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MFN 07-237, Supplement 1 Docket No. 52-010

August 17, 2007

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **NEDO-33201, Revision 2, "ESBWR Probabilistic Risk Assessment," Sections 8,9,10, and 16 and Marked Up Table 7.2-5, Rev 2**

Enclosure 1 contains the subject partial ESBWR Probabilistic Risk Assessment (PRA) document NEDO-33201, Revision 2. The Enclosure also contains a marked up version of Table 7.2-5 of NEDO 33201, Revision 2. Additional sections will be provided on September 14, 2007 with the remaining PRA sections provided by September 28, 2007 as detailed in Reference 1.

Should you have any questions about the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

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NRO

References:

1. MFN 07-200, Letter from James C. Kinsey to U.S. Nuclear Regulatory Commission, *Integrated Plan and Schedule – ESBWR Design Certification Application,* April 19, 2007

Enclosure:

1. MFN 07-237 – NEDO-33201, Revision 2, "ESBWR Probabilistic Risk Assessment:" Enclosure 1, Attachment 1

Section	– 8	Containment Performance
Appendix	– 8A	Quantification of Containment Event Trees
appendix	– 8B	Representative Sequence Results
Appendix	– 8C	Containment Penetration Screening Analysis
Appendix	– 8D	Equipment Survivability Analysis
Section	– 9	Source Terms
Appendix	– 9A	Additional Documentation of MAAP cases
Section	– 10	Consequence Analysis
Section	– 16	Shutdown Risk

MFN 07-237 – List of Changes. Enclosure 1, Attachment 2

MFN 07-237 – Table 7.2-5, Rev 2 – Markup. Enclosure 1, Attachment 3

cc:	AE Cubbage	USNRC (with enclosure)	
	GB Stramback	GEH/San Jose (with enclosure)	
	RE Brown	GEH/Wilmington (with enclosure)	
	eDRF Section:	Section 8	0072-1548
		Appendix 8A	0072-7444
		Appendix 8B	0072-7447
		Appendix 8C	0072-6159
		Appendix 8D	0072-7449
		Section 9	0072-1610
		Appendix 9A	0072-8894
		Section 10	0072-1262
		Section 16	0072-7872
		Table 7.2-5 markup	0073-0098

Attachment 1 to Enclosure 1 of MFN 07-237, Supplement 1

NEDO-33201, Revision 2

Section	– 8	Containment Performance
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8 CONTAINMENT PERFORMANCE

8.0 INTRODUCTION

A spectrum of potential containment failure modes has been evaluated for the ESBWR. In Section 7, the potential for a break outside of containment was evaluated. In Section 21, potential ex-vessel steam explosion, direct containment heating, and basemat penetration challenges were evaluated. In this section, the focus is on the containment challenges associated with potential combustible gas deflagration, overpressurization, and bypass. The potential for containment failure due to these challenges is addressed by considering physical characteristics of the containment, notably the inerted condition, and containment structural capability, as well as the reliability of systems engineered to perform the containment functions of "isolation", "vapor suppression", and "heat removal".

Containment failure due to combustible gas deflagration is shown to be unrealistic considering the inerted containment and time period required to generate enough oxygen to create a combustible gas mixture. Deflagration risk associated with de-inerted operation prior to and following shutdown is considered in Section 8.1.4. The probability of containment failure due to overpressure or bypass requires consideration of the reliability of engineered systems used to isolate the containment and mitigate containment pressurization associated with a severe accident. As will be seen, the containment capability is such that the calculated probability of overpressurization can be considered to be negligible.

Consistent with the NRC design certification policy for advanced reactors discussed in Reference 8.0-1, the containment response has been evaluated for a 24-hour period following the onset of core damage. To provide additional insight on the containment performance objective discussed in the reference, containment effectiveness will be quantified to demonstrate that the containment provides a reliable barrier to radionuclide release after a severe accident.

Section 8.1 discusses the potential for combustible gas deflagration. Section 8.2 evaluates the probability of containment overpressurization and bypass. Section 8.3 presents the computer simulation results of containment response to overpressurization challenges. Section 8.4 summarizes key insights from the evaluation. Appendix 8A quantifies the frequency of all release categories. Appendix 8B displays additional documentation of the representative containment analysis sequences. Appendix 8C provides the screening analysis to support quantification of the containment isolation system probability. Sections 19B and 19C of the DCD document the containment ultimate strength calculation.

The results developed in this section, as well as from Section 21, are used to develop conservative source terms in Section 9 for use in the offsite consequence analysis. The offsite consequence analysis is presented in Section 10.

Table 8.0-1 summarizes acronyms and terminology used in this section.

Table 8.0-1

Acronyms and Terminology

General	
ADS	Automatic Depressurization System
BiMAC	Basemat Internal Melt Arrest and Coolability (Device)
CCI	Core Concrete Interaction
CET	Containment Event Tree
FAPCS	Fuel and Auxiliary Pools Cooling System
GDCS	Gravity Driven Cooling System
ICS	Isolation Condenser System
PCCS	Passive Containment Cooling System
VB	Vacuum Breaker
Sequence Nomenclature	
MLi	Medium LOCA (GDCS injection line)
T	Transient (e.g., MSIV closure, loss of AC)
T-AT	Transient with failure to insert negative reactivity.
nCHR	no Containment Heat Removal
nD	no Deluge
nDP	no Depressurization
nIN	no core Injection
nVB	no Vacuum Breaker (vacuum breaker failure to close)
Containment Release Categories	
BOC	Break Outside of Containment (Connecting RPV to environment)
BYP	Containment Bypass (Connecting containment to environment)
FR	Filtered Release (Through controlled suppression pool venting)
OP	Overpressure (General category)
OPW1	Overpressure due to failure of short-term containment heat removal
OPW2	Overpressure due to failure of long-term containment heat removal
OPVB	Overpressure due to failure of Vacuum Breaker
TSL	Technical Specification Leakage
CCIW	Wet Core-Concrete Interaction (Overpressure failure)
CCID	Dry Core-Concrete Interaction (Overpressure failure)
EVE	Ex-Vessel Steam Explosion
DCH	Direct Containment Heating

8.1 POTENTIAL FOR FAILURE DUE TO COMBUSTIBLE GAS DEFLAGRATION

Because the ESBWR containment is inerted during normal operation, the prevention of a combustible gas deflagration is assured in the short term following a severe accident. In the longer term there would be an increase in the oxygen concentration resulting from the continued radiolytic decomposition of the water in the containment. Because the possibility of a combustible gas condition is oxygen limited for an inerted containment, it is important to evaluate the containment oxygen concentration versus time following a severe accident to assure that there will be sufficient time to implement severe accident management (SAM) actions. It is desirable to have at least a 24-hour period following an accident to allow for SAM implementation. This section discusses the rate at which post-accident oxygen will be generated by radiolysis in the ESBWR containment following a severe accident. Of particular interest is the amount of time that, in the absence of mitigation SAM actions, is required to generate a combustible containment atmosphere in the presence of a large hydrogen release. A combustible atmosphere is assumed to be 5% oxygen by volume and qualifies as a de-inerted containment.

8.1.1 Background

The rate of gas production from radiolysis depends upon the power decay profile and the amount of fission products released to the coolant. Appendix A of Standard Review Plan (SRP) Section 6.2.5 (Reference 8.1-1) provides a methodology for calculation of radiolytic hydrogen and oxygen generation. The analysis results discussed herein were developed in a manner that is consistent with the guidance provided in SRP 6.2.5 and Regulatory Guide 1.7 (Reference 8.1-2).

There are unique design features of the ESBWR that are important with respect to the determination of post-accident radiolytic gas concentrations. In the post-accident period, the ESBWR does not utilize active systems for core cooling and decay heat removal. As indicated earlier, for a design basis loss-of-coolant accident (LOCA), the automatic depressurization system (ADS) would depressurize the reactor vessel and the gravity driven cooling system (GDCS) would provide gravity driven flow into the vessel for emergency core cooling. The core would be subcooled initially and then it would saturate resulting in steam flow out of the vessel and into the containment. The passive containment cooling system (PCCS) heat exchangers would remove the energy by condensing the steam. This would be the post-accident mode and the core coolant would be boiling throughout this period.

A similar situation would exist for a severe accident that results in a core melt followed by reactor vessel failure. In this case, the GDCS liquid would be covering the melted core material in the lower drywell, with an initial period of subcooling followed by steaming. The PCCS heat exchangers would be removing the energy in the same manner as described above for a design basis LOCA.

To ensure that non-condensable gases do not degrade the heat transfer efficiency of the PCCS heat exchangers, vents to the suppression pool are provided. The suppression pool vents connect the heat exchanger drums to the suppression pool; non-condensable gas flow is driven by the positive DW to WW pressure differential. The calculation of post-accident radiolytic oxygen generation accounts for this movement, and assumes that the majority of non-condensable gases move to the suppression pool after they are formed in the drywell.

The effect of the core coolant boiling is to strip dissolved gases out of the liquid phase resulting in a higher level of radiolytic decomposition. This effect was accounted for in the analysis.

8.1.2 Analysis Assumptions

The analysis of the radiolytic oxygen concentration in containment was performed consistent with the methodology of Appendix A to SRP 6.2.5 and Regulatory Guide 1.7. Some of the key assumptions are as follows:

- Reactor power was 102% of rated
- $G(O_2)$ = 0.25 molecules/100eV
- Initial containment O_2 concentration = 4%
- Allowed containment O_2 concentration = 5%
- Stripping of drywell non-condensable gases to wetwell vapor space
- Fuel clad-coolant reaction up to 100%
- Iodine release up 100%
- Adequate gas mixing throughout containment

8.1.3 Analysis Results

The analysis results show that, while inerted, the time required for the oxygen concentration to increase to the de-inerting value of 5% is significantly greater than 24 hours for a wide range of fuel clad-coolant interaction and iodine release assumptions up to and including 100%. Thus, the potential for containment failure due to combustible gas deflagration will not be discussed further.

8.1.4 Risk Due to De-Inerted Operation

The ESBWR operates in short de-inerted states prior to and following shutdown. Because combustible gas deflagration cannot be excluded when the containment atmosphere is de-inerted, this analysis conservatively assumes that all core damage events during this de-inerted window will lead to containment failure.. The total time of de-inerted operation per shutdown is limited to 48 hours by Technical Specifications, so 24 hours per year is used to calculate the release contribution. To account for de-inerted operational release, the core damage frequency per day was applied to the "bypass" release category in addition to the separately calculated release frequency.

Additional BYP Frequency = CDF/365 = 3.34E-11

8.2 FREQUENCY OF OVERPRESSURE AND BYPASS RELEASE CATEGORIES

The containment bypass (BYP) failure mode represents the failure to isolate containment before or during a severe accident, thus allowing a radionuclide barrier to be breached. The containment overpressure (OP) failure mode represents the potential for containment pressurization from stored energy and decay heat to exceed the ultimate containment strength. The likelihood of these failure modes was quantified as part of the "Containment Event Trees" (CETs) evaluation. The following potential release categories are the CET end states. The first group depicts containment failure due to containment systems:

- Containment bypass (BYP) represents the condition in which the containment has been bypassed due to failure of the Containment Isolation System. With this failure mode, the containment is assumed to be unavailable as a radionuclide barrier from the start of the severe accident, i.e., the containment isolation function has failed. As a result, there is a direct path from the containment atmosphere to the environment.
- Overpressurization (OPW) represents the condition in which the vapor suppression capability has functioned, but there has been a failure to remove energy from the containment, i.e., the containment heat removal function has failed. Two modes of containment heat removal failure are considered. Short term failure (within 24 hours of accident initiation) is defined as "OPW1" category; long term failure (after 24 hours) is defined as "OPW2".
- Overpressurization due to vacuum breaker failure (OPVB) represents the condition in which a vacuum breaker is open or fails to reclose, which is assumed to defeat the containment's vapor suppression function. In such a situation, containment overpressure occurs earlier than in the OPW failure mode.

There are also release categories that describe containment failure due to phenomenological consequences. These failure modes are discussed and evaluated in Section 21, and are summarized below:

- Wet core-concrete interaction (CCIW) represents containment failure after the deluge system is successful but the core is postulated to have been ejected in a non-coolable geometry. Water does cover the core, but significant core-concrete interaction persists and eventually results in containment overpressurization.
- Dry core-concrete interaction (CCID) represents the failure of the deluge system to inject after the core has been deposited on the BiMAC layer. High levels of aerosols and non-condensables are produced and eventually lead to containment overpressurization.
- Ex-vessel explosion (EVE) represents the reactor pedestal, and subsequent containment, failure as a result of a steam explosion. The analysis assumes that an adequate steam explosion occurs every time the core melts through the RPV into a "high" lower drywell water level. The severe accident analysis in Section 21 shows that ex-vessel explosion due to medium water level is physically unreasonable. However, the remote potential for a steam explosion at "medium" water level will be documented in a Section 11 sensitivity with a split fraction of 1E-3.

- Direct containment heating (DCH) represents containment failure due to the direct deposit of the core on the LDW walls during a high-pressure melt ejection (HPME). Similar to medium water level EVE, this failure mode is found to be remote in the Section 21 analysis, and is excluded from the baseline Level 2 analysis. However, the remote potential for an immediate containment failure due to DCH will be captured in a Section 11 sensitivity by assigning a 1E-3 point estimate in the class III event trees.

Also shown on the CETs are two end states that are not considered containment failure because they do not result in the loss of control of the containment boundary:

- Technical Specification Leakage (TSL) represents the condition in which the containment pressure boundary is intact and the only source term is that associated with the allowable leakage rate, as defined by the Technical Specifications.
- Filtered release (FR) is an end state depicting containment venting under operator control. Such a release results in a much lower radionuclide source term than containment failure because the radionuclide pathway is through the suppression pool, which provides filtering of the radionuclides.

The CETs are discussed in more detail in Subsection 8.2.1.

8.2.1 Containment Event Trees

The CETs used for the Level 2 PRA analysis and described in Section 8 and 8C are logically equivalent to the combination of containment phenomenology event trees (CPET) and containment system event trees (CSET) used in the Section 21 discussion. In Section 21, separation of the two trees allowed a more clear distinction between the two analyses; a single, combined CET simplifies the quantification of the Level 2 PRA itself.

The Level 1 analysis, described in earlier sections, evaluated severe accident sequences with the potential to cause core damage. The core damage frequency associated with each of these sequences is discussed in Appendix 8A. In that appendix, the core damage sequences were grouped according to their similarity and potential containment challenge so that a manageable number of sequences could be evaluated in terms of the containment response. The class definition and contribution of each accident class to the core damage frequency is summarized as follows:

Accident Class	CDF contribution (per year)	Percentage CDF contribution	Class summary
Class I	5.63E-9	46.16	Sequences with RPV failure at low pressure
Class II	4E-11	0.35	Sequences with containment failure preceding core damage
Class III	4.52E-9	37.02	Sequences with RPV failure at high pressure
Class IV	1.87E-9	15.36	Sequences involving failure to insert negative reactivity
Class V	1E-10	1.20	Break outside of containment

Containment event trees were used to evaluate the complete spectrum of potential challenges to containment integrity. Both phenomenological effects and system responses are captured in the CET analyses.

The number of CETs needed to evaluate the containment response to the Level I accident classes, was established with the following considerations:

- Class II sequences, by definition, ultimately result in containment failure prior to core damage; thus, an event tree is not required to evaluate the probability of containment failure.
- Class V sequences involve direct communication between the RPV and environment which renders containment systems, and associated event tree modeling, irrelevant.

Thus, containment event trees were required only to evaluate the containment response to Class I, Class III, and Class IV events.

The CETs were developed by establishing the functions and containment systems that were relevant to mitigating the overpressure and bypass challenges and combining those with the phenomenology discussed in Section 21. The CETs were then constructed using point estimates for phenomenological effects and appropriate logic to account for mitigating system success or failure by establishing the logically possible containment responses. Finally, the end states of the CETs, which are termed "release categories", were defined. The release categories may indicate containment failure or may indicate that the containment has successfully functioned to limit the radionuclide release. These release categories represent meaningfully different outcomes to the containment challenge and are used in the source term evaluation discussed in Section 9.

Review of the CETs indicates that that there is a generally common structure to the trees, with only the phenomenology differing between the initiating, or entry events. Determination of the CET entry event probabilities is discussed in Subsection 8.2.1.1. The containment systems evaluated in the CETs are summarized in Subsection 8.2.1.2 with the associated top events being discussed in Subsection 8.2.1.3. The end states of the trees, which become the release categories for the consequence evaluation, are discussed in Subsection 8.2.1.4. The frequencies associated with the release categories are presented in Subsection 8.2.1.5. Appendix 8A provides additional detail on the release category quantification.

8.2.1.1 CET Entry Events

The CET entry event frequencies are summarized in Table 8.2-1. Note that each accident class may be divided into various subclasses. The subclasses were necessary to reflect the water level in the lower drywell at the time of RPV failure and the associated phenomenological effects that should be considered. For example, accident Class I was divided into Class I_LD (low water level or dry), Class I_M (medium water level), and Class I_H (high water level). Based on Level 1 results, Class III sequences only result in low water level or dry scenarios, so only Class III_LD was considered.

The Class IV (ATWS) sequences experience core damage at high pressure because ADS is inhibited as part of the core damage mitigation effort. However, it is assumed that Emergency Operating Procedures (EOPs) will instruct the operator to depressurize after core damage has occurred in an attempt to preserve containment. It is shown in Appendix 8A that the frequency of ATWS sequences experiencing RPV rupture at high pressure is negligible, so only failures at low pressure were analyzed. Thus, the subclasses for Class IV are IV_LD (ATWS with low water level or dry) IV_M (ATWS with medium water level), and IV_H (ATWS with high water level).

The qualitative binning of Level 1 sequences into water levels is shown in Section 7.

8.2.1.2 Mitigating Systems

The ESBWR includes systems with the capability to prevent or mitigate containment bypass and overpressurization. The systems considered in the evaluation of containment response are summarized below.

Containment Isolation System

The containment isolation system provides for monitoring and isolation of the containment boundary to prevent unacceptable radiological releases during normal, abnormal, and accident conditions.

Isolation Condenser System

The isolation condenser system (ICS) provides the capability to remove decay heat from the RPV. Because the heat exchangers are external to the containment, removal of heat from the RPV also removes energy from the containment. The isolation condensers would be effective primarily when the RPV is at an elevated pressure. The isolation condensers do not condense a significant amount of steam after RPV depressurization and thus, provide little mitigation of a severe accident after RPV depressurization. In high pressure severe accident sequences the ICS has, by definition, failed already in the Level 1 sequence. Consequentially, the ICS was not credited in the severe accident sequence evaluation.

GDCS Deluge and BiMAC

The deluge mode of GDCS operation provides flow through the BiMAC to flood the lower drywell when the temperature in the lower drywell increases enough to be indicative of RPV failure and core debris in the lower drywell. The GDCS deluge system is activated by a combination of thermocouples embedded in the lower drywell floor and temperature sensors in the lower drywell airspace.

By flooding the lower drywell after the introduction of core material, the potential for energetic fuel-coolant interaction at RPV failure is minimized. Covering core debris with water provides scrubbing of fission products released from the debris and cools the corium, thus limiting potential core-concrete interaction. The BiMAC provides additional assurance of debris bed cooling by providing an engineered pathway for water flow through the debris bed.

Containment Heat Removal (PCCS and Suppression Pool Cooling)

Containment heat removal can be provided by either the PCCS or the suppression pool-cooling mode of the FAPCS. For sequences with successful containment heat removal, the thermal-hydraulic analysis assumed that the PCCS was available and that suppression pool cooling was not in operation. This assumption bounds the containment pressure response because the PCCS can only limit pressurization, while suppression pool cooling can limit and reduce containment pressure.

The PCCS receives a steam-gas mixture from the upper drywell atmosphere, condenses the steam using the PCCS pools as a heat sink, and returns the condensate to the GDCS pool. The non-condensable gas is drawn to the suppression pool through a submerged vent line by the pressure differential between the drywell and wetwell. The PCCS is designed to remove decay heat added to the containment after a LOCA, thus maintaining the containment within its pressure limits. Operation of the PCCS requires no support systems and, as illustrated in Section 8.3, there is adequate inventory in the PCCS pools to provide containment heat removal for more than 24 hours after the onset of core damage. Vacuum breakers are required to be leak-tight for effective PCCS operation.

Drywell Spray

In ESBWR, the drywell sprays are designed as a post-event containment clean up and recovery system to limit the exposure to "first-in" personnel after a severe accident. Use of the drywell spray system will not be included as part of the severe accident mitigation guidelines; the drywell sprays are not included in the Level 2 analysis.

Vacuum Breakers

Vapor suppression requires that a pressure differential be maintained between the drywell and the suppression pool. Failure of the vacuum breakers, either due to a preexisting condition or failure to reclose, is assumed to result in loss of the vapor suppression capability. That is, sequences in which vacuum breaker failure occurred were modeled with an open path between the drywell and wetwell airspace. The success criteria for vapor suppression is no more than one vacuum breaker may be stuck open, and at least one vacuum breaker must open to break a positive WW to DW pressure differential.

Manual Containment Overpressure Protection (MCOPS)

To prevent overpressurization failure of the containment as a result of containment heat removal failure, the ESBWR contains a manually controlled vent connecting the suppression chamber gas space to the environment through the reactor building ventilation system. Opening the vent would greatly decrease the magnitude of a potential release in comparison to containment failure by forcing the radionuclide pathway through the suppression pool. As will be shown in Section 8.3, failure of containment heat removal does not cause the containment to pressurize to the point at which venting is likely to be implemented to prevent containment failure in the 24-hour time frame after onset of core damage.

Reactor Building Effects

Fission product releases to the environment through the paths representing "normal" containment leakage, i.e., leakage up to the amount allowed by the Technical Specifications, could be reduced for some sequences if credit were taken for radionuclide removal by the reactor building HVAC system. However, such a source term reduction was not credited in the severe accident sequence modeling. Therefore, the source terms of sequences with only Technical Specification leakage are conservative in that they represent a direct release from the containment to the environment. Sequences in which the drywell failure is at the drywell head seals are also conservative because credit is not taken for refueling pool scrubbing. Sequences with drywell failure at other locations are not significantly affected because the release path bypasses the reactor building or would overwhelm the capacity of the reactor building ventilation system.

8.2.1.3 Top Events

Subsection 8.2.1.1 identified the entry events for the containment system event tree. The next step in constructing CET was to define, as top events, the phenomenological events and system functions needed to assess the containment response to severe accident challenges. The phenomena include ex-vessel explosion, direct containment heating, and dry and wet core concrete interaction. The system functions are "containment isolation", "vapor suppression", "containment heat removal", and "venting".

Defining top events for the recovery of failed systems and for operator actions was considered, but was judged unnecessary, as indicated in the following sections. As a result, the event trees necessary to model the containment response are simple in form. Further, because the Level 2 PRA model is arranged as an extension of the Level 1 accident sequences, the initiator impact of all sequences is implicitly included. That is, the structure of a CET is the same irrespective of the Level 1 initiating event being considered. The trees differ only in the quantification as dependencies on the entry event are reflected in phenomena that must be considered depending on the entry event bin. For example, the class III CET considers the possibility of containment failure due to DCH because the RPV fails at high pressure. The CET structure is provided as Figure 8.2-1 with corresponding event probabilities provided in Table 8.2-1. A discussion of the treatment of system recovery, operator actions and top events follows. Appendix 8A provides additional discussion of the top event probabilities.

8.2.1.3.1 Repair of Failed Systems

Recovery of failed systems was not credited in the severe accident analysis.

8.2.1.3.2 Key Operator Actions

Because of the passive nature of the ESBWR containment systems, there are no operator actions required to support the containment response to a severe accident in the 24-hour period after onset of core damage. The containment isolation system, vacuum breakers, and PCCS do not require operator action to initiate or function. Analyses provided in Section 8.3 will show that operator action is not required to maintain containment heat removal through the PCCS for the 24-hour period after onset of core damage and that containment venting will not be required during that period. Thus, operator actions are considered in the containment evaluation only in terms of:

(1) Action taken as a backup to an automatic action, e.g., to open the connecting valve for PCCS pool makeup if the low-water level signal were to fail.

(2) Action taken to initiate a backup system, e.g., to actuate the FAPCS if the PCCS were unavailable.

(3) Actions requiring a long time period to initiate. For example, the suppression chamber vent is under operator control. As indicated in Section 8.3, there would be a long time period (more than 24 hours) in virtually all scenarios to initiate venting to prevent containment overpressure due to a loss of containment heat removal.

Because these operator actions are redundant to passive system functioning or are required only after a long time period, such actions do not have a significant effect on the probability of containment failure.

8.2.1.3.3 Top Event EVE_DAM

Top event EVE, which considers the probability of containment failure due to ex-vessel steam explosion, is only included in CETs in which the entry events include a medium water level in the lower drywell. The phenomenology analysis in Section 21 considers a steam explosion from a medium water level to be physically unreasonable; baseline results only include EVE contribution from high water level cases. The effect of considering medium-level steam explosions is presented in a sensitivity study in NEDO-33201 Section 11.

The ESBWR RPV design features several small lower head penetrations for equipment such as fine motion control rod drive (FMCRD) units and in-vessel instrumentation. The heavily dominant RPV failure mode in a core melt scenario is expected to be one of these lower head penetrations, which would result in a relatively slow flow of corium from the vessel breach, and thus potential premixtures that are much smaller (reduced energetics) than those considered in the analyses presented here. However, since these complex processes are not well known, we assume conservatively that all high water level cases result in ex-vessel explosions that are large enough to damage the pedestal to an extent that could be considered gross containment failure.

8.2.1.3.4 Top Event DCH_DAM

Top event DCH, which considers the probability of containment failure due to direct containment heating, is only included in CETs for which the entry events are Class III core damage sequences. That is, the RPV fails at high pressure. The phenomenology analysis in Section 21 considers containment failure due to DCH to be physically unreasonable. The effect of considering actual containment failure caused by DCH is presented in a sensitivity study in NEDO-33201 Section 11.

8.2.1.3.5 Top Event BI_SP

Top event BI_SP represents the function of the deluge system of the GDCS. The deluge system is a sub-system of GDCS that is not included in the Level 1 analysis, so a new fault tree was created to address its function. The deluge system is comprised of twelve injection lines, each of which includes a squib valve that must be fired for successful operation. Actuation of the system is performed by a deluge-specific, stand-alone actuation system that does not require any support systems. Failure of the deluge system is assumed to result in containment failure due to dry core-concrete interaction.

8.2.1.3.6 Top Event BI_FN

Top event BI_FN estimates the probability that even if the deluge system functions successfully, the water is not successful in cooling the core. As postulated in Section 21, this node is used to quantify both the uncertainty in the BiMAC design effectiveness, and the potential for the core to be ejected in a non-coolable geometry. Failure of this node is assumed to result in containment failure due to wet core-concrete interaction. Node BI_FN is assigned a probability of 1.00E-2.

8.2.1.3.7 Top Event CIS

Top event CIS, representing the containment isolation system, assesses the probability that the containment has not been isolated and, as a result, there is a pathway from containment into the reactor building or directly into the environment.

Section 4.18 documents containment isolation from the perspective of analyzing pipe breaks outside of containment (BOC) for the Level 1 analysis. As indicated in Appendix 8C, a screening evaluation was performed to identify those containment penetrations that could potentially lead to offsite consequences. The screening analysis found that two systems, Main Steam and Feedwater, require isolation during a severe accident. The same logic is used for these isolations as with the BOC accidents in the Level 1 PRA. This approach addresses both the "failure-to-close" hardware and actuation logic failure modes of the containment isolation system.

If CIS is failed, the event tree path has no additional branching because the containment has been bypassed and operation of the vacuum breakers, containment heat removal or venting functions is irrelevant. The bypass failure is assumed to be present at the onset of core damage and is not recovered for the duration of the sequence.

8.2.1.3.8 Top Event VB

Top event VB models vacuum breaker operation for vapor suppression. The success criteria for VB are the same as DS-TOPVB in the Level 1 analysis: at least one vacuum breaker must open to relieve a vacuum, and two of three must be leak-tight for vapor suppression. If VB were not successful the containment would pressurize relatively quickly because the vapor suppression function is ineffective. The failure probability is a conditional probability derived from fault tree modeling as discussed in Section 4.18.

8.2.1.3.9 Top Event W

Top event W models containment heat removal. The event is partitioned into "short-term" and "long-term" heat removal functions, "W1" and "W2", respectively. The passive PCCS system and the active suppression pool cooling mode of the FAPCS are considered in these nodes. As indicated in Subsection 4.19.2, the PCCS is designed with adequate water in the PCCS pools to mitigate a design basis event for 24 hours after event initiation. Accordingly, event W1 addresses the period from event initiation to 24 hours after event initiation. This is conservative as indicated by Figure 8.2-2, which illustrates that the PCCS pool water level does not drop below the top of the PCCS heat exchangers for over 72 hours after onset of core damage. The failure probability for W1 is a conditional probability derived from fault tree modeling. There is some dependency on the initiating event because the suppression pool cooling system requires power to operate; the initiator impact is addressed in the Level 1 sequences that comprise the CET entry events.

After 24 hours, it is conservatively assumed that the PCCS pool must be replenished by opening valves to the moisture separator storage pools or providing make-up water from the Fire Protection System. Upon connecting the additional pools, there is adequate water to maintain containment heat removal for the longer term, defined as 24 to 72 hours after event initiation. Long-term containment heat removal is modeled as event W2. As with W1, the failure probability for event W2 is a conditional probability derived from fault tree modeling.

The W2 event frequency is dependent on the initiating event because DC power is required to open the valves to the additional water source and the suppression pool cooling system requires AC power to operate.

8.2.1.3.10 Top Event VT

Top event VT models operator action to prevent containment failure by use of a suppression chamber vent path. If Event VT succeeds, the release path is controlled and directed through the suppression pool where significant filtering can occur to reduce the potential source term.

As discussed earlier, operator guidance for controlled venting has not yet been defined. However, insight into the ESBWR passive containment capability, and the need for venting, can be gained by evaluating a severe accident scenario in which there is no containment heat removal (i.e., event W1 is failed). From the Level 1 analysis discussed in Section 7, the sequence that dominates the core damage frequency is a transient in which the RPV is successfully depressurized. For such a sequence, Figure 8.2-3 illustrates that, for a dominant Class I contributor to the core damage frequency, the containment pressurizes to less than 1.0 MPa within 24 hours. As will be shown in Section 8.3, similar results were obtained for representative Class III and IV sequences. Thus, it is very unlikely that controlled venting in the 24-hour period after the onset of core damage will be required to prevent containment overpressure failure for the sequences dominating the core damage frequency. Top event VT contains a dependency on the Level 1 analysis because certain Level 1 sequences include containment venting.

8.2.1.4 Release Categories

Completion of a path through the event tree presented in Figure 8.2-1 provides the necessary information to establish categories for potential radionuclide release to the environment. A release category descriptor for each path is shown in Figure 8.2-1 in the column headed "Rel Cat". The release categories differ in the timing of containment breach and the magnitude of the radionuclide source term. By at least two orders of magnitude, the most likely path through the CET results in an intact containment with the source term being associated with containment leakage up to the limit allowed by Technical Specifications. This release category is termed "TSL". The release categories associated with the CET presented in Appendix 8A are discussed in more detail in the following sections. Drawing on the quantification presented in Appendix 8A, the probability of each release category is summarized in Table 8.2-2.

Direct Containment Heating (DCH)

The release category "Direct Containment Heating" is a result of highly energetic phenomenological effects during RPV failure at high pressure. As discussed in Section 21, containment failure due to DCH is considered physically unreasonable. However, a sensitivity study is documented in Section 11 that assigns a probability of 1E-3 to containment failure upon RPV rupture at high pressure.

Ex-Vessel Explosion (EVE)

The release category "Ex-Vessel Explosion" is a high-energy phenomenon that occurs when the RPV fails at low pressure and the core falls into an appropriate depth of water. The conditions required, and likelihood of, an EVE event are discussed in detail in Section 21. The total frequency of the EVE release category is approximately two orders of magnitude below TSL.

Dry Core-Concrete Interaction (CCID)

Core-concrete interaction is a phenomenological failure mode of the containment discussed in detail in Section 21. The general containment failure mode is overpressure as a result of non-condensable gas generation. The CCID release category occurs when the GDCS deluge system fails its function of water injection to the lower drywell. The frequency of the CCID release category is more than four orders of magnitude below TSL.

Wet Core-Concrete Interaction (CCIW)

The containment failure mode of CCIW is the same as that in CCID. The only difference in the CCIW sequences is that the GDCS deluge system has successfully injected to the lower drywell, but the water is unsuccessful in cooling the ejected core. The split fraction representing this probability is developed in Section 21. The frequency of the CCIW release category is approximately two orders of magnitude below TSL.

Containment Bypass (BYP)

The release category "Bypass" represents those sequences in which containment isolation has not occurred due to failure of the containment isolation system. Thus, the BYP failure mode provides for a direct path from the containment to the environment and results in an earlier environmental release than an overpressure event. Due to the reliability of the containment isolation system, the probability of such a release occurring is approximately three orders of magnitude less than the TSL release category. According to the analysis, the only credible failure mode for the containment isolation system is the common cause failure Reactor Protection System (RPS) components to support MSIV isolation.

Filtered Release (FR)

The release category "Filtered Release" represents those sequences in which the suppression chamber vent is used to control the containment pressure and potential release point. In such a situation, the containment boundary remains under operator control. As a result, the magnitude of the release is much less than if the containment were to fail because the release path is through the suppression pool, which provides significant radionuclide filtering.

As indicated earlier, in the 24-hour period after onset of core damage, the ESBWR containment would likely not require venting even in the absence of containment heat removal for the sequences that dominate the core damage frequency. Although venting is not likely to be required in the 24-hour period after onset of core damage, the option is maintained in the containment event tree. Treating the possibility of FR in this way accounts for uncertainties in the loss of heat removal analysis and containment venting guidance, and provides a conservative estimate of the likelihood of a controlled release. Further, inclusion of venting on the CET allows for modeling a period longer than 24 hours after the onset of core damage. Although the probability of the FR release category was calculated as more than four of magnitude less than the TSL release category, it will still be considered in Section 9 as a potential source term.

Overpressurization (OP)

The release category "Overpressurization" represents those sequences in which there has been inadequate post-accident heat removal resulting in the containment pressure exceeding the ultimate containment strength. Two categories of overpressure failure are considered. The category "OPW" applies to severe accident sequences in which the vapor suppression function is successful and only the containment heat removal function has failed. Both early (OPW1) and late (OPW2) failures of containment heat removal are considered. The category "OPVB" applies to sequences in which the vapor suppression function fails; in that situation, the containment heat removal function is also failed. As indicated in Table 8.2-2, the total probability of the overpressure failure mode (OPW1, OPW2 and OPVB) is about three orders of magnitude less likely than the TSL failure mode. Much of the OP release category frequency is derived from Level 1, Class II sequences. This frequency is eventually excluded from the offsite consequences analysis, as discussed in Subsection 8.3.2.2. Each subcategory is discussed below.

OPVB: The release category "OPVB" applies to sequences in which vacuum breaker failure has occurred. Failure of the vacuum breakers to close, or to be open in a pre-existing condition, results in failure of the containment vapor suppression function. If the vacuum breakers fail to function effectively, the overpressurization occurs fairly early in the severe accident sequence because the vapor suppression function is not effective. The high reliability of the vacuum breakers necessary for the vapor suppression function is demonstrated in that the calculated probability of the OPVB release category is more than three orders of magnitude less than the TSL release category. The Level 1 dependency of this node is captured by using the "DS-TOPVB" gate from the Level 1 model.

OPW1: The release category "OPW1" applies to sequences in which containment heat removal fails within 24 hours after event initiation. In such sequences, vapor suppression has been successful, but the passive PCCS system is unavailable as well as the active FAPCS in suppression pool cooling (SPC) mode, either of which provides the capability to remove energy from the containment. The 24-hour transition point from W1 to W2 was selected to correspond with the design requirement regarding the amount of water available to the PCCS cubicles without connection to a supplemental pool source. The Level 1 dependency is captured by using the same gates, "WP-TOPDHR", "DL-TOPVB", and "WS-TOPSPC" that are used in the Level 1 analysis. Because of the reliability of the SPC mode of the FAPCS and the passive PCCS, the calculated probability of the OPW1 release category is three orders of magnitude less than the TSL release category.

OPW2: The release category "OPW2" applies to sequences in which containment heat removal fails between 24 and 72 hours. In such sequences, the passive PCCS system becomes unavailable after PCCS pool dryout due to failure to connect to supplemental water pools; FAPCS availability was also evaluated at this time. Because of the minimum system requirements to provide additional water to the PCCS pools, long term heat removal (>24 hours) is very reliable. Although the probability of the OPW2 release category is more than four orders of magnitude less than the TSL release category, it will conservatively be considered in Section 9 as a potential source term.

Containment failure due to overpressurization is conservatively modeled as a direct path from the drywell to the environment. Thus, potential uncertainty in the location of the failure point is accommodated by the assumption of a direct path to the environment if the containment is overpressurized.

Technical Specification Leakage (TSL)

The release category "Technical Specification Leakage" represents those sequences in which there is neither a release due to containment failure nor a controlled filtered venting. The TSL release category provides a source term that exceeds that associated with normal operation because of the severe accident conditions within the containment. It is assumed that the leakage area corresponds to the Technical Specification allowable containment leakage rate of 0.5% of containment air volume per day at rated pressure.

The leakage path was conservatively assumed to occur between the drywell atmosphere and environment. Thus, no credit is taken for source term reduction if the leakage could be affected by potential refueling pool scrubbing or the reactor building HVAC system.

8.2.1.5 Release Category Frequency and Containment Effectiveness

The frequencies of the release categories are calculated by quantifying all of the CET with the single top Level 2 model. The individual release categories can be calculated by multiplying the Fussel-Vessely of that release category's marker (flag) by the overall release frequency, or CDF. As seen in Table 8.2-2, the most likely release category is that associated with leakage from an intact containment, TSL. Release categories associated with containment failure due to phenomenological or system failure events are at least two orders of magnitude less likely than the TSL release category.

The release categories associated with containment failure are so much lower than the TSL category, and their calculated probabilities are so low on an absolute basis, that containment failure due to overpressurization or bypass in the 24-hour period after the onset of core damage is not considered credible. Thus, it is clear that the ESBWR provides a reliable barrier to radionuclide release. This conclusion is reflected in the quantification of containment effectiveness. The containment effectiveness can be conservatively quantified as the probability of release category TSL (i.e., an intact containment) divided by the core damage frequency. Using the values from Appendix 8A, Table 8A-3, and applying "ε" for probabilities less than 1E-15,

$$Containment\_Effectiveness = \frac{TSL\_frequency}{Level\_1\_CDF} = \frac{1.12E-8}{1.22E-8} = 0.921$$

Table 8.2-1

Summary of CET Initiating Probabilities

CET	LDW Water Level	Entry Gate*	Total Entry Gate Frequency
Class I – Low (RPV failure at low pressure)	Low/Dry	I_LD	4.94E-09
Class I - High (RPV failure at low pressure, high water level in LDW)	High	I_H	5.88E-10
Class I – Medium (RPV failure at low pressure, high water level in LDW)	Medium	I_M	9.92E-11
Class III (RPV failure at high pressure)	Low/Dry	III_LD	4.51E-09
Class IV – Low (ATWS with RPV failure at low pressure)	Low/Dry	IV_LD	1.85E-09
Class IV – High (ATWS with RPV failure at low pressure, high water level in LDW)	High	IV_H	2.18E-11
Class IV – Medium (ATWS with RPV failure at low pressure, medium water level in LDW)	Medium	IV_M	7.25E-13

*Nomenclature used in event tree quantification provided in Appendix 8A.

Table 8.2-2

CET Release Category Frequencies

Release category	Frequency (per year)*
TSL	1.12E-8
FR	ε
BYP	5.6E-11
OPVB	1.6E-11
OPW1	3.2E-11
OPW2	ε**
CCIW	9.9E-11
CCID	1E-12
EVE	6.10E-10
DCH	0.00
BOC (from Level 1)	1.47E-10

*The frequency is the summed contribution to the release category from all accident classes, as shown in Table 8A-3. BYP is also augmented with frequency from de-inerted operation, as described in 8.1.4.

**Calculated frequencies less than 1E-12 are reported as "ε". The actual calculated number is preserved for input to the offsite consequences analysis in Section 10.

Figure 8.2-1. Containment System Event Tree (Deleted)

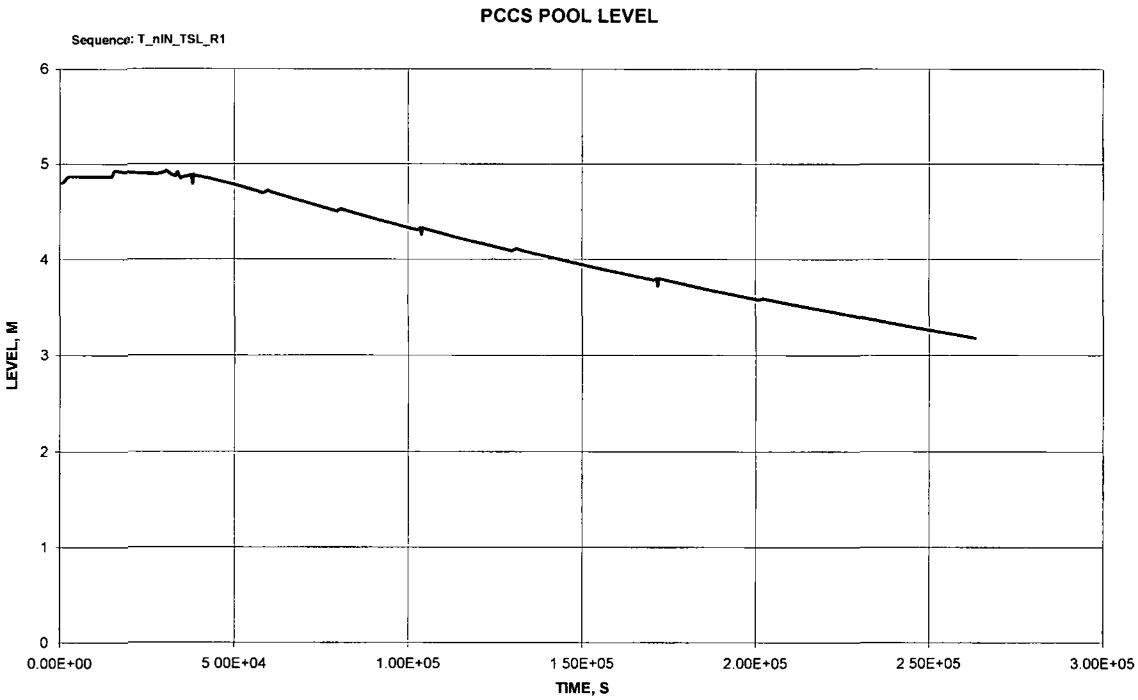

Figure 8.2-2. PCCS heat removal capability for 24 hour period

Example shown is for a dominant Class I sequence, a transient followed by loss of core injection. The PCCS heat exchangers remain covered for more than 24 hours after onset of core damage.

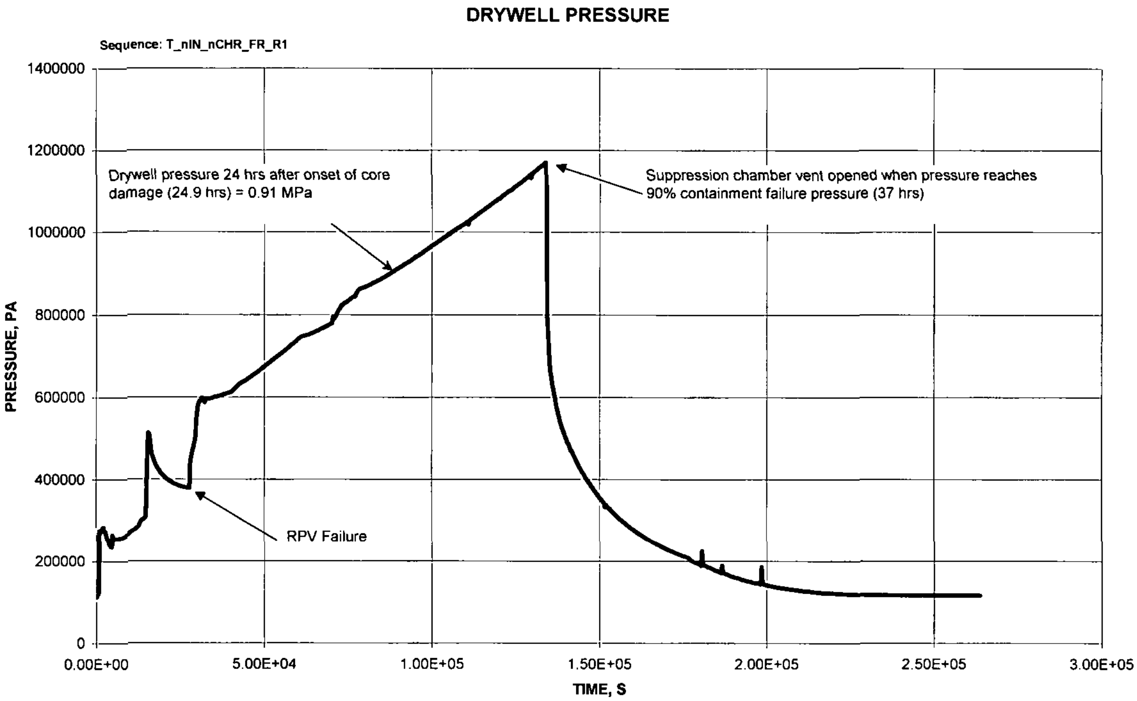

Figure 8.2-3. Containment Pressure with No Containment Heat Removal

Example shown is for a dominant Class I sequence, a transient followed by loss of core injection. With vapor suppression function successful, containment does not pressurize to failure within 24 hours after onset of core damage (8.64E+04 seconds).

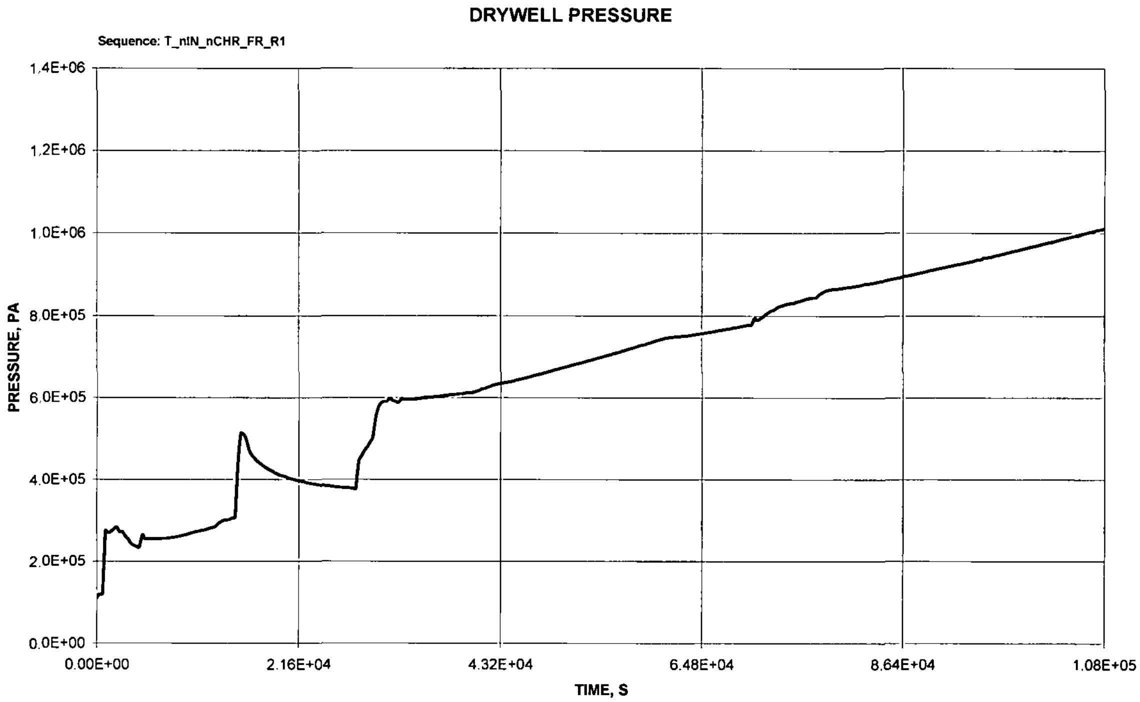

Figure 8.2-4. Drywell Pressure with No Containment Heat Removal

Example shown is for a Class I sequence with a failure of containment heat removal. As shown in the chart, containment does not pressurize to failure within 24 hours after onset of core damage (8.64E+04 seconds).

8.3 CONTAINMENT PERFORMANCE AGAINST OVERPRESSURE

To determine the key characteristics of the containment response to a severe accident, an ESBWR simulation model was developed. The model is used to gain insights into the timing of severe accident progression, the containment pressure-temperature response and ultimately the potential source term if the containment were to fail. As demonstrated in the prior section, the reliability of containment systems designed to mitigate a severe accident is such that the calculated probabilities of containment bypass and overpressure failure due to system failures are so small that they may be considered negligible. Thus, only the TSL and FR release categories are discussed in this section. Hypothetical scenarios in which the containment is bypassed, overpressurizes, or fails due to phenomenological effects are considered in the evaluation of potential source terms, as presented in Section 9.

Analysis of the ultimate strength of the containment indicates that the drywell head is the most likely failure location if the containment were to overpressurize. The analysis also illustrates that the containment pressure capability is a function of temperature. This pressure capability profile was used in the simulation modeling.

Subsection 8.3.1 summarizes the code used for accident simulation. Subsection 8.3.2 provides the simulation results for a spectrum of potential severe accidents representing each accident class. The ultimate containment strength analysis is provided in DCD Sections 19B and 19C.

8.3.1 Simulation Code

The ESBWR was modeled using a computer code capable of modeling the integrated plant response to a severe accident. The code used for this purpose is the Modular Accident Analysis Program code (MAAP), Version 4.0.6, Reference 8.3-1. The code was developed as part of the Industry Degraded Core Rulemaking (IDCOR) program to investigate the physical phenomena that might occur in the event of a severe light water reactor accident leading to core damage, possible RPV failure, and ultimately possible failure of containment integrity and release of fission products to the environment. MAAP development was sponsored by the Atomic Industrial Forum. MAAP includes models for the important phenomena that might occur in a severe light water reactor accident.

MAAP has a long history of use in severe accident analysis, including severe accident analysis of the ABWR as described in Reference 8.3-2, which was based on an earlier version of MAAP. MAAP requires that phenomenological information and plant specific design characteristics be provided in the form of a parameter file. Parameter file inputs related to accident phenomenology were based on the values provided in MAAP sample files, which are maintained for the MAAP Users Group; these values were provided by the code developer. Parametric values related to the ESBWR design were based on review of design documentation information, as it was available in February 2005. In some cases, design information was updated between February and August 2005 when significant design decisions were made.

8.3.2 Sequences Representative of Each Accident Class

As discussed in earlier sections, severe accidents were grouped in five categories in the Level 1 analysis. The Level 1 analysis results were reviewed to identify sequences which were dominant contributors to the core damage frequency. With the exception of Class V accidents, in which the containment is completely bypassed, a single dominant sequence was selected to represent each of the accident classes for detailed modeling. In this way, the containment response to the complete spectrum of accidents contributing to the core damage frequency could be evaluated.

Table 8.3-1 identifies the sequences that were used to represent each accident class. The "core damage sequence descriptor" used in the table derives from the results of the Level 1 analysis. Table 7.2-3 identified the sequences which were significant contributors to the core damage frequency. The representative sequences shown in Table 8.3-1 are based on the Level 1 results presented in Table 7.2-3 and the definitions of the Level 1 sequence bins. For example, Table 7.2-3 indicates that about 80% of the Class I frequency is associated with a stuck open relief valve (T-IORV), a large feedwater LOCA (LL-S-FDWA/B), or loss of feedwater (T-FDW) sequences. From the perspective of modeling the containment response to a severe accident, all Class Isequences can be represented as a transient with loss of injection T-nIN and successful depressurization. A similar approach was used in selecting the representative sequence for the other accident classes. Table 8.3-1 provides a summary description for each representative sequence.

Table 8.3-2 couples the representative core damage sequence with one of the release categories illustrated on the containment system event tree, Figure 8.2-1. The resulting scenario is assigned a "containment response sequence descriptor" to summarize the core damage and containment release information. Recalling that Table 8.2-2 provided the total contribution of all accident classes to each release category frequency, Table 8.3-2 provides additional information by presenting the release category frequency in terms of the contribution from each accident class. As indicated in the table, there is a negligible probability of a core damage sequence resulting in overpressure or bypass failure. However, such hypothetical scenarios are retained for evaluation in Section 9 to assure that a conservative source term is developed.

Graphs of many additional MAAP parameters are shown in Appendix 8B to provide complete documentation of the containment analysis.

8.3.2.1 Class I: Sequences with RPV Failure at Low Pressure

Accident Class I involves sequences in which the RPV fails at low pressure; this accident class represents approximately 46.16% of the core damage frequency. As indicated in Tables 7.2-3 and 7.2-5, the class is dominated by transient sequences in which there is no core injection. Thus, the sequence T_nIN described below was used to evaluate the containment response to Class I events.

8.3.2.1.1 Sequence T_nIN_TSL

The initiating event for the T_nIN sequence is a transient initiated by a loss-of-preferred power. No short or long-term coolant injection to the RPV by the feedwater, CRD or GDCS is available. The ADS functions to reduce the RPV pressure. As stated earlier, heat removal by the isolation condensers is not credited because of the low reactor pressure. Containment heat removal in the short-term is accomplished by successful PCCS functioning; PCCS pool makeup is successful, thus allowing long-term containment heat removal. The GDCS deluge system and BiMAC are available for debris bed cooling. With successful containment isolation, vapor suppression and containment heat removal the containment remains intact. Technical Specification leakage is the only mode of fission product release.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-1a through e show the system behavior throughout the accident sequence.

In this event, the primary system experiences delayed depressurization due to opening of the first ADS-actuated valves at about 655 seconds. The pressure in the containment increases as the drywell is filled with steam and heats up. About thirty minutes into the event, core uncovery occurs which results in fuel rod heatup and melting. Fission products and non-condensable gases are swept into the containment through the DPVs as the core melts. This leads to further heating and pressurization of the drywell air space.

The reactor pressure vessel lower head penetrations fail about 7.8 hours into the event. Corium is deposited on the lower drywell floor, which results in local temperatures that are high enough to cause the GDCS deluge line to open. As a result, the GDCS pool water drains into the lower drywell and covers the debris bed. Because the debris is quenched by the successful GDCS deluge and BiMAC function, significant core-concrete interaction does not occur. Therefore, no significant fission product aerosols or non-condensable gases are generated in the ex-vessel phase of the accident sequence.

Continued heating by debris of the water in the lower drywell leads to the temperature in the overlying water pool reaching saturation. Steam generation in the lower drywell then leads to further increases in the containment pressure until the PCCS heat removal capacity becomes consistent and comparable to the decay heat generated by the core debris. The containment pressure reaches about 0.58 MPa 24 hours after onset of core damage, well below the point at which containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

8.3.2.1.2 Sequence T_nIN_nCHR_FR

Sequence T_nIN_nCHR_FR is the same as the representative Class I sequence T_nIN, except that the containment response differs because containment heat removal has failed. As a result, containment pressurization increases and controlled venting may be implemented to limit the pressure rise and control the radionuclide release location. Specific guidance for the use of the suppression pool vent has not been developed. Indeed, as discussed earlier, venting in the ESBWR does not appear necessary to limit the containment pressure to less than its ultimate strength in the 24-hour period after core damage. The venting scenario is evaluated here to provide insights into vent timing and provide a basis for the FR release category used in the source term evaluation.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-2a through d show the system behavior throughout the accident sequence. The sequence proceeds as discussed in the previous section except that venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength. As indicated, in Figure 8.3-2b, the drywell pressure has reached less than 70% of the ultimate containment strength within 24 hours after onset of core damage; thus venting would not likely be implemented in this time frame. The 90% assumption is met at 32.7 hours, which is about 2.7 hours before containment overpressurization would occur if controlled venting were not implemented.

The sequence demonstrates that venting is not required to prevent containment failure in the 24-hour period after onset of core damage due to a Class I event, even if containment heat removal were unavailable. In such a scenario, there is a long time period after core damage to prepare for venting and take other mitigating actions.

8.3.2.2 Class II: Sequences with Containment Failure Preceding Core Damage

Accident Class II involves sequences in which containment failure precedes RPV failure. After containment failure, RPV makeup capability is assumed to be lost due to the gradual boiloff of water in the passive systems; potential damage to piping connections renders active makeup systems unavailable. As a result, core damage and RPV failure occur after containment failure As shown in representative sequence MLi_nCHR, core damage does not occur during the first 72 hours post-accident. Because core damage is beyond the Level 2 analysis mission time, Class II accident sequences are not included as inputs to the offsite dose consequences in Section 10. The Class II frequency is included in the release frequencies reported in table 8.2-2 so that the entire Level 1 CDF is accounted for..

The sequence MLi_nCHR was selected to represent the containment response to Class II events because the sequence provides containment pressurization due to the break and failure of the containment heat removal function.

8.3.2.2.1 Sequence MLi_nCHR

The initiating event for the sequence MLi_nCHR is a medium LOCA, assumed to occur in the GDCS injection line. Failure of containment heat removal is followed by containment pressurization to its ultimate capacity. Core cooling occurs by gravity feed through the GDCS injection and equalizing lines. Eventually, the water used for RPV makeup is boiled off.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-3a through c show the system behavior throughout the accident sequence. The figures illustrate that the containment pressurizes until the ultimate strength is reached at about 31 hours. The ADS depressurizes the RPV allowing GDCS tanks to drain into the RPV, then into the lower drywell through the break. The shroud water level initially rises in response to the GDCS tank injection, then decays as the GDCS inventory is depleted. The shroud level decreases below the elevation of the break at about six hours. Further, shroud level decrease occurs until flow through the equalizing line begins at about 8.3 hours. Flow from the suppression pool maintains RPV level above the top of active fuel for about 71 hours. Shortly thereafter, core heat up begins.

The results of the sequence simulation indicate that the core damage following containment failure due to loss of containment heat removal does not occur within a 24-hour period after accident initiation. In fact, core temperatures do not reach the point of fuel damage until more than 72 hours after accident initiation. Given the long time for mitigating action to supplement RPV makeup, Class II events are not considered contributors to the offsite consequence analysis However, Class II events are preserved in the release frequency calculation so that the total release is equivalent to CDF.

There are a few Class II accident sequences that do not consider the potential for low-pressure injection during the sequence on the Level 1 event trees. However, if GDCS were considered on these sequences, the failure of GDCS injection resulting in core damage would place the result below the truncation limit. As such, the selected representative sequence is considered to be appropriate.

Class II-a sequences are partitioned into the OPW1, OPW2, and OPVB release categories depending on which system failures resulted in core damage. The Class II-b sequences are binned as release category FR. All Class II frequencies are included in the total frequencies reported in Tables 8.2-2 and 8.3-2.

8.3.2.3 Class III: Sequences with RPV Failure at High Pressure

Accident Class III involves sequences in which the RPV fails at high pressure; this accident class represents approximately 37.02% of the core damage frequency. As indicated in Tables 7.2-3 and 7.2-5, the class is dominated by transient sequences in which there is no core injection. Thus, sequence T_nDP_nIN described below was used to evaluate the containment response to Class III events.

8.3.2.3.1 Sequence T_nDP_nIN_TSL

The initiating event for the sequence T_nDP_nIN is a loss-of-offsite power. The sequence differs from T_nIN in that depressurization fails, although the SRVs remain functional in the relief mode. The ICS was not credited. The CRD and Feedwater systems are unavailable. Because depressurization is unsuccessful, the RPV fails at high pressure, i.e., at the pressure controlled by the relief valve setpoint. GDCS deluge and BiMAC function to cool the debris bed in the lower drywell.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-4a and b summarize the system behavior throughout the accident sequence.

The RPV fails about 6.2 hours. Actuation of the GDCS deluge line and successful BiMAC function prevent significant core-concrete interaction from occurring in the lower drywell. Material dispersed to the upper drywell does not result in significant CCI because the large dispersal area allows the material to be cooled. Continued heating of the water by debris in the lower drywell leads to continued steam generation, which increases containment pressure. The PCCS removes heat from the containment, thus preventing overpressurization. The containment pressure reaches about 0.62 MPa 24 hours after onset of core damage, well below the point at which containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

8.3.2.3.2 Sequence T_nDP_nIN_nCHR_FR

Sequence T_nDP_nIN_nCHR is the same as sequence T_nDP_nIN except that containment heat removal has failed. As a result, containment pressurization increases and controlled venting is implemented to limit the pressure rise and control the radionuclide release point. As indicated earlier, specific guidance for the use of the suppression pool vent has not been developed, thus, venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-5a and b show the system behavior throughout the accident sequence. As indicated, in Figure 8.3-5b, the drywell pressure has reached less than 70% of the ultimate containment strength within 24 hours after onset of core damage; thus venting would not likely be implemented in this time frame. The 90% assumption is met at 42.5 hours after accident initiation, which is about 2.9 hours before containment overpressurization would occur.

The sequence demonstrates that venting is not required to prevent containment failure in the 24-hour period after onset of core damage due to a Class III event, even if containment heat removal were unavailable. In such a scenario, there is a long time period after core damage to prepare for venting and take other mitigating actions.

8.3.2.4 Class IV: Sequences with Failure to Insert Negative Reactivity

Accident Class IV includes sequences that are initiated by an ATWS and followed by failure to initiate negative reactivity. Such sequences represent approximately 15.6% of the core damage frequency. From the Level 1 analysis summarized in Table 7.2-3, the largest Class IV contributor to the core damage frequency is a general transient followed by failure to scram. Thus, the sequence termed T-AT_nIN, which defines the ATWS initiator with no core injection, was selected to evaluate the containment response to Class IV events.

8.3.2.4.1 Sequence T-AT_nIN_TSL

Sequence T-AT_nIN is a general transient followed by ATWS. The standby liquid control system is ineffective or unavailable. The RPV is not initially depressurized because ADS inhibit is successful. To control the ATWS power level, feedwater runback is successful with operator control assumed at the top of active fuel. The PCCS is available, but no active containment heat removal (FAPCS) is assumed.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-6a through c show the system behavior throughout the accident sequence.

In this sequence, feedwater runback is successful. Control of core water level just above the top of active fuel results in a core power level of about 30% full power three minutes after the transient begins. At that time, it is assumed that feedwater is terminated and safety system injection to the RPV does not occur. (System pressure prevents gravity drain from the GDCS and the CRD system is unavailable for forced flow.) Because the ADS inhibit is successful, the RPV is maintained at high pressure, controlled by the SRV setpoint, until the core water level decreases below the point of effective cooling. At that point, manual depressurization is initiated, but injection into the RPV continues to be unsuccessful. RPV failure occurs at about 5.9 hours at low pressure.

Actuation of the GDCS deluge line and successful BiMAC function prevent significant CCI from occurring in the lower drywell. Material dispersed to the upper drywell does not result in significant CCI because the large dispersal area allows the material to be cooled. Continued heating by debris of the water in the lower drywell leads to continued steam generation, which increases containment pressure. The PCCS removes heat from the containment, thus preventing overpressurization. The containment pressure reaches about 0.57 MPa 24 hours after onset of core damage, well below the point at which containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

8.3.2.4.2 Sequence T-AT_nIN_nCHR_FR

Sequence T-AT_nIN_nCHR_FR is the same as sequence T-AT_nIN except that containment heat removal has failed. As a result, containment pressurization increases and controlled venting is implemented to limit the pressure rise and control the radionuclide release point. As indicated earlier, specific guidance for the use of the suppression pool vent has not been developed, thus, venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength.

The key events of the sequence are summarized in Table 8.3-3. Figure 8.3-7a shows the containment response for the accident sequence. As indicated in the figure, the containment pressure 24 hours after the onset of core damage is about 1.1 MPa, within the pressure retaining capability of the containment. The 90% assumption for action to initiate controlled venting is met at about 29 hours after accident initiation.

The sequence demonstrates that venting is not required to prevent containment failure in the 24-hour period after onset of core damage due to a Class IV event, even if containment heat removal were unavailable. In such a scenario, there is a long time period after core damage to prepare for venting and take other mitigating actions.

8.3.2.5 Class V: Sequences with Interfacing LOCA

Because Class V sequences are associated with a direct path from the RPV to the environment the containment response is not relevant to preventing a radionuclide release. The risk of such low probability events is accounted for by defining a release category, "BOC" for break-outside-of-containment, and assigning a frequency in the source term analysis, as discussed in Section 9.0.

Table 8.3-1

Representative Core Damage Sequences

Accident Class	Core Damage Sequence Descriptor	Sequence Summary
I	T_nIN	Transient initiator followed by no short or long-term coolant injection. ADS functions. ICS not credited. PCCS available, but no active containment heat removal (FAPCS). GDCS/BiMAC function successful.
II	MLi_nCHR	Medium liquid line break: GDCS injection line. System is depressurized and injection systems function. Containment heat removal not available.
III	T_nDP_nIN	Transient initiator followed by no short or long-term coolant injection. The RPV is not depressurized; pressure controlled at relief valve setpoint. ICS not credited. PCCS available, but no active containment heat removal (FAPCS). GDCS/BiMAC function successful.
IV	T-AT_nIN	Transient followed by failure to insert negative reactivity. ICS not credited. RPV is not initially depressurized (ADS inhibit successful). SLC is ineffective or unavailable. FW runback is successful. No short or long-term coolant injection. PCCS available, but no active containment heat removal (FAPCS). GDCS/BiMAC function successful.
V	None	No representative sequence assigned for containment evaluation as Class V events involve direct communication between the RPV and environment.

Table 8.3-2

Representative Containment Response Sequence

Containment Response Sequence Descriptor	Release Category	Frequency* (per reactor-year)	Containment Response Summary
T_nIN _TSL	TSL	9.45E-09	Release path from drywell through area associated with Technical Specification leakage. All containment systems function effectively.
_nCHR _FR	FR	ε	Release path through wetwell vent. Containment heat removal function failed.
_nCHR_W1	OPW1	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails early (<24 hrs); no controlled venting.
_nCHR_W2	OPW2	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails late (>24 hrs); no controlled venting.
_nVB	OPVB	6E-12	Release path from drywell through area large enough to depressurize containment. Vapor suppression, containment heat removal and controlled venting functions failed.
_BYP	BYP	2E-12	Release path from drywell through open line connecting drywell atmosphere to environment
_CCIW	CCIW	4.4E-11	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_CCID	CCID	ε	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_EVE	EVE	5.88E-10	Release path from drywell through large area immediately following RPV rupture.
MLi_nCHR	(cdii)	4.2E-11	Release path from drywell through area large enough to depressurize containment. Containment heat removal not available.
T_nDP_nIN_TSL	TSL	4.52E-09	Release path from drywell through area associated with Technical Specification leakage. All containment systems function effectively.
nCHR_FR	FR	ε	Release path through wetwell vent. Containment heat removal function failed.
_nCHR_W1	OPW1	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails early (<24 hrs); no controlled venting.
_nCHR_W2	OPW2	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails late (>24 hrs); no controlled venting.

Table 8.3-2

Representative Containment Response Sequence

Containment Response Sequence Descriptor	Release Category	Frequency* (per reactor-year)	Containment Response Summary
_nVB	OPVB	ε	Release path from drywell through area large enough to depressurize containment. Vapor suppression, containment heat removal and controlled venting functions failed.
_BYP	BYP	4E-12	Release path from drywell through open line connecting drywell atmosphere to environment
_CCIW	CCIW	3.7E-11	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_CCID	CCID	ε	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_DCH	DCH	0.00	Containment failure mode not considered credible.
T-AT_nIN_TSL	TSL	1.84E-09	Release path from drywell through area associated with Technical Specification leakage. All containment systems function effectively.
nCHR_FR	FR	ε	Release path through wetwell vent. Containment heat removal function failed.
_nCHR_W1	OPW1	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails early (<24 hrs); no controlled venting.
_nCHR_W2	OPW2	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails late (>24 hrs); no controlled venting.
_nVB	OPVB	ε	Release path from drywell through area large enough to depressurize containment. Vapor suppression, containment heat removal and controlled venting functions failed.
_BYP	BYP	1.7E-11	Release path from drywell through open line connecting drywell atmosphere to environment
_CCIW	CCIW	1.8E-11	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_CCID	CCID	ε	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_EVE	EVE	2.2E-11	Release path from drywell through large area immediately following RPV rupture.

Table 8.3-2

Representative Containment Response Sequence

Containment Response Sequence Descriptor	Release Category	Frequency* (per reactor-year)	Containment Response Summary

Notes:

"Frequency" indicates contribution from all sequences in accident class, not just the representative sequence. Refer to Table 8A-3 for additional detail regarding release category frequency.

"ε" refers to a calculated frequency of <1.0E-15.

Table 8.3-3

Summary of Results of Severe Accident Sequence Analysis

Sequence Descriptor	RPV Depressurization Initiated (seconds)	Core Uncovered (hours)	Onset of Core Damage (hours)*	RPV Failure (hours)	Deluge Actuated (hour)	Concrete Ablation 24 hrs. after onset of core damage (meters)	Drywell Pressure 24 hrs. after onset of core damage (MPa)	Containment Vent (hours after onset of core damage)
T_nIN _TSL	665	0.50	1.1	7.8	7.8	0.05	0.58	NA
T_nIN_nCHR_FR	661	0.48	1.3	7.7	7.7	0.05	0.92	>24
MLi_nCHR	124	>72	>72	>72	NA	NA	NA	NA
T_nDP_nIN _TSL	NA	0.92	1.7	6.2	6.2	<0.1	0.57	NA
T_nDP_nIN_nCHR _FR	NA	0.93	1.7	6.7	6.7	<0.1	1.01	>24
T-AT_nIN_TSL	1163	0.1	0.77	5.9	6.0	0.1	0.57	NA
T-AT_nIN_nCHR_FR	1161	0.1	0.77	5.7	5.7	<0.1	1.1	>24

Key:

MLi: Medium Liquid break (injection line)

T: Transient

T-AT: Transient without negative reactivity insertion

nCHR: No containment heat removal

nDP: No depressurization

nIN: No injection

FR: Filtered release (controlled vent)

TSL: Technical Specification Leakage

NA: Not Applicable

*Time of maximum core temperature > 2499°K

Figures 8.3-1a through e: Sequence T_nIN_TSL

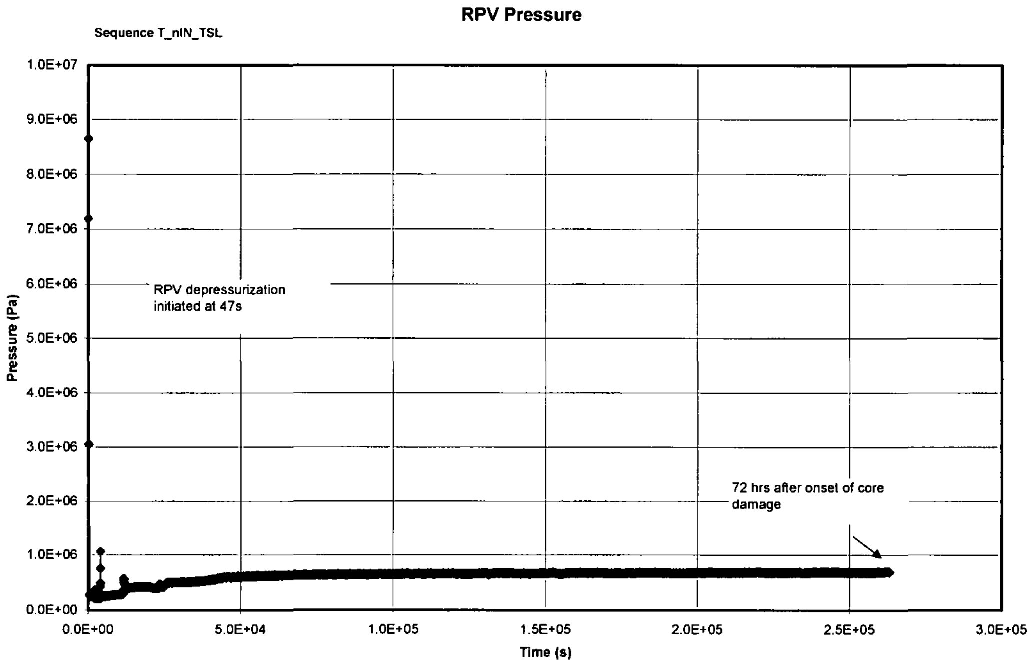

Figure 8.3-1a. T_nIN_TSL: RPV Pressure vs. Time

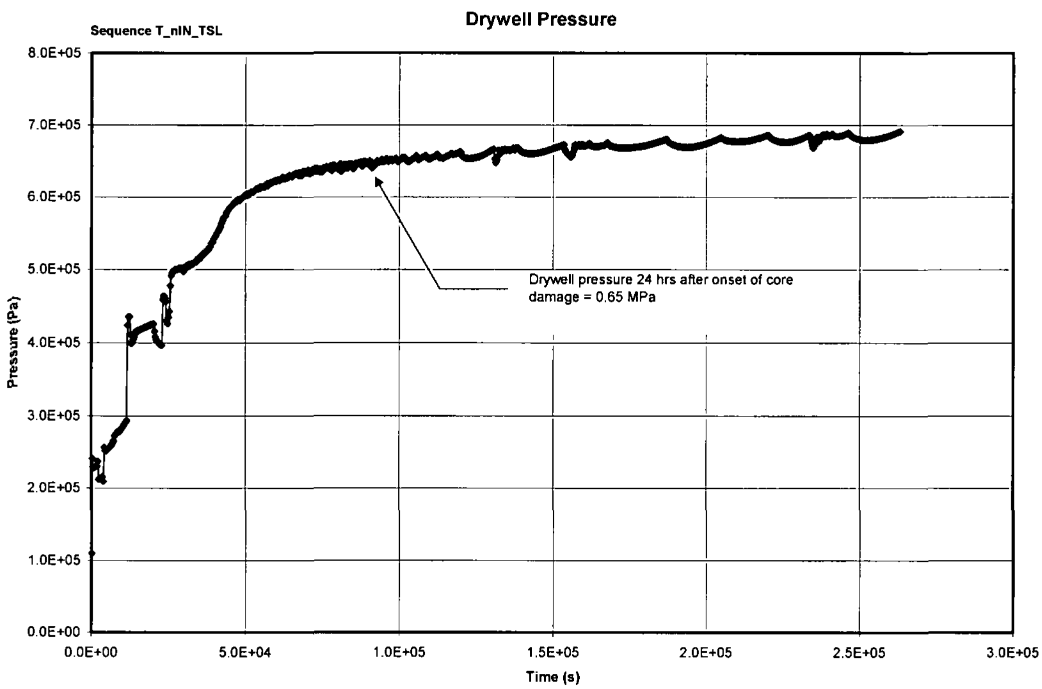

Figure 8.3-1b. T_nIN_TSL: Containment Pressure vs. Time

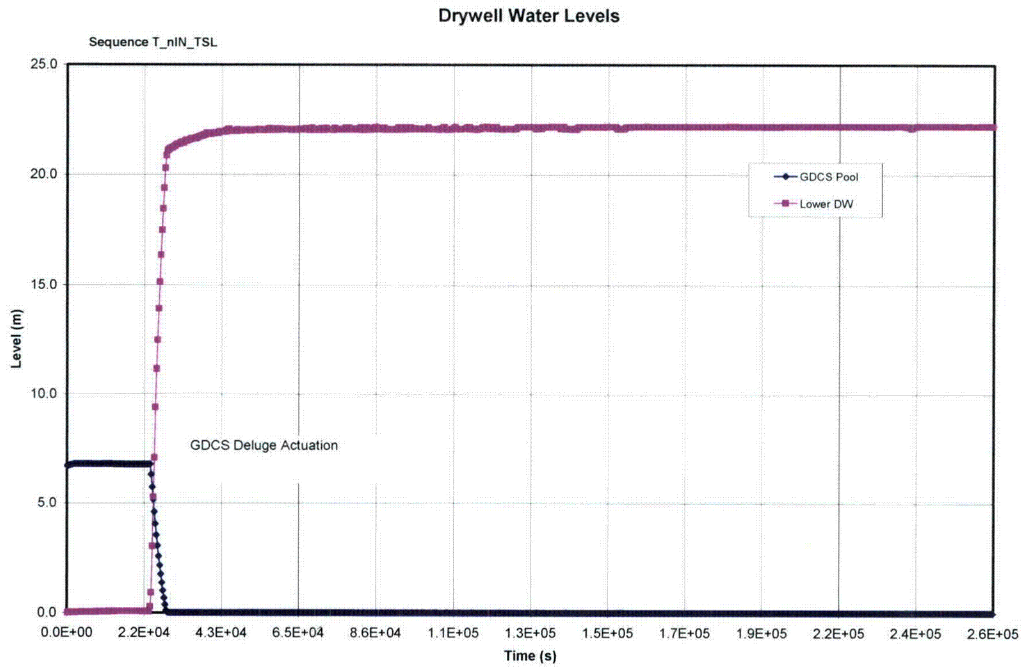

Figure 8.3-1c. T_nIN_TSL: Lower Drywell Temperature vs. Time

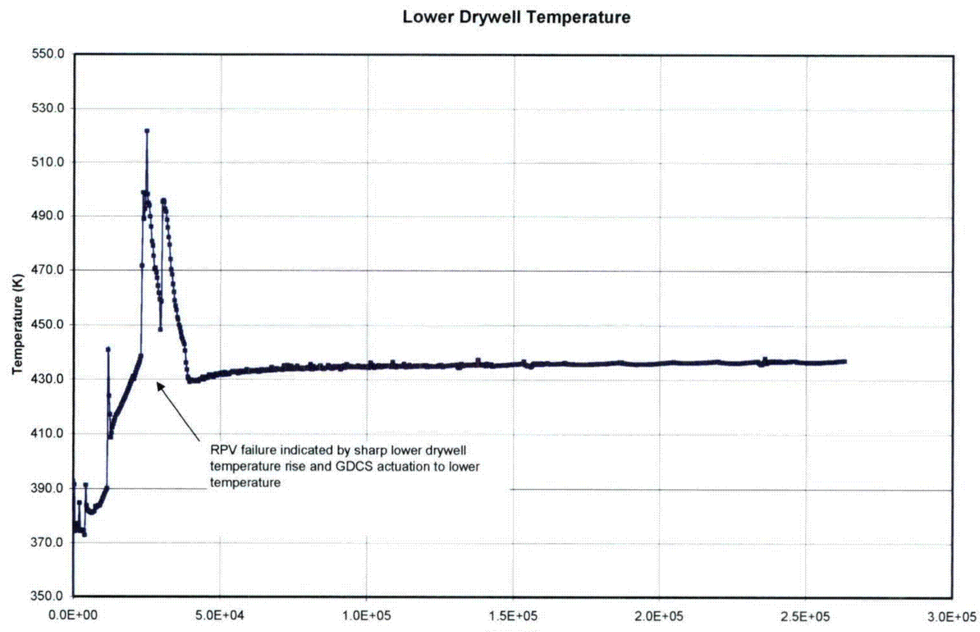

Figure 8.3-1d. T_nIN_TSL: Drywell Water Levels vs. Time

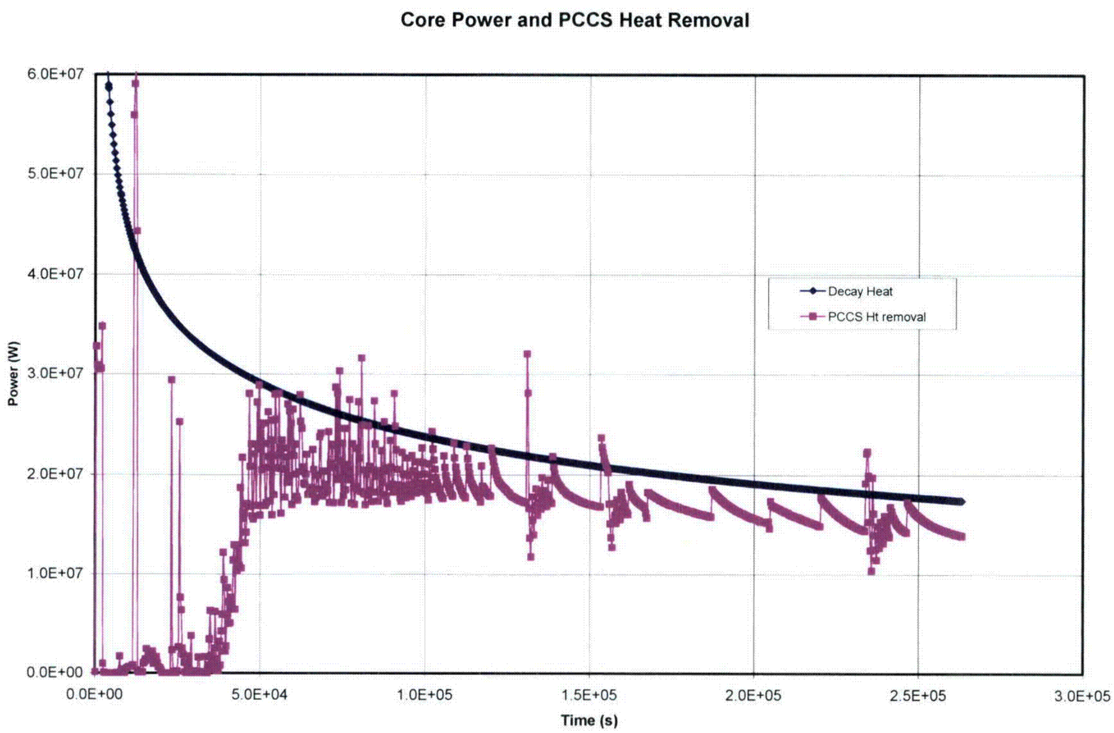

Figure 8.3-1e. T_nIN_TSL: Core Power and PCCS Heat Removal vs. Time

Figures 8.3-2a through d: T_nIN_nCHR_FR

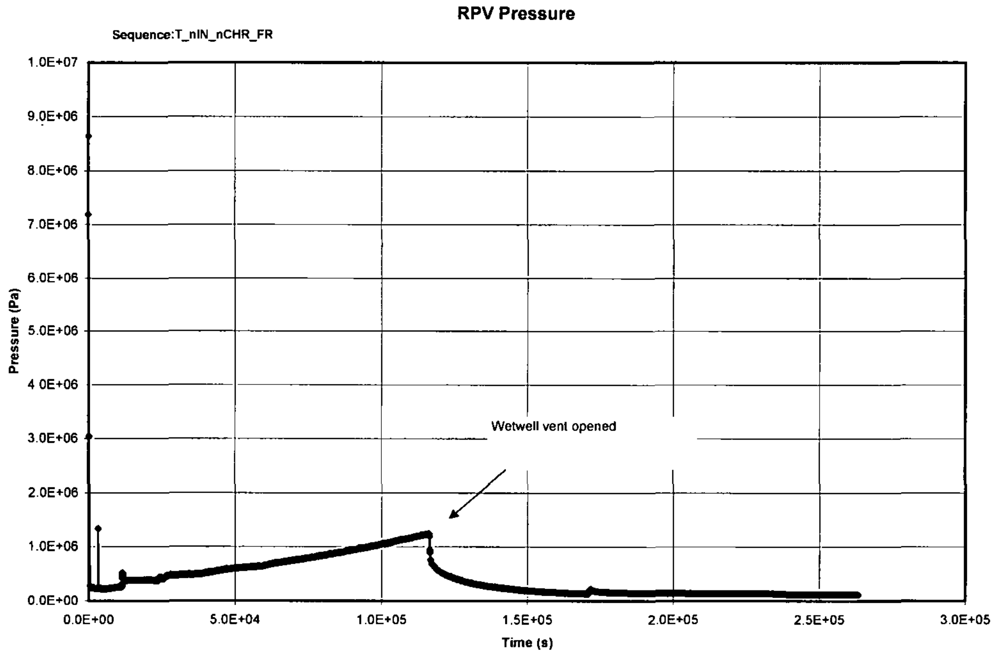

Figure 8.3-2a. T_nIN_nCHR_FR: RPV Pressure vs. Time

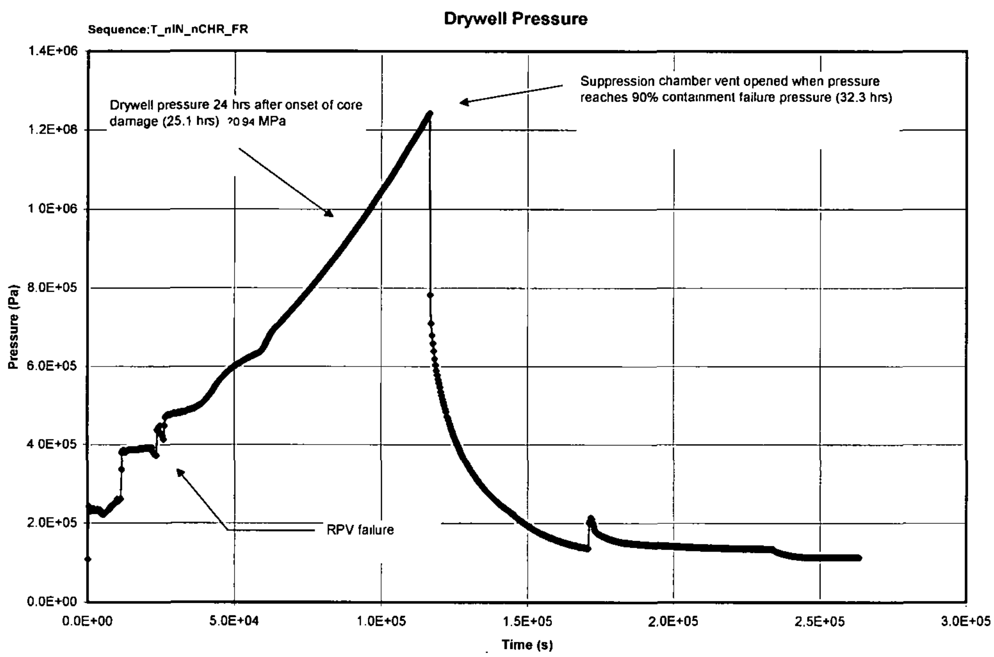

Figure 8.3-2b. T_nIN_nCHR_FR: Containment Pressure vs. Time

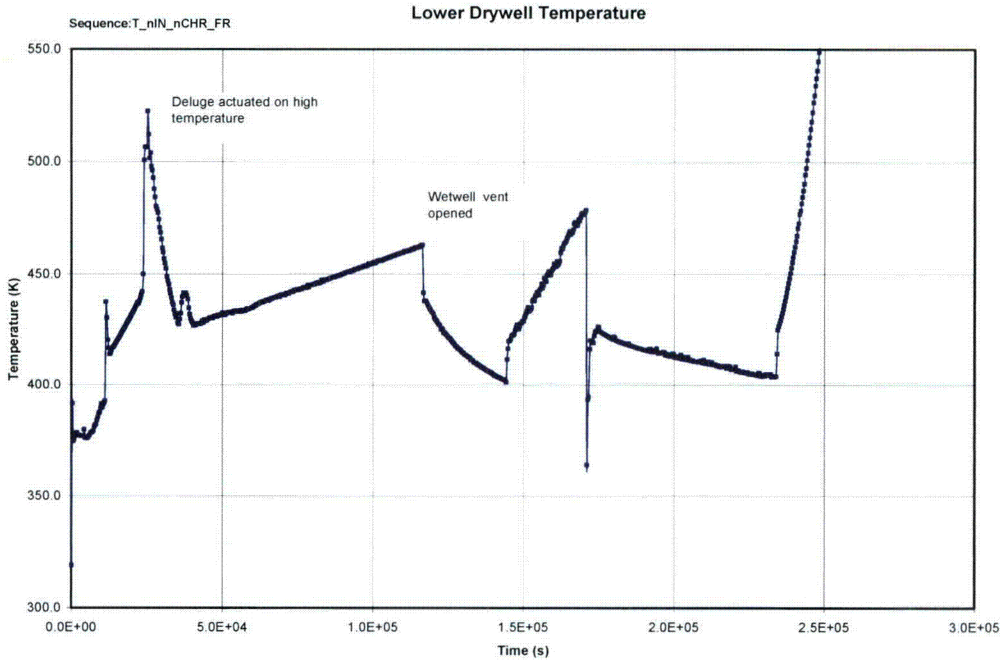

Figure 8.3-2c. T_nIN_nCHR_FR: Lower Drywell Temperature vs. Time

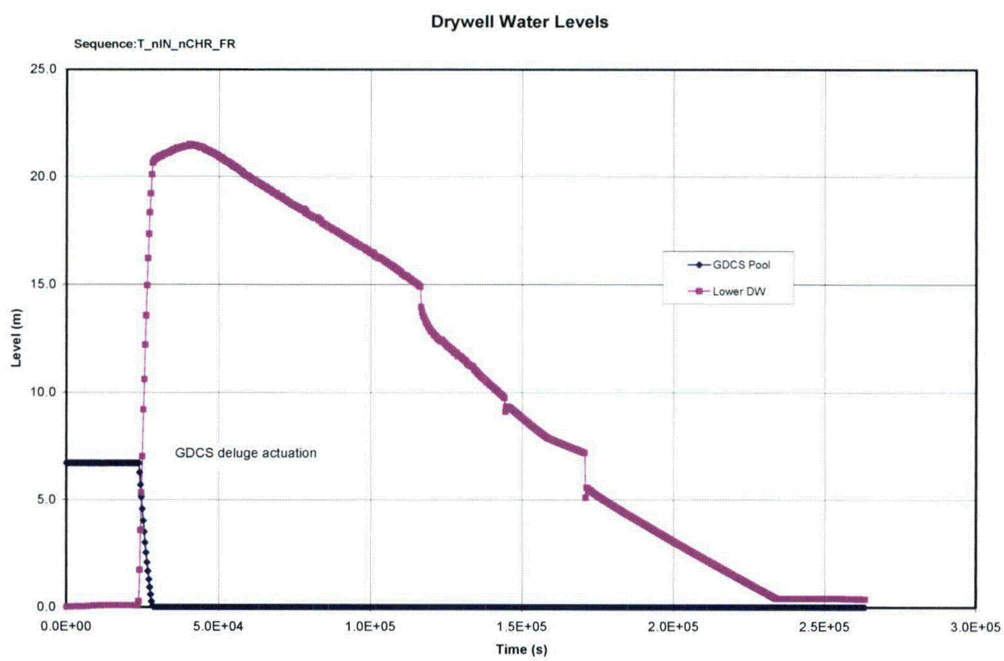

Figure 8.3-2d. T_nIN_nCHR_FR: Drywell Water Levels vs. Time

Figures 8.3-3a through c: MLi_nCHR

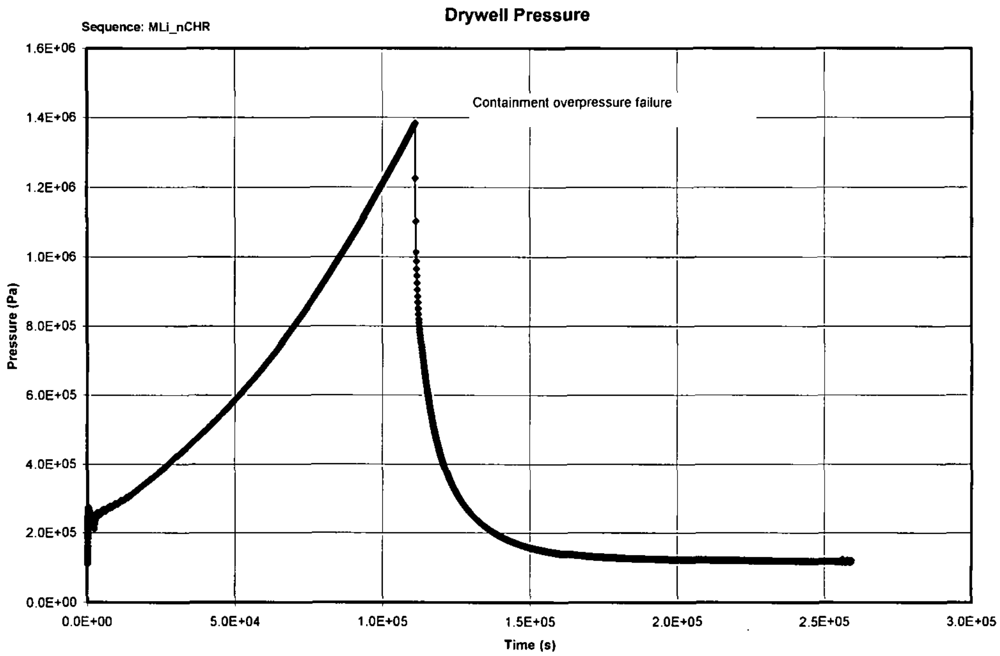

Figure 8.3-3a. MLi_nCHR: Containment Pressure vs. Time

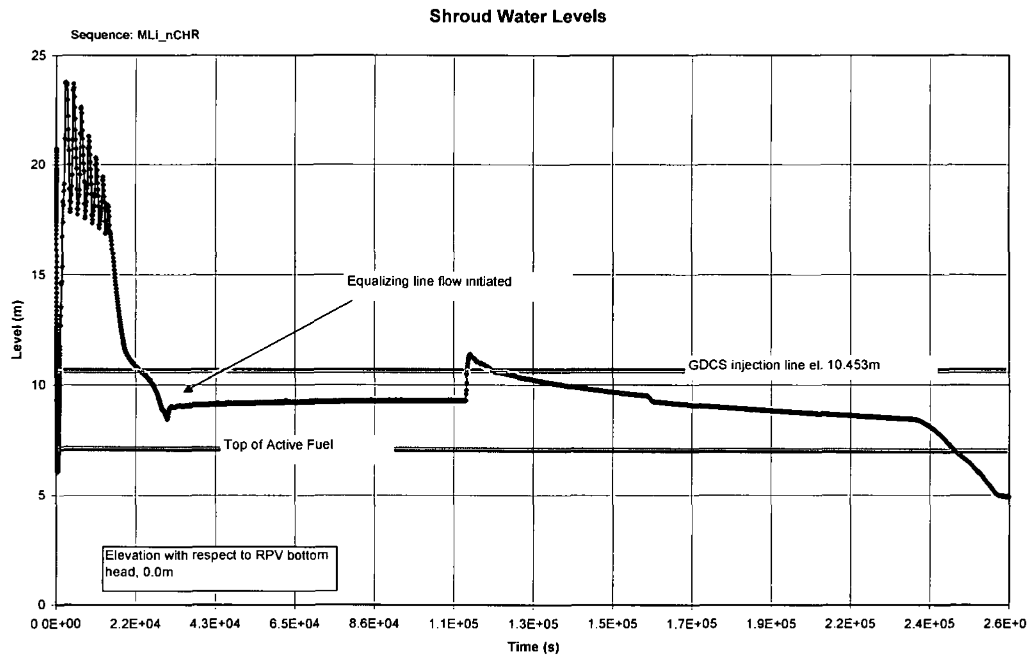

Figure 8.3-3b. MLi_nCHR: Shroud Water Level vs. Time

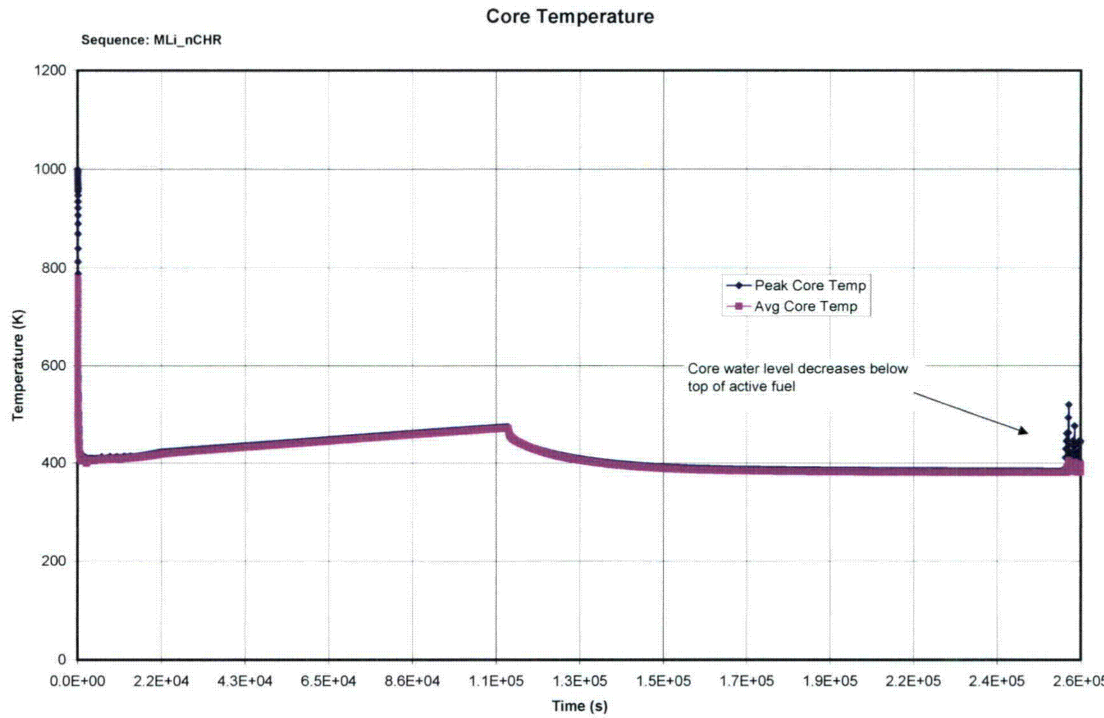

Figure 8.3-3c. MLi_nCHR: Core Heatup vs. Time

Figures 8.3-4a through b: T_nDP_nIN_TSL

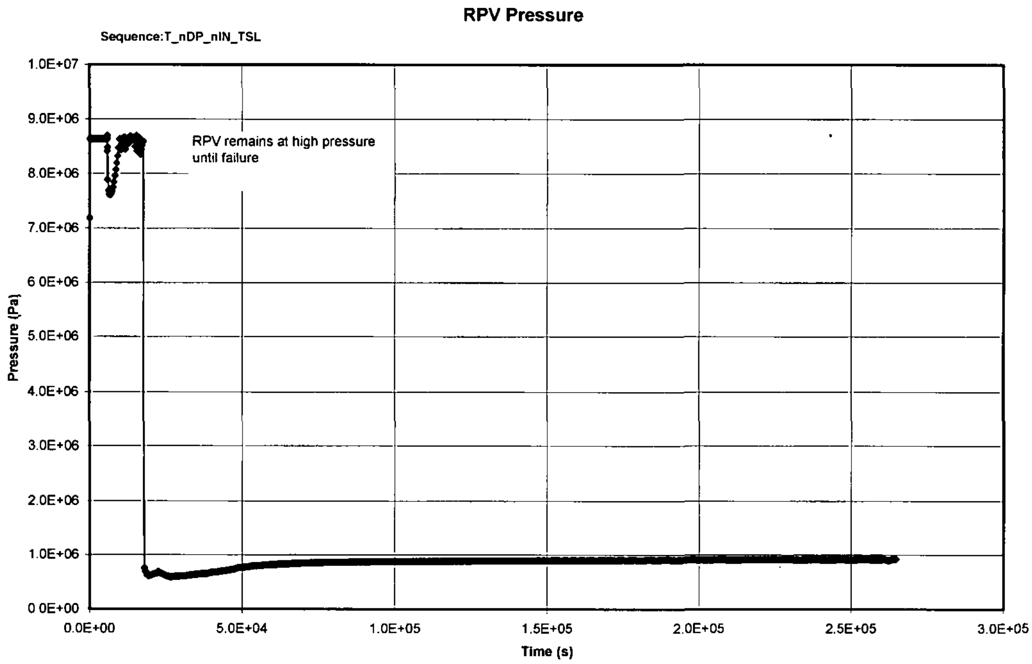

Figure 8.3-4a. T_nDP_nIN_TSL: RPV Pressure vs. Time

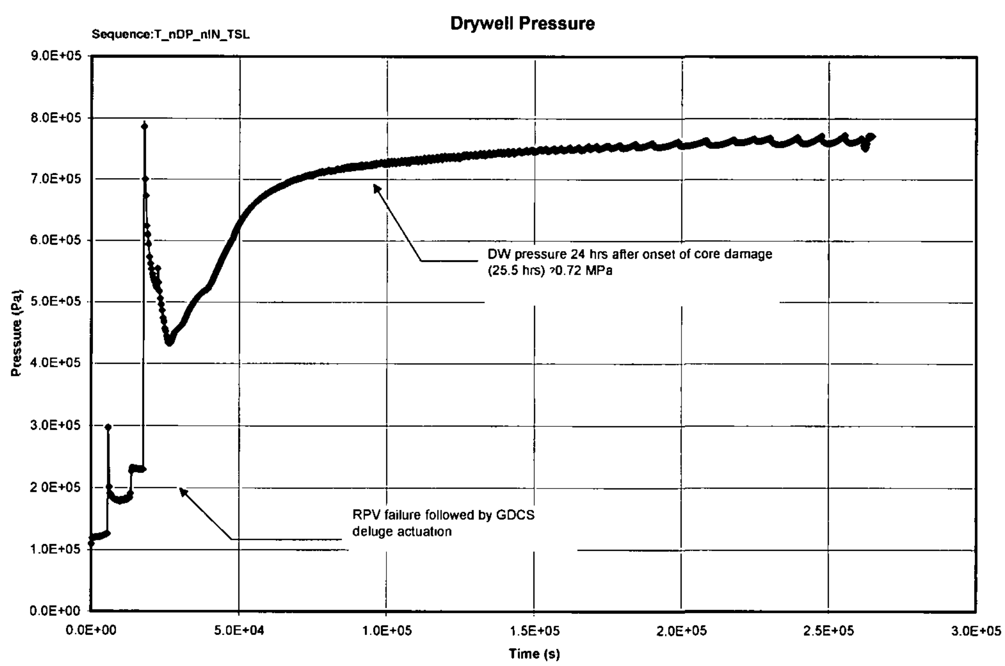

Figure 8.3-4b. T_nDP_nIN_TSL: Containment Pressure vs. Time

Figures 8.3-5a through b: T_nDP_nIN_nCHR_FR

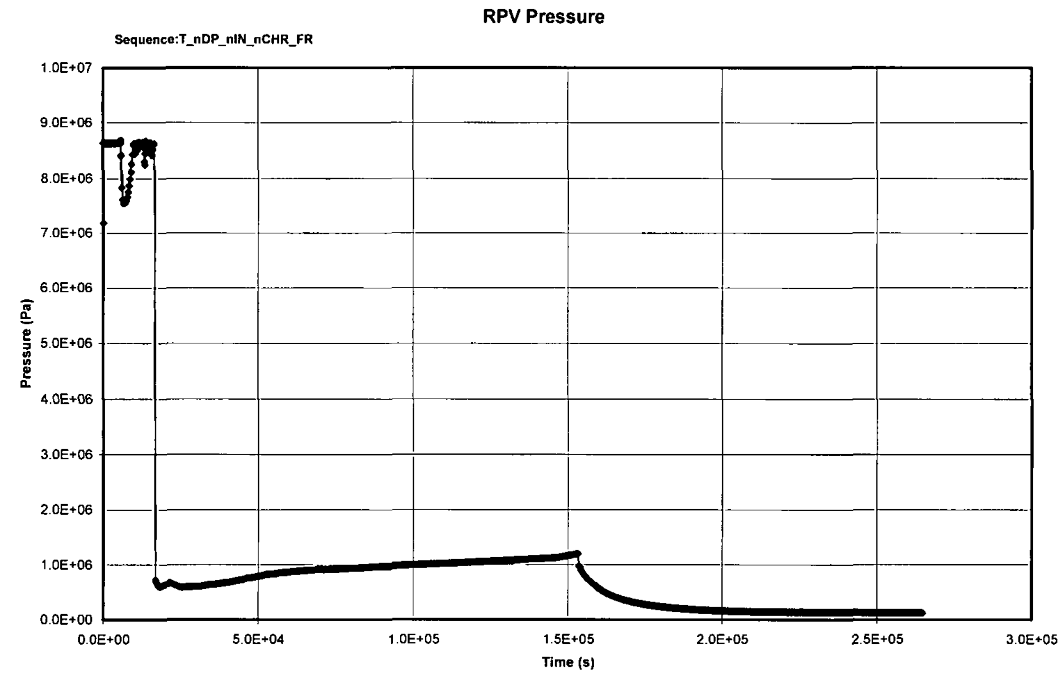

Figure 8.3-5a. T_nDP_nIN_nCHR_FR: RPV Pressure vs. Time

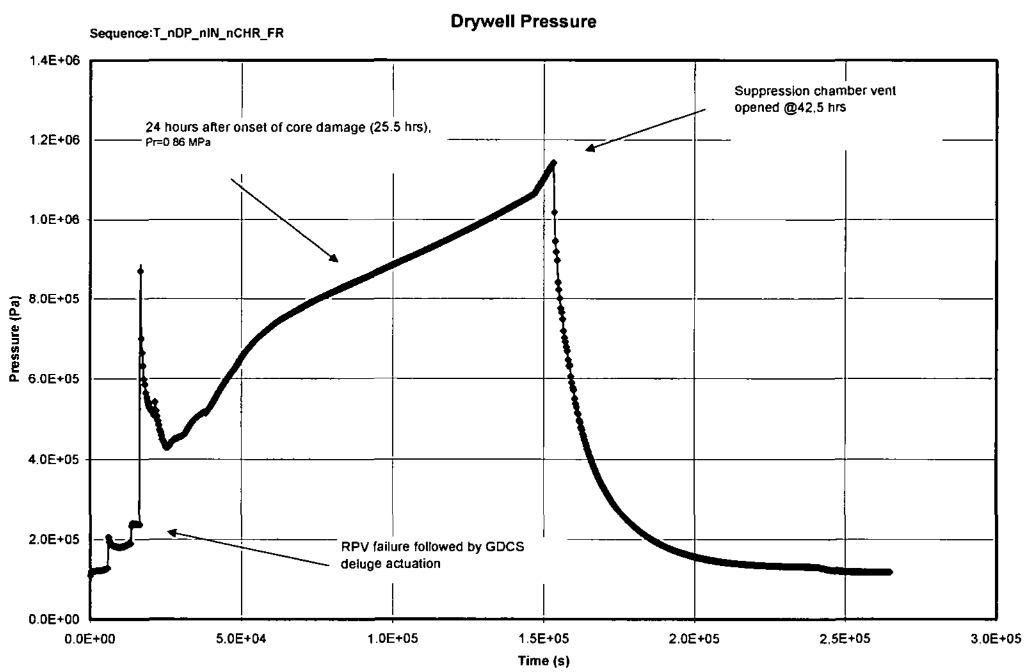

Figure 8.3-5b. T_nDP_nIN_nCHR_FR: Containment Pressure vs. Time

Figures 8.3-6a through c: Sequence T-AT_nIN_TSL

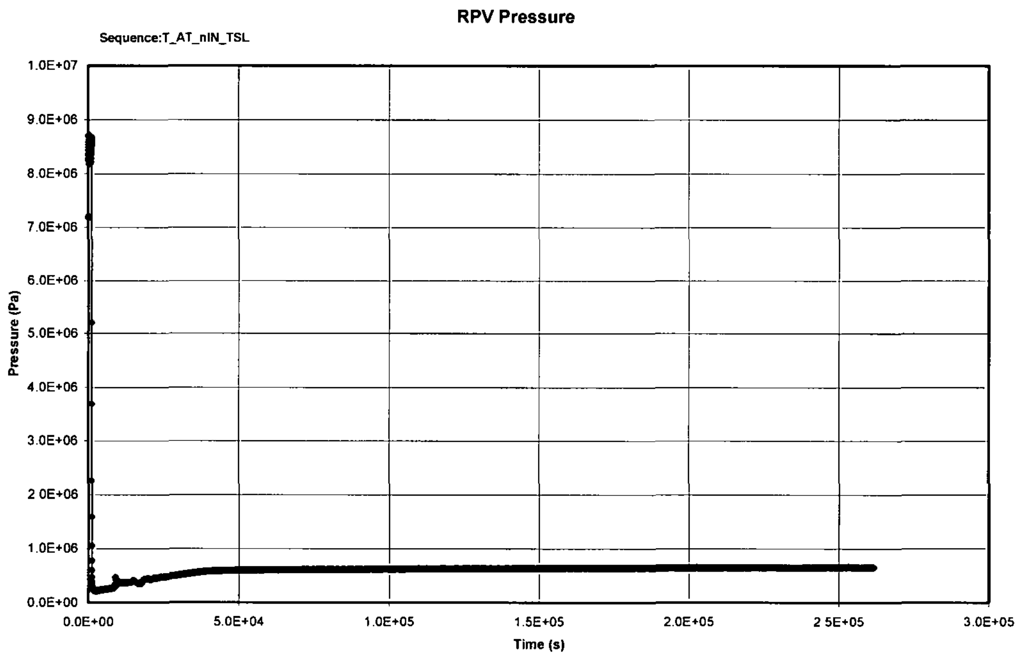

Figure 8.3-6a. T-AT_nIN_TSL: RPV Pressure vs. Time

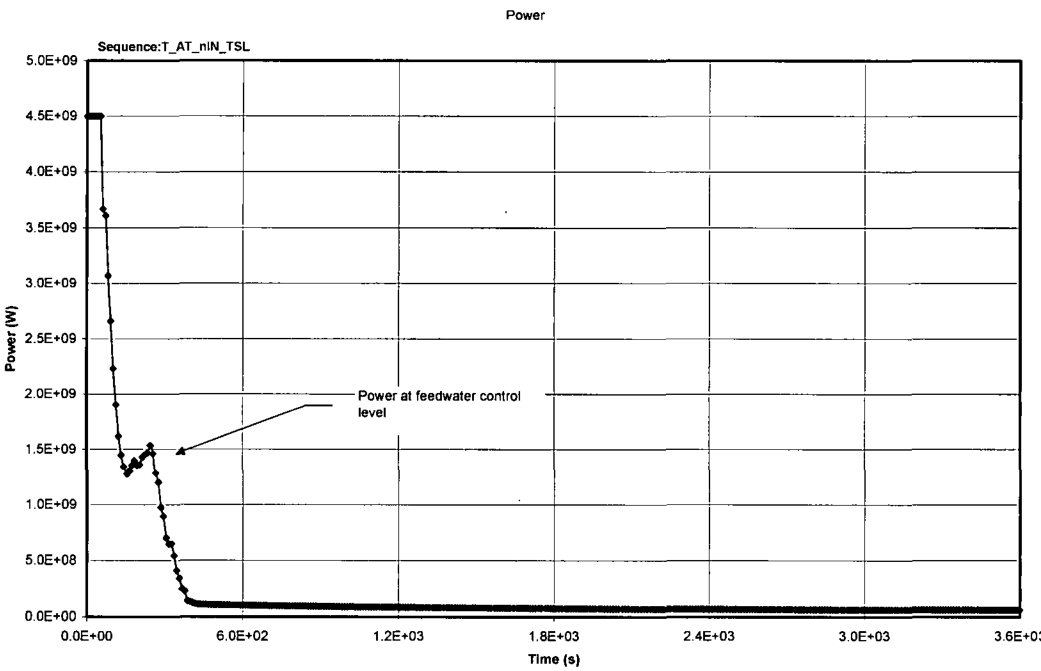

Figure 8.3-6b. T-AT_nIN_TSL: Core Power vs. Time

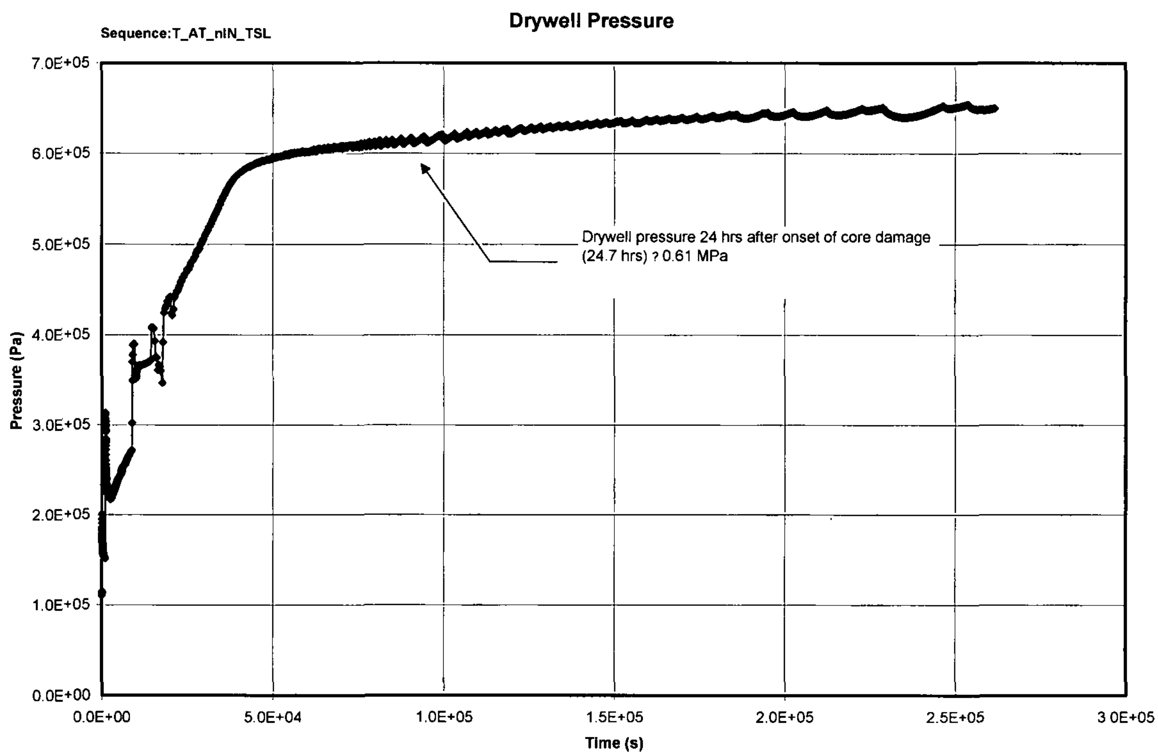

Figure 8.3-6c. T-AT_nIN_TSL: Containment Pressure vs. Time

Figure 8.3-7a: Sequence T-AT_nIN_nCHR_FR

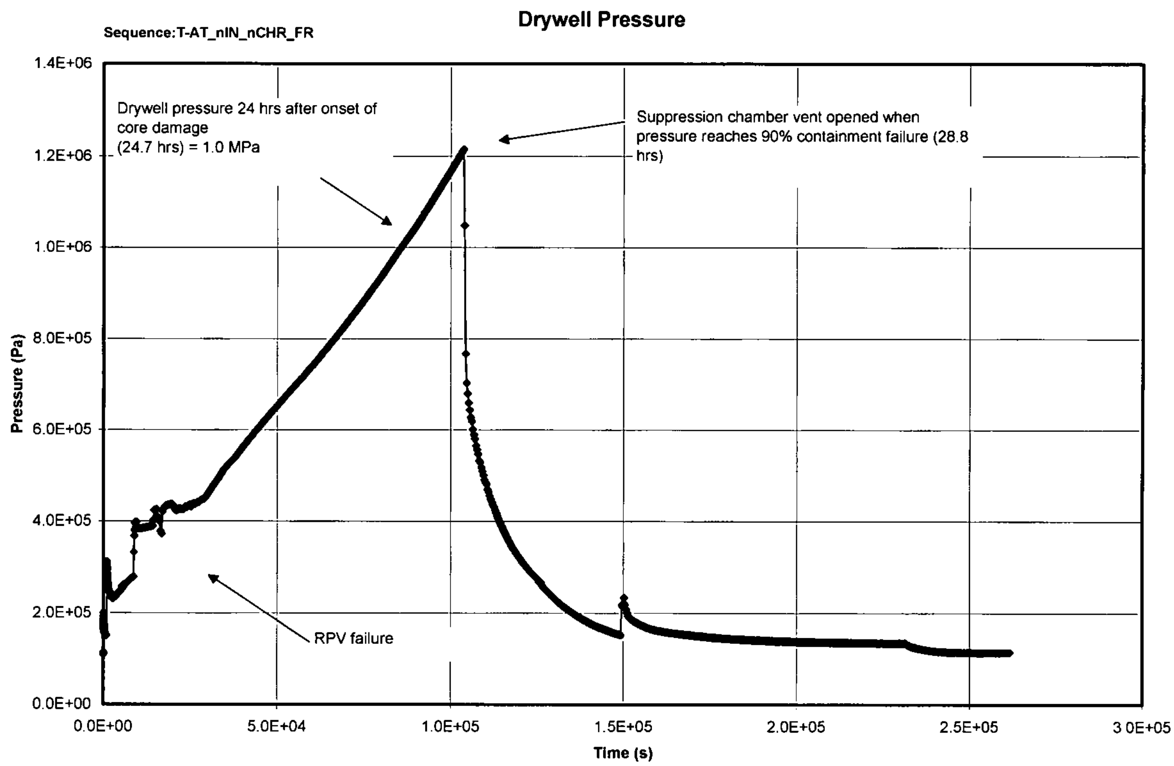

Figure 8.3-7a. T-AT_nIN_nCHR_FR: Containment Pressure vs. Time

8.4 SUMMARY

In this section, the potential for containment failure due to combustible gas generation, containment bypass and overpressurization was evaluated. In addition, the frequency of containment failure events due to the phenomenological events discussed in Section 21 (CCIW, CCID, DCH, EVE) was determined. Because of the ESBWR design and reliability of containment systems, the most likely containment response to a severe accident is associated with successful containment isolation, vapor suppression and containment heat removal. As a result, the containment provides a highly reliable barrier to the release of fission products after a severe accident, with the dominant release category being that defined by Technical Specification leakage (TSL). This conclusion is based on the following insights:

- The combustible gas generation analysis indicated that a combustible gas mixture within containment would not occur within 24 hours after the occurrence of a severe accident during inerted operation. Thus, containment failure by this mechanism during inerted operation is not considered further.
- Containment bypass (BYP) that results in a direct path between the containment atmosphere and environment was evaluated. A containment penetration screening evaluation indicated that there are only a few penetrations that required isolation to prevent significant offsite consequences. All potential leakage paths feature multiple containment isolation valves. Thus, the probability of the bypass failure mode is dominated by common cause hardware failures, resulting in a calculated frequency of containment bypass about three orders of magnitude lower than the TSL release category.
- Containment overpressurization was evaluated in terms of early and late loss of containment heat removal as well as the loss of the vapor suppression function. Total overpressure failure was found to be about three orders of magnitude less likely than the TSL release category after a severe accident, specifically
  - The frequency of loss of containment heat removal in the first 24 hours after accident initiation, release category OPW1, was evaluated to be three orders of magnitude lower than the TSL release category.
  - The frequency of loss of containment heat removal in the period between 24 and 72 hours after accident initiation, release category OPW2, was evaluated to be more than four orders of magnitude lower than the TSL release category.
  - The frequency of vacuum breaker failure, which would result in the shortest time to containment overpressurization because of the loss of the vapor suppression function, release category OPVB, was evaluated to be three orders of magnitude lower than the TSL release category.
- The need for controlled filtered venting, release category FR, in the 24 hour period after onset of core damage was evaluated. The evaluation considered loss of containment heat removal for the spectrum of applicable accident classes. In each representative sequence, operator controlled venting could be implemented to control the containment pressure boundary and potential leak path. In addition to Level 2 scenarios, core damage Class II-b sequences from the Level 1 analysis are classified as filtered release.

However, for the total of Class I, II-b, III, and IV sequences, release category FR was evaluated to be more than four orders of magnitude below the TSL release category.

- Containment failure due to extensive core-concrete interaction (CCI) is postulated to occur due to containment overpressurization by resultant non-condensable gases. Use of the manual wetwell vent is not considered in the PRA because the suppression pool would not reduce the source term from the non-condensables.
  - Wet CCI (CCIW) events, which feature successful actuation of the GDCS deluge system but unsuccessful core cooling, are discussed in Section 21. The frequency of CCIW was found to be approximately two orders of magnitude below that of the TSL release category.
  - The dry CCI (CCID) containment failure scenarios result from a failure of GDCS deluge actuation. The frequency of CCID events was calculated to be over four orders of magnitude lower than TSL.
- The failure of containment due to direct containment heating was excluded from the baseline results based on the discussion in Section 21. However, DCH failures will be considered in a sensitivity study to be documented in Section 11 by assigning a probability of 1E-3 for all sequences in which RPV failure occurs at high pressure.
- The ex-vessel explosion (EVE) containment failure mode is a result of a high-pressure shock to the containment immediately following a steam explosion in the lower drywell. The conditions necessary for a steam explosion as the core melts through the RPV are discussed in Section 21. The total frequency of the EVE release category was calculated to be approximately two orders of magnitude below that of TSL.

Consistent with advanced light water reactor goals established by the NRC, reliability and phenomenological analyses have established that the ESBWR containment maintains its integrity for a 24-hour period after the onset of core damage in a severe accident. An additional insight regarding the ESBWR containment capability can be gained by calculating the "containment effectiveness". The containment effectiveness was calculated as 0.921, which exceeds guidelines provided in Reference 8.0-1 regarding the "conditional containment failure probability".

The release categories and frequencies discussed above will be retained for use in a conservative evaluation of potential source terms, as discussed in Section 9.

8.5 REFERENCES

8.0-1 SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs", April 2, 1993.

8.1-1 NUREG-0800, "Standard Review Plan", Section 6.2.5, "Combustible Gas Control in Containment".

8.1-2 Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident".

8.3-1 "MAAP4 Modular Accident Analysis Program for LWR Power Plants," Transmittal Document for MAAP4 Code Revision MAAP 4.0.6, Rev. 0, Report Number FAI/05-47, prepared for Electric Power Research Institute, 05/05/05.

8.3-2 "ABWR Design Control Document, Tier 2" Section 19.2.

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List of Tables

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8A QUANTIFICATION OF THE LEVEL 2 PRA MODEL

The purpose of this appendix is to present the quantification of the Level 2 PRA. The results are used to determine the conditional containment failure probability (CCFP) and to calculate the frequencies of the various release categories used in the Level 3 PRA.

The CETs used for the Level 2 PRA analysis and described in Section 8 and 8C are logically equivalent to the combination of containment phenomenology event trees (CPET) and containment system event trees (CSET) used in the Section 21 discussion. In Section 21, separation of the two trees allowed a more clear distinction between the two analyses; a single, combined CET simplifies the quantification of the Level 2 PRA itself.

The containment response to phenomenological effects is described in Section 21.1 and the Containment Event Tree (CET) structure is described in Section 8.2. This structure was appended to the Level 1 full power, internal events accident sequences to determine the frequencies of the end states shown on the CETs. In this manner, the total CDF is mapped into appropriate Level 2 end states.

8A.1 BINNING OF LEVEL 1 RESULTS

In order to provide inputs to the CETs, the Level 1 accident sequences are sorted into Level 1 accident class bins. These were used to determine the appropriate phenomenology to consider in the CETs. To address containment susceptibility to steam explosions the Class I and Class IV accident sequences were further divided into bins that specify the water level in the containment at the time of vessel breach.

The Level 1 sequence bins were then used as the initiators to the Level 2 event trees. Effectively, the integrated model is a combination of both the Level 1 and Level 2 PRA models. As such, all initiator impact is preserved throughout the quantification and no special treatment is required for Loss of Preferred Power (LOPP) scenarios.

Table 8A-1 shows the subclass and bin assignment of the sequences above the Level 1 truncation value.

- Class I

  In these cases, core damage occurs when the RPV is at low pressure. All of these sequences remain at low pressure in the RPV through the time of vessel breach.

  The key information needed for the CET is the water level in the lower drywell at the time of vessel breach. If the water is above 1.5 m, failure of the pedestal due to steam explosion cannot be excluded, and therefore is conservatively assumed to occur. If the water level is between 0.7 m and 1.5 m, there is the possibility of a steam explosion but failure of the pedestal is physically unreasonable. Water level below 0.7 m does not present conditions that support a steam explosion. The criteria for determining the water level are presented in Section 7.2.5.

  The following subclasses are defined for quantification:

- Subclass I_LD	Vessel failure occurs at low pressure (<1 MPa), water level in containment is LOW or DRY

- Subclass I_M	Vessel failure occurs at low pressure (<1 MPa), water level in containment is MEDIUM
- Subclass I_H	Vessel failure occurs at low pressure (<1 MPa), water level in containment is HIGH

- Class II

As Class II sequences are defined by containment failure preceding core damage, they are not treated in the containment analysis. However, Class II sequences are binned into individual release categories according to their failure mode. For example, all Class II-b sequences are binned as Filtered Release (FR) releases because the wetwell vent is used as a mitigating system to core damage. Class II-a sequences are binned between OPW1, OPW2, and OPVB release categories according to system failures. Class II sequences are included in the release frequency results, but are not included as an input to the offsite dose calculation because core damage does not occur within 72 hours of accident initiation.

- Class III

In these cases, core damage occurs when the RPV is at high pressure. Water level information is not needed because the high-pressure melt ejection is not subject to the EVE phenomenon.

No additional information is needed for the quantification.

The following subclasses are defined for quantification:

- Subclass III_LD	Vessel failure occurs at high pressure (>1 MPA)

- Class IV

In these cases, core damage sequences are initiated by a failure to reduce reactivity in the core. By the time that the core uncovers, however, the power is essentially shut down due to lack of moderator. The principle difference between these sequences and the Class I or III is the amount of energy transferred to the containment prior to vessel breach. The excess energy is not enough to change the key physics involved in containment failure, so the class can be treated just like the previously defined classes. In the ESBWR Level 1, all Class IV sequences above the truncation limit have depressurization available throughout the sequences, so the core damage is assumed to occur when the RPV is at low pressure. All of these sequences remain at low pressure in the RPV through the time of vessel breach. Therefore, the Class IV sequences use the same quantification model as the Class I.

The following subclasses are defined for quantification:

- Subclass IV_LD	Water level in containment is LOW
- Subclass IV_M	Water level in containment is MEDIUM
- Subclass IV_H	Water level in containment is HIGH

Class IV is retained separately because the timing of key events is somewhat faster than the Class I events.

- Class V

  These are cases where core damage occurs with the RPV open to the environment. No containment event trees are needed. All of these sequences are assigned to the release category of BOC.

8A.2 INTEGRATION OF THE SINGLE TOP LEVEL 2 PRA MODEL

The single-top Level 2 PRA was created with the same basic methodology as the Level 1 model. The key difference is that in the Level 1, each initiator is a basic event with an assigned frequency. In the Level 2, the "initiator" is actually a gate under which the appropriate Level 1 sequences are binned.

To represent the nodes in each of the Level 2 event trees there is either a fault tree or a basic event with a point estimate to represent phenomenological effects. The fault trees may be completely independent of Level 1 sequences (such as the GDCS deluge system), or contain dependencies (such as short term CHR). Integrating the Level 2 PRA with the Level 1 as a single, one-time quantification model allows all dependencies and initiator impact to be correctly reflected in the results.

The Level 2 model integration is as follows:

- The baseline, at-power Level 1 PRA was merged with the Level 1 sequence-binning logic and the new Level 2 fault trees. The gates in the binning logic act as Level 2 event tree initiators, and the fault trees correspond to the nodes in the Level 2 event trees.
- All sequences were imported from the Level 2 event trees. This completes the link from Level 1 initiator to Level 1 sequence, Level 2 initiator, and Level 2 sequence and release category.
- Finally, the Level 2 sequences were merged with the top logic tree for the Level 2 PRA. This top logic tree allows a one-time quantification of the integrated Level 1 / Level 2 model. Many benefits result from the one-top quantification, two of which are decreased quantification time and easily accessible, globally relevant importance measures.[JPH13]

The functions and bases for the Level 2 fault trees are explained in the following sections.

8A.2.1 Debris is Successfully Cooled (BI_FN)

This node is only asked following successful operation of the deluge system. Section 21 identifies the failure of this function as physically unreasonable given successful operation of deluge. This would imply a node probability of less than 10^{-3} for BI_FN, but because the design optimization of the BiMAC has not been completed, the ESBWR PRA assigns a conservative value of 10^{-2} to this node. This value is considered conservative based on the analysis in Section 21.5 and the possibility that the core would be sufficiently spread on the drywell floor to be cooled solely by the overlying pool of water from the deluge system.

8A.2.2 GDCS Deluge Supply to BiMAC Successful (BI_SP)

A complete fault tree analysis was done of the GDCS deluge system. The deluge system is completely independent from all other plant systems and, as a result, all Level 1 sequences. The actuation system is powered by stand-alone batteries and managed by deluge-specific powered logic cabinets (PLCs). There are four main deluge lines, one each from GDCS pools A and D, and two from GDCS pool BC. Each main line forks into three injection lines for a total of 12; each injection line has one squib valve. The success criteria are 6 of 12 injection lines.

8A.2.3 Containment Isolation System (CIS)

Appendix 8C provides a screening of potential containment penetrations that may need to be isolated in a severe accident. The Main Steam lines and Feedwater lines were identified as requiring isolation. In addition, lines that are normally open such as Main Steam, RWCU, and Feedwater are required to close after core damage occurs. To account for these lines, a fault tree was developed for the CIS node to represent the isolation function. Dependencies on the Level 1 model for software function and hardware failures are accounted for in the integrated model.

The placement of node CIS after the phenomenological release nodes in the CETs does not have a significant effect on the Level 2 release category frequency results. The potential for a more severe phenomenological release because of failed containment isolation is screened as below truncation. The effect of increasing the BYP release frequency by placing the CIS node immediately after the initiator in the CET is considered with a sensitivity study in Section 11.

8A.2.4 Containment Intact / Insignificant DCH (DCH_DAM)

Section 21.3 provides the justification for the failure of this function being physically unreasonable. As such, the baseline Level 2 PRA does not consider DCH as a credible failure mode. However, a Section 11 sensitivity study quantifies the potential for DCH by assigning the phenomenon a conservative value of 10^{-3} for all subclasses that experience RPV failure at high pressure.

8A.2.5 Pedestal Intact (EVE_DAM)

Section 21.4 provides the justification for the failure of this function being physically unreasonable in cases where the LDW water level is less than 1.5 m but greater than 0.7 m prior to vessel breach. As such, the baseline Level 2 PRA does not consider EVE from medium water levels to be a credible failure mode. This function will only be asked in the "medium" water level event trees in a Section 11 sensitivity study in which a conservative value of 10^{-3} is assigned for containment failure.

8A.2.6 Water Level Prior to RPV Failure (LD_LVL) – (Deleted)

8A.2.7 Reactor Coolant Boundary Intact (RCB_I)

This branch was not used in the CET quantification. It was identified in Section 21.2 as a "splinter", which means that it is uncertain which path would be followed in any given sequence and a meaningful probability cannot be assigned to the branch. These are treated by solving both paths independently and taking the maximum of the results. In the ESBWR, the path leading to an intact Reactor Coolant Pressure Boundary is more conservative and therefore included in the results.

8A.2.8 Vapor Suppression Function (VB)

This node was modeled using the fault tree from the Level 1 analysis for vapor suppression, "DS-TOPVB". The success criteria are 1 of 3 vacuum breakers must open to break a wetwell to drywell vacuum, and 2 of 3 vacuum breakers must remain closed for vapor suppression.

8A.2.9 Vent Operation (VT)

Air-operated vent valve opening is modeled using the same functional top gate as in the Level 1 analysis: WV-TOPVENT. Re-close of the vent valves is not modeled; the release is assumed to continue once initiated. This captures the function dependency on support systems and Level 1 initiators such as instrument air, the Q-DCIS, and DC power supply. These suppression pool vent valves are opened by remote manual actuation from the control room upon high DW pressure. Using the Level 1 fault tree is conservative because the manual action from Level 1 is based on a time much less than 24 hours. In the Level 2 accident sequence, the operator would have greater than 24 hours, but no operator venting failures appear in the results so the conservatism has negligible impact.

8A.2.10 Containment Heat Removal (W1, Short Term: <24 Hours)

This node is represented by a fault tree with a combination of functions from the Level 1 PRA. The success criteria are that either PCCS, which requires leak tight vacuum breakers, or the suppression pool cooling mode of FAPCS can remove heat in the short term. The dependency of the suppression pool cooling mode of FAPCS on preferred power is captured by the initiator impact in the Level 1 sequences.

8A.2.11 Containment Heat Removal (W2, Long Term: <24 Hours)

This node is identical to the W1 node except that long-term makeup to the PCCS pools by the Fire Protection System is required for PCCS to function.

8A.3 QUANTIFICATION OF THE INTEGRATED MODEL

The single top Level 2 model was quantified at a truncation of 1E-15. The integrated model includes various markers, such as release category and Level 2 sequence, which make the results more versatile for post-processing. The Level 2 event trees are shown in Figures 8A-1 through 8A-7; all of the quantified sequence end state frequencies are shown in Table 8A-3. For each release category, all of the relevant sequence frequencies were totaled to determine the total release category frequency as input to the offsite dose calculation.

To make the sum of the Level 2 release category frequencies match the Level 1 CDF, TSL was calculated by subtracting the total of all non-TSL release categories from the core damage frequency. This conservatively accounts for the small frequency increase associated with the approximations in the quantification process by only reducing the frequency of the "success" path, TSL.

Table 8A-1

Level 1 Sequence Bin Assignments

Acc. Sequence	Initiating Event	CDF [/yr]	Level 1 Class	Level 2 Event Tree	LDW Water Level
T-IORV063	MS-T-IORV	1.79E-09	cdi	I_LD	Low
T-FDW050	%T-FDW	1.12E-09	cdi	I_LD	Low
LL-S-FDWB045	%LL-S-FDWB	5.23E-10	cdi	I_H	High
T-IORV017	MS-T-IORV	4.92E-10	cdi	I_LD	Low
T-LOPP050	T-LOPP	3.64E-10	cdi	I_LD	Low
T-GEN067	T-GEN	1.52E-10	cdi	I_LD	Low
T-FDW060	%T-FDW	8.62E-11	cdi	I_LD	Low
BOC-FDWA027	%BOC-FDWA	7.72E-11	cdi	I_LD	Low
ML-L011	%ML-L	5.39E-11	cdi	I_M	Medium
T-GEN021	T-GEN	4.43E-11	cdi	I_LD	Low
LL-S-FDWA013	%LL-S-FDWA	4.25E-11	cdi	I_H	High
SL-S017	SL-S	2.84E-11	cdi	I_LD	Low
T-LOPP060	T-LOPP	2.73E-11	cdi	I_LD	Low
AT-T-SW004	AT-T-SW	2.62E-11	cdi	I_LD	Low
BOC-FDWB053	%BOC-FDWB	2.46E-11	cdi	I_LD	Low
SL-S063	SL-S	2.35E-11	cdi	I_LD	Low
AT-LOCA005	AT-LOCA	1.98E-11	cdi	I_M	Medium
RVR-014	RVR	1.64E-11	cdi	I_H	High
T-SW037	%T-SW	1.41E-11	cdi	I_LD	Low
LL-S047	LL-S	1.25E-11	cdi	I_LD	Low
BOC-FDWB020	%BOC-FDWB	7.41E-12	cdi	I_LD	Low
SL-L022	%SL-L	4.24E-12	cdi	I_M	Medium
T-SW010	%T-SW	4.23E-12	cdi	I_LD	Low
LL-S049	LL-S	4.16E-12	cdi	I_LD	Low
SL-L068	%SL-L	3.38E-12	cdi	I_M	Medium
BOC-FDWB019	%BOC-FDWB	5.34E-13	cdi	I_LD	Low
BOC-FDWB036	%BOC-FDWB	4.1E-13	cdi	I_LD	Low
T-LOPP033	T-LOPP	3.39E-13	cdi	I_LD	Low
T-SW009	%T-SW	2.35E-13	cdi	I_LD	Low
T-SW029	%T-SW	2.35E-13	cdi	I_LD	Low
AT-T-GEN020	AT-T-GEN	1.33E-11	cdii-a	N/A	
LL-S-FDWB046	%LL-S-FDWB	5.23E-12	cdii-a	N/A	
AT-T-GEN016	AT-T-GEN	4.81E-12	cdii-a	N/A	
AT-T-IORV004	AT-T-IORV	1.31E-12	cdii-a	N/A	

Table 8A-1
Level 1 Sequence Bin Assignments

Acc. Sequence	Initiating Event	CDF [/yr]	Level 1 Class	Level 2 Event Tree	LDW Water Level
T-IORV064	MS-T-IORV	1.17E-12	cdii-a	N/A	
T-IORV013	MS-T-IORV	2.59E-13	cdii-a	N/A	
T-IORV027	MS-T-IORV	2.59E-13	cdii-a	N/A	
AT-T-IORV008	AT-T-IORV	1.88E-13	cdii-a	N/A	
AT-T-GEN021	AT-T-GEN	7.46E-10	cdiii	III_LD	
T-IORV018	MS-T-IORV	7.26E-10	cdiii	III_LD	
T-IORV065	MS-T-IORV	5.75E-10	cdiii	III_LD	
AT-T-LOPP013	AT-T-LOPP	4.46E-10	cdiii	III_LD	
AT-T-FDW013	AT-T-FDW	3.4E-10	cdiii	III_LD	
T-FDW061	%T-FDW	3.13E-10	cdiii	III_LD	
AT-T-IORV009	AT-T-IORV	1.49E-10	cdiii	III_LD	
T-LOPP061	%T-SW	1.29E-10	cdiii	III_LD	
T-GEN022	T-GEN	6.45E-11	cdiii	III_LD	
T-GEN069	T-GEN	5.79E-11	cdiii	III_LD	
BOC-FDWB054	%BOC-FDWB	3.04E-11	cdiii	III_LD	
BOC-FDWA029	%BOC-FDWA	2.55E-11	cdiii	III_LD	
ML-L012	%ML-L	1.78E-11	cdiii	III_LD	
T-SW039	%T-SW	1.73E-11	cdiii	III_LD	
BOC-FDWB021	%BOC-FDWB	1.08E-11	cdiii	III_LD	
SL-S018	SL-S	8.26E-12	cdiii	III_LD	
SL-S065	SL-S	6.69E-12	cdiii	III_LD	
AT-T-GEN012	AT-T-GEN	6.31E-12	cdiii	III_LD	
T-SW011	%T-SW	6.09E-12	cdiii	III_LD	
AT-T-FDW008	AT-T-FDW	2.92E-12	cdiii	III_LD	
SL-L023	%SL-L	1.27E-12	cdiii	III_LD	
SL-L070	%SL-L	1.01E-12	cdiii	III_LD	
AT-T-LOPP008	AT-T-LOPP	8.45E-13	cdiii	III_LD	
AT-T-GEN023	AT-T-GEN	1.3E-09	cdiv	IV_LD	Low
AT-T-GEN026	AT-T-GEN	2.39E-10	cdiv	IV_LD	Low
AT-T-FDW015	AT-T-FDW	1.11E-10	cdiv	IV_LD	Low
LL-S050	LL-S	8.47E-11	cdiv	IV_LD	Low
AT-T-LOPP015	AT-T-LOPP	3.21E-11	cdiv	IV_LD	Low
AT-T-GEN024	AT-T-GEN	2.93E-11	cdiv	IV_LD	Low
AT-T-IORV011	AT-T-IORV	2.58E-11	cdiv	IV_LD	Low
ML-L014	%ML-L	1.89E-11	cdiv	IV_H	High
AT-T-IORV014	AT-T-IORV	4.27E-12	cdiv	IV_LD	Low

Table 8A-1

Level 1 Sequence Bin Assignments

Acc. Sequence	Initiating Event	CDF [/yr]	Level 1 Class	Level 2 Event Tree	LDW Water Level
AT-T-FDW016	AT-T-FDW	2.53E-12	cdiv	IV_LD	Low
LL-S-FDWA016	%LL-S-FDWA	1.39E-12	cdiv	IV_H	High
LL-S-FDWB047	%LL-S-FDWB	1.39E-12	cdiv	IV_H	High
AT-T-SW006	AT-T-SW	7.75E-13	cdiv	IV_LD	Low
AT-T-IORV012	AT-T-IORV	5.31E-13	cdiv	IV_LD	Low
AT-LOCA012	AT-LOCA	5.06E-13	cdiv	IV_M	Medium
AT-T-LOPP016	AT-T-LOPP	4.66E-13	cdiv	IV_LD	Low
AT-T-GEN025	AT-T-GEN	1.4E-13	cdiv	IV_LD	Low
BOC-MS067	%BOC-MS	1.13E-13	cdiv	IV_LD	Low
ML-L013	%ML-L	7.11E-11	cdv	N/A	
BOC-RWCU051	%BOC-RWCU	4.37E-11	cdv	N/A	
T-FDW052	%T-FDW	1.1E-11	cdv	N/A	
BOC-RWCU015	%BOC-RWCU	4.36E-12	cdv	N/A	
T-LOPP052	T-LOPP	3.29E-12	cdv	N/A	
BOC-RWCU049	%BOC-RWCU	1.23E-12	cdv	N/A	
BOC-FDWB105	%BOC-FDWB	6.21E-13	cdv	N/A	
BOC-FDWB103	%BOC-FDWB	1.64E-13	cdv	N/A[JPH26]	

Table 8A-2

Level 2 Results by Sequence

Sequence	Contribution	Frequency	Release Category	% Release Category
I_LD-01	4.010E-01	4.94E-09	TSL	
I_LD-02			FR1	
I_LD-03			OPW2	
I_LD-04			FR2	
I_LD-05			OPW1	
I_LD-06	6.350E-03	6E-12	OPVB	37.5740%
I_LD-07	2.150E-03	2E-12	BYP	3.5374%
I_LD-08	4.660E-02	4.3E-11	CCIW	43.5514%
I_LD-09	4.960E-04	ε	CCID	50.9764%
I_M-01	8.070E-03	1.00E-10	TSL	
I_M-02			FR1	
I_M-03			OPW2	
I_M-04			FR2	
I_M-05			OPW1	
I_M-06			OPVB	
I_M-07			BYP	
I_M-08	8.830E-04	1E-12	CCIW	0.8252%
I_M-09	3.110E-06	ε	CCID	0.3196%
I_M-10			EVE	
I_H-01	6.340E-01	5.88E-10	EVE	96.3526%
III_LD-01	3.670E-01	4.52E-09	TSL	
III_LD-02			FR1	
III_LD-03	2.970E-05	ε	OPW2	35.3993%
III_LD-04			FR2	
III_LD-05			OPW1	
III_LD-06			OPVB	
III_LD-07	3.840E-03	3E-12	BYP	6.3179%
III_LD-08	4.000E-02	3.7E-11	CCIW	37.3832%
III_LD-09	1.960E-04	ε	CCID	20.1439%
III_LD-10			DCH	
IV_LD-01	1.490E-01	1.83E-09	TSL	
IV_LD-02			FR1	
IV_LD-03			OPW2	
IV_LD-04			FR2	
IV_LD-05			OPW1	
IV_LD-06			OPVB	
IV_LD-07	1.870E-02	1.7E-11	BYP	30.7670%
IV_LD-08	1.980E-02	1.8E-11	CCIW	18.5047%
IV_LD-09	2.790E-04	ε	CCID	28.6742%
IV_M-01	5.850E-05	1E-12	TSL	
IV_M-02			FR1	
IV_M-03			OPW2	

Table 8A-2

Level 2 Results by Sequence

Sequence	Contribution	Frequency	Release Category	% Release Category
IV_M-04			FR2	
IV_M-05			OPW1	
IV_M-06			OPVB	
IV_M-07	6.470E-06	ε	BYP	0.0106%
IV_M-08	5.750E-06	ε	CCIW	0.0054%
IV_M-09			CCID	
IV_M-10			EVE	
IV_H-01	2.350E-02	2.2E-11	EVE	3.5714%

Table 8A-3

Level 2 End State Frequencies*

	I_LD	I_M	I_H	III_LD	IV_LD	IV_M	IV_H	II-a	II-b	V	Totals
TSL	4.9E-09	1E-10	-	4.5E-09	1.8E-09	ε	-	-	-	-	1.14E-08
FR			-				-	-	ε	-	ε
OPW2			-	ε			-	ε	-	-	ε
OPW1			-				-	3E-11	-	-	3E-11
OPVB	6E-12		-				-	1E-11	-	-	2E-11
BYP	2E-12		-	4E-12	2E-11	ε	-	-	-	-	2E-11
CCIW	4E-11	ε	-	4E-11	2E-11	ε	-	-	-	-	1E-10
CCID	5E-13	ε	-	ε	ε		-	-	-	-	ε
EVE	-		6E-10				2E-11	-	-	-	6E-10
DCH	-	-	-		-	-	-	-	-	-	0
BOC	-	-	-	-	-	-	-	-	-	1E-10	1E-10

* Note that these are not the final release category frequencies as reported in Section 8, Table 8.2-2. This table displays the raw quantification results, further post-processing such as adding bypass due to de-inerted operation is performed before the final results are complete.

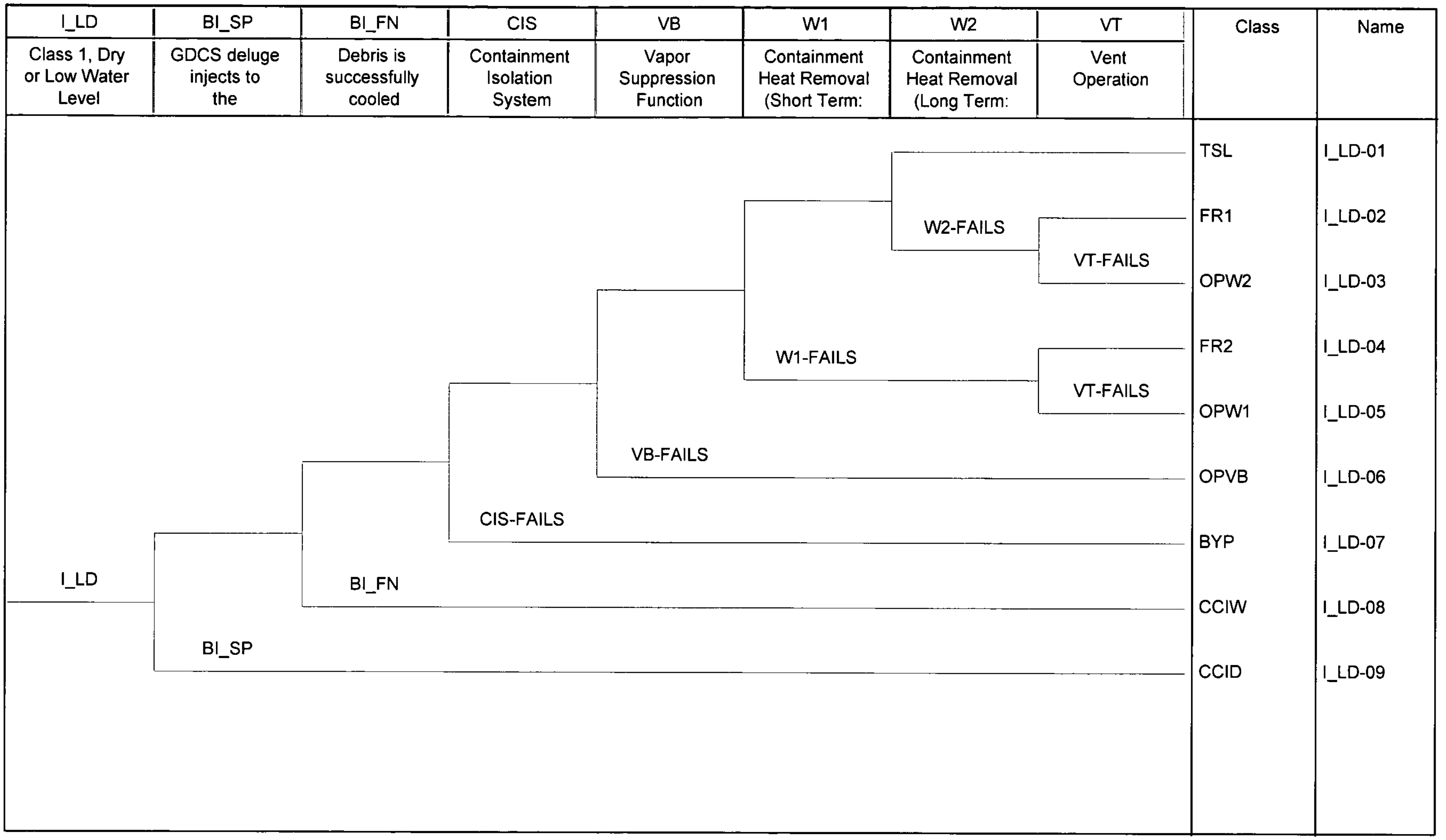

Figure 8A-1. Class I with Low Drywell Water Level CET

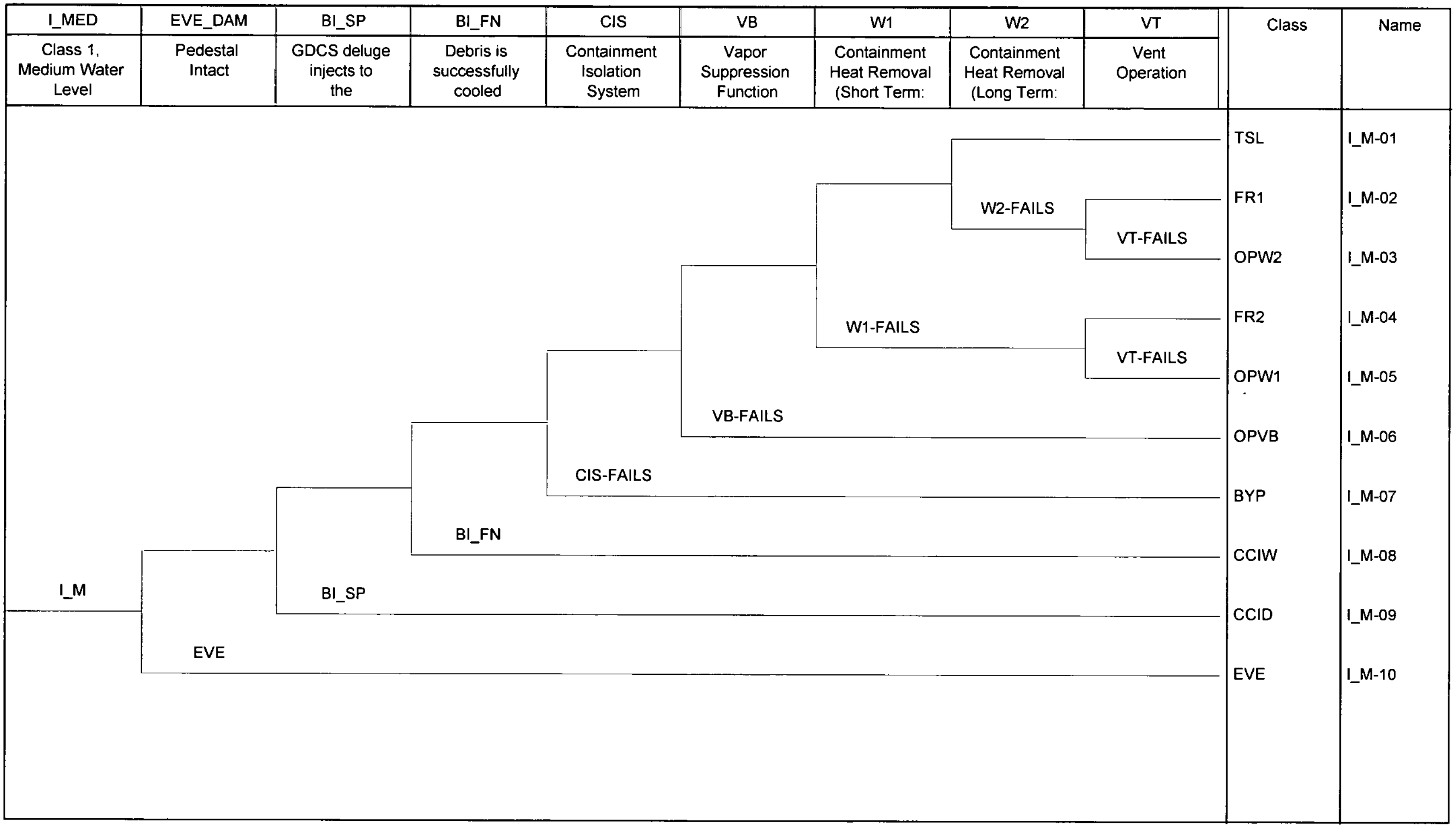

Figure 8A-2. Class I with Medium Drywell Water Level CET

I_HI	EVE_DAM	BI_SP	BI_FN	Class	Name
Class 1, High Water	Pedestal Intact	GDCS deluge injects to	Debris is successfully		
I_H				EVE	I_H-01

Figure 8A-3. Class I with High Drywell Water Level

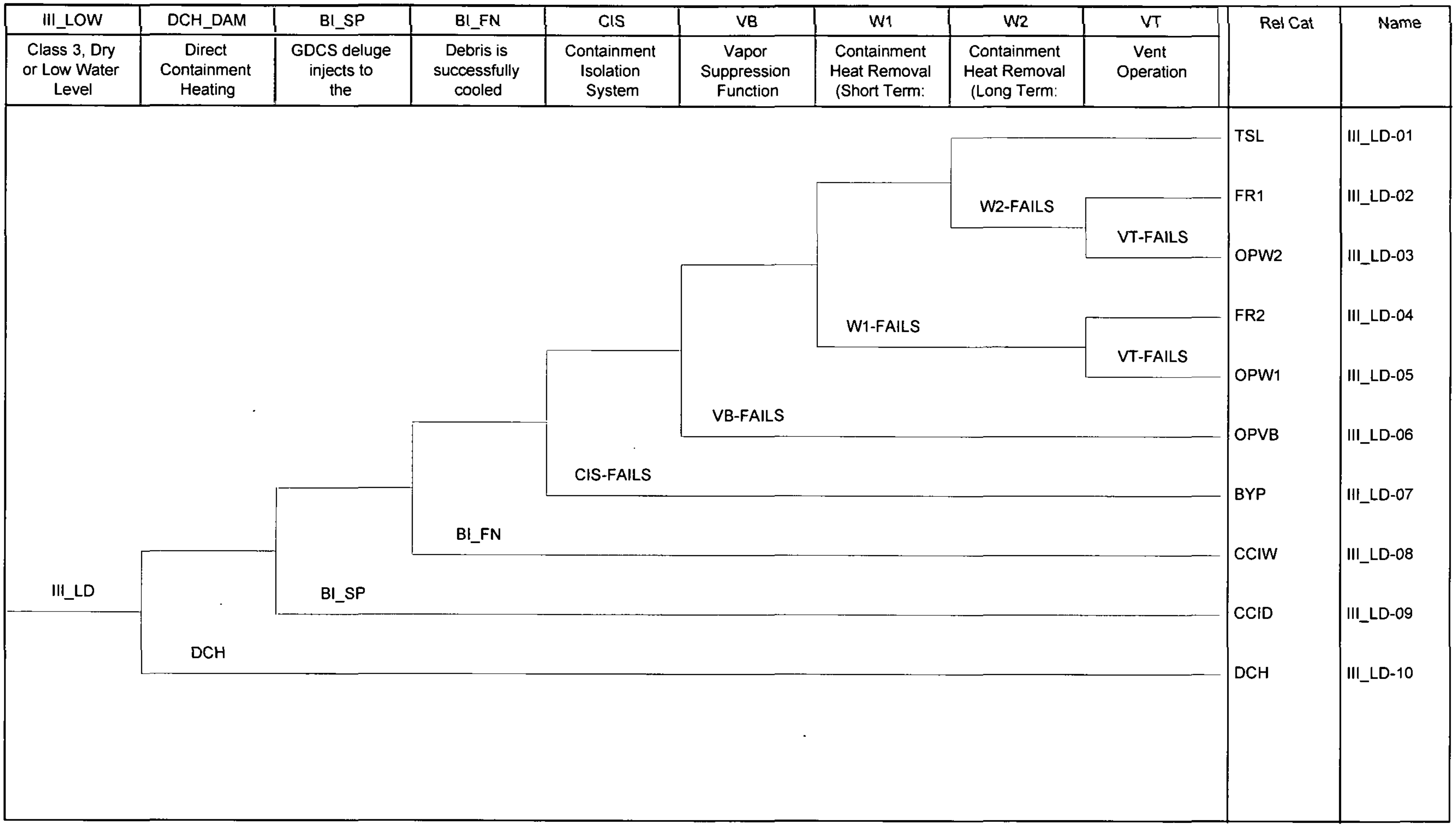

Figure 8A-4. Class III with Low Drywell Water Level CET

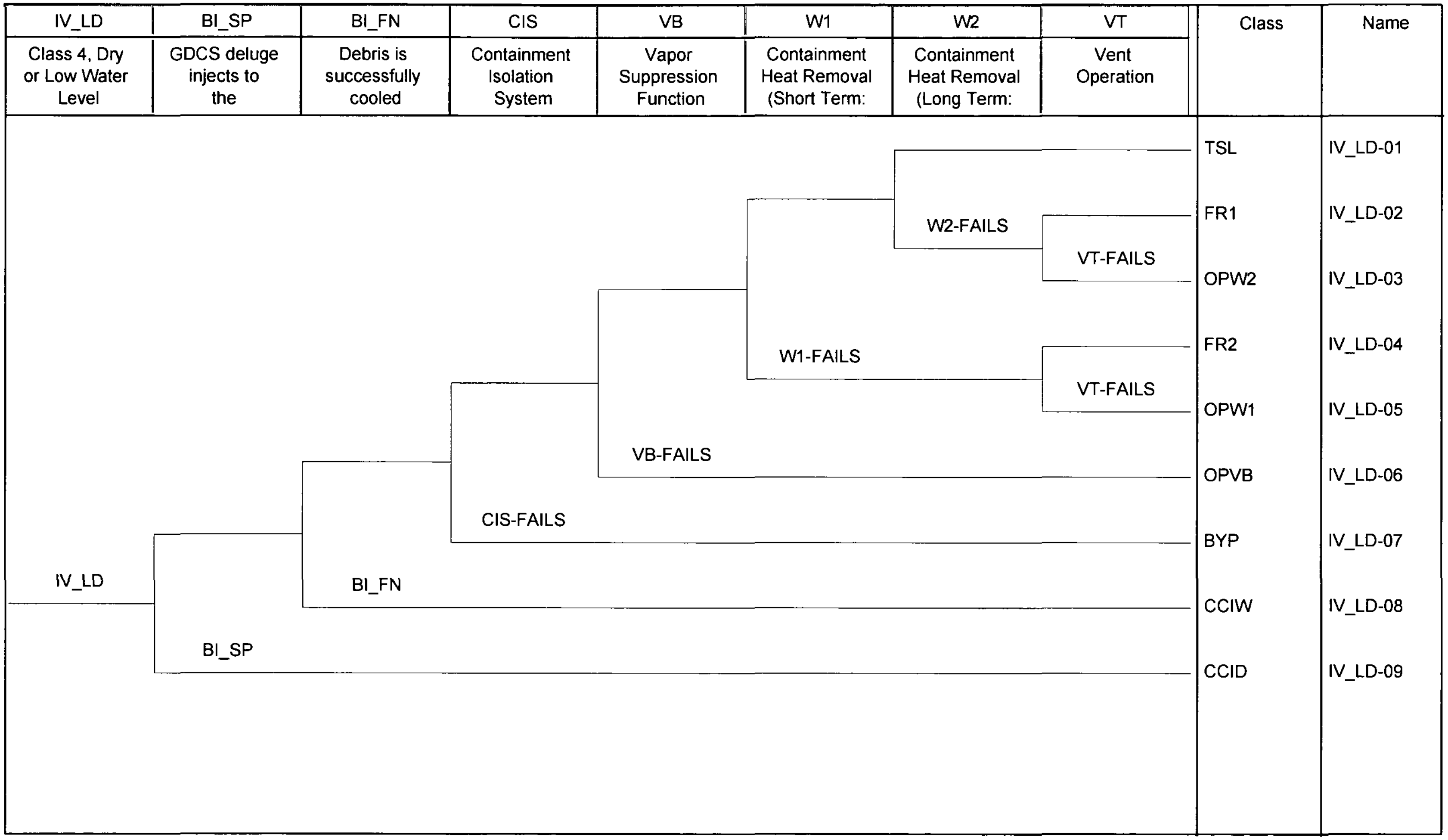

Figure 8A-5. Class IV with Low Drywell Water Level CET

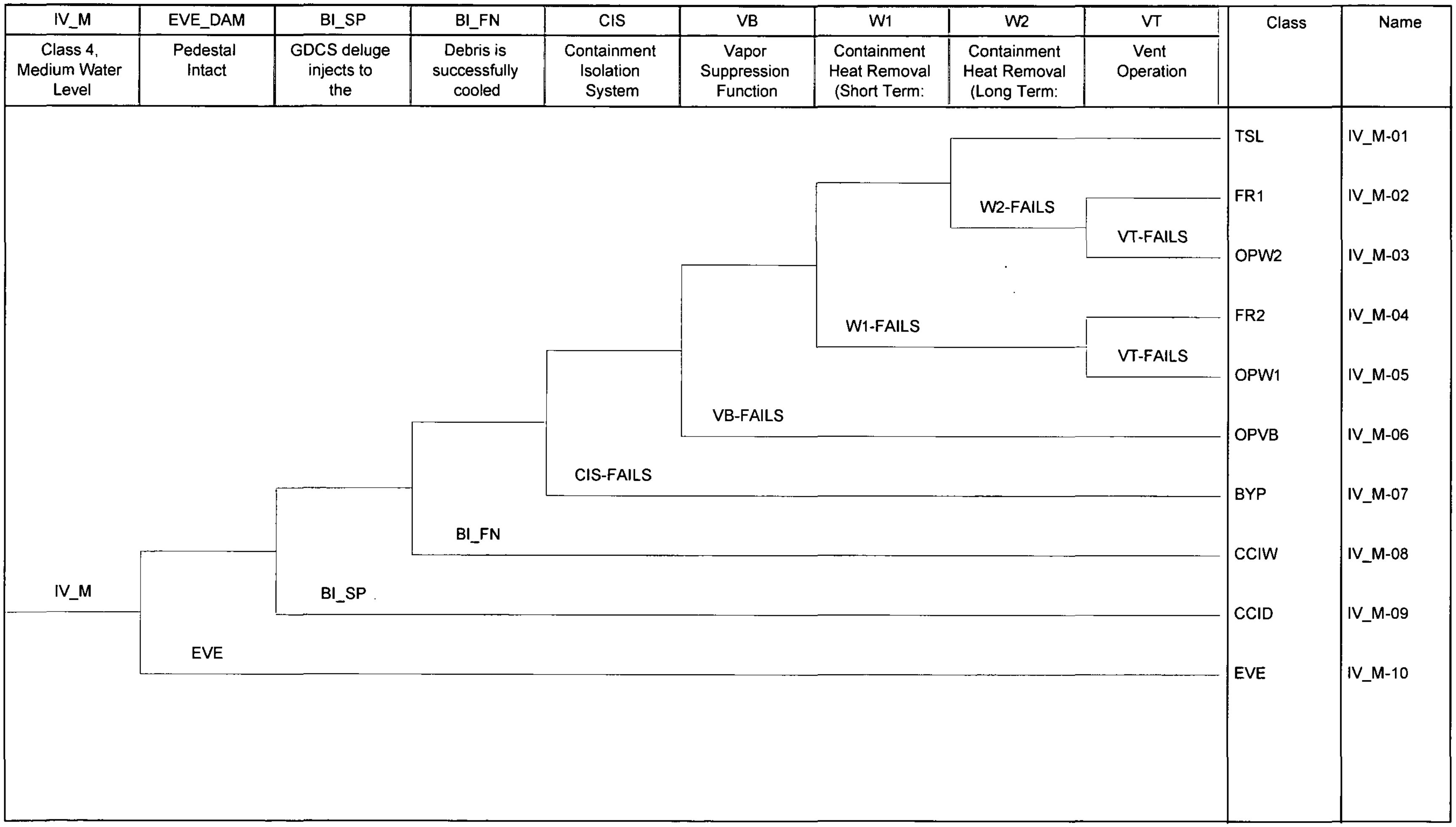

Figure 8A-6. Class IV with Medium Drywell Water Level CET

IV	EVE_DAM	BI_SP	BI_FN	CIS	VB	W1	W2	VT	Class	Name
Class 4, High Water Level	Pedestal Intact	GDCS deluge injects to the	Debris is successfully cooled	Containment Isolation System	Vapor Suppression Function	Containment Heat Removal (Short Term:	Containment Heat Removal (Long Term:	Vent Operation		
IV_H									EVE	IV_H-01

Figure 8A-7. Class IV with High Drywell Water Level CET

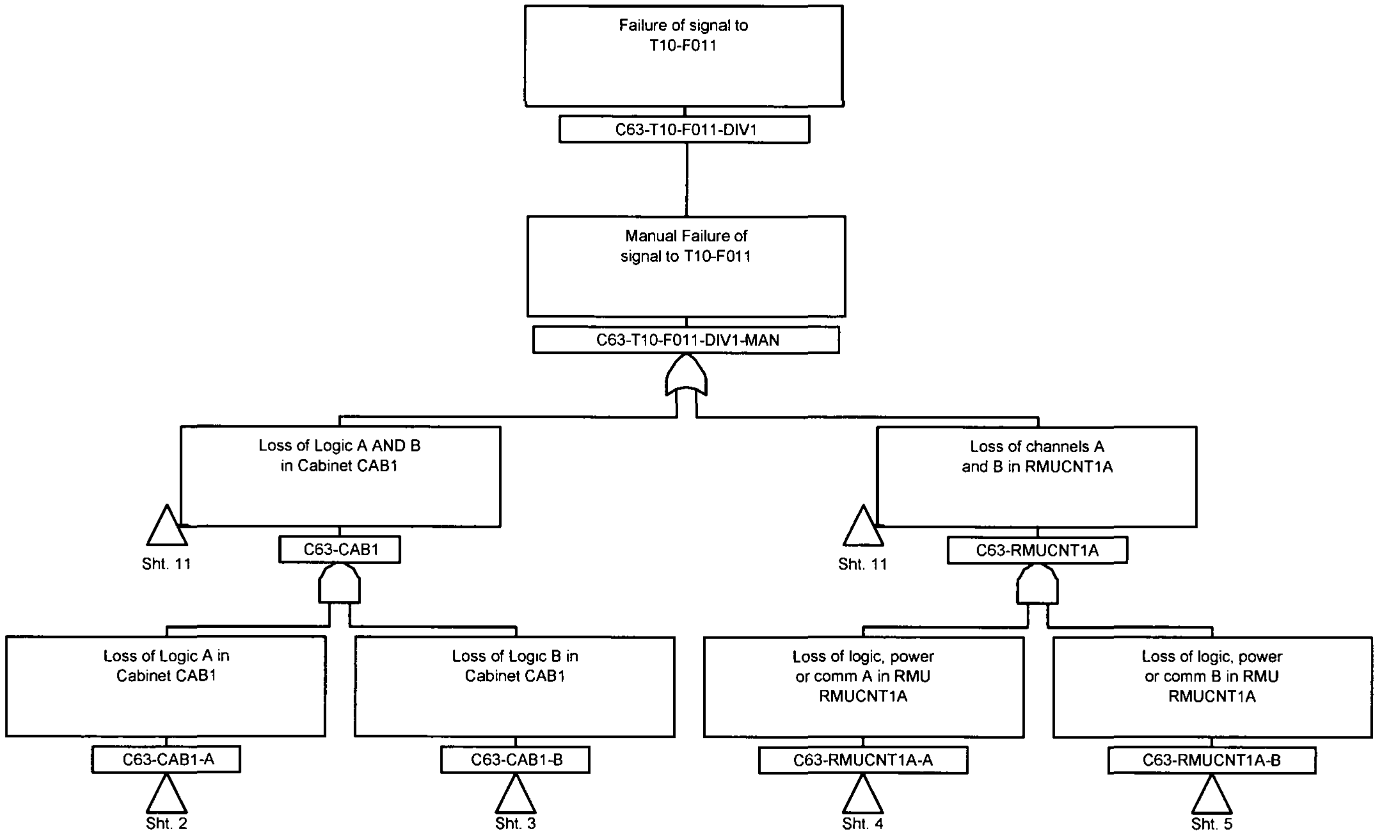

Figure 8A-8. Level 2 Fault Trees
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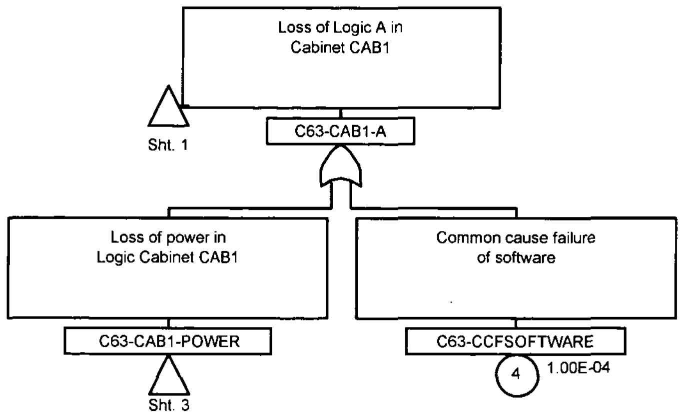

Figure 8A-8. Level 2 Fault Trees

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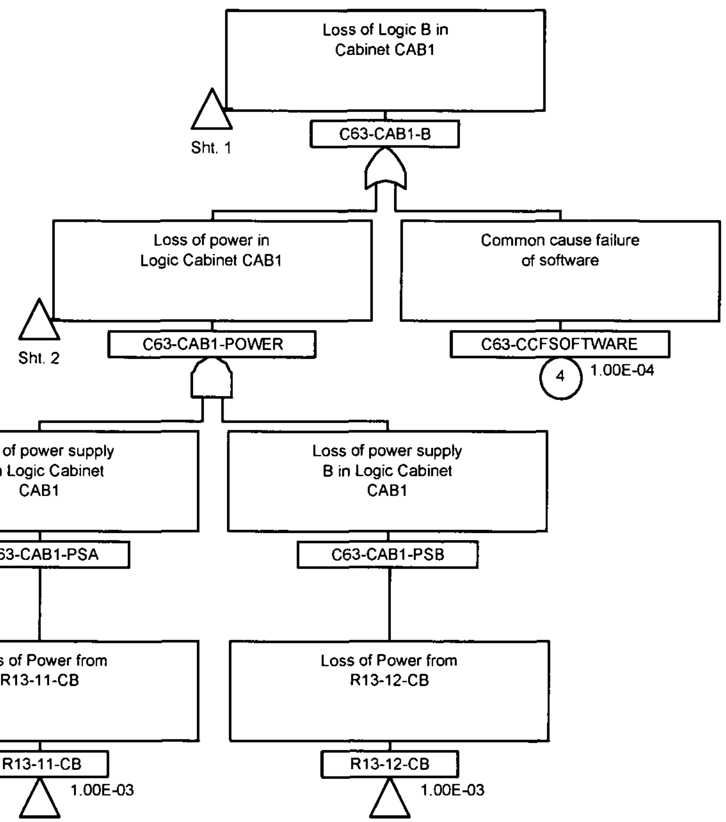

Figure 8A-8. Level 2 Fault Trees
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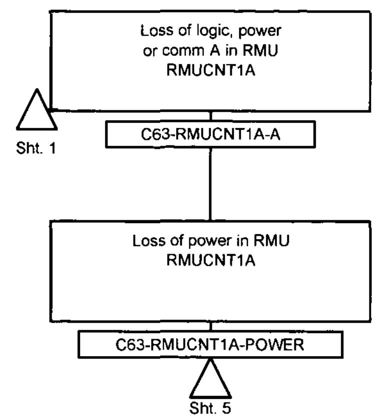

Figure 8A-8. Level 2 Fault Trees
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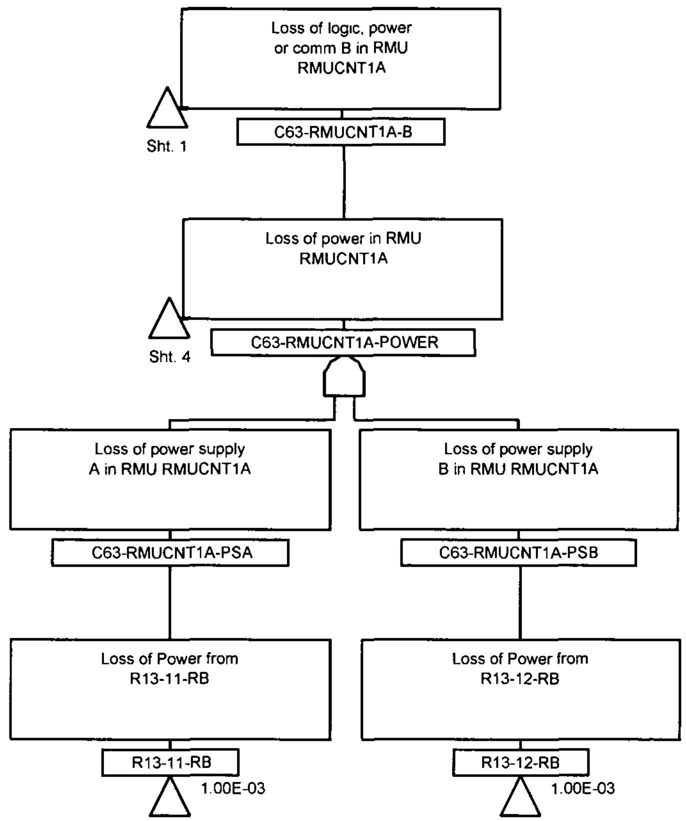

Figure 8A-8. Level 2 Fault Trees
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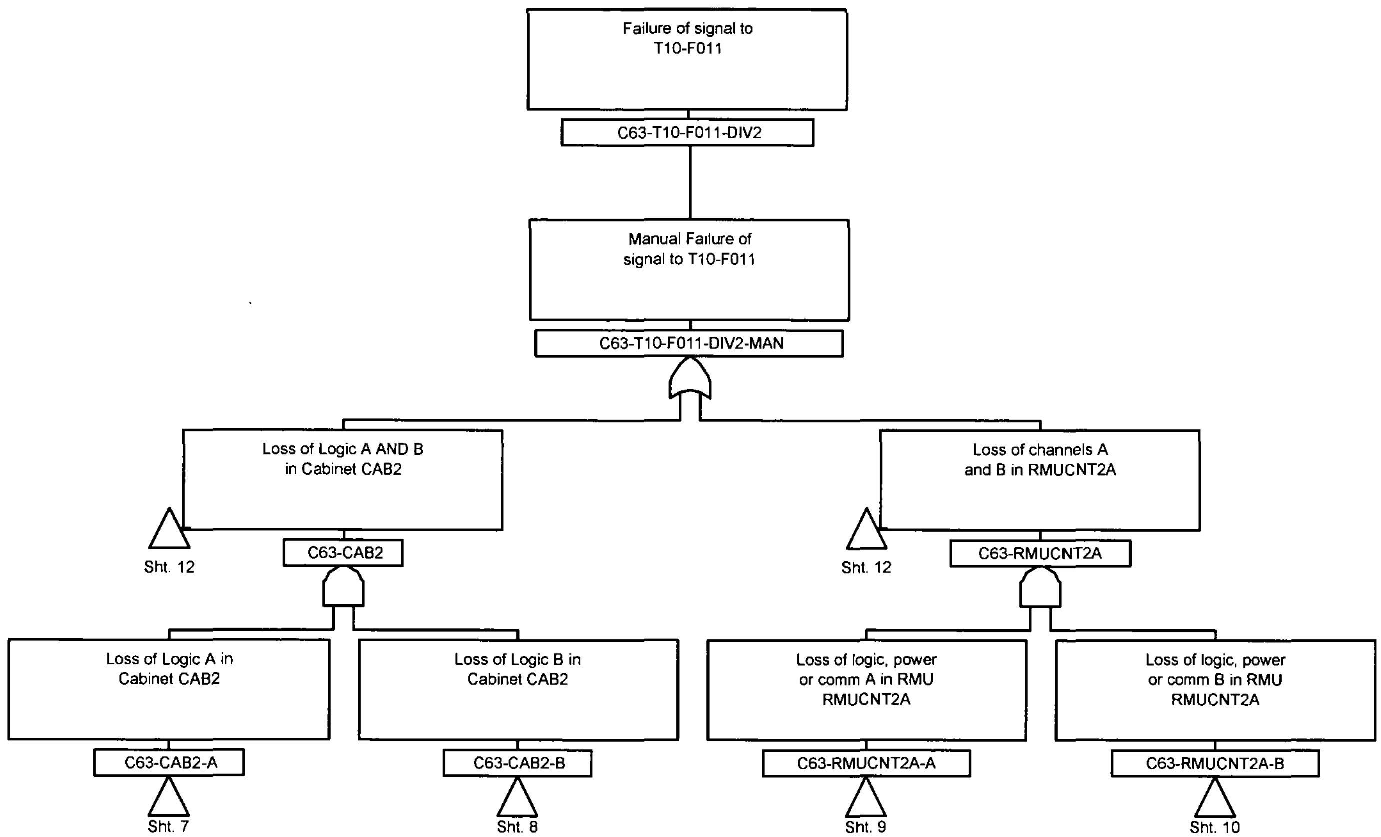

Figure 8A-8. Level 2 Fault Trees
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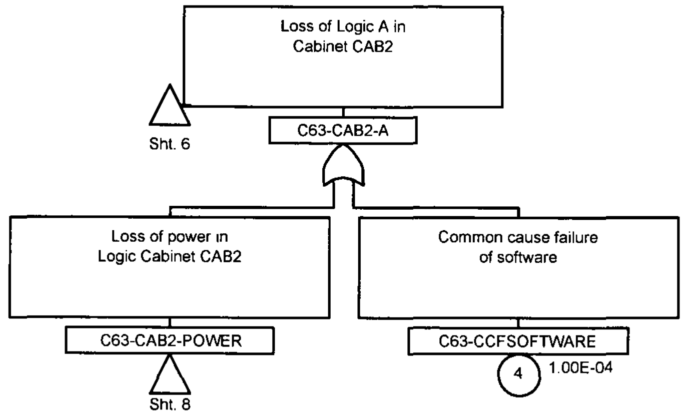

Figure 8A-8. Level 2 Fault Trees
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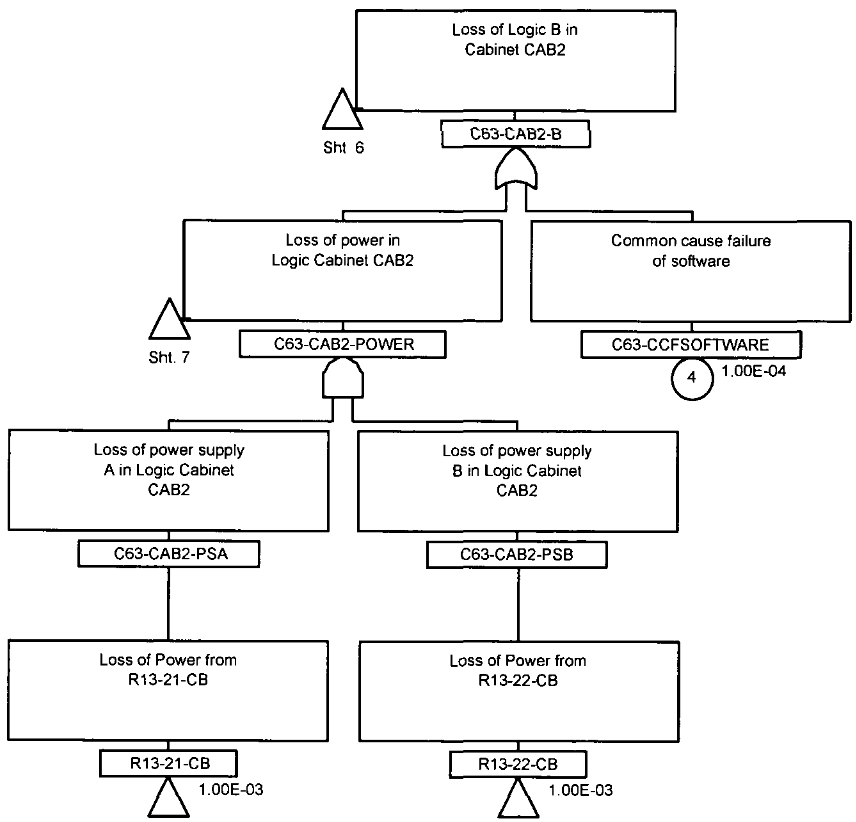

Figure 8A-8. Level 2 Fault Trees

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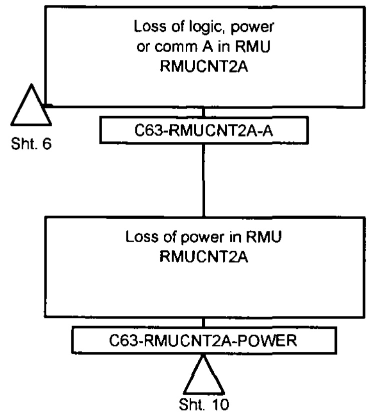

Figure 8A-8. Level 2 Fault Trees

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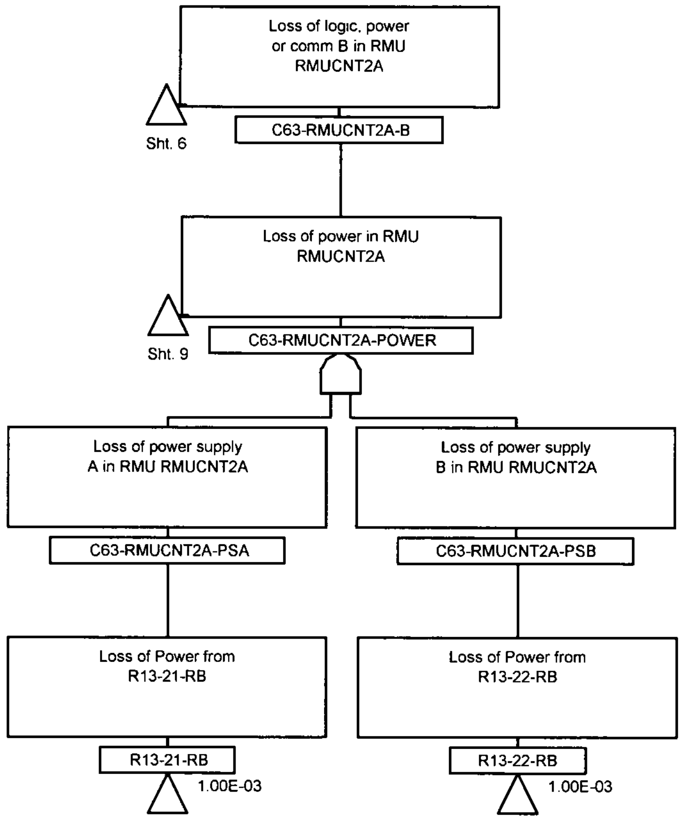

Figure 8A-8. Level 2 Fault Trees

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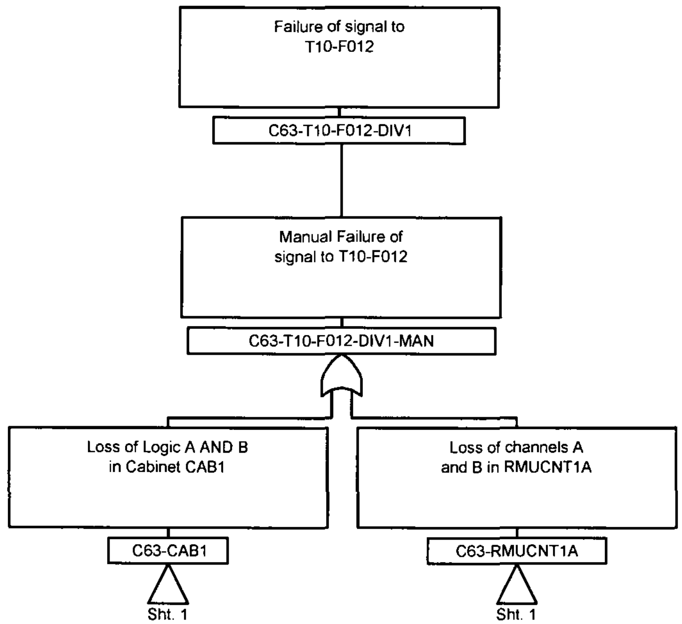

Figure 8A-8. Level 2 Fault Trees

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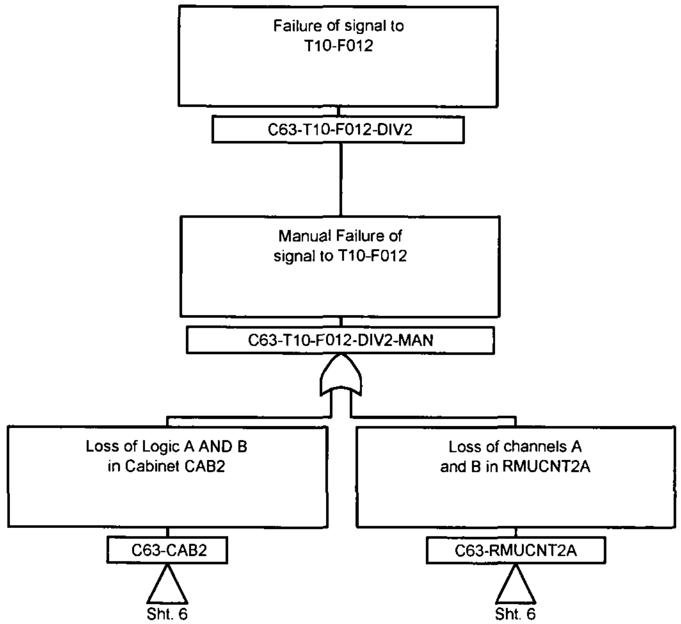

Figure 8A-8. Level 2 Fault Trees

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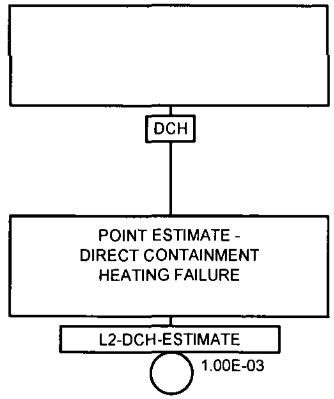

Figure 8A-8. Level 2 Fault Trees
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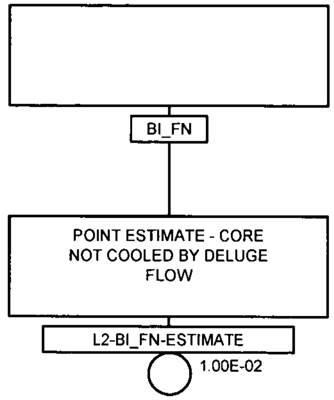

Figure 8A-8. Level 2 Fault Trees

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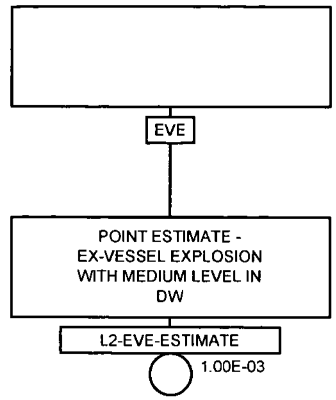

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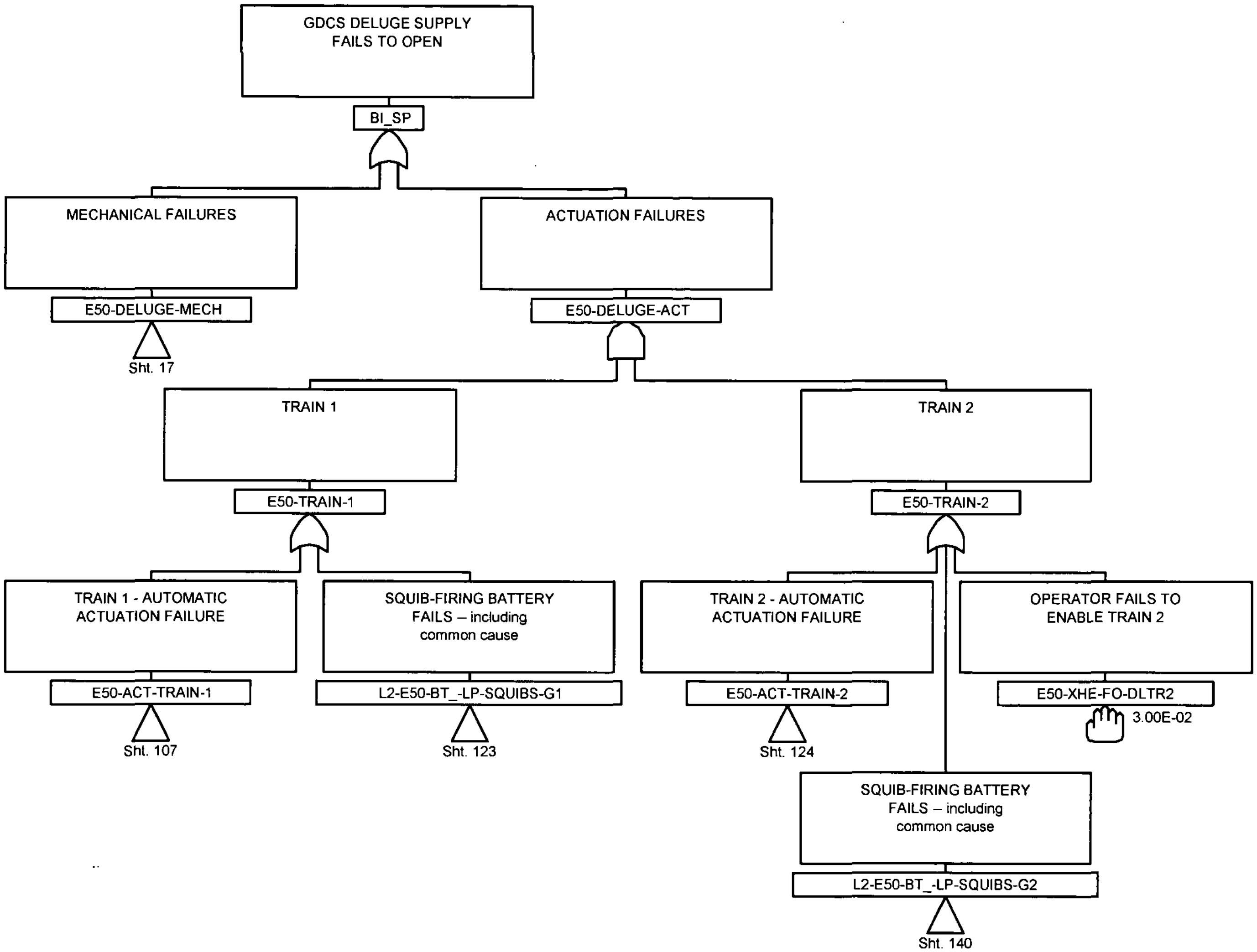

Figure 8A-8. Level 2 Fault Trees
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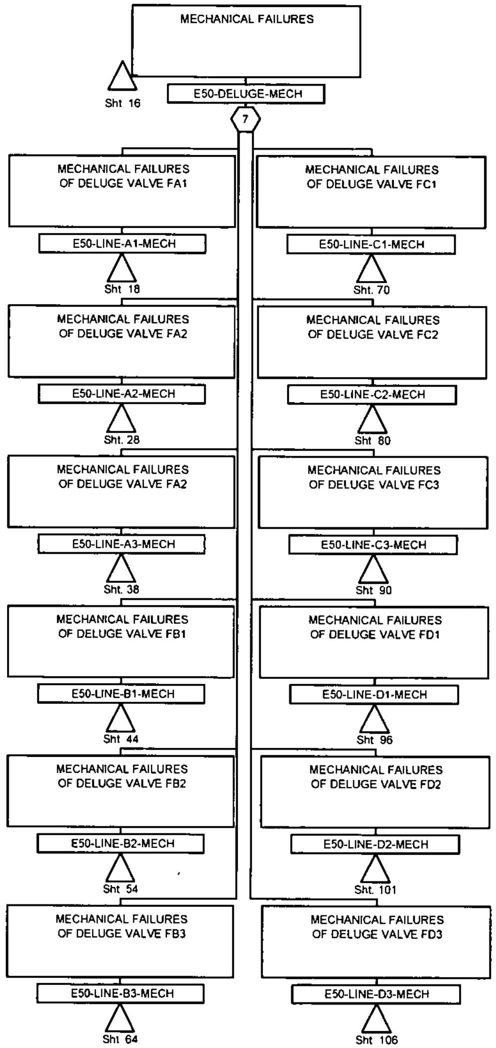

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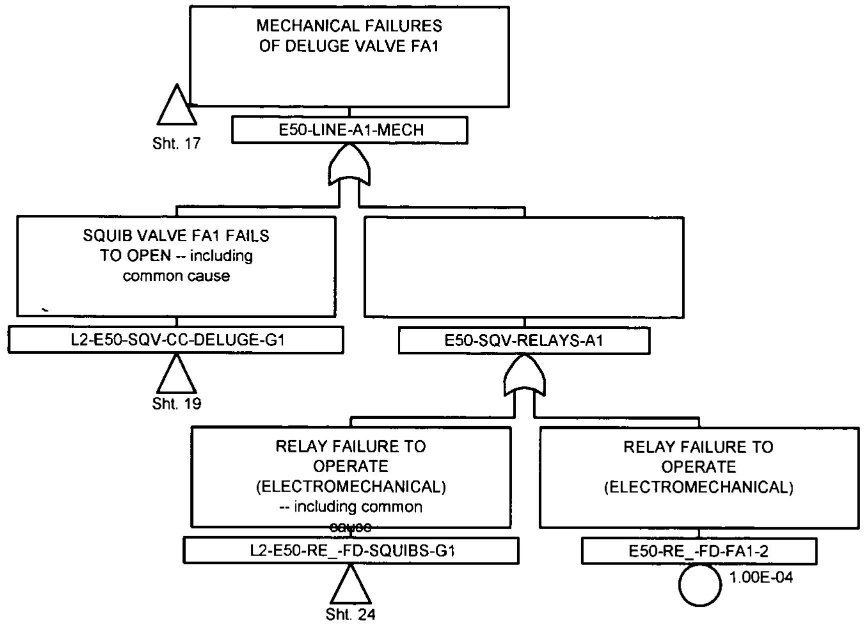

Figure 8A-8. Level 2 Fault Trees

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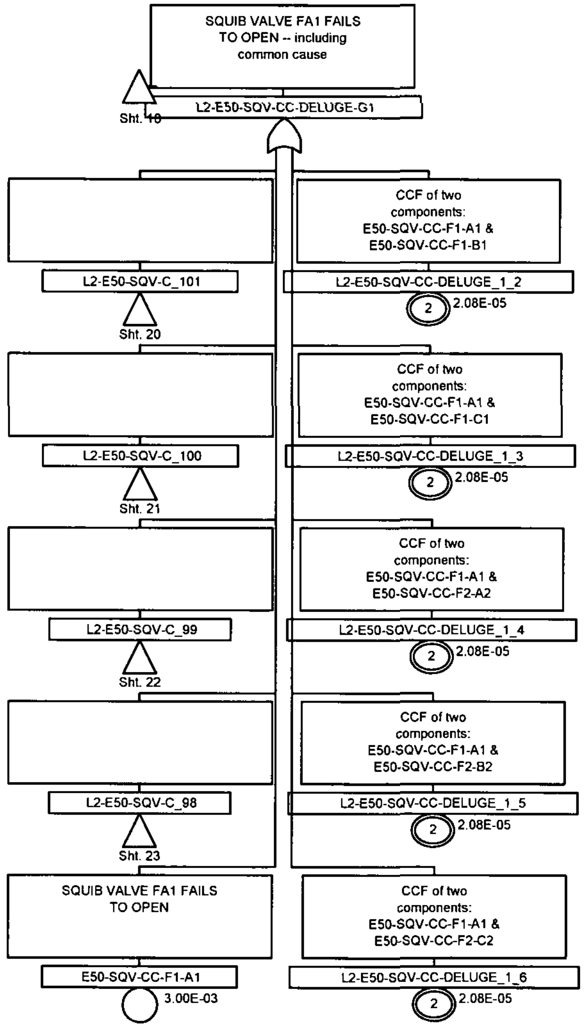

Figure 8A-8. Level 2 Fault Trees

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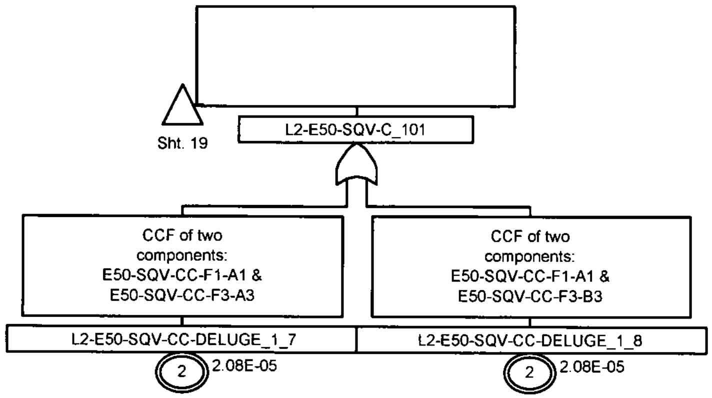

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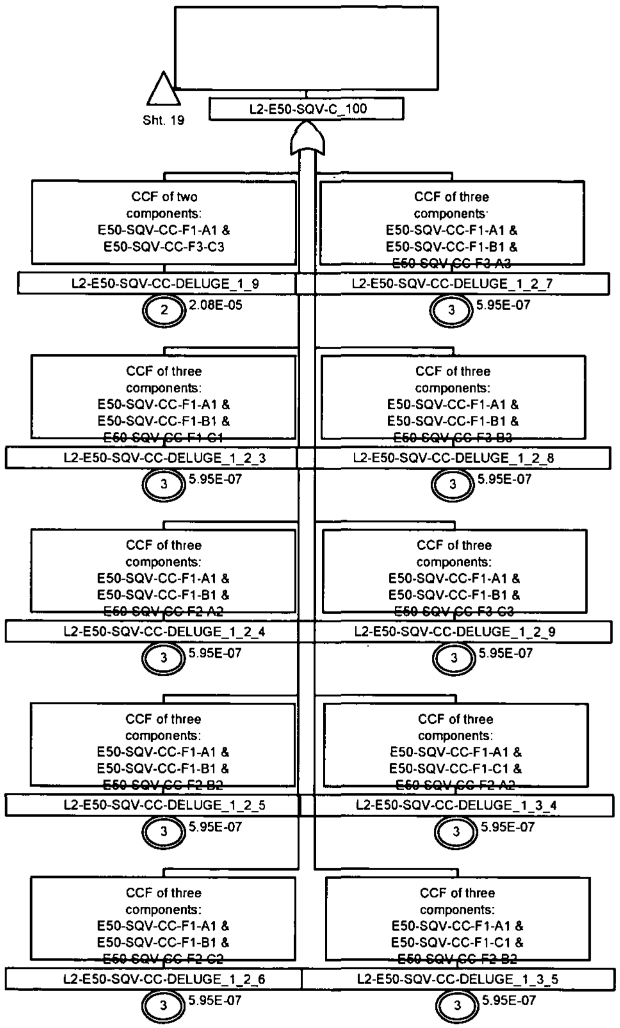

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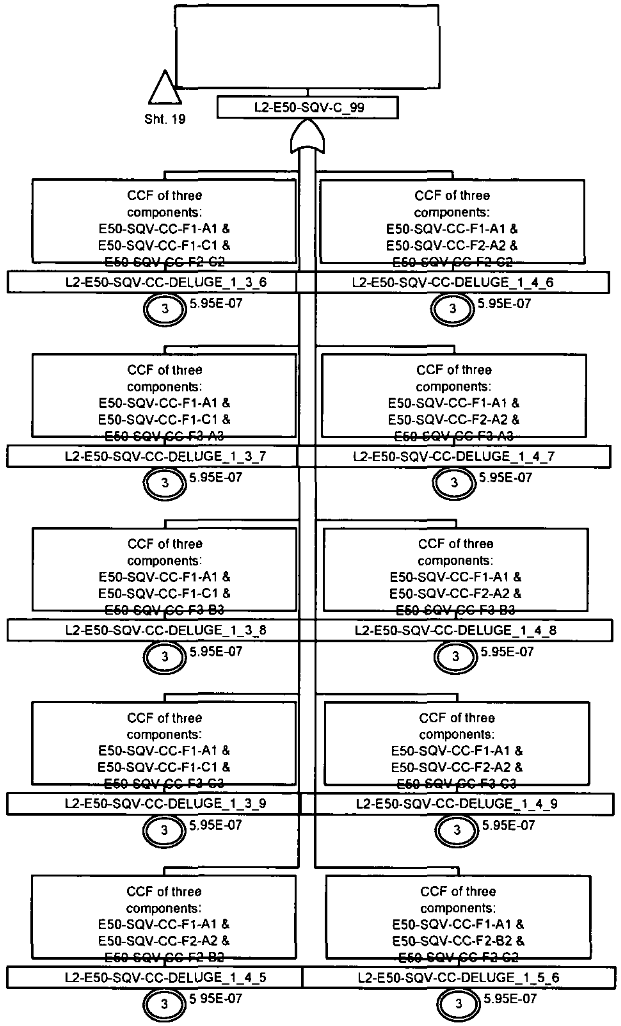

Figure 8A-8. Level 2 Fault Trees

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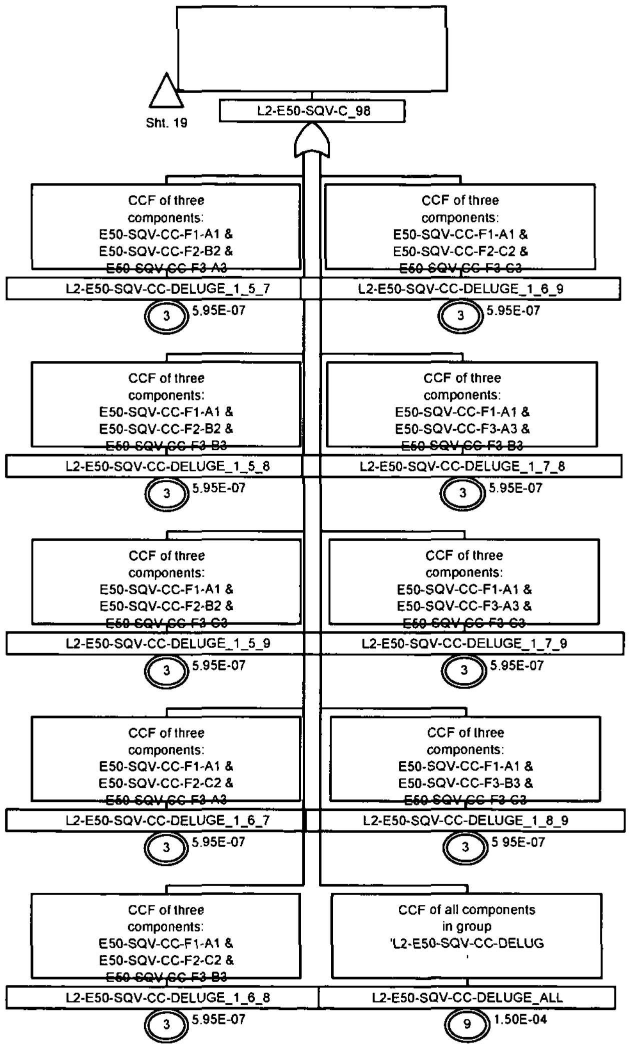

Figure 8A-8. Level 2 Fault Trees

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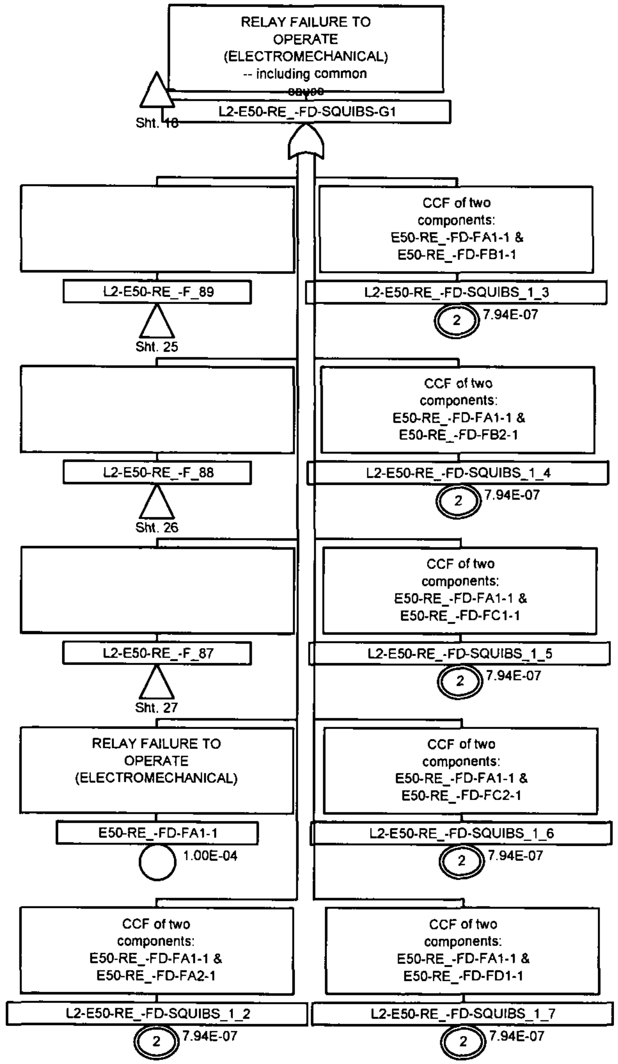

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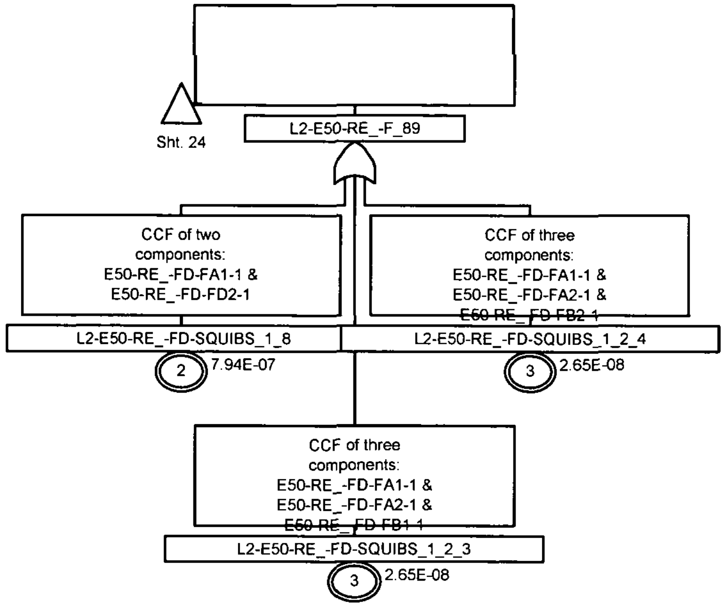

Figure 8A-8. Level 2 Fault Trees

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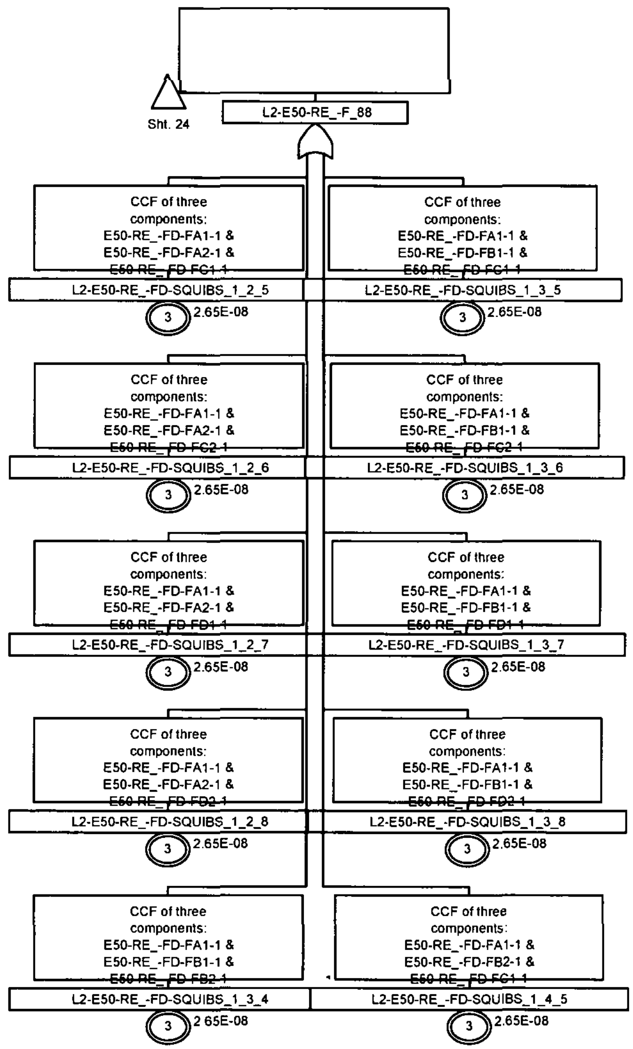

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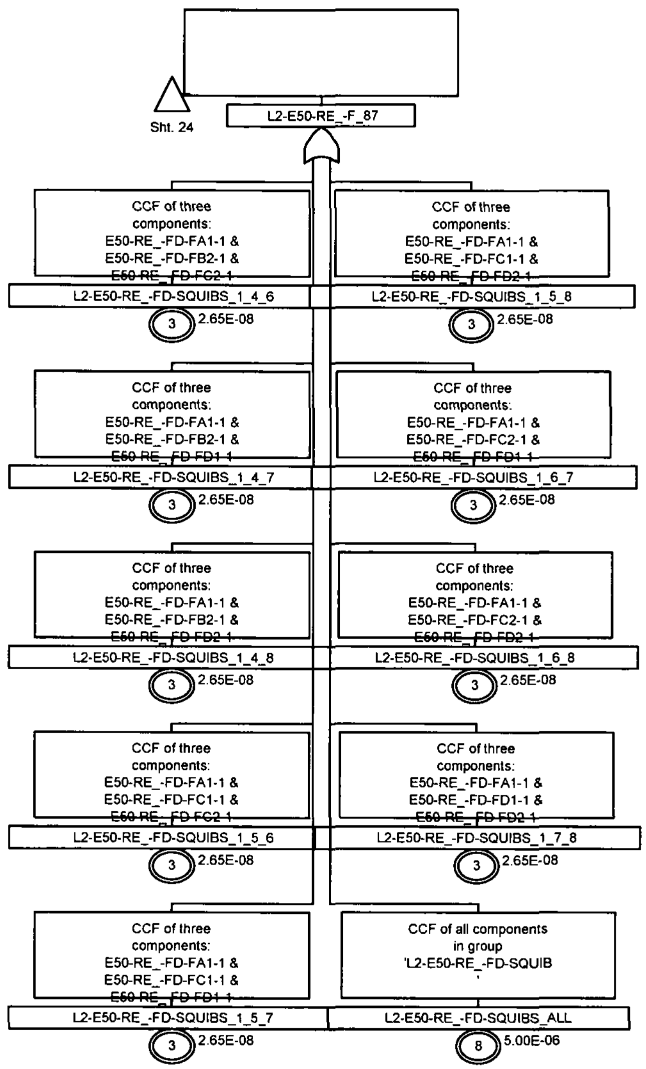

Figure 8A-8. Level 2 Fault Trees

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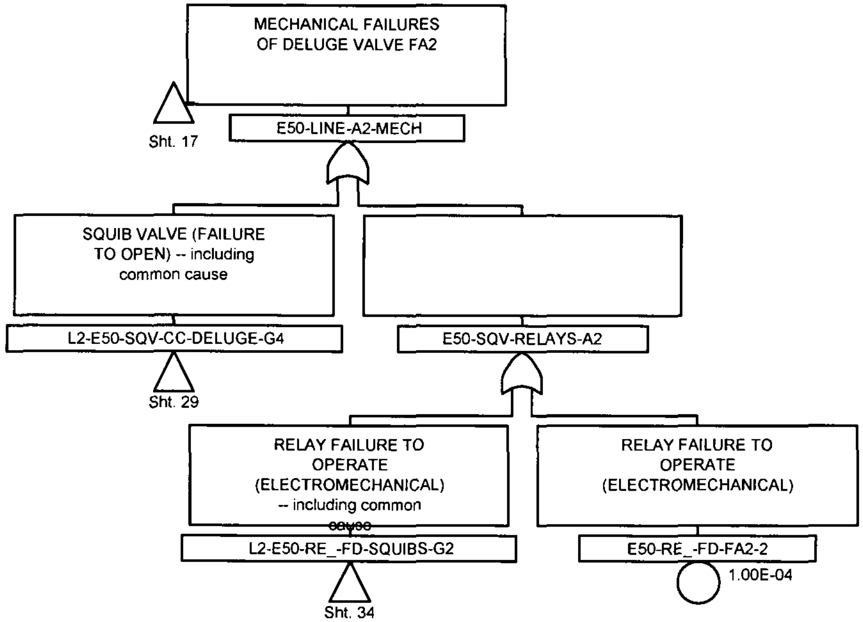

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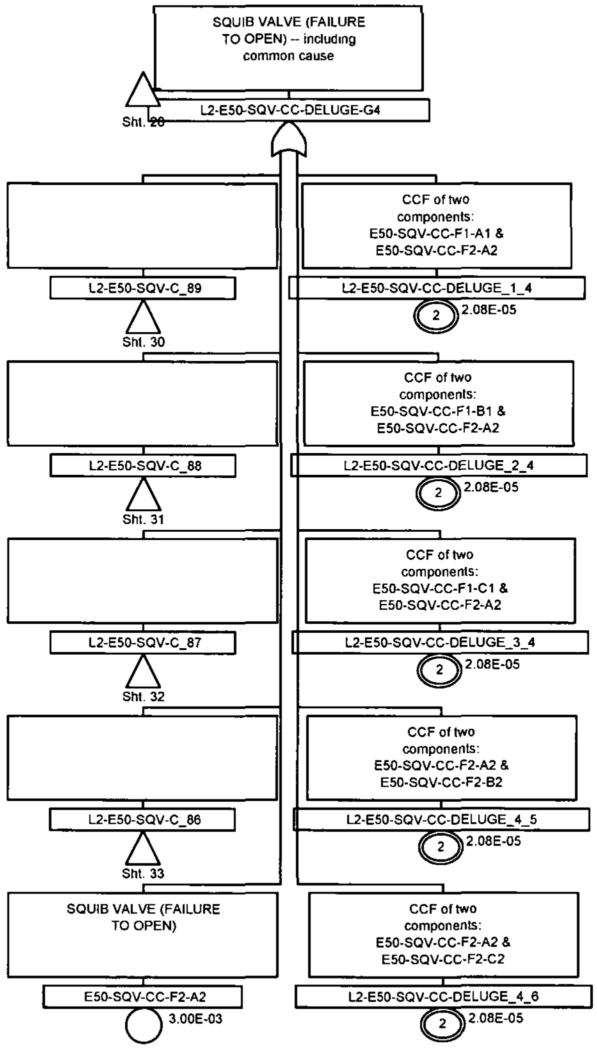

Figure 8A-8. Level 2 Fault Trees

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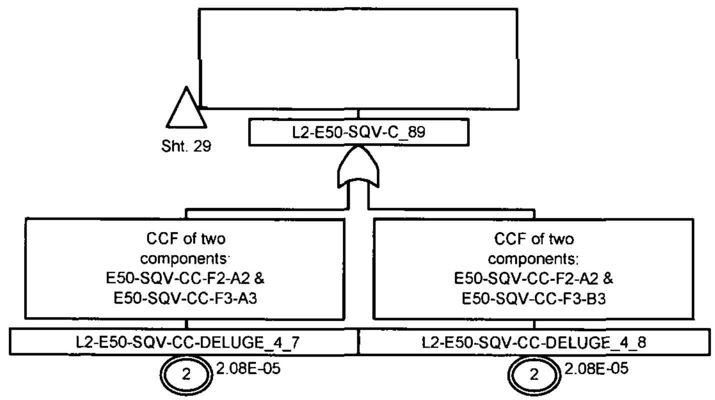

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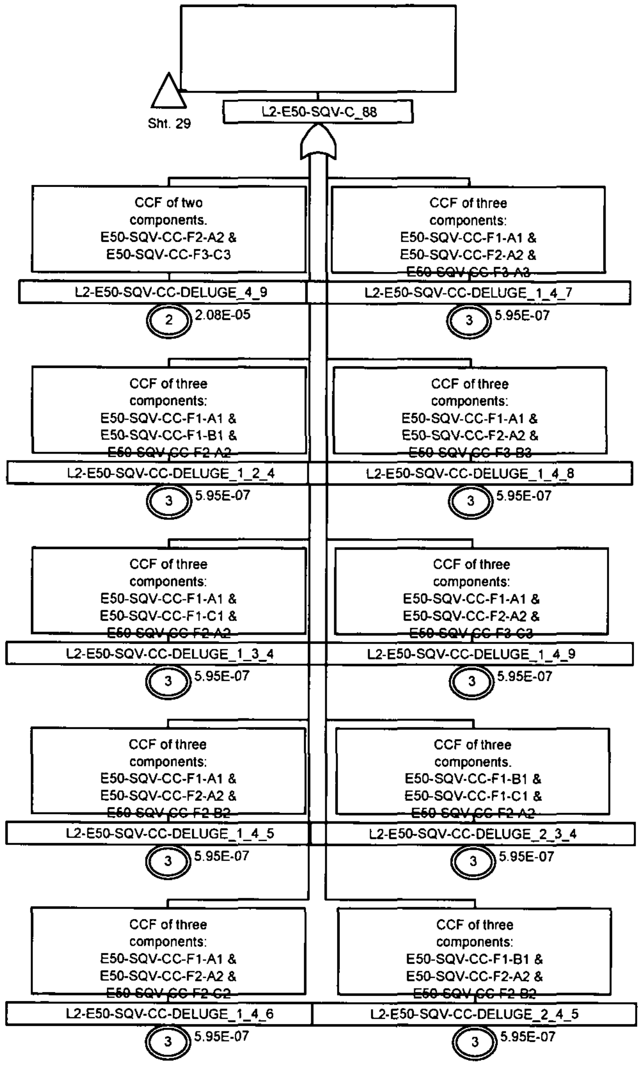

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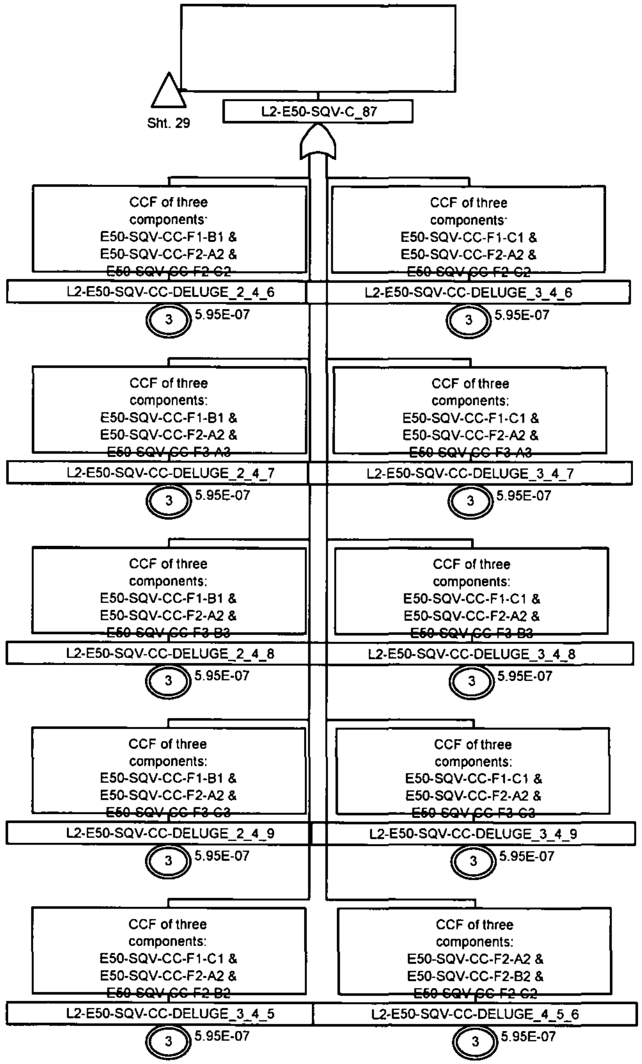

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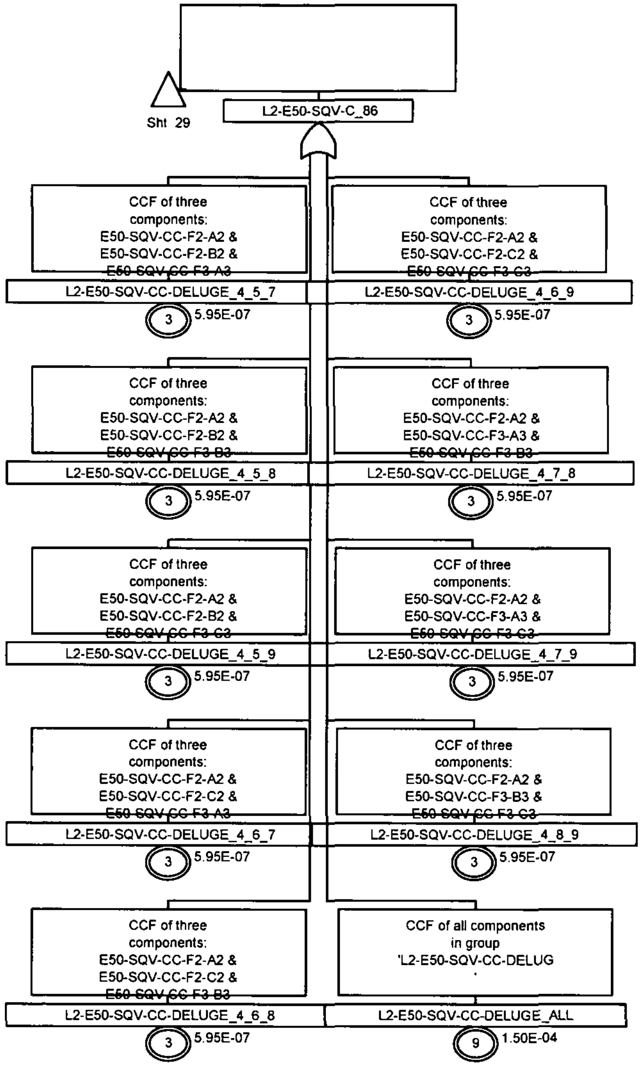

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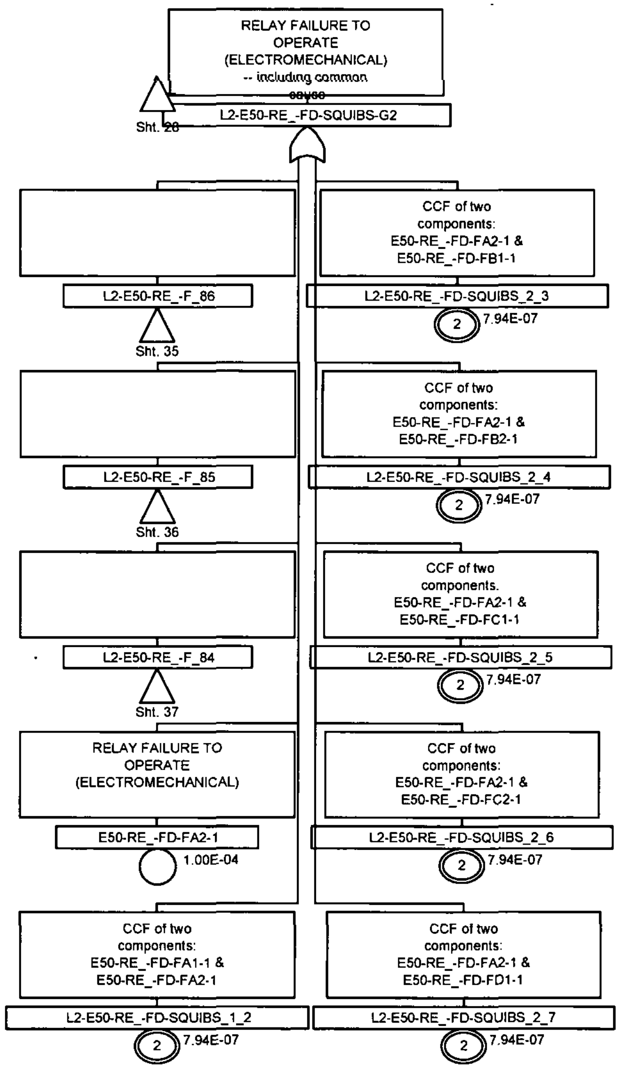

Figure 8A-8. Level 2 Fault Trees
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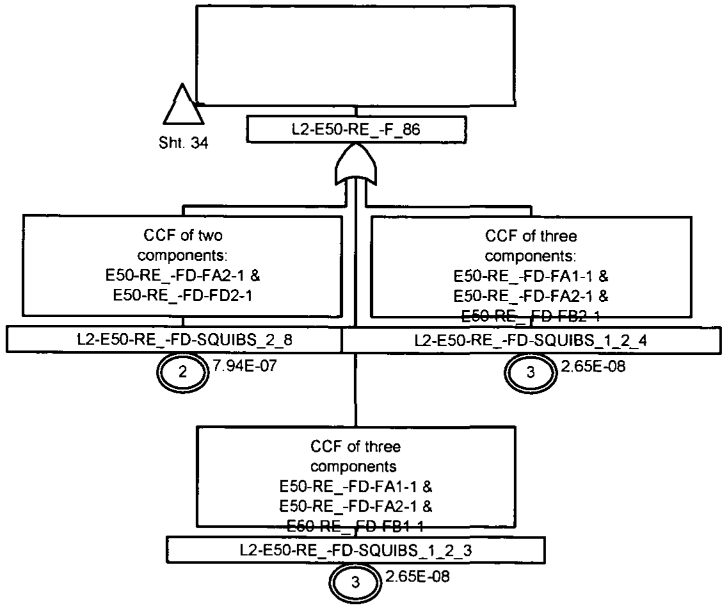

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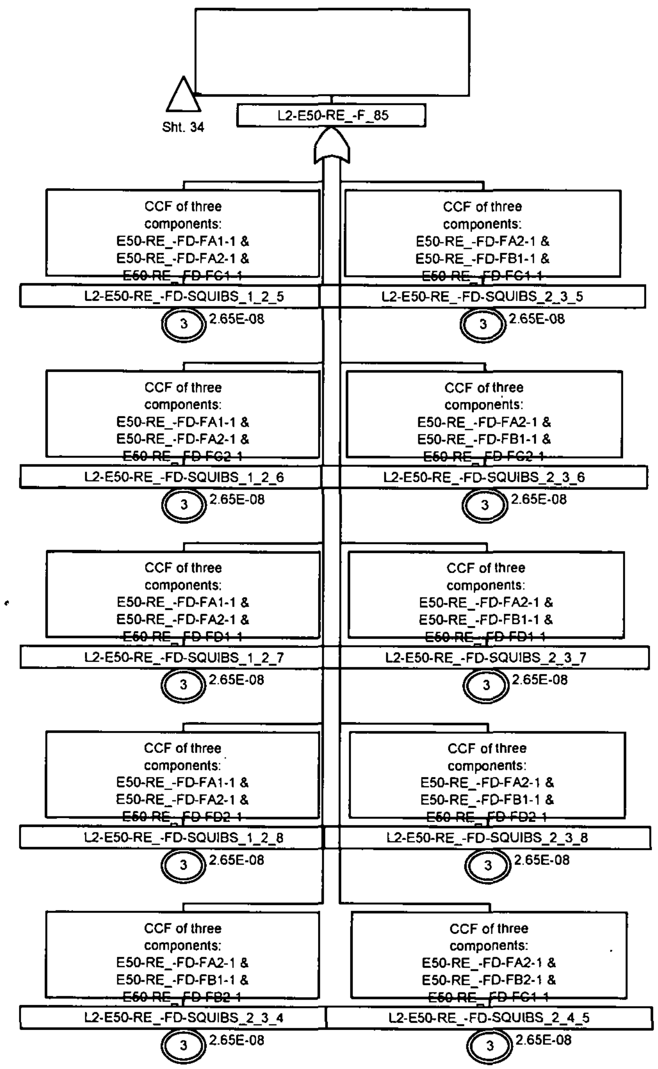

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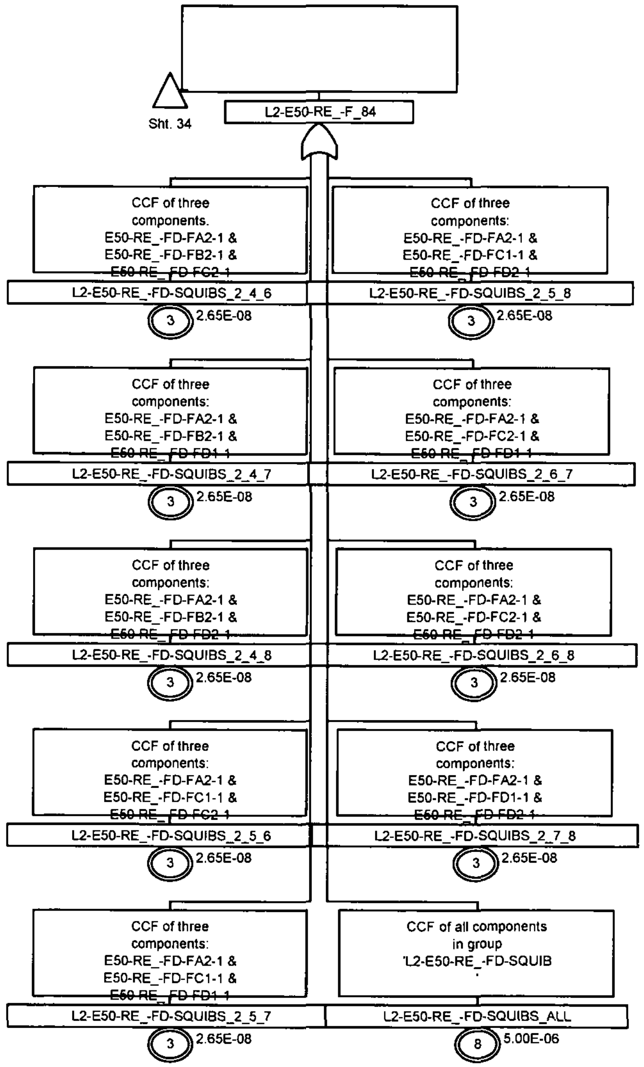

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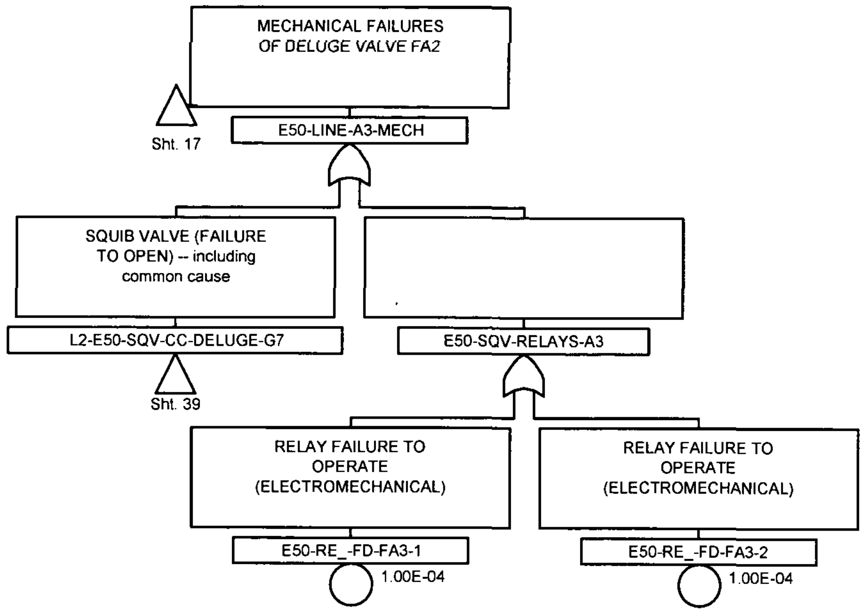

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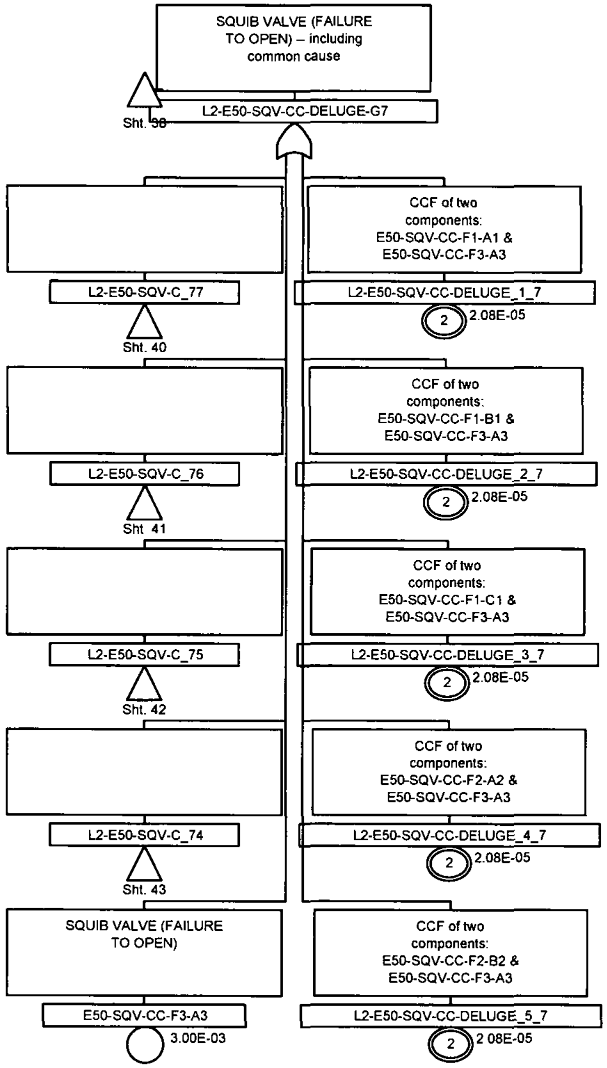

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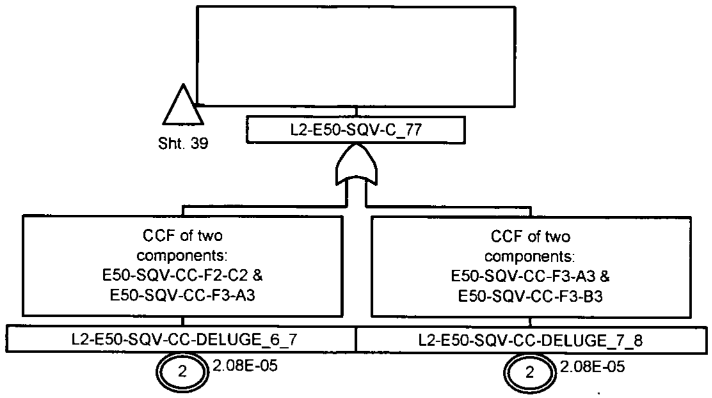

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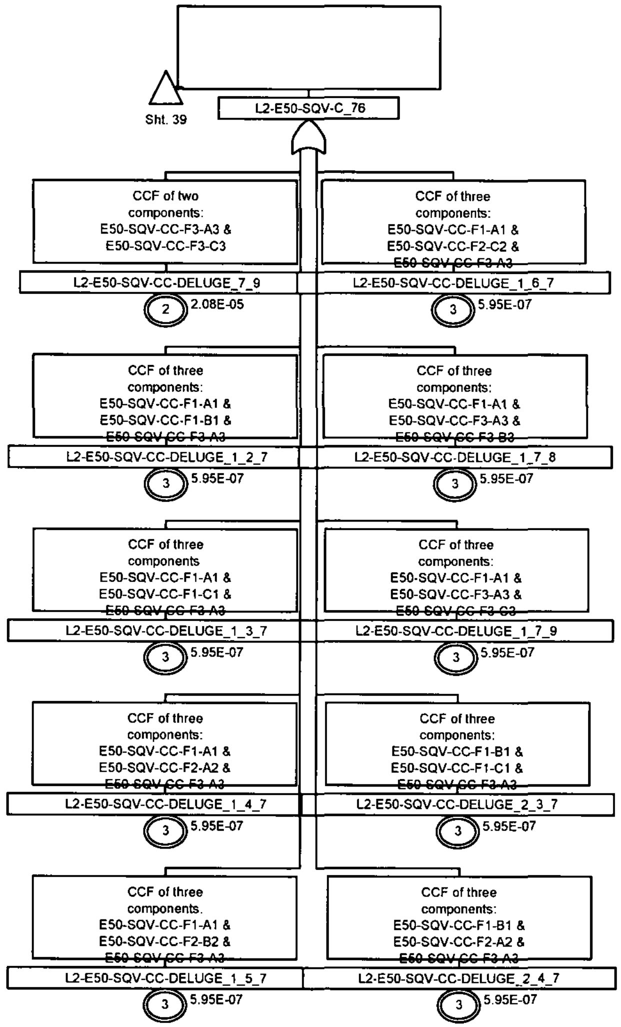

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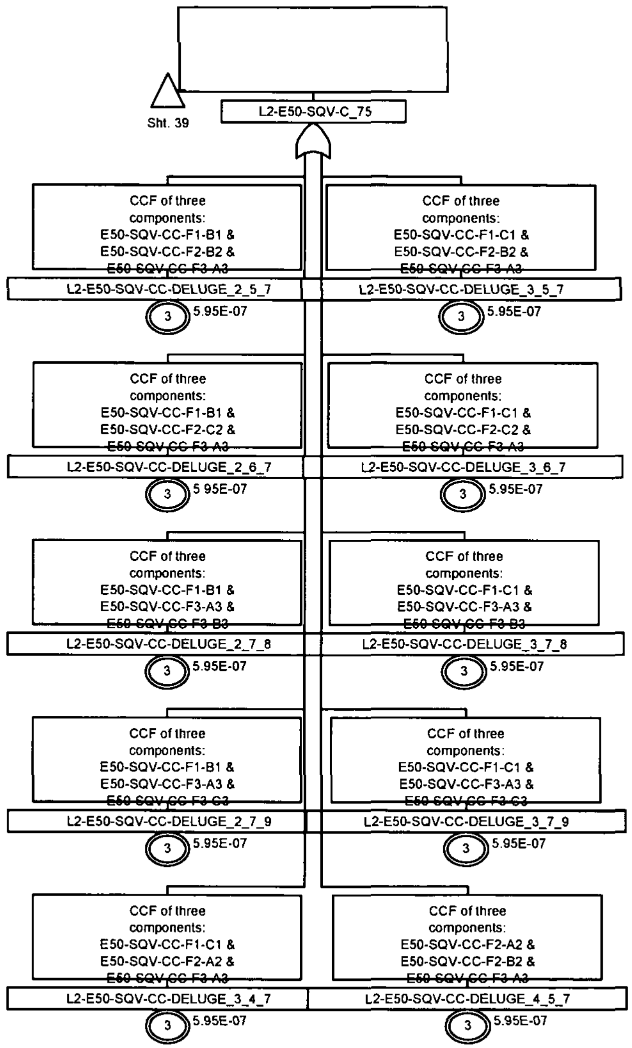

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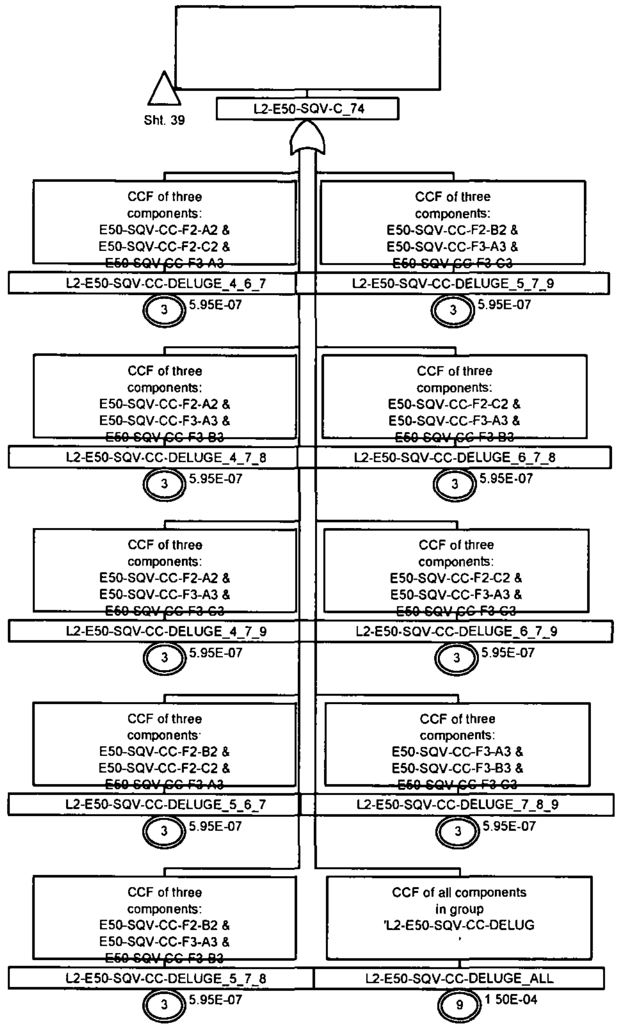

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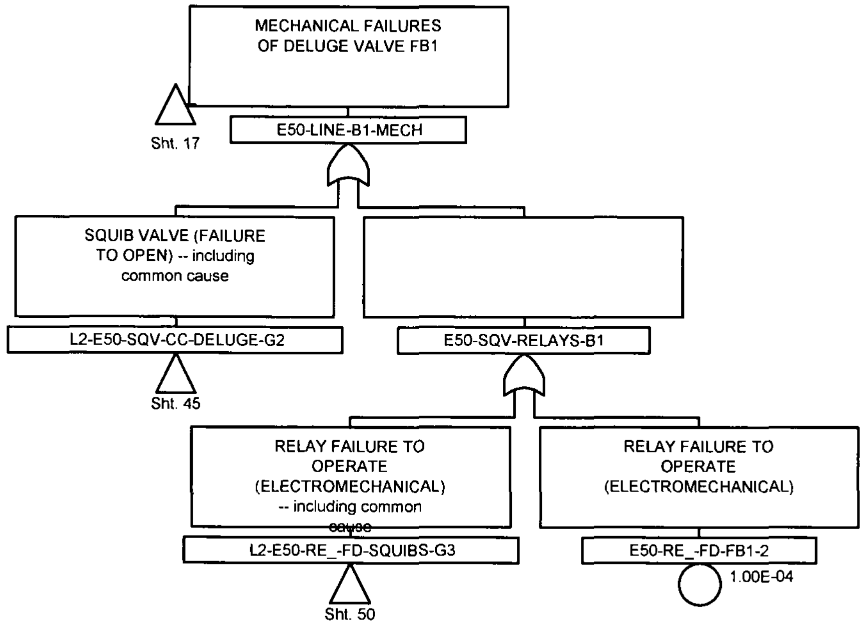

