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Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

February 28, 2006

TVA-BFN-TS-431

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of) Docket No. 50-259
Tennessee Valley Authority)

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-431 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION SPSB-A.11 REGARDING EXTENDED POWER UPRATE - CREDIT FOR NET POSITIVE SUCTION HEAD (TAC NO. MC3812)

This letter provides TVA's supplemental responses to the NRC request for additional information SPSB-A.11 regarding an assessment of the credit for containment overpressure against the five key principles of risk-informed decision making.

On June 28, 2004 (Reference 1), TVA requested a TS change to allow Unit 1 to operate at extended power uprate conditions. As part of this TS change, TVA requested approval to take credit for containment overpressure in order to provide adequate net positive suction head (NPSH) to the Emergency Core Cooling System (ECCS) pumps. On October 3, 2005 (Reference 2), NRC requested TVA provide additional information regarding the ECCS pumps NPSH, including an assessment of the credit for containment overpressure against the five key principles of risk-informed decision making. The requested additional information is provided as Enclosure 1 to this letter. A

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detailed chronology of the correspondence related to the previous approval of NPSH for pre-uprate conditions on Units 2 and 3 is provided in Enclosure 2. A detailed description of plant systems related to the NPSH analysis is provided in Enclosure 3. The supporting risk assessment is provided as Enclosure 4.

The use of containment overpressure to ensure adequate NPSH for ECCS pumps during a limited time after a design basis accident is consistent with NRC staff positions, including Revision 3 of Regulatory Guide 1.82, and is part of the current licensing and design basis for Browns Ferry Units 2 and 3. Crediting containment overpressure results in a small increase in core damage frequency (CDF) and large early release frequency (LERF) of 1.53x10-9/yr. This small increase is well below the guidelines provided in Regulatory Guides 1.174 (10-6/yr for CDF and 10-7/yr for LERF).

TVA has determined that the additional information provided does not affect the no significant hazards considerations associated with the proposed TS changes. The proposed TS changes still qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

If you have any questions about this submittal, please contact me at (256) 729-2636. I declare under penalty of perjury that the foregoing is true and correct. Executed on February 28, 2006.

Sincerely,

William D. Crouch

William D. Crouch
Manager of Licensing
and Industry Affairs

References:

1. TVA letter, dated June 28, 2004, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Proposed Technical Specifications (TS) Change TS-431 - Request for License Amendment -

Extended Power Uprate (EPU) Operation."

2. NRC letter, dated October 3, 2005, "Browns Ferry Nuclear Plant, Unit 1 - Request for Additional Information for Extended Power Uprate (TS-431) (TAC No. MC3812)."

Enclosures:

1. Response to NRC Request for Additional Information Regarding Proposed Technical Specification (TS) TS-431 Extended Power Uprate - Credit for Net Positive Suction Head

2. Detailed Chronology of Correspondence Related to the Previous Approval of NPSH for Pre-uprate Conditions

3. Detailed Description of Plant Systems Related to the NPSH Analysis

4. BFN Extended Power Uprate Containment Overpressure Credit Risk Assessment

cc (Enclosures):
State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, AL 36130-3017

Enclosures
cc (Enclosures):

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-3415

Mr. Malcolm T. Widmann, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

NRC Senior Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

Margaret Chernoff, Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Eva A. Brown, Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY (TVA)

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED TECHNICAL SPECIFICATION (TS) TS-431

EXTENDED POWER UPRATE - CREDIT FOR NET POSITIVE SUCTION HEAD

NRC REQUEST 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 Probabilistic Risk Assessment (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 decision making 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.

TVA RESPONSE

INTRODUCTION

The proposed change for BFN Unit 1 Extended Power Uprate (EPU) includes crediting containment overpressure (COP) in ensuring adequate NPSH to Emergency Core Cooling System (ECCS) pumps following limiting events which cause suppression pool temperature increase. These events are Loss of Coolant Accident (LOCA), Anticipated Transients Without Scram (ATWS), Appendix R and Station Blackout (SBO). COP is defined for BFN as containment pressure in excess of 14.4 PSIA. For the Design Basis Accident (DBA) LOCA, the need to credit COP is due only to consideration of a number of worst case assumptions. More realistic analyses show that elimination of worst case assumptions that have reasonable probability distributions would eliminate the need for COP credit. Results of realistic analyses are presented along with associated probability distributions.

Parameters affecting NPSH were included in a modified PRA model along with probability distributions to show the risk impact associated with reliance on containment integrity and overpressure for ECCS pump NPSH.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" was utilized as a guide for providing risk insights and more realistic analyses to supplement the deterministic analyses and worst case assumptions used in the licensing basis LOCA analysis. These risk insights are used to characterize the degree to which COP is relied upon in the safety design basis.

BACKGROUND

The following provides an abbreviated background for ECCS strainer issues and the use of COP. An in-depth discussion of the regulatory background is provided in Enclosure 2.

Previously, BFN Units 2 and 3 installed new large capacity ECCS strainers to meet the requested actions of NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors." As part of the resolution of Bulletin 96-03, credit for available COP to maintain adequate NPSH following a LOCA was required. BFN requested a change to the licensing basis for Units 2 and 3 in Reference 1 (as supplemented by Reference 2) and received NRC approval for the requested change in Reference 3.

At the time Bulletin 96-03 was resolved for BFN Units 2 and 3, BFN Unit 1 was in an extended shutdown and no actions were taken to resolve Bulletin 96-03 for Unit 1. As part of the restart for Unit 1, large capacity ECCS strainers of the same design as previously installed on Units 2 and 3 have been installed on Unit 1. Credit for available COP to maintain adequate NPSH following a LOCA is also required for Unit 1, the same as Units 2 and 3. Since the intent is to restart Unit 1 at a licensed power level of 120% of original licensed power, NPSH margin analyses were not specifically performed for pre-EPU power levels.

For EPU, BFN is proposing a change in the licensing basis to extend the existing approved credit for COP to provide adequate NPSH following a LOCA for Units 2 and 3 and to apply the same credit for COP to provide adequate NPSH following a LOCA for Unit 1.

Currently for BFN Units 2 and 3, Reference 3 approves the crediting of 3 psi COP for the Residual Heat Removal (RHR) pumps for the first 10 minutes following a LOCA (short-term requirement) and 1 psi COP for the core spray pumps from approximately 5500 to 35000 seconds (about 8.2 hours) following a LOCA (long-term requirement). For EPU, BFN is requesting for all three units approval of 3 psi COP for the RHR pumps for the first 10 minutes following a LOCA (short-term requirement) and 3 psi COP for the core spray pumps from approximately 4,100 to 52,300 seconds (about 13.4 hours) following a LOCA (long-term requirement).

As part of the EPU effort, BFN has also given more consideration for NPSH requirements during Appendix R, ATWS, and SBO events. These events (designated as Special Events at BFN) were not addressed in response to Generic Letters 96-03 and 97-04 and are not addressed in Regulatory Guide (RG) 1.82. Conservative evaluation of these events determined that BFN will credit available containment pressure for the RHR pumps following an SBO, ATWS, and Appendix R events.

SYSTEM DESCRIPTION

The following provides an abbreviated system description. An in-depth description of the BFN containment and ECCS systems is provided in Enclosure 3. The BFN units are BWR-4s with Mark I containments, which incorporate a large torus shaped suppression pool. Four RHR pumps and four Core Spray pumps take suction from the suppression pool through a common ring header which connects to the torus at four locations through a stacked disc strainer mounted on each nozzle. The ECCS ring header is also the alternate suction for the High Pressure Core Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system pumps. The normal suction path for the HPCI and RCIC system pumps is the condensate storage tank (CST).

The four strainers are not associated with individual pump suctions but direct suppression pool water to the common ECCS ring header. Therefore, interaction between operating pumps is considered when determining pump suction pressures.

LOCA EVENT DESCRIPTION

SHORT TERM (T<10 minutes)

The bounding design basis event for determining NPSH margin is a double ended recirculation discharge line break. This event results in maximum suppression pool temperature and maximum total pump flow. The discharge line break is chosen because the low system resistance on the broken line produces the most limiting flow and NPSH for two RHR pumps which are assumed to be pumping into the broken line inside containment. At the beginning of the event, four RHR pumps and four Core Spray pumps start automatically and align to inject to the Reactor Pressure Vessel (RPV). Two RHR pumps inject to the RPV at 10,000 gpm each, two RHR pumps inject through the broken line into the containment at 11,000 gpm each (greater than design flow), and four Core Spray pumps inject to the RPV at 3,125 gpm each. This mode of operation is assumed for 10 minutes consistent with not crediting operator action for 10 minutes. ECCS strainers are assumed to accumulate the maximum equilibrium debris load. During this time suppression pool temperature reaches 155.4°F and only the RHR pumps require credit for COP in order to have sufficient NPSH margin as shown in Figure 1.

LONG TERM (T>10 minutes)

At 10 minutes, operator action is assumed which places the minimum complement of ECCS pumps into modes required for long term cooling. Two Core Spray pumps (one loop) at design flow of 3,125 gpm each are assumed for core cooling, and two RHR pumps in one loop in containment cooling mode at 6,500 gpm each are assumed for pool cooling. Containment spray mode of containment cooling is chosen to minimize available containment pressure. Only two of four RHR pumps are assumed for pool cooling due to single failure considerations. ECCS strainers are assumed to accumulate the maximum equilibrium debris load. During this time suppression pool temperature reaches 187.4°F and only the two Core Spray pumps require credit for COP in order to have sufficient NPSH margin as shown in Figure 2.

REGULATORY GUIDE 1.174 ASSESSMENT

RG 1.174, Section 2, provides the set of five key principles that licensing basis changes are expected to meet:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (Ref. RG 1.175).

5. The impact of the proposed change should be monitored using performance measurement strategies.

1. CURRENT REGULATIONS

On June 28, 2004, TVA requested a TS change to allow Unit 1 to operate at extended power uprate conditions. As part of this TS change, TVA requested approval to take credit for post-accident COP in order to provide adequate NPSH to the ECCS pumps.

TVA has reviewed the requested credit for COP against those aspects of the BFN licensing basis that may be affected by the proposed change, including rules and regulations, the Updated Final Safety Analysis Report (UFSAR), TSs, License Conditions, and licensing commitments. As previously discussed, NRC previously approved the use of COP to maintain adequate ECCS pump NPSH on BFN Units 2 and 3. The use of COP does not invalidate TVA's compliance with 10 CFR 50.54(o), Appendix J to 10 CFR 50, 10 CFR 50.46 and Appendix K to 10 CFR 50. The use of COP is discussed in UFSAR Section 6.5.5.

The approval of credit for post-accident COP is consistent with the NRC's Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, is consistent with NRC staff positions, including Revision 3 of Regulatory Guide 1.82, and is part of the current licensing and design basis for Browns Ferry Units 2 and 3. The credit is supported by the BFN PRAs and the results satisfy the numerical targets contained in NRC Regulatory Guide 1.174. Alternatives which would preclude the need for the use of COP, such as the replacement of pumps or heat exchangers are not practical.

2. DEFENSE-IN-DEPTH

Defense in depth philosophy is maintained by avoiding over reliance on specific features, human actions and assumptions to ensure plant safety. By preserving the function of the ECCS, multiple barriers of fuel cladding and primary containment are maintained. The ECCS functions are being preserved by the proposed plant design and operation. For a LOCA, reliance on COP is only necessary assuming low probability combinations of worst case assumptions governing heatup of the suppression pool. RG 1.174 provides guidance for acceptable methods to assess defense in depth principles. The following addressed the aspects of defense in depth that are potentially impacted by the requested change.

- Capability of Containment to Provide Containment Overpressure

The containment is designed to withstand conditions well in excess of those associated with a DBA. Pre-existing containment leakage is well below that which could defeat maintenance of required COP. At the end of 24 hours, 2 percent leakage results in an approximate 0.3 psi decrease in the 3.4 psig available containment pressure compared with no leakage. The containment is equipped with automatic containment isolation which is designed to single failure criteria. The COP available is the thermodynamic result of the event itself and does not depend on operator actions or systems other than the containment.

- Excess Containment Cooling Capability

  Long-term suppression pool temperature in design basis events is determined crediting only two of the four RHR pumps and heat exchangers. Emergency Operating Instructions (EOIs) dictate using all available RHR pumps for suppression pool cooling. Single failures such as loss of a power supply or failure of containment cooling valves, failure of a service water pump or RHR heat exchanger valves can disable one or two RHR pumps for containment cooling. If no such single failure is assumed in the long term analysis (>10 minutes) then suppression pool temperature remains below 166.4°F with four RHR pumps or 175°F with any three pumps and positive NPSH margin would be maintained long term without COP. These analyses were performed using the same conservative assumptions for input parameters as the licensing basis analysis. Core Spray pumps require credit for COP above 175.8°F. The RHR pumps do not require COP at the peak pool temperature of 187.4°F. The likelihood of failing any two RHR pumps is 8.2E-3.

It can be concluded that defense in depth philosophy is preserved following the proposed change since multiple failures of safety related features would have to be postulated in order to impact ECCS functions. Credit for COP does not rely upon new operator actions or changes to the accident analysis methodologies.

3. SAFETY MARGINS

Analyses for design basis events are performed with established margins added to important parameters to account for uncertainty. Significant parameter margins included in the NPSH analysis were examined and analysis results were obtained using more realistic values. This demonstrates that there is ample margin to ECCS pump functional failure in design basis LOCA events without credit for COP. The following table provides the parameters of interest, the values used in the safety analysis and the associated realistic values.

PARAMETER	LICENSING BASIS VALUE	REALISTIC VALUE	COMMENT
Initial Power	102% Licensed Thermal Power	100% Licensed Thermal Power	Probability of 102% power is 5.0E-3
Decay Heat Model	ANSI 5.1 (plus 2σ)	ANSI 5.1 (w/o 2σ)	
Service Water Temperature	95°F	92°F	Exceedance probability for 92°F is less than 6.0E-2
Initial Suppression Pool Temperature	95°F (TS maximum)	92°F	Exceedance probability for 92°F is 8.25E-2
Heat Exchanger K Value	223 BTU/Hr-°F	241 BTU/Hr-°F 225 BTU/Hr-°F	Based on realistic fouling factor of 0.0020 vs 0.0025 and maximum number of tubes plugged (1.5%) 1.5% tube plugging only
Initial Suppression Pool Volume	121,500 ft^3 (TS minimum)	125,640 ft^3	Nominal value
Containment Heat Sinks	Assumes no heat sinks	Includes realistic heat sinks	Heat sinks are always present but not normally credited

Sensitivity analyses were performed (with selected analyses verified), which are summarized in Table 1. The purpose of these analyses was to identify input parameter combinations where COP was not required (e.g., suppression pool temperature below 175.8°F).

- Sensitivity to RHR Service Water (RHRSW) Temperature

  Suppression pool temperature response was examined as a function of RHRSW temperature which is a seasonal variable. Figure 3 shows Suppression Pool temperature as a function of RHRSW temperature using both licensing basis input values and realistic values. These analyses show that COP is not required for RHRSW temperatures 70°F or below assuming all design basis inputs and 86°F using realistic inputs. The probability of exceeding 70°F is 3.97E-1 and for 86°F is 1.40E-1.

- Realistic Values

  Suppression pool temperature for the DBA-LOCA was evaluated by altering the input parameters to reflect the realistic values given above. Defense in depth assumptions such as RHR pump availability were not changed. This evaluation shows that suppression pool temperature remains below that which COP is required (175.8°F). This is indicated as Case 4a in Table 1 and shows that credit for COP is not required when realistic input values are assumed.

- Margin in Manufacturers Curves for NPSH Required ($NPSH_R$)

  The licensing basis need for COP is based on the conservative assumption in NPSH calculations that the RHR and Core spray pumps will not perform their function at NPSH Available ($NPSH_a$) values less than the manufacturers $NPSH_R$. The values used were derived from manufacturers testing for each pump. Suction pressures were reduced with 3 percent reduction in total dynamic head (TDH) to establish minimum NPSH. At this value, the pumps will operate without degradation.

  BFN RHR pumps are Sulzer-Bingham model 18x24x28 CVIC. Assuming no credit for COP in the limiting short-term LOCA scenario, RHR pumps would be required to be operated for less than 10 minutes at 24.3 feet $NPSH_a$ (broken loop) versus 30 feet $NPSH_R$ or 25.2 feet $NPSH_a$ (intact loop) versus 26 feet $NPSH_R$. Negative NPSH margin of this magnitude for short periods of time will not prevent the RHR pumps from performing long-term in the event. Additional NPSH testing was performed on a BFN RHR pump in 1976 and reported to NRC in Reference 4. In this test, the RHR pump was operated 10,000 GPM (design flow) at approximately 24 feet of NPSH without cavitation and as low as 16 feet without damage. This is compared to 26 feet assumed to be the NPSH limit for the short-term COP requirements for the intact loop at design flow. This demonstrated that the RHR pumps can be operated below the manufacturers curve for at least 10 minutes without damage. This data demonstrated that the RHR pumps have NPSH margin assuming COP is not available. Therefore, in the unlikely event that COP was lost in the short-term LOCA, the function of the RHR pumps would not be affected for the short- and long-term.

By comparison to the RHR pumps, the Core Spray pumps would be challenged in the long-term scenario in the event that COP was lost. Core Spray pumps do not require COP in the short-term (Refer to Figure 1). BFN Core Spray pumps are Sulzer-Bingham model 12x16x14.5 CVDS. Assuming no credit for COP, the Core Spray pumps used for long-term core cooling (>10 minutes) would be expected to operate between 27 feet and 22.6 feet of NPSH verses 27 feet used in NPSH calculations for approximately 13.4 hours as Suppression Pool temperature peaks above 175.8°F during the LOCA. In the unlikely event they become degraded, there is a reasonable likelihood that the affected pumps would still be able to function. In addition, only one of the two Core Spray loops is required to be operated for adequate core cooling and the non-operating Core Spray loop would be available to operators if the operating loop failed after some time period. RHR pumps would also be available in the LPCI mode for core cooling in conjunction with their suppression pool cooling function should all Core Spray pumps become unavailable. COP is not required for RHR pumps in the long-term scenario. Therefore in the unlikely event that COP was lost in the long-term LOCA, the decay heat removal and core cooling functions would be maintained.

It can be concluded that safety margins are preserved following the proposed change. Sensitivity analyses show that COP is not required if realistic inputs are utilized without any changes to the accident analysis methodologies.

4. RISK ASSESSMENT

TVA has evaluated the risk impact of utilizing COP to satisfy the NPSH requirements for RHR and Core Spray pumps to mitigate the consequences of a DBA LOCA. The risk assessment evaluation used the current BFN Unit 1 PRA internal events (including internal floods) model. The evaluation is provided as Enclosure 4 to this letter. The steps taken to perform this risk assessment evaluation were:

1. Evaluate sensitivities to the DBA LOCA accident calculations to determine under what conditions credit for COP is necessary to satisfy low pressure ECCS pump NPSH requirements;

2. Revise all large LOCA accident sequence event trees to make low pressure ECCS pumps dependent upon containment isolation when other plant pre-conditions exist (i.e., Service Water initial high temperature, Suppression Pool initial high temperature);

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

4. Quantify the modified PRA models and determine the change in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF); and

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

Crediting COP resulted in a small increase in CDF and LERF of 1.53 E-9/yr. This small increase was well below the guidelines provided in RG 1.174.

ATWS, SBO, and Appendix R are highly unlikely event scenarios which are defined by failure of multiple features. Failure assumptions in these events are beyond design basis. Additional failures such as loss of containment integrity need not be assumed. Deterministic analyses have shown that COP will be available as thermodynamic result of the event itself provided that containment integrity is maintained. This is acceptable given the low probability of the events.

5. MONITORING

Performance monitoring is performed for parameters important to ECCS NPSH analyses to ensure that assumptions remain valid and that corrective actions are initiated for deficiencies.

