BFN EPU
Containment Overpressure (COP) Credit Risk Assessment

Rev. 2

Performed for:

Tennessee Valley Authority

Performed by:

ERIN Engineering and Research, Inc.

July 10, 2006

Tennessee Valley Authority
Browns Ferry Nuclear (BFN)

BFN EPU
Containment Overpressure (COP)
Credit Risk Assessment

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Prepared by: ____________________ Date: July 10, 2006

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EXECUTIVE SUMMARY

The report documents the risk impact of utilizing containment accident pressure (containment overpressure) to satisfy the net positive suction head (NPSH) requirements for RHR and Core Spray pumps during DBA LOCA, ATWS and SBO accident scenarios.

The risk assessment evaluation uses the current BFN Unit 1 Probabilistic Risk Assessment (PRA) internal events model (including internal flooding). The BFN PRA provides the necessary and sufficient scope and level of detail to allow the calculation of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) changes due to the crediting of containment overpressure in determining sufficient NPSH requirements for the RHR system and Core Spray system emergency core cooling pumps.

The steps taken to perform this risk assessment evaluation are as follows:

1) Evaluate sensitivities to the accident calculations to determine under what conditions credit for COP is required to satisfy low pressure ECCS pump NPSH.

2) Revise accident sequence event trees to make low pressure ECCS pumps dependent upon containment isolation when other plant pre-conditions exist (i.e., SW high temperature, SP initial high temperature, SP low water level).

3) Modify the existing BFN PRA Containment Isolation System fault tree to include the probability of pre-existing containment leakage.

4) Quantify the modified PRA models and determine the following risk metrics:

   - Change in Core Damage Frequency (CDF)
   - Change in Large Early Release Frequency (LERF)

5) Perform modeling sensitivity studies and a parametric uncertainty analysis to assess the variability of the results.

The conclusion of the plant internal events risk associated with this assessment is as follows.

1) Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr. Based on this criteria, the proposed change (i.e., use of COP to satisfy the net positive suction head (NPSH) requirements for RHR and Core Spray pumps) represents a very small change in CDF (2.4E-08/yr).

2) Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of Large Early Release Frequency (LERF) below 10^{-7}/yr. Based on this criteria, the proposed change (i.e., use of COP to satisfy the net positive suction head (NPSH) requirements for RHR and Core Spray pumps) represents a very small change in LERF (2.4E-08/yr).

Section 1
INTRODUCTION

The report documents the risk impact of utilizing containment accident pressure (containment overpressure) to satisfy the net positive suction head (NPSH) requirements for RHR and Core Spray pumps during DBA LOCA, ATWS and SBO accident scenarios.

1.1 BACKGROUND

Tennessee Valley Authority (TVA) submitted the BFN extended power uprate (EPU) license amendment request (LAR) to the NRC in June 2004. In a October 3, 2005 letter to TVA, the NRC requested the following additional information on the EPU LAR:

> *"SPSB-A.11*
>
> *As part of its EPU submittal, the licensee has proposed taking credit (Unit 1) or extending the existing credit (Units 2 and 3) for containment accident pressure to provide adequate net positive suction head (NPSH) to the ECCS pumps. Section 3.1 in Attachment 2 to Matrix 13 of Section 2.1 of RS-001, Revision 0 states that the licensee needs to address the risk impacts of the extended power uprate on functional and system-level success criteria. The staff observes that crediting containment accident pressure affects the PRA success criteria; therefore, the PRA should contain accident sequences involving ECCS pump cavitation due to inadequate containment pressure. Section 1.1 of Regulatory Guide (RG) 1.174 states that licensee-initiated licensing basis change requests that go beyond current staff positions may be evaluated by the staff using traditional engineering analyses as well as a risk-informed approach, and that a licensee may be requested to submit supplemental risk information if such information is not submitted by the licensee. It is necessary to consider risk insights, in addition to the results of traditional engineering analyses, while determining the regulatory acceptability of crediting containment accident pressure.*
>
> *Considering the above discussion, please provide an assessment of the credit for containment accident pressure against the five key principles of risk-informed decisionmaking stated in RG 1.174 and SRP Chapter 19. Specifically, demonstrate that the proposed containment accident*

> *pressure credit meets current regulations, is consistent with the defense-in-depth philosophy, maintains sufficient safety margins, results in an increase in core-damage frequency and risk that is small and consistent with the intent of the Commission's Safety Goal Policy Statement, and will be monitored using performance measurement strategies. With respect to the fourth key principle (small increase in risk), provide a quantitative risk assessment that demonstrates that the proposed containment accident pressure credit meets the numerical risk acceptance guidelines in Section 2.2.4 of RG 1.174. This quantitative risk assessment must include specific containment failure mechanisms (e.g., liner failures, penetration failures, primary containment isolation system failures) that cause a loss of containment pressure and subsequent loss of NPSH to the ECCS pumps."*

Typical of other industry EPU LAR submittals, the BFN EPU LAR includes a request to credit containment accident pressure, also known as containment overpressure (COP), in the determination of net positive suction head (NPSH) for low pressure ECCS systems following design basis events. Also consistent with other industry EPU LAR submittals, the NRC is requesting risk information from licensees regarding the COP credit request.

BFN Units 2 and 3 already have existing approvals for containment overpressure credit.

The need for COP credit requests is driven by the conservative nature of design basis accident calculations. Use of more realistic inputs in such calculations shows that no credit for COP is required. In any event, the request for containment accident pressure credit is a physical aspect that will exist during the postulated design basis and special event accidents. The EPU LAR simply requests to include that existing containment accident pressure in the ECCS pump NPSH calculations. The NRC request is to investigate the impact on risk if the containment accident pressure is not present (e.g., postulated pre-existing primary containment failure) during the postulated scenarios.

The Nuclear Regulatory Commission (NRC) has allowed credit for COP to satisfy NPSH requirements in accordance with Regulatory Guide 1.82 (RG 1.82). Specifically, RG 1.82 Position 2.1.1.2 addresses containment overpressure as follows:

> *"For certain operating BWRs for which the design cannot be practicably altered conformance with Regulatory Position 2.1.1.1 may not be possible. In these cases, no additional containment pressure should be included in the determination of available NPSH than is necessary to preclude pump cavitation. Calculation of available containment pressure should underestimate the expected containment pressure when determining available NPSH for this situation. Calculation of suppression pool water temperature should overestimate the expected temperature when determining available NPSH."*

The proposed change in the BFN license basis regarding credit for COP meets the approved positions of RG 1.82. However, developments between the NRC staff and members of the Advisory Committee on Reactor Safeguards (ACRS) in 2005 regarding proposed language to Revision 4 of RG 1.82 prompted the NRC to request performance of a 'risk-informed' assessment in accordance with NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis".

1.2 SCOPE

This risk assessment addresses principle #4 of the RG 1.174 risk informed structure. Principle #4 of RG 1.174 involves the performance of a risk assessment to show that the impact on the plant core damage frequency (CDF) and large early release frequency (LERF) due to the proposed change is within acceptable ranges, as defined by RG 1.174. The other principles (#1-#3, and #5) are not addressed in this report.

This analysis assesses the CDF and LERF risk impact on the BFN Unit 1 at-power internal events PRA resulting from the COP credit requirement for low pressure ECCS pumps during large LOCA, ATWS and SBO accident scenarios.

External event and shutdown accident risk is assessed on a qualitative basis.

In addition, a review of the BFN Unit 2 and Unit 3 models is performed to show that the results from the Unit 1 BFN PRA apply to Units 2 and 3, as well.

1.3 DEFINITIONS

Accident sequence - a representation in terms of an initiating event followed by a combination of system, function and operator failures or successes, of an accident that can lead to undesired consequences, with a specified end state (e.g., core damage or large early release). An accident sequence may contain many unique variations of events that are similar.

Core damage - uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated and involving enough of the core to cause a significant release.

Core damage frequency - expected number of core damage events per unit of time.

End State - is the set of conditions at the end of an event sequence that characterizes the impact of the sequence on the plant or the environment. End states typically include: success states, core damage sequences, plant damage states for Level 1 sequences, and release categories for Level 2 sequences.

Event tree - a quantifiable, logical network that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails and arrives at either a successful or failed end state.

Initiating Event - An initiating event is any event that perturbs the steady state operation of the plant, if operating, or the steady state operation of the decay heat removal systems during shutdown operations such that a transient is initiated in the plant. Initiating events trigger sequences of events that challenge the plant control and safety systems.

ISLOCA - a LOCA when a breach occurs in a system that interfaces with the RCS, where isolation between the breached system and the RCS fails. An ISLOCA is usually characterized by the over-pressurization of a low-pressure system when subjected to RCS pressure and can result in containment bypass.

Large early release - the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions.

Large early release frequency - expected number of large early releases per unit of time.

Level 1 - identification and quantification of the sequences of events leading to the onset of core damage.

Level 2 - evaluation of containment response to severe accident challenges and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment.

Plant damage state - Plant damage states are collections of accident sequence end states according to plant conditions at the onset of severe core damage. The plant conditions considered are those that determine the capability of the containment to cope with a severe core damage accident. The plant damage states represent the interface between the Level 1 and Level 2 analyses.

Probability - is a numerical measure of a state of knowledge, a degree of belief, or a state of confidence about the outcome of an event.

Probabilistic risk assessment - a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public (also referred to as a probabilistic risk assessment, PRA).

Release category - radiological source term for a given accident sequence that consists of the release fractions for various radionuclide groups (presented as fractions of initial core inventory), and the timing, elevation, and energy of release. The factors addressed in the definition of the release categories include the response of the containment structure, timing, and mode of containment failure; timing, magnitude, and mix of any releases of radioactive material; thermal energy of release; and key factors affecting deposition and filtration of radionuclides. Release categories can be considered the end states of the Level 2 portion of a PRA.

Risk - likelihood (probability) of occurrence of undesirable event, and its level of damage (consequences).

Risk metrics - the quantitative value, obtained from a risk assessment, used to evaluate the results of an application (e.g., CDF or LERF).

Severe accident - an accident that involves extensive core damage and fission product release into the reactor vessel and containment, with potential release to the environment.

Split Fraction - a unitless parameter (i.e., probability) used in quantifying an event tree. It represents the fraction of the time that each possible outcome, or branch, of a particular top event may be expected to occur. Split fractions are, in general, conditional on precursor events. At any branch point, the sum of all the split fractions representing possible outcomes should be unity. (Popular usage equates "split fraction" with the failure probability at any branch [a node] in the event tree.)

1.4 ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ATWS	Anticipated Transient without Scram
BFN	Browns Ferry Nuclear plant
CCF	Common Cause Failure
CDF	Core Damage Frequency
CET	Containment Event Tree
COP	Containment Overpressure
CPPU	Constant Pressure Power Uprate
DBA	Design Basis Accident
DW	Drywell
ECCS	Emergency Core Cooling Systems
EPU	Extended Power Uprate
GE	General Electric
HEP	Human Error Probability
HPCI	High Pressure Core Injection system
HRA	Human Reliability Analysis
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination for External Events

ISLOCA	Interface System Loss of Coolant Accident
La	Maximum Allowable Primary Containment Leakage Rate
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LLOCA	Large LOCA
LOOP	Loss of Offsite Power event
LPCI	Low Pressure Coolant Injection
MAAP	Modular Accident Analysis Program
NPSH	Net Positive Suction Head
NRC	United States Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
RCIC	Reactor Core Isolation Cooling System
RG	Regulatory Guide
RHR	Residual Heat Removal System
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SMA	Seismic Margins Assessment
SP	Suppression Pool
SPC	Suppression Pool Cooling
SW	Service Water
TS	Technical Specifications

TVA	Tennessee Valley Authority
WW	Wetwell

Section 2
APPROACH

This section includes a brief discussion of the analysis approach and the types of inputs used in this risk assessment.

2.1 GENERAL APPROACH

This risk assessment is performed by modification and quantification of the BFN PRA models.

2.1.1 Use of BFN Unit 1 PRA

The current BFN Unit 1 PRA models (BFN model U1050517) are used as input to perform this risk assessment. The Browns Ferry PRA uses widely-accepted PRA techniques for event tree and fault tree analysis. Event trees are constructed to identify core damage and radionuclide release sequences. The event tree "top events" represent systems (and operator actions) that can prevent or mitigate core damage. Fault trees are constructed for each system in order to identify the failure modes. Analysis of component failure rates (including common cause failures) and human error rates is performed to develop the data needed to quantify the fault tree models.

For the purpose of analysis, the Browns Ferry PRA divides the plant systems into two categories:

1. Front-Line Systems, which directly satisfy critical safety functions (e.g., Core Spray and Torus Cooling), and
2. Support Systems, which are needed to support operation of front-line systems (e.g., AC power and service water).

Front-line event trees are linked to the end of the Support System event trees for sequence quantification. This allows definition of the status of all support systems for each sequence before the front-line systems are evaluated. Quantification of the event tree and fault tree models is performed using personal computer version of the RISKMAN code.

The Support System and Front-Line System event trees are "linked" together and solved for the core damage sequences and their frequencies. Each sequence represents an initiating event and combination of Top Event failures that results in core damage. The frequency of each sequence is determined by the event tree structure, the initiating event frequency and the Top Event split fraction probabilities specified by the RISKMAN master frequency file. RISKMAN allows the user to enter the split fraction names and the logic defining the split fractions (i.e., rules) to be selected for a given sequence based on the status of events occurring earlier in the sequence or on the type of initiating event.

2.1.2 PRA Quality

The BFN PRA used as input to this analysis (BFN model U1050517) is of sufficient quality and scope for this application. The BFN Unit 1 PRA is highly detailed, including a wide variety of initiating events (e.g., transients, internal floods, LOCAs inside and outside containment, support system failure initiators), modeled systems, extensive level of detail, operator actions, and common cause events.

The BFN Units 2 and 3 at-power internal events PRAs received a formal industry PRA Peer Review in 1997. All of the "A" and "B" priority comments have been addressed.

Refer to Appendix A for further details concerning the quality of the BFN PRA.

2.2 STEPS TO ANALYSIS

The performance of this risk assessment is best described by the following major analytical steps:

- Assessment of NPSH calculations
- Estimation of pre-existing containment failure probability
- Analysis of relevant plant experience data
- Manipulation and quantification of BFN Unit 1 RISKMAN PRA models
- Comparison to ΔCDF and ΔLERF RG 1.174 acceptance guidelines
- Performance of uncertainty and sensitivity analyses
- Assessment of "Large Late" Release Impact
- Review of BFN Unit 2 and Unit 3 PRAs

Each of these steps is discussed briefly below.

2.2.1 Assessment of NPSH Calculations

The purpose of this task is to develop an understanding of the BFN EPU NPSH calculations that result in the need to credit containment overpressure for LLOCA, ATWS, and SBO accident scenarios.

The need for COP credit requests is driven by the conservative nature of the accident calculations. The NPSH calculations are reviewed and sensitivity calculations performed to determine under what conditions of more realistic inputs is there no need for COP credit in the determination of low pressure ECCS pump NPSH.

2.2.2 Estimation of Pre-Existing Containment Failure Probability

This task involves defining the size of a pre-existing containment failure pathway to be used in the analysis to defeat the COP credit, and then quantifying the probability of occurrence of the un-isolable pre-existing containment failure. The approach to this input parameter calculation will follow EPRI guidelines regarding calculation of pre-existing containment leakage probabilities in support of integrated leak rate test (ILRT) frequency extension LARs (i.e., EPRI Report 1009325, Risk Impact of Extended Integrated Leak Rate Testing Intervals, 12/03).[2] This is the same approach used in the recent Vermont Yankee EPU COP analyses presented to the ACRS in December 2005.

The pre-existing unisolable containment leak probability is combined with the BFN PRA containment isolation failure on demand fault tree (CIL) to develop the likelihood of an unisolated primary containment at t=0 that can defeat the COP credit necessary for the determination of adequate low pressure ECCS pump NPSH.

2.2.3 Analysis of Relevant Plant Experience Data

An unisolated primary containment is not the only determining factor in defeating low pressure ECCS pump NPSH. The DBA LLOCA NPSH calculations show that other extreme low likelihood plant conditions are required at t=0 to result in the need to credit COP in the determination of pump NPSH, such as:

- High initial reactor power level
- High river water temperature
- High initial torus water temperature
- Low initial torus water level

This step involves obtaining plant experience data for river water temperature and torus water temperature and level and performing statistical analysis to determine the probabilities of exceedance.

2.2.4 Manipulation And Quantification of BFN Unit 1 RISKMAN PRA Models

This task is to make the necessary modifications to the BFN Unit 1 RISKMAN-based PRA models to simulate the loss of low pressure ECCS pumps during PRA Large LOCA, ATWS, and SBO scenarios due to inadequate NPSH caused by an unisolated containment coincident with other plant conditions (e.g., high service water temperature).

All large LOCA initiated sequences in the BFN PRA are modified as appropriate (except ISLOCAs and LOCAs outside containment, because these LOCAs result in deposition of decay heat directly outside the containment and not into the suppression pool). This approach to manipulating only LLOCA scenarios is to mirror the DBA accident calculations requiring COP credit. This is consistent with the ACRS observations during the December 2005 Vermont Yankee EPU COP hearings, in which the ACRS commented that they did not prefer the approach of assigning COP credit to all accident sequence types in the PRA simply for the sake of conservatism.

All ATWS sequences in the BFN PRA (i.e., transients, LOOP, and IORV initiated ATWS scenarios) are modified to model the COP credit impact.

SBO accident sequences in the BFN PRA are modified to require COP credit for adequate ECCS NPSH upon recovery of AC power after 4 hours.

The modeling and quantification is performed consistent with common RISKMAN modeling techniques.

2.2.5 Comparison to ΔCDF and ΔLERF RG 1.174 Acceptance Guidelines

The revised BFN Unit 1 PRA models are quantified to determine CDF and LERF. The difference in CDF and LERF between the revised model of this assessment and the BFN Unit 1 PRA base results are then compared to the RG 1.174 risk acceptance guidelines. The RG 1.174 ΔCDF and ΔLERF risk acceptance guidelines are summarized in Figures 2-1 and 2-2, respectively. The boundaries between regions are not necessarily interpreted by the NRC as definitive lines that determine the acceptance or non-acceptance of proposed license amendment requests; however, increasing delta risk is associated with increasing regulatory scrutiny and expectations of compensatory actions and other related risk mitigation strategies.

2.2.6 Performance of Uncertainty and Sensitivity Analyses

To provide context to the variability of the calculated deltaCDF and deltaLERF results, a parametric uncertainty analysis was performed using the RISKMAN software.

2.2.7 Assessment of "Large Late" Release Impact

This task is to perform an assessment of the EPU COP credit impact on BFN Unit 1 PRA "Large Late" radionuclide releases. This task is performed because the ACRS questioned Entergy on this issue during the recent Vermont Yankee EPU ACRS hearings in December 2005.

This aspect of the analysis is for additional information, and does not directly correspond to the RG 1.174 risk acceptance guidelines shown in Figures 2-1 and 2-2.

2.2.8 Review of BFN Unit 2 and Unit 3 PRAs

The base analysis uses the BFN Unit 1 PRA models. This task involves reviewing the BFN Unit 2 and BFN Unit 3 RISKMAN PRA models and associated documentation to determine whether the analysis performed for BFN Unit 1 is also applicable to Unit 2 and Unit 3.

Figure 2-1

RG 1.174 CDF RISK ACCEPTANCE GUIDELINES

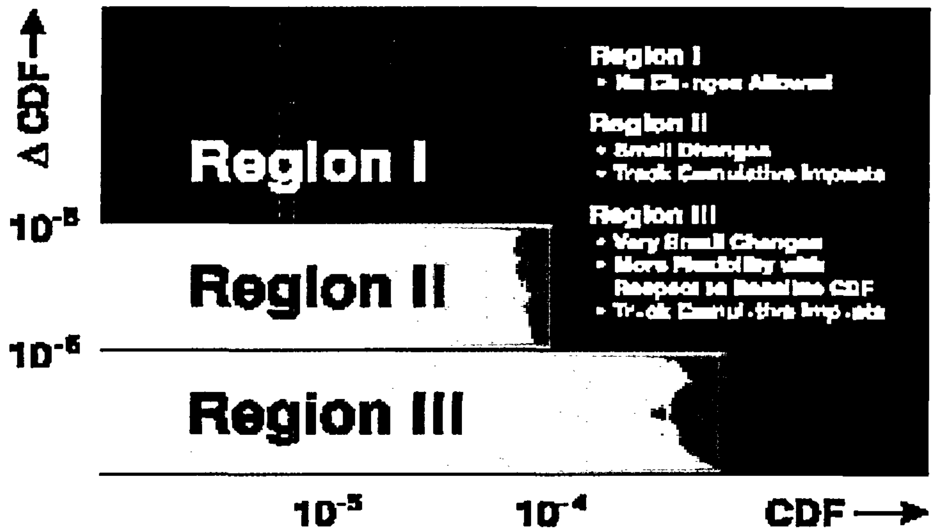
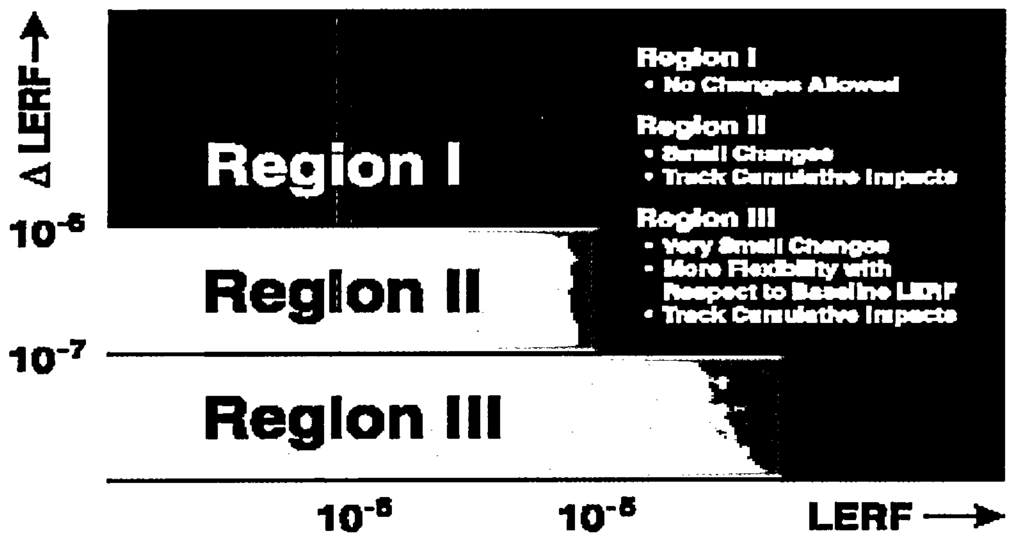


Figure 2-2

RG 1.174 LERF RISK ACCEPTANCE GUIDELINES



Section 3

ANALYSIS

This section highlights the major qualitative and quantitative analytic steps to the analysis.

3.1 ASSESSMENT OF NPSH CALCULATIONS

The purpose of this risk assessment is due to the fact that the conservative nature of the accident calculations result in the need to credit COP in determining adequate low pressure ECCS pump NPSH. Use of more realistic inputs in such calculations can show that no credit for COP is required.

Special events such as ATWS and SBO are not necessarily analyzed using the conservative assumptions as for design basis events such as LOCA. Review of the BFN ATWS and COP NPSH calculations shows that some level of containment overpressure credit would be required even if more realistic inputs are used in the calculations. As such, increasing the degree of realistic treatment in the special events NPSH calculations is not expected to eliminate the need for containment overpressure credit. This is not true for the LLOCA NPSH calculations, which show that use of more realistic values for a variety of input parameters result in showing no need for COP credit. As such, the following discussions address sensitivities to the LLOCA NPSH calculations. The ATWS and SBO scenarios analyzed in this risk assessment assume that COP credit is always required.

The GE DBA LOCA calculation makes the following conservative assumptions, among others, regarding initial plant configuration and operation characteristics:

- Initial reactor power level at 102% EPU
- Decay heat defined by 2 sigma uncertainty

- 2 RHR pumps and 2 RHR heat exchangers in SPC
- All pumps operating at full flow
- River water temperature at 95°F
- Initial suppression pool temperature at 95°F
- Initial SP water volume at minimum technical specification level
- No credit for containment heat sinks

The GE DBA LOCA calculations were reviewed and the following input parameters were identified as those with a potential to significantly impact the DBA analytic conclusions regarding the need for COP credit in NPSH determination:

- Initial reactor power level
- Decay heat
- Number of RHR pumps and heat exchangers in SPC
- River water temperature
- Initial suppression pool temperature
- RHR heat exchanger effectiveness
- Initial suppression pool water volume
- Credit for containment heat sinks

Based on knowledge of the calculations, other inputs such as initial containment air temperature and humidity, have non-significant impacts on the results.

It is recognized that there are numerous different combinations of more realistic calculation inputs that show that COP credit is not necessary for maintenance of low pressure ECCS pump NPSH during LLOCA accidents. To simplify the risk assessment, the different combinations of realistic input sensitivities were maintained at a manageable number. A number of sensitivity calculations were performed to identify key input parameters for use in this risk assessment. The results of these calculations

are shown in Table 3-1 (the shaded cells show those parameters that changed from the base DBA LOCA calculation). [3]

From the results of the LLOCA NPSH sensitivity cases summarized in Table 3-1, the following general conclusions can be made:

- Initial reactor power, decay heat level, initial water temperatures, suppression pool volume, and the number of RHR pumps/HXs in operation are the key determining factors in the analytic conclusions. These factors are evaluated in this risk assessment.
- RHR heat exchanger effectiveness and credit for containment heat sinks also influence the results, but to manage the risk calculation, this assessment takes no probabilistic credit for these issues.
- COP credit is <u>not</u> required for NPSH, even with the conservative DBA calculation inputs, if 4 RHR pumps and associated heat exchangers are in operation (refer to Case 1 in Table 3-1).
- COP credit is <u>not</u> required for NPSH when 3 RHR pumps are in operation event with conservative 102% EPU power and 2 sigma decay heat assumptions, and conservative water temperature and SP volume assumptions (refer to Case 1d in Table 3-1).
- If the plant is operating at an unexpected 102% EPU initial power level with an assumed 2 sigma decay heat, only 2 RHR pumps and heat exchangers are placed in SPC operation, initial SP volume at 123,500 ft^3, and river water temperature is at 68°F, then torus water temperature must be above 87°F to result in the need for COP credit (refer to Case 2f in Table 3-1).
- If the plant is operating at the expected nominal 100% EPU initial power level (2 sigma decay heat not assumed), only 2 RHR pumps and heat exchangers are placed in SPC operation, initial SP volume at 123,500 ft^3, and river water temperature is at 85°F, then torus water initial temperature must be above 86°F to result in the need for COP credit (refer to Case 4i in Table 3-1).

The analytic conclusions are used in this risk assessment to define two plant states that will result in failure of low pressure ECCS pumps on inadequate NPSH during large LOCAs if the containment is unisolated:

- Plant State 1: 102% EPU initial power level, 2 sigma decay heat, 2 RHR pumps and heat exchangers in SPC, initial SP volume at 123,500 ft^3, river water temperature of 68°F, and torus water initial temperature above 87°F.

- Plant State 2: 100% EPU initial power level, nominal decay heat , 2 RHR pumps and heat exchangers in SPC, initial SP volume at 123,500 ft^3, river water temperature of 86°F, and river water initial temperature above 85°F.

These two plant states are used in this risk assessment to model the LLOCA scenarios that can result in loss of low pressure ECCS pumps due to inadequate NPSH when the containment is unisolated. The probability of being in Plant State 1 or Plant State 2 is discussed below in Section 3.2.

Scenarios with 3 or 4 RHR pumps and heat exchangers are not explicitly incorporated into the base case quantification because the risk contribution from such scenarios is non-significant (refer to Section 4.2.2).

3.2 PROBABILITY OF PLANT STATE 1 AND PLANT STATE 2

This section discusses the estimation of the probability of being in Plant State 1 or Plant State 2 during LLOCA scenarios. This assessment is based on the statistical analysis of BFN experience data. Refer to Appendix C for the statistical analysis of variations in BFN river water temperature and torus water temperature and level.

3.2.1 Probability of Plant State 1

The probability of being in Plant State 1 is determined as follows:

- The probability of being at 102% EPU power at the time of the postulated DBA LOCA is modeled as a miscalibration error of an instrument

- If such a miscalibration error occurs, it is assumed that the plant will be operating at 102% and that the operator <u>does not</u> notice other differing plant indications that would cause the operator to re-evaluate the plant condition

- If the plant is operating at 102% power, the decay heat level defined by 2 sigma uncertainty is assumed to occur with a probability of 1.0 (this conservative assumption is to simplify the analysis).

- The probability of river water temperature greater than 68°F is determined from the BFN experience data statistical analysis summarized in Appendix C.

- Given river water temperature 68°F, the conditional probability that the torus water temperature is 87°F is determined from the BFN experience data statistical analysis summarized in Appendix C.

- The probability that suppression pool water level is less than 123,500 ft^3 is also based on the BFN experience data statistical analysis summarized in Appendix C.

The probability of being at 102% power at the time of the accident is modeled as the likelihood of a miscalibrated instrument. Based on review of the pre-initiator human error probability calculations in the BFN Unit 1 PRA Human Reliability Analysis, this risk assessment assumes a nominal human error probability of 5E-3 for miscalibration of an instrument. As such, the probability of being at 102% power at t=0 is taken in this analysis to be 5E-3.

As can be seen from Table C-1, the probability of river water temperature being greater than 68°F at the time of the DBA LOCA is 5.64E-1. As discussed in Section C.2.1, the conditional probability that suppression pool temperature is greater than 87°F is 4.42E-1. As can be seen from Table C-3, the probability of suppression pool water volume being below 123,500 ft^3 at the time of the DBA LOCA is 1.45E-2.

Therefore, the probability of being in Plant State 1 at the time of the DBA LOCA is 5E-3 x 5.64E-1 x 4.42E-1 x 1.45E-2 = 1.8E-5.

3.2.2 Probability of Plant State 2

The probability of being in Plant State 2 is determined as follows:

- The probability of being at 100% EPU power at the time of the postulated DBA LOCA is reasonably assumed to be 1.0
- The probability of river water temperature greater than 85°F is determined from the BFN experience data statistical analysis summarized in Appendix C.
- Given river water temperature of 85°F, the conditional probability that the torus water temperature is 87°F, is taken to be 1.0. This is reasonable (refer to Figure C-1).
- The probability that suppression pool water level is less than 123,500 ft^3 is also based on the BFN experience data statistical analysis summarized in Appendix C.

As can be seen from Table C-1, the probability of river water temperature being greater than 85°F at the time of the DBA LOCA is 1.64E-1. As can be seen from Table C-3, the probability of suppression pool water volume being below 123,500 ft^3 at the time of the DBA LOCA is 1.45E-2.

Therefore, the probability of being in Plant State 2 at the time of the DBA LOCA is 1.64E-1 x 1.0 x 1.45E-2 = 2.4E-3.

3.3 PRE-EXISTING CONTAINMENT FAILURE PROBABILITY

As discussed in Section 2, the approach to this input parameter calculation follows the EPRI guidelines regarding calculation of pre-existing containment leakage probabilities in support of integrated leak rate test (ILRT) frequency extension LARs (i.e., EPRI Report 1009325, Risk Impact of Extended Integrated Leak Rate Testing Intervals, 12/03). [2]

This assessment is provided in Appendix B of this report. As discussed in Appendix B, a pre-existing unisolable containment leakage path of 20La is assumed in the base case quantification of this risk assessment to result in defeating the necessary COP credit. As can be seen from Table B-1, the probability of a 20La pre-existing containment leakage at any given time at power is 1.88E-03.

This low likelihood of a significant pre-existing containment leakage path is consistent with BFN primary containment performance experience. The BFN primary containment performance experience shows BFN containment leakages much less than 20La. Per Reference [1], the BFN Unit 2 and Unit 3 primary containment ILRT results from the most recent tests are as follows:

Unit	Test Date	Containment Leakage (Fraction of La)
2	11/06/94	0.1750
2	03/17/91	0.1254
3	10/10/98	0.1482
3	11/06/95	0.4614

Although the above results are for Units 2 and Units 3, given the similarity in plant design and operation and maintenance practices, the results are reasonably judged to be reflective of BFN Unit 1, as well.

Sensitivity studies to the base case quantification (refer to Section 4) assess the sensitivity of the results to the pre-existing leakage size assumption.

3.4 MODIFICATIONS TO BFN UNIT 1 PRA MODELS

3.4.1 PRA Model Modifications for LLOCAs

As discussed in Section 2, all large LOCA initiated sequences in the BFN PRA are modified as appropriate (except ISLOCAs and LOCAs outside containment, because these LOCAs result in deposition of decay heat directly outside the containment and not into the suppression pool). The following Large LOCA initiated sequences in the BFN Unit 1 PRA were modified:

- Large LOCA – Loop I Core Spray Line Break (LLCA)
- Large LOCA – Loop II Core Spray Line Break (LLCB)
- Large LOCA – Loop A Recirc. Discharge Line Break (LLDA)
- Large LOCA – Loop B Recirc. Discharge Line Break (LLDB)
- Large LOCA – Loop A Recirc. Suction Line Break (LLSA)
- Large LOCA – Loop B Recirc. Suction Line Break (LLSB)
- Other Large LOCA (LLO)

The accident sequence modeling for the above LLOCA initiators was modified as follows:

- A top event for loss of containment integrity (CIL) was added to the beginning of the Level 1 event tree structures
- A top event modeling the additional Plant State pre-conditions (NPSH) was added to the beginning of the Level 1 event tree structures, right after the CIL top event.
- If top events CIL and NPSH are satisfied (i.e., occur), then the RHR pumps and CS pumps are directly failed
- LPCI and LPCS inter-unit crossties are defeated because the pumps crosstied from the Unit 2 would be aligned to the Unit 1 suppression pool and would experience the same NPSH conditions as the Unit 1 pumps.

Refer to Appendix E for print-outs of the revised large LOCA event trees.

The CIL top event is quantified using a fault tree. The fault tree is a modified version of the existing BFN Unit 1 Level 2 PRA containment isolation fault tree. The BFN Unit 1 Level 2 PRA containment isolation fault tree models failure of the containment isolation system on demand given an accident signal. Hardware, power and signal failures for all primary containment penetrations greater than 3" diameter are modeled in the fault tree. To this fault tree structure was added the probability of a pre-existing containment leak size of 20La. Refer to Appendix F for a print-out of the containment isolation fault tree used in this analysis for the CIL node in the large LOCA event trees.

The NPSH top event is also quantified using a fault tree. The NPSH incorporates the fault tree logic to model the probability of being in Plant State 1 or Plant State 2. Refer to Appendix F for a print-out of the fault tree used in this analysis for the NPSH node in the Large LOCA event trees.

3.4.2 PRA Model Modifications for ATWS and SBO

For the ATWS scenarios, COP is modeled as always required for LP ECCS pump NPSH; if COP is unavailable, all LP ECCS pumps drawing from the torus are modeled as failed due to insufficient net positive head. For the SBO scenarios, overpressure is modeled as required after AC power is recovered at t=4 hours.

The following ATWS and SBO initiated sequences in the BFN Unit 1 PRA were modified:

- Turbine Trip ATWS (TTA)
- LOSP ATWS (LOSPA)
- Loss of Condenser Heat Sink ATWS (LOCHSA)
- Inadvertent Opening of SRV ATWS (IOOVA)

- Loss of Feedwater ATWS (LOFWA)
- Loss of Offsite Power (LOSP)
- Loss of 500kV Switchyard to Plant (L500PA)
- Loss of 500kV Switchyard to Unit (L500U)

Similar to the event tree model changes for LLOCA, the ATWS and SBO event trees were modified in order to determine the status of containment integrity (node CIL) prior to questioning the status of low pressure systems drawing from the torus. In the ATWS event trees, failure of the CIL node leads directly to failure of LP ECCS pumps without questioning additional NPSH pre-conditions as is done for the LLOCA scenarios. The same is true for the SBO scenarios, but the scenarios also require COP only after AC power is recovered.

In addition, as discussed previously for the LLOCA scenarios, LPCI and LPCS inter-unit crossties are defeated.

Refer to Appendix E for print-outs of the revised ATWS and SBO event trees.

3.4.3 Quantification of Revised Event Trees

The quantification of the revised model was performed to produce the new CDF. All the new CDF scenarios are those in which the containment is unisolated at t=0 and all RPV injection is lost in the PRA "Early" time frame. Core damage occurs at approximately one hour for the LLOCA and ATWS COP accidents, and in approximately 6 hours for the SBO COP accidents. As such, the additional CDF contributions created by this model manipulation are also all LERF release sequences (i.e., deltaCDF equals deltaLERF). This is a conservative assumption as it assumes that the pre-existing containment leakage of 20La used in the base quantification is representative of a LERF release. Reference [2] determines that a containment leak representative of LERF is >600La.

The quantification results and uncertainty and sensitivity analyses are discussed in Section 4.

The revised BFN Unit 1 PRA RISKMAN model for this base case analysis is archived in file ***U1COP-H*** and saved on the BFN computers along with the other BFN PRA RISKMAN models.

3.5 ASSESSMENT OF LARGE-LATE RELEASES

As discussed above in Section 3.3, all the deltaCDF resulting from this risk assessment also results directly in LERF. As such, there is no increase in Large-Late releases due to scenarios modeling in this risk assessment. Refer to Appendix D for more discussion.