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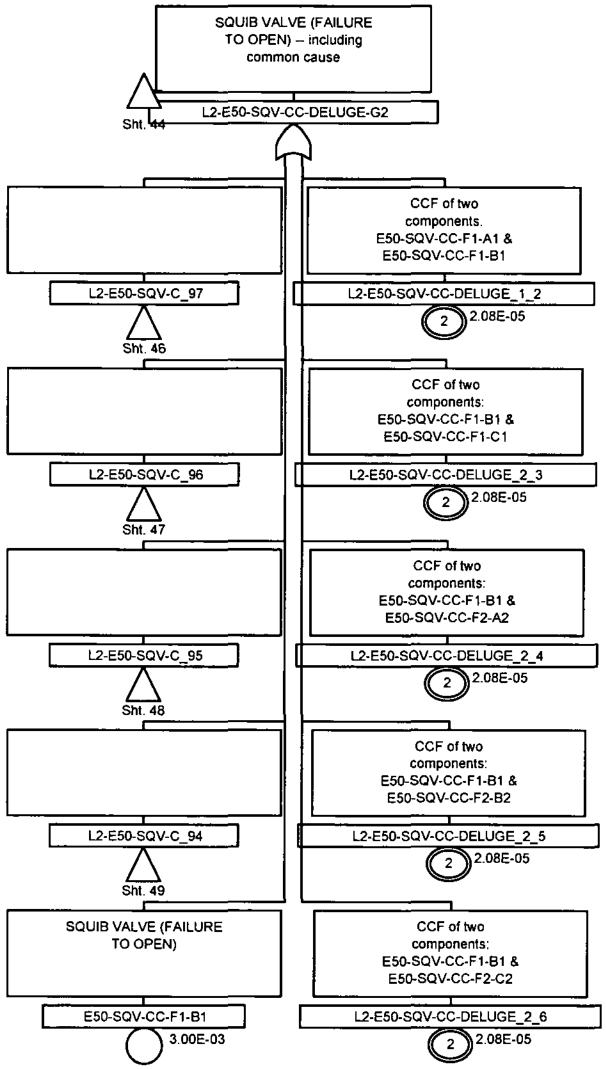

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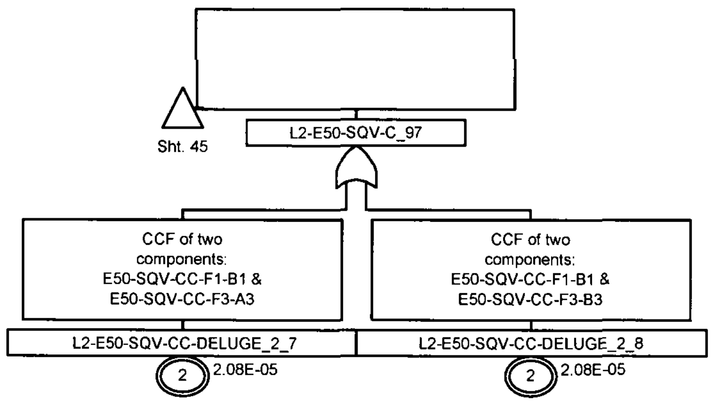

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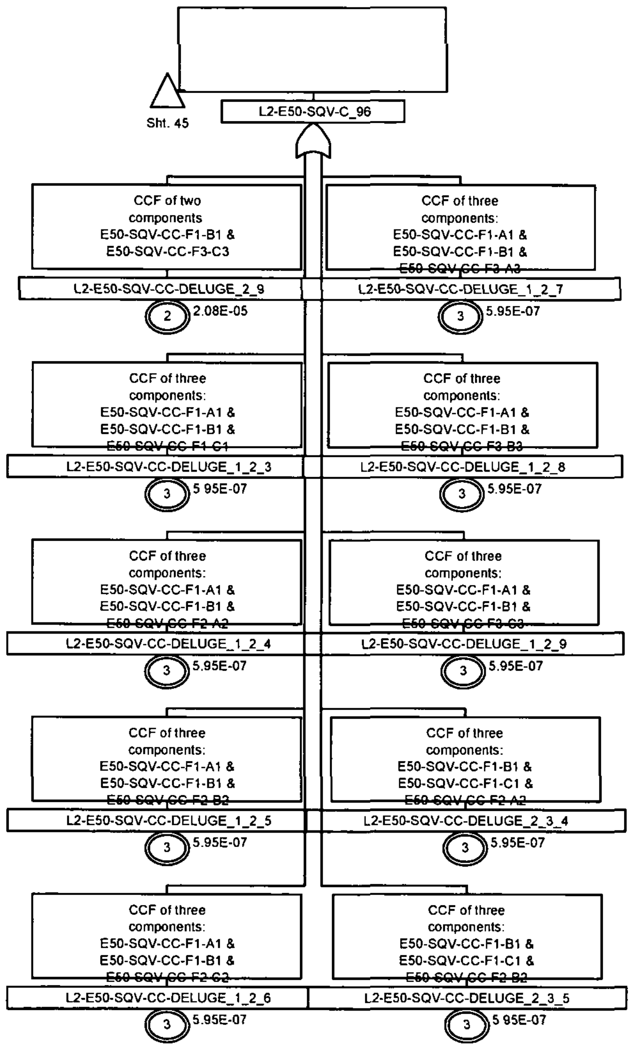

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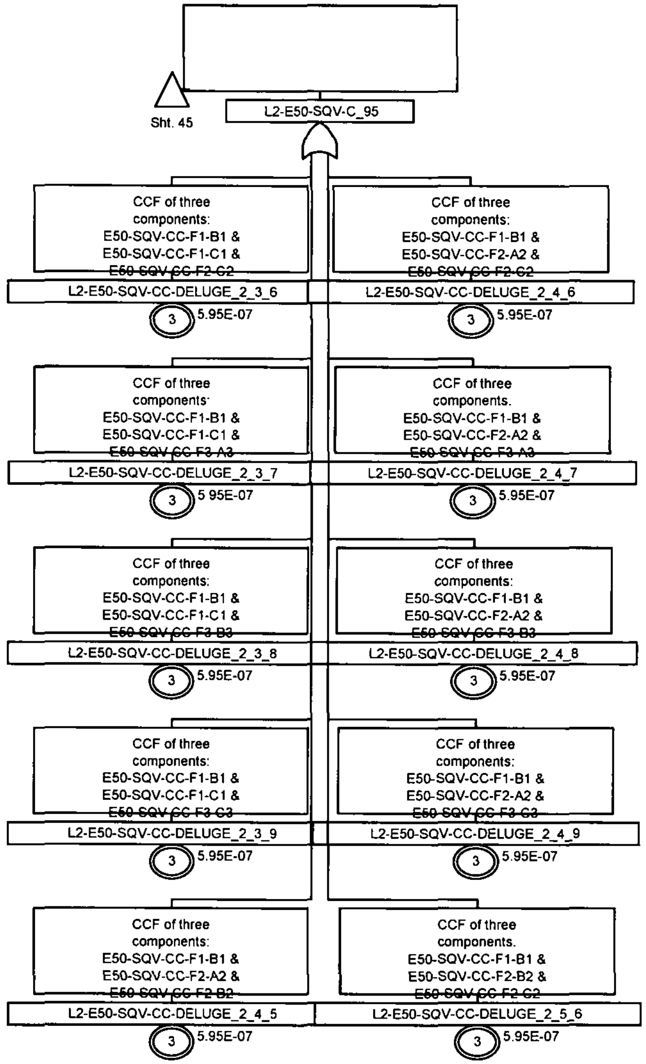

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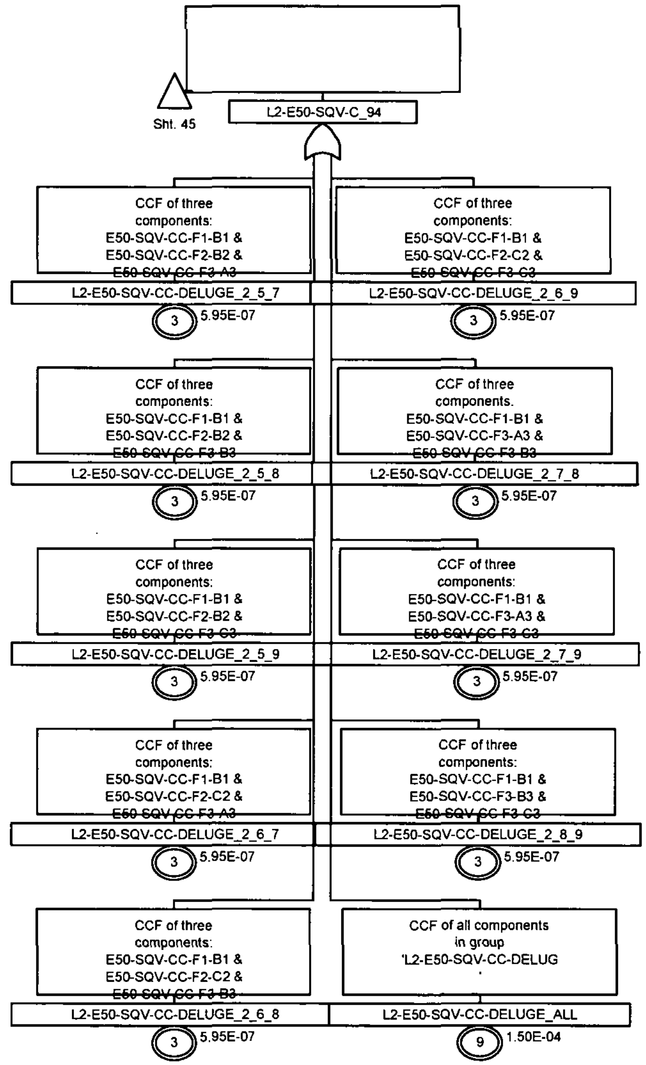

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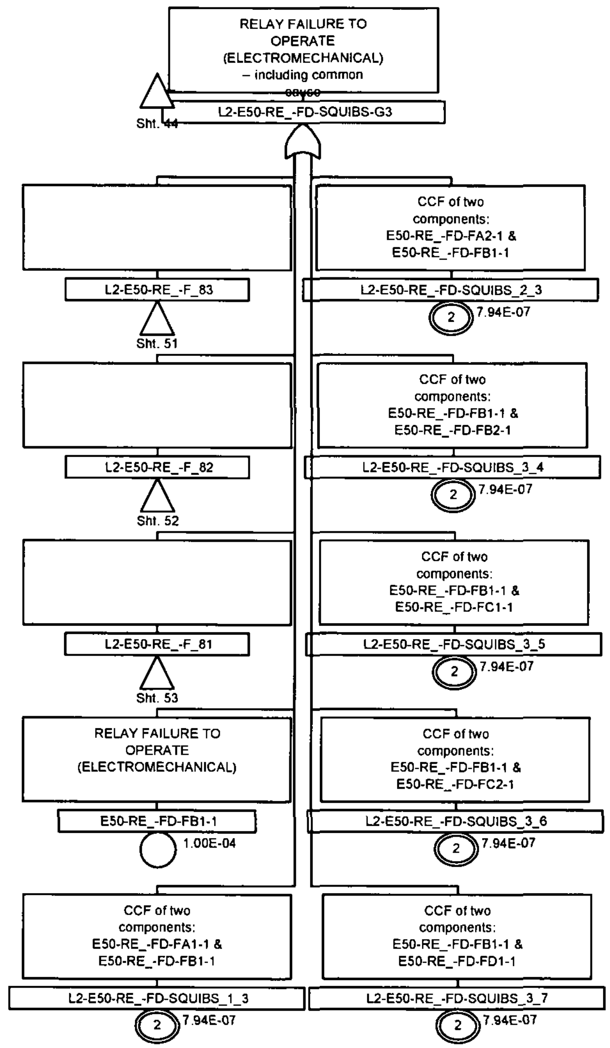

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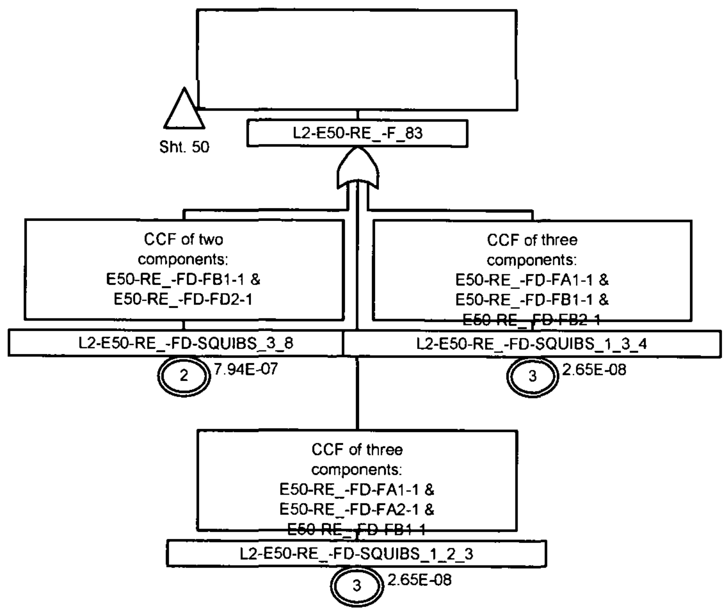

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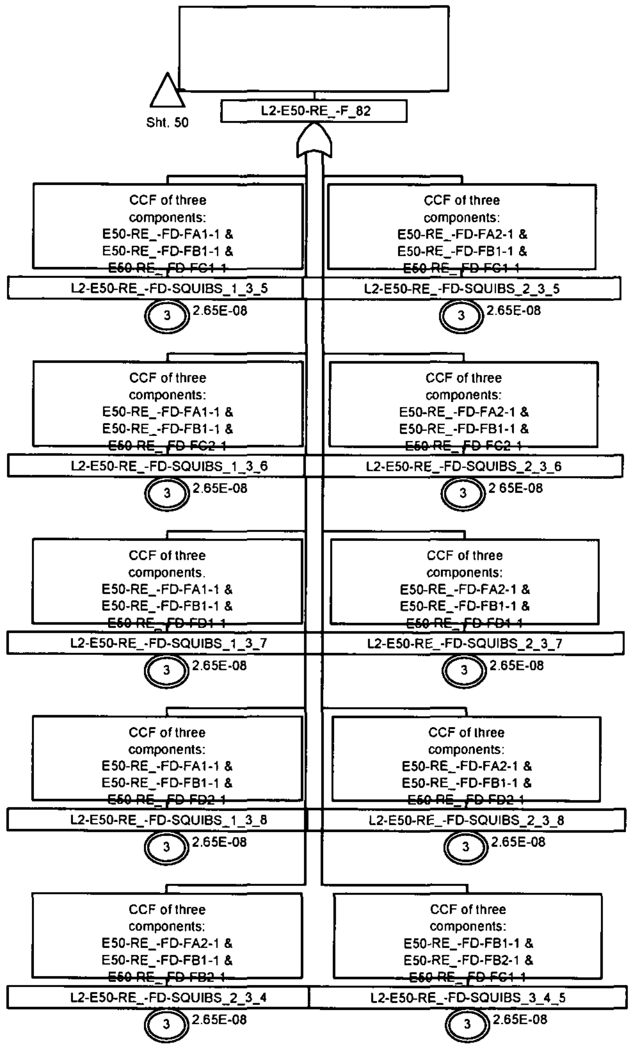

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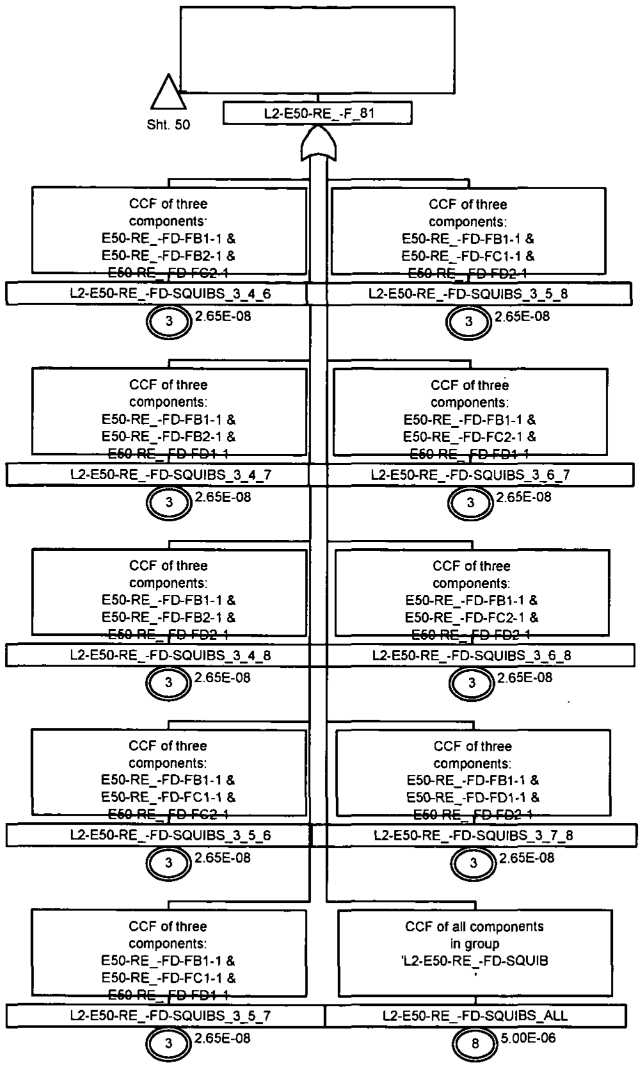

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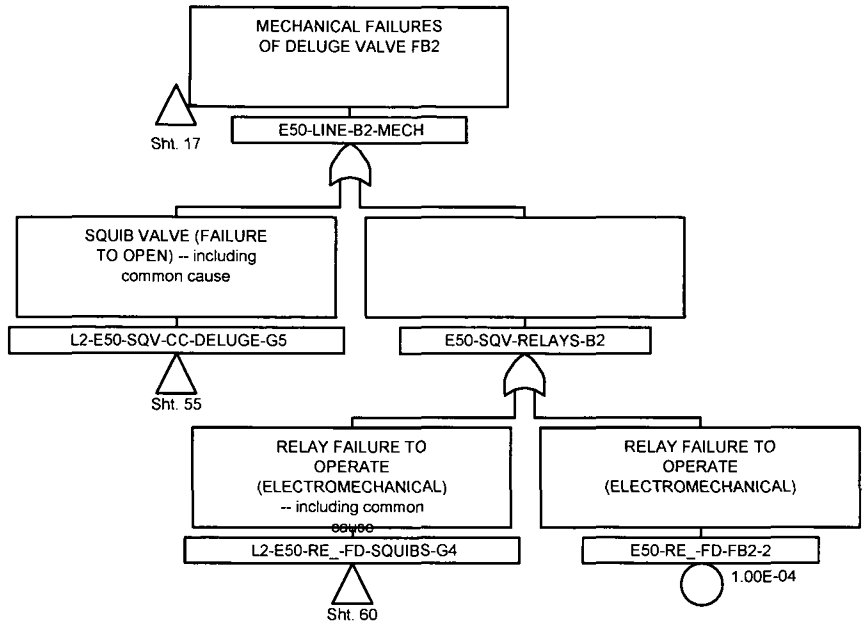

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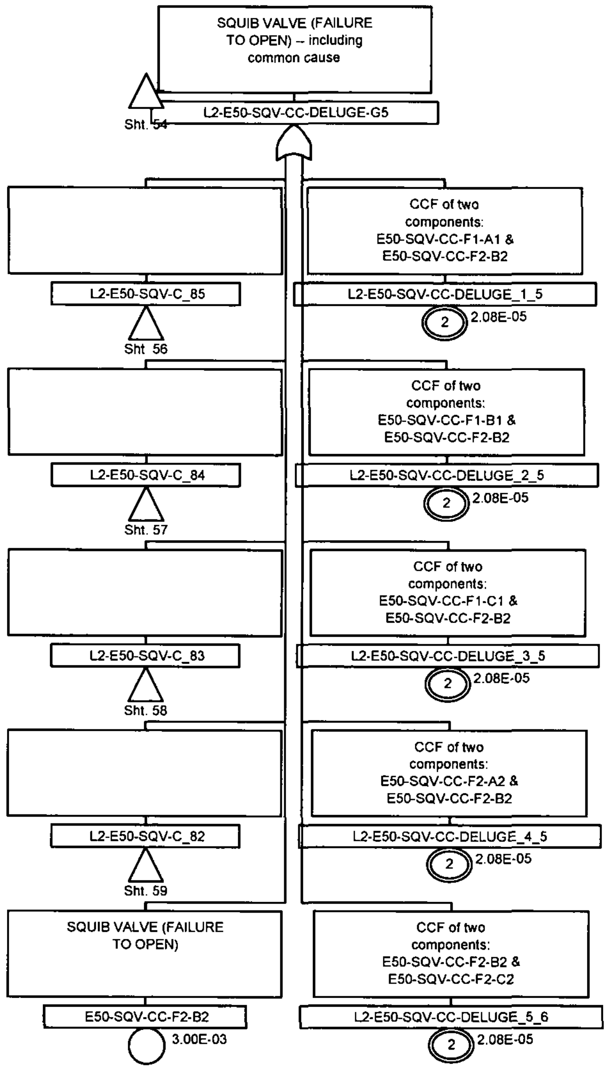

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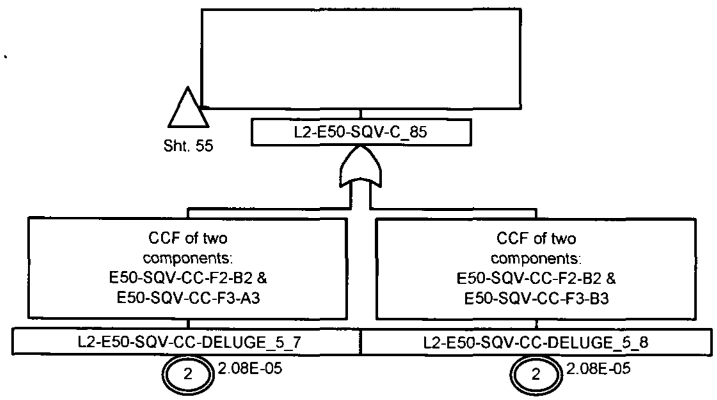

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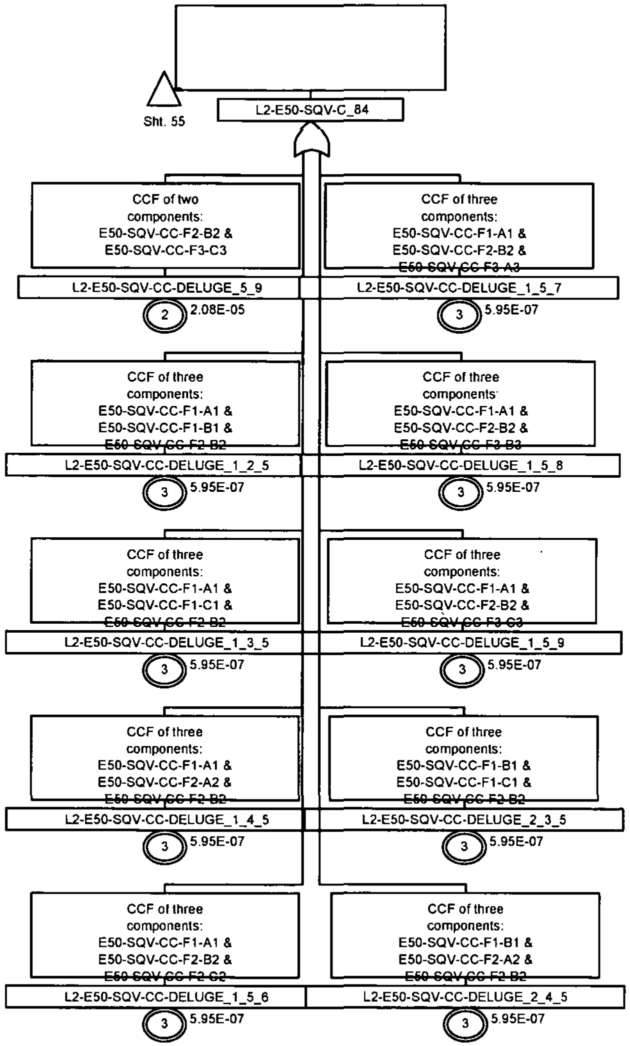

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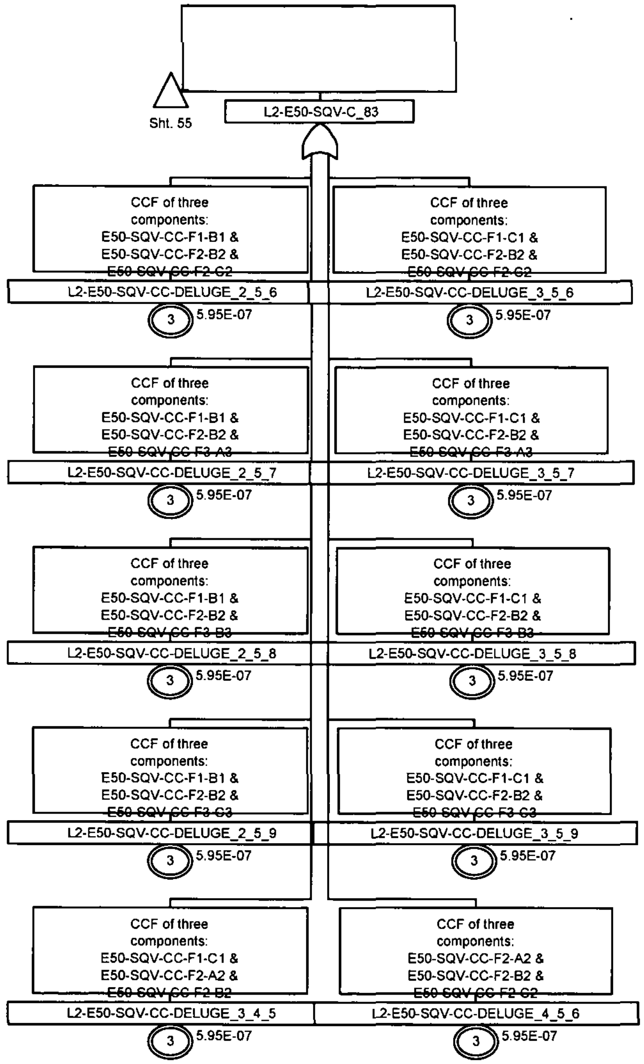

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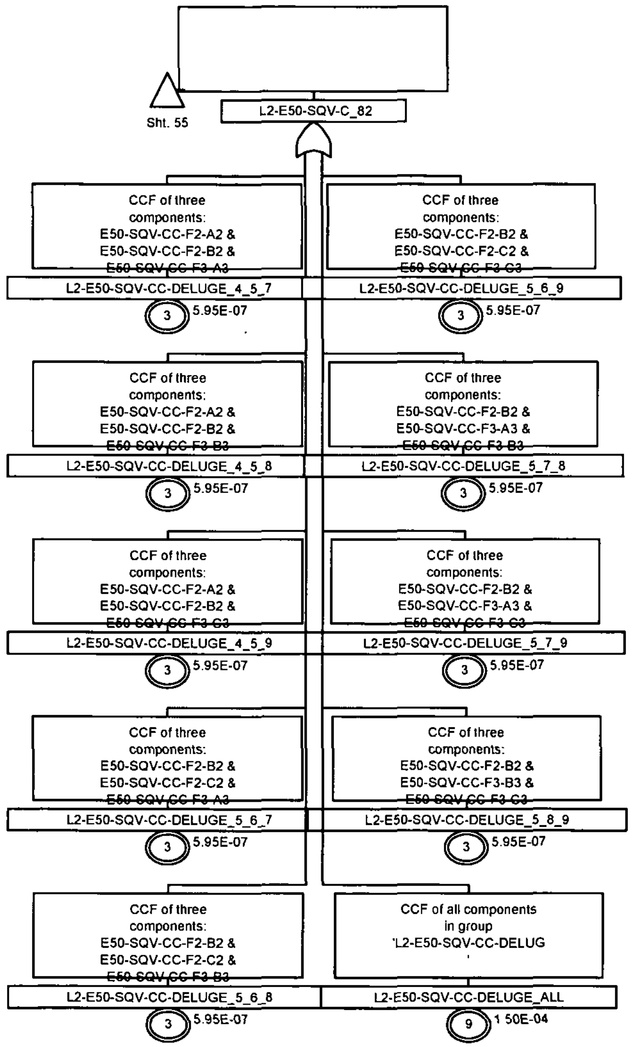

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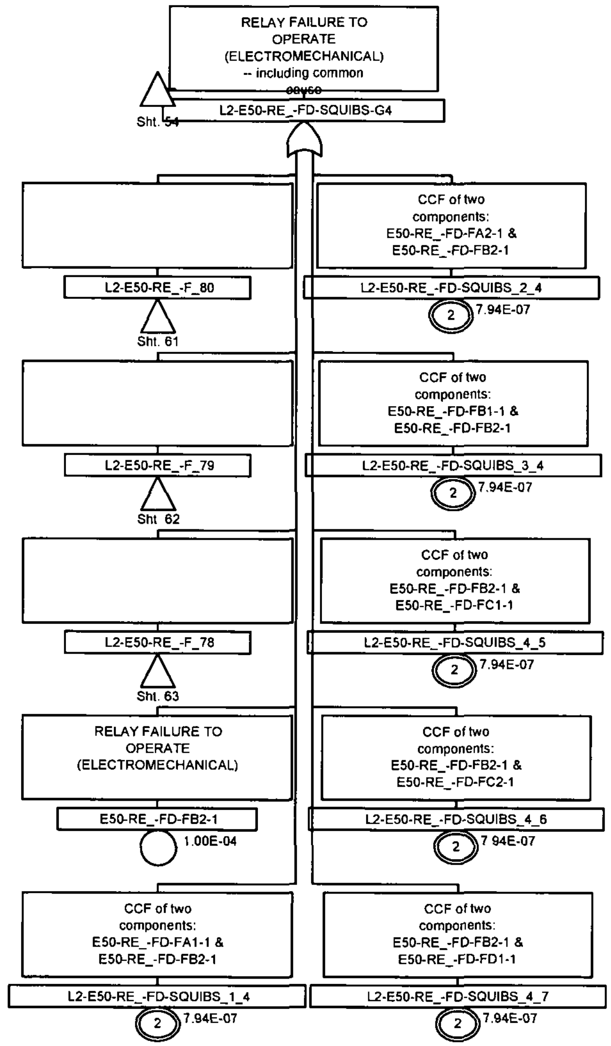

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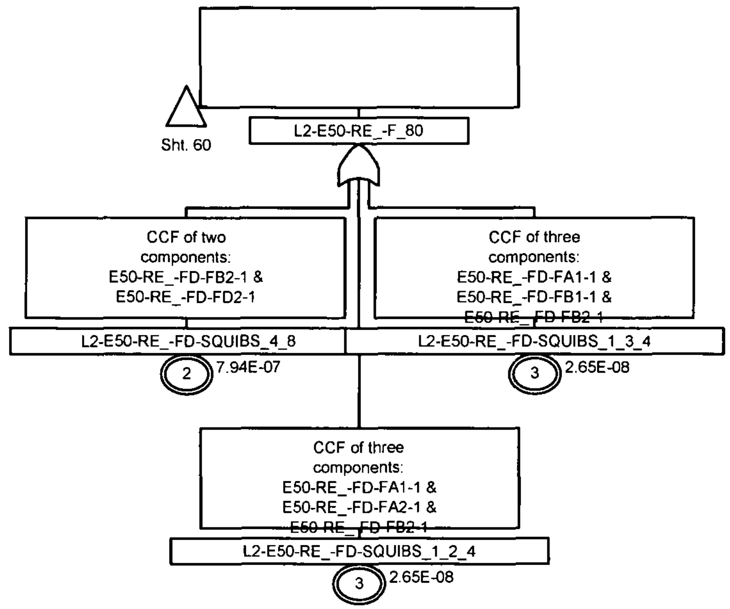

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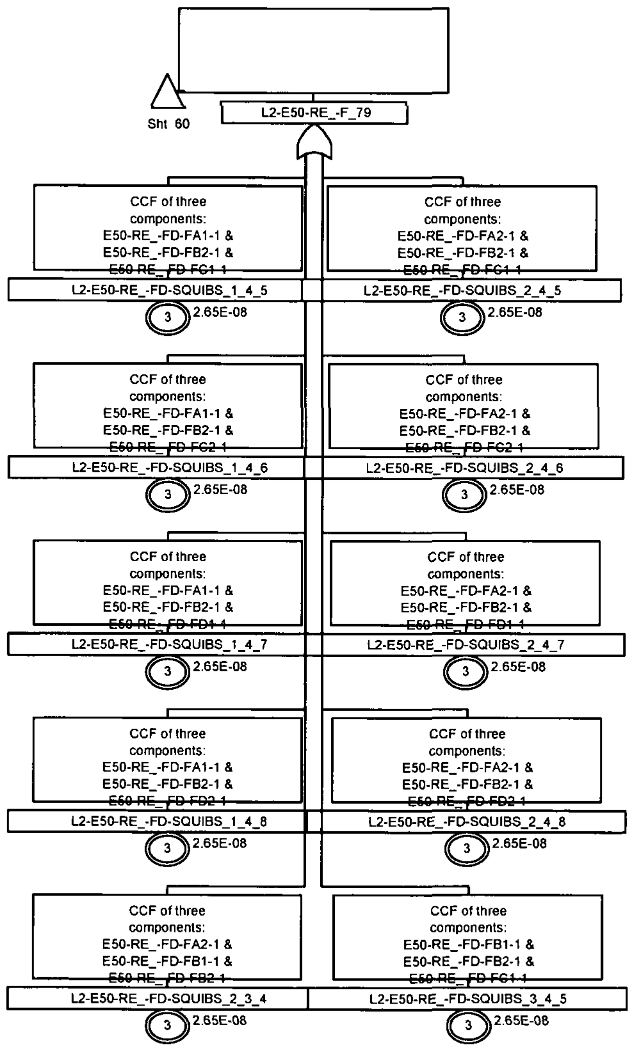

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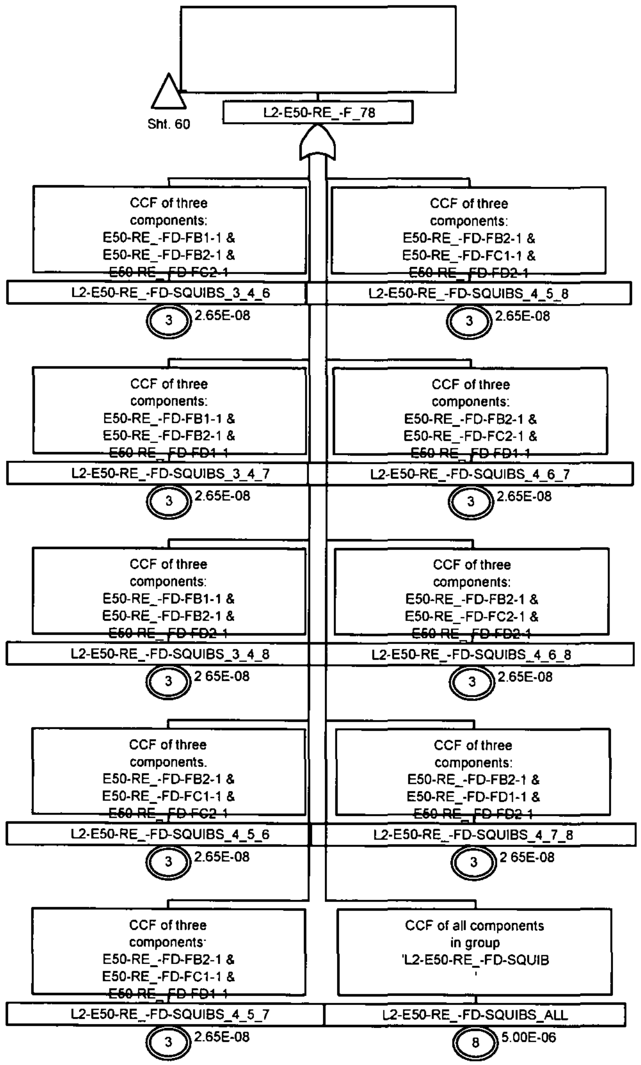

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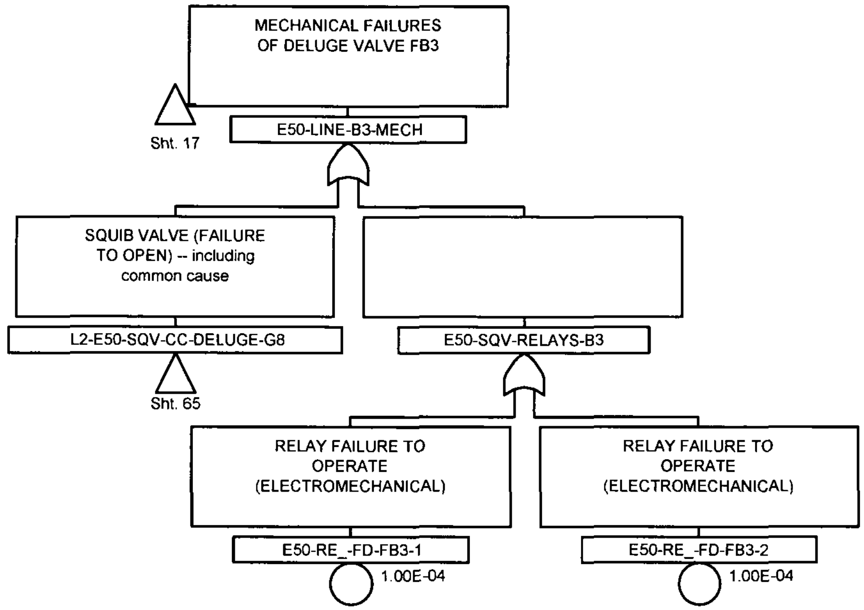

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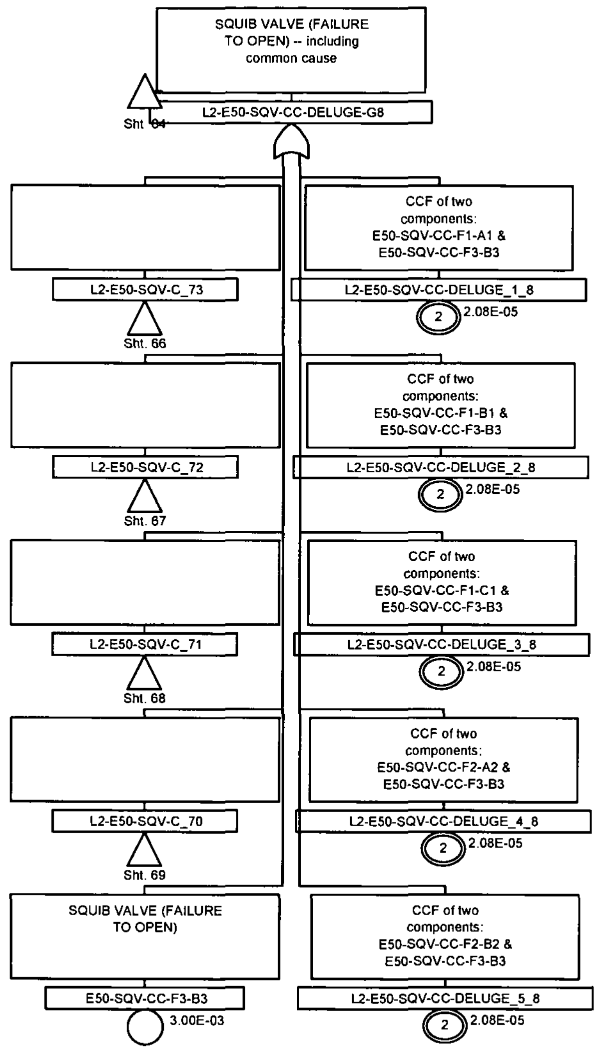

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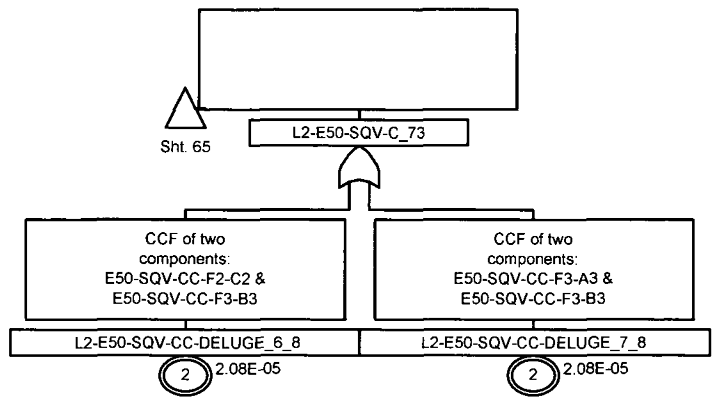

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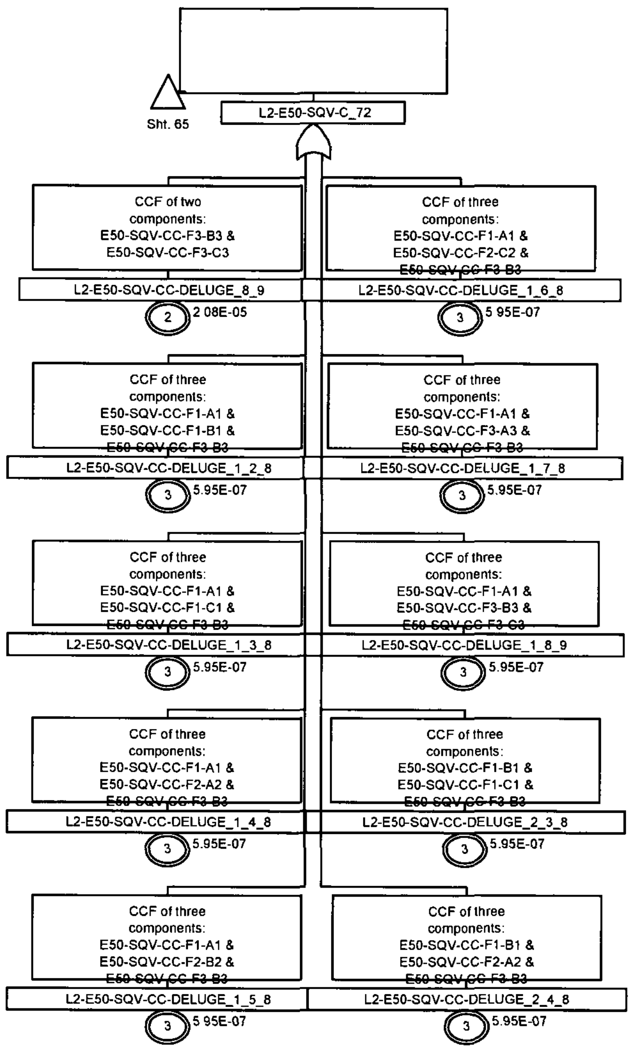

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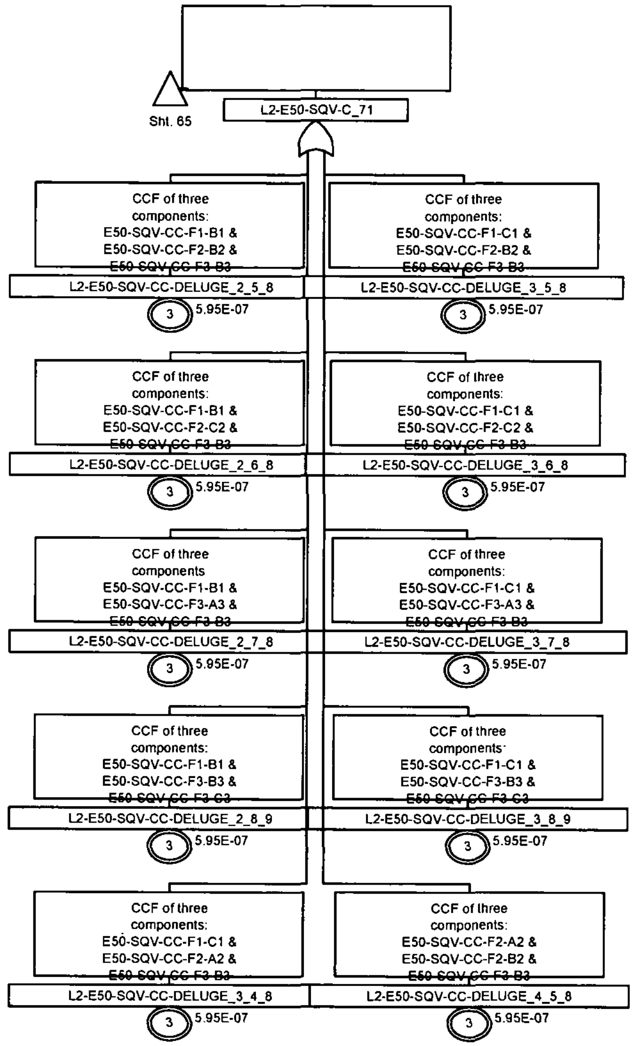

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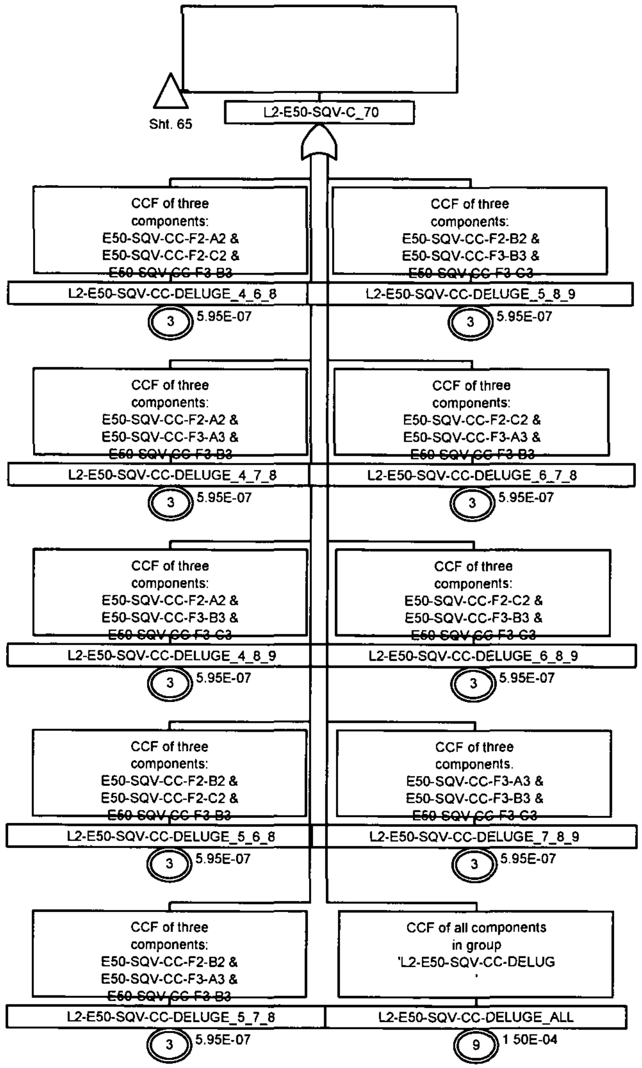

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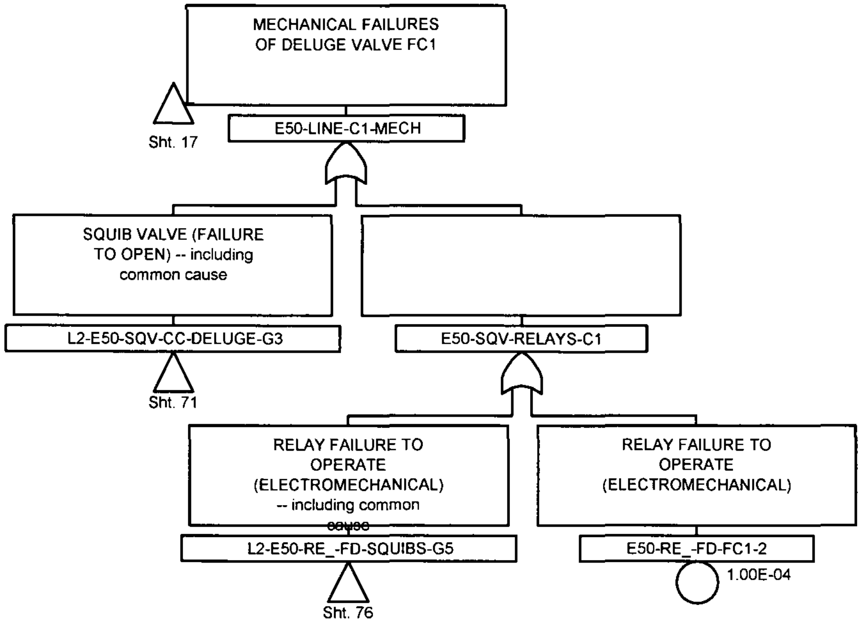

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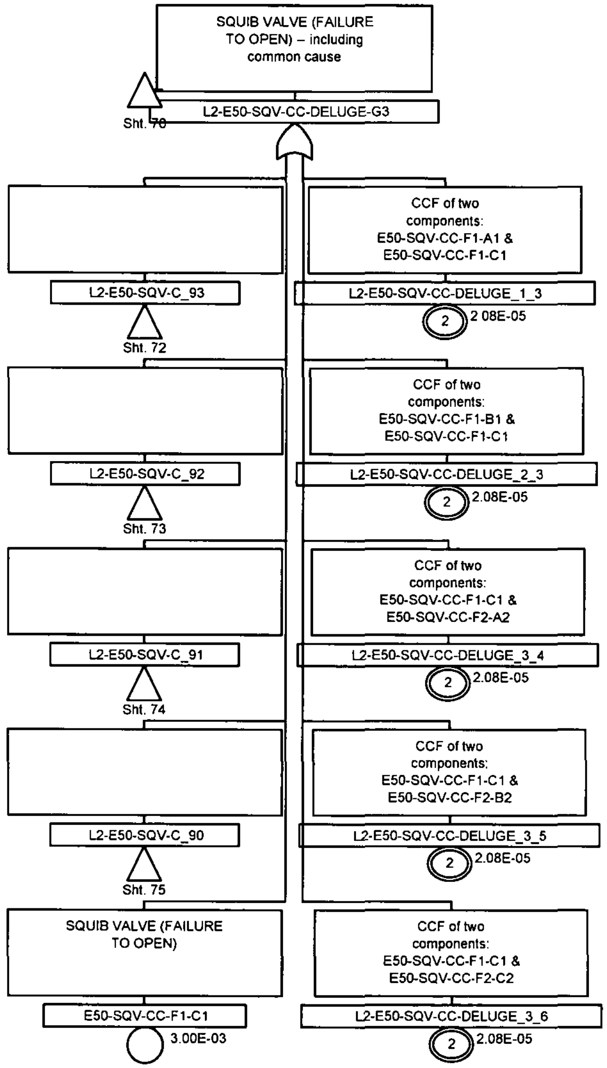

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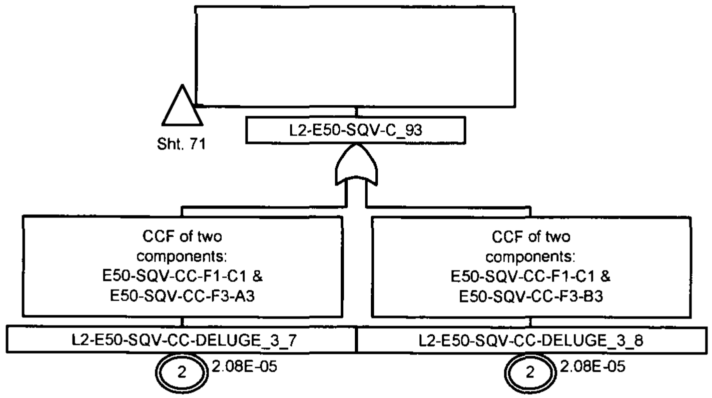

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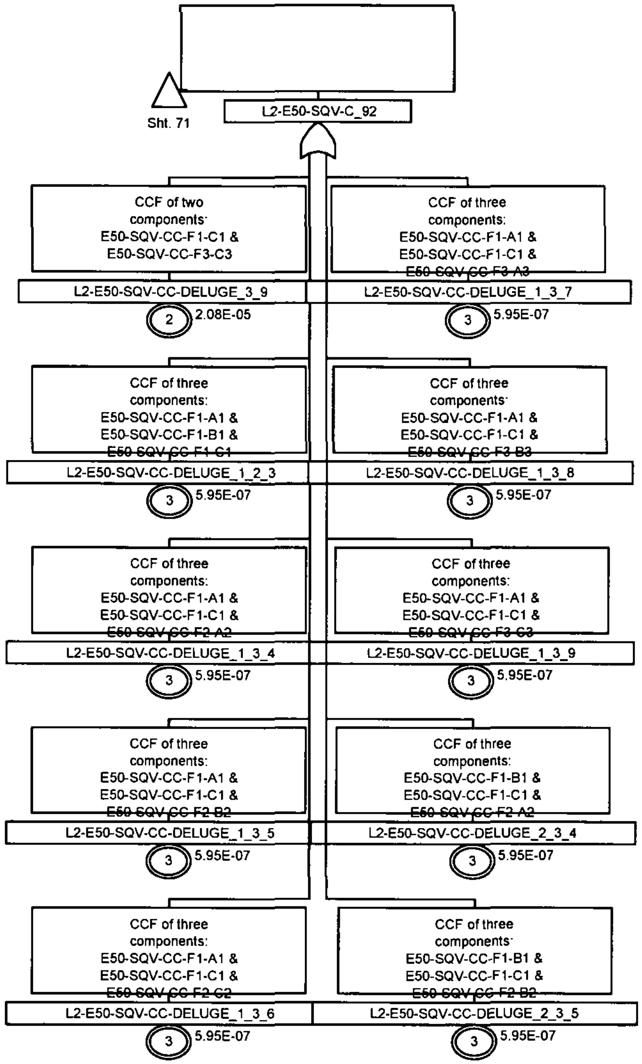

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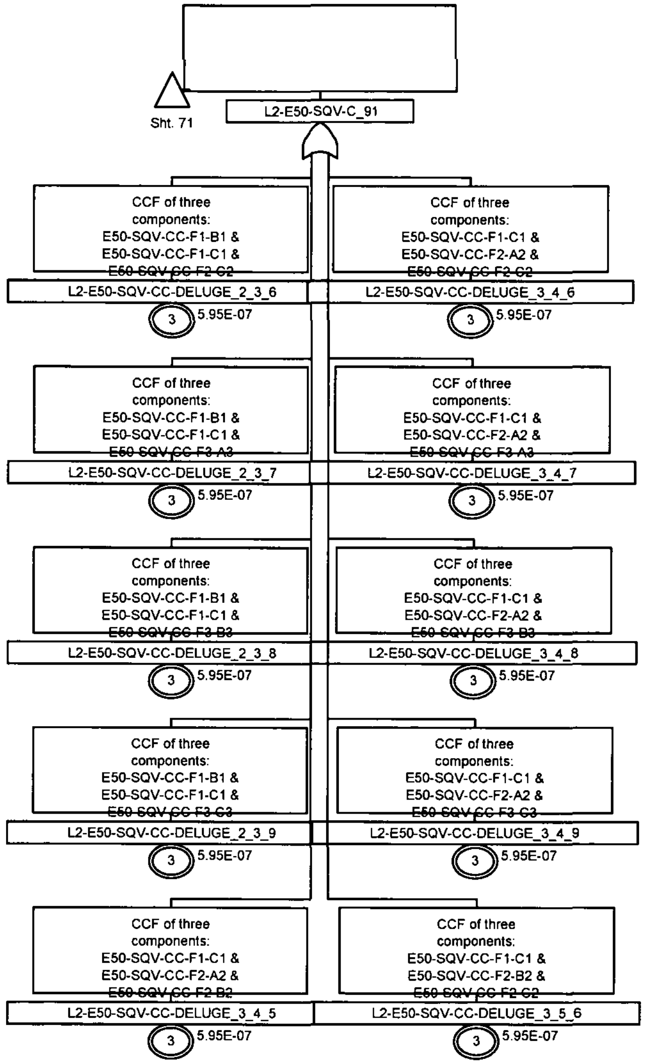

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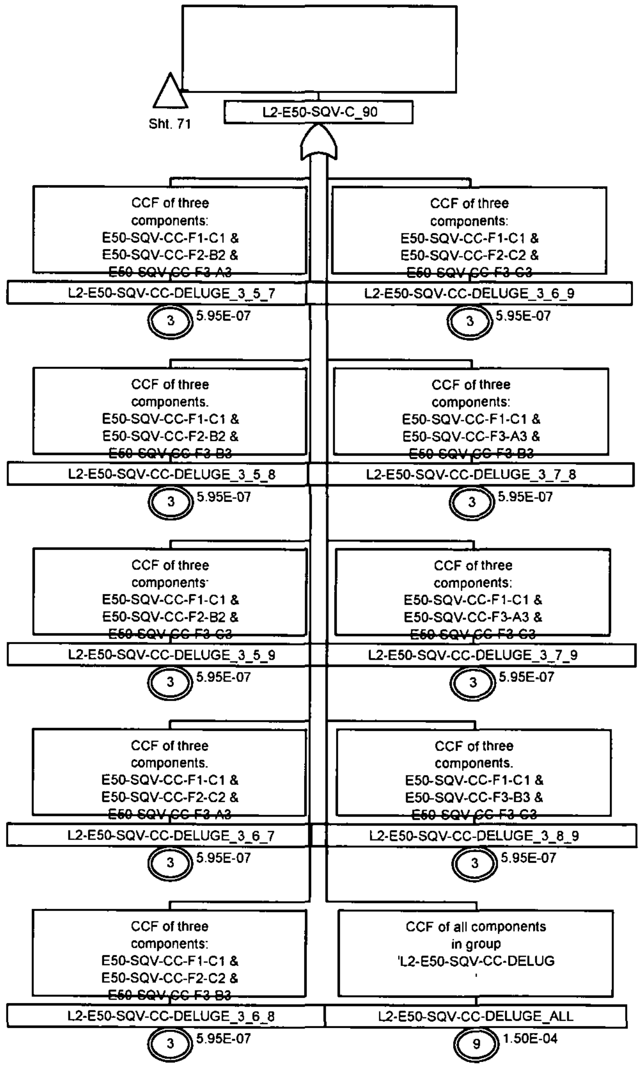

Figure 8A-8. Level 2 Fault Trees

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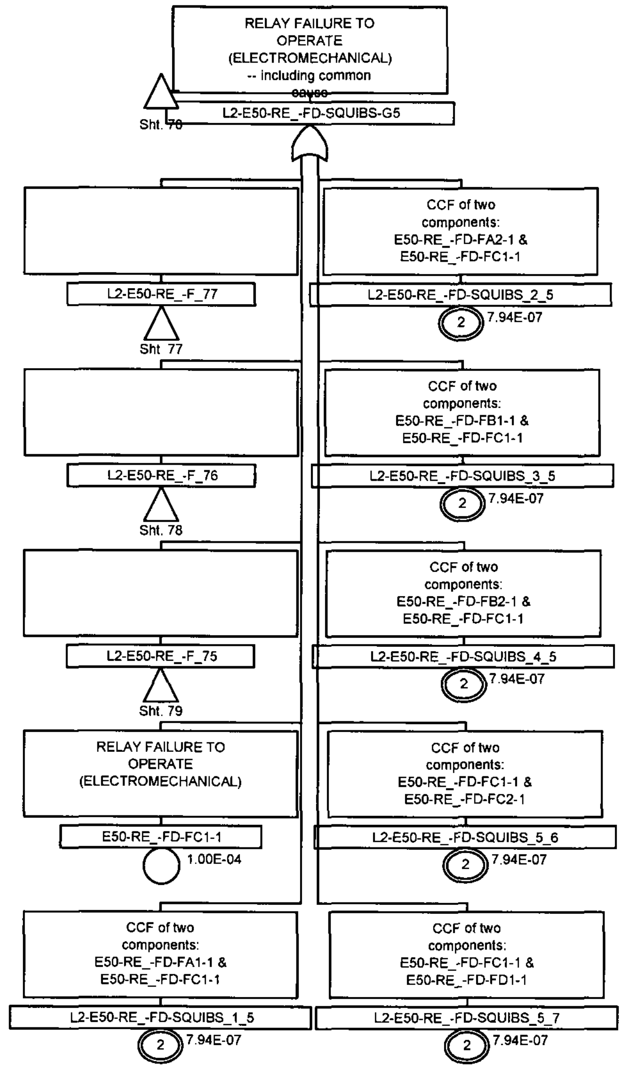

Figure 8A-8. Level 2 Fault Trees

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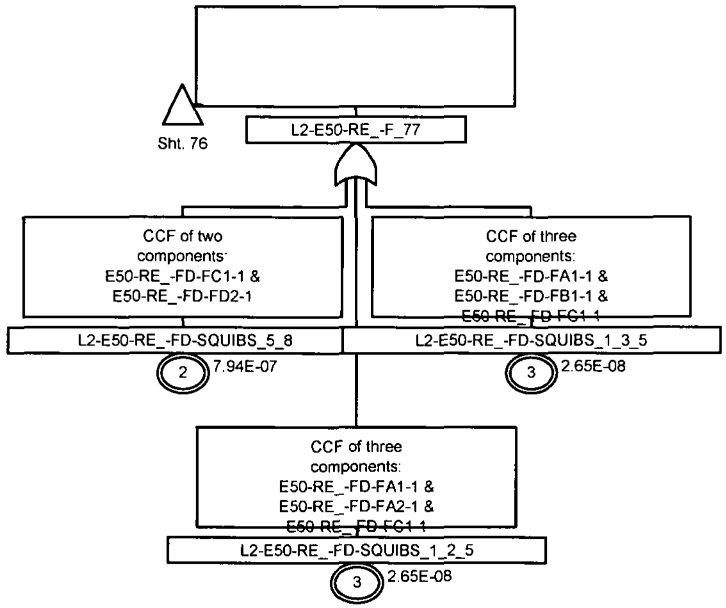

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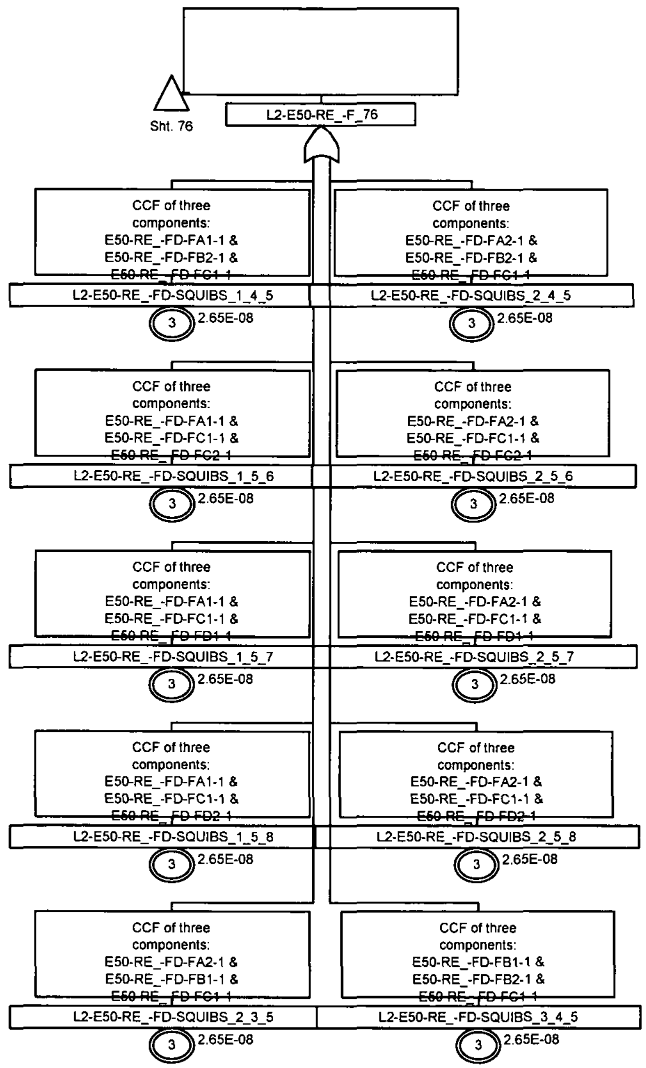

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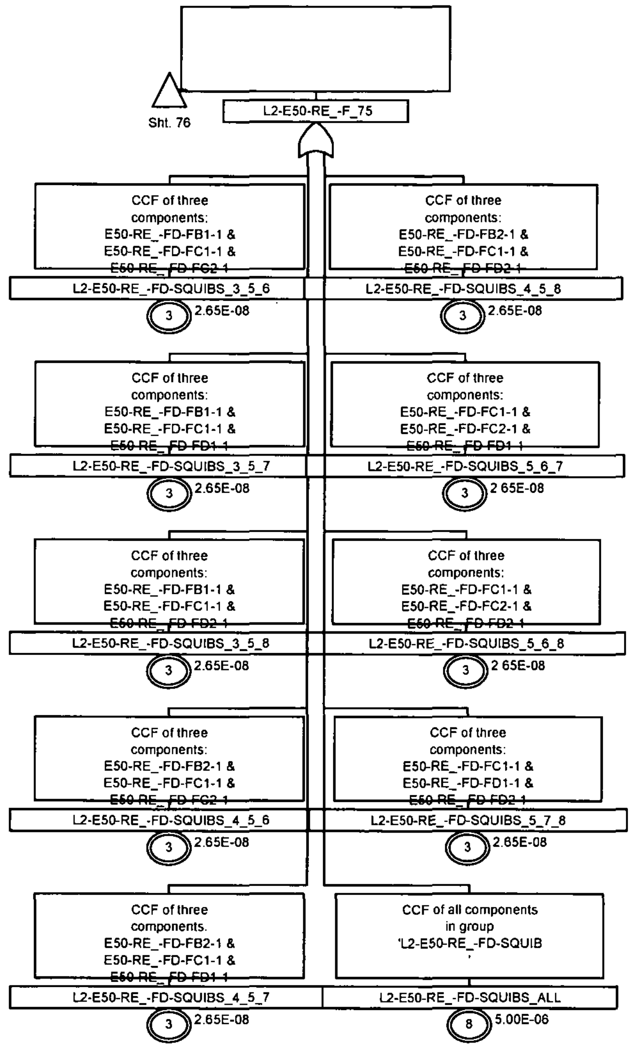

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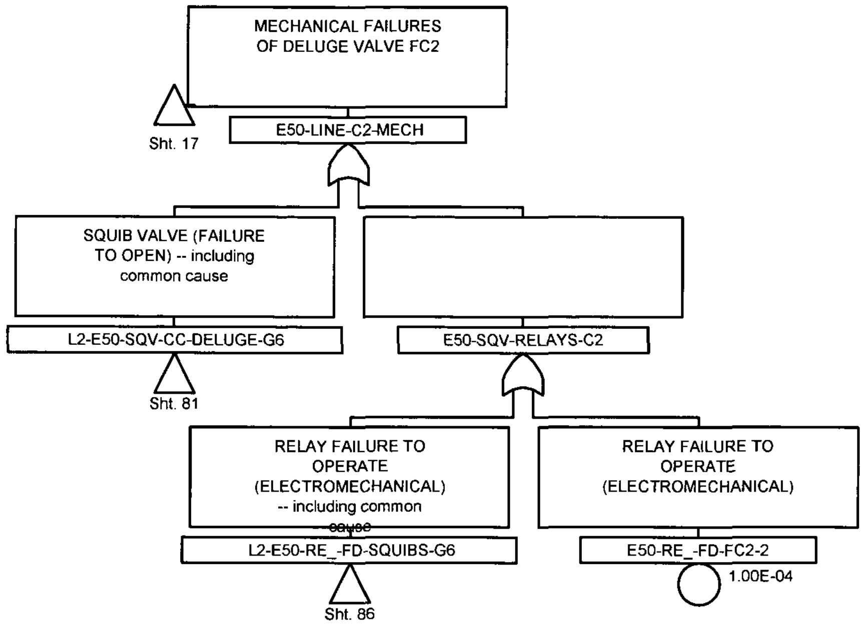

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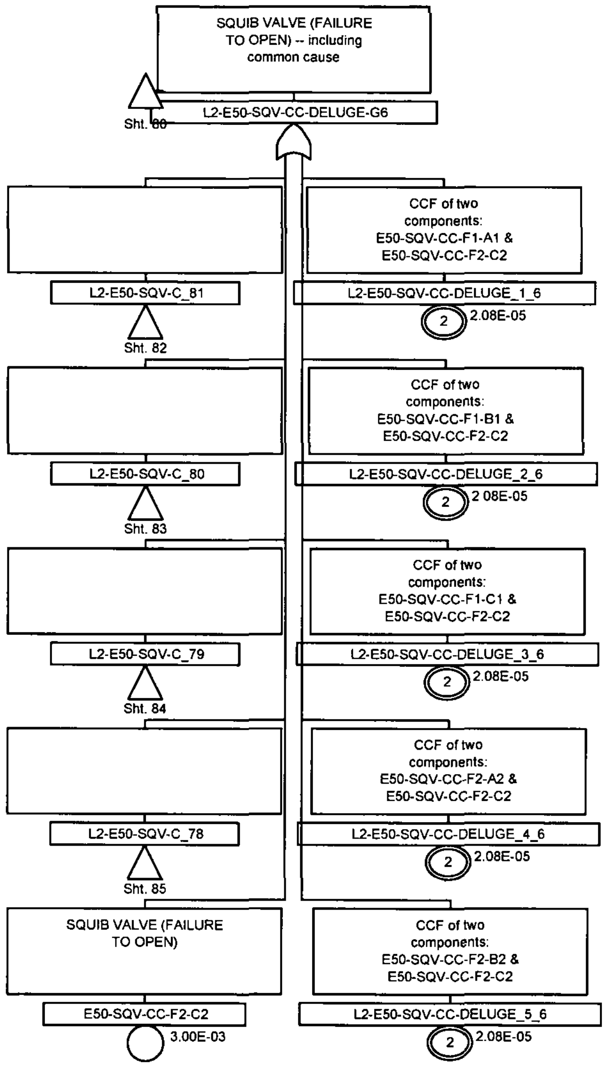

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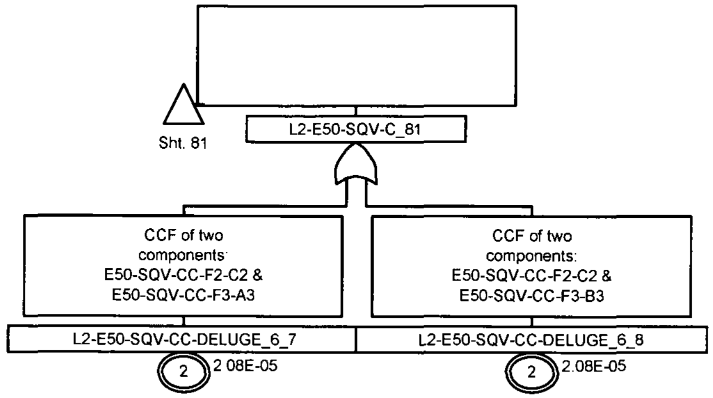

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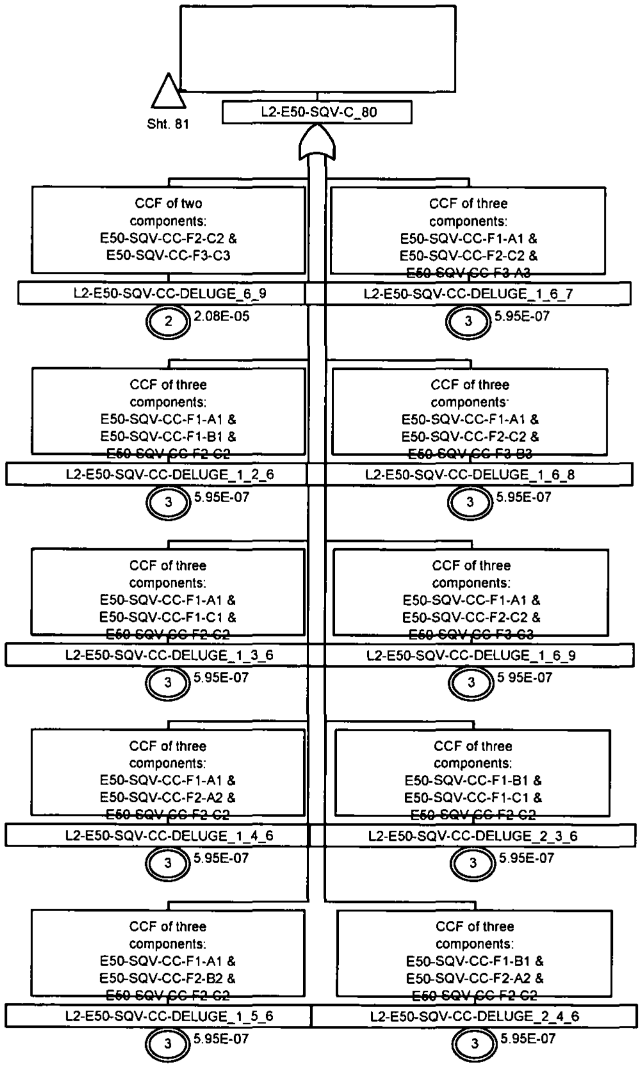

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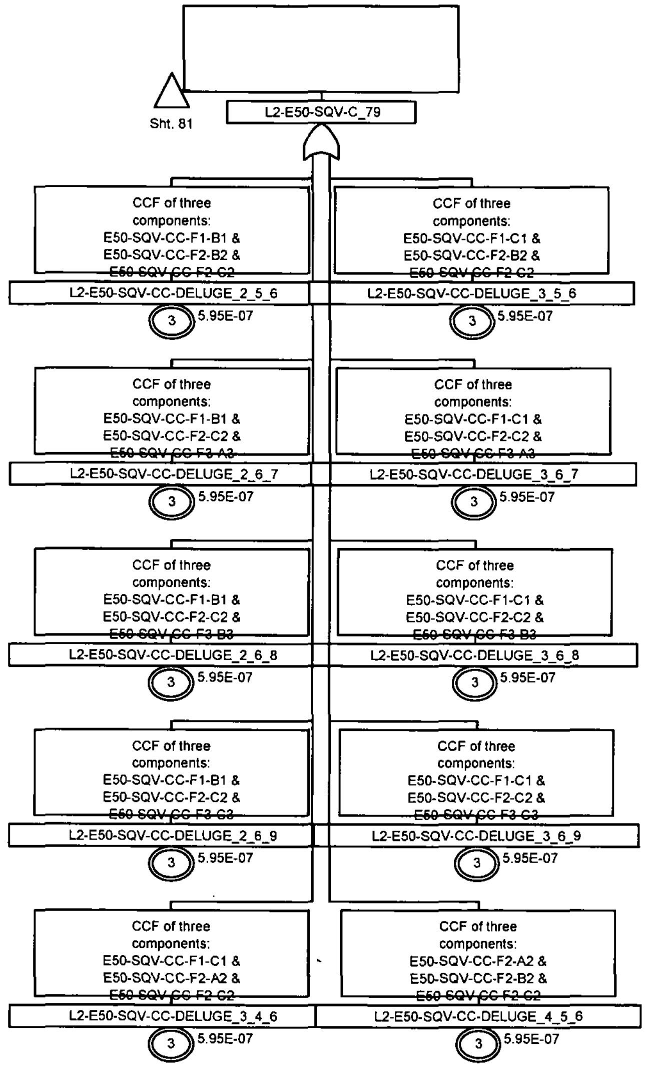

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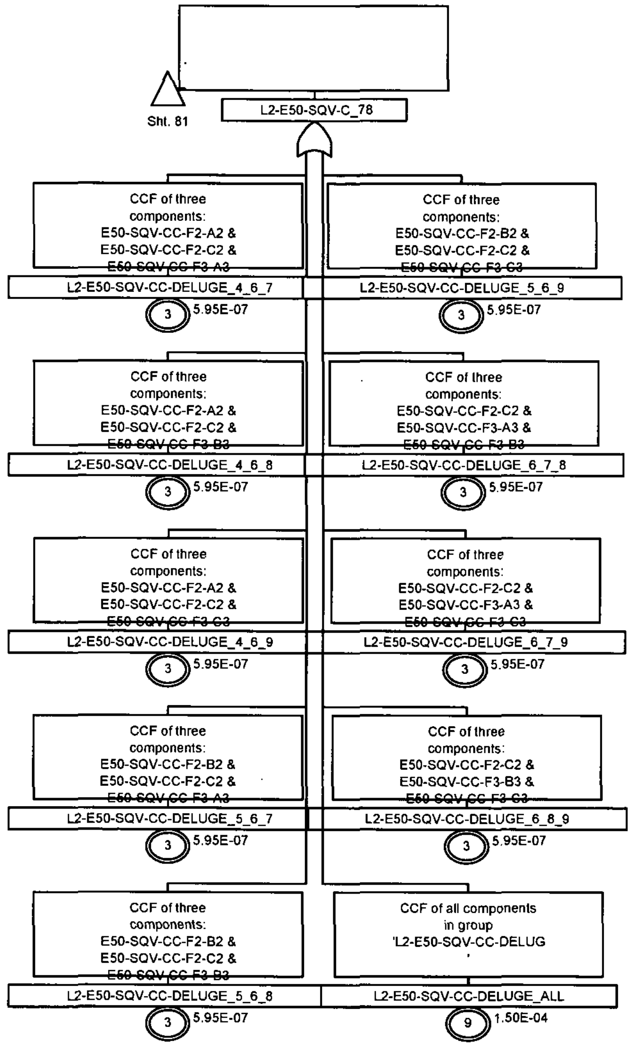

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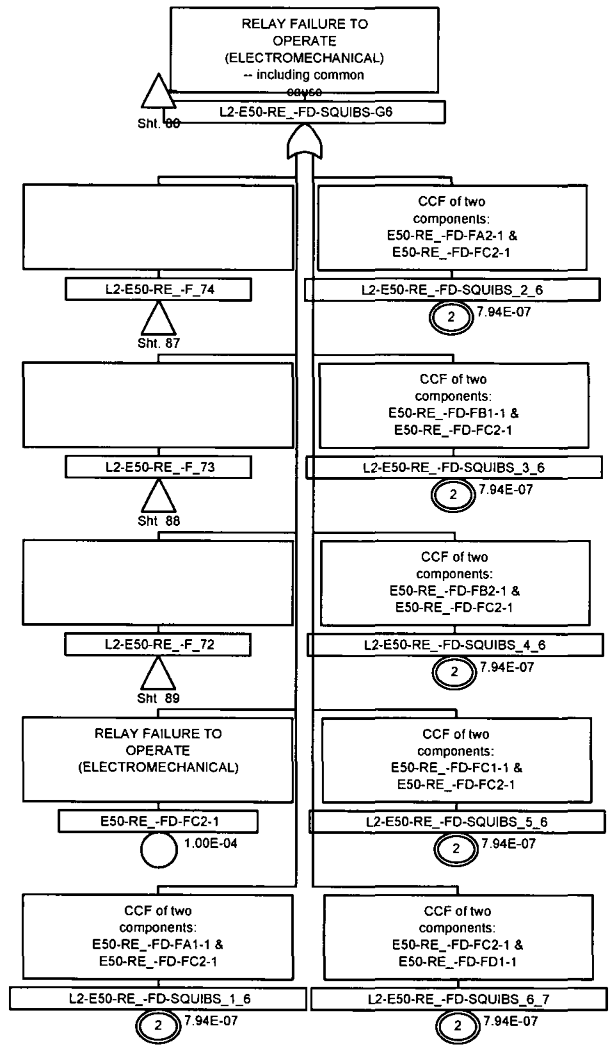

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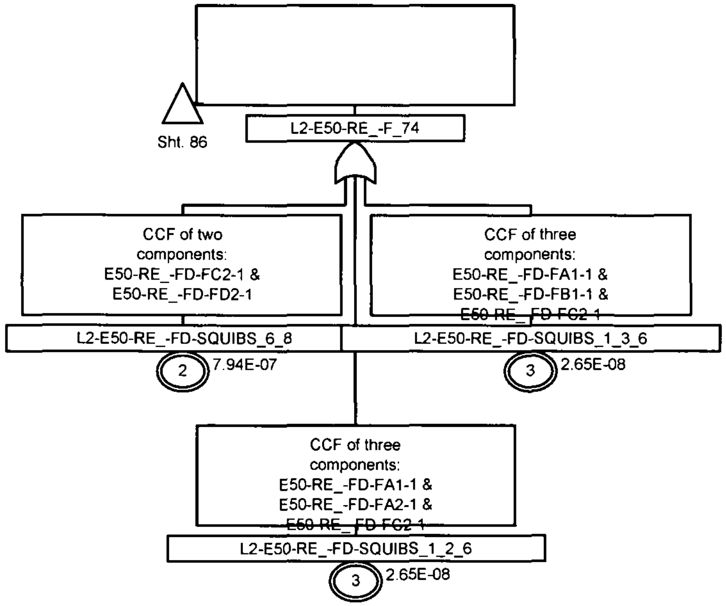

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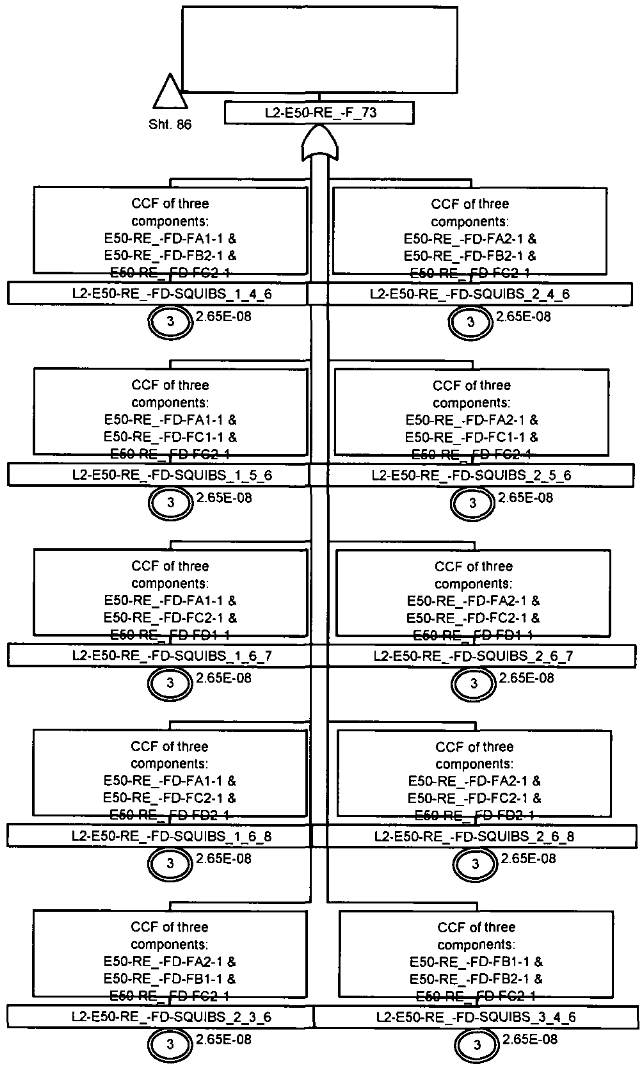

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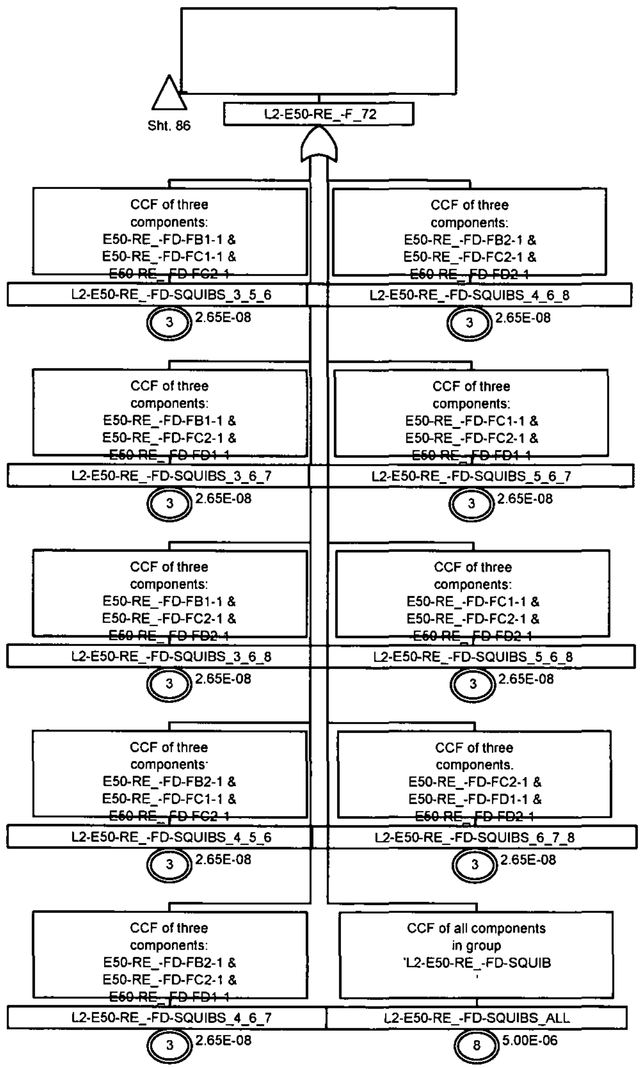

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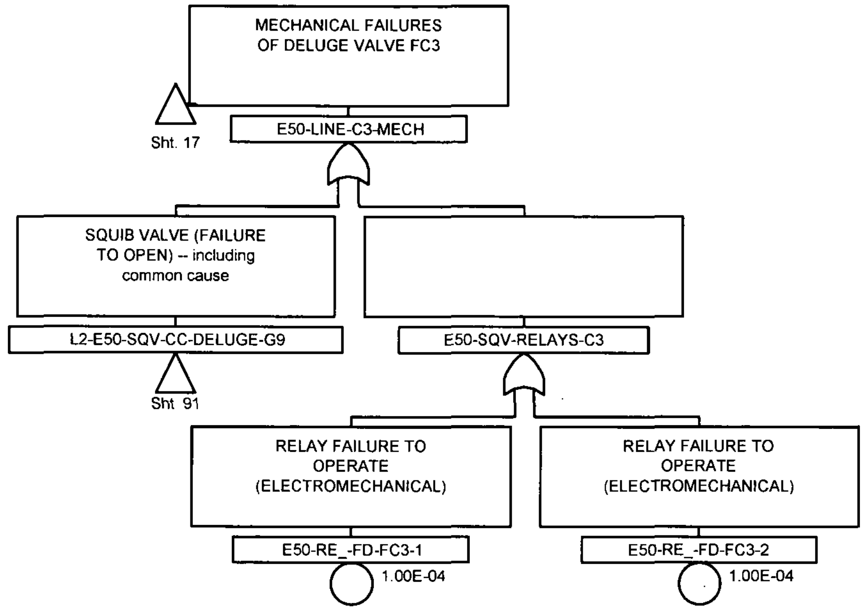

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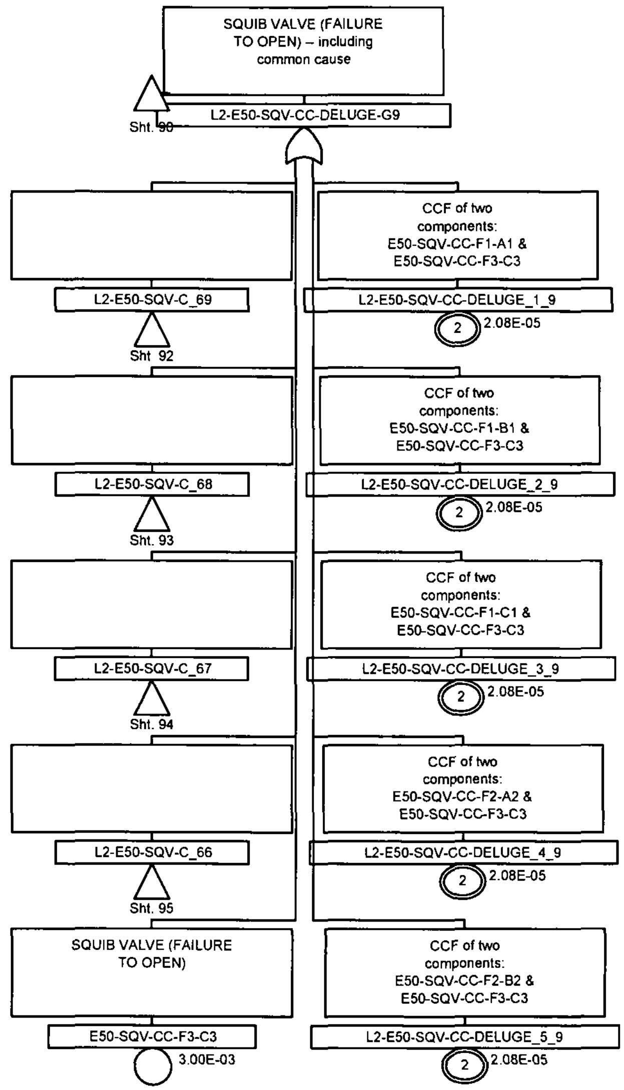

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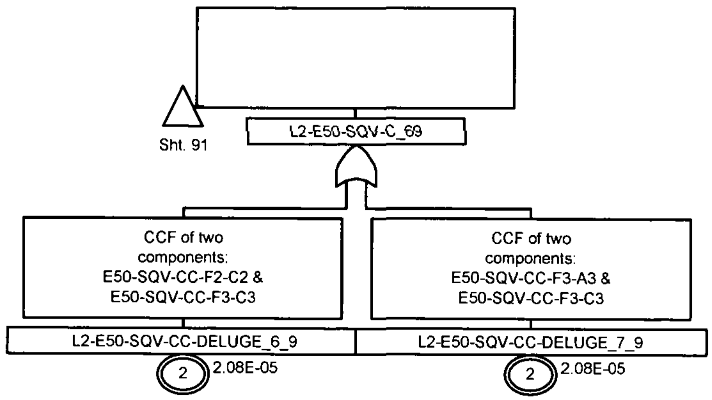

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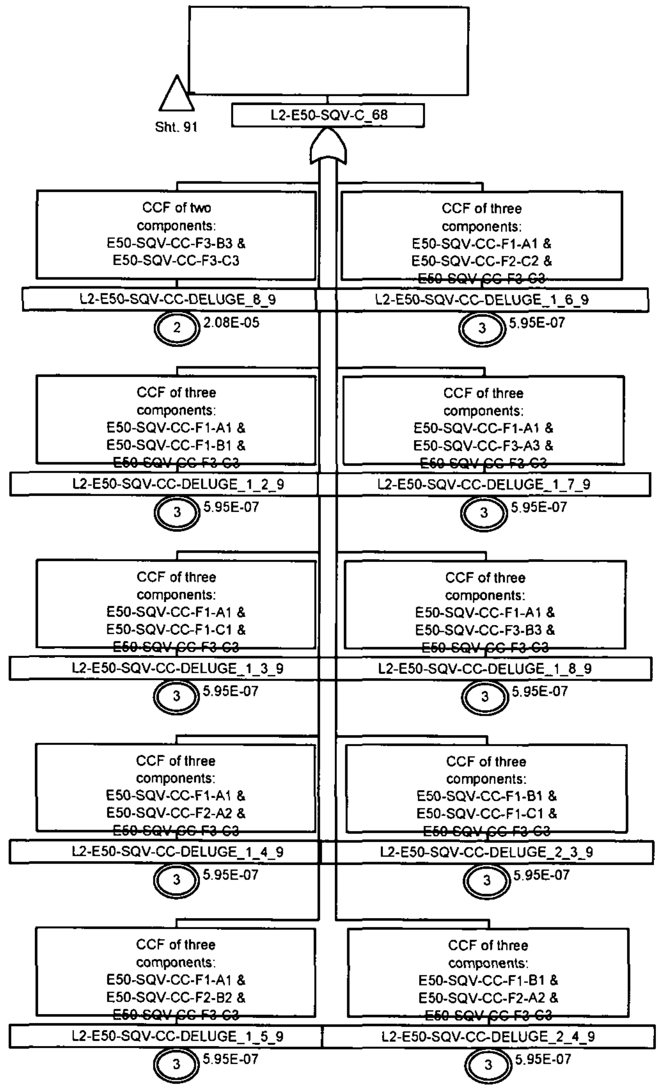

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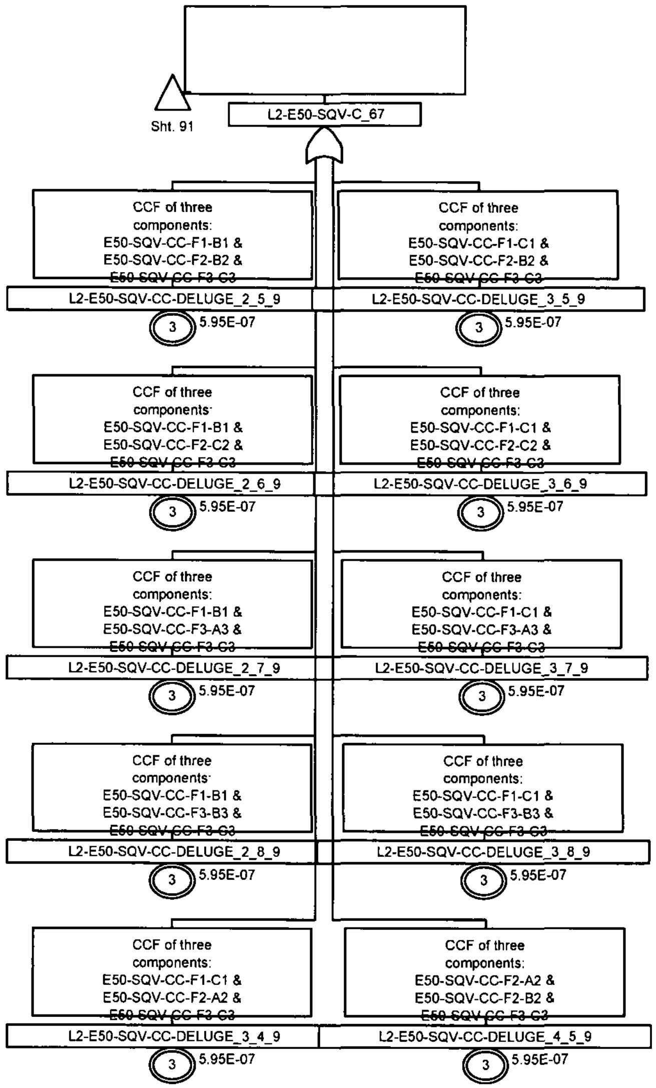

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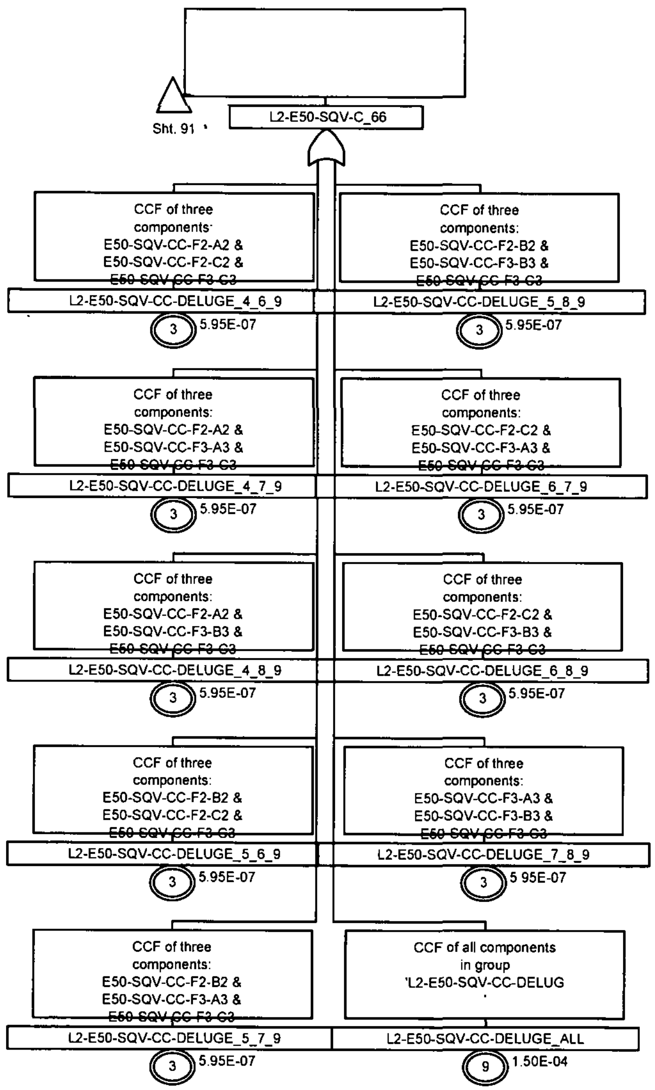

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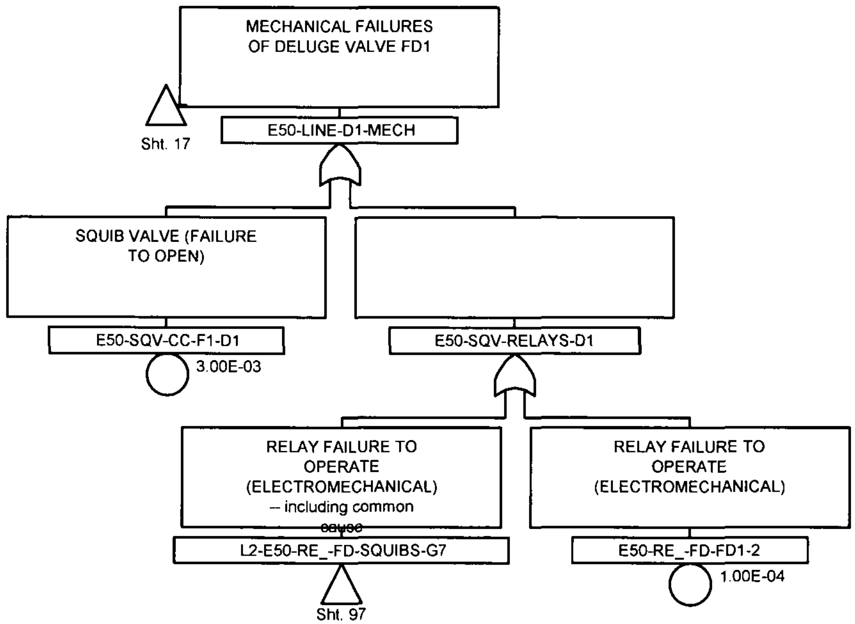

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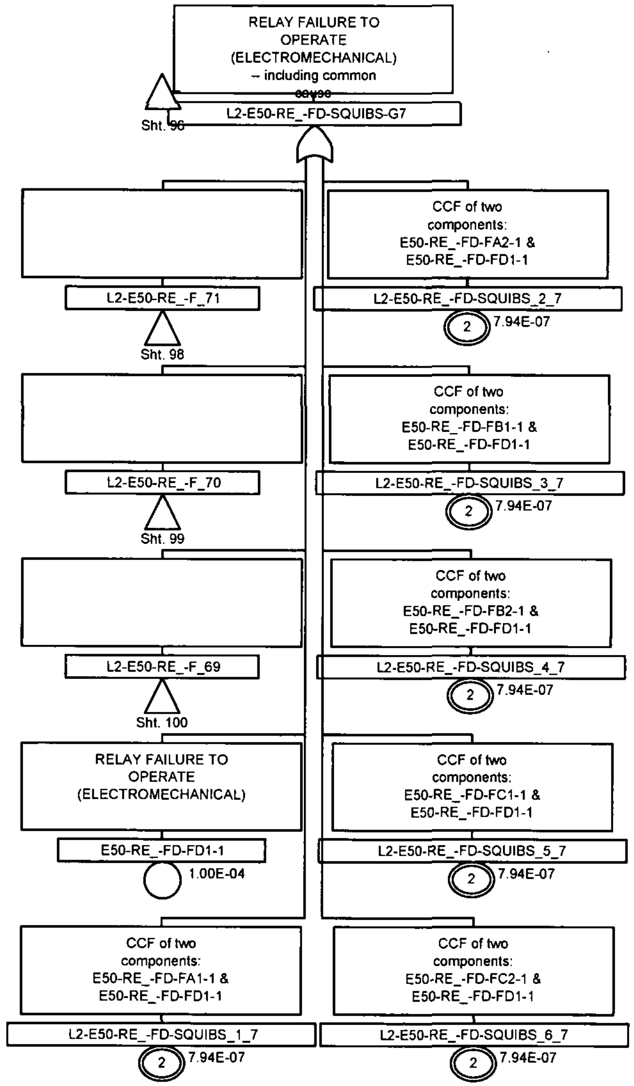

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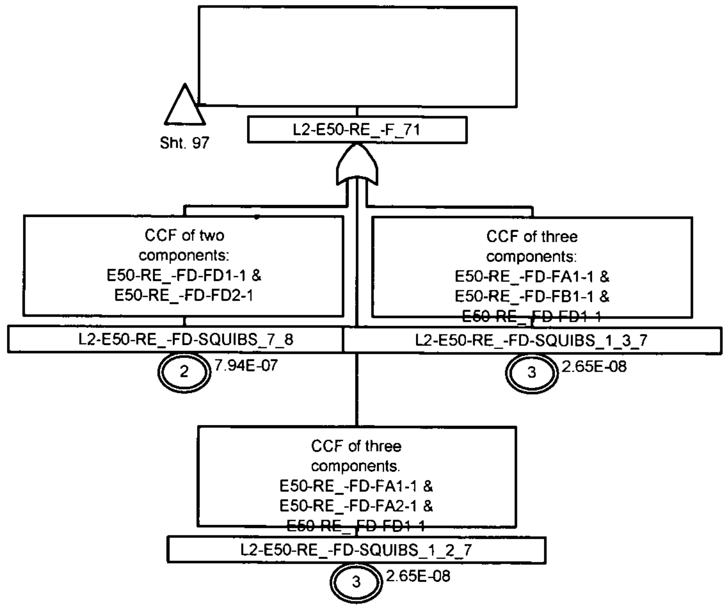

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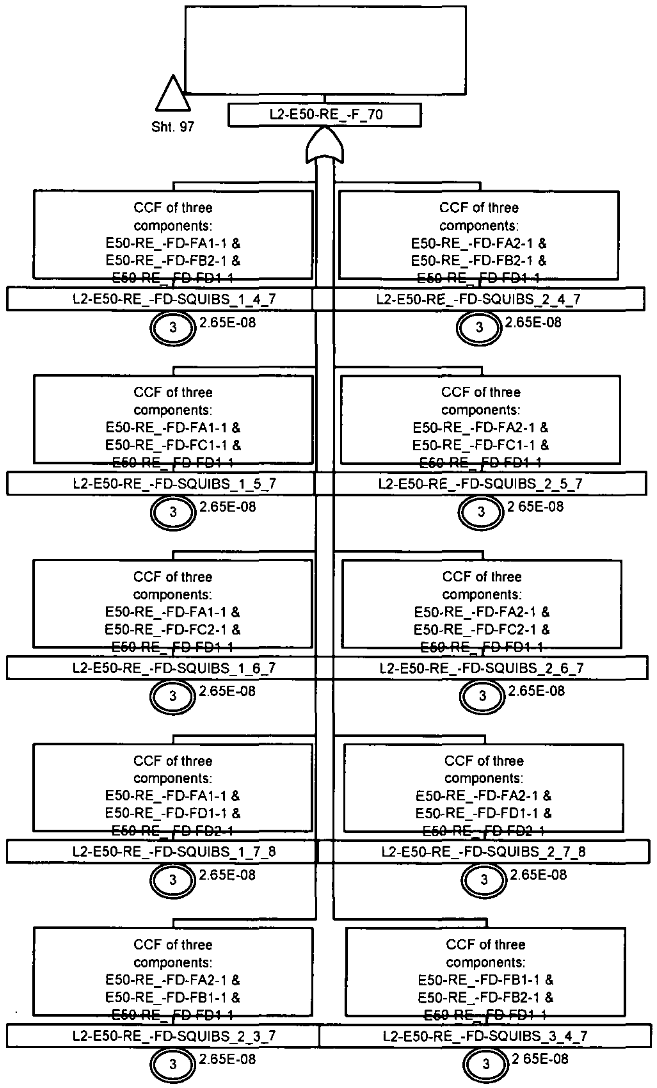

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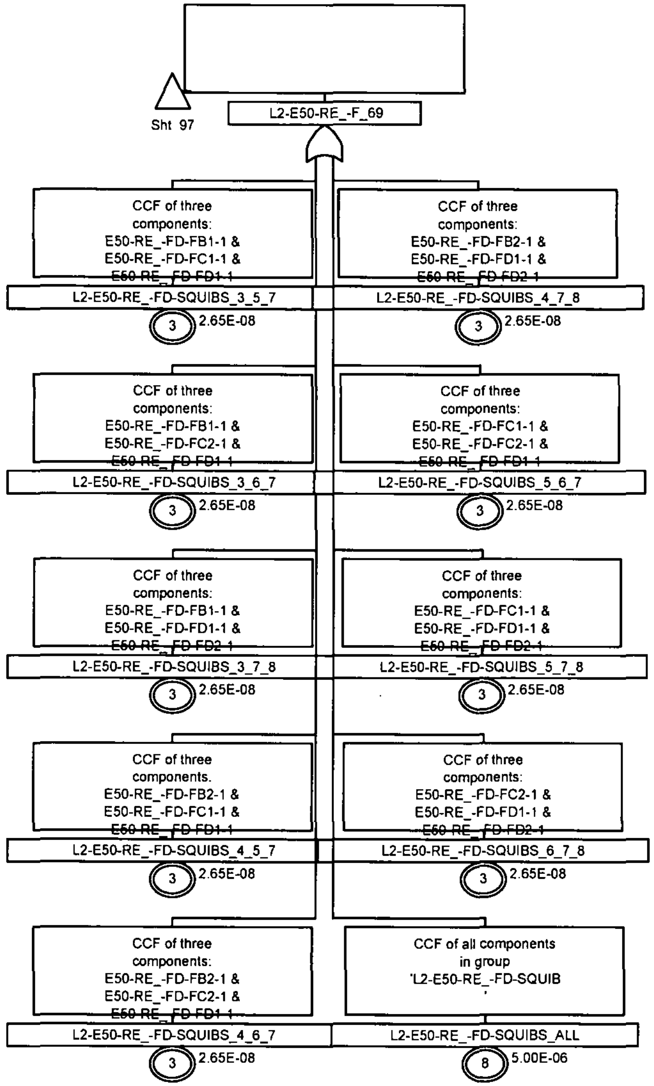

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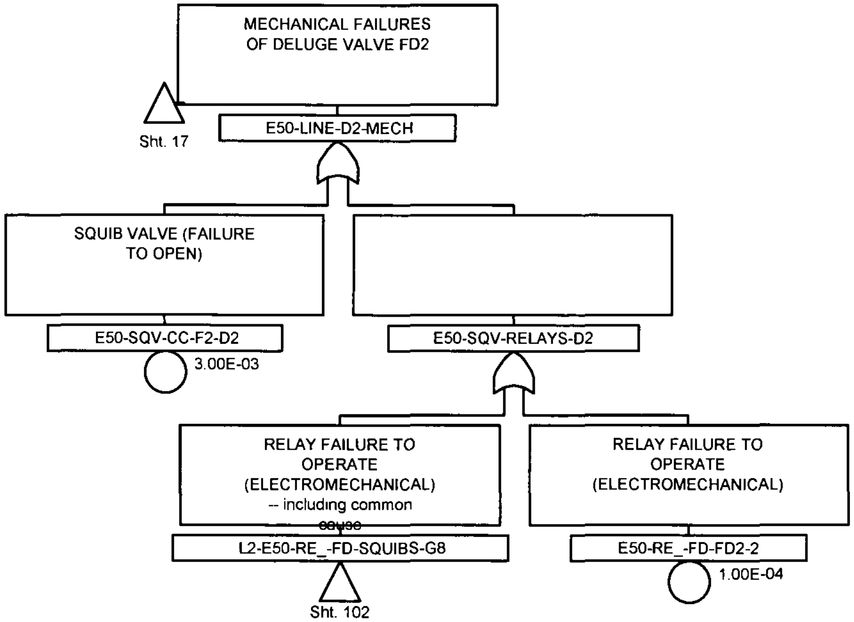

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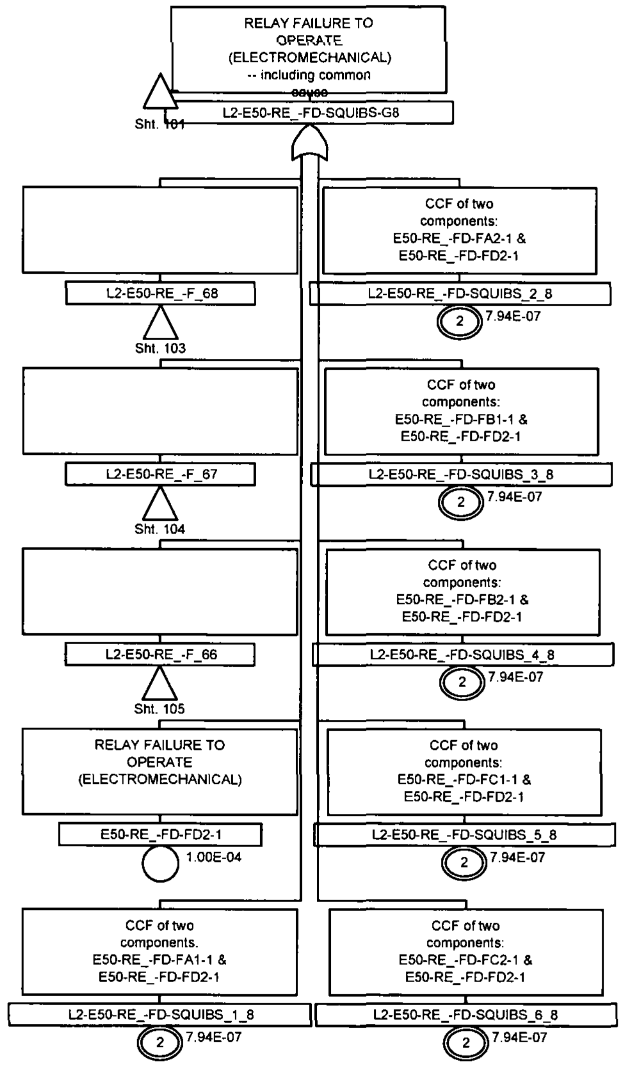

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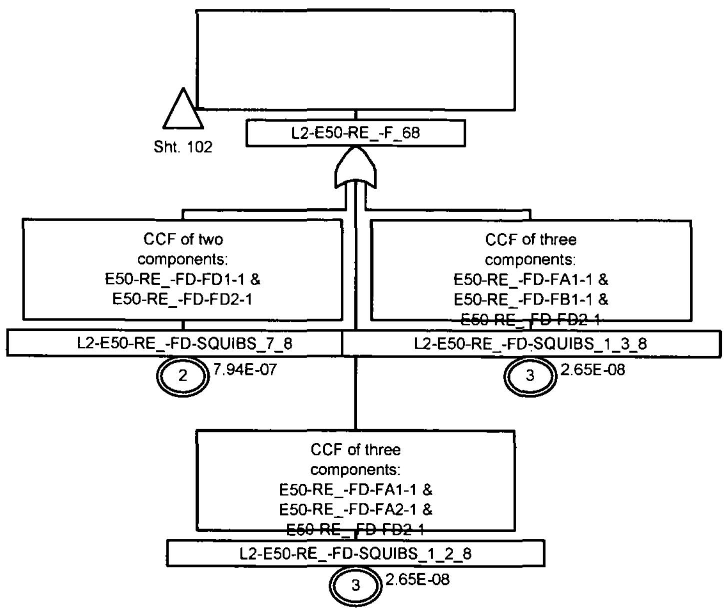

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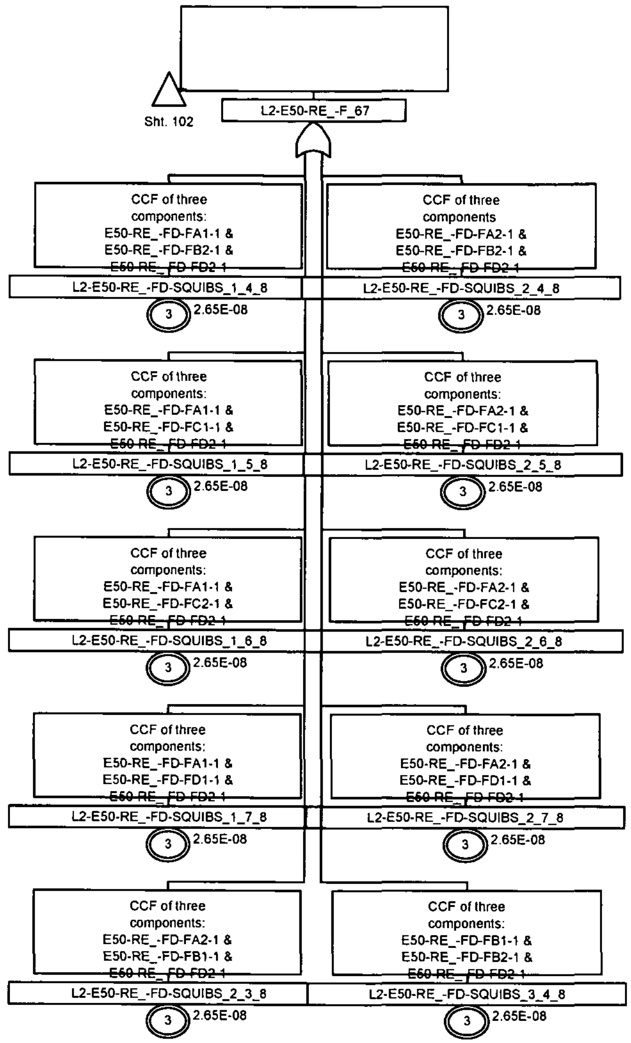

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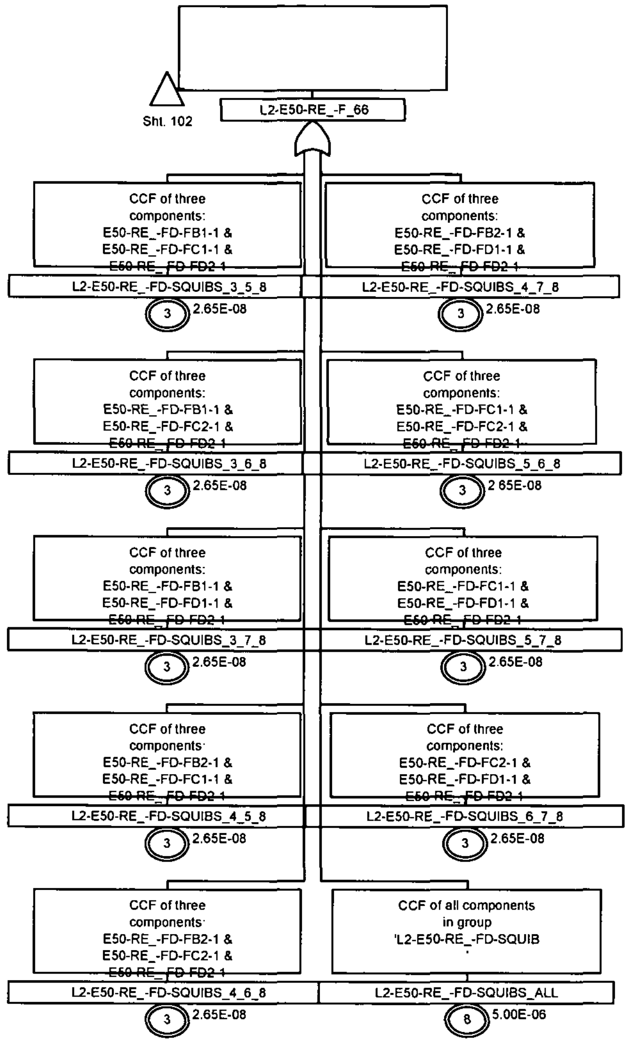

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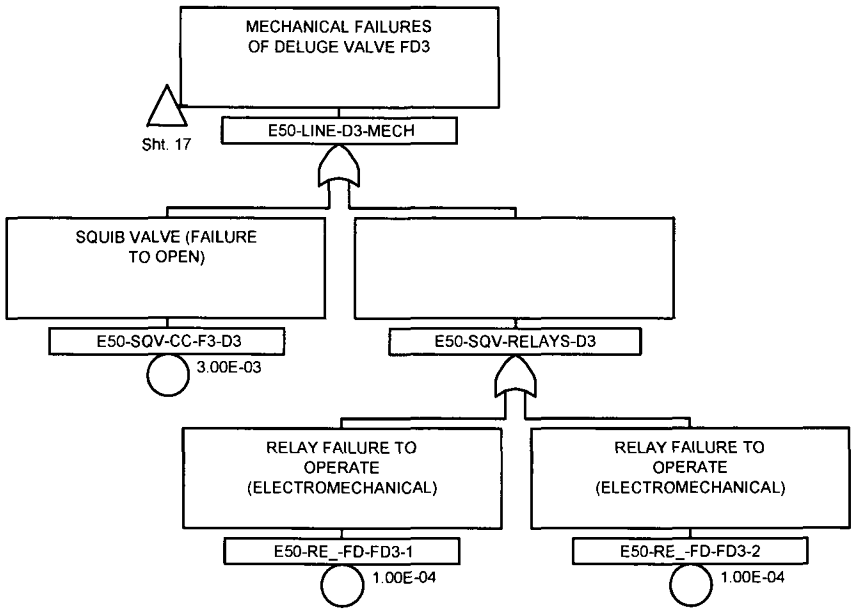

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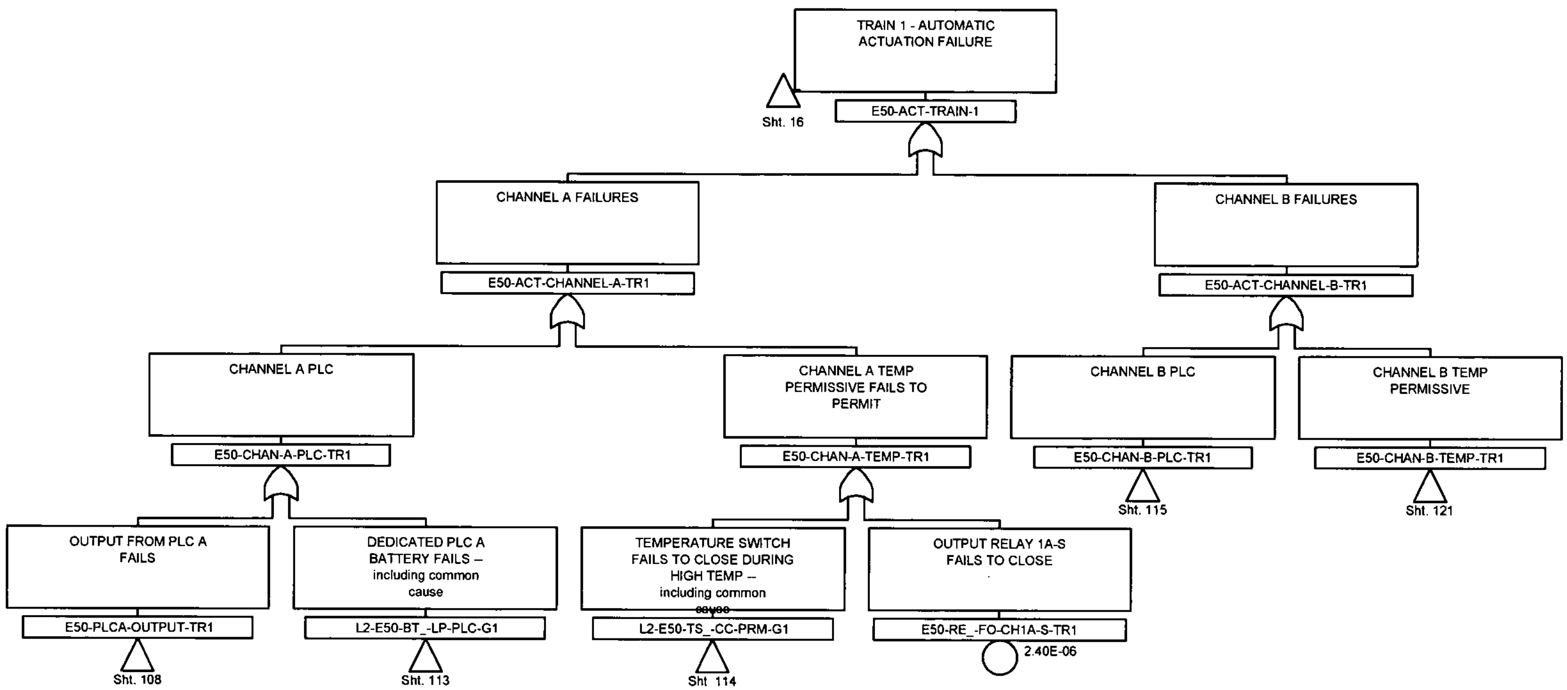

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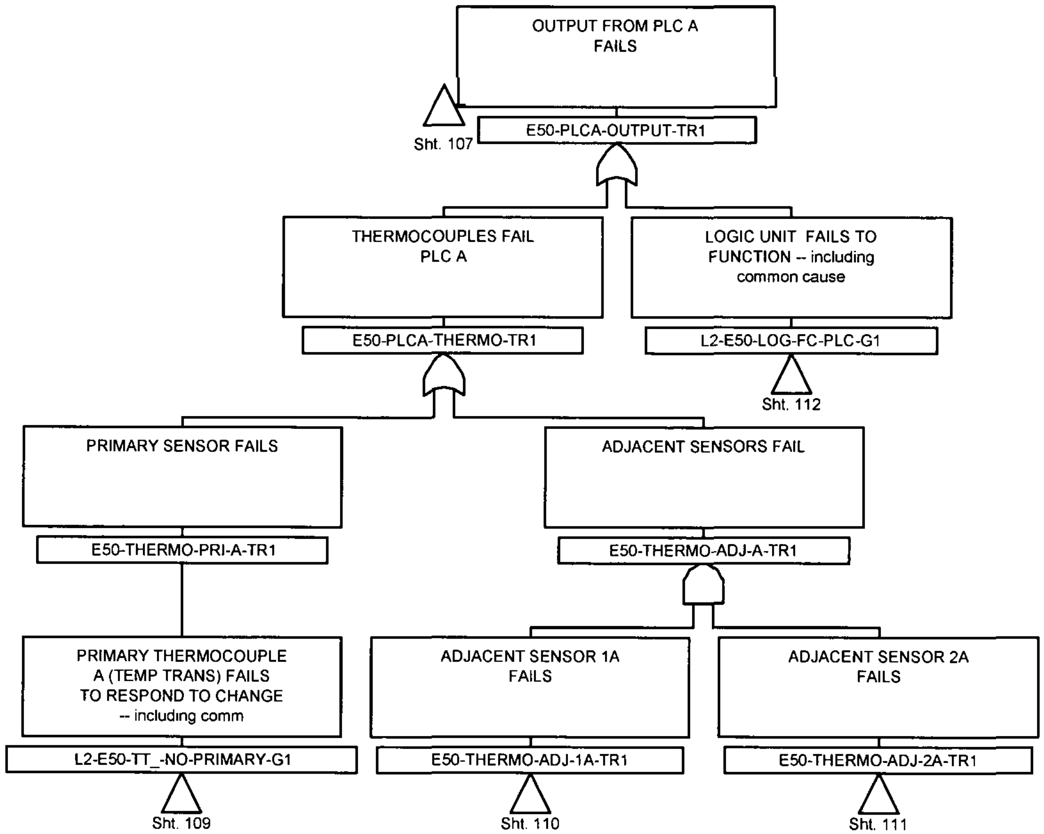

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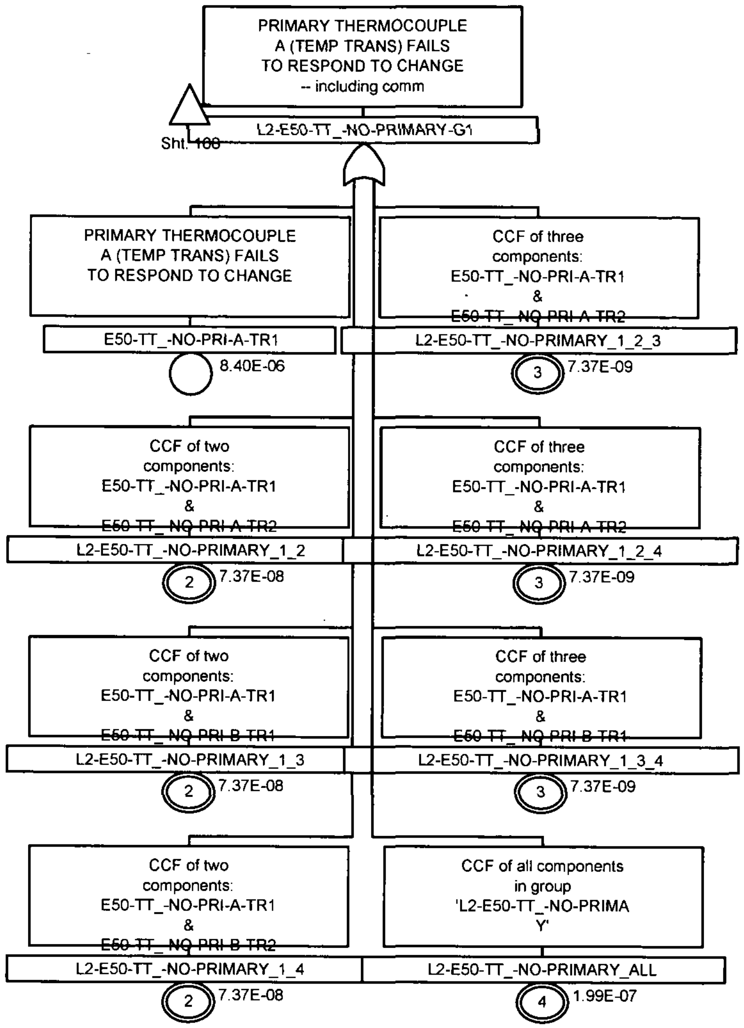

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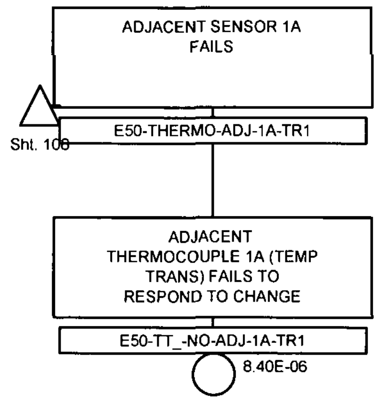

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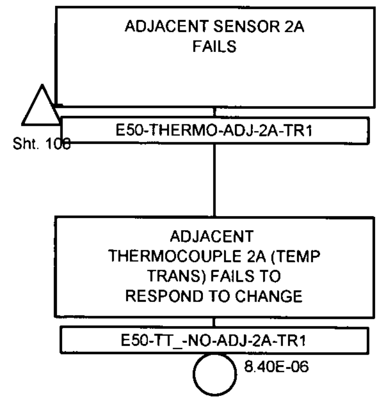

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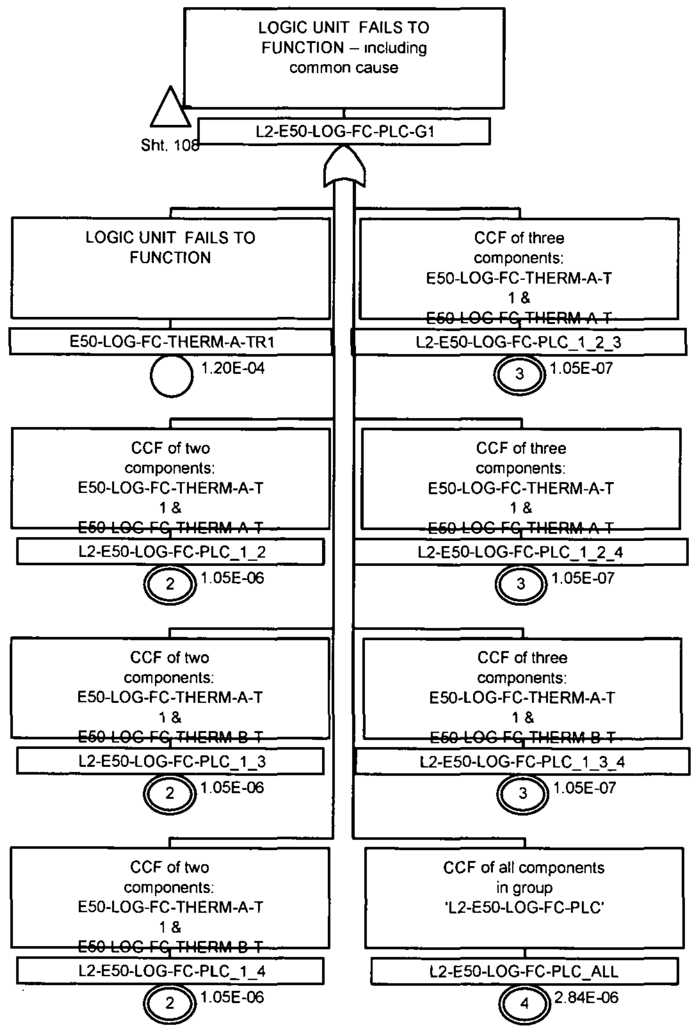

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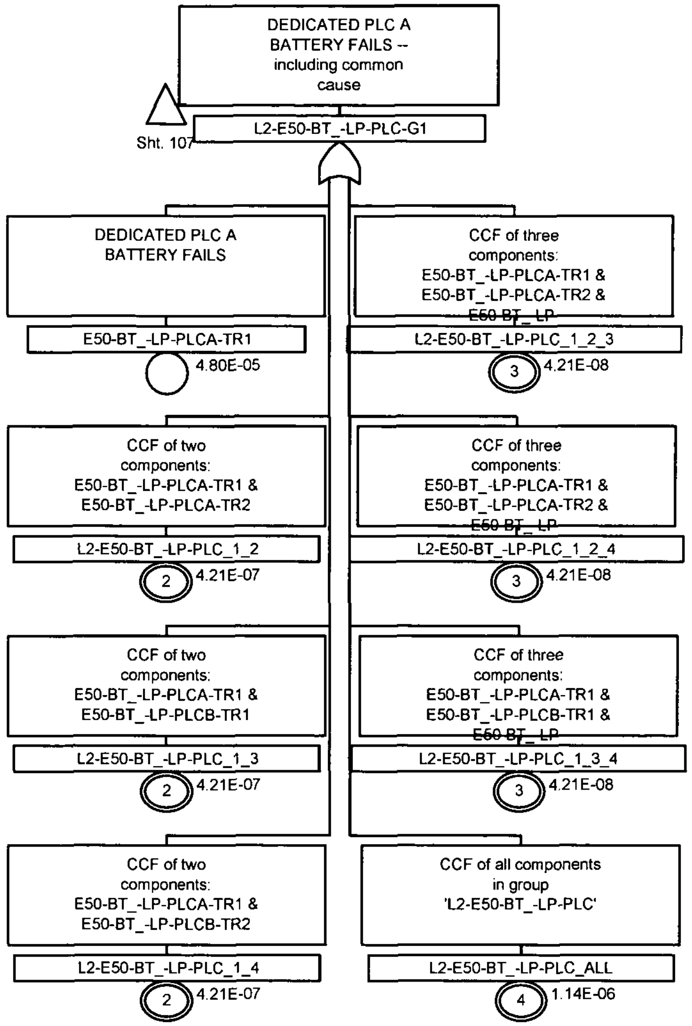

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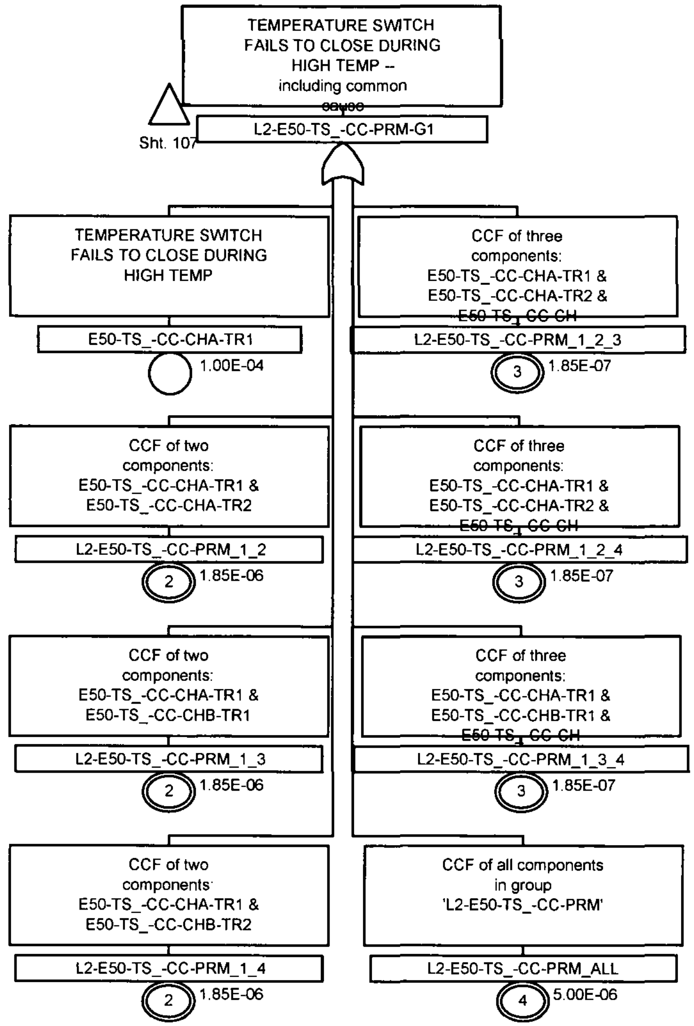

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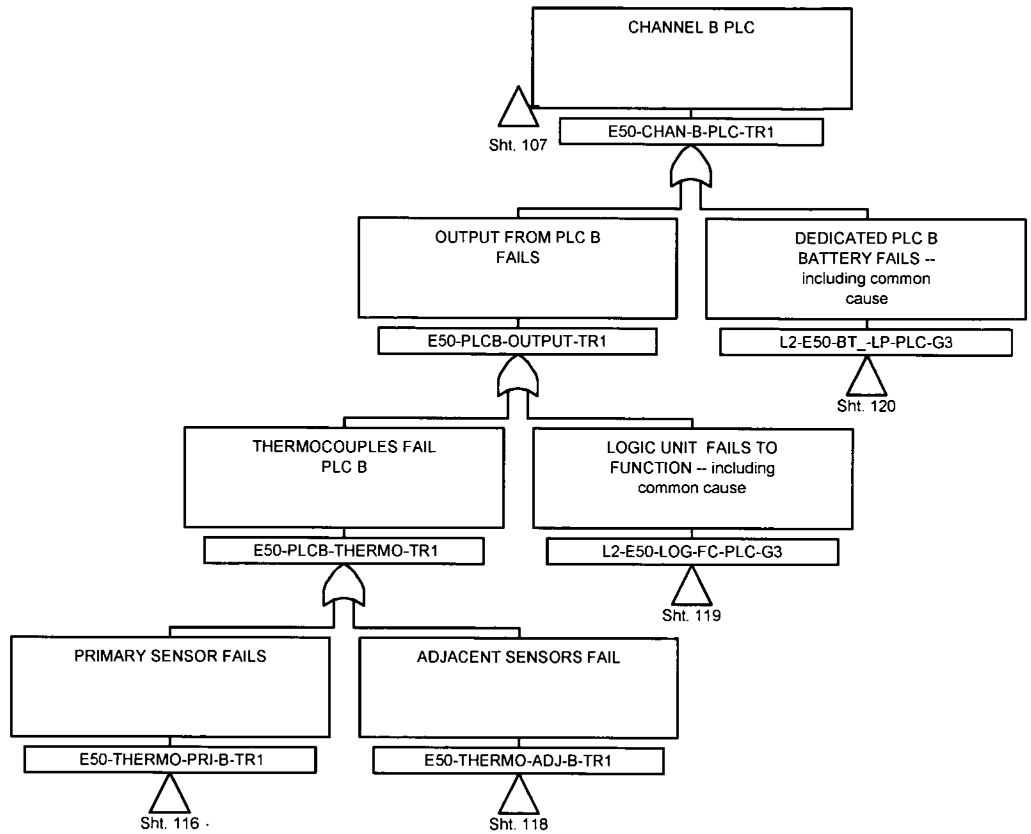

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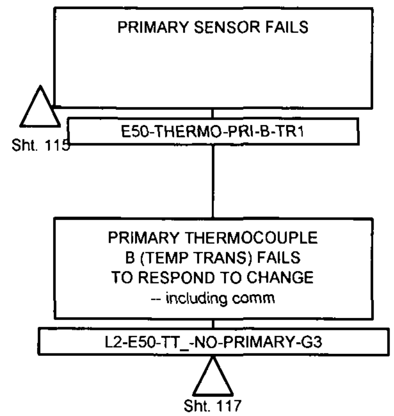

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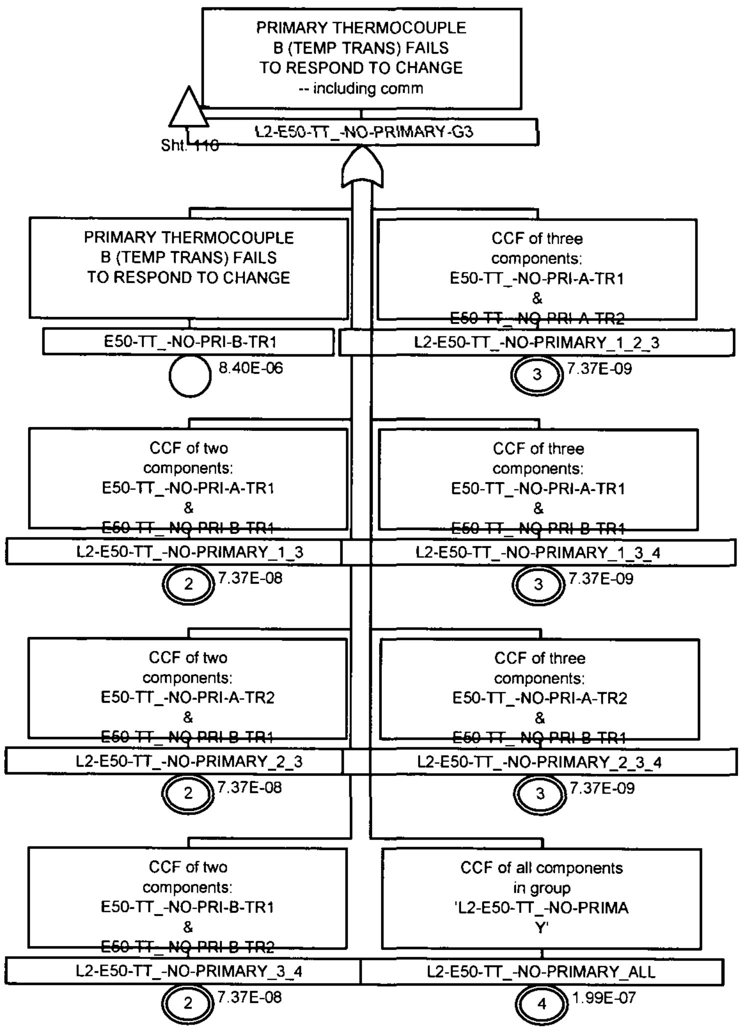

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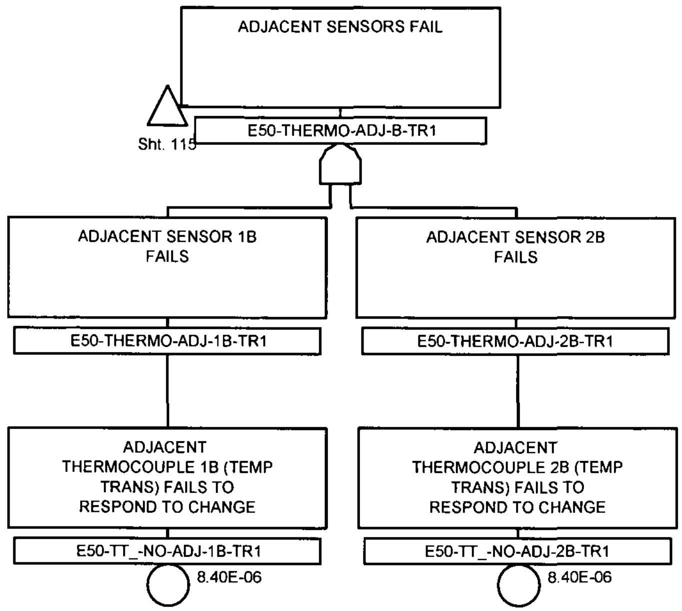

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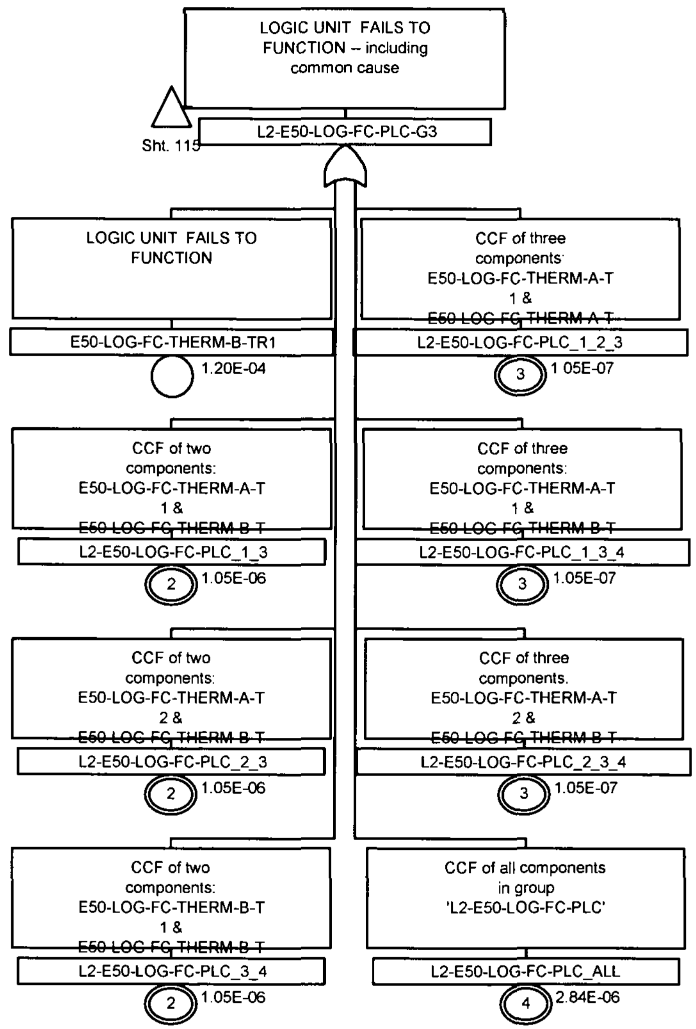

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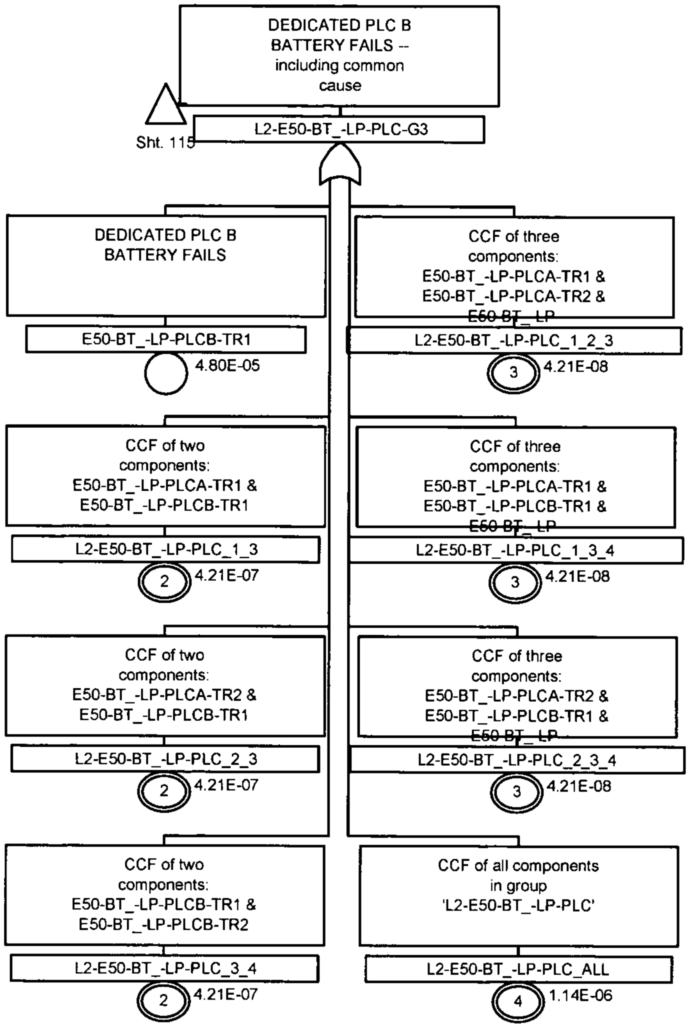

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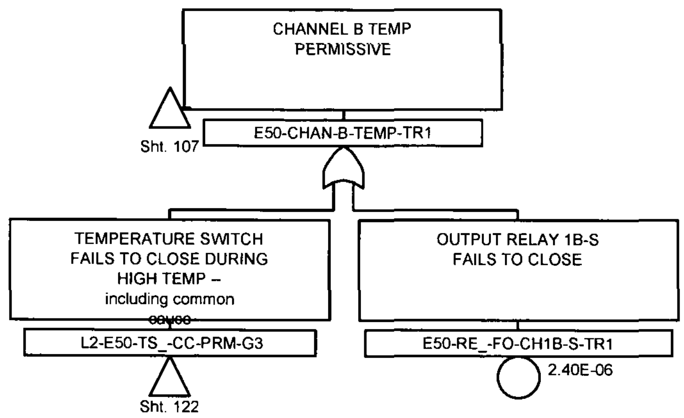

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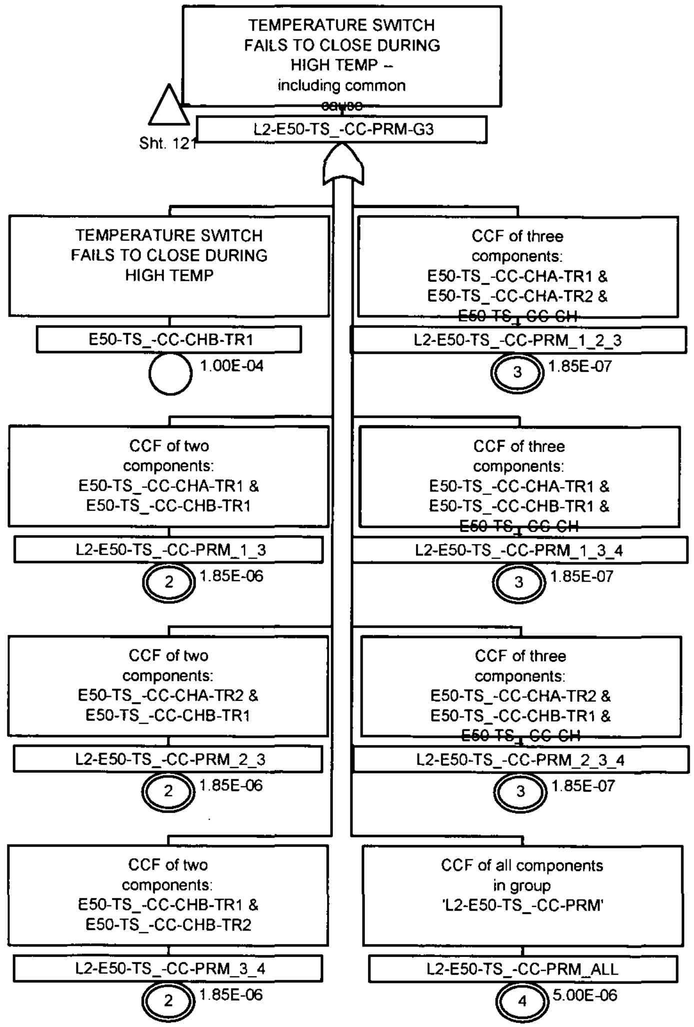

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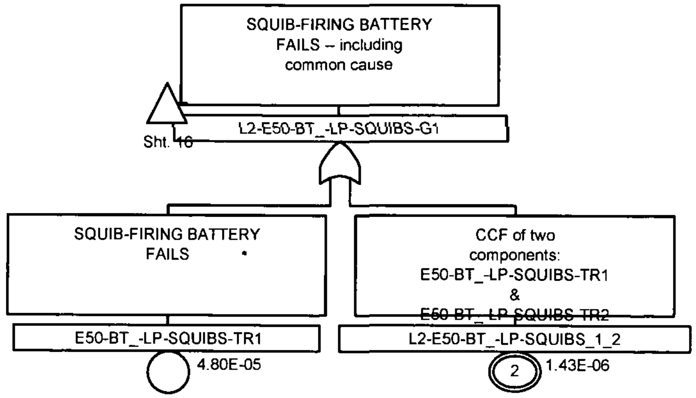

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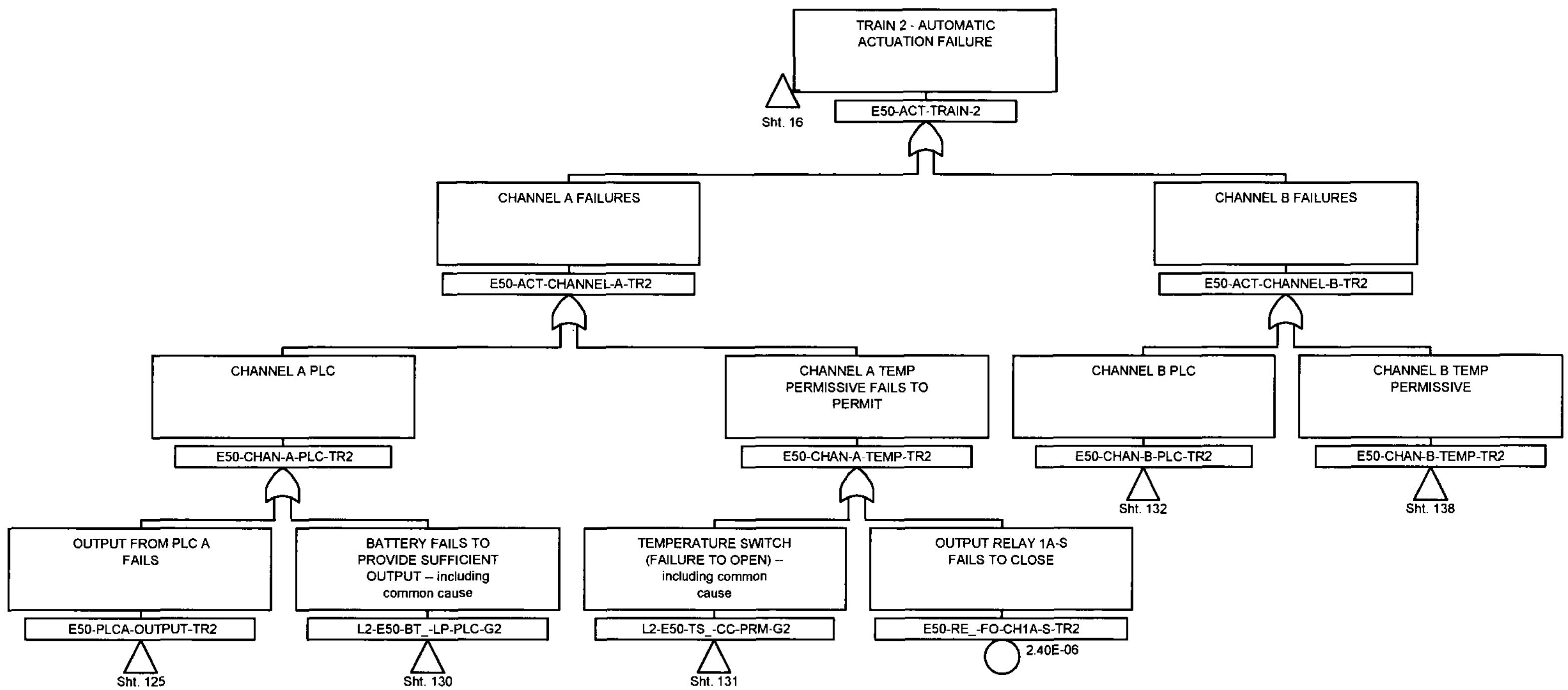

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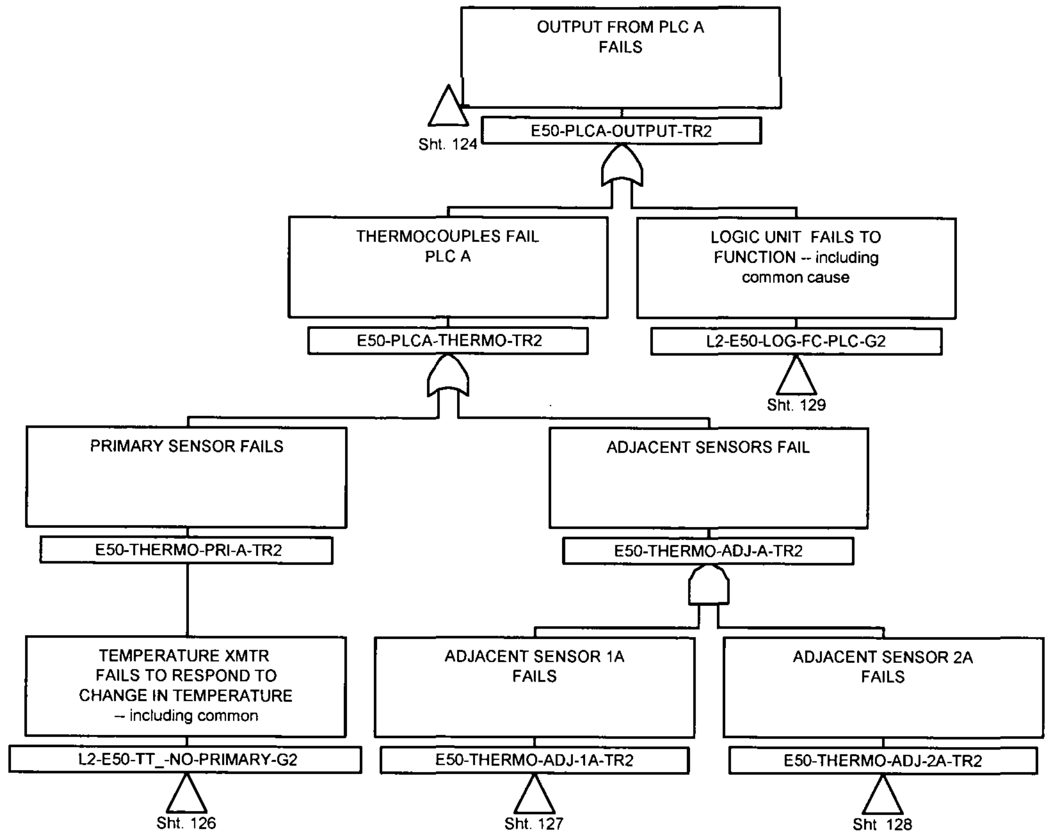

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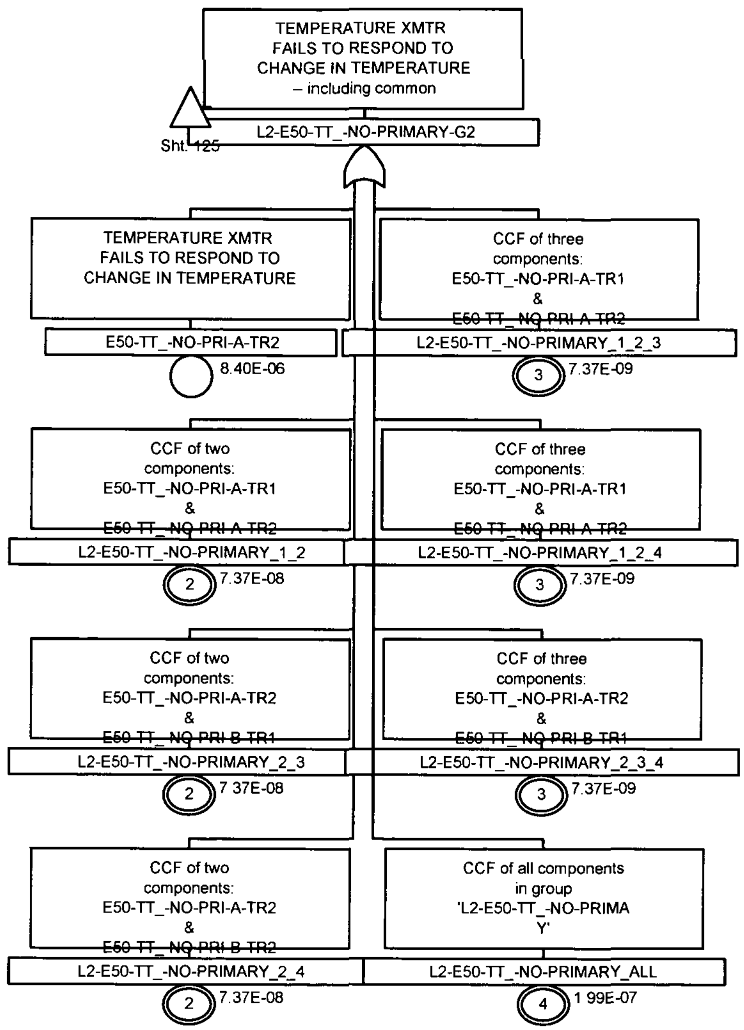

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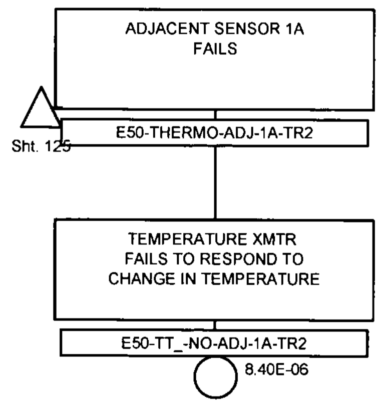

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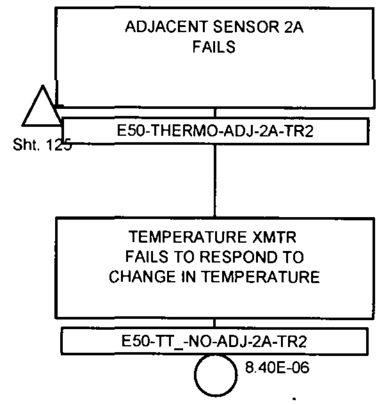

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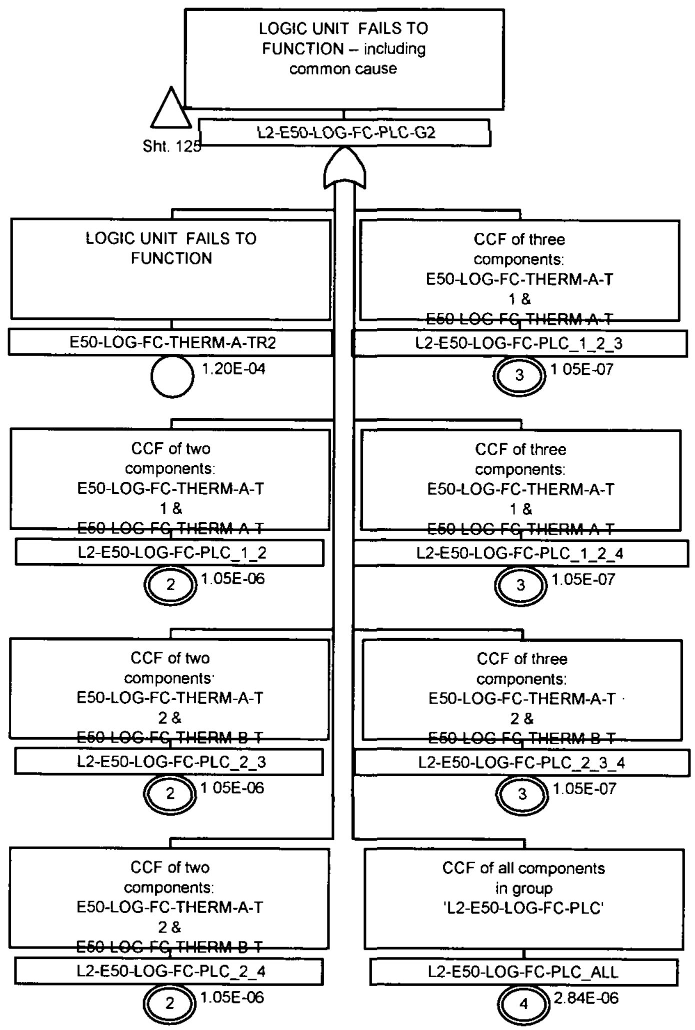

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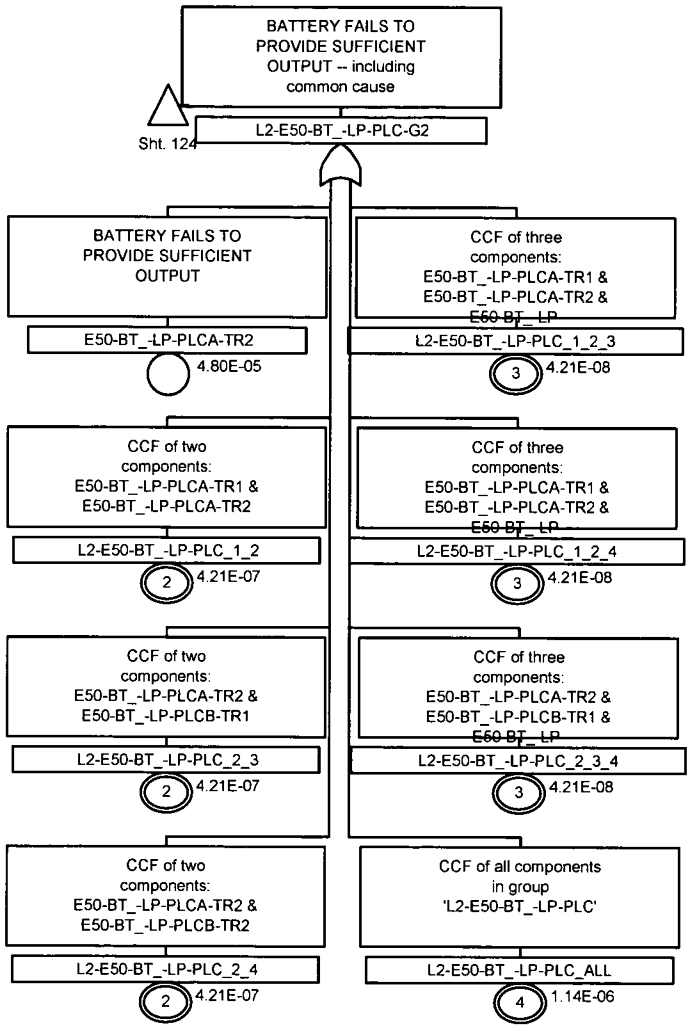

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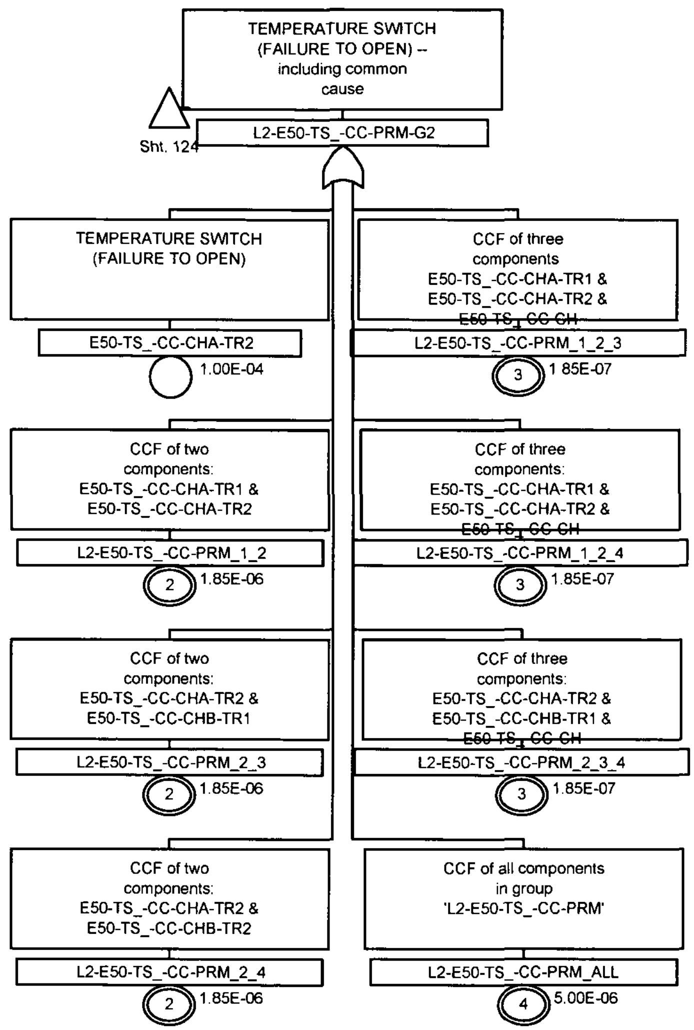

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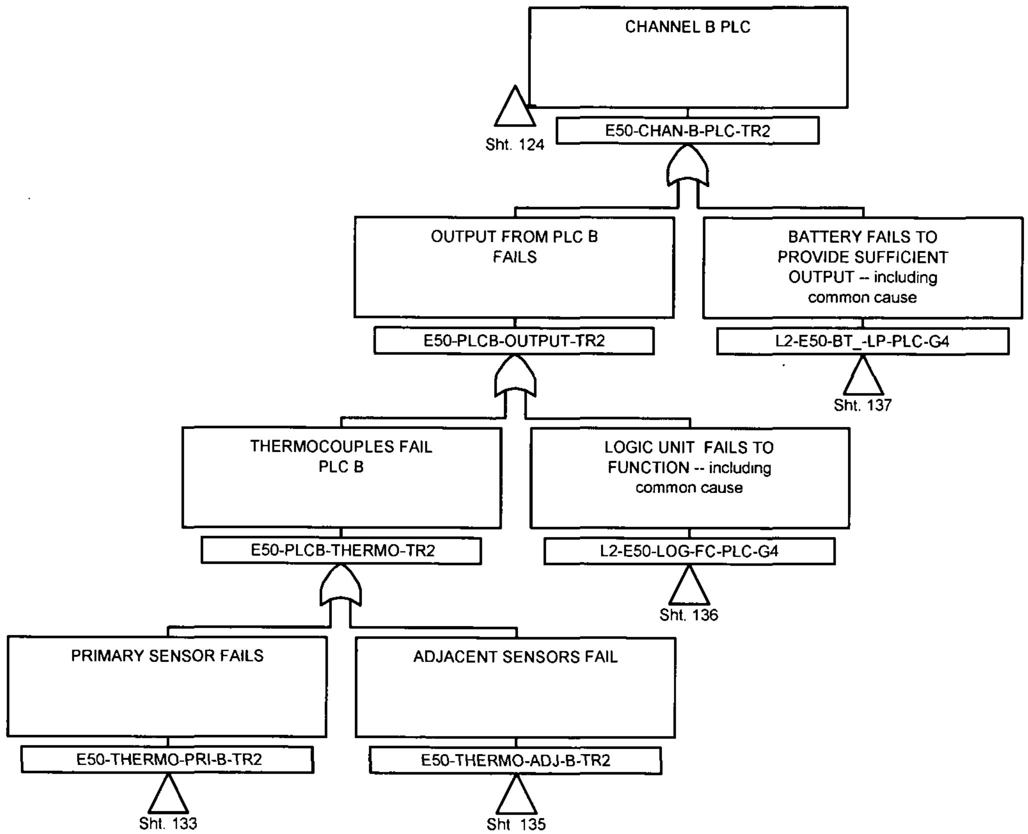

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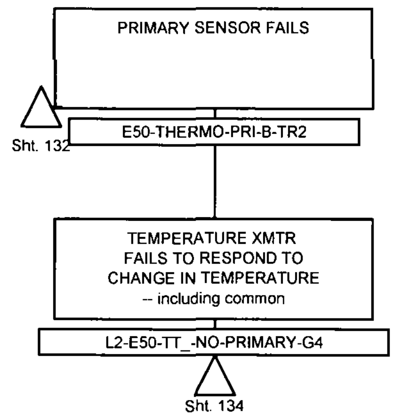

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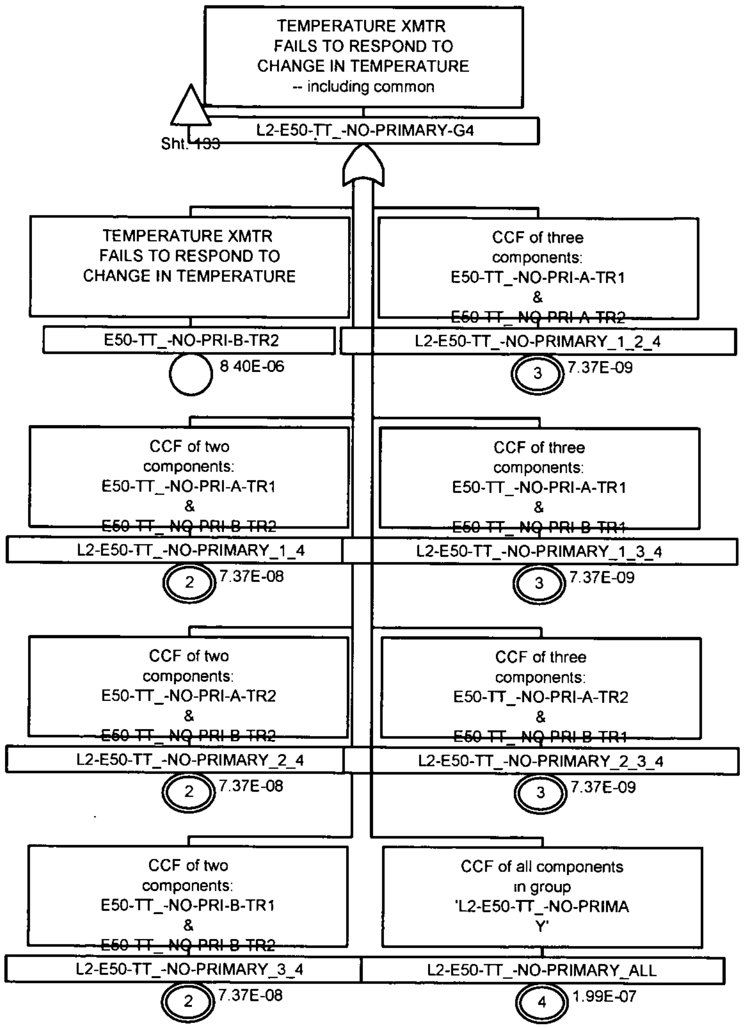

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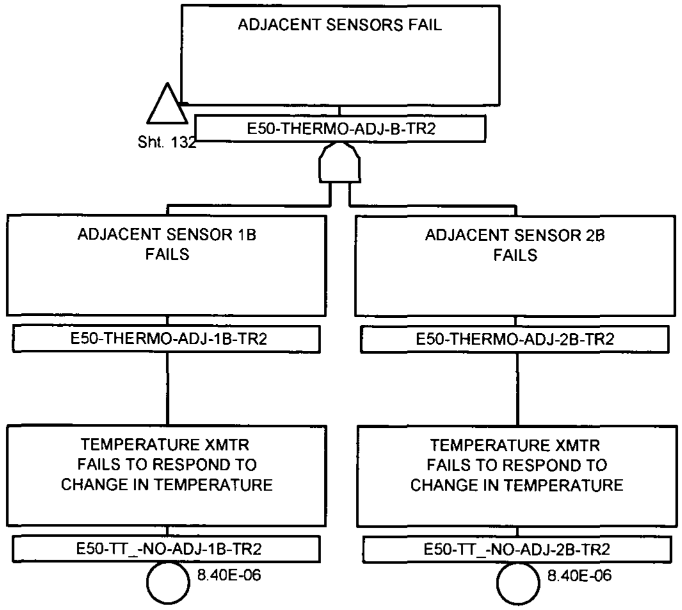

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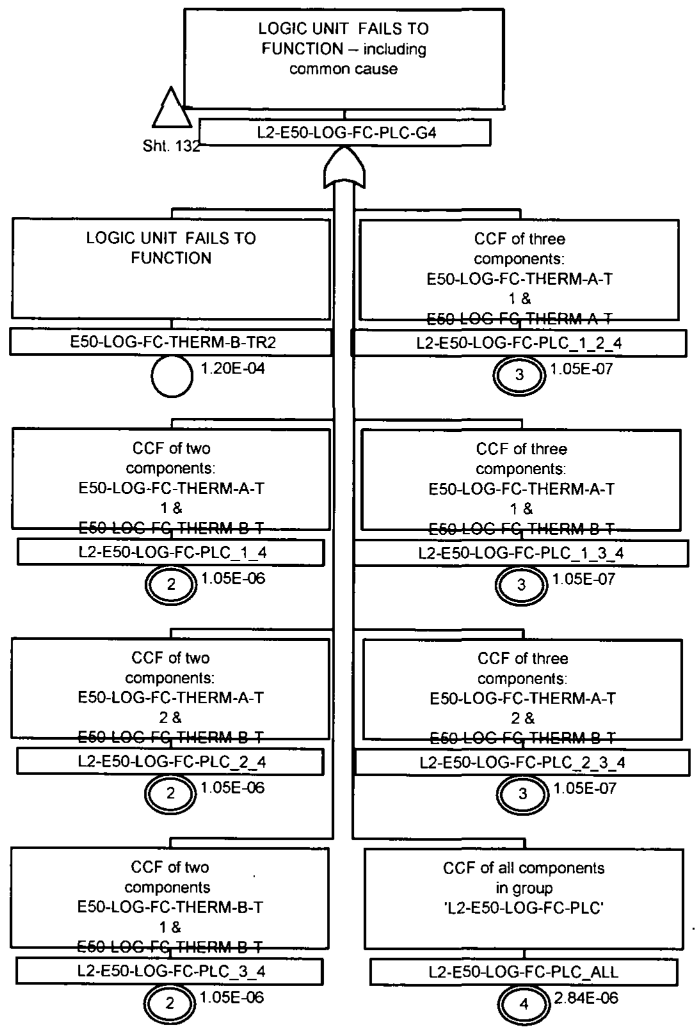

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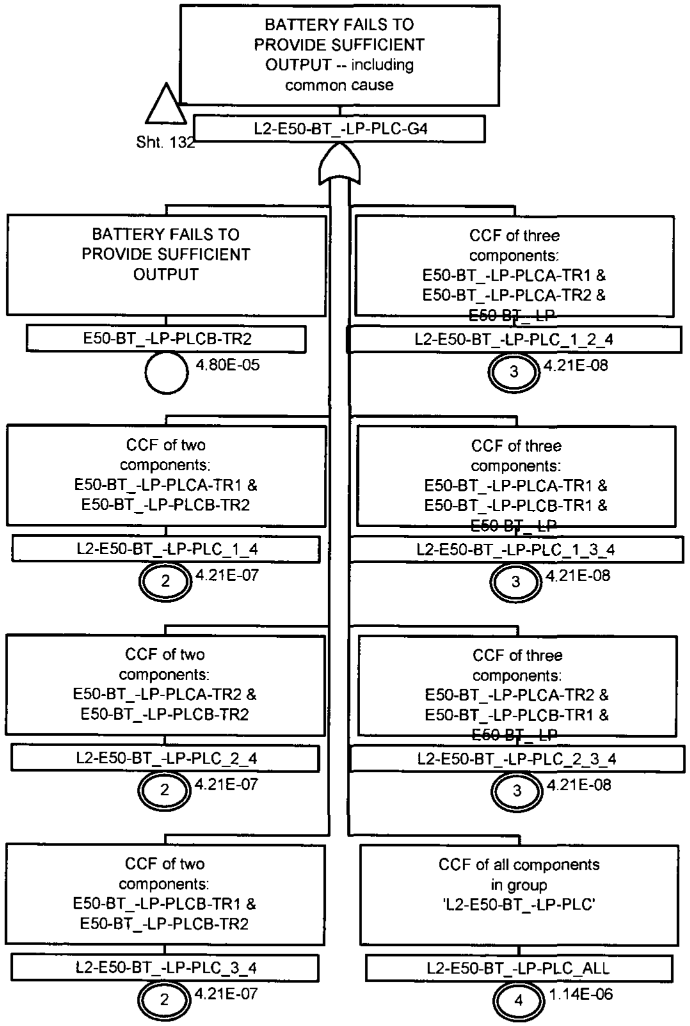

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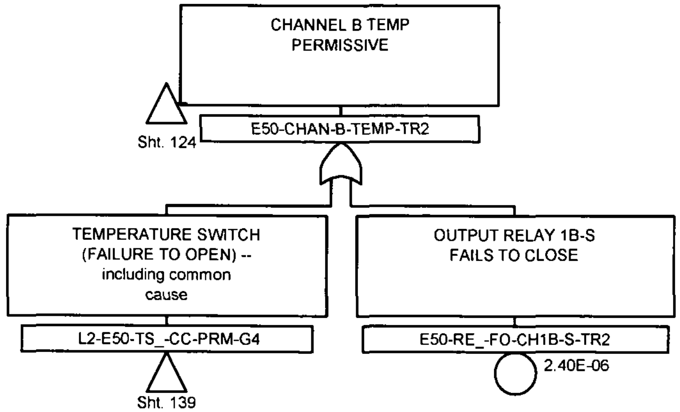

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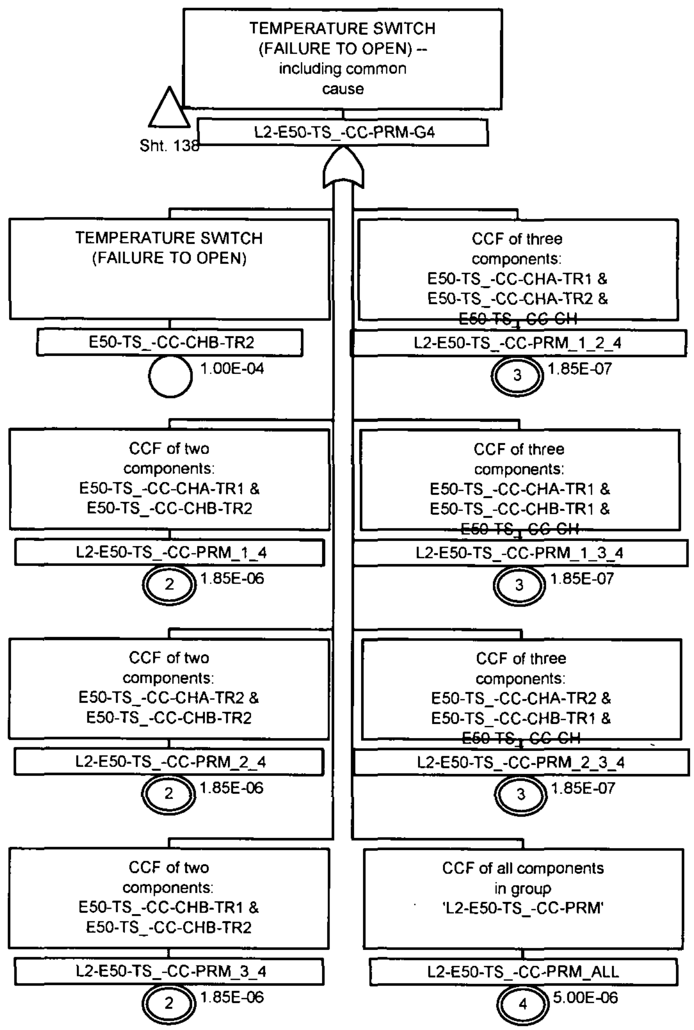

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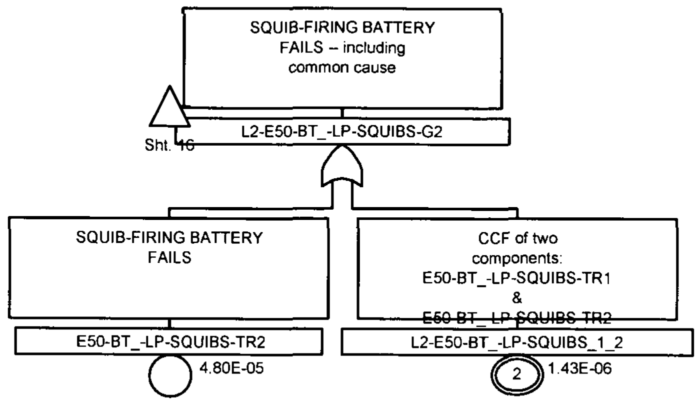

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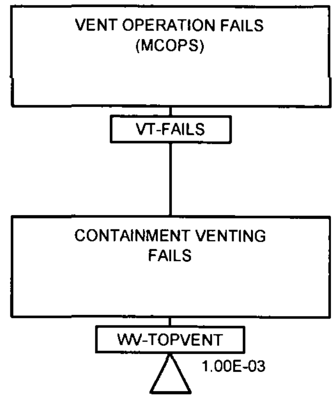

Figure 8A-8. Level 2 Fault Trees
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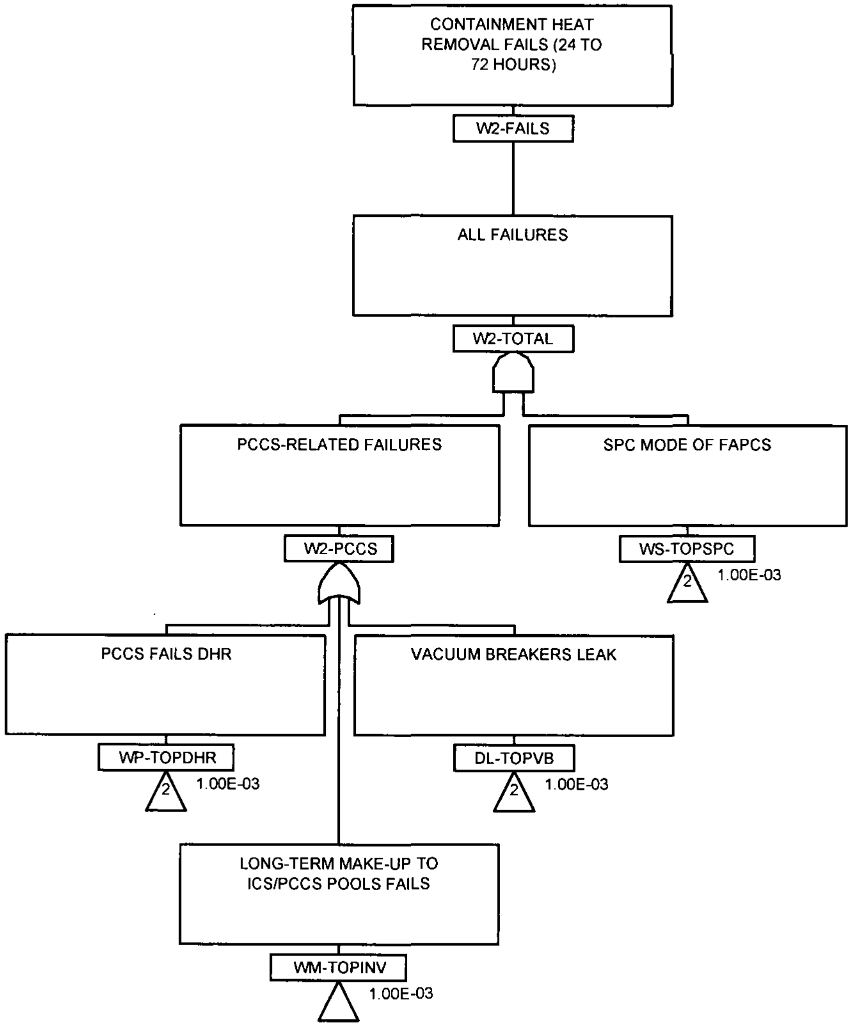

Figure 8A-8. Level 2 Fault Trees
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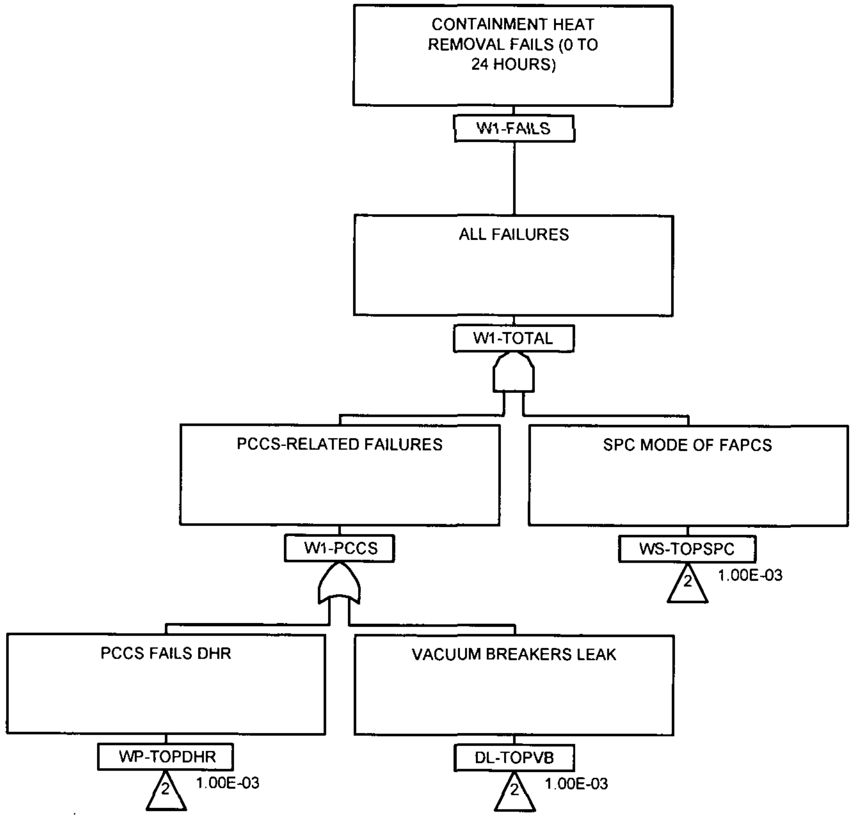

Figure 8A-8. Level 2 Fault Trees
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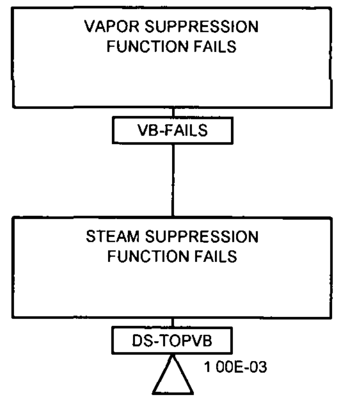

Figure 8A-8. Level 2 Fault Trees
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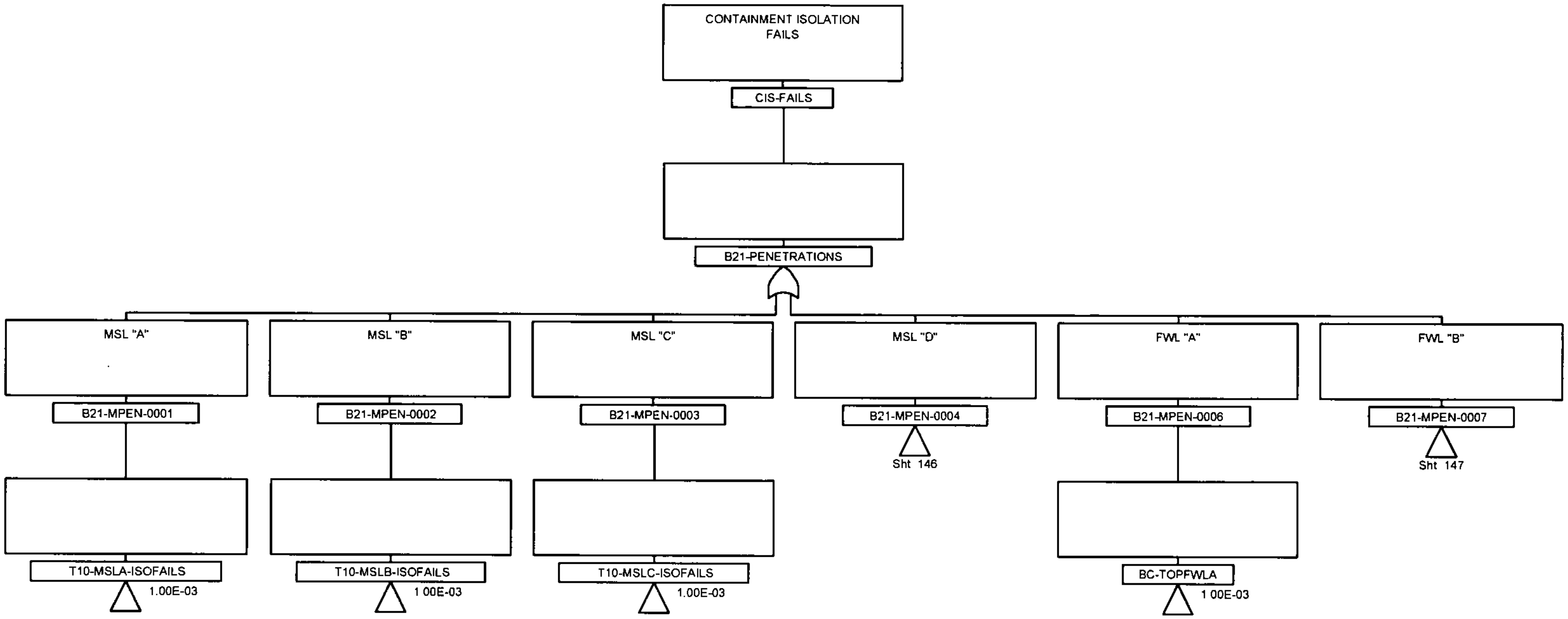

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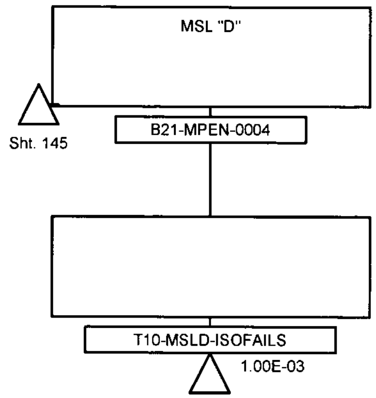

Figure 8A-8. Level 2 Fault Trees
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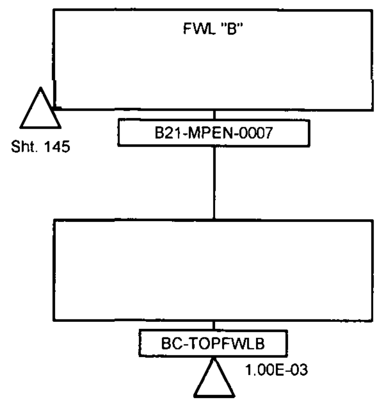

Figure 8A-8. Level 2 Fault Trees
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8B REPRESENTATIVE SEQUENCE RESULTS

The representative sequences analyzed during the containment analysis are shown in this appendix. Both a Technical Specifications Leakage (TSL) case and a Filtered Release (FR) are shown for Class I, Class III, and Class IV severe accident scenarios. Accident classes are defined in Section 7.2.2. A Class II scenario, in which containment failure precedes core damage, is also included. No Class V cases were analyzed because, by definition, containment function is directly bypassed due to an unisolated break outside containment (BOC).

8B.1 – Class I Representative Sequence (T_nIN_TSL)

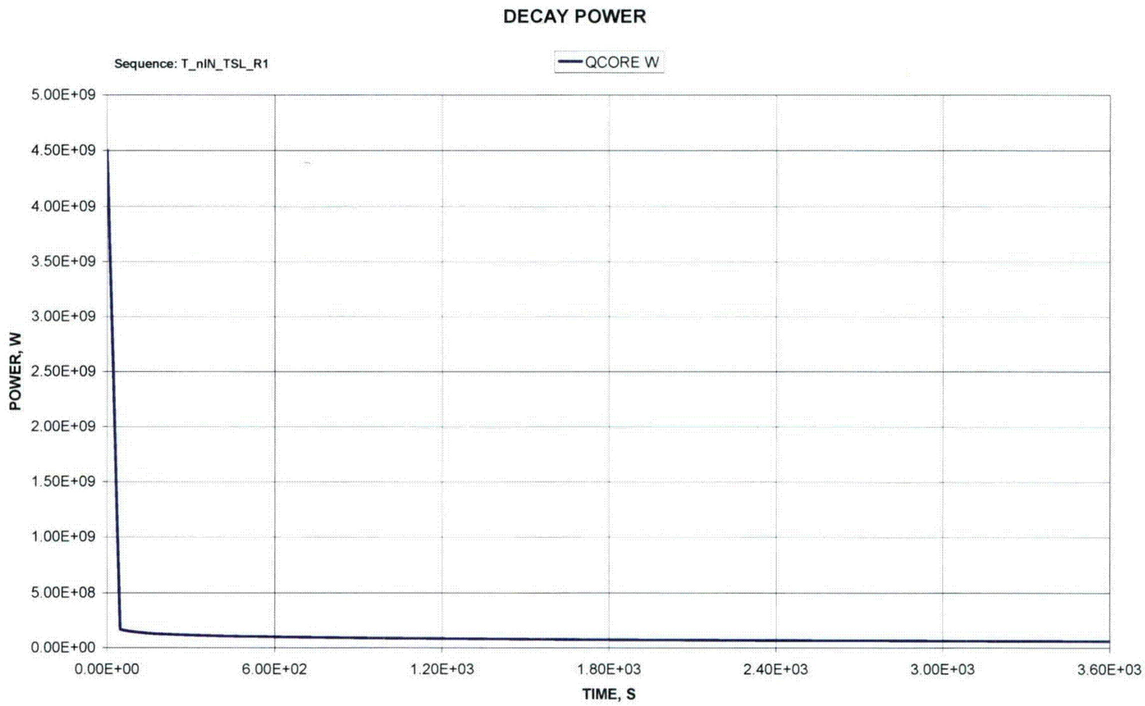

Figure 8B-1a – CDI TSL Core Power

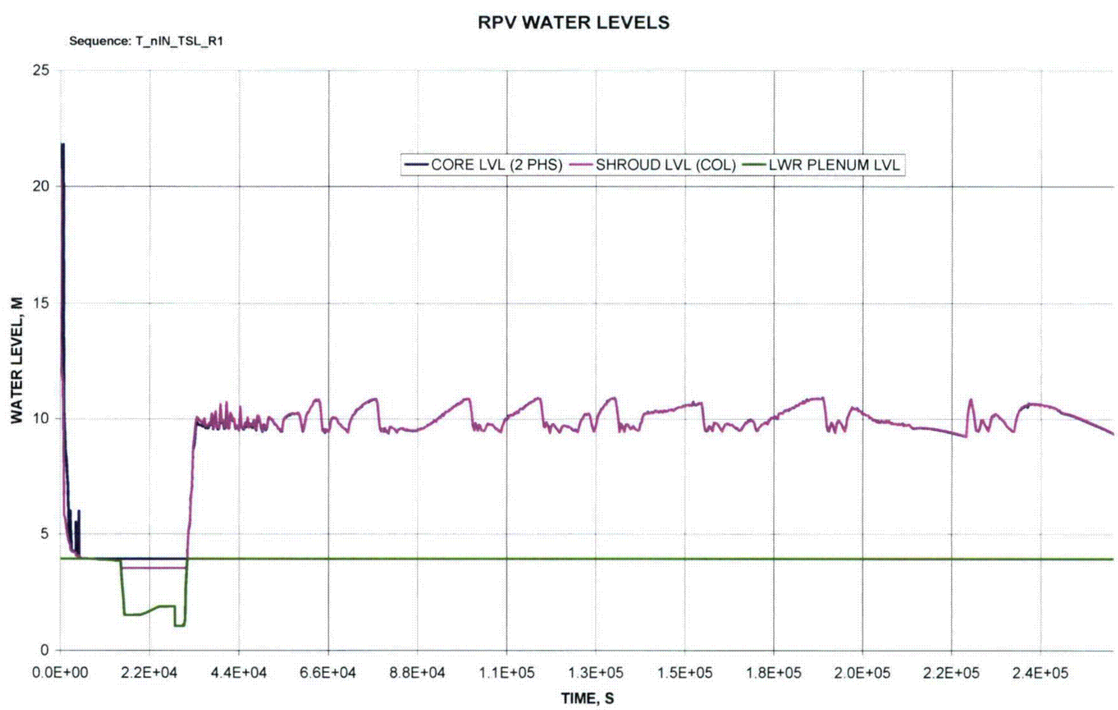

Figure 8B-1b – CDI TSL RPV Water Levels

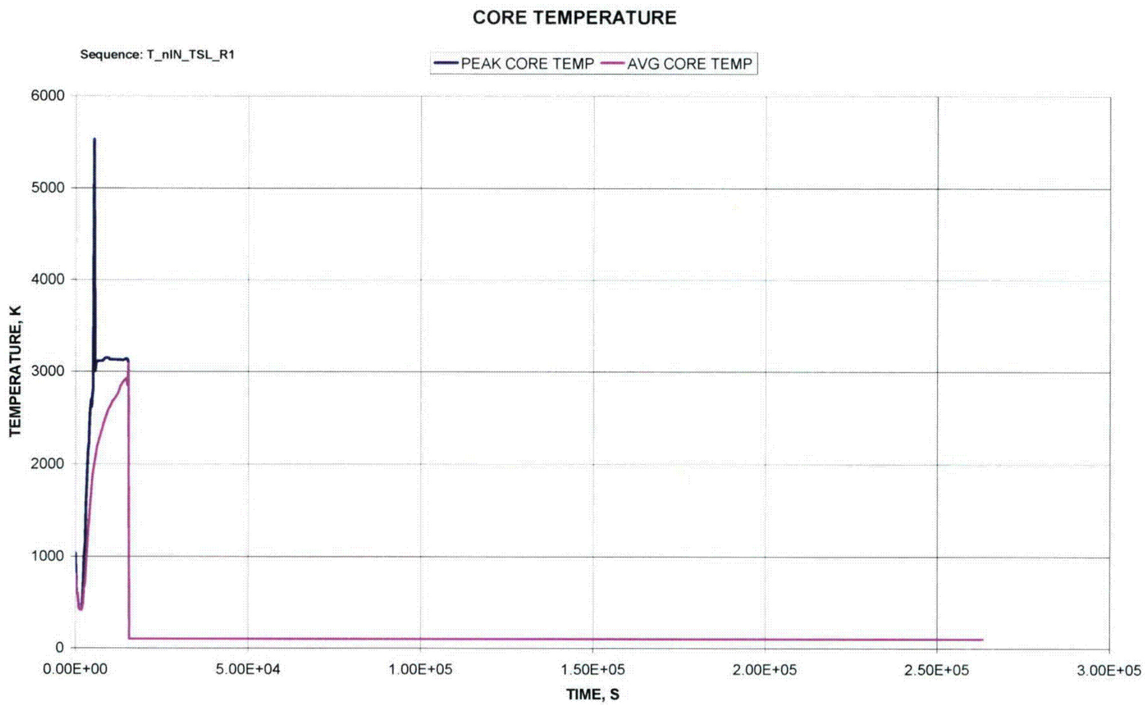

Figure 8B-1c – CDI TSL Core Temperature

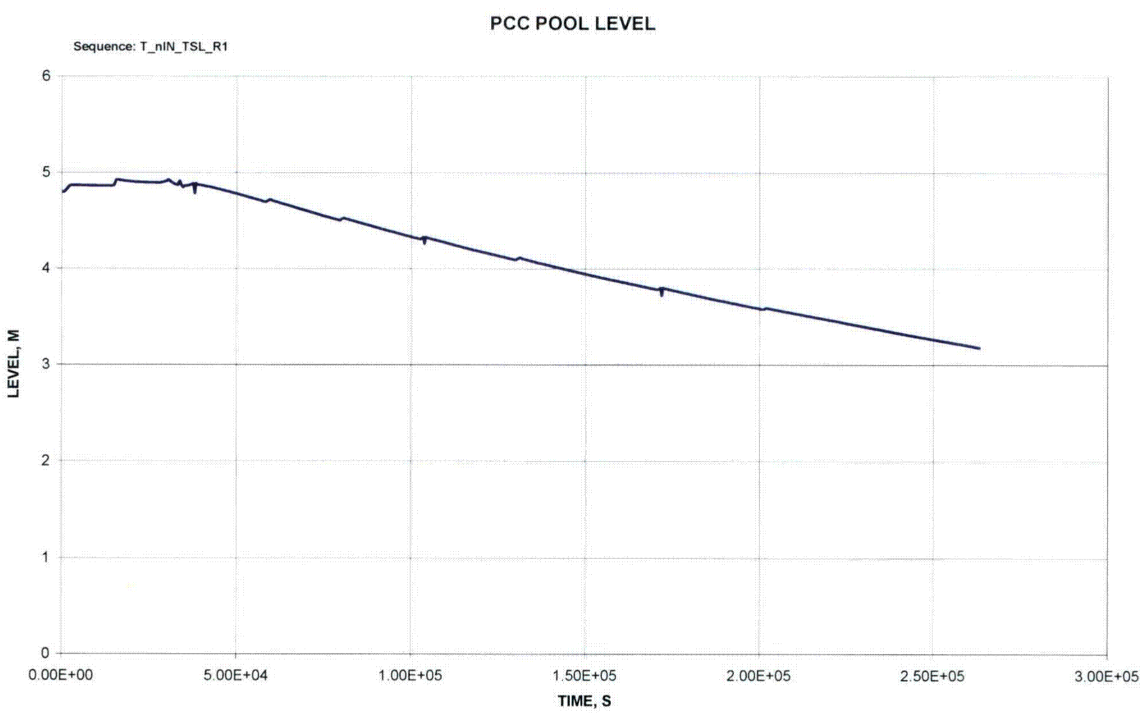

Figure 8B-1d – CDI TSL PCCS Pool Level

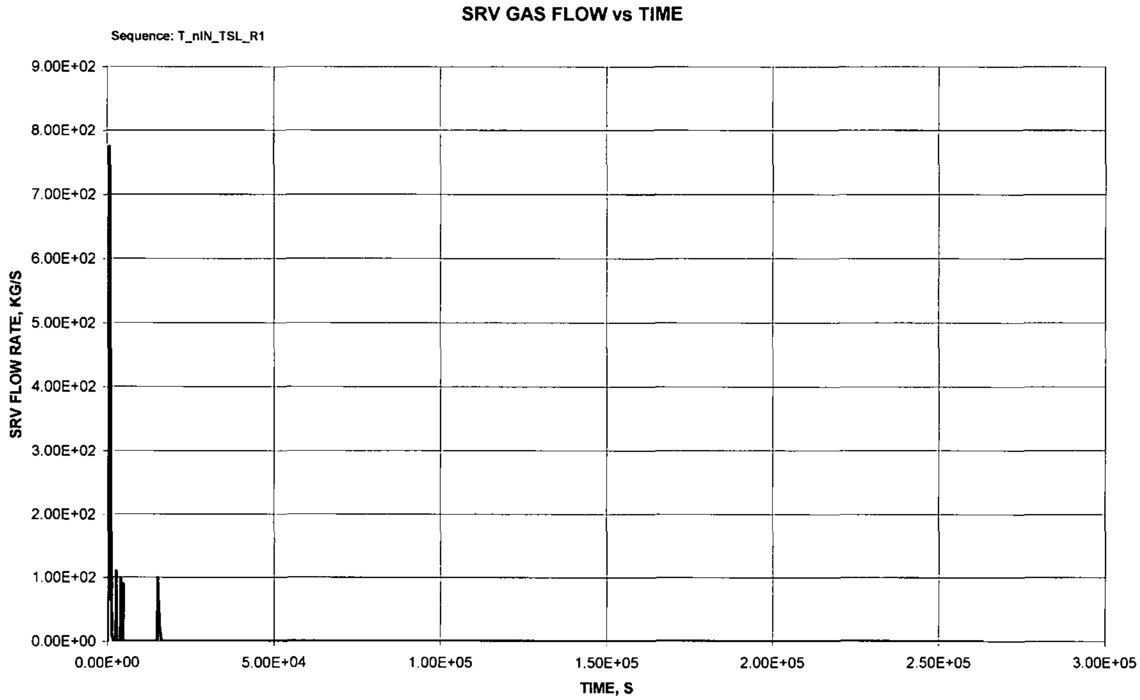

Figure 8B-1e – CDI TSL SRV Gas Flow vs Time

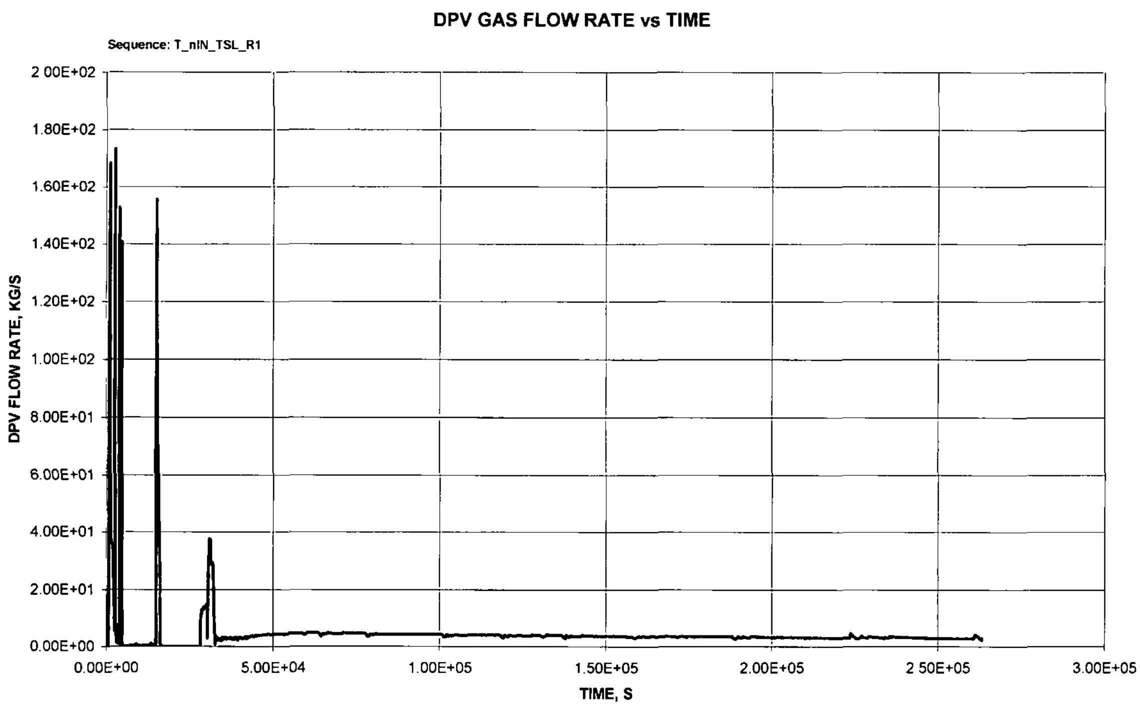

Figure 8B-1f – CDI TSL DPV Gas Flow vs Time

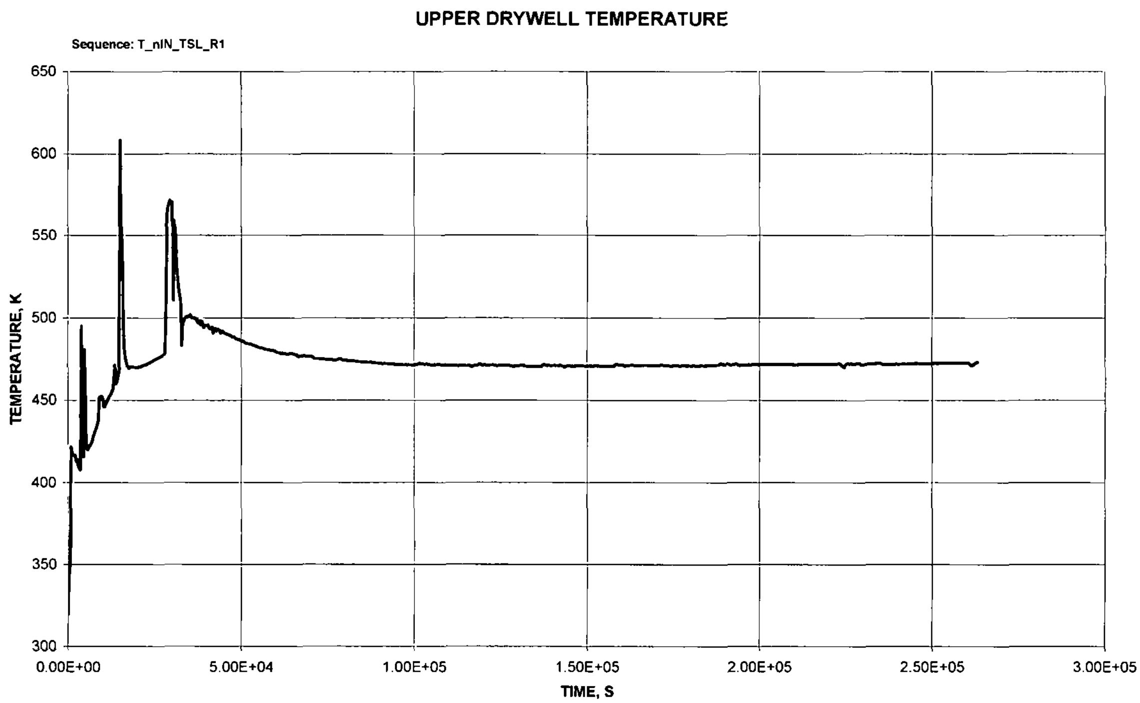

Figure 8B-1g – CDI TSL Drywell Gas Temperature

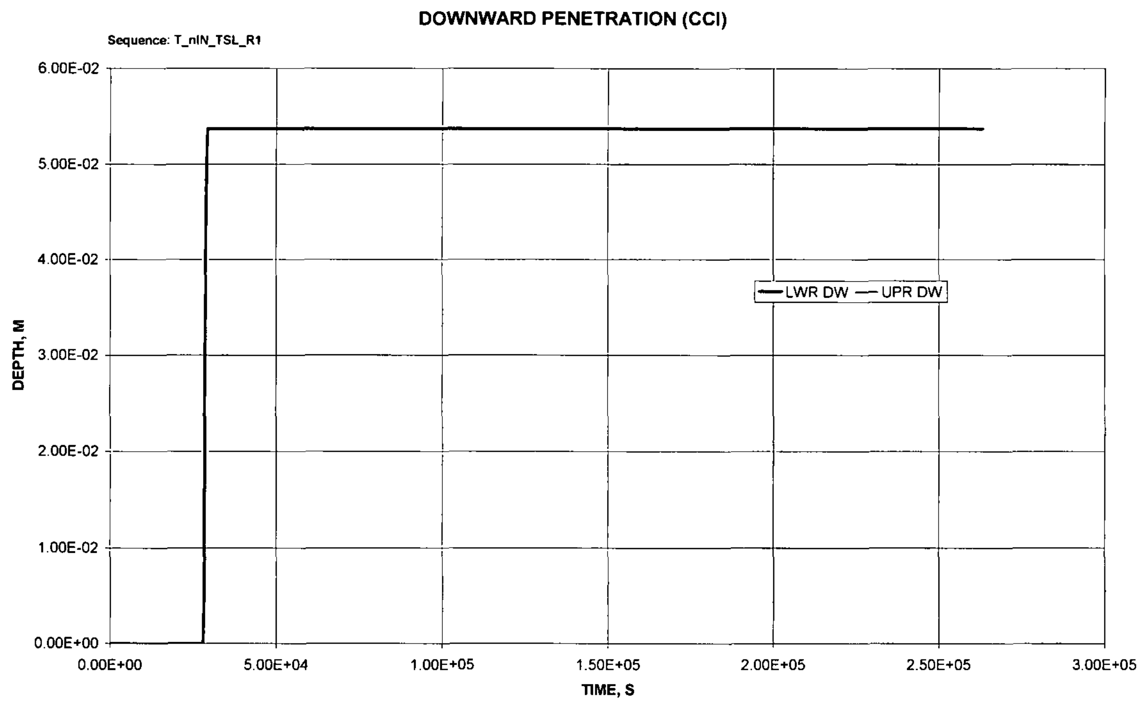

Figure 8B-1h – CDI TSL Downward CCI Penetration

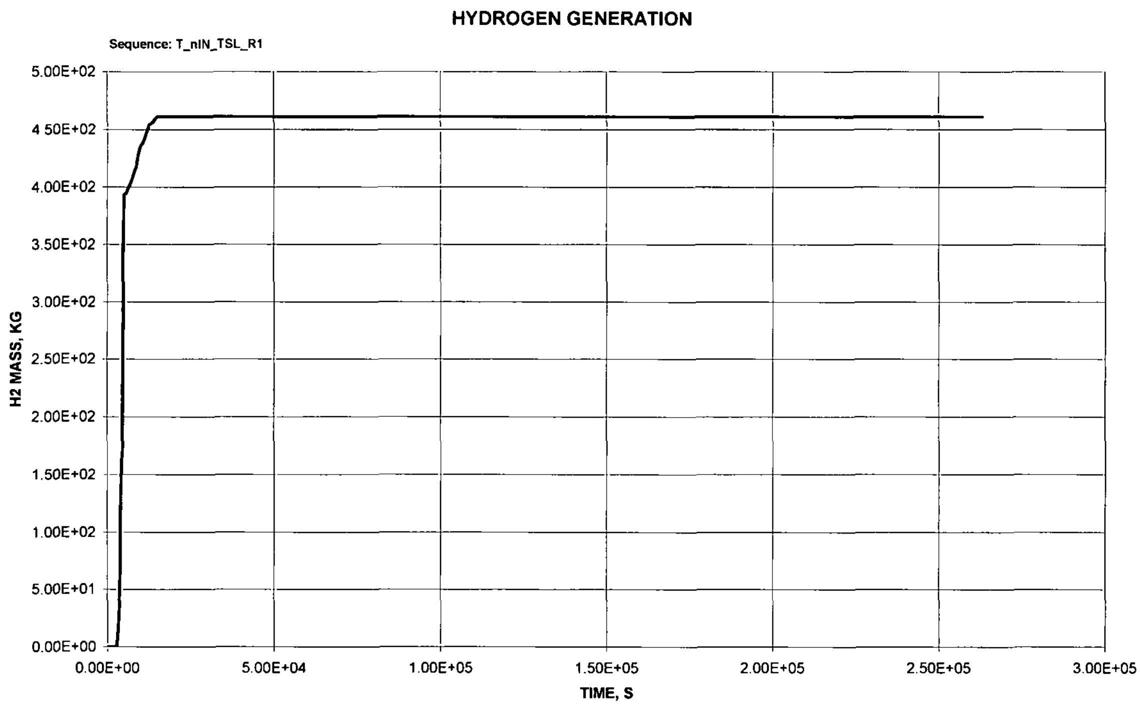

Figure 8B-1i – CDI TSL Hydrogen Generation

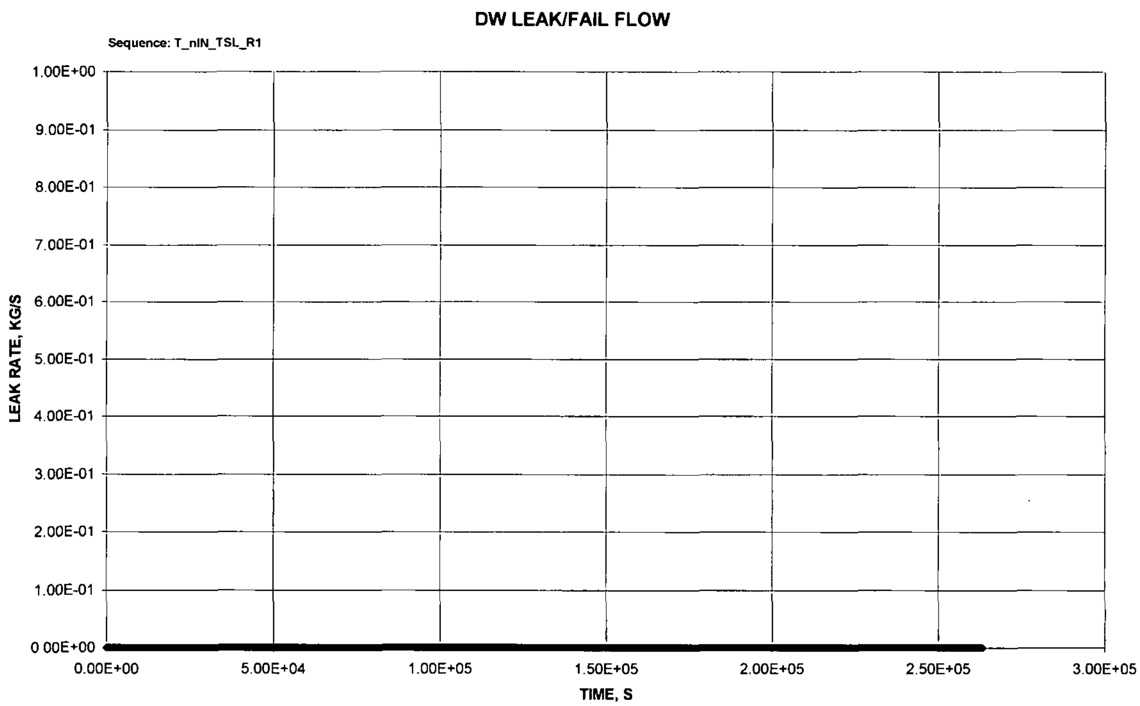

Figure 8B-1j – CDI TSL Drywell Leak/Failure Flow Rate

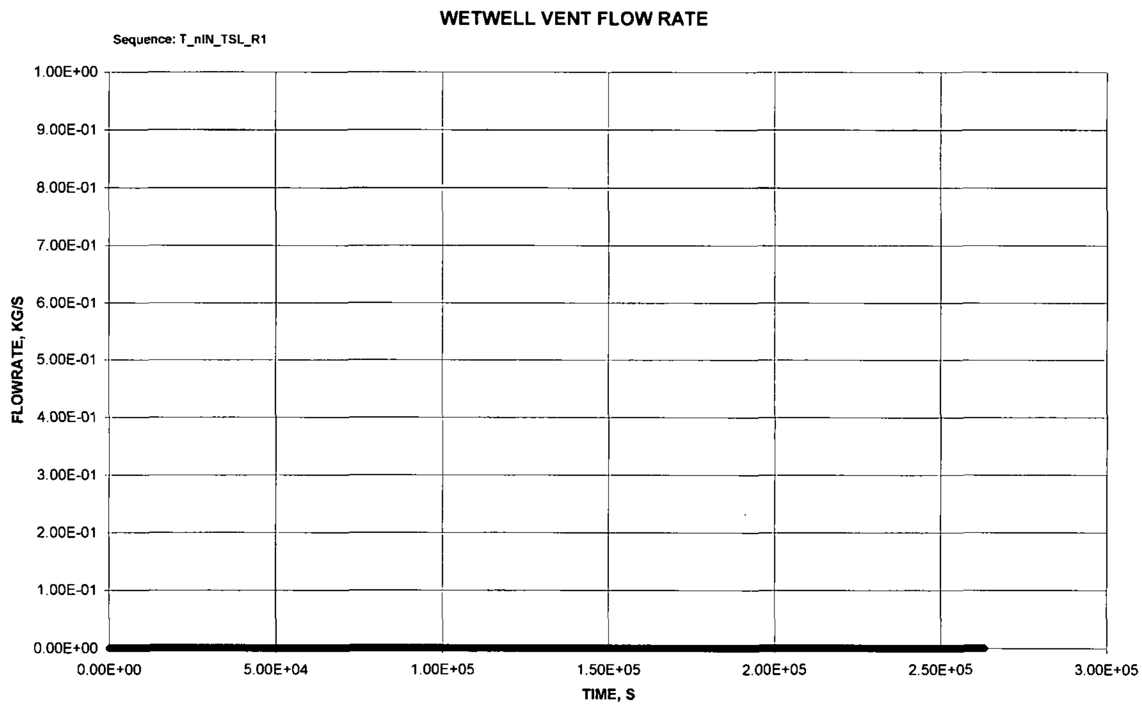

Figure 8B-1k – CDI TSL Wetwell Vent Flow Rate

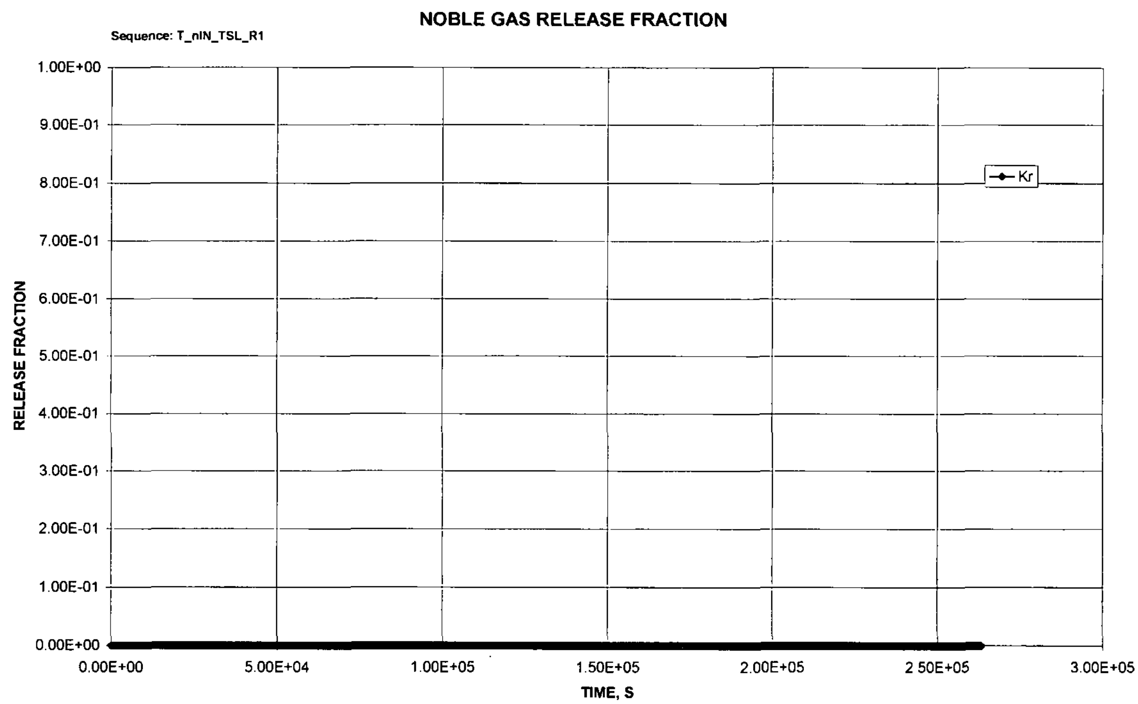

Figure 8B-1l – CDI TSL Noble Gas Release Fraction

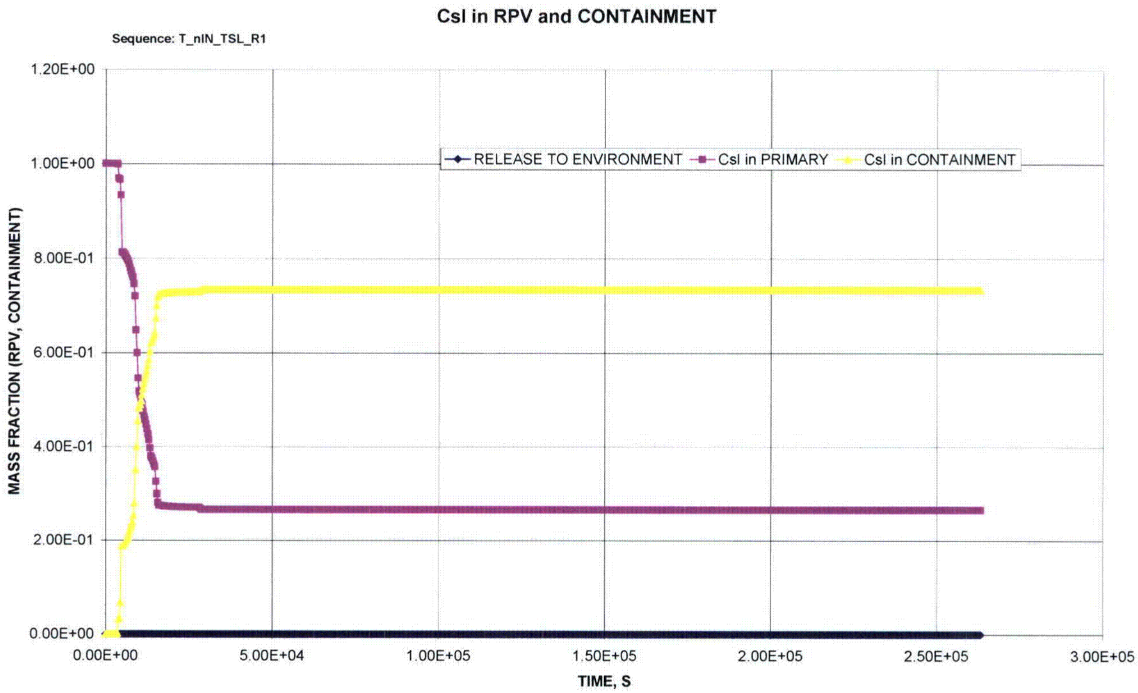

Figure 8B-1m – CDI TSL CsI Distribution

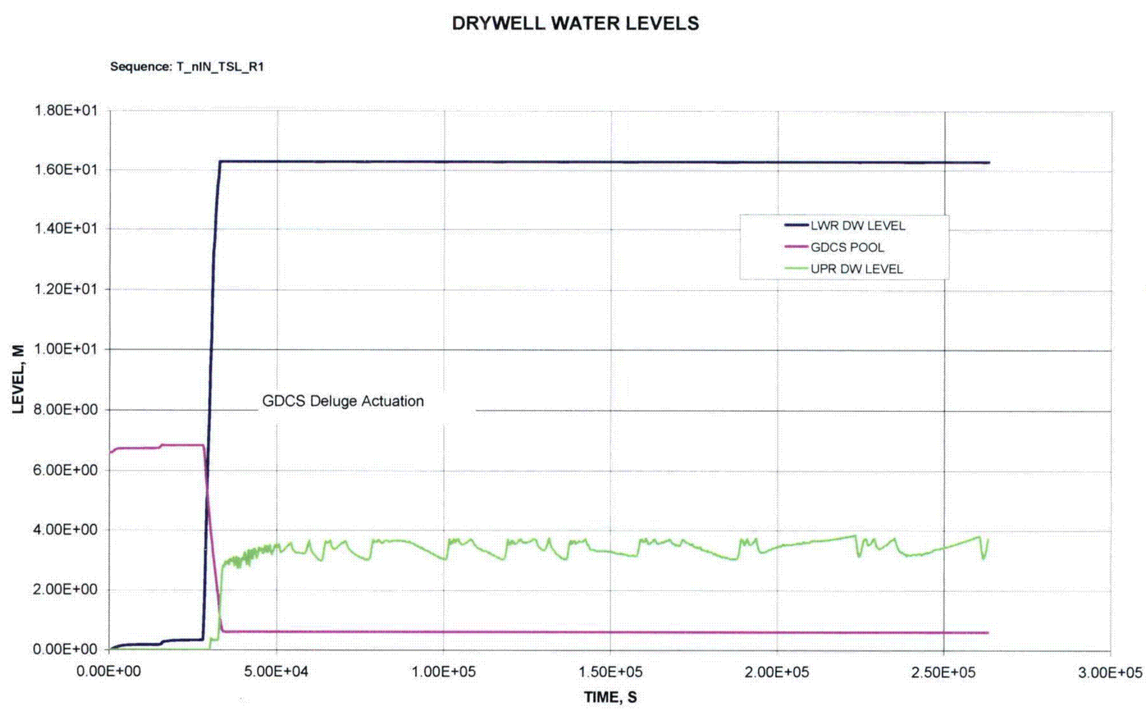

Figure 8B-1n – CDI TSL Drywell Water Levels

8B.2 – Class I Representative Sequence (T_nIN_nCHR_FR)

