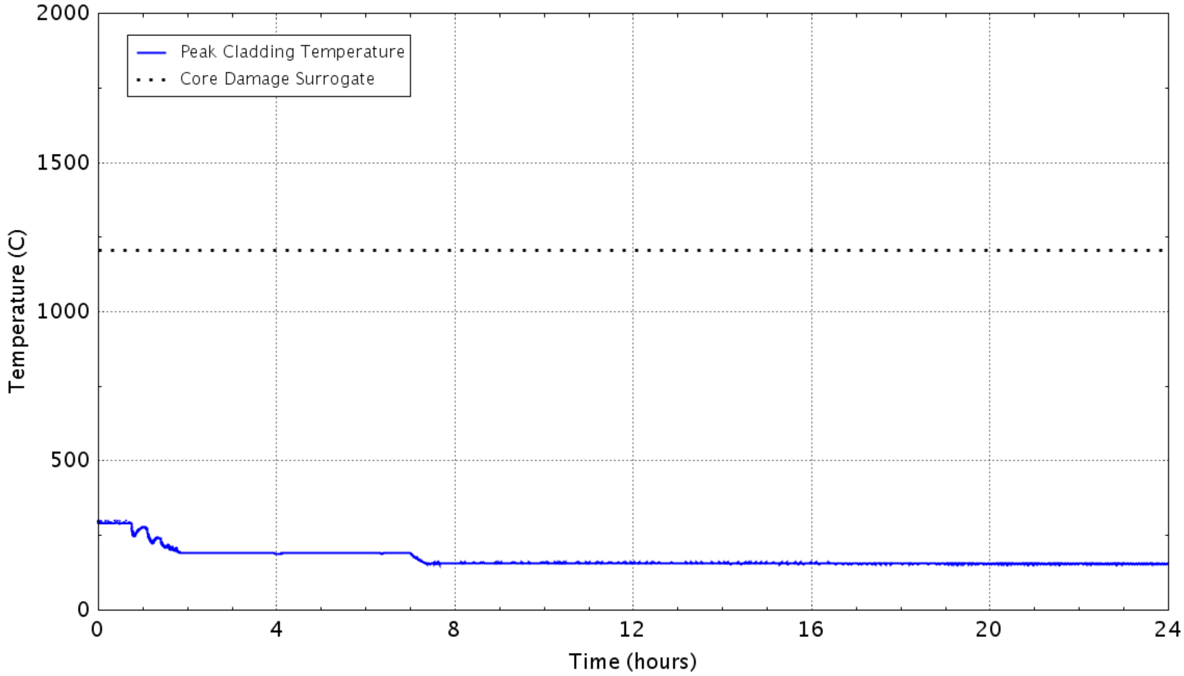

Figure G - 132 Peak temperature of the fuel cladding as a function of time

G.1.12 Case 12: LOOPGR-38-9, RCIC Injection from Wetwell Initially with Swap to FLEX Injection at 7 hrs., Initial Wetwell Level Decreased to 10.4 ft., 200 gpm Recirculation Pump Seal Leakage

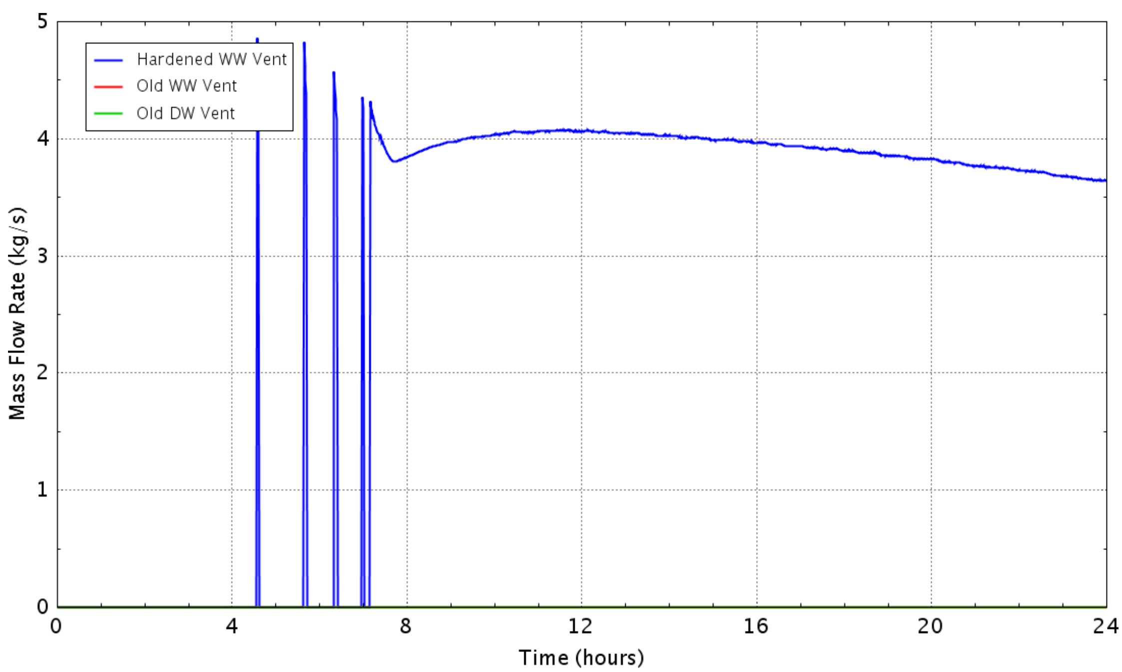

Figure G – 133 Flow rate of the containment vents

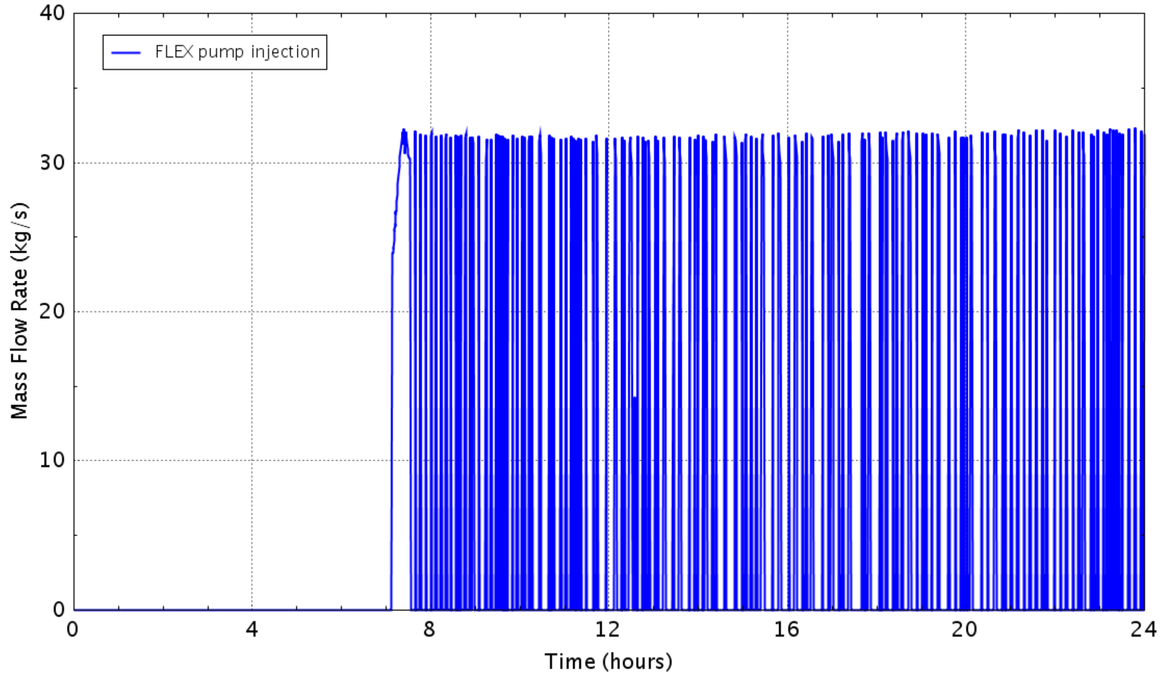

Figure G - 134 Flow rate of the FLEX pump

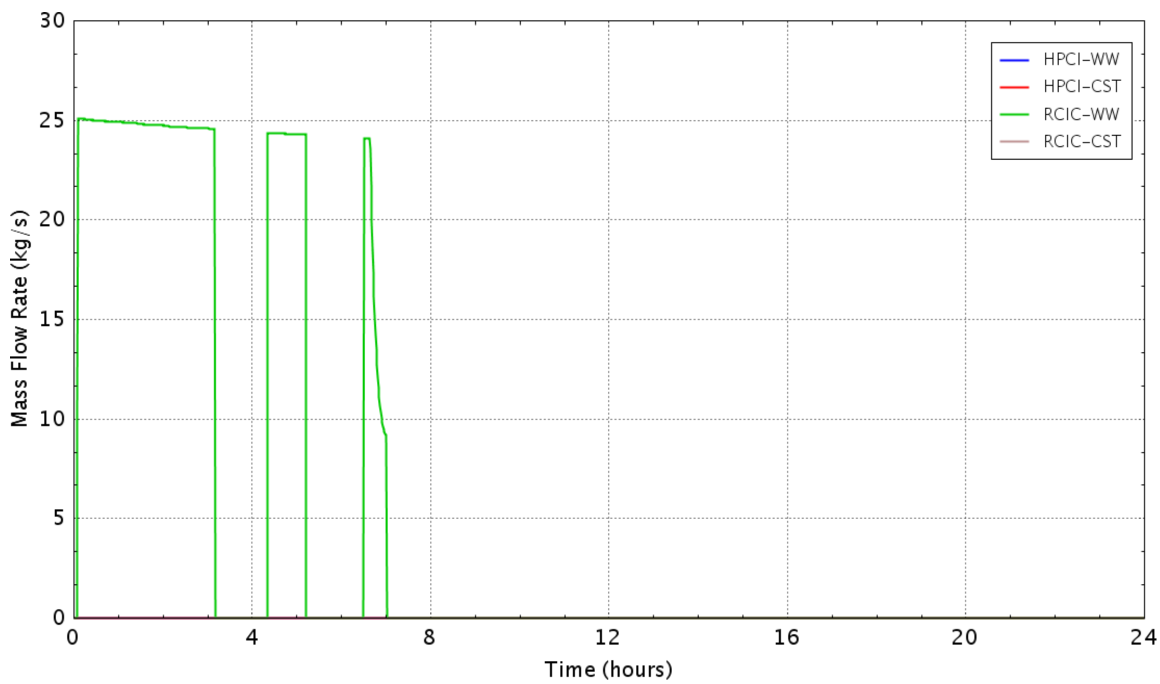

Figure G - 135 Flow rate of the HPCI/RCIC pumps

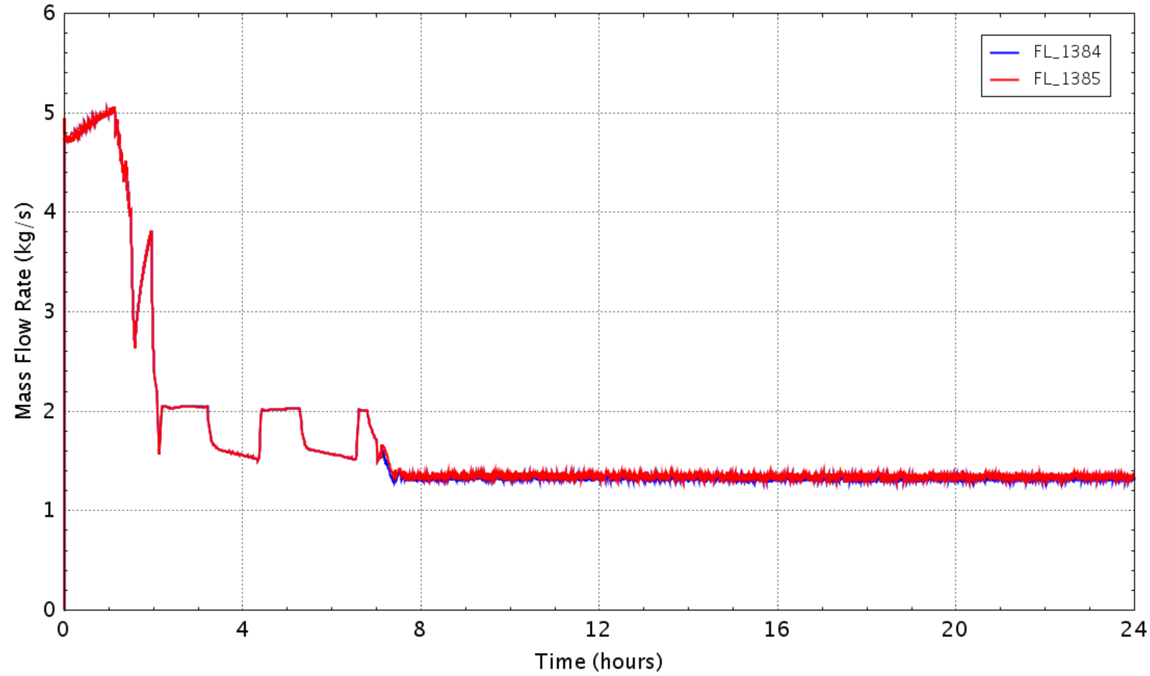

Figure G - 136 Flow rate of the recirculating pump seal leakage

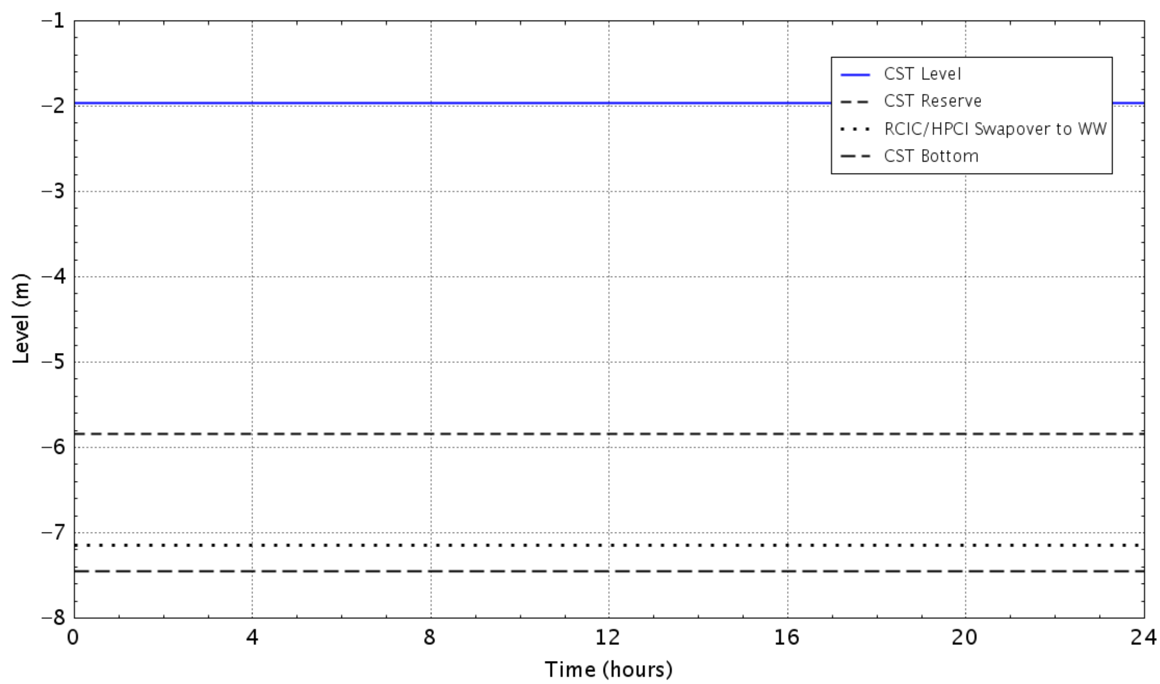

Figure G - 137 Water level in the CST

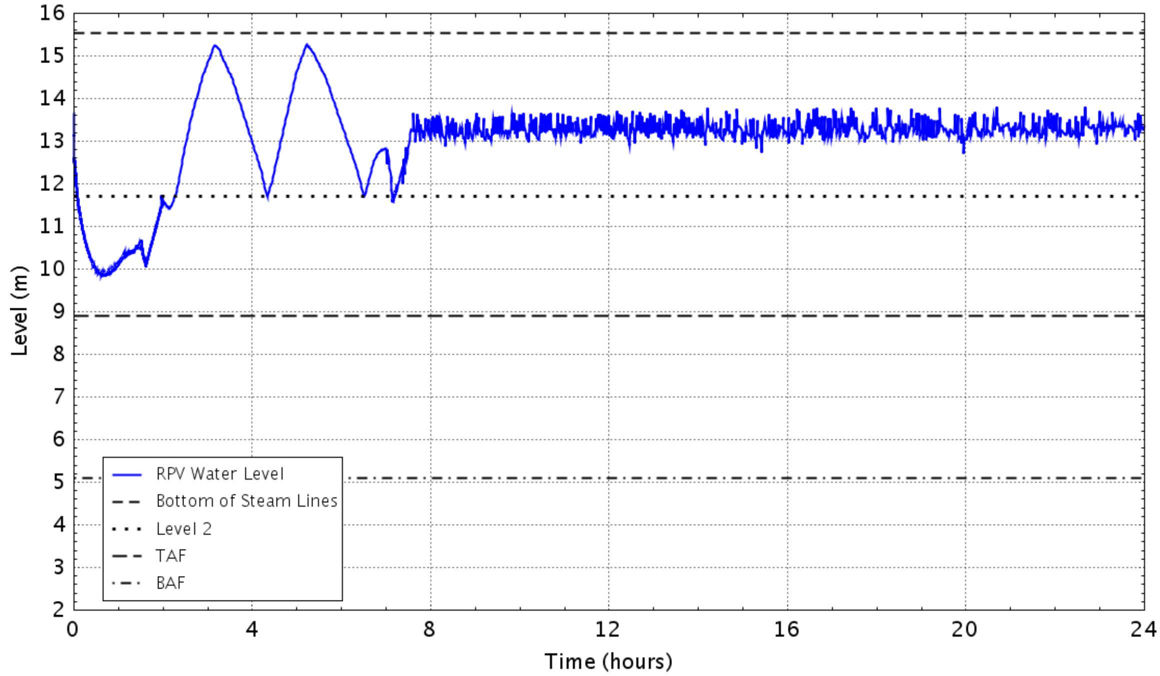

Figure G - 138 RPV Downcomer water level

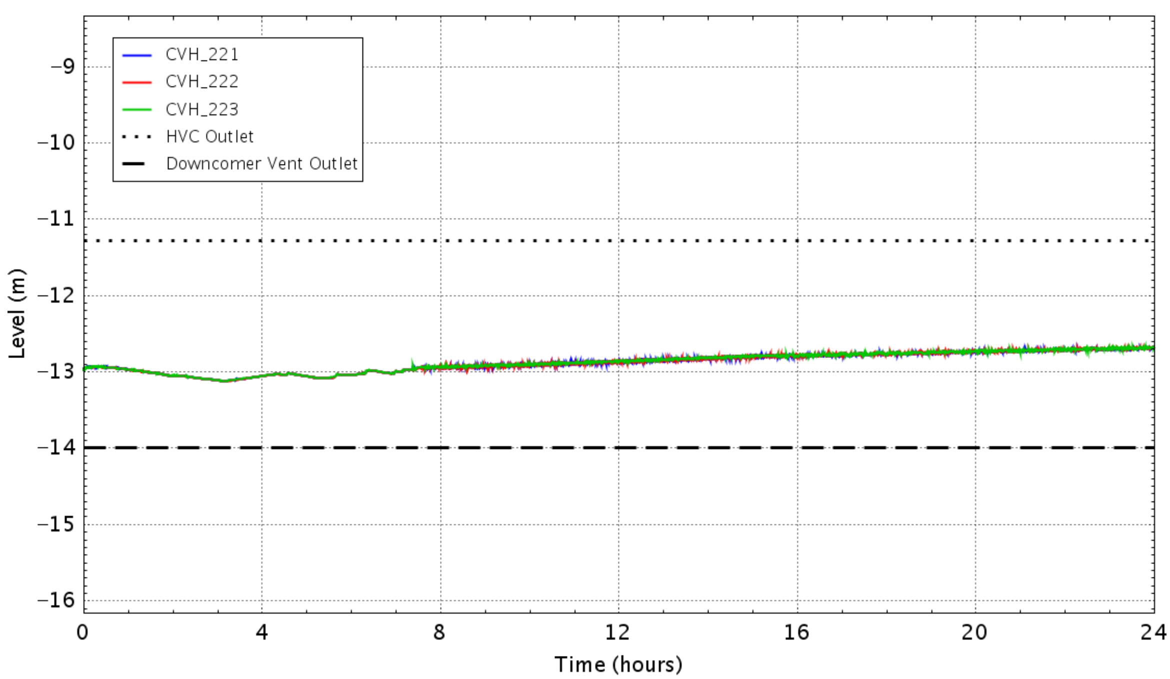

Figure G - 139 Water level in the wetwell

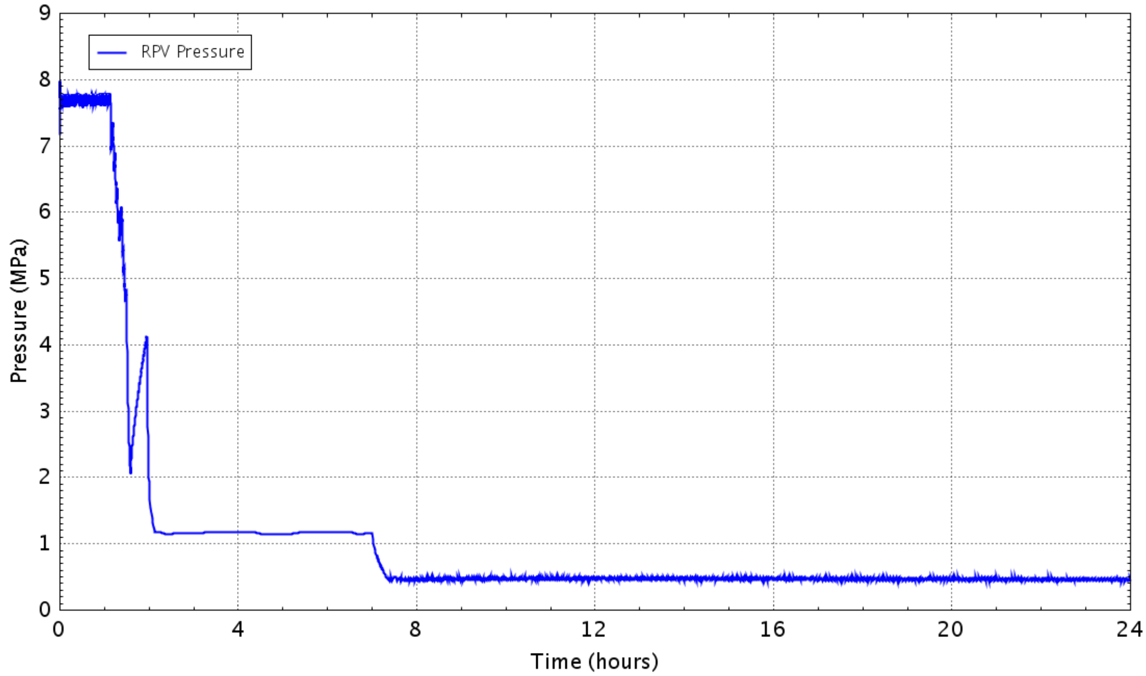

Figure G - 140 Pressure in the RPV

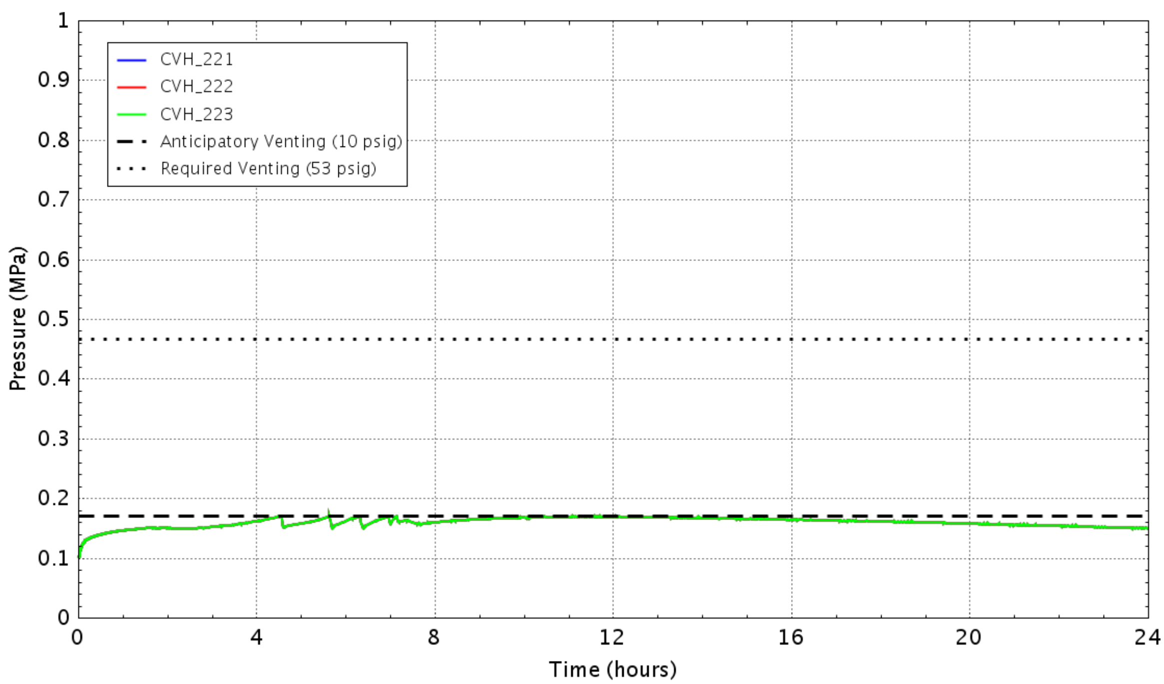

Figure G - 141 Pressure in the wetwell

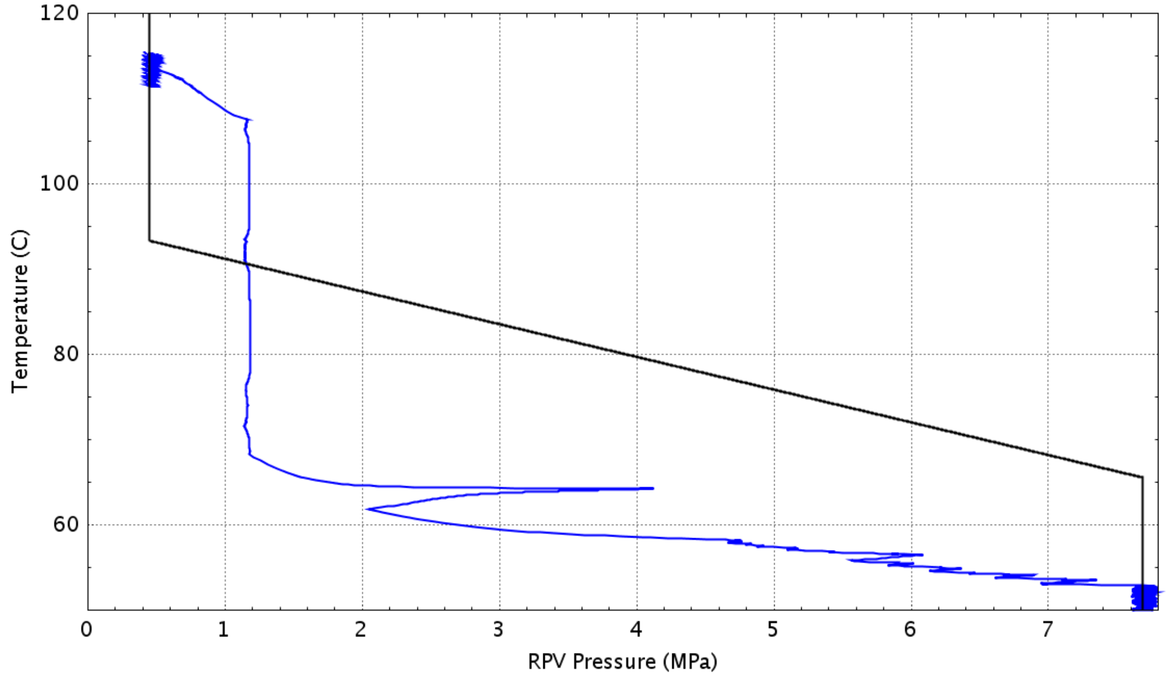

Figure G - 142 Plant status relative to the HCL curve (Graph 4 of the EOPs)

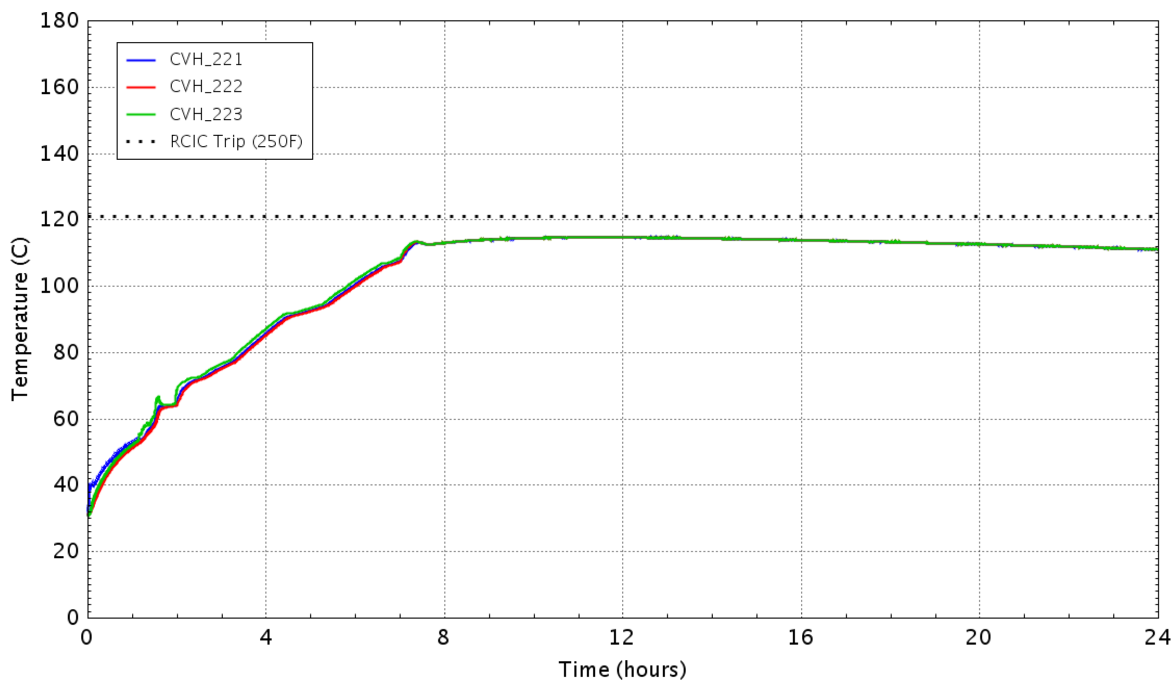

Figure G - 143 Water temperature in the wetwell

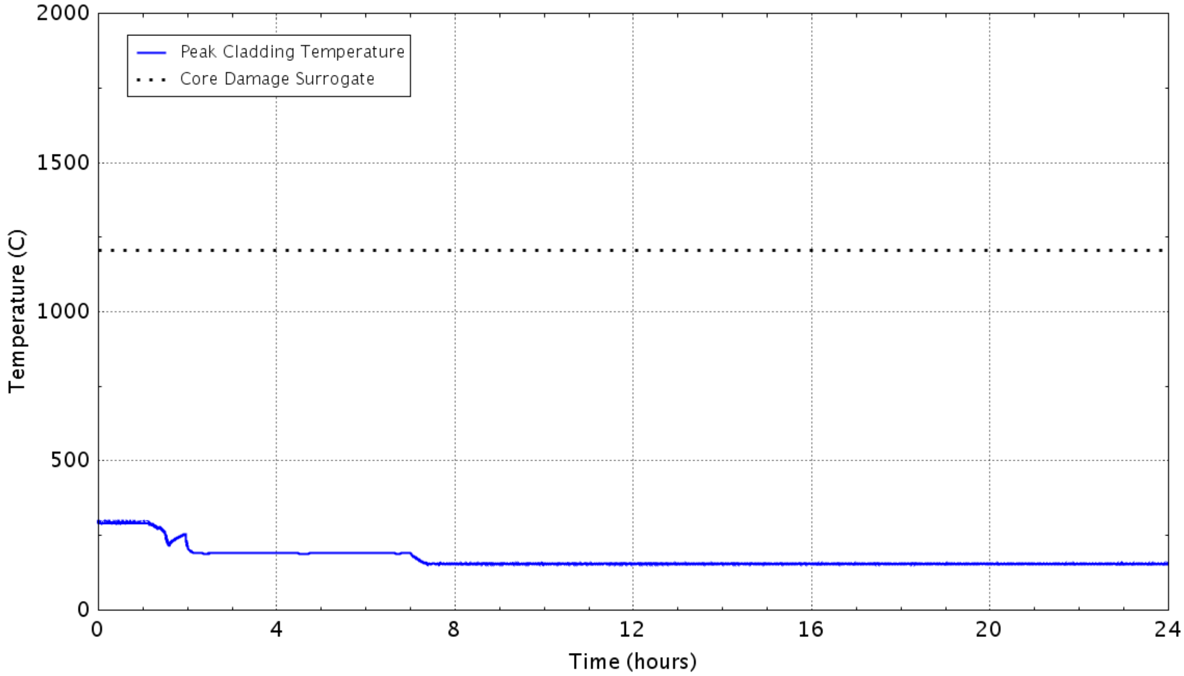

Figure G - 144 Peak temperature of the fuel cladding as a function of time

G.2 LOMFW Scenarios

G.2.1 Case 13: LOMFW-25, RCIC Lost at 4 hrs., Nominal CRDHS Injection, Nominal Recirculation Pump Seal Leakage

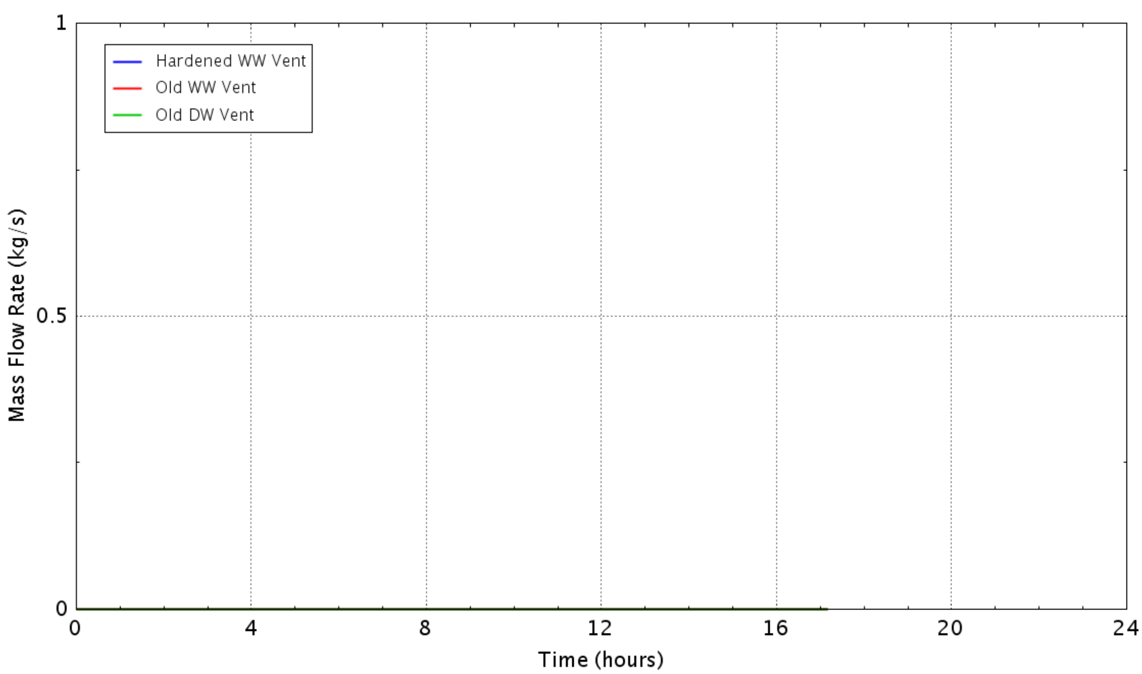

Figure G - 145 Flow rate of the containment vents

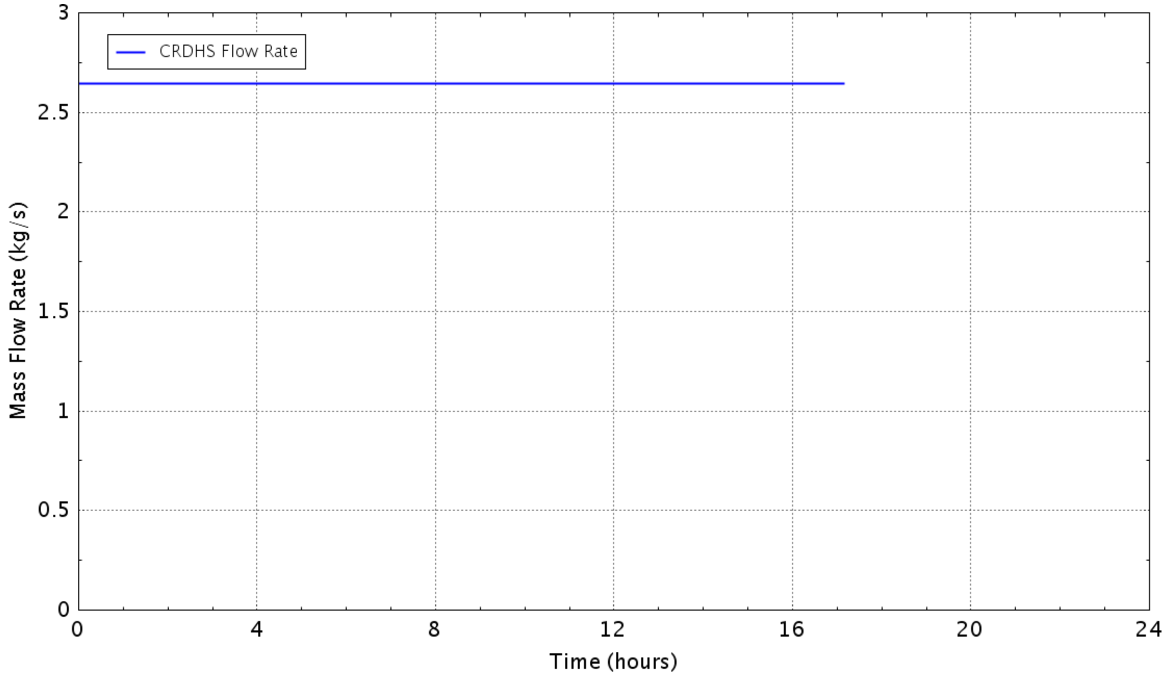

Figure G - 146 Flow rate of the control rod drive hydraulic system

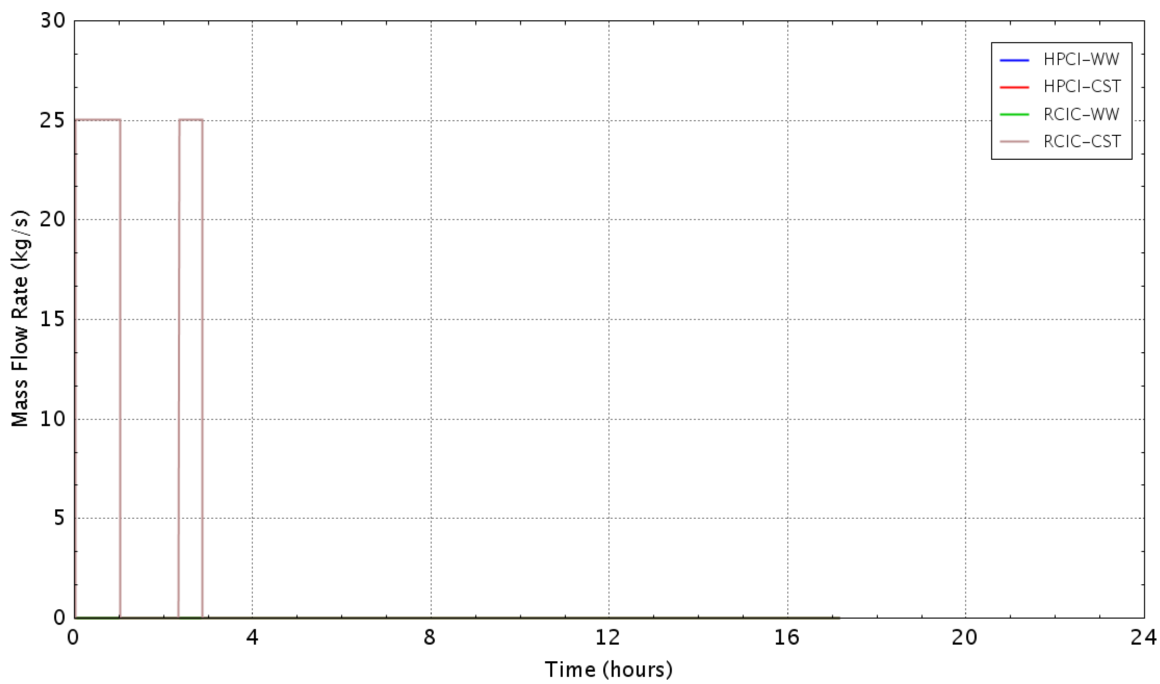

Figure G - 147 Flow rate of the HPCI/RCIC pumps

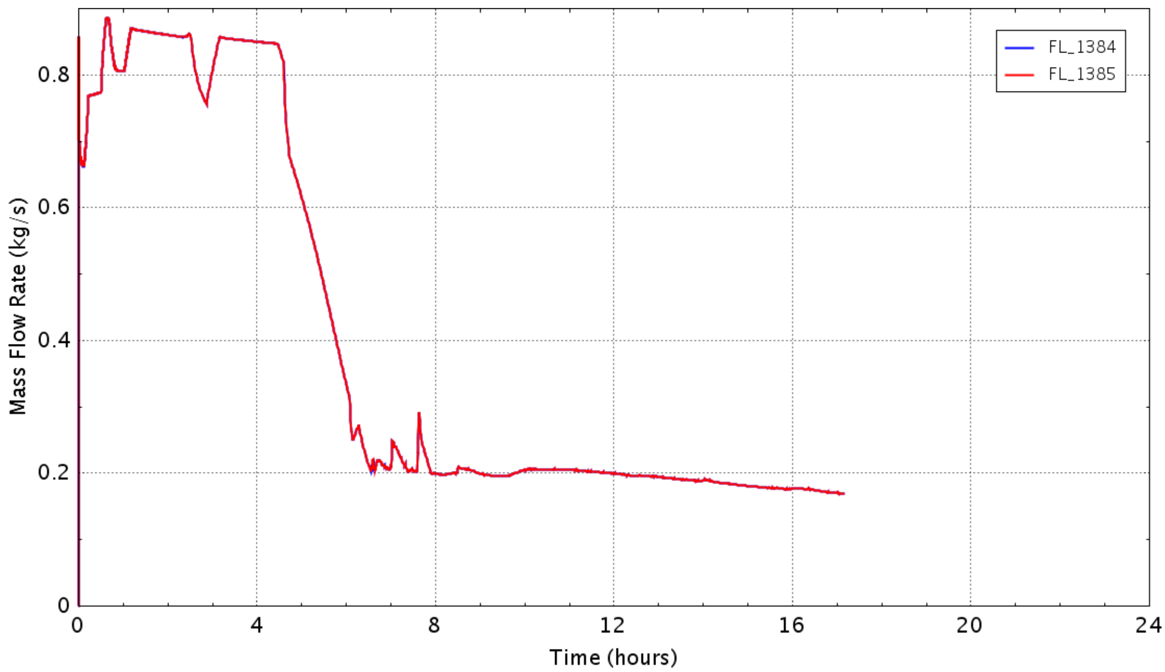

Figure G - 148 Flow rate of the recirculating pump seal leakage

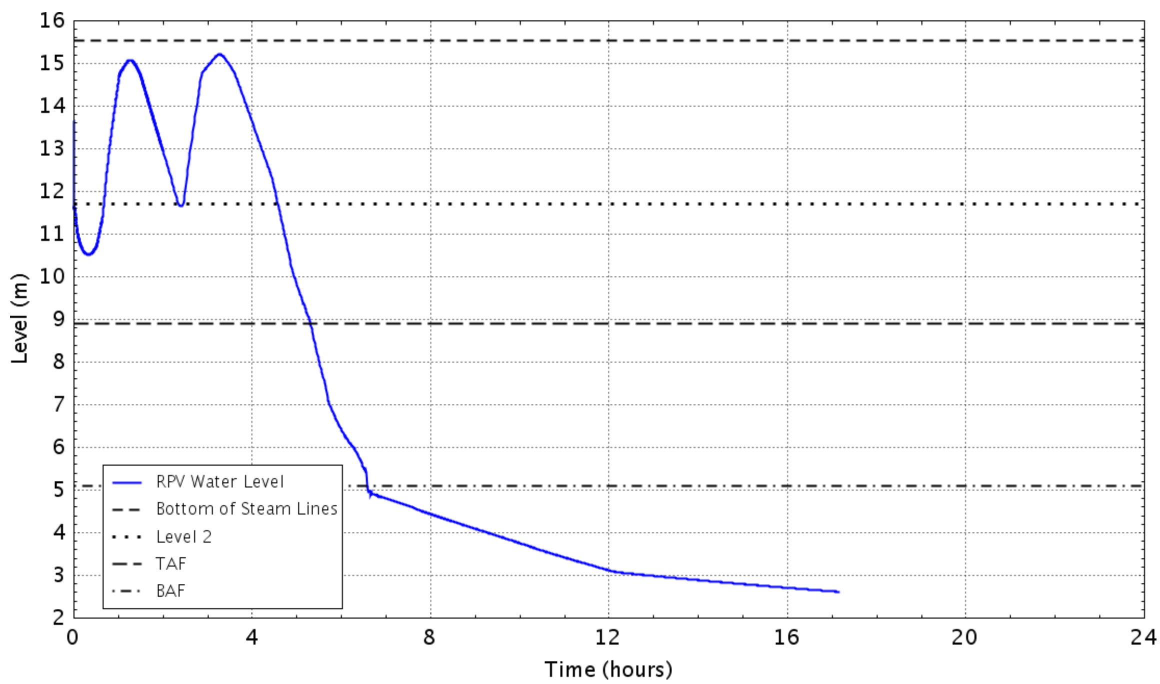

Figure G - 149 RPV Downcomer water level

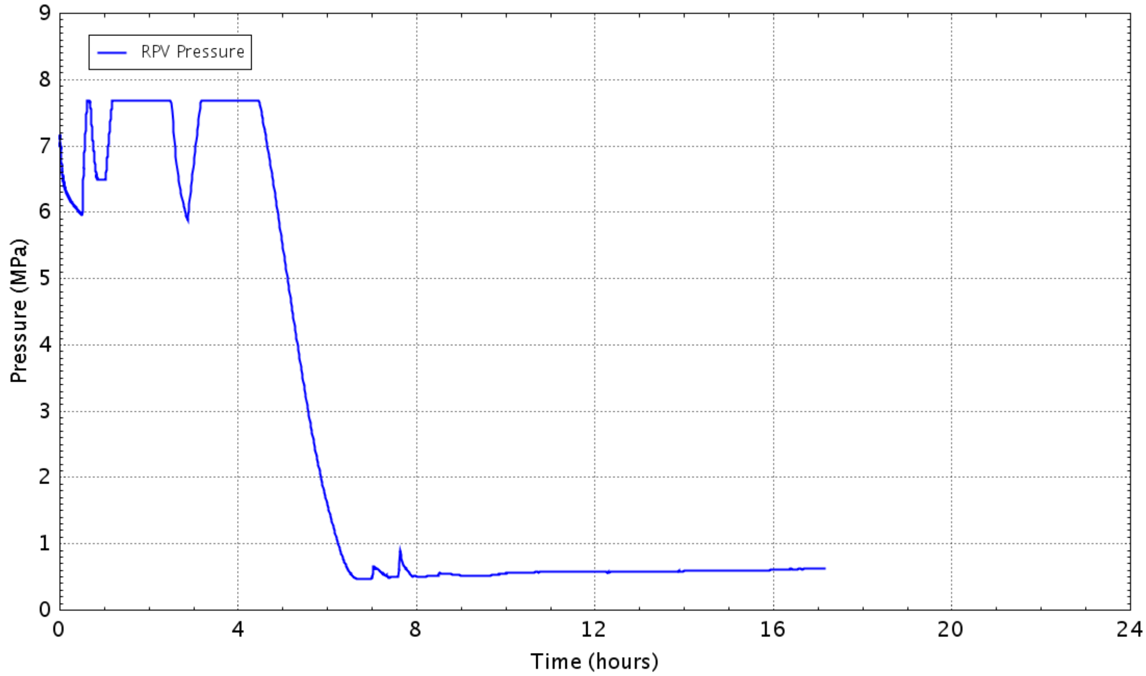

Figure G - 150 Pressure in the RPV

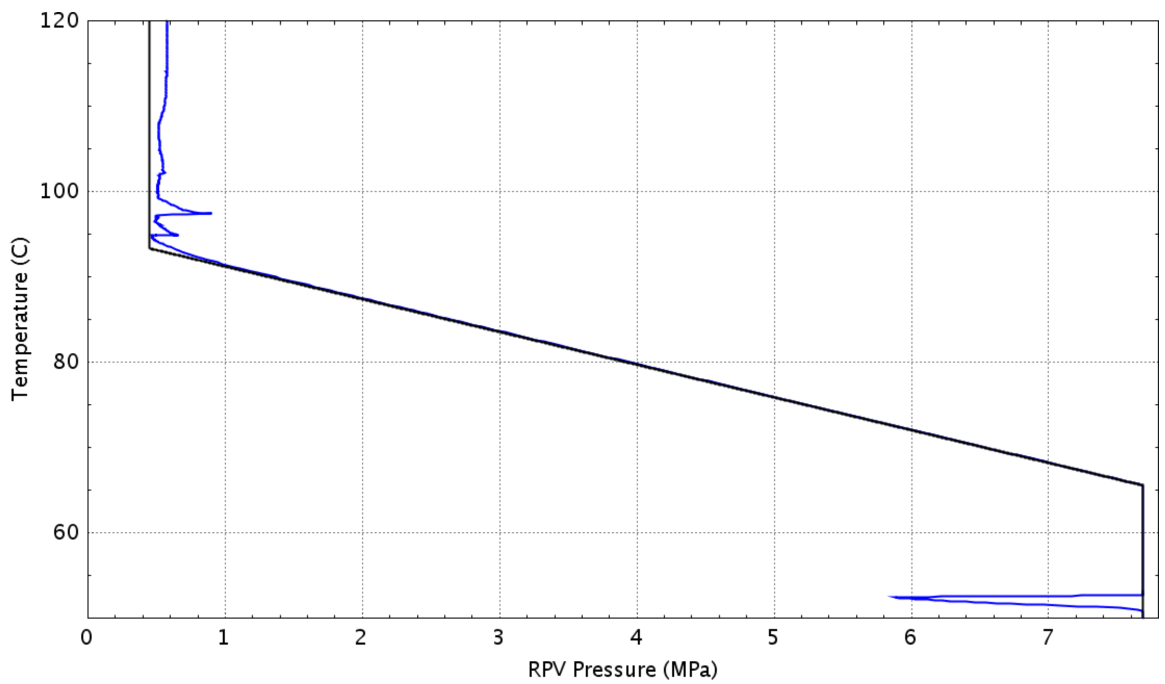

Figure G - 151 Plant status relative to the HCL curve (Graph 4 of the EOPs)

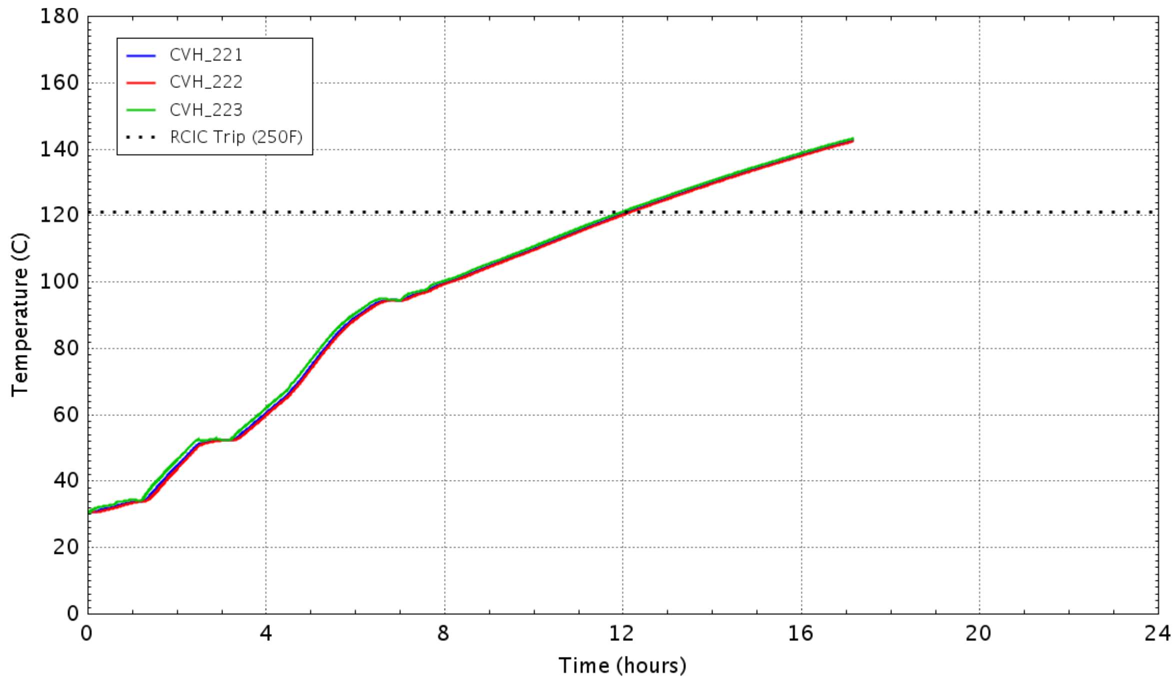

Figure G - 152 Water temperature in the wetwell

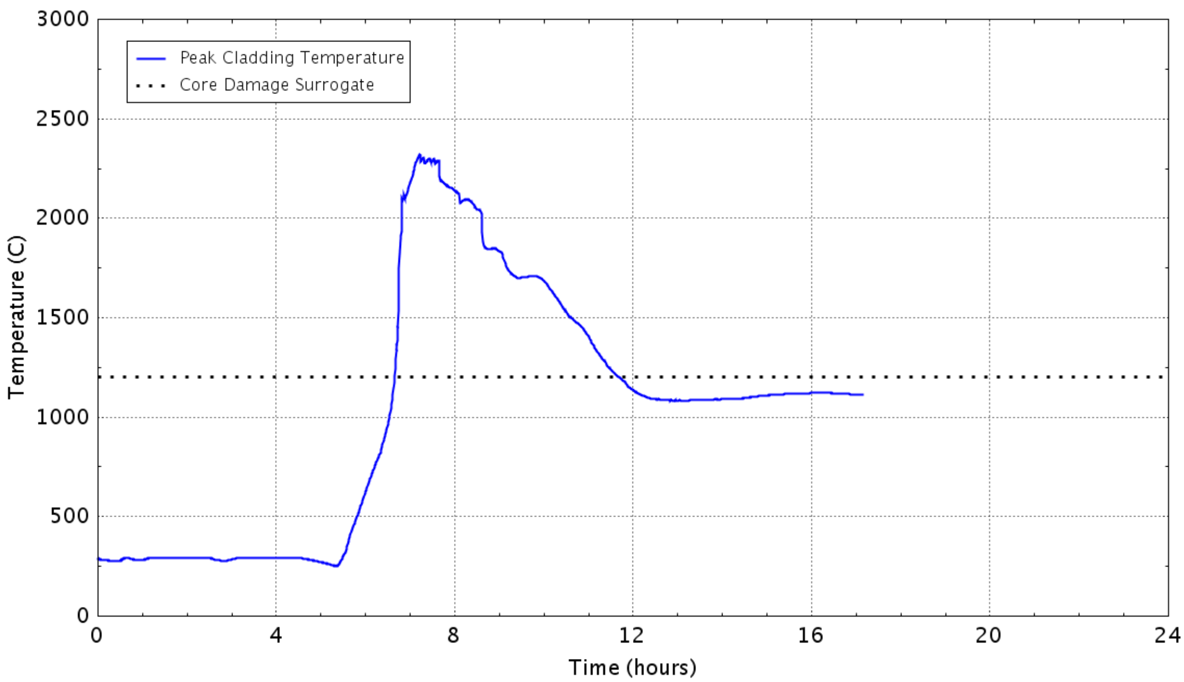

Figure G - 153 Peak temperature of the fuel cladding as a function of time

G.2.2 Case 14: LOMFW-25, RCIC Lost at 4 hrs., Nominal CRDHS Injection, 200 gpm Recirculation Pump Seal Leakage

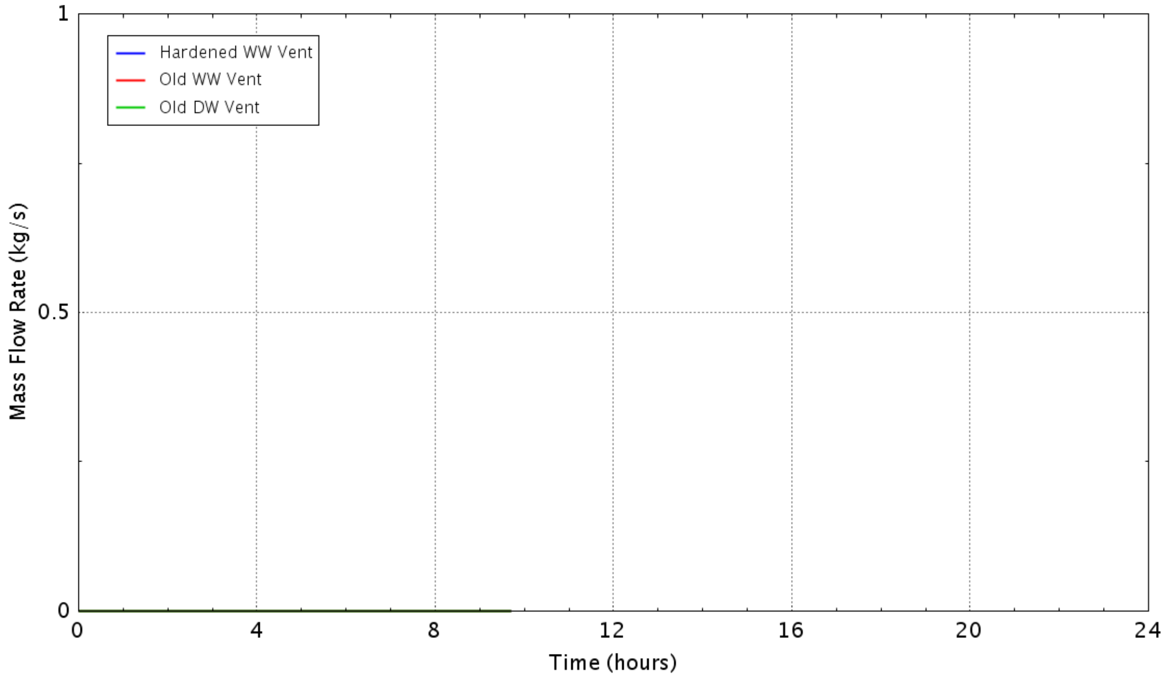

Figure G - 154 Flow rate of the containment vents

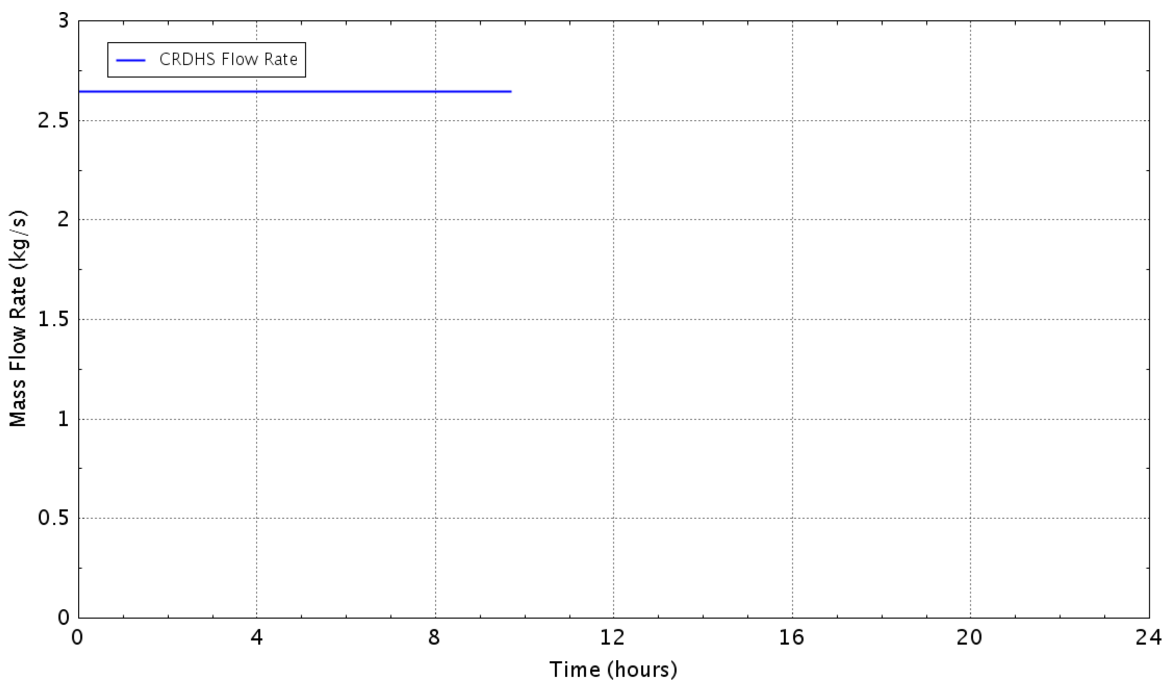

Figure G - 155 Flow rate of the control rod drive hydraulic system

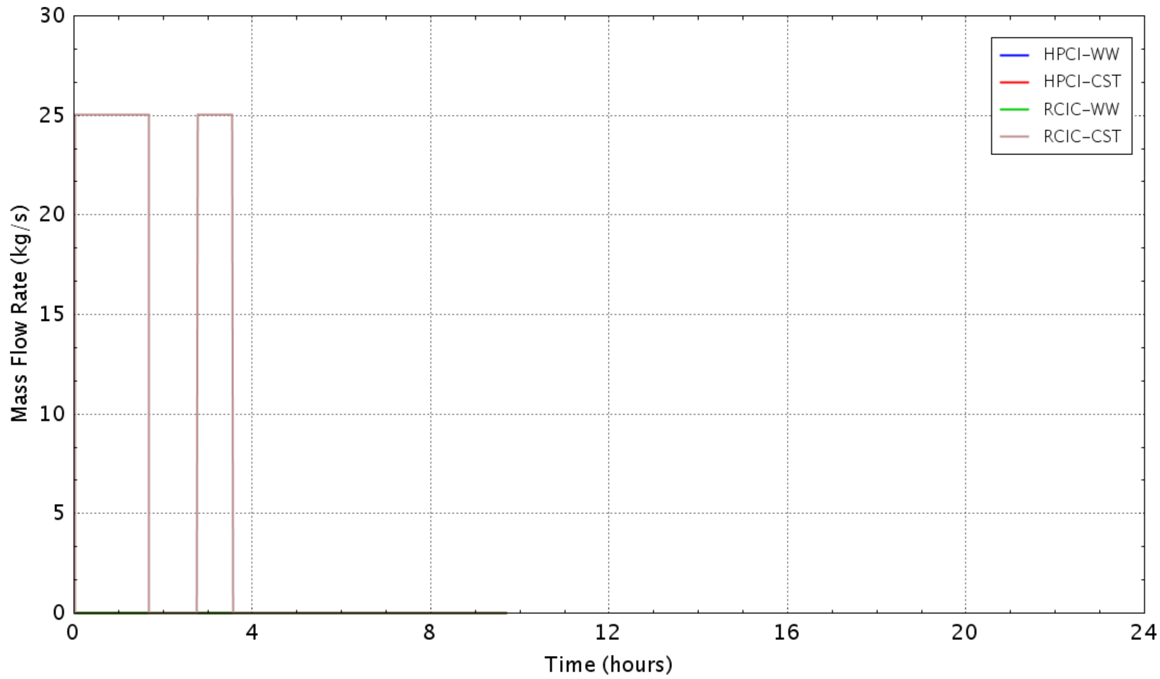

Figure G - 156 Flow rate of the HPCI/RCIC pumps

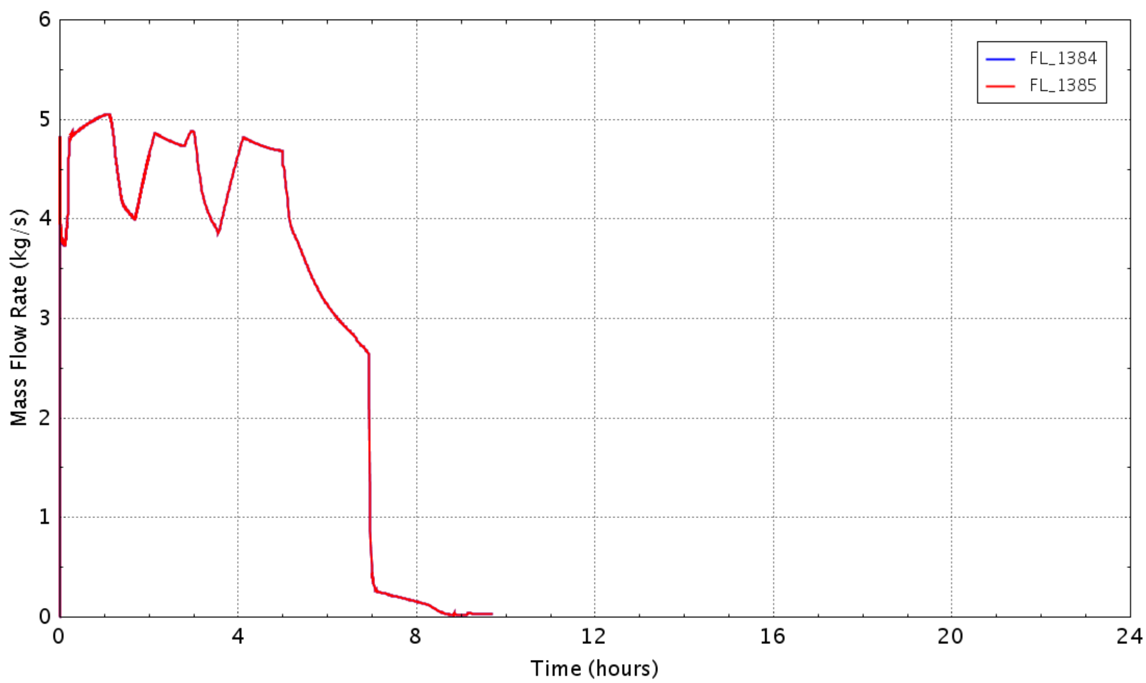

Figure G - 157 Flow rate of the recirculating pump seal leakage

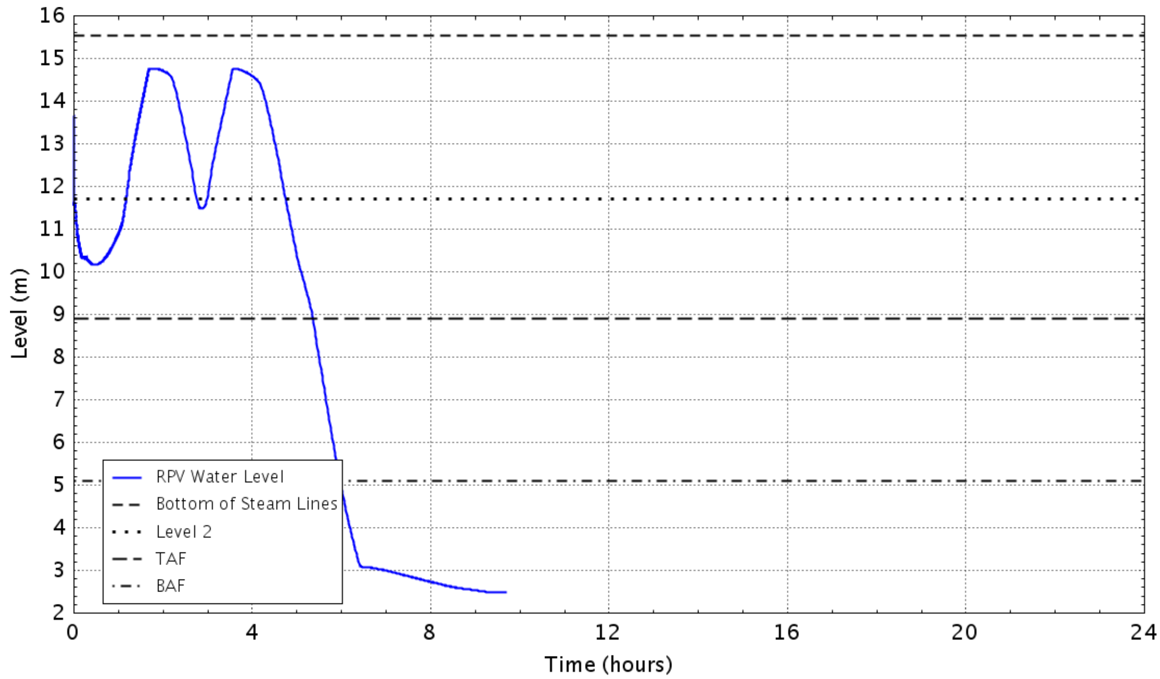

Figure G - 158 RPV Downcomer water level

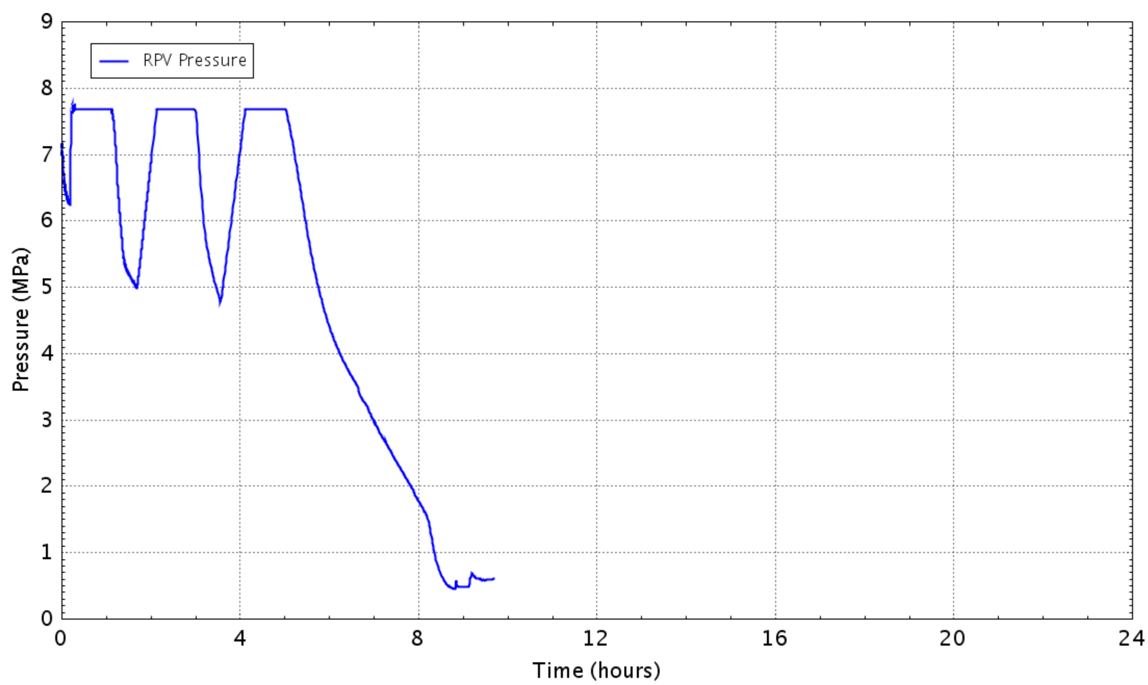

Figure G - 159 Pressure in the RPV

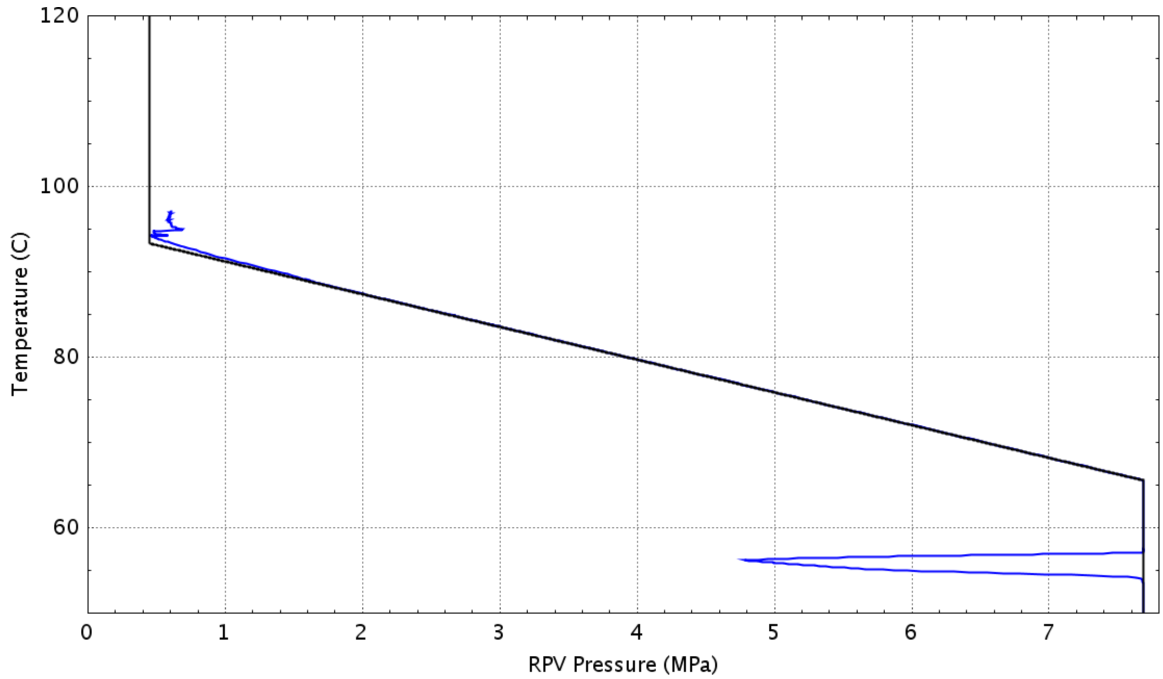

Figure G - 160Plant status relative to the HCL curve (Graph 4 of the EOPs)

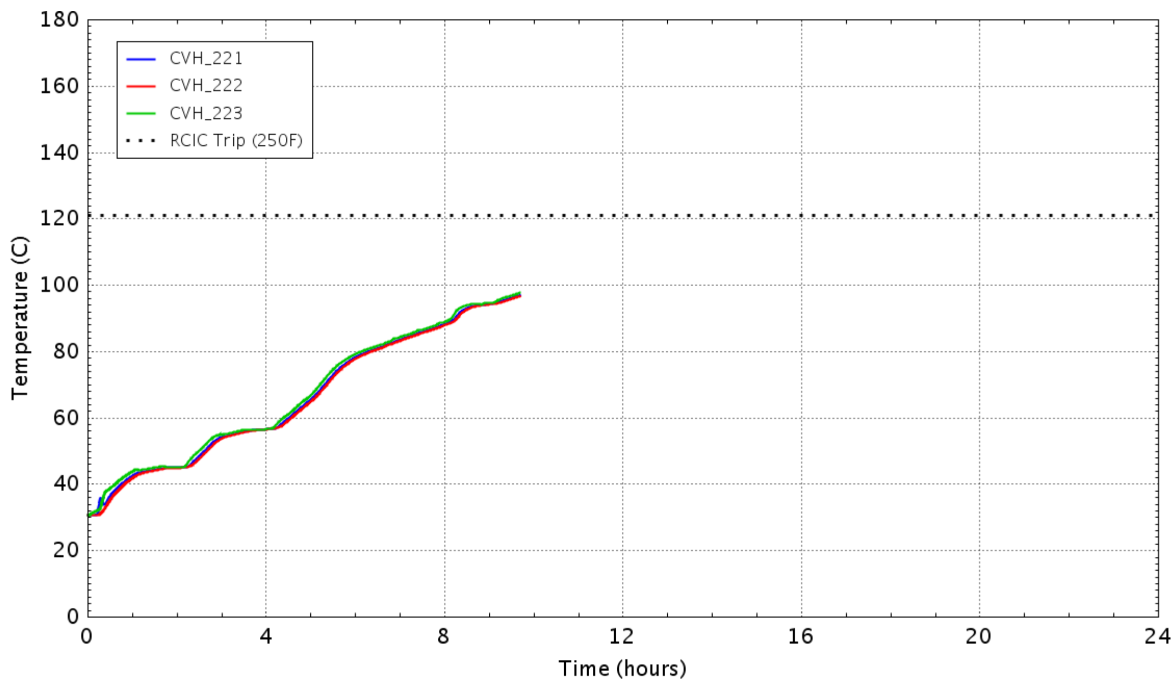

Figure G - 161 Water temperature in the wetwell

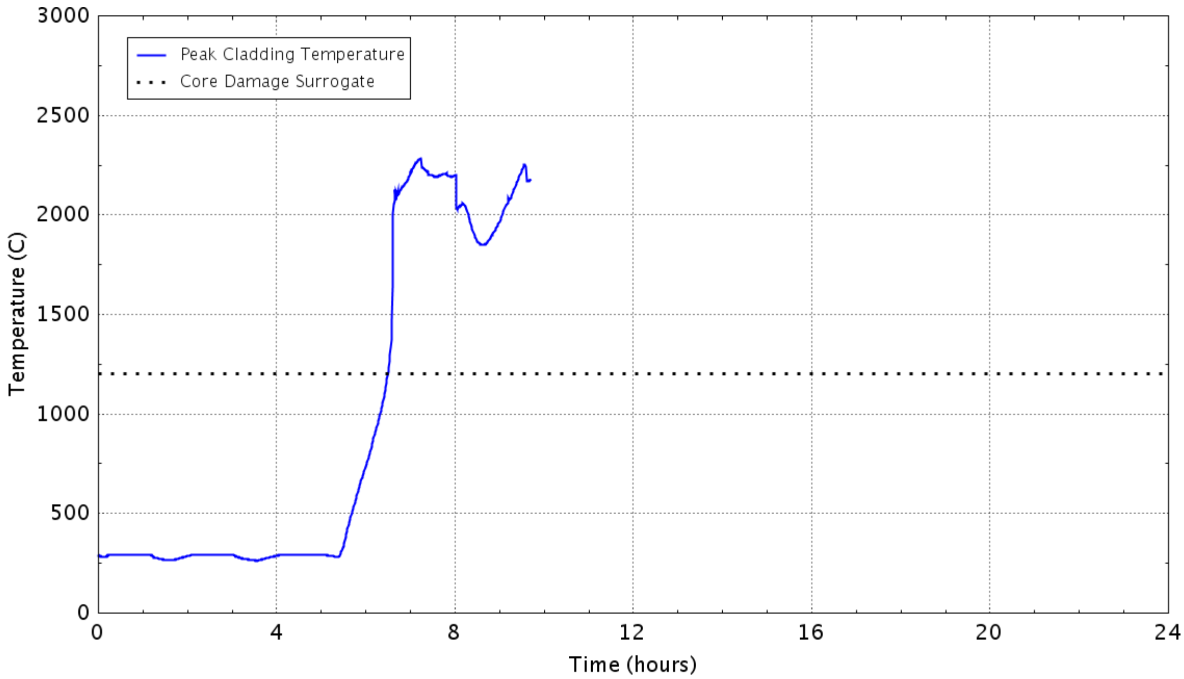

Figure G - 162 Peak temperature of the fuel cladding as a function of time

G.2.3 Case 15: LOMFW-25, RCIC Lost at 4 hrs., One Train ofCRDHS Injection, Nominal Recirculation Pump Seal Leakage

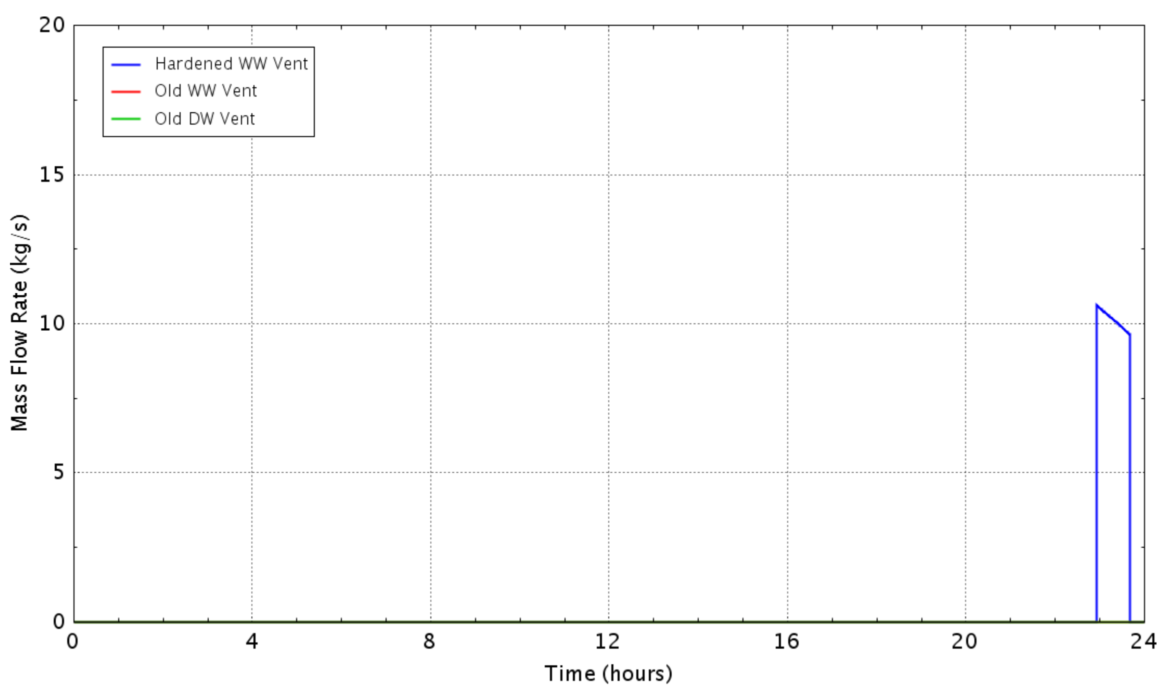

Figure G - 163 Flow rate of the containment vents

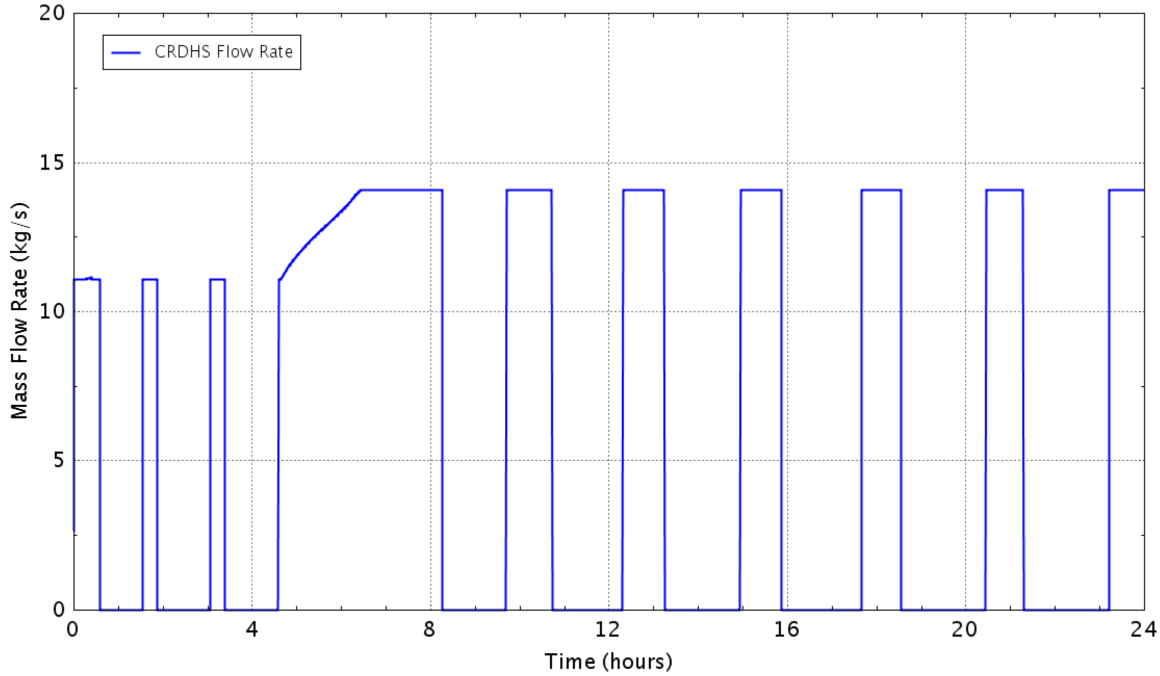

Figure G - 164 Flow rate of the control rod drive hydraulic system

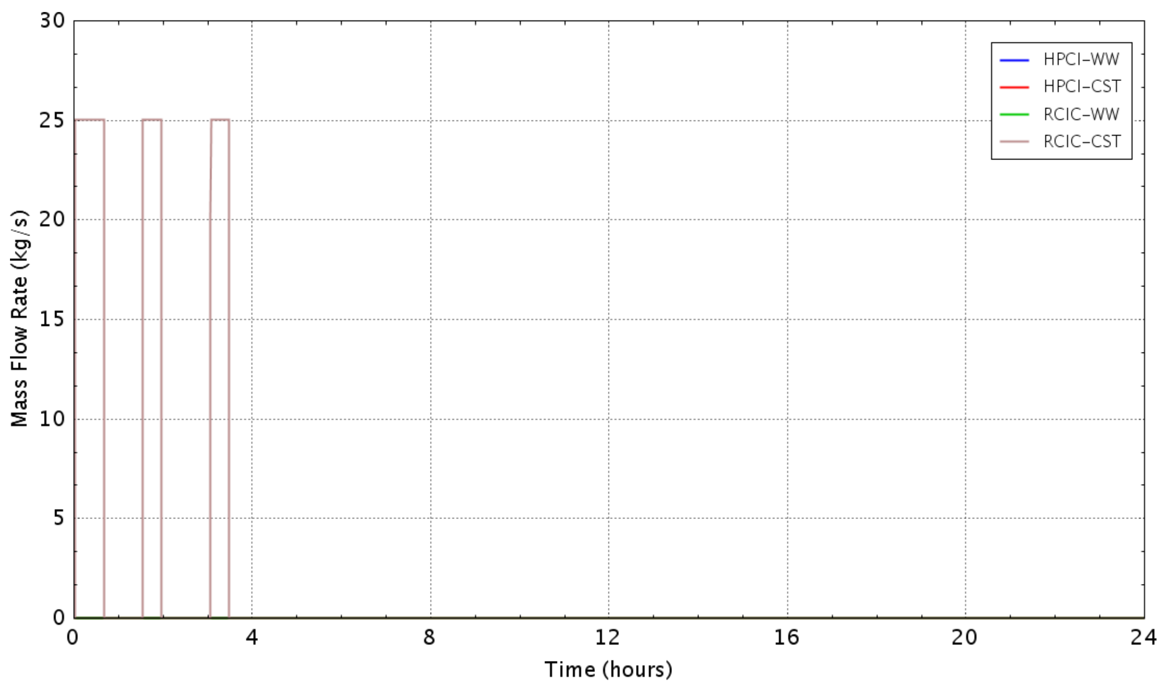

Figure G - 165 Flow rate of the HPCI/RCIC pumps

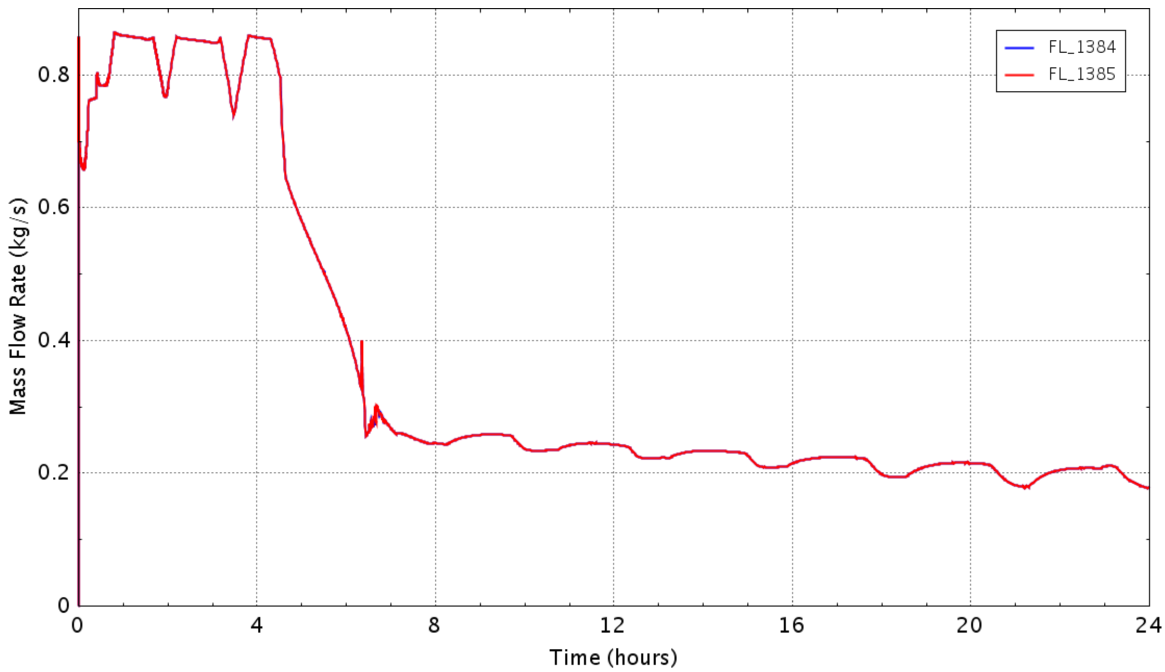

Figure G - 166 Flow rate of the recirculating pump seal leakage

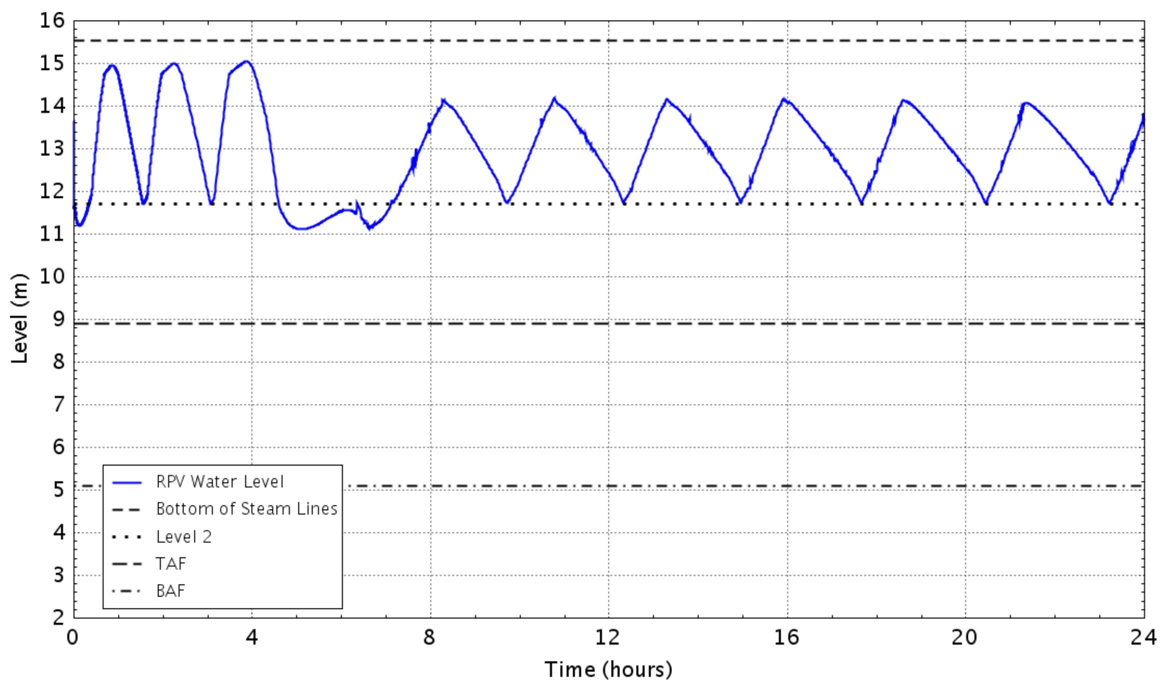

Figure G - 167 RPV Downcomer water level

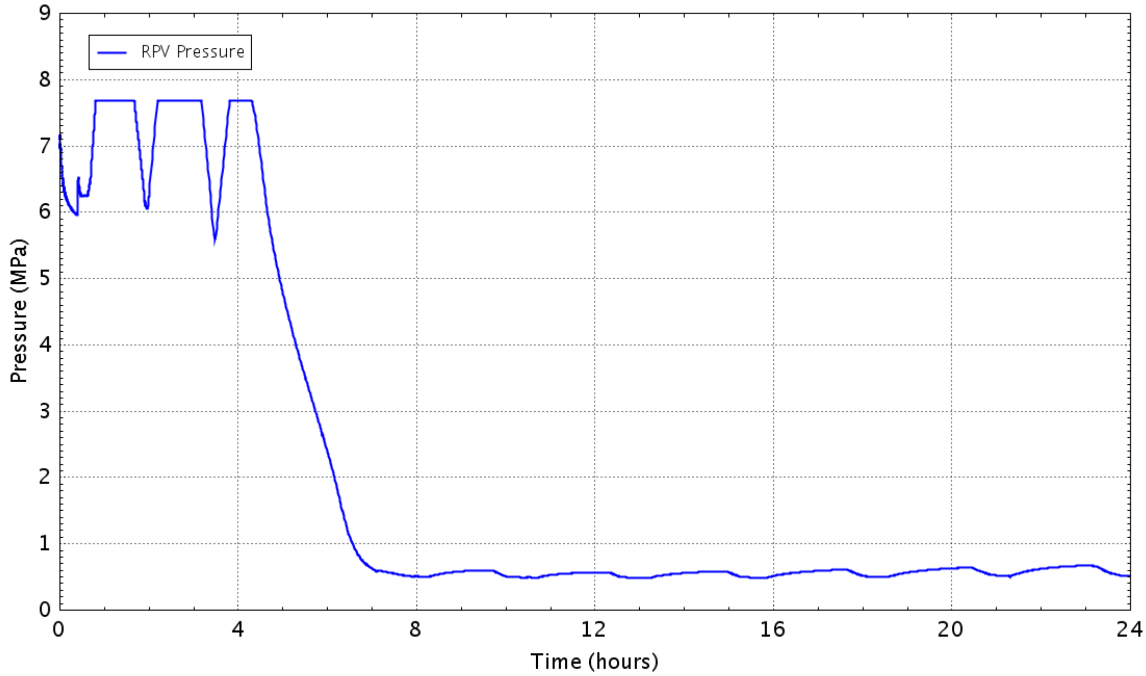

Figure G - 168 Pressure in the RPV

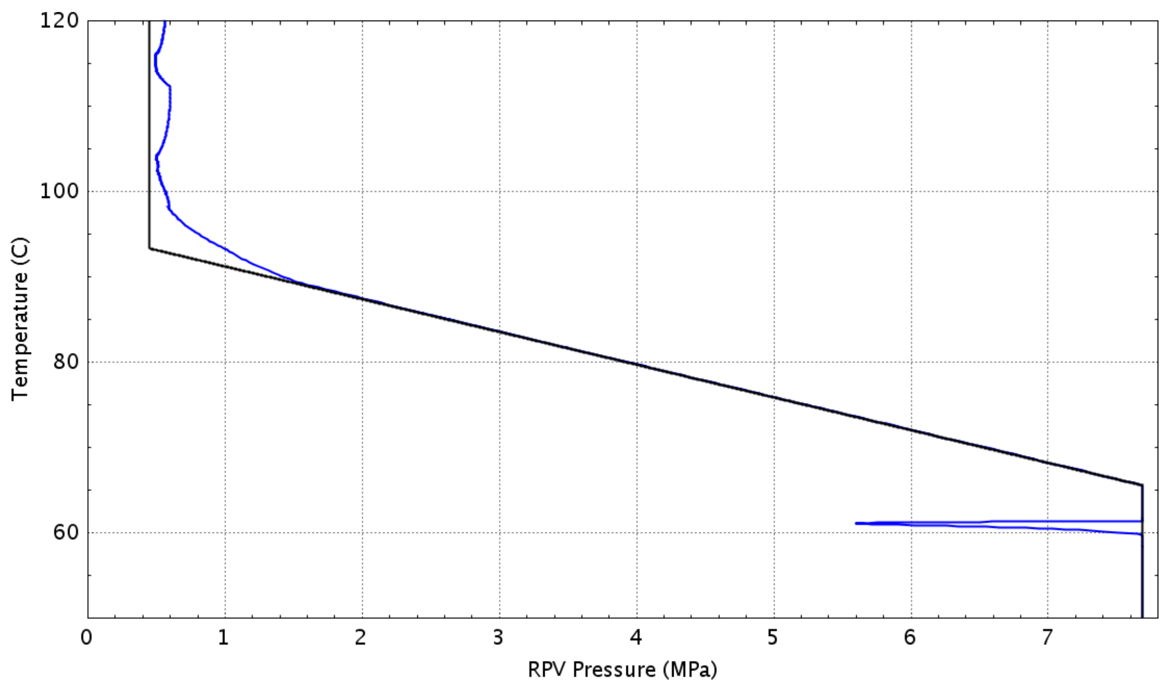

Figure G - 169 Plant status relative to the HCL curve (Graph 4 of the EOPs)