Table 3-1

SUMMARY OF LLOCA NPSH DETERMINISTIC CALCULATIONS(5)

Case(1)	Case Description	Initial Power	Decay Heat	Service Water (SW) Initial Temp (°F)	Suppression Pool (SP) Initial Temp (°F)	Number of RHR pumps in Operation	RHR and CS Pump Flow Rate Per Pump	Number of RHR Heat Exchangers in Operation	Number of RHRSW pumps in Operation	RHRSW Pump Flow Rate Per Pump (gpm)	RHR Heat Exchanger K Value (BTU per sec – °F)	Core Spray Pumps in Operation	Initial SP Water Volume (ft³)	ECCS Strainer Debris Loading	Credit for Containment Heat Sinks	Peak SP Temp (°F)	Containment Overpressure Credit Required(3)
Base Case(2) (GE)	EPU Licensing Calculation – DBA LOCA	102% EPU	ANSI 5.1 w/2σ	95	95	2	Full design	2	2	4000	223	2	121,500	Yes	No	187.3	Yes
Case 1(2) (GE)	DBA Calculation but No Single Failure	102% EPU	ANSI 5.1 w/2σ	95	95	4	Full design	4	4	4000	223	4	121,500	Yes	No	166.4	No
Case 1a(2) (GE)	DBA Calculation but 3 RHR Pumps in Suppression Pool Cooling	102% EPU	ANSI 5.1 w/2σ	95	95	3	Full design	3	3	4000	223	4	121,500	Yes	No	175.0	Yes
Case 1a (TVA) [This case is benchmarked against Case 1a (GE)]	DBA Calculation but 3 RHR Pumps in Suppression Pool Cooling	102% EPU	ANSI 5.1 w/2σ	95	95	3	Full design	3	3	4000	223	4	121,500	Yes	No	175.0	Yes
Case 1b (TVA)	100% Initial Power, RHRSW 89°F, 3 Pumps in Suppression Pool Cooling, K Value 225, 4 CS Pumps	100% EPU	ANSI 5.1 w/o 2σ	89	95	3	Full design	3	3	4000	225	4	121,500	Yes	No	171.0	No
Case 1c (TVA)	100% Initial Power, RHRSW 90°F, 3 Pumps in Suppression Pool Cooling, K Value 225, 4 CS Pumps, Nominal SP WL	100% EPU	ANSI 5.1 w/o 2σ	90	95	3	Full design	3	3	4000	225	4	123,250	Yes	No	170.5	No

Table 3-1

SUMMARY OF LLOCA NPSH DETERMINISTIC CALCULATIONS(5)

Case(1)	Case Description	Initial Power	Decay Heat	Service Water (SW) Initial Temp (ºF)	Suppression Pool (SP) Initial Temp (ºF)	Number of RHR pumps in Operation	RHR and CS Pump Flow Rate Per Pump	Number of RHR Heat Exchangers in Operation	Number of RHRSW pumps in Operation	RHRSW Pump Flow Rate Per Pump (gpm)	RHR Heat Exchanger K Value (BTU per sec – ºF)	Core Spray Pumps in Operation	Initial SP Water Volume (ft³)	ECCS Strainer Debris Loading	Credit for Containment Heat Sinks	Peak SP Temp (ºF)	Containment Overpressure Credit Required(3)
Case 1d (TVA)	DBA Calculation but RHRSW 90ºF, SP Initial Temp 91ºF, 3 Pumps in Suppression Pool Cooling, K Value 225, 4 CS Pumps	102% EPU	ANSI 5.1 w / 2σ	[redacted]	[redacted]	[redacted]	Full design	[redacted]	[redacted]	4000	[redacted]	[redacted]	121,500	Yes	No	171.0	No
Case 1e (TVA)	DBA Calculation but RHRSW 92ºF, SP Initial Temp 90ºF, 3 Pumps in Suppression Pool Cooling, K Value 225, 4 CS Pumps	102% EPU	ANSI 5.1 w / 2σ	[redacted]	[redacted]	[redacted]	Full design	[redacted]	[redacted]	4000	[redacted]	[redacted]	121,500	Yes	No	171.1(4)	No
Case 2 (GE)	DBA Calculation but Initial SW Temperature = 85ºF	102% EPU	ANSI 5.1 w/2σ	[redacted]	95	2	Full design	2	2	4000	223	2	121,500	Yes	No	182.0	Yes
Case 2 (TVA) [This case is benchmarked against Case 2 (GE)]	DBA Calculation but SW Temperature = 85ºF	102% EPU	ANSI 5.1 w/2σ	[redacted]	95	2	Full design	2	2	4000	223	2	121,500	Yes	No	182.2	Yes
Case 2a (GE)	DBA Calculation but Initial SW Temperature = 75ºF	102% EPU	ANSI 5.1 w/2σ	[redacted]	95	2	Full design	2	2	4000	223	2	121,500	Yes	No	177.6	Yes
Case 2b(2) (GE)	DBA Calculation but Initial SW Temperature = 70ºF	102% EPU	ANSI 5.1 w/2σ	[redacted]	95	2	Full design	2	2	4000	223	2	121,500	Yes	No	175.9	Yes
Case 2c (GE)	DBA Calculation but Initial SW Temperature = 65ºF	102% EPU	ANSI 5.1 w/2σ	[redacted]	95	2	Full design	2	2	4000	223	2	121,500	Yes	No	174.3	Yes

Table 3-1

SUMMARY OF LLOCA NPSH DETERMINISTIC CALCULATIONS(5)

Case(1)	Case Description	Initial Power	Decay Heat	Service Water (SW) Initial Temp (°F)	Suppression Pool (SP) Initial Temp (°F)	Number of RHR pumps in Operation	RHR and CS Pump Flow Rate Per Pump	Number of RHR Heat Exchangers in Operation	Number of RHRSW pumps in Operation	RHRSW Pump Flow Rate Per Pump (gpm)	RHR Heat Exchanger K Value (BTU per sec – °F)	Core Spray Pumps in Operation	Initial SP Water Volume (ft³)	ECCS Strainer Debris Loading	Credit for Containment Heat Sinks	Peak SP Temp (°F)	Containment Overpressure Credit Required(3)
Case 2d (TVA)	DBA Calculation but SW Temperature = 65°F, SP Initial Temp 88°F, Nominal SP WL	102% EPU	ANSI 5.1 w/2σ	[redacted]	[redacted]	2	Full design	2	2	4000	223	2	[redacted]	Yes	No	170.6	No
Case 2e (TVA)	DBA Calculation but SW Temperature = 65°F, SP Initial Temp 87°F	102% EPU	ANSI 5.1 w/2σ	[redacted]	[redacted]	2	Full design	2	2	4000	223	2	121,500	Yes	No	170.7	No
Case 2f (TVA)	DBA Calculation but SW Temperature = 68°F, SP Initial Temp 87°F, Nominal SP WL	102% EPU	ANSI 5.1 w/2σ	[redacted]	[redacted]	2	Full design	2	2	4000	223	2	[redacted]	Yes	No	171.1(4)	No
Case 3 (GE)	DBA Calculation but Initial SP Temperature = 85°F	102% EPU	ANSI 5.1 w/2σ	95	[redacted]	2	Full design	2	2	4000	223	2	121,500	Yes	No	183.8	Yes
Case 4 (GE)	100% Initial Power, Minimum SP Level, and No Heat Sink Credit	[redacted]	[redacted]	[redacted]	[redacted]	2	Full design	2	2	4000	[redacted]	2	121,500	Yes	No	177.0	Yes
Case 4 (TVA) [This case is bench-marked against Case 4 (GE)]	100% Initial Power, Minimum SP Level, and No Heat Sink Credit	[redacted]	[redacted]	[redacted]	[redacted]	2	Full design	2	2	4000	[redacted]	2	121,500	Yes	No	177.1	Yes
Case 4a (GE)	100% Initial Power, Nominal SP Level, and Heat Sink Credit	[redacted]	[redacted]	[redacted]	[redacted]	2	Full design	2	2	4000	[redacted]	2	[redacted]	Yes	[redacted]	174.7	Yes

Table 3-1
SUMMARY OF LLOCA NPSH DETERMINISTIC CALCULATIONS(5)

Case(1)	Case Description	Initial Power	Decay Heat	Service Water (SW) Initial Temp (°F)	Suppression Pool (SP) Initial Temp (°F)	Number of RHR pumps in Operation	RHR and CS Pump Flow Rate Per Pump	Number of RHR Heat Exchangers in Operation	Number of RHRSW pumps in Operation	RHRSW Pump Flow Rate Per Pump (gpm)	RHR Heat Exchanger K Value (BTU per sec – °F)	Core Spray Pumps in Operation	Initial SP Water Volume (ft³)	ECCS Strainer Debris Loading	Credit for Containment Heat Sinks	Peak SP Temp (°F)	Containment Overpressure Credit Required(3)
Case 4b(2) (GE)	100% Initial Power, Minimum SP Level, and Heat Sink Credit	100% EPU	ANSI 5.1 w/ 2σ	92	92	2	Full design	2	2	4000	225	2	121,500	Yes	Yes	178.9	Yes
Case 4c(2) (GE)	100% Initial Power, Minimum SP Level, Heat Sink Credit, and SW Temp. that results in Peak SP Temp. equal to/less than 176°F	100% EPU	ANSI 5.1 w/ 2σ	86	92	2	Full design	2	2	4000	225	2	121,500	Yes	Yes	175.8	Yes
Case 4d (TVA)	100% Initial Power, RHRSW 86°F, SP Initial Temp 92°F, K Value 225	100% EPU	ANSI 5.1 w/o 2σ	86	92	2	Full design	2	2	4000	225	2	121,500	Yes	No	177.0	Yes
Case 4e (TVA)	100% Initial Power, RHRSW 86°F, SP Initial Temp 90°F, K Value 225	100% EPU	ANSI 5.1 w/o 2σ	86	90	2	Full design	2	2	4000	225	2	121,500	Yes	No	176.1	Yes
Case 4f (TVA)	100% Initial Power, RHRSW 86°F, SP Initial Temp 90°F, K Value 225, Nominal SP WL	100% EPU	ANSI 5.1 w/o 2σ	86	90	2	Full design	2	2	4000	225	2	123,250	Yes	No	175.6	Yes
Case 4g (TVA)	100% Initial Power, RHRSW 86°F, SP Initial Temp 90°F, K Value 241, Nominal SP WL	100% EPU	ANSI 5.1 w/o 2σ	86	90	2	Full design	2	2	4000	241	2	123,250	Yes	No	173.1	Yes
Case 4h (TVA)	100% Initial Power, RHRSW 85°F, SP Initial Temp 90°F, K Value 225, Nominal SP WL	100% EPU	ANSI 5.1 w/o 2σ	85	90	2	Full design	2	2	4000	225	2	123,250	Yes	No	175.1	Yes

Table 3-1

SUMMARY OF LLOCA NPSH DETERMINISTIC CALCULATIONS(5)

Case(1)	Case Description	Initial Power	Decay Heat	Service Water (SW) Initial Temp (°F)	Suppression Pool (SP) Initial Temp (°F)	Number of RHR pumps in Operation	RHR and CS Pump Flow Rate Per Pump	Number of RHR Heat Exchangers in Operation	Number of RHRSW pumps in Operation	RHRSW Pump Flow Rate Per Pump (gpm)	RHR Heat Exchanger K Value (BTU per sec – °F)	Core Spray Pumps in Operation	Initial SP Water Volume (ft³)	ECCS Strainer Debris Loading	Credit for Containment Heat Sinks	Peak SP Temp (°F)	Containment Overpressure Credit Required(3)
Case 4i (TVA)	100% Initial Power, RHRSW 85°F, SP Initial Temp 86°F, K Value 241, Nominal SP WL	100% EPU	ANSI 5.1 w/o 2σ	85	86	2	Full design	2	2	4000	241	2	123,250	Yes	No	170.8	No
Case 4j (TVA)	100% Initial Power, RHRSW 85°F, SP Initial Temp 88°F, K Value 241, Nominal SP WL	100% EPU	ANSI 5.1 w/o 2σ	85	88	2	Full design	2	2	4000	241	2	125,640	Yes	No	171.0	No

Notes to Table 3-1:

(1) Column information includes designation of organization that performed the calculation.
(2) Case verified by formal analysis.
(3) COP credit required for peak suppression pool temperature of 171°F.
(4) This value is acceptable for demonstrating sensitivity analysis results.
(5) Shaded areas in the table "highlight" differences from the Base Case.

Section 4

RESULTS

4.1 QUANTITATIVE RESULTS

The results of the base quantification of this risk assessment case are summarized in Table 4-1.

As discussed in Section 3, the additional CDF contributions created by this model manipulation are also all LERF release sequences (i.e., deltaCDF equals deltaLERF).

These very low results are expected and are well within the RG 1.174 guidelines (refer to Figures 2-1 and 2-2) for "very small" risk impact. If greater detail was included to address some of the conservative assumptions in this risk assessment (e.g., 2 sigma decay heat assumed with a probability of 1.0 given 102% EPU power exists; refer to Section 3.2), the deltaCDF and deltaLERF would be even lower.

4.2 UNCERTAINTY ANALYSIS

To provide additional information for the decision making process, the risk assessment provided here is supplemented by parametric uncertainty analysis and quantitative and qualitative sensitivity studies to assess the sensitivity of the calculated risk results.

Uncertainty is categorized here into the following three types, consistent with PRA industry literature:

- Parametric
- Modeling
- Completeness

Parametric uncertainties are those related to the values of the fundamental parameters of the PRA model, such as equipment failure rates, initiating event frequencies, and human error probabilities. Typical of standard industry practices, the parametric uncertainty aspect is assessed here by performing a Monte Carlo parametric uncertainty propagation analysis. Probability distributions are assigned to each parameter value, and a Monte Carlo sampling code is used to sample each parameter and propagate the parametric distributions through to the final results. The parametric uncertainty analysis and associated results are discussed further below.

Modeling uncertainty is focused on the structure and assumptions inherent in the risk model. The structure of mathematical models used to represent scenarios and phenomena of interest is a source of uncertainty, due to the fact that models are a simplified representation of a real-world system. Model uncertainty is addressed here by the identification and quantification of focused sensitivity studies. The model uncertainty analysis and associated results are discussed further below.

Completeness uncertainty is primarily concerned with scope limitations. Scope limitations are addressed here by the qualitative assessment of the impact on the conclusions if external events and shutdown risk contributors are also considered. The completeness uncertainty analysis is discussed further below.

4.2.1 Parametric Uncertainty Analysis

The parametric uncertainty analysis for this risk assessment was performed using the RISKMAN computer program to calculate probability distributions and determine the uncertainty in the accident frequency estimate.

RISKMAN has three analysis modules: Data Analysis Module, System Analysis Module, and Event Tree Analysis Module. Appropriate probability distributions for each uncertain parameter in the analysis is determined and included in the Data Module. The System

Module combines the individual failure rates, maintenance, and common cause parameters into the split fraction frequencies that will be used by the Event Tree Module. A Monte Carlo routine is used with the complete distributions to calculate the split fraction frequencies. Event trees are quantified and linked together in the Event Module. The important sequences from the results of the Event Tree Module are used in another Monte Carlo sampling step to propagate the split fraction uncertainties and obtain the uncertainties in the overall results.

The descriptive statistics calculated by RISKMAN for the total core damage frequency of the plant caused by internal events include:

- Mean of the sample
- Variance of the sample
- 5th, 50th, and 95th percentiles of the sample

The parametric uncertainty associated with delta core damage frequency calculated in this assessment is presented as a comparison of the RISKMAN calculated CDF uncertainty statistics for the Unit 1 base EPU PRA and the Unit 1 EPU COP Credit LLOCA quantification. The results are shown in Table 4-2.

It should be cautioned that this distribution is developed via Monte Carlo (random) sampling, and as such it is dependent upon the number of samples and the initial numerical seed values of the sampling routine. Neither the initial seeds nor the number of samples used for the model of record are known. Consequently, some variation from the base model statistics is expected. Taking these cautions into consideration, a comparison of the distributions by percentiles shows little if any change. Based on this result, parametric uncertainty analysis for the ATWS and SBO accidents is not necessary as the conclusion would be the same (i.e., very little distribution change, such that delta CDF and delta LERF results would remain well within the RG 1.174 guidelines for "very small" risk impact).

4.2.2 Modeling Uncertainty Analysis

As stated previously, modeling uncertainty is concerned with the sensitivity of the results due to uncertainties in the structure and assumptions in the logic model. Modeling uncertainty has not been explicitly treated in many PRAs, and is still an evolving area of analysis. The PRA industry is currently investigating methods for performing modeling uncertainty analysis. EPRI has developed a guideline for modeling uncertainty that is still in draft form and undergoing pilot testing. The EPRI approach that is currently being tested takes the rational approach of identifying key sources of modeling uncertainty and then performing appropriate sensitivity calculations. This approach is taken here.

The modeling issues selected here for assessment are those related to the risk assessment of the containment overpressure credit. This assessment does not involve investigating modeling uncertainty with regard to the overall BFN PRA. The modeling issues identified for sensitivity analysis are:

- Pre-existing containment leakage size and associated probability
- Calculation of containment isolation system failure
- Assessment of power and water temperature and level pre-conditions
- Number of RHR pumps and heat exchangers in SPC

Pre-Existing Containment Leakage Size/Probability

The base case analysis assumes a pre-existing containment leakage pathway leakage size of 20La that would result in defeat of the necessary containment overpressure credit.

A larger pre-existing leak size of 100La, consistent with the EPRI 1009325 recommended assumption for a "large" leak, is used in this sensitivity to defeat the necessary COP credit. From EPRI 1009325, the probability of a pre-existing 100La containment leakage pathway at any given time at power is 2.47E-04.

Calculation of Containment Isolation System Failure

The base case quantification uses the containment isolation system failure fault tree logic to represent failure of the containment isolation system. The fault tree specifically analyzes primary containment penetrations greater than 3" diameter. This modeling sensitivity case expands the scope of the containment isolation fault tree to include smaller lines as potential defeats of COP credit. This sensitivity is performed by increasing by a factor of 10 the failure probability associated with the containment isolation system. Refer to Table F-1 for the CIL event tree node failure probability used.

Assessment of Power and Water Temperature and Level Pre-conditions

This is a conservative sensitivity that assumes that all that is necessary for failure of the low pressure ECCS pumps due to inadequate NPSH during a large LOCA is an unisolated containment. This sensitivity is performed by assuming the other pre-conditions represented by the top event NSPH exist with a probability of 1.0.

Number of RHR pumps and heat exchangers in SPC

The base case LLOCA COP credit quantification addresses the situation in which 2 or less RHR pumps and heat exchangers are operating in SPC mode. The likelihood of failing any two RHR pumps during the 24-hr PRA mission time is approximately 8.2E-3. The likelihood of an unisolated containment given an accident initiator is approximately 2.2E-3, and the likelihood of other necessary extreme plant conditions (e.g., high river temperature, high reactor power, reduced suppression pool water level) existing at the

time of the LLOCA is approximately 2.4E-3. As such, the base quantification results in an approximate 4.3E-8 conditional probability, given a LLOCA, of loss of low pressure ECCS pumps due to insufficient NPSH due to inadequate COP.

This sensitivity discusses the risk impact of also explicitly quantifying LLOCA scenarios with only 1 or no RHR pumps failed. Such scenarios are not explicitly included in the base quantification because their risk contribution is non-significant, as shown by the sensitivities discussed here. As shown in Table 3-1, even with very conservative assumptions, if 3 or more RHR pumps and heat exchangers are operating in SPC mode during a LLOCA, there is no need for containment overpressure. To result in a need for COP credit in such cases would require even more conservative input assumptions than the 2 RHR pump scenario. As such, the additional risk from such scenarios is non-significant compared to the 2 RHR pump case explicitly modeled in this analysis.

An estimate of the deltaCDF risk contribution for the scenario with 3 RHR pumps in SPC operation can be approximated as follows (refer to Case 1d in Table 3-1):

- Sum of BFN PRA Large LOCA initiator frequencies: 3E-5/yr
- Likelihood of failure of 1 RHR pump or 1 RHR heat exchanger during the 24-hr PRA mission time: 1.00E-2 (nominal estimate)
- Probability of 102% EPU initial power level: 5E-3 (same as base analysis)
- Probability of containment isolation failure given an accident initiator: 3E-3 (nominal from base analysis)
- Probability of river water temperature ≥90°F at any given time: 9E-2 (nominal value based on Table C-1. Although the river temperature has not exceeded 90°F based on the collected plant data, statistically there is a non-zero likelihood of such a temperature).
- Conditional probability that suppression pool water temperature ≥ 91°F given river water temperature ≥ 90°F: 1.0 (refer to Figure C-1).
- No probabilistic credit for low suppression pool volume or low heat exchanger effectiveness is taken here.

- deltaCDF contribution for 3 RHR pump case: 3E-5 x 1E-2 x 5E-3 x 3E-3 x 9E-2 x 1.0 = ~ 4E-13/yr

This additional contribution to the calculated deltaCDF from a 3 RHR pump LLOCA case is non-significant in comparison to the 2 RHR pump LLOCA case.

An estimate of the deltaCDF risk contribution for the scenario with 4 RHR pumps in operation can be approximated as follows (refer to Case 1 of Table 3-1):

- Sum of BFN PRA Large LOCA initiator frequencies: 3E-5/yr
- Likelihood of 4 RHR pumps and 4 heat exchangers in SPC during Large LOCA: 1.0 (nominal estimate)
- Probability of 102% EPU initial power level: 5E-3 (same as base analysis)
- Probability of containment isolation failure given an accident initiator: 3E-3 (nominal from base analysis)
- Probability of river water temperature $\geq$ 100°F at any given time: 1E-3 (estimate based on Table C-1. Although the river temperature has not exceeded 90°F based on the collected plant data, statistically there is a non-zero likelihood of such a temperature). 100°F is assumed here as the river water temperature at which COP credit is required (refer to Case 1 of Table 3-1).
- Conditional probability that suppression pool water temperature $\geq$ 95°F given river water temperature $\geq$ 100°F: 1.0 (refer to Figure C-1).
- No probabilistic credit for low suppression pool volume or low heat exchanger effectiveness is taken here.
- deltaCDF contribution for 3 RHR pump case: 3.1E-5 x 1.0 x 5E-3 x 3E-3 x 1E-3 x 1.0 = ~5E-13/yr

Similar to the 3 pump case discussed previously, this additional contribution to the calculated deltaCDF from a 4 RHR pump LLOCA case is non-significant in comparison to the 2 RHR pump LLOCA case.

Summary of Modeling Uncertainty Results

The modeling uncertainty sensitivity cases are summarized in Table 4-3.

4.2.3 Completeness Uncertainty Analysis

As stated previously, completeness uncertainty is addressed here by the qualitative assessment of the impact on the conclusions if external events and shutdown risk contributors are also considered.

Seismic

The BFN seismic risk analysis was performed as part of the Individual Plant Examination of External Events (IPEEE). BFN performed a seismic margins assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequency sequences were quantified as part of the seismic risk evaluation.

The conclusions of the SMA are judged to be unaffected by the EPU or the containment overpressure credit issue. The EPU has little or no impact on the seismic qualifications of the systems, structures and components (SSCs). Specifically, the power uprate results in additional thermal energy stored in the RPV, but the additional blowdown loads on the RPV and containment given a coincident seismic event, are judged not to alter the results of the SMA.

The decrease in time available for operator actions, and the associated increases in calculated HEPs, is judged to have a non-significant impact on seismic-induced risk. Industry BWR seismic PSAs have typically shown (e.g., Peach Bottom NUREG-1150 study; Limerick Generating Station Severe Accident Risk Assessment; NUREG/CR-

4448) that seismic risk is overwhelmingly dominated by seismic induced equipment and structural failures. Seismic induced failures of containment are low likelihood scenarios, and such postulated scenarios are moot for the COP question because they would be analyzed in a seismic PRA as core damage scenarios directly.

Based on the above discussion, it is judged that seismic issues do not significantly impact the decision making for the BFN EPU and containment overpressure credit.

Internal Fires

The BFN fire risk analysis was performed as part of the Individual Plant Examination of External Events (IPEEE). BFN performed a screening methodology using the EPRI FIVE (Fire Induced Vulnerability Evaluation) methodology.

Like most plants, BFN currently does not maintain a fire PRA. However, given the very low risk impact of the COP credit, even if fire risk was explicitly quantified the conclusions of this risk assessment are not expected to change, i.e., the risk impact is very small.

Other External Hazards

In addition to seismic events and internal fires, the BFN IPEEE Submittal analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Floods
- Transportation and Nearby Facility Accidents
- Other External Hazards

The BFN IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that BFN meets the applicable NRC Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards. As such, these other external hazards are judged not to significantly impact the decision making for the BFN EPU and containment overpressure credit.

Shutdown Risk

As discussed in the BFN EPU submittal, shutdown risk is a non-significant contributor to the risk profile of the proposed EPU. The credit for containment overpressure is not required for accident sequences occurring during shutdown. As such, shutdown risk does not influence the decision making for the BFN EPU containment overpressure credit.

4.3 APPLICABILITY TO BFN UNIT 2 AND UNIT 3

This risk assessment was performed using the BFN Unit 1 PRA. To assess the applicability of the Unit 1 results to BFN Units 2 and 3, the BFN Unit 3 PRA was reviewed. The Unit 3 PRA was explicitly reviewed because it has a higher base CDF than the Unit 2 PRA due to fewer inter-unit crosstie capabilities than Unit 2.

Review of the Unit 3 PRA models did not identify any differences that would make the Unit 1 PRA results and conclusions not applicable to Units 2 and 3. As further evidence, the Unit 3 PRA was modified in a similar manner as the Unit 1 sensitivity Case #2 and the Unit 3 LLOCA scenarios were quantified to determine the ΔCDF impact. The result for Unit 3 was a deltaCDF of 1.9E-9/yr, which is comparable to the U-1 LLOCA COP delta CDF contribution of 1.5E-9/yr for sensitivity case #2 . The revised BFN Unit 3 PRA RISKMAN model supporting this review is archived in file

U3COP2-9 and saved on the BFN computers along with the other BFN PRA RISKMAN models.

Given the above, the results for the Unit 1 PRA risk assessment are comparable to the Units 2 and 3 PRAs.

The U2/U3 assessment discussed in this sub-section was performed for the Rev. 0 analysis. Given the similar results obtained in Rev. 2 analysis using the U-1 model, the U2/U3 assessment discussed above was not re-performed as the conclusion would be the same.

Table 4-1

BASE CASE RESULTS

TYPE	DESCRIPTION	Total CDF	Total LERF	ΔCDF[(2), (3)]	ΔLERF[(2), (3)]
LLOCA[(1)]	Large LOCAs. All large LOCA initiated scenarios (except ISLOCAs and LOCAs Outside Containment, because these result in deposition of decay heat directly outside the containment and not into the suppression pool).	1.77E-06	4.41E-07	1.39E-09	1.39E-09
ATWS[(1)]	Transient without SCRAM. All PRA ATWS scenarios (i.e., transients, LOOP, and IORV ATWS scenarios) modified to require COP credit. Low pressure ECCS pumps failed if containment isolation is failed.	1.77E-06	4.48E-07	8.17E-09	8.17E-09
SBO[(1)]	Station black out with recovery of power after 4 hours. Low pressure ECCS pumps failed when AC power recovered if containment isolation is failed.	1.78E-06	4.54E-07	1.47E-08	1.47E-08
	TOTAL Results for LLOCA, ATWS and SBO	1.79E-06	4.64E-07	2.43E-08	2.43E-08

Notes:

(1) The results in the top three rows are for each identified group of accident scenarios quantified in isolation and the resulting impact on CDF and LERF. The combined CDF and LERF impact for all three accident scenario types is provided in the bottom row.

(2) The ΔCDF and ΔLERF values are with respect to the BFN Unit 1 PRA model of record CDF of 1.767E-6/yr and LERF of 4.397E-7/yr.

(3) The results presented above are conservative due to the nature of the RISKMAN quantification. The addition of new nodes or top events to event trees (as is done in this analysis) causes previously existing sequences to split into two or more new sequences. The quantification initiator cutoff limit in the COP calculations was reduced (from the base cutoff of 1E-12 to 1E-13) to capture the new sequences added to the model. The reduced cutoff limit in the revised model captures the new low frequency sequences, but also results in capturing sequences that are truncated in the base BFN model; as such, the resultant ΔCDF and ΔLERF values (which are calculated as the new PRA value minus the base PRA value) shown here are overstated.

Table 4-2

PARAMETRIC UNCERTAINTY ANALYSIS RESULTS

Statistic	BFN Unit 1 Base CDF	BFN Unit 1 COP LLOCA CDF[1]
5%	4.71E-7	5.15E-7
50%	1.23E-6	1.23E-6
MEAN	1.77E-6	1.77E-6
95%	4.72E-6	4.47E-6

Notes:

(1) Parametric uncertainty analysis performed on the LLOCA accident sequence impact. Similar results expected for ATWS and SBO sequences (i.e., little change, such that delta CDF and delta LERF results would remain well within the RG 1.174 guidelines for "very small" risk impact).

Table 4-3

SUMMARY OF SENSITIVITY QUANTIFICATIONS

Case	Description	CDF	LERF	ΔCDF[(2), (3)]	ΔLERF[(2), (3)]
Base[(1)]	Base Case Quantification (20 La leak size)	1.791E-06	4.640E-07	2.4E-08	2.4E-08
1[(1)]	Pre-Existing Containment Leakage Sufficient to Fail COP Credit Defined by 100La	1.771E-06	4.441E-07	4.4E-09	4.4E-09
2[(1)]	Assume Low Suppression Pool Water Volume (123,500 ft3) Exists 100% of the Time	1.791E-06	4.642E-07	2.4E-08	2.4E-08
3[(1)]	Expansion of Containment Isolation fault tree to Encompass Smaller Lines (approximate by multiplying Cont. Isol. failure probability by 10x)	1.793E-06	4.656E-07	2.6E-08	2.6E-08
4[(1)]	Assume Initial Power Level and Water Temperature and Level Pre-Conditions Exist 100% of the Time	1.793E-06	4.661E-07	2.6E-08	2.6E-08
5[(1)]	Combination of Cases #3 and #4	1.798E-06	4.708E-07	3.1E-08	3.1E-08
6	Incorporation of "3-RHR pumps in SPC" and "4-RHR pumps in SPC" loss of NPSH scenarios	1.791E-06	4.640E-07	2.4E-08	2.4E-08

Notes:

(1) Scenarios with failure of 2 or more RHR pumps and associated heat exchangers in SPC are explicitly analyzed in these cases. As shown in Case 6, explicit incorporation of scenarios with 0 or 1 RHR pumps in SPC failed has a negligible impact on the results.

(2) The ΔCDF and ΔLERF values are with respect to the BFN Unit 1 PRA model of record CDF of 1.767E-6/yr and LERF of 4.397E-7/yr.

(3) The results presented above are conservative due to the nature of the RISKMAN quantification. The addition of new nodes or top events to event trees (as is done in this analysis) causes previously existing sequences to split into two or more new sequences. The quantification initiator cutoff limit in the COP calculations was reduced (from the base cutoff of 1E-12 to 1E-13) to capture the new sequences added to the model. The reduced cutoff limit in the revised model captures the new low frequency sequences, but also results in capturing sequences that are truncated in the base BFN model; as such, the resultant ΔCDF and ΔLERF values (which are calculated as the new PRA value minus the base PRA value) shown here are overstated.

Section 5

CONCLUSIONS

The report documents the risk impact of utilizing containment accident pressure (containment overpressure) to satisfy the net positive suction head (NPSH) requirements for RHR and Core Spray pumps during DBA LOCA, ATWS and SBO accident scenarios.

The need for COP credit requests is driven by the conservative nature of accident calculations. Use of more realistic inputs in such calculations shows that no credit for COP is required.

The conclusions of the plant internal events risk associated with this assessment are as follows.

1) Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr. Based on this criteria, the proposed change (i.e., use of COP to satisfy the net positive suction head (NPSH) requirements for RHR and Core Spray pumps) represents a very small change in CDF (2.4E-08/yr).

2) Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of Large Early Release Frequency (LERF) below 10^{-7}/yr. Based on this criteria, the proposed change (i.e., use of COP to satisfy the net positive suction head (NPSH) requirements for RHR and Core Spray pumps) represents a very small change in LERF (2.4E-08/yr).

These results are well within the guideline of RG 1.174 for a "very small" risk increase. Even when modeling uncertainty and parametric uncertainty, and external event scenarios are considered, the risk increase is small. As such, the credit for COP in

determining adequate NPSH for low pressure ECCS pumps during DBA LOCA, ATWS and SBO accidents is acceptable from a risk perspective.

The conclusion that the risk impact from the EPU COP credit is very small, applies to BFN Unit 1 as well as BFN Units 2 and 3.

REFERENCES

[1] "Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Technical Specifications (TS) Change 448 – One-Time Frequency Extension For Containment Integrated Leakage Rate Test (ILRT) Interval", TVA-BFN-TS-448, July 8, 2004.

[2] Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, EPRI Report 1009325, Final Report, December 2003.

[3] "Project Task Report – Browns Ferry Units 1, 2 & 3 EPU, RAI Response – NPSH Sensitivity Studies", GE Nuclear Energy, GE-NE-0000-0050-00443-R0-Draft, February 2006.

[4] Letter from G.B. Wallis (Chairman, ACRS) to N.J. Diaz (Chairman, NRC), "Vermont Yankee Extended Power Uprate", ACRSR-2174, January 4, 2006.

Appendix A
PRA QUALITY

The BFN Unit 1 EPU PRA was used in this analysis for the base case quantification as it was recently updated consistent with the ASME PRA Standard and it is representative of each of the three BFN unit PRAs. The following discusses the quality of the BFN Unit 1 PRA models used in performing the risk assessment crediting containment overpressure for RHR and Core Spray pump NPSH requirements:

- Level of detail in PRA
- Maintenance of the PRA
- Comprehensive Critical Reviews

A.1 LEVEL OF DETAIL

The BFN Unit 1 PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events.

The PRA model (Level 1 and Level 2) used for the containment overpressure risk assessment was the most recent internal events risk model for the BFN Unit 1 plant at EPU conditions (BFN model U1050517). The BFN PRA models adopts the large event tree / small fault tree approach and use the support state methodology, contained in the RISKMAN code, for quantifying core damage frequency.

The PRA model contains the following modeling attributes.

A.1.1 Initiating Events

The BFN at-power PRA explicitly models a large number of internal initiating events:

- General transients
- LOCAs
- Support system failures
- Internal Flooding events

The initiating events explicitly modeled in the BFN at-power PRA are summarized in Table A-1. The number of internal initiating events modeled in the BFN at-power PRA is similar to or greater than the majority of U.S. BWR PRAs currently in use.

A.1.2 System Models

The BFN at-power PRA explicitly models a large number of frontline and support systems that are credited in the accident sequence analyses. The BFN systems explicitly modeled in the BFN at-power PRA are summarized in Table A-2. The number and level of detail of plant systems modeled in the BFN at-power PRA is equal to or greater than the majority of U.S. BWR PRAs currently in use.

A.1.3 Operator Actions

The BFN at-power PRA explicitly models a large number of operator actions:

- Pre-Initiator actions
- Post-Initiator actions
- Recovery Actions
- Dependent Human Actions

Approximately fifty operator actions are explicitly modeled in the BFN PRA. A summary table of the individual actions modeled is not provided here.

The human error probabilities for the actions are modeled with accepted industry HRA techniques.

The BFN PRA includes an explicit assessment of the dependence of post-initiator operator actions. The approach used to assess the level of dependence between operator actions is based on the method presented in the NUREG/CR-1278 and EPRI TR-100259.

The number of operator actions modeled in the BFN at-power PRA, and the level of detail of the HRA, is consistent with that of other U.S. BWR PRAs currently in use.

A.1.4 Common Cause Events

The BFN at-power PRA explicitly models a large number of common cause component failures. Approximately two thousand common cause terms are included in the BFN Unit 1 PRA. Given the large number of CCF terms modeled in the BFN at-power internal events PRA, a summary table of them is not provided here. The number and level of detail of common cause component failures modeled in the BFN at-power PRA is equal to or greater than the majority of U.S. BWR PRAs currently in use.

A.1.5 Level 2 PRA

The BFN Unit 1 Level 2 PRA is designed to calculate the LERF frequency consistent with NRC Regulatory Guidance (e.g. Reg. Guides 1.174 and 1.177) and the PRA Application Guide.

The Level 2 PRA model is a containment event tree (CET) that takes as input the core damage accident sequences and then questions the following issues applicable to LERF:

- Primary containment isolation
- RPV depressurization post-core damage
- Recovery of damaged core in-vessel
- Energetic containment failure phenomena at or about time of RPV breach
- Injection established to drywell for ex-vessel core debris cooling/scrubbing
- Containment flooding
- Drywell failure location
- Wetwell failure location
- Effectiveness of secondary containment in release scrubbing

The following aspects of the Level 2 model reflect the more than adequate level of detail and scope:

1. Dependencies from Level 1 accidents are carried forward directly into the Level 2 by transfer of sequences to ensure that their effects on Level 2 response are accurately treated.
2. Key phenomena identified by the NRC and industry for inclusion in BWR Level 2 LERF analyses are treated explicitly within the model.
3. The model quantification truncation is sufficiently low to ensure adequate convergence of the LERF frequency.

A.2 MAINTENANCE OF PRA

The BFN PRA models and documentation are maintained living and are routinely updated to reflect the current plant configuration following refueling outages and to reflect the accumulation of additional plant operating history and component failure data.

The PRA Update Report is evaluated for updating every other refueling outage. The administrative guidance for this activity is contained in a TVA Procedure.

In addition, the PRA models are routinely implemented and studied by plant PRA personnel in the performance of their duties. Potential model modifications or enhancements are itemized and maintained for further investigation and subsequent implementation, if warranted. Potential modifications identified as significant to the results or applications may be implemented in the model at the time the change occurs if their impact is significant enough to warrant.

A.2.1 History of BFN PRA Models

The current BFN Unit 1 PRA is the model used for this analysis. The BFN Unit 1 PRA was initially developed in June 2004 using the guidance in the ASME PRA Standard, and to incorporate the latest plant configuration (including EPU) and operating experience data. The Unit 1 PRA was then subsequently updated in August 2005. The Unit 1 PRA was developed using the BFN Unit 2 and Unit 3 PRAs as a starting point. The BFN Unit 2 and Unit 3 PRAs have been updated numerous times since the original IPE Submittal. The BFN Unit 2 PRA revisions are summarized below:

Original BFN IPE Submittal	9/92
Revision to address plant changes and incorporate BFN IE and EDG experience data	8/94
Revision to ensure consistency with the BFN Multi-Unit PRA	4/95
Revision to address PER BFPER 970754	10/97
2002 PRA Update	3/02
2004 PRA Update (includes conditions to reflect EPU)	6/04
2005 Update	8/05

A.3 COMPREHENSIVE CRITICAL REVIEWS

As described above, the BFN Unit 1 PRA used in this analysis was built on more than 10 years of analysis effort and experience associated with the Unit 2 and 3 PRAs.

During November 1997, TVA participated in a PRA Peer Review Certification of the Browns Ferry Unit 2 and 3 PRAs administered under the auspices of the BWROG Peer Certification Committee. The purpose of the peer review process is to establish a method of assessing the technical quality of the PRA for its potential applications. The elements of the PRA reviewed are summarized in Tables A-3 through A-4.

The Peer Review evaluation process utilized a tiered approach using standardized checklists allowing a detailed review of the elements and the sub-elements of the Browns Ferry PSAs to identify strengths and areas that need improvement. The review system used allowed the Peer Review team to focus on technical issues and to issue their assessment results in the form of a "grade" of 1 through 4 on a PRA sub-element level. To reasonably span the spectrum of potential PRA applications, the four grades of certification as defined by the BWROG document "Report to the Industry on PRA Peer Review Certification Process - Pilot Plant Results" were employed.

During the Unit 2 and 3 PSAs updates in 2003, the significant findings (i.e., designated as Level A or B) from the Peer Certification were resolved, resulting in the PRA elements now having a minimum certification grade of 3. The Unit 1 PRA used in this analysis has incorporated the findings of the Units 2 and 3 PSA Peer Review. The previously conducted Peer Review was effectively an administrative and technical Peer Review of the Unit 1 PRA. Similar models, processes, policies, approaches, reviews, and management oversight were utilized to develop the Unit 1 PRA.