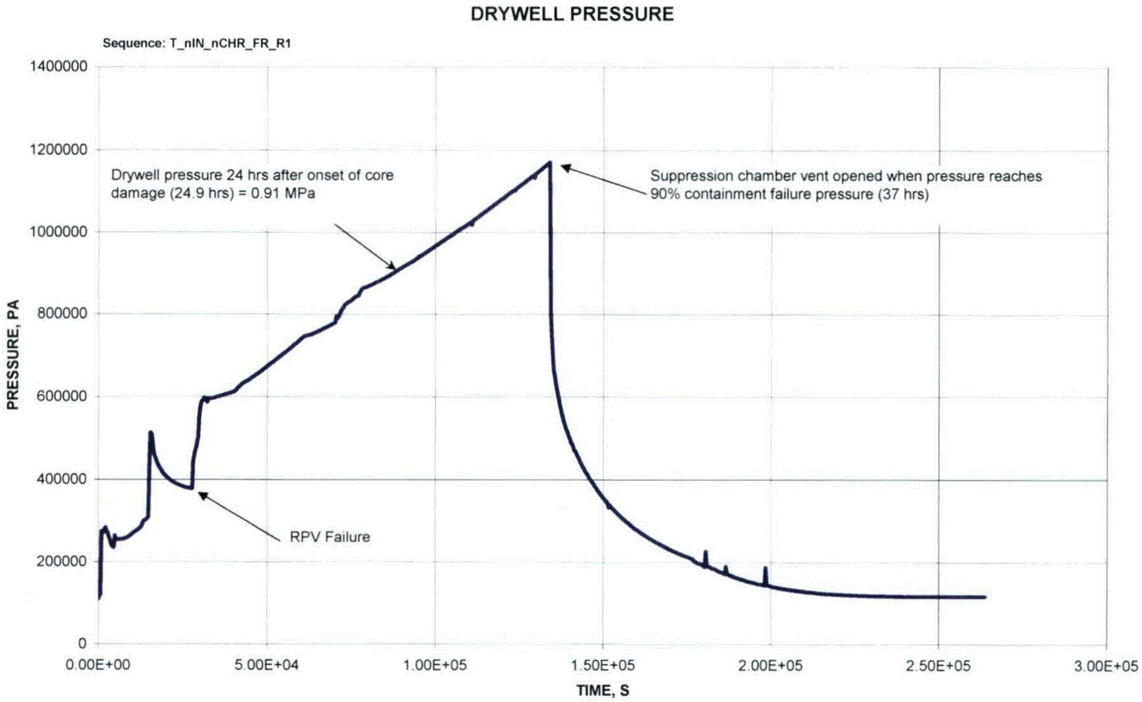

Figure 8B-2a – CDI FR Containment Pressures

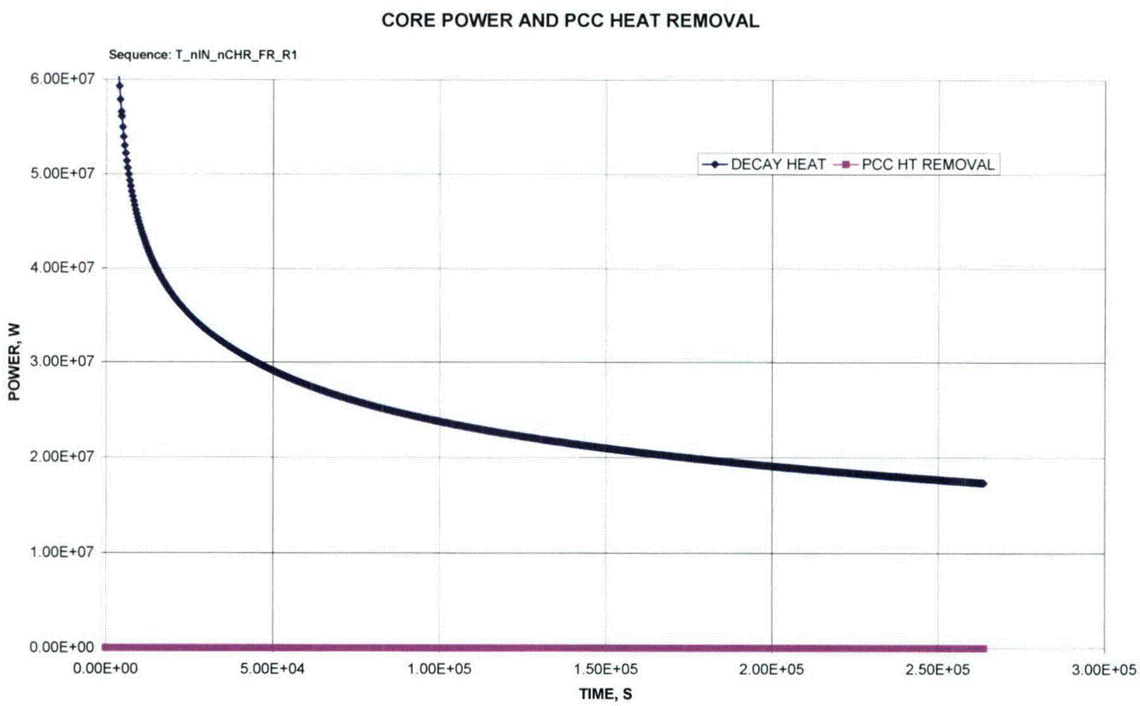

Figure 8B-2b – CDI FR Core Power and PCCS Heat Removal

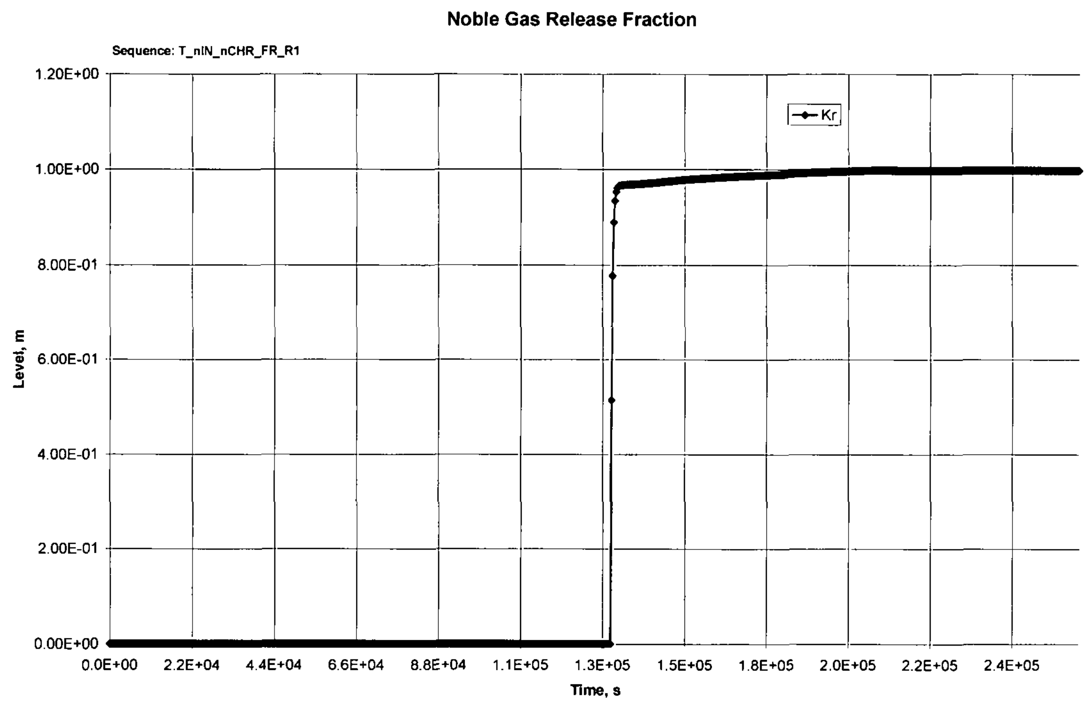

Figure 8B-2c – CDI FR Noble Gas Release Fraction

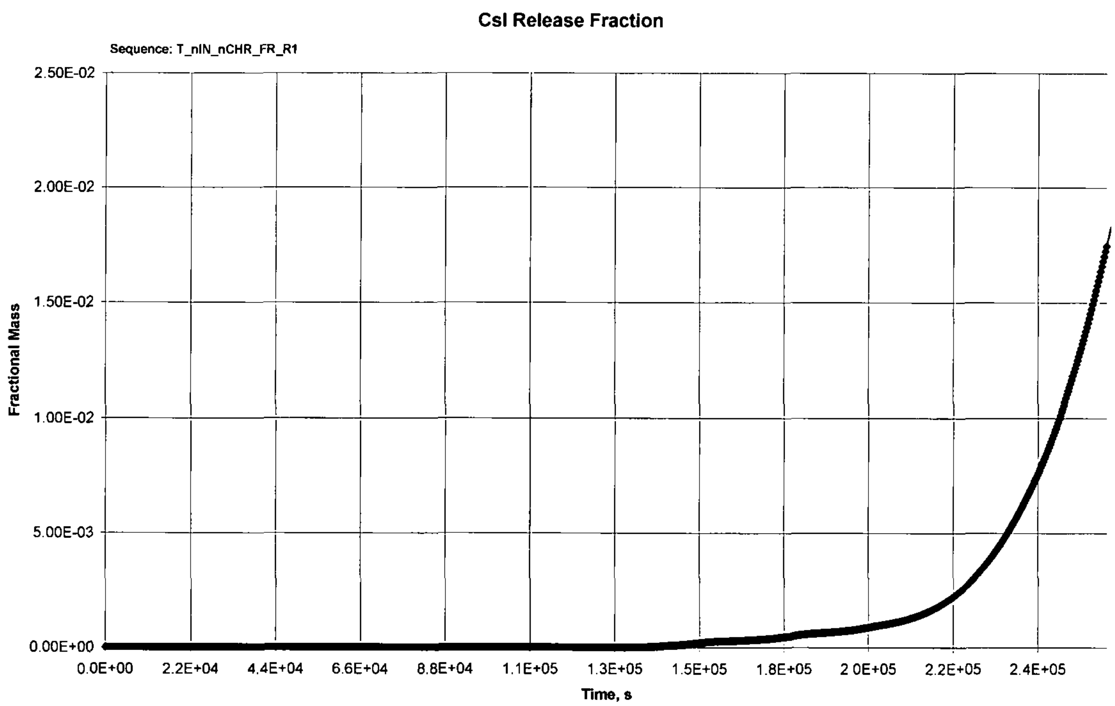

Figure 8B-2d – CDI FR CsI Release Fraction

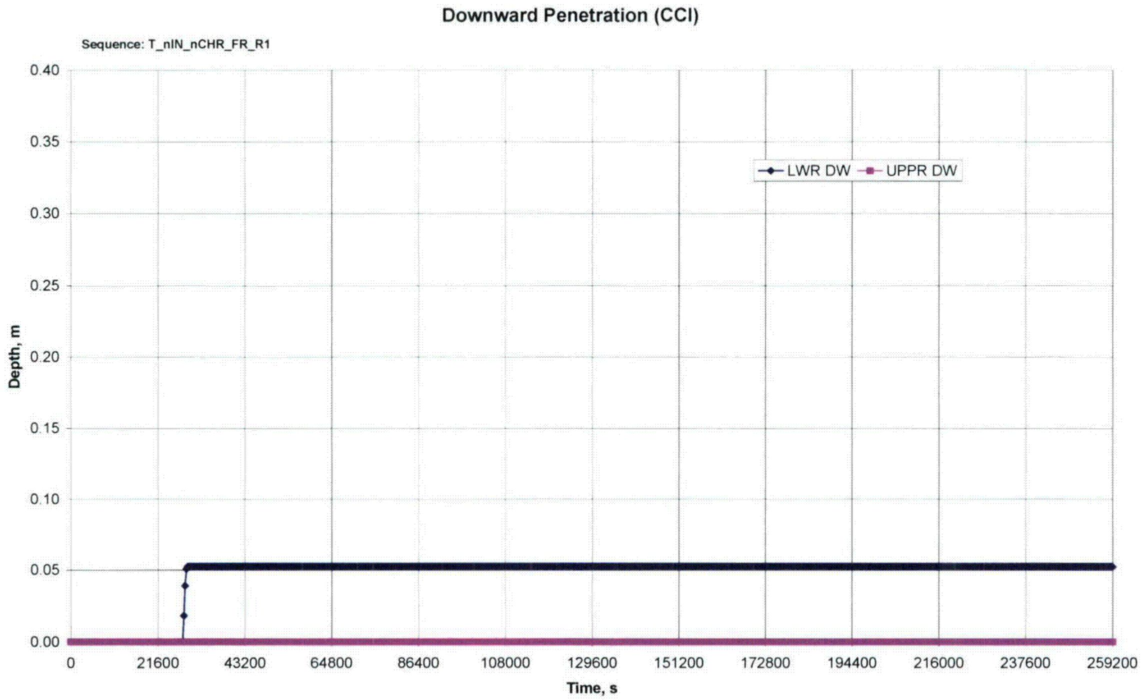

Figure 8B-2e – CDI FR Core Concrete Interaction

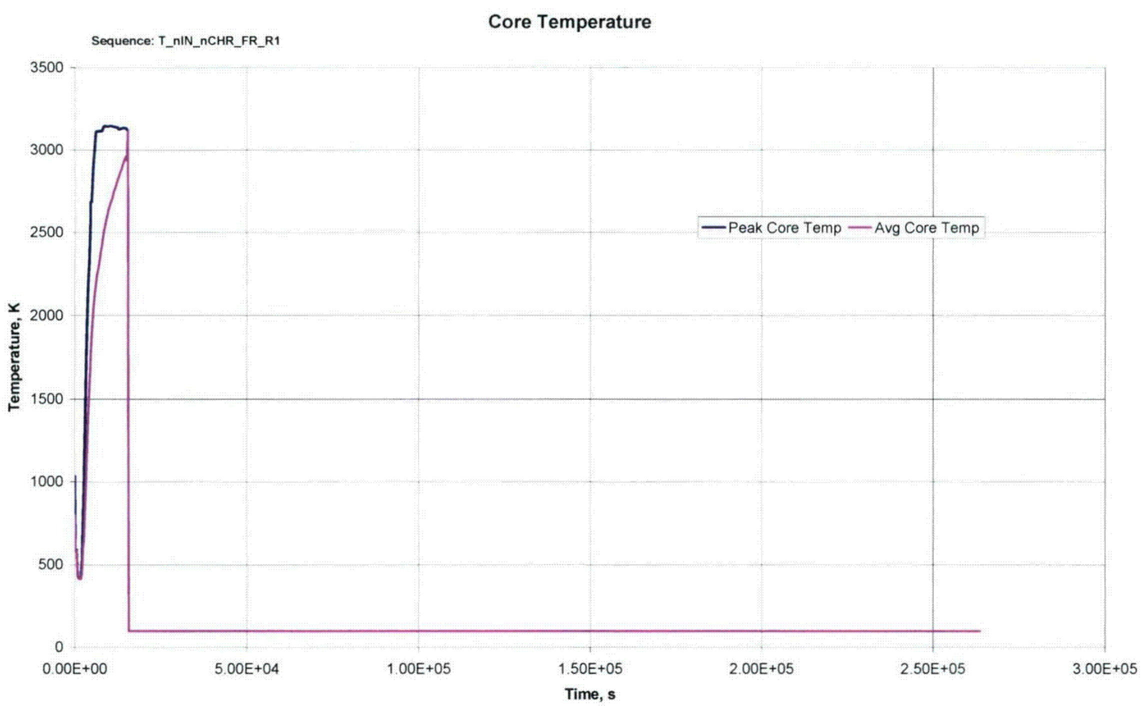

Figure 8B-2f – CDI FR Core Temperature

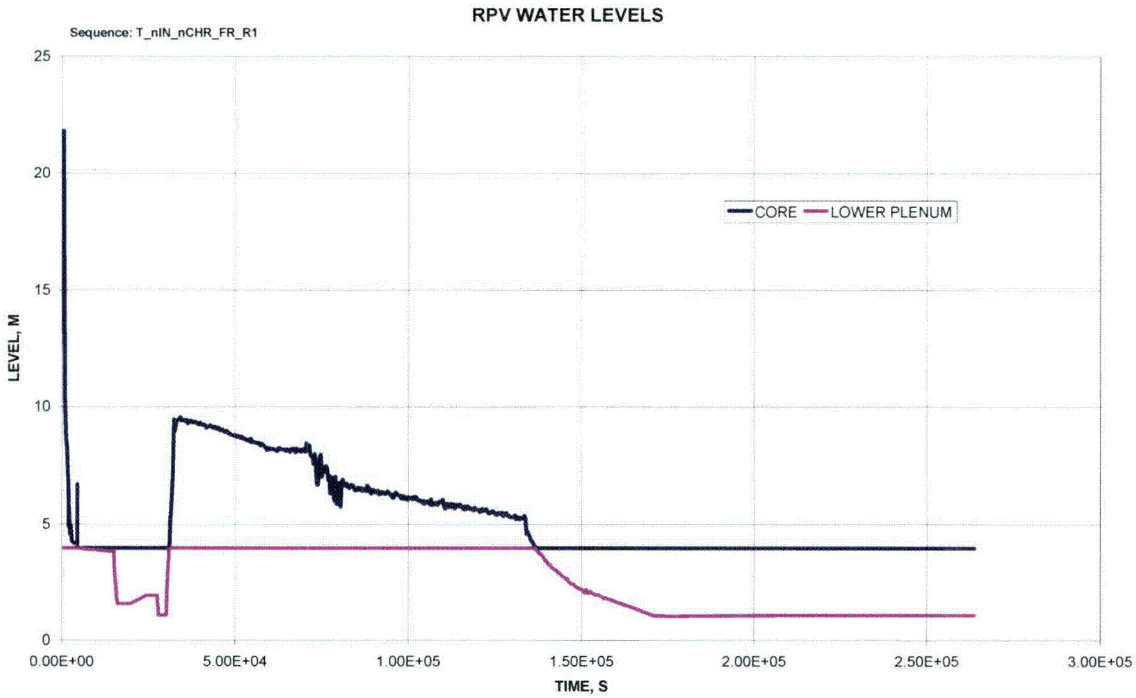

Figure 8B-2g – CDI FR RPV Water Levels

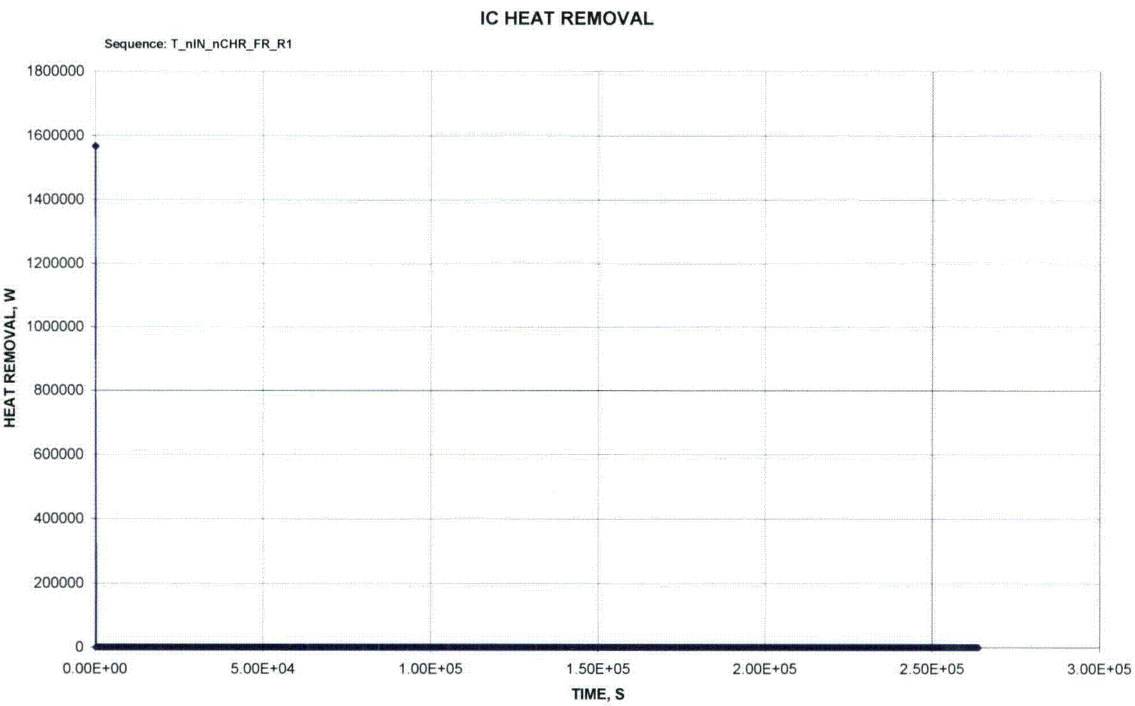

Figure 8B-2h – CDI FR ICS Heat Removal

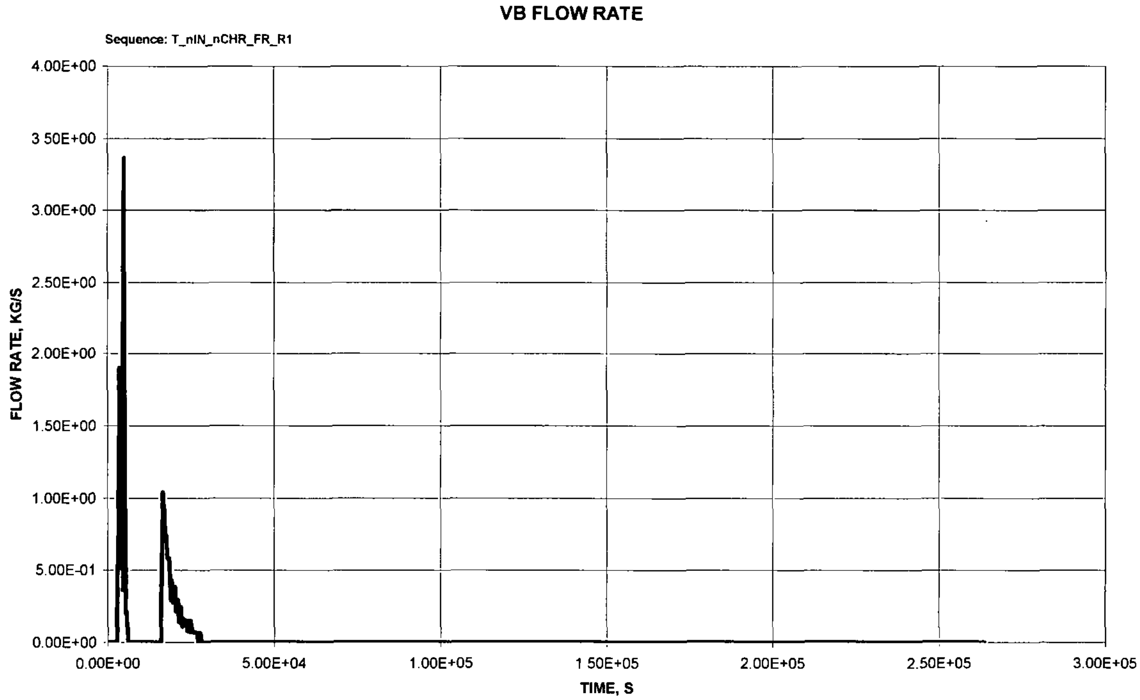

Figure 8B-2i – CDI FR Vacuum Breaker Flow Rate

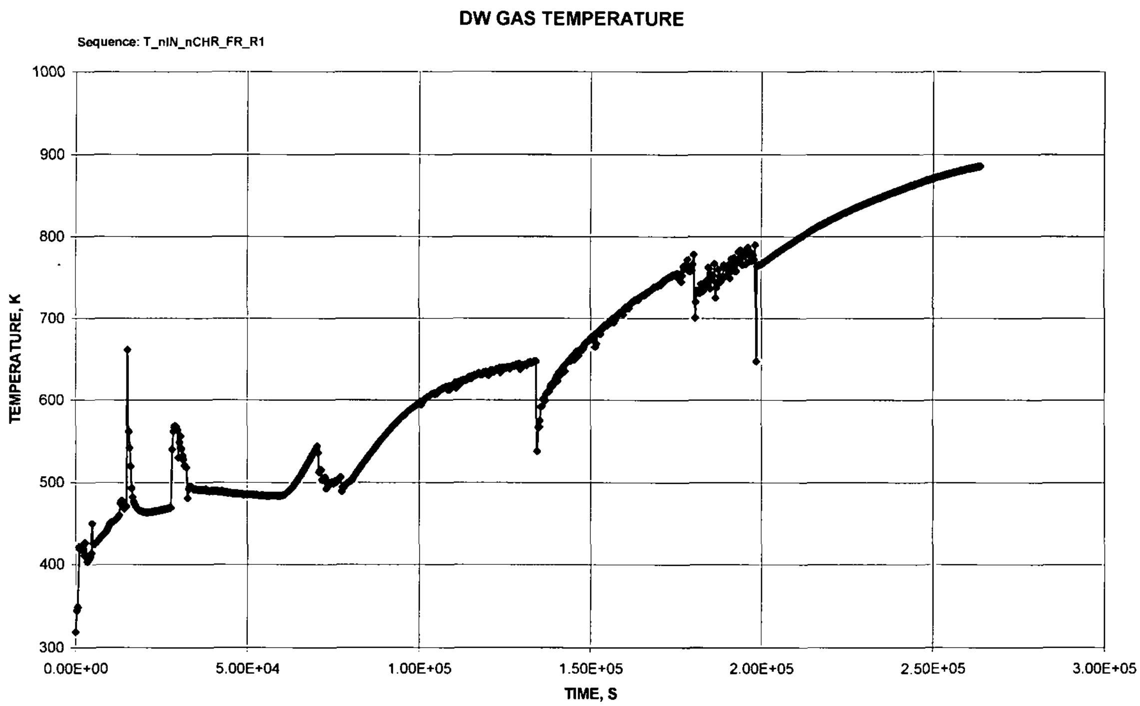

Figure 8B-2j – CDI FR Drywell Gas Temperature

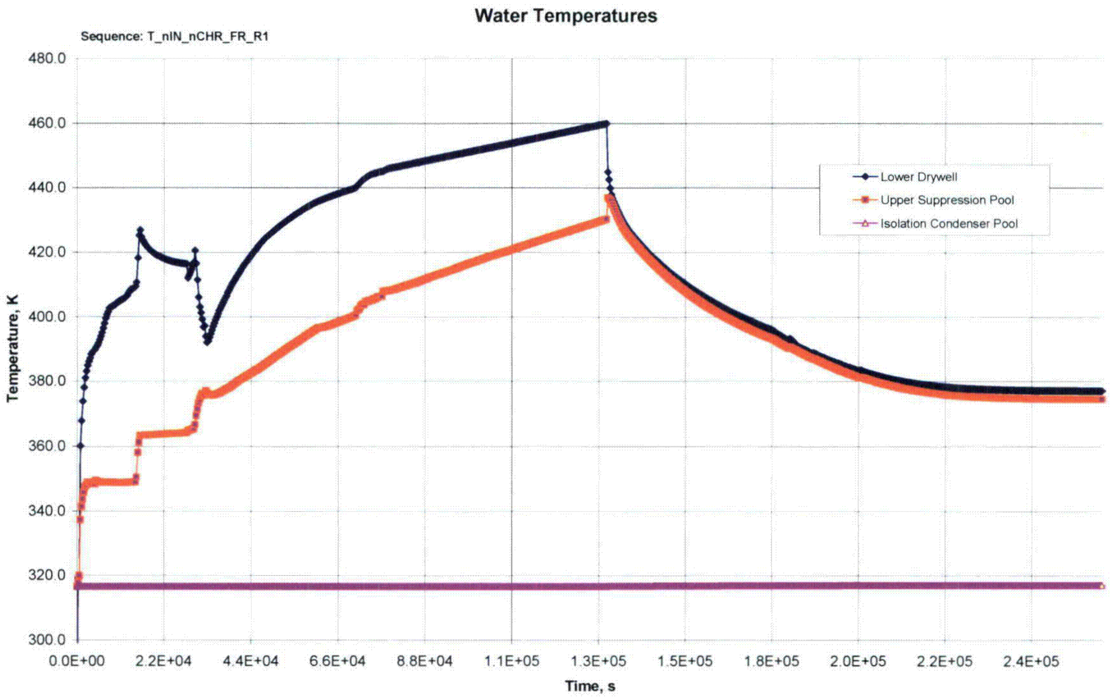

Figure 8B-2k – CDI FR Water Temperatures

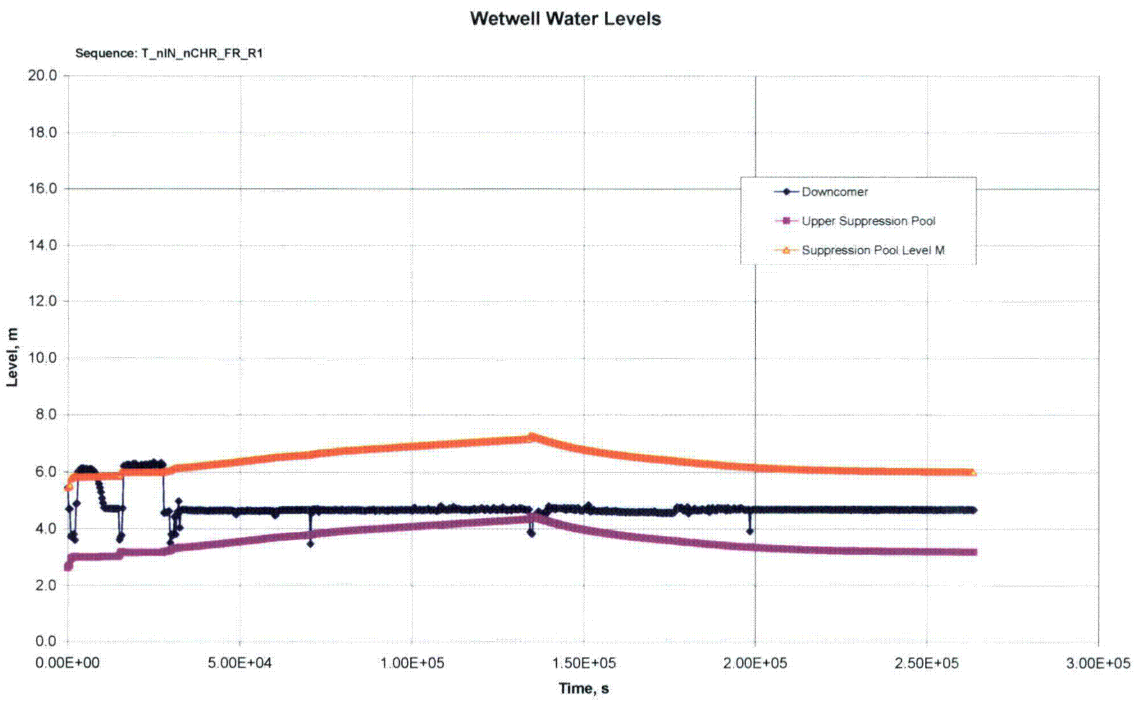

Figure 8B-2l – CDI FR Wetwell Water Levels

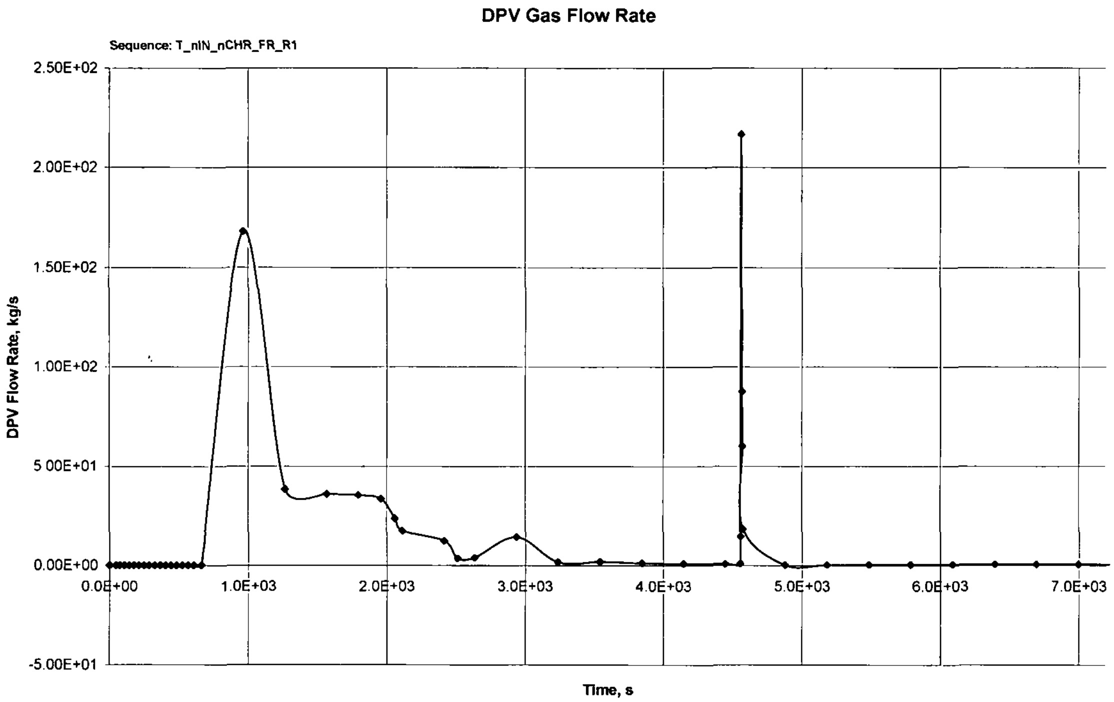

Figure 8B-2m – CDI FR SRV Gas Flow Rate vs Time

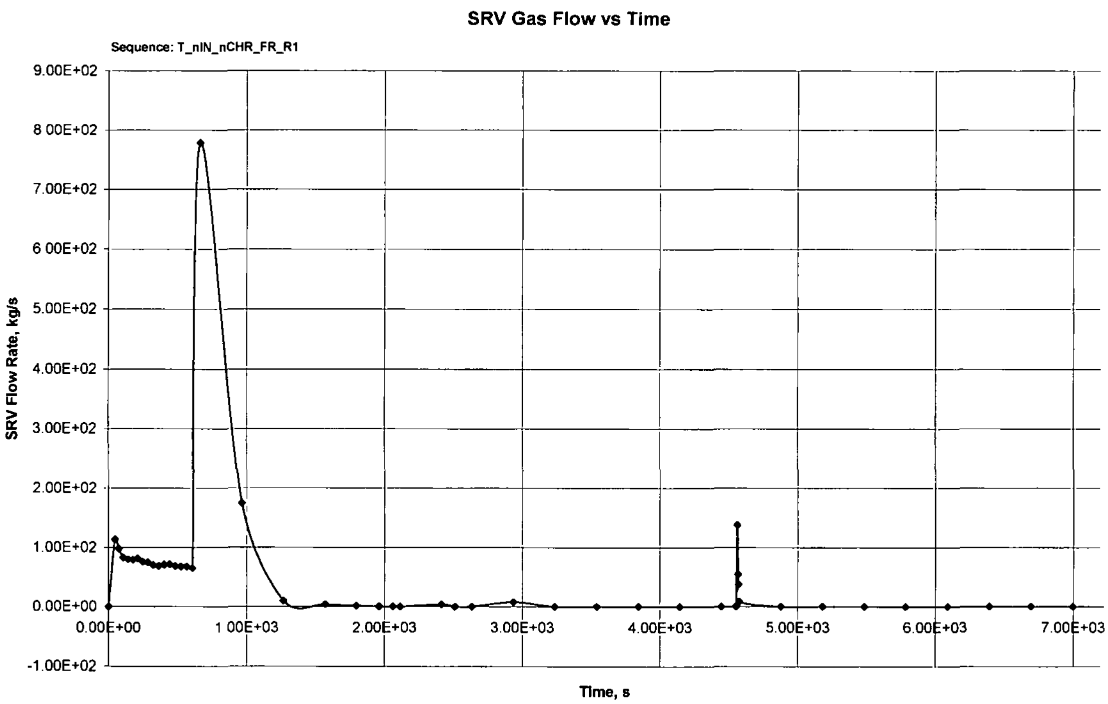

Figure 8B-2n – CDI FR DPV Gas Flow Rate vs Time

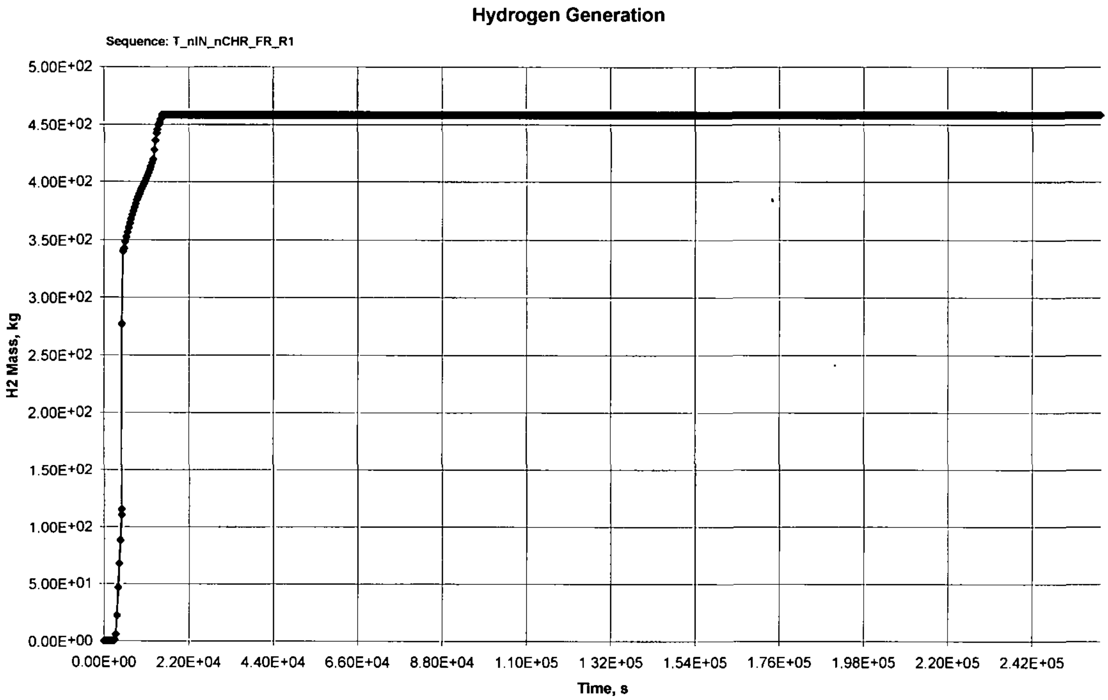

Figure 8B-2o – CDI FR Hydrogen Generation

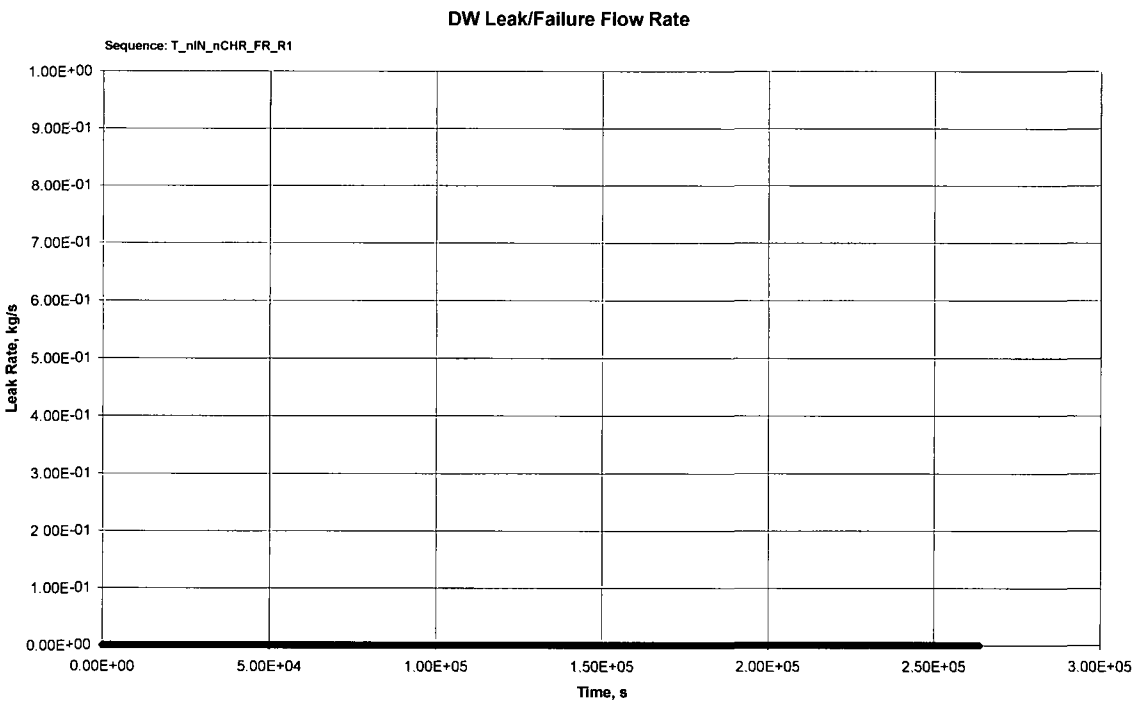

Figure 8B-2p – CDI FR Drywell Leak/Failure Flow Rate

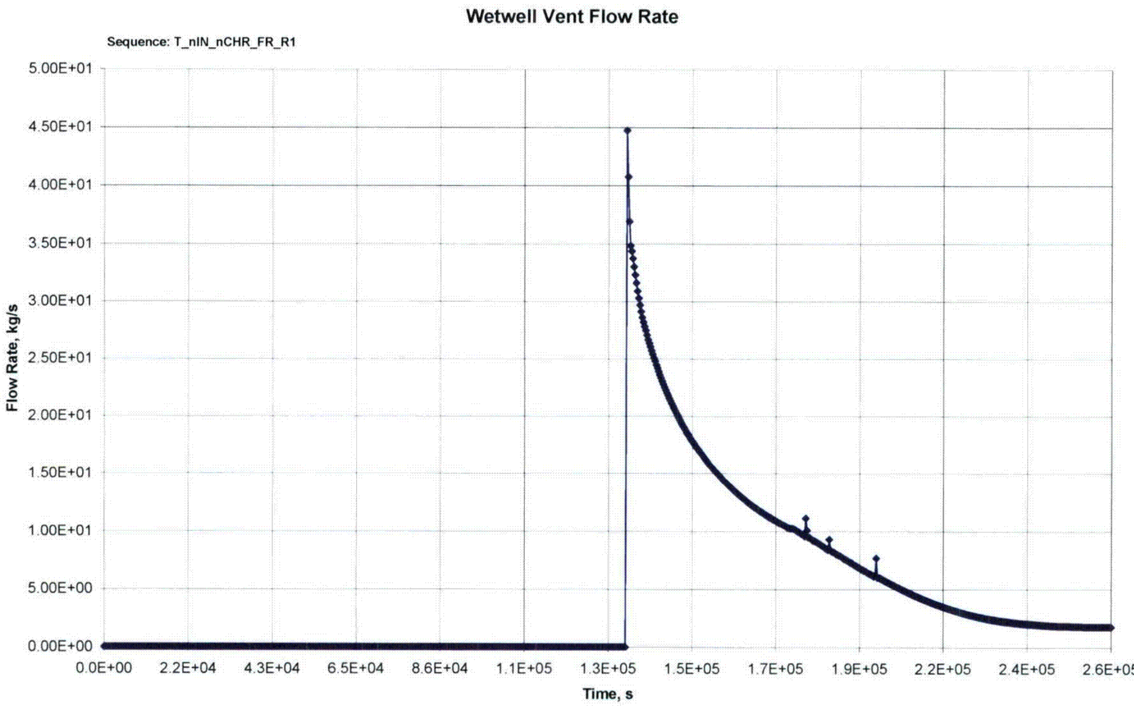

Figure 8B-2q – CDI FR Wetwell Vent Flow Rate

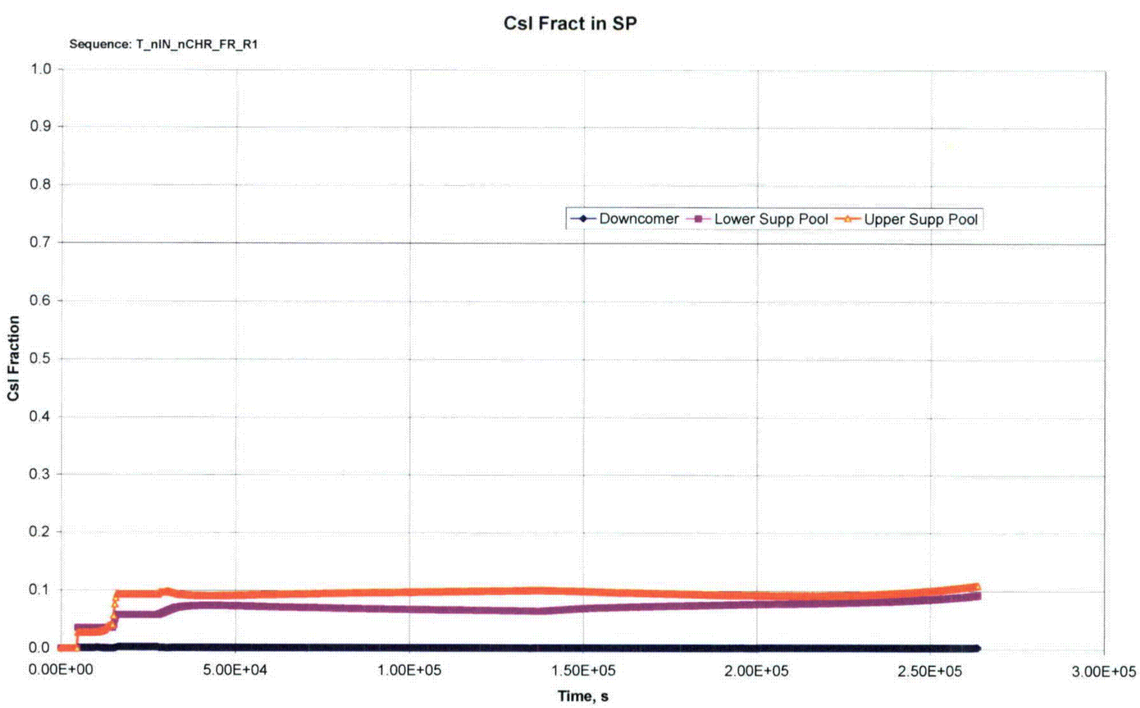

Figure 8B-2r – CDI FR CsI Fraction in Suppression Pool

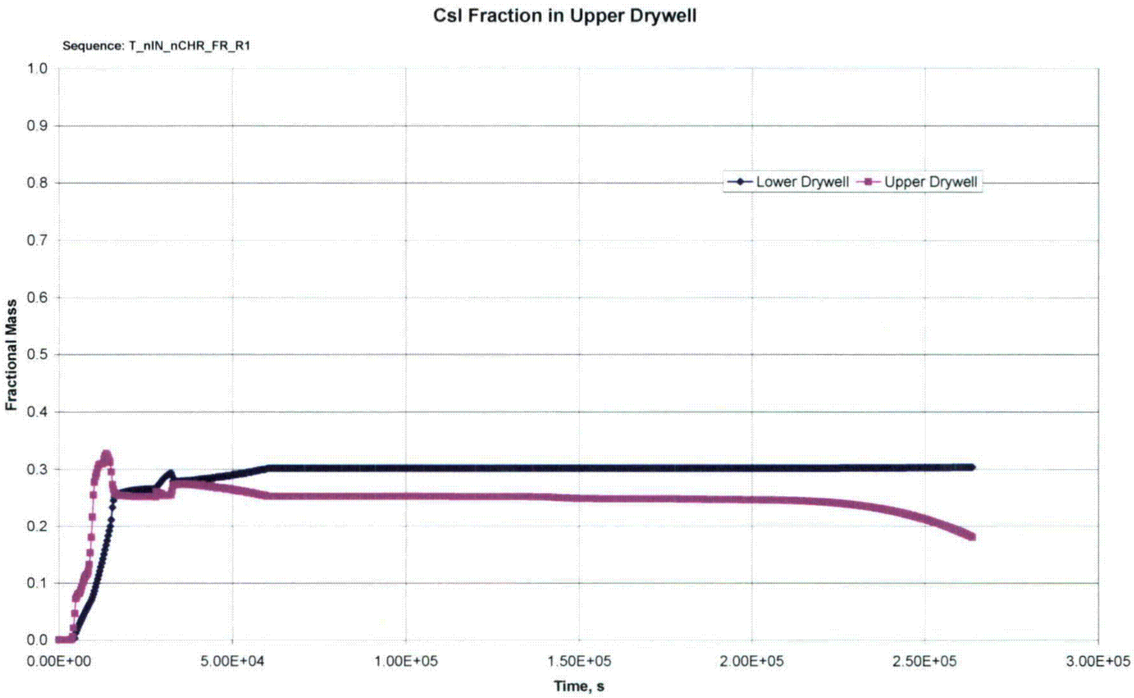

Figure 8B-2s – CDI FR CsI Fraction in Upper Drywell

8B.3 – Class II Representative Sequence (MLi_nCHR)

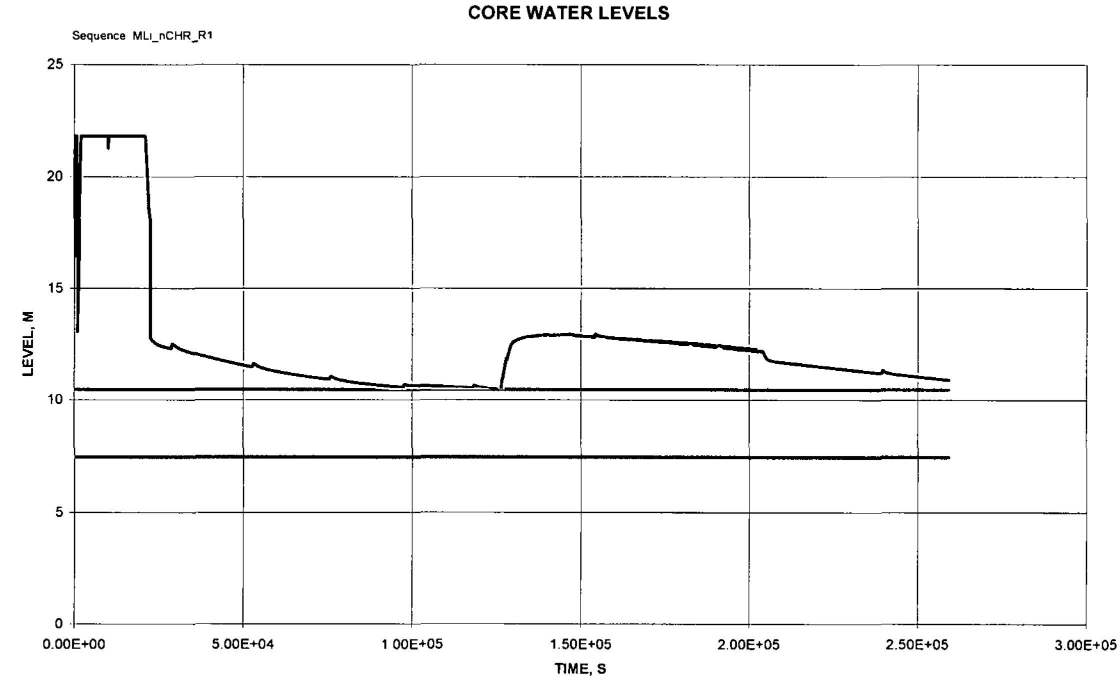

Figure 8B-3a – CDII Core Water Level

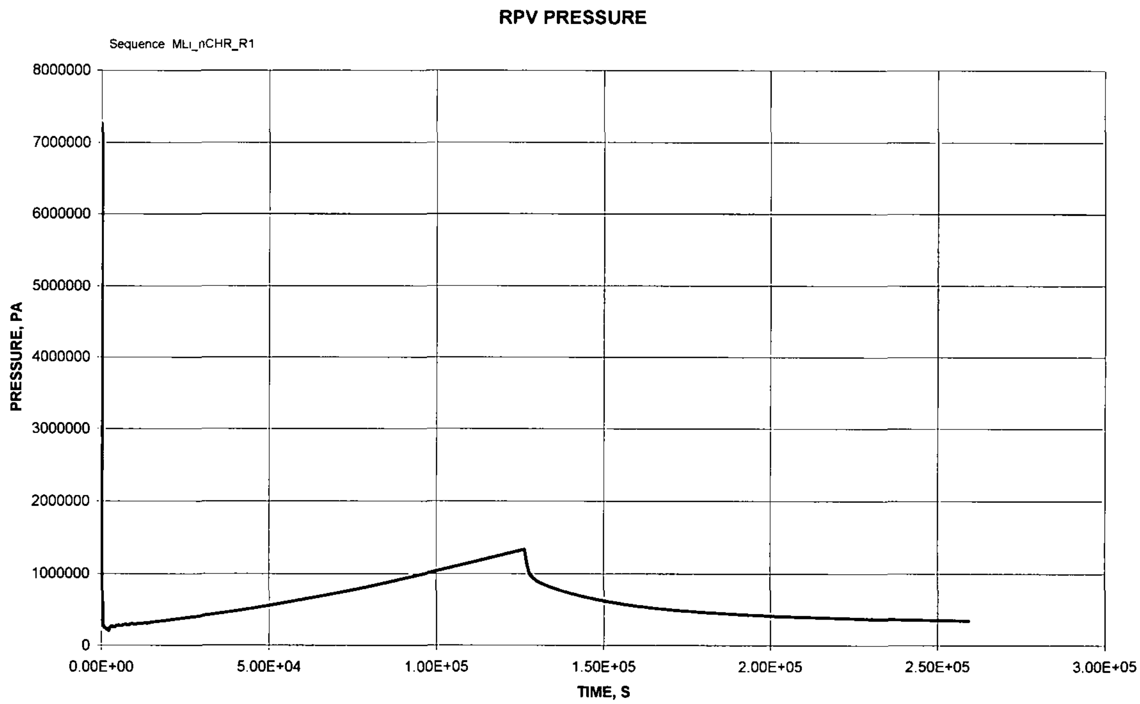

Figure 8B-3b – CDII RPV Pressure

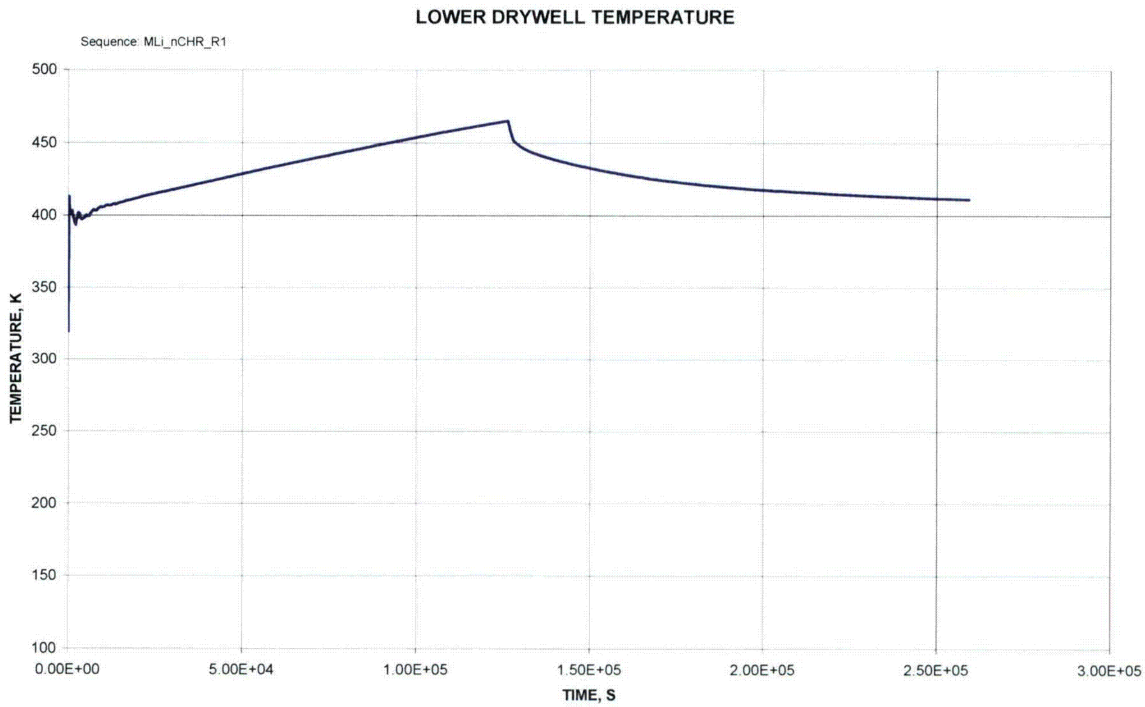

Figure 8B-3c – CDII LDW Temperature

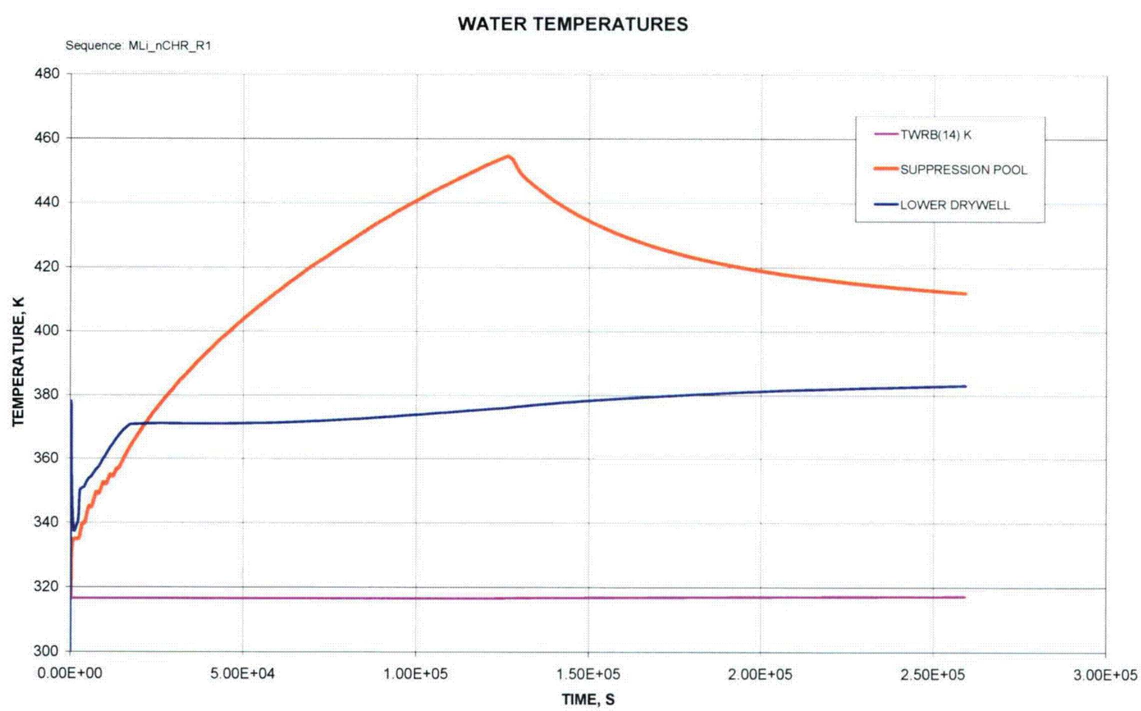

Figure 8B-3d – CDII Water Temperatures

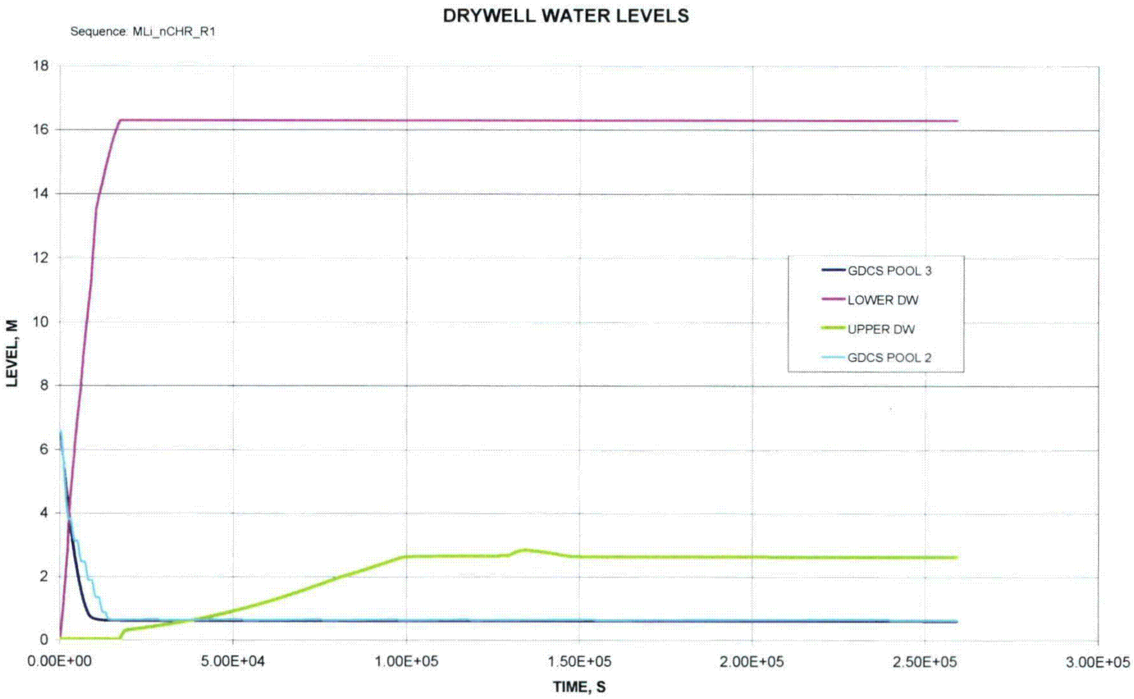

Figure 8B-3e – CDII Drywell Water Levels

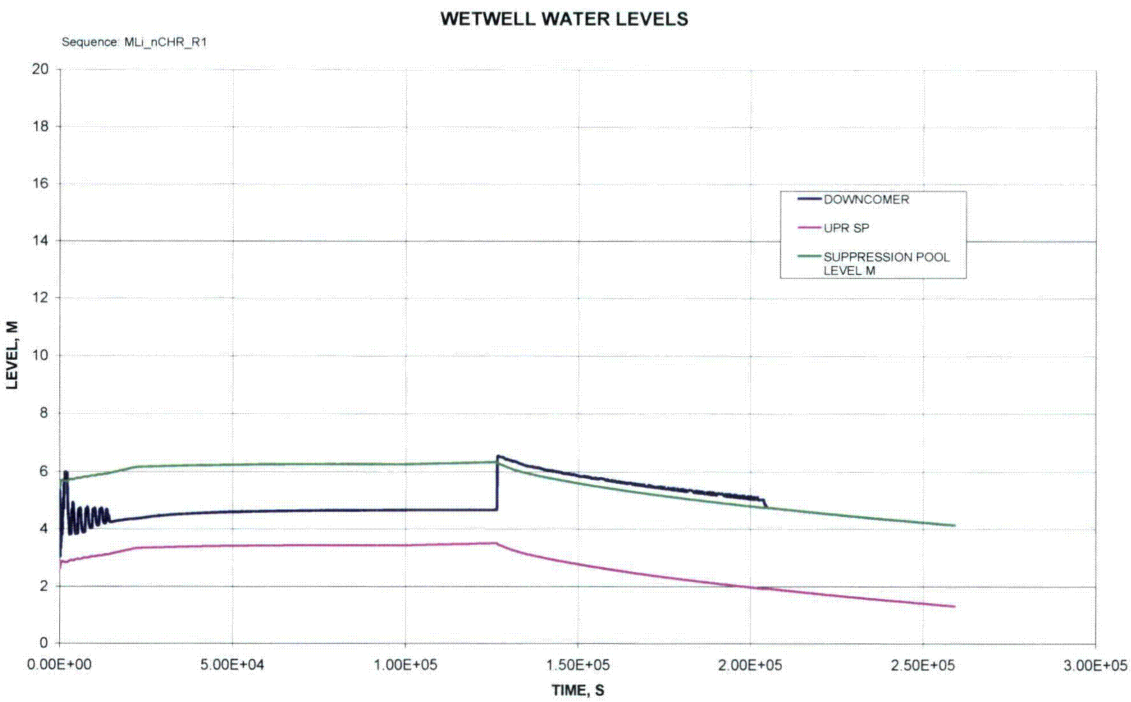

Figure 8B-3f – CDII Wetwell Water Levels

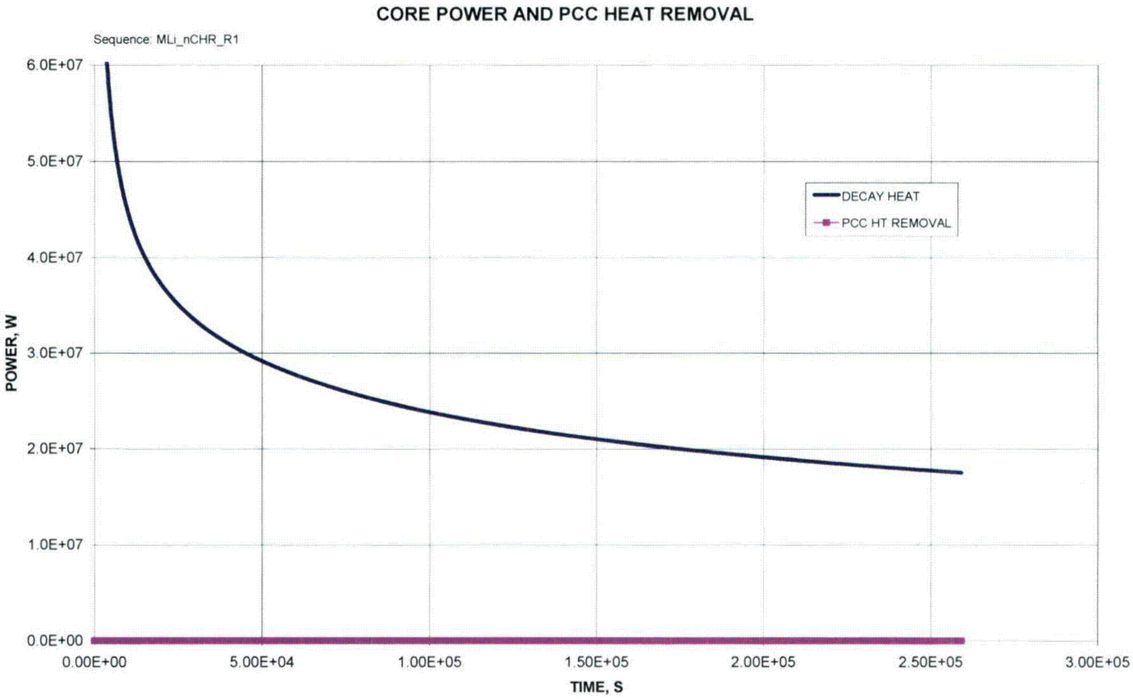

Figure 8B-3g – CDII Core Power and PCCS Heat Removal

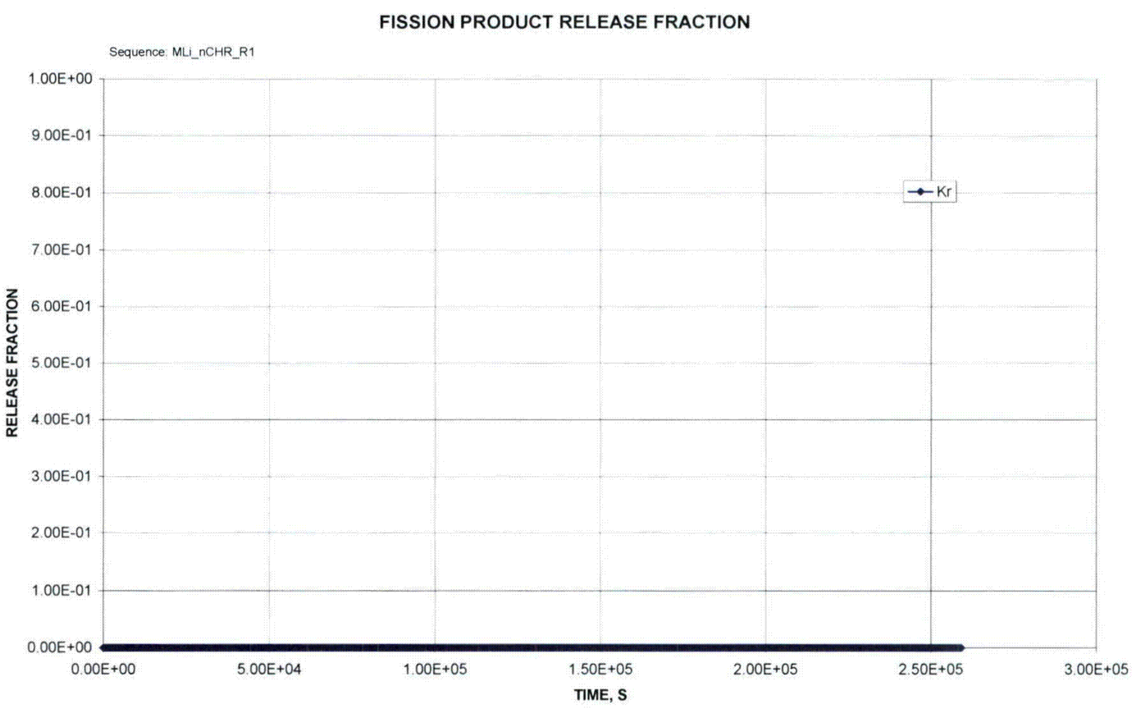

Figure 8B-3h – CDII Noble Gases Release Fraction

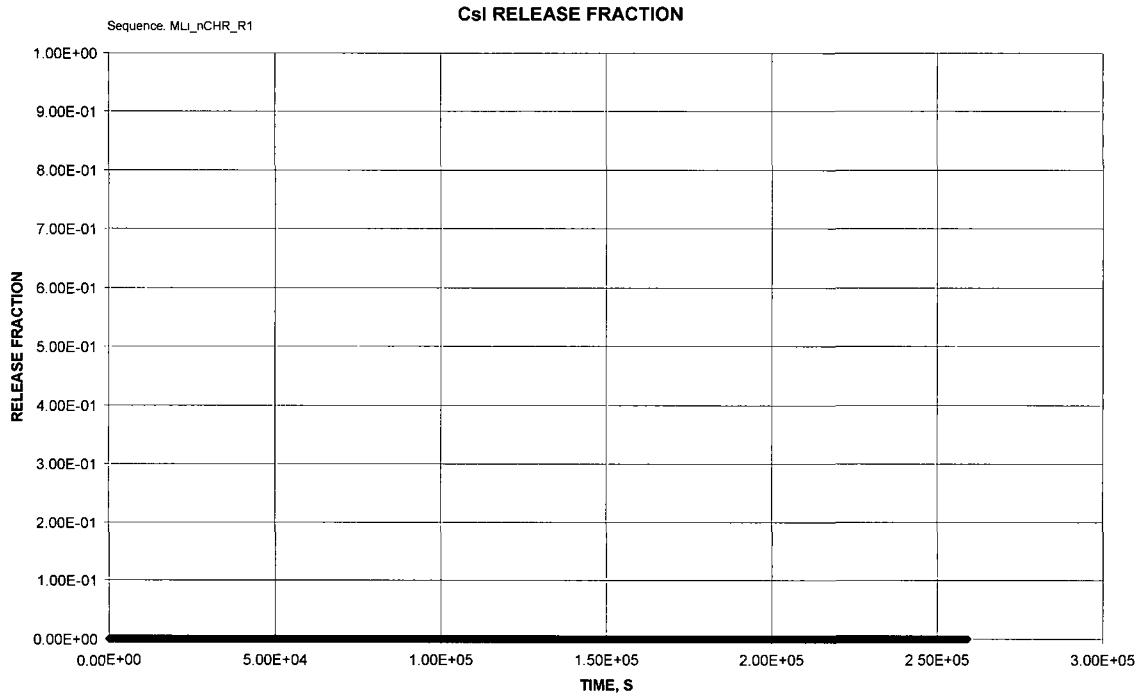

Figure 8B-3i – CDII CsI Release Fraction

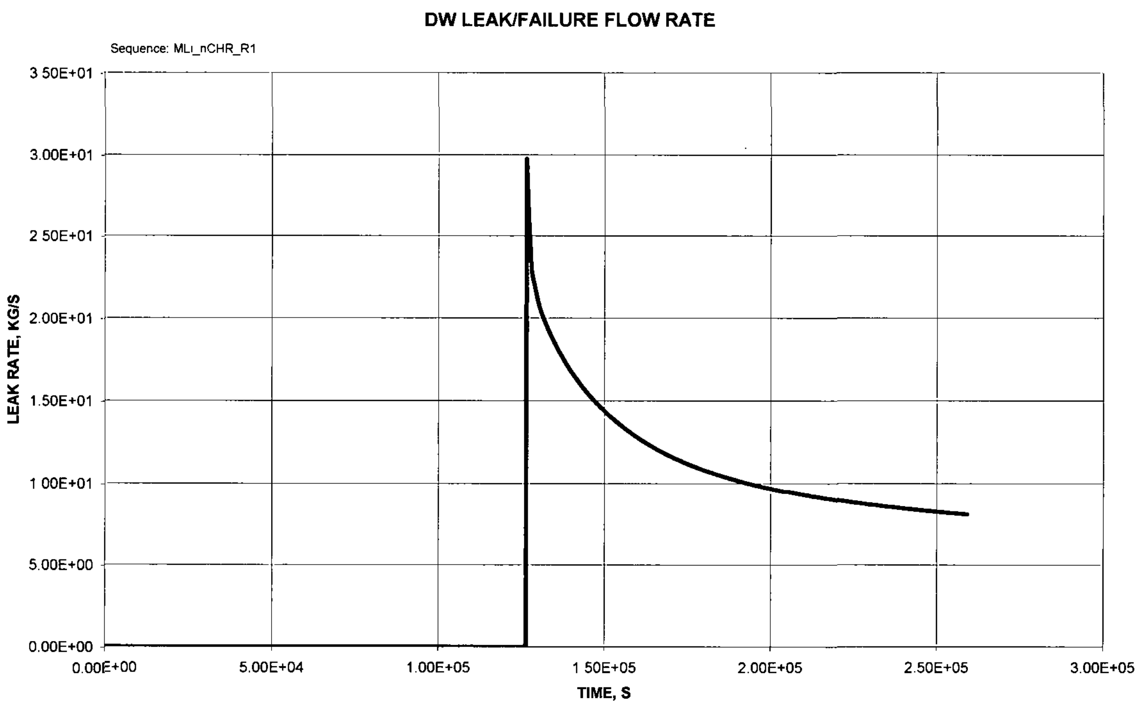

Figure 8B-3j – CDII DW Leak/Failure Flow Rate

8B.4 – Class III Representative Sequence (T_nDP_nIN_TSL)

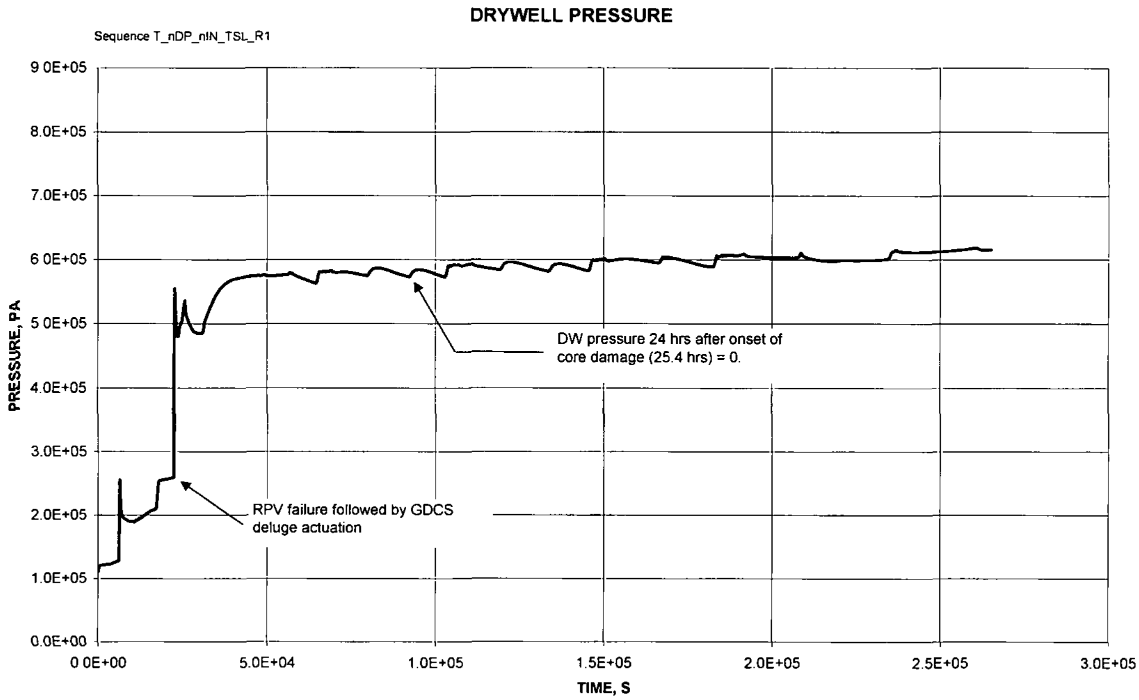

Figure 8B-4a – CDIII TSL Drywell Pressure

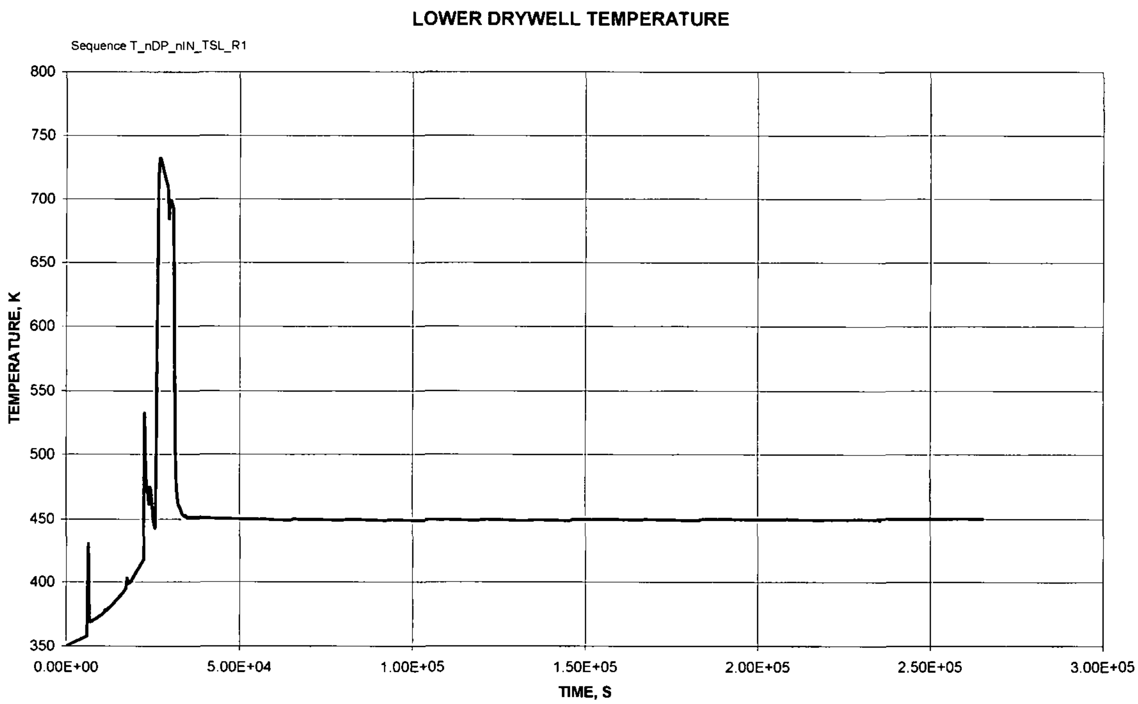

Figure 8B-4b – CDIII TSL Lower Drywell Temperature

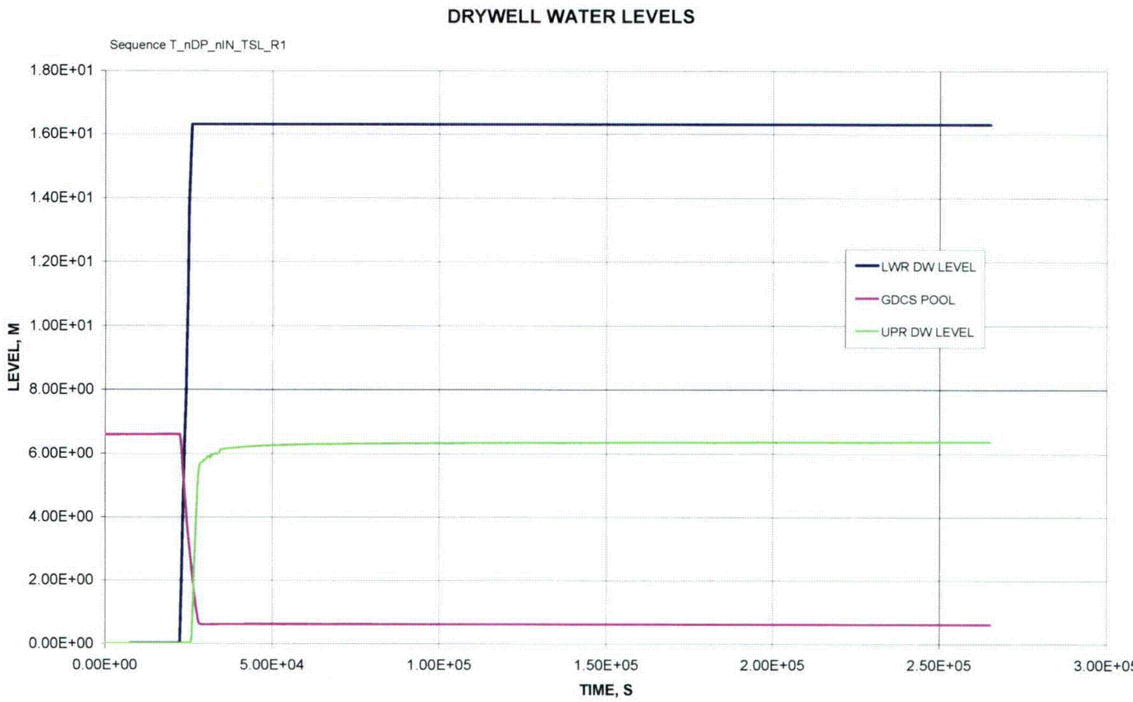

Figure 8B-4c – CDIII TSL Drywell Water Levels

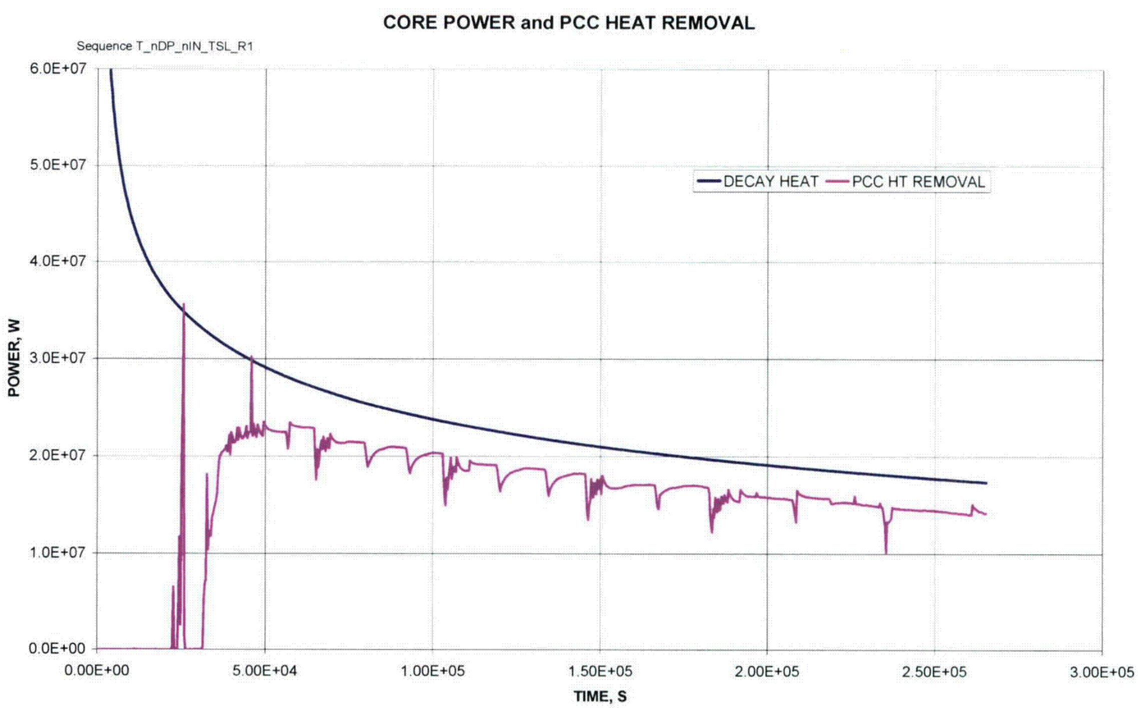

Figure 8B-4d – CDIII TSL Core Power and PCCS Heat Removal

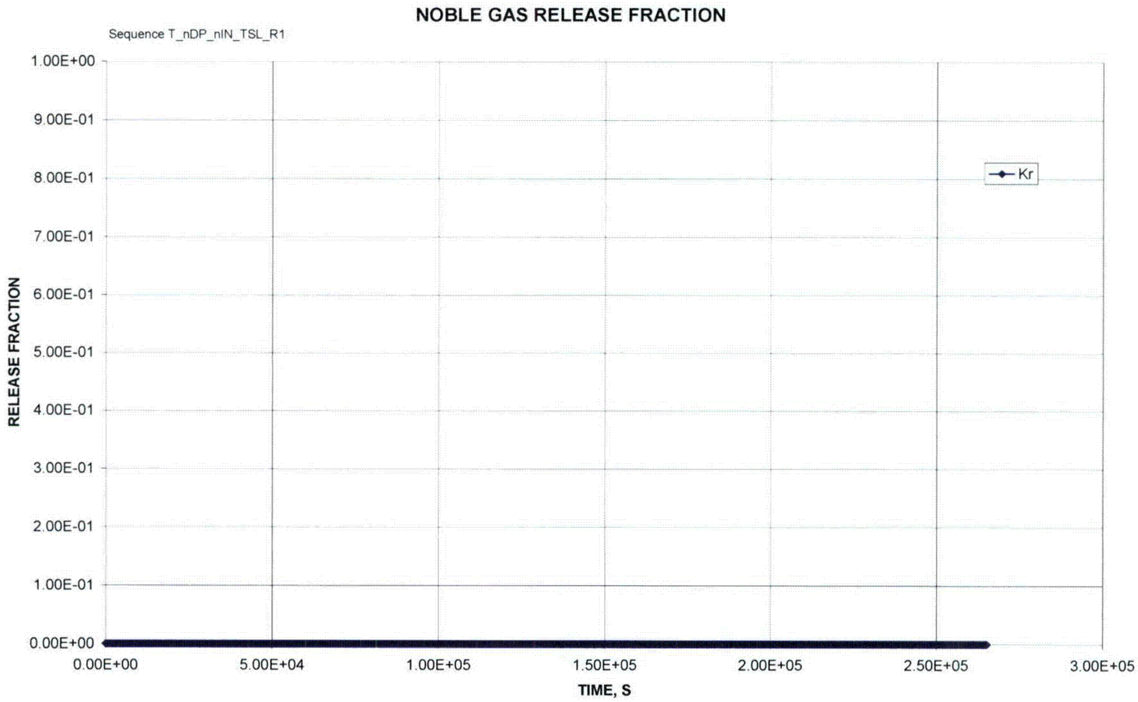

Figure 8B-4e – CDIII TSL Noble Gas Release Fraction

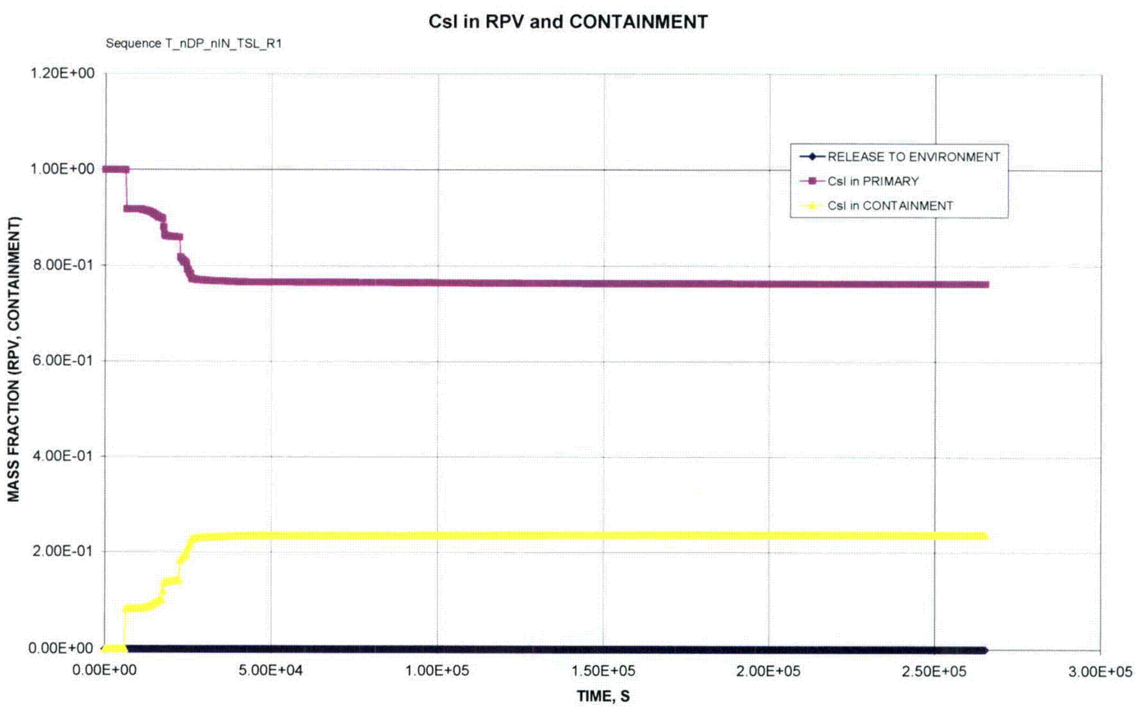

Figure 8B-4f – CDIII TSL CsI Distribution

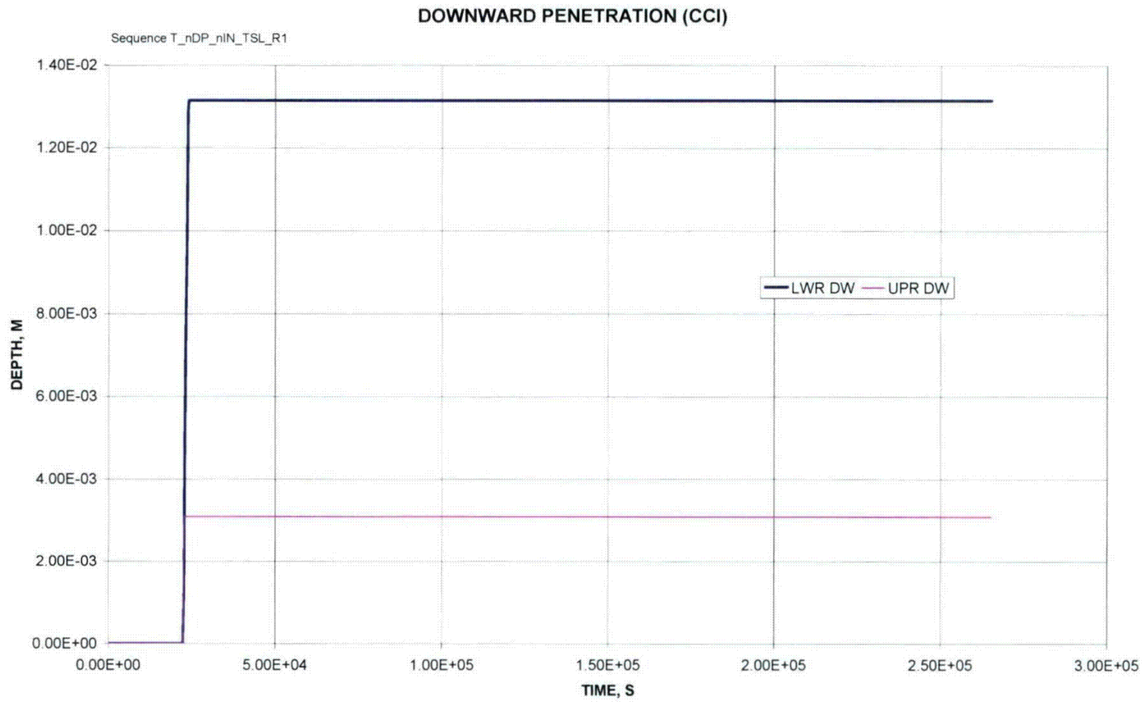

Figure 8B-4g – CDIII TSL Downward CCI Penetration

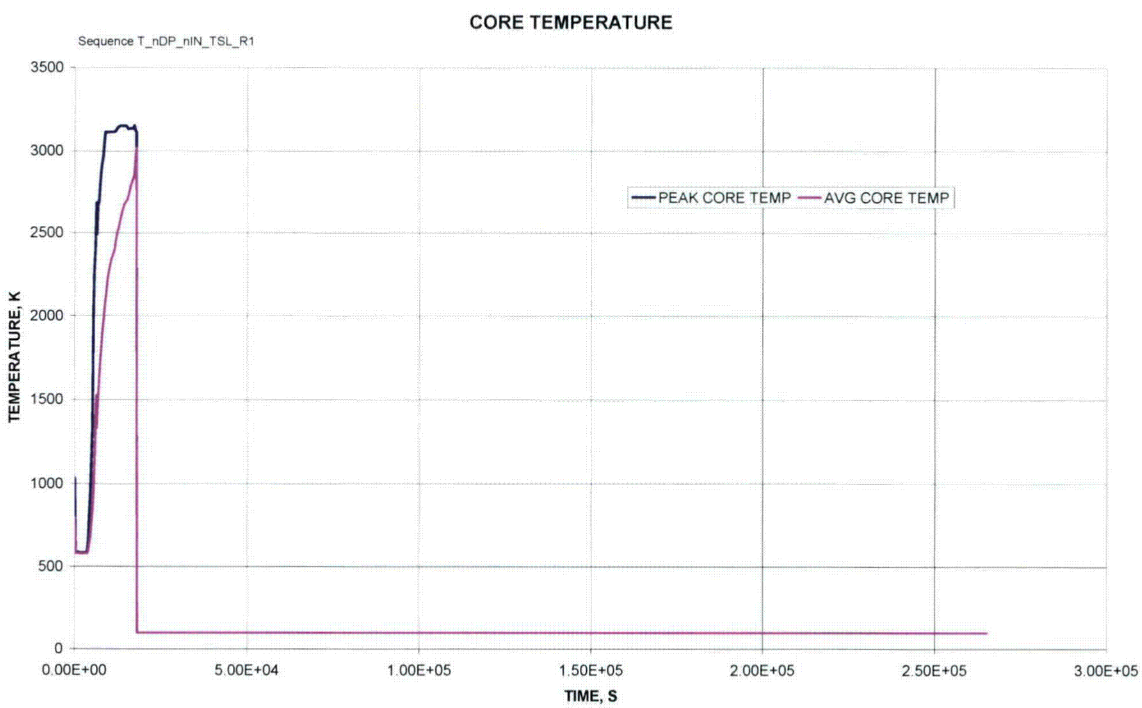

Figure 8B-4h – CDIII TSL Core Temperature

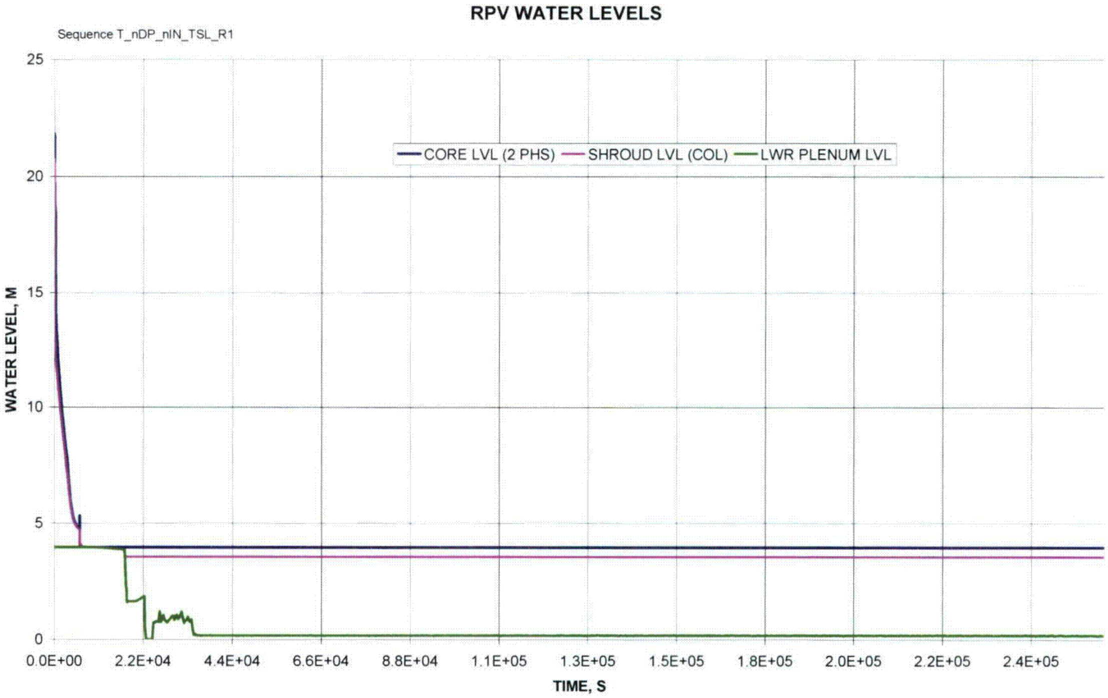

Figure 8B-4i – CDIII TSL RPV Water Levels

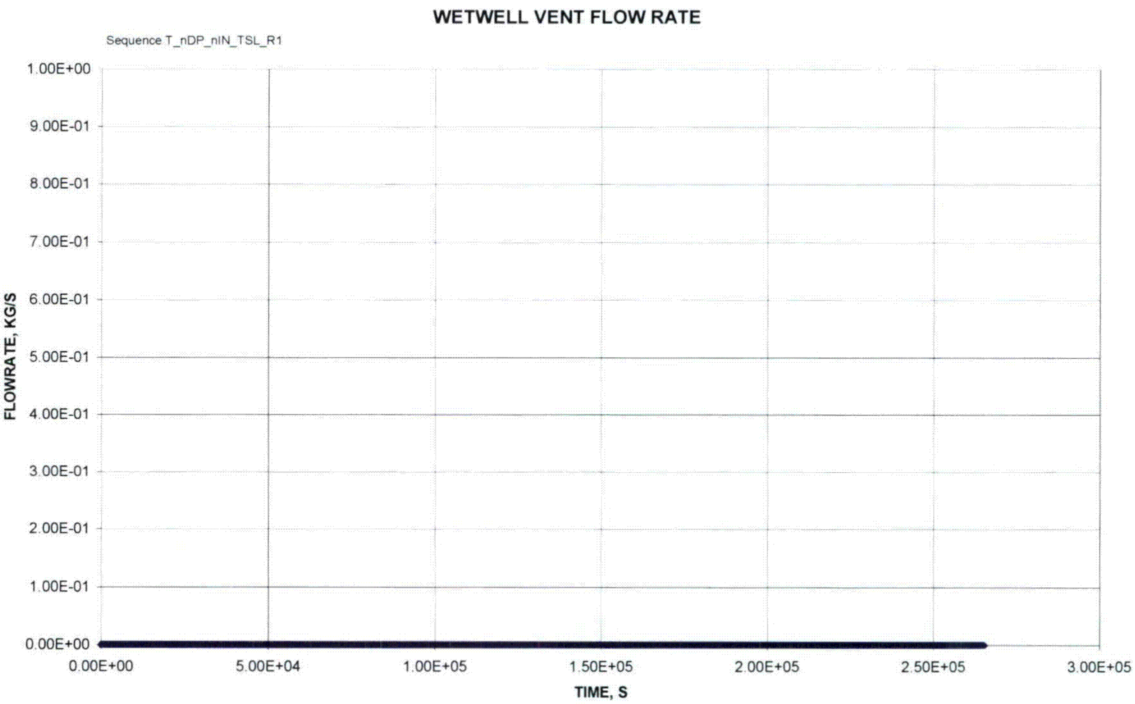

Figure 8B-4j – CDIII TSL Wetwell Vent Flow Rate

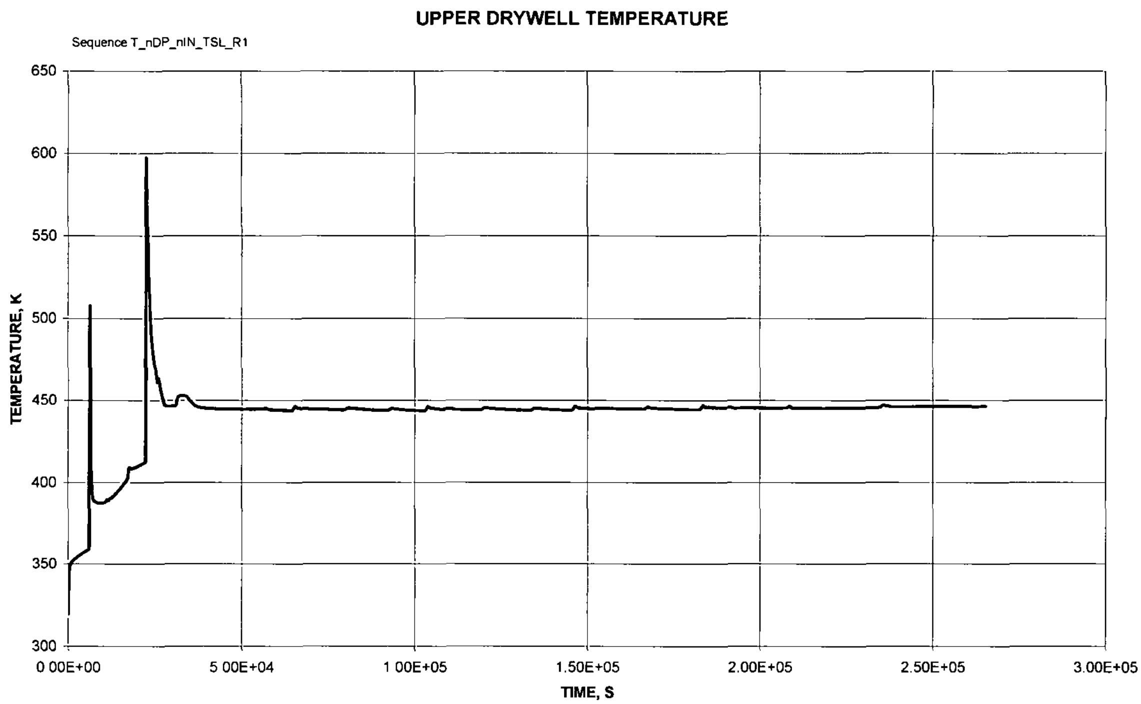

Figure 8B-4k – CDIII TSL Upper Drywell Gas Temperature

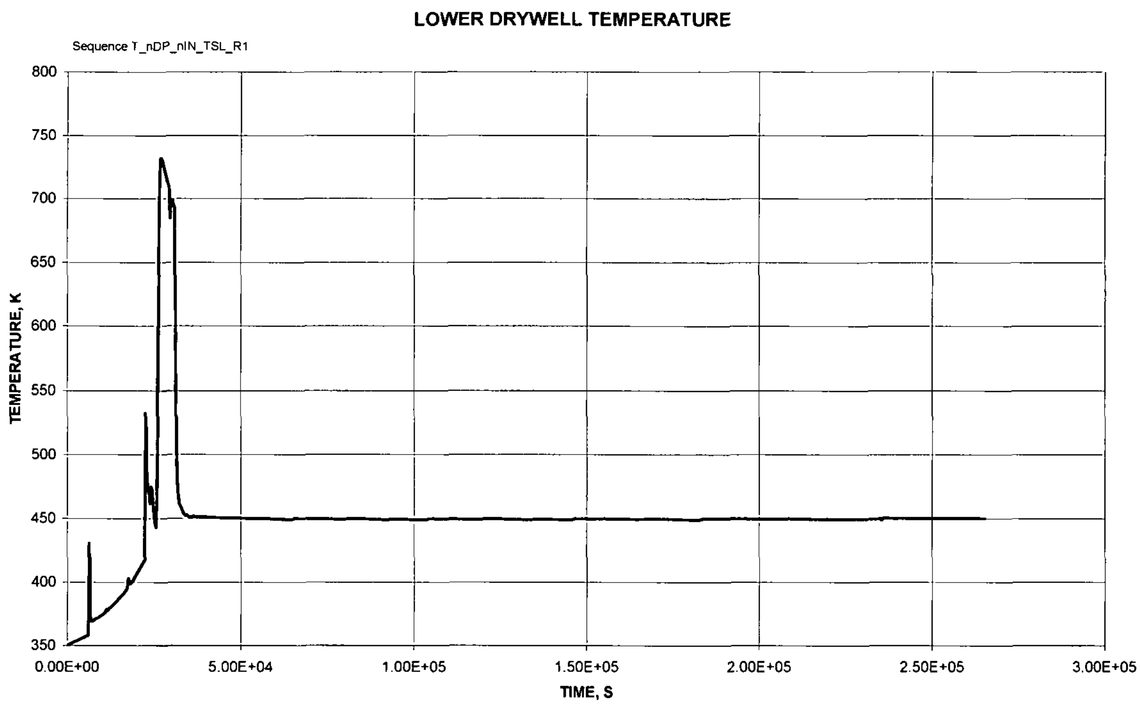

Figure 8B-4l – CDIII TSL Lower Drywell Gas Temperature

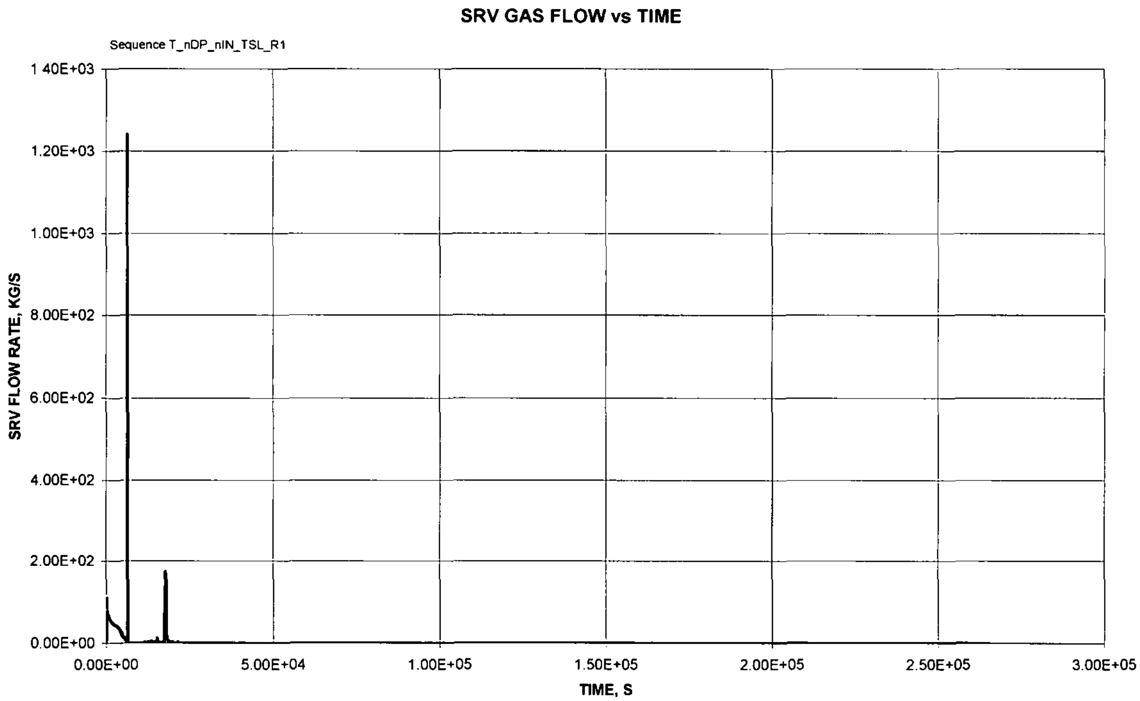

Figure 8B-4m – CDIII TSL SRV Gas Flow Rate

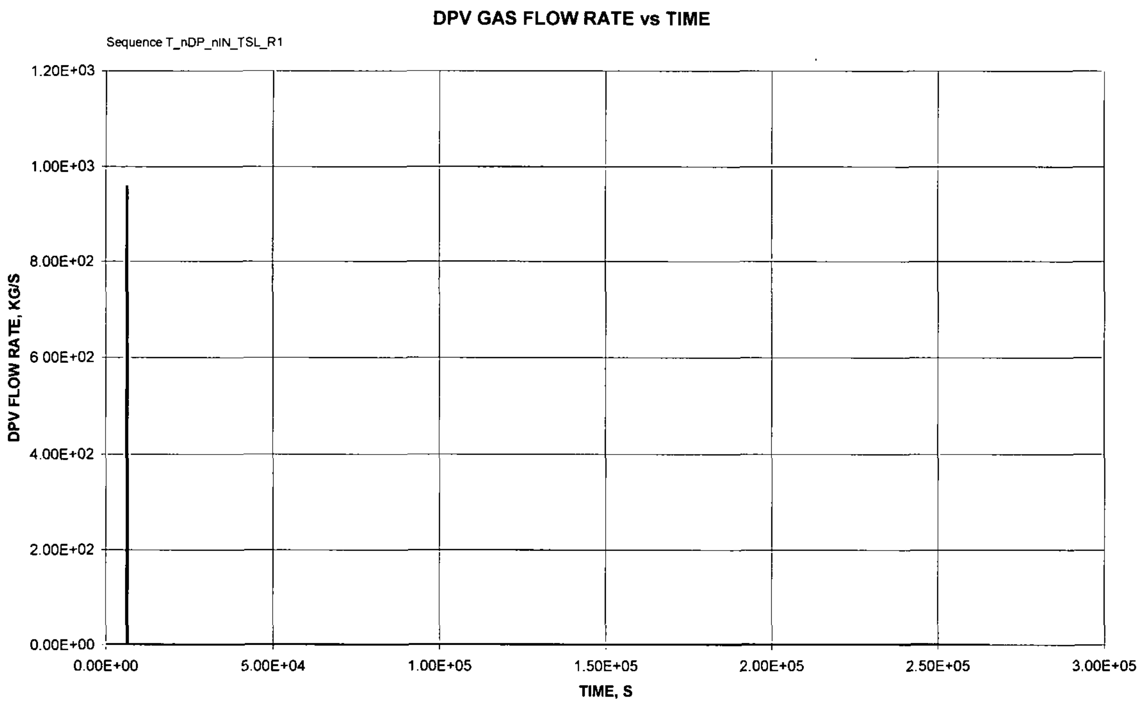

Figure 8B-4n – CDIII TSL DPV Gas Flow Rate

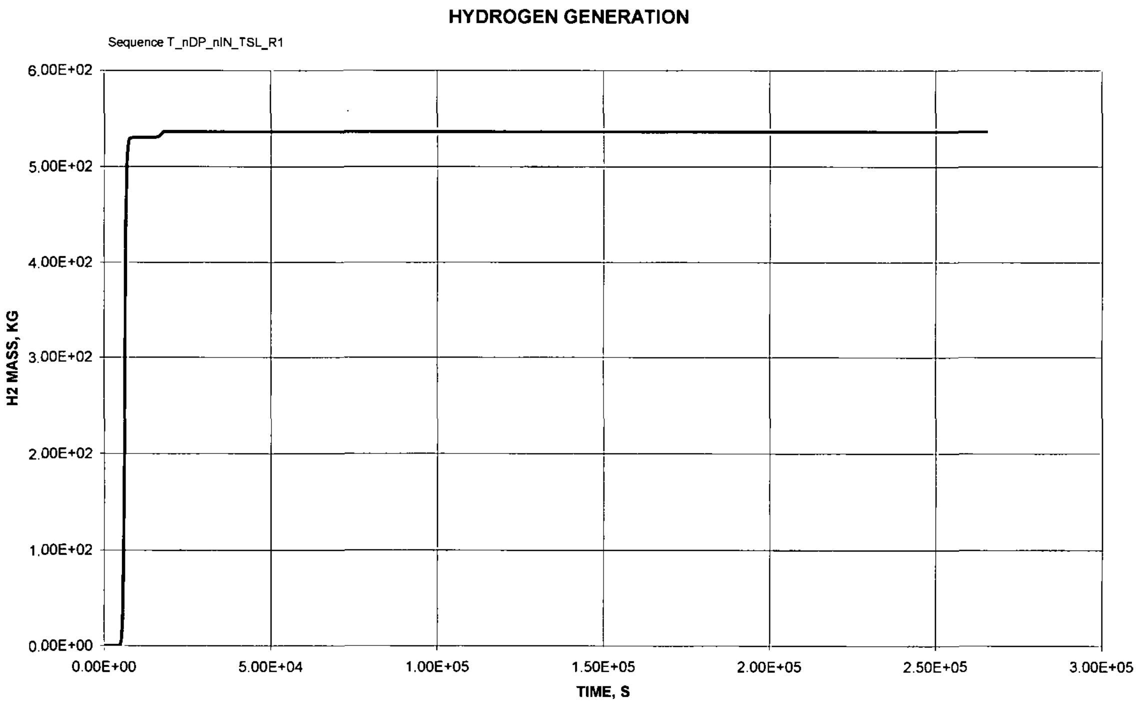

Figure 8B-4o – CDIII TSL Hydrogen Generation

8B.5 – Class III Representative Sequence (T_nDP_nIN_nCHR_FR)

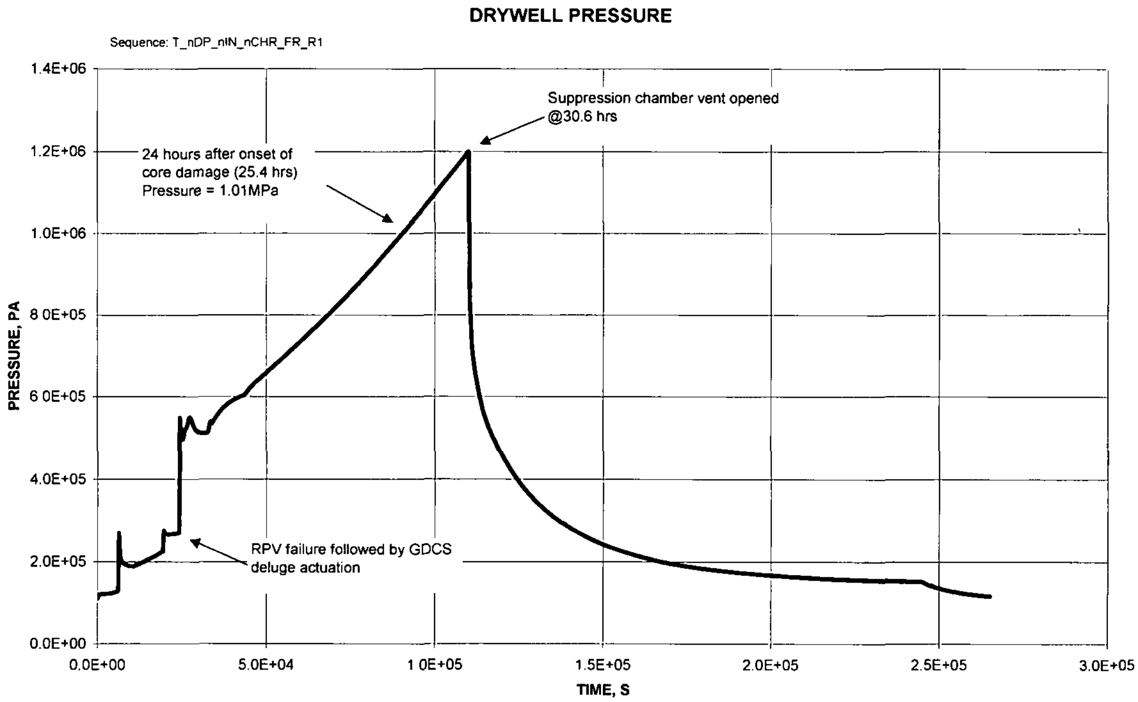

Figure 8B-5a – CDIII FR Drywell Pressure

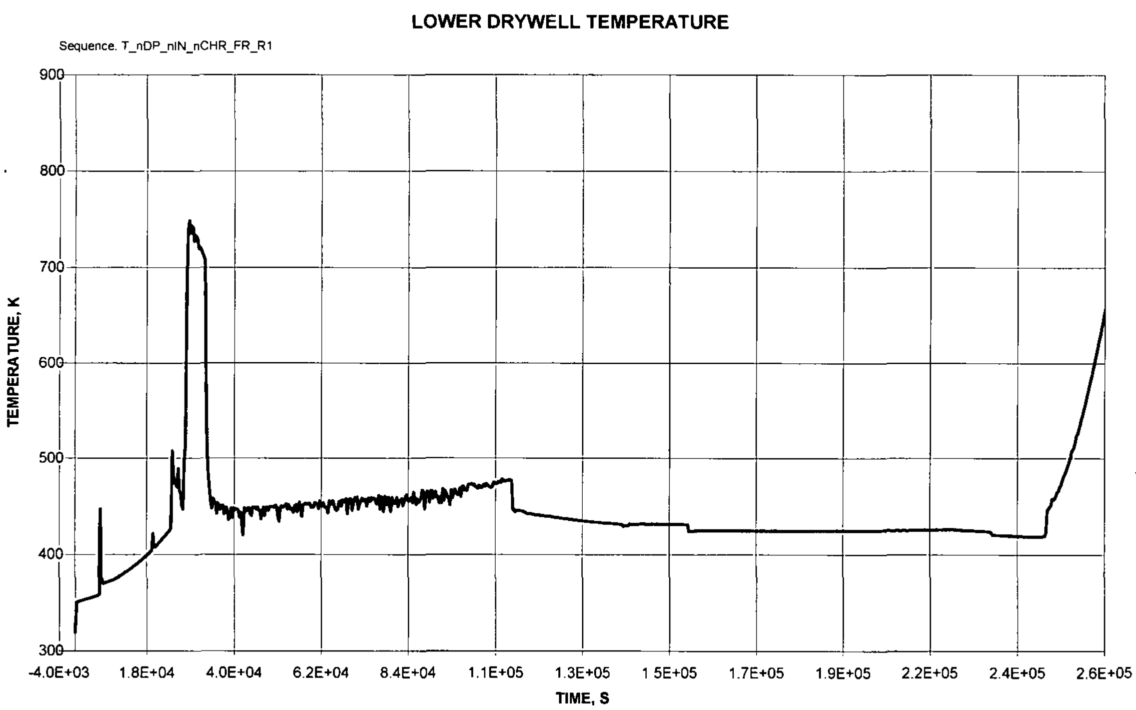

Figure 8B-5b – CDIII FR Lower Drywell Temperature

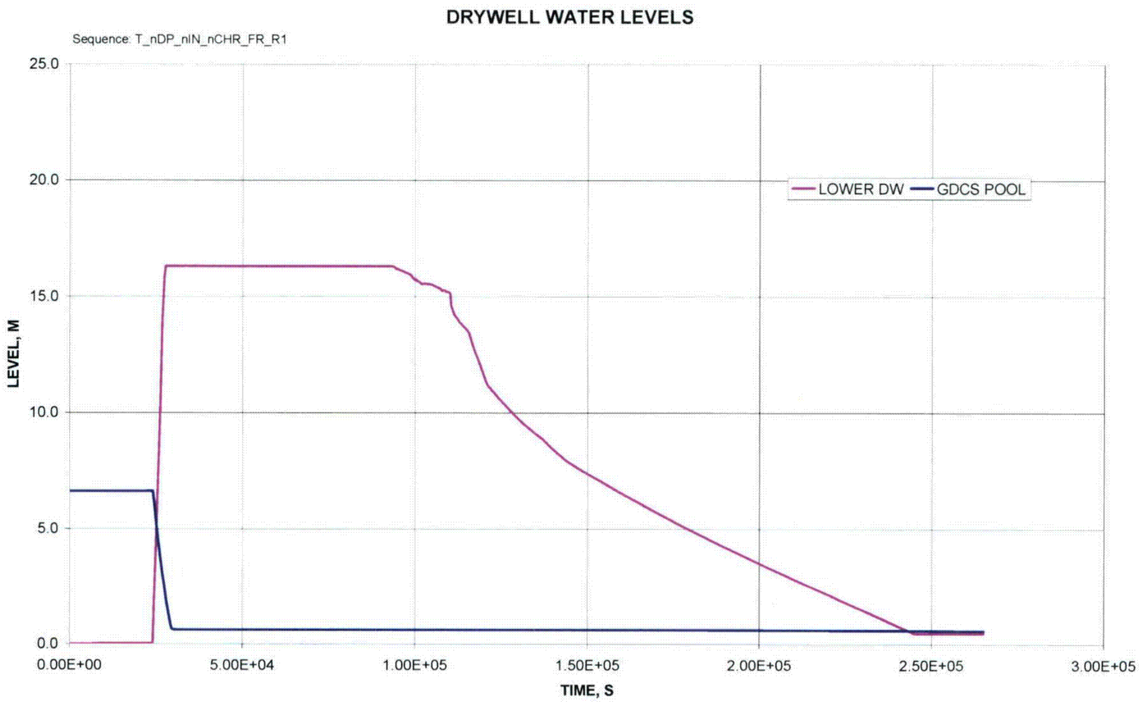

Figure 8B-5c – CDIII FR Drywell Water Levels

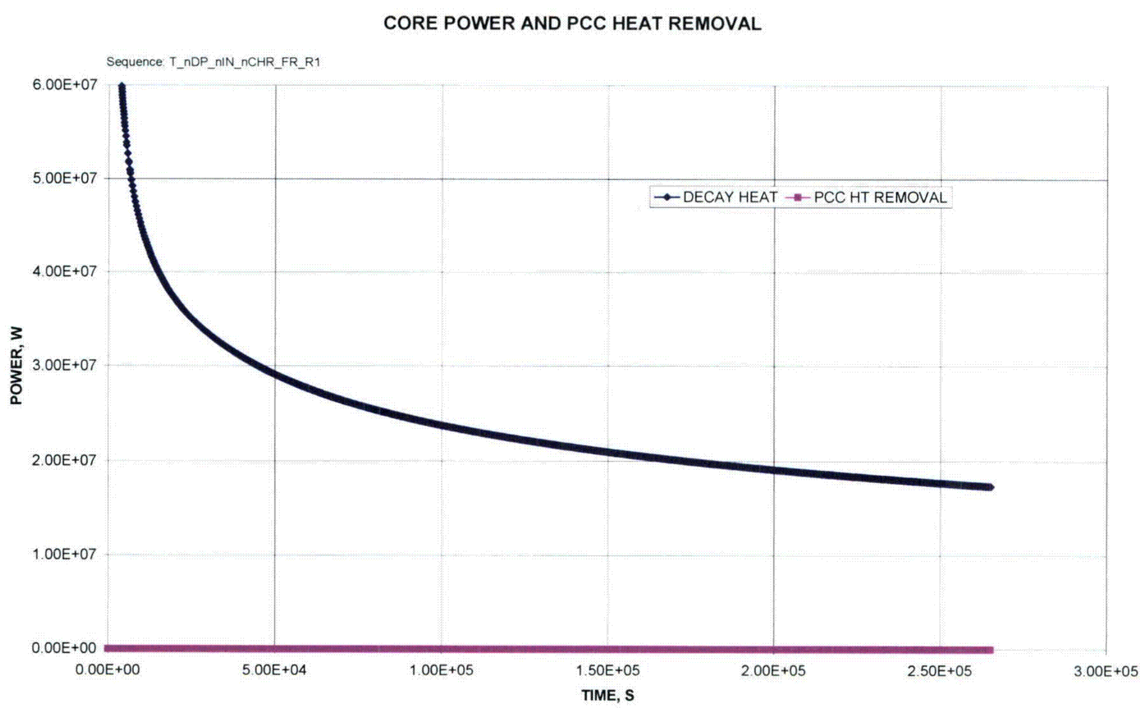

Figure 8B-5d – CDIII FR Core Power and PCCS Heat Removal

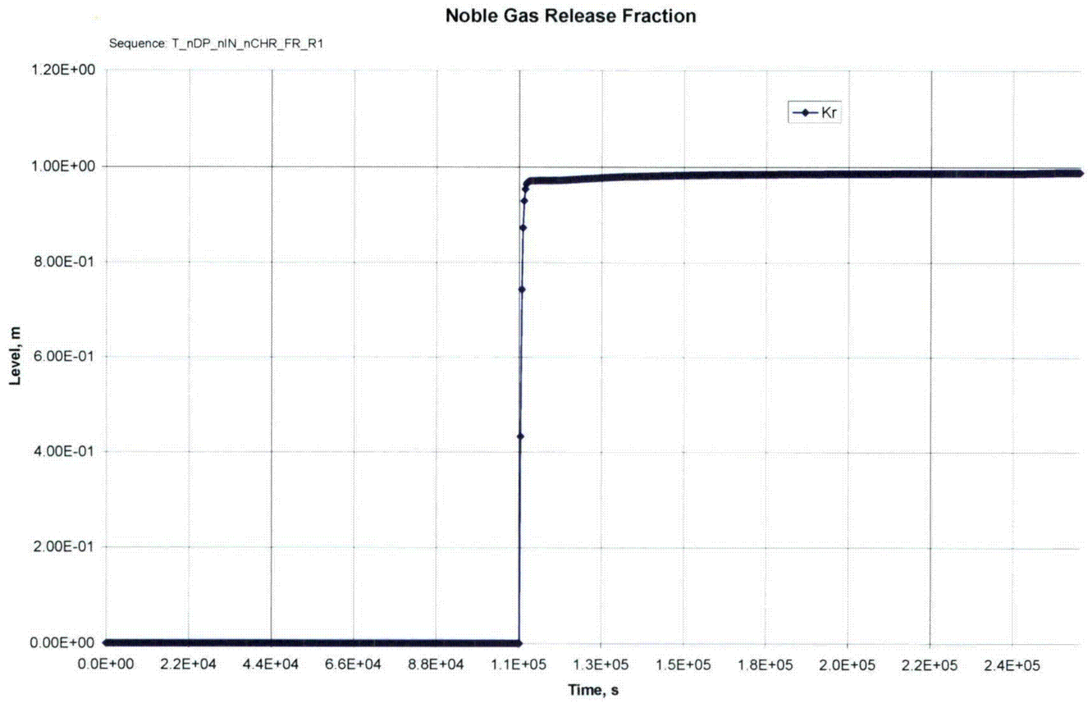

Figure 8B-5e – CDIII FR Noble Gas Release Fraction

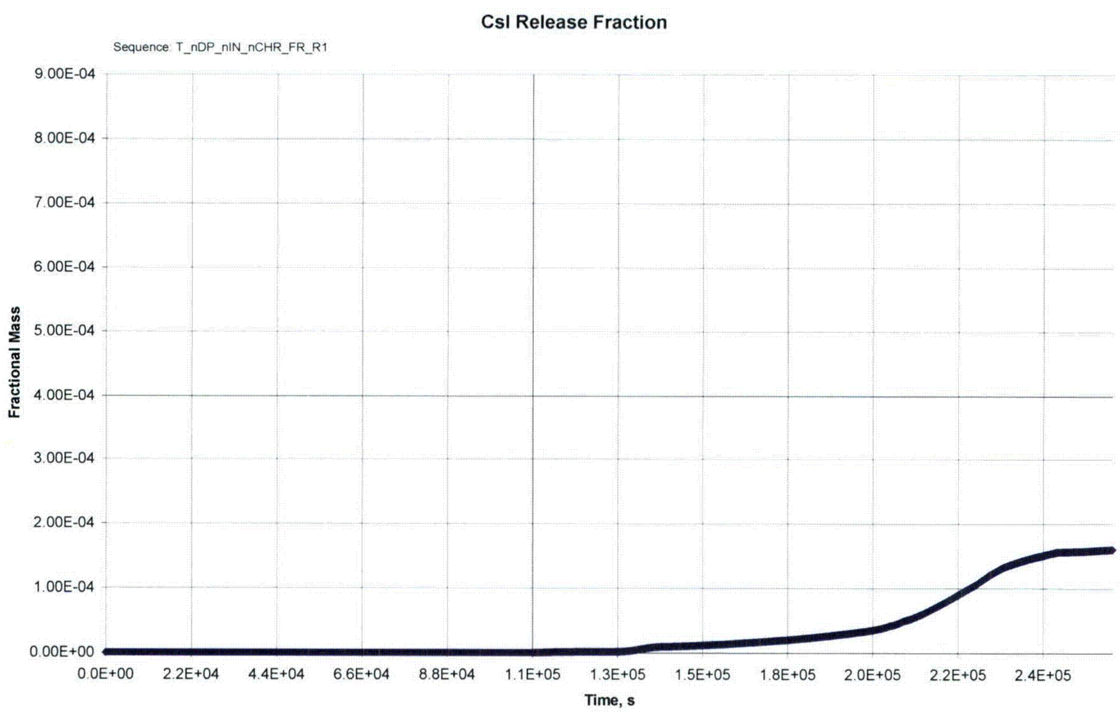

Figure 8B-5f – CDIII FR CsI Release Fraction

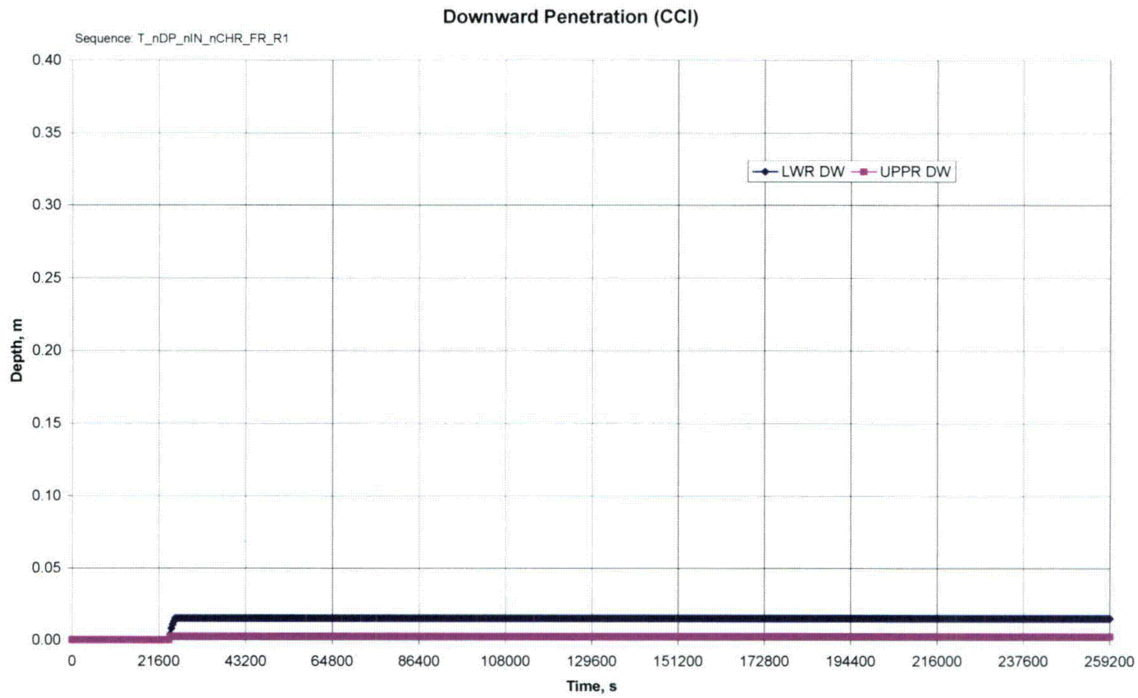

Figure 8B-5g – CDIII FR CCI Penetration

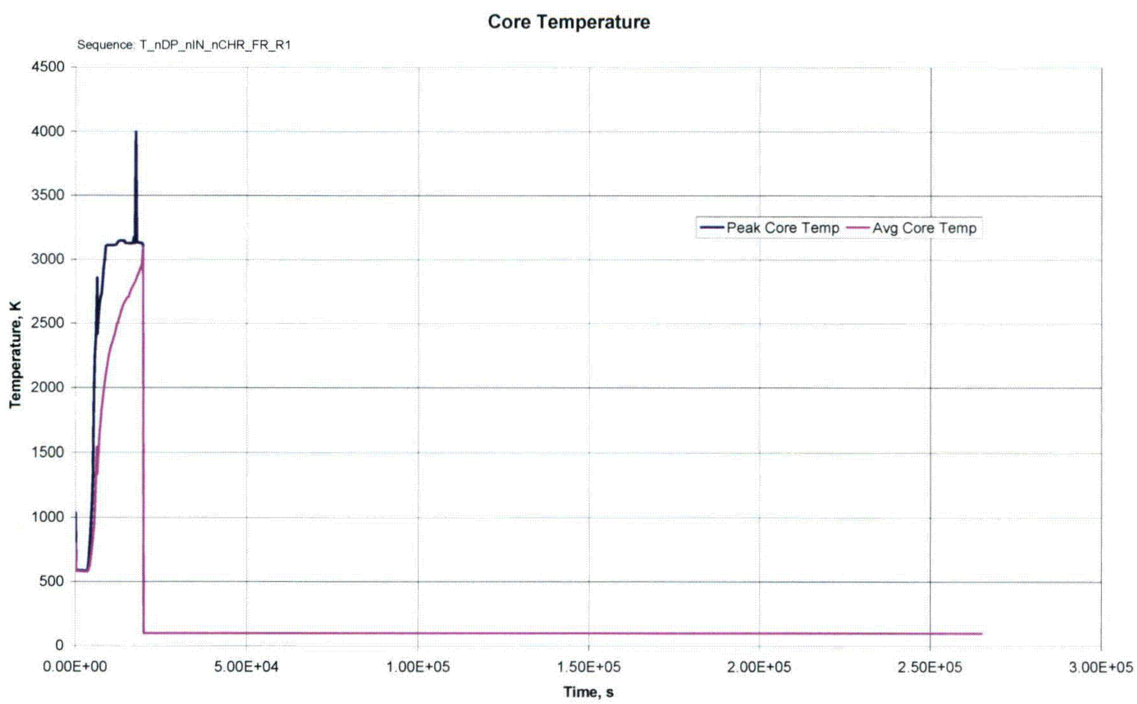

Figure 8B-5h – CDIII FR Core Temperature

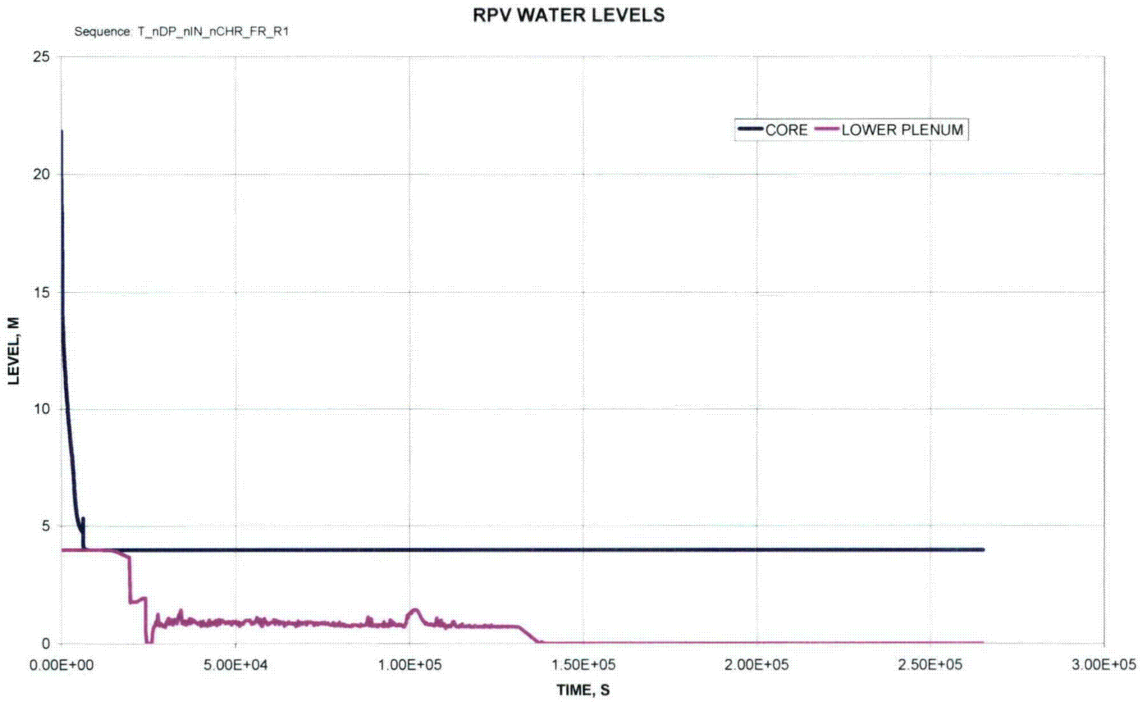

Figure 8B-5i – CDIII FR RPV Water Levels

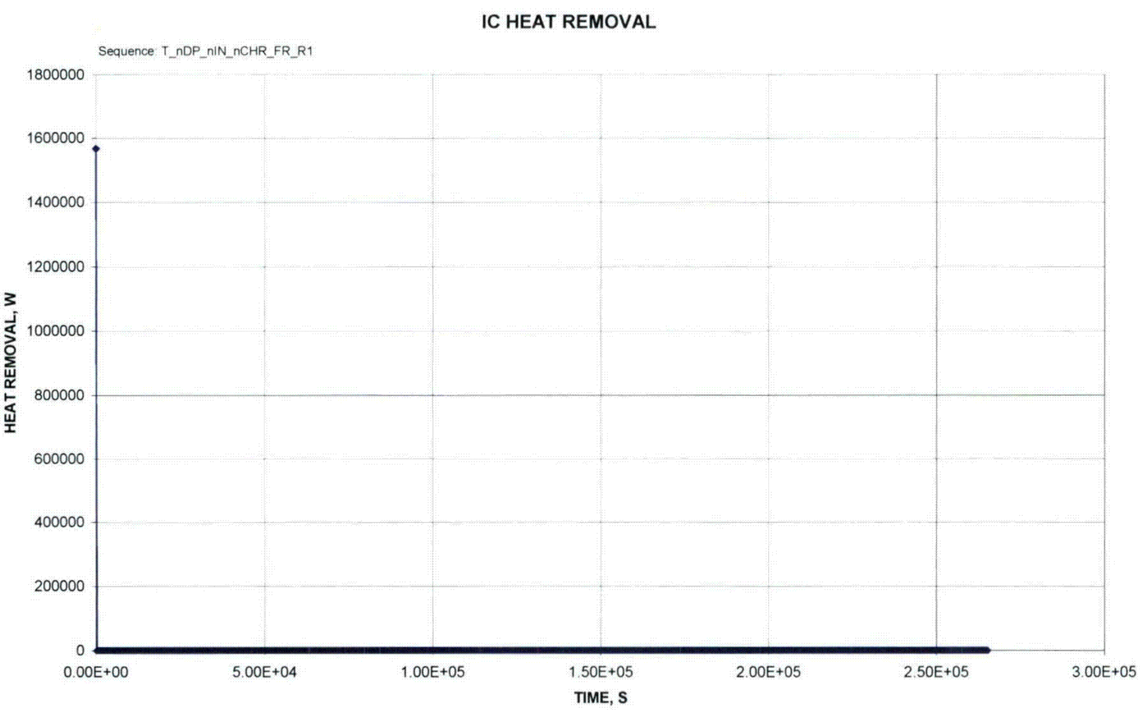

Figure 8B-5j – CDIII FR ICS Heat Removal

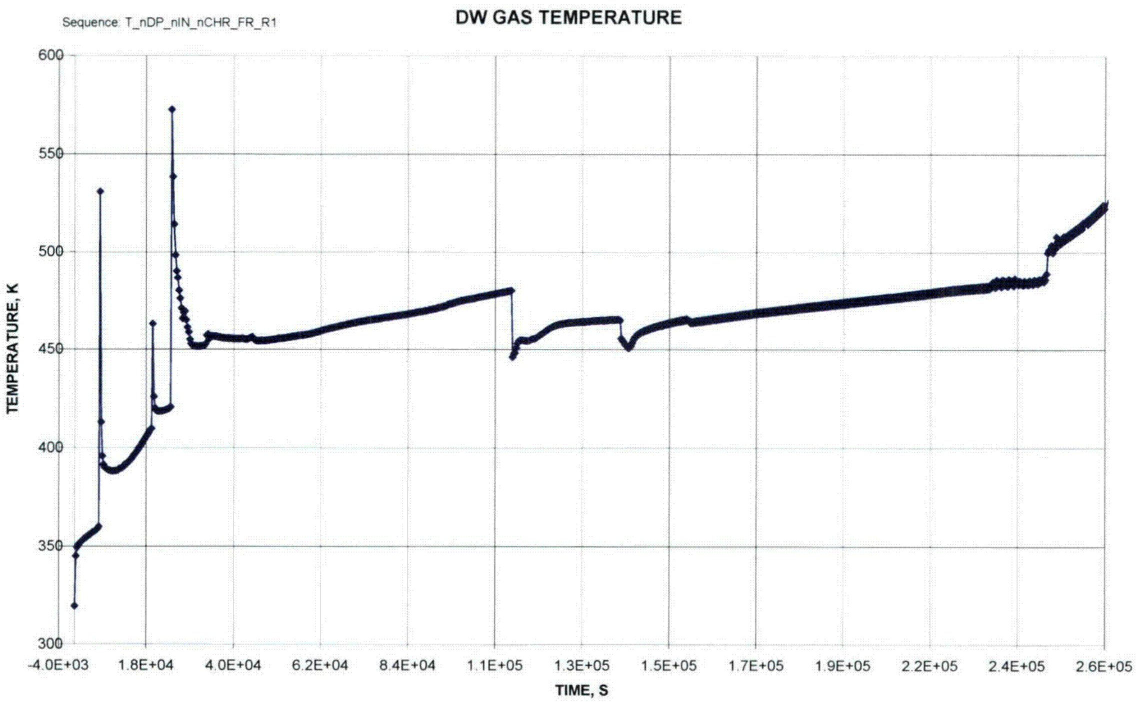

Figure 8B-5k – CDIII FR Drywell Temperature

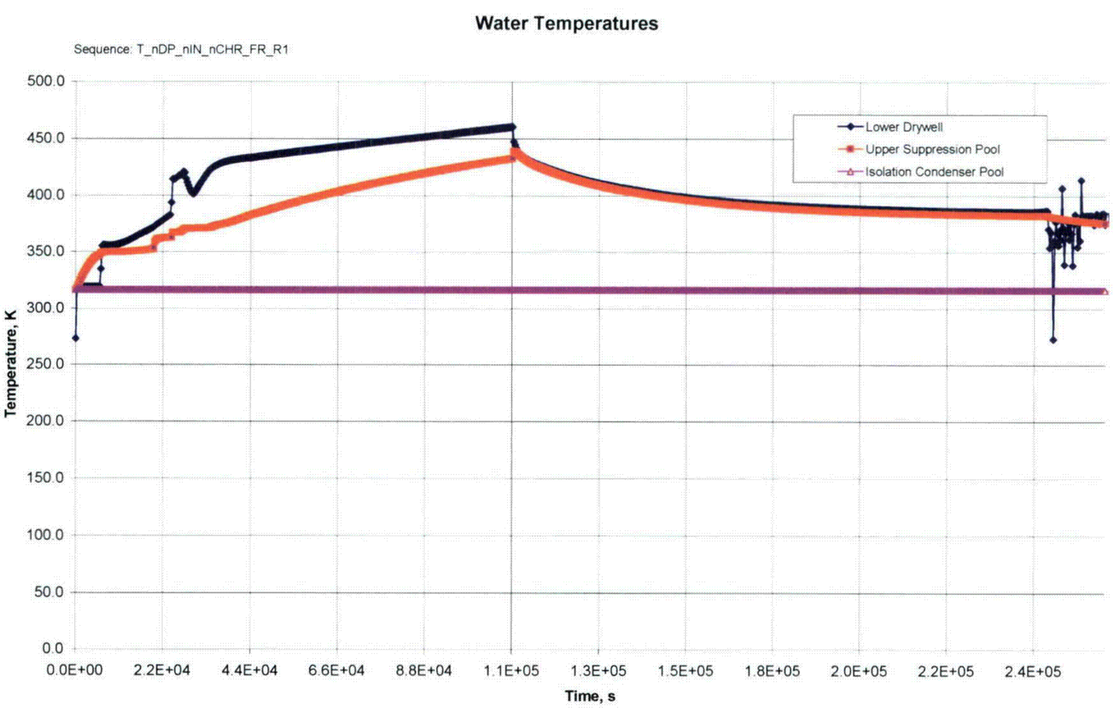

Figure 8B-5l – CDIII FR Water Temperatures

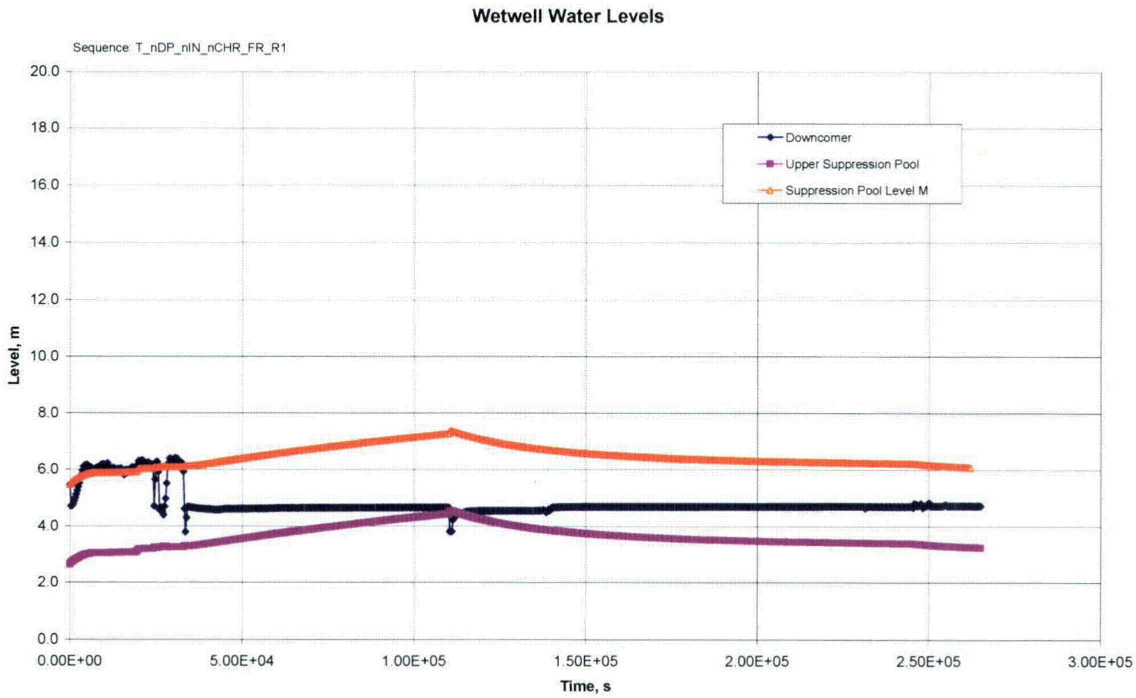

Figure 8B-5m – CDIII FR Wetwell Water Levels

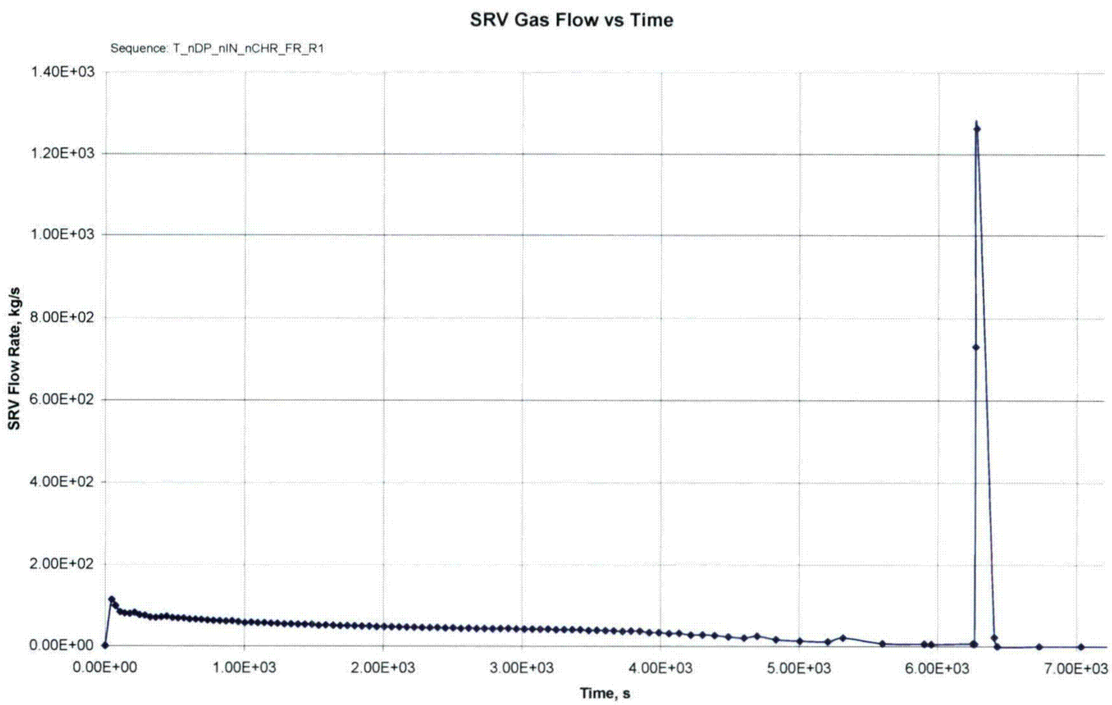

Figure 8B-5n – CDIII FR SRV Flow Rate

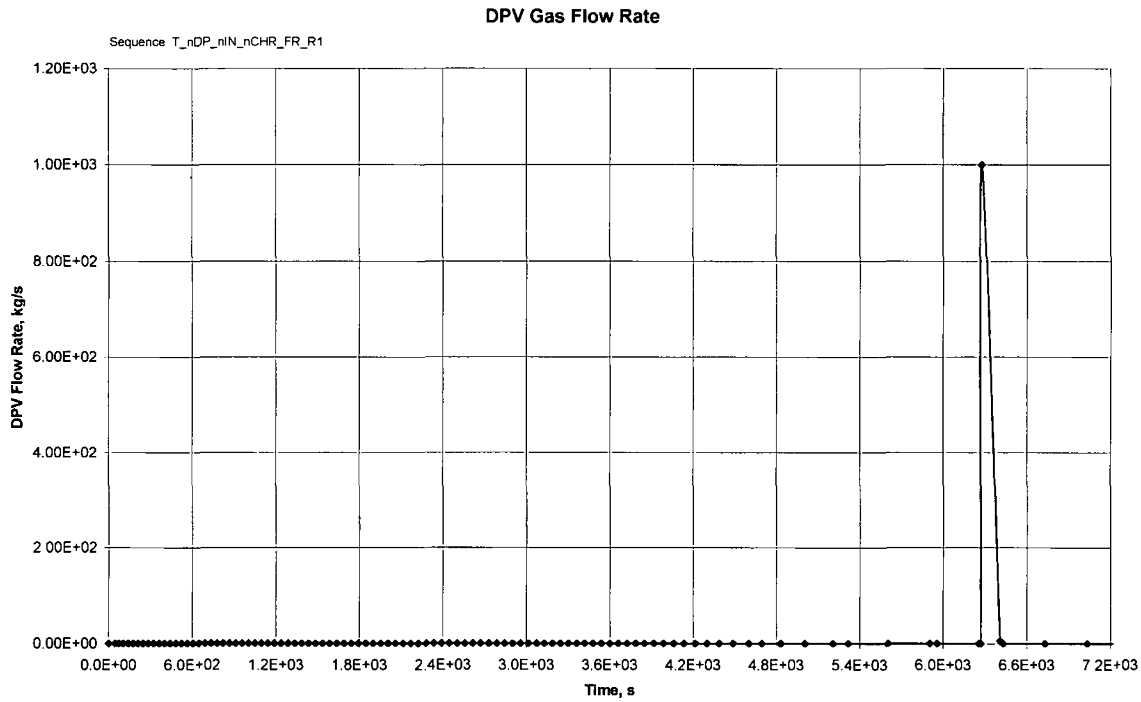

Figure 8B-5o – CDIII FR DPV Flow Rate

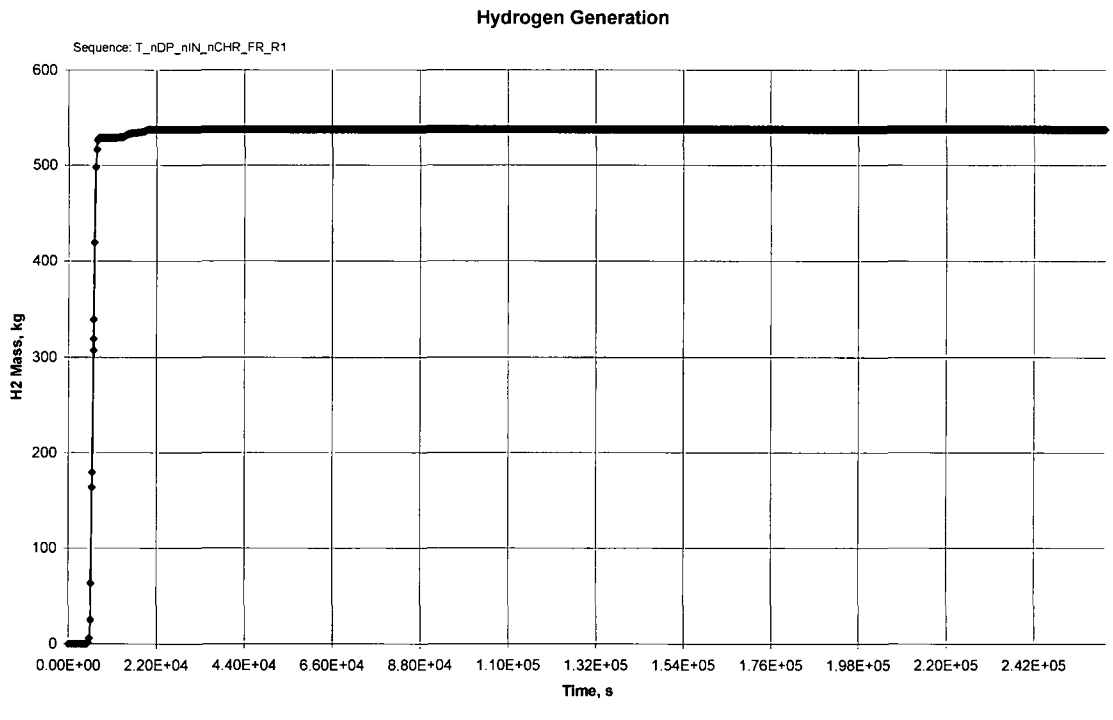

Figure 8B-5p – CDIII FR Core Hydrogen Generation

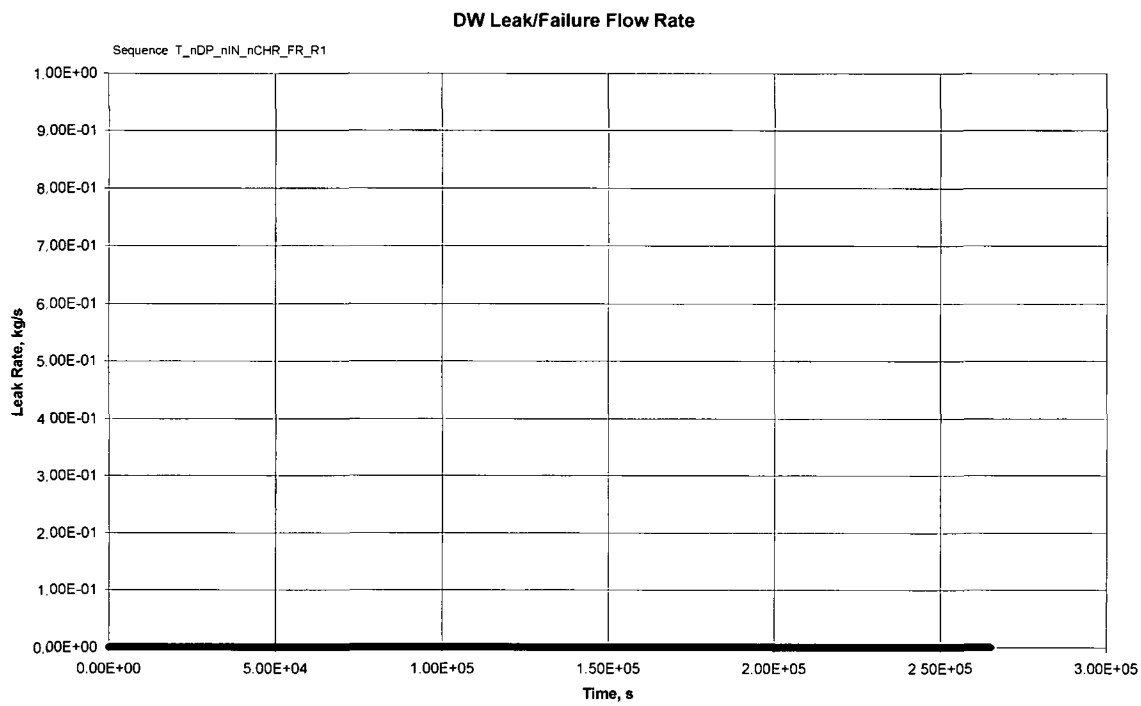

Figure 8B-5q – CDIII FR Drywell Leak/Failure Flow Rate

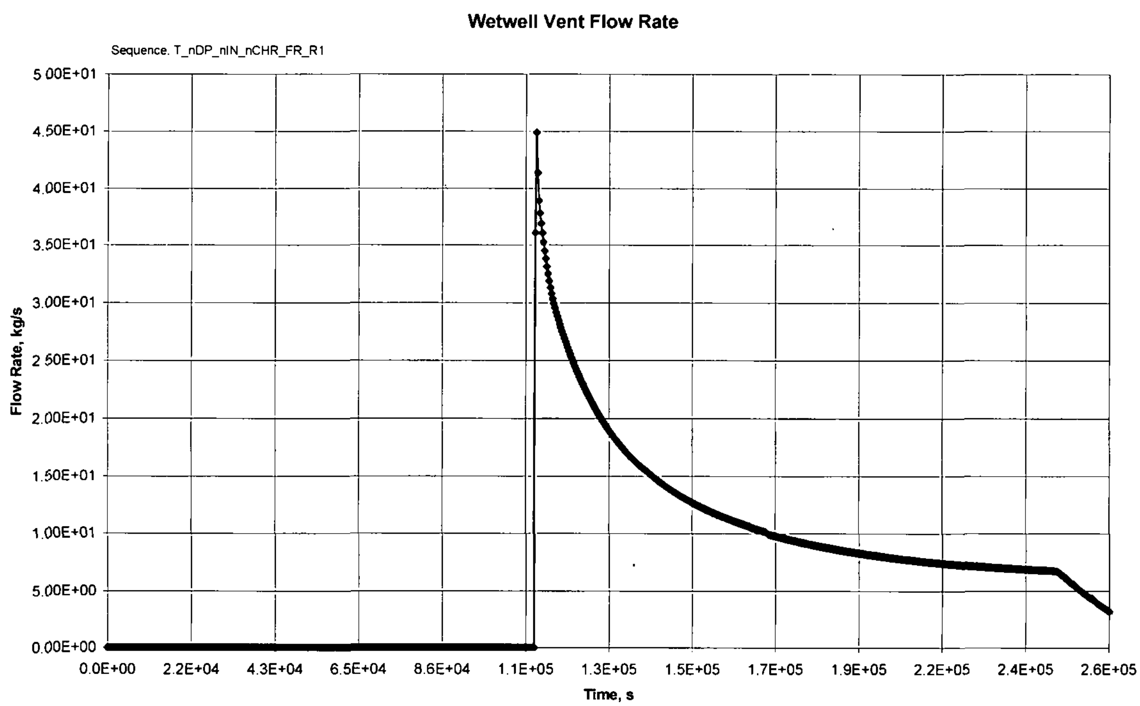

Figure 8B-5r – CDIII FR Wetwell Vent Flow Rate

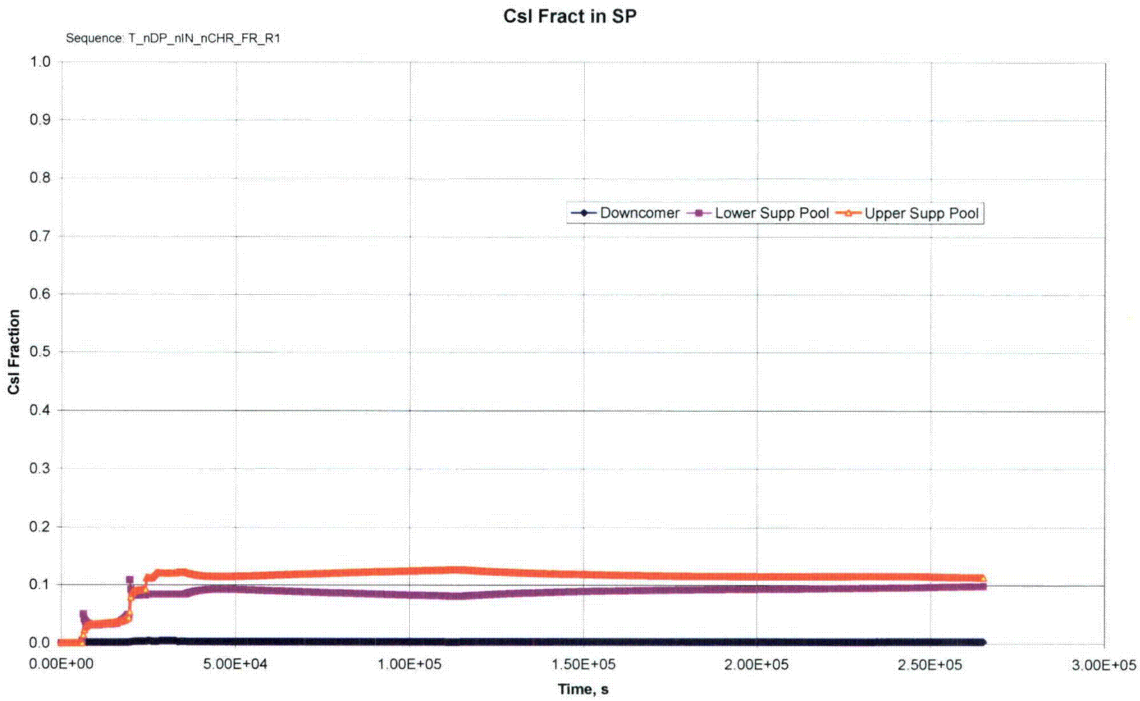

Figure 8B-5s – CDIII FR Suppression Pool CsI Fraction

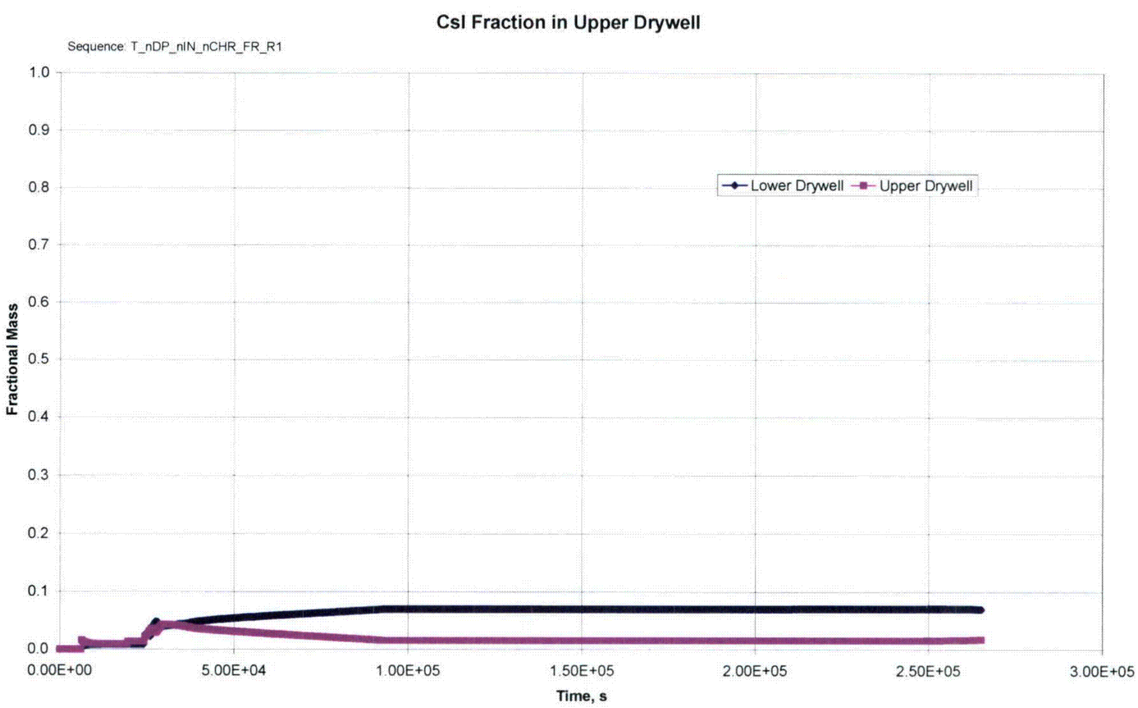

Figure 8B-5t – CDIII FR Drywell CsI Fraction

8B.6 – Class IV Representative Sequence (T_AT_nIN_TSL)

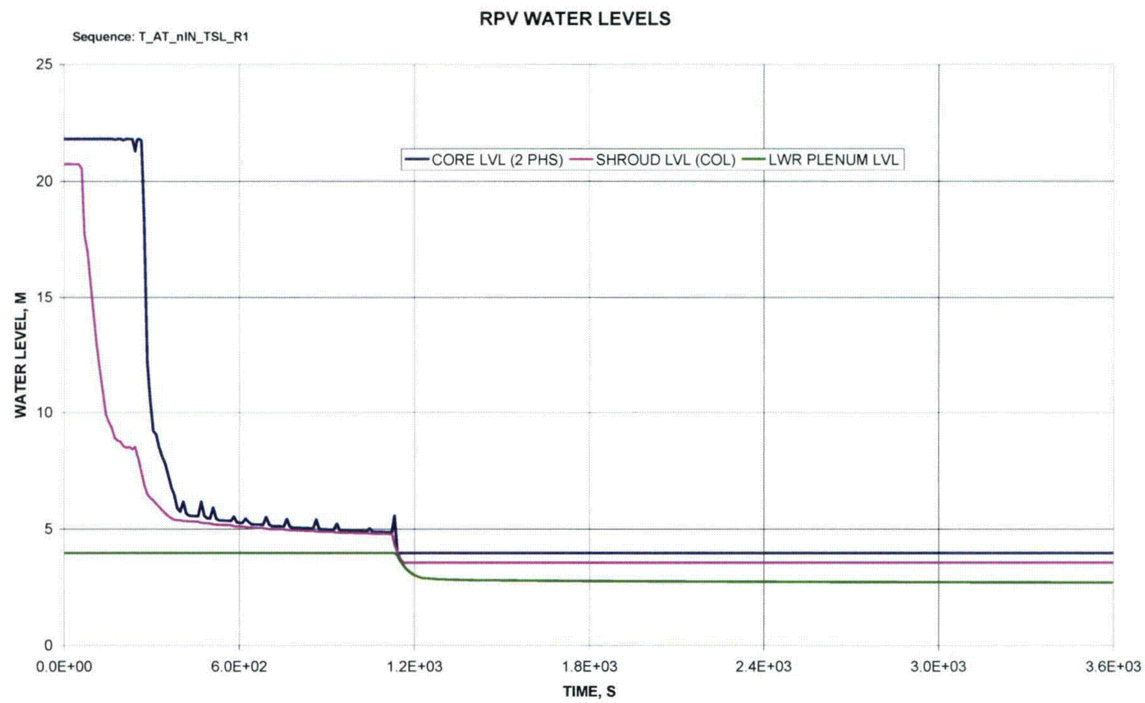

Figure 8B-6a – CDIV TSL RPV Water Levels

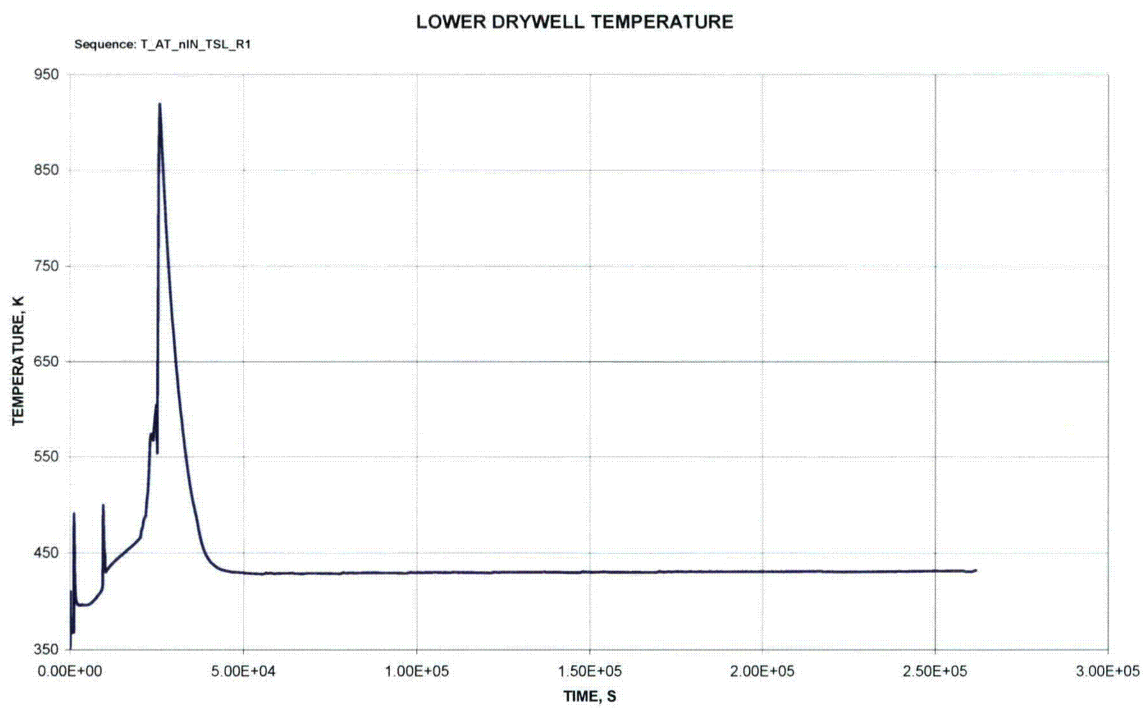

Figure 8B-6b – CDIV TSL Lower Drywell Temperature

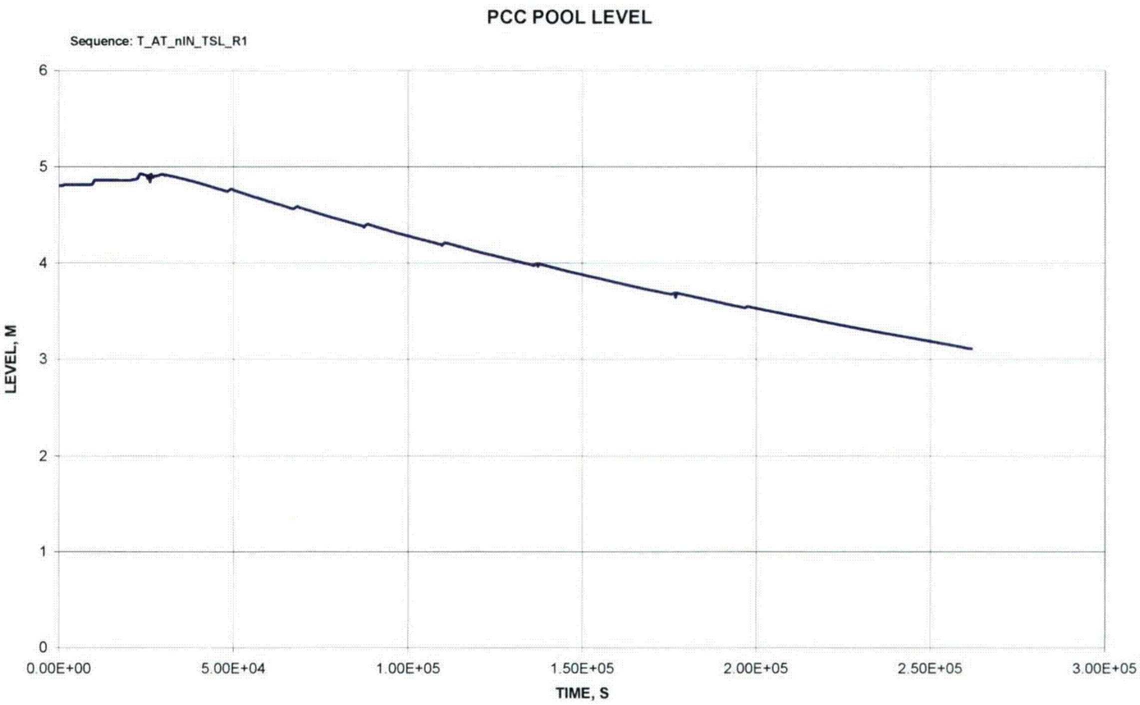

Figure 8B-6c – CDIV TSL PCCS Pool Level

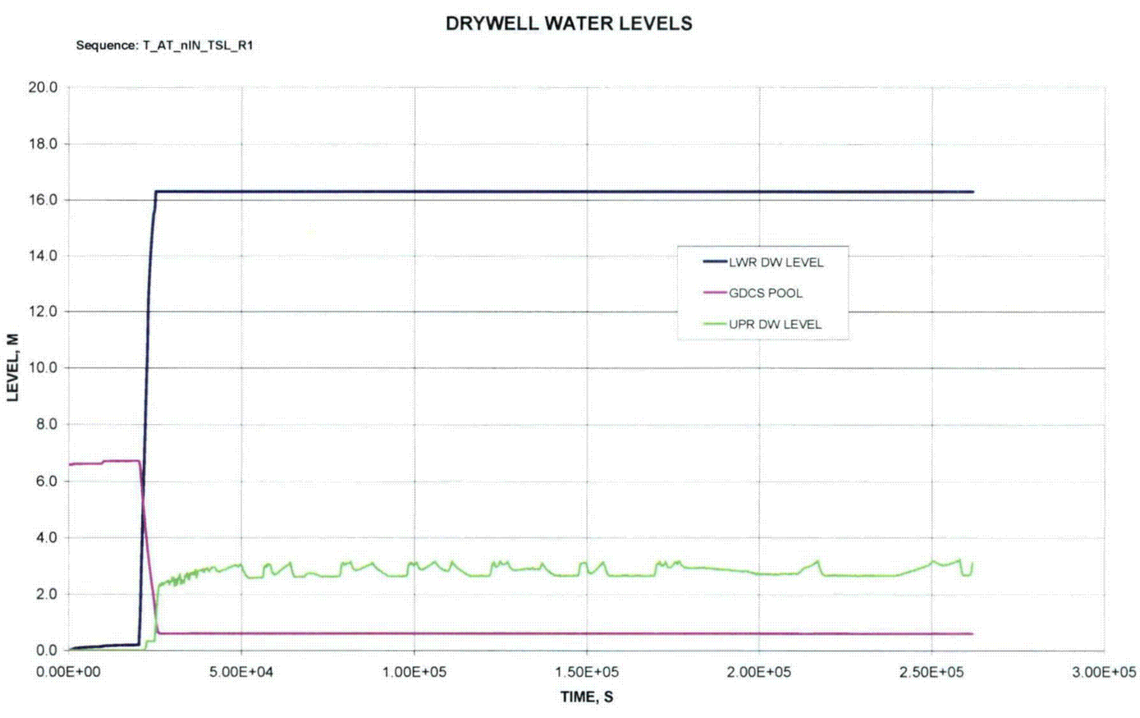

Figure 8B-6d – CDIV TSL Drywell Water Levels

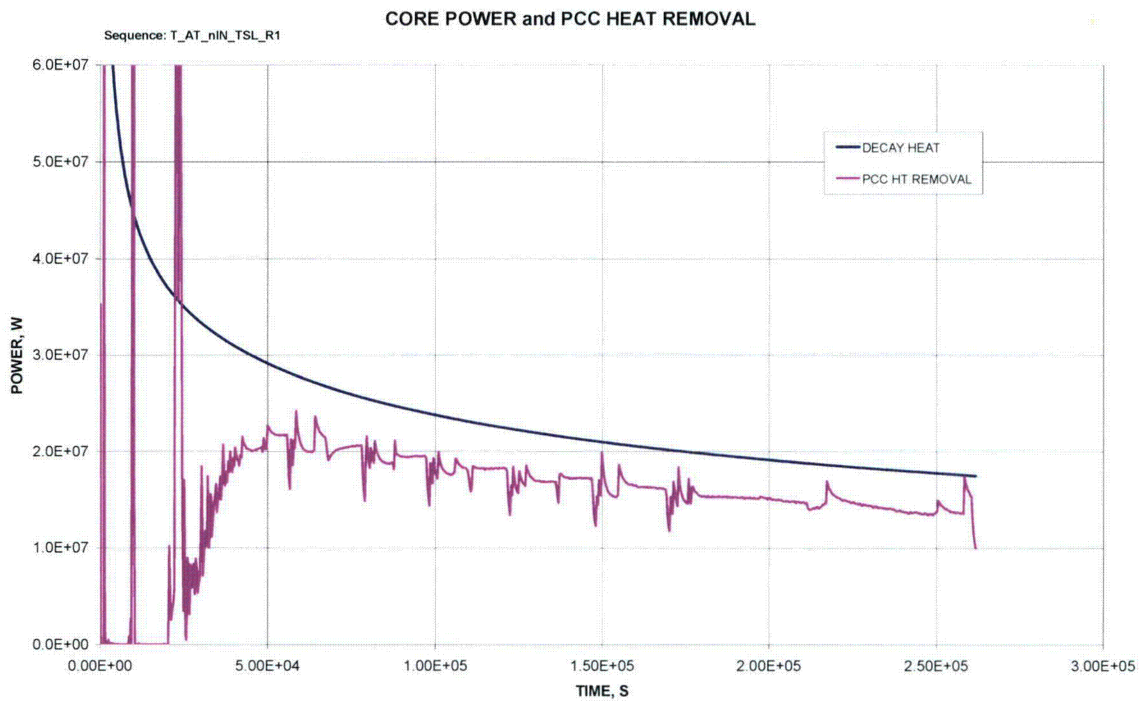

Figure 8B-6e – CDIV TSL Core Power and PCCS Heat Removal

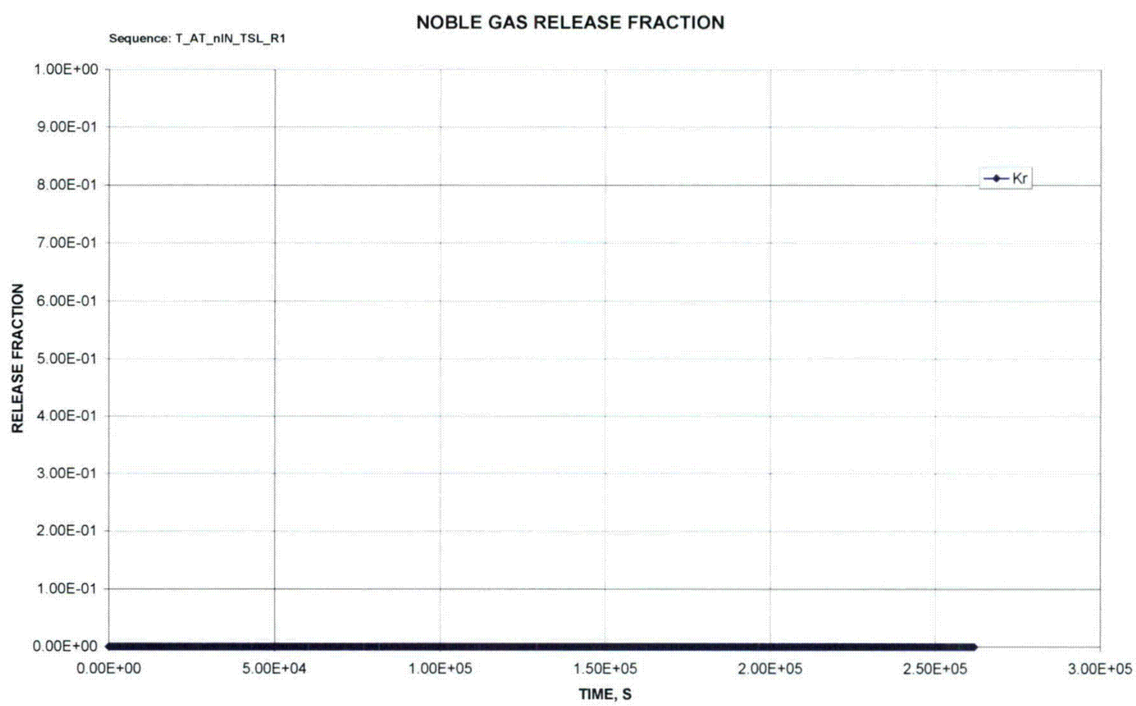

Figure 8B-6f – CDIV TSL Noble Gas Release Fraction

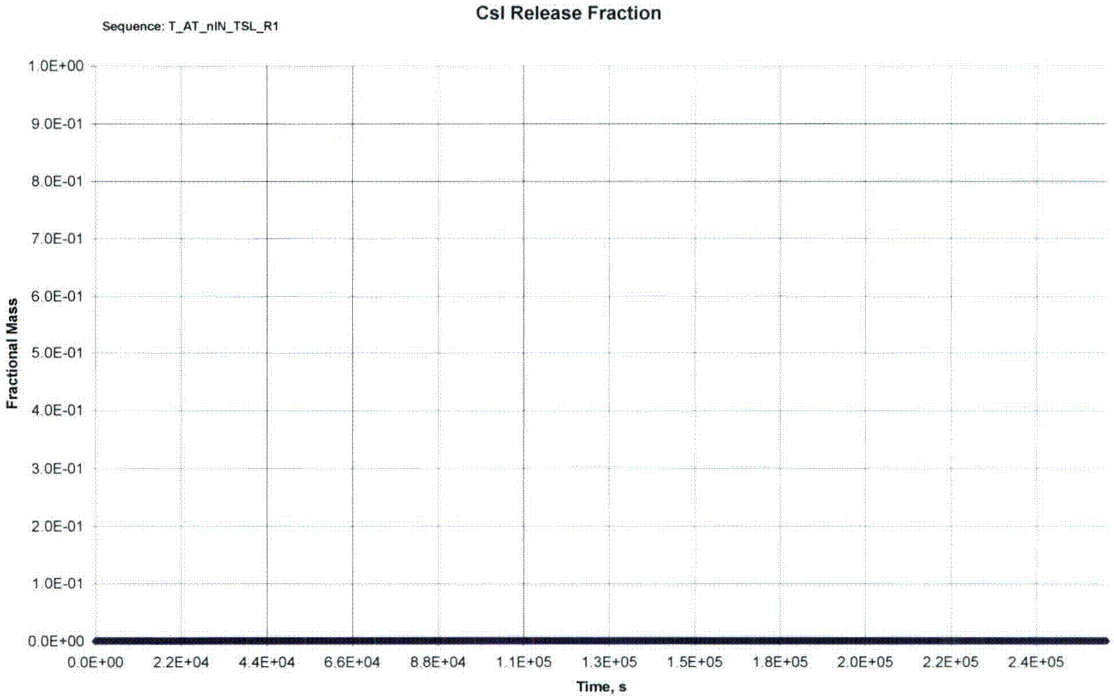

Figure 8B-6g – CDIV TSL CsI Release Fraction

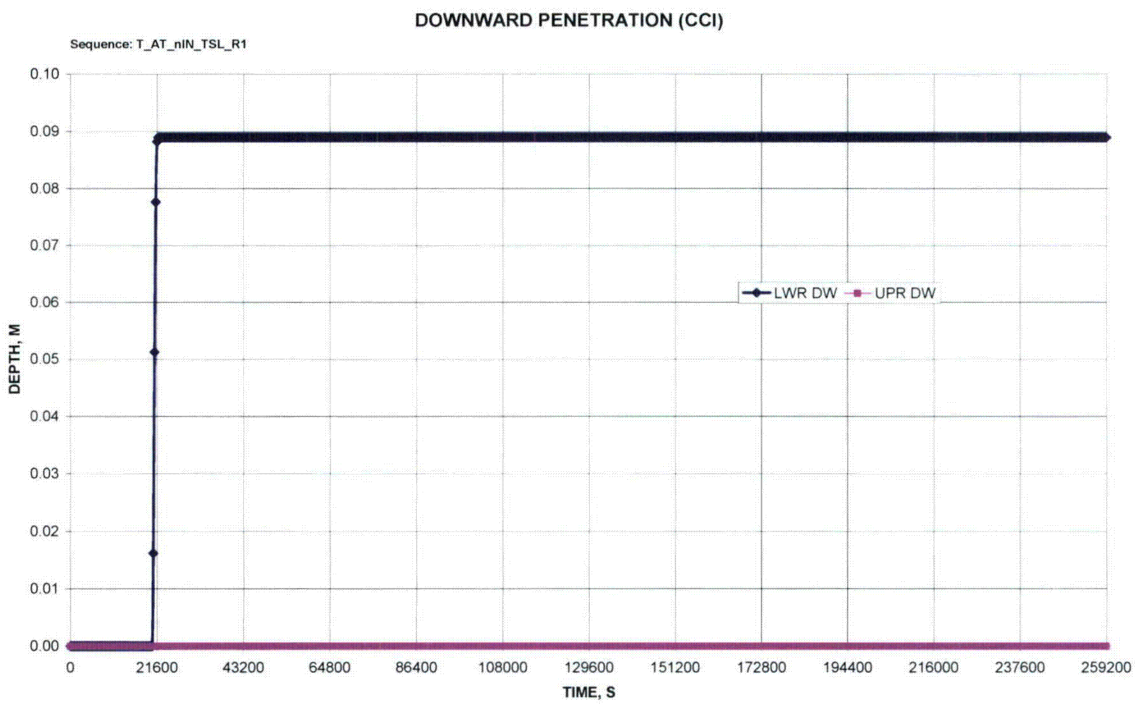

Figure 8B-6h – CDIV TSL CCI Downward Penetration

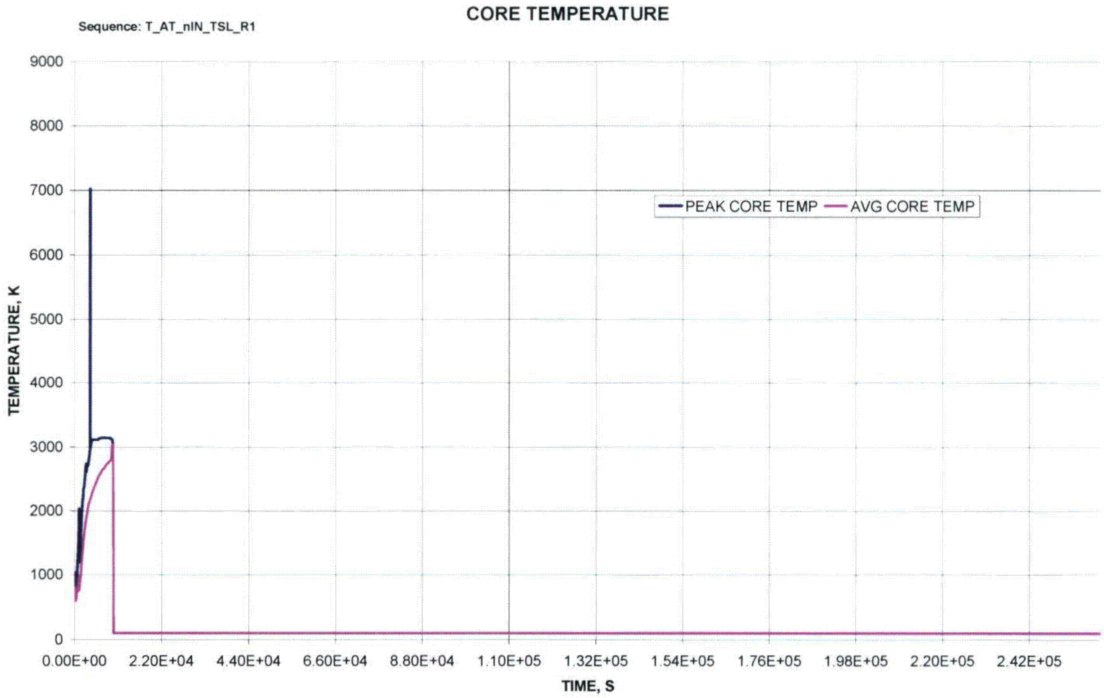

Figure 8B-6i – CDIV TSL Core Temperature

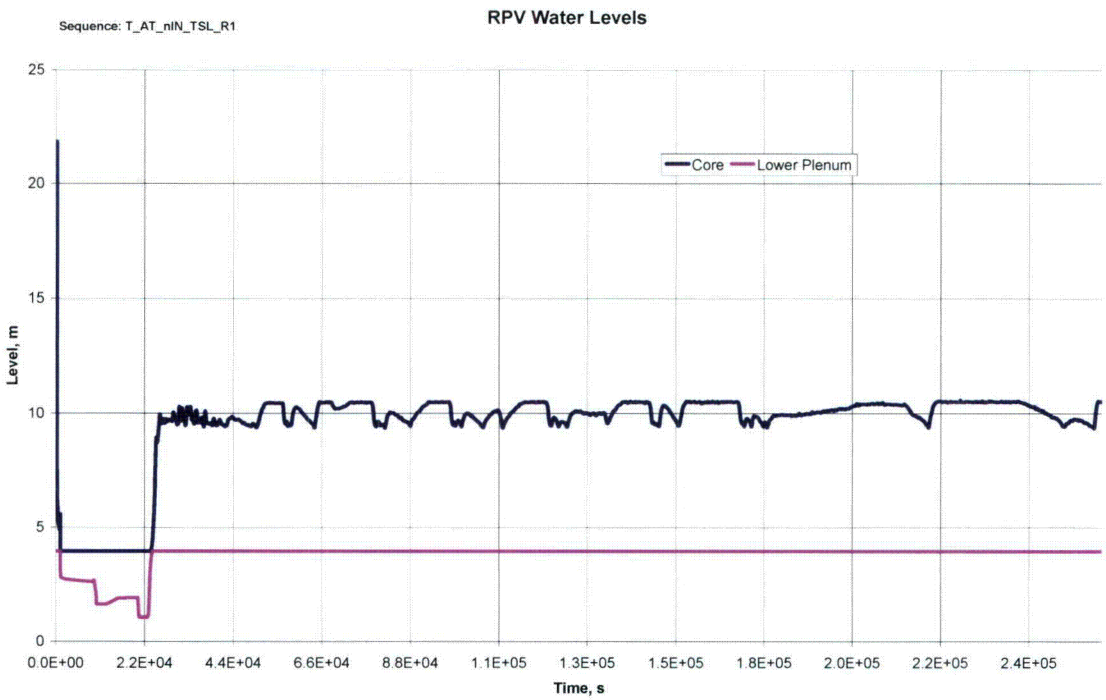

Figure 8B-6j – CDIV TSL RPV Water Levels

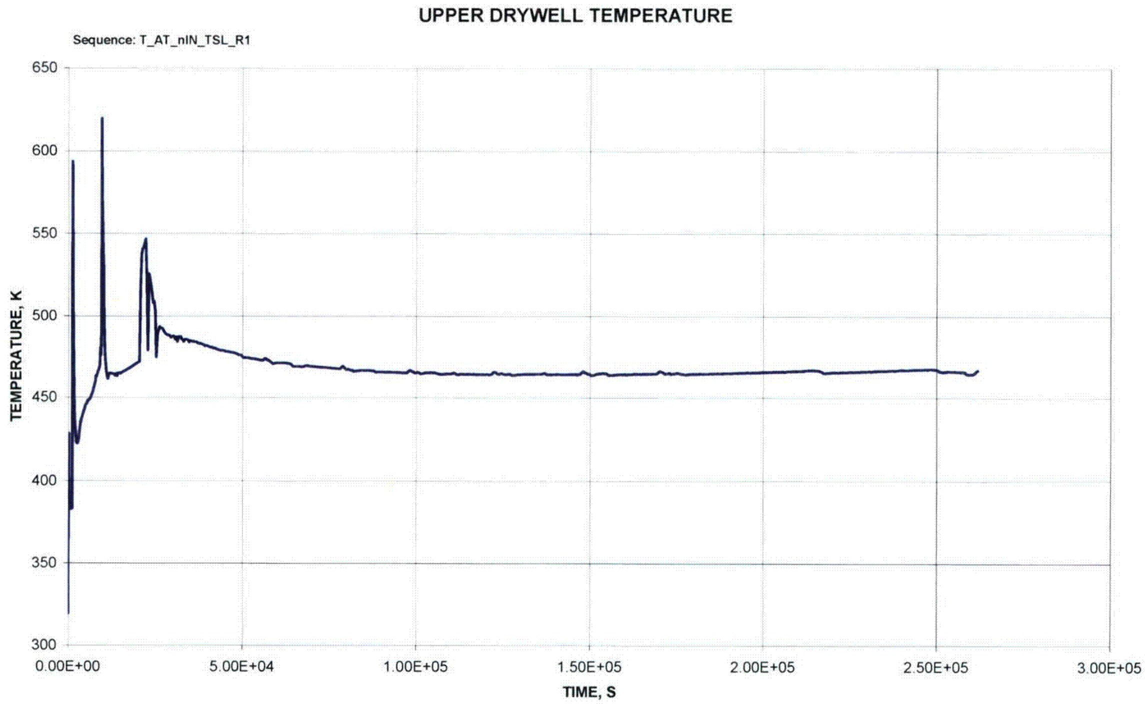

Figure 8B-6k – CDIV TSL Upper Drywell Temperature

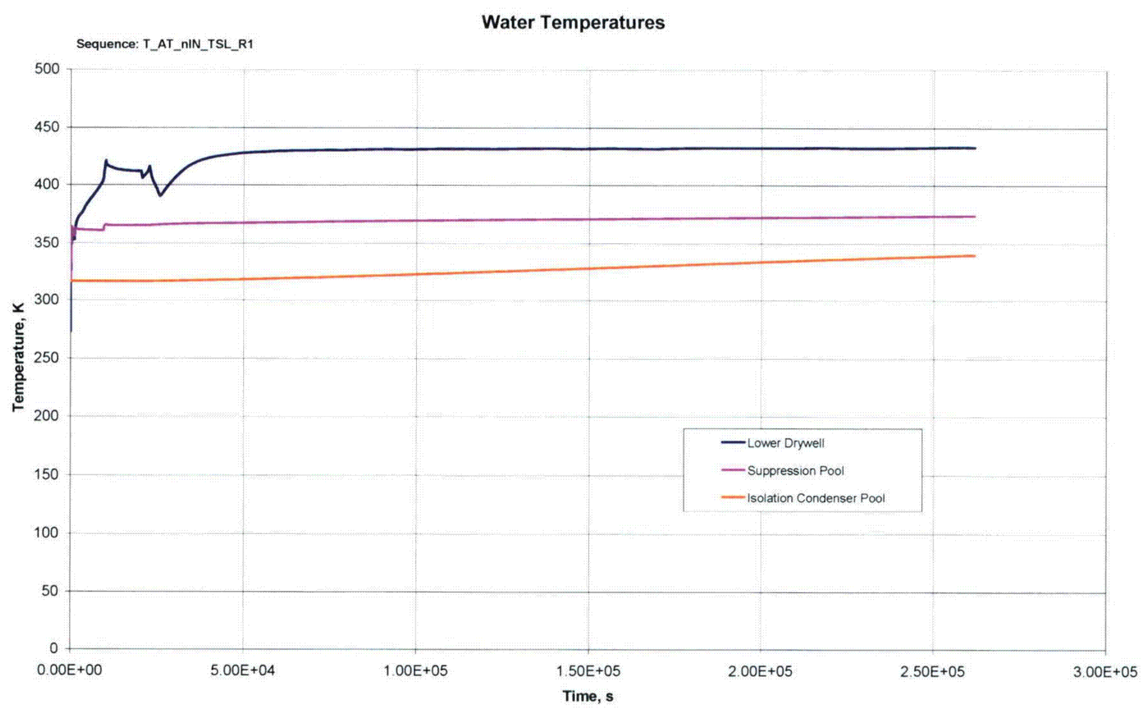

Figure 8B-6l – CDIV TSL Water Temperatures

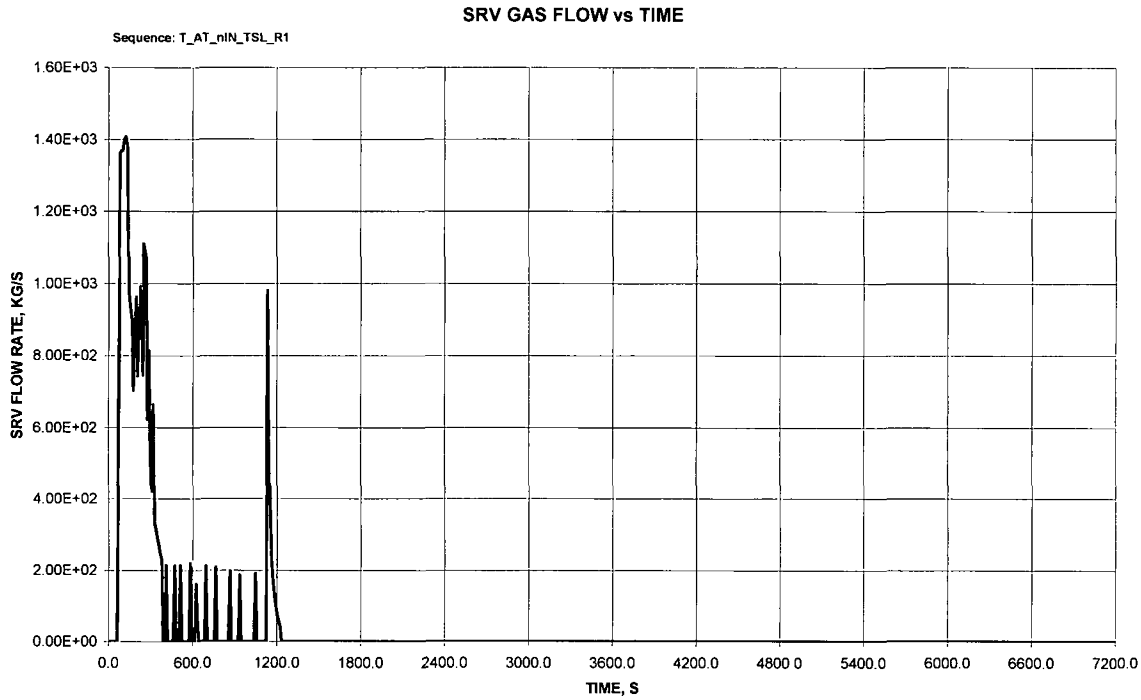

Figure 8B-6m – CDIV TSL SRV Flow Rate

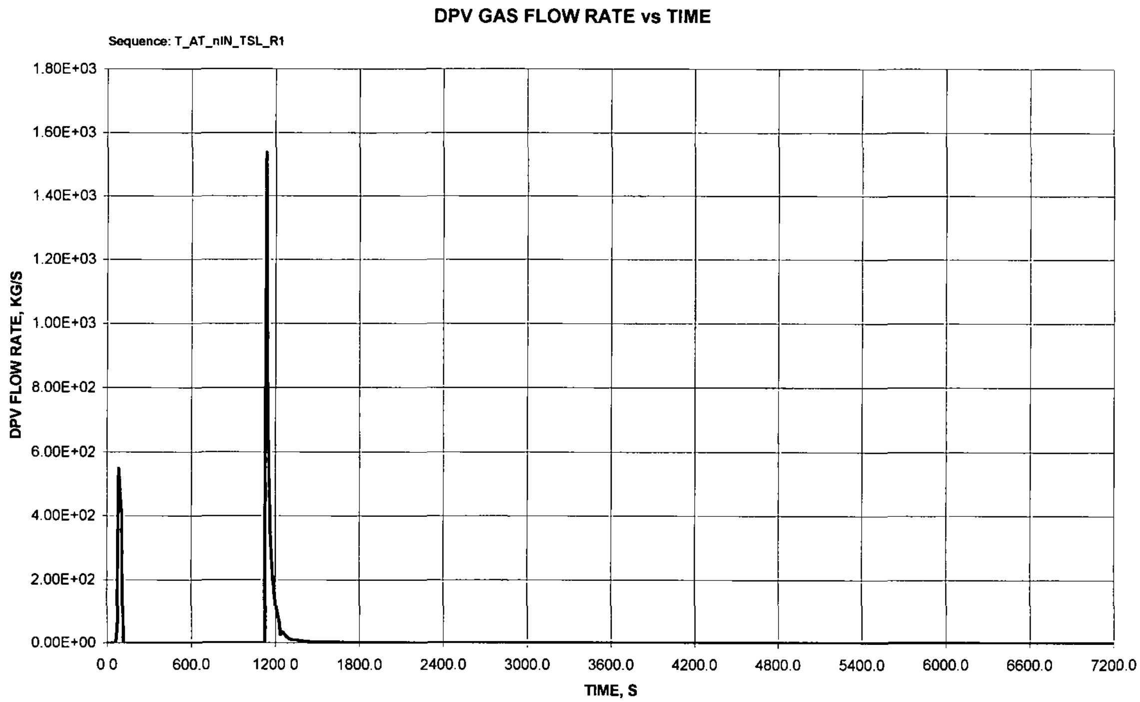

Figure 8B-6n – CDIV TSL DPV Flow Rate

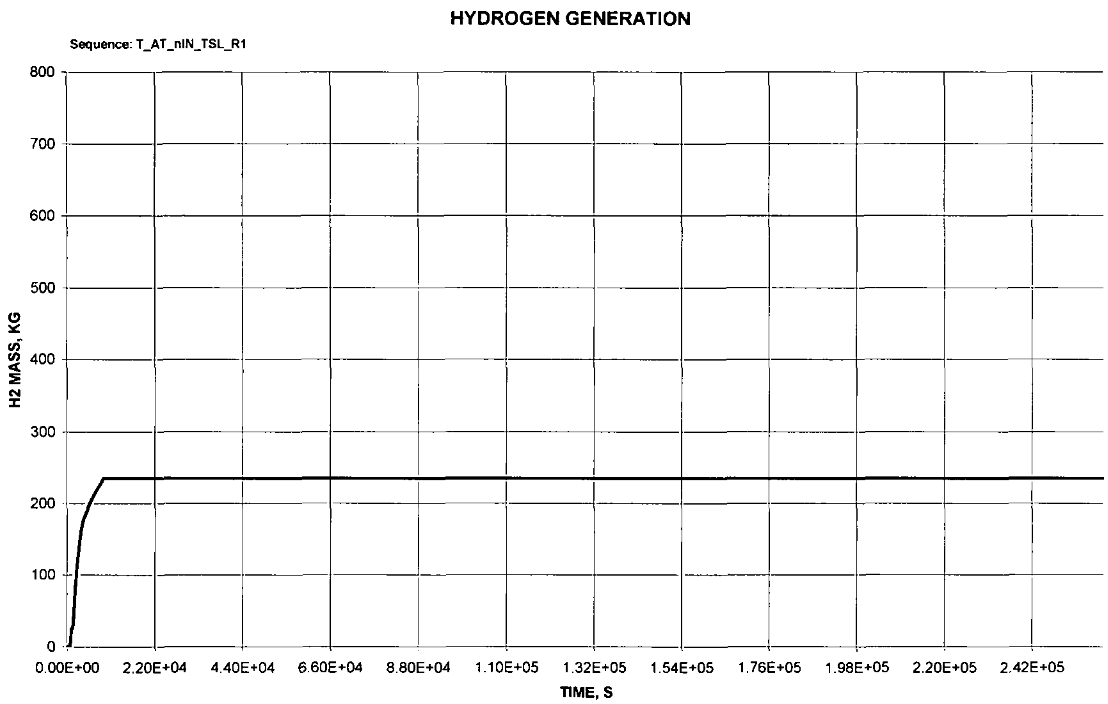

Figure 8B-6o – CDIV TSL Core Hydrogen Generation

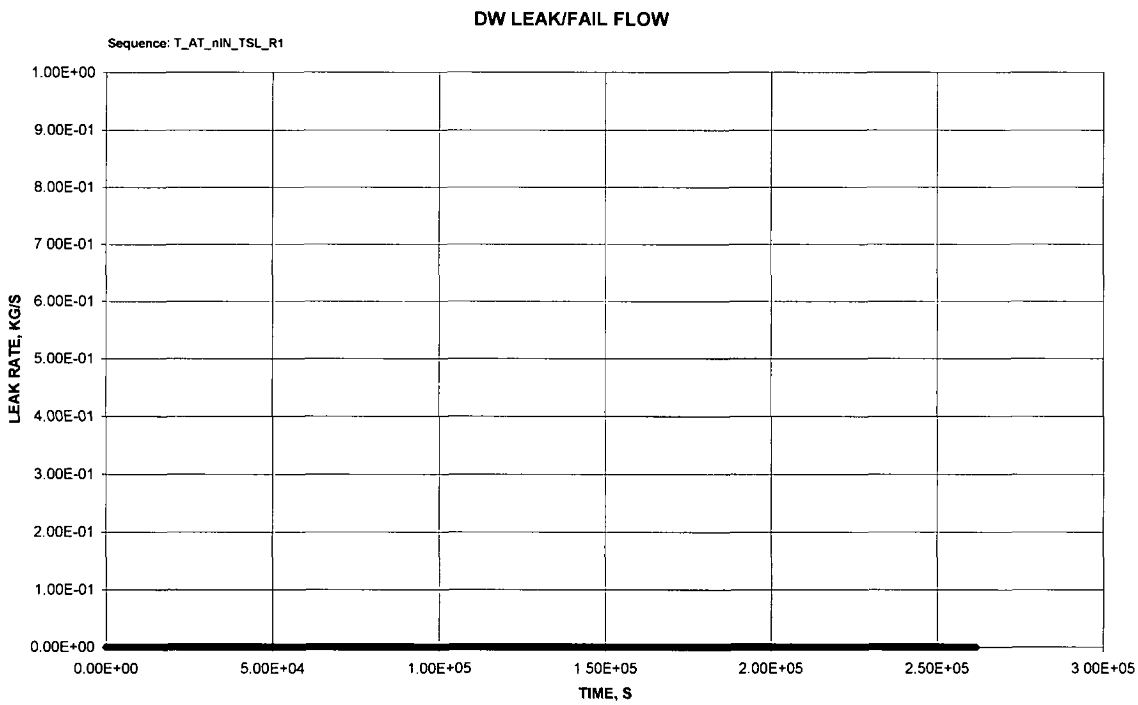

Figure 8B-6p – CDIV TSL Drywell Leak/Failure Flow Rate

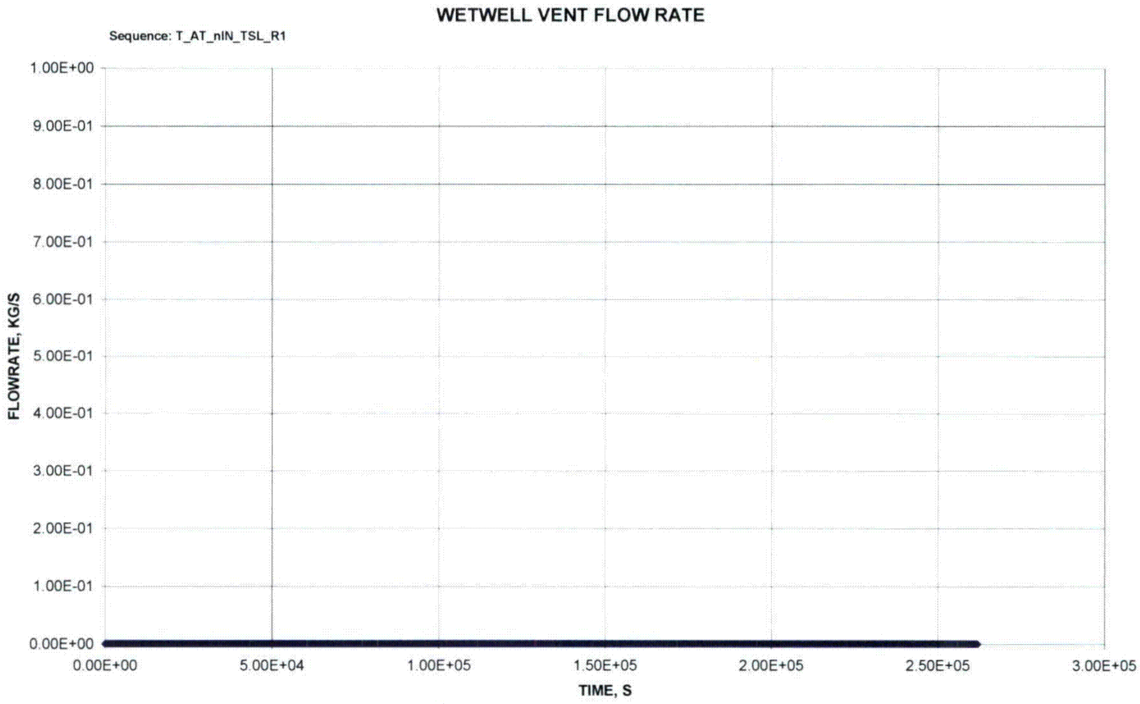

Figure 8B-6q – CDIV TSL Wetwell Vent Flow Rate

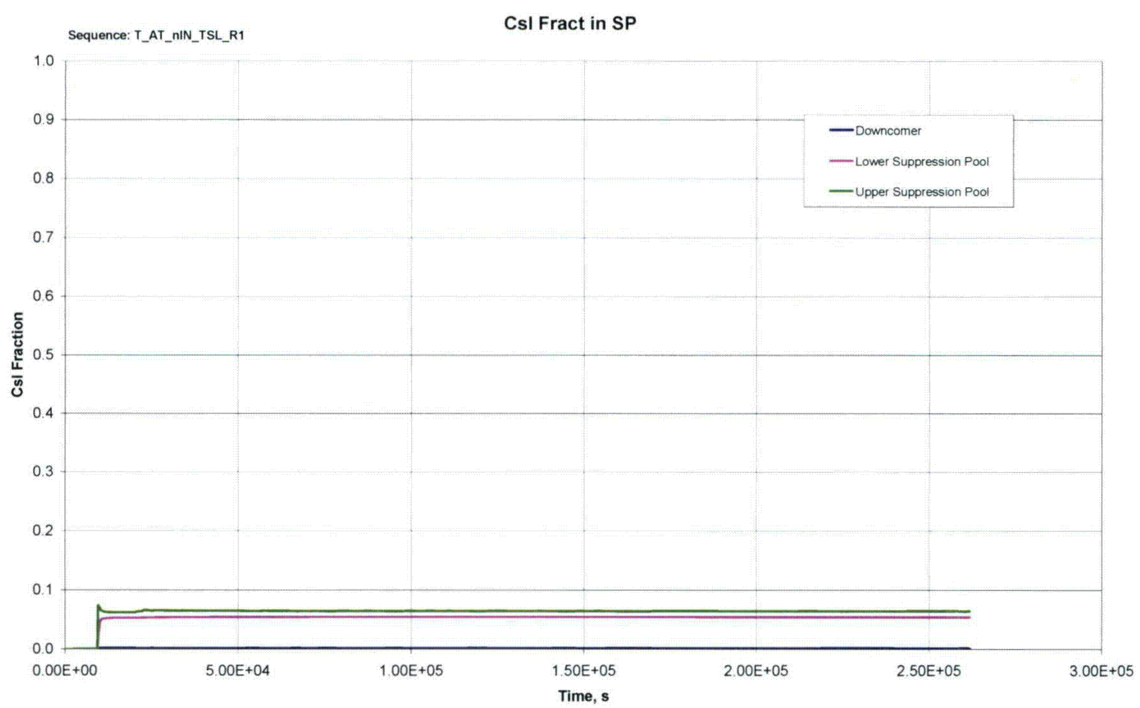

Figure 8B-6r – CDIV TSL Suppression Pool CsI Fraction

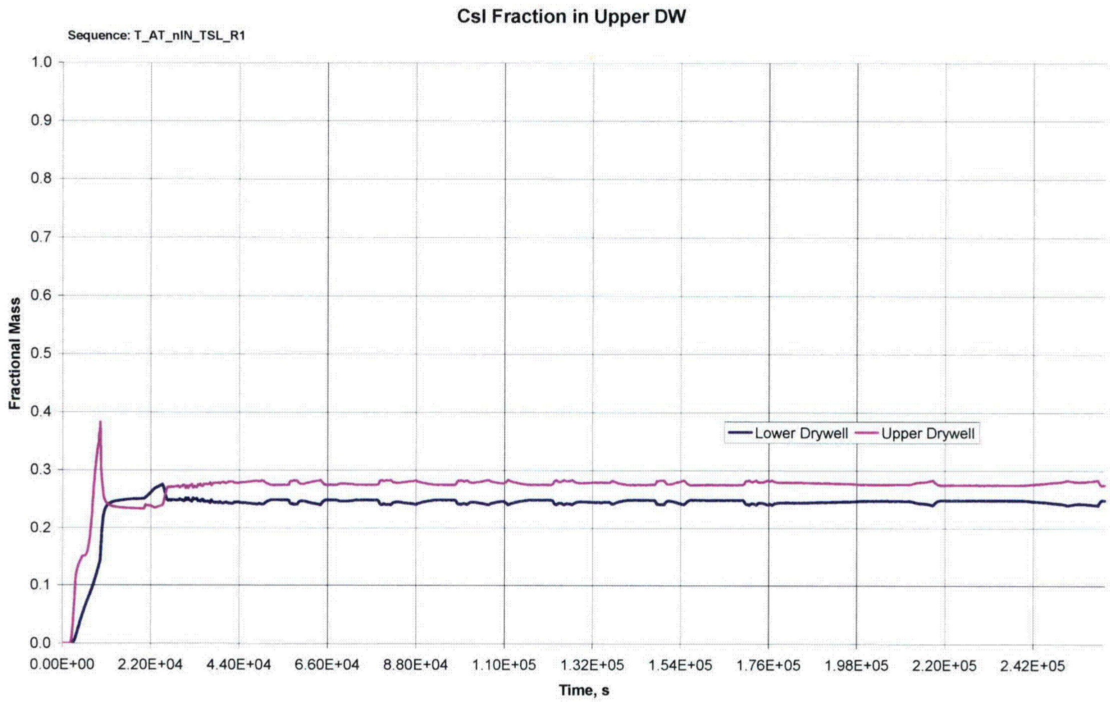

Figure 8B-6s – CDIV TSL Drywell CsI Fraction

8B.7 – Class IV Representative Sequence (T_AT_nIN_nCHR_FR)

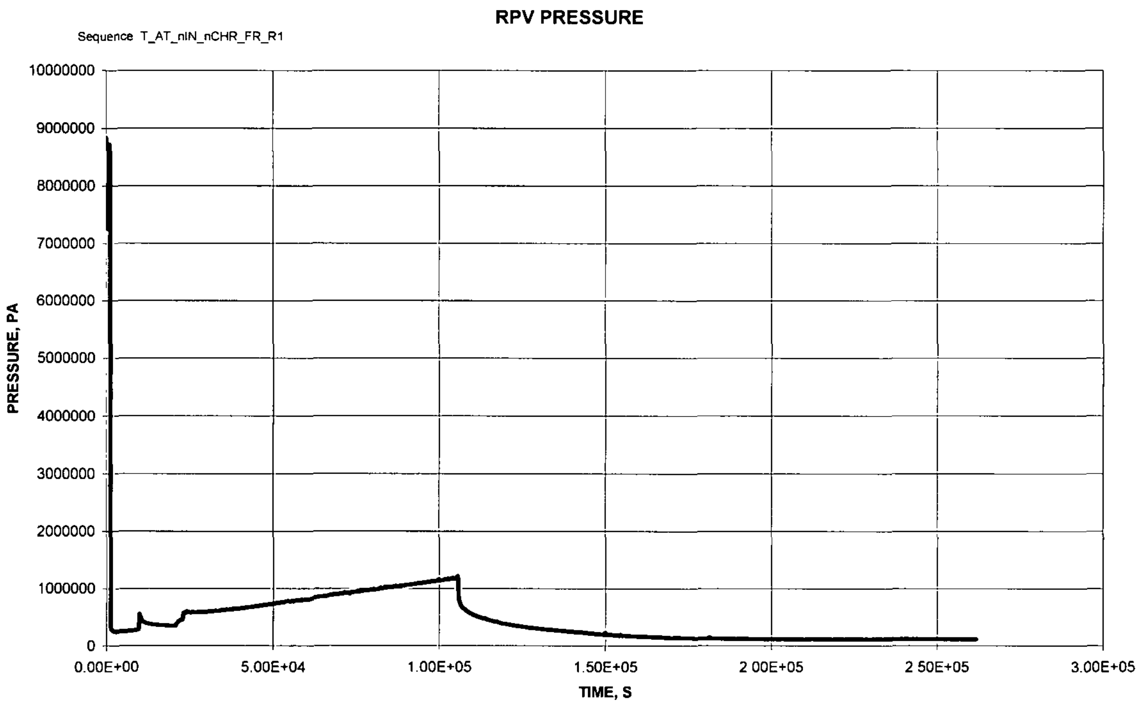

Figure 8B-7a – CDIV FR RPV Pressure

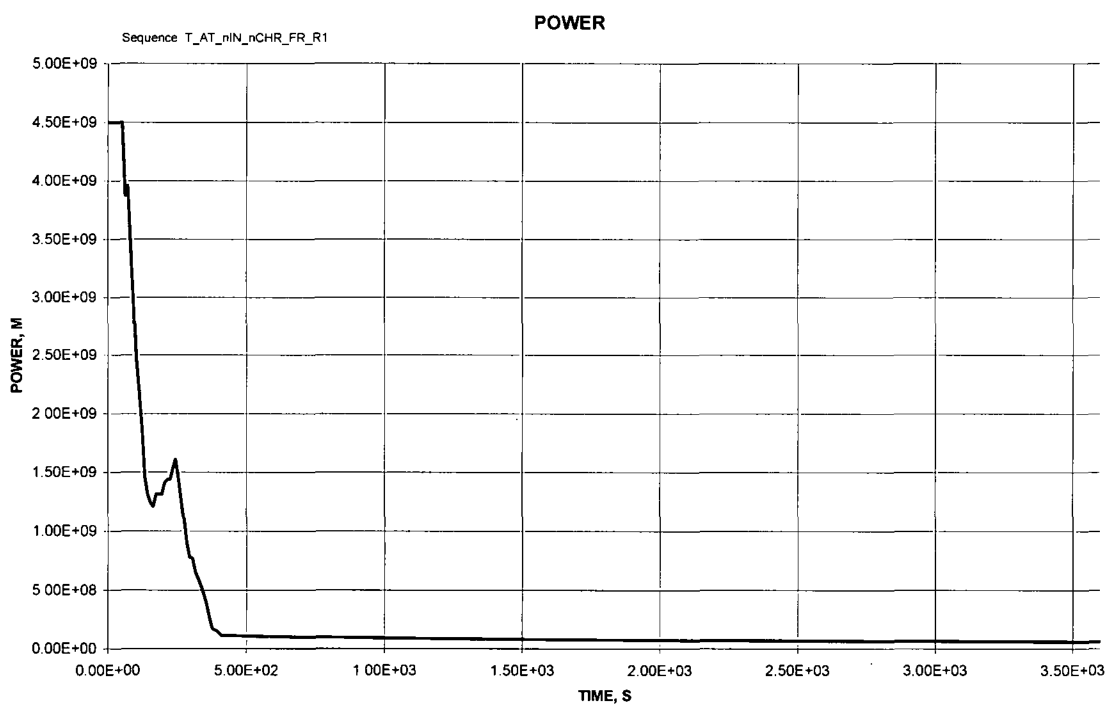

Figure 8B-7b – CDIV FR Core Power Level

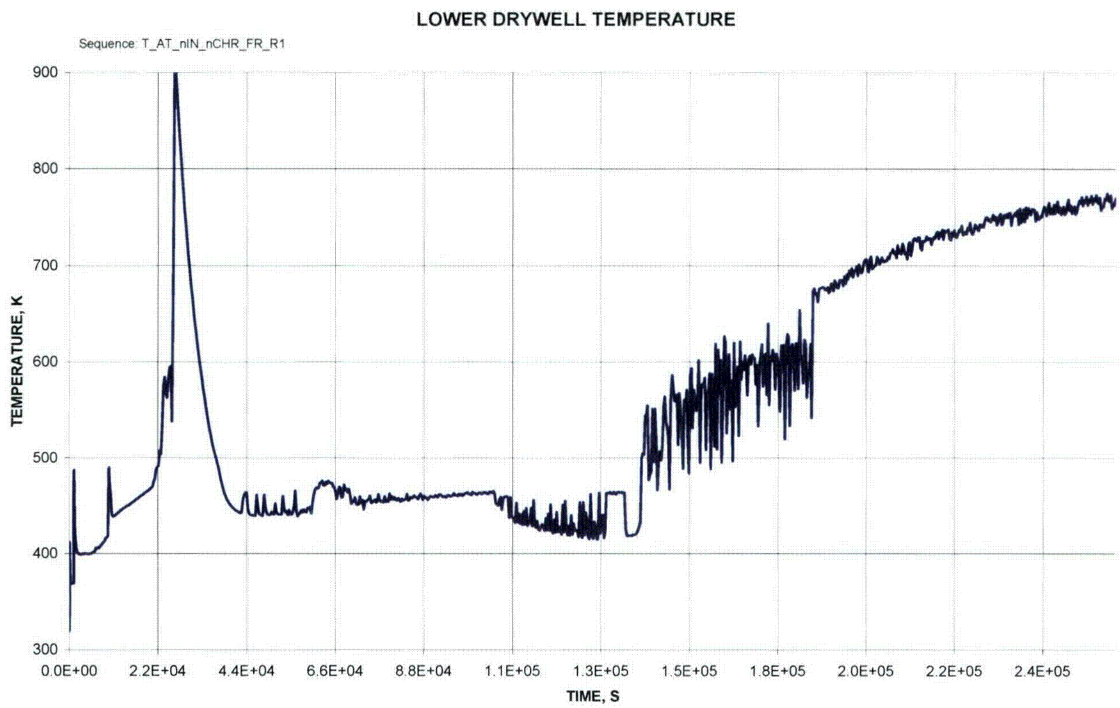

Figure 8B-7c – CDIV FR Lower Drywell Temperature

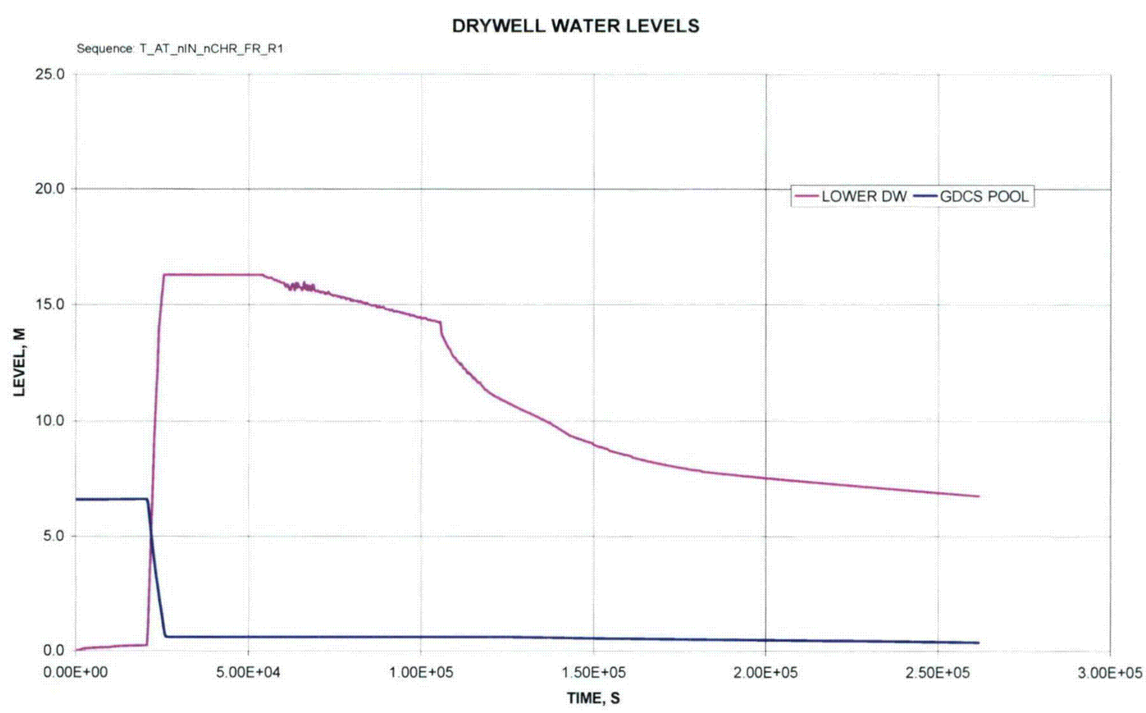

Figure 8B-7d – CDIV FR Drywell Water Levels

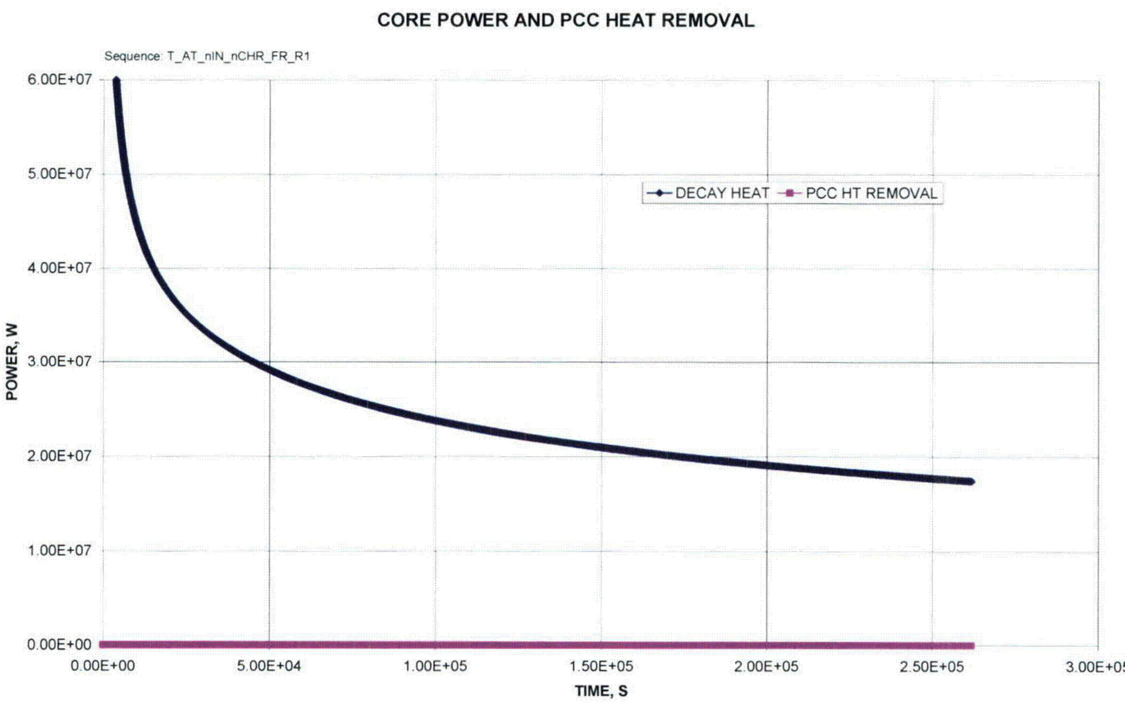

Figure 8B-7e – CDIV FR Core Power and PCCS Heat Removal

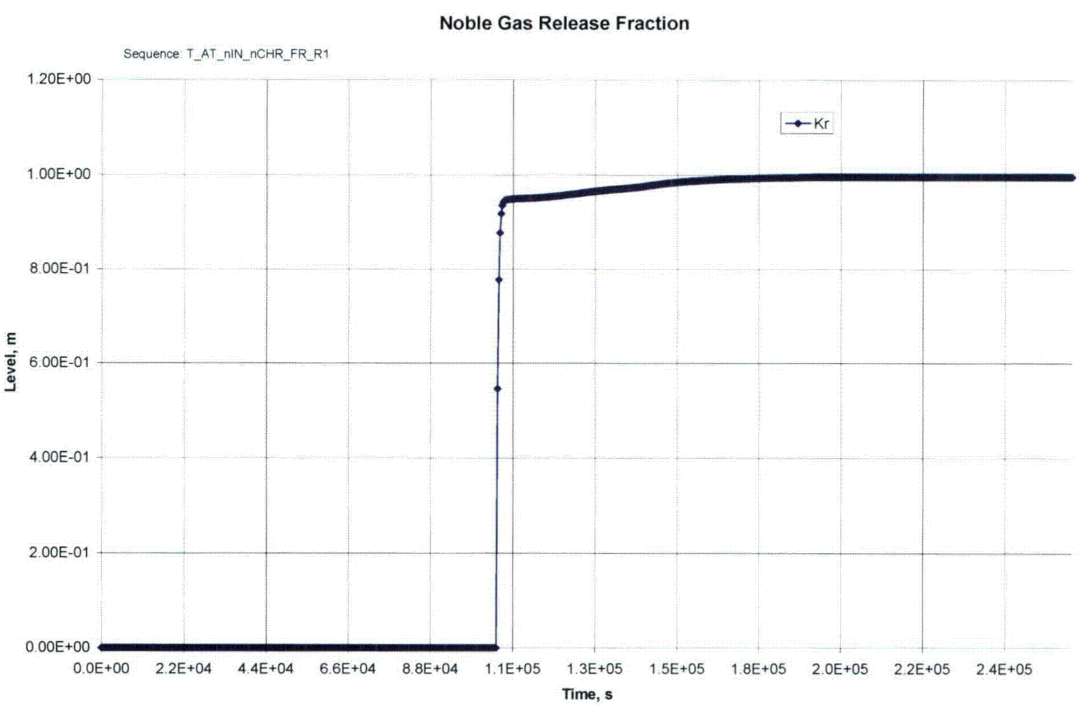

Figure 8B-7f – CDIV FR Noble Gas Release Fraction

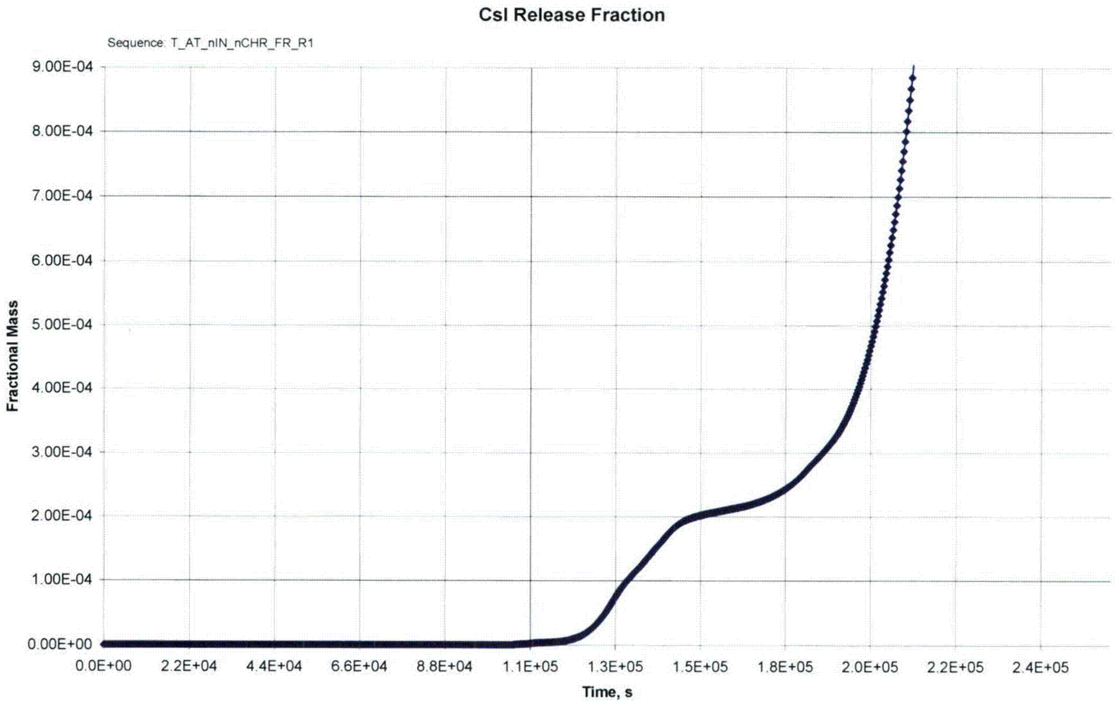

Figure 8B-7g – CDIV FR CsI Release Fraction

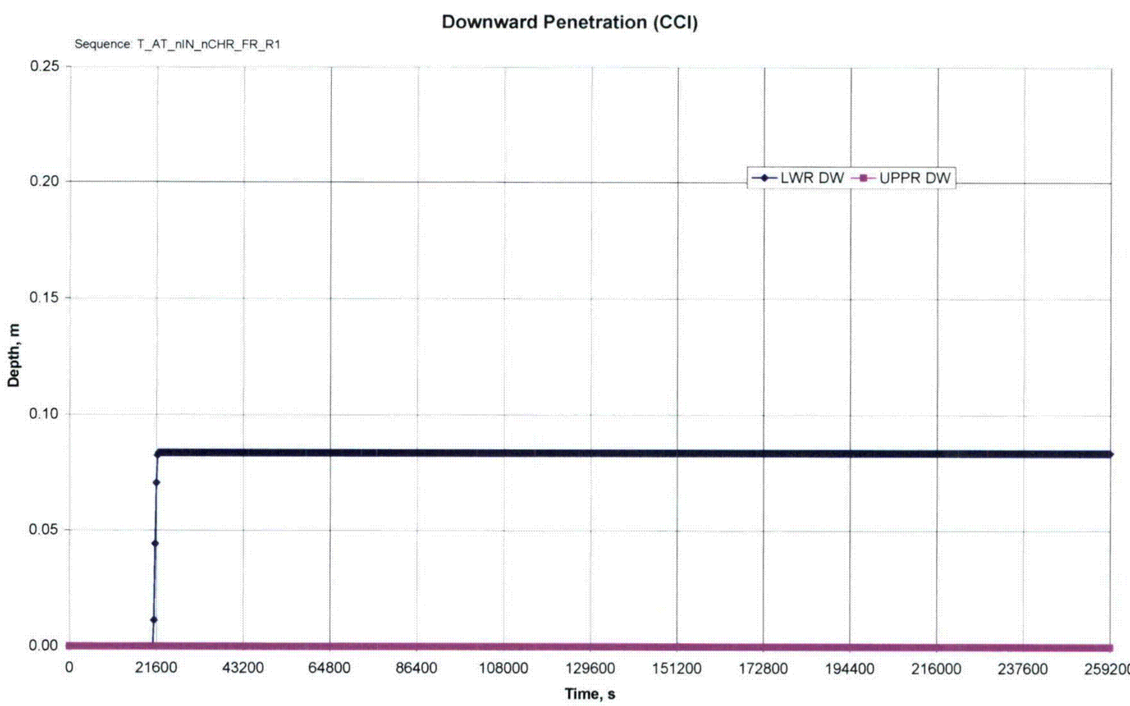

Figure 8B-7h – CDIV FR Downward CCI Penetration

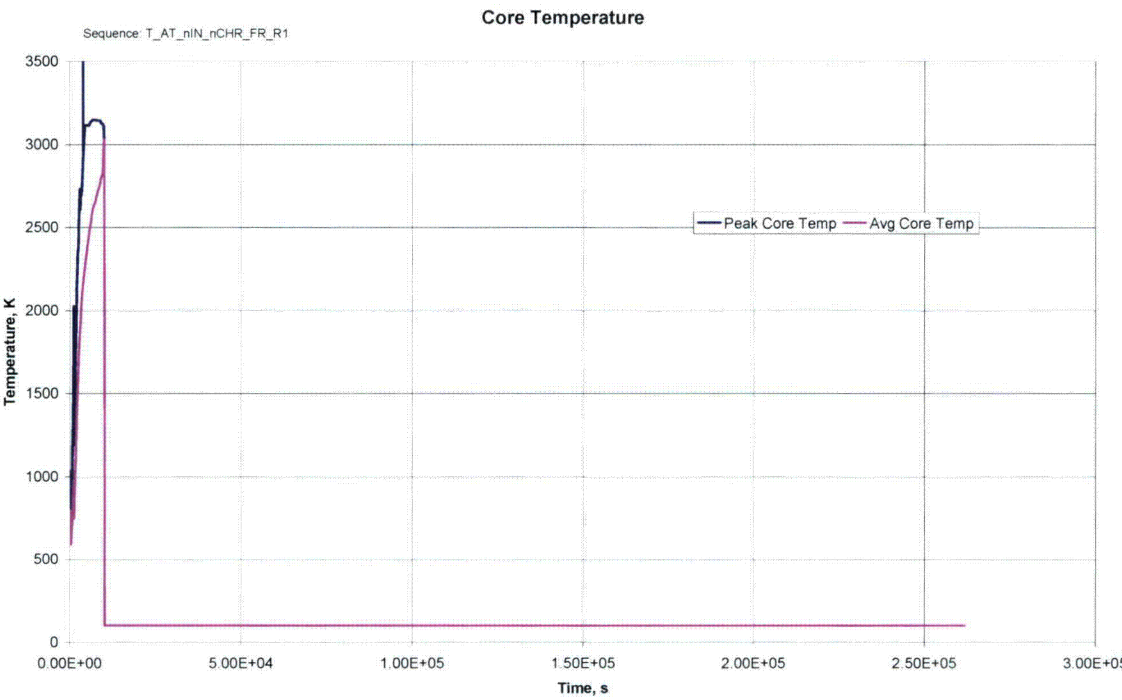

Figure 8B-7i – CDIV FR Core Temperature

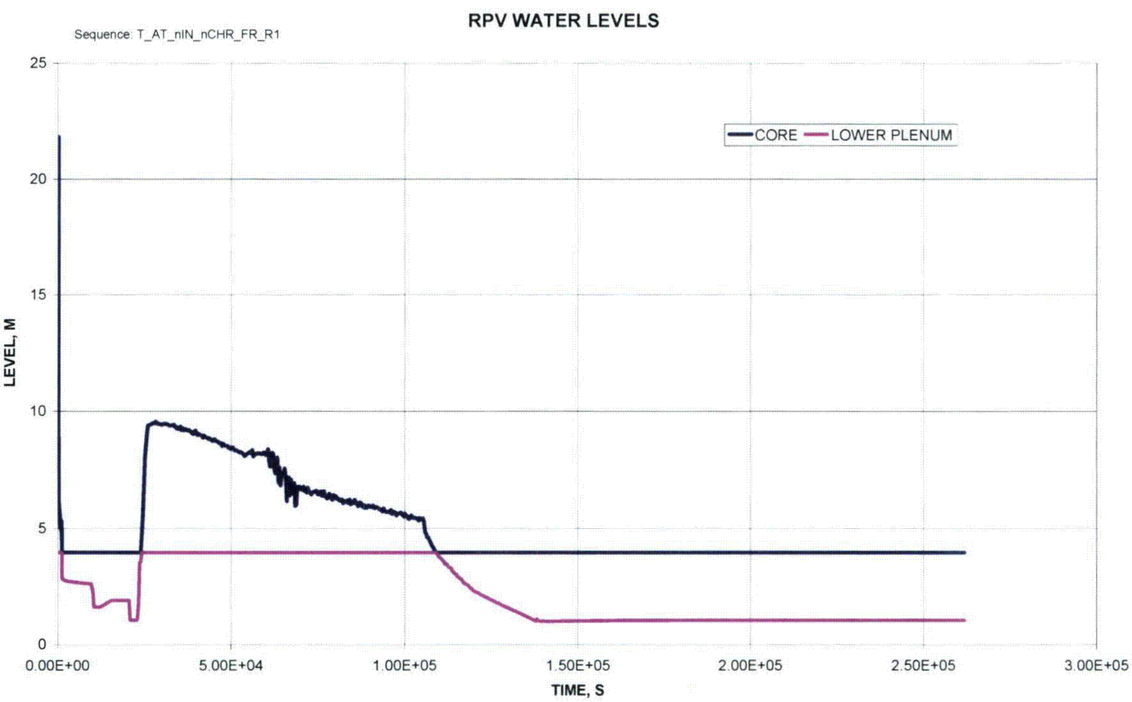

Figure 8B-7j – CDIV FR RPV Water Levels

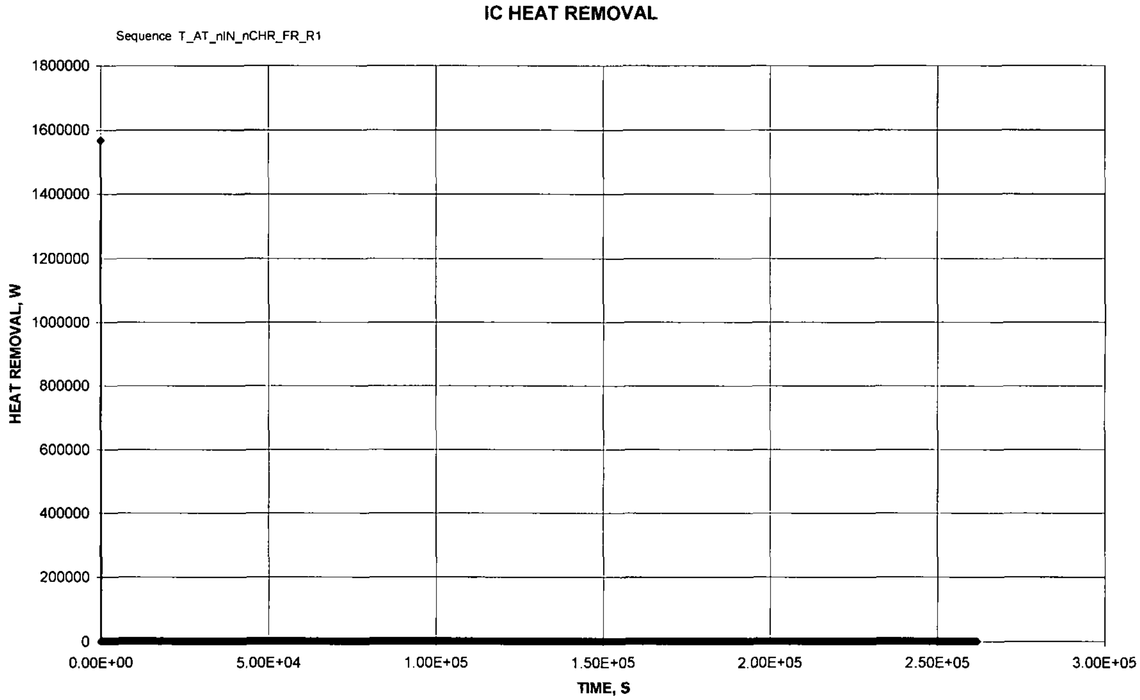

Figure 8B-7k – CDIV FR ICS Heat Removal

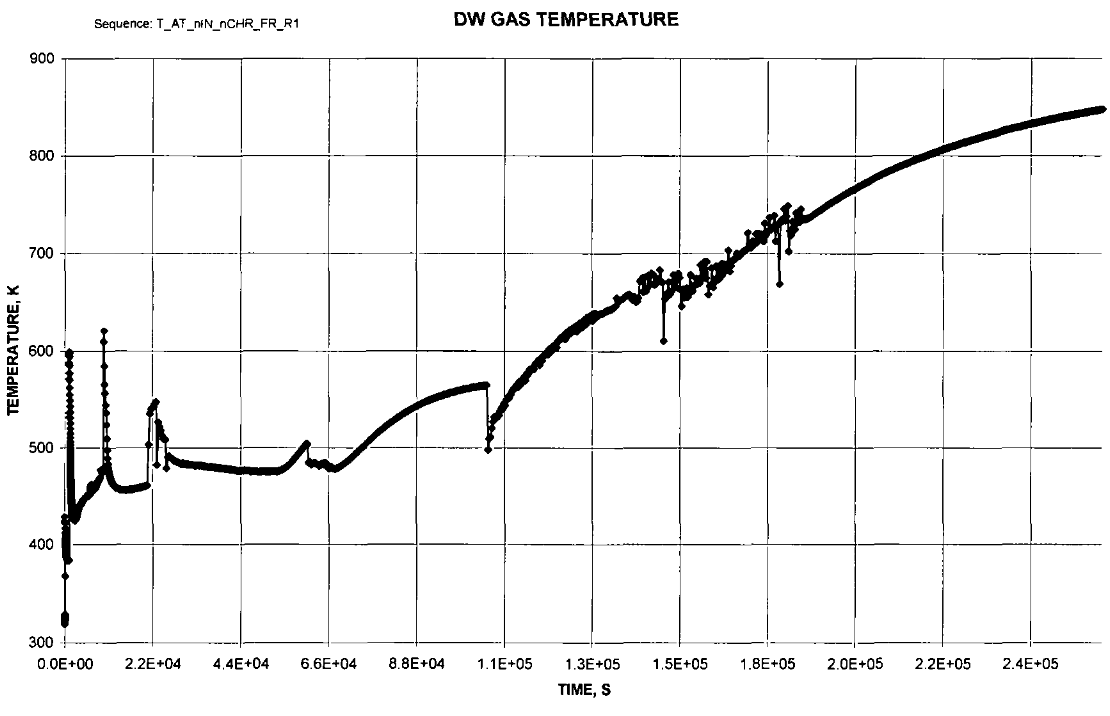

Figure 8B-7l – CDIV FR Drywell Gas Temperature

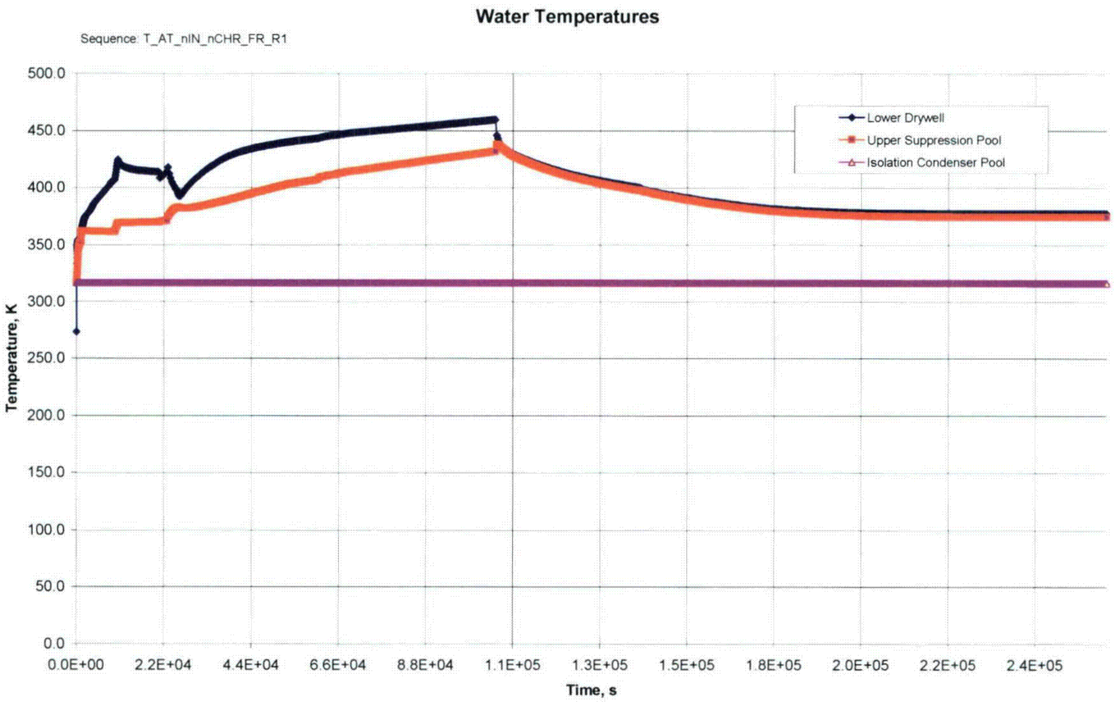

Figure 8B-7m – CDIV FR Water Temperatures

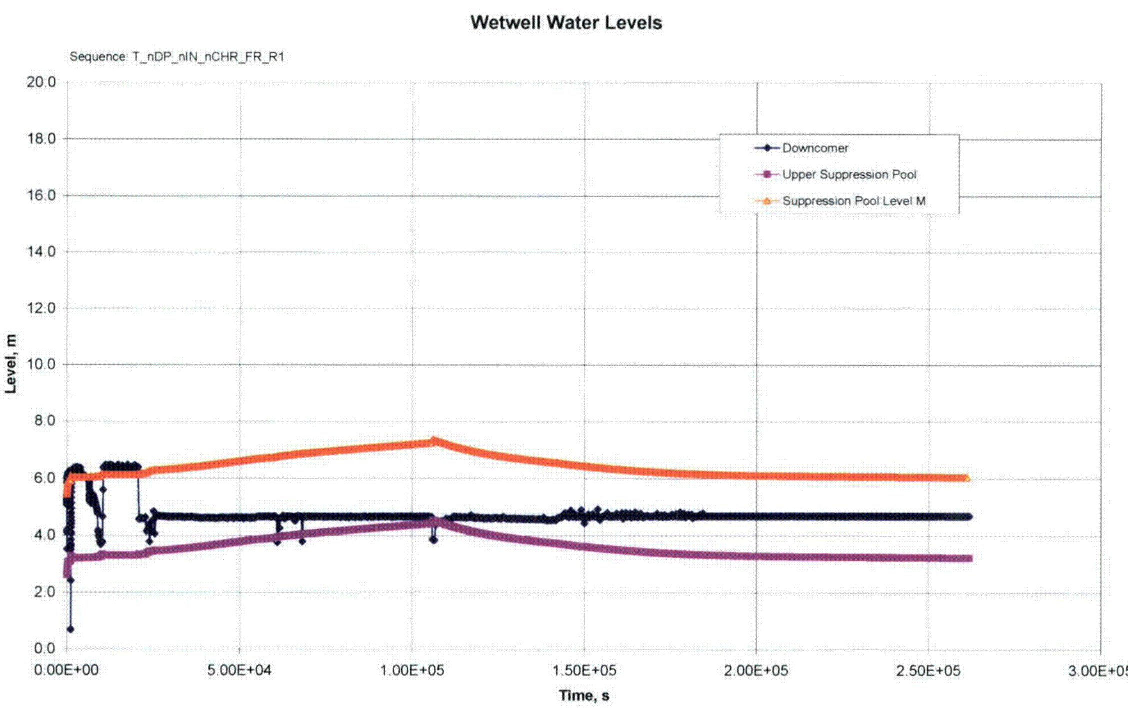

Figure 8B-7n – CDIV FR Wetwell Water Levels

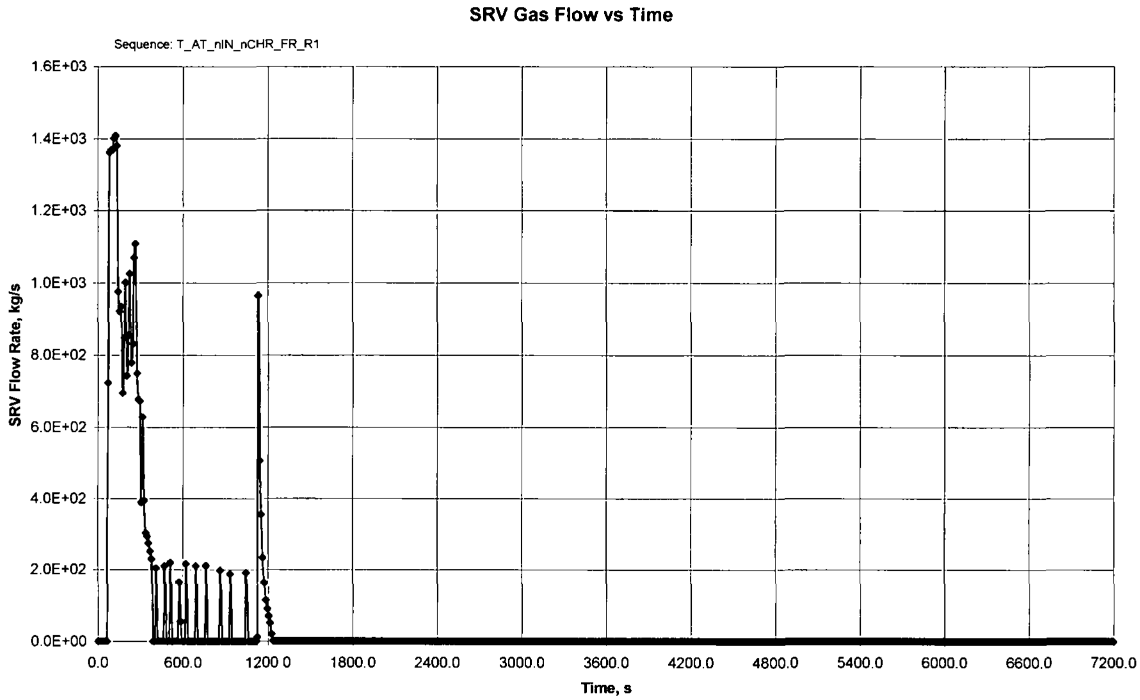

Figure 8B-7o – CDIV FR SRV Gas Flow Rate

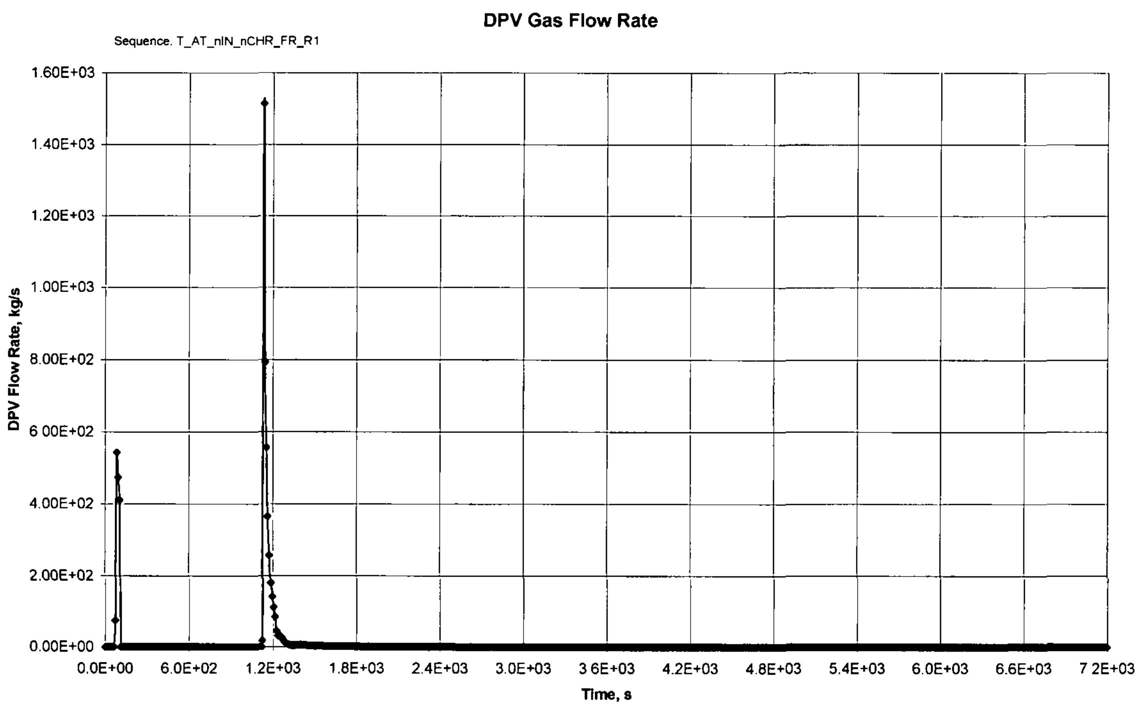

Figure 8B-7p – CDIV FR DPV Gas Flow Rate

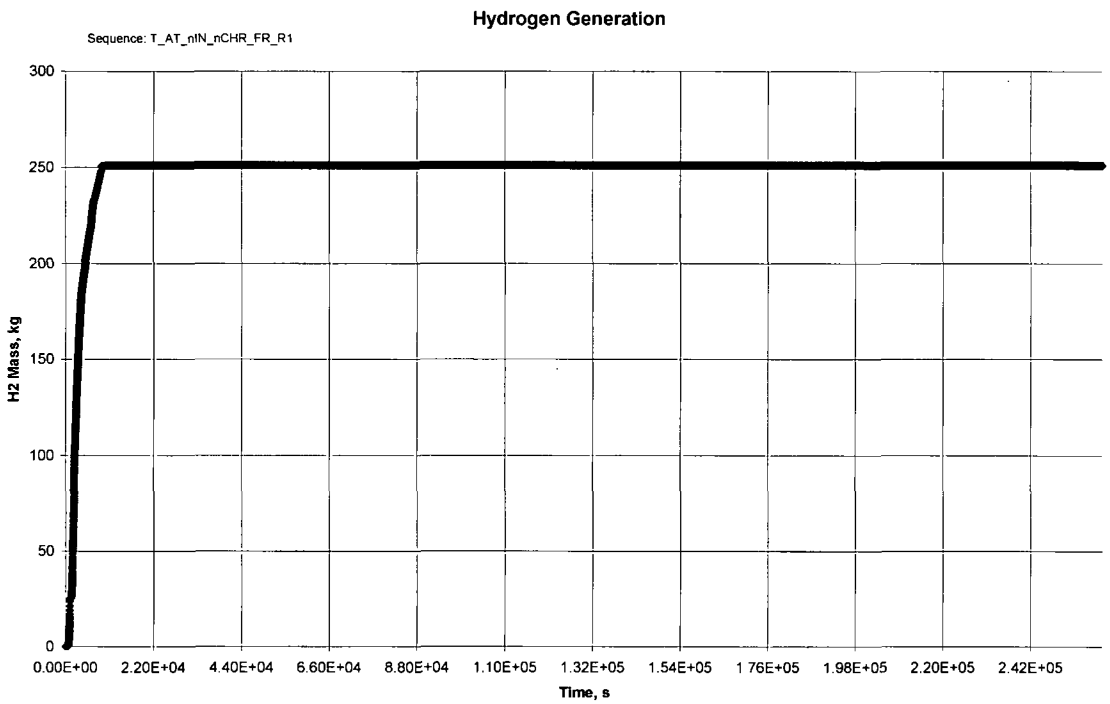

Figure 8B-7q – CDIV FR Core Hydrogen Generation

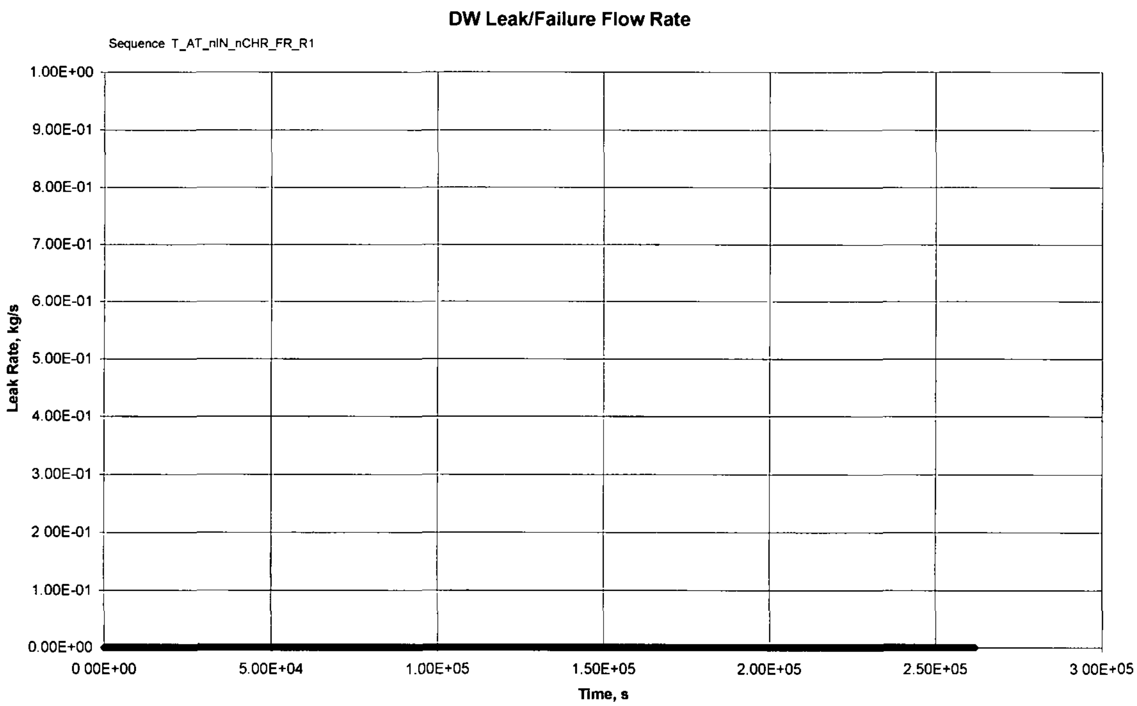

Figure 8B-7r – CDIV FR Drywell Leak/Failure Flow Rate

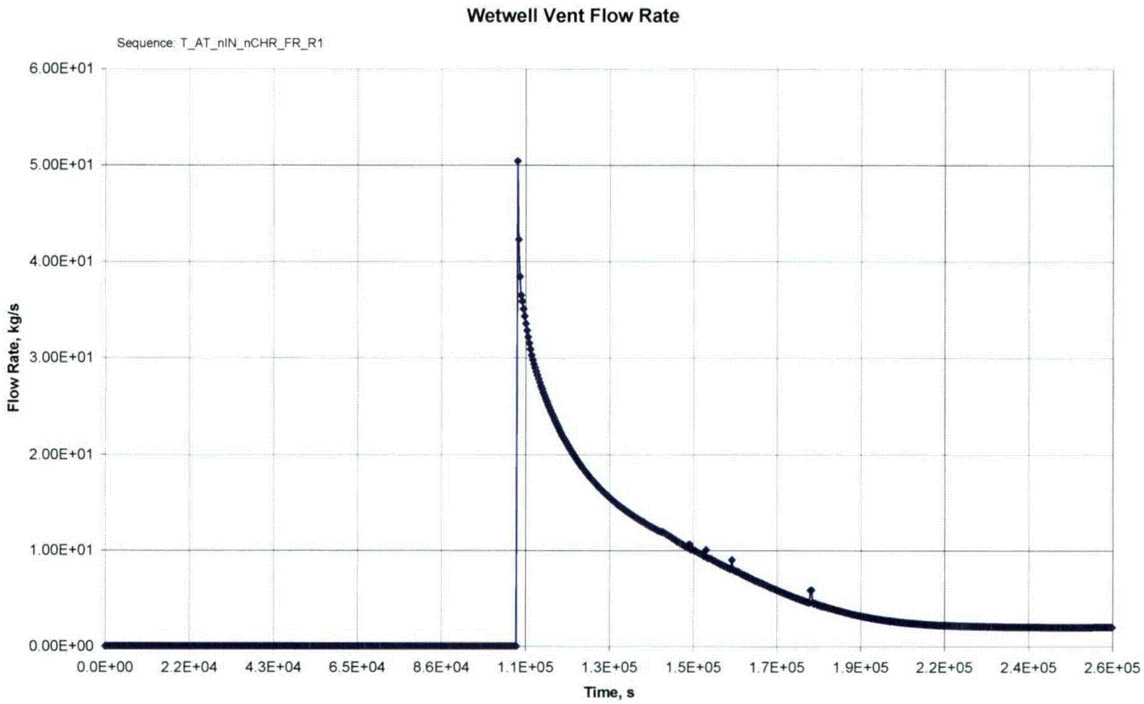

Figure 8B-7s – CDIV FR Wetwell Vent Flow Rate

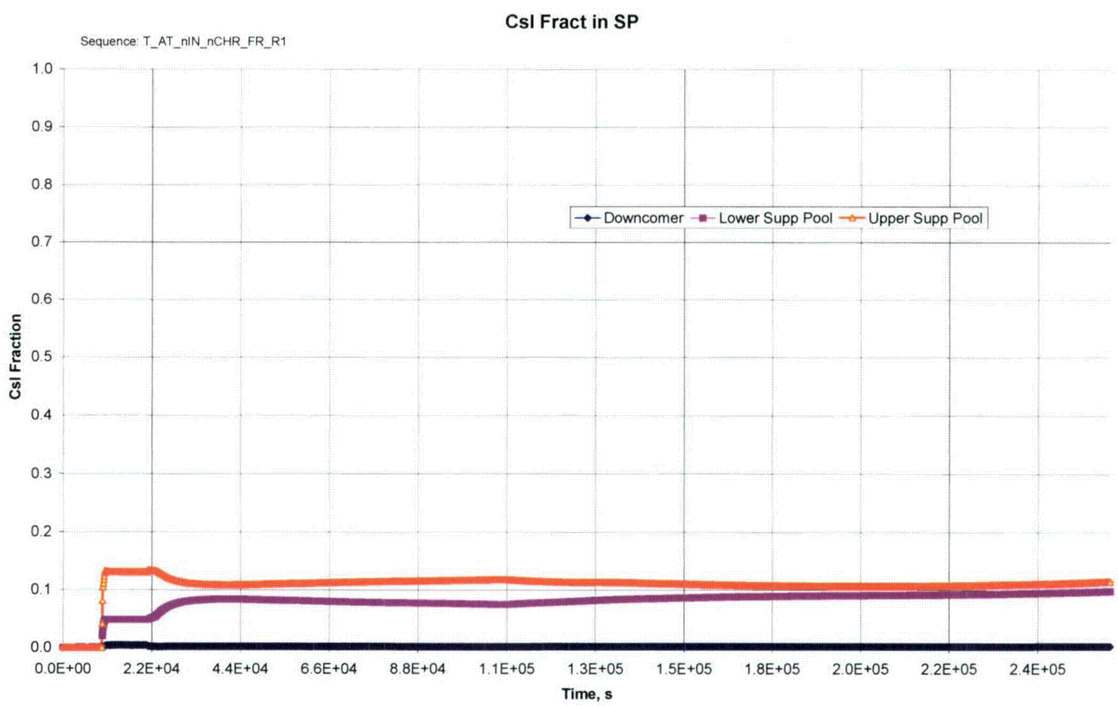

Figure 8B-7t – CDIV FR Suppression Pool CsI Fraction

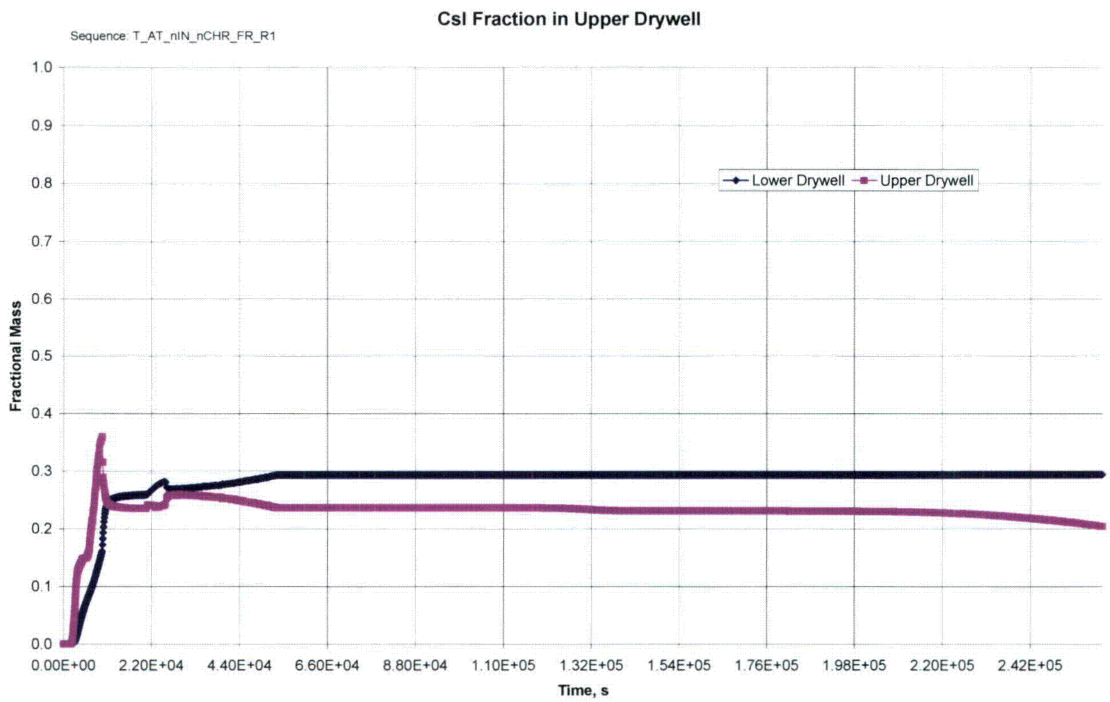

Figure 8B-7u – CDIV FR Upper Drywell CsI Fraction

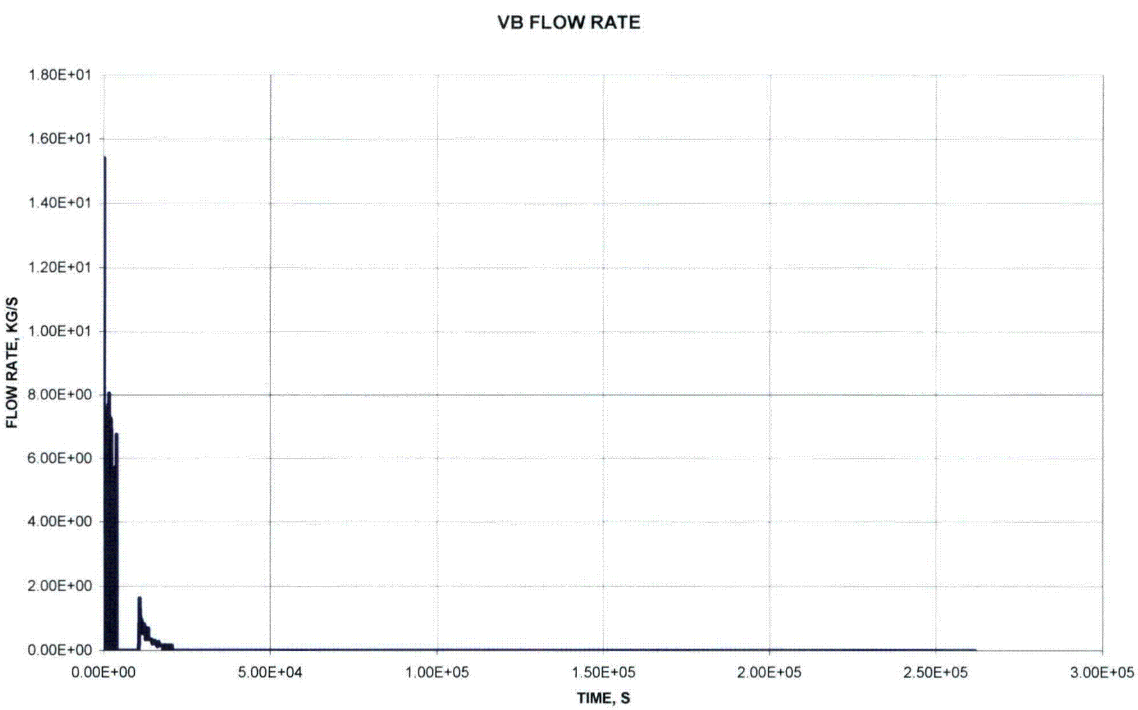

Figure 8B-7v – CDIV FR Vacuum Breaker Flow Rate

Table 8B-1

Summary of Stresses and Strains (Deleted)

Table 8B-2

Summary of Pressure Capabilities of Various Components of the RCCV and the Drywell Head (Deleted)

Table B.8-3

Probability of Failure Versus Pressure

(Median Pressure = 1.632 MPa gauge, β_c= .16) (Deleted)

Figure 8B-1. ANSYS Model (Deleted)

Figure 8B-2. Torispherical Head Buckling Test Data (Deleted)

Figure 8B-3. Torispherical Head Buckling Test Data Statistical Distribution (Deleted)

Figure 8B-4. Definition of Squeeze for Seals (Deleted)

Figure 8B-5. Cumulative Containment Failure Probability (Deleted)

9 SOURCE TERMS

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9 SOURCE TERMS

As discussed in Sections 8 and 21, the containment response to a severe accident is depicted by the end states of containment event trees. These end states become the "release categories" that are used to characterize potential source terms. The source terms will be used in the offsite consequence analysis presented in Section 10.