- Containment Integrity Monitoring

  During normal power operations, the containment is inerted with nitrogen and maintained at greater than or equal to 1.1 psi positive pressure relative to the suppression chamber in accordance with TS 3.6.2.6. Technical Requirements Manual 3.6.5 limits nitrogen makeup to 542 scfh and is determined every 24 hours. This would identify any pre-existing leak in the drywell portion of containment.

  10 CFR 50.54(o) and 10 CFR Part 50 Appendix J require leak rate testing of the containment structure, penetrations and isolation valves at the maximum predicted LOCA pressure. Containment leak rate testing tests containment penetrations and limits total leakage to < 0.6La. La is two weight percent per day at 50.6 PSIG. Available containment pressure is calculated assuming two weight percent per day throughout the event which is conservative.

  10 CFR 50.55a(ii)B requires periodic in-service examination of the containment structure in accordance with the American Society of Mechanical Engineers Code.

- NPSH Monitoring

  The EOIs include precautionary statements warning the operator that continuous operation of the low pressure injection system pumps with inadequate NPSH may result in pump damage or pump inoperability and that reducing containment pressure may affect pump NPSH. The operator is instructed to monitor NPSH using an NPSH limit curve, showing pump flow versus suppression pool temperature for various suppression pool pressures. The EOIs also list additional indications of inadequate NPSH. Operators are trained on these procedures as part of their periodic re-qualification program.

RG 1.174 CONCLUSION

The use of COP to ensure an adequate NPSH for ECCS pumps during a limited time after a design basis accident is consistent with NRC staff positions, including Revision 3 of RG 1.82, and is part of the current licensing and design basis for BFN Units 2 and 3. Alternatives which would preclude the need for the use of COP, such as the replacement of pumps or heat exchangers are not practical. Deterministic evaluations and analyses, which were performed in accordance with regulatory requirements, have demonstrated that an adequate level of protection is maintained.

Even though the use of COP was requested on a deterministic basis, a risk-informed assessment was performed in accordance with the guidelines contained in RG 1.174, Revision 1. In summary, a defense-in-depth philosophy is maintained by avoiding an over reliance on specific features, human actions, or assumptions to ensure safety. Safety margins are maintained since realistic analyses demonstrate that adequate NPSH exists for the ECCS pumps without crediting COP. Crediting COP results in a small increase in CDF and LERF of 1.53×10^{-9}/yr. This small increase is well below the guidelines provided in RG 1.174 (10^{-6}/yr for CDF and 10^{-7}/yr for LERF). The integrity of the primary containment and the associated primary containment isolation valves are monitored using diverse performance measurement strategies that ensure the detection and correction of adverse conditions.

REFERENCES

1. TVA letter, T.E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - License Amendment Regarding Use of Containment Overpressure for Emergency Core Cooling System (ECCS) Pump Net Positive Suction Head (NPSH) Analyses," September 4, 1998.

2. TVA letter, T.E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Response to Request for Additional Information (RAI) Relating to Units 2 and 3 License Amendment Regarding Use of Containment Overpressure for Emergency Core Cooling System (ECCS) Pump Net Positive Suction Head (NPSH) Analyses," November 25, 1998.

3. NRC letter, W.O. Long to TVA, "Browns Ferry Nuclear Plants, Units 2 and 3 - Issuance of Amendments Regarding Crediting of Containment Overpressure for Net Positive Suction Head Calculations for Emergency Core Cooling Pumps (TAC Nos. MA3492 and MA3493)," September 3, 1999.

4. TVA letter, J.E. Gilleland to NRC, "Browns Ferry Nuclear Plant Unit 3 - Reportable Deficiency - Potential for RHR Pump Operation in Excess of Design Runout - IE Control No. H01172F2," May 21, 1976.

TABLE 1
SENSITIVITY ANALYSES OF VARIOUS REALISTIC INPUT PARAMETERS
DURING THE DESIGN BASIS ACCIDENT LOSS OF COOLANT ACCIDENT LONG-TERM PHASE

Case	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	Core Spray Pumps in Operation	Initial SP Water Volume	ECCS Strainer Debris Loading	Credit for Containment Heat Sinks	Peak SP Temp (°F)	Containment Overpressure Credit Required
Base Case*	EPU Licensing Calculation – DBA LOCA	102% EPU	ANSI 5.1 w/2σ	95	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	187.3	Yes
Case 1*	No Single Failure	102% EPU	ANSI 5.1 w/2σ	95	95	4	Full design	4	4	4000	223	4	Minimum	Yes	No	166.4	No
Case 1a*	3 Pumps in SPC	102% EPU	ANSI 5.1 w/2σ	95	95	3	Full design	3	3	4000	223	4	Minimum	Yes	No	175.0	No
Case 2	DBA Calculation but Initial SW Temperature = 85°F	102% EPU	ANSI 5.1 w/2σ	85	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	182.0	Yes
Case 2a	DBA Calculation but Initial SW Temperature = 75°F	102% EPU	ANSI 5.1 w/2σ	75	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	177.6	Yes
Case 2b*	DBA Calculation but Initial SW Temperature = 70°F	102% EPU	ANSI 5.1 w/2σ	70	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	175.9**	No
Case 2c	DBA Calculation but Initial SW Temperature = 65°F	102% EPU	ANSI 5.1 w/2σ	65	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	174.3	No
Case 3	DBA Calculation but Initial SP Temperature = 85°F	102% EPU	ANSI 5.1 w/2σ	95	85	2	Full design	2	2	4000	223	2	Minimum	Yes	No	183.8	Yes
Case 4	100% Initial Power, Minimum SP Level, and No Heat Sink Credit	100% EPU	ANSI 5.1 w/o 2σ	92	92	2	Full design	2	2	4000	241	2	Minimum	Yes	No	177.0	Yes
Case 4a	100% Initial Power, Nominal SP Level, and Heat Sink Credit	100% EPU	ANSI 5.1 w/o 2σ	92	92	2	Full design	2	2	4000	241	2	Nominal	Yes	Yes	174.7	No
Case 4b*	100% Initial Power, Minimum SP Level, and Heat Sink Credit	100% EPU	ANSI 5.1 w/o 2σ	92	92	2	Full design	2	2	4000	225	2	Minimum	Yes	Yes	178.9	Yes
Case 4c*	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/o 2σ	88	92	2	Full design	2	2	4000	225	2	Minimum	Yes	Yes	175.8	No

*- Case verified by formal analyses. ** - This value is acceptable for demonstrating sensitivity analysis results.

FIGURE 1
NET POSITIVE SUCTION HEAD REQUIREMENTS FOR DESIGN BASIS LOSS OF COOLANT ACCIDENT – SHORT TERM

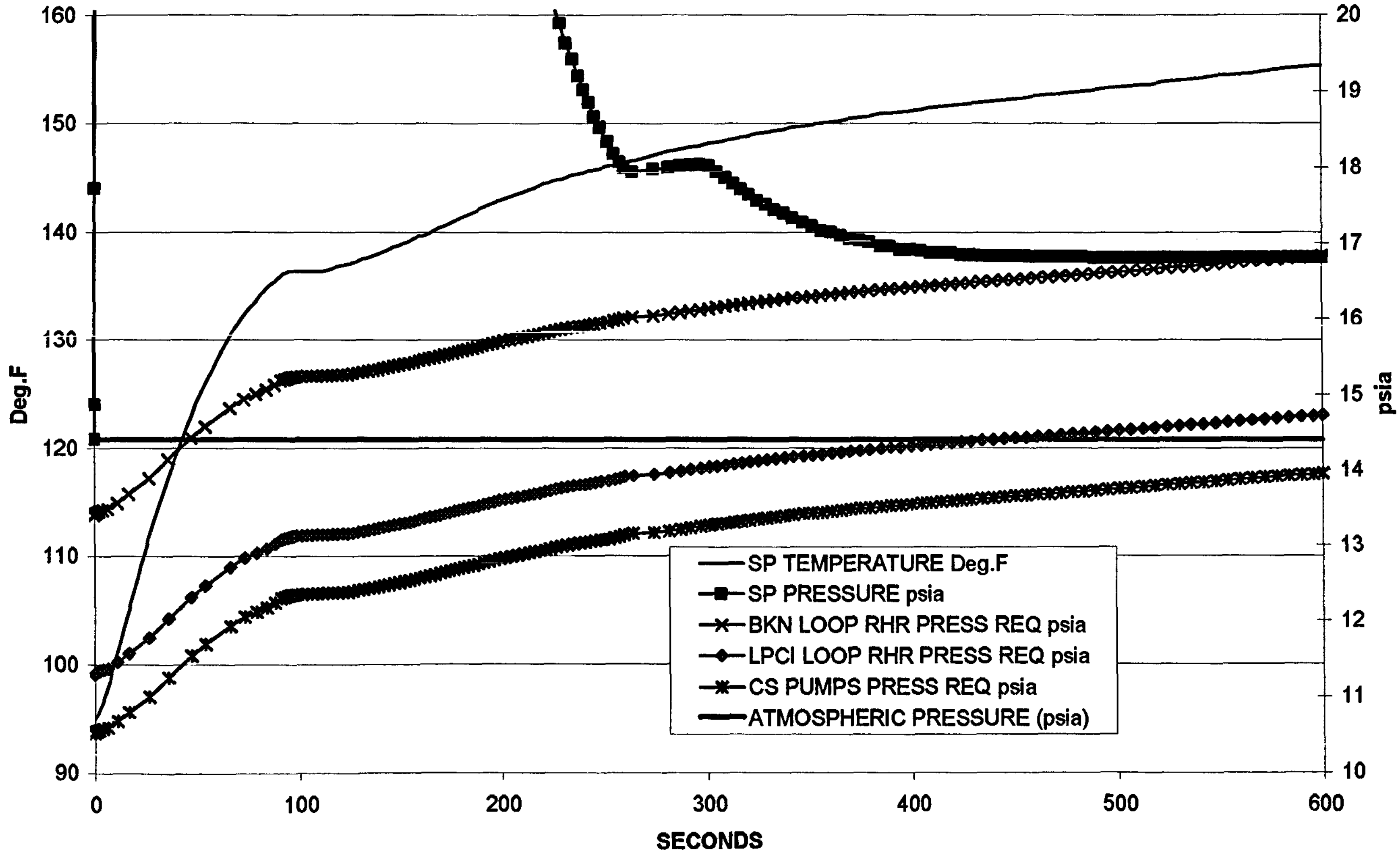


FIGURE 2
NET POSITIVE SUCTION HEAD REQUIREMENTS FOR DESIGN BASIS LOSS OF COOLANT ACCIDENT – LONG TERM

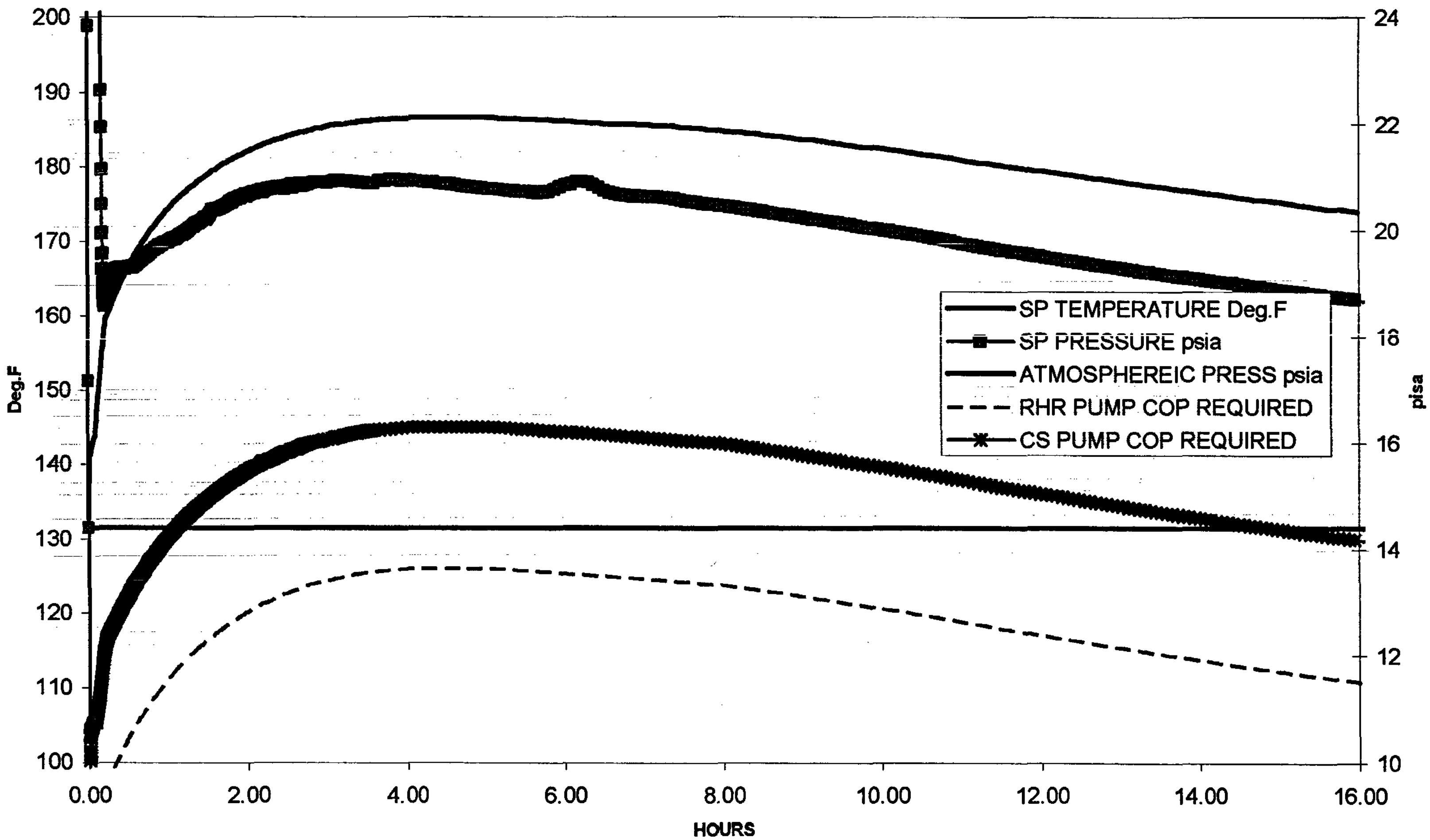
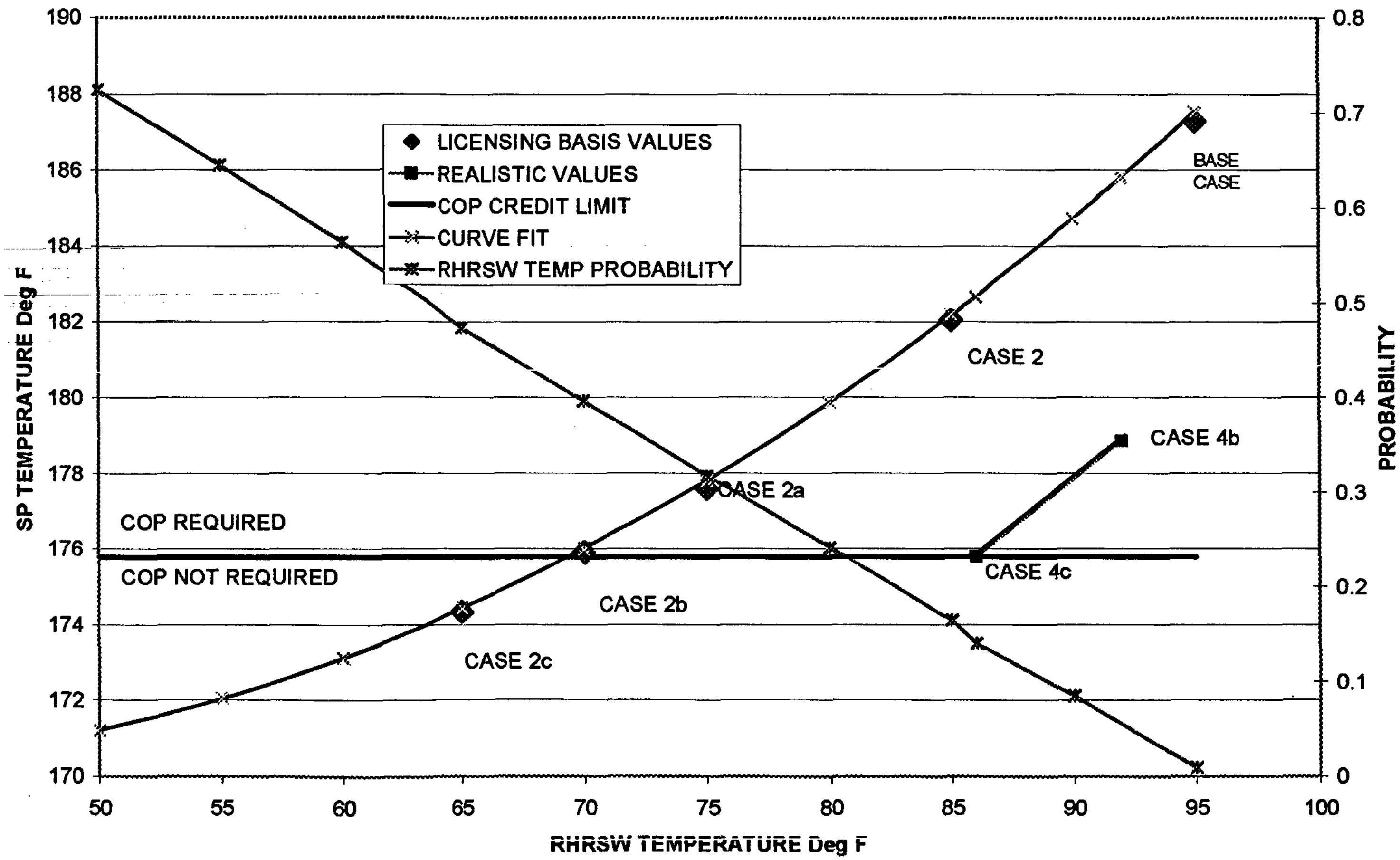


FIGURE 3
CONTAINMENT OVERPRESSURE SENSITIVITY TO RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) TEMPERATURE AND NET POSITIVE SUCTION HEAD SENSITIVITY TO RHRSW TEMPERATURE



ENCLOSURE 2
TENNESSEE VALLEY AUTHORITY (TVA)
BROWNS FERRY NUCLEAR PLANT UNIT 1
DETAILED CHRONOLOGY OF CORRESPONDENCE RELATED TO THE PREVIOUS APPROVAL OF NPSH FOR PRE-UPRATE CONDITIONS

Following a postulated Loss of Coolant Accident (LOCA), the Residual Heat Removal (RHR) and Low Pressure Core Spray (LPCS) pumps operate to provide the required core and containment cooling. The use of containment overpressure to maintain adequate pump net positive suction head (NPSH) is required to ensure essential pump operation. The limiting NPSH conditions occur during either short-term or long-term post-LOCA pump operation depending on the total pump flow rates, debris loading on the suction strainers, and suppression pool temperature. As chronicled below, credit for containment overpressure (up to 3 psi short-term for the RHR pumps and 1 psi long-term for the LPCS pumps) was extensively reviewed and subsequently approved by NRC.

On May 6, 1996, the NRC issued NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," (Reference 1). That bulletin was issued following events at several operating reactors where clogging of containment cooling pump suction strainers adversely impacted pump operation. As a result the NRC requested licensees to take actions to protect Emergency Core Cooling System (ECCS) pump strainers from clogging, and ensure pumps have adequate NPSH to fulfill their function.

By letter dated July 25, 1997, TVA responded to NRC Bulletin 96-03 (Reference 2). That letter outlined its proposed actions for resolution of NRC'S concerns for loss of ECCS following a Design Basis LOCA. To ensure adequate ECCS NPSH during and following accidents, TVA stated it planned to install larger capacity passive strainers and credit for a containment pressure in excess of atmosphere for a short period of time. TVA indicated that it would implement appropriate modifications to BFN Unit 1 prior to its restart.