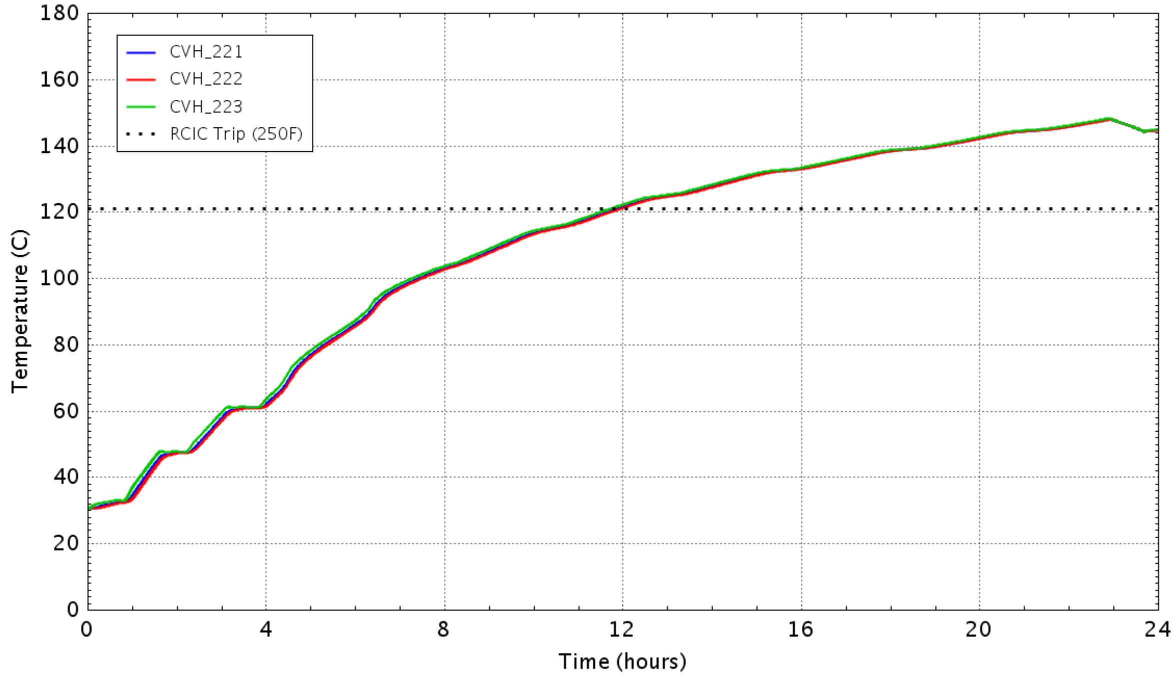

Figure G - 170 Water temperature in the wetwell

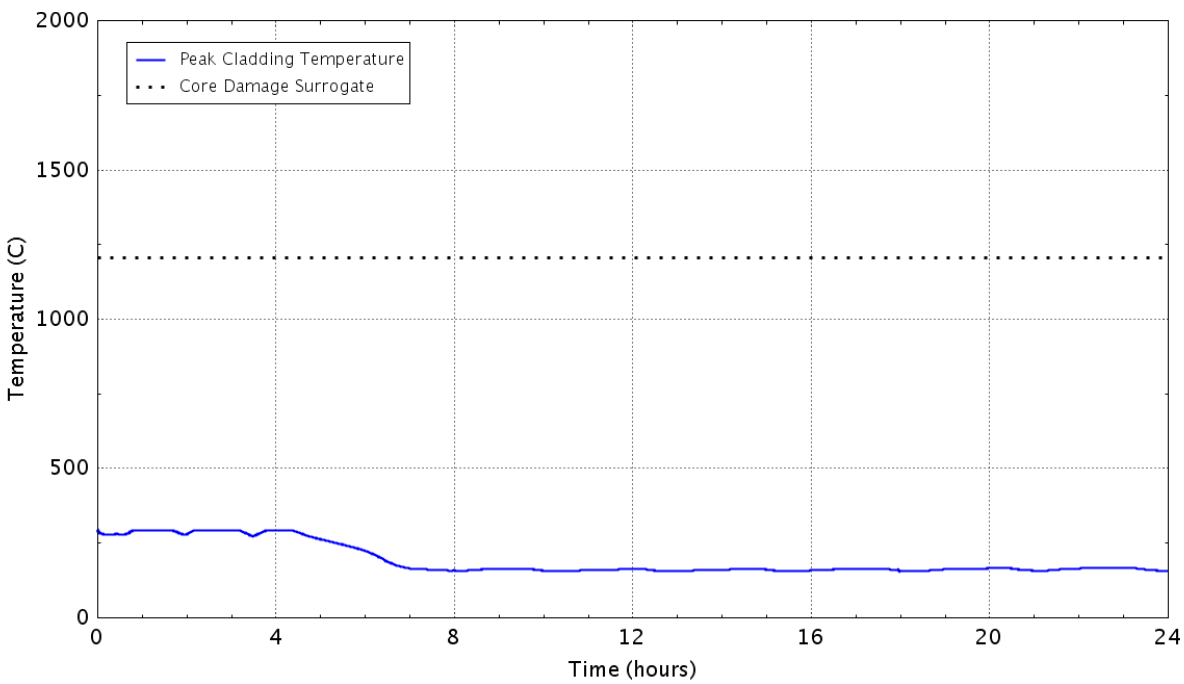

Figure G - 171 Peak temperature of the fuel cladding as a function of time

G.2.4 Case 16: LOMFW-25, RCIC Lost at 4 hrs., One Train ofCRDHS Injection, 200 gpm Recirculation Pump Seal Leakage

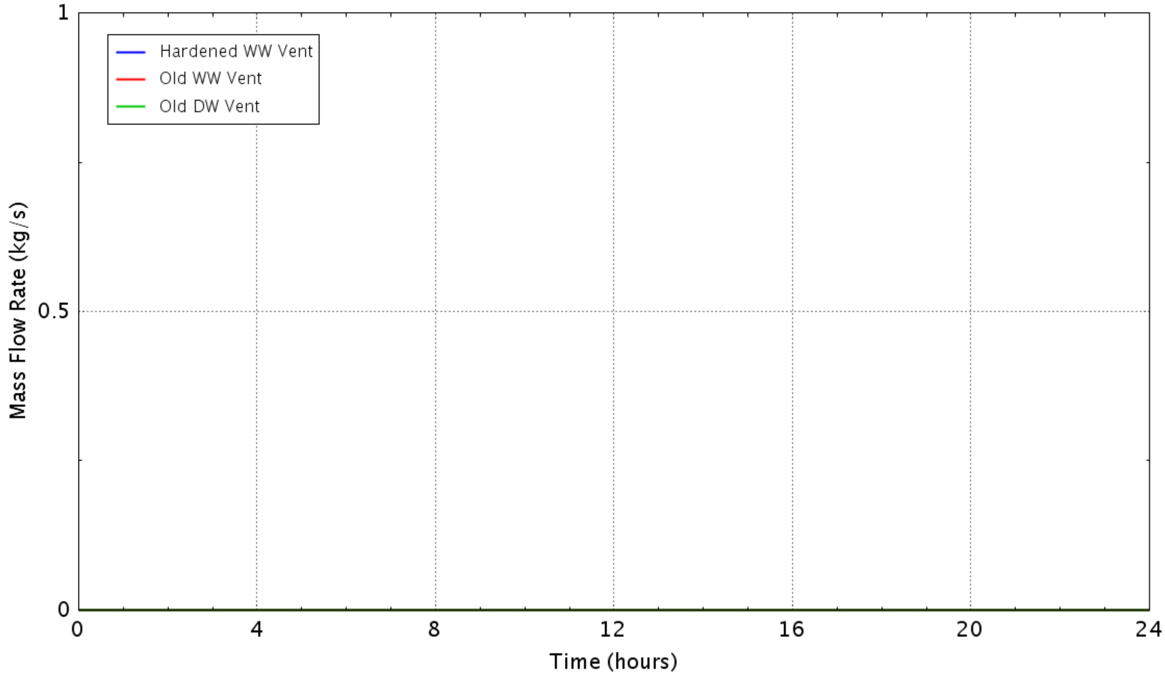

Figure G - 172 Flow rate of the containment vents

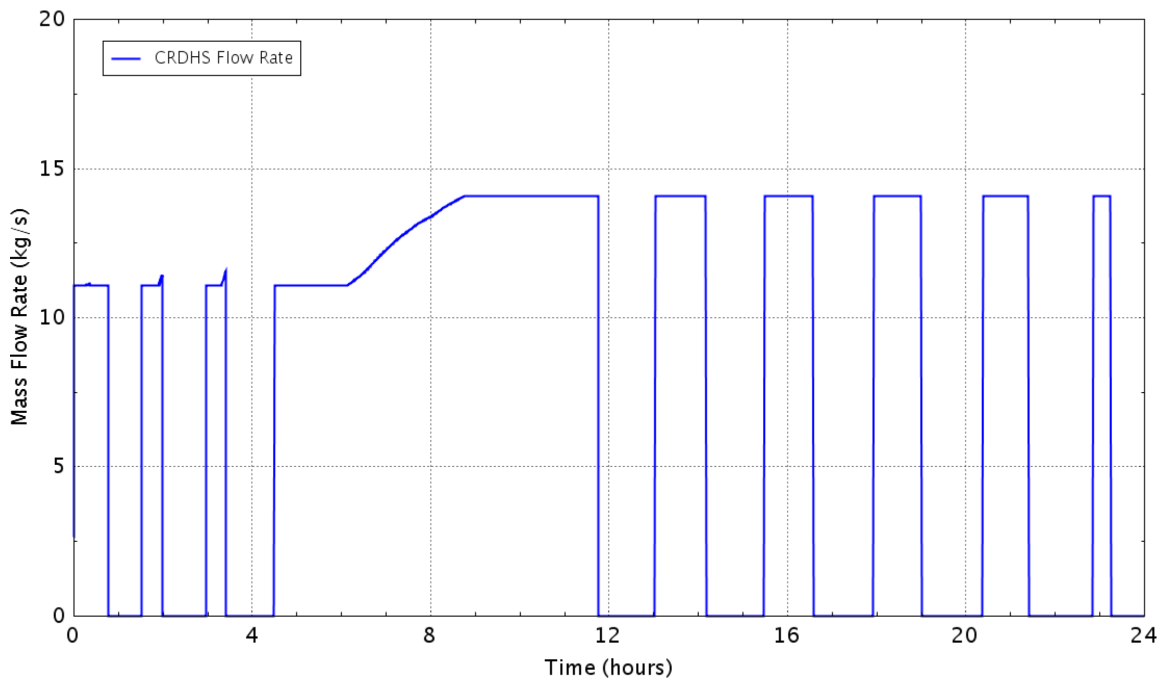

Figure G - 173 Flow rate of the control rod drive hydraulic system

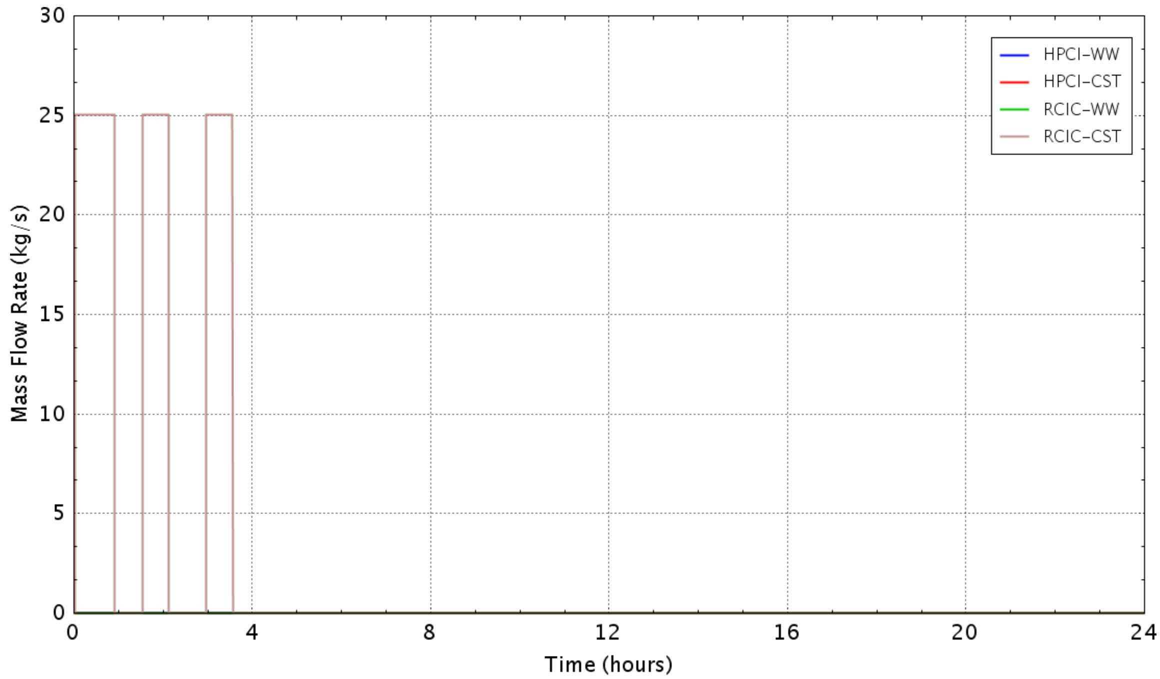

Figure G - 174 Flow rate of the HPCI/RCIC pumps

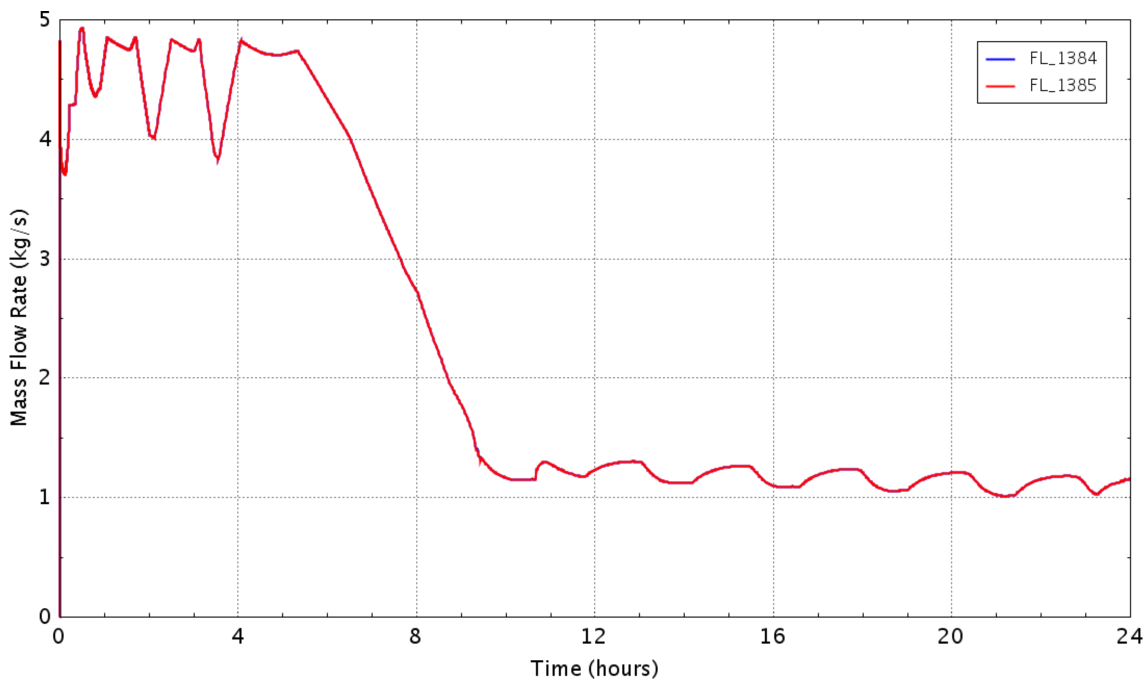

Figure G - 175 Flow rate of the recirculating pump seal leakage

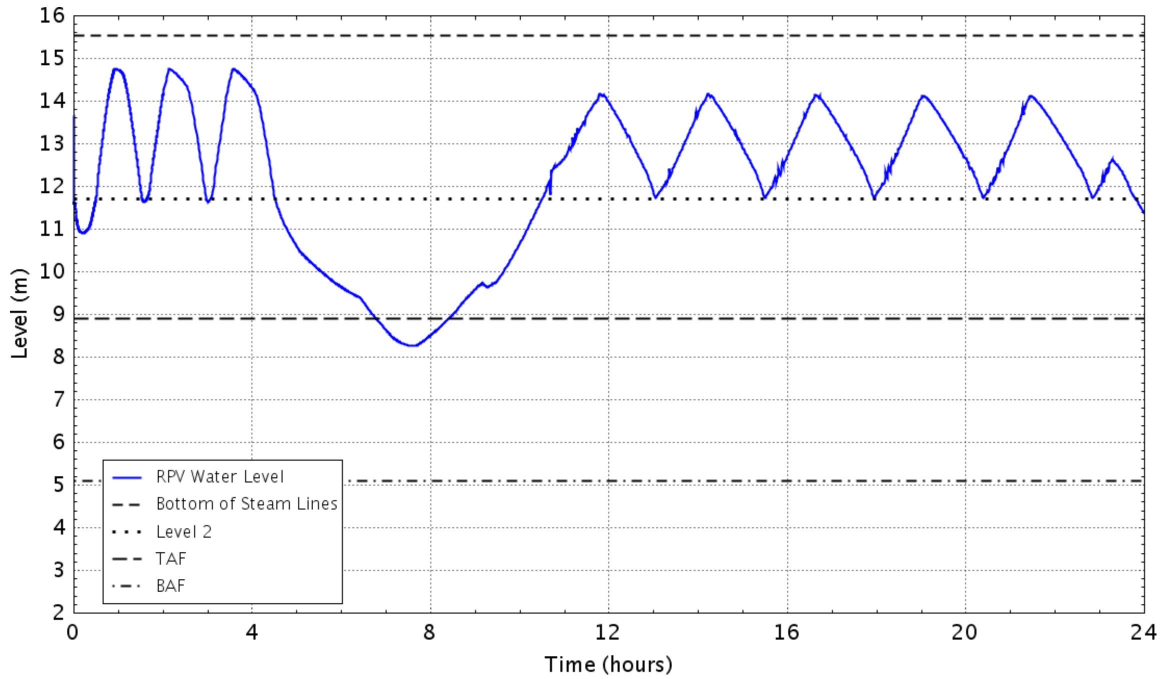

Figure G - 176 RPV Downcomer water level

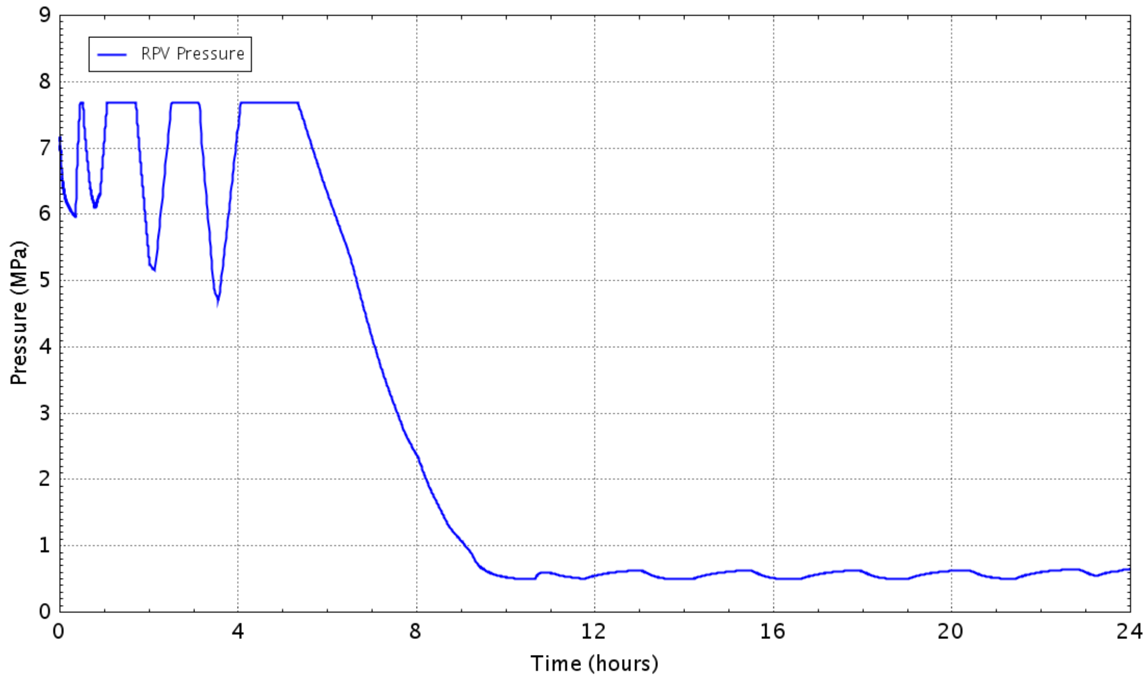

Figure G - 177 Pressure in the RPV

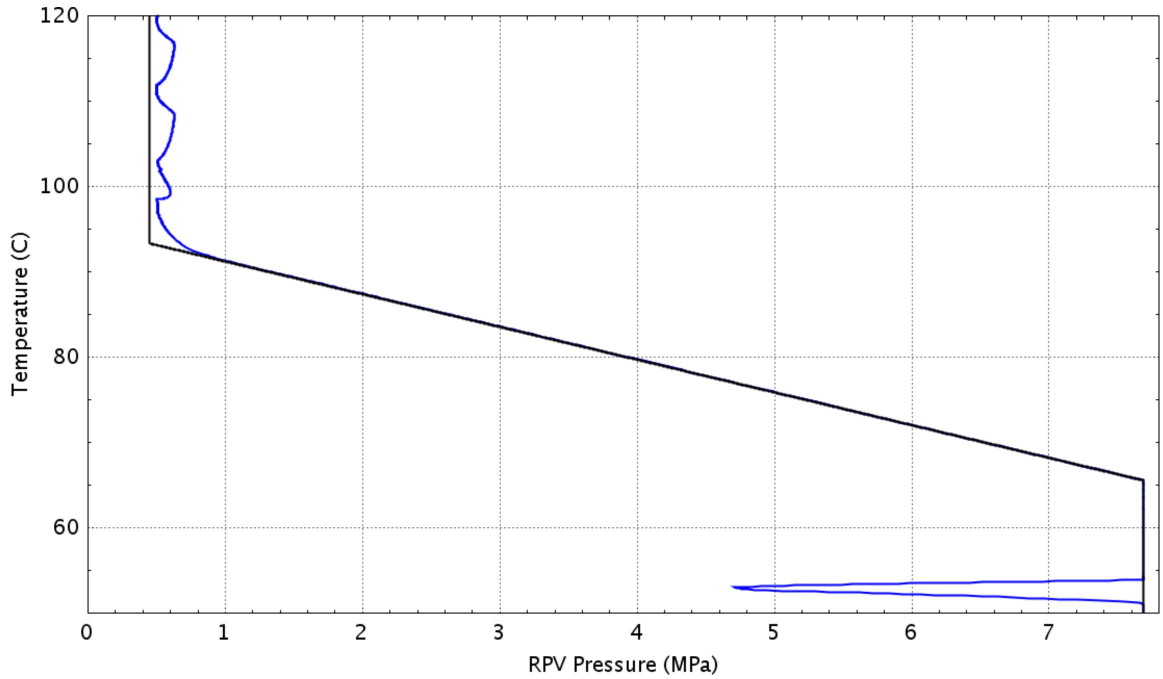

Figure G - 178 Plant status relative to the HCL curve (Graph 4 of the EOPs)

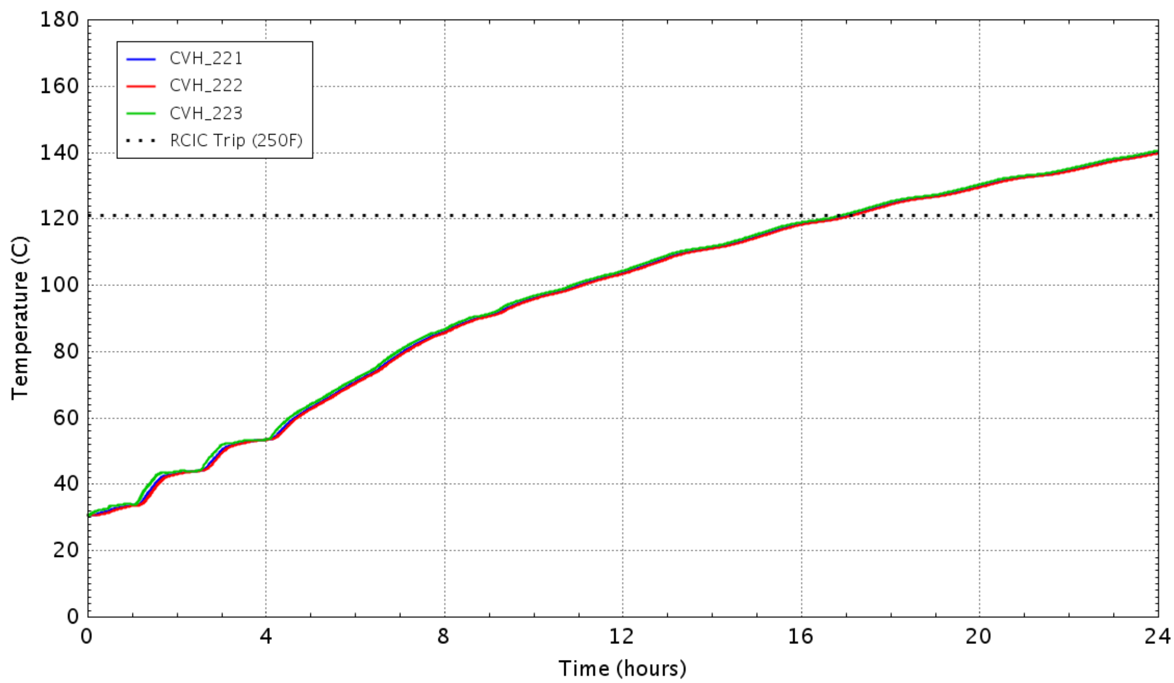

Figure G - 179 Water temperature in the wetwell

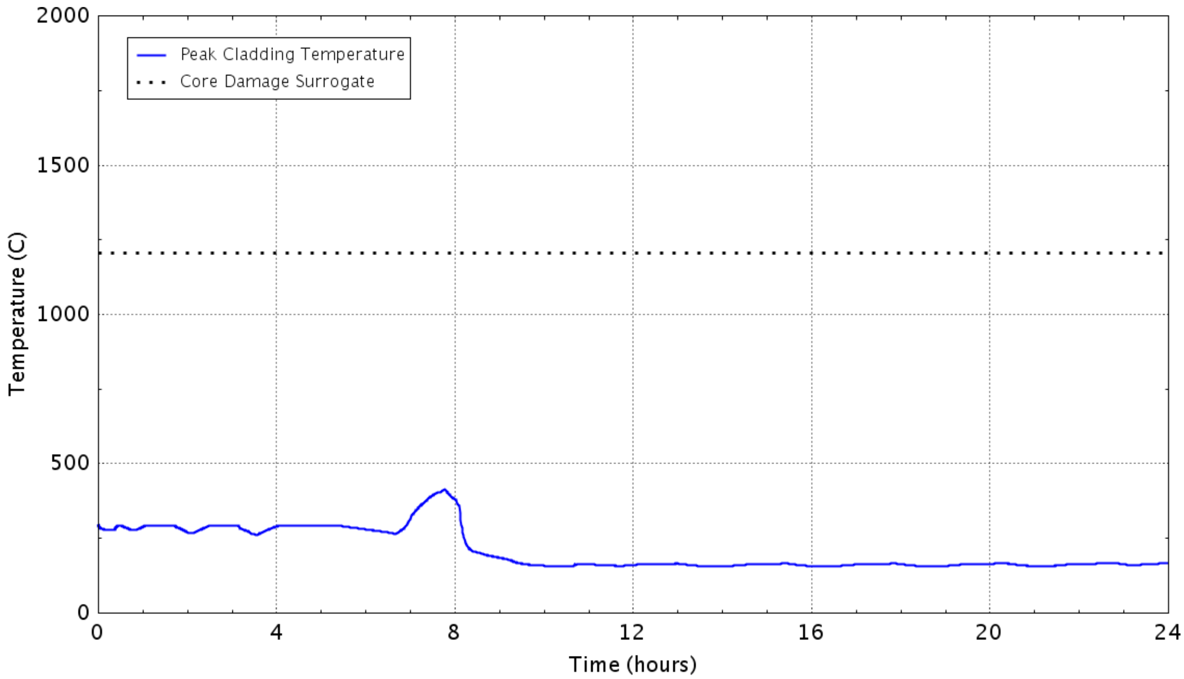

Figure G - 180 Peak temperature of the fuel cladding as a function of time

G.2.5 Case 17: LOMFW-25, RCIC Lost at 6 hrs., Nominal CRDHS Injection, Nominal Recirculation Pump Seal Leakage

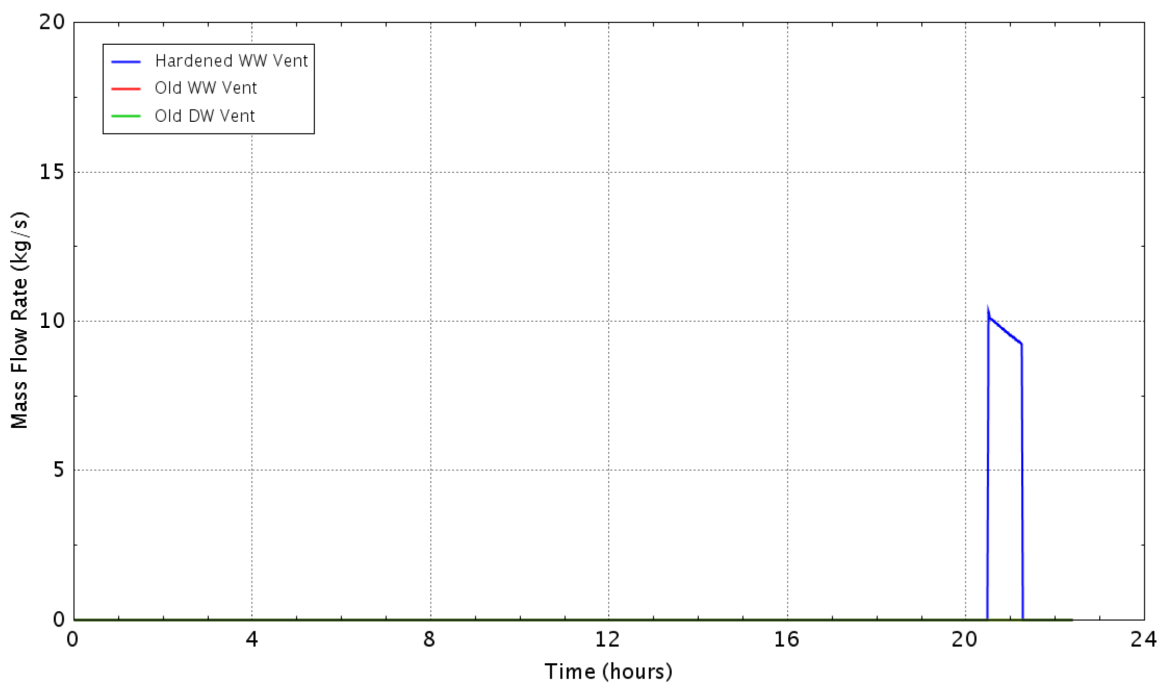

Figure G - 181 Flow rate of the containment vents

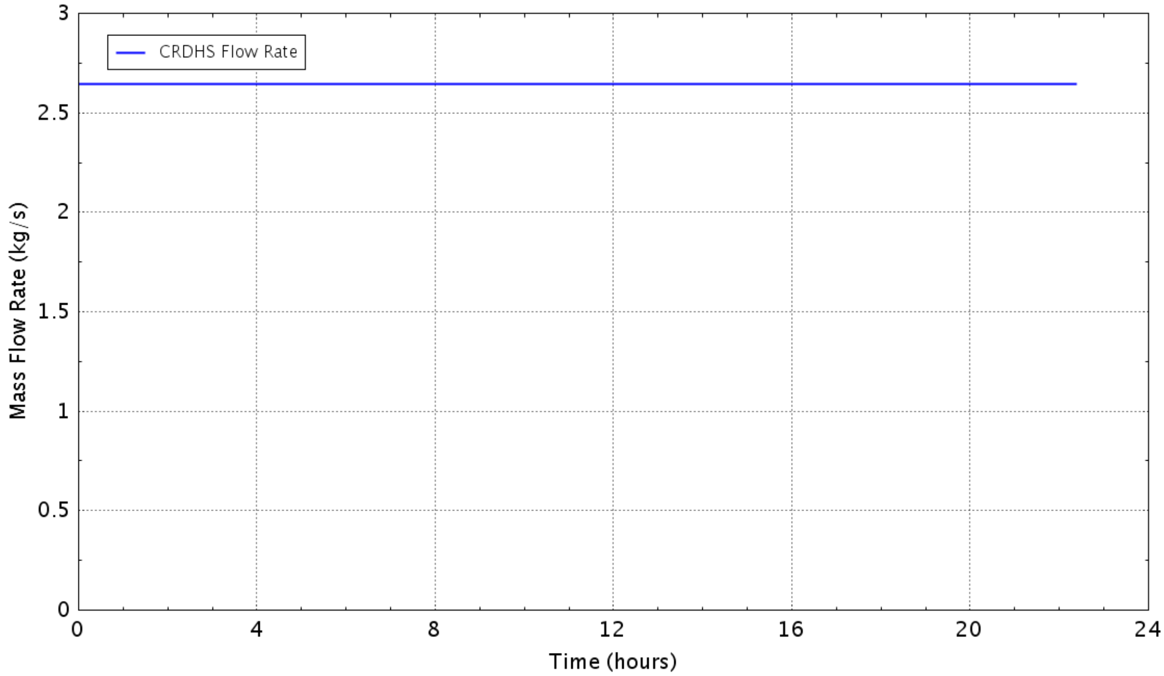

Figure G - 182 Flow rate of the control rod drive hydraulic system

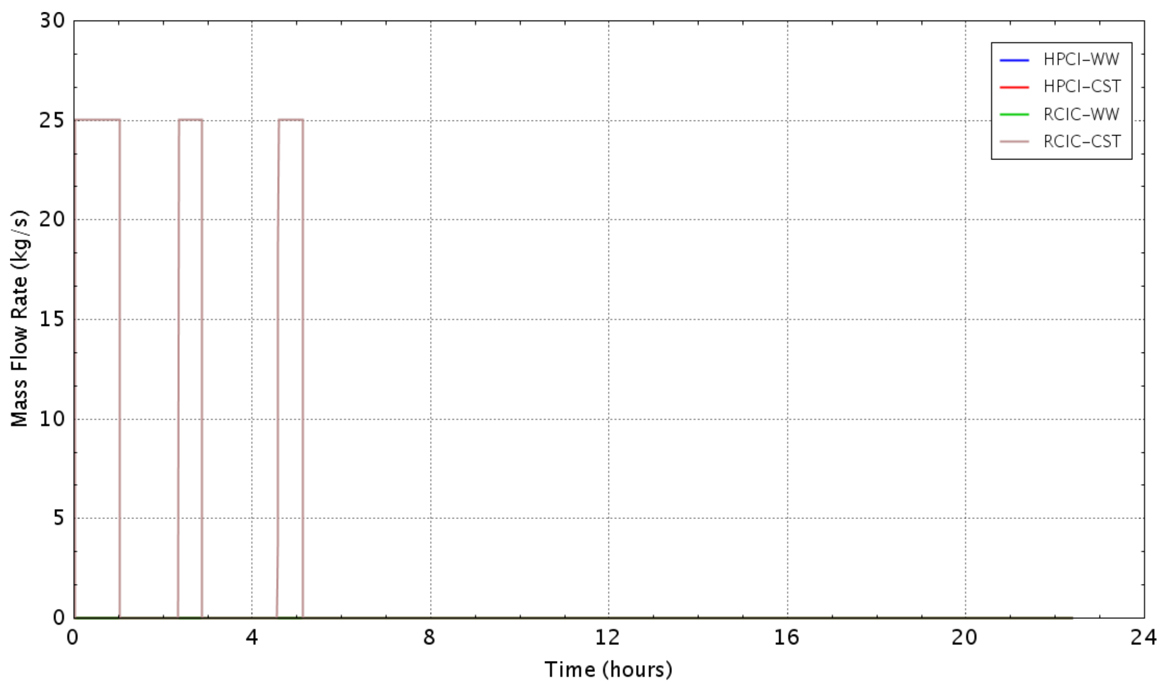

Figure G - 183 Flow rate of the HPCI/RCIC pumps

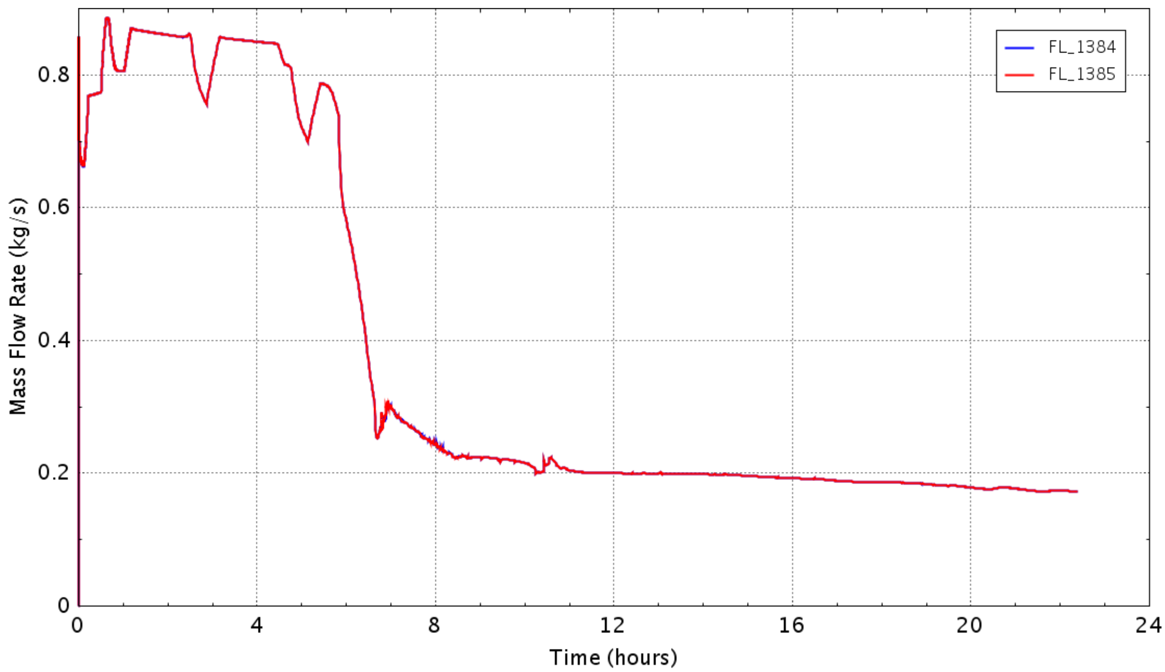

Figure G - 184 Flow rate of the recirculating pump seal leakage

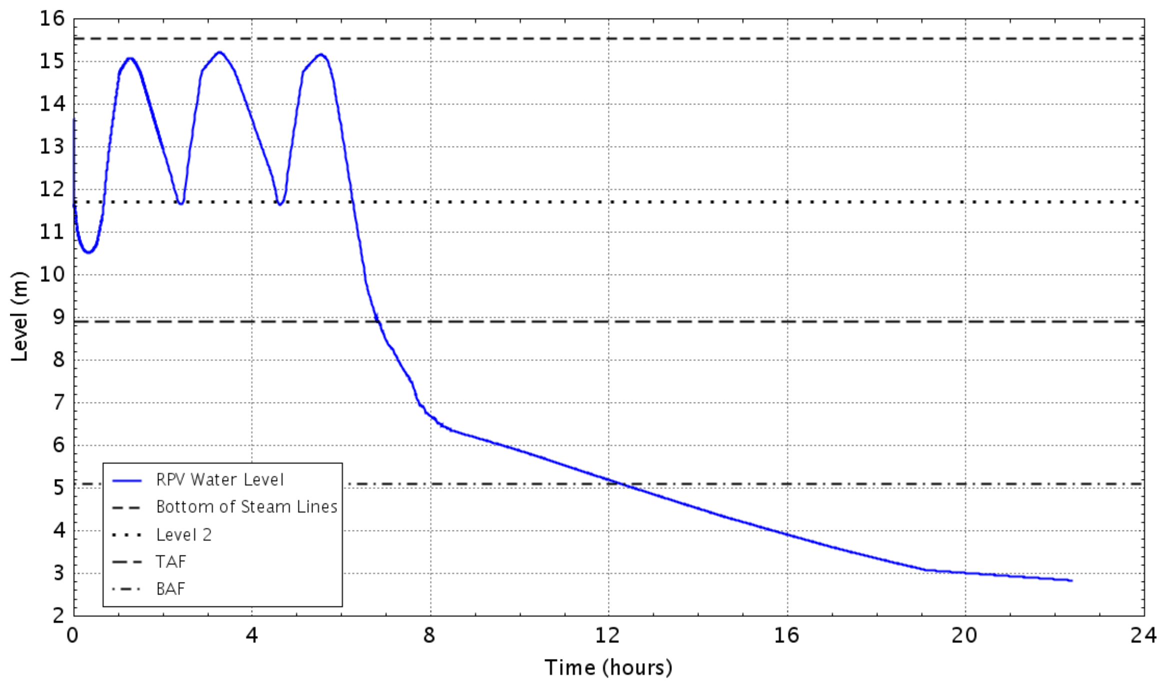

Figure G - 185 RPV Downcomer water level

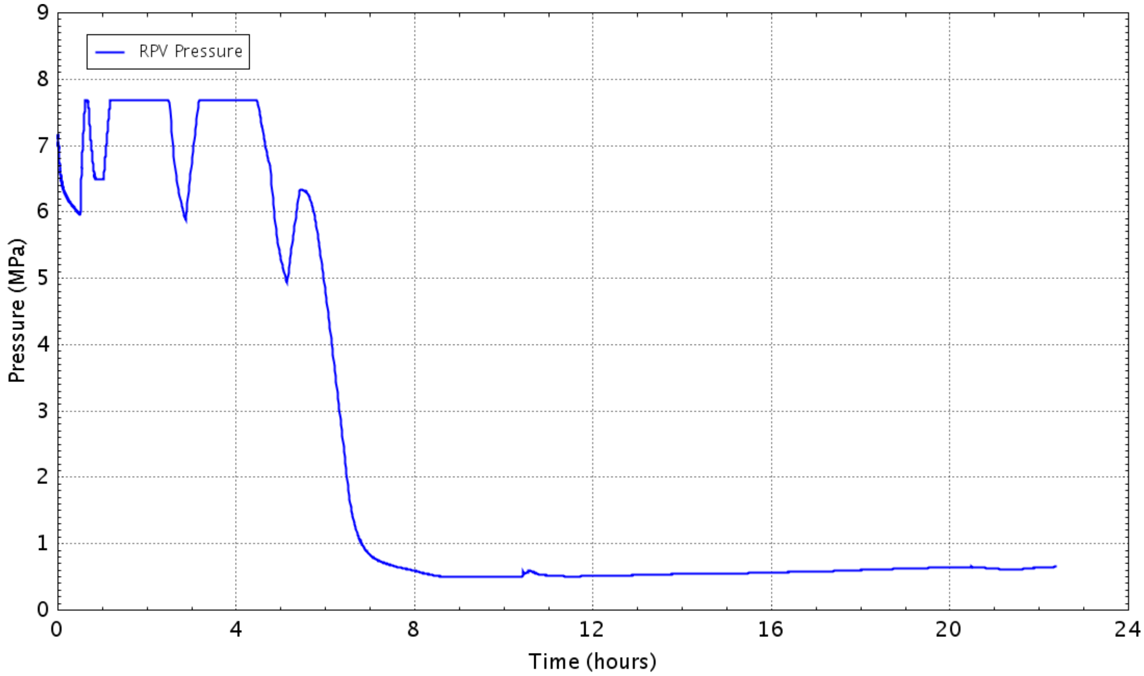

Figure G - 186 Pressure in the RPV

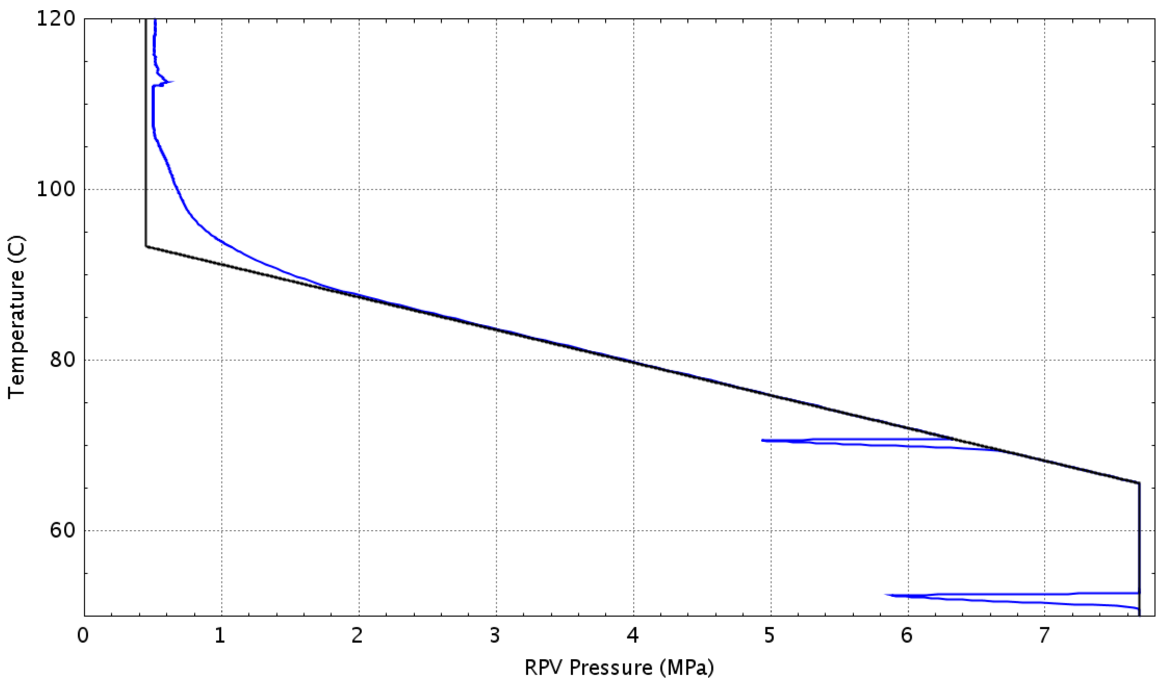

Figure G - 187 Plant status relative to the HCL curve (Graph 4 of the EOPs)

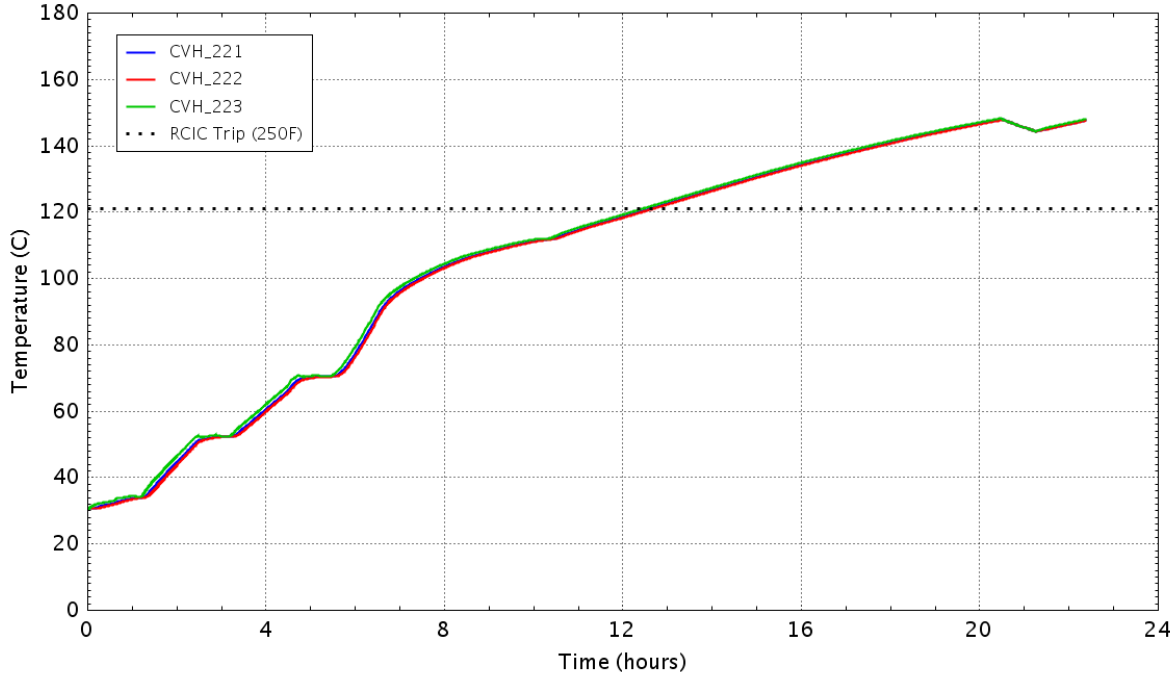

Figure G - 188 Water temperature in the wetwell

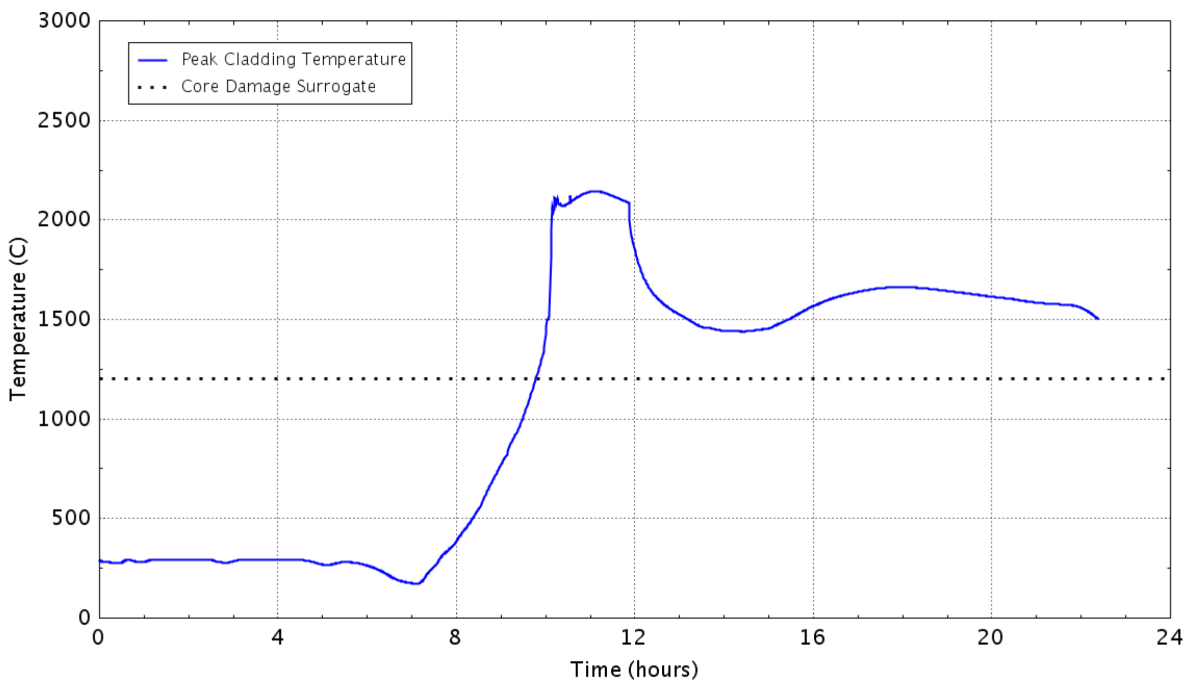

Figure G - 189 Peak temperature of the fuel cladding as a function of time

G.2.6 Case 18: LOMFW-25, RCIC Lost at 6 hrs., Nominal CRDHS Injection, 200 gpm Recirculation Pump Seal Leakage

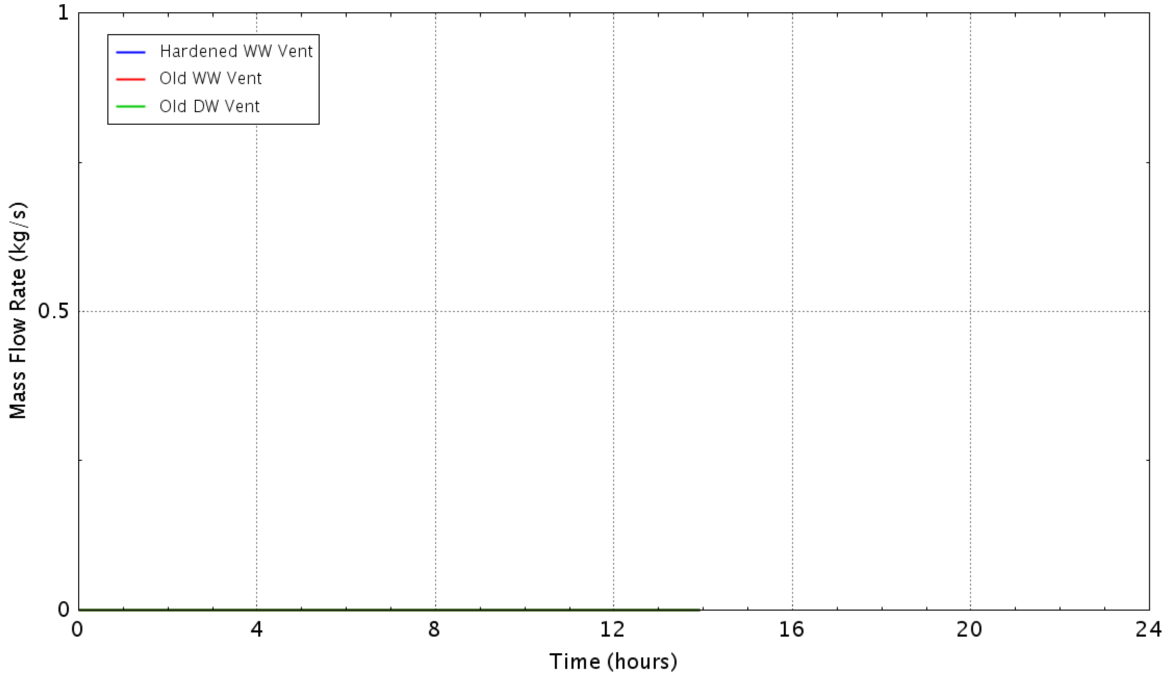

Figure G - 190 Flow rate of the containment vents

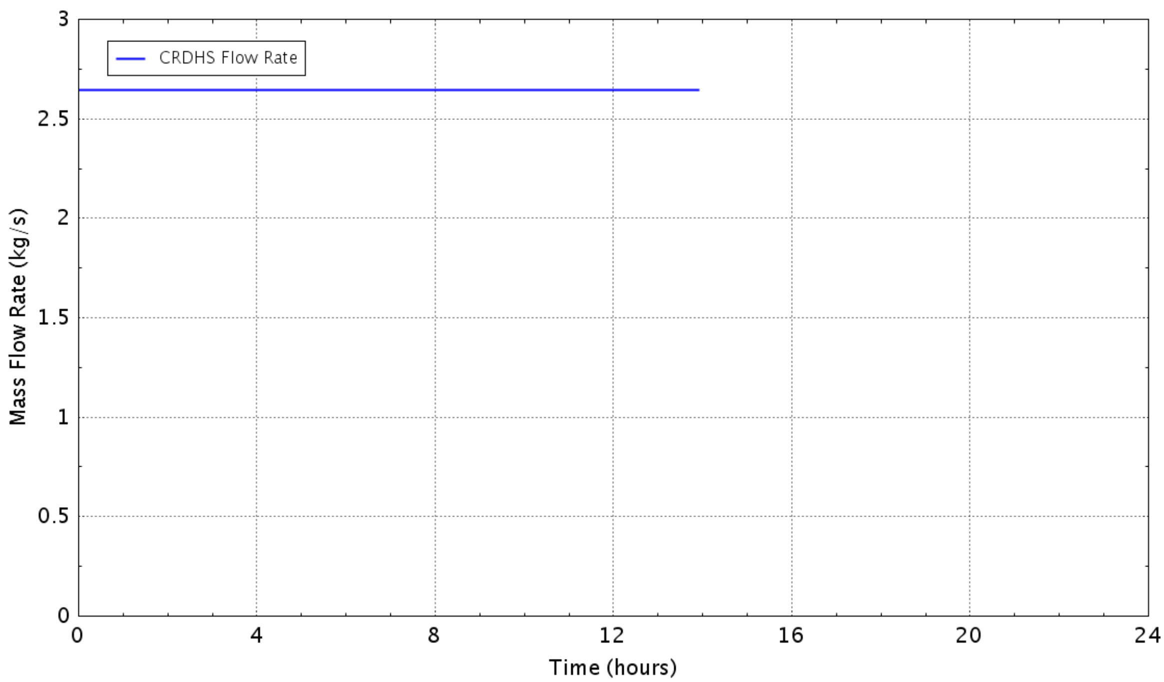

Figure G - 191 Flow rate of the control rod drive hydraulic system

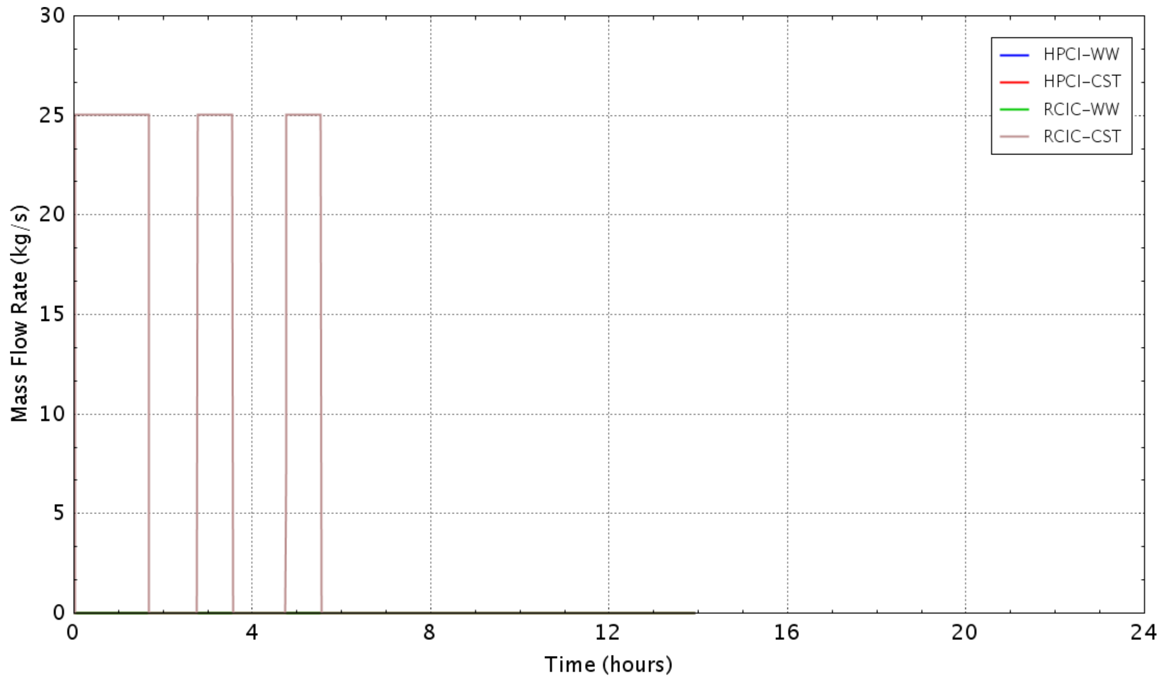

Figure G - 192 Flow rate of the HPCI/RCIC pumps

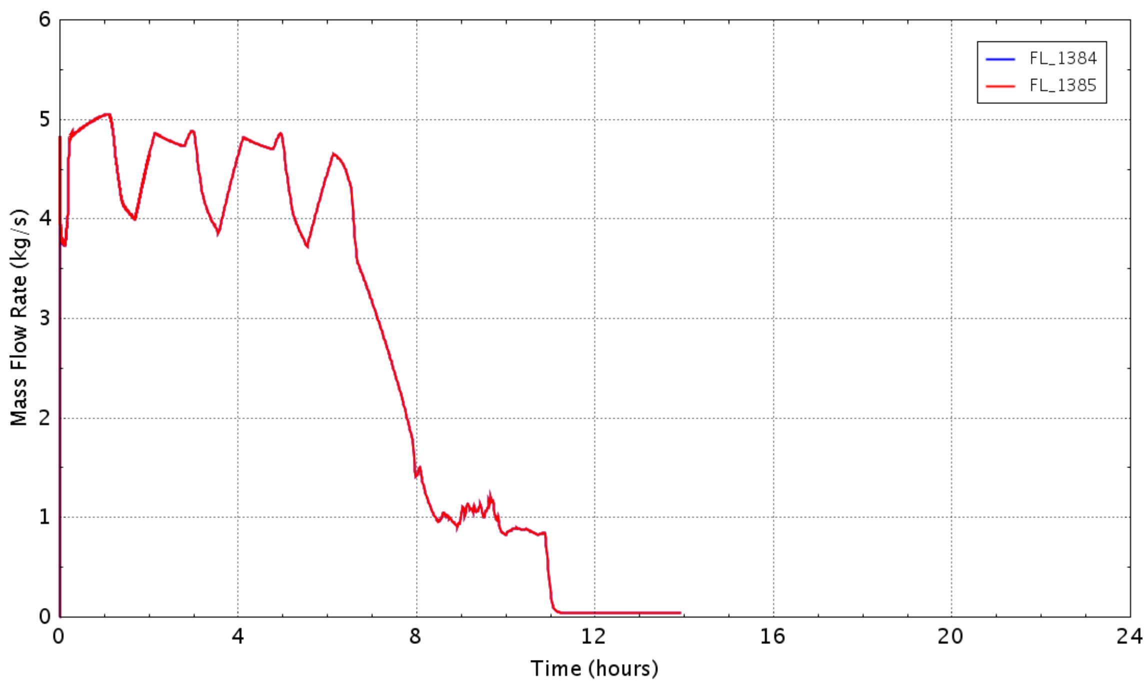

Figure G - 193 Flow rate of the recirculating pump seal leakage

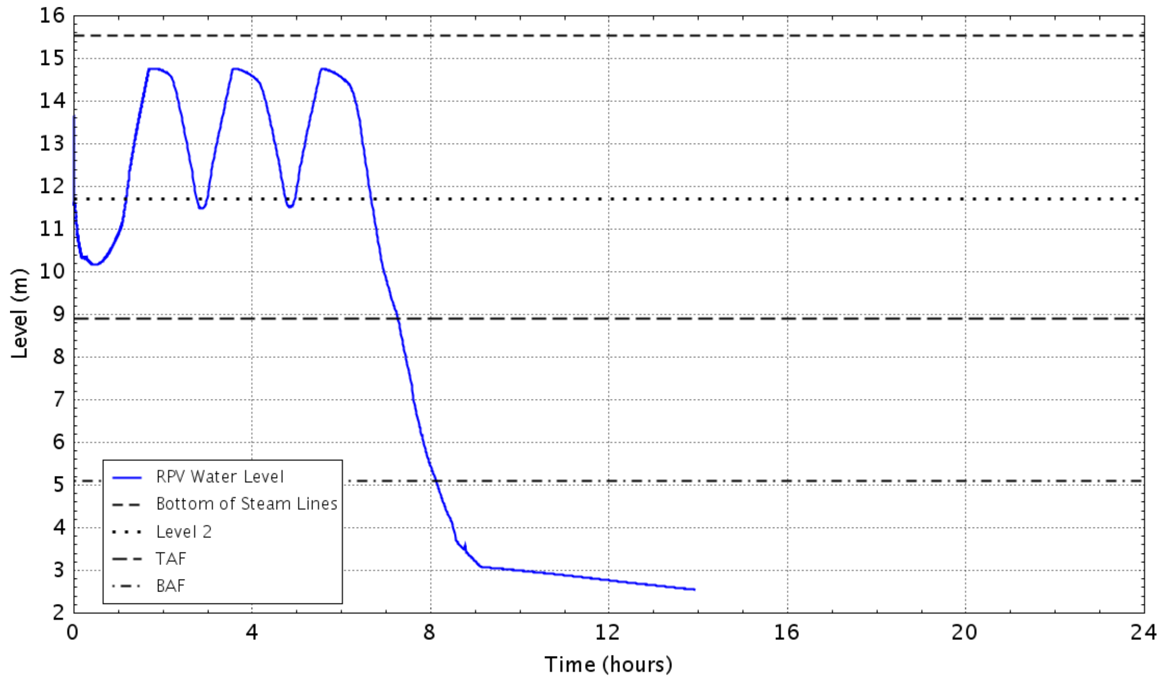

Figure G - 194 RPV Downcomer water level

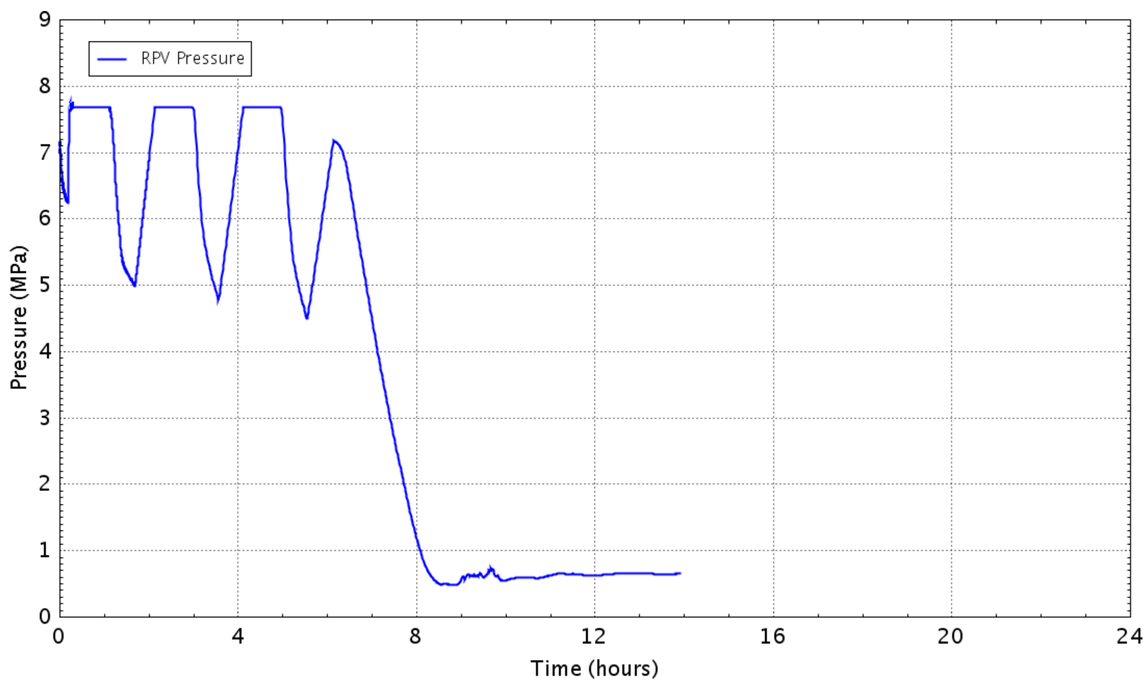

Figure G - 195 Pressure in the RPV

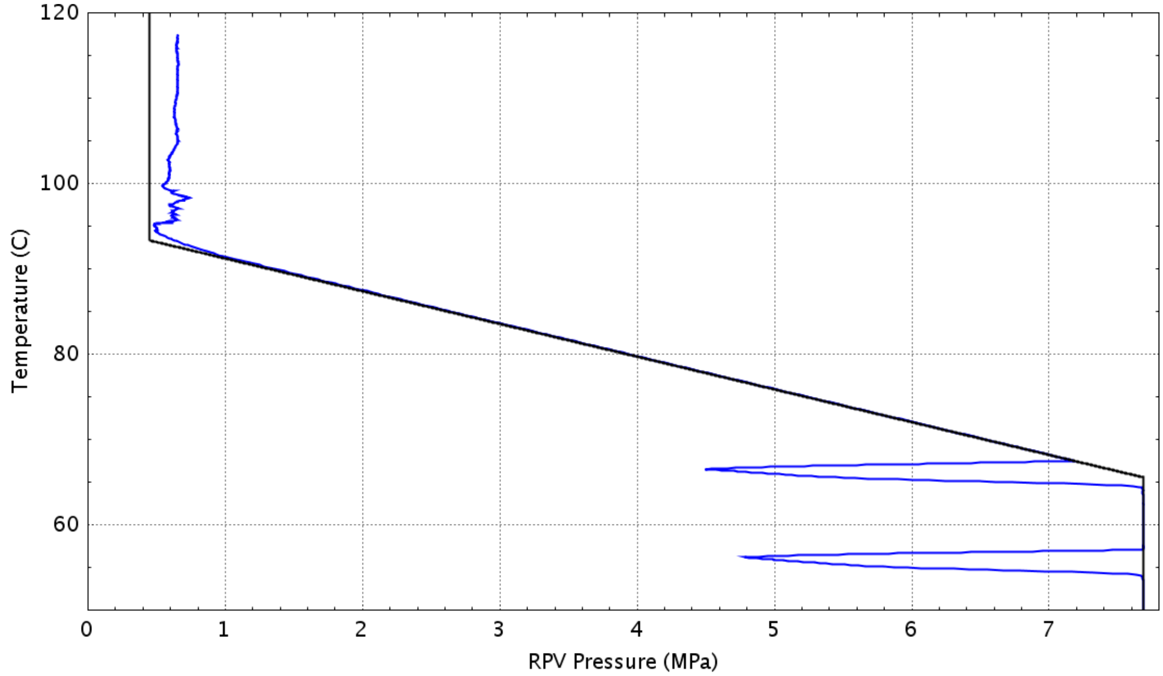

Figure G - 196 Plant status relative to the HCL curve (Graph 4 of the EOPs)

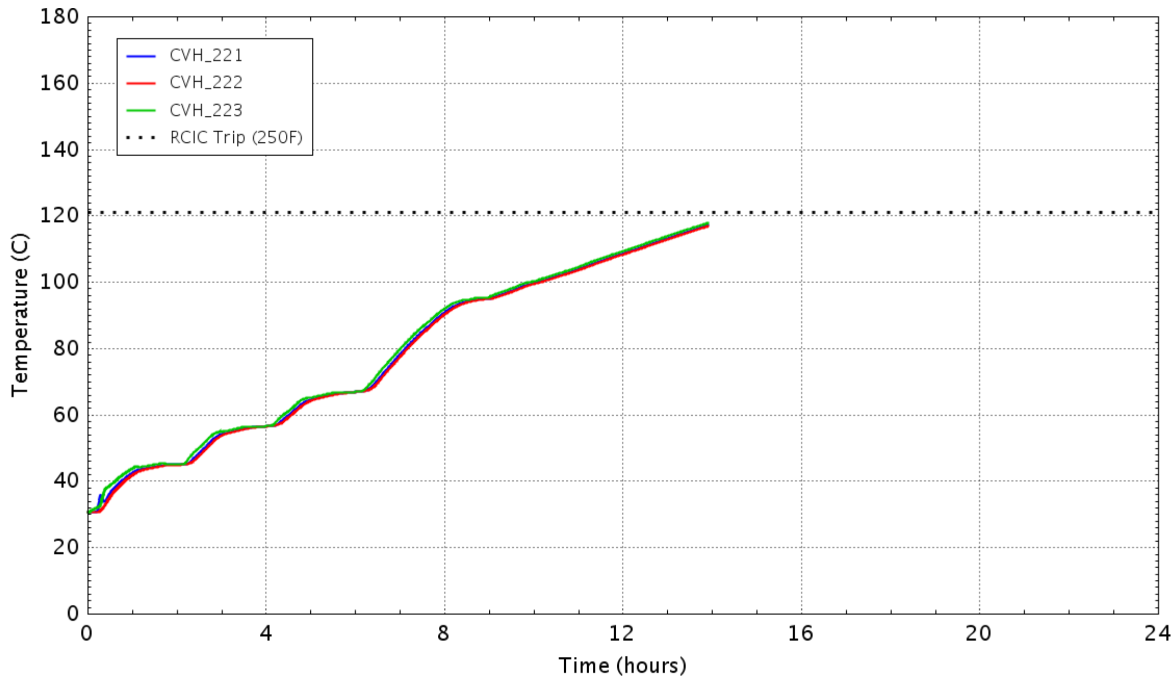

Figure G - 197 Water temperature in the wetwell

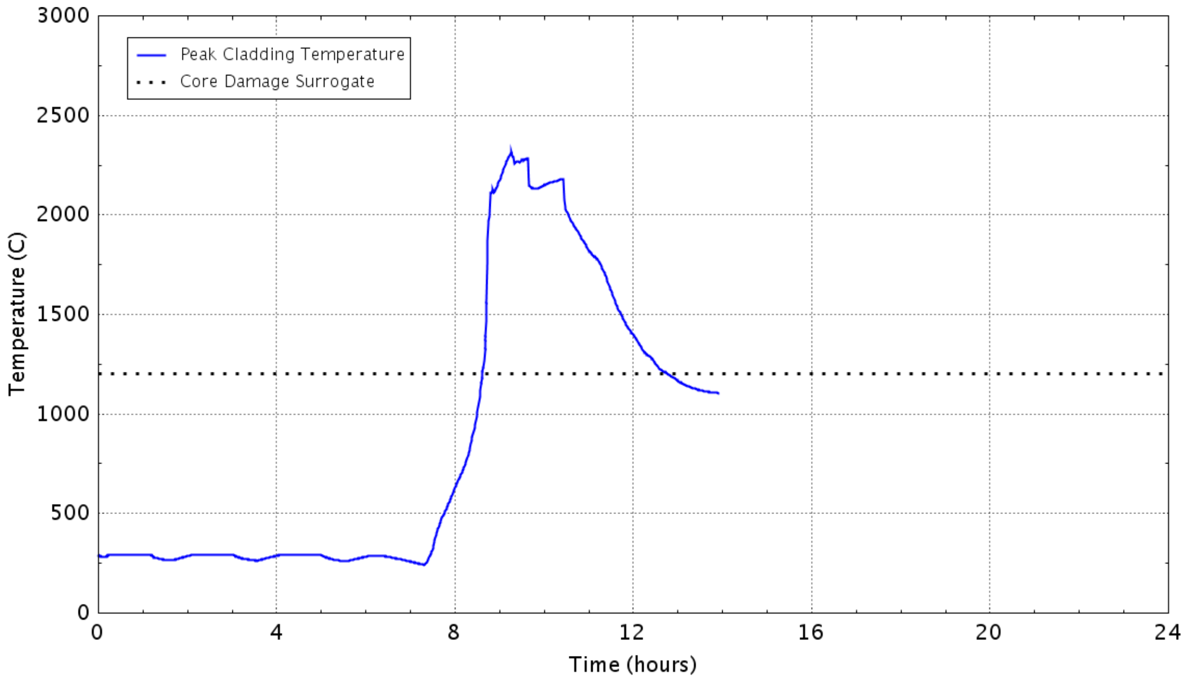

Figure G - 198 Peak temperature of the fuel cladding as a function of time

G.2.7 Case 19: LOMFW-25, RCIC Lost at 6 hrs., One Train ofCRDHS Injection, Nominal Recirculation Pump Seal Leakage