Table 9-1 summarizes the ESBWR release categories and associated frequencies. As indicated in the table, the release category "TSL", which depicts an intact containment with only leakage providing a source term, is the most likely release category. Other release categories have much lower calculated frequencies. For conservatism, a truncation frequency was used to represent some of these release categories. Specifically, if the calculated probability of the category was less than 10^{-15}, the truncation value of 10^{-15} was carried forward for the consequence evaluation.

The source term evaluation was performed with the MAAP computer code, which produces the distribution of radionuclides released to the environment as a function of time. Each release category is represented by one or two severe accident sequence that was selected and modeled to represent the group of potential severe accidents that could be associated with that release category. In some cases, both low pressure and high pressure classes were selected for the same release category to represent broader and more thorough contribution of accident sequences. If multi sequences were selected for a given release category, each sequence is weighted by its sequence frequency contribution to the sequence class.

The selection is based on several factors, including the frequency of the various sequences that lead to the end state and ensuring that the associated source term calculations are reasonably bounding.

The selected sequence provides a conservative basis for the source term quantification. The following sections describe the representative sequences and the bases for choosing them. As indicated in the following sections, conservative assumptions were typically made to account for analytical and phenomenological uncertainties. Table 9-1 includes the representative MAAP sequences as well as the time of initial release, and cumulative release fractions of noble gas and CsI at 24 and 72 hours after onset of core damage. Tables 9-2 and 9-3 provide the radionuclide release spectrum for 24 and 72 hours after onset of core damage, respectively.

Appendix 9A presents additional documentation on MAAP cases used for source term calculations

9.1 BREAK OUTSIDE OF CONTAINMENT (BOC)

The release category "Break Outside-of-Containment" represents sequences in which the RPV communicates directly with the environment due to an unisolated piping break that connects the RPV directly to an area outside of containment. From the Level 1 PRA, three outside-containment break locations contributed to the core damage frequency: breaks in a feedwater line, breaks in a Main Steam Line (MSL) and breaks in a RWCU/SDC line. The RWCU/SDC break event tree includes both a mid-level connection to the RPV and a lower head drain line connection. Although the largest contribution to outside-containment break is associated with the feedwater line, selecting the RWCU/SDC pipe break is conservative because its lower elevation in the RPV results in a more rapid loss of coolant inventory. Both the mid-level location and the lower drain line location were selected to represent the BOC release category.

Therefore, the representative sequences for this category is "BOCsd_nIN" and "BOCdr_nIN. This are unisolated break outside of containment in the shutdown cooling piping followed by no injection into the RPV. In these scenarios, the release begin at the onset of fuel damage and proceeds directly to the environment.

The third BOC class, Main steam line breaks, are a full order of magnitude less likely than the FDW breaks (Table 7.2-1). Also, MSL breaks are connected to the RPV at high elevation, and as such do not result in as bounding a scenario as the RWCU/SDS lines.

9.2 CONTAINMENT BYPASS (BYP)

The release category "Bypass" represents those sequences in which containment isolation has not occurred due to failure of the Containment Isolation System (CIS) function. Thus, there is a direct path from the containment atmosphere to the environment when the severe accident is initiated.

To determine the source term, a large diameter pipe opening was assumed from the time of accident initiation. Sequences in which the RPV is depressurized generally result in an earlier time to core uncovery than those involving failure to depressurize. As a result, the source term is generated earlier and the containment radionuclide concentration is developed earlier because of the path through the DPVs into containment. Both a low pressure sequence and high pressure sequence are selected to represent a thorough cross-section of the sequences. Because of the reliability of the deluge system (i.e., the probability of BYP with failed deluge is below the truncation level), the representative sequences are modeled with deluge success and are termed as "T_nIN_BYP" and "T_nDP_nIN_BYP". In these scenarios, the releases begin at the onset of fuel damage and proceeds directly to the environment.

9.3 CORE-CONCRETE INTERACTION DRY (CCID)

The release category "Core-Concrete Interaction-Dry" applies to sequences in which the containment fails due to core concrete interaction and the lower drywell debris bed is uncovered i.e., the deluge function is unsuccessful.

In these sequences, the core-concrete interaction is not limited by water cooling the debris bed, nor is the radionuclide release limited by the potential scrubbing action of an overlying water pool. Sequences in which the RPV is not depressurized may result in earlier RPV failure, thus initiating earlier CCI. To represent more accurate risk contribution, both a low pressure sequence and a high pressure sequence were selected to represent the CCID source term category. The sequences are termed as "T_nIN_nD_CCID" and "T_nDP_nIN_nD_CCID" to indicate transients failure of injection and the deluge functions.

9.4 CORE-CONCRETE INTERACTION-WET (CCIW)

The release category "Core-Concrete Interaction-Wet" applies to sequences in which the containment fails due to core concrete interaction even though the lower drywell debris bed is covered with water. In such sequences, the deluge system has functioned to cover the debris bed with water, but the BiMAC is not successful in assuring debris bed cooling. The extent of water penetration into the debris bed, independent of the BiMAC, and thus, the potential for debris bed cooling, is subject to assumption. In the worst-case hypothetical condition, the debris bed is impermeable by the overlying water pool and the extent of CCI could approach that of a dry debris bed. To address this uncertainty associated with the debris bed coolability, the debris bed was modeled as being impermeable, thus maximizing the core-concrete interaction that could occur with an overlying water pool. Unlike the CCID release category, the overlying water pool is present, which provides the potential for scrubbing of the radionuclides evolved from the debris bed.

The representative sequences are termed as "T_nIN_CCIW" and "T_nDP_nIN_CCIW" and differ from the representative CCID sequences only in that the deluge system functions.

9.5 DIRECT CONTAINMENT HEATING (DCH)

The release category "Direct Containment Heating" applies to sequences in which the RPV fails at high pressure and a significant DCH event occurs. From Section 21.3, catastrophic containment failure due to DCH is physically unreasonable. Local damage to the liner in the lower drywell will be studied as a sensitivity case in Section 11. As such no DCH sequence is selected for the baseline case.

9.6 EX-VESSEL STEAM EXPLOSION (EVE)

The release category "Ex-vessel Steam Explosion" applies to sequences in which the RPV fails at low pressure and a significant steam explosion occurs. As indicated in Section 21.4, containment leak tightness and failure of the BiMAC function is physically unreasonable for all but 1% of the sequences contributing to the core damage frequency. A conservative approach was used to develop the source term associated with an EVE, specifically:

- Liner damage was assumed to be significant enough to result in containment depressurization, which occurs at the time of RPV failure,
- No credit was taken for mitigation of the release; i.e., liner damage was assumed to result in direct communication with the environment, and
- Due to uncertainties about potential equipment damage and the distribution of water through containment after the EVE, no credit is taken for a lower drywell water pool that would minimize the source term.

The dominant Class I sequence, a transient with no injection and successful RPV depressurization, provided the basis for this category. To address the preceding points, the sequence was modeled with deluge failure and containment failure occurring at the time of RPV failure. The representative sequence is termed "T_nIN_nD_EVE".

9.6.1 EVE (CCIW*) (Deleted)

9.7 FILTERED RELEASE (FR)

The ESBWR design includes the potential to manually vent the containment from the suppression chamber air space. This action may be implemented to limit the containment pressure increase if containment heat removal fails or core-concrete interaction generates enough non-condensables to overpressurize the containment. Venting the suppression chamber forces the radionuclides through the suppression pool, which reduces the magnitude of the source term.

To represent the FR category, a sequence with failure to insert negative reactivity was conservatively selected because such a sequence would pressurize containment more quickly than the much more probable non-ATWS sequences. The sequence assumes RPV failure at low pressure, consistent with the discussion in Section 8.2.1.1. Operator guidance regarding venting has not been developed, but it is assumed that venting would be delayed until containment integrity is threatened. The analysis assumes that venting does not occur until the containment pressure reaches 90% of the containment ultimate strength. No credit was given in the analysis for closing the vent after reducing the containment pressure. The representative sequence is termed "T-AT_nIN _nCHR_FR".

9.8 OVERPRESSURE-VACUUM BREAKER (OPVB)

The release category "OPVB" applies to sequences in which vacuum breaker failure has occurred. Failure of vacuum breakers to close, or to be open in a pre-existing condition, results in failure of the containment pressure suppression function, which in turn also fails containment heat removal. Thus, such sequences would be expected to result in an earlier release than overpressure sequences with failure of containment heat removal alone.

To represent more broad contribution of both high and low pressure sequences, two representative sequences are selected for this category. The event trees illustrate that the OPVB category is logically reached only if deluge/BiMAC function successfully. Thus, the sequences termed as "T_nDP_nIN_VB" and "T_nIN_VB" are used to represent the OPVB release category.

9.9 OVERPRESSURE- EARLY CONTAINMENT HEAT REMOVAL LOSS (OPW1)

The release category "OPW1" applies to sequences in which containment heat removal fails within 24 hours after event initiation. A sequence with RPV failure at high pressure was selected to represent this release category because RPV failure generally occurs earlier than if the vessel were depressurized and the loss of containment heat removal failure probability is higher than for low pressure sequences. Thus, the representative sequence becomes "T_nDP_nIN_nCHR_W1". Containment heat removal is conservatively assumed to be unavailable for the duration of the sequence.

9.10 OVERPRESSURE- LATE CONTAINMENT HEAT REMOVAL LOSS (OPW2)

The release category "OPW2" applies to sequences in which containment heat removal fails after the period covered by OPW1 (post-24 hours) and up to 72 hours after onset of core damage. In such sequences, the passive PCCS system becomes unavailable after 24 hours due to failure to connect to a supplemental water pool; FAPCS availability is also evaluated at this time. The representative sequence is the same as that used for OPW1 except that containment heat removal is terminated at 24 hours after event initiation, consistent with the PCCS design basis. The representative sequence is termed "T_nDP_nIN_nCHR_W2".

9.11 TECHNICAL SPECIFICATION LEAKAGE (TSL)

The category "Technical Specification Leakage" applies to sequences in which the containment is intact and the only release is due to the maximum leak rate allowed by Technical Specifications. Sequence T_AT_nIN_TSL was selected as representative of this category because the core damage time is relatively early for ATWS sequences. For additional conservatism, the area of containment leakage corresponding to the maximum allowable Technical Specification leak rate was doubled to produce the representative source term used for this release category. The representative source term is termed "T-AT_nIN_TSL2x".

9.12 SUMMARY

Potential release categories were defined in Sections 8 and 21. The source terms associated with each release category were developed using MAAP simulations of a representative sequence. Conservative assumptions were used in the selection and simulation of the representative sequence. Table 9-1 summarizes the release category, representative sequence and the cumulative release fractions for noble gases and CsI. Table 9-2 provides source terms for the period 24 hours after onset of core damage. Table 9-3 provides source terms for the period 72 hours after onset of core damage. The source terms and associated release category frequencies are used in the offsite consequence analysis described in Section 10.

Table 9-1

Release Categories

Source Term	Release Category	MAAP CASE	Total Release Frequency (per year)	Time of Plume Release (hr)	NG Release Fraction 24 hrs after onset of core damage	CsI Release Fraction 24 hrs after onse of core damage	NG Release Fraction 72 hrs after onset of core damage	CsI Release Fraction 72 hrs after onse of core damage
1	BOC	BOCsd_nIN_R1	1.47E-10	0.7	9.7E-01	7.0E-01	9.8E-01	7.0E-01
2		BOCdr_nIN_R1		0.6	2.4E-01	1.1E-01	2.6E-01	1.3E-01
3	BYP	T_nIN_BYP_R1	5.6E-11	0.7	9.5E-01	2.1E-01	9.7E-01	3.0E-01
4		T_nDP_nIN_BYP_R1		1.3	5.3E-01	3.3E-02	6.8E-01	3.5E-02
5	CCID	T_nIN_nD_CCID_R1	ε	25.8	0.0	0.0	9.1E-01	6.2E-02
6		T_nDP_nIN_nD_CCID_R1		16.0	9.1E-01	6.7E-02	9.6E-01	3.5E-01
7	CCIW	T_nIN_CCIW_R1	9.9E-11	25.6	0.0	0.0	8.9E-01	1.6E-05
8		T_nDP_nIN_CCIW_R1		18.4	6.4E-01	1.2E-04	8.3E-01	1.1E-02
9	EVE	T_nIN_nD_EVE_R1	6.10E-10	7.4	8.3E-01	2.8E-02	8.3E-01	1.5E-01
10	FR	T-AT_nIN_nCHR_FR_R1	ε	28.9	0.0E+0	0.0E+0	1.0E+00	6.1E-03
11	OPVB	T_nDP_nIN_VB_R1	6E-12	13.8	4.3E-01	1.33E-04	9.6E-01	4.1E-03
12		T_nIN_VB_R1		8.7	8.6E-01	5.0E-03	1.0E+00	1.5E-02
13	OPW1	T_nDP_nIN_nCHR_W1_R1	ε	34.2	0.0	0.0	1.0E+00	1.5E-02
14	OPW2	T_nDP_nIN_nCHR_W2_R1	ε	53.1	0.0E+0	0.0E+0	1.0E+00	1.5E-02
15	TSL	T_AT_nIN_TSL2x_R1	1.12E-08	0.5	2.7E-03	1.6E-04	2.7E-03	1.6E-04

ε Less than 1E-12

Table 9-2

Radionuclide Source Terms

(Release Fraction 24 hours after onset of core damage)

Source Term	Xe/Kr	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
1	9.7E-01	7.0E-01	4.6E-01	1.3E-02	1.7E-01	3.6E-01	3.1E-02	2.5E-04	1.2E-03	4.6E-01	6.4E-04	3.0E-06
2	2.4E-01	1.1E-01	1.2E-01	4.5E-04	1.6E-02	3.3E-02	2.0E-03	3.1E-05	1.4E-04	5.7E-02	1.1E-06	1.0E-06
3	9.5E-01	2.1E-01	1.3E-01	4.6E-03	6.2E-02	1.0E-01	1.3E-02	1.8E-04	8.5E-04	1.9E-01	5.1E-04	5.5E-06
4	5.3E-01	3.3E-02	2.0E-03	4.1E-02	2.3E-02	1.2E-02	4.0E-02	4.1E-02	4.1E-02	7.2E-02	3.6E-04	3.4E-06
5	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
6	9.1E-01	6.7E-02	5.0E-02	7.8E-07	3.4E-07	2.4E-02	7.2E-06	3.6E-07	4.7E-07	7.1E-02	1.0E-07	1.8E-07
7	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
8	6.4E-01	1.2E-04	1.9E-05	2.6E-06	2.1E-06	5.0E-05	2.7E-06	2.5E-06	2.5E-06	1.6E-04	2.4E-07	8.4E-10
9	8.3E-01	2.8E-02	7.0E-02	1.7E-03	6.5E-05	1.3E-01	7.2E-04	4.9E-05	6.6E-04	1.9E-01	4.9E-04	3.3E-06
10	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
11	4.3E-01	1.3E-04	2.0E-05	1.9E-06	1.6E-06	9.0E-05	2.0E-06	1.8E-06	1.8E-06	1.3E-04	3.6E-06	1.7E-10
12	8.6E-01	5.0E-03	8.6E-05	1.2E-05	2.3E-06	1.2E-03	6.1E-06	1.2E-06	8.7E-06	2.2E-02	3.8E-05	8.1E-08
13	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
14	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
15	2.7E-03	1.6E-04	9.9E-05	2.6E-06	6.2E-05	5.9E-05	1.3E-05	1.1E-07	3.7E-07	1.6E-04	7.5E-10	3.3E-10

Table 9-3

Radionuclide Source Terms

(Release Fraction 72 hours after onset of core damage)

Source Term	Xe/Kr	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
1	9.8E-01	7.0E-01	4.6E-01	1.3E-02	1.7E-01	3.7E-01	3.1E-02	2.5E-04	1.2E-03	5.0E-01	6.5E-04	3.0E-06
2	2.6E-01	1.3E-01	1.2E-01	4.5E-04	1.6E-02	3.6E-02	2.0E-03	3.1E-05	1.4E-04	6.0E-02	1.3E-06	1.0E-06
3	9.7E-01	3.0E-01	1.3E-01	4.6E-03	6.2E-02	1.2E-01	1.3E-02	1.8E-04	8.5E-04	3.1E-01	5.1E-04	5.5E-06
4	6.8E-01	3.5E-02	6.1E-03	4.1E-02	2.3E-02	2.5E-02	4.0E-02	4.1E-02	4.1E-02	7.5E-02	3.8E-04	3.4E-06
5	9.1E-01	6.2E-02	7.6E-02	1.1E-07	3.2E-07	1.4E-01	4.0E-06	6.9E-09	1.3E-08	3.7E-02	2.7E-07	3.6E-08
6	9.6E-01	3.5E-01	7.7E-02	8.1E-07	4.0E-07	6.2E-02	1.1E-05	3.6E-07	4.7E-07	1.4E-01	1.3E-07	2.0E-07
7	8.9E-01	1.6E-05	7.8E-07	3.3E-08	2.1E-07	2.8E-05	1.3E-07	2.2E-09	1.2E-08	3.5E-02	7.6E-07	5.0E-10
8	8.3E-01	1.1E-02	1.1E-02	2.7E-06	2.2E-06	2.8E-02	2.8E-06	2.6E-06	2.6E-06	8.1E-03	4.7E-07	1.2E-09
9	8.3E-01	1.5E-01	1.5E-01	1.7E-03	6.5E-05	2.3E-01	7.5E-04	4.9E-05	6.6E-04	2.8E-01	4.9E-04	3.4E-06
10	1.0E+00	6.1E-03	2.6E-04	7.1E-09	3.3E-08	4.0E-03	3.5E-08	5.1E-10	2.2E-09	1.6E-01	2.3E-05	1.5E-11
11	9.6E-01	4.1E-03	7.0E-03	7.5E-06	1.6E-06	1.1E-02	4.9E-06	1.8E-06	1.9E-06	6.1E-02	1.8E-05	2.5E-10
12	1.0E+00	1.5E-02	1.9E-03	1.2E-05	2.3E-06	6.8E-03	6.1E-06	1.2E-06	8.7E-06	3.3E-01	4.6E-05	8.1E-08
13	9.9E-01	3.7E-04	9.9E-04	5.6E-08	9.3E-08	8.9E-03	7.1E-08	5.4E-08	5.4E-08	3.0E-03	1.6E-07	5.7E-13
14	9.7E-01	8.5E-05	3.7E-05	1.1E-08	7.2E-09	7.6E-04	1.1E-08	1.0E-08	1.0E-08	5.1E-03	7.4E-08	3.2E-13
15	2.7E-03	1.6E-04	9.9E-05	2.6E-06	6.2E-05	5.9E-05	1.3E-05	1.1E-07	3.7E-07	1.7E-04	7.6E-10	3.3E-10

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9A REPRESENTATIVE SEQUENCES

The representative sequences analyzed during the source term calculation are shown in this appendix. Each release category has one or two release categories, as discussed in Section 9.0.

9A.1 BOC (Mid-Level1 RWCU Line) Representative Sequence (BOCsd_nIN_R1)

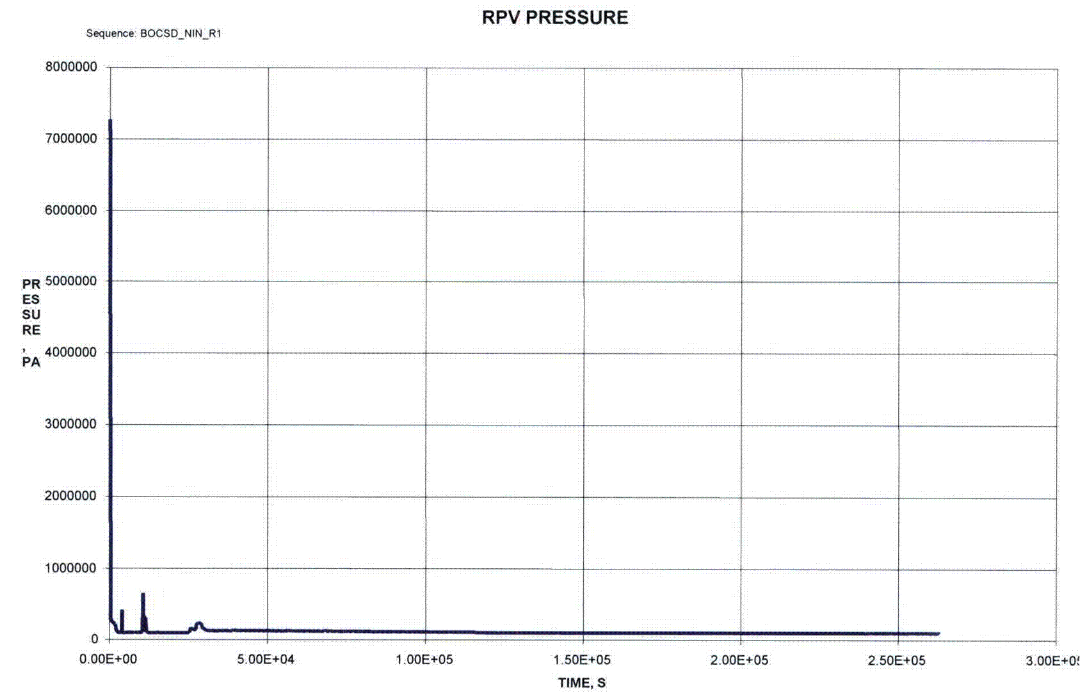


Figure 9A-1a - BOCsd_nIN_R1 RPV Pressure

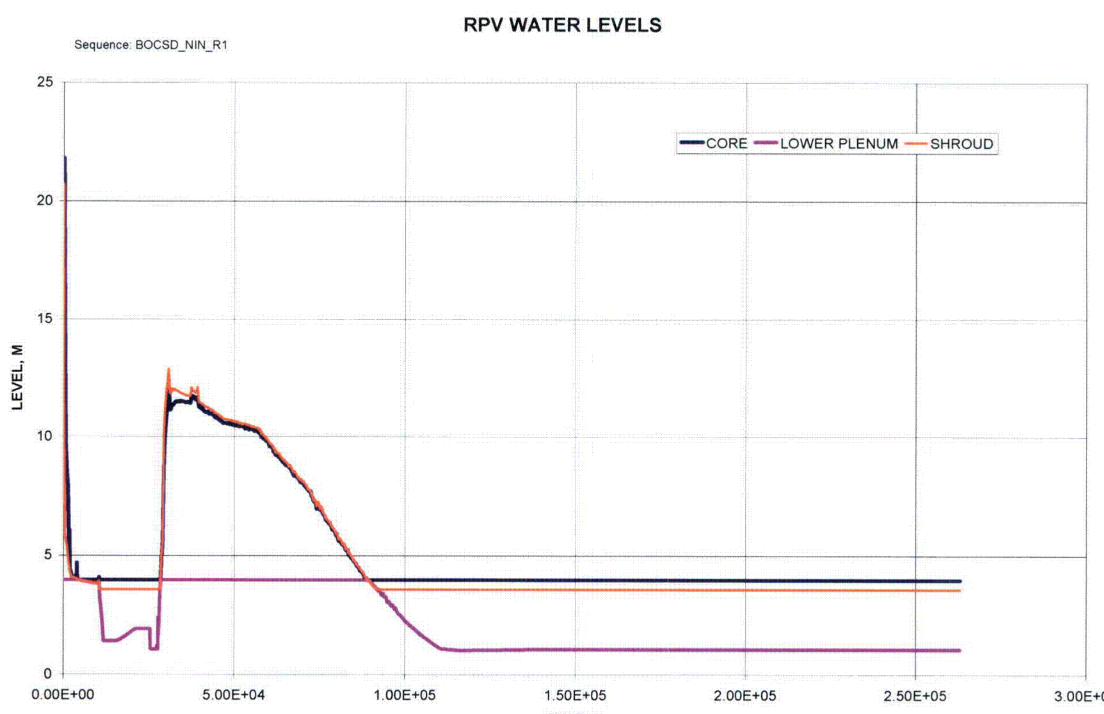


Figure 9A-1b - BOCsd_nIN_R1 RPV Water Levels

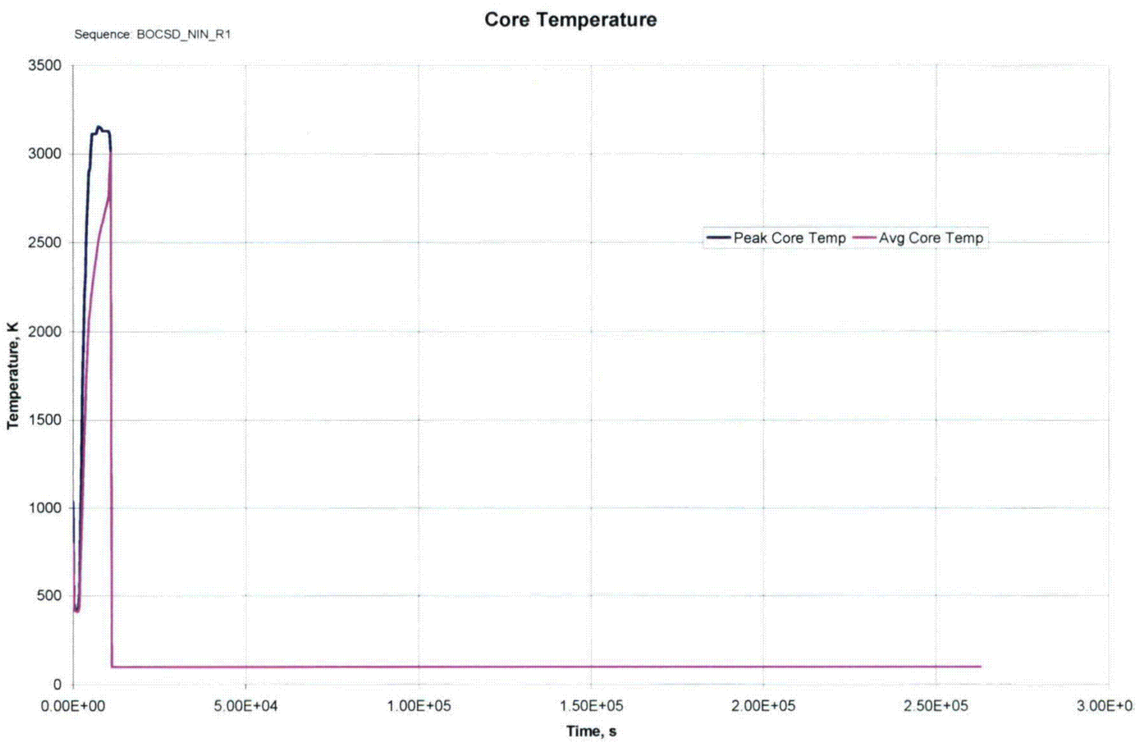


Figure 9A-1c - BOCsd_nIN_R1 Core Temperature

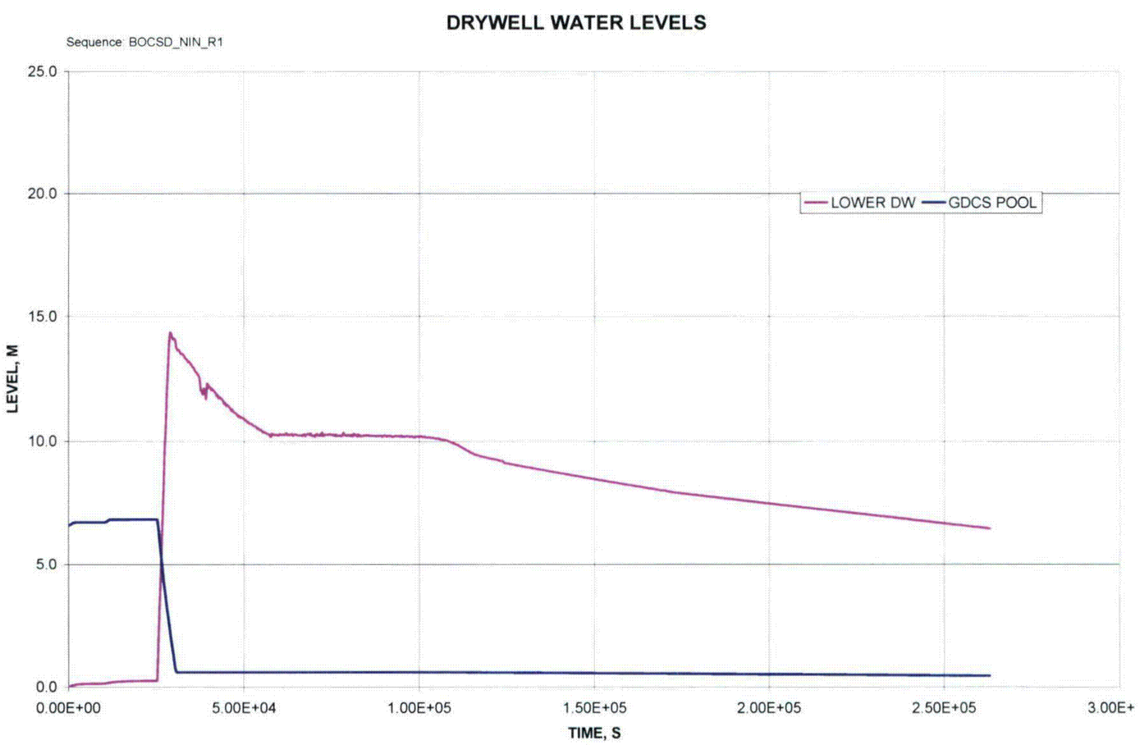


Figure 9A-1d - BOCsd_nIN_R1 Drywell Water Levels

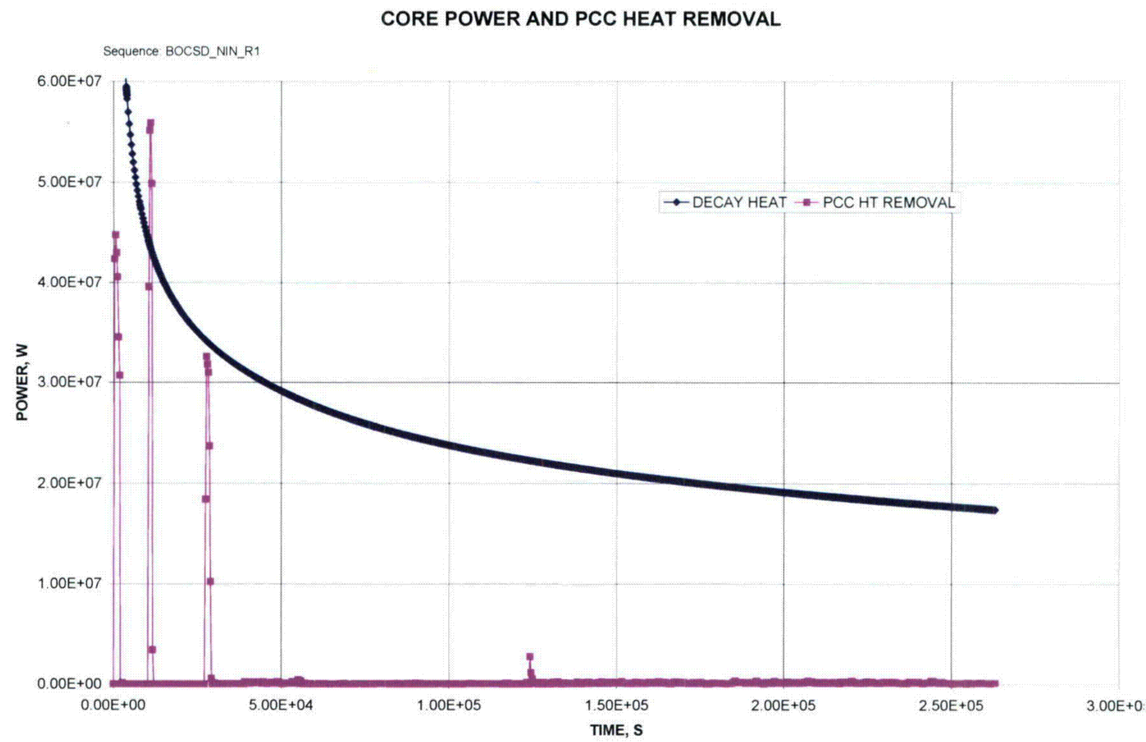


Figure 9A-1e - BOCsd_nIN_R1 Core Power and PCC Heat Removal

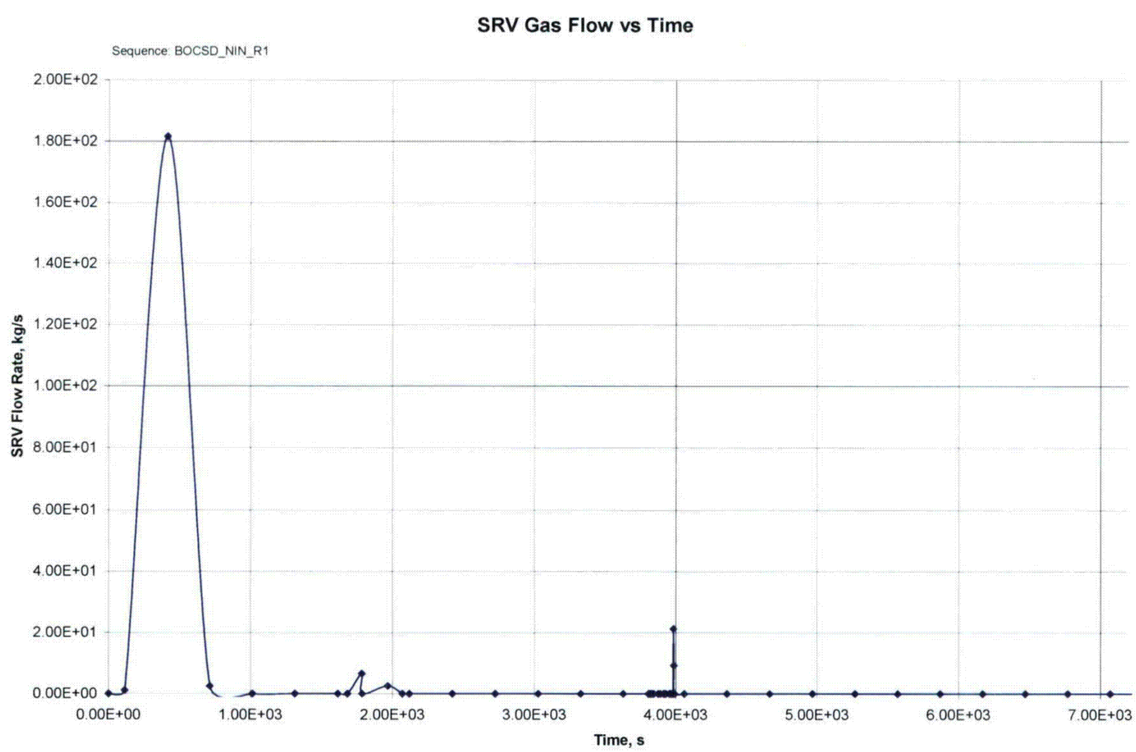


Figure 9A-1f - BOCsd_nIN_R1 SRV Gas Low vs. Time

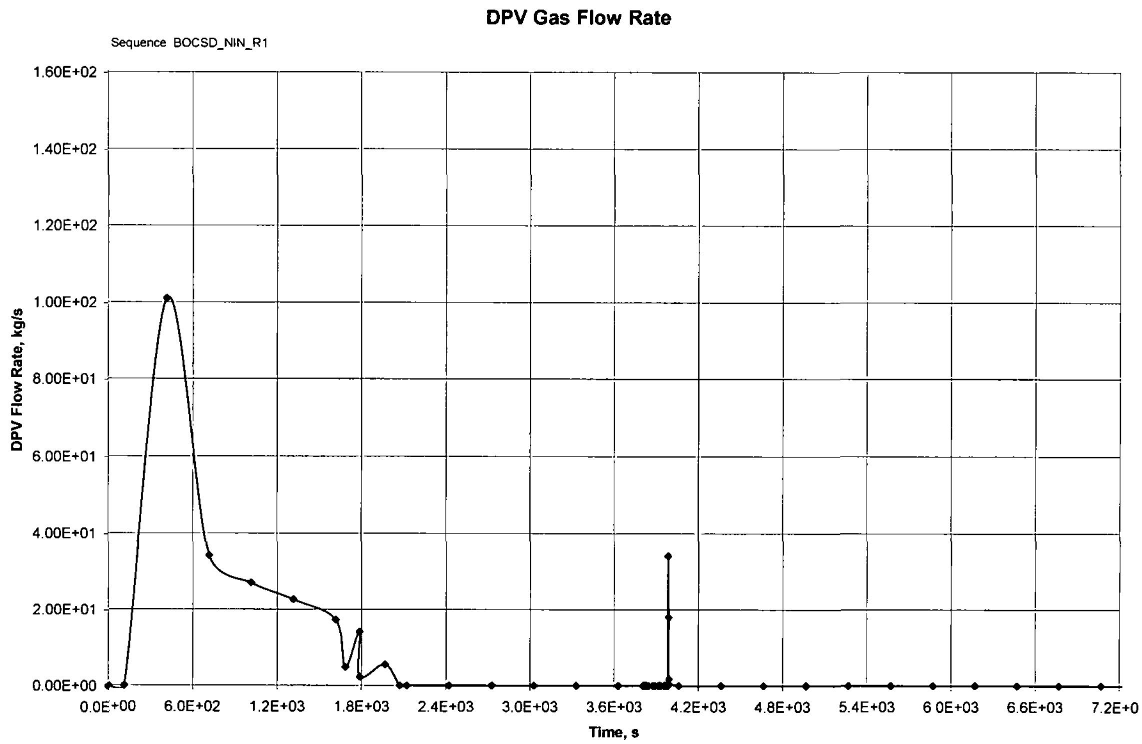


Figure 9A-1g - BOCsd_nIN_R1 DPV Gas Flow Rate

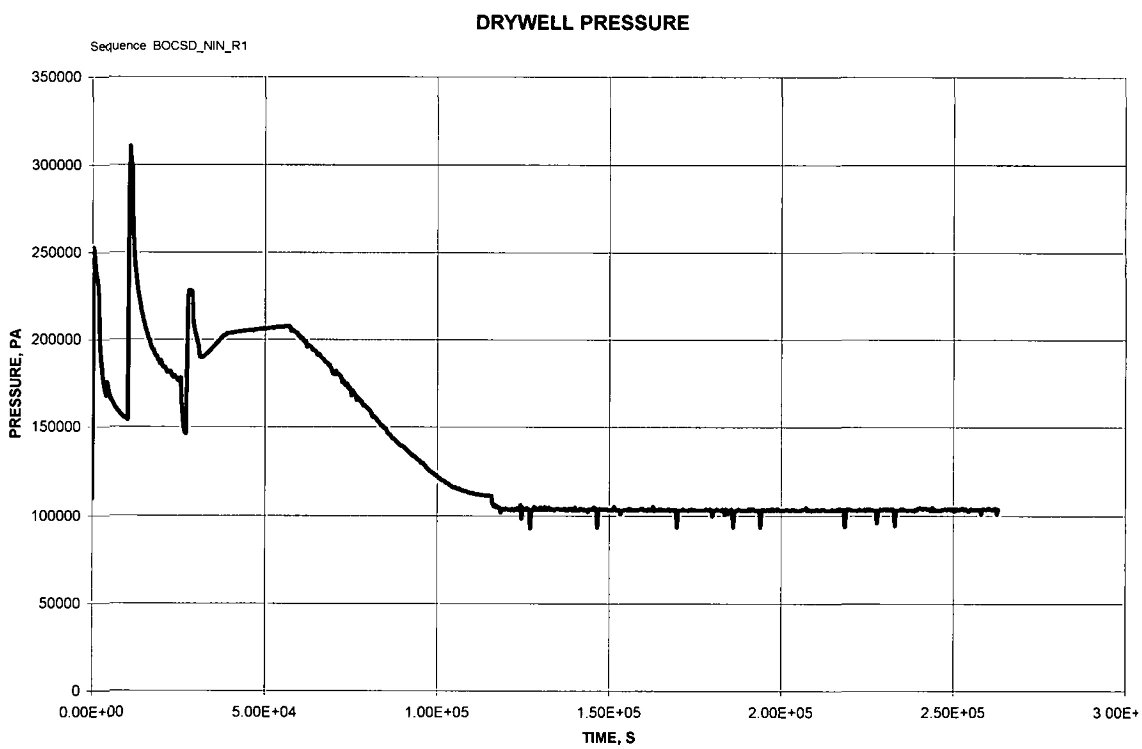


Figure 9A-1h - BOCsd_nIN_R1 Drywell Pressure

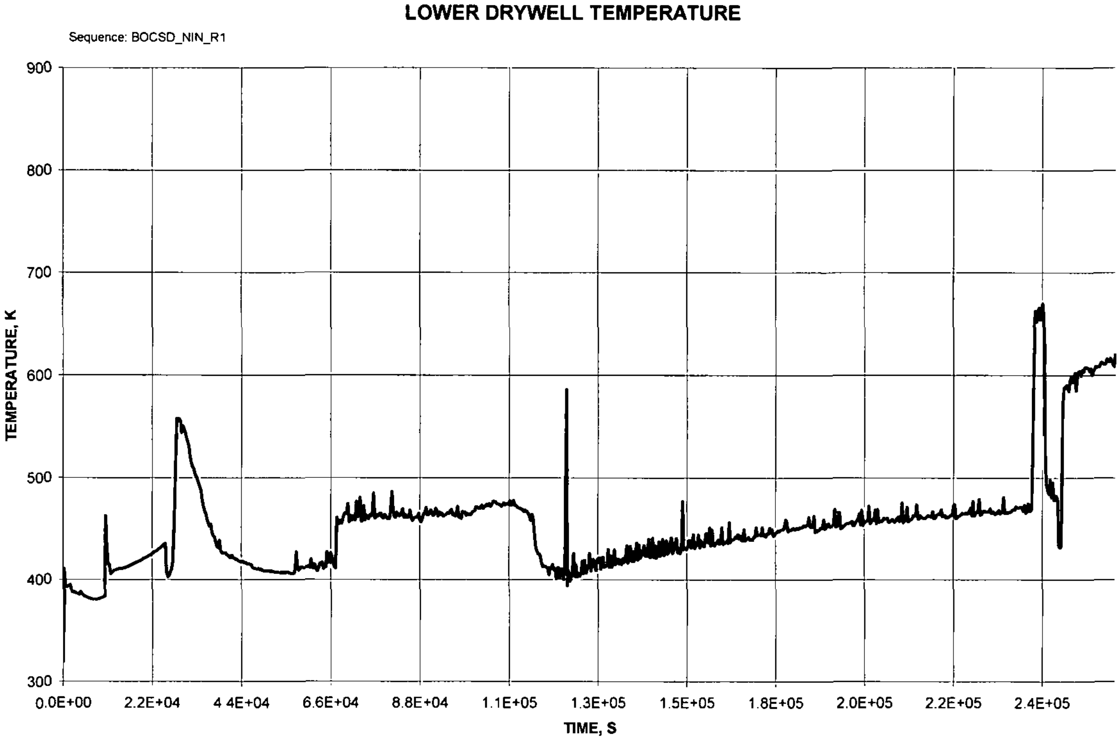


Figure 9A-1i - BOCsd_nIN_R1 Lower Drywell Temperature

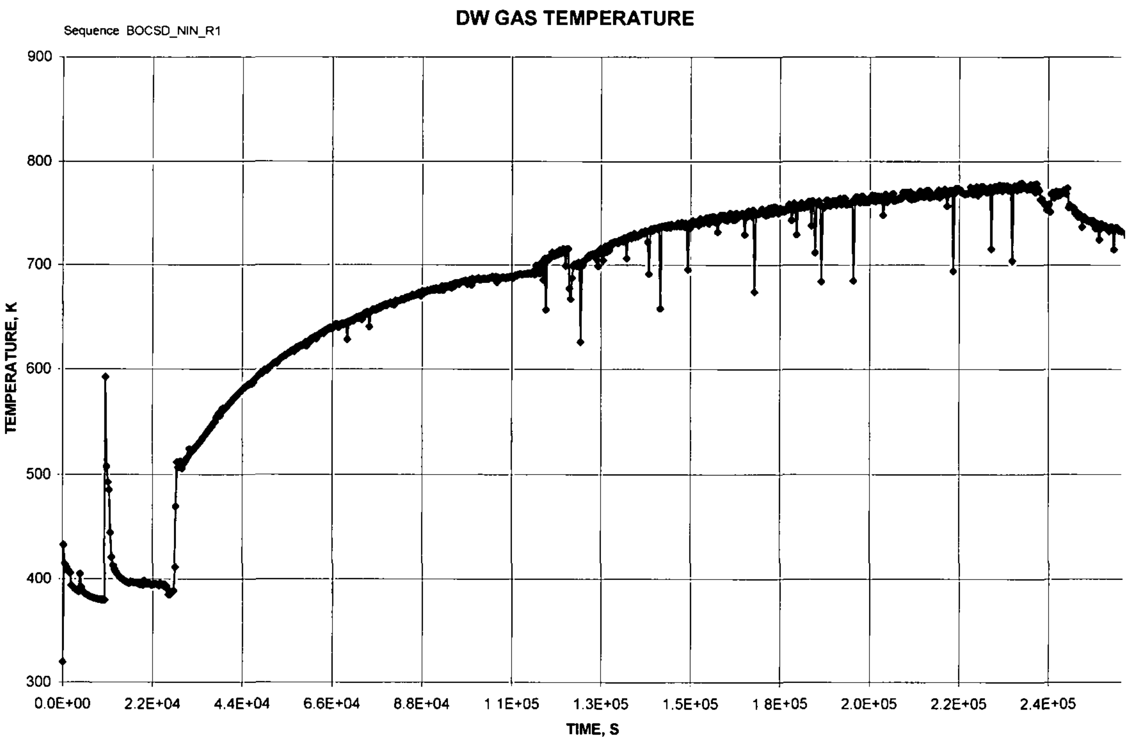


Figure 9A-1j - BOCsd_nIN_R1 DW Gas Temperature

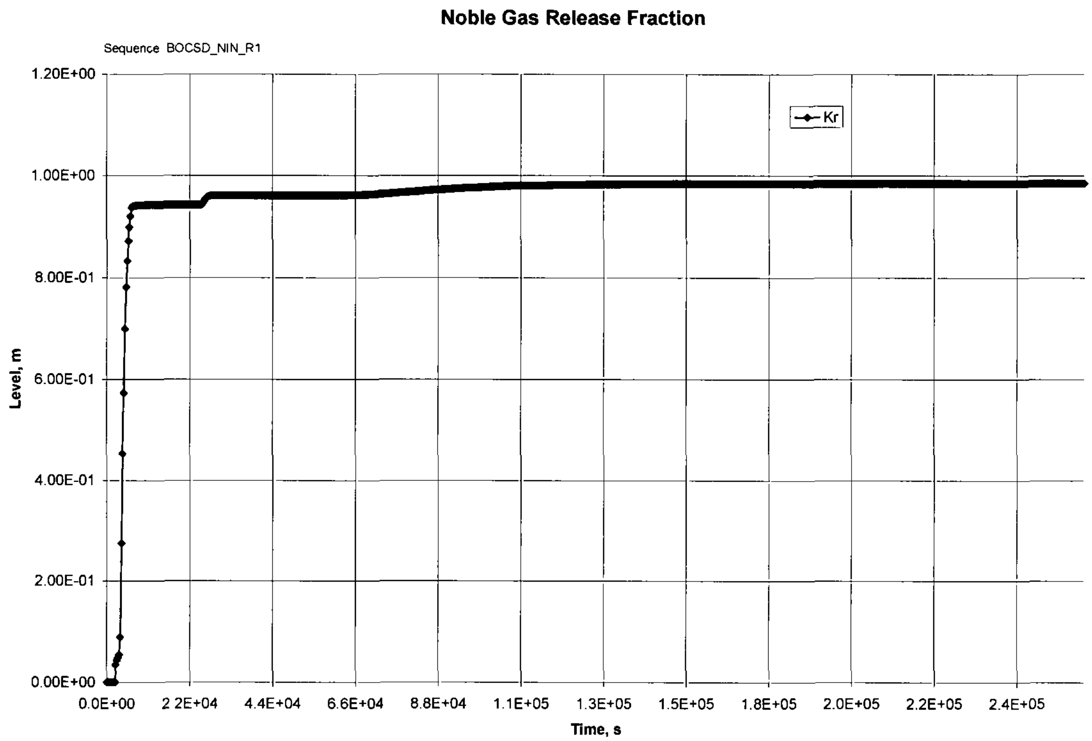


Figure 9A-1k - BOCsd_nIN_R1 Noble Gas Release Fraction

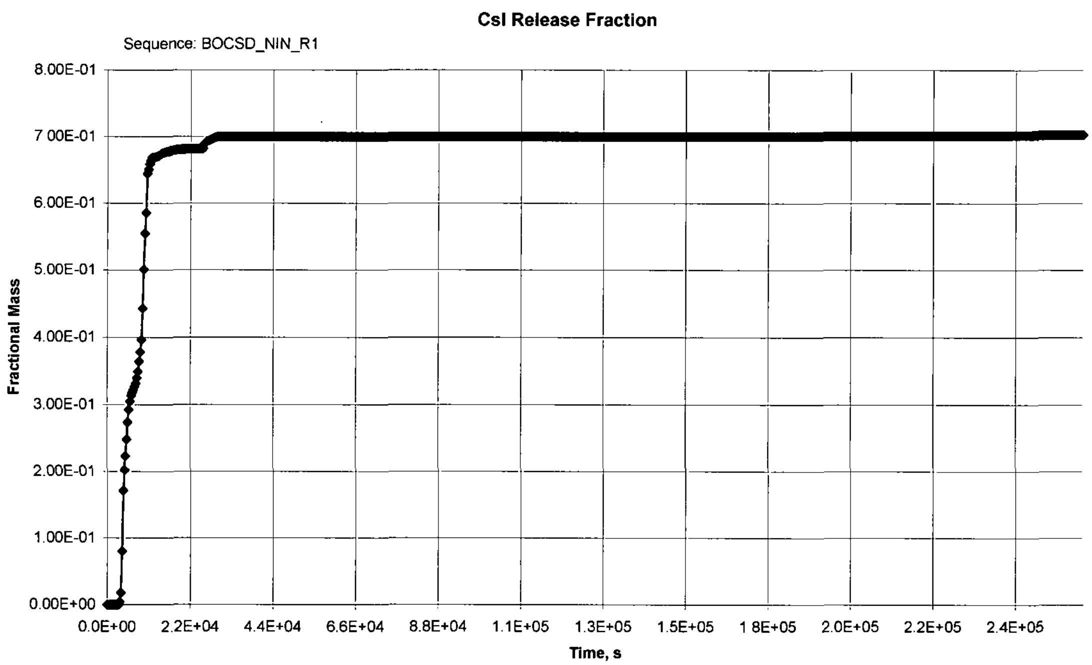


Figure 9A-1l - BOCsd_nIN_R1 CsI Release Fraction

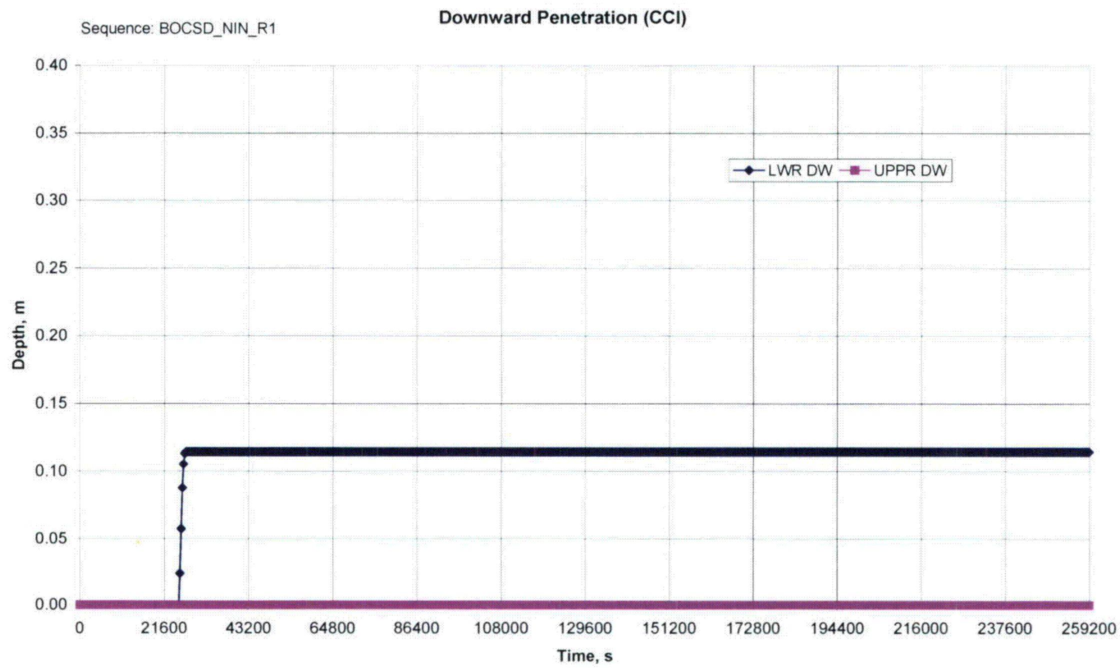


Figure 9A-1m - BOCsd_nIN_R1 Downward Penetration (CCI)

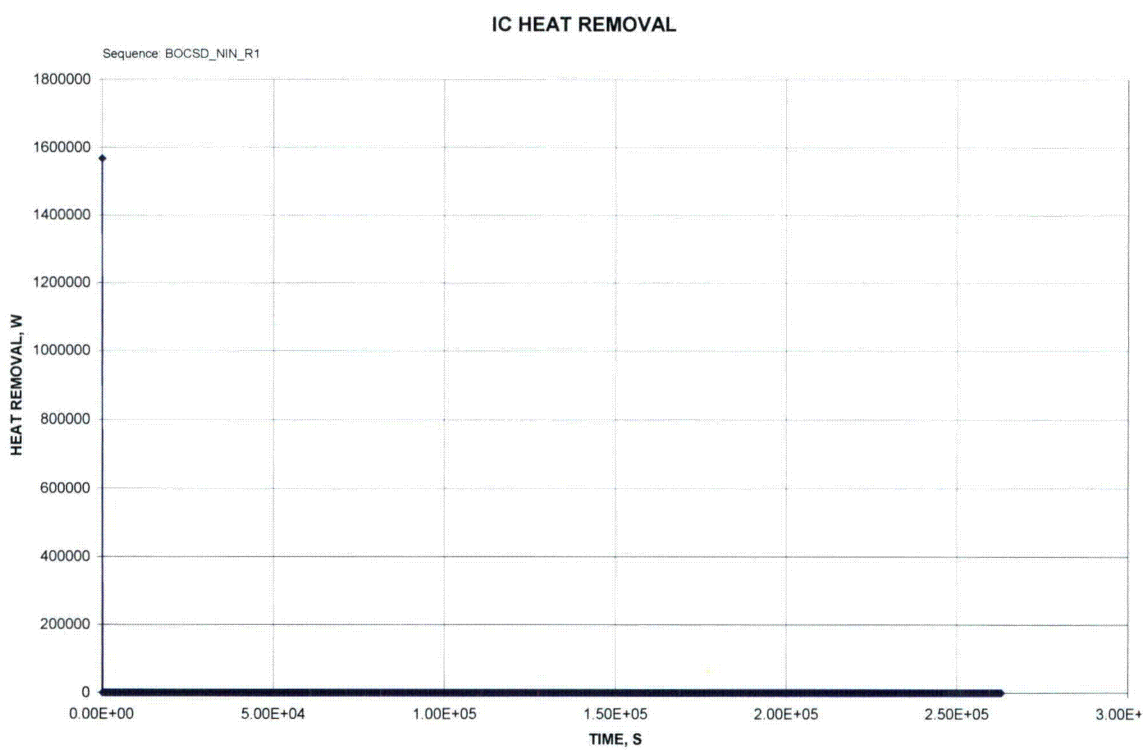


Figure 9A-1m - BOCsd_nIN_R1 IC Heat Removal

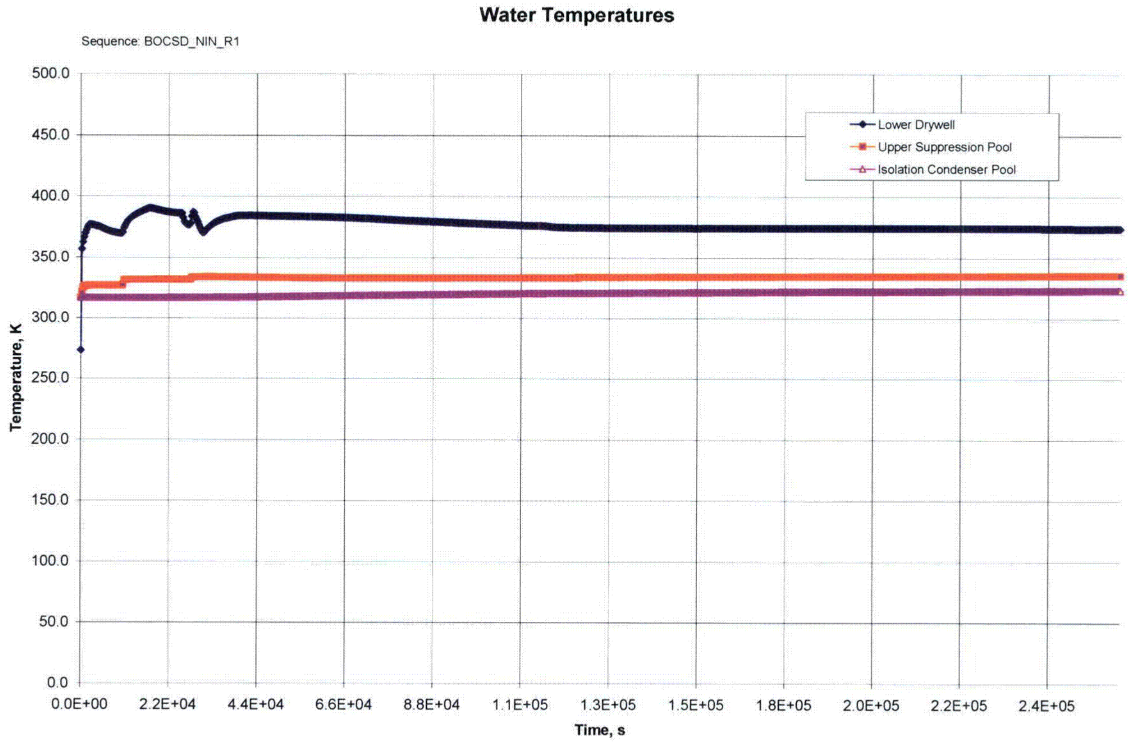


Figure 9A-1o - BOCsd_nIN_R1 Water Temperatures

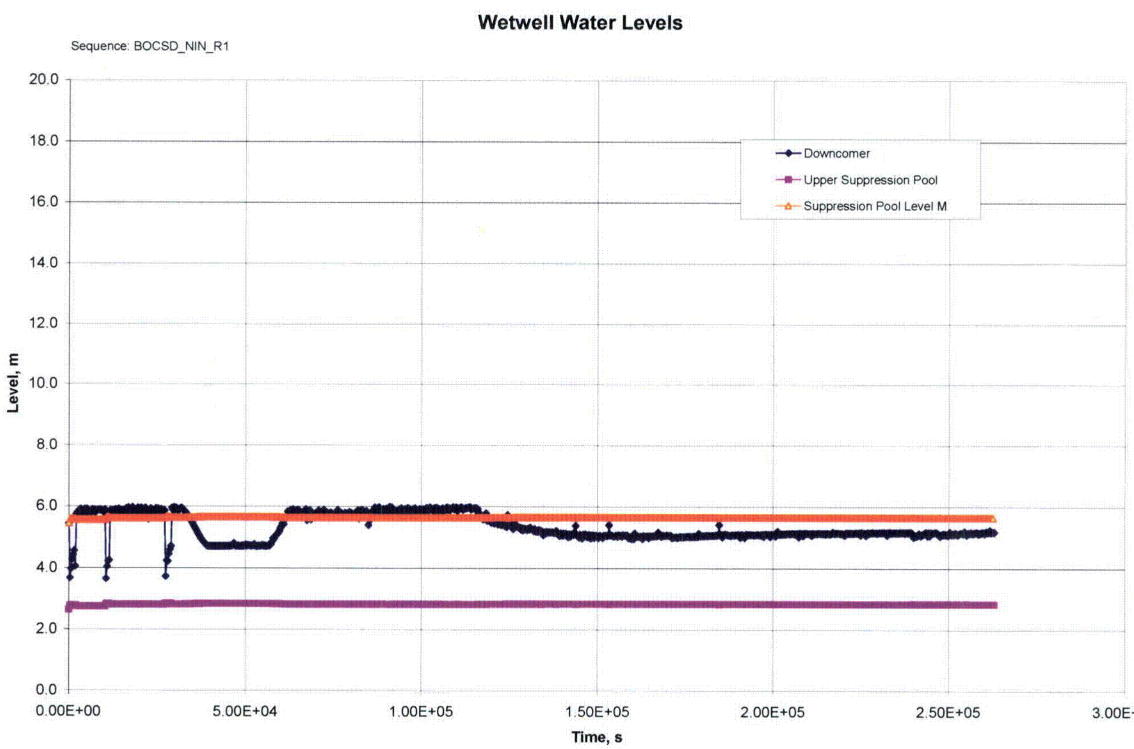


Figure 9A-1p - BOCsd_nIN_R1 Wetwell Water Levels

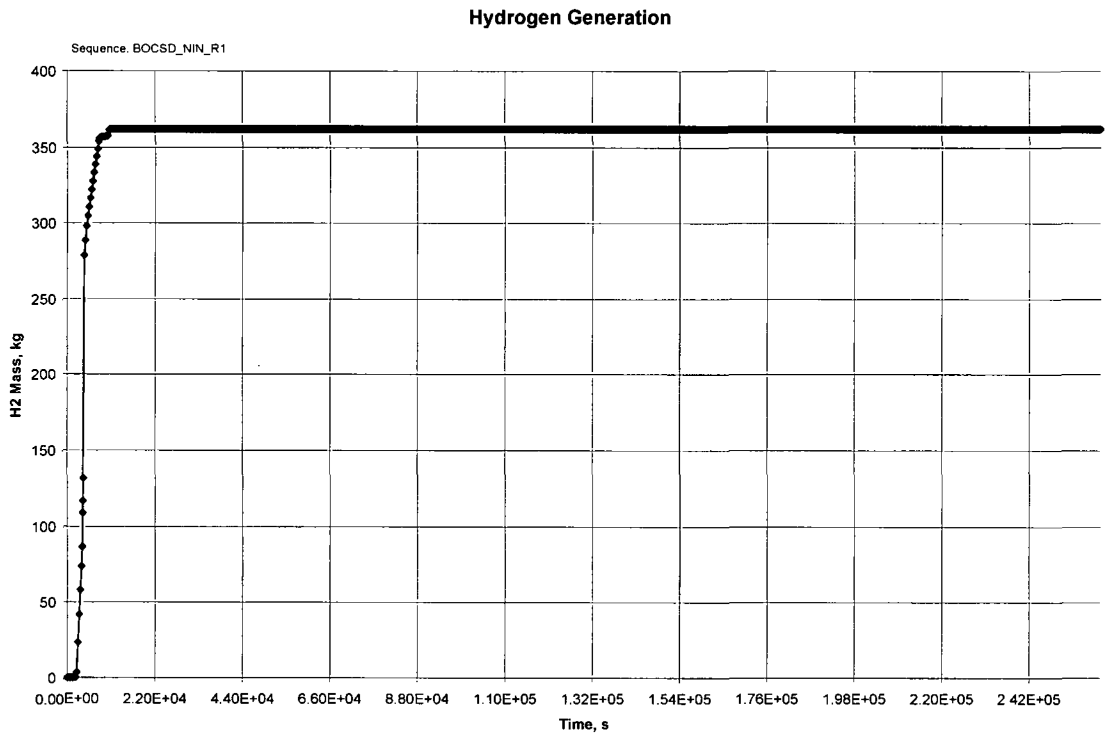


Figure 9A-1q - BOCsd_nIN_R1 Hydrogen Generation

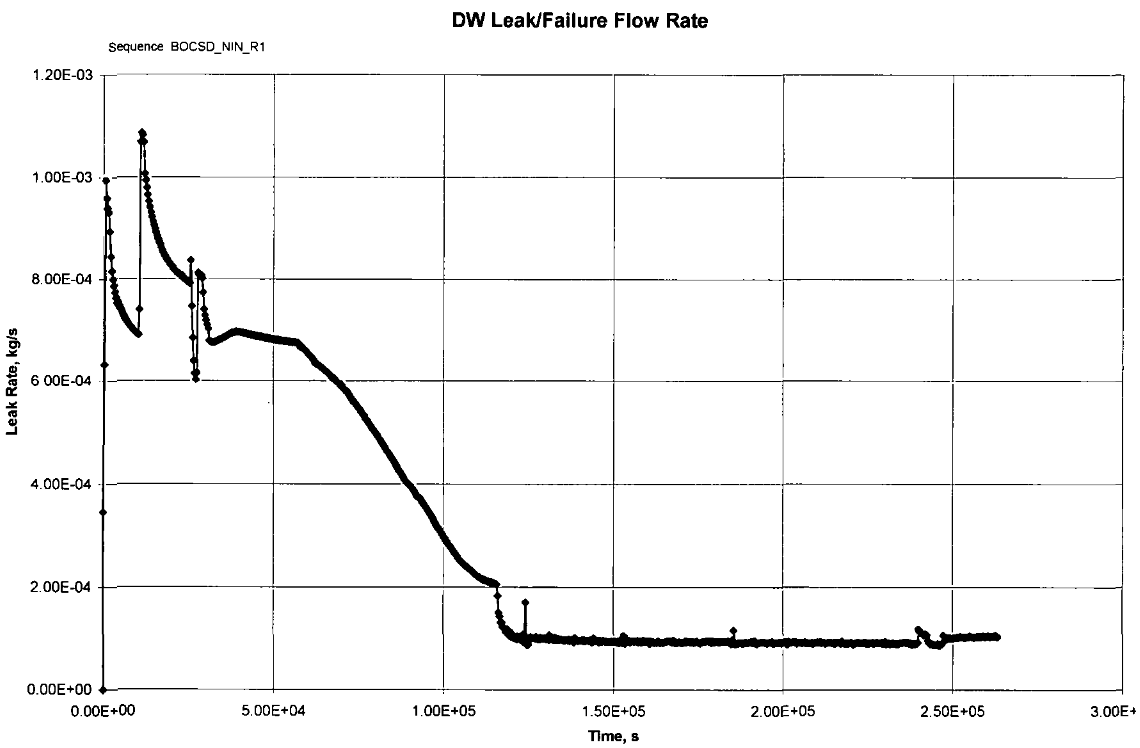


Figure 9A-1r - BOCsd_nIN_R1 DW Leak/Failure Flow Rate

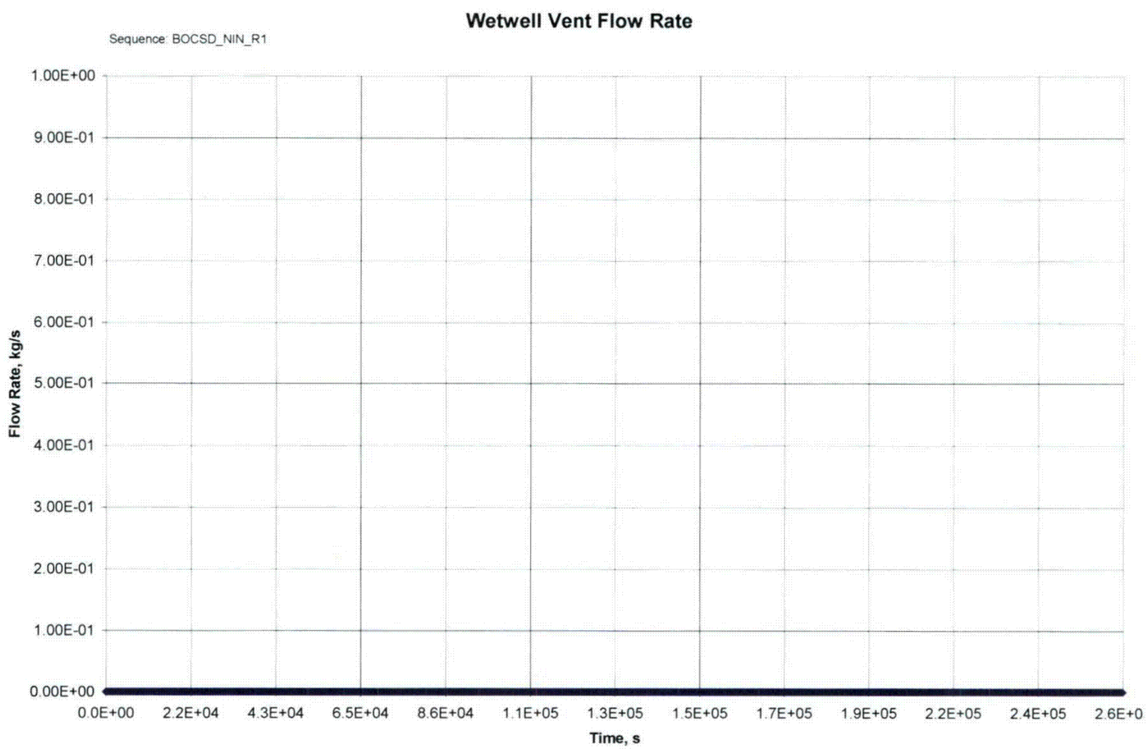


Figure 9A-1s - BOCsd_nIN_R1 Wetwell Vent Flow Rate

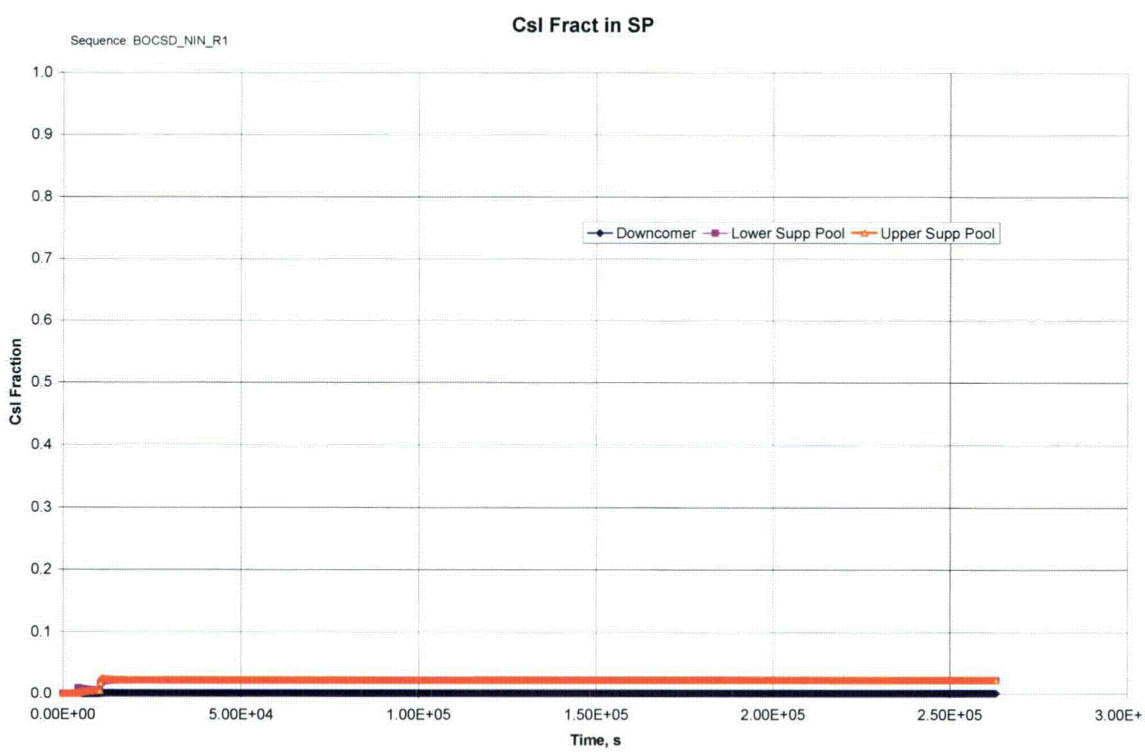


Figure 9A-1t- BOCsd_nIN_R1 CsI Fract in SP

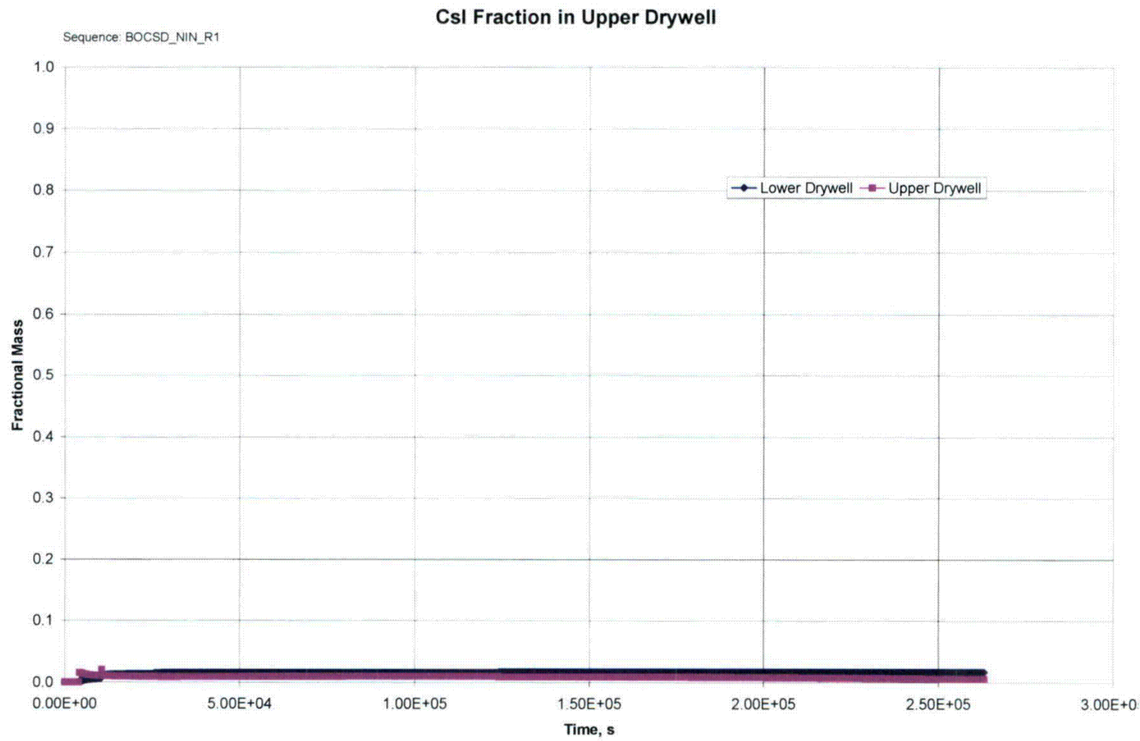


Figure 9A-1u - BOCsd_nIN_R1 CsI Fraction in Upper Drywell

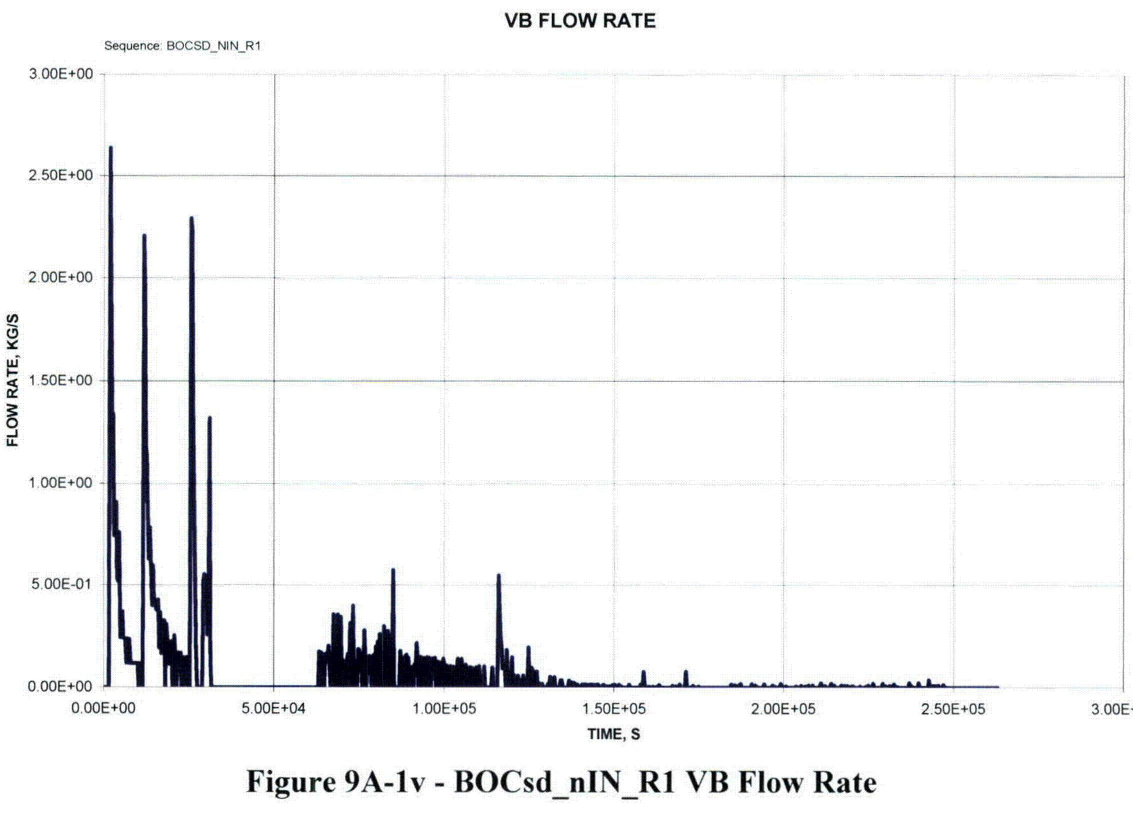


Figure 9A-1v - BOCsd_nIN_R1 VB Flow Rate

9A.2 BOC (Low-Level RWCU Line)
Representative Sequence BOCdr_nIN_R1

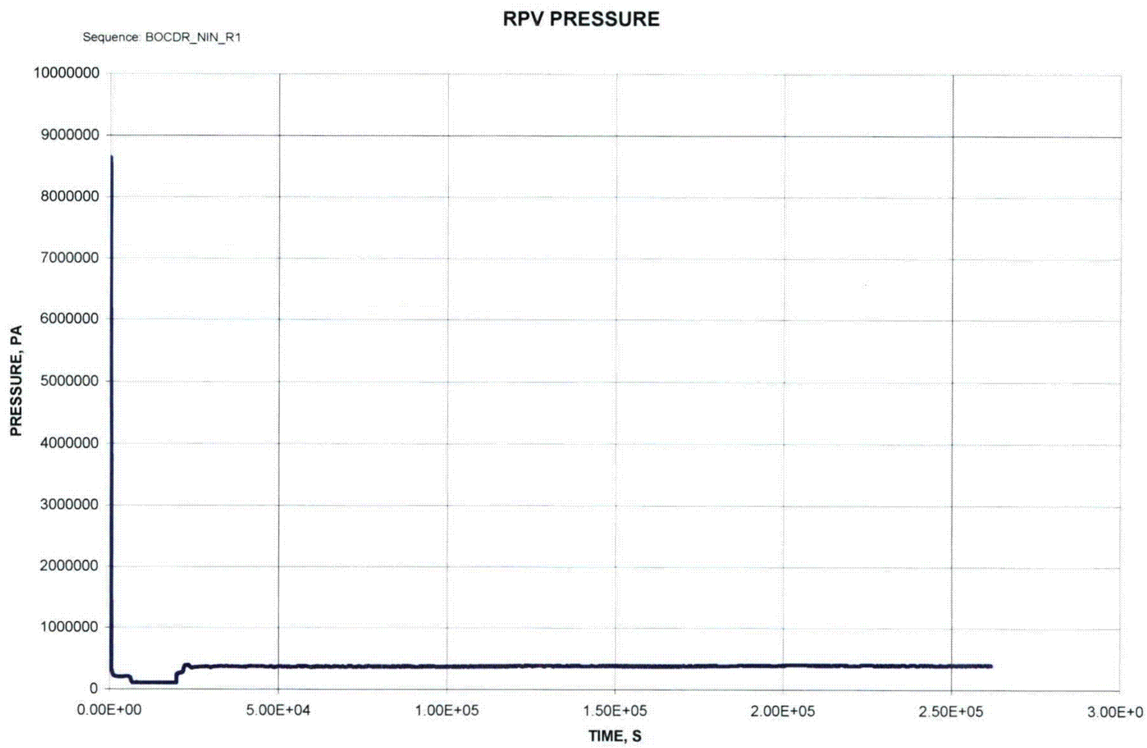


Figure 9A-2a - BOCdr_nIN_R1 RPV Pressure

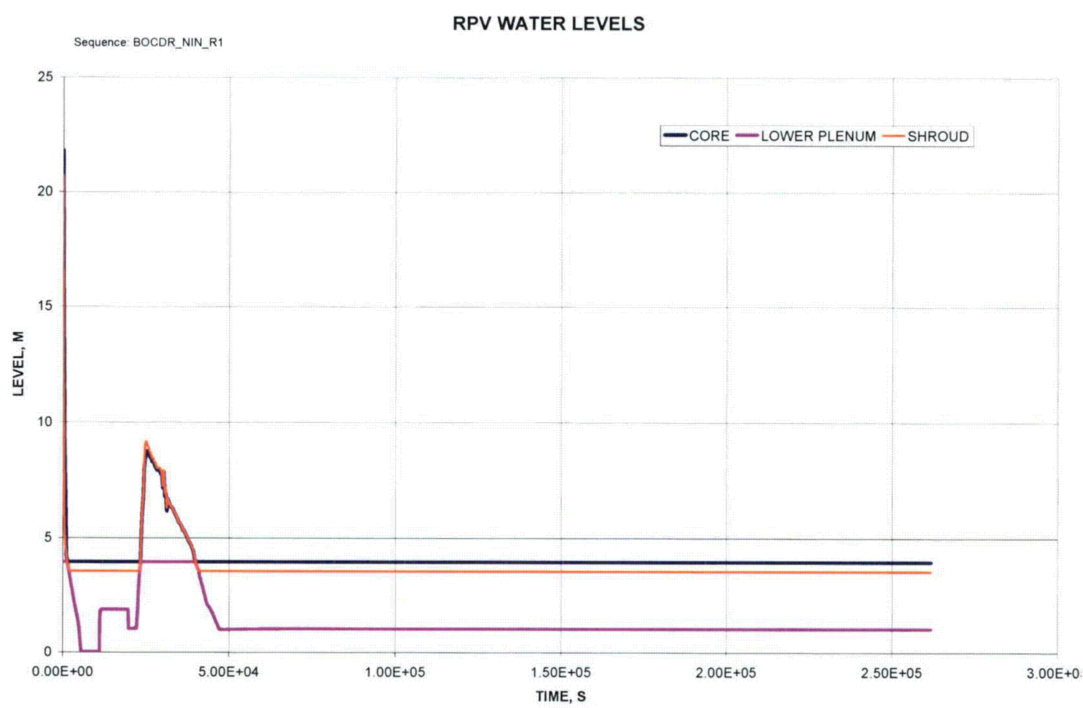


Figure 9A-2b - BOCdr_nIN_R1 RPV Water Levels

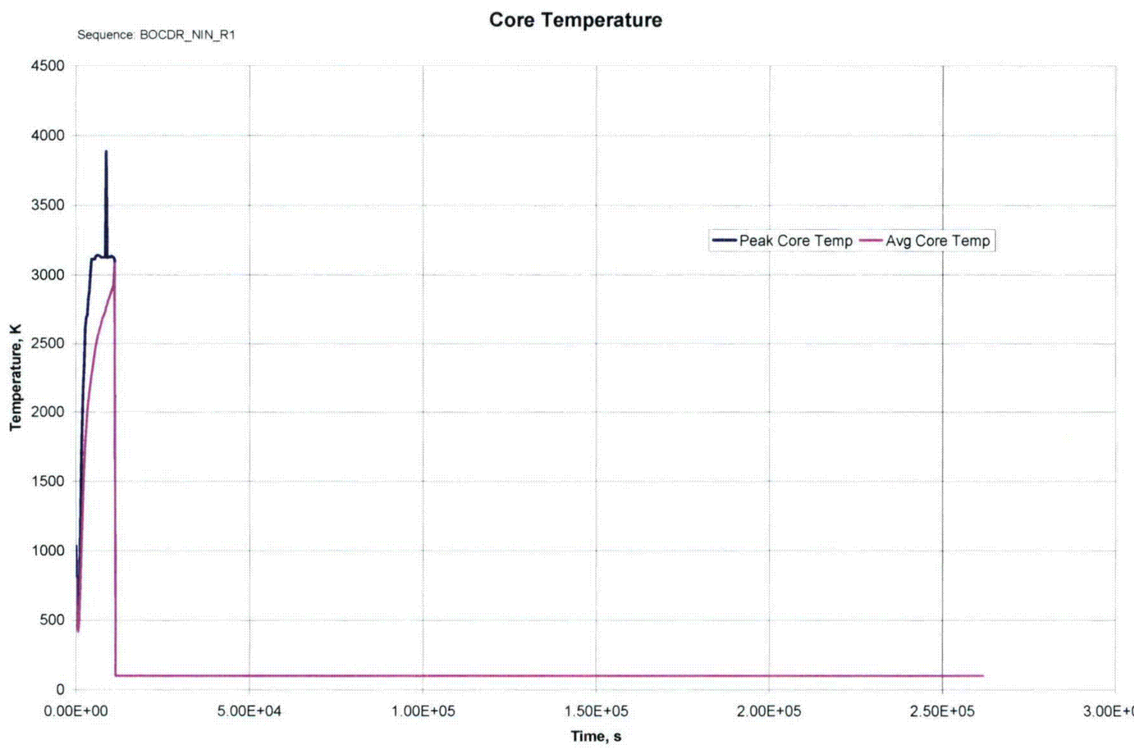


Figure 9A-2c - BOCdr_nIN_R1 Core Temperature

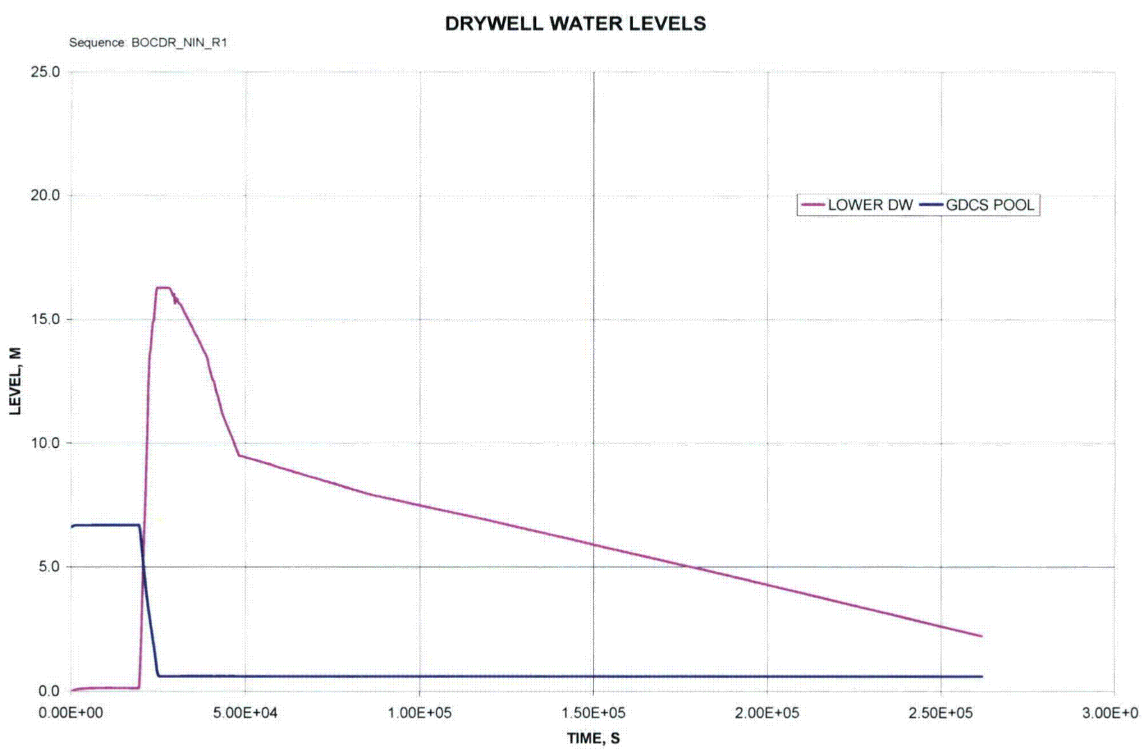


Figure 9A-2d - BOCdr_nIN_R1 Drywell Water Levels

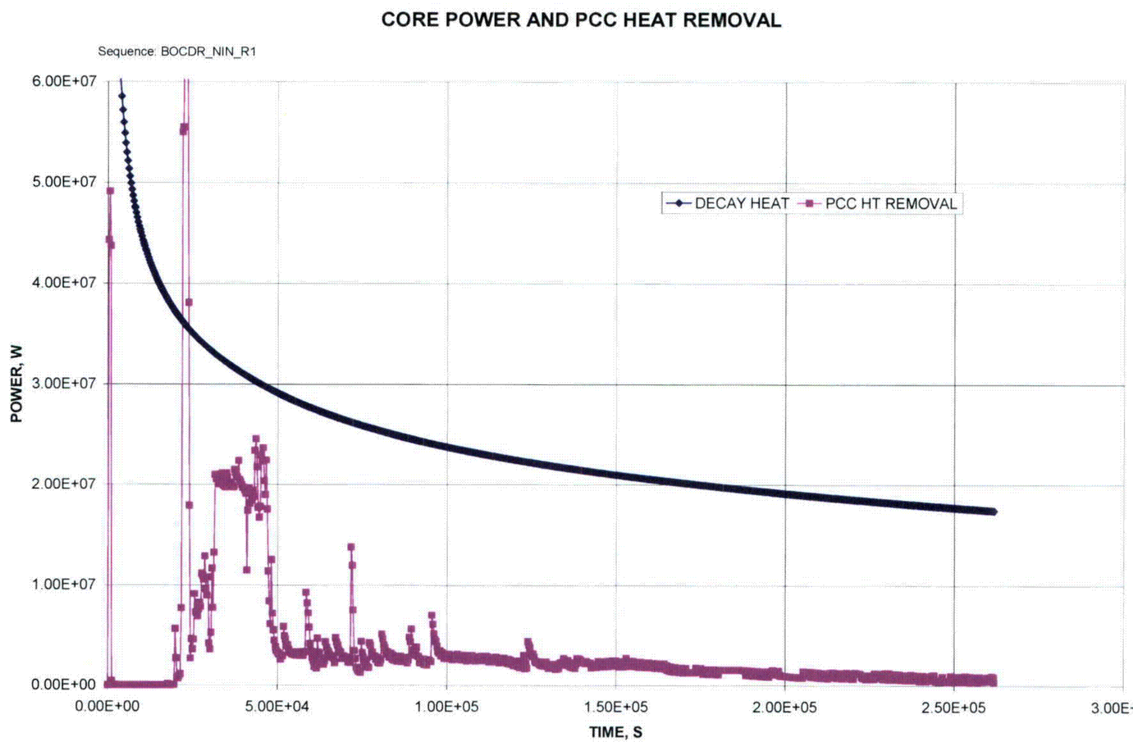


Figure9A-2e - BOCdr_nIN_R1 Core Power and PCC Heat Removal

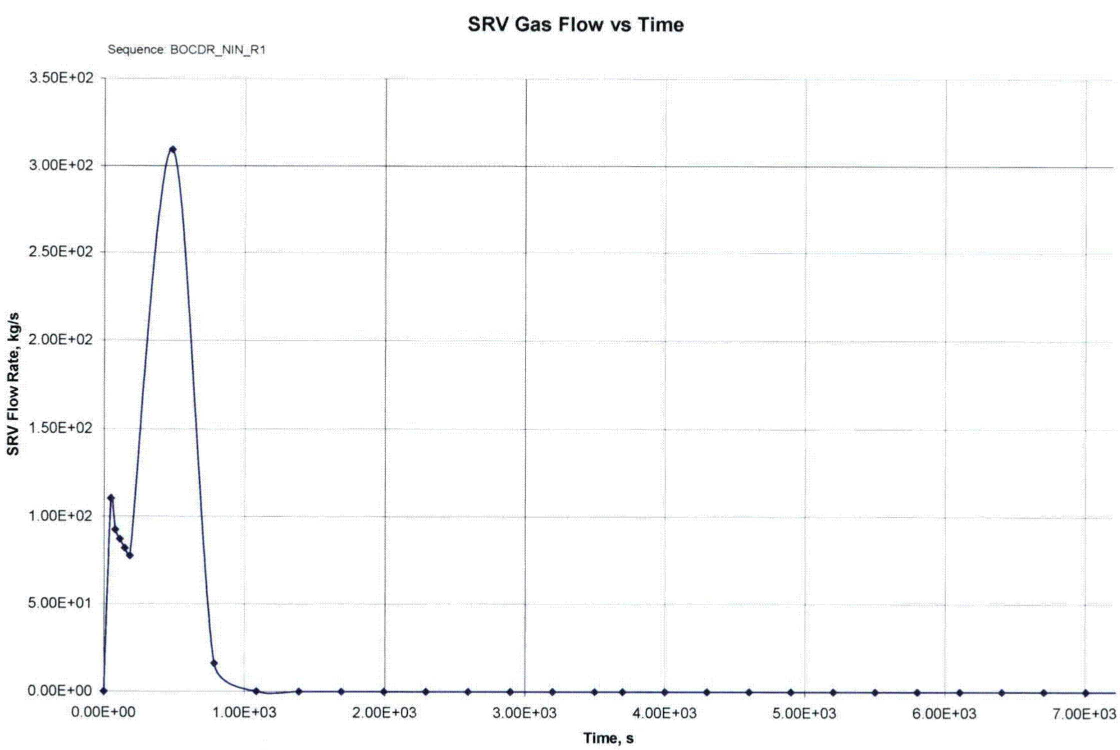


Figure 9A-2f - BOCdr_nIN_R1 SRV Gas Flow vs. Time

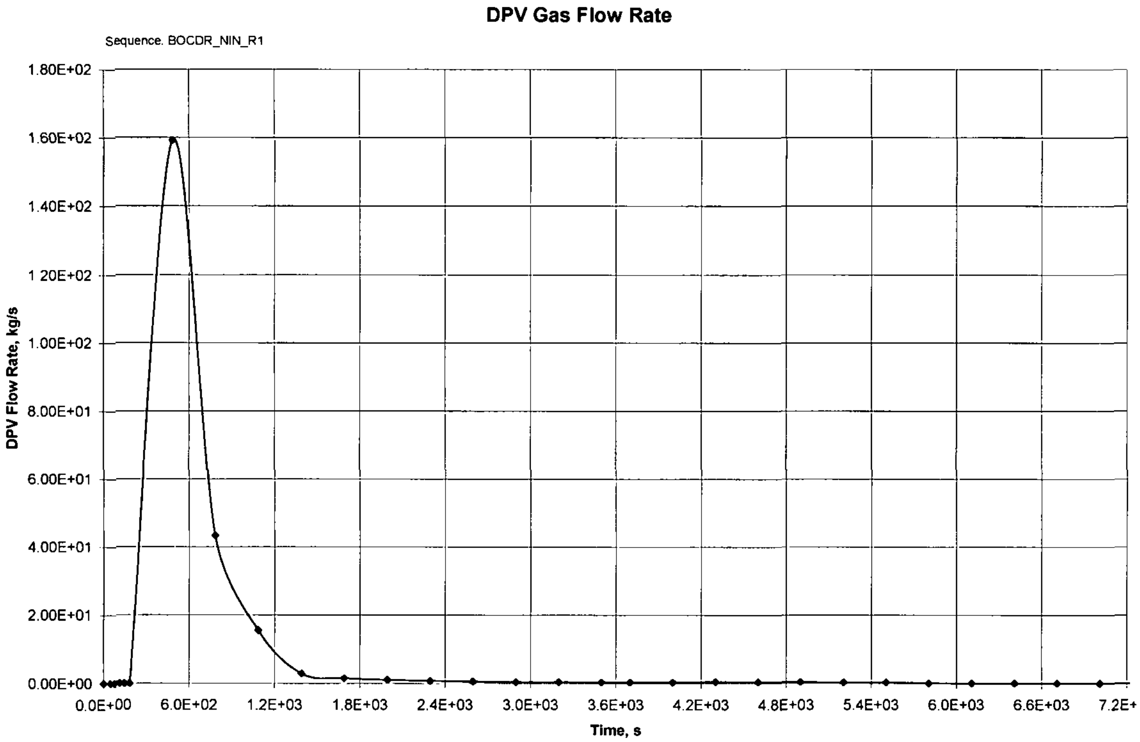


Figure 9A-2g - BOCdr_nIN_R1 DPV Gas Flow Rate

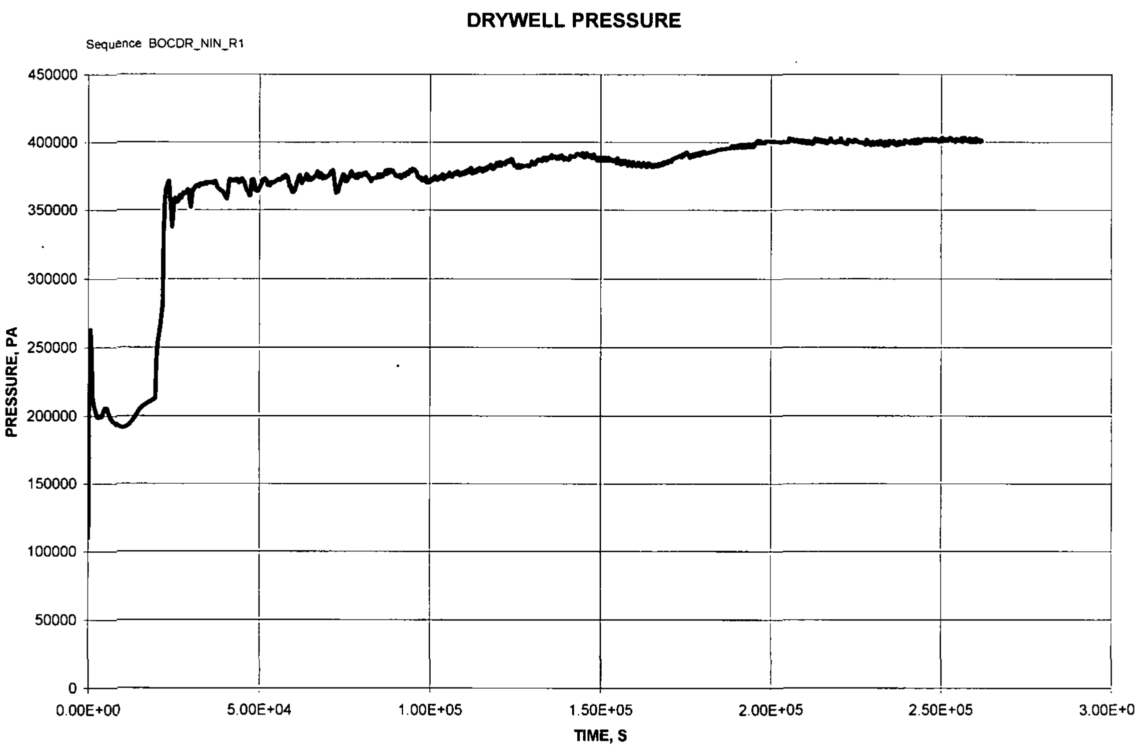


Figure 9A-2h - BOCdr_nIN_R1 Drywell Pressure

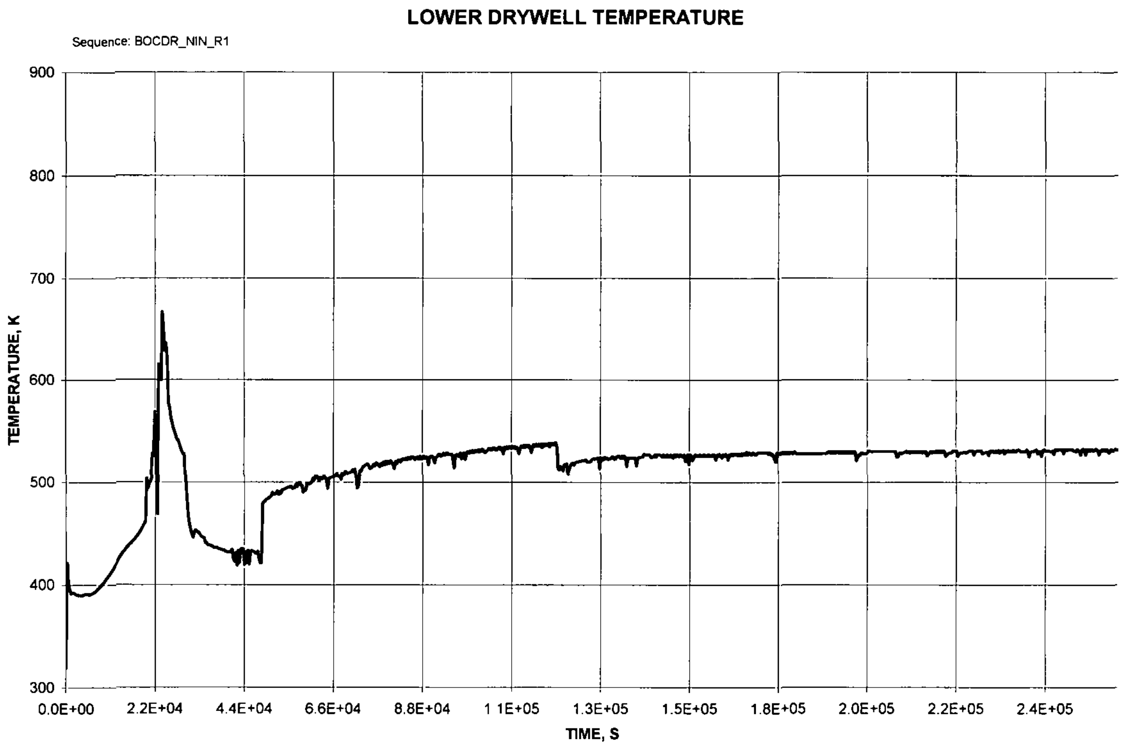


Figure 9A-2i - BOCdr_nIN_R1 Lower Drywell Temperature

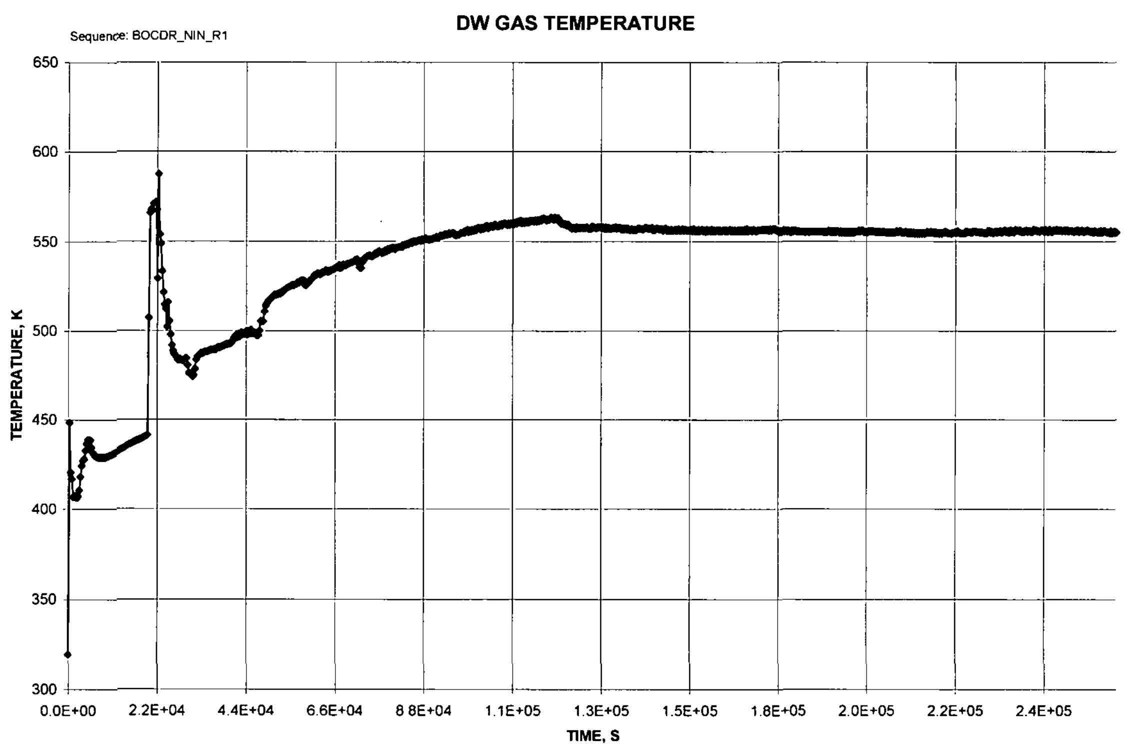


Figure 9A-2j - BOCdr_nIN_R1 DW Gas Temperature

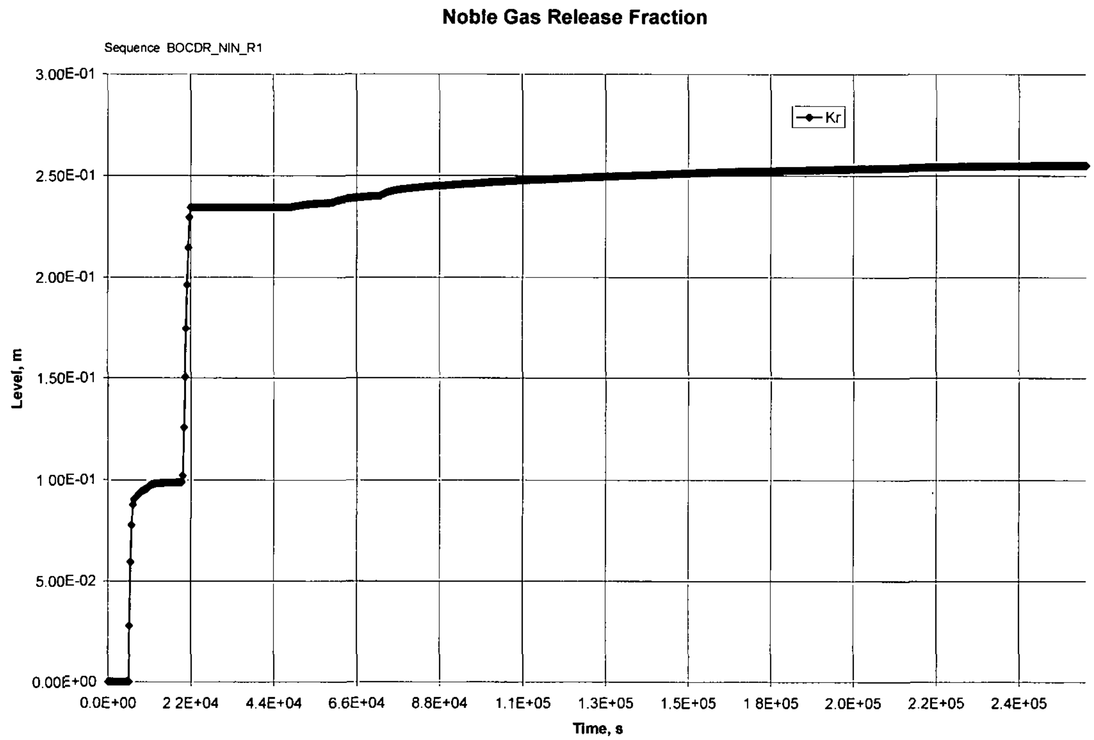


Figure 9A-2k - BOCdr_nIN_R1 Noble Gas Release Fraction

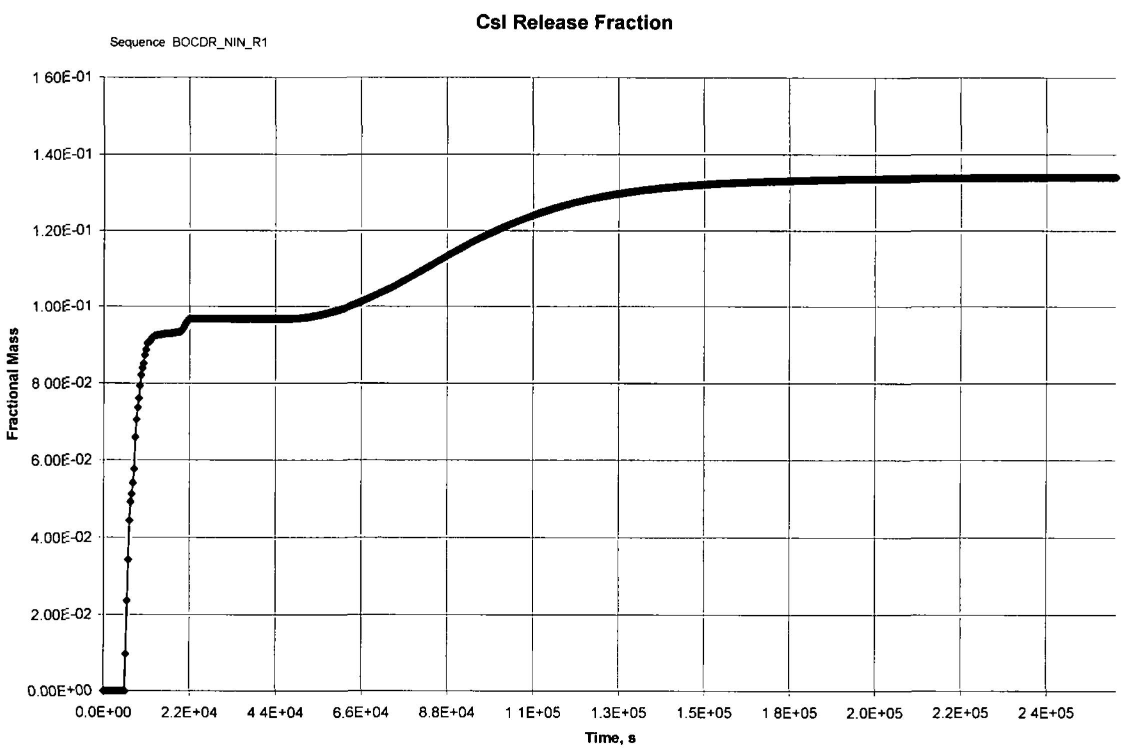


Figure 9A-2l - BOCdr_nIN_R1 CsI Release Fraction

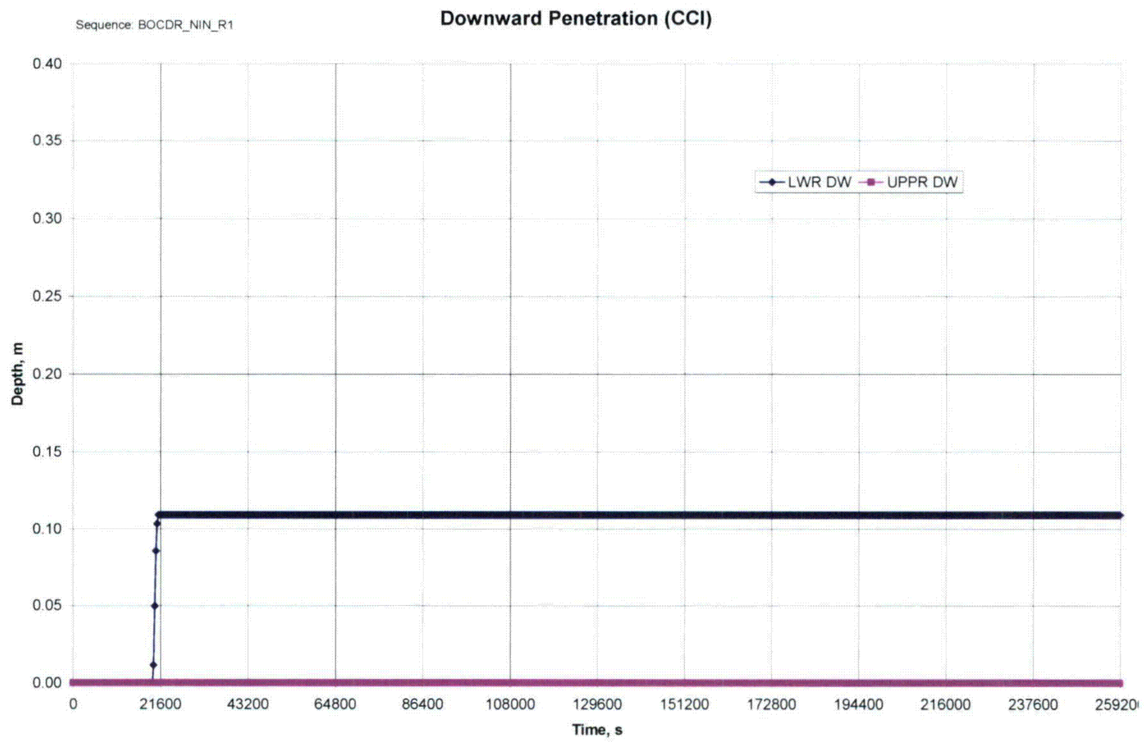


Figure 9A-2m - BOCdr_nIN_R1 Downward Penetration

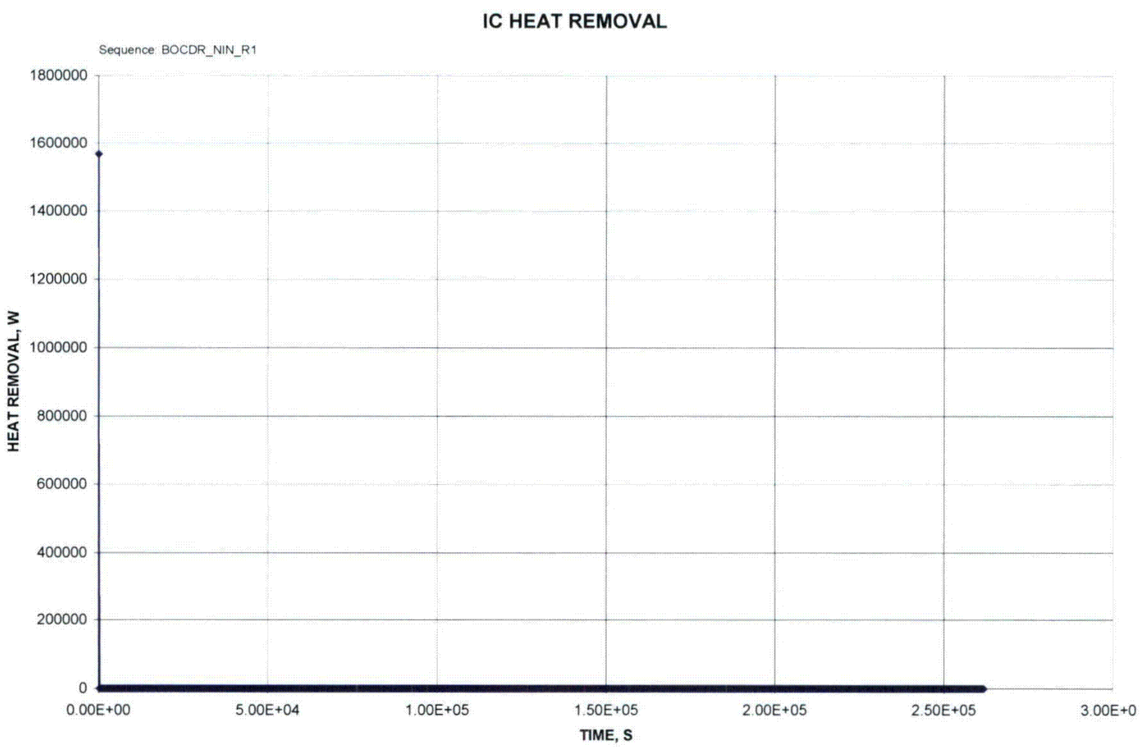


Figure 9A-2n - BOCdr_nIN_R1 IC Heat Removal

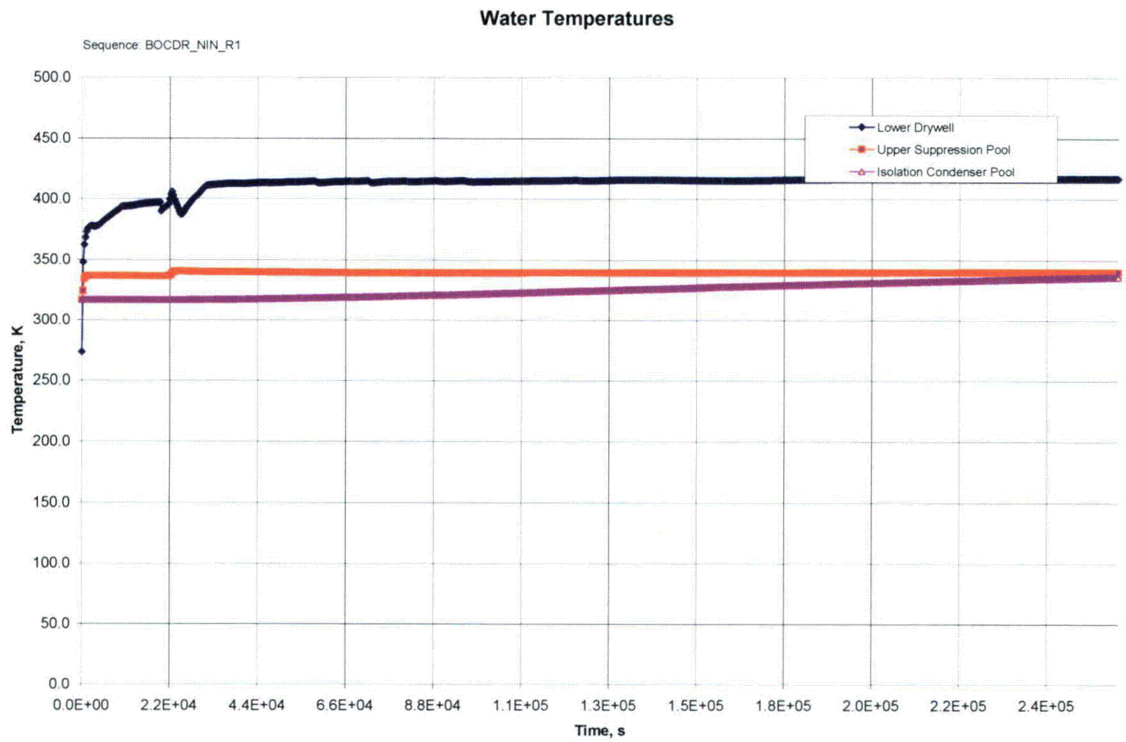


Figure 9A-2o- BOCdr_nIN_R1 Water Temperatures

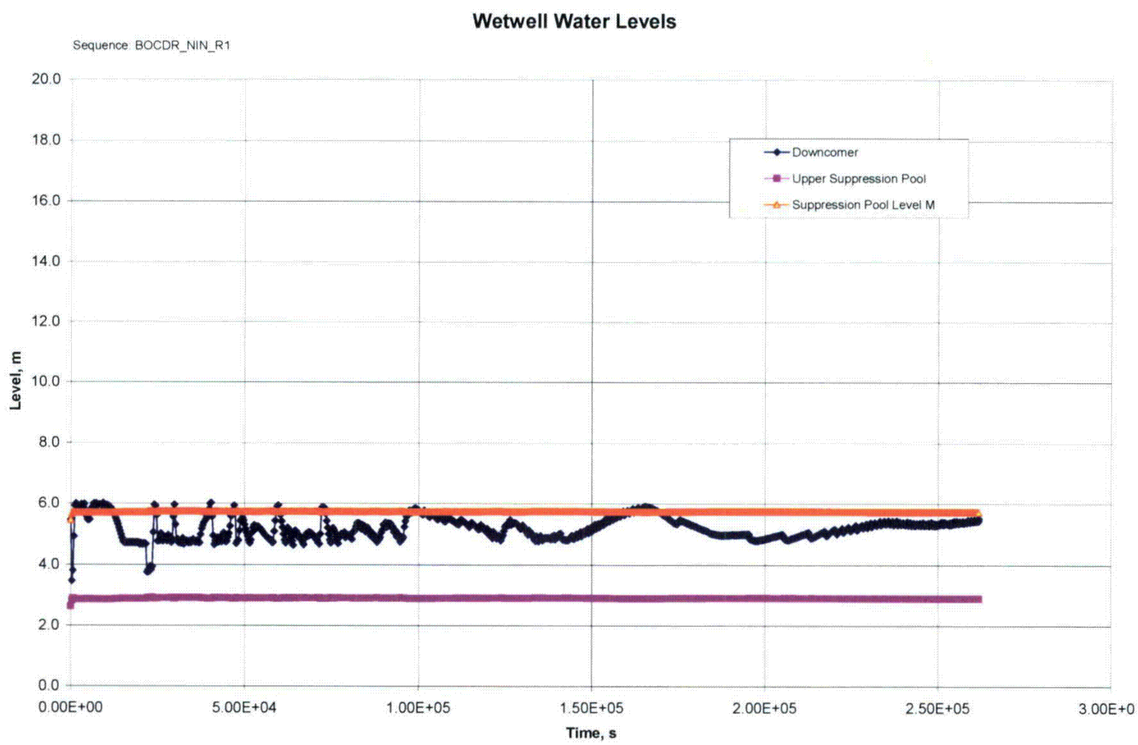


Figure 9A-2p - BOCdr_nIN_R1 Wetwell Water Levels

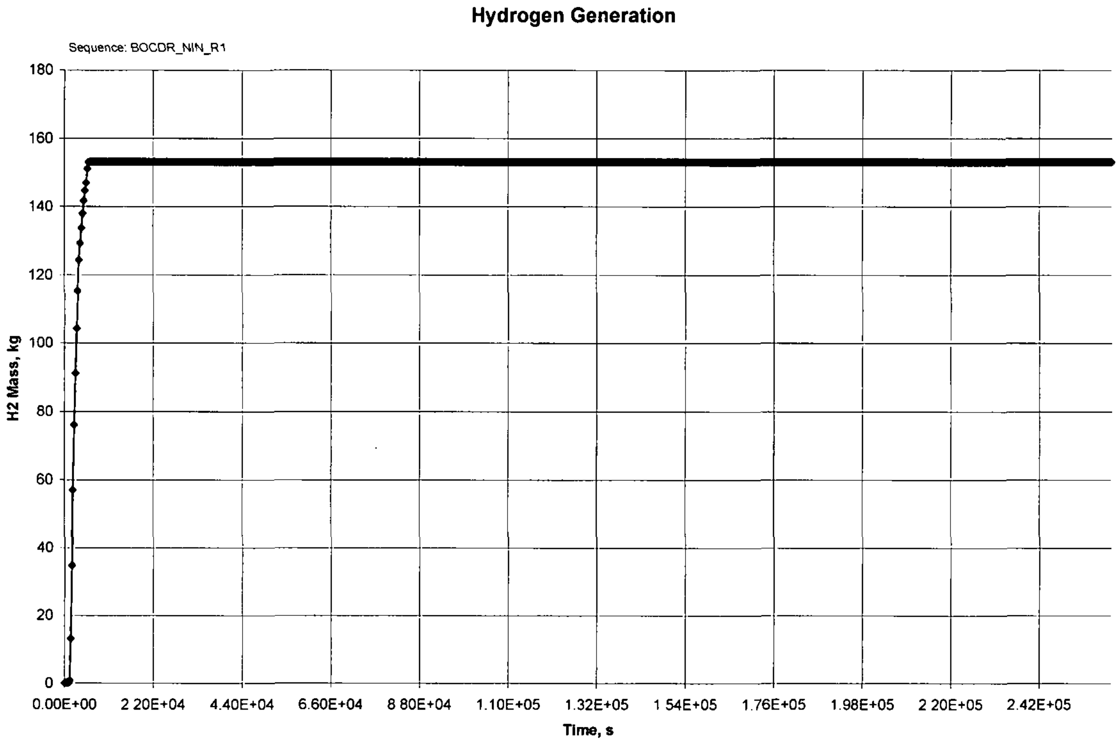


Figure 9A-2q - BOCdr_nIN_R1 Hydrogen Generation

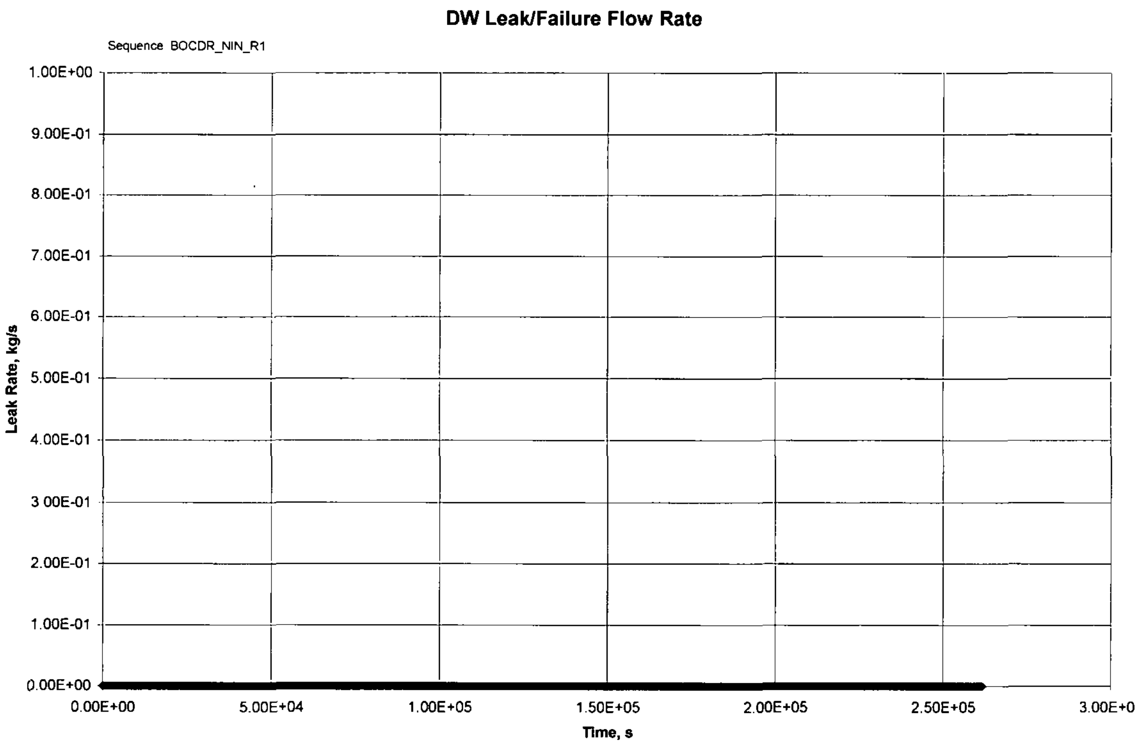


Figure 9A-2r - BOCdr_nIN_R1 DW Leak/Failure Flow Rate

Figure 9A-2s - BOCdr_nIN_R1 Wetwell Vent Flow Rate

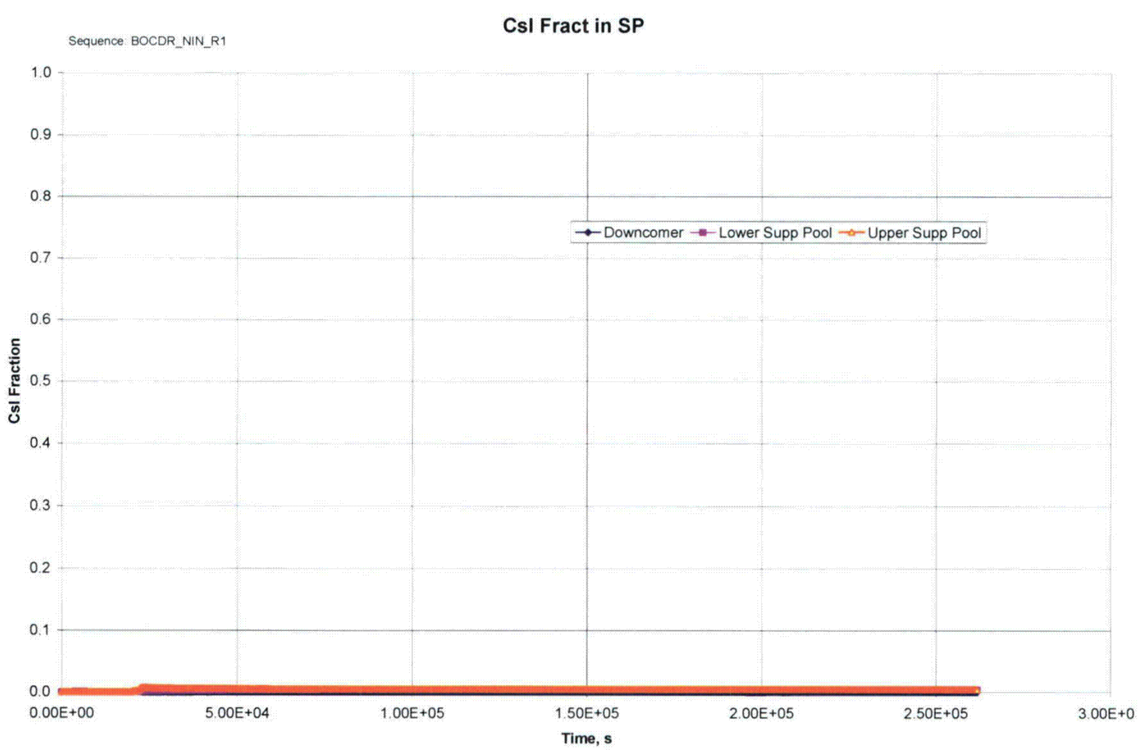


Figure 9A-2t - BOCdr_nIN_R1 CsI Fract in SP

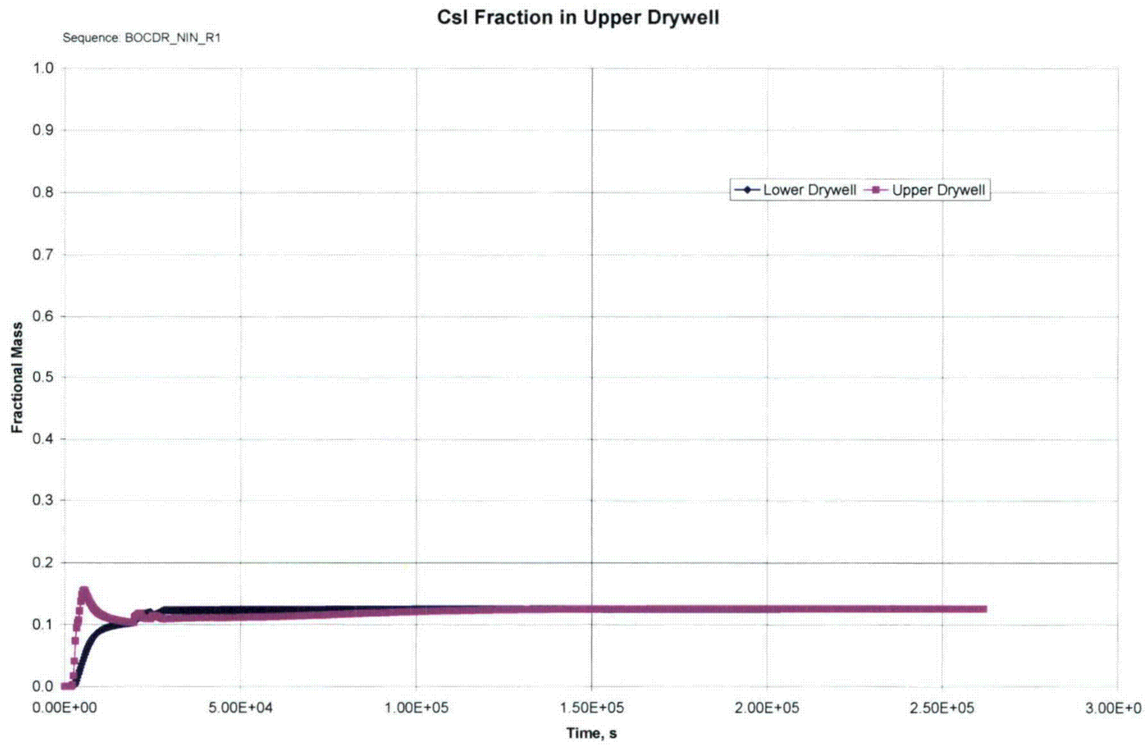


Figure 9A-2u - BOCdr_nIN_R1 CsI Fraction in Upper Drywell

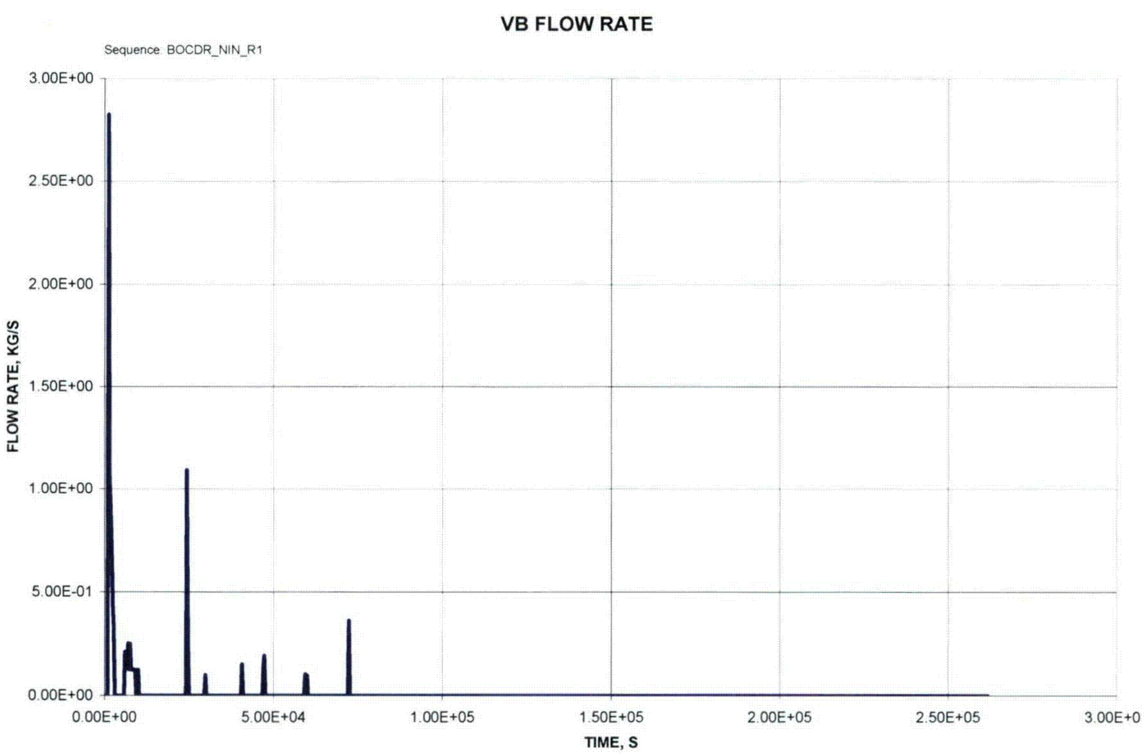


Figure 9A-2v - BOCdr_nIN_R1 VB Flow Rate

9A.3 Bypass Representative Sequence Low Pressure (T_nIN_BYP_R1)

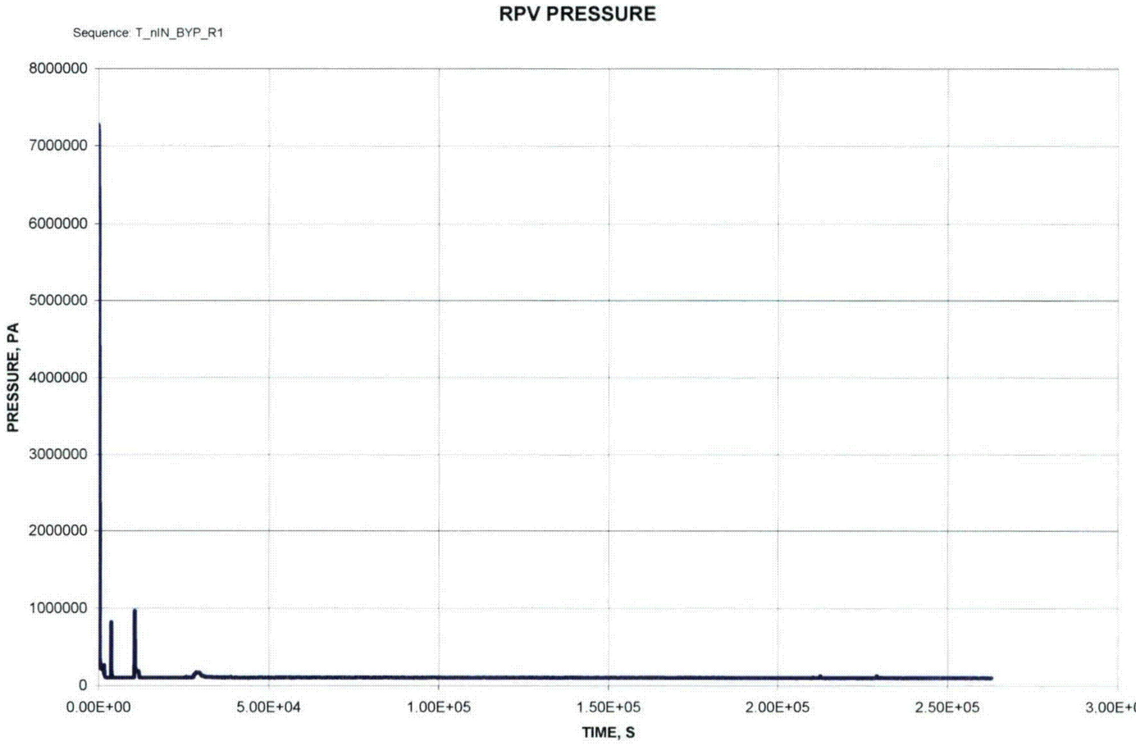


Figure 9A-3a - T_nIN_BYP_R1 RPV Pressure

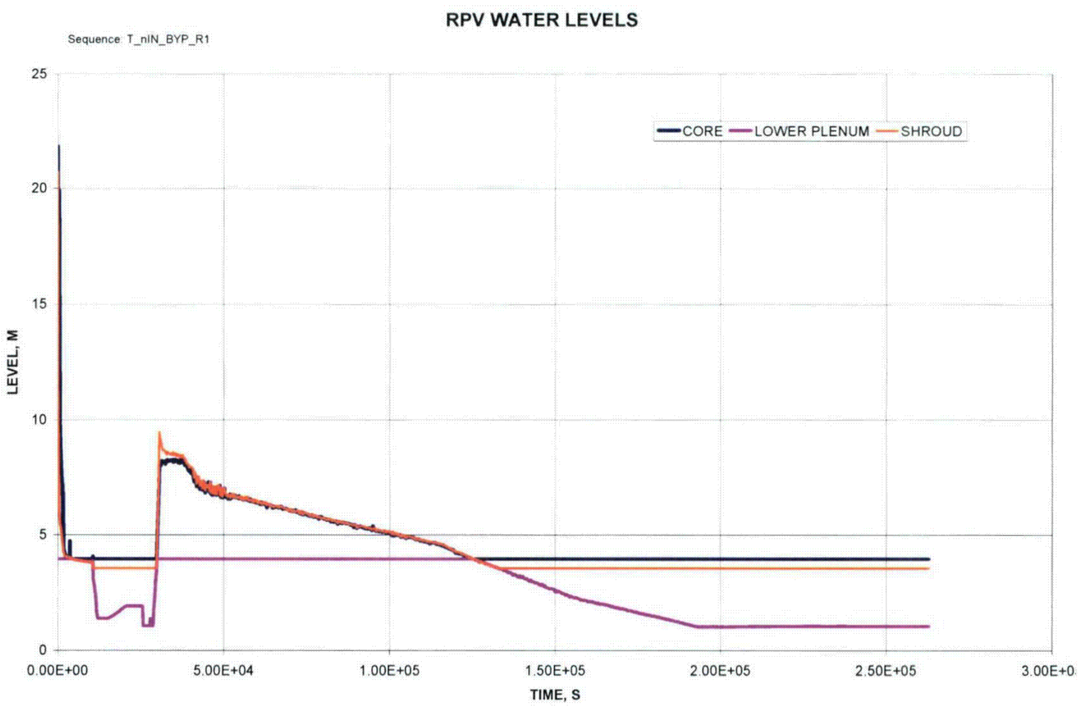


Figure 9A-3b - T_nIN_BYP_R1 RPV Water Levels

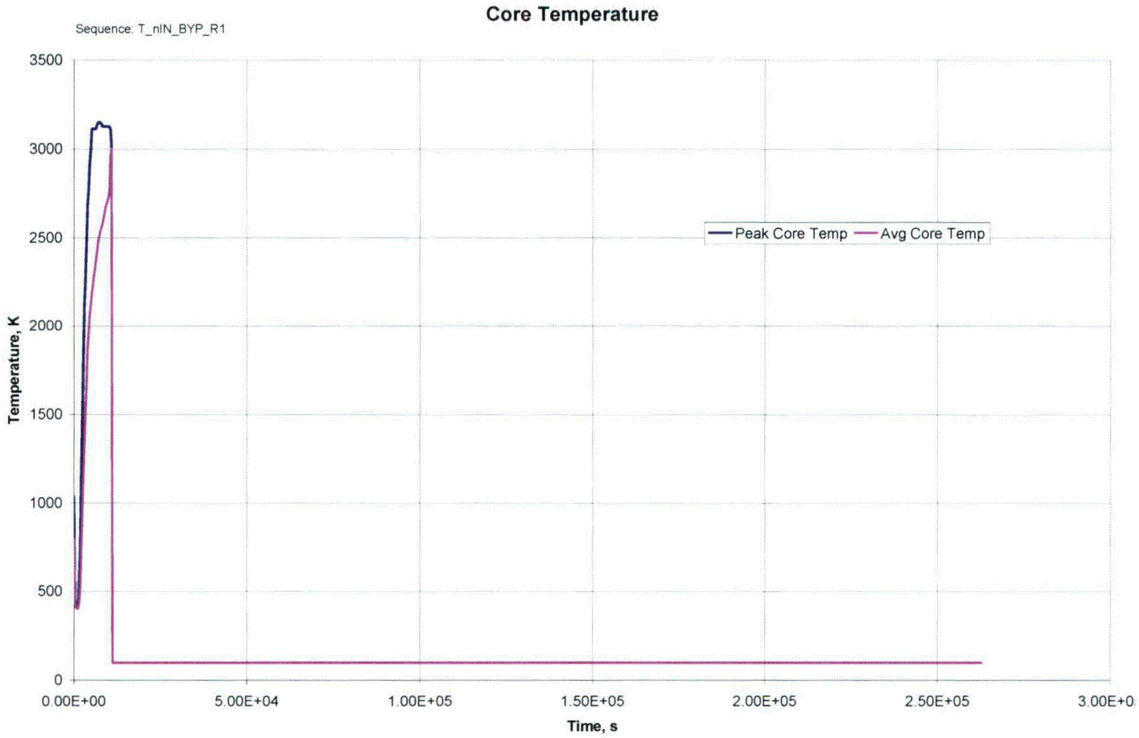


Figure 9A-3c - T_nIN_BYP_R1 Core Temperature

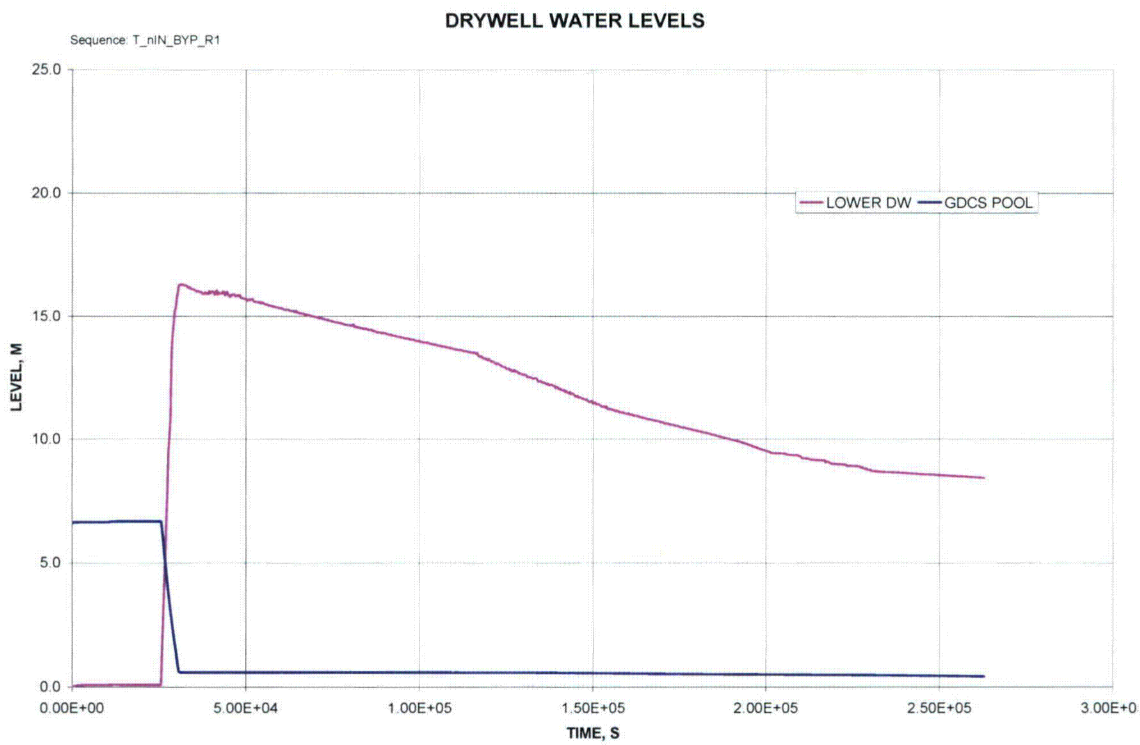


Figure 9A-3d - T_nIN_BYP_R1 Drywell Water Levels

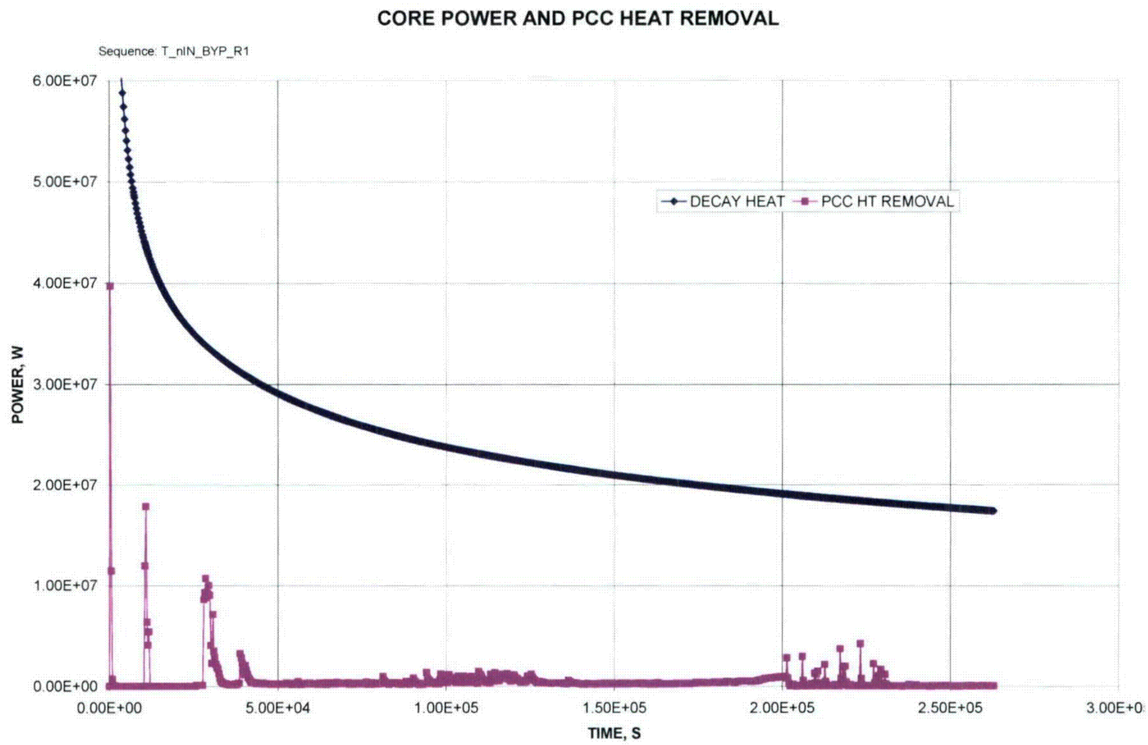


Figure 9A-3e - T_nIN_BYP_R1 Core Power and PCC Heat Removal

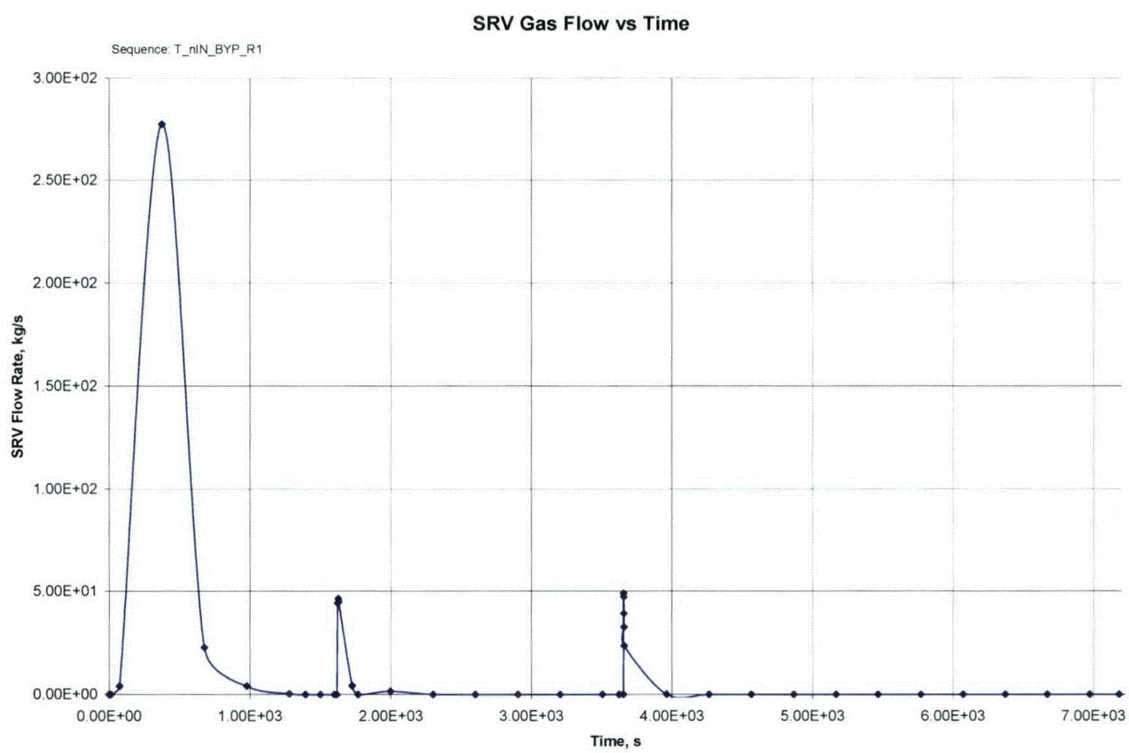


Figure 9A-3f - T_nIN_BYP_R1 SRV Gas Flow vs Time

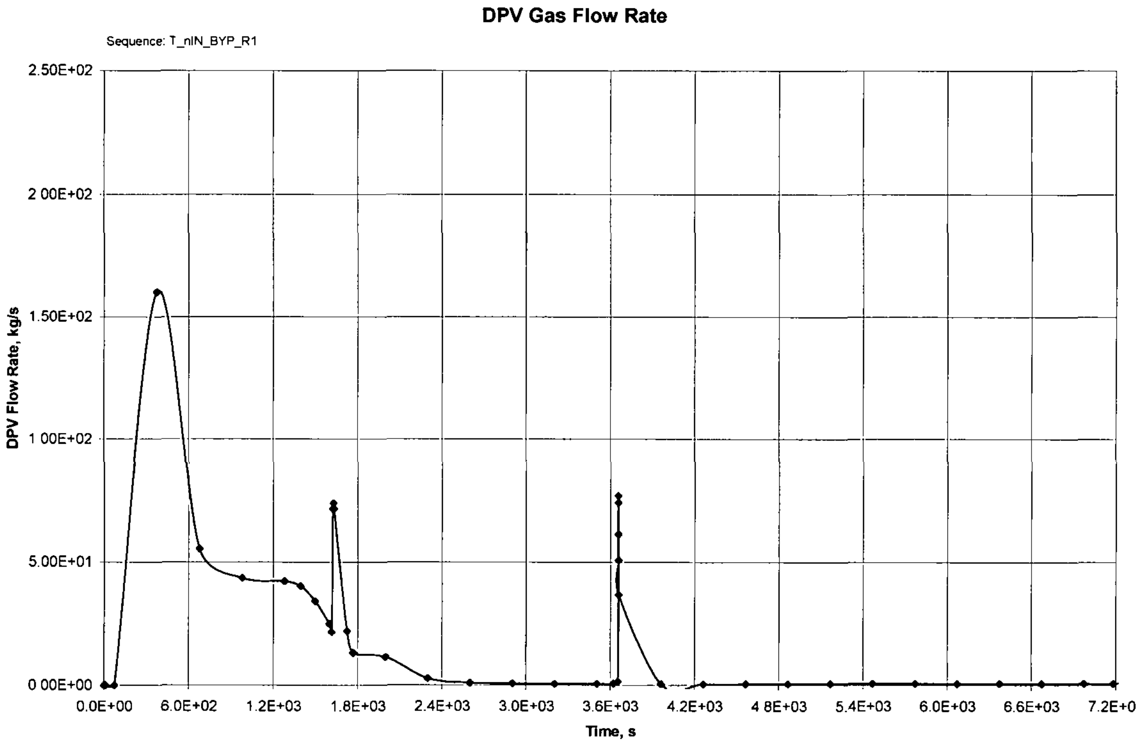


Figure 9A-3g - T_nIN_BYP_R1 DPV Gas Flow Rate

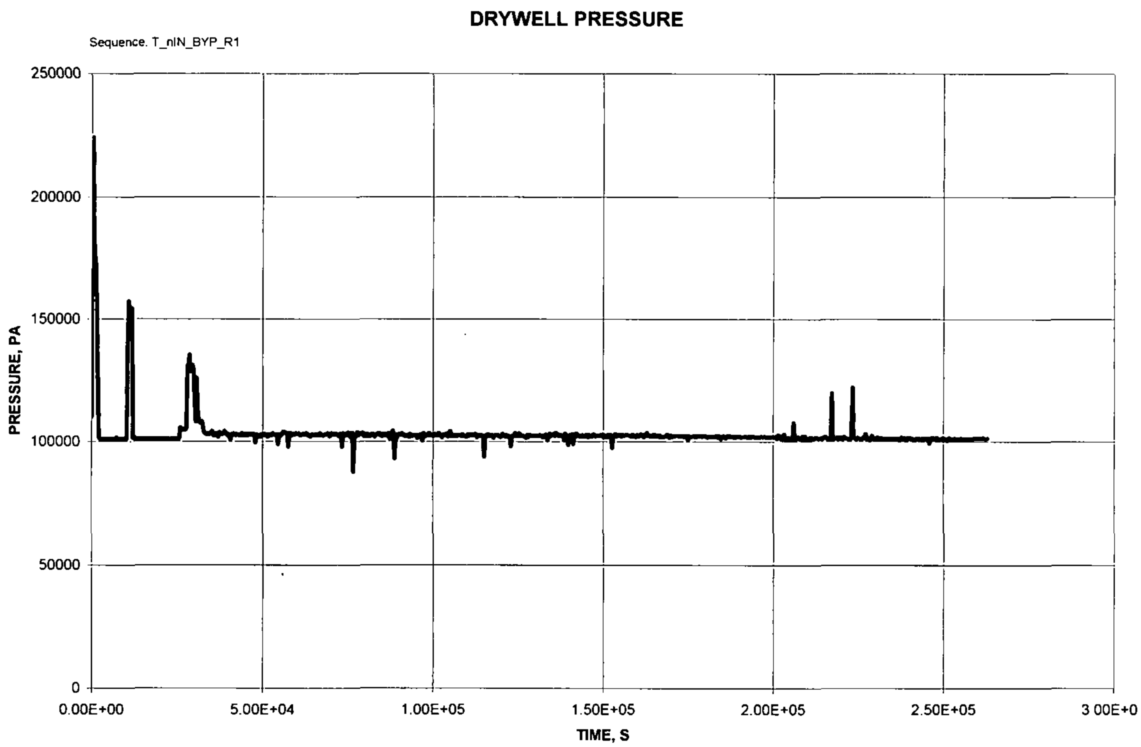


Figure 9A-3h - T_nIN_BYP_R1 Drywell Pressure

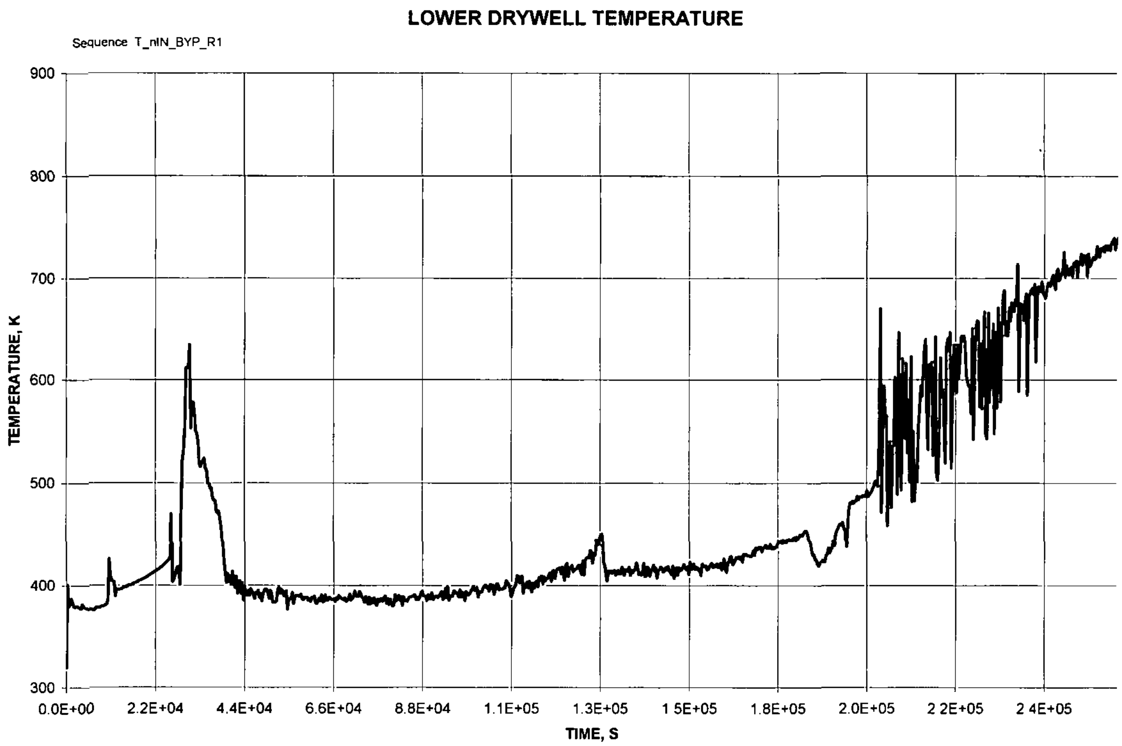


Figure 9A-3i - T_nIN_BYP_R1 Lower Drywell Temperature

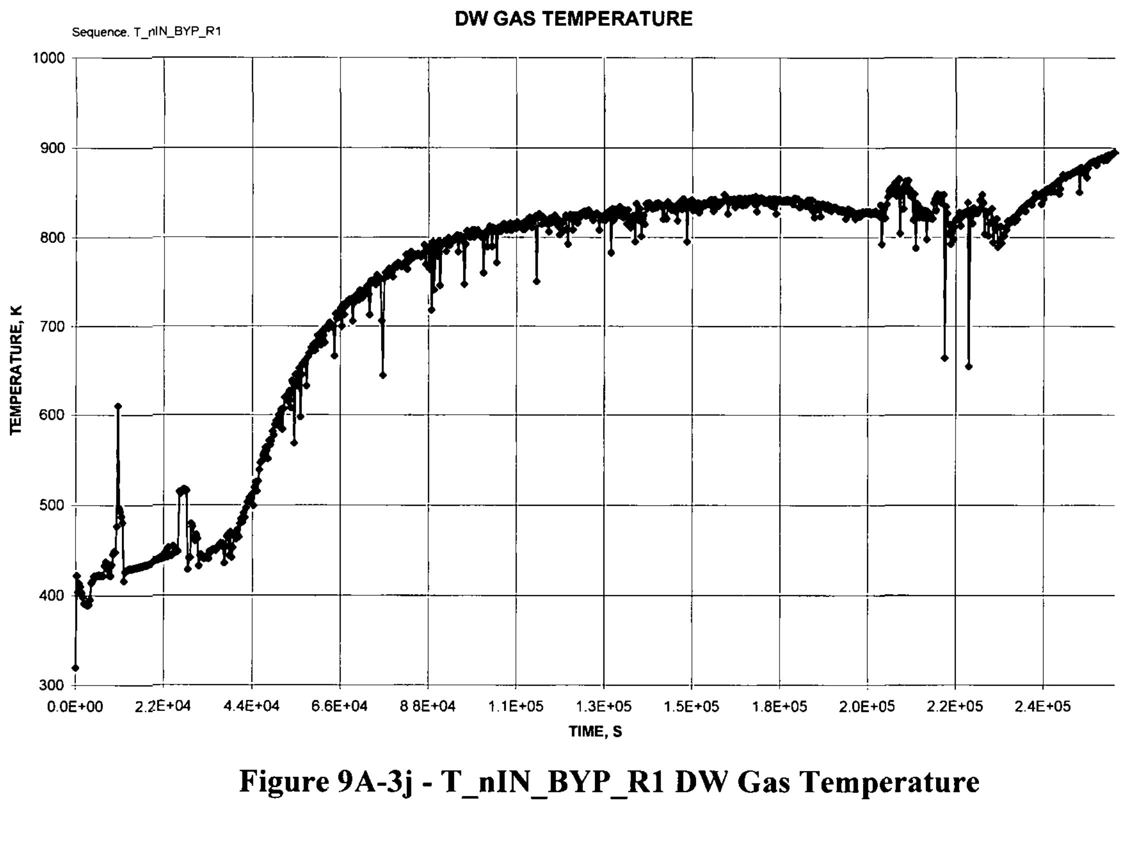


Figure 9A-3j - T_nIN_BYP_R1 DW Gas Temperature

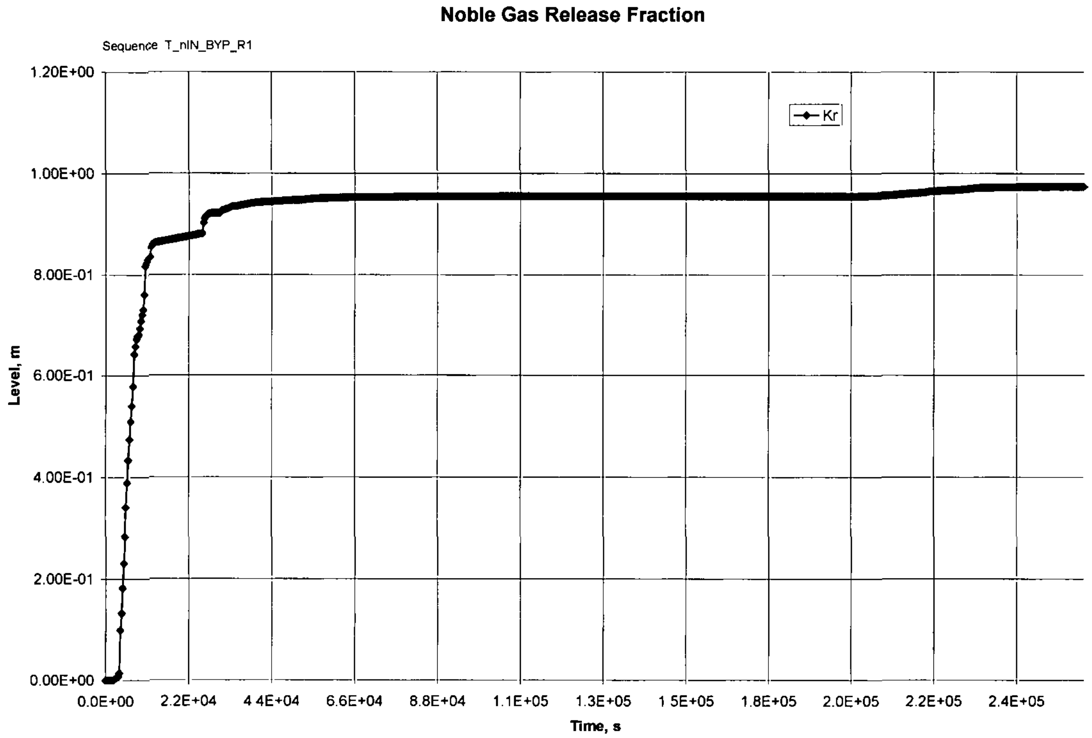


Figure 9A-3k - T_nIN_BYP_R1 Noble Gas Release Fraction

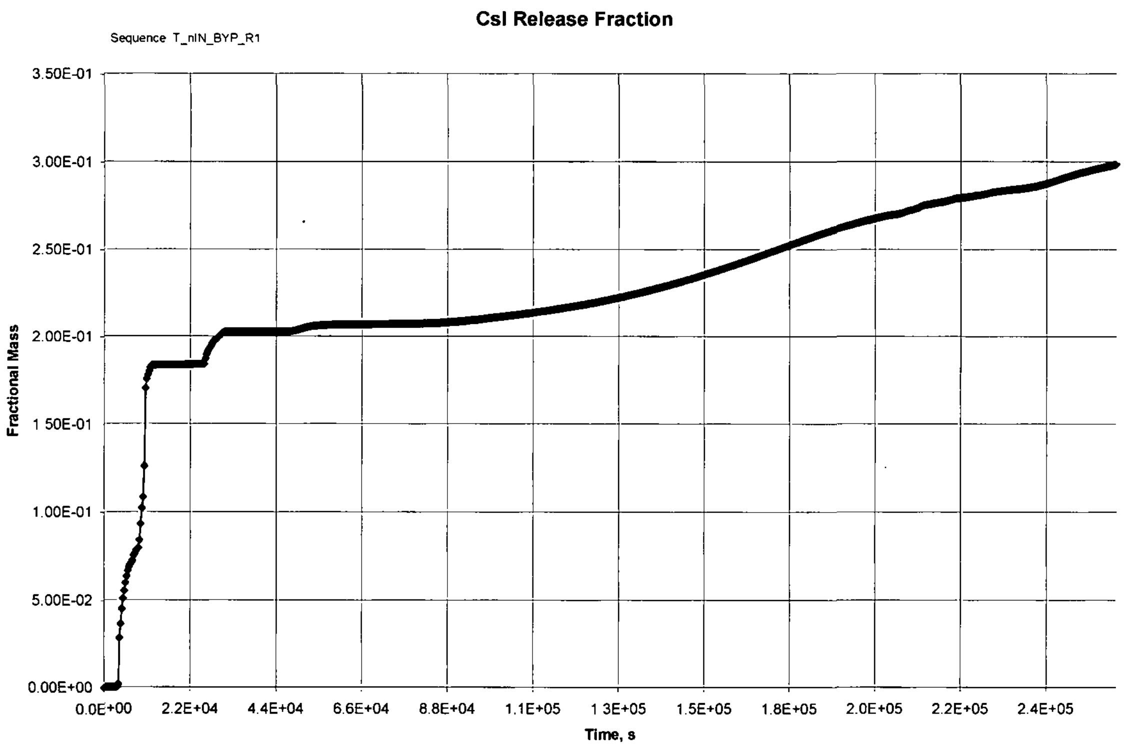


Figure 9A-3l – T_nIN_BYP_R1 CsI Release Fraction

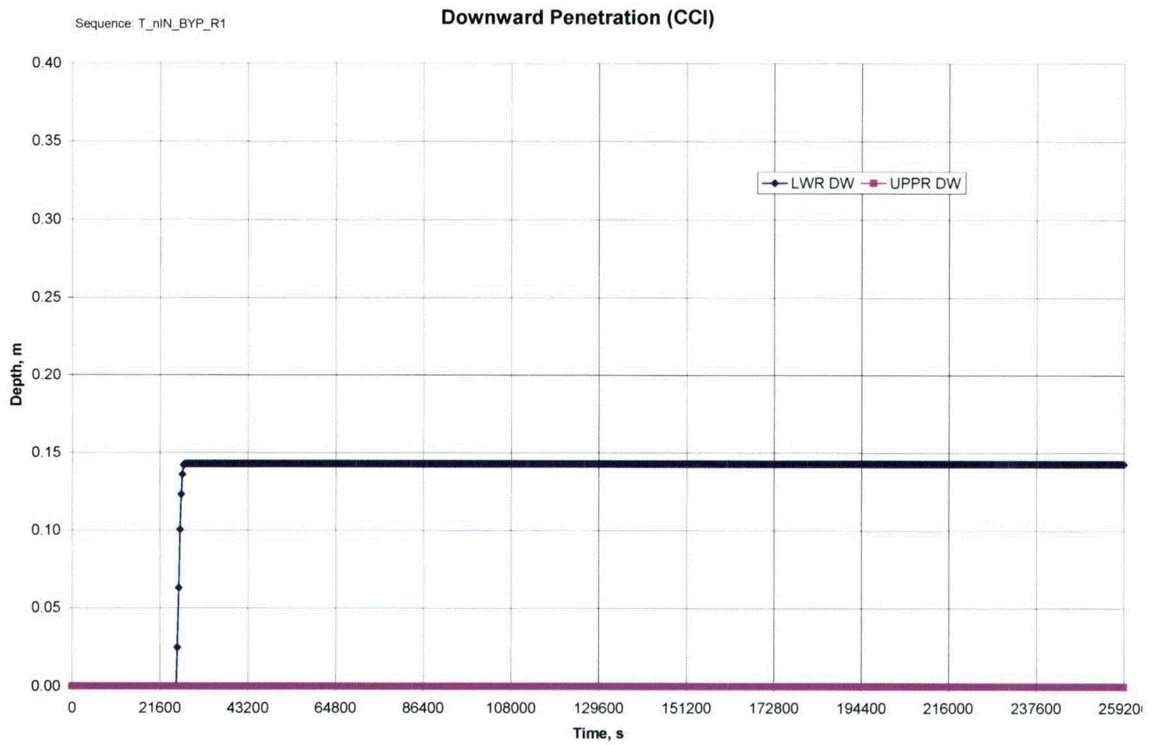


Figure 9A-3m – T_nIN_BYP_R1 Downward Penetration

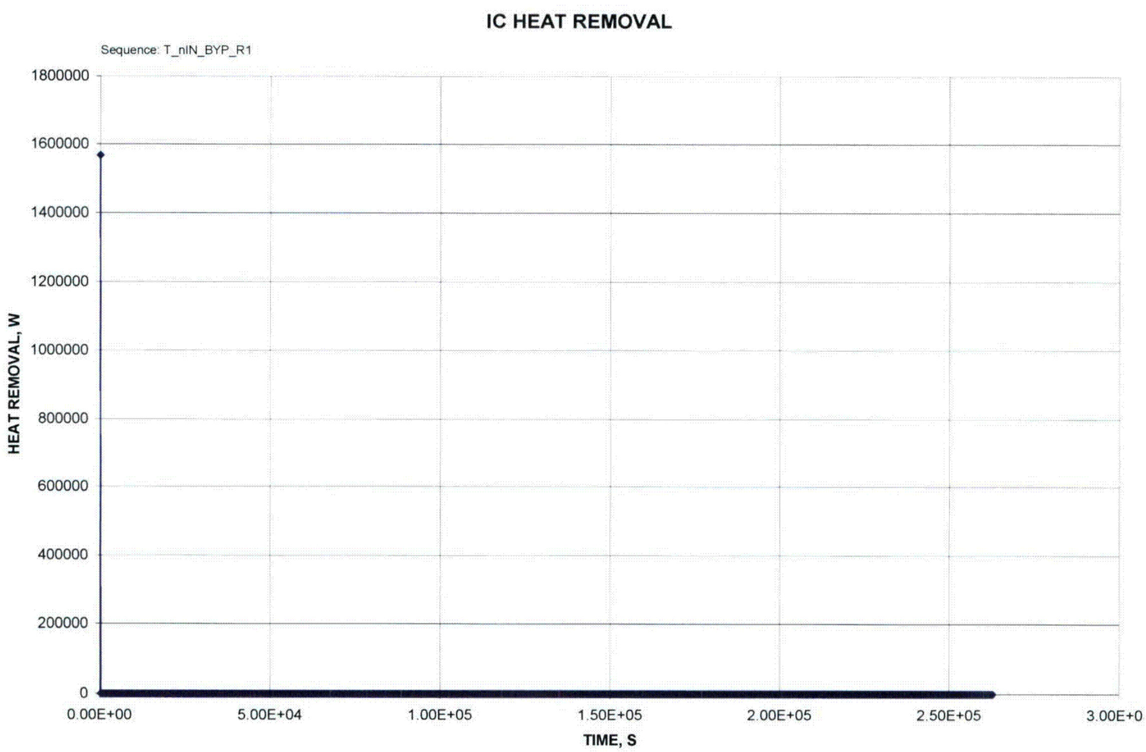


Figure 9A-3n – T_nIN_BYP_R1 IC Heat Removal

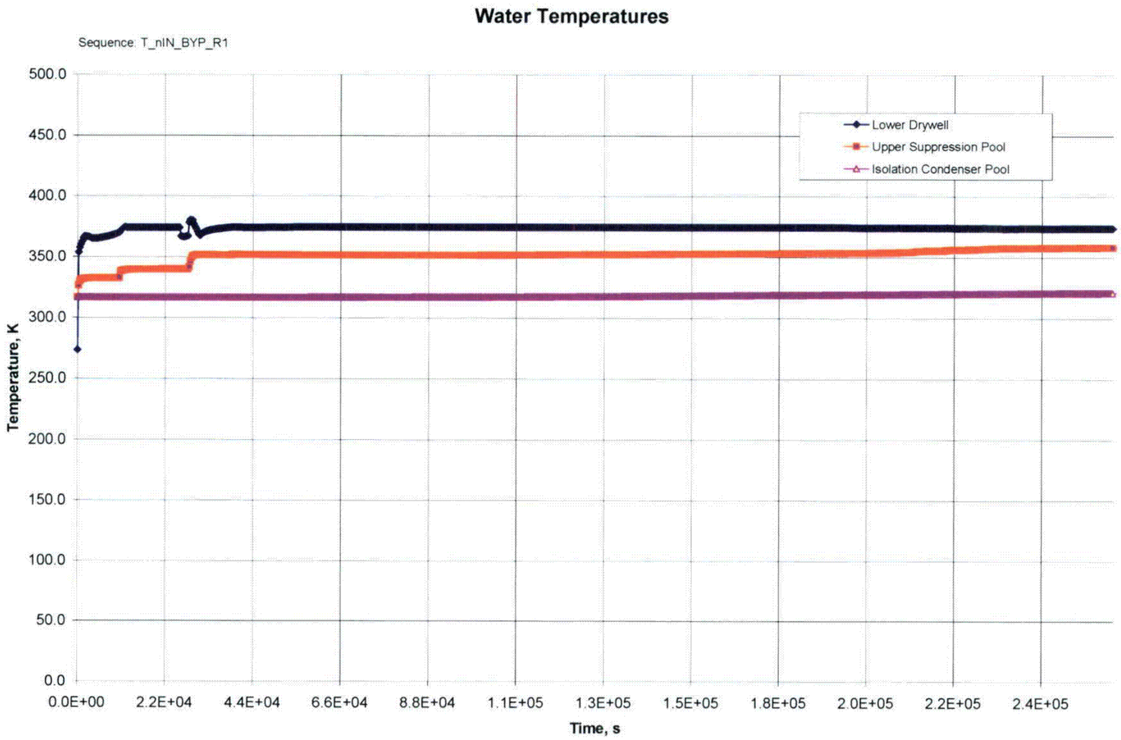


Figure 9A-3o – T_nIN_BYP_R1 Water Temperature

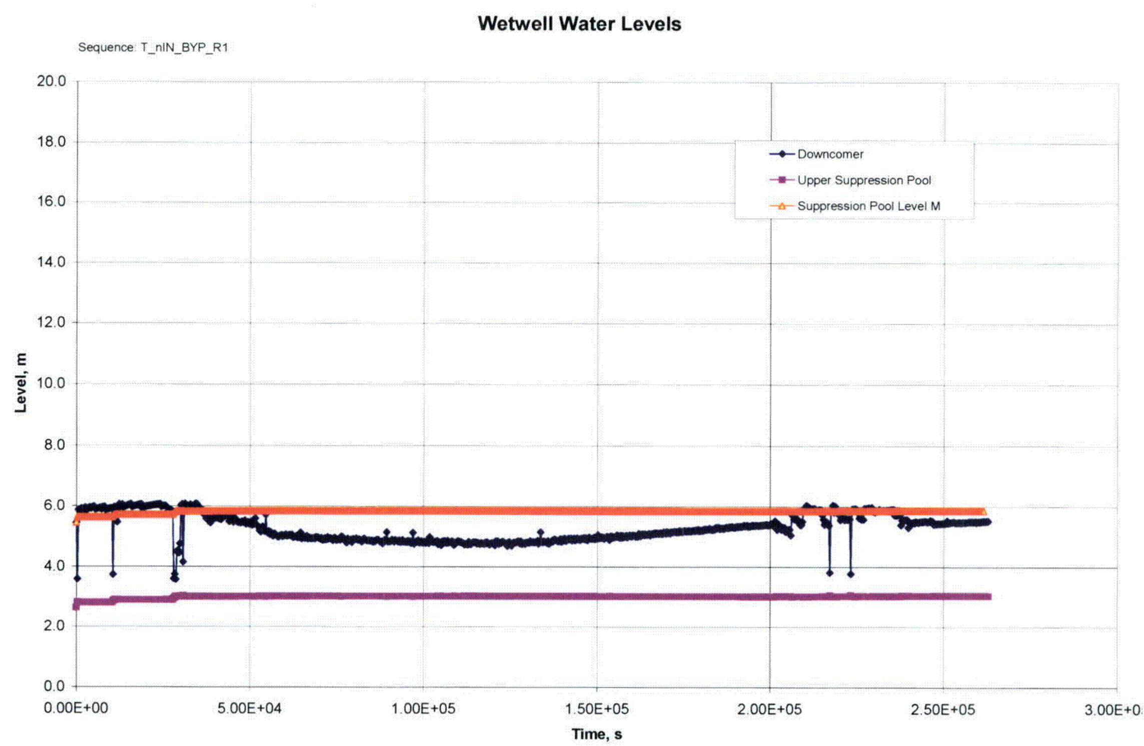


Figure 9A-3p – T_nIN_BYP_R1 Wetwell Water Levels

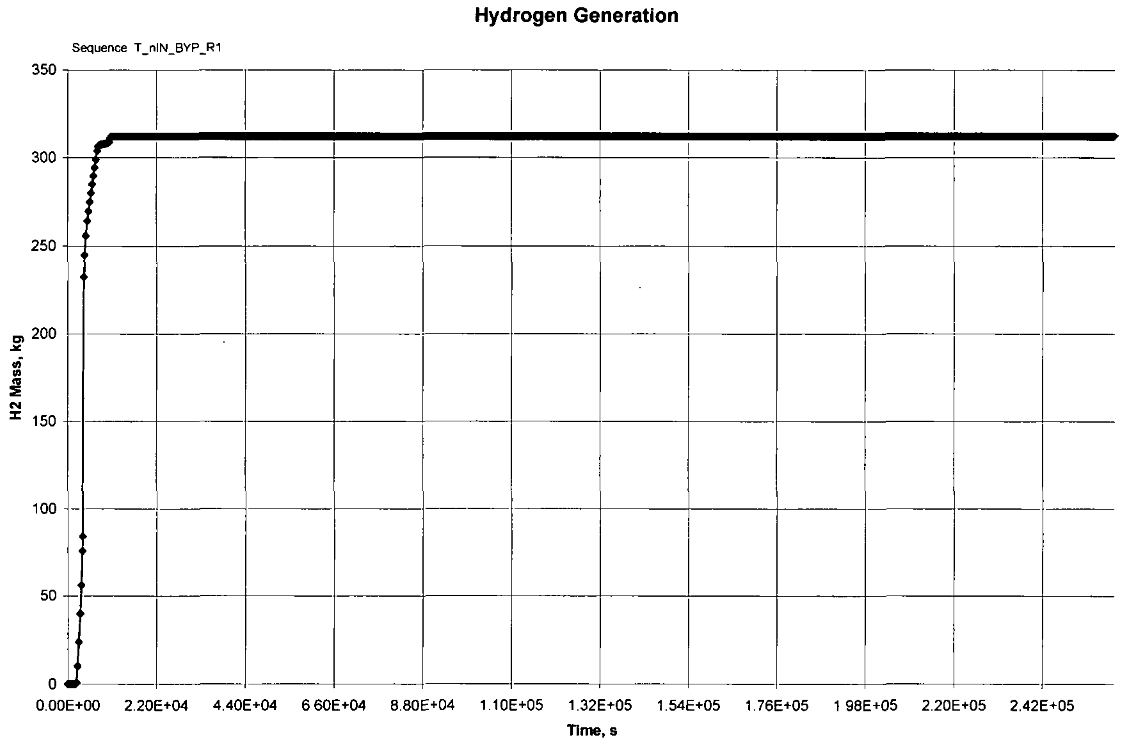


Figure 9A-3q – T_nIN_BYP_R1 Hydrogen Generation

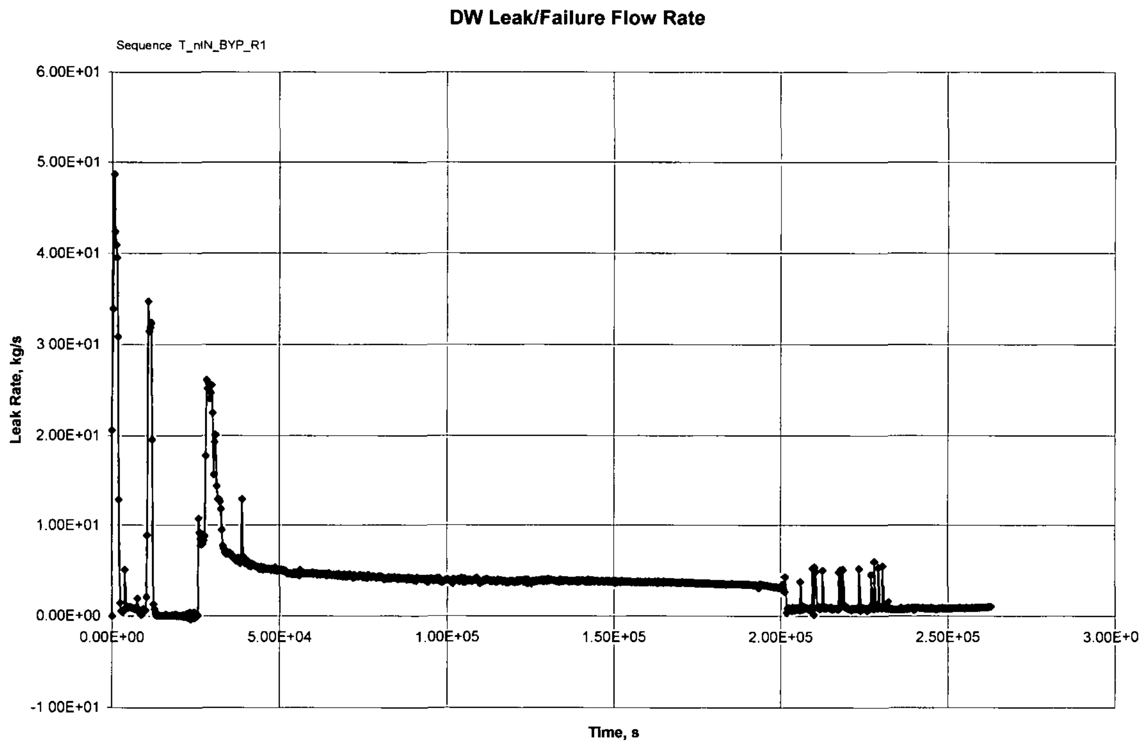


Figure 9A-3r – T_nIN_BYP_R1 DW Leak/Failure Flow Rate

Figure 9A-3s – T_nIN_BYP_R1 Wetwell Vent Flow Rate

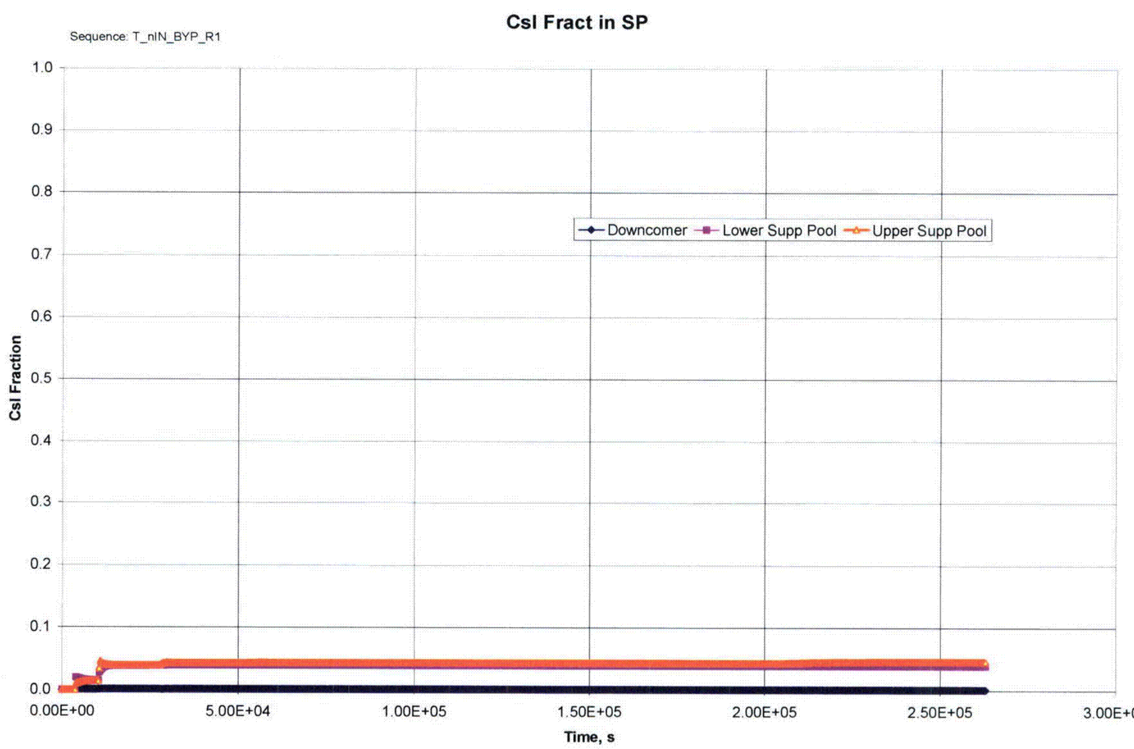


Figure 9A-3t – T_nIN_BYP_R1 CsI Fract in SP

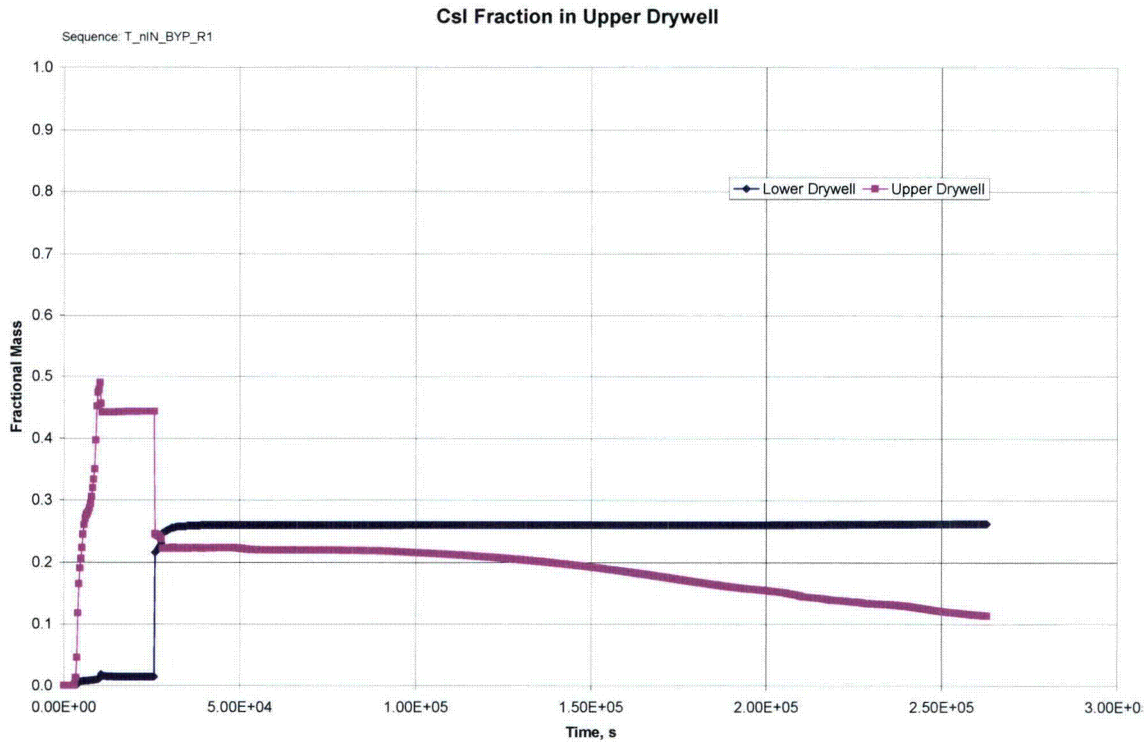


Figure 9A-3u – T_nIN_BYP_R1 CsI Fraction in Upper Drywell

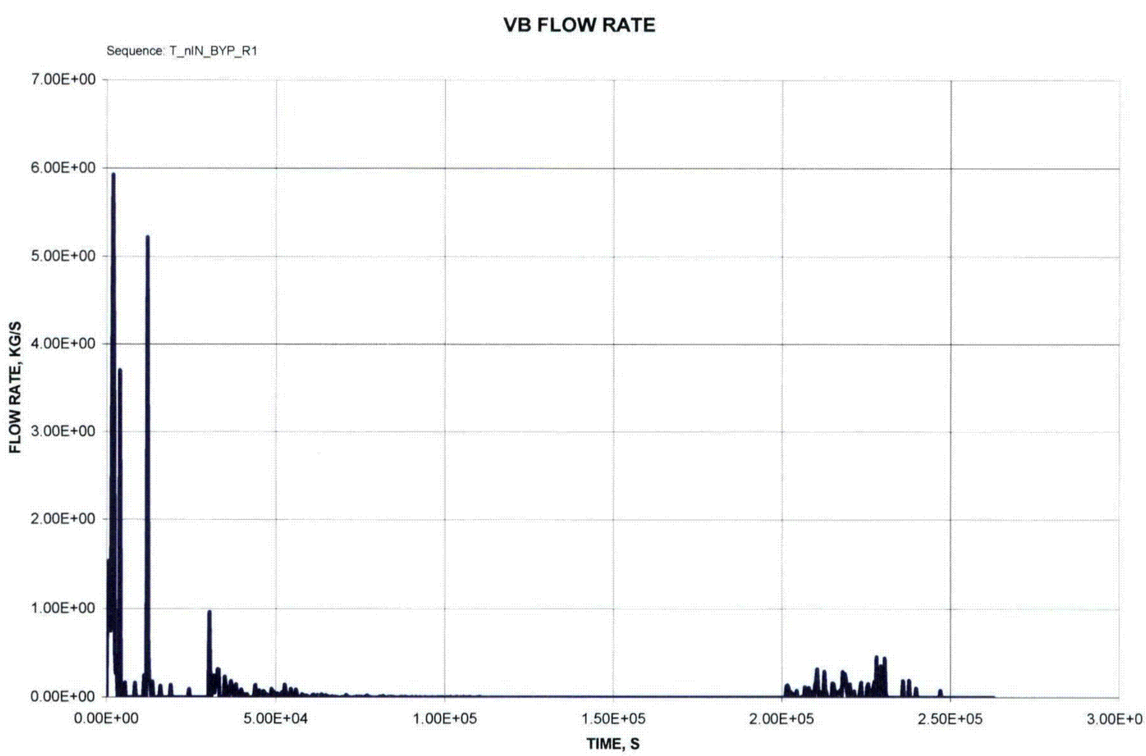


Figure 9A-3v – T_nIN_BYP_R1 VB Flow Rate

9A.4 Bypass RS (High Pressure) T_nDP_nIN_BYP_R1

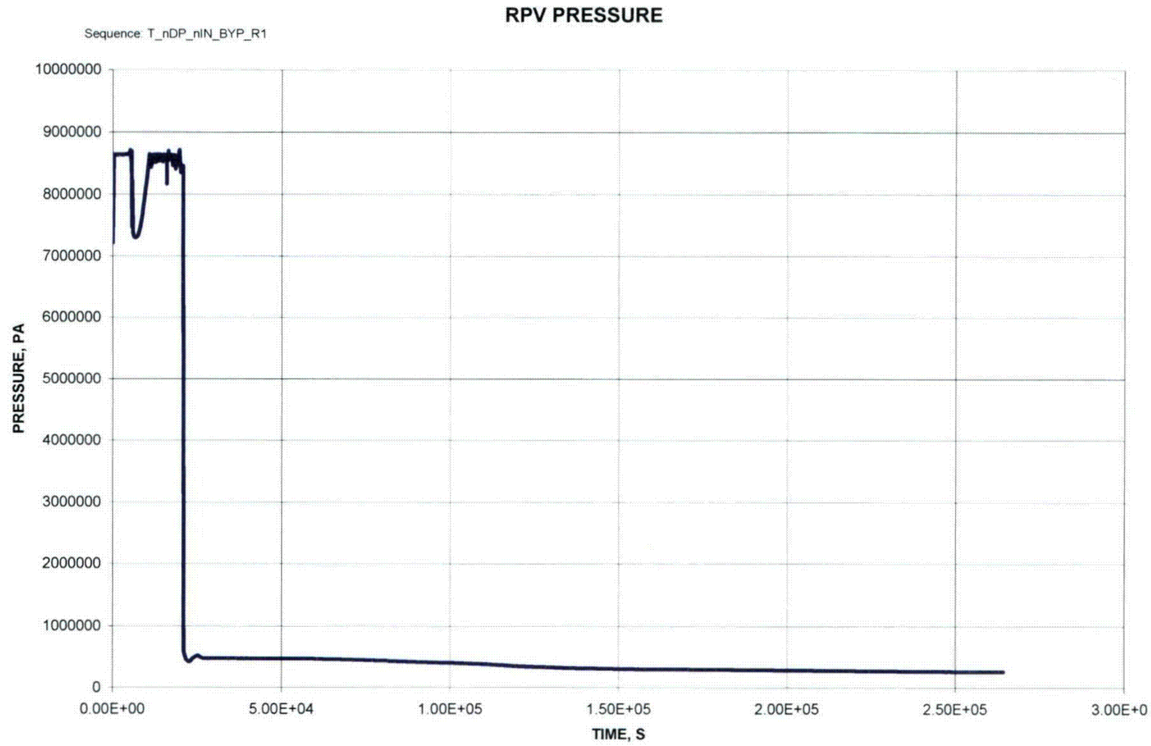


Figure 9A-4a - T_nDP_nIN_BYP_R1 RPV Pressure

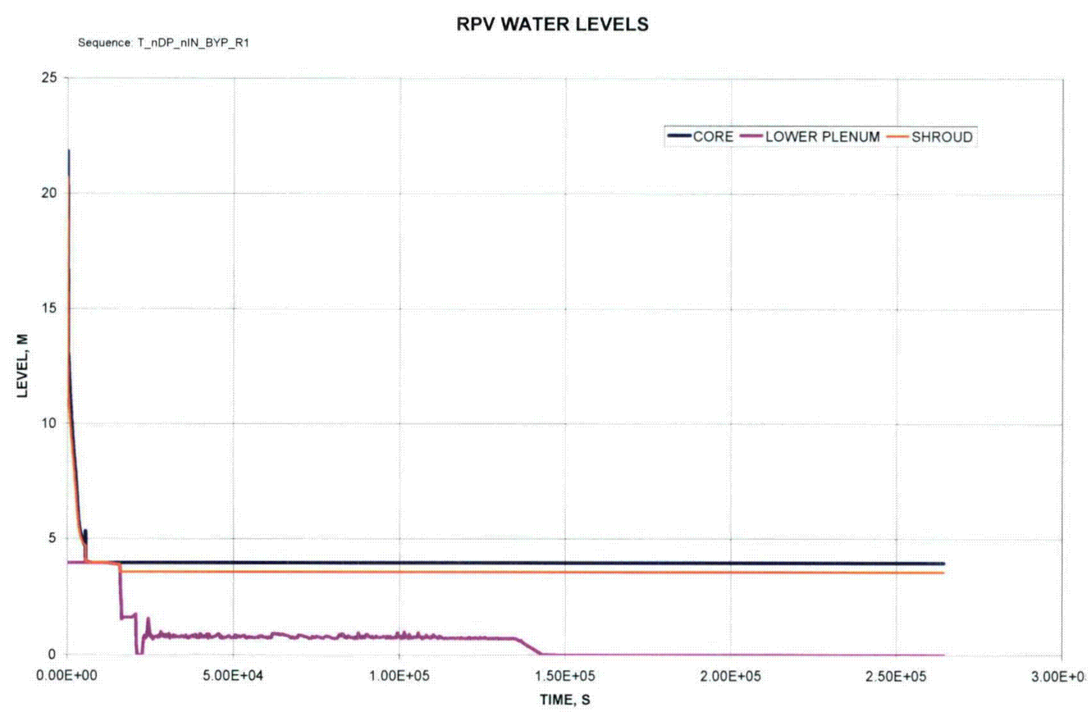


Figure 9A-4b - T_nDP_nIN_BYP_R1 RPV Water Levels

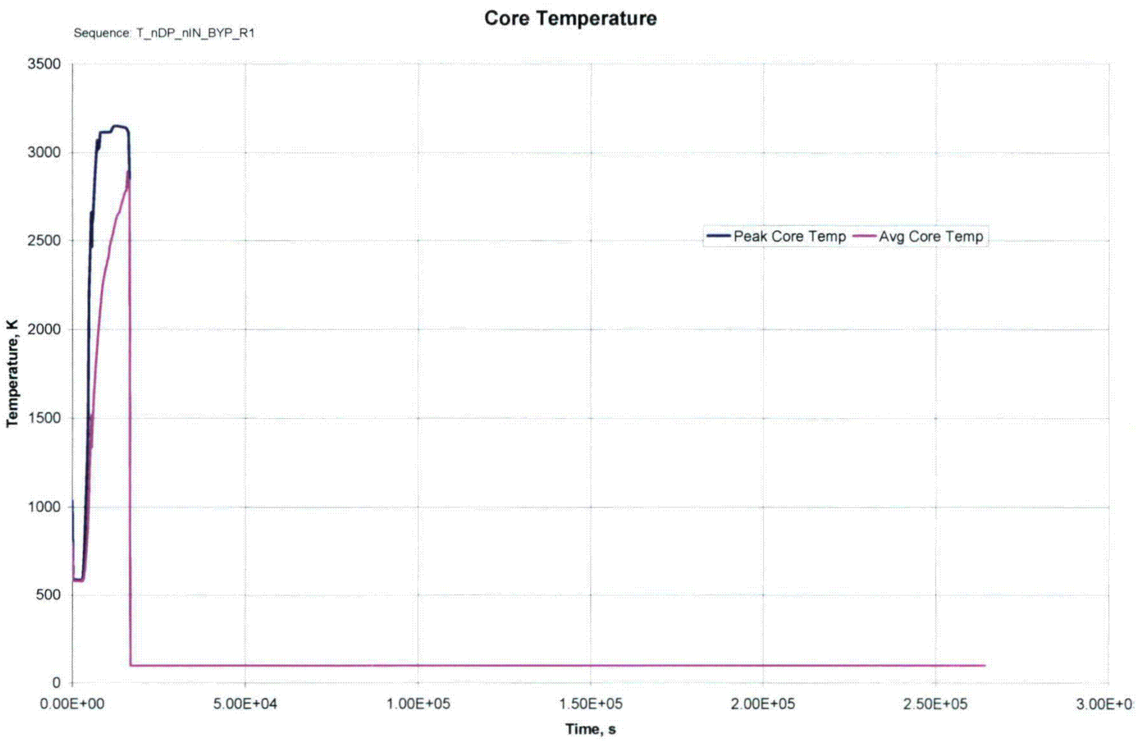


Figure 9A-4c - T_nDP_nIN_BYP_R1 Core Temperature

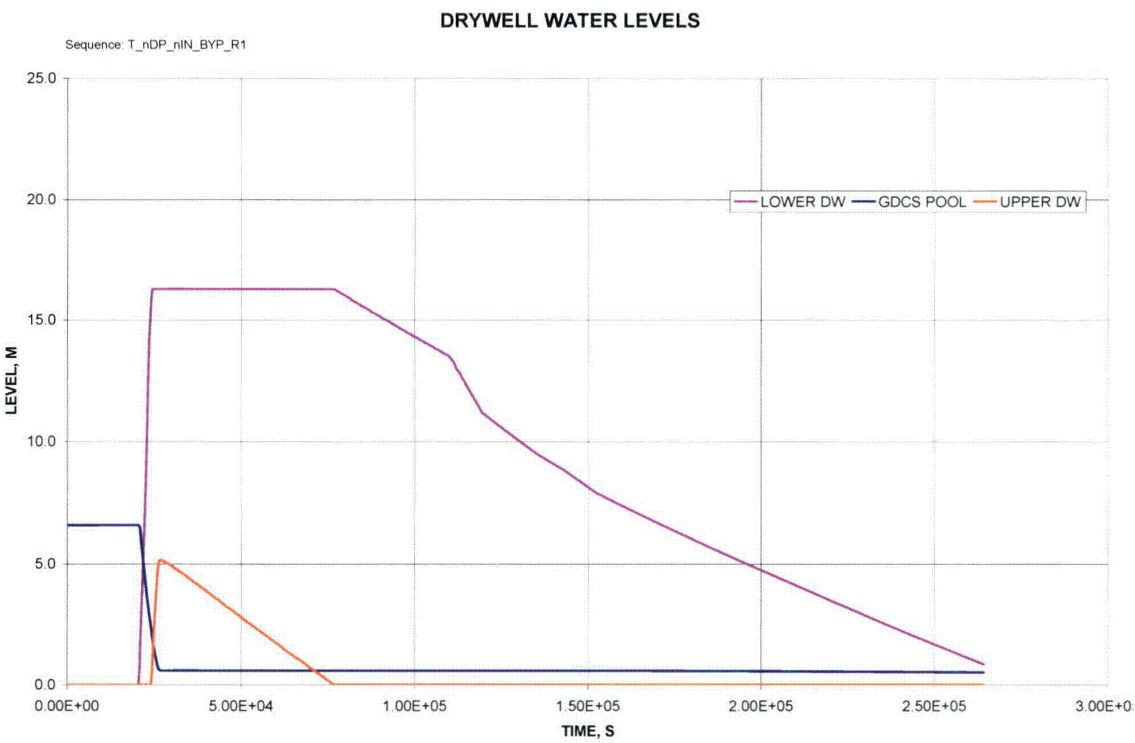


Figure 9A-4d - T_nDP_nIN_BYP_R1 Drywell Water Levels

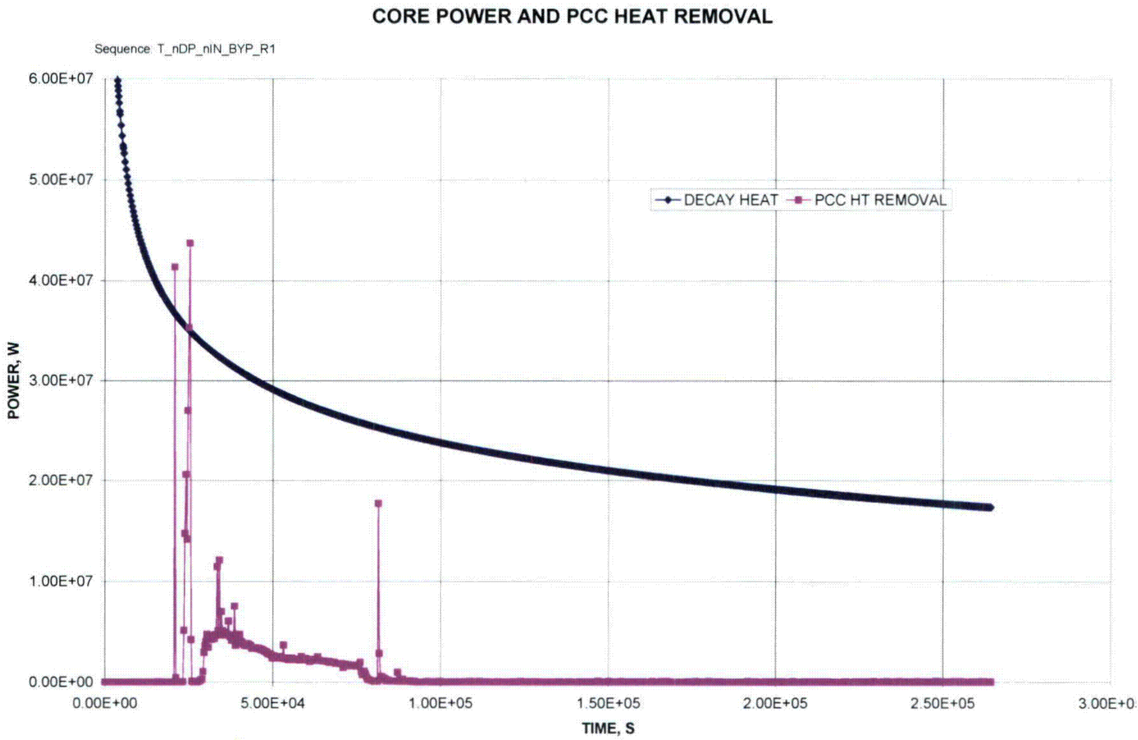


Figure 9A-4e - T_nDP_nIN_BYP_R1 Core Power and PCC Heat Removal

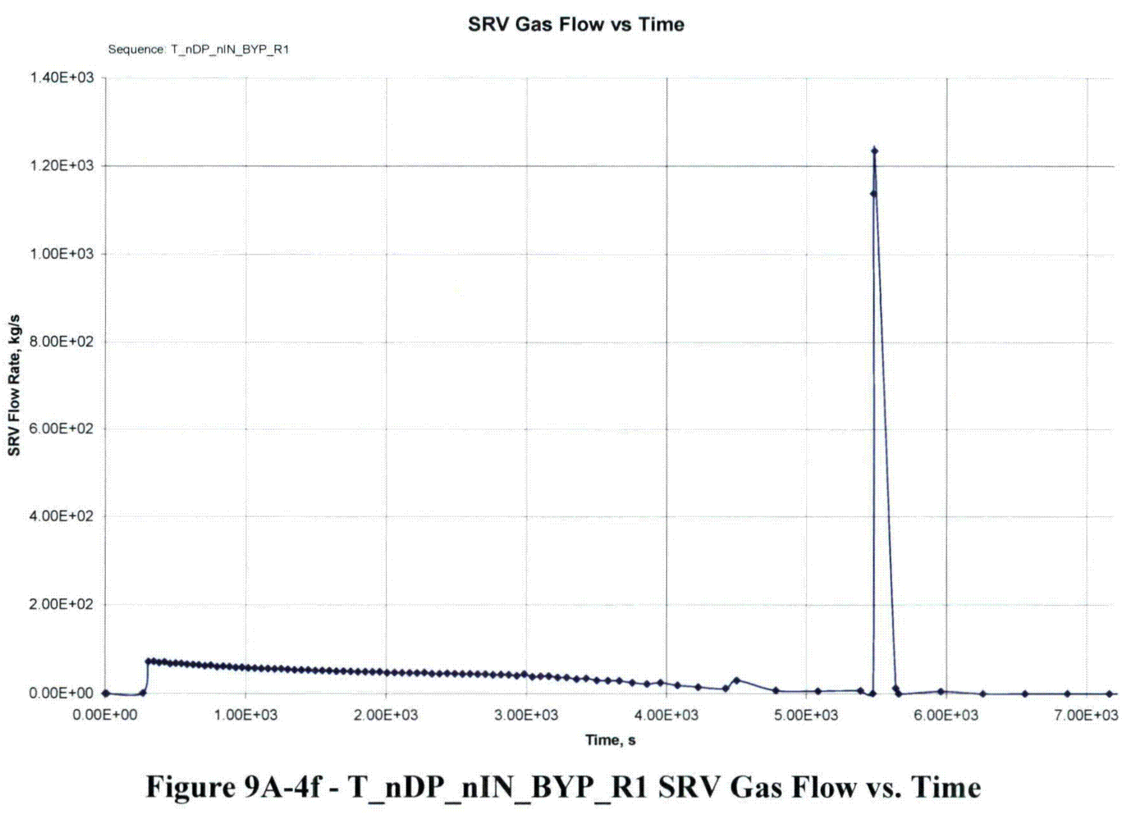


Figure 9A-4f - T_nDP_nIN_BYP_R1 SRV Gas Flow vs. Time

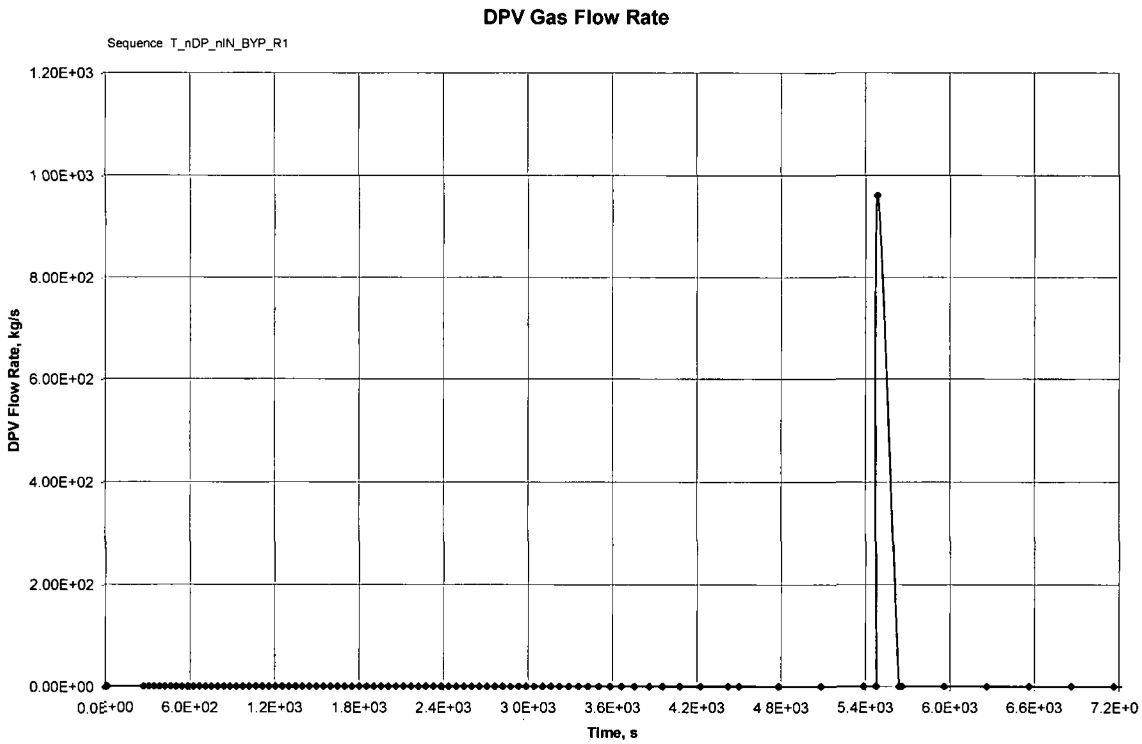


Figure 9A-4g - T_nDP_nIN_BYP_R1 DPV Gas Flow Rate

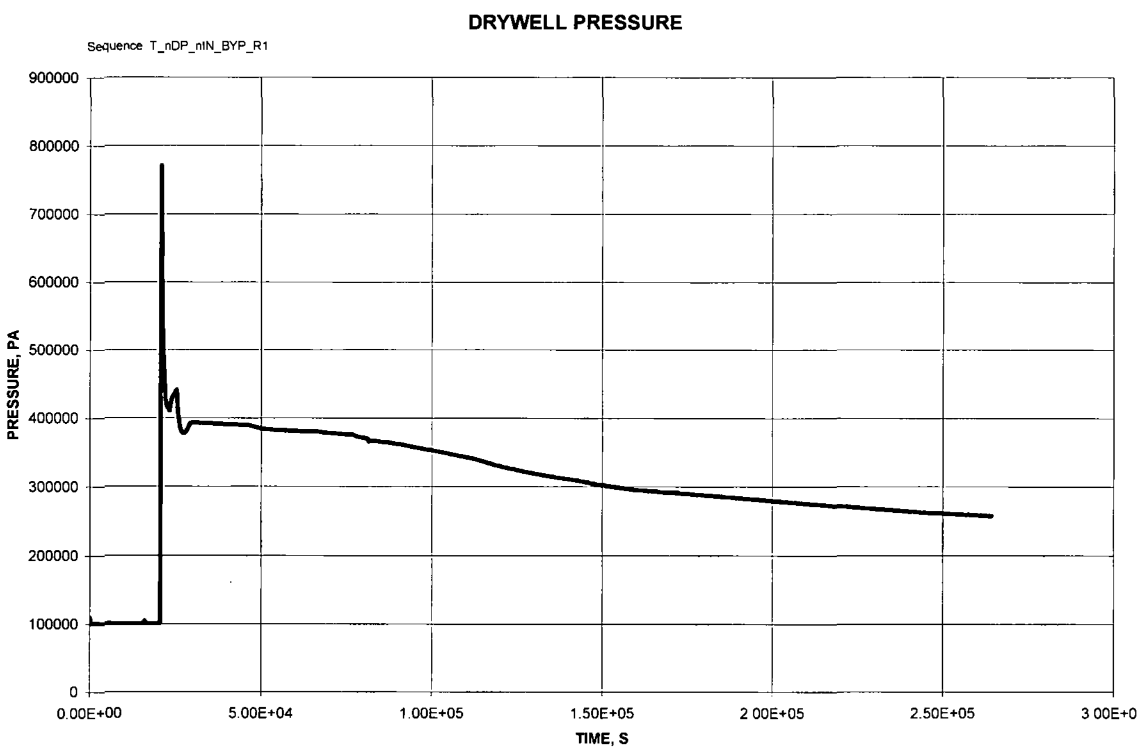


Figure 9A-4h - T_nDP_nIN_BYP_R1 Drywell Pressure

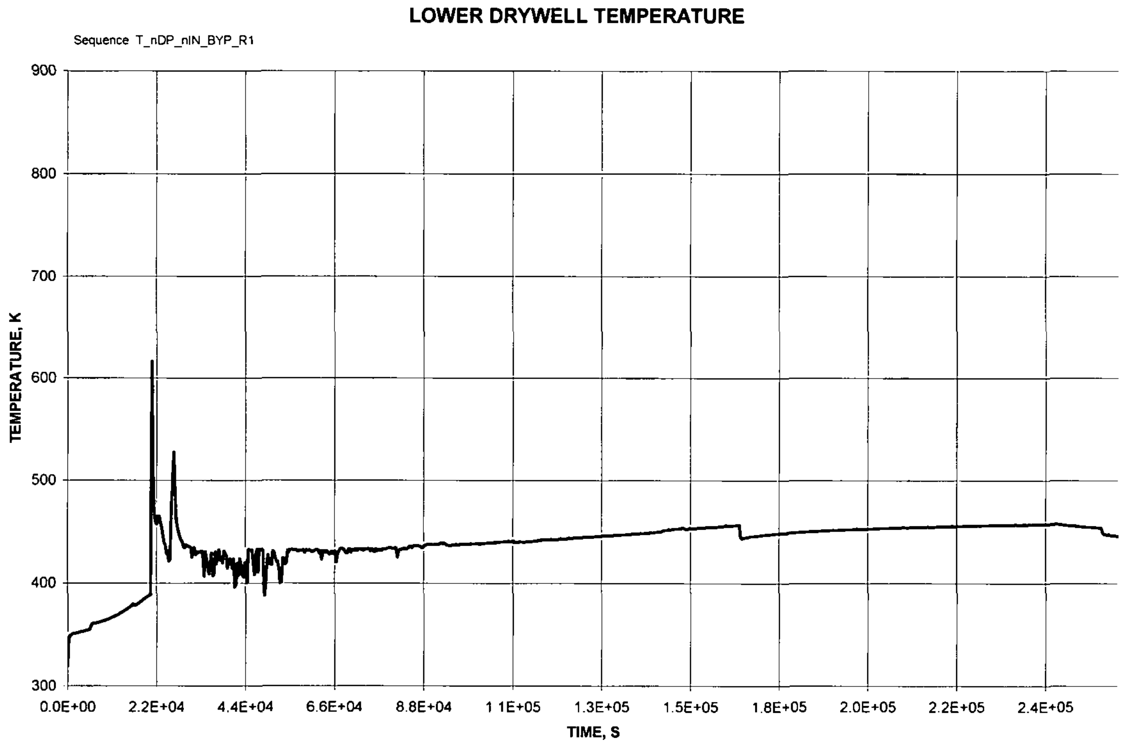


Figure 9A-4i - T_nDP_nIN_BYP_R1 Lower Drywell Temperature

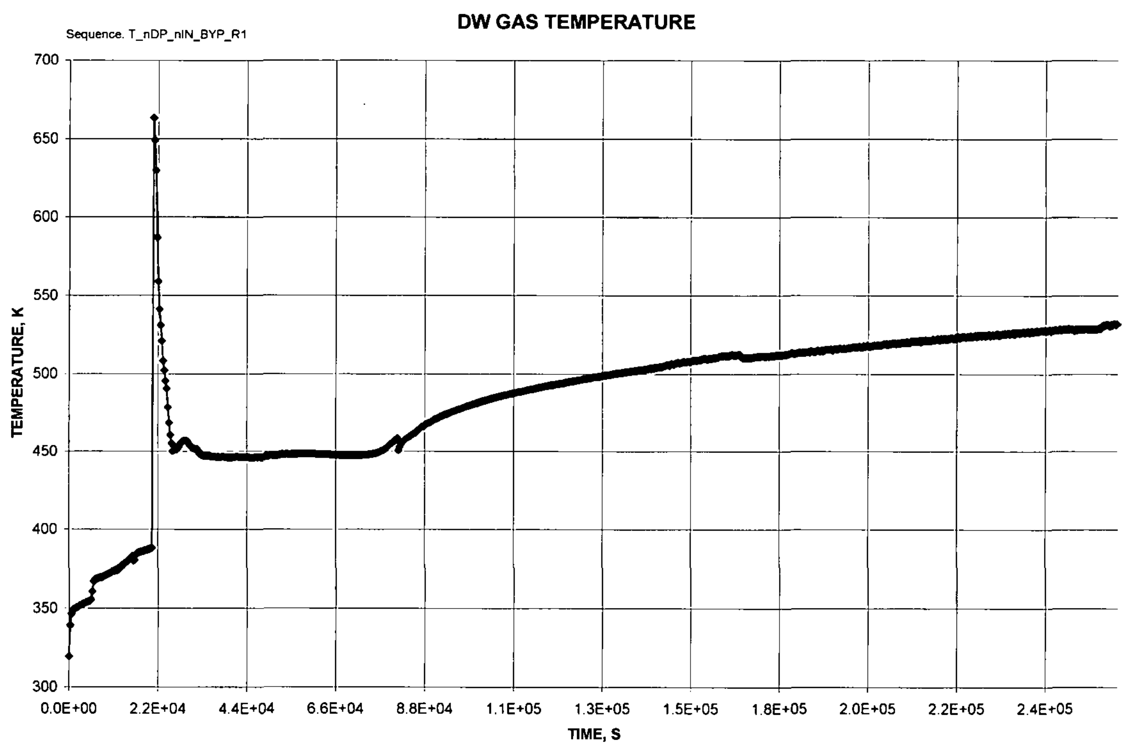


Figure 9A-4j - T_nDP_nIN_BYP_R1 DW Gas Temperature

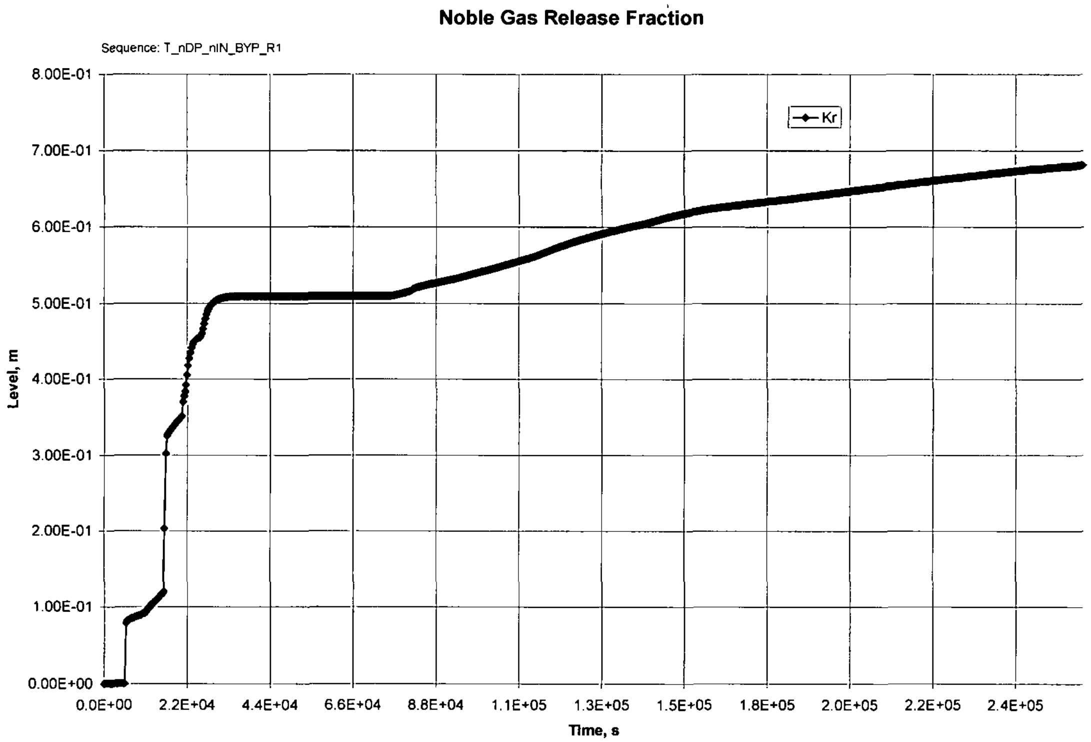


Figure 9A-4k - T_nDP_nIN_BYP_R1 Noble Gas Release Fraction

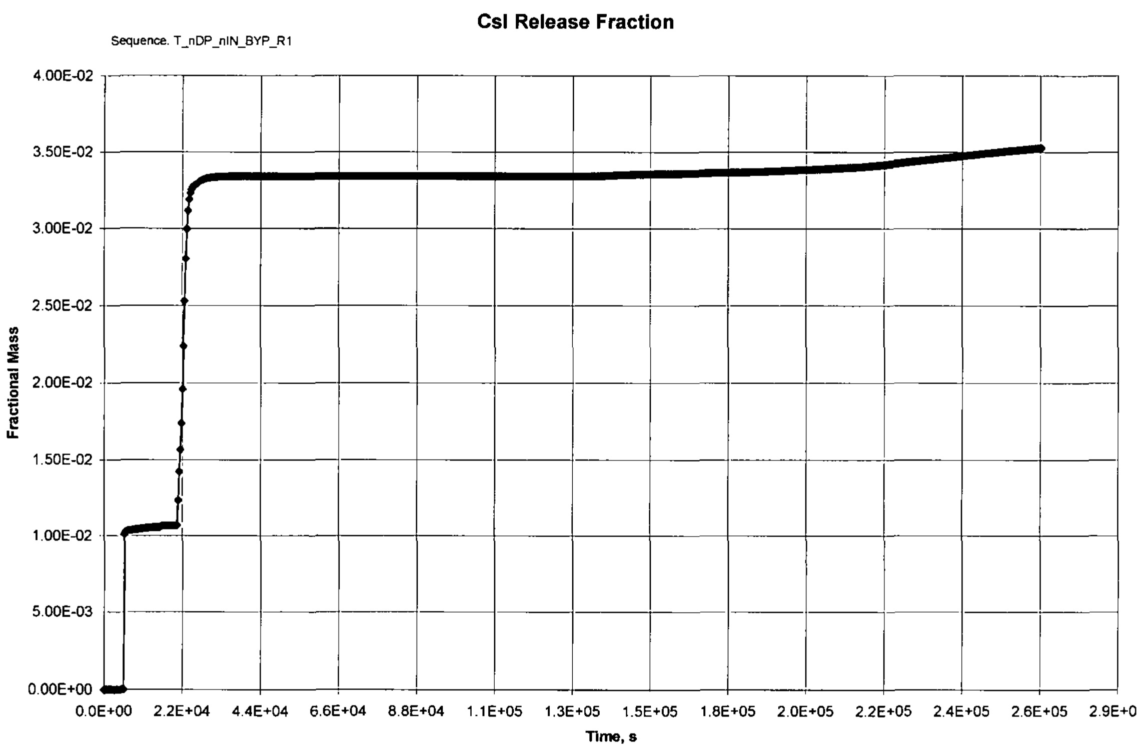


Figure 9A-4l - T_nDP_nIN_BYP_R1 CsI Release Fraction

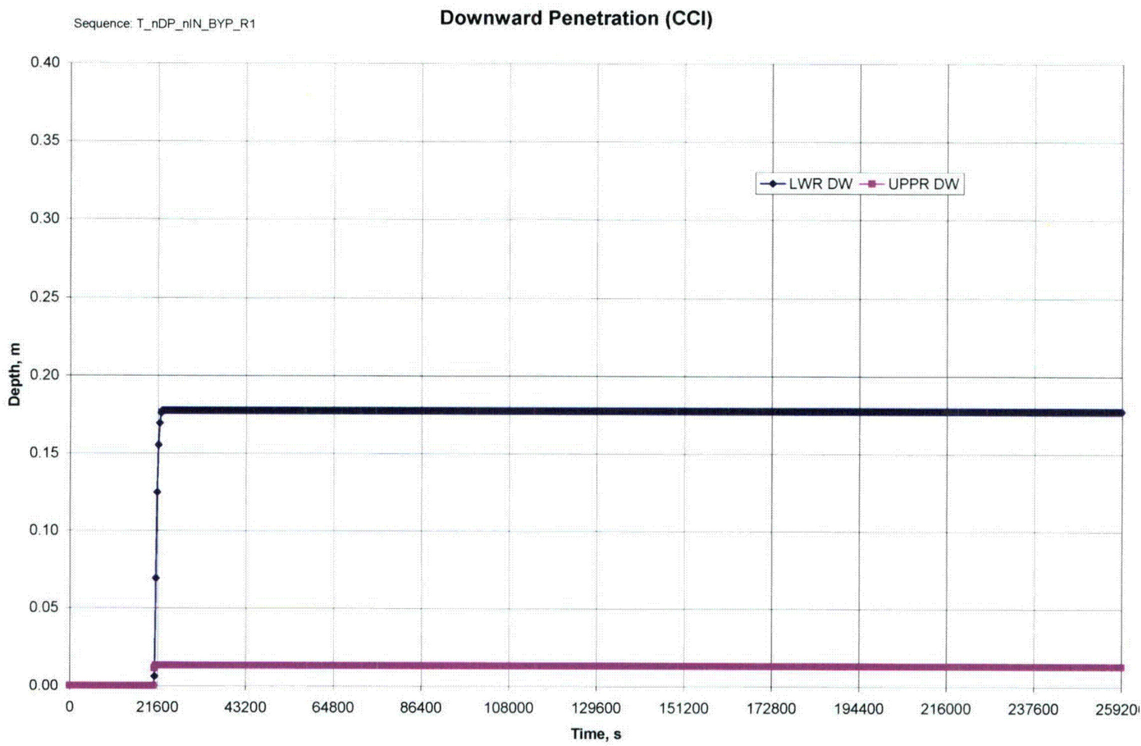


Figure 9A-4m - T_nDP_nIN_BYP_R1 Downward Penetration (CCI)