By letter dated August 25, 1997, TVA supplemented its July 25, 1997 response to NRC Bulletin 96-03 (Reference 3). TVA indicated that pursuant to discussions with the NRC staff, it was preparing a license amendment request to allow crediting containment overpressure to ensure adequate ECCS pump NPSH during and following accidents. TVA also indicated that the NRC had previously approved crediting containment overpressure for ensuring ECCS NPSH as part of the BFN original licensing basis.

By letter dated October 7, 1997, the NRC issued Generic Letter (GL) 97-04, "Assurance of Sufficient Net Positive Suction Head (NPSH) for Emergency Core Cooling and Containment Heat Removal Pumps," (Reference 4). GL 97-04 requested that licensees review their design basis analyses used to determine the available NPSH for the ECCS and containment heat removal pumps that take suction from the containment following a design basis LOCA, and to provide specific information used therein. GL 97-04 requested, in part, that licensees specify whether credit is taken in their ECCS NPSH analyses for containment overpressure, and if so, identify the amount of overpressure needed and the minimum overpressure available.

TVA provided its 90-day response to GL 97-04 with a letter dated January 5, 1998 (Reference 5). In that letter, TVA indicated that BFN Unit 1 was at that time shut down and defueled. Accordingly, TVA indicated that it would evaluate BFN Unit 1 ECCS and containment cooling pump NPSH prior to its restart. TVA summarized actions taken and planned in response to NRC Bulletin 96-03, provided a description of containment debris analyses performed for BFN Units 2 and 3, and reiterated its intent to submit a license amendment request to support credit for containment overpressure. That submittal also provided required and available BFN Units 2 and 3 ECCS pump NPSH, and assumed a containment overpressure of 2 psig for the limiting case. By letter dated June 11, 1998, the NRC closed GL 97-04 for BFN Units 2 and 3 (Reference 6).

On September 4, 1998, TVA submitted a request to change the BFN Units 2 and 3 license basis to permit the use of available containment overpressure for ECCS pump NPSH (Reference 7). On November 25, 1998, in response to a verbal NRC request for additional information, TVA provided (Reference 8):

- The short- and long-term NPSH calculations for the RHR and LPCS pumps;

- Supporting information for these calculations;

- An explanation as to how the analysis at pre-power uprate conditions bounds the uprated conditions;

- A rationale for why the analysis assumed a design flow rate for the LPCS pumps when one RHR pump is in a runout condition;

- A discussion of the requested overpressure value; and

- Graphs showing the NPSH required for the RHR and LPCS pumps versus time and available containment pressure.

On September 3, 1999, NRC approved the use of containment overpressure to maintain adequate ECCS pump NPSH on BFN Units 2 and 3 (Reference 9). The NRC approved 3 psi for the short-term and 1 psi for the long-term period from 5,500 to 35,000 seconds (approximately 92 minutes to 9.7 hours).

By letter dated November 15, 1999, the NRC closed Bulletin 96-03 for BFN Units 2 and 3 (Reference 10). That closure acknowledged actions taken by TVA to address the potential for ECCS suction strainer clogging, and acknowledged closure of the containment overpressure issue for BFN Units 2 and 3 with issuance of corresponding amendments on September 3, 1999.

By letter dated May 6, 2004 (Reference 11), TVA submitted its response to NRC Generic Letter 97-04 for BFN Unit 1. In its response, TVA provided a description of the BFN Unit 1 ECCS pump NPSH analyses performed, key assumptions used, and the ECCS NPSH requirements assuming operations at Extended Power Uprate (EPU) conditions, and modification to the ECCS suction strainers in response to NRC Bulletin 96-03. As stated in Reference 11, BFN Unit 1 requires a credit of 3 psig of containment overpressure to ensure adequate NPSH.

On June 28, 2004 (Reference 12), TVA requested a TS change to allow Unit 1 to operate at extended power uprate conditions. As part of this TS change, TVA requested approval to take credit for containment overpressure in order to provide adequate NPSH to the ECCS pumps. Specifically, TVA requested approval to credit 3 psi containment overpressure for the RHR pumps for the first 10 minutes following a LOCA (short-term requirement) and 3 psi containment overpressure for the LPCS pumps from approximately 4,100 to 52,300 seconds (about 13.4 hours) following a LOCA (long-term requirement).

REFERENCES

1. NRC letter, NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," May 6, 1996.

2. TVA letter, T.E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - NRC Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction (ECCS) Strainers by Debris in Boiling Water Reactors (TAC Nos. M96135, M96136, M96137)," July 25, 1997.

3. TVA letter, T.E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) Unit 2 - NRC Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors (TAC No. M96136)," August 25, 1997.

4. NRC Letter, J. W. Roe to All Licensees, NRC Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," October 7, 1997.

5. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Response To NRC Generic Letter (GL) 97-04, Assurance of Sufficient Net Positive Suction Head (NPSH) for Emergency Core Cooling and Containment Heat Removal Pumps," January 5, 1998.

6. NRC letter, A. W. De Agazio to O. J. Zeringue, "Browns Ferry Nuclear Plant, Units 2 And 3-Completion of Licensing Action For Generic Letter 97-04 (TAC NOS. M99964 AND M99965)," June 11, 1998.

7. TVA letter, T.E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - License Amendment Regarding Use of Containment Overpressure for Emergency Core Cooling System (ECCS) Pump Net Positive Suction Head (NPSH) Analyses," September 4, 1998.

8. TVA letter, T.E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Response to Request for Additional Information (RAI) Relating to Units 2 and 3 License Amendment Regarding Use of Containment Overpressure for Emergency Core Cooling System (ECCS) Pump Net Positive Suction Head (NPSH) Analyses," November 25, 1998.

9. NRC letter, W.O. Long to TVA, "Browns Ferry Nuclear Plants, Units 2 and 3 - Issuance of Amendments Regarding Crediting of Containment Overpressure for Net Positive Suction Head Calculations for Emergency Core Cooling Pumps (TAC Nos. MA3492 and MA3493)," September 3, 1999.

10. NRC letter, W. O. Long to J. A. Scalice, "Browns Ferry Nuclear Plant Units 2 and 3, Completion of Licensing Actions for Bulletin 96-06, 'Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,' Dated May 6, 1996 (TAC NOS. M96135, M96136 and M96137)," November 15, 1999.

11. TVA Letter, "Browns Ferry Nuclear Plant (BFN) Unit 1 - Response to NRC Generic Letter (GL) 97-04, Assurance of Sufficient Net Positive Suction Head (NPSH) For Emergency Core Cooling And Containment Heat Removal Pumps," May 6, 2004.

12. TVA letter, "Browns Ferry Nuclear Plant (BFN) - Unit 1- Proposed Technical Specifications (TS) Change TS-431 - Request for License Amendment - Extended Power Uprate (EPU) Operation," dated June 28, 2004.

ENCLOSURE 3
TENNESSEE VALLEY AUTHORITY (TVA)
BROWNS FERRY NUCLEAR PLANT UNIT 1
DETAILED DESCRIPTION OF PLANT SYSTEMS
RELATED TO THE NPSH ANALYSIS

Each BFN unit employs a pressure suppression containment system which houses the reactor vessel, the reactor coolant recirculation loops, and other branch connections of the Reactor Primary System. The pressure suppression system consists of a drywell, a pressure suppression chamber (alternatively referred to as the torus or wetwell) which stores a large volume of water, a connecting vent system between the drywell and the suppression chamber, isolation valves, containment cooling systems, equipment for establishing and maintaining a pressure differential between the drywell and pressure suppression chamber, and other service equipment.

The drywell is a steel pressure vessel with a spherical lower portion 67 feet in diameter, and a cylindrical upper portion 38 feet 6 inches in diameter. The overall height is approximately 115 feet. In the event of a process system piping failure within the drywell, reactor water and steam would be released into the drywell air space. The resulting increased drywell pressure would then force a mixture of air, steam, and water through the vents into the pool of water which is stored in the suppression chamber. The steam would condense rapidly and completely in the suppression chamber, resulting in rapid pressure reduction in the drywell. Air that is transferred to the suppression chamber pressurizes the chamber and is subsequently vented to the drywell to equalize the pressure between the two vessels.

The pressure suppression chamber is a steal pressure vessel in the shape of a torus below and encircling the drywell, with a centerline diameter of approximately 111 feet and a cross-sectional diameter of 31 feet. Large vent pipes form a connection between the drywell and the pressure suppression chamber. A total of eight circular vent pipes are provided, each having a diameter of 6.75 feet.

A 30-inch diameter Emergency Core Cooling System (ECCS) suction header circumscribes the suppression chamber. Four 30-inch diameter tees are used to connect the suction header to the suppression chamber. Four strainers on connecting lines between the suction header and the suppression chamber have been provided. The suction lines from the Residual Heat Removal (RHR), High Pressure Coolant Injection (HPCI), Low Pressure Core Spray (LPCS), and Reactor Core Isolation Cooling (RCIC) systems are supplied from this header. The four strainers are not individually associated with separate pump suctions but direct suppression pool water to the common ECCS ring header. Therefore interaction between operating pumps are considered when determining suction losses. The normal suction path for the HPCI and RCIC system pumps is the Condensate Storage Tank. Figure 1 provides a general overview of the primary containment.

As shown in Figure 2, the BFN ECCS consists of the following:

- HPCI;
- Automatic Depressurization System (ADS);
- LPCS; and
- Low Pressure Coolant Injection (LPCI), which is an operating mode of RHR.

The ECCS subsystems are designed to limit clad temperature over the complete spectrum of possible break sizes in the nuclear system process barrier, including the design basis break. The design basis break is defined as the complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel (i.e., one of the recirculation loop pipes) with displacement of the ends so that blowdown occurs from both ends.

The low-pressure ECCS consists of LPCS and LPCI. The LPCS consists of two independent loops. Each loop consists of two pumps, a spray sparger inside the core shroud and above the core, piping and valves to convey water from the pressure suppression pool to the sparger, and the associated controls and instrumentation. When the system is actuated, water is taken from the pressure suppression pool. Flow then passes through a normally open motor-operated valve in the suction line to each 50 percent capacity pump.

The RHR System is designed for five modes of operation (i.e., shutdown cooling; containment spray and suppression pool cooling; LPCI; standby cooling; and supplemental fuel pool cooling). During LPCI operation, the four RHR pumps take suction from the pressure suppression pool and discharge to the reactor vessel into the core region through both of the recirculation loops. Two pumps discharge to each recirculation loop.

An important consideration in the operation of the LPCS and RHR pumps is the available net positive suction head (NPSH). Adequate available NPSH is important in ensuring that the pump will deliver the flow assumed in the safety analyses at the expected discharge pressure. In order to ensure acceptable flow and discharge pressure, the available NPSH must be equal to or greater than the required NPSH. The required NPSH is a function of the pump design and is determined by the pump vendor.

The available NPSH is calculated from the equation:

$$\text{Available NPSH} = h_{atm} + h_{static} - h_{loss} - h_{vapor}$$

where:

h_{atm} = head on the surface of the suppression pool

h_{static} = the head due to the difference in elevation between the suppression pool surface and the centerline of the pump suction

h_{loss} = the head loss due to fluid friction, fittings in the flow path from the suppression pool to the pump, and the suction strainers which prevent ingestion of debris into the pumps

h_{vapor} = head due to the vapor pressure of the suppression pool water at the suppression pool water temperature

The increase in power from extended power uprate results in increased decay heat, and a subsequent increase in the suppression pool temperature following the design basis Loss of Coolant Accident. The increased water temperature reduces the available NPSH of the RHR pumps and the LPCS pumps since the vapor pressure of the suppression pool water (or h_{vapor}) increases. The reduction in available NPSH is mitigated, where necessary, by crediting the containment accident pressure, that is, by increasing h_{atm}.

FIGURE 1
GENERAL CONTAINMENT LAYOUT

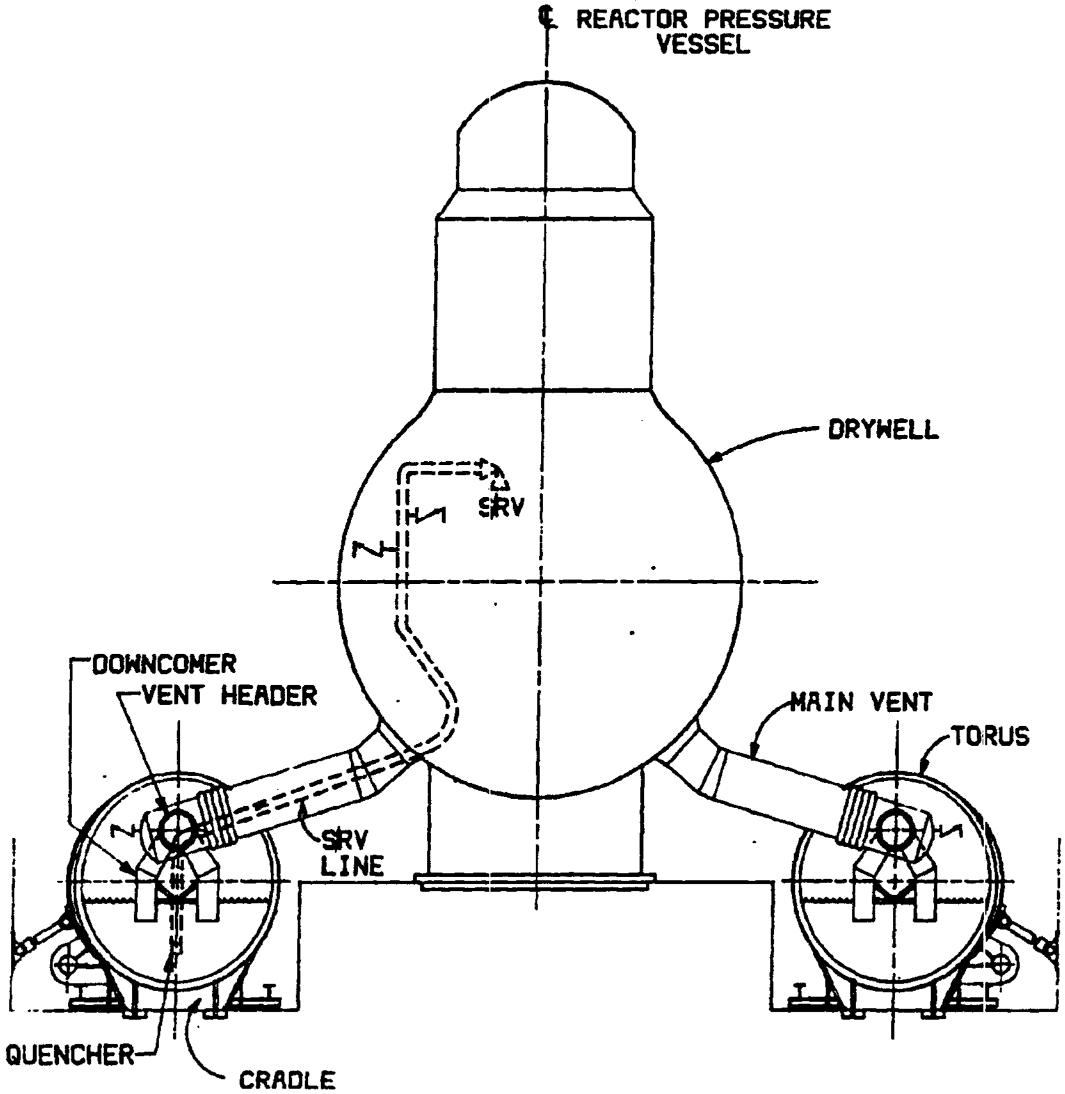
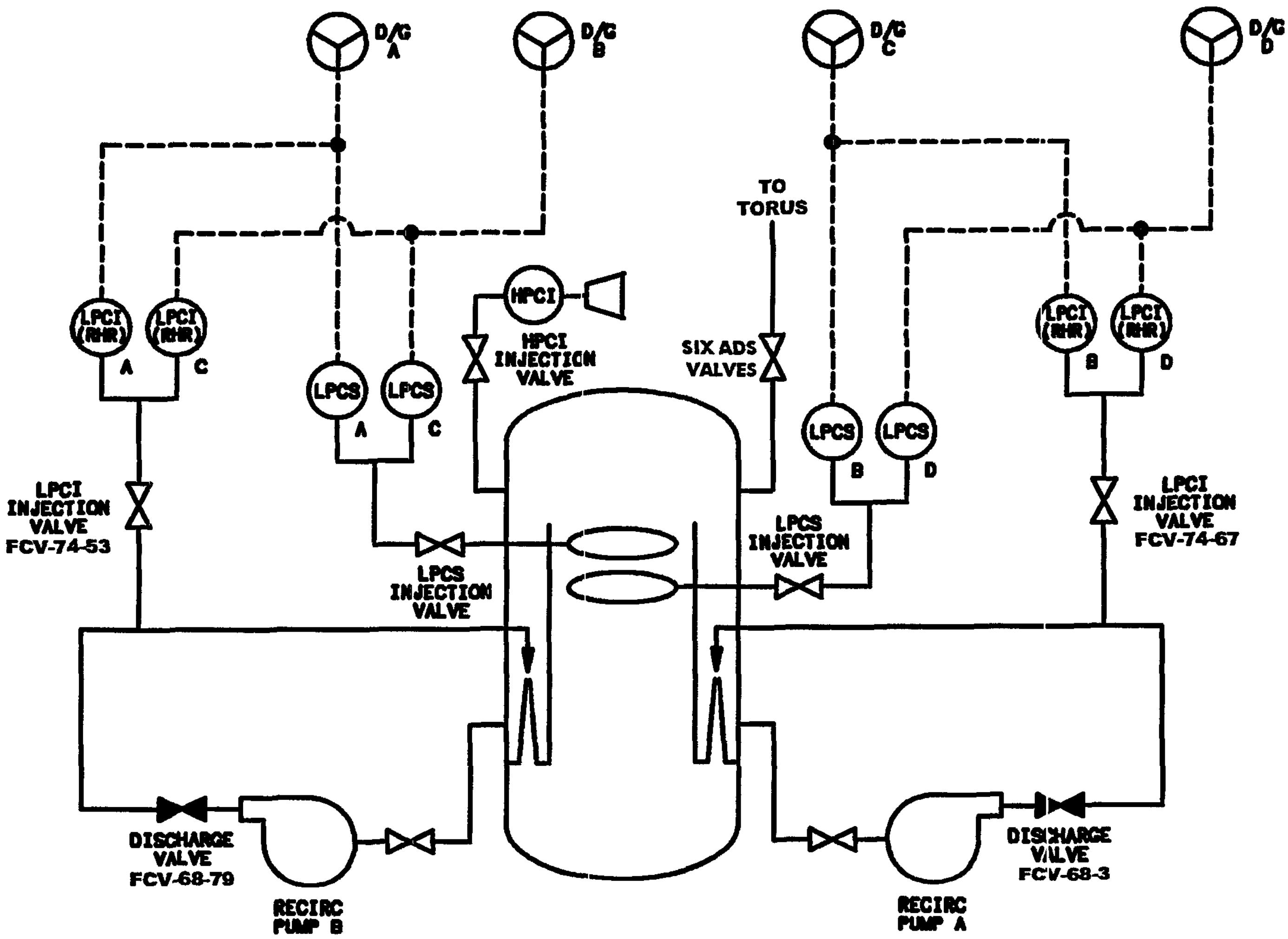


FIGURE 2

LAYOUT OF THE EMERGENCY CORE COOLING SYSTEM



ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY (TVA)

BROWNS FERRY NUCLEAR PLANT UNIT 1

EXTENDED POWER UPRATE

CONTAINMENT OVERPRESSURE CREDIT RISK ASSESSMENT

BFN EPU
Containment Overpressure (COP) Credit Risk Assessment

Performed for:

Tennessee Valley Authority

Performed by:

ERIN Engineering and Research, Inc.