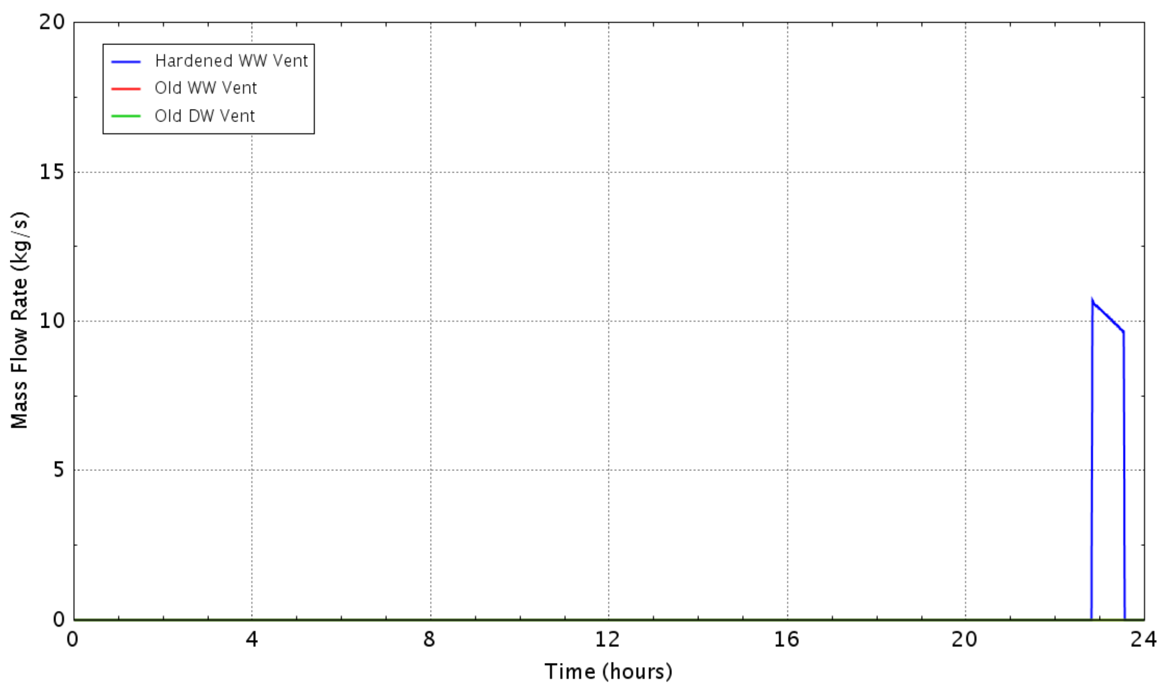

Figure G - 199 Flow rate of the containment vents

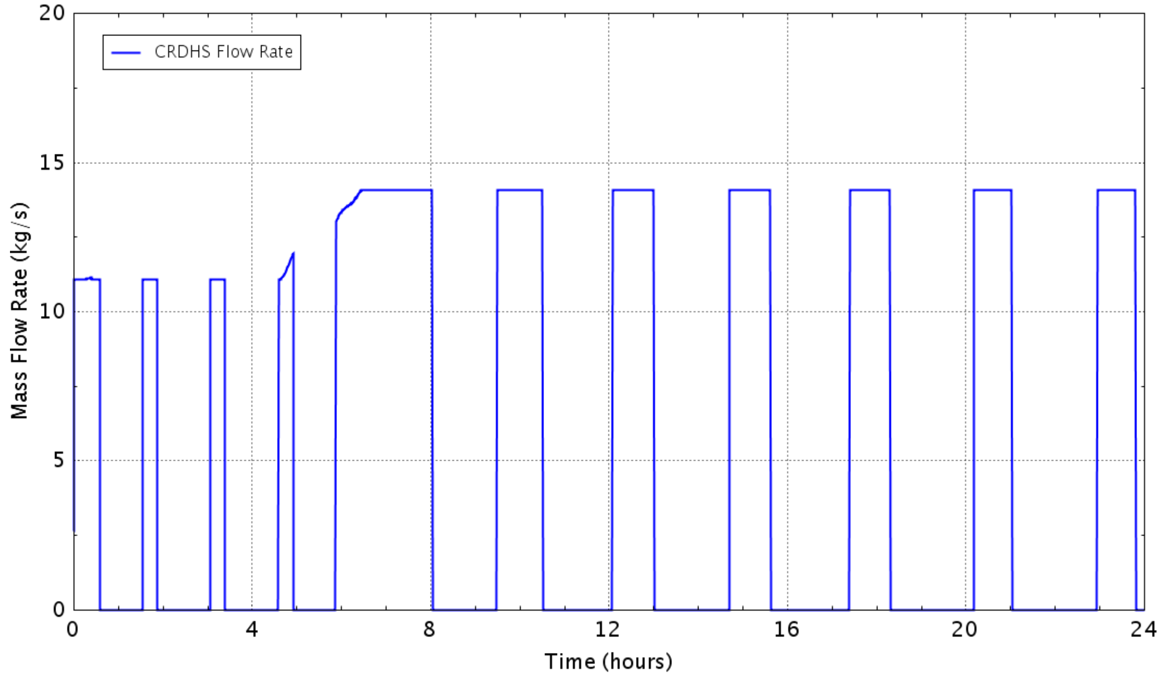

Figure G - 200 Flow rate of the control rod drive hydraulic system

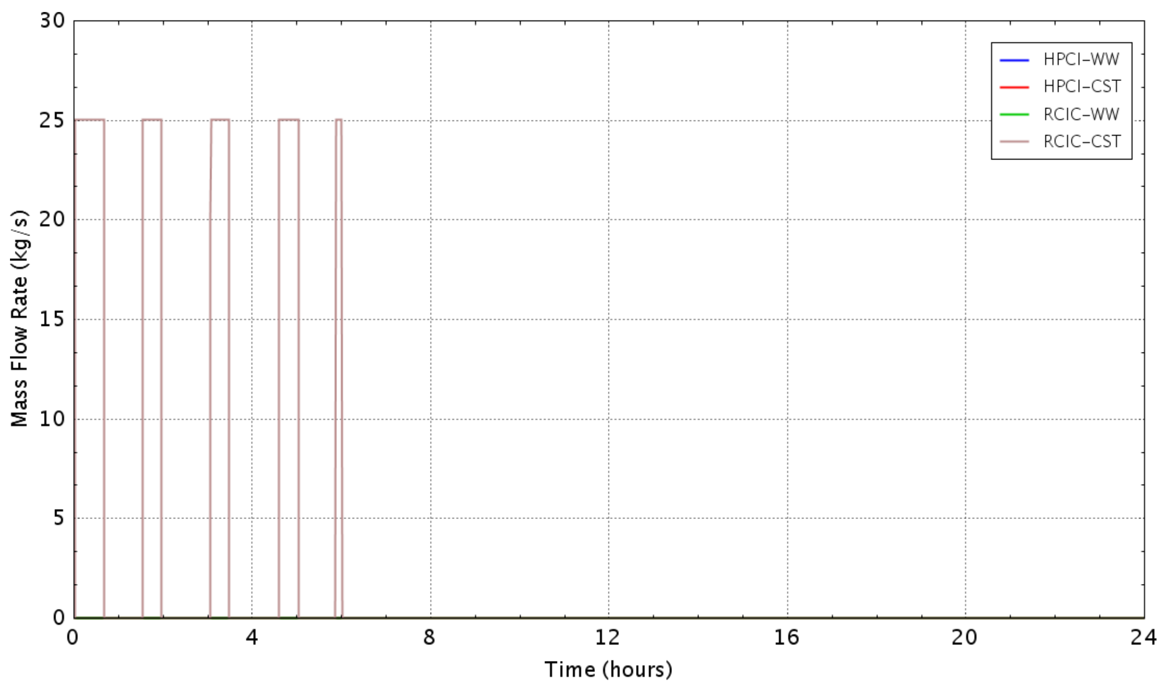

Figure G - 201 Flow rate of the HPCI/RCIC pumps

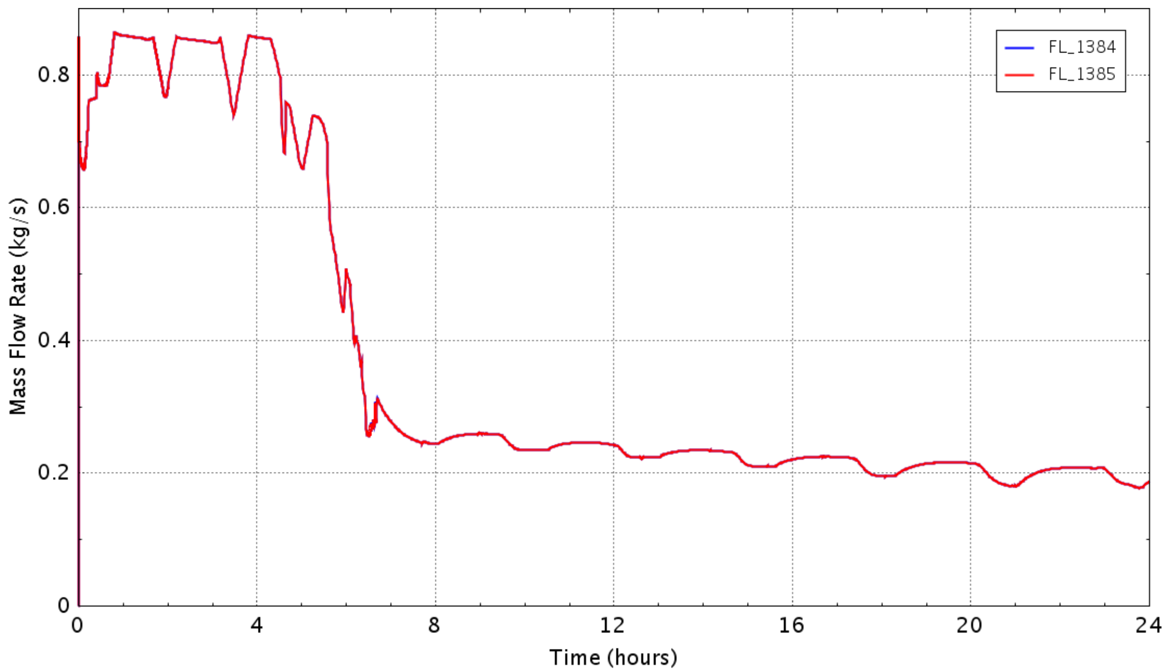

Figure G - 202 Flow rate of the recirculating pump seal leakage

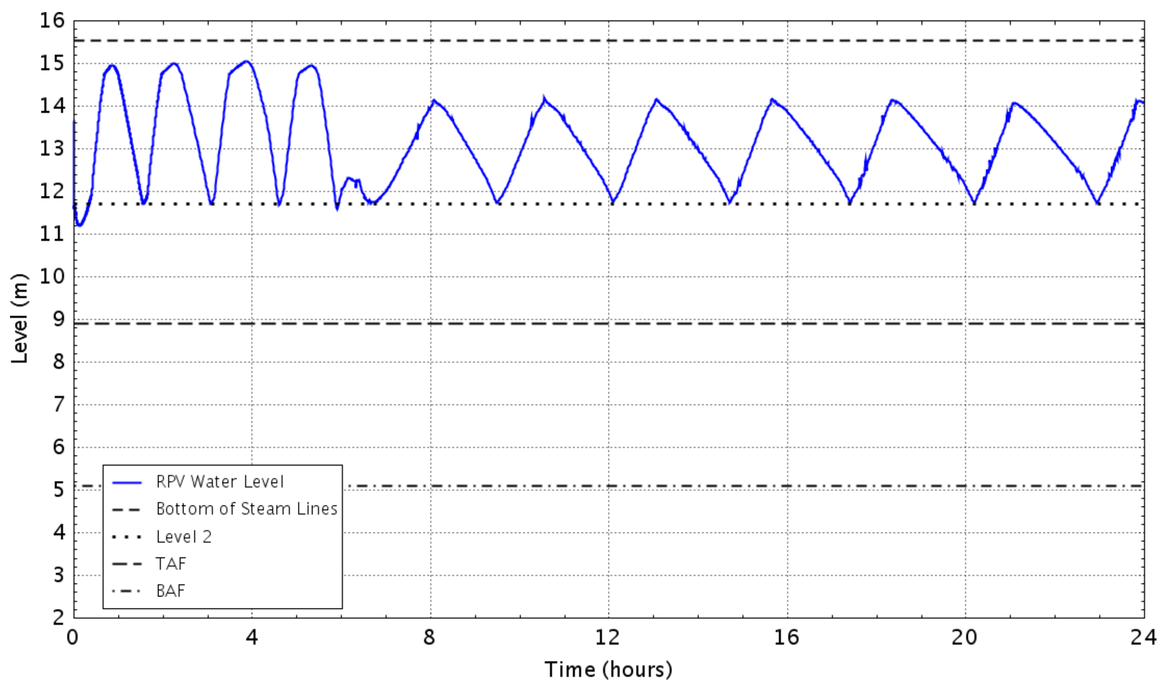

Figure G - 203 RPV Downcomer water level

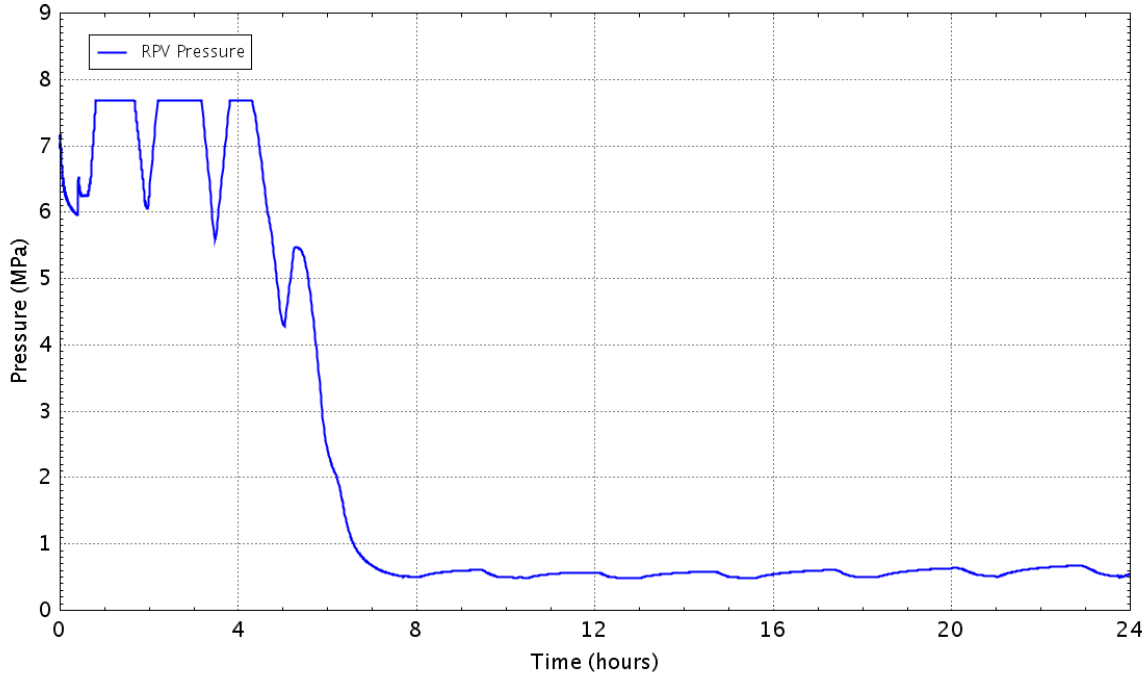

Figure G - 204 Pressure in the RPV

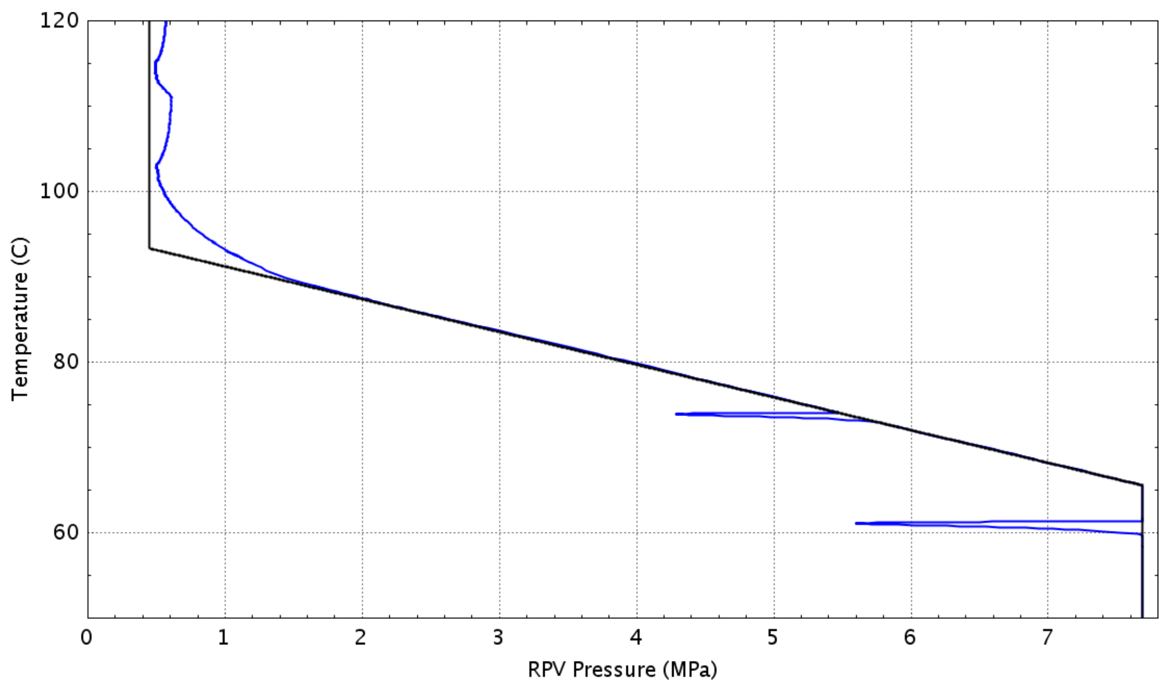

Figure G - 205 Plant status relative to the HCL curve (Graph 4 of the EOPs)

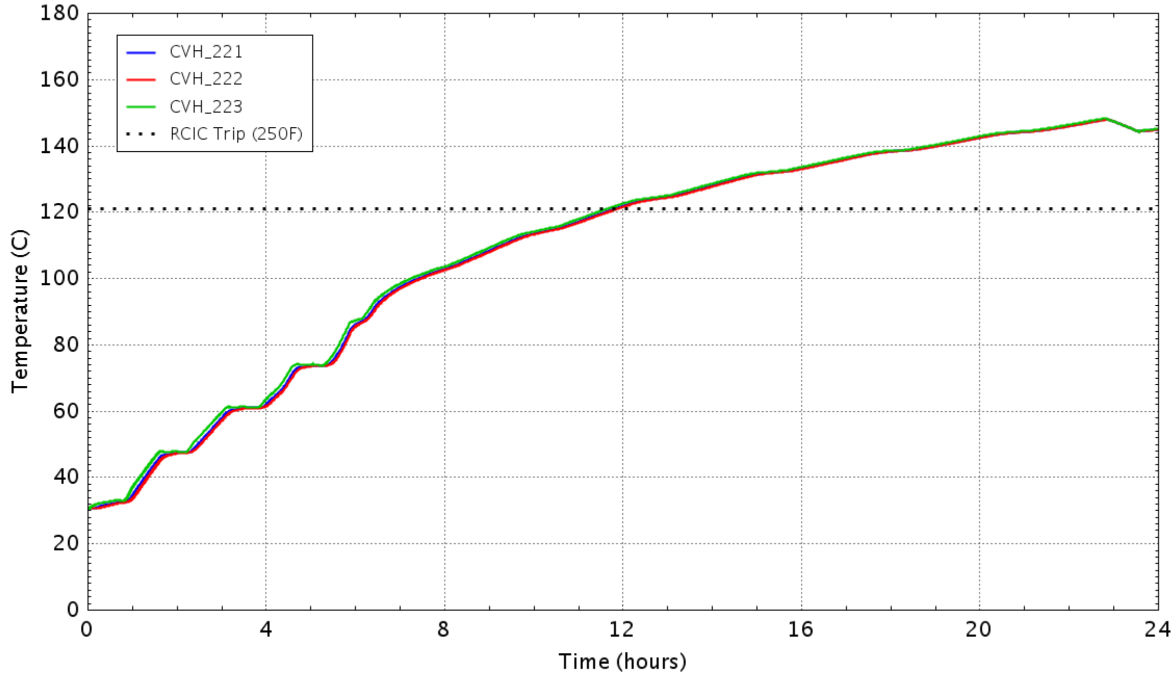

Figure G - 206 Water temperature in the wetwell

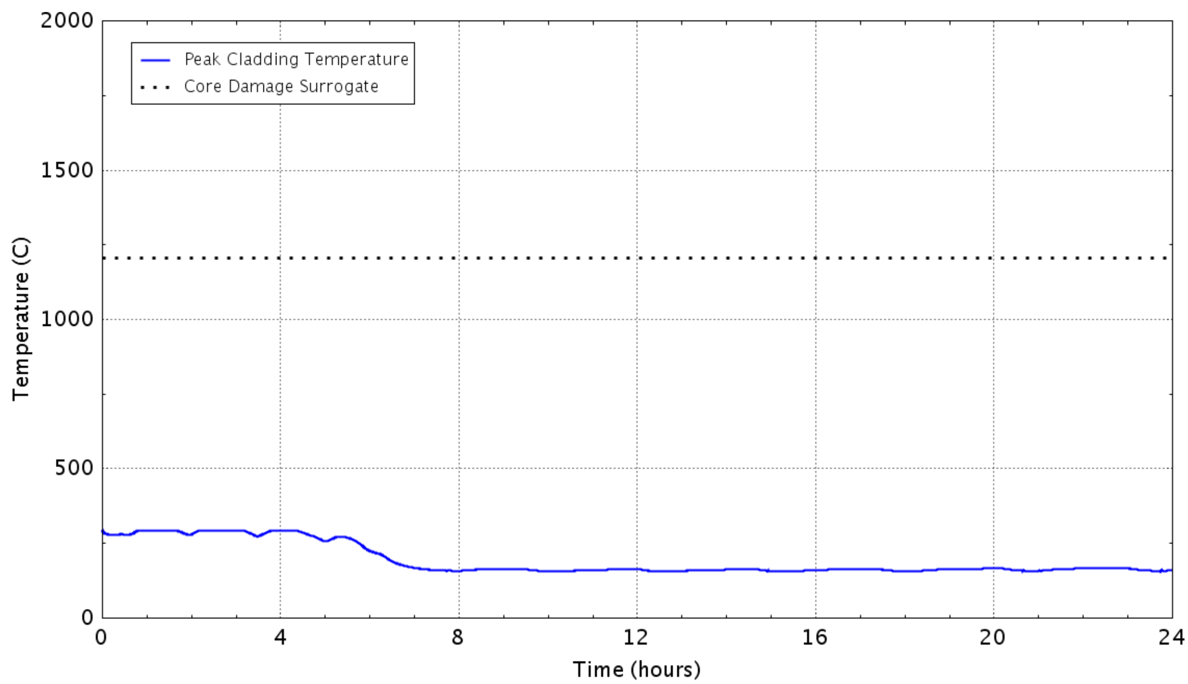

Figure G - 207 Peak temperature of the fuel cladding as a function of time

G.2.8 Case 20: LOMFW-25, RCIC Lost at 6 hrs., One Train ofCRDHS Injection, 200 gpm Recirculation Pump Seal Leakage

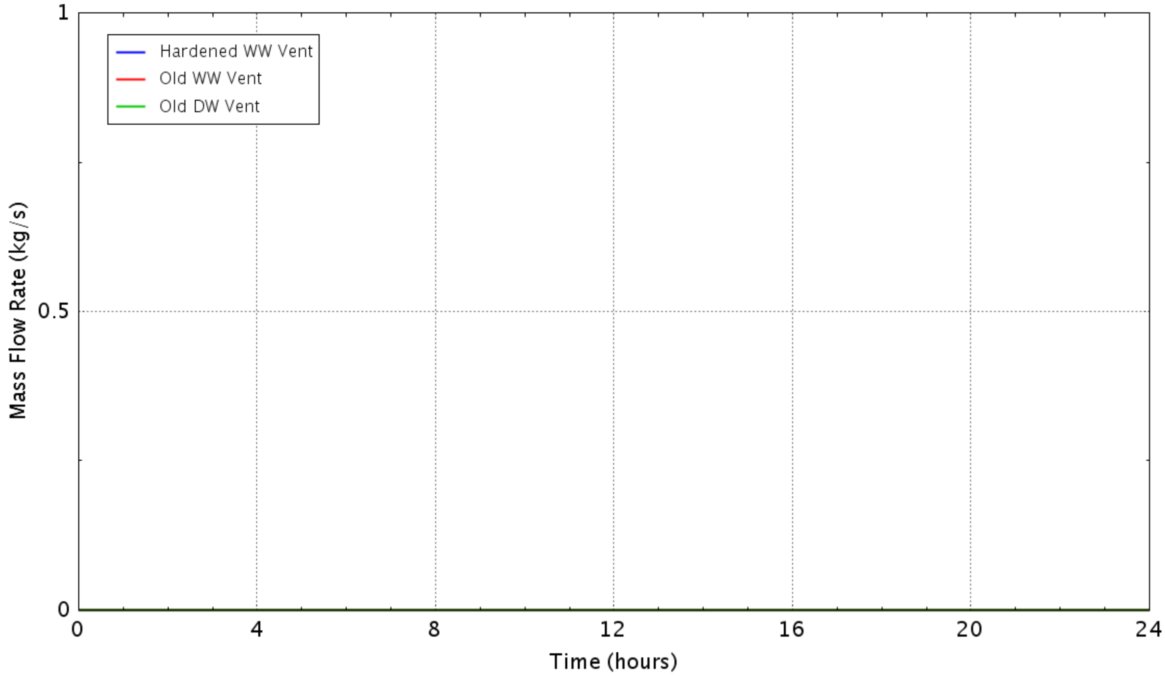

Figure G - 208 Flow rate of the containment vents

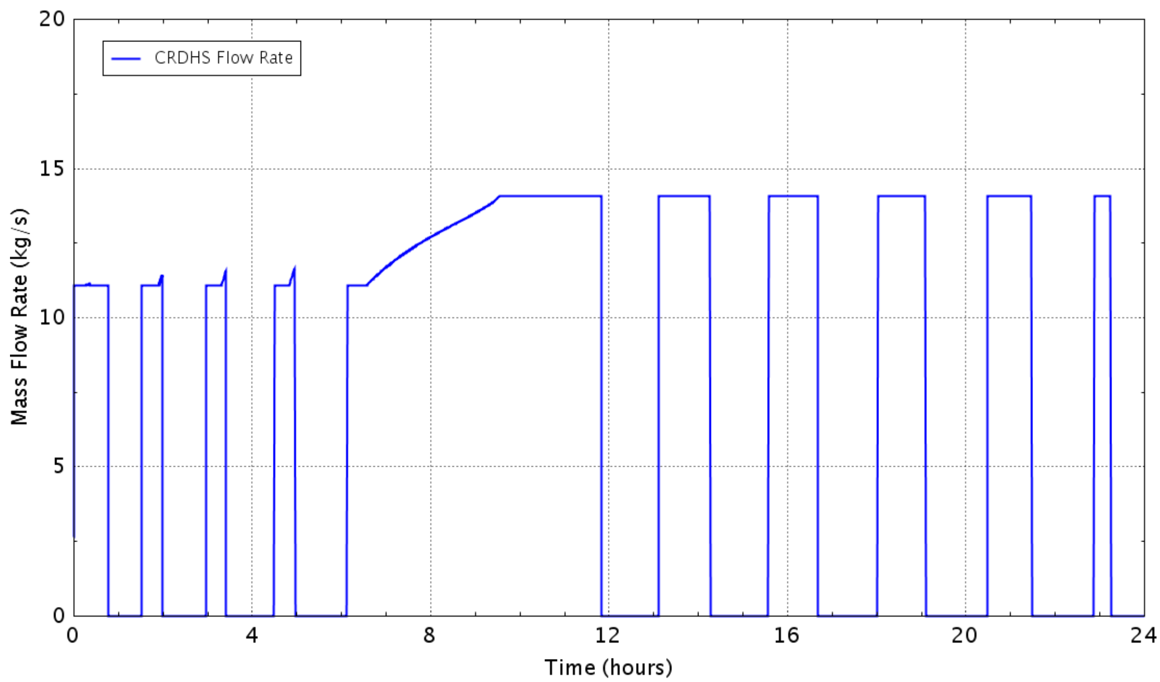

Figure G - 209 Flow rate of the control rod drive hydraulic system

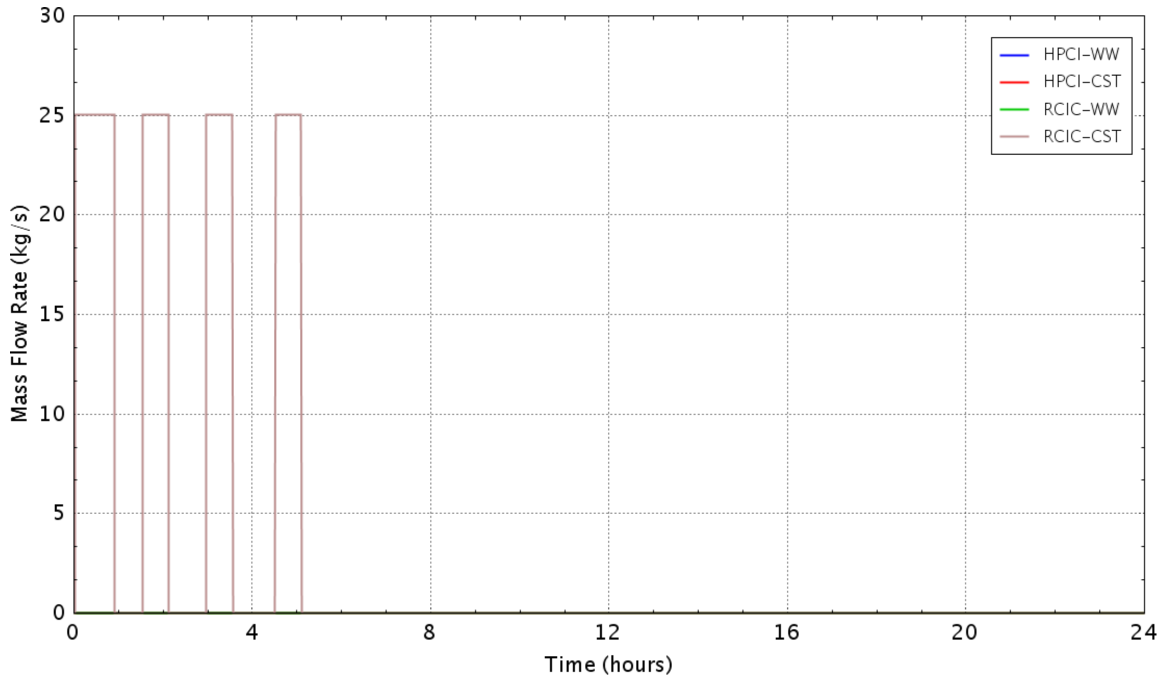

Figure G - 210 Flow rate of the HPCI/RCIC pumps

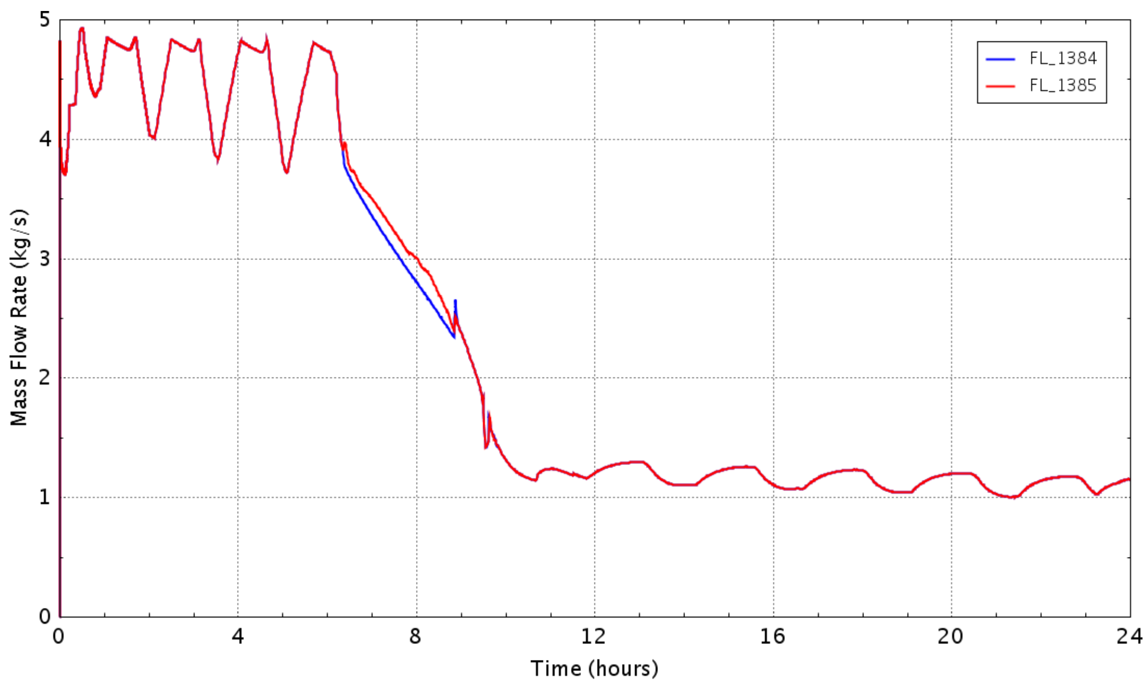

Figure G - 211 Flow rate of the recirculating pump seal leakage

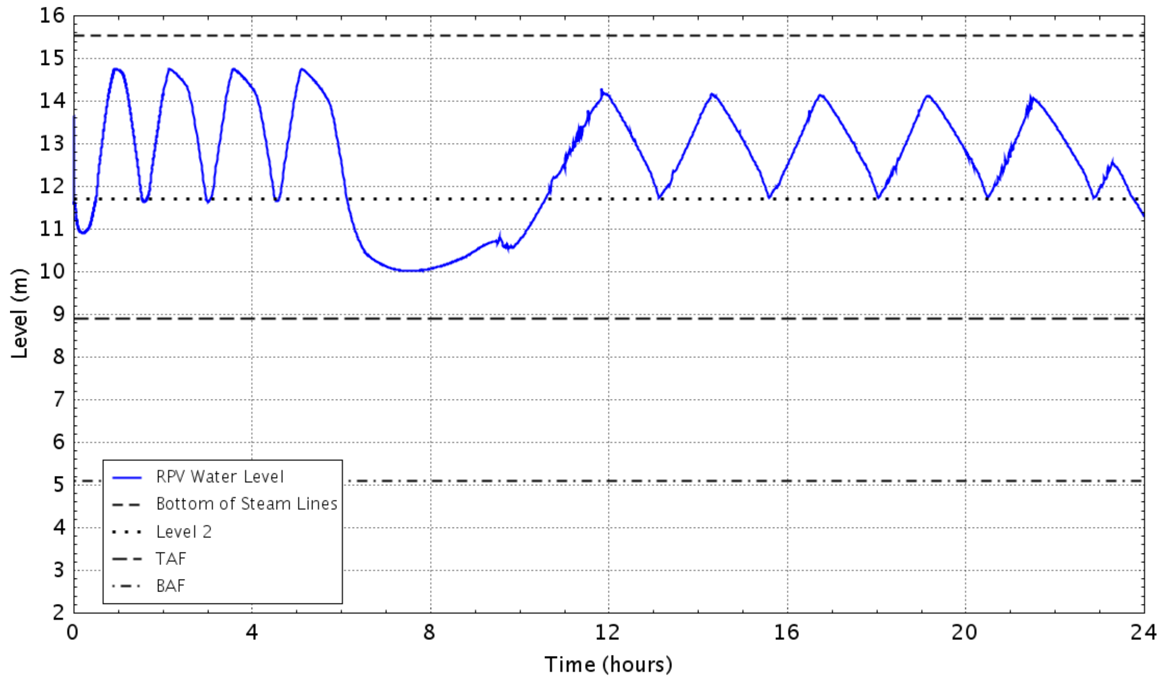

Figure G - 212 RPV Downcomer water level

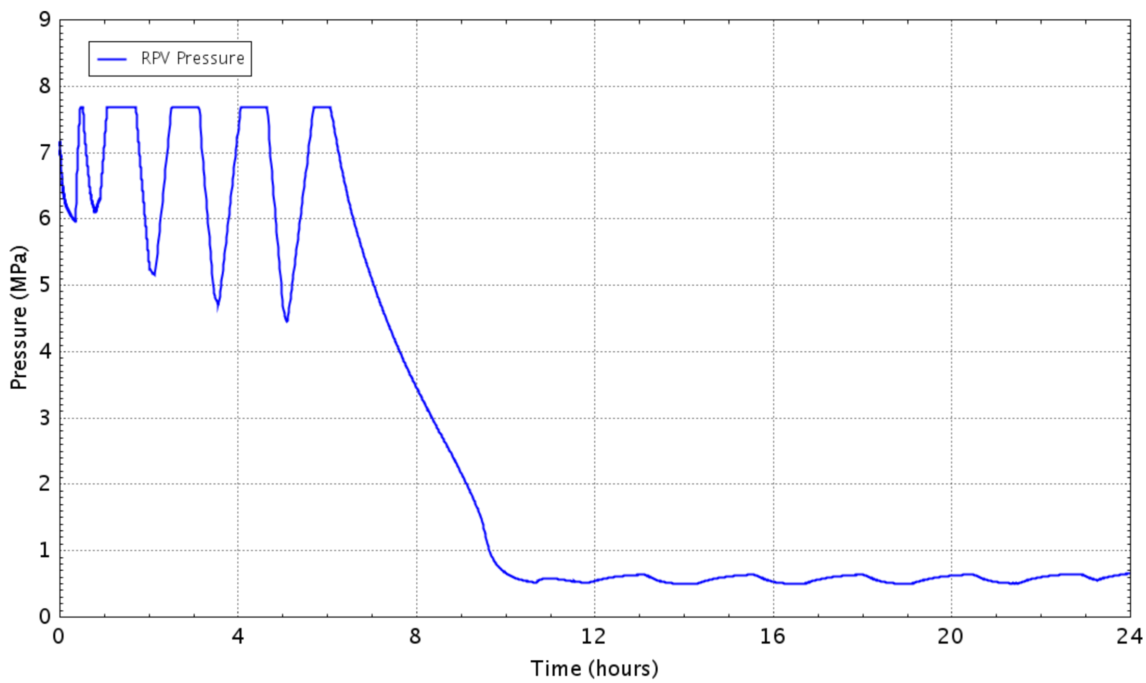

Figure G - 213 Pressure in the RPV

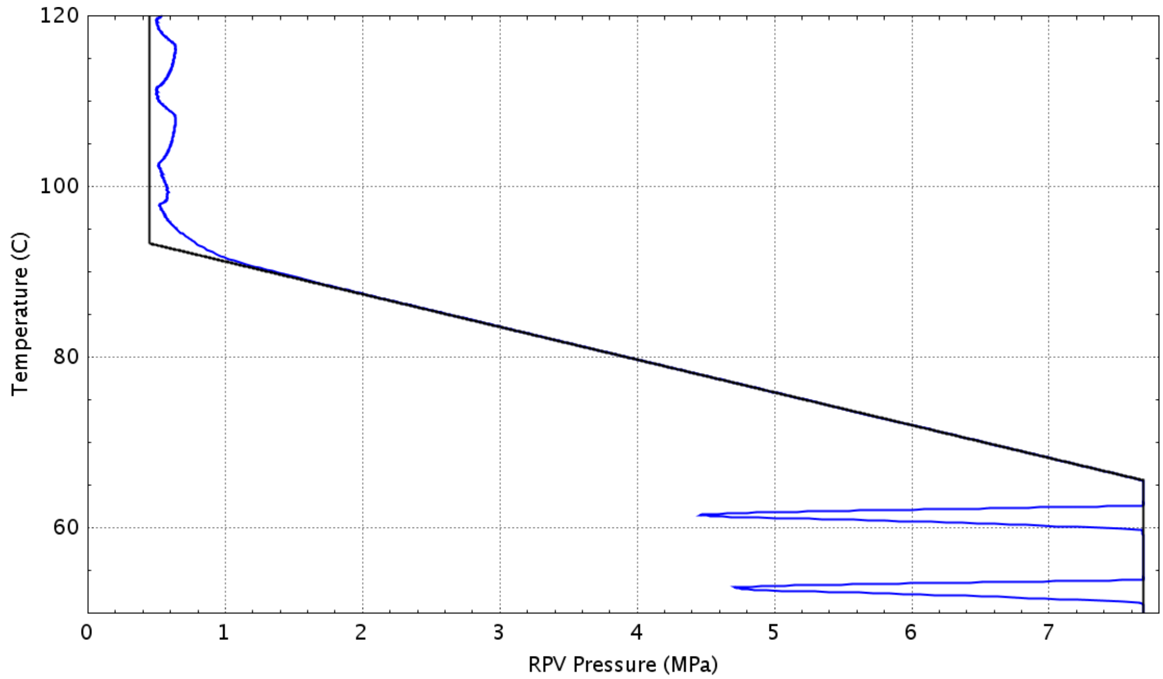

Figure G - 214 Plant status relative to the HCL curve (Graph 4 of the EOPs)

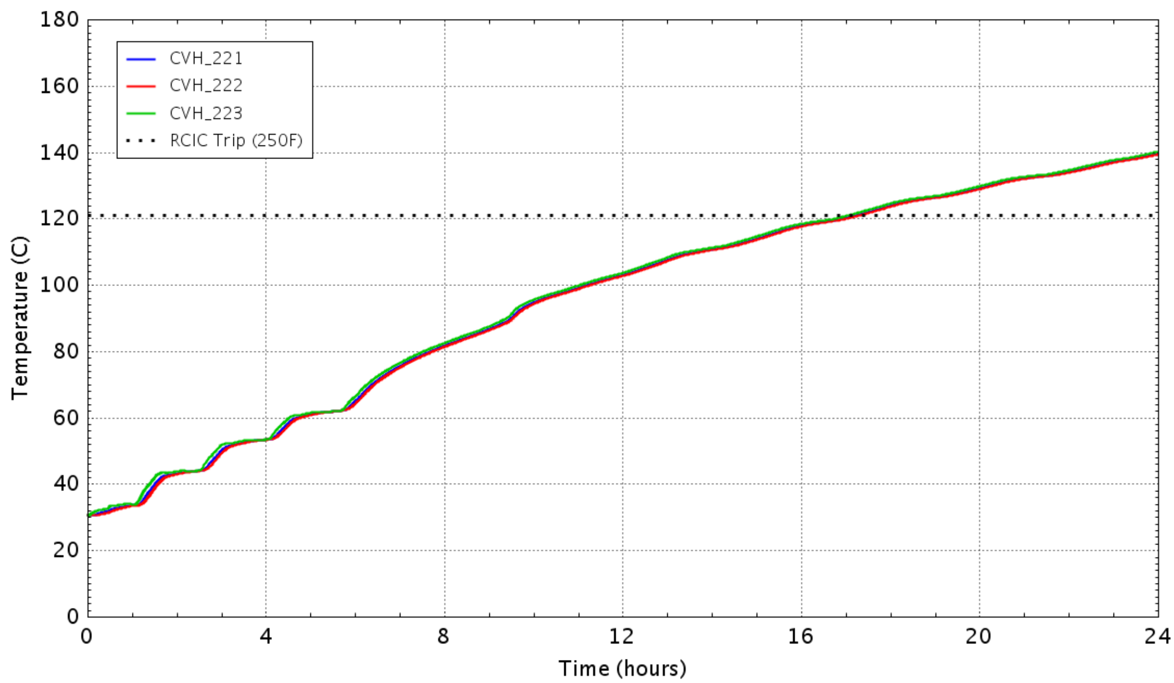

Figure G - 215 Water temperature in the wetwell

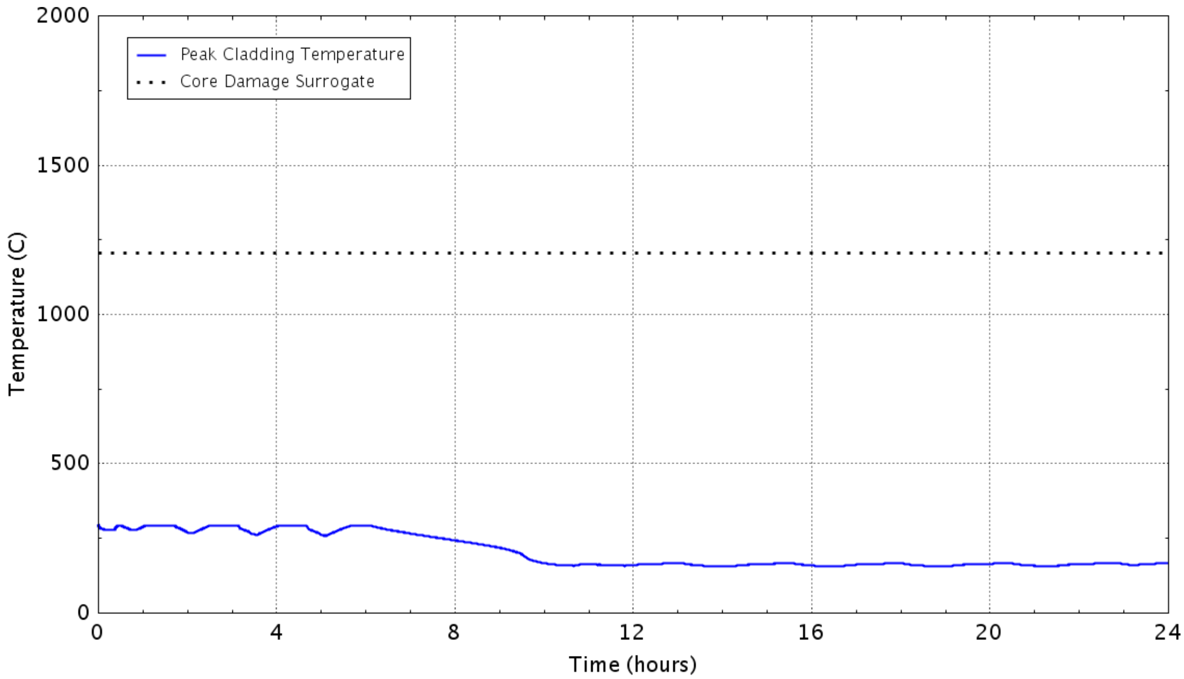

Figure G - 216 Peak temperature of the fuel cladding as a function of time

G.3 Sensitivity Analyses

G.3.1 Case 1a: Sensitivity to LOOPGR-38-9 Case 1 with HPCI Available Instead of RCIC

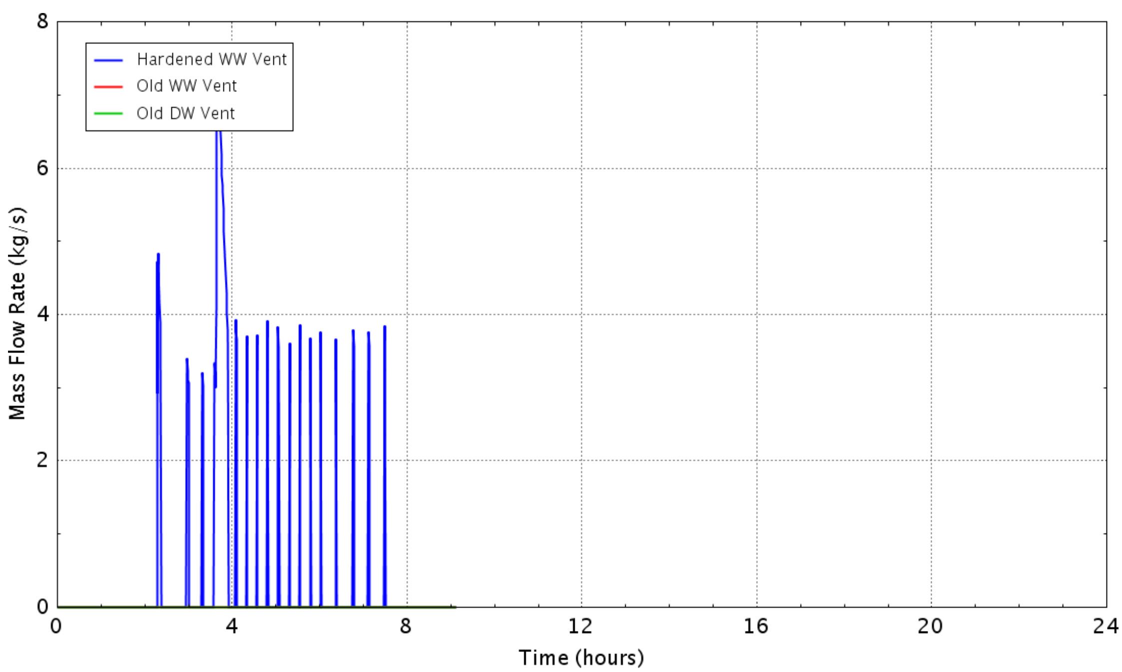

Figure G – 217 Flow rate of the containment vents

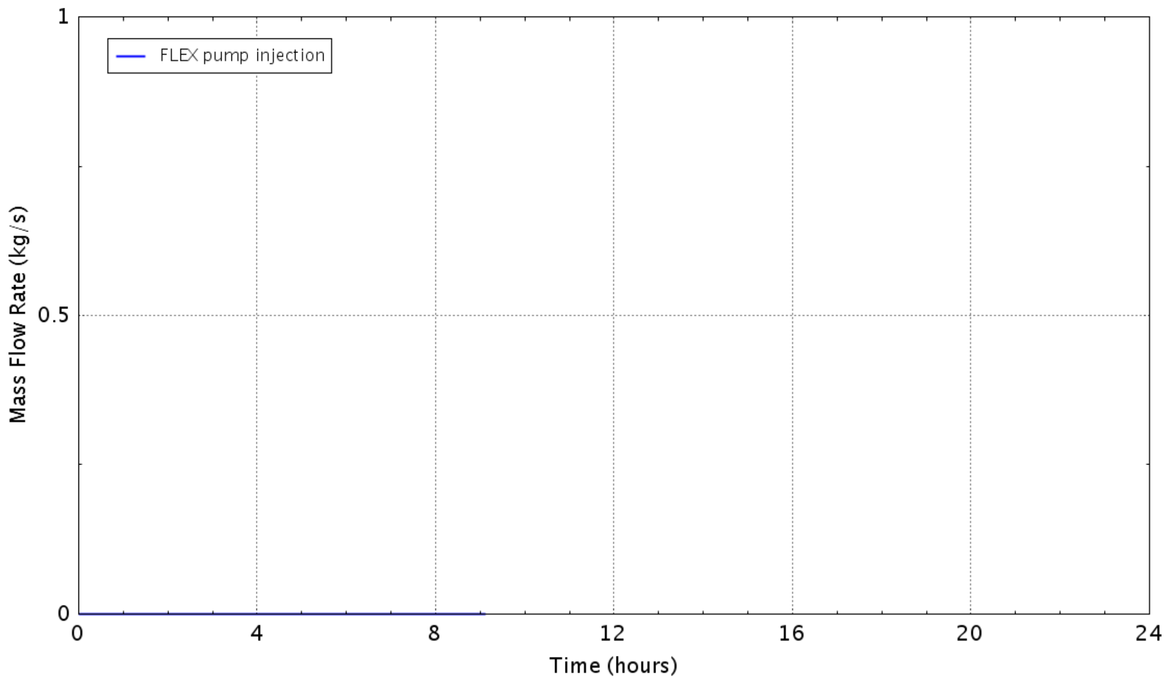

Figure G – 218 Flow rate of the FLEX pump

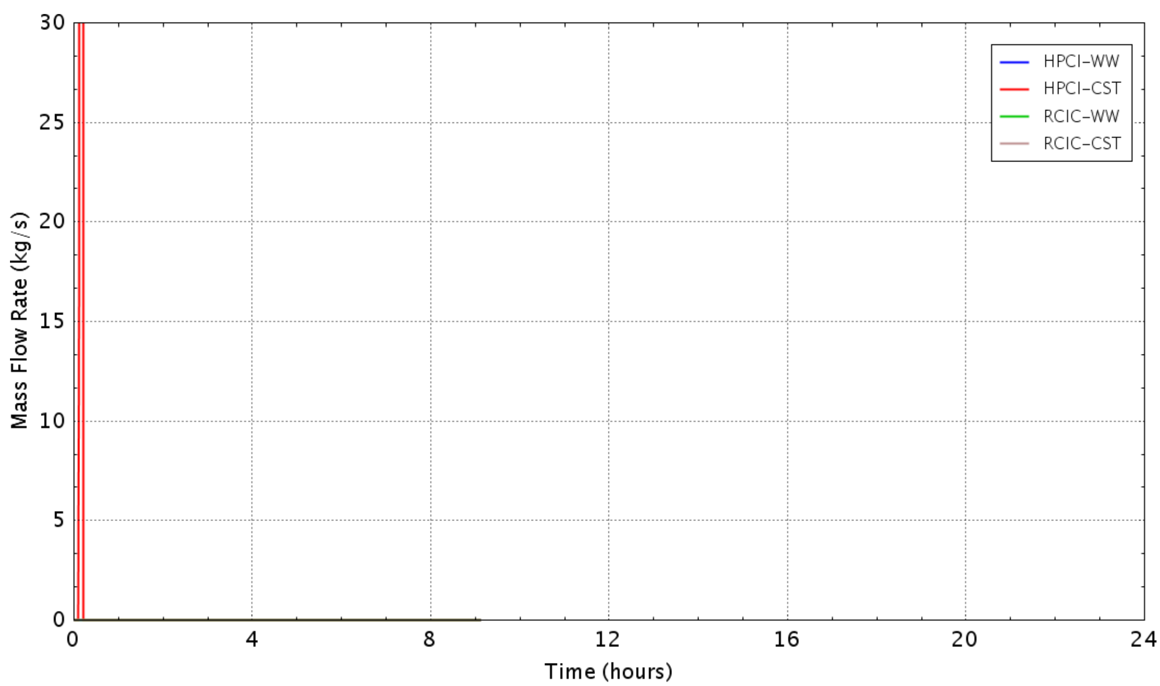

Figure G – 219 Flow rate of the HPCI/RCIC pumps

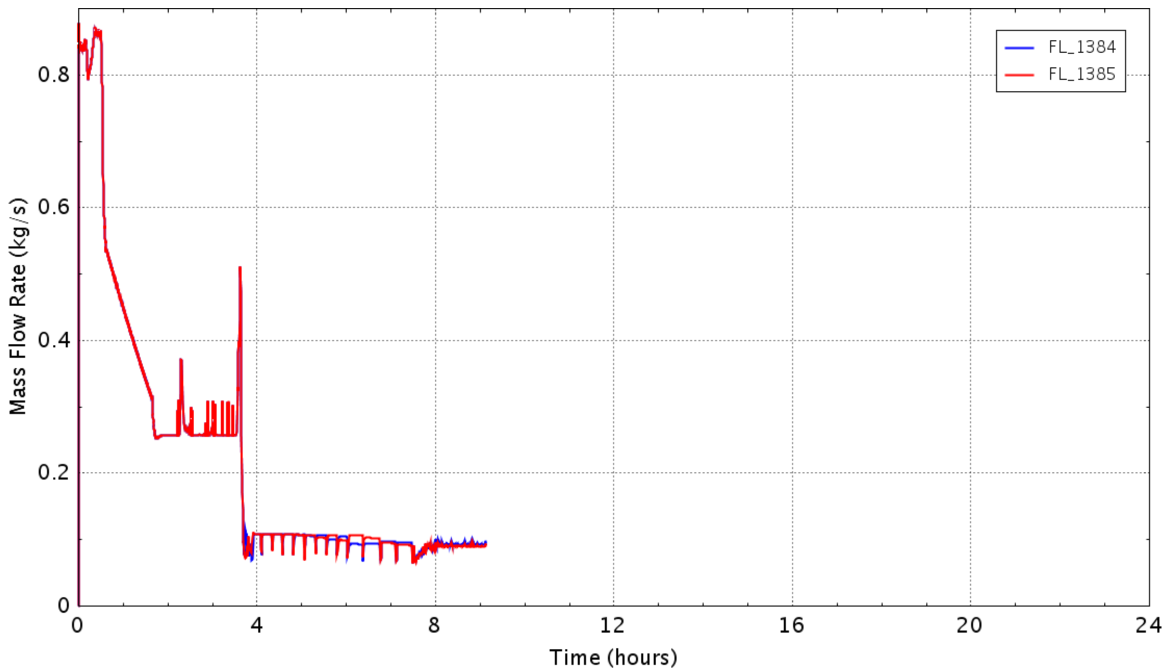

Figure G – 220 Flow rate of the recirculating pump seal leakage

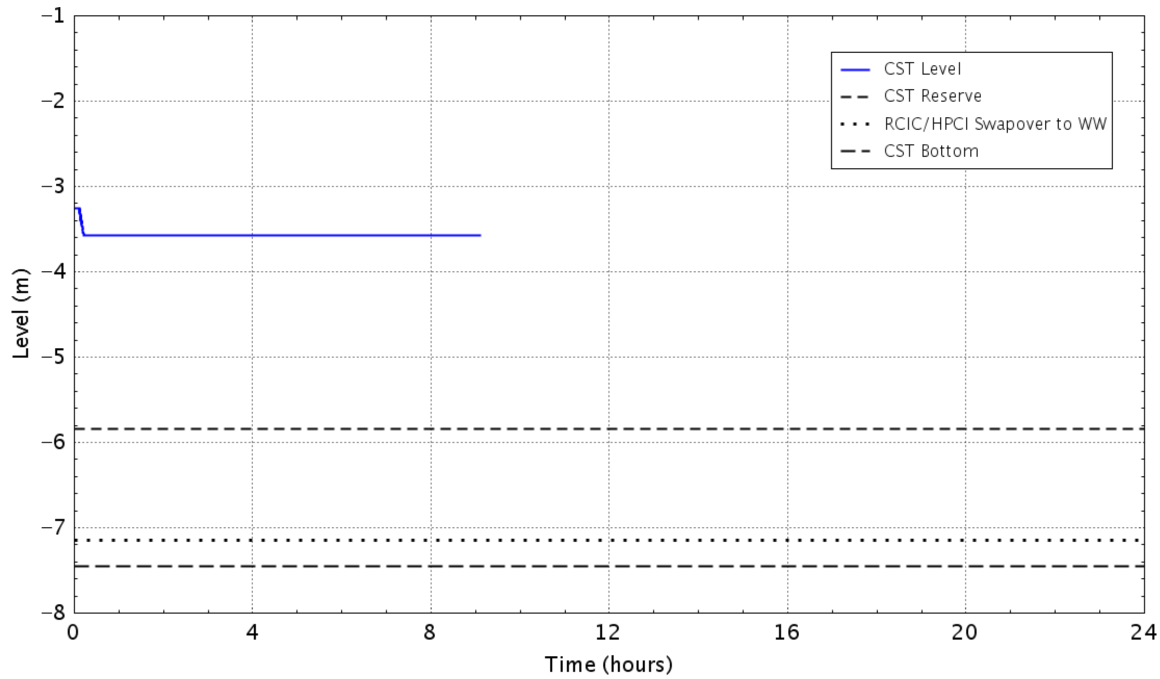

Figure G – 221 Water level in the CST

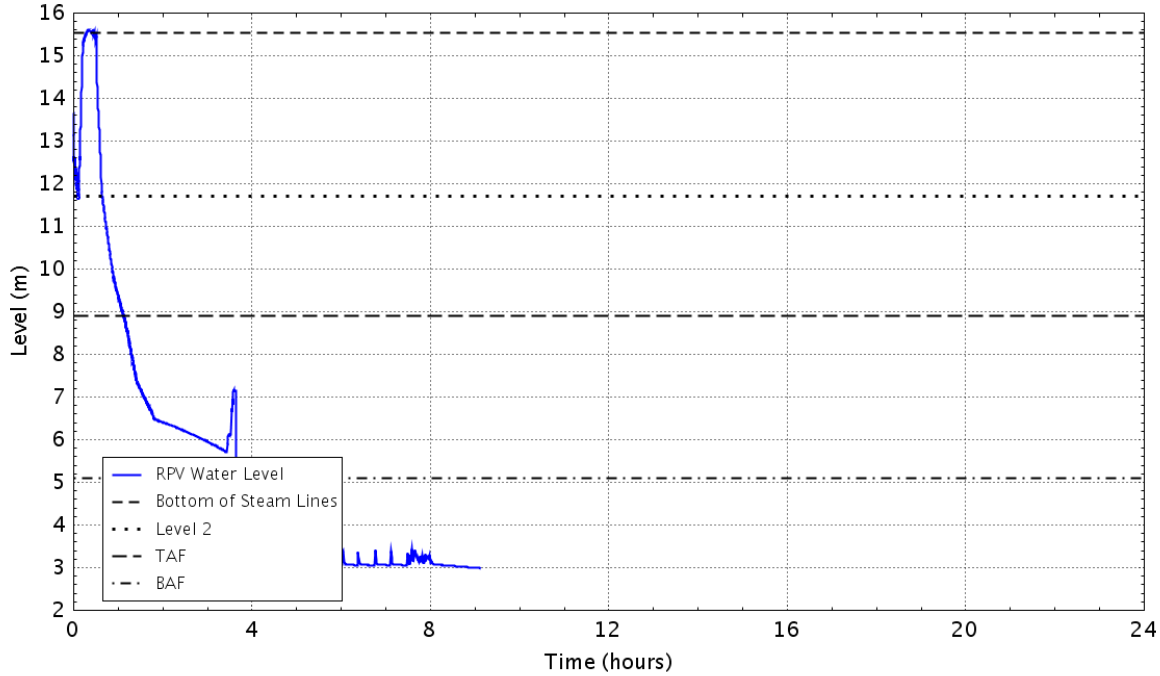

Figure G – 222 RPV Downcomer water level

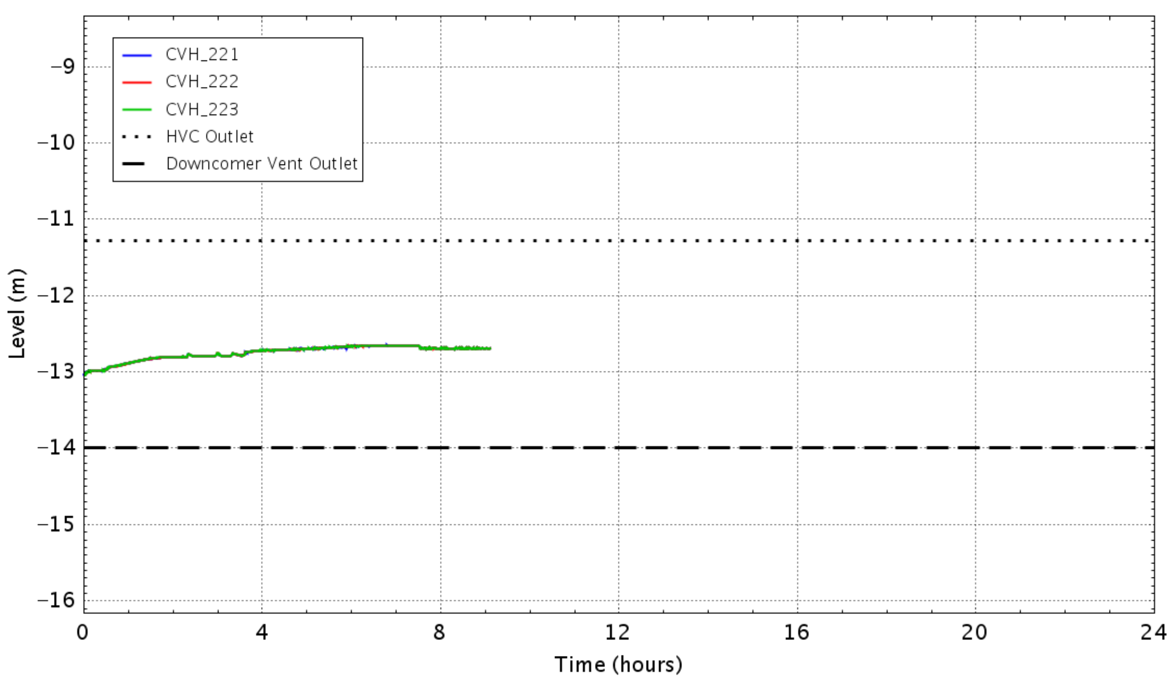

Figure G – 223 Water level in the wetwell

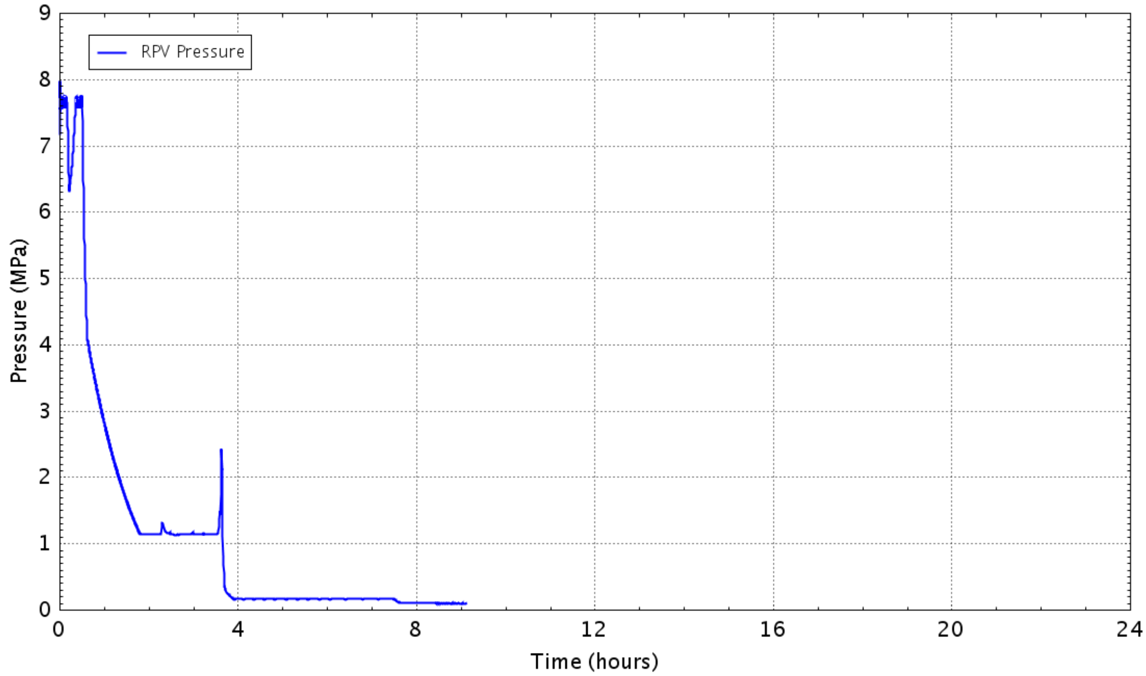

Figure G – 224 Pressure in the RPV

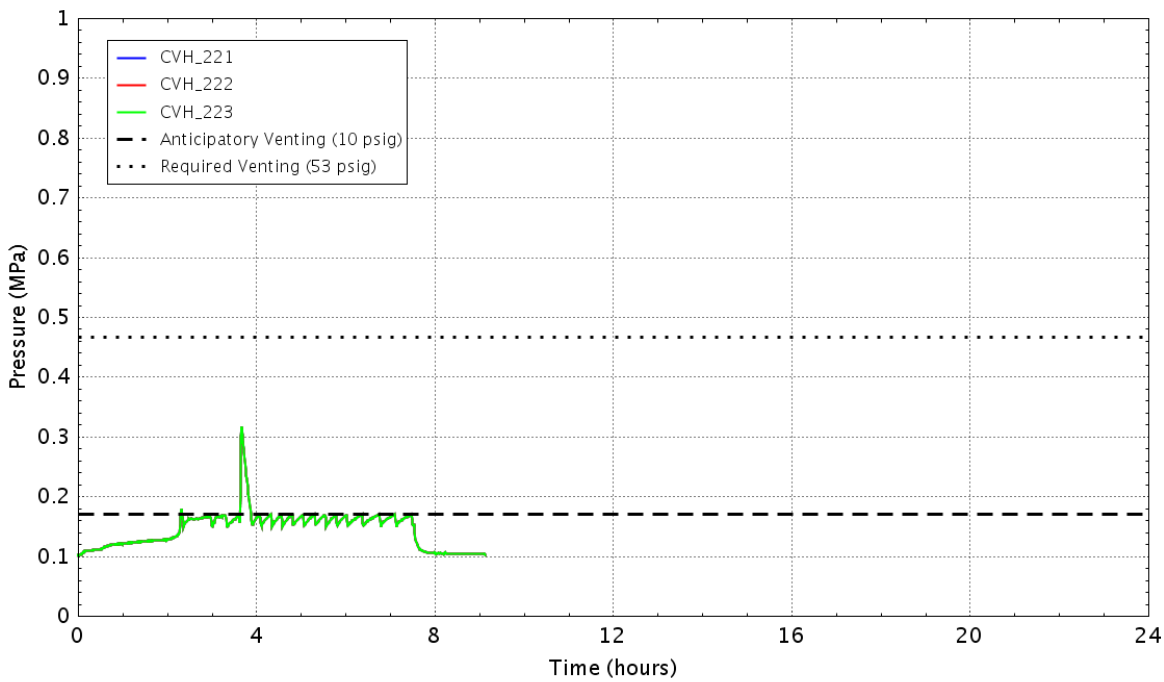

Figure G – 225 Pressure in the wetwell

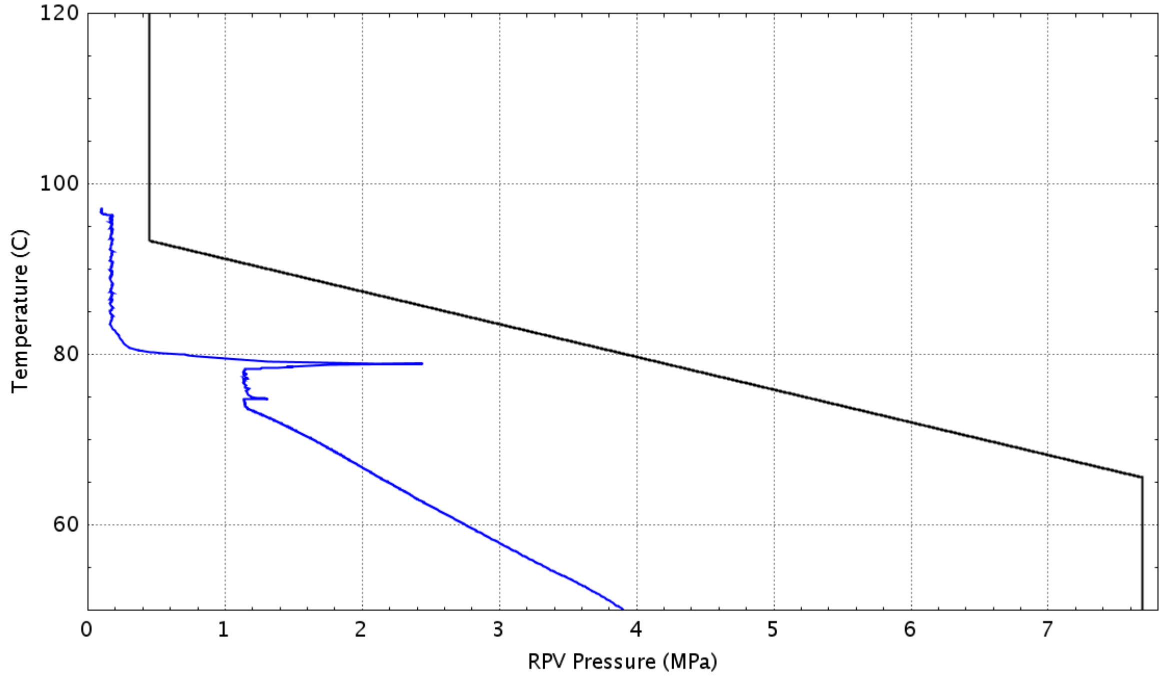

Figure G – 226 Plant status relative to the HCL curve (Graph 4 of the EOPs)

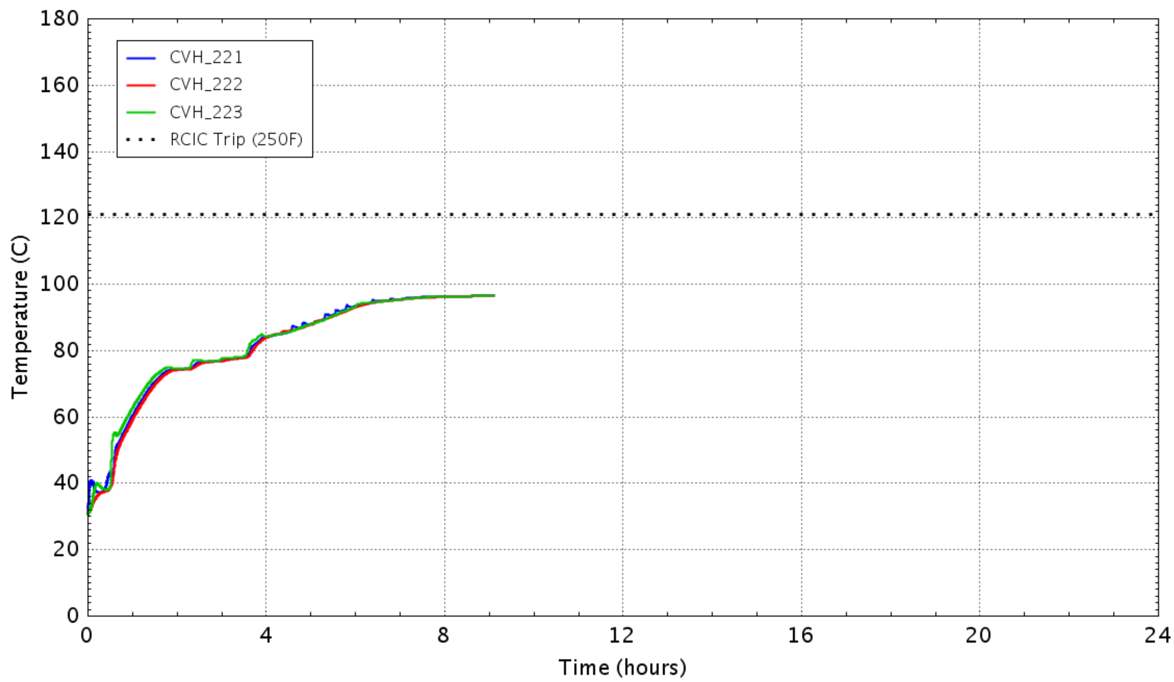

Figure G – 227 Water temperature in the wetwell

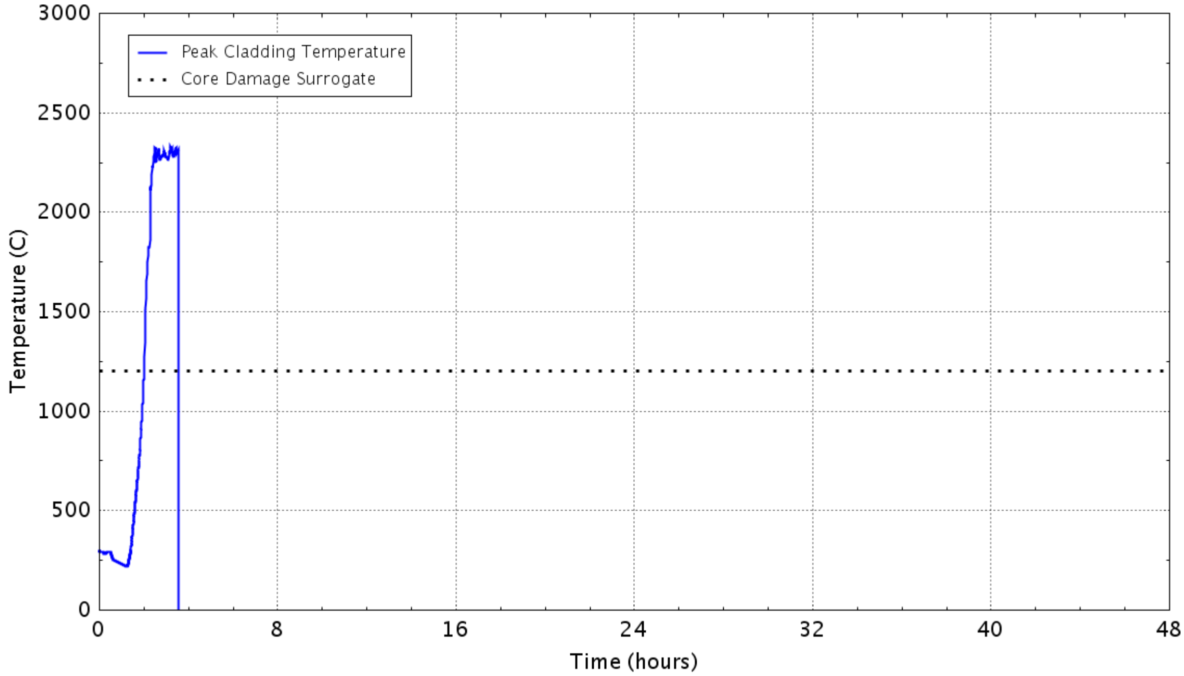

Figure G – 228 Peak temperature of the fuel cladding as a function of time

G.3.2 Case 5a: Sensitivity to LOOPGR-38-9 Case 5 with HPCI Available Instead of RCIC

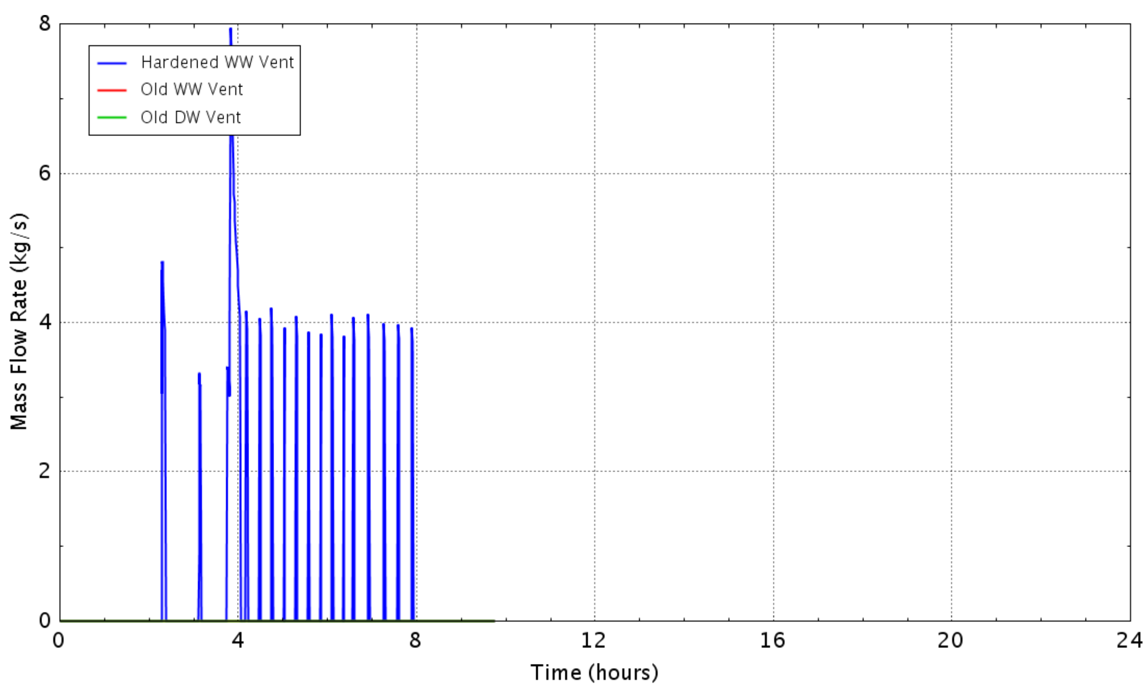

Figure G – 229 Flow rate of the containment vents

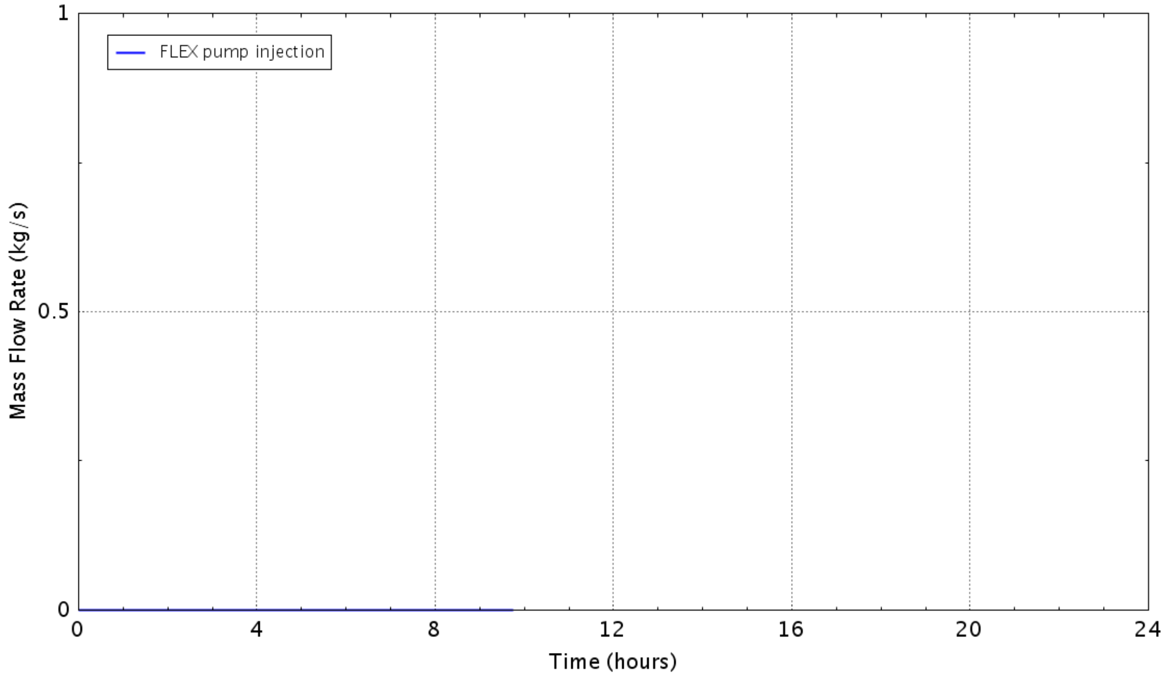

Figure G – 230 Flow rate of the FLEX pump

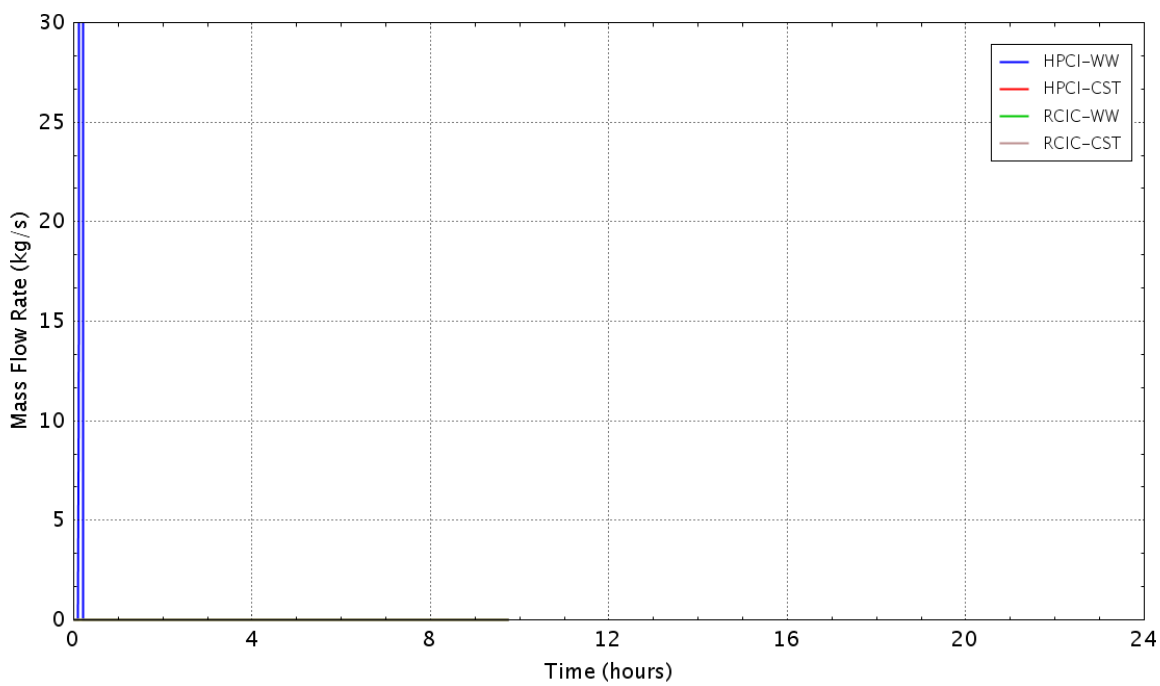

Figure G – 231 Flow rate of the HPCI/RCIC pumps

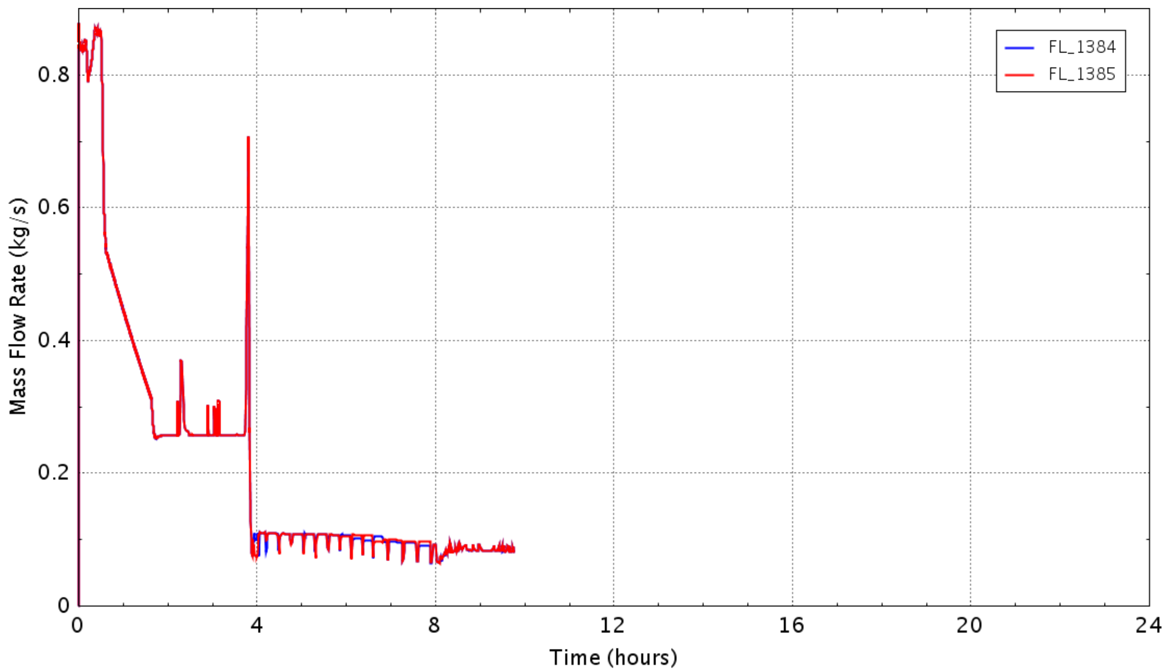

Figure G – 232 Flow rate of the recirculating pump seal leakage

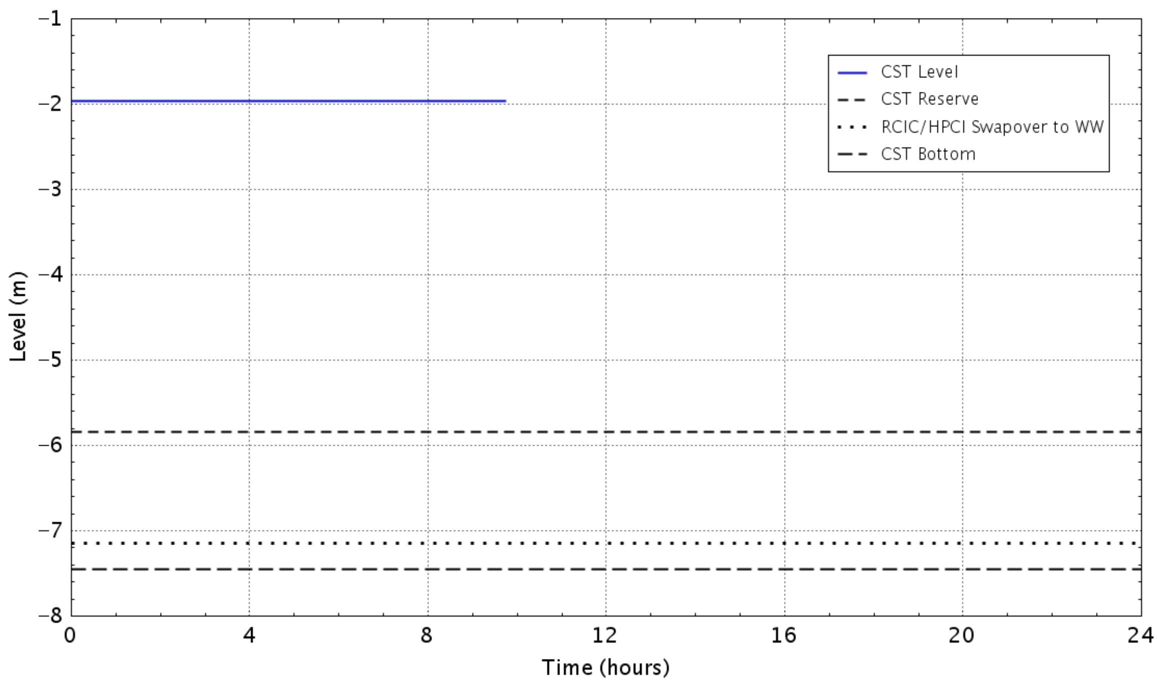

Figure G – 233 Water level in the CST

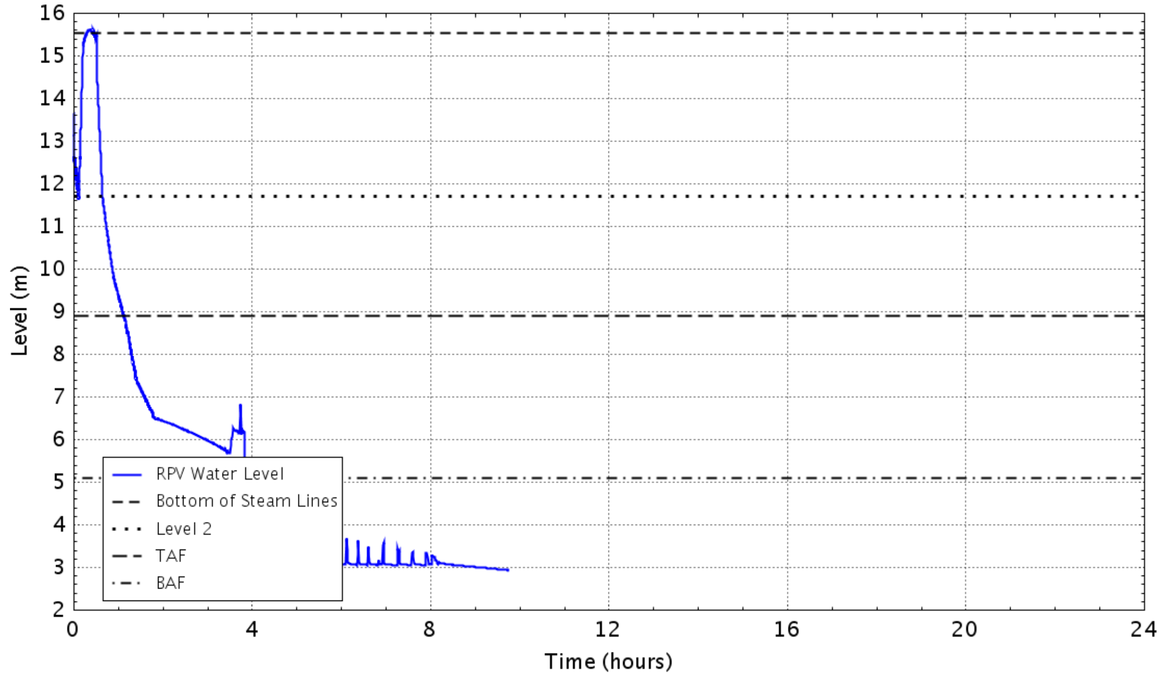

Figure G – 234 RPV Downcomer water level

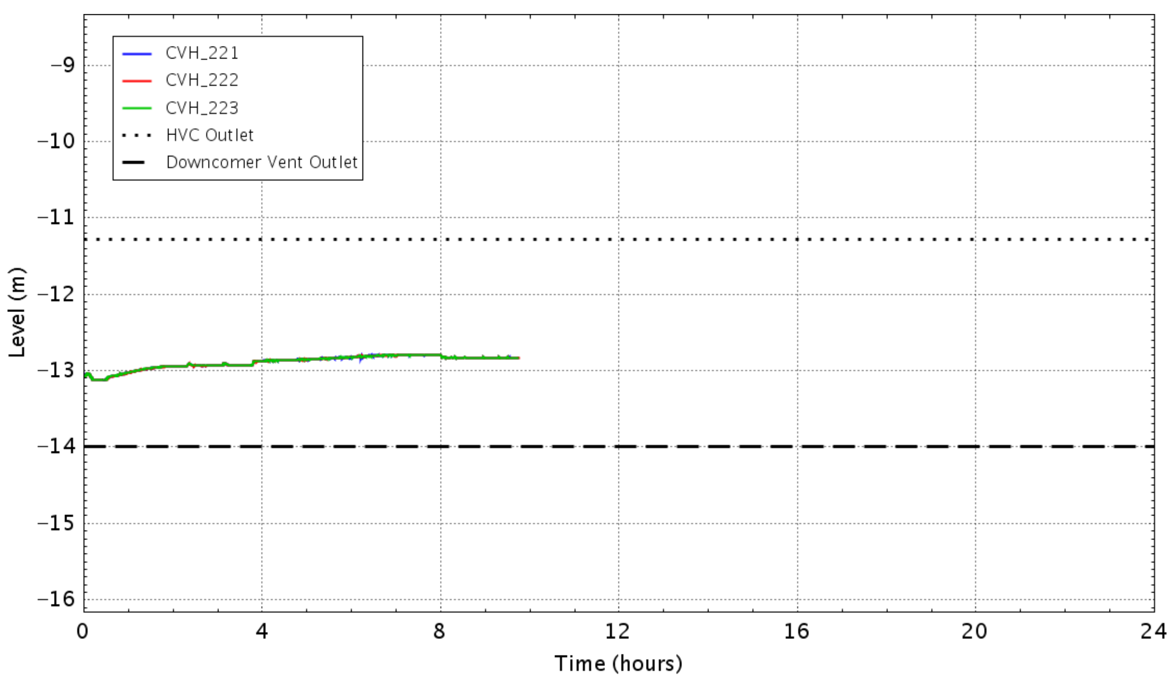

Figure G – 235 Water level in the wetwell

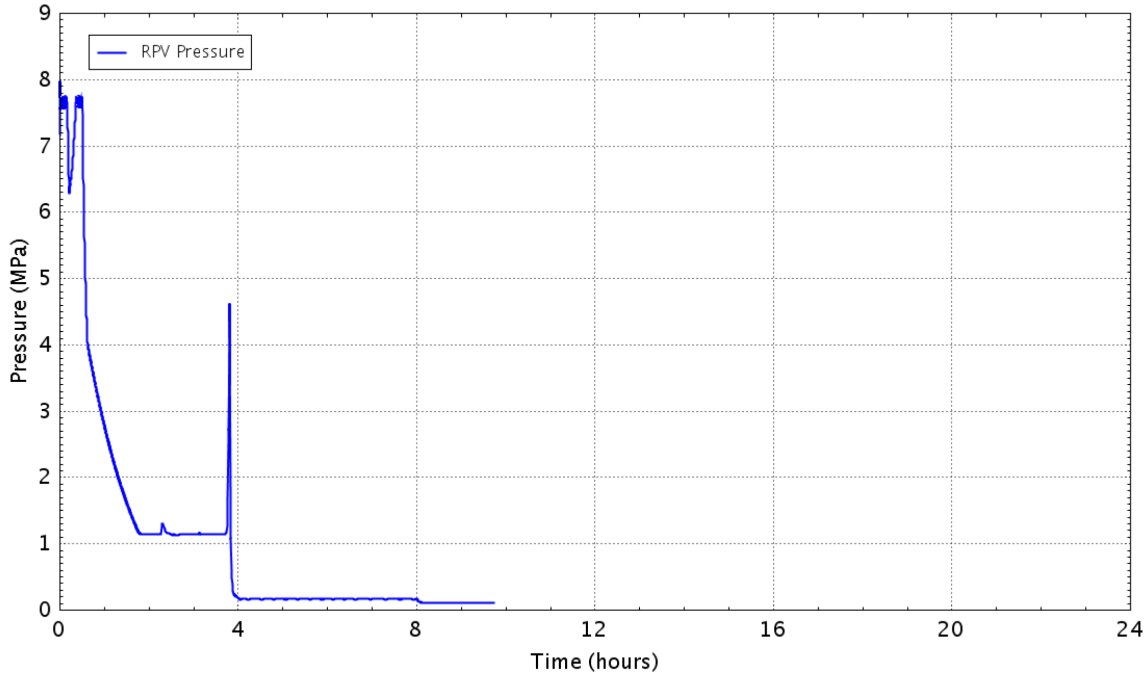

Figure G – 236 Pressure in the RPV

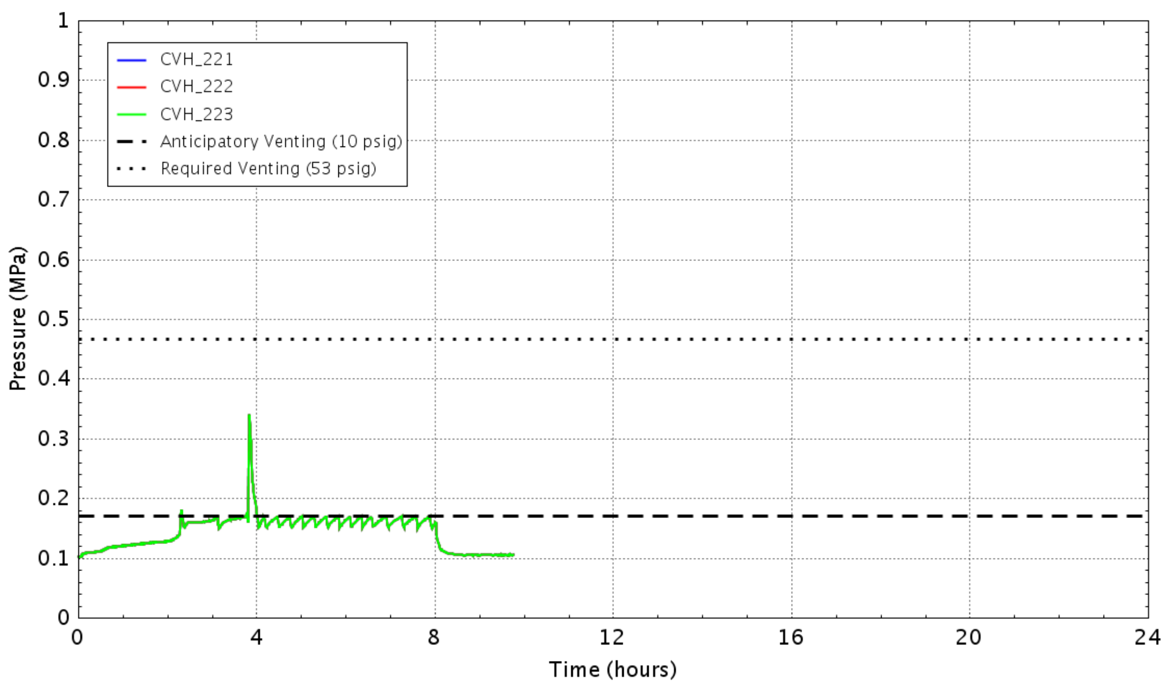

Figure G – 237 Pressure in the wetwell

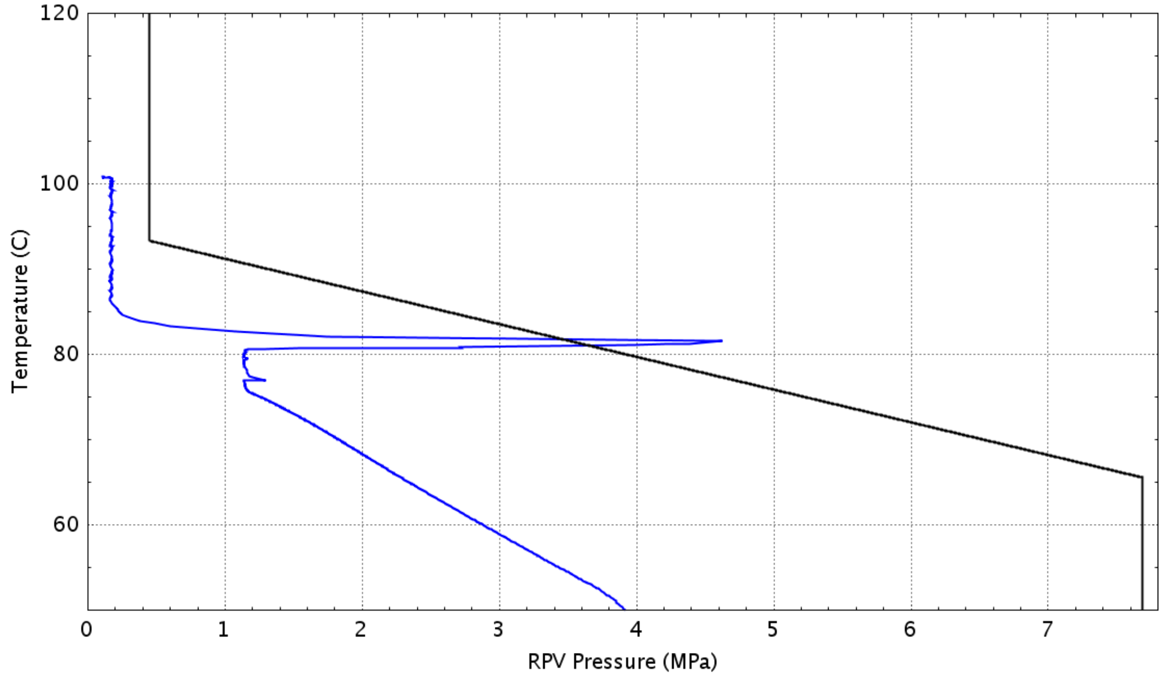

Figure G – 238 Plant status relative to the HCL curve (Graph 4 of the EOPs)