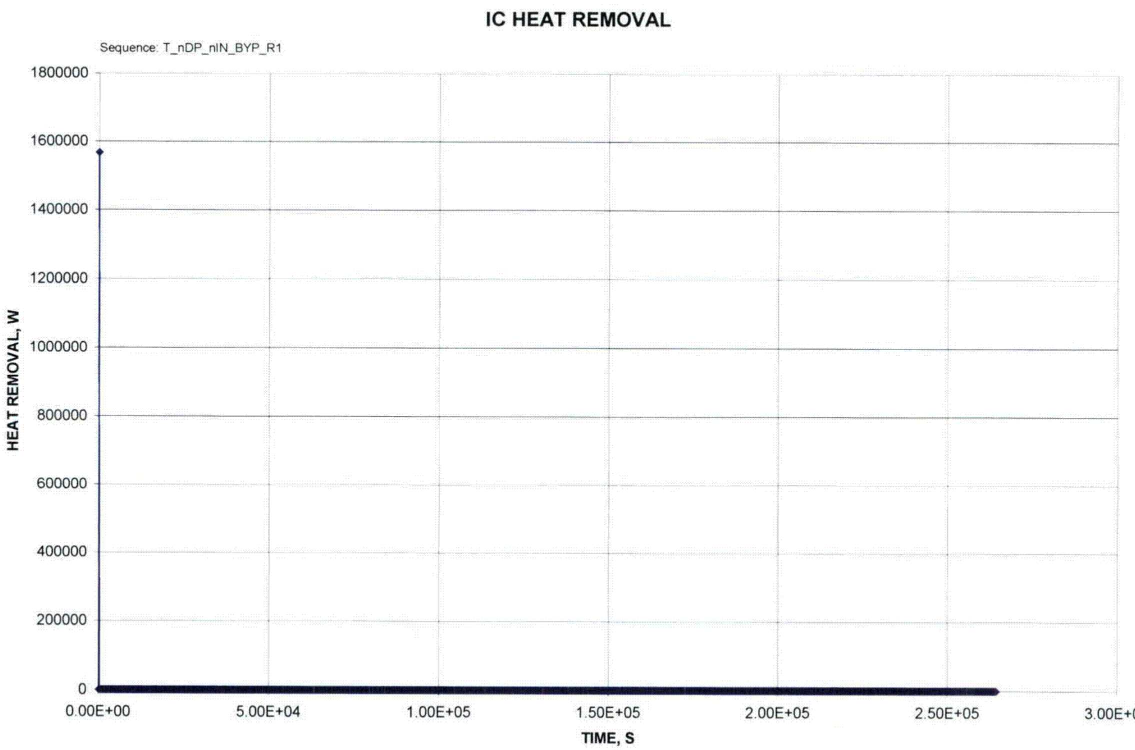


Figure 9A-4n - T_nDP_nIN_BYP_R1 IC Heat Removal

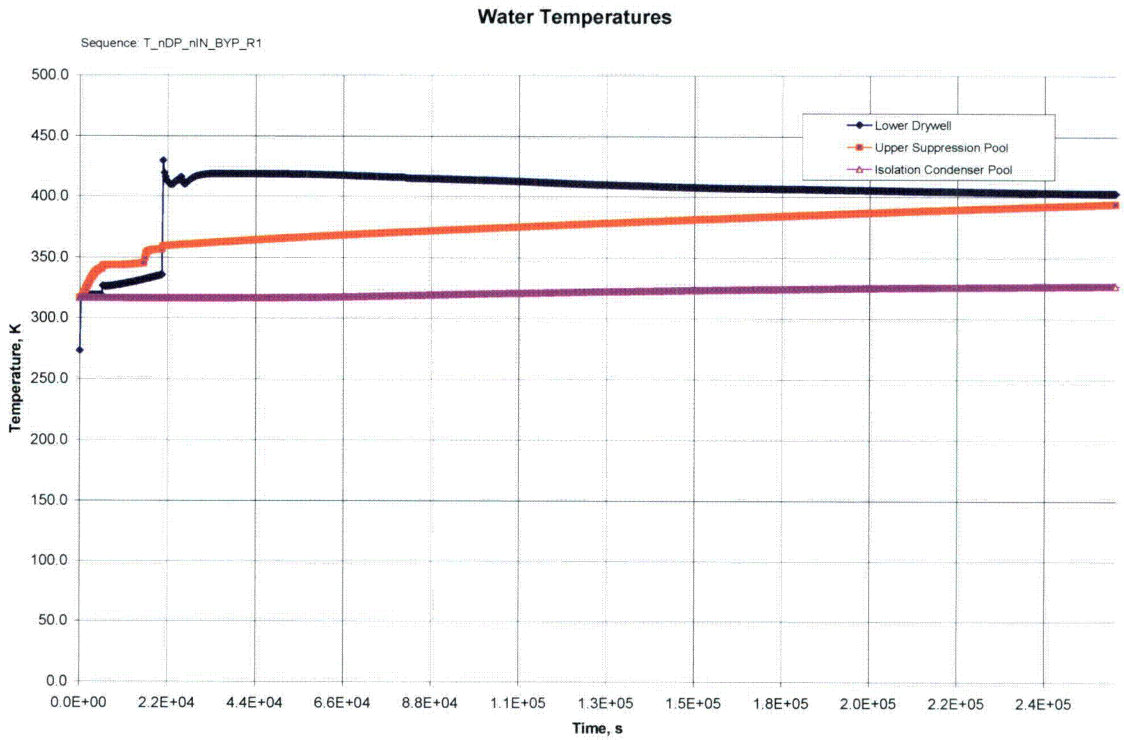


Figure 9A-4o - T_nDP_nIN_BYP_R1 IC Water Temperatures

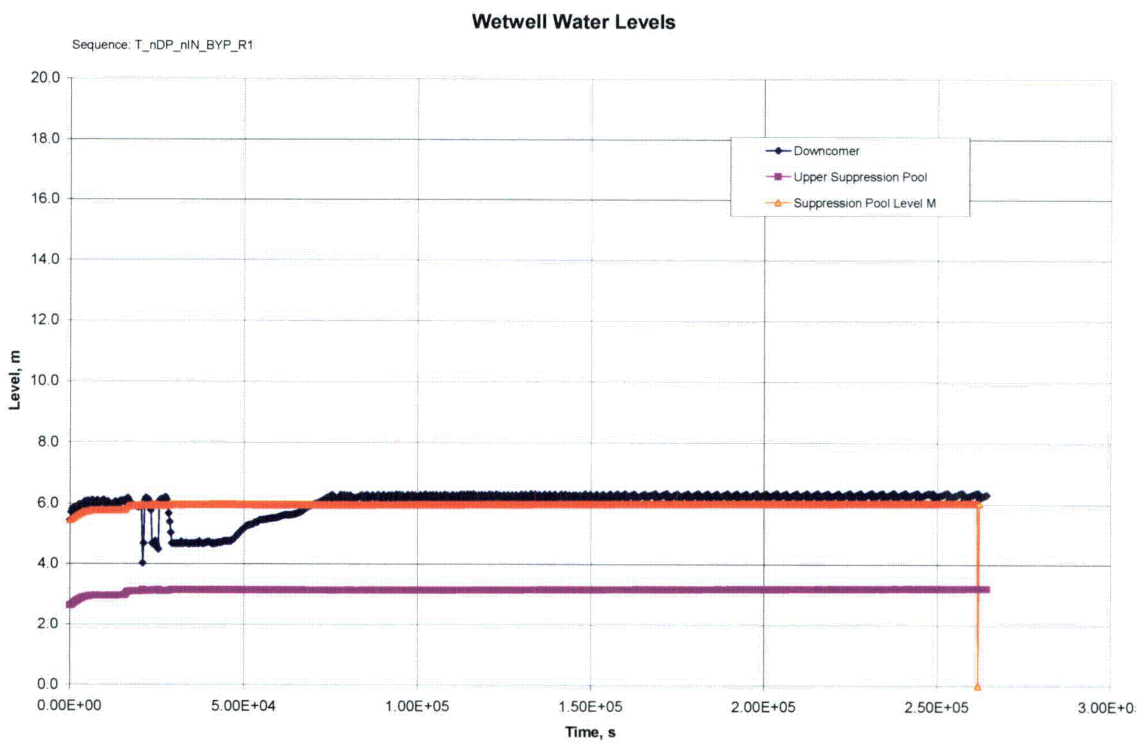


Figure 9A-4p - T_nDP_nIN_BYP_R1 Wetwell Water Levels

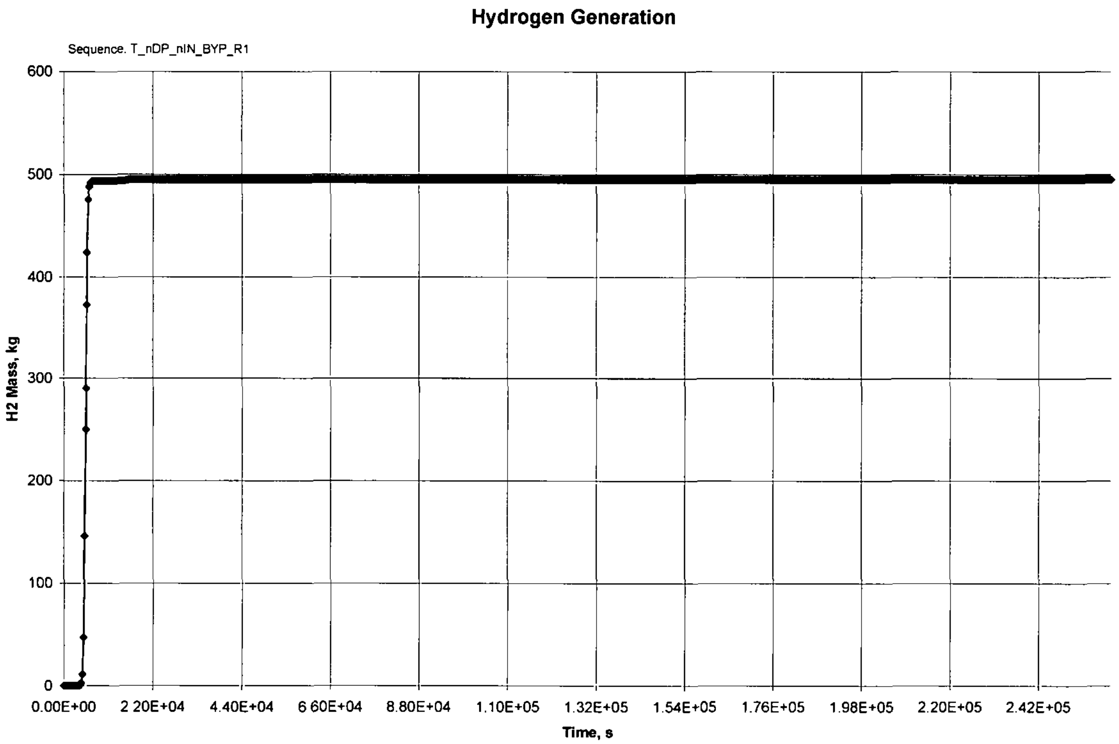


Figure 9A-4q - T_nDP_nIN_BYP_R1 Hydrogen Generation

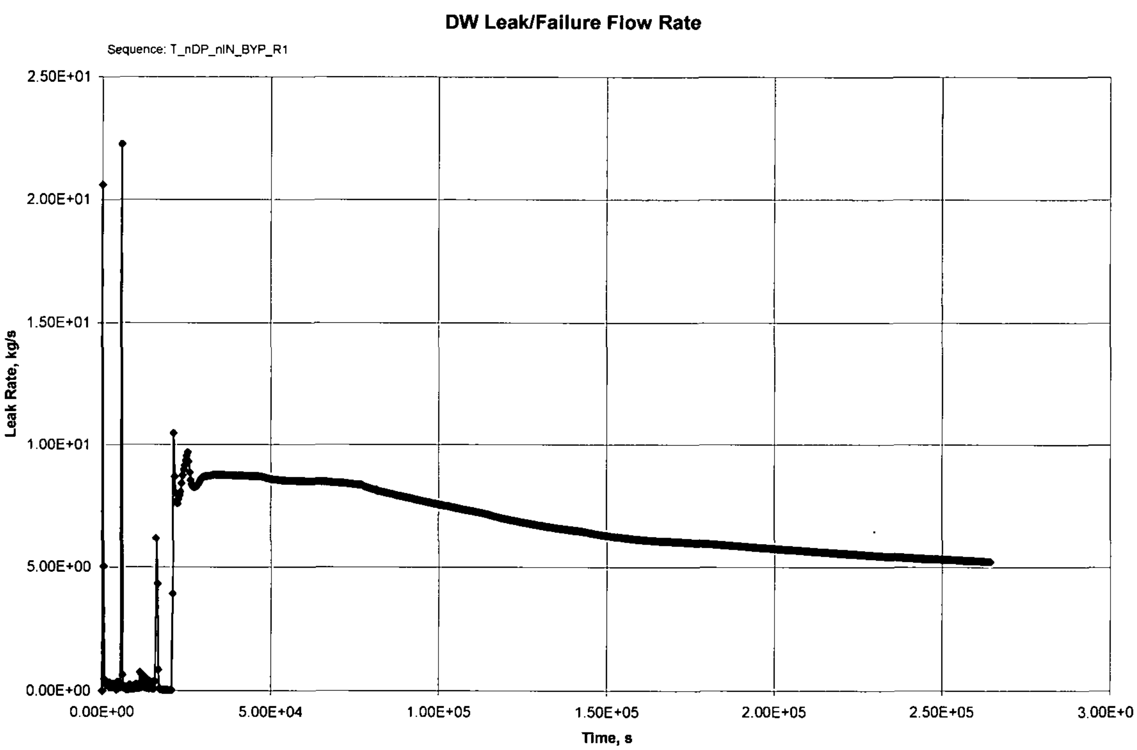


Figure 9A-4r - T_nDP_nIN_BYP_R1 DW Leak/Failure Flow Rate

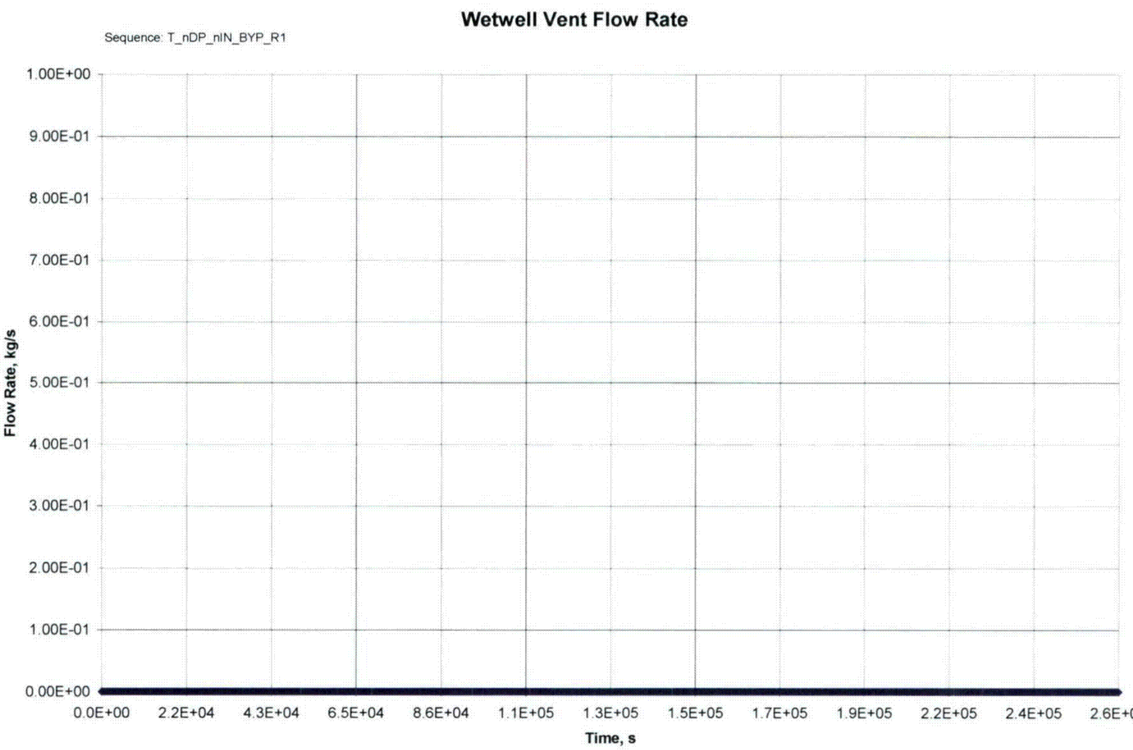


Figure 9A-4s - T_nDP_nIN_BYP_R1 Wetwell Vent Flow Rate

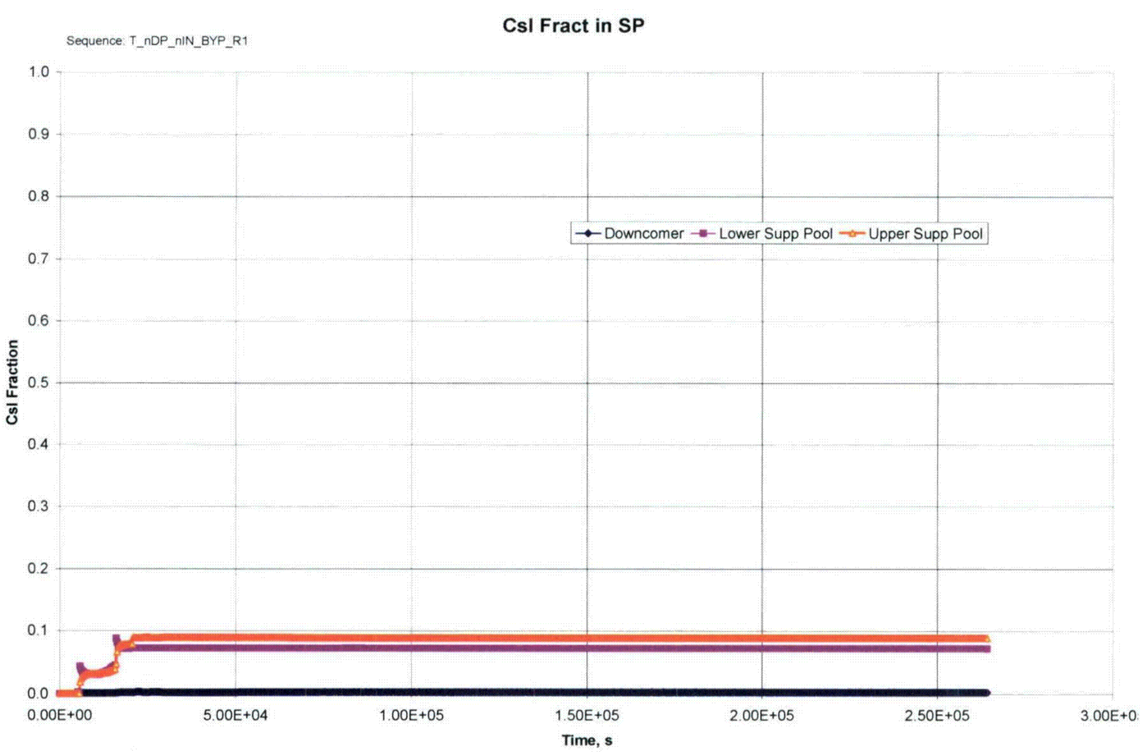


Figure 9A-4t - T_nDP_nIN_BYP_R1 CsI Fract in SP

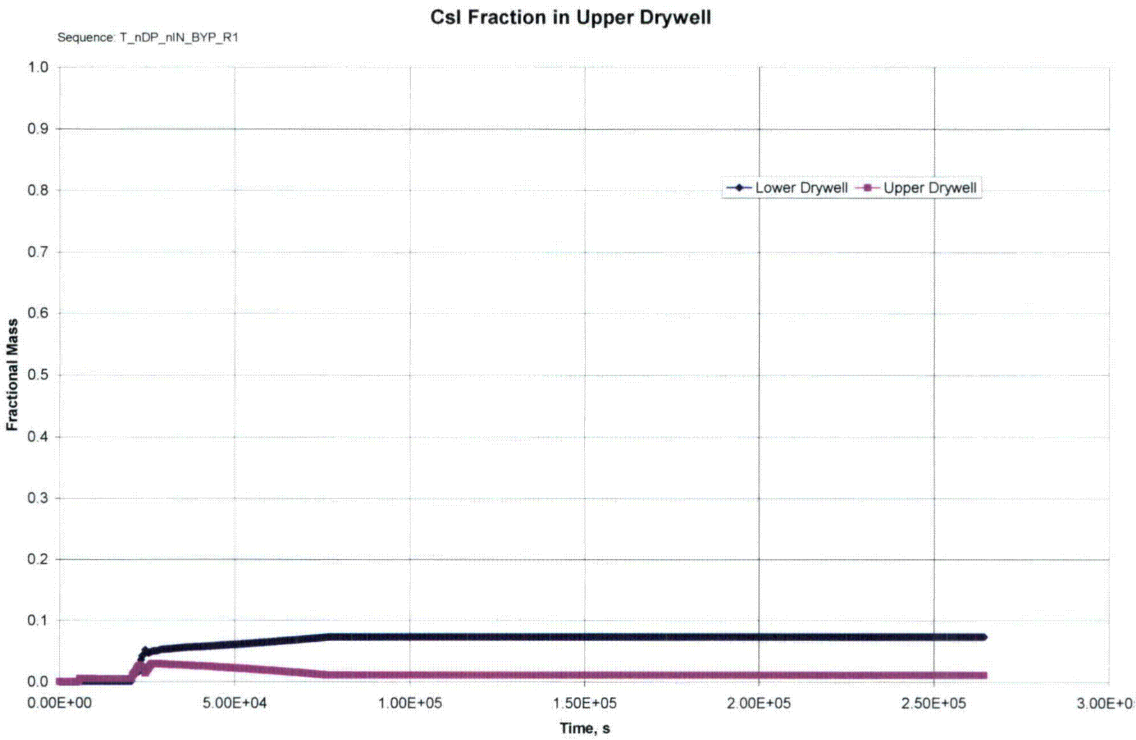


Figure 9A-4u - T_nDP_nIN_BYP_R1 CsI Fraction in Upper Drywell

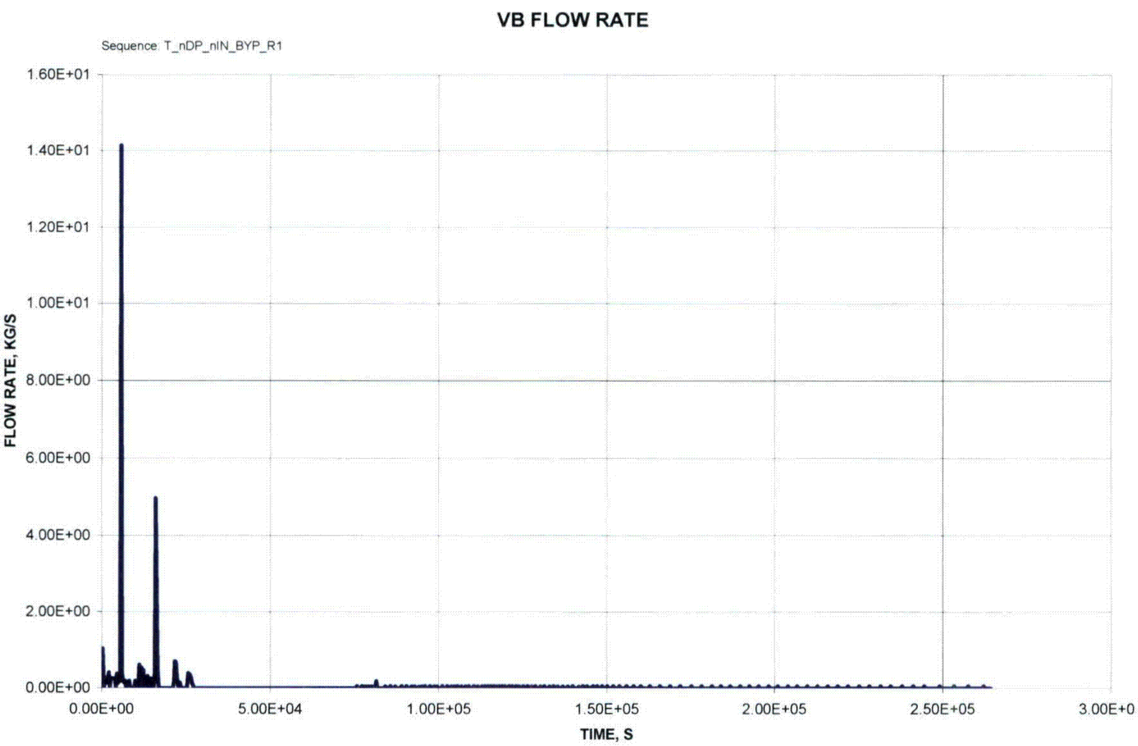


Figure 9A-4v - T_nDP_nIN_BYP_R1 VB Flow Rate

9A.5 Core Concrete Interaction (Dry) - High Pressure T_nIN_nD_CCID_R1

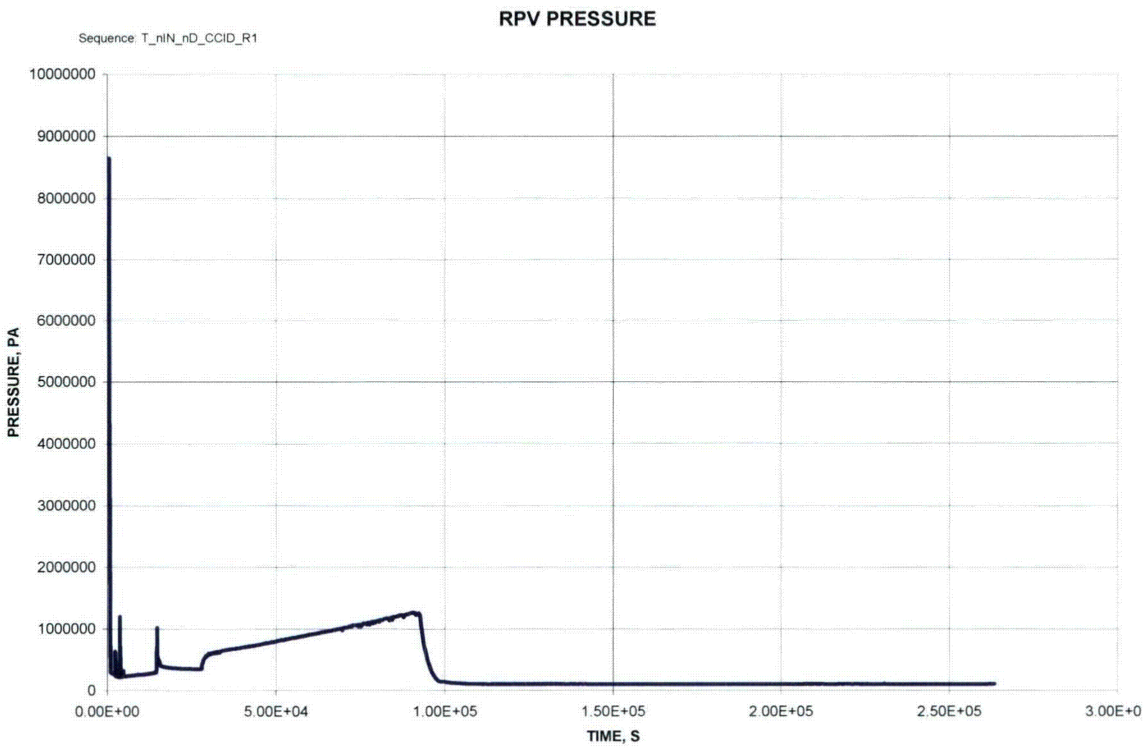


Figure 9A-5a - T_nIN_nD_CCID_R1 RPV Pressure

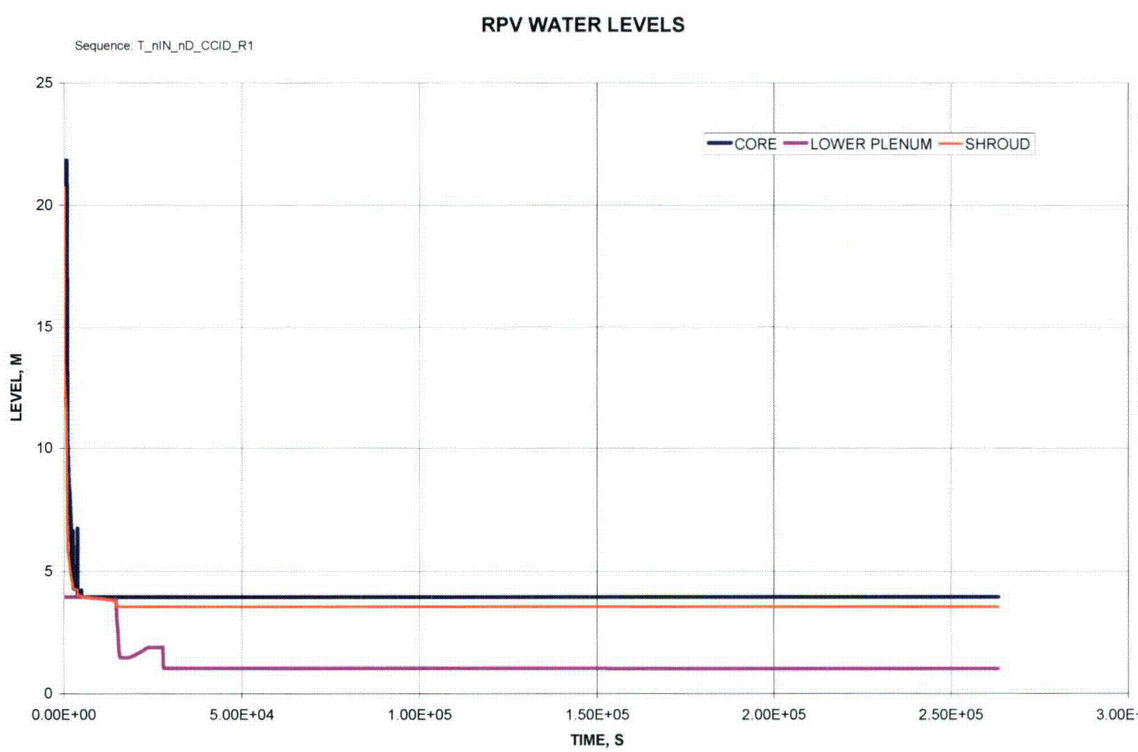


Figure 9A-5b - T_nIN_nD_CCID_R1 RPV Water Levels

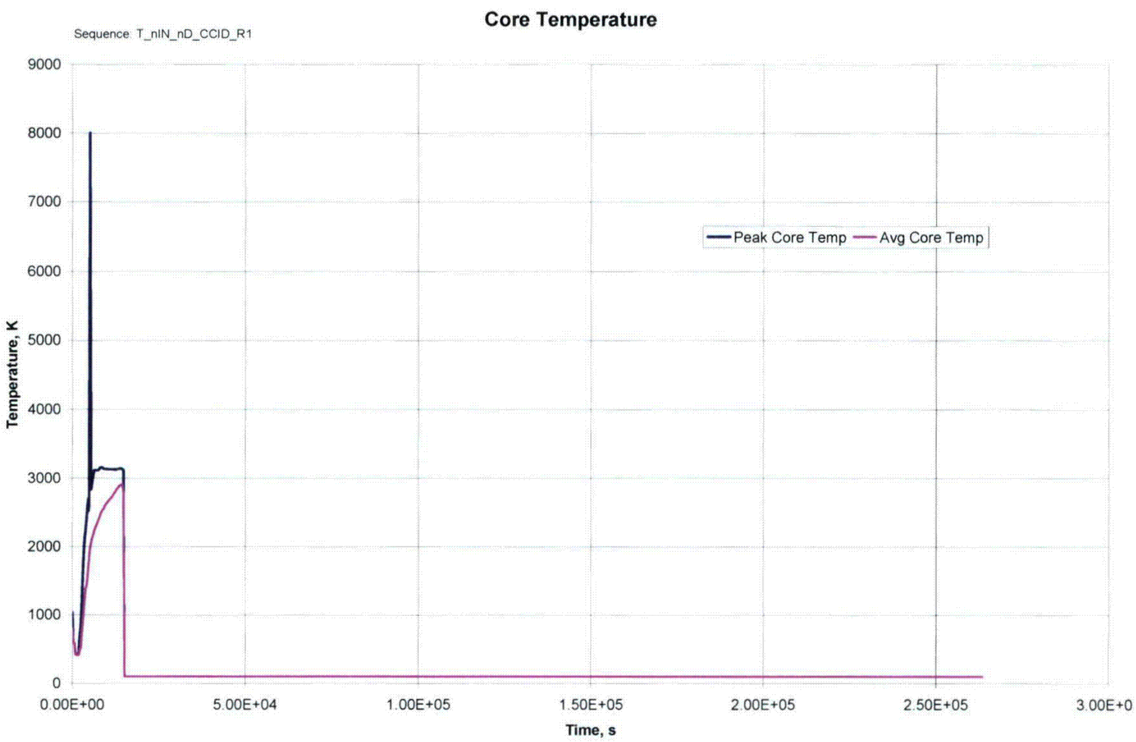


Figure 9A-5c - T_nIN_nD_CCID_R1 Core Temperature

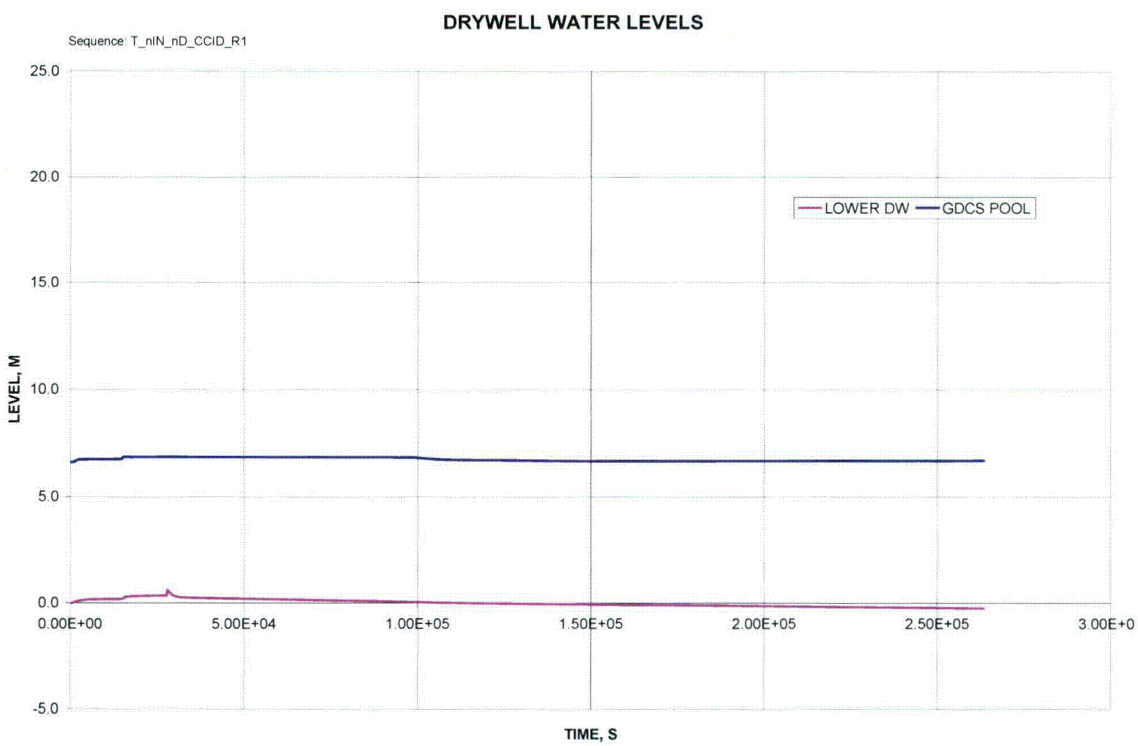


Figure 9A-5d - T_nIN_nD_CCID_R1 Drywell Water Levels

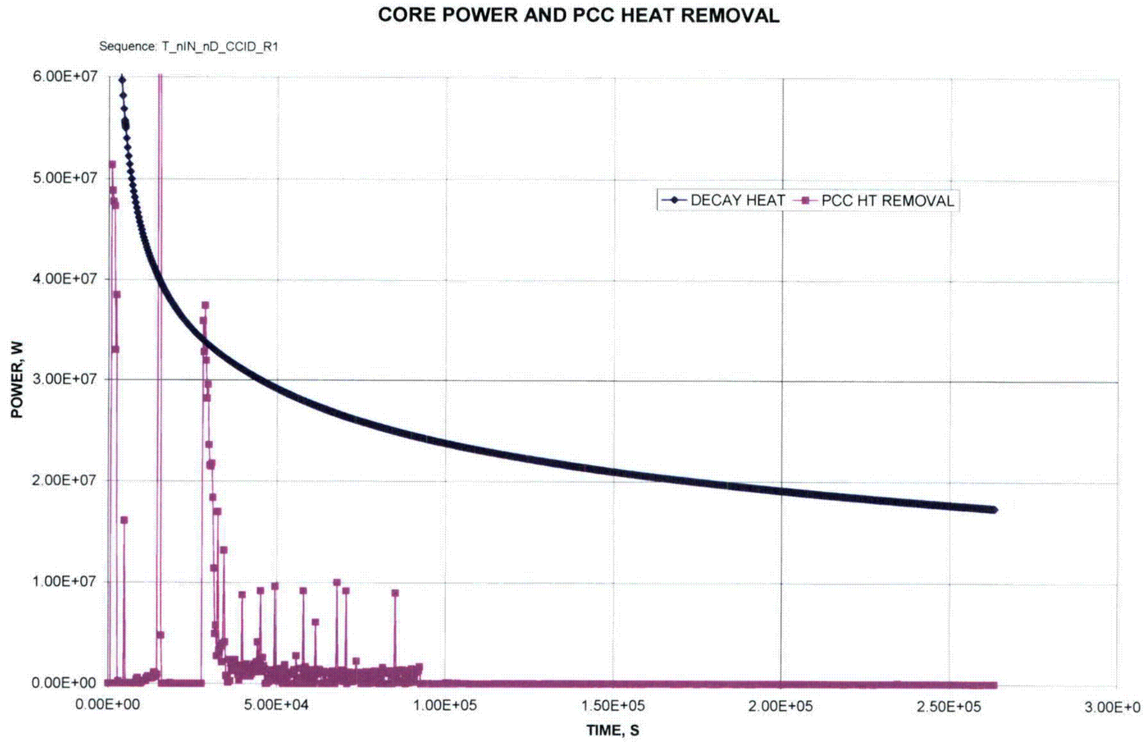


Figure 9A-5e - T_nIN_nD_CCID_R1 Core Power and PCCS Heat Removal

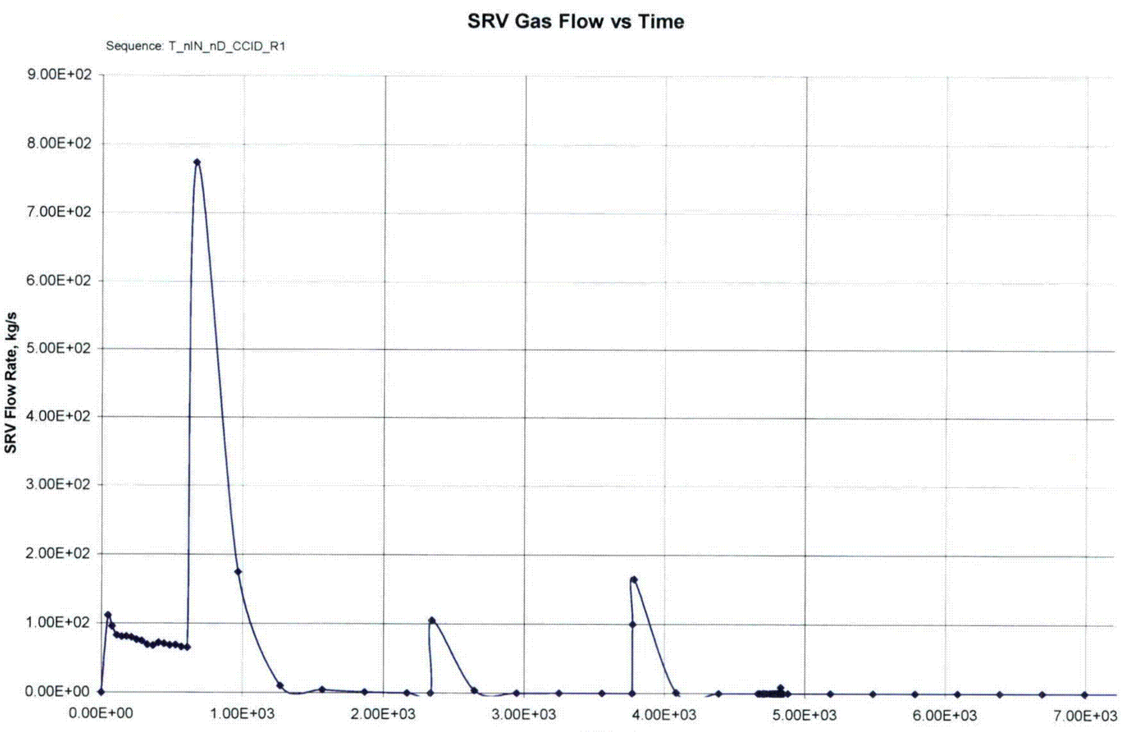


Figure 9A-5f - T_nIN_nD_CCID_R1 SRV Gas Flow vs Time

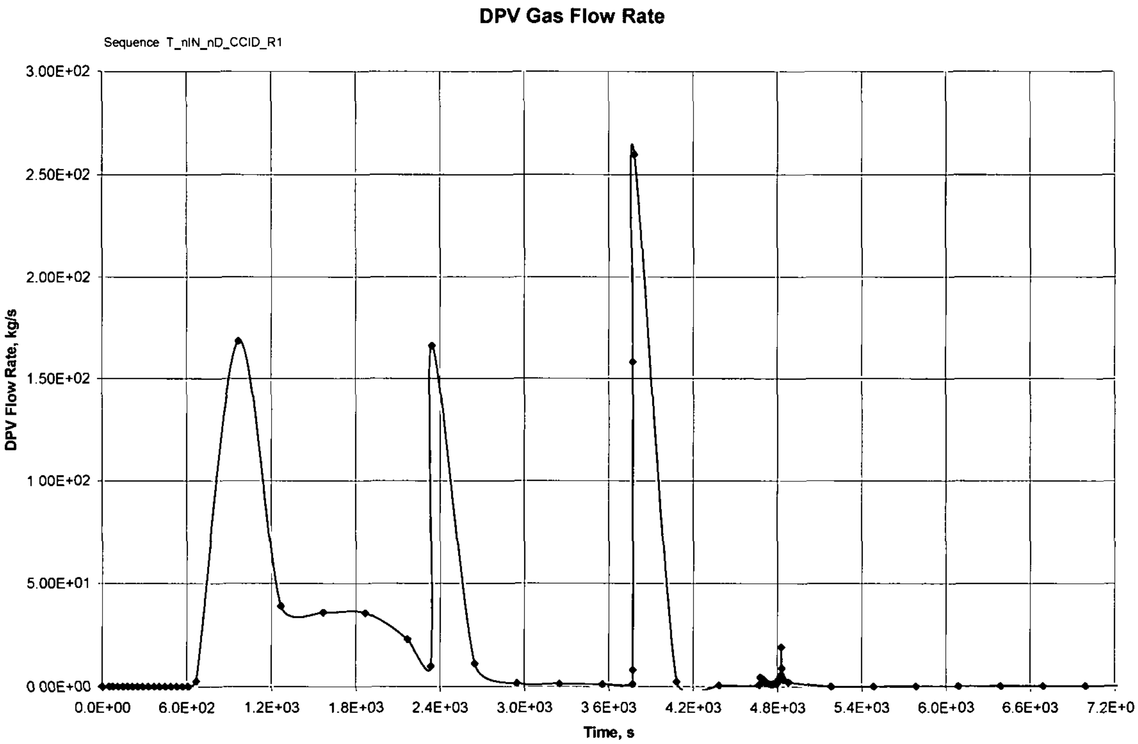


Figure 9A-5g - T_nIN_nD_CCID_R1 DPV Gas Flow Rate

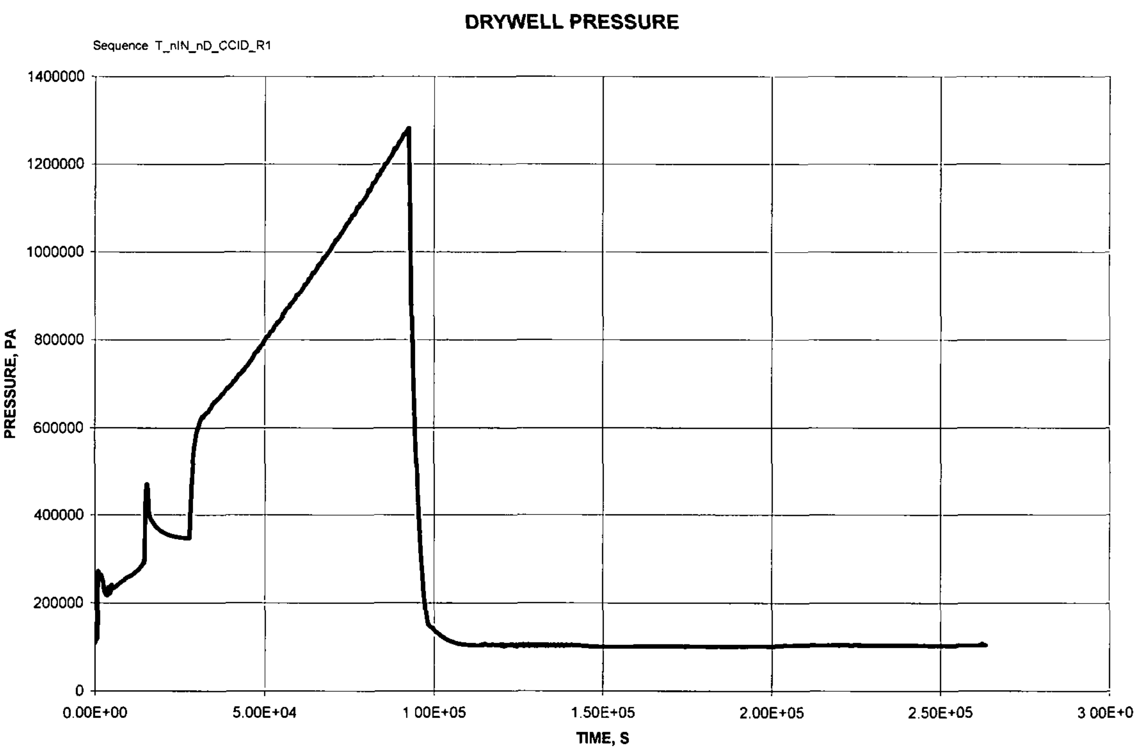


Figure 9A-5h - T_nIN_nD_CCID_R1 Drywell Pressure

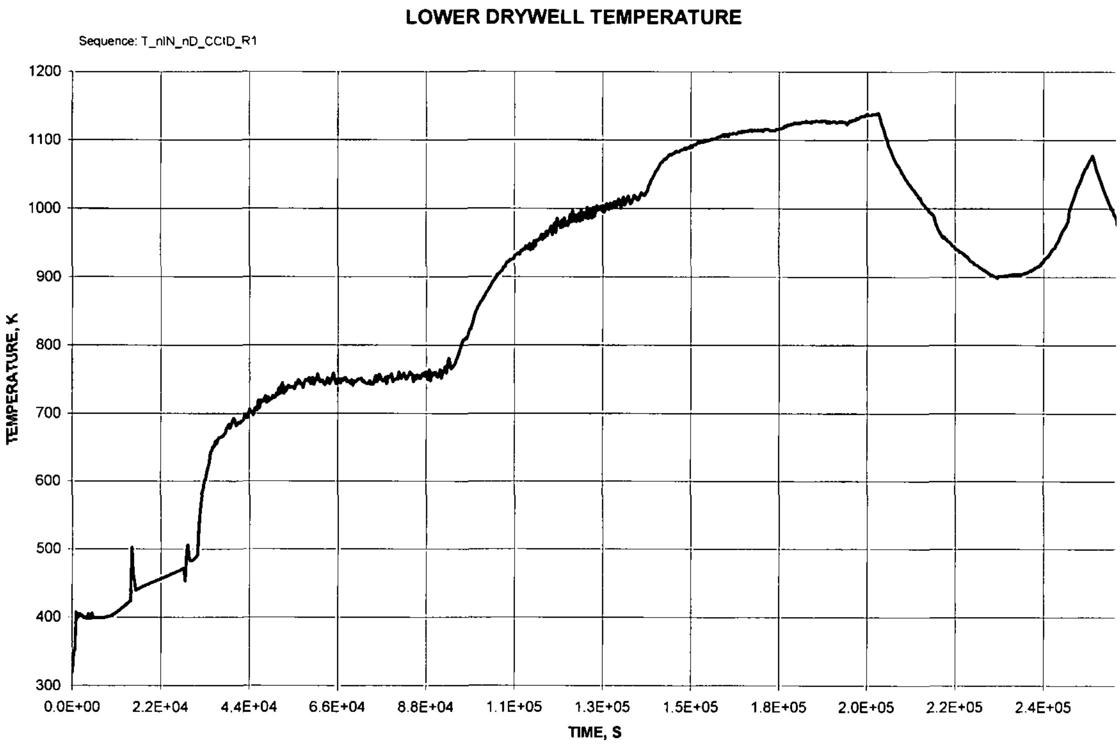


Figure 9A-5i - T_nIN_nD_CCID_R1 Lower Drywell Temperature

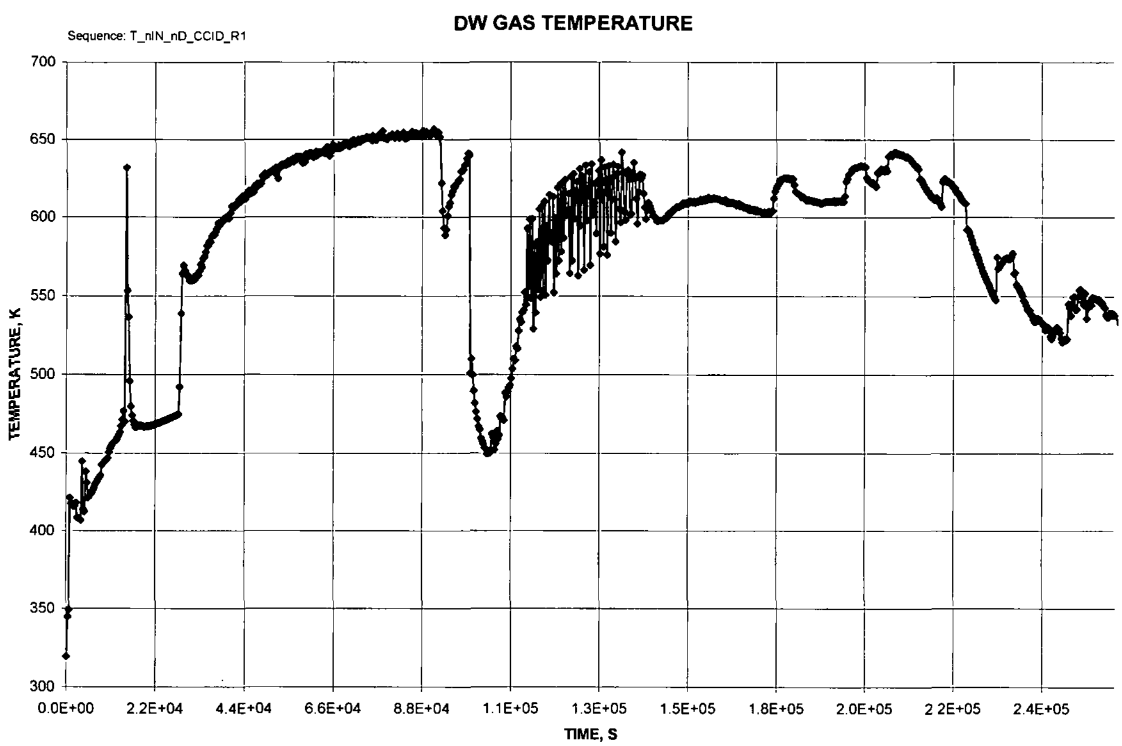


Figure 9A-5j - T_nIN_nD_CCID_R1 DW Gas Temperature

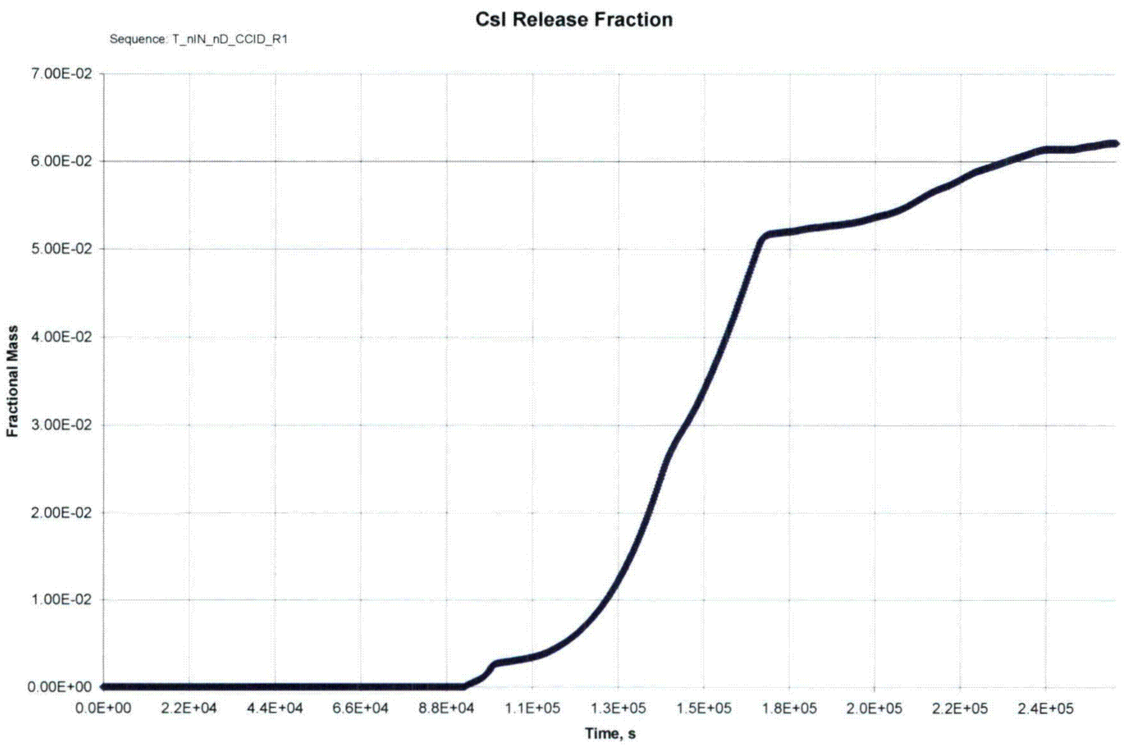


Figure 9A-5l - T_nIN_nD_CCID_R1 CsI Release Fraction

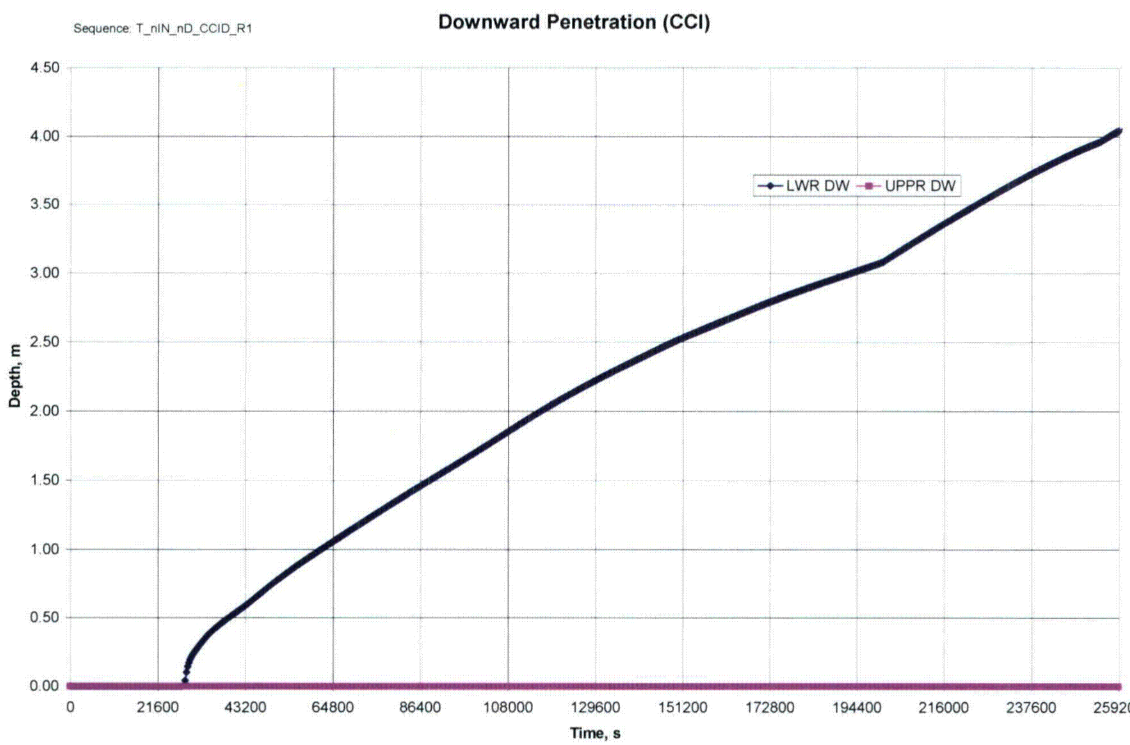


Figure 9A-5m - T_nIN_nD_CCID_R1 Downward Penetration

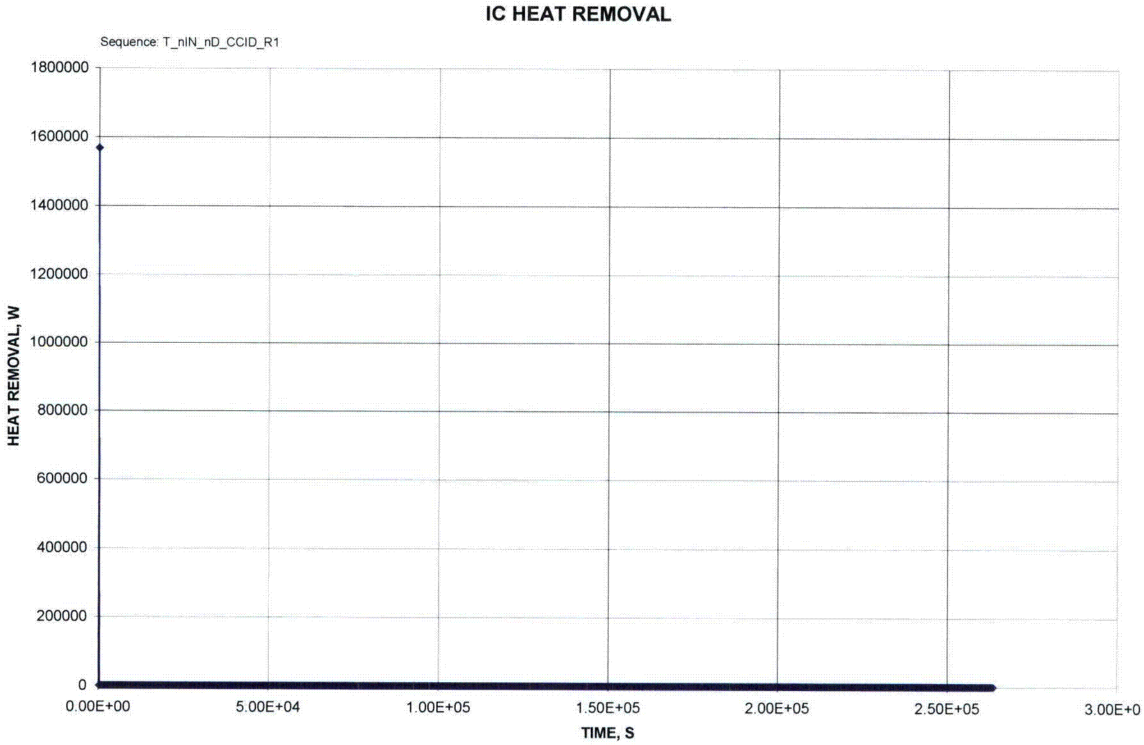


Figure 9A-5n - T_nIN_nD_CCID_R1 ICS Heat Removal

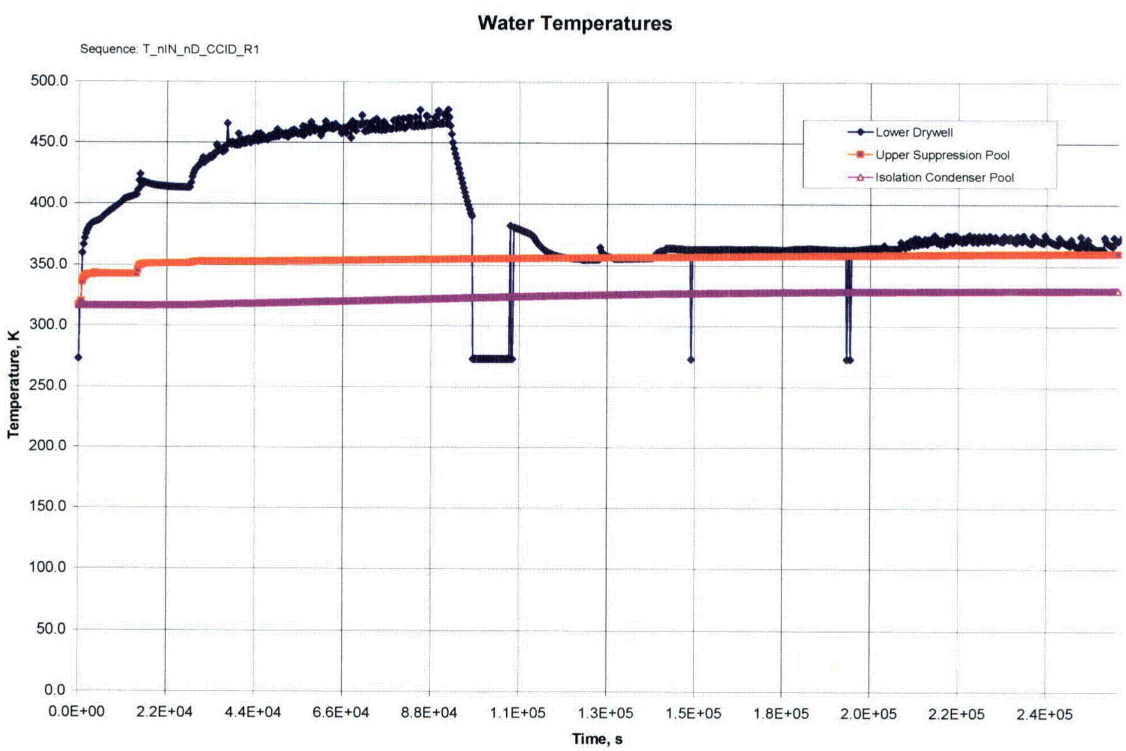


Figure 9A-5o - T_nIN_nD_CCID_R1 Water Temperatures

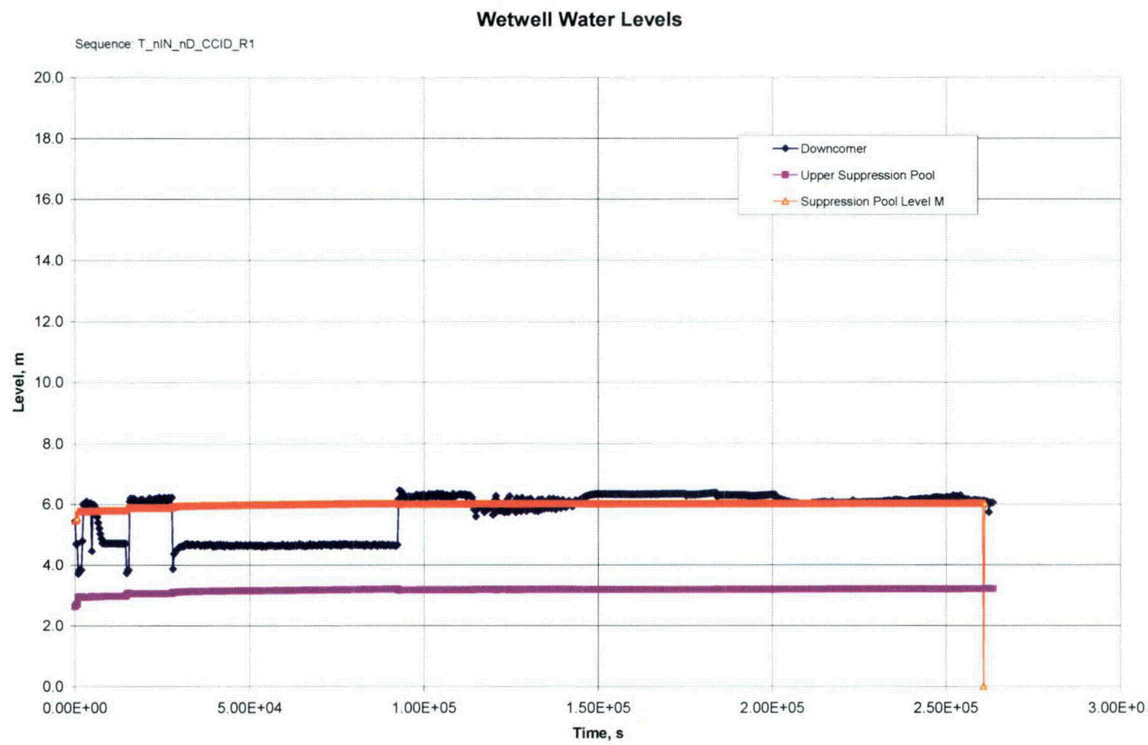


Figure 9A-5p - T_nIN_nD_CCID_R1 Wetwell Water Levels

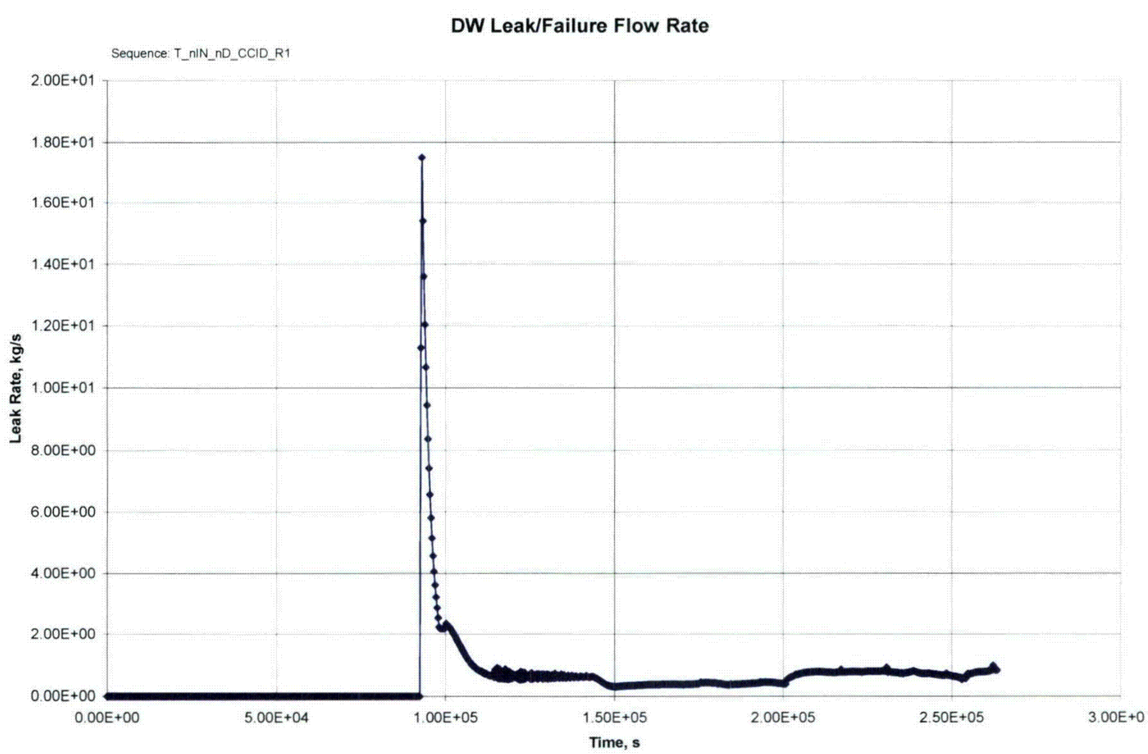


Figure 9A-5r - T_nIN_nD_CCID_R1 DW Leak/Failure Flow Rate

Figure 9A-5s - T_nIN_nD_CCID_R1 Wetwell Vent Flow Rate

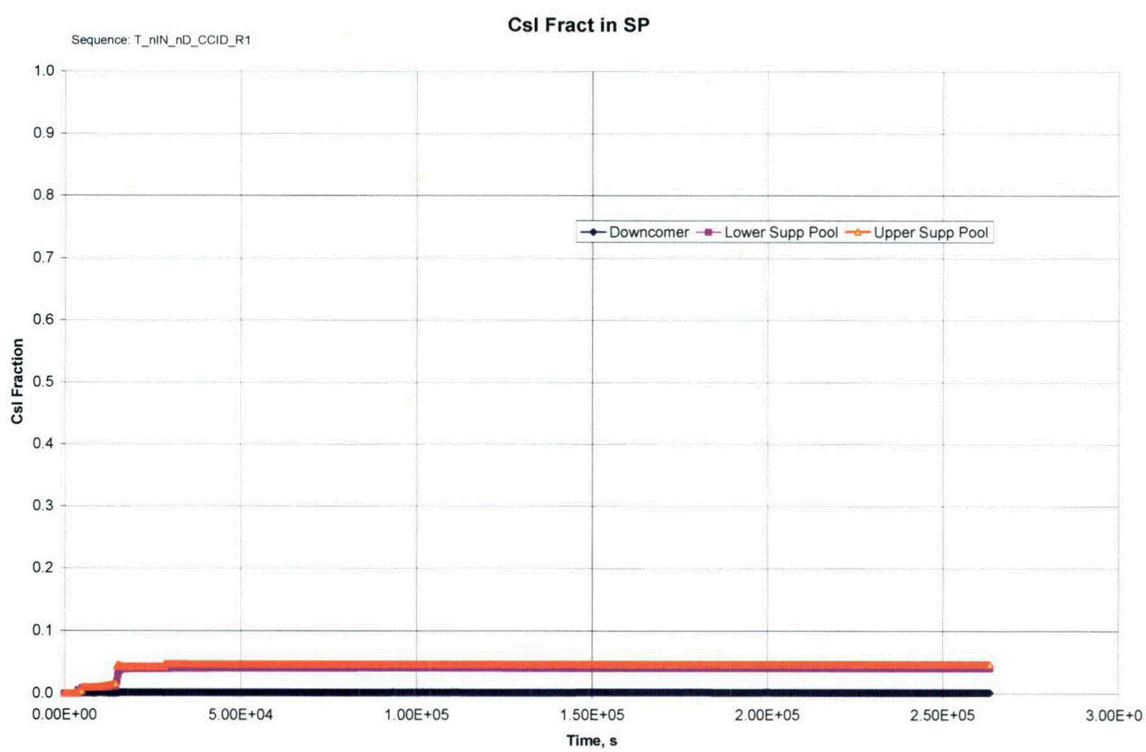


Figure 9A-5t - T_nIN_nD_CCID_R1 CsI Fract in SP

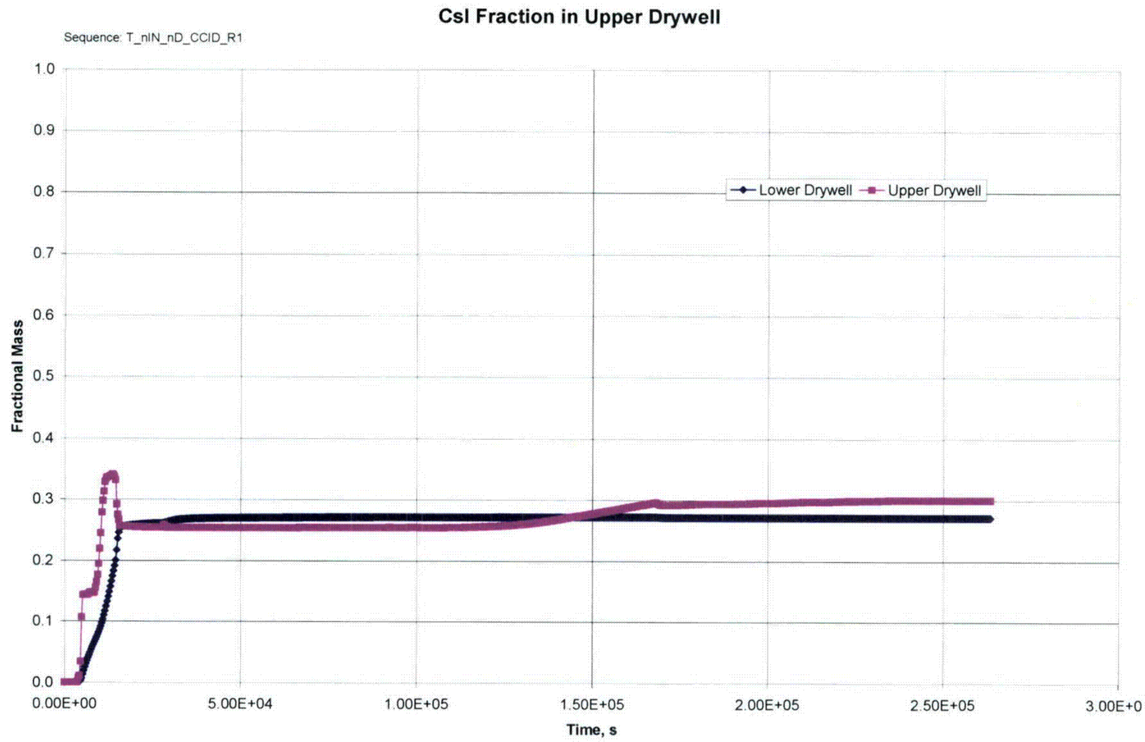


Figure 9A-5u - T_nIN_nD_CCID_R1 CsI Fraction in Upper Drywell

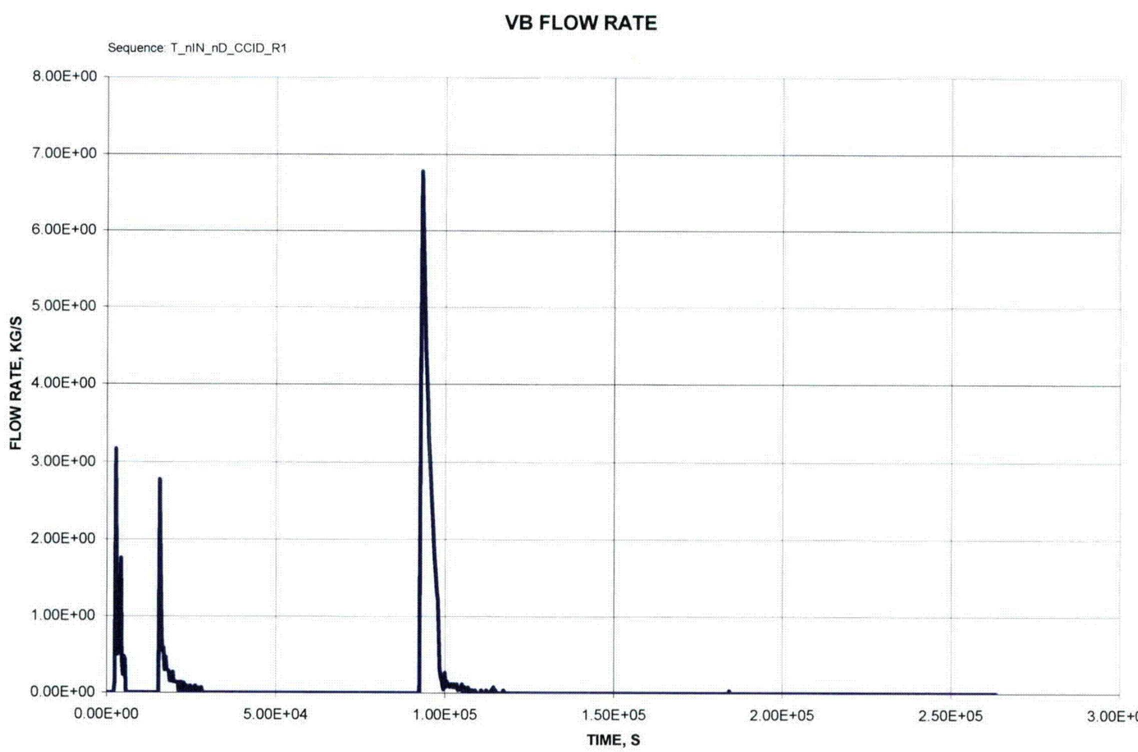


Figure 9A-5v - T_nIN_nD_CCID_R1 VB Flow Rate

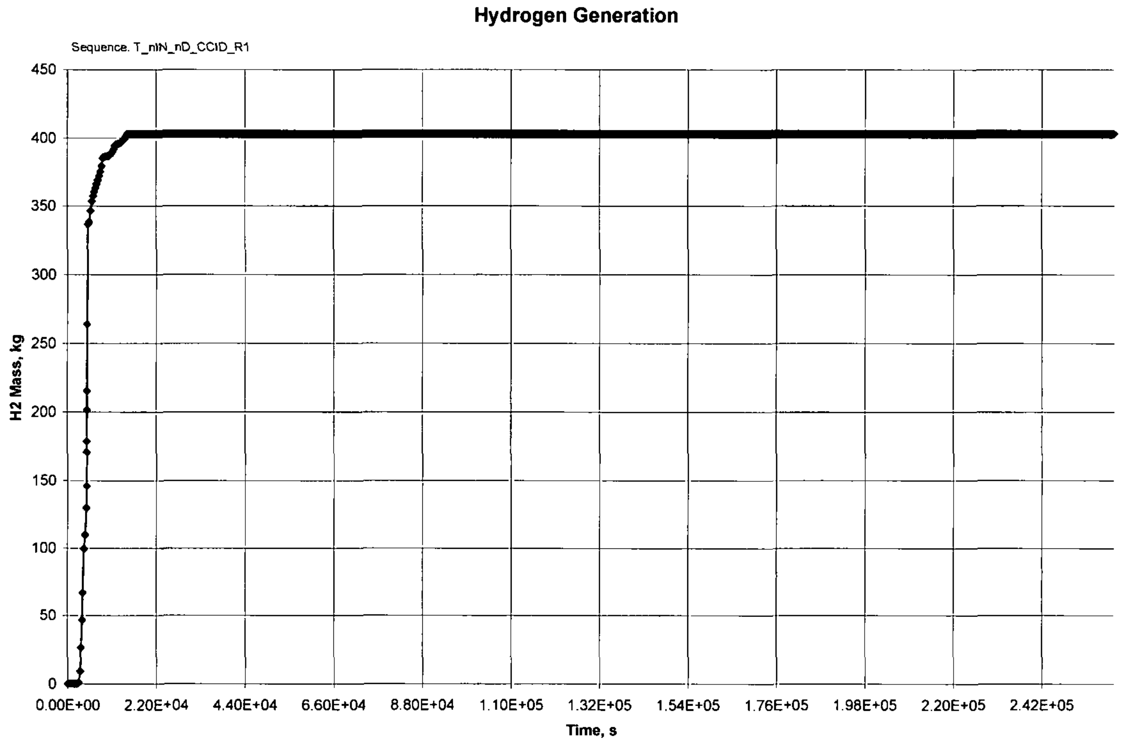


Figure 9A-5w - T_nIN_nD_CCID_R1 Hydrogen Generation

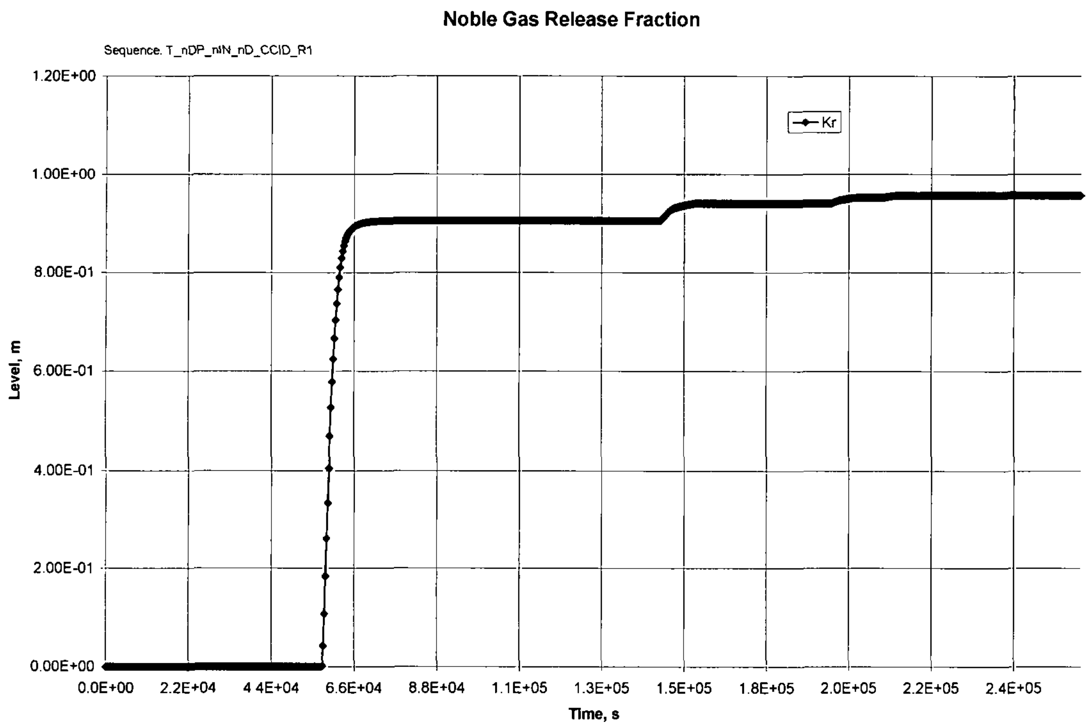


Figure 9A-5x - T_nIN_nD_CCID_R1 Noble Gas Release Fraction

9A.6 Core Concrete Interaction (Dry) – High Pressure T_nDP_nIN_nD_CCID_R1

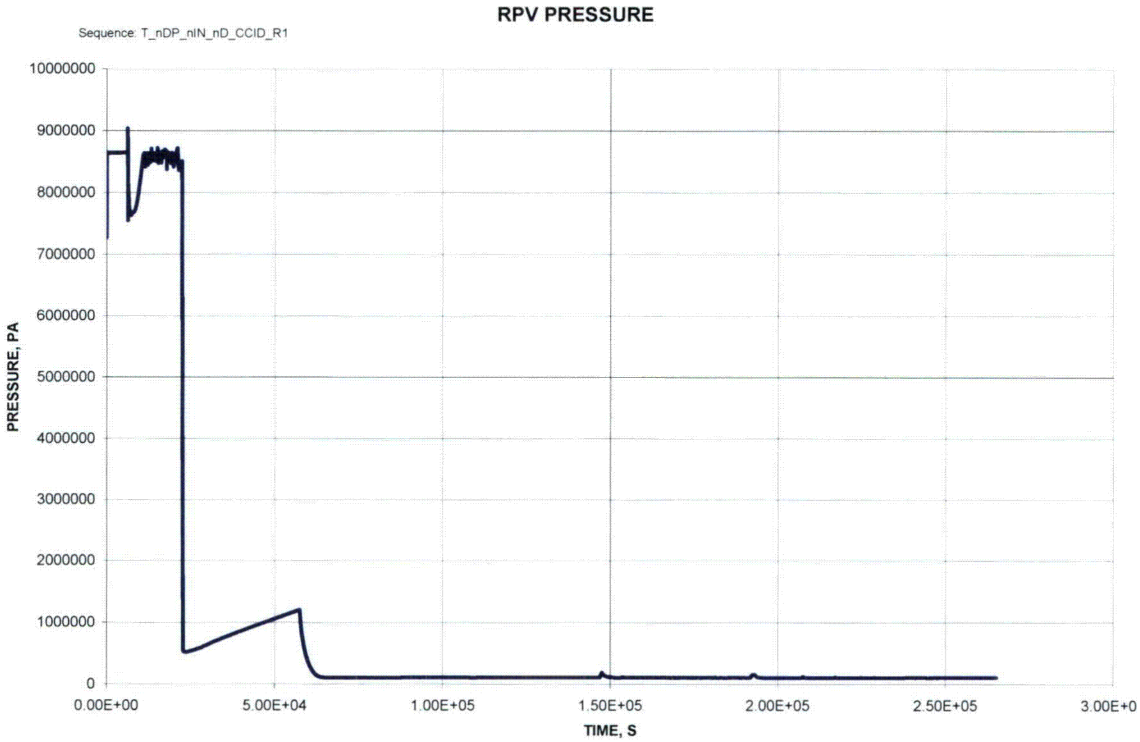


Figure 9A-6a - T_nDP_nIN_nD_CCID_R1 RPV Pressure

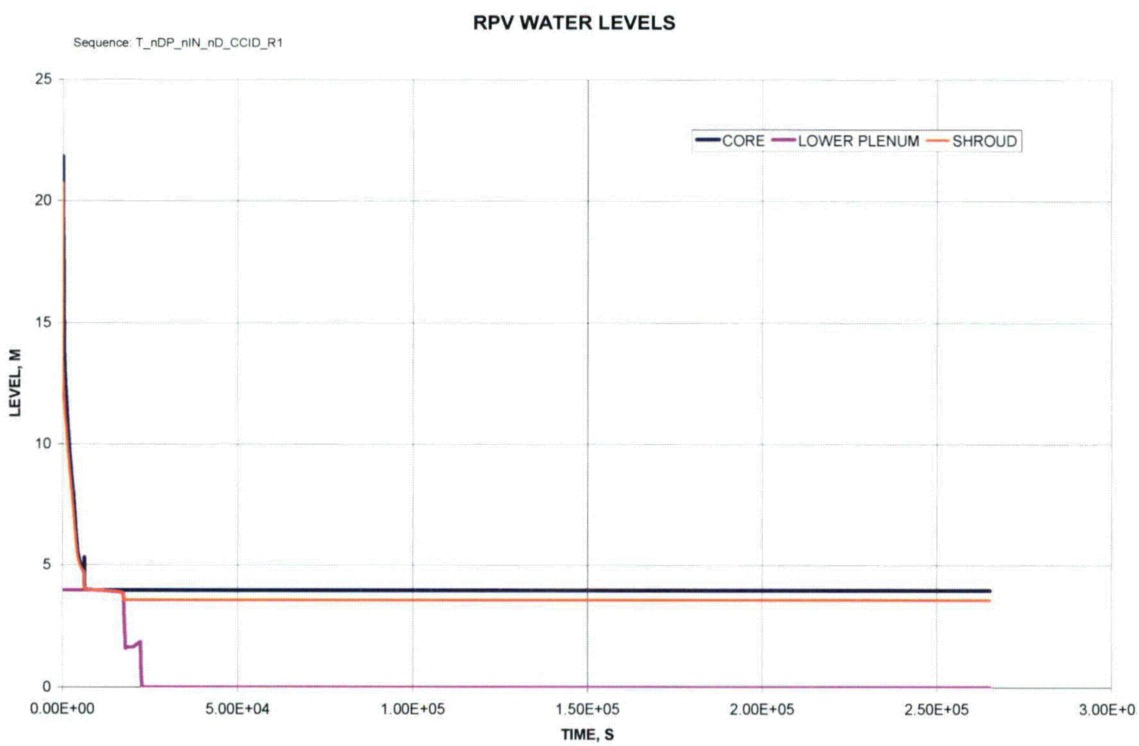


Figure 9A-6b - T_nDP_nIN_nD_CCID_R1 RPV Water Levels

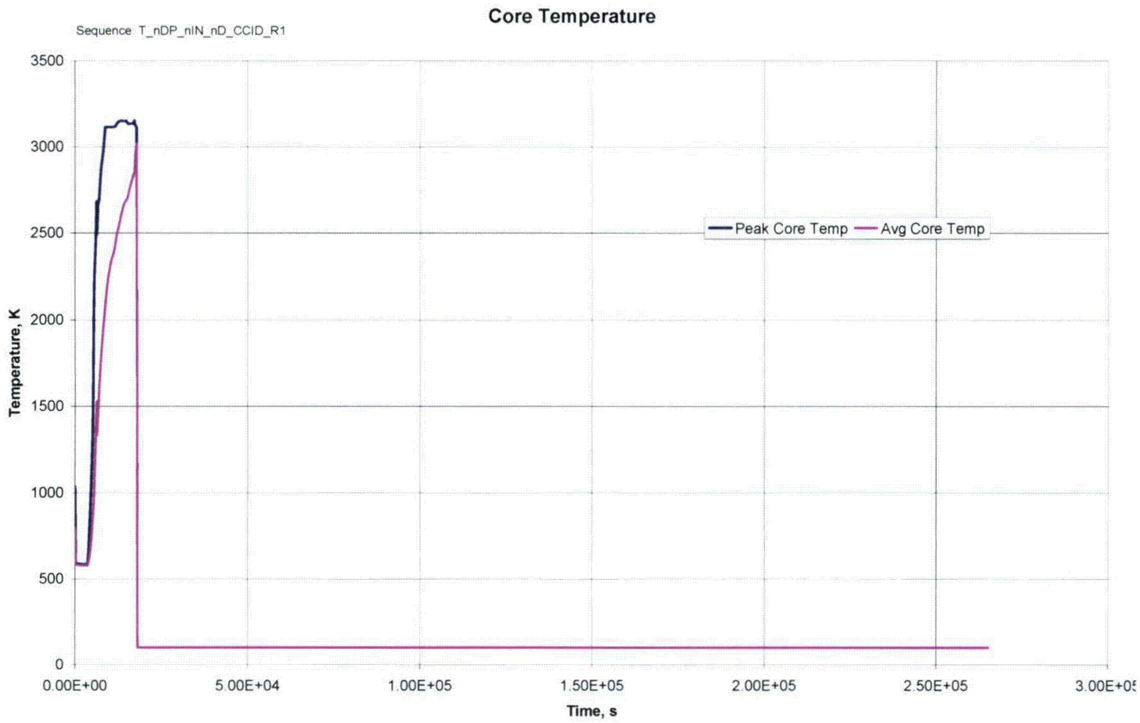


Figure 9A-6c - T_nDP_nIN_nD_CCID_R1 Core Temperature

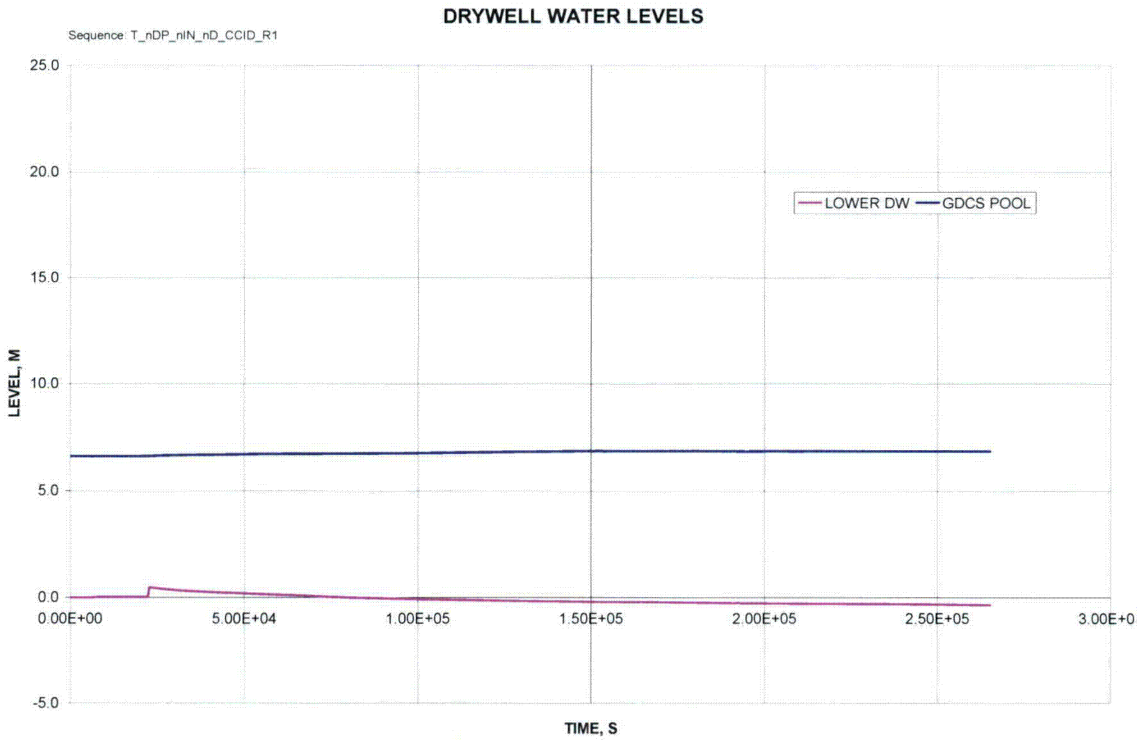


Figure 9A-6d - T_nDP_nIN_nD_CCID_R1 Drywell Water Levels

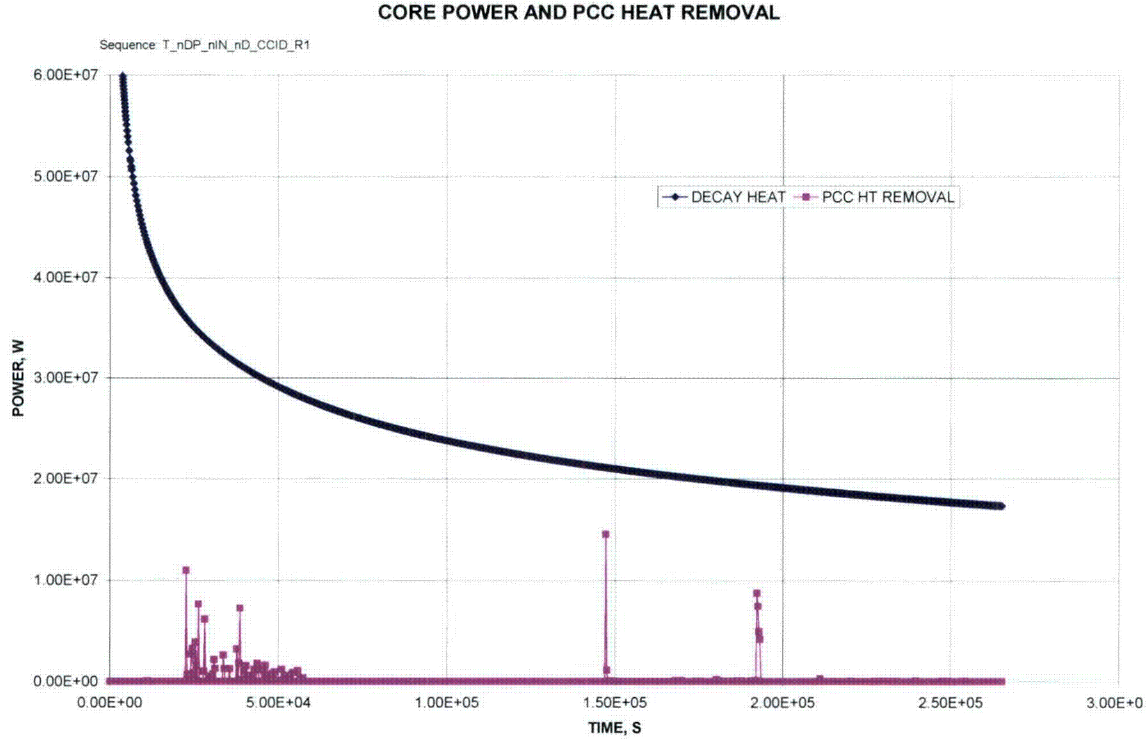


Figure 9A-6e - T_nDP_nIN_nD_CCID_R1 Core Power and PCC Heat Removal

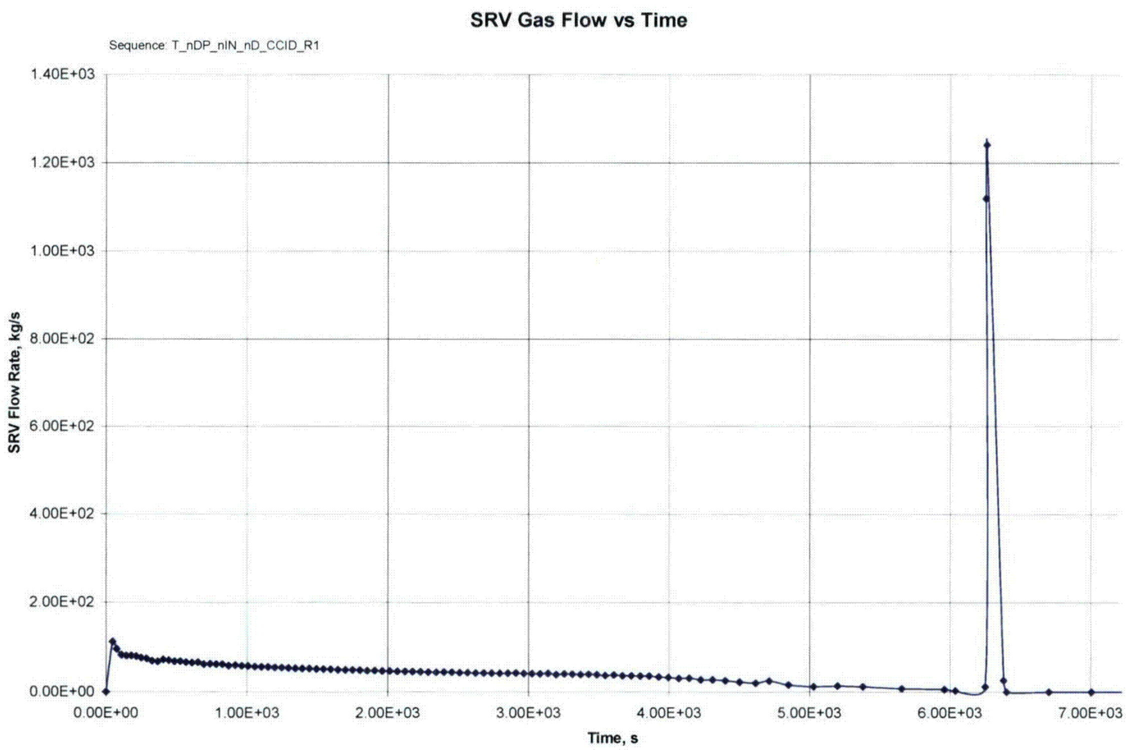


Figure 9A-6f - T_nDP_nIN_nD_CCID_R1 SRB Gas Flow vs Time

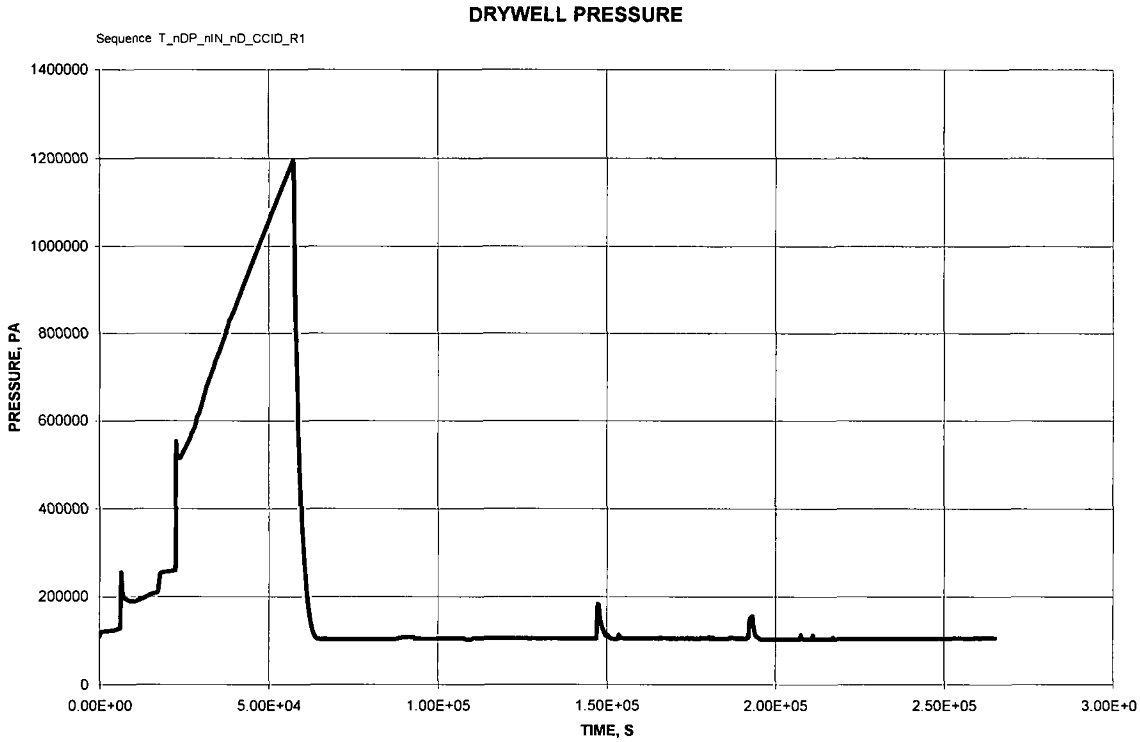


Figure 9A-6g - T_nDP_nIN_nD_CCID_R1 Drywell Pressure

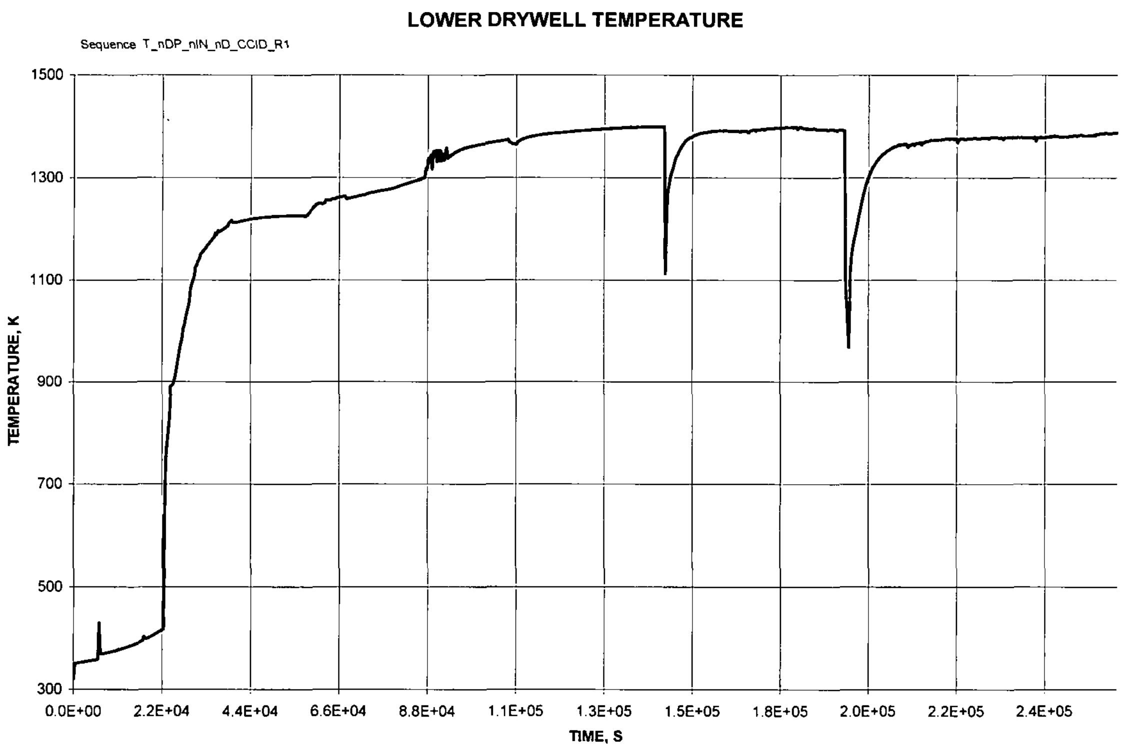


Figure 9A-6h - T_nDP_nIN_nD_CCID_R1 Lower Drywell Temperature

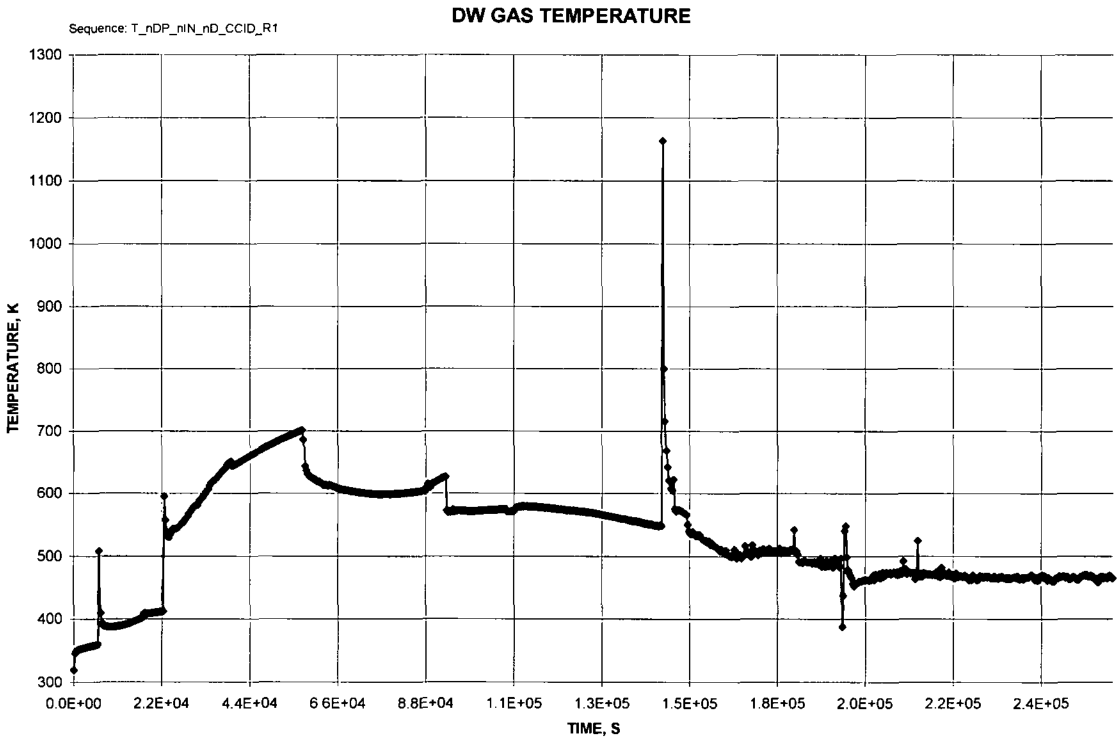


Figure 9A-6i - T_nDP_nIN_nD_CCID_R1 DW Gas Temperature

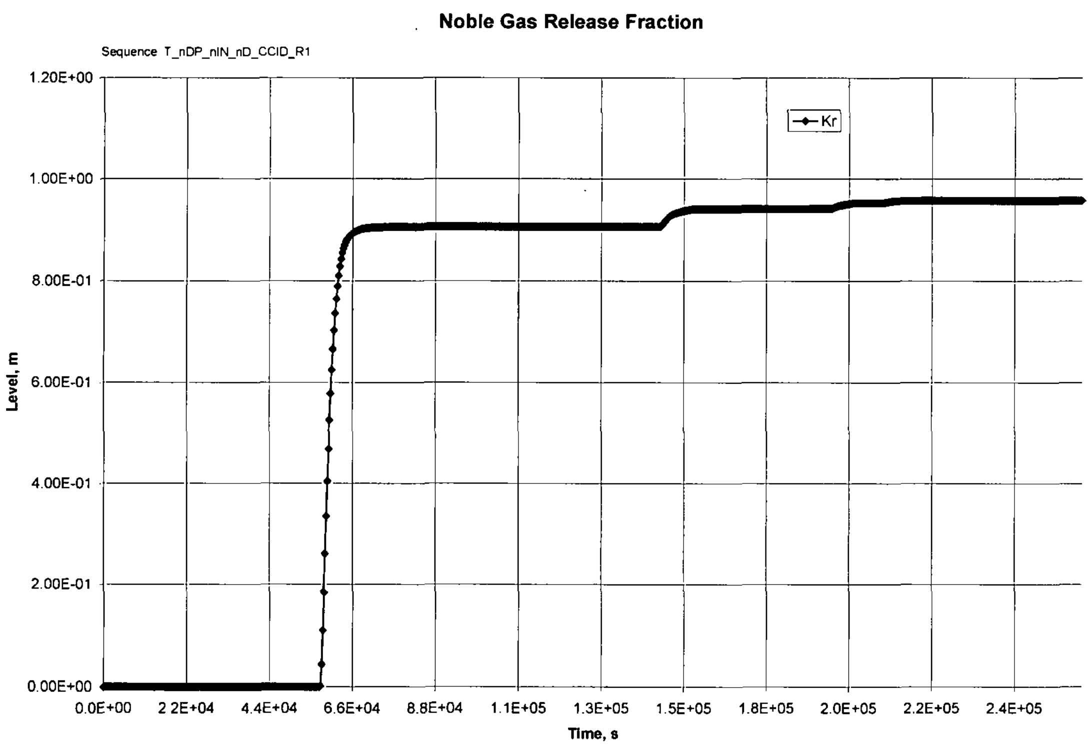


Figure 9A-6j- T_nDP_nIN_nD_CCID_R1 Noble Gas Release Fraction

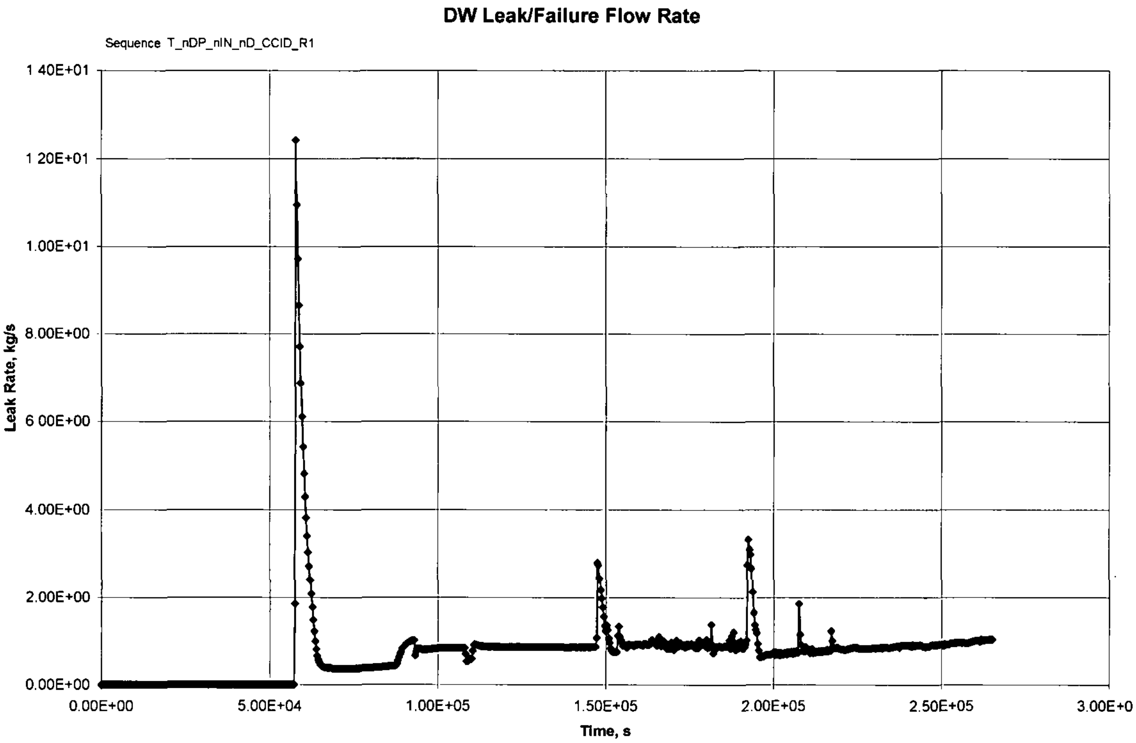


Figure 9A-6q - T_nDP_nIN_nD_CCID_R1 DW Leak/Failure Flow Rate

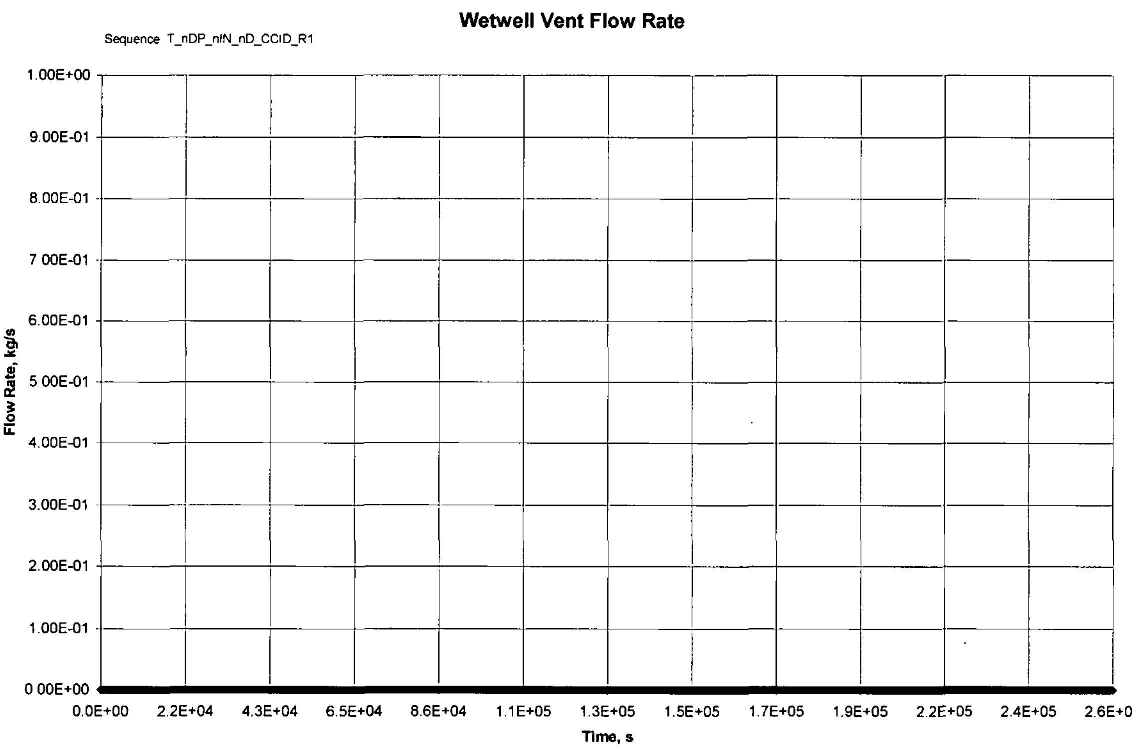


Figure 9A-6r - T_nDP_nIN_nD_CCID_R1 Wetwell Vent Flow Rate

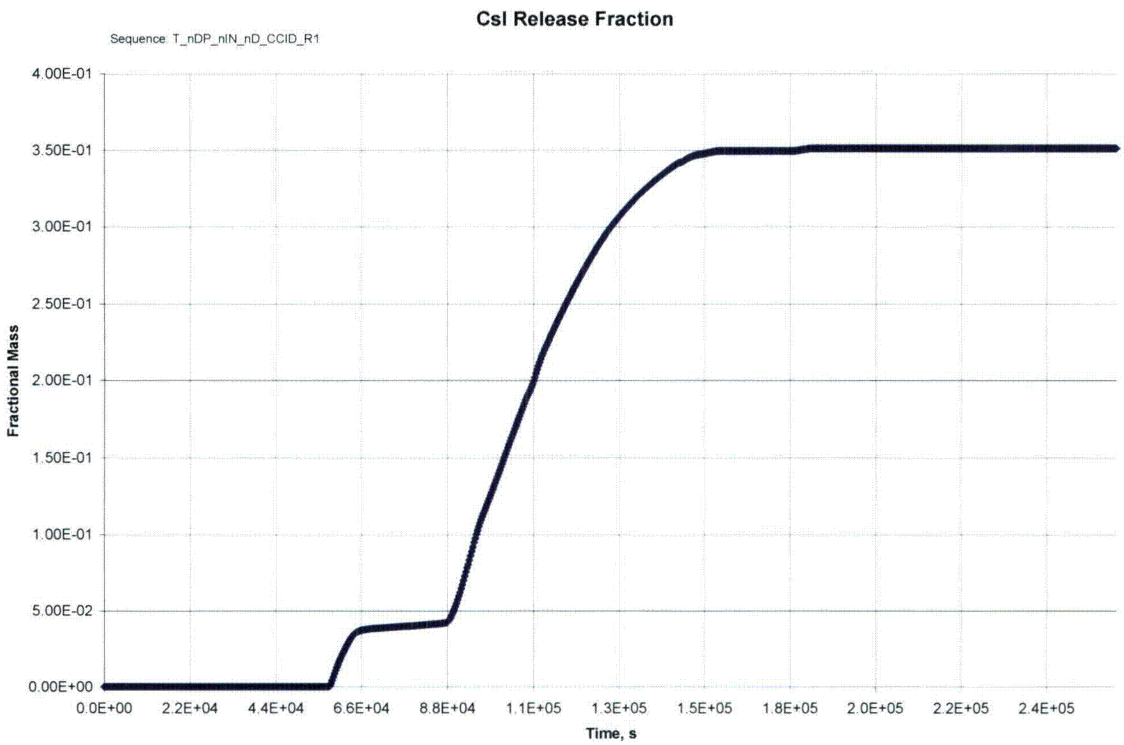


Figure 9A-6k - T_nDP_nIN_nD_CCID_R1 CsI Release Fraction

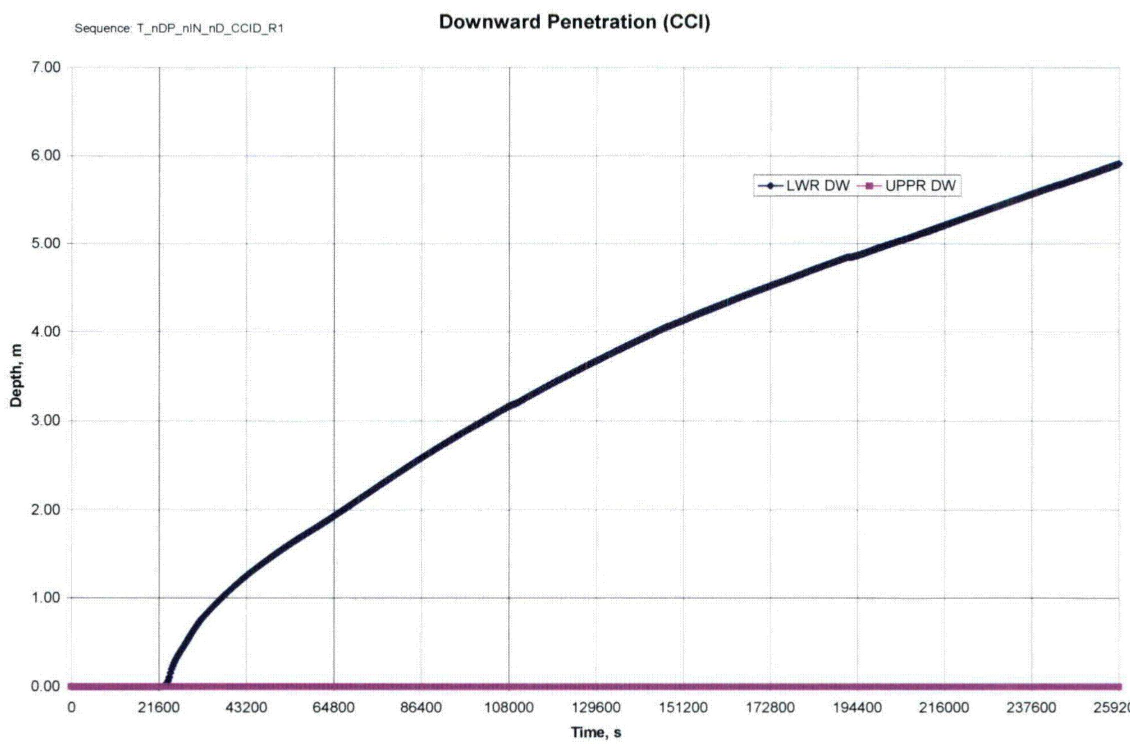


Figure 9A-6l - T_nDP_nIN_nD_CCID_R1 Downward Penetration (CCI)

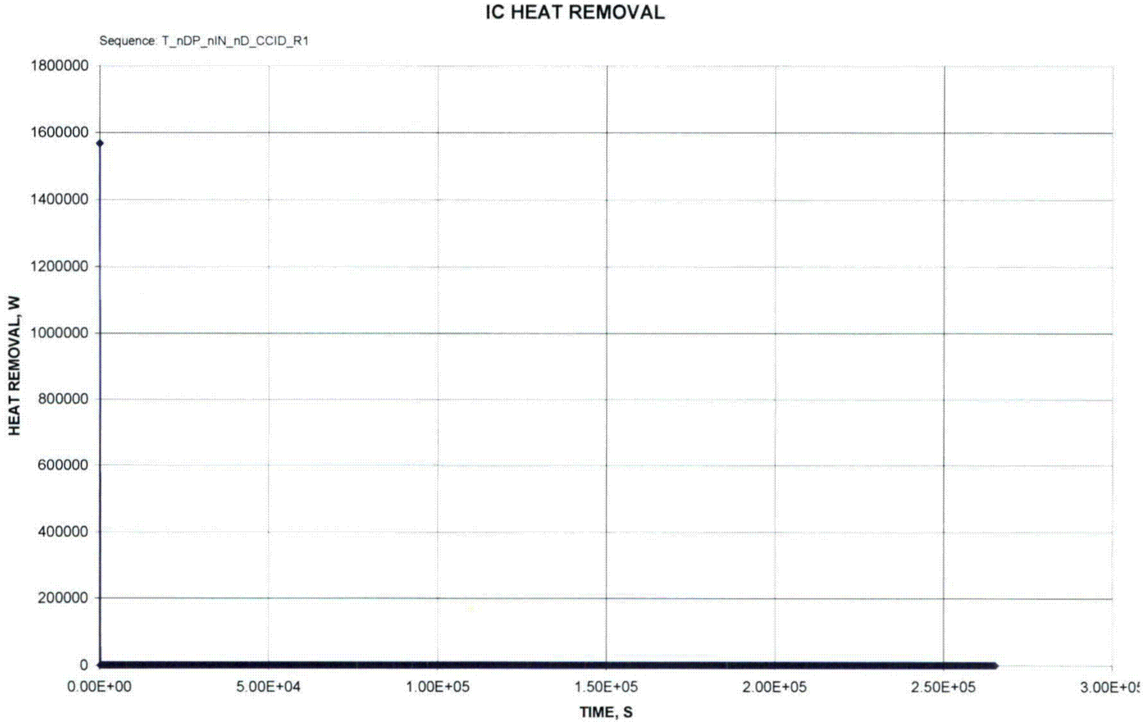


Figure 9A-6m - T_nDP_nIN_nD_CCID_R1 ICS Heat Removal

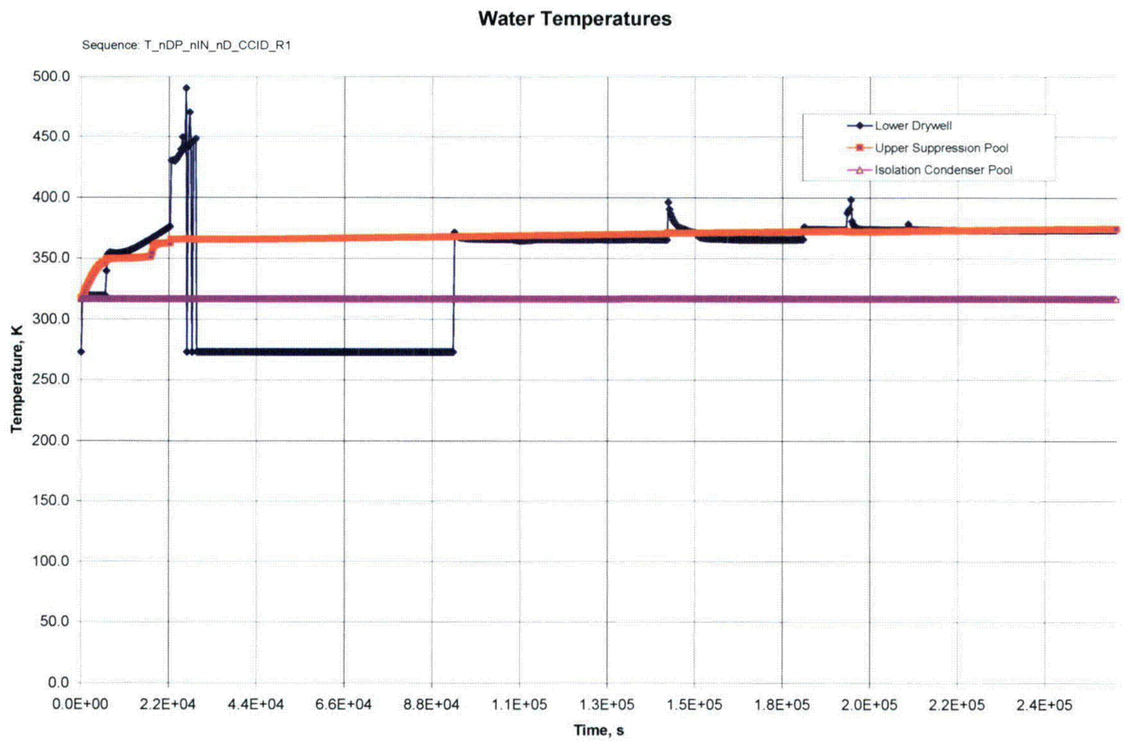


Figure 9A-6n - T_nDP_nIN_nD_CCID_R1 Water Temperatures

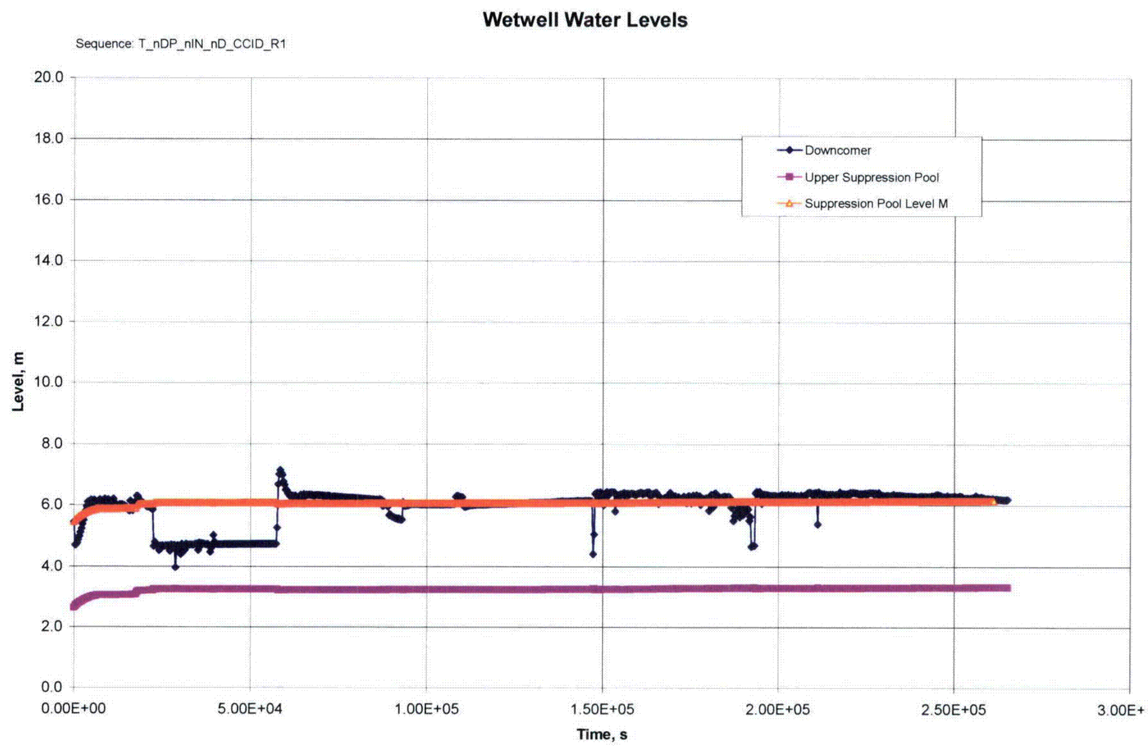


Figure 9A-6o - T_nDP_nIN_nD_CCID_R1 6 Wetwell Water Levels

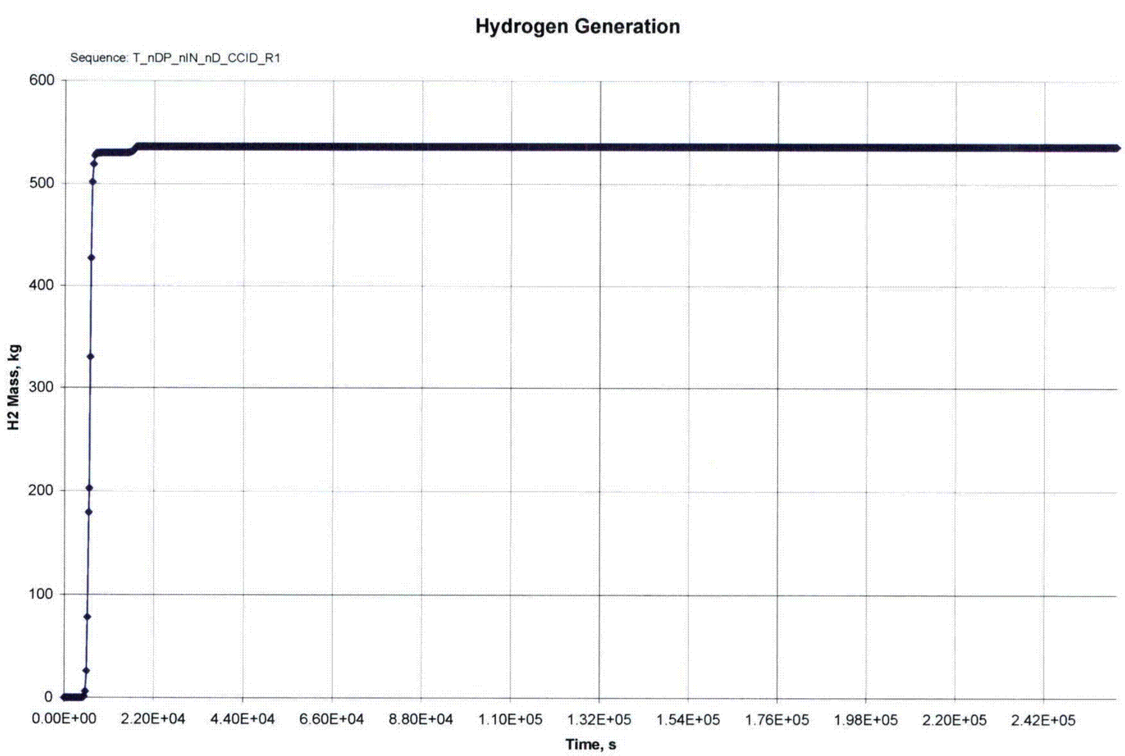


Figure 9A-6p - T_nDP_nIN_nD_CCID_R1 Hydrogen Generation

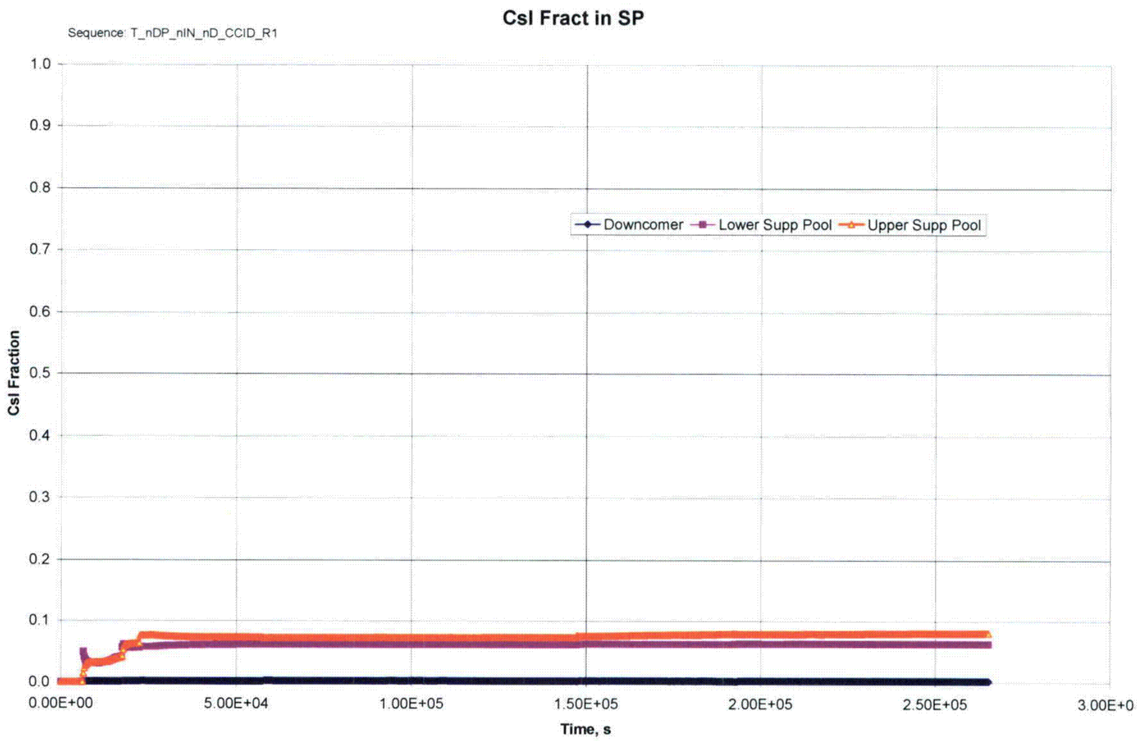


Figure 9A-6s - T_nDP_nIN_nD_CCID_R1 CsI Fract in SP

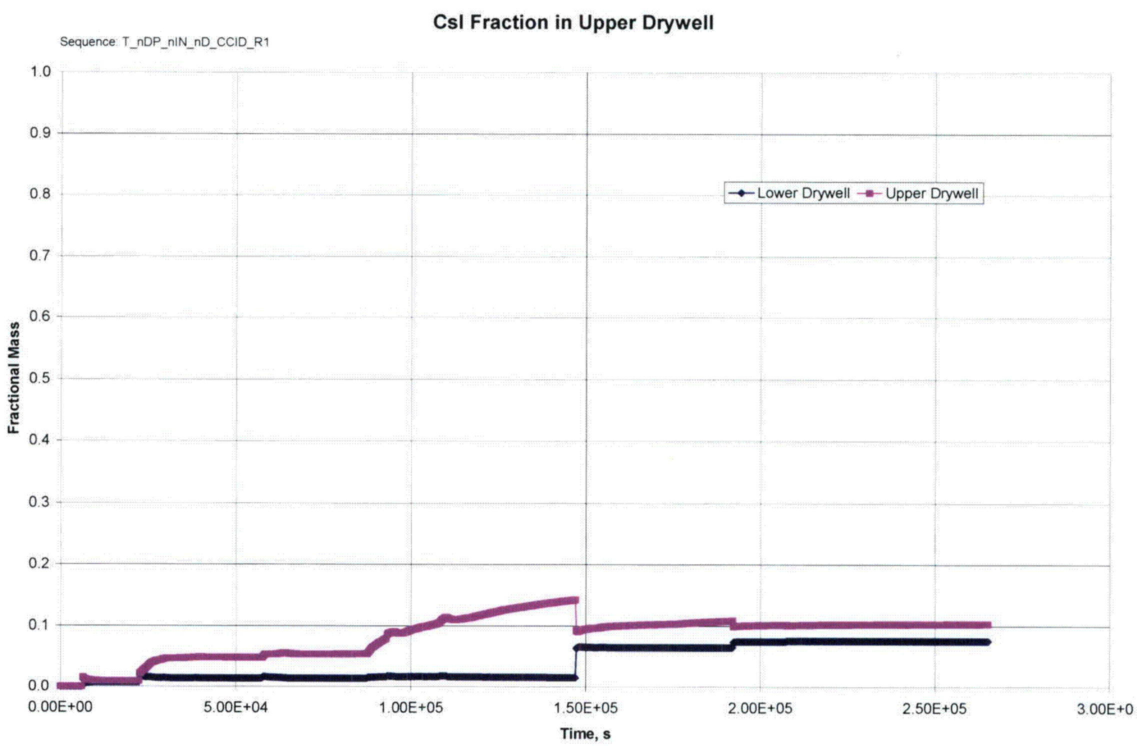


Figure 9A-6t - T_nDP_nIN_nD_CCID_R1 CsI Fraction in Upper Drywell

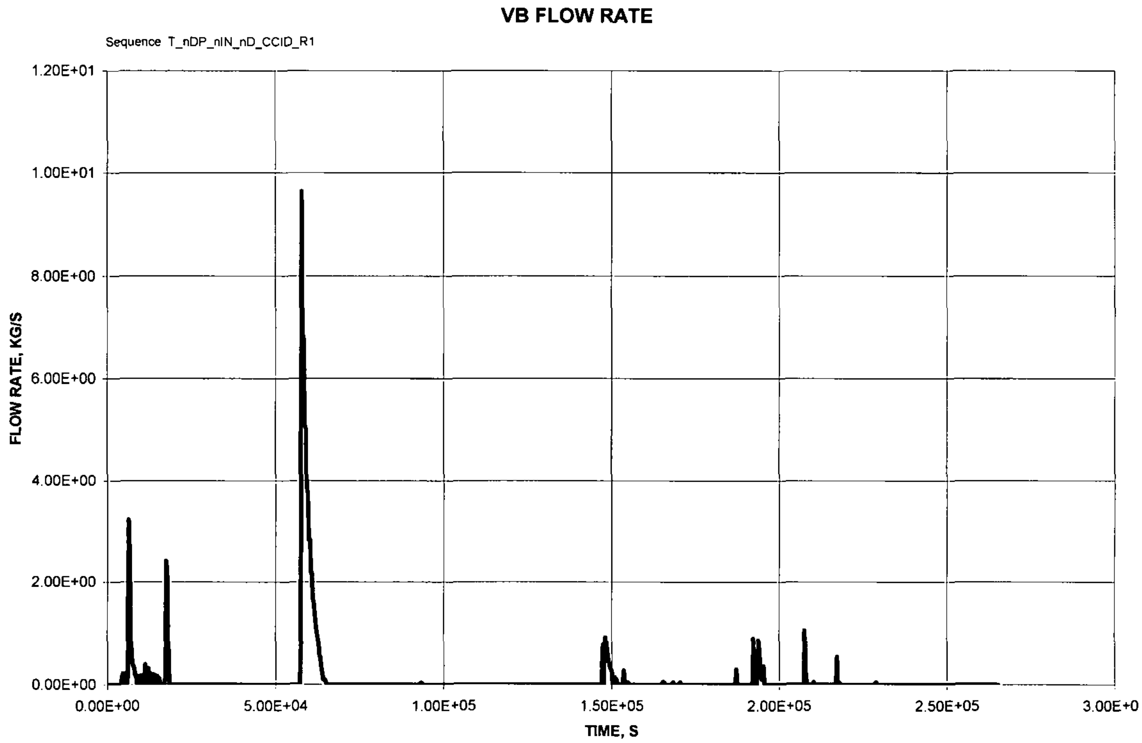


Figure 9A-6u - T_nDP_nIN_nD_CCID_R1 VB Flow Rate

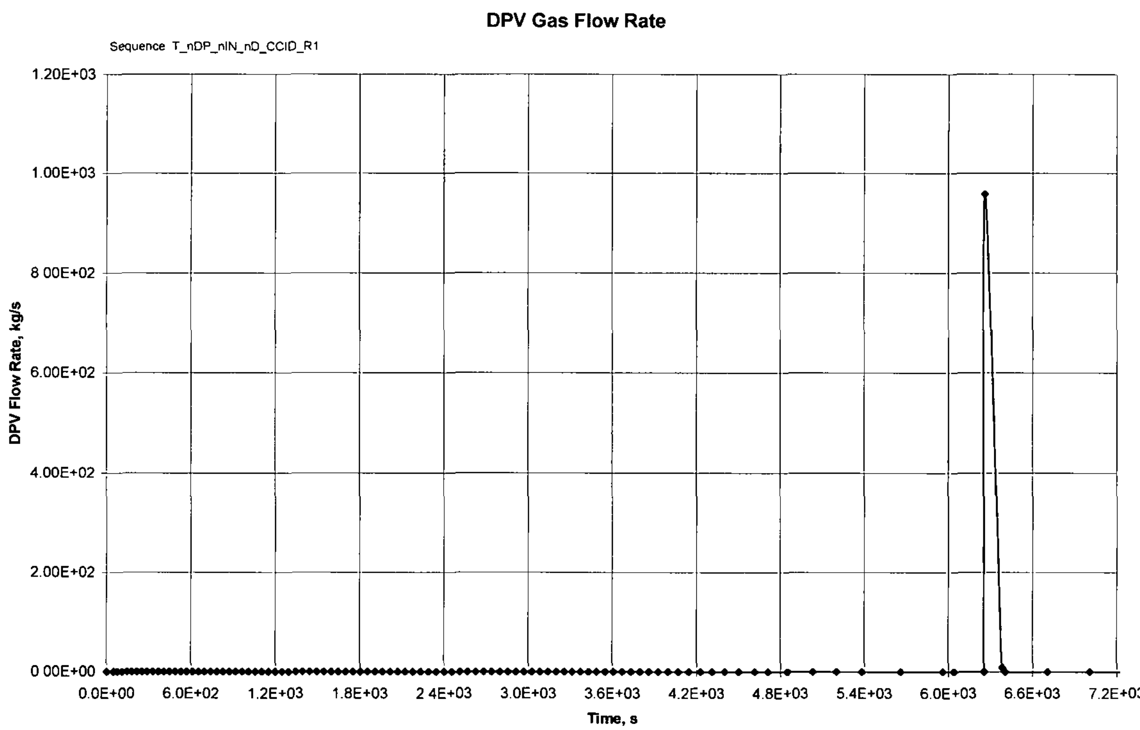


Figure 9A-6v – T_nDP_nIN_nD_CCID_R1 DPV Flow Rate

9A.7 Core Concrete Interaction (Wet) – Low Pressure T_nIN_CCIW_R1

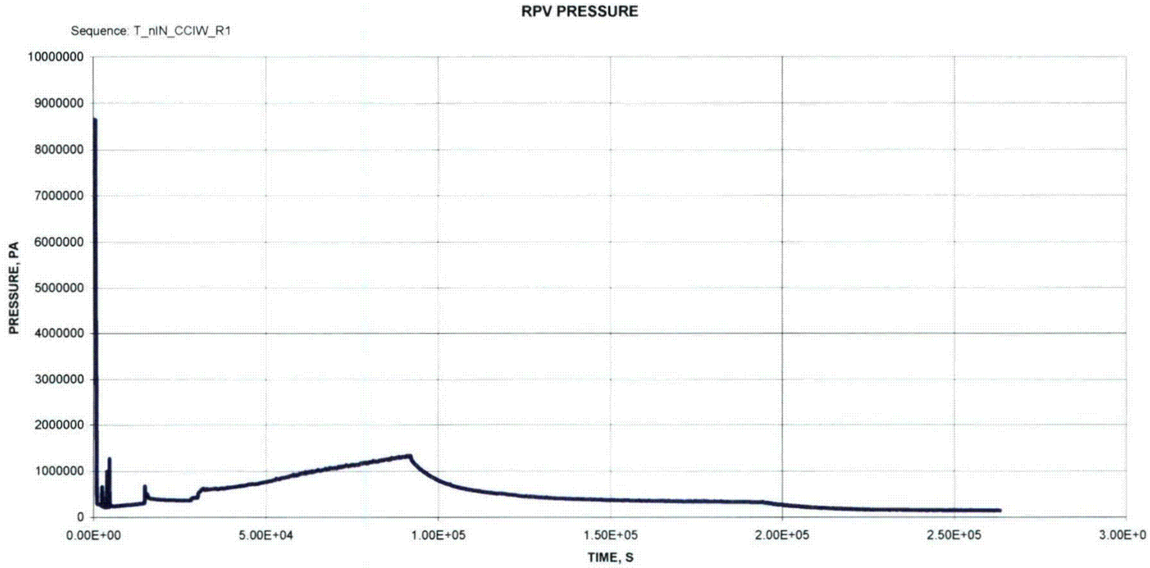


Figure 9A-7a - T_nIN_CCIW_R1 RPV Pressure

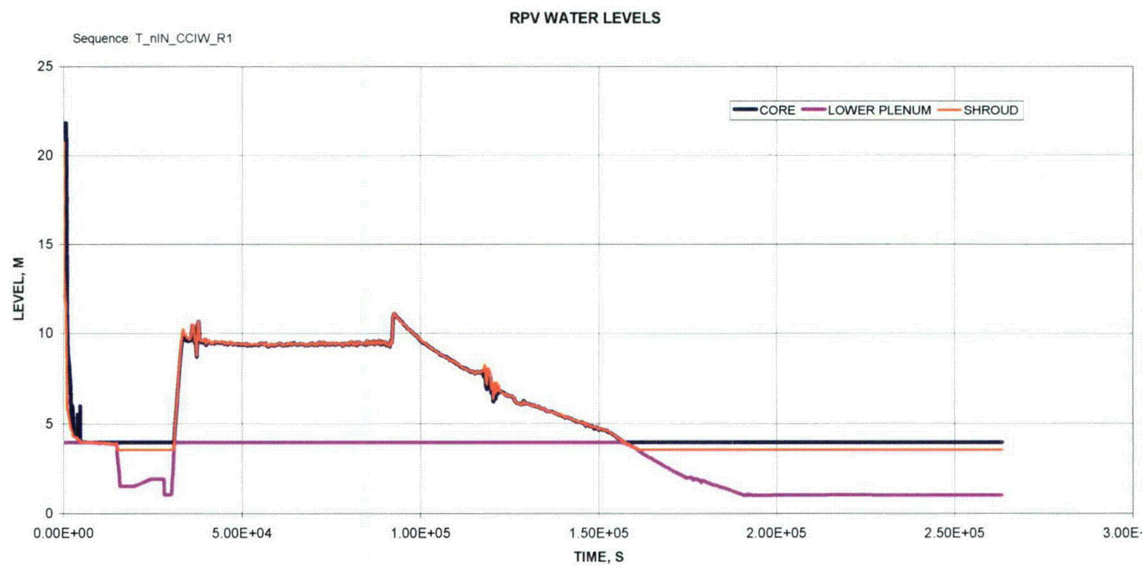


Figure 9A-7b - T_nIN_CCIW_R1 RPV Water Level

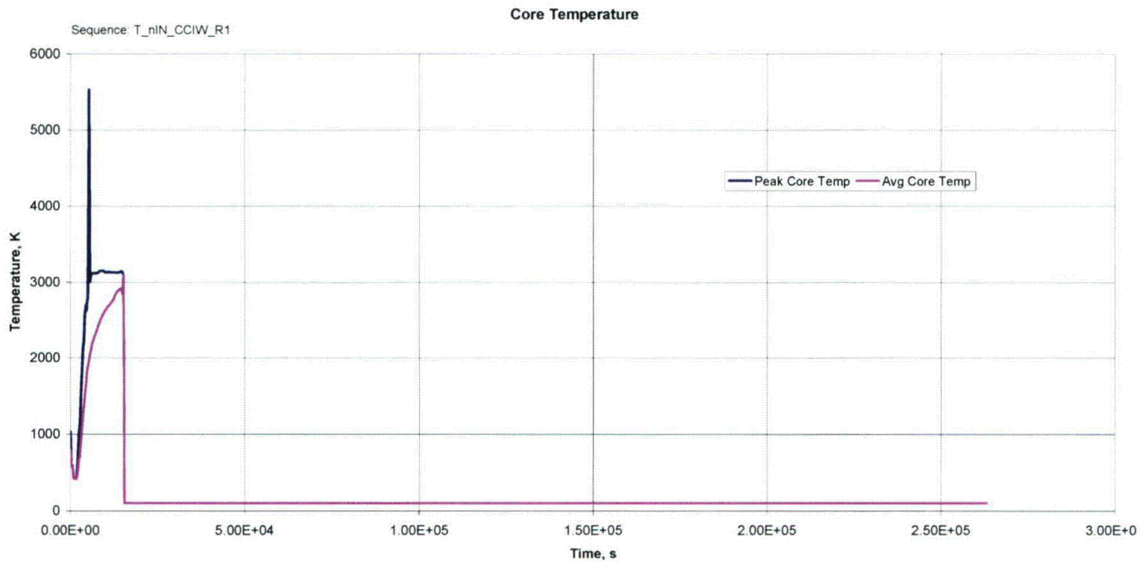


Figure 9A-7c - T_nIN_CCIW_R1 Core Temperature

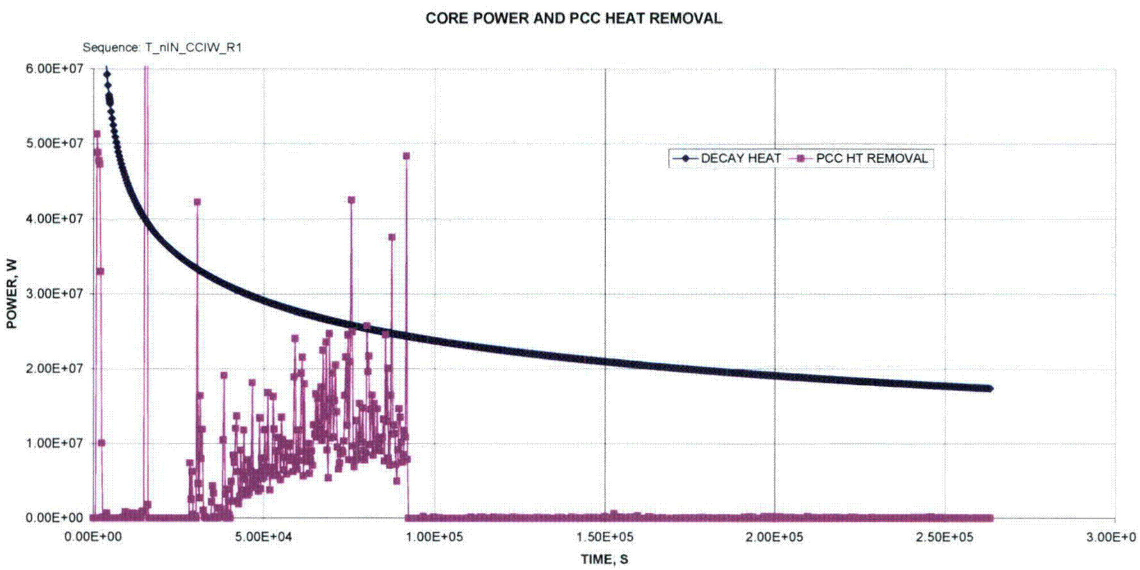


Figure 9A-7d - T_nIN_CCIW_R1 Core Power and PCC Heat Removal

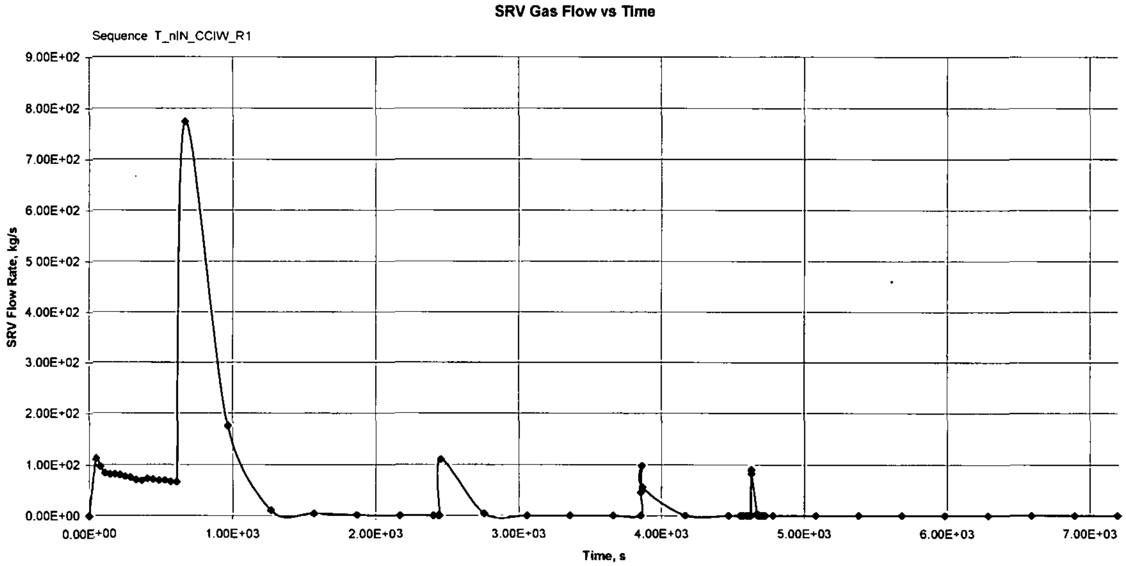


Figure 9A-7e - T_nIN_CCIW_R1 SRV Gas Flow vs Time

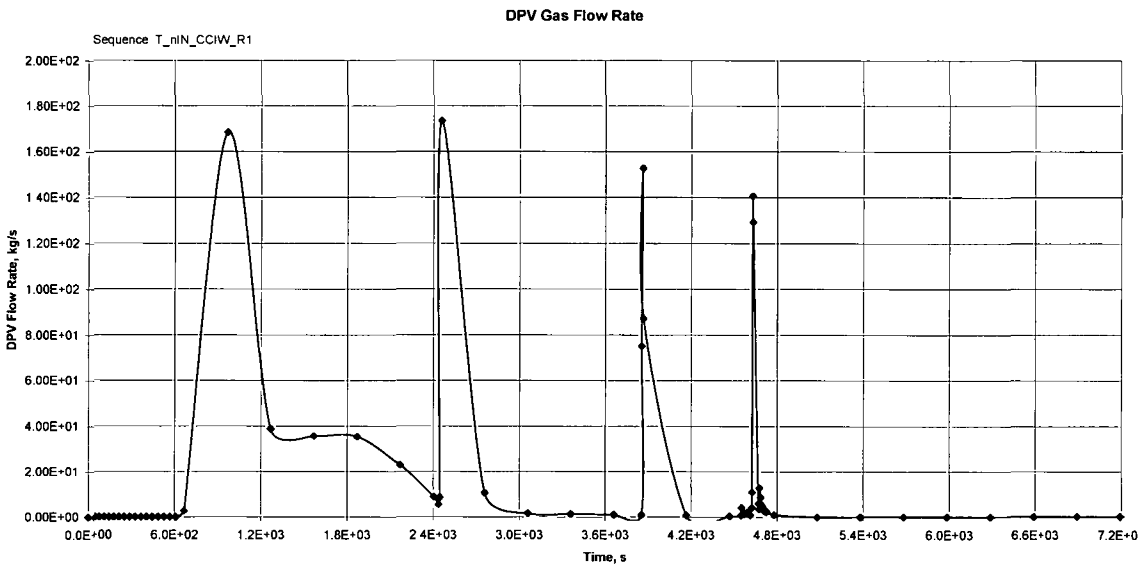


Figure 9A-7f – T_nIN_CCIW_R1 DPV Gas Flow Rate

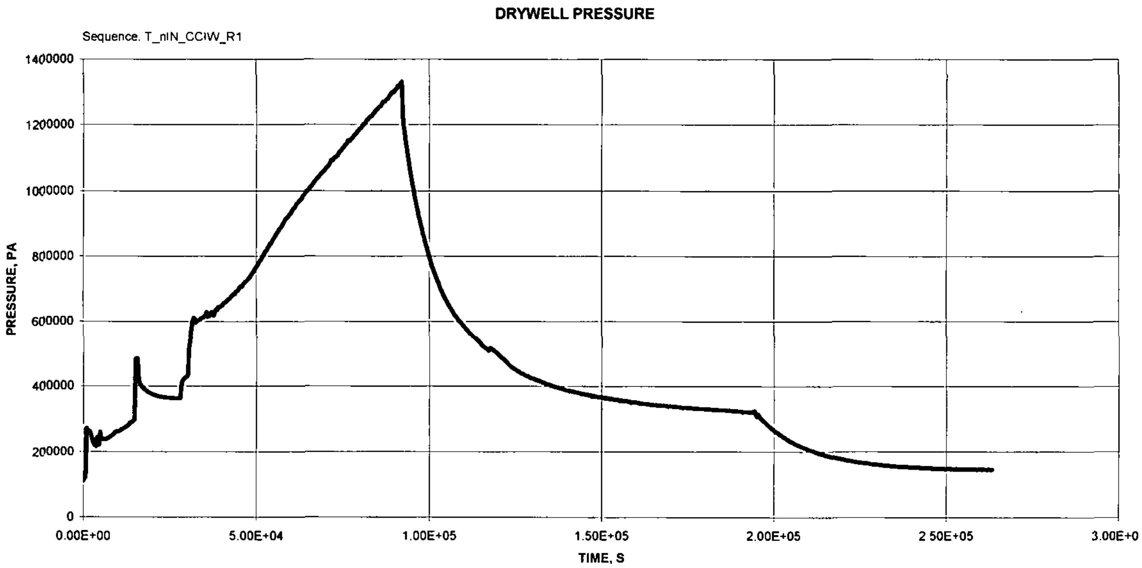


Figure 9A-7g - T_nIN_CCIW_R1 Drywell Pressure

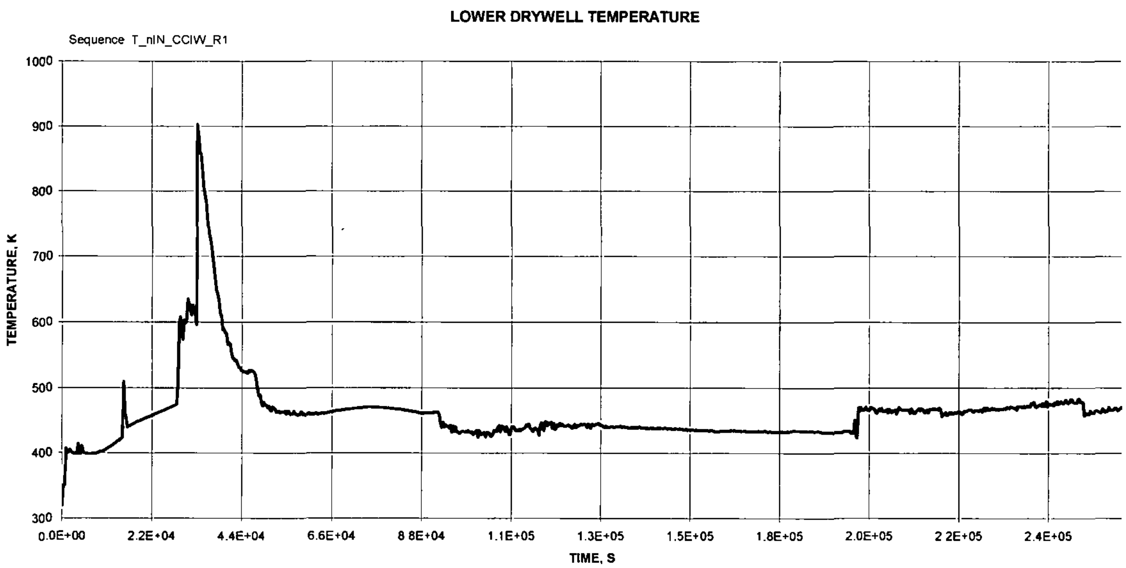


Figure 9A-7h - T_nIN_CCIW_R1 Lower Drywell Temperature

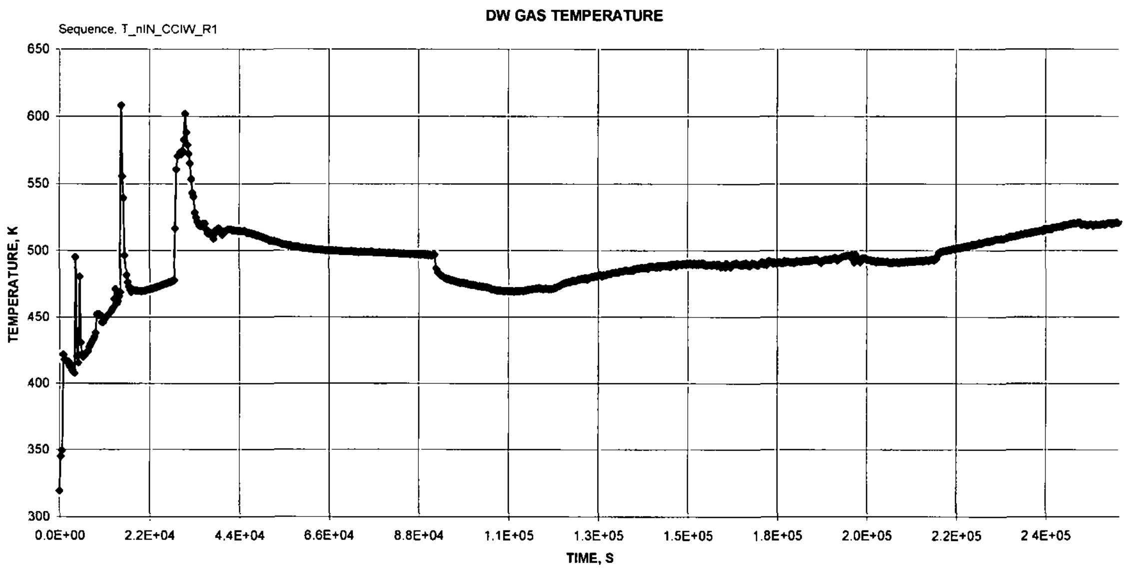


Figure 9A-7i - T_nIN_CCIW_R1 DW Gas Temperature

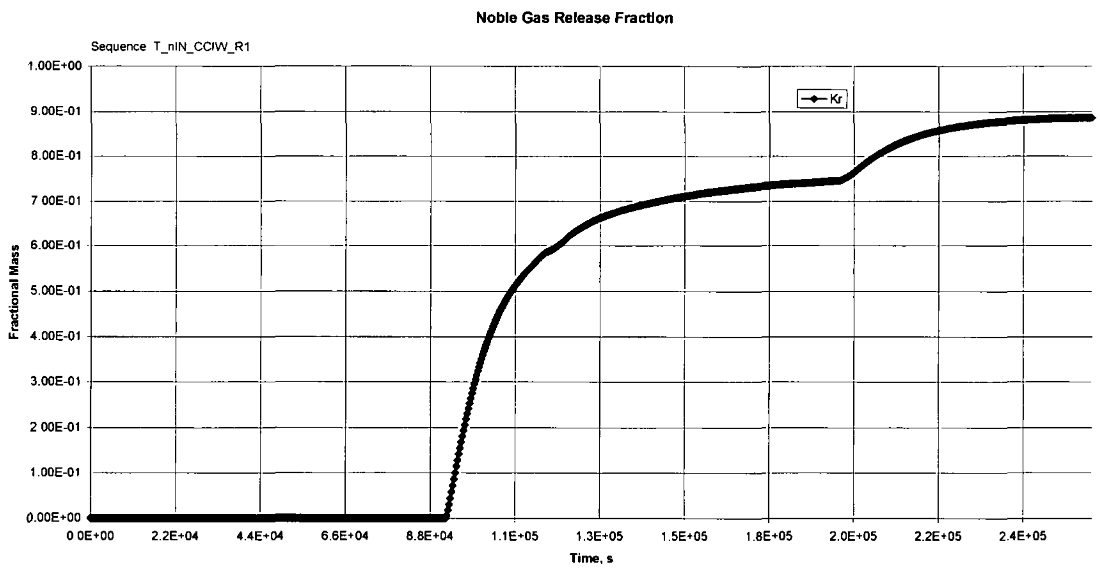


Figure 9A-7j - T_nIN_CCIW_R1 Noble Gas Release Fraction

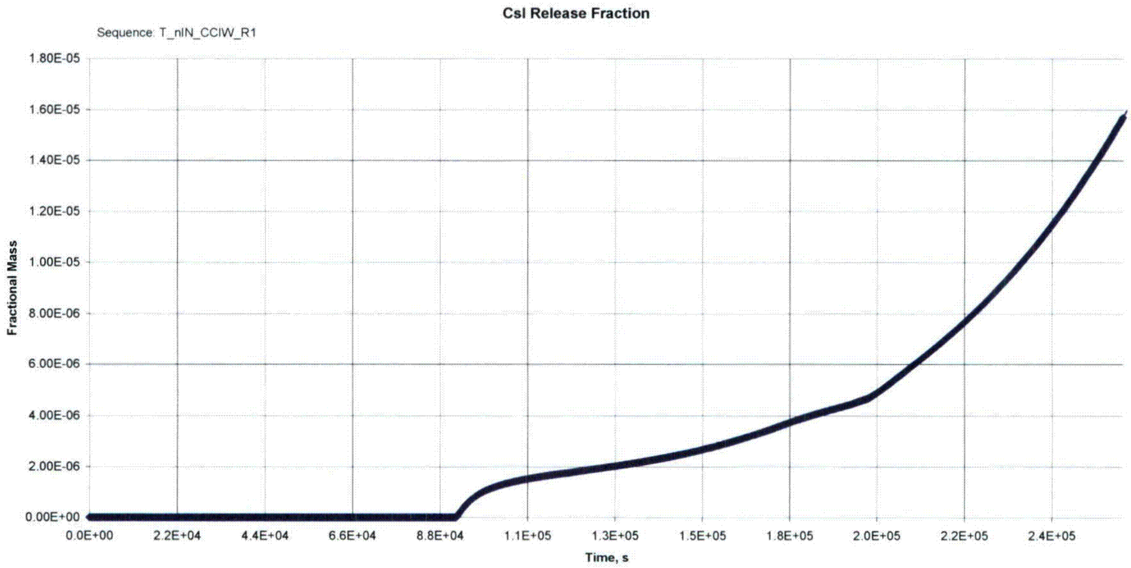


Figure 9A-7k - T_nIN_CCIW_R1 CsI Release Fraction

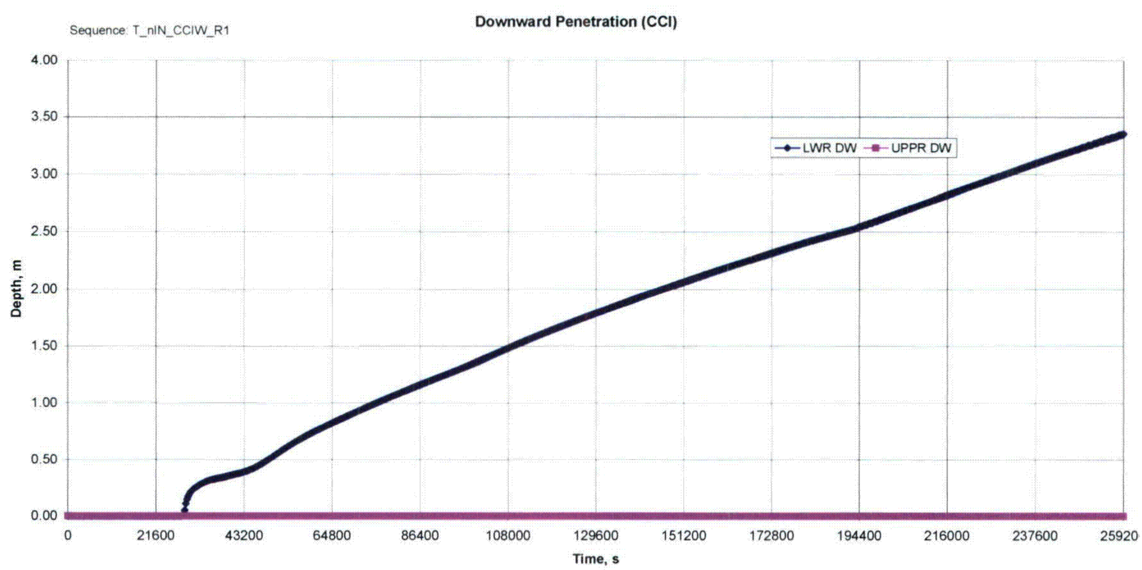


Figure 9A-7l - T_nIN_CCIW_R1 Downward Penetration (CCI)

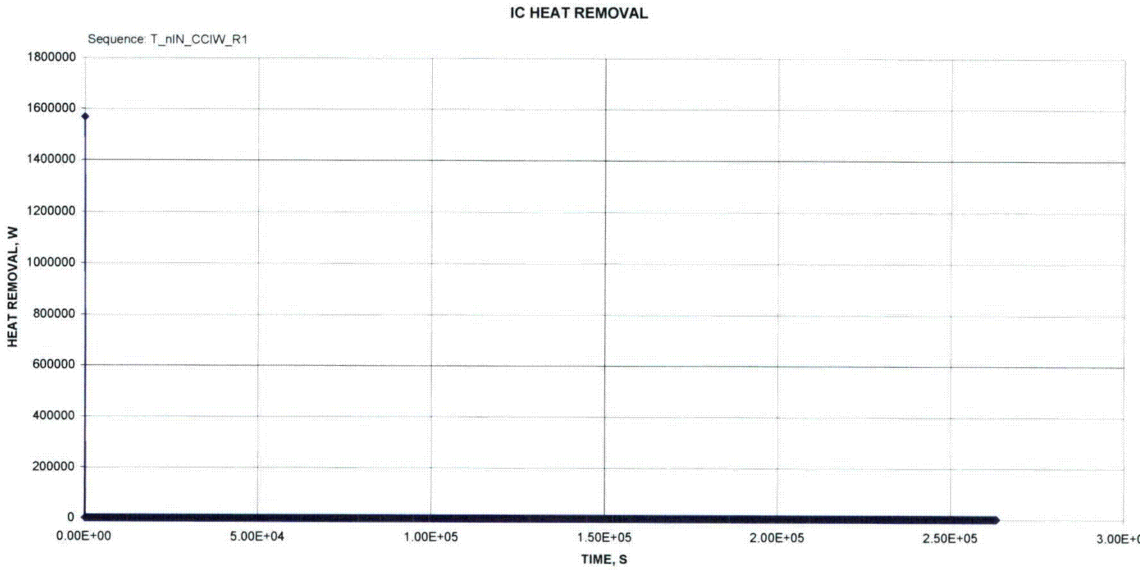


Figure 9A-7m - T_nIN_CCIW_R1 ICS Heat Removal

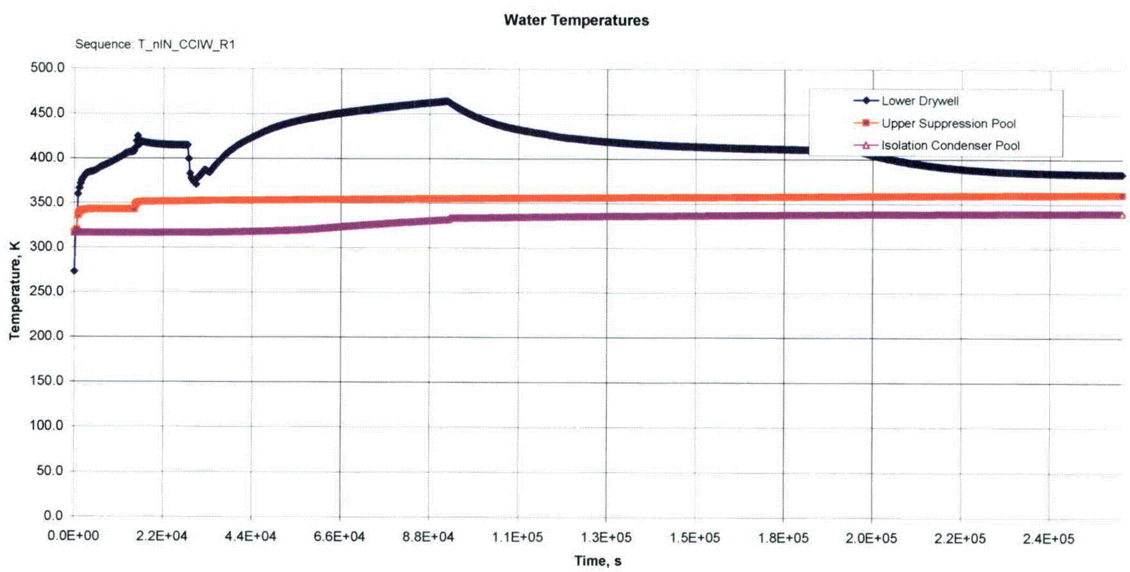


Figure 9A-7n - T_nIN_CCIW_R1 Water Temperature

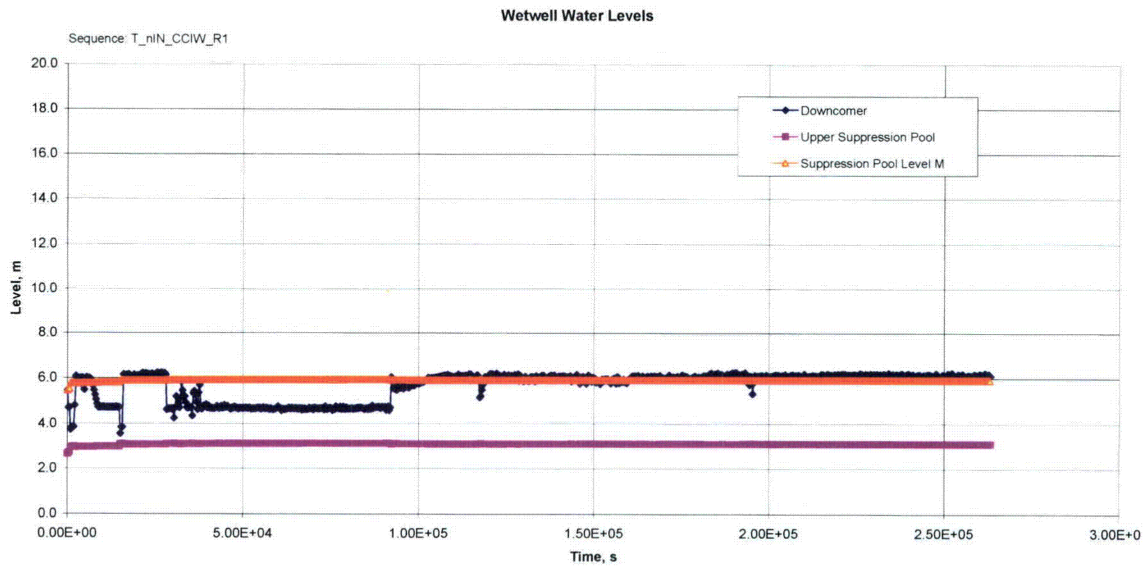


Figure 9A-7o - T_nIN_CCIW_R1 Wetwell Water Levels

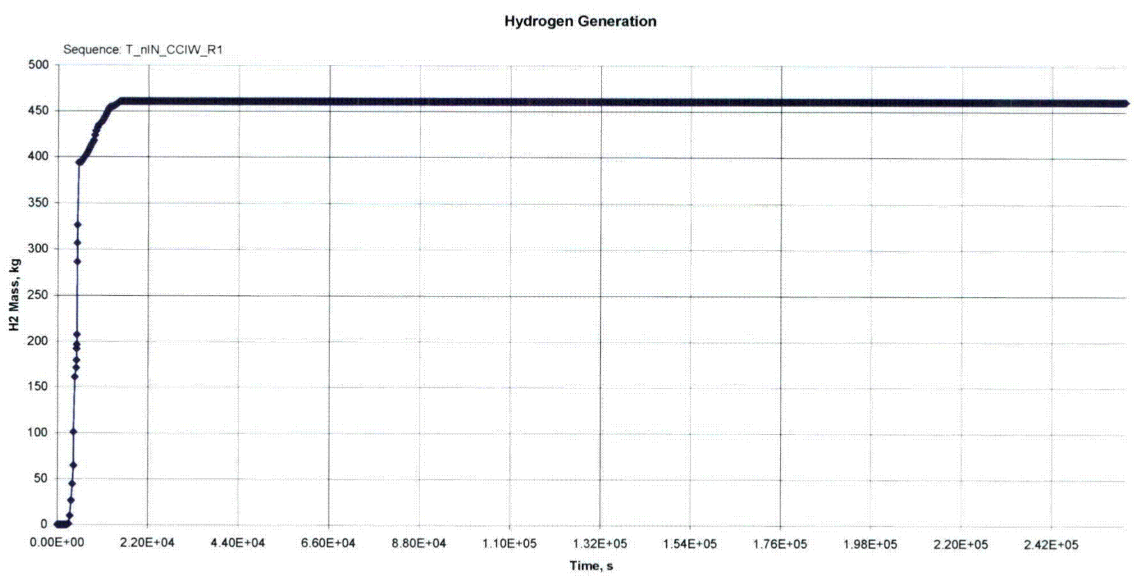


Figure 9A-7p - T_nIN_CCIW_R1 Hydrogen Generation

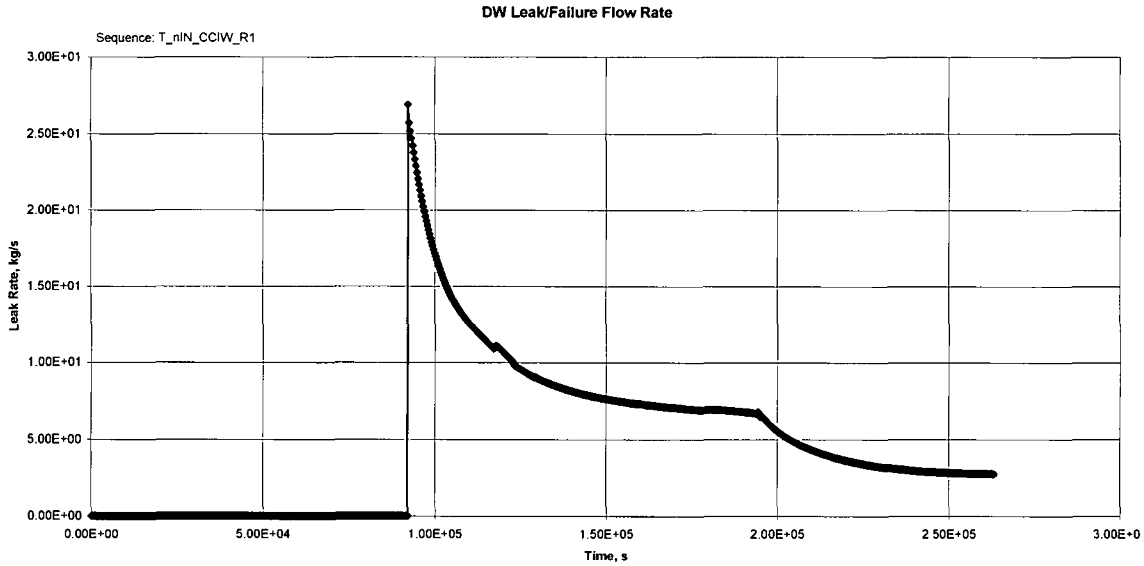


Figure 9A-7q - T_nIN_CCIW_R1 DW Leak/Failure Row Rate

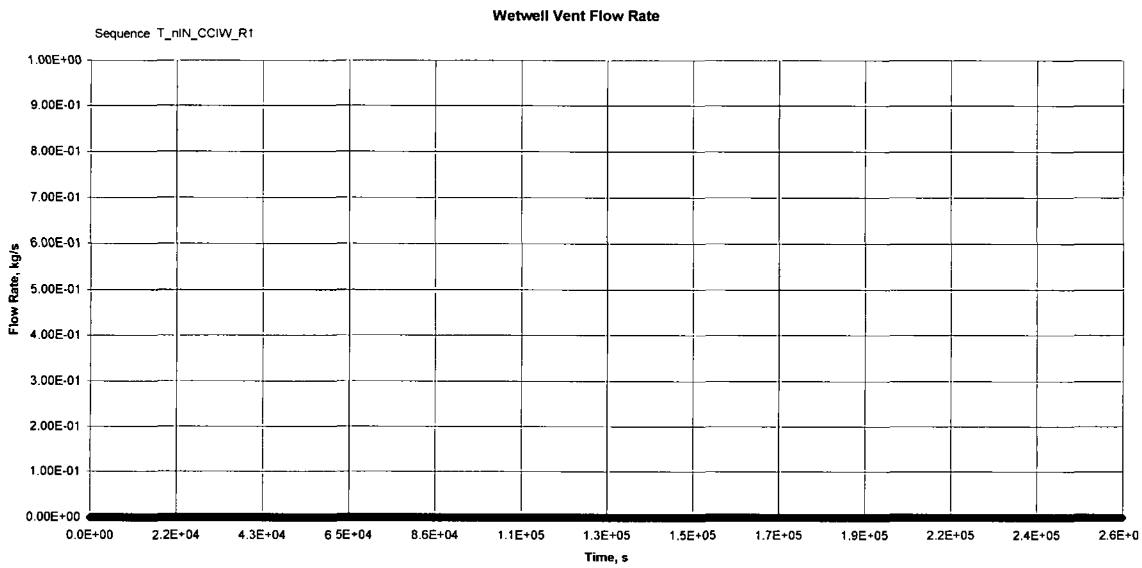


Figure 9A-7r - T_nIN_CCIW_R1 Wetwell Vent Flow Rate

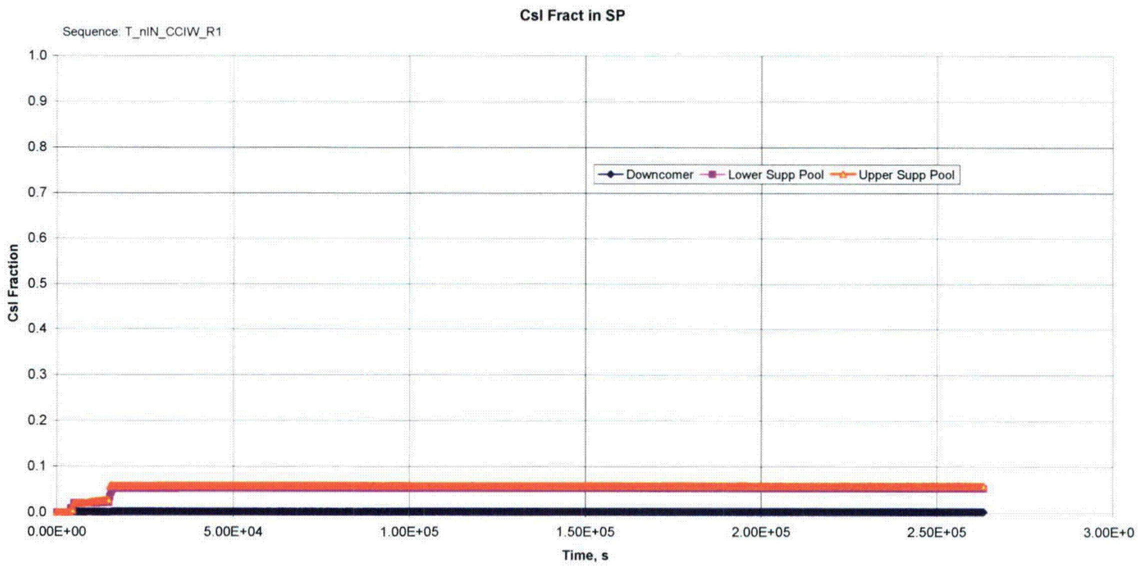


Figure 9A-7s - T_nIN_CCIW_R1 CsI Fract in SP

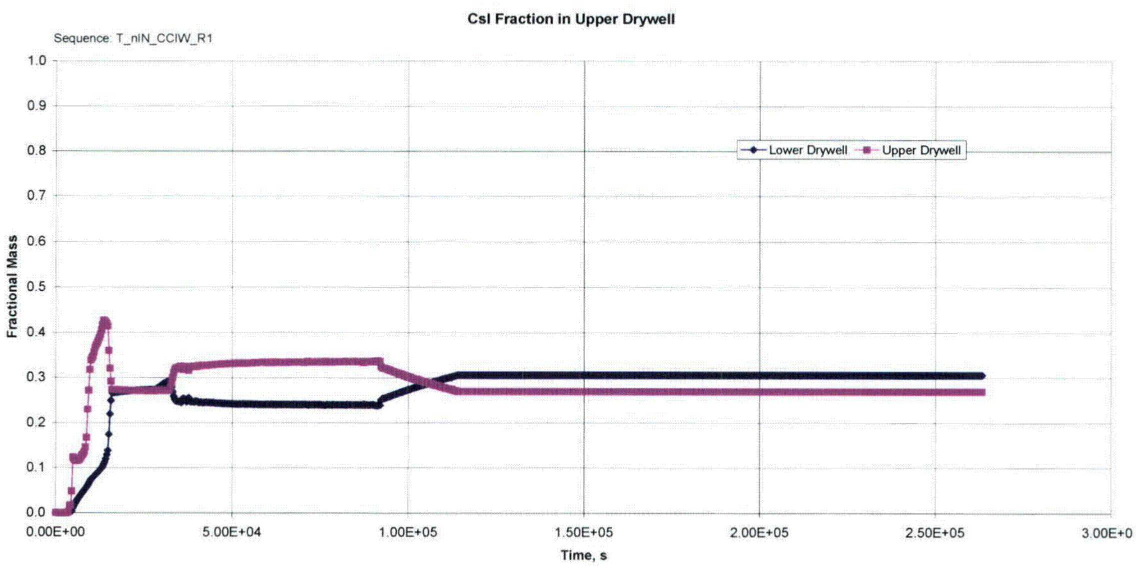


Figure 9A-7t - T_nIN_CCIW_R1 CsI Fraction in Upper Drywell

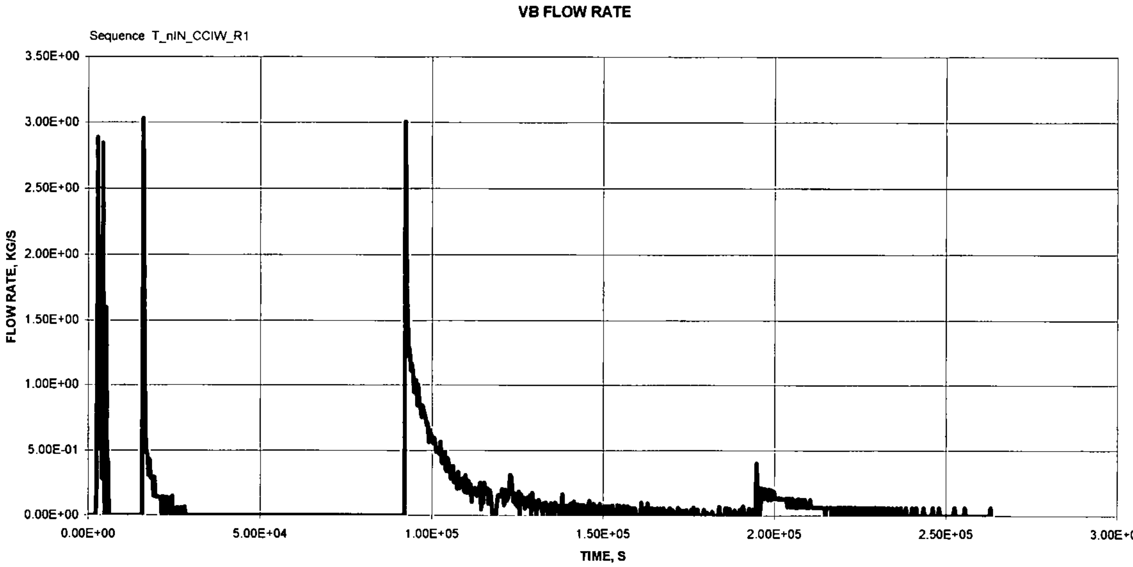


Figure 9A-7u - T_nIN_CCIW_R1 VB Flow Rate

9A.8 Core Concrete Interaction (Wet) – High Pressure T_nDP_nIN_CCIW_R1

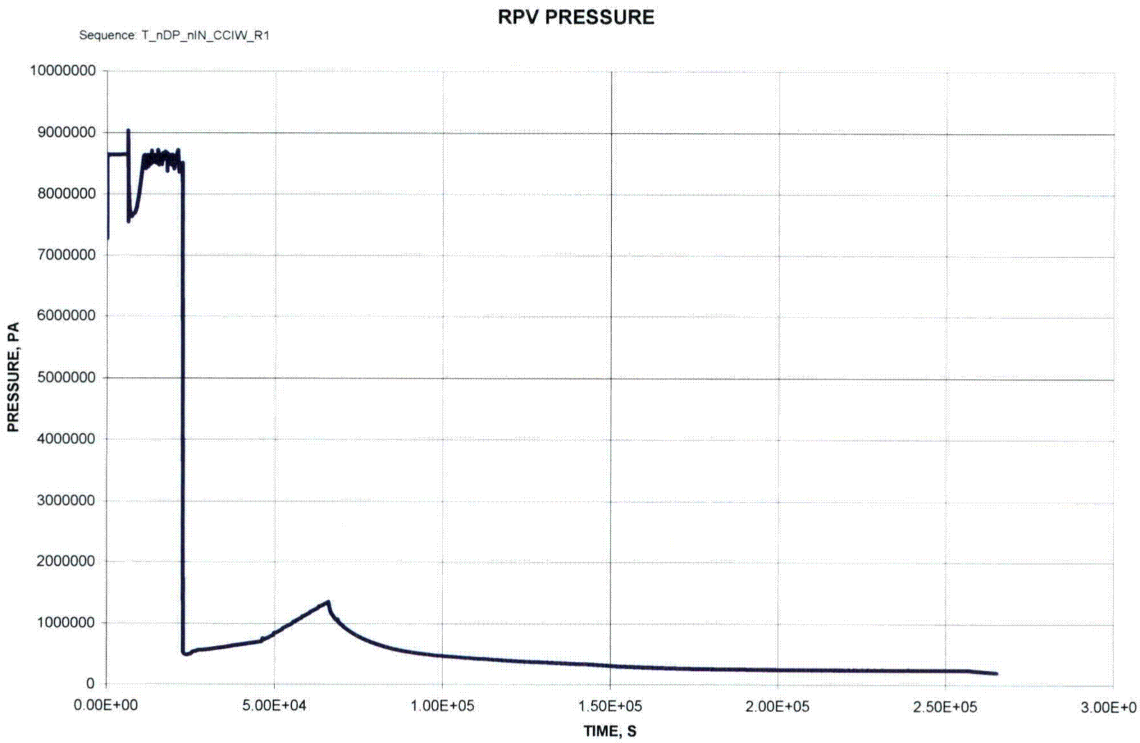


Figure 9A-8a - T_nDP_nIN_CCIW_R1 RPV Pressure

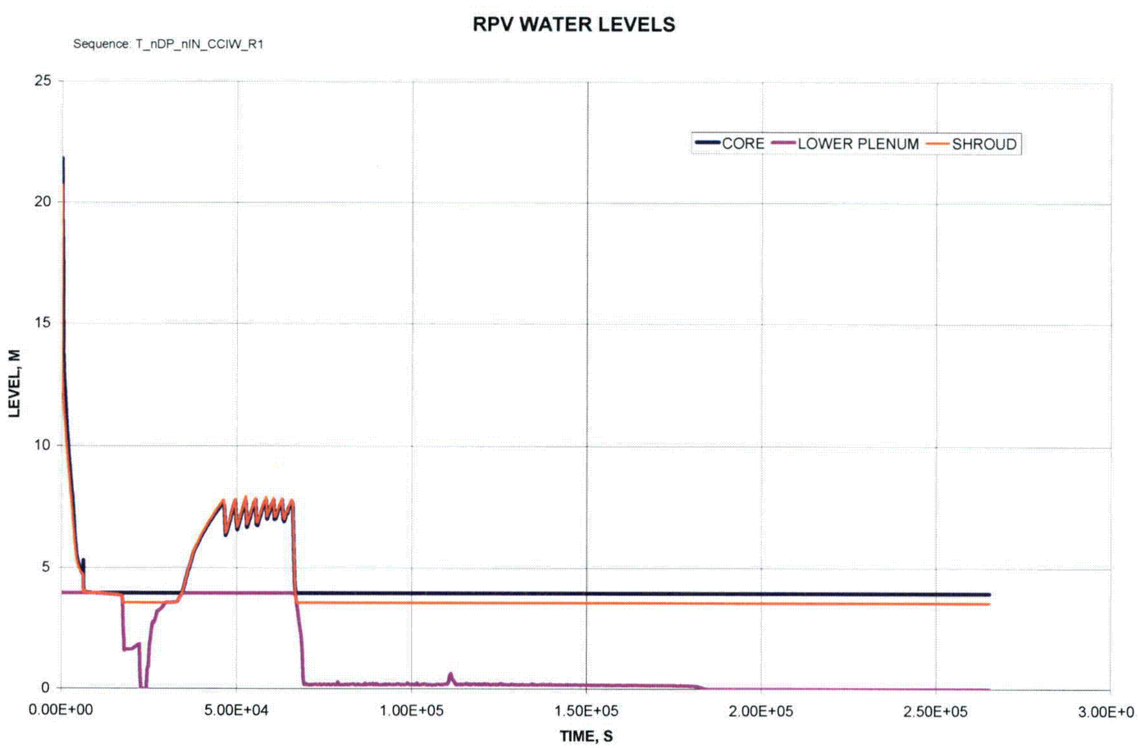


Figure 9A-8b - T_nDP_nIN_CCIW_R1 RPV Water Levels

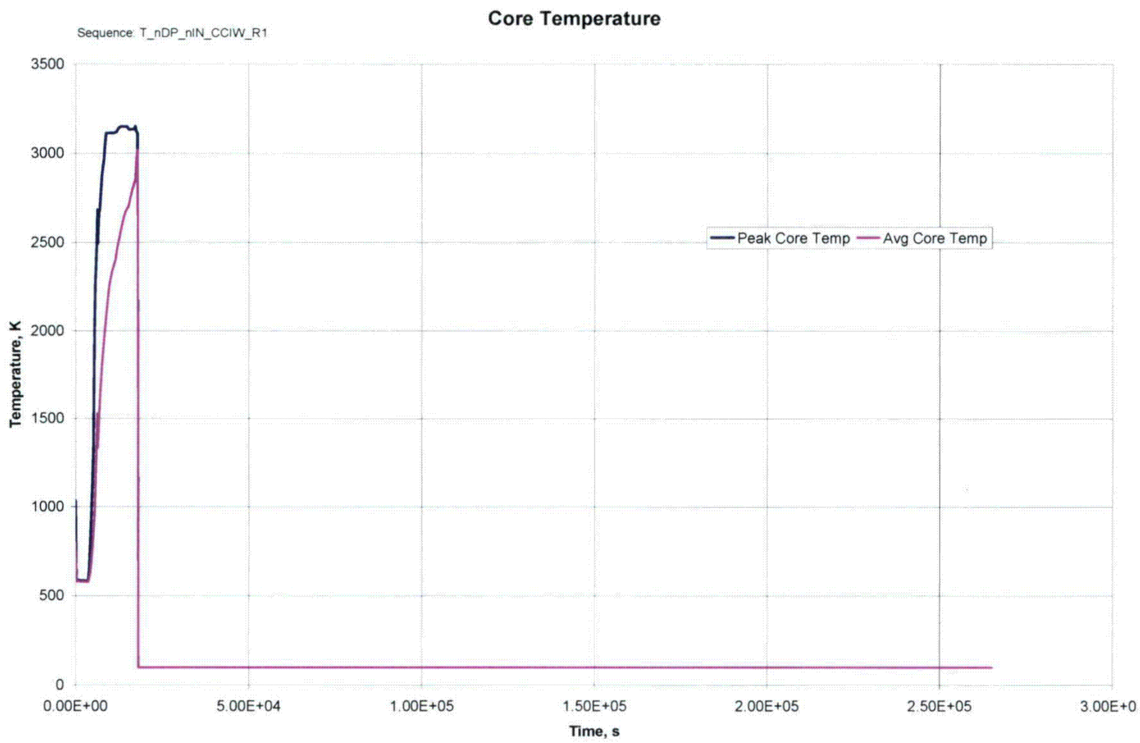


Figure 9A-8c - T_nDP_nIN_CCIW_R1 Core Temperature

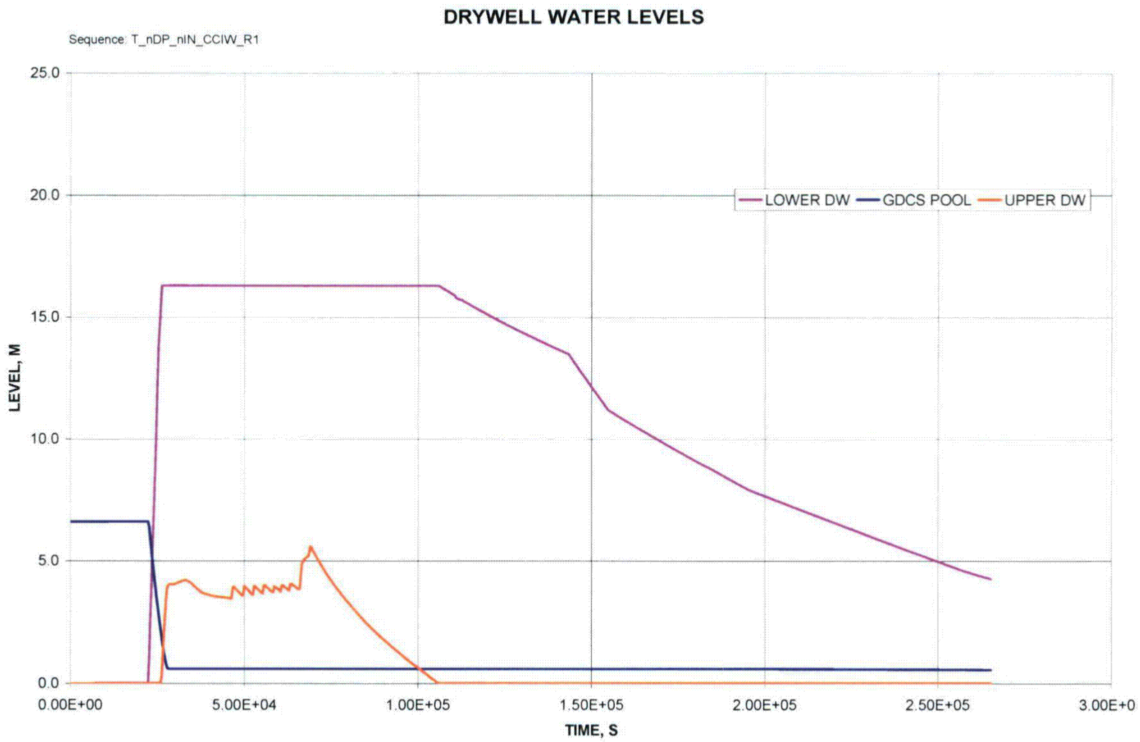


Figure 9A-8d - T_nDP_nIN_CCIW_R1 Drywell Water Levels

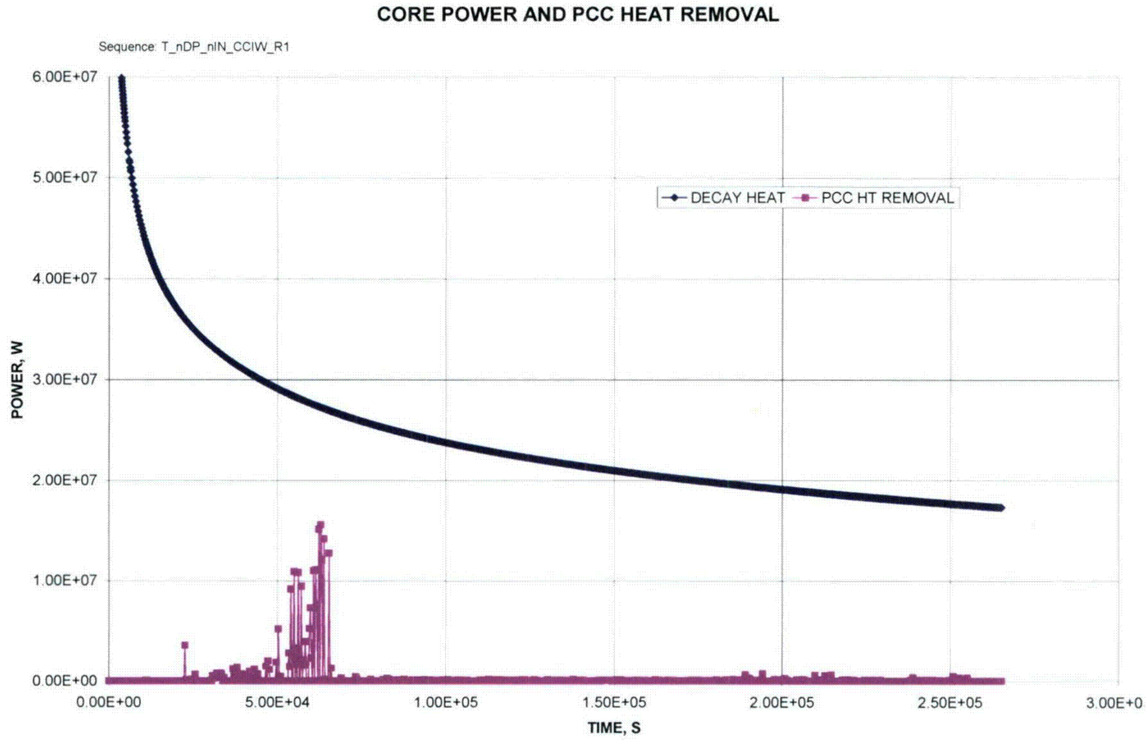


Figure 9A-8e - T_nDP_nIN_CCIW_R1 Core Power and PCC Heat Removal

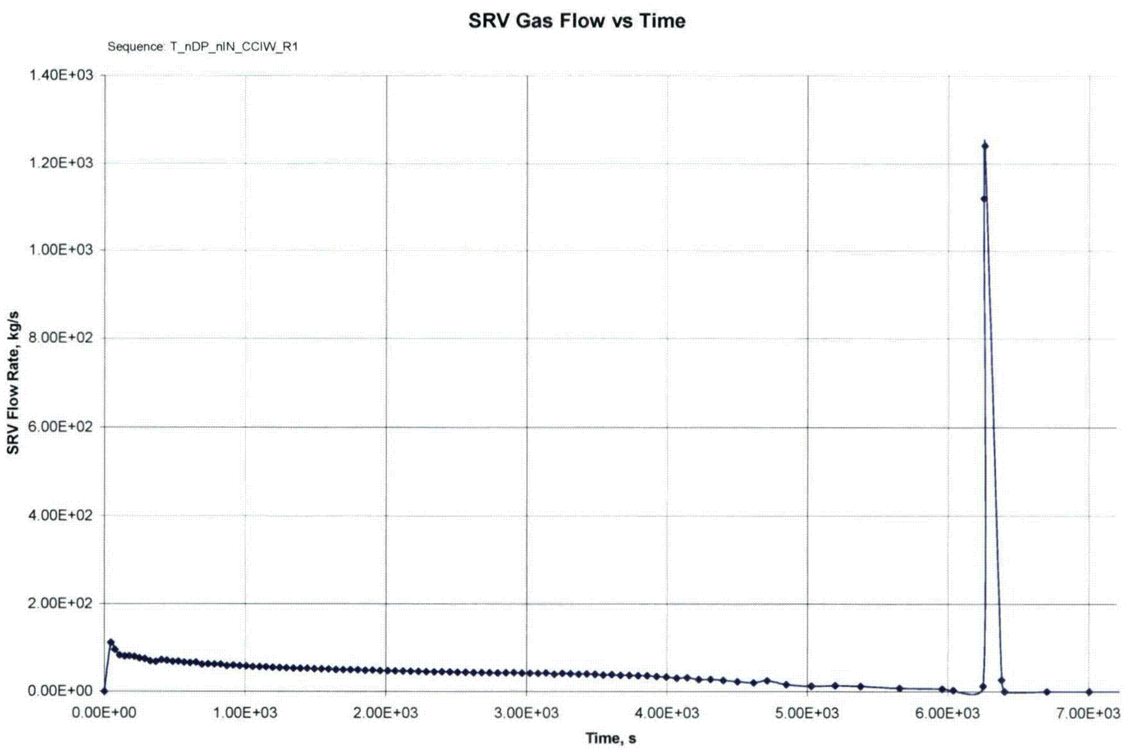


Figure 9A-8f - T_nDP_nIN_CCIW_R1 SRV Gas Flow vs. Time

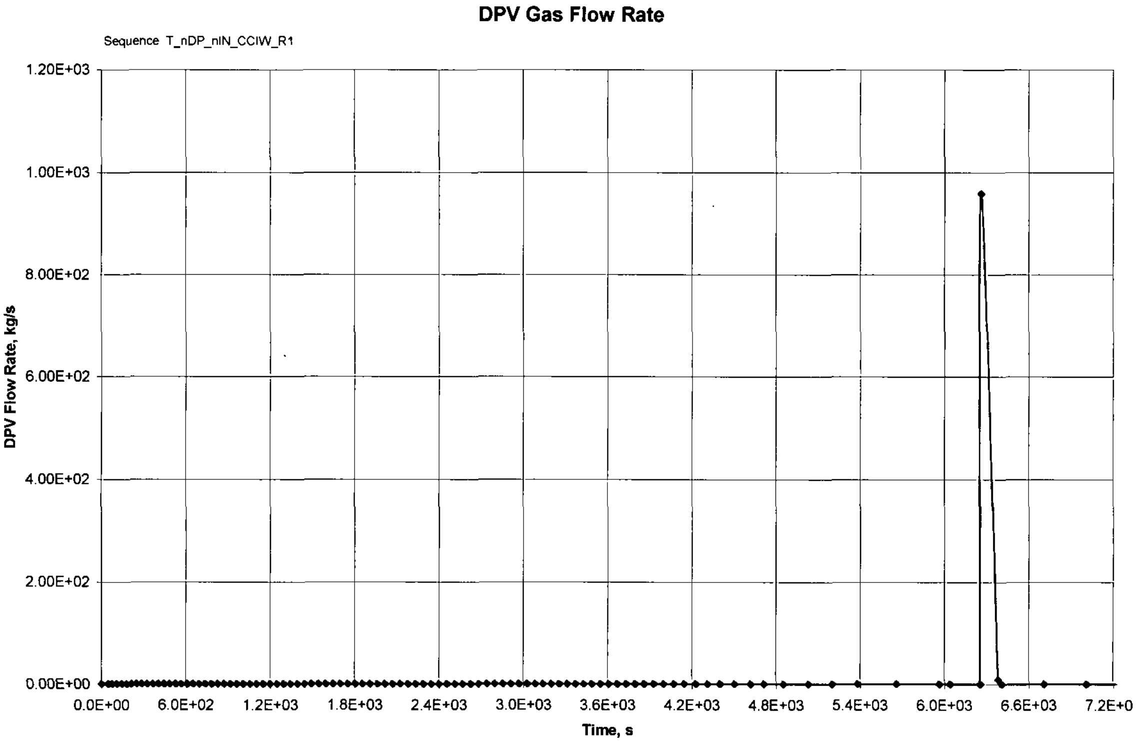


Figure 9A-8g - T_nDP_nIN_CCIW_R1 DPV Gas Flow Rate

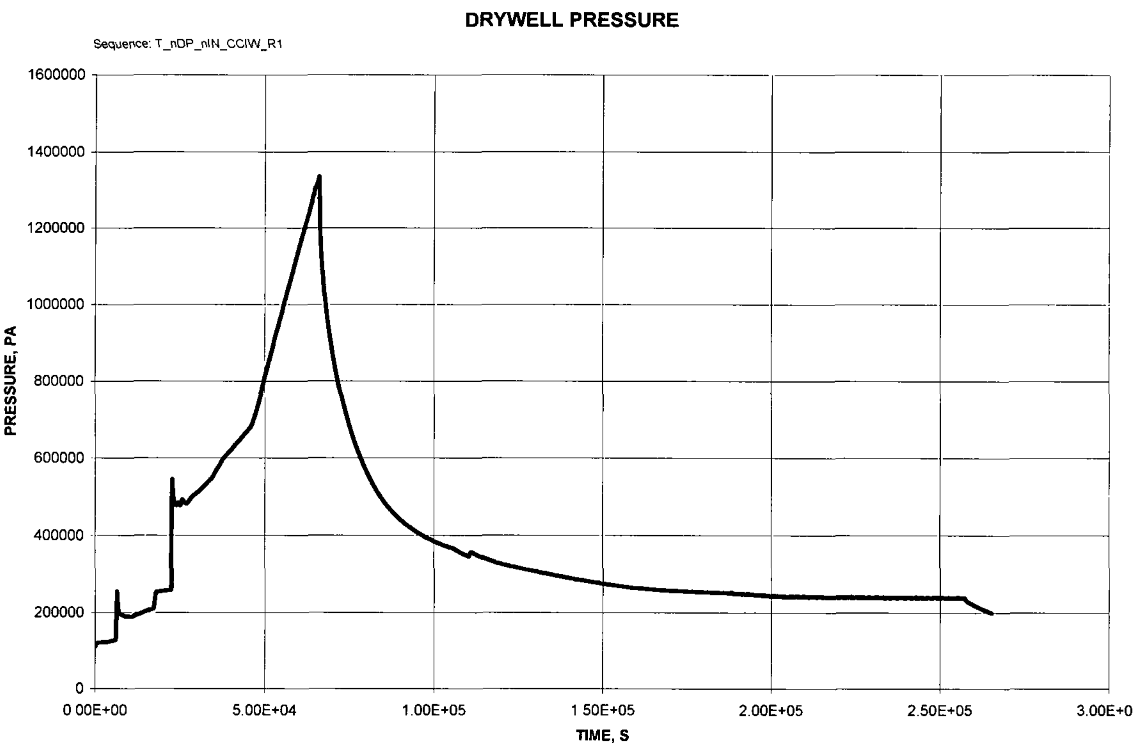


Figure 9A-8h - T_nDP_nIN_CCIW_R1 Drywell Pressure

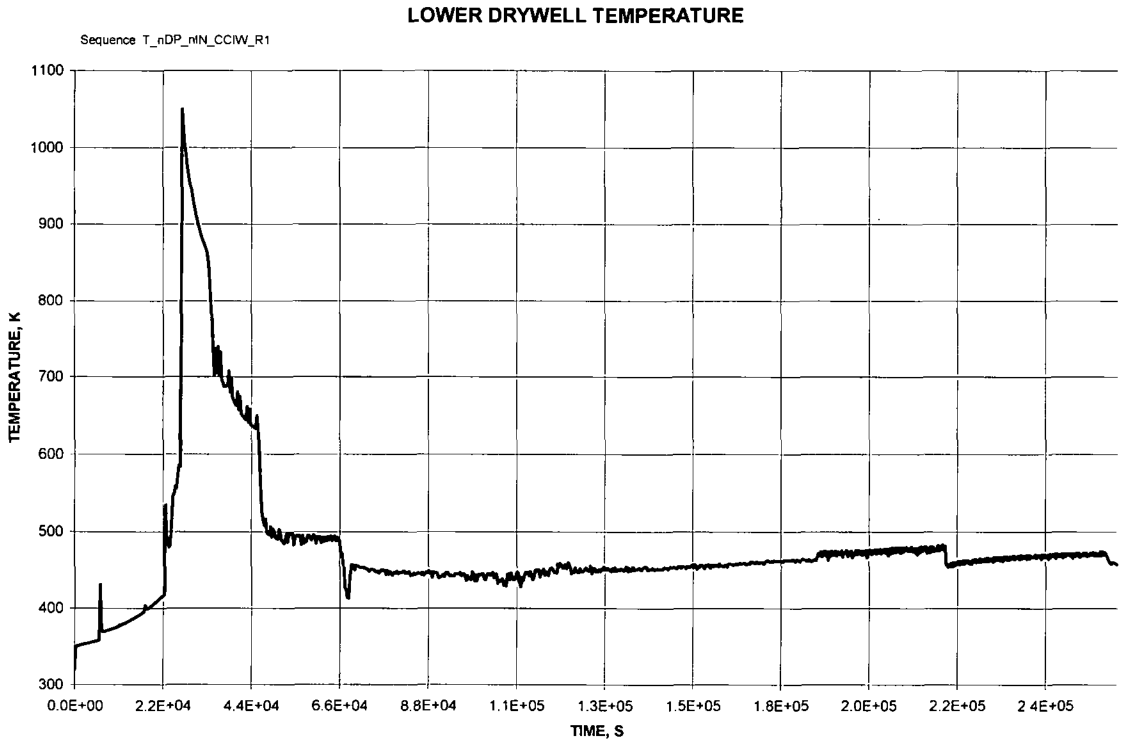


Figure 9A-8i - T_nDP_nIN_CCIW_R1 Lower Drywell Temperature

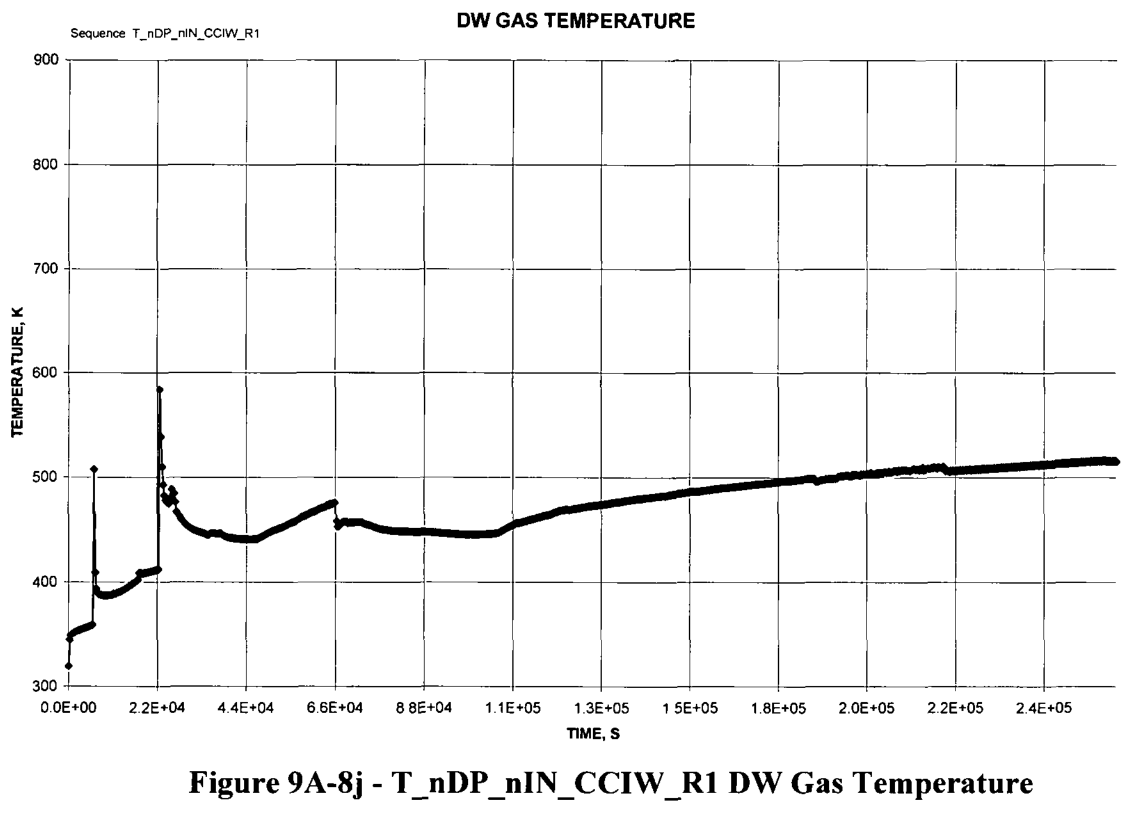


Figure 9A-8j - T_nDP_nIN_CCIW_R1 DW Gas Temperature

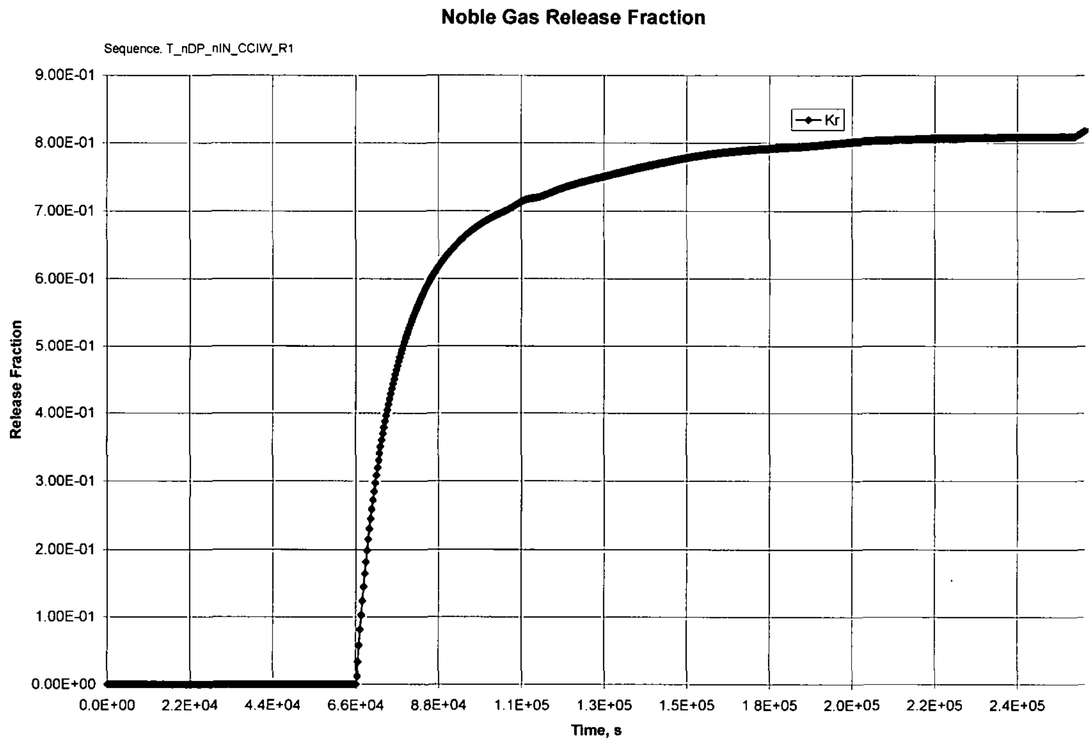


Figure 9A-8k - T_nDP_nIN_CCIW_R1 Noble Gas Release Fraction

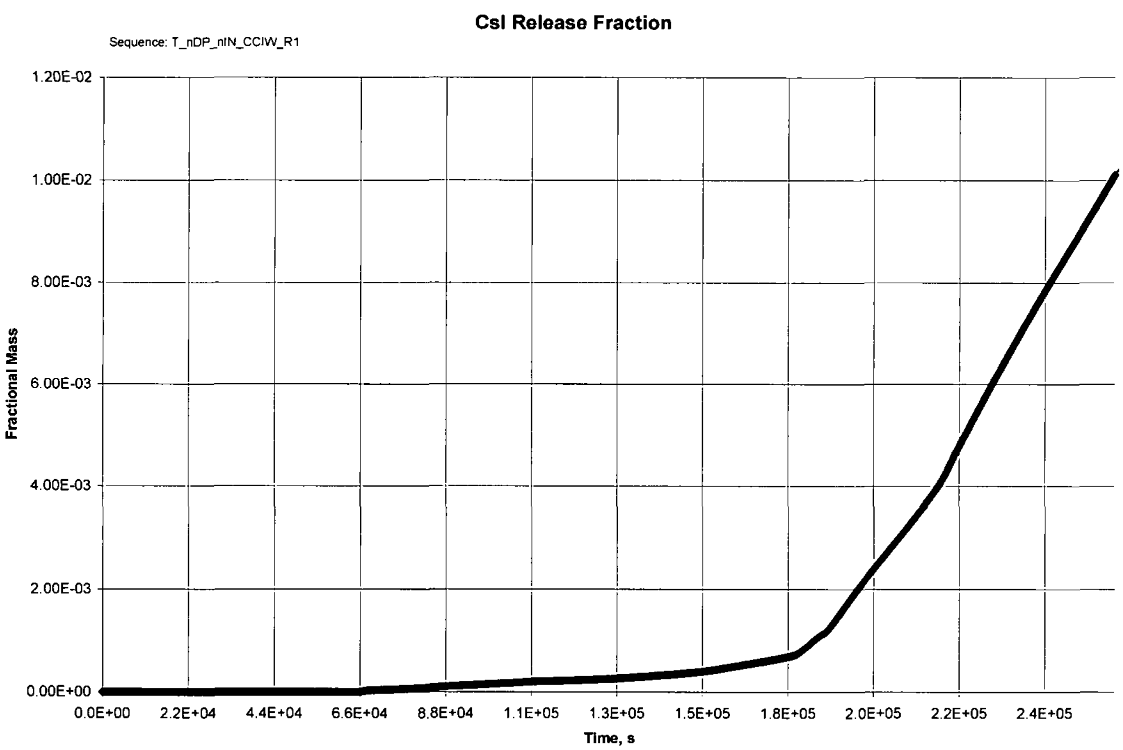


Figure 9A-8l - T_nDP_nIN_CCIW_R1 CsI Release Fraction

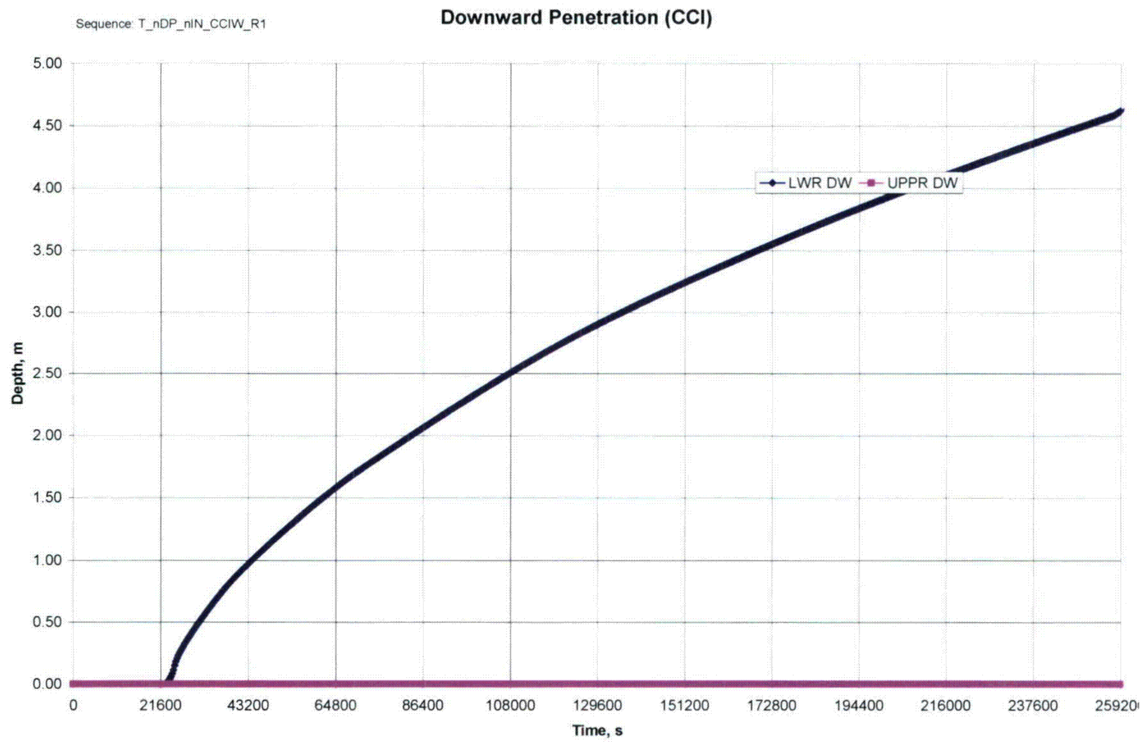


Figure 9A-8m - T_nDP_nIN_CCIW_R1 Downward Penetration (CCI)