A.4 PRA QUALITY SUMMARY

The quality of modeling and documentation of the BFN PRA models has been demonstrated by the foregoing discussions on the following aspects:

- Level of detail in PRA
- Maintenance of the PRA
- Comprehensive Critical Reviews

The BFN Unit 1 Level 1 and Level 2 PRAs provide the necessary and sufficient scope and level of detail to allow the calculation of CDF and LERF changes due to the risk assessment requiring containment overpressure for sufficient NPSH for the low pressure ECCS pumps.

Table A-1

INITIATING EVENTS FOR BFN PRA

Initiator Category	Mean Frequency (events per year)
Transient Initiator Categories	
Inadvertent Opening of One SRV	1.36E-2
Spurious Scram at Power	8.76E-2
Loss of 500kV Switchyard to Plant	1.02E-2
Loss of 500kV Switchyard to Unit	2.37E-2
Loss of Instrumentation and Control Bus 1A	4.27E-3
Loss of Instrumentation and Control Bus 1B	4.27E-3
Total Loss of Condensate Flow	9.45E-3
Partial Loss of Condensate Flow	1.93E-2
MSIV Closure	5.52E-2
Turbine Bypass Unavailable	1.95E-3
Loss of Condenser Vacuum	9.70E-2
Total Loss of Feedwater	2.58E-2
Partial Loss of Feedwater	2.47E-1
Loss of Plant Control Air	1.20E-2
Loss of Offsite Power	7.87E-3
Loss of Raw Cooling Water	7.95E-3
Momentary Loss of Offsite Power	7.57E-3
Turbine Trip	5.50E-1
High Pressure Trip	4.29E-2
Excessive Feedwater Flow	2.78E-2
Other Transients	8.60E-2
ATWS Categories	
Turbine Trip ATWS	5.50E-1
LOSP ATWS	7.87E-3
Loss of Condenser Heat Sink ATWS	1.52E-1
Inadvertent Opening of SRV ATWS	1.36E-2
Loss of Feedwater ATWS	3.02E-1
LOCA Initiator Categories	
Breaks Outside Containment	6.67E-4
Excessive LOCA (reactor vessel failure)	9.39E-9
Interfacing Systems LOCA	3.15E-5

Table A-1

INITIATING EVENTS FOR BFN PRA

Initiator Category	**Mean Frequency (events per year)**
Large LOCA – Core Spray Line Break	
Loop I	1.68E-6
Loop II	1.68E-6
Large LOCA – Recirculation Discharge Line Break	
Loop A	1.18E-5
Loop B	1.18E-5
Large LOCA – Recirculation Suction Line Break	
Loop A	8.39E-7
Loop B	8.39E-7
Other Large LOCA	8.39E-7
Medium LOCA Inside Containment	3.80E-5
Small LOCA Inside Containment	4.75E-4
Very Small LOCA Inside Containment	5.76E-3
Internal Flooding Initiator Categories	
EECW Flood in Reactor Building – shutdown units	1.20E-3
EECW Flood in Reactor Building – operating unit	1.85E-6
Flood from the Condensate Storage Tank	1.22E-4
Flood from the Torus	1.22E-4
Large Turbine Building Flood	3.65E-3
Small Turbine Building Flood	1.65E-2

Table A-2
BFN PRA MODELED SYSTEMS

120V and 250V DC Electric Power
AC Electric Power
ARI and RPT
Condensate Storage Tank
Condensate System
Containment Atmospheric Dilution
Control Rod Drive Hydraulic
Core Spray System
Drywell Control Air
Emergency Diesel Generators
Emergency Equipment Cooling Water
Feedwater System
Fire Protection System (for alternative RPV injection)
Hardened Wetwell Vent
High Pressure Coolant Injection
Main Steam System
Plant Air Systems
Primary Containment Isolation
Raw Cooling Water
Reactor Building Closed Cooling Water
Reactor Core Isolation Cooling
Reactor Protection System
Recirculation System
Residual Heat Removal System
RHR Service Water
Secondary Containment Isolation
Shared Actuation Instrumentation System
SRVs / ADS
Standby Gas Treatment System
Standby Liquid Control System

Table A-2

BFN PRA MODELED SYSTEMS

Suppression Pool / Vapor Suppression

Turbine Bypass and Main Condenser

Table A-3

PRA PEER REVIEW TECHNICAL ELEMENTS FOR LEVEL 1

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Initiating Events	• Guidance Documents for Initiating Event Analysis • Groupings - Transient - LOCA - Support System/Special - ISLOCA - Break Outside Containment - Internal Floods • Subsumed Events • Data • Documentation
Accident Sequence Evaluation (Event Trees)	• Guidance on Development of Event Trees • Event Trees (Accident Scenario Evaluation) - Transients - SBO - LOCA - ATWS - Special - ISLOCA/BOC - Internal Floods • Success Criteria and Bases • Interface with EOPs/AOPs • Accident Sequence Plant Damage States • Documentation

Table A-3

PRA PEER REVIEW TECHNICAL ELEMENTS FOR LEVEL 1

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Thermal Hydraulic Analysis	• Guidance Document • Best Estimate Calculations (e.g., MAAP) • Generic Assessments • FSAR - Chapter 15 • Room Heat Up Calculations • Documentation
System Analysis (Fault Trees)	• System Analysis Guidance Document(s) • System Models - Structure of models - Level of Detail - Success Criteria - Nomenclature - Data (see Data Input) - Dependencies (see Dependency Element) - Assumptions • Documentation of System Notebooks

Table A-3

PRA PEER REVIEW TECHNICAL ELEMENTS FOR LEVEL 1

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Data Analysis	• Guidance • Component Failure Probabilities • System/Train Maintenance Unavailabilities • Common Cause Failure Probabilities • Unique Unavailabilities or Modeling Items - AC Recovery - Scram System - EDG Mission Time - Repair and Recovery Model - SORV - LOOP Given Transient - BOP Unavailability - Pipe Rupture Failure Probability • Documentation
Human Reliability Analysis	• Guidance • Pre-Initiator Human Actions - Identification - Analysis - Quantification • Post-Initiator Human Actions and Recovery - Identification - Analysis - Quantification • Dependence among Actions • Documentation

Table A-3

PRA PEER REVIEW TECHNICAL ELEMENTS FOR LEVEL 1

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Dependencies	• Guidance Document on Dependency Treatment • Intersystem Dependencies • Treatment of Human Interactions (see also HRA) • Treatment of Common Cause • Treatment of Spatial Dependencies • Walkdown Results • Documentation
Structural Capability	• Guidance • RPV Capability (pressure and temperature) - ATWS - Transient • Containment (pressure and temperature) • Reactor Building • Pipe Overpressurization for ISLOCA • Documentation
Quantification/Results Interpretation	• Guidance • Computer Code • Simplified Model (e.g., cutset model usage) • Dominant Sequences/Cutsets • Non-Dominant Sequences/Cutsets • Recovery Analysis • Truncation • Uncertainty • Results Summary

Table A-4

PRA CERTIFICATION TECHNICAL ELEMENTS FOR LEVEL 2

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Containment Performance Analysis	• Guidance Document • Success Criteria • L1/L2 Interface • Phenomena Considered • Important HEPs • Containment Capability Assessment • End state Definition • LERF Definition • CETs • Documentation

Table A-5

PRA CERTIFICATION TECHNICAL ELEMENTS
FOR MAINTENANCE AND UPDATE PROCESS

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Maintenance and Update Process	• Guidance Document • Input - Monitoring and Collecting New Information • Model Control • PRA Maintenance and Update Process • Evaluation of Results • Re-evaluation of Past PRA Applications • Documentation

Appendix B

PROBABILITY OF PRE-EXISTING CONTAINMENT LEAKAGE

Containment failures that may be postulated to defeat the containment overpressure credit include containment isolation system failures (refer to Appendix D) and pre-existing unisolable containment leakage pathways. The pre-existing containment leakage probability used in this analysis is obtained from EPRI 1009325, Risk Impact of Assessment of Extended Integrated Leak Rate Testing Intervals.[2] This is the same approach as used in the recent 2005 Vermont Yankee EPU COP analyses, and accepted by the NRC and ACRS. [4]

EPRI 1009325 provides a framework for assessing the risk impact for extending integrated leak rate test (ILRT) surveillance intervals. EPRI 1009325 includes a compilation of industry containment leakage events, from which an assessment was performed of the likelihood of a pre-existing unisolable containment leakage pathway.

A total of seventy-one (71) containment leakage or degraded liner events were compiled. Approximately half (32 of the 71 events) had identified leakage rates of less than or equal to 1La (i.e., the Technical Specification containment allowed leakage rate). None of the 71 events had identified leakage rates greater than 21La. EPRI 1009325 employed industry experts to review and categorize the industry events, and then various statistical methods were used to assess the data. The resulting probabilities as a function of pre-existing leakage size are summarized here in Table B-1.

The EPRI 1009325 study used 100La as a conservative estimate of the leakage size that would represent a large early release pathway consistent with the LERF risk measure, but estimated that leakages greater than 600La are a more realistic representation of a large early release.

This analysis is not concerned per se about the size of a leakage pathway that would represent a LERF release, but rather a leakage size that would defeat the containment overpressure credit. Given the low likelihood of such a leakage, the exact size is not key to this risk assessment, and no detailed calculation of the exact hole size is performed here. The recent COP risk assessment for the Vermont Yankee Mark I BWR plant, presented to the ACRS in November and December 2005, determined a leakage size of 27La using the conservative 10CFR50, Appendix K containment analysis approach. Earlier ILRT industry guidance (NEI Interim Guidance – see Ref. 10 of EPRI 1009325) conservatively recommended use of 10La to represent "small" containment leakages and 35La to represent "large" containment leakages.

Given the above, the base analysis here assumes 20La as the size of a pre-existing containment leakage pathway sufficient to defeat the containment overpressure credit. Such a hole size does not realistically represent a LERF release (based on EPRI 1009325) and is also believed (based on the VY hole size estimate) to be on the low end of a hole size that would preclude containment overpressure credit. As can be seen from Table B-1, the probability of a 20La pre-existing containment leakage at any given time at power is 1.88E-03.

Sensitivity studies to the base case quantification (refer to Section 4) assess the sensitivity of the results to the pre-existing leakage size assumption.

Table B-1

PROBABILITY OF PRE-EXISTING UNISOLABLE CONTAINMENT LEAK [2]
(as a Function of Leakage Size)(1)

Leakage Size (La)	Mean Probability of Occurrence
1	2.65E-02
2	1.59E-02
5	7.42E-03
10	3.88E-03
20	1.88E-03
35	9.86E-04
50	6.33E-04
100	2.47E-04
200	8.57E-05
500	1.75E-05
600	1.24E-05

Notes:

(1) Reference [2] recommends these values for use for both BWRs and PWRs. Reference [2] makes no specific allowance for the fact that inerted BWRs, such as BFN, could be argued to have lower probabilities of significant pre-existing containment leakages.

Appendix C
ASSESSMENT OF BROWNS FERRY DATA

Variations in river and suppression pool water temperatures, and the suppression pool level at the Browns Ferry plant were statistically analyzed. The purpose of this data assessment is to estimate for use in the risk assessment the realistic probability that the water temperatures and level will exceed a given value, i.e. the probability of exceedance.

C.1 BFN EXPERIENCE DATA

The following sets of river water inlet daily temperature, suppression pool water daily temperature, and suppression pool daily level data were obtained and reviewed:

Data	Unit	Data Period	Years
River Water Temperature and Suppression Pool Temperature	2	01/01/00 – 01/31/06	6.1
	3	02/01/03 – 01/31/06	3.0
Suppression Pool Level	2	01/01/00 – 01/31/06	6.1
	3	02/01/03 – 01/31/06	3.0

The river water temperature data from the above units is not pooled because river temperature is dependent upon the seasonal cycle in weather and is not independent between the units. Use of data for SW inlet temperatures from multiple units would incorrectly assume the sets of data are independent when in fact they are directly dependent upon weather and the common river source. As such, the statistical assessment of the river water temperature variation uses the largest set of data (i.e., the 6.1 years of data from the Unit 2 river water inlet).

As the torus water temperature has a high dependence on river water temperature for most of the year, the assessment of the torus temperature variability also is based on the 6.1 year data set from Unit 2.

The variation in torus level as experienced by Units 2 and 3 can approximate the level range expected to be seen in Unit 1. As such, the statistical assessment of suppression pool level is based on the level data sets from both units. This creates the largest pool of data and will best approximate the variation in level expected from Unit 1 once it begins operation.

C.2 STATISTICAL ANALYSIS OF TEMPERATURE DATA

The chronological variation in river water temperature and torus water temperature is plotted together on the graph shown in Figure C-1. As can be seen from Figure C-1, the torus water temperature is always equal to or higher than the river water temperature. Also, the river water temperatures and torus temperatures are closely correlated in the warmer months when river water temperature is above approximately 70°F.

The 6.1 years of temperature data was categorized into 5-degree temperature bins ranging from 50°F to 99°F degrees. The resulting histograms are shown in Figures C-2 and C-3. Figure C-2 presents histogram for the river water temperature and Figure C-3 presents the histogram for the torus water temperature.

The histogram information was then used in a statistical analysis software package (Crystal Ball, a MS Excel add-in, developed by Decisioneering, Inc. of Denver, CO) to approximate a distribution of the expected range in temperature.

The Crystal Ball software automatically tests a number of curve fits. The best fit for the temperature data is a normal distribution that is truncated at user-defined upper and

lower bounds. If upper and lower bounds are not defined, the tails of the curve fit distribution extend to unrealistic values (e.g., river water and torus water temperatures below 0°F degrees). To constrain the distributions, the following user-defined upper and lower bounds were used:

- River water temperature lower bound of 32°F (no data points in the 6.1 years of data reached 32°F, only a single data point reached 35°F)
- River water temperature upper bound of 95°F (no data points in the 6.1 years of data exceeded 90°F)
- Torus water temperature lower bound of 55°F (no data points in the 6.1 years of data reached lower than 57°F)
- Torus water temperature upper bound of 95°F (only a single data point in the 6.1 years of data reached 93°F)

The Crystal Ball software statistical results for the river water temperature and torus water temperature variations are provided in Figures C-4 and C-5, respectively.

The statistical results are also summarized in the form of exceedance probability as a function of temperature in Figures C-6 and C-7. The information is also presented in tabular form, Tables C-1 and C-2. As discussed previously, the river water and the torus water temperature variations are not independent; as such, the exceedance frequencies are not independent (i.e., they should not be multiplied together directly to determine the probability of exceeding a particular temperature in the river AND at the same time exceeding particular temperature in the torus).

C.2.1 Conditional Probability of Torus Water Temperature

One of the parameters used in this risk assessment is the conditional probability that the torus water temperature is greater than or equal to 87°F given river water temperature is greater than or equal to 68°F. Plant data for Units 2 and 3 were reviewed to

determine this conditional probability. The same data period used for the river water and torus temperature is used in this calculation and both units worth of data is pooled. A simple likelihood estimate was performed. The following table lists the number of data records where river water temperature was greater than 68°F, and of those records, the number of records where the torus temperature exceeded 87°F.

	River >= 68F	Torus>=87F	Cond Prob.
Unit 2	1103	512	4.6E-1
Unit 3	566	225	4.0E-1
Combined	1669	737	4.42E-1

As the table shows, the likelihood of the torus being greater than 87°F when the river temperature is greater than 68°F is 4.42E-1.

C.3 STATISTICAL ANALYSIS OF SP LEVEL DATA

The 9.1 years of Browns Ferry Unit 2 and Unit 3 suppression pool level data was categorized into 0.25 inch water level bins ranging from -1.00 inches to -6.25 inches. Browns Ferry operating instructions require that suppression pool water level remain between these values. The plant is not allowed to remain at power if suppression pool water level falls outside this range. Data points far outside the -1.00 to -6.25 inch range are not included in the statistical analysis because they reflect levels experienced when the plant was shutdown (which is a plant state inapplicable to this risk assessment). Approximately 53 level data points were not included.

The resulting suppression pool level histogram is shown in Figure C-8.

The histogram was then input into the Crystal Ball software tool to approximate a distribution of the expected range in suppression pool level. The Crystal Ball software statistical results for suppression pool level variations are provided in Figure C-9.

The statistical results are also summarized in the form of probability as a function of suppression pool level in Figures C-10. The information is also presented in tabular form in Table C-3.

Figure C-1

CHRONOLOGICAL VARIABILITY IN RIVER WATER AND TORUS WATER TEMPERATURES

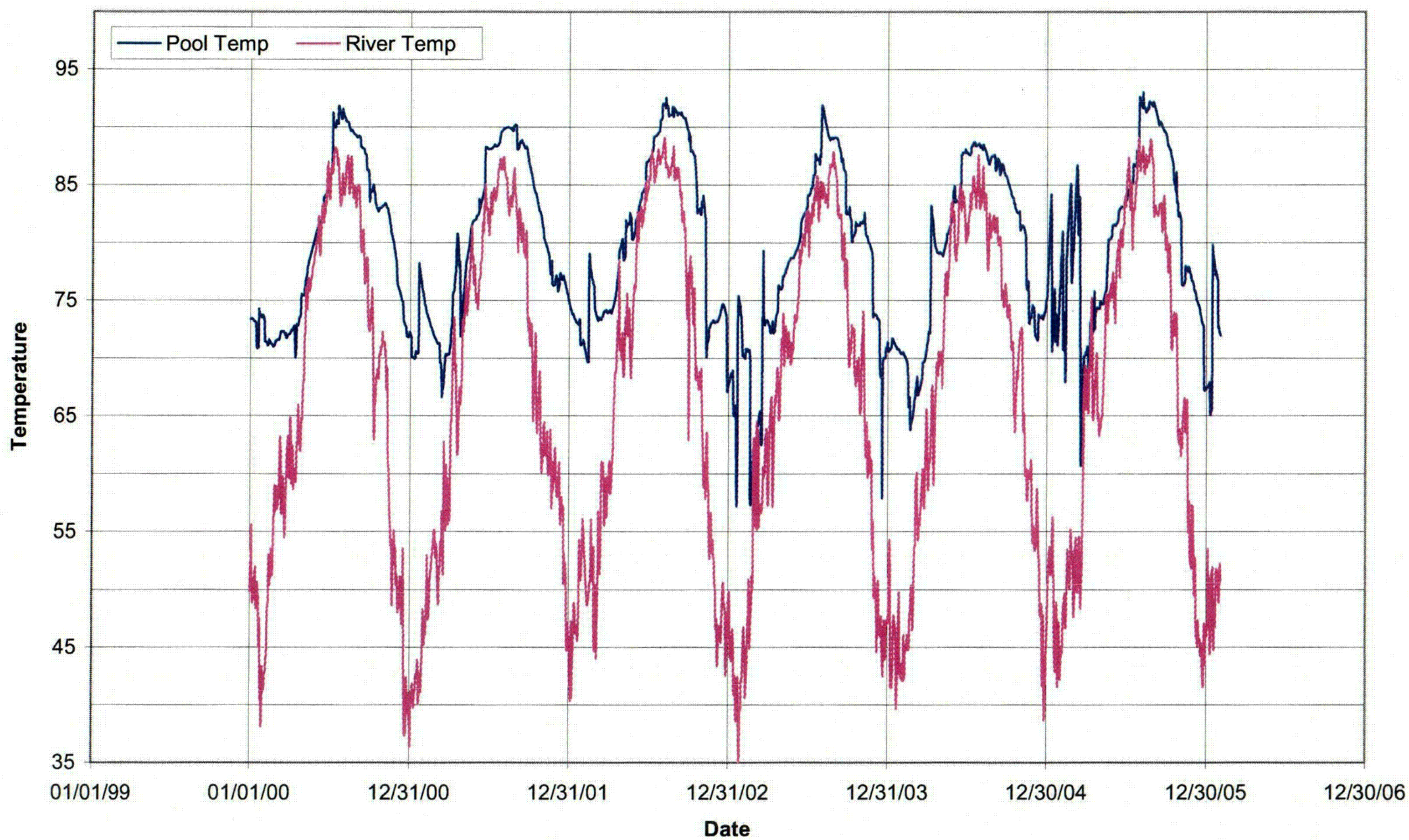


Figure C-2

RIVER WATER TEMPERATURE HISTOGRAM

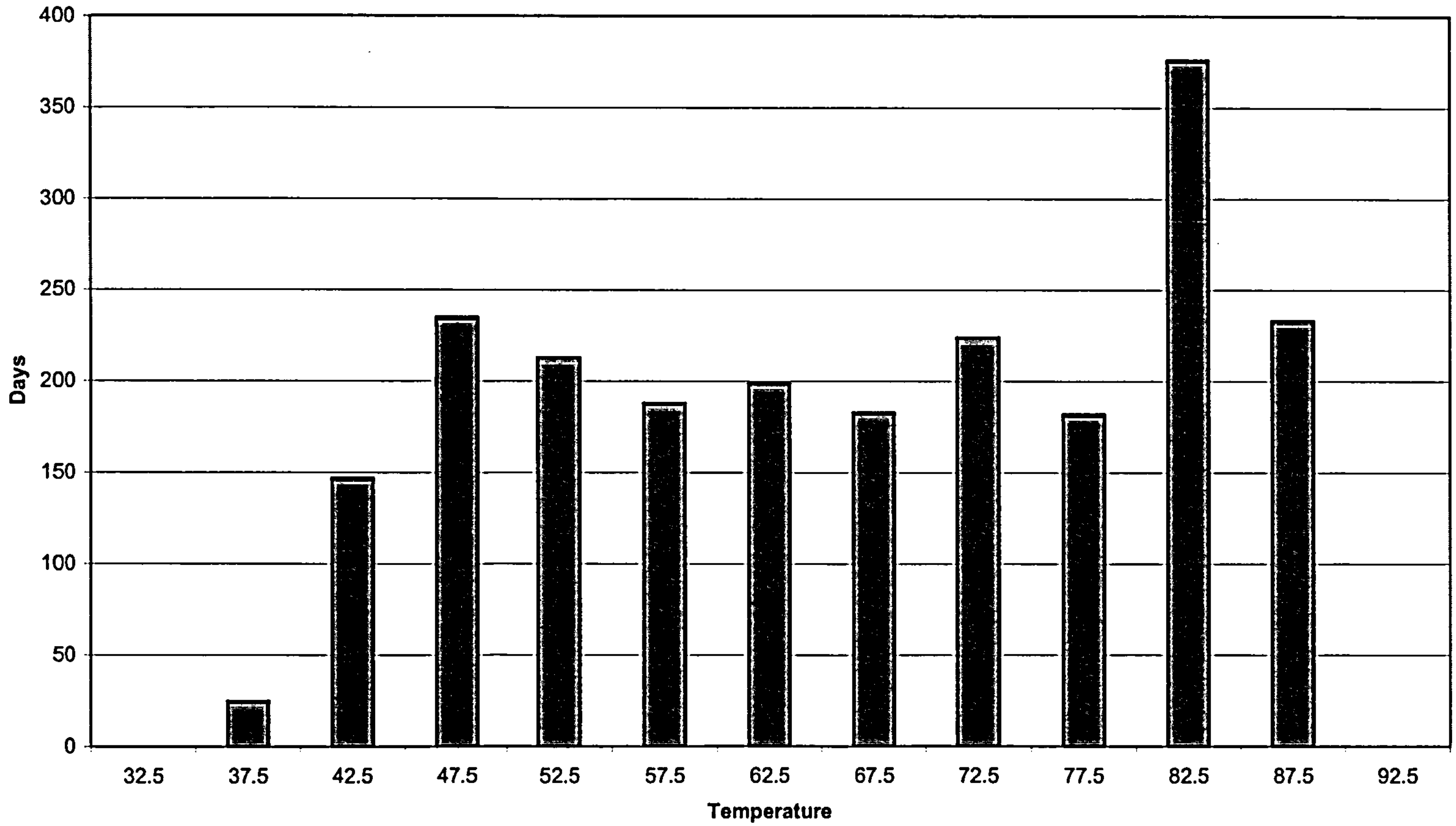


Figure C-3

TORUS TEMPERATURE HISTOGRAM

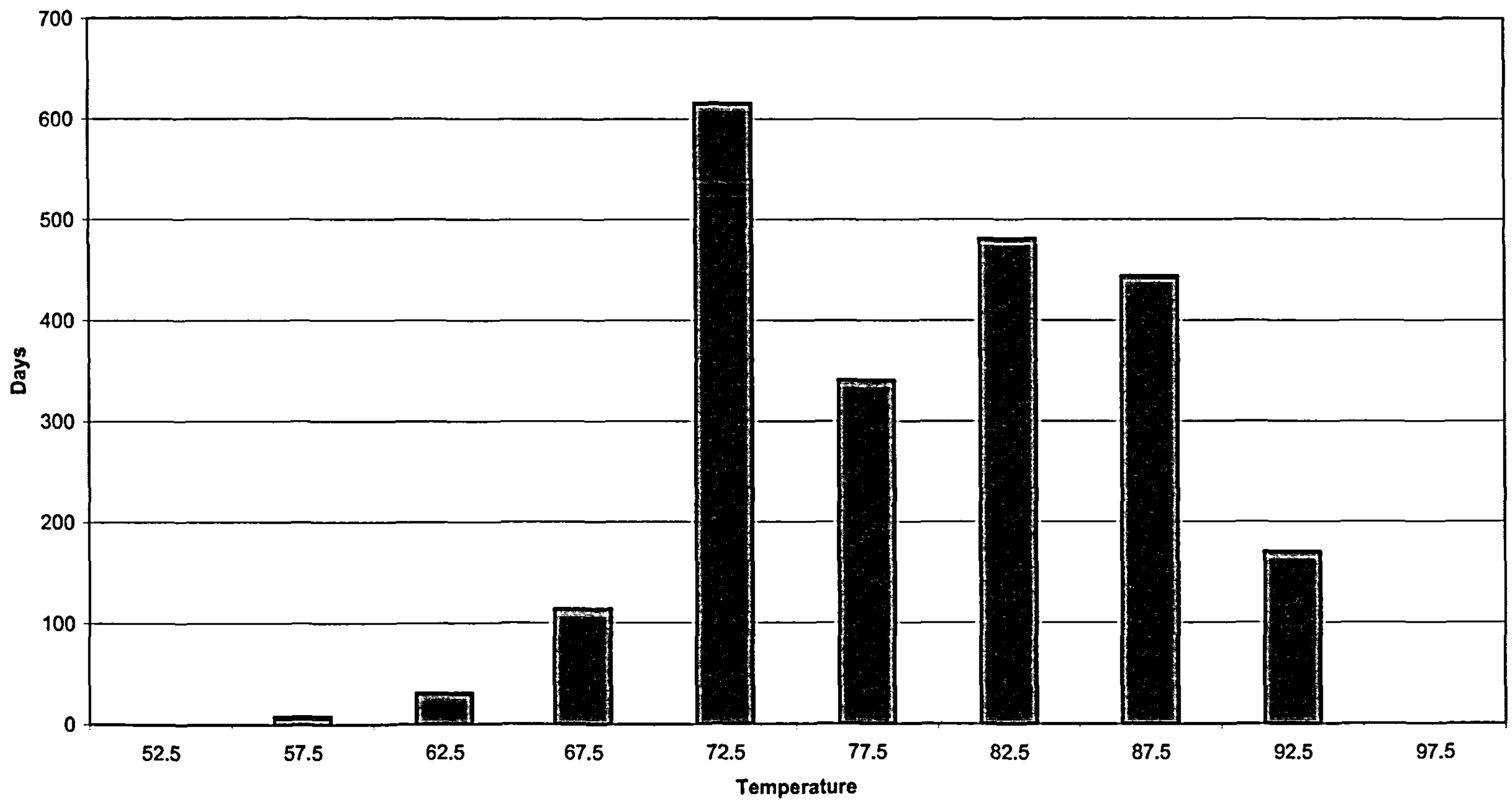


Figure C-4

STATISTICAL RESULTS FOR RIVER WATER TEMPERATURE VARIATION

Crystal Ball Report

Simulation started on 2/6/06 at 7:09:56

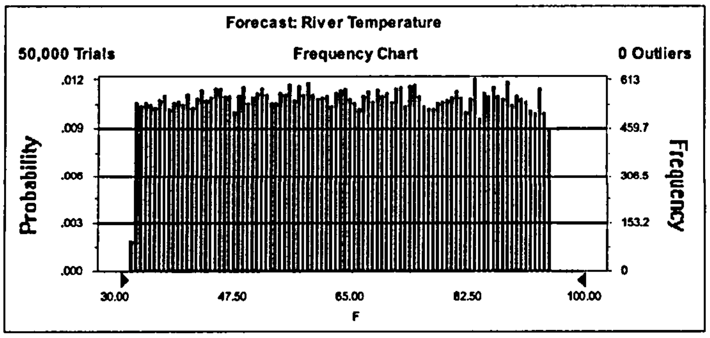
Simulation stopped on 2/6/06 at 7:11:44

Forecast: River Temperature **Cell: G18**

Summary:

Display Range is from 30.00 to 100.00 F
Entire Range is from 32.00 to 95.00 F
After 50,000 Trials, the Std. Error of the Mean is 0.08

Statistics:	Value
Trials	50000
Mean	63.50
Median	63.41
Mode	---
Standard Deviation	18.07
Variance	326.51
Skewness	0.00
Kurtosis	1.81
Coeff. of Variability	0.28
Range Minimum	32.00
Range Maximum	95.00
Range Width	63.00
Mean Std. Error	0.08



Percentiles:

Percentile	F
0.0%	32.00
2.5%	33.60
5.0%	35.25
50.0%	63.41
95.0%	91.69
97.5%	93.32
100.0%	95.00

Figure C-5

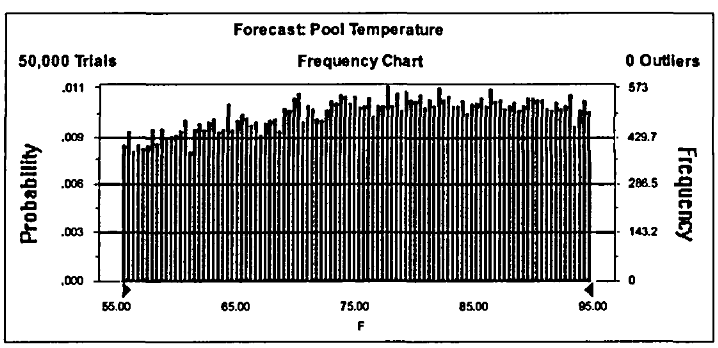
STATISTICAL RESULTS FOR TORUS WATER TEMPERATURE VARIATION

Crystal Ball Report
Simulation started on 2/6/06 at 7:09:56
Simulation stopped on 2/6/06 at 7:11:44

Forecast: Pool Temperature **Cell: C15**

Summary:
Display Range is from 55.00 to 95.00 F
Entire Range is from 55.00 to 95.00 F
After 50,000 Trials, the Std. Error of the Mean is 0.05

Statistics:	Value
Trials	50000
Mean	75.75
Median	76.06
Mode	---
Standard Deviation	11.30
Variance	127.65
Skewness	-0.08
Kurtosis	1.85
Coeff. of Variability	0.15
Range Minimum	55.00
Range Maximum	95.00
Range Width	40.00
Mean Std. Error	0.05



Percentiles:

Percentile	F
0.0%	55.00
2.5%	56.22
5.0%	57.46
50.0%	76.06
95.0%	93.04
97.5%	94.02
100.0%	95.00

Figure C-6

RIVER WATER TEMPERATURE EXCEEDANCE PROBABILITY

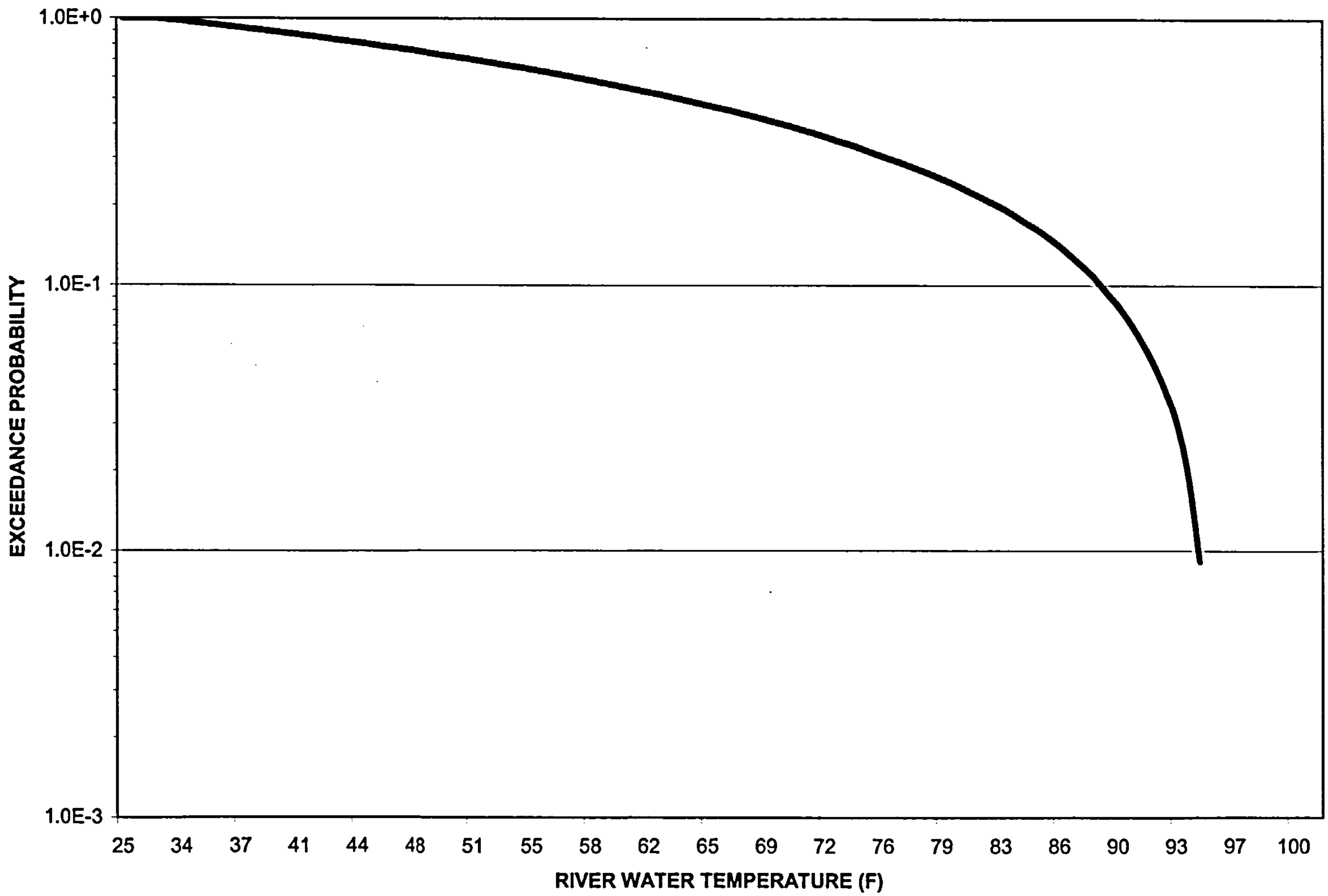


Figure C-7

TORUS WATER TEMPERATURE EXCEEDANCE PROBABILITY

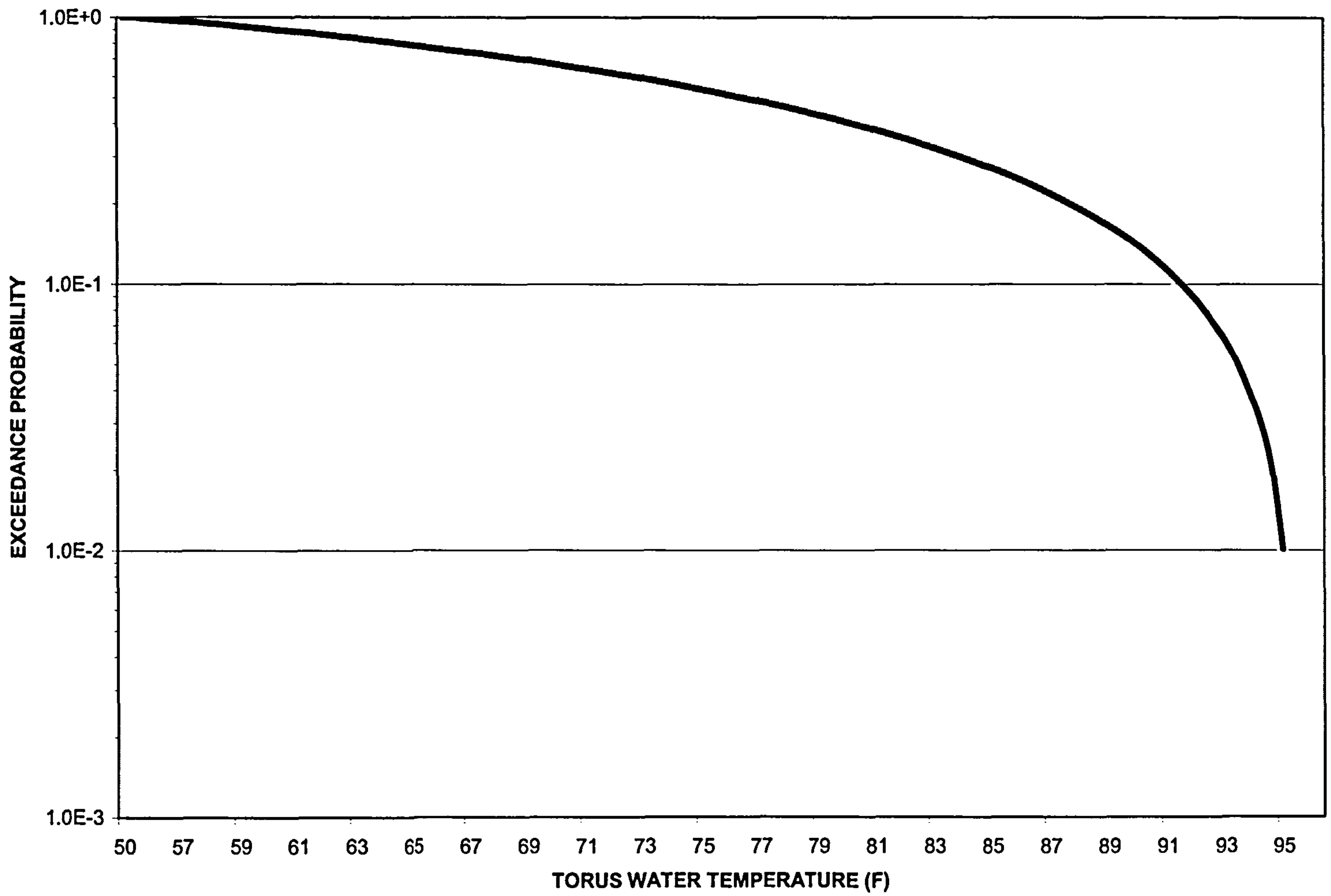


Figure C-8

SUPPRESSION POOL LEVEL HISTOGRAM

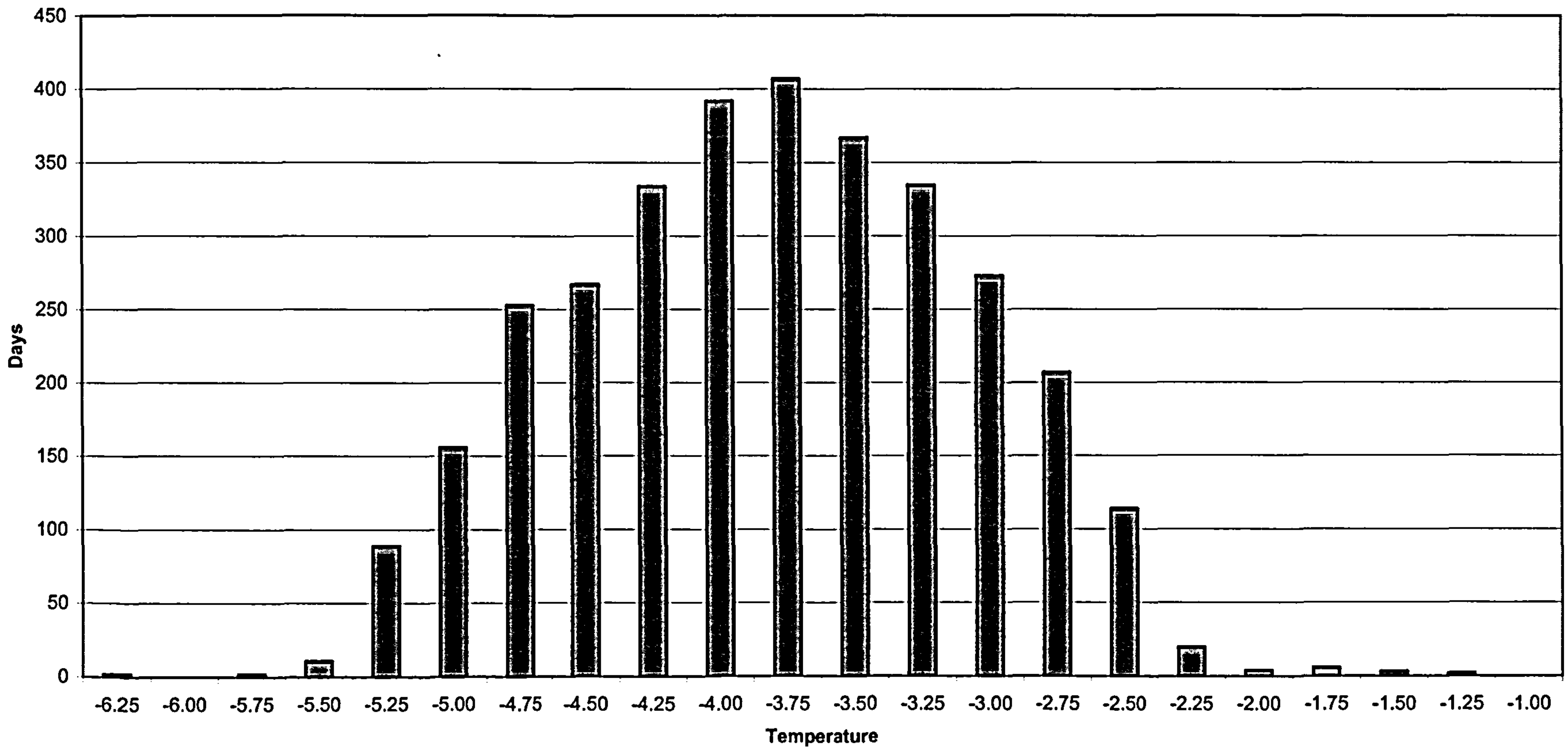


Figure C-9

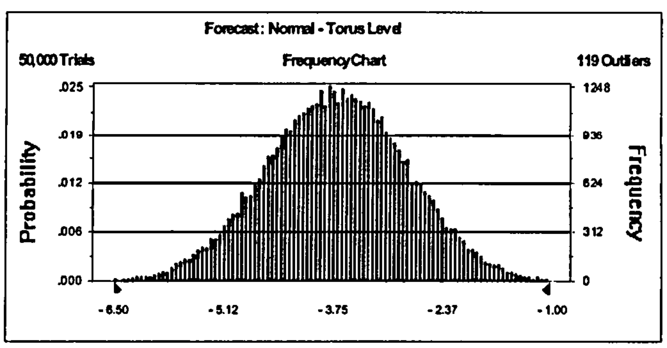
STATISTICAL RESULTS FOR TORUS WATER LEVEL VARIATION