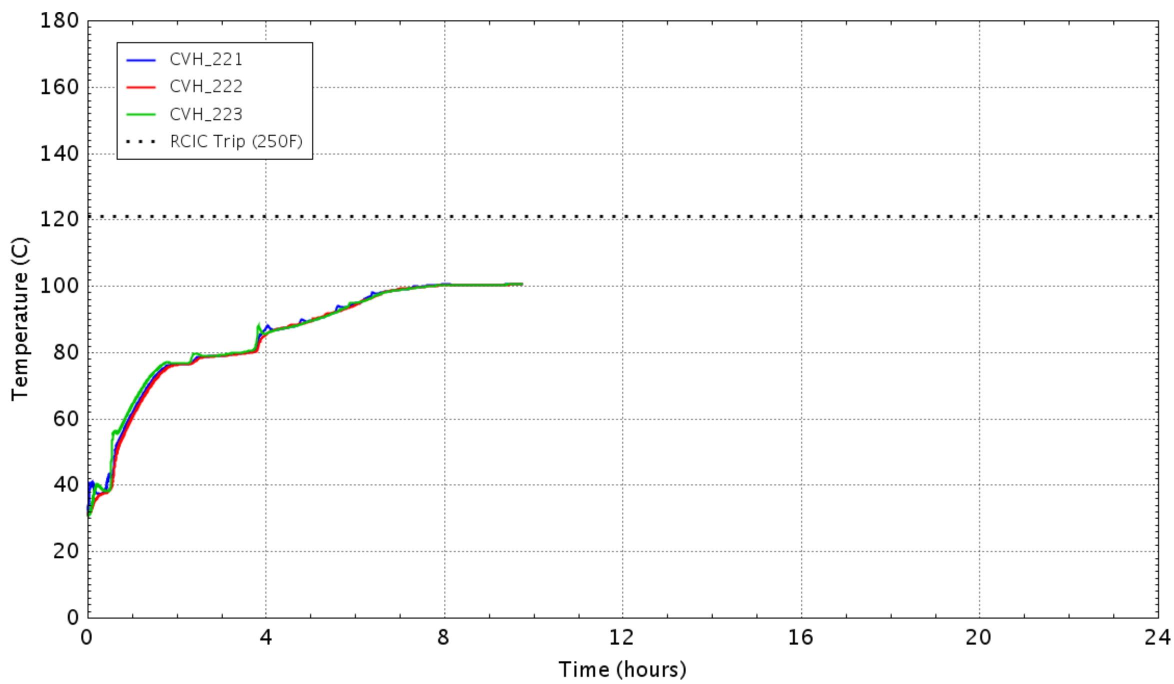

Figure G – 239 Water temperature in the wetwell

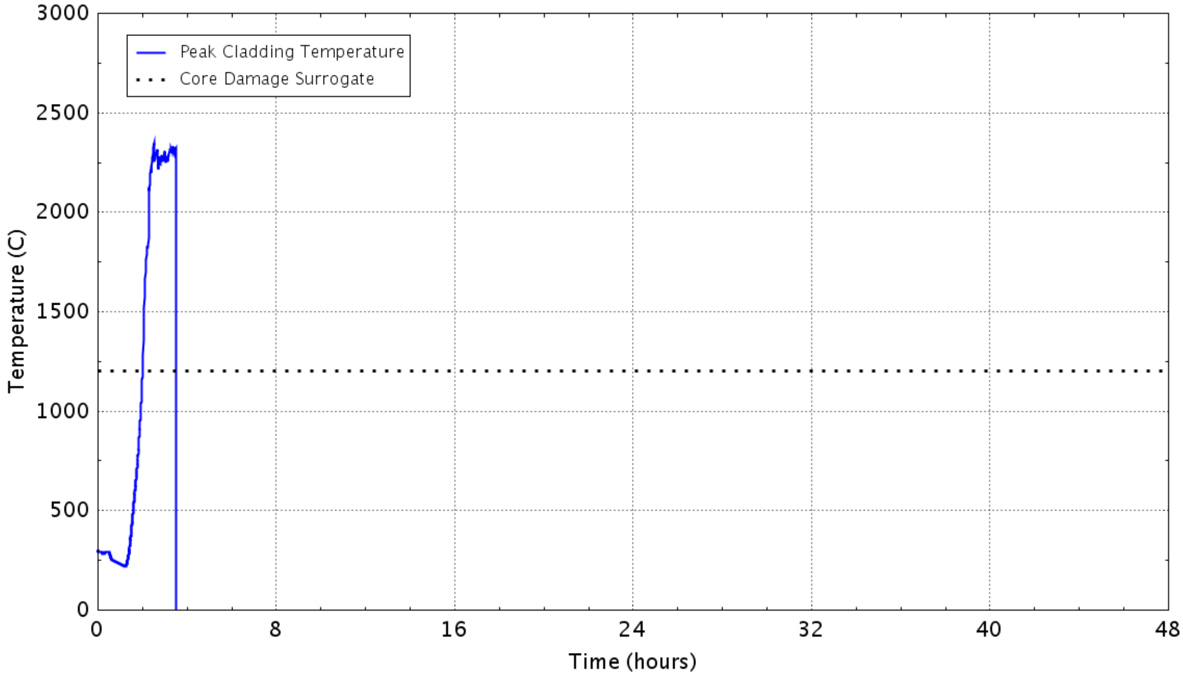

Figure G – 240 Peak temperature of the fuel cladding as a function of time

G.3.3 Case 5b: Sensitivity to LOOPGR-38-9 Case 5 with RCIC Lostwhen NPSH Below the Rated Head

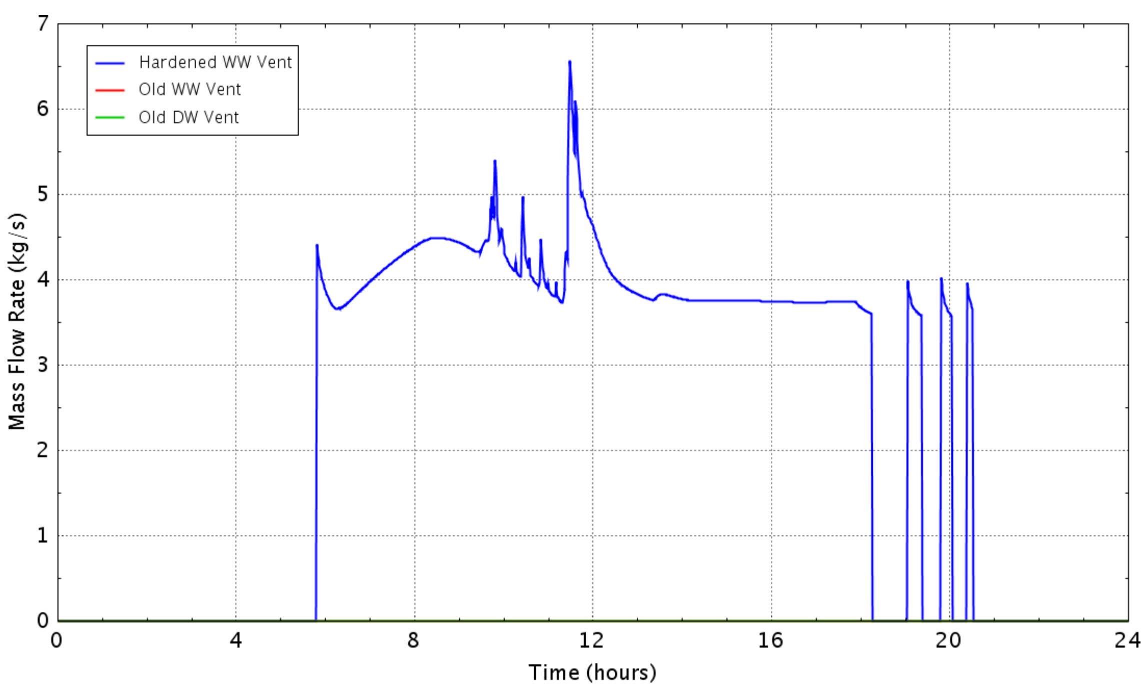

Figure G – 241 Flow rate of the containment vents

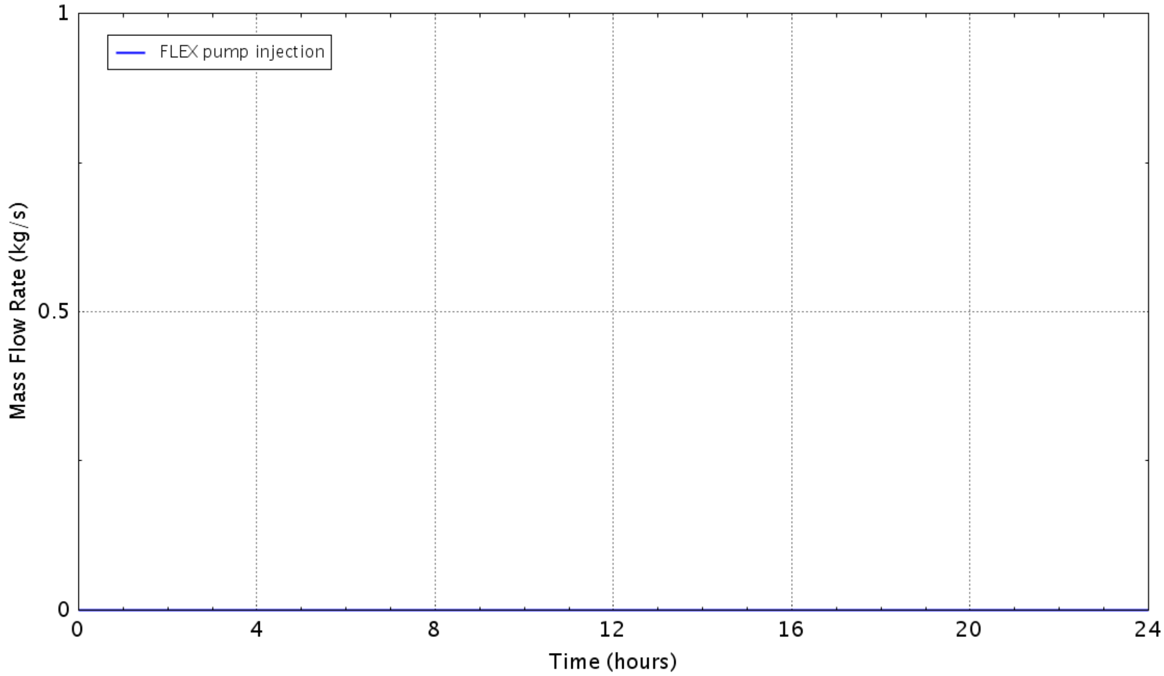

Figure G – 242 Flow rate of the FLEX pump

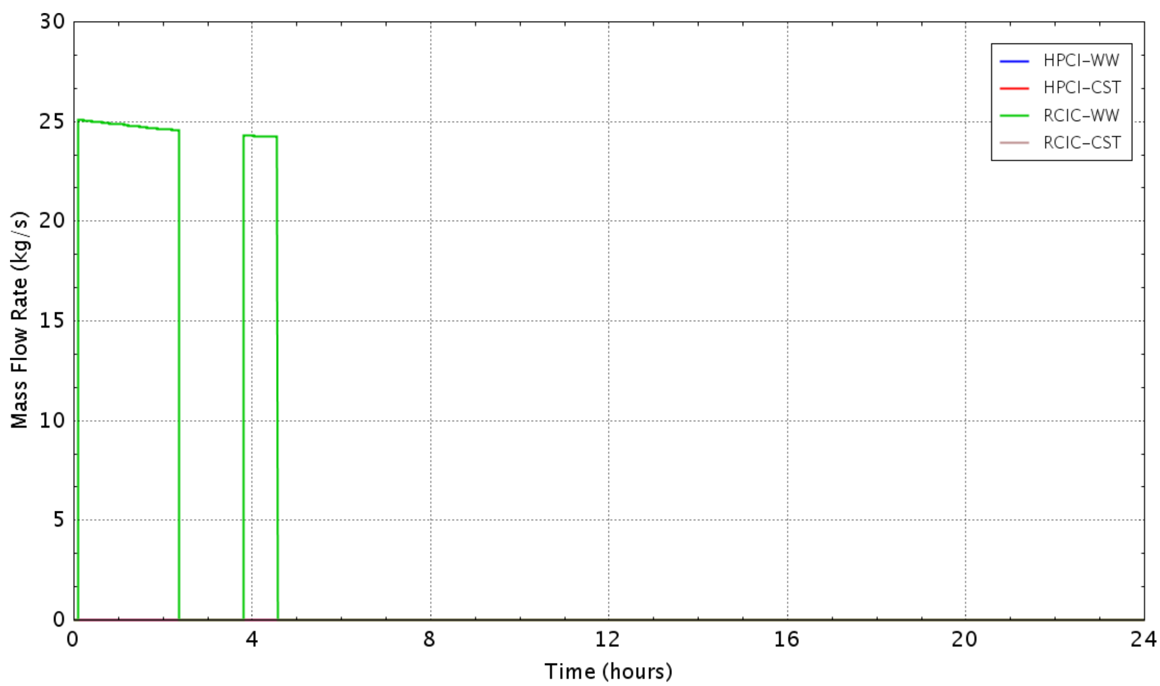

Figure G – 243 Flow rate of the HPCI/RCIC pumps

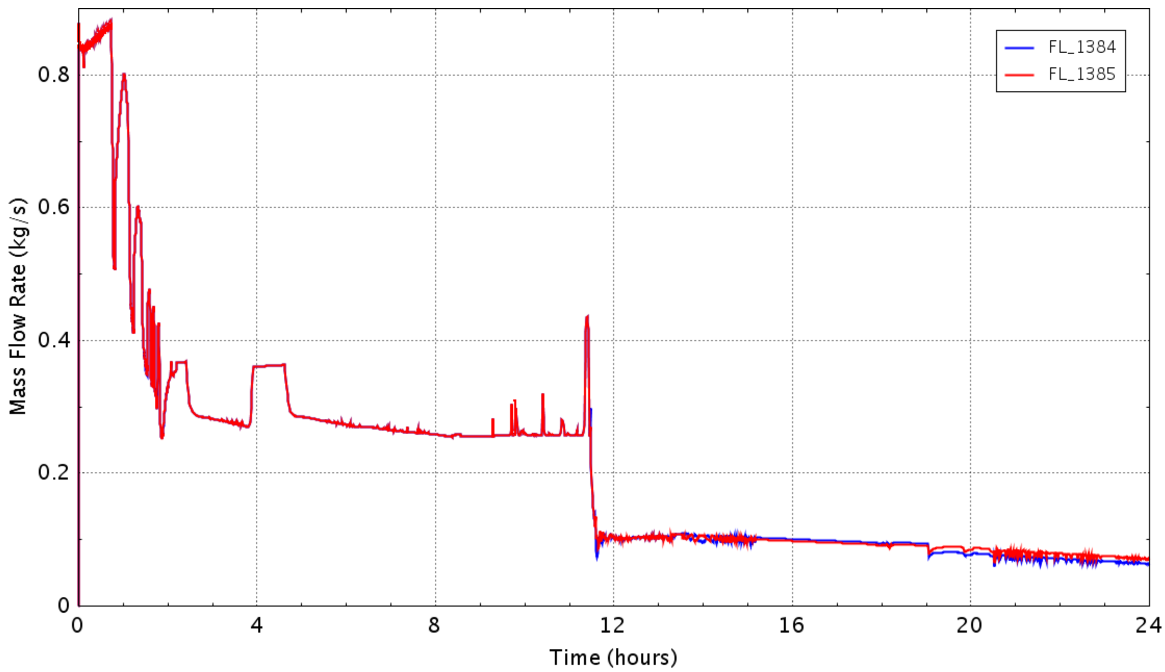

Figure G – 244 Flow rate of the recirculating pump seal leakage

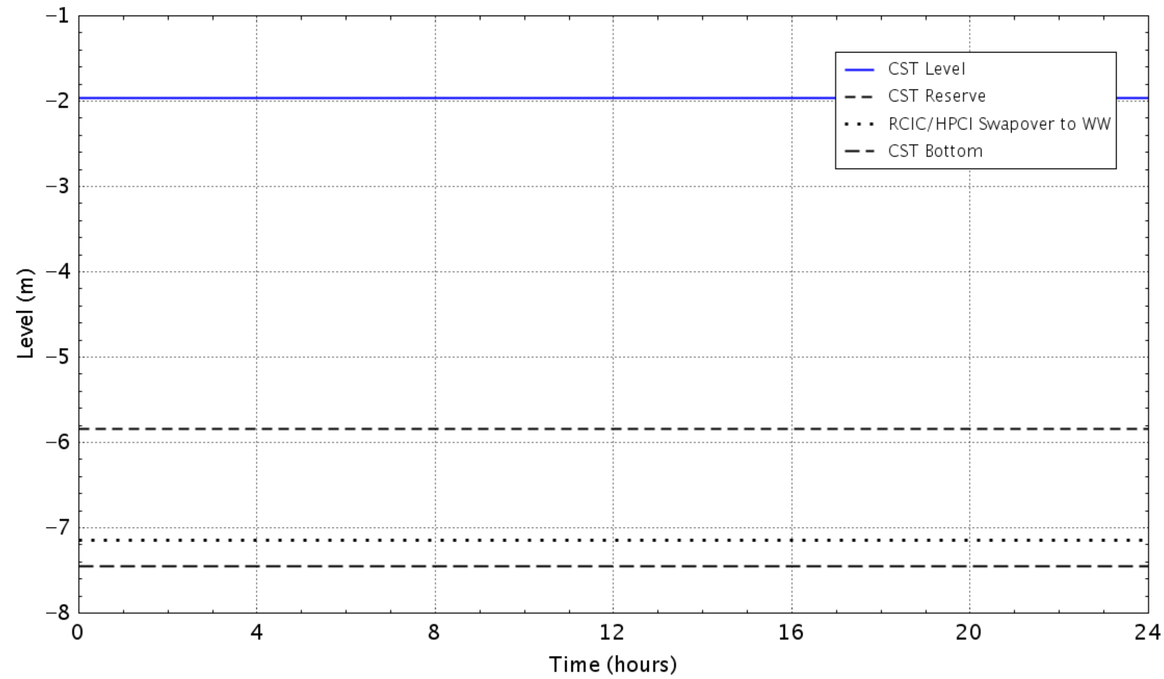

Figure G – 245 Water level in the CST

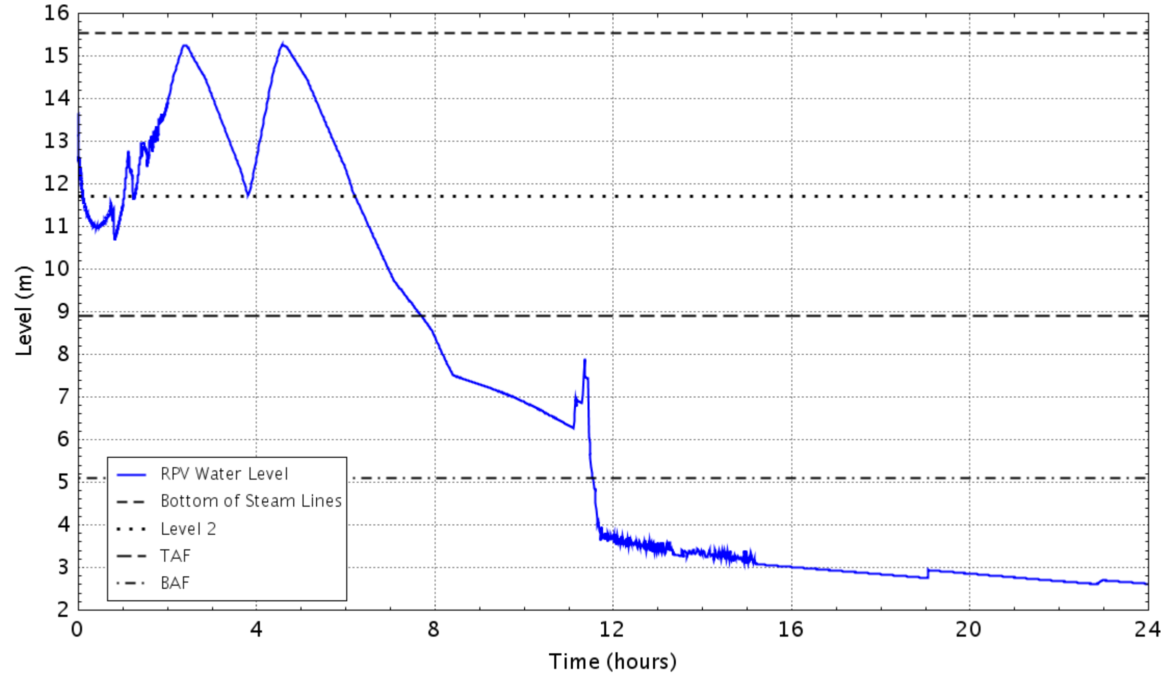

Figure G – 246 RPV Downcomer water level

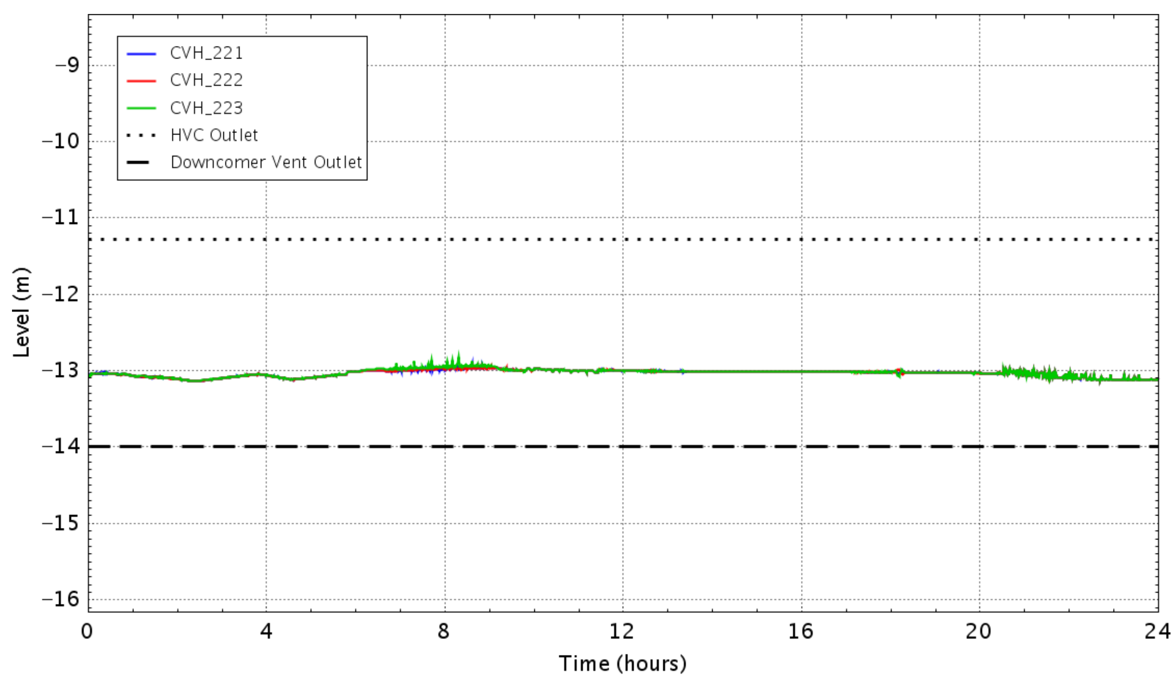

Figure G – 247 Water level in the wetwell

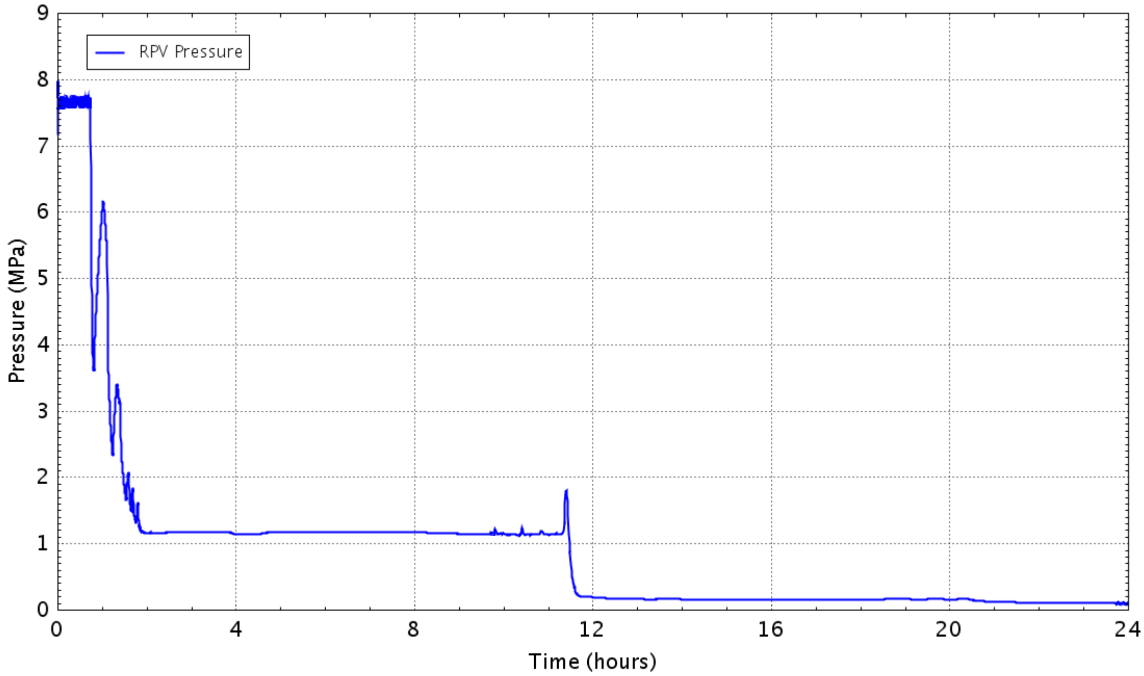

Figure G – 248 Pressure in the RPV

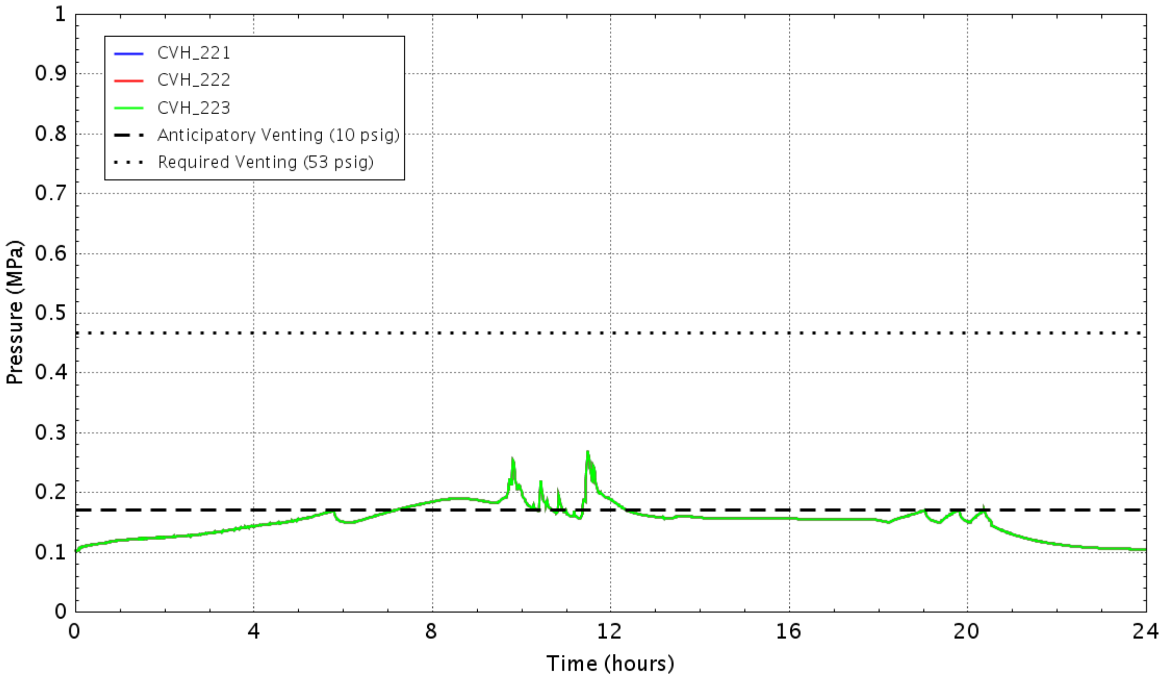

Figure G – 249 Pressure in the wetwell

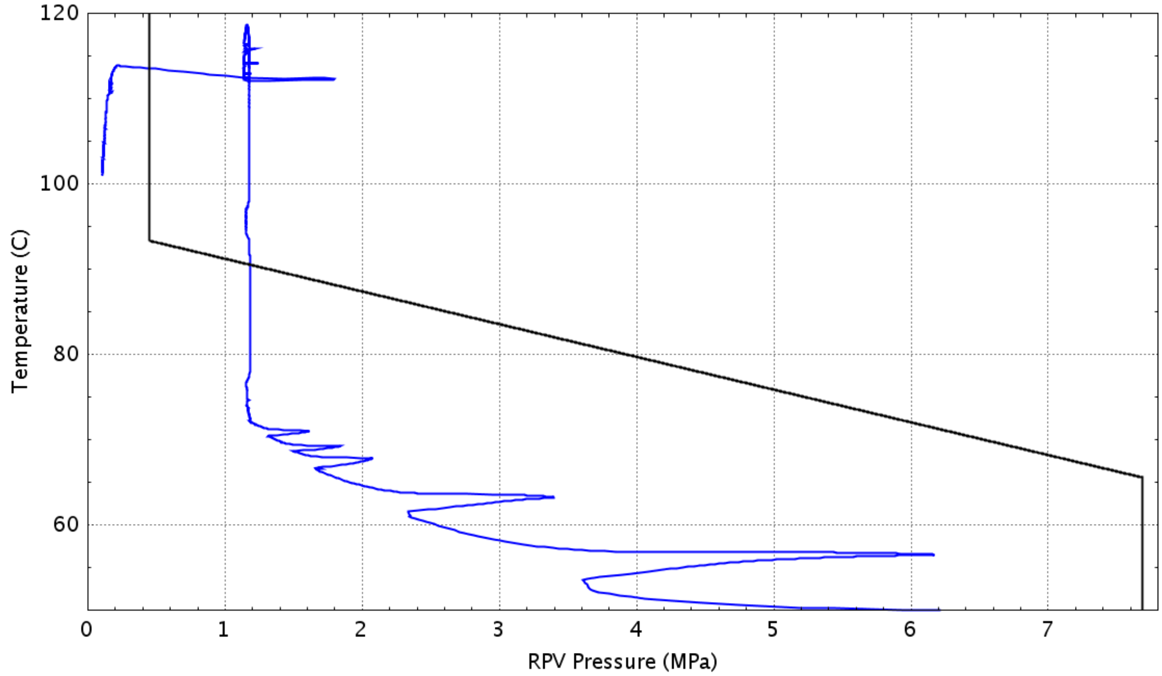

Figure G – 250 Plant status relative to the HCL curve (Graph 4 of the EOPs)

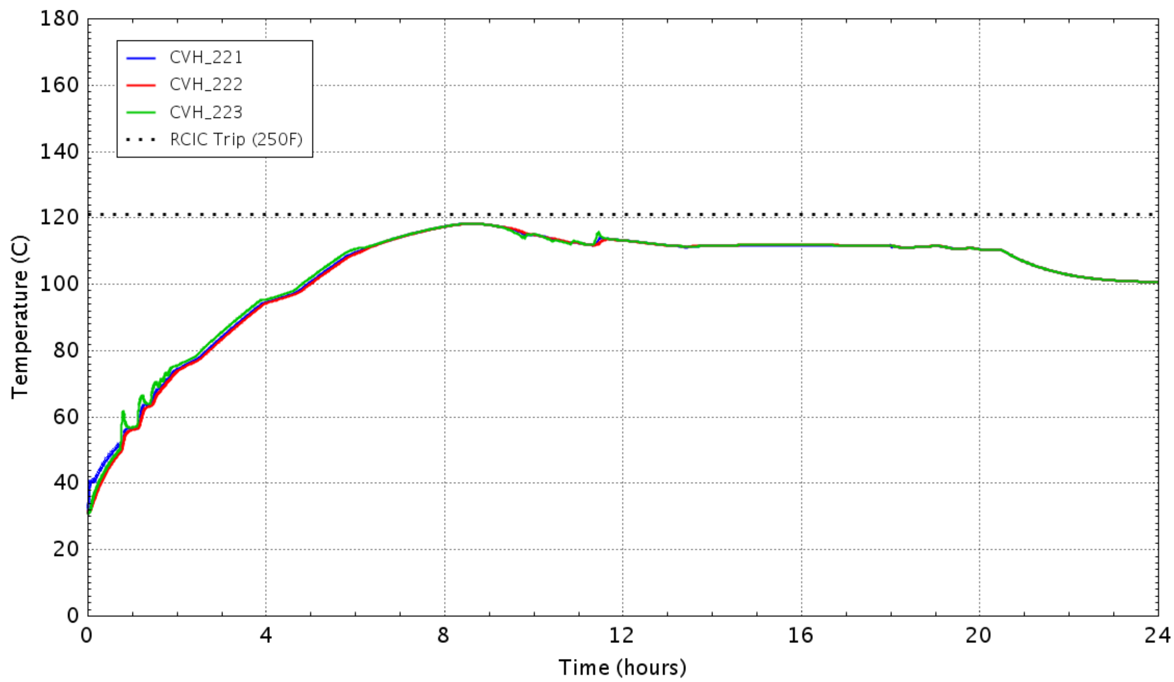

Figure G – 251 Water temperature in the wetwell

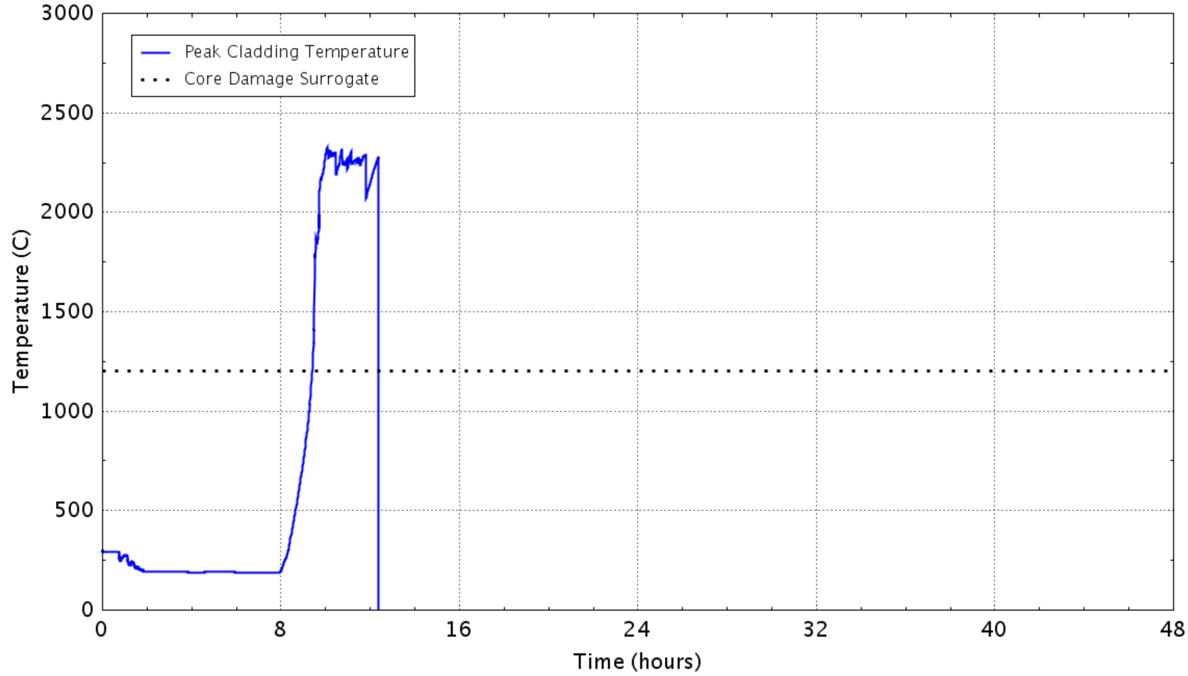

Figure G – 252 Peak temperature of the fuel cladding as a function of time

G.3.4 Case 11a: Sensitivity to LOOPGR-38-9 Case 11 with RCICLost when NPSH Below the Rated Head

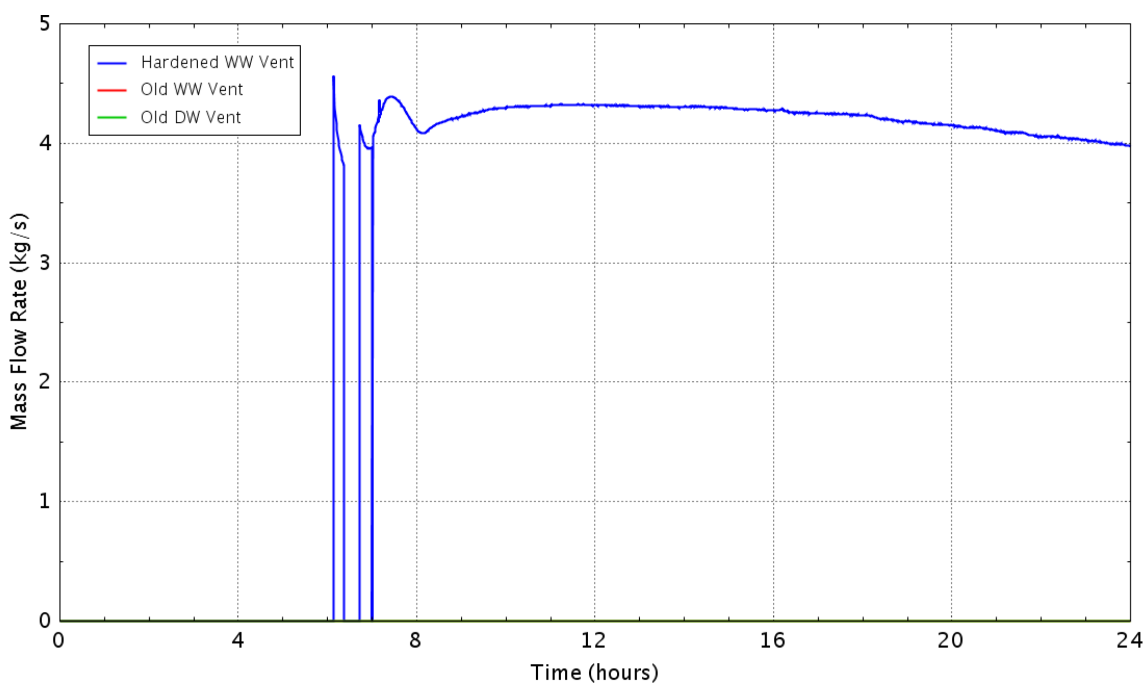

Figure G – 253 Flow rate of the containment vents

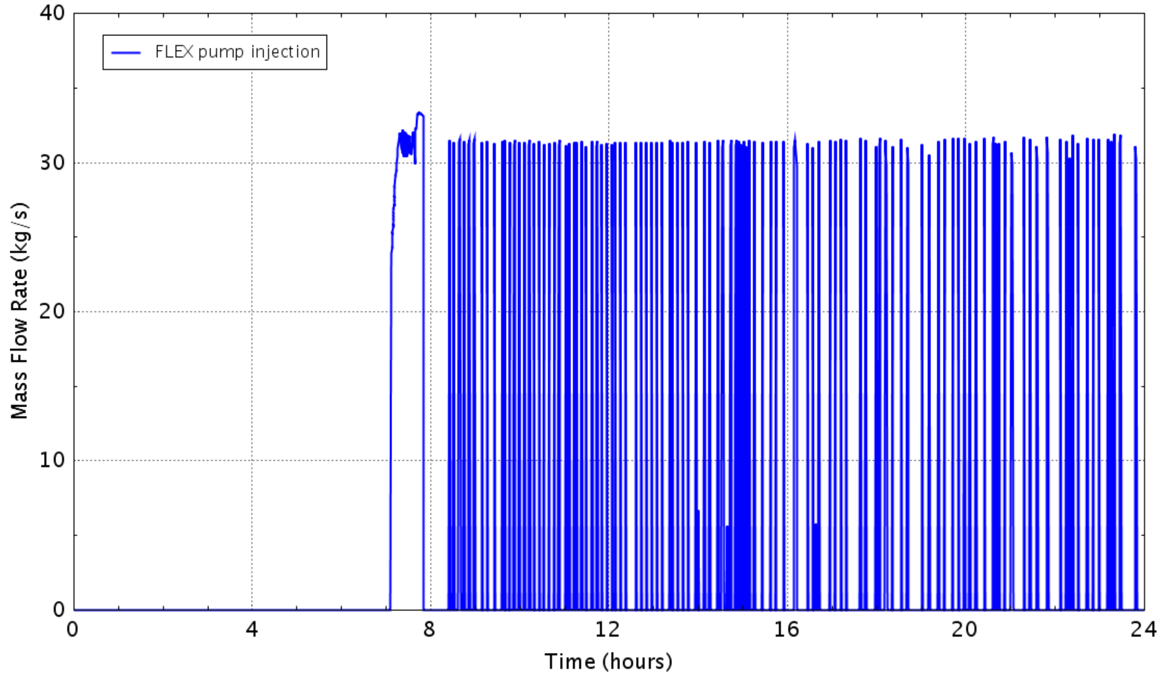

Figure G – 254 Flow rate of the FLEX pump

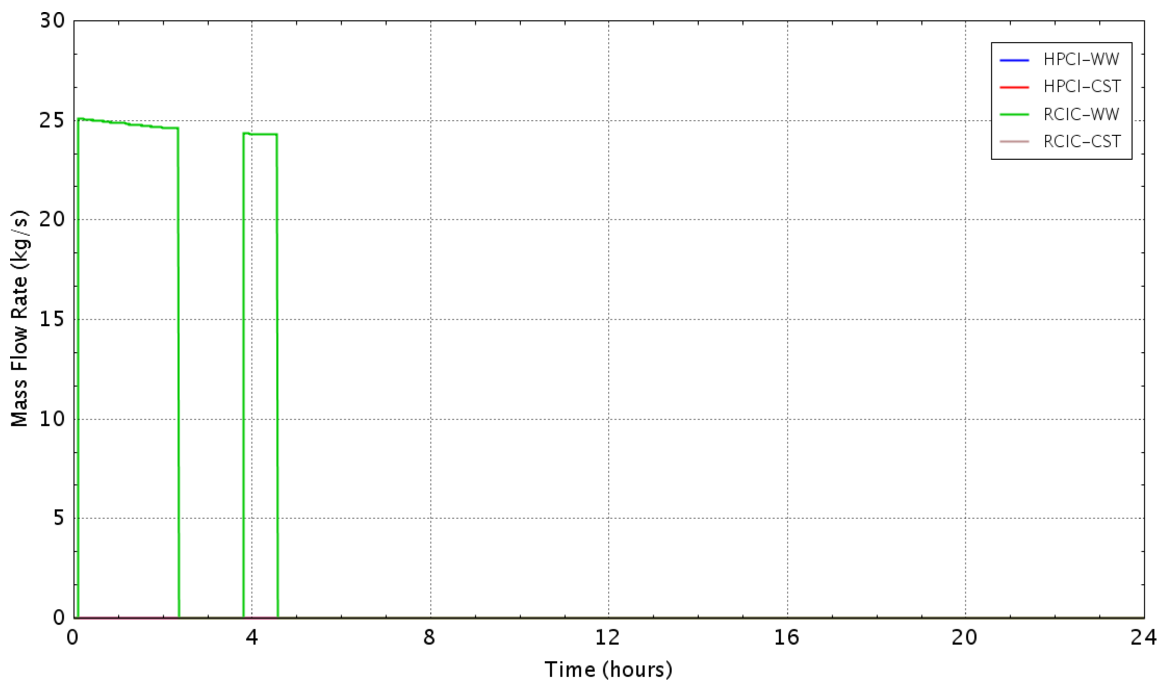

Figure G – 255 Flow rate of the HPCI/RCIC pumps

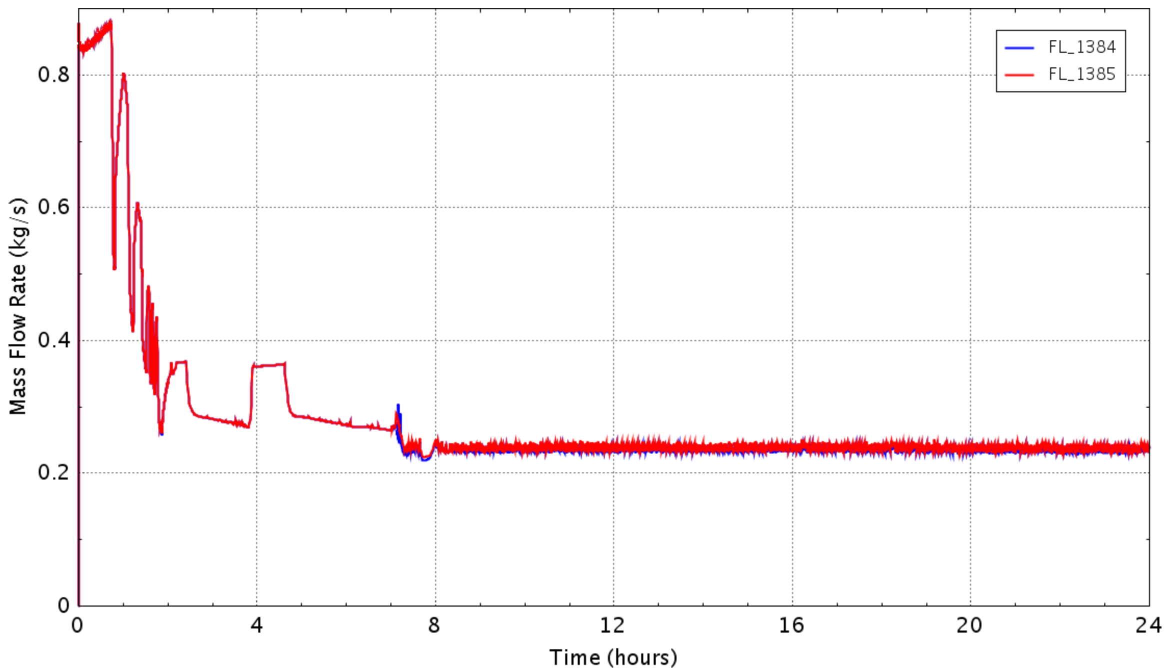

Figure G – 256 Flow rate of the recirculating pump seal leakage

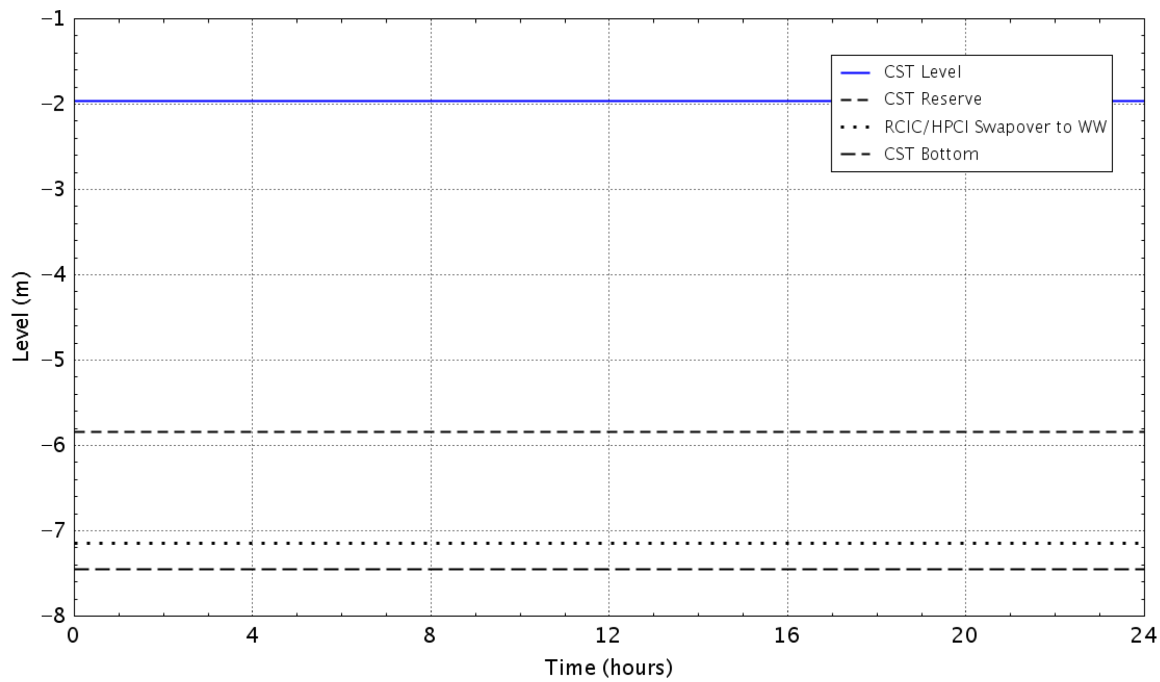

Figure G – 257 Water level in the CST

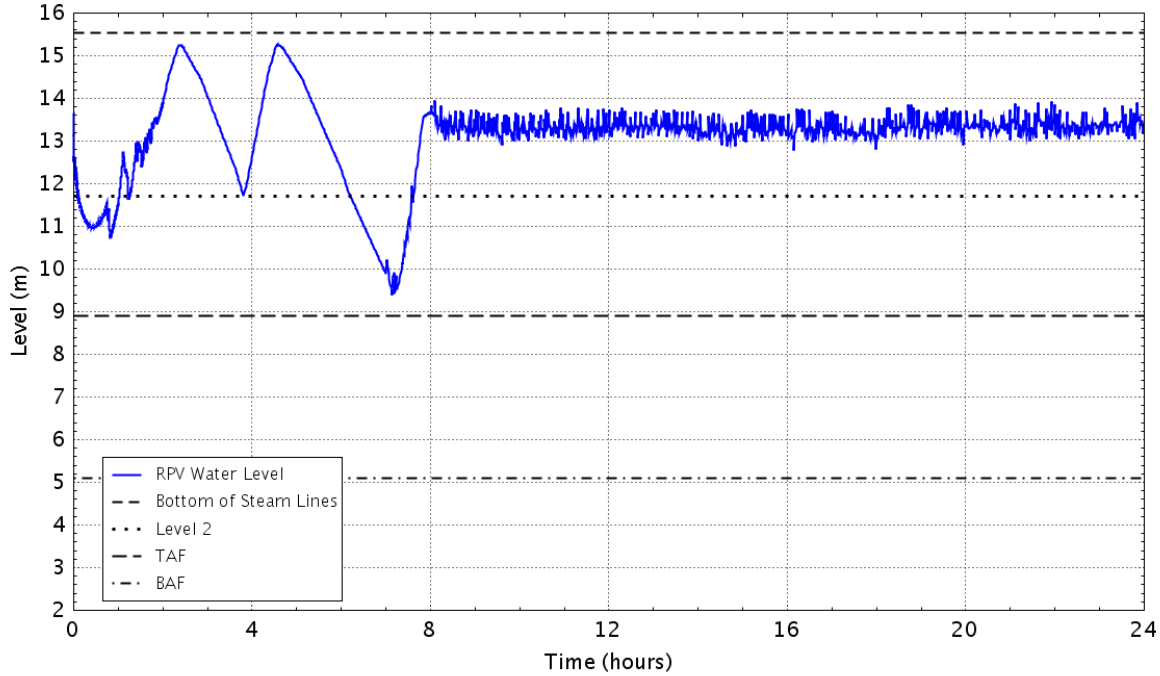

Figure G – 258 RPV Downcomer water level

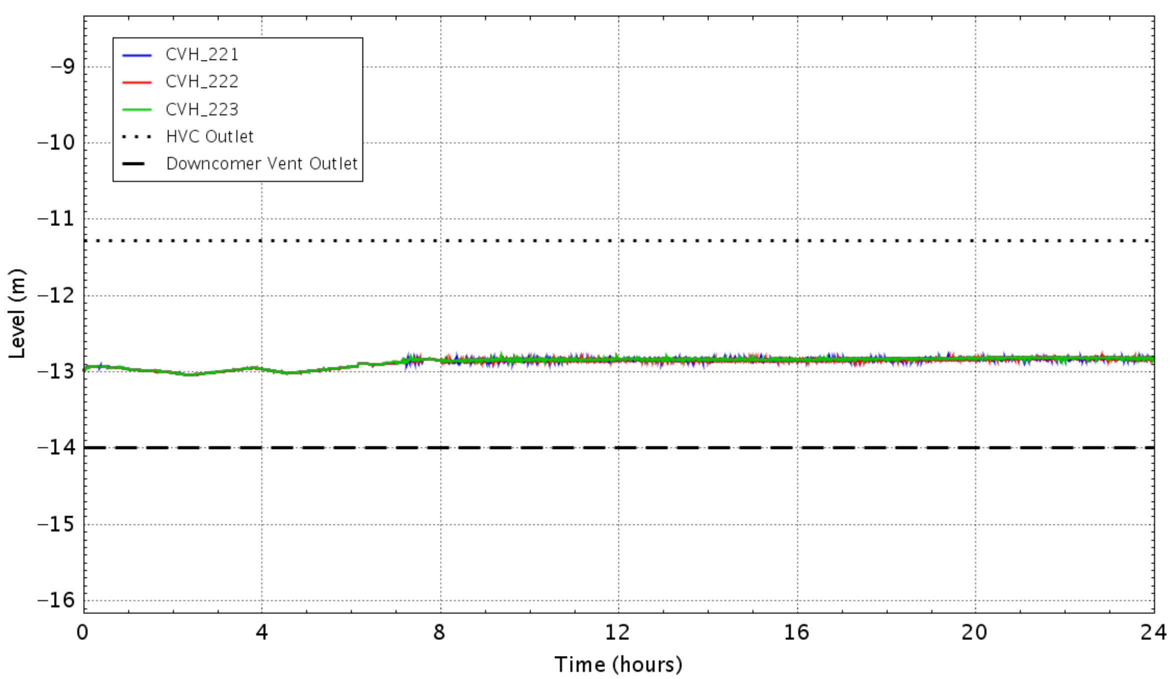

Figure G – 259 Water level in the wetwell

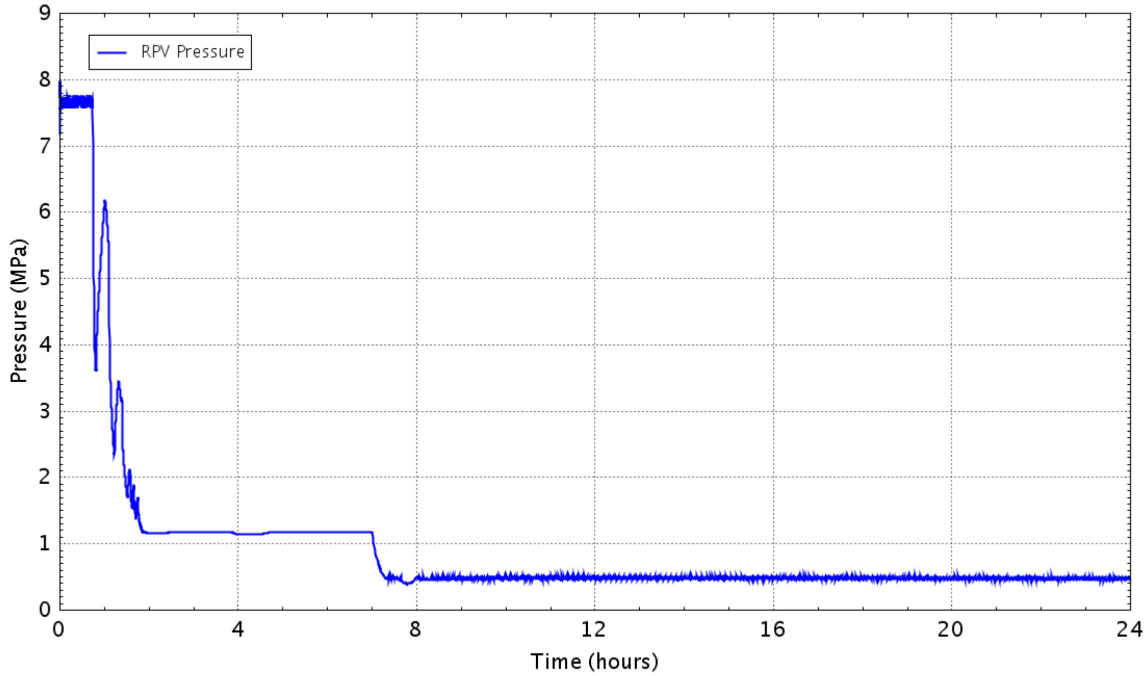

Figure G – 260 Pressure in the RPV

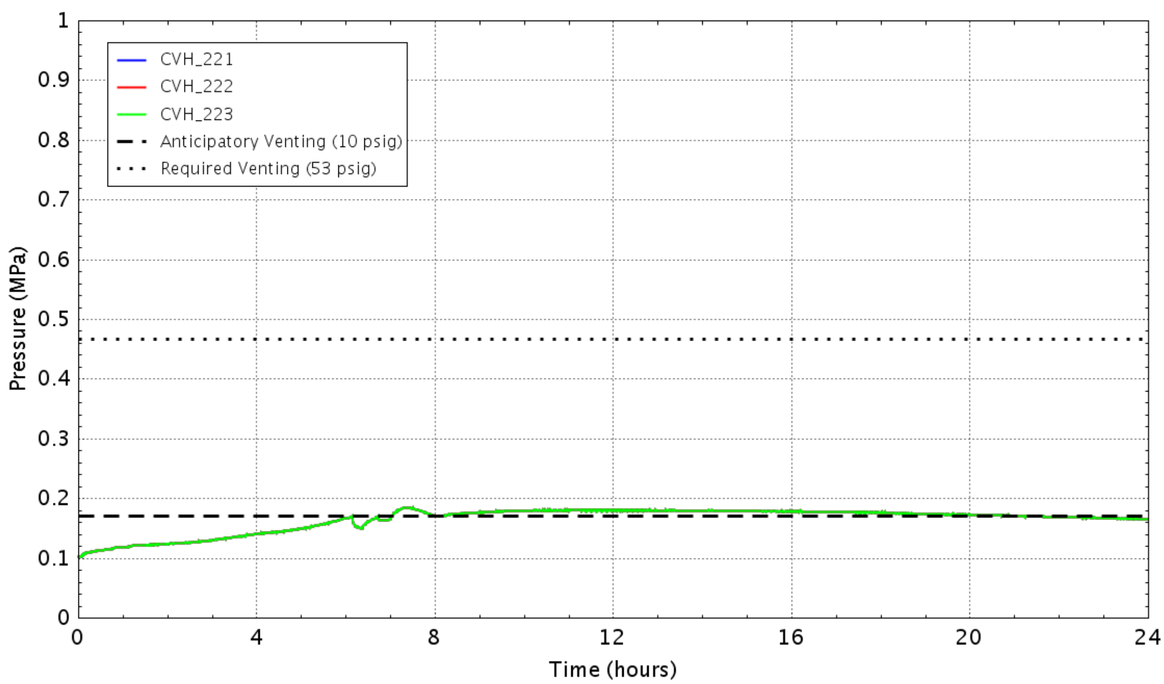

Figure G – 261 Pressure in the wetwell

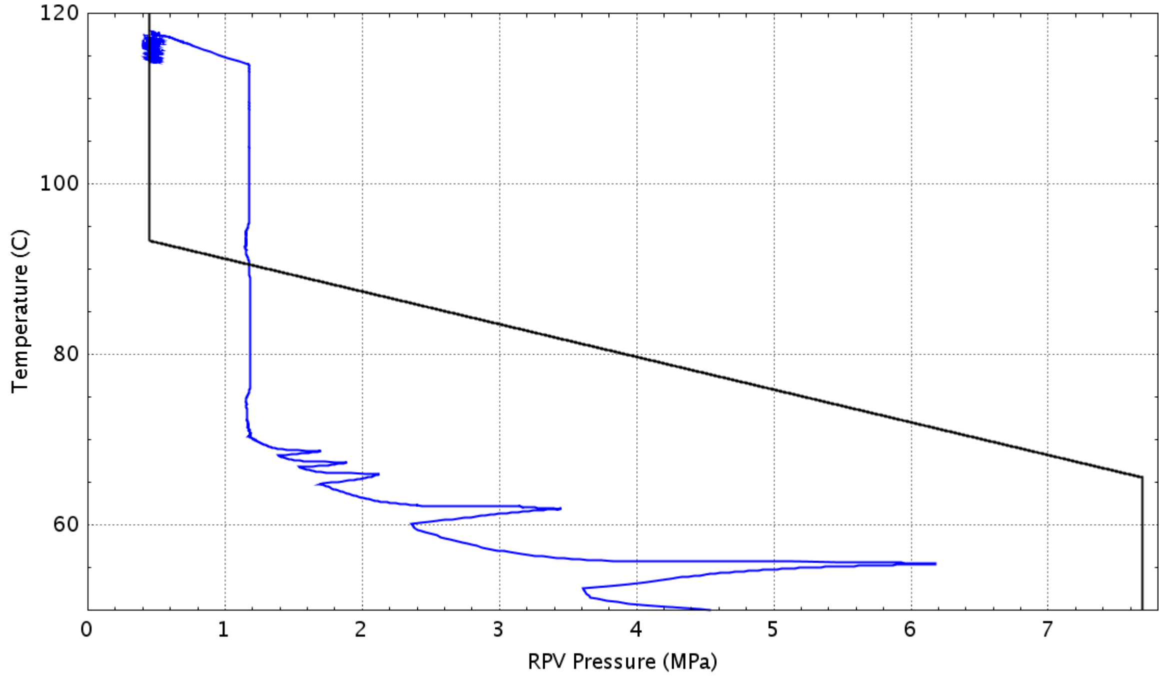

Figure G – 262 Plant status relative to the HCL curve (Graph 4 of the EOPs)

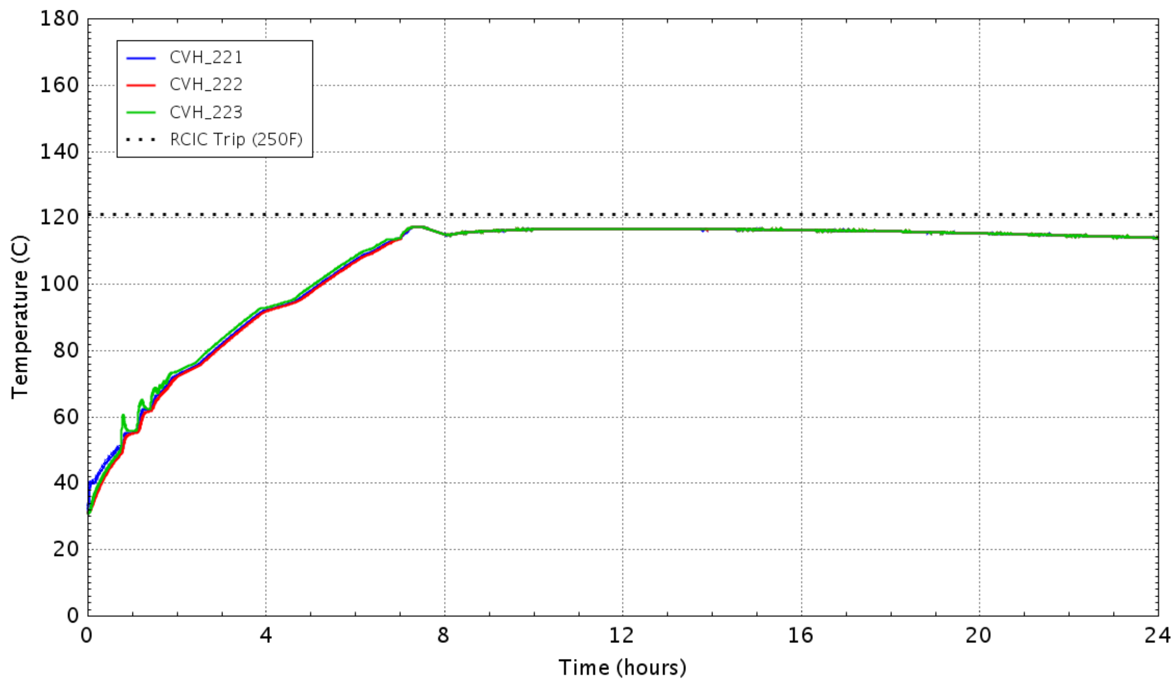

Figure G – 263 Water temperature in the wetwell

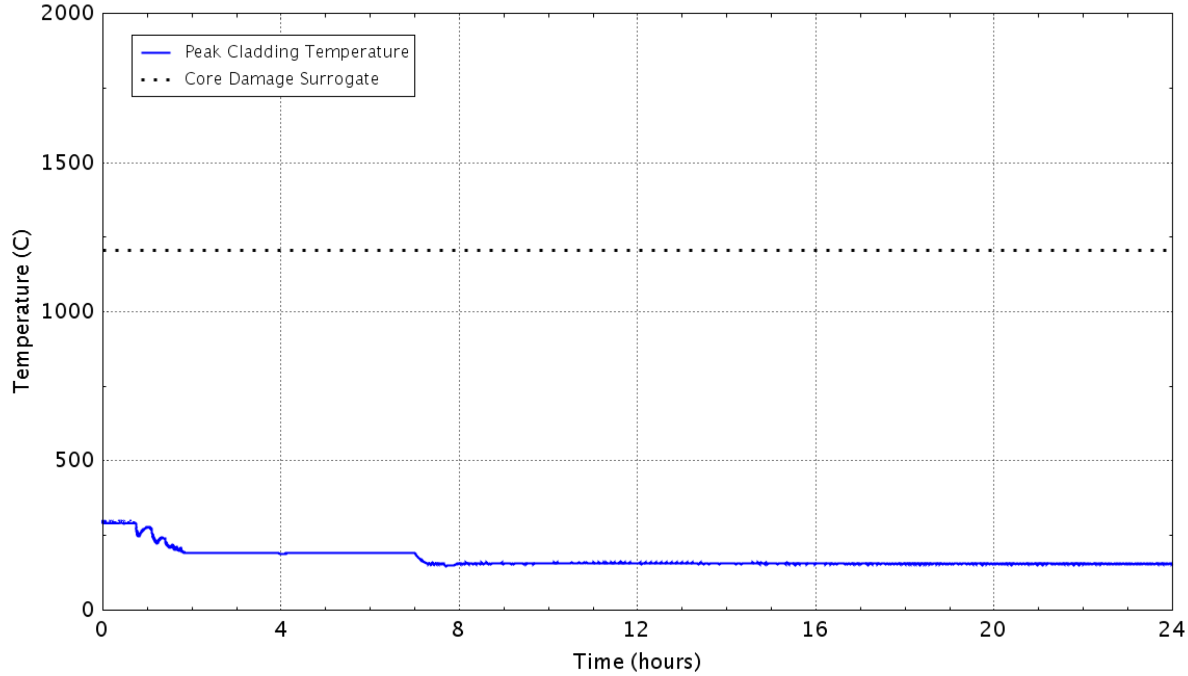

Figure G – 264 Peak temperature of the fuel cladding as a function of time

G.3.5 Case 15a: Sensitivity to LOMFW-25 Case 15 with RCIC Injection from the Wetwell

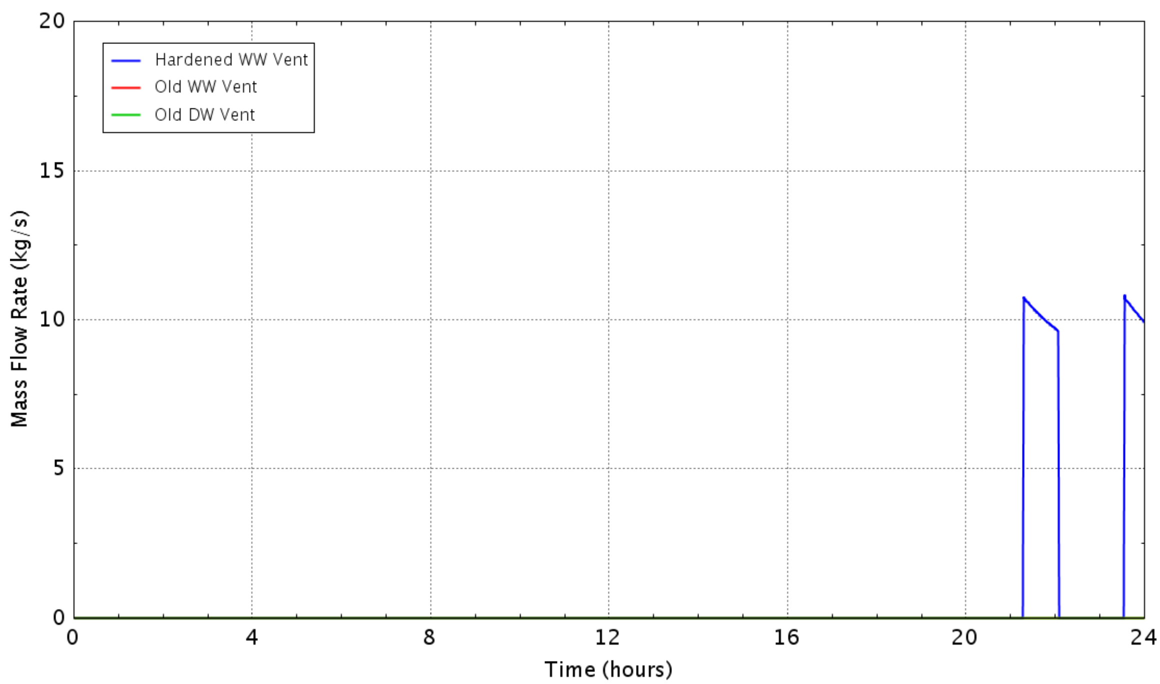

Figure G – 265 Flow rate of the containment vents

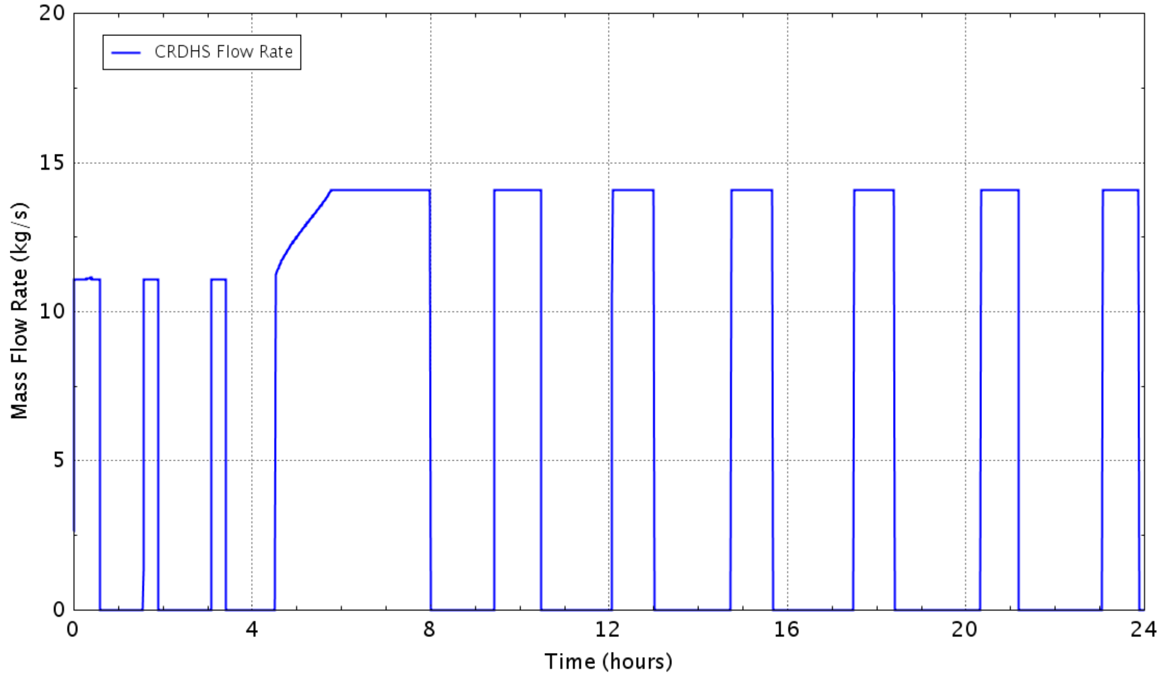

Figure G – 266 Flow rate of the control rod drive hydraulic system

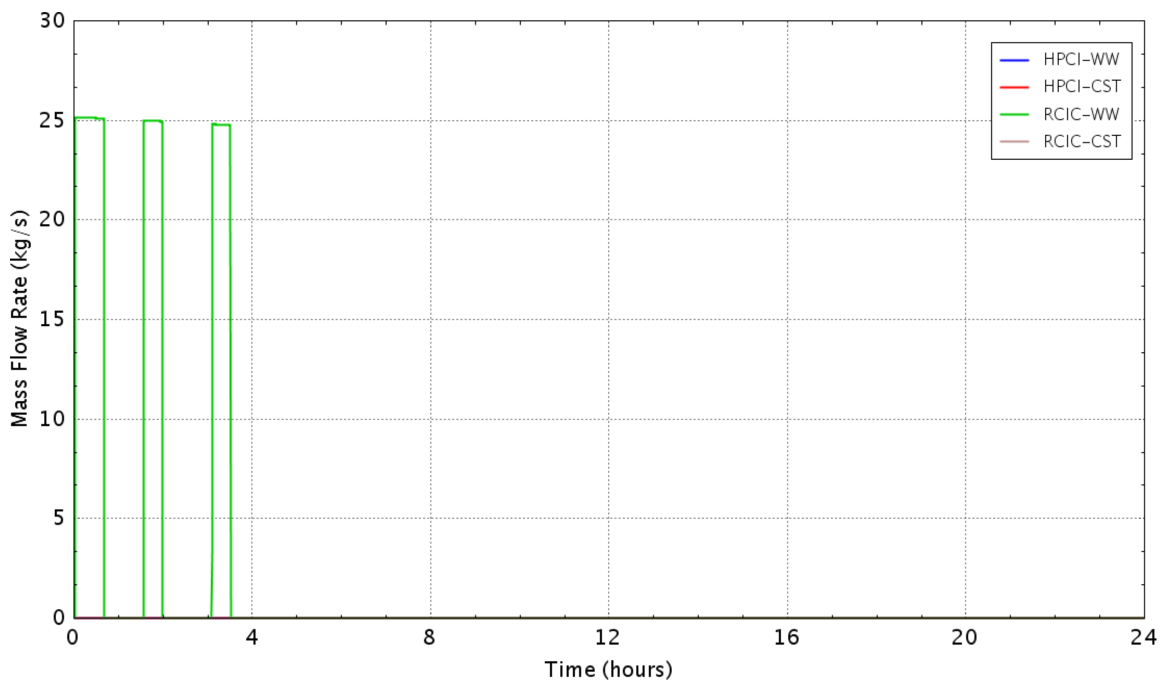

Figure G – 267 Flow rate of the HPCI/RCIC pumps

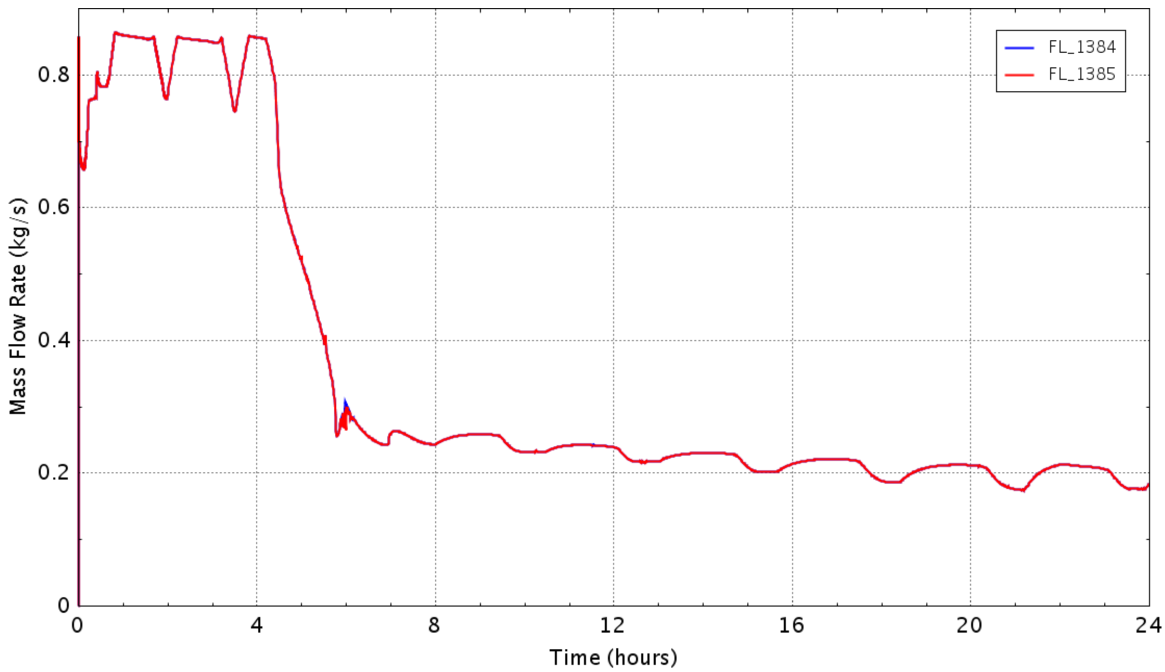

Figure G – 268 Flow rate of the recirculating pump seal leakage

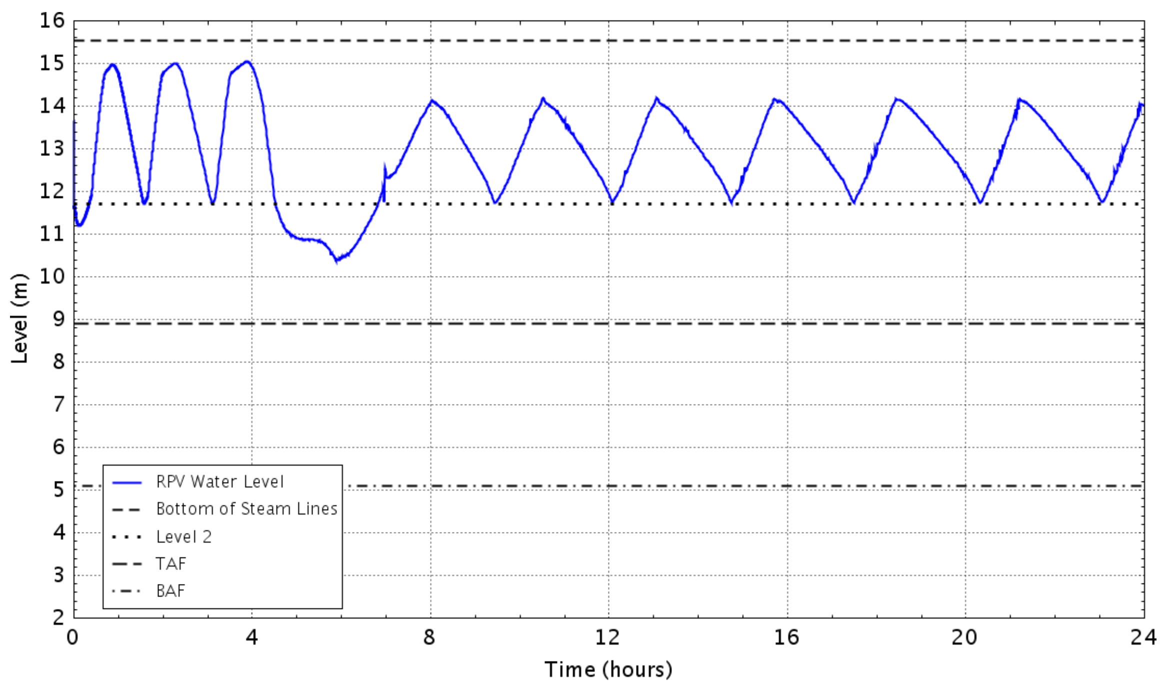

Figure G – 269 RPV Downcomer water level

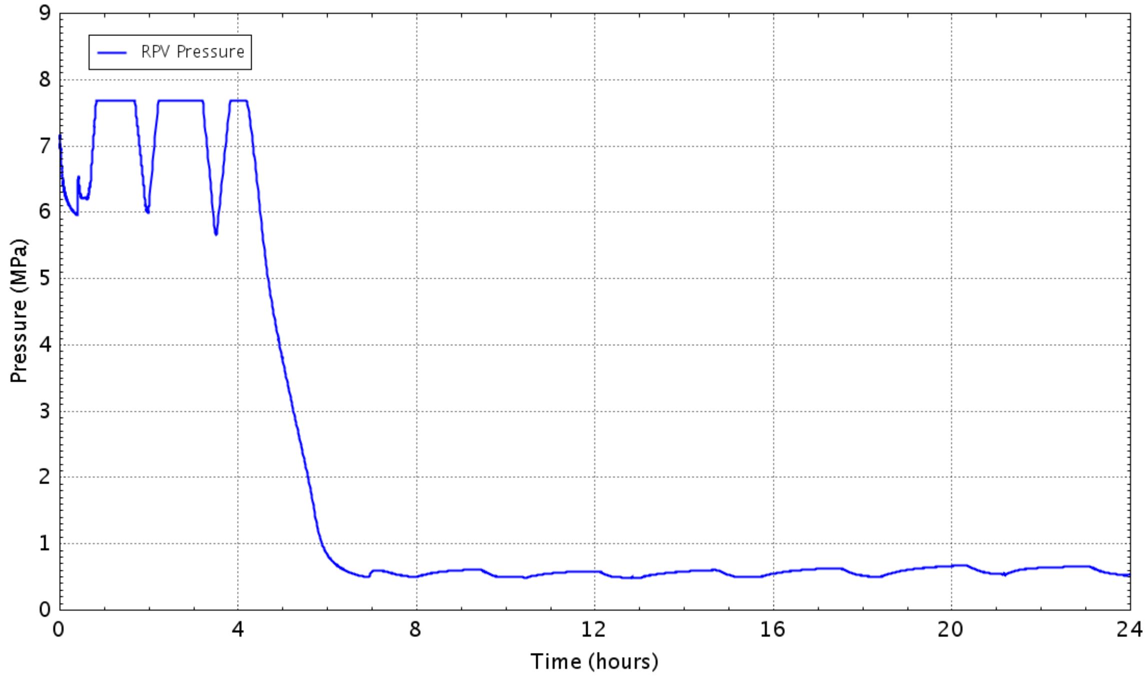

Figure G – 270 Pressure in the RPV

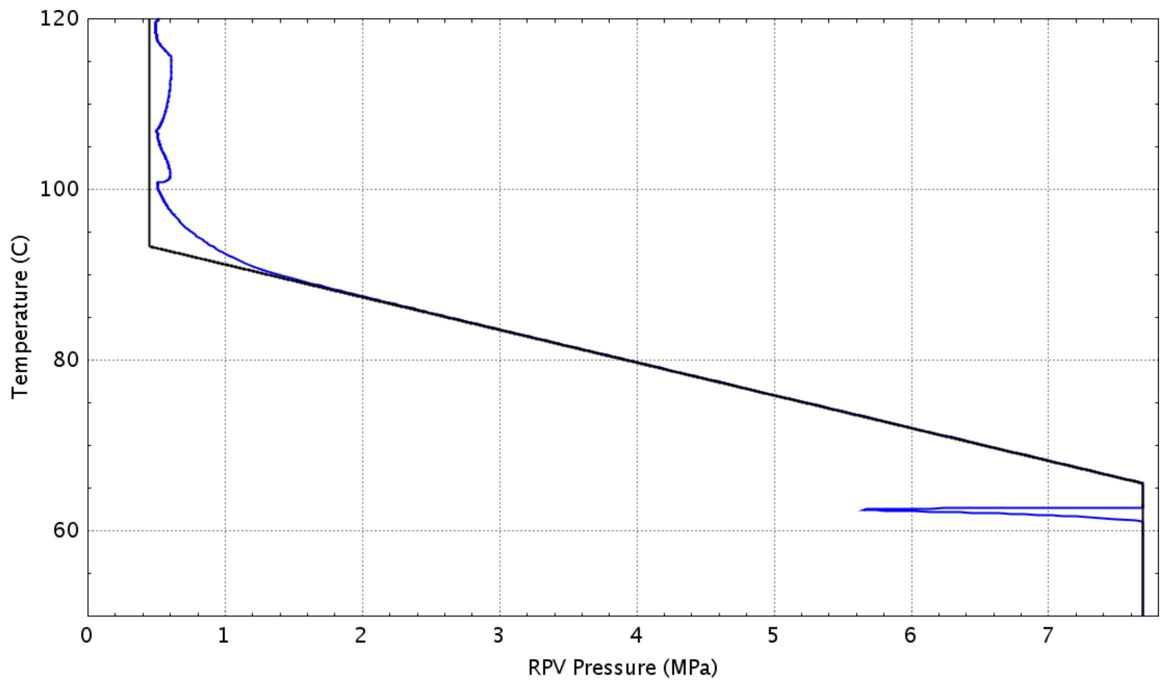

Figure G – 271 Plant status relative to the HCL curve (Graph 4 of the EOPs)