February 27, 2006

Tennessee Valley Authority
Browns Ferry Nuclear (BFN)

BFN EPU
Containment Overpressure (COP)
Credit Risk Assessment

Prepared by: ______________________ Date: Feb. 27, 2006
Vincent M. Andersen

Reviewed by: ______________________ Date: Feb. 27, 2006
David Bidwell

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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 LOCAs.

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 DBA LOCA accident calculations to determine under what conditions credit for COP is required to satisfy low pressure ECCS pump NPSH.

2) Revise all large LOCA 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).

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 (1.53E-09/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 (1.53E-09/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 LOCAs.

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 BFN EPU LAR requests containment overpressure credit for BFN Unit 1 for DBA LLOCA accidents.

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 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 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
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 DBA 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 DBA Calculations

The purpose of this task is to develop an understanding of the BFN EPU design basis LLOCA calculations that result in the need to credit 3 psig containment overpressure credit.

The need for COP credit requests is driven by the conservative nature of design basis accident calculations. The DBA LOCA 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 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 and the following two key water temperature conditions:

- High river water temperature
- High initial torus water temperature

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

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 scenarios due to inadequate NPSH caused by an unisolated containment and other extreme 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.

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

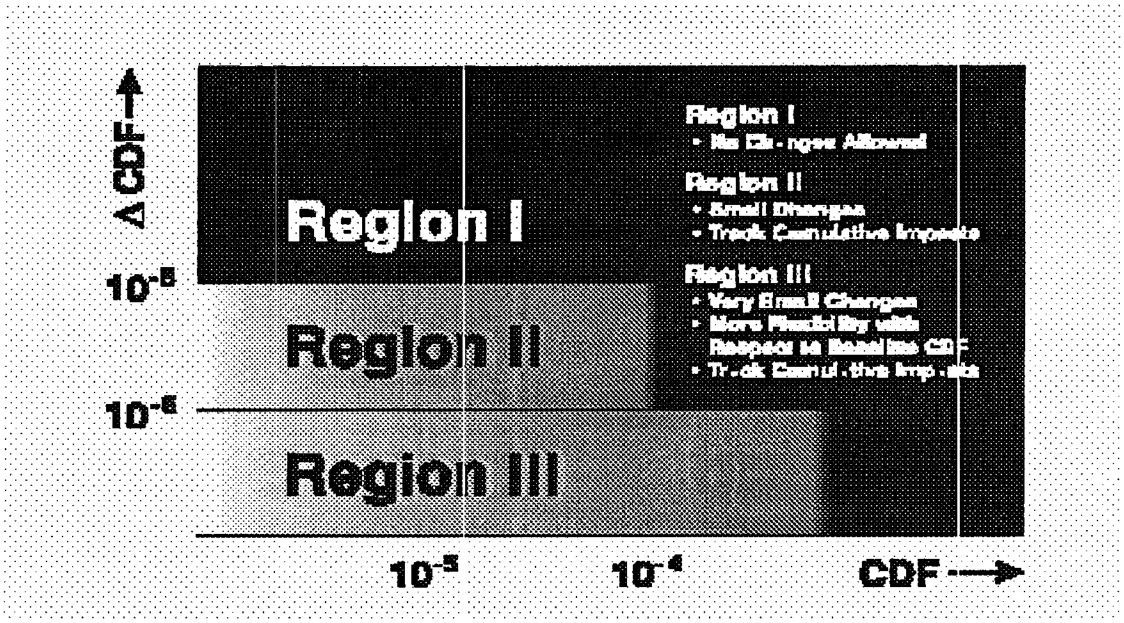
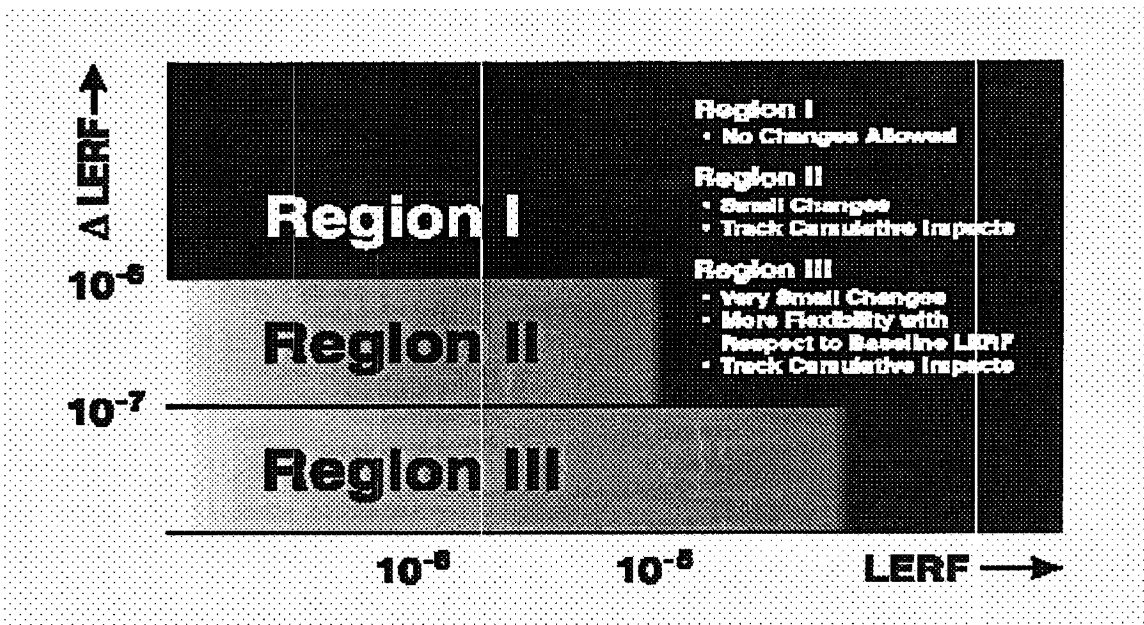


Figure 2-2

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 DBA CALCULATIONS

The purpose of this risk assessment is due to the fact that the conservative nature of design basis 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 shows that no credit for COP is 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
- 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. To simplify the risk assessment, the different combinations of realistic input sensitivities were maintained at a manageable number. Eleven 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 sensitivity cases summarized in Table 3-1, the following general conclusions can be made:

- Initial reactor power, decay heat level, and initial water temperatures are the key determining factors in the analytic conclusions
- COP credit is <u>not</u> required for NPSH, even with the conservative DBA calculation inputs, if 3 or 4 RHR pumps and associated heat exchangers are in operation (refer to Cases 1 and 1a 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, and initial torus water temperature is at the high temperature of 95°F, then river water

temperature must be above 70°F to result in the need for COP credit (refer to Case 2b 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, and initial torus water temperature is taken as 92°F, then river water temperature must be above 86°F to result in the need for COP credit (refer to Case 4c 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 torus water temperature of 95°F, and river water temperature above 70°F
- Plant State 2: 100% EPU initial power level, nominal decay heat , 2 RHR pumps and heat exchangers in SPC, initial torus water temperature of 92°F, and river water temperature above 86°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.

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. 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 and torus water temperatures.

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 does not 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 70°F is determined from the BFN experience data statistical analysis summarized in Appendix C.
- If the above conditions are satisfied, it assumed that the torus water temperature is 95°F, with a probability of 1.0 (this conservative assumption is to simplify the analysis).

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 exceeding 70°F is 4.0E-1.

Therefore, the probability of being in Plant State 1 is 5E-3 x 0.40 = 2E-3.

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 86°F is determined from the BFN experience data statistical analysis summarized in Appendix C.
- If the above conditions are satisfied, it assumed that the torus water temperature is 92°F, with a probability of 1.0 (this conservative assumption is to simplify the analysis).

As can be seen from Table C-1, the probability of river water temperature exceeding 86°F is 1.4E-1.

Therefore, the probability of being in Plant State 2 is 1.4E-1 x 1.0 = 1.4E-1.

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 35La 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 the 35La pre-existing containment leakage used in this base case analysis is 9.86E-04.

This low likelihood of a significant pre-existing containment leakage path is consistent with BFN primary containment performance experience. Neither BFN nor the BWR industry has experienced a 35La pre-existing containment leakage event. The BFN primary containment performance experience shows BFN containment leakages much less than 35La. 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

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

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 35La. 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.

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, all RPV injection is lost early, and core damage occurs at approximately one hour. 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 35La 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 ***U1COP2-9*** 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 COP DETERMINISTIC CALCULATIONS

Case	Case Description	Initial Power	Decay Heat	SW Temp (F)	SP Initial Temp (F)	Number of RHR pumps in Operation	RHR and CS Pump Flow Rate Per Pump	Number of RHR HX in Operation	Number of RHRSW pumps in Operation	RHRSW Pump Flow Rate Per Pump (gpm)	RHR Heat Exchanger K Value	Core Spray Pumps in Operation	Initial SP Water Volume	ECCS Strainer Debris Loading	Credit for Containment Heat Sinks	Peak SP Temp (F)	COP Credit Required
Base Case	EPU Licensing Calculation – DBA LOCA	102% EPU	ANSI 5.1 w/2σ	95	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	187.3	Yes
Case 1	No Single Failure	102% EPU	ANSI 5.1 w/2σ	95	95	4	Full design	4	4	4000	223	4	Minimum	Yes	No	166.4	No
Case 1a	3 Pumps in SPC	102% EPU	ANSI 5.1 w/2σ	95	95	3	Full design	3	3	4000	223	4	Minimum	Yes	No	175.0	No
Case 2	DBA Calculation but SW Temperature = 85F	102% EPU	ANSI 5.1 w/2σ	85	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	182.0	Yes
Case 2a	DBA Calculation but SW Temperature = 75F	102% EPU	ANSI 5.1 w/2σ	75	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	177.6	Yes
Case 2b	DBA Calculation but SW Temperature = 70F	102% EPU	ANSI 5.1 w/2σ	70	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	175.9	No
Case 2c	DBA Calculation but SW Temperature = 65F	102% EPU	ANSI 5.1 w/2σ	65	95	2	Full design	2	2	4000	223	2	Minimum	Yes	No	174.3	No
Case 3	DBA Calculation but SP Temperature = 85F	102% EPU	ANSI 5.1 w/2σ	95	85	2	Full design	2	2	4000	223	2	Minimum	Yes	No	183.8	Yes
Case 4	100% Initial Power, Minimum SP Level, and No Heat Sink Credit	100% EPU	ANSI 5.1 w/o 2σ	92	92	2	Full design	2	2	4000	241	2	Minimum	Yes	No	177.0	Yes
Case 4a	100% Initial Power, Nominal SP Level, and Heat Sink Credit	100% EPU	ANSI 5.1 w/o 2σ	92	92	2	Full design	2	2	4000	241	2	Nominal	Yes	Yes	174.7	No

Table 3-1

SUMMARY OF COP DETERMINISTIC CALCULATIONS

Case	Case Description	Initial Power	Decay Heat	SW Temp (F)	SP Initial Temp (F)	Number of RHR pumps in Operation	RHR and CS Pump Flow Rate Per Pump	Number of RHR HX in Operation	Number of RHRSW pumps in Operation	RHRSW Pump Flow Rate Per Pump (gpm)	RHR Heat Exchanger K Value	Core Spray Pumps in Operation	Initial SP Water Volume	ECCS Strainer Debris Loading	Credit for Containment Heat Sinks	Peak SP Temp (F)	COP Credit Required
Case 4b	100% Initial Power, Minimum SP Level, and Heat Sink Credit	100% EPU	ANSI 5.1 w/o 2σ	92	92	2	Full design	2	2	4000	225	2	Minimum	Yes	Yes	178.9	Yes
Case 4c	100% Initial Power, Minimum SP Level, Heat Sink Credit, and SW Temp. that results in Peak SP Temp. equal to/less than 176F	100% EPU	ANSI 5.1 w/o 2σ	86	92	2	Full design	2	2	4000	225	2	Minimum	Yes	Yes	175.8	No

Section 4

RESULTS

4.1 QUANTITATIVE RESULTS

The results of the base quantification of this risk assessment for the 35 L_a case are as follows:

- deltaCDF: 1.42E-9/yr
- deltaLERF: 1.42E-9/yr

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 assumptive 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 two cases (i.e., the Unit 1 base EPU PRA and the EPU COP Credit base case quantification). The results are shown in Table 4-1. Table 4-1 summarizes the CDF uncertainty distribution statistics for the Unit 1 PRA and for the COP credit base quantification.

As can be seen from the parametric uncertainty results summarized in Table 4-1, even when considering the parametric uncertainty the risk impact is small. The statistics show that CDF has not changed while the distribution of CDF for the COP study has narrowed slightly: the 5%ile increased slightly while the 95%ile decreased slightly.

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.

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 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 35La that would result in defeat of the necessary containment overpressure credit during a DBA LOCA. The following two modeling sensitivity cases are identified to assess the variability of the risk results to the assumed pre-existing containment leakage size:

- A smaller, even more conservative, pre-existing leak size of 20La is assumed in this sensitivity to result in defeat of the necessary COP credit. From EPRI 1009325, the probability of a pre-existing 20La containment leakage pathway is 1.88E-03.
- 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 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 all the split fraction solutions for the containment isolation system fault tree.

<u>Assessment of Power and Water Temperature Pre-conditions</u>

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 (e.g., river water temperature greater than 86°F) exist with a probability of 1.0.

<u>Number of RHR pumps and heat exchangers in SPC</u>

The base case 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 is approximately 8.2E-3. The likelihood of an unisolated containment is approximately 1.4E-3 and the likelihood of other necessary extreme plant conditions (e.g., high river temperature, high reactor power) existing at the time of the LLOCA is approximately 0.14. As such, the base quantification results in an approximate 1.6E-6 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 scenarios with only 1 or no RHR pumps failed. Such scenarios are not explicitly included in the base quantification because their risk contribution is negligible, as shown by the sensitivities discussed here. As shown in Table 3-1, even with design basis conservative assumptions, if 3 or more RHR pumps and heat exchangers are operating in SPC, 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 negligible 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:

- Sum of BFN PRA Large LOCA initiator frequencies: 3.10E-5/yr
- Likelihood of failure of 1 RHR pump or 1 RHR heat exchanger: 1.00E-2 (nominal estimate)
- Probability of 102% EPU initial power level: 5E-3 (same as base analysis)
- Probability of containment isolation failure: 7E-3 (nominal from base analysis)
- Probability of river water temperature >~96°F: 9E-3 (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). 96°F is assumed here as the temperature at which COP credit is required (refer to Case 1a of Table 3-1).
- deltaCDF contribution for 3 RHR pump case: 3.1E-5 x 1E-2 x 5E-3 x 9E-3 = ~1E-13/yr

This additional contribution to the calculated deltaCDF from a 3 RHR pump case is negligible in comparison to the 2 RHR pump case.

An estimate of the deltaCDF risk contribution for the scenario with 4 RHR pumps in operation can be approximated as follows:

- Sum of BFN PRA Large LOCA initiator frequencies: 3.10E-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: 7E-3 (nominal from base analysis)
- Probability of river water temperature >~100°F: 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 temperature at which COP credit is required (refer to Case 1 of Table 3-1).

- deltaCDF contribution for 3 RHR pump case: 3.1E-5 x 1.0 x 5E-3 x 7E-3 x 1E-3 = ~1E-12/yr

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

<u>Summary of Modeling Uncertainty Results</u>

The modeling uncertainty sensitivity cases are summarized in Table 4-2.

4.2.3 <u>Completeness Uncertainty Analysis</u>

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.

Table 4-1

PARAMETRIC UNCERTAINTY ANALYSIS RESULTS

Statistic	BFN Unit 1 Base CDF	COP Risk Assessment CDF
5%	4.71E-7	4.73E-7
50%	1.23E-6	1.21E-6
MEAN	1.77E-6	1.77E-6
95%	4.72E-6	4.69E-6

Table 4-2

SUMMARY OF SENSITIVITY QUANTIFICATIONS

Case	Description	CDF	LERF	ΔCDF	ΔLERF
Base[(1)]	Base Case Quantification	1.77E-06	4.41E-07	1.42E-09	1.42E-09
1[(1)]	Pre-Existing Containment Leakage Sufficient to Fail COP Credit Defined by 100La (probability = 2.47E-4)	1.77E-06	4.41E-07	1.33E-09	1.33E-09
2[(1), (2)]	Pre-Existing Containment Leakage Sufficient to Fail COP Credit Defined by 20La (probability = 1.88E-3)	1.77E-06	4.41E-07	1.53E-09	1.53E-09
3[(1)]	Expansion of Containment Isolation fault tree to Encompass Smaller Lines (approximate by multiplying Cont. Isol. failure probability by 10x)	1.77E-06	4.42E-07	2.05E-09	2.05E-09
4[(1)]	Assume Initial Power Level and Water Temperature Pre-Conditions Exist 100% of the Time	1.77E-06	4.42E-07	2.66E-09	2.66E-09
5[(1)]	Combination of Cases #2, #3 and #4	1.77E-06	4.48E-07	8.33E-09	8.33E-09
6	Incorporation of "3-RHR pumps in SPC" and "4-RHR pumps in SPC" loss of NPSH scenarios	1.77E-06	4.41E-07	1.42E-09	1.42E-09

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) Case 2, 20L_a containment leakage size, is the case used as the basis for the Conclusions of this study (refer to Section 5).

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 quantified to determine the ΔCDF impact. The result for Unit 3 was a deltaCDF of 1.9E-9/yr. 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.

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 LOCAs.

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.

The conclusions of this risk assessment are based on the conservative $20L_a$ assumed containment leakage size (refer to Case 2 of Table 4-2). 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 (1.53E-09/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 (1.53E-09/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 LOCAs is acceptable from a risk perspective.

The general conclusions that the risk impact from the COP credit for DBA LOCAs 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 35La 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 the 35La pre-existing containment leakage used in this base case analysis is 9.86E-04.

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 RIVER WATER AND SP WATER TEMPERATURE VARIATION

The BFN river and torus water temperatures were analyzed to statistically model variability in temperature. The purpose of this data assessment is to estimate for use in the risk assessment the realistic probability that these temperatures will exceed a given value, i.e. the probability of exceedance.

C.1 BFN EXPERIENCE DATA

The following sets of river water inlet and torus water daily temperature data were obtained and reviewed:

Unit	Data Period	Years
2	01/01/00 – 01/31/06	6.1
3	02/01/03 – 01/31/06	3.0

Data for suppression pool water level for the above time periods were also obtained. However, statistical assessment of the variation in pool level was not pursued as the small variation in pool level has a non-significant impact on the COP / NPSH calculations.

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.

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).