Crystal Ball Report
Simulation started on 3/7/06 at 15:33:31
Simulation stopped on 3/7/06 at 15:58:17

Forecast: Normal - Torus Level **Cell: F3**

Summary:
Display Range is from -6.50 to -1.00
Entire Range is from -7.71 to -0.11
After 50,000 Trials, the Std. Error of the Mean is 0.00

Statistics:	Value
Trials	50000
Mean	-3.68
Median	-3.68
Mode	---
Standard Deviation	0.90
Variance	0.81
Skewness	-0.01
Kurtosis	3.00
Coeff. of Variability	0.27
Range Minimum	-7.71
Range Maximum	-0.11
Range Width	7.60
Mean Std. Error	0.00



Percentiles:

Percentile	Value
0.0%	-7.71
2.5%	-5.45
5.0%	-5.16
50.0%	-3.68
95.0%	-2.20
97.5%	-1.92
100.0%	-0.11

Figure C-10

SUPPRESSION POOL WATER LEVEL PROBABILITY
(Probability that Level Below Value of Interest)

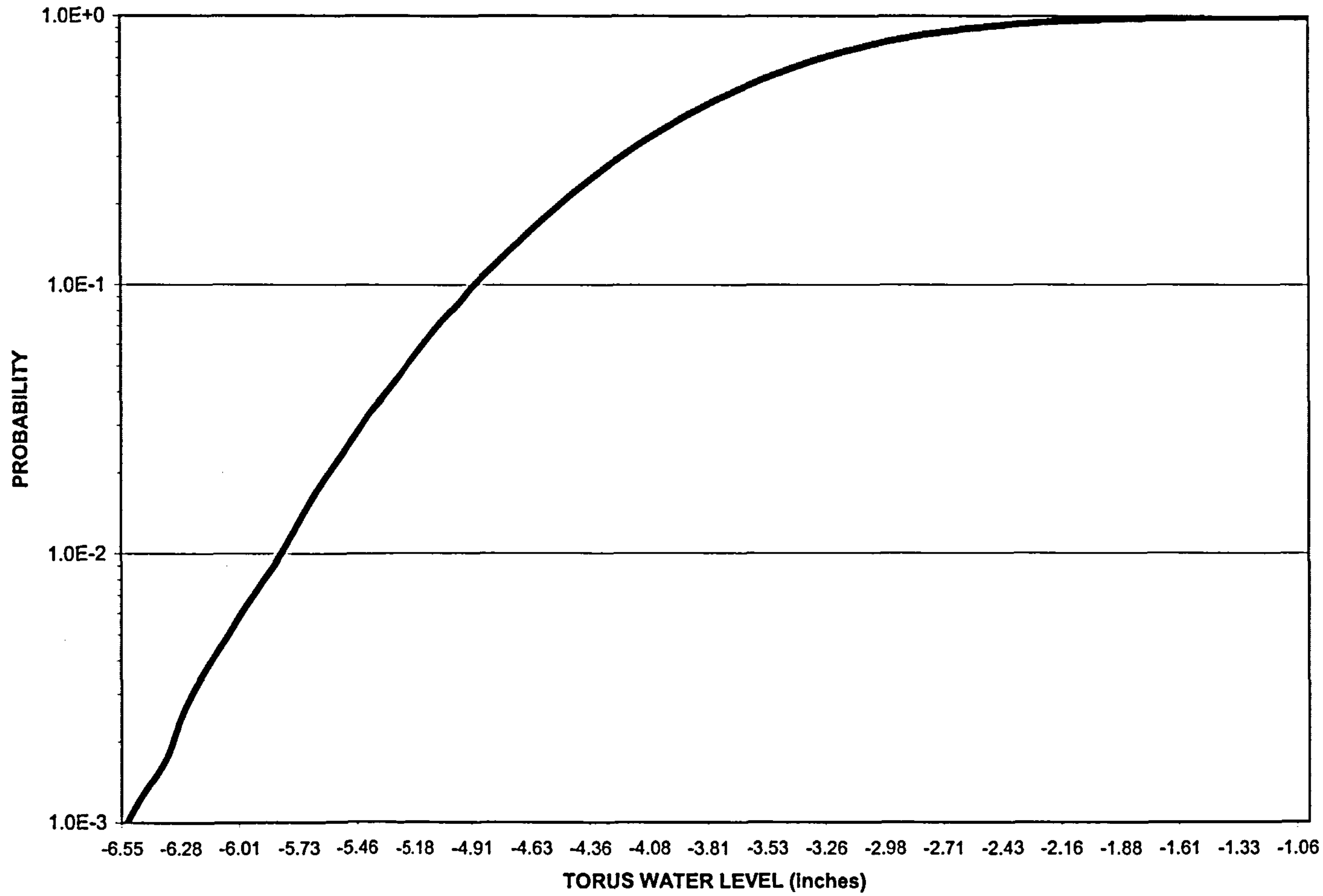


Table C-1

RIVER WATER TEMPERATURE EXCEEDANCE PROBABILITIES

Temperature (°F)	Exceedance Probability
30	1.00E+00
35	9.55E-01
40	8.80E-01
45	8.02E-01
50	7.24E-01
55	6.45E-01
60	5.64E-01
65	4.74E-01
70	3.97E-01
75	3.17E-01
80	2.41E-01
85	1.64E-01
86	1.40E-01
90	8.46E-02
95	9.15E-03
100	0.00E+00

Table C-2

TORUS WATER TEMPERATURE EXCEEDANCE PROBABILITIES

Temperature (°F)	Exceedance Probability
30	1.00E+00
35	1.00E+00
40	1.00E+00
45	1.00E+00
50	1.00E+00
55	1.00E+00
60	8.90E-01
65	7.79E-01
70	6.63E-01
75	5.28E-01
80	4.01E-01
85	2.62E-01
90	1.35E-01
92	8.25E-02
95	1.01E-02
100	0.00E+00

Table C-3

SUPPRESSION POOL WATER LEVEL PROBABILITY
(Probability that Level Below Value of Interest)

Level (inches)	Probability
-6.50	1.10E-03
-6.45	1.30E-03
-6.39	1.50E-03
-6.34	1.80E-03
-6.28	2.40E-03
-6.23	3.00E-03
-6.17	3.60E-03
-6.12	4.20E-03
-6.06	4.90E-03
-6.01	5.80E-03
-5.95	6.80E-03
-5.90	7.90E-03
-5.84	9.10E-03
-5.79	1.08E-02
-5.73	1.30E-02
-5.70	1.45E-02(1)
-5.68	1.55E-02
-5.62	1.83E-02
-5.57	2.11E-02
-5.51	2.44E-02
-5.46	2.84E-02
-5.40	3.28E-02
-5.35	3.71E-02
-5.29	4.24E-02
-5.24	4.78E-02
-5.18	5.38E-02
-5.13	6.09E-02
-5.07	6.88E-02
-5.02	7.73E-02
-4.96	8.60E-02
-4.91	9.72E-02

Table C-3

SUPPRESSION POOL WATER LEVEL PROBABILITY
(Probability that Level Below Value of Interest)

Level (inches)	Probability
-4.85	1.08E-01
-4.80	1.19E-01
-4.74	1.31E-01
-4.69	1.44E-01
-4.63	1.58E-01
-4.58	1.74E-01
-4.52	1.90E-01
-4.47	2.07E-01
-4.41	2.26E-01
-4.36	2.45E-01
-4.30	2.64E-01
-4.25	2.85E-01
-4.19	3.07E-01
-4.14	3.29E-01
-4.08	3.51E-01
-4.03	3.74E-01
-3.97	3.96E-01
-3.92	4.21E-01
-3.86	4.44E-01
-3.81	4.69E-01
-3.75	4.93E-01
-3.70	5.16E-01
-3.64	5.41E-01
-3.59	5.64E-01
-3.53	5.88E-01
-3.48	6.12E-01
-3.42	6.35E-01
-3.37	6.58E-01
-3.31	6.81E-01
-3.26	7.03E-01
-3.20	7.24E-01

Table C-3

SUPPRESSION POOL WATER LEVEL PROBABILITY
(Probability that Level Below Value of Interest)

Level (inches)	Probability
-3.15	7.45E-01
-3.09	7.64E-01
-3.04	7.83E-01
-2.98	8.00E-01
-2.93	8.17E-01
-2.87	8.32E-01
-2.82	8.47E-01
-2.76	8.60E-01
-2.71	8.72E-01
-2.65	8.85E-01
-2.60	8.96E-01
-2.54	9.07E-01
-2.49	9.18E-01
-2.43	9.27E-01
-2.38	9.35E-01
-2.32	9.42E-01
-2.27	9.48E-01
-2.21	9.55E-01
-2.16	9.61E-01
-2.10	9.66E-01
-2.05	9.70E-01
-1.99	9.74E-01
-1.94	9.78E-01
-1.88	9.81E-01
-1.83	9.83E-01
-1.77	9.86E-01
-1.72	9.88E-01
-1.66	9.90E-01
-1.61	9.92E-01
-1.55	9.93E-01
-1.50	9.94E-01

Table C-3

SUPPRESSION POOL WATER LEVEL PROBABILITY
(Probability that Level Below Value of Interest)

Level (inches)	Probability
-1.44	9.95E-01
-1.39	9.96E-01
-1.33	9.96E-01
-1.28	9.97E-01
-1.22	9.98E-01
-1.17	9.98E-01
-1.11	9.98E-01
-1.06	9.99E-01
-1.00	1.00E+00

Note to Table C-3:

(1) A conservative probability value corresponding to -5.70" (123,500 ft^3) instead of -5.90" (123,250 ft^3) was used in the base case quantification.

Appendix D
LARGE-LATE RELEASE IMPACT

In the November-December 2005 ACRS meetings concerning the Vermont Yankee EPU and COP credit risk assessments, the ACRS questioned the impact on Large-Late releases from EPU and COP credit. The following discussion is provided to address this question for the BFN COP credit risk assessment.

D.1 OVERVIEW OF BFN PRA RELEASE CATEGORIZATION

The spectrum of possible radionuclide release scenarios in the BFN Level 2 PRA is represented by a discrete set of release categories or bins. Typical of industry PRAs, the BFN release categories are defined by the following two key attributes:

- Timing of the release
- Magnitude of the release

D.1.1 Timing Categorization

Three timing categories are used, as follows:

1)	Early (E)	Less than 6 hours from accident initiation
2)	Intermediate (I)	Greater than or equal to 6 hours, but less than 24 hours
3)	Late (L)	Greater than or equal to 24 hours.

The definition of the timing categories is relative to the timing of the declaration of a General Emergency and based upon past experience concerning offsite accident response:

- 0-6 hours is conservatively assumed to include cases in which minimal offsite protective measures have been observed to be performed in non-nuclear accidents.
- 6-24 hours is a time frame in which much of the offsite nuclear plant protective measures can be assured to be accomplished.
- >24 hours are times at which the offsite measures can be assumed to be fully effective.

Magnitude Categorization

The BFN Level 2 PRA defines the following radionuclide release magnitude classifications:

1) High (H) - A radionuclide release of sufficient magnitude to have the potential to cause prompt fatalities.
2) Medium or Moderate (M) - A radionuclide release of sufficient magnitude to cause near-term health effects.
3) Low (L) - A radionuclide release with the potential for latent health effects.
4) Low-Low (LL) - A radionuclide release with undetectable or minor health effects.
5) Negligible (OK) - A radionuclide release that is less than or equal to the containment design base leakage.

The definition of the source terms levels distinguishing each of these release severity categories is based on the review of existing consequence analyses performed in previous industry studies, PRAs and NRC studies containing detailed consequence modeling. The BFN Level 2 PRA uses cesium as the measure of the source term magnitude because it delivers a substantial fraction of the total whole body population dose. This approach is typical of most industry PRAs.

In terms of fraction of core inventory CsI released, the BFN release magnitude classification is as follows:

Release Magnitude	Fraction of Release CsI Fission Products
High	greater than 10%
Medium/Moderate	1 to 10%
Low	0.1 to 1.0%
Low-Low	less than 0.1%
Negligible	much less than 0.1%

D.2 EPU COP CREDIT IMPACT ON LARGE-LATE

Based on the preceding discussions, it can be seen that "Large-Late" scenarios are termed High-Late releases in BFN Level 2 PRA terminology and are defined as releases occurring after 24 hrs and with a magnitude of >10% CsI.

For this risk assessment it is not necessary to perform any explicit quantification of the Level 2 PRA to determine the effect on large-late releases, i.e., the scenarios of interest in this analysis are never late releases, in fact they are all always Early releases.

The scenarios of interest in this risk assessment are very low frequency postulated scenarios that were not explicitly incorporated into the BFN base PRA. These scenarios are defined by containment isolation failure at t=0, leading to assumed loss of NPSH to the ECCS pumps in the short term and leading to core damage in approximately one hour (for the LLOCA and ATWS accidents) to approximately six hours (for the SBO accidents).

In summary, there is no change in the frequency of Large-Late releases due to the credit of COP in DBA LOCA, ATWS and SBO scenarios.

Appendix E
REVISED EVENT TREES

This appendix provides print-outs of the BFN Unit 1 PRA modified event trees used in this analysis. In addition, the RISKMAN software event tree "rules" and "macros" for these revised event trees are also provided in this appendix. These print-outs are provided at the end of this appendix.

E.1 EVENT TREE REVISIONS

The following are details of the changes made to the BFN Unit 1 PRA RISKMAN models for this risk assessment.

E.1.1 LLOCA Event Tree Changes

The Level 1 large LOCA event trees were modified for this risk assessment to question the status of containment integrity first in the tree. In addition, a second node was added to the large LOCA event trees to question the probability of extreme plant conditions (e.g., high river water temperature). These nodes are then used to fail the RHR and CS pumps for scenarios with 2 or less RHR pumps in SPC.

In order to ensure that only the large LOCA initiators are affected by the event tree changes, several of the existing event trees were renamed. In addition, because the containment isolation top event CIL is located in the containment event tree CET1, it too was renamed. The event tree names were revised as follows:

Original Event Tree	New Event Tree	Description
CET1	CETN1	Containment Event Tree 1
LLCS	LLCSN	Core Spray LLOCA Event Tree
LLRD	LLDSN	Recirc Discharge LLOCA Event Tree
LLO	LLON	Other Large LOCA Event Tree
LLRS	LLSN	Recirc Suction LLOCA Event Tree

In the containment event tree, top event CIL was replaced with a dummy top event, CILDUM, which is a switch whose branches depends on CIL, now moved into the large LOCA event trees. Two split fractions were developed for CILDUM, one for success (CILDS) and one for failure (CILDF). The branches of CILDUM depend on CIL, which is traced via macro CILFAIL. Macro CILFAIL is a logical TRUE if top event CIL=F, otherwise it is FALSE. If CILFAIL is TRUE, that is if CIL fails, then the failed branch of CILDUM is assigned via split fraction CILDF (1.00E+00). Otherwise, the success branch is assigned via split fraction CILDS (0.00E+00).

The purpose of installing dummy top event CILDUM is to preserve the containment event tree structure (i.e., the RISKMAN software allows use of a specific top event name only once in an accident sequence structure). All top events that are asked in the base model if CIL fails are still asked; those that are not normally asked are not asked in this sensitivity case.

In each of the large LOCA event trees, top event CIL was added as the left most top event, and top event NPSH was added as the next top event to the right. In this way, the original event tree structure is preserved because CIL transfers to NPSH which transfers to the original first top of each event tree. CIL models containment isolation failure probability, and top event NPSH models the probability of other key plant conditions existing at the time of the accident (i.e., high reactor power, high RW and SP water temperatures, low SP level).

The existing CIL fault tree was modified to add the probability of a pre-existing containment leak; a basic event (CONDPRE) was inserted just under the top 'OR' gate of the CIL fault tree. The CONDPRE basic event is set to different values depending on the size of the leak rate assumed in the base quantification and in sensitivity cases (refer to Table 4-2 and to Appendix F).

Top event NPSH has two split fractions, NPSH1 and NPSHS. The latter is used to filter out large LOCA sequences where 3 or more RHR pumps are running. The status of the RHR pumps and heat exchangers is tracked via an existing macro in the event tree RHRET. Split fraction NPSH1 is the split fraction probability resulting from quantification of the NPSH fault tree (refer to Appendix F). Refer to Section 4.2.2 where scenarios with more than 2 RHR pumps in SPC are analyzed as a sensitivity case.

When both top events CIL and NPSH fail, conditions are present such that the model assumes there is insufficient NPSH for the low pressure pumps to operate during a large LOCA. RISKMAN rules were added to assign guaranteed failure split fractions for top events: CS, LPCI, LPCII, SPI and SPII. A macro was created (NPSHLOST, defined as CIL=F*NPSH=F) and defined in each large LOCA event tree. The macro was then added to the split fraction rule for each guaranteed failed split fraction for the desired top event. Note that drywell spray failure is captured by the event tree structure (i.e., if LPCI loops I and II are failed, then drywell spray is never asked in the event trees).

In addition, LPCI and LPCS inter-unit crossties are defeated because the pumps crosstied from the Unit 2 would be aligned to the Unit 1 suppression pool and would experience the same NPSH conditions as the Unit 1 pumps.

E.1.2 ATWS and SBO Event Tree Changes

For the ATWS scenarios, COP is modeled as always required for LP ECCS pump NPSH; if COP is unavailable, all LP ECCS pumps drawing from the torus are modeled as failed due to insufficient net positive head. For the SBO scenarios, overpressure is modeled as required after AC power is recovered at t=4 hours.

Similar to the event tree model changes for LLOCA, the ATWS and SBO event trees were modified in order to determine the status of containment integrity prior to

questioning the status of low pressure systems drawing from the torus. Each of the original event trees was copied and renamed (appending an N) for use in the analysis.

Original Event Tree	New Event Tree	Description
ATWS3	ATWS3N	ATWS Event Tree
ATWS4	ATWS4N	ATWS Event Tree
LPGTET	LPGTETN	Low Pressure General Transient Event Tree

Note:

1. Event trees ATWS1 and ATWS2 exist in the BFN PRA, but they do not contain nodes for LP ECCS pumps and thus do not require modification for this risk assessment.

2. It was not necessary to modify event tree HPGTET, High Pressure General Transient, for this risk assessment.

The same revised containment event tree discussed previously for the LLOCA scenarios is also used for the ATWS and SBO scenarios.

The containment isolation top event (CIL) added to the above revised ATWS and SBO event trees is the same one discussed previously for the LLOCA scenarios. The event tree split fraction rules were modified to fail the low pressure systems (top events) if the containment isolation top event fails (CIL).

In addition, as discussed previously for the LLOCA scenarios, LPCI and LPCS inter-unit crossties are defeated.

An additional requirement was used for modeling COP credit impacts for SBO scenarios. Two top events model recovery of AC power: 1) one top event (EPR30) models AC recovery at t=30 minutes; and 2) another top event (EPR6) models AC recovery at t=6 hours. The BFN is not currently designed with 4-hr SBO scenarios; as such, the 6-hr SBO sequences are used as a surrogate to model the COP impact on SBO sequences after AC power recovery at t=4hrs. Event tree node EPR6 (included in

event tree HPGTET) is checked for success prior to requiring COP (i.e., SBO sequences with failure of AC recovery are not modified to question COP issues).

To quantify the impact of the COP requirement on ATWS and SBO, the following ATWS and SBO initiating events were quantified through the revised event tree structures discussed above:

ATWS Initiators	**LOSP Initiators**
IOOVA	LOSP
LOCHSA	L500PA
LOFWA	L500U
LOSPA	
TTA	

MODEL Name: U1ERIN
Event Tree: LLCSN.ETI

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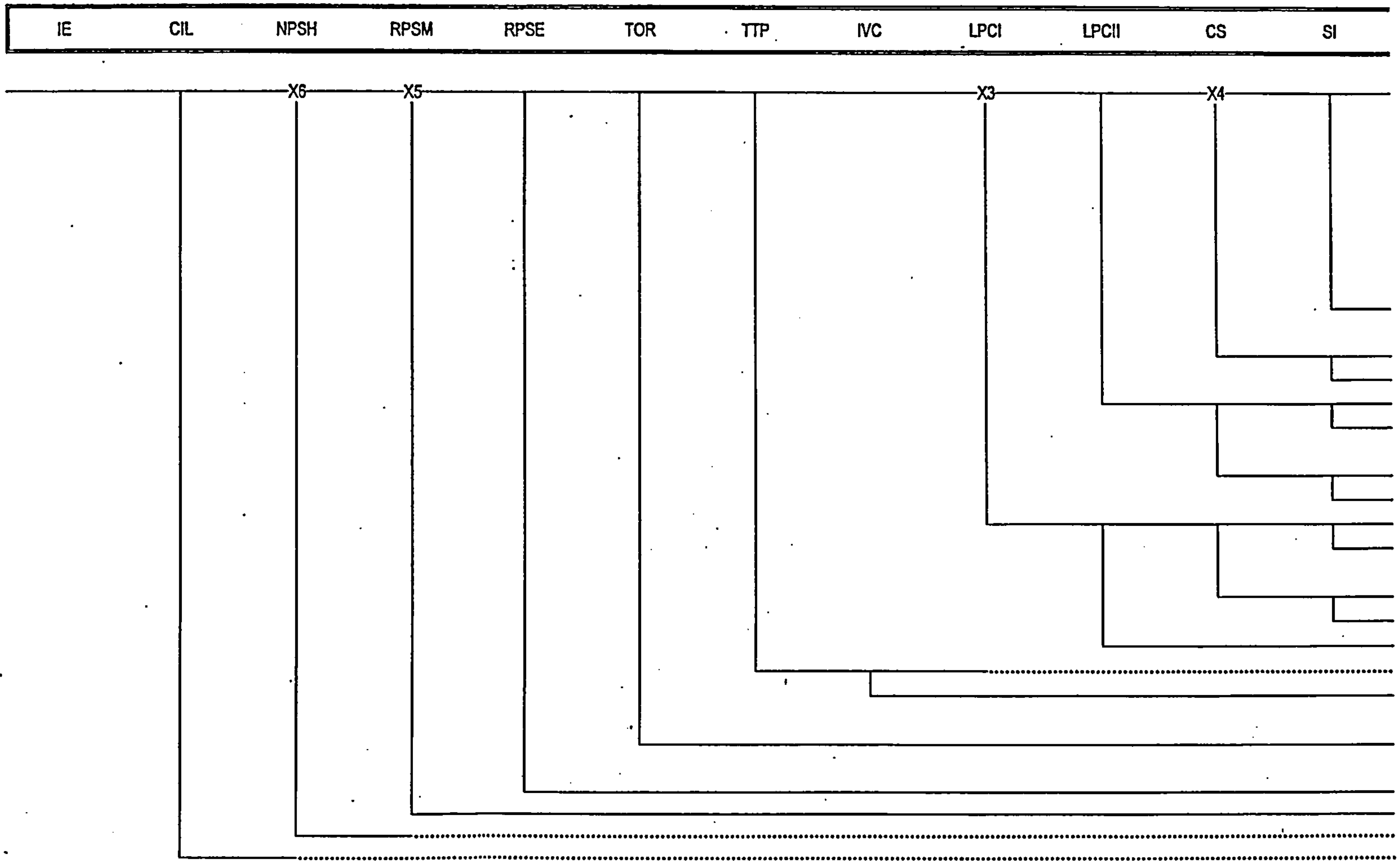
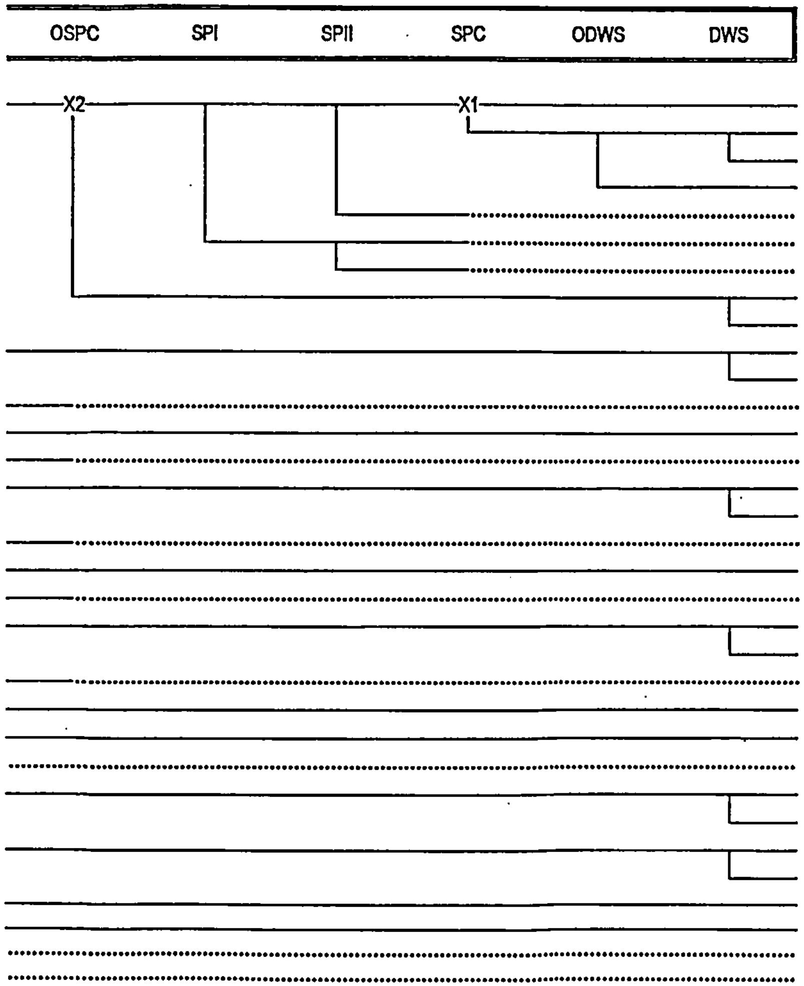
12:22:35 February 14, 2006



MODEL Name: U1ERIN
Event Tree: LLCSN.ETI



X#	B#	S#
	1	1
	2	2
	3	3
	4	4
X1	5	5-8
X1	6	9-12
X1	7	13-16
	8	17
	9	18
	10	19
	11	20
X2	12	21-38
	13	39
X2	14	40-57
	15	58
	16	59
X2	17	60-77
	18	78
X2	19	79-96
	20	97
	21	98
X2	22	99-116
	23	117
	24	118
X3	25	119-236
	26	237
	27	238
	28	239
	29	240
	30	241
	31	242
X5	32	243-484
X6	33	485-968

Model Name: U1COP2-9
Top Events for Event Tree: LLCSN

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Top Event Name	Description
CIL	PRIMARY CONTAINMENT ISOLATION FAILURE - LARGE (=>3 INCHES)
NPSH	CONDITIONS PREVENTING NPSH FOR LLOCA
RPSM	MECHANICAL PORTION OF RPS SUCCESSFUL
RPSE	ELECTRICAL PORTION OF RPS (NUREG-5500 BASIS)
TOR	PRESSURE SUPPRESSION POOL
TTP	TURBINE TRIP
IVC	CLOSURE OF MSIVS
LPCI	LPCI LOOP I
LPCII	LPC LOOP II
CS	CORE SPRAY SYSTEM
SI	LOGIC SWITCH FOR SUFFICIENT INJECTION
OSPC	OPERATOR ALIGNS SUPPRESSION POOL COOLING
SPI	SUPPRESSION POOL COOLING HARDWARE - LOOP I
SPII	SUPPRESSION POOL COOLING HARDWARE - LOOP II
SPC	LOGIC SWITCH FOR SUPPRESSION POOL COOLING WITH U1 RHR
ODWS	OPERATOR ALIGNS DRYWELL SPRAY
DWS	DRYWELL SPRAY HARDWARE

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: LLCSN

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SF	Split Fraction Assignment Rule
CIL1	PCA=S*(DWP=S + LVP=S)
CIL2	PCA=F*(DWP=S + LVP=S)
CILF	DWP=F*LVP=F
NPSHS	RHR1*RHR2*RHR3 + RHR1*RHR2*RHR4 + RHR1*RHR3*RHR4 + RHR2*RHR3*RHR4 + RHR1*RHR2*RHR3*RHR4 Comments IF 3 OR MORE PUMPS ARE AVAILABLE WE DON'T NEED COP FOR ECCS NPSH
NPSH1	INIT=LLCA + INIT=LLCB + INIT=LLDA + INIT=LLDB + INIT=LLO + INIT=LLSA + INIT=LLSB
NPSHS	1
RPSMS	1
RPSEO	1
TOR1	1
TTP1	BB5=S*DI=S
TTP2	BB5=S*DI=F
TTP3	BB5=F*DI=S
TTPF	1
IVC1	1
LPCIF	-LPCISUP + NPSHLOST
LPCI2	LPCISUP Comments MANUAL LPCI START NOT CREDITED LLOCAS; ODD SPLIT FRACTION SWOULD APPLY
LPCIIF	-LPCIISUP + NPSHLOST
LPCII2	LPCI=S
LPCII4	-LPCISUP
LPCII6	LPCI=F*LPCISUP
CSF	INIT=LLCA*(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F + CASSIG +DW=F*LV=F+RB=F+ -EECW) + INIT=LLCB*(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+ -EECW) + NPSHLOST
CS2	INIT=LLCB*-(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+ -EECW)
CS2B	INIT=LLCA*-(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+ -EECW)
CSF	1

Model Name: U1COP2-9

Split Fraction Assignment Rule for Event Tree: LLCSN

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SF	Split Fraction Assignment Rule
SIS	LPCI=S*RPA=S*RPC=S + LPCII=S*RPB=S*RPD=S + LPCI=S*LPCII=S*((RPA=S+RPC=S) + (RPB=S+RPD=S) + CS=S) Comments ANY TWO RHR PUMPS OR CS FROMTHE UNBROKEN LOOP
SIF	1
OSPC1	RPSM=S*RPSE=S
OSPCF	1
SPIF	OSPC=F + RE=F + NPSHLOST
SPI2	RE=S*RC=S*(RPA=S*HXA=S + RPC=S*HXC=S)
SPIF	1
SPIIF	OSPC=F + RF=F + NPSHLOST
SPII4	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=S
SPII5	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*RE=S
SPII6	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*RE=F
SPIIF	1
SPCF	-(SPI=S)*-(SPII=S)
SPCS	SPI=S*(RPA=S*HXA=S + RPC=S*HXC=S) + SPII=S*(RPB=S*HXB=S+RPD=S*HXD=S)
SPCF	1
ODWS1	1
DWSF	PX1=F*PX2=F + (RPA=F*RPC=F +RH=F+NOGB) * (RPB=F*RPD=F+RI=F + NOGD)
DWS1	PX1=S*PX2=S*(RPA=S+RPC=S)*-NOGB*(RPB=S+RPD=S)*-NOGD
DWS2	(RPA=F*RPC=F +RH=F+NOGB+PX1=F) * (RPB=F*RPD=F+RI=F + NOGD+PX2=F)
DWSF	1

Model Name: U1COP2-9
Macro for Event Tree: LLCSN

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Page 1

Macro	Macro Rule / Comments
ALTINJRHSW	RPSM=B THIS MACRO IS NEEDED IN THE CETS
ALTINJU2X	RPSM=B THIS MACRO IS NEEDED IN THE CETS
BUCKET	RPSM=B
CILFAIL	CIL=F
CLASS1A	RPSM=B
CLASS1B	RPSM=B
CLASS1BE	RPSM=B
CLASS1BL	RPSM=B
CLASS1C	RPSM=B
CLASS1D	RPSM=B
CLASS1E	RPSM=B
CLASS2	RPSM=B
CLASS2A	RPSM=B
CLASS2L	SPC=F + OSPC=F
CLASS2T	RPSM=B
CLASS2V	RPSM=B
CLASS3A	RPSM=B
CLASS3B	RPSM=B

Model Name: U1COP2-9
Macro for Event Tree: LLCSN

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Page 2

Macro	Macro Rule / Comments
CLASS3C	-(SI=S)+ -(TTP=S+IVC=S)
CLASS3D	-(TOR=S)
CLASS4	RPSM=F
CLASS5	-(TTP=S)*-(IVC=S)
DWSPRAY	DWS=S THIS MACRO IS NEEDED IN THE CETS
EMDEPHDWR	RPSM=B THIS MACRO IS NEEDED IN THE CETS
HIGH	RPSM=B
HPI	RPSM=B
LOW	INIT=LLCA + INIT=LLCB
LPCIISUP	RF=S*((NPII=S*DW=S) + LV=S)
LPCISUP	RE=S*((NPI=S*DW=S) + LV=S) LOOP I LPCI SUPPORT
LPI	SI=S
NOACREC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOCD	RPSM=S * TOR=S*(TTP=S+IVC=S)*SI=S*SPC=S
NODC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NORV	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOSRV	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NPSHLOST	CIL=F*NPSH=F