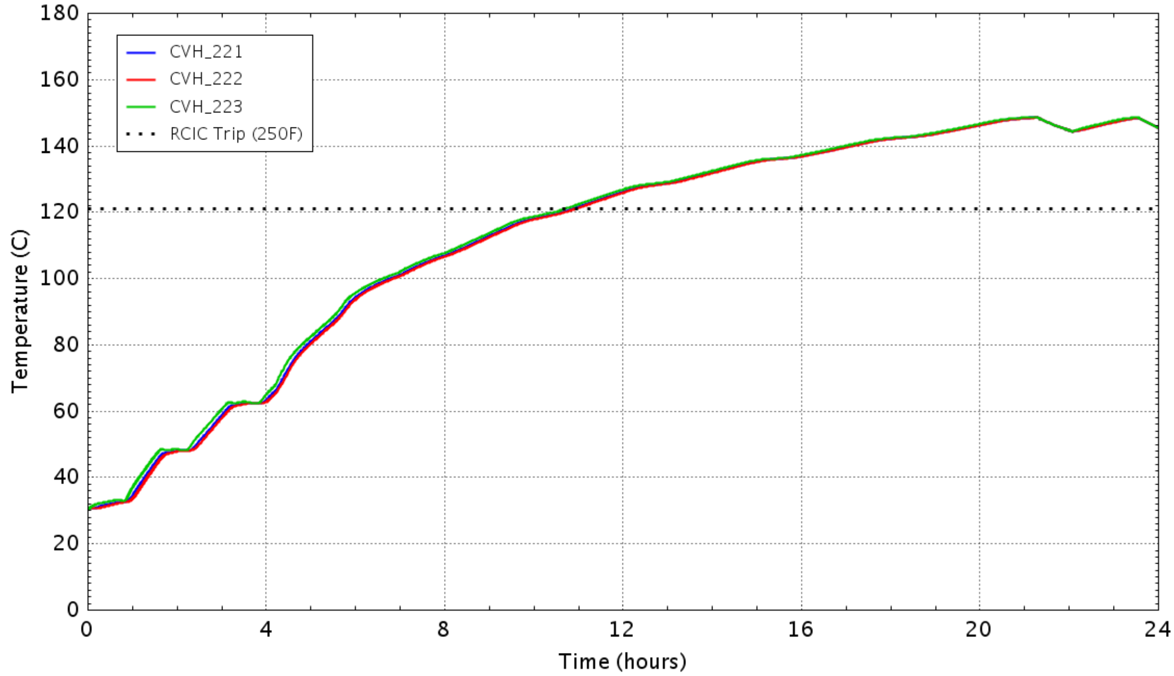

Figure G – 272 Water temperature in the wetwell

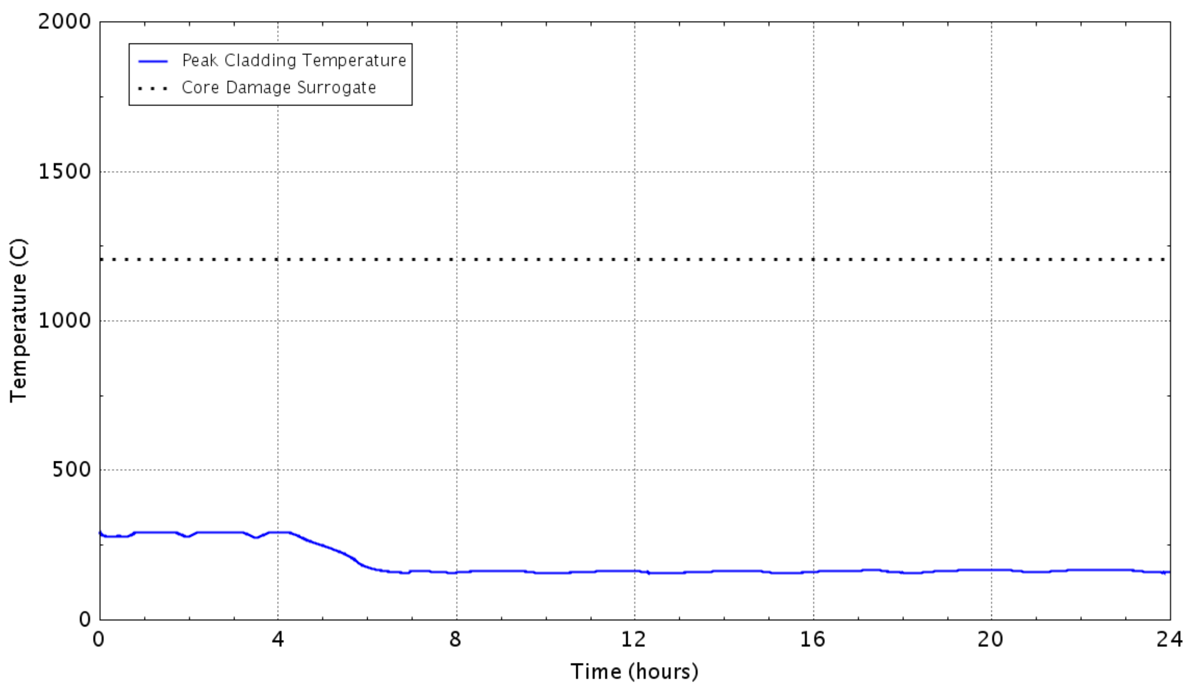

Figure G – 273 Peak temperature of the fuel cladding as a function of time

G.3.6 Case 15b: Sensitivity to LOMFW-25 Case 15 with RPV Depressurization Ending at 150psig

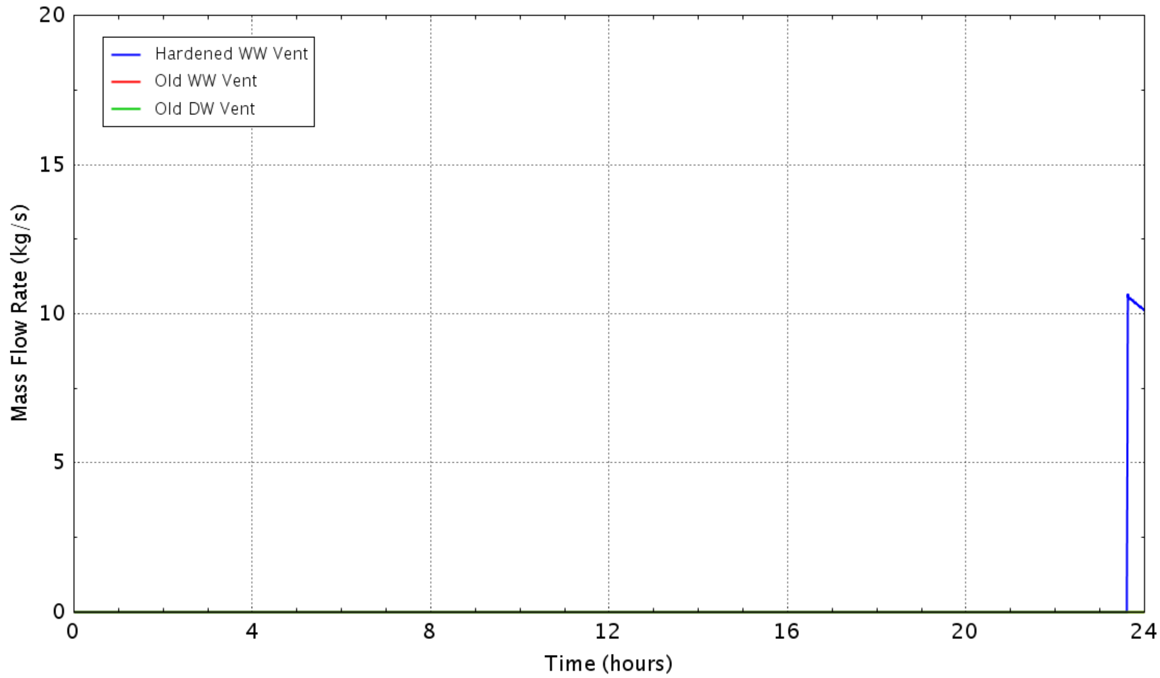

Figure G – 274 Flow rate of the containment vents

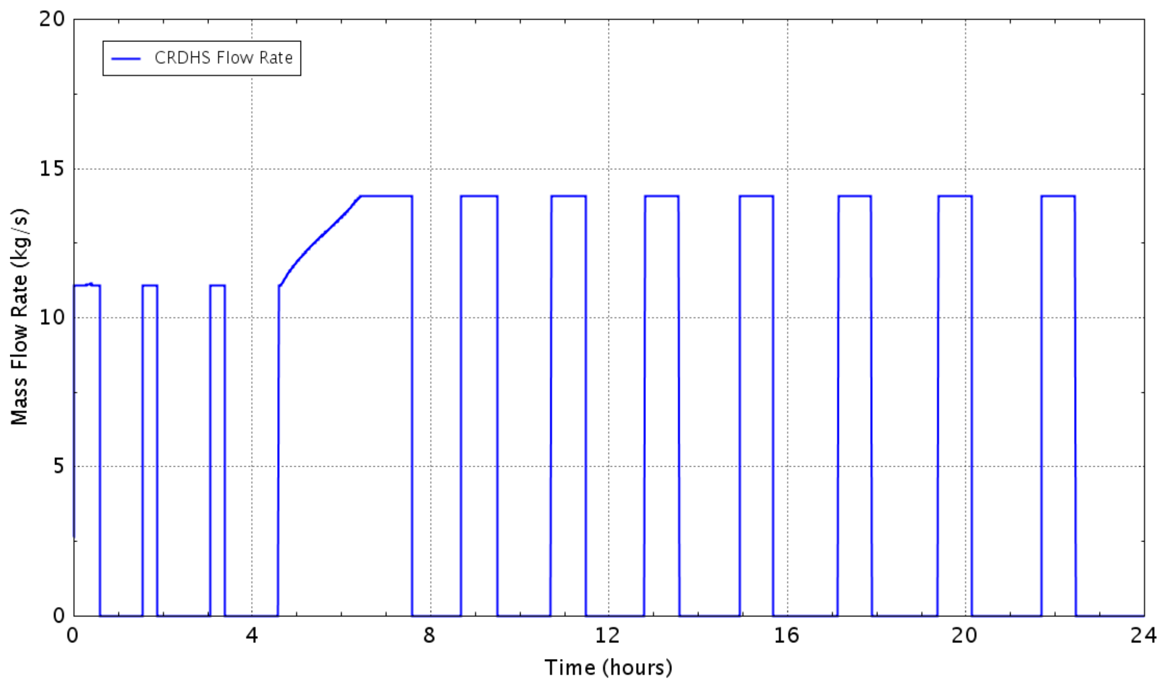

Figure G – 275 Flow rate of the control rod drive hydraulic system

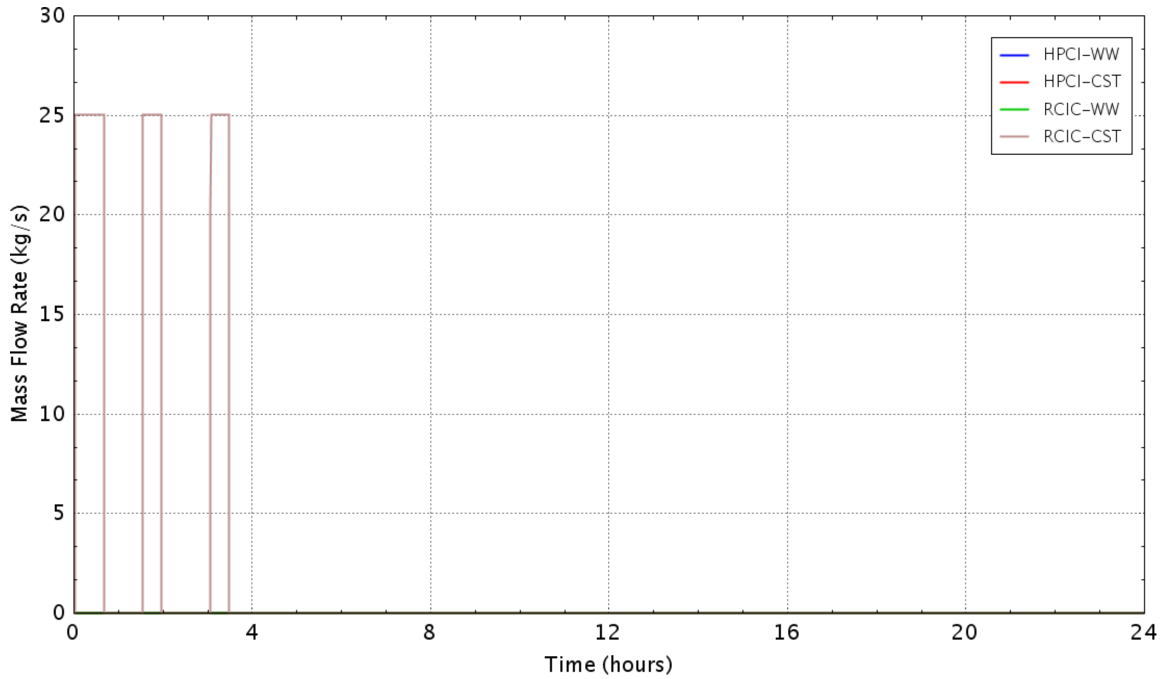

Figure G – 276 Flow rate of the HPCI/RCIC pumps

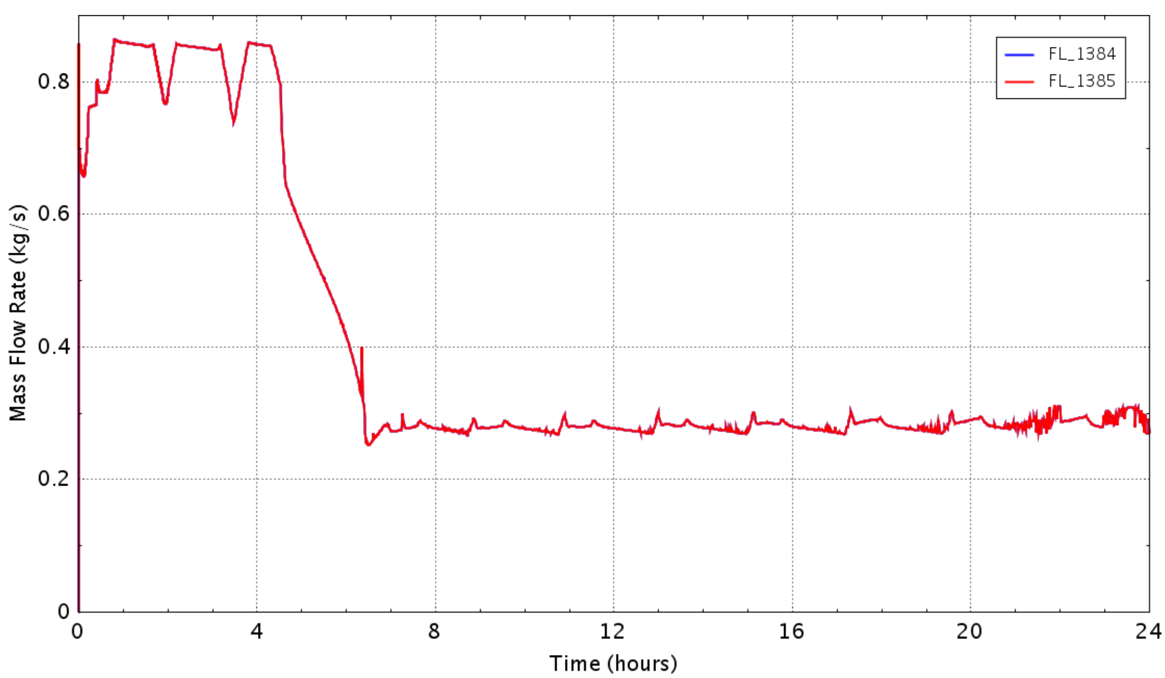

Figure G – 277 Flow rate of the recirculating pump seal leakage

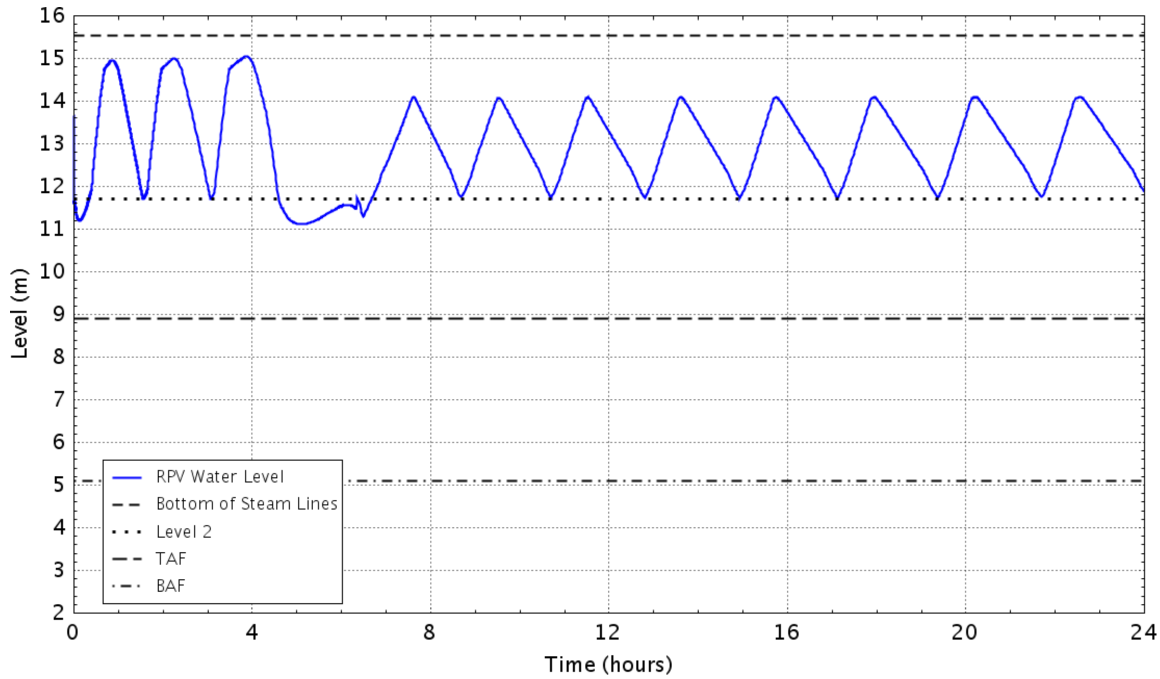

Figure G – 278 RPV Downcomer water level

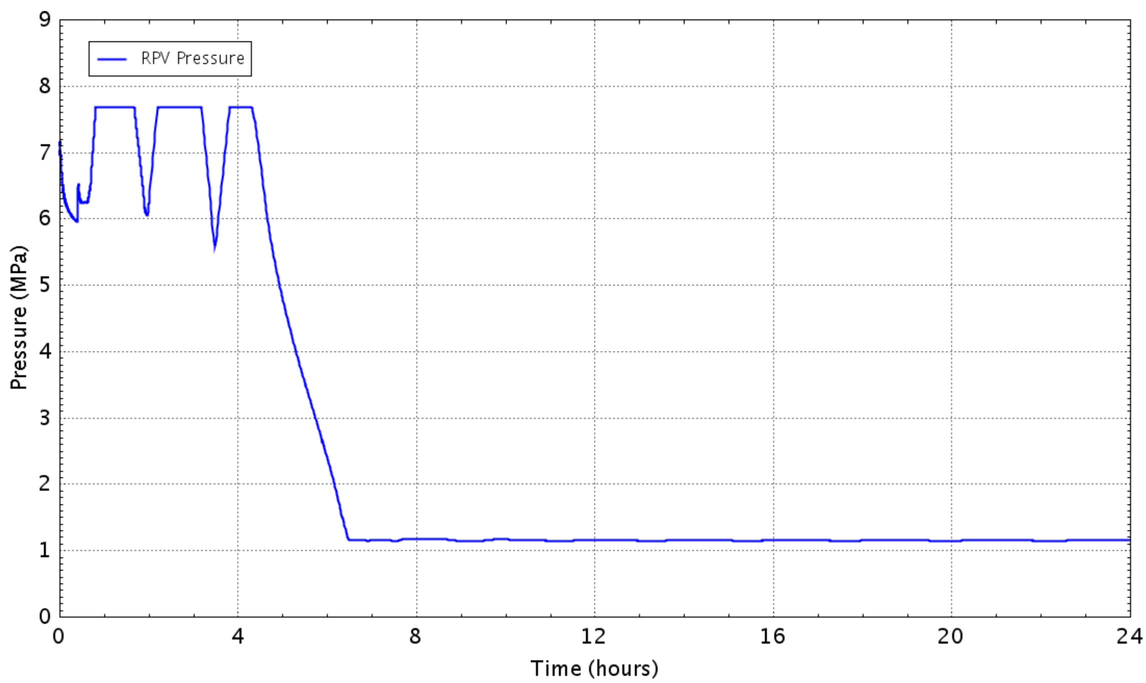

Figure G – 279 Pressure in the RPV

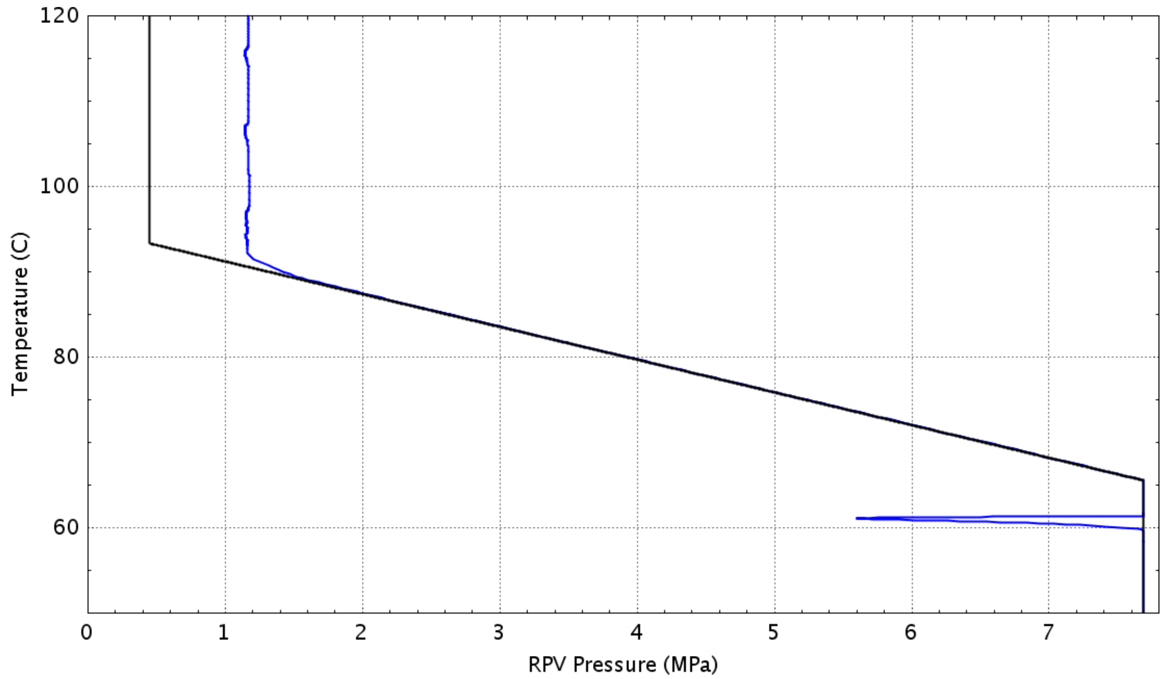

Figure G – 280 Plant status relative to the HCL curve (Graph 4 of the EOPs)

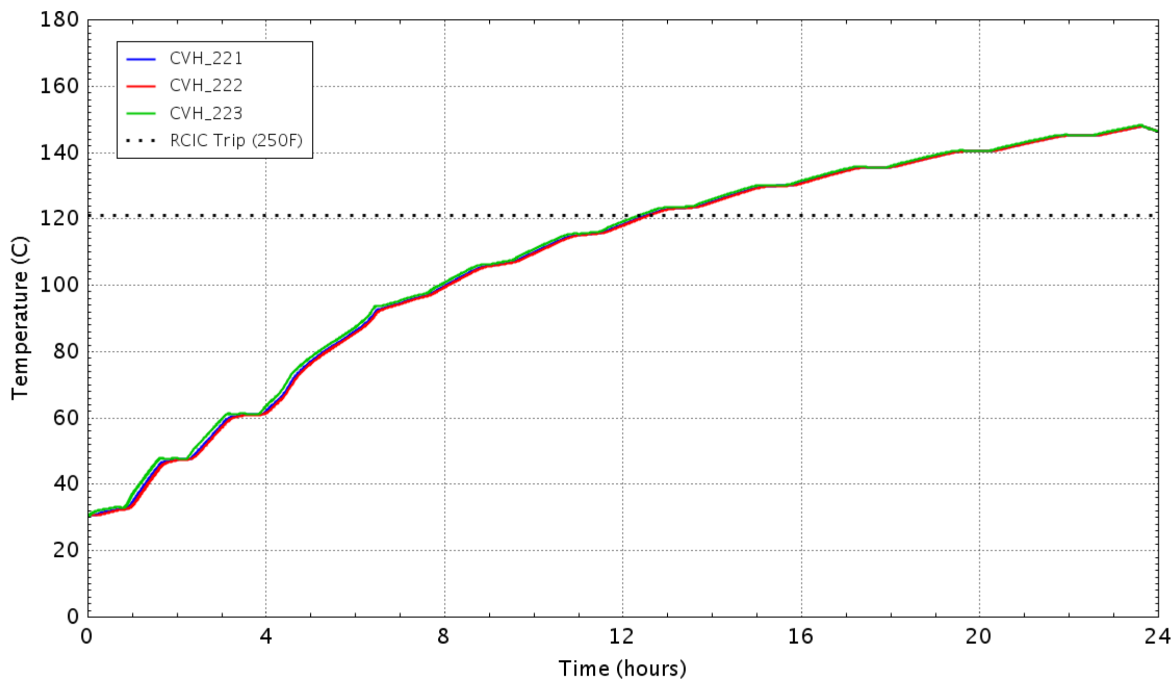

Figure G – 281 Water temperature in the wetwell

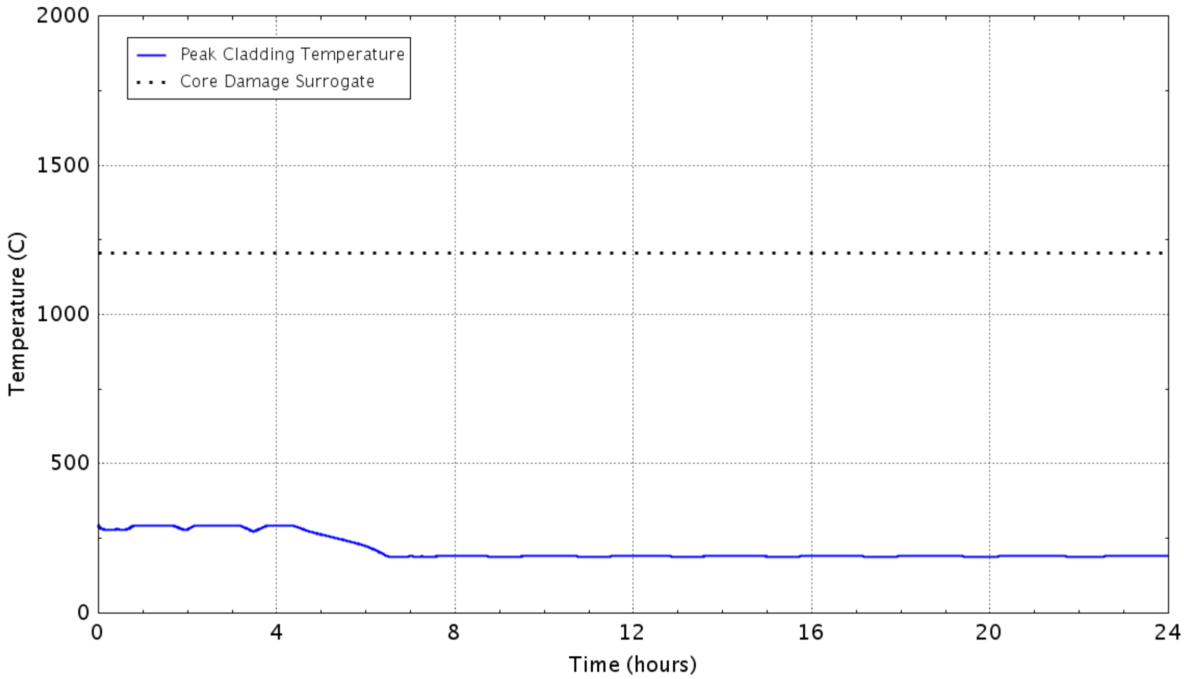

Figure G – 282 Peak temperature of the fuel cladding as a function of time

G.3.7 Case 15c: Sensitivity to LOMFW-25 Case 15 with RPV Emergency Depressurization when the HCL Curve is Reached

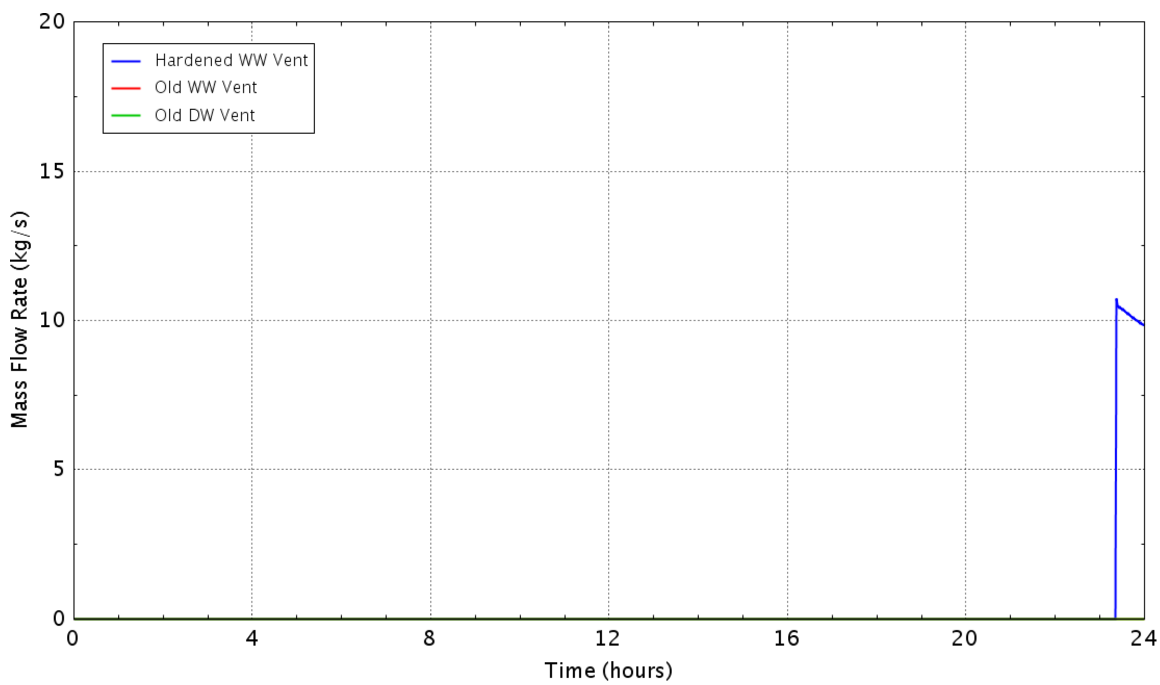

Figure G – 283 Flow rate of the containment vents

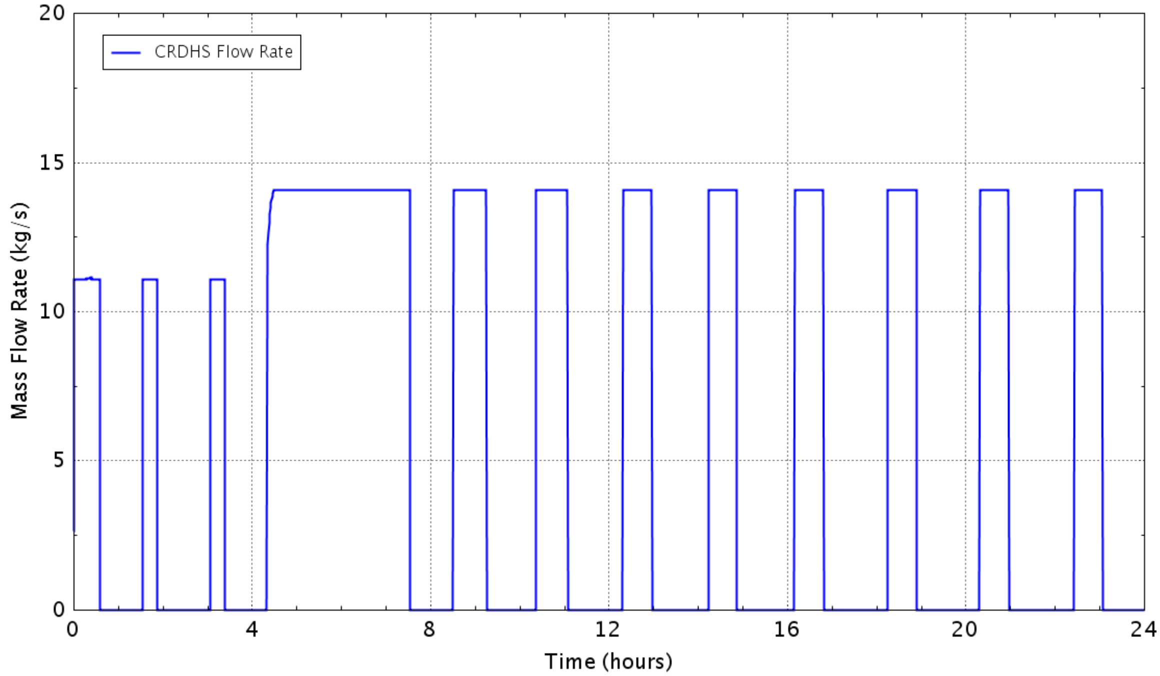

Figure G – 284 Flow rate of the control rod drive hydraulic system

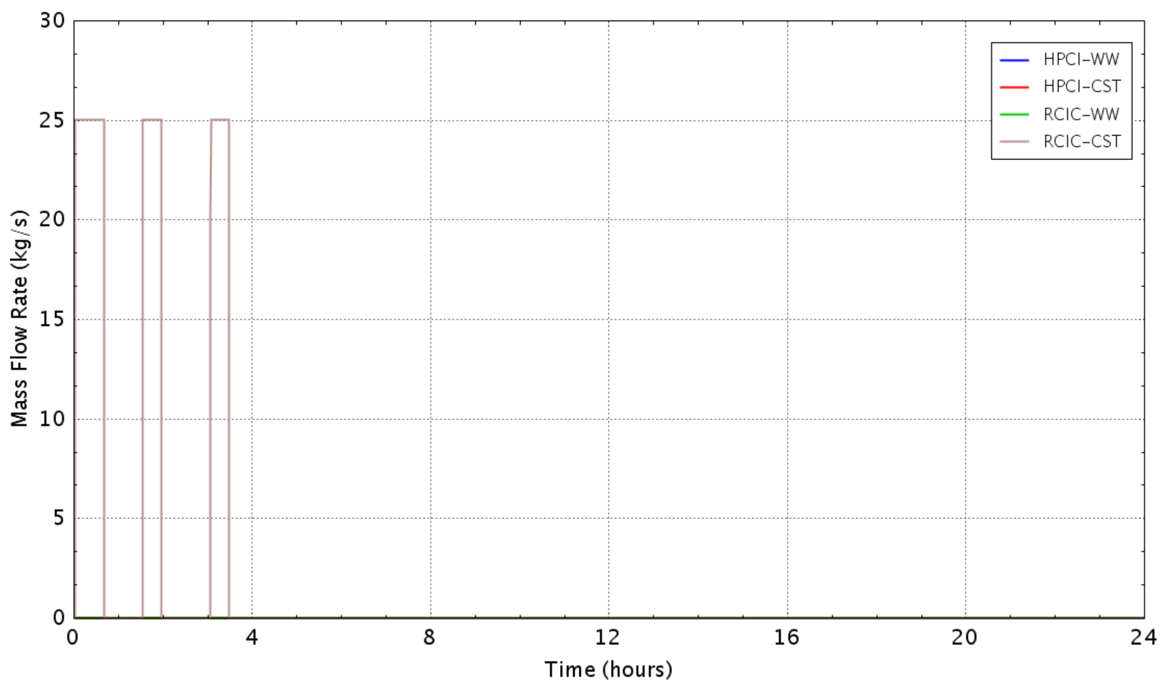

Figure G – 285 Flow rate of the HPCI/RCIC pumps

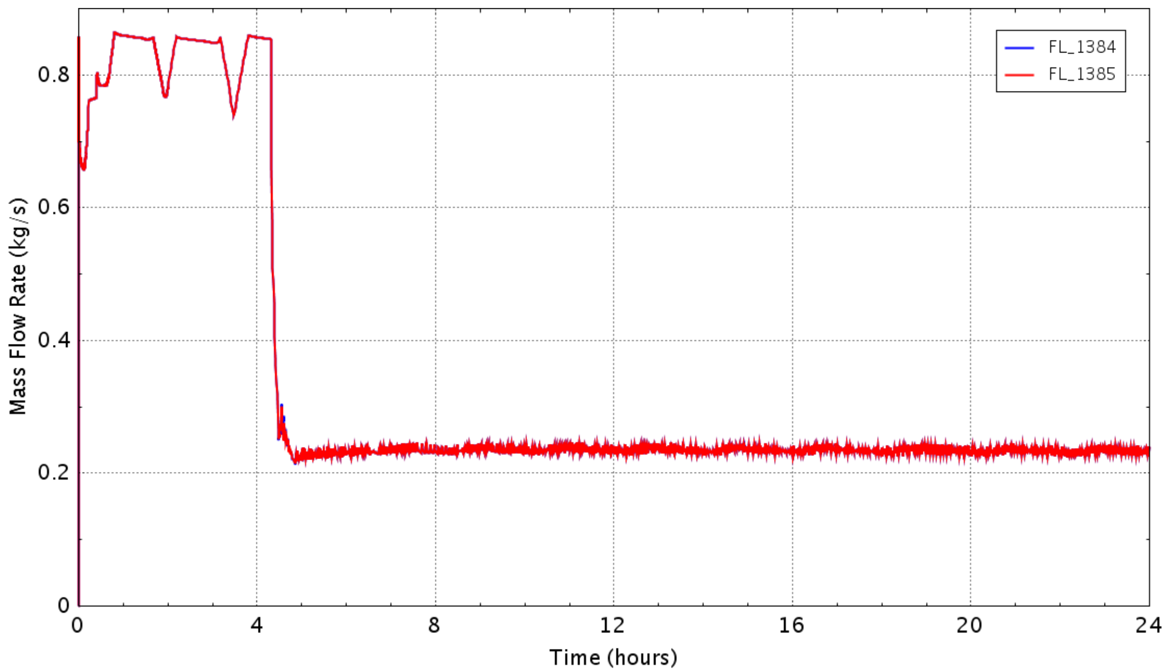

Figure G – 286 Flow rate of the recirculating pump seal leakage

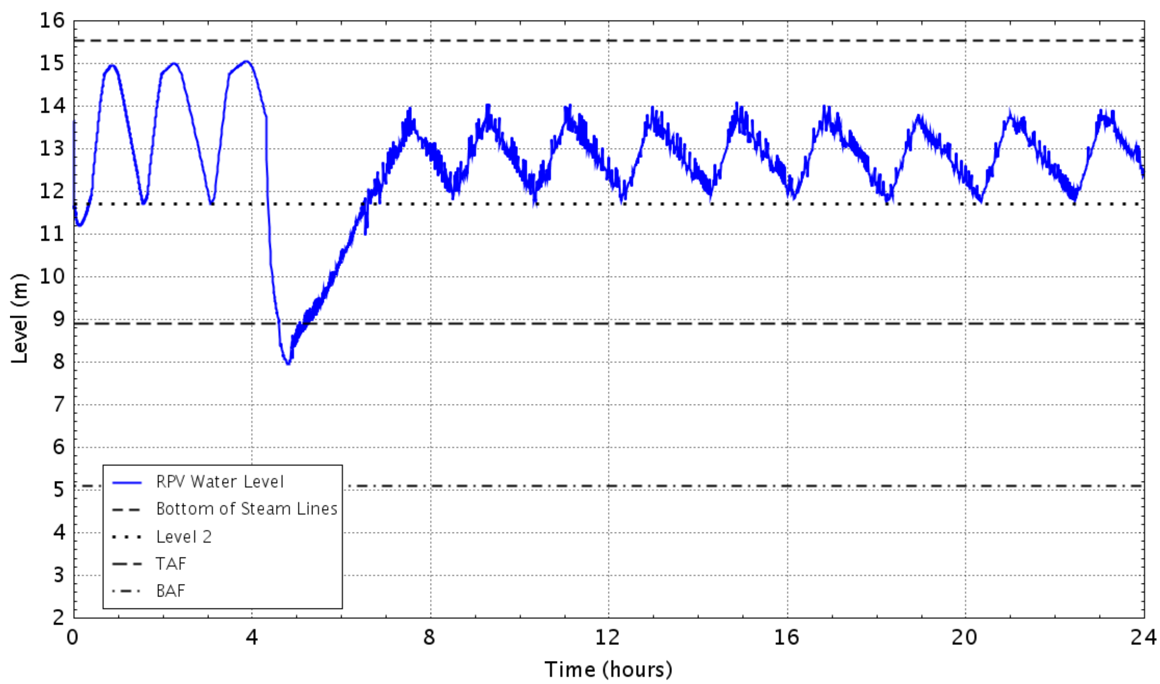

Figure G – 287 RPV Downcomer water level

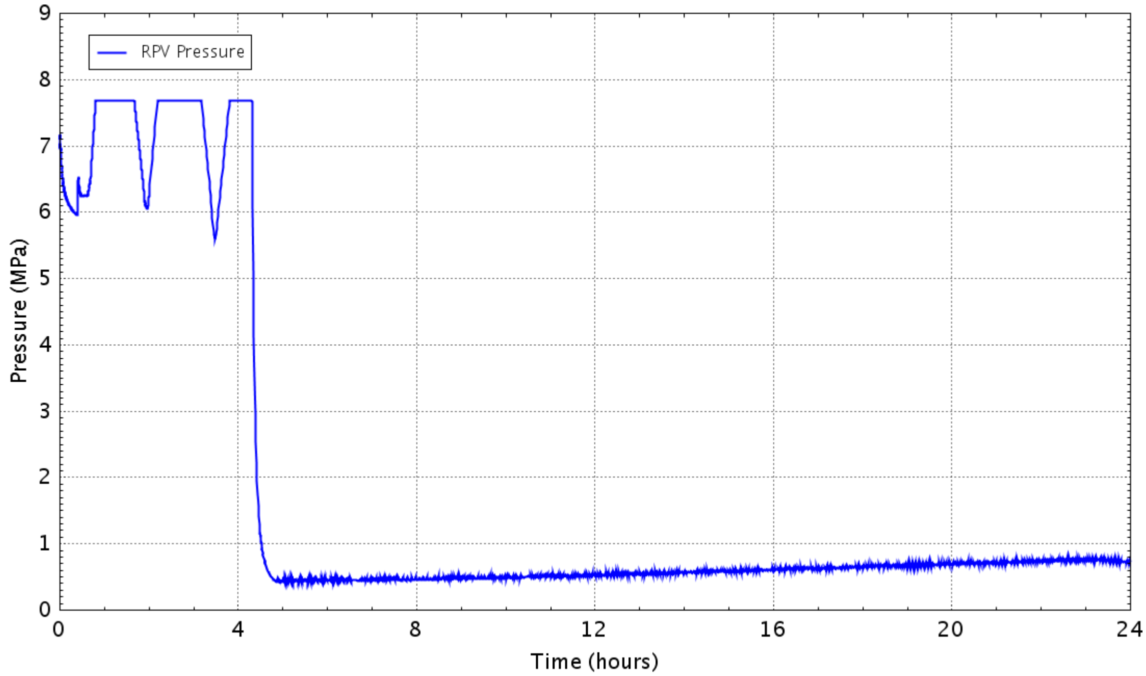

Figure G – 288 Pressure in the RPV

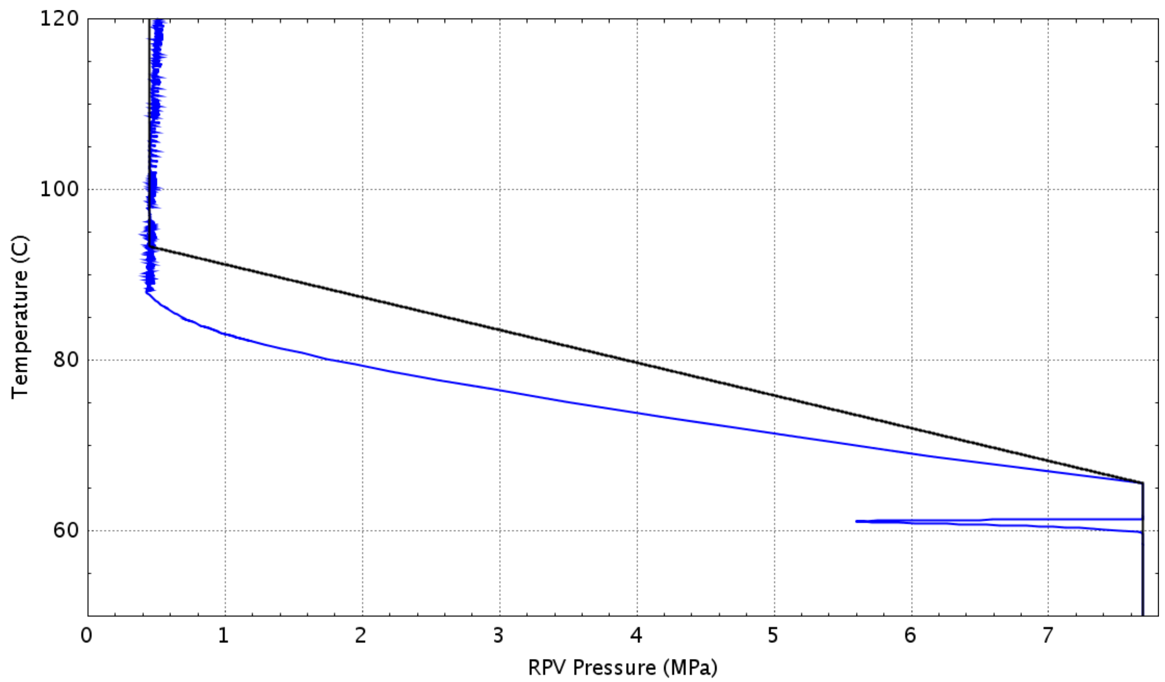

Figure G – 289 Plant status relative to the HCL curve (Graph 4 of the EOPs)

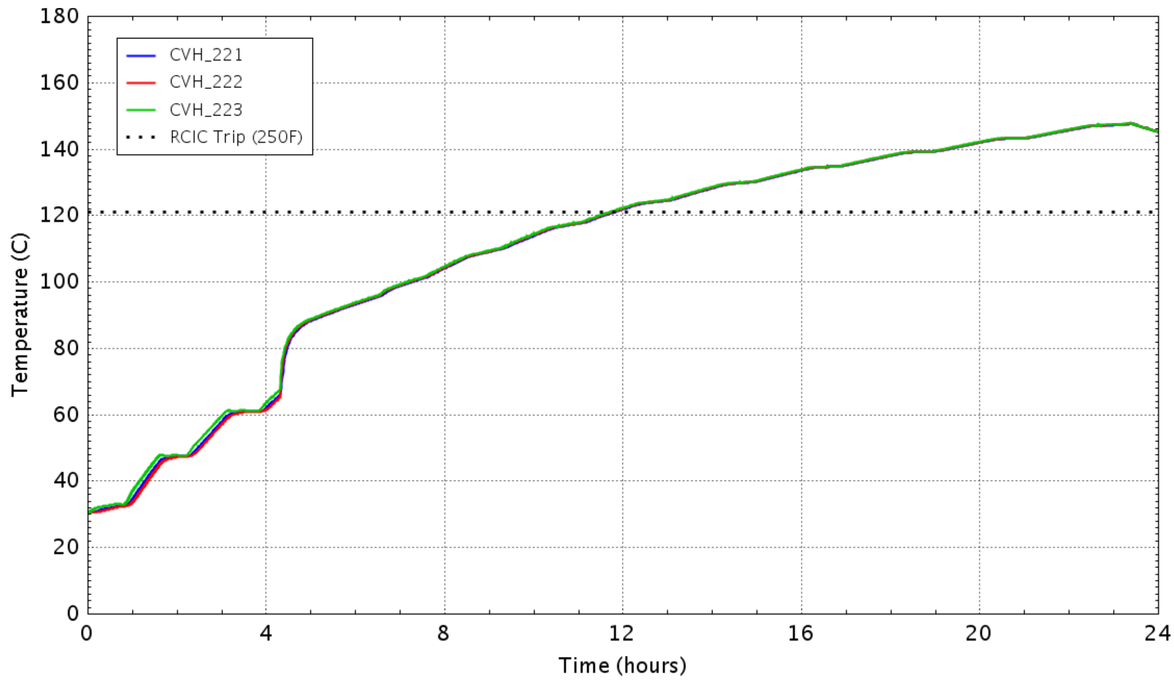

Figure G – 290 Water temperature in the wetwell

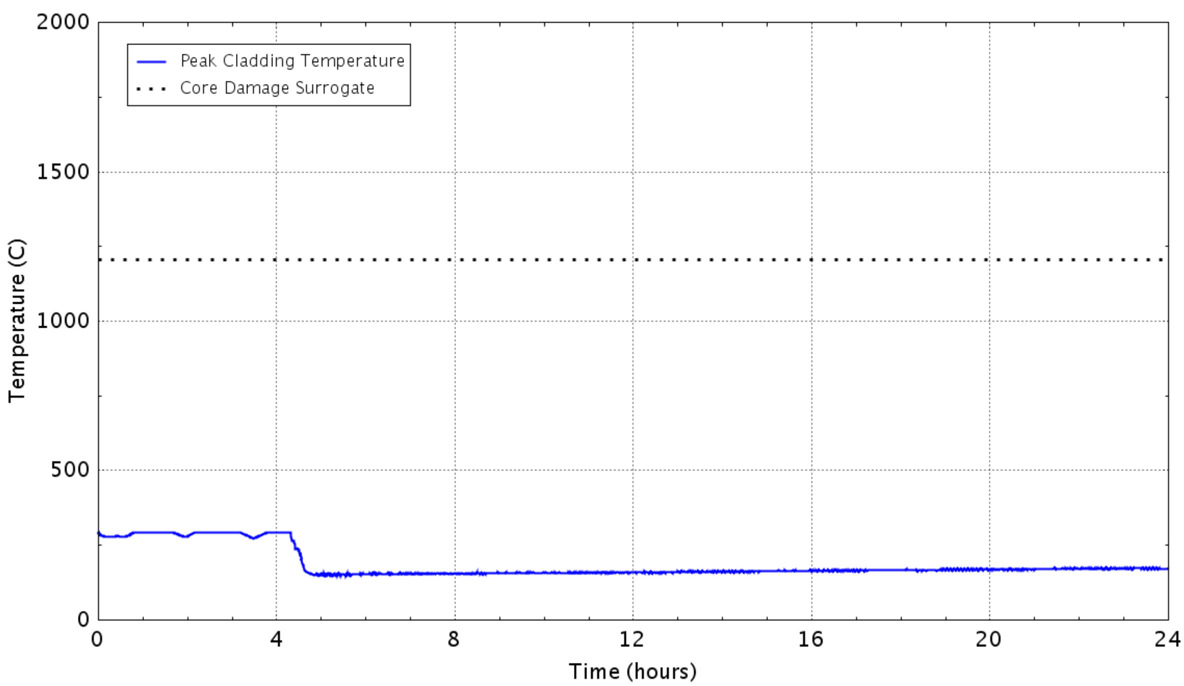

Figure G – 291 Peak temperature of the fuel cladding as a function of time

G.3.8 Case 15d: Sensitivity to LOMFW-25 Case 15 with MSIV Closureat the Start of the Transient

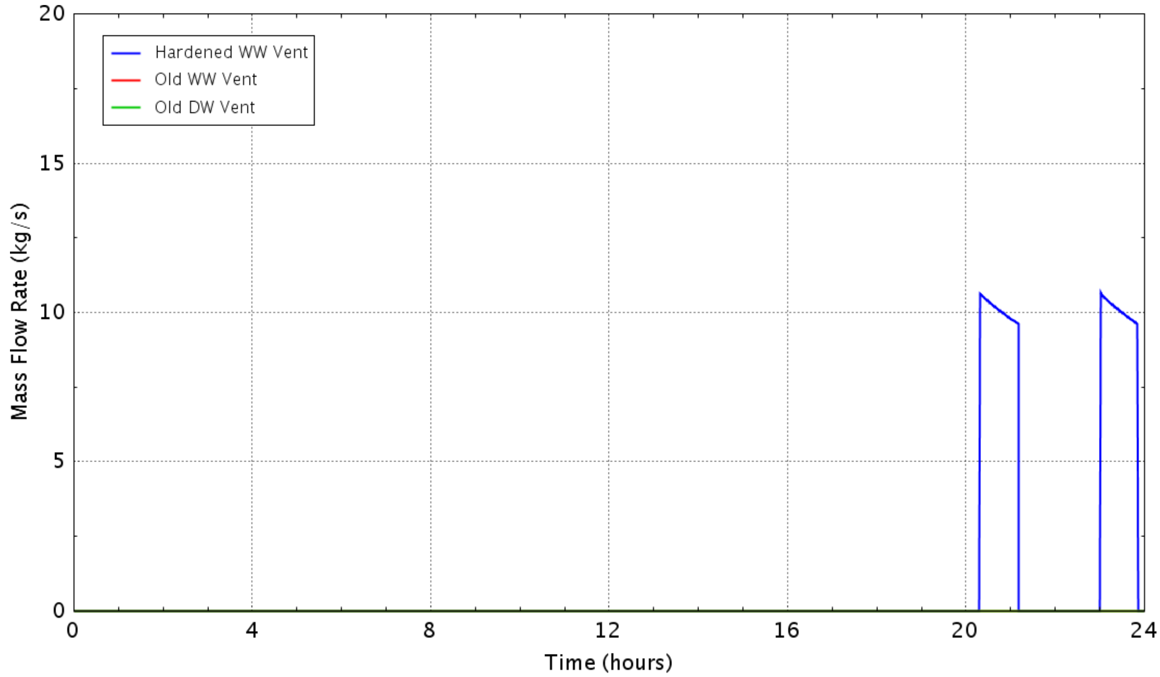

Figure G – 292 Flow rate of the containment vents

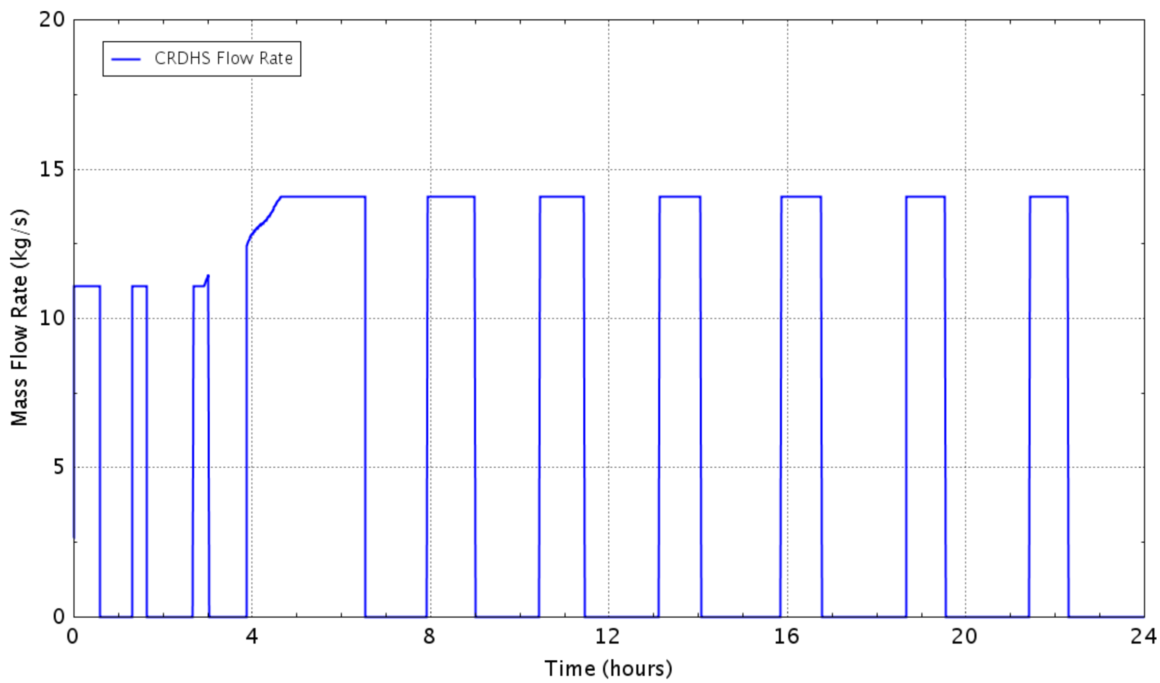

Figure G – 293 Flow rate of the control rod drive hydraulic system

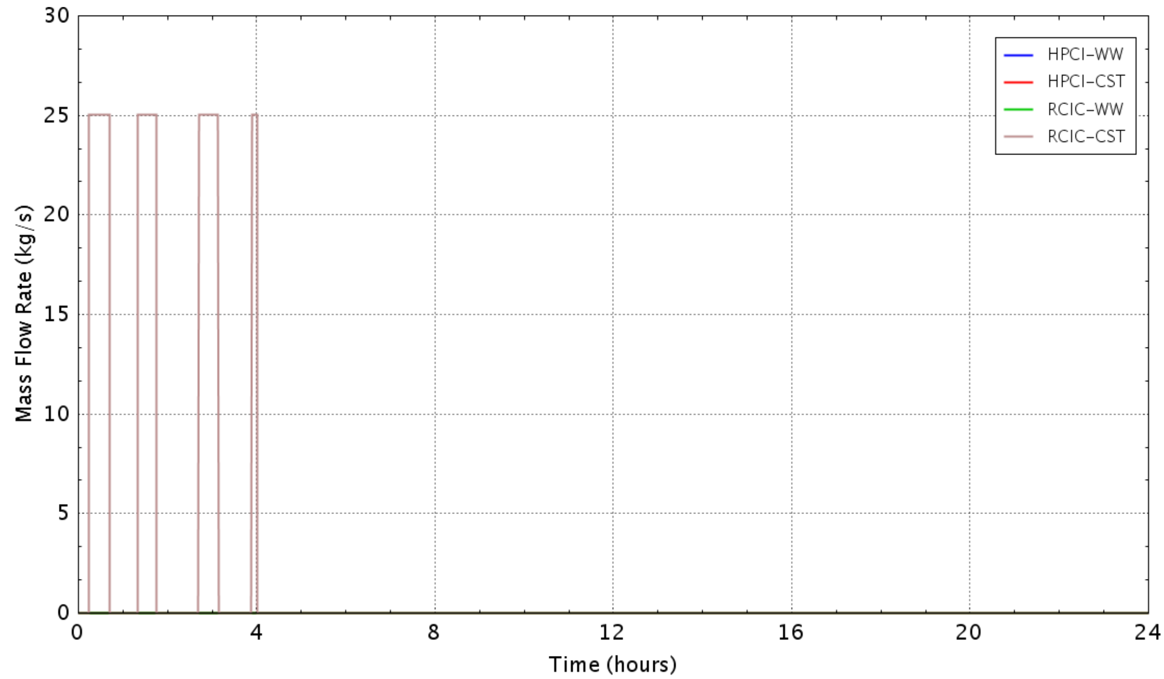

Figure G – 294 Flow rate of the HPCI/RCIC pumps

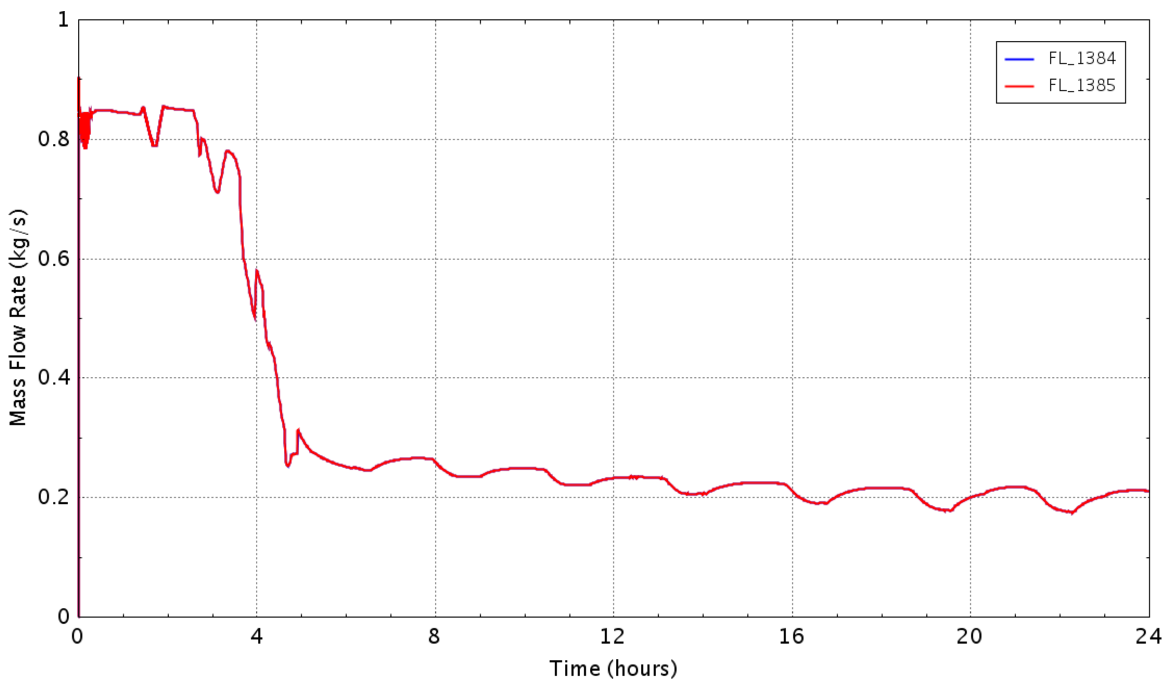

Figure G – 295 Flow rate of the recirculating pump seal leakage

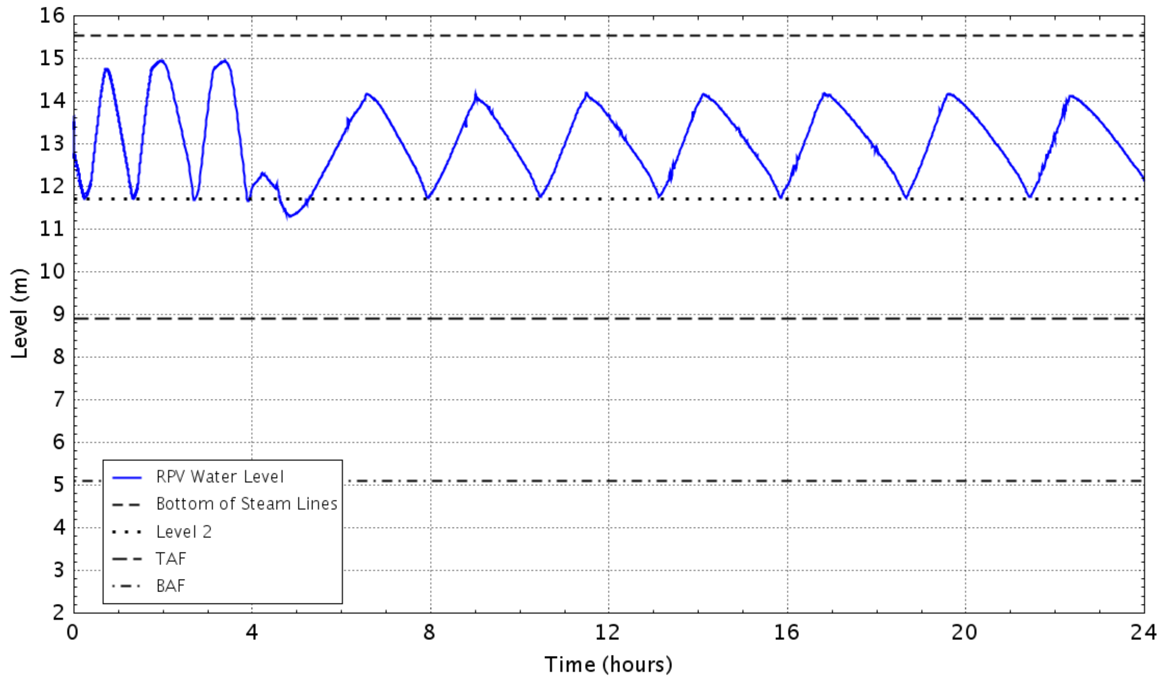

Figure G – 296 RPV Downcomer water level

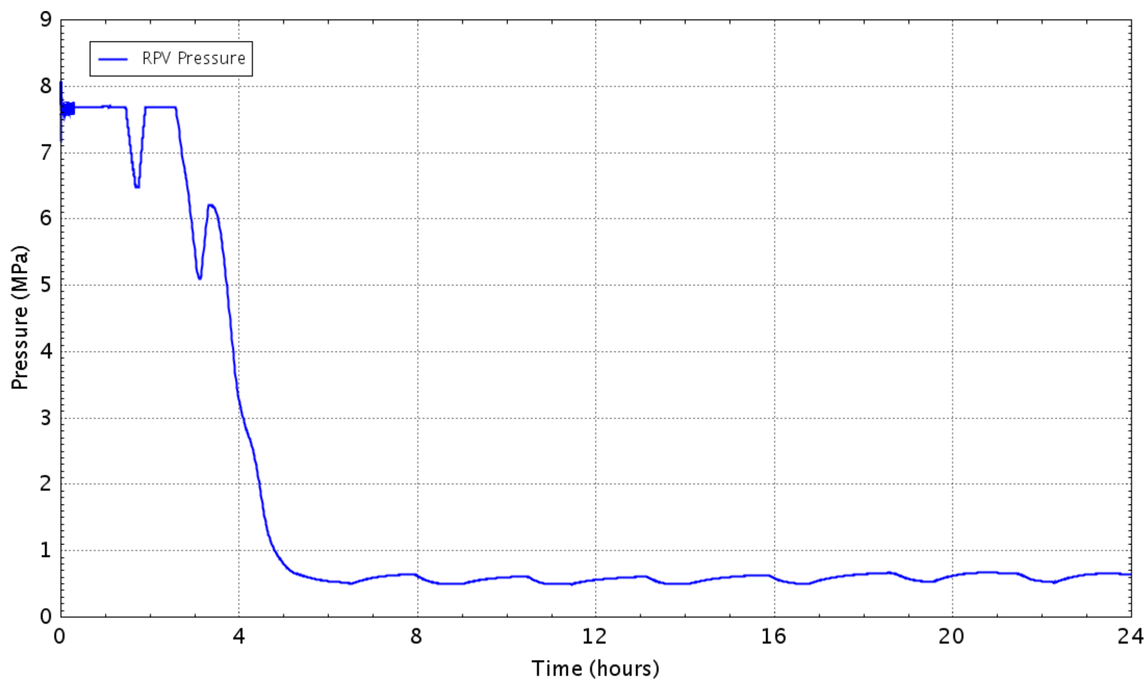

Figure G – 297 Pressure in the RPV

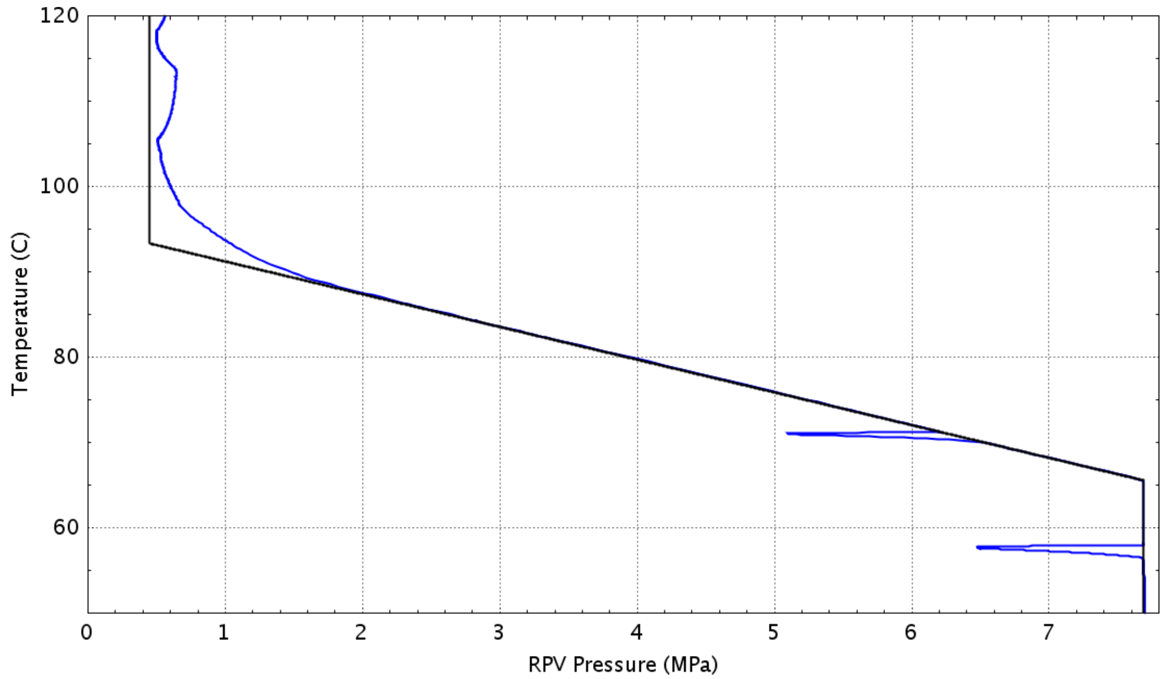

Figure G – 298 Plant status relative to the HCL curve (Graph 4 of the EOPs)

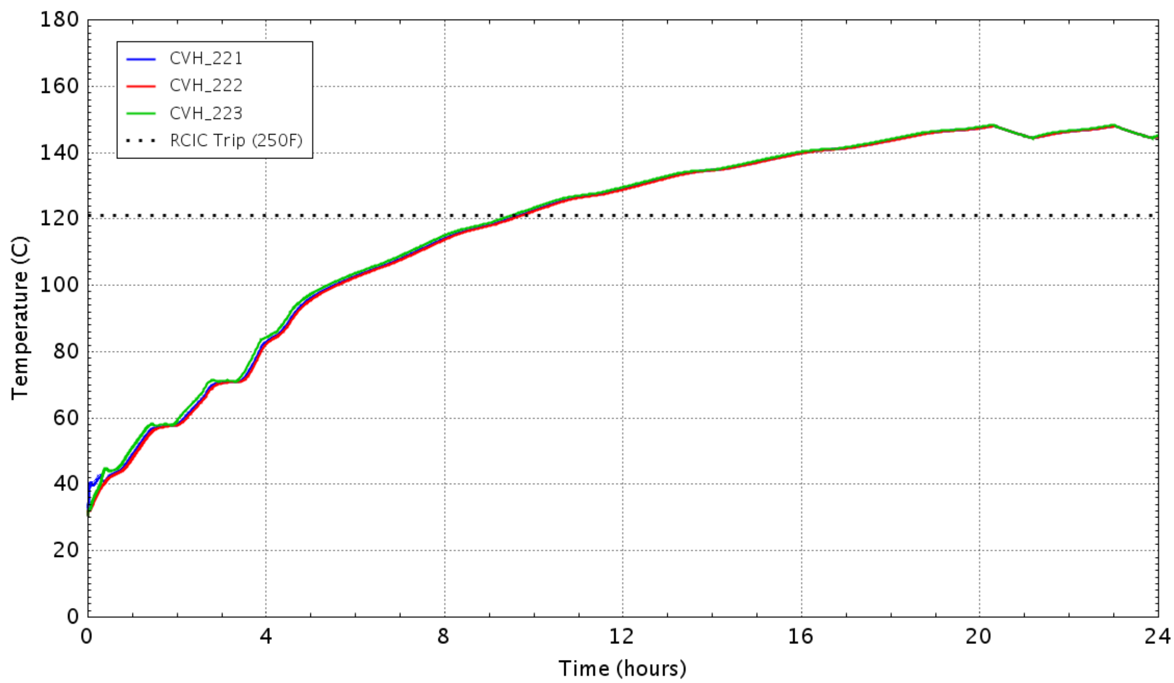

Figure G – 299 Water temperature in the wetwell

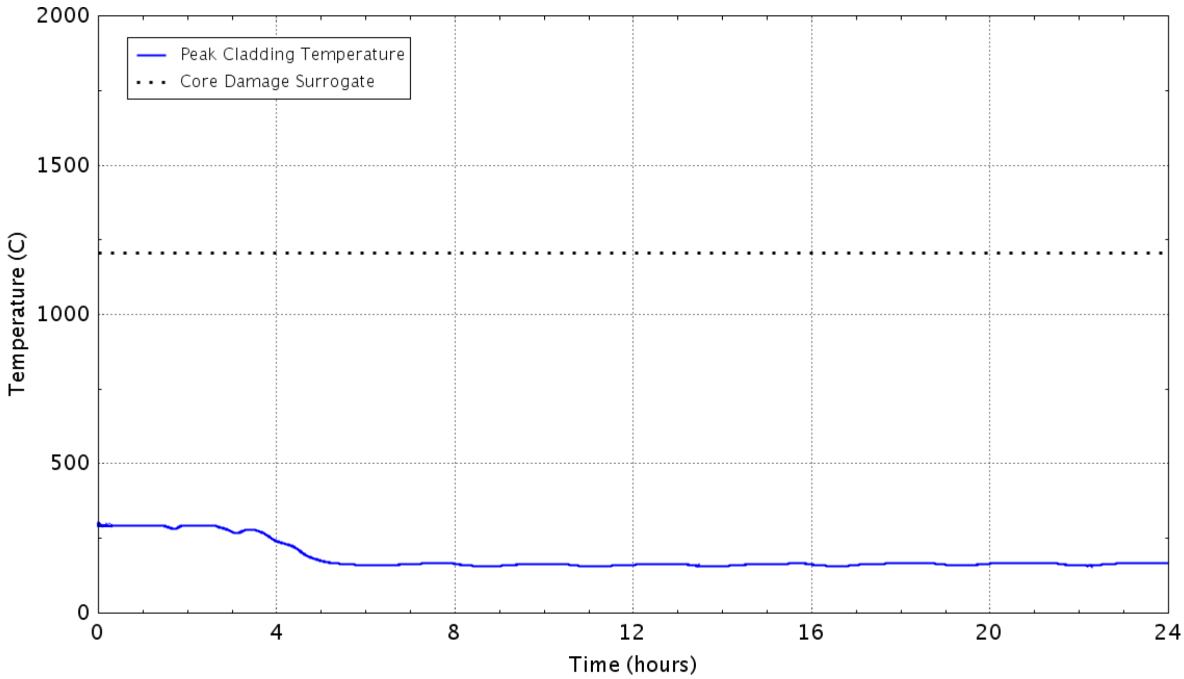

Figure G – 300 Peak temperature of the fuel cladding as a function of time

G.3.9 Case 16a: Sensitivity to LOMFW-25 Case 16 with MSIV Closureat the Start of the Transient

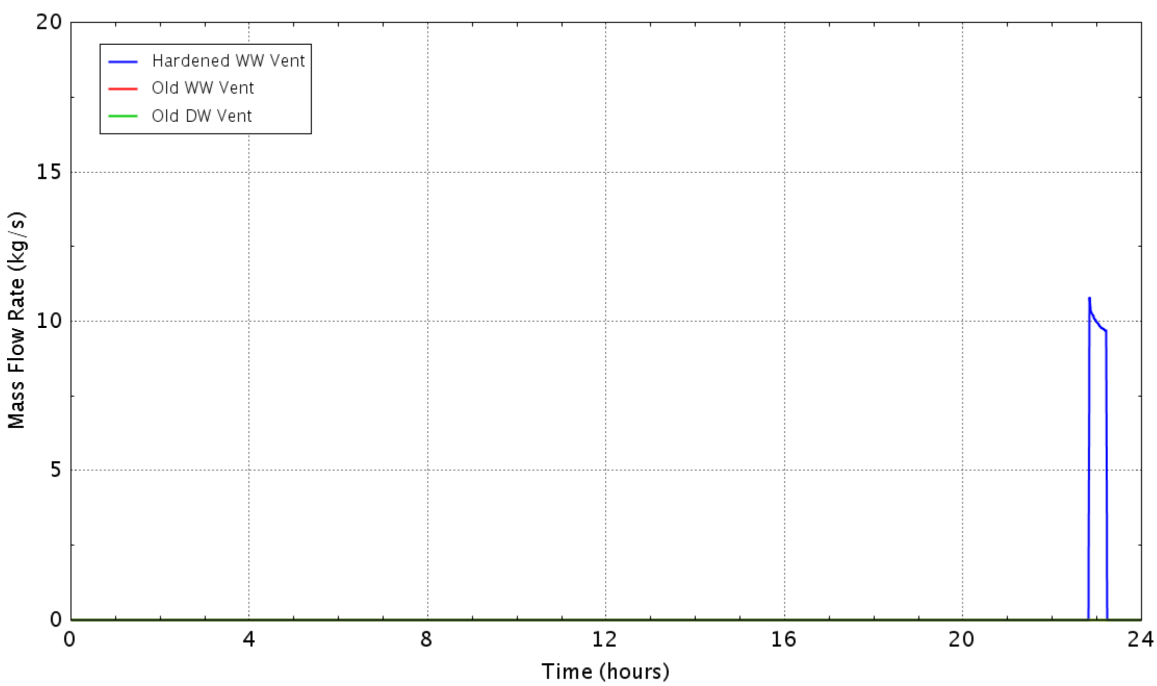

Figure G – 301 Flow rate of the containment vents

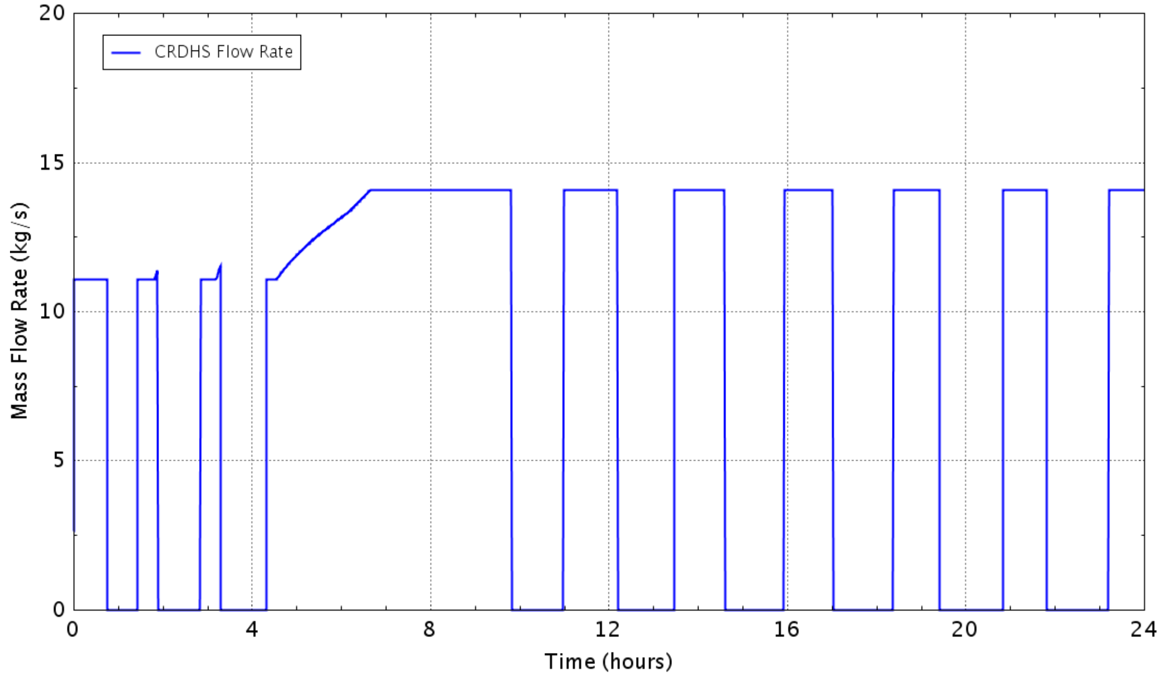

Figure G – 302 Flow rate of the control rod drive hydraulic system

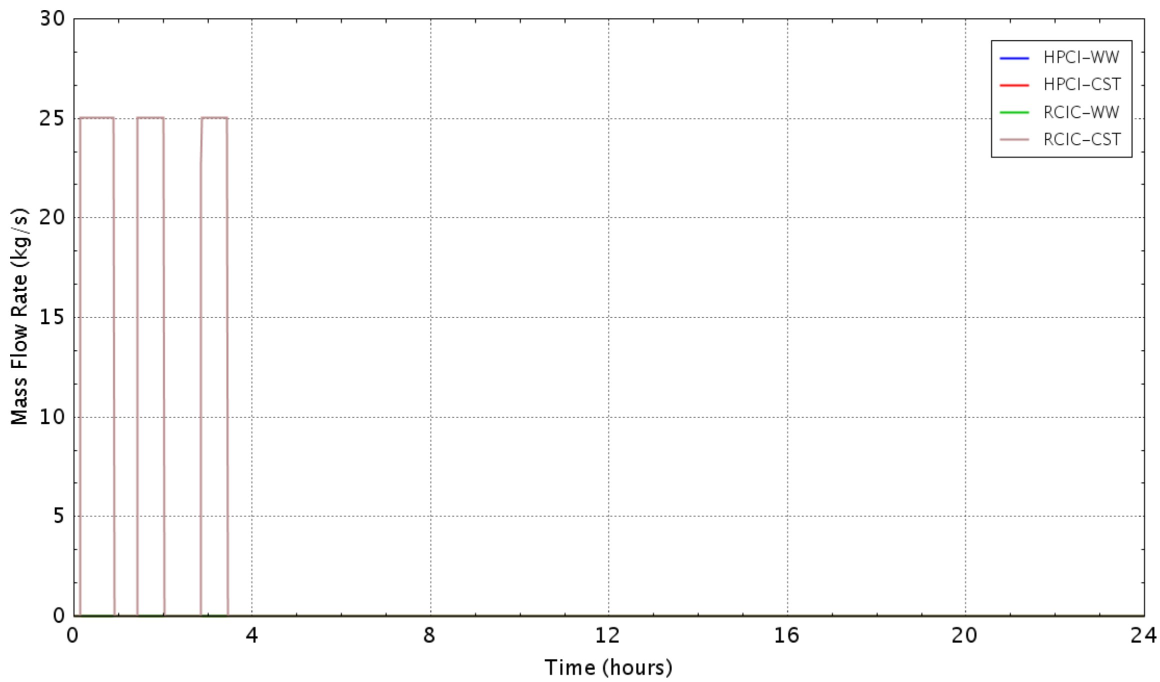

Figure G – 303 Flow rate of the HPCI/RCIC pumps

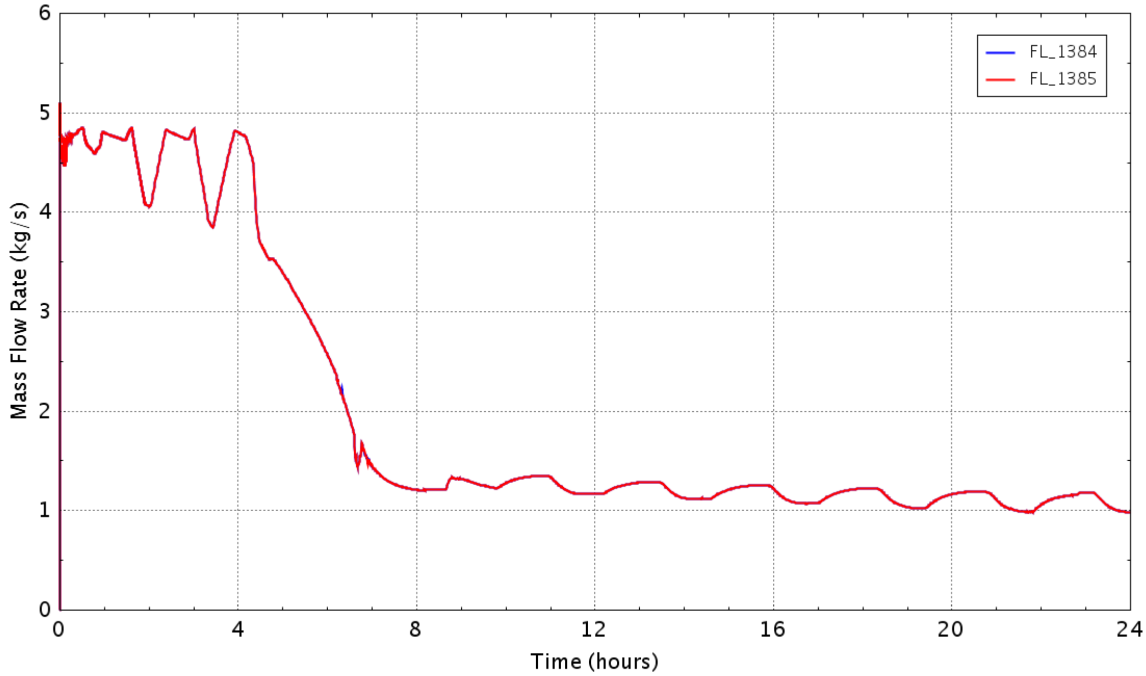

Figure G – 304 Flow rate of the recirculating pump seal leakage

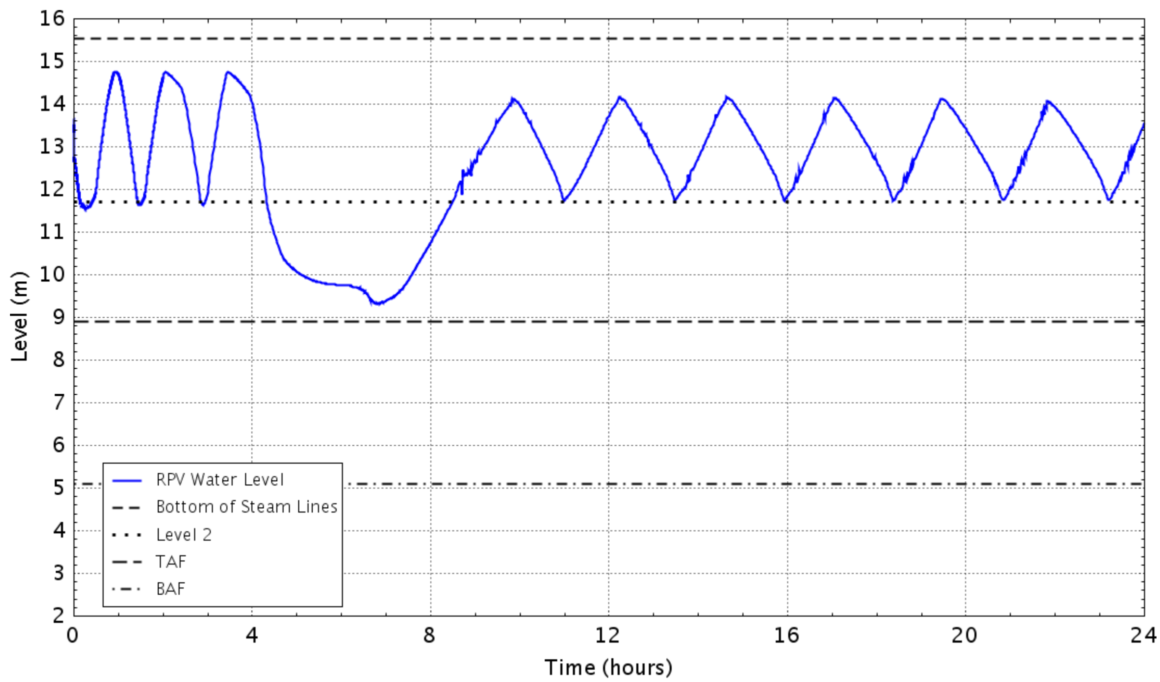

Figure G – 305 RPV Downcomer water level

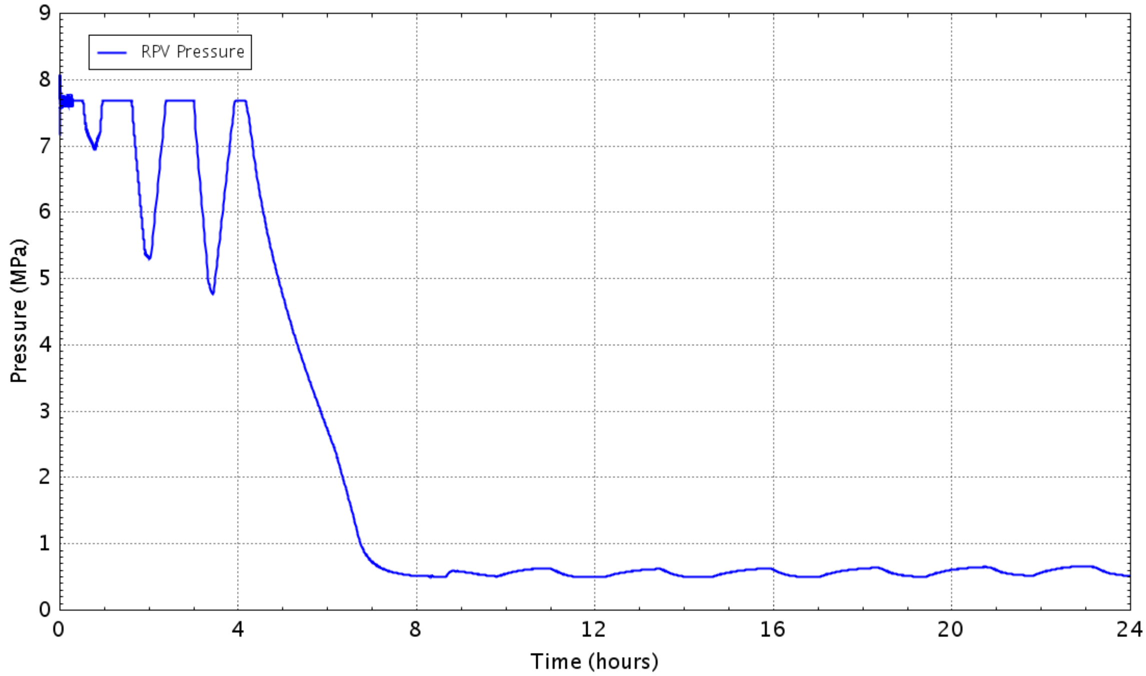

Figure G – 306 Pressure in the RPV

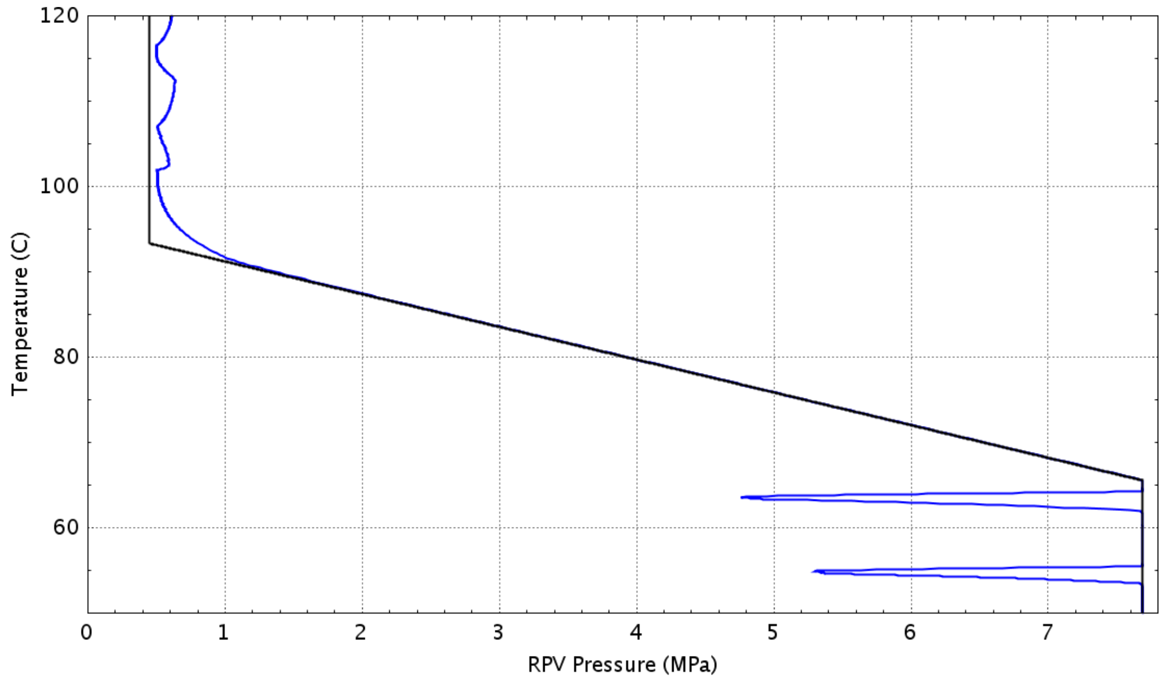

Figure G – 307 Plant status relative to the HCL curve (Graph 4 of the EOPs)