Figure C-1

CHRONOLOGICAL VARIABILITY IN RIVER WATER AND TORUS WATER TEMPERATURES

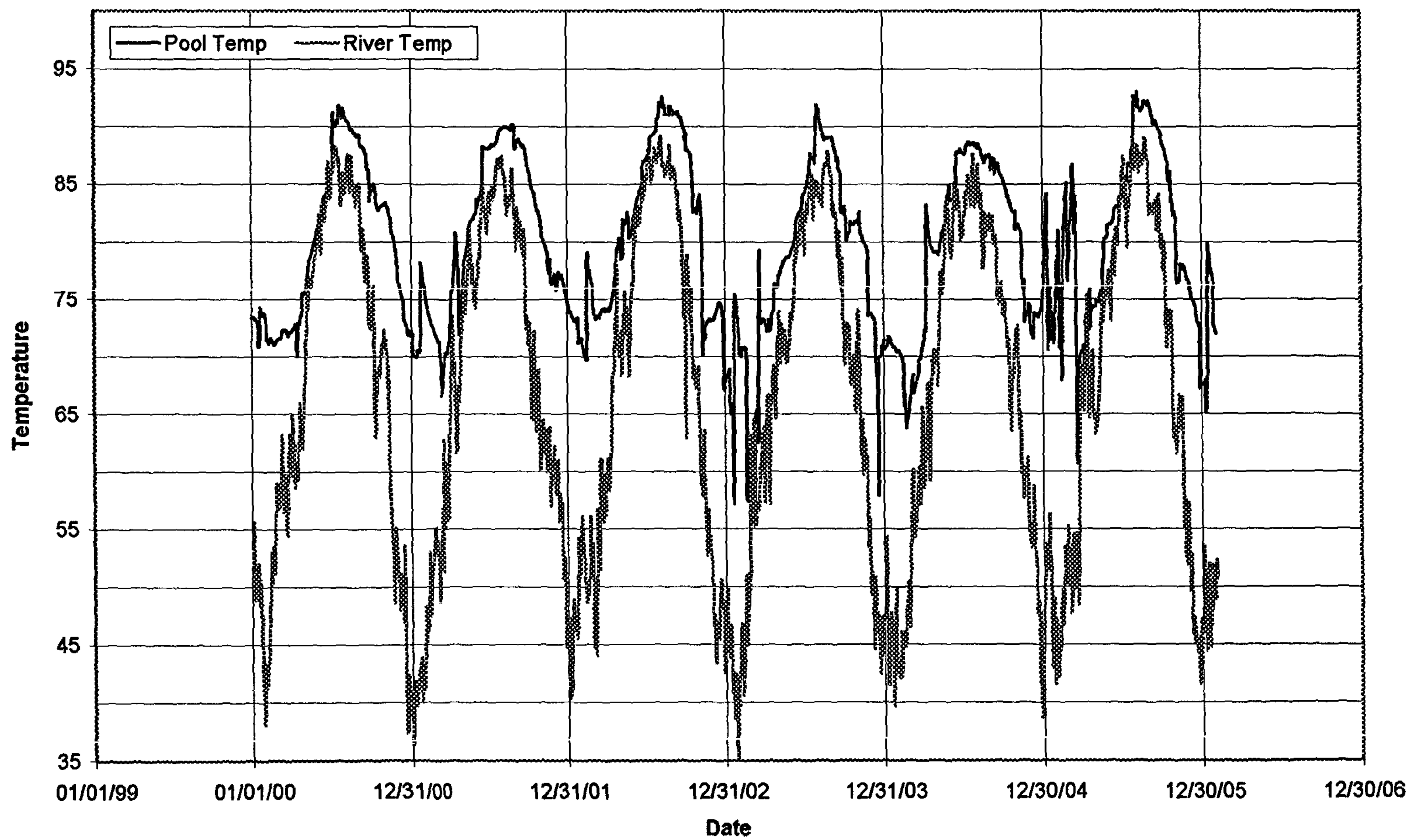


Figure C-2

RIVER WATER TEMPERATURE HISTOGRAM

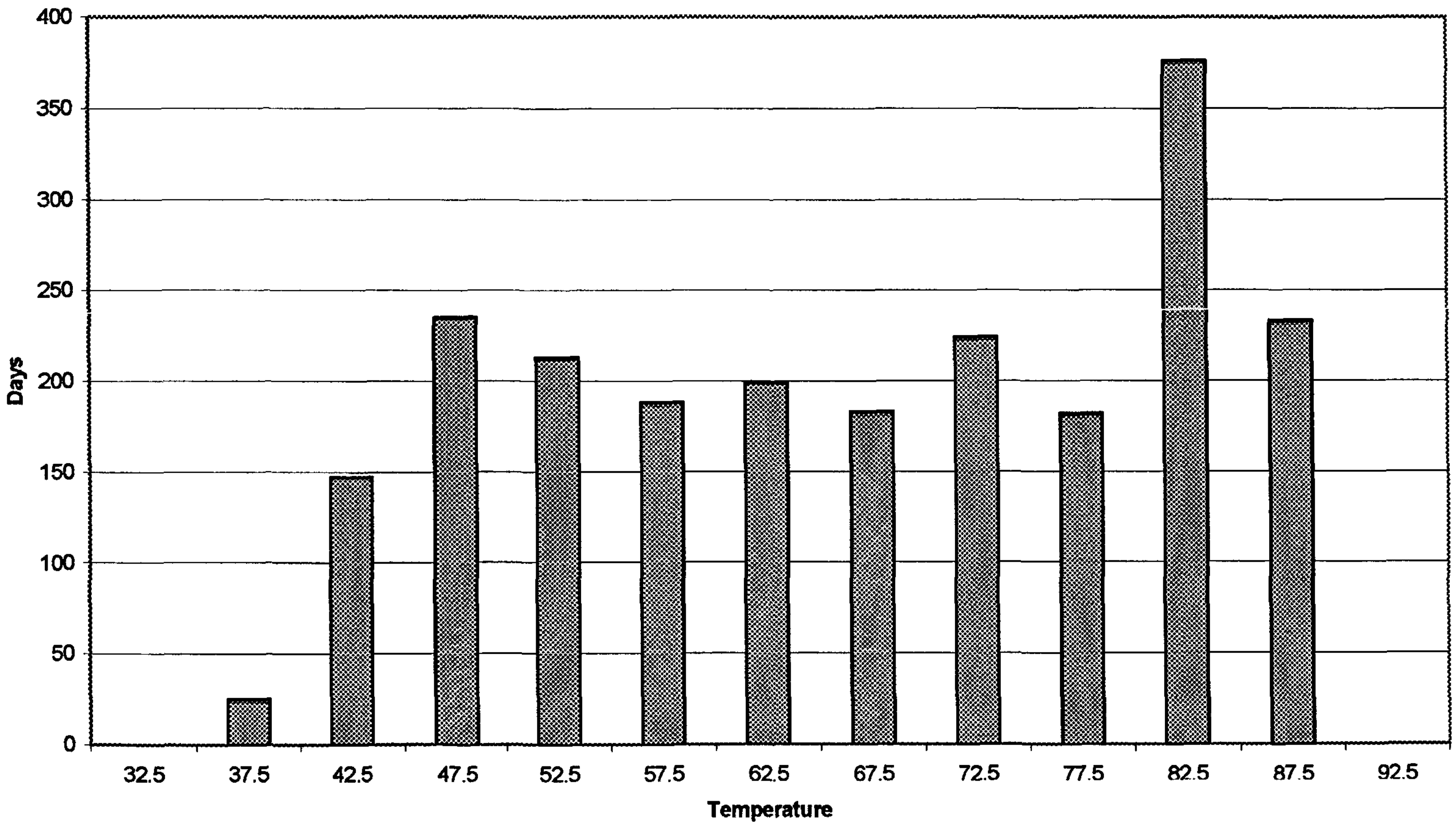


Figure C-3

TORUS TEMPERATURE HISTOGRAM

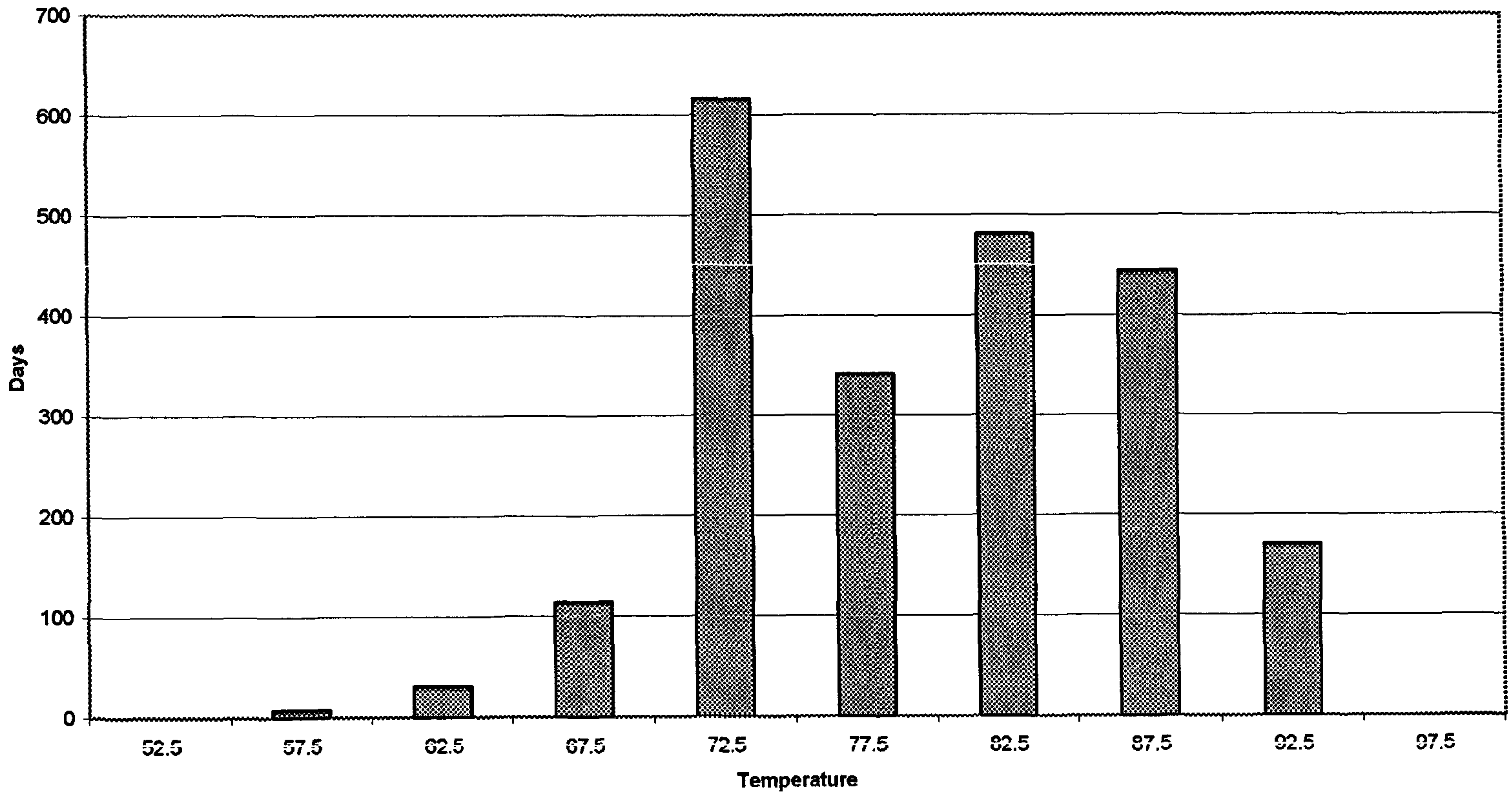


Figure C-4

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: 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

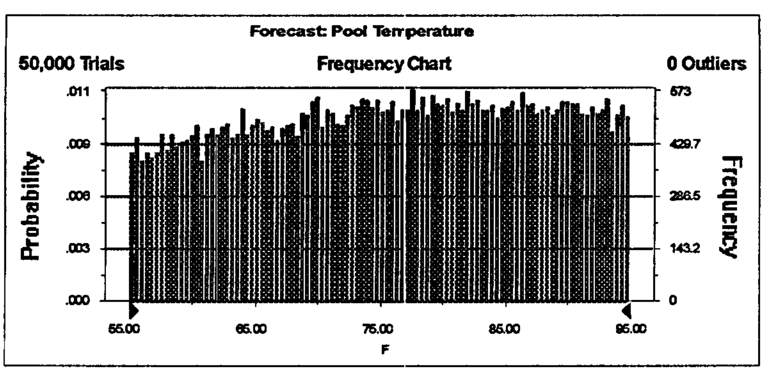


Figure C-5

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: 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

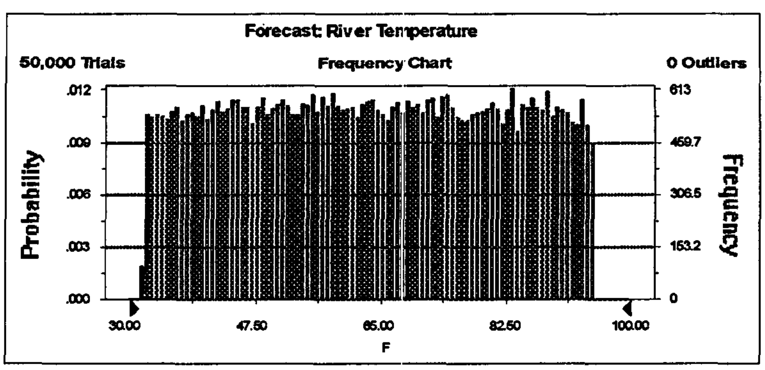


Figure C-6

RIVER WATER TEMPERATURE EXCEEDANCE PROBABILITY

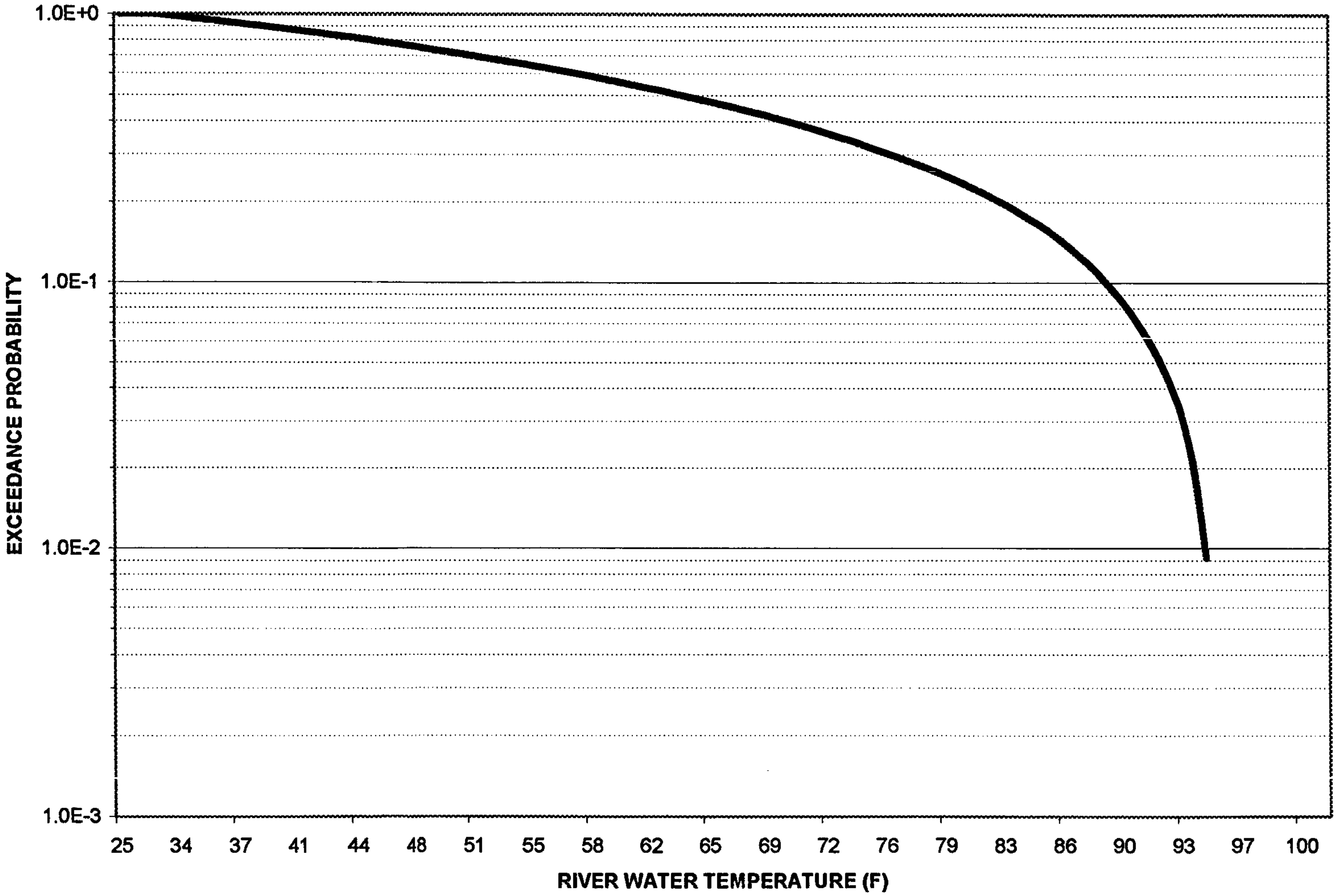


Figure C-7

TORUS WATER TEMPERATURE EXCEEDANCE PROBABILITY

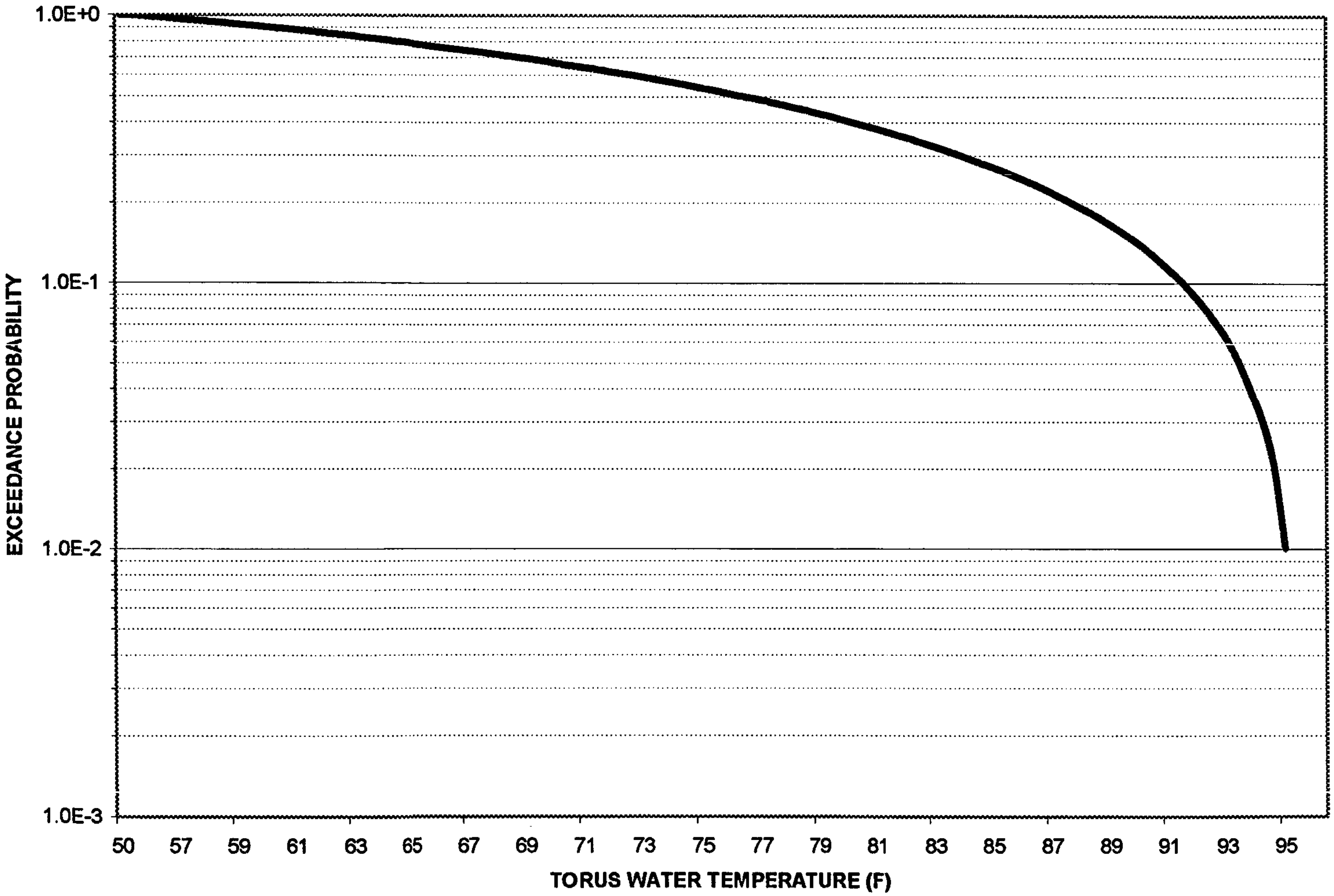


Table C-1

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

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 LLOCA 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 24hrs 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.