Model Name: U1COP2-9
Macro for Event Tree: LLCSN

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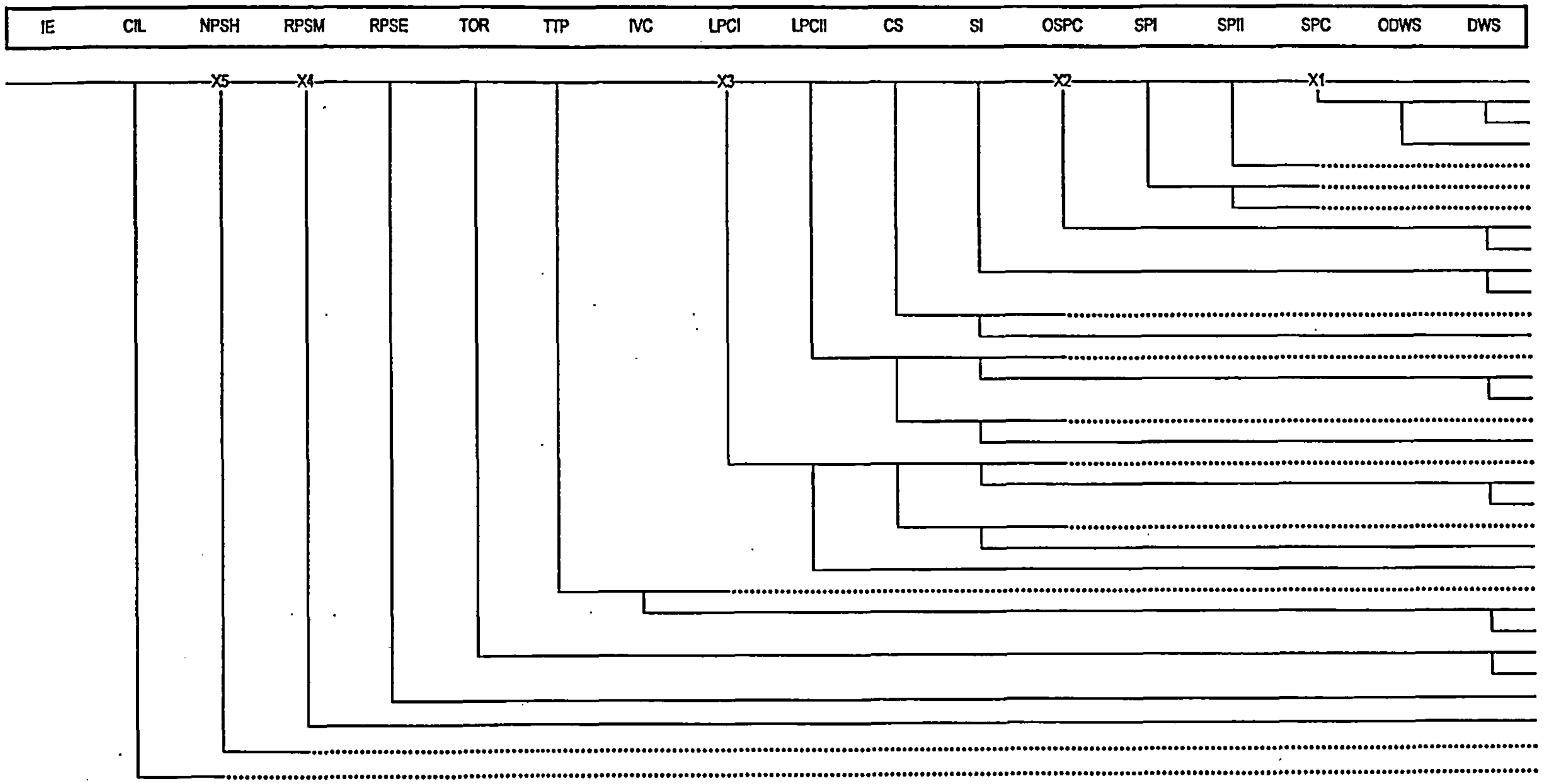
Macro	Macro Rule / Comments
OPDEPL1	RPSM=B THIS MACRO IS NEEDED IN THE CETS
RHRSPCOOL	SPC=S
SORV	RPSM=S LARGE LOCAS ARE ALWAYS DEPRESSURIZEED

MODEL Name: U1ERIN
Event Tree: LLON.ETI

13:37:12 February 16, 2006



MODEL Name: U1ERIN
Event Tree: LLON.ETI

X#	B#	S#
	1	1
	2	2
	3	3
	4	4
X1	5	5-8
X1	6	9-12
X1	7	13-16
	8	17
	9	18
	10	19
	11	20
X2	12	21-38
	13	39
X2	14	40-57
	15	58
	16	59
X2	17	60-77
	18	78
X2	19	79-96
	20	97
	21	98
X2	22	99-116
	23	117
	24	118
X3	25	119-236
	26	237
	27	238
	28	239
	29	240
	30	241
	31	242
X4	32	243-484
X5	33	485-968

Model Name: U1COP2-9
Top Events for Event Tree: LLON

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Page 1

Top Event Name	Description
CIL	PRIMARY CONTAINMENT ISOLATION FAILURE - LARGE (=>3 INCHES)
NPSH	CONDITIONS PREVENTING NPSH FOR LLOCA
RPSM	MECHANICAL PORTION OF RPS SUCCESSFUL
RPSE	ELECTRICAL PORTION OF RPS (NUREG-5500 BASIS)
TOR	PRESSURE SUPPRESSION POOL
TTP	TURBINE TRIP
IVC	CLOSURE OF MSIVS
LPCI	LPCI LOOP I
LPCII	LPC LOOP II
CS	CORE SPRAY SYSTEM
SI	LOGIC SWITCH FOR SUFFICIENT INJECTION
OSPC	OPERATOR ALIGNS SUPPRESSION POOL COOLING
SPI	SUPPRESSION POOL COOLING HARDWARE - LOOP I
SPII	SUPPRESSION POOL COOLING HARDWARE - LOOP II
SPC	LOGIC SWITCH FOR SUPPRESSION POOL COOLING WITH U1 RHR
ODWS	OPERATOR ALIGNS DRYWELL SPRAY
DWS	DRYWELL SPRAY HARDWARE

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: LLON

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Page 1

SF	Split Fraction Assignment Rule
CIL1	PCA=S*(DWP=S + LVP=S)
CIL2	PCA=F*(DWP=S + LVP=S)
CILF	DWP=F*LVP=F
NPSHS	RHR1*RHR2*RHR3 + RHR1*RHR2*RHR4 + RHR1*RHR3*RHR4 + RHR2*RHR3*RHR4 + RHR1*RHR2*RHR3*RHR4 Comments IF 3 OR MORE PUMPS ARE AVAILABLE WE DON'T NEED COP FOR ECCS NPSH
NPSH1	INIT=LLCA + INIT=LLCB + INIT=LLDA + INIT=LLDB + INIT=LLO + INIT=LLSA + INIT=LLSB
NPSHS	1
RPSMS	1
RPSEO	1
TOR1	1
TTP1	BB5=S*DI=S
TTP2	BB5=S*DI=F
TTP3	BB5=F*DI=S
TTPF	1
IVC1	1
LPCIF	-LPCISUP + NPSHLOST
LPCI2	LPCISUP Comments MANUAL LPCI START NOT CREDITED LLOCAS; ODD SPLIT FRACTION SWOULD APPLY
LPCIIF	-LPCIISUP + NPSHLOST
LPCII2	LPCI=S
LPCII4	-LPCISUP
LPCII6	LPCI=F*LPCISUP
CSF	(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+ -EECW) * (RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+ -EECW) + NPSHLOST
CS2	-(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+ -EECW)
CS2B	-(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+ -EECW)
CSF	1
SIS	LPCI=S*(RPA=S+RPC=S) + LPCII=S*(RPB=S+RPD=S) + CS=S

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: LLON

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Page 2

SF	Split Fraction Assignment Rule
	Comments ANY TWO RHR PUMPS OR CS FROMTHE UNBROKEN LOOP
SIF	1
OSPC1	RPSM=S*RPSE=S
OSPCF	1
SPIF	RE=F + OSPC=F + NPSHLOST
SPI2	1
SPIIF	OSPC=F + RF=F + NPSHLOST
SPII4	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=S
SPII5	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*RE=S
SPII6	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*RE=F
SPIIF	1
SPCF	-(SPI=S)*-(SPII=S)
SPCS	SPI=S*(RPA=S*HXA=S + RPC=S*HXC=S) + SPII=S*(RPB=S*HXB=S+RPD=S*HXD=S)
SPCF	1
ODWS1	1
DWSF	PX1=F*PX2=F + (RPA=F*RPC=F +RH=F+NOGB) * (RPB=F*RPD=F+RI=F + NOGD)
DWS1	PX1=S*PX2=S*(RPA=S+RPC=S)*-NOGB*(RPB=S+RPD=S)*-NOGD
DWS2	(RPA=F*RPC=F +RH=F+NOGB+PX1=F) * (RPB=F*RPD=F+RI=F + NOGD+PX2=F)
DWSF	1

Model Name: U1COP2-9
Macro for Event Tree: LLON

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Page 1

Macro	Macro Rule / Comments
ALTINJRHSW	RPSM=B THIS MACRO IS NEEDED IN THE CETS
ALTINJU2X	RPSM=B THIS MACRO IS NEEDED IN THE CETS
BUCKET	RPSM=B
CILFAIL	CIL=F
CLASS1A	RPSM=B
CLASS1B	RPSM=B
CLASS1BE	RPSM=B
CLASS1BL	RPSM=B
CLASS1C	RPSM=B
CLASS1D	RPSM=B
CLASS1E	RPSM=B
CLASS2	RPSM=B
CLASS2A	RPSM=B
CLASS2L	OSPC=F+ SPC=F
CLASS2T	RPSM=B
CLASS2V	RPSM=B
CLASS3A	RPSM=B
CLASS3B	RPSM=B

Model Name: U1COP2-9
Macro for Event Tree: LLON

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Page 2

Macro	Macro Rule / Comments
CLASS3C	-(SI=S)+ -(TTP=S+IVC=S)
CLASS3D	-(TOR=S)
CLASS4	RPSM=F
CLASS5	-(TTP=S)*-(IVC=S)
DWSPRAY	DWS=S THIS MACRO IS NEEDED IN THE CETS
EMDEPHDWR	RPSM=B THIS MACRO IS NEEDED IN THE CETS
HIGH	RPSM=B
HPI	RPSM=B.
LOW	INIT=LLO
LPCIISUP	RF=S*((NPII=S*DW=S) + LV=S)
LPCISUP	RE=S*((NPI=S*DW=S) + LV=S) LOOP I LPCI SUPPORT
LPI	SI=S
NOACREC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOCD	RPSM=S * TOR=S*(TTP=S+IVC=S)*SI=S*SPC=S
NODC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NORV	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOSRV	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NPSHLOST	CIL=F*NPSH=F

Model Name: U1COP2-9
Macro for Event Tree: LLON

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Page 3

Macro	Macro Rule / Comments
OPDEPL1	RPSM=B THIS MACRO IS NEEDED IN THE CETS
RHRSPCOOL	SPC=S
SORV	RPSM=S LARGE LOCAS ARE ALWAYS DEPRESSURIZEED

MODEL Name: U1ERIN

Event Tree: LLRDN.ETI

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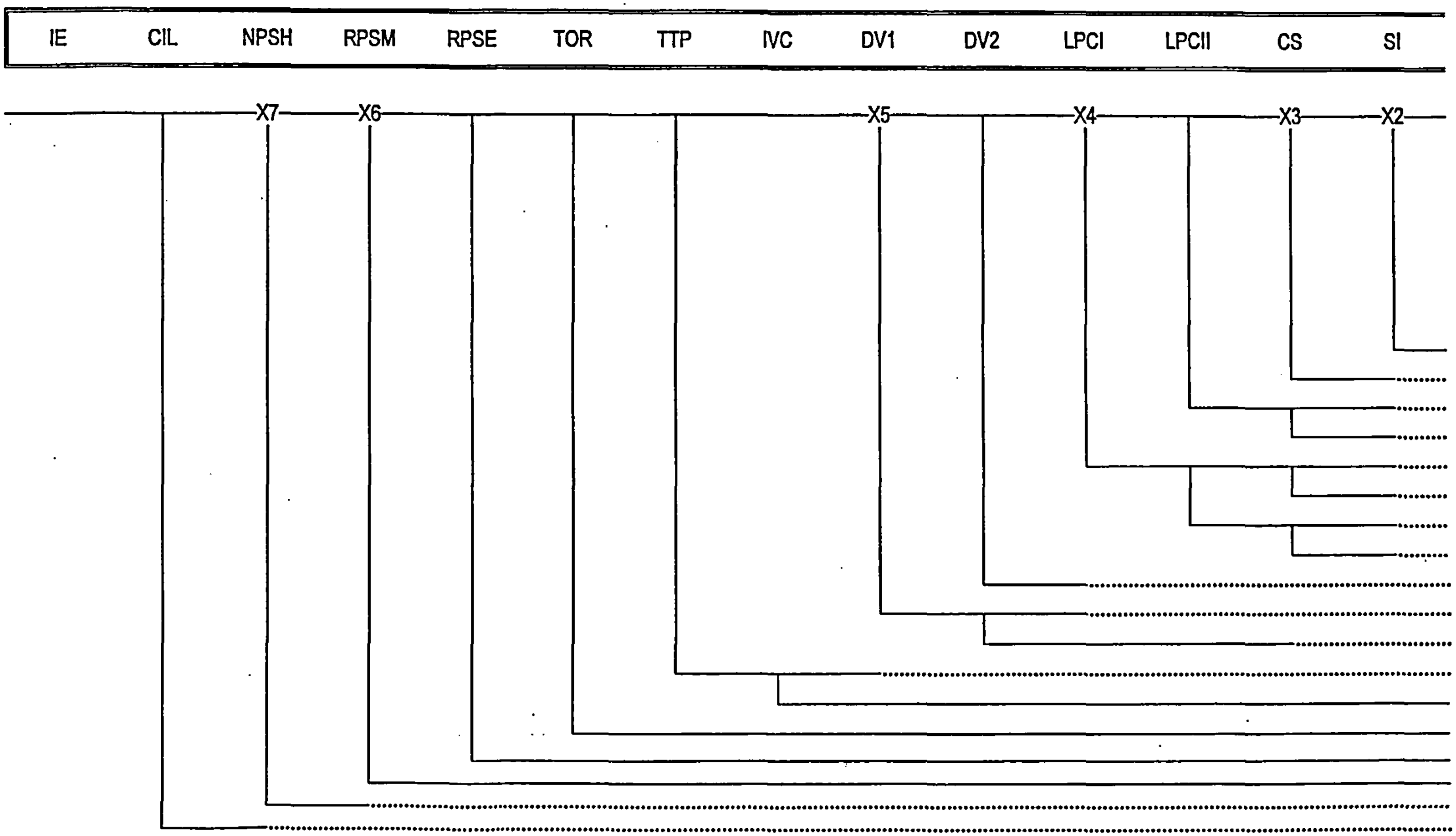
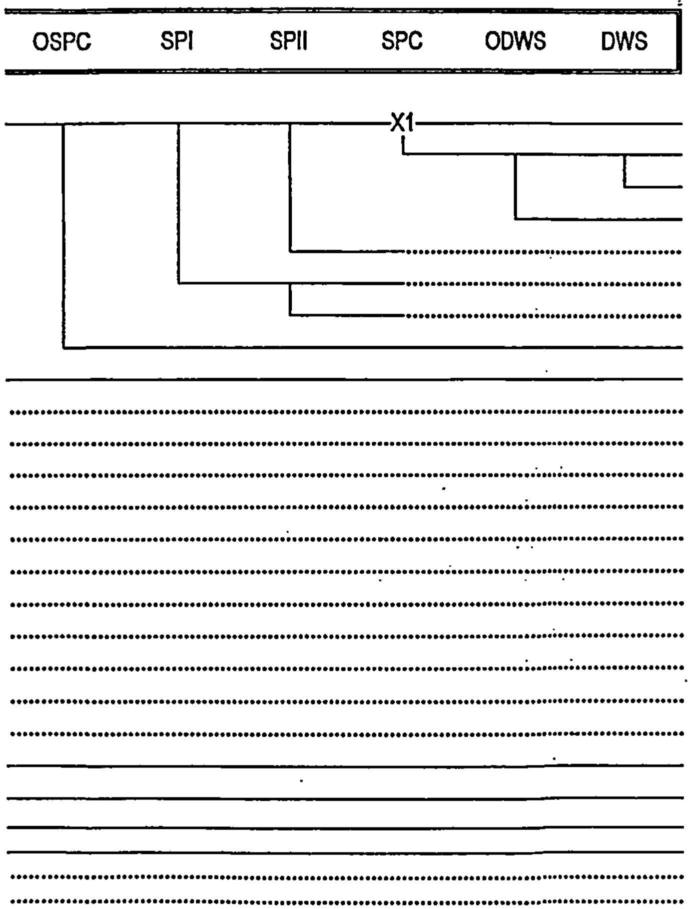
13:37:46 February 16, 2006



MODEL Name: U1ERIN

Event Tree: LLRDN.ETI

13:37:46 February 16, 2006



X#	B#	S#
	1	1
	2	2
	3	3
	4	4
X1	5	5-8
X1	6	9-12
X1	7	13-16
	8	17
	9	18
X2	10	19-36
X2	11	37-54
X2	12	55-72
X2	13	73-90
X2	14	91-108
X2	15	109-126
X2	16	127-144
X4	17	145-288
X4	18	289-432
X3	19	433-468
X5	20	469-936
	21	937
	22	938
	23	939
	24	940
X6	25	941-1880
X7	26	1881-3760

Model Name: U1COP2-9
Top Events for Event Tree: LLRDN

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Page 1

Top Event Name	Description
CIL	PRIMARY CONTAINMENT ISOLATION FAILURE - LARGE (=>3 INCHES)
NPSH	CONDITIONS PREVENTING NPSH FOR LLOCA
RPSM	MECHANICAL PORTION OF RPS SUCCESSFUL
RPSE	ELECTRICAL PORTION OF RPS (NUREG-5500 BASIS)
TOR	PRESSURE SUPPRESSION POOL
TTP	TURBINE TRIP
IVC	CLOSURE OF MSIVS
DV1	LOOP I RECIRCULATION DISCHARGE VALVE CLOSURE
DV2	LOOP II RECIRCULATION DISCHARGE VALVE CLOSURE
LPCI	LPCI LOOP I
LPCII	LPC LOOP II
CS	CORE SPRAY SYSTEM
SI	LOGIC SWITCH FOR SUFFICIENT INJECTION
OSPC	OPERATOR ALIGNS SUPPRESSION POOL COOLING
SPI	SUPPRESSION POOL COOLING HARDWARE - LOOP I
SPII	SUPPRESSION POOL COOLING HARDWARE - LOOP II
SPC	LOGIC SWITCH FOR SUPPRESSION POOL COOLING WITH U1 RHR
ODWS	OPERATOR ALIGNS DRYWELL SPRAY
DWS	DRYWELL SPRAY HARDWARE

Model Name: U1COP2-9

Split Fraction Assignment Rule for Event Tree: LLRDN

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Page 1

SF	Split Fraction Assignment Rule
CIL1	PCA=S*(DWP=S + LVP=S)
CIL2	PCA=F*(DWP=S + LVP=S)
CILF	DWP=F*LVP=F
NPSHS	RHR1*RHR2*RHR3 + RHR1*RHR2*RHR4 + RHR1*RHR3*RHR4 + RHR2*RHR3*RHR4 + RHR1*RHR2*RHR3*RHR4 Comments IF 3 OR MORE PUMPS ARE AVAILABLE WE DON'T NEED COP FOR ECCS NPSH
NPSH1	INIT=LLCA + INIT=LLCB + INIT=LLDA + INIT=LLDB + INIT=LLO + INIT=LLSA + INIT=LLSB
NPSHS	1
RPSMS	1
RPSEO	1
TOR1	1
TTP1	BB5=S*DI=S
TTP2	BB5=S*DI=F
TTP3	BB5=F*DI=S
TTPF	1
IVC1	1
DV1F	RE=F+RB=F*RC=F+NH1=F*NH2=F+DW=F*LV=F
DV11	DW=S*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV12	DW=S*LV=S*NH1=S*NH2=S*(RB=F+RC=F)
DV13	DW=S*LV=F*NH1=S*NH2=S*RB=S*RC=S
DV14	DW=F*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV15	DW=S*LV=S*(NH1=F+NH2=F)*RB=S*RC=S
DV1F	1
DV2F	RF=F+RB=F*RC=F+NH1=F*NH2=F+DW=F*LV=F
DV25	RE=F*DV1=F*DW=S*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV21	DV1=S*DW=S*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV22	DV1=F*DW=S*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV24	RE=F*DV1=F*DW=S*LV=S*NH1=S*NH2=S*(RB=F+RC=F)

Model Name: U1COP2-9

Split Fraction Assignment Rule for Event Tree: LLRDN

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Page 2

SF	Split Fraction Assignment Rule
DV23	DV1=S*DW=S*LV=S*NH1=S*NH2=S*(RB=F+RC=F)
DV24	DV1=F*DW=S*LV=S*NH1=S*NH2=S*(RB=F+RC=F)
DV27	RE=F*DV1=F*DW=S*LV=F*NH1=S*NH2=S*RB=S*RC=S
DV28	DV1=S*DW=S*LV=F*NH1=S*NH2=S*RB=S*RC=S
DV29	DV1=F*DW=S*LV=F*NH1=S*NH2=S*RB=S*RC=S
DV2A	RE=F*DV1=F*DW=F*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV2B	DV1=S*DW=F*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV2C	DV1=F*DW=F*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV2D	RE=F*DV1=F*DW=S*LV=S*(NH1=F+NH2=F)*RB=S*RC=S
DV2E	DV1=S*DW=S*LV=S*(NH1=F+NH2=F)*RB=S*RC=S
DV2G	DV1=F*DW=S*LV=S*(NH1=F+NH2=F)*RB=S*RC=S
DV2F	1
LPCIF	-LPCISUP+ DV1=F*DV2=F + NPSHLOST
LPCI2	LPCISUP
LPCIIF	-LPCIISUP +DV1=F*DV2=F + NPSHLOST
LPCII2	LPCI=S
LPCII4	-LPCISUP
LPCII6	LPCI=F*LPCISUP
LPCIIF	1
CSF	(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+EA=F*EB=F*EC=F + EA=F*EB=F*ED=F + EA=F*EC=F*ED=F + EB=F*EC=F*ED=F)*(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+ EA=F*EB=F*EC=F + EA=F*EB=F*ED=F + EA=F*EC=F*ED=F + EB=F*EC=F*ED=F) + NPSHLOST
CS1	-(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+EA=F*EB=F*EC=F + EA=F*EB=F*ED=F + EA=F*EC=F*ED=F + EB=F*EC=F*ED=F)*-(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+EA=F*EB=F*EC=F + EA=F*EB=F*ED=F + EA=F*EC=F*ED=F + EB=F*EC=F*ED=F)
CS2	-(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+EA=F*EB=F*EC=F + EA=F*EB=F*ED=F + EA=F*EC=F*ED=F + EB=F*EC=F*ED=F)*(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+EA=F*EB=F*EC=F + EA=F*EB=F*ED=F + EA=F*EC=F*ED=F + EB=F*EC=F*ED=F)

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: LLRDN

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SF	Split Fraction Assignment Rule
CS2B	(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+EA=F*EB=F*EC=F + EA=F*EB=F*ED=F + EA=F*EC=F*ED=F + EB=F*EC=F*ED=F)*-(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+EA=F*EB=F*EC=F + EA=F*EB=F*ED=F + EA=F*EC=F*ED=F + EB=F*EC=F*ED=F)
CSF	1 Comments Core Spray Loop II Pipe Break Large LOCA
SIS	CS=S + LPCI=S*(RPA=S + RPB=S) + LPCII=S*(RPB=S + RPD=S)
SIF	1
OSPC1	RPSM=S*RPSE=S
OSPCF	1
SPIF	RE=F + OSPC=F + NPSHLOST
SPI2	1
SPIIF	OSPC=F + RF=F + NPSHLOST
SPII4	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=S
SPII5	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*RE=S
SPII6	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*RE=F
SPIIF	1
SPCF	-(SPI=S)*-(SPII=S)
SPCS	SPI=S*(RPA=S*HXA=S + RPC=S*HXC=S) + SPII=S*(RPB=S*HXB=S+RPD=S*HXD=S)
SPCF	1
ODWS1	1
DWSF	PX1=F*PX2=F + (RPA=F*RPC=F +RH=F+NOGB) * (RPB=F*RPD=F+RI=F + NOGD)
DWS1	PX1=S*PX2=S*(RPA=S+RPC=S)*-NOGB*(RPB=S+RPD=S)*-NOGD
DWS2	(RPA=F*RPC=F +RH=F+NOGB+PX1=F) * (RPB=F*RPD=F+RI=F + NOGD+PX2=F)
DWSF	1

Model Name: U1COP2-9
Macro for Event Tree: LLRDN

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Page 1

Macro	Macro Rule / Comments
ALTINJRHSW	RPSM=B THIS MACRO IS NEEDED IN THE CETS
ALTINJU2X	RPSM=B THIS MACRO IS NEEDED IN THE CETS
BUCKET	RPSM=B
CILFAIL	CIL=F
CLASS1A	RPSM=B
CLASS1B	RPSM=B
CLASS1BE	RPSM=B
CLASS1BL	RPSM=B
CLASS1C	RPSM=B
CLASS1D	RPSM=B
CLASS1E	RPSM=B
CLASS2	RPSM=B
CLASS2A	RPSM=B
CLASS2L	OSPC=F + SPC=F
CLASS2T	RPSM=B
CLASS2V	RPSM=B
CLASS3A	RPSM=B
CLASS3B	RPSM=B

Model Name: U1COP2-9
Macro for Event Tree: LLRDN

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Macro	Macro Rule / Comments
CLASS3C	-(SI=S)+ -(TTP=S+IVC=S)
CLASS3D	-(TOR=S)
CLASS4	RPSM=F
CLASS5	-(TTP=S)*-(IVC=S)
DWSPRAY	DWS=S THIS MACRO IS NEEDED IN THE CETS
EMDEPHDWR	RPSM=B THIS MACRO IS NEEDED IN THE CETS
HIGH	RPSM=B
HPI	RPSM=B
LOW	INIT=LLDA + INIT=LLDB
LPCIISUP	RF=S*((NPII=S*DW=S) + LV=S)
LPCISUP	RE=S*((NPI=S*DW=S) + LV=S) LOOP I LPCI SUPPORT
LPI	SI=S
NOACREC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOCD	RPSM=S * TOR=S*(TTP=S+IVC=S)*SI=S*SPC=S
NODC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NORV	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOSRV	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NPSHLOST	CIL=F*NPSH=F

Model Name: U1COP2-9

Macro for Event Tree: LLRDN

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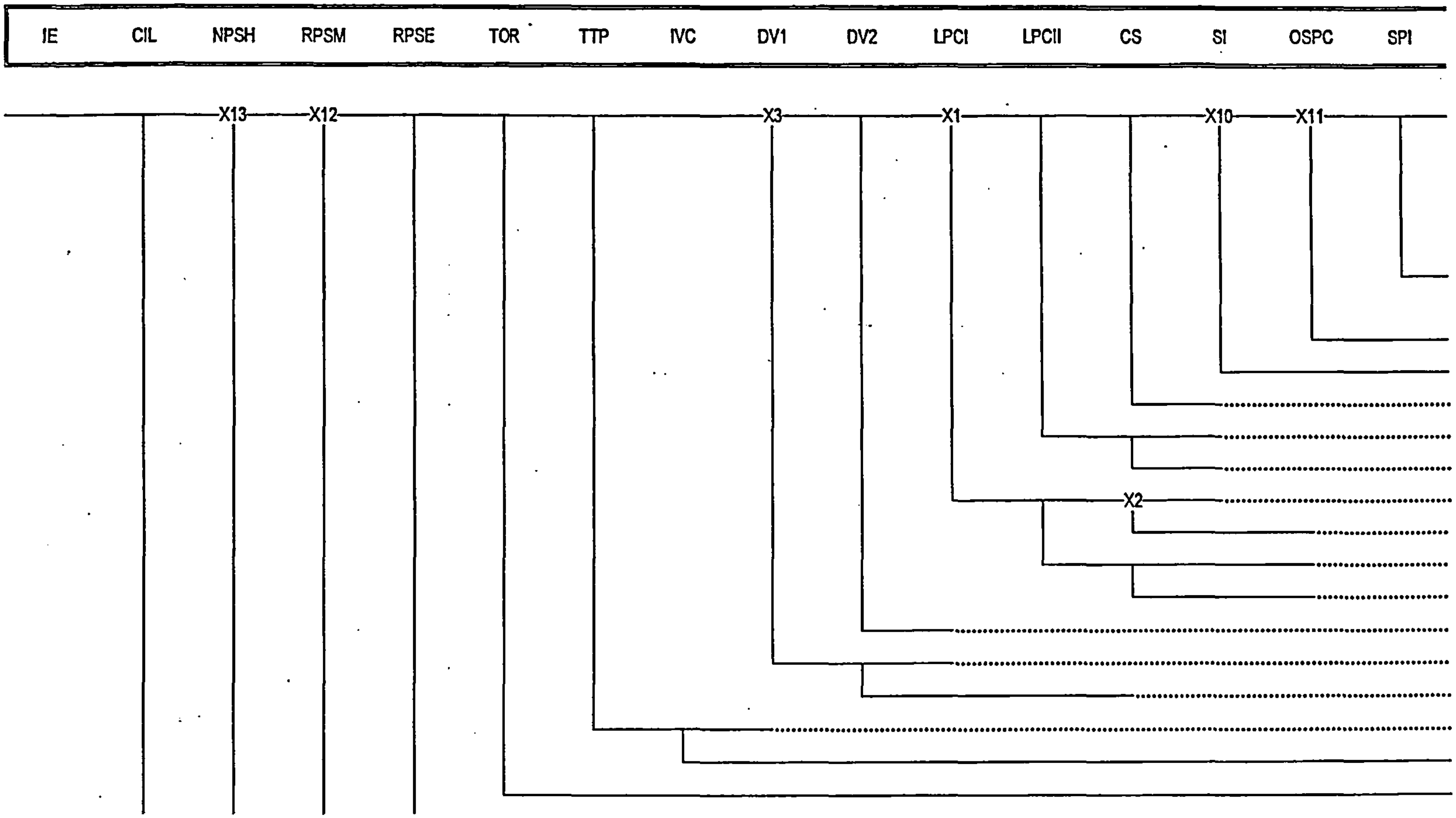
Macro	Macro Rule / Comments
OPDEPL1	RPSM=B THIS MACRO IS NEEDED IN THE CETS
RHRSPCOOL	OSPC=F + SPC=F
SORV	RPSM=S LARGE LOCAS ARE ALWAYS DEPRESSURIZEED

MODEL Name: U1ERIN

Event Tree: LLRSN.ETI

13:38:20 February 16, 2006



MODEL Name: U1ERIN

Event Tree: LLRSN.ETI

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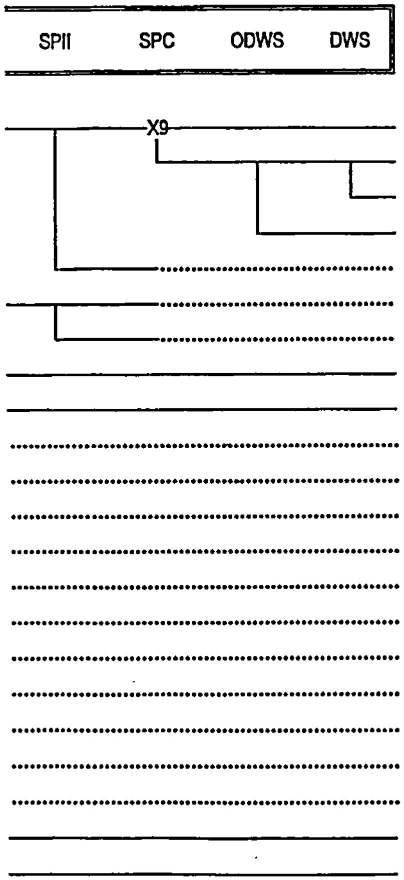

X#	B#	S#
	1	1
	2	2
	3	3
	4	4
X9	5	5-8
X9	6	9-12
X9	7	13-16
	8	17
	9	18
X10	10	19-36
X10	11	37-54
X10	12	55-72
X10	13	73-90
X11	14	91-107
X11	15	108-124
X11	16	125-141
X1	17	142-282
X1	18	283-423
X2	19	424-458
X3	20	459-916
	21	917
	22	918

MODEL Name: U1ERIN

Event Tree: LLRSN.ETI

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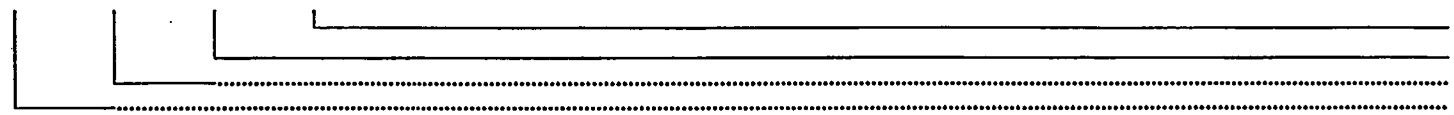
IE	CIL	NPSH	RPSM	RPSE	TOR	TTP	IVC	DV1	DV2	LPCI	LPCII	CS	SI	OSPC	SPI



MODEL Name: U1ERIN

Event Tree: LLRSN.ETI

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SPII	SPC	ODWS	DWS	X#	B#	S#
					23	919
					24	920
				X12	25	921-1840
				X13	26	1841-3680

Model Name: U1COP2-9
Top Events for Event Tree: LLRSN

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Top Event Name	Description
CIL	PRIMARY CONTAINMENT ISOLATION FAILURE - LARGE (=>3 INCHES)
NPSH	CONDITIONS PREVENTING NPSH FOR LLOCA
RPSM	MECHANICAL PORTION OF RPS SUCCESSFUL
RPSE	ELECTRICAL PORTION OF RPS (NUREG-5500 BASIS)
TOR	PRESSURE SUPPRESSION POOL
TTP	TURBINE TRIP
IVC	CLOSURE OF MSIVS
DV1	LOOP I RECIRCULATION DISCHARGE VALVE CLOSURE
DV2	LOOP II RECIRCULATION DISCHARGE VALVE CLOSURE
LPCI	LPCI LOOP I
LPCII	LPC LOOP II
CS	CORE SPRAY SYSTEM
SI	LOGIC SWITCH FOR SUFFICIENT INJECTION
OSPC	OPERATOR ALIGNS SUPPRESSION POOL COOLING
SPI	SUPPRESSION POOL COOLING HARDWARE - LOOP I
SPII	SUPPRESSION POOL COOLING HARDWARE - LOOP II
SPC	LOGIC SWITCH FOR SUPPRESSION POOL COOLING WITH U1 RHR
ODWS	OPERATOR ALIGNS DRYWELL SPRAY
DWS	DRYWELL SPRAY HARDWARE

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: LLRSN

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SF	Split Fraction Assignment Rule
CIL1	PCA=S*(DWP=S + LVP=S)
CIL2	PCA=F*(DWP=S + LVP=S)
CILF	DWP=F*LVP=F
NPSHS	RHR1*RHR2*RHR3 + RHR1*RHR2*RHR4 + RHR1*RHR3*RHR4 + RHR2*RHR3*RHR4 + RHR1*RHR2*RHR3*RHR4 Comments IF 3 OR MORE PUMPS ARE AVAILABLE WE DON'T NEED COP FOR ECCS NPSH
NPSH1	INIT=LLCA + INIT=LLCB + INIT=LLDA + INIT=LLDB + INIT=LLO + INIT=LLSA + INIT=LLSB
NPSHS	1
RPSMS	1
RPSE0	1
TOR1	1
TTP1	BB5=S*DI=S
TTP2	BB5=S*DI=F
TTP3	BB5=F*DI=S
TTPF	1
IVC1	1
DV1F	RE=F+RB=F*RC=F+NH1=F*NH2=F+DW=F*LV=F
DV11	DW=S*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV12	DW=S*LV=S*NH1=S*NH2=S*(RB=F+RC=F)
DV13	DW=S*LV=F*NH1=S*NH2=S*RB=S*RC=S
DV14	DW=F*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV15	DW=S*LV=S*(NH1=F+NH2=F)*RB=S*RC=S
DV1F	1
DV2F	RF=F+RB=F*RC=F+NH1=F*NH2=F+DW=F*LV=F
DV25	RE=F*DV1=F*DW=S*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV21	DV1=S*DW=S*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV22	DV1=F*DW=S*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV24	RE=F*DV1=F*DW=S*LV=S*NH1=S*NH2=S*(RB=F+RC=F)

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: LLRSN

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SF	Split Fraction Assignment Rule
DV23	DV1=S*DW=S*LV=S*NH1=S*NH2=S*(RB=F+RC=F)
DV24	DV1=F*DW=S*LV=S*NH1=S*NH2=S*(RB=F+RC=F)
DV27	RE=F*DV1=F*DW=S*LV=F*NH1=S*NH2=S*RB=S*RC=S
DV28	DV1=S*DW=S*LV=F*NH1=S*NH2=S*RB=S*RC=S
DV29	DV1=F*DW=S*LV=F*NH1=S*NH2=S*RB=S*RC=S
DV2A	RE=F*DV1=F*DW=F*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV2B	DV1=S*DW=F*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV2C	DV1=F*DW=F*LV=S*NH1=S*NH2=S*RB=S*RC=S
DV2D	RE=F*DV1=F*DW=S*LV=S*(NH1=F+NH2=F)*RB=S*RC=S
DV2E	DV1=S*DW=S*LV=S*(NH1=F+NH2=F)*RB=S*RC=S
DV2G	DV1=F*DW=S*LV=S*(NH1=F+NH2=F)*RB=S*RC=S
DV2F	1
LPCIF	RE=F + DV1=F + NPSHLOST
LPCI2	1
LPCIIF	RF=F + DV2=F + NPSHLOST
LPCII2	LPCI=S
LPCII4	RE=F
LPCII6	LPCI=F*RE=S
LPCIIF	1
CSF	(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+-EECW)*(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+ -EECW) + NPSHLOST
CS1	-(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+ -EECW)*-(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+ -EECW)
CS2	-(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+ -EECW)*(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+ -EECW)
CS2B	(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+-EECW)*-(RF=F+AC=F+DB=F+AD=F+DD=F+NPII=F+ CASSIG+DW=F*LV=F+RB=F+-EECW)
CSF	1 Comments Core Spray Loop II Pipe Break Large LOCA
SIS	LPCI=S*RPA=S*RPC=S + LPCII=S*RPB=S*RPD=S + LPCI=S*LPCII=S*(RPA=S+RPC=S)*(RPB=S+RPD=S)

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: LLRSN
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SF	Split Fraction Assignment Rule
SIF	I
OSPC1	RPSM=S*RPSE=S
OSPCF	1
SPIF	RE=F + OSPC=F + NPSHLOST
SPI2	1
SPIIF	OSPC=F + RF=F + NPSHLOST
SPII4	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=S
SPII5	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*RE=S
SPII6	(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*RE=F
SPIIF	1
SPCF	-(SPI=S)*-(SPII=S)
SPCS	SPI=S*(RPA=S*HXA=S + RPC=S*HXC=S) + SPII=S*(RPB=S*HXB=S+RPD=S*HXD=S)
SPCF	1
ODWS1	1
DWSF	PX1=F*PX2=F + (RPA=F*RPC=F +RH=F+NOGB) * (RPB=F*RPD=F+RI=F + NOGD)
DWS1	PX1=S*PX2=S*(RPA=S+RPC=S)*-NOGB*(RPB=S+RPD=S)*-NOGD
DWS2	(RPA=F*RPC=F +RH=F+NOGB+PX1=F) * (RPB=F*RPD=F+RI=F + NOGD+PX2=F)
DWSF	1