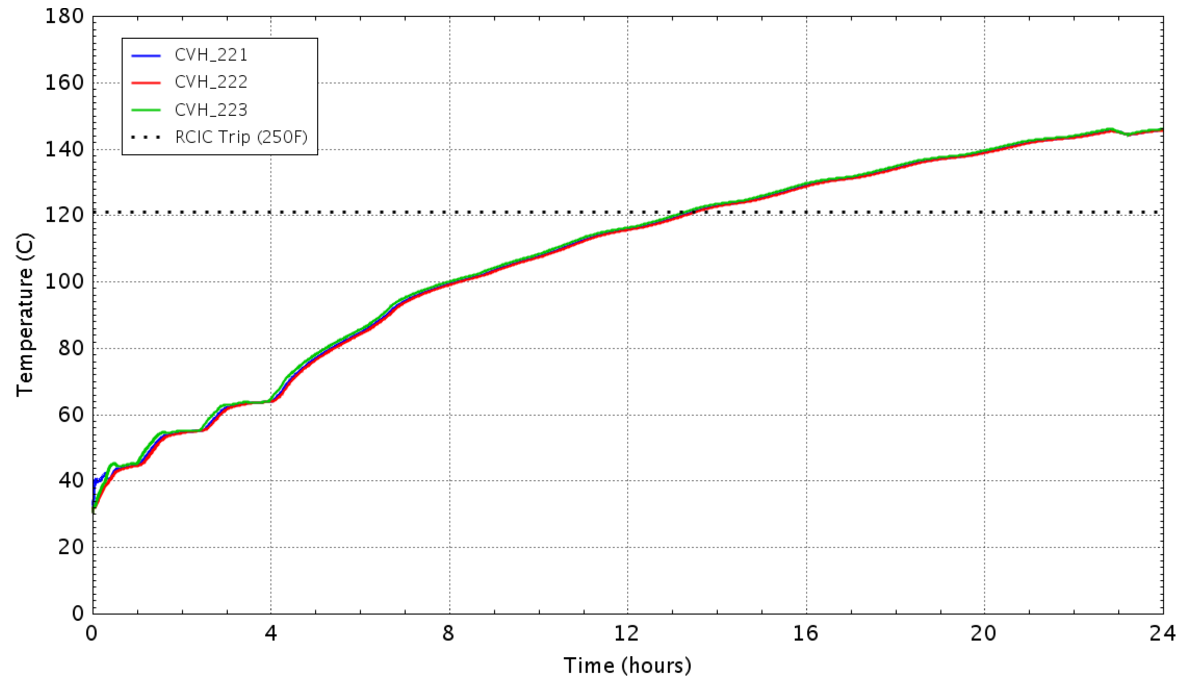

Figure G – 308 Water temperature in the wetwell

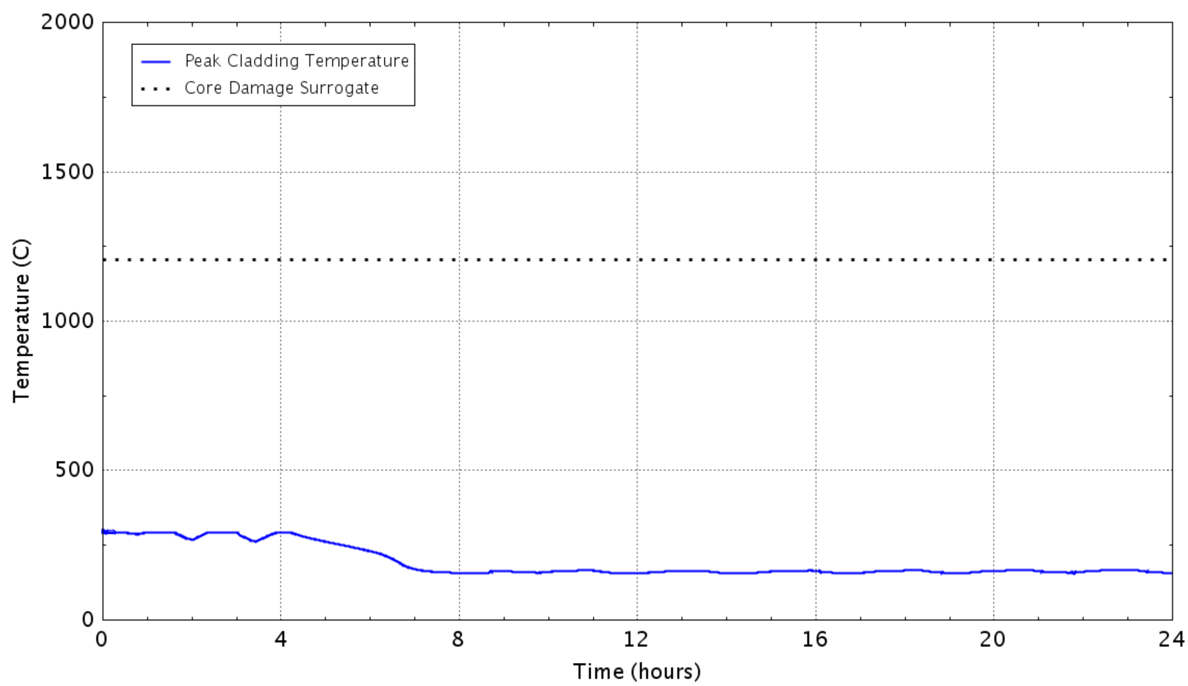

Figure G – 309 Peak temperature of the fuel cladding as a function of time

G.3.10 Case 17a: Sensitivity to LOMFW-25 Case 17 with HPCI Available Instead of RCIC

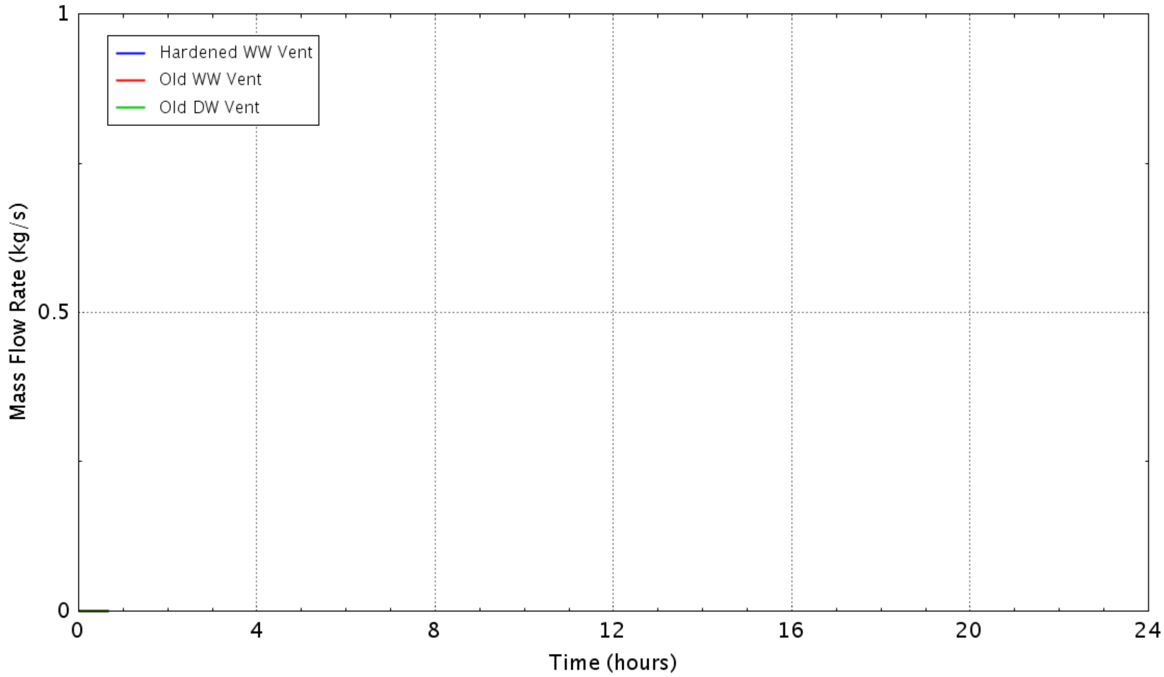

Figure G – 310 Flow rate of the containment vents

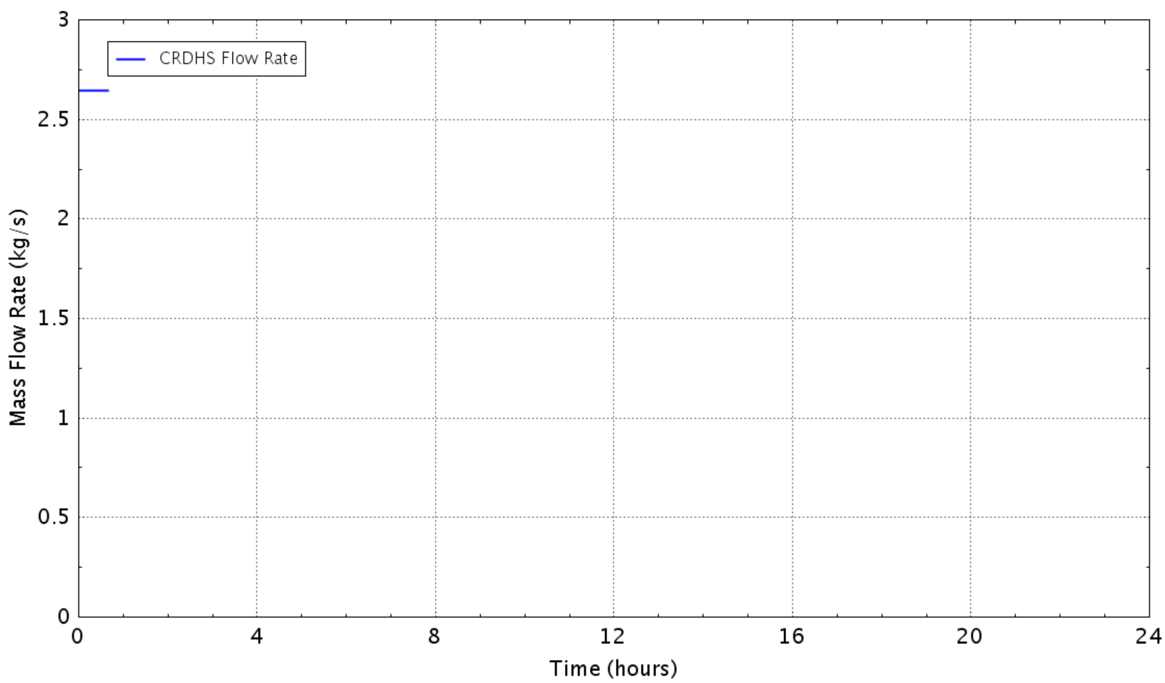

Figure G – 311 Flow rate of the control rod drive hydraulic system

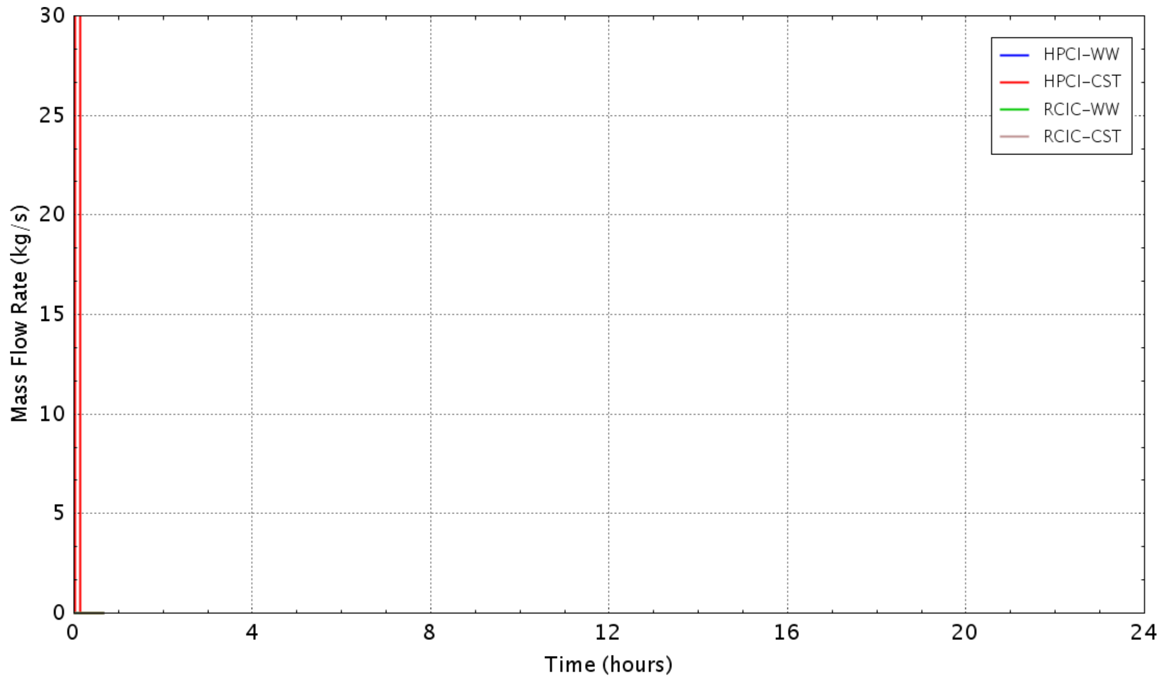

Figure G – 312 Flow rate of the HPCI/RCIC pumps

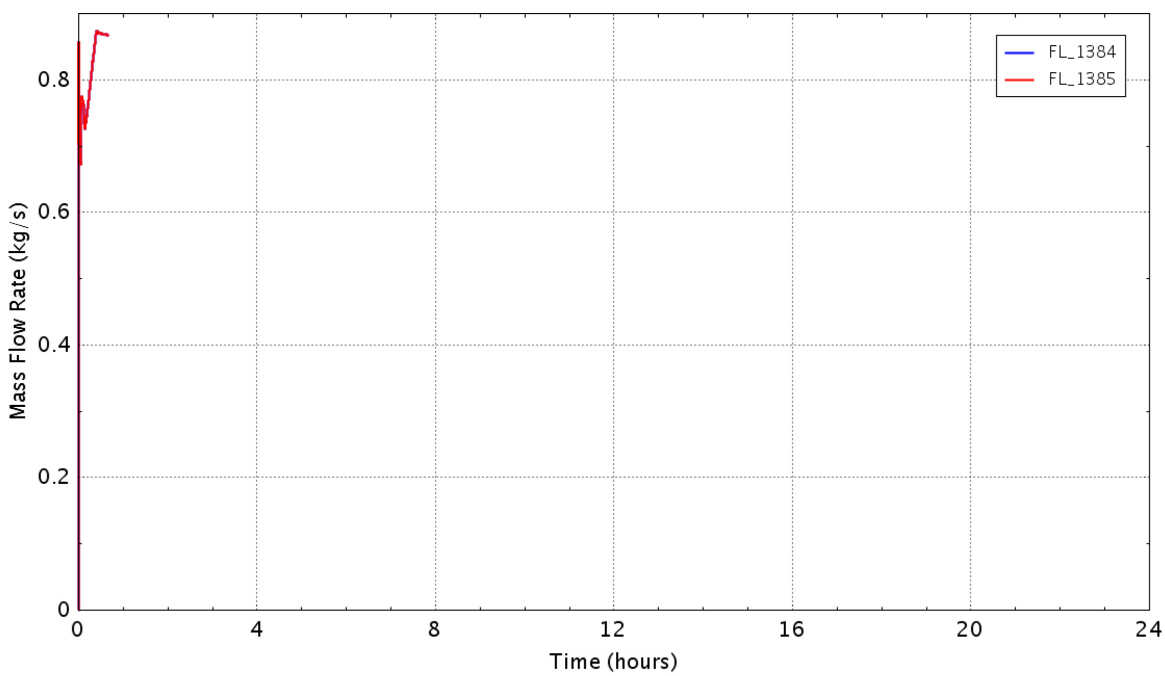

Figure G – 313 Flow rate of the recirculating pump seal leakage

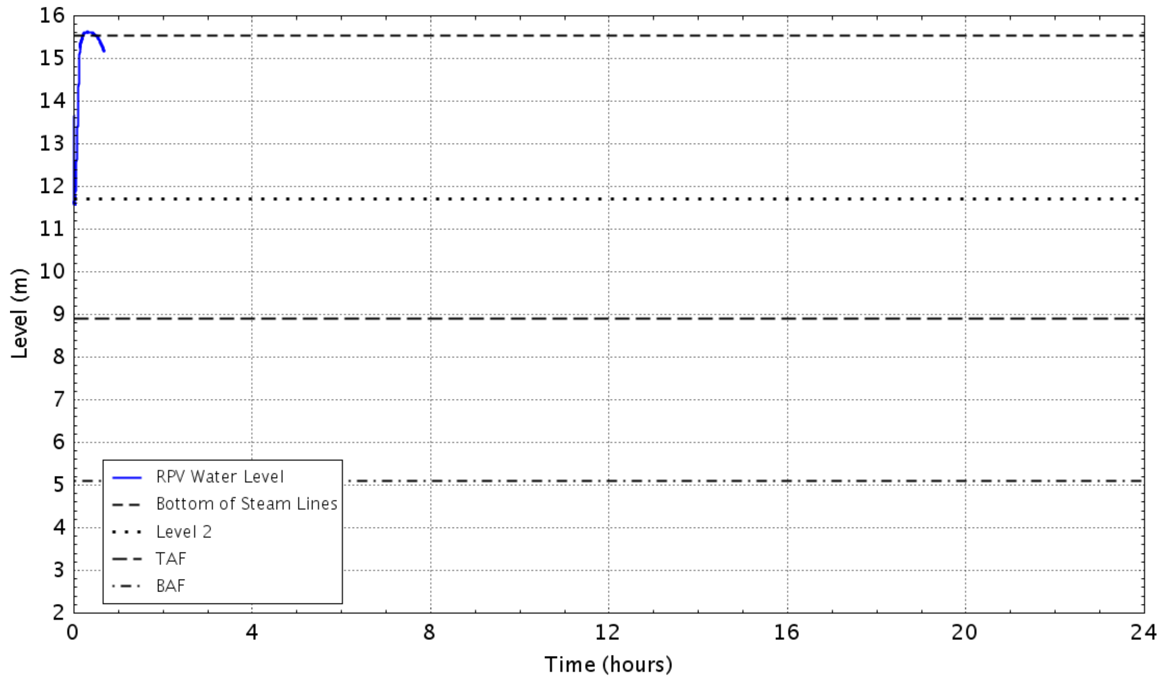

Figure G – 314 RPV Downcomer water level

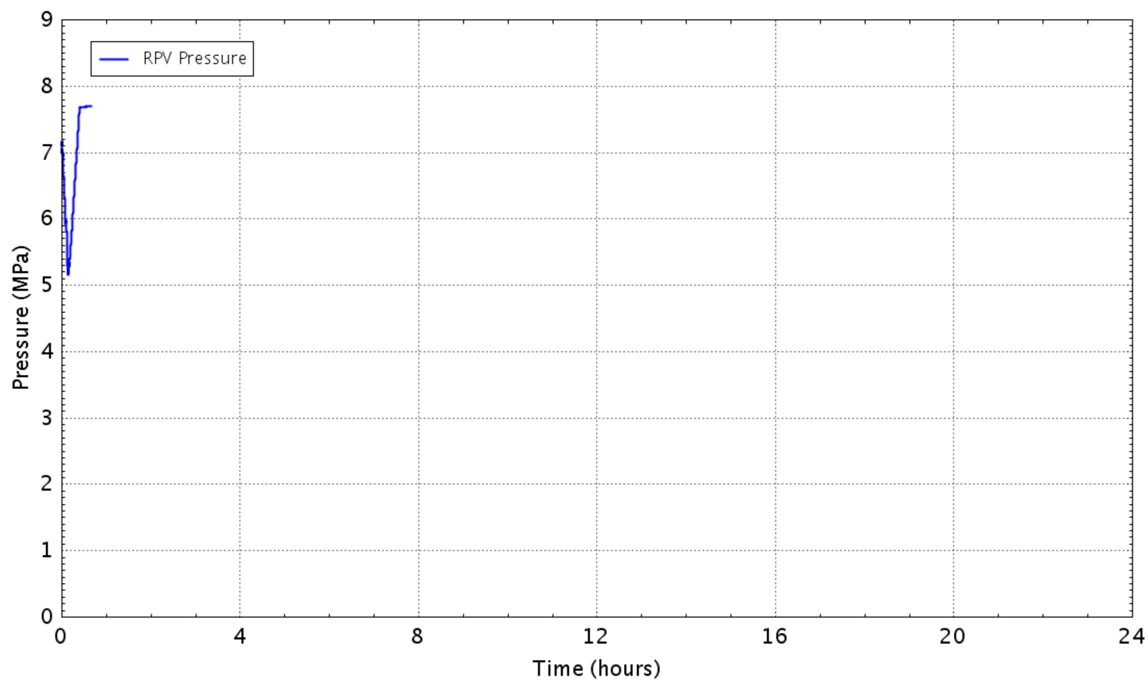

Figure G – 315 Pressure in the RPV

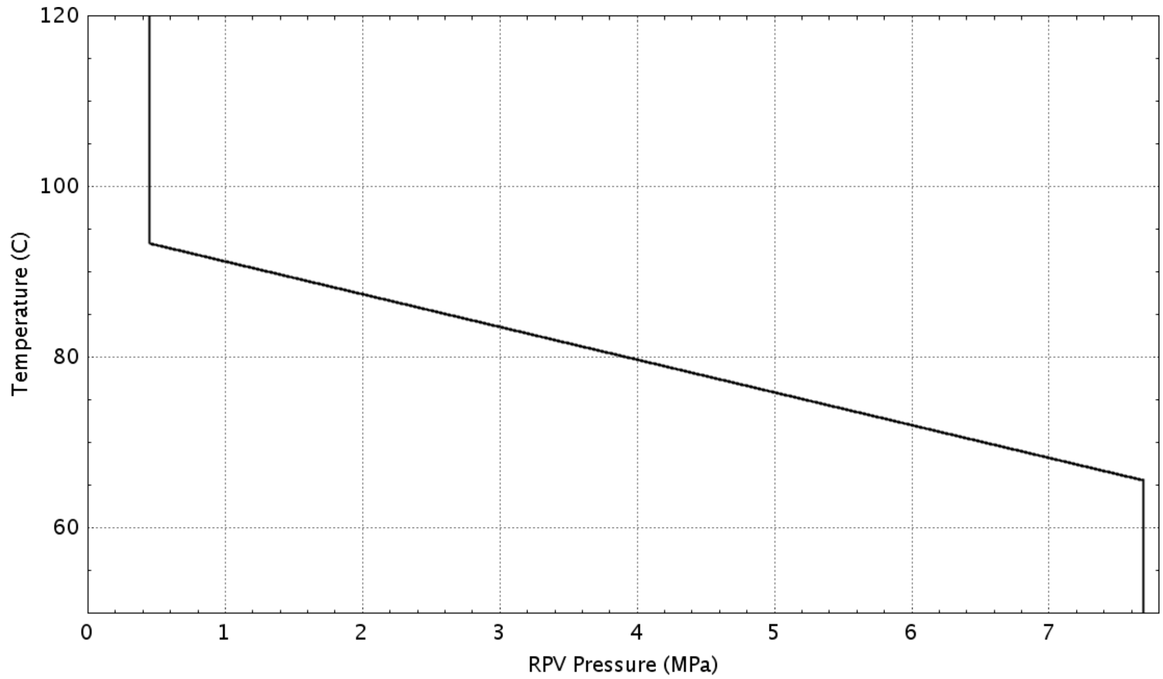

Figure G – 316 Plant status relative to the HCL curve (Graph 4 of the EOPs)

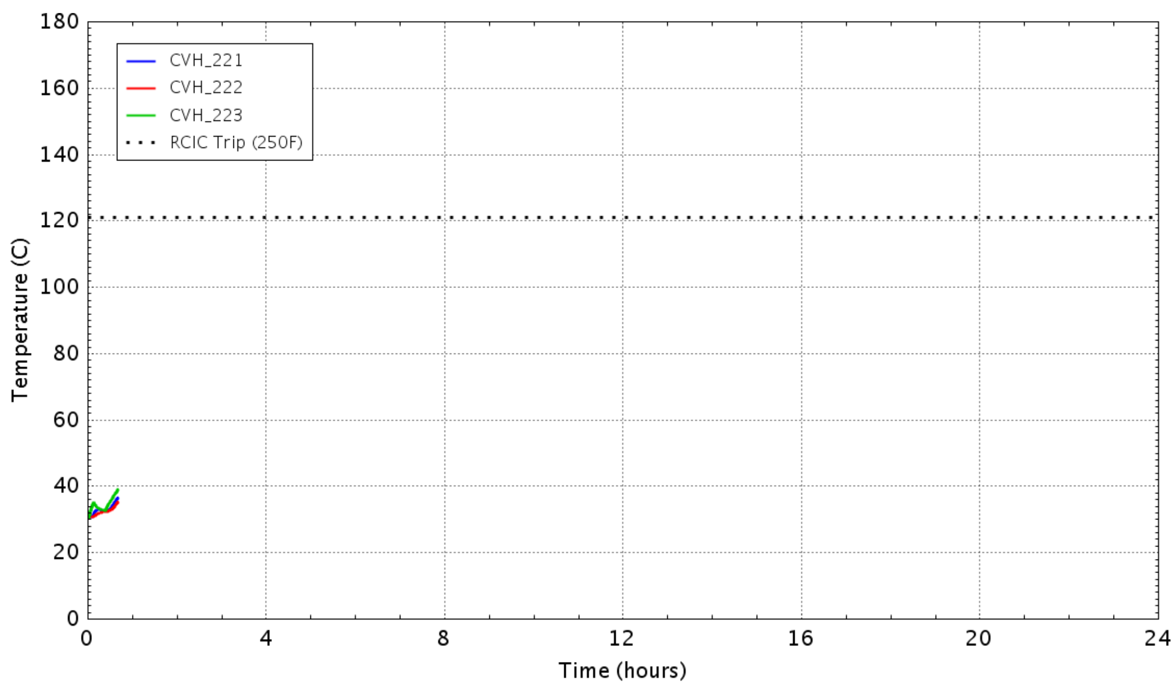

Figure G – 317 Water temperature in the wetwell

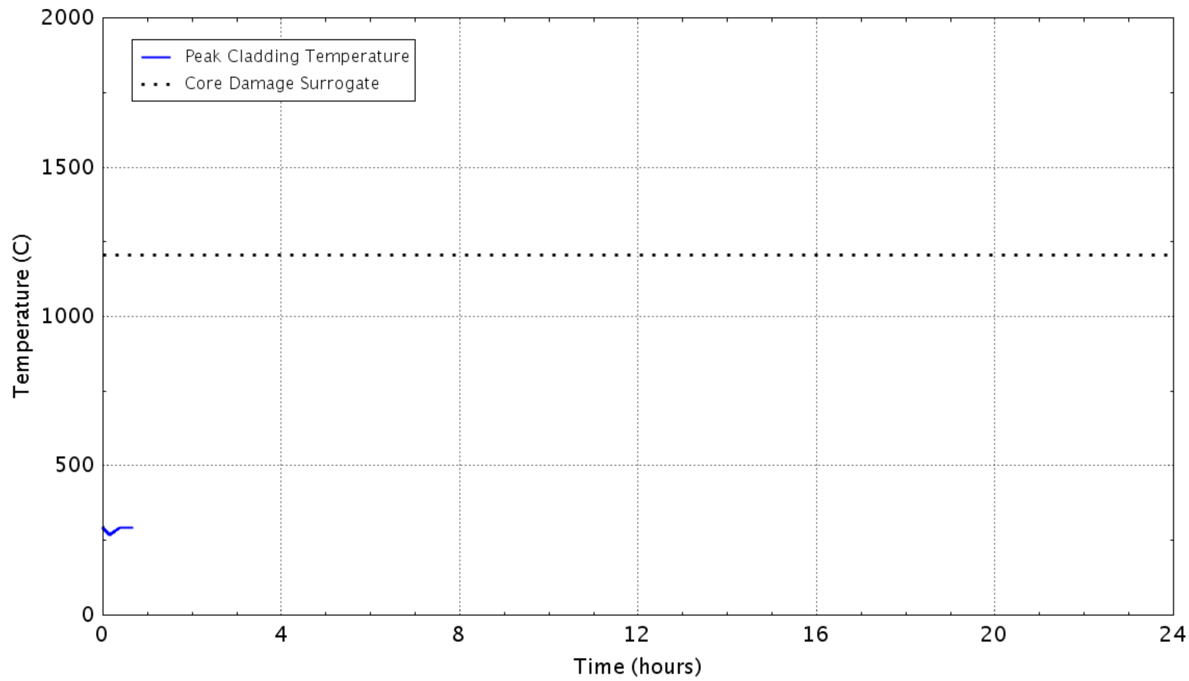

Figure G – 318 Peak temperature of the fuel cladding as a function of time

G.3.11 Case 17b: Sensitivity to LOMFW-25 Case 17 with an SRVFailing Open at 270 Cycles

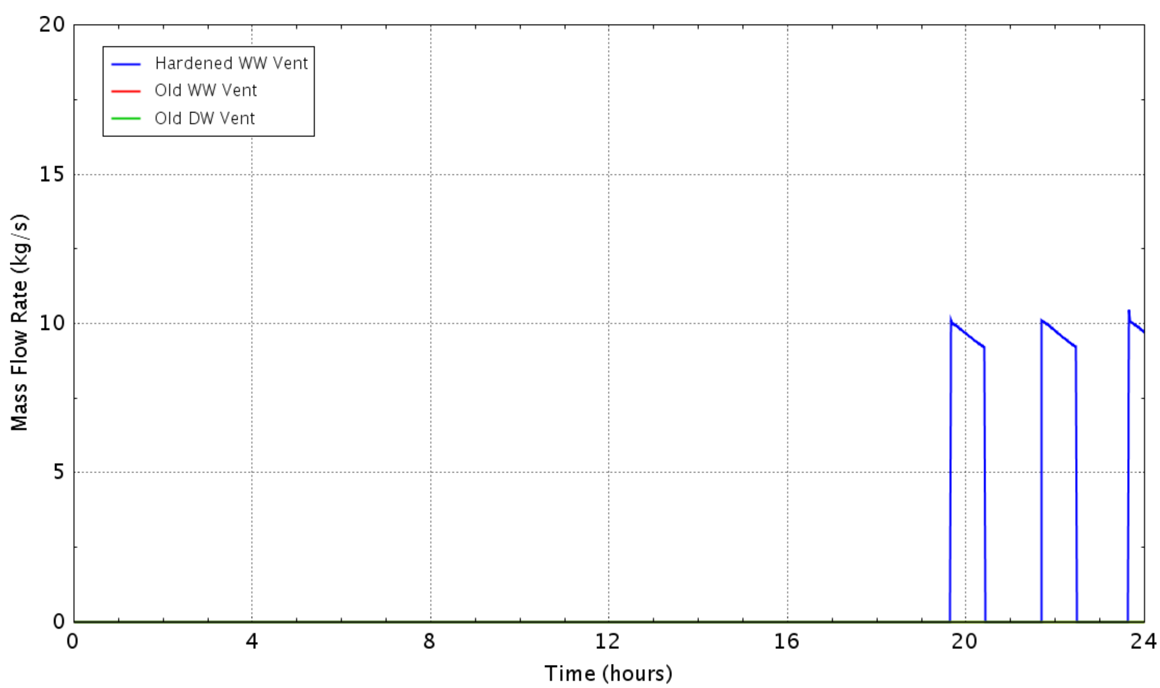

Figure G – 319 Flow rate of the containment vents

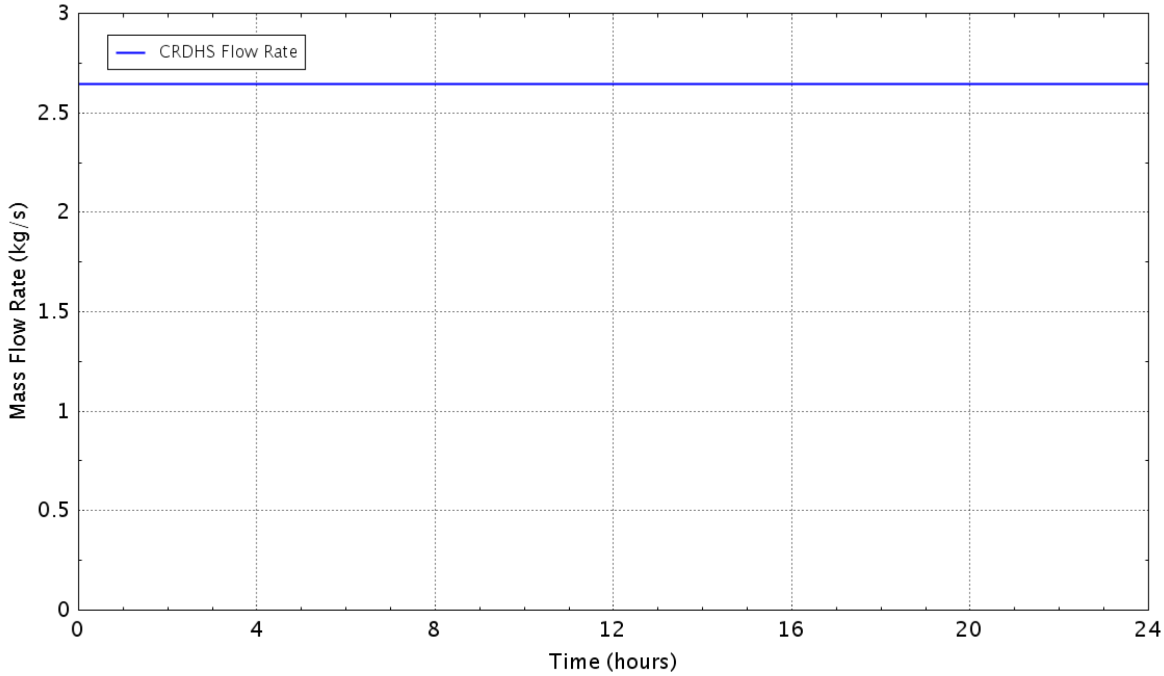

Figure G – 320 Flow rate of the control rod drive hydraulic system

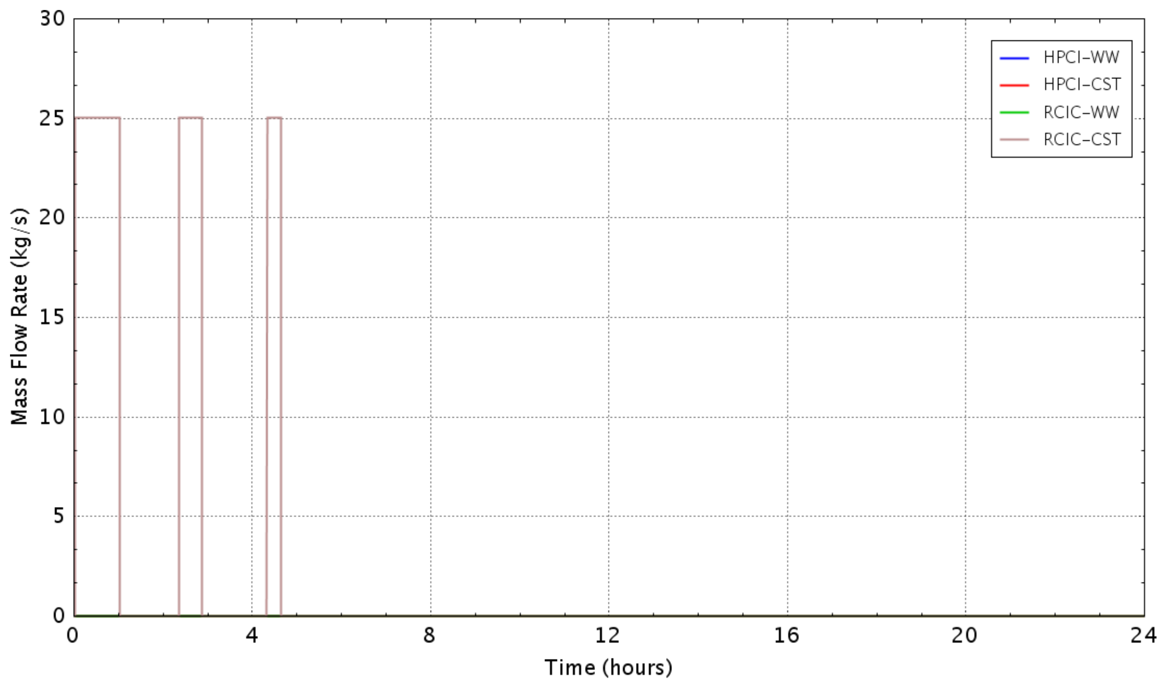

Figure G – 321 Flow rate of the HPCI/RCIC pumps

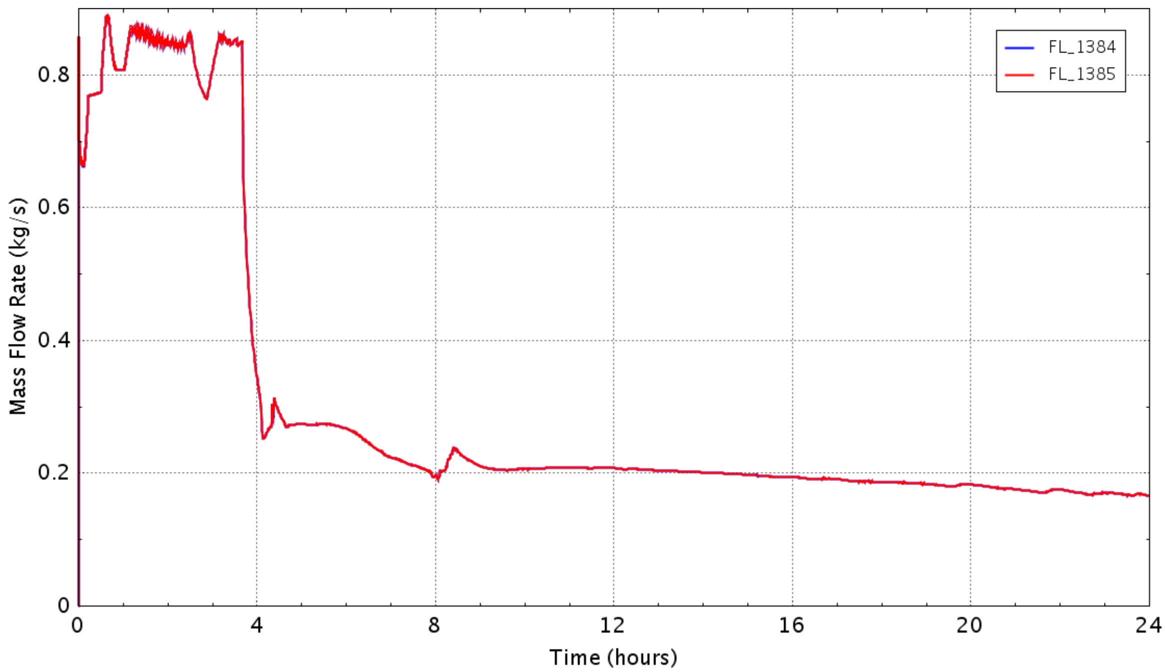

Figure G – 322 Flow rate of the recirculating pump seal leakage

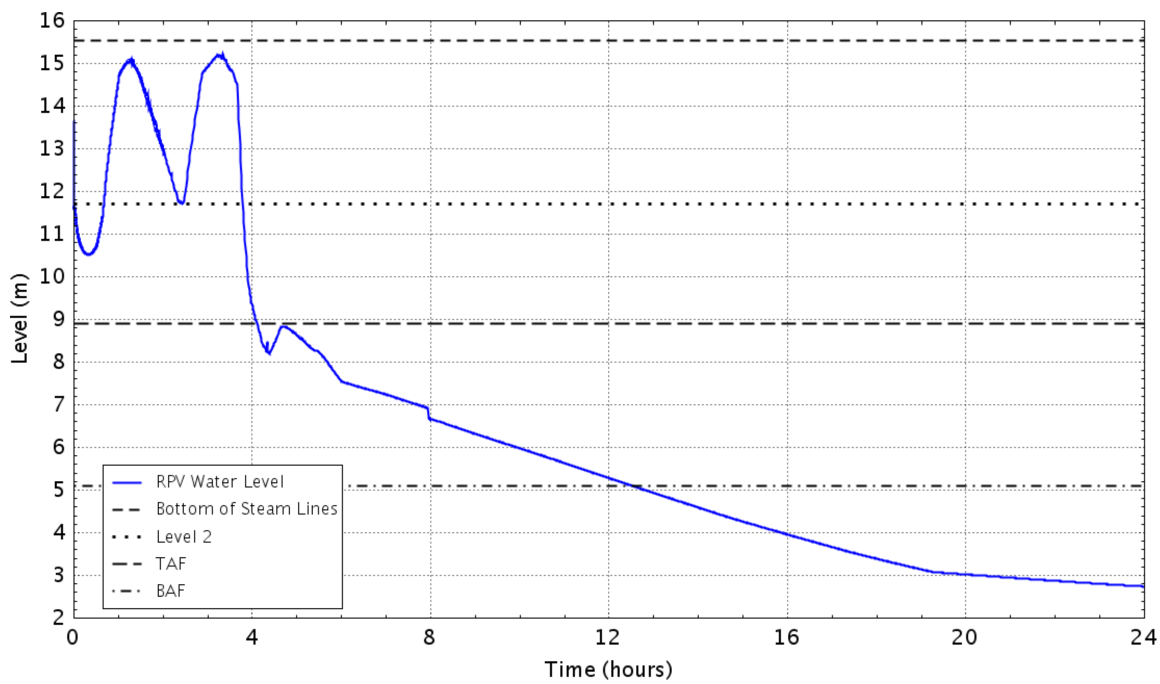

Figure G – 323 RPV Downcomer water level

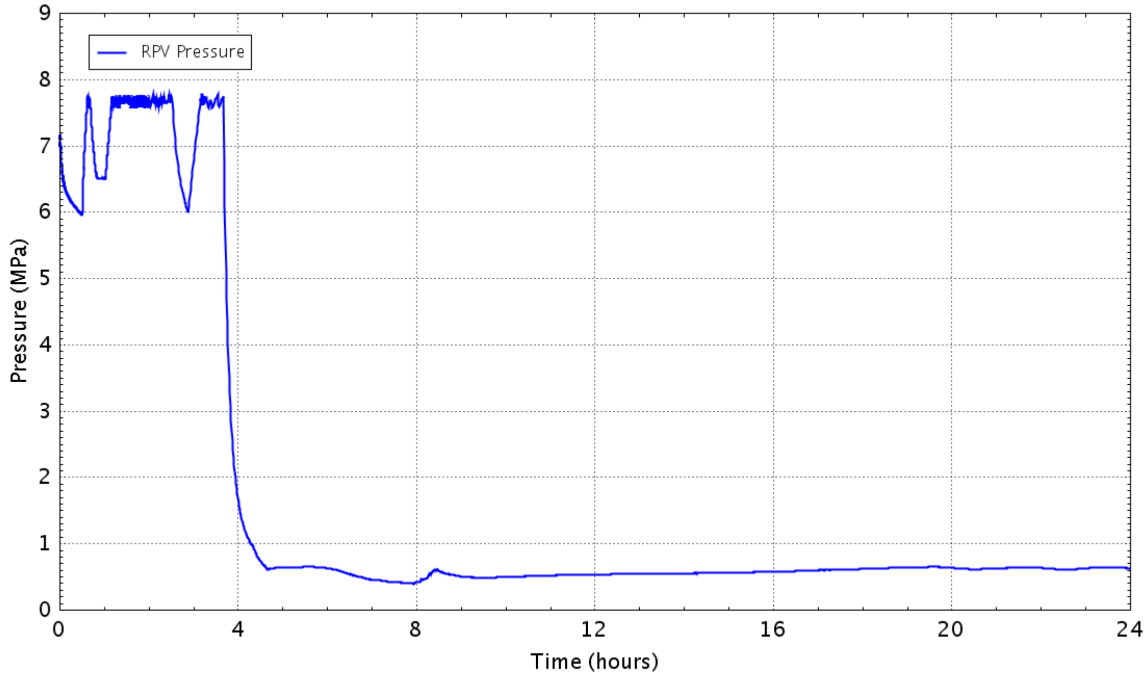

Figure G – 324 Pressure in the RPV

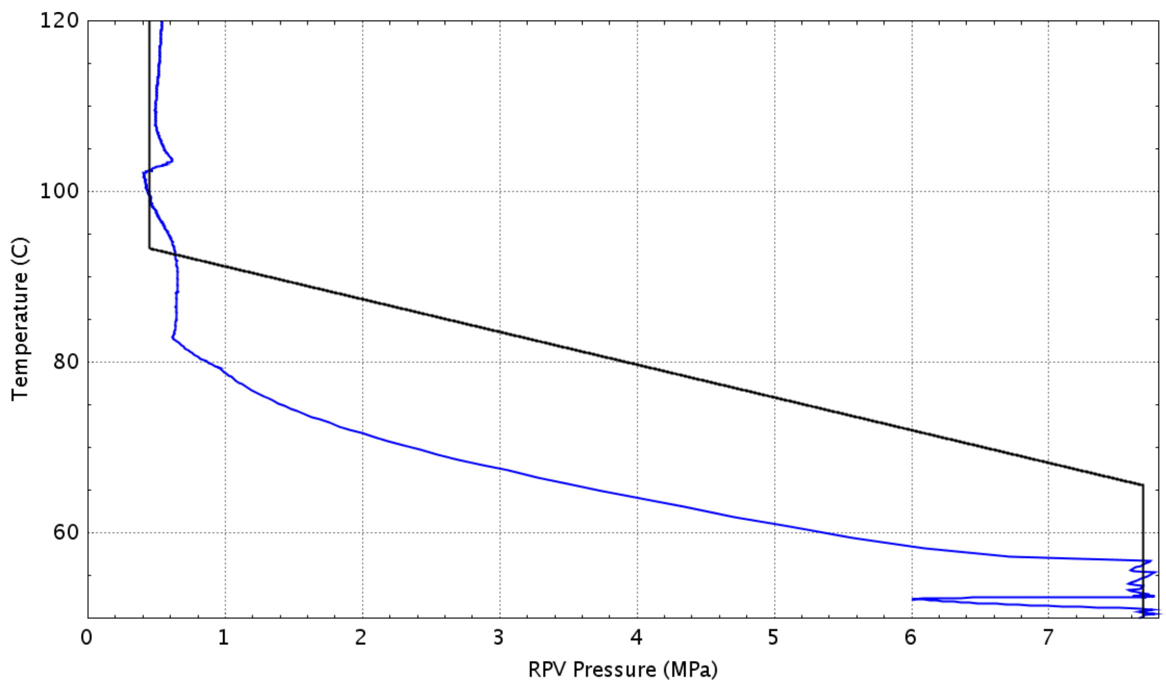

Figure G – 325 Plant status relative to the HCL curve (Graph 4 of the EOPs)

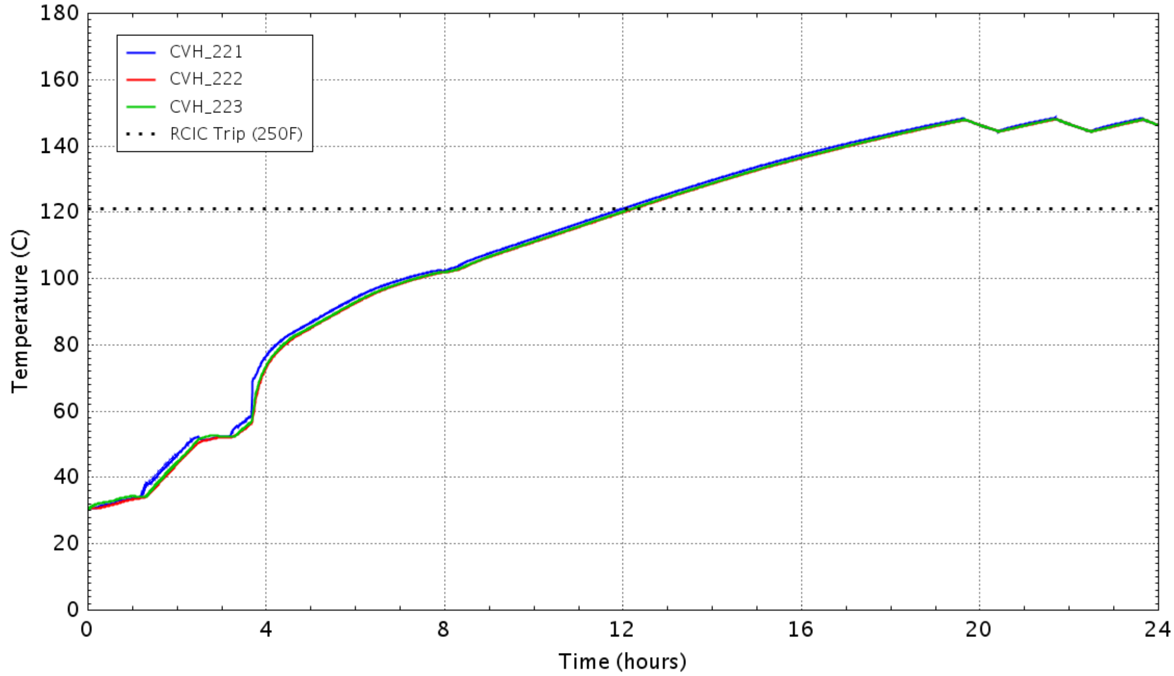

Figure G – 326 Water temperature in the wetwell

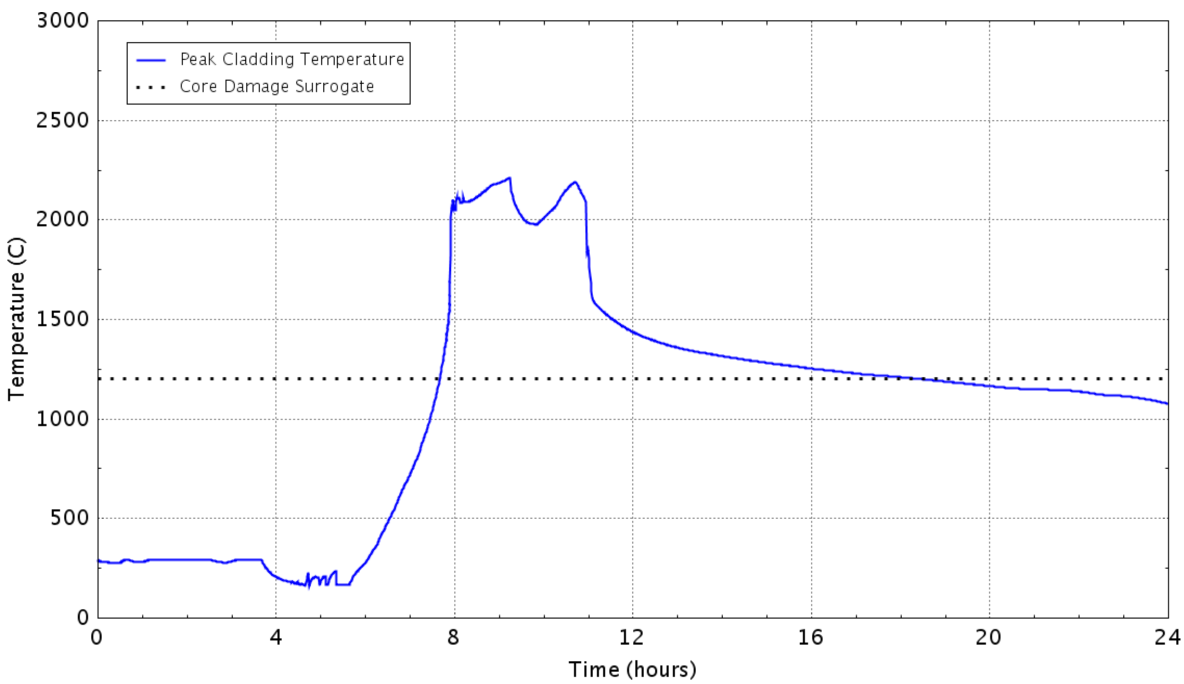

Figure G – 327 Peak temperature of the fuel cladding as a function of time

G.3.12 Case 19a: Sensitivity to LOMFW-25 Case 19 with RCIC Injection from the Wetwell

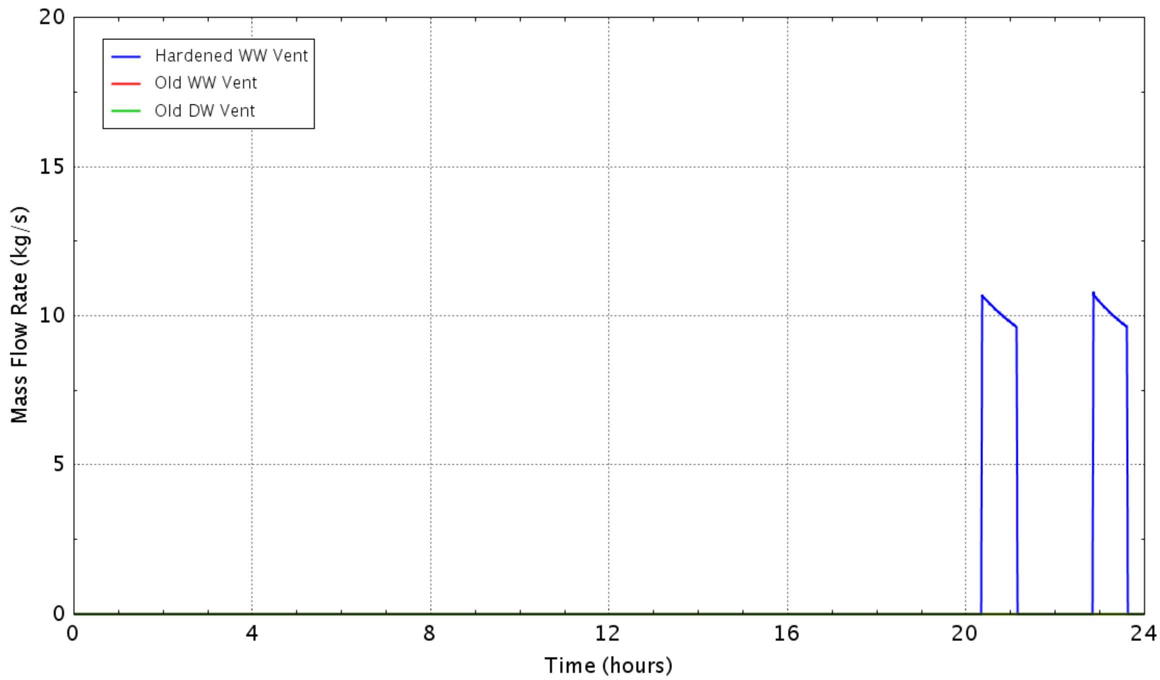

Figure G – 328 Flow rate of the containment vents

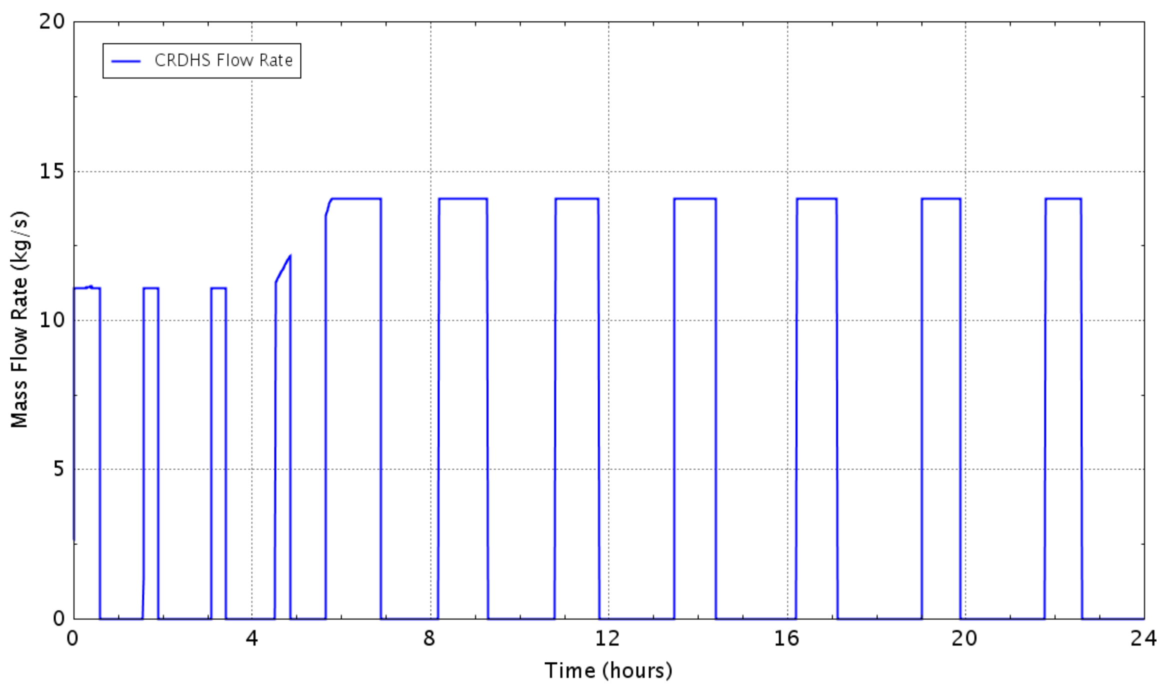

Figure G – 329 Flow rate of the control rod drive hydraulic system

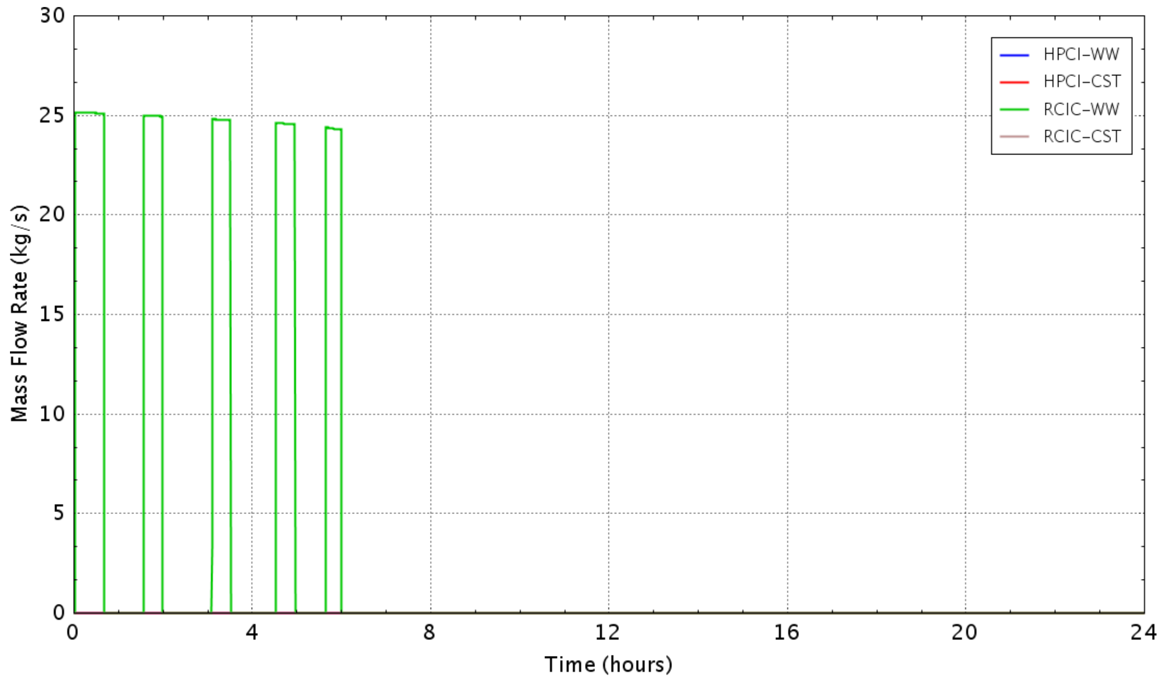

Figure G – 330 Flow rate of the HPCI/RCIC pumps

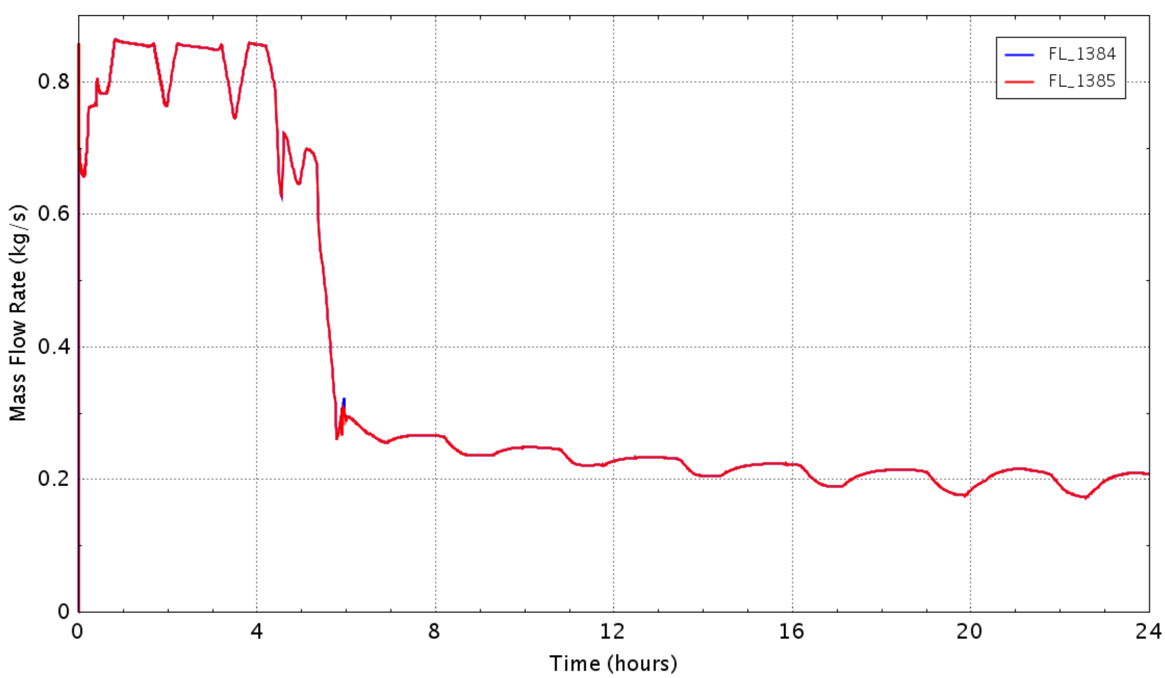

Figure G – 331 Flow rate of the recirculating pump seal leakage

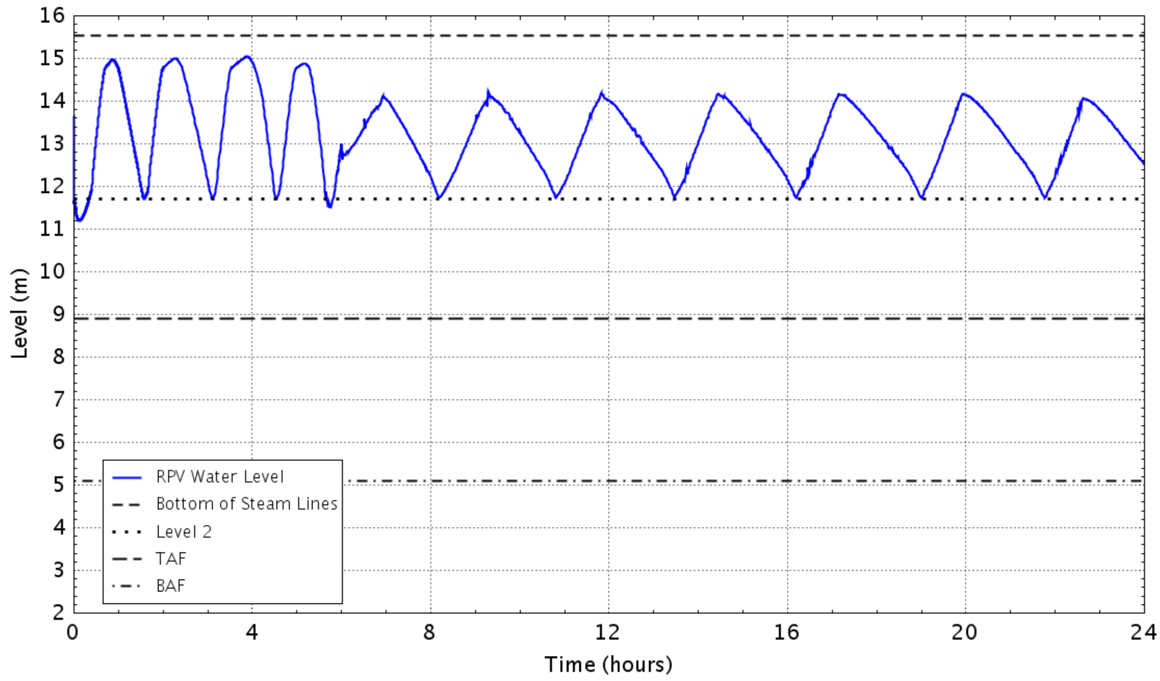

Figure G – 332 RPV Downcomer water level

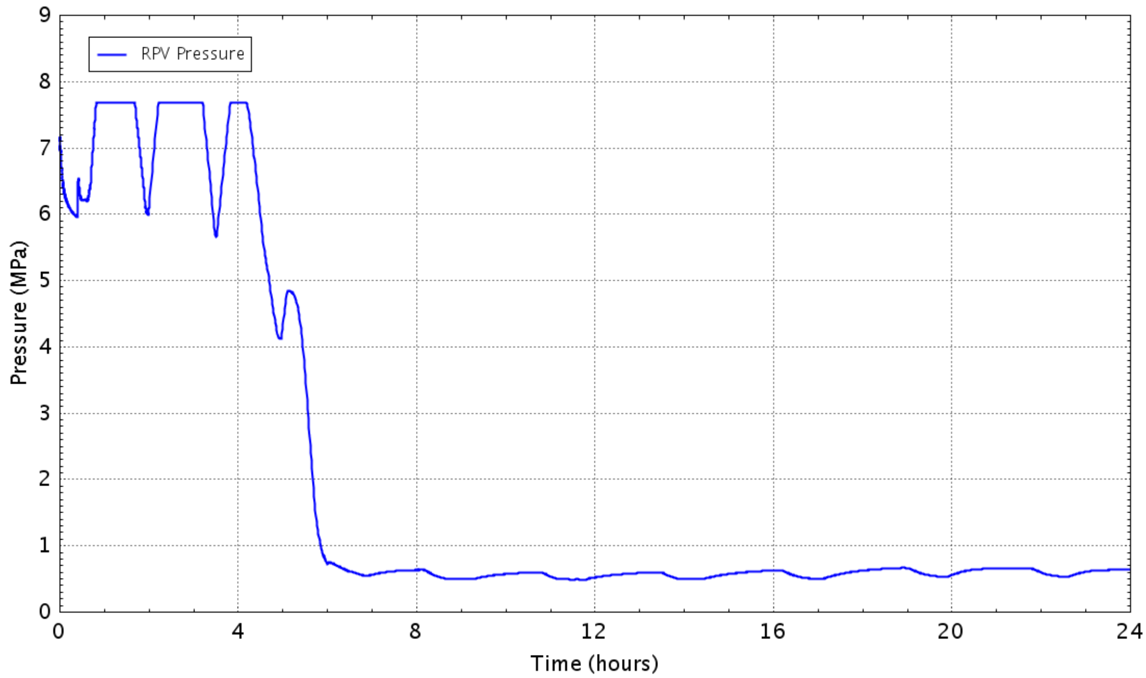

Figure G – 333 Pressure in the RPV

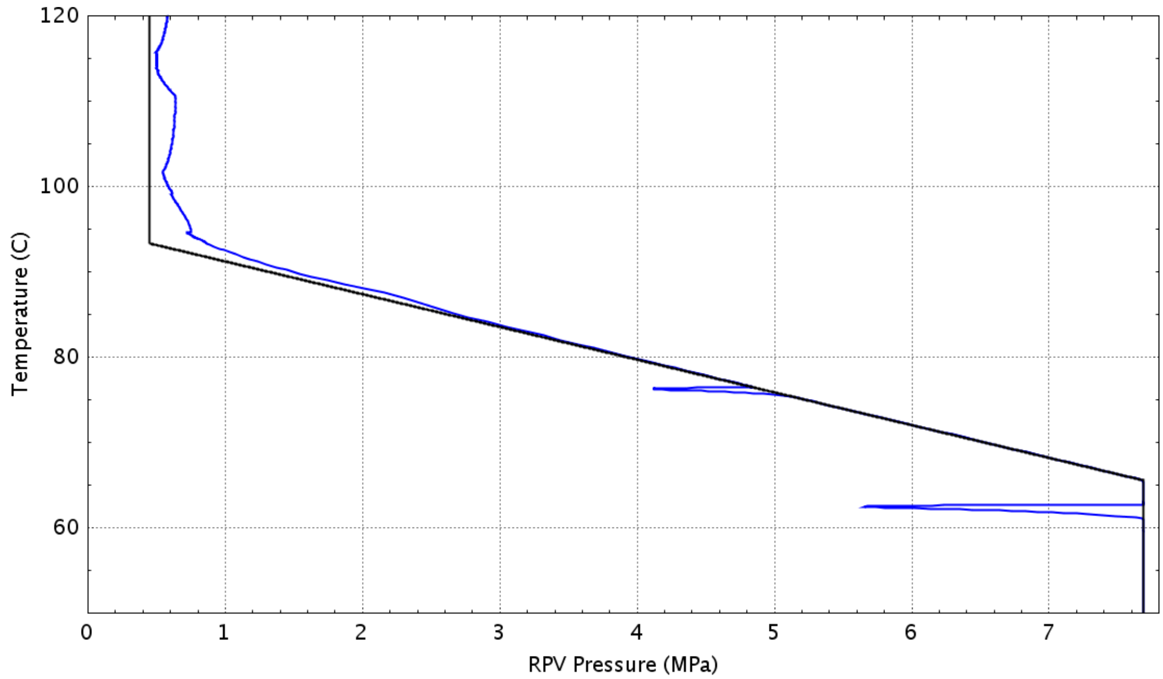

Figure G – 334 Plant status relative to the HCL curve (Graph 4 of the EOPs)

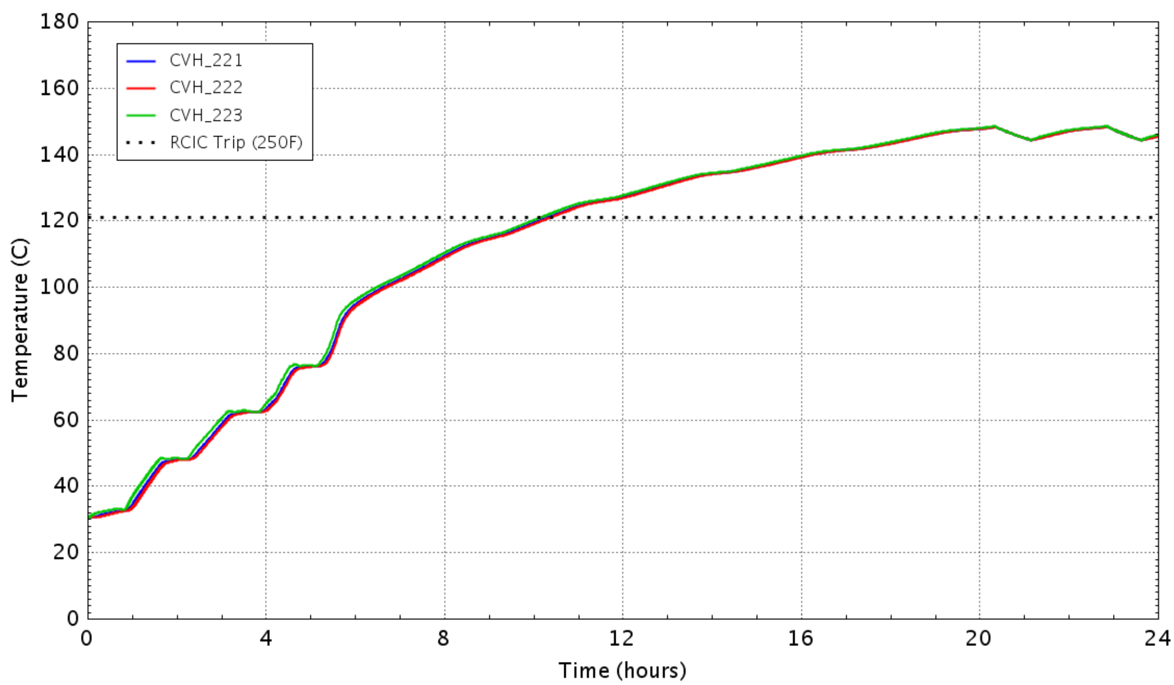

Figure G – 335 Water temperature in the wetwell

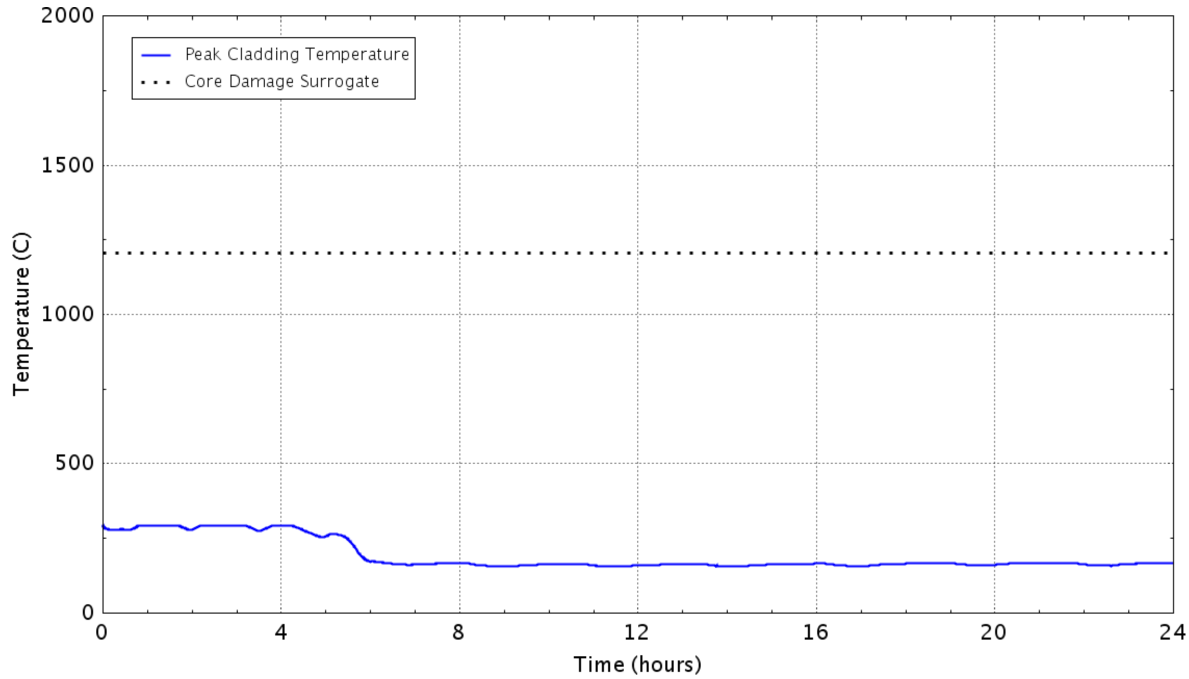

Figure G – 336 Peak temperature of the fuel cladding as a function of time

G.3.13 Case 19b: Sensitivity to LOMFW-25 Case 19 with RPV Emergency Depressurization when the HCL Curve is Reached

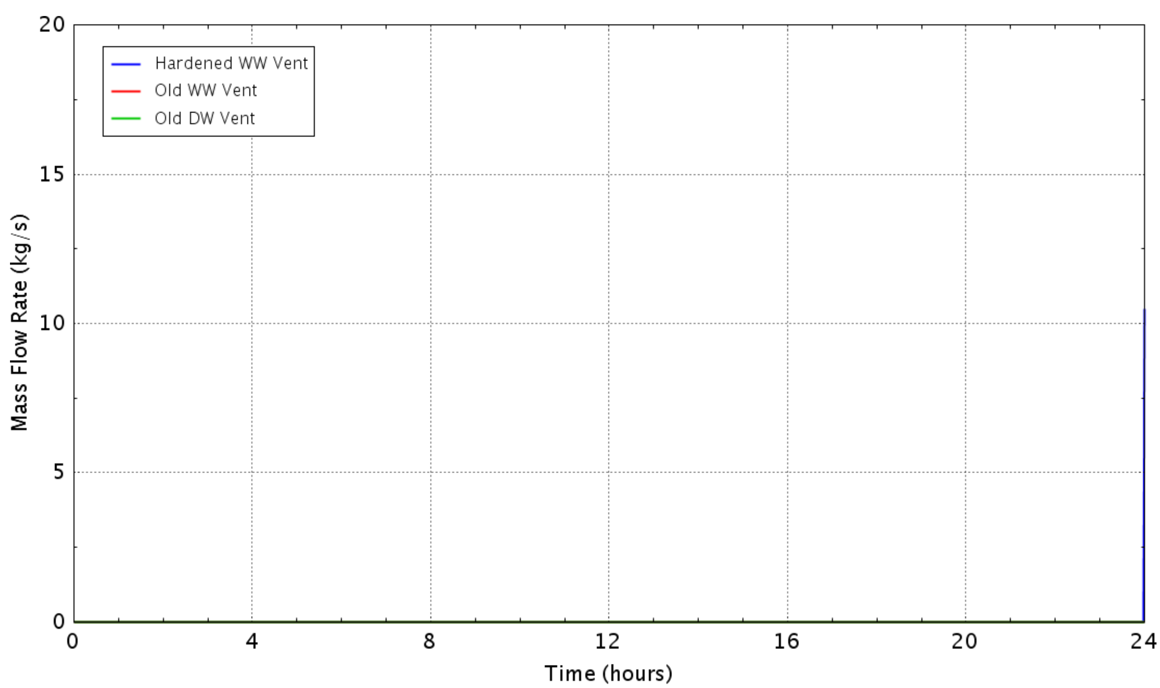

Figure G – 337 Flow rate of the containment vents

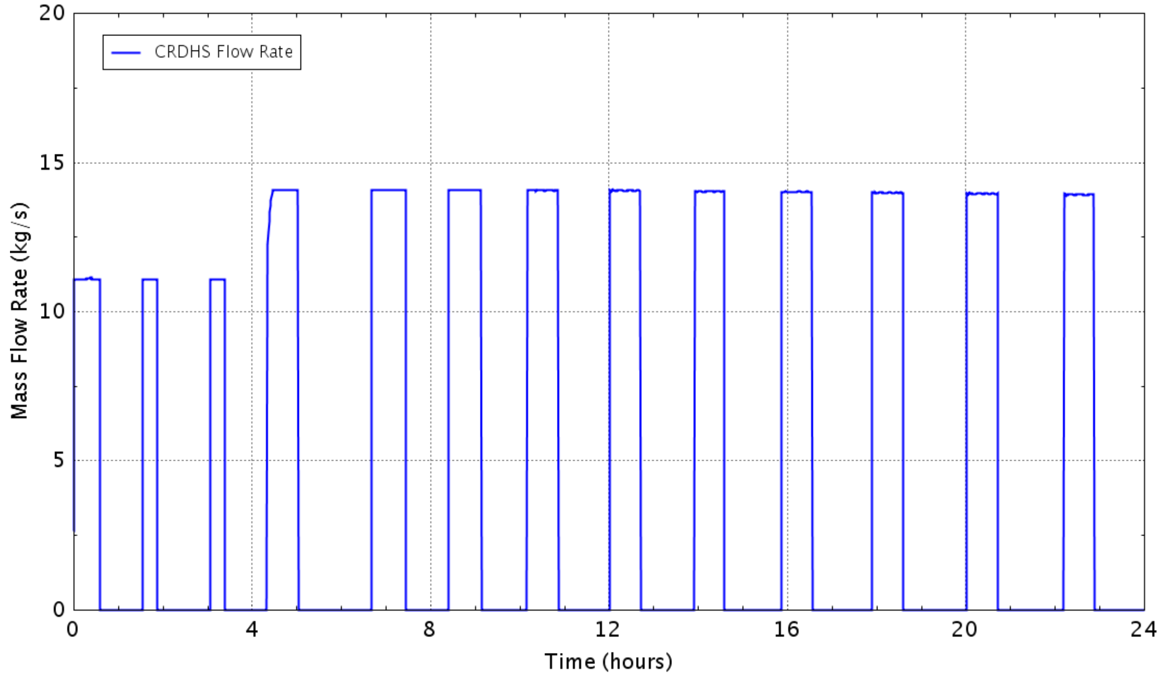

Figure G – 338 Flow rate of the control rod drive hydraulic system

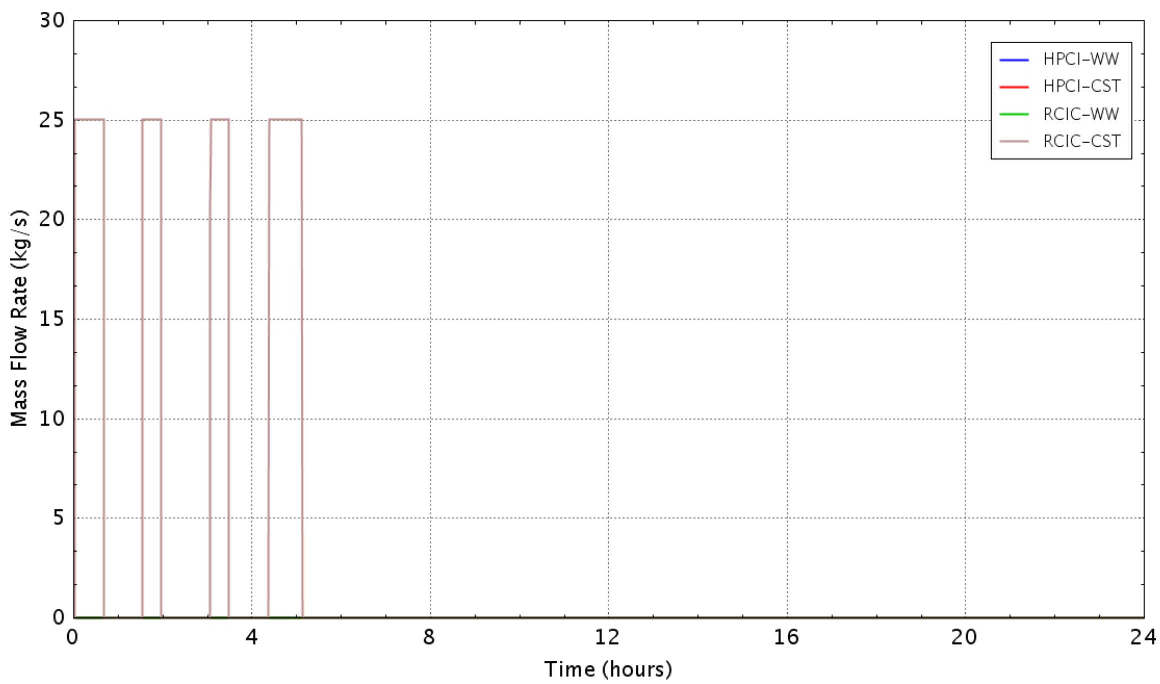

Figure G – 339 Flow rate of the HPCI/RCIC pumps

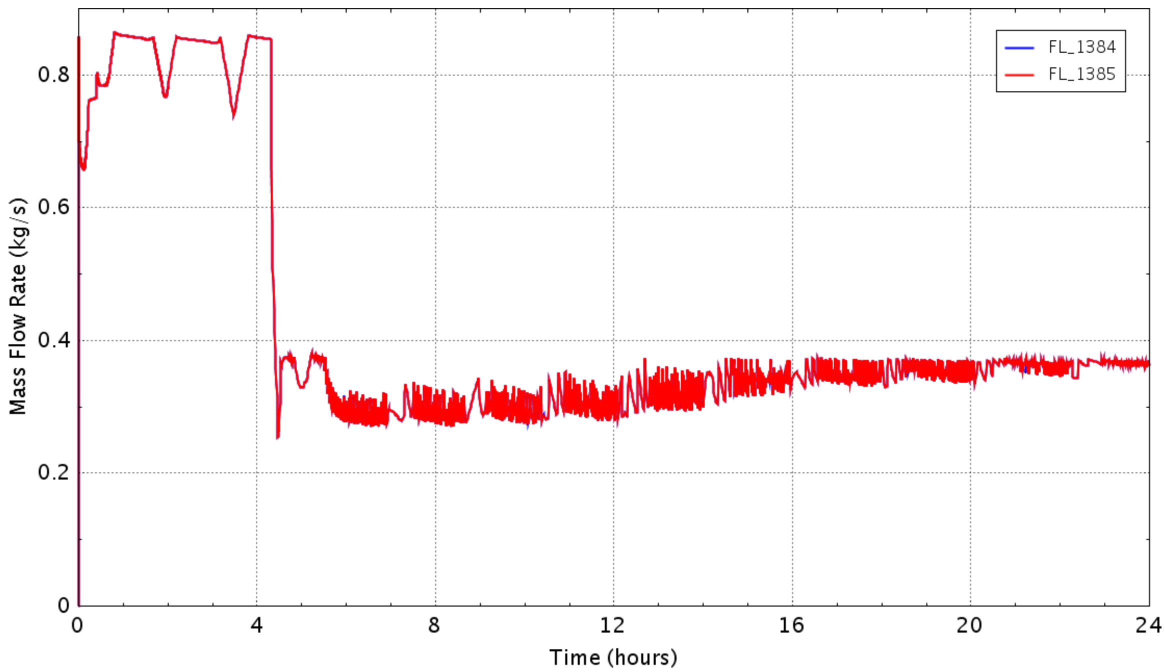

Figure G – 340 Flow rate of the recirculating pump seal leakage

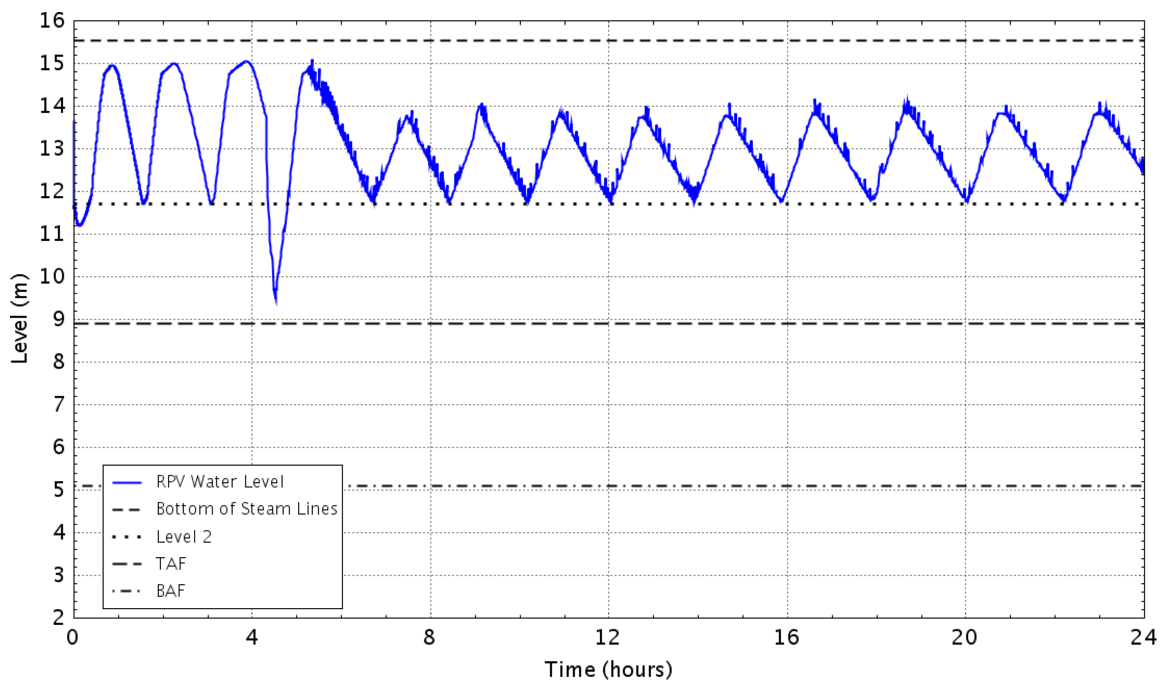

Figure G – 341 RPV Downcomer water level

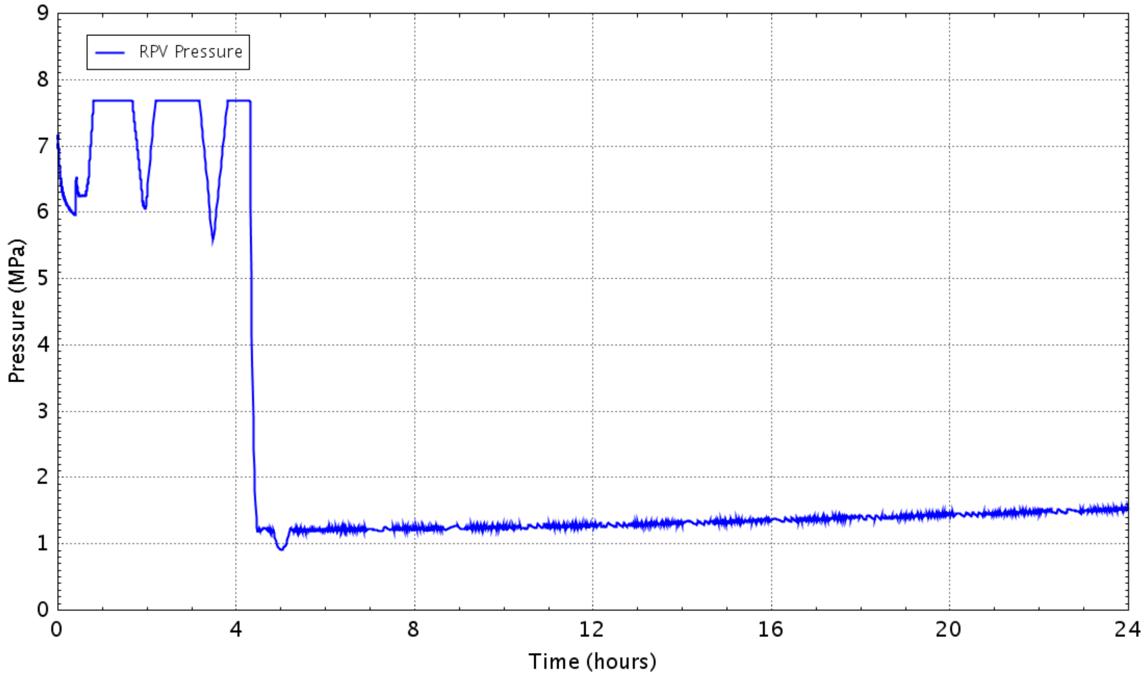

Figure G – 342 Pressure in the RPV

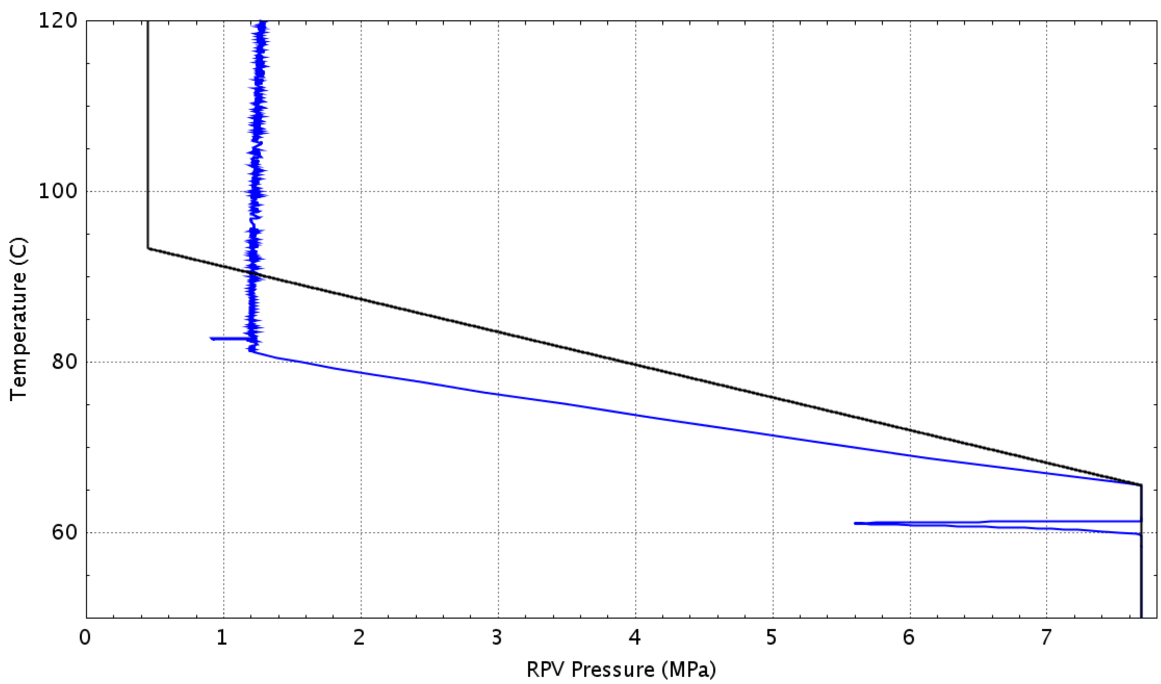

Figure G – 343 Plant status relative to the HCL curve (Graph 4 of the EOPs)