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

E.1 MODEL CHANGES

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

The BFN Unit 1 PRA model of record was modified for this risk assessment to question the status of containment integrity first in the Level 1 large LOCA event trees. 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.

The scope of the analysis is limited to large LOCA accidents. 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. 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 penetrations greater than 3 inches, and top event NPSH models the probability of reactor power at 102% as well as river water temperature greater than 86F. Top event NPSH has two split fractions NPSH1 and NPSHS (success, equal to 0.00E+00). The latter is applied for all initiators other than those modeling large LOCAs. The existing CIL fault tree was modified to add the probability of a pre-existing containment leak; a basic event was inserted just under the top 'OR' gate of the CIL fault tree. The basic event is set to different values depending on the size of the leak rate assumed. See Table 4-2 for the sensitivity cases and associated pre-existing leak size. The values used and the resultant CIL split fraction values are listed below:

Sensitivity Case	Leak Size	Leak Probability	CIL Split Fractions[1]
Base	35 La	9.86E-04	1.36E-03
1	100 La	2.47E-04	6.22E-04
2	20 La	1.88E-03	2.25E-03
3	Base CIL split fractions X 10, plus pre-existing leak 35 La	9.86E-04	6.37E-03
4	35 La	9.86E-04	1.36E-03
5	Base CIL split fractions X 10, plus pre-existing leak 20 La	1.88E-03	7.37E-03

Note:
[1]All support split fraction. Degraded state split fraction is also affected but not shown.

Top event NPSH models the probability that the plant is at 102% reactor power with 86F river water, 'OR' the reactor is at the nominal 100% reactor power level with river water greater than 70F. The probability that the plant is at 102% power is modeled using a miscalibration human error probability taken from a similar action documented in the existing BFN Unit 1 PRA Human Reliability Analysis (see event ZHECCL, instrument calibration error, Control Room). The probability that the river water is either greater than 70F or greater than 86F is developed in the data analysis (refer to Appendix C).

Top event NPSH has two split fractions, NPSH1 and NPSHS. The latter is used to filter out sequences where greater than 3 RHR pumps are running. This latter pass-through split fraction is used to exclude the cases where sufficient RHR pumps are cooling the torus such that containment overpressure is not necessary (per DBA calculations) for the success of the RHR and CS pumps. 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 default split fraction. 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).

MODEL Name: U1ERIN

Event Tree: LLCSN.ETI

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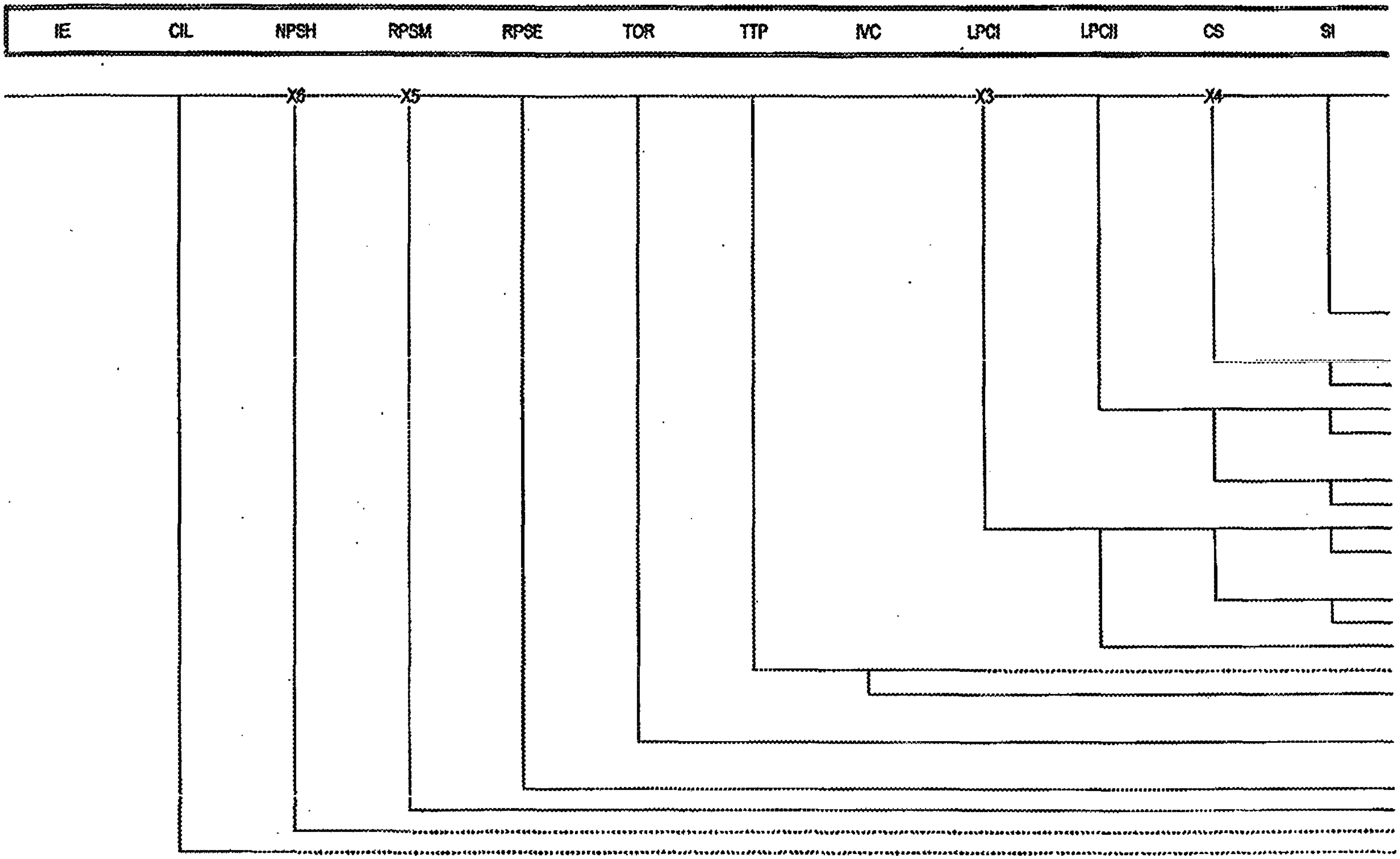

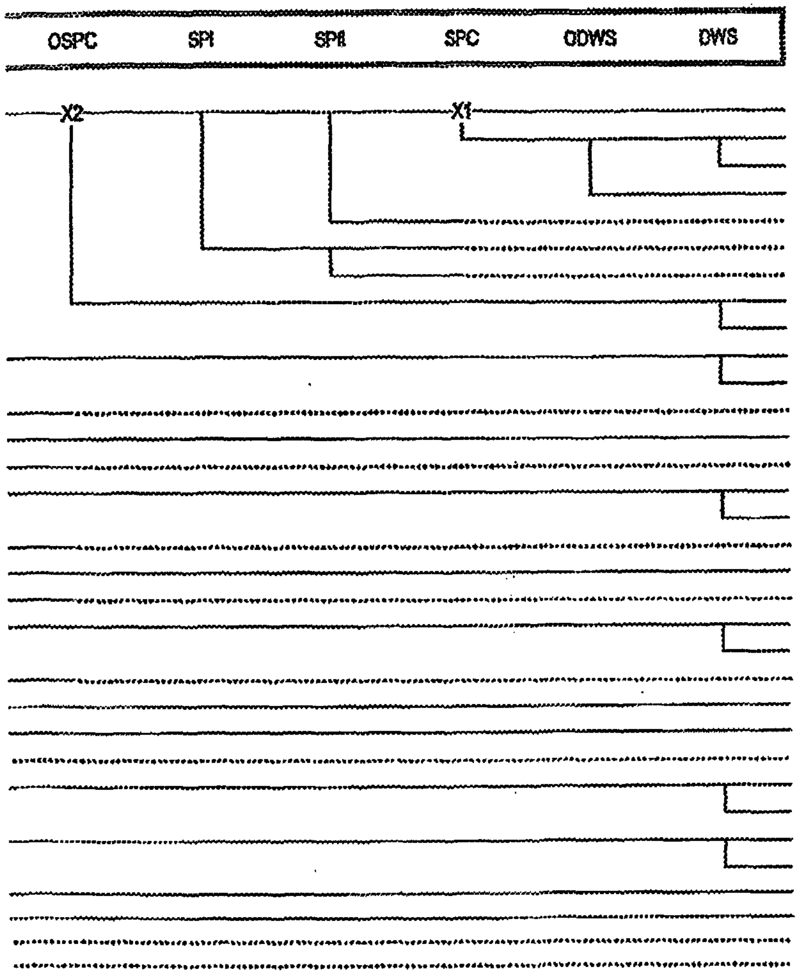

X#	B#	S#
	1	1
	2	2
	3	3
	4	4
X1	5	5-8
X1	6	9-12
X1	7	13-16
	8	17
	9	18
	10	19
	11	20
X2	12	21-38
	13	39
X2	14	40-57
	15	58
	16	59
X2	17	60-77
	18	78
X2	19	79-96
	20	97
	21	98
X2	22	99-116
	23	117
	24	118
X3	25	119-236
	26	237
	27	238
	28	239
	29	240
	30	241
	31	242
X5	32	243-484
X6	33	485-968

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
NPSH3	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
RPSSO	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 SHOULD 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+RA=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)
CSP	1

Model Name: U1COP2-S

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*RPSS=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	FX1=F*FX2=F + (RPA=F*RPC=F +RH=F+NOGB) * (RPB=F*RPD=F+RI=F + NOGD)
DWS1	FX1=S*FX2=S*(RPA=S+RPC=S)*~NOGB*(RPB=S+RPD=S)*~NOGD
DWS2	(RPA=F*RPC=F +RH=F+NOGB+FX1=F) * (RPB=F*RPD=F+RI=F + NOGD+FX2=F)
DWSF	1

Model Name: U1COP2-9

Macro for Event Tree: LLCSN

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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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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=S THIS MACRO IS NEEDED IN THE CETS
HIGH	RPSM=S
HPI	RPSM=S
LOW	INIT=LLCR + INIT=LLCS
LPCIISUP	RE=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=S THIS MACRO IS NEEDED IN THE CETS
NOCD	RPSM=S * TOR=S*(TTP=S+IVC=S)*SI=S*SPC=S
NODC	RPSM=S THIS MACRO IS NEEDED IN THE CETS
NORV	RPSM=S THIS MACRO IS NEEDED IN THE CETS
NOSRV	RPSM=S 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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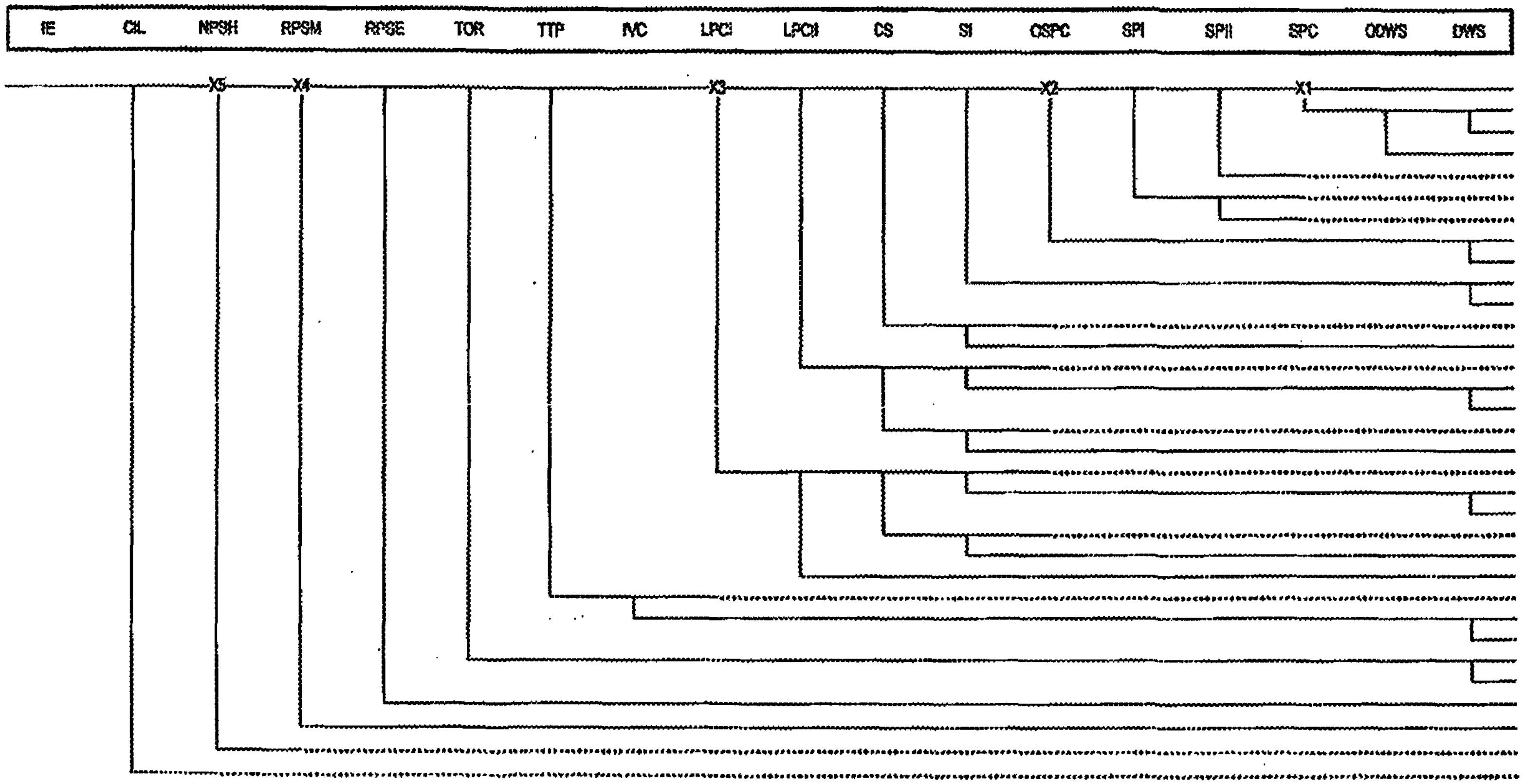
Macro	Macro Rule / Comments
OPDEPL1	RPSM=S THIS MACRO IS NEEDED IN THE CETS
RHRSPCOOL	SPC=S
SORV	RPSM=S LARGE LOCAS ARE ALWAYS DEPRESSURIZED

MODEL Name: U1ERIN

Event Tree: LLON.ETI

13:37:12 February 16, 2006



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	3	3
	4	4
X1	5	5-8
X1	6	9-12
X1	7	13-16
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	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
RPSED	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 SHOULD 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+RD=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	RPSN=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
CDWS1	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=T
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
EMDEPRDWR	RPSM=B THIS MACRO IS NEEDED IN THE CETS
HIGH	RPSM=B
HPI	RPSM=B.
LOW	INIT=LLO
LPCIISUP	RF=S*((NPII=S*DN=S) + LV=S)
LPCISUP	RE=S*((NPI=S*DN=S) + LV=S) LOOP I LPCI SUPPORT
LPI	SI=S
NOACREC	RPSM=B THIS MACRO IS NEEDED IN THE CETS
NOCD	RPSM=B * TOR=S*(TTP=S+IVC=S)*SI=S*BPC=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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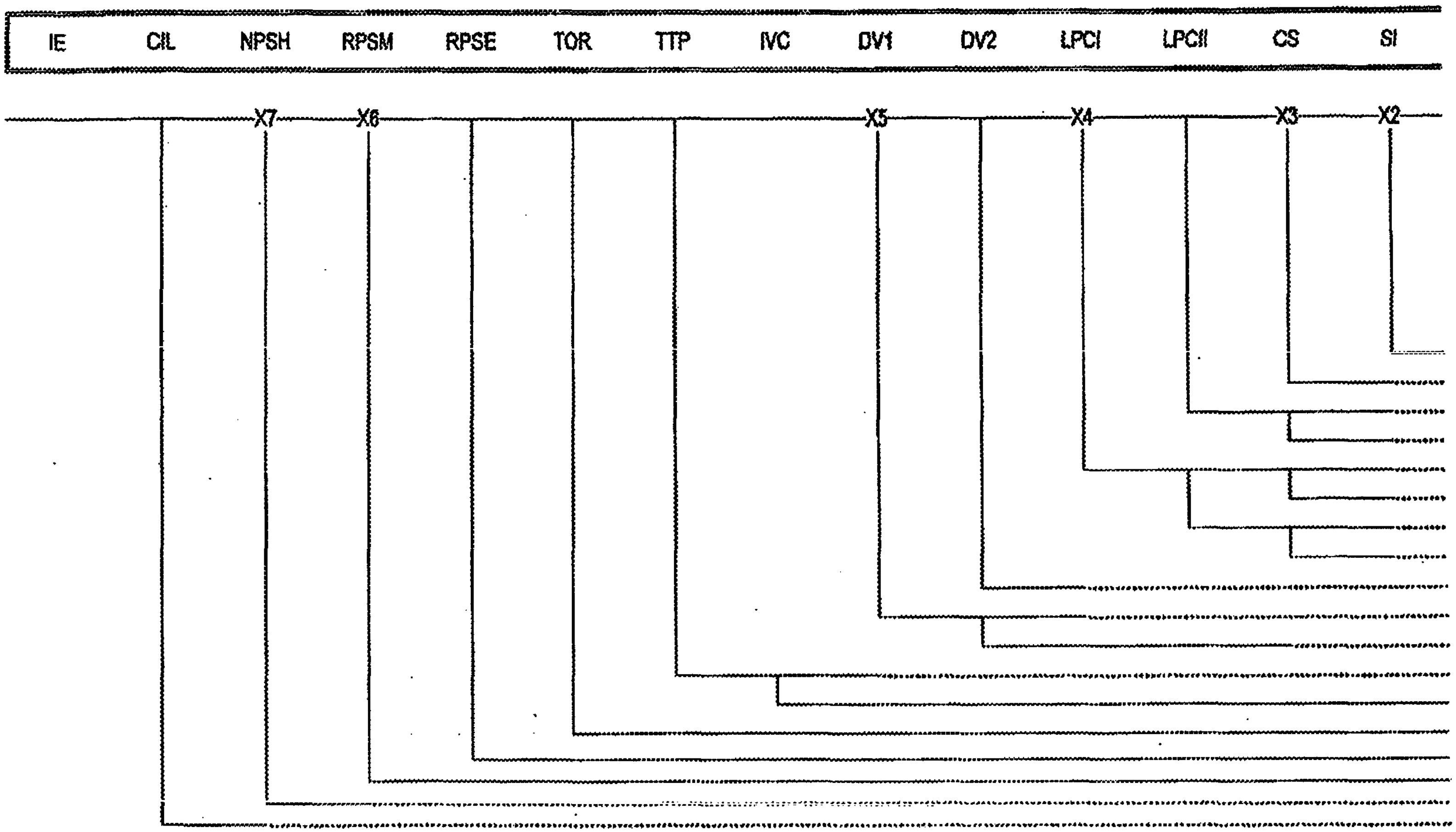
Page 3