Model Name: U1COP2-9
Macro for Event Tree: LLRSN

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Page 1

Macro	Macro Rule / Comments
ALTINJRHSW	RPSM=B THIS MACRO IS NEEDED IN THE CETS
ALTINJU2X	RPSN=B THIS MACRO IS NEEDED IN THE CETS
BUCKET	RPSM=B
CILFAIL	CIL=F
CLASS1A	RPSM=B
CLASS1B	RPSM=B
CLASS1BE	RPSM=B
CLASS1BL	RPSM=B
CLASS1C	RPSM=B
CLASS1D	RPSM=B
CLASS1E	RPSM=B
CLASS2	RPSM=B
CLASS2A	RPSM=B
CLASS2L	OSPC=F + SPC=F
CLASS2T	RPSM=B
CLASS2V	RPSM=B
CLASS3A	RPSM=B
CLASS3B	RPSM=B

Model Name: U1COP2-9
Macro for Event Tree: LLRSN

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Macro	Macro Rule / Comments
CLASS3C	-(SI=S)+ -(TTP=S+IVC=S)
CLASS3D	-(TOR=S)
CLASS4	RPSM=F
CLASS5	-(TTP=S)*-(IVC=S)
DWSPRAY	DWS=S THIS MACRO IS NEEDED IN THE CETS
EECW	EA=S*(EB=S + EC=S + ED=S) + EB=S8*(EC=S + ED=S) + EC=S*ED=S
EMDEPHDWR	RPSM=B THIS.MACRO IS NEEDED IN THE CETS
HIGH	RPSM=B
HPI	RPSM=B
LOW	INIT=LLSA + INIT=LLSB
LPCIISUP	RF=S*((NPII=S*DW=S) + LV=S)
LPCISUP	RE=S*((NPI=S*DW=S) + LV=S) LOOP I LPCI SUPPORT
LPI	SI=S
NOACREC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOCD	RPSM=S * TOR=S*(TTP=S+IVC=S)*SI=S*SPC=S
NODC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NORV	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOSRV	RPSM=B THIS MACRO IS NEEDED IN THE CETS

Model Name: U1COP2-9
Macro for Event Tree: LLRSN

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Macro	Macro Rule / Comments
NPSHLOST	CIL=F*NPSH=F
OPDEPL1	RPSM=B THIS MACRO IS NEEDED IN THE CETS
RHRSPCOOL	SPC=S
SORV	RPSM=S LARGE LOCAS ARE ALWAYS DEPRESSURIZEED

MODEL Name: U1ERIN
Event Tree: CETN1.ETI

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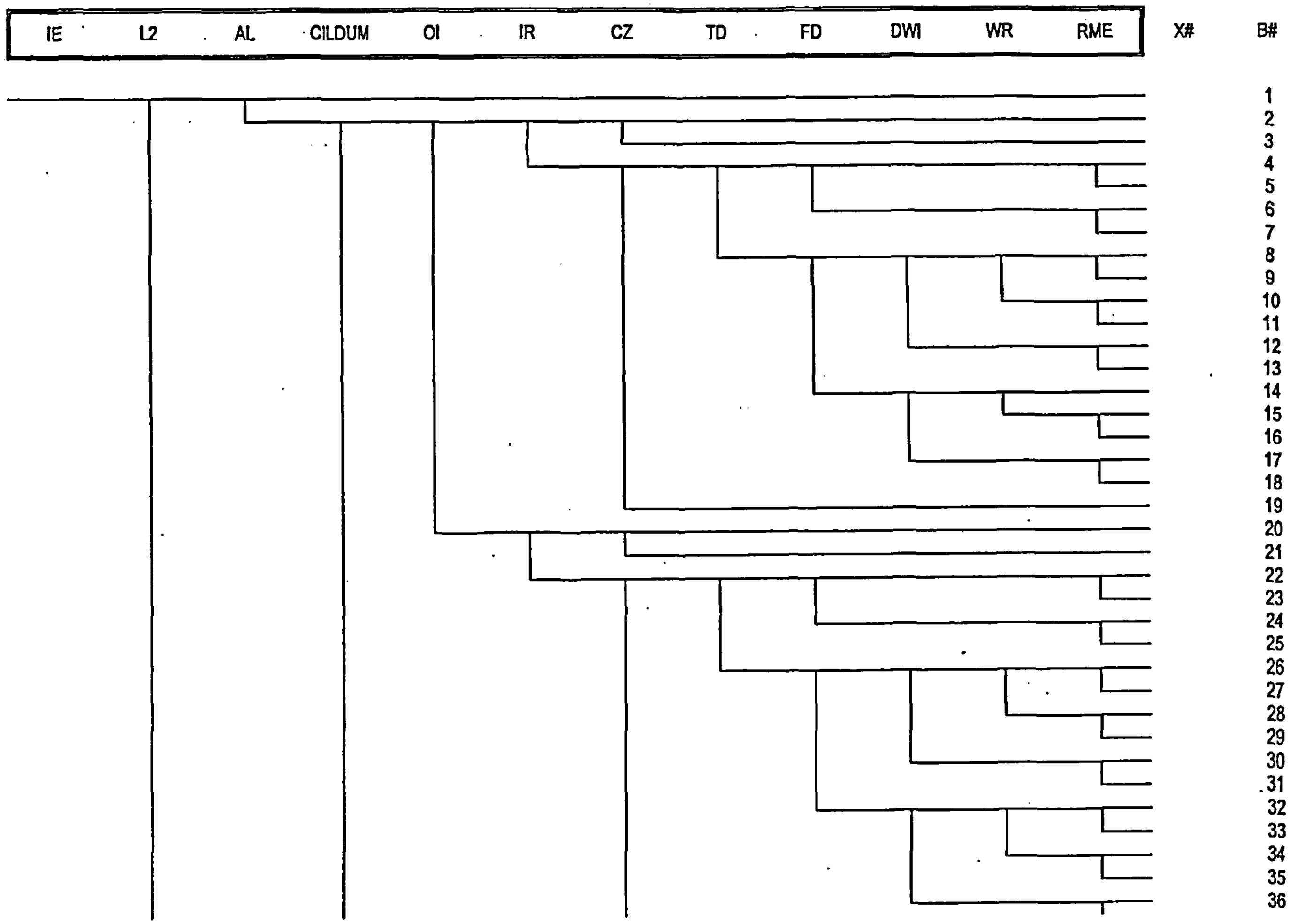
13:36:50 February 16, 2006



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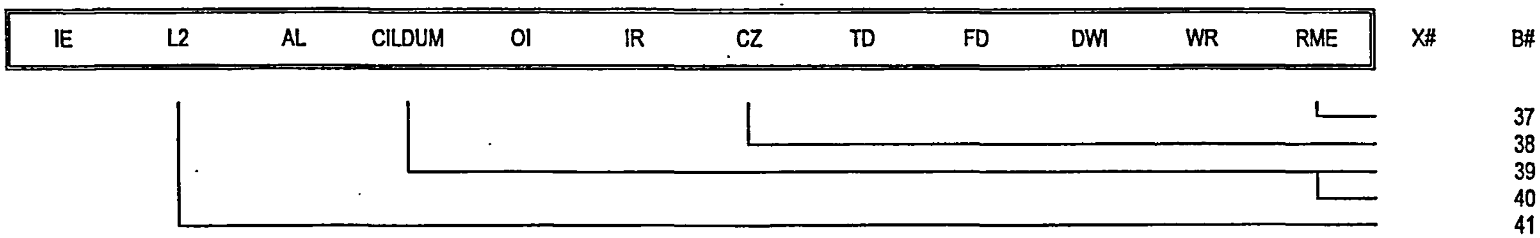
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Model Name: U1COP2-9
Top Events for Event Tree: CETN1

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Top Event Name	Description
L2	LEVEL 2 /LERF RESULTS
AL	CET1 LOGIC NODE FOR CLASS 2 AND CLASS1BL
CILDUM	CIL DUMMY TOP
OI	OPERATORS DEPRESSURIZE RPV (L2)
IR	IN-VESSEL RECOVERY
CZ	CONTAINMENT ISOLATED AND INTACT
TD	INJECTION ESTABLISHED
FD	CONTAINMENT FLOODING
DWI	NO DIRECT DRYWELL RELEASE PATH
WR	WET AIR SPCE FAILURE
RME	CONTAINMENT BUILDING EFFECTIVE

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
L20	1 Comments L20=0 IMPLIES LEVEL 1; L20=1 IMPLIES LEVEL2; USE MFF TO CHANGE
ALF	CLASS1A + CLASS1BE + CLASS1C + CLASS1D + CLASS1E + CLASS3A + CLASS3B + CLASS3C
ALO	NOCD + CLASS1BL + CLASS2A + CLASS2L + CLASS2T + CLASS2V + (CLASS3D + CLASS4 + CLASS5) + BUCKET Comments CLASS 3D AND CLASS 4 ARE EVALUATED FOR LERF
CILDF	CILFAIL
CILDS	1
OIS	CLASS3A + CLASS3B + CLASS3C + LOW
OI1	CLASS2A + CLASS2T + NORV*(CLASS1A + CLASS1BE + CLASS1BL+ CLASS1C) + CLASS1B*(NOACREC + NODC)
OI4	CLASS1B
OI3	-OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
OI2	OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
IR1	OI=F*(CLASS1A + CLASS1C)
IR3	CLASS1BE
IR4	CLASS1BL
IR5	OI=F*CLASS1D
IR6	OI=S*CLASS1D Comments the irginal U1 L2 model
IR7	OI=F*CLASS1E
IR8	OI=S*CLASS1E
IR2	OI=S Comments LOW PRESSURE INJECTION IMPLICIT
IRF	1
CZ2	IR=F*OI=S
CZ4	IR=F*OI=F
CZ1	IR=S*OI=S
CZ3	IR=S*OI=F

Model Name: U1COP2-9

Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
CZF	1
TD1	CLASS1E
TD2	OI=S*DWSPRAY
TD3	-(OI=B)*CLASS1BE
TD4	-(OI=B)*CLASS1BL
TD8	OI=F*CLASS1A
TDF	1
FD1	ALTINJRHSW + DWSPRAY
FD2	TD=S*(CLASS1A + CLASS1BE + CLASS1BL + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD3	TD=F*(CLASS1A + CLASS1C + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD4	TD=F*(CLASS1BE + CLASS1BL)
DWIF	1
WR1	DW=S
RME8	CLASS1BL Comments TD=S*DWSPRAY*RHRSPCOOL This was an assumption that resulted in 100 RBE
RME7	OI=F
RME6	OI=S*TD=S*FD=S*DWS=S
RME5	OI=S*TD=S*FD=S*DWS=F
RME4	OI=S*TD=S*FD=F
RME3	OI=S*TD=F*FD=F
RMEF	1
L20	1 Comments L20=0 IMPLIES LEVEL 1; L20=1 IMPLIES LEVEL2; USE MFF TO CHANGE
ALF	CLASS1A + CLASS1BE + CLASS1C + CLASS1D + CLASS1E + CLASS3A + CLASS3B + CLASS3C
AL0	NOCD + CLASS1BL + CLASS2A + CLASS2L + CLASS2T + CLASS2V + (CLASS3D + CLASS4 + CLASS5) + BUCKET Comments CLASS 3D AND CLASS 4 ARE EVALUATED FOR LERF
CILDF	CILFAIL
CILDS	1

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
OIS	CLASS3A + CLASS3B + CLASS3C + LOW
OI1	CLASS2A + CLASS2T + NORV*(CLASS1A + CLASS1BE + CLASS1BL+ CLASS1C) + CLASS1B*(NOACREC + NODC)
OI4	CLASS1B
OI3	-OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
OI2	OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
IR1	OI=F*(CLASS1A + CLASS1C)
IR3	CLASS1BE
IR4	CLASS1BL
IR5	OI=F*CLASS1D
IR6	OI=S*CLASS1D Comments the irginal U1 L2 model
IR7	OI=F*CLASS1E
IR8	OI=S*CLASS1E
IR2	OI=S Comments LOW PRESSURE INJECTION IMPLICIT
IRF	1
CZ2	IR=F*OI=S
CZ4	IR=F*OI=F
CZ1	IR=S*OI=S
CZ3	IR=S*OI=F
CZF	1
TD1	CLASS1E
TD2	OI=S*DWSPRAY
TD3	-(OI=B)*CLASS1BE
TD4	-(OI=B)*CLASS1BL
TD8	OI=F*CLASS1A
TDF	1

Model Name: U1COP2-9

Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
FD1	ALTINJRHSW + DWSPRAY
FD2	TD=S*(CLASS1A + CLASS1BE + CLASS1BL + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD3	TD=F*(CLASS1A + CLASS1C + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD4	TD=F*(CLASS1BE + CLASS1BL)
DWIF	1
WR1	DW=S
RME8	CLASS1BL Comments TD=S*DWSPRAY*RHRSPCOOL This was an assumption that resulted in 100 RBE
RME7	OI=F
RME6	OI=S*TD=S*FD=S*DWS=S
RME5	OI=S*TD=S*FD=S*DWS=F
RME4	OI=S*TD=S*FD=F
RME3	OI=S*TD=F*FD=F
RMEF	1
L20	1 Comments L20=0 IMPLIES LEVEL 1; L20=1 IMPLIES LEVEL2; USE MFF TO CHANGE
ALF	CLASS1A + CLASS1BE + CLASS1C + CLASS1D + CLASS1E + CLASS3A + CLASS3B + CLASS3C
ALO	NOCD + CLASS1BL + CLASS2A + CLASS2L + CLASS2T + CLASS2V + (CLASS3D + CLASS4 + CLASS5) + BUCKET Comments CLASS 3D AND CLASS 4 ARE EVALUATED FOR LERF
CILDF	CILFAIL
CILDS	1
OIS	CLASS3A + CLASS3B + CLASS3C + LOW
OI1	CLASS2A + CLASS2T + NORV*(CLASS1A + CLASS1BE + CLASS1BL+ CLASS1C) + CLASS1B*(NOACREC + NODC)
OI4	CLASS1B
OI3	-OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
OI2	OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
IR1	OI=F*(CLASS1A + CLASS1C)
IR3	CLASS1BE
IR4	CLASS1BL
IR5	OI=F*CLASS1D
IR6	OI=S*CLASS1D Comments the irginal U1 L2 model
IR7	OI=F*CLASS1E
IR8	OI=S*CLASS1E
IR2	OI=S Comments LOW PRESSURE INJECTION IMPLICIT
IRF	1
CZ2	IR=F*OI=S
CZ4	IR=F*OI=F
CZ1	IR=S*OI=S
CZ3	IR=S*OI=F
CZF	1
TD1	CLASS1E
TD2	OI=S*DWSPRAY
TD3	-(OI=B)*CLASS1BE
TD4	-(OI=B)*CLASS1BL
TD8	OI=F*CLASS1A
TDF	1
FD1	ALTINJRHSW + DWSPRAY
FD2	TD=S*(CLASS1A + CLASS1BE + CLASS1BL + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD3	TD=F*(CLASS1A + CLASS1C + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD4	TD=F*(CLASS1BE + CLASS1BL)
DWIF	1
WR1	DW=S
RME8	CLASS1BL

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
	Comments TD=S*DWSPRAY*RHRSPCOOL This was an assumption that resulted in 100 RBE
RME7	OI=F
RME6	OI=S*TD=S*FD=S*DWS=S
RME5	OI=S*TD=S*FD=S*DWS=F
RME4	OI=S*TD=S*FD=F
RME3	OI=S*TD=F*FD=F
RMEF	1
L20	1 Comments L20=0 IMPLIES LEVEL 1; L20=1 IMPLIES LEVEL2; USE MFF TO CHANGE
ALF	CLASS1A + CLASS1BE + CLASS1C + CLASS1D + CLASS1E + CLASS3A + CLASS3B + CLASS3C
ALO	NOCD + CLASS1BL + CLASS2A + CLASS2L + CLASS2T + CLASS2V + (CLASS3D + CLASS4 + CLASS5) + BUCKET Comments CLASS 3D AND CLASS 4 ARE EVALUATED FOR LERF
CILDF	CILFAIL
CILDS	1
OIS	CLASS3A + CLASS3B + CLASS3C + LOW
OI1	CLASS2A + CLASS2T + NORV*(CLASS1A + CLASS1BE + CLASS1BL+ CLASS1C) + CLASS1B*(NOACREC + NODC)
OI4	CLASS1B
OI3	-OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
OI2	OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
IR1	OI=F*(CLASS1A + CLASS1C)
IR3	CLASS1BE
IR4	CLASS1BL
IR5	OI=F*CLASS1D
IR6	OI=S*CLASS1D Comments the irginal U1 L2 model
IR7	OI=F*CLASS1E
IR8	OI=S*CLASS1E

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
IR2	OI=S Comments LOW PRESSURE INJECTION IMPLICIT
IRF	1
CZ2	IR=F*OI=S
CZ4	IR=F*OI=F
CZ1	IR=S*OI=S
CZ3	IR=S*OI=F
CZF	1
TD1	CLASS1E
TD2	OI=S*DWSPRAY
TD3	-(OI=B)*CLASS1BE
TD4	-(OI=B)*CLASS1BL
TD8	OI=F*CLASS1A
TDF	1
FD1	ALTINJRHSW + DWSPRAY
FD2	TD=S*(CLASS1A + CLASS1BE + CLASS1BL + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD3	TD=F*(CLASS1A + CLASS1C + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD4	TD=F*(CLASS1BE + CLASS1BL)
DWIF	1
WR1	DW=S
RME8	CLASS1BL Comments TD=S*DWSPRAY*RHRSPCOOL This was an assumption that resulted in 100 RBE
RME7	OI=F
RME6	OI=S*TD=S*FD=S*DWS=S
RME5	OI=S*TD=S*FD=S*DWS=F
RME4	OI=S*TD=S*FD=F
RME3	OI=S*TD=F*FD=F
RMEF	1
L20	1

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
	Comments L20=0 IMPLIES LEVEL 1; L20=1 IMPLIES LEVEL2; USE MFF TO CHANGE
ALF	CLASS1A + CLASS1BE + CLASS1C + CLASS1D + CLASS1E + CLASS3A + CLASS3B + CLASS3C
AL0	NOCD + CLASS1BL + CLASS2A + CLASS2L + CLASS2T + CLASS2V + (CLASS3D + CLASS4 + CLASS5) + BUCKET Comments CLASS 3D AND CLASS 4 ARE EVALUATED FOR LERF
CILDF	CILFAIL
CILDS	1
OIS	CLASS3A + CLASS3B + CLASS3C + LOW
OI1	CLASS2A + CLASS2T + NORV*(CLASS1A + CLASS1BE + CLASS1BL+ CLASS1C) + CLASS1B*(NOACREC + NODC)
OI4	CLASS1B
OI3	-OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
OI2	OPDEPL1*(CLASS1A + CLASS1C + CLASS1D) Comments change! hIGH PRESSURE LERF
IR1	OI=F*(CLASS1A + CLASS1C)
IR3	CLASS1BE
IR4	CLASS1BL
IR5	OI=F*CLASS1D
IR6	OI=S*CLASS1D Comments the irginal U1 L2 model
IR7	OI=F*CLASS1E
IR8	OI=S*CLASS1E
IR2	OI=S Comments LOW PRESSURE INJECTION IMPLICIT
IRF	1
CZ2	IR=F*OI=S
CZ4	IR=F*OI=F
CZ1	IR=S*OI=S
CZ3	IR=S*OI=F
CZF	1

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: CETN1

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SF	Split Fraction Assignment Rule
TD1	CLASS1E
TD2	OI=S*DWSPRAY
TD3	-(OI=B)*CLASS1BE
TD4	-(OI=B)*CLASS1BL
TD8	OI=F*CLASS1A
TDF	1
FD1	ALTINJRHSW + DWSPRAY
FD2	TD=S*(CLASS1A + CLASS1BE + CLASS1BL + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD3	TD=F*(CLASS1A + CLASS1C + CLASS1D + CLASS3A + CLASS3B + CLASS3C)
FD4	TD=F*(CLASS1BE + CLASS1BL)
DWIF	1
WR1	DW=S
RME8	CLASS1BL Comments TD=S*DWSPRAY*RHRSPCOOL This was an assumption that resulted in 100 RBE
RME7	OI=F
RME6	OI=S*TD=S*FD=S*DWS=S
RME5	OI=S*TD=S*FD=S*DWS=F
RME4	OI=S*TD=S*FD=F
RME3	OI=S*TD=F*FD=F
RMEF	1

Model Name: U1COP2-9
Macro for Event Tree: CETN1

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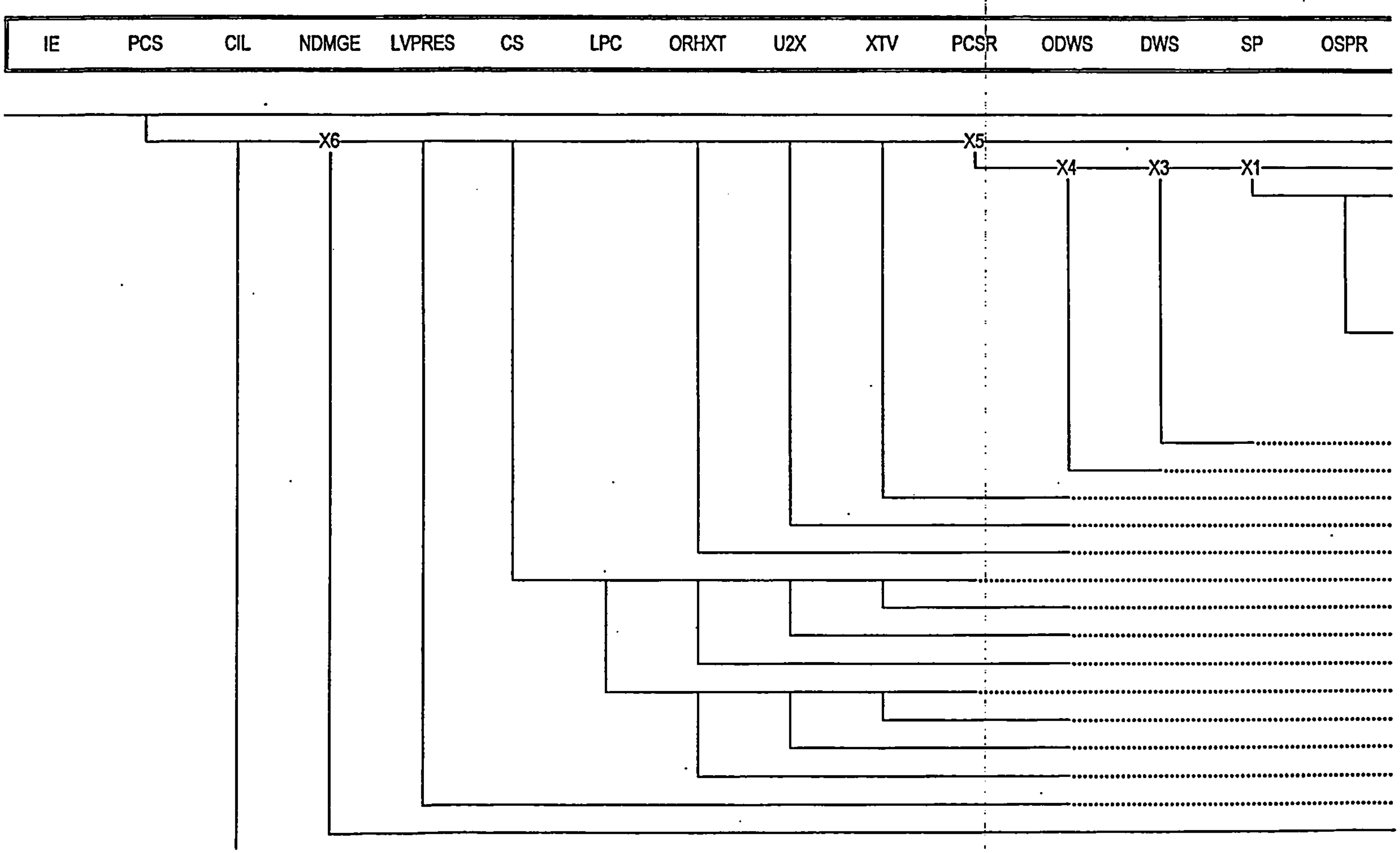
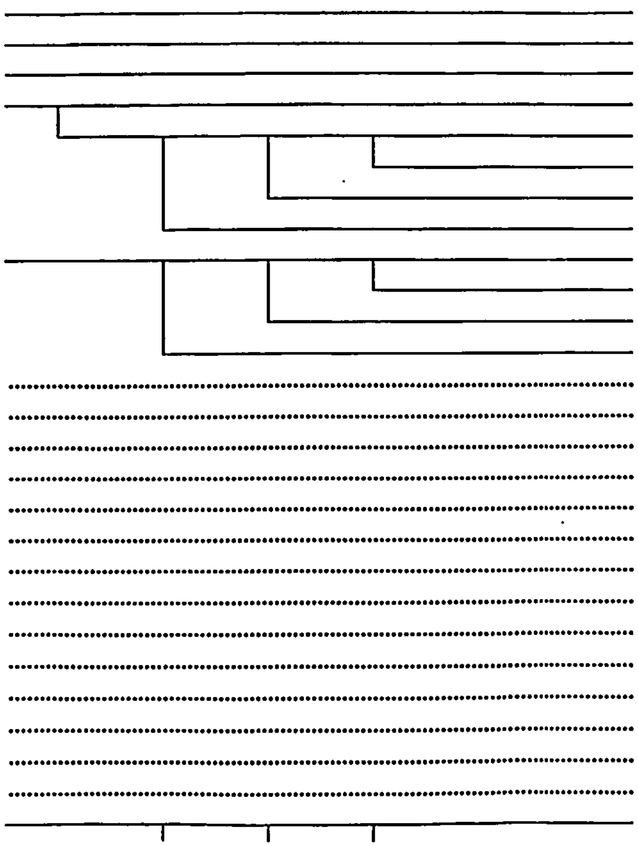
Macro	Macro Rule / Comments
C1C3LERF	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)
	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)
	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)
	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)
	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)

MODEL Name:
Event Tree: LPGTETN.ETI

11:33:48 June 27, 2006





SPR	OWWV	VNT	CRD	OAI	AVI	X#	B#	S#
							1	1
							2	2
							3	3
							4	4
							5	5
							6	6
							7	7
							8	8
							9	9
							10	10
							11	11
							12	12
						X1	13	13-22
						X3	14	23-42
						X4	15	43-82
						X4	16	83-122
						X4	17	123-162
						X5	18	163-203
						X4	19	204-243
						X4	20	244-283
						X4	21	284-323
						X5	22	324-364
						X4	23	365-404
						X4	24	405-444
						X4	25	445-484
						X4	26	485-524
							27	525

MODEL Name:

Event Tree: LPGTETN.ETI

11:33:48 June 27, 2006

IE	PCS	CIL	NDMGE	LVPRES	CS	LPC	ORHXT	U2X	XTV	PCSR	ODWS	DWS	SP	OSPR

MODEL Name:

Event Tree: LPGTETN.ETI

11:33:48 June 27, 2006

SPR	OWWV	VNT	CRD	OAI	AVI	X#	B#	S#
							28	526
							29	527
							30	528
						X6	31	529-1055

Model Name: U1COP2-9
Macro for Event Tree: LPGTETN

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Macro	Macro Rule / Comments
ALTINJRHSW	RPSM=B
BUCKET	-CLASS1A*-CLASS1B*-CLASS1BE*-CLASS1BL*-CLASS1C*-CLASS1D*-CLASS1E*-CLASS2A*-CLASS2L*-CLASS2T*-CLASS2V*-CLASS3A*-CLASS3B*-CLASS3C *-CLASS3D *-CLASS4*-CLASS5 THIS WAS SEPARATED FROM NOCD TO TEST WHAT IS MISSING. IT WILL BE USED IN THE DAMAGE STATES
CILFAIL	CIL=F
CLASS1A	-NOCD1*NDMGE=S*(ORVD=F+ RVD=F + OHPC=F*OHPR=F)*LVPRES=F*-CLASS5 NDMGE=S*RPSM=S*RPSE=S*-(FWH=S)*(-(HPI=S)*-(RCI=S)+-(OHPC=S)*-(OHPR=S))*LVPRES=F
CLASS1B	RPSM=B
CLASS1BE	-NOCD1*RPSM=S*RPSE=S*OG5=F*DGC=F*EPR30=F*-(TTP=F)*-(IVC=F)*(HPI=F*RCI=F + -(OHPC=S)*-(OHPR=S))*-CLASS5*-CLASS1A
CLASS1BL	-NOCD1*RPSM=S*RPSE=S*OG5=F*DGC=F*EPR6=F*-(TTP=F)*-(IVC=F)*-(HPI=F*RCI=F + -(OHPC=S)*-(OHPR=S))*-CLASS5*-CLASS1A*-CLASS1BE
CLASS1C	RPSM=B
CLASS1D	-NOCD1*RPSM=S*RPSE=S*LVPRES=S*(-(LPC=S)*-(CS=S)+ -(OLPC=S))*-CLASS5*-CLASS1A*-CLASS1BE*-CLASS1BL
CLASS1E	-NOCD1*RPSM=S*RPSE=S*LVPRES=F*DE=F*DH=F*DG=F*-CLASS5*-CLASS1A*-CLASS1BE*-CLASS1BL*-CLASS1D
CLASS2A	-NOCD1*RPSM=S*RPSE=S*NDMGE=S*-(SP=S*SPRHR+SPR=S*SPRHR)*-(CND=S+PCSR=S)*-CLASS5*-CLASS1A*-CLASS1BE*-CLASS1BL*-CLASS1D*-CLASS1E
CLASS2L	-NOCD1*RPSM=S*RPSE=S*(INIT=SLOCA + RVC=SORV1 + RVC=SORV2)*NDMGE=S*-(SP=S*SPRHR+SPR=S*SPRHR)*-(CND=S+PCSR=S)*-CLASS5*-CLASS1A*-CLASS1BE*-CLASS1BL*-CLASS1D*-CLASS1E*-CLASS2A
CLASS2T	RPSM=B
CLASS2V	RPSM=B
CLASS3A	-NOCD1*RPSM=B

Model Name: U1COP2-9
Macro for Event Tree: LPGTETN

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Macro	Macro Rule / Comments
CLASS3B	-NOCD1*RPSM=S*RPSE=S*(INIT=SLOCA+RVC=SORV1)*LVPRES=F*-CLASS5*-CLASS1A*-CLASS1BE*-CLASS1BL*-CLASS1D*-CLASS1E*-CLASS2A*-CLASS2L
CLASS3C	-NOCD1*RPSM=S*RPSE=S*(RVC=SORV1*HPI=S + RVC=SORV2)*(CS=S+LPC=S)*((SP=S+SPR=S)*SPRHR)*-CLASS1A*-CLASS1BE*-CLASS1BL*-CLASS1D*-CLASS1E*-CLASS2A*-CLASS2L*-CLASS3B
CLASS3D	-NOCD1*RPSM=S*RPSE=S*(INIT=SLOCA+RVC=SORV1+RVC=SORV2)*-(TOR=S)*-CLASS1A*-CLASS1BE*-CLASS1BL*-CLASS1D*-CLASS1E*-CLASS2A*-CLASS2L*-CLASS3B*-CLASS3C
CLASS4	RPSM=B
CLASS5	-NOCD1*TTP=F*IVC=F
CSASUP	AAOK*DA=S*RERCVRY*(EECW=S+EECWR=S) Core Spray pump A support (sans actuation)
CSBSUP	ABOK*DC=S*RFRCVRY*(EECW=S+EECWR=S)*-(CASTRAN*RVD=S) Core Spray pump B support (sans actuation)
CSCSUP	ACOK*DB=S*RERCVRY*(EECW=S+EECWR=S) Core Spray pump C support (sans actuation)
CSDSUP	ADOK*DD=S*RFRCVRY*(EECW=S+EECWR=S)*-(CASTRAN*RVD=S) Core Spray pump B support (sans actuation)
DWSPRAY	DWS=S
EMDEPHDWR	-(RVD=S)*(RB=F*RC=F*RD=F + EPR6=F)
HIGH	-LOW
HR6ONLY	(RCI=S + HPI=S)*L8F=S*(OHPC=S+OHPR=S)
HRLPT	RVC=SORV1 + RVC=SORV2 + INIT=SLOCA HPCI/RCIC LOW PRESSURE TRIP;
HXASW	SW1+SW2+SW3+SW4+SW5+SW6+SW7+SW8+SW9+SW10+SW11+SW12+SW13+SW14+SW15+SW16+SW17+SW18+SW19+SW20+SW21+SW22+SW23+SW24+SW25+SW26+SW27+SW28+SW29+SW30+SW31+SW32+SW33+SW34+SW35+SW36
HXBSW	SW1+SW2+SW7+SW8+SW9+SW10+SW11+SW12+SW13+SW14+SW15+SW22+SW23+SW24+SW25+SW26+SW27+SW28+SW29+SW30+SW37+SW38+SW39+SW40+SW41+SW42+SW43+SW44+SW45+SW46+SW47+SW48+SW49+SW50+SW51+SW52

Model Name: U1COP2-9
Macro for Event Tree: LPGTETN

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Macro	Macro Rule / Comments
HXCSW	SW3+SW4+SW8+SW9+SW12+SW13+SW16+SW17+SW18+SW19+SW20+SW23+SW24+SW27+SW28+SW31+SW32+SW33+SW34+SW35+SW37+SW38+SW41+SW42+SW43+SW44+SW45+SW47+SW48+SW49+SW50+SW51+SW53+SW54+SW55+SW56
HXDSW	SW5+SW6+SW10+SW11+SW14+SW15+SW17+SW18+SW19+SW20+SW21+SW25+SW26+SW29+SW30+SW32+SW33+SW34+SW35+SW36+SW39+SW40+SW42+SW43+SW44+SW45+SW46+SW48+SW49+SW50+SW51+SW52+SW53+SW54+SW55+SW56
LOW	LVPRES=S
LPCI2	RVC=SORV1 + RVC=SORV2 + INIT=SLOCA *Conditions where 2 RHR Pumps/HXs required for suppression pool cooling*
LPI	LPC=S
NCDCRDLT	(RCI=S+HPI=S)*(OHPR=S+OHPC=S)*CRD=S*((SP=S+SPR=S)*SPRHR+VNT=S) At six hours, CRD is capable of removing decay heat (kevel control)
NCDHRLT	-HRLPT* ((HPL=S+RCL=S) *(SP=S+SPR=S)*SPRHR) HPCI/RCIC used for shutdown, HPCI nor RCI tripped due to low pressure; long term injection with suppression pool pressure control
NCDLVPRES	(LVPRES=S+RVD=S)*(CND=S + PCSR=S + (CS=S+LPC=S + -MULTIT*XTV=S)*((SP=S+SPR=S)*SPRHR)) Rmoved OLPC=S; CRD can be used at 4 hrs or OLPC is recoverable
NCDSORV	(RVC=SORV1*(HPI=S+RCI=S) + RVC=SORV2) *(CS=S+LPC=S)*((SP=S+SPR=S)*SPRHR)
NOACREC	EPR6=F
NOCD	NOCD1
NOCD1	RPSM=S*RPSE=S* -(INIT=IOOVA+ INIT=LOCHSA+INIT=LOFWA + INIT=LOSPA+INIT=TTA) * (NCDHRLT + NCDSORV + NCDLVPRES + NCDCRDLT + FWSD=S + PCSR=S) NONATWS TRANSIENTS
NOCDU2X	-MULTIT*LVPRES=S*XTV=S*VNT=S
NODC	DE=F*DG=F*DH=F
NORV	RVC=SORV0*-(RVD=S) RVC=SORV0*-(RVD=S)
OPDEPL1	ORVD=S

Model Name: U1COP2-9
Macro for Event Tree: LPGTETN

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Macro	Macro Rule / Comments
RHRSPCOOL	SP=S + SPR=S
SORV	RVC=SORV1 + RVC=SORV2
SPRHR	RPA=S*HXA=S*(HXASW+OG5=F*EPR6=S)+RPB=S*HXB=S*(HXBSW+OG5=F*EPR6=S)+RPC=S*HXC=S*(HXCSW+OG5=F*EPR6=S)+RPD=S*HXD=S*(HXDSW + OG5=F*EPR6=S) + XTV=S*-MULTIT
SW1	SW1A=S*SW2A=S*SW1B=S
SW10	SW1A=S*SW1B=S*SW1D=S
SW11	SW1A=S*SW1B=S*SW2D=S
SW12	SW1A=S*SW2B=S*SW1C=S
SW13	SW1A=S*SW2B=S*SW2C=S
SW14	SW1A=S*SW2B=S*SW1D=S
SW15	SW1A=S*SW2B=S*SW2D=S
SW16	SW1A=S*SW1C=S*SW2C=S
SW17	SW1A=S*SW1C=S*SW1D=S
SW18	SW1A=S*SW1C=S*SW2D=S
SW19	SW1A=S*SW2C=S*SW1D=S
SW2	SW1A=S*SW2A=S*SW2B=S
SW20	SW1A=S*SW2C=S*SW2D=S
SW21	SW1A=S*SW1D=S*SW2D=S
SW22	SW2A=S*SW1B=S*SW2B=S

Model Name: U1COP2-9
Macro for Event Tree: LPGTETN

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Macro	Macro Rule / Comments
SW23	SW2A=S*SW1B=S*SW1C=S
SW24	SW2A=S*SW1B=S*SW2C=S
SW25	SW2A=S*SW1B=S*SW1D=S
SW26	SW2A=S*SW1B=S*SW2D=S
SW27	SW2A=S*SW2B=S*SW1C=S
SW28	SW2A=S*SW2B=S*SW2C=S
SW29	SW2A=S*SW2B=S*SW1D=S
SW3	SW1A=S*SW2A=S*SW1C=S
SW30	SW2A=S*SW2B=S*SW2D=S
SW31	SW2A=S*SW1C=S*SW2C=S
SW32	SW2A=S*SW1C=S*SW1D=S
SW33	SW2A=S*SW1C=S*SW2D=S
SW34	SW2A=S*SW2C=S*SW1D=S
SW35	SW2A=S*SW2C=S*SW2D=S
SW36	SW2A=S*SW1D=S*SW2D=S
SW37	SW1B=S*SW2B=S*SW1C=S
SW38	SW1B=S*SW2B=S*SW2C=S
SW39	SW1B=S*SW2B=S*SW1D=S