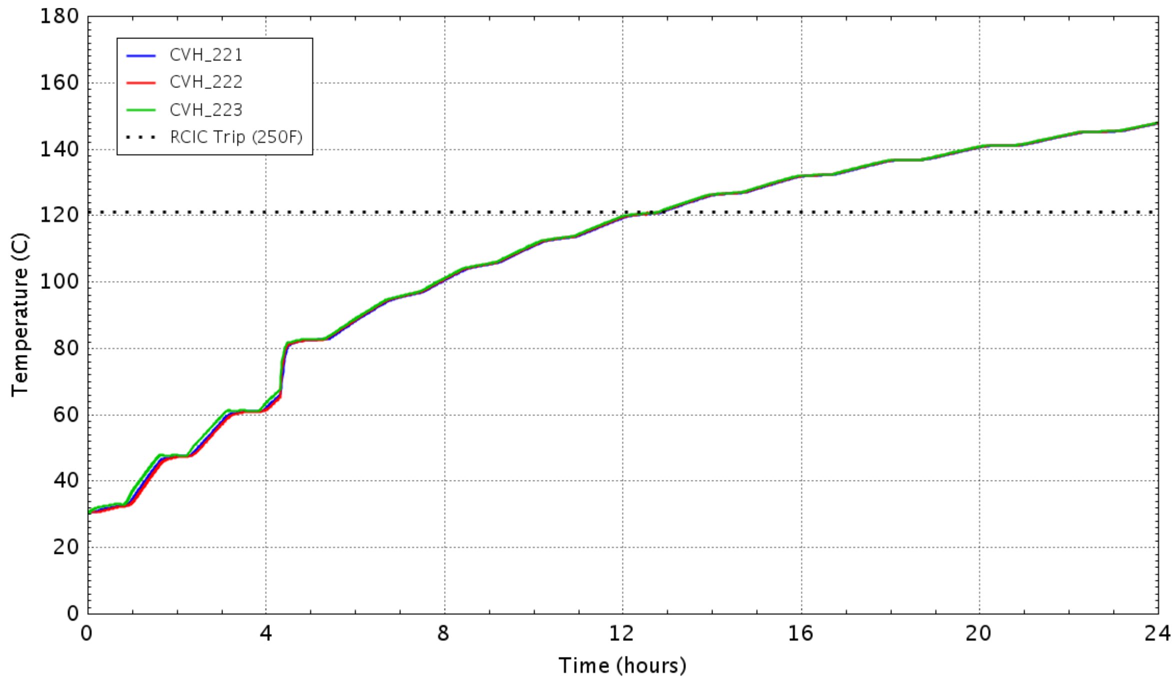

Figure G – 344 Water temperature in the wetwell

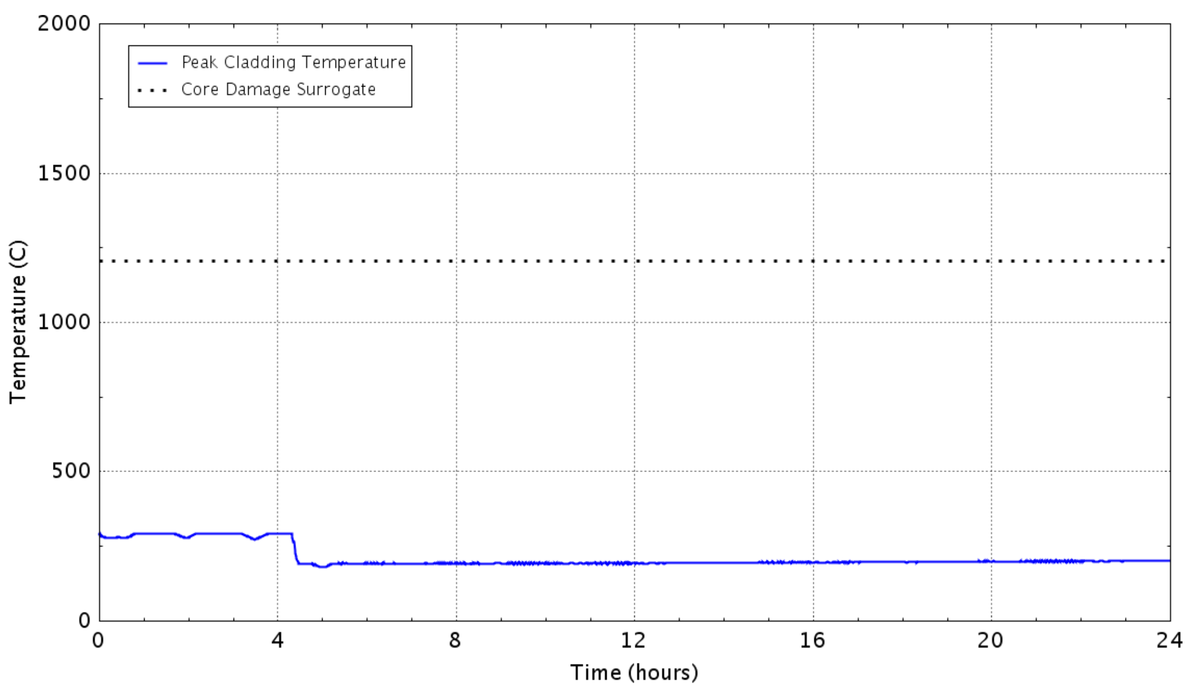

Figure G – 345 Peak temperature of the fuel cladding as a function of time

G.3.14 Case 19c: Sensitivity to LOMFW-25 Case 19 with MSIVClosure at the Start of the Transient

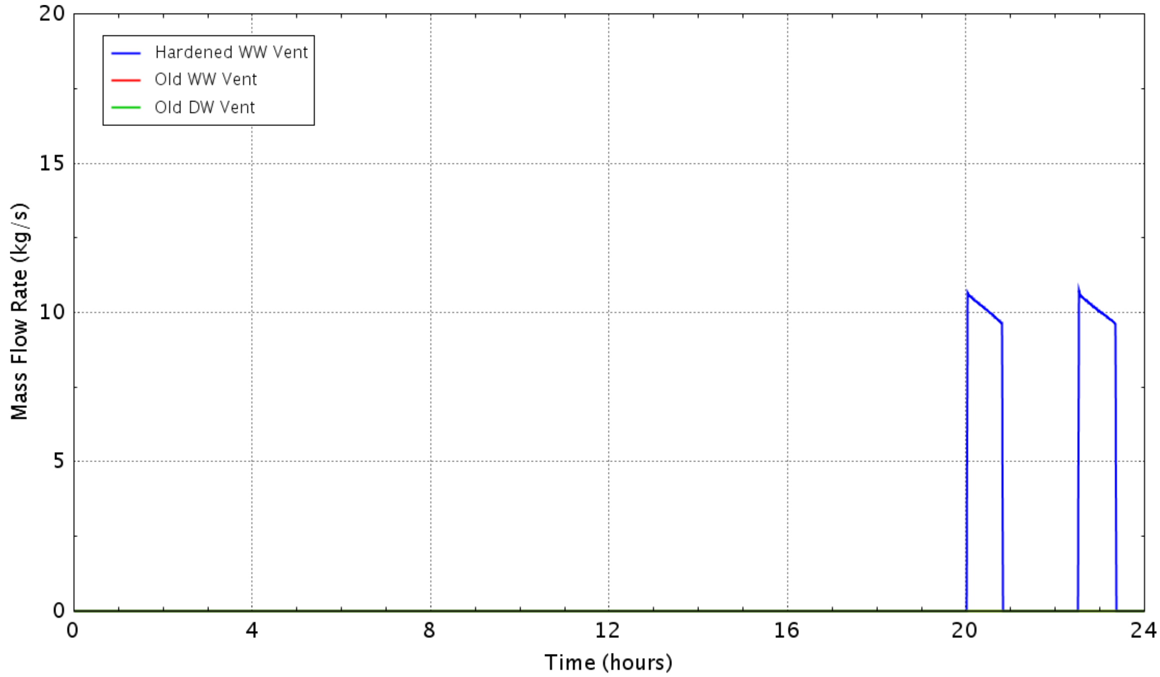

Figure G – 346 Flow rate of the containment vents

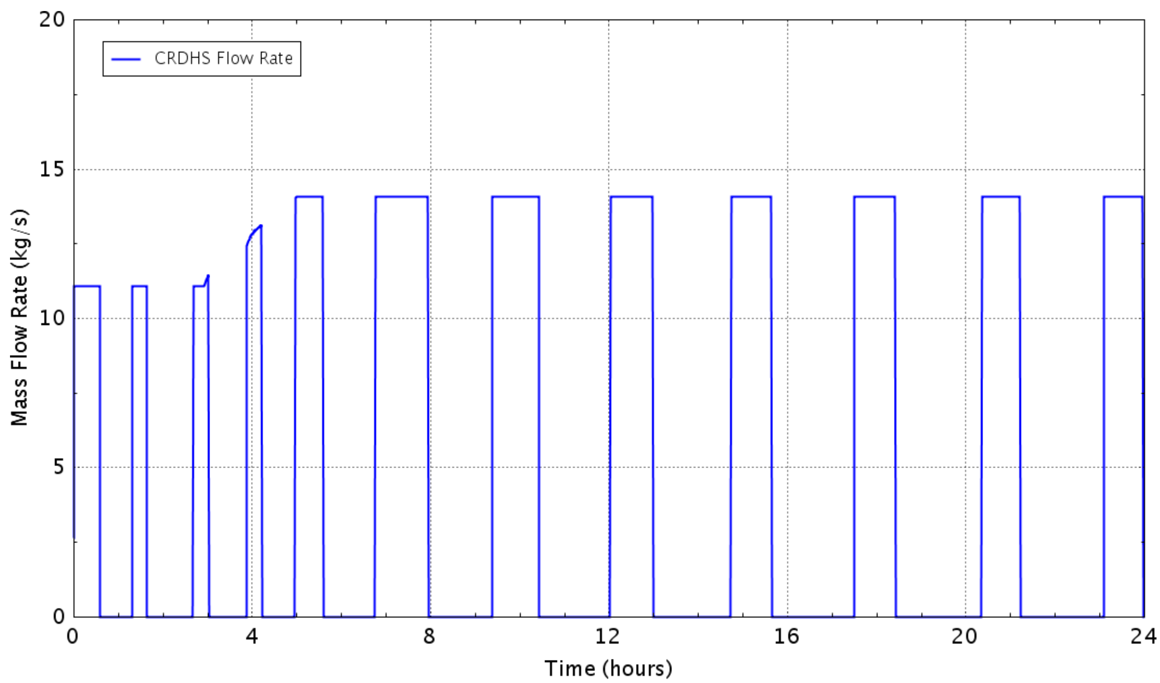

Figure G – 347 Flow rate of the control rod drive hydraulic system

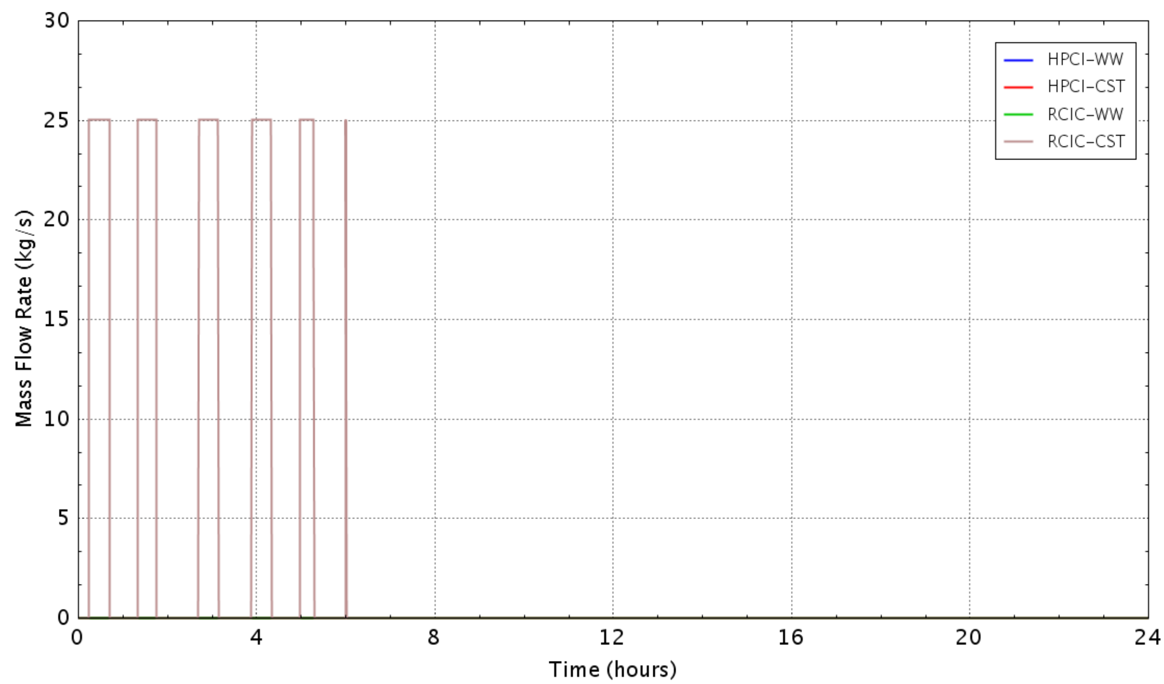

Figure G – 348 Flow rate of the HPCI/RCIC pumps

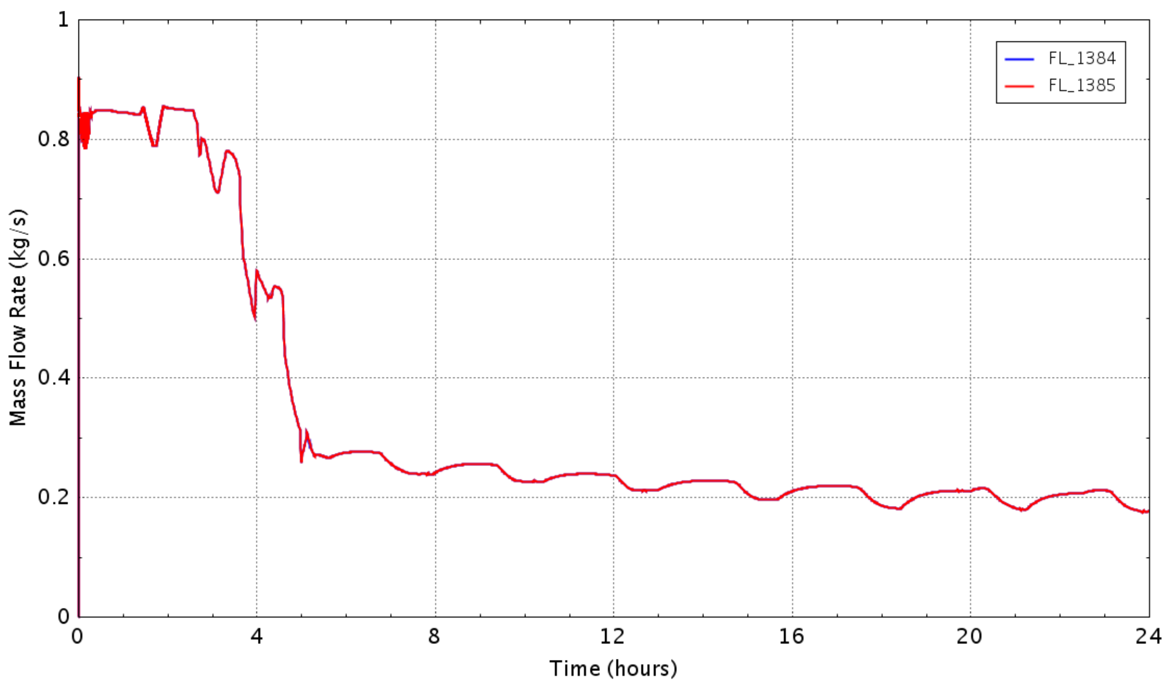

Figure G – 349 Flow rate of the recirculating pump seal leakage

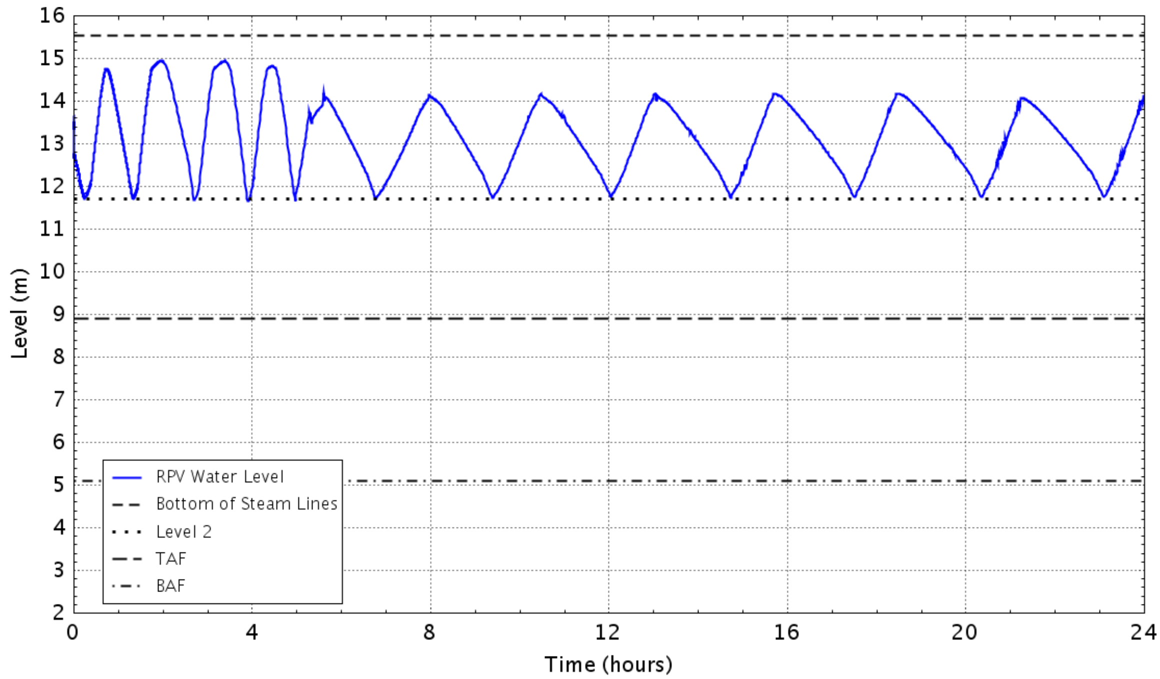

Figure G – 350 RPV Downcomer water level

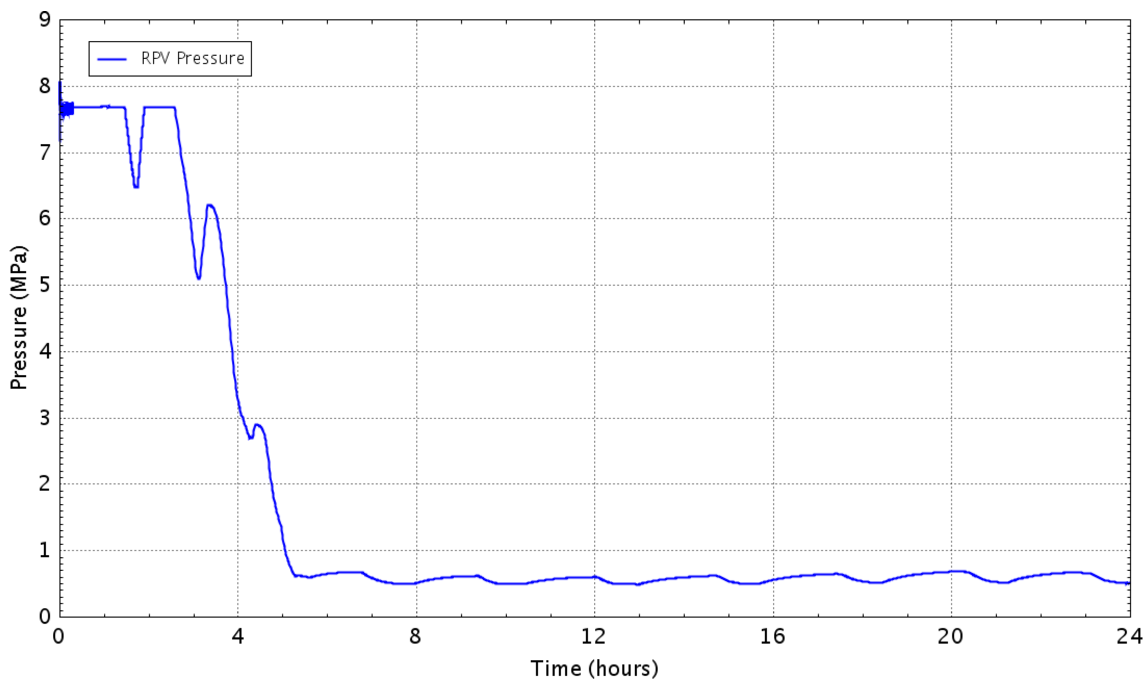

Figure G – 351 Pressure in the RPV

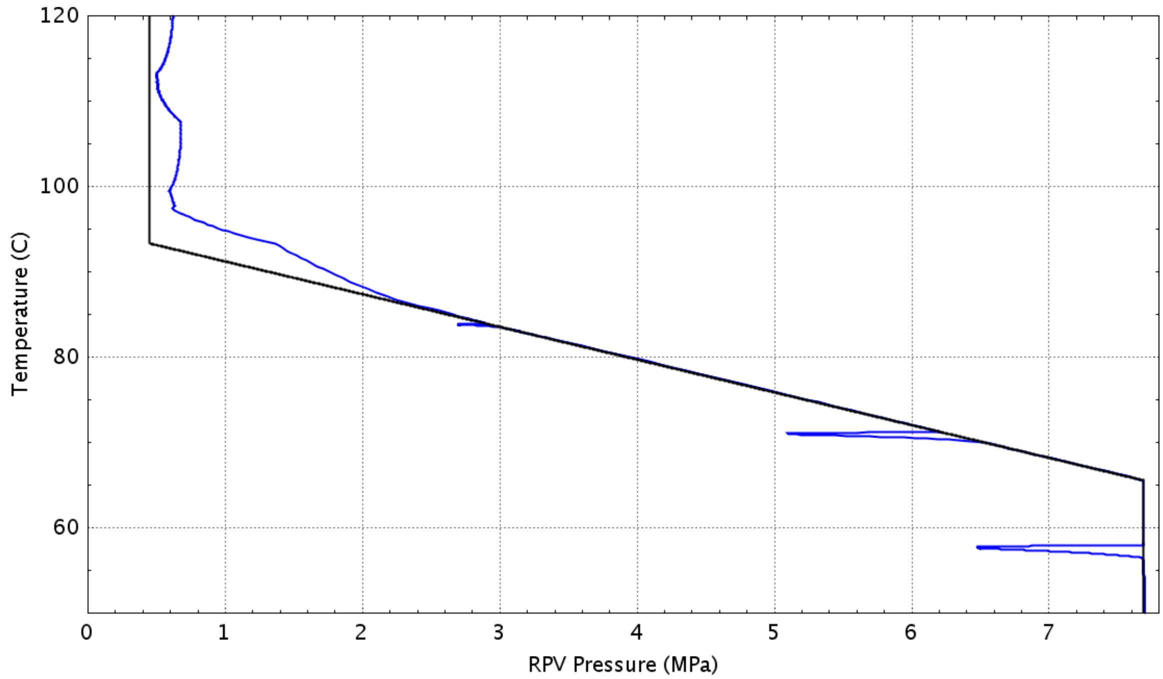

Figure G – 352 Plant status relative to the HCL curve (Graph 4 of the EOPs)

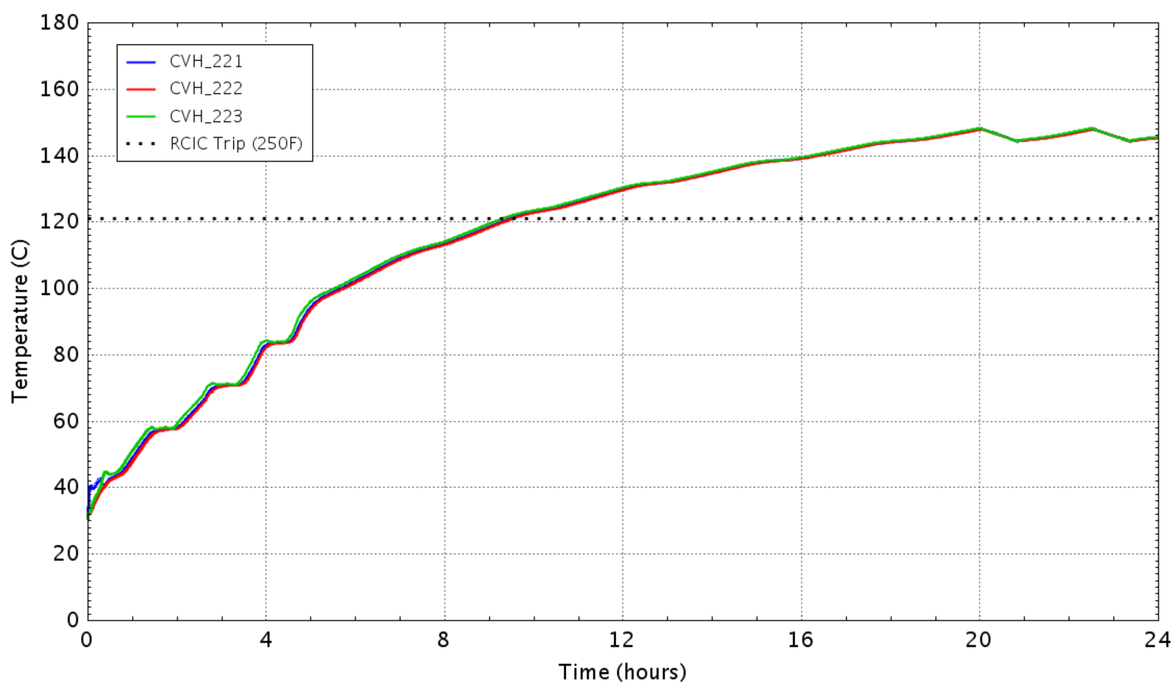

Figure G – 353 Water temperature in the wetwell

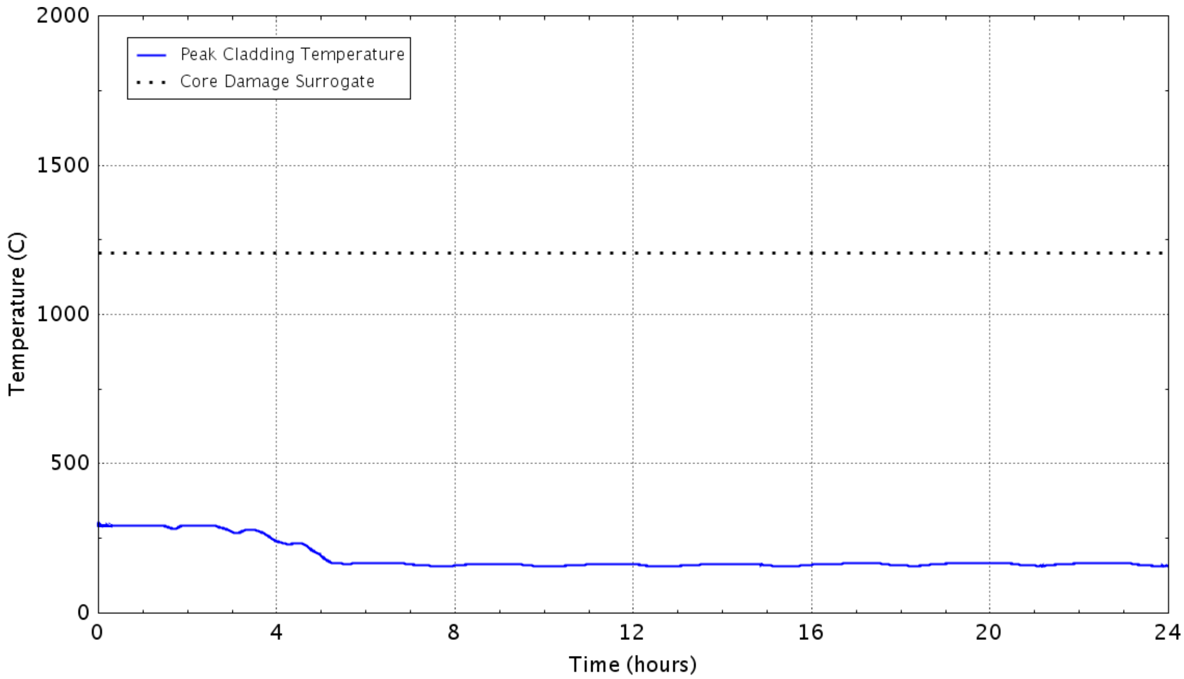

Figure G – 354 Peak temperature of the fuel cladding as a function of time

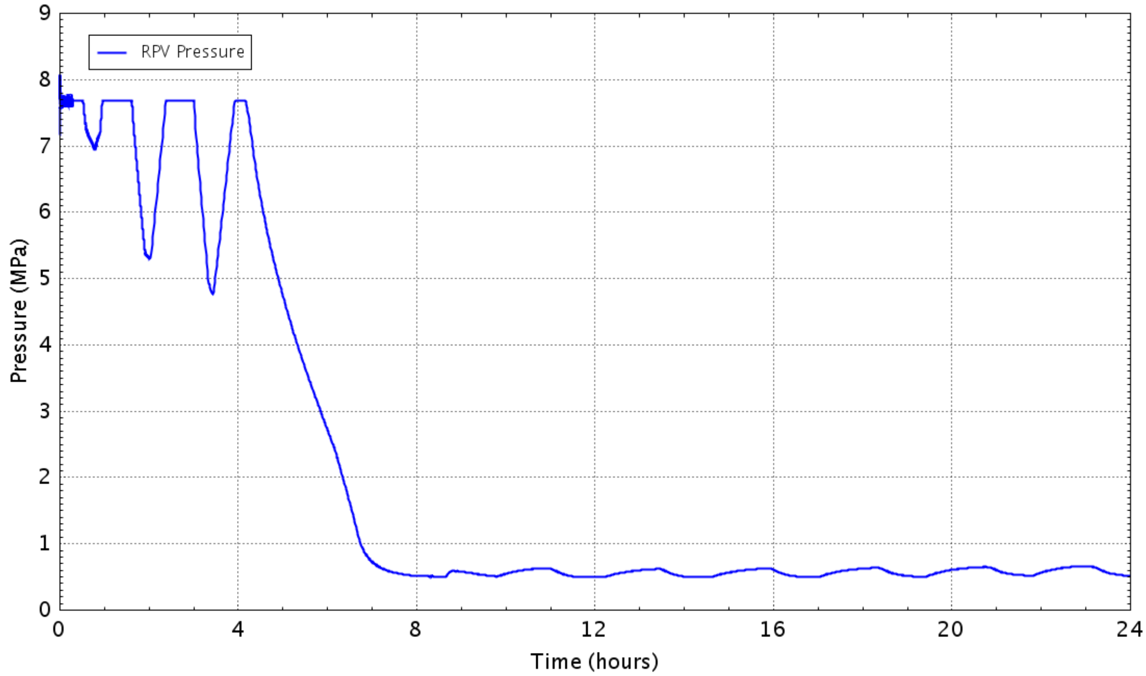

Figure G – 306 Pressure in the RPV

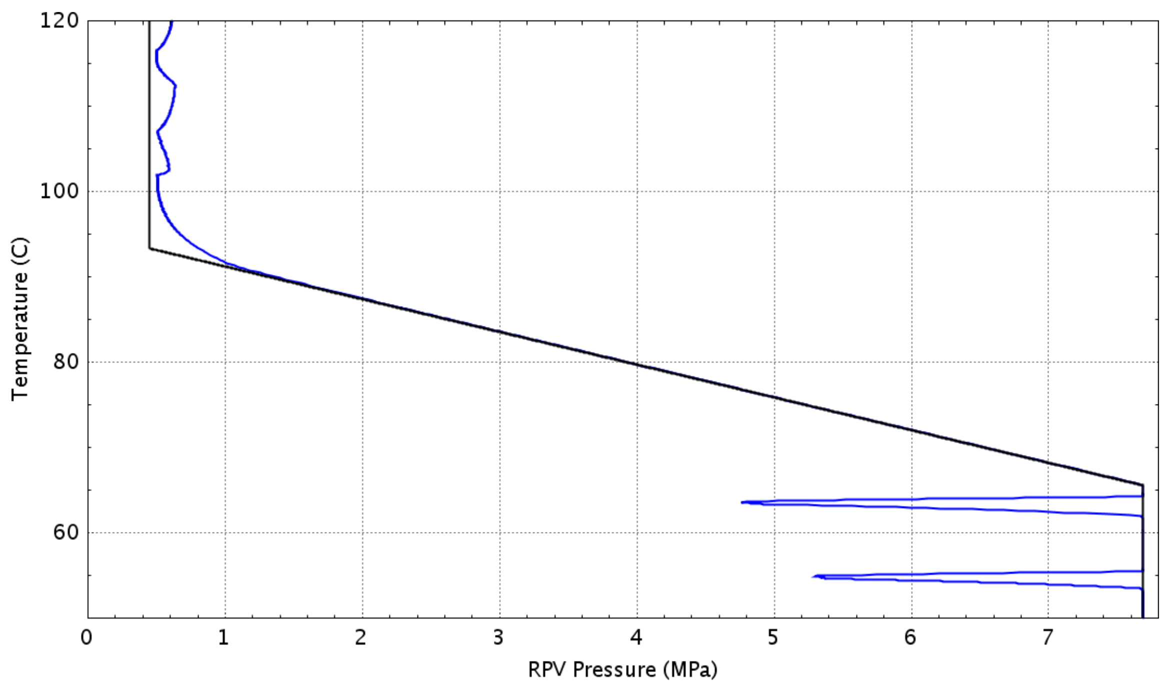

Figure G – 307 Plant status relative to the HCL curve (Graph 4 of the EOPs)

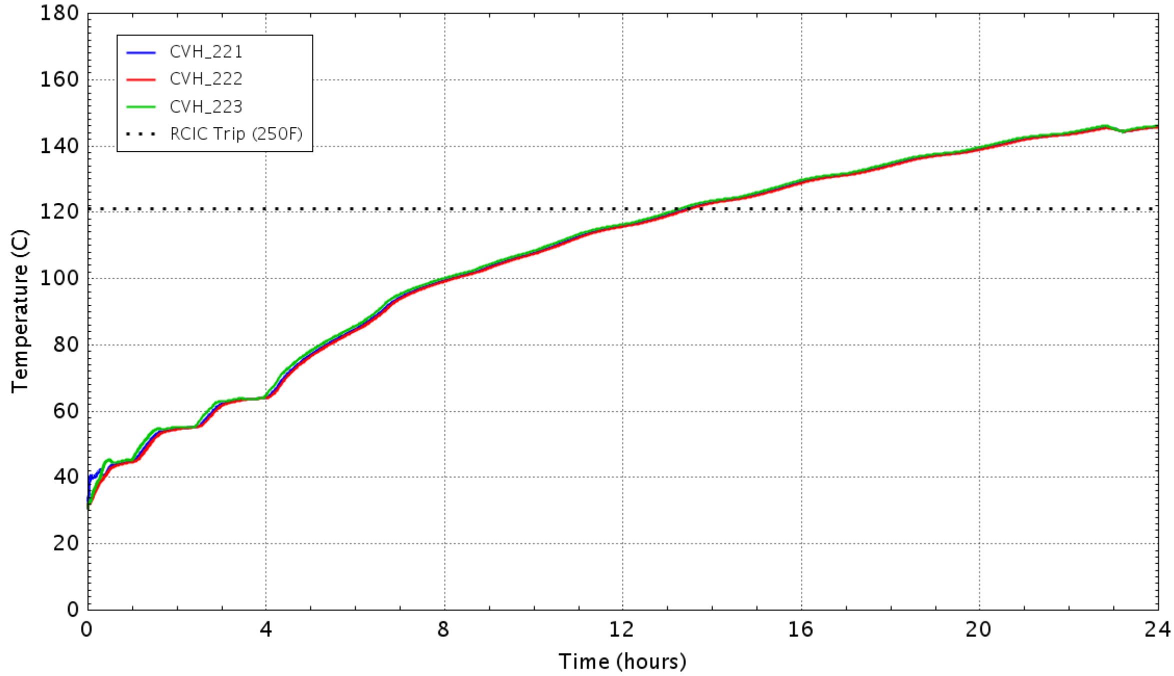

Figure G – 308 Water temperature in the wetwell

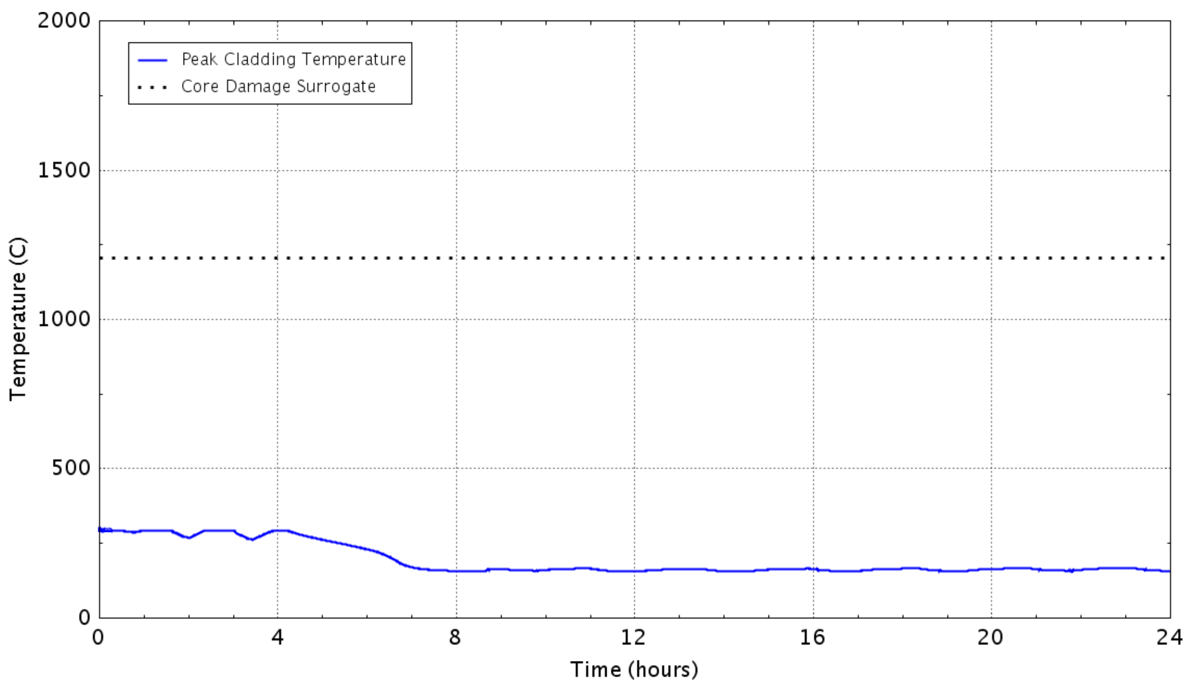

Figure G – 309 Peak temperature of the fuel cladding as a function of time

G.3.10 Case 17a: Sensitivity to LOMFW-25 Case 17 with HPCI Available Instead of RCIC

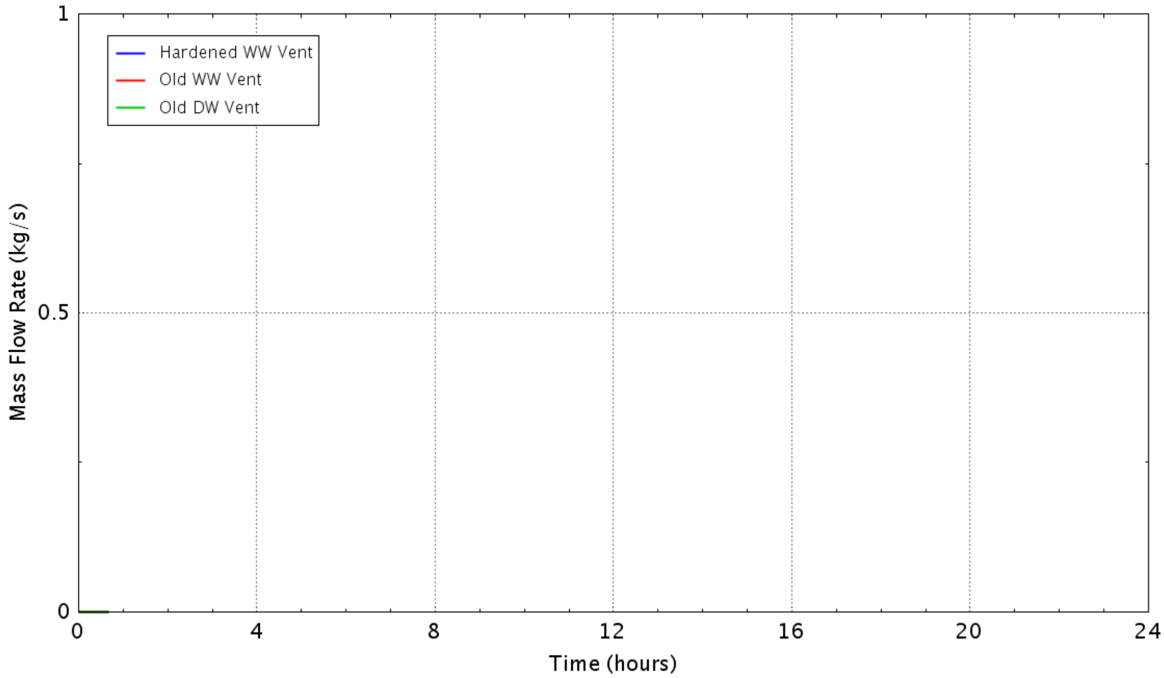

Figure G – 310 Flow rate of the containment vents

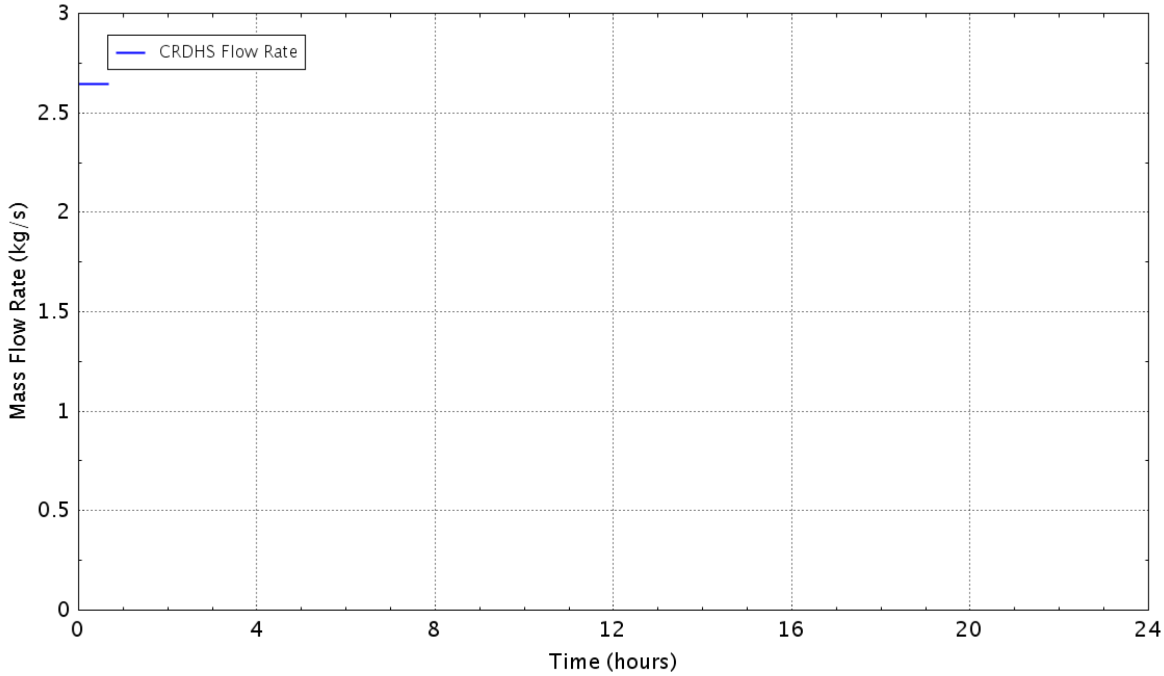

Figure G – 311 Flow rate of the control rod drive hydraulic system

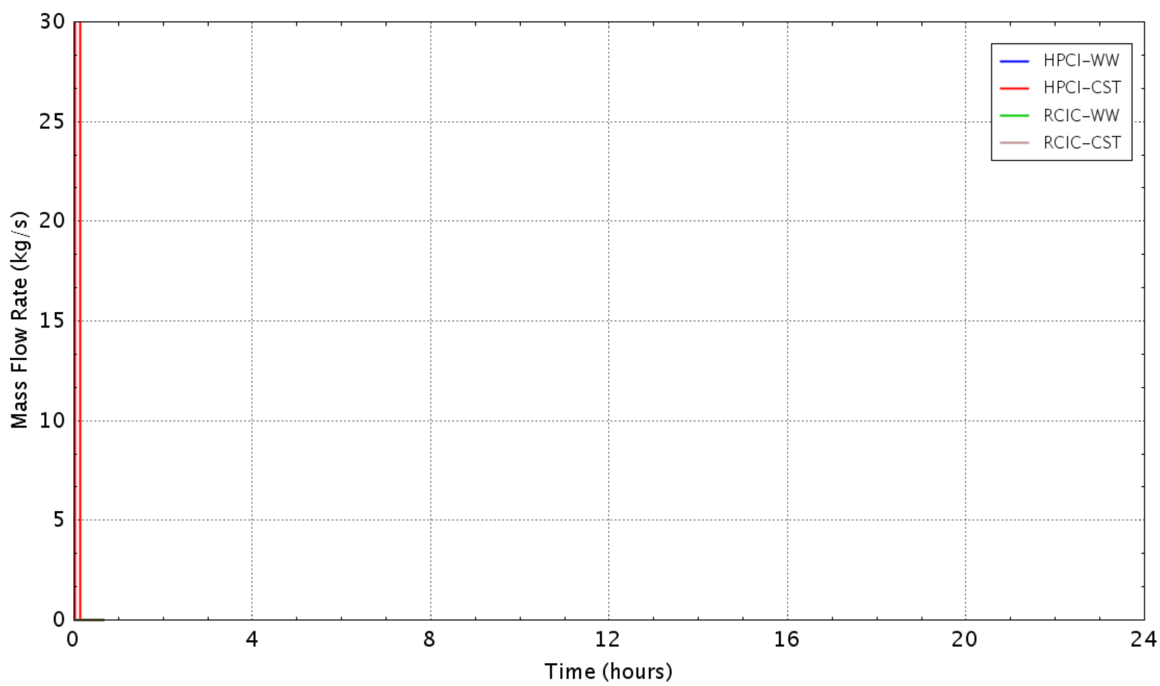

Figure G – 312 Flow rate of the HPCI/RCIC pumps

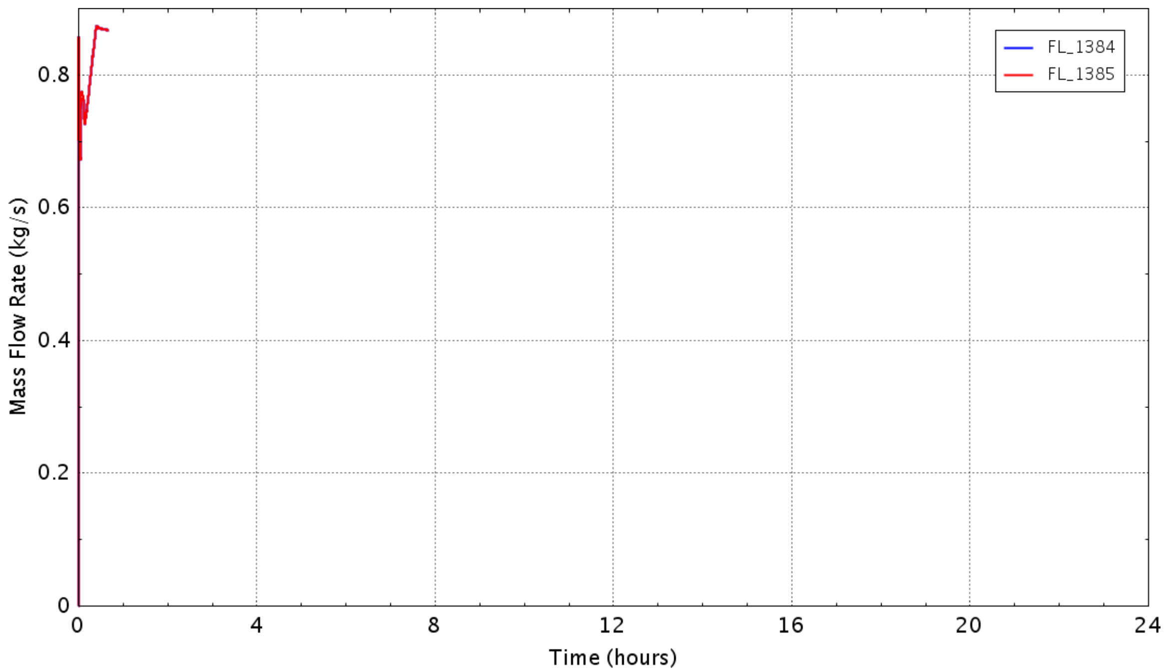

Figure G – 313 Flow rate of the recirculating pump seal leakage

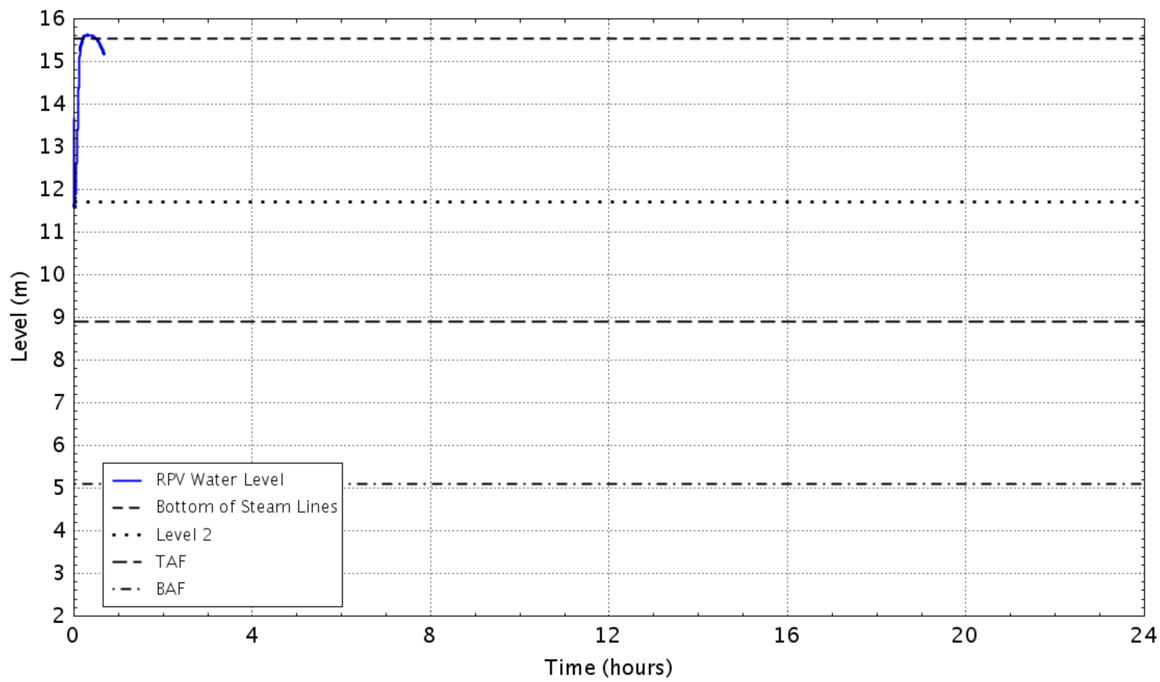

Figure G – 314 RPV Downcomer water level

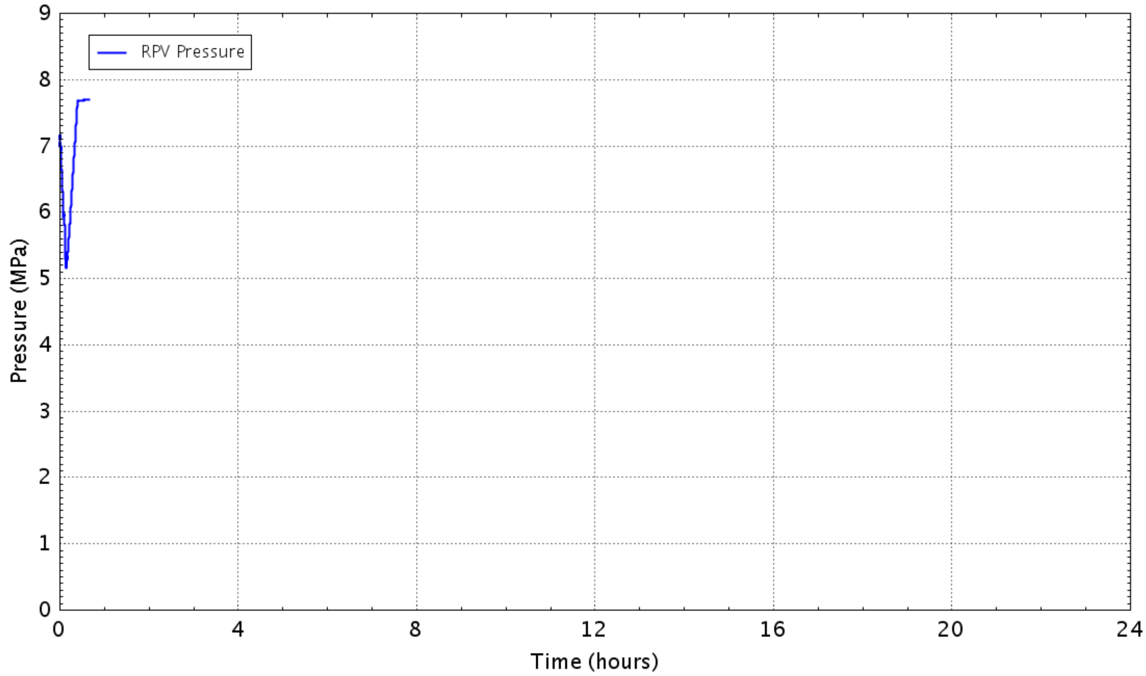

Figure G – 315 Pressure in the RPV

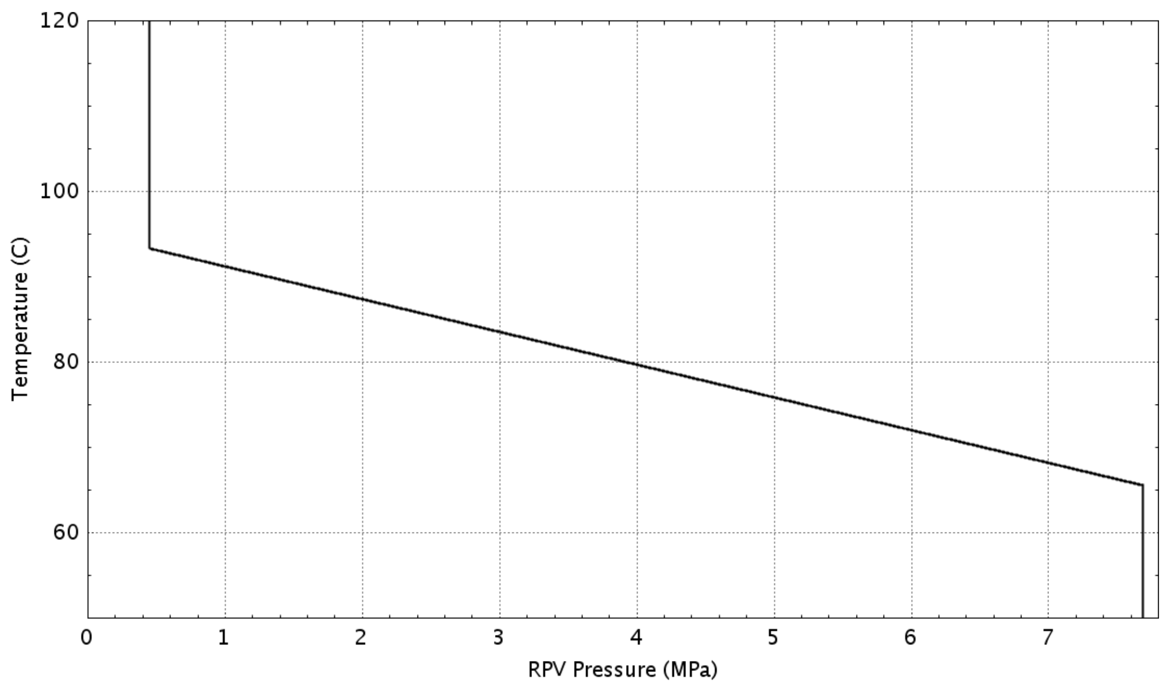

Figure G – 316 Plant status relative to the HCL curve (Graph 4 of the EOPs)

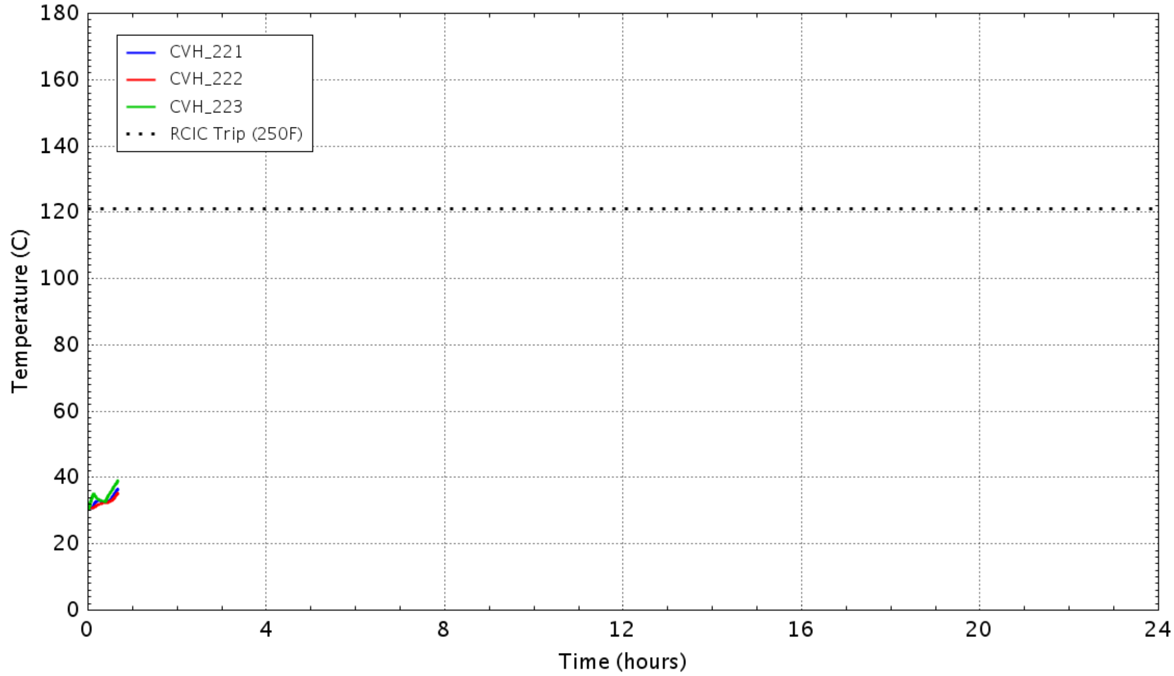

Figure G – 317 Water temperature in the wetwell

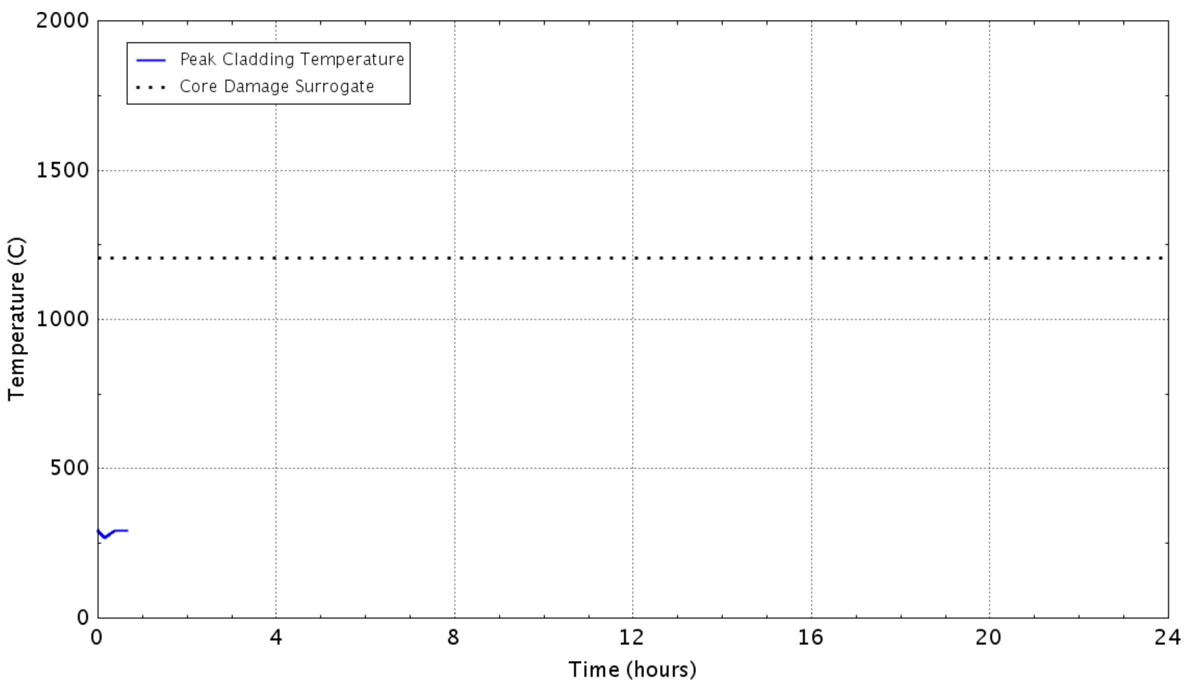

Figure G – 318 Peak temperature of the fuel cladding as a function of time

G.3.11 Case 17b: Sensitivity to LOMFW-25 Case 17 with an SRV Failing Open at270 Cycles

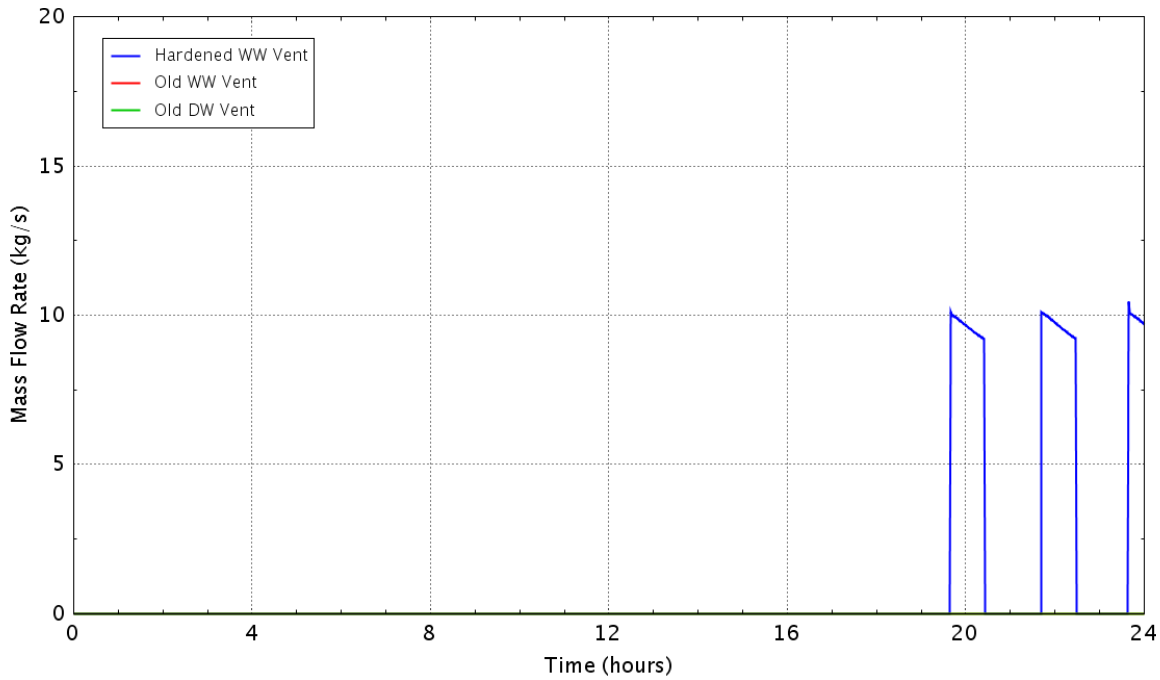

Figure G – 319 Flow rate of the containment vents

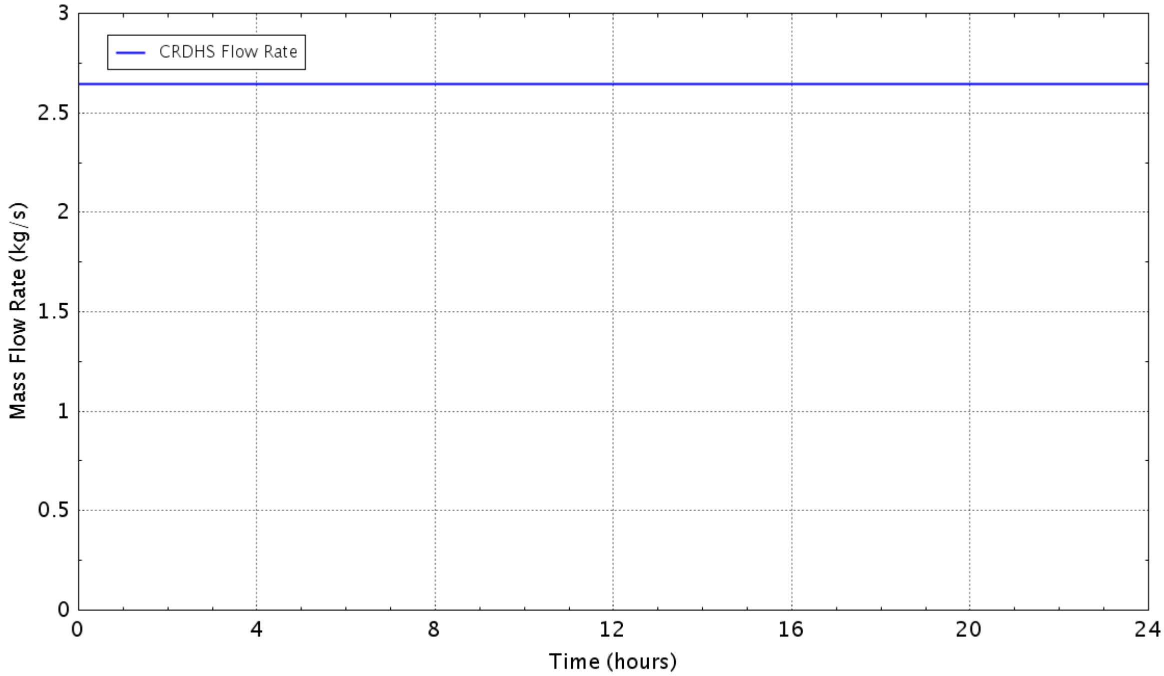

Figure G – 320 Flow rate of the control rod drive hydraulic system

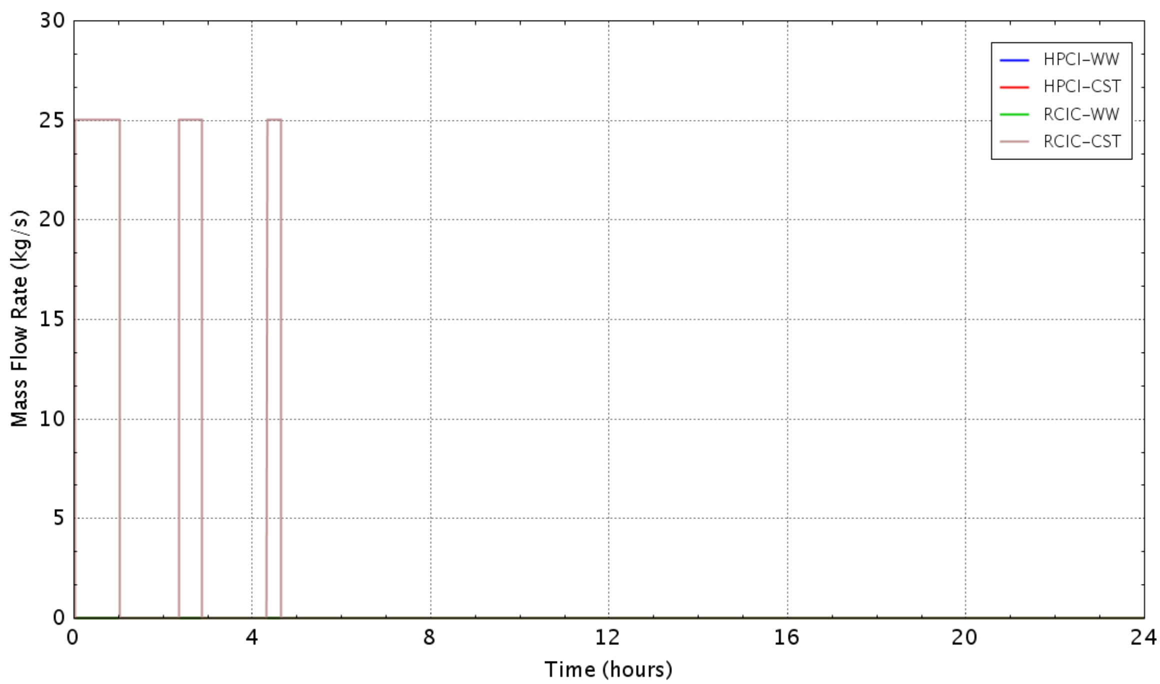

Figure G – 321 Flow rate of the HPCI/RCIC pumps

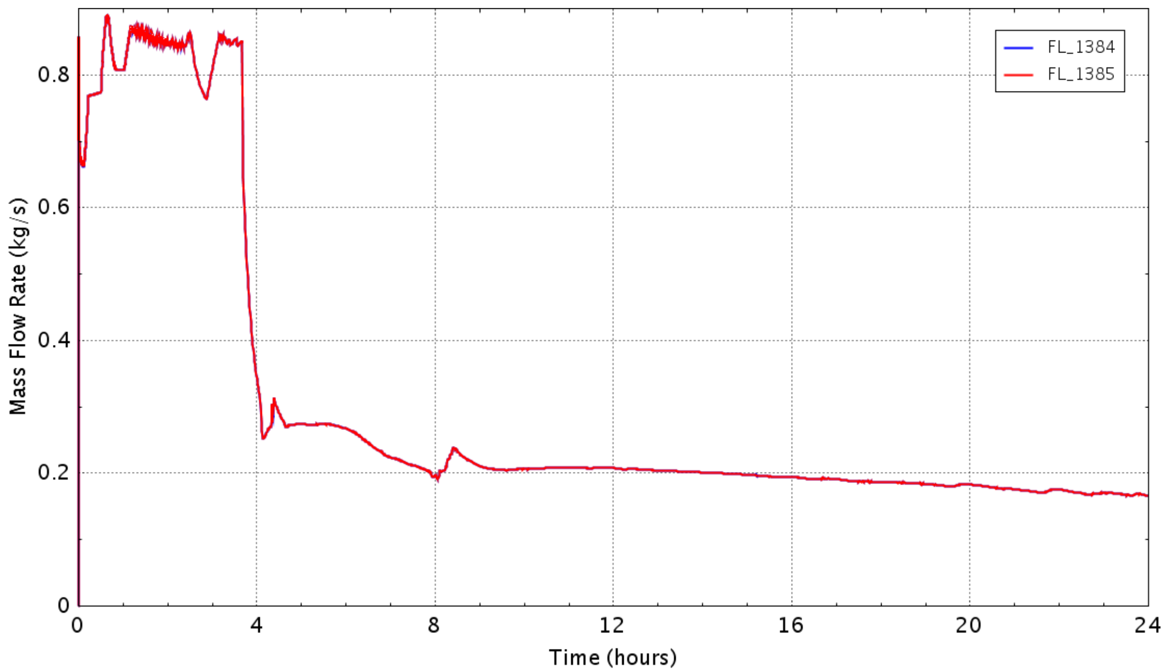

Figure G – 322 Flow rate of the recirculating pump seal leakage

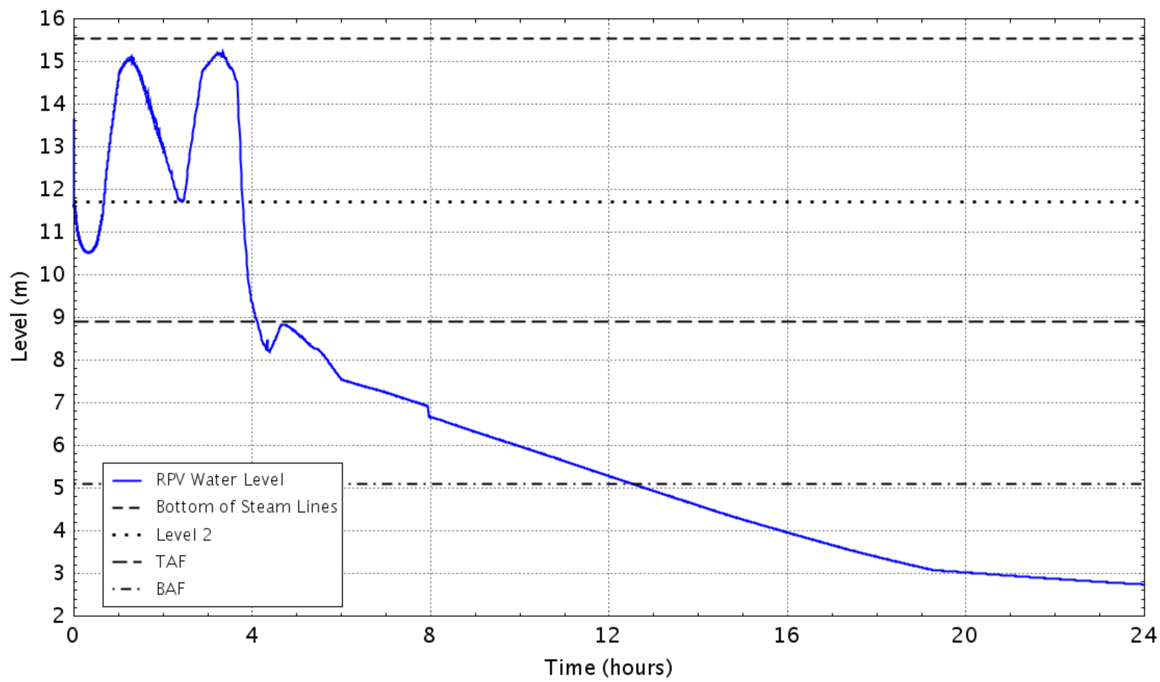

Figure G – 323 RPV Downcomer water level

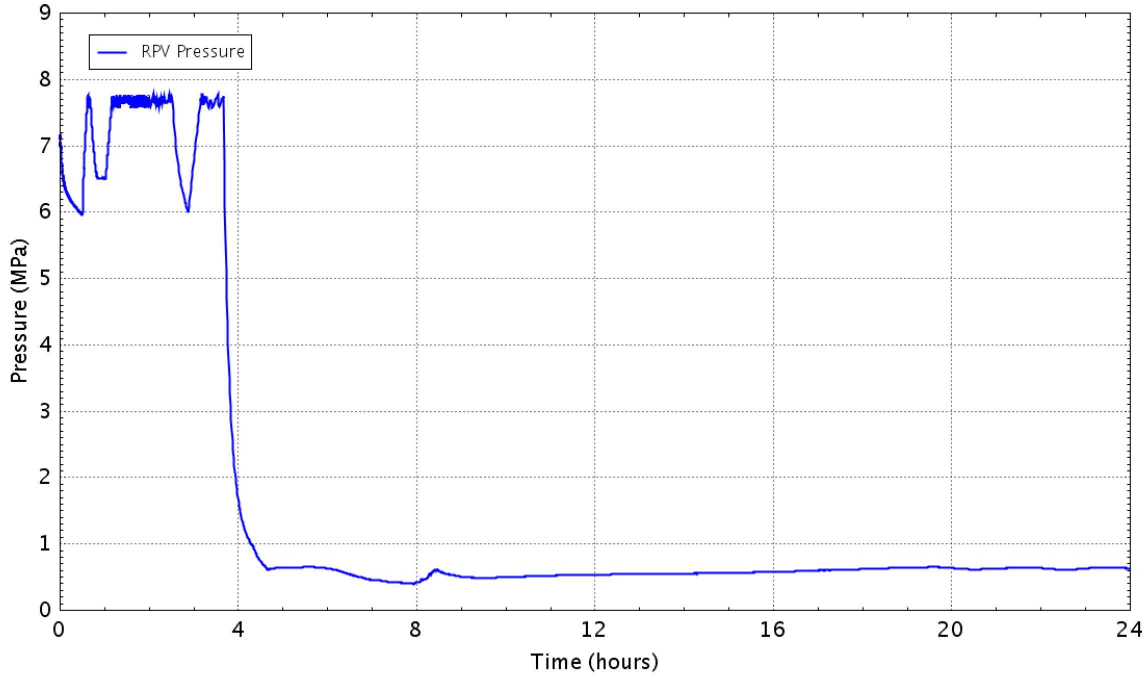

Figure G – 324 Pressure in the RPV

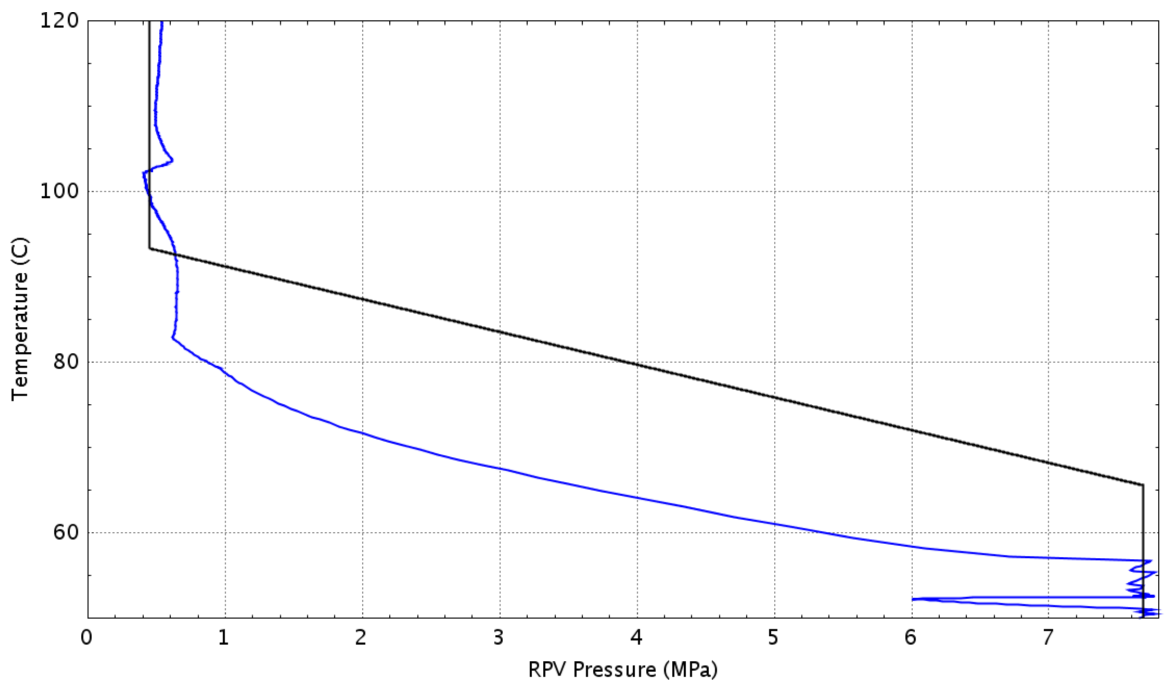

Figure G – 325 Plant status relative to the HCL curve (Graph 4 of the EOPs)

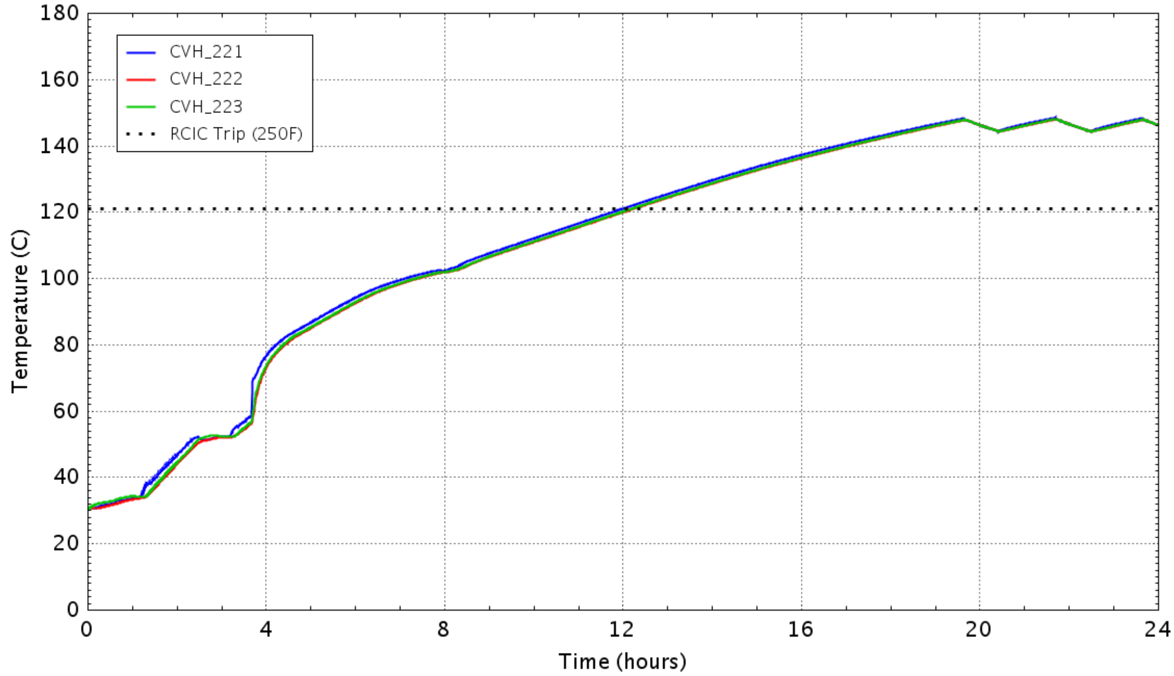

Figure G – 326 Water temperature in the wetwell

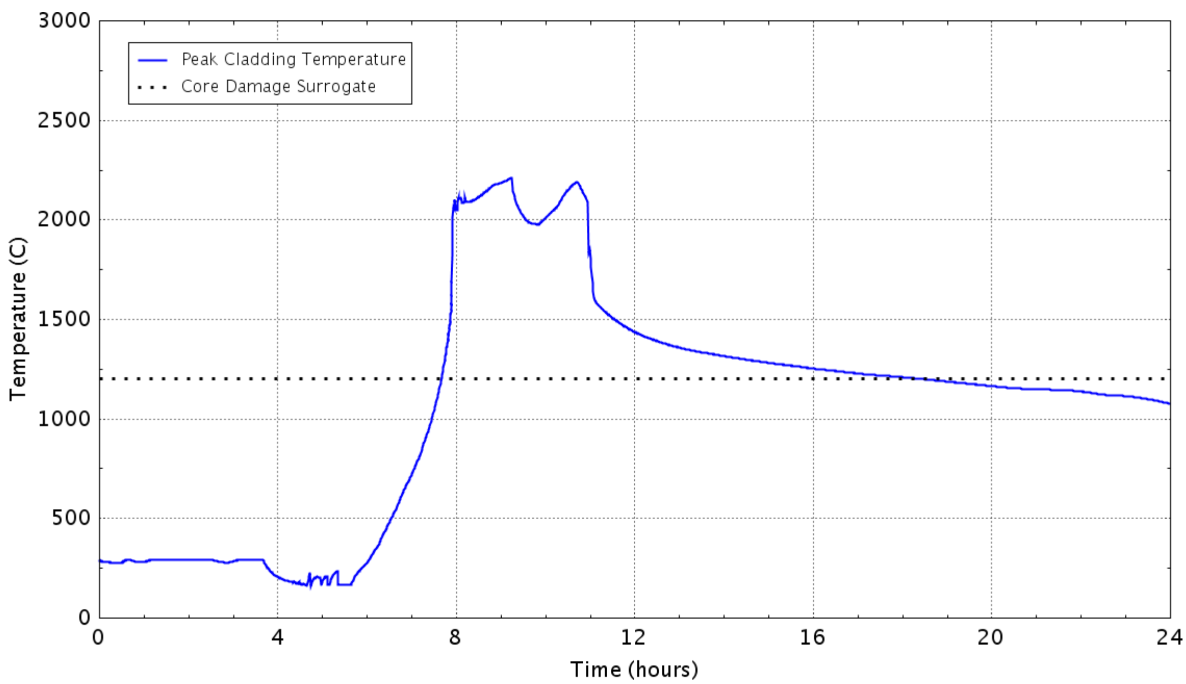

Figure G – 327 Peak temperature of the fuel cladding as a function of time

G.3.12 Case 19a: Sensitivity to LOMFW-25 Case 19 with RCIC Injection from the Wetwell