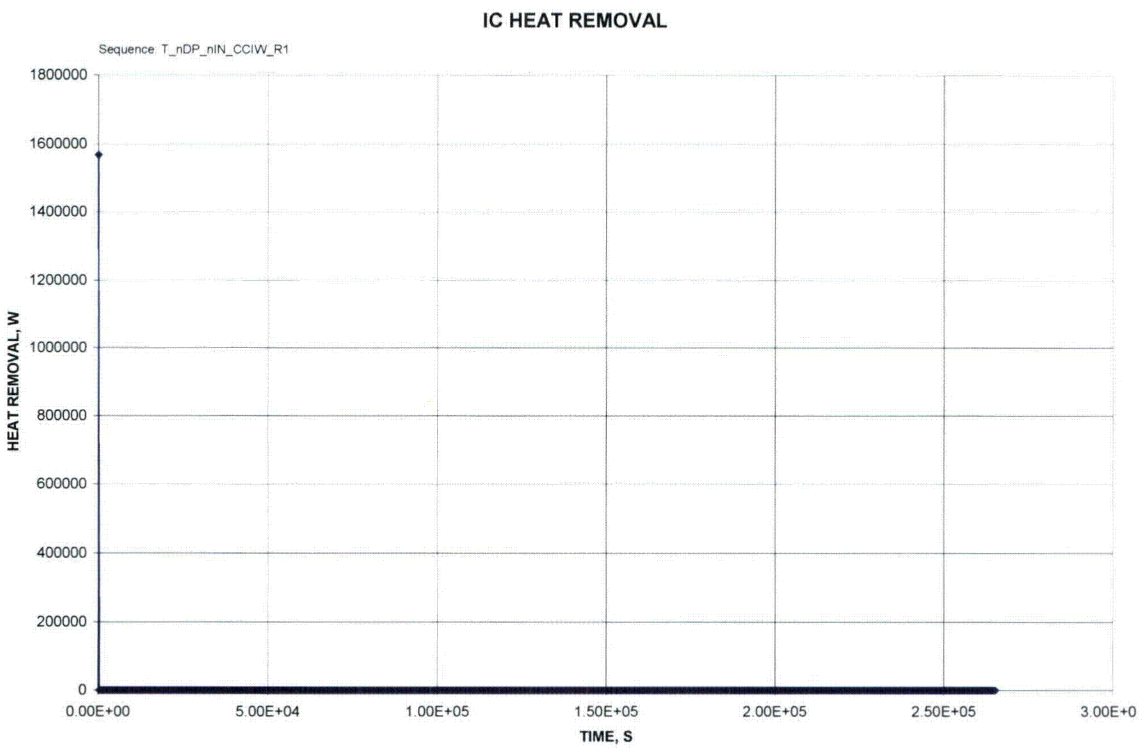


Figure 9A-8n - T_nDP_nIN_CCIW_R1 ICS Heat Removal

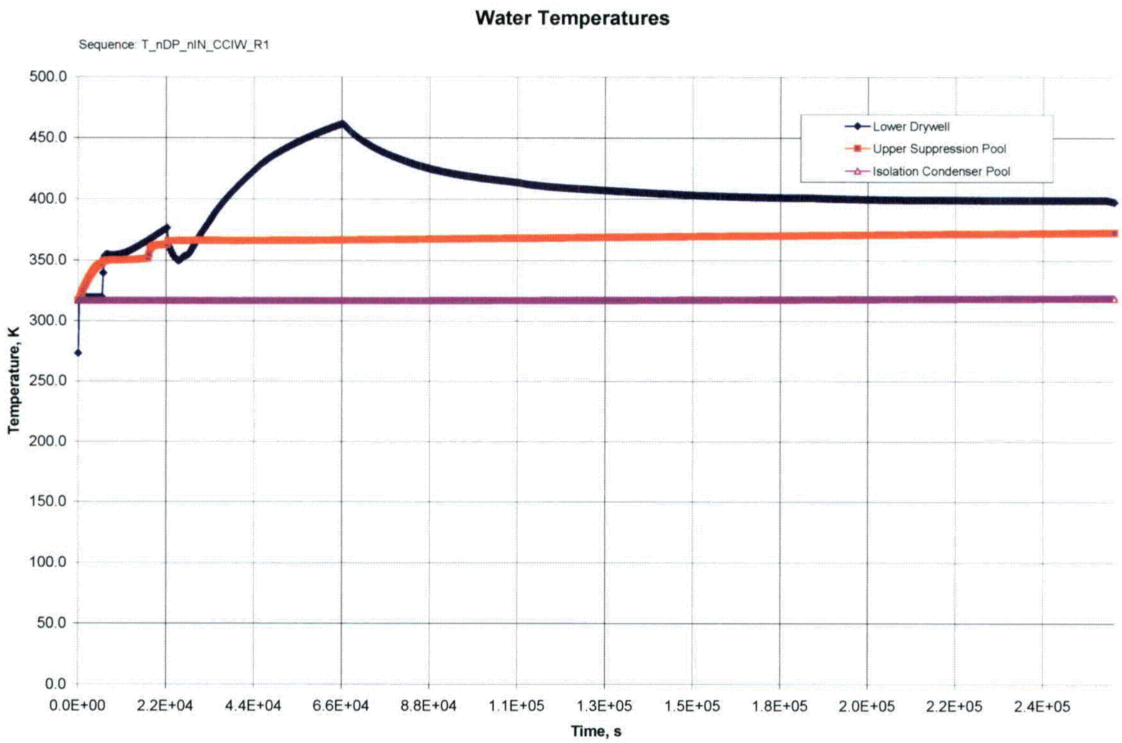


Figure 9A-8o - T_nDP_nIN_CCIW_R1 Water Temperatures

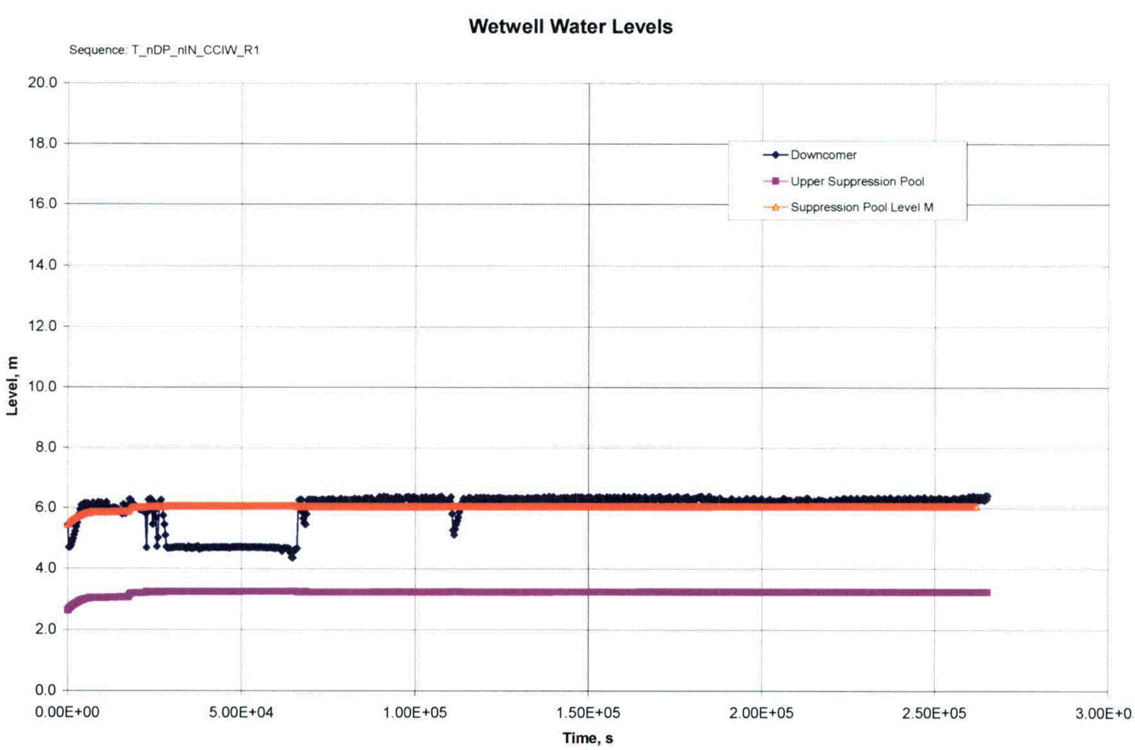


Figure 9A-8p - T_nDP_nIN_CCIW_R1 Wetwell Water Levels

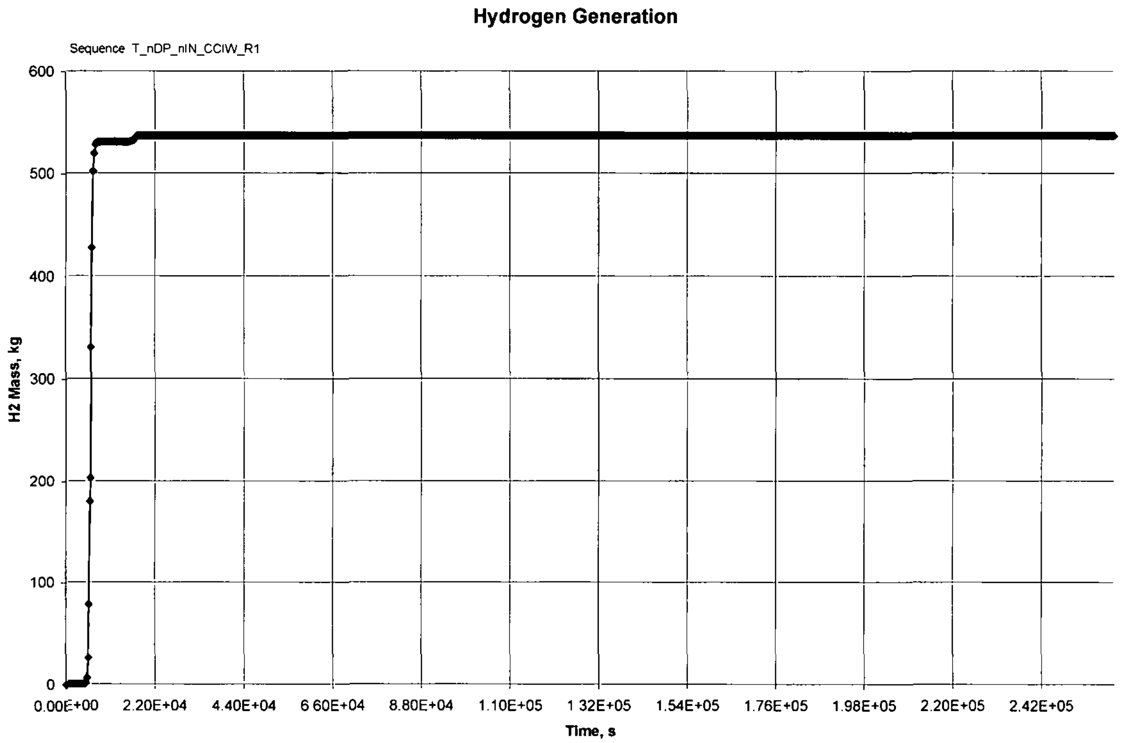


Figure 9A-8q - T_nDP_nIN_CCIW_R1 Hydrogen Generation

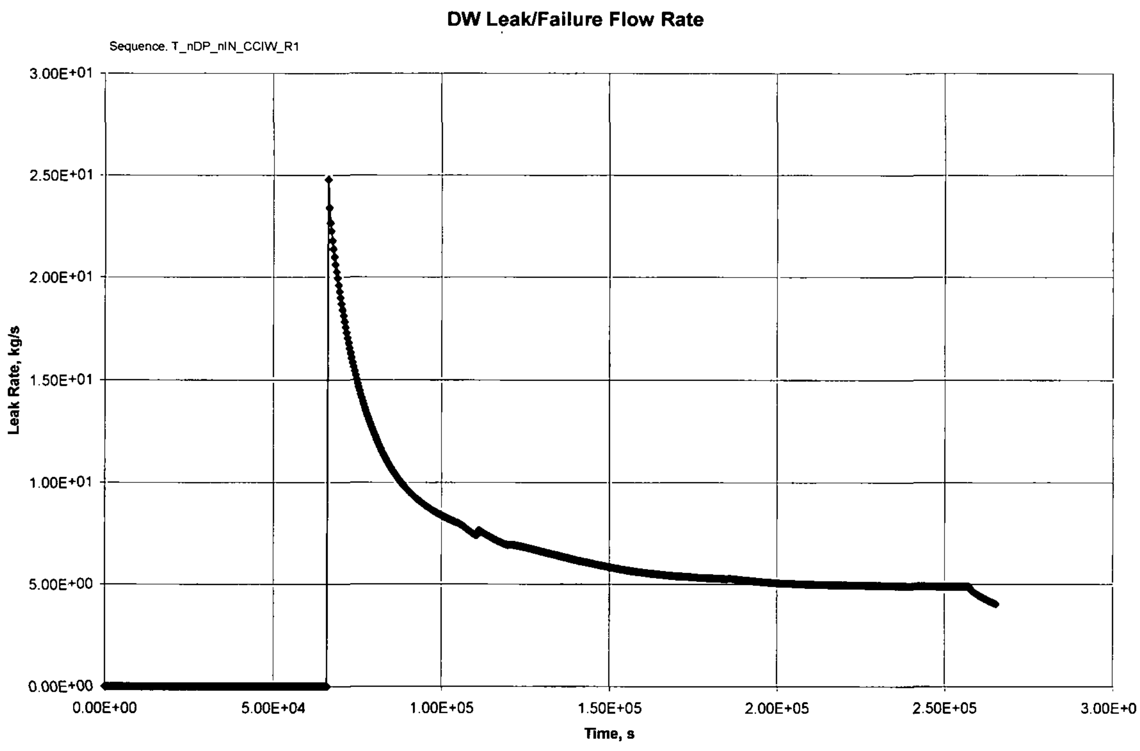


Figure 9A-8r - T_nDP_nIN_CCIW_R1 DW Leak/Failure Flow Rate

Figure 9A-8s - T_nDP_nIN_CCIW_R1 Wetwell Vent Flow Rate

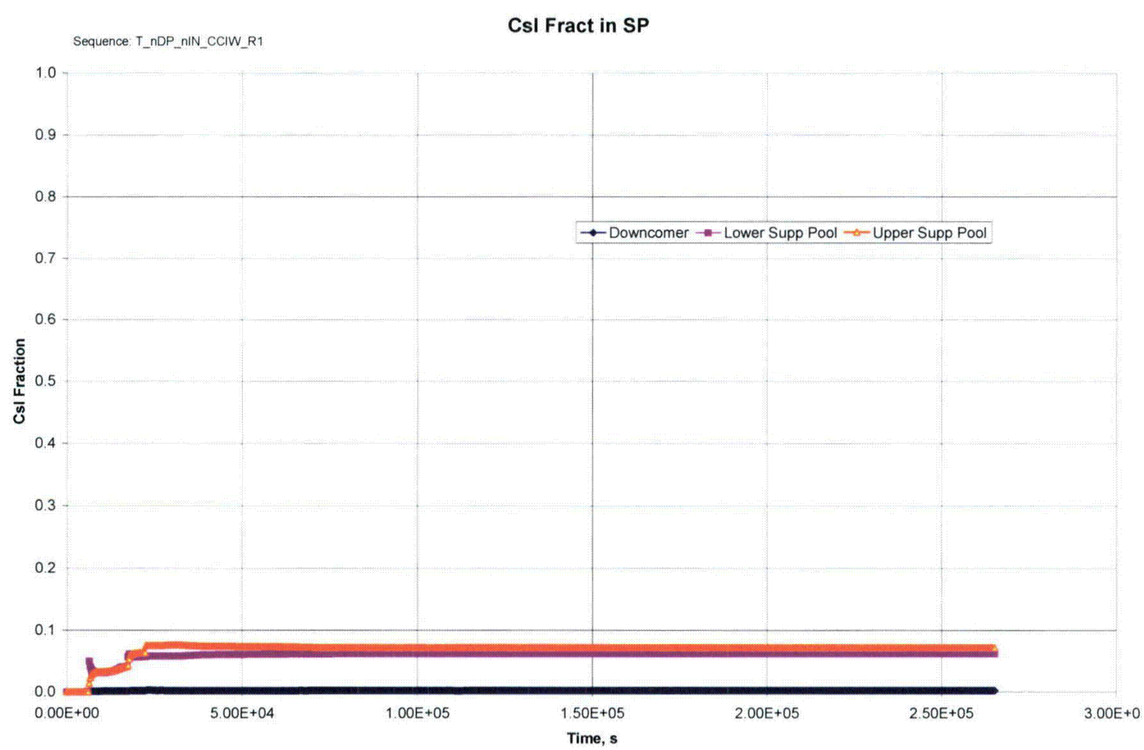


Figure 9A-8t - T_nDP_nIN_CCIW_R1 CsI Fraction in SP

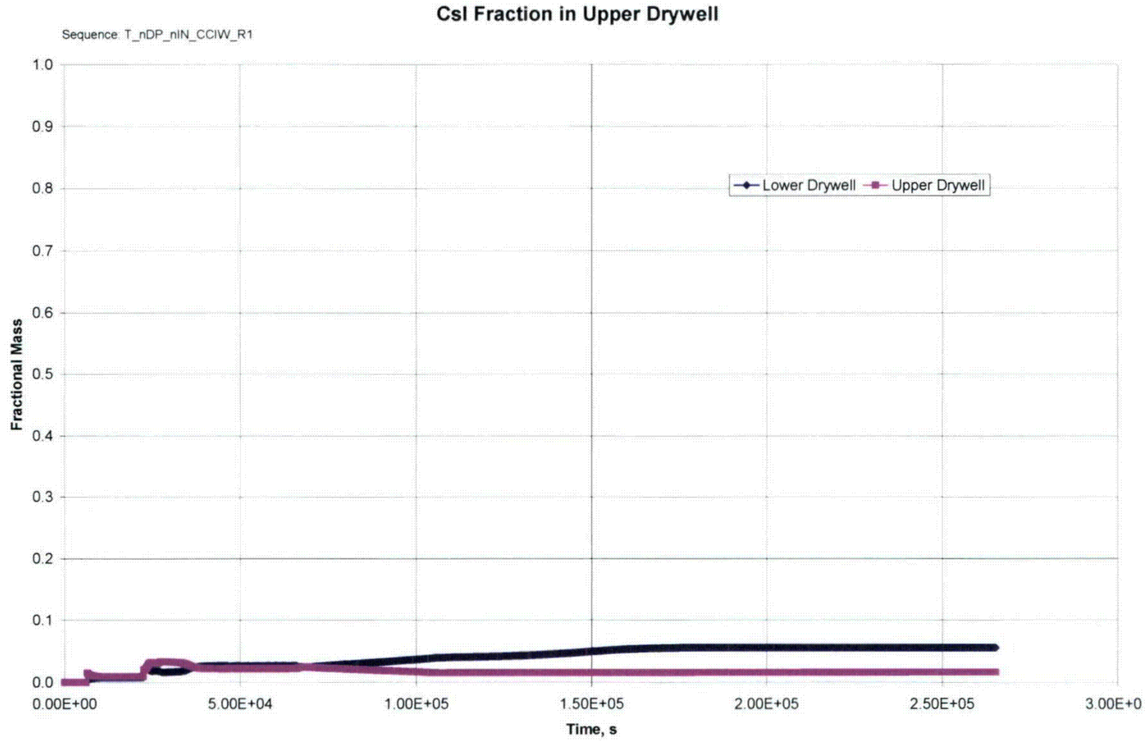


Figure 9A-8u - T_nDP_nIN_CCIW_R1 Csl Fraction in Upper Drywell

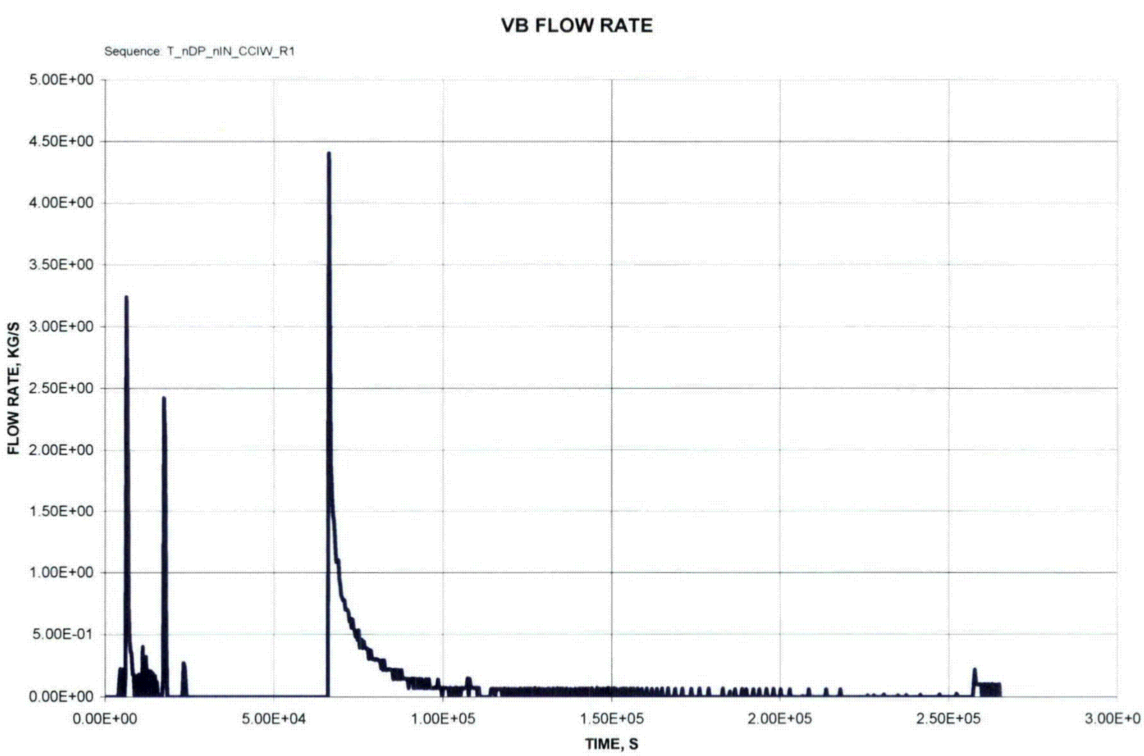


Figure 9A-8v - T_nDP_nIN_CCIW_R1 VB Flowrate

9A.9 Direct Containment Heating T_nDP_nIN_nD_DCH_R1

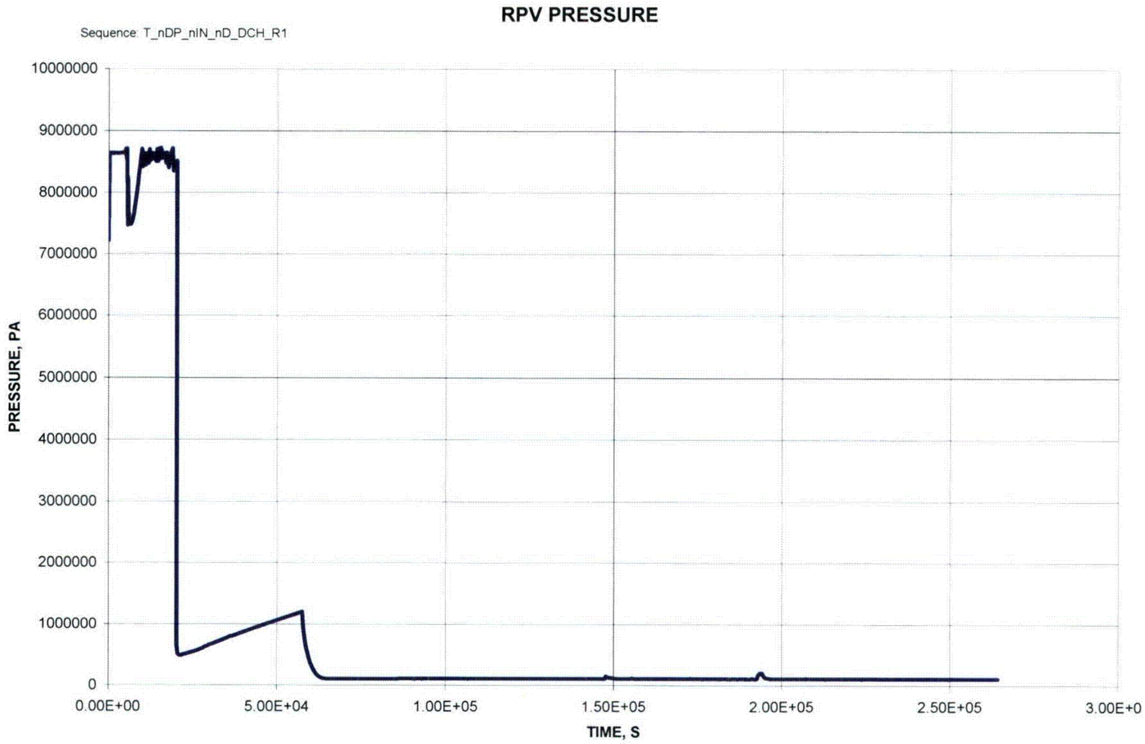


Figure 9A-9a - T_nDP_nIN_nD_DCH_R1 RPV Pressure

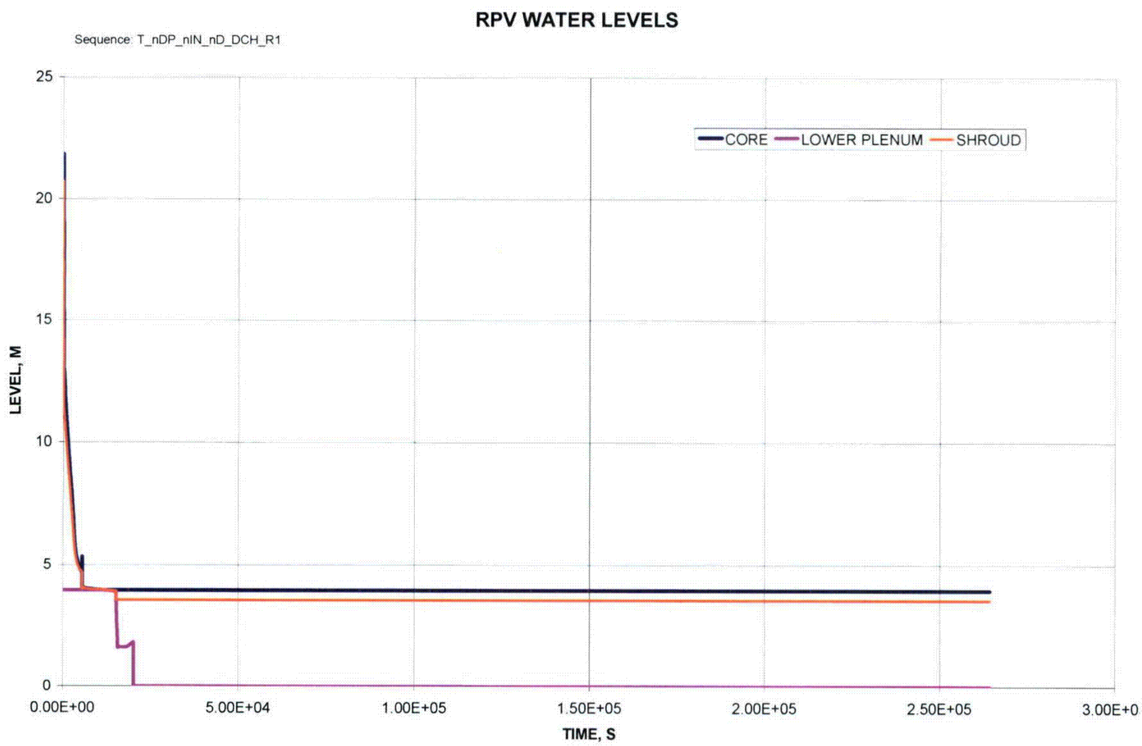


Figure 9A-9b - T_nDP_nIN_nD_DCH_R1 RPV Water Levels

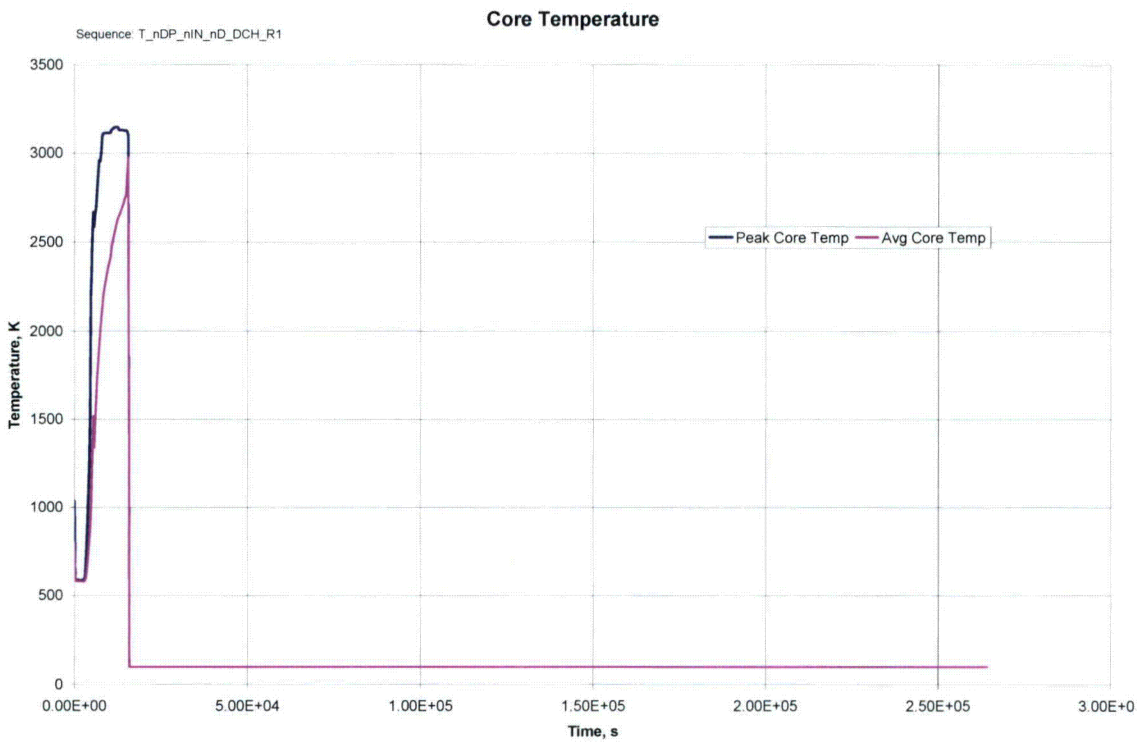


Figure 9A-9c - T_nDP_nIN_nD_DCH_R1 Core Temperature

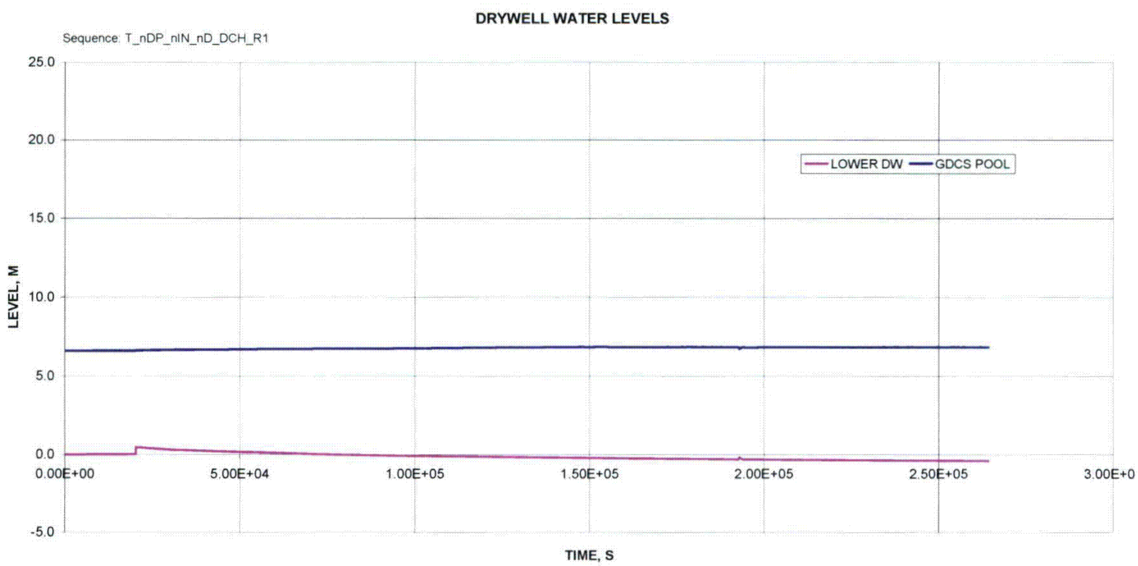


Figure 9A-9d - T_nDP_nIN_nD_DCH_R1 Drywell Water Levels

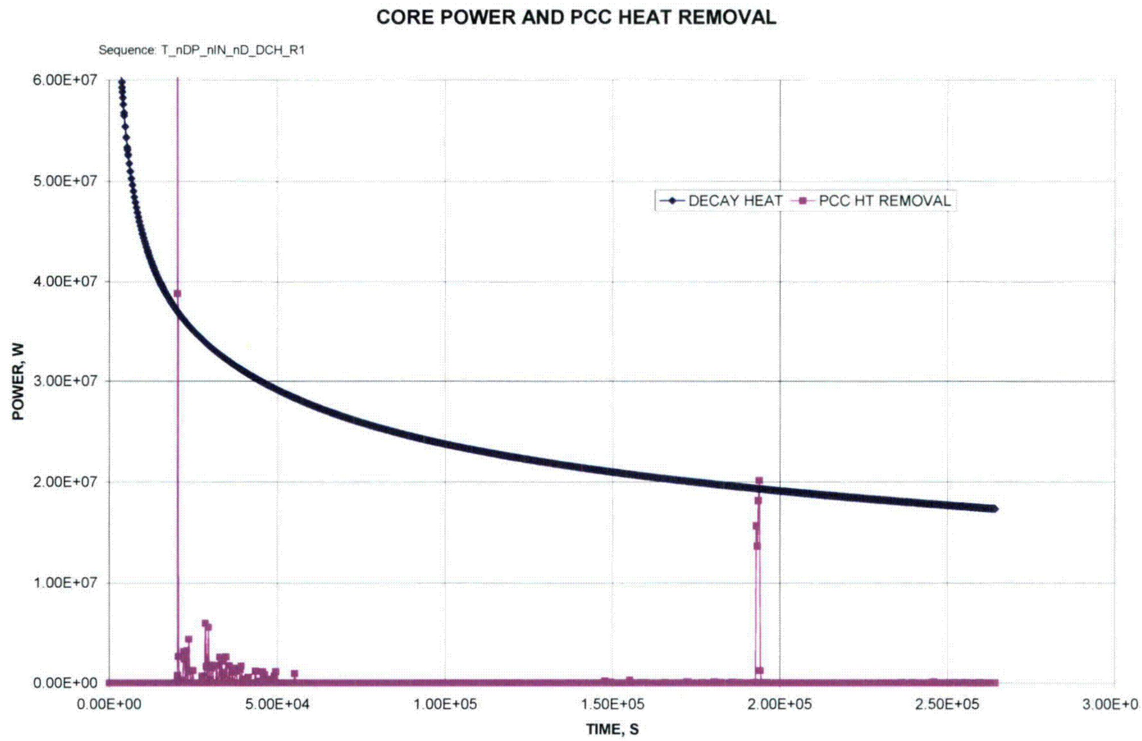


Figure 9A-9e - T_nDP_nIN_nD_DCH_R1 Core Power and PCCS Heat Removal

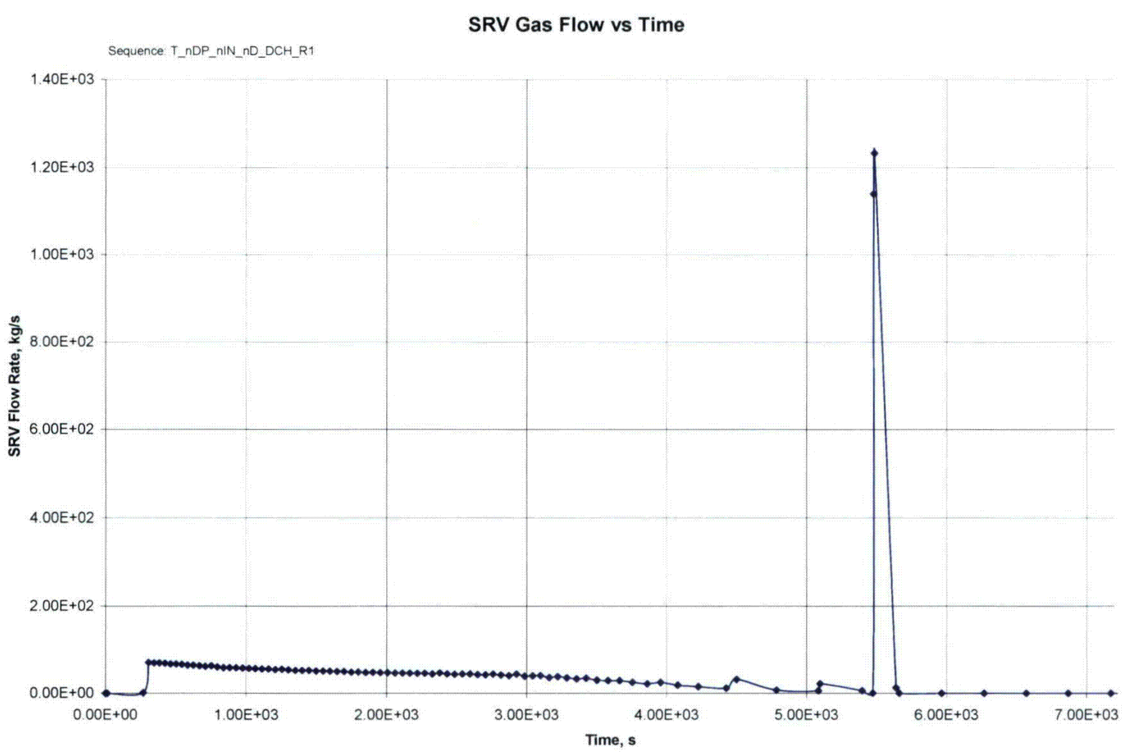


Figure 9A-9f - T_nDP_nIN_nD_DCH_R1 SRV Gas Flow vs. Time

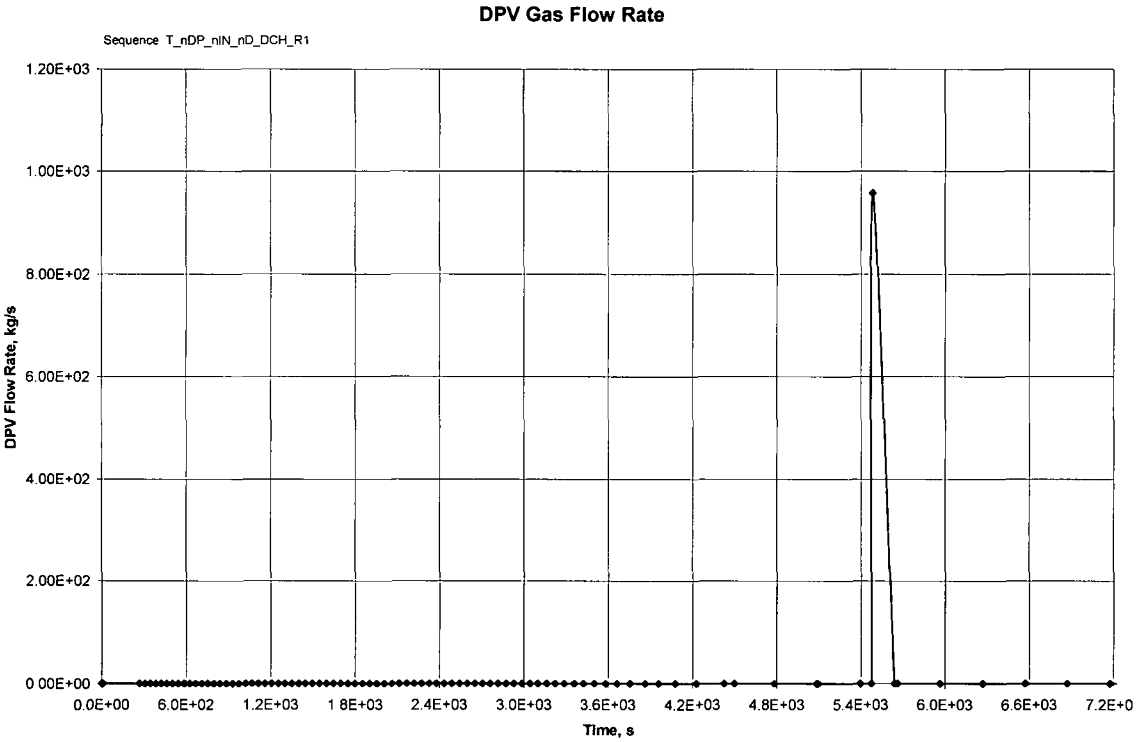


Figure 9A-9g - T_nDP_nIN_nD_DCH_R1 DPV Gas Flow Rate

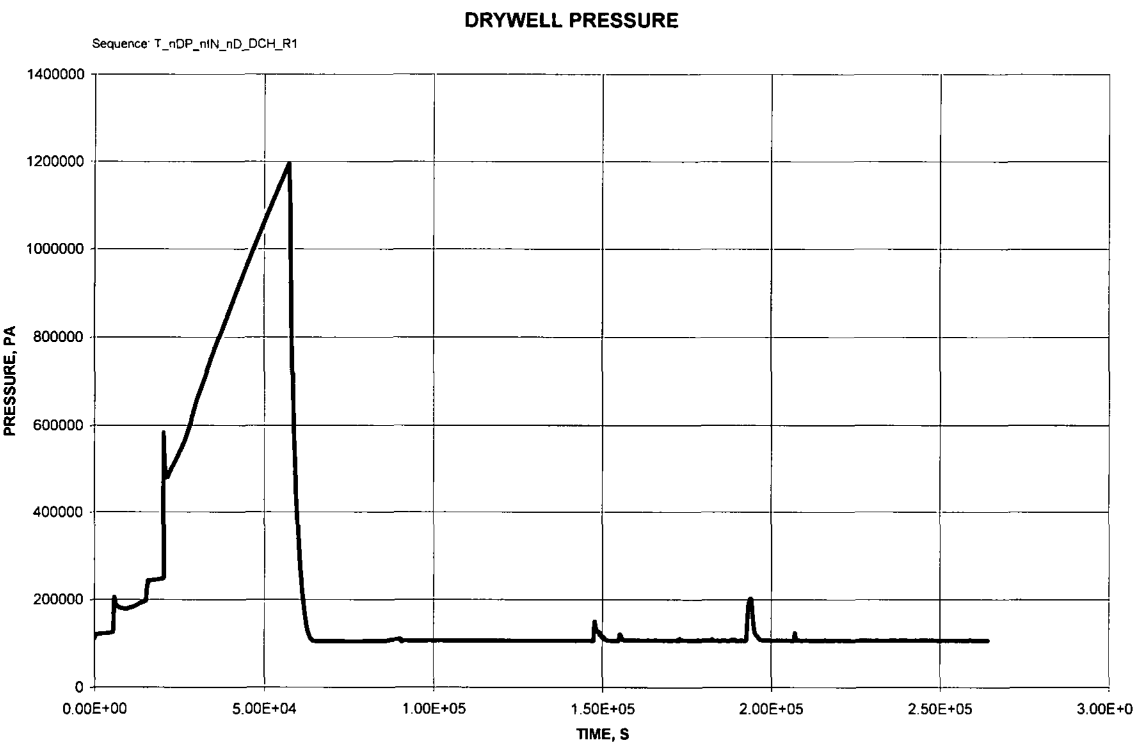


Figure 9A-9h - T_nDP_nIN_nD_DCH_R1 Drywell Pressure

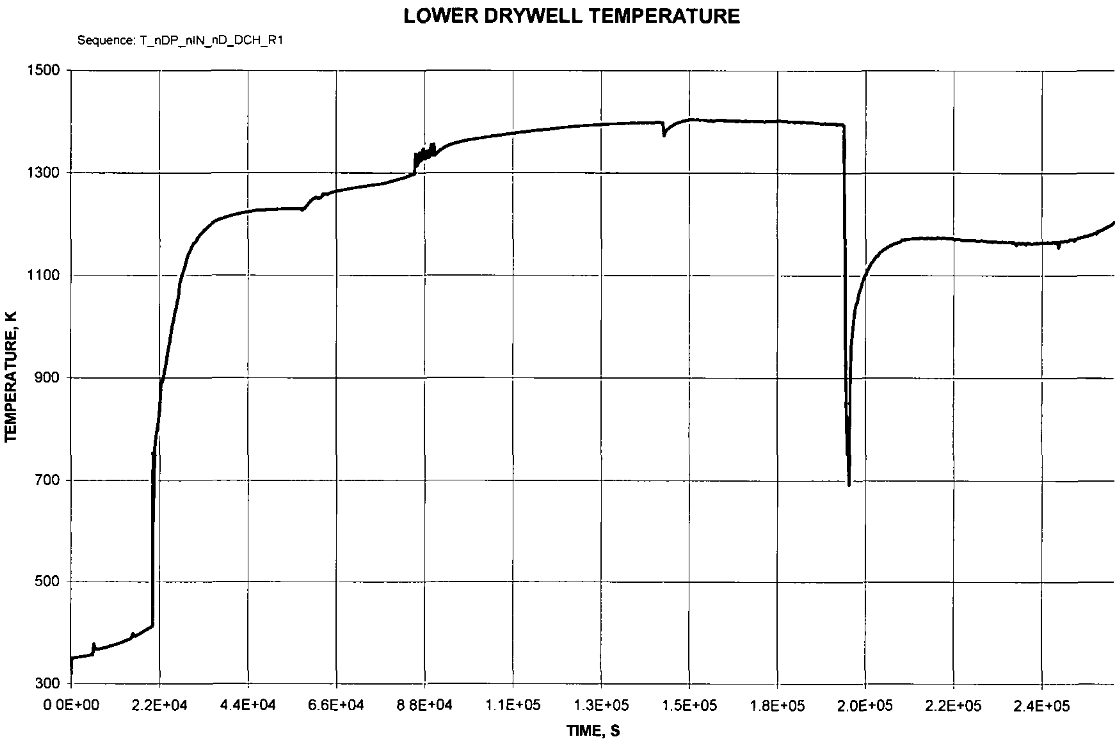


Figure 9A-9i - T_nDP_nIN_nD_DCH_R1 Lower Drywell Temperature

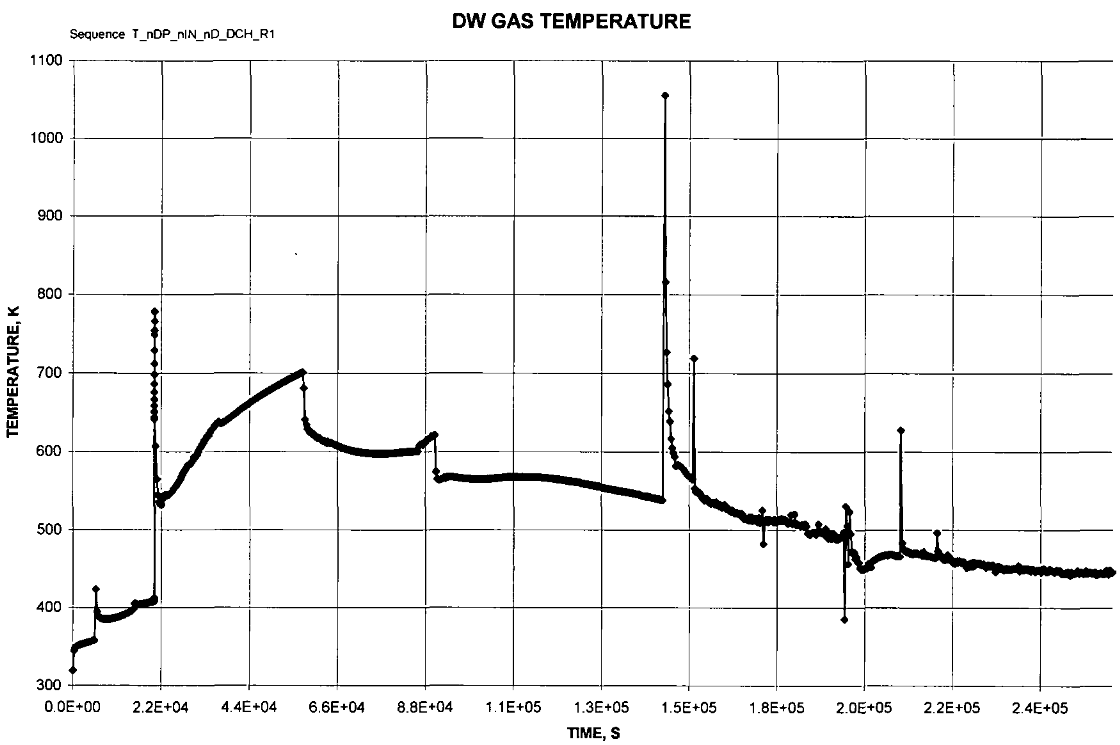


Figure 9A-9j - T_nDP_nIN_nD_DCH_R1 DW Gas Temperature

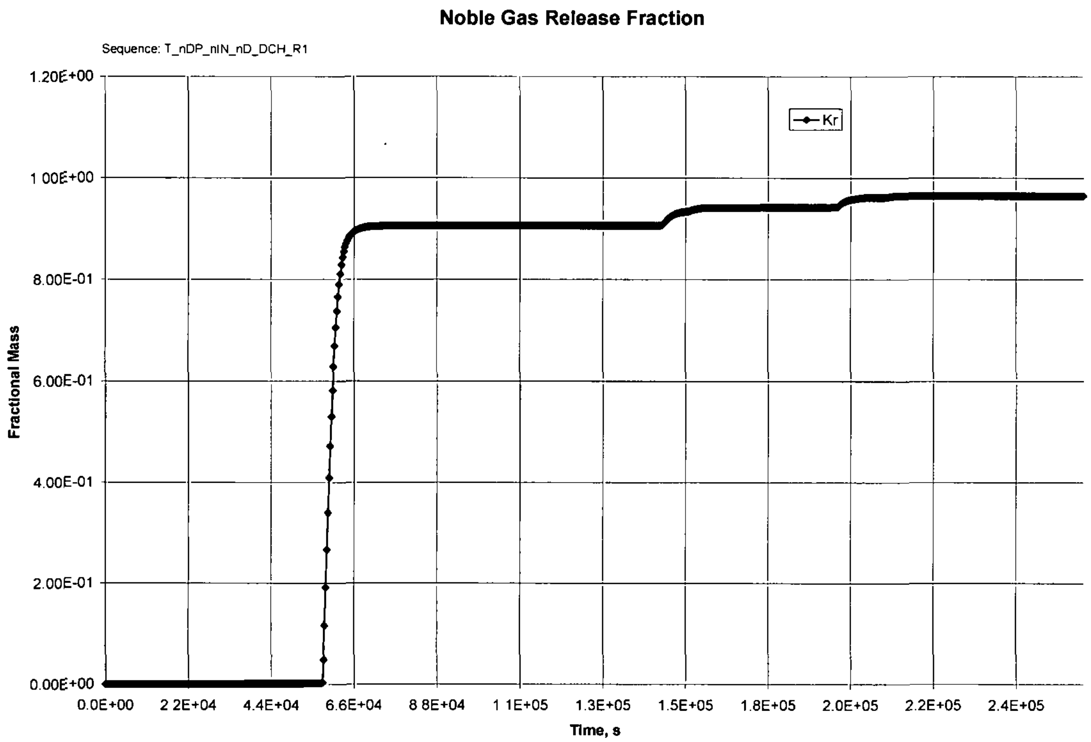


Figure 9A-9k - T_nDP_nIN_nD_DCH_R1 Noble Gas Release Fraction

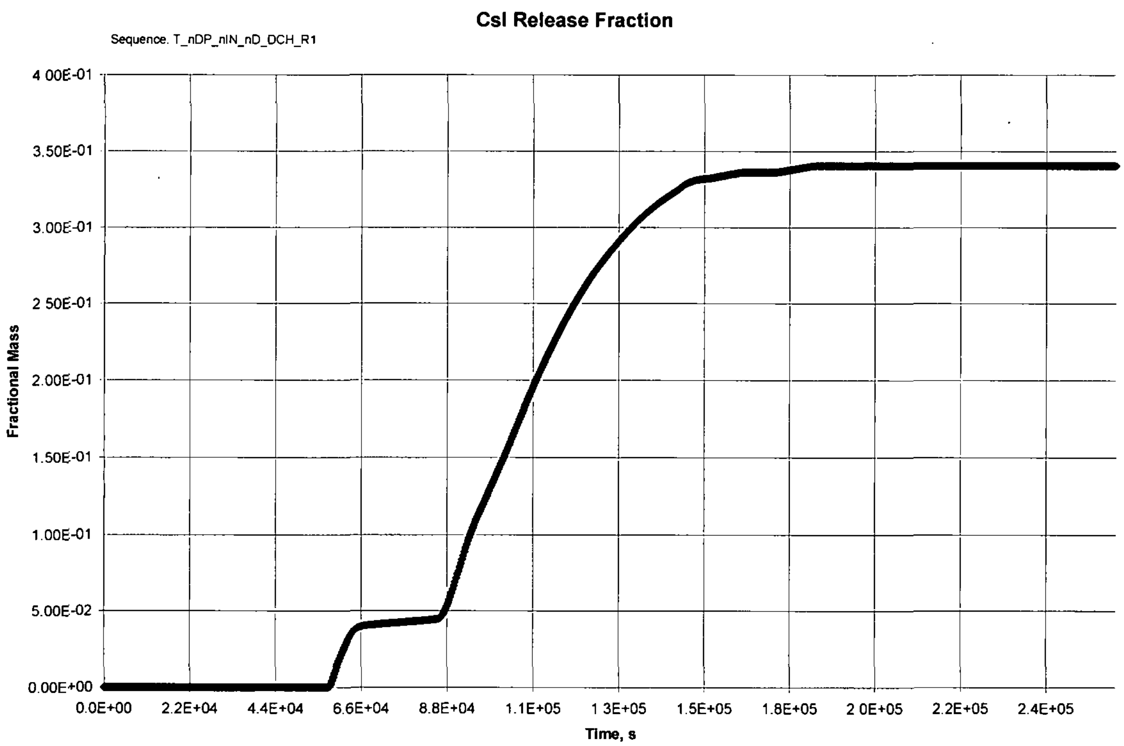


Figure 9A-9l - T_nDP_nIN_nD_DCH_R1 CsI Release Fraction

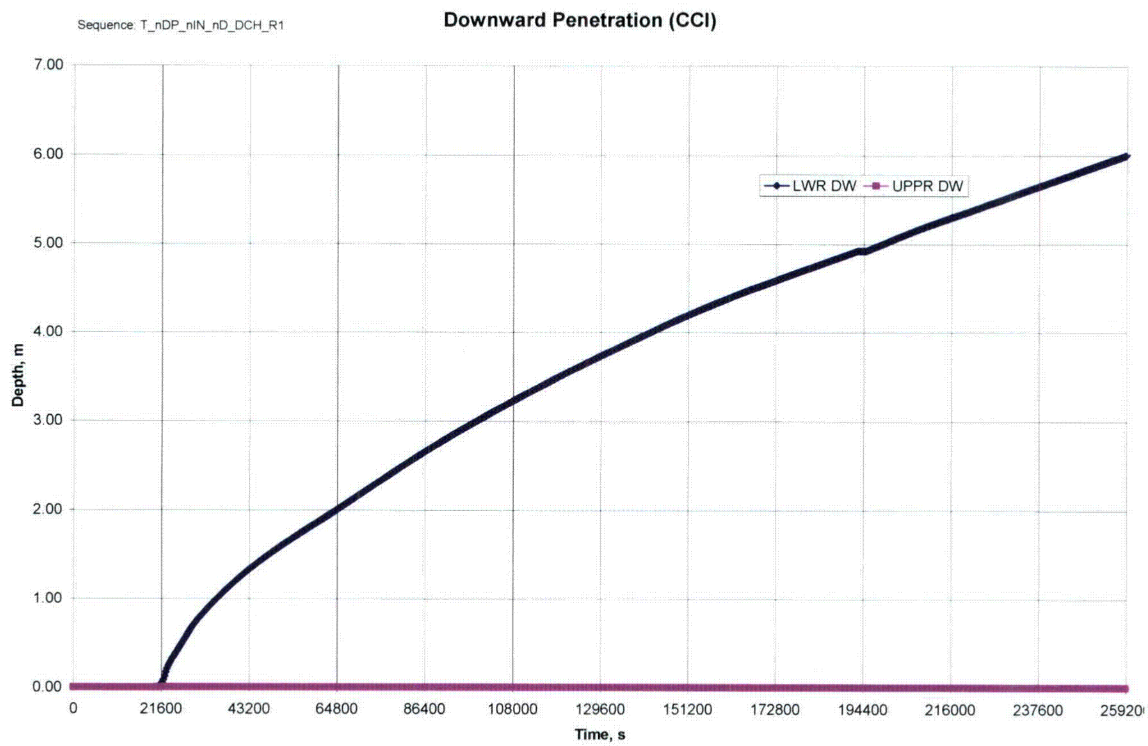


Figure 9A-9m - T_nDP_nIN_nD_DCH_R1 Downward Penetration (CCI)

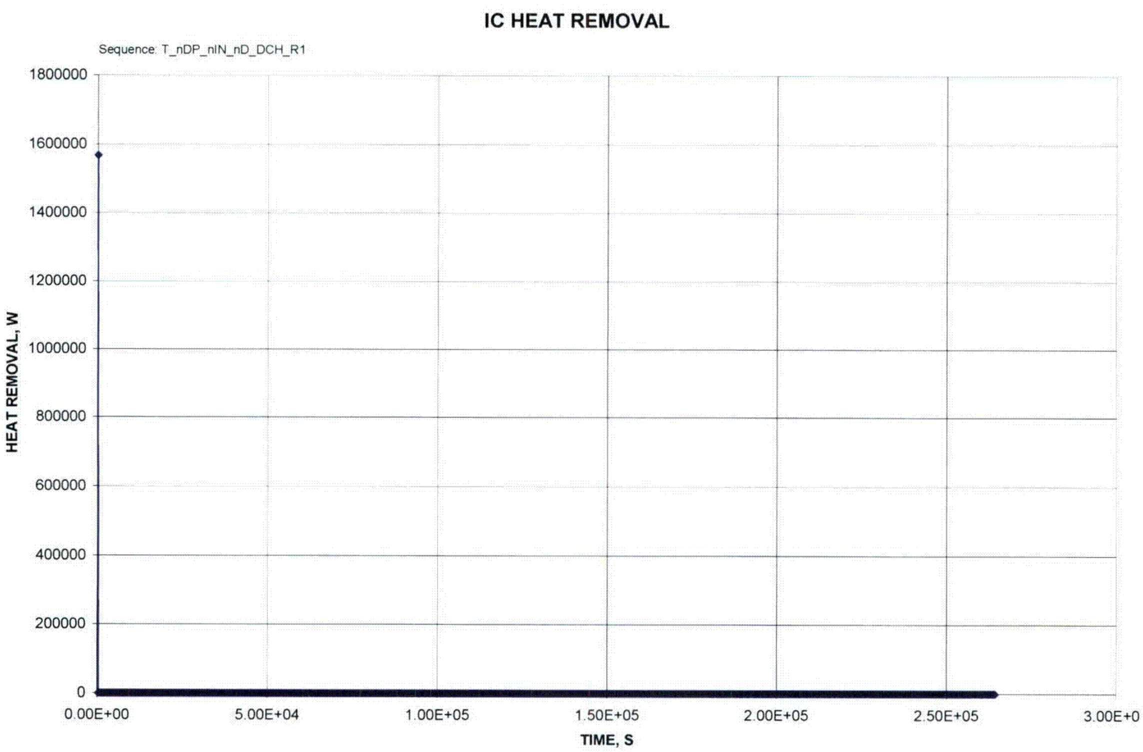


Figure 9A-9n - T_nDP_nIN_nD_DCH_R1 ICS Heat Removal

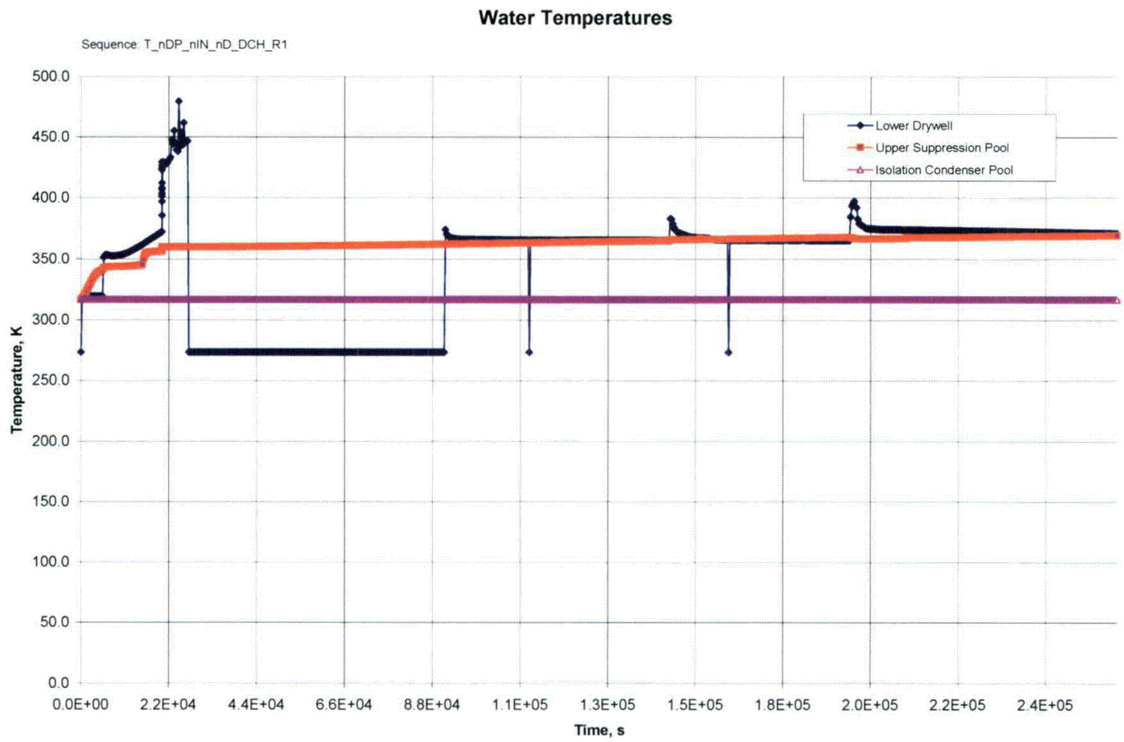


Figure 9A-9o - T_nDP_nIN_nD_DCH_R1 Water Temperatures

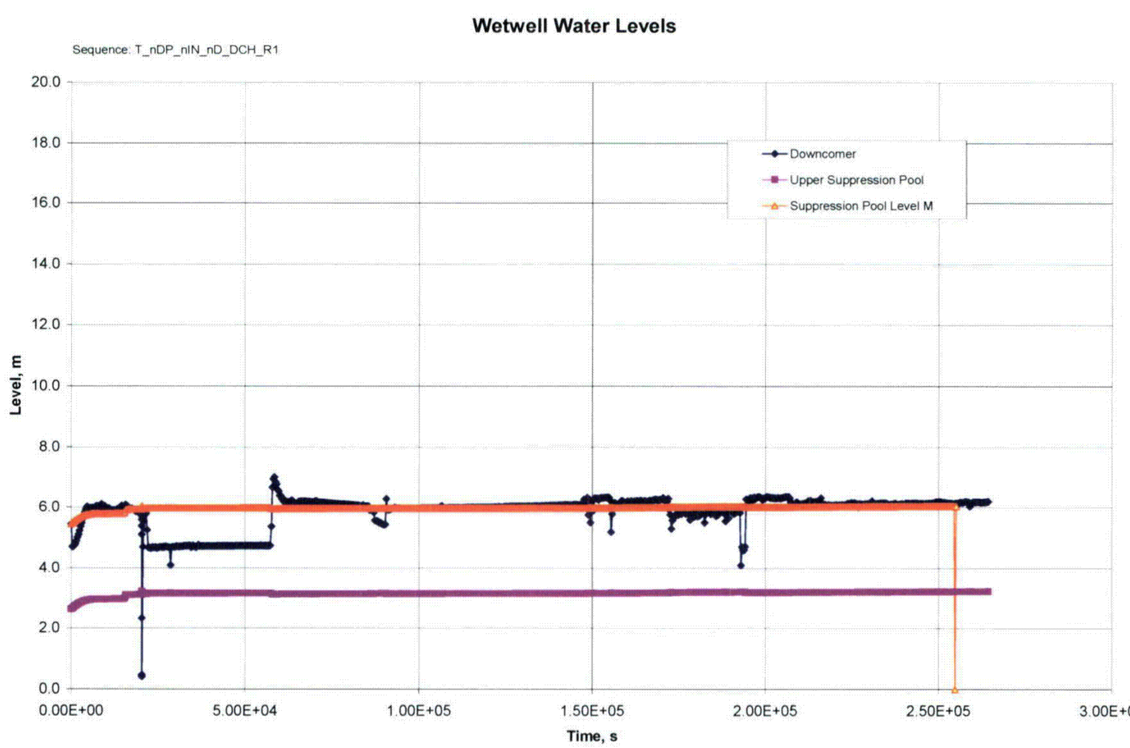


Figure 9A-9p - T_nDP_nIN_nD_DCH_R1 Wetwell Water Levels

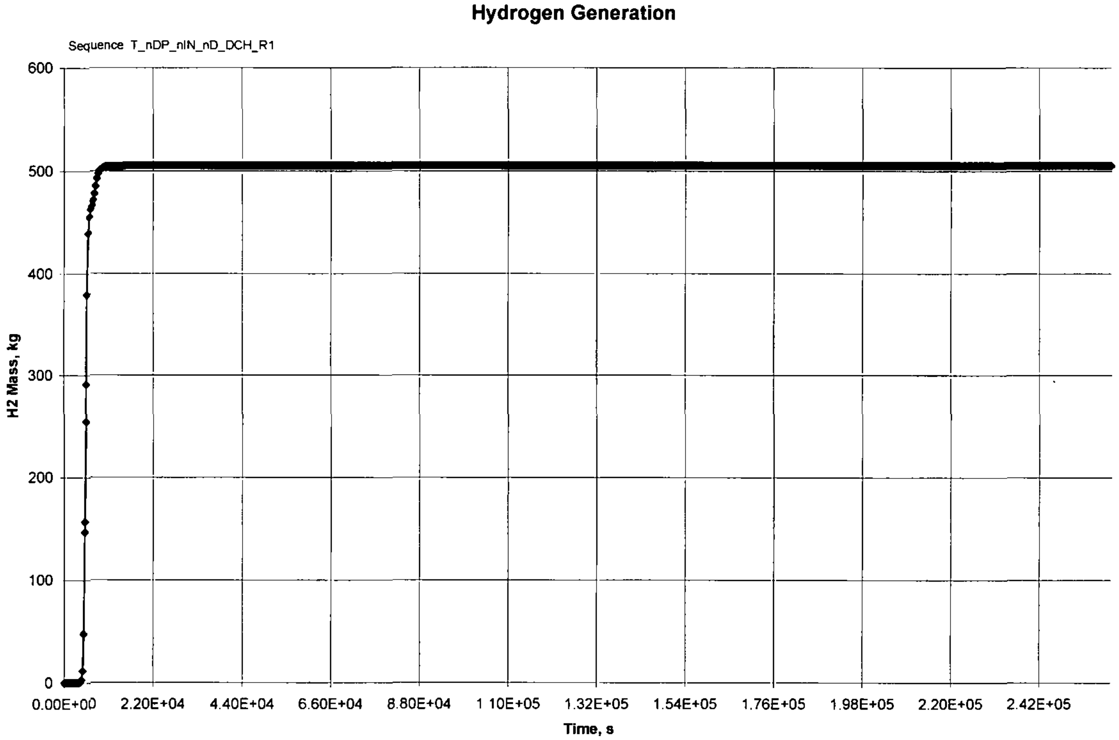


Figure 9A-9q - T_nDP_nIN_nD_DCH_R1 Hydrogen Generation

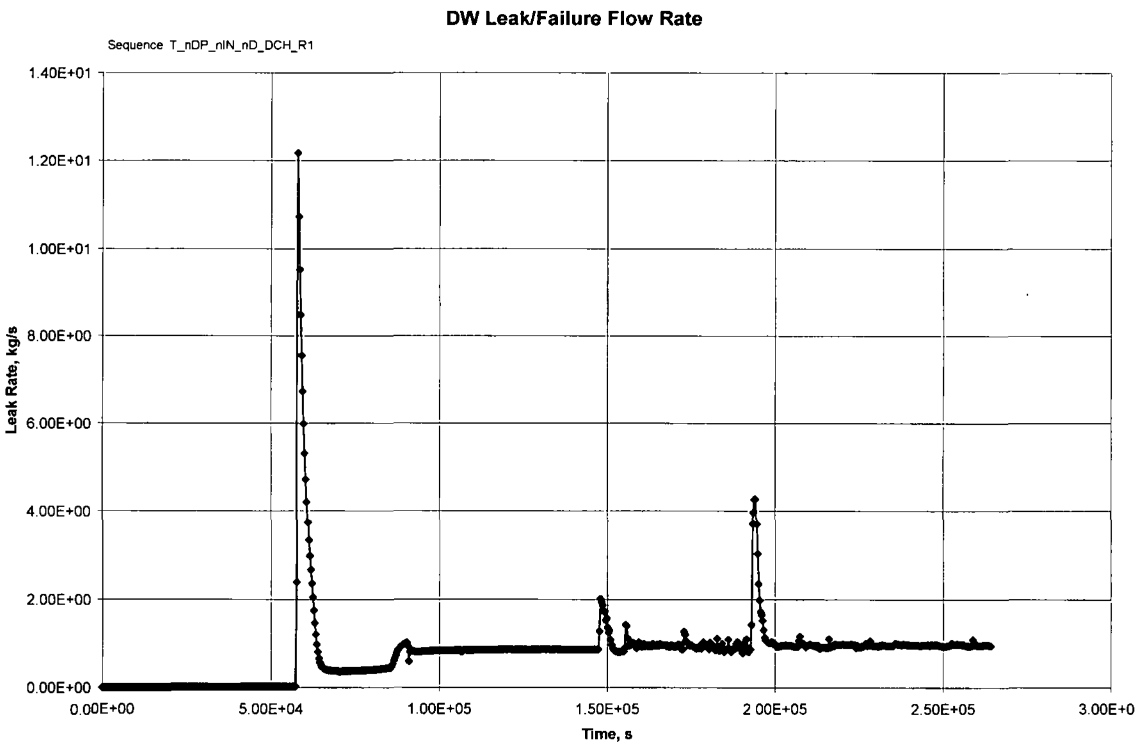


Figure 9A-9r - T_nDP_nIN_nD_DCH_R1 DW Leak/Failure Flow Rate

Figure 9A-9s - T_nDP_nIN_nD_DCH_R1 Wetwell Vent Flow Rate

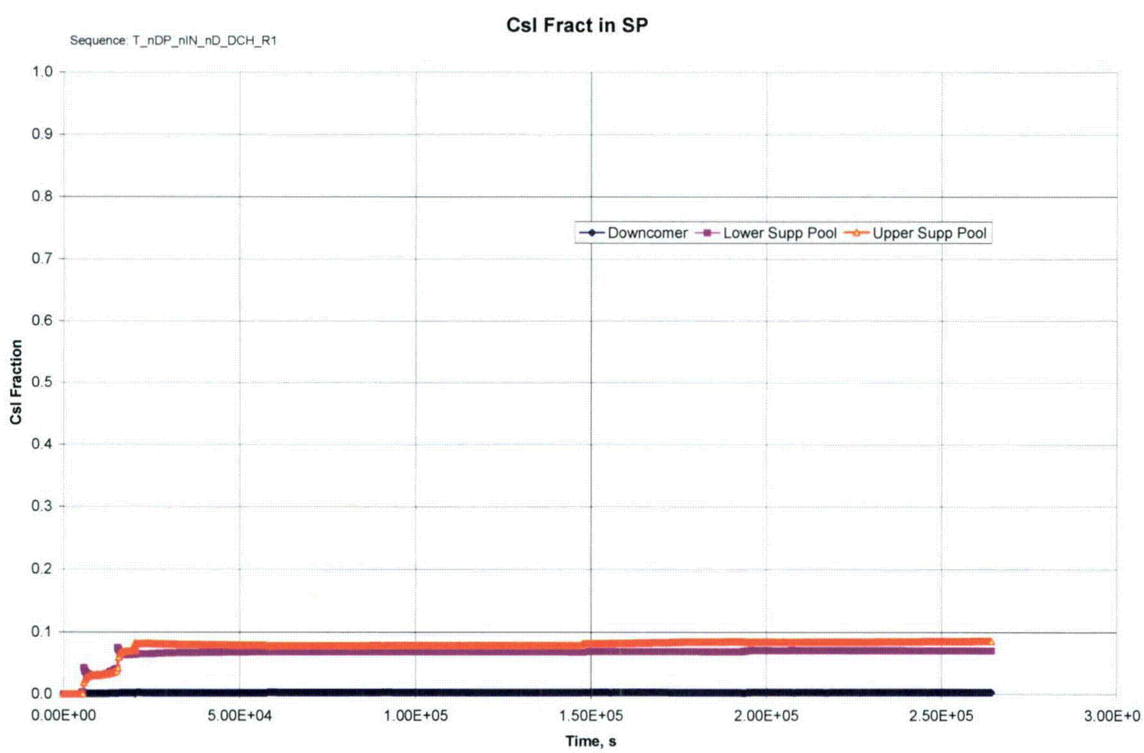


Figure 9A-9t - T_nDP_nIN_nD_DCH_R1 CsL Fraction in SP

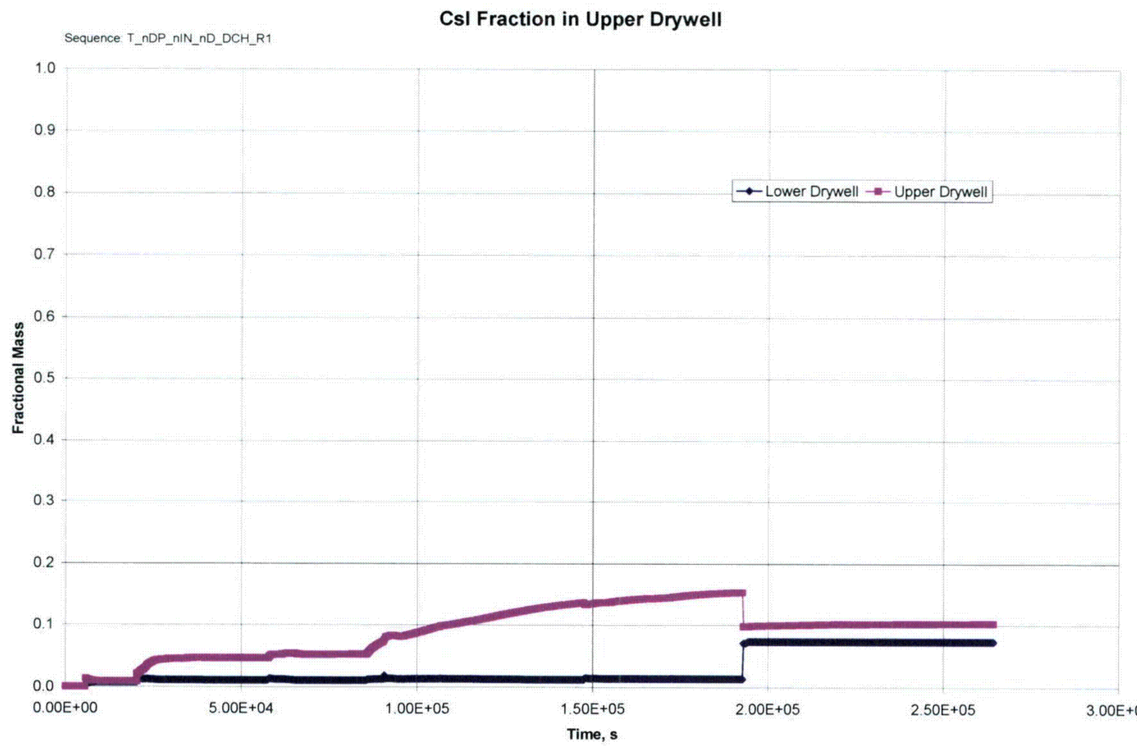


Figure 9A-9u - T_nDP_nIN_nD_DCH_R1 CsI Fraction in Upper Drywell

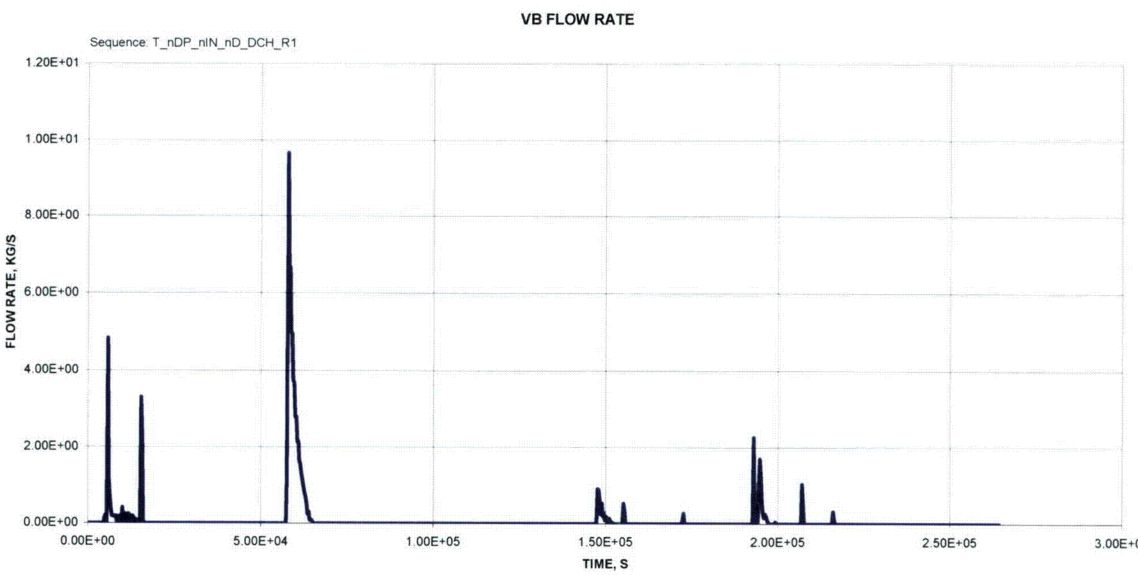


Figure 9A-9v - T_nDP_nIN_nD_DCH_R1 VB Flow Rate

9A.10 Ex-Vessel Explosion T_nIN_nD_EVE_R1

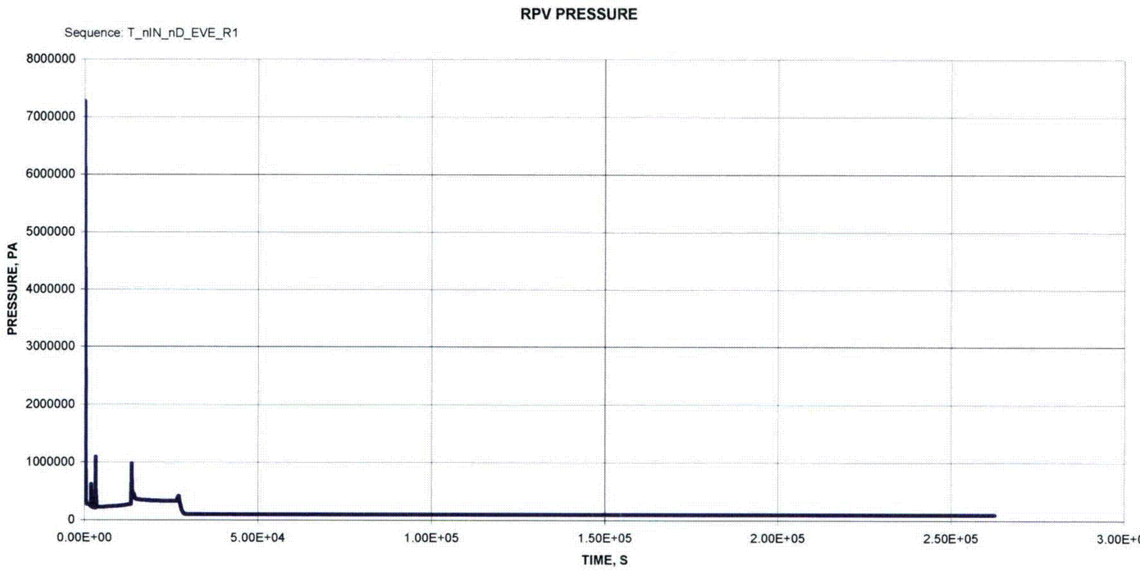


Figure 9A-10a - T_nIN_nD_EVE_R1 RPV Pressure

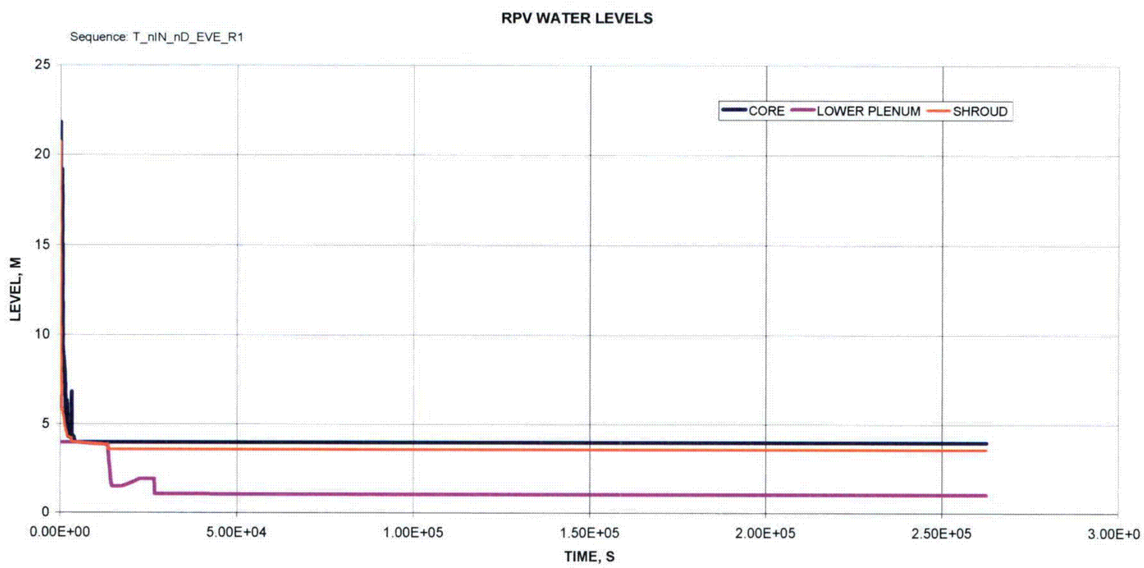


Figure 9A-10b - T_nIN_nD_EVE_R1 RPV Water Levels

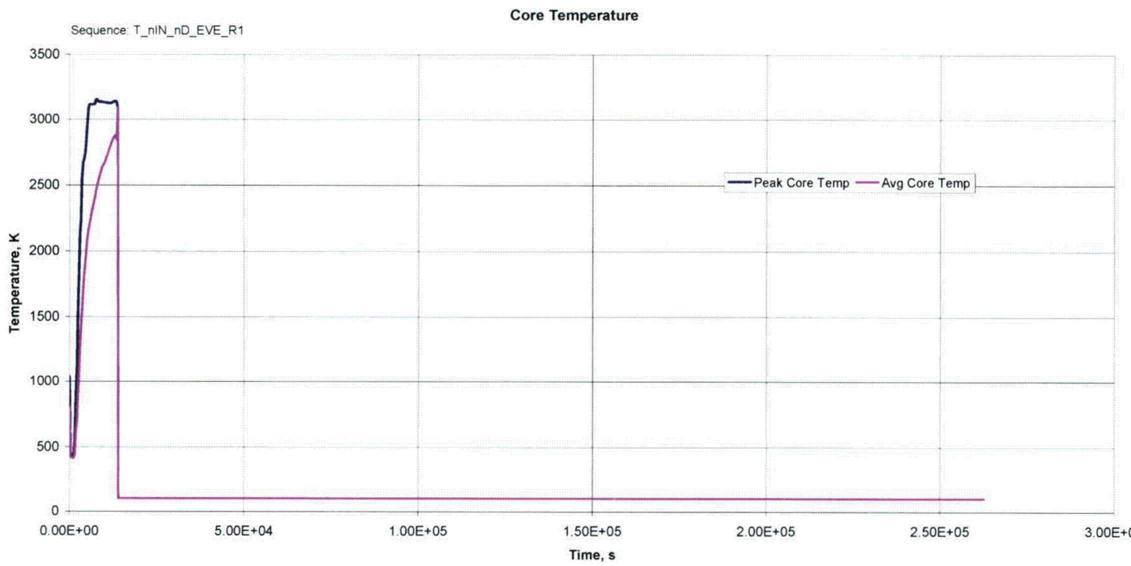


Figure 9A-10c - T_nIN_nD_EVE_R1 RPV Core Temperature

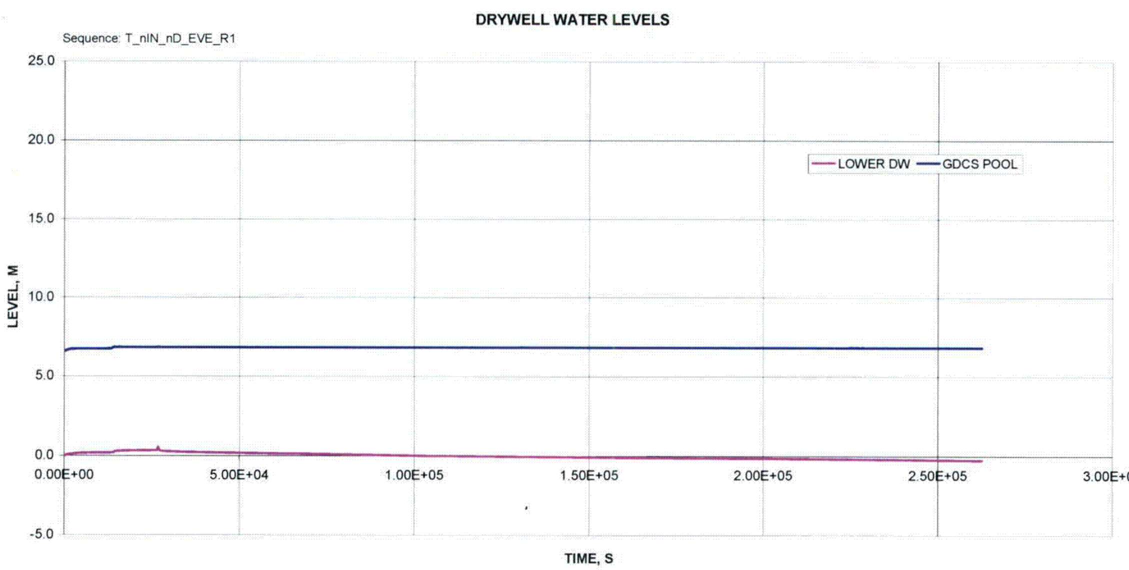


Figure 9A-10d - T_nIN_nD_EVE_R1 RPV Drywell Water Levels

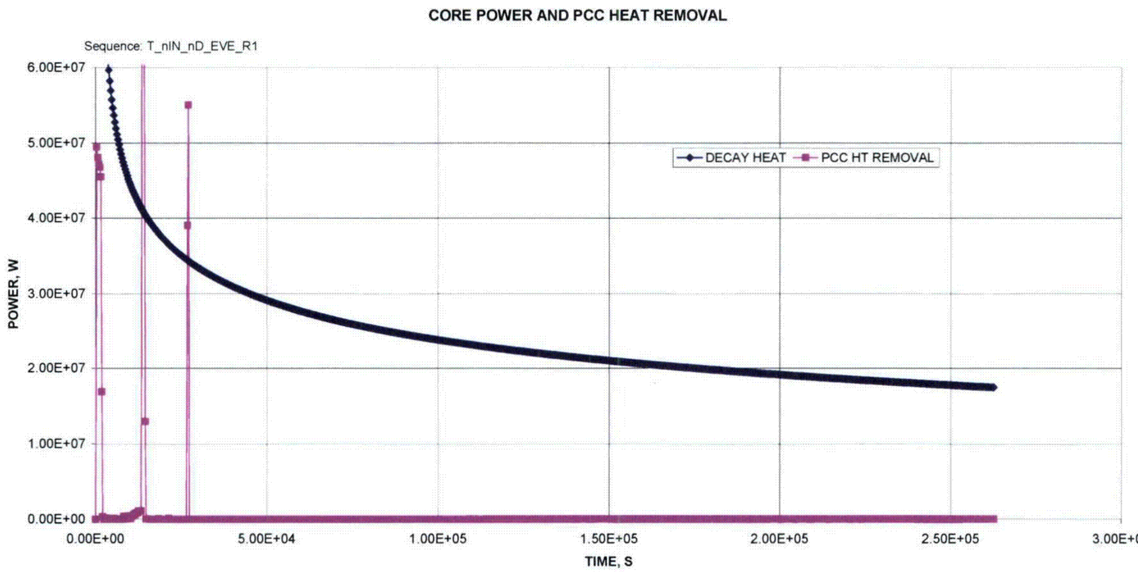


Figure 9A-10e - T_nIN_nD_EVE_R1 RPV Core Power and PCCS Heat Removal

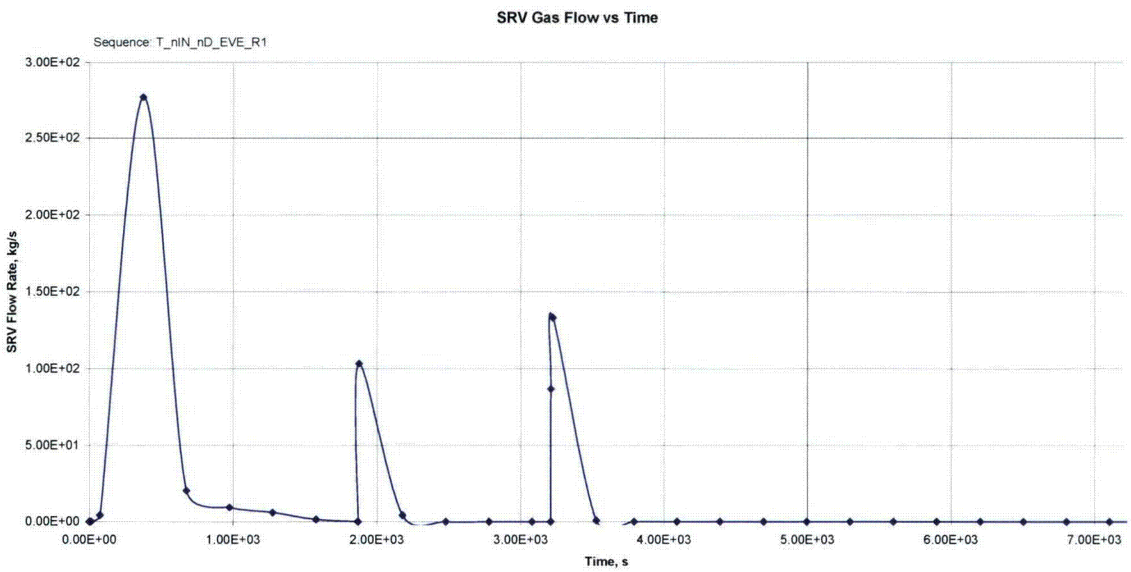


Figure 9A-10f - T_nIN_nD_EVE_R1 RPV SRV Gas Row vs. Time

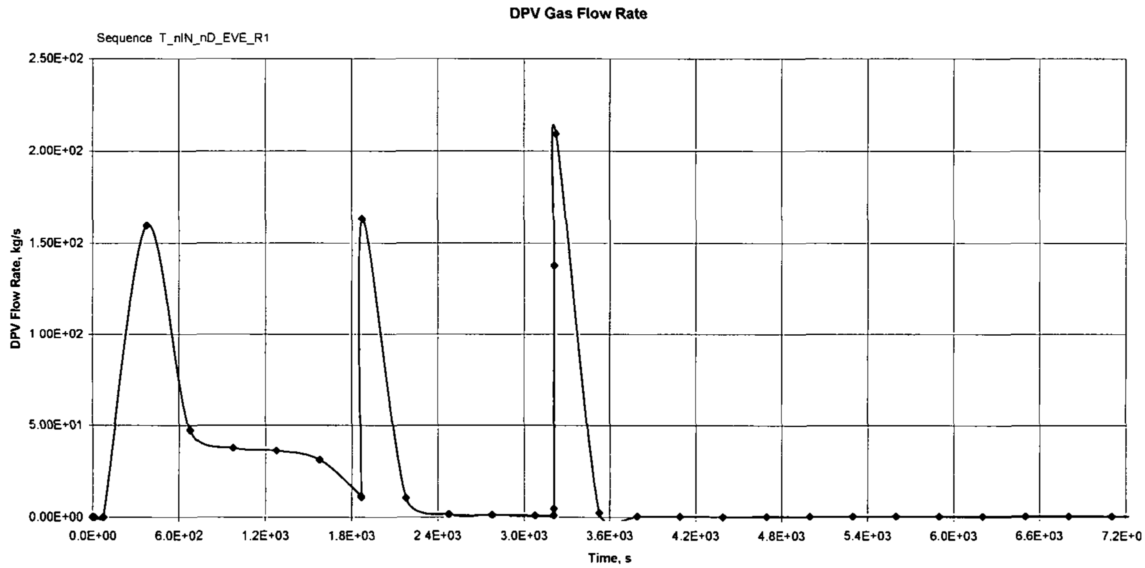


Figure 9A-10G - T_nIN_nD_EVE_R1 RPV DPV Gas Flow Rate

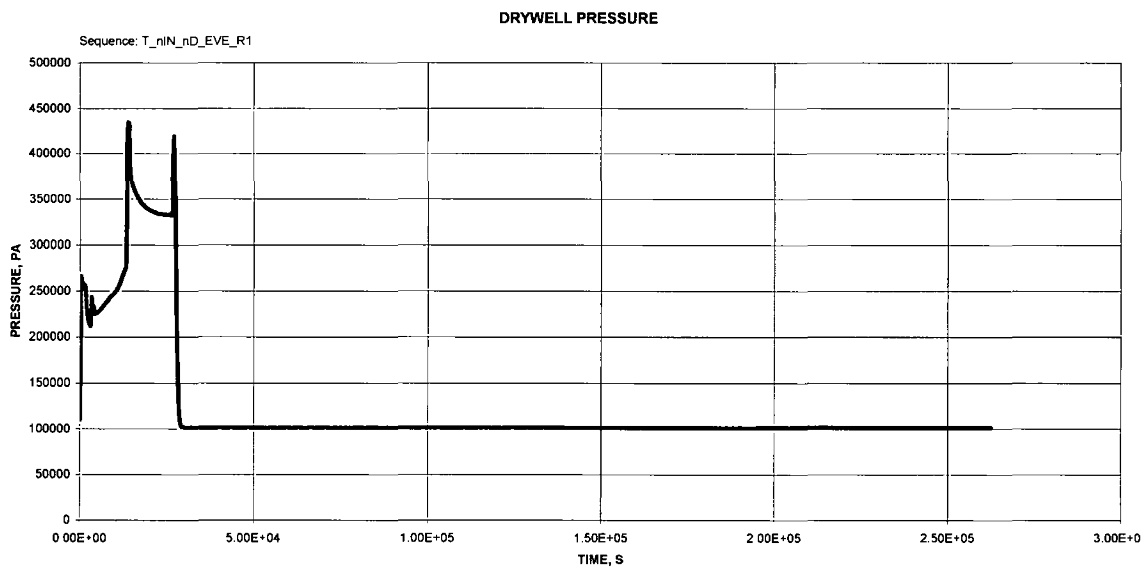


Figure 9A-10h - T_nIN_nD_EVE_R1 RPV Drywell Pressure

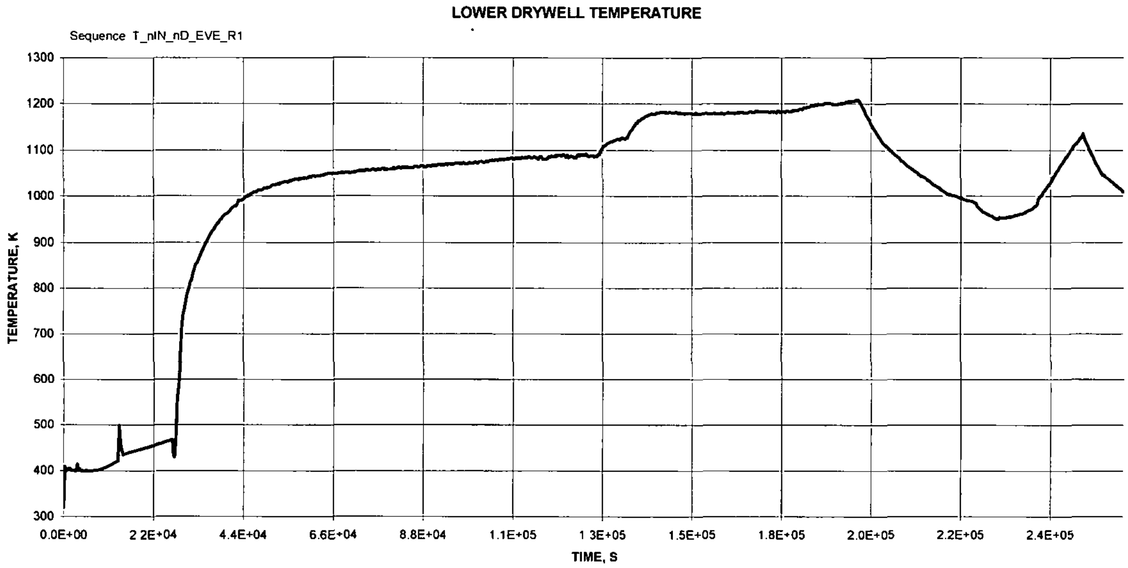


Figure 9A-10i - T_nIN_nD_EVE_R1 RPV Lower Drywell Temperature

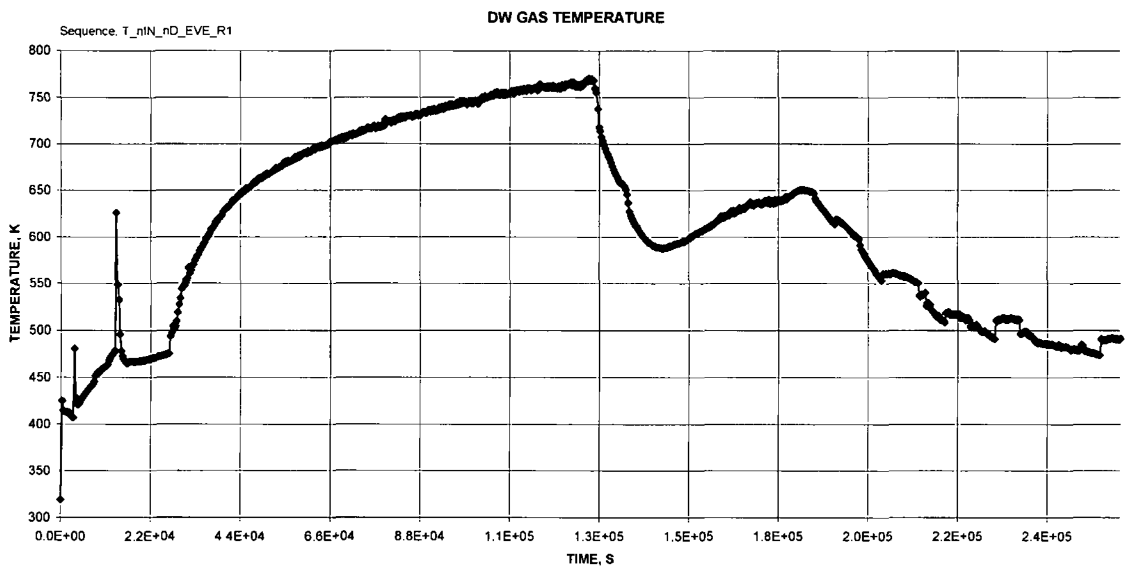


Figure 9A-10h - T_nIN_nD_EVE_R1 RPV DW Gas Temperature

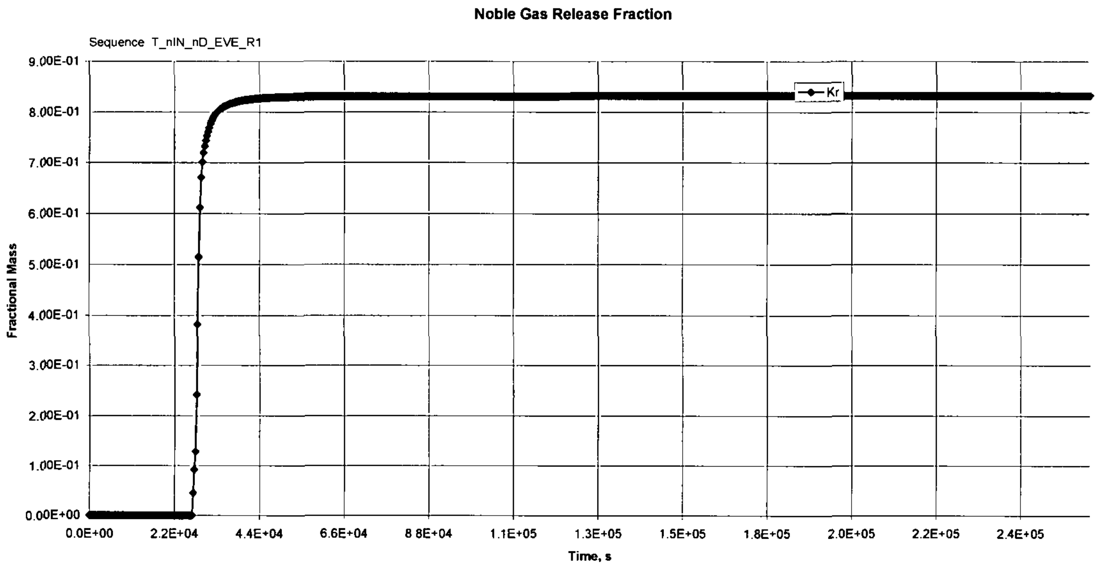


Figure 9A-10k - T_nIN_nD_EVE_R1 RPV Noble Gas Release Fraction

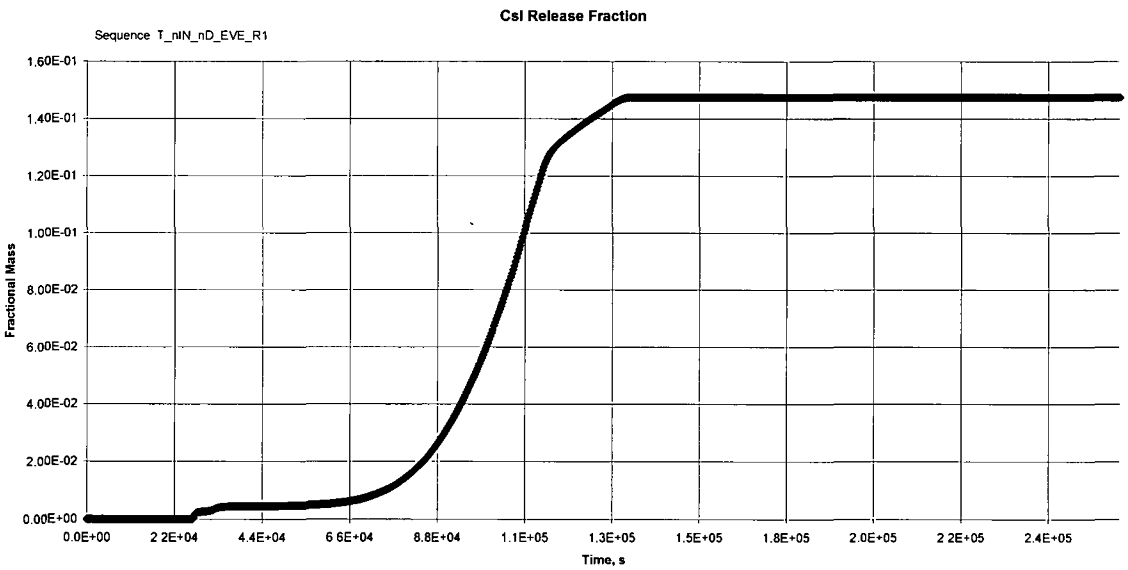


Figure 9A-10l - T_nIN_nD_EVE_R1 RPV CsI Release Fraction

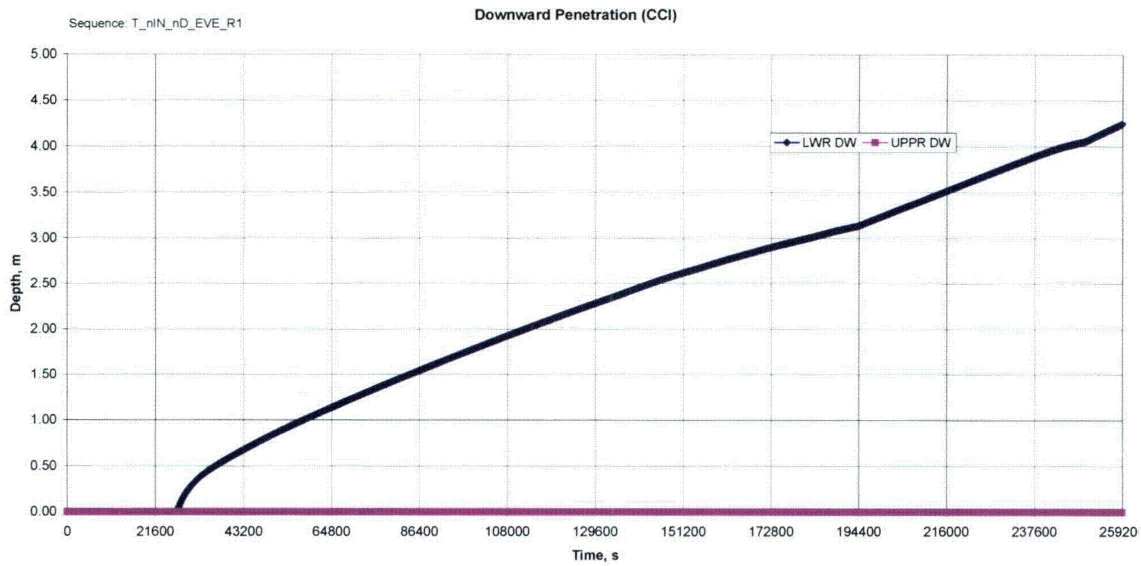


Figure 9A-10m - T_nIN_nD_EVE_R1 RPV Downward Penetration (CCI)

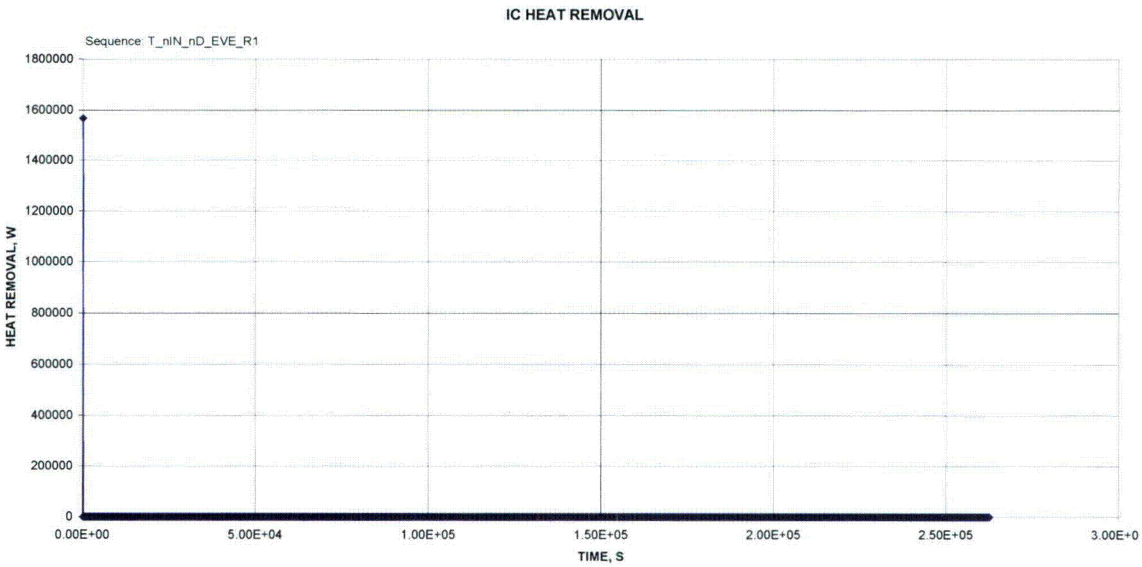


Figure 9A-10n - T_nIN_nD_EVE_R1 RPV ICS Heat Removal

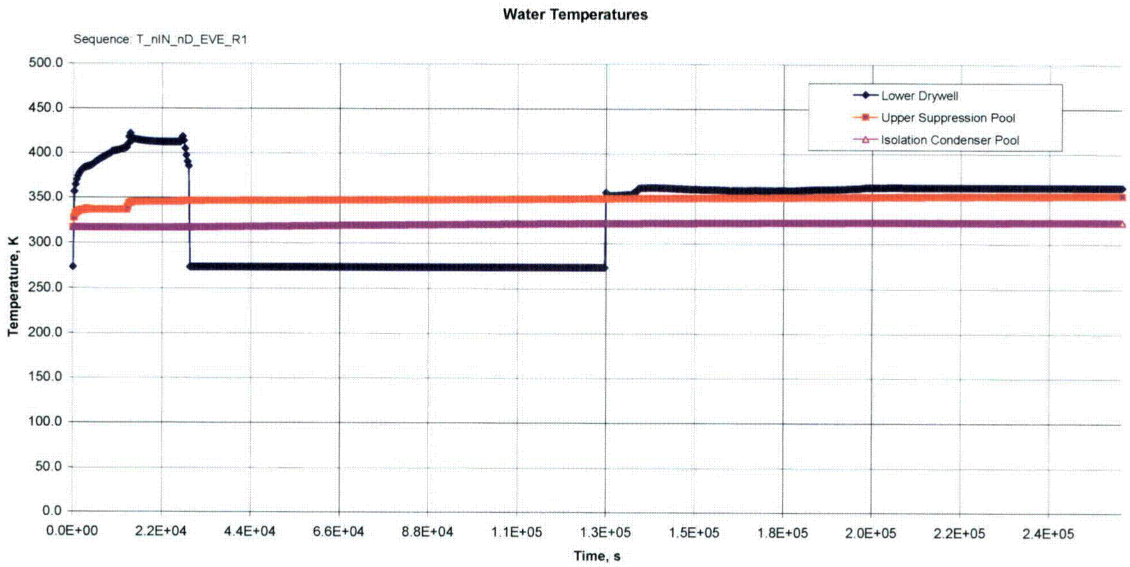


Figure 9A-10o - T_nIN_nD_EVE_R1 RPV Water Temperatures

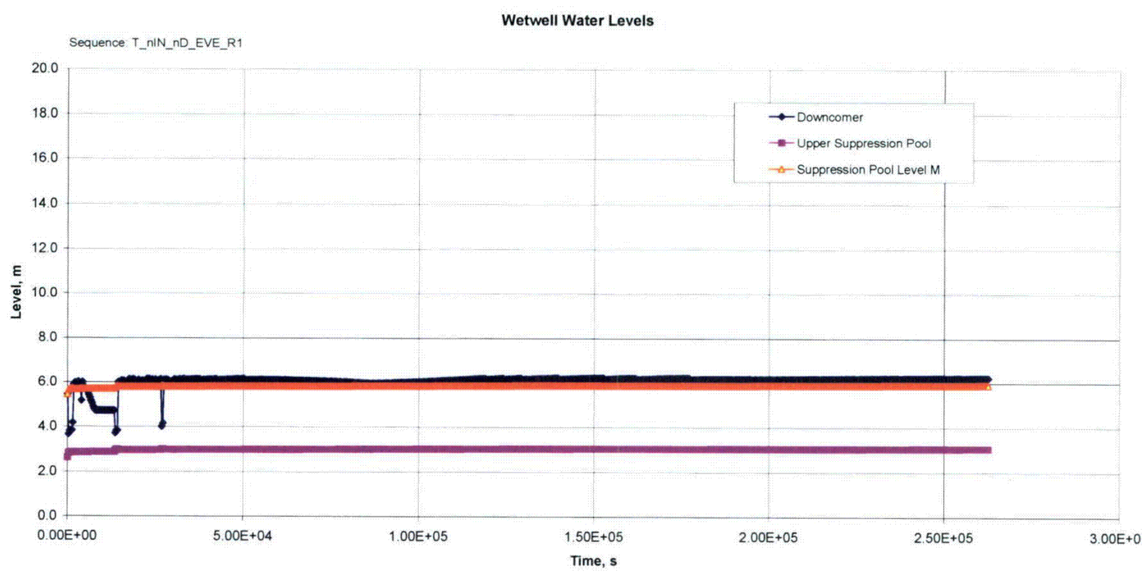


Figure 9A-10p - T_nIN_nD_EVE_R1 RPV Wetwell Water Levels

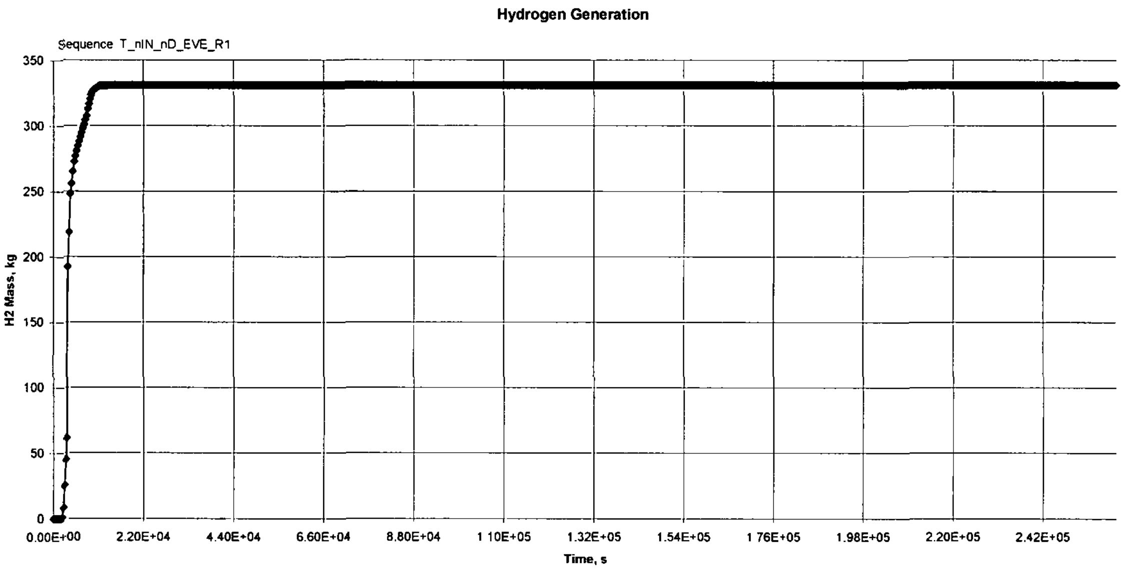


Figure 9A-10aq - T_nIN_nD_EVE_R1 RPV Hydrogen Generation

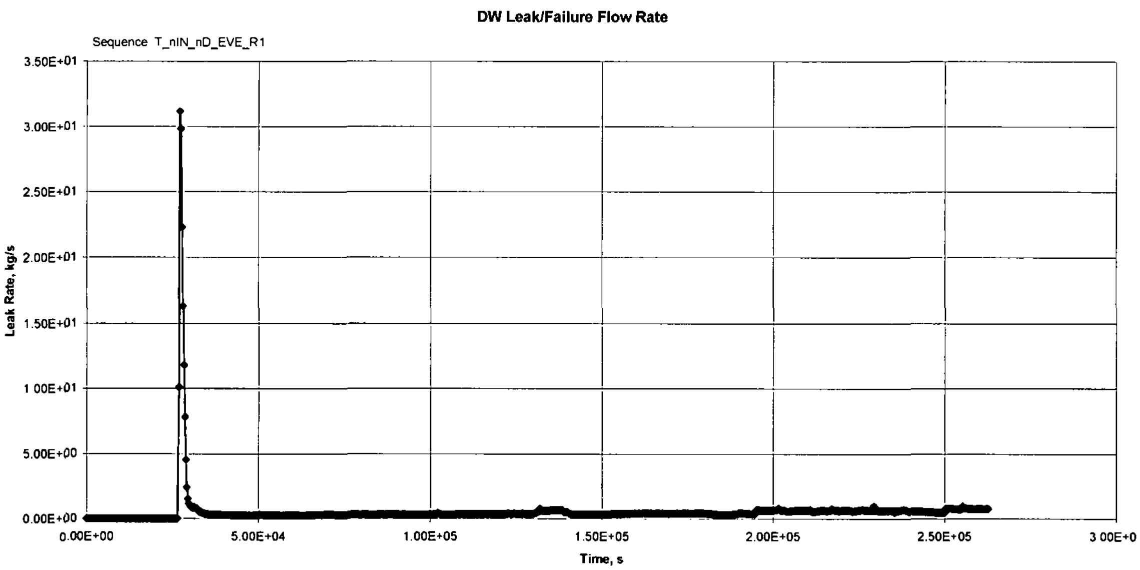


Figure 9A-10r - T_nIN_nD_EVE_R1 RPV DW Leak/Failure Flow Rate

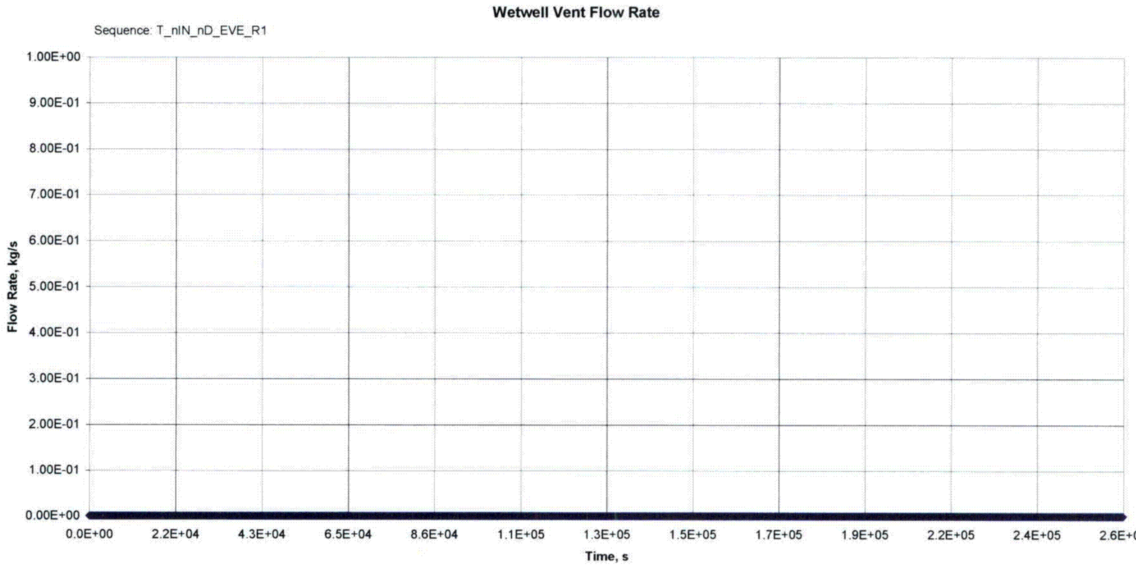


Figure 9A-10s - T_nIN_nD_EVE_R1 RPV Wetwell Vent Flow Rate

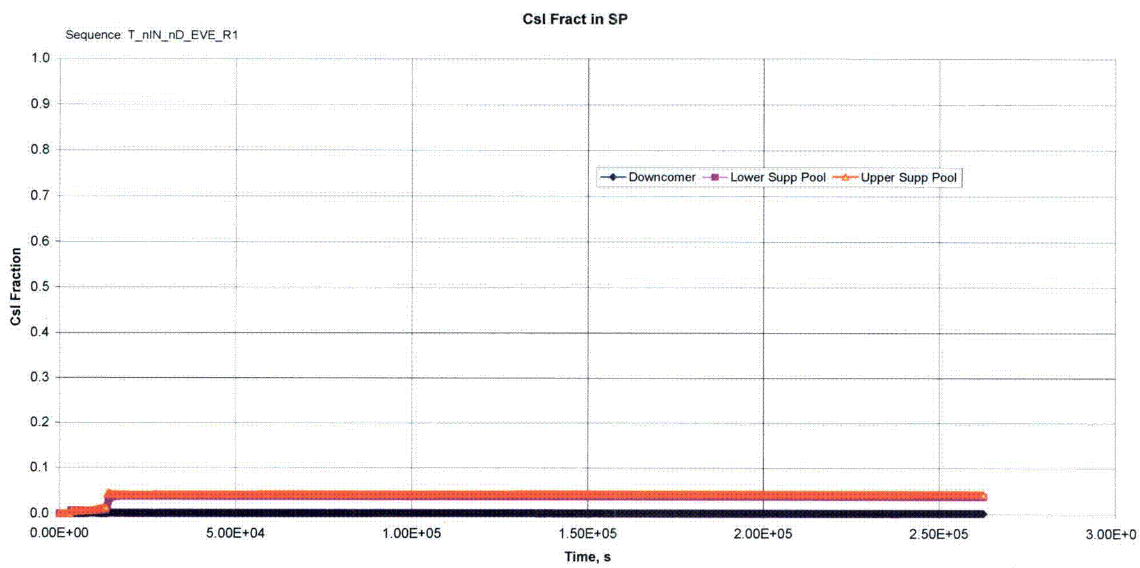


Figure 9A-10t - T_nIN_nD_EVE_R1 RPV CsI Fraction in SP

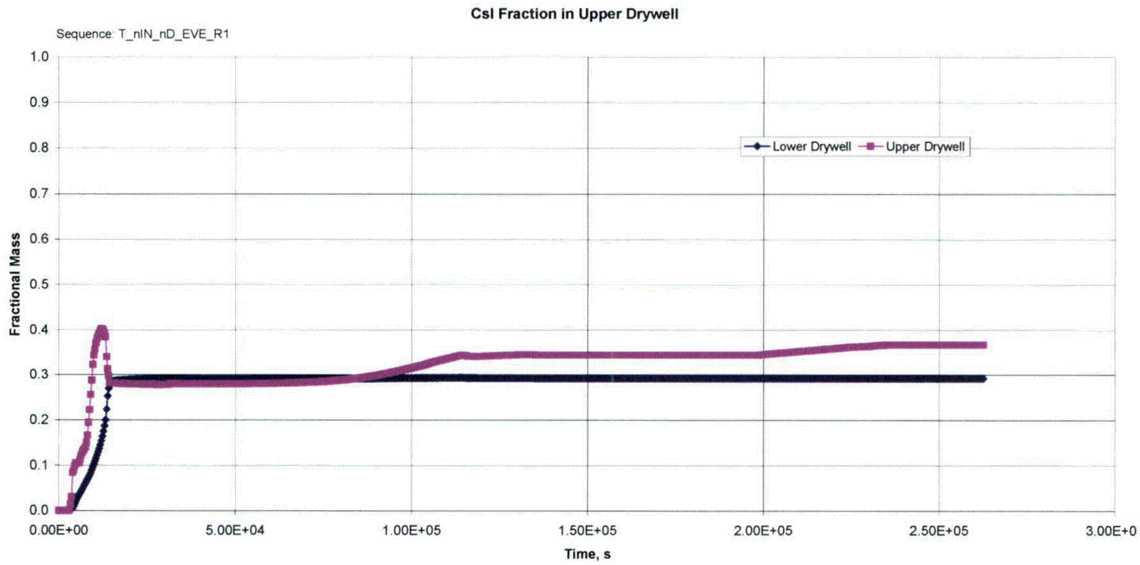


Figure 9A-10u - T_nIN_nD_EVE_R1 RPV CsI Fraction in Upper Drywell

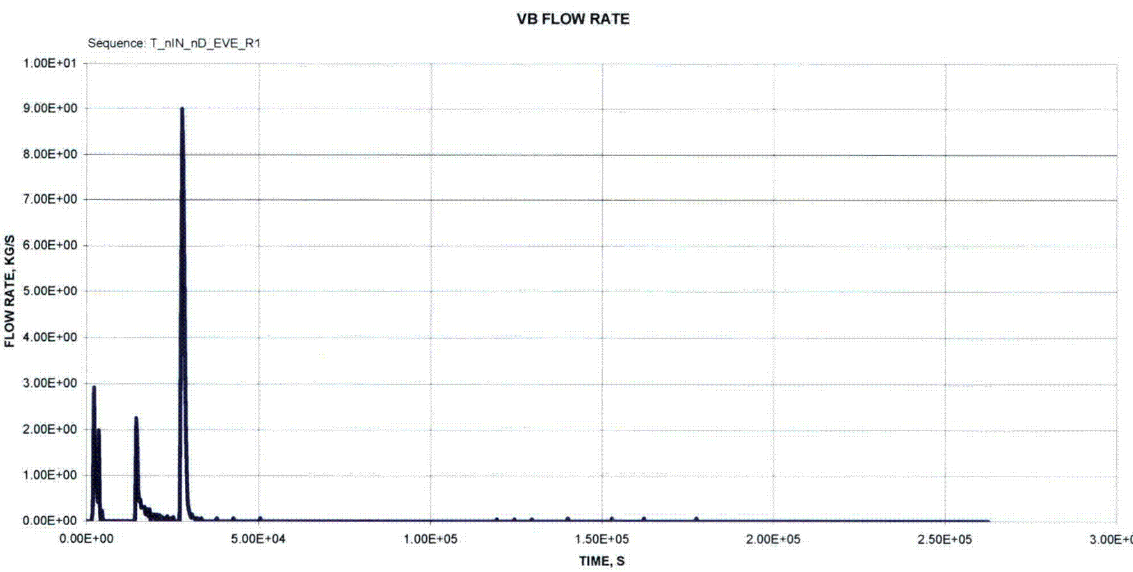


Figure 9A-10v - T_nIN_nD_EVE_R1 RPV VB Flow Rate

9A.11 Filtered Release T-AT_nIN_nCHR_FR_R1

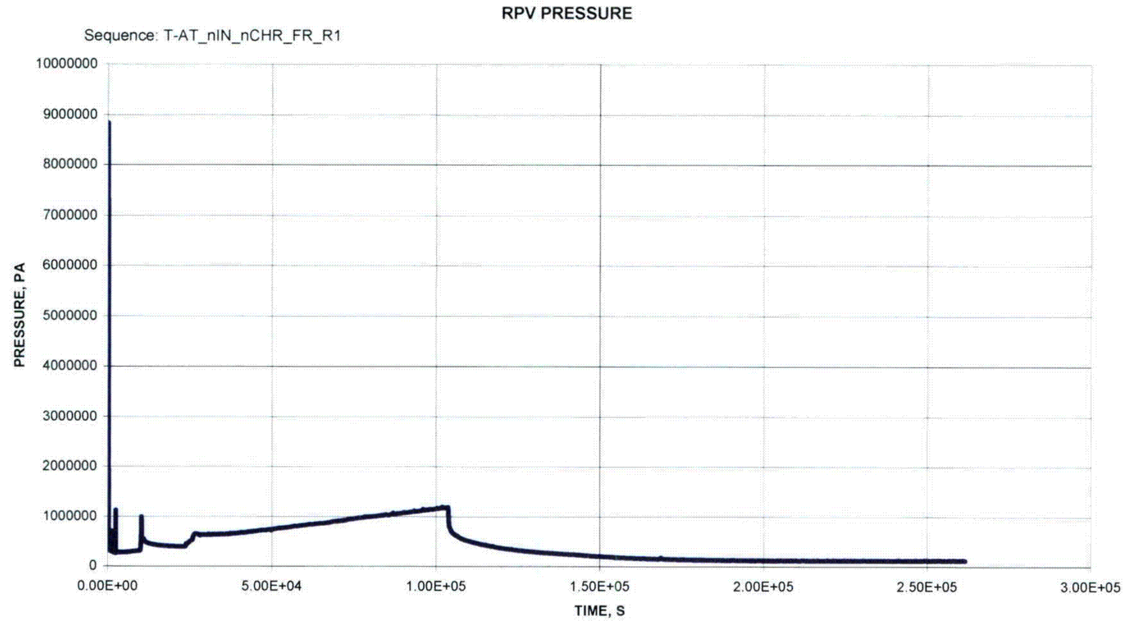


Figure 9A-11a - T-AT_nIN_nCHR_FR_R1 RPV Pressure

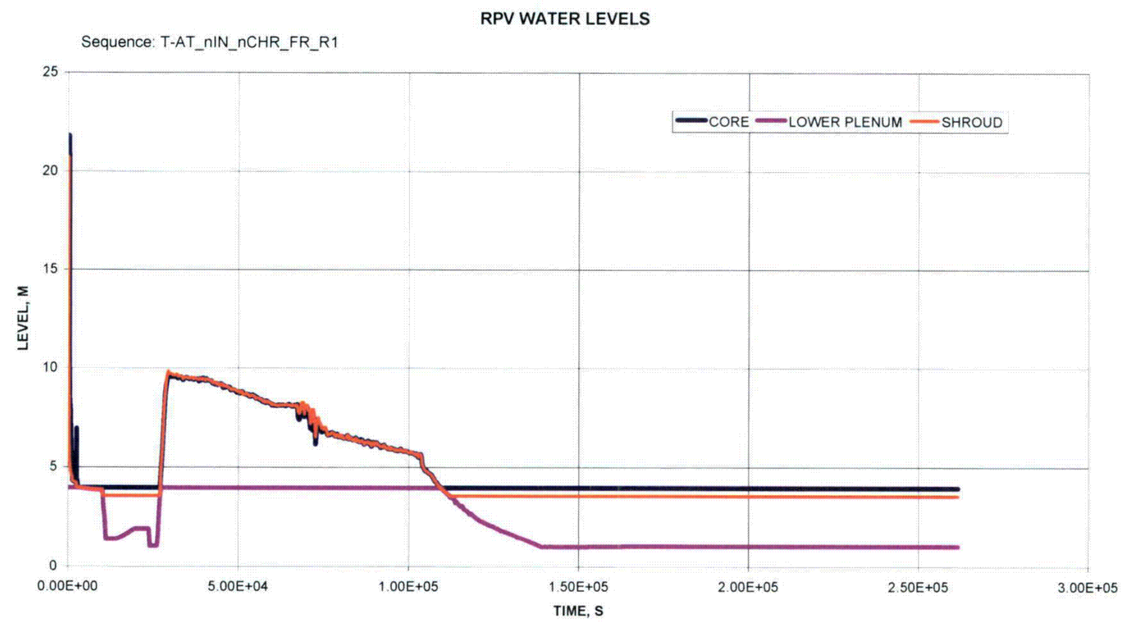


Figure 9A-11a - T-AT_nIN_nCHR_FR_R1 RPV Water Levels

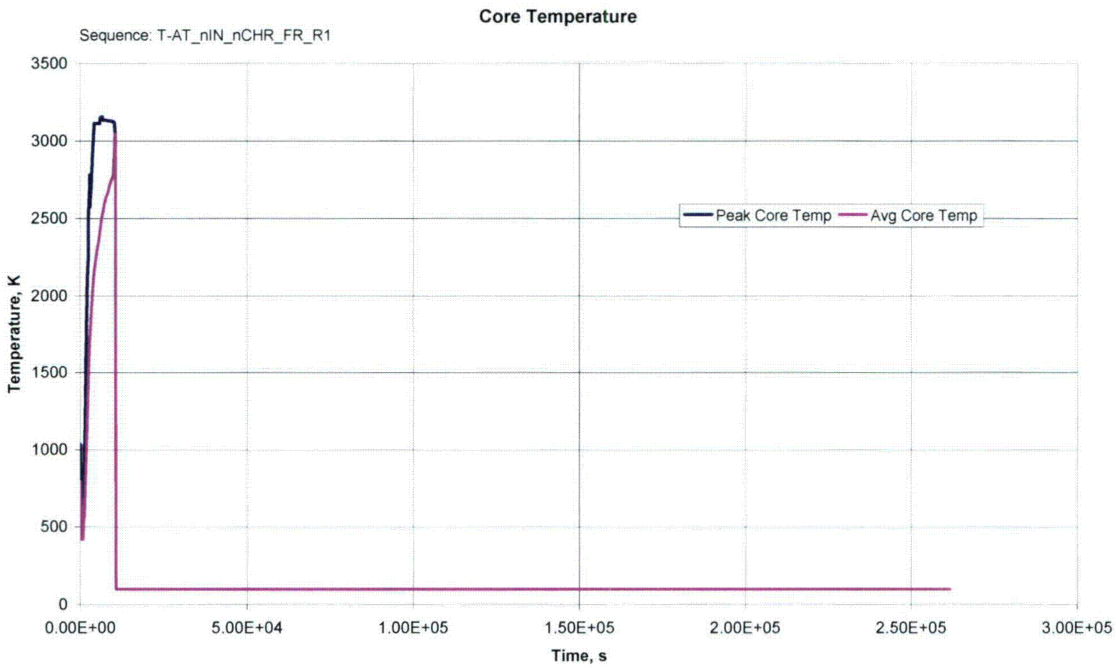


Figure 9A-11c - T-AT_nIN_nCHR_FR_R1 Core Temperature

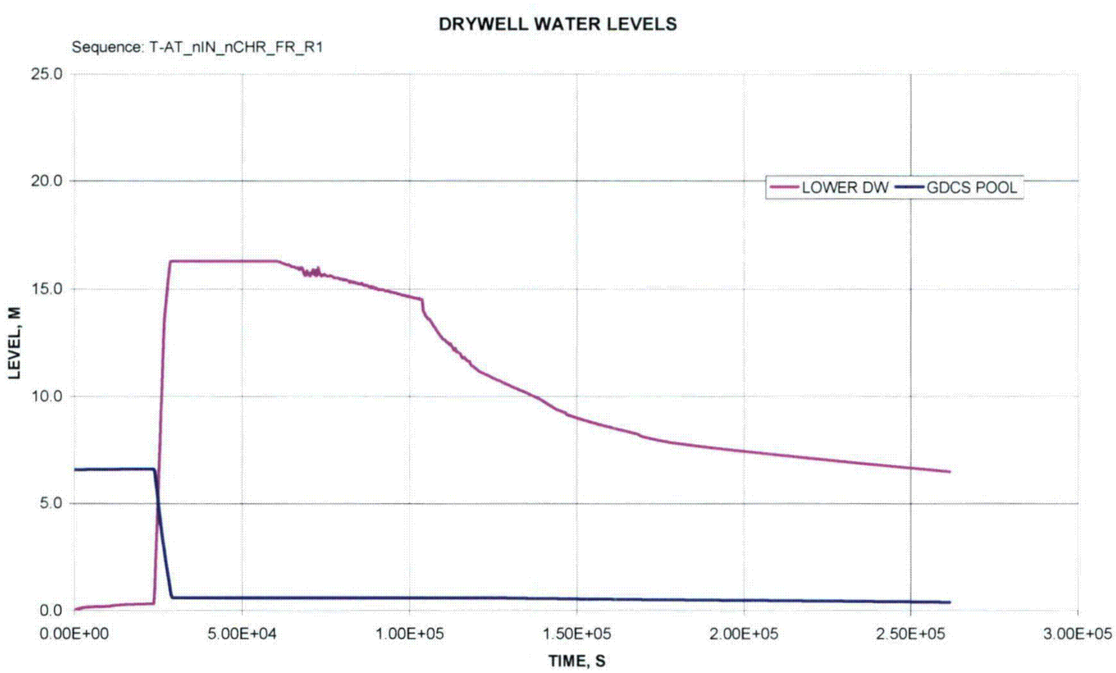


Figure 9A-11d - T-AT_nIN_nCHR_FR_R1 Drywell Water Levels

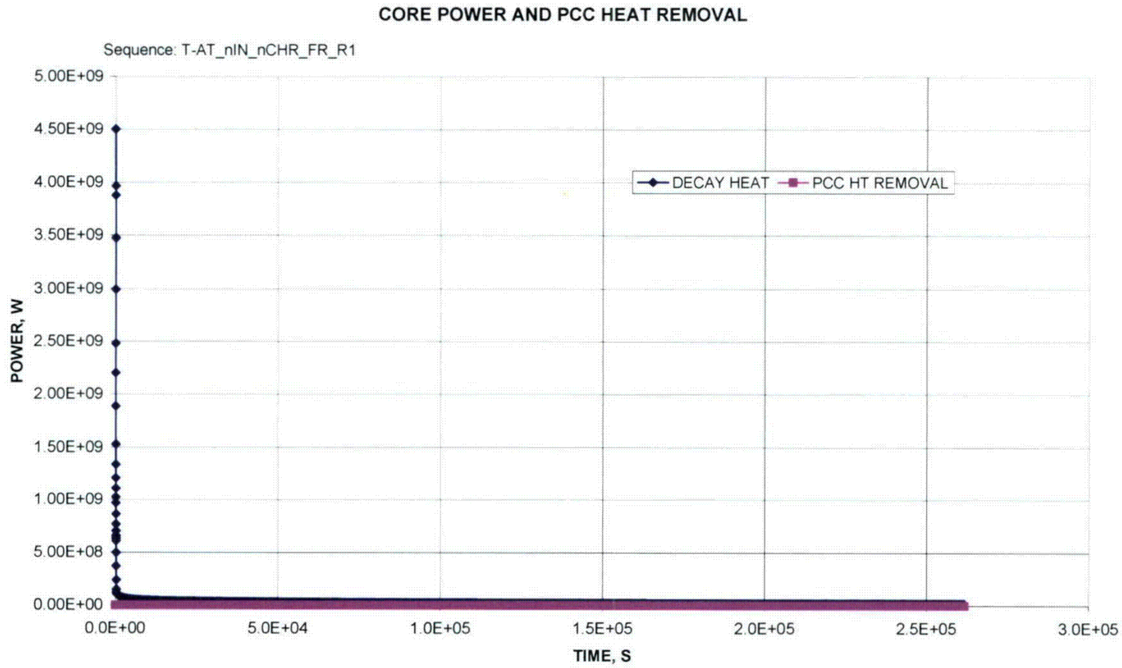


Figure 9A-11e - T-AT_nIN_nCHR_FR_R1 Core Power and PCCS Heat Removal

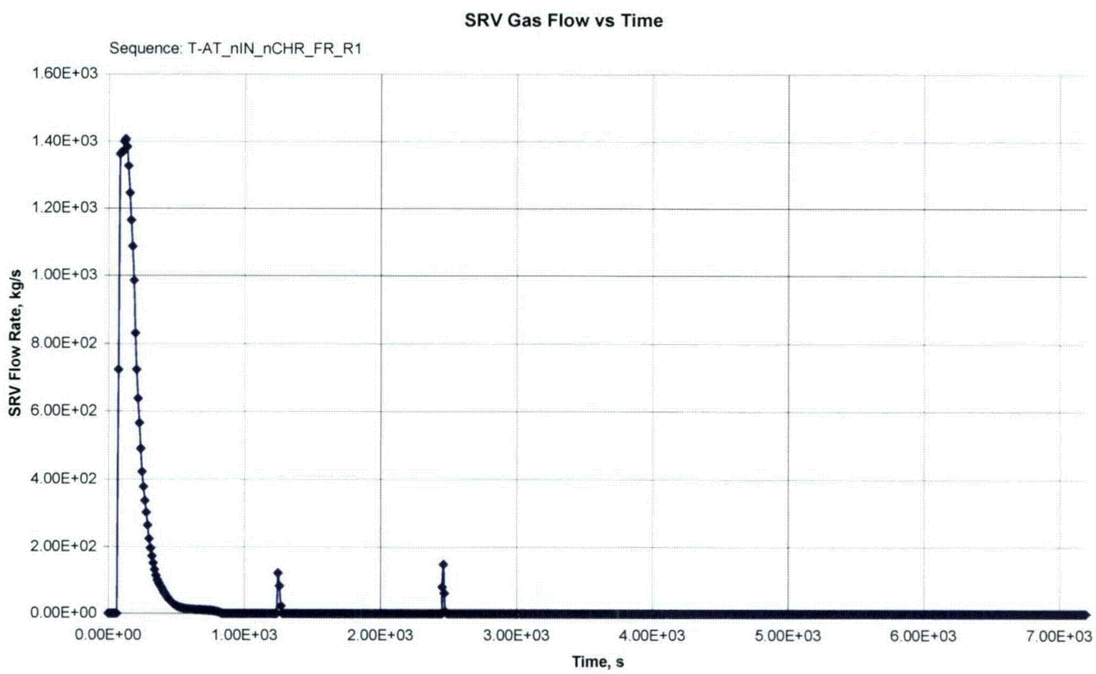


Figure 9A-11f - T-AT_nIN_nCHR_FR_R1 SRV Gas Flow vs. Time

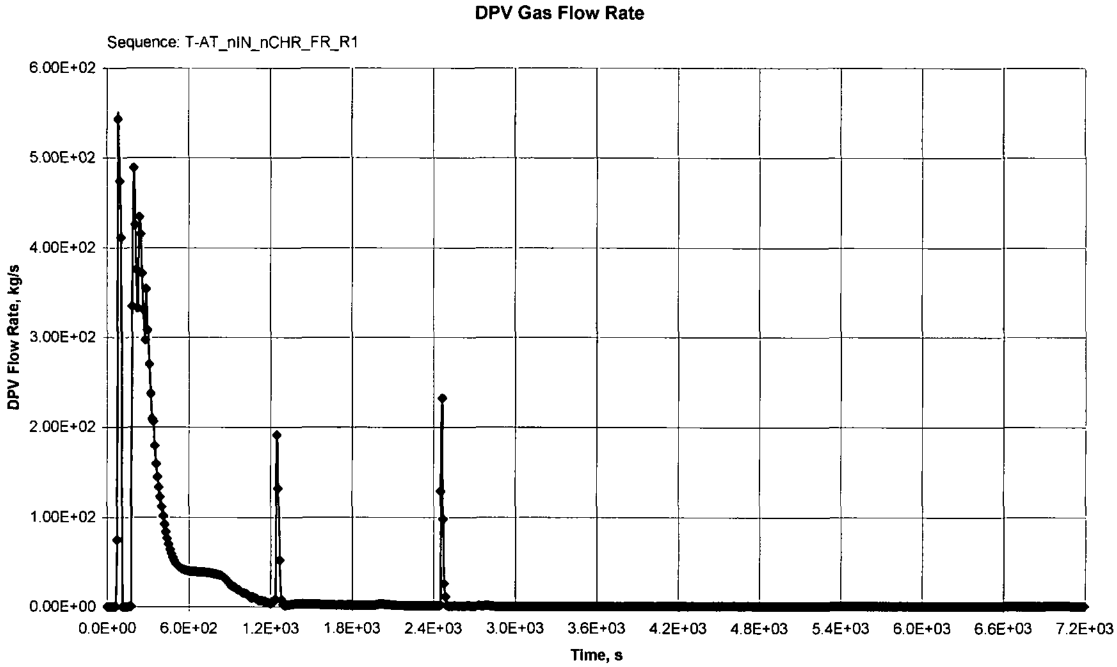


Figure 9A-11g - T-AT_nIN_nCHR_FR_R1 DPV Gas Flow Rate

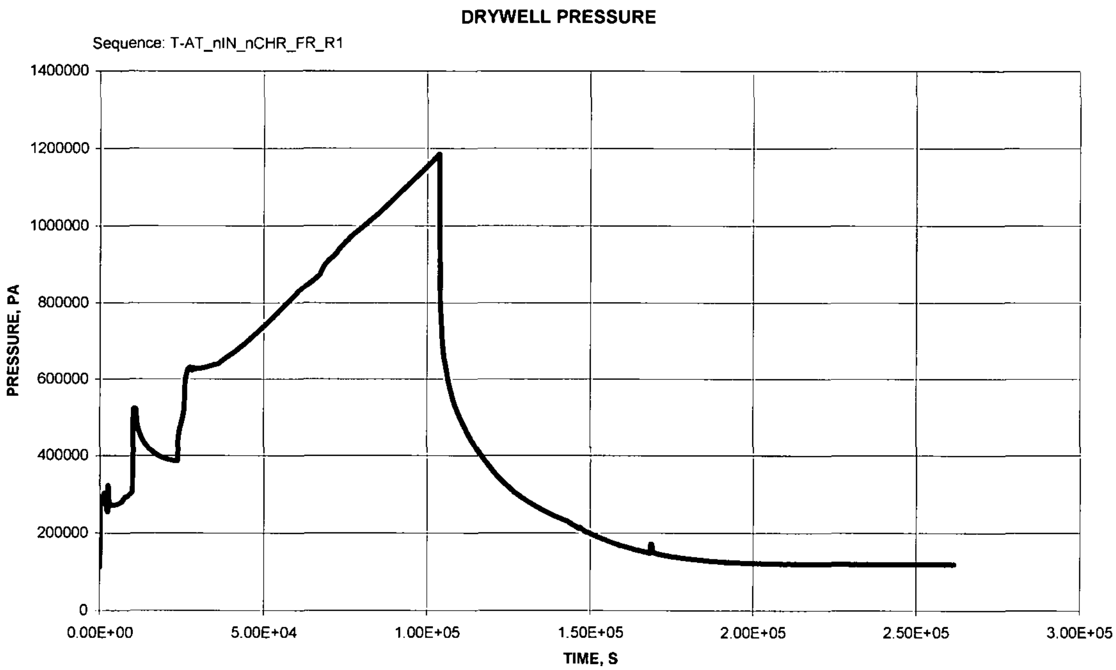


Figure 9A-11h - T-AT_nIN_nCHR_FR_R1 Drywell Pressure

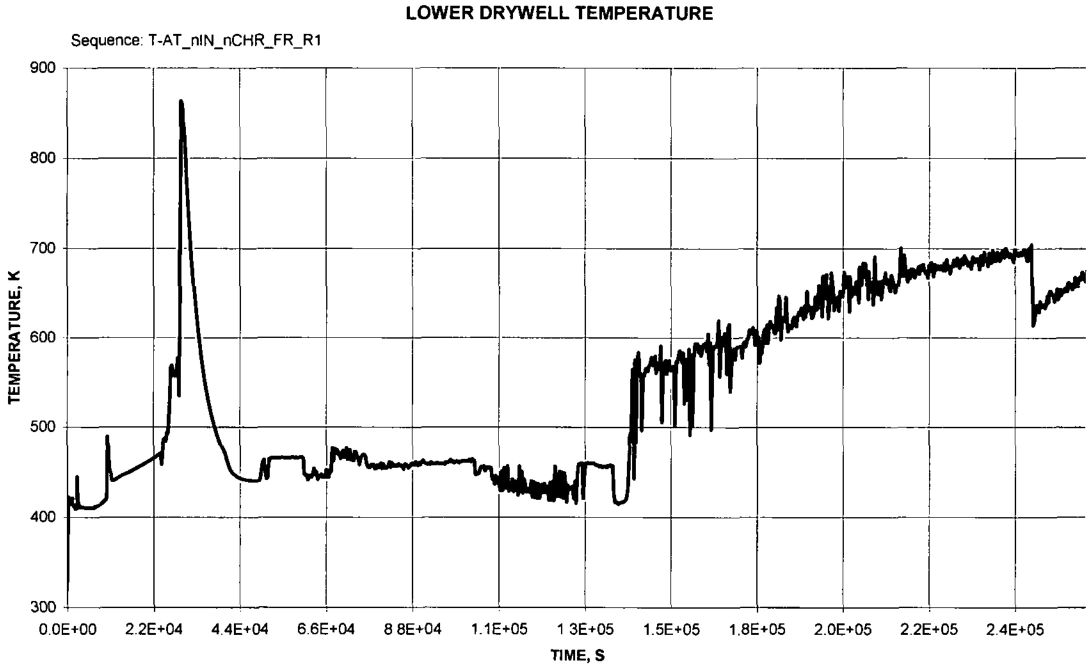


Figure 9A-11i - T-AT_nIN_nCHR_FR_R1 Lower Drywell Temperature

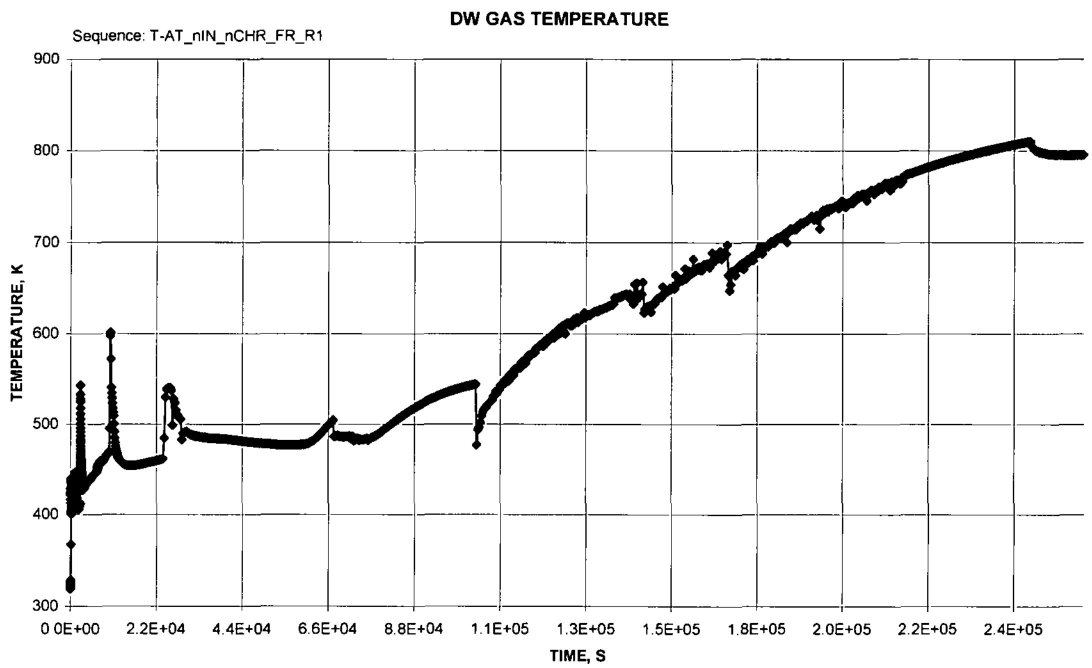


Figure 9A-11j - T-AT_nIN_nCHR_FR_R1 DW Gas Temperature

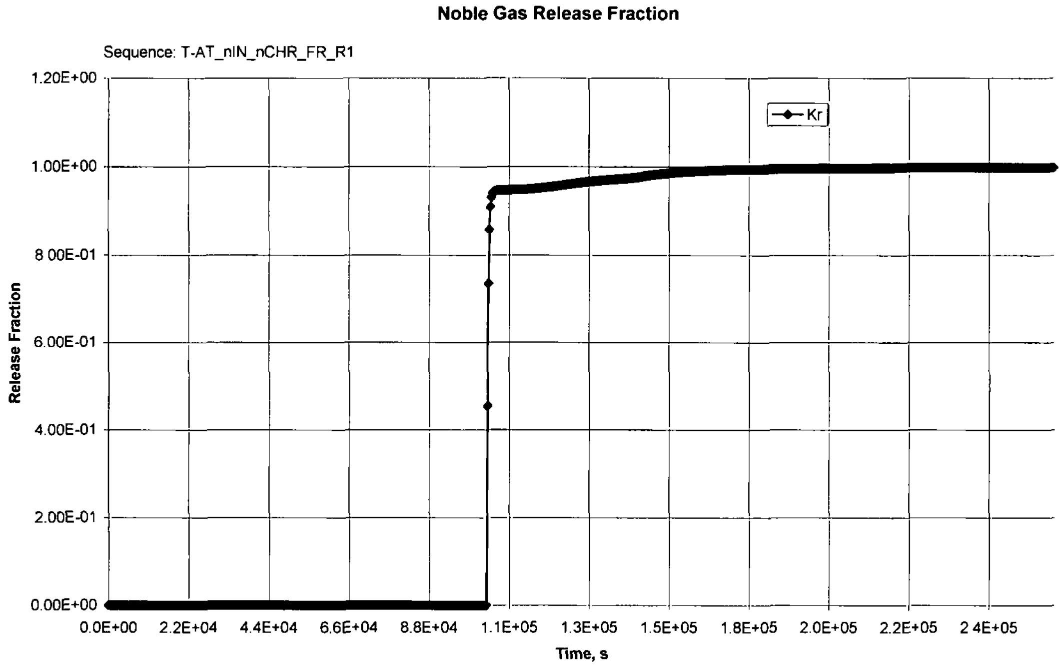


Figure 9A-11k - T-AT_nIN_nCHR_FR_R1 Noble Gas Release Fraction

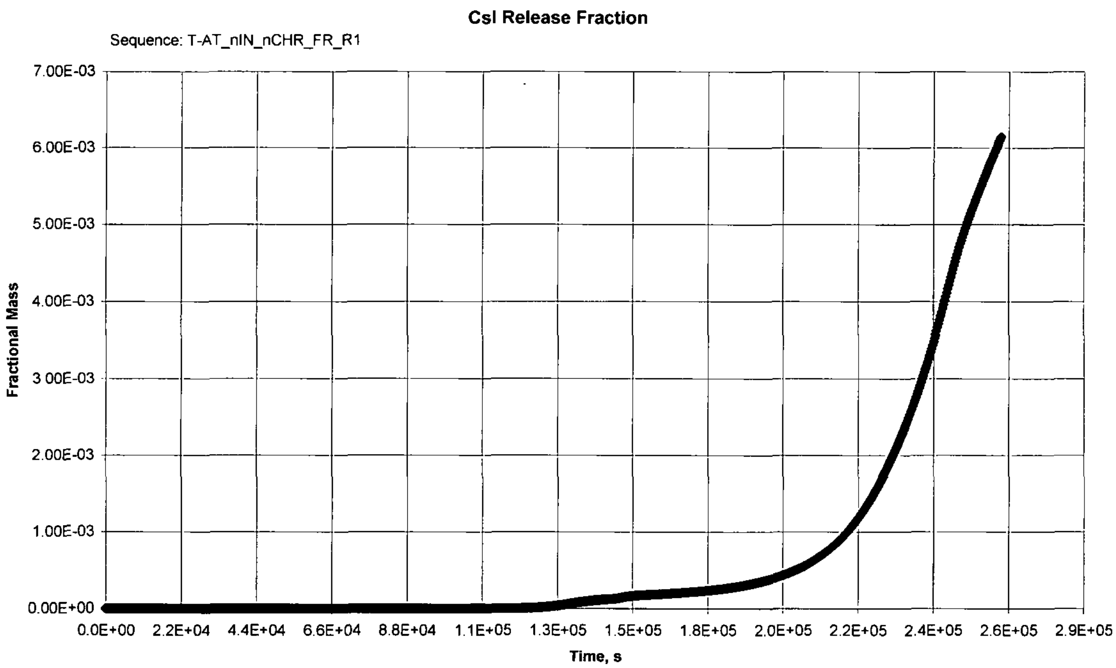


Figure 9A-11l - T-AT_nIN_nCHR_FR_R1 CsI Release Fraction

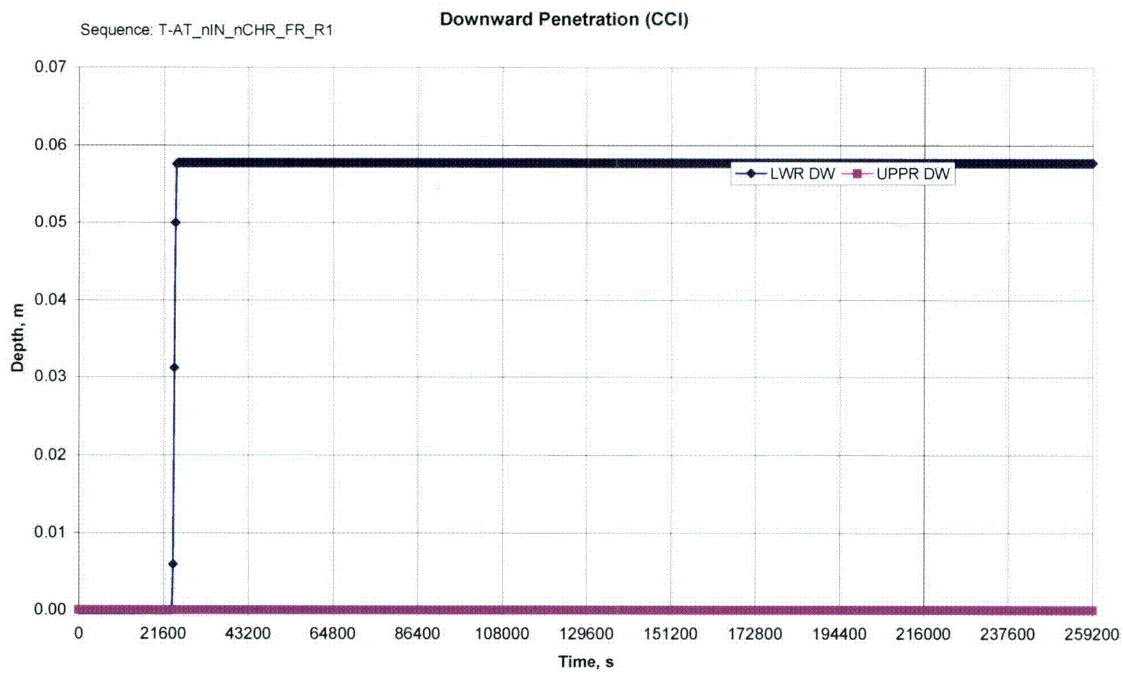
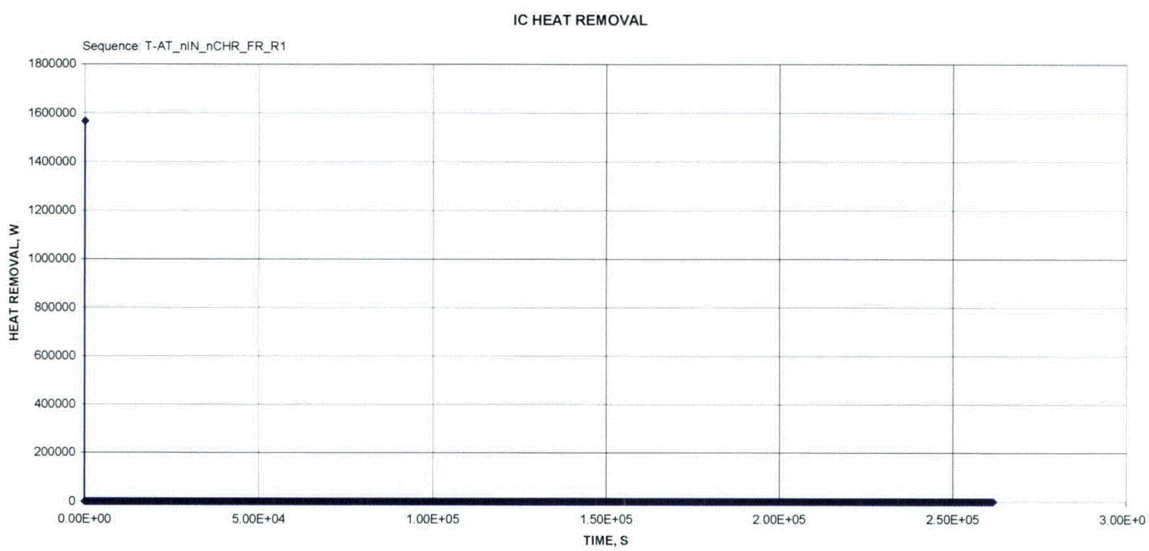


Figure 9A-11m - T-AT_nIN_nCHR_FR_R1 Downward Penetration (CCI)

Figure 9A-11n - T-AT_nIN_nCHR_FR_R1 ICS Heat Removal



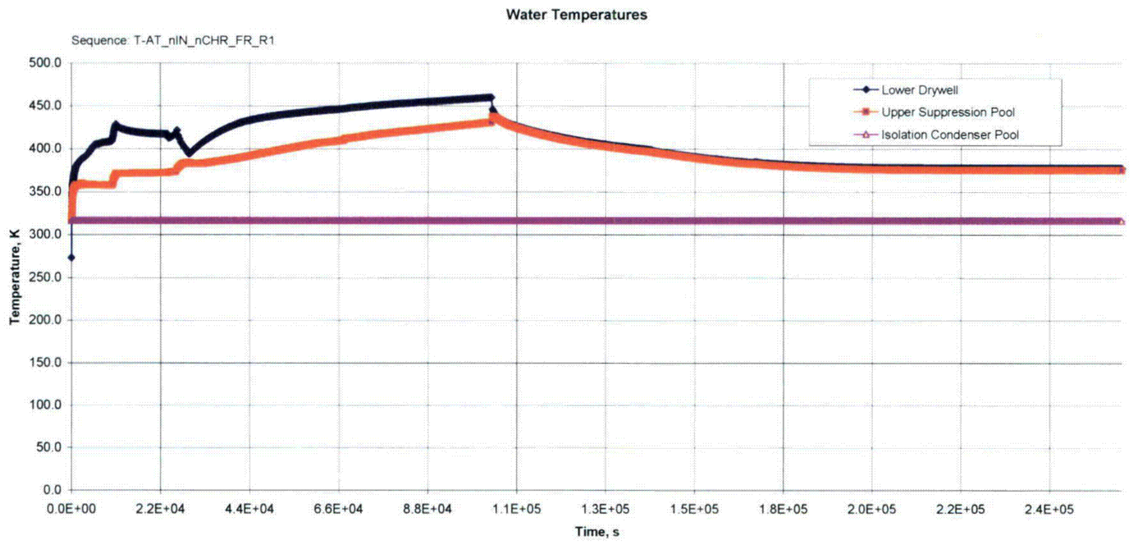


Figure 9A-11o - T-AT_nIN_nCHR_FR_R1 Water Temperature

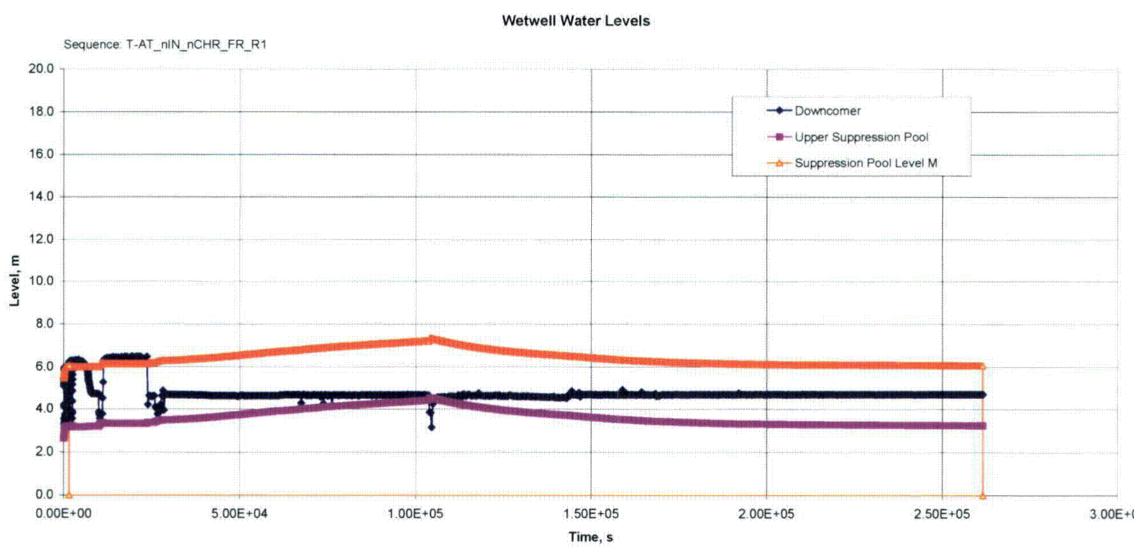


Figure 9A-11p - T-AT_nIN_nCHR_FR_R1 Wetwell Water Levels

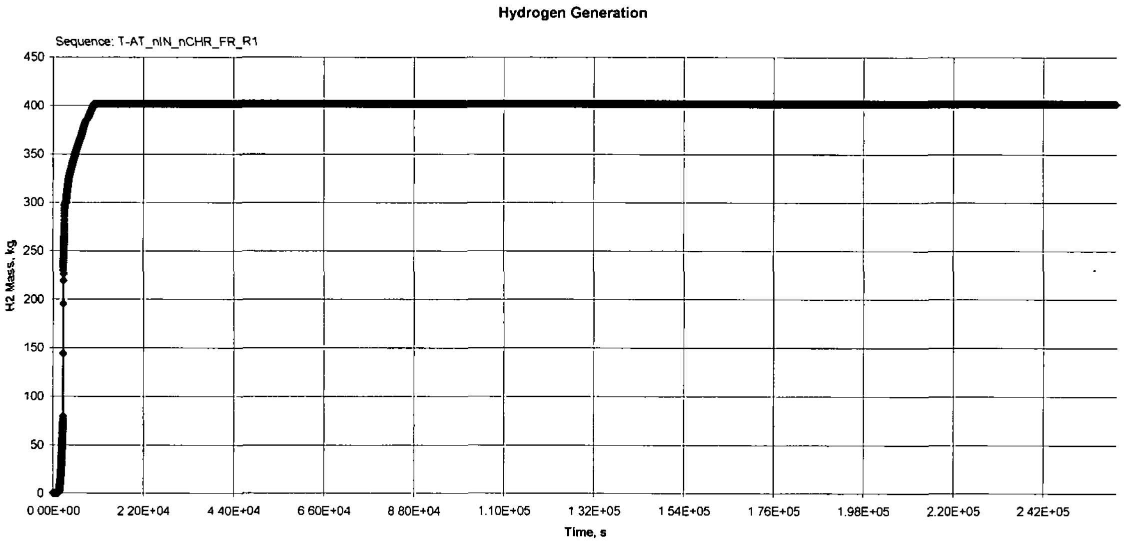


Figure 9A-11q - T-AT_nIN_nCHR_FR_R1 Hydrogen Generation

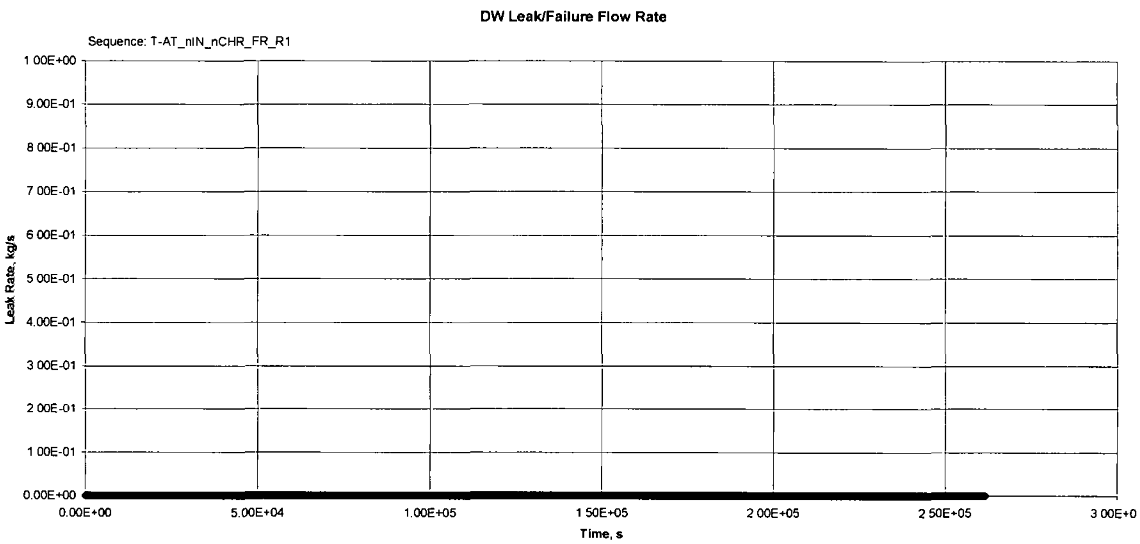


Figure 9A-11r - T-AT_nIN_nCHR_FR_R1 DW Leak/Failure Flow Rate

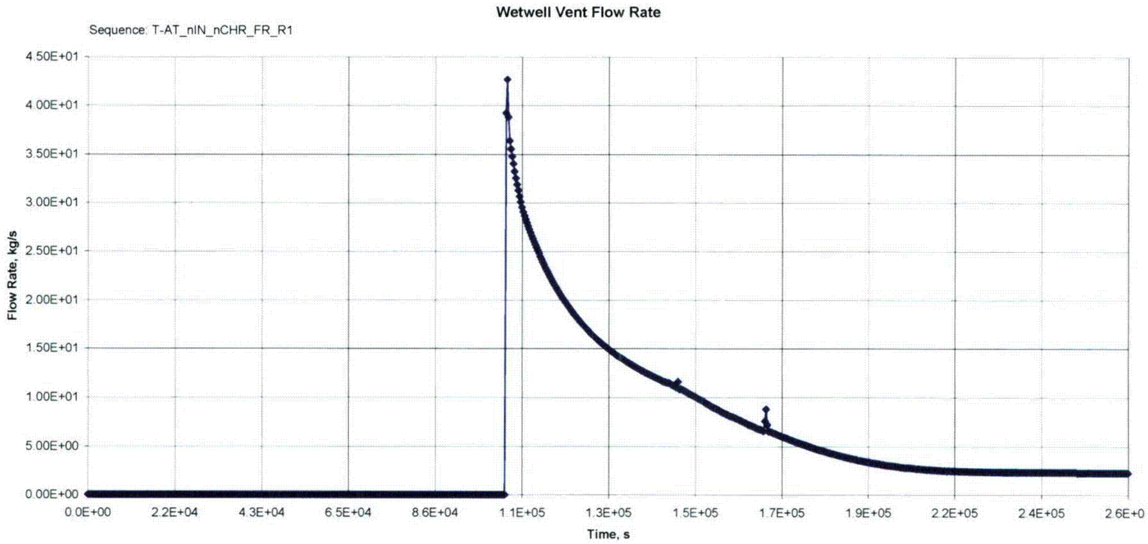


Figure 9A-11s - T-AT_nIN_nCHR_FR_R1 Wetwell Vent Flow Rate

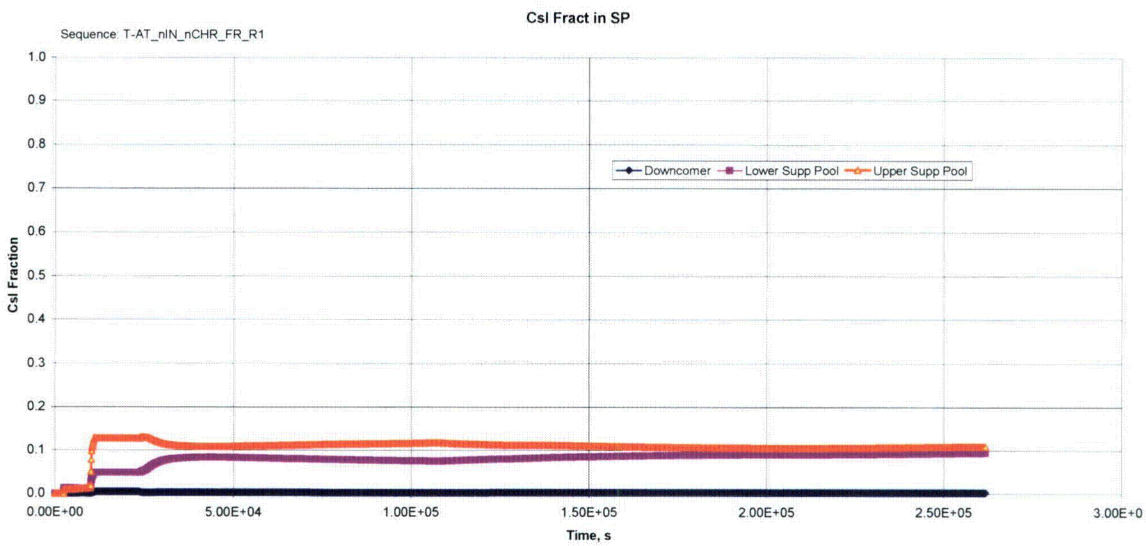


Figure 9A-11t - T-AT_nIN_nCHR_FR_R1 CsI Fraction in SP

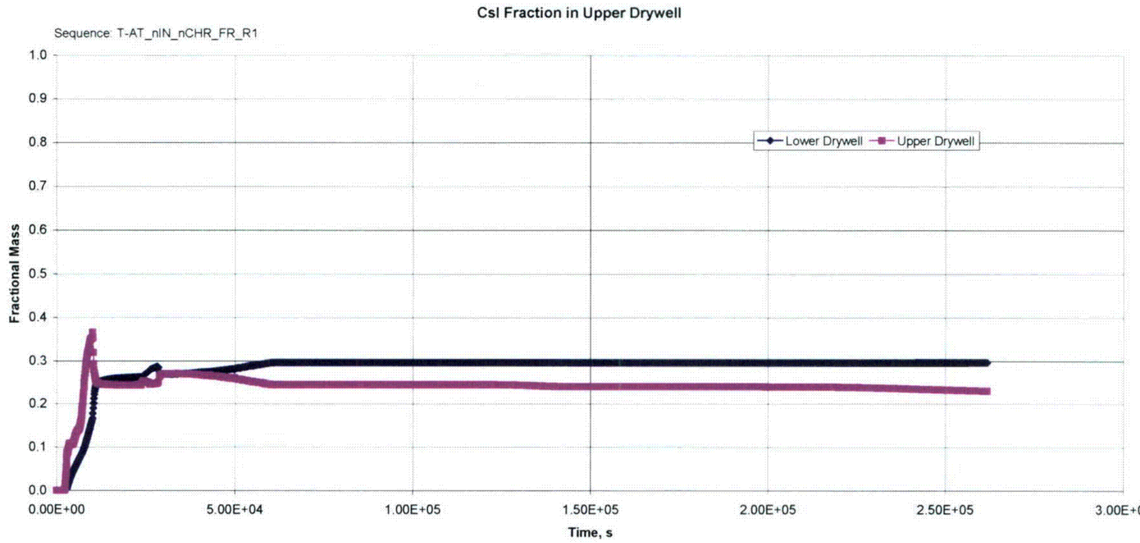


Figure 9A-11u - T-AT_nIN_nCHR_FR_R1 CsI Fraction in Upper Drywell

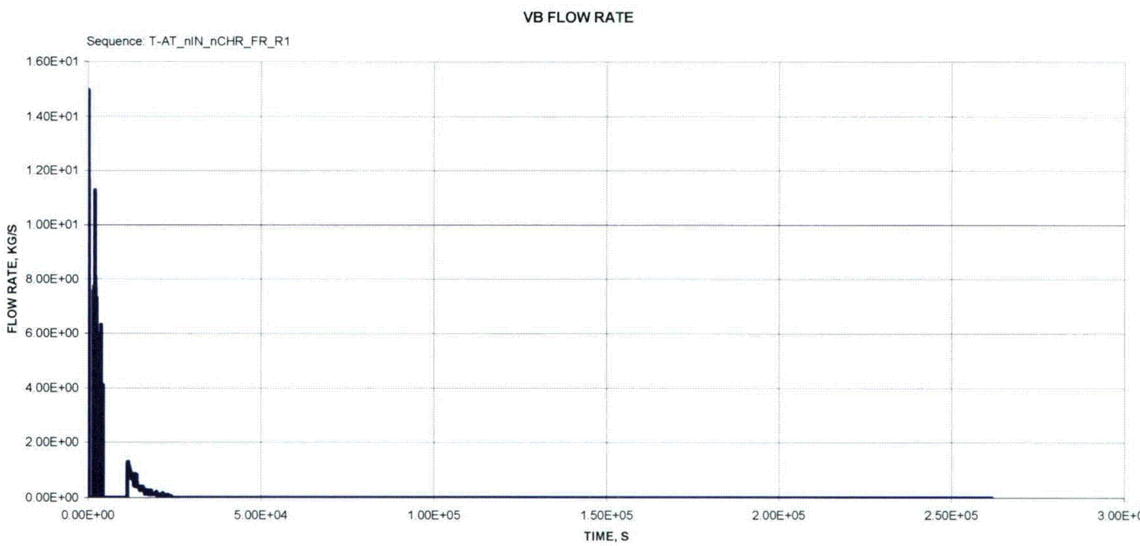


Figure 9A-11v - T-AT_nIN_nCHR_FR_R1 VB Flow Rate

9A.12 Filtered Release (Vent @ 50% Cont. U.S.) T-AT_nIN_nCHR_FR50_R1

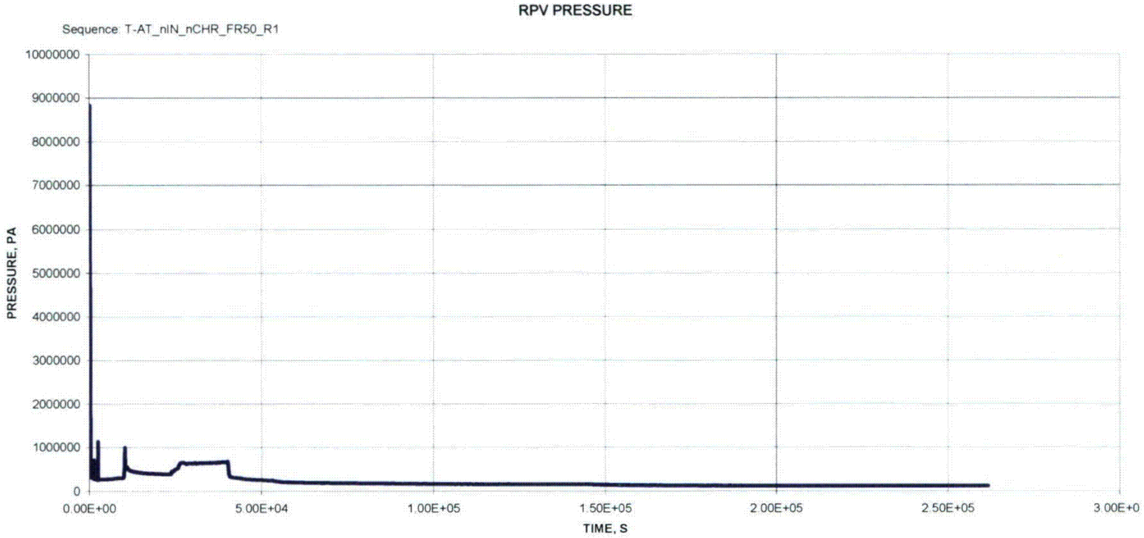


Figure 9A-12a - T-AT_nIN_nCHR_FR50_R1 RPV Pressure

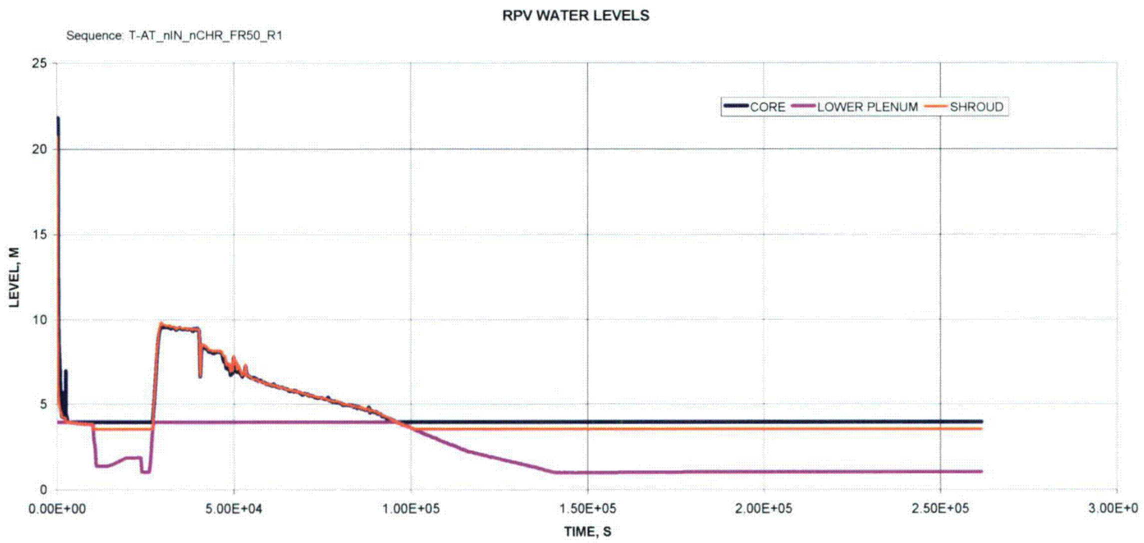


Figure 9A-12b - T-AT_nIN_nCHR_FR50_R1 RPV Water Levels

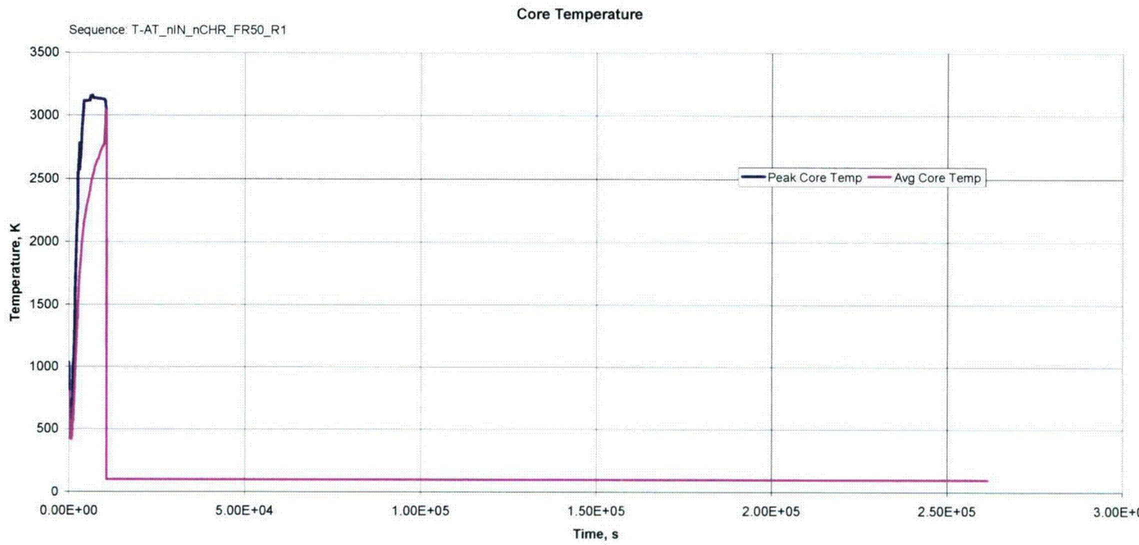


Figure 9A-12c - T-AT_nIN_nCHR_FR50_R1 Core Temperature

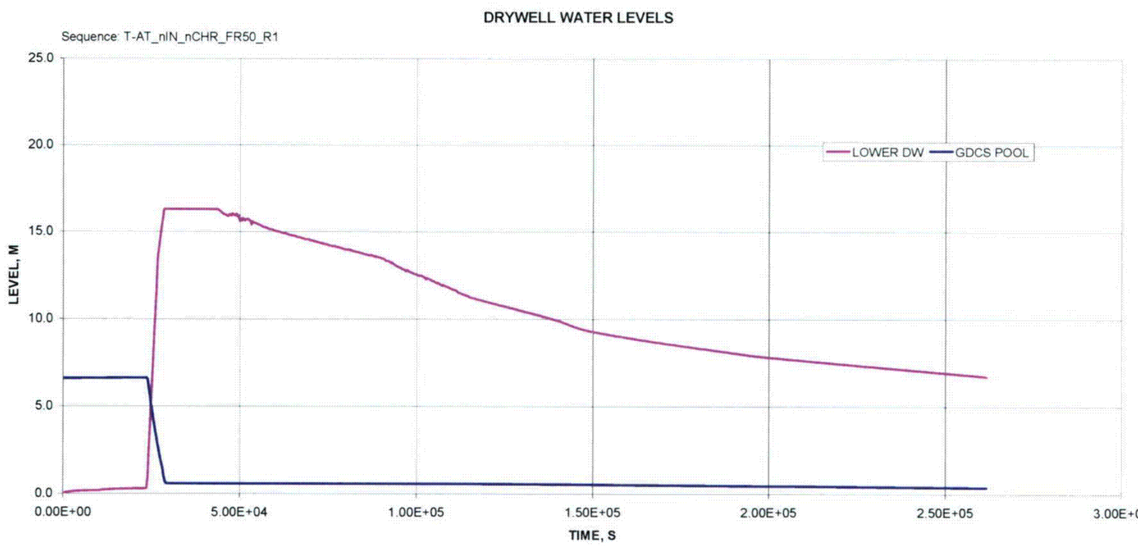


Figure 9A-12d - T-AT_nIN_nCHR_FR50_R1 Drywell Water Levels

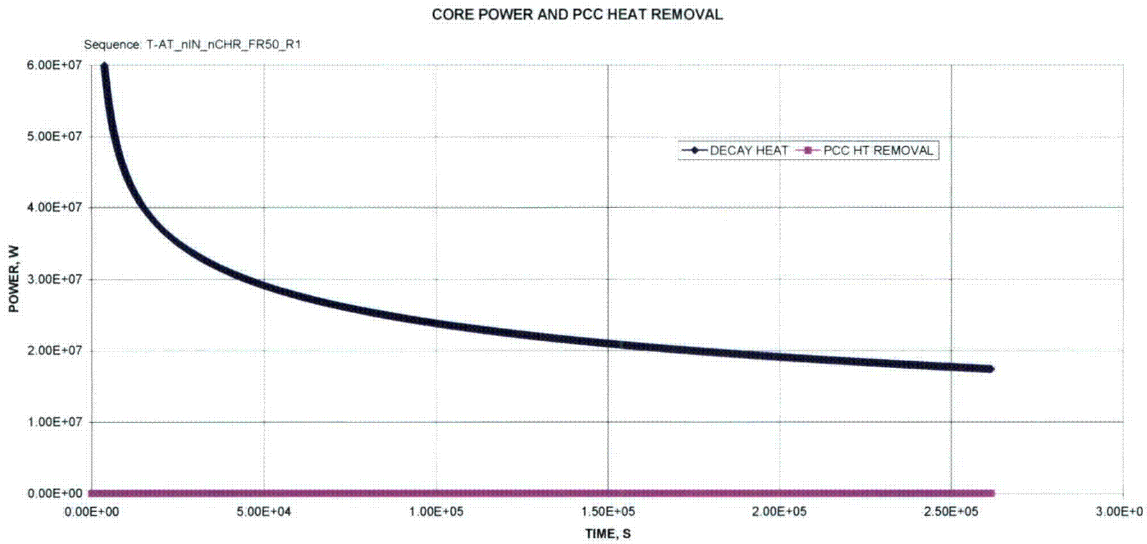


Figure 9A-12e - T-AT_nIN_nCHR_FR50_R1 Core Power and PCCS Heat Removal

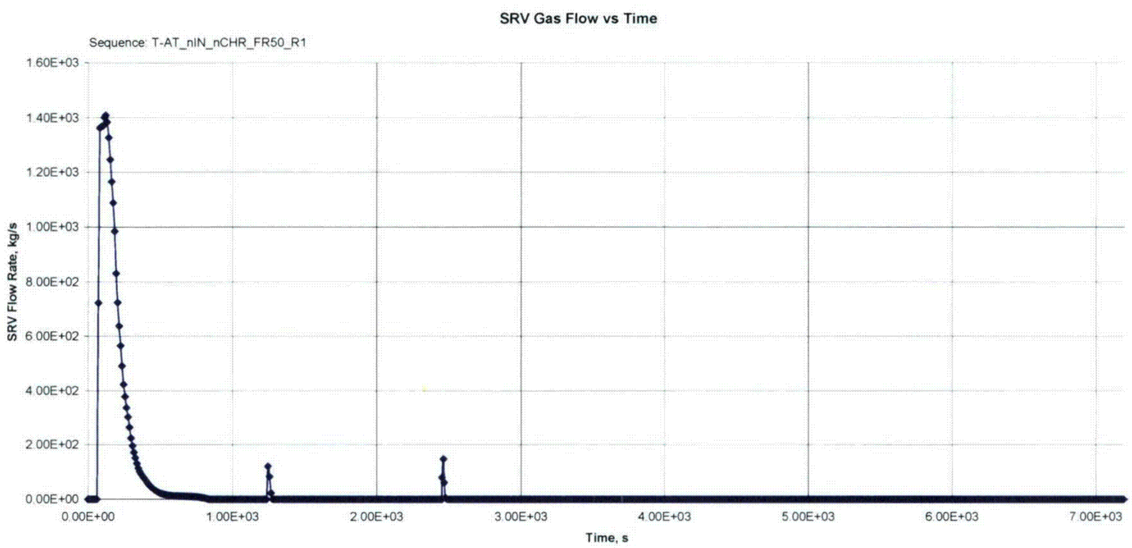


Figure 9A-12f - T-AT_nIN_nCHR_FR50_R1 SRV Gas Flow vs. Time

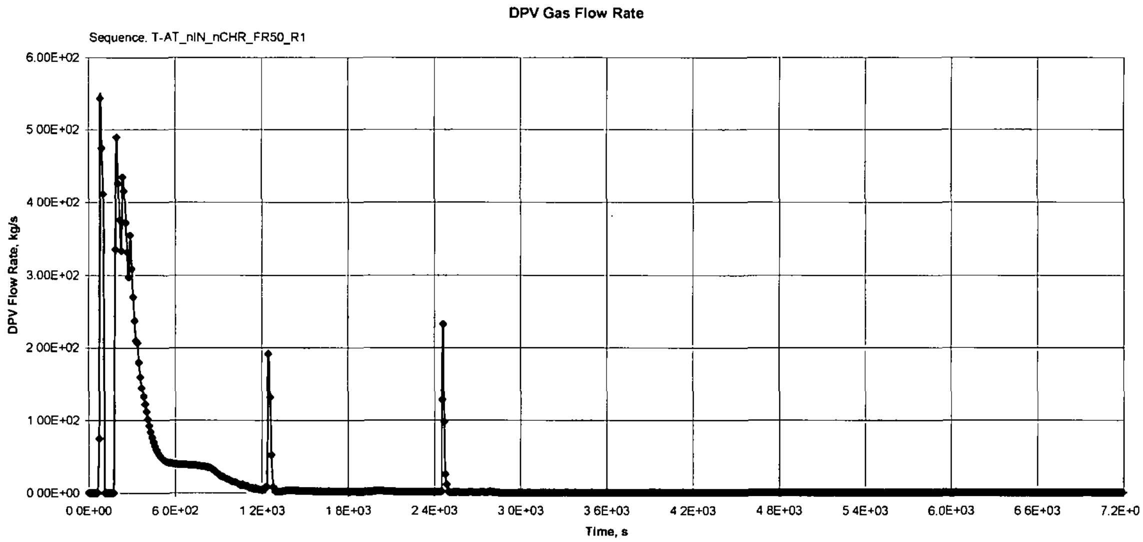


Figure 9A-12g - T-AT_nIN_nCHR_FR50_R1 DPV Gas Flow Rate

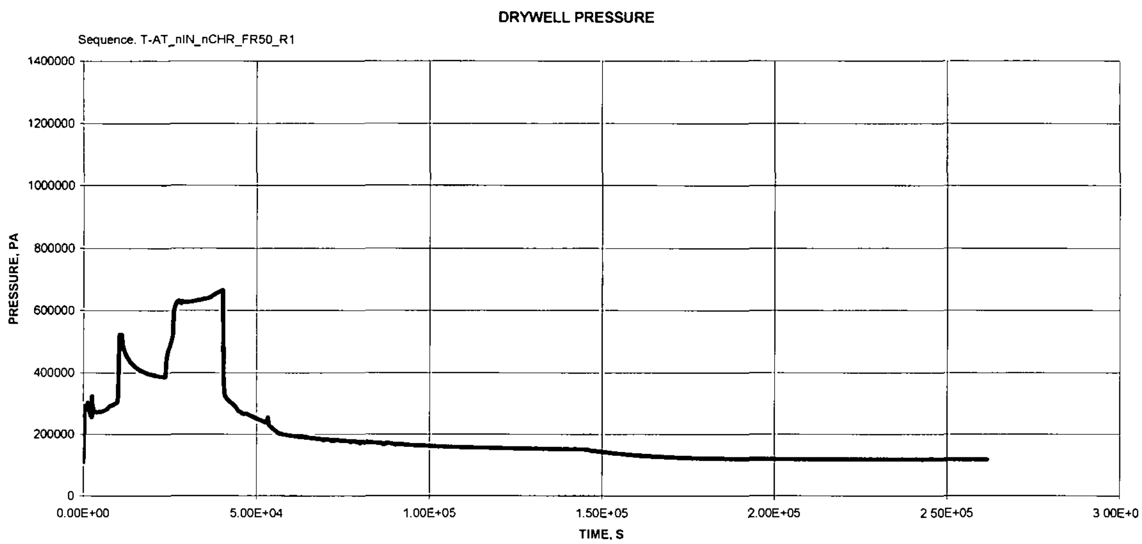


Figure 9A-12h - T-AT_nIN_nCHR_FR50_R1 Drywell Pressure

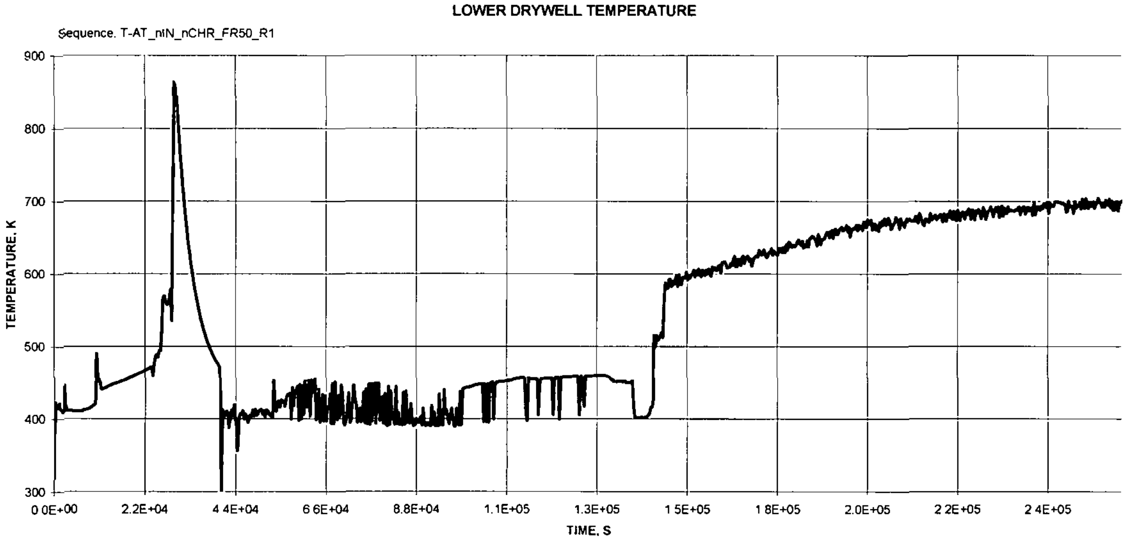


Figure 9A-12i - T-AT_nIN_nCHR_FR50_R1 Lower Drywell Temperature

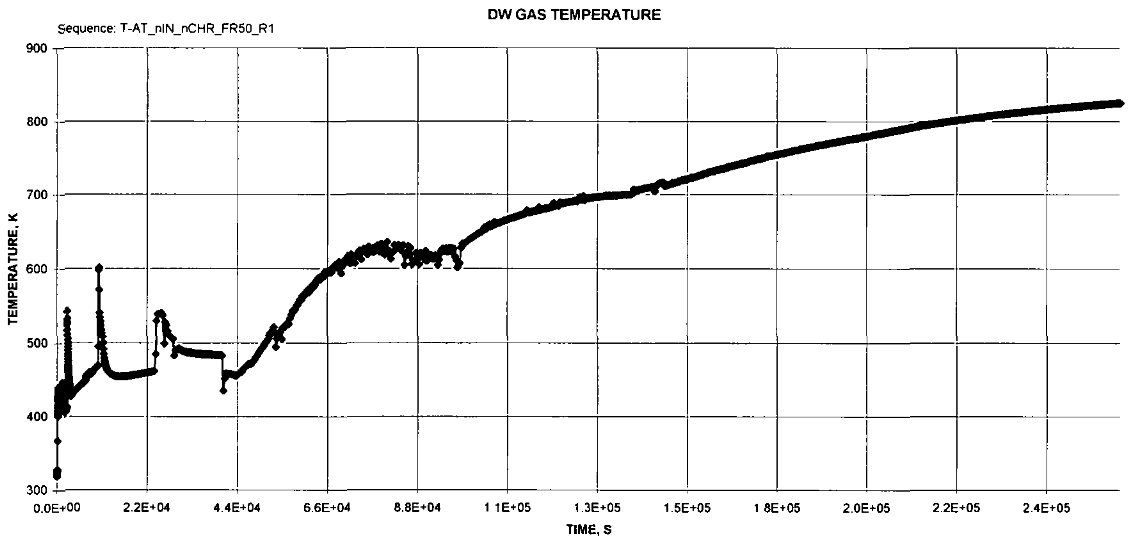


Figure 9A-12j - T-AT_nIN_nCHR_FR50_R1 DW Gas Temperature

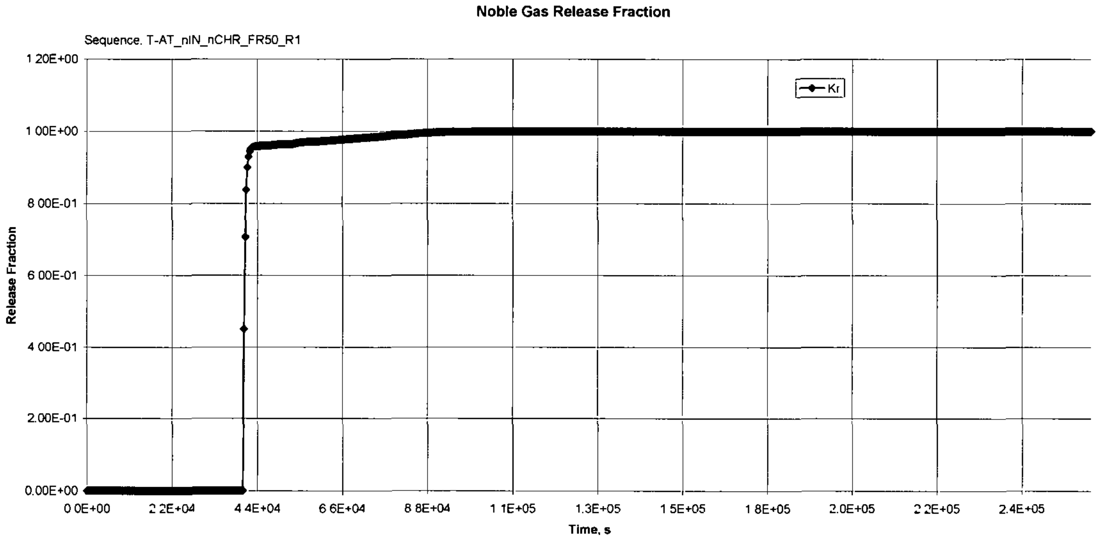


Figure 9A-12k - T-AT_nIN_nCHR_FR50_R1 Noble Gas Release Fraction

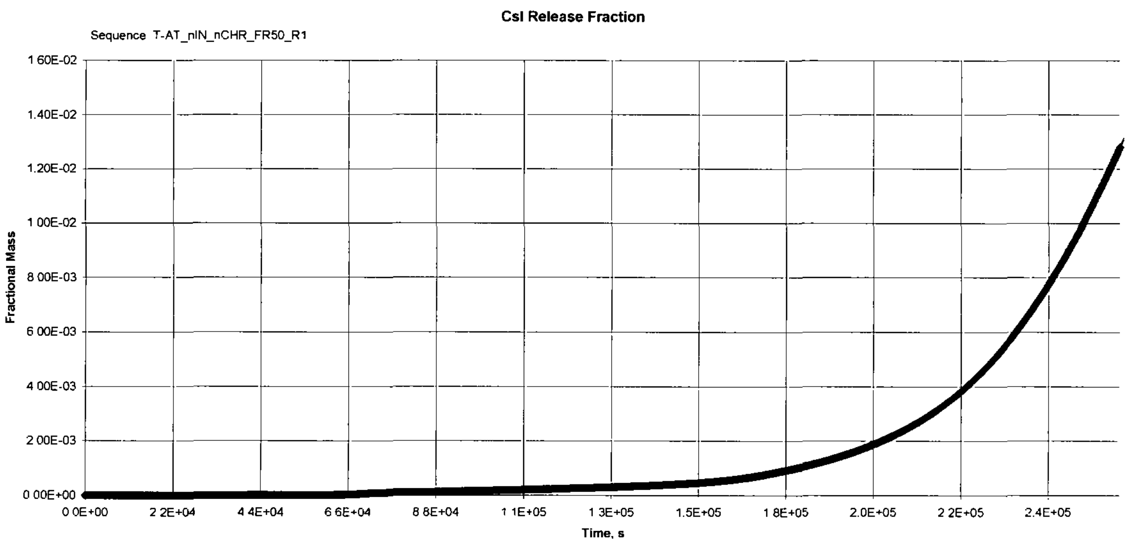


Figure 9A-12l - T-AT_nIN_nCHR_FR50_R1 CsI Release Fraction

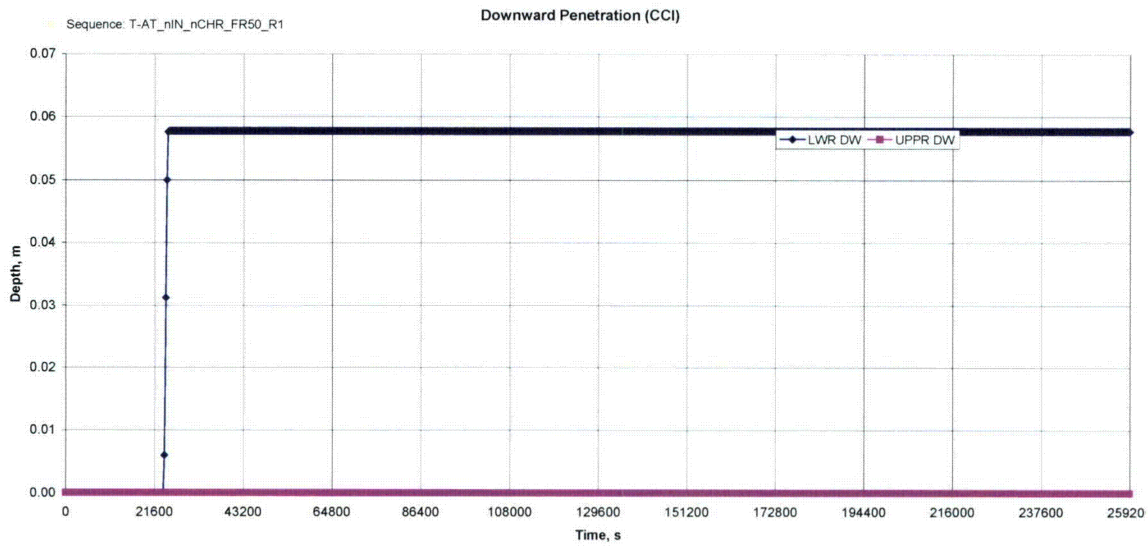


Figure 9A-12m - T-AT_nIN_nCHR_FR50_R1 Downward Penetration (CCI)

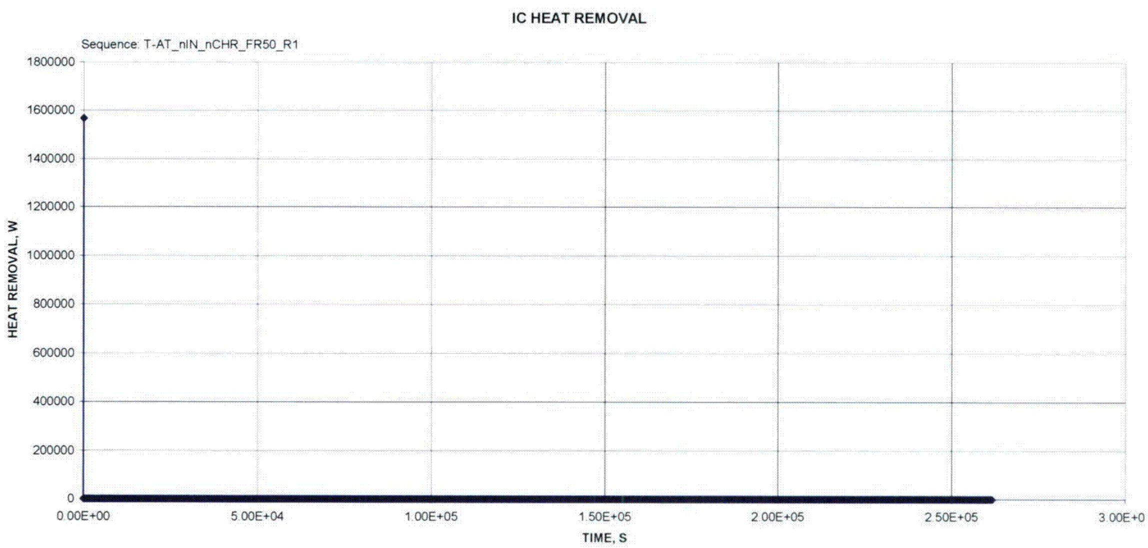


Figure 9A-12n - T-AT_nIN_nCHR_FR50_R1 ICS Heat Removal

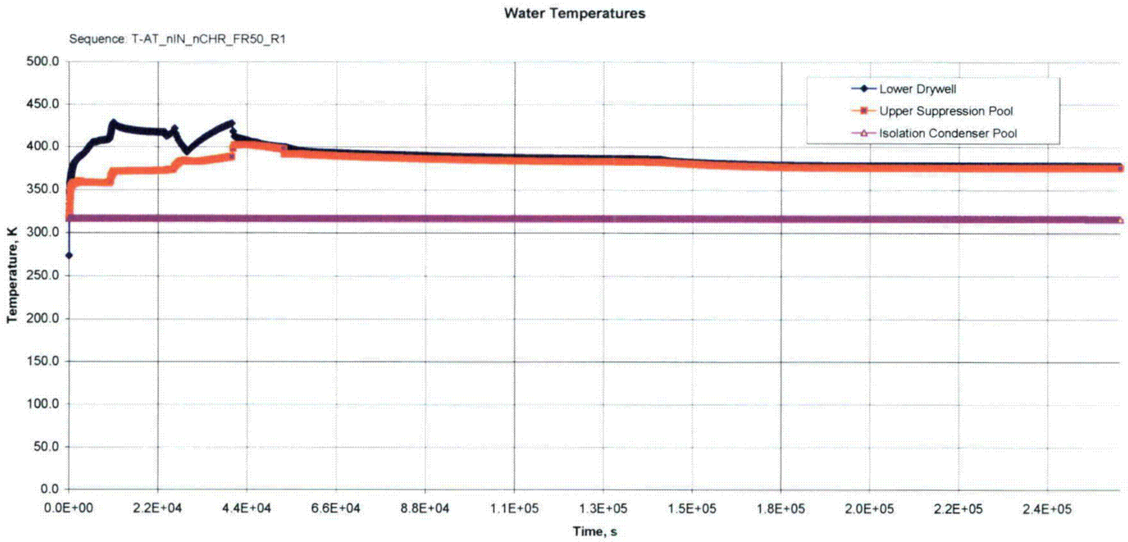


Figure 9A-12o - T-AT_nIN_nCHR_FR50_R1 Water Temperature

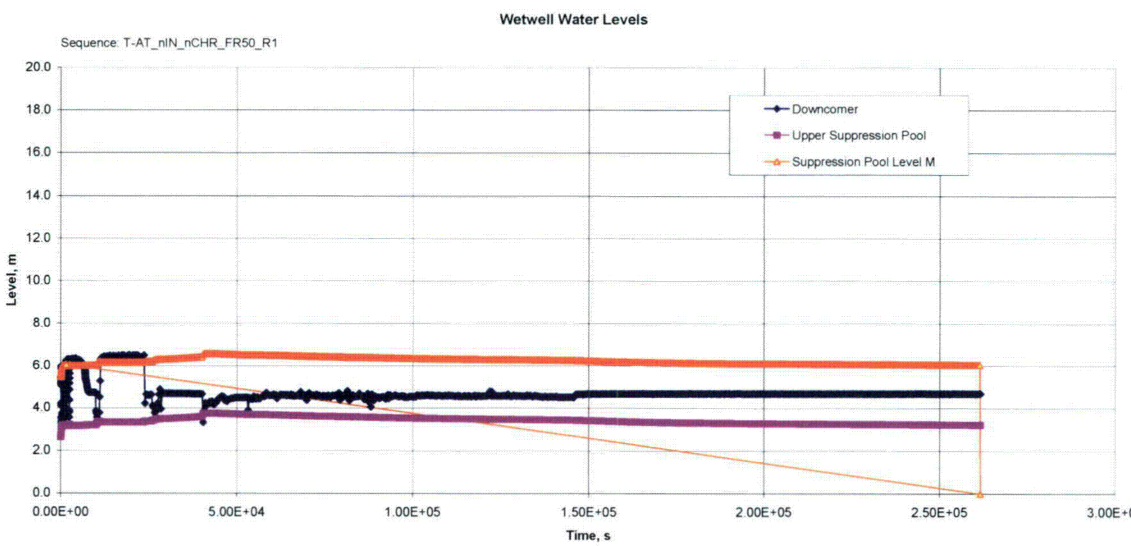


Figure 9A-12p - T-AT_nIN_nCHR_FR50_R1 Wetwell Water Levels

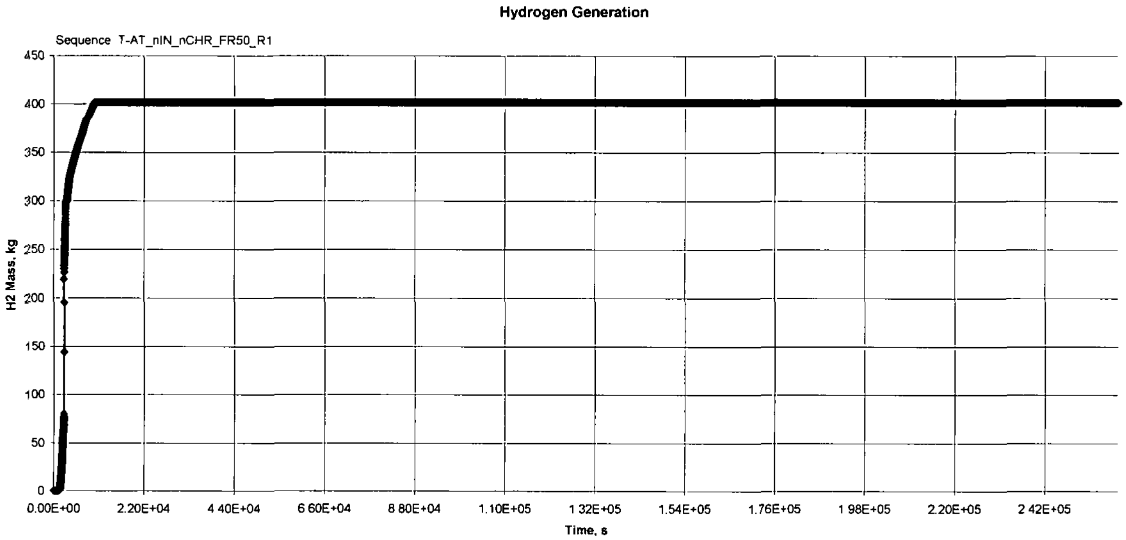


Figure 9A-12q - T-AT_nIN_nCHR_FR50_R1 Hydrogen Generation

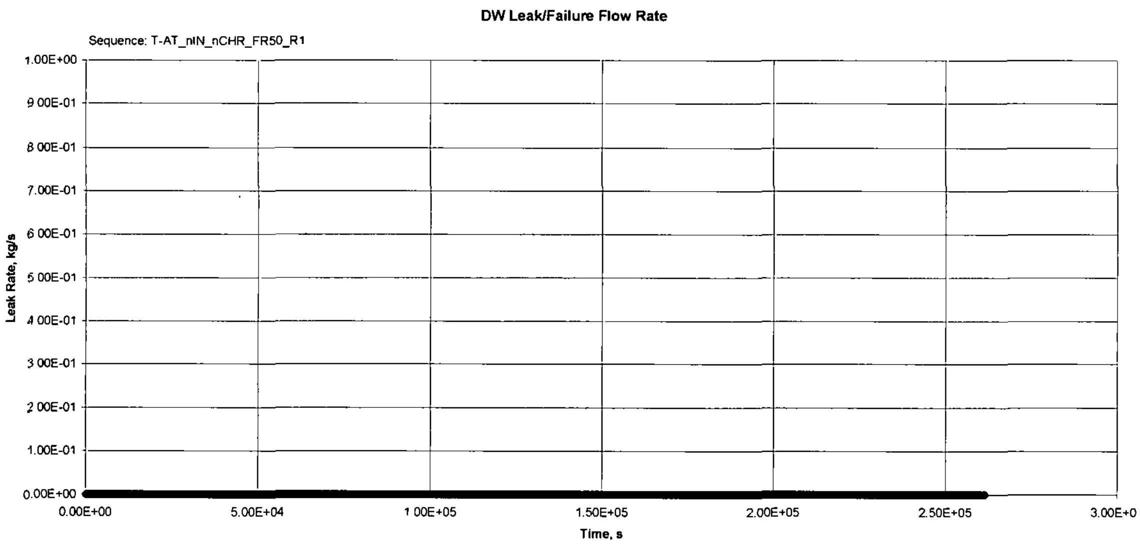


Figure 9A-12r - T-AT_nIN_nCHR_FR50_R1 DW Leak/Failure Flow Rate

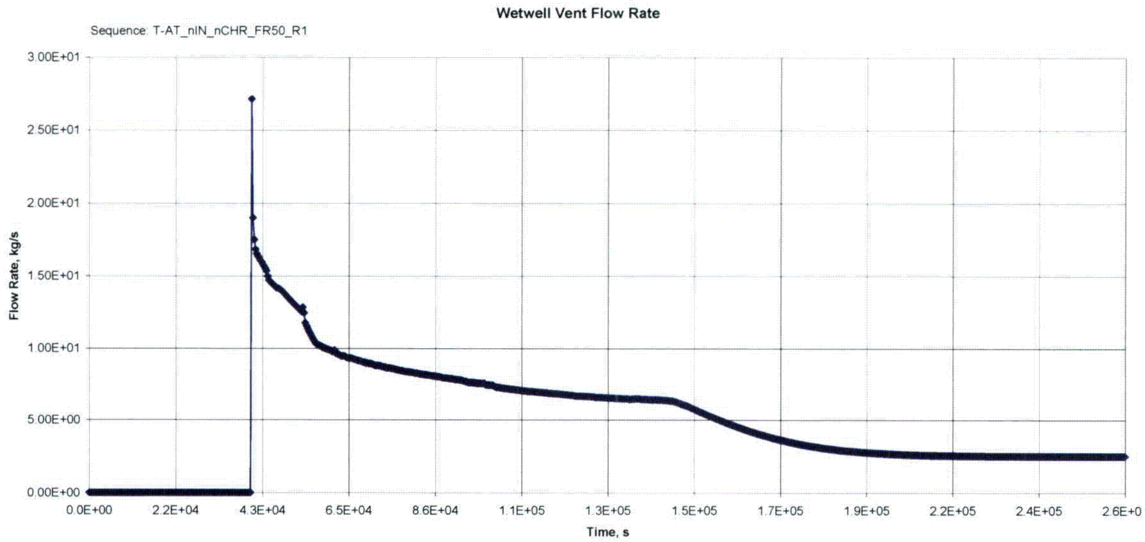


Figure 9A-12s - T-AT_nIN_nCHR_FR50_R1 Wetwell Vent Flow Rate

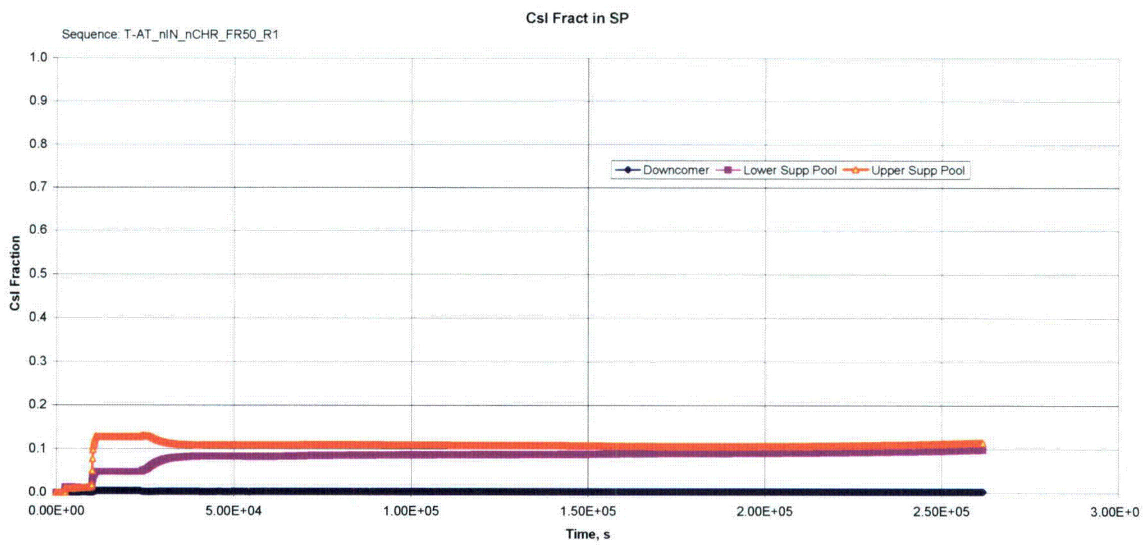


Figure 9A-12t - T-AT_nIN_nCHR_FR50_R1 CsI Fraction in SP

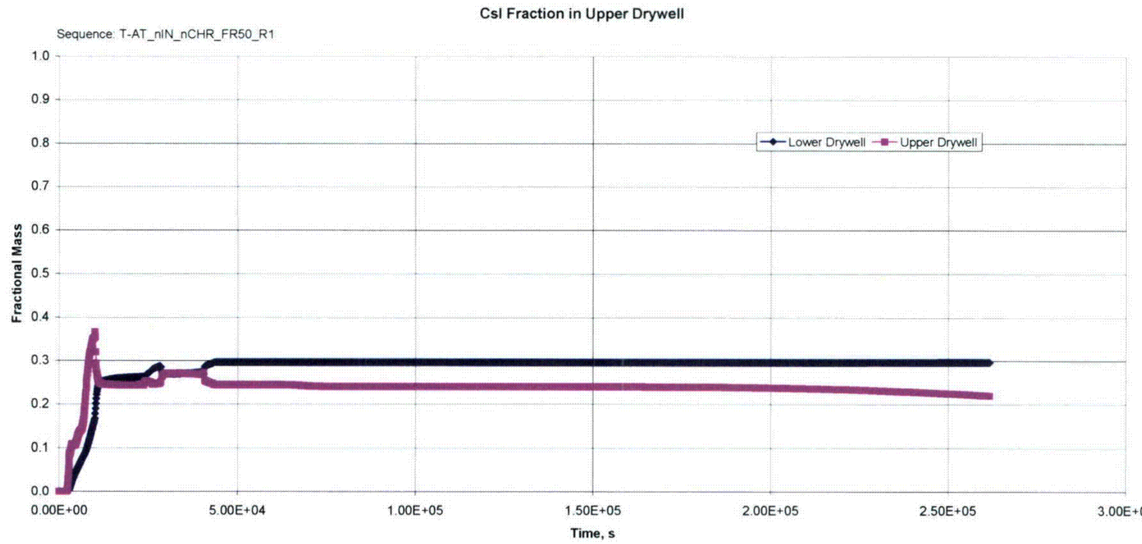


Figure 9A-12u - T-AT_nIN_nCHR_FR50_R1 CsI Fraction in Upper Drywell

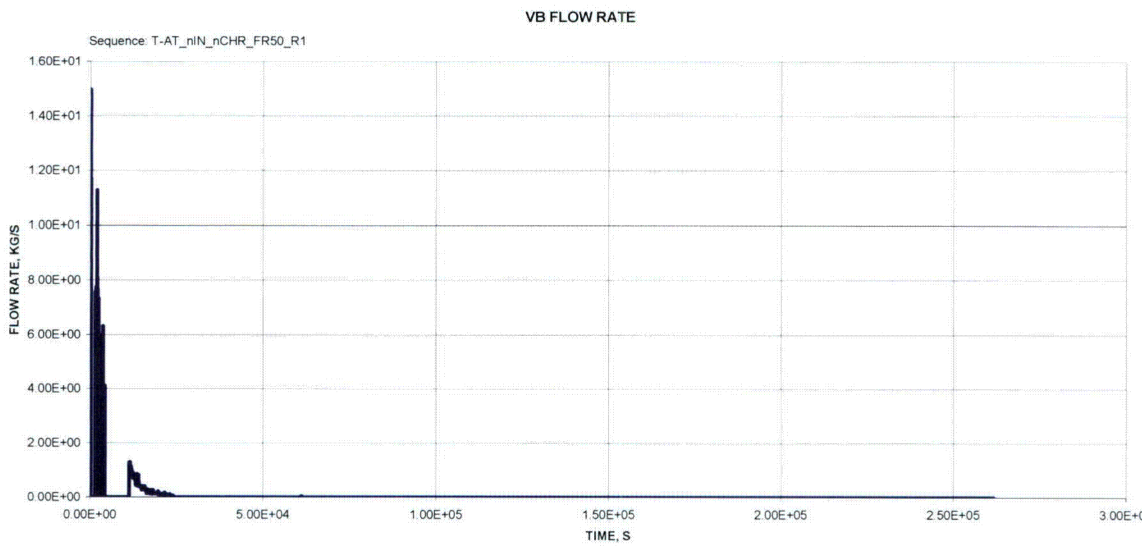


Figure 9A-12v - T-AT_nIN_nCHR_FR50_R1 VB Flow Rate

9A.13 Containment Overpressure (VB) - High Pressure T_nDP_nIN_VB_R1

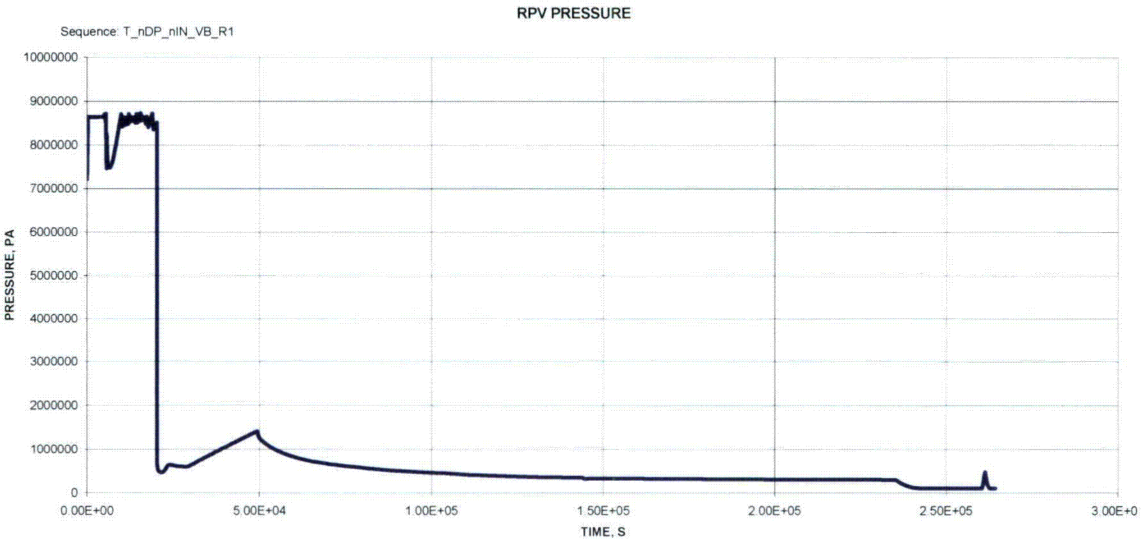


Figure 9A-13a – T_nDP_nIN_VB_R1 RPV Pressure

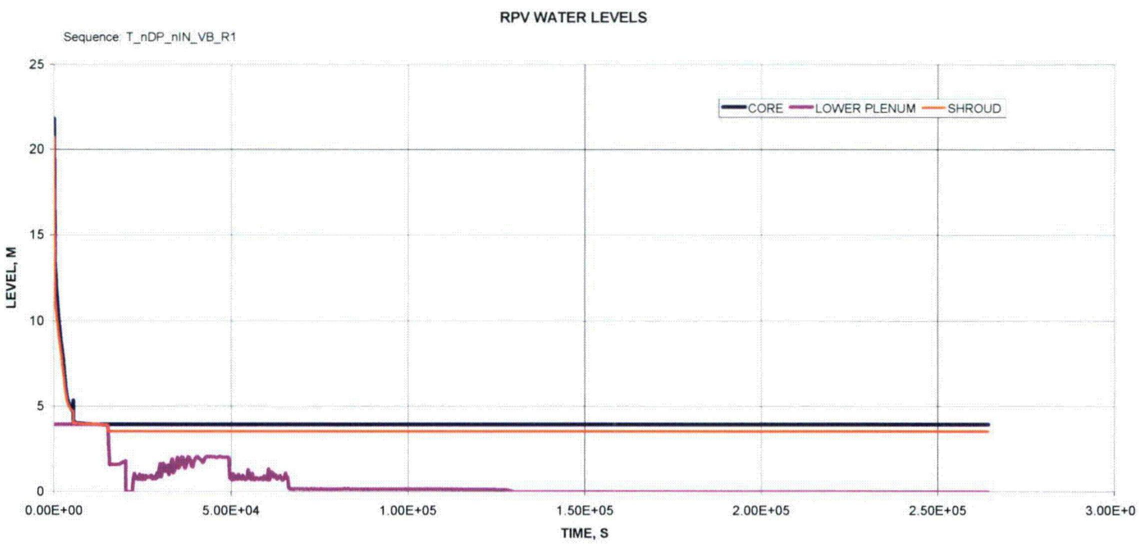


Figure 9A-13b – T_nDP_nIN_VB_R1 RPV Water Levels

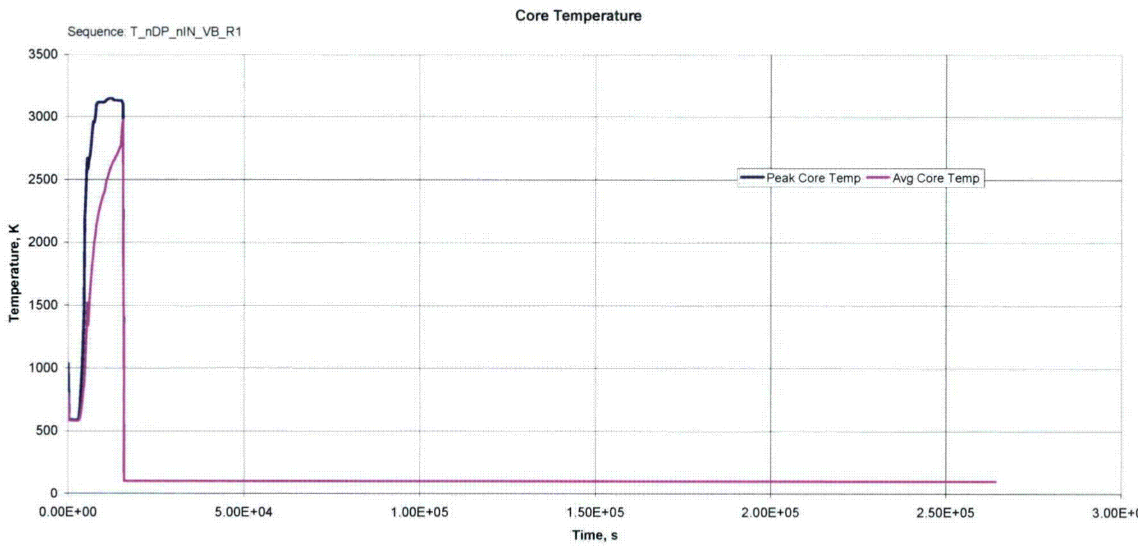


Figure 9A-13c – T_nDP_nIN_VB_R1 Core Temperature

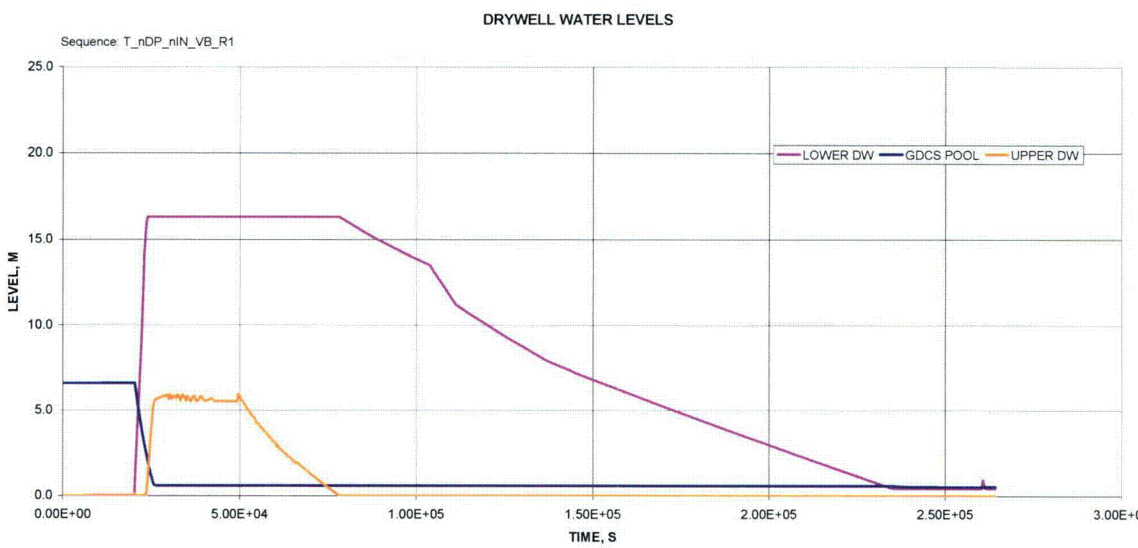


Figure 9A-13d – T_nDP_nIN_VB_R1 Drywell Water Levels

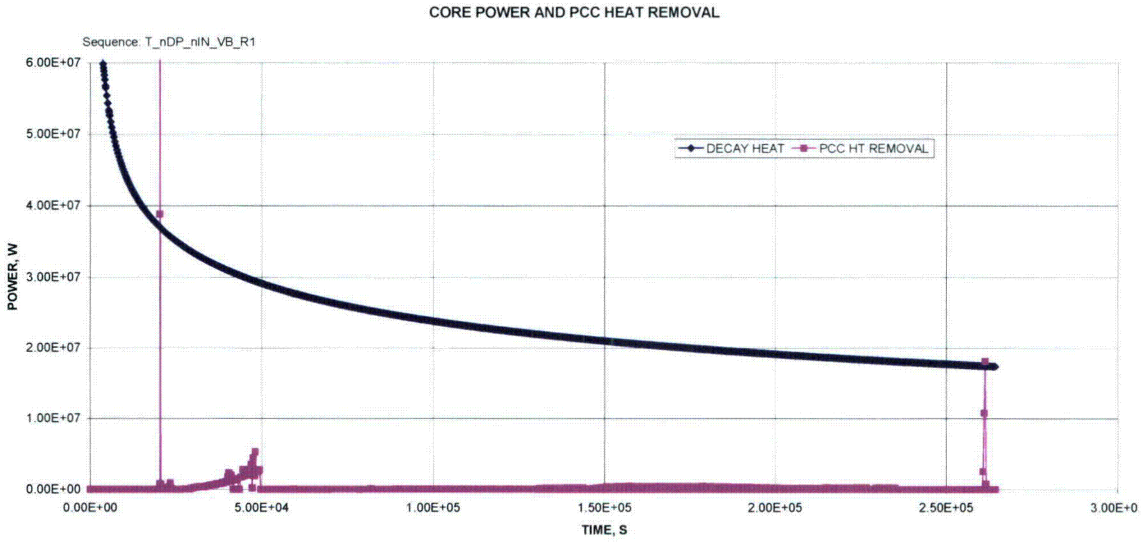


Figure 9A-13e – T_nDP_nIN_VB_R1 Core Power and PCCS Heat Removal

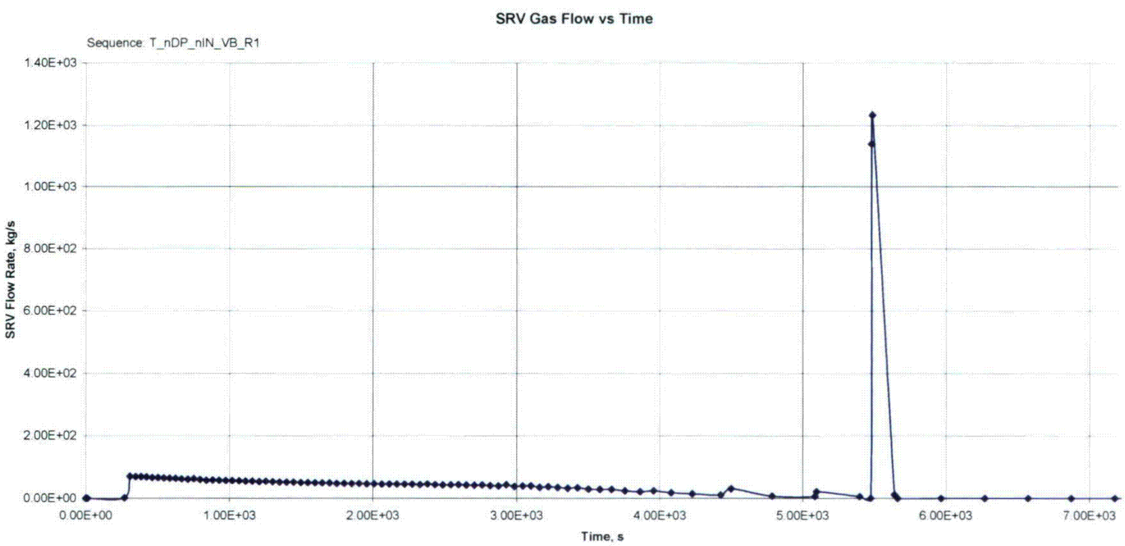


Figure 9A-13f – T_nDP_nIN_VB_R1 SRV Gas Flow Rate

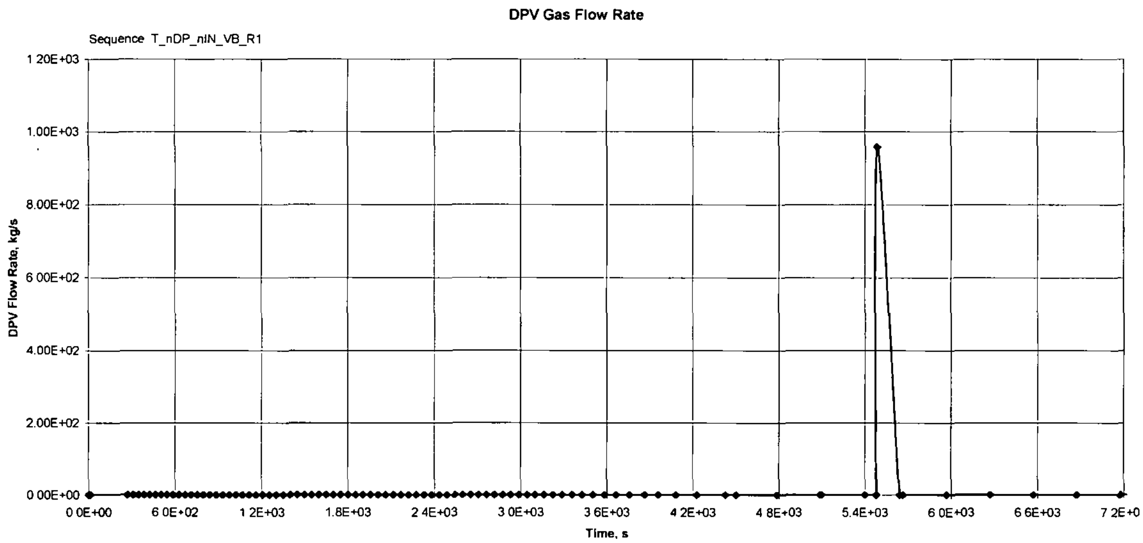


Figure 9A-13g – T_nDP_nIN_VB_R1 DPV Gas Flow Rate

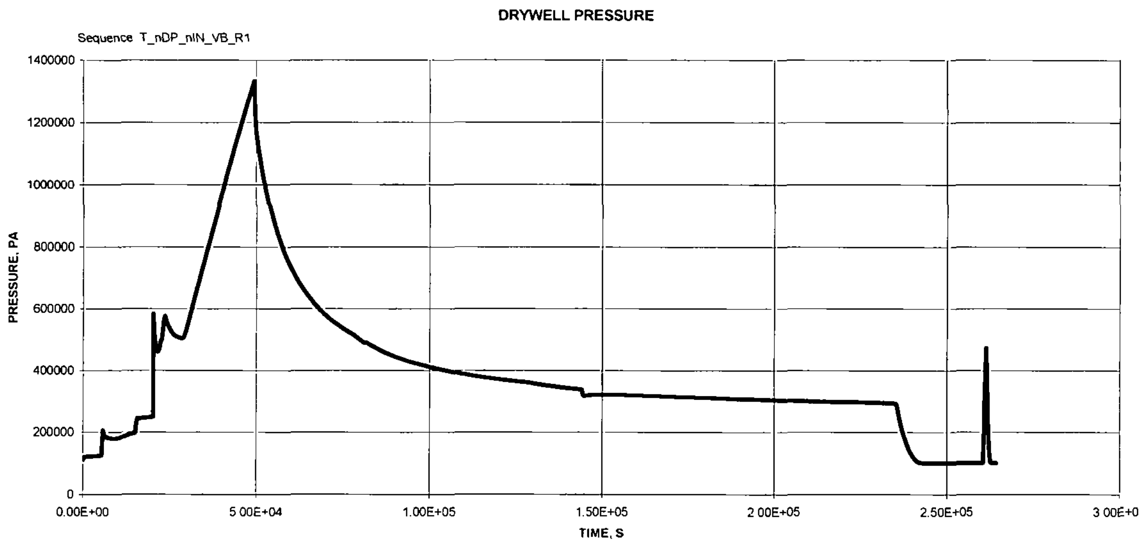


Figure 9A-13h – T_nDP_nIN_VB_R1 Drywell Pressure

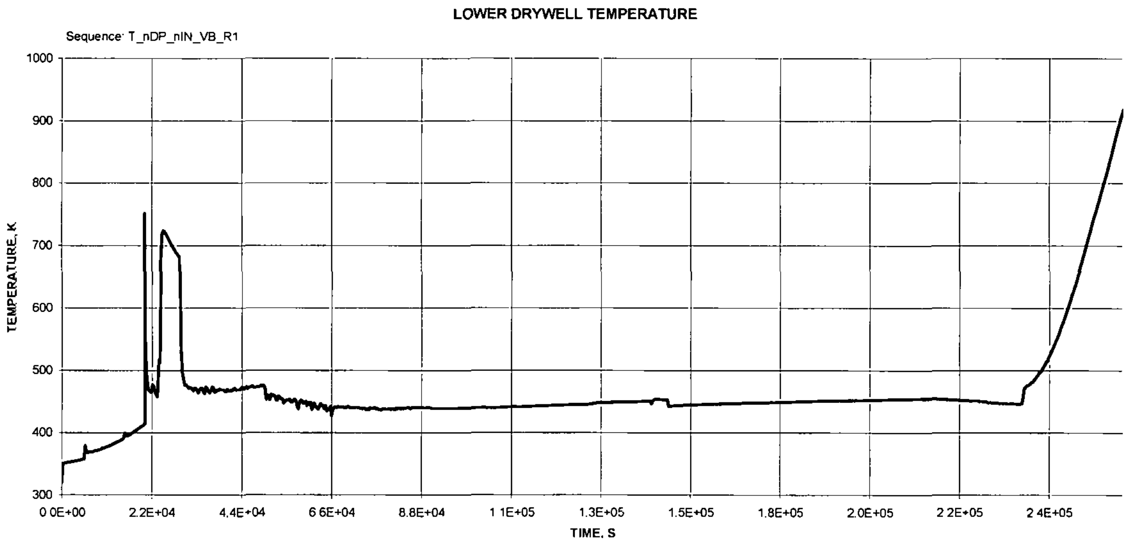


Figure 9A-13i – T_nDP_nIN_VB_R1 Lower Drywell Temperature

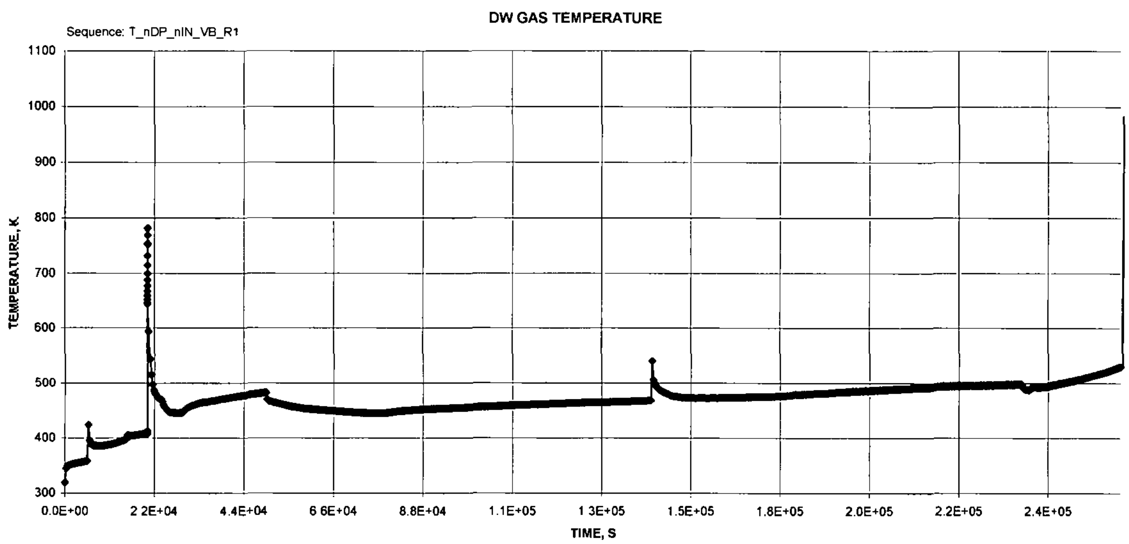


Figure 9A-13j – T_nDP_nIN_VB_R1 Drywell Gas Temperature

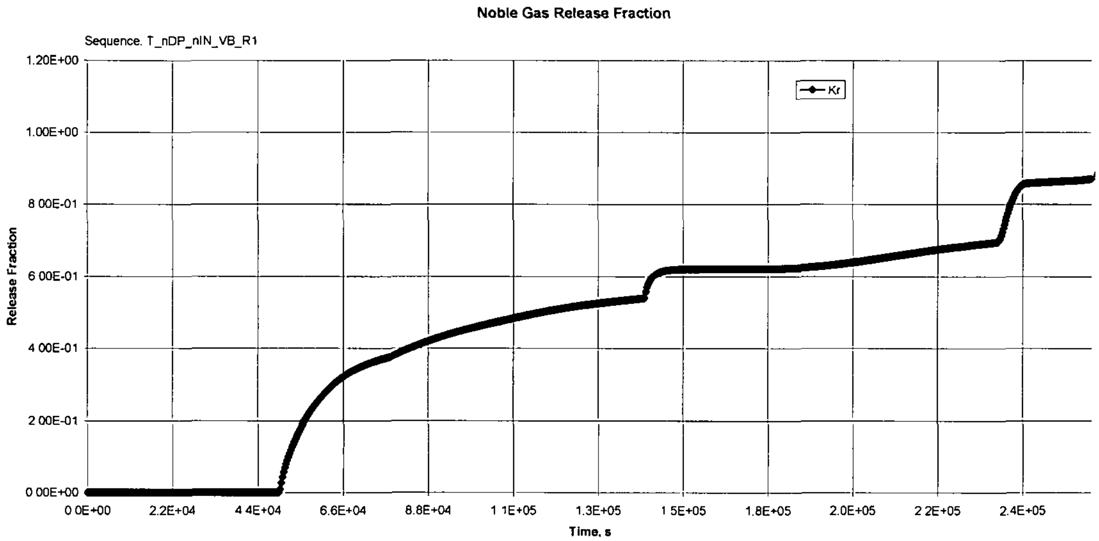


Figure 9A-13k – T_nDP_nIN_VB_R1 Noble Gas Release Fraction

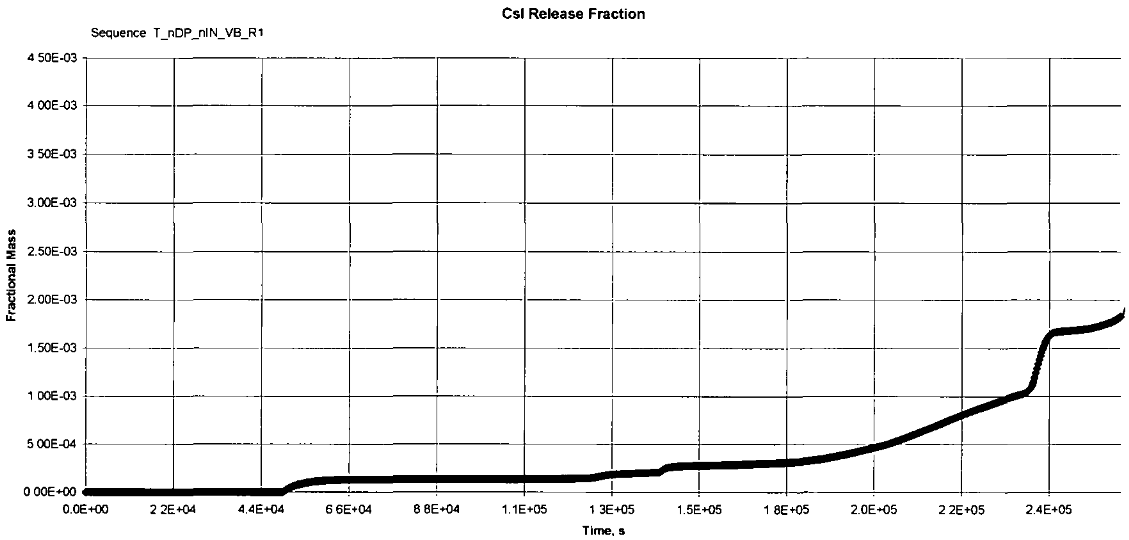


Figure 9A-13l – T_nDP_nIN_VB_R1 CsI Release Fraction

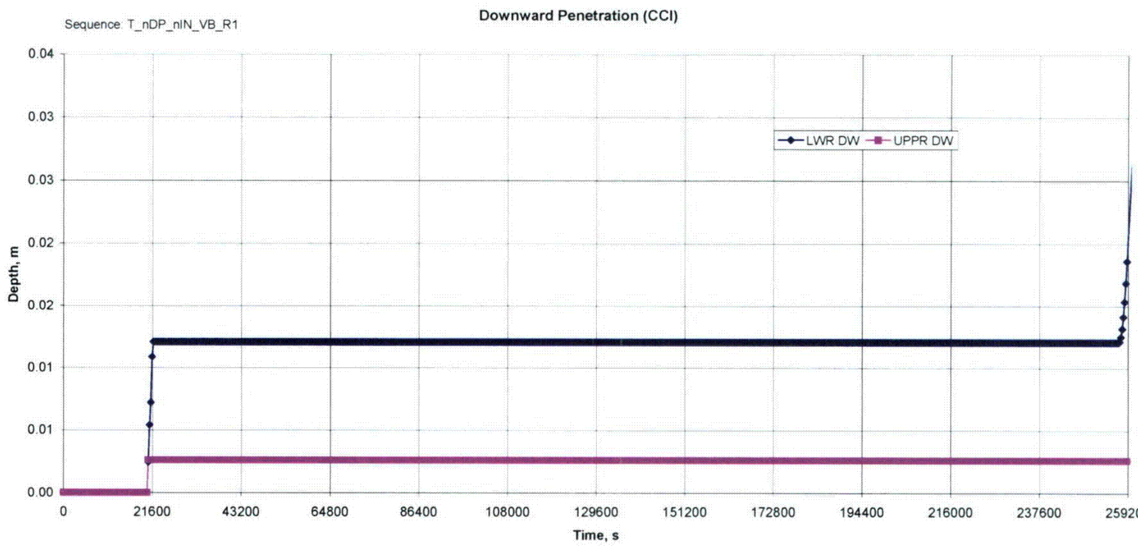


Figure 9A-13m – T_nDP_nIN_VB_R1 Downward Penetration (CCI)