Model Name: U1COP2-9
Macro for Event Tree: LPGTETN

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Macro	Macro Rule / Comments
SW4	SW1A=S*SW2A=S*SW2C=S
SW40	SW1B=S*SW2B=S*SW2D=S
SW41	SW1B=S*SW1C=S*SW2C=S
SW42	SW1B=S*SW1C=S*SW1D=S
SW43	SW1B=S*SW1C=S*SW2D=S
SW44	SW1B=S*SW2C=S*SW1D=S
SW45	SW1B=S*SW2C=S*SW2D=S
SW46	SW1B=S*SW1D=S*SW2D=S
SW47	SW2B=S*SW1C=S*SW2C=S
SW48	SW2B=S*SW1C=S*SW1D=S
SW49	SW2B=S*SW1C=S*SW2D=S
SW5	SW1A=S*SW2A=S*SW1D=S
SW50	SW2B=S*SW2C=S*SW1D=S
SW51	SW2B=S*SW2C=S*SW2D=S
SW52	SW2B=S*SW1D=S*SW2D=S
SW53	SW1C=S*SW2C=S*SW1D=S
SW54	SW1C=S*SW2C=S*SW2D=S
SW55	SW1C=S*SW1D=S*SW2D=S

Model Name: U1COP2-9
Macro for Event Tree: LPGTETN

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Macro	Macro Rule / Comments
SW56	SW2C=S*SW1D=S*SW2D=S
SW6	SW1A=S*SW2A=S*SW2D=S
SW7	SW1A=S*SW1B=S*SW2B=S
SW8	SW1A=S*SW1B=S*SW1C=S
SW9	SW1A=S*SW1B=S*SW2C=S

Model Name: U1COP2-9

Split Fraction Assignment Rule for Event Tree: LPGTETN

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SF	Split Fraction Assignment Rule
PCSS	FWSD=S + (INIT=IOOVA+ INIT=LOCHSA+INIT=LOFWA + INIT=LOSPA+INIT=TTA) Comments Successful if the main condenser, condensate, and feedwater hardware (sans level control) are functional, and the feedwater controller is functional or operators control level or a Level 8 trip occurs with successful operator acti
PCSF	FWSD=F
NDMGES	1
PCSRF	INIT=LOSP + RVC=SORV1 + RVC=SORV2 + FWH=F*(RCI=F*HPI=F+OHPC=F*OHPR=F)*(ORVD=F + RVD=F)
PCSR6	INIT=L5000+ INIT=L500PA
PCSR5	INIT=TBU
PCSR4	INIT=LOPA
PCSR3	INIT=IMSIV
PCSR2	INIT=TLFW + INIT=TLCF
PCSR1	INIT=FLRB3S
PCSRF	1
LVPRS	(HPI=S + RCI=S)*(OHPC=S+OHPR=S) + RVC=SORV1 + RVC=SORV2 + RVD=S Comments The vessel is at low pressure if HPCI ran for six hours, emergency depressurization, or a stuck open SRV or 2 SORVs
LVPRF	1
CSF	(-CSASUP*-CSCSUP + EECW=F*EECWR=F +TOR=F + -(LV=S)*(-(DW=S)+-(NPI=S))*-(OLPC=S)) * (-CSBSUP*-CSDSUP+(EECW=S+EECWR=S) +TOR=F+ -(LV=S)*(-(DW=F)+ NPI=S)*-(OLPC=S)) + INIT=FLRB2 + INIT=FLRB3S + CILFAIL Comments CS fails due to NPSH if there is a containment breach (CIL=F)
CS3	(CSASUP*CSCSUP*(LV=S+DW=S*NPI=S+OLPC=S)) * (CSBSUP*CSDSUP*LV=S*(DW=S*NPII=S+OLPC=S)) * (EECW=S+EECWR=S)*TOR=S
CS4	(CSASUP*CSCSUP*LV=S*DW=S*NPI=S) * (CSBSUP*CSDSUP*LV=S*DW=S*NPII=S) * (EECW=S+EECWR=S)*TOR=S *OLPC=F
CS5	(CSASUP*CSBSUP*CSCSUP*-CSDSUP + CSASUP*CSBSUP*-CSCSUP*CSDSUP + CSASUP*-CSBSUP*CSCSUP*CSDSUP + -CSASUP*CSBSUP*CSCSUP*-CSDSUP) * (LV=S+DW=S*NPI=S*NPII=S + OLPC=S) Comments 3 pumps supported with OLPC=S
CS5A	(CSASUP*CSBSUP*CSCSUP*-CSDSUP + CSASUP*CSBSUP*-CSCSUP*CSDSUP + CSASUP*-CSBSUP*CSCSUP*CSDSUP + -CSASUP*CSBSUP*CSCSUP*-CSDSUP) * (LV=S+DW=S*NPI=S*NPII=S)*OLPC=F Comments 3 pumps supported with OLPC=S
CS6	CSASUP*CSCSUP*(LV=S+DW=S*NPI=S + OLPC=S) + CSBSUP*CSDSUP*(LV=S+DW=S*NPII=S + OLPC=S)

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SF	Split Fraction Assignment Rule
	Comments Conditions for all support already asked so LOOPI*-LOOPII + -LOOPI*LOOPII not necessary
CS7	CSASUP*CSCSUP*(LV=S+DW=S*NPI=S)*OLPC=F + CSBSUP*CSDSUP*(LV=S+DW=S*NPII=S)*OLPC=F
CS8	(LV=S + DW=S*NPI=S*NPII=S + OLPC=S) * (CSASUP + CSCSUP)*(CSBSUP + CSDSUP) Comments Support only for 2 pumps in different loops; heirarchy used - negatives not shown
CS8A	OLPC=F*(LV=S + DW=S*NPI=S*NPII=S) * (CSASUP + CSCSUP)*(CSBSUP + CSDSUP)
CS9	(LV=S + DW=S*NPI=S+ OLPC=S) * (CSASUP + CSCSUP) + (LV=S + DW=S*NPII=S + OLPC=S) * (CSBSUP + CSDSUP) Comments Support for 1 pump only; heirarchy used - negatives not shown
CS9A	OLPC=F*(LV=S + DW=S*NPI=S) * (CSASUP + CSCSUP) + OLPC=F*(LV=S + DW=S*NPII=S) * (CSBSUP + CSDSUP) Comments Support for 1 pump only; heirarchy used - negatives not shown
LPCF	CS=B + LV=F*DW=F*NPI=F*NPII=F*OLPC=F + -RERCVRY*-RFRCVRY + CILFAIL Comments LPC fails due to NPSH if there is a containment breach (CIL=F)
LPC1	RERCVRY*RFRCVRY*OLPC=S
LPC2	RERCVRY*RFRCVRY*-(OLPC=S)
LPC3	RFRCVRY*-RERCVRY*OLPC=S
LPC4	RFRCVRY*-RERCVRY*-(OLPC=S)
LPC5	RERCVRY*-RFRCVRY*OLPC=S
LPC6	RERCVRY*-RFRCVRY*-(OLPC=S)
LPCF	1
ORHXTS	HXA=S*HXB=S*HXC=S+ HXA=S*HXB=S*HXD=S + HXA=S*HXC=S*HXD=S + HXB=S*HXC=S*HXD=S
ORHXTS	RF=F+RH=F+EECW=F*EECWR=F+RF=F*RI=F+(AB=F+DC=F+SW1C=F*SW2C=F)*(AA=F+DA=F+SW1A=F*SW2A=F) + INIT=FLRB1 Comments PASS THROUGH IF SUPPORT FOR XTIE NOT AVAILABLE
ORHXT1	1
U2XF	RF=F+RH=F+EECW=F*EECWR=F+RF=F*RI=F+(AB=F+DC=F+-HXCSW)*(AA=F+DA=F+HXASW) + INIT=FLRB1 + CILFAIL Comments U2X crosstie fails due to NPSH if there is a containment breach (CIL=F)
U2X5	(AA=F+DA=F+ -HXASW)*RI=F
U2X6	(AB=F+DC=F+ -HXCSW)*RI=F
U2X3	(AA=F+DA=F+ -HXASW)*RI=S
U2X4	(AB=F+DC=F+ -HXCSW)*RI=S

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SF	Split Fraction Assignment Rule
U2X2	RI=F
U2X1	1
XTVS	HXA=S*HXB=S*HXC=S+ HXA=S*HXB=S*HXD=S + HXA=S*HXC=S*HXD=S + HXB=S*HXC=S*HXD=S
XTV1	RF=S
XTVF	1
PCSRF	INIT=LOSP + RVC=SORV1 + RVC=SORV2 + FWH=F*(RCI=F*HPI=F+OHPC=F*OHPR=F)*(ORVD=F + RVD=F) + CS=F*LPC=F
PCSR6	INIT=L500U+ INIT=L500PA
PCSR5	INIT=TBU
PCSR4	INIT=LOPA
PCSR3	INIT=IMSIV
PCSR2	INIT=TLFW + INIT=TLCF
PCSR1	INIT=FLRB3S
PCSRF	1
ODWS1	1
DWSF	PX1=F*PX2=F + ((RPA=F+HXA=F)*(RPC=F+HXC=F) +RE=F+NOGA)*((RPB=F+HXC=F)*(RPD=F+HXC=F) +RF=F+ NOGC)+ODWS=F + CILFAIL Comments DWS fails due to NPSH if there is a containment breach (CIL=F)
DWS1	PX1=S*PX2=S * (RPA=S*HXA=S + RPC=S*HXC=S)*RERCVRY*GA=S*(RPB=S*HXB=S+RPD=S*HXD=S)*RFRCVRY*GC=S*ODWS=S
DWS2	(PX1=S*(RPA=S*HXA=S + RPC=S*HXC=S)*RERCVRY*GA=S + PX2=S*(RPB=S*HXB=S+RPD=S*HXD=S)*RFRCVRY*GC=S)*ODWS=S
DWSF	1
SPF	OSPC=F + CILFAIL Comments SP fails due to NPSH if there is a containment breach (CIL=F)
SP1	-(LPC=S)*RERCVRY*RFRCVRY*(RPA=S*HXA=S + RPC=S*HXC=S)*(RPB=S*HXB=S + RPD=S*HXD=S + XTV=S*-MULTIT) Comments Only RMOV 1A and 1B boards are needed
SP2	-(LPC=S)*(-(RERCVRY*(RPA=S*HXA=S + RPC=S*HXC=S))*RFRCVRY*(RPB=S*HXB=S + RPD=S*HXD=S +XTV=S*-MULTIT) + RERCVRY*(RPA=S*HXA=S + RPC=S*HXC=S)*-(RFRCVRY*(RPB=S*HXB=S + RPD=S*HXD=S + XTV=S*-MULTIT)))
SP3	LPC=S*RERCVRY*RFRCVRY*RCOK*RBOK*(RPA=S*HXA=S + RPC=S*HXC=S)*(RPB=S*HXB=S + RPD=S*HXD=S + XTV=S*-MULTIT)

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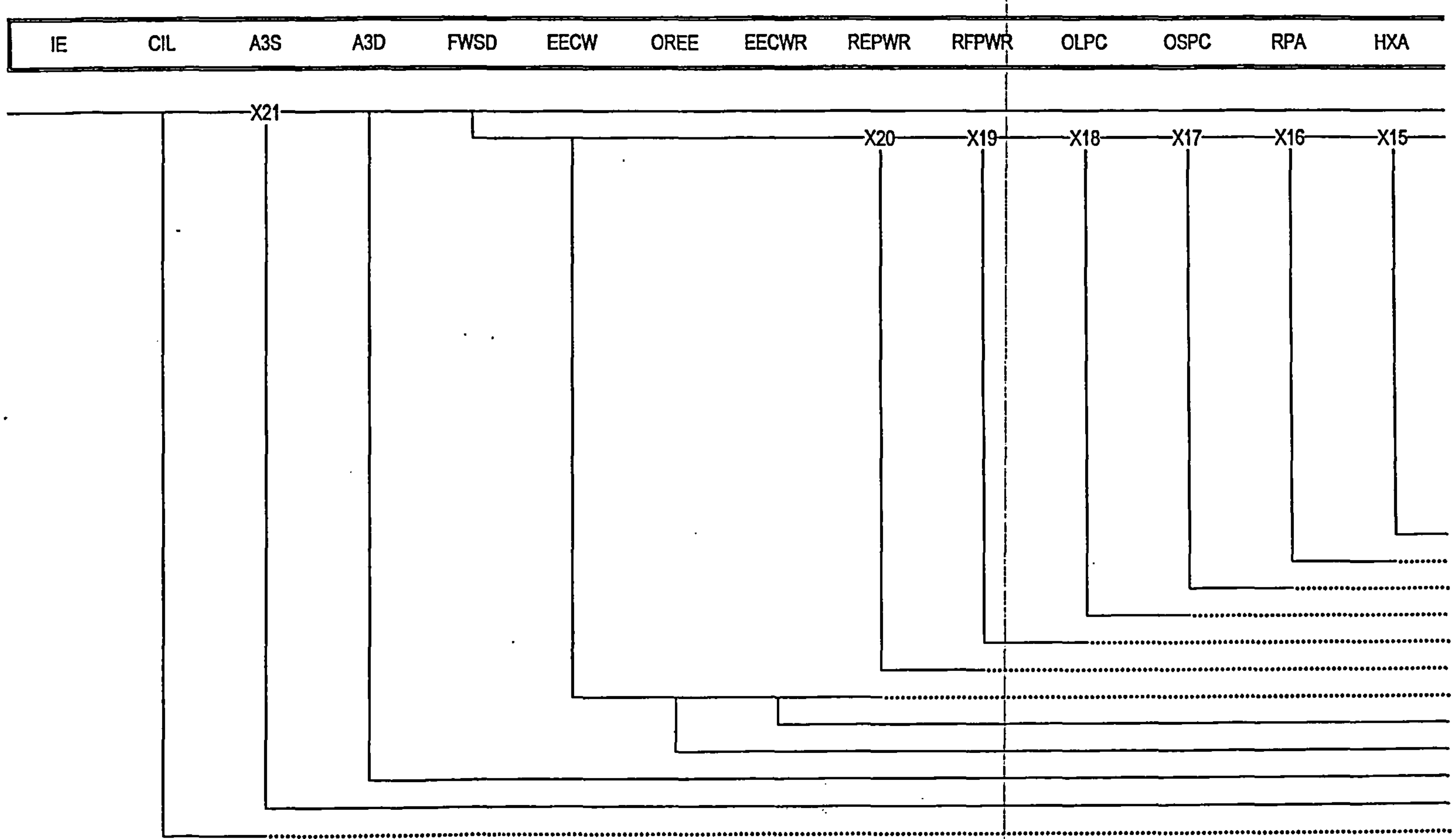
SF	Split Fraction Assignment Rule
SP4	LPC=S*(-(RERCVRY*RCOK*(RPA=S*HXA=S + RPC=S*HXC=S))*RFRCVRY*RBOK*(RPB=S*HXB=S + RPD=S*HXD=S + XTV=S*-MULTIT) + RERCVRY*RCOK*(-(RFRCVRY*RBOK*(RPB=S*HXB=S + RPD=S*HXD=S + XTV=S*-MULTIT))*(RPA=S*HXA=S + RPC=S*HXC=S)))
SP5	RBOK*RCOK*RERCVRY*RFRCVRY*(RPA=S*HXA=S + RPC=S*HXC=S)*(RPB=S*HXB=S + RPD=S*HXD=S + XTV=S*-MULTIT)
SPF	1
OSPRF	(RPA=F+HXA=F)*(RPB=F+HXB=F)*(RPC=F+HXC=F)*(RPD=F+HXC=F) + OSPC=F Comments Support for SPR not available
OSPR1	1
SPRF	(RPA=F+HXA=F)*(RPB=F+HXB=F)*(RPC=F+HXC=F)*(RPD=F+HXC=F) + OSPC=F + CILFAIL Comments SPR fails due to NPSH if there is a containment breach (CIL=F)
SPR1	RPSM=S Comments This has been simplified and is slightly conservative
SPRF	1
OWWVS	(HXASW+HXBSW+HXCSW+HXDSW)*(SP=S+ SPR=S) + OG5=F*EPR6=S
OWWV1	1
VNTNN	(HXASW+HXBSW+HXCSW+HXDSW)*(SP=S+ SPR=S) + OG5=F*EPR6=S
VNTF	OWWV=F
VNT1	RBOK*RCOK*PCA=S
VNT2	OWWV=S
VNTF	1
CRD2	(CST=S + RCW=S)*UB41C=S + OG5=F*EPR6=S*CST=S
CRD3	(CST=S + RCW=S)*UB41C=F*AA=S + OG5=F*EPR6=S*CST=S
CRDF	1
OAIF	1
AVIF	SW1D=F+SW2D=F+RF=F
AVI1	1

MODEL Name:
Event Tree: ATWS3N.ETI

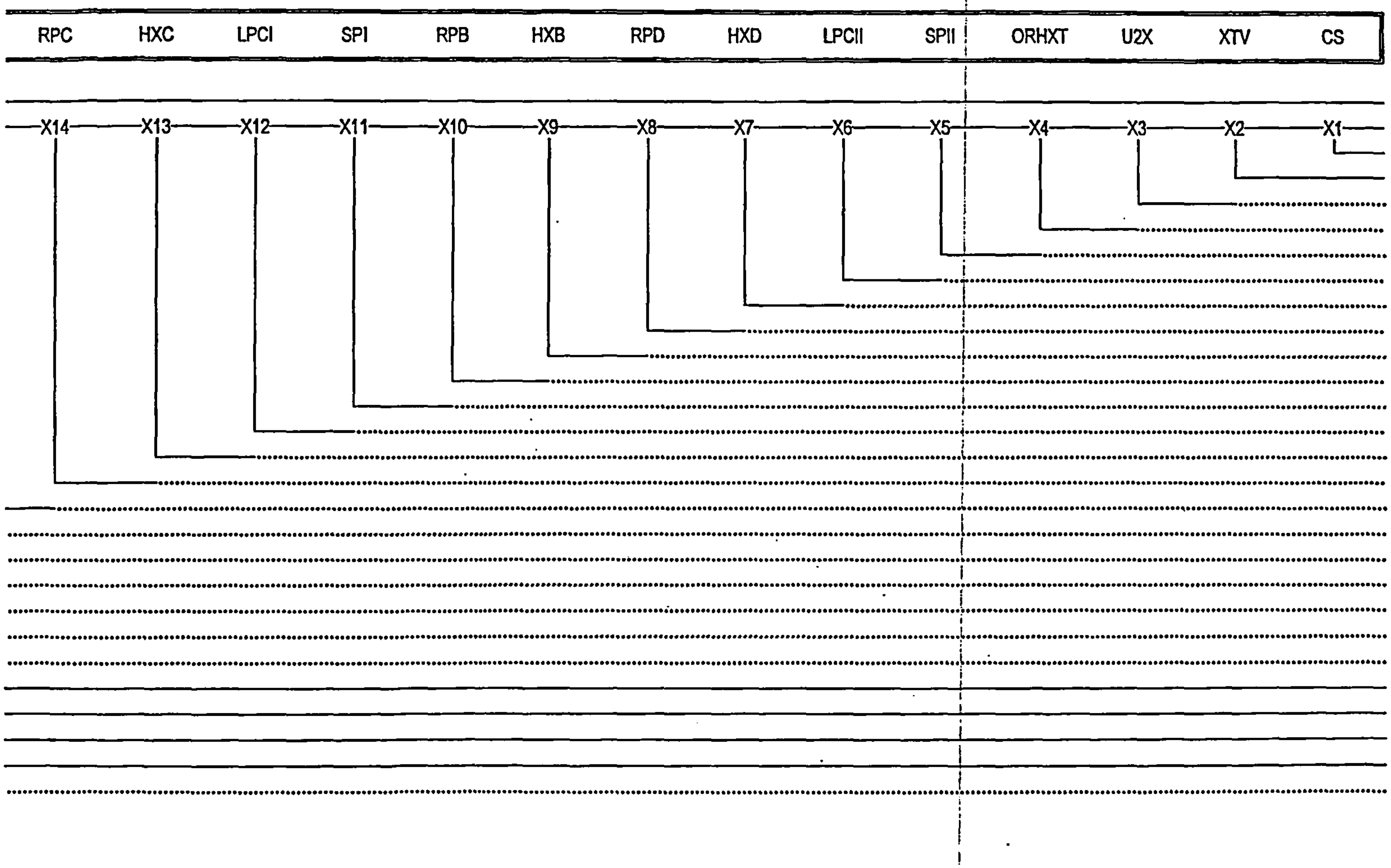
11:42:21 June 27, 2006



MODEL Name:

Event Tree: ATWS3N.ETI

11:42:21 June 27, 2006



MODEL Name:

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X#	B#	S#
	1	1
	2	2
	3	3
	4	4
X2	5	5-7
X3	6	8-13
X4	7	14-25
X5	8	26-49
X6	9	50-97
X7	10	98-193
X8	11	194-385
X9	12	386-769
X10	13	770-1537
X11	14	1538-3073
X12	15	3074-6145
X13	16	6146-12289
X14	17	12290-24577
X15	18	24578-49153
X16	19	49154-98305
X17	20	98306-196609
X18	21	196610-393217
X19	22	393218-786433
X20	23	786434-1572865
	24	1572866
	25	1572867
	26	1572868
	27	1572869
X21	28	1572870-3145738

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SF	Split Fraction Assignment Rule
A3SY	A2S=S*(OSV=S*MCD=S*CND=S*FWH=S)*TOR=S*OAL=S*OTAF=S*OHPC=S + A2S=F Comments RECALL SUCCESS IS DOWN BRANCH!
A3SN	1 Comments SUCCESS CANNOT BE DETERMINED IN ATWS2
A3DY	A2D=F + HPI=F*(OSV=F + MCD=F +CND=F + FWH=F + L8F=F) + TOR=F + OAL=F + OTAF=F + OHPC=F Comments UNCHANGED FROM ATWS2
A3DN	1
FWSDF	INIT=LOCHSA
FWSDS	(RVC=SORV0+RVC=SORV1)*MCD=S*CND=S*FWH=S
FWSDF	1
EECWF	EA=F*EB=F*EC=F+EA=F*EB=F*ED=F+EA=F*EC=F*ED=F+EB=F*EC=F*ED=F
EECWS	1
OREENN	GA=F*GB=F*GC=F*GD=F*GE=F*GF=F*GG=F*GH=F*-SWING1C*-SWING1D
OREE2	SWING1C*SWING1D
OREE1	SWING1C+SWING1D
EECWRS	SWING1C*SWING1D*SW1C=S*SW1D=S + SWING1C*-SWING1D*SW1C=S + -SWING1C*SWING1D*SW1D=S
EECWRF	1
REPWRS	RQOK Comments 480 V RMOV BOARD 1A IS RECEIVING PWR FROM 480 V SD BD 1A
REPWRF	-RQOK Comments 480 V RMOV BOARD 1A IS NOT RECEIVING PWR FROM 480 V SD BD 1A
RFPWRS	RROK Comments 480 V RMOV BOARD 1A IS RECEIVING PWR FROM 480 V SD BD 1B
RFPWRF	-RROK Comments 480 V RMOV BOARD 1A IS NOT RECEIVING PWR FROM 480 V SD BD 1B
OLPCNN	HPI=S + RCI=S
OLPC1	1 Comments OPERATOR INITIATES LPC (STARTS PUMPS) HUGE QUESTION HERE ON PWER WXCURSION
OSPC2	1
RPAF	-RPASUP+ INIT=FLRB2 + INIT=FLRB3S + CILFAIL

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SF	Split Fraction Assignment Rule
	Comments RPA fails due to NPSH if there is a containment breach (CIL=F)
RPA1	RPASUP
RPAF	1
HXAF	-HXASUP
HXA1	HXASUP
~~HXAF~~	~~1~~
RPCF	-RPCSUP + INIT=FLRB2 + INIT=FLRB3S + CILFAIL Comments RPC fails due to NPSH if there is a containment breach (CIL=F)
RPC1	RPCSUP*RPA=S
RPC3	RPCSUP*-RPASUP
RPC2	RPCSUP*RPASUP*RPA=F
RPCF	1
HXC1	HXA=S
HXC2	HXA=F*HXASUP
HXC3	HXA=F*-HXASUP
HXCF	1
SPIF	OSPC=F + REPWR=F
SPI1	NOLPCI*REPWR=S*(RPA=S*HXA=S + RPC=S*HXC=S) Comments Only RMOV 1A and 1B boards are needed
SPI2	-NOLPCI*REPWR=S*RC=S*(RPA=S*HXA=S + RPC=S*HXC=S)
SPIF	1
LPCIF	-LPCISUP + CILFAIL Comments LPCI fails due to NPSH if there is a containment breach (CIL=F)
LPCI2	LPCISUP Comments MANUAL LPCI START NOT CREDITED LLOCAS; ODD SPLIT FRACTION SWOULD APPLY
LPCIIF	-LPCIISUP + CILFAIL Comments LPCII fails due to NPSH if there is a containment breach (CIL=F)
LPCII2	LPCI=S
LPCII4	-LPCISUP
LPCII6	LPCI=F*LPCISUP

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SF	Split Fraction Assignment Rule
RPBF	-RPBSUP+ INIT=FLRB2 + INIT=FLRB3S + CILFAIL Comments RPB fails due to NPSH if there is a containment breach (CIL=F)
RPB6	RPBSUP*(-RPASUP*RPC=F*RPCSUP+RPA=F*RPCSUP*(RPC=B+-RPCSUP))
RPB5	RPBSUP*(-RPASUP*RPC=S+RPA=S*(RPC=B+-RPCSUP))
RPB4	RPBSUP*-RPASUP*(RPC=B+-(RPCSUP))
~~RPB3~~	~~RPBSUP*RF=S*RB=S*RPA=F*RPC=F*SIGII~~
RPB2	RPBSUP*(RPA=S*RPC=F+RPA=F*RPC=S)
RPB1	RPBSUP*RPA=S*RPC=S
RPBF	1
HXBF	-HXBSUP
HXB1	HXA=S*HXC=S
HXB6	-HXASUP*-HXCSUP
HXB5	HXASUP*HXA=F*HXCSUP*HXC=F
HXB4	HXA=F*-HXASUP*HXC=F*HXCSUP + HXA=F*HXASUP*HXC=F*-HXCSUP
HXB3	HXA=F*-HXASUP*HXC=S + HXA=S*HXC=F*-HXCSUP
HXB2	HXA=F*HXASUP*HXB=S + HXA=S*HXB=F*HXBSUP
HXBF	1
RPDF	-RPDSUP+ INIT=FLRB2 + INIT=FLRB3S + CILFAIL Comments RPD fails due to NPSH if there is a containment breach (CIL=F)
RPD10	RPDSUP*(RPA=F*(RPC=F*-RPBSUP+(RPC=B+-RPCSUP)*RPB=F)+-RPASUP*RPC=F*RPB=F)
RPD9	RPDSUP*(RPA=S*(RPC=F*RPCSUP*-RPBSUP+(RPC=B+-RPCSUP)*RPB=F*RPBSUP)+RPC=S*(RPA=F*RPASUP*-RPBSUP+-RPASUP*RPB=F*RPBSUP)+RPB=S*(RPA=F*RPASUP*(RPC=B+-(RPCSUP)+-(RPASUP)*RPC=F*RPCSUP)))
RPD8	RPDSUP*(RPA=S*(RPC=S*-(RPBSUP)+(RPC=B+-(RPCSUP)*RPB=S))+-(RPASUP)*RPC=S*RPB=S)
RPD7	RPDSUP*(-(RPASUP)*(RPC=F*RPCSUP*-(RPBSUP)+(RPC=B+-(RPCSUP))*RPB=F*RPBSUP)+RPA=F*RPASUP*(RPC=B+-(RPCSUP))*-(RPBSUP))
RPD6	RPDSUP*(-(RPASUP)*(RPC=S*-(RPBSUP)+(RPC=B+-(RPCSUP))*RPB=S)+RPA=S*(RPC=B+-(RPCSUP))*-(RPBSUP))
RPD5	RPDSUP*-(RPASUP)*(RPC=B+-(RPCSUP))*-(RPBSUP)
RPD4	RPDSUP*RPA=F*RPASUP*RPC=F*RPCSUP*RPB=F*RPBSUP

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SF	Split Fraction Assignment Rule
RPD3	RPDSUP*(RPA=F*RPASUP*(RPC=F*RPCSUP*RPB=S+RPC=S*RPB=F*RPBSUP)+RPA=S*RPC=F*RPCSUP*RPB=F*RPBSUP)
RPD2	RPDSUP*(RPA=F*RPASUP*RPC=S*RPB=S+RPA=S*RPC=F*RPCSUP*RPB=S+RPA=S*RPC=S*RPB=F*RPBSUP)
RPD1	RPDSUP*RPA=S*RPC=S*RPB=S
RPDF	1
HXDF	-HXDSUP
HXD10	-HXASUP*-HXCSUP*-HXBSUP
HXD9	-HXASUP*-HXCSUP*HXBSUP*HXB=F + -HXASUP*HXCSUP*HXC=F*-HXBSUP + HXASUP*HXA=F*-HXCSUP*-HXBSUP
HXD8	-HXASUP*-HXCSUP*HXB=S + -HXASUP*HXC=S*-HXBSUP + HXA=S*-HXCSUP*-HXBSUP
HXD7	HXASUP*HXCSUP*HXBSUP*HXA=F*HXC=F*HXB=F
HXD6	HXASUP*HXA=F*HXCSUP*HXC=F*-HXBSUP + HXASUP*HXA=F*-HXCSUP*HXBSUP*HXB=F + -HXASUP*HXCSUP*HXC=F*HXBSUP*HXB=F
HXD5	HXASUP*HXA=F*HXCSUP*HXC=F*HXB=S+ HXASUP*HXA=F*HXC=S*HXBSUP*HXB=F + HXA=S*HXCSUP*HXC=F*HXBSUP*HXB=F
HXD4	-HXASUP*HXCSUP*HXC=F*HXB=S + HXASUP*HXA=F*HXC=S*-HXBSUP + HXA=S*HXCSUP*HXC=F*-HXBSUP
HXD3	HXASUP*HXA=F*HXC=S*HXB=S + HXA=S*HXC=S*HXBSUP*HXB=F + HXA=S*HXCSUP*HXC=F*HXB=S
HXD2	-HXASUP*HXC=S*HXB=S + HXA=S*HXC=S*-HXBSUP + HXA=S*-HXCSUP*HXB=S
HXD1	HXA=S*HXC=S*HXB=S
HXDF	1
SPIIF	OSPC=F + RFPWR=F + CILFAIL *Comments* · *SPII fails due to NPSH if there is a containment breach (CIL=F)*
SPII1	NOLPCI*RFPWR=S*(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=S
SPII2	NOLPCI*RFPWR=S*(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*REPWR=S
SPII3	NOLPCI*RFPWR=S*(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*REPWR=F
SPII4	-NOLPCI*RFPWR=S*(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=S
SPII5	-NOLPCI*RFPWR=S*(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*REPWR=S
SPII6	-NOLPCI*RFPWR=S*(RPB=S*HXB=S + RPD=S*HXD=S)*SPI=F*REPWR=F
SPIIF	1
LPCIIF	-LPCIISUP

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SF	Split Fraction Assignment Rule
LPCII2	LPCI=S
LPCII4	-LPCISUP
LPCII6	LPCI=F*LPCISUP
ORHXTS	HXA=S*HXB=S*HXC=S+ HXA=S*HXB=S*HXD=S + HXA=S*HXC=S*HXD=S + HXB=S*HXC=S*HXD=S
ORHXTS	RF=F+RH=F+EECW=F*EECWR=F+RF=F*RI=F+(AB=F+DC=F+SW1C=F*SW2C=F)*(AA=F+DA=F+SW1A =F*SW2A=F) + INIT=FLRB1 Comments PASS THROUGH IF SUPPORT FOR XTIE NOT AVAILABLE
ORHXT1	1
U2XF	RF=F+RH=F+EECW=F*EECWR=F+RF=F*RI=F+(AB=F+DC=F+-HXCSW)*(AA=F+DA=F+HXASW) + INIT=FLRB1 + CILFAIL Comments U2X fails due to NPSH if there is a containment breach (CIL=F)
U2X5	(AA=F+DA=F+ -HXASW)*RI=F
U2X6	(AB=F+DC=F+ -HXCSW)*RI=F
U2X3	(AA=F+DA=F+ -HXASW)*RI=S
U2X4	(AB=F+DC=F+ -HXCSW)*RI=S
U2X2	RI=F
U2X1	1
XTVS	HXA=S*HXB=S*HXC=S+ HXA=S*HXB=S*HXD=S + HXA=S*HXC=S*HXD=S + HXB=S*HXC=S*HXD=S
XTV1	RF=S
XTVF	1
CSF	(-CSASUP*-CSCSUP + EECW=F*EECWR=F +TOR=F + -(LV=S)*(-(DW=S)+-(NPI=S))*-(OLPC=S)) * (-CSBSUP*-CSDSUP+(EECW=S+EECWR=S) +TOR=F+ -(LV=S)*(-(DW=F)+ NPI=S)*-(OLPC=S)) + INIT=FLRB2 + INIT=FLRB3S + CILFAIL Comments CS fails due to NPSH if there is a containment breach (CIL=F)
CS3	(CSASUP*CSCSUP*(LV=S+DW=S*NPI=S+OLPC=S)) * (CSBSUP*CSDSUP*LV=S*(DW=S*NPII=S+OLPC=S)) * (EECW=S+EECWR=S)*TOR=S
CS4	(CSASUP*CSCSUP*LV=S*DW=S*NPI=S) * (CSBSUP*CSDSUP*LV=S*DW=S*NPII=S) * (EECW=S+EECWR=S)*TOR=S *OLPC=F
CS5	(CSASUP*CSBSUP*CSCSUP*-CSDSUP + CSASUP*CSBSUP*-CSCSUP*CSDSUP + CSASUP*-CSBSUP*CSCSUP*CSDSUP + -CSASUP*CSBSUP*CSCSUP*-CSDSUP) * (LV=S+DW=S*NPI=S*NPII=S + OLPC=S) Comments 3 pumps supported with OLPC=S
CS5A	(CSASUP*CSBSUP*CSCSUP*-CSDSUP + CSASUP*CSBSUP*-CSCSUP*CSDSUP + CSASUP*-CSBSUP*CSCSUP*CSDSUP + -CSASUP*CSBSUP*CSCSUP*-CSDSUP) * (LV=S+DW=S*NPI=S*NPII=S)*OLPC=F

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: ATWS3N

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SF	Split Fraction Assignment Rule
	Comments 3 pumps supported with OLPC=S
CS6	CSASUP*CSCSUP*(LV=S+DW=S*NPI=S + OLPC=S) + CSBSUP*CSDSUP*(LV=S+DW=S*NPII=S + OLPC=S) Comments Conditions for all support already asked so LOOPI*-LOOPII + -LOOPI*LOOPII not necessary
CS7	CSASUP*CSCSUP*(LV=S+DW=S*NPI=S)*OLPC=F + CSBSUP*CSDSUP*(LV=S+DW=S*NPII=S)*OLPC=F
CS8	(LV=S + DW=S*NPI=S*NPII=S + OLPC=S) * (CSASUP + CSCSUP)*(CSBSUP + CSDSUP) Comments Support only for 2 pumps in different loops; heirarchy used - negatives not shown
CS8A	OLPC=F*(LV=S + DW=S*NPI=S*NPII=S) * (CSASUP + CSCSUP)*(CSBSUP + CSDSUP)
CS9	(LV=S + DW=S*NPI=S+ OLPC=S) * (CSASUP + CSCSUP) + (LV=S + DW=S*NPII=S + OLPC=S) * (CSBSUP + CSDSUP) Comments Support for 1 pump only; heirarchy used - negatives not shown
CS9A	OLPC=F*(LV=S + DW=S*NPI=S) * (CSASUP + CSCSUP) + OLPC=F*(LV=S + DW=S*NPII=S) * (CSBSUP + CSDSUP) Comments Support for 1 pump only; heirarchy used - negatives not shown

Model Name: U1COP2-9
Macro for Event Tree: ATWS3N

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Macro	Macro Rule / Comments
CILFAIL	CIL=F
CSASUP	AAOK*DA=S*REPWR=S*(EECW=S+EECWR=S) Core Spray pump A support (sans actuation)
CSBSUP	ABOK*DC=S*RFPWR=S*(EECW=S+EECWR=S) Core Spray pump B support (sans actuation)
CSCSUP	ACOK*DB=S*REPWR=S*(EECW=S+EECWR=S) Core Spray pump C support (sans actuation)
CSDSUP	ADOK*DD=S*RFPWR=S*(EECW=S+EECWR=S) Core Spray pump B support (sans actuation)
HRLPT	RVC=SORV1 + RVC=SORV2 + RVD=F + INIT=SLOCA HPCI/RCIC LOW PRESSURE TRIP;
HWFHXA	HXA=F*HXASUP
HWFHXB	HXB=F*HXBSUP
HWFHXC	HXC=F*HXCSUP
HWFHXD	HXD=F*HXDSUP (sans actuation)
HWFRPA	RPA=F*RPASUP
HWFRPB	RPB=F*RPBSUP
HWFRPC	RPC=F*RPCSUP
HWFRPD	RPD=F*RPDSUP
HXAB	RH=F+SW2A=F*SW1A=F+NOGB+HXA=B+RPA=F
HXASUP	REPWR=S*HXASW*RPA=S
HXBB	RI=F+SW2B=F*SW1B=F+NOGD+HXB=B+RPB=F
HXBSUP	RFPWR=S*HXBSW*RPB=S

Model Name: U1COP2-9
Macro for Event Tree: ATWS3N

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Macro	Macro Rule / Comments
HXCB	RH=F+SW2C=F*SW1C=F+NOGB+HXC=B+RPC=F
HXCSUP	REPWR=S*HXCSW*RPC=S
HXDSUP	RFPWR=S*HXDSW*RPD=S
LOOPIIRHR	RHR2+RHR4+INIT=LLDA*CS=F
LOOPIRHR	RHR1+RHR3+INIT=LLDB*CS=F
LPCIISUP	RF=S*((NPII=S*DW=S) + LV=S)
LPCISUP	RE=S*((NPI=S*DW=S) + LV=S) LOOP I LPCI SUPPORT
NOGA	GA=S*-(EECW=S+EECWR=S)
NOGB	GB=S*-(EECW=S+EECWR=S)
NOGC	GC=S*-(EECW=S+EECWR=S)
NOGD	GD=S*-(EECW=S+EECWR=S)
NOGE	GE=S*-(EECW=S+EECWR=S)
NOGF	GF=S*-(EECW=S+EECWR=S)
NOGG	GG=S*-(EECW=S+EECWR=S)
NOGH	GH=S*-(EECW=S+EECWR=S)
NOLPCI	MCD=F*(RVC=SORV0+RVC=SORV1)*HPI=S
RHOK	(ABOK+AC=S+ADOK)*INIT=LOSP+RH=S+EPR6=S*OG5=F
RHRSW1	SW2B=S+SW1B=S+SW2D=S+SW1D=S

Model Name: U1COP2-9
Macro for Event Tree: ATWS3N

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Macro	Macro Rule / Comments
RIOK	(ABOK+AC=S+ADOK)*INIT=LOSP+RI=S+EPR6=S*OG5=F
RPASUP	AAOK*DA=S*(RCW=S+EECW=S+EECWR=S)*REPWR=S*(SIGI*RCOK+OLPC=S + OSPC=S)
RPBSUP	ACOK*DB=S*(RCW=S+EECW=S+EECWR=S)*RFPWR=S*(SIGII*RBOK+OLPC=S+ OSPC=S)
RPCSUP	ABOK*DC=S*(RCW=S+EECW=S+EECWR=S)*REPWR=S*(SIGI*(RBOK+RCOK) +OLPC=S+OSPC=S)
RPDSUP	ADOK*DD=S*(RCW=S+EECW=S+EECWR=S)*RFPWR=S*(SIGII*(RBOK+RCOK) +OLPC=S+OSPC=S)
SIG3	(LV=S+DW=S)
SWING1C	SW1C=S*(EA=F*EB=F*EC=F*ED=F + EA=F*EB=F*EC=F*ED=S + EA=F*EB=S*EC=F*ED=F + EA=S*EB=F*EC=F*ED=F)
SWING1D	SW1D=S*(EA=F*EB=F*EC=F*ED=F + EA=F*EB=F*EC=S*ED=F + EA=F*EB=S*EC=F*ED=F + EA=S*EB=F*EC=F*ED=F)

ATWS4N Event Tree

- No ATWS4N event tree structure included here, as no tree structure modifications were made for this analysis (i.e., tree structure same as base BFN PRA ATWS4 event tree).
- No ATWS4N macros print-out provided here as no new macros were defined for this tree for this analysis.