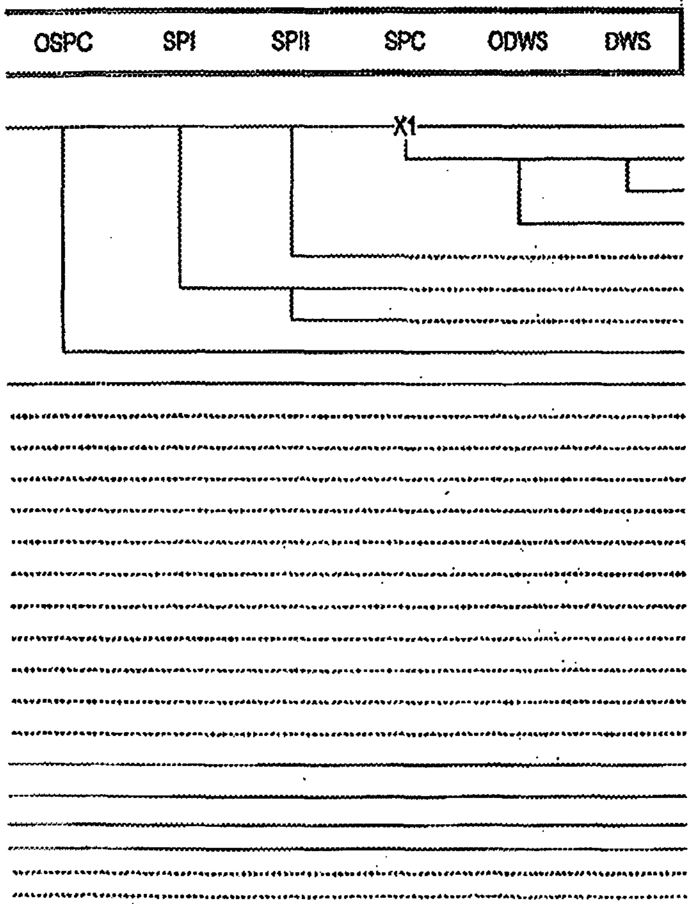
Macro	Macro Rule / Comments
OPDEPL1	RESM=S THIS MACRO IS NEEDED IN THE CETS
RHRSPCOOL	SPC=S
SORV	RPSM=S LARGE LOCAS ARE ALWAYS DEPRESSURIZEED

MODEL Name: U1ERIN

Event Tree: LLRDN.ETI

13:37:46 February 16, 2006





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	3	3
	4	4
X1	5	5-8
X1	6	9-12
X1	7	13-16
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	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: U1COP2-5
Top Events for Event Tree: LLRDN

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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: 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=LLOA + INIT=LLOB + INIT=LLO + INIT=LLSA + INIT=LLSB
NPSHS	1
RPSMG	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+RE=F*RC=F+NH1=F*NH2=F+DW=F*LV=F
DV2S	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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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
CS2S	(RE=F+AA=F+DA=F+AB=F+DC=F+NPI=F+DW=F*LV=F+RC=F+EB=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*RPSB=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

5:08 PM 2/9/2006
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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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
EMDEPRDWR	RPSM=B THIS MACRO IS NEEDED IN THE CETS
HIGH	RPSM=B
HPI	RPSM=B
LOW	INIT=LLDA + INIT=LLDB
LPCIIISUP	RP=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*SPO=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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Page 3

Macro	Macro Rule / Comments
OPDEPL1	RPSM=S 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

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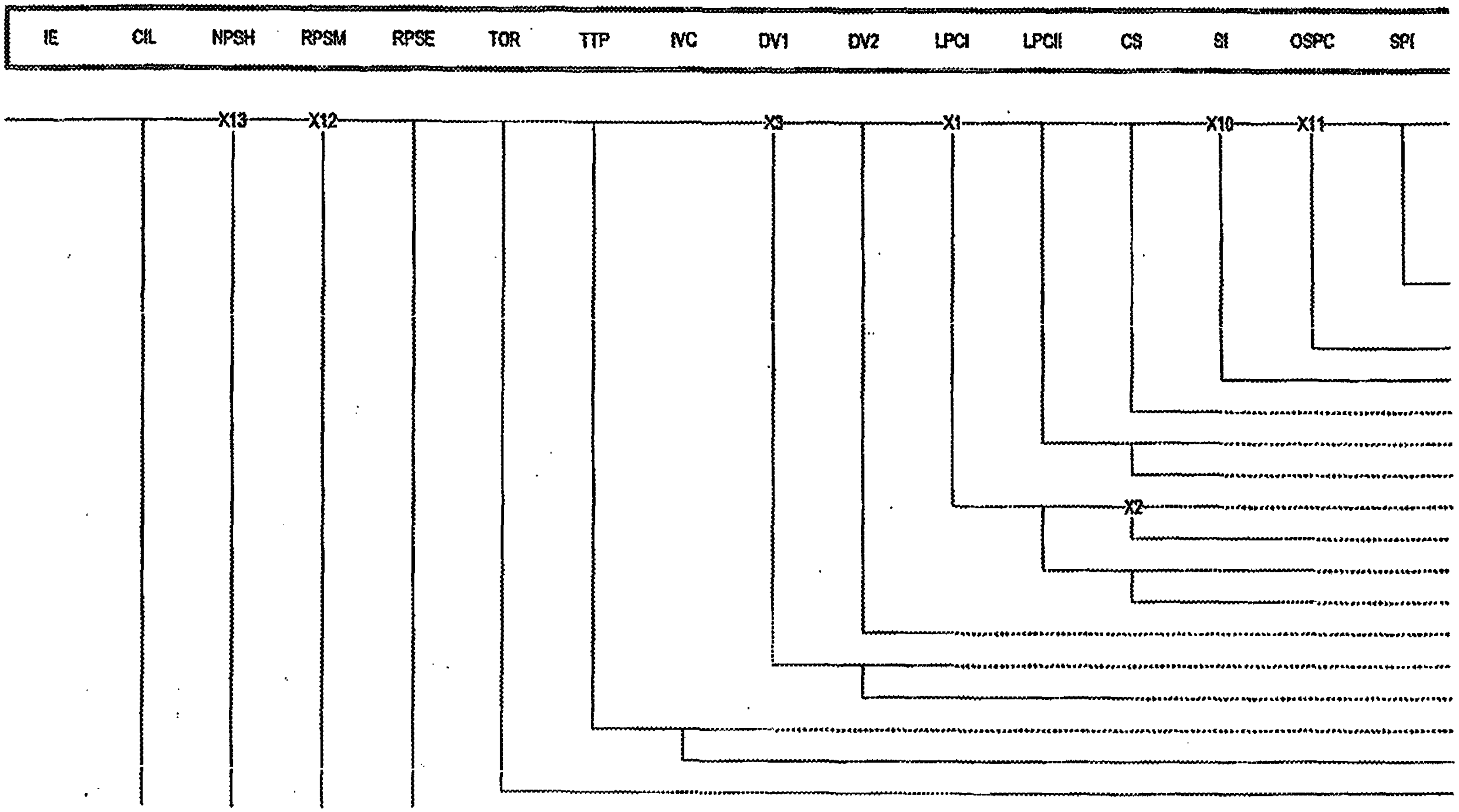
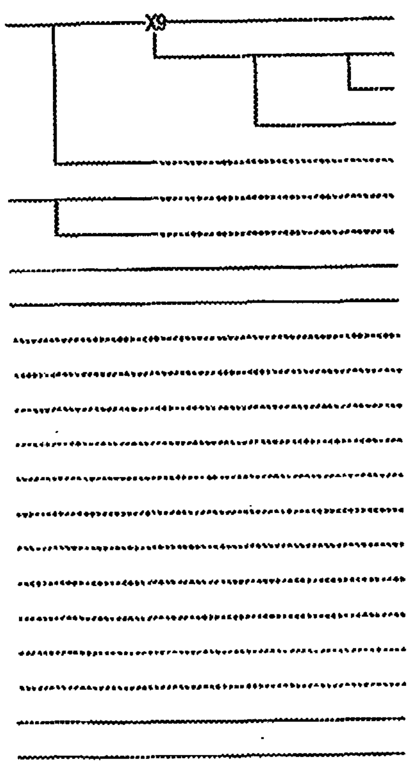
13:38:20 February 16, 2006



MODEL Name: U1ERIN

Event Tree: LLRSN.ETI

SPII	SPC	ODWS	DWS	X#	B#	S#



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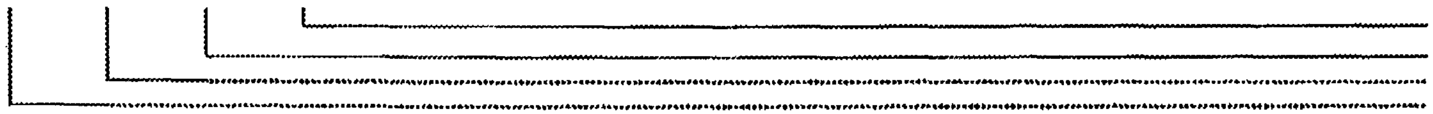
	3	3
	4	4
X9	5	5-8
X9	6	9-12
X9	7	13-16
	8	17
	9	18
X10	10	19-36
X10	11	37-54
X10	12	55-72
X10	13	73-90
X11	14	91-107
X11	15	108-124
X11	16	125-141
X1	17	142-282
X1	18	283-423
X2	19	424-458
X3	20	459-916
	21	917
	22	918

Event Tree: LLRSN.ETI

13:38:20 February 16, 2006

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



MODEL Name: U1ERIN

Event Tree: LLRSN.ETI

13:38:20 February 16, 2006

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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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: LLRSN

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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=LLOA + INIT=LLOB + INIT=LLO + INIT=LLSA + INIT=LLSB
NPSHS	1
RPSMS	1
RPSEC	1
TOR1	1
TTP1	RB5=S*DI=S
TTP2	RB5=S*DI=F
TTP3	RB5=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
DV1P	1
DV2F	RE=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	1
OSPC1	RPSM=S*RPSS=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 +RE=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 +RE=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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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: U1COPZ-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	DNS=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*DN=S) + LV=S)
LPCISUP	RE=S*((NPI=S*DN=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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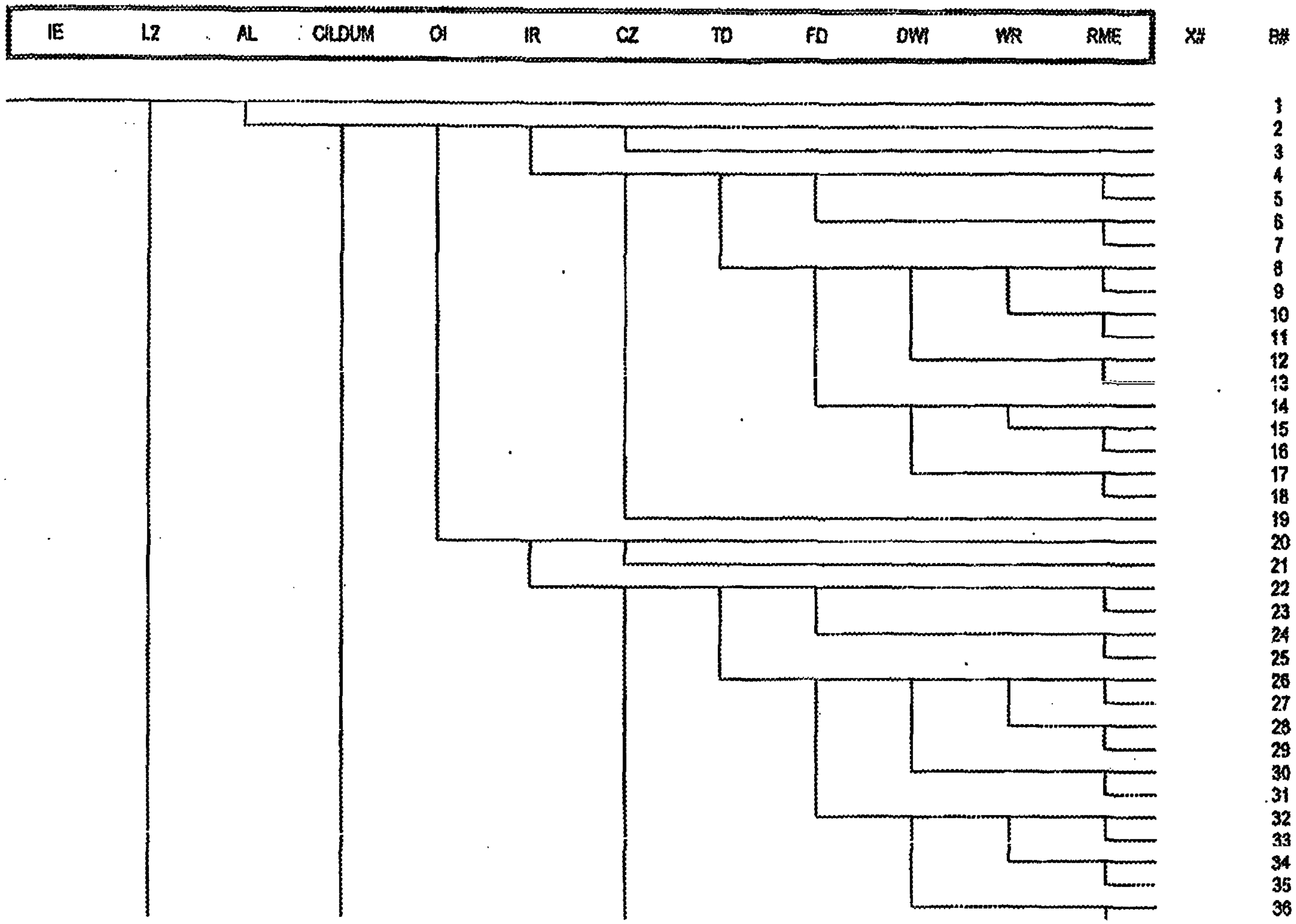
Macro	Macro Rule / Comments
NPSHLOST	CIL=F*NPSH=F
OPDEPL1	RPSM=S THIS MACRO IS NEEDED IN THE CETS
RHRSPCOOL	SPC=S
BORV	RPSM=S LARGE LOCAS ARE ALWAYS DEPRESSURIZEED

MODEL Name: U1ERIN

Event Tree: CETN1.ETI

13:38:50 February 16, 2006



MODEL Name: U1ERIN

Event Tree: CETN1.ETI

13:36:50 February 16, 2006

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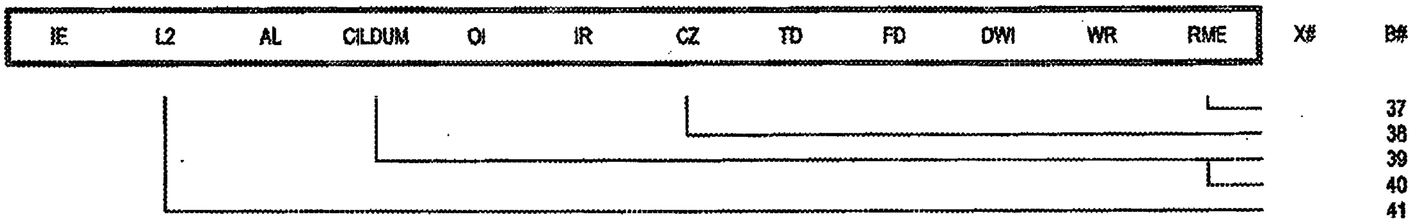
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L2
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CILDUM
OI
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CZ
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Model Name: U1COP2-9

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
RNE	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
CSF	1
TD1	CLASS1E
TD2	OI=S*DWSPRAY
TD3	-(OI=S)*CLASS1BE
TD4	-(OI=S)*CLASS1BL
TDB	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
ALU	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
ALC	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
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
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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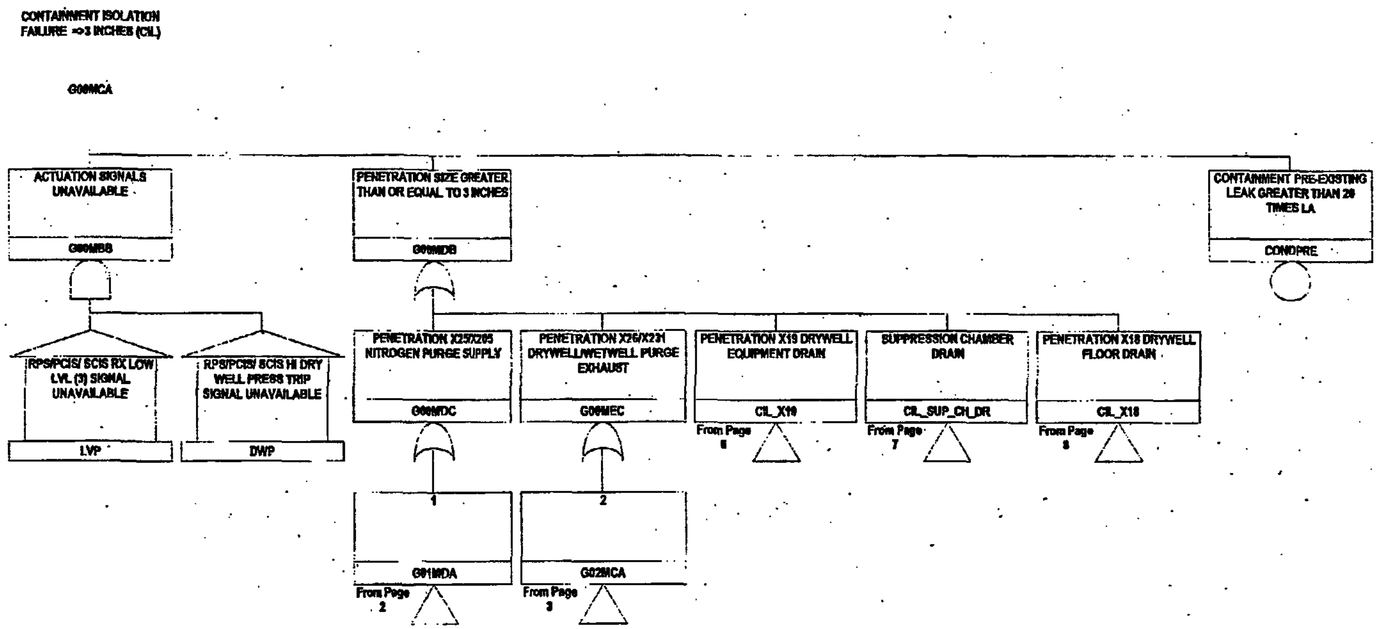
Macro	Macro Rule / Comments
C1C3LERF	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)
	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)
	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)
	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)
	CZ=F + RME=F*(CILFAIL+DWI=F+IR=F*TD=S*FD=S)