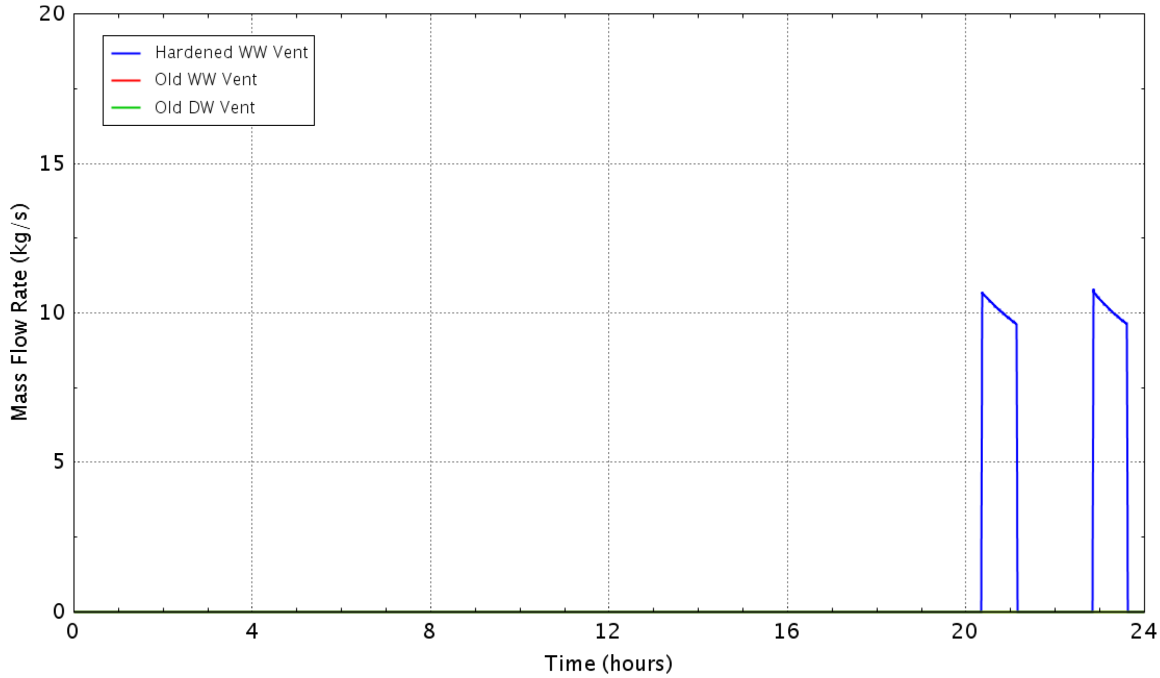

Figure G – 328 Flow rate of the containment vents

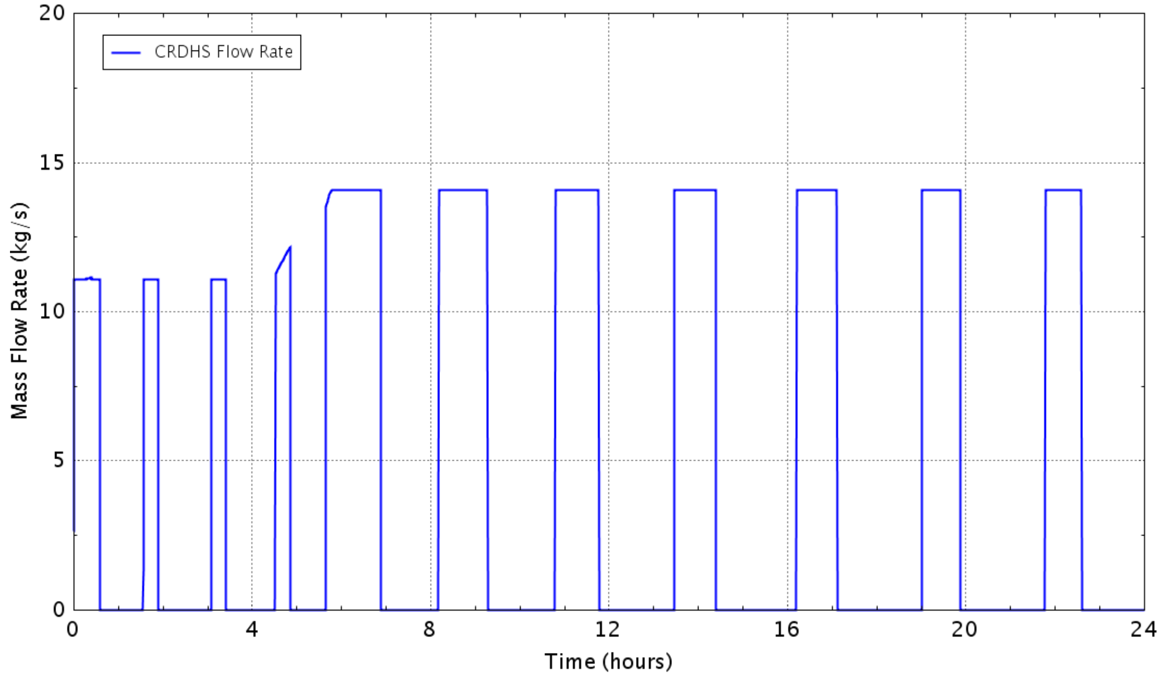

Figure G – 329 Flow rate of the control rod drive hydraulic system

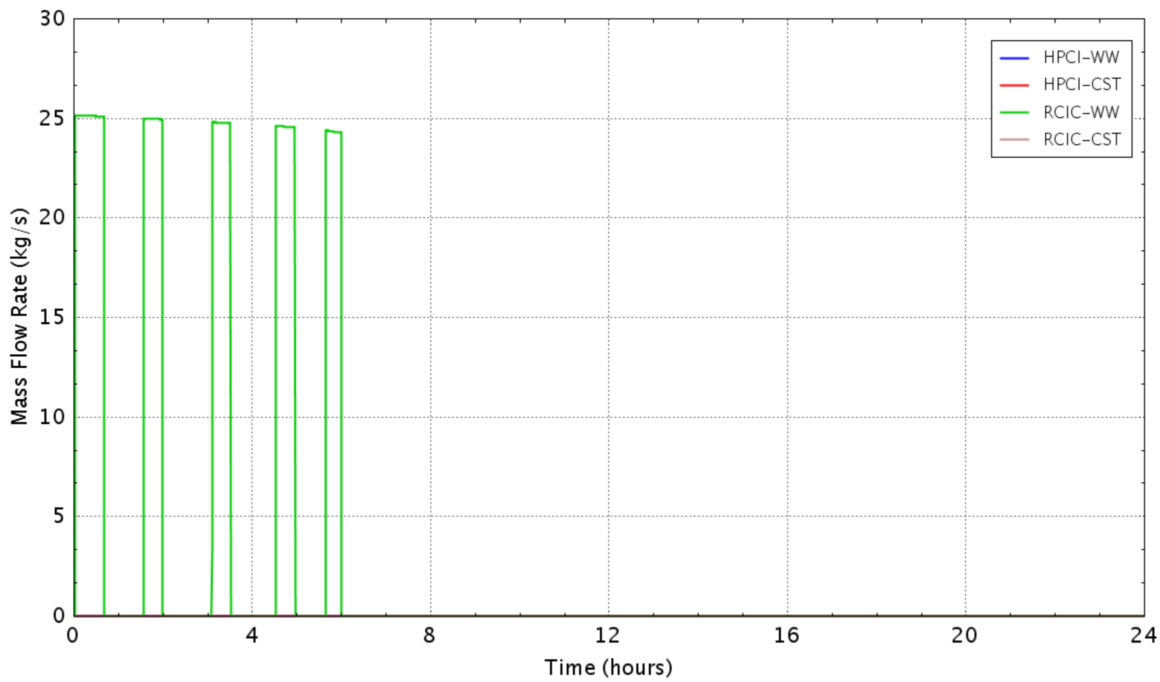

Figure G – 330 Flow rate of the HPCI/RCIC pumps

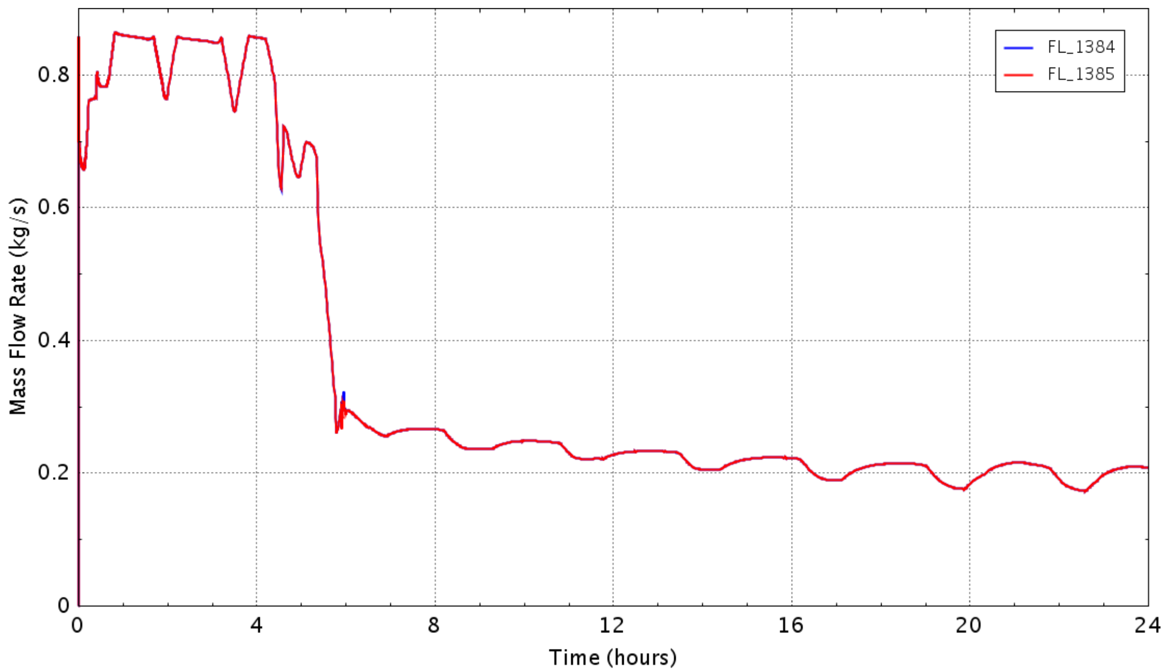

Figure G – 331 Flow rate of the recirculating pump seal leakage

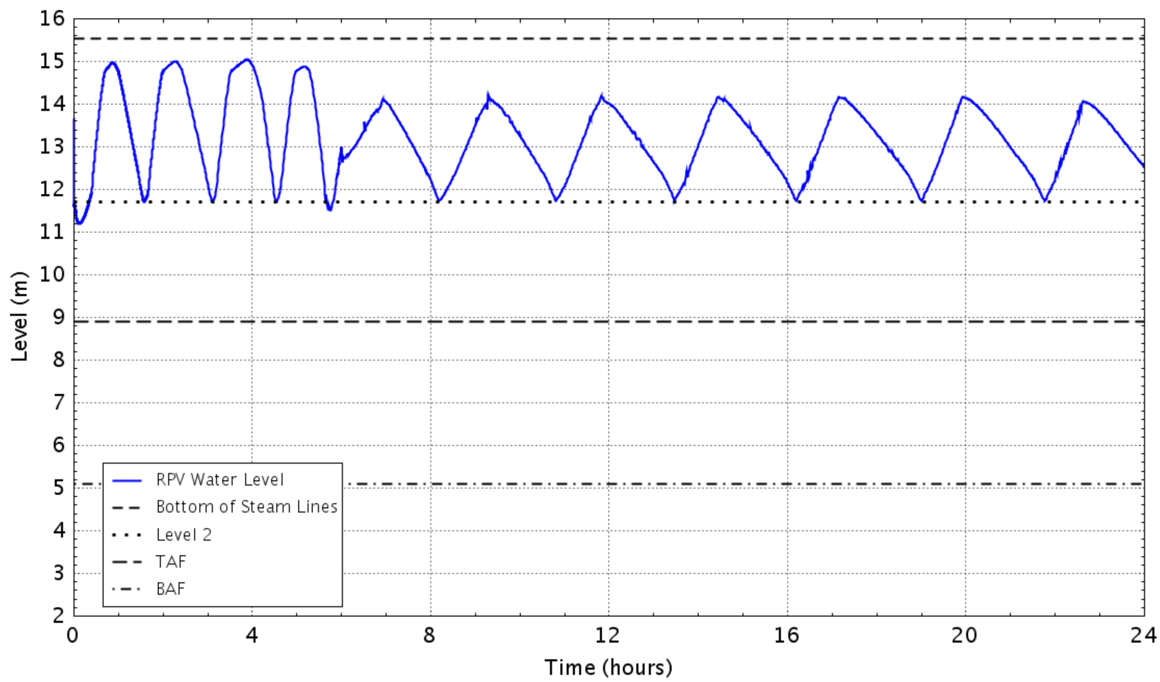

Figure G – 332 RPV Downcomer water level

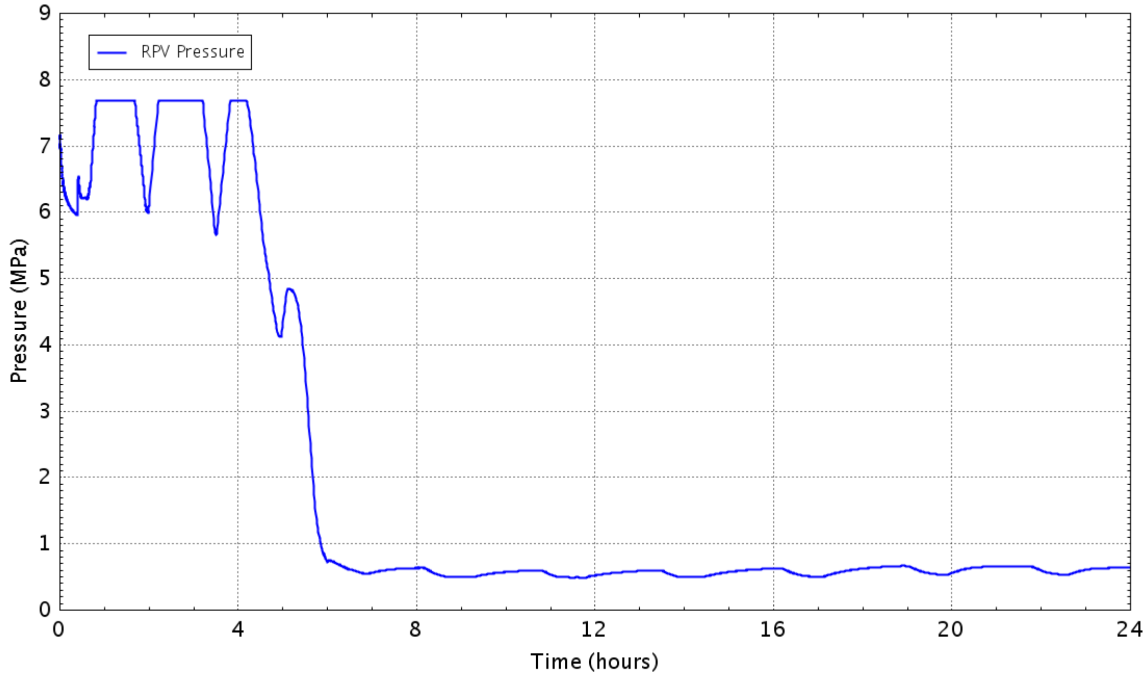

Figure G – 333 Pressure in the RPV

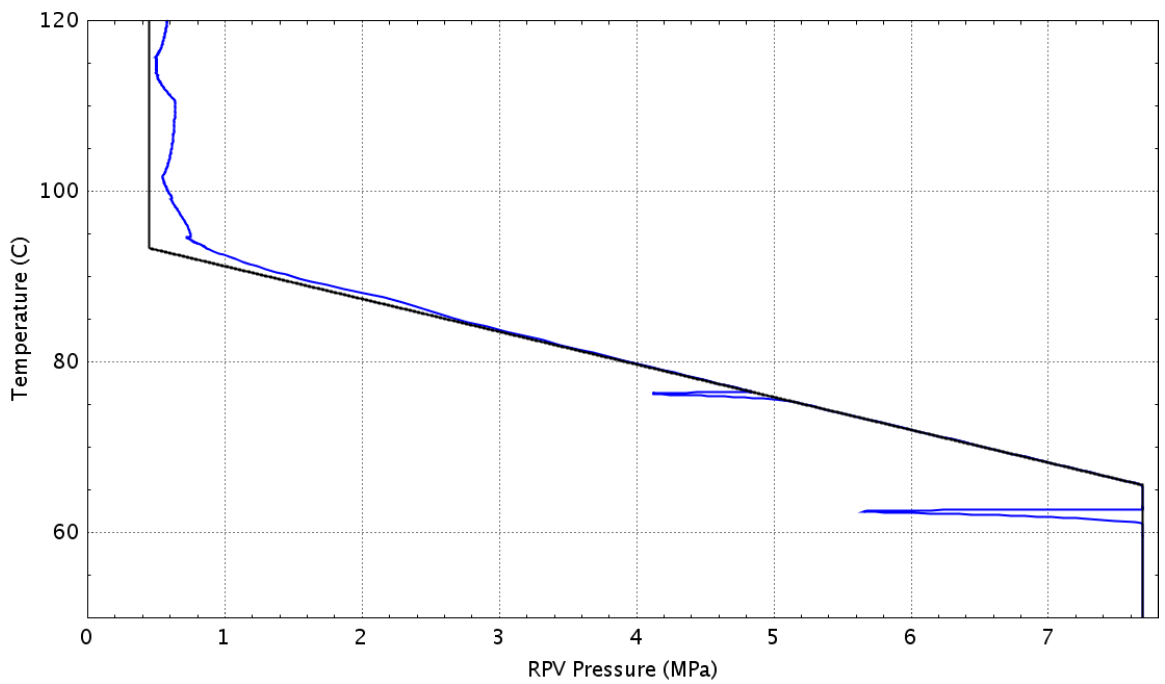

Figure G – 334 Plant status relative to the HCL curve (Graph 4 of the EOPs)

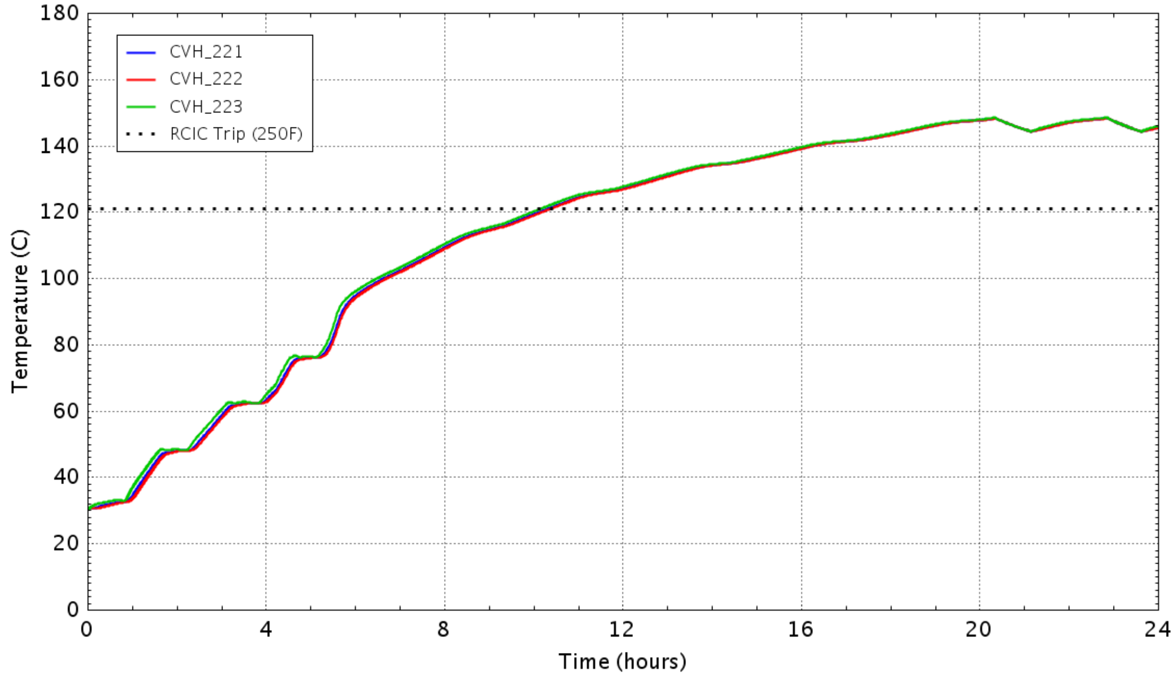

Figure G – 335 Water temperature in the wetwell

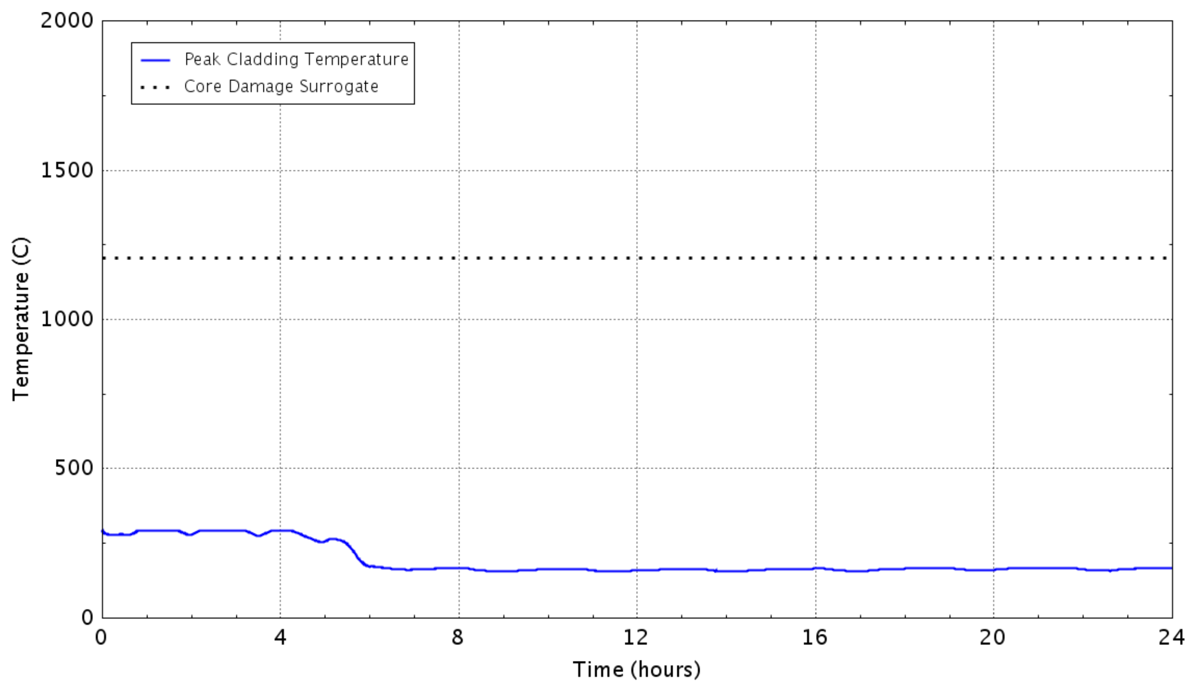

Figure G - 336 Peak temperature of the fuel cladding as a function of time

G.3.13 Case 19b: Sensitivity to LOMFW-25 Case 19 with RPV Emergency Depressurization when the HCL Curve is Reached

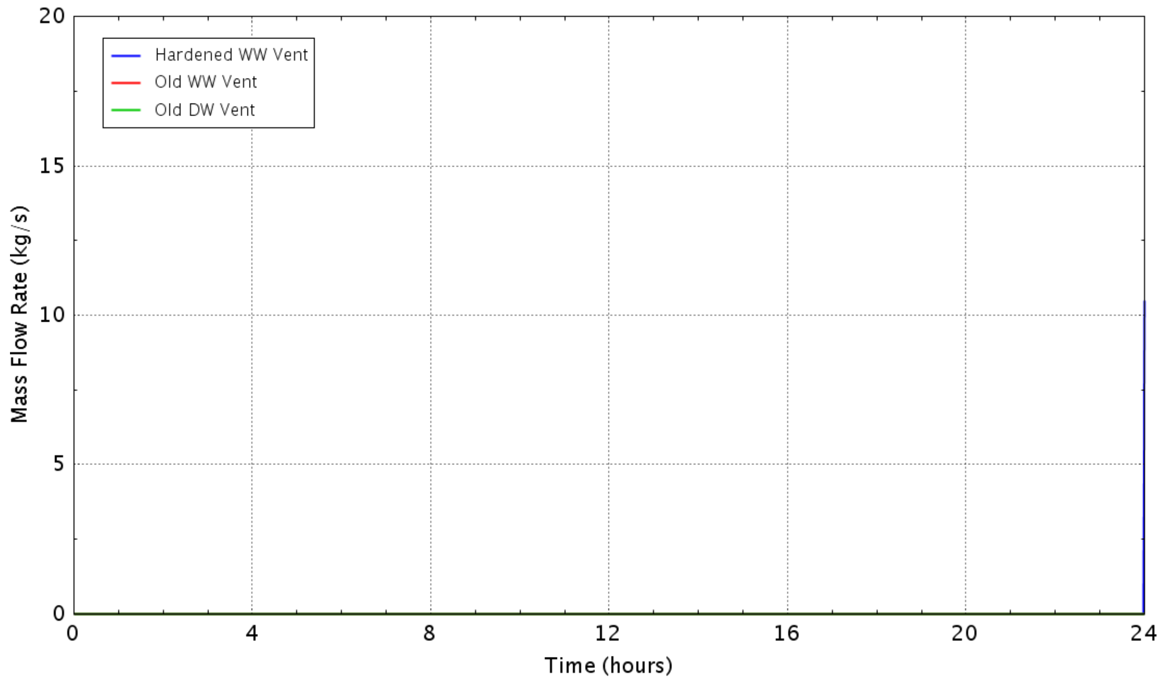

Figure G – 337 Flow rate of the containment vents

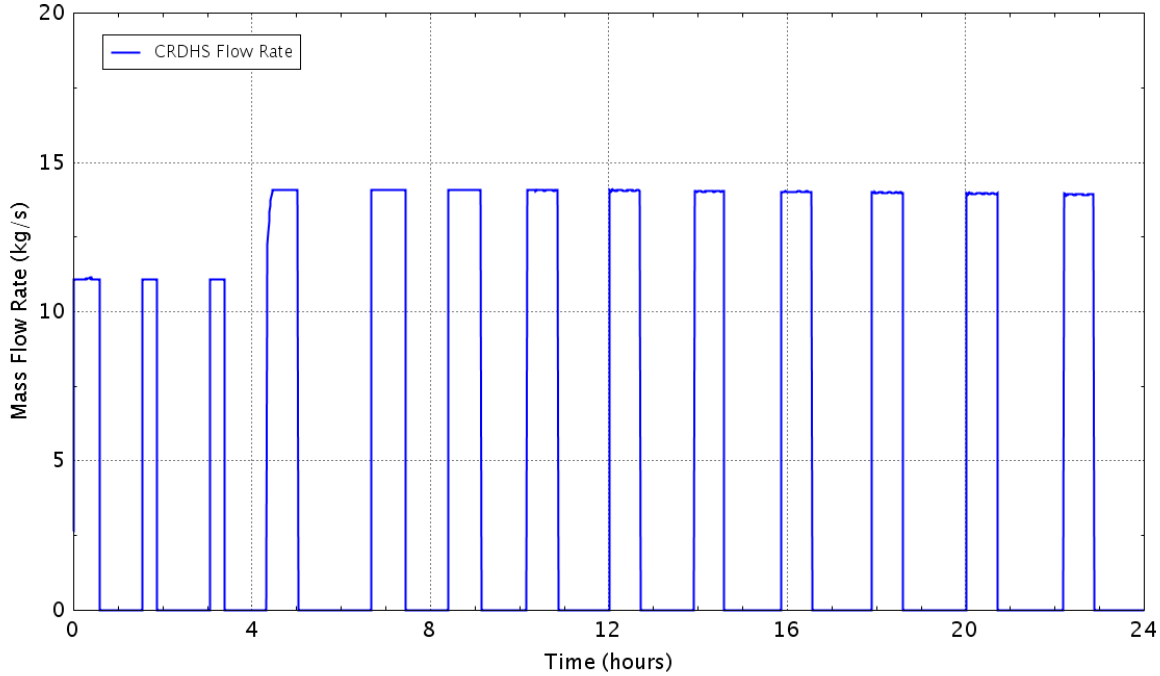

Figure G – 338 Flow rate of the control rod drive hydraulic system

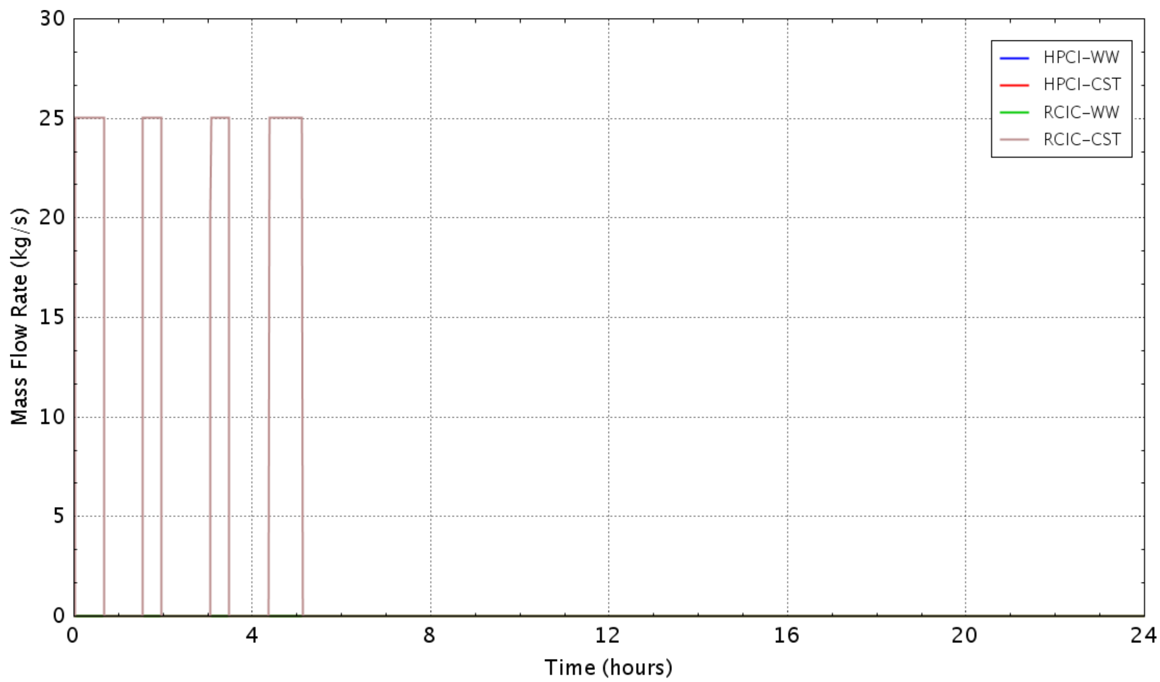

Figure G – 339 Flow rate of the HPCI/RCIC pumps

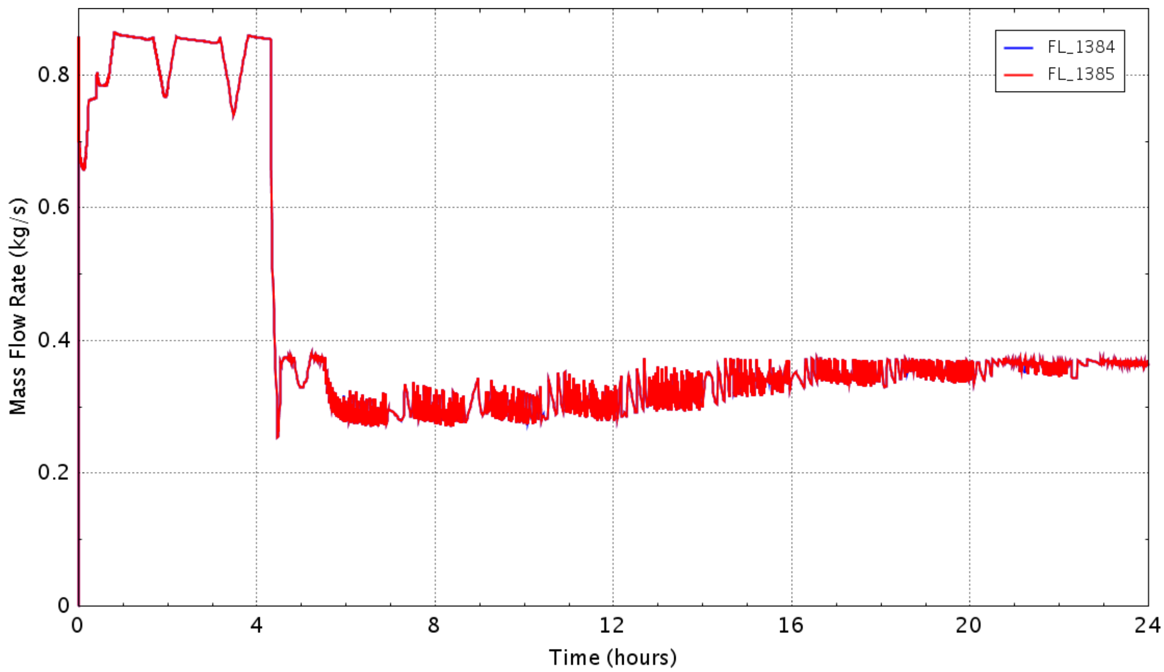

Figure G – 340 Flow rate of the recirculating pump seal leakage

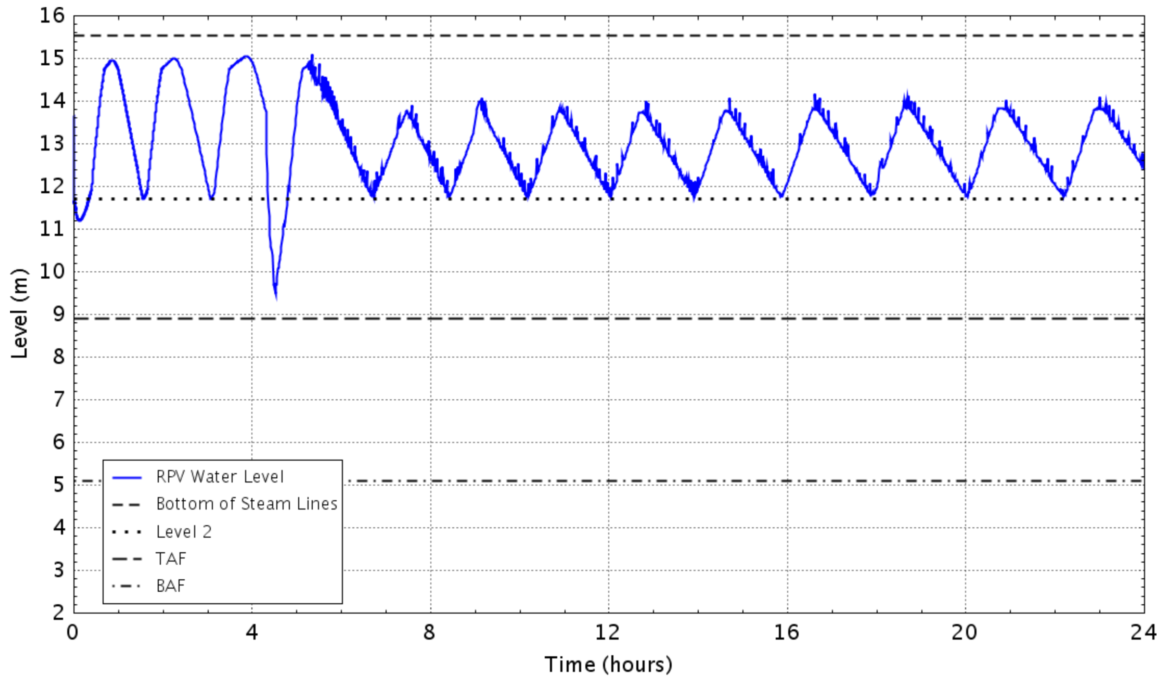

Figure G – 341 RPV Downcomer water level

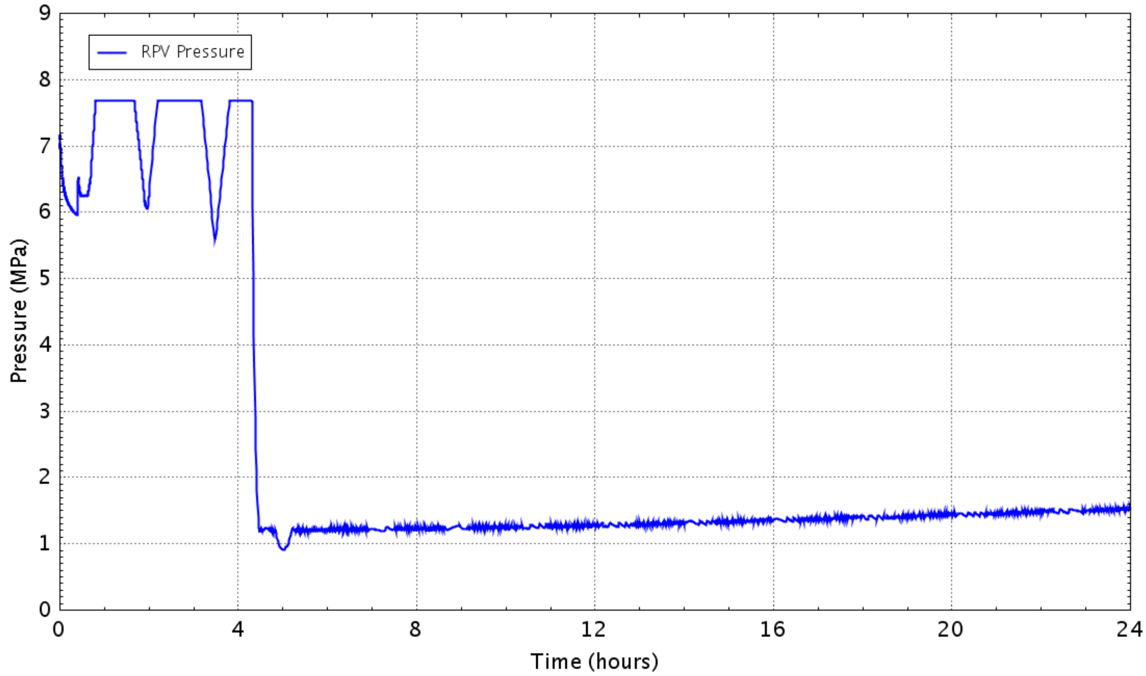

Figure G – 342 Pressure in the RPV

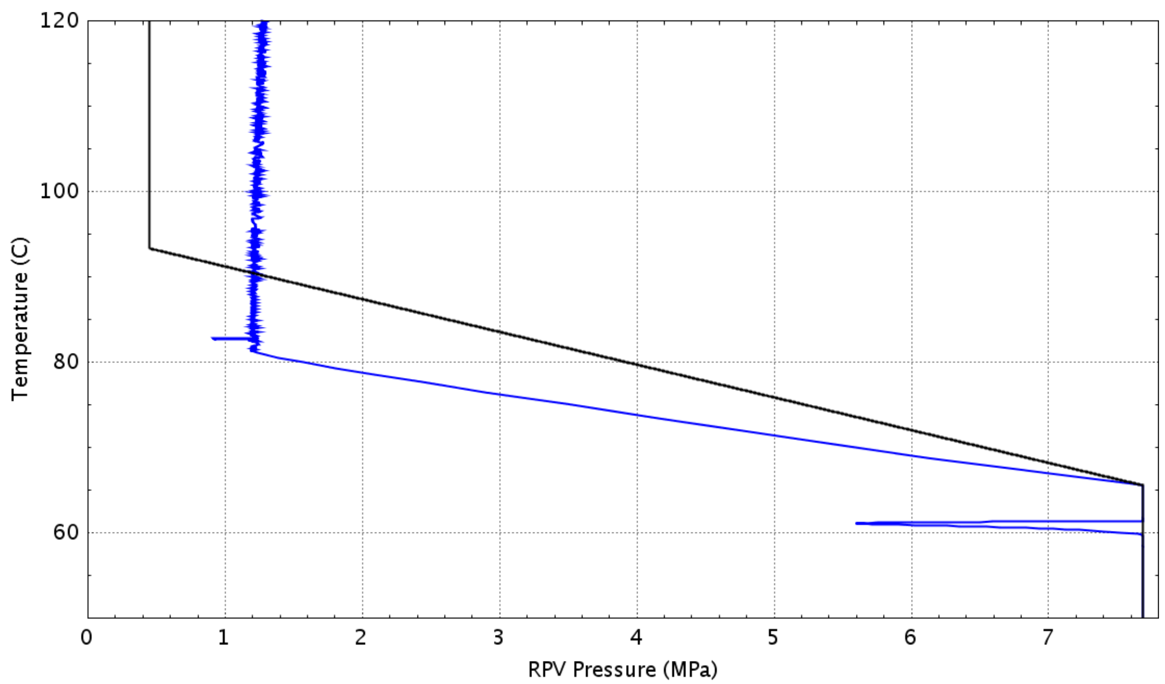

Figure G – 343 Plant status relative to the HCL curve (Graph 4 of the EOPs)

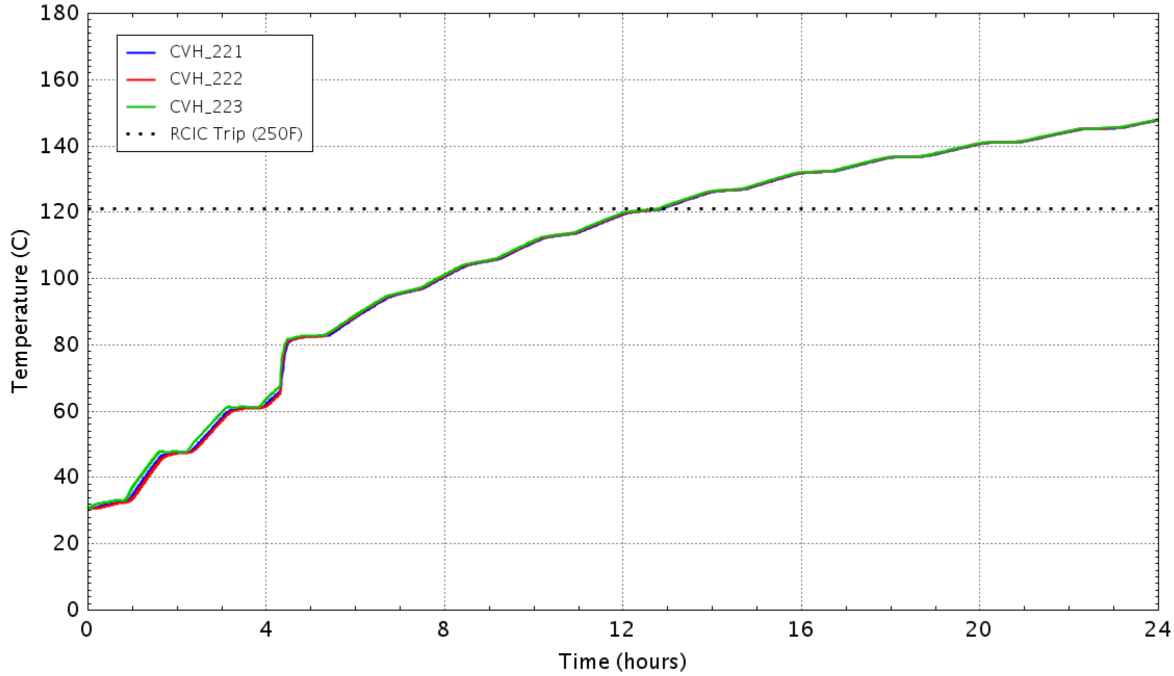

Figure G – 344 Water temperature in the wetwell

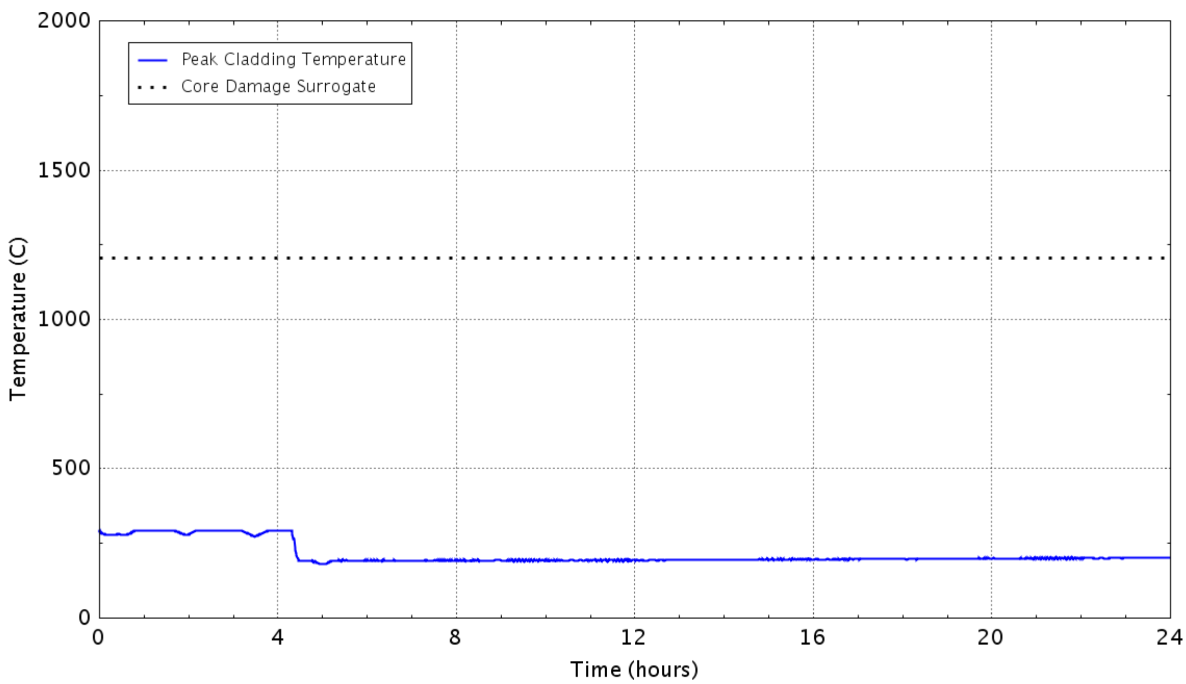

Figure G – 345 Peak temperature of the fuel cladding as a function of time

G.3.14 Case 19c: Sensitivity to LOMFW-25 Case 19 with MSIV Closure at the Start of the Transient

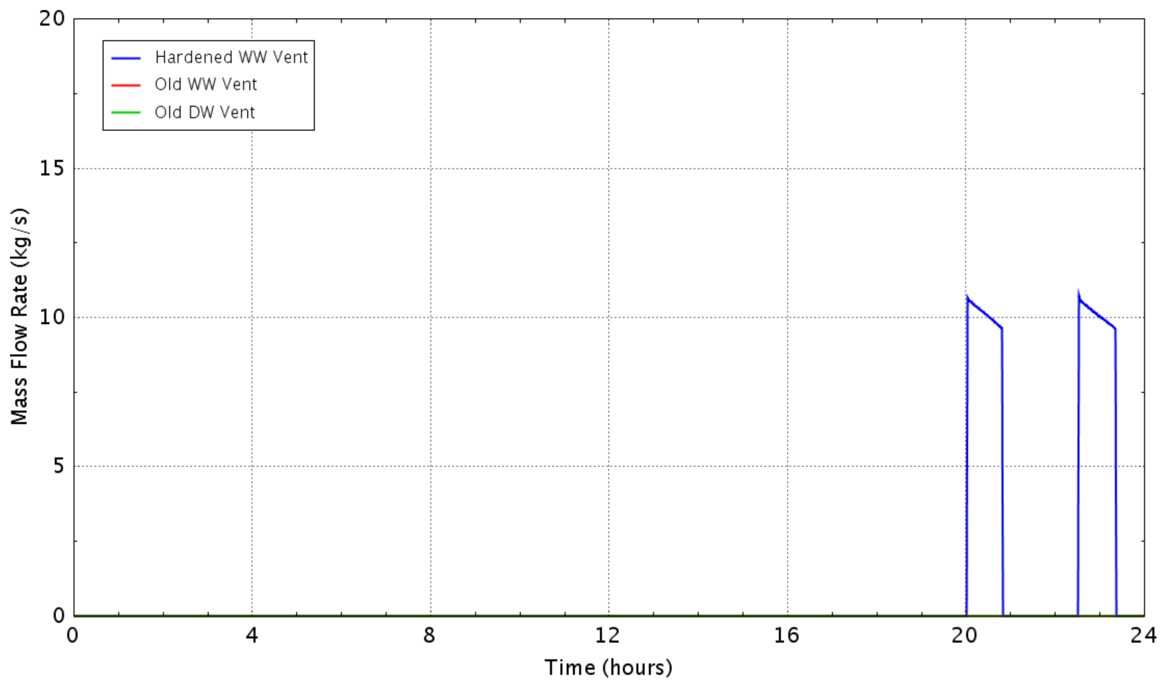

Figure G – 346 Flow rate of the containment vents

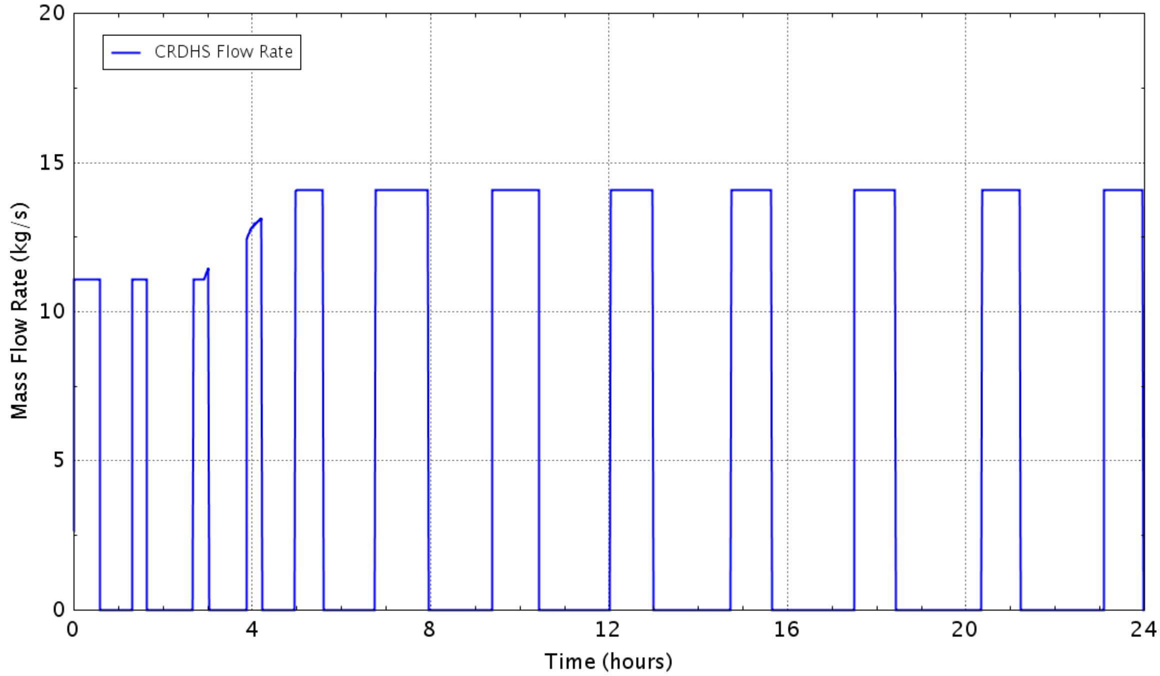

Figure G – 347 Flow rate of the control rod drive hydraulic system

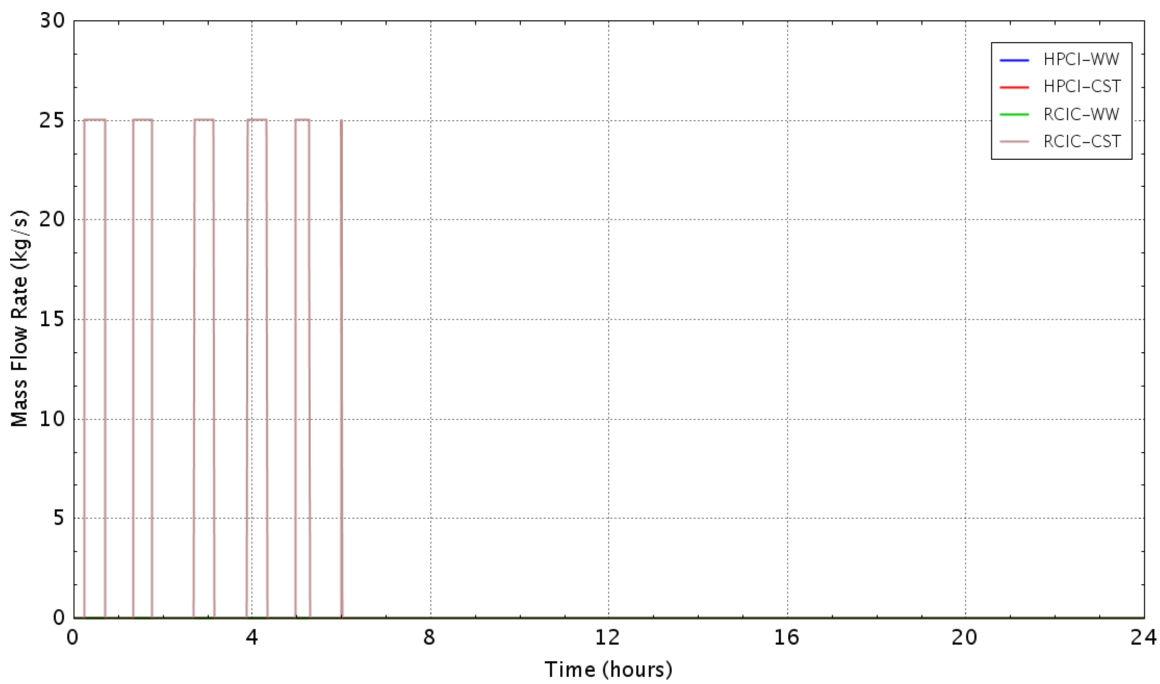

Figure G – 348 Flow rate of the HPCI/RCIC pumps

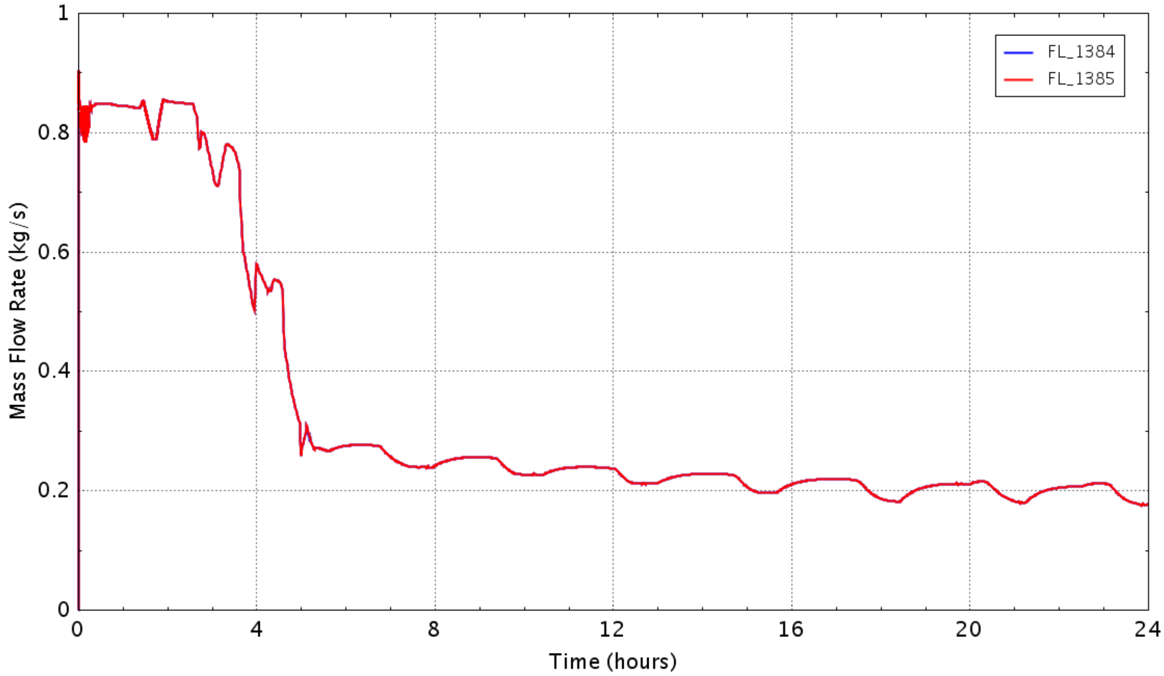

Figure G – 349 Flow rate of the recirculating pump seal leakage

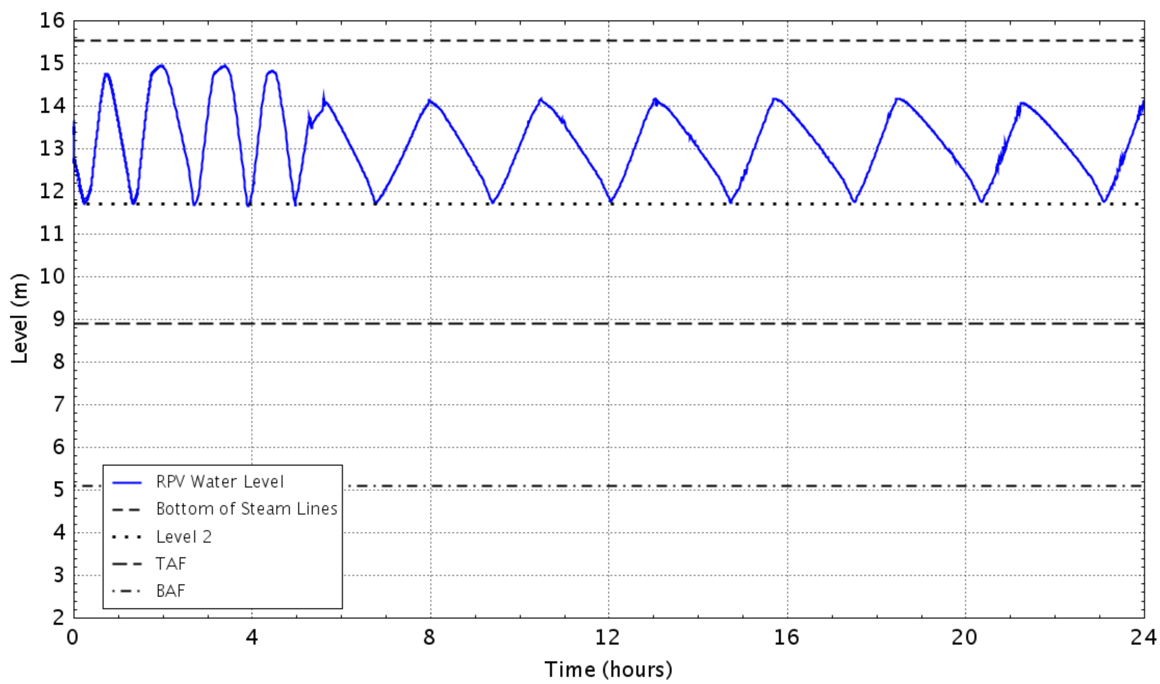

Figure G – 350 RPV Downcomer water level

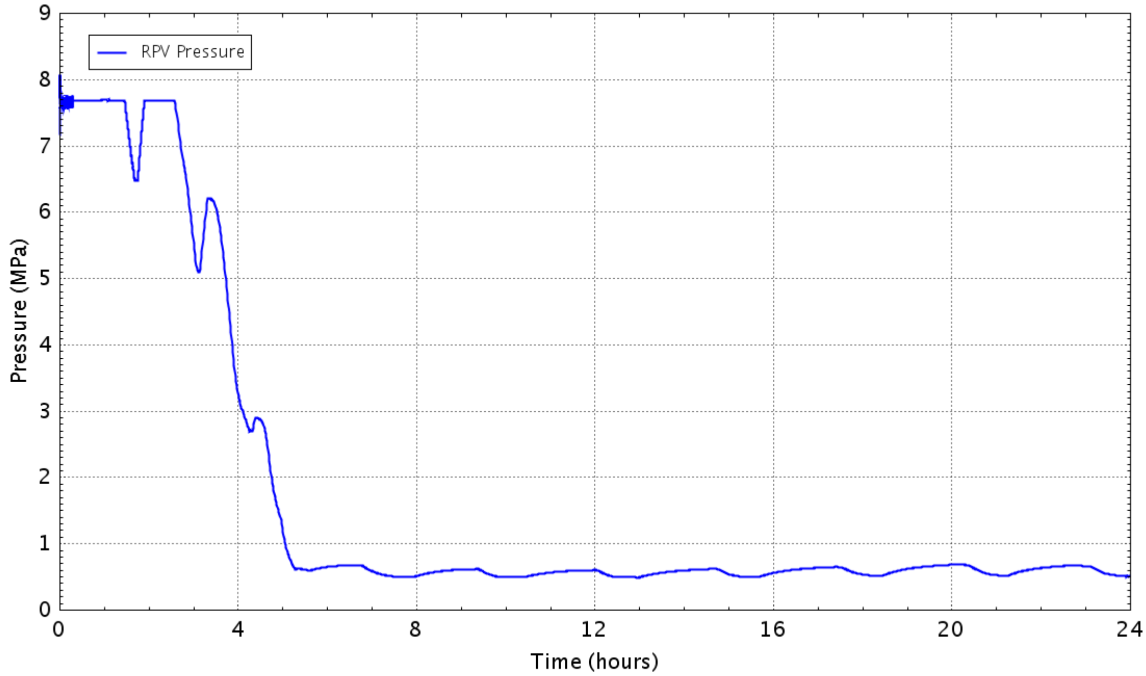

Figure G – 351 Pressure in the RPV

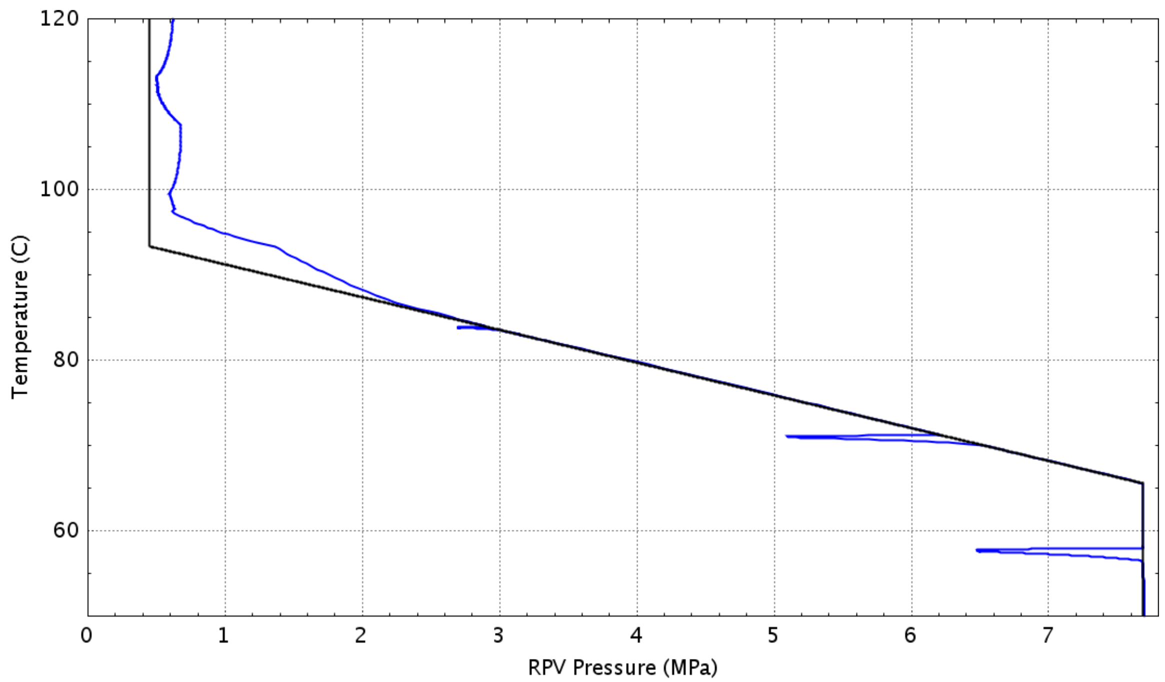

Figure G – 352 Plant status relative to the HCL curve (Graph 4 of the EOPs)

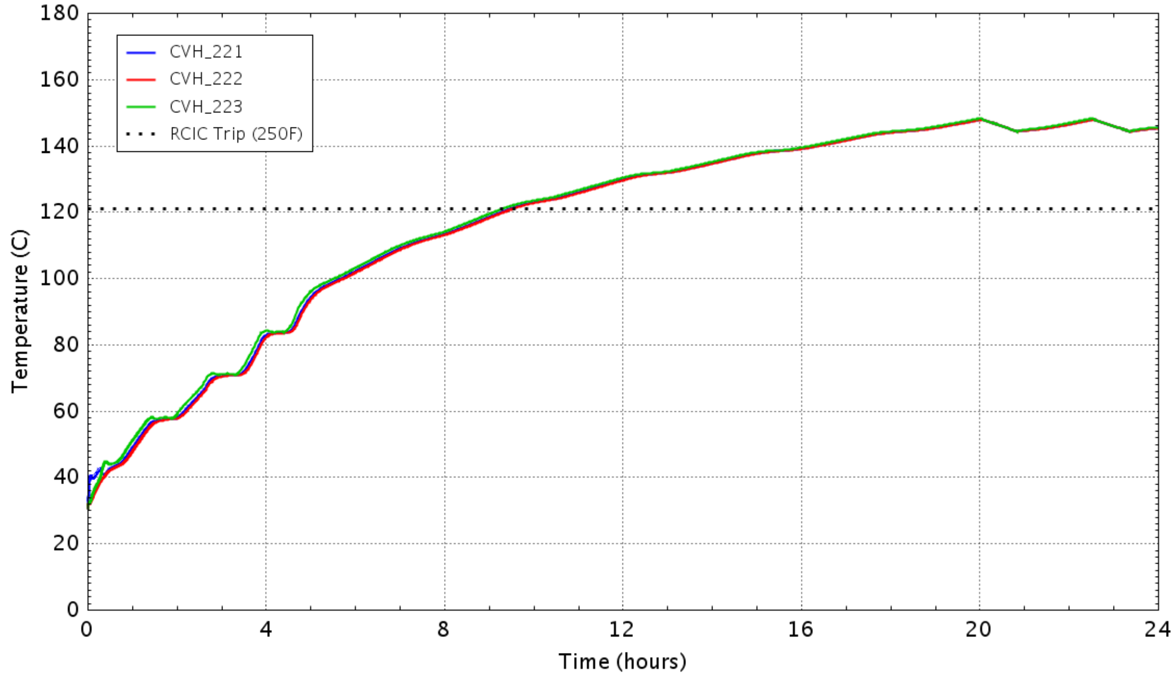

Figure G – 353 Water temperature in the wetwell

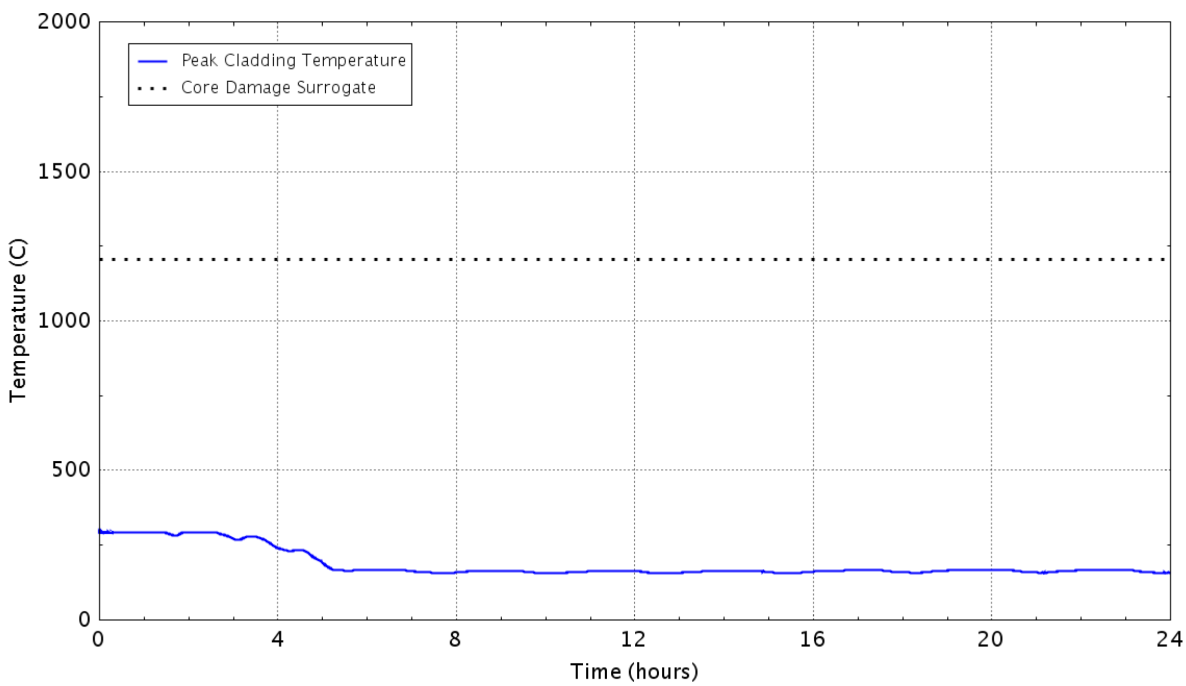

Figure G - 354 Peak temperature of the fuel cladding as a function of time

APPENDIX H
EVENT TREES

EVENT TREES

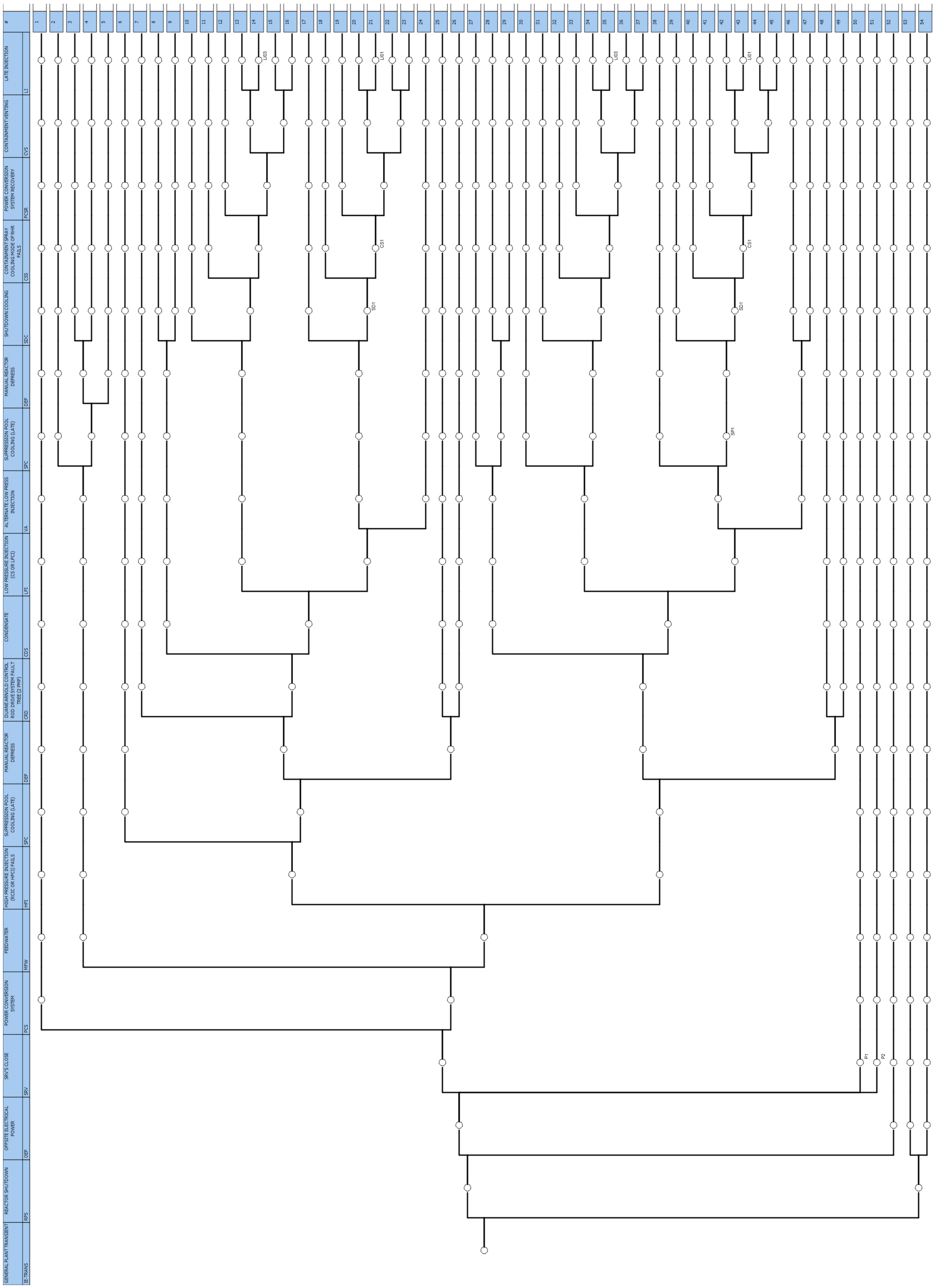

Figure H-1 TRANS event tree—Duane Arnold Energy Center (DAEC)

EVENT TREES (continued)

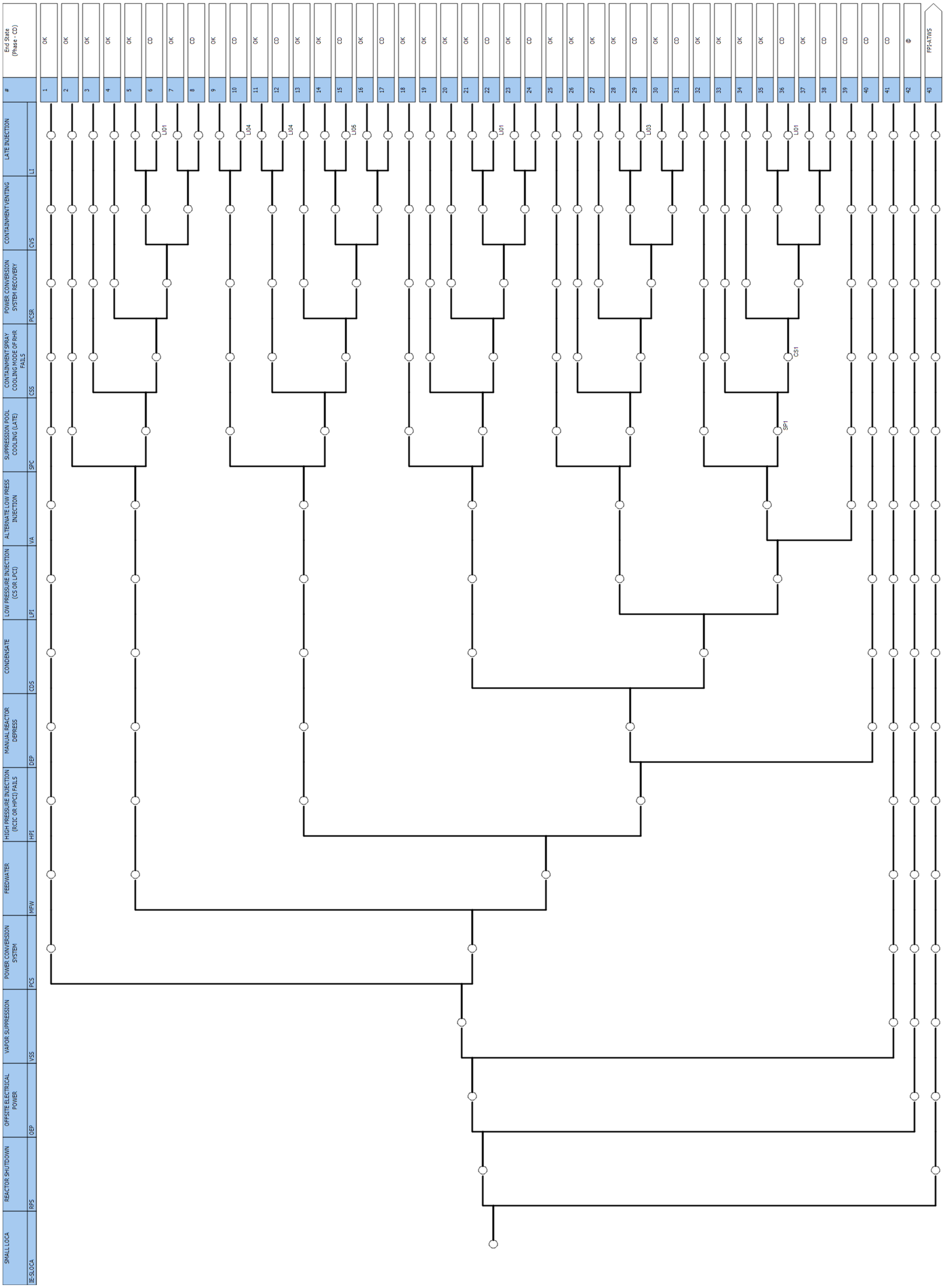

Figure H-2 Small loss-of-coolant accident event tree—DAEC

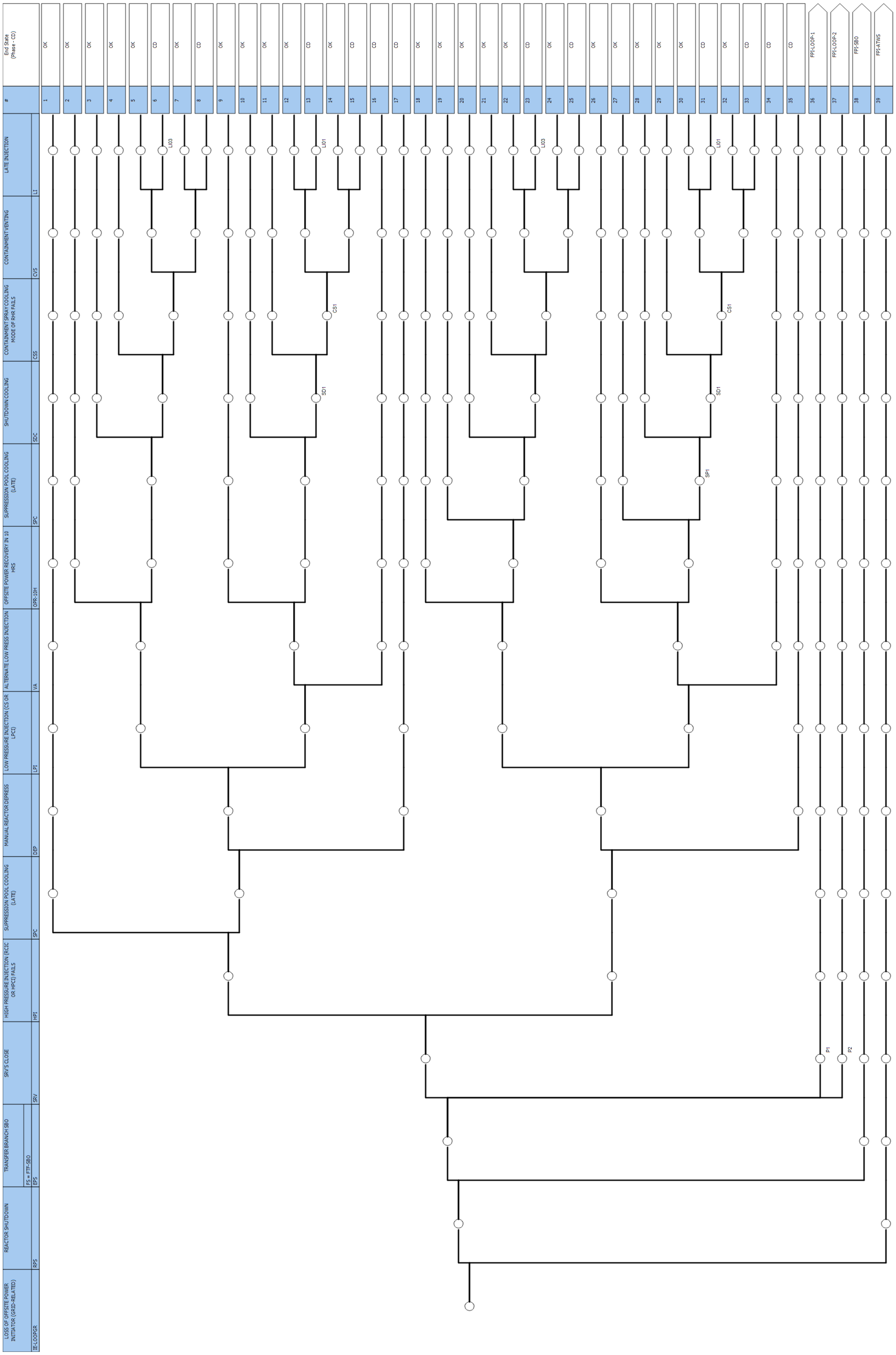

Figure H-3 Loss of offsite power (grid-related) event tree—DAEC

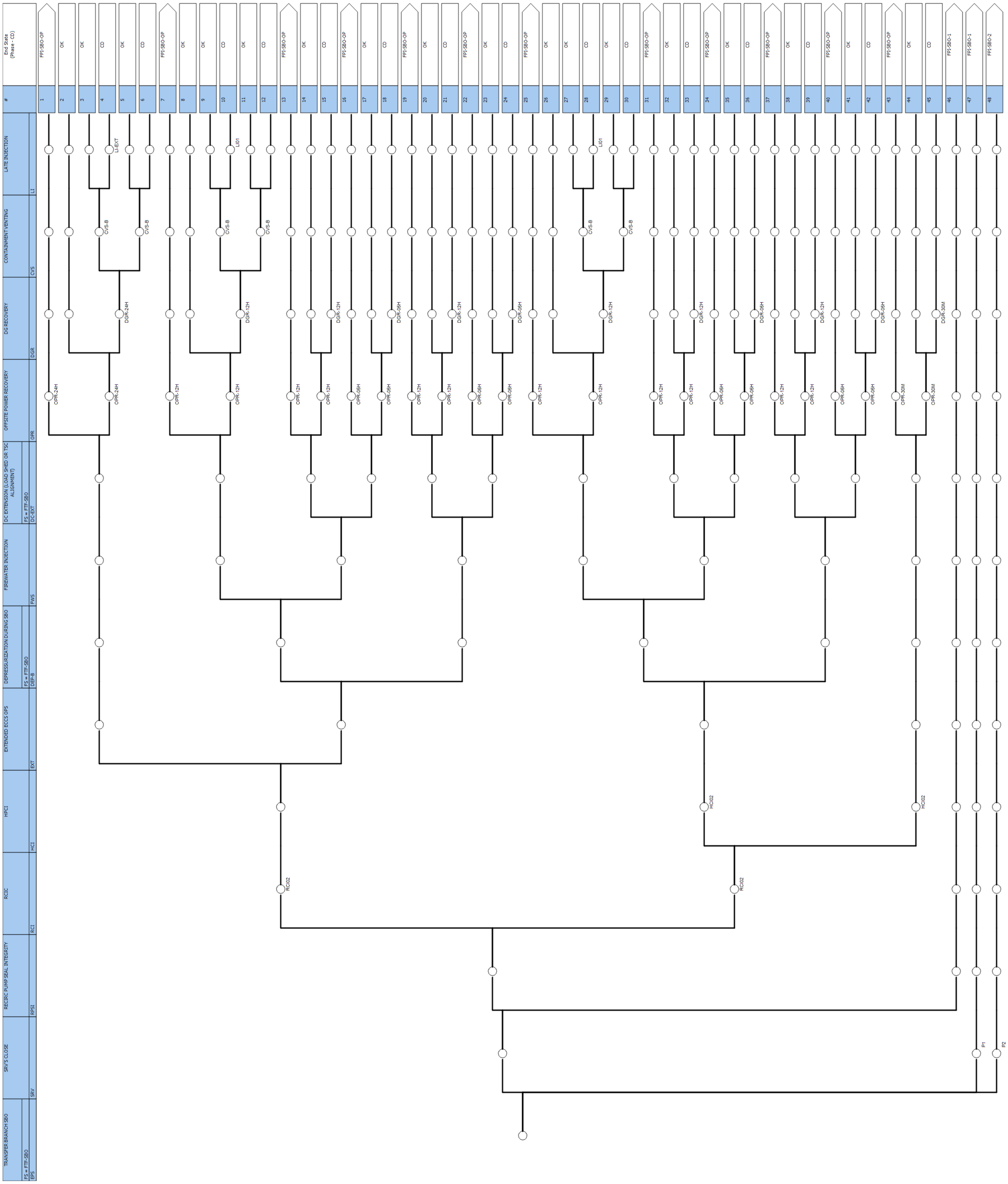

Figure H-4 Station blackout (SBO) event tree—DAEC

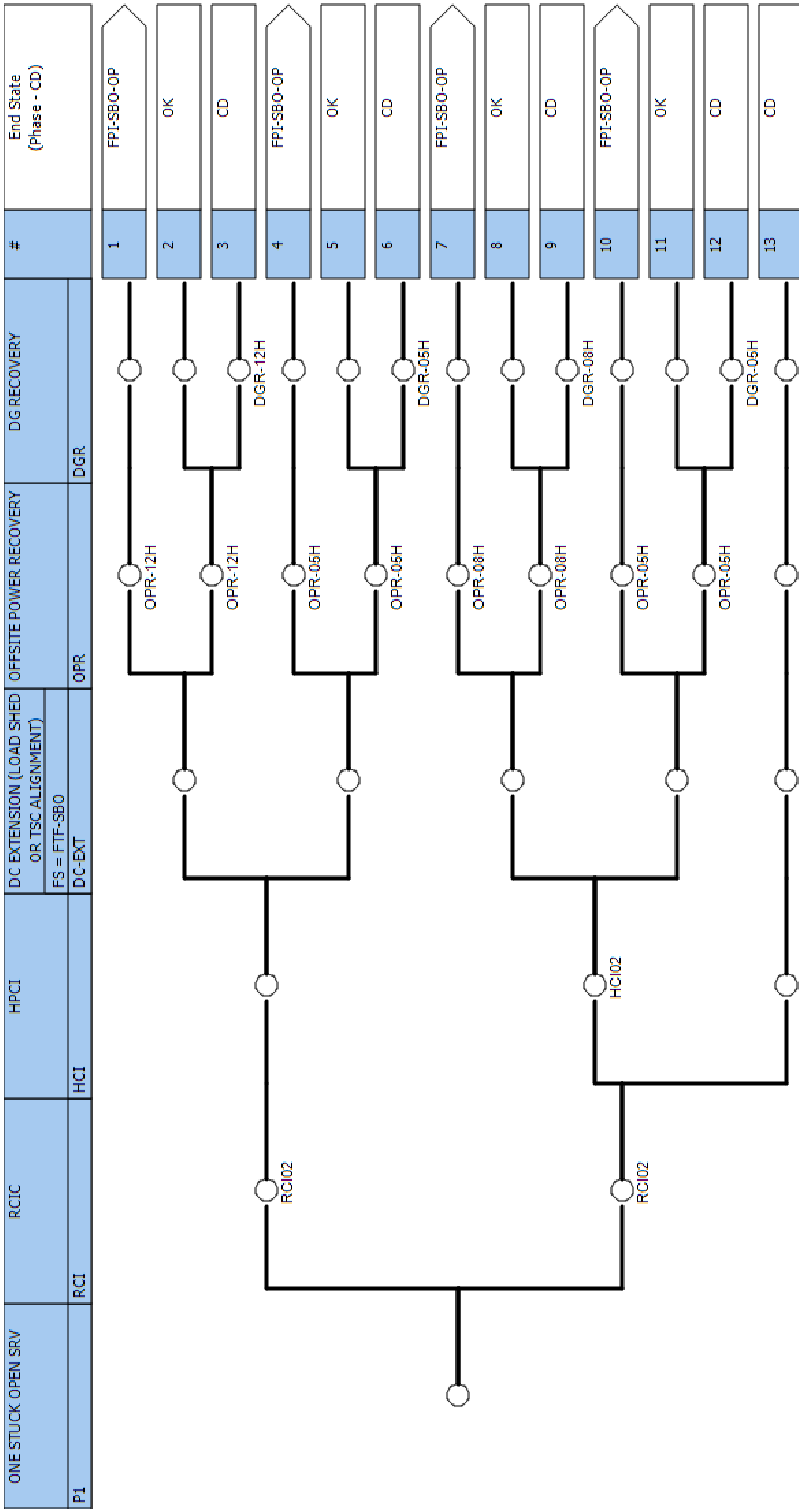

Figure H- 5 SBO-1 event tree—DAEC

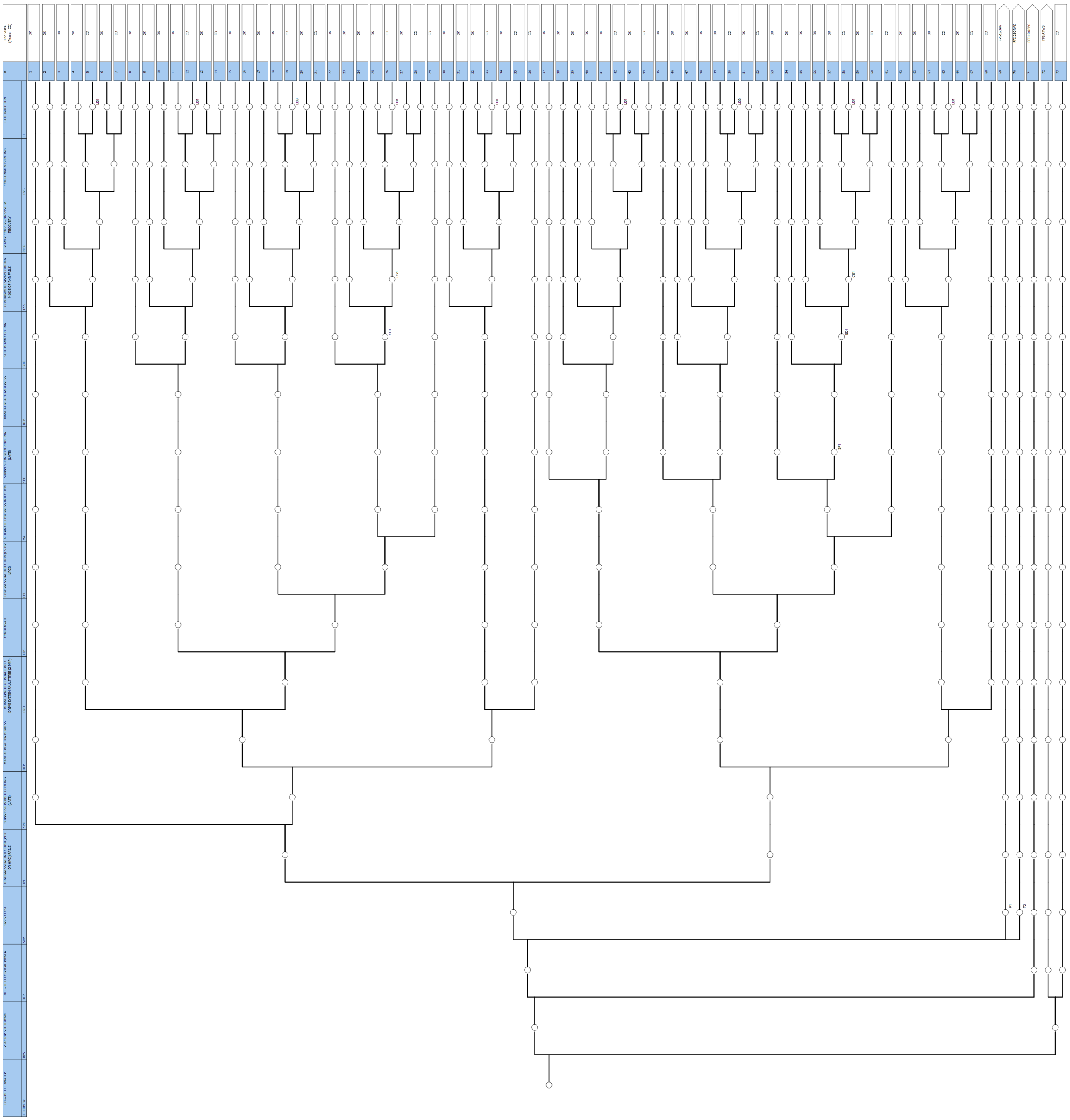

Figure H-6 Loss of main feedwater event tree—DAEC

NRC FORM 335 (12-2010) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-2236 Volume 3
2. TITLE AND SUBTITLE Confirmatory Thermal-Hydraulic Analysis To Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Duane Arnold Appendices F - H	3. DATE REPORT PUBLISHED MONTH: October YEAR: 2020 4. FIN OR GRANT NUMBER
5. AUTHOR(S) S. Campbell, S. Dennis, D. Helton, M. Khatib-Rahbar,A. Krall, L.Kozak, R. Buell, F. Sock, J. Pires	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Office of Nuclear Regulatory Research
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10. SUPPLEMENTARY NOTES

S. Dennis

11. ABSTRACT (200 words or less)

This report extends the work documented in NUREG-2187, “Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron Unit 1,” issued January 2016, to the Duane Arnold Energy Center. 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 for system performance and operator timing found in the agency’s probabilistic risk assessment 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.

This report first describes major assumptions used in this study. It then describes the major plant characteristics for the Duane Arnold Energy Center, 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 success criteria, the licensee’s success criteria, or other generic studies.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
PRA, Success Criteria, MELCOR, thermal-hydraulic	unlimited
	14. SECURITY CLASSIFICATION
	(This Page) unclassified
	(This Report) unclassified
	15. NUMBER OF PAGES
	16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

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NUREG-2236, Volume 3
Appendices F - H

Confirmatory Thermal-Hydraulic Analysis To Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Duane Arnold

October 2020