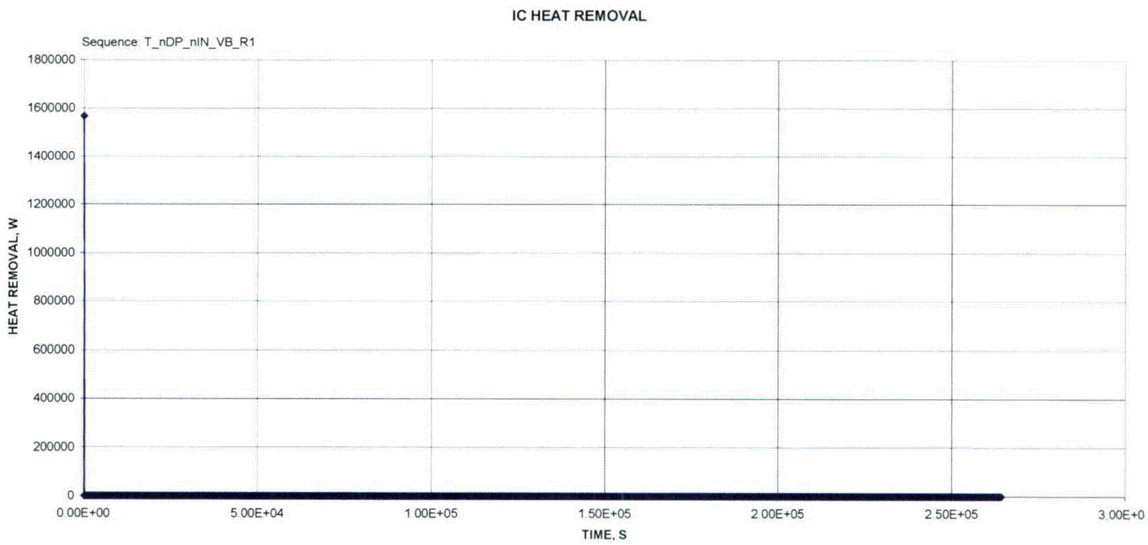


Figure 9A-13n – T_nDP_nIN_VB_R1 ICS Heat Removal

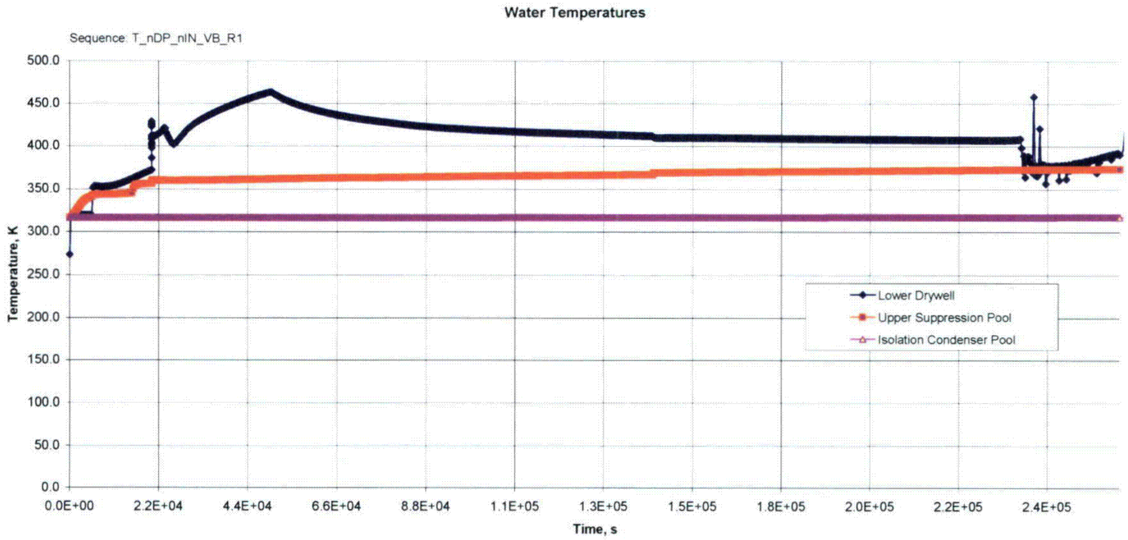


Figure 9A-13o – T_nDP_nIN_VB_R1 Water Temperatures

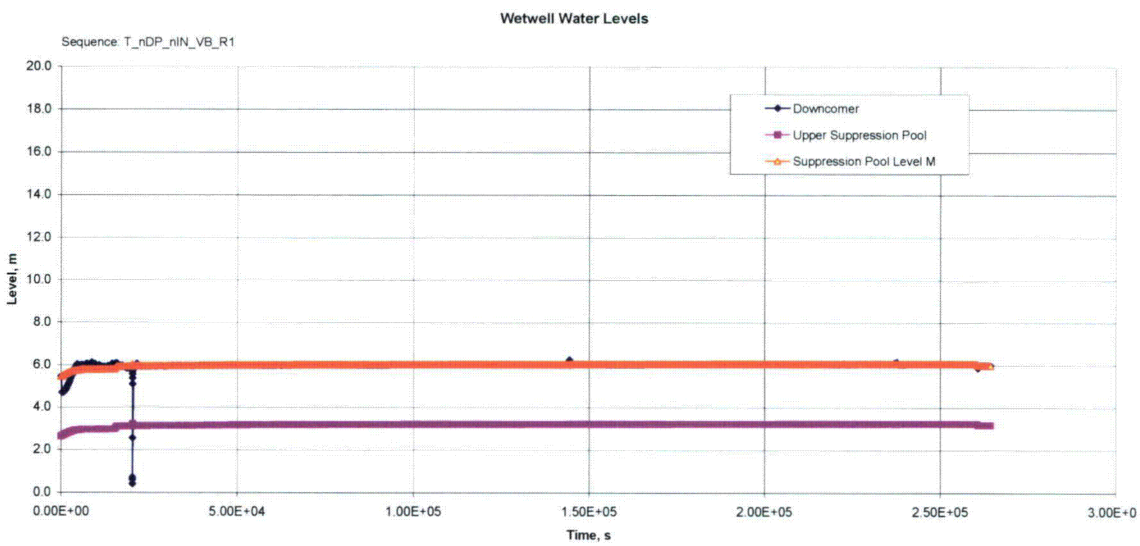


Figure 9A-13p – T_nDP_nIN_VB_R1 Wetwell Water Levels

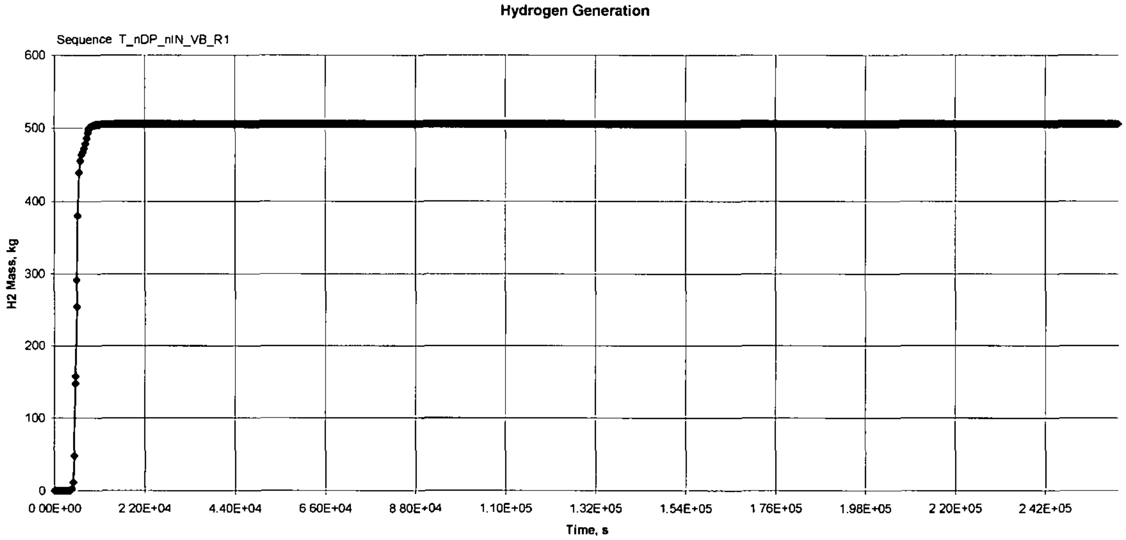


Figure 9A-13q – T_nDP_nIN_VB_R1 Hydrogen Generation

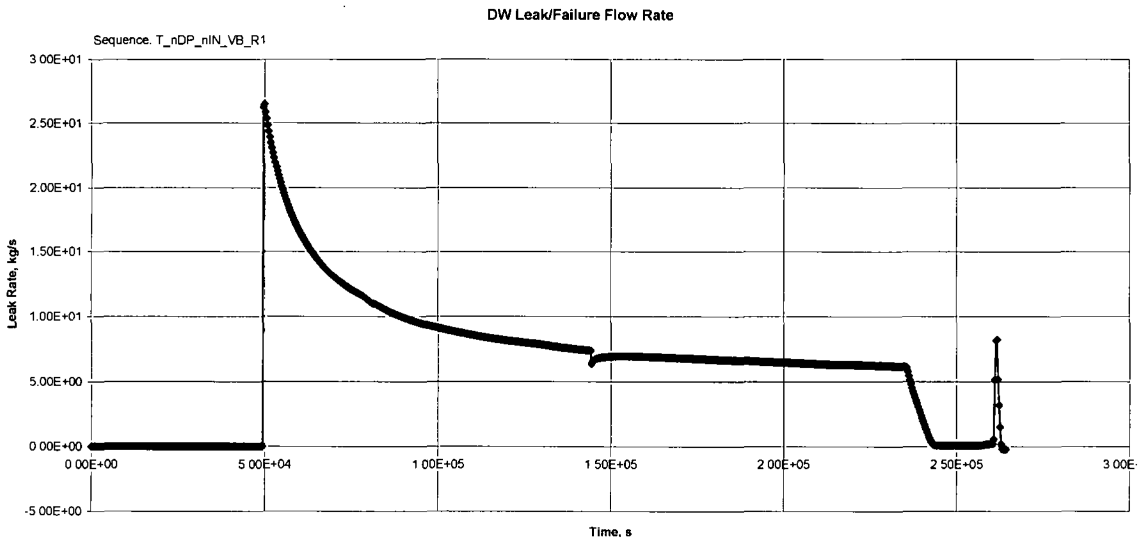


Figure 9A-13r – T_nDP_nIN_VB_R1 Drywell Leak/Failure Flow Rate

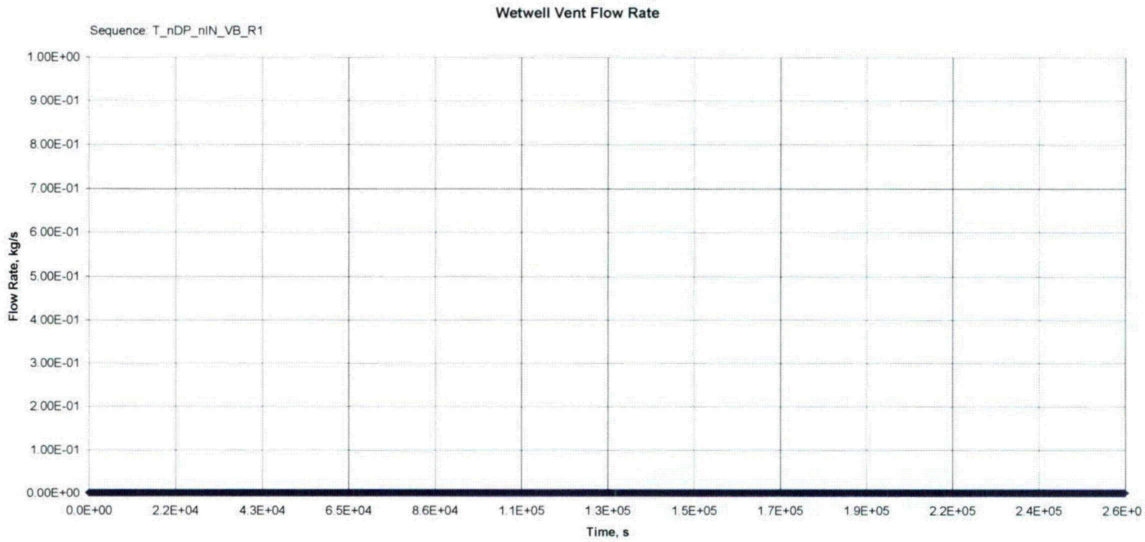


Figure 9A-13s – T_nDP_nIN_VB_R1 Wetwell Vent Flow Rate

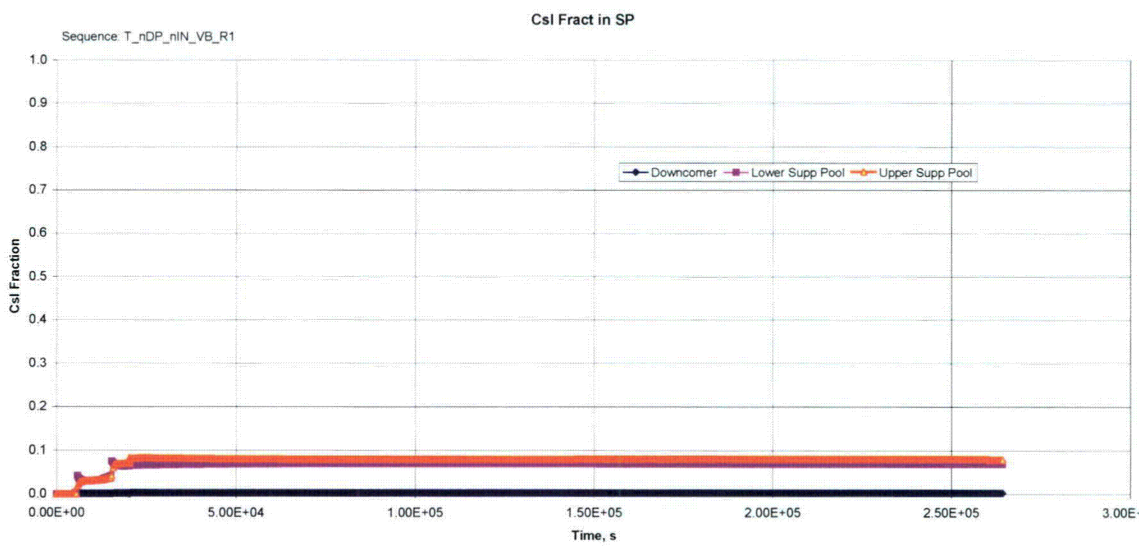


Figure 9A-13t – T_nDP_nIN_VB_R1 CsI Fraction in Suppression Pool

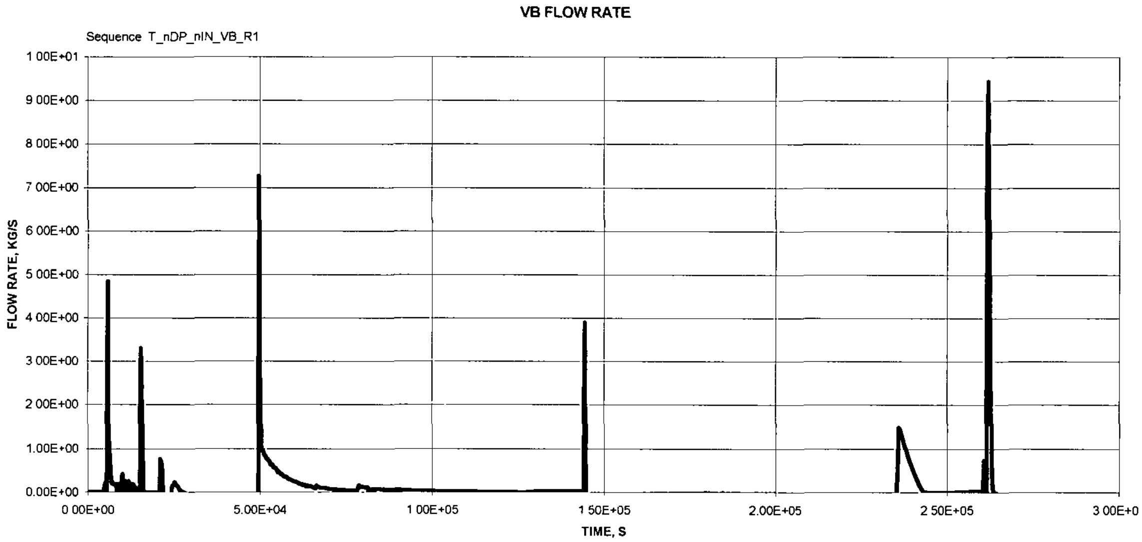


Figure 9A-13u – T_nDP_nIN_VB_R1 Vacuum Breaker Flow Rate

9A.14 Containment Overpressure (VB) - Low Pressure T_nIN_VB_R1

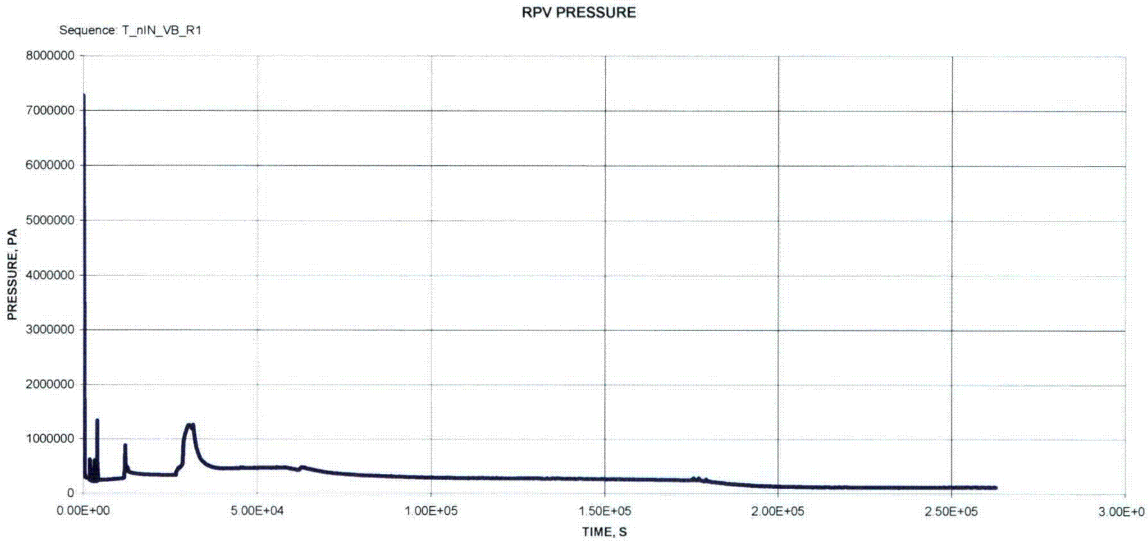


Figure 9A-14a – T_nIN_VB_R1 RPV Pressure

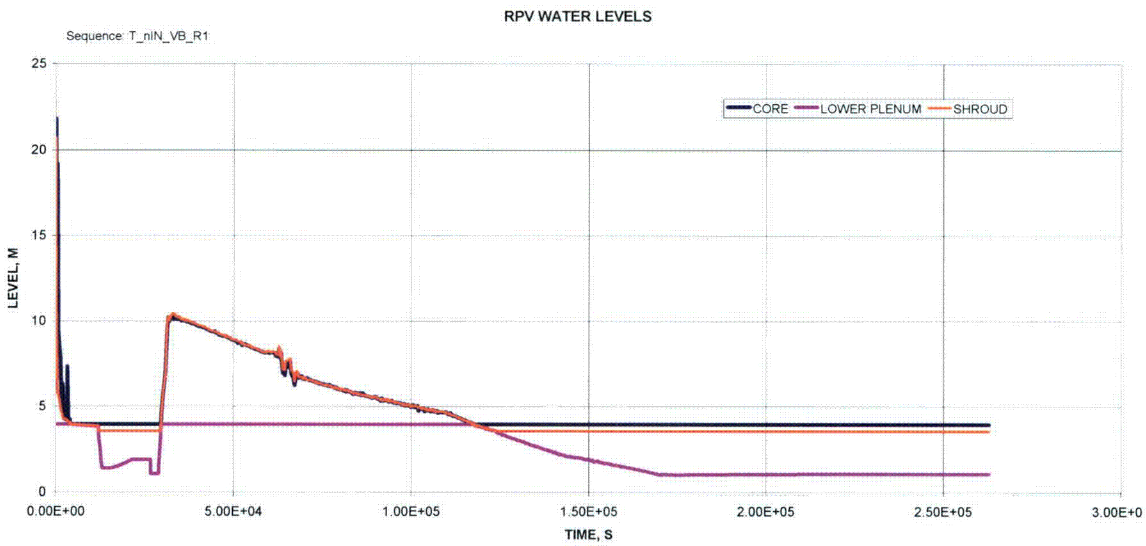


Figure 9A-14b – T_nIN_VB_R1 RPV Water Levels

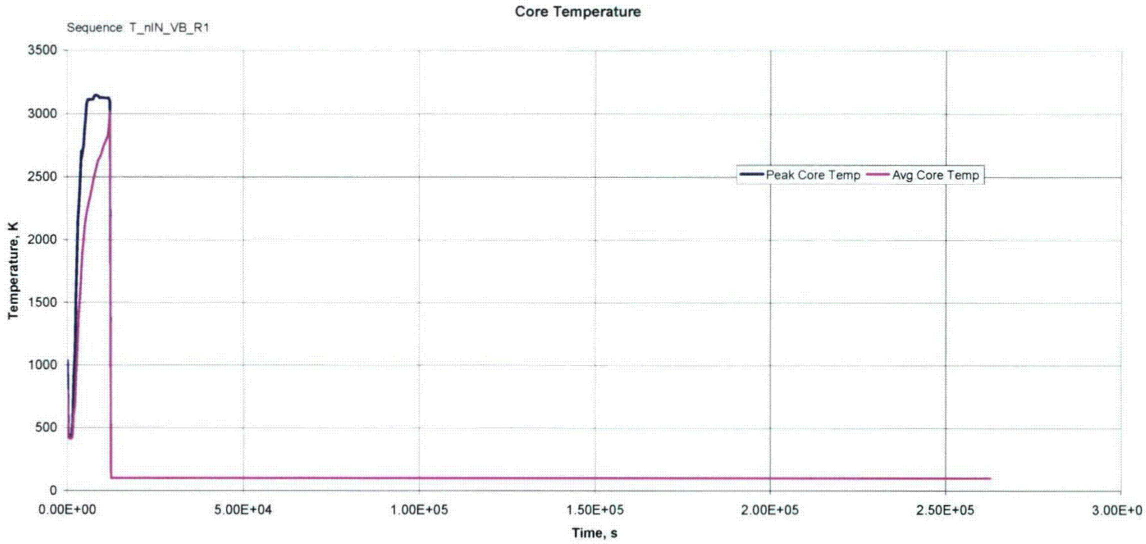


Figure 9A-14c – T_nIN_VB_R1 Core Temperature

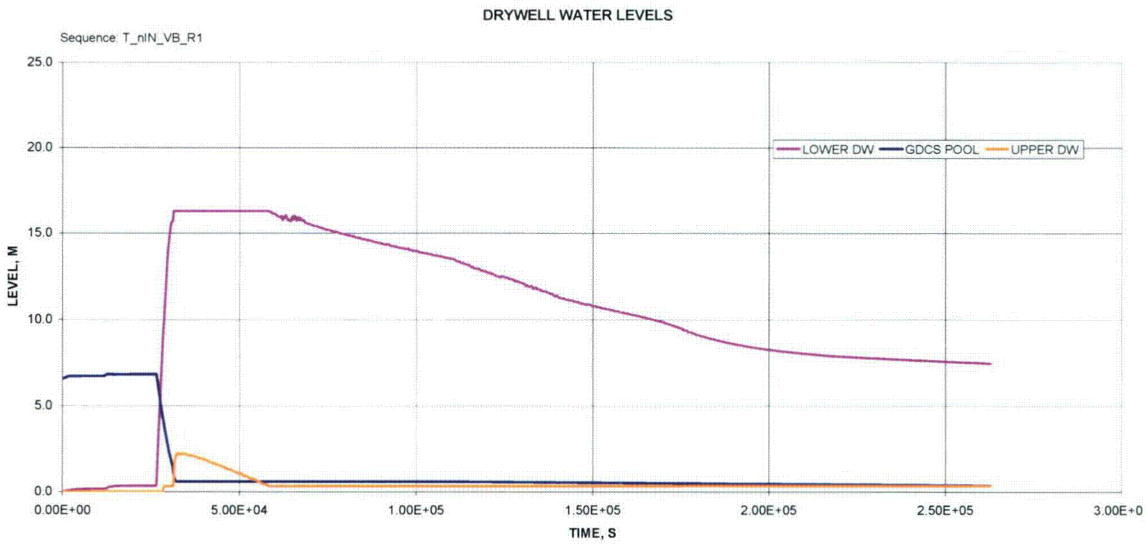


Figure 9A-14d – T_nIN_VB_R1 Drywell Water Levels

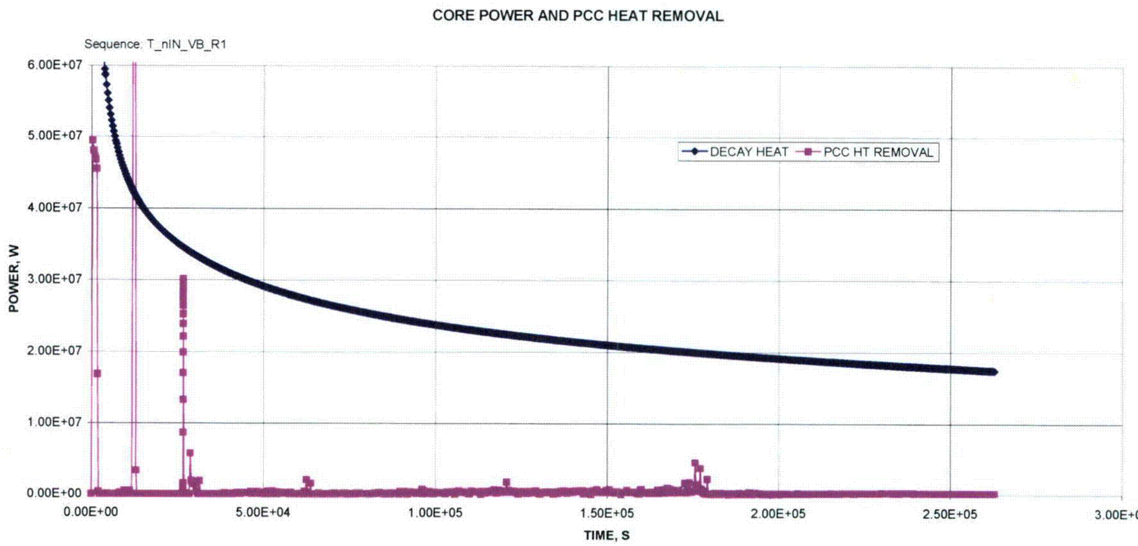


Figure 9A-14e – T_nIN_VB_R1 Core Power and PCCS Heat Removal

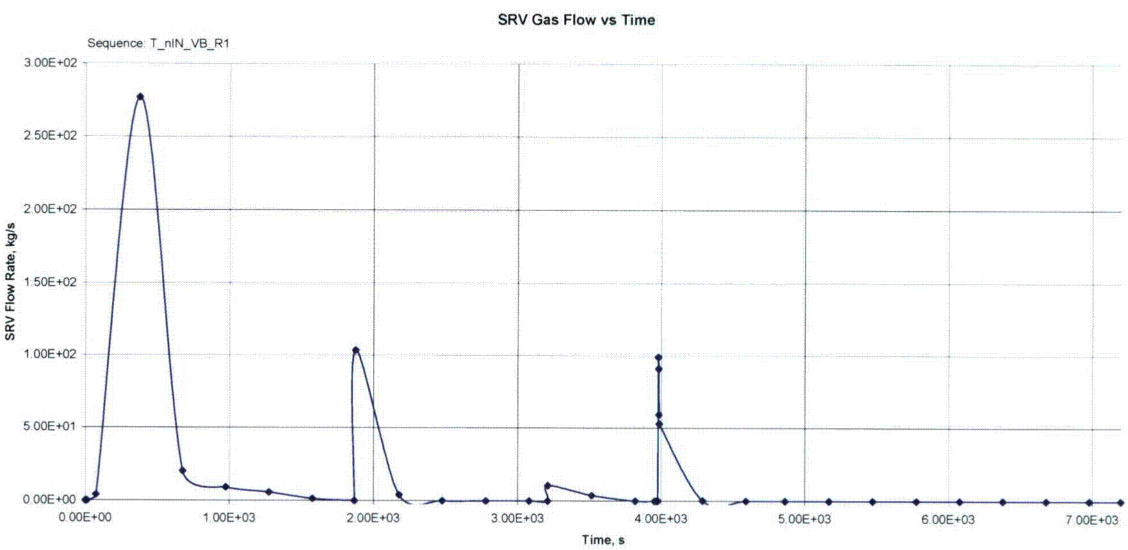


Figure 9A-14f – T_nIN_VB_R1 SRV Gas Flow Rate

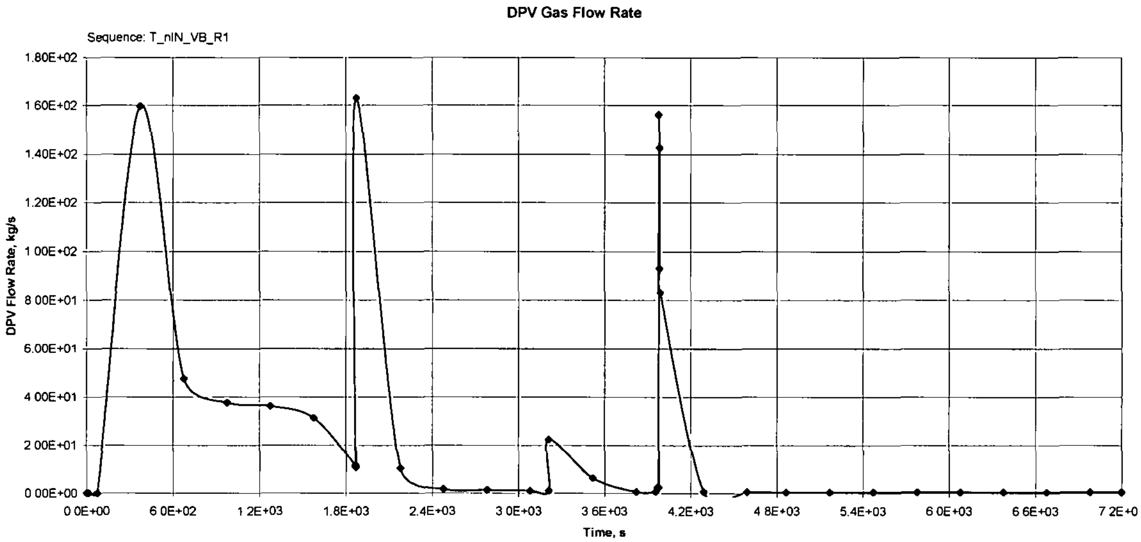


Figure 9A-14g – T_nIN_VB_R1 DPV Gas Flow Rate

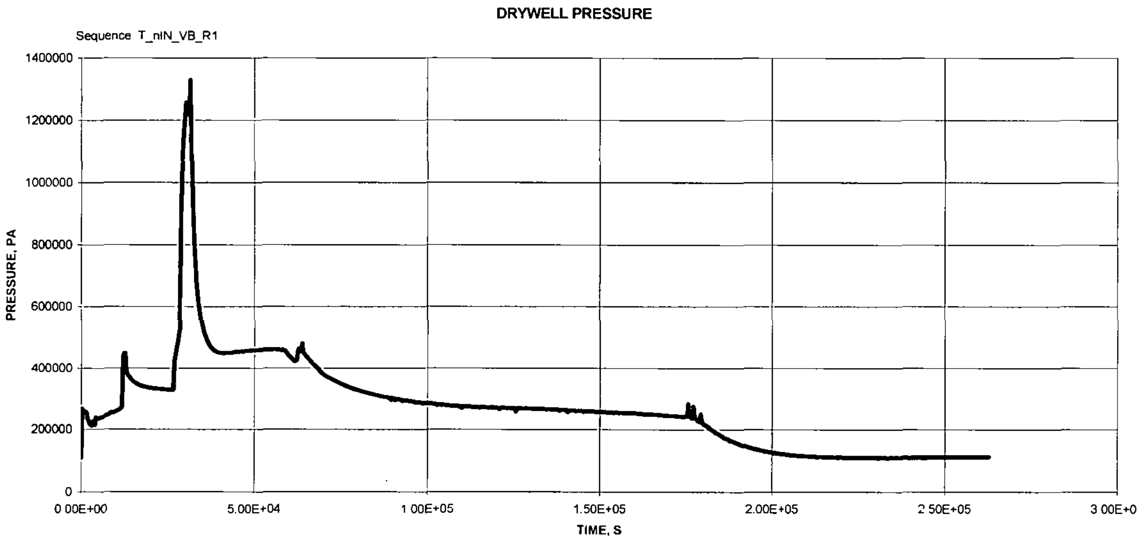


Figure 9A-14h – T_nIN_VB_R1 Drywell Pressure

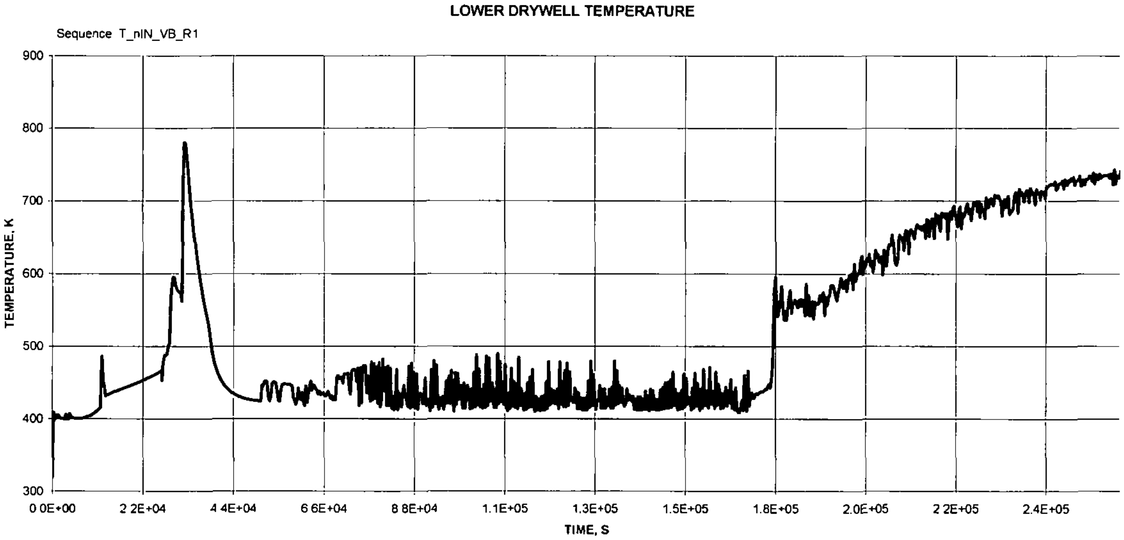


Figure 9A-14i – T_nIN_VB_R1 Lower Drywell Temperature

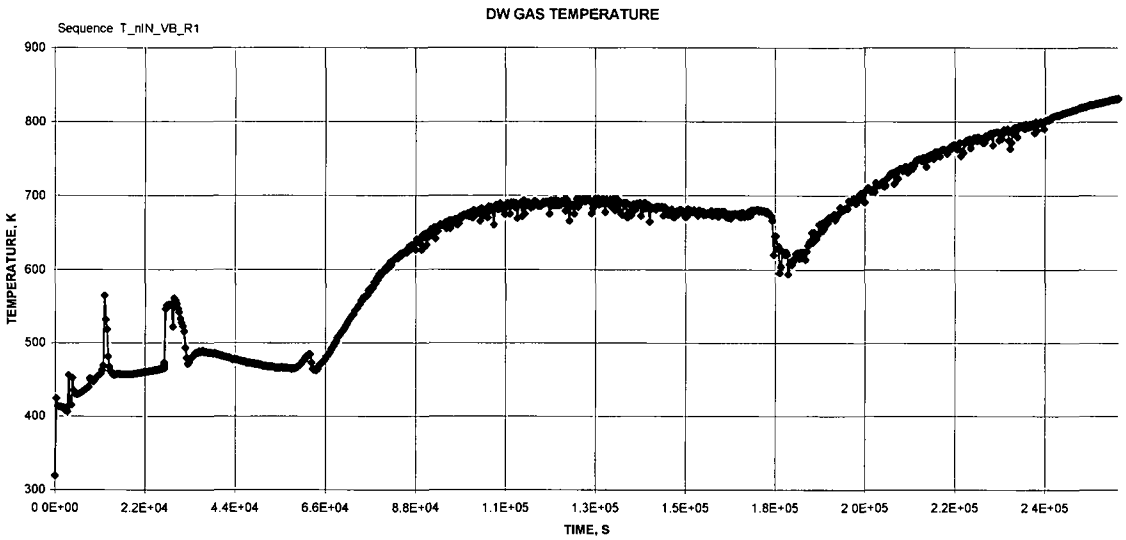


Figure 9A-14j – T_nIN_VB_R1 Drywell Gas Temperature

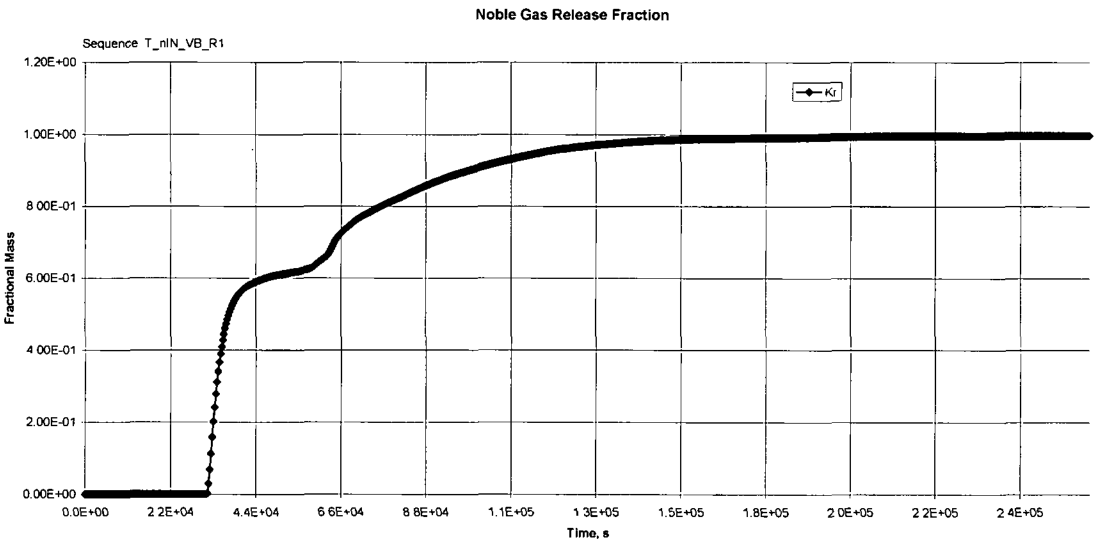


Figure 9A-14k – T_nIN_VB_R1 Noble Gas Release Fraction

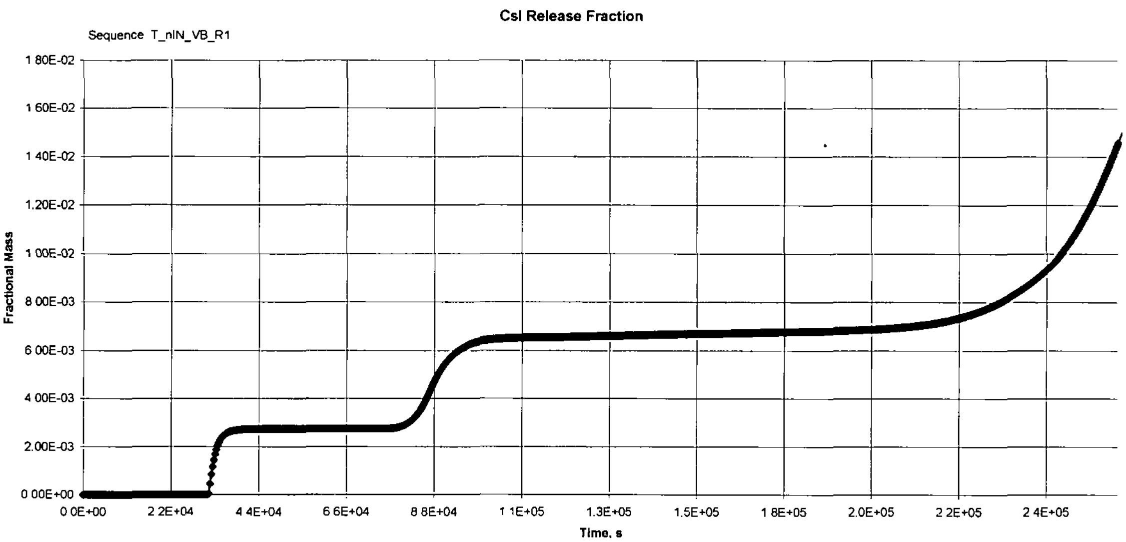


Figure 9A-14l – T_nIN_VB_R1 CsI Release Fraction

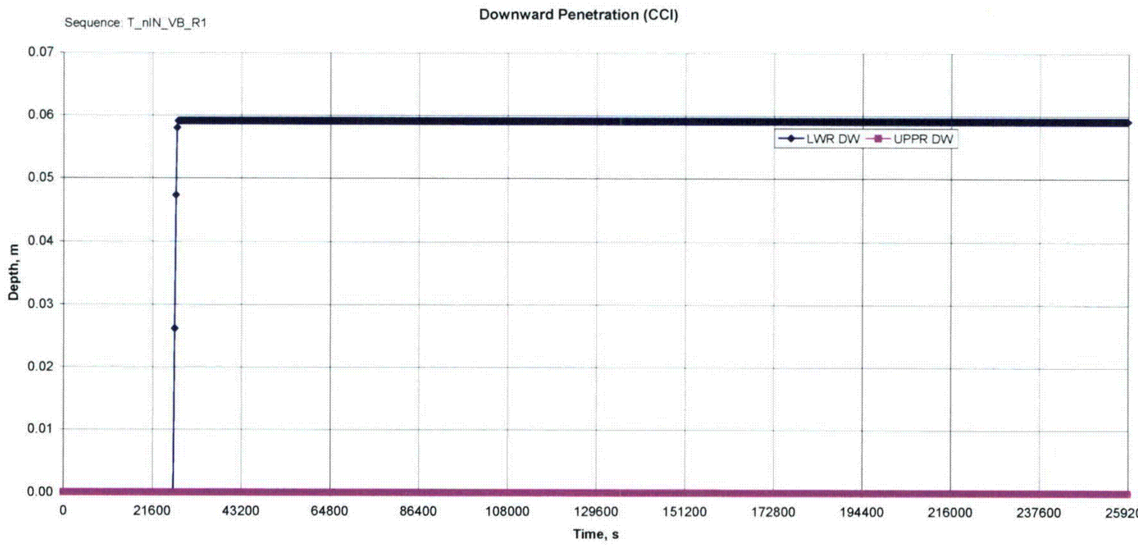


Figure 9A-14m – T_nIN_VB_R1 Downward Penetration (CCI)

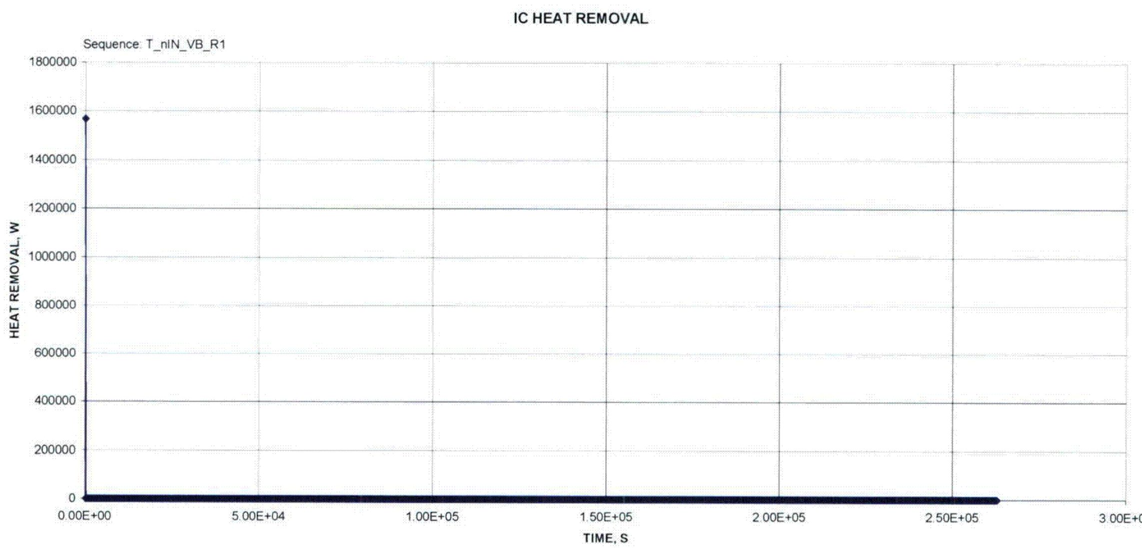


Figure 9A-14n – T_nIN_VB_R1 ICS Heat Removal

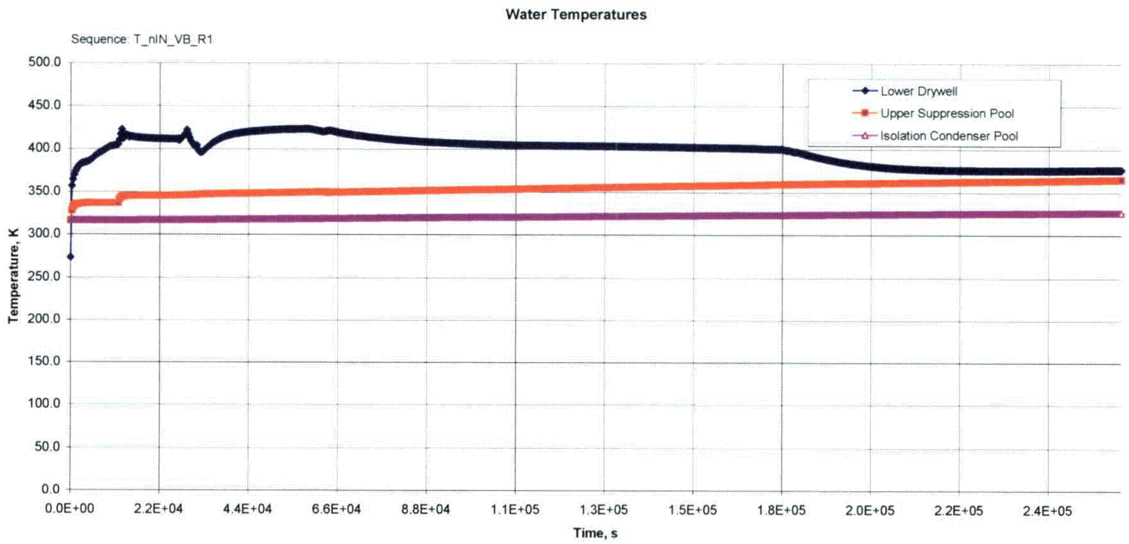


Figure 9A-14o – T_nIN_VB_R1 Water Temperatures

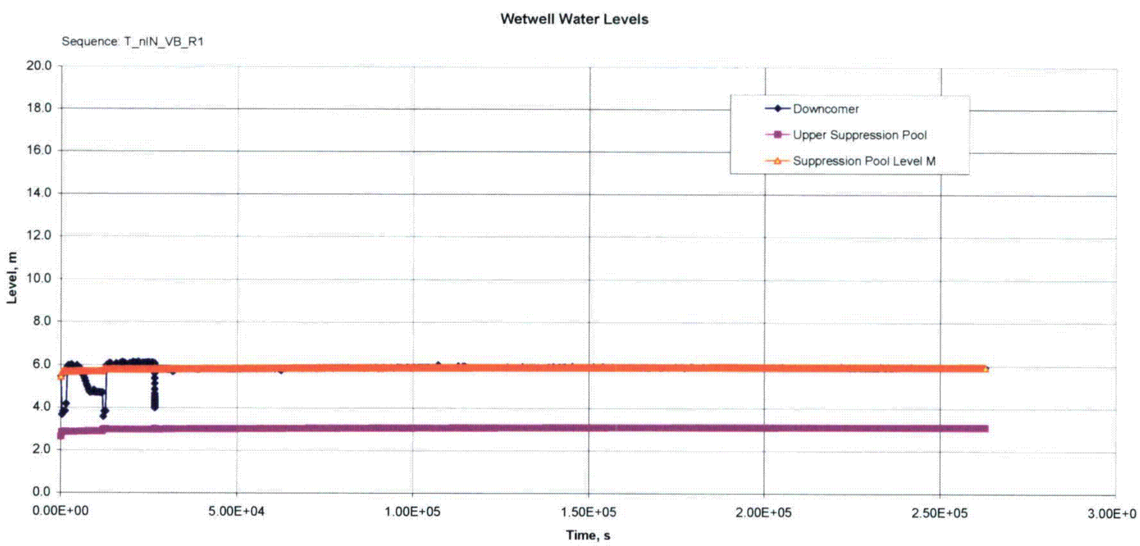


Figure 9A-14p – T_nIN_VB_R1 Wetwell Water Levels

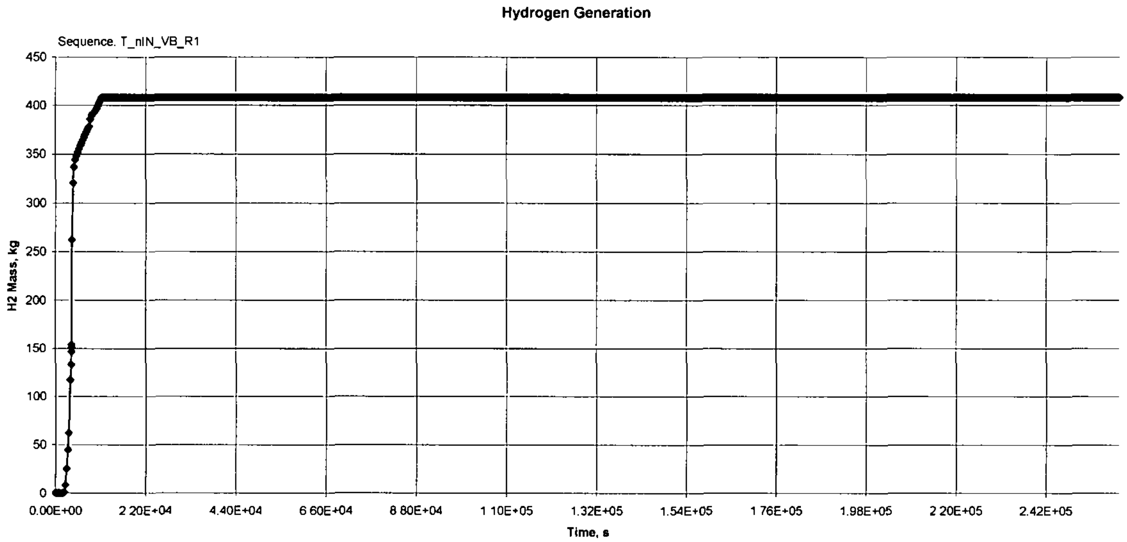


Figure 9A-14q – T_nIN_VB_R1 Hydrogen Generation

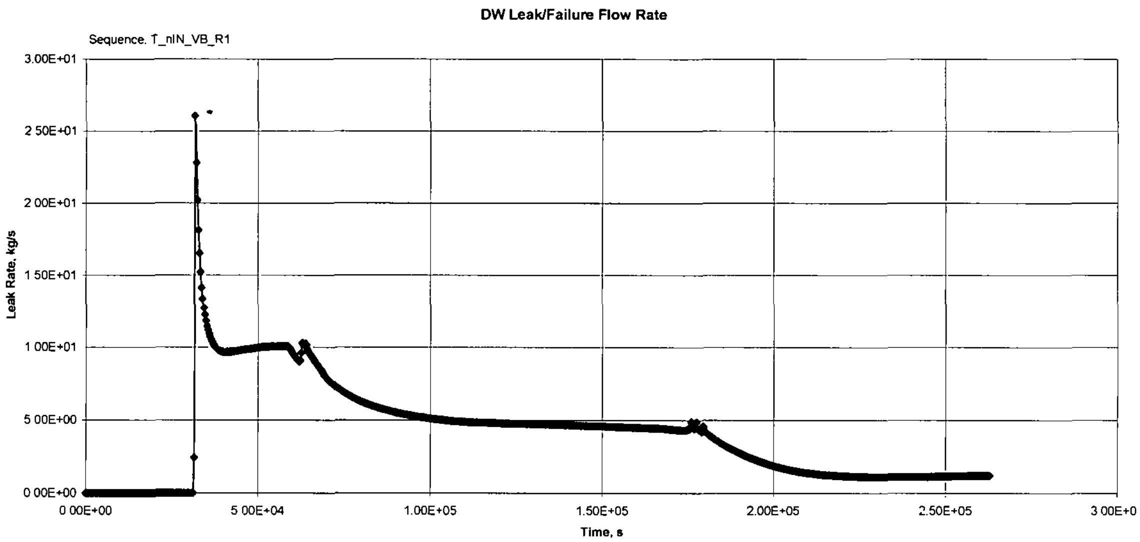


Figure 9A-14r – T_nIN_VB_R1 Drywell Leak/Failure Flow Rate

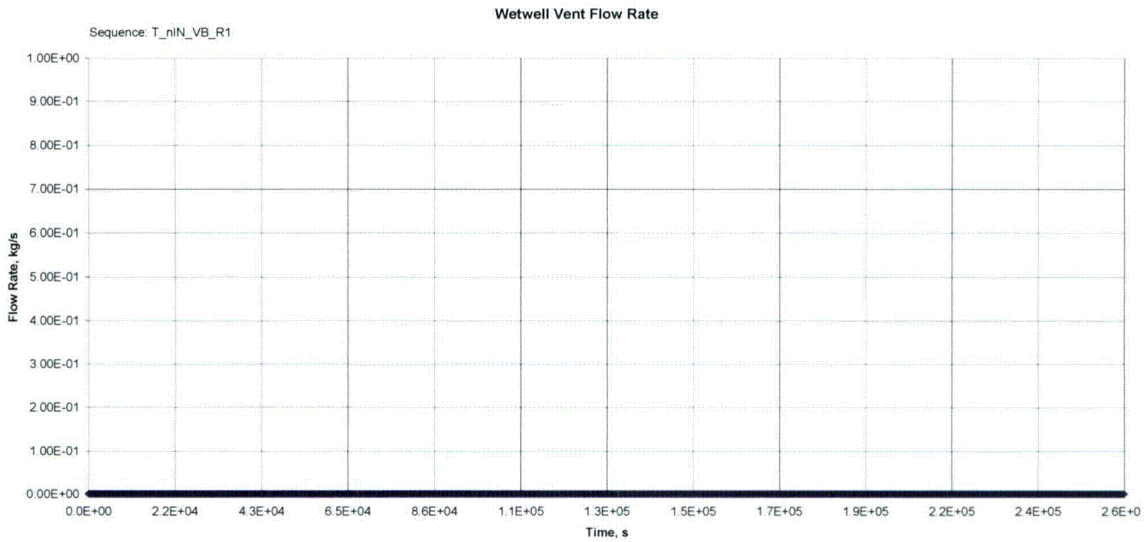


Figure 9A-14s – T_nIN_VB_R1 Wetwell Vent Flow Rate

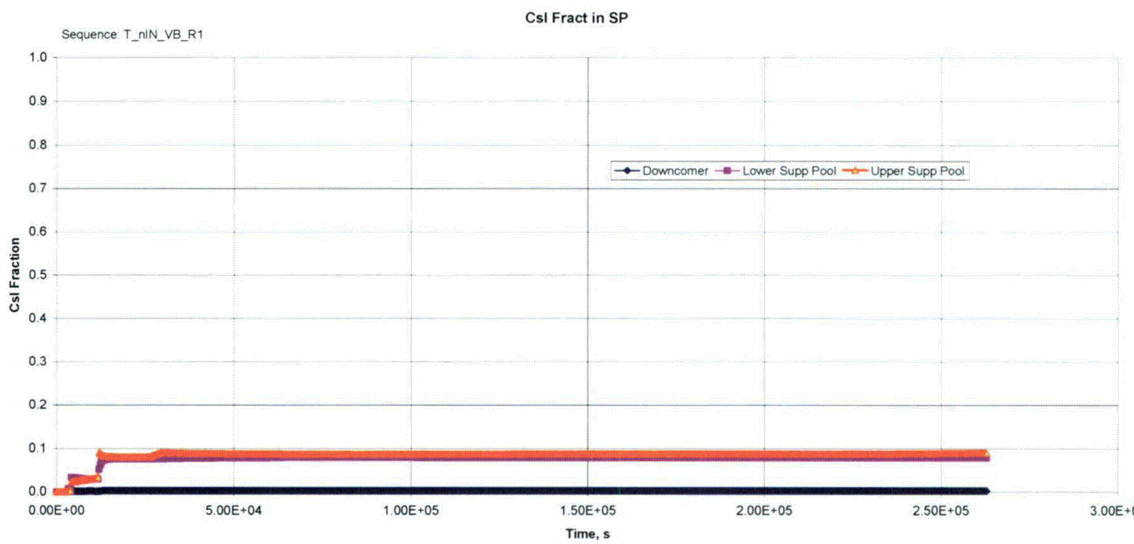


Figure 9A-14t – T_nIN_VB_R1 CsI Release Fraction

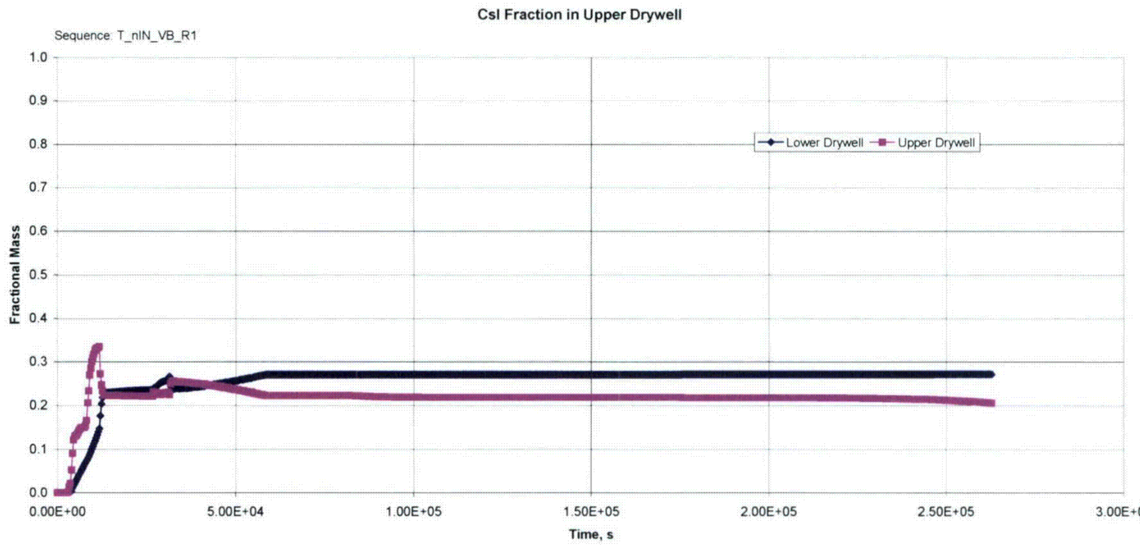


Figure 9A-14u – T_nIN_VB_R1 CsI Fraction in Upper Drywell

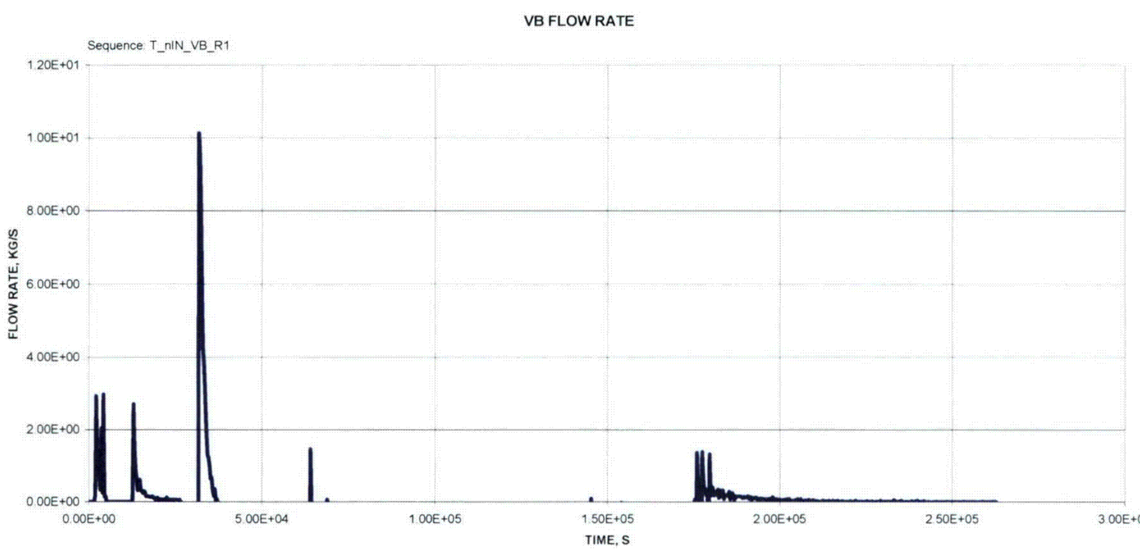


Figure 9A-14v – T_nIN_VB_R1 Vacuum Breaker Flow Rate

9A.15 Containment Overpressure (W1) T_nDP_nIN_nCHR_W1_R1

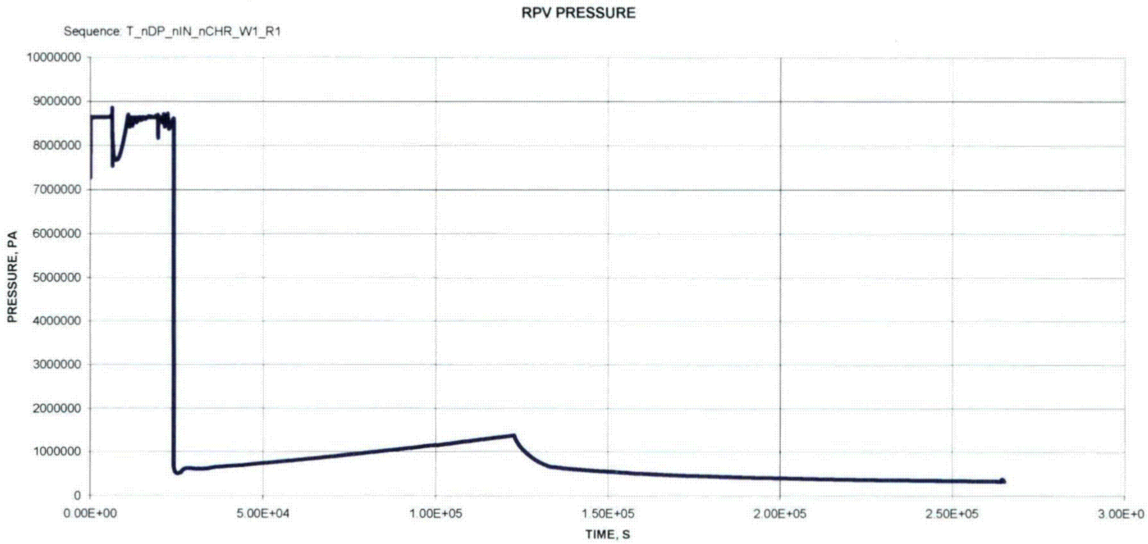


Figure 9A-15a – T_nDP_nIN_nCHR_W1_R1 RPV Pressure

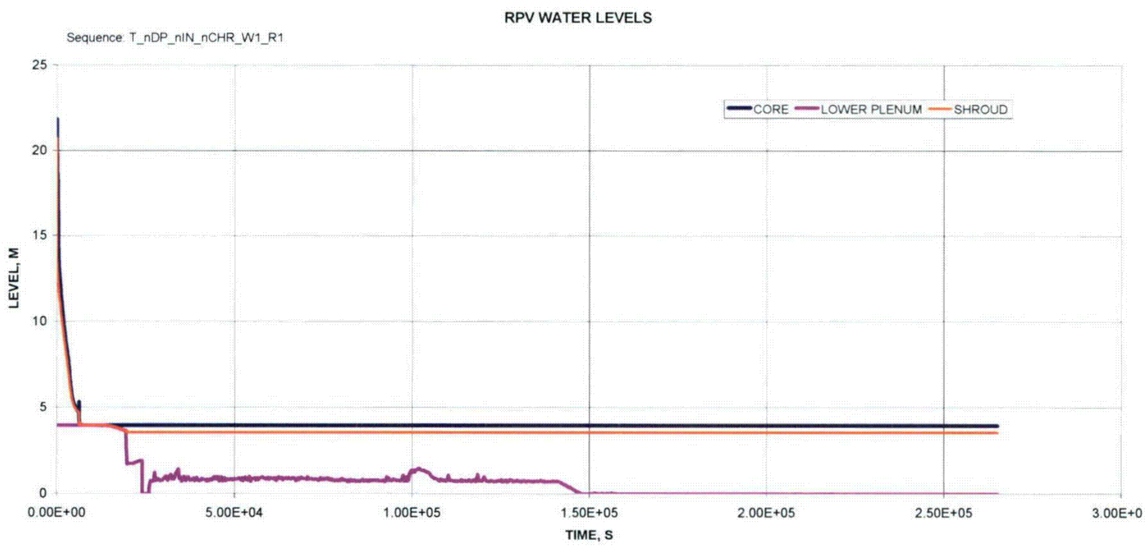


Figure 9A-15b – T_nDP_nIN_nCHR_W1_R1 RPV Water Levels

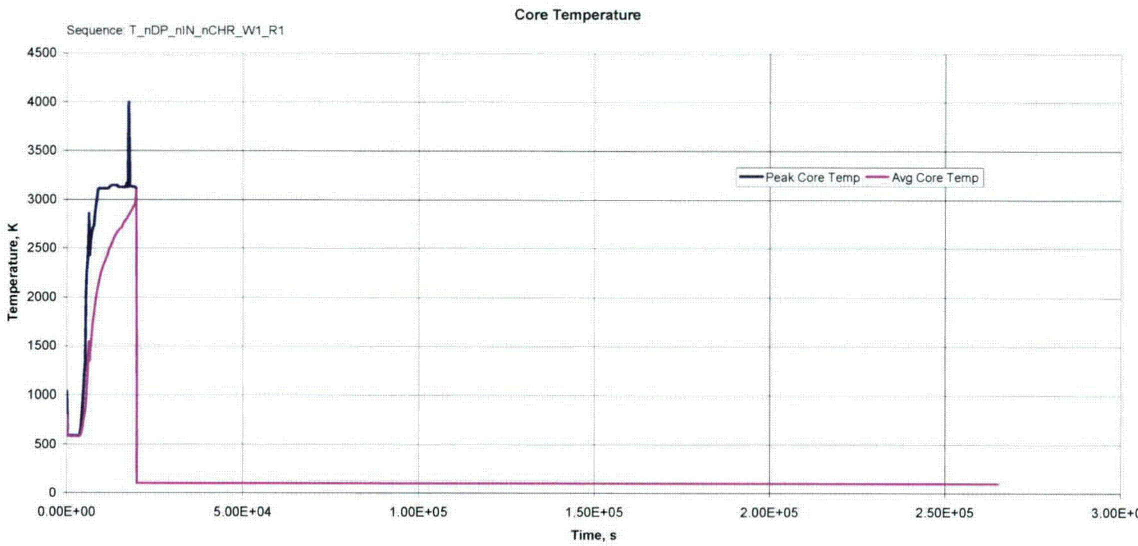


Figure 9A-15c – T_nDP_nIN_nCHR_W1_R1 Core Temperature

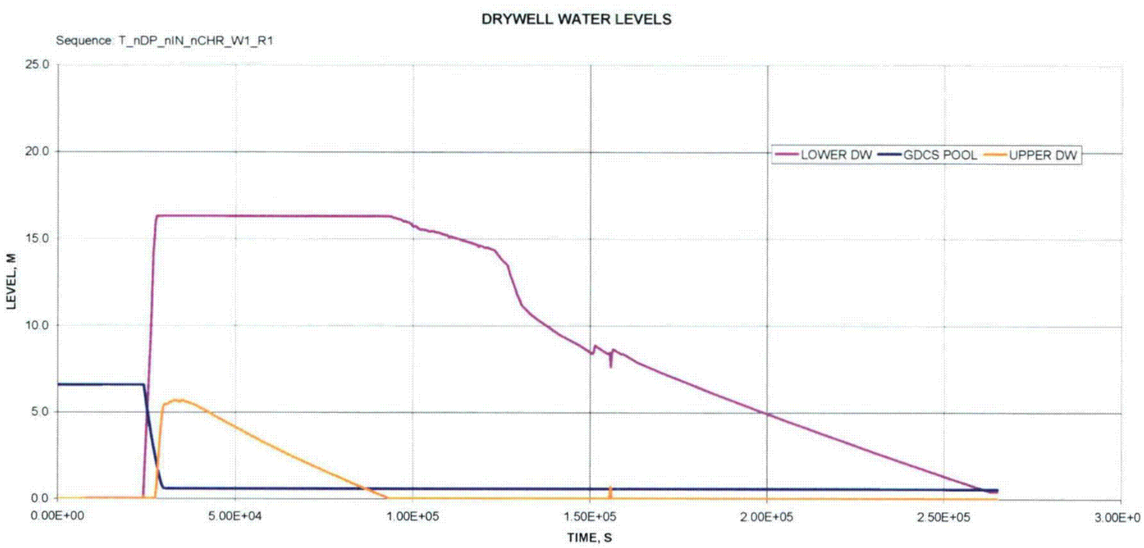


Figure 9A-15d – T_nDP_nIN_nCHR_W1_R1 Drywell Water Levels

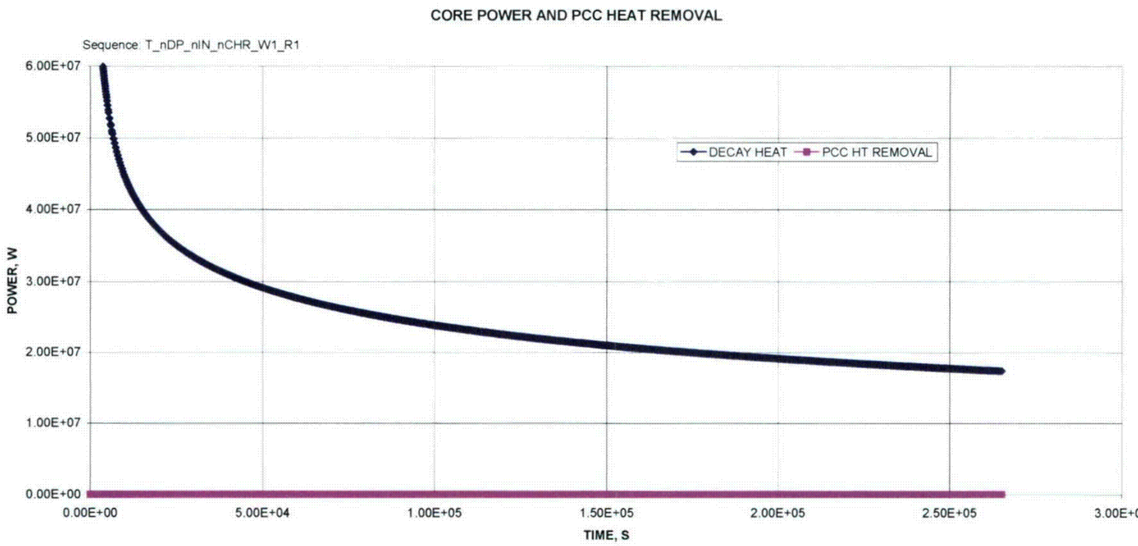


Figure 9A-15e – T_nDP_nIN_nCHR_W1_R1 Core Power and PCCS Heat Removal

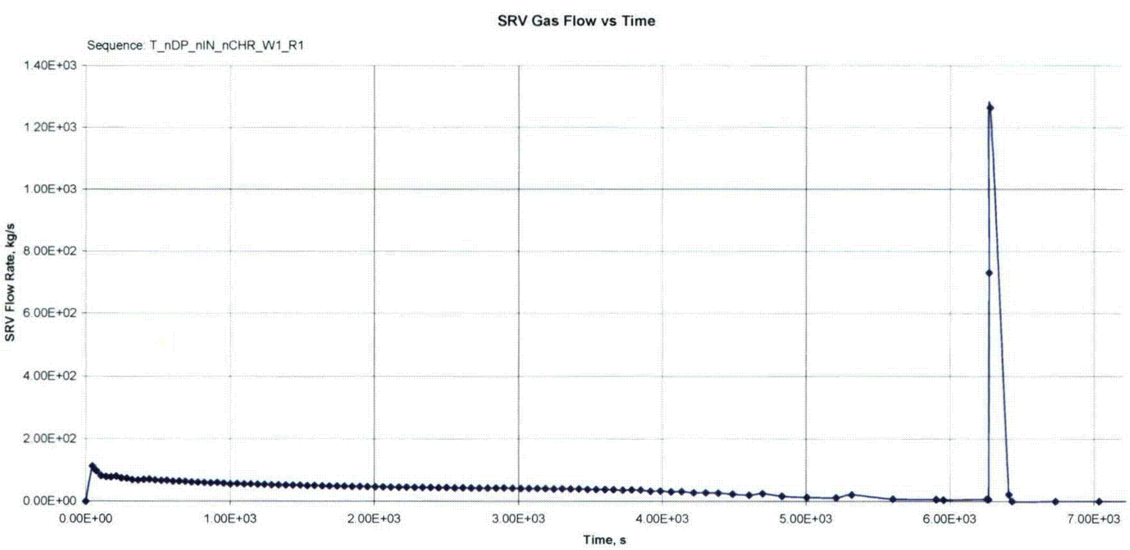


Figure 9A-15f – T_nDP_nIN_nCHR_W1_R1 SRV Gas Flow Rate

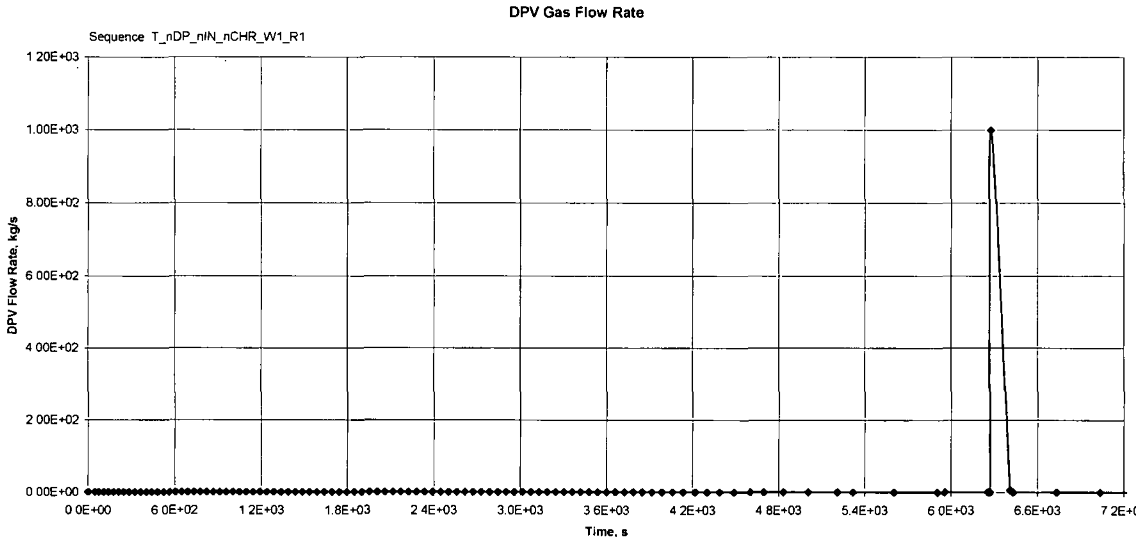


Figure 9A-15g – T_nDP_nIN_nCHR_W1_R1 DPV Gas Flow Rate

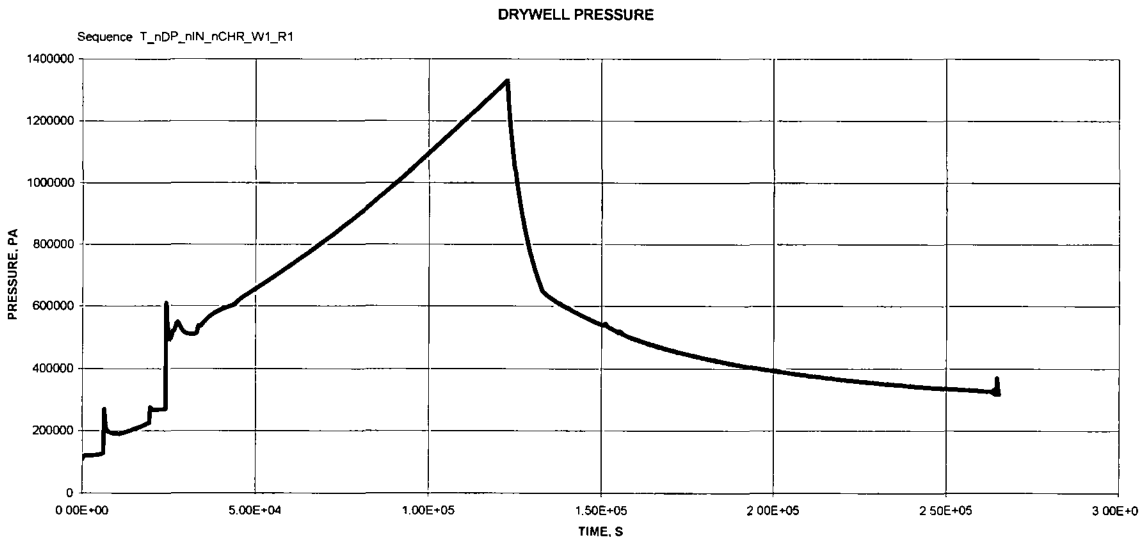


Figure 9A-15h – T_nDP_nIN_nCHR_W1_R1 Drywell Pressure

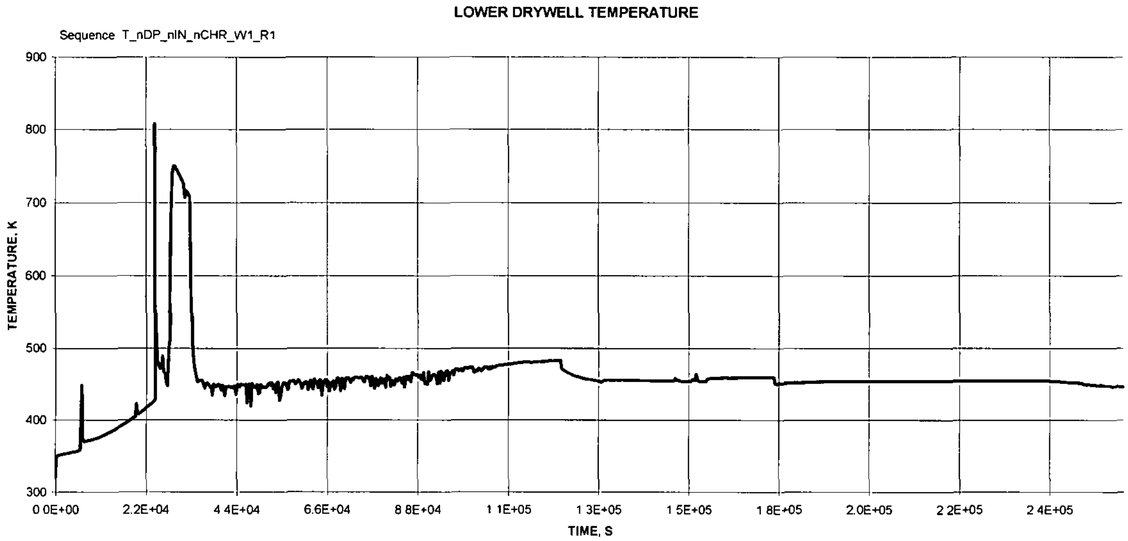


Figure 9A-15i – T_nDP_nIN_nCHR_W1_R1 Lower Drywell Temperature

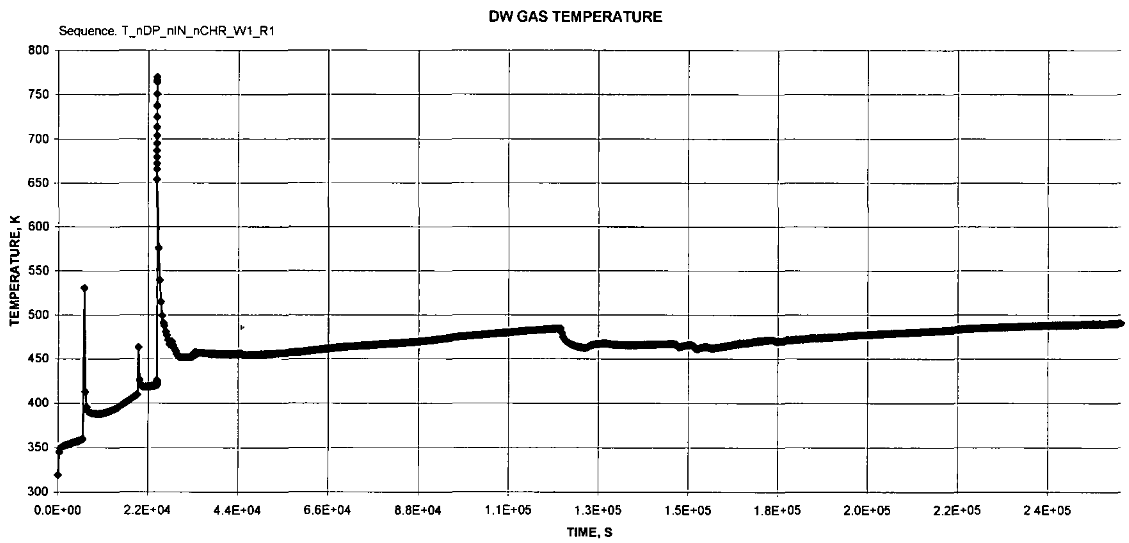


Figure 9A-15j – T_nDP_nIN_nCHR_W1_R1 Drywell Gas Temperature

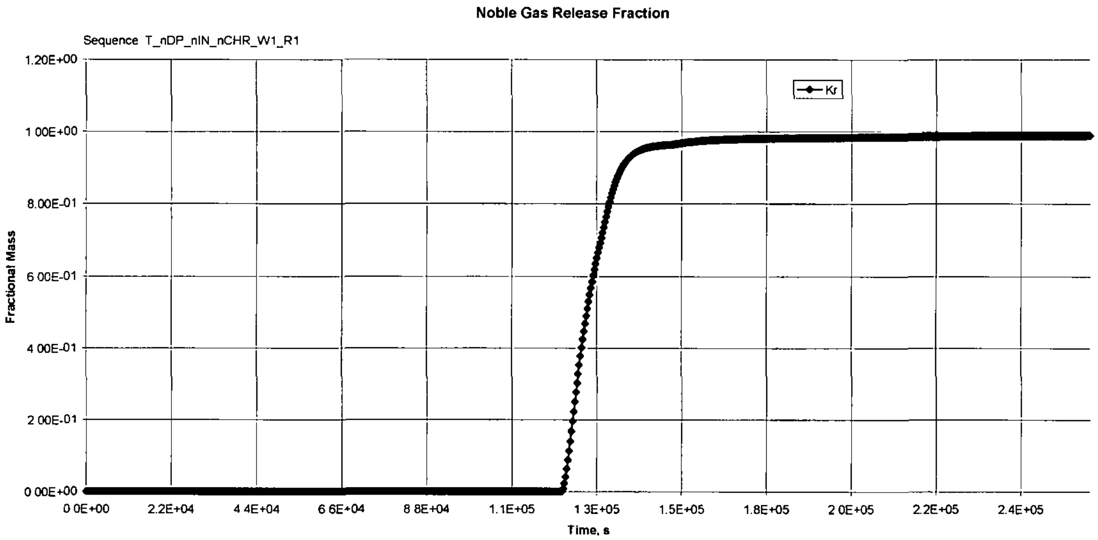


Figure 9A-15k – T_nDP_nIN_nCHR_W1_R1 Noble Gas Release Fraction

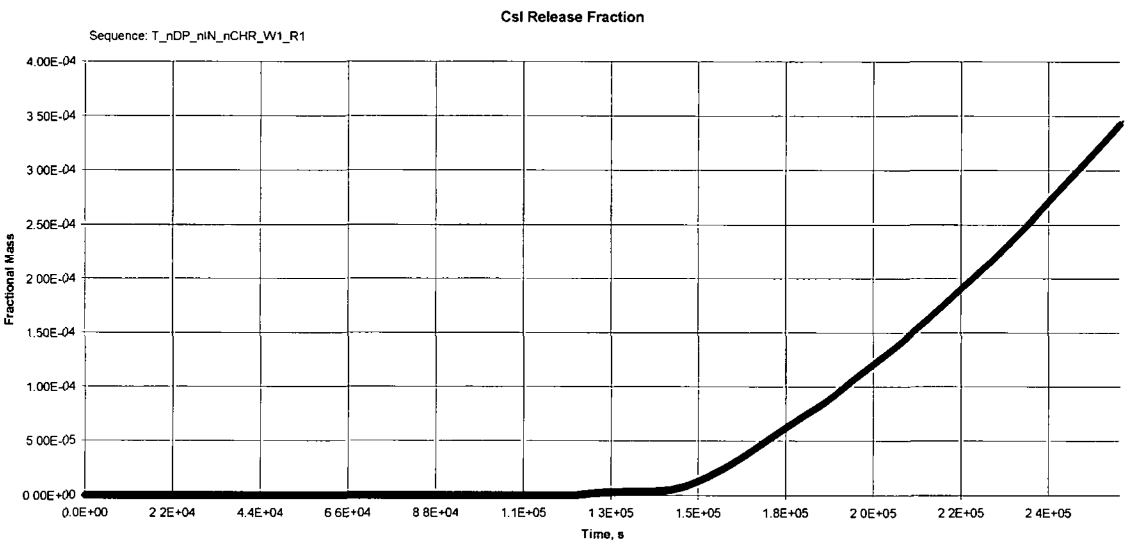


Figure 9A-15l – T_nDP_nIN_nCHR_W1_R1 CsI Release Fraction

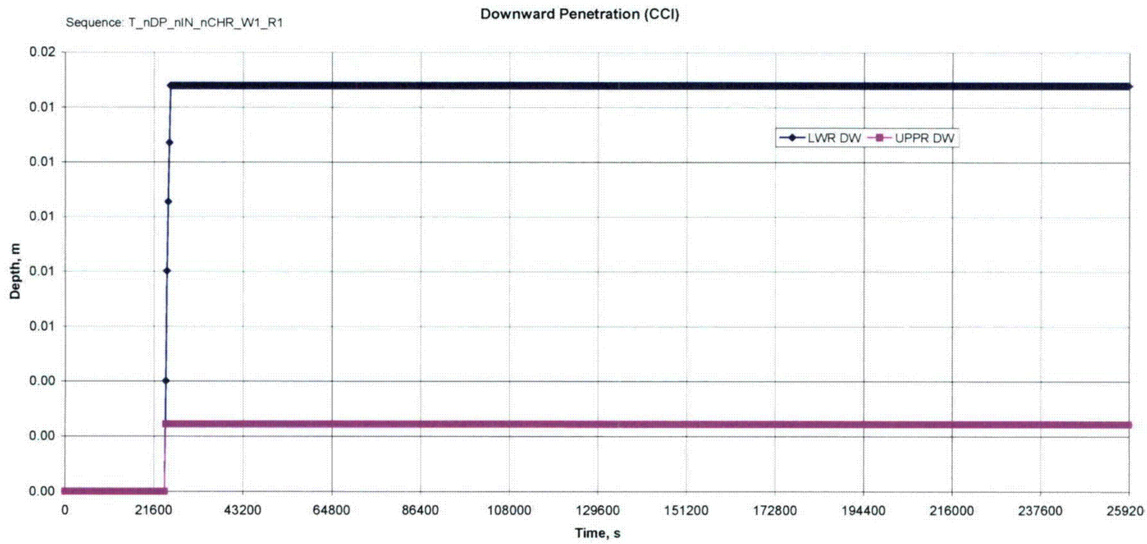


Figure 9A-15m – T_nDP_nIN_nCHR_W1_R1 Downward Penetration (CCI)

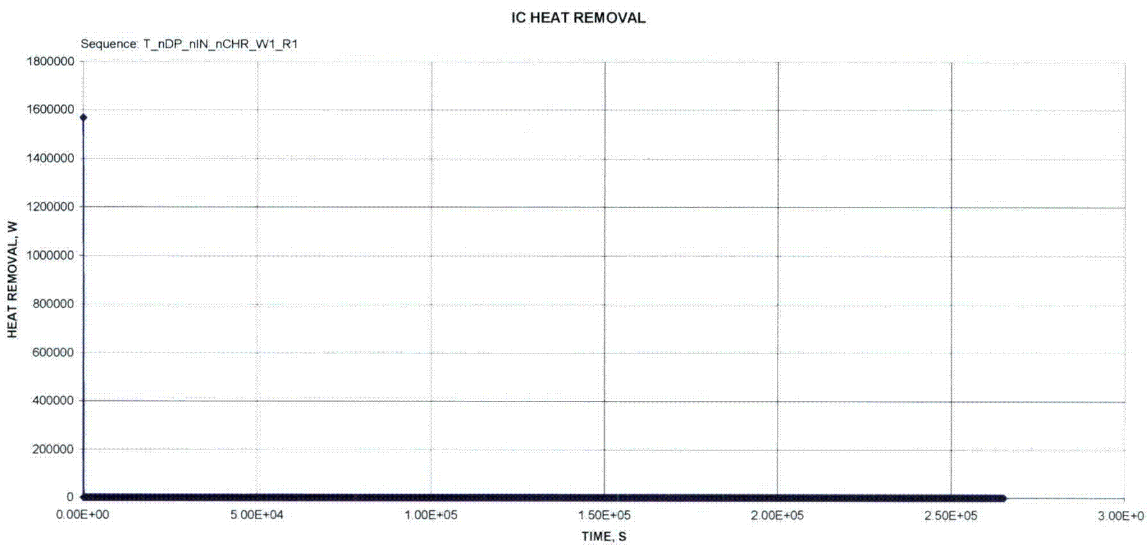


Figure 9A-15n – T_nDP_nIN_nCHR_W1_R1 ICS Heat Removal

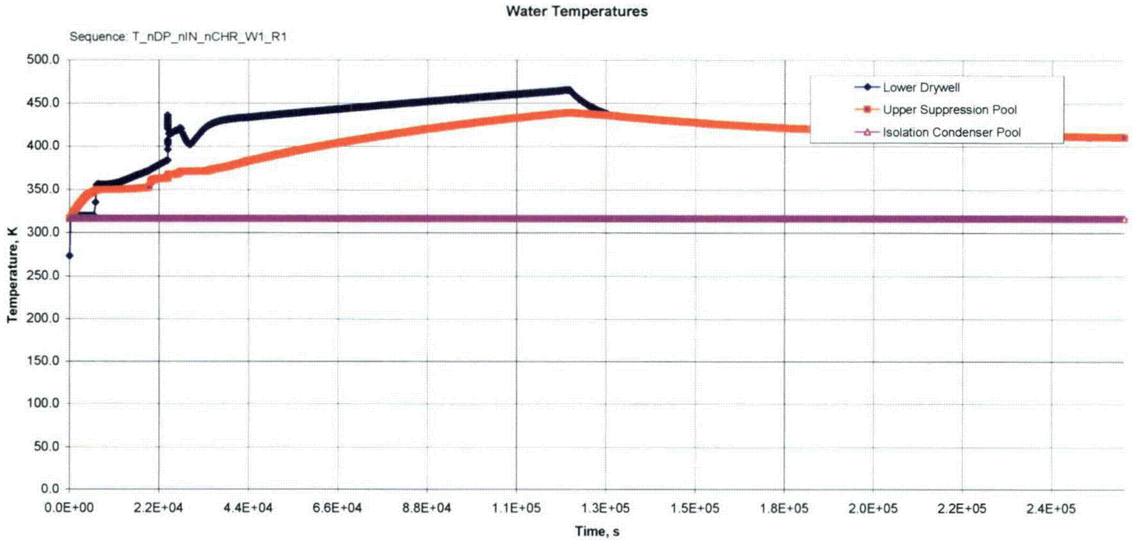


Figure 9A-15o – T_nDP_nIN_nCHR_W1_R1 Water Temperatures

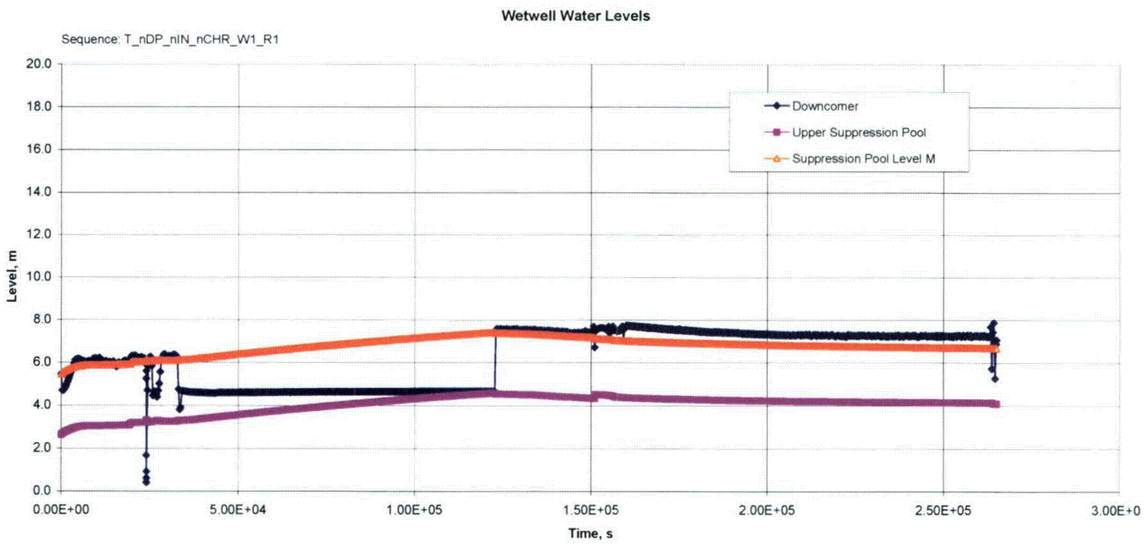


Figure 9A-15p – T_nDP_nIN_nCHR_W1_R1 Wetwell Water Levels

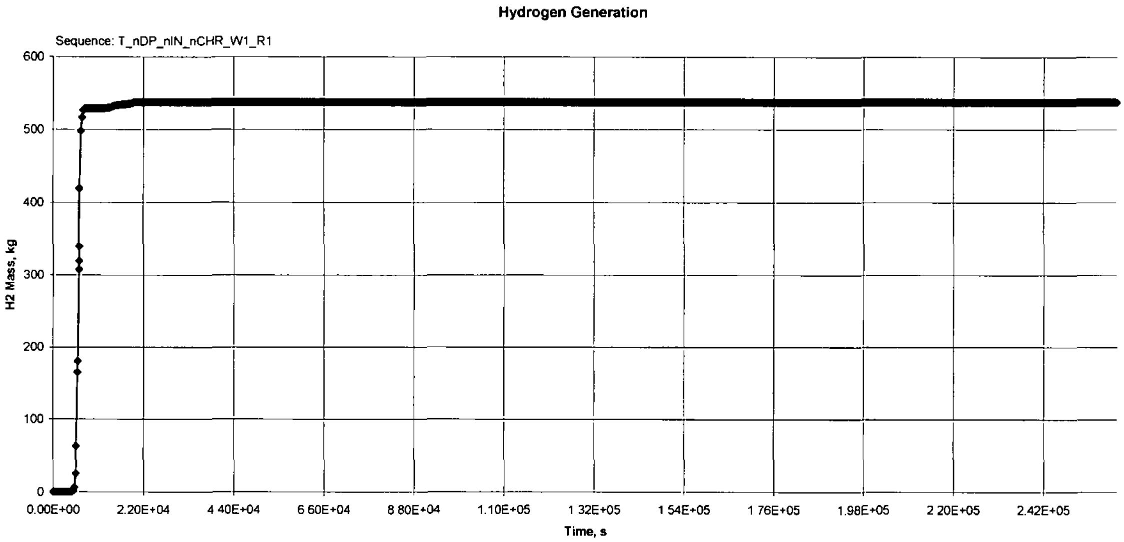


Figure 9A-15q – T_nDP_nIN_nCHR_W1_R1 Hydrogen Generation

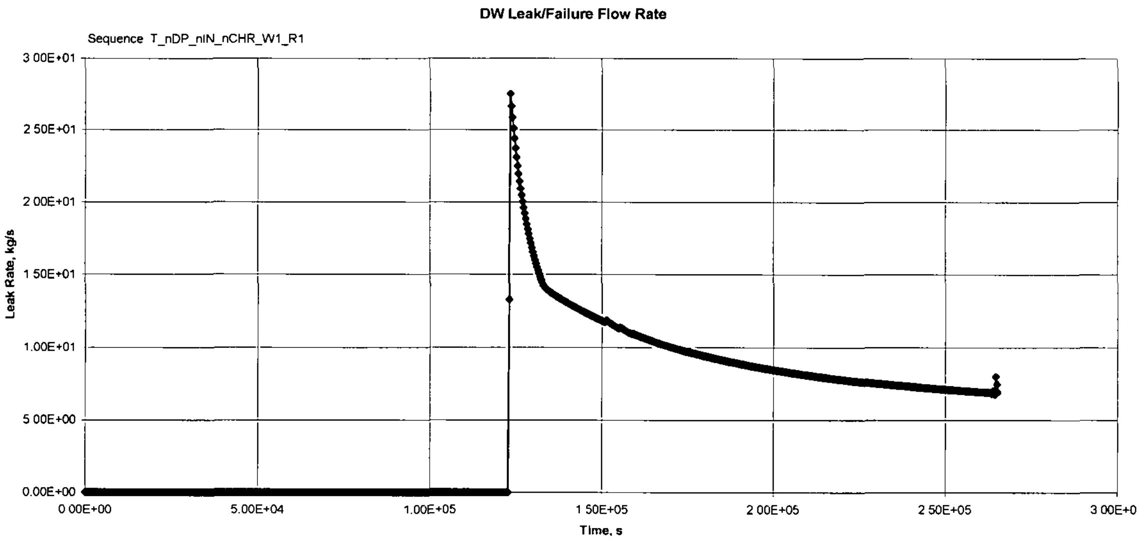


Figure 9A-15r – T_nDP_nIN_nCHR_W1_R1 Drywell Leak/Failure Flow Rate

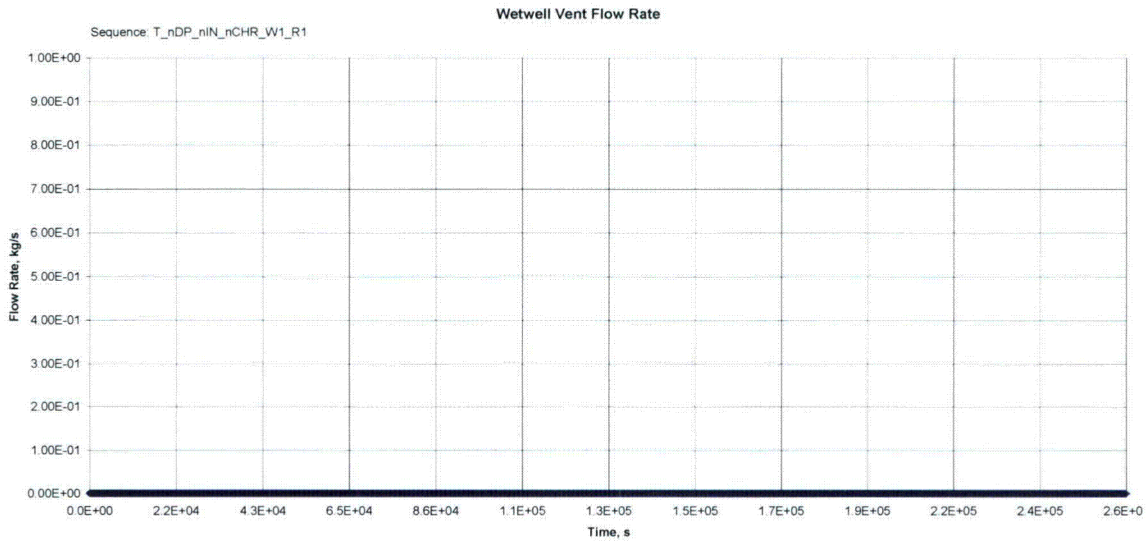


Figure 9A-15s – T_nDP_nIN_nCHR_W1_R1 Wetwell Vent Flow Rate

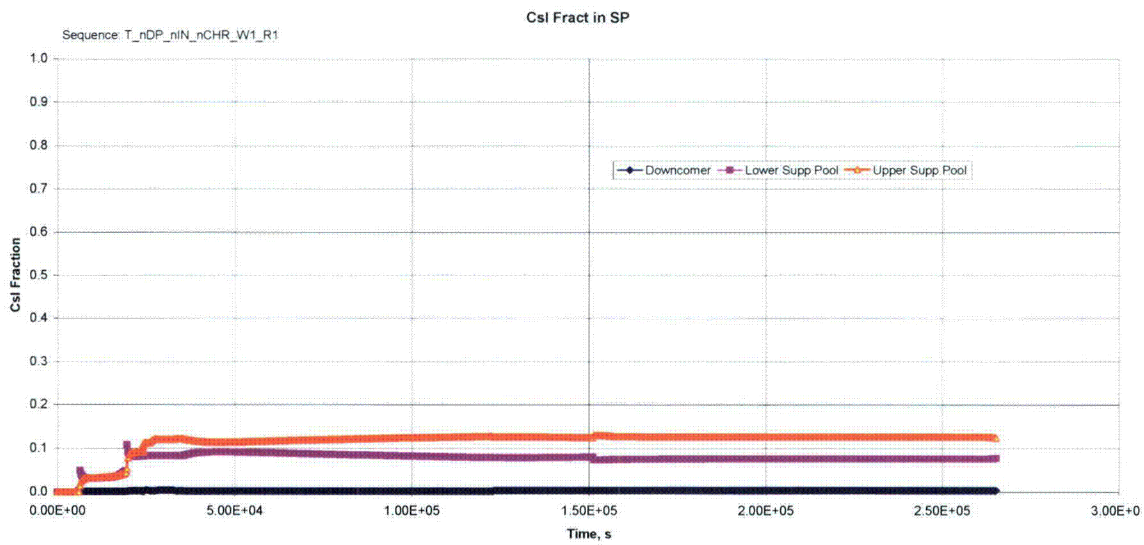


Figure 9A-15t – T_nDP_nIN_nCHR_W1_R1 CsI Fraction in Suppression Pool

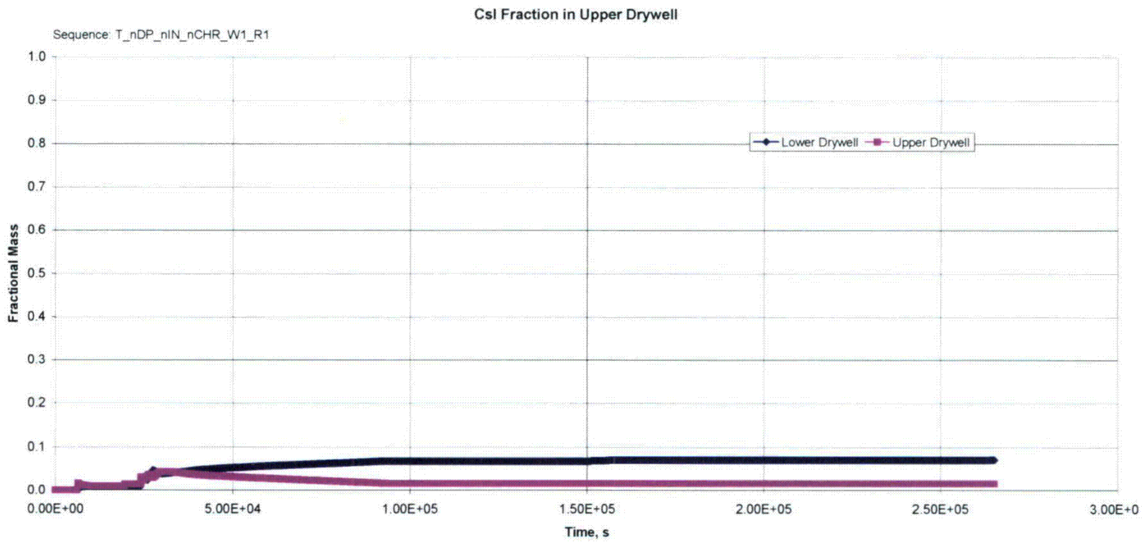


Figure 9A-15u – T_nDP_nIN_nCHR_W1_R1 CsI Fraction in Upper Drywell

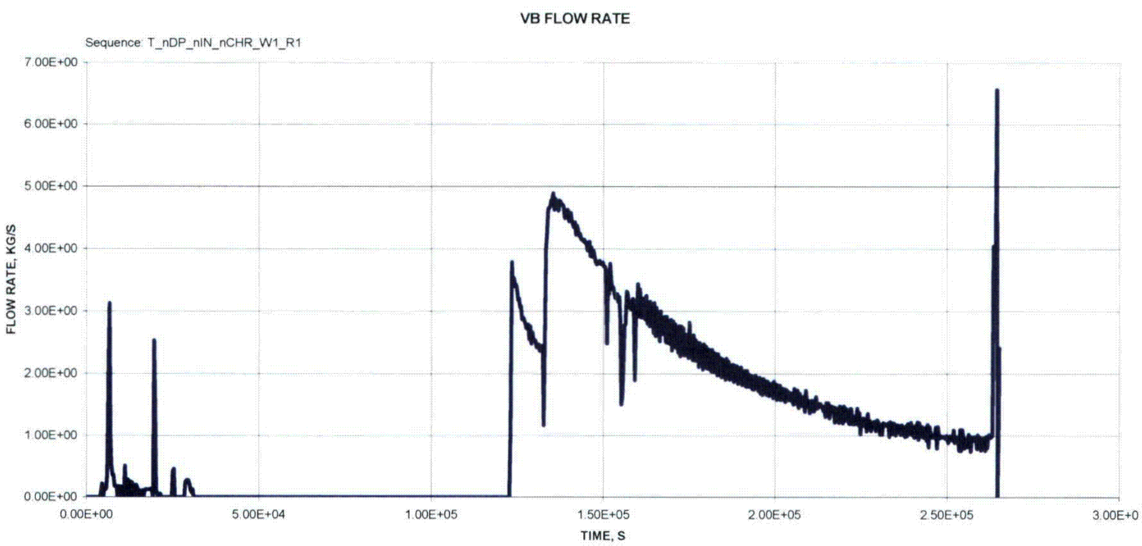


Figure 9A-15v – T_nDP_nIN_nCHR_W1_R1 Vacuum Breaker Flow Rate

9A.16 Containment Overpressure (W2) T_nDP_nIN_nCHR_W2_R1

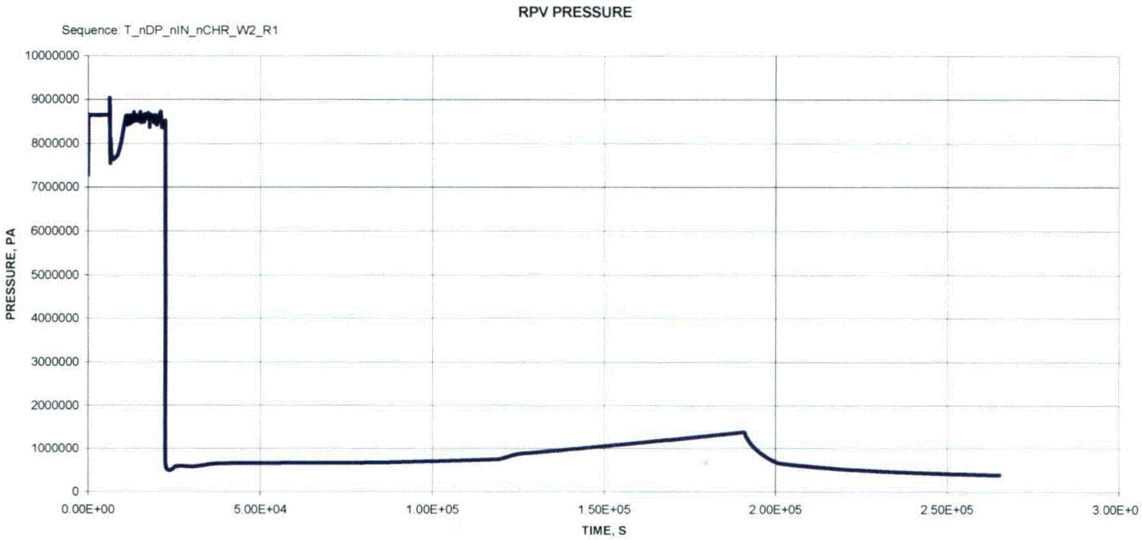


Figure 9A-16a – T_nDP_nIN_nCHR_W2_R1 RPV Pressure

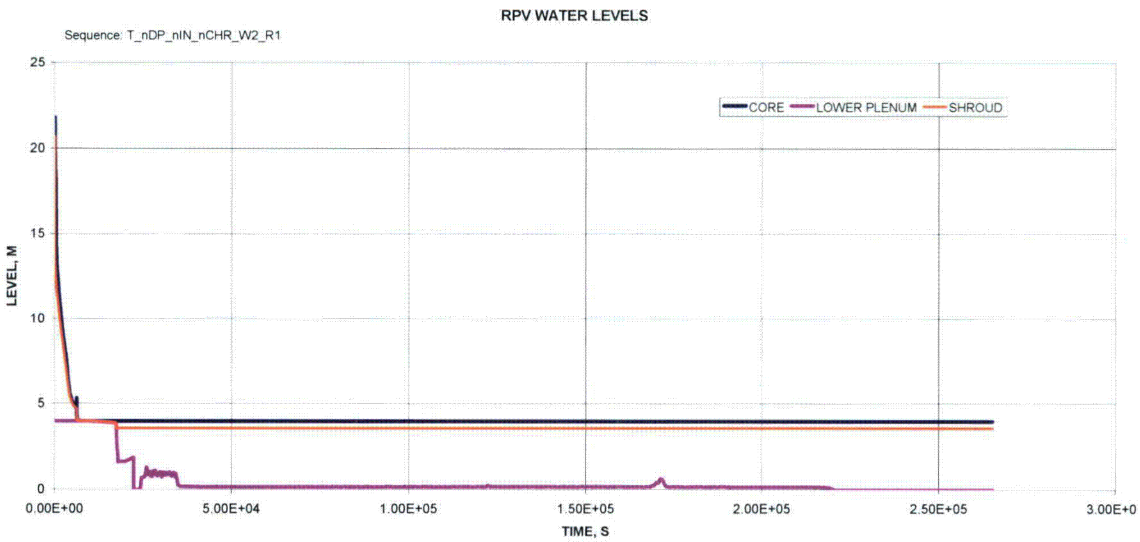


Figure 9A-16b – T_nDP_nIN_nCHR_W2_R1 RPV Water Levels

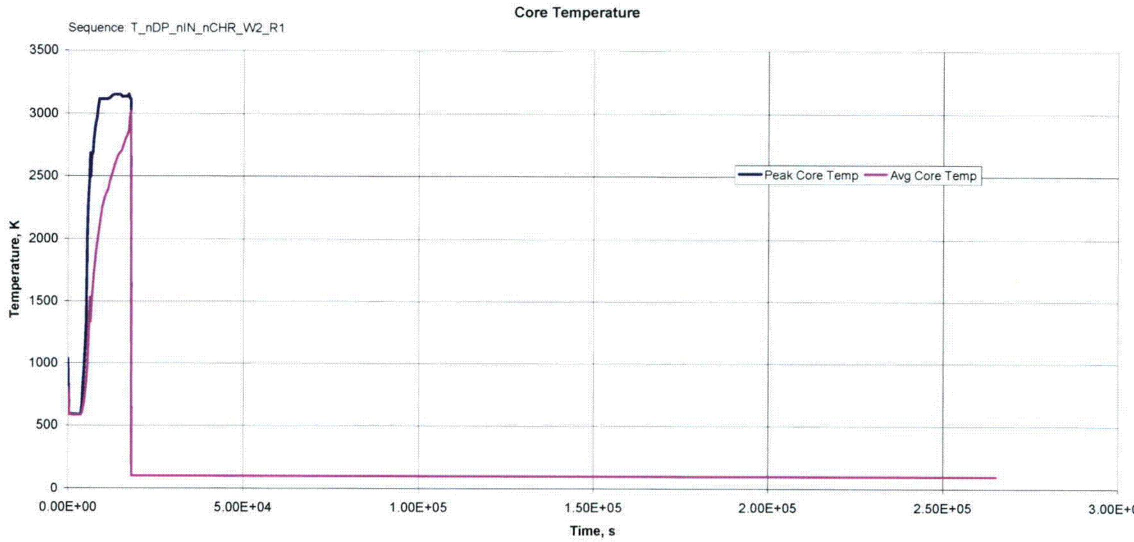


Figure 9A-16c – T_nDP_nIN_nCHR_W2_R1 Core Temperature

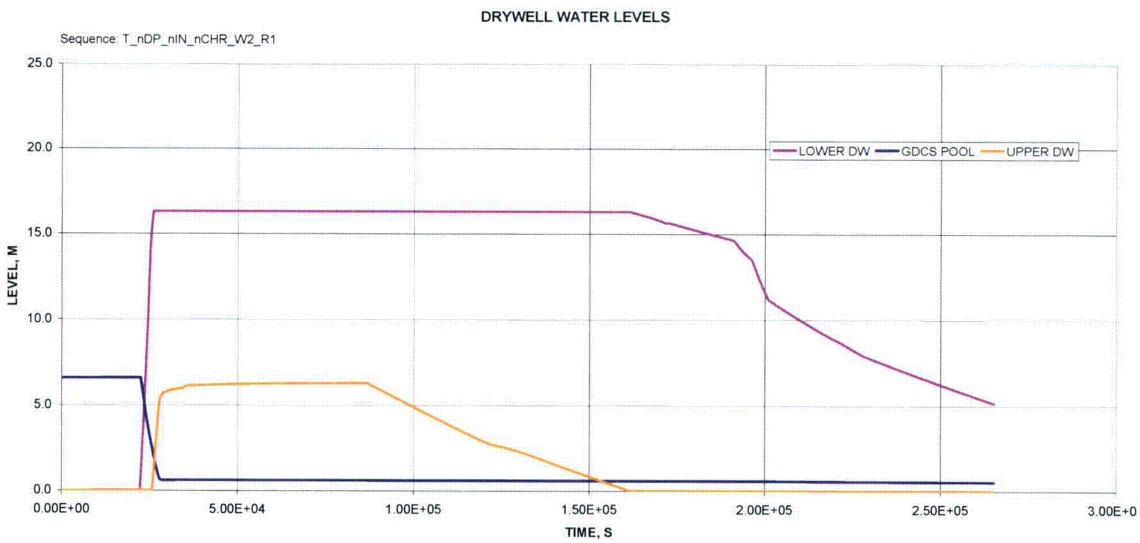


Figure 9A-16d – T_nDP_nIN_nCHR_W2_R1 Drywell Water Levels

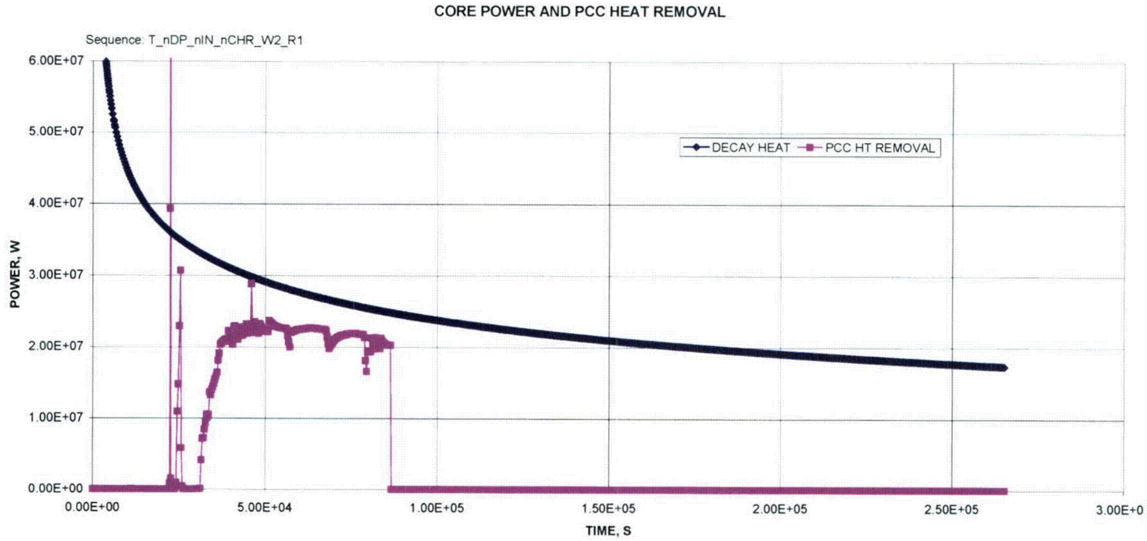


Figure 9A-16e – T_nDP_nIN_nCHR_W2_R1 Core Power and PCCS Heat Removal

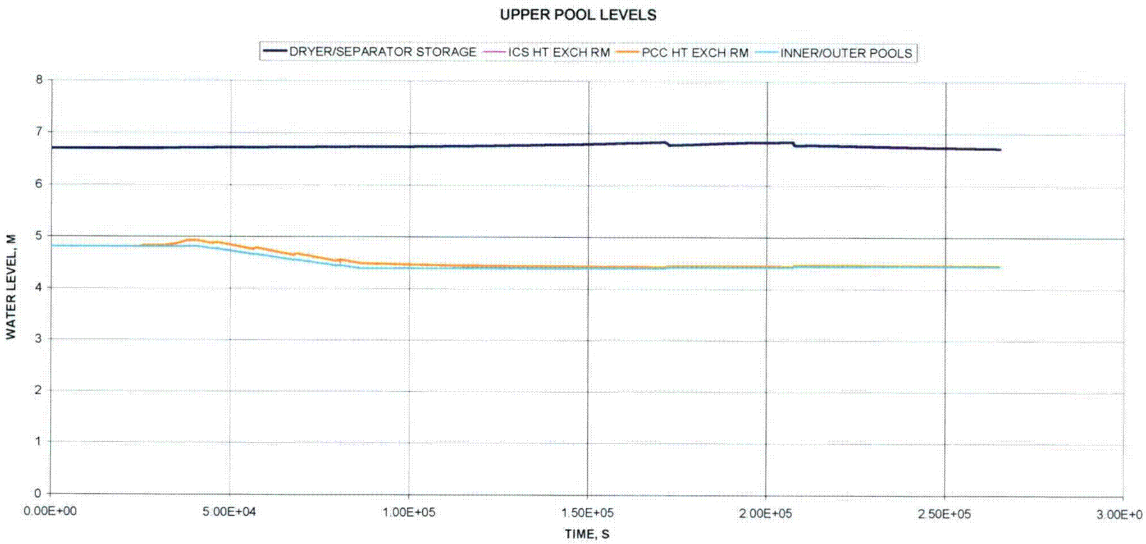


Figure 9A-16f – T_nDP_nIN_nCHR_W2_R1 Upper Pool Water Levels

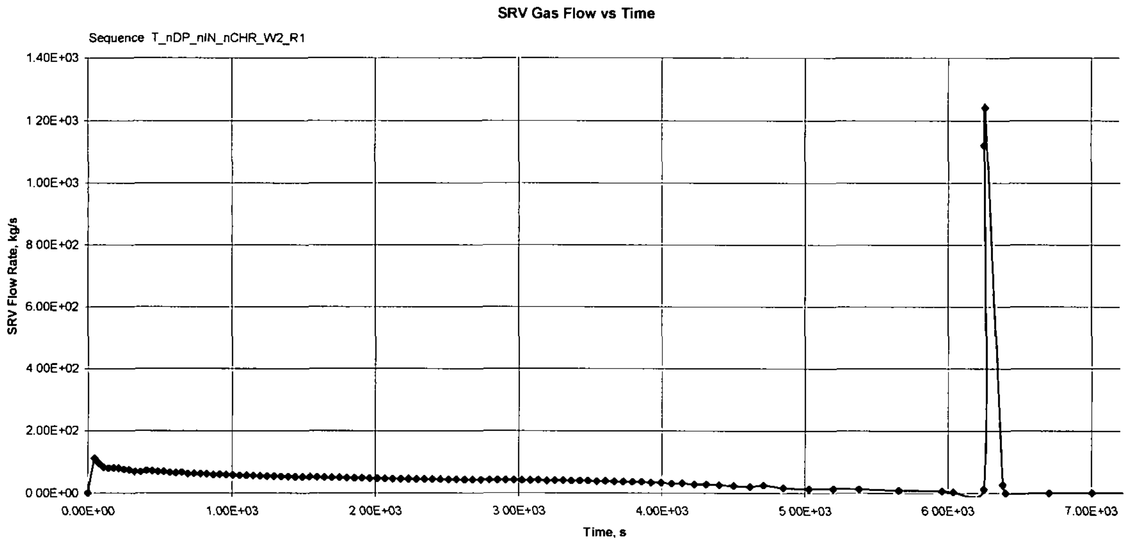


Figure 9A-16g – T_nDP_nIN_nCHR_W2_R1 SRV Gas Flow Rate

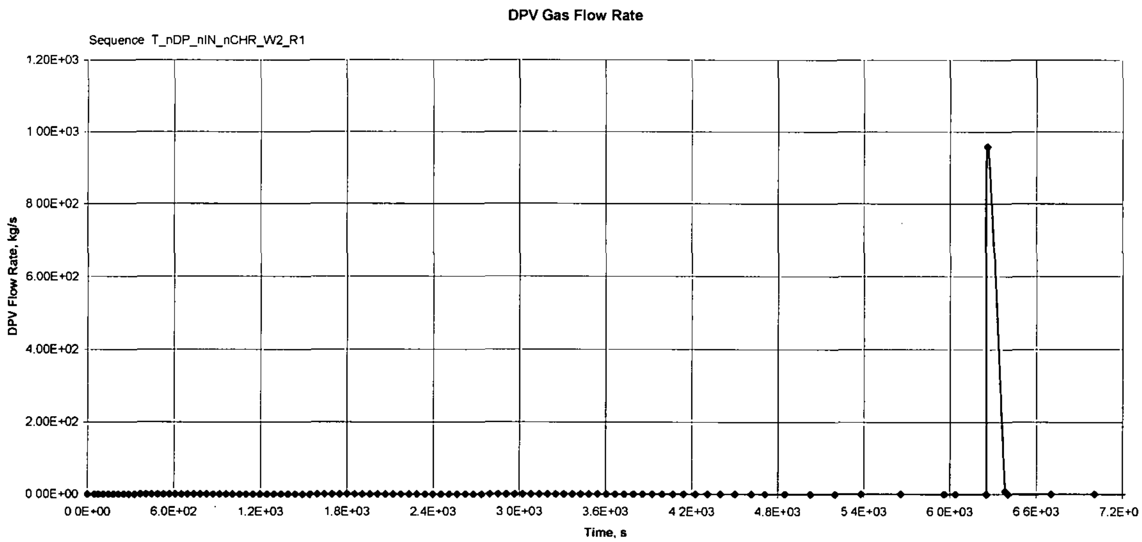


Figure 9A-16h – T_nDP_nIN_nCHR_W2_R1 DPV Gas Flow Rate

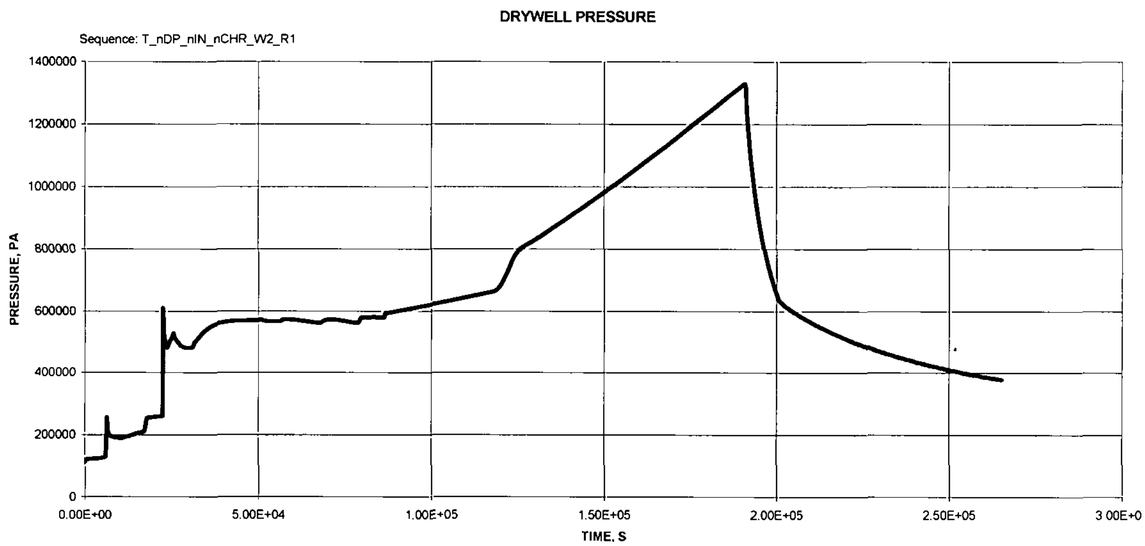


Figure 9A-16i – T_nDP_nIN_nCHR_W2_R1 Drywell Pressure

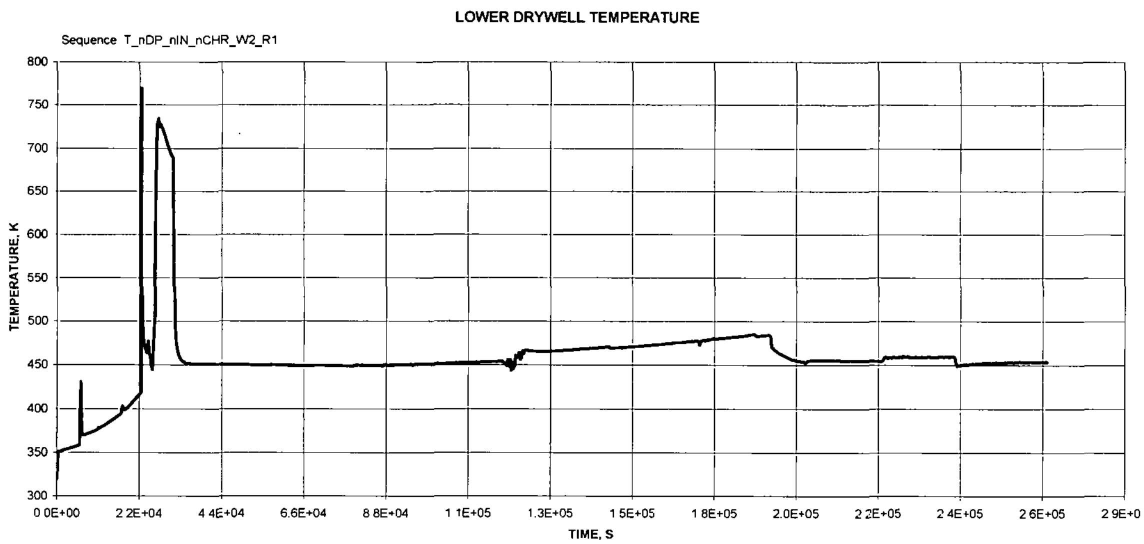


Figure 9A-16j – T_nDP_nIN_nCHR_W2_R1 Lower Drywell Temperature

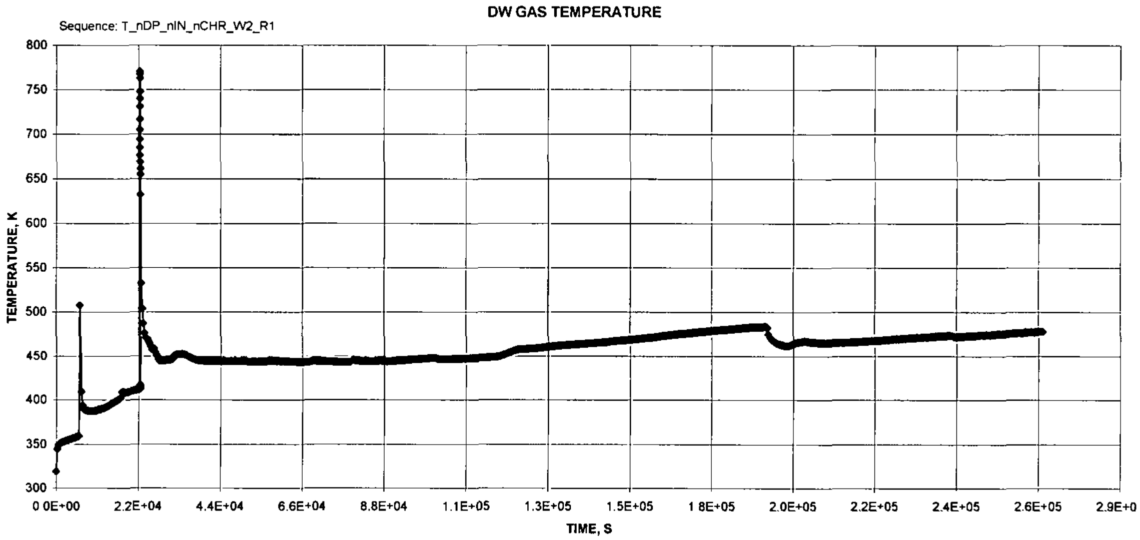


Figure 9A-16k – T_nDP_nIN_nCHR_W2_R1 Drywell Gas Temperature

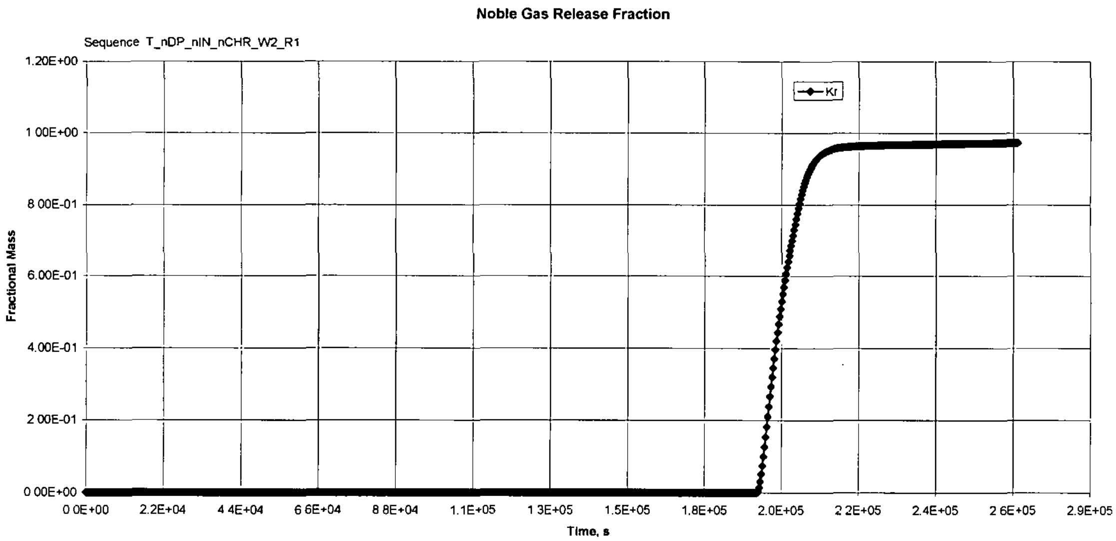


Figure 9A-16l – T_nDP_nIN_nCHR_W2_R1 Noble Gas Release Fraction

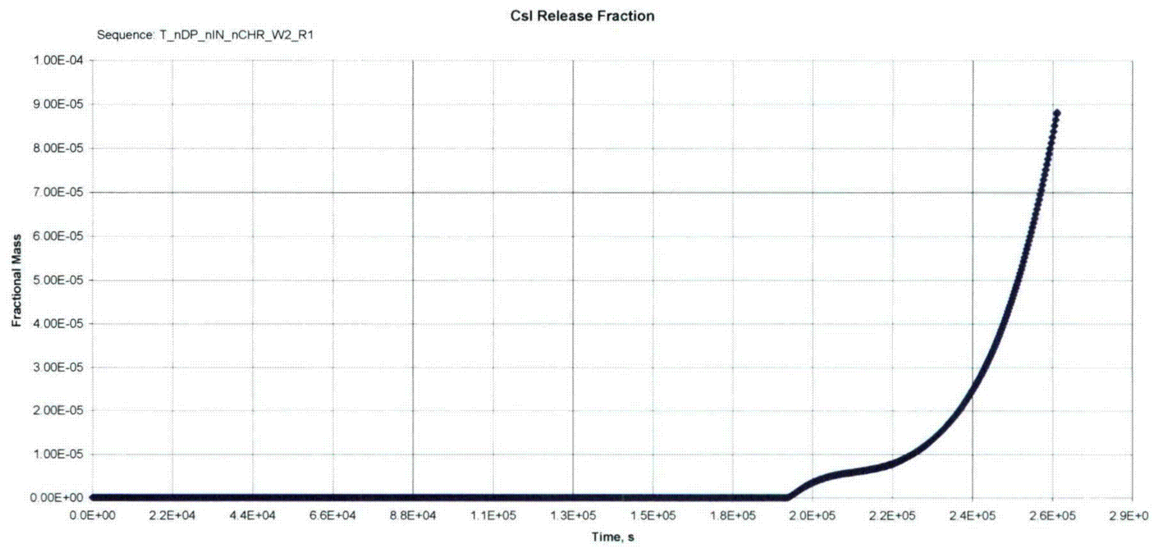


Figure 9A-16m – T_nDP_nIN_nCHR_W2_R1 CsI Release Fraction

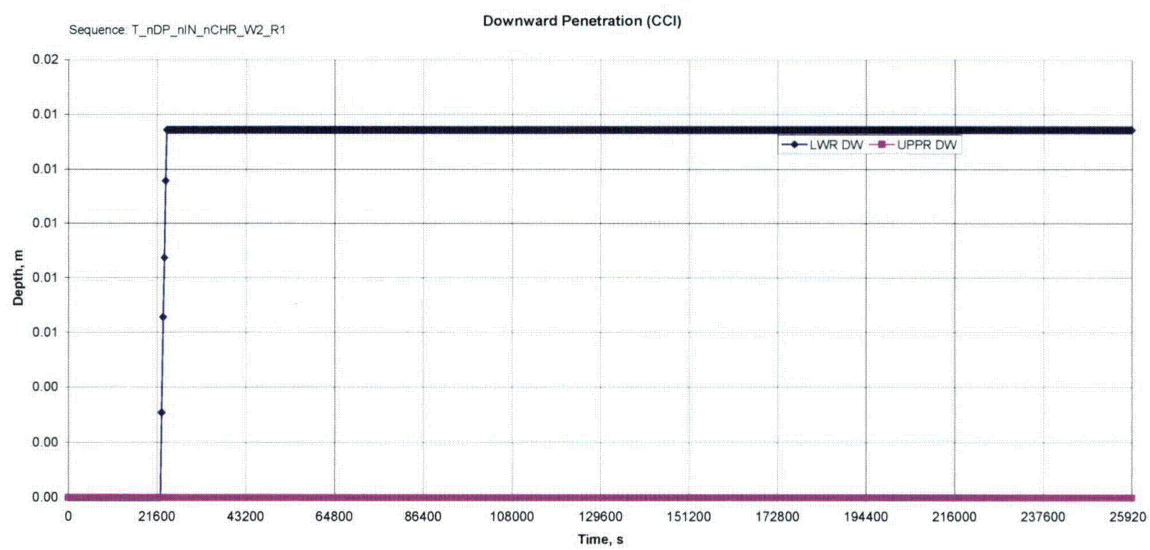


Figure 9A-16n – T_nDP_nIN_nCHR_W2_R1 Downward Penetration (CCI)

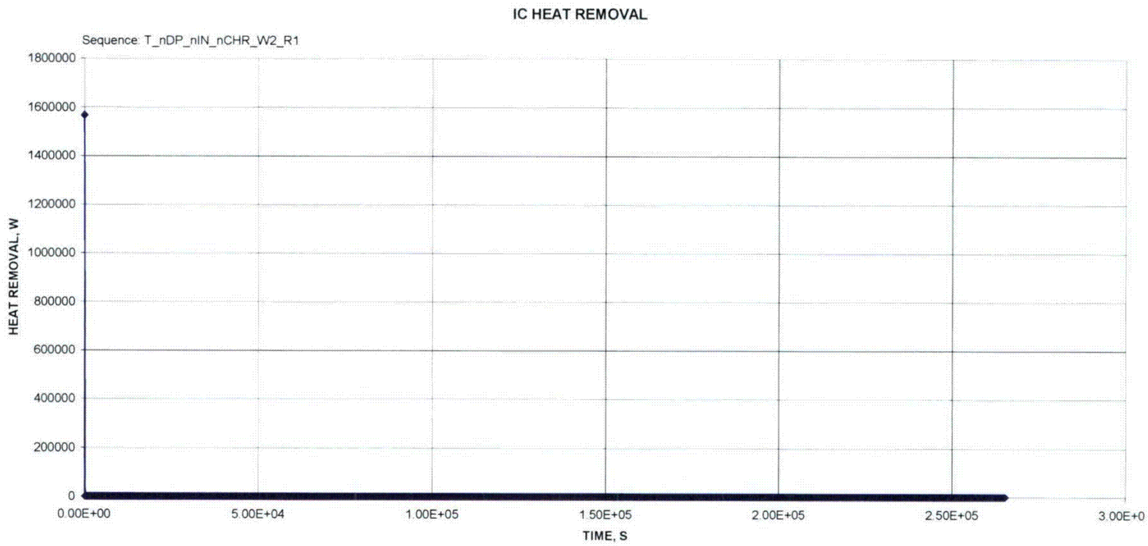


Figure 9A-16o – T_nDP_nIN_nCHR_W2_R1 ICS Heat Removal

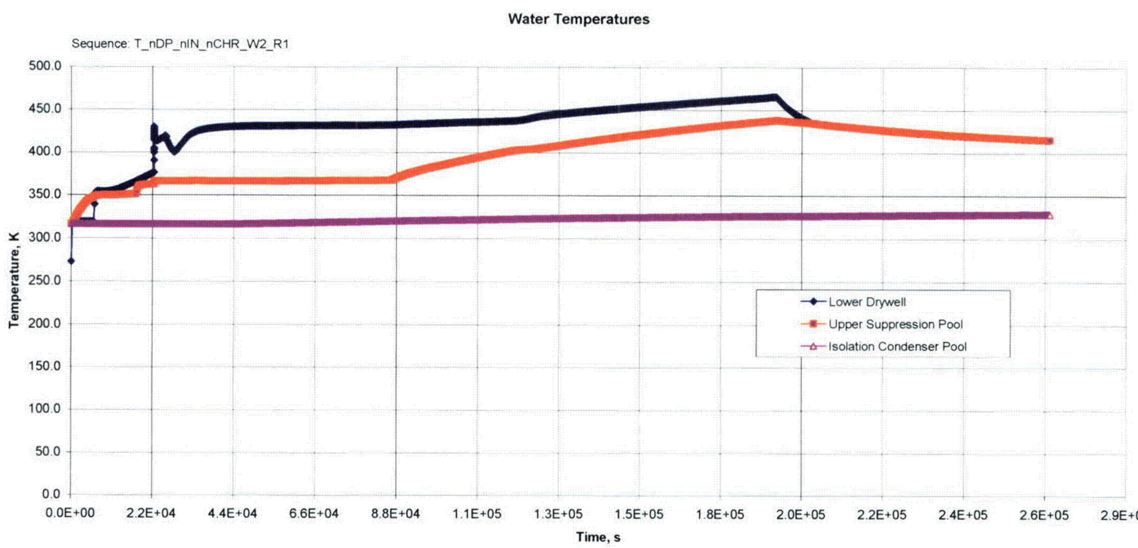


Figure 9A-16p – T_nDP_nIN_nCHR_W2_R1 Water Temperatures

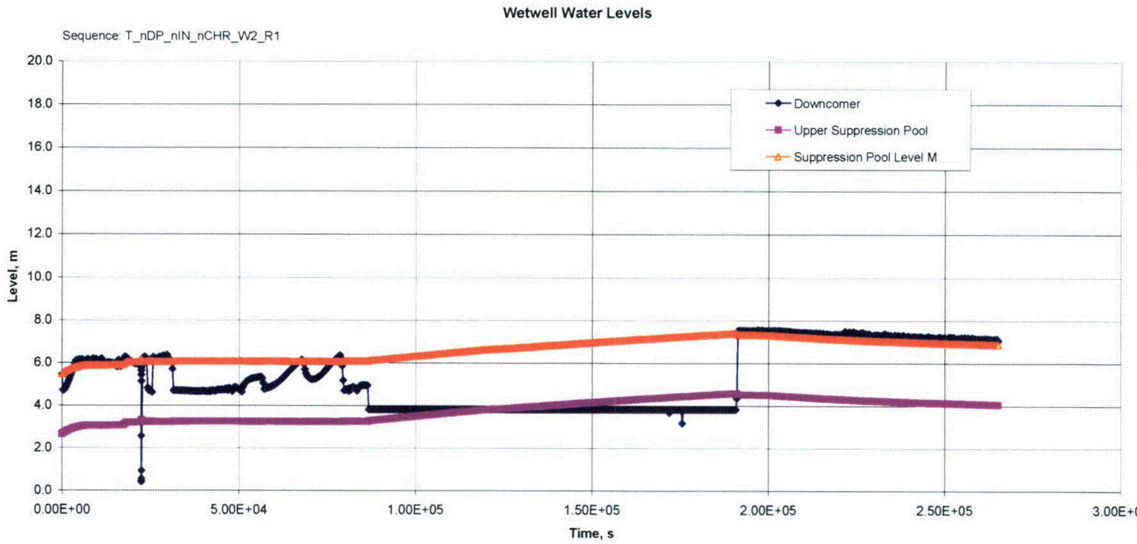


Figure 9A-16q – T_nDP_nIN_nCHR_W2_R1 Wetwell Water Levels

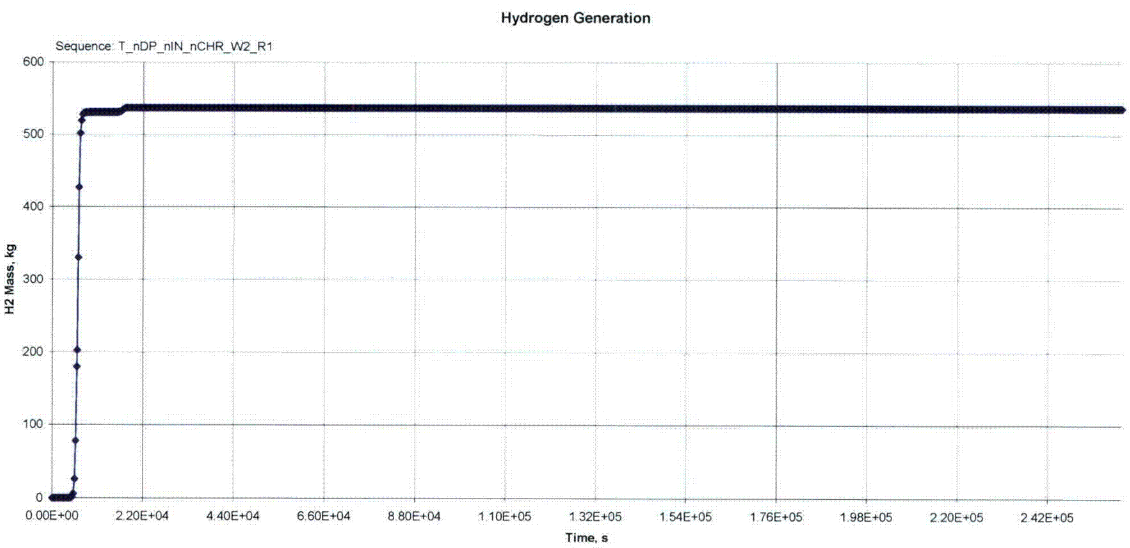


Figure 9A-16r – T_nDP_nIN_nCHR_W2_R1 Hydrogen Generation

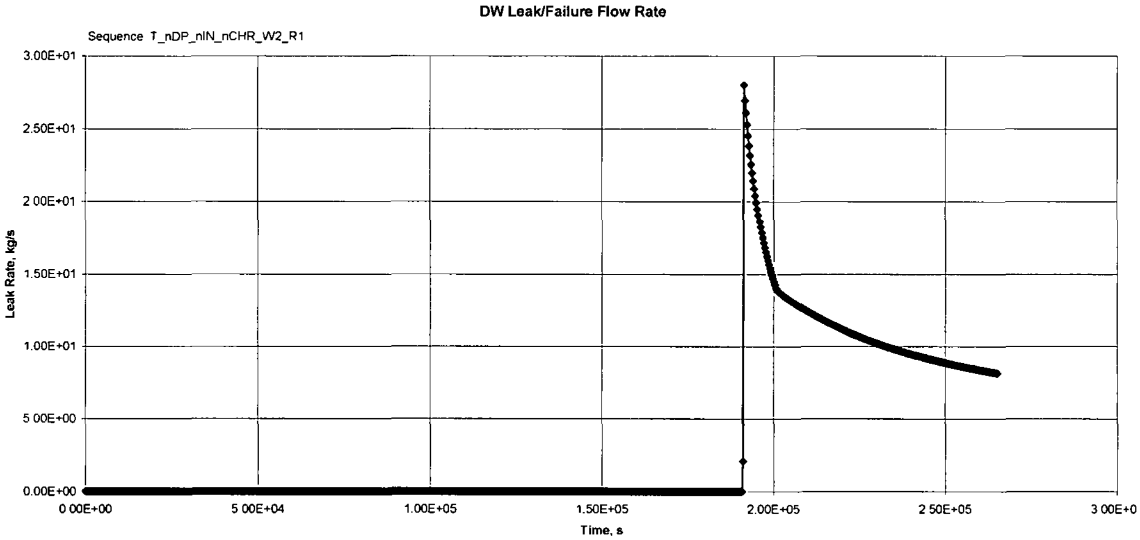


Figure 9A-16s – T_nDP_nIN_nCHR_W2_R1 Drywell Leak/Failure Flow Rate

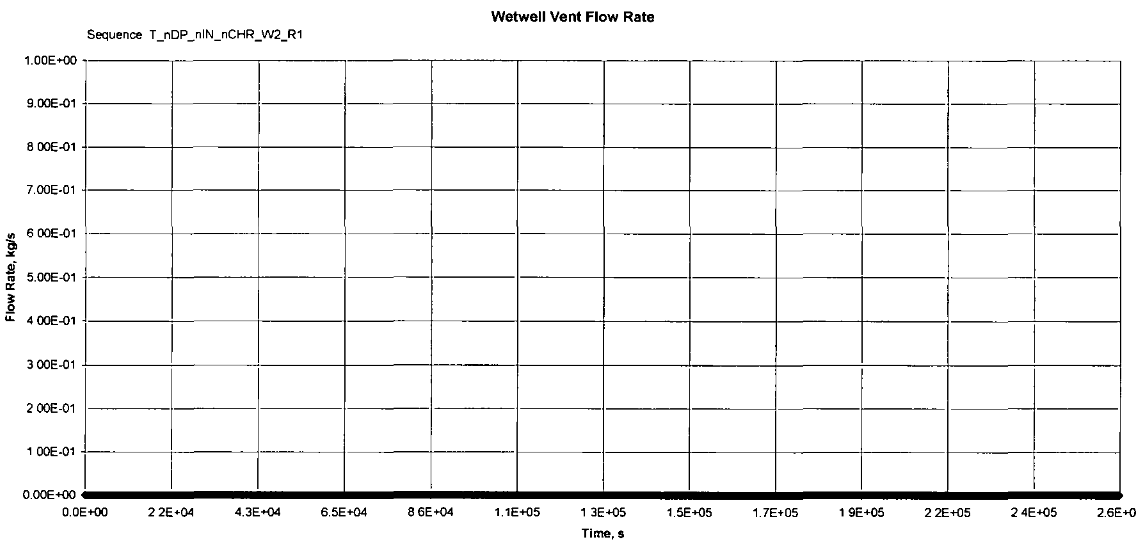


Figure 9A-16t – T_nDP_nIN_nCHR_W2_R1 Wetwell Vent Flow Rate

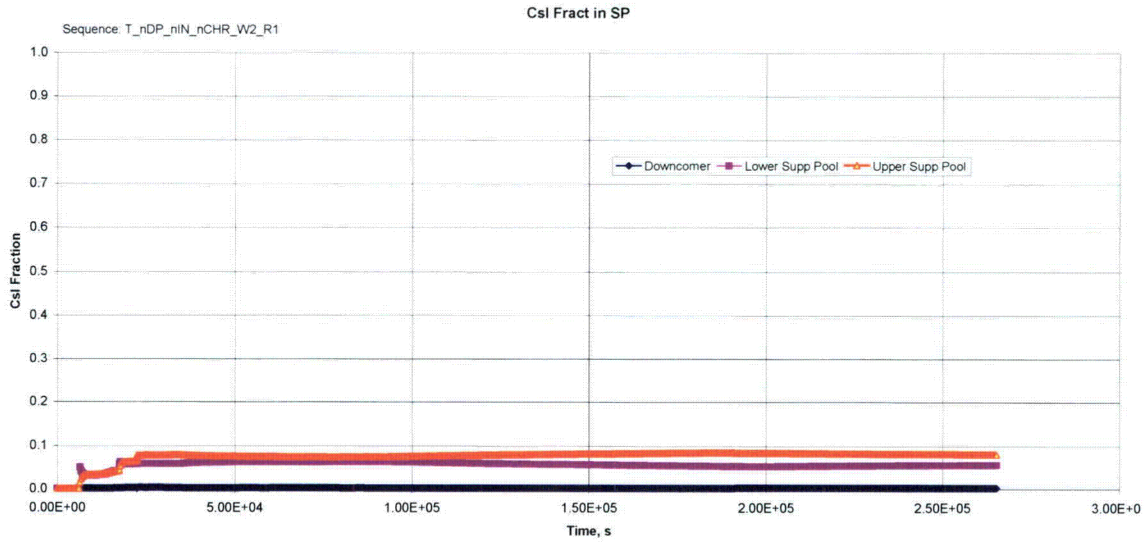


Figure 9A-16u – T_nDP_nIN_nCHR_W2_R1 CsI Fraction in Suppression Pool

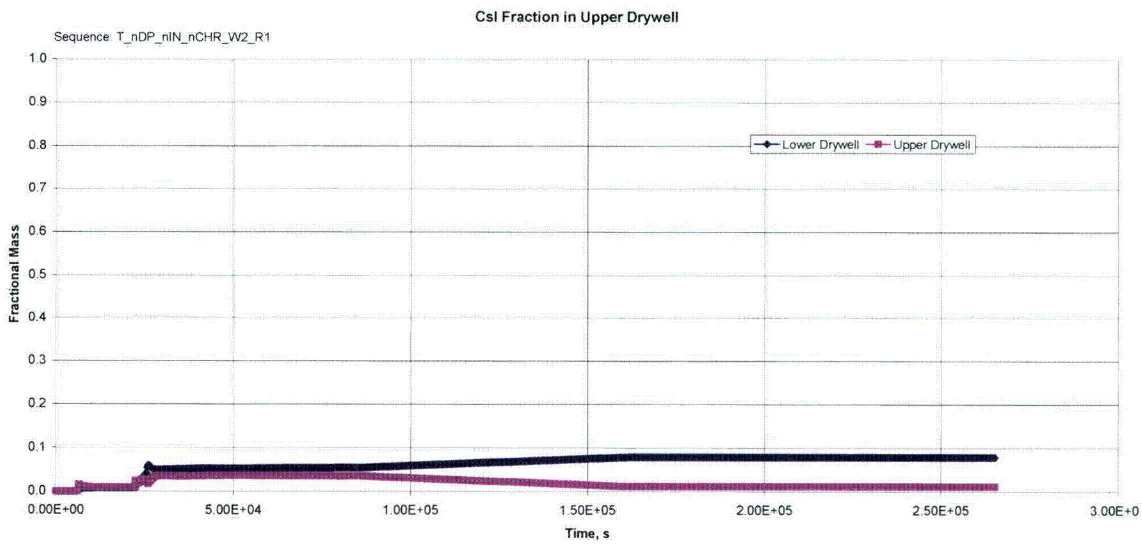


Figure 9A-16v – T_nDP_nIN_nCHR_W2_R1 CsI Fraction in Upper Drywell

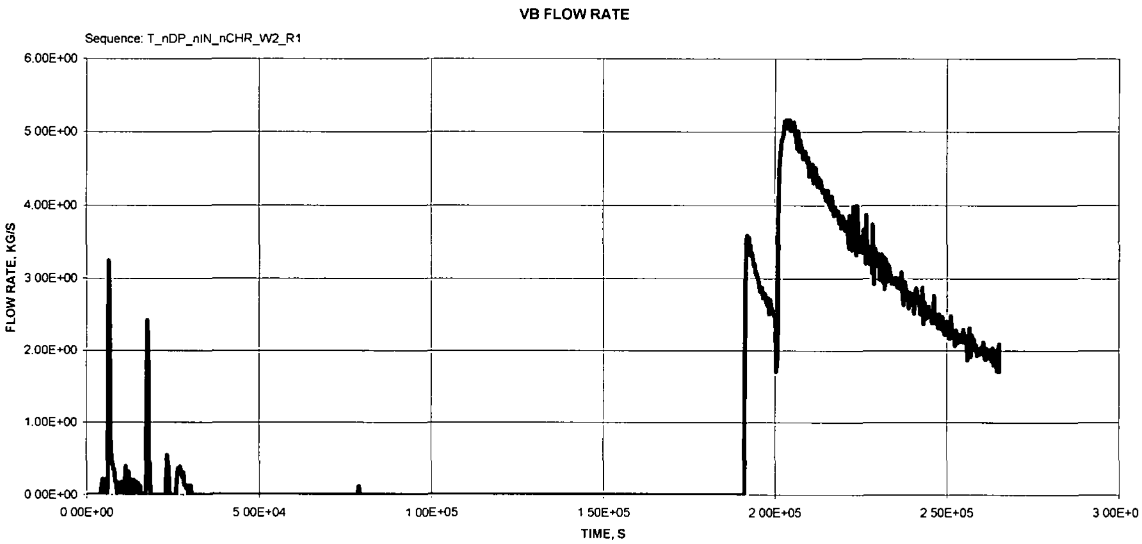


Figure 9A-16w – T_nDP_nIN_nCHR_W2_R1 Vacuum Breaker Flow Rate

9A.17 Technical Specifications Leakage T_AT_nIN_TSL2x_R1

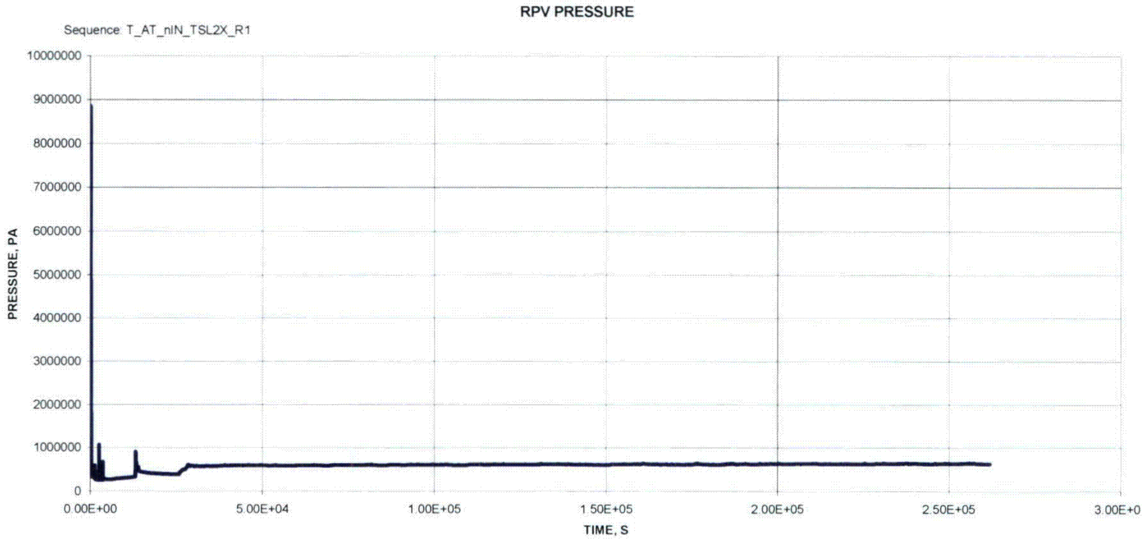


Figure 9A-17a – T_AT_nIN_TSL2x_R1 RPV Pressure

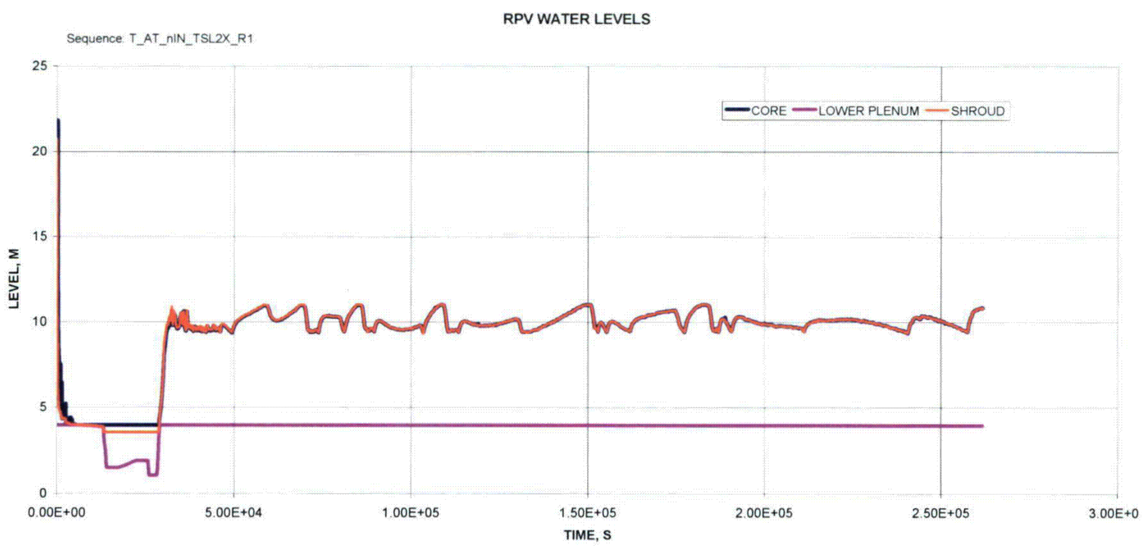


Figure 9A-17b – T_AT_nIN_TSL2x_R1 RPV Water Levels

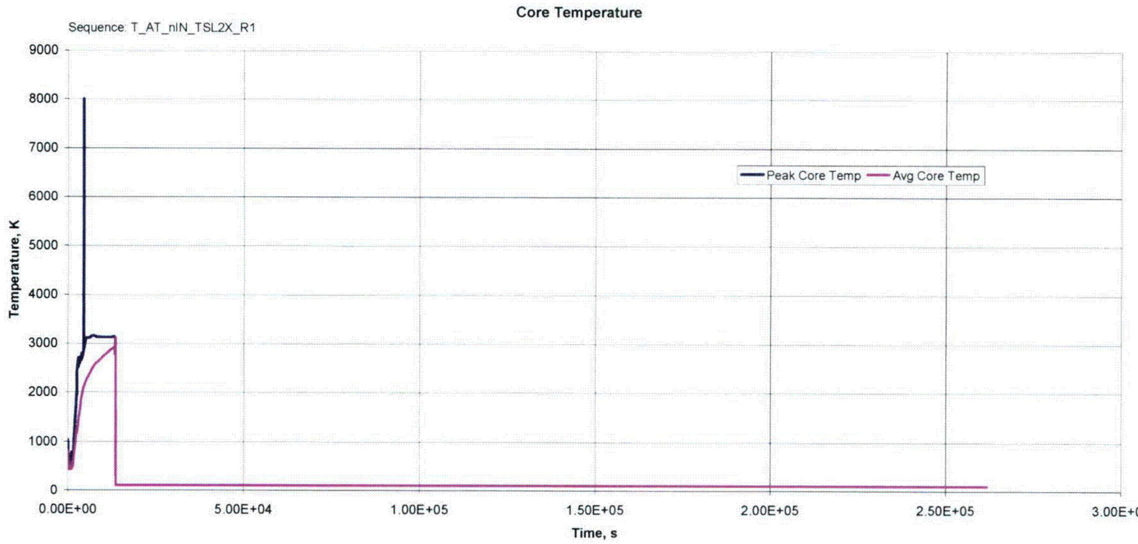


Figure 9A-17c – T_AT_nIN_TSL2x_R1 Core Temperature

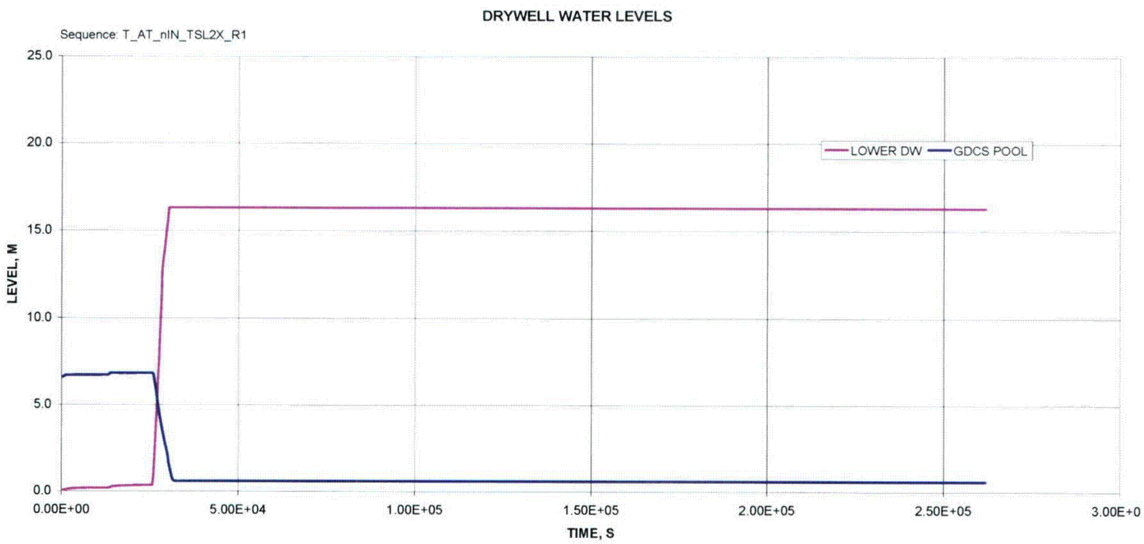


Figure 9A-17d – T_AT_nIN_TSL2x_R1 Drywell Water Levels

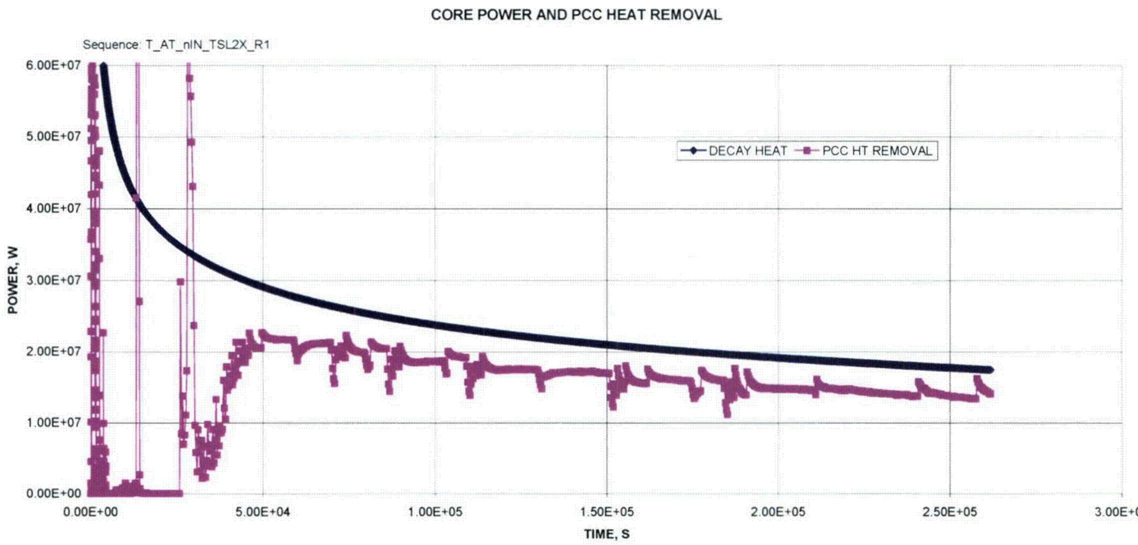


Figure 9A-17e – T_AT_nIN_TSL2x_R1 Core Power and PCCS Heat Removal

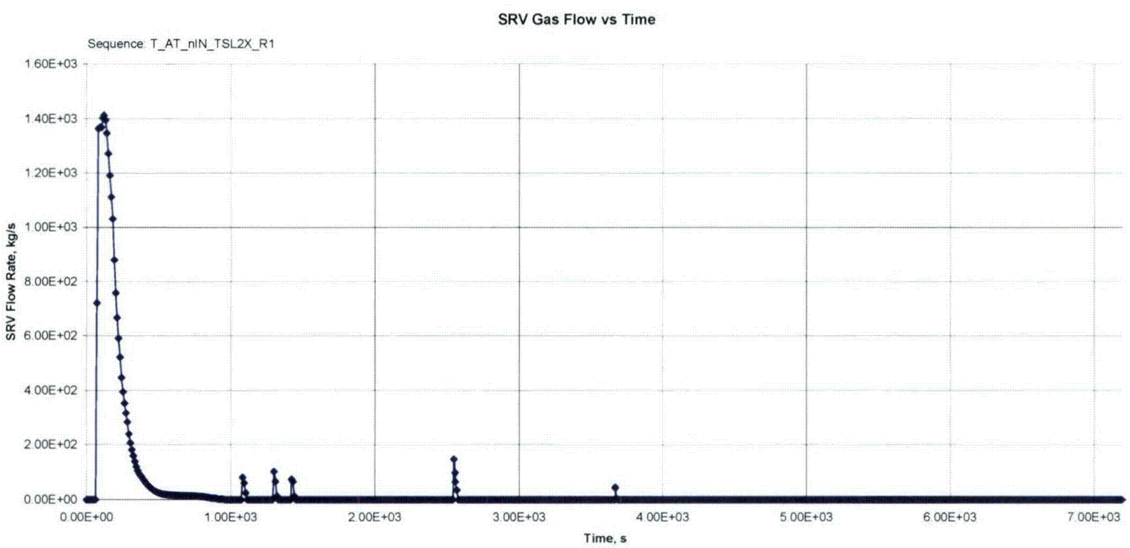


Figure 9A-17f – T_AT_nIN_TSL2x_R1 SRV Gas Flow Rate

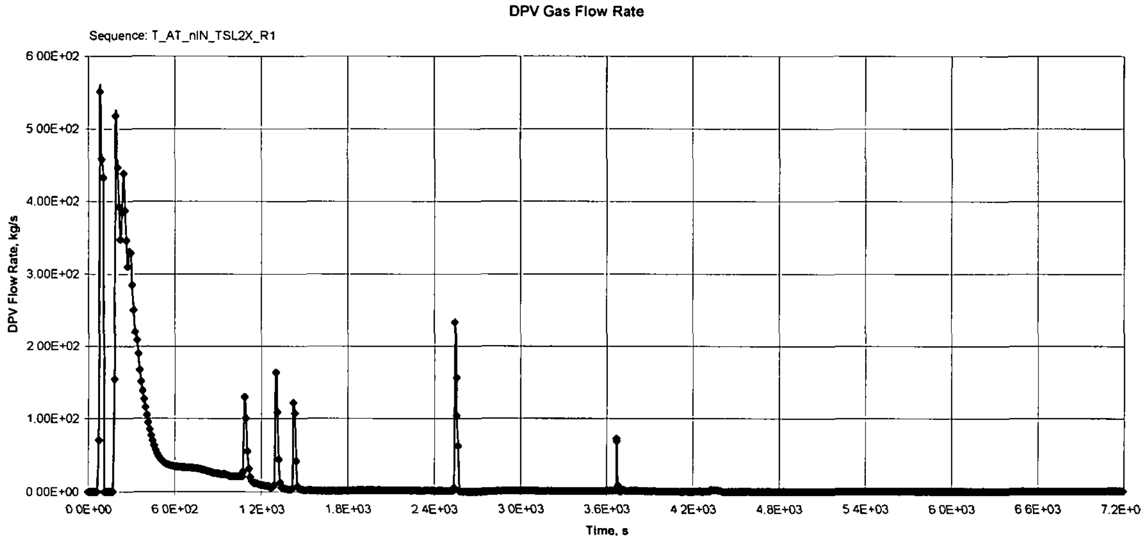


Figure 9A-17g – T_AT_nIN_TSL2x_R1 DPV Gas Flow Rate

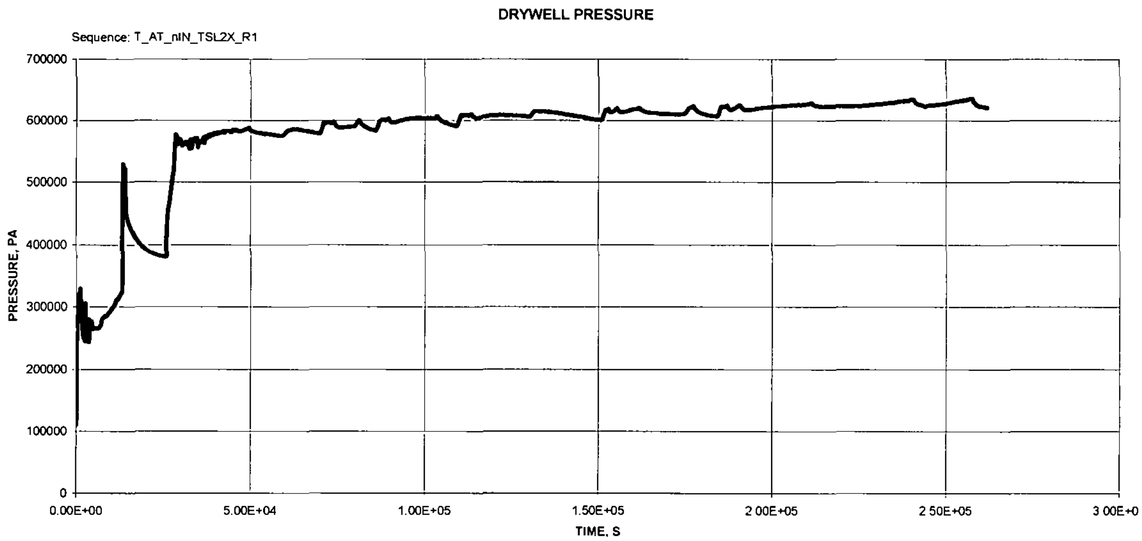


Figure 9A-17h – T_AT_nIN_TSL2x_R1 Drywell Pressure

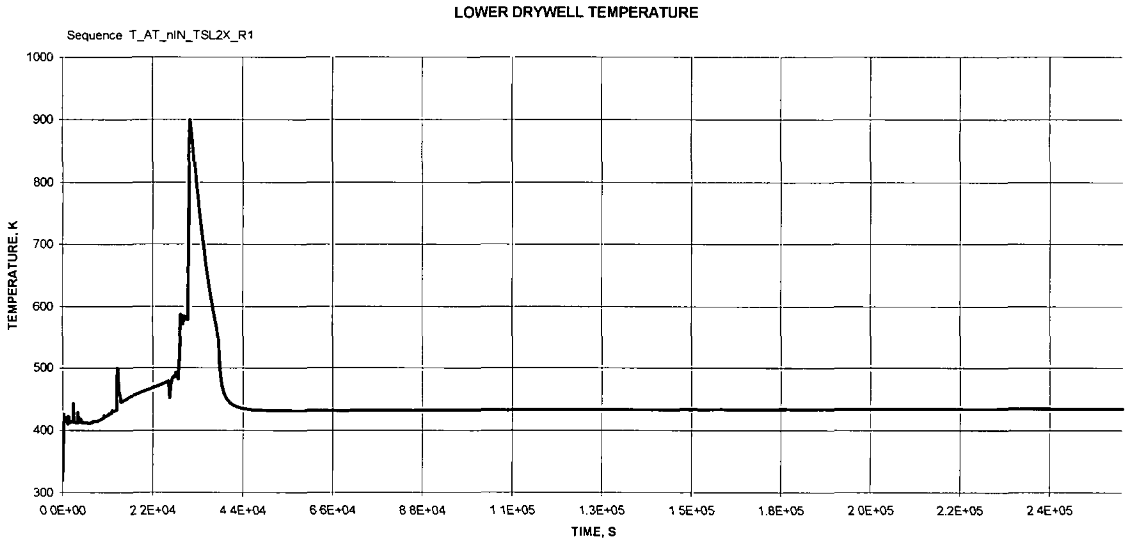


Figure 9A-17i – T_AT_nIN_TSL2x_R1 Lower Drywell Temperature

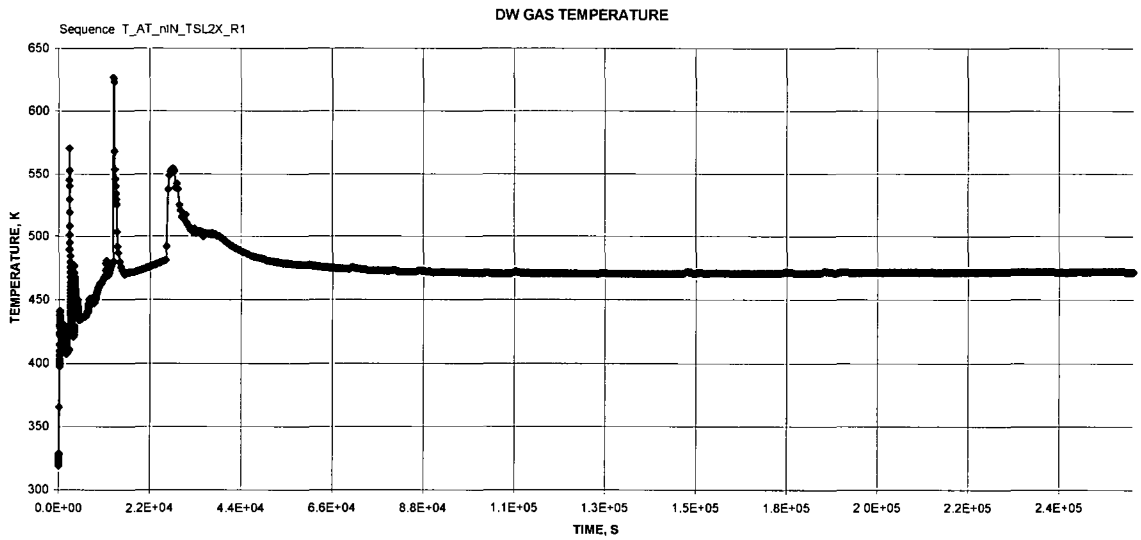


Figure 9A-17j – T_AT_nIN_TSL2x_R1 Drywell Gas Temperature

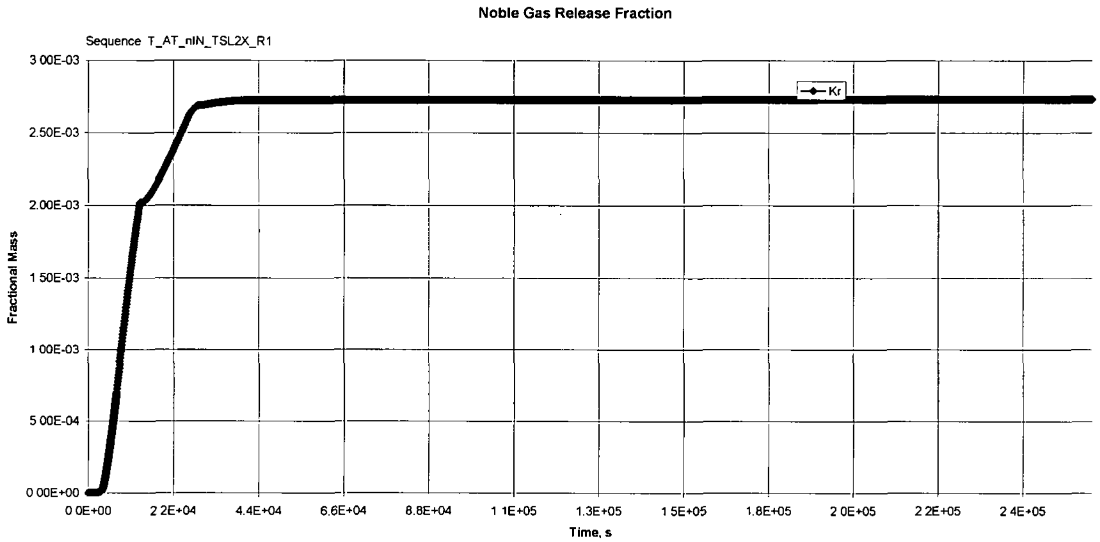


Figure 9A-17k – T_AT_nIN_TSL2x_R1 Noble Gas Release Fraction

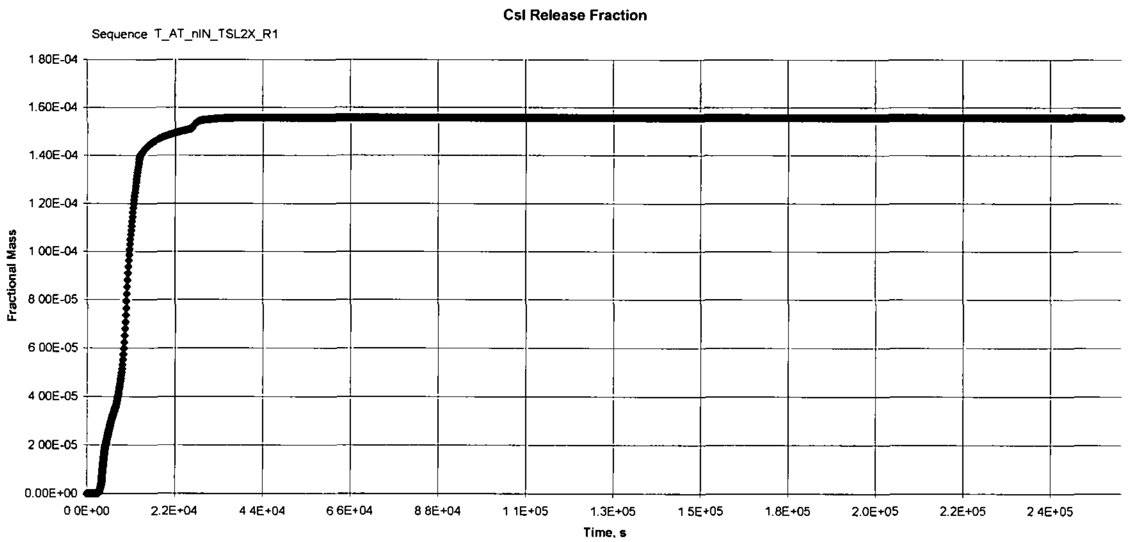


Figure 9A-17l – T_AT_nIN_TSL2x_R1 CsI Release Fraction

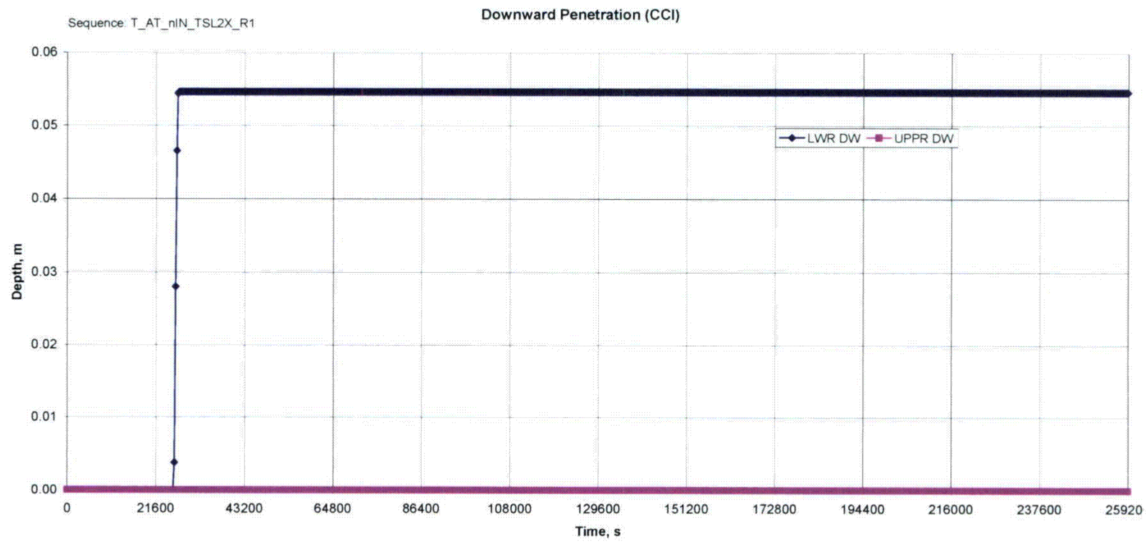


Figure 9A-17m – T_AT_nIN_TSL2x_R1 Downward Penetration (CCI)

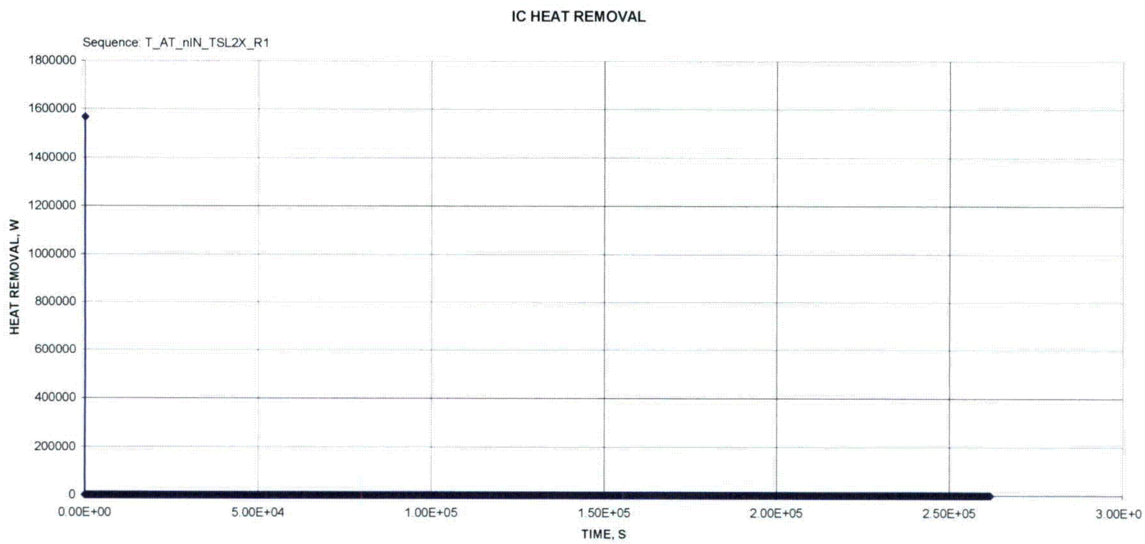


Figure 9A-17n – T_AT_nIN_TSL2x_R1 ICS Heat Removal

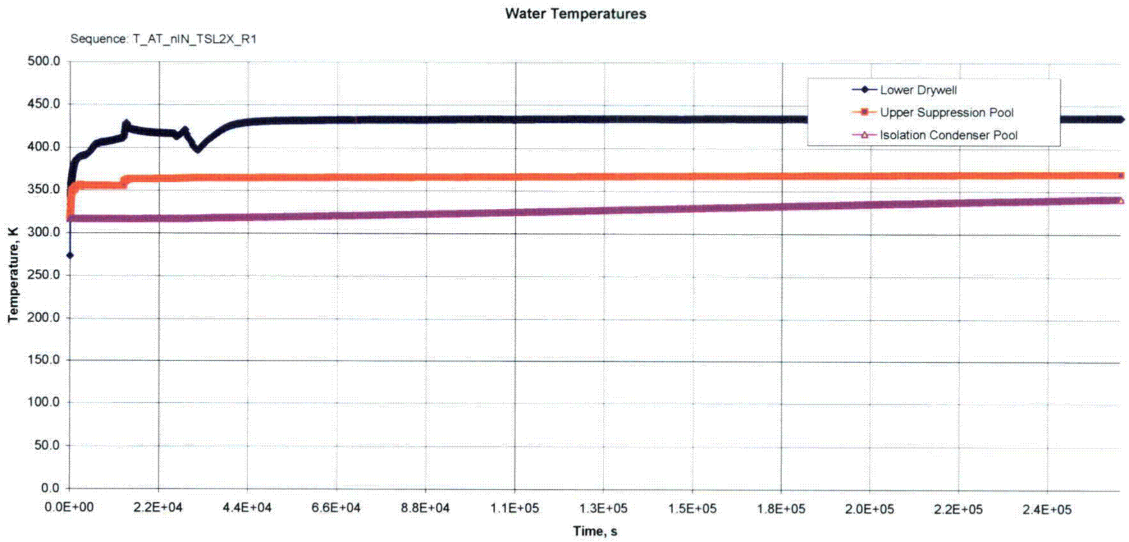


Figure 9A-17o – T_AT_nIN_TSL2x_R1 Water Temperatures

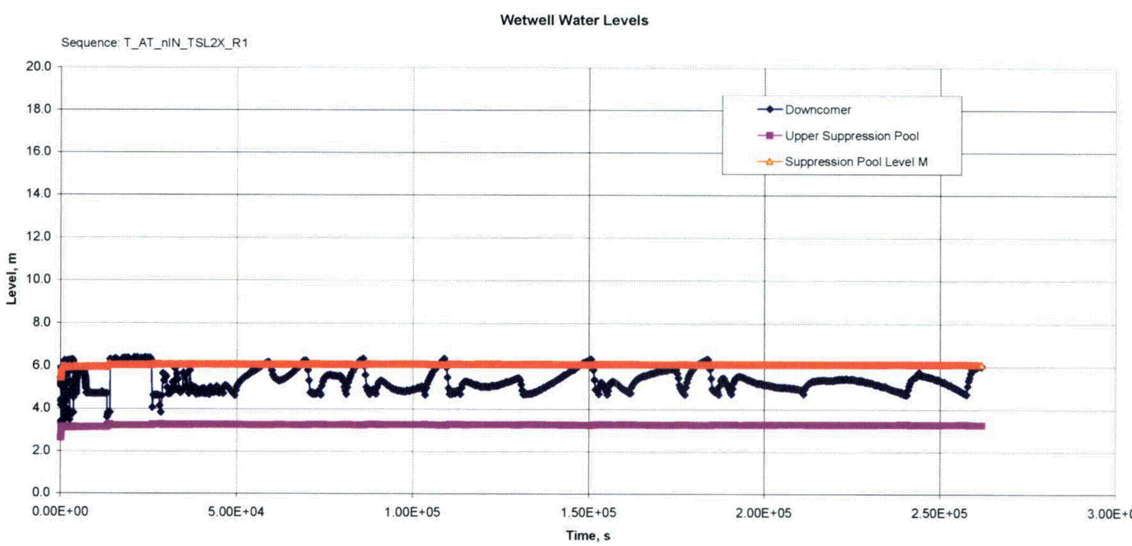


Figure 9A-17p – T_AT_nIN_TSL2x_R1 Wetwell Water Levels

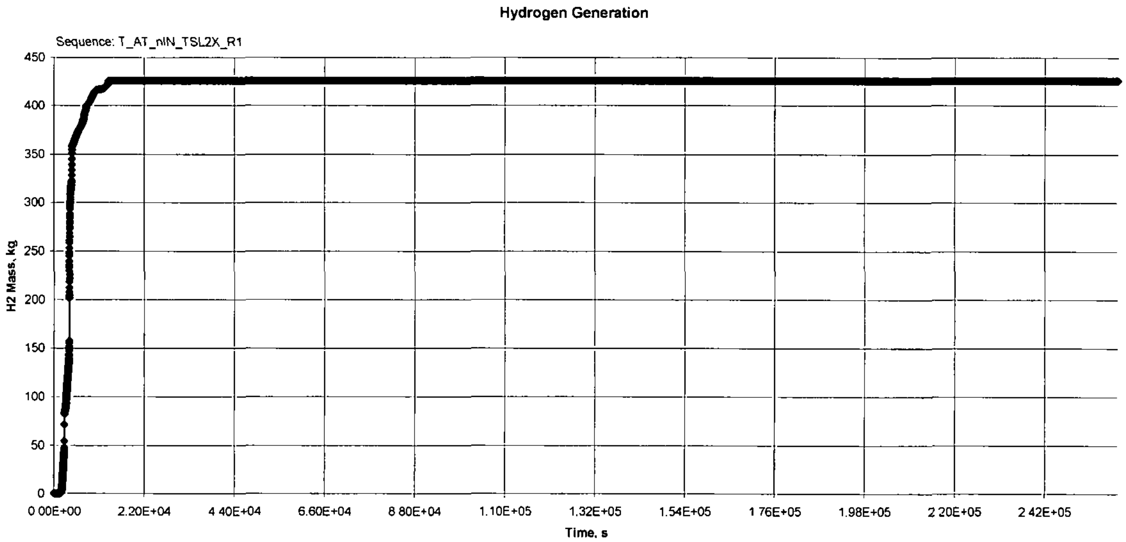


Figure 9A-17q – T_AT_nIN_TSL2x_R1 Hydrogen Generation

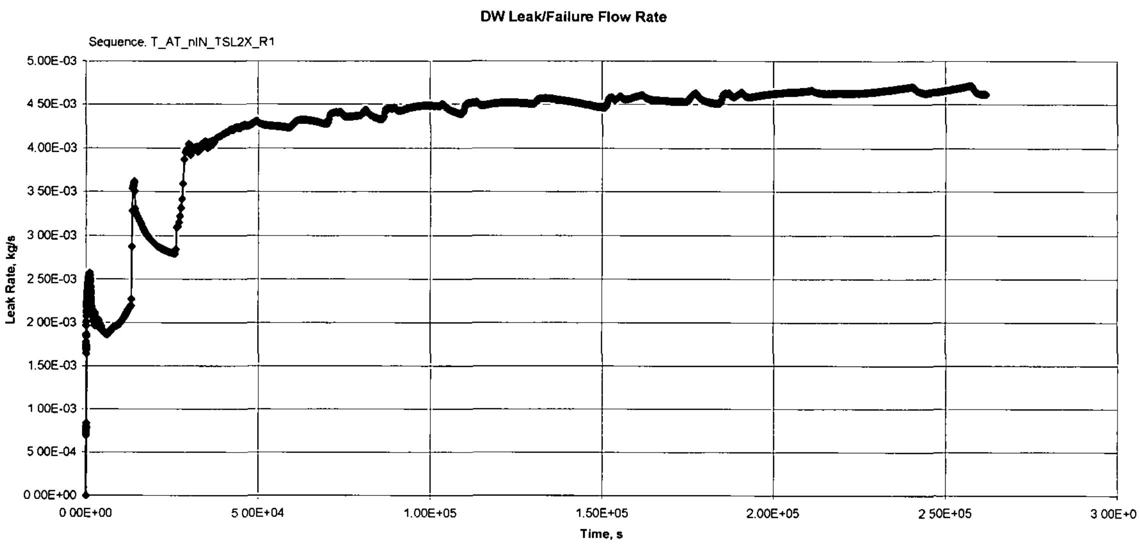


Figure 9A-17r – T_AT_nIN_TSL2x_R1 Drywell Leak/Failure Flow Rate

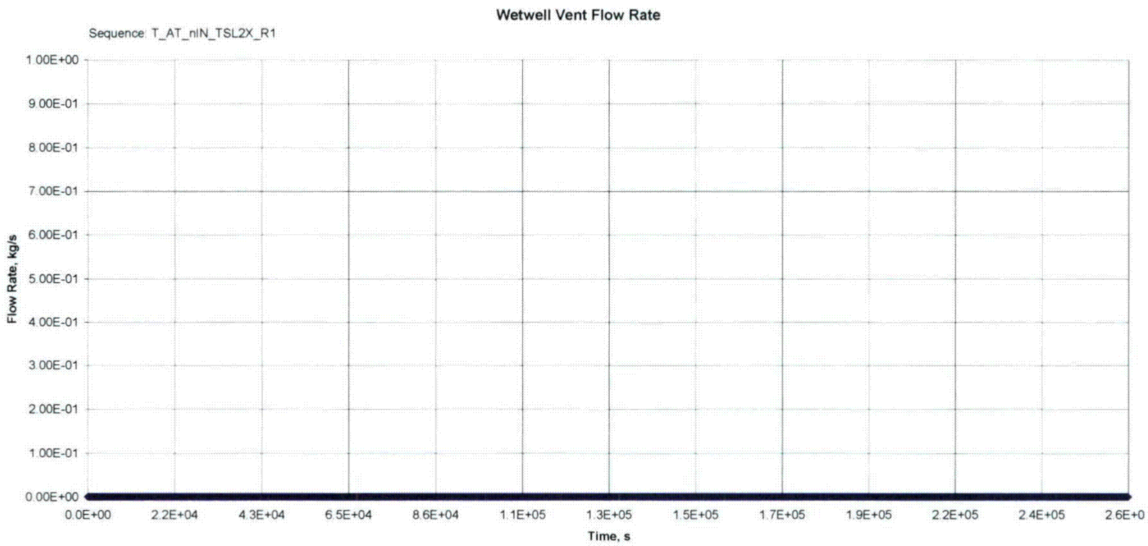


Figure 9A-17s – T_AT_nIN_TSL2x_R1 Wetwell Vent Flow Rate

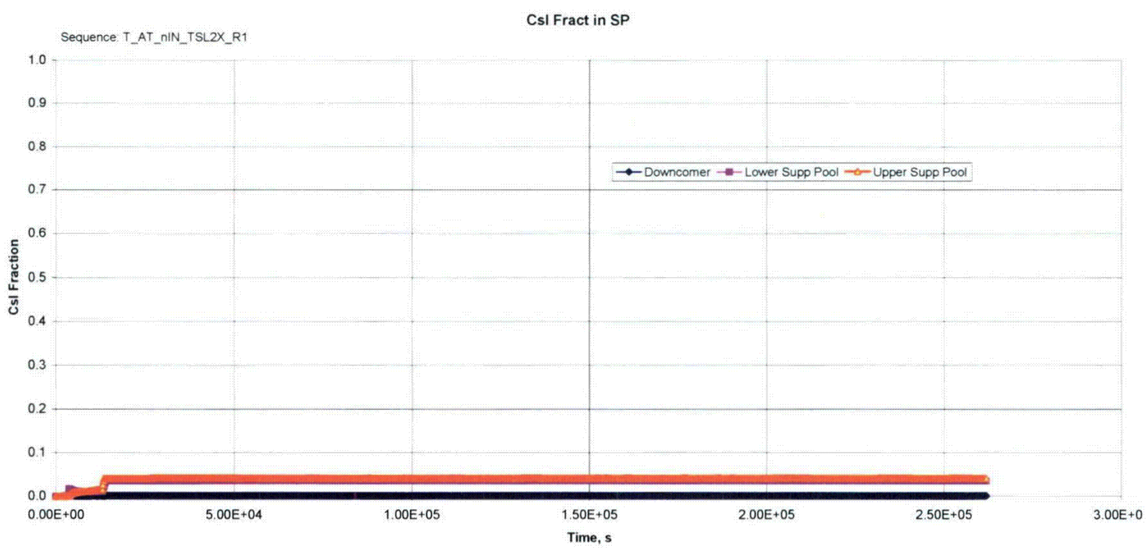


Figure 9A-17t – T_AT_nIN_TSL2x_R1 CsI Fraction in Suppression Pool

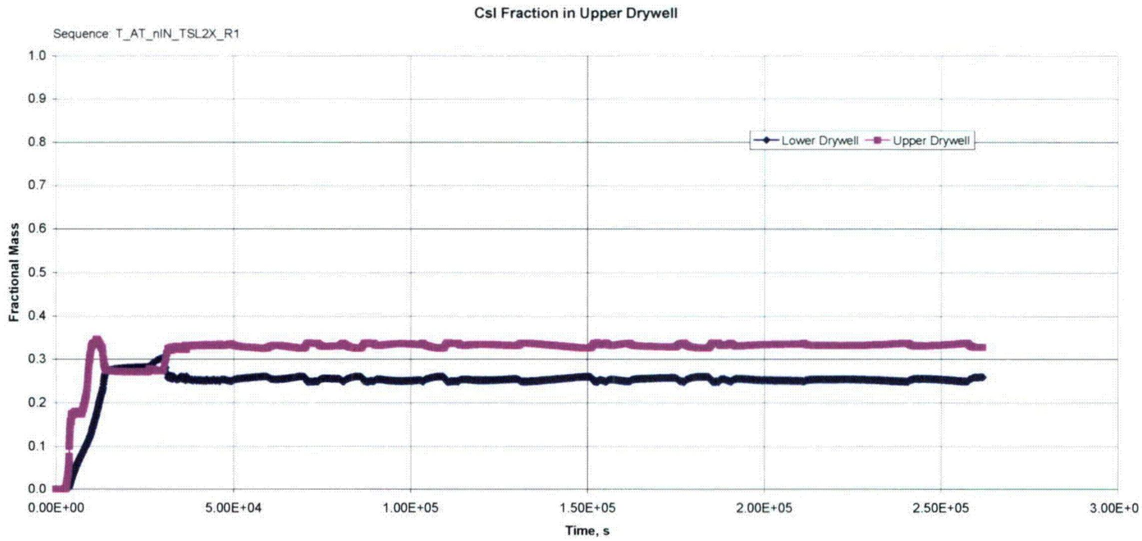


Figure 9A-17u – T_AT_nIN_TSL2x_R1 CsI Fraction in Upper Drywell

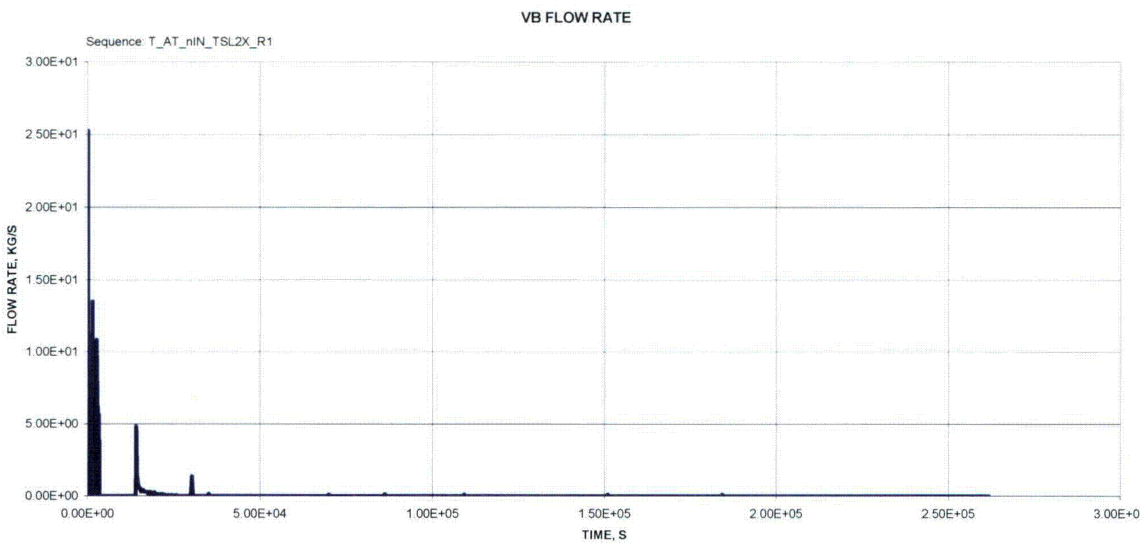


Figure 9A-17v – T_AT_nIN_TSL2x_R1 Vacuum Breaker Flow Rate

10 CONSEQUENCE ANALYSIS

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10 CONSEQUENCE ANALYSIS

10.1 INTRODUCTION

This section describes the offsite consequence evaluation (Level 3 analysis). Key inputs and assumptions are described. The calculated results are compared to consequence related goals to determine if the goals are satisfied.

The MACCS2 Version 1.13.1 computer code (Reference 10-1) is used to determine the consequences of potential reactor accidents. The MACCS2 code evaluates offsite dose and consequences such as early fatality risk and latent cancer fatality risk for each source term (i.e., radionuclide release category) over a range of possible weather conditions and evacuation assumptions. The MACCS2 code model is described in Reference 10-1. The rationale for site related input selection is presented in Section 10.2. Other more generic input parameters for the MACCS2 analysis are based on "Sample Problem A" of Reference 10-1. ESBWR specific reference data from the plant performance analysis in Section 8 and Section 9 are used as MACCS2 inputs as presented in Subsection 10.3. The calculated consequence results are compared to the goals in Subsection 10.5-7b. In Section 10.5, a sensitivity study is summarized. Table 10.2-1 through 10.5-7b present the MACCS2 base case and sensitivity study results. Figure 10.4-1 presents the exceedance probability as a function of population dose.

10.2 SITE ASSUMPTIONS

The evaluation of the offsite consequences of a reactor accident uses generic site parameters (e.g., weather, population, land use).

The subsections below describe the rationale for the selection of site meteorology, population, and evacuation parameters. The following tables present these inputs:

Table	Inputs
10.2-1	Population Density
10.2-2	Shielding and Exposure Parameters

10.2.1 Meteorology

For this study, a meteorological condition comparable with ALWR URD (Reference 10-6) meteorological reference data set is used. For base case study, weather category bin sampling approach of the meteorological data is used as was done in the URD. An hourly sampling approach of the meteorological data is presented in the sensitivity study.

10.2.2 Population

For the ESBWR consequence evaluation, the SANDIA Siting Study population density data (Table 3-2 of Reference 10-4) is used to develop a uniform population density corresponding to each spatial interval. The population distribution is developed for distances to 0.5, 1, 2, 3, 4, 5, 10, 20, 30, 40 and 50 miles from the site.

The three offsite consequence goals defined for the ESBWR are concerned with consequences within 10 miles of the site; therefore, a bounding 0-10 mile population density is used. The maximum 0-10 mile population distribution value from the "all" sites column of Table 3-2 of Reference 10-4 is used for the ESBWR consequence evaluation and is provided in Table 10.2-1. As can be seen from Table 10-1, the 0-5 mile population density is larger than the 5-10 mile population density and is used in this bounding analysis as a constant uniform density. This approach provides a more bounding 0-10 mile population density than that provided in the ALWR URD (Reference 10-6).

10.2.3 Evacuation

Many evacuation related characteristics (local roads, population demographics, emergency services) are site specific. The evacuation parameters used in this study are conservative assumptions in that no evacuation or relocation assumed and no sheltering is assumed. The public is assumed to continue normal activity during the reactor accident in this bounding analysis. Shielding and exposure values used for normal activity are the standard MACCS2 assumptions and are provided in Table 10.2-2.

Table 10.2-2 provides the following information for people engaged in normal activity:

- Cloudshine Shielding Factor – Fraction of cloudshine dose received from direct external exposure to the plume
- Inhalation Protection Factor – Fraction of inhalation dose received from cloud inhalation
- Breathing Rate – Breathing rate for people in normal activity
- Skin Protection Factor – Fraction of skin dose received from material deposited on skin
- Groundshine Shielding Factor – Fraction of groundshine dose received from material deposited on the ground

Table 10.2-1

Population Distribution

Radial Interval	Maximum Population
	All Sites People per sq. km.(per sq. mi.)
0-8.1 km (0-5 mi)	305 (790)
8.1-16.1 km (5-10 mi)	270 (700)
16.1-32.2 km (10-20 mi)	282 (730)
32.2-48.3 km (20-30 mi)	772 (2000)
48.3-80.5 km (30-50 mi)	965 (2500)

Data taken from Reference 10-4, Table 3-2.

The 0-5 mile population density (790 people per square mile) is used in the ESBWR bounding analysis as a uniform density for all radial intervals in the 0-50 mile region.

Table 10.2-2

Shielding and Exposure Data

MACCS2 Parameter	Normal Activity Value
Cloudshine Shielding Factor	7.50E-01
Inhalation Protection Factor	4.10E-01
Breathing Rate (m^3/sec)	2.66E-04
Skin Protection Factor	4.10E-01
Groundshine Shielding Factor	3.30E-01

See Subsection 10.2.3 for additional description of parameters in this table.

All values are based on Reference 10-1

10.3 MACCS2 RADIONUCLIDE RELEASE INPUT DATA

10.3.1 MACCS2 Radionuclide Release Input Data

ESBWR specific radionuclide release data is used in this analysis to model the dispersion of a plume of material released to the environment during a reactor accident.

The following tables present these inputs:

Table	Inputs
10.3-1	Building Data for Meteorological Modeling of Wake Effects
10.3-2	Core Inventory Parameters
10.3-3a	Reactor Accident Release Parameters 24 Hours After the Onset of Core Damage
10.3-3b	Reactor Accident Release Parameters 72 Hours After the Onset of Core Damage
10.3-4	Nuclide Release Categories

10.3.2 ESBWR Release Parameters

ESBWR specific parameters are used for wake effect data, core inventory, and reactor thermal power. The width and height of the building wake are used by MACCS2 to model the initial plume dimensions. These parameters for the ESBWR are provided in Table 10.3-1.

The equilibrium core inventory and reactor thermal power used in this analysis are ESBWR specific and are provided in Table 10.3-2. These parameters are used to determine the inventory of each nuclide in the core at accident initiation.

10.3.3 Input to MACCS2 from MAAP

The severe accident sequence analysis results provide input parameters to the MACCS2 code and are described here and are shown in Table 10.3-3a and Table 10.3-3b. Table 10.3-3a provides the release parameters 24 hours after the onset of core damage, and Table 10.3-3b provides the release parameters 72 hours after the onset of core damage. The severe accident sequence analysis performed using the MAAP code is further described in Section 8. The representative MAAP cases used as MACCS2 inputs are summarized in Section 9. Important input release characteristics include the nuclide release time, duration, and release fraction. The MAAP cases are used to develop source terms for each release category for the consequence analysis. Tables 10.3-3a and 10.3-3b describe the source terms and corresponding radionuclide release categories used for the MACCS2 analysis.

For each source term, which represents a release category from Section 8, the following data are used (Table 10.3-3a and Table 10.3-3b):

- Source Term – Source term developed from the severe accident analysis that characterizes the release category. The source terms are summarized in Section 9.
- Release Category – Release category represented by the source term
- MAAP Case – Severe accident sequence analysis results which are used to develop each source term. Section 8 and Section 9 provide a summary of the MAAP cases.
- Release Frequency – The frequency per year associated with the radionuclide release category. The release frequencies are calculated in Section 8.
- Time of Plume Release – Time from reactor trip (time of accident initiation) until the time of the modeled plume release to the atmosphere. This parameter is based on the severe accident analysis discussed in Section 9 results and is approximately the time when the CsI release from containment begins.
- Duration of Release - Duration of release of radionuclides from the plant is used to determine the dispersion of the release cloud. Each MAAP case for the ESBWR was performed for 72 hours after the onset of core damage. MACCS2 limits the duration of an individual plume to a maximum of 10 hours. Source terms in which the release flattens out after a short time (i.e., less than 10 hours) are characterized by a release duration corresponding to the time the release starts to the time the release flattens out. Each release fraction is reviewed in determining the release duration, with special attention given to the nuclides with the greatest offsite consequence impacts (i.e., iodine and cesium).
- NG – Release fraction of Noble gases from containment to the environment.
- CsI – Release fraction of Iodine from containment to the environment.

For this assessment no warning time is assumed due to no evacuation is credited. This is the time between official notification of public and release of radioactivity from the plant.

For each source term, the release is modeled to occur at ground level. The thermal content of the plume is assumed to be the same as ambient. These assumptions are conservative for early fatalities based on Reference 10-4.

A sensitivity study is presented in Section 10.5 on elevated release, buoyant plume rise, and meteorological conditions.

MAAP provides results for twelve (12) nuclide release fractions from containment to the atmosphere. These nuclide release fractions are related to the MACCS2 release groups as shown in Table 10.3-4.

Table 10.3-1

Site and Reactor Data for Meteorological Modeling

Parameter	Measurement, m (ft)
Reactor Building Length	49.0 (160)
Reactor Building Width	49.0 (160)
Reactor Building Height	48.0 (157)
Fuel Building Length	49.0 (160)
Fuel Building Height	24.0 (78)

Table 10.3-2

ESBWR Core Inventory

ESBWR Core Power is 4500 MWt			
Nuclide	**Bq/MWt**	**Nuclide**	**Bq/MWt**
Co-58	5.10E+12	Te-131m	1.42E+14
Co-60	4.92E+12	Te-132	1.41E+15
Kr-85	1.23E+13	I-131	9.90E+14
Kr-85m	2.73E+14	I-132	1.44E+15
Kr-87	5.27E+14	I-133	2.04E+15
Kr-88	7.42E+14	I-134	2.25E+15
Rb-86	2.35E+12	I-135	1.91E+15
Sr-89	9.93E+14	Xe-133	2.03E+15
Sr-90	9.76E+13	Xe-135	6.72E+14
Sr-91	1.25E+15	Cs-134	1.98E+14
Sr-92	1.34E+15	Cs-136	6.89E+13
Y-90	1.04E+14	Cs-137	1.28E+14
Y-91	1.27E+15	Ba-139	1.84E+15
Y-92	1.35E+15	Ba-140	1.77E+15
Y-93	1.55E+15	La-140	1.82E+15
Zr-95	1.79E+15	La-141	1.68E+15
Zr-97	1.85E+15	La-142	1.62E+15
Nb-95	1.80E+15	Ce-141	1.68E+15
Mo-99	1.90E+15	Ce-143	1.56E+15
Tc-99m	1.68E+15	Ce-144	1.36E+15
Ru-103	1.50E+15	Pr-143	1.53E+15
Ru-105	1.00E+15	Nd-147	6.69E+14
Ru-106	5.21E+14	Np-239	1.93E+16
Rh-105	9.10E+14	Pu-238	3.34E+12
Sb-127	1.03E+14	Pu-239	4.02E+11

Table 10.3-2

ESBWR Core Inventory

ESBWR Core Power is 4500 MWt			
Nuclide	**Bq/MWt**	**Nuclide**	**Bq/MWt**
Sb-129	3.15E+14	Pu-240	5.21E+11
Te-127	1.05E+14	Pu-241	1.51E+14
Te-127m	1.37E+13	Am-241	1.70E+11
Te-129	3.10E+14	Cm-242	4.01E+13
Te-129m	4.60E+13	Cm-244	1.94E+12

Table 10.3-3a

Event Release Parameter

24 Hours After the Onset of Core Damage

Source Term (1), (2)	Release Category	MAAP CASE	Relative Fraction (5)	Total Release Frequency (per year)	Time of Plume Release (hr)	Duration of Release (hr)(3)	NG (4) Release Fraction	CsI (4) Release Fraction
1	BOC	BOCsd_nIN_R1	0.500	1.47E-10	0.7	7.45	9.7E-01	7.0E-01
2		BOCdr_nIN_R1	0.500		0.6	5.5	2.4E-01	1.1E-01
3	BYP	T_nIN_BYP_R1	0.026	5.6E-11	0.7	8	9.5E-01	2.1E-01
4		T_nDP_nIN_BYP_R1	0.974		1.3	7.5	5.3E-01	3.3E-02
5	CCID	T_nIN_nD_CCID_R1	0.717	ε	25.8	10	0.0	0.0
6		T_nDP_nIN_nD_CCID_R1	0.283		16.0	2.6	9.1E-01	6.7E-02
7	CCIW	T_nIN_CCIW_R1	0.538	9.9E-11	25.6	10	0.0	0.0
8		T_nDP_nIN_CCIW_R1	0.462		18.4	10	6.4E-01	1.2E-04
9	EVE	T_nIN_nD_EVE_R1	1.000	6.10E-10	7.4	10	8.3E-01	2.8E-02
10	FR	T-AT_nIN_nCHR_FR_R1	1.000	ε	28.9	10	0.0	0.0
11	OPVB	T_nDP_nIN_VB_R1	0.050	6E-12	13.8	10	4.3E-01	1.33E-04
12		T_nIN_VB_R1	0.950		8.7	3	8.6E-01	5.0E-03
13	OPW1	T_nDP_nIN_nCHR_W1_R1	1.000	ε	34.2	10	0.0	0.0
14	OPW2	T_nDP_nIN_nCHR_W2_R1	1.000	ε	53.1	10	0.0	0.0
15	TSL	T_AT_nIN_TSL2x_R1	1.000	1.12E-08	0.5	7.5	2.7E-03	1.6E-04

Notes to Table 10.3-3a

(1) See Subsection 10.3.3 for definition of parameters in this table.

(2) For this bounding analysis, release height is ground level and release sensible heat is same as ambient.

(3) The release parameters are based on the 24 hours after the onset of core damage value. Each MAAP case for the ESBWR was performed for 72 hours after the onset of core damage. MACCS2 limits the duration of an individual plume to a maximum of 10 hours. Source terms in which the release flattens out after a short time (i.e., less than 10 hours) are characterized by a release duration corresponding to the time the release starts to the time the release flattens out. ,The nuclides with the greatest offsite consequences (i.e., Iodine and Cesium) are conservatively used.

(4) Noble Gases (NG) and Cesium Iodine (CsI) release fractions are the cumulative release fractions at 24 hours after the onset of core damage.

(5) The relative fraction is the relative contribution of each of the representative sequences to their release category.

ε Less than 1E-12

Table 10.3-3b

Event Release Parameter

72 Hours After the Onset of Core Damage

Source Term (1), (2)	Release Category	MAAP CASE	Relative Fraction (5)	Total Release Frequency (per year)	Time of Plume Release (hr)	Duration of Release (hr)(3)	NG (4) Release Fraction	CsI (4) Release Fraction
1	BOC	BOCsd_nIN_R1	0.500	1.47E-10	0.7	7.45	9.8E-01	7.0E-01
2		BOCdr_nIN_R1	0.500		0.6	5.5	2.6E-01	1.3E-01
3	BYP	T_nIN_BYP_R1	0.026	5.6E-11	0.7	8	9.7E-01	3.0E-01
4		T_nDP_nIN_BYP_R1	0.974		1.3	7.5	6.8E-01	3.5E-02
5	CCID	T_nIN_nD_CCID_R1	0.717	ε	25.8	10	9.1E-01	6.2E-02
6		T_nDP_nIN_nD_CCID_R1	0.283		16.0	2.6	9.6E-01	3.5E-01
7	CCIW	T_nIN_CCIW_R1	0.538	9.9E-11	25.6	10	8.9E-01	1.6E-05
8		T_nDP_nIN_CCIW_R1	0.462		18.4	10	8.3E-01	1.1E-02
9	EVE	T_nIN_nD_EVE_R1	1.000	6.10E-10	7.4	10	8.3E-01	1.5E-01
10	FR	T-AT_nIN_nCHR_FR_R1	1.000	ε	28.9	10	1.0E+00	6.1E-03
11	OPVB	T_nDP_nIN_VB_R1	0.050	6E-12	13.8	10	9.6E-01	4.1E-03
12		T_nIN_VB_R1	0.950		8.7	3	1.0E+00	1.5E-02
13	OPW1	T_nDP_nIN_nCHR_W1_R1	1.000	ε	34.2	10	1.0E+00	1.5E-02
14	OPW2	T_nDP_nIN_nCHR_W2_R1	1.000	ε	53.1	10	1.0E+00	1.5E-02
15	TSL	T_AT_nIN_TSL2x_R1	1.000	1.12E-08	0.5	7.5	2.7E-03	1.6E-04

Notes to Table 10.3-3b

(1) See Subsection 10.3.3 for definition of parameters in this table.

(2) For this bounding analysis, release height is ground level and release sensible heat is same as ambient.

(3) Each MAAP case for the ESBWR was performed for 72 hours after the onset of core damage. MACCS2 limits the duration of an individual plume to a maximum of 10 hours. Source terms in which the release flattens out after a short time (i.e., less than 10 hours) are characterized by a release duration corresponding to the time the release starts to the time the release flattens out. In general, the nuclides with the greatest offsite consequences (i.e., Iodine and Cesium) are conservatively used.

(4) Noble Gases (NG) and Cesium Iodine (CsI) release fractions are the cumulative release fractions at 72 hours after the onset of core damage.

(5) The relative fraction is the relative contribution of each of the representative sequences to their release category

ε Less than 1E-12

Table 10.3-4

MACCS2 Release Groups vs. ESBWR Release Groups

MACCS2 Release Groups	MAAP Release Groups	MAAP Output Parameter
1-Xe/Kr	Noble gases	FREL (1)
2-I	CsI	FREL (2)
73-Cs	CsOH	FREL (6)
4-Te	TeO2 (1) (Sb (1) & Te2 (2) fractions are included)	FREL (3), FREL (10) and FREL (11)
5-Sr	SrO	FREL (4)
6-Ru	MoO2 (Mo is in Ru MACCS category)	FREL (5)
7-La	La2O3	FREL (8)
8-Ce	CeO2 (included UO2 (2) in this category)	FREL (9) and FREL (12)
9-Ba	BaO	FREL (7)

(1) The larger release fraction of TeO2 and Sb is used as input into MACCS2.
(2) Te2 and UO2 release fractions are negligible.

10.4 COMPARISON OF RESULTS TO GOALS

10.4.1 Goals

Three major offsite consequence-related goals are established in the GE ESBWR Licensing Review Bases based on the NRC Safety Goal Policy Statement. These goals are:

(1) **Individual Risk Goal**

The risk to an average individual in the "vicinity" of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one tenth of one percent (0.1%) of the sum of "prompt fatality risks" resulting from other accidents to which members of the U.S. Population are generally exposed.

As noted in the Safety Goal Policy statement, "vicinity" is defined as the area within 1.61 km (1 mile) of the plant site boundary. "Prompt Fatality Risks" are defined as those risks to which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities. Such risks are the sum of risks that result in fatalities from such activities as driving, household chores, occupational activities, etc.

For this evaluation, the sum of prompt fatality risks is taken as the U.S. accidental death risk value of 39.1 deaths per 100,000 people per year based upon Reference 10-7.

As a design objective, the Individual Risk Goal is conservatively set to be one order of magnitude lower, which gives a Individual Risk Goal of 3.9E-8 at 0-1 mile.

(2) **Societal Risk Goal**

The risk to the population in the area "near" a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one tenth of one percent (0.1%) of the sum of the "cancer fatality risks" resulting from all other causes. As noted in the Safety Goal Policy Statement, "near" is defined as within 16.1 km (10 miles) of the plant. The "cancer fatality risk" is taken as 169 deaths per 100,000 people per year based upon 1983 statistics in Reference 10-8.

Similar to the Individual Risk Goal, for design objective, the Societal Risk Goal is set to be 1.7E-7 at 0-10 mile

(3) **Radiation Dose Goal**

The probability of exceeding a whole body dose of 0.25 Sv at a distance of 805 m (one half mile) from the reactor shall be less than one in a million per reactor year.

The design objective for the probability of having 0.25 Sv at 0.5 mile is set at less than 1E-which is an order of magnitude lower than the NRC dose goal.7. The calculated ESBWR consequence results are compared to these goals in the following subsection.

10.4.2 Results

The mean results from the offsite consequence analysis for each source term are shown in Table 10.4-1a and Table 10.4-1b.

Table 10.4-1a provides the results 24 hours after the onset of core damage, and Table 10.4-1b provides the results 72 hours after the onset of core damage.

The 24 hour mission time after the onset of core damage is the typical time used for probabilistic risk analysis. The 72 hours mission time is the conservative time used for passive ESBWR design evaluation.

These results are multiplied by the annual release frequency for each source term and then summed to obtain the risk weighted mean consequence results. These results are compared to the consequence goals identified in Subsection 10.4.1 and summarized in Table 10.4-2.

A plot of whole body dose at a distance of 805 m (one half mile) against cumulative probability is shown in Figure 10.4-1. As can be seen, the whole body dose at 805m (0.5 miles) over the entire dose spectrum from 0.1 Sv to >100 Sv is well below the goal of 1E-6/yr exceedance frequency.

The individual risk and societal risk goals are maintained with sufficient margin as shown in Table 10.4-2 and all the risk measures are several orders of magnitude lower than the risk goals.

Based upon these results, the ESBWR meets the established consequence related goals with substantial margin.

Table 10.4-1a
MACCS2 Results by Source Term
24 Hour After Onset of Core Damage

Source Term (10)	Individual Risk (0-1 mile) (1)	Weighted Individual Risk (per year) (2)	Weighted Individual Risk Contribution (%) (3)	Societal Risk (0-10 miles) (4)	Weighted Societal Risk (per year) (5)	Weighted Societal Risk Contribution (%) (6)	Probability of Dose > .2 Sv (0-0.5 mile)(7)	Weighted Prob of Exceedance (per year) (8)	Weighted Dose Contribution (%) (9)
1	1.09E-01	8E-12	10.88%	1.88E-02	1E-12	14.83%	1.00E+00	7.4E-11	3.62%
2	7.96E-02	6E-12	7.95%	7.35E-03	ε	5.80%	1.00E+00	7.4E-11	3.62%
3	9.56E-02	ε	0.19%	1.39E-02	ε	0.22%	1.00E+00	1E-12	0.07%
4	9.17E-02	5E-12	6.81%	4.32E-02	2E-12	25.36%	1.00E+00	5.5E-11	2.70%
5	0	0	0.00%	0	0	0.00%	0	0	0.00%
6	5.64E-02	ε	0.02%	3.27E-03	ε	0.01%	1.00E+00	ε	0.01%
7	0	0	0.00%	0	0	0.00%	0	0	0.00%
8	0	0	0.00%	7.26E-05	ε	0.04%	5.62E-02	3E-12	0.13%
9	8.95E-02	5.5E-11	73.92%	6.79E-03	4E-12	44.30%	1.00E+00	6.10E-10	29.97%
10	0	0	0.00%	0	0	0.00%	0	0	0.00%
11	0	0	0.00%	9.02E-05	ε	0.00%	2.93E-01	ε	0.00%
12	3.10E-02	ε	0.23%	1.22E-03	ε	0.07%	8.06E-01	5E-12	0.22%
13	0	0	0.00%	0	0	0.00%	0	0	0.00%
14	0	0	0.00%	0	0	0.00%	0	0	0.00%
15	0	0	0.00%	7.80E-05	ε	9.38%	1.08E-01	1.21E-09	59.66%
Total	--	7.4E-11	100.00%	--	9E-12	100.00%	--	2.03E-09	100.00%

Notes to Table 10.4-1a

(1) The individual risk is calculated as the total number of early fatalities within one mile divided by the total one mile population

(2) The weighted individual risk is the individual risk per year and is calculated as the product of the release category release frequency and the release category individual risk.

(3) The weighted individual risk contribution is the percentage of a release category's weighted individual risk to the total weighted individual risk.

(4) The societal risk is calculated as the total number of latent fatalities within ten miles divided by the total ten mile population.

(5) The weighted societal risk is the societal risk per year and is calculated as the product of the release category release frequency and the release category societal risk.

(6) The weighted societal risk contribution is the percentage of a release category's weighted societal risk to the total weighted societal risk.

(7) The probability of dose greater than 0.2 Sv is obtained from the MACCS2 output file and is provided in the form of CCDF tables.

(8) The weighted probability of exceedance is the probability of exceeding a dose greater than 0.2 Sv per year and is calculated as the product of the release category release frequency and the release category MACCS2 probability of dose greater than 0.2 Sv.

(9) The weighted probability of exceedance contribution is the percentage of a release category's weighted societal risk to the total weighted societal risk

(10) The source term definition is the same as defined in Table 10.3-3a.

ε Less than 1E-12.

Table 10.4-1b
MACCS2 Results by Source Term
72 Hour After Onset of Core Damage

Source Term [10]	Individual Risk (0-1 mile) [1]	Weighted Individual Risk (per year) [2]	Weighted Individual Risk Contribution (%) [3]	Societal Risk (0-10 miles) [4]	Weighted Societal Risk (per year) [5]	Weighted Societal Risk Contribution (%) [6]	Probability of Dose > .2 Sv (0-0.5 mile) [7]	Weighted Prob of Exceedance (per year) [8]	Weighted Dose Contribution (%) [9]
1	1.10E-01	8E-12	9.89%	1.85E-02	1E-12	12.64%	1.00E+00	7.4E-11	3.52%
2	8.04E-02	ε	7.23%	7.51E-03	ε	5.13%	1.00E+00	7.4E-11	3.52%
3	9.98E-02	ε	0.18%	1.42E-02	ε	0.19%	1.00E+00	1E-12	0.07%
4	9.21E-02	5E-12	6.17%	4.29E-02	2E-12	21.82%	1.00E+00	5.5E-11	2.62%
5	7.52E-02	ε	0.06%	4.53E-03	ε	0.03%	1.00E+00	ε	0.03%
6	6.85E-02	ε	0.02%	5.94E-03	ε	0.01%	1.00E+00	ε	0.01%
7	3.83E-02	2E-12	2.49%	1.28E-03	ε	0.63%	3.06E-01	1.6E-11	0.78%
8	2.77E-02	1E-12	1.55%	1.57E-03	ε	0.67%	1.00E+00	4.6E-11	2.19%
9	9.66E-02	5.9E-11	71.87%	8.91E-03	5E-12	50.37%	1.00E+00	6.10E-10	29.09%
10	7.87E-02	ε	0.00%	4.19E-03	ε	0.00%	9.36E-01	ε	0.00%
11	5.99E-02	ε	0.02%	2.71E-03	ε	0.01%	1.00E+00	ε	0.01%
12	7.57E-02	ε	0.52%	6.05E-03	ε	0.31%	9.82E-01	5E-12	0.26%
13	3.65E-03	ε	0.00%	7.29E-04	ε	0.00%	1.00E+00	ε	0.00%
14	1.28E-03	ε	0.00%	3.79E-04	ε	0.00%	8.55E-01	ε	0.00%
15	0	0	0.00%	7.85E-05	ε	8.18%	1.08E-01	1.21E-09	57.91%
Total		8.2E-11	100.00%	--	1.1E-11	100.00%	--	2.10E-09	100.00%

Notes to Table 10.4-1b

(1) The individual risk is calculated as the total number of early fatalities within one mile divided by the total one mile population

(2) The weighted individual risk is the individual risk per year and is calculated as the product of the release category release frequency and the release category individual risk.

(3) The weighted individual risk contribution is the percentage of a release category's weighted individual risk to the total weighted individual risk.

(4) The societal risk is calculated as the total number of latent fatalities within ten miles divided by the total ten mile population.

(5) The weighted societal risk is the societal risk per year and is calculated as the product of the release category release frequency and the release category societal risk.

(6) The weighted societal risk contribution is the percentage of a release category's weighted societal risk to the total weighted societal risk.

(7) The probability of dose greater than 0.2 Sv is obtained from the MACCS2 output file and is provided in the form of CCDF tables.

(8) The weighted probability of exceedance is the probability of exceeding a dose greater than 0.2 Sv per year and is calculated as the product of the release category release frequency and the release category MACCS2 probability of dose greater than 0.2 Sv.

(9) The weighted probability of exceedance contribution is the percentage of a release category's weighted societal risk to the total weighted societal risk

(10) The source term definition is the same as defined in Table 10.3-3a.

ε Less than 1E-12..

Table 10.4-2
Baseline Consequence Goals and Results

Goal	Risk Goal	ESBWR 24 Hours After Onset of Core Damage (Ground Release)	Safety Goal Achieved 24 Hours After the Onset of Core Damage	ESBWR 72 Hours After Onset of Core Damage (Ground Release)	Safety Goal Achieved 72 Hours After the Onset of Core Damage
Individual Risk (0 – 1 Mile)	$<3.9\times10^{-7}$ (0.1%)	7.4E-11	YES	8.2E-11	YES
Societal Risk (0 – 10 Mile)	$<1.7\times10^{-6}$ (0.1%)	9E-12	YES	1.1E-11	YES
Radiation Dose Probability at 0.25 Sv (0 – 0.5 Mile)	$<10^{-6}$	2.03E-09	YES	2.10E-09	YES

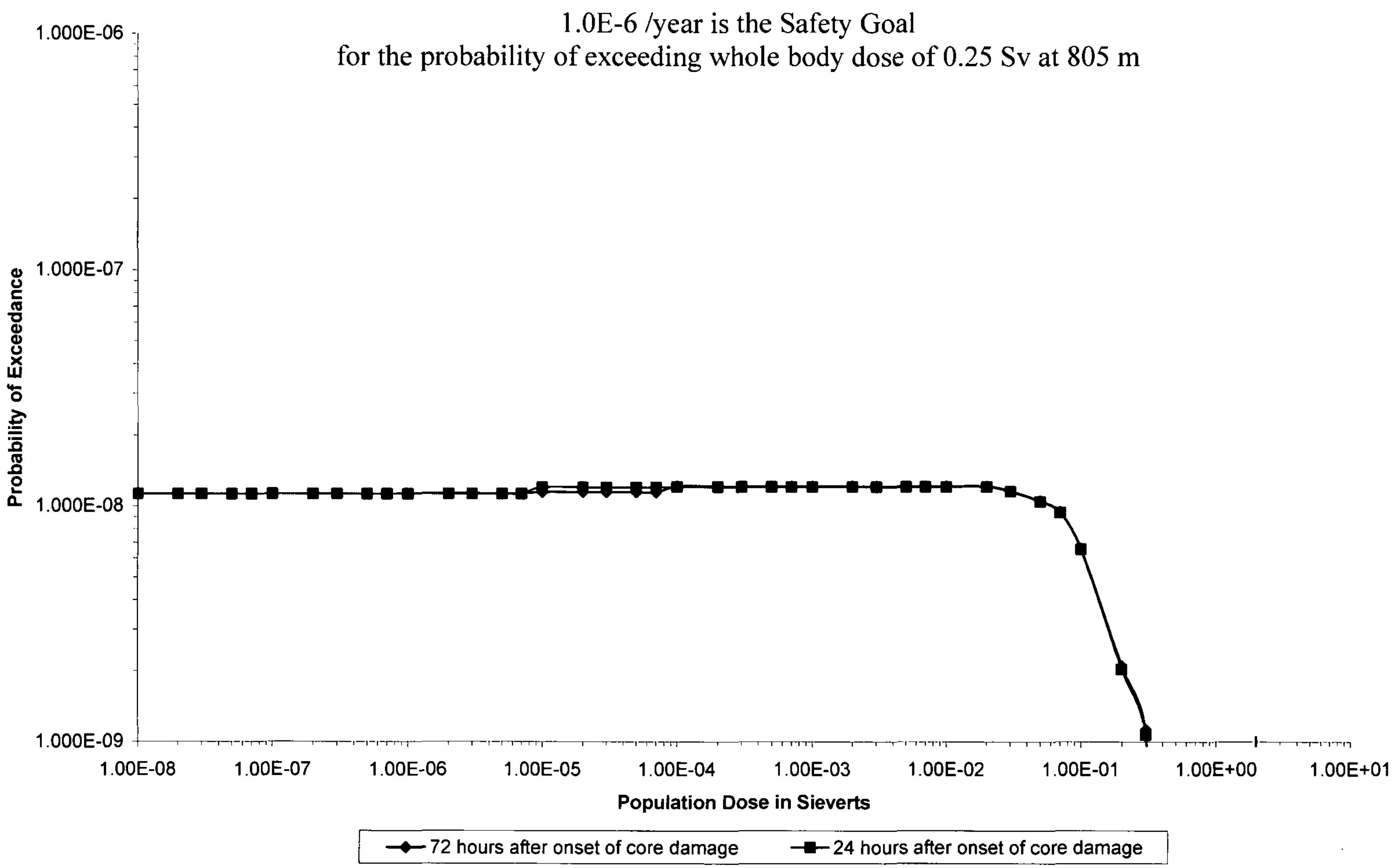

Figure 10.4-1. Whole Body Dose at 805 m (0.5 Mile) as Probability of Exceedance

*The goal of a maximum probability of 1E-6 is well above the entire dose range at 0.5 mile.

10.5 SENSITIVITY STUDY

For this sensitivity study, two meteorological conditions are studied. The first is used for the ESBWR Level 3 base case study and is comparable with the of ALWR URD meteorological reference data.

The second, the sensitivity meteorological condition case, represents a narrower distribution condition. The narrower distribution can represent conservative radiological consequences in certain wind sectors and with certain stability classes. The goal of the sensitivity study is to reveal the radiological consequence insights with regard to the three risk goals.

Elevated release with and without buoyant plume energy rise is studied along with sensitivity on population density.

The sensitivity study results show that the three risk goals stated in Subsection 10.4.1 are bounded with substantial margin. In addition, the design goals stated in Subsection 10.4.1, are set at one tenth of the NRC risk goals.

The sensitivity study results show that the baseline case is sufficiently bounding to allow the variation of inputs and assumptions while maintaining several orders of magnitude of design margin relative to the NRC risk goals Table 10.5-1 through Table 10.5-7b present the sensitivity study results.

Table 10.5-1 shows the sensitivity study results summary. As shown in Table 10.5-1, the three NRC risk goals and the conservative design risk goals are adequate to envelop the variations of MACCS2 input parameters and assumptions, with several orders of magnitude margin. The results also indicate that the variation of certain MACCS2 input parameters, such as the meteorological conditions, would result in minute changes in relation to the three risk goals measures. However, the magnitude of changes due to these input parameter variations are still well bounded by, or still several orders of magnitude lower than, the risk goals. Though the three risk goals are not applicable at 50 miles, to show the impact of elevated release, Table 10.5-2a and 10.5-2b show that the population dose at 50 miles do not vary much for ground vs. elevated release for 24 hour and 72 hour mission time. The risk insights obtained via ground release modeling at 50 miles does not change even with elevated release modeling.

Table 10.5-3a and 10.5-3b show the ground vs. elevated release at 0 to 10 miles for 24 hour and 72 hour mission time. Similar to Table 10.5-2a and 10.5-2b, at 0-10 miles, the risk goals measures are still within the same order of magnitude between ground and elevated releases. The importance here is the risk insights are maintained whether modeled as ground or elevated releases.

Table 10.5-4 shows the ground release results with hourly meteorological data sampling of 72 hour mission time. The intent of this case is to show that the risk insights obtained with different meteorological data sampling method (more conservative hourly sampling method vs. binning sampling method) does not change the risk goal measures results and conclusions.

Table 10.5-5a and 10.5-5b show the ground release with sensitivity meteorological data for 24 hour and 72 hour mission time. The sensitivity meteorological data represents a different and narrower distribution comparing with the base case meteorological data set. The risk goal measures obtained from this different meteorological condition show that both the NRC risk goals and the design risk goals are well maintained with good margin. The important risk insight is the public health and safety is well maintained as shown from measures of the various risk goals. The intent of the design goals implemented is to maintain the ESBWR design to have a radiological risk as low as reasonably achievable approach.

Table 10.5-6a and 10.5-6b show the elevated release with sensitivity meteorological data for 24 hour and 72 hour mission time. Similar to results presented in Table 10.5-5a and Table 10.5-5b, the elevated release approach is added in Table 10.5-6a and 10.5-6b. Already has a very small high point release probability, the results indicate again that, the ESBWR design maintains a high confidence to protect the health and safety of the public the risk measures are well within the goal settings.

Table 10.5-7a and 10.5-7b show the elevated release with the sensitivity meteorological data and buoyant plume rise for both 24 hour and 72 hour mission times. To further support the conclusion, a buoyant plume energy is added to the cases presented in Table 10.5-6a and 10.5-6b. As shown, the added plume rise does not alter the conclusion and risk insights, and again, the results show that public health and safety is well maintained by the ESBWR design.

Table 10.5-1

Sensitivity Case Results Summary

Design Goal	Risk Goal	Design Goal (10% of Risk Goal)	Meteorological Data [1], [2]	24 Hrs. After Onset of Core Damage (Ground Release)	24 Hrs. After Onset of Core Damage (Elevated Release)	24 Hrs. After Onset of Core Damage (Elevated With Buoyant)	72 Hrs. After Onset of Core Damage (Ground Release)	72 Hrs. After Onset of Core Damage (Ground Release with Hourly Sampling)	72 Hrs. After Onset of Core Damage (Elevated Release)	72 Hrs. After Onset of Core Damage (Elevated Release with Buoyant)	Within Design Goal
Individual Risk (0 – 1 Mile)	$<3.9\text{x}10^{-7}$	$<3.9\text{x}10^{-8}$	Case 1	7.4E-11	7.0E-11		8.2E-11	8.2E-11	7.9E-11		YES
			Case 2	9.1E-11	8.8E-11	7.9E-11	9.8E-11		7.9E-11	8.7E-11	YES
Societal Risk (0 – 10 Mile)	$<1.7\text{x}10^{-6}$	$<1.7\text{x}10^{-7}$	Case 1	9E-12	1.0E-11		1.1E-11	1.1E-11	1.2E-11		YES
			Case 2	1.2E-11	1.3E-11	1.4E-11	1.4E-11		1.2E-11	1.6E-11	YES
Radiation Dose Probability at 0.25 Sv (0 – 0.5 Mile)	$<10^{-6}$	$<10^{-7}$	Case 1	2.03E-09	1.22E-09		2.10E-09	2.09E-09	1.28E-09		YES
			Case 2	1.41E-09	1.05E-09	1.05E-09	1.49E-09		1.28E-09	1.10E-09	YES

Notes to Table 10.5-1

(1) Case 1 is using the base meteorological reference data as stated in Subsection 10.2.1.

(2) Case 2 is using a sensitivity meteorological reference data, which has a narrower meteorological distribution

Table 10.5-2a

Sensitivity Case

Ground vs. Elevated Release at 0-50 miles (24 hrs)

Source Term (24 Hrs)	Population Dose (50 miles) (Ground)	Weighted Population Dose (50 mile, Ground)	Population Dose (0-50 miles) (Elevated)	Weighted Population Dose (50 mile, Elevated)
1	5.65E+05	4.16E-05	6.10E+05	4.50E-05
2	1.11E+05	8.18E-06	1.20E+05	8.84E-06
3	2.29E+05	3.35E-07	2.48E+05	3.63E-07
4	5.53E+05	3.03E-05	6.05E+05	3.32E-05
5	0	0	0	0
6	5.80E+04	1.48E-08	6.33E+04	1.62E-08
7	0	0	0	0
8	7.46E+02	3.42E-08	8.08E+02	3.70E-08
9	1.35E+05	8.23E-05	1.46E+05	8.90E-05
10	0	0	0	0
11	9.90E+02	2.91E-10	1.09E+03	3.21E-10
12	1.36E+04	7.60E-08	1.47E+04	8.22E-08
13	0	0	0	0
14	0	0	0	0
15	7.85E+02	8.82E-06	8.97E+02	1.01E-05
Total		1.72E-04		1.87E-04

Table 10.5-2b

Sensitivity Case

Ground vs. Elevated Release at 0-50 miles (72 hrs)

Source Term (24 Hrs)	Population Dose (50 miles) (Ground)	Weighted Population Dose (50 mile, Ground)	Population Dose (0-50 miles) (Elevated)	Weighted Population Dose (50 mile, Elevated)
1	5.78E+05	4.26E-05	6.25E+05	4.61E-05
2	1.17E+05	8.62E-06	1.27E+05	9.36E-06
3	2.89E+05	4.23E-07	3.13E+05	4.58E-07
4	5.58E+05	3.06E-05	6.09E+05	3.34E-05
5	9.93E+04	6.42E-08	1.08E+05	6.98E-08
6	1.32E+05	3.37E-08	1.42E+05	3.62E-08
7	1.02E+04	5.44E-07	1.10E+04	5.87E-07
8	4.35E+04	1.99E-06	4.71E+04	2.16E-06
9	2.01E+05	1.23E-04	2.17E+05	1.32E-04
10	5.40E+04	5.4E-11	5.90E+04	5.9E-11
11	4.32E+04	1.27E-08	4.66E+04	1.37E-08
12	1.13E+05	6.32E-07	1.23E+05	6.88E-07
13	2.34E+04	2.3E-11	2.58E+04	2.6E-11
14	5.41E+03	5E-12	6.10E+03	6E-12
15	7.89E+02	8.87E-06	9.01E+02	1.01E-05
Total		2.17E-04		2.35E-04

Table 10.5-3a

Sensitivity Case

Ground vs. Elevated Release at 0-10 miles (24 hrs)

Source Term (24 Hrs)	Weighted Individual Risk (0-1 mile) (Ground)	Weighted Individual Risk (0-1 mile) (Elevated)	Weighted Societal Risk (0-10 mile) (Ground)	Weighted Societal Risk (0-10 mile) (Elevated)	Weighted Probability of Exceedance of 0.2 Sv (0-0.5 mile) (Ground)	Weighted Probability of Exceedance of 0.2 Sv (0-0.5 mile) (Elevated)
1	8E-12	8E-12	1E-12	1E-12	7.4E-11	7.4E-11
2	6E-12	6E-12	ε	ε	7.4E-11	7.4E-11
3	ε	ε	ε	ε	1E-12	1E-12
4	5E-12	5E-12	2E-12	3E-12	5.5E-11	5.5E-11
5	0	0	0	0	0	0
6	ε	ε	ε	ε	ε	ε
7	0	0	0	0	0	0
8	0	0	ε	ε	3E-12	2E-12
9	5.5E-11	5.2E-11	4E-12	5E-12	6.10E-10	6.10E-10
10	0	0	0	0	0	0
11	0	0	ε	ε	ε	ε
12	ε	ε	ε	ε	5E-12	4E-12
13	0	0	0	0	0	0
14	0	0	0	0	0	0
15	0	0	ε	ε	1.21E-09	4.05E-10
Total	7.4E-11	7.0E-11	9E-12	1.0E-11	2.03E-09	1.22E-09

ε Less than 1E-12.

Table 10.5-3b[yg22]

Sensitivity Case

Ground vs. Elevated Release at 0-10 miles (72 hrs)

Source Term (24 Hrs)	Weighted Individual Risk (0-1 mile) (Ground)	Weighted Individual Risk (0-1 mile) (Elevated)	Weighted Societal Risk (0-10 mile) (Ground)	Weighted Societal Risk (0-10 mile) (Elevated)	Weighted Probability of Exceedance of 0.2 Sv (0-0.5 mile) (Ground)	Weighted Probability of Exceedance of 0.2 Sv (0-0.5 mile) (Elevated)
1	8E-12	8E-12	1E-12	1E-12	7.4E-11	7.4E-11
2	6E-12	6E-12	ε	ε	7.4E-11	7.4E-11
3	ε	ε	ε	ε	1E-12	1E-12
4	5E-12	5E-12	2E-12	2E-12	5.5E-11	5.5E-11
5	ε	ε	ε	ε	ε	ε
6	ε	ε	ε	ε	ε	ε
7	2E-12	1E-12	ε	ε	1.6E-11	6E-12
8	1E-12	ε	ε	ε	4.6E-11	4.6E-11
9	5.9E-11	5.8E-11	5E-12	6E-12	6.10E-10	6.10E-10
10	ε	ε	ε	ε	ε	ε
11	ε	ε	ε	ε	ε	ε
12	ε	ε	ε	ε	5E-12	5E-12
13	ε	ε	ε	ε	ε	ε
14	ε	ε	ε	ε	ε	ε
15	0	0	ε	ε	1.21E-09	4.05E-10
Total	8.2E-11	7.9E-11	1.1E-11	1.2E-11	2.10E-09	1.28E-09

ε Less than 1E-12.

Table 10.5-4

Sensitivity Case

Ground Release with Hourly Met Data Sampling (72 hrs)

Source Term (72 Hrs)	Individual Risk (0-1 mile)	Weighted Individual Risk (per year)	Weighted Individual Risk Contribution (%)	Societal Risk (0-10 miles)	Weighted Societal Risk (per year)	Weighted Societal Risk Contribution (%)	Probability of Dose > .2 Sv (0-0.5 mile)	Weighted Probability of Exceedance (per year)	Weighted Dose Contribution (%)	Release Fraction
1	1.09E-01	8E-12	9.83%	1.83E-02	1E-12	12.57%	1.00E+00	7.4E-11	3.53%	0.5
2	8.06E-02	6E-12	7.27%	7.42E-03	ε	5.10%	1.00E+00	7.4E-11	3.53%	0.5
3	9.90E-02	ε	0.18%	1.41E-02	ε	0.19%	1.00E+00	1E-12	0.07%	0.026
4	9.20E-02	5E-12	6.18%	4.26E-02	2E-12	21.79%	1.00E+00	5.5E-11	2.63%	0.974
5	7.66E-02	ε	0.06%	4.49E-03	ε	0.03%	1.00E+00	ε	0.03%	0.717
6	6.83E-02	ε	0.02%	5.89E-03	ε	0.01%	1.00E+00	ε	0.01%	0.283
7	3.91E-02	2E-12	2.55%	1.29E-03	ε	0.64%	3.22E-01	1.7E-11	0.82%	0.538
8	2.84E-02	1E-12	1.59%	1.59E-03	ε	0.68%	1.00E+00	4.6E-11	2.20%	0.462
Note 1	9.57E-02	ε	0.00%	8.39E-03	ε	0.00%	1.00E+00	ε	0.00%	1
9	9.62E-02	5.9E-11	71.78%	8.88E-03	5E-12	50.48%	1.00E+00	6.10E-10	29.23%	1
10	8.00E-02	ε	0.00%	4.16E-03	ε	0.00%	9.48E-01	ε	0.00%	0.5
Note 2	8.36E-02	ε	0.00%	4.71E-03	ε	0.00%	9.96E-01	ε	0.00%	0.5
11	6.11E-02	ε	0.02%	2.70E-03	ε	0.01%	1.00E+00	ε	0.01%	0.05
12	7.51E-02	ε	0.51%	6.00E-03	ε	0.31%	9.93E-01	6E-12	0.27%	0.95
13	3.45E-03	ε	0.00%	7.35E-04	ε	0.00%	1.00E+00	ε	0.00%	1
14	1.16E-03	ε	0.00%	3.81E-04	ε	0.00%	8.38E-01	ε	0.00%	1
15	0	0	0.00%	7.82E-05	ε	8.19%	1.07E-01	1.20E-09	57.66%	1
Total	--	8.2E-11	100.00%	--	1.1E-11	100.00%	--	2.09E-09	100.00%	

Notes to Table 10.5-4

Note 1 The DCH case is added for the sensitivity study.

Note 2 The T-AT_nIN_nCHR_FR50_R1 case is added for the sensitivity study

ε Less than 1E-12.

Table 10.5-5a[yg23]

Sensitivity Case

Ground Release With Sensitivity Met Data (24 hrs)

Source Term (24 Hrs)	Individual Risk (0-1 mile)	Weighted Individual Risk (per year)	Weighted Individual Risk Contribution (%)	Societal Risk (0-10 miles)	Weighted Societal Risk (per year)	Weighted Societal Risk Contribution (%)	Probability of Dose > .2 Sv (0-0.5 mile)	Weighted Probability of x (per year)	Weighted Dose Contribution (%)	Release Fraction
1	1.16E-01	9E-12	9.37%	1.77E-02	1E-12	10.50%	1.00E+00	7.4E-11	5.21%	0.5
2	9.64E-02	7E-12	7.79%	1.06E-02	ε	6.29%	1.00E+00	7.4E-11	5.21%	0.5
3	1.15E-01	ε	0.18%	1.94E-02	ε	0.23%	1.00E+00	1E-12	0.10%	0.026
4	1.13E-01	6E-12	6.80%	6.19E-02	3E-12	27.34%	1.00E+00	5.5E-11	3.88%	0.974
5	0	0	0.00%	0	0	0.00%	0	0	0.00%	0.717
6	7.49E-02	ε	0.02%	4.56E-03	ε	0.01%	1.00E+00	ε	0.02%	0.283
7	0	0	0.00%	0	0	0.00%	0	0	0.00%	0.538
8	0	0	0.00%	8.89E-05	ε	0.03%	2.18E-02	ε	0.07%	0.462
Note 1	1.11E-01	ε	0.00%	7.88E-03	ε	0.00%	1.00E+00	ε	0.00%	1
9	1.13E-01	6.9E-11	75.56%	9.43E-03	6E-12	46.29%	1.00E+00	6.10E-10	43.13%	1
10	0	0	0.00%	0	0	0.00%	0	0	0.00%	0.5
Note 2	3.00E-02	ε	0.00%	1.15E-03	ε	0.00%	2.51E-01	ε	0.00%	0.5
11	0	0	0.00%	1.13E-04	ε	0.00%	1.53E-01	ε	0.00%	0.05
12	4.41E-02	ε	0.27%	1.67E-03	9.34E-1	0.08%	9.87E-01	6E-12	0.39%	0.95
13	0	0	0.00%	0	0	0.00%	0	0	0.00%	1
14	0	0	0.00%	0	0	0.00%	0	0	0.00%	1
15	0	0	0.00%	1.02E-04	1E-12	9.23%	5.28E-02	5.93E-10	41.98%	1
Total	--	9.1E-11	100.00%	--	1.2E-11	100.00%	--	1.41E-09	100.00%	

Notes to Table 10.5-5a

Note 1 The DCH case is added for the sensitivity study.

Note 2 The T-AT_nIN_nCHR_FR50_R1 case is added for the sensitivity study.

ε Less than 1E-12

Table 10.5-5b[yg24]

Sensitivity Case

Ground Release With Sensitivity Met Data (72 hrs)

Source Term (72 Hrs)	Individual Risk (0-1 mile)	Weighted Individual Risk (per year)	Weighted Individual Risk Contribution (%)	Societal Risk (0-10 miles)	Weighted Societal Risk (per year)	Weighted Societal Risk Contribution (%)	Probability of Dose > .2 Sv (0-0.5 mile)	Weighted Probability of Exceedance (per year)	Weighted Dose Contribution (%)	Release Fraction
1	1 16E-01	9E-12	8 77%	1.69E-02	1E-12	8.73%	1.00E+00	7 4E-11	4 95%	0.5
2	9.66E-02	7E-12	7.30%	1.08E-02	ε	5.58%	1.00E+00	7 4E-11	4.95%	0.5
3	1.16E-01	ε	0.17%	1 85E-02	ε	0 19%	1.00E+00	1E-12	0.10%	0 026
4	1.14E-01	6E-12	6.41%	6 14E-02	3E-12	23.61%	1.00E+00	5.5E-11	3.68%	0.974
5	1.02E-01	ε	0 07%	6.29E-03	ε	0.03%	1.00E+00	ε	0 04%	0.717
6	8.00E-02	ε	0 02%	8.31E-03	ε	0.01%	1.00E+00	ε	0 02%	0 283
7	5.50E-02	3E-12	3.01%	1.79E-03	ε	0.67%	4.03E-01	2 2E-11	1.44%	0.538
8	3 72E-02	2E-12	1.75%	2 03E-03	ε	0.65%	1.00E+00	4.6E-11	3.08%	0.462
Note 1	1 15E-01	ε	0.00%	1.15E-02	ε	0.00%	1.00E+00	ε	0.00%	1
9	1.15E-01	7.0E-11	71.92%	1.22E-02	7E-12	52.15%	1.00E+00	6.10E-10	40.95%	1
10	1.06E-01	ε	0.00%	6.02E-03	ε	0.00%	9 95E-01	ε	0 00%	0 5
Note 2	1.09E-01	ε	0.00%	6.84E-03	ε	0.00%	1 00E+00	ε	0.00%	0.5
11	8.36E-02	ε	0.03%	3 81E-03	ε	0.01%	1.00E+00	ε	0.02%	0.05
12	9.61E-02	ε	0.55%	8 35E-03	ε	0 33%	9.98E-01	6E-12	0.37%	0 95
13	4.94E-04	ε	0.00%	8.74E-04	ε	0.00%	1.00E+00	ε	0 00%	1
14	9.91E-05	ε	0.00%	4.31E-04	ε	0.00%	9 97E-01	ε	0 00%	1
15	0	0	0.00%	1 02E-04	1E-12	8.04%	5.35E-02	6 01E-10	40 39%	1
Total	--	9 8E-11	100.00%	–	1.4E-11	100.00%	–	1.49E-09	100.00%	--

Notes to Table 10.5-5b

Note 1 The DCH case is added for the sensitivity study.