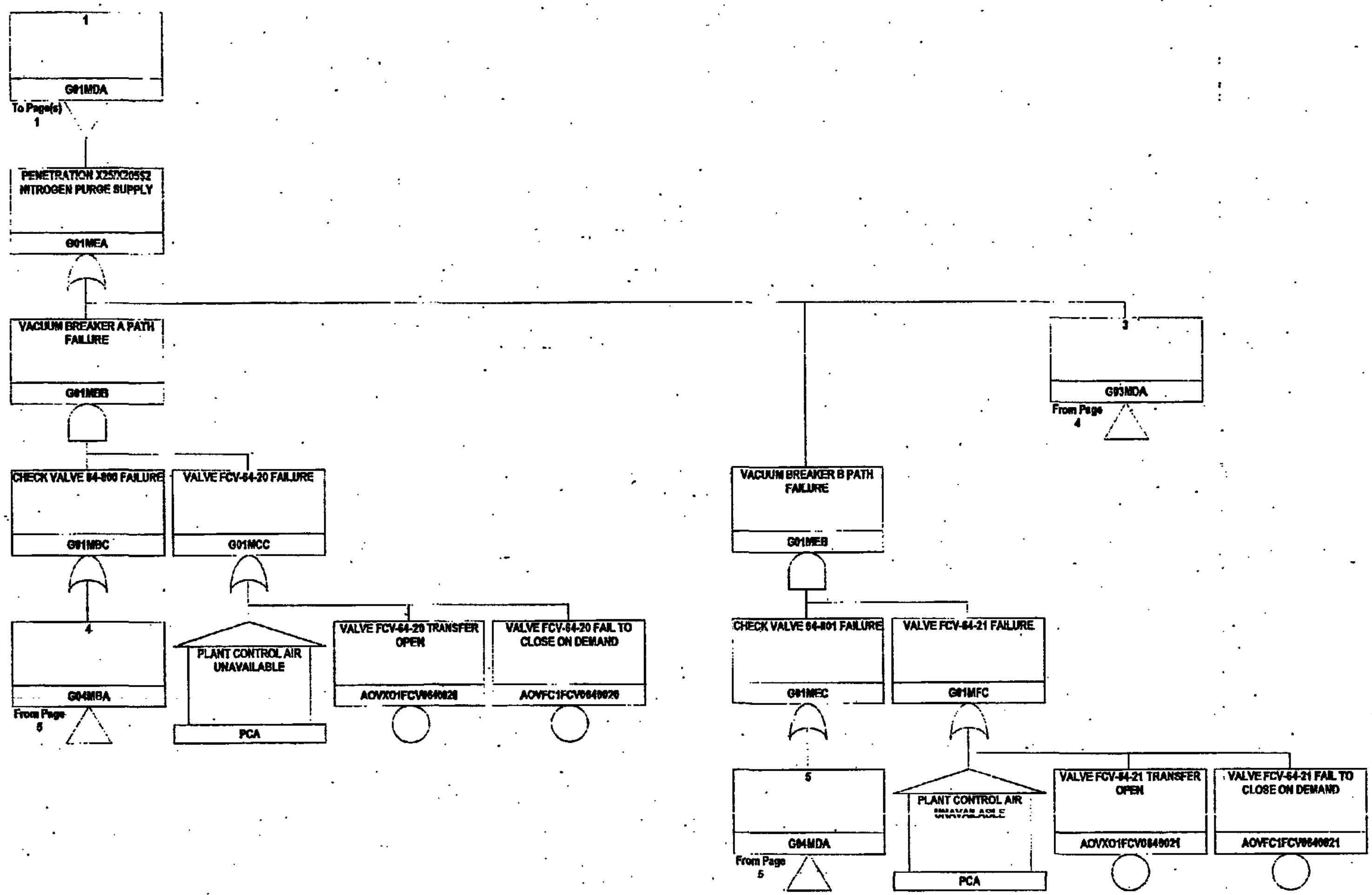
Appendix F
FAULT TREES

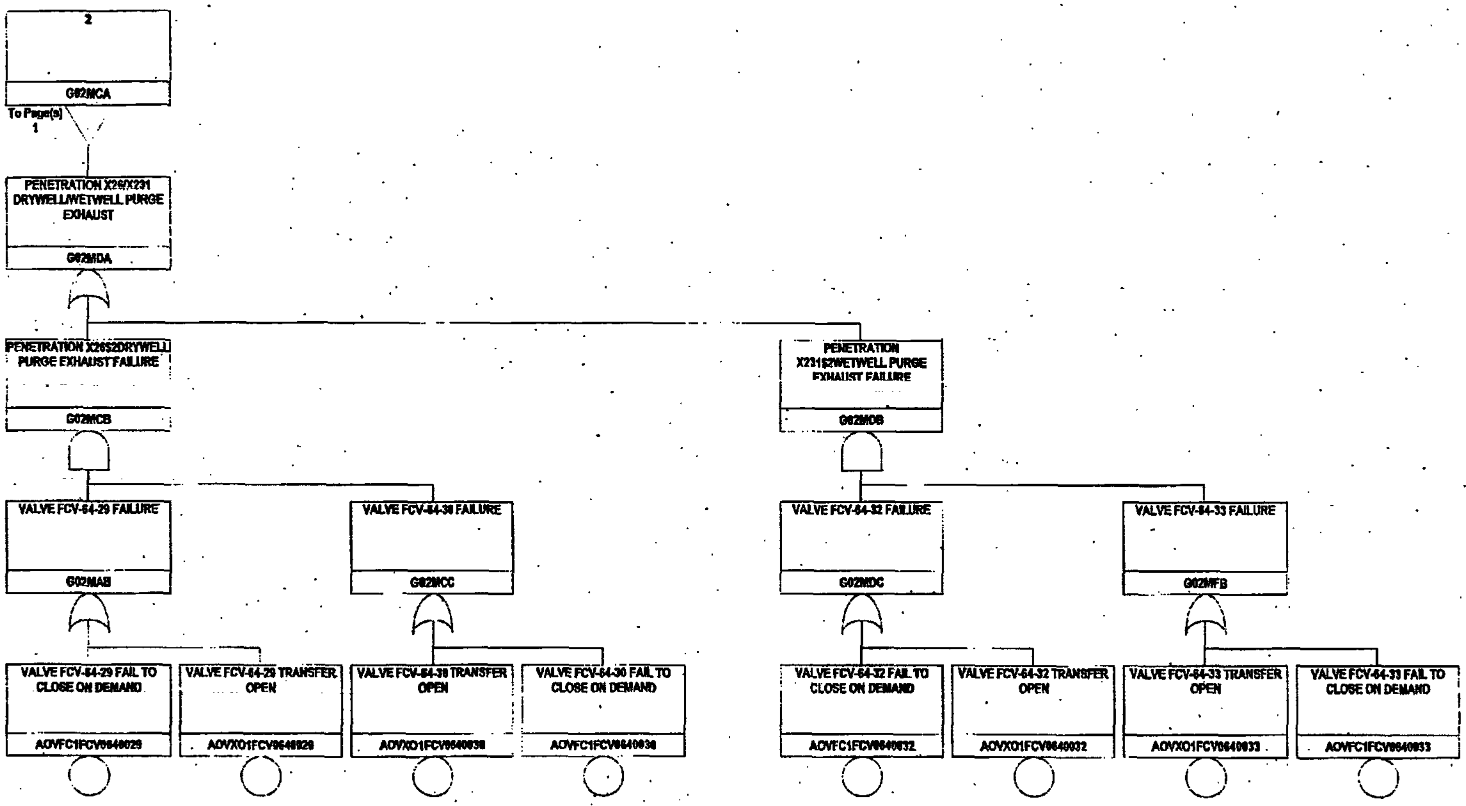
This appendix provides print-outs of the BFN Unit 1 PRA modified containment isolation fault tree and the NPSH fault tree used in this analysis.

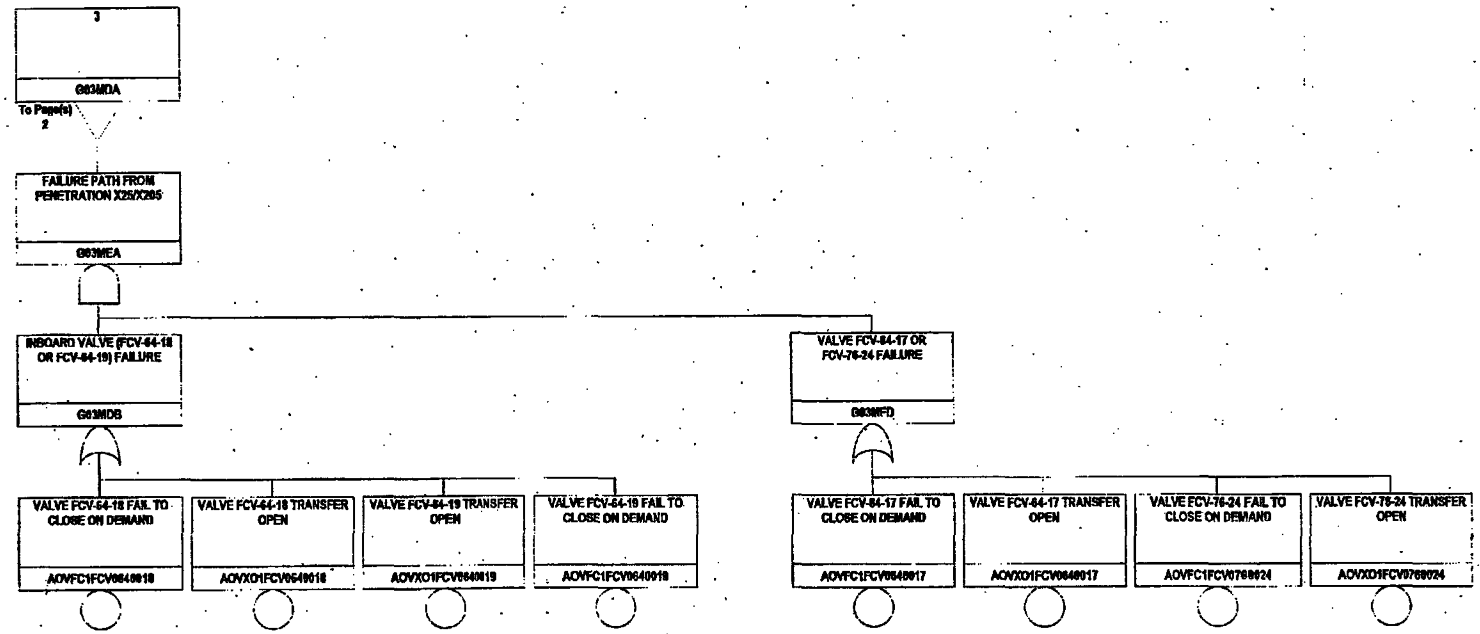


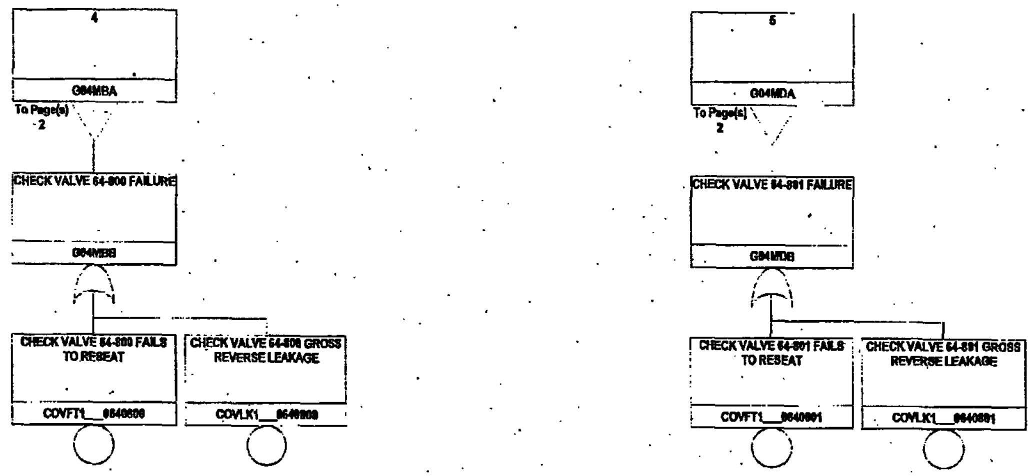
CONTAINMENT ISOLATION FAILURE =>3 INCHES (CIL)
G00MCA
ACTUATION SIGNALS UNAVAILABLE
G00MBB
PENETRATION SIZE GREATER THAN OR EQUAL TO 3 INCHES
G00MDB
CONTAINMENT PRE-EXISTING LEAK GREATER THAN 20 TIMES LA
CONDPRE
RPS/PCIS/ SCIS RX LOW LVL (3) SIGNAL UNAVAILABLE
LVP
RPS/PCIS/ SCIS HI DRY WELL PRESS TRIP 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

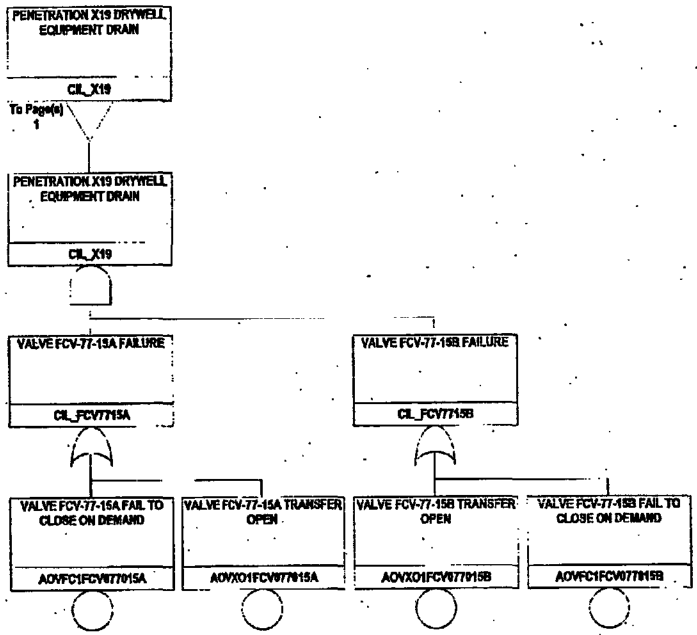
F-2

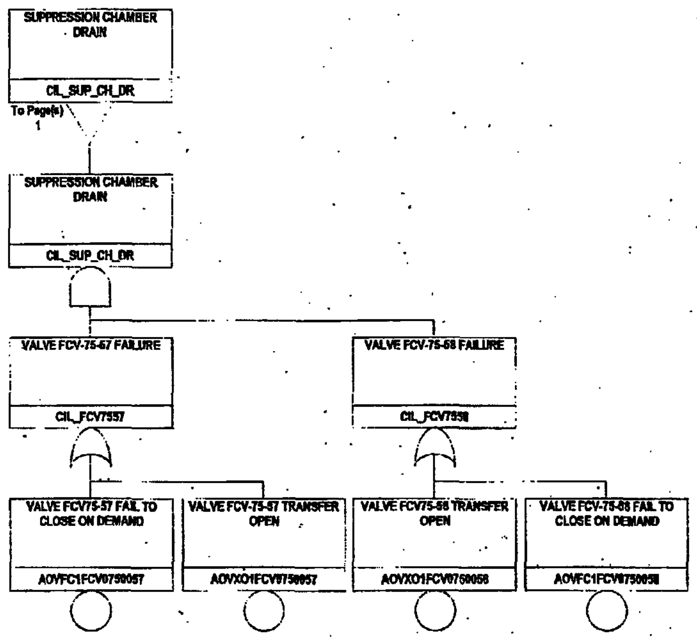

1
G01MDA
To Page(s)
1
PENETRATION X25?X205S2 NITROGEN PURGE SUPPLY
G01MEA
VACUUM BREAKER A PATH FAILURE
G01MEB
CHECK VALVE 64-800 FAILURE
G01MBC
VALVE FCV-64-20 FAILURE
G01MCC
4
G04MBA
From Page
5
PLANT CONTROL AIR UNAVAILABLE
PCA
VALVE FCV-64-20 TRANSFER OPEN
AOVXO1FCV0640020
VALVE FCV-64-20 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640020
VACUUM BREAKER B PATH FAILURE
G01MEB
CHECK VALVE 64-801 FAILURE
G01MEC
VALVE FCV-64-21 FAILURE
G01MFC
5
G04MDA
From Page
5
PLANT CONTROL AIR
PCA
VALVE FCV-64-21 TRANSFER OPEN
AOVXO1FCV0640021
VALVE FCV-64-21 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640021
3
G03MDA
From Page
4


2
G02MCA
To Page(s) 1
PENETRATION X26/X231 DRYWELL/WETWELL PURGE EXHAUST
G02MDA
PENETRATION X26S2DRYWELL PURGE EXHAUST FAILURE
G02MCB
PENETRATION X231S2WETWELL PURGE EXHAUST FAILURE
G02MDB
VALVE FCV-64-29 FAILURE
G02MAB
VALVE FCV-64-30 FAILURE
G02MCC
VALVE FCV-64-32 FAILURE
G02MDC
VALVE FCV-64-33 FAILURE
G02MFB
VALVE FCV-64-29 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640029
VALVE FCV-64-29 TRANSFER OPEN
AOVXO1FCV0640029
VALVE FCV-64-30 TRANSFER OPEN
AOVXO1FCV0640030
VALVE FCV-64-30 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640030
VALVE FCV-64-32 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640032
VALVE FCV-64-32 TRANSFER OPEN
AOVXO1FCV0640032
VALVE FCV-64-33 TRANSFER OPEN
AOVXO1FCV0640033
VALVE FCV-64-33 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640033


3
G03MDA
To Page(s) 2
FAILURE PATH FROM PENETRATION X25/X205
G03MEA
INBOARD VALVE (FCV-64-18 OR FCV-64-19) FAILURE
G03MDB
VALVE FCV-64-17 OR FCV-76-24 FAILURE
G03MFD
VALVE FCV-64-18 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640018
VALVE FCV-64-18 TRANSFER OPEN
AOVXO1FCV0640018
VALVE FCV-64-19 TRANSFER OPEN
AOVXO1FCV0640019
VALVE FCV-64-19 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640019
VALVE FCV-64-17 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0640017
VALVE FCV-64-17 TRANSFER OPEN
AOVXO1FCV0640017
VALVE FCV-76-24 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0760024
VALVE FCV-76-24 TRANSFER OPEN
AOVXO1FCV0760024


4
G04MBA
To Page(s)
2
CHECK VALVE 64-800 FAILURE
G04MBB
CHECK VALVE 64-800 FAILS TO RESEAT
COVFT1___6640800
CHECK VALVE 64-800 GROSS REVERSE LEAKAGE
COVLK1___6640800
5
G04MDA
To Page(s)
2
CHECK VALVE 64-801 FAILURE
G04MDB
CHECK VALVE 64-801 FAILS TO RESEAT
COVFT1___6640801
CHECK VALVE 64-801 GROSS REVERSE LEAKAGE
COVLK1___6640801

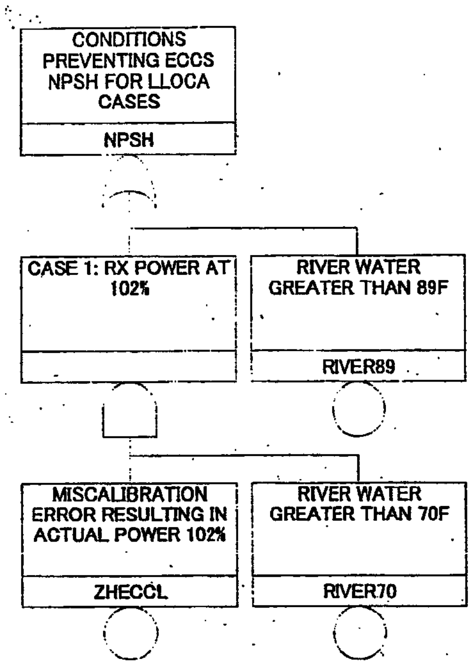

PENETRATION X19 DRYWELL EQUIPMENT DRAIN
CIL_X19
To Page(s) 1
PENETRATION X19 DRYWELL EQUIPMENT DRAIN
CIL_X19
VALVE FCV-77-15A FAILURE
CIL_FCV7715A
VALVE FCV-77-15B FAILURE
CIL_FCV7715B
VALVE FCV-77-15A FAIL TO CLOSE ON DEMAND
AOVFC1FCV077015A
VALVE FCV-77-15A TRANSFER OPEN
AOVXO1FCV077015A
VALVE FCV-77-15B TRANSFER OPEN
AOVXO1FCV077015B
VALVE FCV-77-15B FAIL TO CLOSE ON DEMAND
AOVFC1FCV077015B



SUPPRESSION CHAMBER DRAIN
CIL_SUP_CH_DR
To Page(s) 1
SUPPRESSION CHAMBER DRAIN
CIL_SUP_CH_DR
VALVE FCV-75-57 FAILURE
CIL_FCV7557
VALVE FCV-75-58 FAILURE
CIL_FCV7558
VALVE FCV75-57 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0750057
VALVE FCV-75-57 TRANSFER OPEN
AOVXO1FCV0750057
VALVE FCV75-58 TRANSFER OPEN
AOVXO1FCV0750058
VALVE FCV-75-58 FAIL TO CLOSE ON DEMAND
AOVFC1FCV0750058

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_DR	3	TRANSFER_OUT
CIL_SUP_CH_DR	1	TRANSFER_IN
CIL_SUP_CH_DR	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%
RIVER WATER GREATER THAN 89F
RIVER89
MISCALIBRATION ERROR RESULTING IN ACTUAL POWER 102%
ZHECCL
RIVER WATER GREATER THAN 70F
RIVER70

SYMBOL NAME	P#	SYMBOL TYPE
	1	AND_GATE
NPSH	1	OR_GATE
RIVER70	1	BASIC_EVENT
RIVER89	1	BASIC_EVENT
ZHECCL	1	BASIC_EVENT