Model Name: U1COP2-9
Split Fraction Assignment Rule for Event Tree: ATWS4N

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SF	Split Fraction Assignment Rule
A4SY	A3S=F + (LPCI=S + LPCII=S)*(SPI=S + SPII=S + XTV=S) Comments RECALL SUCCESS IS DOWN BRANCH!
A4SN	1
A4DY	A3D=F
A4DN	1
OWWV1	1
VNT1	RBOK*RCOK*PCA=S
VNT2	OWWV=S
VNTF	1
CRD2	(CST=S + RCW=S)*UB41C=S + OG5=F*EPR6=S*CST=S
CRD3	(CST=S + RCW=S)*UB41C=F*AA=S + OG5=F*EPR6=S*CST=S
CRDF	1
ODWS1	1
DWSF	PX1=F*PX2=F + ((RPA=F+HXA=F)*(RPC=F+HXC=F) +RE=F+NOGA)*((RPB=F+HXC=F)*(RPD=F+HXC=F) +RF=F+ NOGC)+ODWS=F + CILFAIL Comments DWS fails due to NPSH if there is a containment breach (CIL=F)
DWS1	PX1=S*PX2=S * (RPA=S*HXA=S + RPC=S*HXC=S)*REPWR=S*GA=S*(RPB=S*HXB=S+RPD=S*HXD=S)*RFPWR=S*GC=S*ODWS=S
DWS2	(PX1=S*(RPA=S*HXA=S + RPC=S*HXC=S)*REPWR=S*GA=S + PX2=S*(RPB=S*HXB=S+RPD=S*HXD=S)*RFPWR=S*GC=S)*ODWS=S
DWSF	1

Appendix F
REVISED FAULT TREES

This appendix provides print-outs of the BFN Unit 1 PRA modified containment isolation (CIL) fault tree and the NPSH fault tree used in this analysis. These print-outs are provided at the end of this appendix.

F.1 FAULT TREE REVISIONS

The following two BFN Unit 1 PRA RISKMAN fault tree models were revised for this risk assessment:

- Containment Isolation Failure (CIL)
- Conditions Preventing ECCS NPSH for LLOCA Cases (NPSH)

F.1.1 CIL Fault Tree Revisions

The BFN Unit 1 PRA existing CIL (Containment Isolation Failure) fault tree was modified to add the probability of a pre-existing containment leak; a basic event (CONDPRE) was inserted just under the top 'OR' gate of the CIL fault tree. The remainder of the CIL event tree models containment isolation system failure on demand given an accident.

The CONDPRE basic event probability is based on a 20La leak rate (refer to Table B-1) for the base case quantification. This event is modified for use in different sensitivity studies.

The containment isolation failure portion of the CIL fault tree is not modified in this risk analysis. Note that one of the quantification sensitivity studies investigates the risk impact if more containment penetrations are explicitly analyzed. However, this

sensitivity was addressed by modifying the CONDPRE basic event probability to mimic the impact (refer to Table F-1).

The value of the CONDPRE basic event and associated CIL top event frequency for each quantification case is summarized in Table F-1.

F.1.2 NPSH Fault Tree Revisions

The NPSH (Conditions Preventing ECCS NPSH for LLOCA Cases) fault tree was created for this risk assessment. The NPSH fault tree models the other (i.e., in addition to containment isolation failure modeled by the CIL fault tree) plant conditions that are necessary in order to require COP credit for LLOCA scenarios.

The NPSH fault tree is an "OR" gate structure that models the two Plant States used in this analysis (refer to Sections 3.1 and 3.2). One side of the NPSH fault tree models the probability of plant conditions when the plant is assumed to be at the DBA assumed power level of 102% EPU reactor power. The other side of the NSPH fault tree models the probability of plant conditions when the plant is assumed to be the nominal 100% reactor power level.

The probability that the plant is at 102% power is modeled using a miscalibration human error probability basic event (ZHECCL) taken from a similar action documented in the existing BFN Unit 1 PRA Human Reliability Analysis for Control Room instrument calibration error.

The NPSH fault tree also includes the following basic events that model the likelihood of exceeding specific river water and suppression pool water temperatures:

- "Exceedance Prob for River Water >68F" (RIVER68)
- "Given RW>68F, Cond Prob SP Water >87F" (CPSP87RW68)

- "Exceedance Prob for River Water >85F" (RIVER85)
- "Given RW>85F, Cond Prob SP Water >86F" (CPSP86RW85)

The NPSH fault tree also includes a basic event (SPLVL123K) that models the probability that the suppression pool water level is at or below 123,500 ft3 at the start of the accident.

The probabilities of the above temperature and level basic events are based on analysis of BFN plant data (refer to Appendix C).

The values of the NPSH fault tree basic events and associated NSPH top event probability for each quantification case are summarized in Table F-2.

Table F-1

CIL FAULT TREE RESULTS FOR EACH QUANTIFICATION CASE

Quantification Case	CONDPRE Basic Event		CIL Split Fraction Probability(1)
	Probability	Leak Size	
Base	1.88E-03	20La	2.25E-03
1	2.47E-04	100La	6.22E-04
2	Same as Base	Same as Base	2.25E-03
3	5.217E-03(2)	Same as Base	5.59E-03
4	Same as Base	Same as Base	2.25E-03
5	5.217E-03(2)	Same as Base	5.59E-03

Notes to Table F-1:

(1) "All Support Systems Available" split fraction. "Degraded Support State" split fraction is also affected but is not shown here.

(2) In these sensitivity cases the pre-existing containment leak rate is maintained at the base value of 20La, but the sensitivity issue of increasing the detail of the containment isolation system failure modeling to include smaller lines is addressed here by increasing the CONDPRE basic event probability. This surrogate approach is taken for simplicity. Rather than re-designing the containment isolation system fault tree logic, the probability of the containment isolation system portion of the tree (3.71E-4) is increased by a factor of 10x and the CONDPRE basic event value is modified and used as a surrogate to result in the new top event probability.

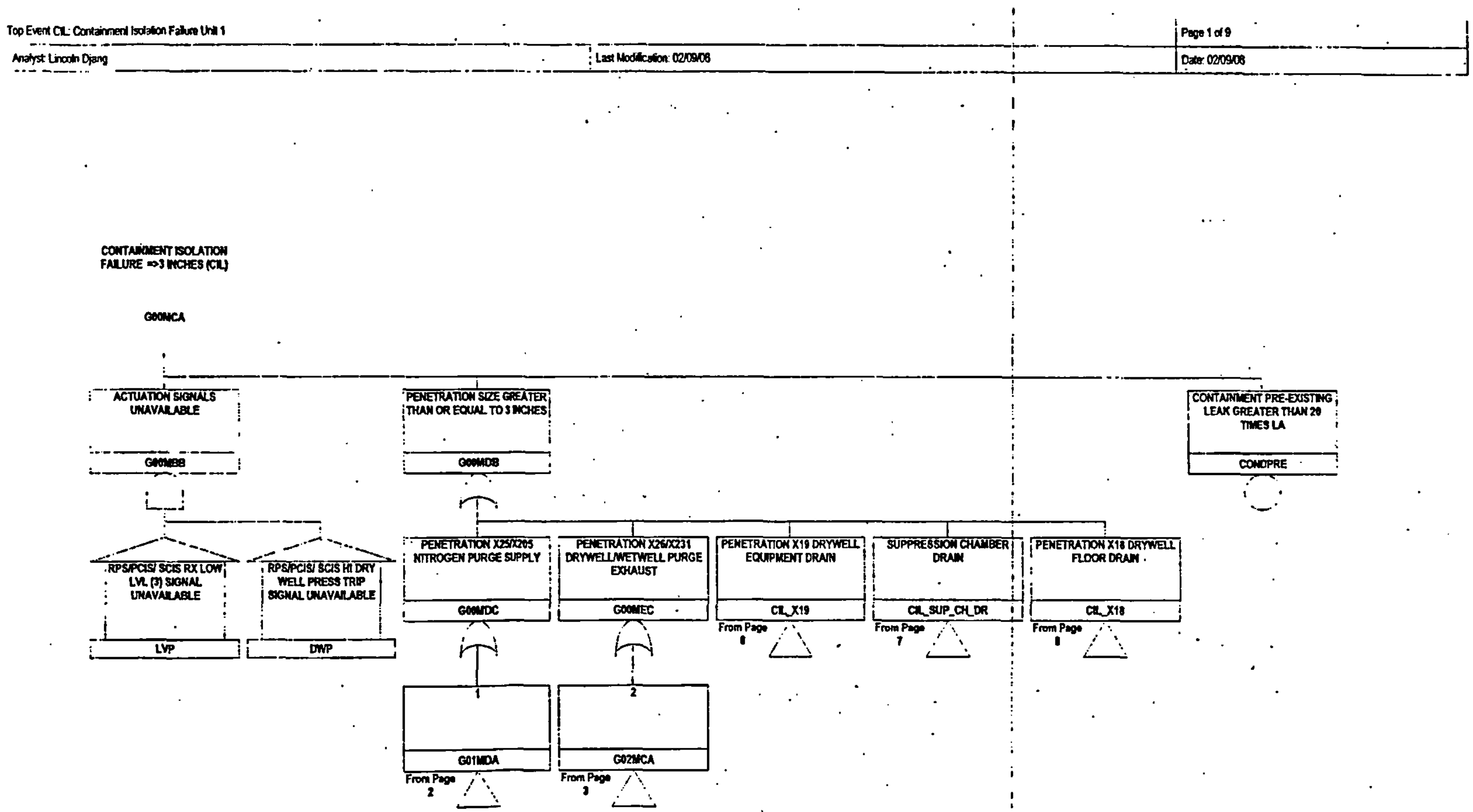
Table F-2

NPSH FAULT TREE RESULTS FOR EACH QUANTIFICATION CASE

Quantification Case	Basic Event Probabilities						NPSH Split Fraction Probability
	ZHECCL	RIVER68	CPSP87RW68	SPLVL123K	RIVER85	CPSP86RW85	
Base	5.00E-03	5.64E-01	4.42E-01	1.45E-02	1.64E-01	1.00	2.38E-03
1	Same as Base	Same as Base	Same as Base	Same as Base	Same as Base	Same as Base	2.38E-03
2	Same as Base	Same as Base	Same as Base	1.00	Same as Base	Same as Base	1.64E-01
3	Same as Base	Same as Base	Same as Base	Same as Base	Same as Base	Same as Base	2.38E-03
4	n/a(1)	n/a(1)	n/a(1)	n/a(1)	n/a(1)	n/a(1)	1.0E+00
5	n/a(1)	n/a(1)	n/a(1)	n/a(1)	n/a(1)	n/a(1)	1.0E+00

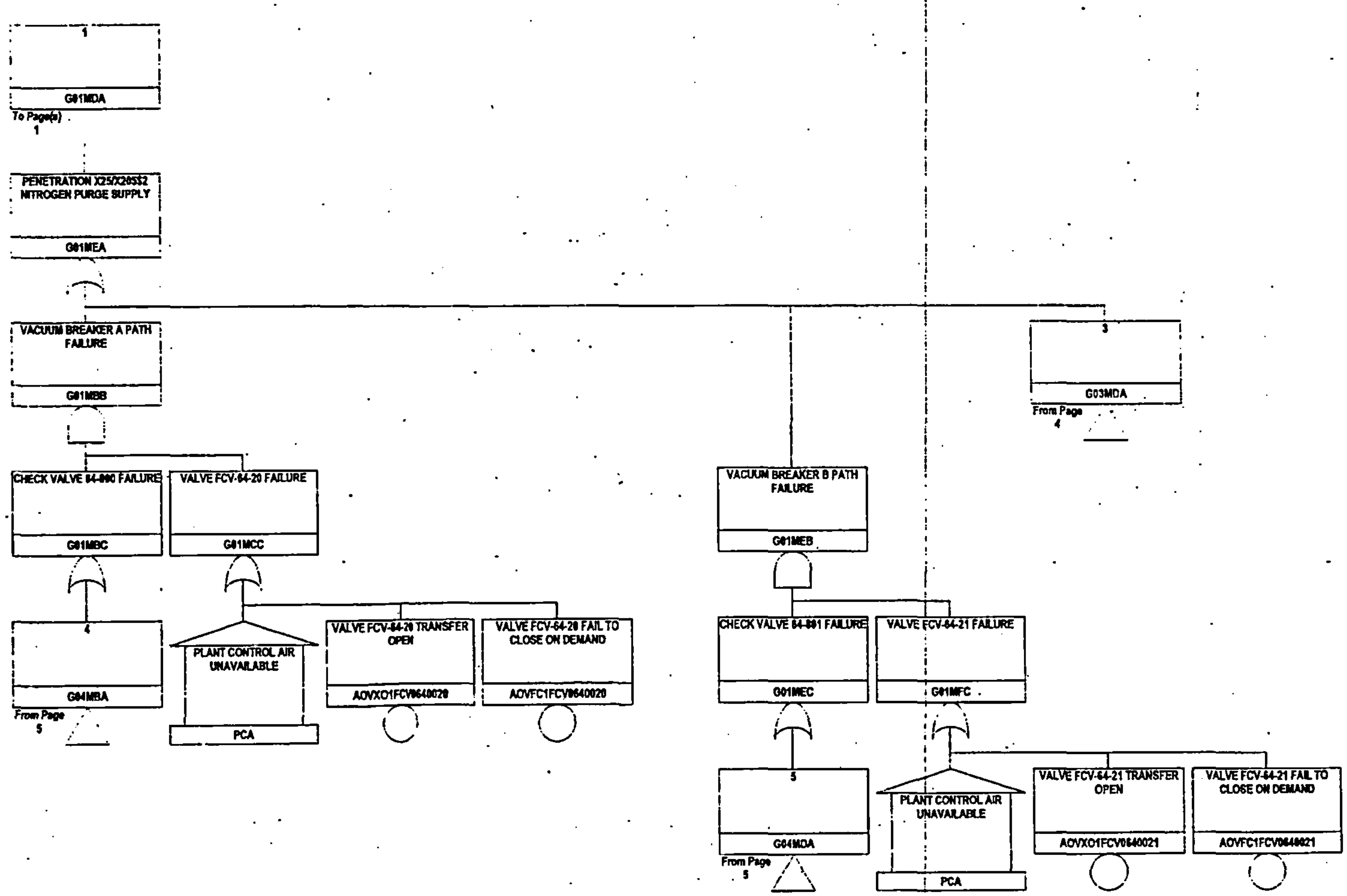
Notes to Table F-2:

(1) In these sensitivity cases the NPSH split fraction is simply set to 1.0.

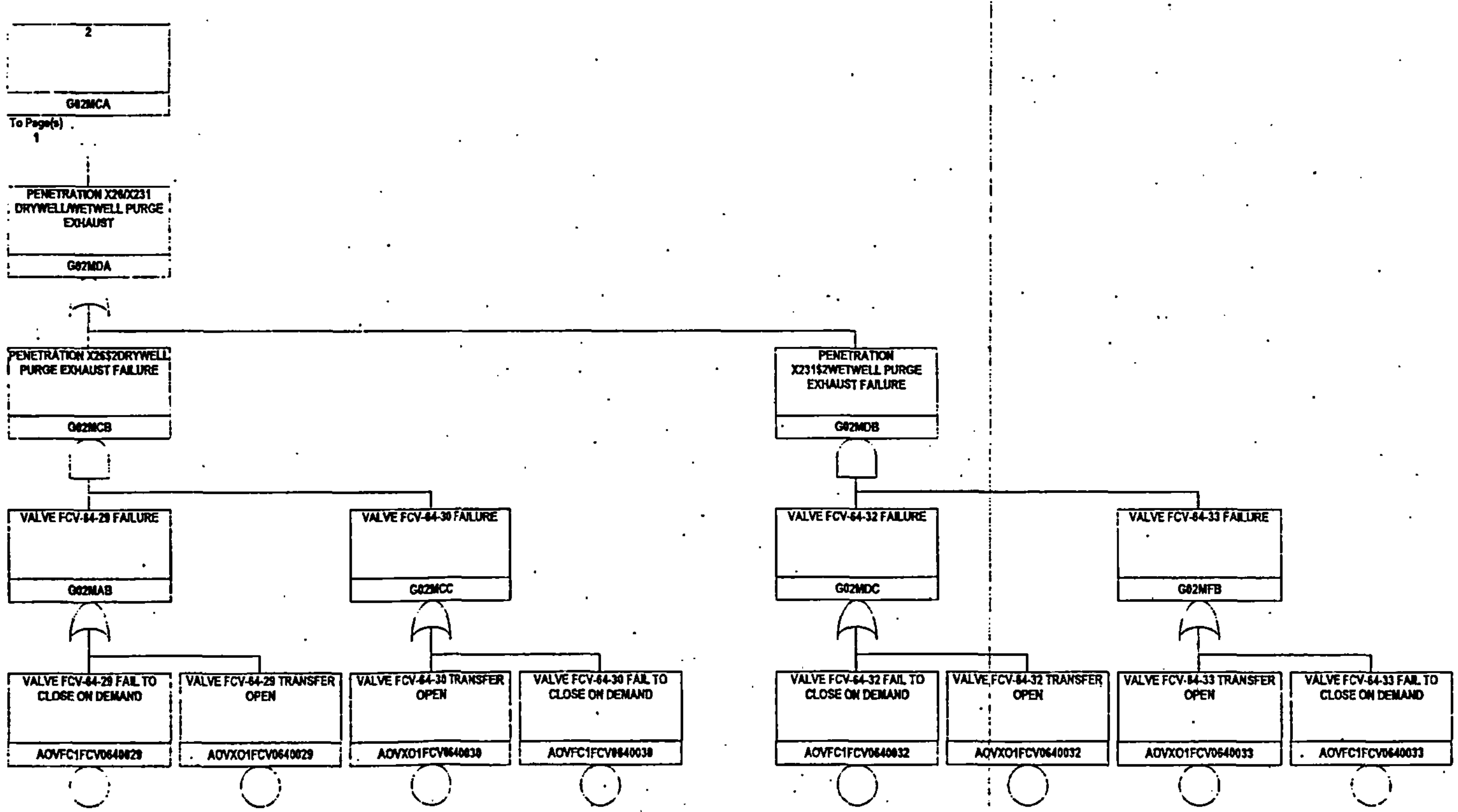

Top Event CIL: Containment Isolation Failure Unit 1
Page 1 of 9
Analyst: Lincoln Djang
Last Modification: 02/09/06
Date: 02/09/06
CONTAINMENT ISOLATION FAILURE =>3 INCHES (CIL)
G00NCA
ACTUATION SIGNALS UNAVAILABLE
G00MBB
PENETRATION SIZE GREATER THAN OR EQUAL TO 3 INCHES
G00MDB
CONTAINMENT PRE-EXISTING LEAK GREATER THAN 20 TIMES LA
CONPRE
RPS/PCIS/ SCIS RX LOW LVL (3) SIGNAL UNAVAILABLE
LVP
RPS/PCIS/ SCIS HI DRY WELL PRESS TRIP SIGNAL UNAVAILABLE
DWP
PENETRATION X25/X205 NITROGEN PURGE SUPPLY
G00MDC
PENETRATION X26/X231 DRYWELL/WETWELL PURGE EXHAUST
G00MEC
PENETRATION X19 DRYWELL EQUIPMENT DRAIN
CIL_X19
From Page 6
SUPPRESSION CHAMBER DRAIN
CIL_SUP_CH_DR
From Page 7
PENETRATION X18 DRYWELL FLOOR DRAIN
CIL_X18
From Page 8
1
G01MDA
From Page 2
2
G02MCA
From Page 3

Top Event CIL: Containment Isolation Failure Unit 1 | Page 2 of 9

Analyst: Lincoln Dyang | Last Modification: 02/09/06 | Date: 02/09/06

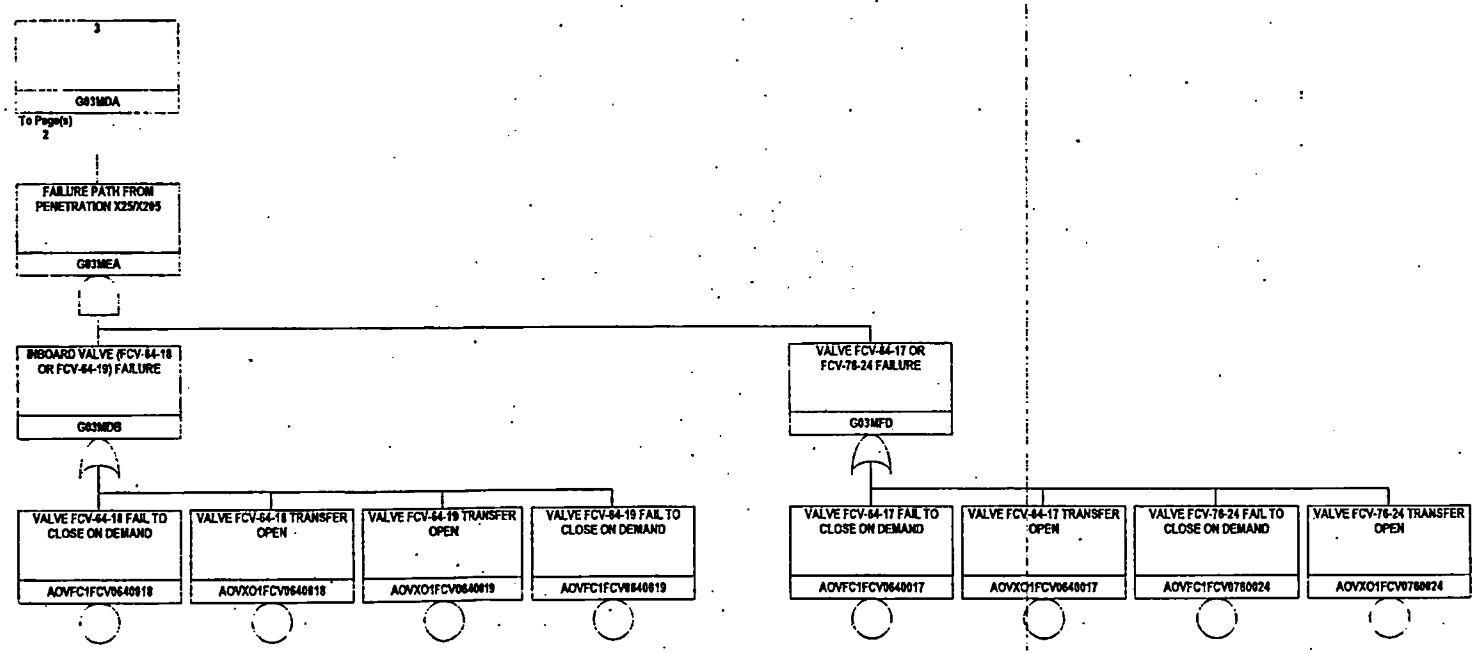


Top Event CIL: Containment Isolation Failure Unit 1 | Page 3 of 9
Analyst: Lincoln Djang | Last Modification: 02/09/06 | Date: 02/09/06

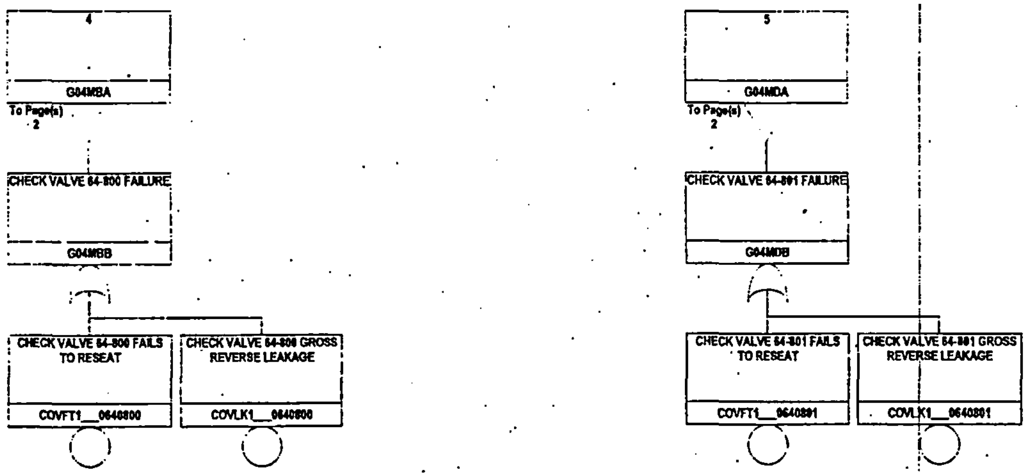


Top Event CIL: Containment Isolation Failure Unit 1

Analyst: Lincoln Djang | Last Modification: 02/09/06 | Page 4 of 9 | Date: 02/09/06



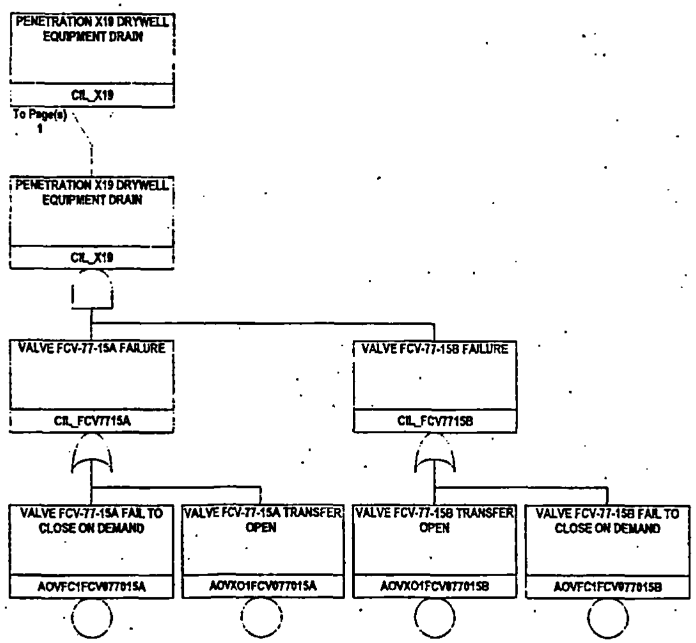
Top Event CIL: Containment Isolation Failure Unit 1		Page 5 of 9
Analyst: Lincoln Djang	Last Modification: 02/09/06	Date: 02/09/06



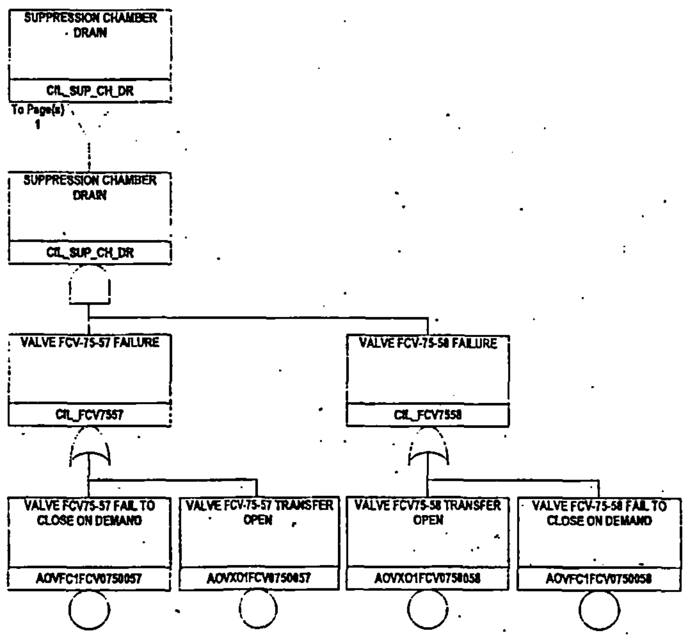
Top Event CIL: Containment Isolation Failure Unit 1

Page 6 of 9

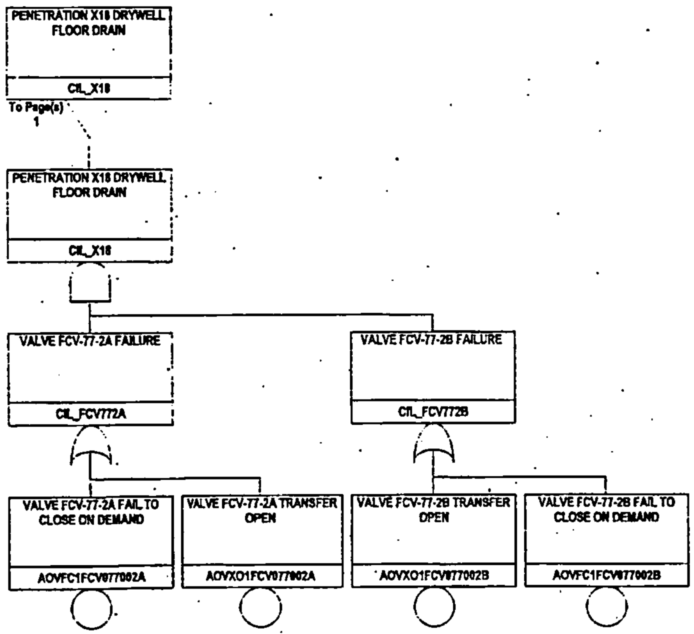
Analyst: Lincoln Djang | Last Modification: 02/09/06 | Date: 02/09/06



Top Event CIL: Containment Isolation Failure Unit 1		Page 7 of 9
Analyst: Lincoln Djang	Last Modification: 02/09/06	Date: 02/09/06



Top Event CIL: Containment Isolation Failure Unit 1		Page 8 of 9
Analyst: Lincoln Djang	Last Modification: 02/09/06	Date: 02/09/06

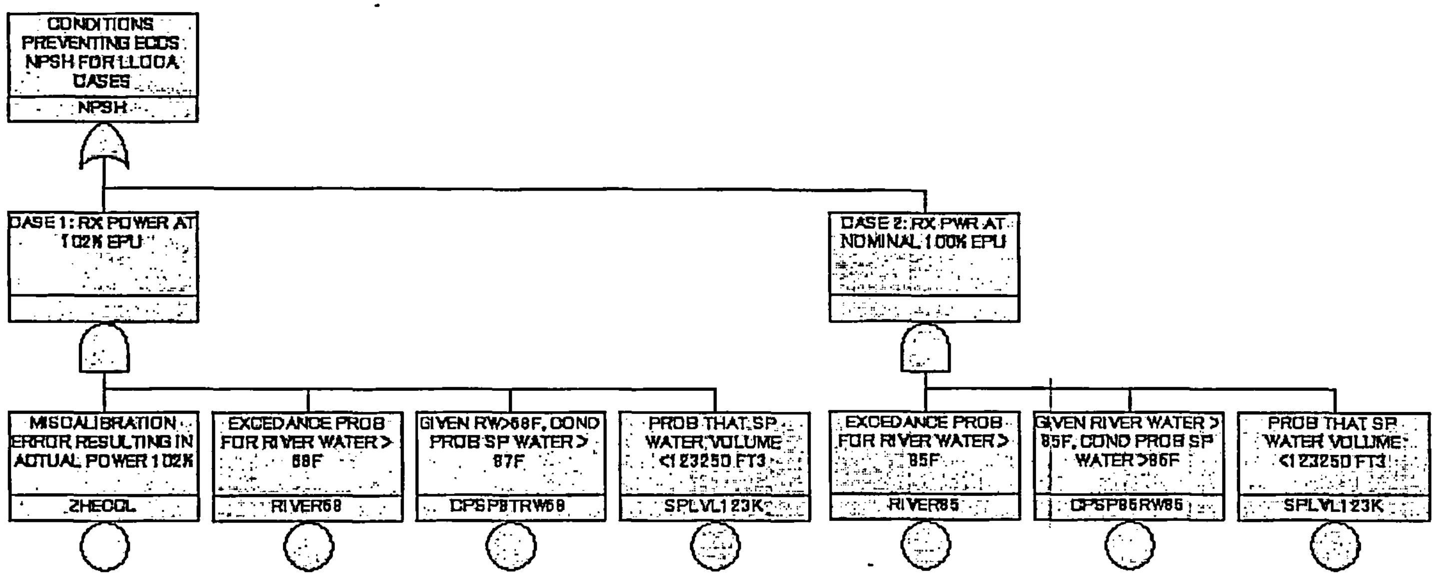


Top Event CIL: Containment Isolation Failure Unit 1 | Page 9 of 9

Analyst: Lincoln Djang | Last Modification: 02/03/03 | Date: 02/09/06

SYMBOL NAME	P#	SYMBOL TYPE
AOVFC1FCV0640017	2	BASIC_EVENT
AOVFC1FCV0640018	2	BASIC_EVENT
AOVFC1FCV0640019	2	BASIC_EVENT
AOVFC1FCV0640020	2	BASIC_EVENT
AOVFC1FCV0640021	2	BASIC_EVENT
AOVFC1FCV0640029	3	BASIC_EVENT
AOVFC1FCV0640030	3	BASIC_EVENT
AOVFC1FCV0640032	3	BASIC_EVENT
AOVFC1FCV0640033	3	BASIC_EVENT
AOVFC1FCV0750057	3	BASIC_EVENT
AOVFC1FCV0750058	3	BASIC_EVENT
AOVFC1FCV0760024	2	BASIC_EVENT
AOVFC1FCV077002A	4	BASIC_EVENT
AOVFC1FCV077002B	4	BASIC_EVENT
AOVFC1FCV077015A	4	BASIC_EVENT
AOVFC1FCV077015B	4	BASIC_EVENT
AOVXO1FCV0640017	2	BASIC_EVENT
AOVXO1FCV0640018	2	BASIC_EVENT
AOVXO1FCV0640019	2	BASIC_EVENT
AOVXO1FCV0640020	2	BASIC_EVENT
AOVXO1FCV0640021	2	BASIC_EVENT
AOVXO1FCV0640029	3	BASIC_EVENT
AOVXO1FCV0640030	3	BASIC_EVENT
AOVXO1FCV0640032	3	BASIC_EVENT
AOVXO1FCV0640033	3	BASIC_EVENT
AOVXO1FCV0750057	3	BASIC_EVENT
AOVXO1FCV0750058	3	BASIC_EVENT
AOVXO1FCV0760024	2	BASIC_EVENT
AOVXO1FCV077002A	4	BASIC_EVENT
AOVXO1FCV077002B	4	BASIC_EVENT
AOVXO1FCV077015A	4	BASIC_EVENT
AOVXO1FCV077015B	4	BASIC_EVENT
CIL_FCV7557	3	OR_GATE
CIL_FCV7558	3	OR_GATE
CIL_FCV7715A	4	OR_GATE
CIL_FCV7715B	4	OR_GATE
CIL_FCV772A	4	OR_GATE
CIL_FCV772B	4	OR_GATE
CIL_SUP_CH_OR	3	TRANSFER_OUT
CIL_SUP_CH_OR	1	TRANSFER_IN
CIL_SUP_CH_OR	3	AND_GATE
CIL_X18	4	TRANSFER_OUT
CIL_X18	1	TRANSFER_IN
CIL_X18	4	AND_GATE
CIL_X19	4	TRANSFER_OUT
CIL_X19	4	AND_GATE
CIL_X19	1	TRANSFER_IN
CONDPRE	1	BASIC_EVENT
COVFT1__0640800	3	BASIC_EVENT
COVFT1__0640801	3	BASIC_EVENT

SYMBOL NAME	P#	SYMBOL TYPE
COVLK1__0640800	3	BASIC_EVENT
COVLK1__0640801	3	BASIC_EVENT
DWP	1	HOUSE_EVENT
G00MBB	1	AND_GATE
G00MDB	1	OR_GATE
G00MDC	1	OR_GATE
G00MEC	1	OR_GATE
G01MBB	2	AND_GATE
G01MBC	2	OR_GATE
G01MCC	2	OR_GATE
G01MDA	2	TRANSFER_OUT
G01MDA	1	TRANSFER_IN
G01MEA	2	OR_GATE
G01MEB	2	AND_GATE
G01MEC	2	OR_GATE
G01MFC	2	OR_GATE
G02MAB	3	OR_GATE
G02MCA	3	TRANSFER_OUT
G02MCA	1	TRANSFER_IN
G02MCB	3	AND_GATE
G02MCC	3	OR_GATE
G02MDA	3	OR_GATE
G02MDB	3	AND_GATE
G02MDC	3	OR_GATE
G02MFB	3	OR_GATE
G03MDA	2	TRANSFER_OUT
G03MDA	2	TRANSFER_IN
G03MDB	2	OR_GATE
G03MEA	2	AND_GATE
G03MFD	2	OR_GATE
G04MBA	3	TRANSFER_OUT
G04MBA	2	TRANSFER_IN
G04MBB	3	OR_GATE
G04MDA	3	TRANSFER_OUT
G04MDA	2	TRANSFER_IN
G04MDB	3	OR_GATE
LVP	1	HOUSE_EVENT
PCA	2	HOUSE_EVENT
PCA	2	HOUSE_EVENT


CONDITIONS PREVENTING ECCS NPSH FOR LLOCA CASES
NPSH
CASE 1: RX POWER AT 102% EPU
CASE 2: RX PWR AT NOMINAL 100% EPU
MISCALIBRATION ERROR RESULTING IN ACTUAL POWER 102%
ZHECOL
EXCEDANCE PROB FOR RIVER WATER > 88F
RIVER88
GIVEN RW>88F, COND PROB SP WATER > 87F
CPSP87RW88
PROB THAT SP WATER VOLUME <123250 FT3
SPLVL123K
EXCEDANCE PROB FOR RIVER WATER > 85F
RIVER85
GIVEN RIVER WATER > 85F, COND PROB SP WATER >86F
CPSP86RW85
PROB THAT SP WATER VOLUME <123250 FT3
SPLVL123K