Note 2 The T-AT_nIN_nCHR_FR50_R1 case is added for the sensitivity study.

e Less than 1E-12.

Table 10.5-6a

Sensitivity Case

Elevated Release With Sensitivity Met Data (24 hrs)

Source Term (24 Hrs)	Individual Risk (0-1 mile)	Weighted Individual Risk (per year)	Weighted Individual Risk Contribution (%)	Societal Risk (0-10 miles)	Weighted Societal Risk (per year)	Weighted Societal Risk Contribution (%)	Probability of Dose > .2 Sv (0-0.5 mile)	Weighted Probability of Exceedance (per year)	Weighted Dose Contribution (%)	Release Fraction
1	1.16E-01	9E-12	9.77%	1.60E-02	1E-12	8.78%	1.00E+00	7.4E-11	7 03%	0.5
2	9.23E-02	7E-12	7.78%	1.17E-02	ε	6.42%	1.00E+00	7.4E-11	7 03%	0.5
3	1 12E-01	ε	0.19%	2.10E-02	ε	0 23%	1.00E+00	1E-12	0 14%	0.026
4	1.08E-01	6E-12	6.77%	6.82E-02	3E-12	27 85%	1.00E+00	5.5E-11	5 23%	0.974
5	0	0	0.00%	0	0	0.00%	0	0	0.00%	0.717
6	6.74E-02	ε	0 02%	5.08E-03	ε	0 01%	1.00E+00	ε	0.02%	0.283
7	0	0	0 00%	0	0	0 00%	0	0	0.00%	0.538
8	0	0	0.00%	8.94E-05	ε	0.03%	6.24E-03	ε	0.03%	0.462
Note 1	1.03E-01	ε	0.00%	8.76E-03	ε	0.00%	1.00E+00	ε	0 00%	1
9	1.08E-01	6.6E-11	75.28%	1 05E-02	6E-12	47.66%	1 00E+00	6 10E-10	58 16%	1
10	0	0	0.00%	0	0	0.00%	0	0	0.00%	0.5
Note 2	1 47E-02	ε	0.00%	1 20E-03	ε	0.00%	4.35E-02	ε	0.00%	0.5
11	0	0	0 00%	1.17E-04	ε	0 00%	2.89E-02	ε	0.00%	0.05
12	2.93E-02	ε	0.19%	1.81E-03	ε	0 08%	9.22E-01	5E-12	0.49%	0.95
13	0	0	0.00%	0	0	0.00%	0	0	0.00%	1
14	0	0	0.00%	0	0	0.00%	0	0	0.00%	1
15	0	0	0.00%	1 07E-04	1E-12	8.95%	2.04E-02	2 29E-10	21 87%	1
Total	--	8.8E-11	100.00%	--	1.3E-11	100.00%	--	1 05E-09	100.00%	--

Notes to Table 10.5-6a

Note 1 The DCH case is added for the sensitivity study.

Note 2 The T-AT_nIN_nCHR_FR50_R1 case is added for the sensitivity study.

ε Less than 1E-12

Table 10.5-6b

Sensitivity Case

Elevated Release With Sensitivity Met Data (72 hrs)

Source Term (72 Hrs)	Individual Risk (0-1 mile)	Weighted Individual Risk (per year)	Weighted Individual Risk Contribution (%)	Societal Risk (0-10 miles)	Weighted Societal Risk (per year)	Weighted Societal Risk Contribution (%)	Probability of Dose > .2 Sv (0-0.5 mile)	Weighted Probability of Exceedance (per year)	Weighted Dose Contribution (%)	Release Fraction
1	1.10E-01	8E-12	10.30%	1.74E-02	1E-12	11.12%	1.00E+00	7.4E-11	5.77%	0.5
2	7.58E-02	6E-12	7.10%	8.75E-03	ε	5.59%	1.00E+00	7.4E-11	5.77%	0.5
3	9.84E-02	ε	0.18%	1.44E-02	ε	0.18%	1.00E+00	1E-12	0.11%	0.026
4	8.78E-02	5E-12	6.12%	4.77E-02	3E-12	22.70%	1.00E+00	5.5E-11	4.30%	0.974
5	6.65E-02	ε	0.05%	5.36E-03	ε	0.03%	1.00E+00	ε	0.05%	0.717
6	6.56E-02	ε	0.02%	6.65E-03	ε	0.01%	1.00E+00	ε	0.02%	0.283
7	2.52E-02	1E-12	1.71%	1.56E-03	ε	0.72%	1.15E-01	6E-12	0.48%	0.538
8	1.62E-02	ε	0.94%	1.78E-03	ε	0.71%	1.00E+00	4.6E-11	3.59%	0.462
Note 1	9.34E-02	ε	0.00%	9.13E-03	ε	0.00%	1.00E+00	ε	0.00%	1
9	9.43E-02	5.8E-11	73.04%	9.51E-03	6E-12	50.31%	1.00E+00	6.10E-10	47.76%	1
10	7.10E-02	ε	0.00%	4.98E-03	ε	0.00%	9.36E-01	ε	0.00%	0.5
Note 2	7.56E-02	ε	0.00%	5.62E-03	ε	0.00%	9.88E-01	ε	0.00%	0.5
11	4.67E-02	ε	0.02%	3.22E-03	ε	0.01%	1.00E+00	ε	0.02%	0.05
12	7.37E-02	ε	0.52%	6.33E-03	ε	0.31%	9.82E-01	5E-12	0.43%	0.95
13	2.02E-04	ε	0.00%	7.63E-04	ε	0.00%	1.00E+00	ε	0.00%	1
14	3.98E-05	ε	0.00%	3.94E-04	ε	0.00%	6.49E-01	ε	0.00%	1
15	0	0	0.00%	8.51E-05	ε	8.30%	3.60E-02	4.05E-10	31.69%	1
Total	--	7.9E-11	100.00%	--	1.2E-11	100.00%	--	1.28E-09	100.00%	--

Notes to Table 10.5-6b

Note 1 The DCH case is added for the sensitivity study.

Note 2 The T-AT_nIN_nCHR_FR50_R1 case is added for the sensitivity study

ε Less than 1E-12

Table 10.5-7a[yg26]

Sensitivity Case

Elevated Release Sensitivity Met Data & Plume (24 hrs)

Source Term (24 Hrs)	Individual Risk (0-1 mile)	Weighted Individual Risk (per year)	Weighted Individual Risk Contribution (%)	Societal Risk (0-10 miles)	Weighted Societal Risk (per year)	Weighted Societal Risk Contribution (%)	Probability of Dose > .2 Sv (0-0.5 mile)	Weighted of Exceedance (per year)	Weighted Dose Contribution (%)	Release Fraction
1	1.15E-01	8E-12	10.72%	1.56E-02	1E-12	8.22%	1.00E+00	7.4E-11	7.04%	0.5
2	8.31E-02	6E-12	7.74%	1.23E-02	ε	6.48%	1.00E+00	7.4E-11	7.04%	0.5
3	1.04E-01	ε	0.19%	2.17E-02	ε	0.23%	1.00E+00	1E-12	0.14%	0.026
4	9.82E-02	5E-12	6.81%	7.15E-02	4E-12	28.05%	1.00E+00	5.5E-11	5.24%	0.974
5	0	0	0.00%	0	0	0.00%	0	0	0.00%	0.717
6	5.42E-02	ε	0.02%	5.34E-03	ε	0.01%	1.00E+00	ε	0.02%	0.283
7	0	0	0.00%	0	0	0.00%	0	0	0.00%	0.538
8	0	0	0.00%	8.71E-05	ε	0.03%	6.24E-03	ε	0.03%	0.462
Note 1	8.99E-02	ε	0.00%	9.22E-03	ε	0.00%	1.00E+00	ε	0.00%	1
9	9.65E-02	5.9E-11	74.41%	1.11E-02	6E-12	48.40%	1.00E+00	6.10E-10	58.24%	1
10	0	0	0.00%	0	0	0.00%	0	0	0.00%	0.5
Note 2	3.97E-03	ε	0.00%	1.12E-03	ε	0.00%	1.08E-02	ε	0.00%	0.5
11	0	0	0.00%	1.16E-04	ε	0.00%	4.24E-02	ε	0.00%	0.05
12	1.58E-02	ε	0.11%	1.79E-03	ε	0.07%	6.71E-01	4E-12	0.36%	0.95
13	0	0	0.00%	0	0	0.00%	0	0	0.00%	1
14	0	0	0.00%	0	0	0.00%	0	0	0.00%	1
15	0	0	0.00%	1.06E-04	1E-12	8.52%	2.04E-02	2.29E-10	21.90%	1
Total	--	7.9E-11	100.00%	--	1.4E-11	100.00%	--	1.05E-09	100.00%	--

Notes to Table 10.5-7a

Note 1 The DCH case is added for the sensitivity study

Note 2 The T-AT_nIN_nCHR_FR50_R1 case is added for the sensitivity study.

ε Less than 1E-12

Table 10.5-7b

Sensitivity Case

Elevated Release Sensitivity Met Data & Plume (72 hrs)

Source Term (72 Hrs)	Individual Risk (0-1 mile)	Weighted Individual Risk (per year)	Weighted Individual Risk Contribution (%)	Societal Risk (0-10 miles)	Weighted Societal Risk (per year)	Weighted Societal Risk Contribution (%)	Probability of Dose > .2 Sv (0-0.5 mile)	Weighted Probability of Exceedance (per year)	Weighted Dose Contribution (%)	Release Fraction
1	1.15E-01	8E-12	9.77%	1.49E-02	1E-12	7.04%	1.00E+00	7.4E-11	6.68%	0.5
2	8.42E-02	6E-12	7.15%	1.25E-02	ε	5.91%	1.00E+00	7.4E-11	6.68%	0.5
3	1.09E-01	ε	0.18%	1.86E-02	ε	0.17%	1.00E+00	1E-12	0.13%	0.026
4	9.89E-02	5E-12	6.25%	7.05E-02	4E-12	24.80%	1.00E+00	5.5E-11	4.98%	0.974
5	7.11E-02	ε	0.05%	7.50E-03	ε	0.03%	1.00E+00	ε	0.06%	0.717
6	7.37E-02	ε	0.02%	9.52E-03	ε	0.02%	1.00E+00	ε	0.02%	0.283
7	1.85E-02	ε	1.14%	1.94E-03	ε	0.66%	1.38E-01	7E-12	0.67%	0.538
8	6.87E-03	ε	0.36%	2.05E-03	ε	0.60%	1.00E+00	4.6E-11	4.15%	0.462
9	1.06E-01	ε	0.00%	1.30E-02	ε	0.00%	1.00E+00	ε	0.00%	1
10	1.06E-01	6.5E-11	74.50%	1.35E-02	8E-12	52.79%	1.00E+00	6.10E-10	55.30%	1
11	7.77E-02	ε	0.00%	7.18E-03	ε	0.00%	9.95E-01	ε	0.00%	0.5
12	8.45E-02	ε	0.00%	8.05E-03	ε	0.00%	1.00E+00	ε	0.00%	0.5
13	4.57E-02	ε	0.02%	4.38E-03	ε	0.01%	1.00E+00	ε	0.03%	0.05
14	8.44E-02	ε	0.54%	9.09E-03	ε	0.33%	9.98E-01	6E-12	0.51%	0.95
15	5.91E-06	ε	0.00%	8.29E-04	ε	0.00%	1.00E+00	ε	0.00%	1
16	8.58E-07	ε	0.00%	4.07E-04	ε	0.00%	4.79E-01	ε	0.00%	1
17	0	0	0.00%	1.06E-04	1E-12	7.64%	2.04E-02	2.29E-10	20.79%	1
Total	--	8.7E-11	100.00%	--	1.6E-11	100.00%	--	1.10E-09	100.00%	--

Notes to Table 10.5-7b

Note 1 The DCH case is added for the sensitivity study.

Note 2 The T-AT_nIN_nCHR_FR50_R1 case is added for the sensitivity study

ε Less than 1E-12

10.6 REFERENCES

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10-4 Aldrich, D.C., et al, Technical Guidance for Siting Criteria Development NUREG/CR-2239, December 1982.

10-5 Criteria for preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654.

10-6 Murley, T.E., Advanced Boiling Water Reactor Licensing Review Bases, Project No. 671, August 7, 1987.

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16 SHUTDOWN RISK

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16 SHUTDOWN RISK

16.1 INTRODUCTION

A detailed PRA is performed to determine the core damage frequency during shutdown. Loss of the Reactor Water Cleanup/Shutdown Cooling System, Loss of Reactor Component Cooling Water System, Loss of Plant Service Water System, and Loss of Preferred Power are all investigated. Additionally, the Core Damage Frequency (CDF) due to drain down of the RPV or Loss of Coolant Accidents (LOCAs) during shutdown is evaluated. Fault trees and event trees are used to determine the shutdown CDF for each event analyzed.

The evaluation encompasses plant operation in cold shutdown and refueling modes (Modes 5 & 6). This evaluation addresses conditions for which there is fuel in the RPV. It includes all aspects of the NSSS, the containment, and all systems that support operation of the NSSS and containment.

The scope of the Shutdown PRA is that of a Level 1 PRA. The different accident sequences are classified according to whether the core is damaged or not. All shutdown core damage sequences are assumed to lead directly to a release of radionuclides to the environment (containment is assumed to be open at the time of the initiating event).

The critical safety functions essential to the shutdown model are Decay Heat Removal and Inventory Control. Containment is open for much of the analysis, and containment integrity is not relevant to any modeled functions. Reactivity Control and Spent Fuel Pool Cooling are assumed to have no significant impact on the shutdown model. Power availability is modeled for its impact on decay heat removal. Loss of power is evaluated as an initiating event, and power dependencies for systems are included in the model.

The following subsections discuss the shutdown PRA modeling methodology, data sources, modeling assumptions, and the results of the data analyses for inclusion in the shutdown PRA model. Shutdown PRA analyses during external events are covered in individual external events Sections 12 (Fire), 13 (Flood) and 14 (High Wind).

16.2 PLANT CONFIGURATION IN SHUTDOWN

The differences between the shutdown PRA and the power operation PRA are due to the following:

- Plant operating mode,
- Time after shutdown,
- RPV and containment status,
- Water levels and temperatures,
- Fuel location, and
- Availability of required systems.

This analysis covers the ESBWR risk associated with a refueling outage. The systems modeled are evaluated based on anticipated activities associated with refueling operations.

To develop a suitable shutdown model, multiple bounding plant configurations are defined with similar characteristics in relation to the residual heat, the availability of systems, and the RPV water levels.

The outage plant operating mode is used to define the initial plant condition for individual accident sequence quantification.

Once the outage plant configurations have been defined, the duration of each one is estimated to determine its contribution to the overall calculation of annual core damage frequency. The duration is expressed in hours per refueling outage.

16.2.1 Definition of Plant Shutdown Configurations

The list below shows the Technical Specification definitions of the plant modes. The list is from DCD Chapter 16, Table 1.1-1.

Mode	Title	Reactor Mode Switch Position	Average RCS Temp °C (°F)
1	Power Operation	Run	NA
2	Startup	Refuel or Startup	NA
3	Hot Shutdown	Shutdown	> 215.6 (420)
4	Stable Shutdown	Shutdown	≤ 215.6 (420) and > 93.3 (200)
5	Cold Shutdown	Shutdown	≤ 93.3 (200)
6	Refueling	Shutdown or Refuel	NA

When shutting down the reactor, the transition from Mode 5 to Mode 6 begins when one or more reactor vessel head closure bolts is less than fully tensioned.

The shutdown PRA considers the following outage plant configurations as representative of the possible plant configurations during shutdown.

- Mode 4 (Stable Shutdown) – bounded by full power PRA

- Mode 5 (cold shutdown)
- Mode 5 Open (cold shutdown with containment open)
- Mode 6-Unflooded (refueling)
- Mode 6-Flooded (refueling)

Figure 16.2-1 displays the duration of the different outage plant configurations considered in the ESBWR Shutdown PRA. The following paragraphs describe each of these configurations, detailing the vessel pressure and temperature conditions, as well as the assumed duration and status of RWCU/SDCS and other decay heat removal systems.

16.2.1.1 Mode 4 –Stable Shutdown

This is the cooldown phase to bring the plant to cold shutdown. The reactor mode switch is in the shutdown position. It begins after control rod insertion is completed. Operation of the reactor mode switch from one position to another bypasses Reactor Protection System trips and channels and automatically alters Neutron Monitoring System trip setpoints in accordance with the reactor conditions implied by the given position of the mode switch (Reference DCD Chapter 7).

Decay and sensible heat are removed through the Main Condenser and/or Isolation Condenser. Approximately one-half hour after control rod insertion, both RWCU/SDCS trains are operating, with the regenerative heat exchangers bypassed and pumps running at reduced speed to avoid exceeding the RCCWS design cold leg temperature. The control rod drive system is in service to provide makeup water for the reactor coolant contraction.

The duration of this mode is assumed to be 8 hours.

Containment is de-inerted but integrity is maintained during this mode.

The initial RPV conditions (pressure and temperature) for Mode 4 are the same as power operating values. A review of the Technical Specifications show that almost all credited systems in the PRA have the same Tech Spec requirements for Modes 1 through 4. The CDF contribution of this mode (as well as Modes 1, 2 & 3) is bounded by the full power PRA.

16.2.1.2 Mode 5 – Cold Shutdown

The Tech Spec defined Mode 5 begins when the Reactor Coolant Temperature drops too or below 93.3 °C (200 °F) while the plant is cooling/shutting down. The ESBWR shutdown PRA treats Mode 5 slightly differently. The defined conditions are the same, but the mode itself is subdivided into two sections, with and without an open containment. 'Mode 5' for the shutdown PRA is the portion of the Tech Spec defined mode with the upper drywell head still in place.

This mode occurs twice for each shutdown. It occurs once during cool down, and again after refueling once the reactor head and upper drywell head are replaced before startup.

The total duration of the Tech Spec defined Mode 5 is 240 hours per shutdown. Of that, there are 88 hours before refueling and 152 hours between refueling and startup (see Figure 16.2-1). For the shutdown PRA analysis, 192 hours is the assumed duration of Mode 5. The remaining 48 hours are evaluated as Mode 5 Open, which the subset of the Tech Spec defined mode.

For Mode 5, the reactor mode switch is in the shutdown position. Prior to entering Mode 5 from Mode 4, the heat removal requirements are transferred to the RWCU/SDCS. The Main Condenser and circulating water pumps are removed from service and the use of the isolation condensers is terminated. For the entire duration of Mode 5, all decay heat removal is through the RWCU/SDC system.

Both RWCU/SDCS trains run in parallel, with regenerative heat exchangers bypassed and the pump speed gradually increasing up to the maximum flow rate.

Containment is opened at some time during this mode. The Tech Spec defined mode ends once removal of the reactor head bolts begins. For the shutdown PRA analysis, this mode ends when the upper drywell bolts are removed (which opens containment to the reactor building).

RPV conditions in this mode are assumed to be a pressure of 0.75 MPa (109 psia) and a temperature of 93.3 °C (200 °F). Both values are the assumed high values for temperature and pressure at the transition from Mode 4 to Mode 5.

16.2.1.3 Mode 5 Open – Cold Shutdown

This mode is not a Tech spec defined Mode, and actually includes a period of time from two separate Tech Spec defined Modes. Mode 5 Open is essentially the same as Mode 5 with the exception being that there is no intact containment. The reactor vessel head is still on, but the containment is open.

This analysis assumes that the duration of this mode is approximately 48 hours per refueling outage. That allows for 24 hours to remove the drywell head and the reactor head, and 24 hours to re-attach both after refueling.

Part of the Mode 5 Open period is actually part of the Tech Spec defined Mode 6. According to the Tech Spec mode definitions, Mode 6 begins when one or more reactor vessel head closure bolts is less than fully tensioned. Mode 5 Open sequences consider pressure relief in the model. Mode 6 sequences do not since the RPV head is removed for the majority of the mode. Due to the Tech Spec definition, there is a small period of time that is technically Mode 6, but where the vessel head may still provide a pressure seal. The period of Mode 6 with the vessel head still on is included in the Mode 5 Open event trees and analysis.

16.2.1.4 Mode 6 – Refueling (Unflooded)

In this configuration the reactor head is assumed fully removed, and the reactor well is not flooded. The reactor vessel is open to the reactor building.

As soon as the reactor coolant temperature reaches 49 °C (120 °F), reactor head removal operations may start. Prerequisites required to remove reactor head, such as reactor well drain or drywell head removal occur during the cooldown phase.

Decay heat removal is provided by the RWCU/SDCS. At the start of this mode, both trains are expected to be running. Later, only one is required to keep the reactor coolant temperature within limits.

Though the description of this mode uses the word 'unflooded,' the reactor water level is maintained well above the normal level for Modes 1-5. Water level in this mode is maintained near the flange connection for the vessel head to minimize dose in the reactor building. It is only unflooded relative to the refueling water level, which includes flooding up of the reactor well.

The duration of this mode is assumed to be approximately 59 hours, including the period before refueling and the period after refueling.

16.2.1.5 Mode 6 – Refueling (Flooded)

The plant enters this configuration after the reactor well flooding is completed.

Decay heat removal is provided by RWCU/SDCS, with only one train running much of the time in this mode. The FAPCS, operating in the reactor well cooling mode, can be used also to cool the reactor. If required, FAPCS can be operated in Reactor Well Cooling mode. In this mode, water from the reactor well is directed to the FAPCS heat exchanger to ensure adequate cooling of the upper layer of the reactor well water. It is expected that FAPCS operates in this mode 8 hours in every refueling outage.

The duration of Mode 6- Flooded is assumed to be approximately 240 hours (10 days).

In this configuration, the reactor head is removed and the reactor well is flooded.

The RPV is at atmospheric pressure and the reactor coolant temperature is maintained between 54 °C (150 °F) and 51 °C (124 °F). The reactor vessel is assumed to be open to the reactor building.

16.2.2 Mission Time

For the quantification of core damage frequency, the mission time is assumed to be 24 hours. However, the availability of inventories of water and power sufficient to ensure core cooling for longer time periods is also considered.

16.3 INITIATING EVENTS

The purpose of this subsection is to determine the initiating events that challenge the critical safety functions (e.g., heat removal, inventory control) during shutdown operations. A shutdown initiating event is defined as any event that provokes a disturbance in the stable state of the plant and that requires some kind of action to prevent damage to the core.

Subsection 16.3.1 discusses the shutdown critical safety functions and the shutdown initiating events that challenge these critical safety functions. Subsection 16.3.2 presents the analysis of the initiating events considered in the Shutdown PRA. Frequencies for initiating events are estimated in Subsection 16.3.3 and the recovery actions credited are discussed in Subsection 16.3.4.

16.3.1 Shutdown Critical Safety Functions

The primary critical safety functions accounted for in the Level 1 internal event shutdown evaluation are the following:

- Decay Heat Removal (DHR)
- Reactor Coolant System Inventory Control

No explicit treatment is necessary for the remaining critical safety functions for the following reasons:

- Spent fuel cooling: This function will be maintained during shutdown modes just as it will during full power modes. It is assumed to have no significant impact to the shutdown model
- Reactivity control: All control rods are fully inserted for the duration of the modeled modes. ATWS is not an issue. Reactivity control is assumed to have no significant impact on the shutdown PRA model.
- Power availability: Modeled as it is in the full power model. 'Loss of Preferred Power' is a shutdown initiating event due to it leading to a loss of decay heat removal.
- Containment: Containment is open for much of the shutdown model, and it not credited in the model for the time it is maintained. All core damage sequences modeled in the shutdown PRA are assumed to lead to a direct containment bypass.

16.3.1.1 Decay Heat Removal

The decay heat removal function during all shutdown modes of operation is provided by the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDCS) operating in shutdown cooling mode. In Mode 6 with the reactor well flooded, the Fuel and Auxiliary Pools Cooling System (FAPCS) may be used as an alternative.

At the beginning of every shutdown period, both RWCU/SDCS trains will be running, with pumps varying their speed in order to meet the cooldown rate objectives. Once in Mode 6, before completing reactor cavity flooding, only one train is required. Though two trains are generally running all of Mode 5, only one train is required to prevent reactor coolant boiling.

Two trains are required to lower coolant temperature from 200 °F to the desired refueling temperature of 120 °F.

If the reactor well is flooded (Mode 6-Flooded), the risk associated to loss of decay heat removal has been judged to be negligible because of the following:

- In addition to RWCU/SDCS, FAPCS can be aligned to cool the reactor well water, constituting a valid alternative for RWCU/SDCS, thus reducing the probability of losing the decay heat removal function.
- The large amount of water stored above the core assures core cooling during a long period of time. This time would be significantly longer than 24 hours. This long period could be used to establish an adequate path from an external water source to the reactor well. CRD pumps, FAPCS pumps, condensate pumps, or firewater pumps could provide this makeup function. The long period of time available makes it practically certain that sufficient inventory can be supplied.

Therefore, the loss of decay heat removal is not analyzed in detail for the case when the reactor well is flooded (Mode 6-Flooded).

For the other shutdown modes (Modes 5 and 6 with the reactor well unflooded), it is assumed that one RWCU/SDCS train is sufficient to remove decay heat to prevent reactor coolant boiling.

It is assumed that both RWCU/SDCS trains are running, because the time periods in which only one is running occurs when the reactor well is flooded. Consequently, failure of one of the trains is not considered an initiating event.

16.3.1.2 Reactor Coolant System Inventory Control

This critical safety function is defined as maintenance of the RCS inventory at a level sufficient to sustain decay heat removal.

LOCA and RPV draindown events can potentially challenge this critical safety function. They can occur as a result of:

- Random pipe breaks within the RCS (including breaks related to maintenance or refueling operations).
- Misalignment or leaks of systems connected to the RPV.
- Leakage during FMCRD replacement.

16.3.1.2.1 Pipe breaks

Three different cases are analyzed, depending on whether the reactor vessel head is installed or not, and depending on whether or not containment is intact or not.

The frequency of these events is expected to be lower than at full power, due to the reduced vessel pressure and temperature. For example, the Grand Gulf Shutdown Study (Reference 16-1) reports that the large LOCA frequency for shutdown events is a factor of ten lower than the frequency for the full power case. LOCA frequencies for the shutdown analysis are estimated by:

- Using the associated frequencies from the Level 1 model,
- Adjusting the value based on the duration of the Mode (time in the mode versus the yearly frequency applied to the Level 1 value), and
- Reducing the value by a factor of ten.

The difference in conditions between at power and shutdown is the primary reason for the frequency reduction. The order of magnitude reduction in LOCA frequency method is borrowed from NUREG/CR-6143, Vol. 2, Part 1 A.

Additional basis for the reduction includes:

- ISI and other pipe failure analyses show leak versus break ratio is likely 100 - 1 or greater for small pipe. Leaks in lines (Instrument & drain lines) would not be LOCA (less than 50 gpm) and would not be initiating event.
- In IS-LOCA analysis, without RPS pressure, IS-LOCA is not credible. During shutdown pressure is significantly less than during power operation.
- Shutdown LOCA frequencies at European plants are calculated using reduction factors to account for smaller pressure and temperature. (Reference 16-8)

16.3.1.2.1.1 Pipe Breaks in Mode 6

With the vessel head removed, as long as the RPV level is above Level 3 (L3), it is assumed that RWCU/SDCS provides adequate core cooling. Natural circulation of coolant inside the vessel is not challenged because L3 is above the bottom of the steam separators. As such, any break above L3 during Mode 6 does not constitute a shutdown initiating event, as RWCU/SDCS will continue to ensure normal core cooling and the core will remain covered.

However, if RPV level drops to L3, RWCU/SDCS pumps receive a runback signal, slowing down to cleanup mode flow rate. In addition, once water level in the vessel falls below the bottom of the steam separators, natural circulation is not assured and the core cooling function of RWCU/SDCS may not be adequate. As such, breaks below L3 are included in the analysis as shutdown initiating events.

Breaks below L3 are divided into the following two categories: Breaks outside containment and inside containment.

<u>Breaks Outside Containment</u>

Breaks outside containment can originate only in RWCU/SDCS piping, because this is the only system that removes reactor coolant from the containment in Mode 6. The rest of the RPV vessel piping is isolated.

The RWCU/SDCS containment penetrations have redundant and automatic power-operated containment isolation valves that close on signals from the leak detection and isolation system and the reactor protection system. An additional, diverse non-safety isolation of the RWCU/SDCS system provides protection in the event of a break outside containment.

The RWCU/SDCS return to the feedwater lines are each provided with redundant check valves in series located in the Main Steam Tunnel. A single-power operated isolation valve in each line is located upstream of the check valves and inside the Reactor Building. The FAPCS and CRDS connections are downstream of the two check valves. A postulated break in the RWCU/SDCS piping system inside the Reactor Building, which would otherwise allow reactor coolant to flow backwards through main feedwater lines and to spill into the Reactor Building, will be isolated by either redundant RWCU/SDCS check valves or feedwater check valves even if a single failure of one check valve is assumed.

Therefore, the shutdown PRA considers RWCU/SDCS breaks outside containment to be negligible risk contributors and does not analyze them further. This is consistent with the at-power PRA which shows the CDF contribution from RWCU/SDCS breaks outside containment to be negligible.

Breaks Inside Containment

For breaks inside containment, coolant flows through the break to the lower drywell. Decay heat removal is achieved in this case by allowing reactor coolant boiling and then venting the steam to the atmosphere (i.e., the drywell head is removed in Mode 6). To maintain adequate water level in the vessel, a water supply to the vessel is required.

If a break is located below TAF, to reach a safe core cooling condition, it is necessary to flood the drywell and the vessel up to a level above the TAF.

The lower drywell is equipped with a personnel hatch and with an equipment hatch to allow access to the containment for personnel and equipment. These hatches are closed during normal operation, but they may be open during refueling. A manual recovery action to close these two hatches is required for successful drywell flooding (see recovery analysis below).

Two different cases are considered for breaks inside containment below TAF during Mode 6:

- Reactor well flooded (Mode 6-Flooded):

  If the reactor well is flooded, the water inventory stored above the core is assumed to be sufficient to flood the drywell and the vessel well above the TAF if the two lower drywell access hatches are closed at the time of the event or they are manually closed before the water level in the drywell reaches the elevation of the hatches.

- Reactor well unflooded (Mode 6-Unflooded):

  If the well is not flooded, the water inventory stored above the core is assumed to be insufficient to cover the core, and additional coolant supply is required

As discussed previously, only pipe breaks below RPV Level 3 (L3) are considered shutdown Mode 6 initiating events. Therefore, breaks in main steam lines, DPVs, and instrument lines above L3 are not considered as shutdown initiating events.

Based on the discussions above, and the line breaks identified in Table 2.3-1, the following line break categories are identified as potential shutdown LOCA initiators:

- Feedwater LOCA – As a simplifying and conservative assumption, the entire feedwater LOCA frequency is applied to feedwater Line A in the shutdown analysis. In the internal events analysis, the frequency is evenly split between the two lines (A & B). In the shutdown analysis feedwater line A is the more limiting break. A feedwater Line A break would disable one half of RWCU/SDC, and all FAPCS and FPS injection. A feedwater Line B break would disable the other half of RWCU/SDC and all of CRD injection.

- GDCS injection line LOCA - This event degrades the passive inventory control system.
- LOCA other than feedwater and GDCS – This is for line breaks above TAF other than GDCS injection or feedwater lines (see Table 16.3-3b and Table 2.3-1 for specific lines).
- Instrument Line LOCA below TAF – All LOCAs below TAF during shutdown require closure of lower drywell hatch. The hatch can be opened during shutdown. If a break occurs in the lower drywell and the hatch is not closed, core damage is assumed to occur (once the water level reaches the bottom of the hatch, it is assumed that the door can not be closed and the leak not isolated).
- RWCU/SDC drain line LOCA below TAF. The RWCU/SDC system has drain lines that are below the reactor core. A break in these lines is limiting because it has the potential to drain the vessel to below the fuel. Closure of the lower drywell hatch is required to mitigate this event.

16.3.1.2.1.2 Mode 5 and Mode 5 Open

The same LOCA scenarios modeled for Mode 6 are modeled for Mode 5 with and without an intact containment. Steam Line LOCAs and DPV line LOCAs are assumed to pose negligible risk. The LOCA frequencies for these are lower than GDCS and feedwater, and the resulting LOCAs don't disable any makeup function.

16.3.1.2.2 RPV Draindown events due to misalignments

The ESBWR design has significantly reduced the number of potential RPV draindown pathways due to postulated system misalignment during shutdown conditions.

In particular, as compared to Residual Heat Removal System in current BWRs, the RWCU/SDCS in the ESBWR does not have the potential for diverting RPV inventory to the suppression pool through the SP suction, return, or spray lines. RWCU/SDCS does not provide any drywell spray function, so the potential RPV draindown through drywell spray does not exist. In addition, the absence of recirculation lines in the ESBWR design further reduces the potential RPV draining paths.

The only operating system that has the potential to drain the RPV during this mode of shutdown is RWCU/SDCS. This system is connected to the RPV during shutdown and it is used to discharge excess reactor coolant to the main condenser or to the radwaste system during startup, shutdown and hot standby conditions.

An evaluation of system pipe drawings showed two potential draindown paths due to misalignment. Both lines have 20mm diameters and are assumed to be too small to be considered an initiating event.

16.3.1.2.3 FMCRD replacement

Draining the RPV during FMCRD maintenance has been evaluated, but is not considered a shutdown PRA initiating event. With the tools used for the actions, and the controls in place to monitor the evolution, the chance of a significant RPV leak due to the activity is assumed to be negligible.

The procedure for removal of the FMCRD for maintenance or replacement is similar to previous BWR product lines. The control rod is first withdrawn to the full-out position. During removal of the lower housing (spool piece) following removal of the position indicator probes and motor unit, the control rod backseats onto the control rod guide tube. This metal-to-metal contact provides the seal that prevents draining of reactor water when the FMCRD is subsequently lowered out of the CRD housing. The control rod normally remains in this backseated condition at all times with the FMCRD out; however, in the unlikely event it also has to be removed, a temporary blind flange is first installed on the end of the CRD housing to prevent draining of reactor water.

If the operator inadvertently removes the control rod after FMCRD is out without first installing the temporary blind flange, or conversely, inadvertently removes the FMCRD after first removing the control rod, an un-isolable opening in the bottom of the reactor is created, resulting in drainage of reactor water. The possibility of inadvertent reactor drain-down by this means is considered remote for the following reasons:

- Procedural controls similar to those of current BWRs provide the primary means for prevention. Current BWR operating experience demonstrates this to be an acceptable approach. There has been no instance of an inadvertent drain-down of reactor water due to simultaneous CRD and control rod removal.
- During drive removal operations, personnel are required to monitor under the RPV for water leakage out of the CRD housing. Abnormal or excessive leakage occurring after only a partial lowering of the FMCRD within its housing indicates the absence of the full metal-to-metal seal between the control rod and control rod guide tube required for full drive removal. In this event, the FMCRD can then be raised back into its installed position to stop the leakage and allow corrective action.

The COL applicant shall develop maintenance procedures with provisions to prohibit coincident removal of the control rod and CRD of the same assembly. In addition, the COL applicant shall develop contingency procedures to provide core and spent fuel cooling capability and mitigative actions during CRD replacement with fuel in the vessel.

The FMCRD design also allows for separate removal of the motor unit, position indicator probe (PIP), separation indicator probe (SIP) and spool piece for maintenance during plant outages without disturbing the upper assembly of the drive. While these FMCRD components are removed for servicing, the associated control rod is maintained in the fully inserted position by one of two mechanical locking devices that prevent rotation of the ball screw and drive shaft.

The first anti-rotation device is engaged when the motor unit consisting of the induction motor, reduction gear, brake and position signal detector is removed. It is a spring-actuated locking cam located on the bottom of the spool piece. When the motor unit is lowered away from the spool piece, the locking cam is released from its normally retracted position and engaged by spring force with gear teeth on the bottom of the magnetic coupling outer rotor, thereby locking the shaft in place.

With the motor unit removed, the locking cam can be visually checked from below the drive to verify that it is properly engaged. When the vessel head is removed, another means of verification of proper locking is for the operator to view the top of the control rod from over the reactor vessel. If the top of the control rod is visible at its normal full in position, it provides both direct indication that the control rod remains fully inserted and additional assurance that the ball screw is restrained from reverse rotation. The drive shaft remains locked in this manner until the motor unit is reattached to the spool piece. During motor installation, a release pin on the motor unit pushes up a plunger linked to the locking cam as the motor unit is raised into contact with the spool piece. The release pin forces the locking cam away from the teeth on the bottom of the magnetic coupling outer rotor and into the normally retracted, unlocked position.

The second anti-rotation device is engaged when the spool piece is removed from the FMCRD. As described in DCD Subsection 4.6.2.1.3, this device is a spline arrangement between the ball screw lower coupling and the middle flange backseat. When removing and lowering the spool piece, the weight of the ball screw, hollow piston and control rod provides a vertical force in the downward direction that brings the two splines together. This locks the ball screw into the backseat and prevents reverse rotation. As with the first anti-rotation device, proper engagement of this device can be visually checked from below the drive. If the splines do not completely lock together, there is indication of this because the ball screw does not seat against the backseat and there is a small gap for leakage of water. If this should occur, removal of the spool piece can be discontinued and corrective action taken. If there is no leakage, it confirms that the splines are properly locked together. Also as in the case of the first anti-rotation device, visual observation of the top of the control rod from over the reactor vessel provides another means for verifying proper locking of the ball screw. The ball screw remains locked in this position until the spool piece is reattached to the FMCRD. During spool piece installation, the end of the drive shaft fits into a seat on the end of the ball screw. As the ball screw piece is raised off the middle flange backseat, the anti-rotation splines disengage and the weight of the ball screw, hollow piston and control rod is transferred to the spool piece assembly.

16.3.2 Identification of Initiating Events

The identification of the shutdown initiating events for inclusion in the ESBWR shutdown risk assessment is based on:

- Review of past shutdown PRAs,
- Review of the ESBWR full power PRA initiators, and
- Consideration of the ESBWR design and configuration during shutdown.

The potential initiator scenarios are described in Section 16.3.1. The resulting list of initiating event types during shutdown (and as a function of critical safety function) is presented in Table 16.3-1.

16.3.3 Frequency of Initiating Events

Initiating event frequencies are quantified based on a review of BWR operating experience as well as ESBWR specific evaluations.

16.3.3.1 Loss of Both Operating RWCU/SDCS Trains

The main components of the system are the pumps, heat exchangers, demineralizers, valves and piping.

RWCU/SDCS is connected to nonsafety-related standby AC power (diesel generators) allowing it to perform its reactor cooling function when the preferred power source is not available.

In addition to AC power, RWCU/SDCS requires the operation of the Reactor Component Cooling Water System (RCCWS) and the Plant Service Water System (PSWS) in order to remove decay heat.

The unavailability of the RWCU/SDCS system can occur for the following general reasons:

- Failure of both RWCU/SDCS trains.
- Isolation of the RWCU/SDCS, caused by RPV low level (Level 3 causes RWCU/SDCS pump runback and Level 2 causes loss of suction pressure), SLCS initiation, LD&IS signals, high temperature in main steam tunnel or high system flow.
- Loss of Preferred Power (LOPP).
- Loss of RCCWS or PSWS.

The Loss of RWCU/SDCS shutdown initiating event is defined by failure of both trains, either due to RWCU/SDCS component failures or by automatic closure of isolation valves. LOPP and Loss of RCCWS/PSWS are modeled as separate shutdown initiating events.

Automatic closure of RWCU/SDCS isolation valves can be initiated by the following signals:

- High RWCU/SDCS flow,
- Low reactor water level (level 2),
- High temperature in main steam tunnel, and
- Leak Detection and Isolation System signals that indicate a break in the RWCU system.

ESBWR logic design uses four divisions of power backed up by safety-related batteries. Therefore, loss of power to the logic is highly unlikely. Three divisional logic power supply failures are required to initiate the SDC isolation; as such, this RWCU/SDC failure mode is non-significant compared to loss of support systems or mechanical failures.

The initiating frequency for Loss of RWCU/SDCS is calculated from a combination of several common cause failures in the RWCU/SDC logic. The events contributing to the Loss of RWCU/SDC are listed in Table 16.3-1.

16.3.3.2 Loss of Preferred Power

Loss of Preferred Power (LOPP) may happen as a result of severe weather, grid failures or switchyard faults.

The LOPP shutdown initiator frequency is calculated using the loss of offsite power during shutdown data documented in NUREG/CR-5496. (Reference 16-2)

16.3.3.3 Loss of RCCWS/PSWS

This initiating event is the loss of the Reactor Component Cooling Water System (RCCWS) or the loss of the Plant Service Water System (PSWS) supporting the RWCU/SDCS operating in shutdown cooling mode. This initiating event poses a DHR challenge and renders the RWCU/SDCS system unavailable.

The frequency of this initiator is based on the Loss of PSWS initiating event frequency calculated for the at-power PRA.

16.3.3.4 Shutdown LOCA Events

The following LOCA initiators are quantified in the shutdown PRA:

- Mode 5
  - Break in one of the GDCS injection lines
  - LOCA in FW-A
  - LOCA other than FW or GDCS
  - LOCA below TAF in RWCW/SDC drain lines
  - LOCA below TAF in instrument lines
- Mode 5 Open
  - Break in one of the GDCS injection lines
  - LOCA in FW-A
  - LOCA other than FW or GDCS
  - LOCA below TAF in RWCW/SDC drain lines
  - LOCA below TAF in instrument lines
- Mode 6 Unflooded
  - Break in one of the GDCS injection lines
  - LOCA in FW-A
  - LOCA other than FW or GDCS
  - LOCA below TAF in RWCW/SDC drain lines
  - LOCA below TAF in instrument lines
- Mode 6 Flooded
  - LOCA below TAF in RWCW/SDC drain lines
  - LOCA below TAF in instrument lines

The shutdown LOCA frequencies are based on the at-power RCS line break frequencies (refer to Chapter 2, Table 2.3-1). The at-power RCS line break frequencies are reduced by a factor of 10 for the shutdown PRA to reflect the low pressure and temperature during shutdown conditions.

16.3.4 Recovery Actions

This section documents the calculation of time-dependent, post-initiator recovery probabilities used in the ESBWR Shutdown PRA. The recovery events addressed in this section are those that terminate or mitigate the initiating event before a safety function is challenged. Unlike at full-power conditions, during shutdown modes extended time can be available to terminate the initiating event. This justifies the quantification of recovery possibilities.

Recovery events are analyzed for the following initiating events:

- Loss of both operating RWCU/SDCS trains: Operators recover at least one of the two failed trains.
- Loss of RCCWS/PSWS: Operators recover the failed equipment.
- LOCA Below TAF (Mode 6): Operators close the two lower drywell hatches if they are open.

The analysis of these events is performed using the BWR industry data from References 16-3, 16-4, 16-5 and 16-6. Industry data is included here based on operating experience in Modes 5 and 6.

The operating experience events in each category are analyzed to determine the time elapsed before the initiating event was terminated. Because of the limited data for extended durations, an additional assumed event with a recovery time of 20 hours is added to all distributions.

It is then assumed that the time to recovery is a random variable following a lognormal probability density distribution. The operating experience recover y events are listed in Table 16.3-4, and the distribution from the data is shown in figure 16.3-1.

16.3.4.1 Recovery of RWCU/SDCS

The most functionally similar system to RWCU/SDCS in current BWRs is the Residual Heat Removal system.

Events involving the loss of a running RHR pump at BWRs are identified from References 16-3, 16-4 and 16-5, and are shown in Table 16.3-4. However, due to the lack of BWR events specifying the duration, PWR events (see Reference 16-6) are also used in the recovery analysis, excluding those occurring during reduced inventory conditions. Using the methodology described above, the lognormal distribution parameters m and s are determined to be 2.73 and 2.08, respectively. The resulting recovery probability curve as a function of time for the Loss of RWCU/SDCS initiator is provided in Figure 16.3-1.

Based on the most limiting Mode 5 conditions, the time to boil following a loss of decay heat removal (loss of RWCU/SDC) would be approximately 4 hours. This time does not account for the time following boiling for level to drop to L3. This time is based on a bounding estimate of the decay heat at Mode 5 and the specific configuration of the ESBWR vessel and internals. The analysis assumed the following conditions at the moment decay heat removal was lost:

- Temperature of 93.3 °C (200 °F) – Mode 5 maximum temperature
- Atmospheric pressure at 14.7 psia – minimum Mode 5 pressure during or immediately prior to vessel head removal

- Decay heat at 3.89E-7 Watts – decay heat 5.3 hours after shutdown. Mode 5 is not likely to begin until at least 8 hours following shutdown.
- Normal water level – water level will be gradually increased during Mode 5 up to the vessel head flange. This level is likely the minimum water level for all of Mode 5.

The same calculation also estimates the time to boil for Mode 6 Unflooded conditions. This analysis estimates the time to boil for the most limiting conditions of Mode 6 is approximately 5.3 hours.

For Mode 5, the time available for operators to recover RWCU/SDC is 4 hours.

For Mode 6-Unflooded, the available time is 5.7 hours.

These two time estimates are also included on the graph in Figure 16.3-1. A vertical line on the 'time' axis at each value has been added to the plot. Both lines intercept the operating experience distribution for SDC recovery between the 0.8 and 0.9 values on the recovery axis. This establishes that recovery of RWCU/SDC for the ESBWR during shutdown is likely between 80% and 90% successful given the estimated times. Or, that the failure to recover shutdown cooling has a failure rate of 0.2 or less.

The previous revision of the ESBWR internal events shutdown PRA estimated the failure to recover RWCU/SDC probabilities to be 0.229 for Mode 5 and 0.218 for Mode 6. This revision of the shutdown model retains these failure rates for the recovery actions. The numbers are relatively close to what the updated analysis estimates. The new estimates have a slightly lower failure probability, but use of the old values remains conservative.

The non-recovery probabilities for these two cases are shown in Table 16.3-5.

16.3.4.2 Recovery of RCCWS/PSWS

The RCCWS and PSWS are used during shutdown to remove decay heat and other heat loads. The main components in the system are pumps and heat exchangers. It is assumed that the time to recover from a Loss of RCCWS/PSWS shutdown initiator follows the same probability density distribution as the time to recover the RWCU/SDCS.

The allowable time frame for recovery is taken to be the same as that for Loss of RWCU/SDCS.

For Mode 5, the available time is 1.4 hours.

For Mode 6-Unflooded, the available time is 1.57 hours.

The non-recovery probabilities for these two cases are shown in Table 16.3-5.

16.3.4.3 Close Lower Drywell Hatches

An equipment hatch for removal of equipment during maintenance and an air lock for entry of personnel are provided in both the lower and upper drywell. These access openings are sealed under normal plant operation but are opened when the plant is shut down for refueling. Credit is not given for closing the doors after reactor coolant overflows through the hatches. Therefore, the time available for closing both hatches depends on the volume of the lower drywell under the bottom edge of the hatches, on the size of the break, and on the water level in the vessel or reactor well above the break.

This action is required for the LOCA below TAF initiators during shutdown. These LOCAs involve breaks in the RWCU/SDC drain lines and instrument lines.

The flow through the break is assumed constant and equal to $S\sqrt{2gh}$, where S is the break area, h is the height of water above the break, and g is the acceleration due to gravity.

Both break locations, in RWCU/SDCS drain lines and instrument lines are evaluated. The calculation assumes the reactor well is not flooded at the time of the break. Table 16.3-6 summarizes the calculation.

Detection of the event will be immediate if personnel are present in the lower drywell. If this is not the case, it is assumed that an alarm on drywell sump high level is available in the control room.

Closure of the lower drywell hatches is covered in the Availabilities Control Manual (ACM). The ability to close the hatch is covered for shutdown, and immediate action is required if hatch closure is unavailable for any reason. "ACLCO 3.6.2 - The lower drywell personnel air lock and lower drywell equipment hatch shall be AVAILABLE for closure. APPLICABILITY: MODES 5 and 6, during operations with a potential for draining the reactor vessel."

Once the event has been detected, the plant operator must correctly diagnose the situation, make the decision to close the hatches, gain access to elevation – 6400 mm in the reactor building, and manually close the equipment hatch and the personnel air lock. It is assumed that during the outage, personnel will be continuously located in the area of the doors.

Probabilities of 1.0E-1 and 1.0E-2 are assigned for failing to close the DW hatches for breaks in RWCU/SDCS drain lines and instrument line breaks, respectively. A sensitivity analysis on these events will be run and the results section will discuss how various values for these actions change the results.

16.4 EVENT TREES

The shutdown PRA event trees are shown in Figures 16.4-1 through 16.4-26.

The event tree construction takes into account the following aspects:

- Chronological order of system actuation
- Grouping of mitigating systems by safety functions

Descriptions of all event tree headings used are provided below.

The success criteria used in the different events are reported under the description of the event heading below in Section 16.4.1.

The accident sequence end state nomenclature is the same as in the full power PRA:

- OK: The core is successfully cooled and the containment is intact. There is no core damage in these events.
- CD I: The containment is intact when core damage occurs and the RPV is at low pressure.
- CD II: The containment fails while the core is successfully cooled, leading to subsequent core damage.
- CD III: The containment is intact when core damage occurs and the RPV is at high pressure.
- C IV: ATWS
- CD V: The containment is bypassed at the time of core damage.

All core damage sequences in the shutdown model are assumed to be category CD-V. This end state is certain in Mode 5 Open and Mode 6 due to the upper drywell head being removed. For Mode 5, this treatment is conservative.

16.4.1 Event Tree Headings

The following is a list of event tree headings used in the ESBWR Shutdown PRA and a brief description of each. Most of the headings are the same ones used in the full power PRA model. Any changes made to the logic for shutdown are described below in section 16.5.

B32-3LOOPSFAIL-Isolation Condensers (IC)

This is the same ICS heading as is used for the full power model.

Regardless of initial RPV pressure, if decay heat is not removed, the Isolation Condenser System is initiated automatically on high reactor pressure, or later on RPV water level 2.

The IC function is able to prevent RPV Level 1 from being reached if:

- The initial RPV water inventory is above Level 3
- There is little or no leakage from the RPV.

The maximum RCPB leak rate within the Technical Specification during full power operation is assumed to be insufficient to decrease the level to the point where an ADS signal occurs, even if high pressure RPV makeup is not established during the sequence mission time. Therefore, failure of the IC function due to leaks is considered a low probability.

The success criterion of this function is the operation of both operable (2/2) ICs during the sequence mission time. The Tech Specs for Mode 5 only require 2 out of the 4 ICs be available.

MS-TOP18-At Least 1 SRV Open

This is the same heading as is used for the full power model.

If the IC function fails, the RPV pressure will increase up to the SRVs setpoint. The success criterion for this function is the automatic operation of at least 1 SRV. Failure of this function is conservatively assumed to lead to core damage.

The possibility of a stuck open relief valve is not modeled. Because no credit is given for the IC function after the opening of an SRV.

MS-TOP2-At Least 2 SRVs Open

This heading is not in the full power model, it is unique to the shutdown PRA.

If no high pressure injection system is available, it is necessary to depressurize the RPV to allow FAPCS or FPS injection to the RPV.

Success of this function requires the operator to manually open at least 2 SRVs.

The time available to the operator to manually initiate RPV depressurization is defined by the time when RPV level falls below L2 to the time when the ADS system will automatically initiate (i.e., at RPV Level 1).

UD-TOPINJ2-High Pressure Injection Systems

This is the same heading as is used for the full power model.

Water level can be maintained above RPV Level 1 by the CRD system. Under normal conditions, CRD automatically actuates when coolant drops to RPV level 2. For shutdown analysis, it is assumed operator action is required to initiate CRD injection. Successful CRD injection into the reactor vessel requires 1 CRD pump taking suction from the CST.

UD-TOPINJ-High Pressure Injections Systems

This is the same heading as is used for the full power model.

This is CRD system top when both pumps are required for success. In the above heading, only one of two pumps is needed. In cases where this heading appears, both CRD pumps are needed to meet the success criteria.

VL-TOPINJ-FAPCS-Low Pressure Injection System 1

VM-TOPINJ-FPS-Low Pressure Injection System 2

These two headings are the same ones used for the full power model.

After successful RPV depressurization, either FAPCS or FPS can fulfill the core cooling function when configured in the RPV injection mode. Both systems require manual actuation.

The time available to the operator to manually initiate either of these two systems is defined by the time when RPV pressure has been sufficiently reduced to the time when the ADS system will automatically initiate (i.e., at RPV Level 1). Success of either function requires the operator to manually align at least 1 FAPCS or FPS train in RPV injection mode.

FAPCS requires RCCWS/PSWS for heat exchanger cooling. FPS does not require cooling.

VL-TOPINJL-FAPCS-Low Pressure Injection System 1 after ADS

VM-TOPINJL-FPS-Low Pressure Injection System 2 after ADS

These two headings are the same ones used for the full power model.

These event tree nodes model initiation of low-pressure injection following automatic RPV depressurization by the ADS system. As in the pre-ADS nodes, either 1 train of FAPCS or FPS operating in the RPV injection is needed for success. However, the time available to the operator to perform the manual alignment is different in the scenario with automatic ADS.

VI-TOPINJ-Gravity Driven Cooling System

This heading is also used for the full power model.

If all the active low pressure injection systems are unavailable after successful RPV depressurization, the passive GDCS system will automatically inject water into the RPV. The opening of at least 2 GDCS lines and the discharge of at least 1 GDCS pool accomplish short term cooling. One equalizing line must be opened for long term core cooling.

The success criteria for this function are the discharge of at least 2 lines and 1 GDCS pool and the opening of at least 1 equalizing line.

WR-TOPSDC-RWCU/SDC Restart

This heading is not in the full power model, it is unique to the shutdown PRA.

After a LOPP event, the RWCU/SDCS pumps are tripped and the decay heat removal function is temporarily unavailable. Operator action to restart the RCWU/SDCS trains on the diesel generator power is included in the fault tree logic.

The success criterion is that at least one RWCW/SDCS train successfully restarts after a LOPP event and operates during the sequence mission time.

XD-TOPDPV-ADS-At Least 4 DPVs Open Automatically

This is the same heading as is used for the full power model.

The success criterion for this function is that at least 4 DPVs automatically open.

Recovery Actions

The operator has the opportunity to recover the lost functions (RWCU, PSWS) before any safety function is challenged. See Section 16-3 above for the time available and the recovery failure probabilities. The following recovery actions are in the event trees:

R-M6-G31-RWCU/SDC Recovery in Mode 6

R-PSWS-6-PSWS Recovery in Mode 6

R-M5-G31-RWCU/SDC Recovery in Mode 5

R-M5-PSWS-Recovery of RCCWS/PSWS (Mode 5)

<u>Close the lower drywell hatches</u>

<u>DWH-1</u> Close Drywell Hatch – Instrument line LOCA

<u>DWH-2</u> Close Drywell Hatch – RWCU/SDC drain line LOCA

If not closed at the time of the initiating event, the operator must close the lower drywell hatches to prevent flooding of reactor building lower levels. See Subsection 16.3.4.3 for details on time available for this action and the recovery failure probability.

16.4.2 Loss of Decay Heat Removal Event Trees

Initiators leading to a loss of decay heat removal function are grouped into scenarios occurring during Mode 5 Mode 5 Open, or Mode 6 Unflooded. Given the three initiating event types leading to loss of DHR scenarios (Loss of RWCU/SDC, Loss of RCCWS/PSWS and Loss of Preferred Power), nine shutdown loss of DHR event trees are analyzed. The event trees for Mode 5 and Mode 5 Open are practically identical. The credited systems, and sequence order are the same for both modes.

16.4.2.1 Loss of DHR in Mode 5 or Mode 5 Open

Following loss of the decay heat removal function, pressure and temperature in the RPV gradually increase. Due to reduced initial pressure and temperature during shutdown conditions, the operator has the opportunity to recover the lost function before the pressure reaches the SRV setpoint.

If recovery of decay heat removal is not possible, the Isolation Condenser (IC) function will be initiated on RPV high pressure or low RPV water level (Level 2). This function provides short term as well as the long term core cooling.

If ICS fails, RPV pressure increase leads to SRV opening. Steam generated by decay heat is then discharged to the suppression pool, where it is condensed. As the level inside the RPV decreases, high pressure makeup is required to keep the core covered.

Failure of all of the SRVs to open could be postulated to lead to a vessel rupture scenario. Even if mitigation is still possible, core damage is assumed.

Control Rod Drive system (CRD) can fulfill the high-pressure makeup function. CRD is assumed to need operator actuation during shutdown.

If high pressure makeup fails, low pressure makeup is required. The opening of two SRVs enables the injection modes of either FAPCS or the Fire Protection System (FPS).

If low pressure injection systems fail after manual depressurization with 2 SRVs, the ADS will actuate and the short term and long term core cooling functions are performed with 2 of 4 lines from the Gravity Driven Cooling System (GDCS), 2 of 3 GDCS pools and the opening of at least one equalizing line. The equalizing line will permit effective RPV flooding with the suppression pool water, as long as at least 4 DPVs have opened. If for some reason GDCS cannot inject into the depressurized reactor, either FAPCS or FPS injection mode can support the short term and long term core cooling functions.

16.4.2.1.1 Loss of Both RWCU/SDCS Trains (Mode 5 & Mode 5 Open)

For specific details about the event trees for these initiating events, refer to Figure(s) 16.4-2 and 16.4-5. The description given above corresponds to this case. The event tree headings associated with this event tree are listed here:

- R-M5-G31 - Recovery of RWCU/SDCS (Mode 5/Mode 5 Open)
- B32-3LOOPSFAIL - Isolation Condensers
- MS-TOP18 – At Least 1 SRV Open
- UD-TOPINJ2 - CDR – High Pressure Injection Systems
- MS-TOP2 – At Least 2 SRVs Open
- VL-TOPINJ - FAPCS – Low Pressure Injection System 1
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- XD-TOPDPV - ADS - At Least 4 DPVs Open Automatically
- VL-TOPINJ1 - FAPCSL - Low Pressure Injection System1 after ADS
- VM-TOPINJ - FPS - Low Pressure Injection System 2 after ADS
- VI-TOPINJ – Gravity Driven Cooling System

This tree results in 5 core damage sequences. For the specific logic associated with each, see the attached figures (16.4-2 and 16.4-5).

16.4.2.1.2 Loss of Preferred Power (Mode 5 & Mode 5 Open)

The event trees for these initiating events are shown in Figures 16.4-3 and 16.4-6. The event tree heading descriptions above for loss of DHR in Mode 5 are applicable to this case. This event tree is very similar to the Loss of RWCU/SDC event tree, though no credit is given for power recovery. Every system credited has an alternative power supply to ensure availability (FAPCS/CRD/RWCU – diesel generator, FPS has two diesel driven pumps).

- B43-3LOOPSFAIl – Isolation Condensers
- MS-TOP18 – At Least 1 SRV Open
- WR-TOPSDC – RWCU/SDC Restart after LOPP
- UD-TOPINJ2 - CDR – High Pressure Injection Systems
- MS-TOP2 – At Least 2 SRVs Open
- VL-TOPINJ - FAPCS – Low Pressure Injection System 1
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- XD-TOPDPV - ADS - At Least 4 DPVs Open Automatically
- VL-TOPINJ1 - FAPCSL - Low Pressure Injection System1 after ADS
- VM-TOPINJ - FPS - Low Pressure Injection System 2 after ADS

- VI-TOPINJ – Gravity Driven Cooling System

These trees results in 5 core damage sequences each. For the specific logic associated with each, see the attached figures (16.4-3 and 16.4-6).

16.4.2.1.3 Loss of RCCWS/PSWS (Mode 5 & Mode 5 Unflooded)

The event trees for these initiating events are shown in Figures 16.4-1 and 16.4-4. The event tree heading descriptions above apply to this tree as well, except for those systems that are unavailable because they rely on RCCWS/PSWS: CRD: and FAPCS.

- R-M5-PSWS - Recovery of RSWS/RCCWS (Mode 5/Mode 5 Open)
- B32-3LOOPSFAIL - Isolation Condensers
- MS-TOP18 – At Least 1 SRV Open
- MS-TOP2 – At Least 2 SRVs Open
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- XD-TOPDPV - ADS - At Least 4 DPVs Open Automatically
- VM-TOPINJ - FPS - Low Pressure Injection System 2 after ADS
- VI-TOPINJ – Gravity Driven Cooling System

These trees results in 5 core damage sequences each. For the specific logic associated with each, see the attached figures (16.4-1 and 16.4-4).

16.4.2.2 Loss of DHR in Mode 6 (Unflooded)

In this case, the RPV is open (i.e., the RPV head has been removed, Mode 6), the containment is also open, and the reactor well is assumed to be drained, so the water level is at the elevation of the vessel flange.

A considerable amount of cool water is above the core, allowing the operator significant time to recover failed equipment or systems before coolant boil-off reduces the water level in the vessel, causing safety system actuations.

However, if the decay heat function cannot be recovered, RPV coolant temperature rises, eventually reaching boiling conditions. Makeup water to the RPV is then needed to prevent core damage. Four systems are allowed for the makeup function. Three of them require manual initiation by the operator: FAPCS in injection mode and FPS in injection mode, as well as CRD (assumed to need manual action during shutdown). The one, GDCS automatically initiates on RPV Level 1.

16.4.2.2.1 Loss of Both RWCU/SDCS Trains (Mode 6)

The event tree for this initiating event is shown in Figure 16.4-8. All headings for this event tree have been described previously. The headings associated with this initiating event are:

- R-M6-G31 - RWCU/SDC recovery in Mode 6
- UD-TOPINJ2 - CDR – Control Rod Drive Pump

- VL-TOPINJ - FAPCS – Low Pressure Injection System 1
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- VI-TOPINJ – Gravity Driven Cooling System

This trees results in only one core damage sequence. It is the result of all credited systems failing to function. For the specific logic associated the event tree, see the attached figure (16.4-8).

16.4.2.2.2 Loss of Preferred Power (Mode 6)

The event tree for this initiating event is shown in Figure 16.4-9. The descriptions above for loss of DHR in Mode 6 apply to this tree as well. RWCU/SDC is credited in this event tree. Recovery of the system does require a manual action to restart the pump on diesel generator power. The headings associated with this initiating event are:

- WR-TOPSDC – RWCU/SDC Restart after LOPP
- UD-TOPINJ2 - CDR – Control Rod Drive Pump
- VL-TOPINJ - FAPCS – Low Pressure Injection System 1
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- VI-TOPINJ – Gravity Driven Cooling System

This trees results in only one core damage sequence. It is the result of all credited systems failing to function. For the specific logic associated the event tree, see the attached figure (16.4-9).

16.4.2.2.3 Loss of RCCWS/PSWS (Mode 6)

The event tree for this initiating event is shown in Figure 16.4-7. The descriptions above for loss of DHR apply to this tree as well, except that some systems are unavailable because they rely upon RCCWS/PSWS. CRD and FAPCS. FPS and GDCS remain the only makeup systems available. The headings associated with this initiating event are:

- R-PSWS-6 - PSWS Recovery in Mode 6
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- VI-TOPINJ – Gravity Driven Cooling System

This trees results in only one core damage sequence. It is the result of all credited systems failing to function. For the specific logic associated the event tree, see the attached figure (16.4-7).

16.4.3 Loss of Coolant Accidents

16.4.3.1 Shutdown LOCAs (Mode 5, Mode 5 Open, & Mode 6Unflooded)

Due to low pressure and temperature of shutdown conditions, all LOCAs are liquid breaks. All LOCAs in this section are analyzed for three modes (Mode 5, Mode 5 Open, and Mode 6

Unflooded). The evolution of the accident and the systems available for mitigation depend on the break location.

Four break locations are analyzed:

- Break in a GDCS injection line
- Break in feedwater line A
- Break above TAF other than GDCS or FW
- Breaks below TAF.

As soon as the break takes place, the liquid coolant flows into the lower drywell, driven by gravity and hydrostatic pressure. If insufficient coolant makeup is provided to the vessel, water level decreases from the vessel flange down to the break elevation. Only break elevations below L3 analyzed; for breaks above L3, the RWCU/SDCS continues removing the decay heat, and no safety function is directly challenged.

Once the level falls below L3, decay heat removal is lost, as the RWCU/SDCS pumps receive a runback signal, and natural circulation inside the vessel is lost when water level drops below the separators skirts.

It is assumed for breaks above TAF that providing makeup to the vessel and allowing coolant boiling is an effective method for core cooling. CRD, FAPCS, Fire Protection and GDCS are considered for water makeup.

For breaks below TAF, the drywell has to be flooded up to an elevation above TAF to reach a safe core cooling condition. The personnel and equipment access hatches to the lower drywell could be open during shutdown, a recovery action to close these doors is modeled.

16.4.3.1.1 LOCA in GDCS Line (Mode 5, Mode 5 Open, Mode 6, Unflooded)

The event trees for this initiating event, in each analyzed mode are shown in Figures 16.4-10, 16.4-15, and 16.4-20. Each event tree has the same top headings and each one has one core damage sequence (all decay heat removal/inventory makeup systems fail). The headings associated with this initiating event are:

- UD-TOPINJ - CDR – Both Control Rod Drive Pumps
- VL-TOPINJ - FAPCS – Low Pressure Injection System 1
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- VI-TOPINJ – Gravity Driven Cooling System

For the specific logic associated these event trees, see the attached figures (16.4-10, 16.4-15, and 16.4-20).

16.4.3.1.2 LOCA in FDW-A Line

The event trees for this initiating event, in each analyzed mode are shown in Figures 16.4-11, 16.4-16, and 16.2-21. Each event tree has the same top headings and each one has only one core damage sequence (all decay heat removal/inventory makeup systems fail). FAPCS and FPS injection into the RPV is through feedwater Line A. With those systems unavailable, only CRD and GDCS are credited in these event trees. The headings associated with this initiating event are:

- UD-TOPINJ - CDR – Both Control Rod Drive Pumps
- VI-TOPINJ – Gravity Driven Cooling System

For the specific logic associated these event trees, see the attached figures (16.4-11, 16.4-16, and 16.4-21).

16.4.3.1.3 LOCA above TAF other than GDCS or FDW-A

The event trees for this initiating event are shown in Figures 16.4-12, 16.4-17, and 16.4-22. These event trees are the same as the ones listed above for GDCS line LOCAs. The available systems are the same, and the only difference is the initiating event and initiating event frequency. The headings associated with this initiating event are:

- UD-TOPINJ - CDR – Both Control Rod Drive Pumps
- VL-TOPINJ - FAPCS – Low Pressure Injection System 1
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- VI-TOPINJ – Gravity Driven Cooling System

For the specific logic associated these event trees, see the attached figures (16.4-12, 16.4-17, and 16.4-22).

16.4.3.1.4 LOCA below TAF in RWCU/SDC Drain Lines

The event trees for this initiating event are shown in Figures 16.4-14, 16.4-19, and 16.4-24. In all below TAF LOCA events, closure of the lower drywell hatches is required to prevent core damage. For these event trees, if the lower drywell hatch is closed, makeup to the RPV is still required to prevent core damage. The headings associated with this initiating event are:

DWH-1 - Close Drywell Hatch – RWCU/SDC drain line LOCA

- UD-TOPINJ - CDR – Both Control Rod Drive Pumps
- VL-TOPINJ - FAPCS – Low Pressure Injection System 1
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- VI-TOPINJ – Gravity Driven Cooling System

Each event tree associated with this initiator has two core damage sequences. One occurs when the drywell hatch closure fails. The other one is when the hatch is successfully closed, but no makeup is provided. For the specific logic associated these event trees, see the attached figures (16.4-14, 16.4-19, and 16.4-24).

If not closed at the time of the initiating event, the operator must close the lower drywell hatches to prevent flooding of reactor building lower levels. See Section 16.3.4.3 for details on time available for this action and the recovery failure probability.

16.4.3.1.5 LOCA Below TAF in Instrument Lines

The event tree for these initiating events are shown in Figures 16.4-13, 16.4-18, and 16.4-23. The description is the same as the preceding case, except that heading DWH-2 is used instead of DWH-1 to account for the longer time available to the operator to close the drywell hatches. The headings associated with this initiating event are:

- DWH-2 - Close Drywell Hatch – Instrument Line LOCA
- UD-TOPINJ - CDR – Both Control Rod Drive Pumps
- VL-TOPINJ - FAPCS – Low Pressure Injection System 1
- VM-TOPINJ - FPS – Low Pressure Injection System 2
- VI-TOPINJ – Gravity Driven Cooling System

Each event tree associated with this initiator has two core damage sequences. For the specific logic associated these event trees, see the attached figures (16.4-13, 16.4-18, and 16.4-23).

16.4.3.2 LOCAs in Mode 6 (Flooded)

Breaks below TAF are analyzed for LOCAs in Mode 6-Flooded. The same two below TAF break locations are analyzed:

- LOCA below TAF in RWCU.SDC Drain Lines
- LOCA below TAF in Instrument Lines

The event trees for these two cases are provided in Figures 16.6-25 and 16.6-26, respectively. Each event tree has a single top event modeling failure to close the lower drywell hatches. The failure probabilities for this node are the same as that described previously. Failure to close the lower drywell hatch is modeled as directly leading to a core damage scenario. The scenario with successful closure of the lower drywell hatches does not result in a core damage end state (and no additional top events are questioned) because sufficient water exists above the break to flood the containment above TAF.

16.5 SYSTEM ANALYSIS

This section describes the fault trees used in the Shutdown PRA evaluation. The unavailability of a system to perform its safety function on demand is evaluated by fault tree analysis.

The necessary fault trees are identified following construction of the event trees. These fault trees represent the nodes included in the event trees.

Maximum use is made of the fault trees developed for the Full Power PRA. Potential differences between the full power and the shutdown fault tree models may result from:

- Differences in maintenance unavailabilities
- Differences in success criteria between full power and shutdown condition
- Differences in initial system configuration between full power and shutdown condition
- Differences in human actions

Maintenance events used in the full power model are not adjusted for the shutdown. This treatment is conservative. Technical Specification requirements are different for some systems during shutdown. For example, four GDCS injection lines are allowed to be out of service in Mode 5. Any maintenance is likely to be on one of the out of service lines.

The following paragraphs discuss the cases where modifications were made to the full power fault tree logic to reflect the shutdown conditions.

16.5.1 Reactor Water Cleanup / Shutdown Cooling System

The logic used fro RWCU in the shutdown model is the same as is used in the full power model with only slight variation. The RWCU/SDC function modeled in shutdown is the restart of at least one RWCU train after a LOPP event, including operator action to re-start the system (it is assumed to not be automatically sequenced on upon diesel generator startup). The full power model has the system in cleanup mode and has several valve position changes modeled to switch the system to the shutdown cooling mode. These position changes are not required in shutdown since the system is in SDC mode for the duration of shutdown. Several of the basic events associate with the changing of the system from cleanup mode to SDC mode have been flagged to FALSE. The system is operating in SDC mode for all of shutdown, and only needs to restart following a LOPP. The failures flagged out of the model are not credible failures during shutdown.

Maintenance is expected to be performed mostly during full power operation, when only one train is operating. Nevertheless, the same maintenance unavailability as used for the full power PRA is conservatively used in the shutdown evaluation.

16.5.2 Control Rod Drive System

It is assumed that during the entire shutdown period a single CRD pump is running (providing purge flow to FMCRD and/or the RWCU/SDC pumps) and the second pump is in standby, which is the same initial configuration as at full power.

Additionally, it is assumed that no automatic initiation of CRD injection is available and alignment in RPV injection mode requires operator action. The same action in the fault tree for the full power model is used for the shutdown analysis. The automatic actuation of the system on RPV Level 2 is removed from the shutdown analysis with the use of flag files. In the full power fault tree, there is AND gate under system actuation. One branch of the AND gate has the automatic actuation and all the logic associated with it. The other branch of the AND gate has the operator action to initiated CRD injection. For the shutdown analysis, the gate with the automatic actuation is set to TRUE, so that the system is entirely dependant on the operator action.

16.5.3 Gravity Driven Cooling System

Technical Specification requirements for the GDCS during shutdown are different than during full power operations. For shutdown modes (5 and 6) only 4 out of the 8 GDCS injection lines are required to be available, and only 2 of 3 pools are required available. The Level 1 fault tree for GDCS is used for the shutdown analysis with one of the pools and four of the injection lines set to TRUE (failed). Success of GDCS in shutdown is 2/4 injection lines and 1 of 2 pools (as opposed to 2/8 injection lines and 2/3 pools during full power).

16.5.4 Isolation Condenser System

Technical Specification requirements for the ICS are different than during full power operations. The system will not function in Mode 6 with the vessel head removed and it has not been credited in any Mode 6 event trees. During Mode 5, only 2 out of the 4 IC units are required to be available. The Level 1 fault tree for ICS is used for the shutdown analysis with two of the four pools TRUE (failed). Success of ICS in shutdown is proper functioning of 2/2 ICS (as opposed to 2/4 during full power).

16.5.5 Fuel and Auxiliary Pool Cooling System (FAPCS)

The Level 1 fault trees for FAPCS (both before and after ADS) are left unchanged and used in the shutdown analysis. The system is required to operate during shutdown in fuel pool cooling mode, and it is the primary shutdown cooling backup for RWCU/SDC.

The system has no automatic actuation of low pressure injection and is dependent on operator initiation for all functions credited in the shutdown analysis.

16.5.6 Fire Protection System (FPS)

The Level 1 fault trees for FPS Low Pressure Injection (both before and after ADS) are left unchanged and used in the shutdown analysis. Fire Protection requirements at the plant are not likely dependant on the reactor modes.

The system has no automatic actuation of low pressure injection, and is dependant on operator initiation for all functions credited in the shutdown analysis.

16.5.7 18/18 SRVs Fail to Open in Relief Mode

This logic is modeled in the full power PRA. It represents the failure of all 18/18 SRVs to open in overpressure protection during a transient that challenges the PRV pressure. For overpressure protection, 1 of 18 SRVs must open in relief mode. The only cutset that exists for the logic (when the individual system is quantified with a truncation of 1E-16) is the common cause failure of both SRV groups. There is one group of 10 and one group of 8 SRVs. The two groups are diverse enough not to be common cause grouped together. The one cutset that shows up has a value of 2.2 E-13. For the shutdown analysis, this top event was treated as a point estimate with a failure rate of 1E-12.

16.5.8 MS-TOP2 – At Least 2 SRVs Open

This logic is also treated with a point estimate for the shutdown analysis. Manual depressurization is modeled in the Level 1 analysis. Failure of the operator action to depressurize is two orders of magnitude higher than any equipment failures. To simplify modeling, this whole heading is treated as a point estimate with a value of 5E-2 (one failure in twenty). That value is higher than the operator action screening value used in Chapter 4 Section 4.01 (Depressurization System) for the manual depressurization logic

16.6 QUANTIFICATION RESULTS

The shutdown accident sequence analysis models the impacts on the following two critical safety functions during shutdown:

- Decay Heat Removal (DHR)
- Reactor Coolant System Inventory Control

Initiating event types, and associated frequencies, are identified that challenge the above critical safety functions (refer to Section 16.3). Event trees are developed specific to the shutdown configurations (refer to Section 16.4) and the system fault tree analysis is based on at-power fault tree models adjusted to match shutdown conditions (refer to Section 16.5). The model development and quantification is performed using CAFTA and FORTE. The quantification and all sensitivities were performed at a truncation limit of 1E-15/yr.

16.6.1 Baseline Shutdown PRA Results

The core damage frequency results of the ESBWR shutdown risk analysis are summarized in the following table:

- Shutdown CDF by Initiating Event and Operating Mode (Table 16.6-1)

As can be seen from these tables, the calculated shutdown CFD is 8.77E-09/yr.

The top 200 cutsets for the shutdown CDF are provided in Table 16.6-3.

The risk importance measures for the shutdown CDF are provided in Table 16.6-4.

A list of sequences that result in core damage is provided in Table 16.4-5.

The cutsets from the dominant sequences are listed in table 16.4-6.

All evaluated shutdown core damage events are assumed to result in a large release because of the potential for the containment being open during the outage. CCFP is not affected because the containment is not being used as a mitigating system during shutdown.

DCD Chapter 19, Section 19.2.1 states the following goals for a design-specific PRA:

The design-specific PRA results and insights are compared against the following goals (note: these are goals and not regulatory requirements) and address how the plant features properly balance severe accident prevention and mitigation:

- Determine how the risk associated with the design compares against the Commission's goals of less than 1E-4/yr for core damage frequency (CDF).
- Determine how the risk associated with the design compares against the Commission's goals of less than 1E-6/yr for large release frequency (LRF).

The ESBWR shutdown results are well within these documented goals:

	CDF	LERF	
ESBWR Shutdown:	8.77E-9	8.77E-9*	(* assumes all events bypass containment)
Chapter 19 goals:	1.0E-4	1.0E-6	

The above result and all others presented below are presented on a yearly basis for comparison with NRC goals. To obtain a yearly shutdown risk with a once every two year refueling schedule, all initiating event frequencies were multiplied by a factor of 0.5.

The presented shutdown results should not be added to the full power results for a combined one year CDF value. There are large differences in uncertainty between the two models and simple addition of the two may provide unreasonable results.

16.6.2 Shutdown Sensitivity

The following sections show various sensitivity analyses run with the shutdown PRA.

16.6.2.1 LOCA Frequency

Due to the lower temperatures and pressures in the RPV during shutdown, a reduction factor was applied to LOCA frequencies for the shutdown PRA. The basis for the reduction is in Section 16.3.1.2.1 above. This sensitivity case shows the results with no reduction factor applied.

Baseline Results = 8.77E-9/yr

Sensitivity Results = 8.67E-8/yr

This case results in an order of magnitude increase in core damage frequency. Contribution from LOCA sequences makes up over 98% of the baseline result. The results for the entire shutdown are entirely dependant on LOCA frequencies and how they are determined essentially determines the results.

Even with no reduction factor applied to LOCA frequencies, the results are still well below the NRC stated goals for a design specific PRA.

16.6.2.2 Lower Drywell Hatch Sensitivity

Almost as important to the shutdown PRA results as LOCA frequencies is closure of the lower drywell hatch. Failure to close the hatch in the event of a lower drywell LOCA is assumed to lead to core damage. Two below TAF LOCA events are evaluated in each mode for a total of eight. The top eight cutsets are those LOCA events with drywell hatch closure failure. Those eight cutsets account for 98% of the total shutdown CDF.

Two hatch closure events are included in the model. In one case (instrument line LOCA) 360 minutes are available to close the hatch; in the other case (RWCU drain line) 90 minutes are available. Both times are based on the time available to close the hatch given a worst-case pipe break. Screening values have been used in the baseline case for the operator action to close the hatch. A failure probability of 0.01 is applied to the case with 360 minutes available for the action. A failure probability of 0.1 is applied to the case with 90 minutes available.

Three sensitivity cases have been analyzed.

Case 1 applies a higher failure rate for both hatch closure events. DWH-1 (90 minutes) and DWH-2 (360 minutes) are both given a 0.5 failure probability (50% failure rate).

Case 1 Results = 3.40E-7/yr

More than an order of magnitude increase compared to the baseline shutdown CDF. Though the DWH-2 value is a factor of ten lower than the DWH-1 number in the base case, the DWH-2 event is in four of the top six cutsets (Instrument line LOCA frequencies are quite a bit higher than RWCU drain line ones). The increase in the DWH-2 number leads to a big increase in shutdown CDF.

Case 2 gives both values the DWH-2 screening value (0.01 – one failure in 100).

Case 2 Results = 6.97E-9/yr

A decrease compared to the base case, but not a large one. Due to the LOCA frequency associated with the instrument line LOCA (two orders of magnitude higher than RWCU drain line), the Instrument line LOCA scenarios are the ones that contribute the most.

Case 3 assumes no lower drywell entry is allowed until Mode 6. This eliminates the Mode 5 and Mode 5 Open sequences that include drywell hatch closure.

Case 2 Results = 4.94E-9/yr

Case results in CDF about half of the base case. Due to the impact the lower drywell LOCA events has on the results, any restrictions put on entering (ensuring the hatch closed) would result in a decrease in CDF.

As is the case with the LOCA frequencies, any results are going to be a direct reflection of the numbers used for the action to close the lower drywell hatch. The two events are in only eight sequences, but those are the top eight cutsets and account for 98% of the total shutdown CDF.

16.6.2.3 Operator Action Sensitivity

During shutdown, the plant relies on operators much more for accident mitigation than during at power conditions. Several systems have no automatic actuation and rely on operators to initiate (FPS, FAPCS, CRD). Human actions are the only barrier between the initiating events and core damage for LOCA events below the TAF. Also adding to the contribution of human errors during shutdown is the fact that many systems that don't require human action are in a reduced capacity during shutdown conditions (IC having only 2 out of 4 loops available).

Two operator action sensitivity cases are evaluated.

Case 1 has all recover actions set to TRUE (failed). This eliminates several systems from possible accident mitigation. CRD (during shutdown), FAPCS, FPS, and manual depressurization are all completely dependant on human action to initiate. RWCU/SDC also requires operator action following LOPP.

Case 1 Results = 1.04E-6/yr

Results show more than a two order of magnitude increase in CDF over the baseline case. The plant has features to safeguard against possible transients, but during shutdown, the arrangement of the plant requires operators to take action to put those systems into service. For many sequences in this case, the CDF value is equal to the initiating event value.

Case 2 has all recovery actions given a much lower than usual human error probability: 1.0E-3. This number is about an order of magnitude lower than most modeled human actions. It shows how CDF could be affected if credit is taken for very effective operator response to transients.

Case 2 Results = 7.00E-10/yr

Case results in decrease in CDF of approximately one order of magnitude compared to the base shutdown case. The top cutsets in this case are still dominated by human errors. Even with the reduced failure values, the human errors are still generally higher than the common cause equipment failures that show up in the top cutsets with the human errors.

The Shutdown PRA is somewhat sensitive to human errors. Many of the plants passive safety features are unavailable due to the conditions of shutdown (PCCS, ICS in Mode 6). In Mode 6, only one system that is not entirely dependant on operators is available to prevent CDF (GDCS).

16.7 INISGHTS FROM SHUTDOWN PRA

By far, the greatest contribution to shutdown risk comes from breaks in lines connected to the vessel below TAF. In these cases, the lower drywell equipment hatch or personnel hatch is likely to be open to facilitate work in the lower drywell. Although the frequency of these events is very low, there is only one method for mitigation – manual closure of the hatch(es).

In order to minimize the risk from these scenarios, refueling outages must be conducted in a judicious manner. Whenever the hatches are open, procedures shall require personnel to be available and in close proximity to the hatches, with the purpose of providing fast closure of the containment in the event of a water leak. Other measures can be taken, including temporary installation of equipment to aid in closing the hatch or to minimize the flooding rate in the lower drywell. The next largest contributions to shutdown risk are due to losses of preferred power (LOPP) or loss of all service water (PSWS) during Mode 6 Unflooded. These scenarios are higher than other shutdown ones due to ICS being unavailable in Mode 6.

LOCA events above the TAF do not contribute much at all to the entire overall shutdown CDF. The highest sequence for any LOCA above the fuel has a value of 3E-13, which is four orders of magnitude below the highest cutset, and two orders of magnitude below the highest non-LOCA cutsets.

16.8 CONCLUSIONS

The main conclusion that can be drawn from the ESBWR shutdown risk analysis is that the ESBWR design provides a robust, passive means for preventing shutdown core damage events and the results are well below the NRC goals for CDF and LERF. The key risk insights are:

- Shutdown process should provide assurance that the equipment and personnel hatches in the lower drywell can be isolated in the event of a leak.
- Operator and staff training ensures shutdown transient identification and mitigation is stressed. Few fully automated systems are credited in the shutdown transient response. Human action is key in preventing core damage.

	CDF	LERF	
ESBWR Shutdown:	8.77E-9	8.77E-9*	(* assumes all events bypass containment)
Chapter 19/NRC goals:	1.0E-4	1.0E-6	

16.9 REFERENCES

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16-6. Residual Heat Removal Experience Review and Safety Analysis: Pressurized Water Reactors, NSAC-52, January 1983

16-7. ABWR Standard Safety Analysis Report, 23A6100, August 1996

16-8. NEA/CSNI/R (2005)11/Vol. 121-Sep-2005 "Improving Low Power and Shutdown PSA Methods and Data to Permit Better Risk Comparison and Trade-Off Decision-Making"/ Nuclear Energy Agency (NEA) - Committee on the Safety of Nuclear Installations

Table 16.3-1

Shutdown Initiating Events Challenging Critical Safety Functions

Critical Safety Function	Initiating Event
Decay Heat Removal	Loss of Both RWCU/SDCS trains
	Loss of Preferred Power
	Loss of RCCWS/PSWS
Reactor Coolant System Inventory Control (Modes 5 and 6)	GDCS LOCA
	Feedwater LOCA
	LOCA other than GDCS or feedwater
	Instrument Line LOCA below TAF
	RWCU drain line below TAF

Table 16.3-2a

ESBWR Shutdown PRA Initiating Event Types

Initiating Event	Applicable Modes
Loss of Both Operating RWCU/SDCS Trains	5, 5O, 6U
Loss of Preferred Power	5, 5O, 6U
Loss of RCCWS/PSWS	5, 5O, 6U
LOCA in GDCS Line	5, 5O, 6U
LOCA in FW-A Line	5, 5O, 6U
LOCA other than GDCS/FW	5, 5O, 6U
LOCA Below TAF in RWCU/SDC Drain Lines	All
LOCA Below TAF in Instrument Lines	All
Mode Descriptions 5 – Mode 5 5O – Mode 5 Open 6U – Mode 6 Unfolded 6F- Mode 6 Flooded	

Table 16.3-2b
ESBWR Shutdown PRA Event Trees

Event Tree / Initiating Events	Mode	Figure
Loss of PSWS	5	16.4-1
Loss of RWCU/SDC	5	16.4-2
Loss of Pref Power	5	16.4-3
Loss of PSWS	5 open	16.4-4
Loss of RWCU/SDC	5 open	16.4-5
Loss of Pref Power	5 open	16.4-6
Loss of PSWS	6 Unflooded	16.4-7
Loss of RWCU/SDC	6 Unflooded	16.4-8
Loss of Pref Power	6 Unflooded	16.4-9
LOCA - RWCU/SDC below TAF	5	16.4-10
LOCA - Other than FW or GDCS	5	16.4-11
LOCA - FW Line A	5	16.4-12
LOCA - Instrument Line below TAF	5	16.4-13
LOCA - GDCS	5	16.4-14
LOCA - RWCU/SDC below TAF	5 open	16.4-15
LOCA - Other than FW or GDCS	5 open	16.4-16
LOCA - FW Line A	5 open	16.4-17
LOCA - Instrument Line below TAF	5 open	16.4-18
LOCA - GDCS	5 open	16.4-19
LOCA - RWCU/SDC below TAF	6 Unflooded	16.4-20
LOCA - Other than FW or GDCS	6 Unflooded	16.4-21
LOCA - FW Line A	6 Unflooded	16.4-22
LOCA - Instrument Line below TAF	6 Unflooded	16.4-23
LOCA - GDCS	6 Unflooded	16.4-24
LOCA - Instrument Line below TAF	6 Flooded	16.4-25
LOCA - RWCU/SDC below TAF	6 Flooded	16.4-26

Table 16.3-3a

ESBWR Shutdown PRA Initiating Event Frequencies

Initiating Events	Mode	Value	Units	Source	Value/hour	Hours in Mode	per shutdown value	Yearly value (0.5 shutdowns/year)	Values with LOCA correction
Loss of PSWS	5	9.70E-04	events/year	1	1.11E-07	192	2.13E-05	**1.06E-05**	
Loss of RWCU/SDC	5	3.15E-5	/hour	3	3.15E-5	192	6.55E-3	**3.02E-03**	
Loss of Pref Power	5	2.20E-05	/hour	2	2.20E-05	192	4.22E-03	**2.11E-03**	
Loss of PSWS	5 open	9.70E-04	events/year	1	1.11E-07	48	5.32E-06	**2.66E-06**	
Loss of RWCU/SDC	5 open	3.15E-5	/hour	3	3.15E-5	48	1.51E-3	**7.56E-04**	
Loss of Pref Power	5 open	2.20E-05	/hour	2	2.20E-05	48	1.06E-03	**5.28E-04**	
Loss of PSWS	6 unflooded	9.70E-04	events/year	1	1.11E-07	60	6.64E-06	**3.32E-06**	
Loss of RWCU/SDC	6 unflooded	3.15E-5	/hour	3	3.15E-5	60	1.89E-3	**9.45E-04**	
Loss of Pref Power	6 unflooded	2.20E-05	/hour	2	2.20E-05	60	1.32E-03	**6.60E-04**	
LOCA - RWCU/SDC below TAF	5	6.51E-06	events/year	5	7.43E-10	192	1.43E-07	7.13E-08	**7.13E-09**
LOCA - Other than FW or GDCS	5	1.68E-04	events/year	8	1.92E-08	192	3.68E-06	1.84E-06	**1.84E-07**
LOCA - FW Line A	5	1.10E-05	events/year	6	1.26E-09	192	2.41E-07	1.21E-07	**1.21E-08**
LOCA - Instrument Line below TAF	5	2.14E-04	events/year	4	2.44E-08	192	4.69E-06	2.35E-06	**2.35E-07**
LOCA - GDCS	5	1.27E-05	events/year	7	1.45E-09	192	2.78E-07	1.39E-07	**1.39E-08**
LOCA - RWCU/SDC below TAF	5 open	6.51E-06	events/year	5	7.43E-10	48	3.57E-08	1.78E-08	**1.78E-09**
LOCA - Other than FW or GDCS	5 open	1.68E-04	events/year	8	1.92E-08	48	9.21E-07	4.60E-07	**4.60E-08**
LOCA - FW Line A	5 open	1.10E-05	events/year	6	1.26E-09	48	6.03E-08	3.01E-08	**3.01E-09**
LOCA - Instrument Line below TAF	5 open	2.14E-04	events/year	4	2.44E-08	48	1.17E-06	5.86E-07	**5.86E-08**
LOCA - GDCS	5 open	1.27E-05	events/year	7	1.45E-09	48	6.96E-08	3.48E-08	**3.48E-09**
LOCA - RWCU/SDC below TAF	6 unflooded	6.51E-06	events/year	5	7.43E-10	60	4.46E-08	2.23E-08	**2.23E-09**
LOCA - Other than FW or GDCS	6 unflooded	1.68E-04	events/year	8	1.92E-08	60	1.15E-06	5.75E-07	**5.75E-08**

Table 16.3-3a

ESBWR Shutdown PRA Initiating Event Frequencies

Initiating Events	Mode	Value	Units	Source	Value/hour	Hours in Mode	per shutdown value	Yearly value (0.5 shutdowns/year)	Values with LOCA correction
LOCA - FW Line A	6 unflooded	1.10E-05	events/year	6	1.26E-09	60	7.53E-08	3.77E-08	**3.77E-09**
LOCA - Instrument Line below TAF	6 unflooded	2.14E-04	events/year	4	2.44E-08	60	1.47E-06	7.33E-07	**7.33E-08**
LOCA - GDCS	6 unflooded	1.27E-05	events/year	7	1.45E-09	60	8.70E-08	4.35E-08	**4.35E-09**
LOCA - Instrument Line below TAF	6 Flooded	2.14E-04	events/year	4	2.44E-08	240	5.86E-06	2.93E-06	**2.93E-07**
LOCA - RWCU/SDC below TAF	6 Flooded	6.51E-06	events/year	5	7.43E-10	240	1.78E-07	8.92E-08	**8.92E-09**

Note: See Table below for source details

Notes:

(1) As discussed in Section 16.2, the time in each Operating Mode is assumed to be as follows: Mode 5, 238 hrs; Mode 6-Unflooded, 59 hrs; and Mode 6-Flooded, 241 hrs.

(2) The shutdown initiating event frequencies per year are calculated using the hourly frequencies of Table 16.3-1 and multiplying by the hours in the corresponding Operating Mode. An additional 0.5 factor is applied given that shutdown is expected to occur every two years.

Table 16.3-3b

ESBWR Initiating Event Frequency Sources

	Sources
1	NEDO-33201 Rev 2, Chapter 2, Table 2.3-3 "Complete Loss of PSWS"
2	NUREG/CR-5496 (Reference 16-2) shutdown loss of offsite power frequency on a per hour basis.
3	NEDO-33201 Rev 2, Chapter 4, Section 4.08 (Sum of 6 RWCU CCF events that can cause failure to run)*
4	NEDO-33201 Rev 2, Chapter 2, Table 2.3-1 (il)
5	NEDO-33201 Rev 2, Chapter 2, Table 2.3-1 (g)
6	NEDO-33201 Rev 2, Chapter 2, Table 2.3-1 (c)
7	NEDO-33201 Rev 2, Chapter 2, Table 2.3-1 (f)
8	NEDO-33201 Rev 2, Chapter 2, Table 2.3-1 (d, e, fl, h, i2)
*	CCF events include: 3 of 3 Pumps fail to run, AOV/NOV Spurious Transfer, MOVs fail to close, and suction flow transmitters fail low

Table 16.3-4

BWR and PWR Loss of Running RHR Pump Events During Shutdown

SITE	CAUSE	REFERENCE	TIME (hours)
AG	PUMP TRIP AFTER TRANSFER OF RPS POWER SUPPLY	NSAC-157	0.02
DAVIS BESSE 1	PUMP FAILED TO START	NSAC-52	0.07
CALVERT CLIFFS 1	LPSI ACTUATION CAUSES PUMP TRIP	NSAC-52	0.25
CALVERT CLIFFS 2	LPSI ACTUATION CAUSES PUMP TRIP	NSAC-52	0.38
272/89-019	GAS BINDING (N2) DURING ACCUMULATOR DISCHARGE VALVE TESTING	AEOD/S93-05	0.70
CALVERT CLIFFS 2	CAVITATION	NSAC-52	2.00
BRUNSWICK 2	START FAILURE	NSAC-88	--
PEACH BOTTOM 3	PUMP TRIP (OVERCURRENT)	NSAC-88	--
BRUNSWICK 2	HIGH PUMP MOTOR RUNNING TEMP	NSAC-88	--
HATCH 1	HIGH RHR PUMP BEARING TEMP	NSAC-88	--
BRUNSWICK 2	LOSS OF PUMP SHAFT SEAL COOLING SYSTEM	NSAC-88	--
GRAND GULF	COOLING FAN FAILURE	NSAC-88	--
BROWNS FERRY 1	PUMP WINDING FAILURE (OVERCURRENT TRIP)	NSAC-88	--
BI	PUMP TRIP DUE TO WORN IMPELLER EYE WEAR RING	NSAC-157	--
DH	PUMP TRIP DUE TO HYDRAULIC TRANSIENT WHEN ONE PUMP STOPPED	NSAC-157	--
AI	PUMP TRIP DUE TO LOGIC POWER SUPPLY DISTURBANCE	NSAC-157	--
271/91-009	AIR BINDING OF PUMP DUE TO LOW RCS LEVEL	AEOD/S93-05	--

Table 16.3-5

Recovery Actions Failure Probabilities

Recovery Event	Mode	Time Allowable (hrs)	Failure Probability
Recovery of RWCU/SDCS	5	4	2.29E-01
	6-Unflooded	5.3	2.18E-01
Recovery of RCCWS/PSWS	5	4	2.29E-01
	6-Unflooded	5.3	2.18E-01
Close the lower drywell hatches (RWCU break)	6	1.5	1.00E-01
Close the lower drywell hatches (instr. line break)	6	6.0	1.00E-02

Table 16.3-6

Time Available to Close Lower DW Hatches

Parameter	**Break in RWCU/SDCS drainlines**	**Break in instrument lines below TAF**	**Units**
Lower drywell radius, R	5600		mm
Height from lower drywell to bottom of hatch	2400		mm
Volume of lower drywell below hatch, $V = \pi R^2 H$	236		m^3
Height of water above break, h	25000		mm
Break diameter, d	50	25	mm
Flow rate through break, $Q = (\pi d^2/4)\sqrt{2gh}$	157	39	m^3/hr
Time available[(1)], t = V/Q	91	362	min

Note:

(1) This is the time available to close the lower DW hatches (if open) before flood level in containment reaches the bottom elevation of the hatch opening. Closure of both the equipment hatch and personal hatch can be done form outside the lower drywell/containment.

(2) No credit is given to the two lower drywell sump pumps. Operating sump pumps would give personnel more time to close hatches.

Table 16.5-1

System Maintenance Unavailabilities During Shutdown

System	Mode 5	Mode 6	Observations
ICS	As at full power	Out of service	Maintenance expected in Mode 6, with vessel depressurized.
RWCU/SDC	As at full power	As at full power	1 or 2 trains operating. Maintenance expected to be carried out during full power on one train while the other train operates in RWCU mode.
ADS/SRV	As at full power	Out of service	Maintenance expected in Mode 6, with vessel depressurized.
Feedwater and condensate	Higher maintenance unavailability	Higher maintenance unavailability	Not credited in shutdown PRA analysis though it may be available.
CRD (injection mode)	As at full power	As at full power	Maintenance can be executed during full power on one train while the other train is in operation or on standby.
FAPCS	As at full power	As at full power	Maintenance can be executed during full power as well as during shutdown on one train while the other is operating or on standby.
FPS	As at full power	As at full power	Maintenance can be executed during full power as well as during shutdown on one train while the other is operating or on standby.
GDCS	As in full power	Higher maintenance unavailability	Maintenance during shutdown will be performed on lines and pools allowed out of service by Tech specs.
PCCS	As at full power	Out of service	Not credited in shutdown PRA analysis though it may be available for Mode 5 while containment is still intact. Maintenance expected in Mode 6, with vessel depressurized and containment open
RCCWS	As at full power	As at full power	Maintenance can be executed during full power as well as during shutdown
PSWS	As at full power	As at full power	Maintenance can be executed during full power as well as during shutdown
Service/Instrument Air	As at full power	As at full power	Maintenance can be executed during full power as well as during shutdown
HPNSS	As at full power	As at full power	Maintenance can be executed during full power as well as during shutdown
13.8 kV Power Distribution	As at full power	As at full power	It is assumed that any bus, power center or MCC can be maintained during power operation.
Diesel Generators	As at full power	As at full power	It is assumed that one DG at a time can be maintained during power operation

Table 16.5-1

System Maintenance Unavailabilities During Shutdown

System	Mode 5	Mode 6	Observations
Uninterruptible AC Power	As at full power	As at full power	It is assumed that any bus, power center or MCC can be maintained during power operation.
250V DC Power	As at full power	As at full power	It is assumed that any division can be maintained during power operation.

Table 16.6-1

Shutdown CDF by Initiating Event and by Mode of Operation

Initiating Event	CDF (Per Year)	% Of Total	Mode	CDF (Per Year)
Loss of PSWS	ε	0.00%	5	3.07E-09
Loss of RWCU/SDC	5.00E-12	0.10%		
Loss of Pref Power	1.00E-12	0.00%		
LOCA - RWCU/SDC below TAF	7.13E-10	8.10%		
LOCA - Other than FW or GDCS	ε	0.00%		
LOCA - FW Line A	ε	0.00%		
LOCA - Instrument Line below TAF	2.35E-09	26.80%		
LOCA - GDCS	ε	0.00%		
Loss of PSWS	ε	0.00%	5 Open	7.66E-10
Loss of RWCU/SDC	1.00E-12	0.00%		
Loss of Pref Power	1.00E-12	0.00%		
LOCA - RWCU/SDC below TAF	1.78E-10	2.00%		
LOCA - Other than FW or GDCS	ε	0.00%		
LOCA - FW Line A	ε	0.00%		
LOCA - Instrument Line below TAF	5.86E-10	6.70%		
LOCA - GDCS	ε	0.00%		
Loss of PSWS	2.60E-11	0.30%	6 Unflooded	1.12E-09
Loss of RWCU/SDC	2.00E-12	0.00%		
Loss of Pref Power	1.37E-10	1.60%		
LOCA - RWCU/SDC below TAF	2.23E-10	2.50%		
LOCA - Other than FW or GDCS	ε	0.00%		
LOCA - FW Line A	ε	0.00%		
LOCA - Instrument Line below TAF	7.33E-10	8.30%		
LOCA - GDCS	ε	0.00%		
LOCA - Instrument Line below TAF	2.93E-09	33.40%	6 Flooded	3.82E-09
LOCA - RWCU/SDC below TAF	8.92E-10	10.20%		
Total	**8.77E-09**			**8.77E-09**

Table 16.6-2

Shutdown CDF by Accident Class (Deleted)

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
1	2.93E-09	2.93E-07	%M6F_LOCA_I	INSTRUMENT LINE LOCA IN MODE 6 FLOODED
		1.00E-02	DWH-2	FAILURE TO CLOSE DRYWELL HATCH
2	2.35E-09	2.35E-07	%M5-LOCA-I	LOCA - INSTRUMENT LINE BELOW TAF MODE 5
		1.00E-02	DWH-2	FAILURE TO CLOSE DRYWELL HATCH
3	8.92E-10	8.92E-09	%M6F_LOCA_R	RWCU LOCA IN MODE 6 FLOODED
		1.00E-01	DWH-1	CLOSE LOWER DRYWELL HATCH
4	7.33E-10	7.33E-08	%M6U_LOCA-I	INSTRUMENT LINE LOCA - MODE 6 UNFLOODED
		1.00E-02	DWH-2	FAILURE TO CLOSE DRYWELL HATCH
5	7.13E-10	7.13E-09	%M5-LOCA-RW	LOCA - RWCU BELOW TAF
		1.00E-01	DWH-1	CLOSE LOWER DRYWELL HATCH
6	5.86E-10	5.86E-08	%M5O_LOCA_I	LOCA - INSTRUMENT LINE - MODE 5 OPEN
		1.00E-02	DWH-2	FAILURE TO CLOSE DRYWELL HATCH
7	2.23E-10	2.23E-09	%M6U_LOCA-RW	RWCU LOCA - MODE 6 UNFLOODED
		1.00E-01	DWH-1	CLOSE LOWER DRYWELL HATCH
8	1.78E-10	1.78E-09	%M5O_LOCA_R	LOCA - RWCU DRAINLINE MODE 5 OPEN
		1.00E-01	DWH-1	CLOSE LOWER DRYWELL HATCH
9	1.05E-11	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
10	5.45E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
11	5.25E-12	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
12	4.96E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
13	3.50E-12	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
14	2.72E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
15	2.48E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
16	2.06E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
17	2.06E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
18	2.06E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
19	2.06E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
20	2.06E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
21	2.06E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
22	1.87E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
23	1.87E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
24	1.87E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
25	1.87E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
26	1.87E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
27	1.87E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
28	1.85E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.93E-04	P21-ACV-OO-F0016_1_2	CCF of two components: P21-ACV-OO-F016A & P21-ACV-OO-F016B

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
29	1.81E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
30	1.75E-12	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
31	1.65E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
32	1.03E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
33	1.03E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
34	1.03E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
35	1.03E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
36	1.03E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
37	1.03E-12	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
				NO SLCS
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
38	9.60E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-04	C62-CCFSOFTWARE	Common cause failure of software
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
39	9.60E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-04	C62-CCFSOFTWARE_S	Common cause failure of software, for spurious
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
40	9.33E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
41	9.33E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
42	9.33E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
43	9.33E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
44	9.33E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
45	9.33E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
46	9.25E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.93E-04	P21-ACV-OO-F0016_1_2	CCF of two components: P21-ACV-OO-F016A & P21-ACV-OO-F016B
47	9.06E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
				DEPRESSURIZATION
48	8.24E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
49	6.84E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
50	6.84E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
51	6.84E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
				DEPRESSURIZATION
52	6.84E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
53	6.84E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
54	6.84E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
55	6.80E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020A	MOTOR OPER. VALVE F020A FAILS TO OPEN
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
				NO SLCS
56	6.80E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020B	MOTOR OPER. VALVE F020B FAILS TO OPEN
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
57	6.27E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.69E-03	C12-MP_-FS-C001B	MOTOR-DRIVEN PUMP C001B FAILS TO START
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
58	6.22E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
59	6.22E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
60	6.22E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
61	6.22E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
62	6.22E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
63	6.22E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
64	6.18E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020A	MOTOR OPER. VALVE F020A FAILS TO OPEN
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
65	6.18E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020B	MOTOR OPER. VALVE F020B FAILS TO OPEN
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
66	6.16E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.93E-04	P21-ACV-OO-F0016_1_2	CCF of two components: P21-ACV-OO-F016A & P21-ACV-OO-F016B
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
67	5.71E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.69E-03	C12-MP_-FS-C001B	MOTOR-DRIVEN PUMP C001B FAILS TO START
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
68	5.10E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-03	C12-SYS-TM-TRAINB	TRAIN B IN MAINTENANCE
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
69	4.96E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
70	4.79E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-04	C62-CCFSOFTWARE	Common cause failure of software
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
71	4.79E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-04	C62-CCFSOFTWARE_S	Common cause failure of software, for spurious
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
72	4.64E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-03	C12-SYS-TM-TRAINB	TRAIN B IN MAINTENANCE
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
73	4.08E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		2.40E-03	C12-MP_-FS-C001BOIL	MOTOR-DRIVEN AUX. OIL PUMP FOR C001B FAILS TO START
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
74	3.98E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
75	3.98E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
76	3.98E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
77	3.98E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
78	3.98E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		4.84E-02	C12-BV_-RE-F064	MISPOSITION OF VALVE F064
		4.84E-02	C12-BV_-RE-F065	MISPOSITION OF LOCKED OPEN VALVE F065
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
79	3.71E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		2.40E-03	C12-MP_-FS-C001BOIL	MOTOR-DRIVEN AUX. OIL PUMP FOR C001B FAILS TO START
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
80	3.62E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
81	3.62E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
82	3.62E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
83	3.62E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
84	3.62E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F064	MISPOSITION OF VALVE F064
		4.84E-02	C12-BV_-RE-F065	MISPOSITION OF LOCKED OPEN VALVE F065
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
85	3.50E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
		1.61E-03	U43-XHE-FO-LPCI	OPERATOR FAILS TO ACTUATE U43 IN LPCI MODE
86	3.41E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
87	3.41E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
88	3.41E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
89	3.41E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
90	3.41E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
91	3.41E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
92	3.39E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020A	MOTOR OPER. VALVE F020A FAILS TO OPEN
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
93	3.39E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020B	MOTOR OPER. VALVE F020B FAILS TO OPEN
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
94	3.19E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-04	C62-CCFSOFTWARE	Common cause failure of software
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
				DEPRESSURIZATION
95	3.19E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-04	C62-CCFSOFTWARE_S	Common cause failure of software, for spurious
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
96	3.13E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.69E-03	C12-MP_-FS-C001B	MOTOR-DRIVEN PUMP C001B FAILS TO START
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
97	3.11E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
98	3.11E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
99	3.11E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
100	3.11E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
101	3.11E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
102	3.11E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
103	3.09E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020A	MOTOR OPER. VALVE F020A FAILS TO OPEN
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
104	3.09E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020B	MOTOR OPER. VALVE F020B FAILS TO OPEN
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
105	3.08E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.93E-04	P21-ACV-OO-F0016_1_2	CCF of two components: P21-ACV-OO-F016A & P21-ACV-OO-F016B
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
106	2.85E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.69E-03	C12-MP_-FS-C001B	MOTOR-DRIVEN PUMP C001B FAILS TO START
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
107	2.54E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-03	C12-SYS-TM-TRAINB	TRAIN B IN MAINTENANCE
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
108	2.48E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
109	2.47E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		7.05E-06	E50-UV_OC_1_4_5	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003D & E50-UV_-OC-F003E
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
110	2.47E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		7.05E-06	E50-UV_OC_1_4_6	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003D & E50-UV_-OC-F003F
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
111	2.47E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		7.05E-06	E50-UV_OC_1_5_6	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003E & E50-UV_-OC-F003F
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
112	2.47E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		7.05E-06	E50-UV_OC_4_5_6	CCF of three components: E50-UV_-OC-F003D & E50-UV_-OC-F003E & E50-UV_-OC-F003F
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
113	2.31E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-03	C12-SYS-TM-TRAINB	TRAIN B IN MAINTENANCE
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
114	2.26E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020A	MOTOR OPER. VALVE F020A FAILS TO OPEN
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
115	2.26E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020B	MOTOR OPER. VALVE F020B FAILS TO OPEN
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
116	2.09E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.69E-03	C12-MP_-FS-C001B	MOTOR-DRIVEN PUMP C001B FAILS TO START
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
117	2.06E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020A	MOTOR OPER. VALVE F020A FAILS TO OPEN
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
118	2.06E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020B	MOTOR OPER. VALVE F020B FAILS TO OPEN
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
119	2.04E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		2.40E-03	C12-MP_-FS-C001BOIL	MOTOR-DRIVEN AUX. OIL PUMP FOR C001B FAILS TO START
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
120	1.99E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
121	1.99E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
122	1.99E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
123	1.99E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
124	1.99E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F064	MISPOSITION OF VALVE F064
		4.84E-02	C12-BV_-RE-F065	MISPOSITION OF LOCKED OPEN VALVE F065
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
125	1.90E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.69E-03	C12-MP_-FS-C001B	MOTOR-DRIVEN PUMP C001B FAILS TO START
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
126	1.88E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		1.75E-02	E50-UV_-OC-F003A	CHECK VALVE F003A FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003D	CHECK VALVE F003D FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003E	CHECK VALVE F003E FAILS TO REMAIN OPEN OR PLUG
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
127	1.88E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		1.75E-02	E50-UV_-OC-F003A	CHECK VALVE F003A FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003D	CHECK VALVE F003D FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003F	CHECK VALVE F003F FAILS TO REMAIN OPEN OR PLUG
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
128	1.88E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		1.75E-02	E50-UV_-OC-F003A	CHECK VALVE F003A FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003E	CHECK VALVE F003E FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003F	CHECK VALVE F003F FAILS TO REMAIN OPEN OR PLUG
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
129	1.88E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		1.75E-02	E50-UV_-OC-F003D	CHECK VALVE F003D FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003E	CHECK VALVE F003E FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003F	CHECK VALVE F003F FAILS TO REMAIN OPEN OR PLUG

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
130	1.87E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
131	1.87E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
132	1.87E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
133	1.87E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
134	1.87E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
135	1.87E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.21E-02	P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
136	1.85E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		2.40E-03	C12-MP_-FS-C001BOIL	MOTOR-DRIVEN AUX. OIL PUMP FOR C001B FAILS TO START
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
137	1.81E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
138	1.81E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
139	1.81E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
140	1.81E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
141	1.81E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F064	MISPOSITION OF VALVE F064
		4.84E-02	C12-BV_-RE-F065	MISPOSITION OF LOCKED OPEN VALVE F065
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
142	1.75E-13	3.32E-06	%M6U_LPSWS	LOSS OF SERVICE WATER - MODE 6 UNFLOODED
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		2.18E-01	R-PSWS-6	SERVICE WATER RECOVERY
		1.61E-03	U43-XHE-FO-LPCI	OPERATOR FAILS TO ACTUATE U43 IN LPCI MODE
143	1.74E-13	1.71E-06	%M6U_G31	LOSS OF RWCU - MODE 6 UNFLOODED

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.18E-01	R-M6-G31	FAILURE TO RECOVER RWCU/SDC
144	1.70E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-03	C62-UNDEVSPUR5	Undeveloped spurious hardware failure
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
145	1.70E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-03	C62-UNDEVSPUR7	Undeveloped spurious hardware failure
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
146	1.70E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-03	C63-UNDEVSPUR126	Undeveloped spurious hardware failure
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
147	1.70E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-03	C63-UNDEVSPUR127	Undeveloped spurious hardware failure
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
148	1.70E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-03	C12-SYS-TM-TRAINB	TRAIN B IN MAINTENANCE
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
149	1.65E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
150	1.59E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-04	C62-CCFSOFTWARE	Common cause failure of software
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
151	1.59E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-04	C62-CCFSOFTWARE_S	Common cause failure of software, for spurious
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
152	1.55E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.00E-03	C62-UNDEVSPUR5	Undeveloped spurious hardware failure
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
153	1.55E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-03	C62-UNDEVSPUR7	Undeveloped spurious hardware failure
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
154	1.55E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-03	C63-UNDEVSPUR126	Undeveloped spurious hardware failure
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
155	1.55E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.00E-03	C63-UNDEVSPUR127	Undeveloped spurious hardware failure
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
156	1.54E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-03	C12-SYS-TM-TRAINB	TRAIN B IN MAINTENANCE
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
157	1.36E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		2.40E-03	C12-MP_-FS-C001BOIL	MOTOR-DRIVEN AUX. OIL PUMP FOR C001B FAILS TO START
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
158	1.32E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
159	1.32E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
160	1.32E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
161	1.32E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
162	1.32E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F064	MISPOSITION OF VALVE F064
		4.84E-02	C12-BV_-RE-F065	MISPOSITION OF LOCKED OPEN VALVE F065
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
163	1.31E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		3.00E-03	R10-LOSP-EPRI	CONSEQENTIAL LOSS OF PREFERRED OFFSITE POWER DUE TO A TRANSIENT
		4.54E-03	R21-DG_-FR-CCF_1_2	CCF of two components: R21-DG_-FR-DGA & R21-DG_-FR-DGB
164	1.28E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		7.05E-06	E50-UV_OC_1_4_5	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003D & E50-UV_-OC-F003E
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
165	1.28E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		7.05E-06	E50-UV_OC_1_4_6	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003D & E50-UV_-OC-F003F
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
166	1.28E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		7.05E-06	E50-UV_OC_1_5_6	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003E & E50-UV_-OC-F003F
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
167	1.28E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		7.05E-06	E50-UV_OC_4_5_6	CCF of three components: E50-UV_-OC-F003D & E50-UV_-OC-F003E & E50-UV_-OC-F003F
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
168	1.23E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		2.40E-03	C12-MP_-FS-C001BOIL	MOTOR-DRIVEN AUX. OIL PUMP FOR C001B FAILS TO START
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
169	1.21E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.26E-05	P21-AHU-FR_1_2	CCF of two components: P21-AHU-FR-RCCWA & P21-AHU-FR-RCCWB
170	1.20E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
171	1.20E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
172	1.20E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
173	1.20E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
174	1.20E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.84E-02	C12-BV_-RE-F064	MISPOSITION OF VALVE F064
		4.84E-02	C12-BV_-RE-F065	MISPOSITION OF LOCKED OPEN VALVE F065
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
175	1.17E-13	1.21E-08	%M5-LOCA-FW	LOCA - FEEDWATER - MODE 5
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
176	1.16E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		7.05E-06	E50-UV_OC_1_4_5	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003D & E50-UV_-OC-F003E
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
177	1.16E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		7.05E-06	E50-UV_OC_1_4_6	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003D & E50-UV_-OC-F003F
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
178	1.16E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		7.05E-06	E50-UV_OC_1_5_6	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003E & E50-UV_-OC-F003F

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
179	1.16E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		7.05E-06	E50-UV_OC_4_5_6	CCF of three components: E50-UV_-OC-F003D & E50-UV_-OC-F003E & E50-UV_-OC-F003F
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
180	1.15E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		4.00E-03	P21-MOV-CC-F034B	MOV P21-F034B FROM RCCWS TO RWCU/SDC HX-B FAILS TO OPEN
		3.00E-03	R10-LOSP-EPRI	CONSEQENTIAL LOSS OF PREFERRED OFFSITE POWER DUE TO A TRANSIENT
181	1.15E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.20E-05	P41-FAN-FR_ALL	CCF of all components in group 'P41-FAN-FR'
182	1.13E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020A	MOTOR OPER. VALVE F020A FAILS TO OPEN
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
183	1.13E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020B	MOTOR OPER. VALVE F020B FAILS TO OPEN
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
184	1.10E-13	2.35E-07	%M5-LOCA-I	LOCA - INSTRUMENT LINE BELOW TAF MODE 5
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
185	1.09E-13	3.27E-04	%M5-G31	LOSS OF RWCU/SDC MODE 5
		1.00E-04	C63-CCFSOFTWARE_S	Common cause failure of software, for spurious
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		2.29E-01	R-M5-G31	RWCU/SDC RECOVERY
186	1.04E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.69E-03	C12-MP_-FS-C001B	MOTOR-DRIVEN PUMP C001B FAILS TO START
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
187	1.03E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020A	MOTOR OPER. VALVE F020A FAILS TO OPEN
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
188	1.03E-13	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		4.00E-03	C12-MOV-CC-F020B	MOTOR OPER. VALVE F020B FAILS TO OPEN

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
189	9.95E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F003A	MISPOSITION OF VALVE FOO3A
		4.84E-02	C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
190	9.95E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F003A	MISPOSITION OF VALVE FOO3A
		4.84E-02	C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
191	9.95E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F003B	MISPOSITION OF VALVE F003B
		4.84E-02	C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
192	9.95E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F003B	MISPOSITION OF VALVE F003B
		4.84E-02	C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
193	9.73E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.75E-02	E50-UV_-OC-F003A	CHECK VALVE F003A FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003D	CHECK VALVE F003D FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003E	CHECK VALVE F003E FAILS TO REMAIN OPEN OR PLUG
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
194	9.73E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.75E-02	E50-UV_-OC-F003A	CHECK VALVE F003A FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003D	CHECK VALVE F003D FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003F	CHECK VALVE F003F FAILS TO REMAIN OPEN OR PLUG
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
195	9.73E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.75E-02	E50-UV_-OC-F003A	CHECK VALVE F003A FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003E	CHECK VALVE F003E FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003F	CHECK VALVE F003F FAILS TO REMAIN OPEN OR PLUG
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
196	9.73E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.21E-02	C12-XHE-FO-LEVEL2	Operator fails to back-up CRD actuation
		1.75E-02	E50-UV_-OC-F003D	CHECK VALVE F003D FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003E	CHECK VALVE F003E FAILS TO REMAIN OPEN OR PLUG
		1.75E-02	E50-UV_-OC-F003F	CHECK VALVE F003F FAILS TO REMAIN OPEN OR PLUG
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.77E-02	G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
197	9.55E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.00E-04	E50-UV_OC_ALL	CCF of all components in group 'E50-UV_OC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		3.00E-03	R10-LOSP-EPRI	CONSEQENTIAL LOSS OF PREFERRED OFFSITE POWER DUE TO A TRANSIENT
		5.76E-02	R21-DG_-FR-DGA	DIESEL GENERATOR "A" FAILS TO RUN GIVEN START
		5.76E-02	R21-DG_-FR-DGB	DIESEL GENERATOR "B" FAILS TO RUN GIVEN START
198	9.48E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		3.69E-03	C12-MP_-FS-C001B	MOTOR-DRIVEN PUMP C001B FAILS TO START
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		1.61E-02	P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
		1.61E-02	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
199	9.33E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED
		1.21E-02	C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026
200	9.33E-14	6.60E-04	%M6U_LOPP	LOSS OF PREF POWER - MODE 6 UNFLOODED

Table 16.6-3

Internal Events Shutdown PRA Cutset Report

Cutsets with Descriptions Report
Internal Events Shutdown
Core Damage Frequency = 8.77E-09
Top 200 Cutsets

#	Cutset Prob	Event Prob	Event	Description
		1.21E-02	C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
		1.50E-04	E50-SQV-CC_ALL	CCF of all components in group 'E50-SQV-CC'
		4.84E-02	G21-BV_-RE-F334	MISPOSITION OF VALVE F334
		1.61E-03	P54-XHE-FO-F026	OPERATOR FAIL TO REOPEN F026

Table 16.6-4

Internal Events Shutdown PRA Importance Measure Report

F-V and RAW Importance Measures Report

(F-V = Fussell-Vesely Importance Measure; RAW = Risk Achievement Worth Importance Measure)

Internal Event Shutdown Full Power

Core Damage Frequency = 8.71E-09

Event Name	Probability	Fus Ves	RAW	Description
DWH-2	1.00E-02	7.51E-01	75.4	FAILURE TO CLOSE DRYWELL HATCH
DWH-1	1.00E-01	2.28E-01	3.06	CLOSE LOWER DRYWELL HATCH
G21-BV_-RE-F334	4.84E-02	1.48E-02	1.29	MISPOSITION OF VALVE F334
E50-UV_OC_ALL	3.00E-04	1.14E-02	38.53	CCF of all components in group 'E50-UV_OC'
G31-XHE-FO-SDC	1.77E-02	6.76E-03	1.37	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
P54-XHE-FO-REOPEN	1.61E-02	6.14E-03	1.37	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
E50-SQV-CC_ALL	1.50E-04	5.65E-03	38.3	CCF of all components in group 'E50-SQV-CC'
XXX-XHE-FO-LPMAKEUP	1.61E-02	4.88E-03	1.29	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
C12-XHE-FO-LEVEL2	3.21E-02	3.35E-03	1.1	Operator fails to back-up CRD actuation
R-PSWS-6	2.18E-01	2.98E-03	1.01	SERVICE WATER RECOVERY
C12-BV_-RE-F021A	1.21E-02	1.22E-03	1.1	MISPOSITION OF VALVE F021A
P21-BV_-RE-F049A	1.21E-02	1.22E-03	1.1	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
P21-BV_-RE-F050A	1.21E-02	1.22E-03	1.1	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
C12-BV_-RE-F021B	1.21E-02	1.22E-03	1.1	MISPOSITION OF VALVE F021B
P21-BV_-RE-F049B	1.21E-02	1.22E-03	1.1	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
P21-BV_-RE-F050B	1.21E-02	1.22E-03	1.1	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
R10-LOSP-EPRI	3.00E-03	1.11E-03	1.34	CONSEQENTIAL LOSS OF PREFERRED OFFSITE POWER DUE TO A TRANSIENT
E50-UV_-OC-F003D	1.75E-02	8.90E-04	1.05	CHECK VALVE F003D FAILS TO REMAIN OPEN OR PLUG
E50-UV_-OC-F003A	1.75E-02	8.74E-04	1.05	CHECK VALVE F003A FAILS TO REMAIN OPEN OR PLUG
E50-UV_-OC-F003E	1.75E-02	8.74E-04	1.05	CHECK VALVE F003E FAILS TO REMAIN OPEN OR PLUG
E50-UV_-OC-F003F	1.75E-02	8.74E-04	1.05	CHECK VALVE F003F FAILS TO REMAIN OPEN OR PLUG
R-M5-G31	2.29E-01	6.81E-04	1	RWCU/SDC RECOVERY
P54-XHE-FO-F026	1.61E-03	5.98E-04	1.36	OPERATOR FAIL TO REOPEN F026
C12-BV_-RE-F013A	4.84E-02	5.62E-04	1.01	MISPOSITION OF VALVE F013A

Table 16.6-4

Internal Events Shutdown PRA Importance Measure Report

F-V and RAW Importance Measures Report

(F-V = Fussell-Vesely Importance Measure; RAW = Risk Achievement Worth Importance Measure)

Internal Event Shutdown Full Power

Core Damage Frequency = 8.71E-09

Event Name	Probability	Fus Ves	RAW	Description
C12-BV_-RE-F013B	4.84E-02	5.62E-04	1.01	MISPOSITION OF VALVE F013B
C12-BV_-RE-F015A	4.84E-02	5.62E-04	1.01	MISPOSITION OF VALVE F015A
C12-BV_-RE-F015B	4.84E-02	5.62E-04	1.01	MISPOSITION OF VALVE F015B
B21-SQV-CC_ALL	1.50E-04	5.40E-04	4.41	CCF of all components in group 'B21-SQV-CC'
P21-ACV-OO-F0016_1_2	1.93E-04	5.37E-04	3.76	CCF of two components: P21-ACV-OO-F016A & P21-ACV-OO-F016B
C12-MOV-CC-F020A	4.00E-03	4.00E-04	1.1	MOTOR OPER. VALVE F020A FAILS TO OPEN
C12-MOV-CC-F020B	4.00E-03	4.00E-04	1.1	MOTOR OPER. VALVE F020B FAILS TO OPEN
C12-MP_-FS-C001B	3.69E-03	3.67E-04	1.1	MOTOR-DRIVEN PUMP C001B FAILS TO START
C62-CCFSOFTWARE	1.00E-04	3.03E-04	4	Common cause failure of software
C12-SYS-TM-TRAINB	3.00E-03	2.99E-04	1.1	TRAIN B IN MAINTENANCE
C62-CCFSOFTWARE_S	1.00E-04	2.78E-04	3.76	Common cause failure of software, for spurious
C12-BV_-RE-F065	4.84E-02	2.65E-04	1.01	MISPOSITION OF LOCKED OPEN VALVE F065
E50-UV_OC_1_4_5	7.05E-06	2.54E-04	36.18	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003D & E50-UV_-OC-F003E
E50-UV_OC_1_4_6	7.05E-06	2.54E-04	36.18	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003D & E50-UV_-OC-F003F
E50-UV_OC_1_5_6	7.05E-06	2.54E-04	36.18	CCF of three components: E50-UV_-OC-F003A & E50-UV_-OC-F003E & E50-UV_-OC-F003F
E50-UV_OC_4_5_6	7.05E-06	2.54E-04	36.18	CCF of three components: E50-UV_-OC-F003D & E50-UV_-OC-F003E & E50-UV_-OC-F003F
C12-BV_-RE-F064	4.84E-02	2.42E-04	1	MISPOSITION OF VALVE F064
C12-MP_-FS-C001BOIL	2.40E-03	2.39E-04	1.1	MOTOR-DRIVEN AUX. OIL PUMP FOR C001B FAILS TO START
MS-TOP2	5.00E-02	2.39E-04	1	TWO DPVs FAIL TO OPEN
R-M6-G31	2.18E-01	2.04E-04	1	FAILURE TO RECOVER RWCU/SDC
R21-DG_-FR-DGA	5.76E-02	1.96E-04	1	DIESEL GENERATOR "A" FAILS TO RUN GIVEN START
R21-DG_-FR-DGB	5.76E-02	1.60E-04	1	DIESEL GENERATOR "B" FAILS TO RUN GIVEN START
E50-SQV-CC-F002D	3.00E-03	1.44E-04	1.04	SQUIB VALVE F002D FAILS TO OPERATE
E50-SQV-CC-F002A	3.00E-03	1.42E-04	1.04	SQUIB VALVE F002A FAILS TO OPERATE
E50-SQV-CC-F002E	3.00E-03	1.42E-04	1.04	SQUIB VALVE F002E FAILS TO OPERATE

Table 16.6-4

Internal Events Shutdown PRA Importance Measure Report

F-V and RAW Importance Measures Report

(F-V = Fussell-Vesely Importance Measure; RAW = Risk Achievement Worth Importance Measure)

Internal Event Shutdown Full Power

Core Damage Frequency = 8.71E-09

Event Name	Probability	Fus Ves	RAW	Description
E50-SQV-CC-F002F	3.00E-03	1.42E-04	1.04	SQUIB VALVE F002F FAILS TO OPERATE
C12-BV_-RE-F003A	1.21E-02	1.36E-04	1.01	MISPOSITION OF VALVE FOO3A
C12-BV_-RE-F003B	1.21E-02	1.36E-04	1.01	MISPOSITION OF VALVE F003B
R21-DG_-TM-DGA	4.60E-02	1.34E-04	1	STANDBY DIESEL GENERATOR "A" IN MAINTENANCE
C63-UNDEVSPUR58	1.00E-03	1.33E-04	1.12	Undeveloped spurious hardware failure
C63-UNDEVSPUR59	1.00E-03	1.33E-04	1.12	Undeveloped spurious hardware failure
C63-UNDEVSPUR62	1.00E-03	1.33E-04	1.12	Undeveloped spurious hardware failure
C63-UNDEVSPUR63	1.00E-03	1.33E-04	1.12	Undeveloped spurious hardware failure
C63-UNDEVSPUR66	1.00E-03	1.33E-04	1.12	Undeveloped spurious hardware failure
C63-UNDEVSPUR67	1.00E-03	1.33E-04	1.12	Undeveloped spurious hardware failure
C63-UNDEVSPUR70	1.00E-03	1.33E-04	1.12	Undeveloped spurious hardware failure
C63-UNDEVSPUR71	1.00E-03	1.33E-04	1.12	Undeveloped spurious hardware failure
R-PSWS-5	2.18E-01	1.17E-04	1	FAILURE TO RECOVER SERVICE WATER
U43-XHE-FO-LPCI	1.61E-03	1.16E-04	1.07	OPERATOR FAILS TO ACTUATE U43 IN LPCI MODE
R21-DG_-TM-DGB	4.60E-02	1.06E-04	1	STANDBY DIESEL GENERATOR "B" IN MAINTENANCE
C62-UNDEVSPUR5	1.00E-03	9.75E-05	1.1	Undeveloped spurious hardware failure
C62-UNDEVSPUR7	1.00E-03	9.75E-05	1.1	Undeveloped spurious hardware failure
C63-UNDEVSPUR126	1.00E-03	9.75E-05	1.1	Undeveloped spurious hardware failure
C63-UNDEVSPUR127	1.00E-03	9.75E-05	1.1	Undeveloped spurious hardware failure
C63-CCFSOFTWARE_S	1.00E-04	9.09E-05	1.89	Common cause failure of software, for spurious
P41-SYS-FC-HVACPSW-A	1.00E-03	5.07E-05	1.05	PSW-A ROOM COOLING FAILURE
P41-SYS-FC-HVACPSW-B	1.00E-03	5.07E-05	1.05	PSW-B ROOM COOLING FAILURE
R21-DG_-FS-DGA	1.40E-02	4.50E-05	1	DG-A FAILS TO START AND LOAD
C12-MOV-CC-F014A	4.00E-03	4.32E-05	1.01	MOTOR OPER. VALVE F014A FAILS TO OPEN
C12-MOV-CC-F014B	4.00E-03	4.32E-05	1.01	MOTOR OPER. VALVE F014B FAILS TO OPEN

Table 16.6-4

Internal Events Shutdown PRA Importance Measure Report

F-V and RAW Importance Measures Report

(F-V = Fussell-Vesely Importance Measure; RAW = Risk Achievement Worth Importance Measure)

Internal Event Shutdown Full Power

Core Damage Frequency = 8.71E-09

Event Name	Probability	Fus Ves	RAW	Description
P21-AHU-FS-RCCWA	6.00E-03	4.18E-05	1.01	AIR HANDLING UNIT RCCWS ROOM A FAILS TO START
R21-DG_-FR-CCF_1_2	4.54E-03	4.04E-05	1.01	CCF of two components: R21-DG_-FR-DGA & R21-DG_-FR-DGB
P21-MOV-CC-F034B	4.00E-03	3.84E-05	1.01	MOV P21-F034B FROM RCCWS TO RWCU/SDC HX-B FAILS TO OPEN
P52-CMP-FS-C001B	2.00E-02	3.81E-05	1	MOTOR-DRIVEN AIR COMPRESS. C001B FAILS TO START
P41-TRN-RE-PUMP2B	8.07E-03	3.74E-05	1	FAILURE TO RESTORE PSW PUMP 2B
P41-TRN-RE-PUMP2A	8.07E-03	3.73E-05	1	FAILURE TO RESTORE PSW PUMP 2A
P21-AHU-FS-RCCWB	6.00E-03	3.71E-05	1.01	AIR HANDLING UNIT RCCWS ROOM B FAILS TO START
R21-DG_-FS-DGB	1.40E-02	3.64E-05	1	DG-B FAILS TO START AND LOAD
C62-UNDEVSPUR97	1.00E-03	3.46E-05	1.03	Undeveloped spurious hardware failure
P21-AHU-FR_1_2	1.26E-05	3.43E-05	3.66	CCF of two components: P21-AHU-FR-RCCWA & P21-AHU-FR-RCCWB
C62-UNDEVSPUR99	1.00E-03	3.35E-05	1.03	Undeveloped spurious hardware failure
G31-ACV-CC-F019A	2.00E-03	3.27E-05	1.01	DEMINERALIZER BYPASS VALVE F019A FAILS TO OPEN
P41-FAN-FR_ALL	1.20E-05	3.26E-05	3.65	CCF of all components in group 'P41-FAN-FR'
E50-SQV-CC_1_4	2.38E-05	3.19E-05	2.25	CCF of two components: E50-SQV-CC-F002A & E50-SQV-CC-F002D
E50-SQV-CC_1_5	2.38E-05	3.19E-05	2.25	CCF of two components: E50-SQV-CC-F002A & E50-SQV-CC-F002E
E50-SQV-CC_1_6	2.38E-05	3.19E-05	2.25	CCF of two components: E50-SQV-CC-F002A & E50-SQV-CC-F002F
E50-SQV-CC_4_5	2.38E-05	3.19E-05	2.25	CCF of two components: E50-SQV-CC-F002D & E50-SQV-CC-F002E
E50-SQV-CC_4_6	2.38E-05	3.19E-05	2.25	CCF of two components: E50-SQV-CC-F002D & E50-SQV-CC-F002F
E50-SQV-CC_5_6	2.38E-05	3.19E-05	2.25	CCF of two components: E50-SQV-CC-F002E & E50-SQV-CC-F002F
G31-NST-TM-B	9.00E-03	3.18E-05	1	RWCU/SDCS TRAIN B IN MAINTENANCE OR OUT OF SERVICE
R-PSWS	2.29E-01	2.93E-05	1	SERVICE WATER RECOVERY
P21-ACV-OO-F0004	2.00E-03	2.89E-05	1.01	AIR OPERATED VALVE F0004 FAILS TO CLOSE
P21-ACV-OO-F0007	2.00E-03	2.89E-05	1.01	AIR OPERATED VALVE F0007 FAILS TO CLOSE
P21-ACV-OO-F0020	2.00E-03	2.89E-05	1.01	AIR OPERATED VALVE F0020 FAILS TO CLOSE
P21-ACV-OO-F0027	2.00E-03	2.89E-05	1.01	AIR OPERATED VALVE F0027 FAILS TO CLOSE

Table 16.6-5

Shutdown PRA Core Damage Sequences

Event Tree / Initiating Events	Mode	Core damage sequences
Loss of PSWS	5	M5-LPSW-6 M5-LPSW-7 M5-LPSW-10 M5-LPSW-11 M5-LPSW-12
Loss of RWCU/SDC	5	M5-LRWC-9 M5-LRWC-10 M5-LRWC-14 M5-LRWC-15 M5-LRWC-16
Loss of Pref Power	5	M5-LOP-9 M5-LOP-10 M5-LOP-14 M5-LOP-15 M5-LOP-16
Loss of PSWS	5 open	M5O-LPSW-6 M5O-LPSW-7 M5O-LPSW-10 M5O-LPSW-11 M5O-LPSW-12
Loss of RWCU/SDC	5 open	M5O-LRWC-9 M5O-LRWC-10 M5O-LRWC-14 M5O-LRWC-15 M5O-LRWC-16
Loss of Pref Power	5 open	M5O-LOP-9 M5O-LOP-10 M5O-LOP-14 M5O-LOP-15 M5O-LOP-16
Loss of PSWS	6 Unflooded	M6U-LPSW-4
Loss of RWCU/SDC	6 Unflooded	M6U-G31-6
Loss of Pref Power	6 Unflooded	M6U-LOP-6
LOCA - RWCU/SDC below TAF	5	M5-LOCA-R5 M5-LOCA-R6
LOCA - Other than FW or GDCS	5	M5-LOCA-OT5
LOCA - FW Line A	5	M5-LOCA-F3
LOCA - Instrument Line below TAF	5	M5-LOCA-I5 M5-LOCA-I6
LOCA - GDCS	5	M5-LOCA-G5
LOCA - RWCU/SDC below TAF	5 open	M5O-LOCA-R5

Table 16.6-5

Shutdown PRA Core Damage Sequences

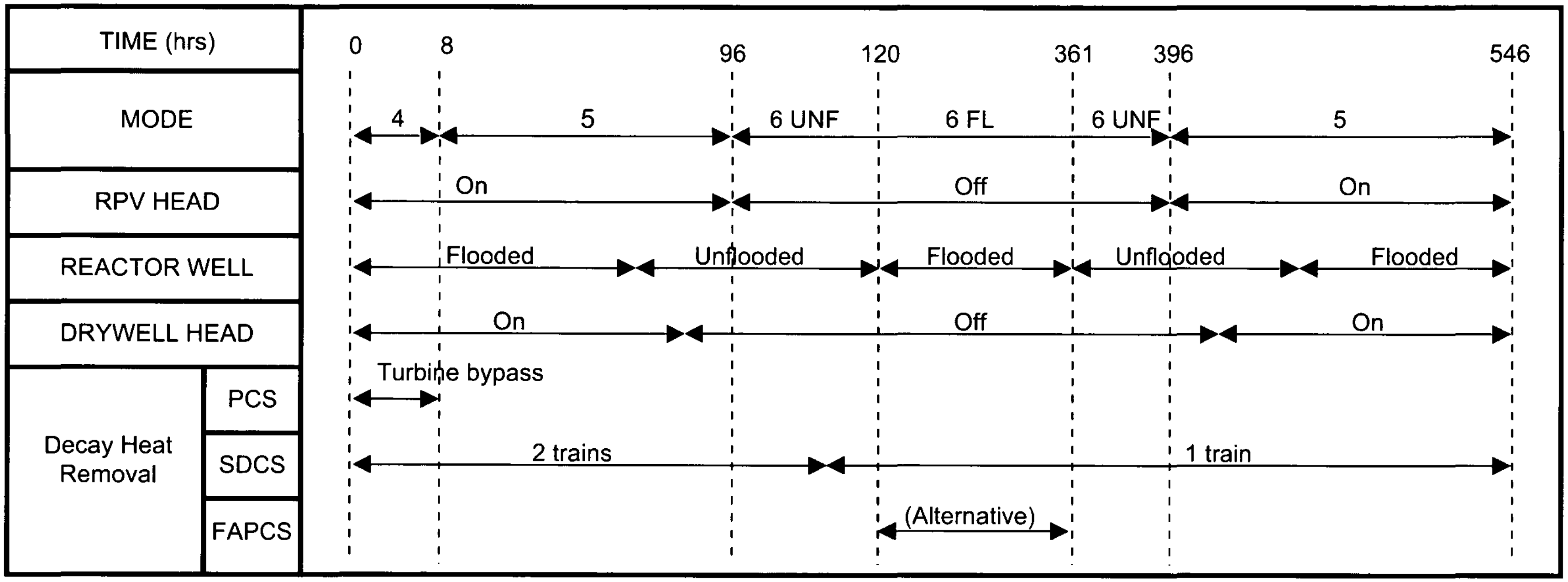
Event Tree / Initiating Events	Mode	Core damage sequences
		M5O-LOCA-R6
LOCA - Other than FW or GDCS	5 open	M5O-LOCA-OT5
LOCA - FW Line A	5 open	M5O-LOCA-F3
LOCA - Instrument Line below TAF	5 open	M5O-LOCA-I5 M5O-LOCA-I6
LOCA - GDCS	5 open	M5O-LOCA-G5
LOCA - RWCU/SDC below TAF	6 Unflooded	M6U-LOCA-R5 M6U-LOCA-R6
LOCA - Other than FW or GDCS	6 Unflooded	M6U-LOCA-OT5
LOCA - FW Line A	6 Unflooded	M6U-LOCA-F3
LOCA - Instrument Line below TAF	6 Unflooded	M6U-LOCA-I5 M6U-LOCA-I6
LOCA - GDCS	6 Unflooded	M6U-LOCA-G5
LOCA - Instrument Line below TAF	6 Flooded	M6F-LOCA-I2
LOCA - RWCU/SDC below TAF	6 Flooded	M6F-LOCA-R2

Table 16.6-6

Shutdown PRA Core Damage Sequences

#	Probability	% of CDF	Sequence	Cutsets	
1	2.93E-09	33.40%	M6F-LOCA-I2	%M6F_LOCA_I	DWH-2
2	2.35E-09	60.20%	M5-LOCA-I6	%M5-LOCA-I	DWH-2
3	8.92E-10	70.40%	M6F-LOCA-R2	%M6F_LOCA_R	DWH-1
4	7.33E-10	78.70%	M6U-LCOA-I6	%M6U_LOCA-I	DWH-2
5	7.13E-10	86.90%	M5-LOCA-R6	%M5-LOCA-RW	DWH-1
6	5.86E-10	93.50%	M5O-LOCA-I6	%M5O_LOCA_I	DWH-2
7	2.23E-10	96.10%	M6U-LCOA-R6	%M6U_LOCA-RW	DWH-1
8	1.78E-10	98.10%	M5O-LOCA-R6	%M5O_LOCA_R	DWH-1

Note: All the dominant cutsets in the internal events shutdown model are from LOCA events below the TAF. All these sequences have only one cutset; the initiating event combined with failure to close the lower drywell hatch.



Note: Node 5 Open is assumed to occur the last 24 hours of Mode 5 before refueling (hours 72-96), and the first 24 hours of Mode 5 following refueling (hours 396-420).

Figure 16.2-1. ESBWR Refuel Outage Plan for Shutdown PRA

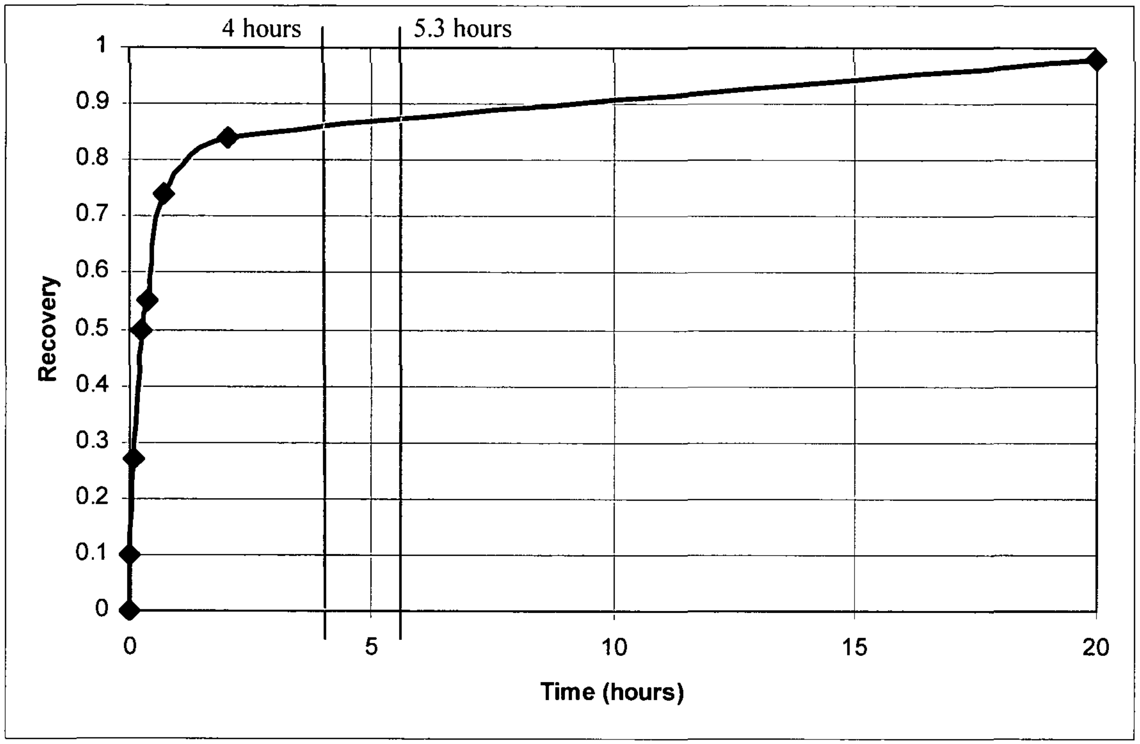


Figure 16.3-1. RWCU/SDCS Recovery Probability (Cumulative)

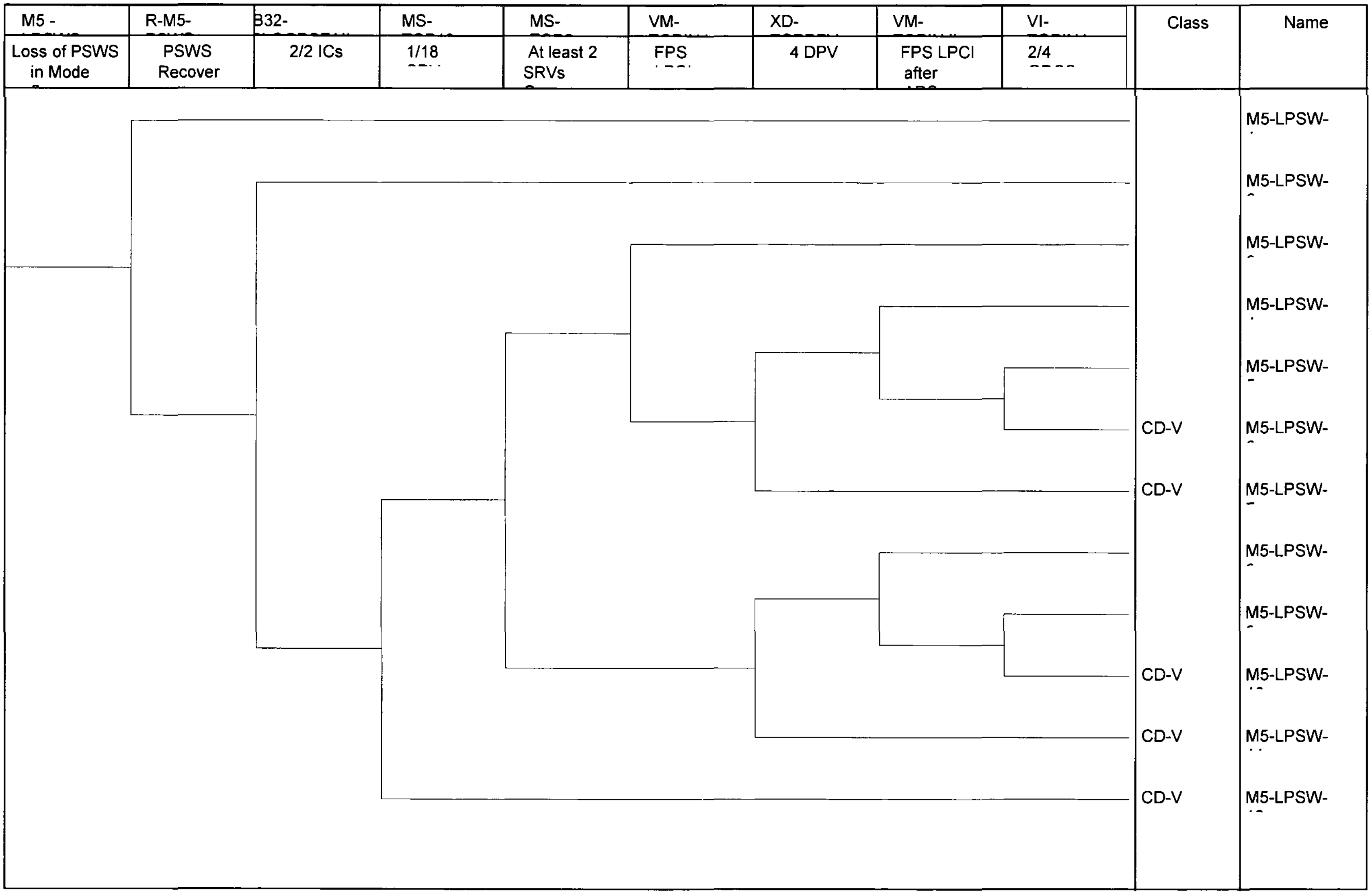


Figure 16.4-1. Loss of all Service Water PSWS/RCCWS (Mode 5)

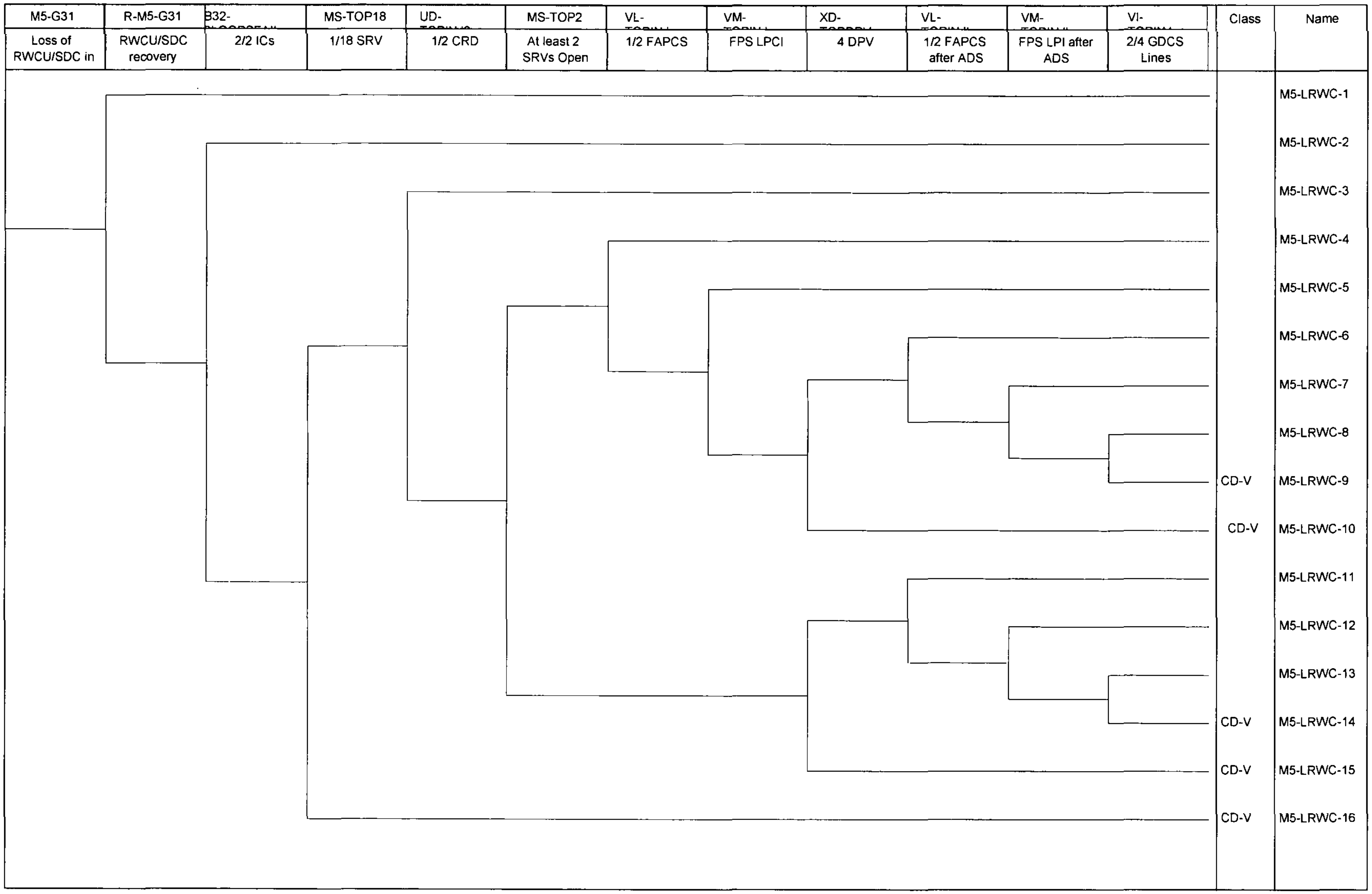


Figure 16.4-2. Loss of Both RWCU/SDCS Trains (Mode 5)

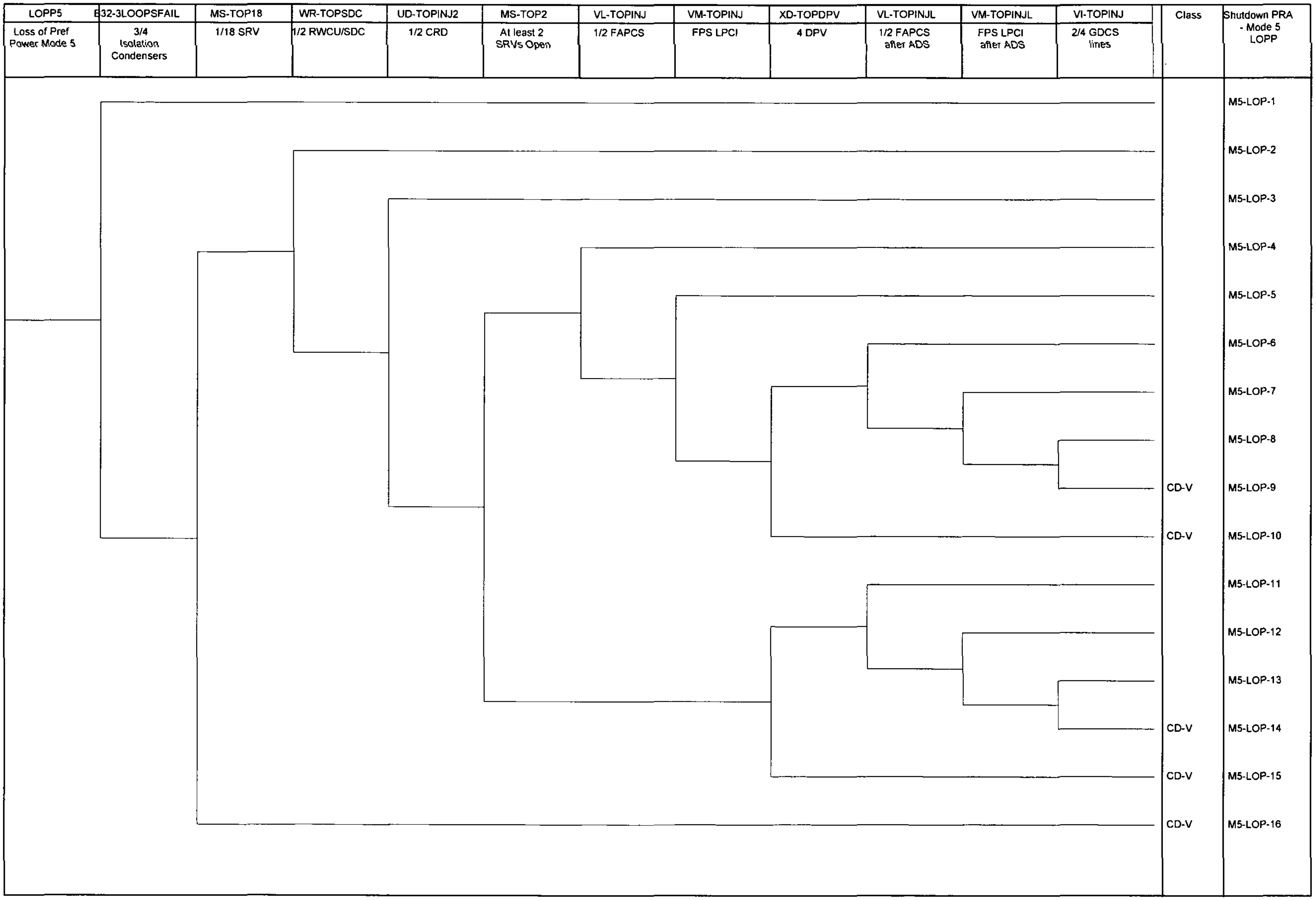


Figure 16.4-3. Loss of Preferred Power (Mode 5)

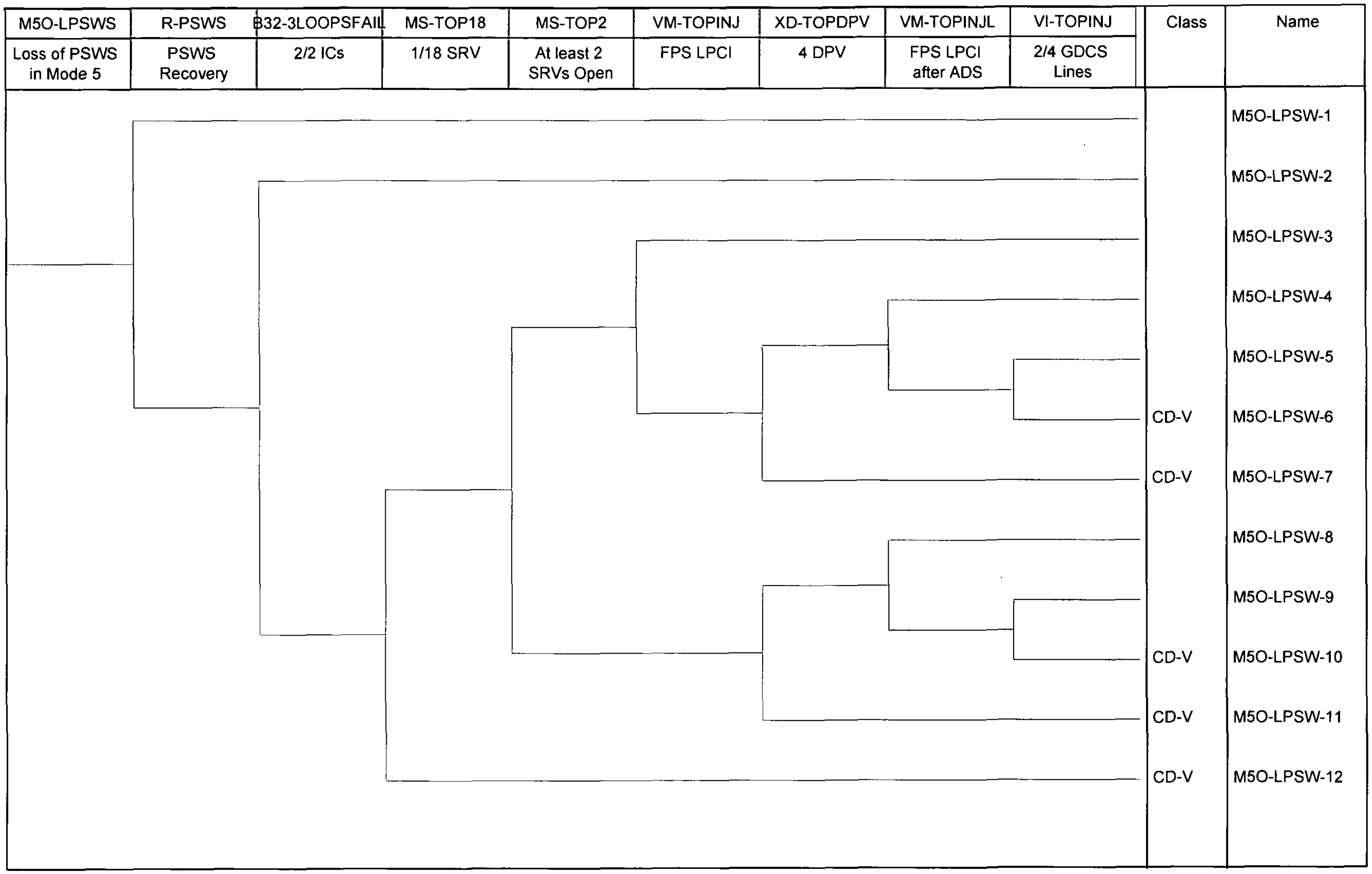
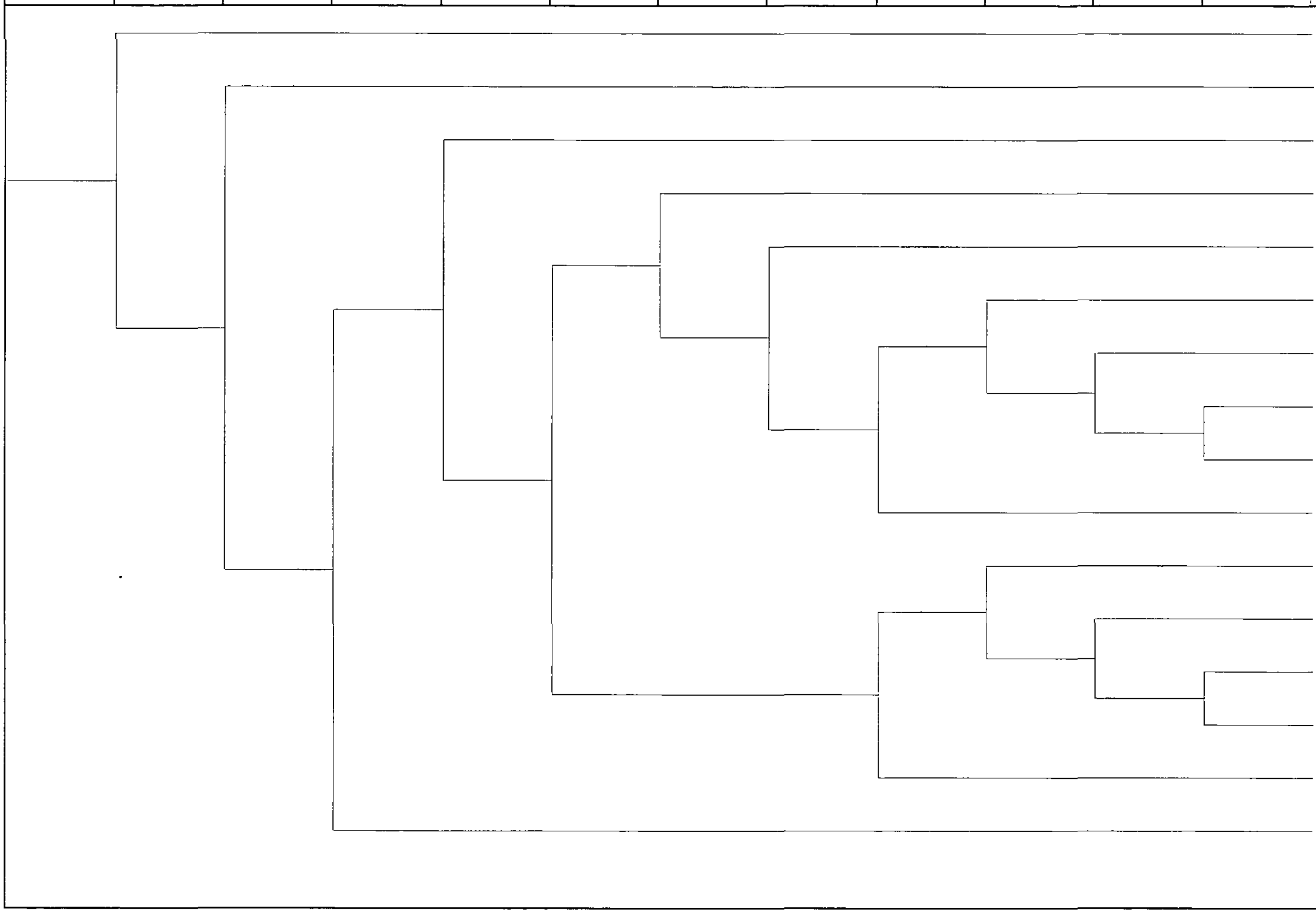


Figure 16.4-4. Loss of all Service Water PSWS/RCCWS (Mode 5 Open)

M5O-G31	R-M5-G31	B32-3LOOPSFAIL	MS-TOP18	UD-TOPINJ2	MS-TOP2	VL-TOPINJ	VM-TOPINJ	XD-TOPDPV	VL-TOPINJL	VM-TOPINJL	VI-TOPINJ
Loss of RWCU/SDC in	RWCU/SDC recovery	2/2 ICs	1/18 SRV	1/2 CRD	At least 2 SRVs Open	1/2 FAPCS	FPS LPCI	4 DPV	1/2 FAPCS after ADS	FPS LPCI after ADS	2/4 GDCS Lines



Class	Name
	M5O-LRWC-1
	M5O-LRWC-2
	M5O-LRWC-3
	M5O-LRWC-4
	M5O-LRWC-5
	M5O-LRWC-6
	M5O-LRWC-7
	M5O-LRWC-8
CD-V	M5O-LRWC-9
CD-V	M5O-LRWC-10
	M5O-LRWC-11
	M5O-LRWC-12
	M5O-LRWC-13
CD-V	M5O-LRWC-14
CD-V	M5O-LRWC-15
CD-V	M5O-LRWC-16

Figure 16.4-5. Loss of Both RWCU/SDCS Trains (Mode 5 Open)

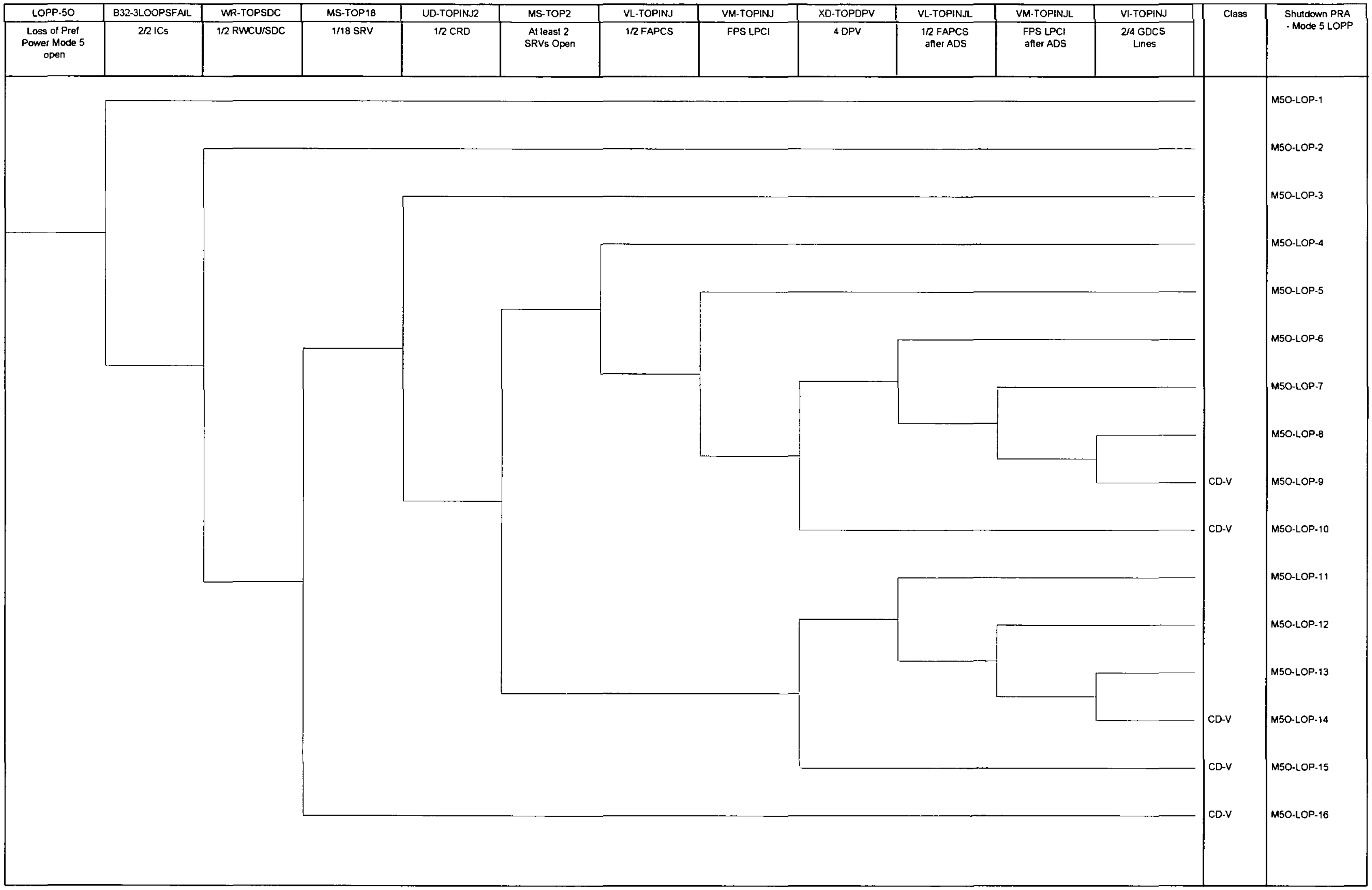


Figure 16.4-6. Loss of Preferred Power (Mode 5 Open)

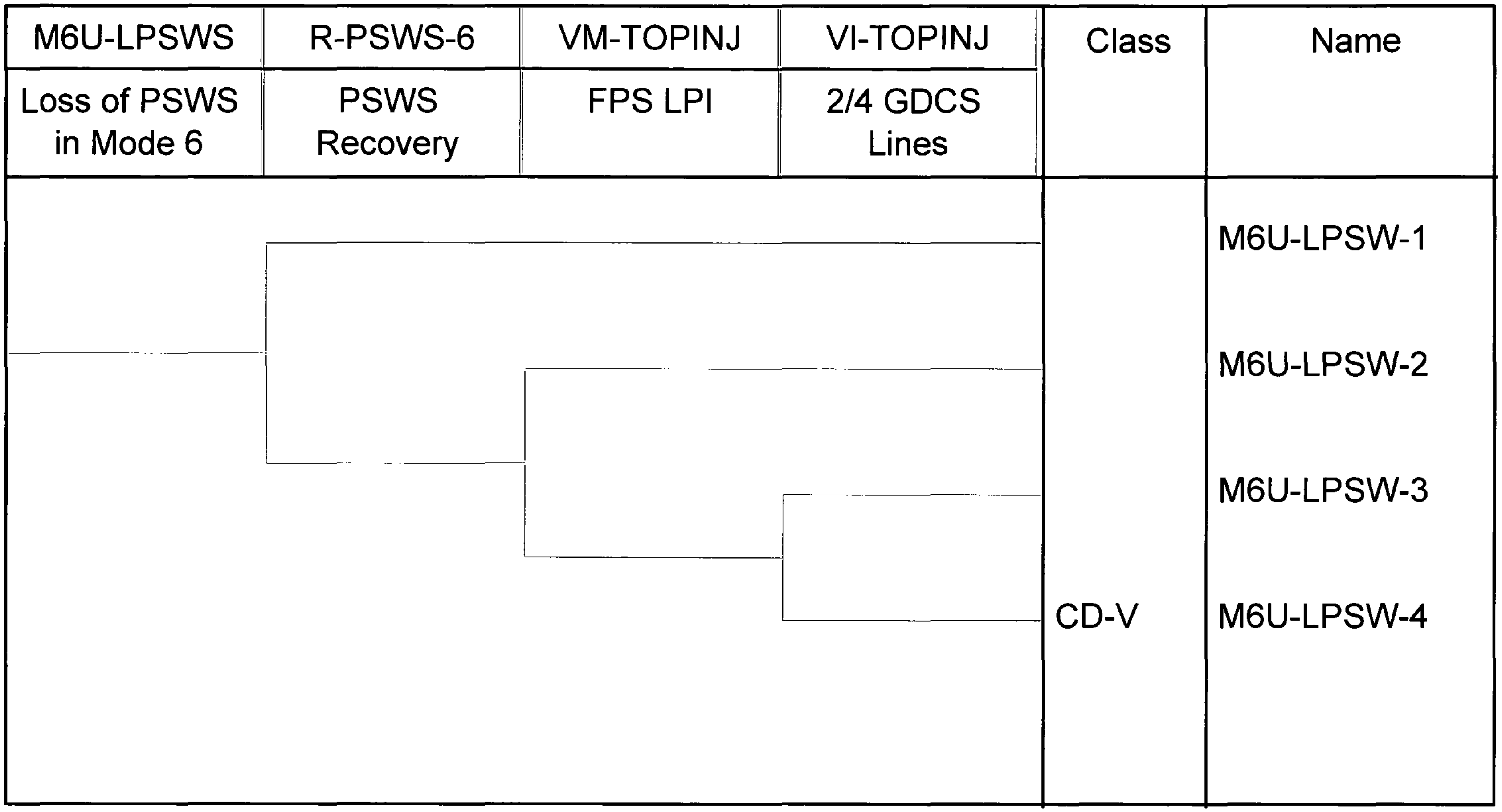


Figure 16.4-7. Loss of all Service Water PSWS/RCCWS (Mode 6 Unflooded)

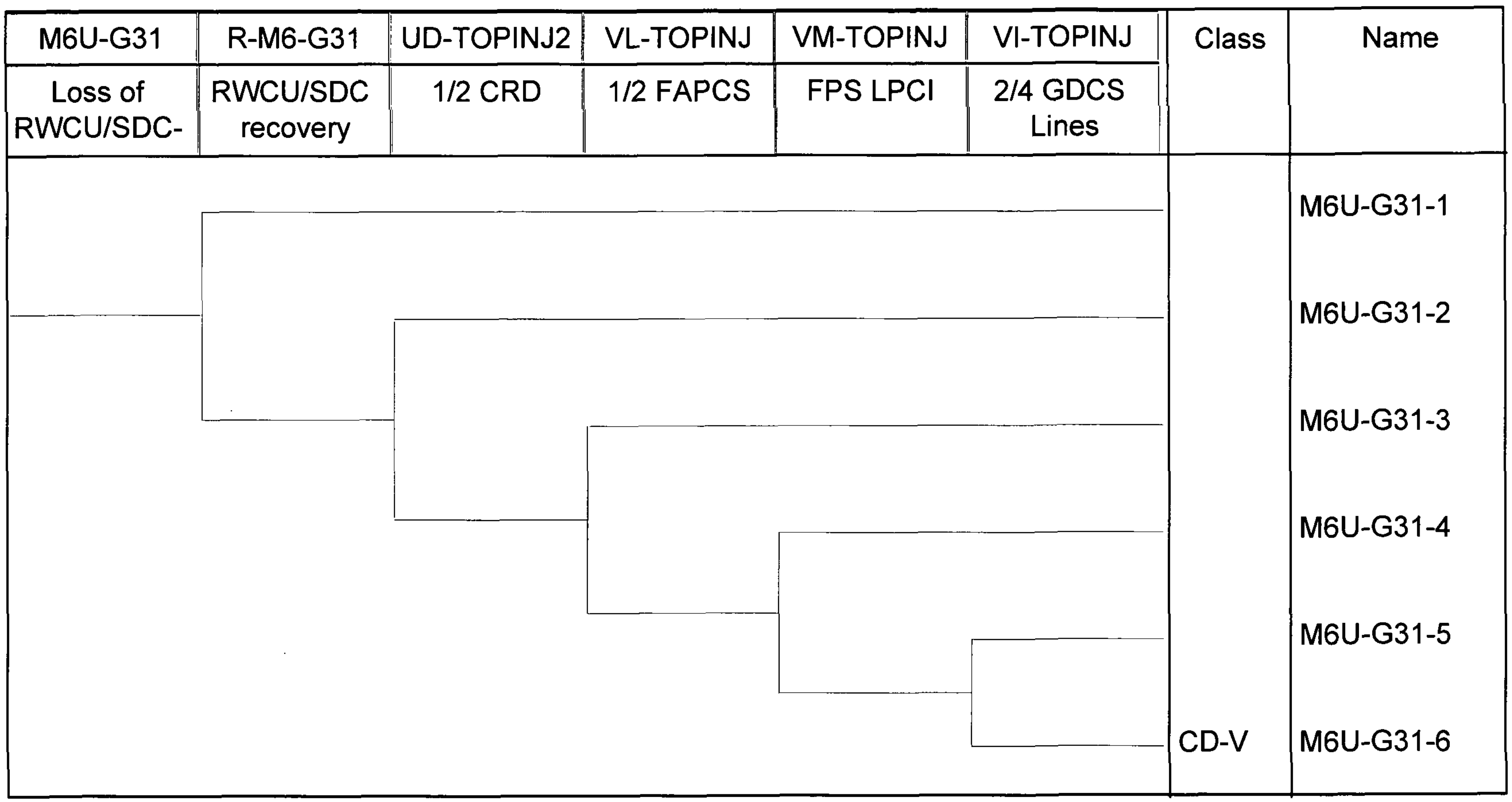


Figure 16.4-8. Loss of Both RWCU/SDCS Trains (Mode 6 Unflooded)

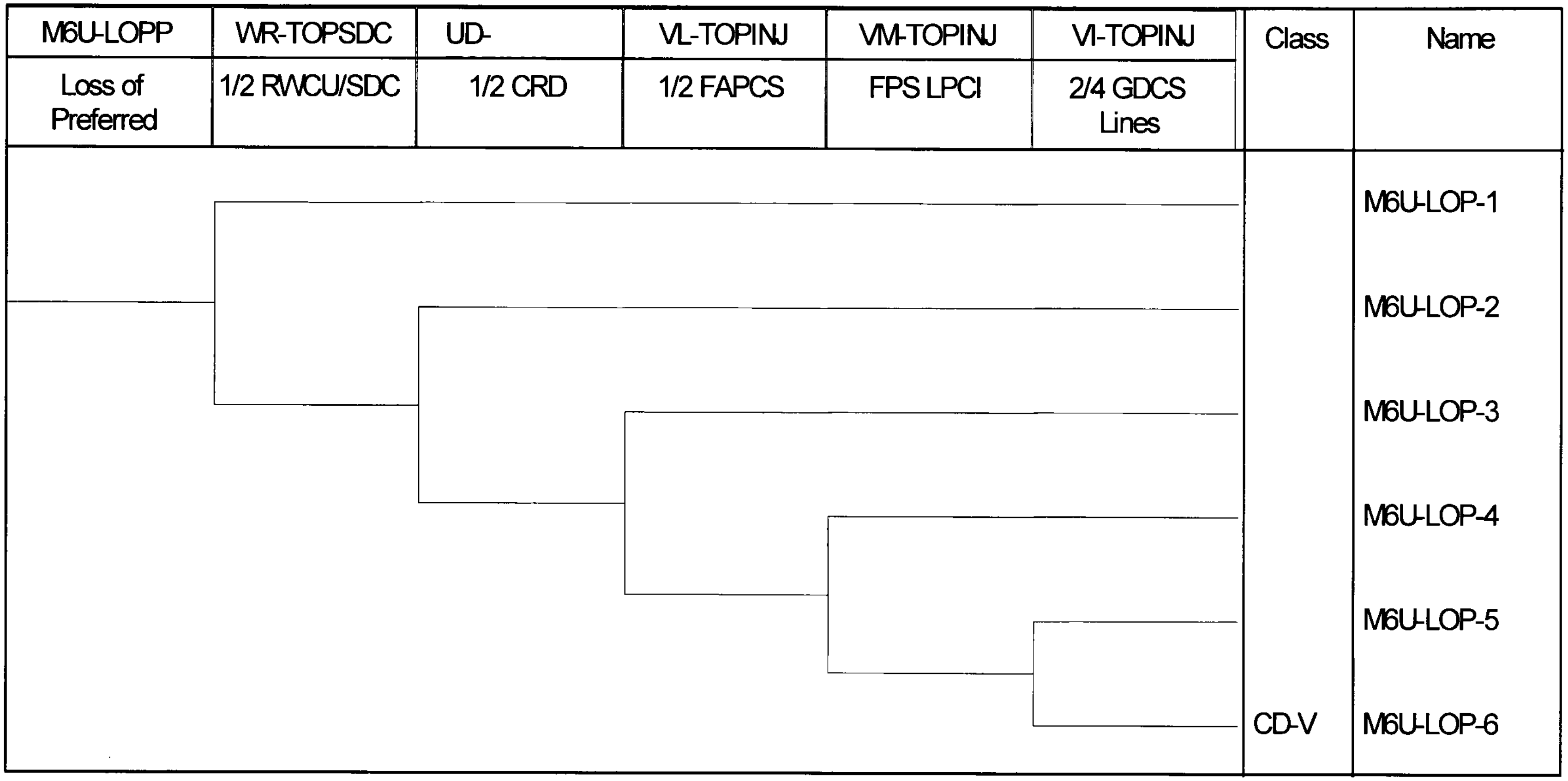


Figure 16.4-9. Loss of Preferred Power (Mode 6 Unflooded)

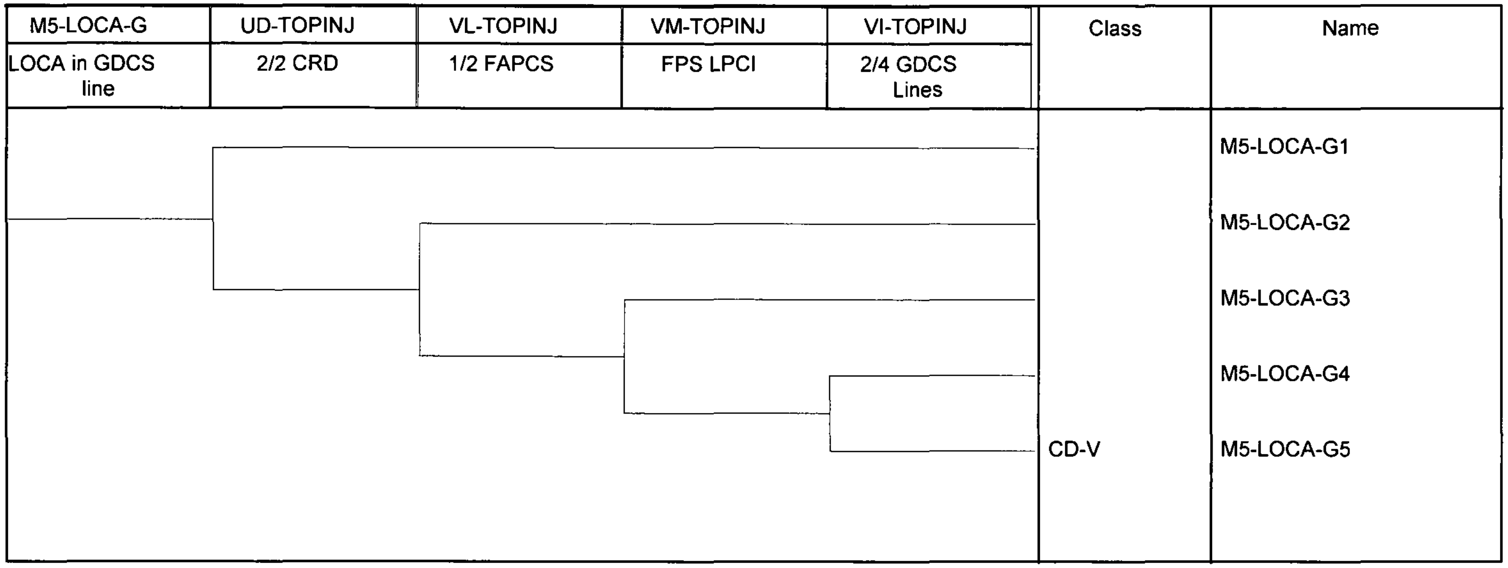


Figure 16.4-10. LOCA in GDCS line (Mode 5)

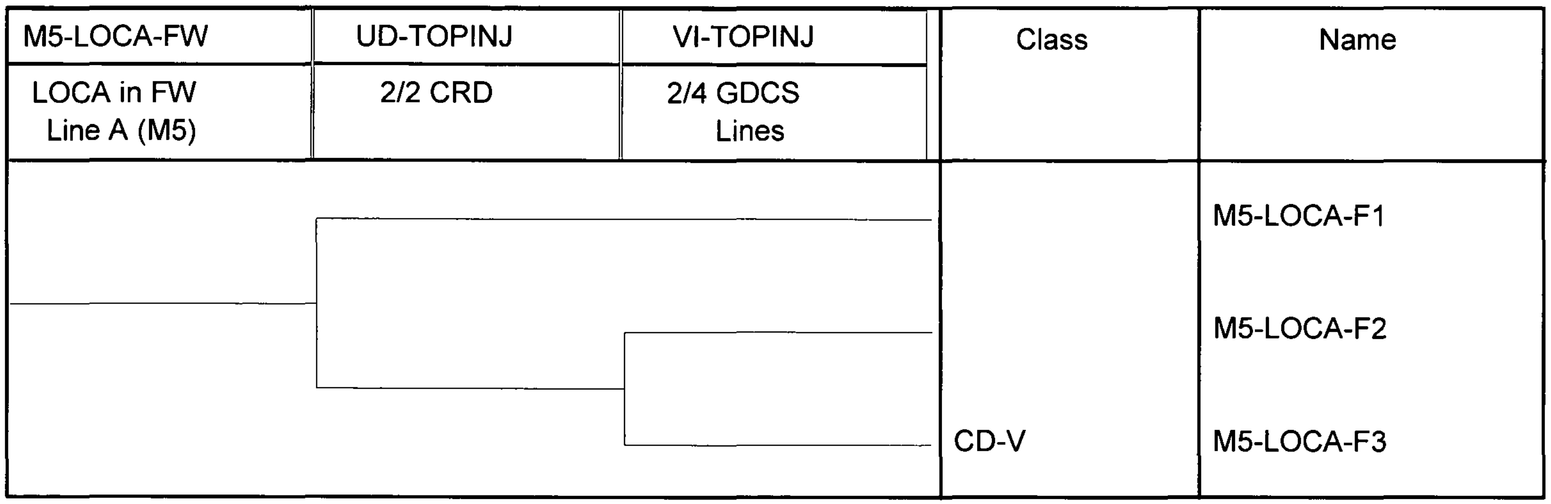


Figure 16.4-11. LOCA in Feedwater line (Mode 5)

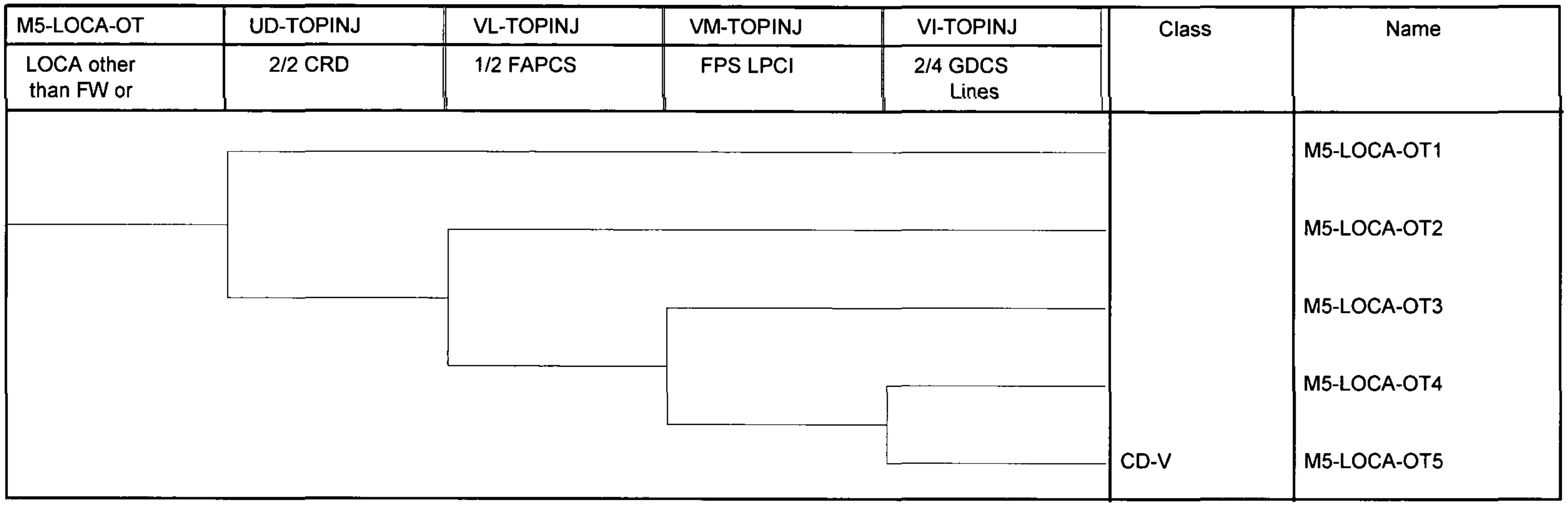


Figure 16.4-12. LOCA other than FW or GDCS (Mode 5)

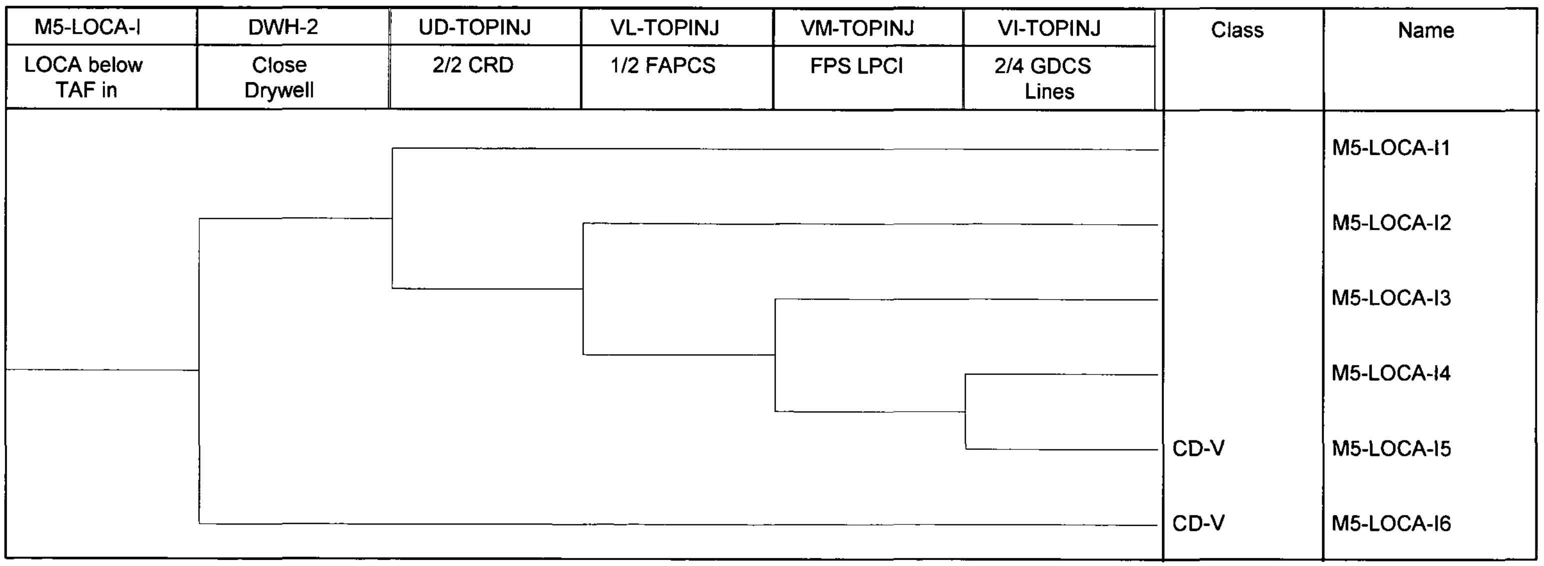


Figure 16.4-13. LOCA below TAF in Instrument line (Mode 5)

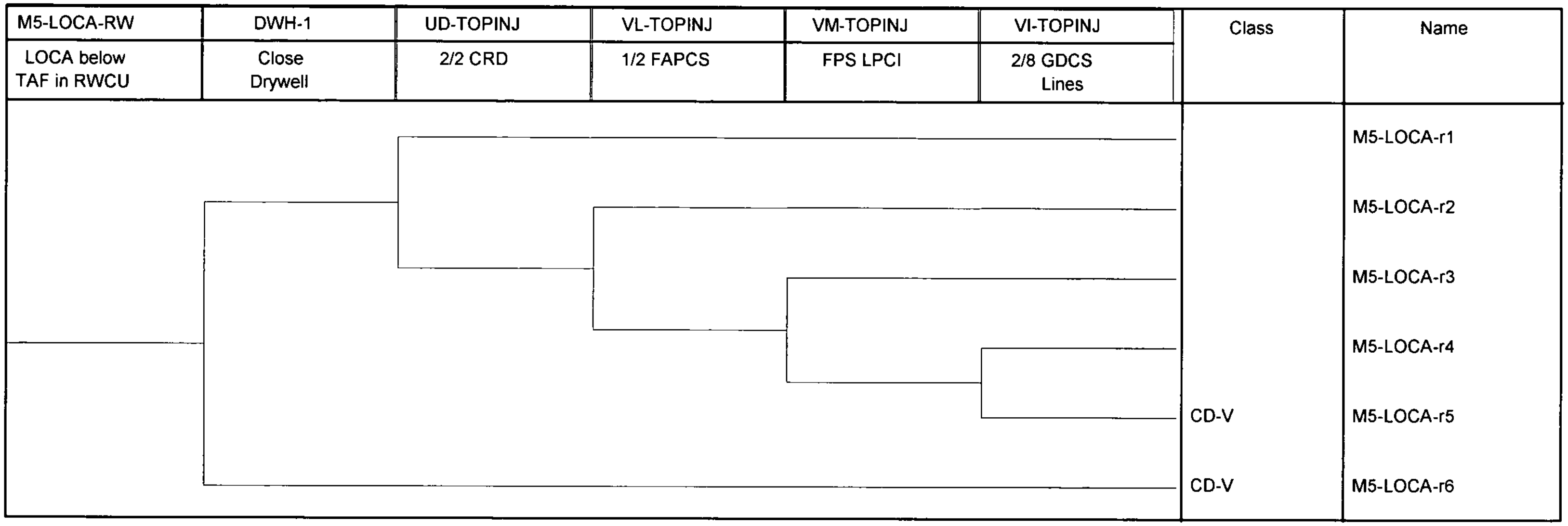


Figure 16.4-14. LOCA below TAF in RWCU drain line (Mode 5)

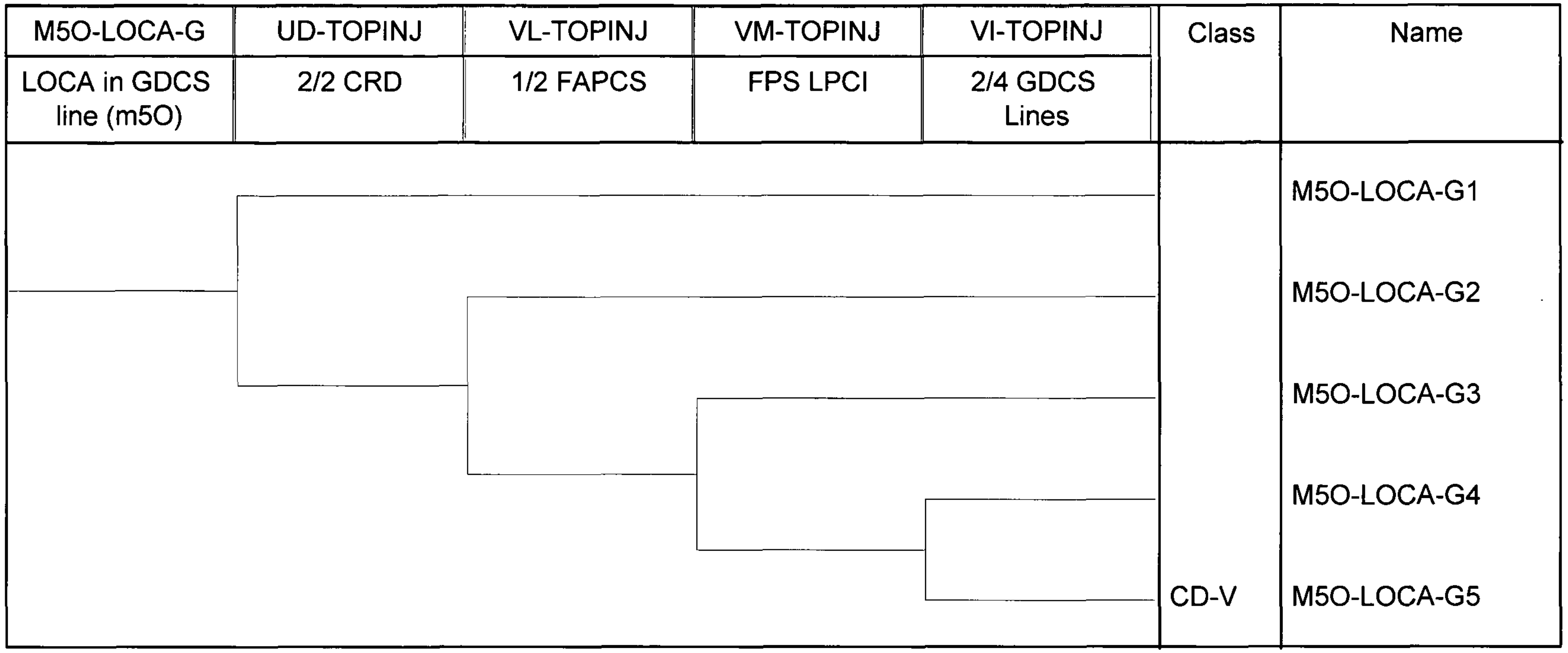


Figure 16.4-15. LOCA in GDCS line (Mode 5 Open)

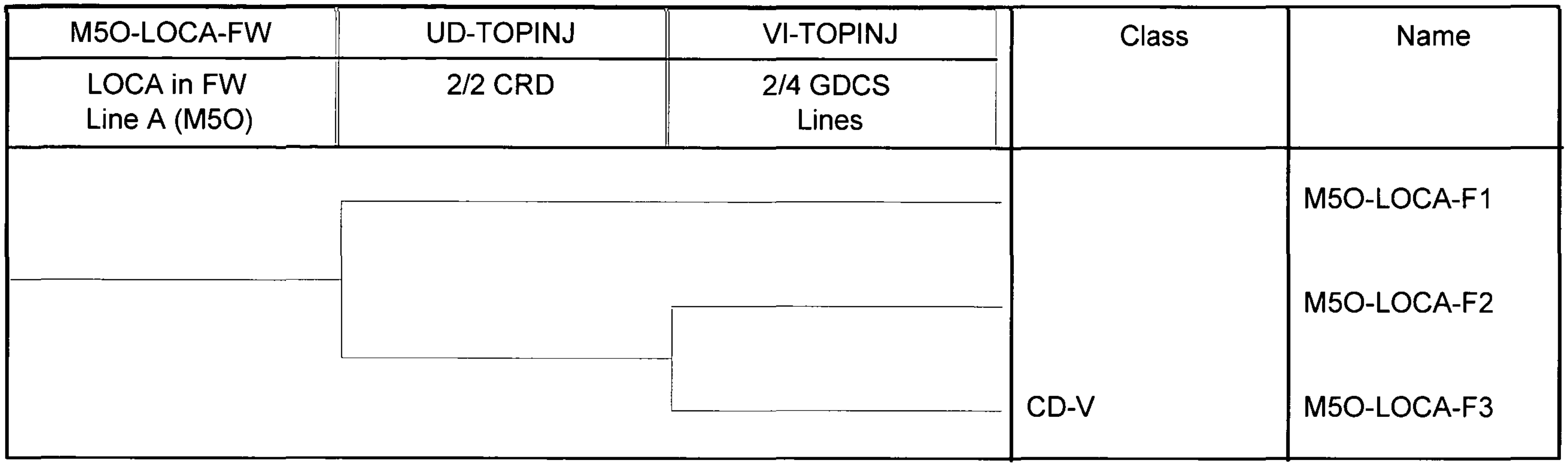


Figure 16.4-16. LOCA in Feedwater line (Mode 5 Open)

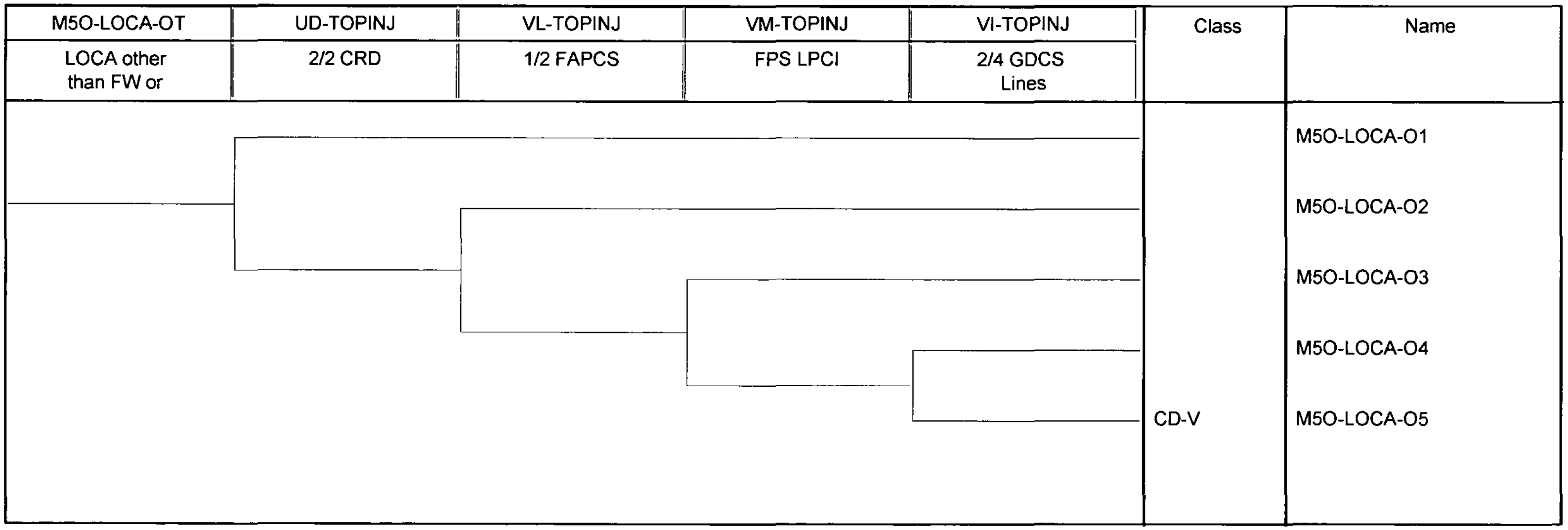


Figure 16.4-17. LOCA in line other than Feedwater or GDCS (Mode 5 Open)

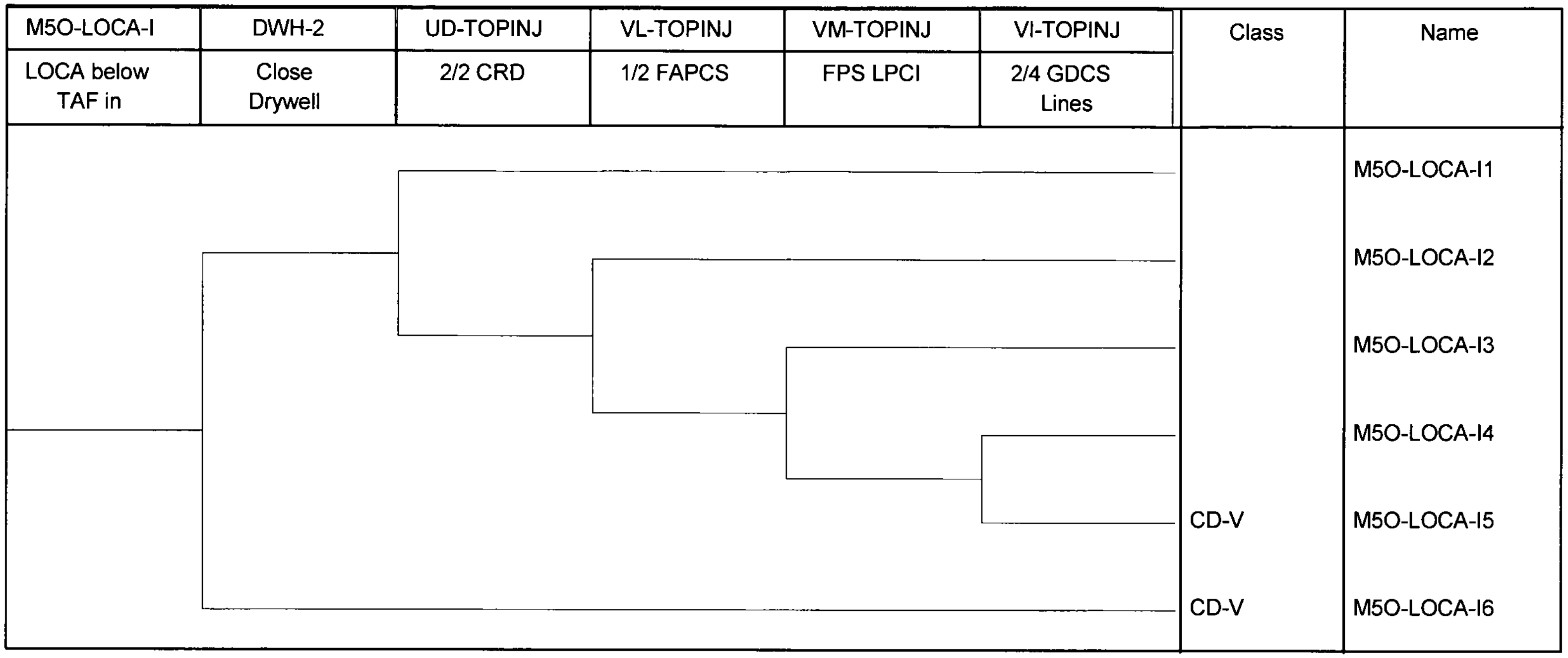


Figure 16.4-18. LOCA below TAF in Instrument line (Mode 5 Open)

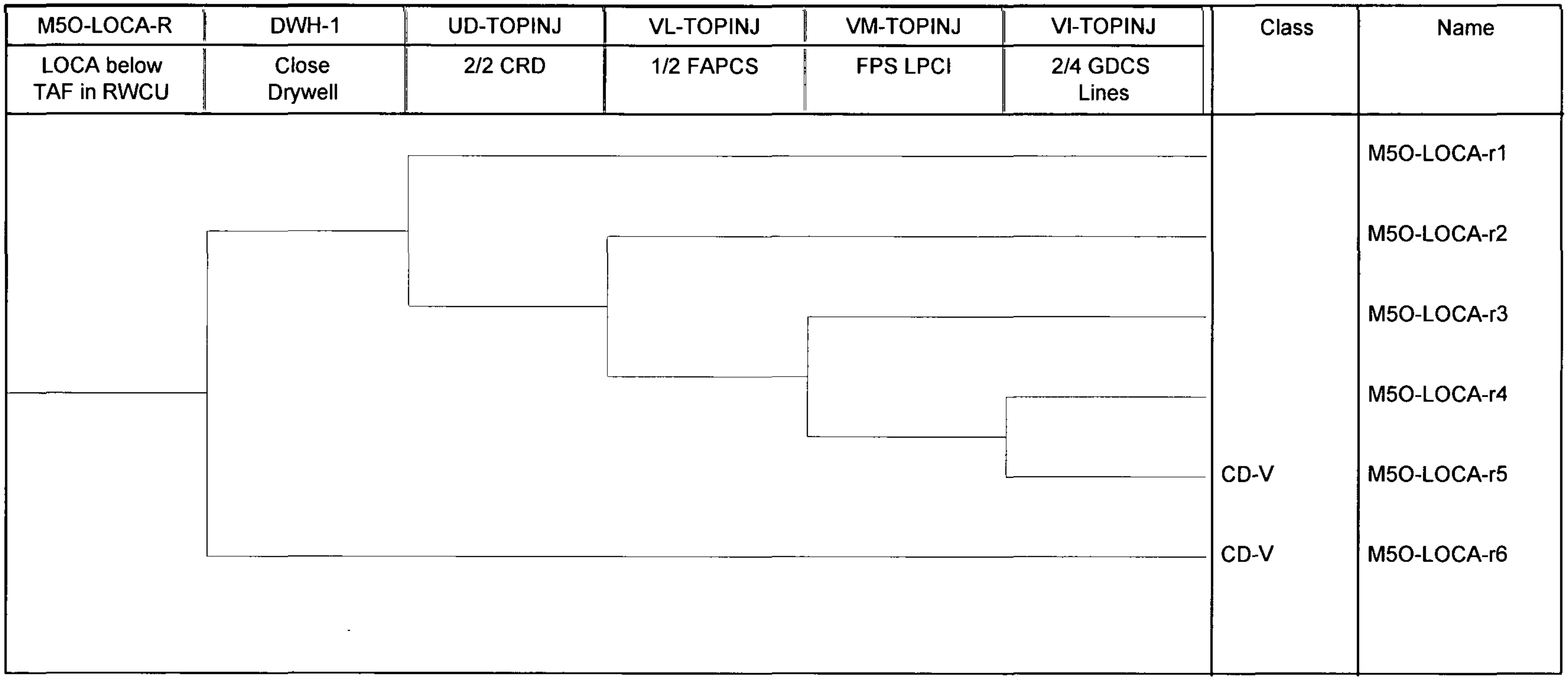


Figure 16.4-19. LOCA below TAF in RWCU drain line (Mode 5 Open)

M6U-LOCA-G	UD-TOPINJ	VL-TOPINJ	VM-TOPINJ	VI-TOPINJ	Class	Name
LOCA in GDCS line	2/2 CRD	1/2 FAPCS	FPS LPCI	2/4 GDCS Lines		
						M6U-LOCA-G1
						M6U-LOCA-G2
						M6U-LOCA-G3
						M6U-LOCA-G4
					CD-V	M6U-LOCA-G5

Figure 16.4-20. LOCA in GDCS line (Mode 6 Unflooded)

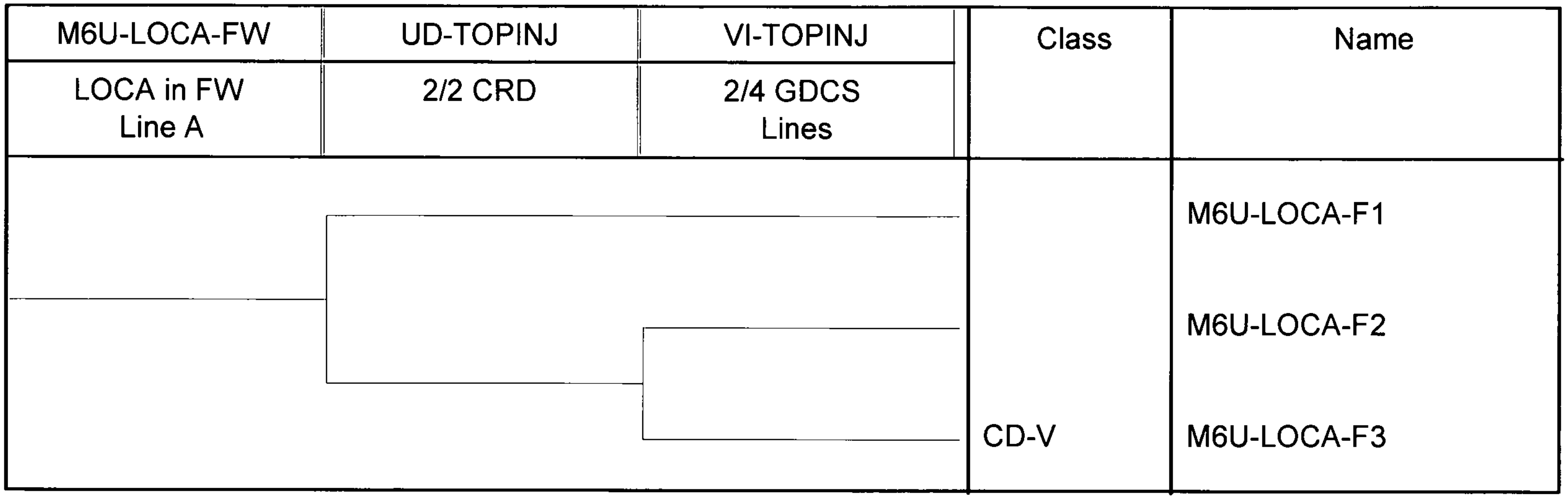


Figure 16.4-21. LOCA in Feedwater line (Mode 6 Unflooded)

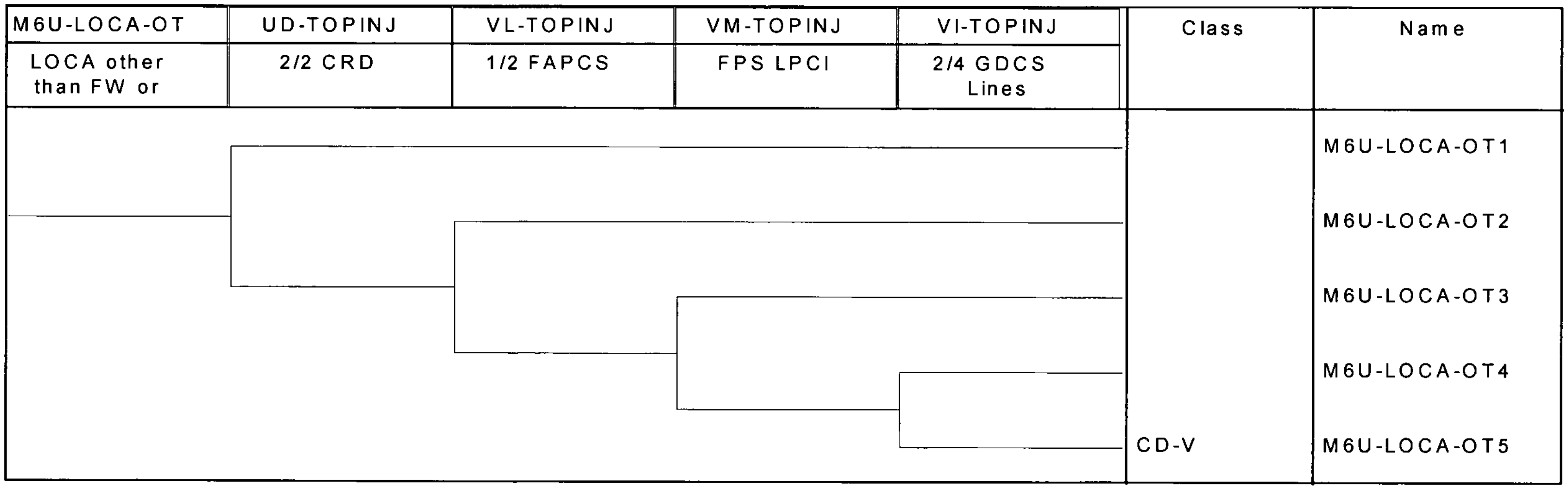


Figure 16.4-22. LOCA in line other than Feedwater or GDCS (Mode 6 Unflooded)

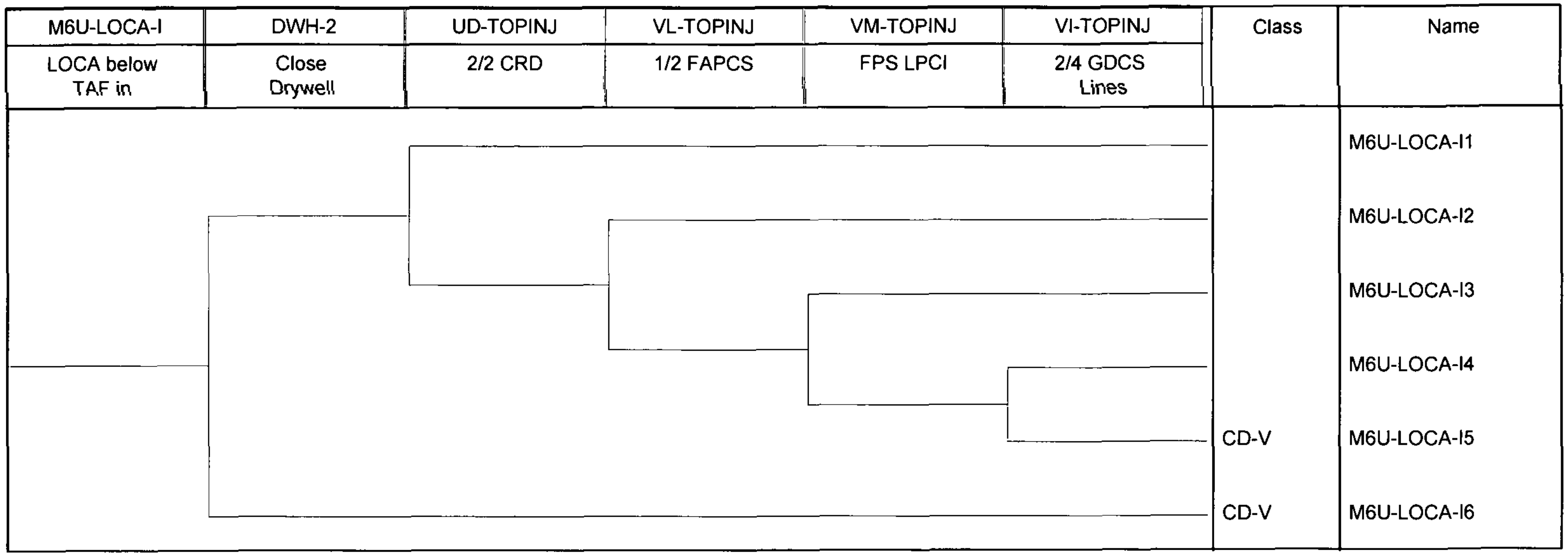


Figure 16.4-23. LOCA below TAF in Instrument line (Mode 6 Unflooded)

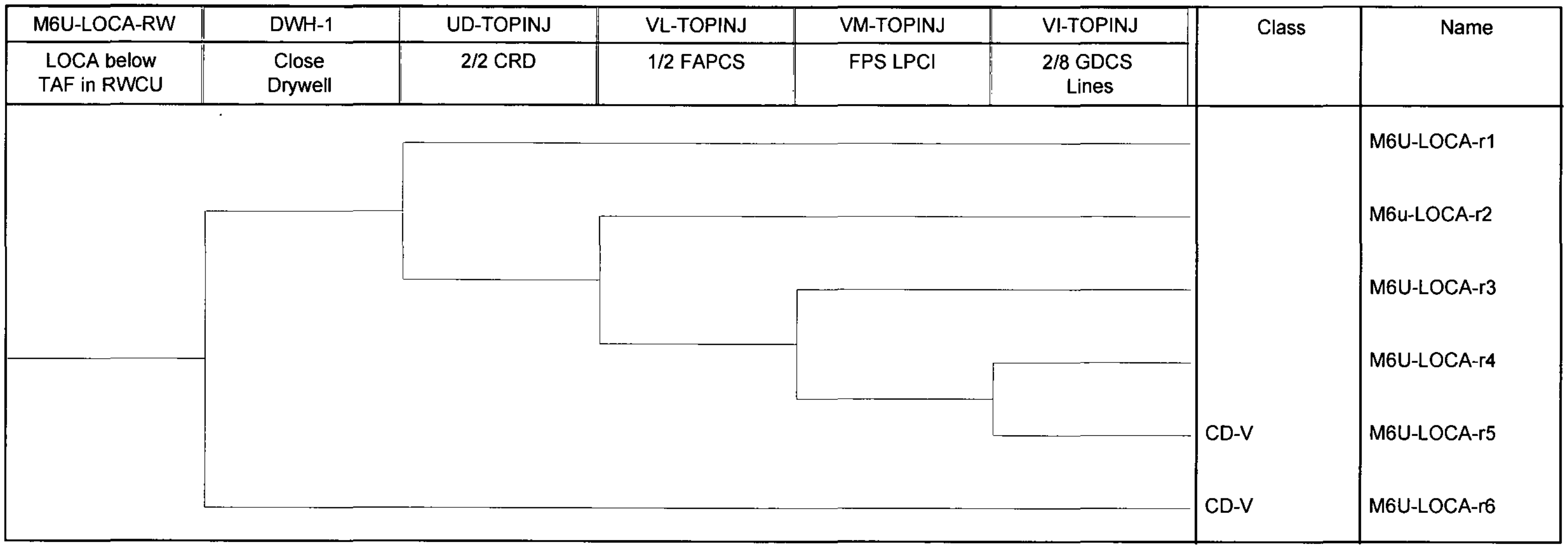


Figure 16.4-24. LOCA below TAF in RWCU drain line (Mode 6 Unflooded)

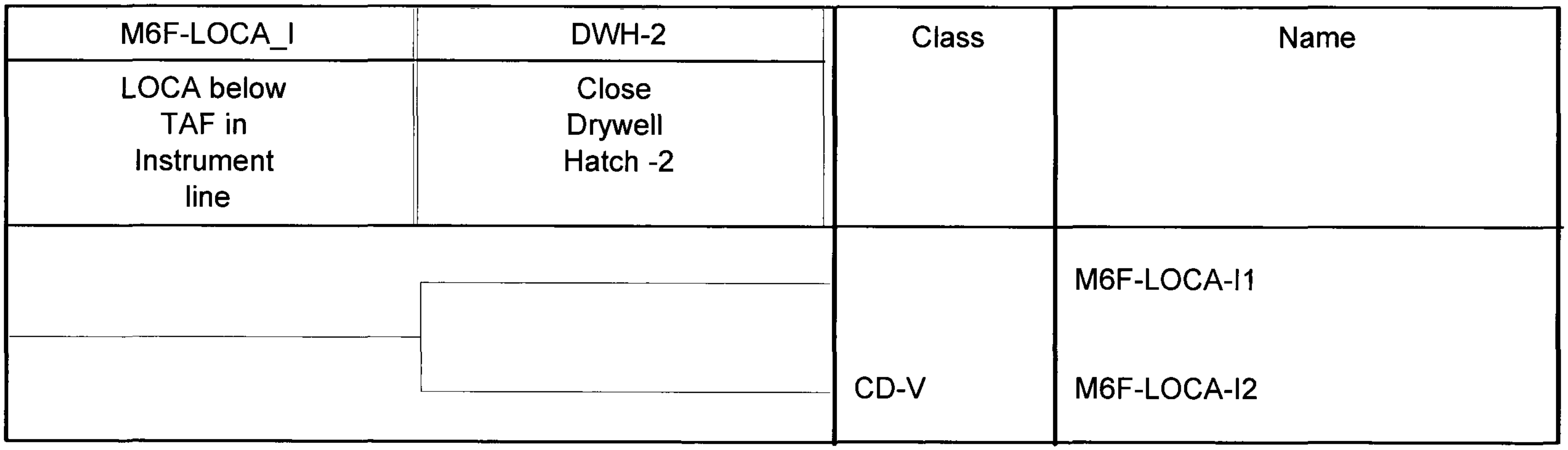


Figure 16.4-25. LOCA below TAF in Instrument line (Mode 6 Flooded)

M6F-LOCA_R	DWH-1	Class	Name
LOCA below TAF in RWCU/SDC drainlines	Close Drywell Hatch -1		
			M6F-LOCA-R1
		CD-V	M6F-LOCA-R2

Figure 16.4-26. LOCA below TAF in RWCU drain line (Mode 6 Flooded)

Figure 16.3-2. "Plant-Centered" LOPP Recovery Probability (Cumulative) - (Deleted)

Figure 16.3-3. "External" LOOP Recovery Probability (Cumulative) - (Deleted)

Figure 16.4-1a. Loss of Both RWCU/SDCS Trains (Mode 5) RWCU5A - (Deleted)

Figure 16.4-1b. Loss of Both RWCU/SDCS Trains (Mode 5) RWCU5B - (Deleted)

Figure 16.4-2. Loss of Both RWCU/SDCS Trains (Mode 6-Unflooded) RWCU6 - (Deleted)

Figure 16.4-3a. Loss of Preferred Power (Mode 5) LOPP5A - (Deleted)

Figure 16.4-3b. Loss of Preferred Power (Mode 5) LOPP5B - (Deleted)

Figure 16.4-4. Loss of Preferred Power (Mode 6-Unflooded) LOPP6 - (Deleted)

Figure 16.4-5a. Loss of RCCWS/PSWS (Mode 5) SW5A - (Deleted)

Figure 16.4-5b. Loss of RCCWS/PSWS (Mode 5) SW5B - (Deleted)

Figure 16.4-6. Loss of RCCWS/PSWS (Mode 6-Unflooded) SW6 - (Deleted)

Figure 16.4-7. LOCA in GDCS Line (Mode 6-Unflooded) LGDCS - (Deleted)

Figure 16.4-8. LOCA in FW-A (Mode 6-Unflooded) LFWA - (Deleted)

Figure 16.4-9. LOCA Other Than FW or GDCS (Mode 6-Unflooded) LOTHER - (Deleted)

Figure 16.4-10. LOCA Below TAF in RWCU Drain Lines (Mode 6-Unflooded) LBTAF1 - (Deleted)

Figure 16.4-11. LOCA Below TAF in Instrument Lines (Mode 6-Unflooded) LBTAF2 - (Deleted)

Figure 16.4-12. LOCA Below TAF in RWCU/SDC Drain Lines (Mode 6-Flooded) LBTAF1F - (Deleted)

Figure 16.4-13. LOCA Below TAF in Instrument Lines (Mode 6-Flooded) LBTAF2F - (Deleted)

Attachment 2 to Enclosure 1 of MFN 07-237, Supplement 1

NEDO-33201, Revision 2 List of Changes

Chapter 8 From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
1.	S8, entire section	Change Containment System Event Trees (CSET) and Containment Phenomenology System Event Trees (CPET) to Containment Event Trees (CET)
2.	S8.0, 1st para	Technical clarification: containment isolation is not passive, neither is suppression pool cooling (a form of "heat removal")
3.	S8.0, 2nd para	Added sentence in response to RAI 19.2-69 and 19.2-70
4.	S8.0, 4th para	Revised containment fragility analysis is now in DCD S19B and 19C
5.	T8.0-1, 4th row	The rev 2 Level 2 model no longer uses separate Containment Phenomenology Event Trees (CPET) and Containment System Event Trees (CET), instead there is one integrated Containment Event Tree (CET) for each class
6.	T8.0-1, last row	These release categories were not discussed in section 8 previously
7.	S8.1.3	for clarification, because the referred to analysis is only valid for inerted operation
8.	S8.1.4	In response to RAI 19.2-69 and RAI 19.2-70 – the potential effect of a severe accident during de-inerted operation has been considered and assigned to the BYP release category
9.	S8.2, 1st para	Technical clarification regarding the completely revised Level 2 model
10	S8.2, 2nd para and bullet lists	New explanation of the phenomenological release categories. In NEDO-33201 Revision 2, Section 8 extends from just a containment systems discussion to an analysis of containment response to severe accident conditions
11	S8.2.1, 1st para	Technical clarification because the Section 21 analysis still uses separate CPET/CSET pairs – however, the two methods are logically equivalent
12	S8.2.1, table	New frequencies for each accident subclass from the Level 1 analysis
13	S8.2.1, 3rd para	Technical clarity – removal of CPET discussion because of model change
14	S8.2.1. bullet list	Sentence removed because Class II accident sequences are no longer ignored

Chapter 8 From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
15	S8.2.1, 5th and 6th para	Technical clarification to be consistent with the new CETs versus the old CPETs/CSETs model
16	S8.2.1.1, 1st para	Containment failure due to phenomenology is approximately 65% of the non-TSL release frequency, plus CPET/CSET quantification is no longer used
17	S8.2.1.1, 2nd para	Technical explanation and basis of the new Level 1 sequence binning process
18	S8.2.1.1, 3rd para	New technical explanation of ATWS severe accident assumptions regarding low RPV pressure at vessel rupture
19	S8.2.1.2, 3rd para	Technical clarification regarding why ICS is not credited
20	S8.2.1.2, 4th para	Technical design change – the temperature sensors in the DW airspace are new additions to the design to help prevent spurious actuation
21	S8.2.1.2, 6th para	Technical clarification – only PCCS is credited in the deterministic analysis, but both PCCS and suppression pool cooling are credited in the probabilistic analysis
22	S8.2.1.2, 7th para	Technical clarification – PCCS does not have any support systems in the traditional sense, but it DOES require leak-tight vacuum breakers
23	S8.2.1.2, 8th para	New technical explanation of the Drywell Spray system, which is not used as a mitigating system in ESBWR
24	S8.2.1.2, 9th para	Technical clarification on the vacuum breaker success criteria
25	S8.2.1.2, 10th para	Venting is not used as a mitigating system in core-concrete interaction sequences because scrubbing is negligible for source term reduction
26	S8.2.1.3, 1st para	Revised technical discussion to reflect updated integrated CETs
27	S8.2.1.3, 2nd para	Technical clarification – explanation of implicit initiator impact as a result of the integrated Level 1 / Level 2 model
28	S8.2.1.3.2, (1)	Deleted. The new vacuum breaker back-up valves no longer require manual actuation upon leak detection – the isolation logic is completely automatic
29	S8.2.1.3.3	New discussion on EVE
30	S8.2.1.3.3, 2nd para	Technical clarification of the postulated RPV failure mode

Chapter 8 From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
		to indicate the high level of conservatism utilized in the current analysis
31	S8.2.1.3.4	New discussion of top event DCH
32	S8.2.1.3.5	New discussion of top event BI_SP, which is represented by a new fault tree analysis
33	S8.2.1.3.6	New discussion of top event BI_FN
34	S8.2.1.3.7	Revised technical discussion of the containment isolation function – DCD R3 contained more detailed information regarding containment penetrations
35	S8.2.1.3.8	Revised technical discussion of top event VB – clarified to confirm that event VB represents the vapor suppression function, not the leak tight function necessary for PCCS operation
36	S8.2.1.3.9, 1st para, 6th sent	Revised thermal-hydraulic analyses indicate that PCCS pool make-up is not necessary until beyond 72 hours when the ICS is not functioning
37	S8.2.1.3.9, 1st para, 8th sent	Technical clarification
38	S8.2.1.4, 1st para 4th sent	Revised technical results based on updated R2 results
39	S8.2.1.4, 2nd – 5th para	New technical discussion regarding the phenomenological release categories / containment failure modes
40	S8.2.1.4, 6th –11th para	Updated technical discussion based on new R2 results
41	S8.2.1.5, 1st para	Updated technical discussion to reflect new quantification method and R2 results
42	S8.2.1.5, 2nd para	New calculation of containment effectiveness based on R2 results and a new truncation of 1E-15 versus 1E-12 from R1
43	T8.2-1	Table updated to reflect new sequence bins and their respective frequencies out of the Level 1 results
44	T8.2-2	Table updated to include ALL release categories, not just the systemic releases as in R1. New results for all release categories included
45		

Chapter 8 From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
46	F8.2-1	Deleted. With the integration of the CPET/CSET structures there is no longer a standard CET architecture.
47	F8.2-2	Figure updated with revised data from MAAP run
48	F8.2-3	Figure updated with revised data from MAAP run
49	F8.2-4	Figure updated with revised data from MAAP run
50	S8.3, 1st para	Technical clarification because phenomenological failure modes are discussed two sentences later
51	S8.3, 3rd para, 3rd sent	Revised to refer to the DCD sections instead of Appendix 8B because the analysis is in the DCD and 8B has been deleted
52	S8.3.2	Technical clarification, since class V sequences are not analyzed
53	S8.3.2, 2nd para	Technical clarification to update the Level 1 PRA results and incorporate the Level 1 accident sequence binning definitions
54	S8.3.2.1	Update with new results from the Level 1, R2 analysis
55	S8.3.2.1.1, 3rd para –5th para	Updated timing from the thermal-hydraulic analysis
56	S8.3.2.1.2, 2nd para	Updated timing from revised MAAP runs
57	S8.3.2.2	Technical justification and explanation for removing the Class II frequency from the offsite dose calculation
58	S8.3.2.2, last para	Deleted. Level 1 PRA results changed
59	S8.3.2.2.1, 3rd para	Technical clarification that class II sequences are NOT included in the offsite dose analysis, but ARE included in the total release frequency
60	S8.3.2.2.1, 5th para .	Technical clarification regarding the release categories to which the Class II sequence are assigned
61	S8.3.2.3	New Level 1 PRA results
62	S8.3.2.3.1	New Level 1 PRA results
63	S8.3.2.4	New Level 1 PRA results
64	S8.3.2.4.1	New Level 1 PRA results

Chapter 8 From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
65	T8.3-2	Table is updated with the phenomenological release categories and updated frequencies for all release categories
66	T8.3-3	Table is updated with new results from the thermal-hydraulic analyses of each representative sequence
67	F8.3-1 through F8.3-7	Updated figures with new MAAP results
68	S8.4, 1st bullet	Section 8 has been updated to include technical discussion of phenomenology in R2
69	S8.4, 2nd bullet	Technical clarification because the combustible gas generation analysis assumes the accident begins when containment is inerted
70	S8.4, 3rd bullet	New, detailed design of containment penetrations in DCD R3 allowed more detailed analysis of the containment isolation function – results updated with Level 2 R2 results
71	S8.4, 4th – 7th bullet	Updated R2 results for the various overpressure release categories
72	S8.4, 8th bullet	New discussion of the filtered release category – the only results come from class II-a sequences, which may eventually be discounted
73	S8.4, 9th para – 13th para	New discussion of the phenomenological releases
74	S8.4, 10th para, 3rd sent	Recalculated containment effectiveness based on R2 results

Chapter 8A From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
75	S8A	Title change
76	S8A, 1st para	Technical clarification – the new Level 2 PRA model is an entire integrated, single-top model, not just a set of containment event trees
77	S8A, 2nd para	Technical clarification relating to the paradigm shift in the Level 2 PRA
78	S8A.1, 1st and 2nd para	Updated methodology for the binning and partitioning of Level 1 sequences
79	S8A.1, Class I bullet	The integrated Level 2 model implicitly includes all initiator impact, so separate CETs for the LOPP scenarios is no longer necessary New sub-class definitions for Level 1, class I accident sequences
80	S8A.1, Class II bullet	There ARE class II accident sequences above truncation from the Level 1 analysis in revision 2 – this binning scheme is still a work in progress though. Technical clarification explaining treatment of class II sequences from the Level 1 analysis
81	S8A.1, Class III bullet	Revised binning based on integrated Level 2 model
82	S8A.1, 4th bullet	New sub-classes for the Level 1, class IV accident sequences
83	S8A.2, 1st – 4th paras	Explanation of the integrated Level 2 PRA, and how it is an extension of the Level 1 PRA. Old explanation of how the "split fraction" CETs were calculated and quantified is no longer applicable
84	S8A.2.2	
85	SA8.2.3	Updated technical discussion – more detailed design of containment penetrations in DCD R3 indicate that some lines do need to be isolated; fault tree was developed as a result
86	S8A.2.3, 2nd para	In response to RAI 19.2-74, a sensitivity will be done in NEDO-33201 Section 11 to consider this effect
87	S8A.2.4	Technical clarification – only high pressure (class III)

		sequences have the potential for DCH
88	S8A.2.5	Technical clarification – now that water level-specific event trees are used, they are specified
89	S8A.2.6	Deleted section entitled "Water Level Prior to RPV Failure (LD_LVL)." No longer a node because of the water level binning.
90	S8A.2.8	The vapor suppression node is now modeled with a fault tree per NEDO-33201 Section 4.18.
91	S8A.2.9	In response to RAI 19.2-73 – the vent operation is now modeled with a fault tree to justify the reliability number and initiator impact is included – HRA event is conservative but does not impact results
92	S8A.2.10	Per R2 model update, containment heat removal is now modeled explicitly with fault trees as opposed to calculated split fractions
93	S8A.2.11	Per R2 model update, containment heat removal is now modeled explicitly with fault trees as opposed to calculated split fractions
94	S8A.3	Completely new quantification methodology as a result of the entirely different Level 2 PRA
95	T8A-1	New sequences as a result of the revised Level 1 PRA – updated frequencies, Level 2 bins, and LDW water levels
96	T8A-2	New table of Level 2 sequence-by-sequence results
97	T8A-3	Table updated to include new Level 2 event trees and R2 results
98	F8A-1 through F8A-7	New CET, combines the old CPET and CSET
99		
10		
10		

Chapter 8B From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
10	S8B	Section deleted. Replaced with MAAP results.

Chapter 8C From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
10	8C, 2nd para	The isolation of main steam and feedwater lines is now explicitly modeled
10	8C, bullet E	Results of a MAAP source term sensitivity regarding containment bypass size. 1" penetrations are deemed insignificant relative to the "standard" containment bypass size
10	T8.C-1	New information to reflect additional design of containment penetrations in DCD R3 and the revised containment penetrations screening analysis

Chapter 9 From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
10	Section 9	Editorial changes.
10	Section 9.1	Editorial changes to reflect Rev 2 of Level 1 model results.
10	Section 9.2	Editorial changes to reflect Rev 2 of Level 1 model results
10	Section 9.3	Editorial changes to reflect Rev 2 of Level 1 model results.
11	Section 9.4	Editorial changes to reflect Rev 2 of Level 1 model results.
11	Section 9.5	DCH is not included based on Chapter 21 results.
11	Section 9.6.1	Not included due to Rev 2 Level 1 results and Chapter 8 results
11	Section 9.8	Editorial changes to reflect Rev 2 of Level 1 model results.
11	Table 9-1	Changed due to Rev 2 of Level 1 results and new MAAP results
11	Table 9-2	Changed due to Rev 2 of Level 1 results and new MAAP results
11	Table 9-3	Changed due to Rev 2 of Level 1 results and new MAAP results

Chapter 10 From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
11	10.1	Editorial. Stated latest version of MACCS2, 1.13.1 is used.
11	10.1	Editorial changes.
11	10.1	Editorial changes.
12	10.3.3	Updated for Technical Clarity
12	10.4	Editorial changes and added design dose goals in addition to NRC risk goals.
12	10.5	Editorial changes and added sensitivity study section
12	Table 10-3	Changed based on proposed addition to Rx building roof.
12	Table 10-3	Editorial changes.
12	Table 10-4	Editorial changes.
12	Table 10-5a	Changed due to new MAAP runs.
12	Table 10-5a	Added more sequences for some of the release categories.
12	Table 10-5b	Changed due to new MAAP runs.
12	Table 10-5b	Added more sequences for some of the release categories.
13	Table 10-7a	Editorial changes and changes due to new MACCS2 runs.
13	Table 10-7b	Editorial changes and changes due to new MACCS2 runs.
13	Table 10-8a	Editorial changes and changes due to new MACCS2 runs.
13	Table 10-8b	Added due to new MACCS2 runs and sensitivity study results.
13	Table 10-9a	Added due to new MACCS2 runs and sensitivity study results.
13	Table 10-9b	Added due to new MACCS2 runs and sensitivity study results.
13	Table 10-10a	Added due to new MACCS2 runs and sensitivity study results.
13	Table 10-10b	Added due to new MACCS2 runs and sensitivity study results.
13	Table 10-12a	Added due to new MACCS2 runs and sensitivity study results.
13	Table 10-12b	Added due to new MACCS2 runs and sensitivity study results.

Chapter 10 From Revision 1 to Revision 2 Change List

Item	Location	Description of Change
14	Table 10-13a	Added due to new MACCS2 runs and sensitivity study results.
14	Table 10-13b	Added due to new MACCS2 runs and sensitivity study results.
14	Table 10-14a	Added due to new MACCS2 runs and sensitivity study results.
14	Table 10-14b	Added due to new MACCS2 runs and sensitivity study results.
14	Figure 10-1	Figure changed due to new MACCS2 results

NEDO 33201 Revision 2 Section 16 Change List

Item	Location	Description of Change
1.	S16.1	Updated introduction section to be more specific and to add discussion of critical safety functions
2.	S16.2 2nd para.	Add to clarify that shutdown model is specific to refueling outages
3.	S16.2.1	Updated in response to NRC RAI 16.1-84. Tech Spec Mode definitions added to text.
4.	S16.2.1.2 and S16.2.1.3	Added mode to analysis to account for Mode 5 with an open containment. Addressed NRC RAI 19.1-81 and 19.1-83
5.	S16.2.1.4	Re worded section to clarify water level is actually higher than Mode 5 even though title is 'unflooded'
6.	S16.2.1.5	Updated to clarify that either FAPCS train is available for cooling reactor well
7.	S16.3.1	Expanded discussion to explain treatment of all critical safety functions
8.	S16.3.1.1	Additional updated information provided for clarity
9.	S16.3.1.2.1	Updated to provide additional justification for LOCA frequency reduction in response to RAI 19.1-121

NEDO 33201 Revision 2 Section 16 Change List

Item	Location	Description of Change
10.	S16.3.1.2.1	Additional updated information provided for clarity with regard to LOCA treatment during shutdown
11.	S16.3.1.2.1.2	LOCA initiators added to Mode 5 in response to RAI 19.1-85
12.	S16.3.1.2.2 4th para.	Additional updated information provided for clarity and in response to RAI 19.1-87
13.	S16.3.1.2.3	Updated to clarify FMCRD replacement methodology in response to RAI 19.1-89
14.	S16.3.3.4	Additional info added to complete table following the addition of LOCA events for Mode 5
15.	S16.3.4.1	Rewrote section on recovery actions to reference an ESBWR specific calculation instead of the ABWR used for the previous revison
16.	S16.3.4.2	Recovery of LOPP no longer credited in shutdown model
17.	S16.3.4.3	Added reference to Availability Controls Manual for drywell hatch closure in response to RAI 19.1-90
18.	S16.4.1	Rewrote entire event tree section to clarify and better organize information
19.	S16.5	Updated system section to add details about system availability during shutdown and reflect Tech Spec requirements for shutdown. Details in updated section assist in response to several NRC RAIs (19.1-91, 19.1-92, 19.1-93, 19.1-95)
20.	S16.6	Updated results section with new results and several sensitivity analyses.
21.	S16.6.2.3	Operator action sensitivity in response to RAI 19.1-99
22.	S16.7	Updated insights based on revised results
23.	S16.8	Updated conclusions section including comparison of results to NRC goals for PRA.
24.	Tables (all)	All Tables updated based on extensive revision to model. Table 16.6-2 is deleted since it is no longer relevant. Tables 16.6-5 (all CD sequences) and 16.6-6 (cut sets from dominant sequences) added as these tables were requested following chapter 4 submittal.

NEDO 33201 Revision 2 Section 16 Change List

Item	Location	Description of Change
25.	Figures (all)	Figure 16.3-1 was updated based on ESBWR specific analysis Every event tree figure updated based on model revision

Attachment 3 to Enclosure 1 of
MFN 07-237, Supplement 1

NEDO-33201, Revision 2
Table 7.2-5, Rev 2 – Markup

Sequence	T-FDW050– Sequence No. 3	
CDF		1.14E-09
% of Class I CDF		20.26%
% of total CDF		9.35%
Initiating event	Loss of Feedwater	

Scram is successful

Isolation Condensers fail to provide overpressure protection

SRVs lift – overpressure protection is successful

All SRVs reclose

ADS is successful using DPVs

DW/WW vacuum breakers are successful - pressure suppression is successful

GDCS fails

Low pressure injection using FAPCS, Firewater and CRD fail

Vessel fails at ~~high~~**low** pressure

Lower drywell water level is LOW

Sequence	T-IORV018– Sequence No. 4	
CDF		**9.02**~~8.78~~E-10
% of Class III CDF		19.~~96~~**98**%
% of total CDF		7.3**9**%
Initiating event	Inadvertent Open Relief Valve	

Scram **success**~~fails~~

Feedwater **injection fails**~~Runback is successful~~

2 CRD fail to restore level

Manual Depressurization using SRVs is successful

Low pressure injection with FAPCS and Firewater fail

ADS fails to depressurize using DPS

Lower drywell water level is LOW

Sequence	AT-T-GEN021– Sequence No. 5	
CDF		**8.78**~~9.02~~E-10
% of Class III CDF		19.4~~5~~**6**%
% of total CDF		7.20%
Initiating event	General Transient (e.g. turbine trip)	

Scram **fails**~~is successful~~

Feedwater **runback success**~~injection fails~~

SRVs lift – overpressure protection is successful, but one or more SRVs sticks open

ADS Inhibit is successful

SLC is successful

Feedwater and CRD fail to maintain reduced level

Lower drywell water level is LOW

Sequence	AT-T-LOPP013– Sequence No. 7	
CDF		**6.8**~~5.25~~E-10
% of Class **III**~~II~~ CDF		15.05%
% of total CDF		5.57%
Initiating event	Loss of Offsite power	

Scram fails

SRVs lift – successful overpressure protection, but one or more SRV sticks open

ADS inhibit is successful

SLC is successful

CRD pumps fail to maintain level (both required)

Lower drywell water level is LOW

Sequence	LL-S-FDWB045– Sequence No. 9	
CDF		**5.25**~~6.80~~E-10
% of Class I CDF		9.32%
% of CDF		4.30%
Initiating event	Large LOCA in Feedwater Line B	

Scram is successful

LOCA depressurizes

DW/WW vacuum breakers are successful – pressure suppression is successful

GDCS fails

Low pressure injection using FAPCS and Firewater fail

Lower drywell water level is HIGH